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Deterministic Analysis of Operational Events in Nuclear Power Plants

Proceedings of a Technical Meeting held in Dubrovnik, Croatia, 23–26 May 2005

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FOREWORD

Computer codes are being used to analyse operational events in nuclear power plants but until now no special attention has been given to the dissemination of the benefits from these analyses. The IAEA's Incident Reporting System contains more than 3000 reported operational events. Even though deterministic analyses were certainly performed for some of them, only a few reports are supplemented by the results of the computer code analysis.

From 23–26 May 2005 a Technical Meeting on Deterministic Analysis of Operational Events in Nuclear Power Plants was organized by the IAEA and held at the International Centre of Croatian Universities in Dubrovnik, Croatia. The objective of the meeting was to provide an international forum for presentations and discussions on how deterministic analysis can be utilized for the evaluation of operational events at nuclear power plants in addition to the traditional root cause evaluation methods.

The IAEA would like to thank N. Cavlina from the Faculty of Electrical Engineering of the University of Zagreb, for organizing and hosting the meeting. The IAEA officer responsible for the meeting was M. Dusic of the Division of Nuclear Installation Safety.

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SUMMARY

1 BACKGROUND

Deterministic analysis is widely used in the area of nuclear engineering. It is typically used for licensing purposes, validation of operating procedures and plant simulators, support to probabilistic safety assessment, support for accident management and emergency planning, analysis of operational events and many others.

Most of these fields are extensively elaborated, but analysis of operational events is still an exception. This publication is intended to address this issue.

For this publication "operational event" is any event, which happened at a nuclear power plant (NPP) leading to an unplanned transient or significantly affecting the plant response characteristics. Such event may be caused by operator errors or equipment failures or some other malfunctions. The range of events could be from an apparently insignificant event to a transient that leads to reactor trip and subsequent actuation of designed safety systems or operator action.

The publication is focused on the NPP operational events, which can be analyzed by: thermal hydraulic codes (e.g. system codes, component and computational fluid dynamics codes), containment analysis codes and neutron kinetics computer codes. Structural analyses and related codes are not within the scope.

The following topics are regarded as important:

- Identification of events to be considered and associated phenomena;
- Methodology of analyses;
- Benefits from the operational event analyses.

The purpose of this meeting was to:

- Draw the attention of the international nuclear operators, designers and regulators, on benefits from analyzing operational events using deterministic computer codes, for enhancement of operational safety and reliability.
- Provide a brief overview of the current status.
- Set the grounds for discussions and further developments in the application of deterministic computer codes for operational event analyses.

The main conclusion from the meeting was that deterministic analysis of operational events are an effective supplement to the traditional root cause methodologies. The main benefits from such analysis are better understanding of the phenomena occurring during the event, precise identification of the event causes, evaluation of the impact of operator action versus automation, determination of the plant safety margins during the event, etc. The results of deterministic analysis of operational events can be used for operational safety enhancement, improvements to operators' training and also for validation and increased confidence in the computer codes and models used.

2 OVERVIEW OF THE CURRENT STATUS

During many years of NPP operation large experience in analysing operational events has been accumulated. In most of the cases only qualitative evaluation of the events was performed. This approach is sometimes not sufficient to fully understand the event. Such cases would require detailed computer modelling, using the experience of best-estimate deterministic analyses.

Deterministic analyses have been applied mainly for licensing purposes, and only in very few cases for operational event analyses. Operational event analyses could be applied both for validation of codes and for better understanding of the unit response to deviations from normal operation.

In the past only a limited number of operational events have been analyzed, for example: reactor trip, reactor coolant pump trip, inadvertent pressurizer spray valve opening, stuck open turbine bypass valve, inadvertent closure of main steam isolation valves etc. These analyses show that in many cases the codes are able to reasonably predict the transients, but some modelling limitations still exist. In some cases, plant measurement recordings were not sufficient for the operational event analyses.

In the IAEA/NEA Incident Reporting System (IRS) more than 3000 operational events are reported and only few of the reports are supported with computer code analysis. Computerised analyses of operational events do not seem to be widely used, or in cases, when such analyses are performed, their results are not widely shared. Therefore, there is a need to disseminate the results and benefits of these analyses.

The demand for deterministic analyses of operational events is supported by the need to avoid recurrence of events by better understanding of the phenomena involved and to validate the adequacy of corrective measures.

Progress in analytical tools, both software and hardware, allows for the performance of more detailed analysis of operational events, bringing several benefits:

- Better understanding of the phenomena, occurring during a specific event and helping to identify the direct and contributing causes;
- Identification of the impact of different contributing factors and conditions (operator and/or automated actions);
- Evaluation of the plant safety margins during the event;
- Operational safety enhancement; improvements in operator training and operating procedures;
- Increase of the confidence in the code and code input models;
- Sharing experience and utilizing external experience;
- Supporting designers by pinpointing weaknesses.

3 IDENTIFICATION OF EVENTS TO BE CONSIDERED AND ASSOCIATED PHENOMENA

A large number of operational events have occurred in nuclear power plants. Most of them were analyzed qualitatively and root causes were established. However, only few events were analyzed by deterministic computer codes.

In most cases, the causes of the event and the involved phenomena were considered to be understood, and detailed analysis of the event was not found necessary. If the event remains within the range of the unit's technical specifications and the response of the unit is correlated to existing analytical predictions, then the event does not need deterministic analysis (although such analyses could be beneficial for code and model validation purposes).

For some events either the operator or the regulator requires deterministic analysis to prove that unit behaviour fits predictions, to confirm safety margins, or for any other reason.

For some operational events it is difficult to identify the particular reason of the event occurrence, or there are multiple possible reasons from which the major one needs to be identified. For such events, detailed deterministic analysis could be beneficial, as it can bring the needed insights and can help to find appropriate corrective measures.

Typical operational events, which need deterministic analysis, are:

- Malfunction of valves, pumps or other components, resulting in complex response of the unit
  To analyse these events, the same codes and similar models are to be used as for safety analysis reports. These events are typically used for code validation, but there is still need to analyse such operational events for deeper understanding of plant behaviour and in order to prove the high quality of analytical predictions.
- Inadequate response of a control or safety system
  Some reported events resulted in inadequate or unexpected response of a control or safety system. To explain such events detailed modelling of the relevant system is necessary. The typical integral plant model used for safety analysis report does not credit control systems.
- Pipeline leakage, rupture or thermal fatigue
  Events initiated by damage of pipelines can be caused by thermo-hydraulic phenomena - e.g. coolant stratification or frequent changes of coolant temperature or pressure, leading to increased thermal stresses and fatigue. Simulation of these events may require specific modelling of local phenomena. Large uncertainties of initial and boundary conditions exist due to limitations of plant monitoring systems.
- Reactivity events
  These events (e.g. control rod or cluster drop) should be analyzed for assessment of effects and for proof of the similarity between design and measured changes in power profile.
- Other events
  List of other operational events, which should be analyzed, depends on reactor type and the country specific practices.

Some of the listed events can challenge the safety margins. Safety importance of these events is reason for deterministic analysis. Analysis should include simulation of the real processes and evaluation of margins to the acceptance criteria.

For deterministic analysis of operational events, the applied codes and input models must be capable to represent various phenomena, e.g:

- Thermo-hydraulics and neutron kinetics phenomena;
- Fast steam condensation in pipelines (up to water hammer);

- Large flow oscillations in long pipelines;
- effects in pipelines and tanks (thermal stratification);
- Other specific phenomena, related to the particular event.

It should be noted, that the selection of appropriate code(s) for analysis of complex events could be an iterative process, e.g. analysis with a system code (like ATHLET, CATHARE, RELAP5, APROS) could reveal the necessity of Computational Fluid Dynamics (CFD) code application.

4 METHODOLOGY OF ANALYSIS

The analysis of operational events involves four main steps:

(1) Establishing sequence of events;
(2) Evaluate the event as it happened;
(3) Evaluate the event as it could happen ("what if" analysis);
(4) Record the results of the event analysis in historical databases.

4.1 Establishing the sequence of events

In the methodology of the Analysis of Operational Events we can distinguish two approaches:

- Construction of dedicated input models (of systems, components) for specific event or group of events. It should be based on the use of validated codes.
- Using NPP analyzers. Using specific (i.e. dedicated to real NPP) NPP analyzer is always recommended because the model confidence is high and the analysis could be completed in a relatively short time (i.e. minimizing analysis costs).

The following recommendations apply for these two approaches:

(1) *Use of recorded data from the plant*
In order to obtain the best results it is recommended to use recorded data from the NPP. These data should be complete, i.e., including all the significant variables, and with the greatest level of confidence and accuracy. The accuracy of measured data and the recording interval should always be known and adequate.

(2) *Derivation of missing information*
For the missing information, engineering judgment and realistic assumptions for the important parameters need to be applied and should be documented. Missing data need to be assessed and derived carefully, preferably by support of sensitivity studies. Their correct derivation is very important in order to obtain a reliable detailed simulation model.

(3) *Realistic modelling*
Any of the above specified approaches has to be based on the state-of-art physical models available. Use of Best-Estimates (BE) codes (or the models implemented in them) is always recommended. They must be extensively validated and able to cover all the relevant physical phenomena for the analysis of the event.

(4) *Analyst qualification*
Analyst should be qualified in the use of BE codes and/or NPP analyzers. In order to minimize errors and wrong modelling of events and systems, he/she should have some years of experience in the use of these tools. It is necessary to have a good knowledge of the modelling of physical phenomena and of the limitations of this modelling. The numerical scheme and its characteristics and limitations should be known too.

(5) *Modelling scope*
All important systems and components, including safety systems, should be modelled. Control and auxiliary systems and secondary circuit should be included in the input model to the relevant extent of detail, depending on the event specifics. Simplifications in the modelling should be limited to the systems that have no influence on the simulated event. Appropriate modelling possibilities for implementation of digital control systems are required. As far as possible, the plant instrumentation and its response characteristics have to be modelled in order to get the most realistic signal representation from the simulation.

The use of interactive capabilities for operator actions (i.e. human-system interface) is recommended particularly for the analyses of the emergency procedures. Operators actions must be reconstructed from plant data and correctly implemented in the analysis tools.

(6) *Plant model validation*
Operational models which could consist of integrated models of thermal-hydraulics, neutron kinetics and control and auxiliary systems (e.g. input decks of RELAP5, ATHLET, CATHARE, APROS, QUABBOX-CUBBOX, PARCS codes) should be verified and validated.

Different validation and verification methods can be used:

- Plausibility test;
- Human expertise and know-how (e.g. from NPP operators);
- Comparison with other previously tested models;
- Graphical comparison with measurement results from NPP for a minimum of two cases;
- Sensitivity Analysis.

The decision about the level of the model validation can be decided by the analyst with or without the support of independent experts.

4.2 Evaluate the event as it happened

Once sufficient agreement between the event and the simulation has been reached, relevant characteristics of the event should be identified. Depending on the specific objective of the analysis, different types of evaluation can be performed.

Aspects to be considered include:

4.2.1 Consistency with design basis accident analysis

It should be identified if the assumptions of the design basis accident analysis (DBAA) have been met during the event. These assumptions include the type of initiating event, range of initial and boundary conditions, operator and automatic actions, additional failures in equipment or human errors. If the event meets the assumptions of DBAA, its consequences, measured in terms of the safety variables used to evaluate acceptance criteria, should not be worse than those resulting from design basis accidents (DBA). Moreover, it should be possible to identify one or more DBA that sufficiently cover the evolution of the analyzed event. Significant safety variables should be checked to confirm that their extreme values remain on the safe side with respect to the corresponding extreme values in the identified DBA. If inconsistencies are found, a revision of the DBAA should be considered.

The occurrence of operational events non adequately covered by DBAs can be an indication of insufficient design requirements for protections. The need of extending the design basis envelope by adding new DBAs or by changing the existing ones should be analyzed.

4.2.2 Analysis of operator actions

The operator actions taken during the event should be clearly identified and evaluated. Agreement with operating procedures, especially with EOPs should be carefully analyzed. Deviations from procedures should be analyzed to determine the reason. If they are found beneficial, revision of operating procedures should be considered. If not, the operator training should be improved. Possible inconsistencies between EOPs, other operating procedures and operator actions assumed in DBAA should be identified.

4.2.3 Identification of exceeded safety limits

An important input to evaluate the safety significance of the event is the identification of safety limits that have been exceeded or approached. If the event is covered by one or more DBAs, the safety limits applicable to the class the DBAs belong to or to a higher one, should not be exceeded. Exceedance of limits applicable to less severity classes is not an indication of poor protection performance, although the occurrence of any event beyond the *anticipated events* class (the lowest severity class) is a sufficient reason to deeply analyze the causes of the event to avoid recurrence. It could result in a reconsideration of the event classification in DBAA, possibly leading to protection design changes.

4.2.4 Identification of failed equipment or unexpected performance

The usefulness of the analysis will be increased if it is possible to assess the level of damage or failure of the equipment involved in the transient. The safety level of the whole plant could in this way be assessed. Furthermore, the NPP operator could be trained to take appropriate actions to avoid such critical situation that could result in the equipment damage.

4.3 Evaluate the event as it could happen (“what if” analysis)

Alternative paths that could have been reasonably expected or that could have special significance should be analyzed. Aspects to be considered include:

- Variation in plant parameters, which could include effects of different operating states or lifetime along the refuelling cycle;
- Variations in time of operator actions;
- Consideration of possible operator errors;
- Additional equipment failures, especially if some equipment has been found close to its qualification limits;

- Effects of reasonable miscalibration of plant instrumentation, especially protection setpoints.

4.3.1 Record the results of the event analysis in historical databases

A detailed reporting of the analysis performed has to be recorded. The report format has to be as much as possible understandable for the wider group of users (operators of other NPP, code analysts, employees of safety departments). The sequence of events should be presented as well as NPP data (measurements throughout the event, calculation results, plant sketches and lay-outs). The list of recommended actions should also be included. An appropriate classification of the events by safety categories, by physical phenomena and by plant components involved should also increase the usefulness of the analysis performed.

5 BENEFITS FROM DETERMINISTIC ANALYSIS OF OPERATIONAL EVENTS

Qualitative analysis of operational events is performed as a rule by each nuclear power plant – mainly for identification of the event causes (root cause analysis) and for evaluation of the operator actions, etc. Usually such analyses are based on the existing records, operator's reports and engineering judgment. Deterministic analysis is carried out only in a few cases, basically for two main purposes:

(1) to reveal the nature of the processes and phenomena which led to the failure by attempting to reconstruct the details of the process, or
(2) to benchmark and validate an existing computer model by taking advantage of a well recorded and understood event.

From the point of view of the NPP operating organisation the deterministic analysis of an event provides the greatest direct benefit when it is carried out as an integral part of the investigation. In such a case – often in an iterative way – the analysis and the other activities in the investigation process mutually enhance each other. In some cases, it is impossible to carry out a complete and successful investigation without detailed modelling of the process.
Thus the benefits of the two applications of computer codes (see also Fig. 1) and models for analysing operational events can be grouped as follows:

- Better understanding of the phenomena, occurring during a specific event and helping to identify the direct and contributing causes;
- Identification of the impact of different contributing factors and conditions;
- Evaluation of the plant safety margins during the event;
- Operational safety enhancement; improvements in operator training and operating procedures;
- Increase of the confidence in the code and code input model;
- Sharing experience and utilizing external experience;
- Supporting designers by pinpointing weaknesses.

The last three items (Figure 1) are more related to the general supporting process behind the operation, thus the benefits they yield are indirect from the point of view of a nuclear operator.

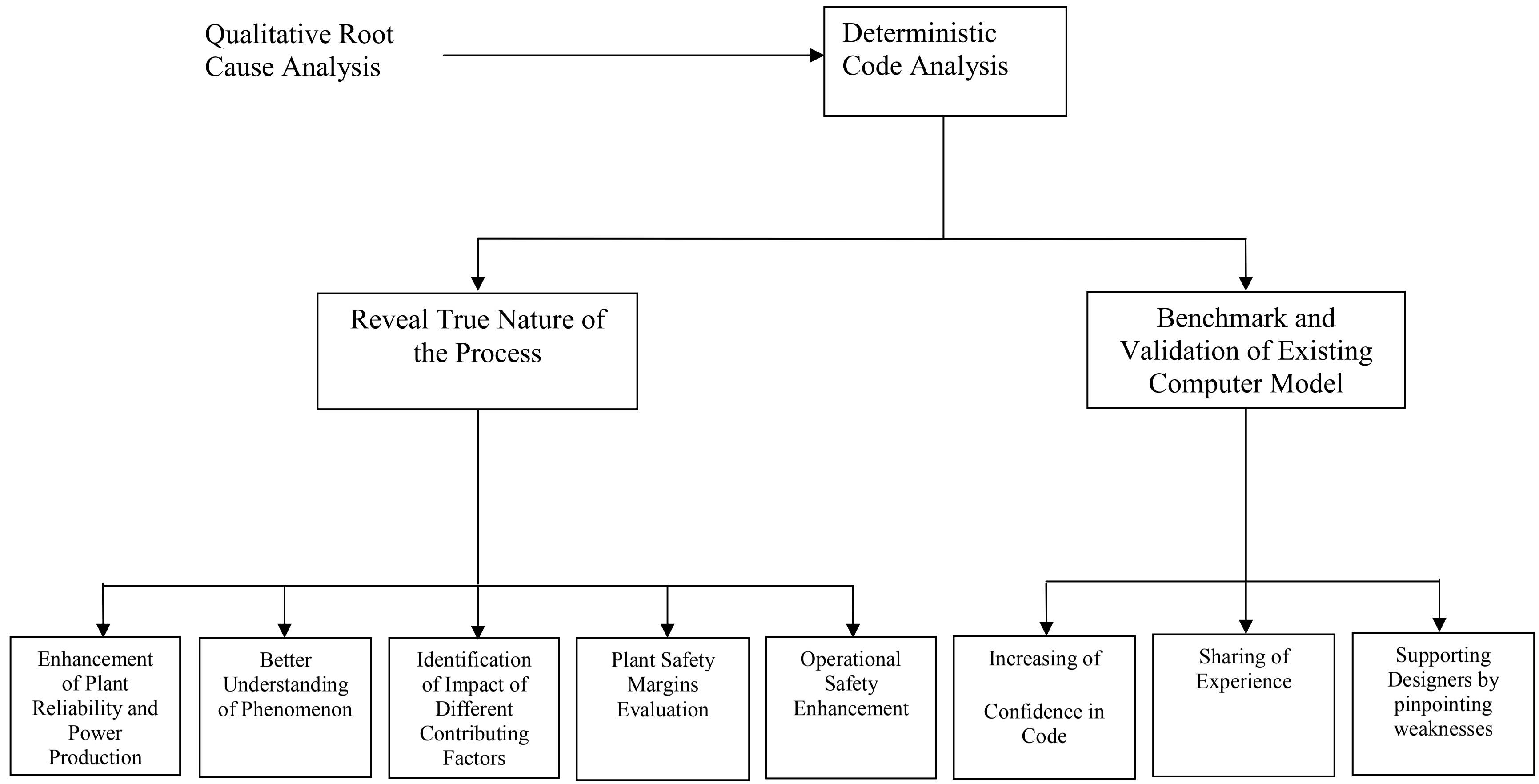

Fig. 1. The benefits of deterministic analysis of operational events.

The other items (Figure 1) are closely related to the event investigation, thus the benefits are equivalent to a well performed event investigation. In this context the deterministic analysis is a tool to enhance the complete investigation and its benefits. Though the focus in case of an event investigation is on safety, its completeness may also improve the plant reliability and production.

A brief discussion for each of these benefits is presented below.

5.1 Better understanding of the phenomena, occurring during a specific event

The plant monitoring systems are not capable to directly measure some parameters, related to some of the event-specific phenomena (e.g. peak cladding temperature, fluid temperature profiles in large/long pipes, flow rates through valves, leaks and orifices, etc.). Consequently, many phenomena and inter-relations, which actually occurred during the event remain hidden or, at best, are subject to guessing and engineering judgment. The application of computer codes and models allows for identification and analysis of such phenomena and relations. For example, a system code calculation could identify and evaluate quite precisely the temperature profile along a long pipe, as well as the relevant core feedback effects. Besides, the computer models are capable to represent in detail the phenomena, which occur for extremely short time and provide estimates for their impact on the overall event development.

Whenever the details of an event are not fully understood on the basis of the available process data and event reports (there is some unexplained parameter behaviour or some apparent contradiction) it is highly recommended to carry out detailed analysis by using computer models.

Sometimes it is difficult to identify the root cause (or important contributing causes) of an operational event because of the incomplete understanding of the process due to the complex inter-relations among the technology, possible control system malfunctions and operator actions. In such cases the computer simulation may provide a reliable source of quantitative information for all factors, which have influenced the event occurrence. This information should also be used for the review of the list of initiating events (or the assumptions thereof), which are analyzed in the Safety Analysis Report of the plant.

5.2 Identification of the impact of different contributing factors and conditions

Once a computer model of the event is prepared and verified, it can also be used for the identification of the contribution of each factor. For example, a sensitivity study could yield a series of possible outcomes of a single initial event – depending on selected actions and inadvertent conditions, which are introduced, removed or modified. Such a result could then be used as a basis for final evaluation of the automated action and operators' performance during the event. Changing some inadvertent conditions to all possible typical cases may help to judge the seriousness of the event (e. g. under some less fortunate conditions the outcome of the same event could yield more serious consequences). Further application of the analysis results could be the review and improvement of the related operation procedures.

5.3 Evaluation of the plant safety margins during the event

In some cases there is interest in a realistic evaluation of the safety margins, i.e. "How close did we get to the actual safety limit?" This question cannot be answered in some cases without detailed model analysis of the case, since very often the actual safety limits are related to such parameters, which can not be measured (e. g. cladding temperature, DNBR, etc.). Whenever such a safety parameter is endangered during the event, the completion of detailed analysis may be indispensable.

When the consequences of the event are not directly measurable, then it is also the only solution to judge the state of the affected components by carrying out detailed analysis. A typical example for such a case is the fatigue analysis of some component after an event, which caused some unplanned thermal stress on some components.

5.4 Operational safety enhancement

It is a general goal of a detailed and thorough event investigation to provide inputs for operational safety enhancement (with the direct goal of avoiding recurrence). In many cases the detailed deterministic analysis of the physical process during the event enhances root cause investigation, and thus directly contributes to the enhancement the operational safety. The computer modelling provides powerful means for explanation and representation of the cause, phenomena, evolution and lessons learned from the event. The complete understanding of an abnormal operational event may require modification of the training material of the operating personnel and/or modification of the procedures. In some cases it may reveal that some possible conditions are not covered in any procedures, thus it is necessary to complement the existing set of procedures. It is especially important in such cases to generalise the analysis to make it possible to identify and resolve the safety issues, related to the event – by analysing all possible additional failures and their consequences.

The results of the analyses (correct course of actions, errors, malfunctions, etc.) may be used in the operators training in order to assure correct response to the same or similar event in the future. Comparison between event "replay" on the plant simulator and the results from computer code calculations could be useful as a means for simulator validation and improvement.

5.5 Increase of the confidence in the code and code input model

The reliable comparison between the computer calculation results and the recorded data is a basis for evaluation of the code capability and model accuracy.

Although code and input model verification and validation is out of the scope of this document, it should be noted that both the plant and the regulatory body appreciate any additional proof for the reliability of the results, produced with codes and models, which are often used for licensing purposes. The analysis could also identify particular deficiencies in the code input model, thus providing basis for its improvement.

5.6 Sharing experience and utilizing external experience

The standard event reports which are circulated through the Incident Reporting System are sometimes not detailed enough to make it possible to gain any direct benefit for other organisations. The detailed analysis reports, however, provide greater opportunity to utilise experience of other organisations. Exchanging detailed data, including boundary conditions is the most direct and effective way of utilizing external experience.

5.7 Supporting designers by pinpointing weaknesses

In case of some events which are related to unexpected conditions, the detailed analysis of the event may also provide recommendations to the plant designers to modify some of the details in order to detect the event in early phase or prevent the escalation of the process, or completely eliminate the possibility of such an undesirable process.

6 FUTURE TRENDS AND RECOMMENDED DEVELOPMENTS

Deterministic analyses of operational events are needed and are expected to become even more important in the future, considering the following:

- Needs to understand in detail the reasons for events occurrence, to get deep insights into the event phenomena and resulting challenges;
- Support of proposals for component replacement and modifications;
- Efforts to include real events into the scope of plant training simulator capabilities;
- Provision of quantitative event characteristics to be used in probabilistic studies and related applications;
- Occurrence of new events, resulting from aging, fatigue of materials and components, etc.

The operational events being the most typical candidates for deterministic analysis are those safety important events, which include complex interaction of several systems.

Deterministic analysis of operational events is expected to benefit from the progress in tools and methodologies. Steadily growing expertise of analytical teams, together with improvements in computer technology and development of codes will facilitate:

- Overall trend to use best estimate codes;
- Ability to apply complex and coupled codes. The increasing trend to use well tested codes interactively connected for simulation of different systems or operational procedures will significantly increase the simulation power;
- Possibility to apply much more detailed nodalization schemes;
- Capability to simulate complicated geometries and sufficiently long processes with CFD codes;
- Use of virtual reality to analyze operator actions.

Expected progress in plant monitoring systems (hardware and software) will be a good contribution to make analysis of real operational events more flexible and less challenging to analysts. It could remove the most typical problems: incompleteness of relevant data for a complex description of the event and provision of all necessary initial and boundary conditions.

Unified analysis methods, able to simultaneously address deterministic and probabilistic aspects of event analysis could be highly beneficial to better evaluate the safety significance of the events.

QUANTITATIVE ASSESSMENT OF MSIV CLOSURE EVENTS IN KRSKO NPP CALCULATED BY RELAP5/MOD3.3

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Abstract. In the past only a limited number of operational events have been analyzed. Recently, it was recognized that deterministic analysis of operational events could improve operational safety of nuclear power plants. The purpose of the study was to qualitatively and quantitatively assess the RELAP5/MOD3.3 calculations of two abnormal events occurred in Krško nuclear power plant (NPP) originating from sudden closure of Main Steam Isolation Valve (MSIV). Both events occurred before the SG replacement in 2000, the first one in September 1995 and the second one in January 1997. For quantitative assessment the fast Fourier transform based method (FFTBM) was used. The obtained measured plant parameters were mostly limited to flows, pressures, temperatures and levels. A full two-loop RELAP5 plant model, developed at Jožef Stefan Institute (JSI), has been used for the analyses. The short-term analysis (50 s) was performed to assess the RELAP5/MOD3.3 code against plant and automatic regulation, protection and safety systems response to the initial event. The long-term calculation of event was also performed. Namely, later course of the transient strongly depends on the operator actions. The results of quantitative analysis performed by FFTBM showed that the plant data were reproduced very well. Larger discrepancies were observed for steam flows and pressurizer level but still acceptable. Very accurate were calculations of hot and cold leg temperatures, and primary and secondary pressures. The results suggest that the RELAP5/MOD3.3 code could reproduce the operational events very well.

1 INTRODUCTION

Available plant data from various abnormal events or incidents are of great importance for assessing large system thermal-hydraulic computer codes as well as for improving operational safety of nuclear power plants. In the paper the RELAP5/MOD3.3 analysis of two abnormal events originating from sudden closure of Main Steam Isolation Valve (MSIV) is presented. They have occurred in nuclear power plant (NPP) Krško, which is a two-loop Westinghouse PWR plant, on September 25th, 1995 at 10:22:06 and January 1st, 1997 at 8:33:30. Both transients were initiated by sudden closure of Main Steam Isolation Valve (MSIV) and were similar. The first event in 1995 was caused by a malfunction in steam generator no.1 (SG1) MSIV regulation, while the second event in 1997 was a consequence of SG2 MSIV stem breach. The second event in 1997 resulted even in slightly faster MSIV closure than in 1995. Both events occurred with original Westinghouse W-D4 steam generators installed. These were already considerably degraded and highly plugged in 1995 (SG1: 18.87 % and SG2: 17.27 %). After extensive SG tube plugging during the 1996 outage the plugging level was reduced (SG1: 16.27 % and SG2: 10.05 %) so the 1997 event occurred at somewhat different plant state.

In the present study the operational events were used for code assessment purposes of RELAP5/MOD3.3. Recently the last frozen version RELAP5/MOD3.3 has been released, before merging with another best-estimate thermal-hydraulic system code TRAC into an integrated thermal-hydraulic code. It is thus of utmost importance to assess models built in RELAP5/MOD3.3 code against real plant transients before the code merger. For quantitative assessment of code predictions the fast Fourier transform based method (FFTBM) was used. Finally, because the data from real plant transients are rather rare, there are of exceptional value, no matter if parts of the primary and secondary system of Krško NPP were replaced during modernization in 2000.

2 FFTBM OVERVIEW

2.1 Original FFTBM

The original FFTBM is briefly described below. For further information on method and its applications the reader is referred to [1] and [2]. For calculation of measurement-prediction discrepancies the experimental signal $F_{\exp}(t)$ and error function $\Delta F(t)$ are needed. The error function in the time domain is defined as $\Delta F(t) = F_{cal}(t) - F_{\exp}(t)$ where $F_{cal}(t)$ is calculated signal. The code accuracy quantification for an individual calculated variable is based on amplitudes of discrete experimental and error signal obtained by fast Fourier transform at frequencies f_n, where (n=0,1,...,2^m) and m is the exponent defining the number of points ($N=2^{m+1}$). These spectra of amplitudes are used for calculation of average amplitude:

$$\mathrm{AA} = \sum_{n=0}^{2^m} \left|\tilde{\Delta F}(f_n)\right| \bigg/ \sum_{n=0}^{2^m} \left|\tilde{F}_{\exp}(f_n)\right|, \tag{1}$$

where $\tilde{\Delta F}(f_n)$ is error function amplitude at frequency f_n and $\tilde{F}_{\exp}(f_n)$ is time function amplitude at frequency f_n. The AA factor can be considered a sort of average fractional error and closer is AA value to zero the more accurate is the result.

The overall picture of the accuracy for the given code calculation is obtained by defining total average amplitude (total accuracy):

$$\mathrm{AA}_{tot} = \sum_{i=1}^{N_{\mathrm{var}}} (\mathrm{AA})_i \cdot \left(w_f\right)_i \quad \text{with} \quad \sum_{i=1}^{N_{\mathrm{var}}} \left(w_f\right)_i = 1, \tag{2}$$

where N_{var} is the number of the variables analyzed (normally from 20 to 25 are selected), and $(\mathrm{AA})_i$, and $(w_f)_i$ are average amplitude and weighting factor for i-th analyzed variable, respectively. Weighting factors were introduced to take into account the importance of each variable and can be found in references for FFTBM (e.g. [2]). For the total accuracy the following criteria were set: $\mathrm{AA}_{tot} \leq 0.3$ characterize very good code predictions, $0.3 < \mathrm{AA}_{tot} \leq 0.5$ characterize good code predictions, $0.5 < \mathrm{AA}_{tot} \leq 0.7$ characterize poor code predictions, and $\mathrm{AA}_{tot} > 0.7$ characterize very poor code predictions. For primary pressure accuracy the criterion is $\mathrm{AA}_{pr} \leq 0.1$.

2.2 Improved FFTBM

Recently new accuracy measures were proposed and tested on IAEA-SPE-4 data [3]. An improved FFTBM with new accuracy measures is described in detail in [4]. Let be i-th variable accuracy (VA):

$$\mathrm{VA}_i = \mathrm{AA}_i \cdot \left(w_f\right)_i \cdot N_{\mathrm{var}}. \tag{3}$$

The minimal variable accuracy (VA_{min}) was defined as:

$$\mathrm{VA}_{\min} = \max\{\mathrm{VA}_i\};\ i = 1\ to\ N_{\mathrm{var}} \tag{4}$$

and also represents the hypothetical total accuracy combined from variables all having the same value of $\mathrm{AA} = \mathrm{VA}_{min}$ (see Eqs. (2) and (3)). Therefore the variable accuracy can be compared to acceptability limits for AA_{tot}.

3 EVENTS AND INPUT MODEL DESCRIPTION

3.1 MSIV events description

Both MSIV events occurred at 100% reactor power. Initial steam flow in SG1 was slightly higher than in SG2 and the corresponding SG1 pressure was slightly lower than in SG2 due to asymmetric SG tube plugging (SG1: 18.87 % and SG2: 17.27 % in 1995 and SG1: 16.27 % and SG2: 10.05 % in 1997). In the first case a malfunction in the SG1 MSIV control circuit caused inadvertent valve closure, while in the second case the valve stem has been broken in the SG2, which also caused sudden valve closure. The flow through the affected SG MSIV stopped in a few seconds in the first case and almost instantaneously in the second case. The affected SG pressure rose up to the SG PORV setpoint shortly after. Since the heat was still generated in the core, the intact SG flow increased rapidly and the pressure dropped. This was detected by the reactor protection system and the safety injection (SI) was triggered. On SI signal the reactor trip and turbine trip signals were generated. Reactor trip caused also main feedwater isolation. The intact SG was also isolated at the same time when SI signal was produced and the pressure in this SG also started to rise. The pressure in both SGs were thus stabilized soon at the SG PORV setpoint. A closer look to the secondary pressure behavior indicated possible steam leakage through a MSIV or through some steamline drain valves.

3.2 RELAP5/MOD3.3 input model description

A full two-loop plant model, developed at Jožef Stefan Institute (JSI), has been used for the analyses (see refs. [5] through [9]). The model includes old Westinghouse D4 type steam generators with assumed 18% U-tubes plugged in both steam generators. Regardless of the fact that the SG plugging level was slightly different at both events, for both transients the same RELAP5/MOD3.3 master input model was used for simulation. It accounted for 18 % SG tube plugging, reflecting the condition for which the plant was licensed before SG replacement and power uprate in 2000. The operator actions modeled were SI reset, chemical and volume control system (CVCS) charging and letdown flow, and auxiliary feedwater (AFW) flow. The CVCS and AFW flows were set to the values matching the plant data.

The utilized model consists of 183 volumes, connected with 200 junctions. Plant structure is represented by 203 heat structures with 705 mesh points, while the reactor protection and regulation systems, safety systems operational logic and plant instrumentation is represented by 109 logical conditions (trips) and 180 control variables.

4 RESULTS

In Figures 1 and 2 only first 50 seconds of the calculation are compared to the plant measured data for both events to assess the RELAP5/MOD3.3 code against automatic regulation, protection and safety systems response to the initial event. In the Figure 3 the long-term calculation of events is shown which strongly depends on the operator actions. The measured data in Figures 1 to 3 are marked with NPP and the year of event while RELAP5/MOD3.3 calculations are marked with R5 and the year of event. Finally, in Table 1 are shown the results of quantitative assessment by FFTBM.

After the initial event the secondary steam flow in the affected steamline was reduced rapidly (Figures 1a and 1b), which caused a sudden pressure rise (Figures 1c and 1d) and SG narrow range (NR) level rise (Figures 1e and 1f). On the contrary, in the intact steamline the steam flow has rapidly increased and the SG pressure drop was observed with SG level increase. After about 2-3 seconds the high steam flow with coincident low steamline pressure has simultaneously triggered SI signal and intact SG MSIV closure. Turbine tripped immediately after the intact MSIV closure, while the reactor scram occurred 1 second later and, followed by main feedwater closure another 1 second later. Figures 1a to 1f show that the 1995 event was slightly slower than the 1997 event. The differences in the time of plant response can be explained by analyzing the details of both initiating events. In 1995 the affected MSIV no.1 was closed smoothly because of the malfunction in regulation system, while in 1997 the stem breach caused uncontrolled oscillatory closing of MSIV no.2. In 1997 a large flow oscillation at the very beginning of the transient caused SG pressure oscillations and the MSIV closure signal was triggered earlier. Since the oscillatory closing of the MSIV could not be modeled in RELAP5/MOD3.3 using motor valve (MTRVLV) component, those differences between the two events could not be demonstrated in the RELAP5/MOD3.3 calculation. Nevertheless, RELAP5/MOD3.3 predicted most of the secondary parameters very close to the plant data.

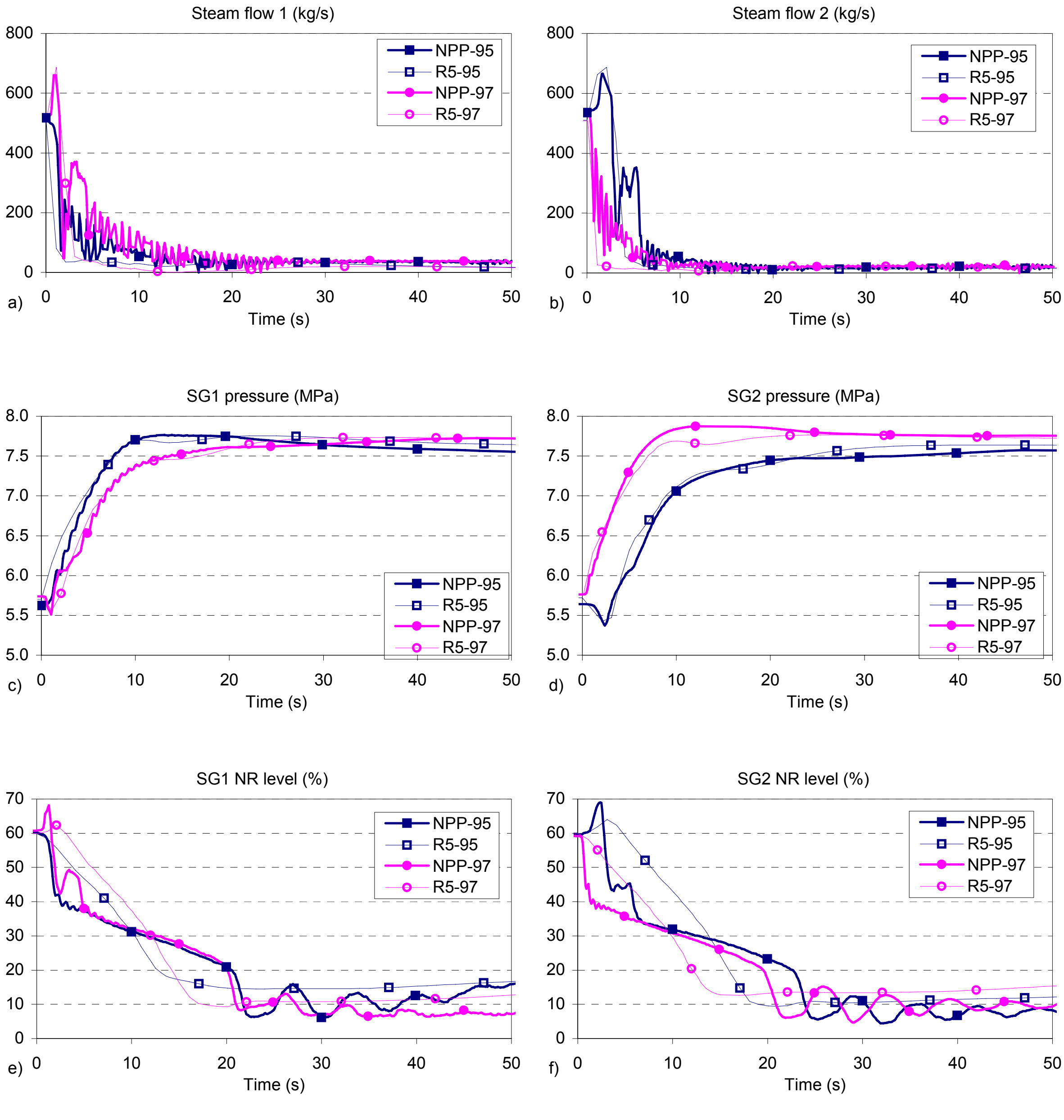


Fig. 1. Short-term comparison of secondary parameters for both MSIV closure events.

For primary parameters shown in Figure 2 the plant measured data are practically the same for both events and similarly the calculated data. The calculated primary pressure rise (Figure 2a) was slightly faster and higher after the initial event than the plant recordings showed. Small disagreement in initial conditions and the expected error band in RELAP5/MOD3.3 prediction could justify relatively small discrepancies. For pressurizer (PRZ) level shown in Figure 2b sharper initial level rise could be observed in calculation in comparison with the plant data, but the trends were correctly predicted. Some discrepancy originates from the difference in initial PRZ level between plant data and RELAP5/MOD3.3 data. Another possible reason for the differences in primary parameters prediction could be in modelling spray flow and reactor coolant pump seal leak flow, which contribute to correct primary pressure and inventory prediction. Those measurements were not available among the plant data. The cold and hot leg temperatures shown in Figures 2c to 2f responded faster in the calculated case. The reason is the delay in measurement system, which is not modelled in RELAP5/MOD3.3 model.

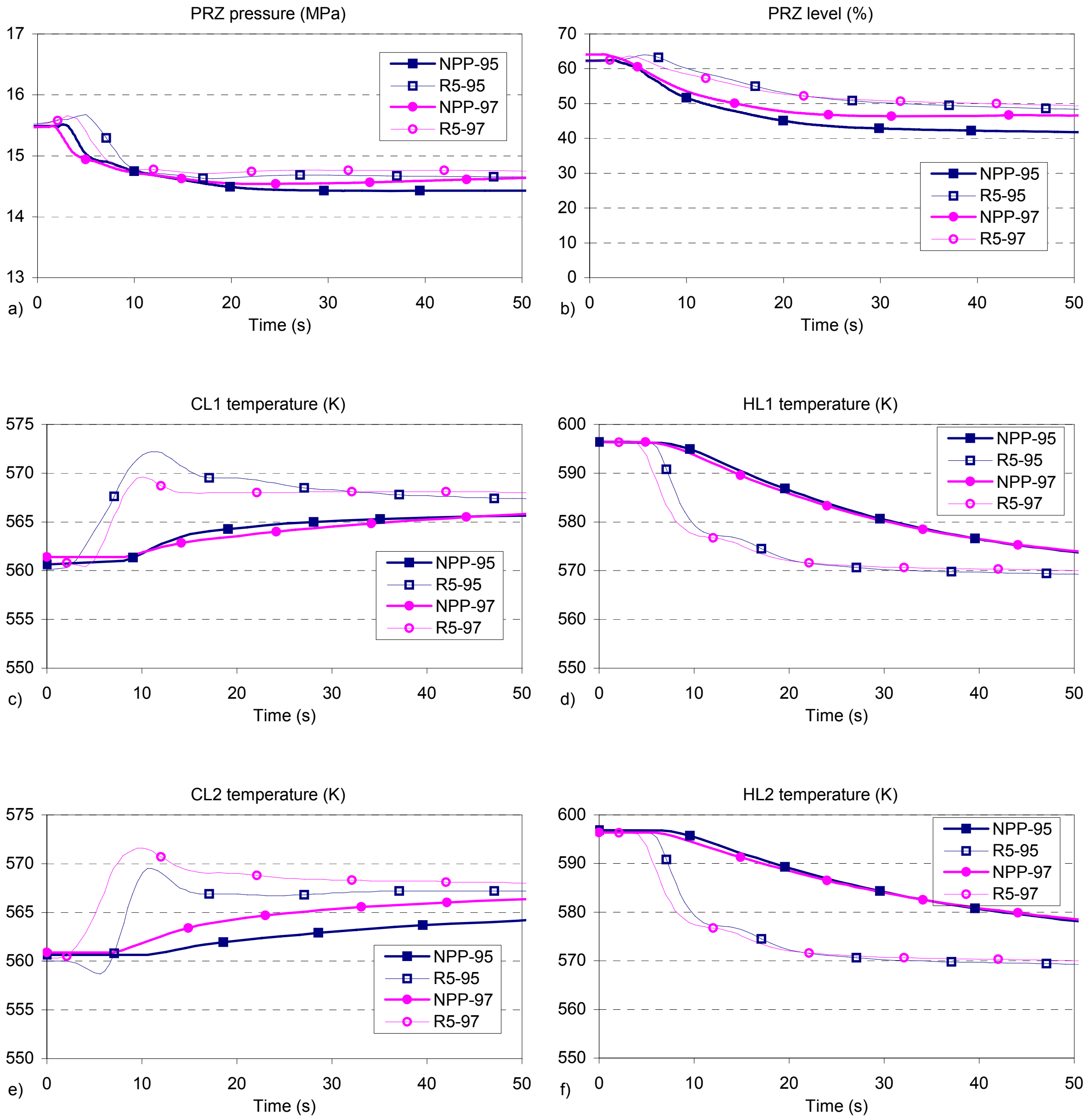
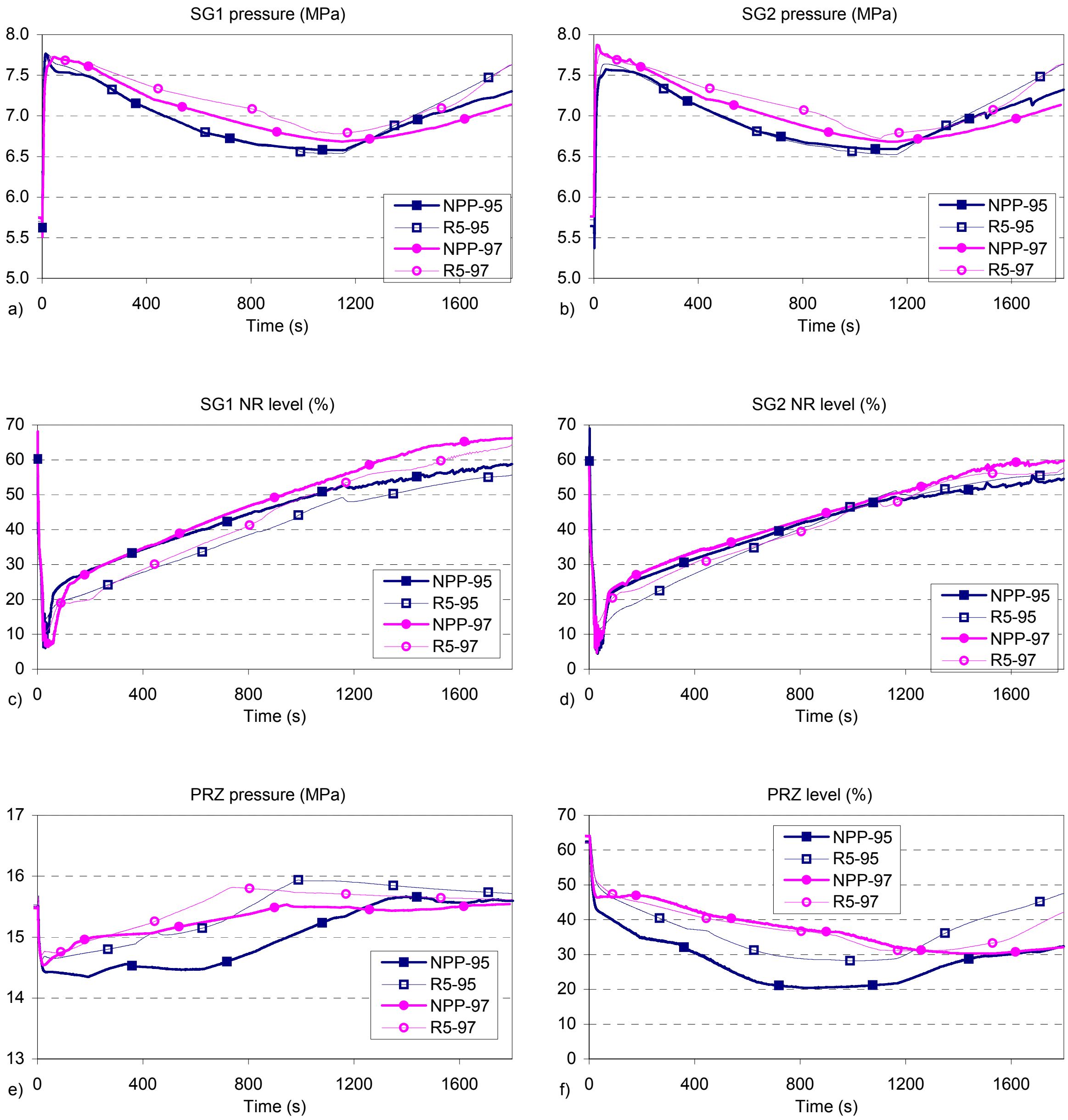


Fig. 2. Short-term comparison of primary parameters for both MSIV closure events.

In Figure 3 the long-term response showed that secondary parameters (Figures 3a to 3d) were matched closely to the plant measured data on the long-term (1800 s), since the AEW flow was tuned to the measured plant data. Besides, the introduced secondary leak was tuned to such values that SG pressures were also matched closely with the plant data on the long-term. The discrepancy in SG NR level originates from the time period before AFW injection at around 110 s. Some larger discrepancies were observed on the primary side for primary pressure (Figure 3e) and PRZ level long-term development (Figure 3f) for the 1995 event simulation, while the agreement is much better in the 1997 event simulation. The long-term temperature prediction is in very good agreement with plant measured data with differences a few K (Figures 3g and 3h).



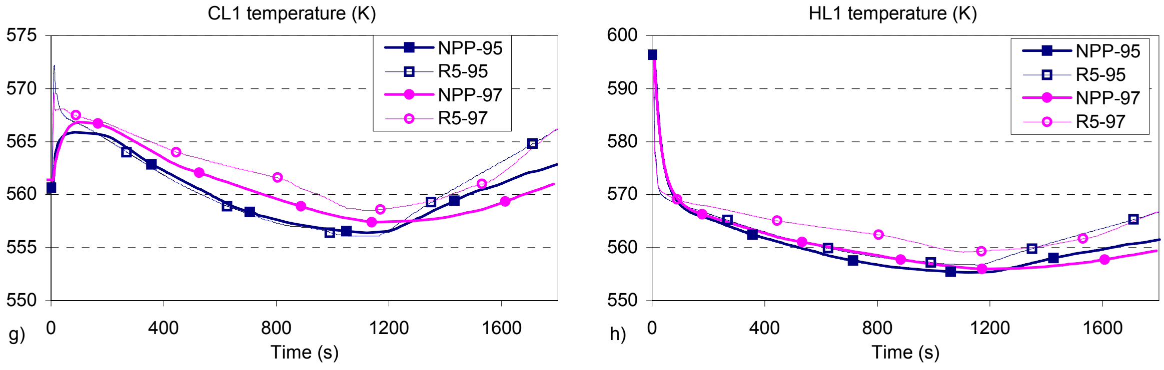


Fig. 3. Long-term comparison of parameters for both MSIV closure events.

Quantitative results for the short and long-term calculations are shown in Table 1. It can be seen that code accuracy expressed in terms of AA for short and long-term calculations is comparable. The exception is PRZ level in 1995 event where accuracy is significantly lower for the long-term calculation. This is in agreement with the qualitative analysis described above. Larger discrepancies were observed for steam flows and levels but still acceptable. However, the contribution of these variables (see values of VA in Table 1) to total accuracy is lesser as AAs show due to smaller weighting factors flows and levels than for pressures and temperatures. Very accurate were calculations of hot and cold leg temperatures, and primary and secondary pressures. When comparing primary and secondary parameters, in short-term analysis higher accuracy as judged by FFTBM was obtained for primary parameters. In long-term analysis this difference was smaller due to less accurate predictions of primary parameters than in the short-term analysis. This finding is in a slight contradiction with the statement made in [9] that primary parameters prediction indicates that matching with the plant data was not exactly as close as for secondary parameters, but still good agreement was reached. It seems that the primary parameters plotted in very narrow range visually gave impression of larger discrepancies while FFTBM objectively (regardless the selected range) quantify the accuracy.

Also when comparing calculations for both MSIV events the accuracy is very similar. When comparing the total accuracy with the main circulation pump (MCP) trip at nearly full power transient in Mochovce VVER [10] the obtained total accuracy for MSIV events is comparable. In the case of Mochovce MCP trip the total accuracy was 0.09 and the largest AAs were calculated for steam flows and steam generator NR levels while the most accurate variables were PRZ pressure and loop temperatures.

TABLE 1. AVERAGE AMPLITUDE AND VARIABLE ACCURACY FOR SHORT- AND LONG-TERM CALCULATIONS

		1995 event				1997 event			
	Time interval	Short-term (0 - 50 s)		Long-term (0 – 1780 s)		Short-term (0 – 50 s)		Long-term (0 – 1780 s)	
		AA	VA	AA	VA	AA	VA	AA	VA
1	Steam flow 1	0.63	0.15	0.58	0.14	0.49	0.12	0.48	0.11
2	Steam flow 2	0.56	0.13	0.53	0.13	0.68	0.16	N.A.	N.A.

		1995 event				1997 event			
3	SG1 pressure	0.05	0.04	0.08	0.07	0.04	0.03	0.13	0.10
4	SG2 pressure	0.05	0.04	0.08	0.06	0.04	0.03	0.13	0.10
5	SG1 NR level	0.37	0.19	0.32	0.16	0.39	0.20	0.33	0.17
6	SG2 NR level	0.43	0.22	0.39	0.20	0.45	0.23	0.32	0.16
7	PRZ pressure	0.05	0.06	0.09	0.11	0.04	0.05	0.07	0.08
8	PRZ level	0.23	0.11	0.42	0.21	0.15	0.08	0.23	0.12
9	CL1 temperature	0.02	0.04	0.02	0.04	0.02	0.03	0.03	0.05
10	CL2 temperature	0.02	0.04	0.02	0.05	0.02	0.04	0.03	0.05
11	HL1 temperature	0.04	0.07	0.05	0.09	0.04	0.07	0.06	0.10
12	HL2 temperature	0.05	0.09	0.05	0.09	0.05	0.09	0.05	0.10
	total	0.10		0.11		0.09		0.10*	

* - based on 11 variables

5 CONCLUSIONS

The qualitative conclusions made in [9] stated that plant data were reproduced very well by RELAP5/MOD3.3, especially secondary parameters matched better because of the MSIV leakage tuning. Some more discrepancies were between plant data and RELAP5/MOD3.3 prediction, but these could be explained by differences in initial conditions and by the expected RELAP5/MOD3.3 modeling accuracy. Some additional uncertainty originates in modeling of certain parameters, which were not measured at the plant and could thus not be verified and compared to the RELAP5/MOD3.3 calculated results. Quantitative results confirmed the qualitative conclusions, which stated that plant data were reproduced very well. Very accurate were calculations of hot and cold leg temperatures, and primary and secondary pressures. In addition it was shown that also primary side parameters were very accurate as judged by FFTBM. This is in slight contradiction with the qualitative conclusions made in [9]. The reason is that the parameters were plotted in very narrow range thus visually giving impression of larger discrepancies while FFTBM objectively quantify the differences. The quantitative results also showed that code accuracy for short and long-term calculations is comparable. Interesting is also the finding that the accuracy of both event calculations is practically the same. Finally, the results of quantitative analysis suggest that operational transients are calculated with higher accuracy than design basis accidents. This was shown also in the past in the quantitative analysis of main circulation pump (MCP) trip transient in Mochovce VVER [10].

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DETERMINISTIC SAFETY ANALYSIS OF OPERATIONAL EVENTS IN UKRAINIAN NUCLEAR POWER PLANTS

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Abstract. At present time a problem of obtaining from NPPs the precise information on operational events in reactor facility, which used then for performing the safety analysis and validation of input models is very topical. The comparative analysis of data from data-processing system (DPS) acquired upon operational events on reactor facilities VVER-1000 (V-320) of KhNPP unit 1 and 2 is cited in given report. Tripping of one and two feed-water pumps (FWP) were selected as operational events. Data of KhNPP unit 1 were acquired from DPS of outmoded type (output of eighties). The problem of KhNPP unit 1 data is in large discontinuity of derived data upon operational events and it has a great effect on accuracy of calculations. Data of KhNPP unit 2 were acquired from modern DPS, which was installed and modernized directly before unit's setting in operation in 2002-2004. In this case we have practically continuous array of data on operational event. Since the first fuel load on KhNPP unit 2 improved fuel assemblies (TVSA) are used, some of which have an embedded temperature detectors. Improved construction of fuel assemblies allow to define the temperature fields and distribution of energy-release in active core more exactly. The data of real operational events were also compared with data acquired during the calculation of presented operational events with use of thermohydraulic code RELAP5. The results of given analysis are used for input models of presented reactor facilities validation and allow to conclude that model of KhNPP unit 2 is performing the analysis of different operational events more precisely. It is possible to make a conclusion as a result of performed comparative analysis to the effect that accuracy and quantity of data are making a great effect on accuracy of further safety analysis of corresponding NPP units and also about the necessity of modernizing of outmoded DPSs for the purpose of increase of calculation accuracy hence for increase of unit's safety in whole.

