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# Probabilistic and deterministic analysis of BDBA in RBMK-1500

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By now, design basis accidents for the RBMK-1500 have been rather thoroughly investigated. Analyses helped to develop and implement a number of safety-ensuring modifications. The further plant safety enhancement requires developing the emergency procedures that would enable to manage beyond design basis accidents by preventing the core damage or mitigating consequences of severe accidents. To develop the emergency guidelines, it is necessary to collect information on initiating events leading to beyond design basis accidents, phenomena taking place during such accidents and possibilities of accident management. Collection of such information would require a lot of deterministic calculations. However, results of probabilistic safety assessment could be used to identify the most important scenarios. The paper presents results of the Ignalina NPP Level 1 and Level 2 probabilistic safety assessments and their use for compiling a list of beyond design basis accidents.

The paper describes the most important phenomena for the management of severe accidents in RBMK-type reactors. The paper also presents the physical processes that occur in an overheated reactor core and discusses its cooling capabilities. The discussion also continues on the processes that occur in the reactor and primary circuit surrounding compartments, and recommendations on their management are presented.

**Key words**: beyond design basis accident, RBMK reactor, accident management, probabilistic safety assessment

## 1. INTRODUCTION

The Ignalina NPP is the only nuclear power plant in Lithuania consisting of two units commissioned in 1983 and 1987. Both units are equipped with channel-type graphite-moderated boiling water reactors RBMK-1500 [1]. Unit 1 of the Ignalina NPP was shutdown for decommissioning at the end of 2004.

The design-based accidents for the RBMK-1500 have been quite thoroughly investigated. Analysis helped to develop and implement a number of modifications that ensure the safety of the power plant. A number of incidents and accidents are considered during the design stage of the NPP. For such events the safety systems and special procedures are developed in order to prevent the accident escalation and radioactivity release to the environment. Nevertheless, there are events that are not foreseen in the design, and they are called Beyond Design Basis Accidents (BDBA). According to IAEA definitions, there are two groups of BDBA: 1) BDBA without core damage, and 2) severe accidents, i. e. accidents that involve core damage. Additional efforts are required to manage these accidents in order to protect the environment from radiation release from the NPP.

All existing capabilities of the plant should be investigated in order to enhance safety, e.g., hook-ups of non-dedicated systems and temporary connections. Such research would enable development of the BDBA guidelines for accident management in order to prevent core damage or to mitigate the consequences of accidents. Such guidelines could be developed only after collecting information regarding initiating events leading to core damage; the phenomena taking place during such accidents are registered and analysed, and the capabilities of the plant equipment, instrumentation and control devices and existing procedures are considered. The following sources of information should be considered for such work [2]:

- Results of probabilistic safety assessment
- Research on severe accident phenomena
- Study of operational experience and precursor events
- Generic studies and analyses done for similar or reference plants

• Review of existing procedures to assess their limitations

• Evaluation of instrumentation behaviour and limitations for accident identification and control

• Evaluation of operating organisation capability in emergency situations

• Plant-specific operational experience

• Generic operational experience (e.g., IAEA database).

Probabilistic Safety Assessment (PSA) plays an important role in the development of accident management program. It is used for compiling the list of BDBA, selection of suitable strategies, development of actual accident management guidance and training.

## 2. PROBABILISTIC SAFETY ASSESSMENT OF THE IGNALINA NPP

Probabilistic safety assessment is a comprehensive, structured approach to the identification of the dominant failure scenarios constituting a conceptual and mathematical tool for deriving numerical estimates of risk. Three levels of PSA are generally recognized. Level 1 comprises assessment and quantification of plant failures leading to core damage. Level 2 includes the assessment of containment response that leads, together with level 1 results, to estimation of radiation release. Level 3 includes the assessment of off-site consequences, which together with the results of level 2 analysis estimates risks to the public.

The Ignalina NPP PSA project was initiated in 1991 and is known as the Barselina project [3]. The project initially was related to a multilateral co-operation among Lithuania, Russia and Sweden. The long-range objective was to establish common perspectives and unified bases for assessment of severe accident risks and needs for remedial measures for the RBMK reactors.

The Level 1 PSA study was developed in five stages and was reviewed by two IAEA review missions in 2000 and 2001. The first level of the Ignalina NPP PSA is the most mature and represents the most recent plant configuration.

The first outline of the Ignalina NPP Level 2 PSA was issued in 1997. The Level 2 PSA study was completed within the international project "Accident Management, Consequence Mitigation and PSA Level 2" in 2001. As this study was the first Level 2 PSA application for RBMK type reactors, a rather conservative approach was adopted and the results abounded in uncertainties.

### 2.1. Results of PSA level 1

The Initiating Events (IE) considered in Level 1 PSA are limited to internal events such as transients, loss of coolant accidents (LOCA) and internal hazards. Event sequences were defined and evaluated for initiating events that could arise in full-power operating conditions and at >50% of full power. The other operational stages are excluded according to the scope of this study. The initiating event screening covers the following main areas:

• LOCA events (both design-based and beyond design basis) are categorised according to size and zone categories. The size categorisation reflects guillotine ruptures. Cracks and leaks are represented by a smaller size in the same zone

• Primary circuit events, including blockage events

• Power conversion system events

• General Common Cause Initiators (CCI), including support systems events

• Plant model analysis of safety function degradation events

• Area events (internal fire, flooding and missiles).

The following events are recognized as potential initiating events, but are excluded from the analysis due to its limited scope:

• Support system events: crane or reloading machine falling on the reactor upper plate; loss of water in the spent fuel pool

• Local events: hydrogen explosion in the Accident Localisation System (ALS)

• External events: airplane crash, flood outside the plant, clogging of the ultimate heat sink water intake; a seismic event. It should be noted that external events were analysed and quantified in the safety analysis report, but it was not integrated with the PSA model.

The next step is to identify the safety functions that prevent core damage. Normally, the plant model is concentrated on the main functions, i.e. front line system functions, and includes these as main headings in the event trees. All support functions, including alarm signal generation and power supply, are modelled with fault trees to which transfers are made from the front line system (safety systems) fault trees.

System analysis was performed using the fault tree technique. In general, functional representation in the system analysis is believed to be good and is highly credible. The overall system analysis covers 18 systems.

Functional block diagrams showing a successful accident protection illustrate the required safety functions. For each IE the safety functions needed to prevent core damage are identified. In the following, accident sequences are developed based on event trees. The event trees present descriptions for the functional events contained in the accident sequence analysis and their success criteria. The time considered in the accident sequence analysis is 24 hours. The accident sequence model for reactor cooling is a phased mission model divided into three time periods:

• Short-term cooling 0–2 minutes

• Medium-term cooling 2 minutes – 1 hour

• Long-term cooling 1 hour – 24 hours.

The following core hazard states after the initiating event were defined for the Level 1 PSA model:

• Safe state (S): No exceeding of the Maximum Design Limits (MDL) or exceeding of the Safe Operation Limits (SOL) in no more than three Fuel Channels (FC)

• Violation (V): exceeding of SOL in more than three FC due to cladding defects (but no FC ruptures at a high pressure)

• Violation (V1, V2): exceeding of MDL in no more than 3 FC, leading to or caused by 1–9 FC ruptures at a high pressure

• Damage (D): exceeding of MDL in 4–90 FC (accompanied by ruptures of no more than 9 FC at a high pressure)

• Severe damage (A): exceeding of MDL in more than 90 FC or rupture of more than 9 FC.

The general results show that probability for the end state "Violation" is in the order of $10^{-2}$ and is dominated by single channel blockage. This probability is based on operational data, as three cases have occurred in RBMK reactors. However, the design of the channel inlet valves was changed to lower the IE frequency.