1 INTRODUCTION

At present time a problem of obtaining from NPPs the precise information on operational events in reactor facility, which used then for performing the safety analysis and validation of input models is very topical.

A lot of factors have an impact on information accuracy: accuracy of measuring systems, operating speed and accuracy of computer systems, etc. Today as a result of strong development of computer technologies the question of operating speed and accuracy of data processing with use of data processing systems (DPS) is not so critical. However it should be mentioned that on NPPs that were built even 8-10 years ago are staying vastly outmoded to today's measures DPSs which work have a great impact on quality of obtained data on operational events. On Ukrainian NPPs are used such outmoded DPSs mainly with the exception of two recently placed into operation units on Rivnenskaya and Khmelnytskaya NPPs.

An influence of quality of data obtained on NPPs from DPSs of different type on the results of validation of RELAP5 models and performing of initial events analysis are considered in given report. Steady-state and operational events validation of KhNPP unit 1 and 2 (outmoded and new DPS correspondingly) were taken up in report. As operational events were chosen ones occurred on mentioned power units: tripping of two feed-water pumps (FWP) on Unit 1 and of one feed-water pump on Unit 2.

There are an objective lack of information on physical phenomena occurred on NSSS, which may be filled up with following:

(1) Using experimental facilities (EF) (particularly integral full-scope facilities). However, cost of creation and performing of experimental researches is rising with EF approaching to full-scale power unit, but from the other side under use of scaling relatively to full-scale object the scaling effects are making a great contribution in accuracy of data extrapolation.
(2) Using data on transients occurred on functioning power units, especially obtained on phase of starting-up and adjustment.

Considering difficulties of access to information on EF for Ukrainian NPPs, absence of detailed information from NPP control equipment is also adversely affect on possibility of using of mentioned data for validation of calculation models without recourse to use of information integral test facilities. In support of this turn up the fact that presence of great amount of cumulative statistical material on incidents at NPPs with VVER-type reactors (excerpts may be extended at the expense of PWR reactors) didn't give an impartial picture of event's passing.

Such conclusion could be made after learning the database of OJSC "KIEP" on accidents at reactor facilities VVER-1000 VVER-440 containing information on incidents during 688 reactor-years of operation of facilities with VVER-type reactors all over the world. From practical point of view available database is having value only with relation to the determination of accidents initial events frequencies for certain reactor types or with relation to study an operational experience and possible prevention of already occurred events by means of taking up the technical-organizational or design decisions. In the view of using of available information for improvement of NSSS calculation models this information is fully inapplicable on account of its scantiness, that mainly related with the low level of documentation of initial events and also with total lack of graphic documentation. Such situation is observing also in database AIRS supported by IAEA, what obviously arise from absence of special demands on level of precision of information.

2 STEADY-STATE CALCULATION

The results of validation steady-state calculations of KhNPP Unit 1 and 2 RELAP5 input models are presented in this section considering changes included in models by the results of validation.

2.1 KhNPP Unit 1 and 2 steady-state calculation results

Basic calculation characteristics of unit's steady-state and rating values of proper operation parameters [1, 2] are presented in Table 1.

TABLE 1. KHNPP UNIT 1 AND 2 STEADY-STATE CHARACTERISTICS

Parameter	**Parameter value**		
	Design data	**Calculation data (Unit 1)**	**Calculation data (Unit 2)**
Reactor heat power, MW	3000+60	3000.0	3000.0
Primary pressure, MPa	15,68 ± 0,3	15.62 - 15.82	15.6- 15.8
Pressurizer pressure, MPa	15,696 ± 0,294	15.59 - 15.78	15.5- 15.7

Parameter	Parameter value		
	Design data	Calculation data (Unit 1)	Calculation data (Unit 2)
Pressurizer level, m	8,77 ± 0,15	8,77±0,11	8.71-8.84
Reactor flow rate, m^3/h	$84800\ ^{+4000}_{-4800}$	85546.0	84951.0
Coolant temperature at the entry point of reactor, ^oC	289,7	290,3	289.4 – 289.6
Coolant temperature at the exit of reactor, ^oC	320	320,2	319.5 – 319.7
Coolant heating, ^oC	30,3	29,89	30.2
Reactor pressure fall, MPa	0,381+0,04	0.3725	0.3825
Active core pressure fall, MPa	0,15	0.143	0.141
MCP pressure fall, MPa	0,66	0.595	0.604
SG pressure fall (in the first circuit) , MPa	0,123	0.136	0.132
Pressure in SGs, MPa	6,3 ± 0.2	6.31–6.33	6.28–6.30
Pressure in main steam drum, MPa	5,98±0,98	5.96	5.98
Water level in SGs, m:			
– cold bottom, 4m range	2.25	2.238	2.18
– level by 1 m range	0.27 ± 0.05	0.27	0.27
Coolant temperature at the entry point of SG, ^oC	320	319.8	319.6
Coolant temperature at the exit of SG, ^oC	289.7	289.6	289.2
Feed water temperature, ^oC	220 ± 5	219.65	220.0
SG steam capacity, kg/s	408.3 ± 28.6	409	410
Total SG capacity, t/h	5870	5853.6	5889.6

The analysis of calculation results reveals that all basic heat engineering, hydraulic and neutron-physical parameters of model are in close fit with design data. This is an indirect confirmation of adequacy of principles and approaches assumed under development of nodalization scheme and basic systems and regulators of KhNPP Unit 1 and 2 models.

3 DYNAMIC VALIDATION OF MODELS

Major task of validation is the identification of calculation model capability to model adequately the different physical phenomena. Since validation is carried out by means of comparison of calculation results with data acquired from measuring devices the necessity in comparison criteria for model capability evaluation arise.

3.1 Two feed-water pumps tripping on KhNPP Unit 1

This accident occurred at KhNPP Unit 1 on 18.11.1997 [3]. Reactor heat power at time of initial event reach 3000 MW. The initial event is the failure in output scheme of FWP-1,2 leading to pumps unloading and correspondingly to abrupt decrease of level in all four SGs. In given section the brief description of incident is presented [3], made up on basis of KhNPP technical documentation and archival data of DPS “Complex Titan 2”.

Phenomenologically given incident can be divided into three phases:

- First phase is characterized by abrupt decrease of SG-1-4 level as a result of FWP tripping with further FPP signal conditioning. Abrupt decrease of SGs level leads to AFWP engaging, MCP tripping and reactor flow rate decrease, emergency protection actuation, TSV closing. This phase lasts first 80 s.
- Second phase(80-200 s) is characterized by abrupt change of heat-hydraulic parameters of first and second circuits as a result of TSV closing.
- Third phase (from 200 s) is characterized by FASDS-C closing (RC11S01 and RC12S01,02), work of AFASDS, first and second circuit parameters stabilization and also SGs level renewal.

In calculation are defined following boundary conditions peculiar to this operational event:

- FWP coastdown was modeled by assignment of coastdown curve, which points were selected from the condition of compliance with chronology of operational event development:: MFWR full opening, moment of signal forming by pressure at FWP head.
- AFASDS work, which wasn't embodied in base model, was modeled as the boundary conditions with use of DPS data.

Calculation time of transient is 1000 s.

In compliance with phenomenology of operational event, acceptance criteria for dynamic validation of calculation model can be formulated in the following way:

(1) For the first phase:
- time intervals of FWP coastdown and of basic regulators operation comply with incident plant data;
- time intervals and turndown of primary and secondary pressure comply with incident plant data;
- time intervals and turndown of reactor power comply with incident plant data;
- time intervals and turndown of SGs levels comply with incident plant data.

(2) For the second phase:
- values and changes of primary coolant temperature comply with incident plant data;
- values and changes of primary and secondary pressure comply with incident plant data;
- time intervals and turndown of Prz and SG levels comply with incident plant data.

(3) For the third phase:
- stabilization of primary and secondary pressure;
- SGs levels renewal.

3.1.1 Calculation results

The dynamics of RELAP5 model parameters in comparison with parameters which were registered in database of DPS "Complex Titan 2" on KhNPP Unit 1 during operational event represented at Figures 1–7. Fact of beginning of FWP unloading assumed as the beginning of operational event.

Chronological consecution of events in the incident and in calculation is cited in Table 2.

TABLE 2. EVENT'S TIME COMPARISON IN CALCULATION AND INCIDENT

Event	**Times**	
	Measurement	**Calculation**
Beginning of calculation	0	0
Main feed-water controller full opening	19	16
SG level decrease on 50 mm from nominal	24	28
RPS signal on FWP tripping by pressure behind adjusting FWP stage	28 (27)	28
FPP snapping on signal of FWP, PP-1, power controller tripping	29	28
AFWP-1,2 engaging by decrease of SG-1-4 level on 100 mm from nominal	33	36
Consecutive MCP-2,3,4 tripping by SG level decrease on 500 mm from nominal	58-60	58
SCRAM on tripping three of four MCPs	61	62
MCP-1 tripping by SG level decrease on 500 mm from nominal	65	62
Pressurizer heaters second group turning on	23	32
Pressurizer heaters third group turning on	25	33
Pressurizer heaters fourth group turning on	28	34
TSV closing	153	134
Pressurizer heaters fourth group turning off	600	450
Pressurizer heaters third group turning off	750	480
Pressurizer heaters second group turning off	never	535
Pressurizer heaters first group turning off	never	never
End of calculation	-	1000

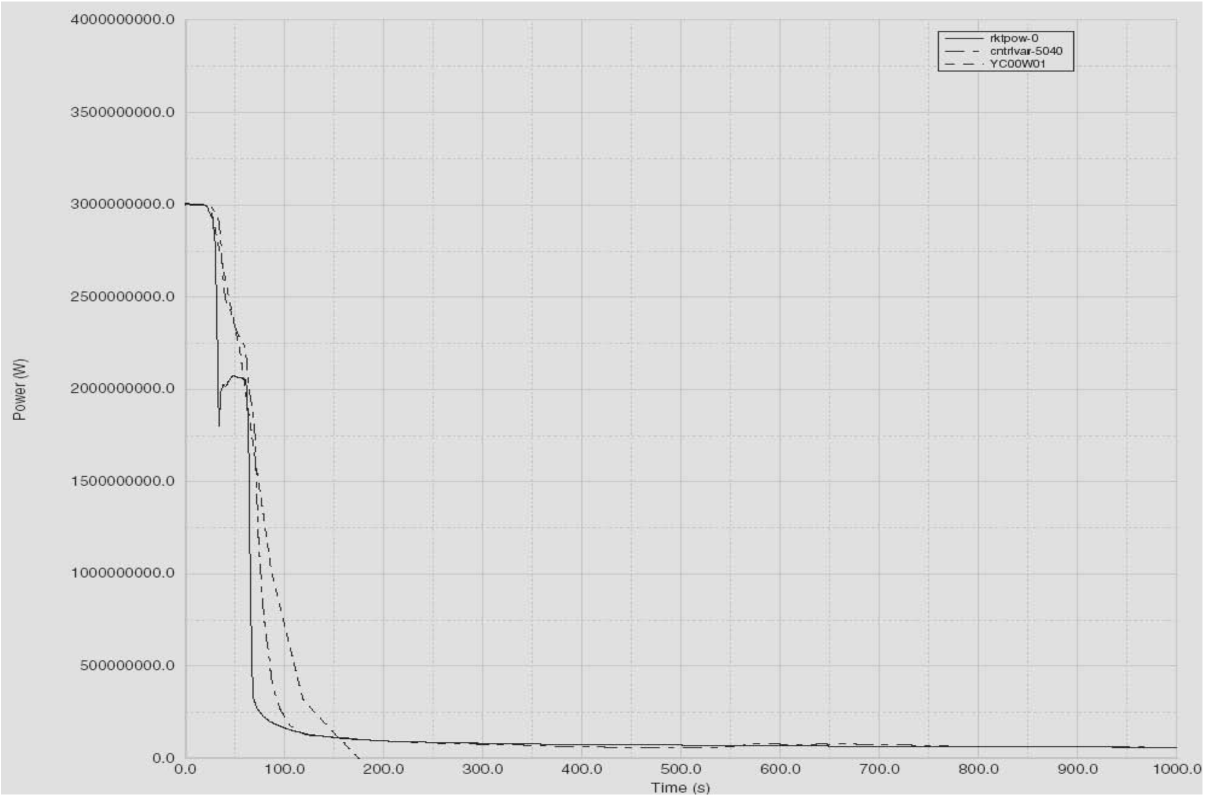

Fig. 1. Reactor heat power.

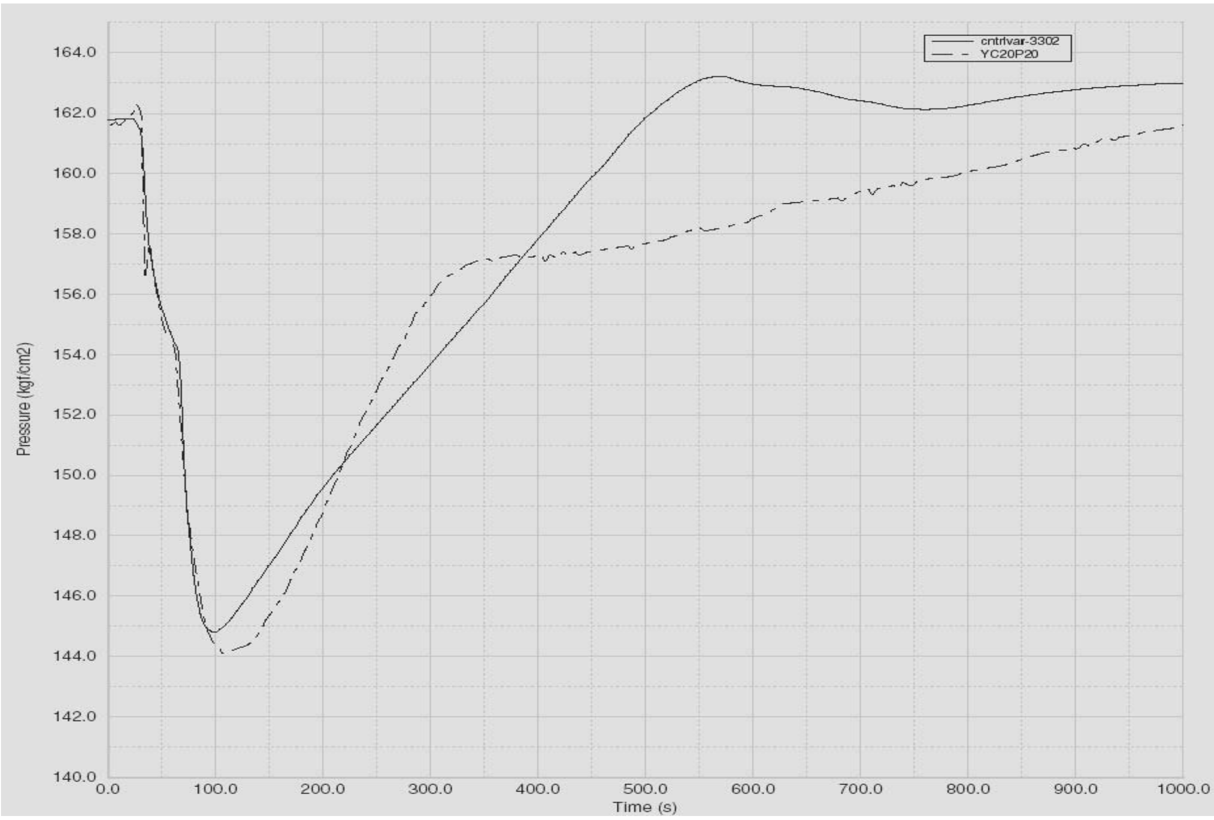

Fig. 2. Primary pressure.

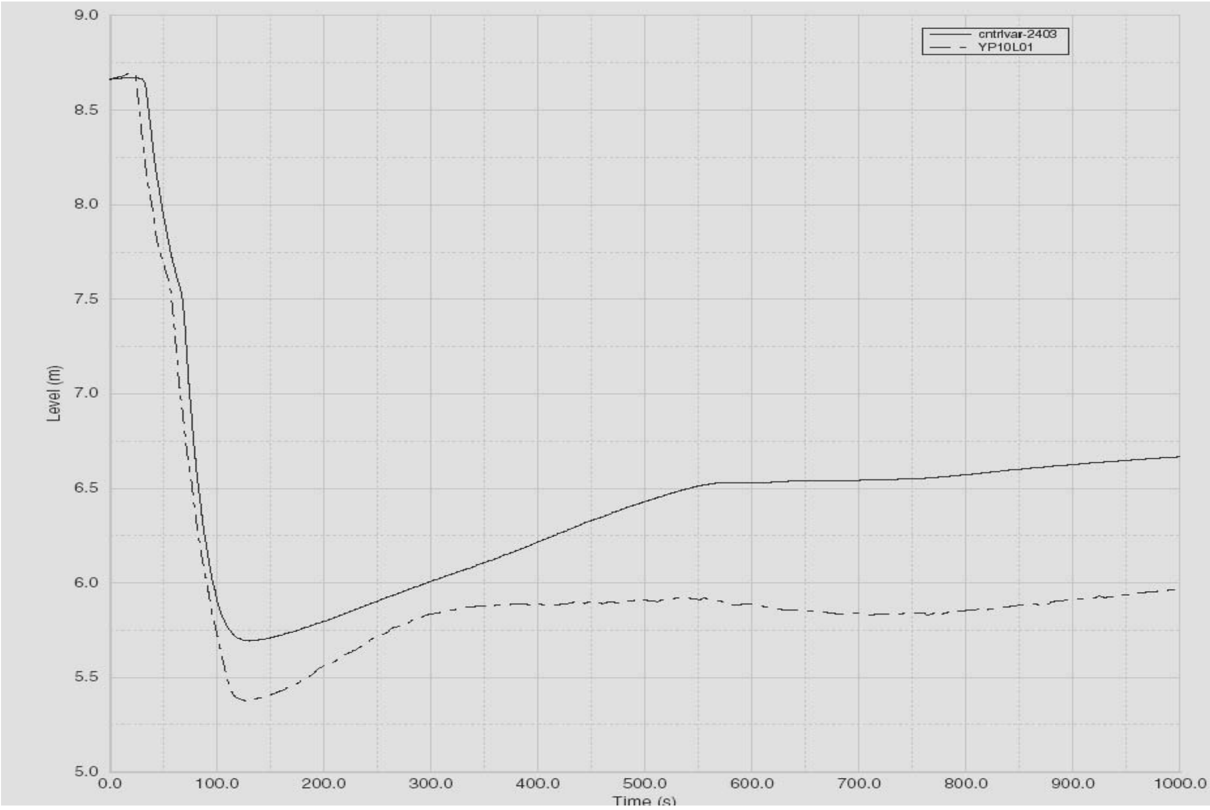

Fig. 3. Pressurizer level.

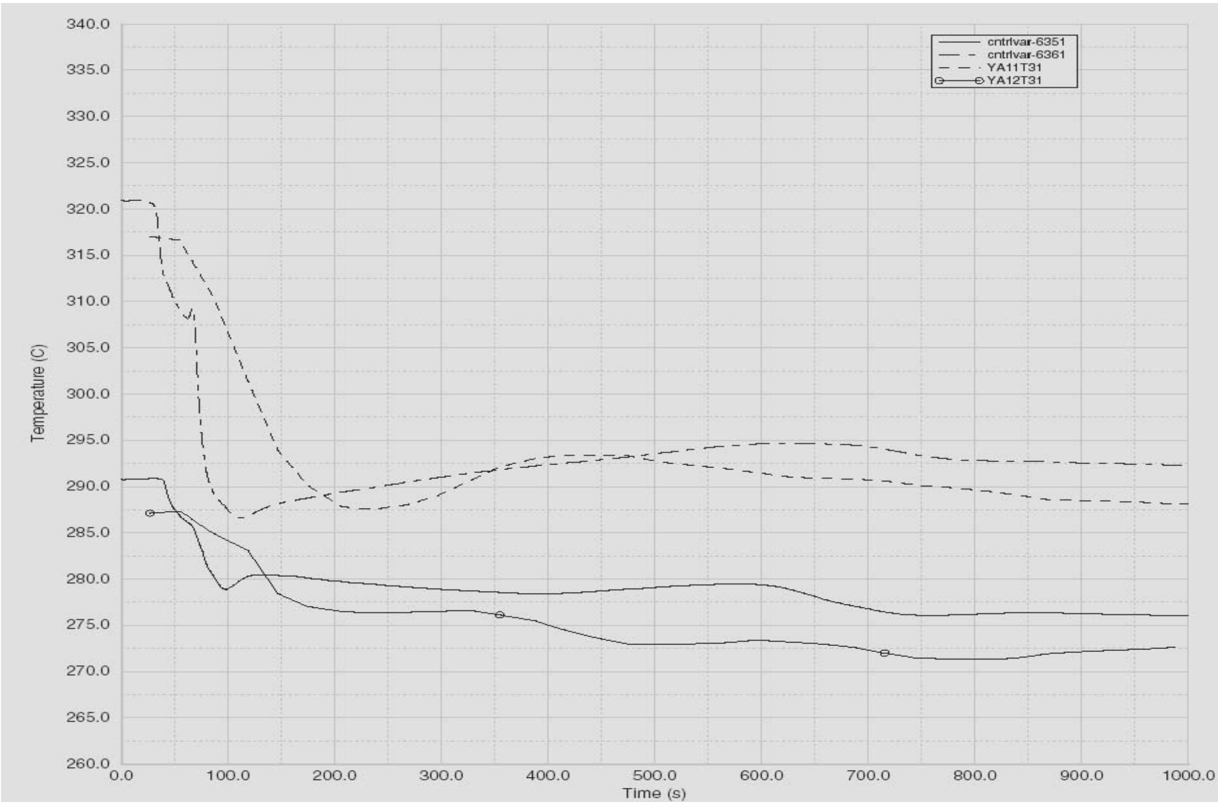

Fig. 4. Temperature of first loop.

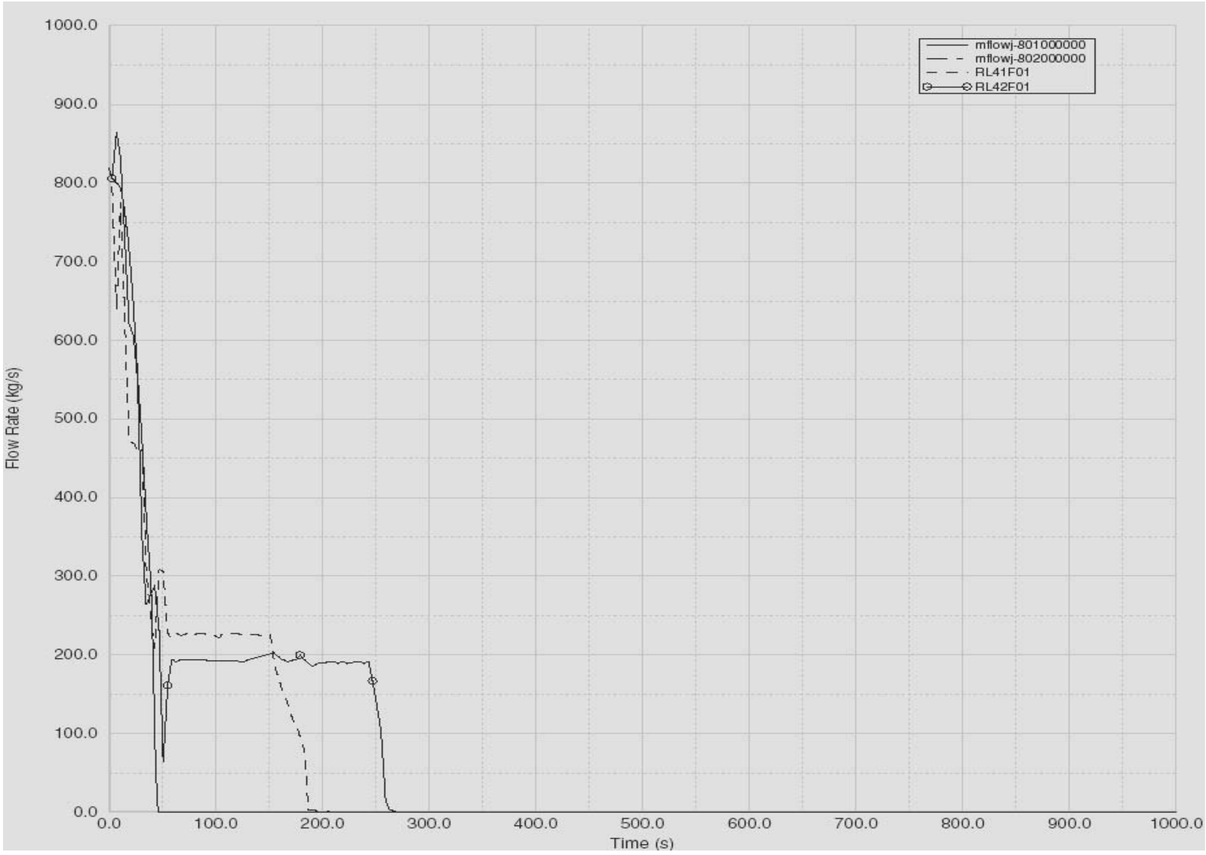

Fig. 5. Feed-water pumps work.

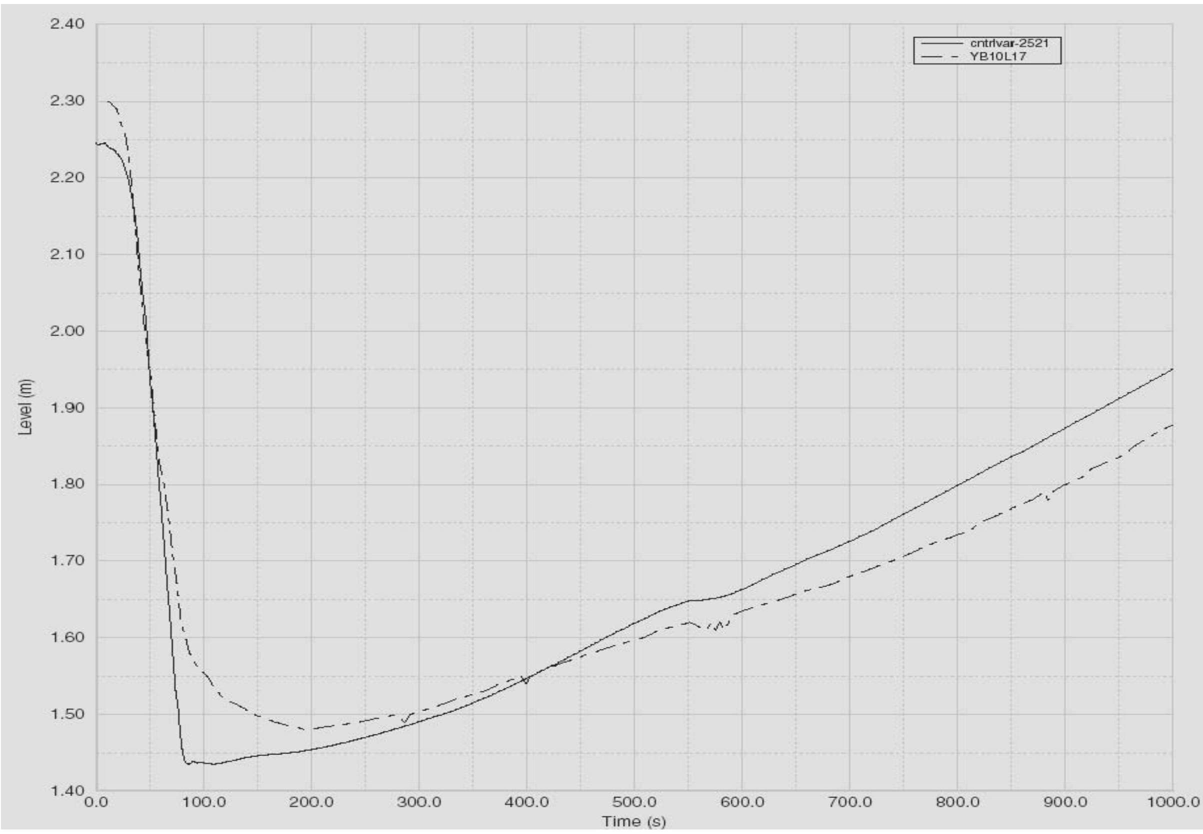

Fig. 6. SG-1 Level (base 4 m).

Fig. 7. Pressure in SG-1.

3.1.2 Conclusions on operational event validation

Relying on analysis of results of transient calculation it is possible to conclude that assumed acceptance criteria are satisfied in whole and that model adequately reconstruct physical phenomena occurred on reactor facility and also the work of systems and equipment of power unit. Absence of operation of systems, which were included in model, allows to judge about correct of given systems modeling.

It should be mentioned that under carrying out the calculations their results were compared with data of DPS "Complex Titan 2" (diagrams of power unit analog parameters history and also data on change of state of discontinuous parameters).

3.2 One of two feed-water pump tripping on KhNPP Unit 2

3.2.1 Description of events consecution

Experiment on one FWP tripping at KhNPP Unit 2 was realized on 07.12.2004 according to [4]. Consecution of events, which take place during operational event, is represented below:

21:00:54	FWP-1 tripping by operator from BCP
21:00:56	FPP signal conditioning
21:00:58	Pressurizer heaters third group turning on
21:01:03	Pressurizer heaters fourth group turning on

21:01:09	AFWP-1,2 engaging
21:01:12	AFASDS-1 opening
21:01:16	AFASDS-2 opening
21:01:57	AFASDS-2 closing
21:02:21	PP-1 signal conditioning
21:02:49	PP-1 signal releasing
21:06:37	Pressurizer heaters fourth group turning off
21:10:10	Pressurizer heaters third group turning off
21:11:15	Pressurizer heaters second group turning off
21:17:34	1000 s from the beginning of transient – range of analysis

3.2.2 Calculation results

Events consecution acquired in calculation and occurred on power unit under operational event are presented in Table 3.

TABLE 3. EVENT'S TIME COMPARISON IN CALCULATION AND EXPERIMENT

Event	**Time**	
	Measurement	**Calculation**
FWP-1 tripping by operator from BCP	0	0*
FPP signal conditioning	2	2
Pressurizer heaters third group turning on	4	6
Pressurizer heaters fourth group turning on	9	7
AFWP-1,2 engaging	15	11
AFASDS-1 opening	18	18*
AFASDS-2 opening	22	22*
AFASDS-2 closing	63	63*
PP-1 signal conditioning	87	79
PP-1 signal releasing	115	94
Pressurizer heaters fourth group turning off	343	330
Pressurizer heaters third group turning off	616	540
Pressurizer heaters second group turning off	681	wasn't turned off
End of calculation	1000	1000
* Event modeling by measured data		

The dynamics of NSSS model parameters in comparison with parameters which were registered by DPS on KhNPP Unit 2 during the transient are represented at Figures 8–14. Beginning of transient is the FWP-1 tripping.

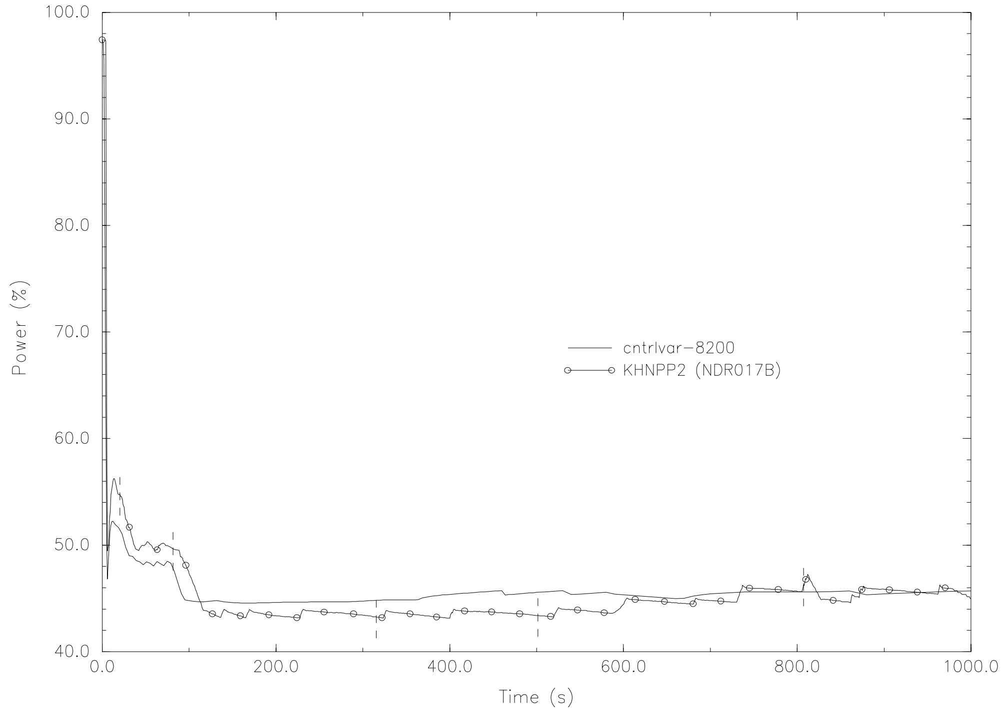

Fig. 8. Neutron power.

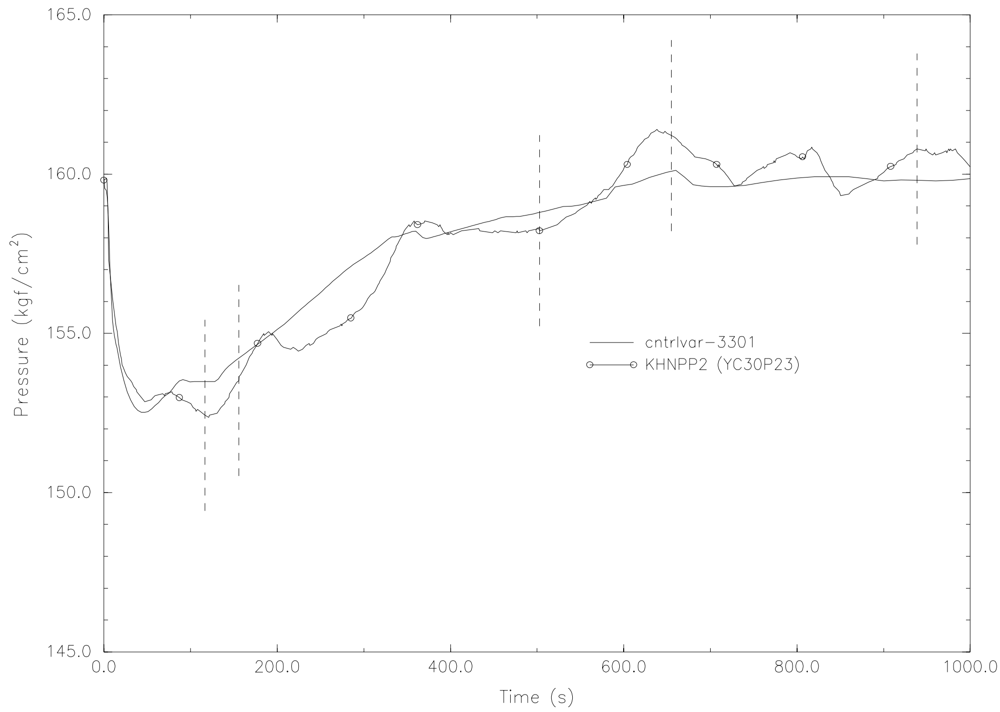

Fig. 9. Primary pressure.

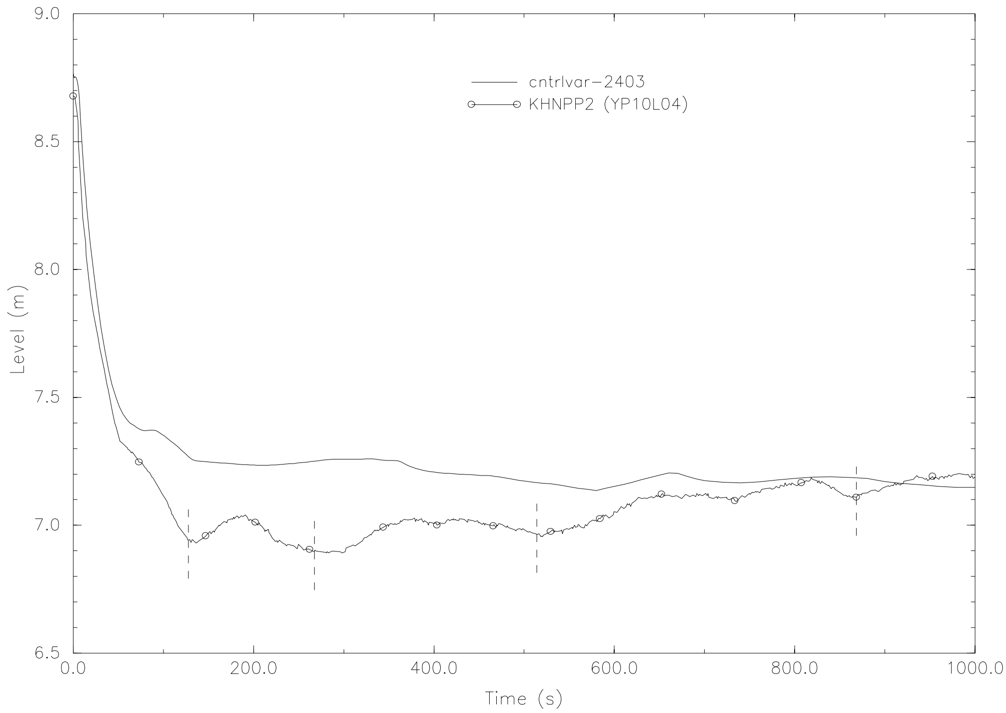

Fig. 10. Pressurizer level.

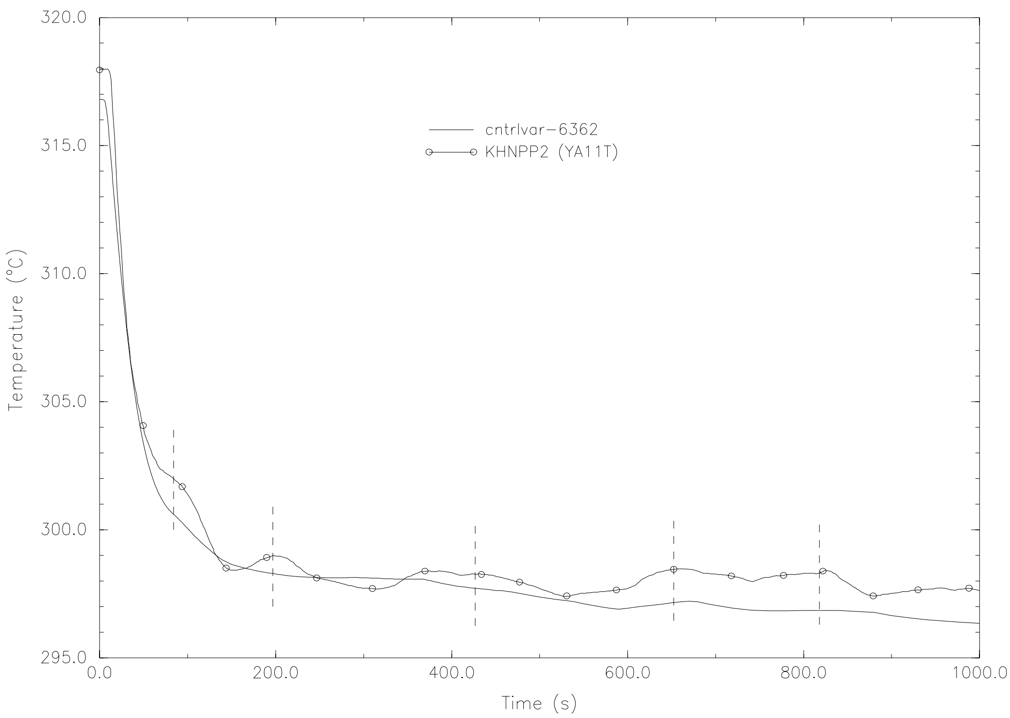

Fig. 11. Coolant temperature in hot leg of first loop.

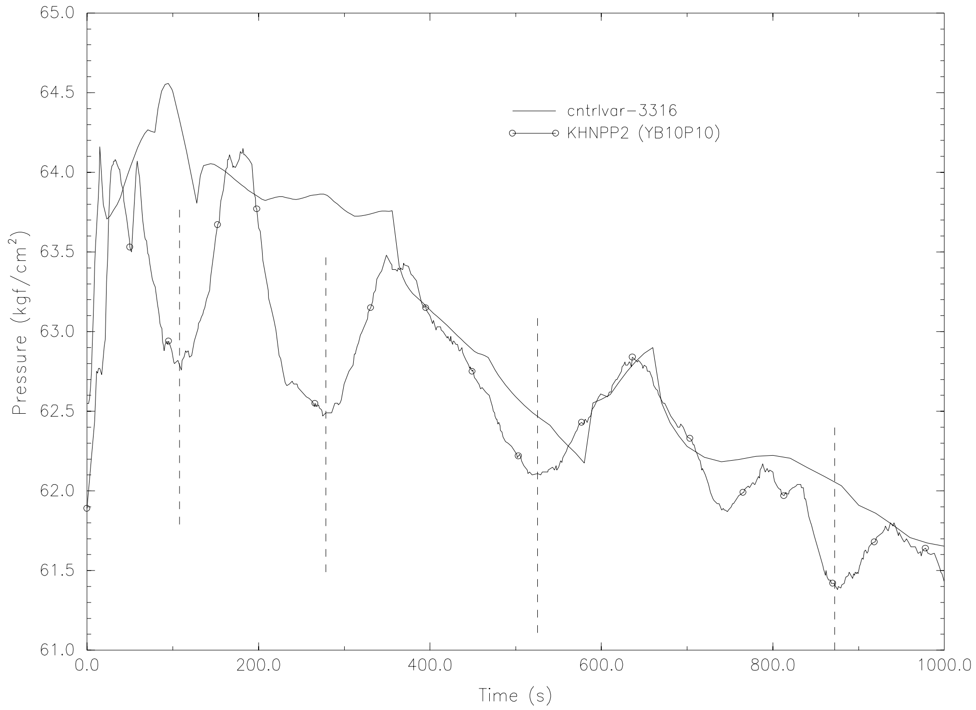

Fig. 12. SG-1 Pressure.

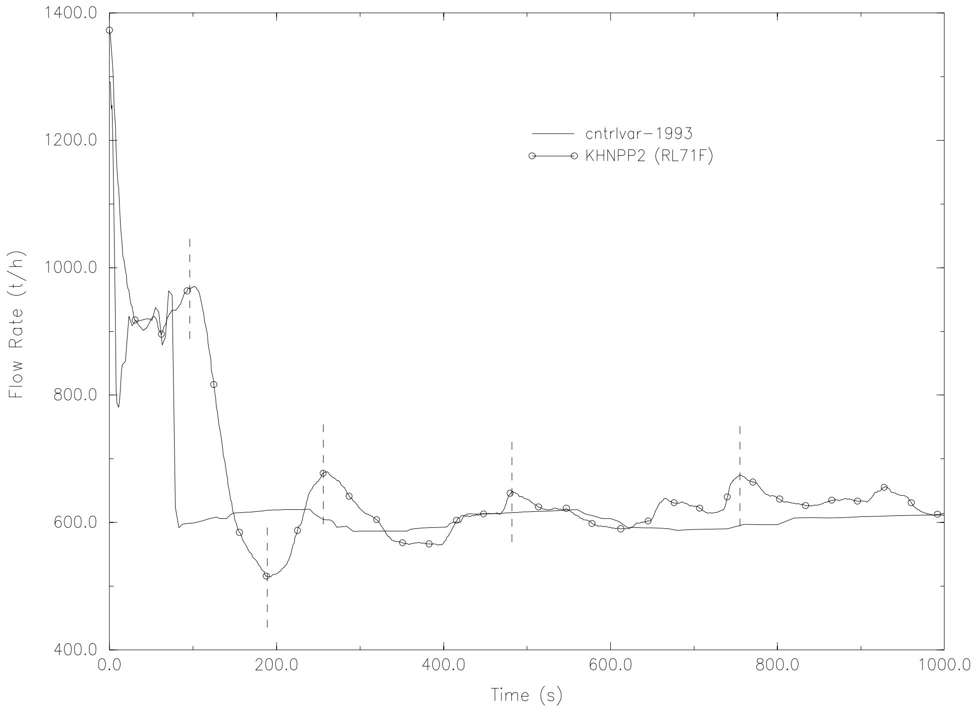

Fig. 13. Feed Water Flow Rate into SG-1.

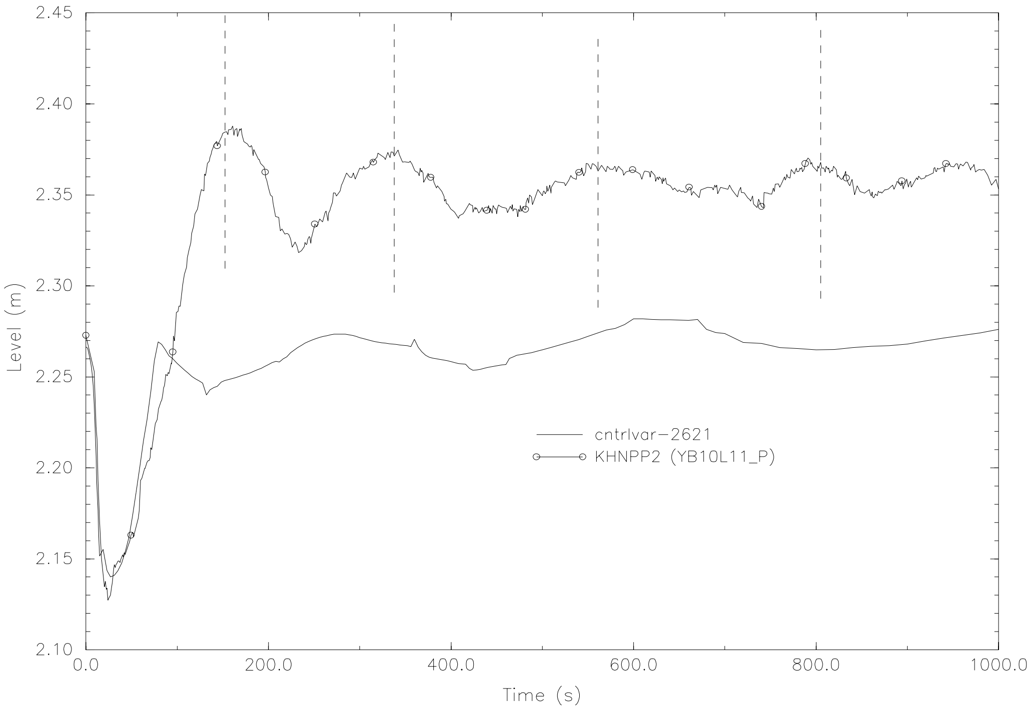

Fig. 14. SG-1 level (base 4 m).

3.2.3 Conclusions on operational event validation

Specific events sequence of chosen incident allowed checking of the accuracy of work of power unit systems, namely:

- to check the accuracy of interlocks and protection of first and second circuits actuation;
- to assess the correctness of used FWP and AFWP models, particularly FWP coastdown model;
- to check reactor power control models (RPL, power controller, emergency protection) and model of pressure compensation system (pressurizer heaters;
- to check the accuracy of SG and Prz level meters modeling;
- to assess the correctness of feed water regulators modeling (MFWR and AFWR);
- to check the accuracy of turbine controller model operation.

Relying on analysis of results of transient calculation it is possible to conclude that assumed acceptance criteria are satisfied in whole and that model adequately reconstruct physical phenomena occurred on reactor facility and also the work of systems and equipment of power unit. Absence of operation of systems, which were included in model, allows judging about correctness of given systems modeling.

However, during calculations were revealed following discrepancies between the calculation results and incident data:

(1) Calculation values of Prz pressure and level in the range 100-300 s are differ slightly from the observed ones in the greater side. It is connected with dynamics of heat sink

from the first to second circuits and with power of Prz heaters which were assumed in model.

(2) Differences in primary pressure increase dynamics in the range 300-600 s related with features of thermalphysic processes modeling in liquid volume of Pressurizer peculiar to one-dimensional codes on the whole. At the same time the calculation parameters have a deviation in the greater side, i.e. have some conservatism in comparison with observed ones at power unit.

So the model of KhNPP Unit 2 provides sufficient compliance of results with experimental data during this operational event.

4 GENERAL CONCLUSIONS

As a result of transient calculations a following total conclusion can be made: assumed acceptance criteria are satisfied in the whole and model adequately reflects physical phenomena occurring on reactor facility and also operation of systems and equipment of power unit.

On the basis of analysis of calculation results of analyzed operational events a conclusion can be made that assumed acceptance criteria are satisfied in the whole and that model adequately represents physical phenomena occurred on reactor facility and also the operation of systems and equipment of power unit.

However, during performing of the calculations following weaknesses of model which result in divergences between the results of calculation results and incident data were revealed:

- Because of reductive modeling of SG feed component (particularly the output regulator and FWP coastdown weren't realized) the feed-water flow rate was set artificially so that the calculation value correspond to data of database DPS "Complex Titan 2".
- Deviations up to 14 % under SG level changes by level meter with 1 m base. It is obviously connected with general problem of heat-hydraulic processes in SG modeling on low levels of power and with features of modelling of compound surge tanks.
- Deviations up to 20 % on position of CV FASDS-C rod during unit unloading that for one's turn determined the divergences on secondary pressures after TSV closing. This discrepancy is determined by accepted reductive approximation of four FASDS-C modeling which operate synchronously by general setting.
- Model doesn't take into account the asymmetry of steam flow from single SG on turbine at low levels of power and it is connected with modeling of four TSV as of one element. This also resulted the discrepancy in time of final TSV closing.

From the obtained data on both operational events Prz heaters groups behave more dynamical that ones on power unit what in given case leads to increase of primary pressure and to more conservative results (i.e. allows to avoid the optimistic estimations in part of basic parameters behaviour what not less appreciable also under using best estimate approach). However it should be mentioned that relative errors of primary pressure is within 3% that is acceptable for similar analysis.

Discrepancies are determined by differences in logic of operation of digital regulators of FASDS-C and by timing data of fit and nodalization of TSVs and also by neglecting of AFASDS operation.

Considering all obtained results a conclusion can be made that KhNPP unit 1 model for code RELAP5 is suitable for analysis performing for obtaining a success criteria within the frameworks of first level PSA and after proper alteration for DBA analysis.

It also should be mentioned that existing on KhNPP unit 1 DPS “Titan 2” was placed into operation at the moment of unit’s putting into operation in 1988 and on this moment it become technically obsolete. This system in steady-state under small changes of parameters gives satisfactory data on NPP operation, however, because of slow enough processors (i286) under transients reveals great discontinuity of output data that raise a question of its adequacy and hence of model validation accuracy and accuracy of further analysis.

Calculation results analysis of steady-state had presented that developed input model of KhNPP Unit 2 for code RELAP5 reflects adequately all real physical processes and phenomena provided by reactor facility design. All basic heat engineering, hydraulic and neutron-physic parameters of model are in good compliance with design data. This is an indirect confirmation of approaches accuracy which were assumed under development of nodalization scheme and of basic systems and regulators of KhNPP Unit 2 model.

On the results of dynamic validation of thermohydraulic KhNPP Unit 2 model it is possible to draw following conclusions:

(1) Model was developed on the basis of design data and complies with design nominal parameters of reactor facility.
(2) Conducted analysis displayed that calculation parameters of first and second circuits have **perfect** and **sufficient** conformity according to criteria of this validation.

Model appears to be adaptable from the point of view of new initial and boundary conditions assignment and also of its adjustment and further working out in detail depending on purposes of executable tasks.

KhNPP Unit 2 DPS was placed in operation on the phase of unit’s putting in operation in 2002-2004 and therefore it has sufficient processing speed for adequate representation of processes not only in steady-state but also during different operational events. Discontinuity of data in this case is small enough that give an opportunity to talk that Unit 2 model corresponds to real behavior of reactor facility during operational events and it can be used for further analysis of the processes on KhNPP Unit 2. Since the first fuel load on KhNPP unit 2 improved fuel assemblies (TVSA) are used, some of which have an embedded temperature detectors. Improved construction of fuel assemblies allow to define the temperature fields and distribution of energy-release in active core more exactly. Use of TVSA also affects kindly on accuracy and completeness of data used for validation of calculation models.

As results of comparison of KhNPP Unit 1 and 2 dynamic validations it is possible to conclude that for increasing of quality and accuracy of safety analysis of all Ukrainian NPPs a replacement of outmoded DPSs on DPSs of new type is necessary and it will be made in the nearest future.

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LIST OF ACRONYMS

AFASDS	-	Auxiliary Fast Acting Steam Dump System
AFWP	-	Auxiliary Feed-water Pump
AFWR	-	Auxiliary Feed Water Regulator
BCP	-	Block Control Pane
CV	-	Control Valve
DPS	-	Data-processing System
EF	-	Experimental Facility
FASDS-C	-	Condenser Fast Acting Steam Dump System
FPP	-	Fast Preventive Protection
FWP	-	Feed-water Pump
KhNPP	-	Khmelnytskaya Nuclear Power Plant
KIEP	-	Kiev Institute “Energoproject”
MCP	-	Main Circular Pump
MFWR	-	Main Feed Water Regulator
NPP	-	Nuclear Power Plant
NSSS	-	Nuclear Steam Supply Station
OJSC	-	Opened Joint-Stock Company
PP	-	Preventive Protection
Prz	-	Pressurizer
RPS	-	Reactor Protection System
SG	-	Steam Generator
TSV	-	Turbine Stop Valve
TVSA	-	Improved Fuel Assembly
VVER	-	Pressurized Water Reactor

DETERMINISTIC ANALYSIS OF EVENTS FOR EVALUATING SAFETY MARGINS AT NPPS – A REGULATORY PERSPECTIVE

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Abstract. Among the various efforts to improve operational safety of nuclear installations, systematic collection, evaluation and feedback of operational experience are considered valuable and effective. Such a system enables all safety related events to be analyzed for determination of the root causes and necessary corrective and preventive action to be taken to avoid their recurrence and to enhance operational safety at Nuclear Power Plants.
The goal of event investigation is to improve overall plant safety and reliability of operations by learning from experience. Although use of probabilistic methodologies is gaining pace, the regulatory authority in Pakistan (PNRA) mainly relies on deterministic analysis for operational safety at NPPs. This covers the determination of safety margins and root causes of events and the necessary evaluations for verification of results of assessments conducted by the NPPs. This paper provides a general overview of events reported by NPP licensees to PNRA and includes examples of evaluations carried out by PNRA with the regulatory perspective. Among the various evaluations carried out by PNRA, one such evaluation of operational events included in this paper as an example relates to control rod drop time of our 300 MW(e) NPP at Chasma in Pakistan. In this event the drop time, during the test, was found to be higher than the required drop time i.e. <2 seconds for two control clusters. For this case, evaluation of the neutronic analysis for most significant design bases condition was conducted by PNRA to assess the safety margins. In addition, PNRA also conducted analysis of the thermal hydraulic characteristics in the core and possible flow induced effects on mechanical components which included determining the effects of transverse flow forces and fluid densities at various temperatures on slow drop movement of the rods.

1 INTRODUCTION

Every engineering activity carries some potential of deviation from normal operation, resulting in events, which could be unexpected and may result in undesirable risk or consequences. To avoid such risks, study and evaluation of abnormal events is considered essential and the depth of evaluation depends on the severity of consequences attached to the activity. Evaluation of events is carried out either by mathematical modelling or by information gathered from operational experience of components, equipment and systems or both.

In the engineering activity as complex as a nuclear power plant, experience of operation is a valuable source of information for learning and improving the safety and reliability of systems. This along with the evaluation of systems provides for introducing necessary modifications for monitoring control and mitigation of abnormalities leading to potential events and to prevent those from becoming accidents. It is essential to collect information in a systematic way that meets agreed thresholds for reporting on events occurring at plants during commissioning, operation, surveillance and maintenance activities, and on deviations from normal performance by systems and personnel which could be precursors of accidents.

It is extremely important that all concerned specifically operators and regulators should acquire an adequate event investigation program and that a multidisciplinary group of trained investigators exists within the respective organizations. It is also vital that senior utility and senior regulatory managers are fully supportive of the program and allocate adequate, dedicated resources to operating experience activities, including the event investigation process.

Information regarding the events reporting requirements and the process adopted by the regulatory body for analysis has been further elaborated by examples.

2 OBJECTIVE

The objective of this paper is to share experience on deterministic analysis of NPPs events with regulatory perspective at international level which include screening, analysis and trending of safety significant events at NPPs.

3 REGULATORY REQUIREMENTS FOR EVENTS REPORTING

According to PNRA Regulations on the "Safety of Nuclear Power Plants Operation" (PAK/913, Rev. 1), clause 19, the holder of an operating license for a nuclear power plant (licensee) in Pakistan has to submit the following event reports.

- An event notification report (ENR) within 24hrs to the regulatory body containing requirements like event title, date and time, operational mode, operating pressure and temperature, brief description, immediate action taken and event consequences (Example of ENR report format used by CNPP is attached as Annex 1).
- A detailed event report for any reportable event within 60 days after the discovery of the event.
- The Detailed Event Report contains:
  - A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence.
  - A clear, specific, narrative description of what occurred so that knowledgeable readers conversant with the design of commercial nuclear power plants, but not familiar with the details of a particular plant, can understand the complete event.
  - The narrative description must include the following specific information as appropriate for the particular event:
    - Plant operating conditions before the event;
    - Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event;
    - Dates and approximate times of occurrences;
    - The cause of each component or system failure or personnel error, if known;
    - The failure mode, mechanism, and effect of each failed component, if known;
    - For failures of components with multiple functions, include a list of systems or secondary functions that were also affected;
    - For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service;
    - The method of discovery of each component or system failure or procedural error
    - For each human performance related root cause, the licensee shall discuss the cause(s) and circumstances;
    - Automatically and manually initiated safety system responses.
  - An assessment of the safety consequences and implications of the event.

- Event causes which include direct or apparent cause, root cause and contributory cause.
- A description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future.
- Reference to any previous similar events at the same plant that are known to the licensee.
- The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information concerning the event and the plant's characteristics.

PNRA sometimes require the licensee to submit specific additional information if PNRA finds that supplementary material is necessary for complete understanding of an unusually complex or significant event. These requests for supplementary information is made in writing.

The reports and copies that licensees are required to submit to PNRA must be of sufficient quality to permit legible reproduction and micrographic processing.

4 SCREENING OF EVENTS

Screening of event information is undertaken to ensure that all significant safety relevant matters are considered and that all applicable lessons learned are taken into account. The screening process is performed to select events for further detailed investigation and analysis. This includes prioritization according to safety significance and recognition of adverse trends.

The quality of screening depends, in part, on engineering judgment, therefore highly experienced and knowledgeable personnel are assigned to this task. Operating organizations have the objective of safe operation, plant availability and commercial performance by identifying the causes of events and thereby avoiding their recurrence by taking necessary corrective, preventive and predictive measures. PNRA reviews the screening of events to have an oversight that can be used to reflect in the inspection program, licensing activities, elaboration of regulations and requirements for safety back fits. Vendor companies can take advantage form the operational experience feedback data to improve their design and manufacture of structures, systems and components.

4.1 The plant level screening

At the plant level, two sources of information are available: internal and external operational experience. Internal events are those that occur at the plant. External information is the experience coming from other plants, either from the same country or a foreign one.

The screening of internal events is carried out promptly to rank the priorities in the event feedback process and follow-up actions. External information is reviewed to determine whether it is applicable for their plant. The inputs considered applicable are provided to the specific groups in the plant (e.g. operations, maintenance) for investigation or information. Results of the screening at a plant level are recorded for evaluation during subsequent periodic self-assessment or peer review. History of the internal as well as external event screening process is maintained and made available to PNRA.

4.2 Regulatory authority level screening

The regulatory body inspects the operating organization/licensee's screening process in order to ensure that the screening is effective in identifying events for analysis. The regulatory body has a more strategic role and monitors the operational experience feedback process to ensure that it is effectively conducted by the operating organizations.

The screening process consists of:

(1) Review of the event reports for immediate implications on the safe operation of NPPs.

(2) Determination of significance of events for impact on plant safety and availability (consequences and ability to learn the lesson.

(3) Review against established thresholds, consistent with the significance of the event, which determine the depth of analysis (or for instance, if only trending should be carried out or a full root cause analysis conducted).

5 INVESTIGATION OF EVENTS

Abnormal events with significant implications on plant safety and availability are investigated to determine the direct and root causes. The investigation, where appropriate, result in clear recommendations to the plant management to take appropriate corrective action without undue delay. Information resulting from such evaluations and investigations are fed back to the plant personnel.

Significant events that would influence the magnitude of an investigation include the following characteristics:

— A significant radiological release or personnel overexposure.
— Plant operation that exceeds LCO, or has not been included in the design basis of the NPP.
— A pattern which is sufficiently complex, unique, or not well enough understood.

In case of severity of an event, the regulatory body conducts an investigation in parallel but independent from the investigation carried out by the operating organization/licensee.

For investigation of events at PNRA, terms of reference typically include the following information:

— Conditions preceding the event;
— Sequence of events;
— Equipment performance and system response;
— Human performance considerations;
— Equipment failures;
— Precursors to the event;
— Plant response and follow-up;
— Radiological considerations;
— Regulatory process considerations;
— Safety significance.

The on-site investigation is conducted by our regional inspectors instantly, to ensure that information is not lost or diminished or that evidence is not removed. It is vital that the on-site investigation does not interfere with the activities carried out by the operational staff.

Interviews are conducted with the staff involved, or witness to the event. A sequence of events is compiled, started immediately and continuously updated as new data is gained.

The investigator prepares a written report and presents it to the PNRA management. In some cases there will be a request for corrective actions commensurate with the identified root causes.

The investigation includes:

- Preparation of status reports and other interim reports documenting significant activities, findings and concerns.
- Ensuring that safety at the incident scene is maintained as appropriate.
- Ensuring that the investigative activities do not result in any adverse impact on the rest of the plant.
- Initiating requests for information, interviews with witnesses, laboratory tests, and technical or administrative support.
- Maintenance of control of information and material collected as part of any investigation.

The object of investigations is not to assign blame or fault, or to recommend or dispense disciplinary actions, either for the plant itself or for the operating staff of the plant. It has been recognized that conducting investigations in such an environment can be conducive to establish the facts that will lead to the identification of root causes and hence the corrective actions to improve safety of equipment and human performance.

6 EVENTS ANALYSIS

Event analysis is prioritized depending on the event significance. The main phases of event analysis are summarized as follows:

- Establishment of the complete event sequence (what happened);
- Determination of the deviations (how it happened);
- Direct cause (why it happened);
- Root cause (why wasn't it prevented);
- Assessment of safety significance (what actions are required) and;
- Identification of corrective actions (how its recurrence can be prevented).

At the regulatory body level, several follow-up activities are carried out after an event analysis. These activities comprise the documentation, dissemination of significant results, monitoring of the implementation of the corrective actions and the assessment of their effectiveness. The regulatory body ensures that operating experience is appropriately analyzed and that lessons to be learned are disseminated.

It is common practice that organizations regularly involved in the evaluation process use standard methods to achieve a consistent approach for the assessment of all events. These standard methods normally make use of different techniques. Each technique may have its particular advantages for cause analysis. Regulatory authority does not normally recommend any specific technique for evaluation unless stated in some regulatory documents.

6.1 Event analysis methodologies

Numerous root cause methodologies, many having a similar basis, have been developed or are under development to address the connection between root causes and corrective actions. Since there is not one best technique to use for all events in all countries, the evaluator selects the most appropriate tool for the event in question, in context of the national capabilities. In Pakistan mainly deterministic and probabilistic methodologies are used for event evaluation. Lately, integrated approach is also gaining pace for regulatory decision making.

Probabilistic Safety Assessment (PSA) provides a systematic approach to determining whether safety systems are adequate, the plant design balanced, the defence in depth requirement been realized and the risk as low as reasonably achievable. These are characteristics of the probabilistic approach which distinguish it from the deterministic approach. PSA include the following:

— To provide insights to supplement those obtained from deterministic safety assessments;
— To identify weaknesses in the design and operation of plants;
— To estimate the risk from plants for comparison with the risk criteria;
— To provide an input into plant specific applications such as the optimization of technical specifications and into operational uses such as maintenance planning;
— To address the phenomena that would occur during core damage and provide insights into how a plant would behave during a severe accident;
— To identify weaknesses in the level of protection provided for severe accidents;
— To identify additional safety systems and accident management measures that would provide further protection against severe accidents;
— To provide an input into emergency preparedness.

PSA is increasingly being used as part of the decision making process to assess the level of safety of nuclear power plants. PSA is now seen as a very useful and often essential tool to support the deterministic analyses. Additionally, many regulatory bodies consider that PSA (especially Level 1 PSA) is sufficiently well developed that it can be used centrally in the regulatory decision making process — referred to as *risk informed regulation*. For these applications to be successful, it will be necessary for the regulatory body (and the utility) to have a high degree of confidence in the PSA.

Deterministic Analysis is the licensing basis of NPPs and for event evaluation it still shares the most important role. Following are the major steps for an evaluation of events using deterministic approach:

— Identification and categorization of events considered in the design basis;
— Analysis of enveloping scenarios;
— Analysis to determine the affected safety functions;
— Analyze barrier integrity to determine the degradation of defence-in-depth;
— Evaluation of the consequences verification that acceptance criteria are met.

Integrated safety assessment (ISA) is a systematic examination of the overall safety level of the plant including processes, items and personnel activities to ensure that all relevant risks have been adequately evaluated and appropriate protective measures have been identified.

7 EXAMPLES

Pakistan is among those countries which have an experience of more than thirty years of nuclear power plant operation. KANUPP, the first nuclear power plant of Pakistan (CANDU type) started its operation in 1972 and after its original design life of thirty years is undergoing its re-licensing.

Some significant incidents / accidents experienced during the 30 year operation of the plant include the following:

- LOCA incident due to surge tank pipe rupture;
- Tube leak in SG #3;
- Small LOCA due to fuelling machine hose rupture;
- Loss of heavy water due to PH6Q- DQ1 transmitter failure;
- Loss of moderator heavy water (~ 34 tons) due to rupture of gasket in motorized valve;
- Fire in one of the oil circuit breaker control cabinet;
- Incident of channel flow blockage and over power trips.

The second power plant CHASNUPP (C-1) a two loop PWR is operating for the last five years and its operation has significantly improved with reducing number of events causing plant trips. So far during operation of C-1, the most important events experienced are:

- Repeated trips of reactor coolant pump;
- Rod drop time of control clusters A1-2 and A1-6 exceeds the technical specification limit;
- Loose connections in engineered features systems;
- Electrical cabinet fire incident;
- Emergency diesel generator damage;
- Operation of safety valve in main steam line due to failure of steam dump system; and;
- Turbine trip due to failure of main feed water control valve, etc.

Detailed analysis of only one event has been presented as an example:

At commissioning stage, rod drop time test was performed. It was observed that rod drop time of control clusters A1-2 and A1-6 exceeds the limit given in technical specification (<2 seconds). To evaluate the safety consequences due to this event an analysis was performed by the operator and was submitted to PNRA for review. PNRA also performed an independent analysis by using deterministic analysis method.

7.1 Analysis for the possibility of mechanical problem in the guide tubes of control rod assemblies A1-2 & A1-6

7.1.1 Status at the time of analysis

According to the information provided by the operator the rod drop time of the two clusters A1-2 and A1-6, did not meet the required drop time limit when the tests were performed in mode-4 and in mode 5.

7.1.2 Event Analysis

Due to the recurring problem of increase in drop time of the A1-2 and A1-6 shutdown (control) assemblies, an analysis was carried out to analyze the possibility of any mechanical deterioration. For this purpose the focus points were:

- Thermal expansion behavior of the material used; and;
- The flow forces exerted on the affected assemblies.

7.1.3 Thermal Expansion of Material

The cladding tube endures high temperature (around 316 °C) during reactor operation. Thus presence of thermal stresses along radial, circumferential and axial direction can be given as:

$$\sigma_r^T = E\ \alpha\Delta t/2(1-\nu)\ [\ln k_r/\ln k + (K_r^2-1)/(k^2-1)]$$

$$\sigma_t^T = E\ \alpha\Delta t/2(1-\nu)\ [(1-\ln k_r)/\ln k - (K_r^2+1)/(k^2-1)]$$

$$\sigma_z^T = E\ \alpha\Delta t/2(1-\nu)\ [(1-2\ln k_r)/\ln k - 2/(k^2-1)]$$

From the available data:

$\nu = 0.31$ at 300°C

$E = 175\text{x}10^3$ MPa at 300°C

$\alpha = 16.98\text{x}10^{-6}$ /°C at 300°C

$\Delta t = 2$°C

k = outer dia/inner dia =10/9 = 1.11

k_r = outer dia/random dia = 10/10 = 1.00 (since we calculated the thermal stresses at outer surface of the cladding. Hence, random diameter was considered equal to the outer diameter).

Therefore:

$\sigma_r^T = 0.0$ MPa

$\sigma_t^T = 4.186$ MPa

$\sigma_z^T = 4.186$ MPa

According to ASME Code Section III Subsection NB-3000 secondary stress:

$$Q = |\sigma_t^T - \sigma_r^T| = 4.186 \text{ MPa}.$$

Therefore secondary stresses were too less as compared to general primary membrane stresses (P_m). Because the loads for calculating P_m had been designed for pressure of 17.16 MPa and the main load considered was the stepping load (78g). Hence as compared to those loads secondary stress load had negligible effect on the material used. Secondly, at cold functional test the temperature and pressure in the assemblies region remain low enough and do not affect the material behavior.

Although according to ASME Code Section III, NB-3124 changes in material properties may occur due to environmental effects. In particular, fast neutron irradiation (> 1MeV) above a certain level may result in significant increase in the brittle fracture transition temperature and deterioration in the resistance to fracture at temperatures above the transition range. Therefore, nozzles or other structural discontinuities in ferritic vessels should preferably not be placed in regions of high neutron flux.

7.1.4 Flow forces exerted on the assemblies

7.1.4.1 Assumptions Used

- Uniform Flow through the core barrel;
- To be on more safe side only control rod assemblies (guide tubes) were considered inside the core barrel;
- Uniform temperature distribution and velocity throughout the core barrel area.

7.1.4.2 Flow forces calculation

The fluctuating flow forces by the coolant flow, converging on the assembly were calculated as given below. Fluid properties and dimensional parameters were:

Density (ζ)		$=732.2$ Kg/m^3
Viscosity (μ)		$=9.42\text{x}10^{-5}$ Kg.m/s
Flow rate (Q)		=16100x2 m^3/hr
Core barrel Outer diameter		=2930 mm
Core barrel inner diameter		=2800 mm
Core barrel thickness	(29300 - 2800)/2	=65 mm

Calculations:

Flow cross sectional area = $(\pi/4)\text{X}(\text{ID})^2$	=6.16 m^2
Velocity (v) = flow rate/area =16100x2/6.16	=1.45 m/s

At hot condition:

Mass flow rate = flow rate x density =16100x2x 732.2	=6549.12 Kg/s
Force = mass flow rate x velocity = 6549.12x1.45	=9496.22 Kg-m/s^2 (N)
Total number of assemblies	=37
Force exerted per assembly =9496.22/37	=256.65 N
Buoyancy force	=110 N
Total reactive force = (256.65 + 110.00)	=366.65 N

At cold condition:

Density (ζ)	=1000 Kg/m^3
Mass flow rate = flow rate x density =16100x2x 1000	=8944.44 Kg/s
Force = mass flow rate x velocity = 8944.44x1.45	=12969.44Kg-m/s^2
	=12969.44 N
Total number of assemblies	=37
Force exerted per assembly =12969.44/37	=350.53 N
Buoyancy force due to coolant	=110 N (including drive shaft)
Total reactive force = (350.53 + 110.00)	=460.53 N

According to design specification, load of control rod i.e. 1471N was higher than the force of 460.53 N (350.53+110) exerted by the coolant flow and buoyancy forces against the rod drop even in cold full flow condition of coolant.

The above calculation indicates that there should be no adverse effect on the rod drop time assuming that the flow remains uniform all over the area having constant velocity and temperature. However, as the locations of assemblies A1-2 and A1-6 are closest to the outlet of core barrel as shown in the Figure 1, about half of the coolant flows towards one exit nozzle and half towards the other nozzle. So the coolant flow in the specified region has to be much higher and hence maximum of the forces exerted due to transversal flow was considered to be 12969.44/2 = 6484.72 N roughly on both those assemblies, which could significantly effect the rods drop time.

Results of tests submitted by C-1 indicated that the rods drop time of those assemblies could be significantly affected by high flow rate of coolant and high transversal flow at the core coolant outlets. The impact of coolant flow and traversal flow was also evident from the fact that for other rods the drop time near outlet nozzles were also little bit higher than those away from the outlet nozzles as shown in Figure 2.

7.1.5 Conclusion

Based on designer report and our rough calculations it can be concluded that the transversal flow forces could be a cause resulting in sticking of rod clusters A1-2 and A1-6.

7.2 Effect of A1-2 and A1-6 control rod drop time extension on results of Chapter 15 of C-1 FSAR

Analysis was performed by experts from designer for this event by considering 3.8 seconds as drop time. Major concern in this case was the minimum DNBR. This was analyzed by using RETRAN code. For cross checking of the analysis done by the designer, PNRA repeated the evaluations by using RELAP code. Comparison shows that the control rod assemblies A1-2 and A1-6 drop time extension do not have significant effect for the design basis events and all consequences of these events remain within the acceptable limits.

7.2.1 Comparison of results for complete loss of flow transient

A complete loss of primary coolant flow results in decrease in heat removal from the core and consequent results in increase in reactor coolant temperature and pressure. If the reactor does not trip promptly, the sub-cooling margin could be lost and DNB would occur. The comparison of results are shown in table-1 below:

TABLE 1. COMPARISON OF RESULTS FOR COMPLETE LOSS OF FLOW TRANSIENT

Designers Results (RETRAN Code)				NRA Results (RELAP Code)
Detail	Drop Time (Sec)	Worth of Rod (Δk/k)	Minimum DNBR	Minimum DNBR
Complete Loss of Flow Transient	2.8	-5.26%	1.9898	1.9468
	3.8	-5.26%		1.9373
	2.8	-4.0%	1.9551	1.9443
	3.8	-4.0%		1.9320

7.2.2 Turbine Trip

A sudden loss of load transient occurring at full power results in over-pressurization of secondary and primary sides. This transient was analyzed to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins. The comparisons of results are shown in table 2 below:

TABLE 2. COMPARISON OF RESULTS FOR TURBINE TRIP TRANSIENT

Designers Results (RETRAN Code)					PNRA Results (RELAP Code)	
Detail	Drop Time (Sec)	Worth of Rod (Δk/k)	Minimum DNBR	Pressurizer Pressure (MPa)	Minimum DNBR	Pressurizer Pressure (MPa)
1. Maximum	2.8	-5.26%	3.9881	16.40	2.3734	16.17

Designers Results (RETRAN Code)					PNRA Results (RELAP Code)	
Reactivity Feedback With Pressurizer Control	3.8				2.3734	16.16
	2.8	-4.0%			2.3734	16.17
	3.8	-4.0%	3.9810	16.40	2.3734	16.17
2. Minimum Reactivity Feedback Without Pressurizer Control	3.8	-5.26%	2.5461	17.59	2.1735	17.04
	3.8	-4.0%	2.5425	17.67	2.1735	17.06

7.2.3 Main steam line break accident analysis

The Main Steam Line Break (MSLB) accident results in rapid cooldown of the primary side and positive reactivity insertion in the core due to moderator temperature feedback effect. If the reactor is in hot shutdown mode, there is a chance that the available shutdown margin is lost due to this positive reactivity insertion and the reactor returns to power. In this case the DNBR may exceed the design limit. For C-1, this limit is 1.45. In the analysis of MSLB the total reactor shutdown margin for the analysis was conservatively assumed as -2.0% (the limit specified by technical specifications). Out of this the worth of A1-2 and A1-6 control rod assemblies was also conservatively assumed to be -1.26% ΔK_{eff} and the remaining worth of the other 34 RCCAs was -0.74% $\Delta K_{eff.}$ Therefore, at the beginning of the transient the shutdown margin was only -0.74% ΔK_{eff}. When safety injection set point reaches, a delay of 2 seconds as per design was considered for reactor shutdown. The A1-2 and A1-6 control rod assemblies would drop to the core bottom within 15 seconds. Event sequence for main steam line break was calculated to be as follows:

CASES	*EVENT*	*TIME (sec.)*
Case 1	Main steam line ruptures	0.00
With offsite power	Low steam pressure attained	0.24
thermal design flow.	SI signal occurs	0.24
	MSIV begins to close	2.24
	Main feedwater isolated	7.24
	A1-2 and A1-6 drop into the core	17.24
	Core criticality attained	56.00
	2000ppm boron reaches core	58.00
	Pressurizer empties	114.00
	Peak nuclear power (19.8%)	154.00
	Minimum DNBR (1.51)	154.00
Case 2	Main steam line ruptures	0.00
Without offsite power	Low steam pressure attained	0.24
thermal design flow.	SI signal occurs	0.24
	MSIV begins to close	2.24
	Main feedwater isolated	7.24

CASES	*EVENT*	*TIME (sec.)*
	A1-2 and A1-6 drop into the core	17.24
	2000ppm boron reaches core	77.30
	Core criticality attained	104.00
	Peak nuclear power (1.37%)	236.00
	Minimum DNBR (20.00)	238.00
	Pressurizer empties	320.00
Case 3	Main steam line ruptures	0.00
With offsite power	Low steam pressure attained	0.24
mechanical design flow.	SI signal occurs	0.24
	MSIV begins to close	2.24
	Main feedwater isolated	7.24
	A1-2 and A1-6 drop into the core	17.24
	Core criticality attained	55.00
	2000ppm boron reaches core	56.10
	Pressurizer empties	85.00
	Peak nuclear power (20.1%)	140.00
	Minimum DNBR (1.62)	140.00

Results of analysis are reflected in Figures-3, 4, and 5. From the above analysis, it could be inferred that the A1-2 and A1-6 control rod assemblies drop time extension does not have significant effect for the events as mentioned in Chapter 15 of FSAR and all consequences of these events would still remain in their acceptable safety margin limit and will not jeopardize the safety of the plant. The available margin in the minimum DNBR and other parameters indicate that no damage of the fuel rod should occur during the transient.

Based on the safety evaluations provided by C-1 and the analysis performed by the regulatory body, the case was presented to the Pakistan Nuclear Regulatory Board (PNRB) which allowed relaxation from the requirements of technical specification for rod drop time of these two control clusters. The drop time of the rods (A1-2 and A1-6) were relaxed to remain within the following limits:

$T < 3.0$ seconds (hot full flow conditions)

$T < 4.0$ seconds (cold full flow conditions)

8 CONCLUSION

Experience has shown that significant events at nuclear power plants (NPPs) worldwide are on decline due to evaluation of events. In this context determination of root causes of events are enabling NPPs to eliminate the direct causes of equipment failure, gross omissions in specific procedures, etc. Such evaluation of events is also helping the regulators, operators and designers to identify and prevent events in nuclear installations.

In-spite of the above, there is a need of a dynamic operational experience process to continue improving nuclear safety, plant performance, and quality. In many cases events also indicate lack of adequate supervision or deficiencies in the safety management of nuclear installation. From this point of view an event has to be taken as an opportunity for learning. A low level event (which includes the near miss) is the discovery of a weakness or a deficiency that would have caused an undesirable effect but did not, due to the existence of one (or more) defence in depth barriers. Individually a low level event may appear to be unimportant; however, collectively these can result in serious incidences due to erosion of safety margins.

Due to evaluation of most of the large events in the safety analysis of NPPs, evaluation of low-level events has become important. There are strong indications that timely corrective actions on trends of declining performance help to avoid further degradation of safety margins. The proof of timely corrective actions and positive response towards safety is that the number of trips of C-1 were fifty four (54) in cycle-1, reduced to fourteen (14) in cycle-2 and reached to three in cycle-3 (all due to external causes). and almost all of them are of INES level "0" or below scale.

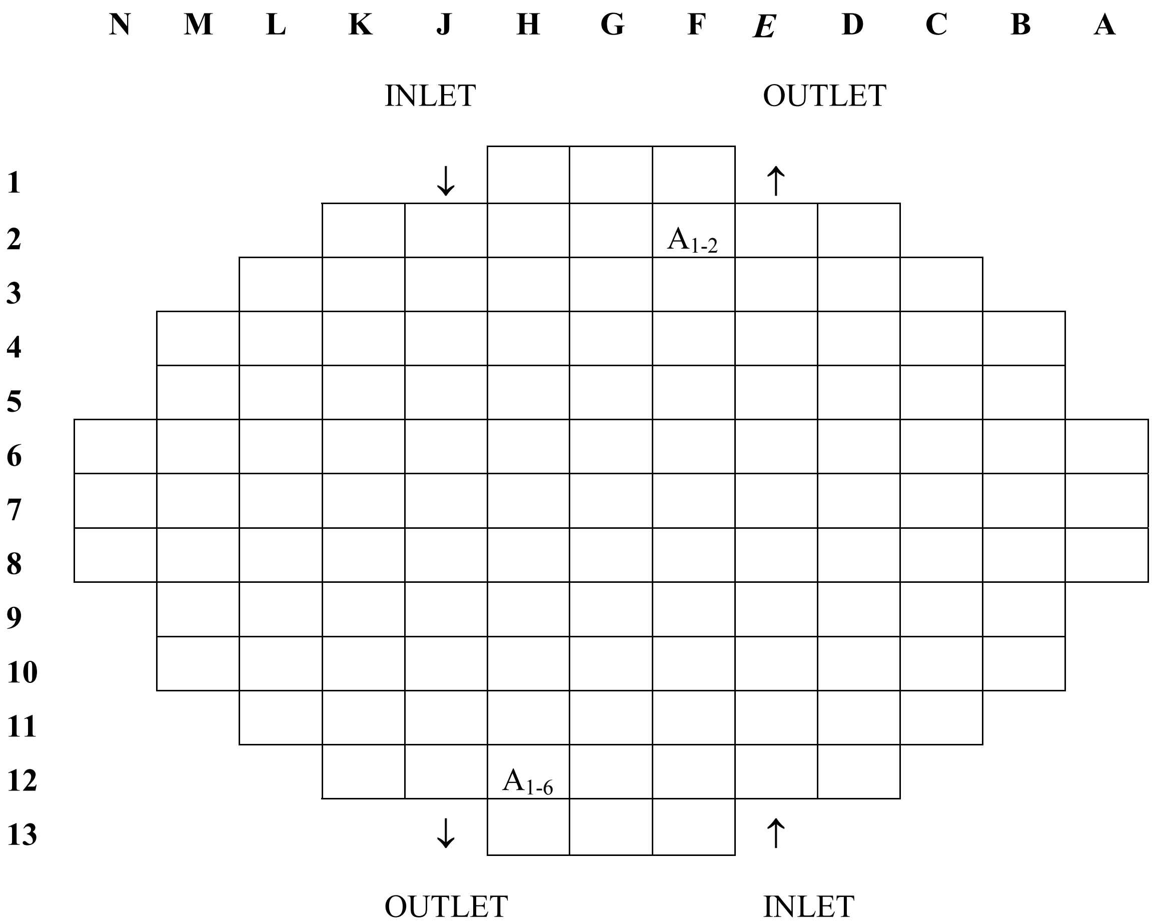

Fig. 1. Location of rod assemblies A1-2 and A1-6 for C-1.

INLET ↓ (between J and H, above row 1) OUTLET ↑ (at E, above row 1)

	N	M	L	K	J	H	G	F	E	D	C	B	A
1													
2						1.39		2.50					
3					1.27		1.33		1.50				
4						1.27		1.36					
5			1.30		1.26		1.26		1.33		1.38		
6		1.24		1.24						1.25		1.25	
7			1.26		1.28		1.25		1.25		1.25		
8		1.27		1.27						1.26		1.26	
9			1.33		1.33		1.29		1.26		1.24		
10						1.37		1.29					
11					1.54		1.36		1.29				
12						2.41		1.35					
13													

OUTLET ↓ INLET ↑

Fig. 2. Rod drop time (to dashpot entry) distribution for C-1. (from CZEC Report).

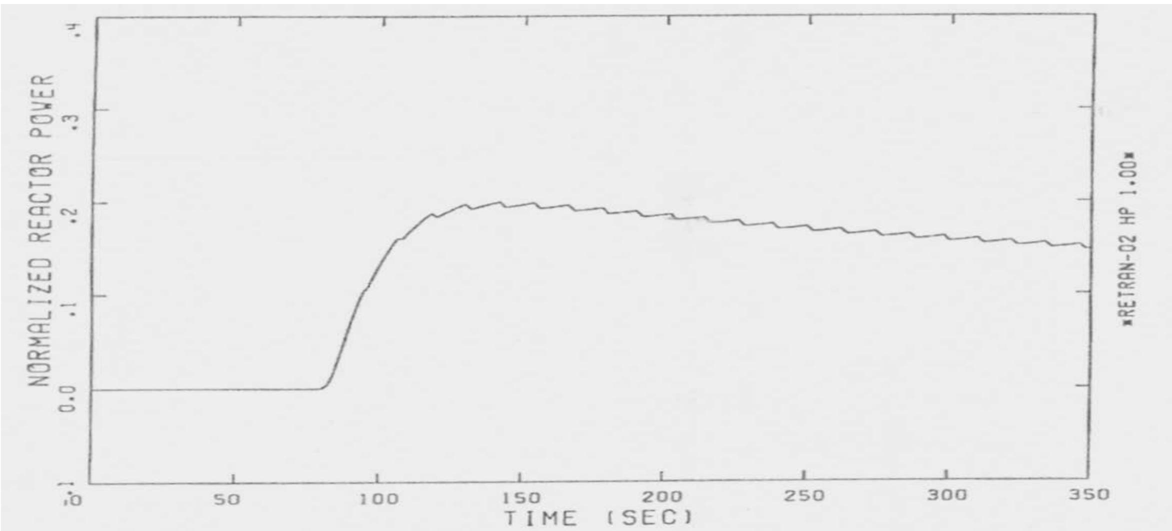

Fig. 3 Main steam line break (case I)-normalized reactor power.

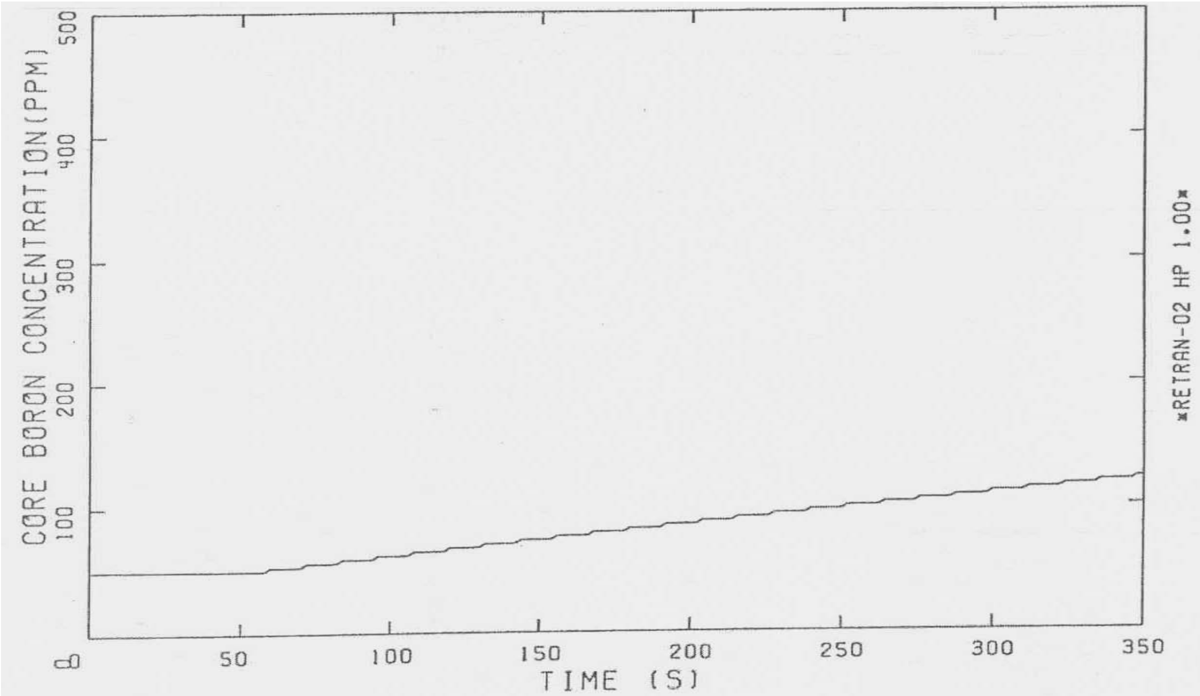

Fig. 4 Main steam line break (Case I)-Core boron concentration.

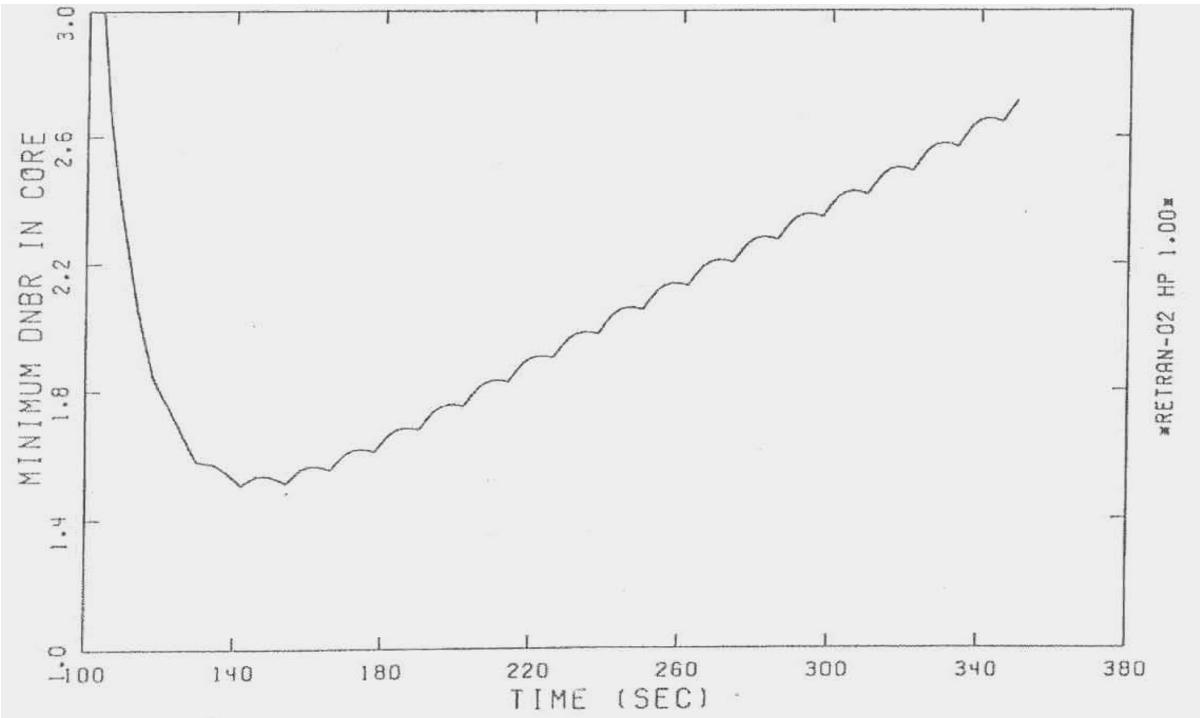

Fig. 5 Main steam line break (Case I) - Core minimum DNBR

Annex 1

EVENT NOTIFICATION REPORT FORM

NAME OF NUCLEAR POWER PLANT & UNIT_______________ EVENT NOTIFICATION REPORT_______________	MONOGRAM

REPORTING CRITERIA		ASSOCIATED PROCEDURE				No. ENR ______________			
EVENT TITLE:									
EVENT TIME & DAY									
TIME:			DAY:			MONTH:		YEAR:	
Reported by NAME:___________________________________ DESIGNATION:___________________________ DEPARTMENT: ______________									
Prior Plant Status		Thermal Power:_______ MW				Electrical Power______ MW			
Tavg _________°C		Pressure:___________ MPa				Operation Mode: ________			
ABSTRACT (MAJOR OCCURRENCES):									
IMMEDIATE ACTION:									
ENSURING STATUS AND CONSEQUENCES:									
SHIFT SUPERVISOR NAME: _______________						Signature: ______________			
REPORTABLE TO		PNRA		IAEA SAFEGUARDS			PLANT MANAGEMENT		
COUNTERSIGNED BY GM/DGM Signature: ___________________									
Notifications to Plant Mnagt	GM	DGM	SM QAD	M(E)	M(M)	M(O)	H(S&HP)	SAR Sect	Any Other

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[12] Informative literature gathered in the file No. DNSRP-NLD-02(05).

THERMAL-HYDRAULIC ANALYSIS FOR RECONSTRUCTING A POTENTIAL PTS EVENT AT THE HUNGARIAN NPP

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Abstract. In relation with a real operational event, when due to some unexpected and unnoticed conditions, strong temperature stratification developed in 3 of the six loops of the unit 1 of Paks NPP. During the incident, which lasted for about 8 hours, in several cases the potentially critical stresses may have developed in the cold legs and also in the reactor vessel. It was extremely important to evaluate the potential consequences of the incident. However, it turned out that that the both the available set of measured information, both the available analysis tools were not completely satisfactory to reveal all the details of the operational sequence. The paper outlines the methodology applied to extract the highest possible information from the available measurements, and also the efforts the set up and apply analytical models do simulate those physical parameters which could not be extracted from the measured information. At the end, the mail lessons to learn from the incident are summarized.

1 SHORT DESCRIPTION OF THE EVENT SEQUENCE

On the 3rd of May of 2004, after completing the regular refueling outage at unit 1 of Paks NPP, the primary system was in the heat-up stage. At around 2 a.m., the system reached the 215^{o}C temperature and the operational pressure (124 bar) of the system, with the six main circulation pumps (MCPs) running. Then a localized fire broke up at the MCP No. 2. Due to a reverse setting of the valves in the MCP fire extinguishing system, the water started to spray first every MCP but the No. 2 in fire. The operator soon realized the situation and manually initiated the extinguishing system at the proper MCP, as well, thus the fire was extinguished in a short time. However, since all the 6 MCPs got wet, the operator had to shut down the MCPs, leaving the system in natural circulation regime. Due to the circumstances (restarting phase after refueling), the residual heat of the core was low: about 1.6 MW, which was only able to maintain only a weak circulation pattern. Due to the actual configuration of the make-up water system, one of the two trains operated in such a regime that without MCP operation, the regenerative heat-exchanger did not receive hot water, therefore three out of the six loops receive unheated make-up water. Due to this, after about 2 hours, a small flow rate (6–8 t/h, distributed for 3 loops) of make-up water of about 40^{o}C cold reached the loops. The operators did not realize the situation for at least three reasons:

- they were engaged in examining the causes and consequences of the MCP fire;
- according to their intention, the problematic branch of the make-up water system was not in operation;
- due to some erroneous wiring, the two temperature gauges which were supposed to show the temperatures at the inlet and the outlet of the regenerative heat-exchanger, respectively, both showed the inlet temperature. Since this error was known to the operators, they did not pay attention to the low temperature.

After about four hours, the operators noticed that the cold leg temperatures of the six loops have shown a highly non-symmetric pattern: in three loops (which are connected to the problematic branch of make-up water system) the delta-T varied between 15 and 30^{o}C, while in the other three loops it was merely 2–3 ^{o}C. Before 6 a.m., the personnel of the replacement operational shift decided to stabilise the natural circulation and therefore closed those loops, which have shown low ΔT, along with loop 1, in which the ΔT was also seemingly much lower then in the remaining two loops. A few minutes after this, the PTS protection signal become active, since the temperature in loop 1 started to decrease rapidly. The protection signal shut off the make-up water pumps and opened the relief valve of the safety valve system of the pressurizer, decreasing the primary pressure to 105 bar. The personnel assumed that the protection signal became active only due to a wrong measurement, thus deactivated the signal and closed the pressurizer relief valve. When the cold leg temperature in loop 1 went further down to about 110^{o}C, they re-opened the loop. After opening the main gate valve (MGV), there was a fast transient, when the loop temperature reached 98^{o}C, then it went sharply up to about 180^{o}C. The operators re-started the make-up water pumps and the feeding of cold water into 3 loops did continue until around 9:30, when finally one of the MCPs was restarted.

Due to the complexity of the physical processes during this event, the details behind the symptoms described above remained unclear for several weeks.

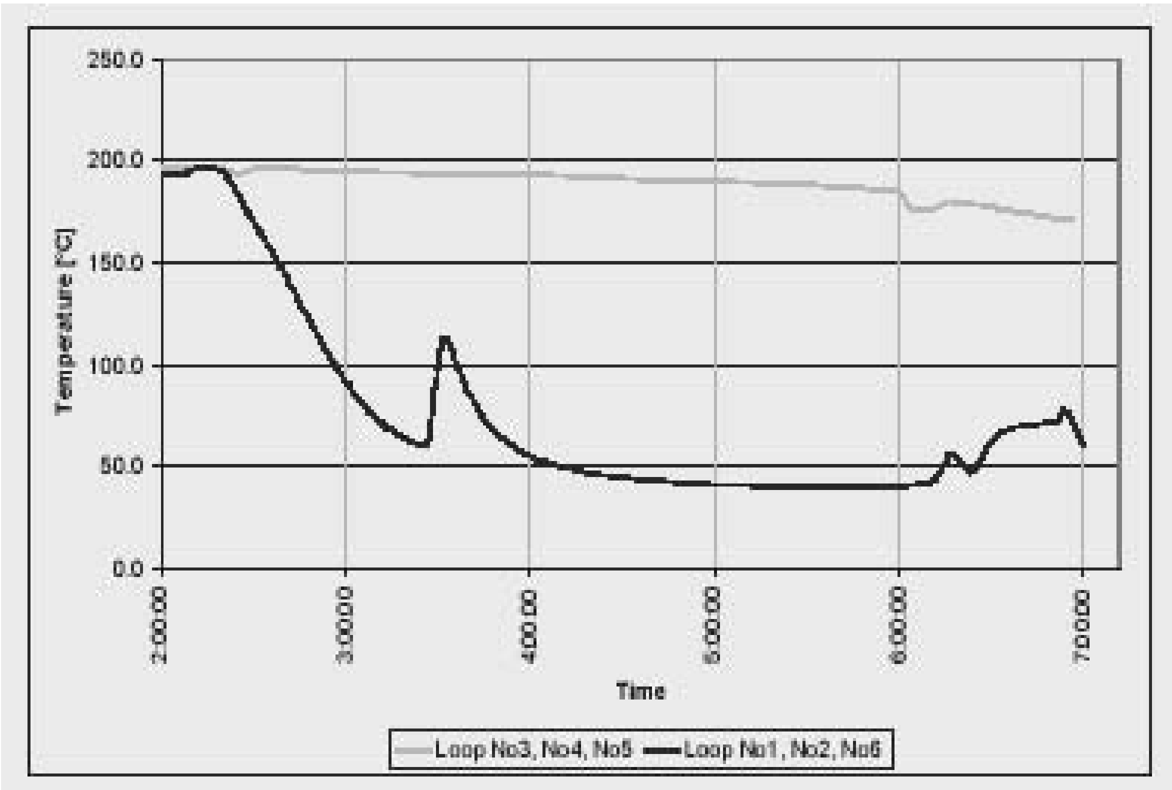

Fig. 1. The temperature behaviour of the two make-up water trains.

2 EFFORTS TO CLARIFY THE TECHNICAL DETAILS OF THE PROCESS

Soon after the event, investigation teams were set up both at the plant and at the Regulator. These teams collected the best possible detailed set of measurement data. From these, it became clear soon that the main reason of the presence of cold water in the loops was the unintended feeding of cold make-up water. It also became clear that the cold water caused stratified flow pattern in the affected loops. This also meant that the protection signal was mistakenly judged as a false alarm by the operators. Note however, that the deactivation of the signal was a good decision, since the continuation of blow-down through the relief valve would endanger the personnel working in the containment to eliminate the consequences of the fire (the rapture disc of the blow-down tank would break soon).

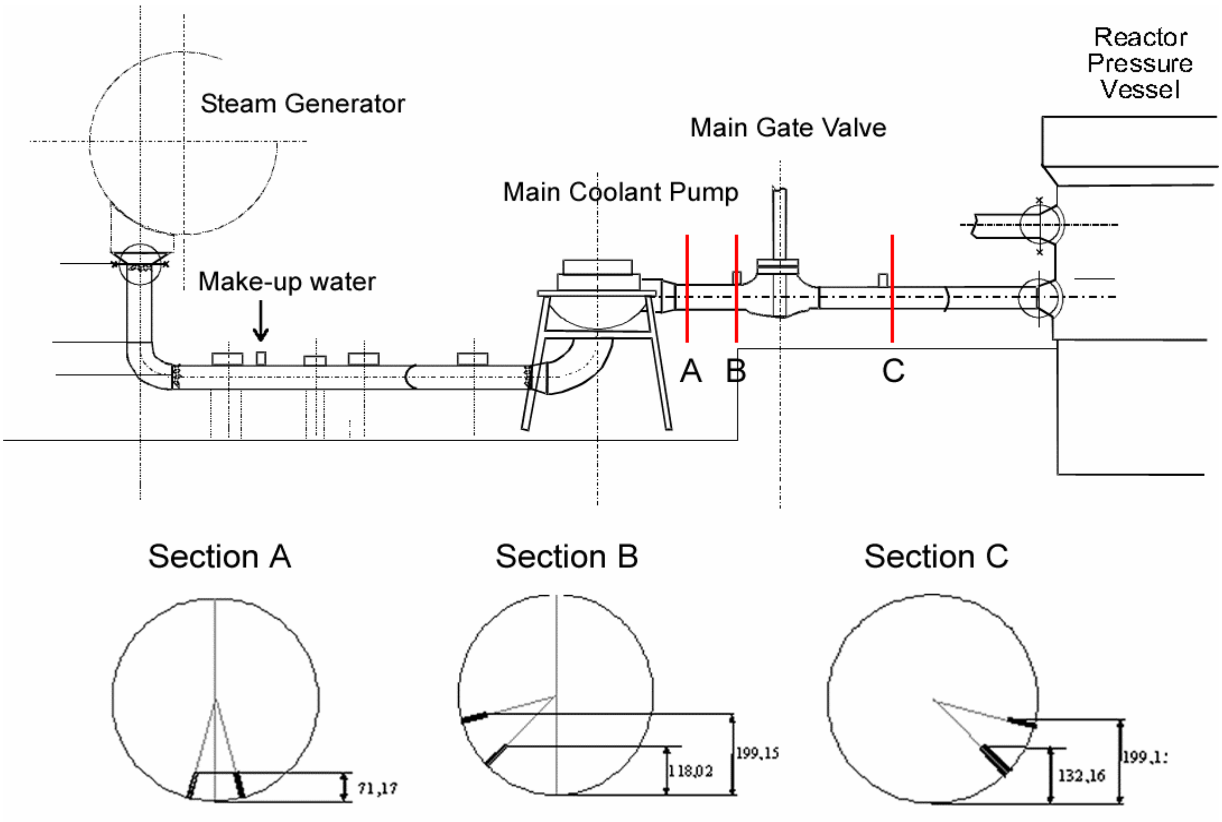

Fig. 2. The scheme of the cold branch of the VVER-440/213 primary loop. The sections A, B, and C represent the possible positions of temperature sensors.

The technical analyses went into three directions:

- it was necessary to fully understand the readings of the different sensors for the duration of the process to be able to provide initial and boundary conditions for the analytic studies;
- analytical thermal-hydraulic calculations were carried out to estimate or reconstruct those parameters which were not measured directly;
- analytical studies to estimate the possible stresses in the affected parts of the primary system.

These three levels of analyses formed a hierarchy: the proper evaluation of measured data provided input for the thermal-hydraulic calculations, while the results of these supplied boundary conditions for the stress calculations.

It was identified that while there is a single value presented in the control room about the cold leg temperature of each loop, these values are automatically selected out of three measurements per loop. Under normal operational conditions, when either forced circulation or intense natural circulation is going on in the loops, the three measurements are very close to each other due to the intensive mixing, therefore the selection of a single, recommended value is a proper practice. In more complicated cases, however, this approach is no longer a proper one. Such cases include the temperature stratification within the loop and the case when the main gate valve (MGV) is closed. Unfortunately, both conditions were present in this process.

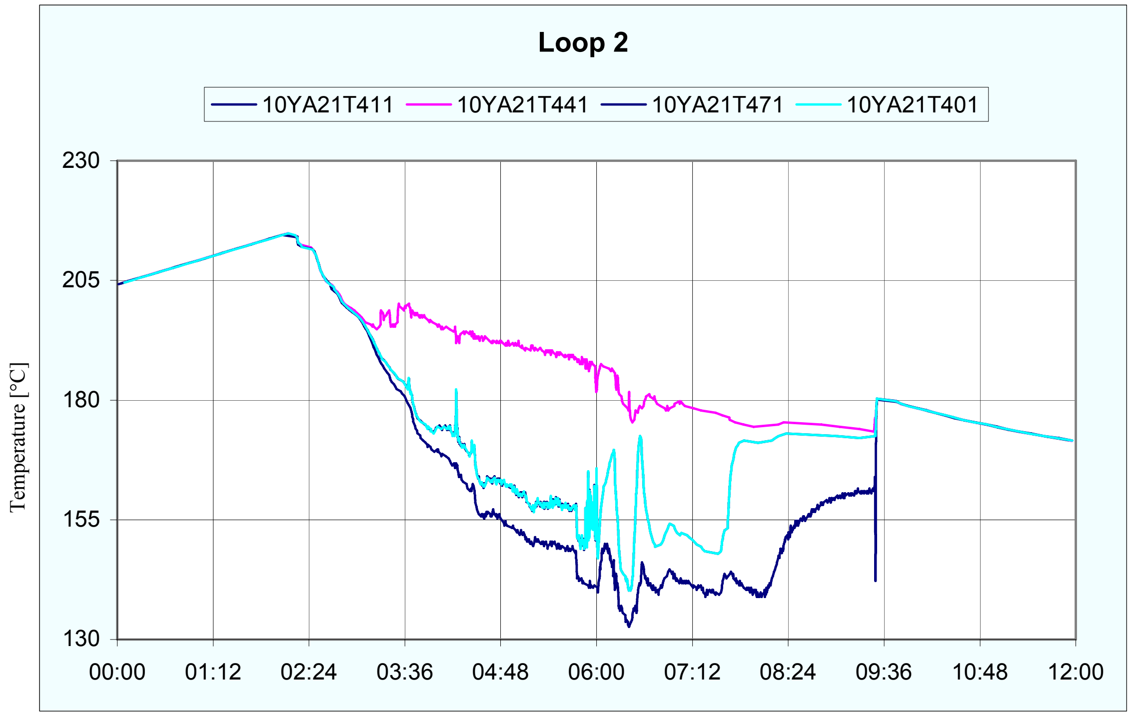

Fig. 3. The trend of the three different temperature sensors in loop 2.

To understand the situation, one needs to look at the scheme of the cold branch of one of the primary loops of a VVER-440/213 reactor. This is presented in figure 2. In the scheme, it is visible that connection point of the make-up water to the loop is at the loop-seal section, i. e. at the lowest point of the branch. The sections marked by A, B, and C represent the cross sections where the temperature sensors can be placed. Since it was regarded that the exact position of the sensors makes no difference from the point of view of the reactor protection logic, the sensors were distributed quasi-randomly among the six possible positions. The individual measurements are available through the process computer, though the actual, exact positioning of the sensors was not available for the operators. From Fig. 1, it is visible that the sensitive tips of the lowest sensors are at 71mm, while the highest is at 199 mm above the bottom of the line of the pipeline. It is also notable that section C is on the side of the reactor pressure vessel (RPV), while sections A and B are on the opposite side of the MGV. In Figure 3, one can see the behaviour of the three different temperature sensors - positioned at three different elevations - in loop 2 throughout the process. The temperature corresponding to the middle curve was presented to the operator. The largest spread between the measured curves was about 52°C. The stratification was obviously very strong.

In Fig. 4, the trend of the three different temperature sensors in loop 1 is shown. It can be noticed that two of the sensors behaved exactly the same way, meaning that they are placed at identical heights. Due to this, the selection logic of the reactor protection system chosen one of these measurements to present for the operator and also to use as protection signal. At 5:45 the MGV of the loop was closed and then the signal of the two sensors which showed the higher values, dropped very sharply and soon coincided with the third one. Since all the three sensors were at the side of the MGV which falls away from the RPV, the transient of closing the valve mixed up the cold water within the closed section of the cold leg and that phenomenon caused the apparent fast cooling. This became also the primary cause of activating the PTS protection signal. It can also be noticed from the curves that after re-opening the MGV (~6:30), the stratification in this loop completely disappeared. This is an obvious sign of developing a healthy natural circulation, which could have been confirmed by the hot leg measurements as well (See figure 5). Note that in loop 2 the stratification remained there (and so it was in loop 6, as well), signifying that the natural circulation did not start in these loops, in spite their open MGVs.

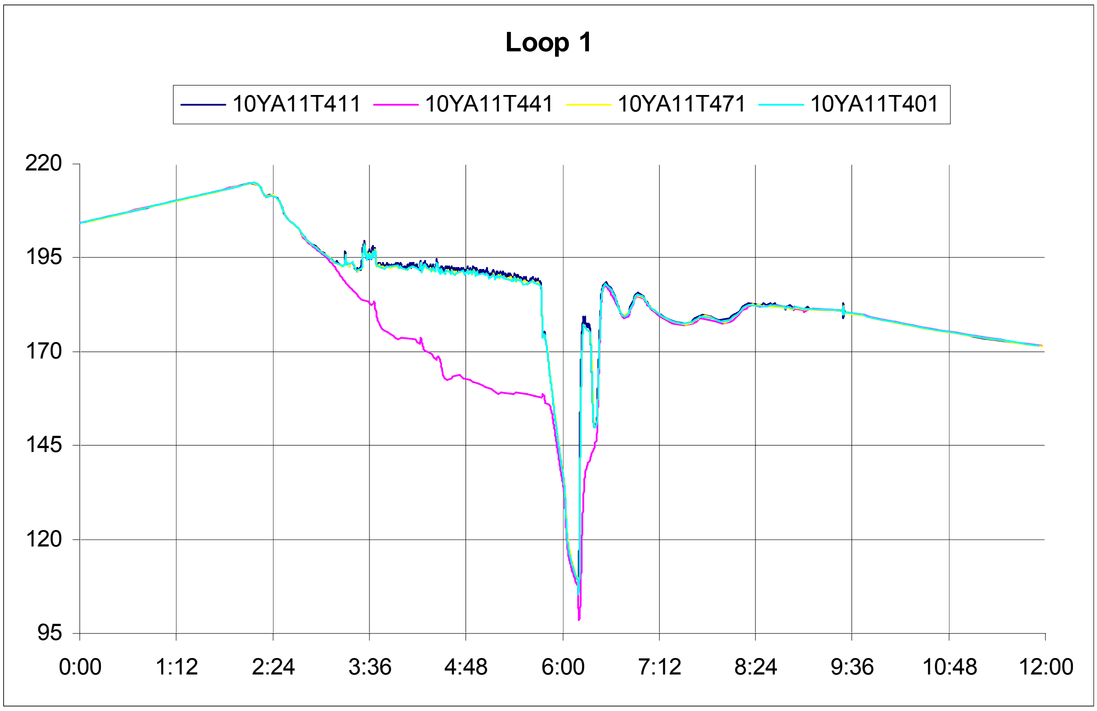

Fig. 4. The trend of temperature measurements in loop 1 during the process.