The "Damage" and "Accident" end states are combined into one "Core Damage" category whose frequency is estimated to be $5.9 \cdot 10^{-6}$ per reactor year. The Lithuanian target value for the core damage frequency (CDF) is $1.0 \cdot 10^{-5}$ per reactor year. The target values for the CDF in Western countries varies between $1.0 \cdot 10^{-4}$ and $1.0 \cdot 10^{-6}$. The estimated CDF includes BDBA and a contribution from the group distribution header (GDH) blockage. The contribution to CDF from various initiating event groups is shown in Fig. 1. The plant risk topography is presented in Fig. 2. The plant risk topography provides estimates of each IE frequency and the corresponding plant barrier against the reactor core damage.

The interpretation of the results is carried out in the form of the main contributor analysis. The contributor analysis suggests a number of improvements that were further analysed in the sensitivity analysis. The study identified the following main plant risk contributors:

• Drum-separator rupture – DS-RUPT (CDF contribution $1.0 \cdot 10^{-6}$). This event is a beyond design basis

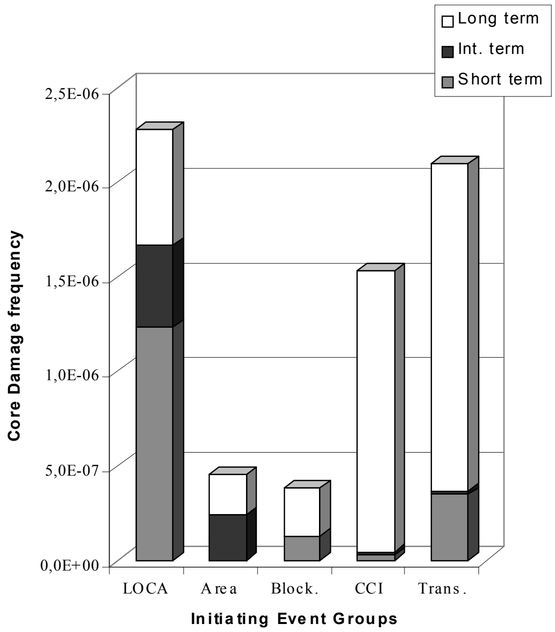


**Fig. 1.** Core Damage Frequencies for Groups of Initiating Events. LOCA – loss of coolant accidents; Area – area events (internal fire and flood); Block. – blockage events; CCI – common cause initiators; Trans. – transient events

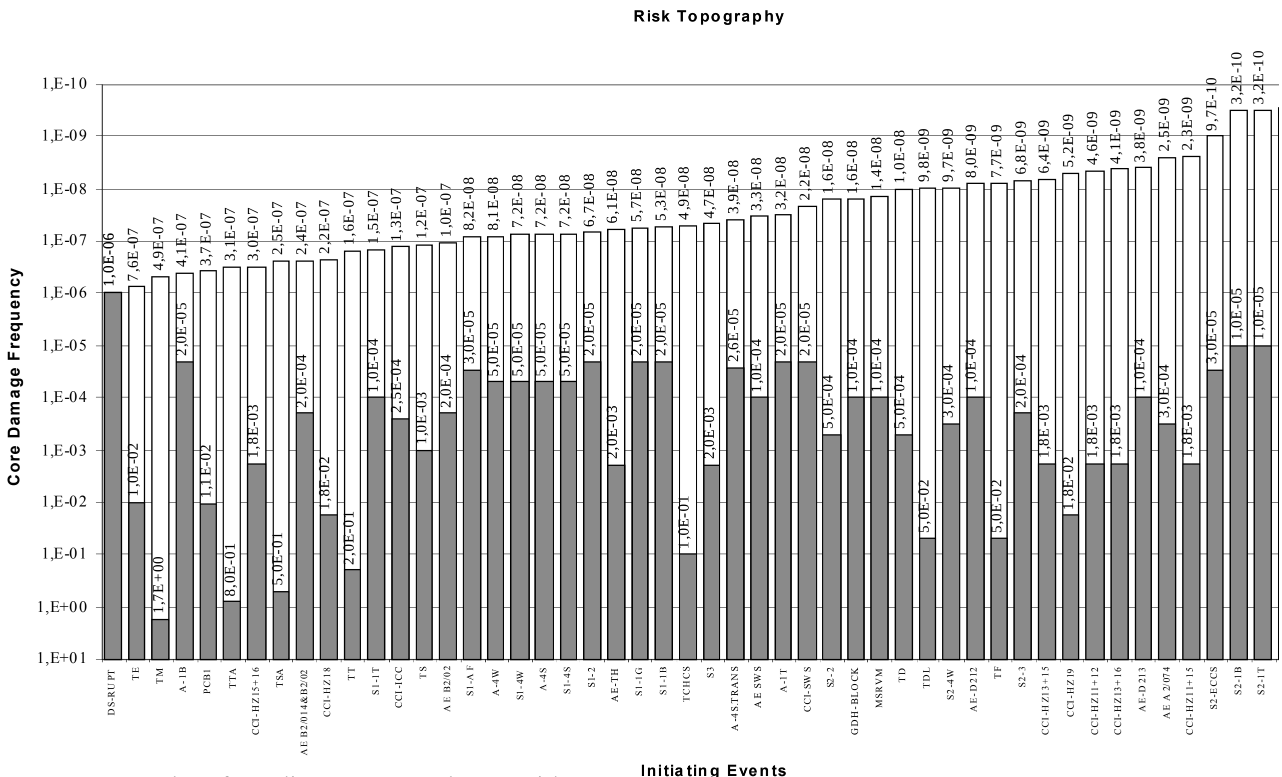


**Fig. 2.** Topography of Ignalina NPP core damage risk

accident. The drum separator rupture is treated as a vessel rupture in vessel-type reactors. The consequence is assumed to be a short-term accident.

• Loss of Offsite Power – TE (CDF contribution $7.6 \cdot 10^{-7}$). The redundancy of the emergency power is six-fold in many situations. The reactor is forgiving in the sense that core damage does not occur during the first hour of a total blackout. The battery depletion time is specified to 60 minutes. A diversification of emergency power or improved accident management procedures would contribute to elimination of the risk.

• Manual Shutdown – TM (CDF contribution $4.9 \cdot 10^{-7}$).

• Large LOCA in Zone 1 at MCP discharge header A-1B (CDF contribution $4.1 \cdot 10^{-7}$).

• Blockage of one Fuel Channel – PCB1 (CDF contribution $3.7 \cdot 10^{-7}$).

• Turbine Trip with condensers available – TTA (CDF contribution $3.1 \cdot 10^{-7}$).

• Loss of safety buses HZ15 and HZ16 – CCI-HZ15+16 (CDF contribution $3.0 \cdot 10^{-7}$).

The plant CDF is dominated by the end state "Accident". This depends on the long-term sequences. Several proposals for further mitigation features, e.g., emergency operating procedures to depressurise the primary circuit and to use non-safety grade means for emergency core cooling, were studied within the scope of PSA Level 2 project.

A comparison of different classes of initiating events, 'D' and 'A' sequences are dominated by transient, common cause initiators and area events, giving half of the CDF contribution. Blockages give a very limited contribution. The contribution of LOCA became also significant, but it is dominated by the beyond design basis LOCA – drum separator rupture – frequency. The other pipe rupture frequencies used in the calculations have a large impact on the LOCA contribution.

The PSA model has been used to demonstrate the impact of various proposed changes of plant design as well as to show the importance of various assumptions used in the PSA. However, uncertainties always exist in the data used for initiating events and component reliability. The impact of some of these uncertainties have been demonstrated and discussed in the sensitivity analysis. The model, however, lacks a comprehensive uncertainty analysis, which should be accomplished, especially for the models giving low risk estimates.

### 2.2. Results of PSA Level 2

Level 2 PSA study extends results of Level 1 PSA to analyse the accident processes after the core damage occurs. Therefore, Level 1 PSA results are recombined to plant damage states (PDS), each serving as a Level 2 PSA initiating event. Then accident progression event trees (APET) are developed and quantified using a computer model. Both Level 1 and Level 2 PSA were integrated into one INPP risk model with a Risk-Spectrum PSA computer tool.