In Figure 5, the hot leg temperature of the loops is drawn, along with the average upper plenum temperature within the RPV. It can be seen clearly that after the closing of the MGVs in the loops 1, 3, 4, and 5, the upper plenum temperature started to rise linearly, while the hot leg temperatures remained lower. This can only be explained by a total shut down of natural circulation through the core for that period. After the MGV in loop 1 was re-opened, the hot leg temperature of loop 1 jumped up and showed a good agreement with the upper plenum temperature, while the temperatures in the other loops did not. Soon after the main circulation pump (MCP) in loop 3 was re-started (~9:30) all the temperatures converged shortly.

The main consequence of the above findings is that around 3:00 a.m., in the loops which received cold make-up water (loops 1,2, and 6), the natural circulation became blocked by the cold water which filled up the loop seal. Before closing the four MGVs (1,3,4, and 5), the natural circulation was going on in three loops (3,4, and 5) which did not receive cold water. The temperature rise was in the range of 2–3^{o}C, due to the low residual heat of the core. When the MGV in loop 1 was re-opened — perhaps due to the transient caused by the valve — the cold contents of the loop seal was pushed into the RPV and a stable natural circulation started in this single loop with a temperature rise of ~8^{o}C.

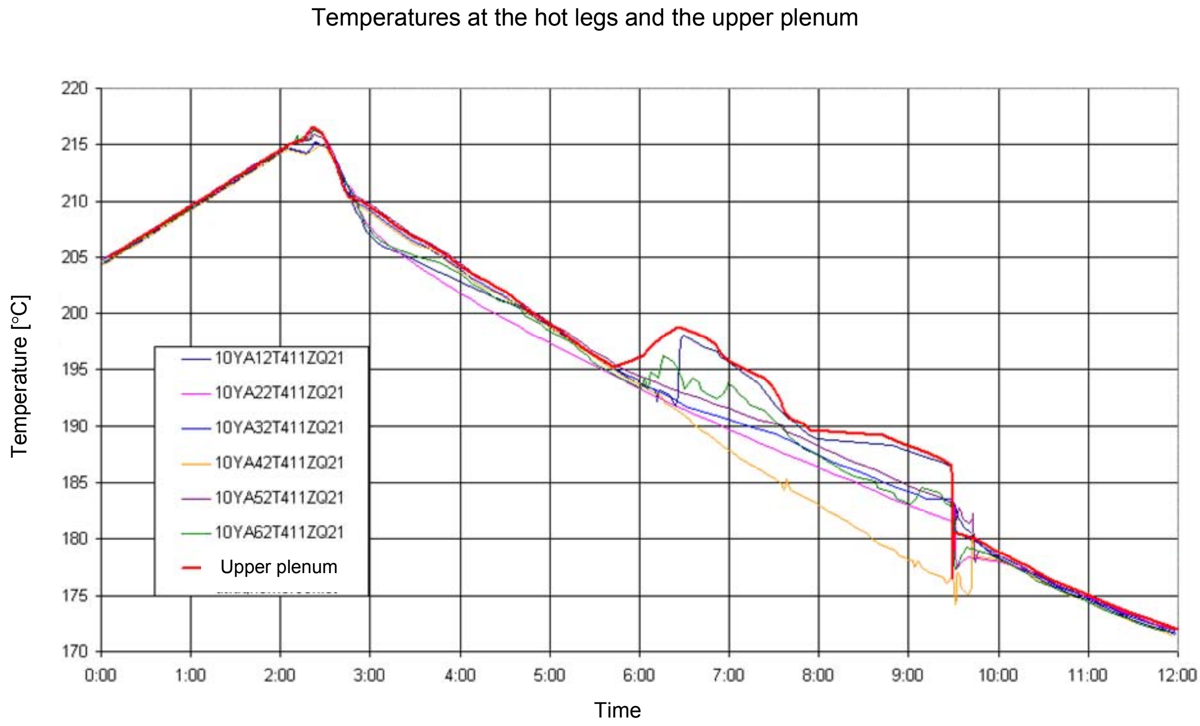

Fig. 5. The run-down of the temperatures at the hot legs of the loops and at the upper plenum of the RPV.

3 SIMULATION EFFORTS

At the licensee, the analyser team chose to apply the REMIX methodology, which had been developed to study the PTS phenomenon, which is potentially caused by the ECC system [1]. In this so called REgion MIXing methodology, semi-empirical, fitted formulae are applied in the characteristic regions of the straight section of the main circulation pipe, of the junction section, and of the downcomer regions. The parameters of the formulae are fitted and verified for the typical range of the operational conditions of the ECCS. By applying the formalism for the actual case, the results have shown no endangering of the critical section of the RPV, i.e. the section which receives high fast neutron fluence, causing embrittlement. It is necessary to note that the actual RPV was about to start the 22nd cycle, therefore due to its embrittlement the transition temperature of the RPV material is estimated to be around 100^{o}C.

In parallel with the analyses at the of the NPP personnel, the Technical Department of the HAEA/NSD set up a suitable APROS model on the basis of the generic Paks-NPP model, which had been developed by the Institute of Nuclear Technologies of BUTE[1]. By using this model it was possible to demonstrate that the incoming cold make-up water fills up the loop seal and blocks completely the natural circulation through the affected loops. Due to basic nature of the APROS code — as being a one-dimensional system code — it was impossible to model the stratified flow in the upper section of the cold leg. The model was also able to predict the correct heat-up on the core (2–3^{o}C) in natural circulation regime with the three remaining loops.

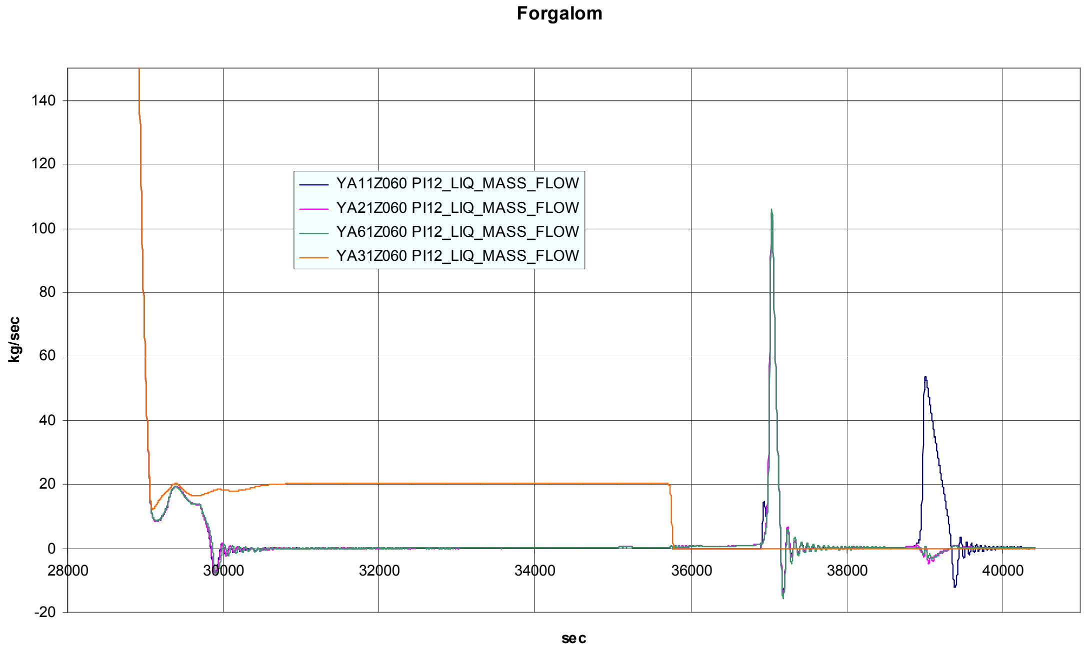

Fig. 6. The loop flow rates according to APROS simulation with cold make-up water supply into 3 loops. The MCPs stop around 29000 sec, the natural circulation is blocked after about 1000 seconds. The MGVs are closed in the circulating loops at 35800 sec and at 37000 sec the MGV in loop one opened, causing a high flow for a short time, when the cold content of the loop seal is pushed into the reactor vessel.

Since the results of the APROS modelling were unable to answer the most stringent question: how much stress was exerted on the RPV wall due to excessive cooling under high pressure; therefore the HAEA/NSD requested one of its technical support organisations, the Institute of Nuclear Technologies of BUTE, to carry out more detailed 3D computational fluid dynamics (CFD) calculations in relation to the case. They have also made APROS calculations, which basically confirmed the HAEA/NSD findings. The Institute of Nuclear Technologies of BUTE uses the CFX code for CFD calculations, and they are already well experienced in modelling the 3D flow in the very detailed and realistically set-up VVER-440/213 RPV model [2], with the main emphasis on the flow distribution in the downcomer. The main results of their calculations in relation with the PTS event have recently been presented at the ICONE conference in Beijing [3]. As the CFD codes in general, such a modelling is extremely demanding with respect the computational time, even by using relatively fast processors (to calculate less then 2 hours of process time required about two weeks of computing). The main result of their analyses was that by assuming 40^{o}C water entering the RPV at the lower edge of inlet nozzle with 6 t/h flow rate continuously for 6600 seconds, the

[1] Budapest University of Technology and Economics

lowest temperature in the cold water plums at the height of the weld 5/6 (which is suffering the highest embrittlement), was around 158°C.

It had to be considered, however, that no matter how detailed the CFD modelling was, still there were quite a lot of things which were not been taken into account. Such problems include the neglecting the interaction of the metallic structure with the coolant, the rough estimation of the flow rate due to the natural circulation, the duration of the cold water influx (~6 hrs).

Due to these circumstances the HAEA/NSD could nether accept the reasoning based on the REMIX results nor the reasoning based on the CFD calculations, that the integrity of the RPV had not been endangered. By that time the reactor was operating in its 22nd cycle, therefore the Authority ordered that should the primary circuit be cooled down for whatever reason, it can not be re-heated without credible proof of its integrity either by unquestionable analytic methods or by sufficiently detailed non-destructive structural material testing.

4 INVESTIGATIONS AT THE LICENSEE AND THE REGULATOR

According to Hungarian law, in case of every event when active protection signal was produced — or due to the conditions, it should have been produced — an event investigation has to be carried out by the licensee and the investigation report has to be submitted to the Regulator for approval. Depending on the significance of the event, the Regulator may also decide to carry out a parallel investigation and the final closing of the investigation occurs when the investigators at the Regulator are able to determine all the important root causes, and contributing causes, making it able to order all the necessary corrective actions. In this case — due to its safety relevance and also due to its complicated nature — the Regulator decided to carry out the parallel investigation. While the investigation at the licensee has already been concluded (after a few iterations), the Regulatory investigation is not closed yet, its conclusion is expected in a few weeks.

The investigation at the Licensee revealed the main points of the event sequence and also revealed the most important causes, behind. In addition to the thermal-hydraulic assessment of the case by using the REMIX model, the Licensee — by utilizing some external specialists — evaluated the possible material behaviour of the affected parts of the primary circuit boundary. These analyses revealed that by assuming the worst possible situation within the pipelines of cold leg, some parts of it may have suffered such a stress which falls outside the safe domain according to the applicable ASME standards. Therefore, it became clear that a detailed structural testing of pipeline structure was unavoidable. Concerning the RPV, the thermal-hydraulic estimations made the Licensee optimistic, but the Regulator demanded to prove the integrity by actual testing.

Concerning the causes, it was identified that apart from the direct causes, one of the most important contributor was of organisational nature: the lack of general procedures which is applicable when a complicated transient occurs in shut-down state. It is also necessary to elaborate some general guideline about the operational strategies when the system enters such a state — without any apparent emergency situation — which is not covered by any detailed procedure.

5 THE MAIN FINDINGS AND LESSONS

The most important technical findings can be listed, as follows:

- a seemingly non-significant mishap (the opposite setting of control valves of the fire extinguishing system) could initiate undesirable complications;
- in the given case, due to significant thermal stresses in the boundary structures of the primary system, the integrity of the primary circuit boundary was endangered in such a way, that it was not even noticed by the operating personnel;
- in case of an unusual and unplanned operational mode, the standard control room panel measurements may mislead the operators;
- strong thermal stratification in the primary loops is highly undesirable;
- cold water in the loop seal is capable to block the natural circulation;
- in case of a strongly 3D flow pattern, the system codes are not able to model the situation;
- the 3D CFD calculations may be very useful to model complicated flow phenomena, but it is questionable to accept their quantitative results for judging a safety case;
- when it is impossible to prove by using analytical methods that the safety margins were not exceeded, then it is unavoidable to apply direct testing methods;
- fortunately, the non-destructive material testing of the loops and the critical belt of the reactor vessel did not reveal any structural damage, therefore the continuation of operation of the reactor was justified.

The most important lessons drawn from the case, along with the related corrective actions are, as follows:

- it is necessary to complement the existing Symptom-based Emergency Operating Procedures by such chapters which are related to cases initiated from different shut down states;
- the simple ways of judging the proper natural circulation are not good enough in every case, it is necessary to train the operators to carry out more complex judgements by using several related measurements (e. g. the upper plenum temperature, the relation of the secondary side temperature and the cold leg temperature, etc.);
- availability direct measurements of the temperature and flow rate of make-up water could have avoid such situations;
- it is not a proper strategy to keep the reactor in a completely unplanned stand-by state for a several hours period, especially if no existing procedure is applicable; the operating personnel should always drive the system into a well controlled state, as soon as possible (e. g. in this case the primary system could have been de-pressurized and cooled down);
- a few years ago — in case of a fully unrelated modification — it was revealed during the simulator testing that under some conditions unheated make-up water may get into the primary loops. For that particular situation the proper measures were introduced to avoid this undesirable situation, but that lesson was not regarded and utilized in a general way. It is necessary to improve the practice of utilisation of internal and external experiences at the plant.

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FULL SCOPE SIMULATOR AS A TOOL FOR ANALYSIS OF OPERATIONAL EVENTS

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Abstract. Operating experience (OE) programme at Temelin NPP is based on Guideline WANO GL 2003-01 – Guidelines for Operating Experience et Nuclear Power Plants and uses HPES (Human Performance Enhancement system) for analysis of operational events. The investigation of the events using traditional OE methods (task analysis, change analysis, barrier analysis and event and casual factors charting) could not be sufficient for several plant transients. For such transients the complementary deterministic analysis should be performed. Temelín full scope simulator could be used for such analysis as a very powerful tool. Even if compared to specialized codes (RELAP, ATHLET, CATHARE, etc.) the nodalization of simulator model is simplified to allow running in real time, the full scope simulator provides quick results with respect to actual state of the units and performed operator's actions with no additional costs.

1 INTRODUCTION

The goal of Temelin NPP internal OE programme is to effectively and efficiently use lessons learned from plant operating experience to improve plant safety and reliability. Learning and applying the lessons from operating experience is an integral part of plant culture and is encouraged by managers throughout the organization. Plant personnel regard operating experience as helpful and important to them and they use this information at every opportunity. Methods of using operating experience are structured to provide applicable information to the right personnel in time to make a difference. When plant personnel analyse the causes of significant events, operating experience is routinely reviewed to determine if and why previous lessons were not effectively learned.

Components of Temelin NPP internal OE programme are the following:

- Identifying and reporting of plant events;
- Screening of plant events;
- Analysis of plant events;
- Corrective actions;
- Assessing how effectively OE is used.

To satisfy the main goal of operating experience programme, i.e. to determining what happened, how it occurred and understanding why it occurred, the following types of event analysis are used:

- Task analysis;
- Change analysis;
- Barrier analysis;
- Event and casuals factors charting.

The investigation of the events using traditional OE methods could not be sufficient for several plant transients. For such transients the complementary deterministic analysis should be performed. Many tools for such analysis can be used, however Temelin full scope simulator cold be used for such analysis as a very powerful tool. In comparison to the specialized codes (RELAP, ATHLET, CATHARE, etc.) the nodalization of simulator model is simplified to allow running in real time. However, the full scope simulator provides quick and relatively accurate results with respect to actual state of the units and performed operator's actions with no additional costs.

2 DETERMINISTIC ANALYSIS OF INTERNAL OPERATIONAL EVENTS

Even if the OE programme is well described and incorporated, some plant events exist, for that additional analyses should be performed to satisfy the main goal of OE, i.e. to prevent the reoccurrence of the events. The reoccurrence of the event is prevented if the direct and root causes are properly determined and corresponding corrective actions are developed and implemented. The plant events that require additional analysis are mostly the plant transients. Two basic approaches to such analyses exist: (1) use of the specialized codes with detail nodalization and (2) run of the transient on the full scope (or display) simulator with simulation of all operators' actions performed during real transient. The both approaches provide different results. Analysis performed by codes with detail nodalization provides large amount of very detail results but based on code user judgment since these codes does not include all relationships and many phenomena and operator's actions should be simulated by input deck modification. On the other hand, the full scope simulator provides less accurate results based on simplified model and nodalization, but includes all relevant operators' actions. Additionally, the simulator run can be performed very quickly after the transient and without additional costs.

3 FULL SCOPE SIMULATOR CAPABILITY

3.1 Temelin full scope simulator

Temelin full scope simulator is made by Czech firm OSC. OSC is a traditional Czech producer of simulators for NPPs since 70's. Temelin simulator centre has got two simulators: Full scope simulator (FS simulator, FSS) and Display simulator (D simulator, DS).

Temelin FS simulator works with more than 600 000 variables, simulates more than 6 500 sensors and around 3 600 active parts (valves, motors, pumps and so on) are simulated.

3.2 History of Temelin full scope simulator

Making, montage a probative operation at supplier‘s facility continued since 22.09.1994 to-15.09.1999. First part of an introduction training of operational staff at supplier‘s facility was led after permission of State office for nuclear safety in years 1998 – 1999. Overtaking tests at supplier‘s facility were realized since 26.07.1999 to 10.09.1999. In September 1999 main parts of simulator were transported from Brno to NPP Temelin. Installation of simulator at NPP Temelin finished 30.9.1999. Tests of simulator at Temelin training center continued since 1.10.1999 to 31.10.1999. First training preparation was started in 1.11.1999. Permission of State office for nuclear safety for starting of operational staff was obtained 25.11.1999. Second part of first operational staff training was realized since 6.12.1999 to 18.2.2000

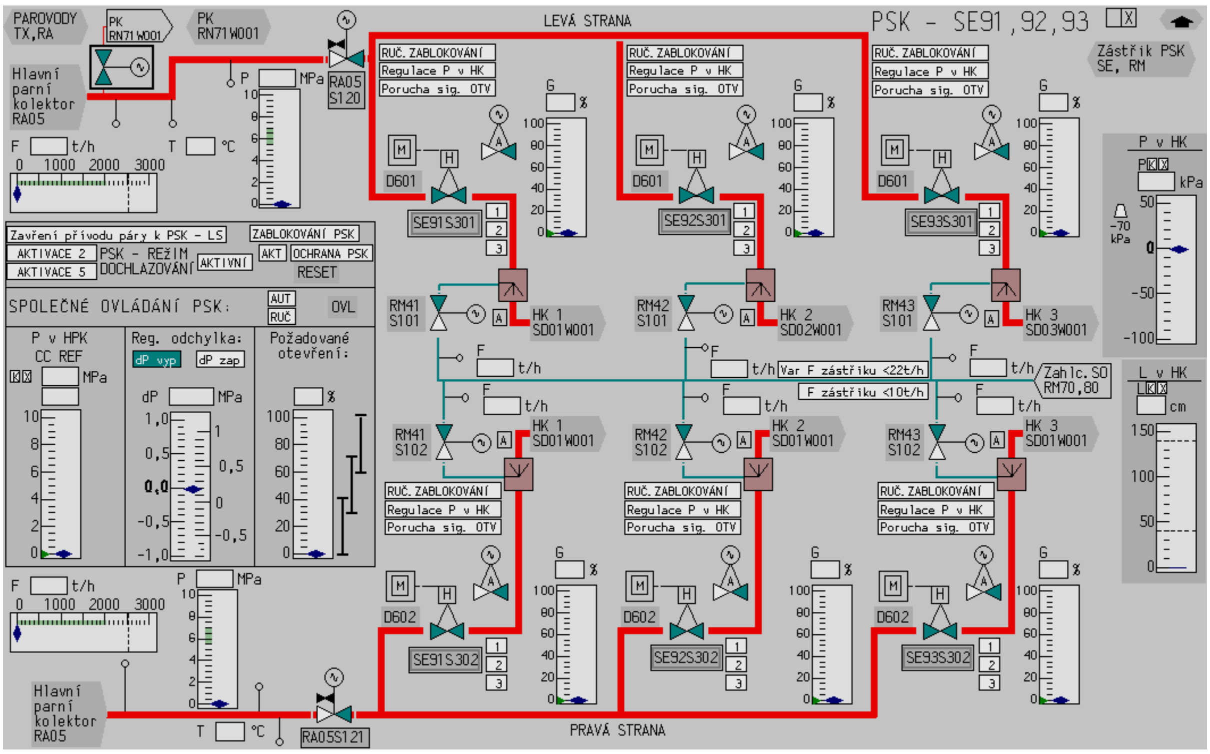
3.3 Adjustment and accessories of FS simulator

- Display simulator was made for support of starting NPP Temelin in 2000. Nowadays D - simulator is used for training;
- Small changes of FSS were made on base of real unit starting tests results starting tests in 2001;
- Education room of simulator was made in 2002. There is an Active Board connected with simulator and NPP net in this room;
- Wide-ranging change of simulation SW according to operational data was realized in frame of upgrade realized since 2003 – 2005;
- System of visual inspection was prepared in years 2003–2004. It is a simulation of camera system including of visualization of leaks;
- Simulators were connected with a terminal placed in Technical Support Centre (TSC) in 2004. This permanent terminal in TSC is dedicated for training of emergency staff. The direct connection of this terminal with FS simulator is used (it means that NPP net isn't used).

4 EXAMPLES OF THE EVENTS ANALYZED ON THE FULL SCOPE SIMULATOR

4.1 Inoperability of one steam dump to condenser valve

During the test of steam dump to condenser valve, one valve was stacked in the close position. Each unit is equipped by six steam dump to condenser valves and after the event only five valves were operable. After the test of the valve hydraulic system it was verified, that oil pressure corresponds to transmitter position and due to the hydraulic system works correctly. The problem was detected on valve control system and reparation was scheduled to the next outage. Further operation of the unit was supposed with five operable valves only.



Even if such operation is foreseen and allowed in the corresponding operating instruction it was decided to perform a simulation of the turbine trip transient with inoperable steam dump to condenser valves.

The purpose of the simulation was to confirm the possibility of operation with one inoperable valve in frame of operational restrictions. These restrictions are said in the table.

No. of operable valves	Allowable reactor power [%]
5	100
4	50
3	30
≤ 2	2

For the test were set these the following admittance criteria:

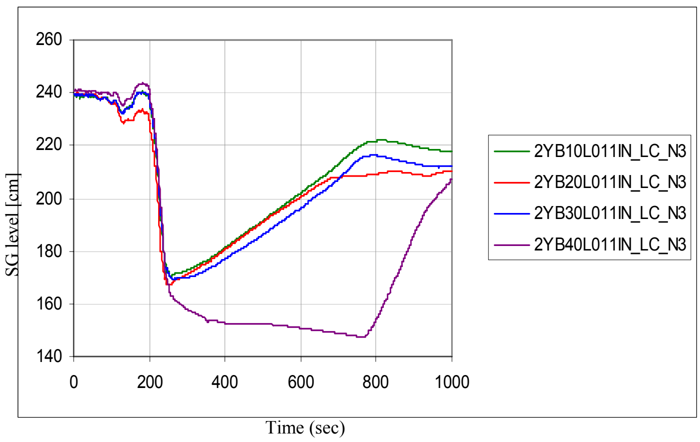
- RT is not initiated (7.74 MPa)
- SDA opening setpoint is not reached (7.6 MPa)

Results of simulation confirmed that operation at power of 100 % N_{NOM} with only 5 operable steam-dump-to-condenser valves was possible. An example of results is said in the table:

N_R [%]	Operable valves	Initial steam pressure [MPa]	Final steam pressure [MPa]	Results of simulation
100	6	6.01	Normal operation – comparable case	
100	5	6.01	6.69/6.84	Acceptable
100	4	6.01	6.88/7.01	Acceptable
100	3	6.01	7.27/7.37	Unacceptable pressure rate
100	2	6.01		RT, SDA
75	6	6.72	Normal operation – comparable case	
75	5	6.72	7.24/7.35	Acceptable
75	4	6.72	7.30/7.41	Acceptable
75	3	6.72	7.37/7.45	Acceptable

4.2 Reactor trip after SG level decrease due to problems in turbine drive FW (TDFW) pumps control system

Reactor trip signal was initiated after decrease of SG levels. The signal “TDFW pumps out of service” was initiated after the problem in TDFW pumps control system that caused TDFW pumps revolution decrease and due to FW flaw was decreased out of the working area. These problems caused rapid decrease of all SG levels and trip of all RCPs and consequently also reactor trip signal on low SG level. SG levels were restored by auxiliary FW pumps and two RCPs were



started. Parameters were stabilized in hot standby conditions. The SG levels during the transient are shown in the picture.

Simulator isn't dedicated for simulation of malfunction in automatic control systems. In his case it was necessary to find a way for obtaining similar response. Suitable way was to fix the steam control valves of working FW pumps from main steam collector (MSC) in closed state and to close steam control valves from turbine extraction to 20%. The time of closing of these control valves has an impact on decreasing of SG level. The best results were obtained with closing time of 120 s.

The purpose of the analysis was to investigate the following impacts:

- The impact of number of functional AFW pumps on decreasing of SG level;
- The impact of uninterrupted work of two main circulation pumps (MCP, main reactor cooling pumps) on decreasing of SGs level.

Results of simulation confirmed that inoperatibility of one AFW pump had no important impact on decrease of SG level. The summary of results is said in the table:

Malfunction insertion time	Closing time of 1SA11,12S351 [s]	Protection of MCP working properly							
		Two AFW pumps are functional				One AFW pump is functional			
		LPG1 [cm]	LPG2 [cm]	LPG3 [cm]	LPG4 [cm]	LPG1 [cm]	LPG2 [cm]	LPG3 [cm]	LPG4 [cm]
+30s	0	156	156	156	150	----	----	----	----
+30s	30	157	157	157	151	----	----	----	----
+30s	60	157	157	157	150	155	155	155	149
+30s	120	165	165	165	154	161	161	161	152
+30s	180	177	177	177	166	174	174	174	164
+60s	180	179	177	177	166	174	174	174	164
Min. LPG - real malfunction		171	166	171	148	----	----	----	----

Results of simulation demonstrated that uninterrupted work of two MCPs was leading to start of EFW pumps on decrease of level in two SGs. The summary of results is said in the table:

Malfunction insertion time	Closing time of 1SA11,12S351 [s]	MCP 1,3 are uninterrupted working Two AFW pumps are functional			
		LPG1 [cm]	LPG2 [cm]	LPG3 [cm]	LPG4 [cm]
+30s	60	138	164	138	160
+30s	120	138	174	138	166
+30s	180	139	186	139	182

5 CONCLUSIONS

Authors of this paper wanted to show that in some reasonable cases the simulator primarily dedicated for training can be used for analyzing of operational events, especially if this simulator was well tested and is in a good correspondence with the real unit.

Advantage of such using of training simulator is:

- The analysis can be done quickly;
- There‘s possibility to respect the real state of technology;
- There‘s possibility to respect the performed operator’s actions and their timing;
- Analysis can be done without other additional expenditures for plant.

Disadvantage of such using of training simulator is:

- In comparison with specialized codes the nodalization is simplified;
- The state of automatic control systems or technology can be different from real unit because later implementation of changes.

DEVELOPMENT OF STATISTICAL METHODOLOGY FOR ANALYSIS OF OPERATIONAL DATA TRENDS ACQUIRED DURING REACTOR OPERATION IN "KOZLODUY" NPP

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Abstract. Deterministic approach requires accurate evaluation of uncertainties. The uncertainties determined as result of computer code calculations have different structure than the uncertainties resulting of parameters measurement on real unit. Understanding of these values is an important issue for validation of deterministic computer code input data. Therefore a special attention must be given to collection and analysis of operational parameters. Since 1996, complex TOFT has been consequently installed on units equipped with WWER-440 reactors of "Kozloduy" NPP. One of the applied approaches for analysis of the accumulated archives, is pure statistical. It is based on methodology adopted from mathematics of communications, where analysis of transmitted electric signals is done in order to get higher level of measurement accuracy. The main considerations of the methodology are described in the paper. It includes description of a way for evaluation of the central moments in form of location, scale and kurtosis. Shannon's entropy, calculated in form of entropy coefficient is evaluated for the purpose of identification the possible probability density function and the ranges of particular parameter uncertainty. As result there are produced cumulative uncertainties of signal transmission, which include: perturbations in measured medium, sensor properties, intermediate signal transformations and the features of output signal. Identification of the uncertainties of important measured parameters during the unit normal operation is performed. There is presented statistical analysis of core inlet/outlet temperatures, pressure of primary circuit and etc. on different operation states, followed by the corresponding conclusions. Six basic states are considered in the analysis. In addition, evaluation of core channels outlet temperatures is done, where outlet thermocouples are analyzed according to the presented methodology.

1 INTRODUCTION

The recent year's improvement of nuclear science and technology led to significant improvement of nuclear safety on reactor units under operation. Created databases of equipment failures, human errors and cost effectiveness are contributing to operational safety improvement. There still can be achieved improving in methods of analysis of data trends, accumulated after registration of values of each parameter, measured in NPPs. Because of the significant number of such parameters on particular unit and the small timesteps between records within 24-hour time interval, an automated statistical analysis is required. On "Kozloduy" NPP a team of engineers put a lot of voluntary efforts on development of such analytical tool. The system of computer applications was launched in operation in 1996 on Unit 2. Consequently the system was installed on units 1, 3 and 4, all of them equipped with VVER-440 reactors. An application of this approach in reactor operation analysis was reported in Reference [1], where data obtained from Russian NPPs was analyzed. Currently the program "STAT," which offers an advanced statistics, is under testing. It is capable to reduce the large trends of data accumulated during 24-hour unit operation to few characteristics, including statistical moments. The approach used in the analysis previously has been applied explicitly for analysis of data transmission on single discrete channel, Reference [5].

[1] Corresponding Author

2 ANALYTICAL METHOD

The computing algorithm might be split on three important steps, Figure 1. The first of them is the determining of the center/location of an unknown statistical distribution (USD). In the analysis must be identified the five possible centers of a statistical distribution. The equations used for evaluation are presented below:

Center of Median (CM):

$$C_M = X_i \ , \tag{1}$$

where (i) is calculated by equation:

$$\sum_{j=1}^{i} p(X_j) = 0.5 \tag{2}$$

Center of Average (CA):

$$C_A = \frac{1}{n}\sum_{i=1}^{n} X_i \ , \tag{3}$$

where n is the total number of observations.
Center of interqartile range (CIQR):

$$C_{IQR} = \frac{X_{i_\beta} + X_{i_\delta}}{2} \tag{4}$$

The values of 25% (X_{i_β}) and 75% (X_{i_δ}) interquantiles are calculated according to the equations:

$$\sum_{j=1}^{i_\delta} p(X_j) = 0.25 \text{ and } \sum_{j=1}^{i_\beta} p(X_j) = 0.75 \tag{5}$$

Center with exclusion of possible errors (CEPE):

$$C_{EPE} = \frac{1}{n_\alpha}\sum_{i \ge i_\delta}^{i \le i_\beta} X_i \text{ Where } n_\alpha = \sum_{i \ge i_\delta}^{i \le i_\beta} n_i \tag{6}$$

Center of range (CR):

$$C_R = \frac{X_{\min} + X_{\max}}{2} \tag{7}$$

The values of $X_{\min}$ and $X_{\max}$ are the maximal and minimal observation in reordered of descending way sequence. According to Reference [2], depending on the shape of statistical distribution, the effectiveness of a particular center is different. If the statistical distribution probability density function (PDF) is characterized by sharp peak type, the CM is more effective than the other centers; for the two-mode statistical PDF most effective center is CIQR; for statistical distributions such as uniform and arcus-sinusiodal, the most effective is the CR. Therefore the conclusion in Reference [2] is: CA is most effective only in case of normal distribution. The authors of Reference [2] recommend usage of all five centers in practical calculation of real discrete observations {X_0. X_1, X_2,…X_n}, the following way: After the calculation, all of the five possible centers are ordered descending way and the center, which takes the middle position, is designated to be the center of an unknown statistical distribution (CUSD), denoted as C_5 (location), Reference [2]. The most deviated values of observation from C_5 can significantly change the value of the CUSD. Therefore an empirical equation is proposed in Reference [2] for removal of these data:

$$t_{\lim} = 1.55 + 0.8\left(\sqrt{\frac{1}{\chi} - 1}\right)\lg\left(\frac{n}{10}\right) \tag{8}$$

The following condition is applied for removal of possible errors [2]:

$$C_5 - \sigma t_{\lim} \le C_5 \le C_5 + \sigma t_{\lim} \tag{9}$$

Where σ is dispersion (scale), calculated as $\sigma = \sqrt{\frac{\sum (X_i - C_5)^2}{(n-1)}}$, Reference [7]. Kontraexess is calculated as $\chi = \frac{1}{\varepsilon'} = \frac{1}{\sqrt{\varepsilon}}$, where ε is the kurtosis (fourth moment). According to the Reference [2], if the kontraexess (χ) is less than 0.515 or more than 0.645 and the center appears as C_5. In case of 0.515 <χ< 0.645 the center is CA [2].
The authors of Reference [6] presented for bivariate Cauchi and Normal distributions, the importance of correctly evaluated centers. There was applied sample mean, sample componentwise median, sample simplicial median and half-space median. It was demonstrated significance of knowledge about the PDF's shape for calculation of the appropriate center C_5.

The second step of the analysis, Figure 1, is the computing of entropy coefficient, ratio between uncertainty range and dispersion. It is followed by identification of the statistical distribution PDF. In this part of analysis, finding of uncertainties during operations on stationary power levels occurs. The major problem of identification of PDF, is calculation of necessary number of bars in histogram. The Reference [2] proposes a way of calculation the minimal and maximal values $m_{\min} \le m \le m_{\max}$ for them, depending on number of observations:

$$m_{\min} = 0.55n^{0.4} \cup m_{\max} = 1.25n^{0.4} \tag{10}$$

The computation of entropy coefficient is performed according to References [2] and [5] for value of number of histogram bars $m = m_{opt}$:

$$k_{ent} = \frac{dn}{2\sigma} 10^{-\frac{1}{n}\sum_{j=1}^{m} n_j \lg n_j}, \tag{11}$$

Where: d is the thickness of histogram bars; σ is the standard deviation; m is the number of bars; n_j is the number of observations in histogram's j^{-th} bar.

Approximate identification of characteristics of PDF, performed empirically in Reference [], is used for approximate identification of the type of PDF as a function $k_{ent} = f(\chi)$.

The algorithm was applied for analysis of recorded parameters during stationary power levels of nuclear reactors VVER-440/ V230 ("Kozloduy" NPP). It was found defined ranges for application, Reference [2] are appropriate for data trends recorded on reactor units in operation. The Reference [2] presents correlations for entropy coefficient calculations as:

$$k_1 = \sqrt{\frac{\pi e}{2} - 5.2\left(\frac{1}{\sqrt{3}} - \chi\right)^2}, \pm 5\%, \tag{12}$$

$$k_2 = \sqrt{\frac{\pi e}{2}} - 69.4\left(\chi - \frac{1}{\sqrt{3}}\right)^3 \text{, } \pm 0.5\% \tag{13}$$

The approximate entropy coefficient is calculated as follows [2]:

$$k^{calc} = \begin{cases} k_1;\ \chi \le 0.58 \\ k_2;\ 0.58 \le \chi \le 0.745 \end{cases} \tag{14}$$

In the cases of $\chi > 0.745$, the current algorithm is unable to produce identification of PDFs. Therefore no stationarity is detected. The ranges for entropy coefficient and kontraexess are calculated with approximate equations [2]:

$$\Delta_{0.9}(\chi) = 1.6\sigma = \frac{1.6\chi\sqrt[4]{\left(\frac{1}{\chi}\right)^2 - 1}}{\sqrt{29n}} \tag{15}$$

$$\Delta_{0.9}(k_{ent}) = \frac{1.6}{k_{ent}\sqrt{k_{ent}n\sqrt{\chi^3}}} \tag{16}$$

Where $\Delta_{0.9}$ denotes uncertainty range, calculated with 90% fidelity for a given parameter (kontraexess, entropy coefficient).

The presented equations imply the following actions: the equations (12) or (13) calculate $k_{\min}^{calc}$ and $k_{\max}^{calc}$ by substitution $\chi+\Delta\chi$ and $\chi-\Delta\chi$, as kontraexess is calculated for a particular data-trend, Equation (14). The condition below must be satisfied for exponential class of probability density distribution [2].

$$k_- = \left(k_{\min}^{calc} - \Delta k\right) \le k_{ent} \le k_+ = \left(k_{\max}^{calc} + \Delta k\right) \tag{17}$$

The following formula can be used for calculation of the power of the exponent [2]:

$$\alpha(\chi) = \frac{1.46}{\left(\ln\left(\frac{1}{\chi} - \frac{2}{9} - 10.7\chi^7\right) - 0.289\right)} \tag{18}$$

The following errors are considered: for $\alpha \in (4\div0.25)$ the error is less than ±0.3%; for $\alpha \in (5\div7)$ the error is 1%, and for $\alpha \in (10\div20)$- the error is ±1.0%, finally, α=29 is the case of $\alpha\to\infty$. The limitations of interval α_{min} and α_{max} are calculated by substitution in equation (18) [2]:

$$\alpha\left(\frac{1}{\chi} - 2\frac{\Delta\chi}{\sqrt{\chi^3}}\right) < \alpha(\chi) < \alpha\left(\frac{1}{\chi} + 2\frac{\Delta\chi}{\sqrt{\chi^3}}\right) \tag{19}$$

If the condition (17) is not satisfied, the following two conditions can be applied:

$$K_{ent} > 1.87 \quad \cup \quad k_{ent} < 1.87 \tag{20}$$

For k_{ent}>1.87 the PDF is assumed to be a composition of exponential and uniform distribution, where the dispersions of uniform (σ_{uni}) and exponential (σ_{exp}) parts are analyzable. Calculated is the ratio $C_p = \frac{\sigma_{uni}}{\sigma_{\exp}}$. The value of C_p can be approximated in the interval of 0.2<χ<0.745 with error less than 5% by the equation [2]:

$$C_{p\pm} = 0.77 + 35.6\left[0.8\left(\sqrt{\frac{\pi e}{2}} - k_\pm\right)^2 + \left(\sqrt{\frac{\pi e}{2}} - k_\pm\right)(\chi - 0.288) + 0.025\chi^2\right] \tag{21}$$

The power of the exponent is calculated as [2]:

$$\alpha_{\pm} = \left[6.87 - 8.71 \frac{(1.57 - 1.9\chi)(k_{\pm} - 1.87)}{(0.7332 - \chi)} \right]^{-\frac{2}{3}} \quad \textbf{(22)}$$

For the case of k_{ent}>1.87 the distribution is assumed to be two-mode. It is analyzed as a composition of discrete and exponential distribution with the same kurtosis. The discrete component is calculated as $C_d = \frac{\sigma_{disc}}{\sigma_{\exp}} = \frac{\sigma_{disc}}{\alpha_{\exp}}$, Reference [2]. The standard error of discrete component is less than 5% when approximation is considered [2]:

$$C_{d\pm} = 0.9\left(\sqrt{\frac{\pi e}{2}} - k_{\pm} \right) + (\chi - 0.06)(0.775 - 0.46\chi) \quad \textbf{(23)}$$

The power of the exponent with a standard error of less than 6% is calculated as [2]:

$$\alpha_{\pm} = 0.31 + \frac{2.12\chi^2 - 2.8\chi + 0.65}{8.83\chi^2 - 9.34\chi + 0.13 + k_{\pm}} \quad \textbf{(24)}$$

The third step, Figure 1, is verification of entropy parameters calculation. The NPPs on power are operated on stationary levels in terms of reactor operation. In statistics, the term "stationary" refers to two cases: strictly stationary (SS), which implies the normal distribution characterized only by first two moments and weakly stationary processes, which are characterized by all the statistical moments (WS). For a SS or WS processes, the autocorrelation function ρ_n and autocovariance γ_n have the properties, References [3] and [4]:

(1) γ_0=Var(X_n); ρ_0=1.
(2) $|\gamma_n|\leq\gamma_0$; $|\rho_n|\leq 1$.
(3) γ_n, ρ_n are symmetric functions regarding OX axis in two dimensional Cartesian coordinates YOX.

The presented above criteria is used for verification of results obtained by two previously described steps, (are the recorded datatrends WS or SS). There has been decided the criteria of WS or SS to be applied according to Reference [3] the following way n→0 when $\rho_n\to 0$ will be obtained:

$$\Phi = \lim \frac{1}{n} \sum_{j=-n+1}^{n-1} \rho_j = 0 \text{, for } n \to \infty \quad \textbf{(25)}$$

The program "STAT" calculates sequence of Equation (25) with consequently increasing number of $j = \{j_1, j_2, j_3\}$, where $j_l = \left\{ \frac{-l(n+1)}{3} .... \frac{l(n+1)}{3} \right\}$ According to Reference [3], If for the consecutive sums (Φ_j) calculated by the Equation (25), is valid $\Phi_j \to 0$, then time average converges to ensemble average (first two moments are enough to describe the PDF, which is normal distribution).

3 DESCRIPTION OF ANALYZED DATATRENDS

The process of data collection is described in Reference [8]. The collected data is stored and tested for detection of false values and recording device malfunctions. The analyzed VVER-440/B230 is unit 2 of "Kozloduy" NPP. Data was acquired during normal operation in 1998. Six different states where considered in current analysis. Their main characteristics are given in Table 1.

TABLE 1. REACTOR STATES, DEPENDING ON ELECTRICAL POWER

No.	Electrical Power	Average Temperature	Duration	Remarks
	MWt/ratio	T_{AC}, K	Records	
1	0.00/0.00	528.75	1440	External Source of Heat
2	46.93/0.11	533.95	1440	One Turbogenerator
3	225.03/0.51	542.35	1440	One Turbogenerator
4	224.81/0.51	543.25	1440	Exchange of Turbogenerator
5	294.63/0.67	548.80	1440	Two Turbogenerators
6	381.06/0.87	555.60	1440	Two Turbogenerators

VVER-440 reactor is designed with two separated turbogenerators. There are considered five basic different states on stationary operational levels of the reactor and one additional, where the power level remains almost the same, but the turbogenerator is exchanged to the other one. Additionally, the State No. 2 should not be considered as stationary for core outlet temperatures, hot leg temperatures, because of increasing level power after 853 records from 0 to 0.29 N_{el}^{nom}. It was added to the other states for the purpose of current algorithm scope evaluation.
The parameters presented in Table 2 are used in this particular analysis. There are included important parameters, related to operation of primary circuit.

TABLE 2. PARAMETERS USED IN CURRENT ANALYSIS

Reactor Component	Parameter	Units, Used in Recordings
Primary Piping System	Temperature of Hot Leg No. 6	Centigrade
	Temperature of Cold Leg No. 6	Centigrade
	Pressure in Cold Leg No.6	Kgf/cm^2
	Temperature of Hot Leg No. 2	Centigrade
	Temperature of Cold Leg No. 2	Centigrade
	Pressure in Cold Leg No.2	Kgf/cm^2
Pressurizer	Temperature in Volume Occupied by Steam	Centigrade
	Temperature in Volume Occupied by Water	Centigrade
Upper plenum	Pressure	Kgf/cm^2
	216 Thermocouples above the fuel assemblies	Centigrade

4 DISCUSSION OF THE RESULTS

Figure 2 presents averaged hot leg (AHL) temperatures for loops No. 2,6 and averaged cold leg (ACL) temperatures for the same loops. The presented curves are in the form[1] of: $T^*_{AHL} = \dfrac{T_{AHL} - T_{AC}}{T_{AC}}$, $T^*_{ACL} = \dfrac{T_{ACL} - T_{AC}}{T_{AC}}$, which characterizes the dimensionless temperature difference for the selected power levels. They are connected with lines. On the plot is demonstrated growing of temperature difference for analyzed states, depending on increasing of power level.

After the analysis, performed by the program "STAT," was done, it was concluded that significance of C_5 application appears in accurate identification of PDFs shape. On Figure 6 are plotted kontraexess and uncertainties for core outlet temperature, calculated for two cases: using central value C_5 and average values C_{AV}. The usage of the CA instead of C_5 might produce less than 1% error.

The calculated values of kontraexess, demonstrated on Figure 3 and Figure 4 show parameters measured on cold and hot legs have initial values between 0.38 ÷0.48. Cold legs konraexesses achieves χ=0.6÷0.645 at ~0.51 N_{el}^{nom} of nominal power from below, while the hot leg kontraesesses achieves χ=0.6 from above (because of transient condition). For both legs it varies in range of 0.58÷0.68 for the rest of the states. The kontraexesses on Figure 5 are characterizing values of pressurizer parameters. For the first state they vary between 0.36÷0.38. For ~0.51 N_{el}^{nom} of nominal power kontraexesses achieve χ=0.65 and for the rest of the states remains within the range of 0.62÷0.68.

Figure 8 shows kontraexesses, calculated for all thermocouples above the reactor core. Their values of no-power state No. 1, vary within ranges of 0.46÷0.56. If the electrical power is increased to 0.11 N_{el}^{nom}, there are separated four groups of thermocouples: three of them with the following ranges of kontraexess: 0.55÷0.58, 0.65÷0.67, 0.70÷0.77. The forth group includes most significant amount of thermocouples. Their kontraexesses vary within range of 0.79÷0.85. There is no possible by the method described above to be identified the approximate PDFs for that group of thermocouples on 0.11 N_{el}^{nom}, because $\chi > 0.745$. Similar behaviour is observed of hot leg temperatures, where kontraexess is equal to 0.8 for 0.11 N_{el}^{nom} Figure 4. Statistical analysis failed to identify PDF because this states is not considered analysable by this algorithm.

As power achieves ~0.51 N_{el}^{nom} all the groups merge kontraexesses within range of 0.55÷0.79. If the electrical power gets value of ~0.865 N_{el}^{nom}, the range of kontraexesses variation is increased to 0.55÷0.76.

[1] Star * denotes dimensionless values

Two groups of factors can influence uncertainties, produced in the measurement channels:

- Thermal hydraulic fluctuations, actuators insensitivity, small reactivity oscillations and etc., Reference [8];
- Transmission of the generated signals, which include transmitter, noise source, receiver and destination [5];

Evaluation of uncertainties of analyzed primary parameters is presented on Figure 3, Figure 4 and Figure 5. The common effect is decreasing of their values with increasing of the reactor power until power level reaches 0.51 N_{el}^{nom}. The uncertainties maintain lower level for values of power 0.67 N_{el}^{nom}, 0.87 N_{el}^{nom}. It can be explained with the improving of the internal reactor feedbacks followed entropy decreasing. Similar conclusion can be made for evaluated uncertainties of thermocouples Figure 9, where besides the general decreasing of the uncertainties, swelling of wideness of uncertainties range occurs until ~0.51 N_{el}^{nom} is achieved. The large uncertainty interval for 0.11 N_{el}^{nom} is due to transient condition in State no. 2.

The combined effects of factors of group A and B on relative values of uncertainties were presented on Figure 2 ÷ Figure 6 in $(k_{ent}\sigma)_i / T_{C_5}$,$(k_{ent}\sigma)_i / T_{AV}$ form. The values of uncertainties, in $(k_{ent}\sigma)_i$ form, are plotted on Figure 7. The increasing of group A uncertainties, leads to increasing of wideness of uncertainty range. For core outlet thermocouples, range of uncertainties grow from 0.002÷0.004 (0.51 N_{el}^{nom}) to 0.001÷0.007 (0.865 N_{el}^{nom}).

On Figure 9 an analysis of uncertainties for the whole core outlet is demonstrated. There were produced characterizing uncertainties for all six states as $k_{ent}\sigma_{ent}$. With increasing of power, the uncertainties are increased in linear law regarding central value of core heat-up temperature. The kontraexess, calculated for temperature measured of thermocouples above the fuel assemblies, remains in range 0.54÷0.58, (0.515<χ<0.645) until 0.11 N_{el}^{nom} is achieved. For higher power level it leaves this zone and center C_5 become important for calculations. This kind of behaviour is explained with increasing influence of separated channel effects, where kontraexess is going out of range characterizing CA (0.48÷0.46).

The separated effects in all heating channels can be described by $\Delta T = f(u_1, u_2, u_3 ... u_o)$, where deviation of central/nominal values are accounted by factors u_o, the following formula was previously derived in Reference [8]:

$$k_{ent}^{\Delta T} = 1 + \frac{k_{ent}\sqrt{\sum_{j=1}^{o}\sigma_j^2}}{T_{C5}^{core} - T_{CL}} = 1 + \frac{k_{ent}^{T}\sigma_{ent}}{\Delta T_0} \qquad \textbf{(26)}$$

Where $k_{ent}^{\Delta T}$ is the entropian core peak factor; k_{ent}^{T} is core outlet temperature entropy coefficient; σ_{ent} dispersion of central core outlet temperatures; σ_j is dispersion for temperatures under influence of j technological factor and $\Delta T_0 = T_{C5}^{core} - T_{CL}$. The maximal core outlet temperature can be estimated with entropian fidelity, using equation (26) [8]:

$$T_{ent} = T_{C5} + k_{ent}\sigma_{ent} \qquad \textbf{(27)}$$

In this part of the analyses, temperatures of the reactor core participate with their central values C_5. After excluding of uncertainties group A and transient conditions, the peak factor is calculated using the equation (26). The peak factor represents maximal deviation from the average core heating with entropian fidelity (27). Figure 10 represents calculated entropian peak factor after temperature values interpolation by polynomials of form $f(x) = a_0 + a_1x + a_2x^2 + a_3x^3 + a_4x^4$. The large values of $k_{ent}^{\Delta T}$, for power levels of $0.06\,N_{el}^{nom}$ came out from evaluated large uncertainties, for state No. 2

5 CONCLUSIONS

The presented algorithm reveals safety features in nuclear reactors, related to evaluation of uncertainties. With increasing of power levels, the values of uncertainties are decreased. However, analysis of other types of reactors must be performed, in order to be done comparison of uncertainties generated during accumulation of operational experience of different reactor designs. The program "STAT" through the values of kontraexess $\chi > 0.745$ detects the transients (demonstrated in state 2).

Detection of entropian values of maximal core outlet temperature helps the complicated cloud of core outlet temperatures to be reduced to: central temperature and maximal entropian temperature, which can improve the quality of safety margin evaluation.

The analysis of VVER-440/230 ("Kozloduy" NPP) parameters, demonstrates the internal features of this particular design. It proves safety, based on internal feedback mechanisms. Automated analysis of parameters can bring additional information to the decision takers for VVER safety evaluation.

Finally, evaluation of uncertainties gives the ranges of each parameter variation during normal reactor operations. In case of deterministic codes validation of certain operational events, the results, produced by the deterministic codes easy can be compared: do they lie within or outside the each particular parameter ranges for particular power level.

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Special thanks to "Kozloduy" NPP management and staff supporting this activity and advising the developers' team for improving of the system through all these years.

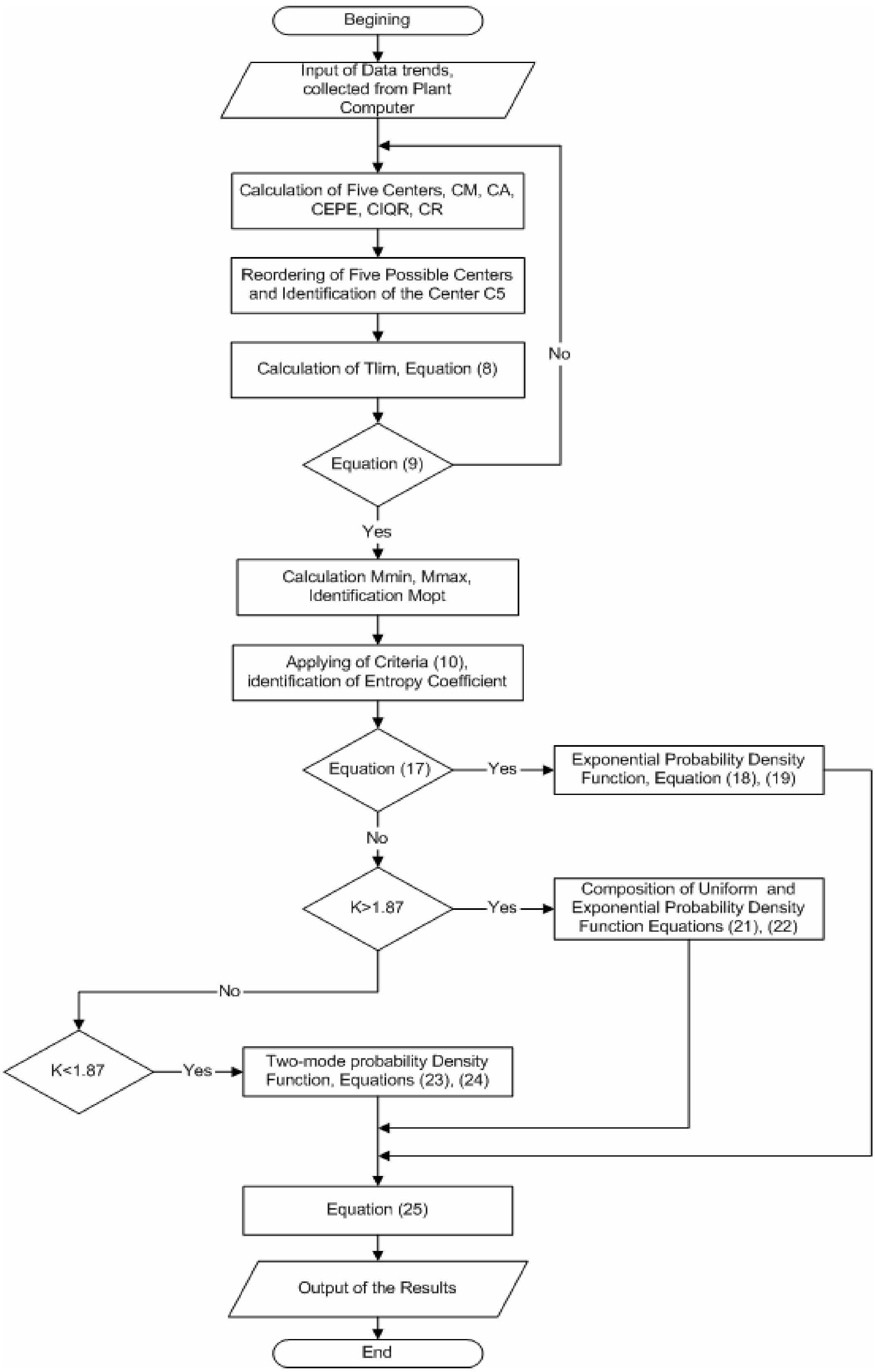

Fig.1. Flow-Chart of STAT computer code.

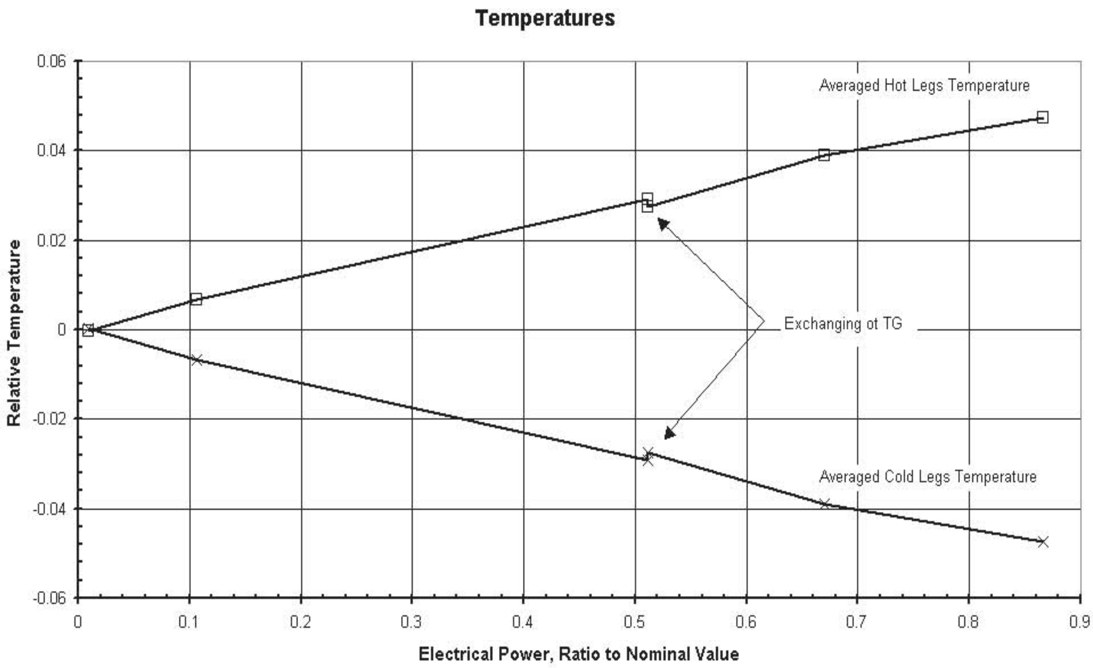

Fig. 2. Relative values of averaged hot and cold legs depending on electrical power.

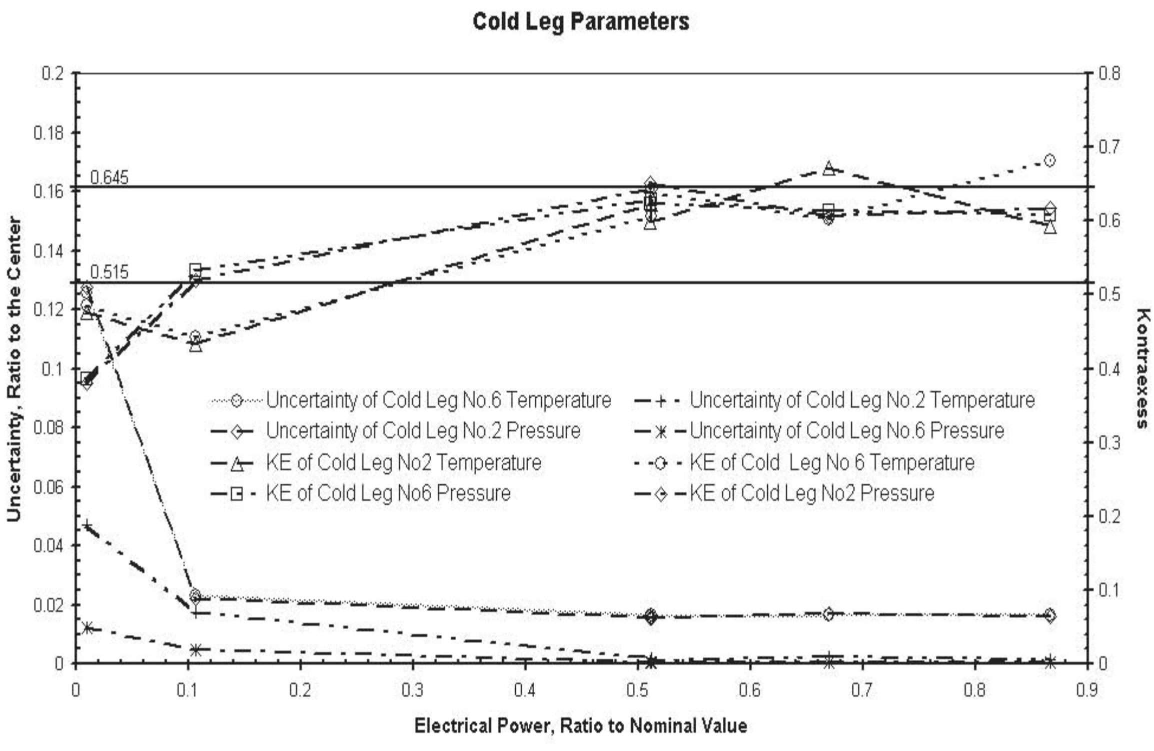

Fig. 3. Kontraexess and uncertainties $(k_{ent}\sigma)_i / T_{C_5}$, $(k_{ent}\sigma)_i / P_{C_5}$ *for Cold Legs No. 2,6.*

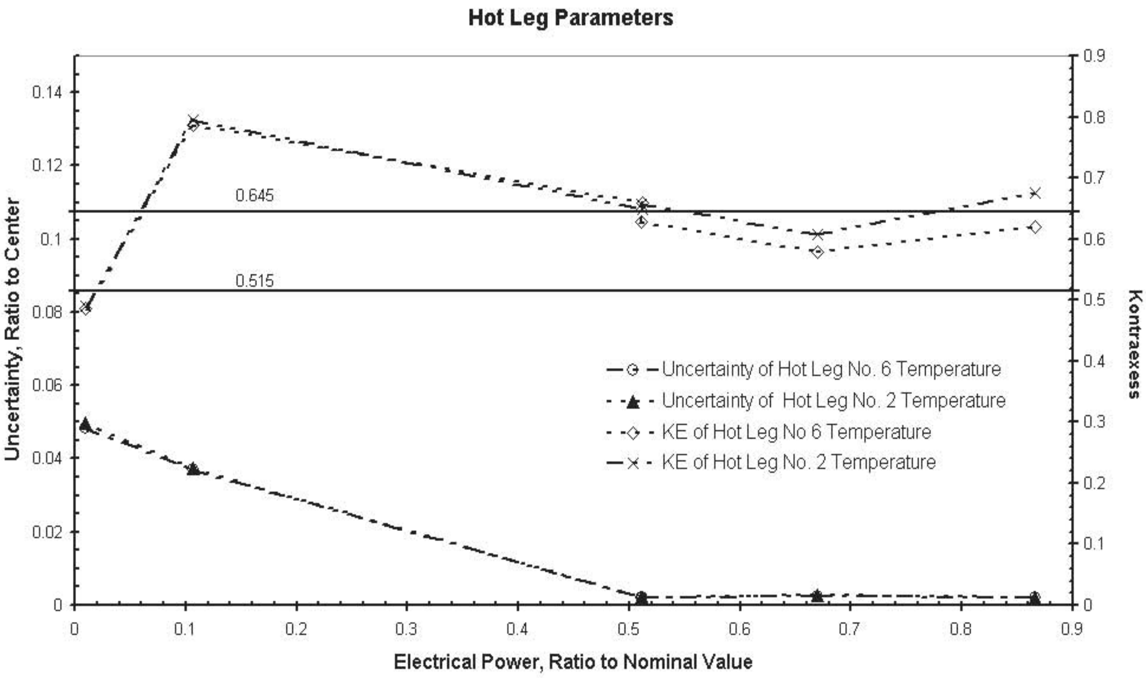

Fig. 4. Kontraexess and uncertainties $(k_{ent}\sigma)_i / T_{C_5}$, $(k_{ent}\sigma)_i / P_{C_5}$ *for hot legs No. 26.*

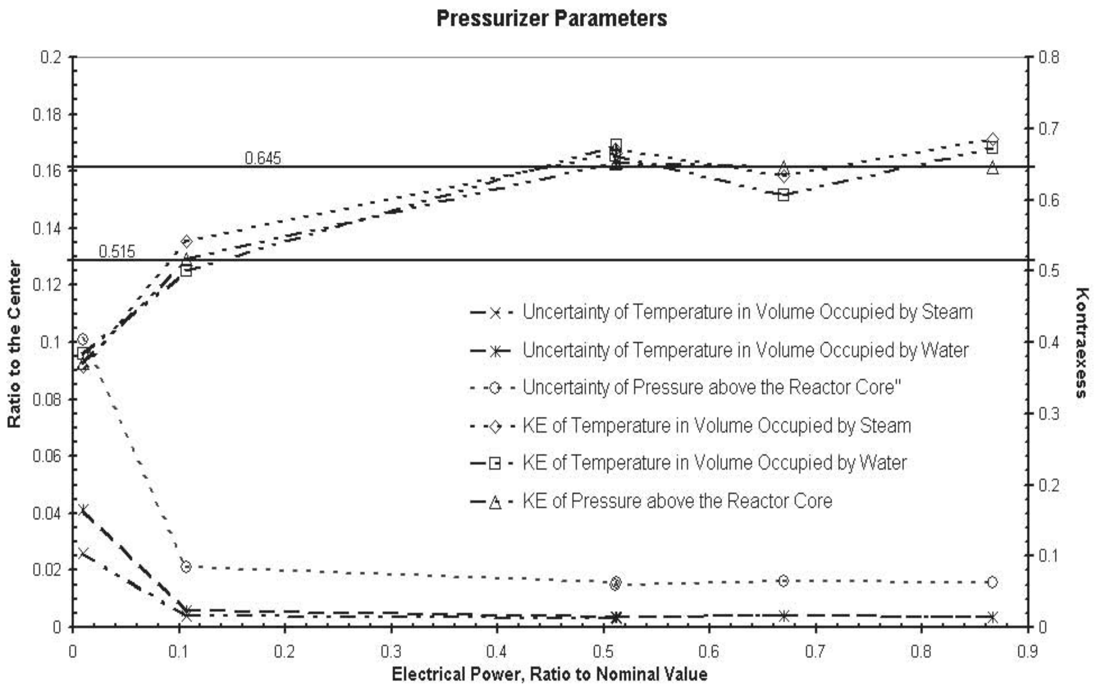

Fig. 5. Kontraexess and uncertainties $(k_{ent}\sigma)_i / T_{C_5}$, $(k_{ent}\sigma)_i / P_{C_5}$ *for pressurizer.*

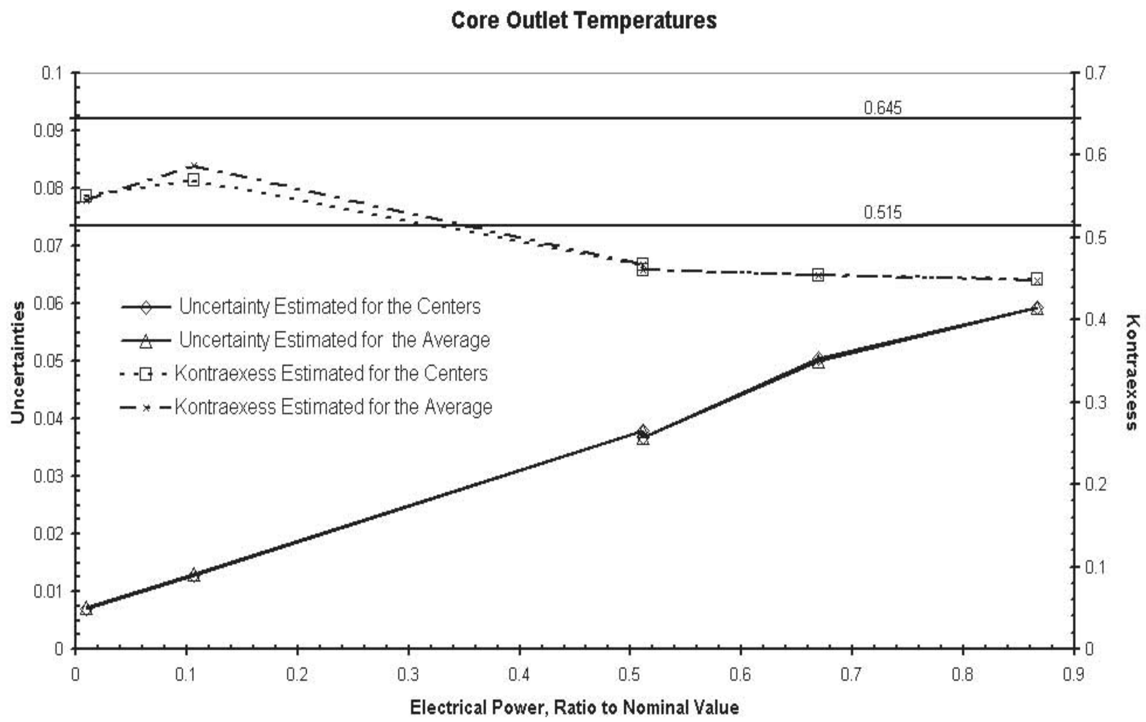

Fig. 6. Kontraexess and uncertainties $(k_{ent}\sigma)_i / T_{C_5}$, $(k_{ent}\sigma)_i / T_{AV}$ *for core outlet temperatures.*

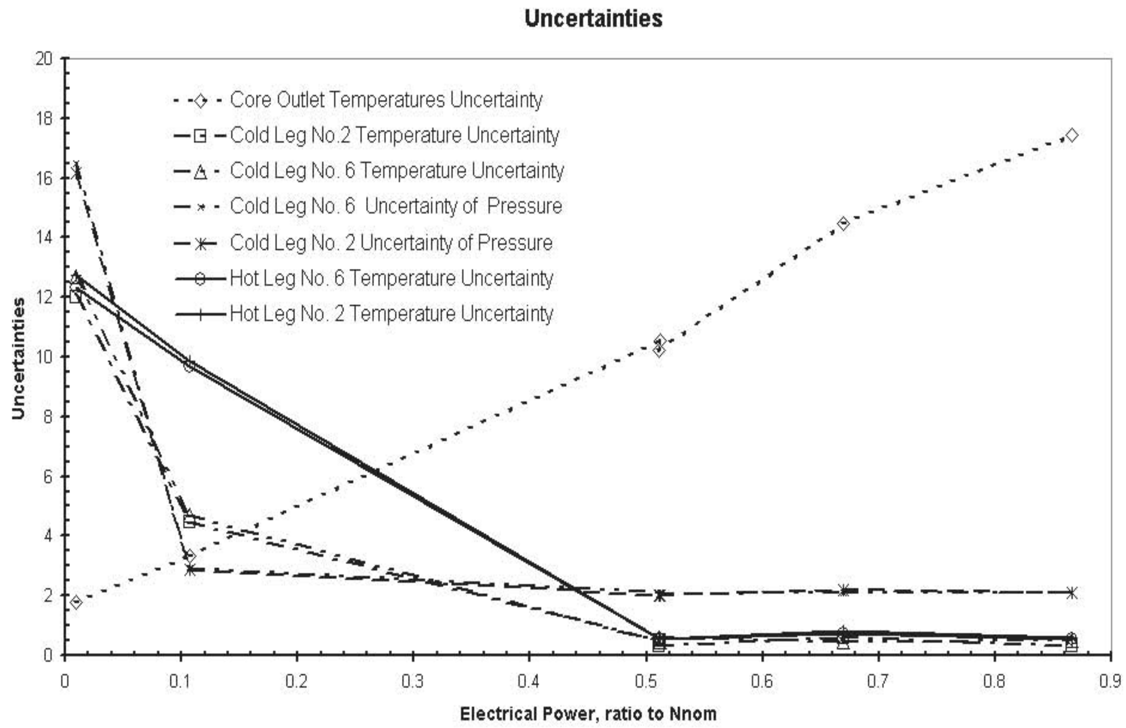

Fig. 7. Uncertainties for parameters.

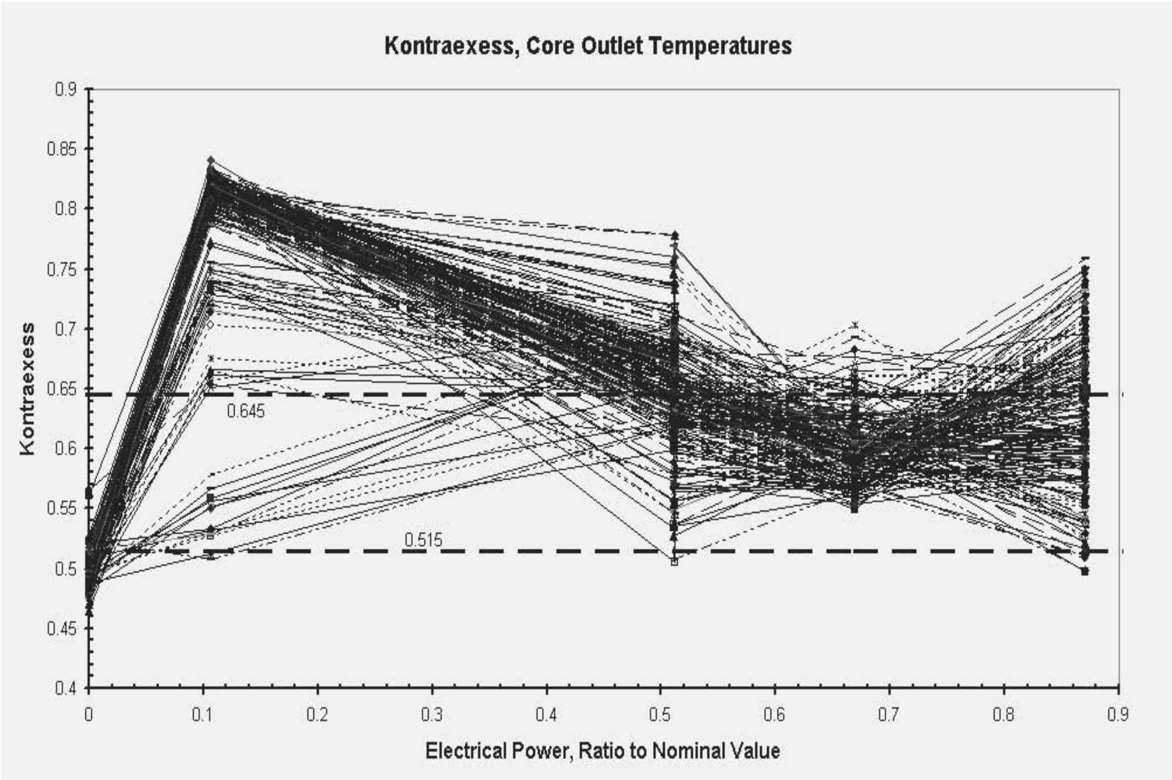

Fig. 8. Kontraexess, calculated for thermocouples, measuring core outlet temperatures.

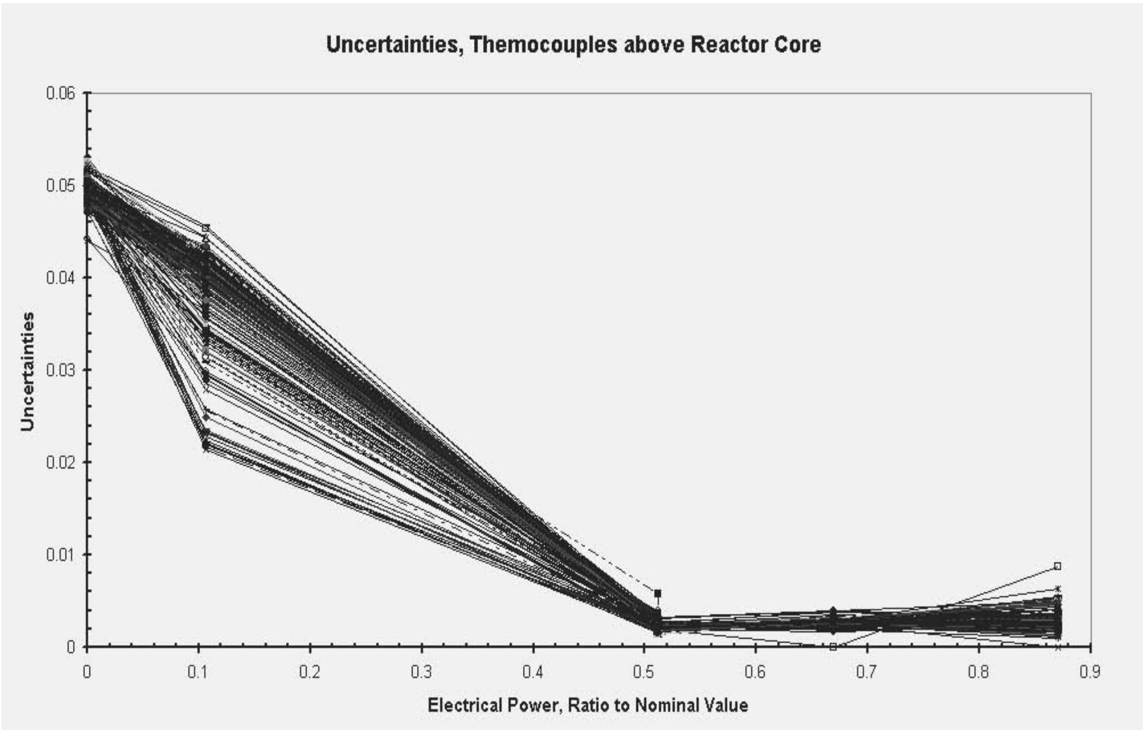

Fig. 9. Uncertainties, calculated for thermocouples, measuring core outlet temperatures.

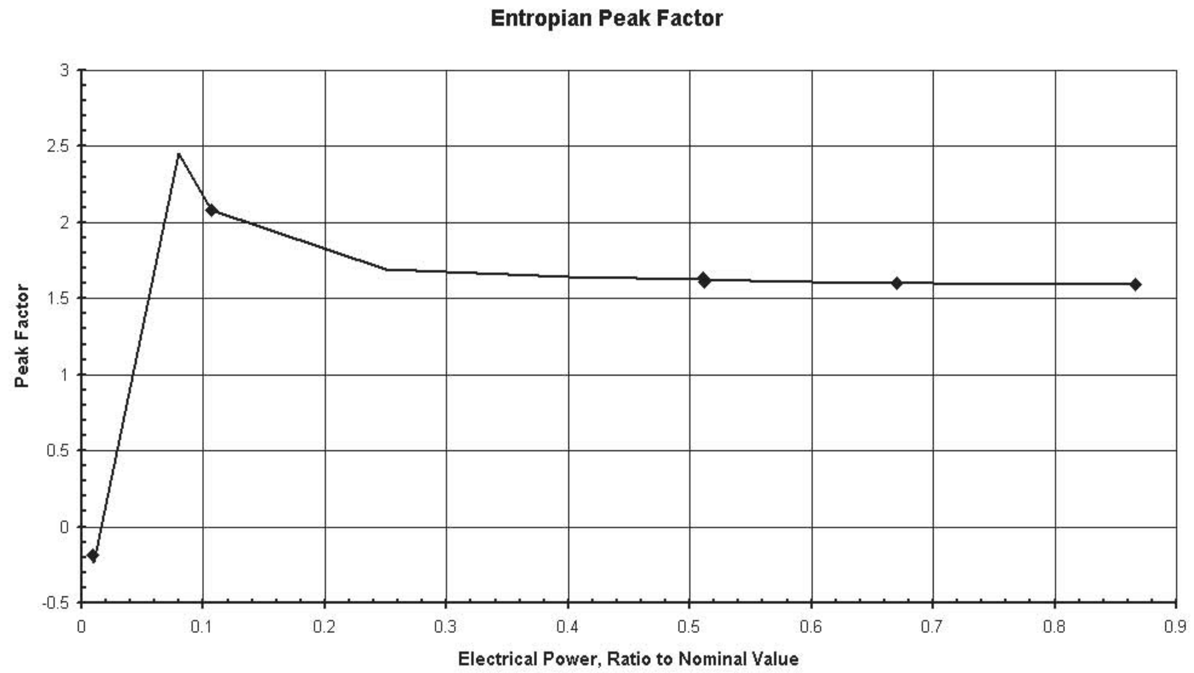

Fig. 10. Calculated entropian value of core peak factor.

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RELAP5/MOD 3.3 ANALYSIS OF REACTOR COOLANT PUMP TRIP EVENT AT NPP KRŠKO

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Abstract. In the paper the results of the RELAP5/MOD 3.3 analysis of the Reactor Coolant Pump (RCP) Trip event at NPP Krško are presented. The event was initiated by an operator action aimed to prevent the RCP 2 bearing damage. The action consisted of a power reduction, that lasted for 50 minutes, followed by a reactor and a subsequent RCP 2 trip when the reactor power was reduced to 28 %. Two minutes after reactor trip, the Main Steam Isolation Valves (MSIVs) were isolated and the steam dump flow was closed. On the secondary side the Steam Generator (SG) pressure rose until SG 1 Safety Valve (SV) 1 opened. The realistic RELAP5/MOD 3.3 analysis has been performed assuming realistic equipment behavior and operator actions. The comparison of the RELAP5/MOD 3.3 results for the realistic analysis with the measurement has shown small differences for the major parameters (nuclear power, average temperature, secondary pressure) for both power reduction transient (100 – 28 %) and pump trip event. Four additional RELAP5/MOD3.3 analyses with different transient scenarios that contribute to Conditional Core Damage Probability (CCDP) in the pump trip event were performed.

1 INTRODUCTION

The RCP Trip event at NPP Krško began on 24.02. 2002. at night when the shift crew decided to reduce the power because of the false temperature increase reading of the upper radial bearing of the RCP 2 had been noticed. After about 50 minutes as the false bearing temperature reading still was high, it was decided to trip the reactor and the RCP 2. At the time of reactor trip, reactor power was equal to 28 %. After manual reactor trip and RCP 2 trip had been completed, there was the noise in the control room. It was concluded that an unexpected steam leak on the secondary side occurred. Therefore, the MSIV isolation was carried out. Consequently, the steam dump flow was terminated. Following the steam dump flow isolation, the only available heat removal from the secondary side was by means of SG relief and safety valves. About 300 sec after reactor and RCP 2 trip, the SG 1 SV 1 opened. According to the plant event report the SG relief valve (SG PORV) did not open at the beginning of the event and the SG 1 SV 1 setpoint was adjusted after first time opening to enable steam relief at lower pressure. After stabilizing the steam line pressure below the SG 1 SV 1 opening setpoint using SG 1 relief valve and with leaking path on the secondary side isolated, the MSIVs were opened. A detailed analysis of the RCP Trip event had found the failure in the indication of the RCP radial bearing temperature detector.

The RELAP5/MOD3.3 analysis of the RCP trip has been performed for the following cases: 1) Realistic event analysis assuming realistic plant behavior and operator actions (power reduction: 100 – 28 %, RCP trip at 28 % power, manual MSIV isolation, SG 1 relief and SV 1 valve, Auxiliary Feedwater (AFW) flow) and 2) Cases that contribute to the CCDP: (a) RCP 2 trip from 102 % power, MSIV isolation, AFW not available, (b) transient scenario from a) and with SG 1 PORV and SG 2 PORV responding and stuck open, (c) transient scenario from (b) but only SG 1 PORV stuck open and (d) transient scenario from (c) and Main Feedwater (MFW) trip at transient begin.

The standard RELAP5/MOD 3.3 nodalization for NPP Krško developed at Faculty of Electrical Engineering and Computing (FER) was used in the analysis, ref. 3 and 4. The RELAP5 model has 469 volumes and 497 junctions. The total number of heat structures is

378 with total number of mesh points of 2107. The developed RELAP5/MOD 3.3 input data set contains the models of the NPP Krško monitoring as well as protection and control systems. The RELAP5 model contains the detailed models of Safety Injection (SI) system, MFW and AFW system as well as of control systems (Automatic rod control, Pressurizer pressure and level control, Steam dump control and Steam generator level control). The initial conditions for the RCP trip event analysis are summarized in Table 1. The initial conditions for the RELAP5 realistic analysis were obtained as a result of power reduction: 100%-28% calculation.

TABLE 1. INITIAL CONDITIONS FOR REACTOR COOLANT PUMP TRIP EVENT AT NPP KRŠKO

Parameter	Unit	Measurement	RELAP5 realistic analysis	RELAP5 CCDP analyses
Pressurizer pressure	MPa	15.5	15.5	15.5
Steam generator pressure	MPa	7.27/7.3	7.23/7.23	6.227/6.226
Cold leg temperature	K	564.02/563.65	563.79/563.78	559.04/558.84
Hot leg temperature	K	575.55/576.0	574.73/574.73	596.56/596.56
RCS average temperature	K	569.79/569.65	569.3/569.3	577.8/577.7
Feedwater temperature	K	452.25/451.05	452.25/451.05	492.8/492.79
Feedwater mass flow rate	kg/s	160.06/158.81	154.79/154.8	551.93/555.27
Main steam line mass flow	kg/s	109.8/125.8	136.48/136.27	551.93/555.26
Pressurizer level	%	36.6	38.9	55.27
SG narrow range level	%	69.9/69.5	68.45/68.47	69.3/69.3
Reactor core power	MW (%)	558.32 (28 %)	558.1 (27.99 %)	2033.88 (102 %)

2 REALISTIC ANALYSIS OF THE REACTOR COOLANT PUMP TRIP EVENT

The results of the realistic RCP trip analysis are summarized in Table 2 and Figure 1 to Figure 4. The realistic RCP trip event analysis consists of the (a) power reduction: 100 – 28 % transient and (b) RCP trip event analysis. The realistic transient analysis has been performed and the results were published in [4]. For the power reduction transient scenario as well as for main feedwater temperature, the realistic plant data obtained from PIS data, [5], were assumed. The power reduction transient lasted for approximately 50 minutes (3020 sec). It was initiated on the secondary side by a reduction of turbine steam flow, Figure 1. There are two control systems that automatically adjust the plant parameters following the turbine power change. The automatic Rod control system maintains the programmed average temperature by inserting reactivity in the core and the SG level control system regulates the main feedwater valve position so that sufficient feedwater flows into SG to maintain the level at a reference value. Following the turbine steam mass flow reduction, the SG secondary pressure increased, Figure 3, while the set point for the RCS average temperature decreased according to coolant average temperature program linearly as the turbine power decreased. Following the control rod insertion according to the control rod speed program, the RCS average temperature, as well as nuclear power, Figure 2 decreased. Calculated initial conditions for the RCP trip event at the end of power reduction: 100%-28% transient are summarized in Table 1. A very good agreement between the calculated and measured values at time = 3020 sec (nuclear power = 28 %) was obtained. The RCP trip transient was initiated by the sequence of the following events: reactor trip, turbine trip and RCP 2 trip. Following the RCP 2 trip, a part of the cold leg flow from the unaffected loop (1) bypassed reactor core and flew to the cold leg of the affected loop (2) so that flow in that loop reversed. Hence, the imbalance in the transferred heat on the secondary side between the two loops was established. Thus, the major part of the heat produced in the core was transferred in the SG 1. Following reactor trip nuclear power was quickly reduced, Figure 2. The average temperature decreased quickly to the value for the Low Tavg & Reactor trip signal, which actuates main feedwater isolation. Owing to steam dump flow, which is determined by steam dump control – turbine trip mode, the secondary pressure was reduced following turbine trip, Figure 3. However, because of MSIV isolation initiated by the operator, the steam dump flow was terminated and the SG pressure increased. Following the MSIV isolation, the only means of heat removal to the secondary side was by SG relief and safety valves. The analysis of the measured data has suggested that the SG 1 relief valve did not open at the set point pressure. Also, nonstandard behavior of the Auxiliary Feedwater (AFW) was observed that resulted in the low SG 1 level, Figure 4. In the RELAP5 model following assumptions were used in order to match the steam leakage and operator actions: 1) The SG 1 relief valve did not respond at the beginning of the event as expected. (2) SG 1 SV 1 valve was closing at the lower closing pressure than specified and additional leakage equal to 8.5 % of the SG SV area was considered after closing signal. After closing the SG 1 SV 1 for the second time the steam line pressure was controlled by the adjustment of the SG 1 PORV set pressure. (3) The AFW flow obtained from the measurement including flow measurement error were used in the analysis. The AFW flow was assumed to start at a time point when SG 1 pressure exceeded 7.8 MPa and SG 1 SV 1 closed after first time opening.

The SG pressure was affected by both heat removal through the SG 1 relief and safety valves and by AFW flow. The role of AFW flow is twofold. First, spray-type injection of cold AFW water efficiently reduces SG pressure due to steam condensation on water droplets. Secondly, the AFW recovers SG inventory so that the heat sink is preserved. In the realistic analysis a very good agreement of the calculation with measurement for both phases (power reduction transient and RCP trip event) were obtained.

TABLE 2. TIME TABLE OF EVENTS FOR THE REALISTIC ANALYSIS OF THE REACTOR COOLANT PUMP TRIP EVENT AT NPP KRŠKO

Event	Measurement Time (sec)	Realistic analysis Time (sec)
Reactor trip, turb. Trip	3020	3020
RCP 2 trip	3023	3023
MSIV isolation	3125	3125
Main feedwater isolation	not available	3026.6
AFW start	3521	3508
SG 1 relief valve opens for the 1st time	not available	4163
SG 1 SV 1 valve opens for the 1st time	3330	3317
SG 1 SV 1 opens for the 2nd time	3781	3796

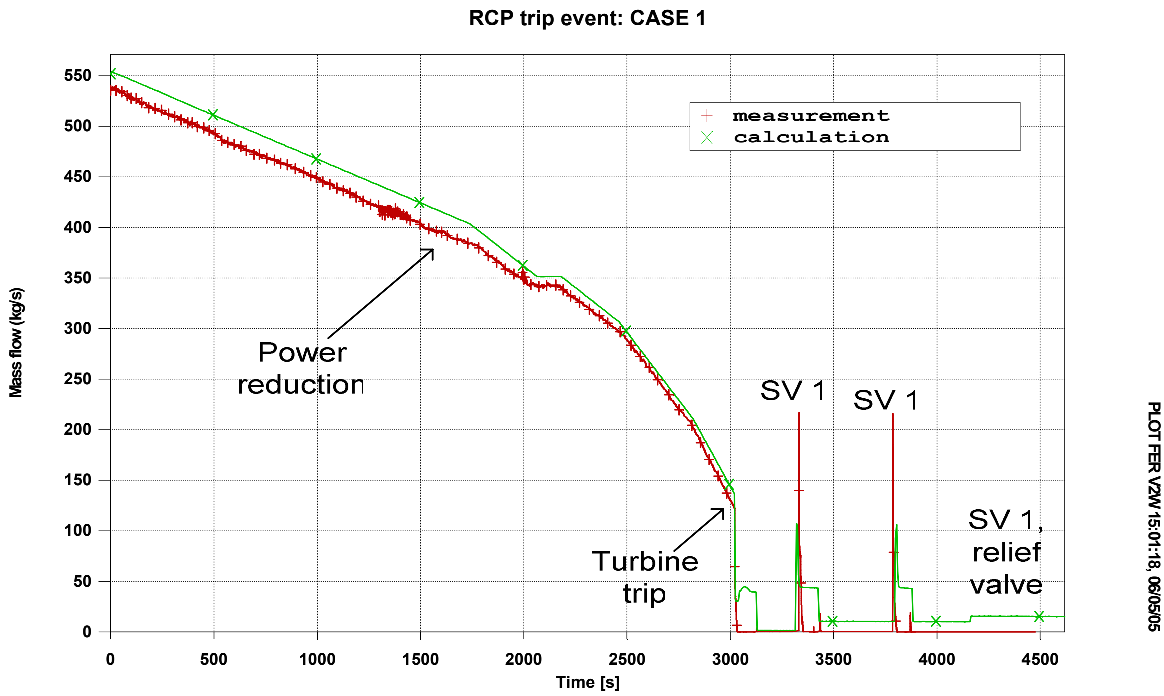

Fig. 1. Realistic analysis: SG 1 steam mass flow, power reduction (0-3020 sec).

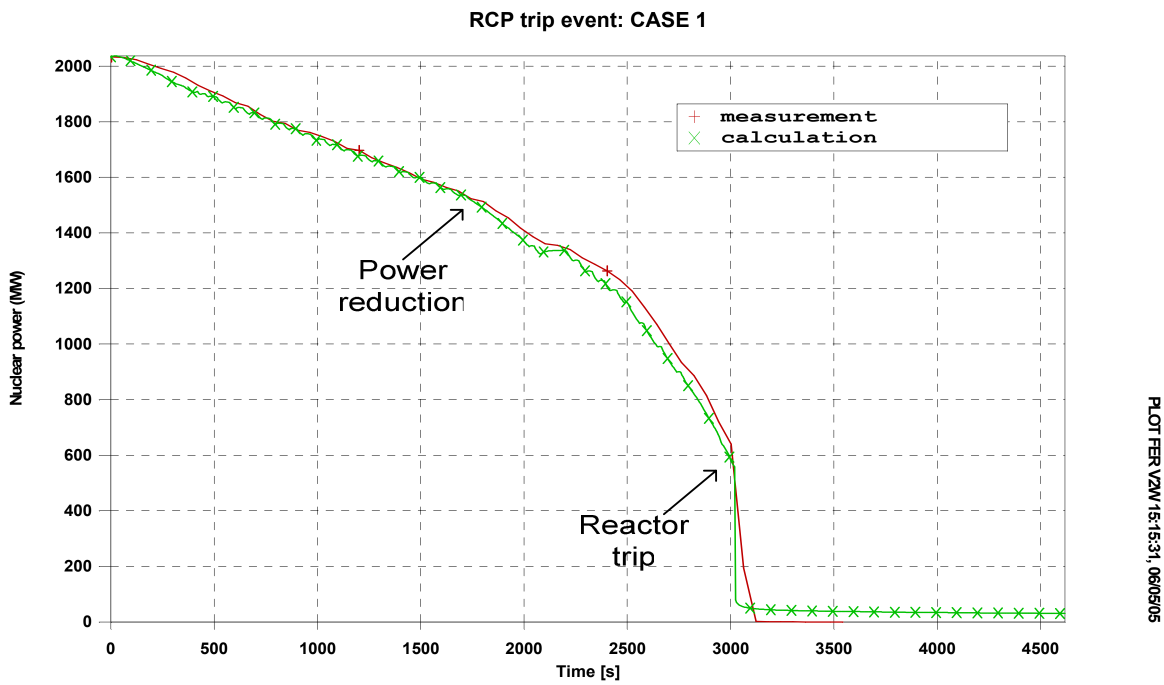

Fig. 2. Realistic analysis: nuclear power, power reduction (0-3020 sec).

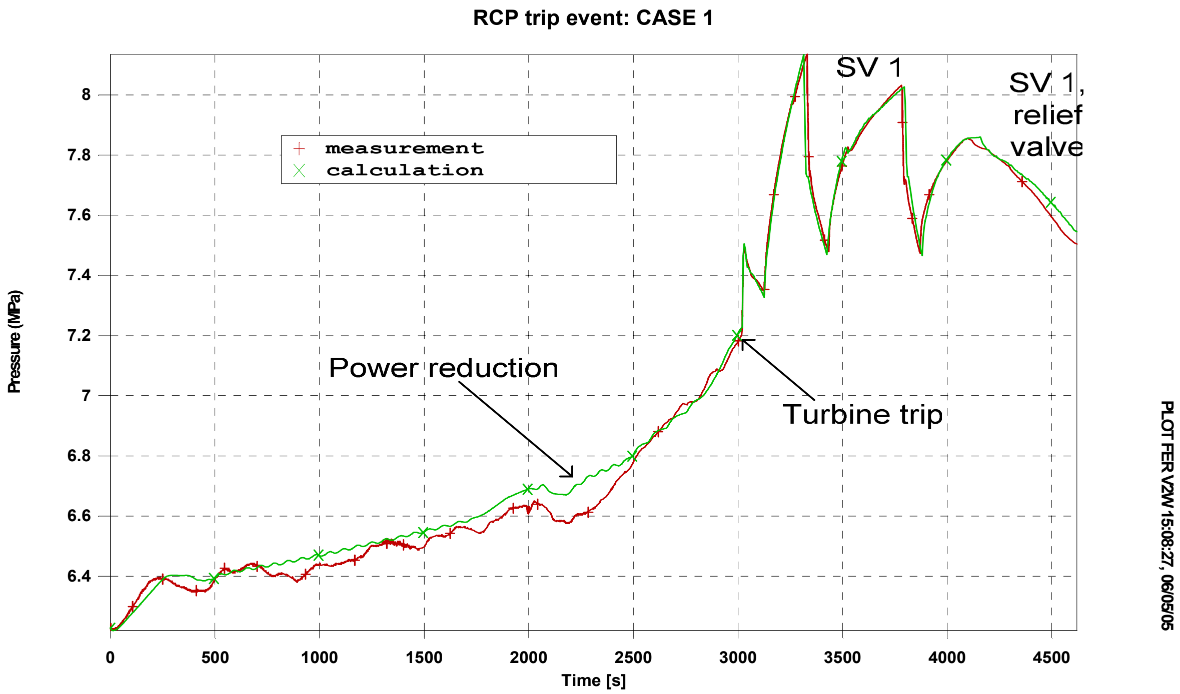

Fig. 3. Realistic analysis: SG 1 secondary pressure, power reduction (0-3020 sec).

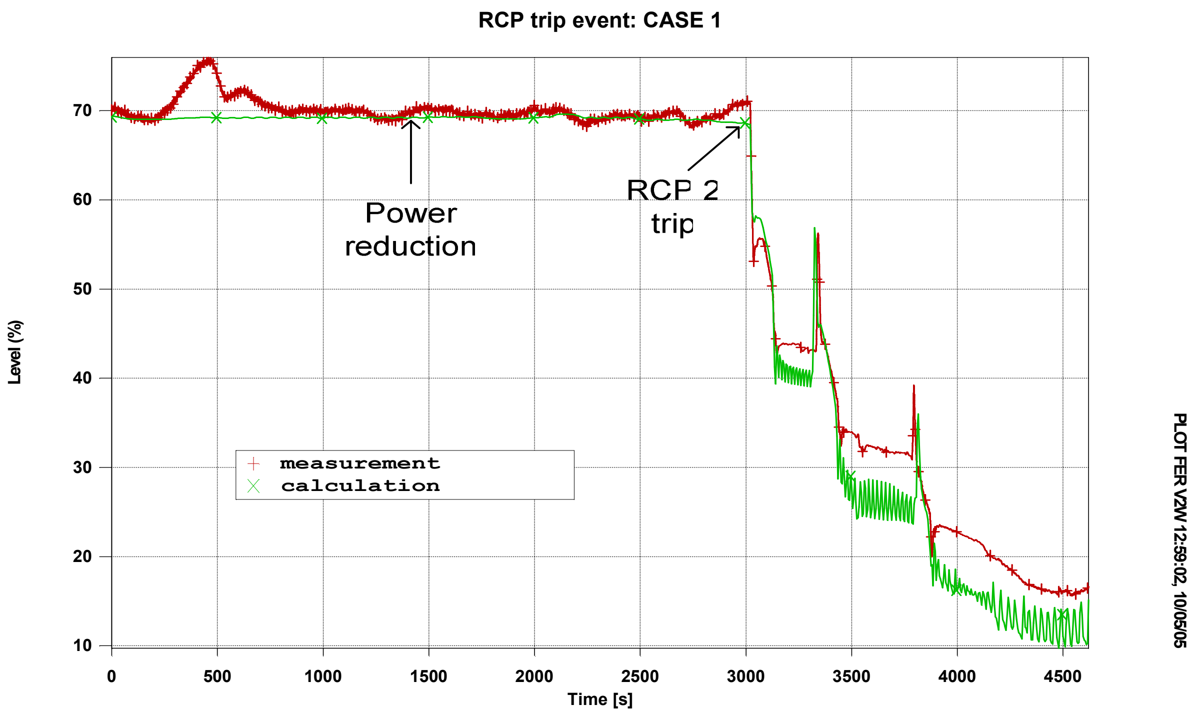

Fig. 4. Realistic analysis: SG 1 level, power reduction (0-3020 sec).

3 ANALYSES OF THE CCDP TRANSIENT SCENARIOS OF REACTOR COOLANT PUMP TRIP EVENT

The realistic analysis of the RCP trip event drew attention to the complexity of the RCP trip event scenario, as well as of interference between inherent plant behavior and operator actions. Additional analyses with different transient scenarios were found necessary to support the resolution of potential safety issues in the event of additional failures. Beside that, a Probability Safety Analysis (PSA) precursor event analysis has been performed for RCP trip event at NPP Krško, ref. [7]. To evaluate the safety significance of the operational events that may constitute important elements of accident sequences potentially leading to unacceptable consequences (precursor events) the concept of Conditional Core Damage Probability (CCDP) is used. The resulting precursor events that contribute to the CCDP according to the PSA evaluation of the RCP trip event are: Secondary side steam line break and the consequential effects and damage, Steam generator PORVs not opening during the initial pressure peak and Malfunction of AFW motor driven pumps 1 & 2.

Following cases based on CCDP transient scenarios have been analyzed using RELAP5/MOD3.3 code: case a: RCP 2 trip from 102 % power, MSIV isolation, AFW not available, case b: transient scenario from a) and with SG 1 PORV and SG 2 PORV responding and stuck open, case c: transient scenario from b) but only SG 1 PORV stuck open and case d: transient scenario from c) and Main Feedwater (MFW) trip at transient begin. Results of the RELAP5/MOD 3.3 calculations of the RCP 2 Trip event for CCDP scenarios are graphically presented in Figure 5through Figure 8. The main events are summarized in Table 3. Since in all analyzed cases no AFW flow was assumed the SG inventory depletes causing a loss in heat sink in all analyzed cases, Table 3, Figure 5. The core dry-out occurs in all analyzed cases as a result of the loss of hear sink. In the case b (both SG PORVs open) a quick decrease of pressurizer pressure, Figure 6, occurred at the beginning of the transient as a result of intensive heat removal thus causing an early Safety injection with a significant amount if injected mass into the RCS, Figure 7. As a consequence, in that case the longest time to core dry-out was obtained, Table 3, Figure 8. The shortest time to core dry-out (6630 sec) was obtained for the case d (1 SG PORV stuck open and early MFW trip).

TABLE 3. TIME TABLE OF MAIN EVENTS FOR THE CCDP ANALYSES

Event	CCDP case a Time (sec)	CCDP case b Time (sec)	CCDP case c Time (sec)	CCDP case d Time (sec)
Reactor trip, turb. trip	0	2.5	2.5	2.5
RCP 2 trip	3	0	0	0
MSIV isolation	105	0	0	0
Main feedwater isolation	57.6 (Reactor trip & Low Tavg)	57.6 (Reactor trip & Low Tavg)	57.6 (Reactor trip & Low Tavg)	0
SG 1/2 low-low level (SG mass<25174 kg)	1746/4458	980/1723	783/4286	172/2467
RCP 1 trip	6995	7185	6778	4722
Safety injection	3348 (Low-2 SL 1 pressure)	538 (Low-2 SL 2 pressure)	1394 (Low-2 SL 1 pressure)	220 (Low-2 SL 1 pressure)
Core dry-out	9050	9200	8790	6630

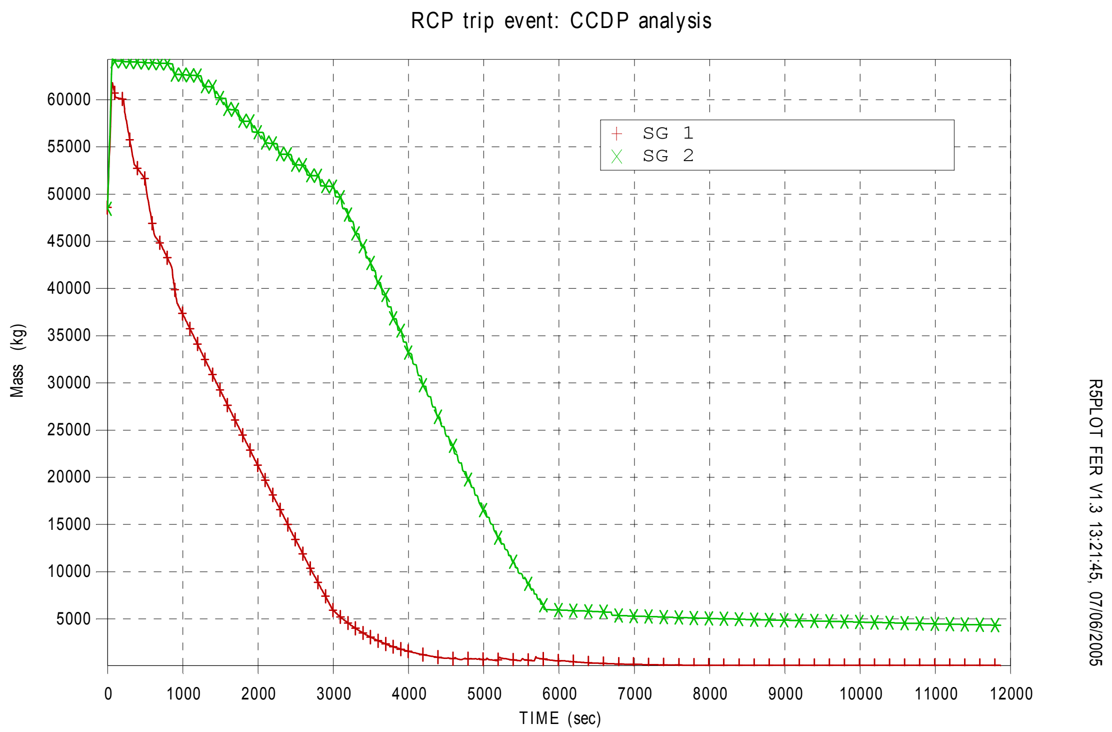

Fig. 5. CCDP case a: SG mass.

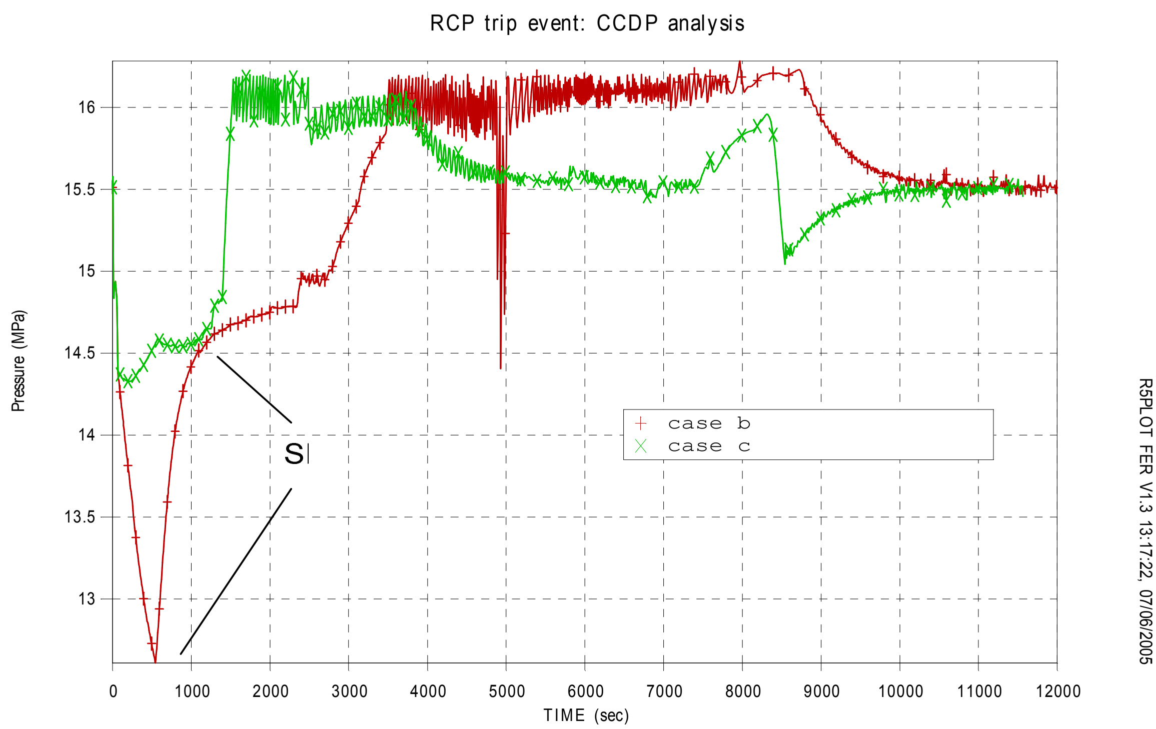

Fig. 6. CCDP case b (SG 1 and SG 2 PORV stuck open), case c (SG 1 PORV stuck open): pressurizer pressure.

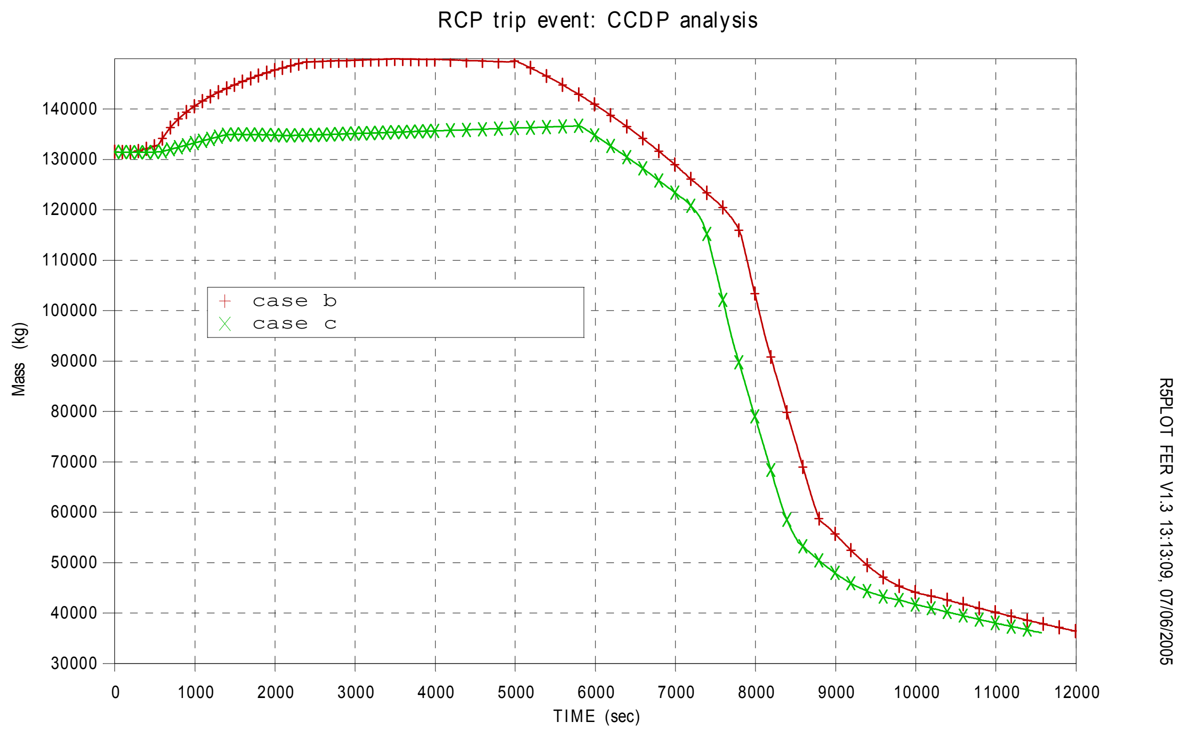

Fig. 7. CCDP case b (SG 1 and SG 2 PORV stuck open), case c (SG 1 PORV stuck open): RCS mass.