For the Level 2 purposes more detailed consequences were required. For each Level 1 accident sequence, a detailed consequence (AS, AI, AL, DS, DI, DL, VS, VI, VL) was assigned, consisting of damage severity and timing factors (e.g., AS means a severe core damage in a short term of 0–2 min). This work was done in the early phase of the Level 1 PSA Phase 5 updating and preparation for the Level 2 analysis.

Totally, 16 PDSs were defined. Each PDS was named by the following format: Y–Z – where Y represents Level

Table 1. **List of analysed plant damage states for RBMK-1500 reactor**

| PDS | Frequency | Description | Comments |
|---|---|---|---|
| A-HP | $2.6 \cdot 10^{-6}$ | Accident in long term (AL) after transients and area events with high pressure | Accident mitigation is possible. In case of successful mitigation it is possible to reduce core damage consequences |
| D-LLZ1 | $3.4 \cdot 10^{-7}$ | Damage in intermediate term after LLOCA in zone 1 | AM actions are not possible due to short time window. APET analysis should be performed |
| V-LLZ1 | $3.4 \cdot 10^{-7}$ | Violation after LLOCA in zone 1 | |
| V-MLZ1 | $1.7 \cdot 10^{-7}$ | Violation after MLOCA in zone 1 | |
| V-MLZ2 | $1.1 \cdot 10^{-6}$ | Violation after MLOCA in zone 2 | |
| S-LLZ1 | $4.0 \cdot 10^{-5}$ | Safe state after large LOCA in zone 1 | Although safe state was reached, large amount of coolant is released to the ALS and can challenge structural integrity of the ALS. In case of ALS damage, reactor safety systems can be affected and even accident state can be reached. APET analysis is essential for these PDS |
| S-MLZ1 | $1.7 \cdot 10^{-4}$ | Safe state after Medium LOCA in zone 1 | |
| S-MLZ2 | $2.0 \cdot 10^{-5}$ | Safe state after medium LOCA in zone 2 | |

1 hazard state and Z – abbreviation of the Level 1 initiating event.

Seven PDSs were selected for a detailed APET analysis, as is shown in Table 1. The remaining eight PDSs were determined as having small releases and environmental consequences or direct releases without containment function intact, and the release category was assigned directly. One PDS (A-HP) was analysed through the accident management (AM) scheme.

In Level 2 analysis for the RBMK-1500 reactor, it was conservatively assumed that accident sequences after all initiating events resulting in a severe core damage in short and intermediate terms (AS, AI) led directly to the highest release category without ALS protection. Accidents in a long term (AL) were grouped into two PDSs, based on the pressure criteria in the primary system. Basically, after all LOCA events, the pressure in the Main Circulation Circuit (MCC) will be low, and for these events the PDS – A-LP is defined. In case of transients and local events, pressure will be kept high, and this leads to the PDS – A-HP. It is assumed that high pressure in the MCC will eventually lead to depressurization, i. e. a rupture of several channels with the highest power. However, this scenario needs a deterministic justification and further investigations. A long-term cooling failure is subject to accident management actions, which could be justified by deterministic analysis.

Table 2. **Release categories**

| | Release, % of core inventory | Amount of radioactivity released to environment | INES scale |
|---|---|---|---|
| INPP1 | <0.003 | Up to 3500 Ci of $^{131}$I | Low release INES 1-3 |
| NPP2 | 0.003-0.2 | Up to 35000 Ci of $^{131}$I | Medium release INES 4-6 |
| INPP3 | >0.2 | More than 35000 Ci of $^{131}$I | Large release INES 7 |

Table 3. **ALS failure probabilities**

| Event | Conditional probability |
|---|---|
| ALS failure in short term due to DBA in zone 1 or GDH rupture in zone 2 | $1.0 \cdot 10^{-2}$ |
| ALS failure in short term due to medium LOCA in zone 2 | $1.0 \cdot 10^{-3}$ |
| Conditional probability of safety system failure given events above | 1.0 |
| ALS failure in long term due to CTCS failure | $1.0 \cdot 10^{-1}$ |
| ALS failure in long term with CTCS available | $1.0 \cdot 10^{-2}$ |