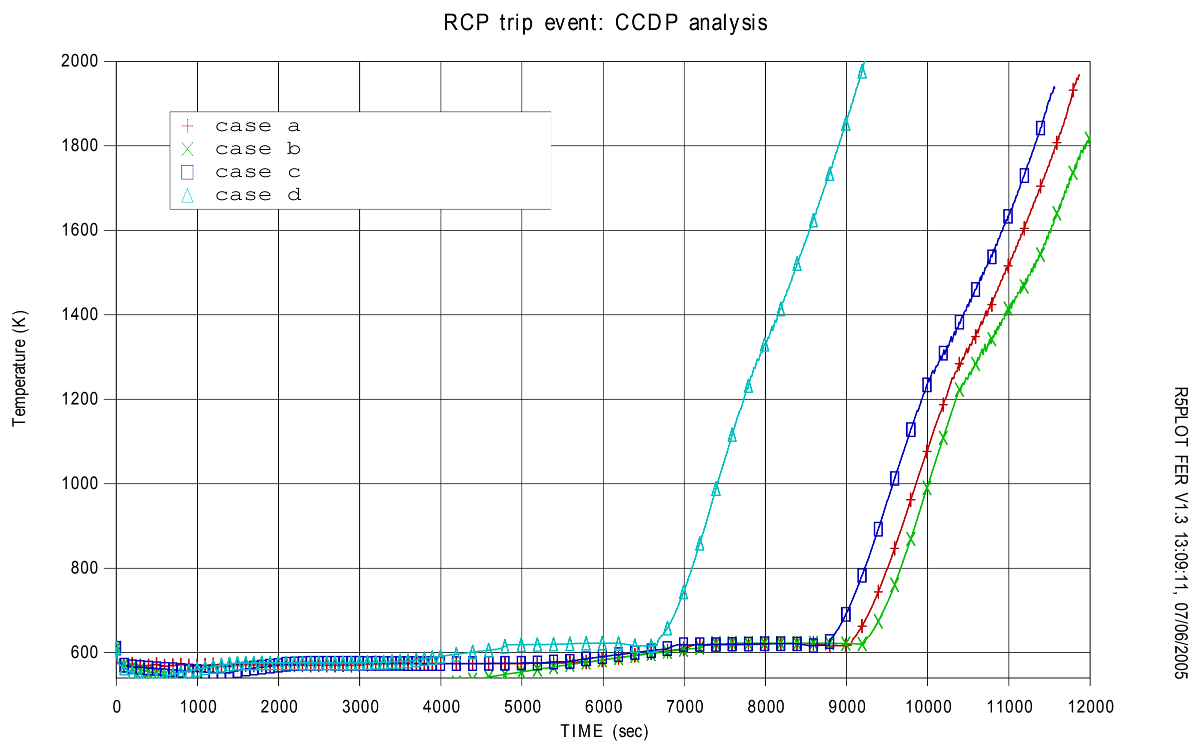

Fig. 8. CCDP analysis: Core cladding temperature (N=11).

4 CONCLUSION

The RELAP5/MOD 3.3 analyses of the RCP Trip event for NPP Krško for realistic and CCDP transient scenarios have been performed. Following conclusions can be drawn from the analyses.

- In the realistic analysis the realistic boundary and initial conditions were assumed. Calculation of the initial conditions for the RCP Trip event, i.e. the power reduction: 100 % – 28 % transient made demands on proper modelling of all control systems in the plant. The obtained differences between measured and calculated values are well within the acceptance criteria.
- The realistic plant behaviour influenced by operator actions has been taken into account in the realistic analysis. All major trends following the RCP Trip event were well predicted by the model. During the simulation the two major effects on transient results were identified: the SG 1 relief or SV 1 open/close behaviour and AFW flow actuation. The major characteristics of the real transient have been reproduced, e.g. the results for SG pressure, SG level, average temperature, etc.
- The presented RELAP5/MOD 3.3 analysis has demonstrated the appropriateness of the developed NPP Krško nodalization with models of the plant control systems for use in the realistic transient simulation
- CCDP RELAP5/MOD3.3 analysis showed that the hypothetical case with 1 SG PORV stuck open and early MFW isolation leads to core dry-out within 2 hours

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DETERMINISTIC ANALYSIS OF MCP TRIP EVENTS AT IGNALINA NPP

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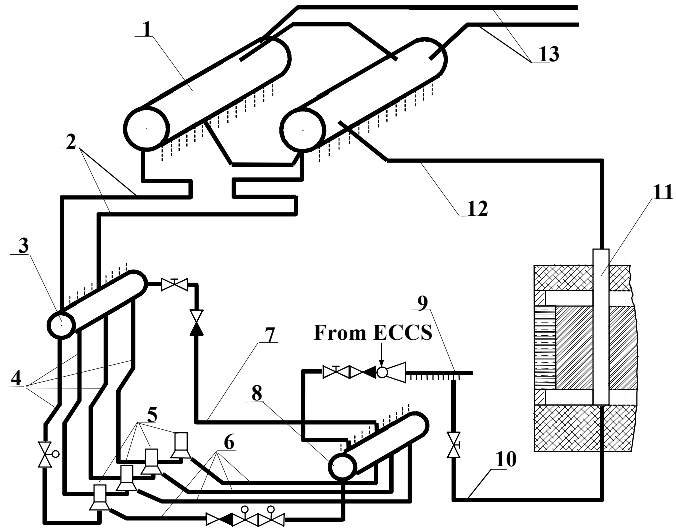
Abstract. For the cooling of water forced circulation through the RBMK-1500 reactor at the Ignalina Nuclear Power Plant eight Main Circulation Pumps (MCPs) are employed. These pumps are joined into groups of four pumps each (three for normal operation and one on standby). There were few events when one or few MCPs were inadvertently tripped. On 14 May, 1996 one MCP at Ignalina Unit 2 was inadvertently tripped. The similar event took place on 23 January, 1998. During this event, the MCP check valve failed to close, causing a recirculation loop to develop by means of a reversed flow through tripped pump. On 31 July, 2000 three MCPs at Ignalina Unit 2 were tripped one after another, due to inadvertent activation of fire protection system. Simultaneous trip of all MCPs occurred on 26 March, 1986. In the case of one MCP trip 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. In the case of all MCPs trip, the coolant during the first few seconds is supplied to the reactor by pumps coastdown. Later the reactor is cooled by natural circulation. This paper presents investigation of one and all MCP trip events at the Ignalina NPP by employing best estimate code RELAP5. For single tripped MCP and simultaneous trip of all MCPs cases uncertainty and sensitivity analysis was performed. For that purpose the GRS (Germany) developed System for Uncertainty and Sensitivity Analysis package was used. Uncertainty analysis of flow energy loss in different parts of the Main Circulation Circuit, initial conditions and code-selected models was performed. On the basis of these analyses recommendations for the improvement of the Ignalina NPP RELAP5 model have been developed.

1 INTRODUCTION

The Ignalina Nuclear Power Plant is a twin-unit with two RBMK-1500, graphite moderated, boiling water, multichannel reactors. Schematic representation of one Main Circulation Circuit loop is given in the Figure 1. The MCC is divided into two halves – the left and right loops. The MCPs (5) are joined in groups of four pumps each (three for normal operation and one on standby). In the all pumps trip case, the coolant during the first few seconds is supplied to the reactor by pumps coastdown, due to high inertia of pump flywheel. Later the natural circulation through the core is established. The MCPs feed common pressure header (8) on each side of the reactor. Each pressure header provides coolant to 20 Group Distribution Headers (9), each of which in turn feeds from 38 to 43 fuel channels (11). The coolant flow rate through individual fuel channels is regulated by mounted in the lower water communication lines Isolating and Control Valves (10). Coolant passing through fuel channels is boiled and part of the water is evaporated. Steam-water mixture through steam-water communication lines (12) flows to Drum Separators. Separated in the DS steam through steam lines (13) is supplied to turbines. A detailed description of the Ignalina NPP can be found in [1].



1 – DS, 2 – downcomers, 3 – MCP suction header, 4 – MCP suction piping, 5 – MCPs, 6 – MCP discharge piping, 7 – bypass line, 8 – MCP pressure header, 9 – GDHs, 10 – lower water communication line, 11 – fuel channel, 12 - steam-water communication line, 13 – steam lines.

Fig. 1. Schematic representation of one loop of the RBMK-1500 MCC.

For the MCP trip events investigation best estimate model of RELAP5 Ignalina NPP RBMK-1500 reactor cooling circuit was developed by Lithuanian Energy Institute. This model includes forced circulation circuit, steam lines and safety systems, necessary for transient and accident processes analysis. Detailed description of RELAP5 nodalization scheme is presented in [2]. Obtained results were compared with measurements from Ignalina NPP. Initial and boundary conditions (coolant pressure, flow rate, feed water temperature, amount of steam for in-house needs, reactor power, flow energy loss in different MCC components) and RELAP5 code models, assumptions and correlations may impact the calculation results uncertainty.

The GRS method SUSA 3.2 [4] for the determination of uncertainties was used for sensitivity and uncertainty calculation. For the one and all MCPs trip analysis the following parameters, whose initial values may have the greatest impact on the simulation results, on the basis of the knowledge from earlier performed benchmarking calculations are selected: 1 - pressure in the drum separator, 2 - coolant flow rate through the MCPs, 3 - feed water temperature, 4 - amount of steam for in-house needs and 5 - reactor thermal power. For the analysis also the following RELAP5 code modelling parameters are selected: 6 - water packing, 7 - vertical stratification, 8 - modified PV term in the equations, 9 - CCFL (counter current flow limit), 10 - thermal front tracking, 11 - mixture level tracking, 12 - non-homogeneous media. Selected RELAP5 code parameters are varied 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 conditions are excluded due to the fact that these parameters do not have impact to the results in such region. In the basic case of calculations some of the code models were disabled. It was due to the fact that they did not have impact to the results. However, in the sensitivity and uncertainty analysis none of the potential contributors to the uncertainty of the results can be excluded. Additionally one parameter, which might impact the coolant flow regime in the reactor, was selected – 13 flow energy loss in MCC components.

2 BEST-ESTIMATE ANALYSIS OF SINGLE MCP TRIP AND ALL MCP TRIP EVENTS

2.1 Single MCP trip analysis

On May 14, 1996 one MCP at Ignalina Unit 2 was inadvertently tripped. 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 was operated in power control mode. Six MCPs were in operation, providing coolant flow 7700 m^3/h to the one and 7866 m^3/h to another MCC loops.

The initiating event was the switch off of two preferred electrical buses. It led to one of six MCPs trip. 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. Reactor power was reduced to 60 % from design (4800 MW_{th}) in response to one MCP trip signal. The turbine generator (which before the accident operated in the power control mode) switched from power control mode to steam separators pressure maintenance mode.

Using RELAP5 model calculated pressures versus real measured plant data are presented in Figure 2. As it is seen from figures, the divergence of initial values of pressure can be explained by measurements errors - initial measured pressure in DS and in pressure header is higher than calculated by RELAP5 model (basic case in Figure) and the initial measured pressure in Suction Header is lower than calculated. To cover these two extremes uncertainty analysis is performed using two-sided tolerance limit (with 0.95 of probability and 0.95 confidence). According to Wilk's formula [3] it is needed to perform at least 93 code runs. In this case 100 runs were performed. In Figure 2 through Figure 5 curves "Maximum" and "Minimum" bound values of all performed 100 calculations: curve "Maximum" represents the maximum values of all 100 runs, while curve "Minimum" – minimum values.

Comparison between calculated flow rates obtained by RELAP5 model and real measured data is presented in Figure 3 through Figure 5. After MCP trip the coolant flow rate through this pump is dropped to zero approximately within 5 seconds (see Figure 4). Coolant flow rate through MCPs of intact MCC loop after one MCP trip increases by ~400 m^3/h (see Figure 3). This increase is due to decrease of the reactor core resistance to coolant flow after reactor power decreasing. The throughput of each of two running pumps increased by ~1500 m^3/h (see Figure 5). However, the total coolant flow through the affected loop decreased from 23500 m^3/h to 19000 m^3/h. In Ignalina NPP measurement of flow rate through MCPs is based on pressure losses measure in the throttling devices. In Figure 3 through Figure 5 the observed spread of measured coolant flow rates through MCPs can be explained by imperfection of measuring devices and information processing system.

In case of one MCP trip the throughput of two running pumps in affected MCC loop increased, however the total coolant flow through affected loop decreased. The main question arises whether this coolant flow rate is sufficient for adequate core cooling. Therefore, from computational results for the analysis coolant flow rate through one running MCP of affected loop was selected. In Figure 5 maximum, minimum values of all performed calculations, basic case calculation results and Ignalina NPP data are presented. All presented calculated parameters are in good agreement with the real plant data, because the most of main measured parameters values are in calculated uncertainty range (Figures 2–5).

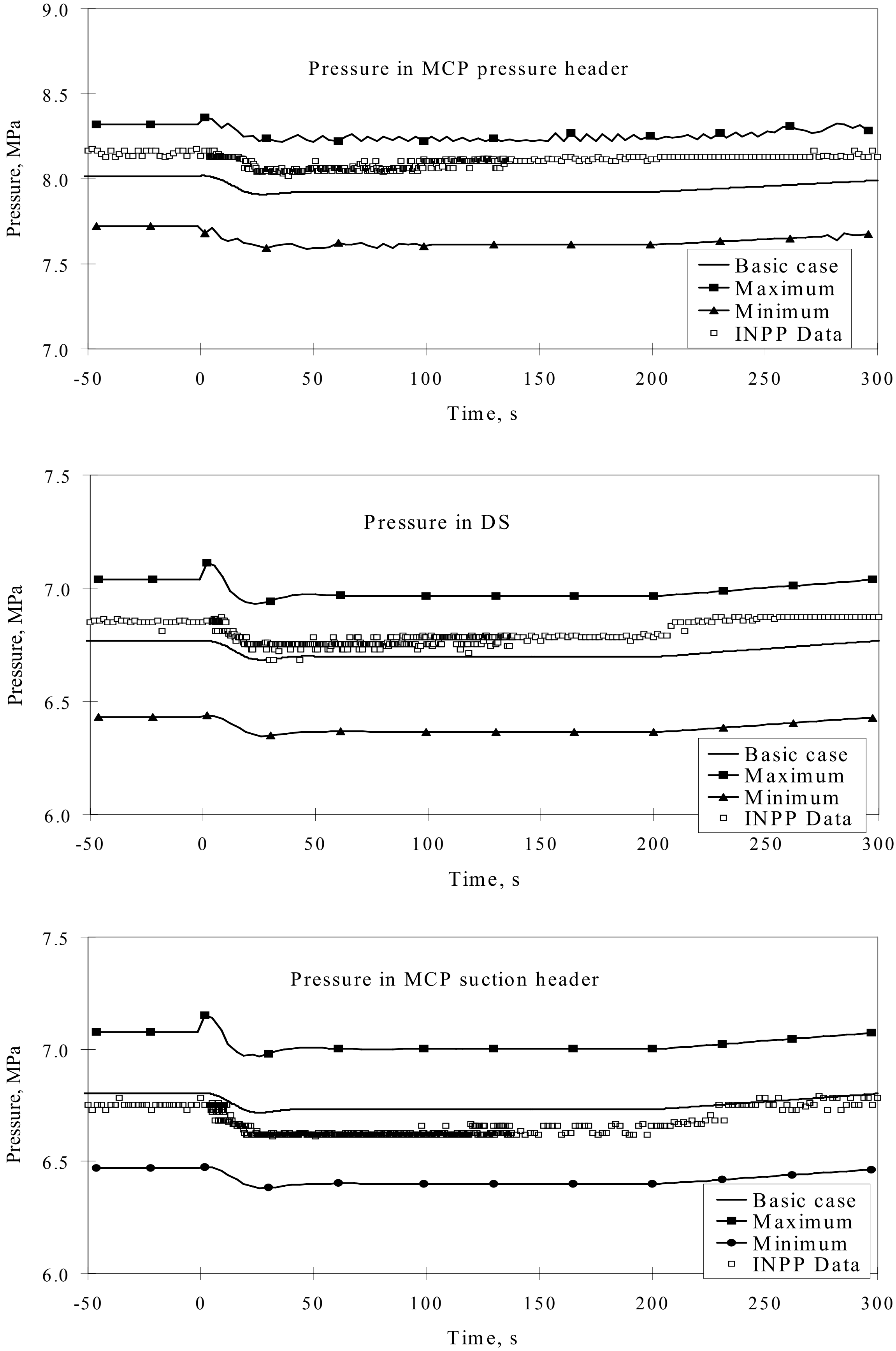


Fig. 2. Pressure in the main circulation circuit.

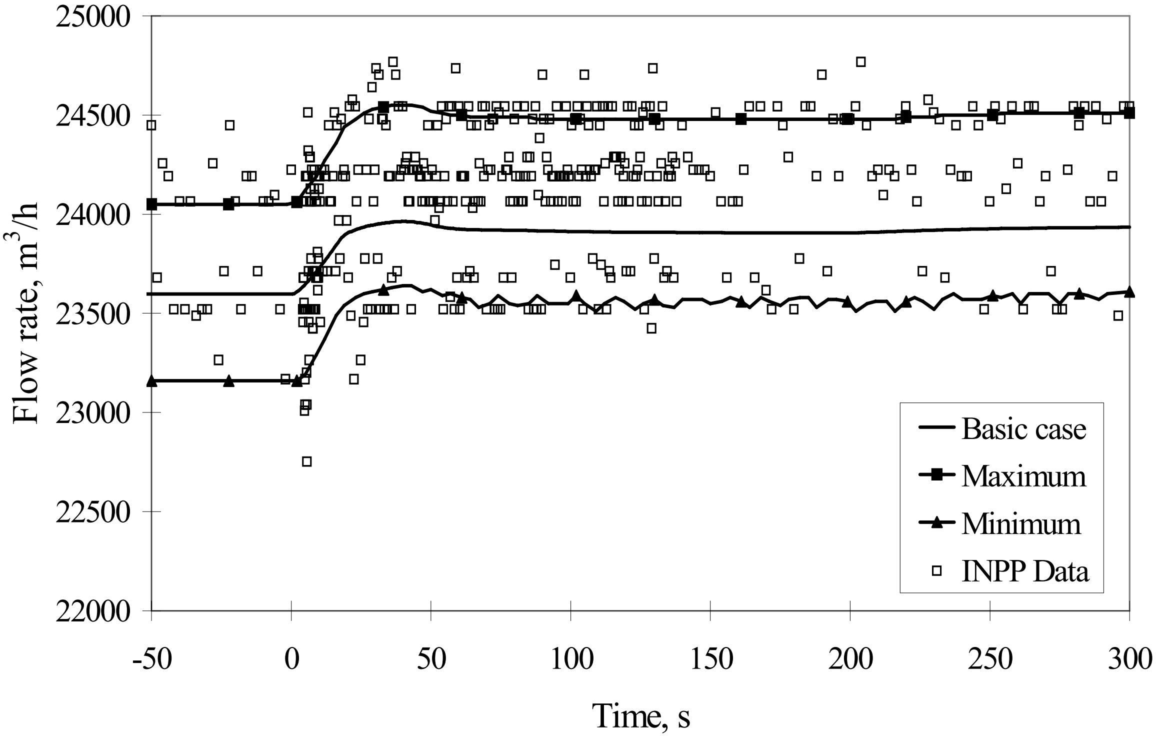


Fig. 3. Coolant flow rate through the MCP of the intact MCC loop.

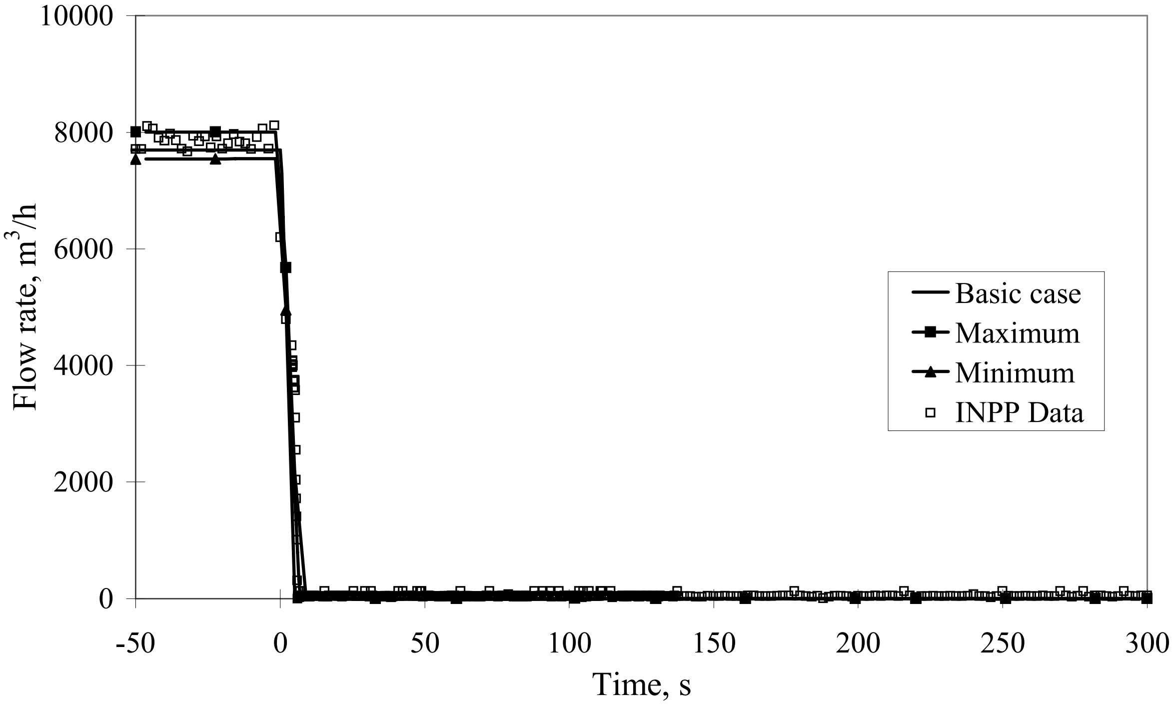


Fig. 4. Coolant flow rate through tripped MCP.

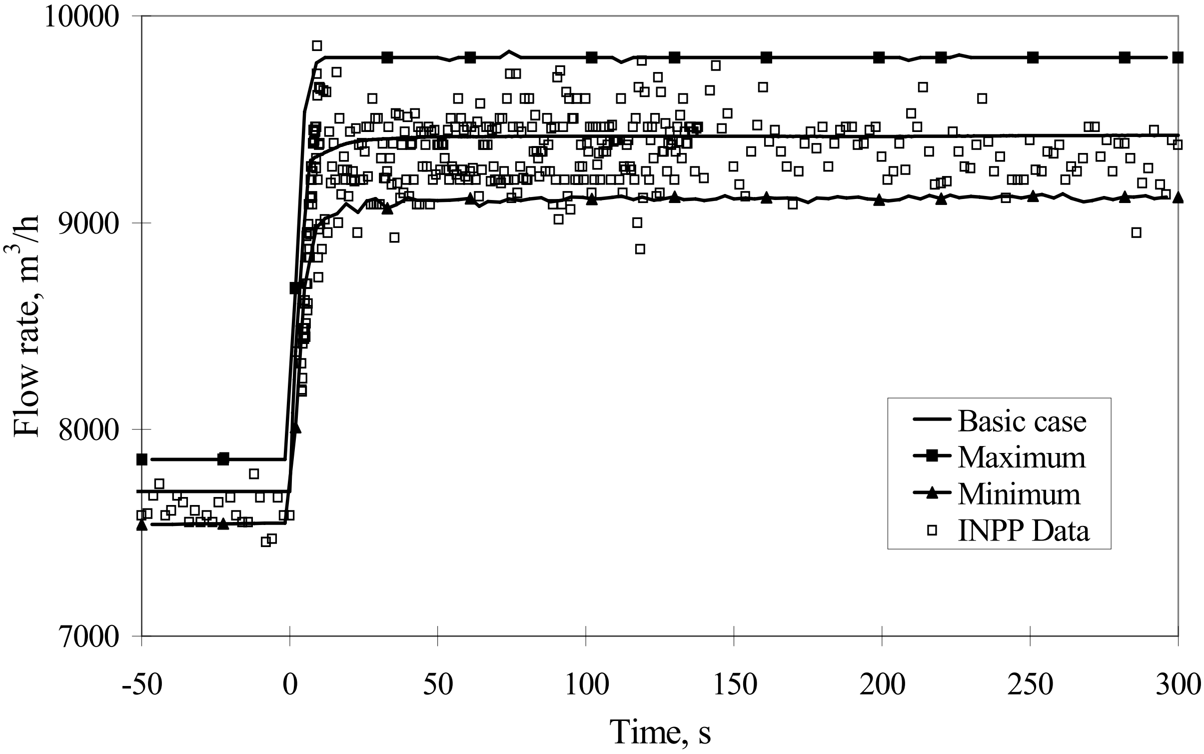


Fig. 5. Coolant flow rate through MCP of affected MCC loop.

Most sensible parameter, which defined possible damage during such type of accidents, is calculated critical heat flux margin coefficient from the side of fuel assembly to coolant. This coefficient is defined as relationship between calculated critical and real heat transfer fluxes. If this critical heat flux margin coefficient is greater than 1, it means that no critical heat flux will be observed in any fuel channel segment and drying of FC will not occur.

As it is seen from Figure 6, the margin to CHF in the intact MCC loop increases after reactor power decrease. The margin to CHF in the affected MCC loop at the beginning of accident slightly decreases due to decreased coolant flow rate. The decrease of coolant flow rate and reactor power affects on CHF in opposite manner. However the influence of power decrease is higher than influence of coolant flow rate decrease. Later heat flux decreases faster than calculated critical heat flux, and margin to CHF starts to increase. During all accident the margin to CHF is greater than 1. The changes in the margin to CHF demonstrate the reactor cooling regime changes after one MCP trip. Therefore, uncertainty analysis was performed for two states of reactor: steady state conditions and trip of one MCP.

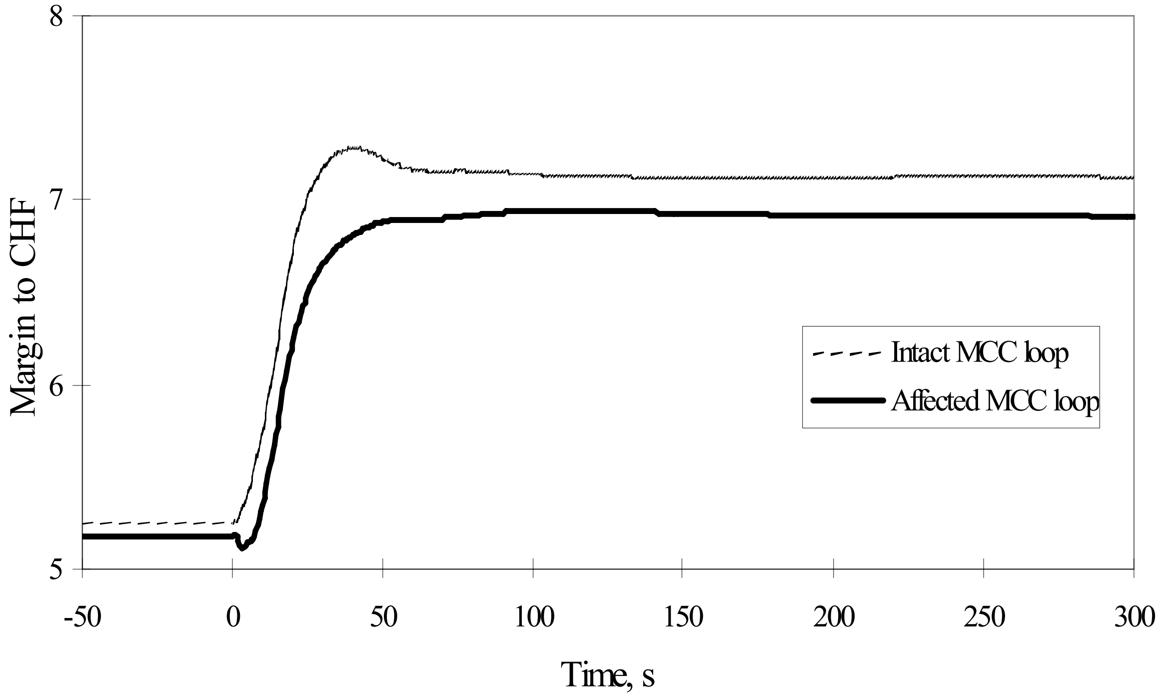


Fig. 6. Margin to CHF.

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

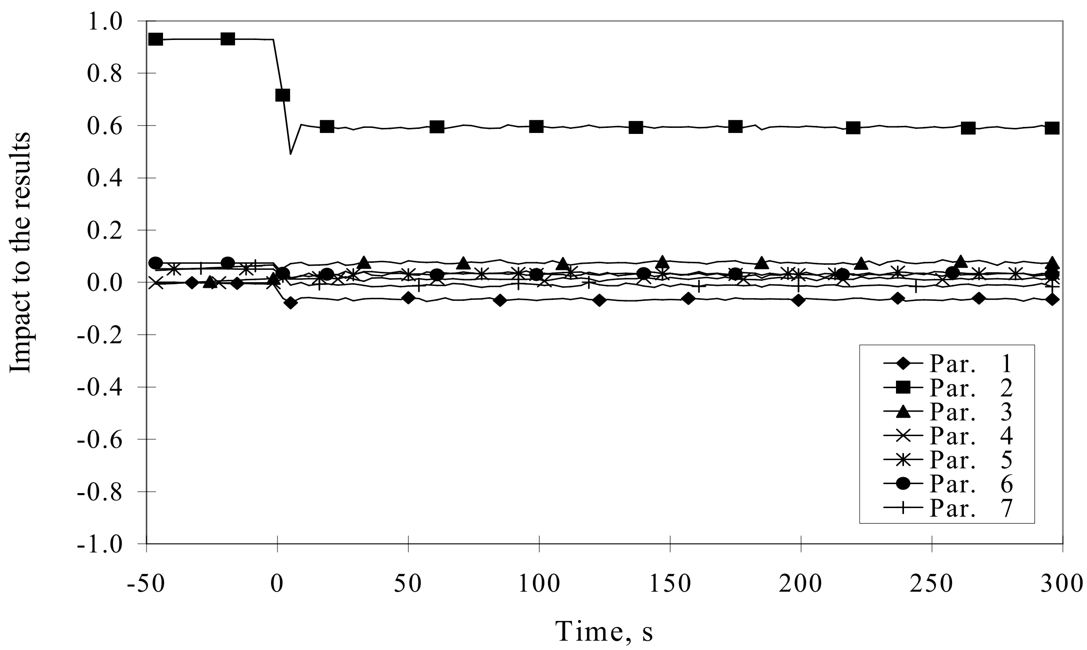


Fig. 7. Impact of parameters No.1 – No.7 to the coolant flow rate through one running MCP of affected MCC loop.

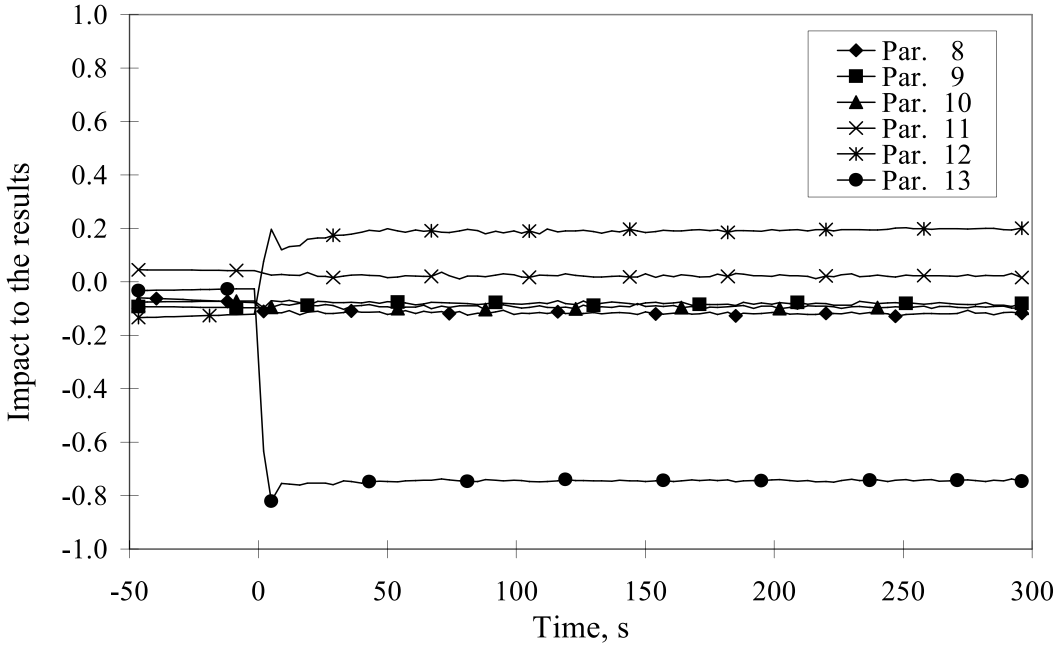


Fig. 8. Impact of parameters No.8 – No.13 to the coolant flow rate through one running MCP of affected MCC loop.

The performed analysis shows that after one MCP trip 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. Increased margin to critical heat flux (Figure 6) in the worst case (lowest flow rate) shows that reactor core is reliably cooled by forced circulation in case of one MCP trip.

2.2 Loss-of-all MCPs event analysis

Loss-of-all-MCPs event occurred at Ignalina NPP on March 26, 1986. During this event all six operating MCPs at Ignalina Unit 1 were tripped simultaneously. Before this event the reactor operated at thermal power level of 4650 MW. In response to multiple pump trips, an emergency protection signal AZ-1 was generated and the reactor was shut down. The MCC flow decreased in response to the MCPs coastdown. Long-term flow was due to natural circulation in the MCC.

RELAP5 analysis results were compared against the plant data. Uncertainty and sensitivity analysis is performed using two-sided tolerance limit (with 0.95 of probability and 0.95 confidence). 100 runs were performed. Data of the flow rate in twelve individual channels during natural circulation regime was available and are presented in Figure 9. The calculated maximum flow rate through maximum loaded FC and calculated minimum flow rate through minimum loaded FC and real plant data showed reasonable agreement. Coolant flow rate through one MCC loop is presented in Figure 10. As it is seen from the figure, measured values are reduced, and become equal 0 after about 115 seconds from the beginning of accident. Comparison of coolant flow rate measures through the fuel channels and through one MCC loop shows that the last measures by employing throttling devices are unreliable when flow rate through one MCC loop decreases down to 7000 m^3/h. The flow rates show a coastdown associated with the loss of forced circulation by the MCPs. The coastdown continues during first 40 seconds from the beginning of the transient. Later, coolant natural circulation was established at flow rate equal to about 15 % of the initial flow.

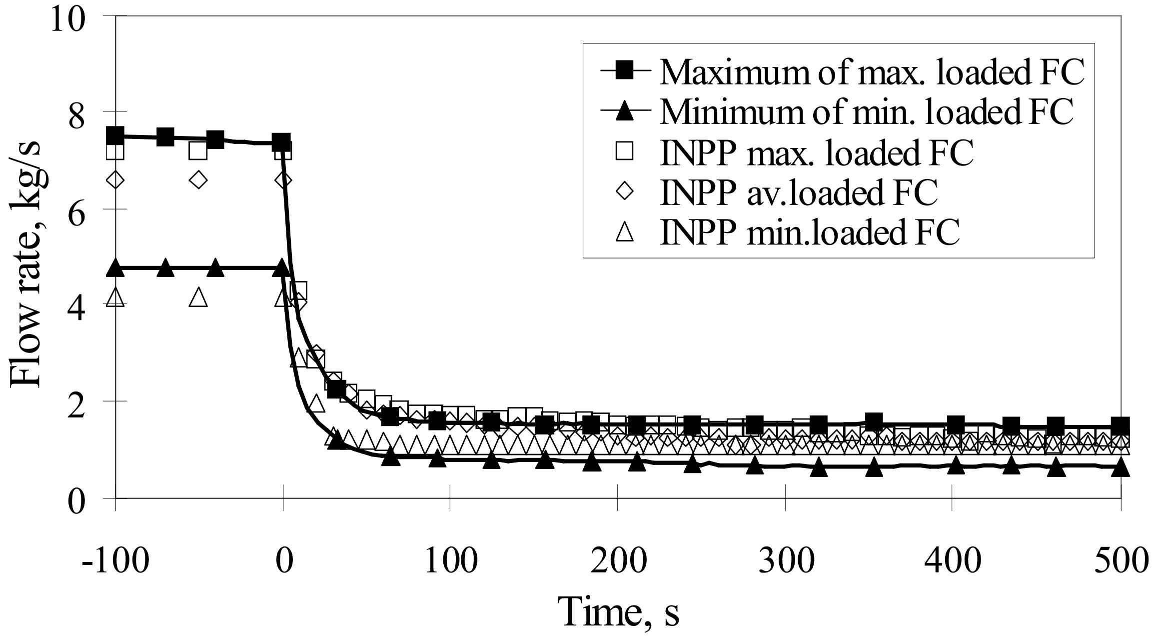


Fig. 9. Coolant flow rate through individual channels.

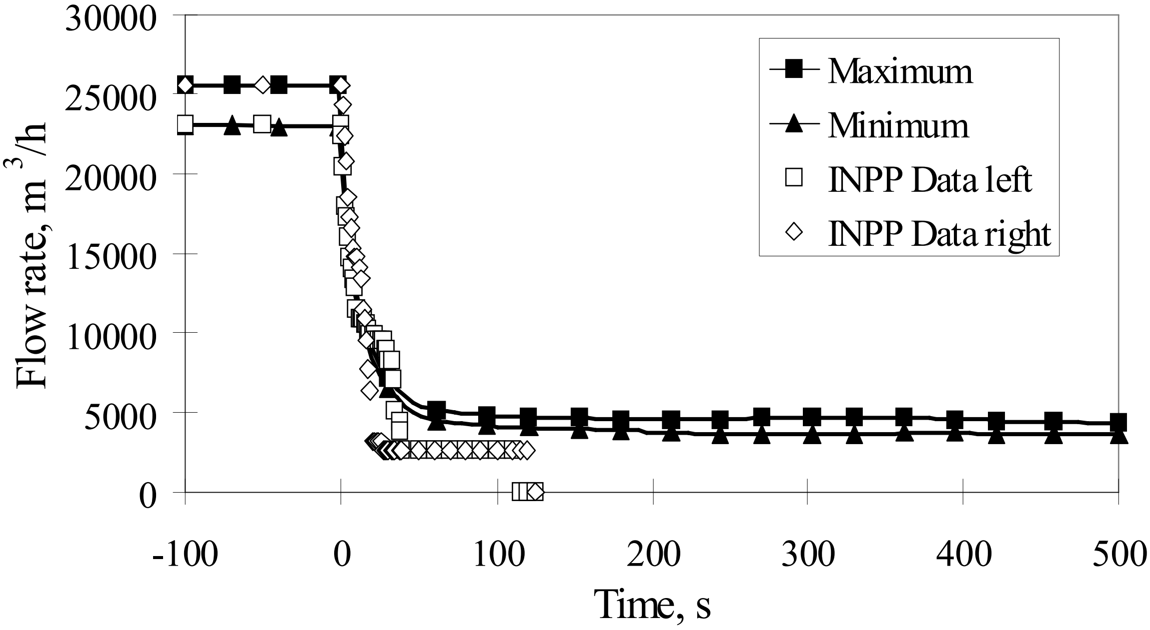


Fig. 10. Coolant flow rate through one MCC loop.

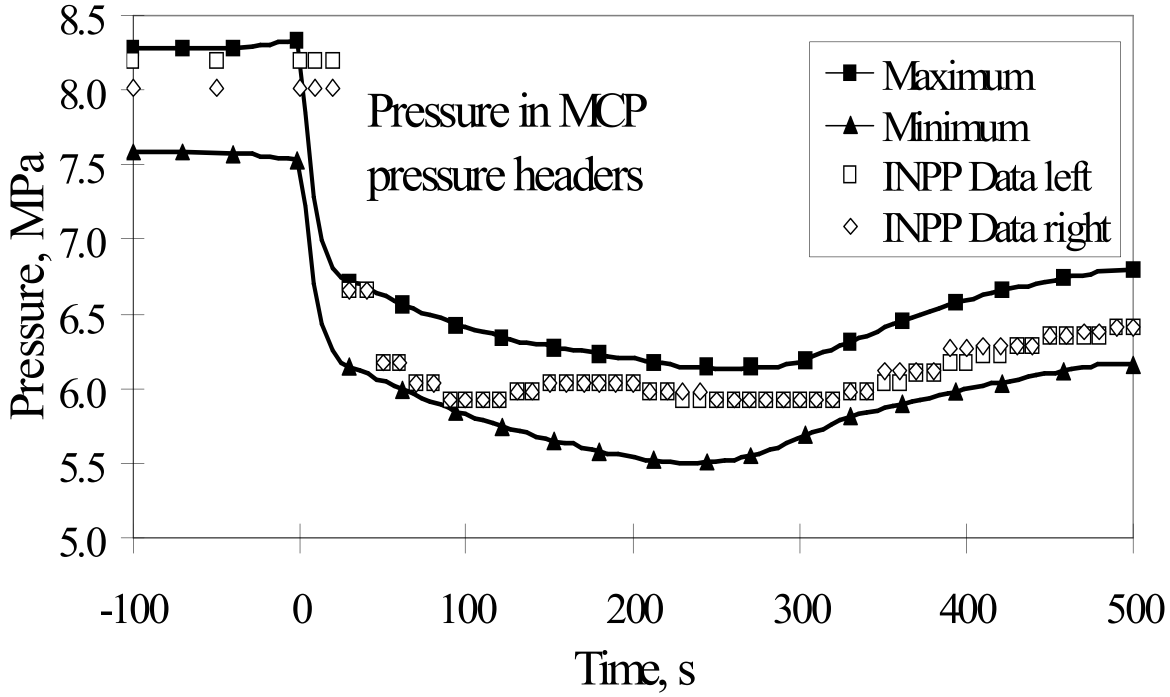
Figure 11 shows pressure in the MCC. Pressure in the pressure header decreases immediately after MCP trip, because the MCP head is decreasing. During first seconds, when reactor is not shutdowned, coolant flow rate decrease through the core causes short-term increase of steam generation. Increase of steam generation causes the short-term pressure increase in the DS. Steam generation in the core decreases and pressure in the MCC also decreases after reactor shutdown. The turbines reloading process is starting immediately after reactor shutdown.

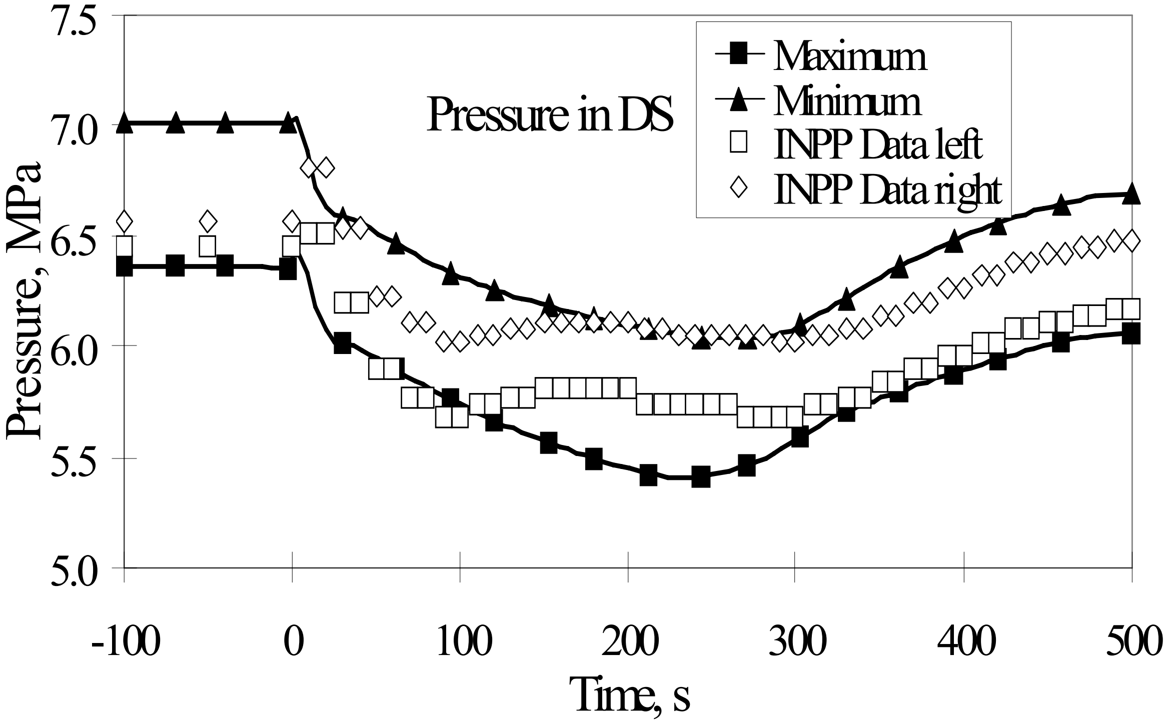
When the turbine control valves are closed, the pressure starts to increase again. Furthest pressure changes depend on amount of steam for in-house needs. As could be seen from the presented comparison, pressure losses in different parts of the MCC are predicted correctly.

From a number of the RELAP5 output results one of the important technological parameters which shows the existence of natural circulation for uncertainty and sensitivity analysis was selected – coolant flow rate through one MCC loop. The performed analysis shows that the largest impact to the calculation results has the selection of homogeneous or non-homogeneous media model (see Figures 12 and 13). Homogeneous media model selection is a non-physical conservative assumption and it is not recommended for best estimate codes.

The parameter-dependent sensitivity analysis shows, that initial plant conditions (coolant flow rate, pressure in the DS, feed water temperature, reactor power) and flow energy loss in different MCC parts (ICV position) have only insignificant influence on natural circulation regime.

The performed uncertainty and sensitivity analysis shows that reactor core is reliably cooled due to natural circulation regime.





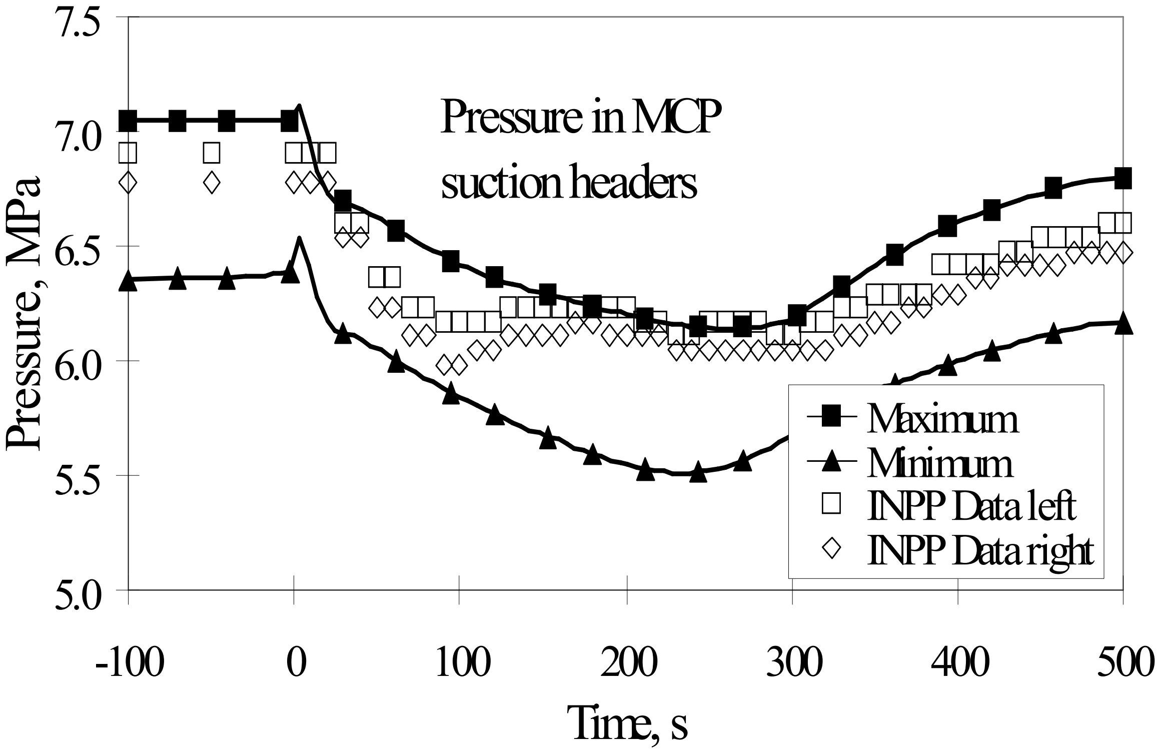


Fig. 11. Pressure in the MCC.

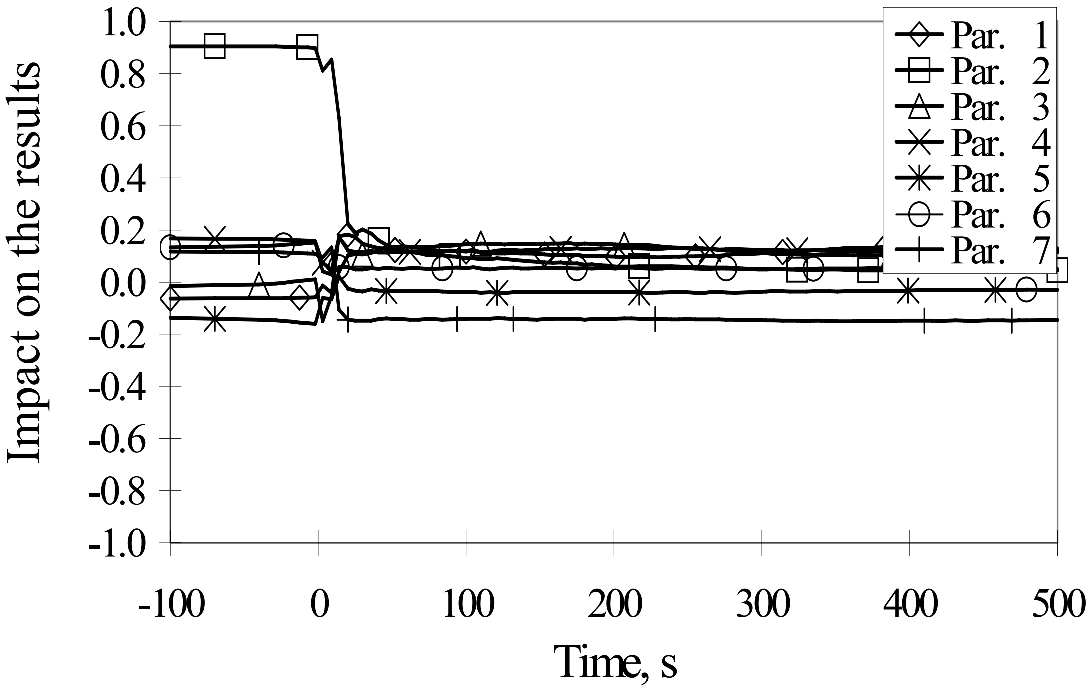


Fig. 12. Impact of parameters No.1 – No.7 to the coolant flow rate through one MCC loop in case of all MCP trip.

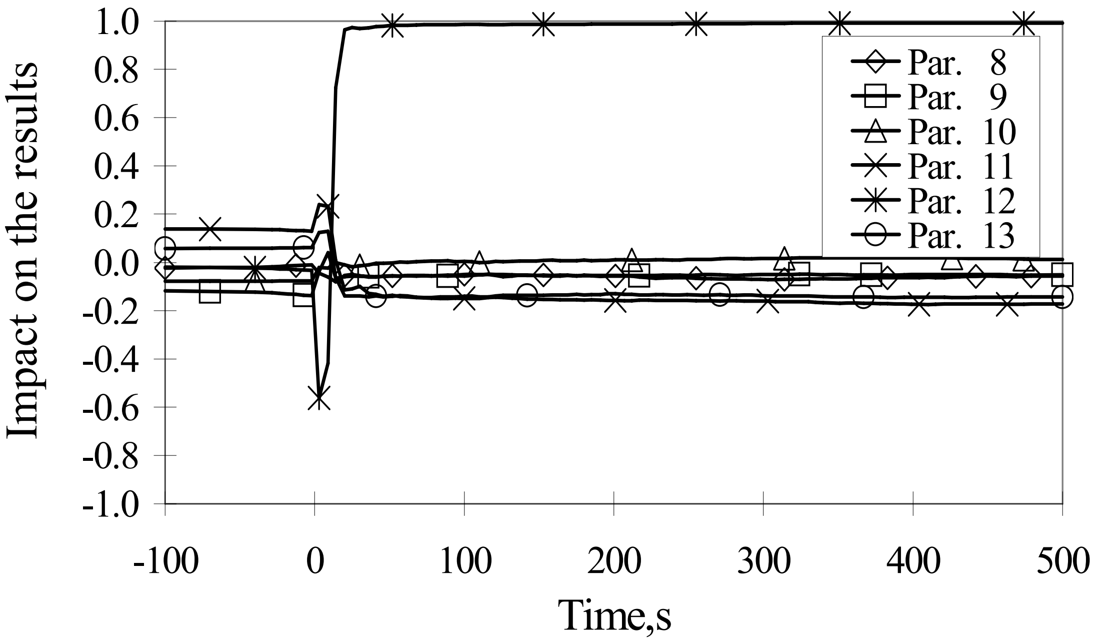


Fig. 13. Impact of parameters No.8 – No.13 to the coolant flow rate through one MCC loop in case of all MCP trip.

3 SINGLE MCP TRIP WITH CHECK VALVE FAILURE AND SEQUENTIAL MCP TRIP ANALYSES

In the previous chapter described analyses showed that the largest impact on the calculation results has the selection of the homogeneous media model in coolant natural circulation case. Because homogeneous media model is too conservative and could be too big source of uncertainties, in accident analyses described in this chapter only the non-homogeneous model was used. Because in case of a single tripped MCP the largest impact to the calculation results uncertainties has the flow energy loss in ICV, in the analysis reactor core was represented by more channel groups. Implementation of these principles ensures smaller uncertainties of calculation results. Therefore, uncertainty and sensitivity analysis for the below presented cases was not performed.

3.1 Single MCP trip with check valve failure modeling

Single MCP trip event, which was accompanied by failure of check valve, took place on January 23, 1998. Before this event the reactor operated at the power level of 3700 MW_{th}. One turbine generator was operated in a pressure maintenance mode and the other was operated in power control mode. Six MCPs were in operation, providing coolant flow 7750 m^3/h (23250 m^3/h to each MCC loop).

At 10:21 one MCP was switched off by mistake. The reactor power was reduced down to 2860 MW_{th} about 14 seconds after the beginning of the accident by emergency protection system. As the flow through the pump dropped to zero (approximately 4 seconds after the beginning of the accident) the check valve downstream of this MCP had to start to close, however, it remained to be open. The operator noticed that coolant flow through the left loop of the reactor calculated as a sum of fuel channel flow rates was 13400 m^3/h while the remaining running pumps provided a coolant flow of 20000 m^3/h. On this base the conclusion has been made that the check valve downstream of the tripped pump failed to close. At 10:23 the operator closed the throttling regulating valve of the tripped MCP. At 10:24 reactor power was reduced manually down to 2100 MW_{th}. At 10:25 the operator decreased reactor power down to 1900 MW_{th} and increased the throughput of two running pumps by opening their throttling-regulating valves.

Analysis results are shown in Figures 14–16. The comparison of calculated and measured pressure behaviour in MCC is presented in Figure 14. Because of the check valve failure, the flow through the tripped pump reversed. At the same time, each of two running pumps increased its throughput because the pressure gradient across the pumps was reduced due to the flow bypass. The increased pump throughput effectively compensated for the reversed flow through the failed valve. The net flow supplied to the affected core side decreased from 21000 m^3/h to 13500 m^3/h (see Figure 15). The throttling-regulating valve of the tripped pump closed after 90 seconds from the beginning of the transient and decreased the reversed flow by 2000 m^3/h. At this moment the real coolant flow through the reactor side reached about 16000 m^3/h. The opening of throttling-regulating valves of running pumps after 220 second increased the coolant flow to 17000 m^3/h.

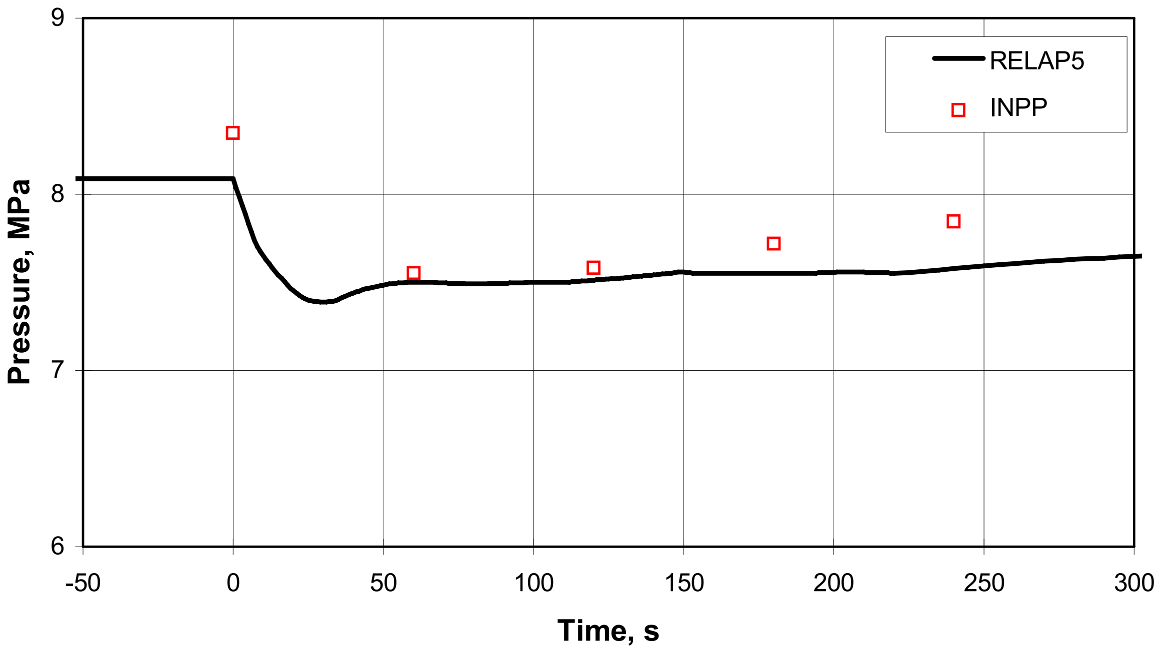


Fig. 14. Single MCP trip with failure of check valve. Pressure in pressure header of affected loop.

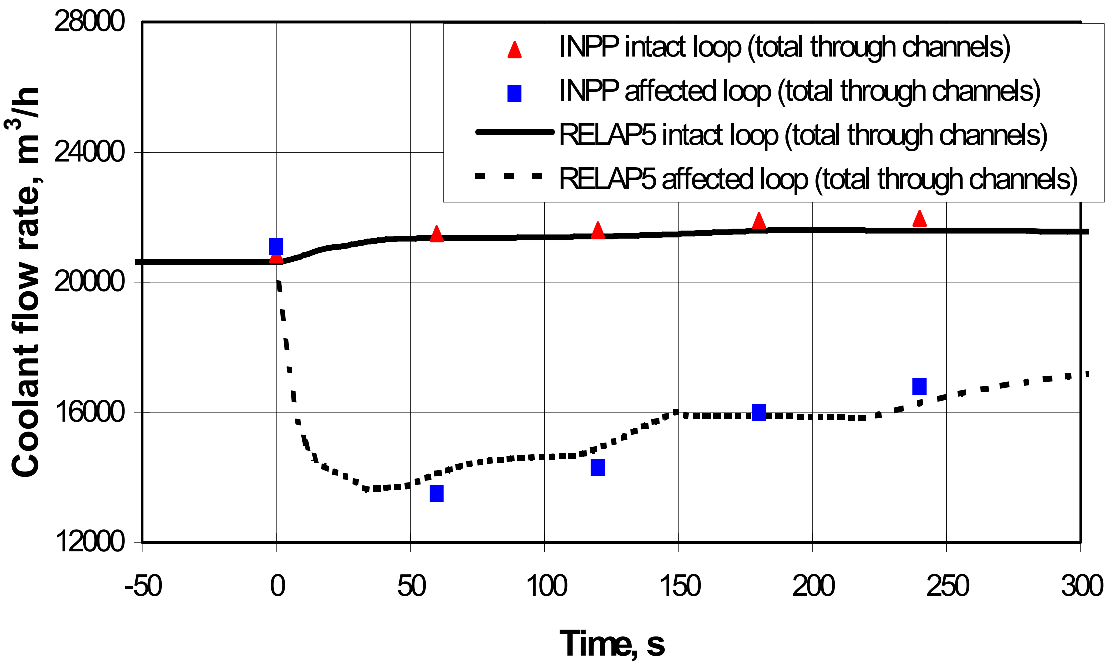


Fig. 15. Single MCP trip with failure of check valve. Coolant flow rate through the MCC loops.

As can be seen from Figure 16, in this case the margin to CHF during the all accident is greater than 1, i.e. reactor core is reliably cooled.

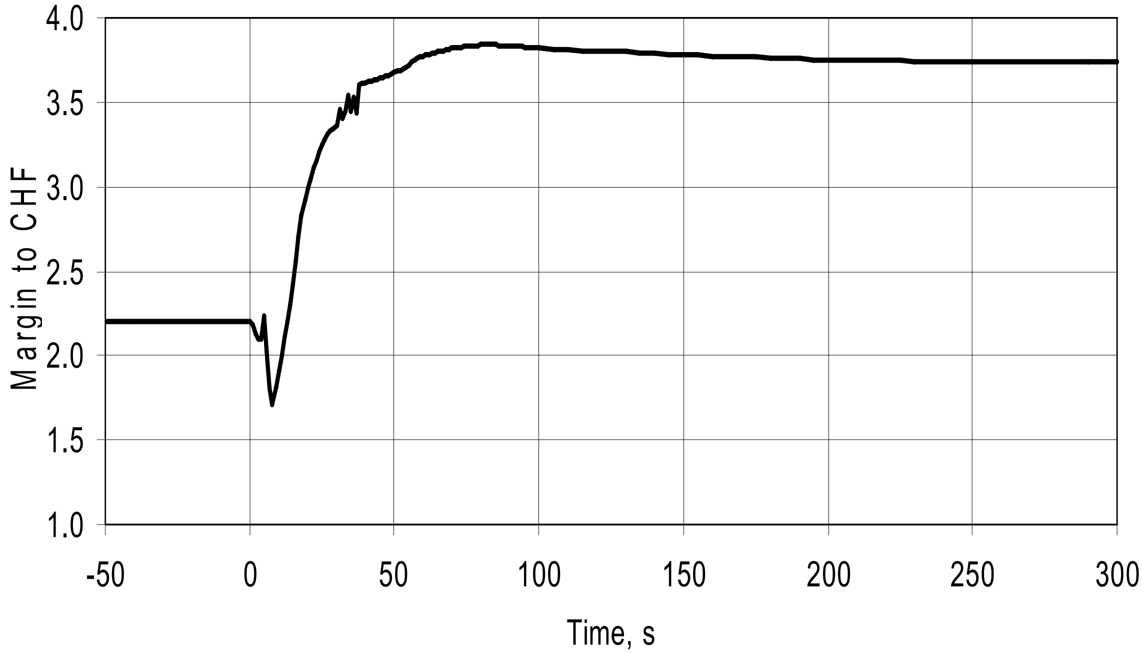
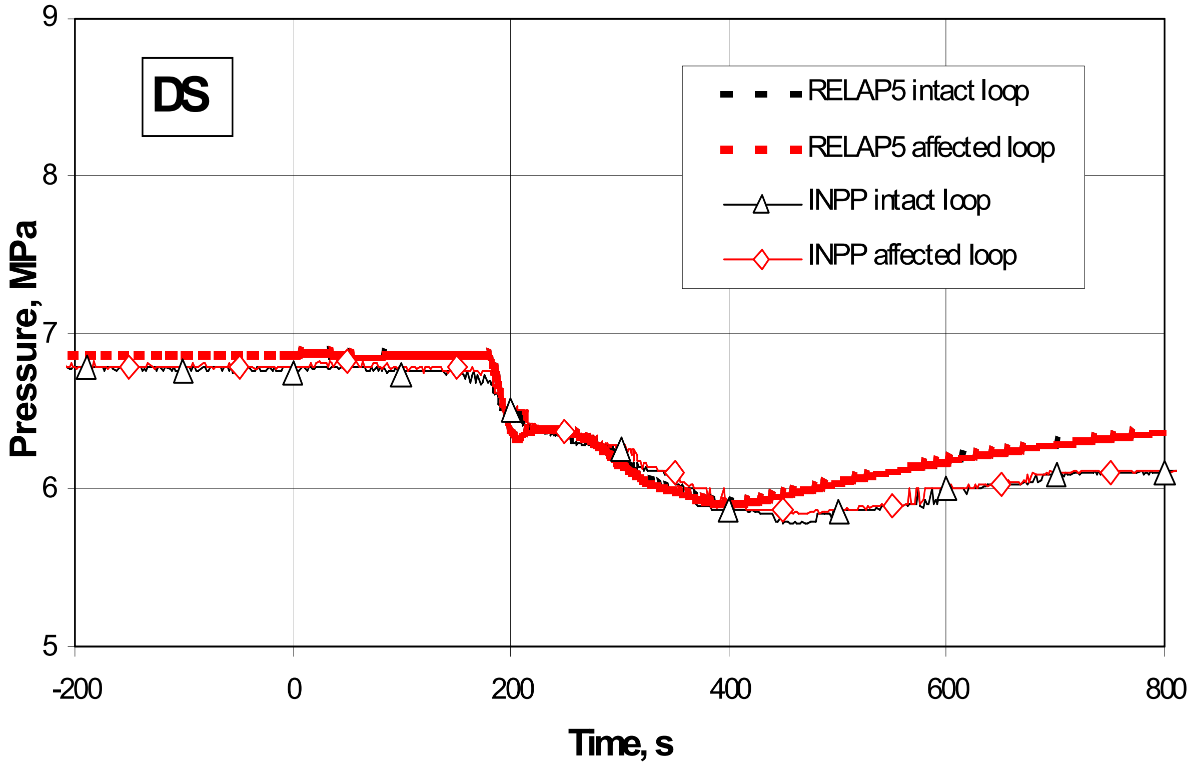


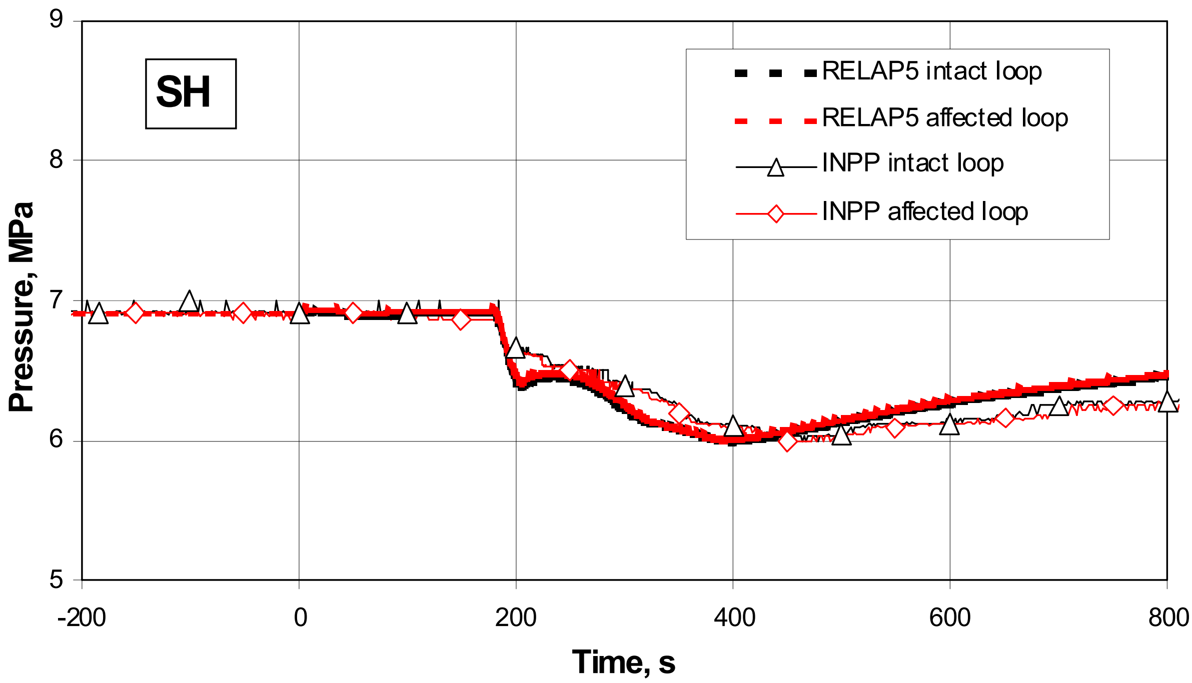
Fig. 16. Single MCP trip with failure of check valve. Margin to CHF.

3.2 Three MCP sequential trip event analysis

Three MCP sequential trip event occurred at Ignalina NPP on August 23, 2000. Before this event the reactor operated at 2300 MW_{th}. At 11:17 due short circuit into the control cable, fire-prevention signal of Ignalina NPP is activated by mistake. This caused that fire-prevention pump provided foam mixture into the MCP compartments of one MCC loop. Foam was found on the cabinets of MCP electric motors control. The short circuit protections were activated. At 11:23 first MCP was switched off. As the core power was less than 2860 MW_{th}, AZ-4 signal was not generated. Still after three minutes, the second MCP of the same MCC loop was switched-off. According to two MCPs trip in one loop of the MCC, AZ-1 signal was generated. According to this signal all CPS rods were inserted within 12 – 14 seconds and the reactor was shutdown. Approximately 20 seconds after AZ-1 initiation, the steam supply for turbine was suspended. At 11:29 the last operating MCP in affected loop of the MCC was switched off. In order to decrease flow rate differences in both loops, operators stopped one pump in intact loop of the MCC.

Analysis results are presented in Figures 17-20. The calculated and measured pressure in the DS, SH and PH agrees well (see Figure 17). Switching off of one MCP leads to the increase of coolant flow rate through other MCPs of the same loop (Figure 18). The second MCP of affected MCC loop was switched off after about 180 s. Output of the only operating pump increases up to 10500 m^3/h. Approximately 400 s after the first MCP was tripped, the last MCP is switched off. Calculations showed that after MCP trip, the coolant flow rate through it decreases smoothly due high inertia of flywheel, pump and motor rotors. In about 60 s after the last MCP trip, the coolant natural circulation in affected loop of the MCC starts. Unfortunately, due to measuring devices insensibility to low coolant flow rates at the Ignalina NPP, the coolant natural circulation was not identified. Coolant flow rate through MCP of intact loop is presented in Figure 19. In this case reactor core also is reliably cooled because the margin to CHF is greater than 1 (see Figure 20).





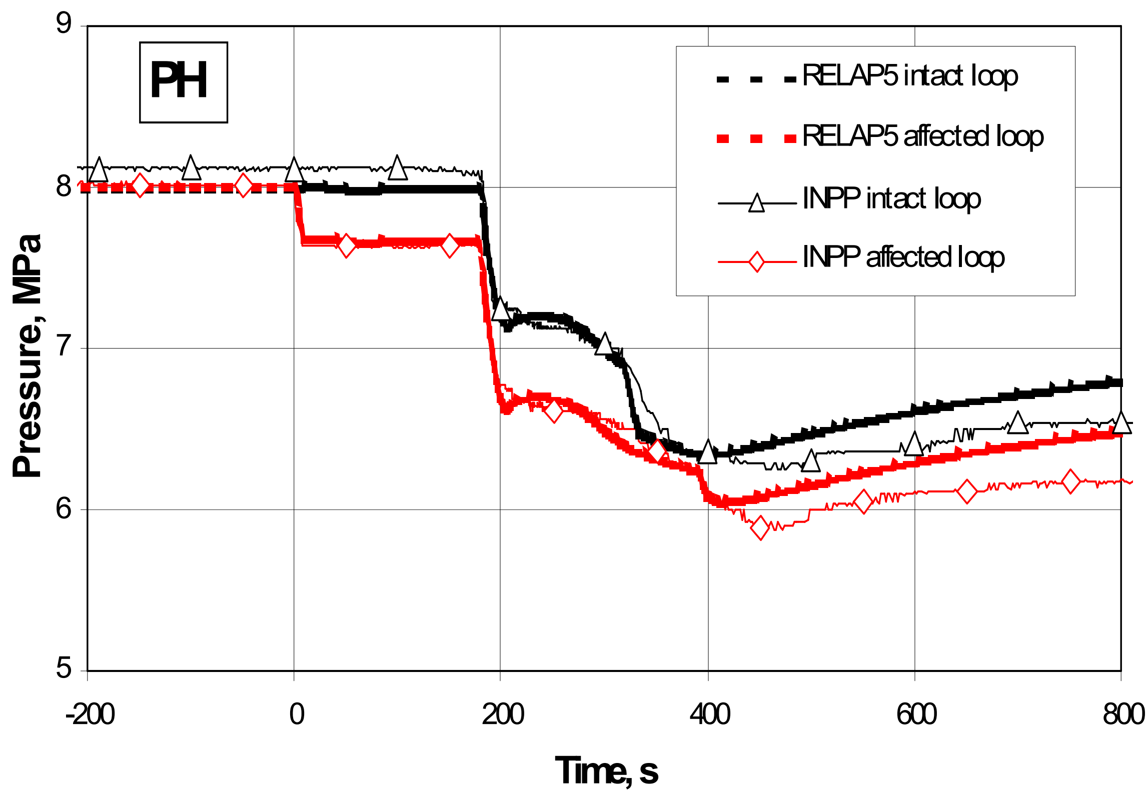


Fig. 17. Three MCP sequential trip. Pressure in the MCC.

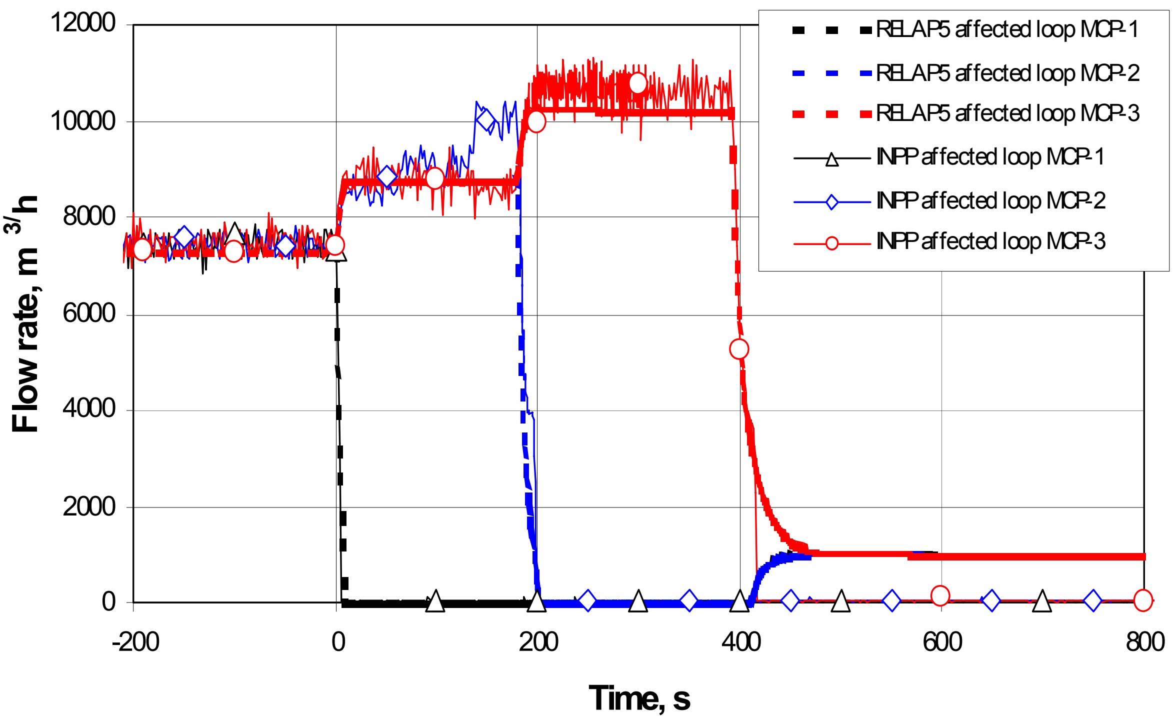


Fig. 18. Three MCP sequential trip. MCP throughput in the affected MCC loop.

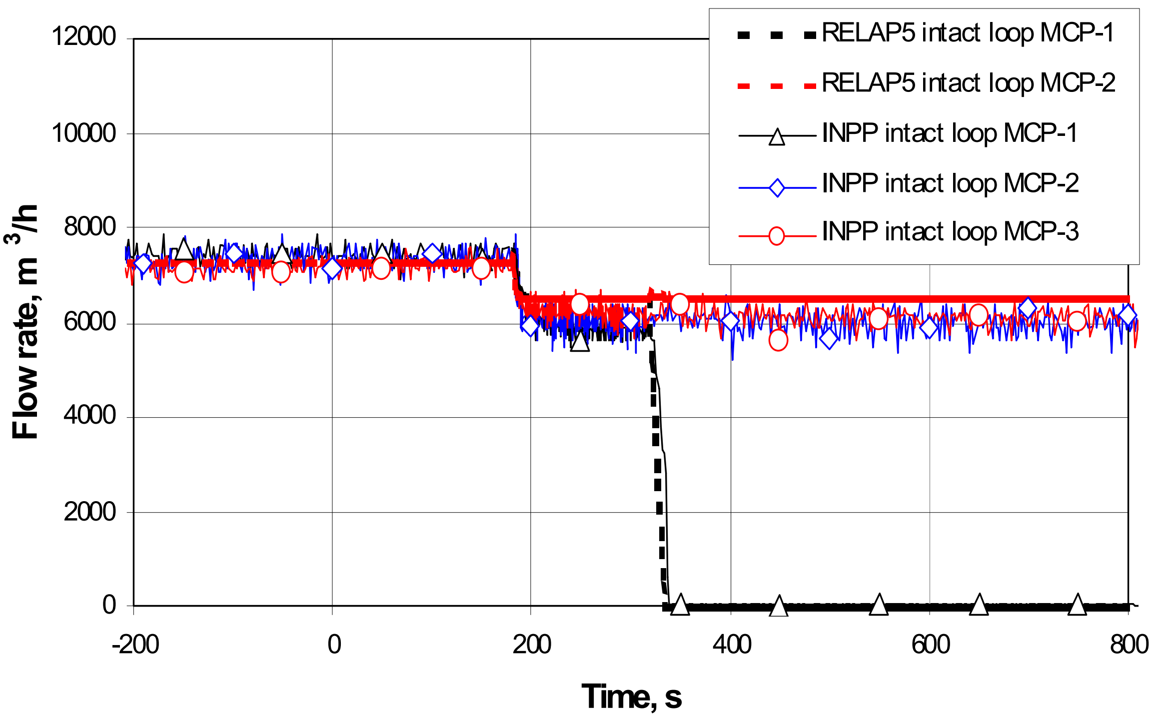


Fig. 19. Three MCP sequential trip. MCP throughput in the intact MCC loop.

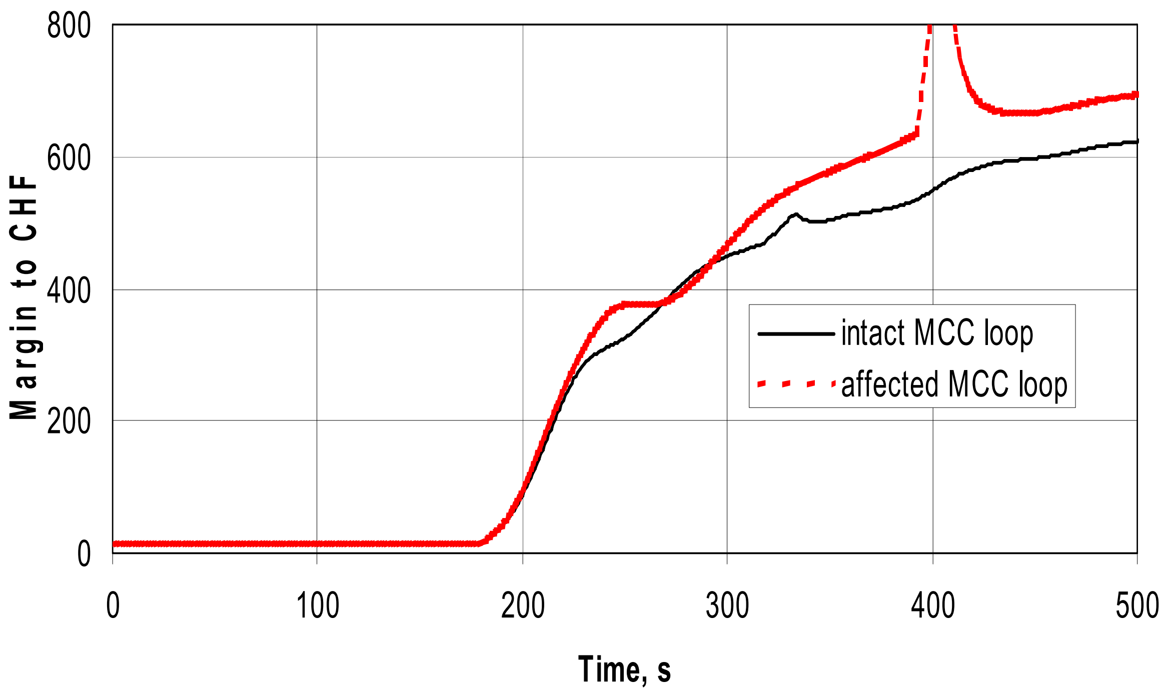


Fig. 20. Three MCP sequential trip. Margin to CHF.

4 CONCLUSIONS

For single tripped MCP and simultaneous trip of all MCPs cases uncertainty and sensitivity analysis was performed. The performed analysis shows that the largest impact on the calculation results has the selection of the homogeneous media model in coolant natural circulation case. Because homogeneous media model is too conservative and could be too big source of uncertainties, during accident analyses the use only of the non-homogeneous model is proposed. Since in case of a single tripped MCP the largest impact on the calculation results uncertainties was the flow energy loss in ICV, it is recommended in cases of forced circulation to model the reactor core with more details, i.e. the core must be represented by possibly more channel groups.

Performed benchmark analysis of MCP trip events showed that the reactor core is reliably cooled even taking into account possible sources of uncertainties.

Subscripts

th thermal

Nomenclature

AZ-1 emergency protection

AZ-4 emergency protection

CHF critical heat flux

DS drum separator

FC fuel channel

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 “Channelled Large Power Reactor”

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INDEPENDENT DETERMINISTIC ANALYSIS OF THE OPERATIONAL EVENT WITH TURBINE VALVE CLOSURE AND ONE ATMOSPHERIC DUMP VALVE STUCK OPEN

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Abstract. The paper presents the results of the independent analysis of the operational event which took place on 07.11.2003 at Unit 1 of Rostov NPP. The event started with switching off the electrical generator of the turbine due to a short cut at the local switching substation. The turbine isolating valves closed to prevent damage of the turbine. The condenser dump valves (BRU-K) and the atmospheric dump valves (BRU-A) opened to release the vapour generated in the steam generators. After the pressure decrease in the steam generators BRU-K and BRU-A closed but one valve stuck opened. The emergency core cooling system was activated automatically. The main circulation pump of the loop corresponding to the steam generator with the stuck BRU-A was tripped. The stuck valve was closed by the operational stuff manually. No safety limits were violated. The analysis of the event was carried out using ATHLET code. A reasonable agreement was achieved between the calculated and measured values.

1 INTRODUCTION

The analysis of an operational event includes several stages. The first one is always a qualitative evaluating of the event. The second step is choosing the instrument for the calculation. The third stage is data handling. The forth one is preparing calculation. And the last stage is summarizing the results and making conclusions. The following chapters of the paper discuss each of these steps by the example of the event taking place at Rostov NPP on 7 November 2003. Of course, these stages are quite relative and doing the analysis we many times got back to some of the previous steps, especially to the first one.

2 QUALITATIVE EVALUATING OF THE EVENT

At the stage of qualitative evaluation we try to understand the consequences of the events. The aim is to determine the peculiarities of the regime and foreseen possible difficulties for the stage of calculation.

At the first stage of the analysis we had a short report on the event, prepared by the plant and several plots, shown below.

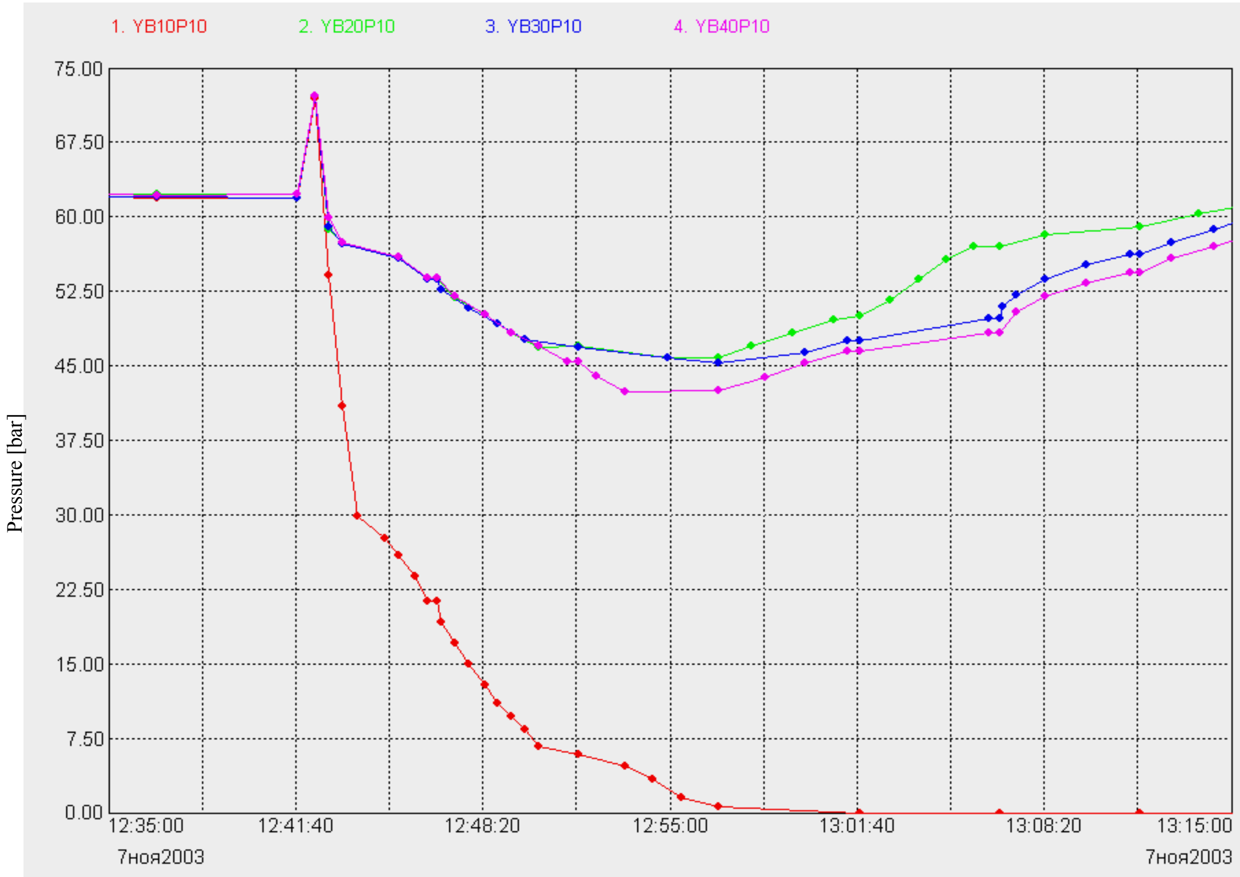

Fig. 1. Preesure in the steam generators.

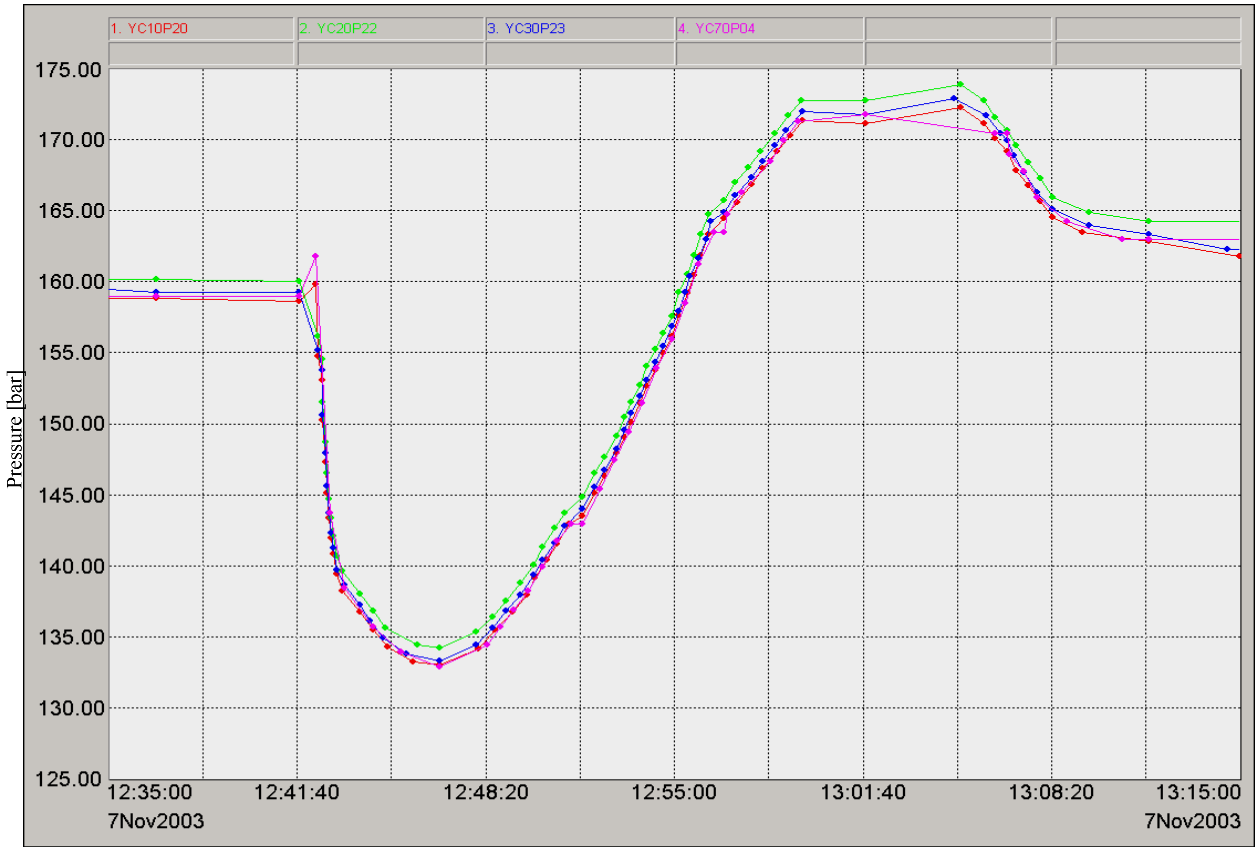

Fig. 2. Pressure in the upper plenum.

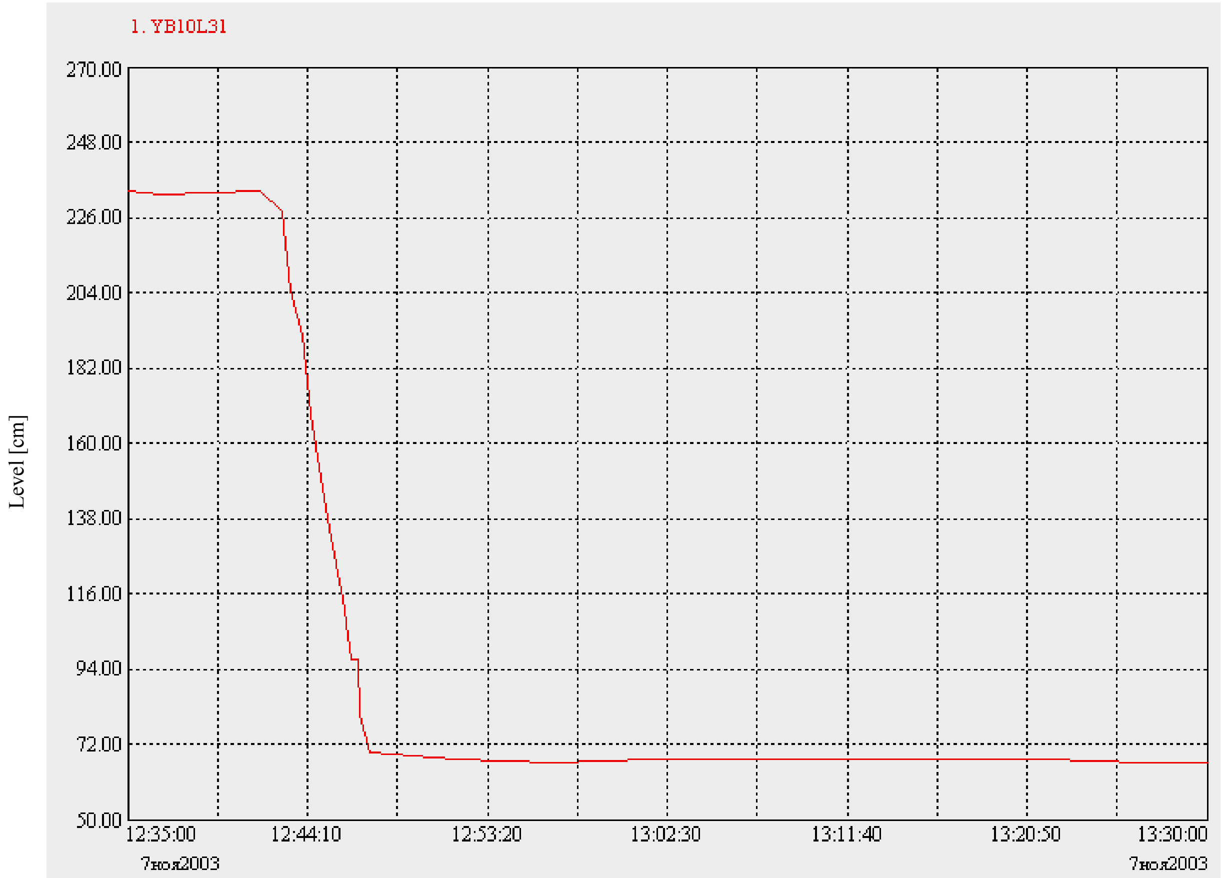

Fig. 3. Level in the affected steam generator.

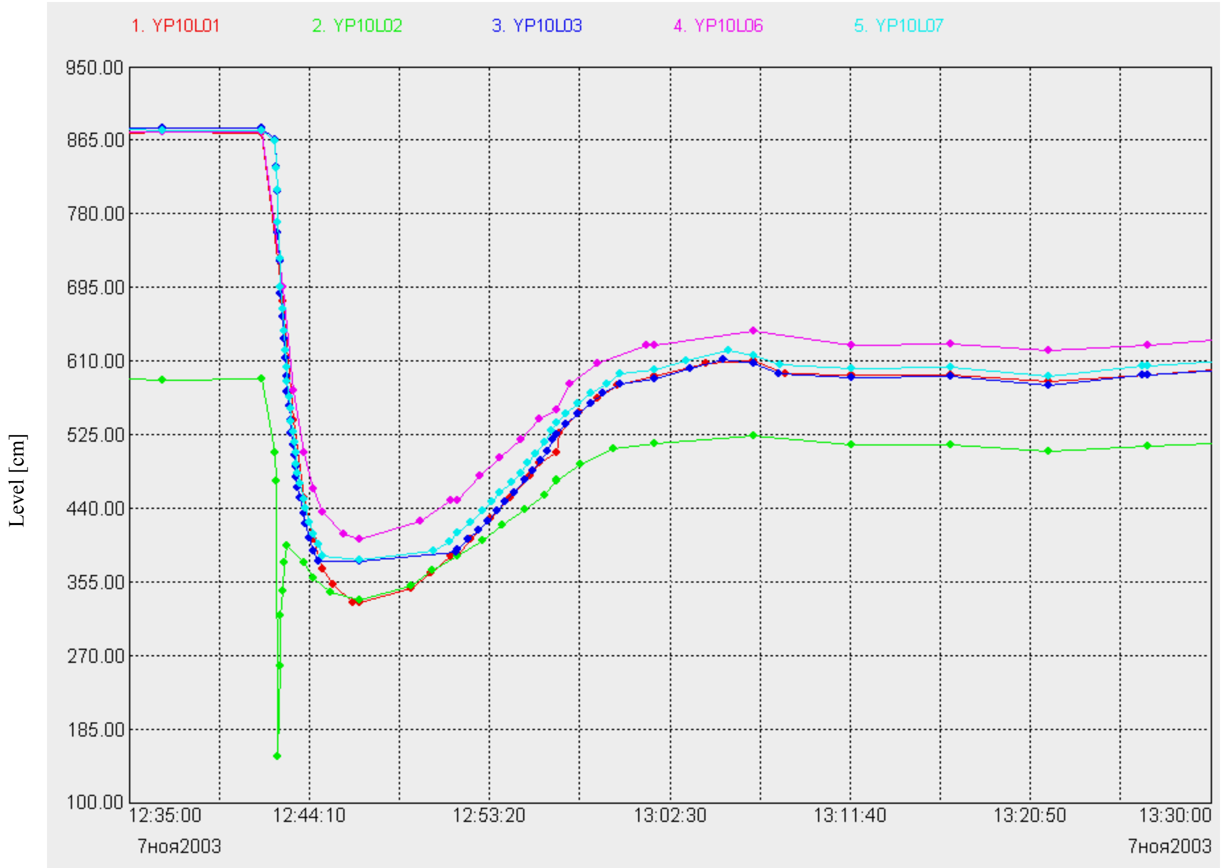

Fig. 4 Level in the pressurizer.

The event started with switching off the electrical generator of the turbine due to a short cut at the local switching substation. The turbine isolating valves closed to prevent damage of the turbine. The condenser dump valves (BRU-K) and the atmospheric dump valves (BRU-A) opened to release the vapor generated in the steam generators. After the pressure decrease in the steam generators BRU-A and BRU-K closed but one BRU-A valve stuck opened. The main circulation pump of the loop corresponding to the affected steam generator was tripped and the leaking steam generator was isolated. Later the rest of the steam generators were isolated too. The stuck valve was closed by the operational stuff manually. After that the drainage line of SG1 was connected to the drainage lines of the other steam generators. Thanks to this possibility it was filled with hot water and strong temperature gradients were avoided.

The following time table presents the scenario of the event.

Table 1. Scenario of the event

Time	Seconds from the beginning of the accident	Events, signals, operator's actions
07.11.2003		
12:41:48	-7	Shortcut at the local substation. Drop of the voltage in the grid.
12:41:55	0	Activation of the electrical protections of the generator. Closure of the turbine isolating valves.
12:41:57	2	P_{MSC}>6.96MPa. The condenser dump valves (BRU-K) open
12:42:04	9	P_{SG}>7.07MPa. The atmospheric dump valves (BRU-A) open
12:42:18	23	P_{SG}<6.28MPa. The atmospheric dump valves (BRU-A) close except for that at SG1.
12:42:19	24	The operator activates EP-I. Reactor scram.
12:42:41	46	Signal "$\Delta P_{CHV\ SG1}$<-0.196MPa & P_{SG}<5.01MPa". Trip of MCP-1.
12:42:45	50	Signal "P_{SG1}<4.9MPa"
12:42:48	53	Signal "Leak in the secondary side ($\Delta T_{S1\text{-}2}$>75°C & P_{SG1}<4.9MPa)"
12:42:50	55	EP-I signal
12:42:51	56	Isolation of SG1.
12:43:03	68	P_{MSC}<5.67MPa. The condenser dump valves (BRU-K) close.
12:49÷13:01	425	Signals "Leak in the secondary side ($\Delta T_{S1\text{-}2}$>75°C & P_{SG}<4.9MPa)" for SG2, SG3, SG4.
16:40		The stuck valve is closed manually at the site.
17:00		The boron concentration in the primary circuit is 16 g/kg.
22:00		Filling SG1 through the drainage valve started
08.11.2003		
15:40		Filling of SG1 through the drainage valve finished. The level in SG1 is 3.8 m.
16:10		Water from the deaerator supplied.
18:00		The level in SG1 is 4 m.

The event was classified as of level 1 according to INES.

As it could be seen in Figure 1, the pressure in the not-affected steam generators continued decreasing even after closing atmospheric and condenser dump valves (the set point for closing BRU-K is 5.67MPa). It was concluded that the reason could be an opening of the dump valves to the all-purpose vapor collector (BRU-SN). BRU-SN is a controller with quite a complicated algorithm of work that maintains the pressure in the all-purpose vapor collector (KSN). It is located at the main steam collector.

3 CHOOSING THE INSTRUMENT FOR THE CALCULATION

This step is closely connected with the previous one and highly depends on the results of it.

In our department we usually use ATHLET code for thermo-hydraulic calculations. It seemed to be a capable instrument for calculating this scenario.

But the instrument for the calculation is not only the code itself. The input deck is also very important. Because the code could be able to model the phenomena taking place, but the modeling of systems and controllers in the input deck may not be sufficient.

In our case the input deck represents the steam lines from the steam generators to the turbine. Turbine itself with the extractions and all-purpose collector are not modeled. The mass flow rate through the valves dumping to the all-purpose vapor collector could be modeled only as a boundary condition. So, we went further being conscious that problems with BRU-SN are to be expected.

4 DATA HANDLING

Data handling is a rather time consuming step. It includes: digitizing plots, searching through a tables and transferring printed values to files, reading handwritten evidence reports of the operators. It is very important to digitize plots very precisely. The program GetData (getdata.com.ru) could be recommended.

It should be mentioned that among the grate amount of values, which are measured, only a small percent is really useful for the calculation. From the other hand, the parameters, dear to everybody who prepares thermo-hydraulic calculations are not available. Values like decay heat, heat transferred in the steam generators, mass flow rates through the valves are obviously not measured at all.

5 PREPARING CALCULATION

This is the longest but most interesting phase of the work. It is a cycle, which includes:

- doing runs with the code;
- analyzing the difference between the calculated and measured values;
- improving the input deck or fitting some parameters.

In the course of the above mentioned cycle the set points for some signals were defined more exactly. Also the laws for turbine isolating valve closure and atmospheric valve opening were varied. Another very important parameter which was tuned is the resistance of the spray line. As it could be seen in Figure 2 the pressure rise in the primary circuit was suppressed by the spray from the cold leg. The spray took place even though all main circulation pumps were tripped.

The results of the calculation after fitting mentioned parameters are shown in Figure 5 ÷ Figure 9.

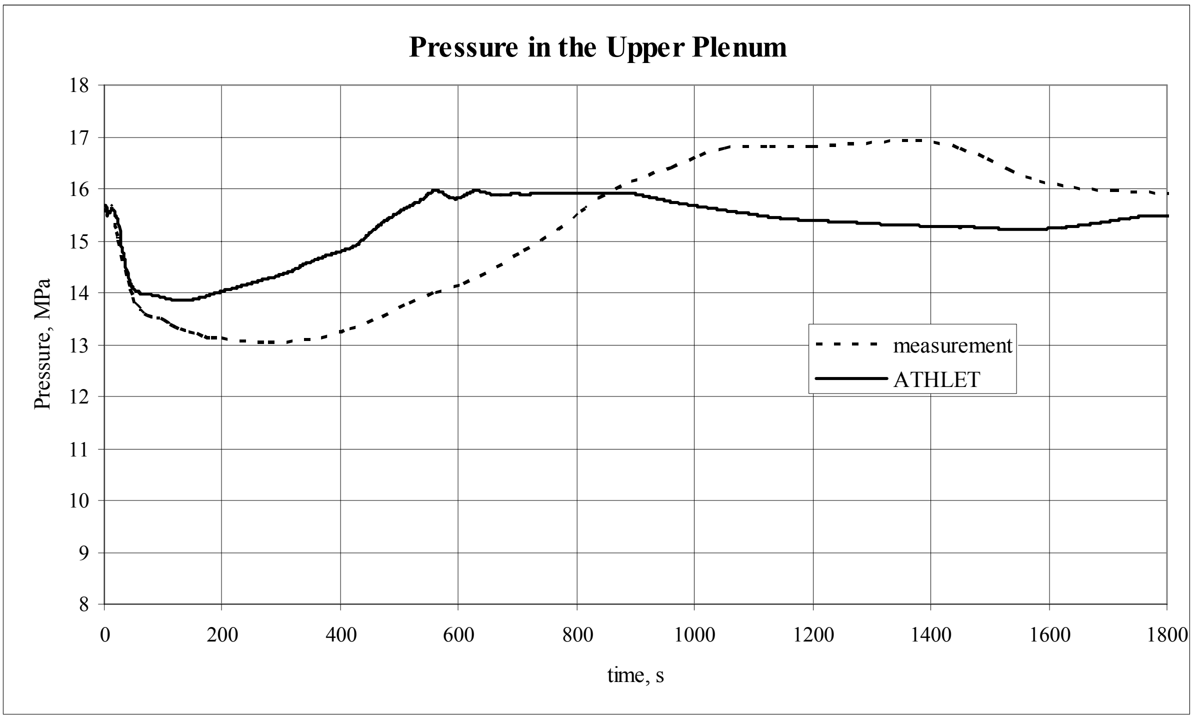

Fig. 5. Pressure in the upper plenum.

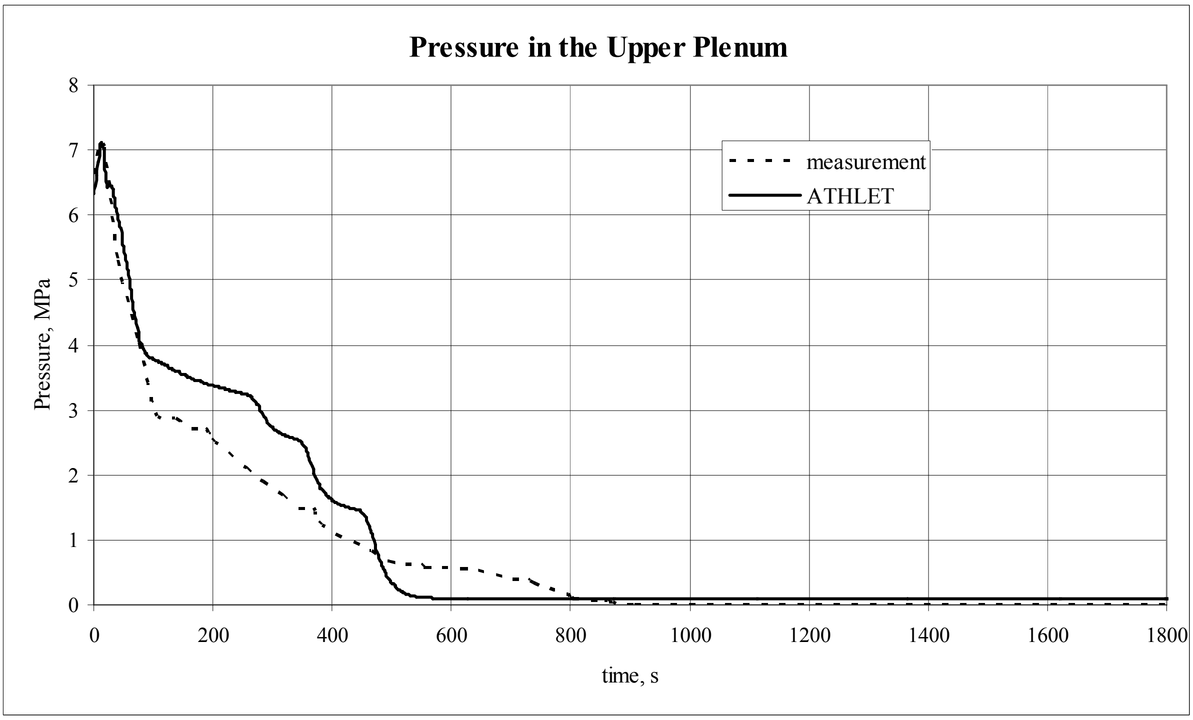

Fig. 6. Pressure in the affected steam generator.

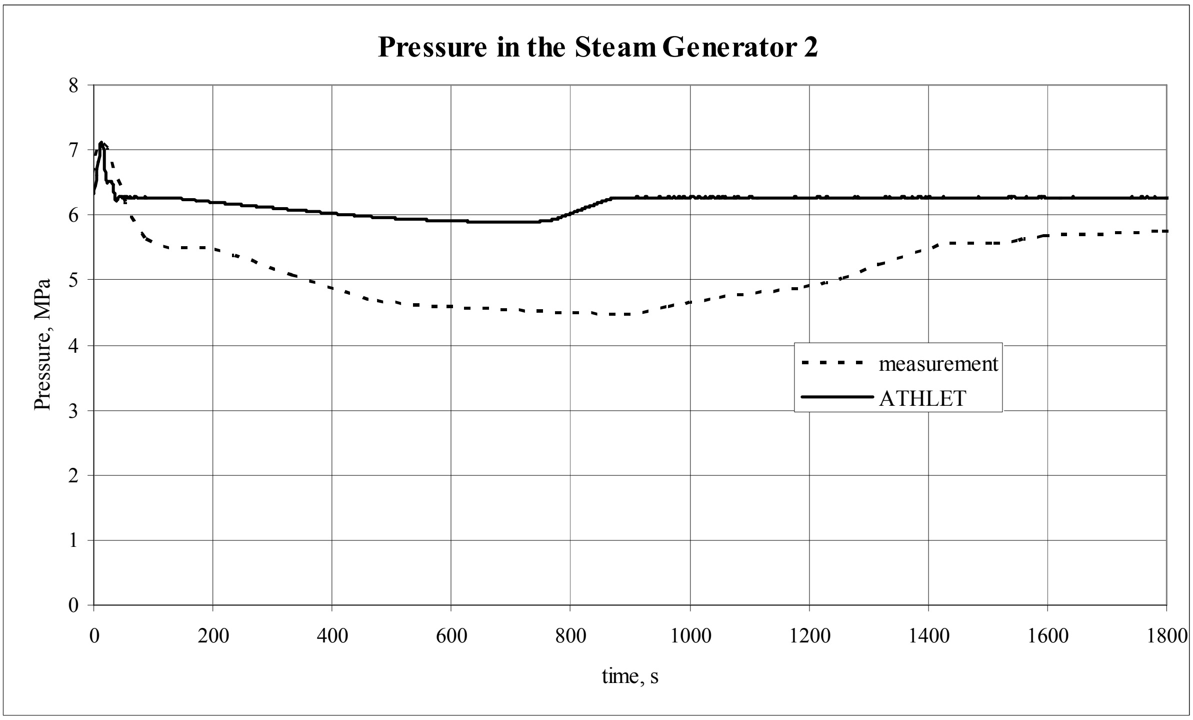

Fig. 7. Pressure in the not-affected steam generators.

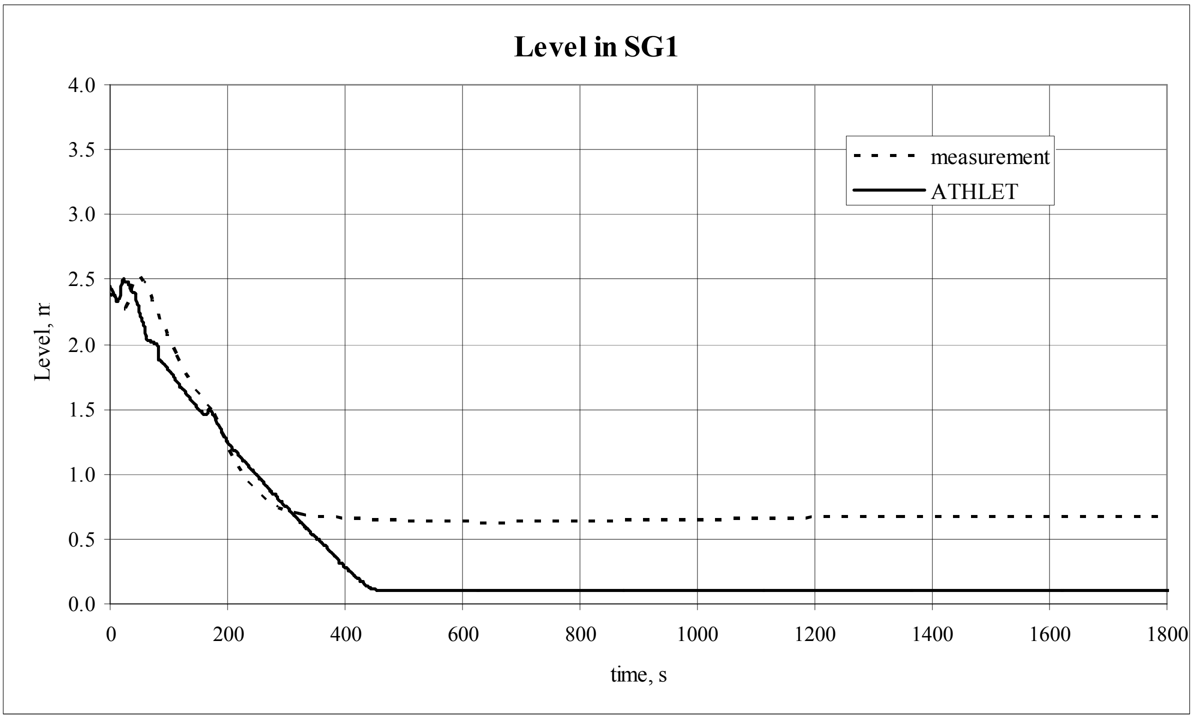

Fig. 8. Level in affected steam generator.

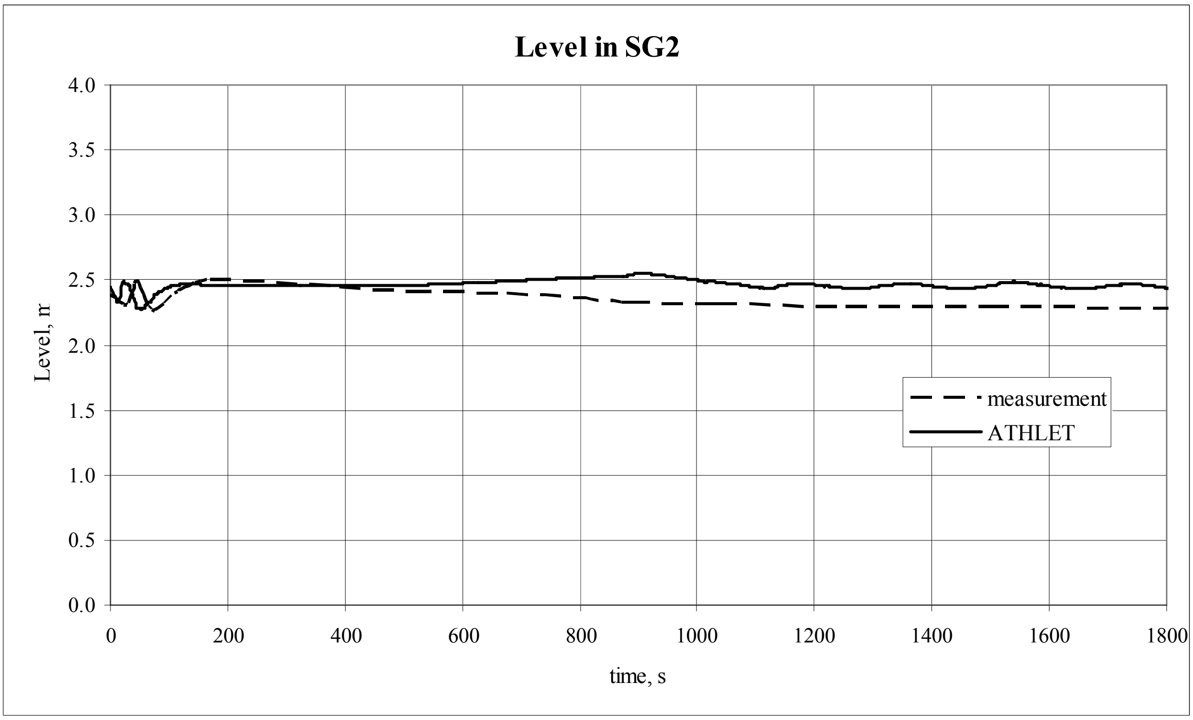

Fig. 9. Level in the not-affected steam generators.

At this point it became obvious that the forecast, made at the stage of qualitative analysis was absolutely correct. The behavior of the dump valves to the all-purpose vapor collector played a significant role in the course of the event. So it was decided to model it. As it was mentioned above, the only possibility was to present it as a boundary condition varying the mass flow rate.

The results of the calculation are shown in Figure 10 and Figure 11. It could be seen that they are much better than in the calculation without modeling BRU-SN.

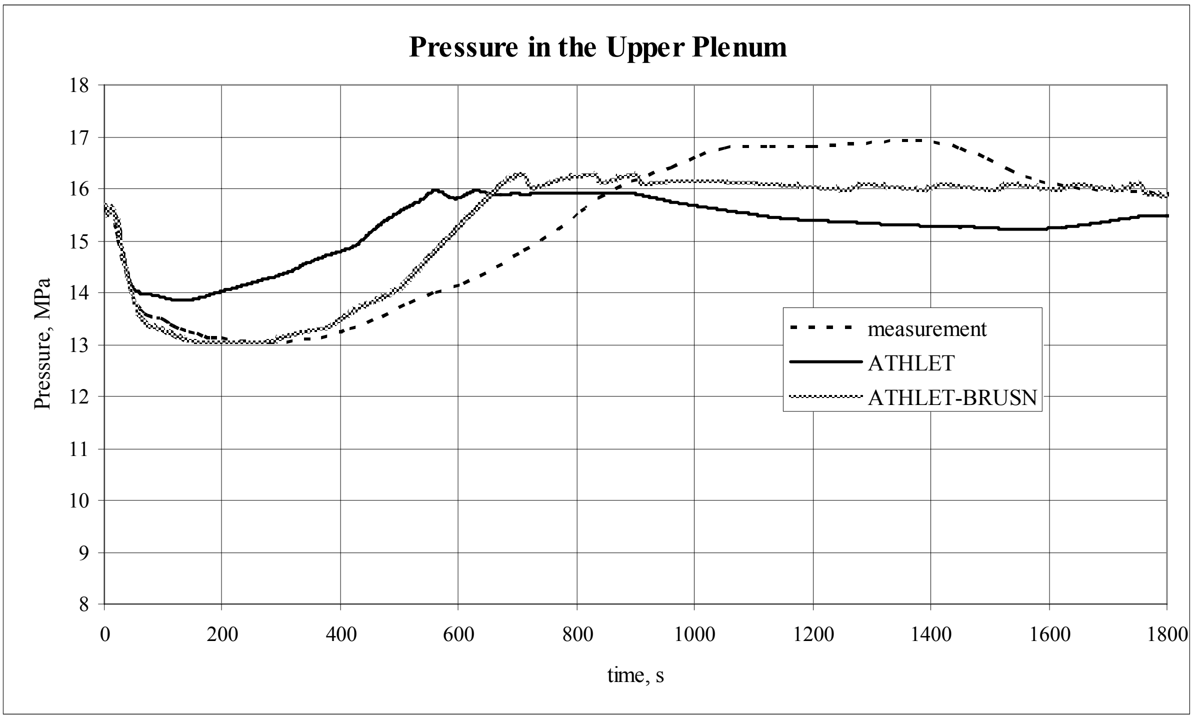

Fig. 10. Pressure in the upper plenum.

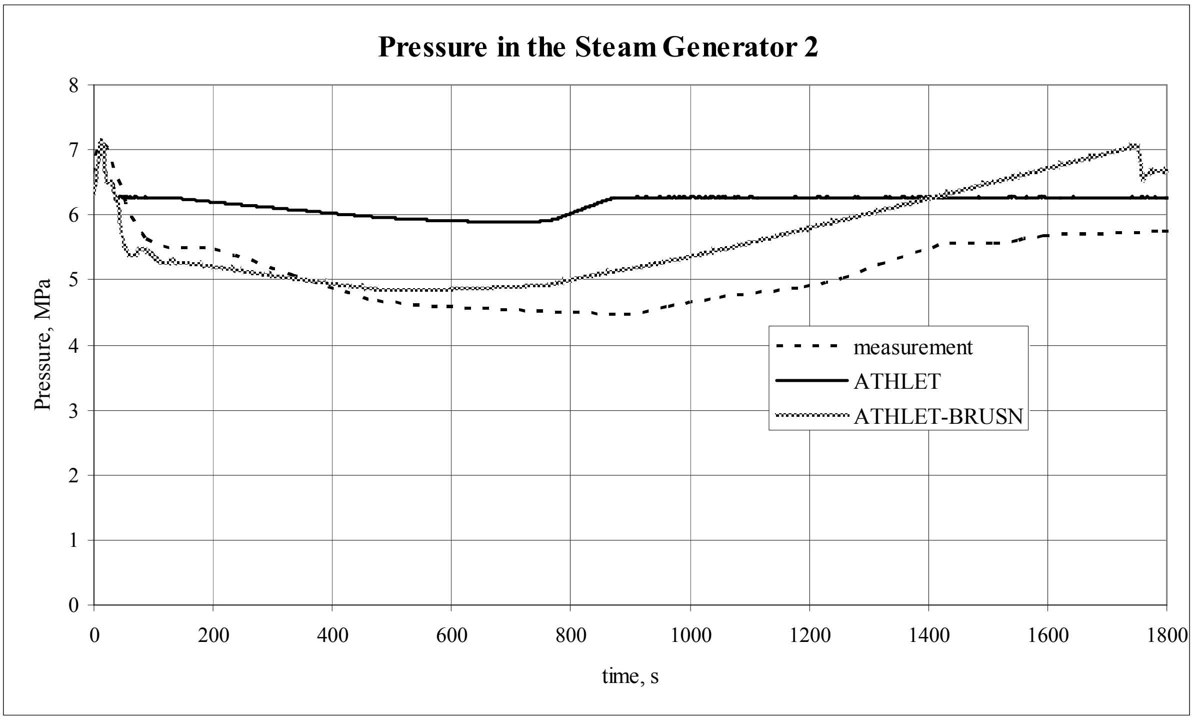

Fig. 11. Pressure in the not-affected steam generators.

Never the less, it should be underlined that fitting the mass flow rate through BRU-SN is quite a different kind of fitting, compared with fitting the resistance of the spray line. The characteristics of the spray line are constant and we used the measured data to get closer to the reality with our model. And what we achieved as a result for this line in this calculation will be used in future calculations. As for the fitted mass flow rate through BRU-SN, it can't be used in other calculations. At least, it can't be used directly. Or, saying it in other wards, no experience is gained by fitting this mass flow rate.

6 SUMMARIZING THE RESULTS

At this final part of the work we summarize the following:

- The input deck was improved, and thanks to that;
- A reasonable agreement was achieved between the measured and calculated data;
- The scope of modeling systems of the secondary side is not enough for precise modeling of the event. Detailed presentation of the turbine part of the scheme is desirable.

In connection with the last conclusion it was decided to calculate the same event using ATHLET/CMS/ATLAS analytical simulator, where the turbine and condense circuit are thoroughly modeled.

7 CONCLUSIONS

The deterministic analysis of an operational event is an extremely useful work. It is important for validation of the instrument used for calculations. In some cases it could serve even for improving the code itself and it is always helpful for validation of the input deck.

The process of analysis is very useful also for improving the skills of the user in understanding different phenomena.

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OVERVIEW OF RECENT DETERMINISTIC THERMOHYDARAULIC ANALYSES OF OPERATIONAL EVENTS FOR SLOVAK NPPS

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Abstract. Based on review of cooperation of the VUJE, Inc. with Slovak NPP operator within the last three years, the paper describes selected operational occurrences, which were required to be analyzed from thermohydraulic point of view. For each event a short problem description is given, followed by information of analytical methodology and approach, as well as overview of the most important findings. The events, described in detail include problems with mechanical fatigue of the pressurizer surge line due to the thermal stratification in the upper part of the line (CFD simulation using FLUENT code), leakage from the primary circuit due to the thermal stress and fatigue of the ECCS pipelines (RELAP5 simulation of the ECCS line section with two back valves) and low-level leakage through the reactor flange sealing (RELAP5 simulation together with activity analysis).

1 INTRODUCTION

VUJE Inc. is the follow-up organization of the research institute created in the seventies to support construction and operation of Slovak nuclear power plants. Through the last thirty years VUJE's main role has been to support Slovak NPP operator providing all special services, including also the whole scope of theoretical and computational analyses. Besides the licensing type analyses, which are the most important services provided, also analyses of real operational occurrences are from time to time required. Such analyses are not very frequent due to the large experience gathered during the years of operation and all measures taken by operator to prevent malfunctions of systems and equipment.

The Slovak units were, from many different reasons, subjects of relatively very deep and detailed safety assessment in the past years. If a safety-important case occurs it is included in standard safety assessment of the unit and is analyzed as a generalized scenario. Therefore, deterministic analyses of real operational cases are typically not directly related to safety important occurrences.

To illustrate typical problems, which were required to be deterministically analyzed, four cases were selected. These cases are examples of originally unexpected operational problems, which challenge the analytical team with untypical scenarios and phenomena and which show problems in fitting analytical models to mostly insufficient description of initial and boundary conditions.

The paper content is limited to deterministic, thermohydraulic analyses, using the RELAP5 and FLUENT codes, although the results were used e.g. by material fatigue analyses or to analyses of control system behavior.

2 SELECTED OPERATIONAL CASES ANALYZED

2.1 Mechanical fatigue of the pressurizer surge line, induced by thermal stratification

2.1.1 Problem description

The problem with excessive mechanical fatigue of the pressurizer surge line was identified by extended fatigue monitoring of selected pipelines of the Bohunice VVER440/V213 units.

Along the surge line, additional thermocouples are installed, grouped in six sections, each containing five sensors along the perimeter of the line. The frequency of reading the data is about one minute. The measured temperatures are recorded and in off-line mode processed to evaluate mechanical fatigue of the pipeline material.

Integral analysis of the fatigue of the upper elbows of the surge line has shown that the fatigue of the material reached such a status, that these parts need to be replaced. The reason was in frequent temperature fluctuations, which had, especially at rector start up, amplitudes of over 100 °C, together with thermal stratification of the coolant (see example in Fig. 1).

In spite of the fact that the elbows have been replaced it was required to understand the reason of the problem, especially why the fluctuations and stratification is a typical status at one unit and not at both, and what countermeasures can by taken to prevent such a problem.

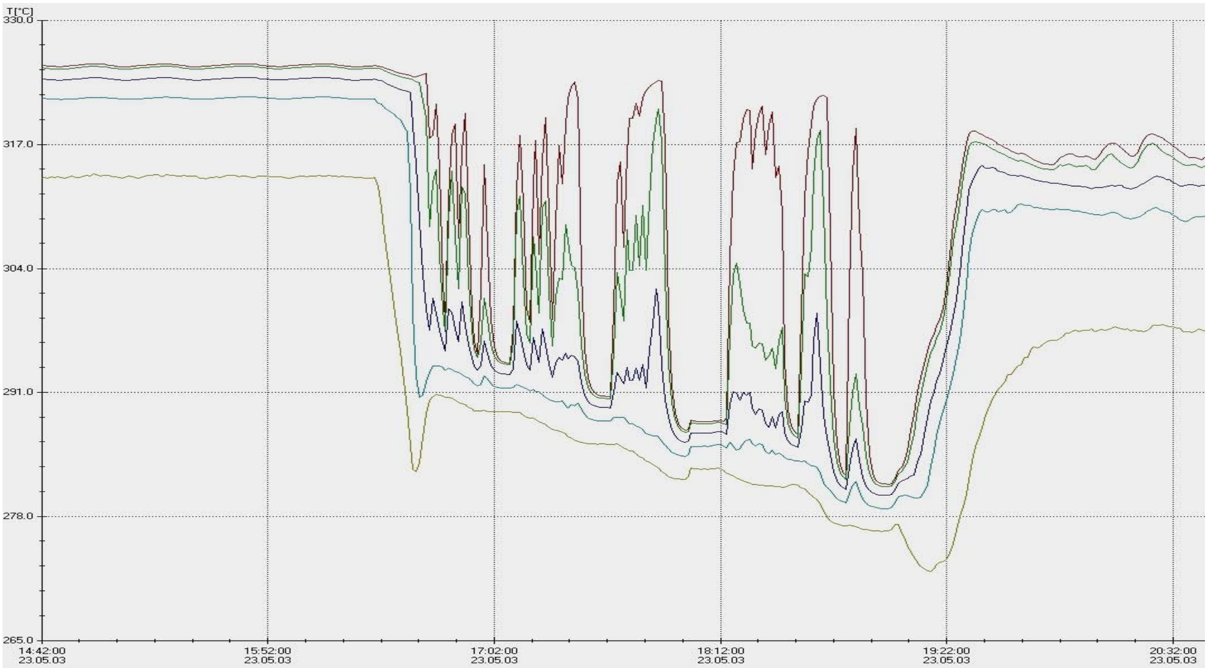


Fig. 1. Temperature of the pipeline surface along selected perimeter, located at upper horizontal part.

2.1.2 Selection of tools and methodology

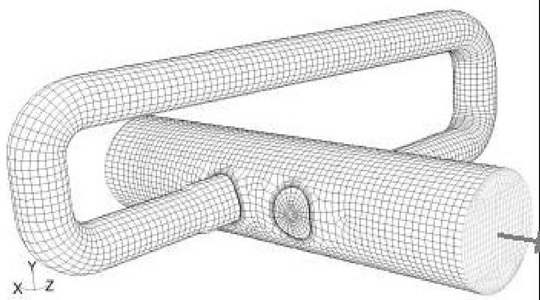
From the description of the problem and from initial analysis of the available data it was obvious that the governing phenomenon was thermal stratification along the upper horizontal section of the surge line. The only well described boundary condition for analysis is the temperature record for all measured points, joined with normal operational measurements of indirectly related parameters as level in pressurizer, temperature and pressure in primary etc. There is no information about coolant flow rate through the surge line or from/to pressurizer.

Flow through the surge line is caused and maintained by pressure difference at the two connections of the surge line to the primary loop, separated along the flow direction by about 30 cm. By flow velocities in order of 10 m.s^{-1}, the local effects could play decisive role. To analyze effects of possible asymmetries of the connection section the computational fluid dynamics (CFD) code FLUENT was selected. Simplified model of the section was developed

(Figure 2) and extended sensitivity study was done varying deviations from ideal symmetry, including weld reaching into the flow of primary coolant in primary loop.

To analyze thermal stratification, interference of outflow from pressurizer and flow through the surge lines, model of the complete surge line was developed (Fig. 3), with pressure difference as boundary conditions at the lower ends at primary loop and flow velocity at the pressurizer outlet, both coupled with coolant temperatures.

Using the surge line model for the FLUENT code, in initial state the model was fitted to the measured data for several selected cases. The main reason was to reach acceptable validity of the model, especially in removing uncertainty in flow rate through the line and from pressurizer. Then, different modes were analyzed and impact to frequency and amplitude of the temperature fluctuations was evaluated.



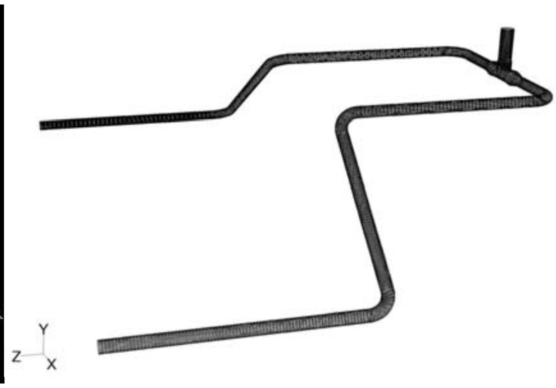


Fig. 2. Simplified problem - connection section. *Fig. 3. Surge line configuration and model.*

2.1.3 Results of analysis and outputs

Thermal stratification at the upper horizontal line was clear from the measurement. Taking into account all available data and indirect evidence, it was identified that the basic reason for stratification is cyclic in/out flow from pressurizer. This flow is caused by pressurizer level control system, which cycles with level around the mean value. Flow from pressurizer is low and almost does not mix with primary coolant, flowing at speed below $0.2 m.s^{-1}$ through the surge line from one primary nozzle to the other.

CFD simulation of the connection part of the surge line to primary loop reveals that the flow through the surge line is very low in symmetrical configuration of the nozzles and that relatively small deviations can lead to significant changes in this flow. Mass flow rate through the line is therefore specific to the individual unit. This explains different behavior of the seemingly identical surge lines at the unit 3 and unit 4 of the Bohunice NPP.

CFD simulation of the surge line in the cyclic flow mode confirms stratification in the level observed by the measurements.

Several measures were suggested to prevent the stratification in cyclic mode, ranging from unit heat-up with stopped circulation pump at the first loop, full opening the pressurizer spray valves to modification of the system and algorithms of the pressurizer level control.

Most limitations to CFD application to the problem analysis follow from possible deviations of boundary conditions (including geometry), not known in sufficient detail for time and location of interest. But — having the model well adjusted to the sufficient set of available operational data — very useful prediction of local phenomena are the main results.

2.2 Rapid long-term cooldown of isolated primary loop, followed by problems with its heating-up by slightly reopening the main isolation valve

2.2.1 Problem description

Due to malfunction of the main circulation pump (MCP) electric protection signal, operator decreased reactor power to 74% as required by Technical Specifications. The cold leg main loop isolation valve (MLIV) was closed and MCP was tripped down. Coolant temperature of cold leg started decreasing in an excessive rate. In operator attempt, in order to heat-up the inactive loop, cold leg MLIV was manually slightly opened (about 40 revolution of "hand wheel"). In spite of expectations of the operator, the cold leg coolant temperature does not increase after time, expected to be sufficient for establishing some circulation through the loop, but even decreased faster. In another attempt operator opened MCP venting. This led to an increase of the temperature at measured points but this is not the searched solution of the problem. The loop was heated up only after several hours of cool-down by about twice more manual opening of the MLIV than in the first attempt.

Although no limits were overstepped and no safety event occurred, the not expected behavior of the unit was reason for detailed analytical study. The reasons of the excessive cool down rate of the loop were searched. Additionally the physical phenomena and technological reasons, resulting in coolant temperature decrease after slight opening of cold leg MLIV needed to be identified and proposals of countermeasures taken to prevent the problem in the future had to be proposed.

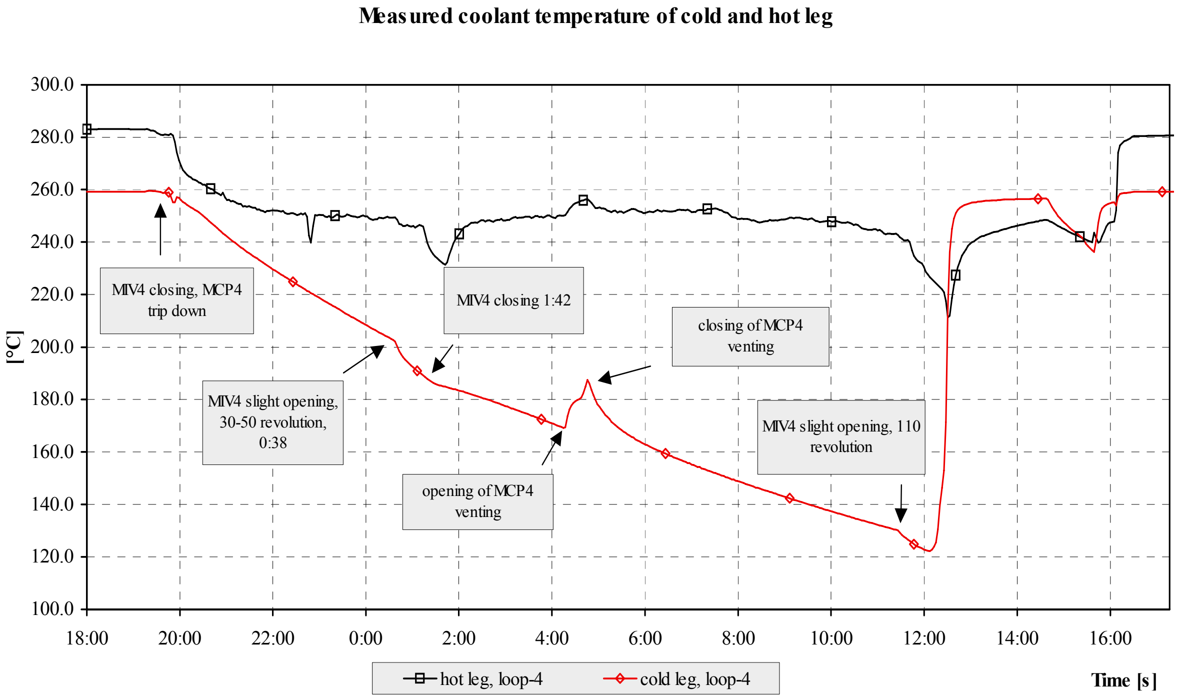


Fig. 4. Coolant temperature evolution during the problem and actions taken by operator.

2.2.2 *Selection of tools and methodology*

Initially the problem seemed to be a candidate for RELAP5 application. Governing phenomenon should be heat losses from coolant to pipeline wall, to MCP body material and MCP cooling circuit. Influence of steam generator should be taken into account. During the preparation, testing and validation of the RELAP model against available data, it was revealed that the effects of thermal stratification, localization of the measuring thermocouples and natural circulation inside the cold leg play the decisive role. As such phenomena cannot be simulated by the RELAP5 code, the FLUENT code was applied. For both codes specific detailed models were developed and validated against available operational data. The models are shown in the Figures 5 and 6.

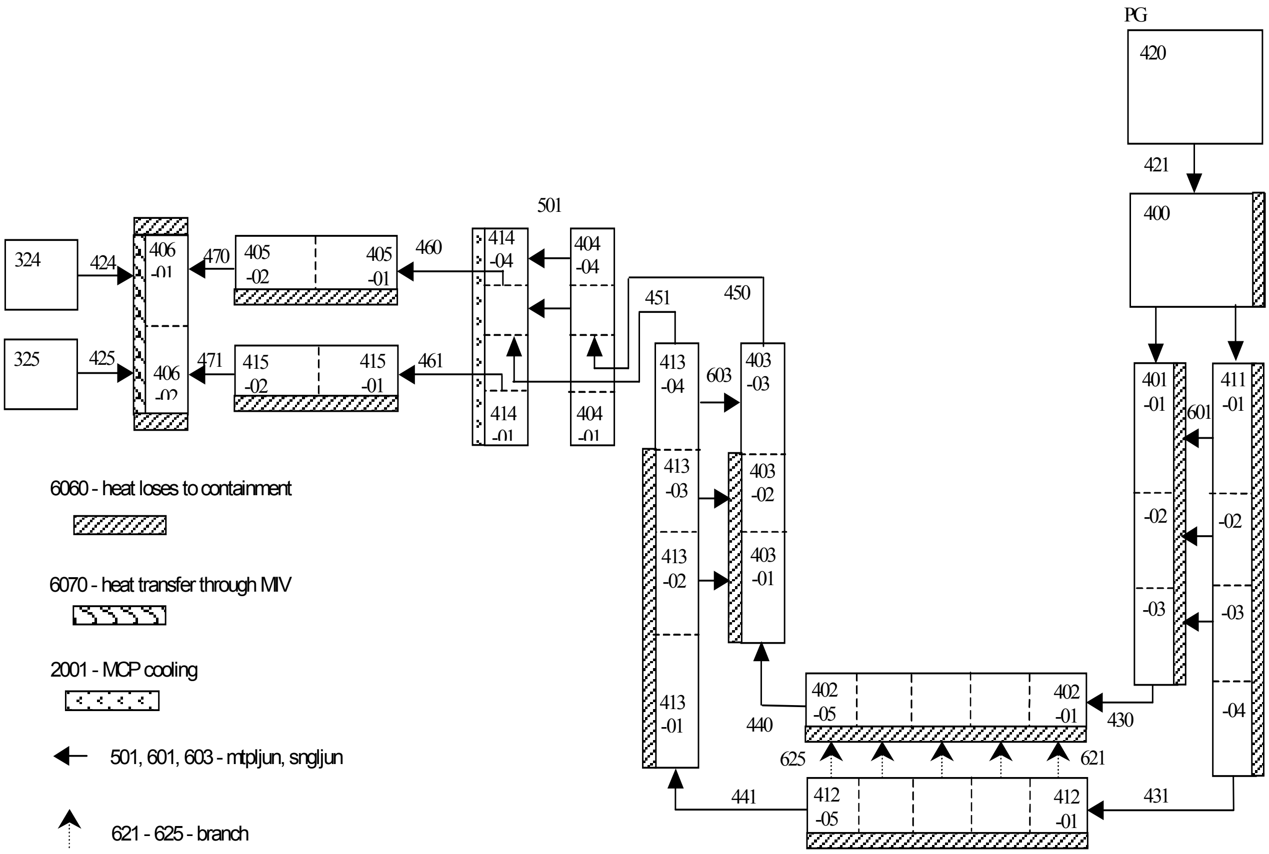


Fig. 5. Model for RELAP5.

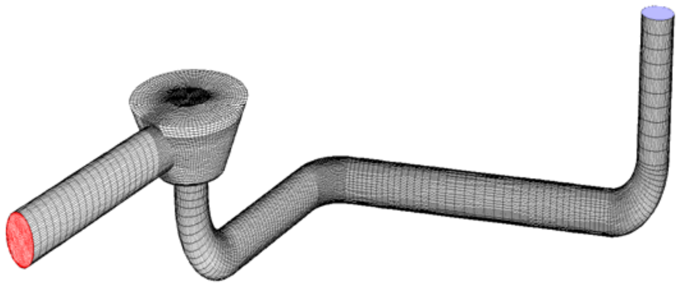


Fig. 6. Model for FLUENT.

2.2.3 Results of analysis and outputs

The calculations proved that most of heat removal from coolant is caused by MCP cooling circuit, other heat losses, including losses through the wall, are much lower. At stagnant flow, internal natural circulation is of the main importance. This internal flow contributes to the relatively good mixing of the coolant in the lower part of the closed cold loop (between main circulation pump and steam generator collector). In the part between MLIV and MCP, stratification is predicted and coolant temperature on the bottom of the pipe is of about 40 °C less than temperature between MCP and SG collector.

The sharper decrease of the coolant temperature after slight opening of the MLIV was caused by relocation of the cold coolant from horizontal upper section. If the operator stays by it, the temperature would increase a bit. But, as the mass flow rate from reactor to inactive loop in such configuration is less than 1.0 m^3/hour, it cannot “flush” water from the cold part.

Corresponding operator training was carried out, presenting all phenomena and influences and results of the analyses. Modification of the MCP cooling circuit operation by isolated loop was suggested to prevent excessive cool down of the isolated loop.

2.3 Suspected low level leakage through the reactor flange sealing

2.3.1 Problem description

The proper tightness of the reactor flange sealing is checked, measuring the pressure in a special annular space between the circular sealings. During reactor start-up and pressurization, unacceptable increase of pressure was identified, but without direct dependency to reactor internal pressure or temperature. Nevertheless, as it could indicate a safety important problem, the start-up was interrupted. Extensive checks of all possible systems and reasons were started. No direct reason was identified, several heat-up and cool-down of the unit were tried, followed by different measures to find and remove the reason, with different impacts to pressure in the flange space. Finally, the space stay at allowed pressure and unit came to power. Due to the potential safety relevance and obvious impact to the economy of the unit, analysis of the problem was required.

2.3.2 Selection of tools and methodology

In spite of the large effort of the staff to find a reason and to cope with the problem, many of the boundary conditions for deterministic analysis remain open. Manual calculations revealed that the reason was with high probability not in insufficient tightness of the sealing. The candidate for the reason was presence of a liquid and/or gasses in the flange space and/or somewhere in the relatively long and complicated line of pressure measurement.

The process is almost stationary. It is governed by heat up of gasses and evaporation of liquid. To take into account differences in temperatures of structures along the flange space and measuring line, RELAP5 model was developed. Large parametric study was performed and compared with the measured data.

2.3.3 Results of analysis and outputs

Results have shown that the pressure increase could be caused by either presence of small amount of coolant in space close to the reactor, heated up by increasing reactor vessel temperature or by a leakage most probably from measuring lines, not from reactor internal side. As the amount of coolant, which is sufficient for measured pressure increase, is extremely low (10^{-5} to 10^{-6} kg/s according the analysis), the location of the leakage is not directly identifiable.

The only suggestion and output from the analysis could be to clean thoroughly both space at reactor flange and measuring lines.

3 CONCLUSIONS

In the last decades of the 20th century, most deterministic analyses of real operational events had been connected with validation of analytical codes and/or models. In this field there is large experience and together with licensing type analyses are close to routine.

Most requirements for analyses of operational events coming from nuclear unit operator are connected with events, very different from typical “safety report” scenarios. It challenges analytical team with additional phenomena, need of development of specific models, justification of the models against available data from monitoring system, and mostly with many uncertainties in boundary conditions.

Analysis of operational event needs realistic approach, detailed justification of the model and scenario and has to answer well-defined questions of the operator. Experience gained during the years of close cooperation between VUJE and Slovak NPP staff shows that such support is extremely important. Deterministic analysis can solve most, but usually not every, questions rose by an unexpected, untypical operational event.

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ANALYSIS OF SAFETY MARGIN MAINTENANCE IN OPERATIONAL EVENTS

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Abstract. An adequate level of safety is achieved in nuclear power plants by a suitable design of protection systems. Safety Analysis Reports typically include a verification that the plant behavior under a set of Design Basis Accidents (DBAs) (including from anticipated operational events to postulated accidents) remains under specified acceptable limits and, therefore, enough safety margin is provided by the protection design. Conclusions of the Design Basis Accident Analysis (DBAA) depend on a set of assumptions on initial and boundary conditions and on system reliability which should be consistent with the operation of the plant. Many of these assumptions are imposed as Technical Specifications or equivalent operation requirements. This kind of safety verification, however, is mainly analytical and only very specific aspects of the protection design can be experimentally verified. Operational events are unique occasions to check, on one hand, if the plant protection behaves as expected and, on the other, if the designed protection is enough to guarantee a sufficient level of safety. There are several mechanisms of safety margin degradation. Some are of dynamic nature, i.e., the plant or protection behavior is not as expected and a loss of margin may occur, eventually leading to limit exceedance. Others are of probabilistic nature, for example, due to a loss of protection reliability leading to a given probability of limit exceedance. Incident analysis should address, to the extent possible, all these mechanisms. As a part of the analysis of operational events it should be identified whether the incident meets the assumptions of the DBA analysis or it is outside the design basis envelope. In the former case, one or more DBAs can be identified as envelope transients. Thus, safety margins during the real event can be verified to be equal to or greater than those demonstrated in the Design Basis Accident Analysis for the envelope transients. In the latter case, it should be identified if one or more of the safety limits used as acceptance criteria for DBAs has been actually exceeded. In any case, an extension of the precursor analysis techniques, usually focused on the eventuality of degradation to severe accident conditions, can be used as a means to evaluate probabilistic mechanisms of safety margin loss.

1 INTRODUCTION

Operation of potentially harmful installations, like nuclear power plants, is only possible if an adequate level of safety is achieved by including in the plant design appropriate protective systems and features aimed at preventing the generation of undesired damage. However, the protection design is particularly difficult due to a number of factors, among them, the wide spectrum of situations to be covered and, in some cases, the lack of knowledge of the damage generating phenomena which may occur in the course of an accident. The eventuality of damage generating events cannot be totally avoided but the protection design should ensure that frequent events will generate small or no damage and that events resulting in high damage generation are very unlikely.

The protection design is based on the consideration of a set of accident scenarios of different severity for which appropriate systems or functions are provided. Although very specific scenarios are used as design basis, they are supposed to cover a broad range of situations and events which hopefully include most of the actual occurrences during the lifetime of the plant. The terms "Design Basis Event", "Design Basis Accident" (DBA) and "Design Basis Envelope" usually refer to the design of automatic protections; however, manually initiated protections also contribute to the achievement of an adequate level of safety if they are systematically designed and adequately implemented in operating procedures.

Safety Analysis Reports typically include the verification of the adequacy of automatic protections. The plant behavior under DBA, ranging from anticipated operational occurrences to hypothetical accidents, is analized to show that it remains under specified acceptable limits (usually referred to as *safety limits*) and, therefore, required safety margins are maintained. The results of the accident analysis depend, nevertheless, on a set of assumptions on initial and boundary conditions, including system availability and reliability which are usually implemented in the form of Technical Specifications or equivalent operating requirements. Plant operation should be made consistent with these assumptions since, otherwise, an acceptable result of the analysis does not guarantee the maintenence of enough safety margin in the real plant. A proper selection of the DBAs allows to conclude that no plant transient matching the assumptions of the analysis will result in a loss of safety margins beyond that of the DBAs. Although some operator actions are given credit in this analysis under very strict conditions, manual actions are, in general, not considered.

This paper discusses how to use the information obtained from operational events to check the protection design by analyzing the consistency between plant behavior and safety analyses.

2 QUALITY AND TESTABILITY OF PROTECTION SYSTEMS

Once the plant with its corresponding protection has been designed, one can wonder about the actual level of safety achieved with this particular design. The answer to this question depends, among other things, on whether the actual plant behavior is consistent with the simulation models used in the accident analysis, on the completeness of the set of DBAs and on the consistency between plant operation and assumptions of the analysis.

Some of these aspects are very difficult to verify. For example, the predicted plant behavior is more difficult to assess as the operating conditions differ from normal operation. In some cases, the only information available for this purpose comes from experimental facilities where the variety of considered situations is necessarily limited and the extrapolation from the facility scale and conditions to the plant accident conditions is, to say the least, difficult. In extreme cases, this information does not even exist.

The completeness of the Design Basis Envelope (i.e., the set of DBAs) is another aspect difficult to verify. The design techniques are necessarily conditioned by the design organization experience and by the knowledge on the plant features that should be taken into consideration. As a consequence, there is always a possibility, especially in new designs, that particular combinations of events and plant conditions that should be considered have been inadvertently left outside the Design Basis Envelope. In addition, if significant design changes are implemented in an operating plant the validity of the Design Basis Envelope should be reanalyzed.

These difficulties show that, unlike other systems, the protection system cannot be fully tested in real conditions in order to confirm the quality of the design or to perform fine tuning of its configuration parameters. Only some partial aspects are tested with real signals, for example during start-up tests. Most of the tests performed on the protection system consist of the verification that its response is as expected but this is done with simulated input signals and without feedback from the plant, since it is not in accident conditions during the test.

The only occasions where the protection system behavior can be evaluated in real conditions is during operational events. Hopefully, its ability to cope with the most severe situations will never be tested but, even with mild operational events, it is possible to obtain very useful indications on the protection system capability to provide an adequate level of safety. A necessary condition to adequately analyze the plant response over an incident is to have a suitable data acquisition system, able to automatically record all the variables and events that could be significant. This should not be a problem in nowadays plants since even those plants originally equipped with poor recording capabilities have the possibility to implement more adequate systems.

3 DEGRADATION MECHANISMS OF SAFETY MARGINS

As explained before, automatic and manual protections are designed to provide sufficient safety margins with respect to unacceptable damage generation. The acceptability of a damage does not only depend on its magnitude but also on the likelihood of this damage to actually occur. Because of that, safety margins should be characterized not only in terms of consequences but also in terms of likelihood (usually measured as expected frequency of occurrence) and both dimensions should be taken into account when evaluating possible degradations of safety margins.

These two dimensions are taken into account in the analysis of DBAs which are grouped into a number of classes according to their expected frequency[1] and different acceptance criteria, i.e., different safety limits, are applied in each class. Any mechanism contributing to increasing the amount of damage generated under given circumstances or the likelihood of damage generating conditions can be considered a loss of safety margins.

Some of the mechanisms for the loss of safety margins are of dynamic nature, i.e., due to a plant or protection behavior different from expected. This kind of margin degradation may result in worse than expected consequences of DBAs or plant transients covered by them, eventually exceeding the applicable acceptance criteria.

Other safety margin losses may result from lack of reliability in protection equipment. In accident analysis, high reliability is assumed for this equipment, based on strict design and qualification requirements and on the application of the single failure criterion. Low reliability equipment is not given credit in accident analysis and, therefore, a degradation in protection reliability is a loss of safety margin.

A third type of degraded safety margin may result from an improper selection of DBAs in such a way that the likelihood of out-of-envelope plant transients becomes too high. This degradation can be caused by deficiencies in the original plant design or by insufficient analysis of the implications of significant plant design changes. Transient scenarios not covered by the Design Basis Envelope do not necessarily result in damage generation or limit violation but it cannot be guaranteed that these eventualities will no occur.

Among these degradations, the two last are of probabilistic nature since the generation of undesired consequences depends on random events like protection failures or the occurrence of particular plant transients. However, they can directly challenge limits and acceptance criteria of the so-called deterministic safety analysis, i.e., the Design Basis Accident Analysis (DBAA). This apparent contradiction only reflects the fact that any approach to the safety problem needs to take into account, explicitly or implicitly, the two dimensions of the problem: consequences and likelihood.

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[1]More exactly, they are classified according to the expected frequency of all the situations they are intended to cover.

The analysis of operational events, understood as protection system tests, should address to the extent possible all the mechanisms of safety margin loss, even when the DBAs are taken as the main reference.

4 ANALYSIS OF OPERATIONAL EVENTS INVOLVING A PLANT TRANSIENT

The term "operational event" is often interpreted as an occurrence taking the plant away from its normal behavior and usually requiring some kind of protective action, i.e., at least a reactor trip. Typically, it includes from the initiating event up the termination of the subsequent transient in a stable safe shutdown state. In next section we are discussing the convenience of extending the concept of operational event to include other cases where the plant operation is not disturbed but we are first going to discuss the usual case which will be referred to as "dynamic operational event" hereafter.

While the occurrence of a dynamic operational event will be evident, it is not always easy to identify the nature of the initiating event (i.e., the immediate cause of the transient), the boundary conditions which may influence the subsequent plant evolution or the undesired consequences that the event could produce. To this aim, the existence of an adequate data acquisition system may be an important help but, in general, the information it provides must be further elaborated.

The first task in the analysis of a dynamic operational event is to understand it, i.e., to identify its immediate causes, phenomena that have occurred during the event evolution, automatic or manual actuations, environmental and other boundary conditions with significant impact on the event, and so on.

As long as the degree of detail in the analysis is more than purely qualitative, the use of computer codes to simulate the incident is a need. The qualification of the simulation models and associated input data describing both the plant and the phenomenology must be deeply discussed; however, for the purposes of this paper we are assuming that a fully qualified simulation model, able to reproduce the incident, is available.

The analysis of dynamic operational events is explained below and summarized in Figure 1.

4.1 Classification of the event

Among the many aspects that can be analyzed in an operational event we are focusing in the maintenance of safety margins, i.e., in the capability of the plant protection to prevent the generation of unacceptable damage. From this point of view we may distinguish between operational events included in the Design Basis Envelope (i.e., matching the assumptions of the DBAA) and those outside the envelope. The main classification criteria are:

- The type of initiating event. Care should be taken in the assessment of this criterion since in some cases the analysis of a particular DBA actually covers other possible initiating events of different nature but with similar consequences. A typical example is the analysis of uncontrolled control rod withdrawal in PWR which actually covers many reactivity insertion events including, for example, some overcooling events.
- Consistency between initial and boundary conditions during the operational event and Technical Specifications or equivalent operating requirements assumed in the DBAA. The initial plant state, the availability of required safety equipment and the correctness of safety system settings should be taken into account.

— Operator actions. Except for some special cases, operator actions are not considered in DBAA. This does not mean that any operator action not considered in the safety analysis takes the event out of the Design Basis Envelope. Only those actions which conflict with some assumption of the analysis result in classifying the event as not included in the Design Basis Envelope.
— Equipment failures or human errors beyond the assumptions of the DBAA. The protection design and the associated accident analysis require the application of the so-called *single failure criterion*. The main implications of this criterion are that multiple independent failures should not be considered as Design Basis Initiating Events and that any single failure in safety related equipment should not prevent the achievement of the required safety function.

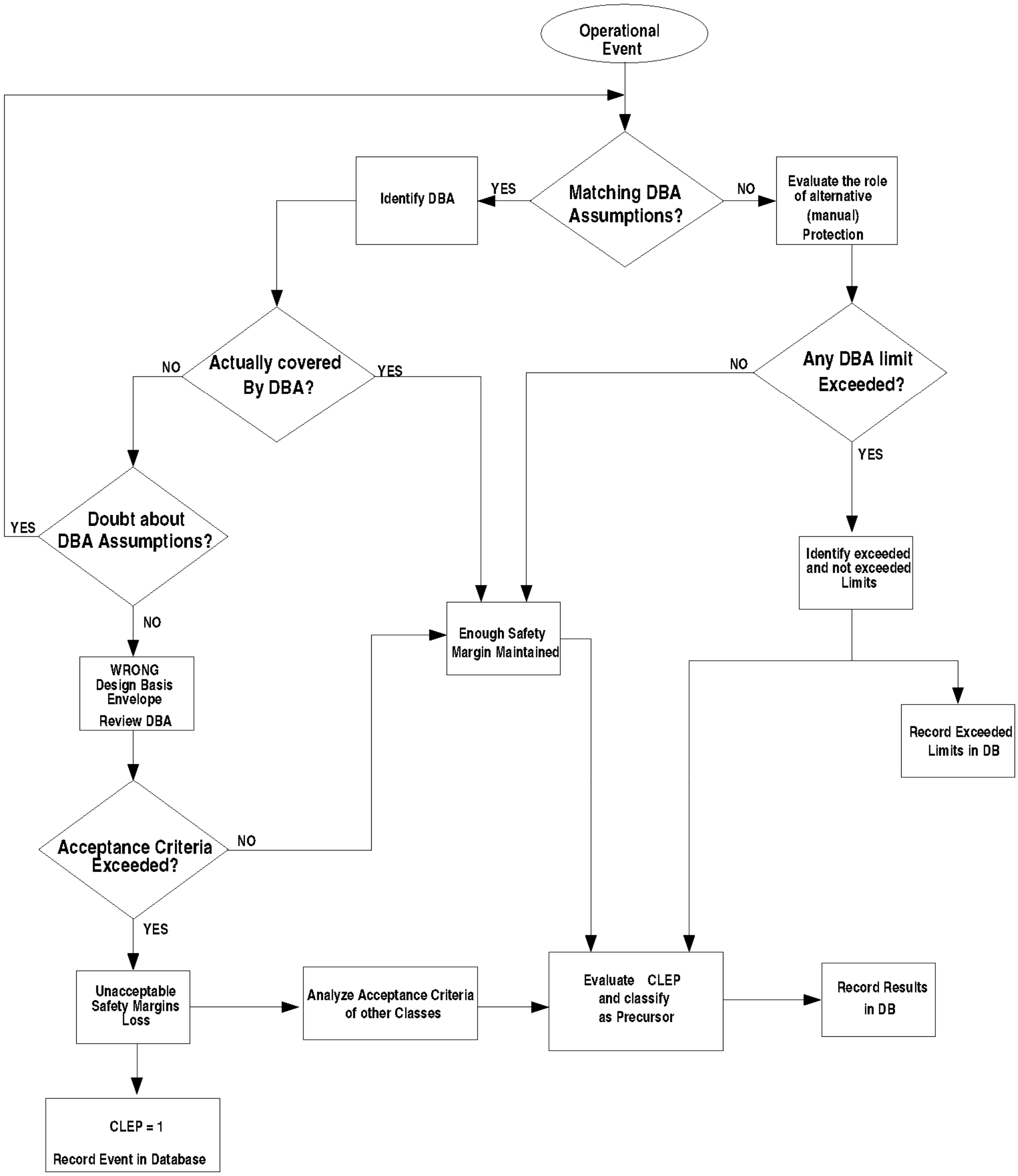

Fig. 1. Flow diagram for the analysis of dynamic operational events.

4.2 Dynamic Safety Margin assessment in operational events within the Design Basis Envelope

If an operational event is found to match the assumptions of the protection design, it will be possible to identify one or more related DBA involving similar safety challenges. It should be pointed out that different aspects of a single operational event may be covered by different DBAs. For example, a loss of external power supply in a PWR can be related with Loss of Primary Flow and with Loss of Feedwater DBAs.

The most important aspect to be evaluated in this case is if the identified DBA(s) is (are) actually an envelope of the operating event. The evolution of significant safety variables should be analyzed to check that their extreme values remain on the safe side with respect to the corresponding extreme values in the DBA(s).

As a result of this analysis it can be found that safety variables during the operational event have exceeded the extreme values given by the DBAA. Should this occur, it indicates that the operational event is not actually covered by the DBAA. It is advisable to reanalyze in this case the compliance with the DBAA assumptions and to reconsider the event classification with respect to the Design Basis Envelope. If this compliance is confirmed it can be concluded that the DBAA must be reviewed. Still, exceeding the results of the DBA does not necessary mean to exceed the applicable safety limits but, if these limits are found exceeded in the incident, it means that an unacceptable loss of safety margins has occurred and reinforces the need to review the protection design and the DBAA.

If the identified applicable limits are not exceeded, it can be concluded that enough safety margins have been maintained during the operational event from the dynamic point of view. However, the probabilistic mechanisms of safety margin degradation should still be analyzed as discussed later.

4.3 Dynamic Safety Margin assessment in operational events outside the Design Basis Envelope

Operational events outside the Design Basis Envelope may involve protective measures, including operator actions, not credited in DBAA. It is important to identify if these kinds of actuations have actually occurred and to determine as accurately as possible the role they have played during the event.

To be outside the Design Basis Envelope does not necessary mean to exceed limits or acceptance criteria of the DBAs. Actually, one of the objectives of the analysis of the operational event is to identify what safety limits have been exceeded. Only if no limit has been exceeded, it can be concluded that enough safety margin has been maintained from the dynamic point of view. Operational events outside the Design Basis Envelope cannot be classified in a Design Basis Class and, therefore, it is not possible to distinguish which limits should be maintained and which should not. Only the evaluation of probabilistic mechanisms of safety margin loss can give some indication of the safety significance of the event.

4.4 Probabilistic Safety Margin assessment of dynamic operational events

When a dynamic operational event occurs, the very fact that a particular safety limit has not been exceeded does not mean that there is enough safety margin with respect to the damage such limit tries to prevent. As an example, we may consider a hypothetical event where the limit exceedance has been prevented by an operator action taken at a particular time point. If the analysis of the event reveals that an eventual small delay in the operator intervention would lead to safety limit exceedance or even to damage generation, it can be concluded that, if a similar situation occurs again, the probability of limit exceedance would be very high. In this case, it cannot be considered that enough safety margin has existed during the operational event.

The best magnitude to measure the probabilistic proximity to some limit or undesired condition during an operational event is the conditional probability of reaching such condition, given the occurrence of the event. The acronym CLEP (Conditional Limit Exceedance Probability) will be used for this measure.

If a particular limit is exceeded during a dynamic operational event, the above defined conditional probability would be clearly 1 and it is equivalent to an identified loss of dynamic safety margin. For those limits not exceeded, credible variants of the operational event can be analyzed to assess whether they would have resulted in safety limit exceedance. These variants may result, for example, from additional equipment failures, alternative operator behavior, including human errors, different actuation times, uncalibrated instruments, differences in initial or boundary conditions, etc. Of course, the probability of these alternatives should be evaluated somehow to get the CLEP of each limit.

According to the calculated conditional probability, the operational event can be classified with respect to each considered limit. Following a classification scheme consistent with that used in PSA based event analysis seems reasonable. For each limit being analyzed, the event can be classified in one out of four classes, namely, *non-significant*, *precursor*, *significant precursor* or *limit exceeded*. The boundaries between these classes depend on the particular limit under consideration. Limits associated to low frequency classes of DBA would have also lower boundaries of conditional probability for event classification. A tentative classification scheme is proposed in Table 1. *Limit exceeded* always corresponds to CLEP = 1 and *non-significant* means that CLEP is lower that the *precursor event* boundary.

TABLE 1. PROPOSED CLASSIFICATION OF OPERATIONAL EVENTS ACCORDING TO CLEP

Type of Safety Limit	Precursor Event	Significant Precursor Event
Applicable to C-II DBA	CLEP $\geq 10^{-2}$	CLEP $\geq 10^{-1}$
Applicable to C-III DBA	CLEP $\geq 10^{-4}$	CLEP $\geq 10^{-2}$
Applicable to C-IV DBA	CLEP $\geq 10^{-6}$	CLEP $\geq 10^{-4}$

The occurrence of an event being classified as precursor or significant precursor with respect to a particular limit is not, by itself, an indication of lack of safety even though it may be indicative of corrective measures needed. It should be taken into account that the calculated probability is conditional to the occurrence of the event. Only if similar events occur at an unexpected high frequency can it be concluded that there could be a safety problem. For this reason, it is very important to keep historical records of operational events including the results of probabilistic safety margin analyses. The number of precursor or significant precursor events with respect to each safety limit can be used as indicators of the plant safety performance.

5 ANALYSIS OF DEGRADED OPERATIONAL CONDITIONS

Some events occurring in operating power plants may have no impact on the plant dynamics unless a later initiating event triggers a plant transient. This kind of latent event affect stand-by systems which should work only under specific conditions. Protection systems are the most typical examples of these systems and, when a latent failure occurs in a protection, the plant safety becomes degraded although no impact on the plant operation can be observed. Latent events are discovered either when the affected system is called for operation or when a surveillance test is performed on that system.

Since no dynamic disturbance occurs, the safety significance of this kind of operational events cannot be evaluated in terms of dynamic safety margin loss. Only the probabilistic mechanisms of safety margin degradation can be evaluated. However, the probabilistic margin assessment methods used in the case of dynamic operational events cannot be directly applied since there is not a reference transient from which different variants are obtained. The plant protection should be able to cope with any possible initiating event and the analysis of latent operational events should be consistent with this principle.

Given the occurrence of a latent event, a particular safety limit will be exceeded only if some plant transient is triggered and the affected system is needed to prevent the exceedance of that limit. The frequency of the possible initiating events (eventually influenced by the latent event) and the probabilities of additional events configuring the subsequent transients are the ingredients for the estimation of the expected conditional frequency of limit exceedance, given the occurrence of the latent failure. The duration of the degraded condition is also important for the evaluation of latent events. The highest difficulty is the estimation of the time when the degradation occurred. The end point, on the contrary, is the time when the failure was corrected, which is usually known. The product of the conditional frequency of limit exceedance times the duration of the degraded condition gives a value of the conditional limit exceedance probability which allows one to classify the event with the same criteria used in the classification of dynamic operational events.

The main difficulty in the analysis of probabilistic safety margin degradation due to latent events lies in the identification of all the paths from possible initiating events leading to limit exceedance where the latent failure is relevant. The extension of PSA techniques to consider any safety limit as sequence success criterion could provide the basis for suitable methods of latent event analysis.

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