The release categories depend on the release paths that could be generally divided into three groups:

• Direct release: release from confinement zone 4 or zone 5 to the environment without getting into the Accident Localisation System (ALS) or from zone 3 if the top metal structure of the reactor is lifted

• Through ALS: release to the environment from zones 1–3. ALS is not bypassed and ALS structures remain intact

• Limited ALS: ALS fails due to overpressure or is bypassed due to a structural leakage of the compartments. In this case, there is a possibility of radioactive products distribution and retention in ALS before and after the condensing pools.

No calculations were performed to calculate the source term for different hazard states. The source terms are rough expert estimates and are subject to high uncertainties. On the basis of the PSA Level 1 hazard states, three release categories are defined in Table 2. Each release category considers PSA Level 1 hazard state and the filter factor of the appropriate release path.

The APET analysis of the RBMK-1500 reactor is simplified and is not typical in a sense that it does not include phenomenological events. The APET block diagram for the PDS – D-LLZ1 is shown in Fig. 3 and is rather representative of the events analysed for the other PDSs. The quantification of function events in APETs is supported by deterministic ALS fragility calculations. The APET considers the reliability of the ALS active systems: condenser tray cooling system, spray availability, and the closure of the venting valve (through fault trees). No equipment recovery was considered.

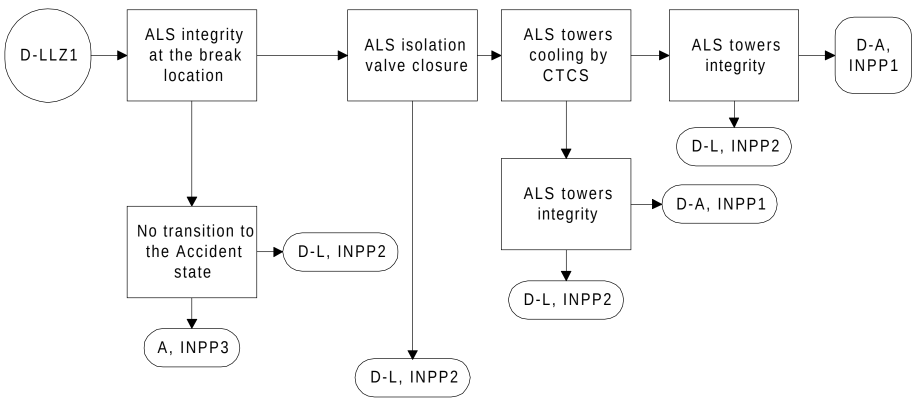


**Fig. 3.** APET block diagram for the PDS–D-LLZ1

The ALS integrity failure probabilities are based on the deterministic maximum pressure and integrity calculations and were assigned by experts in a conservative way (Table 3). The conditional core cooling failure probability resulting in the accident damage state and thus INPP3 release state was conservatively set to unity.

Table 4. **Release frequency without ALS (with ALS)**

| | Frequency without AM | Frequency with AM |
|---|---|---|
| INPP3 | $6.2 \cdot 10^{-6}$ *($6.3 \cdot 10^{-6}$)* | $3.7 \cdot 10^{-6}$ *($3.8 \cdot 10^{-6}$)* |
| INPP2 | $1.5 \cdot 10^{-2}$ (*$2.1 \cdot 10^{-4}$*) | $1.5 \cdot 10^{-2}$ (*$2.1 \cdot 10^{-4}$*) |
| INPP1 | 0 (*$1.5 \cdot 10^{-2}$*) | 0 (*$1.5 \cdot 10^{-2}$*) |

Due to specific features of the RBMK, in most cases it is necessary and possible to avoid the transition to the area of a severe core damage. Level 1 PSA results showed that the main risk contributors of the RBMK-1500 are transients with the loss of long-term reactor cooling, and for many accidents there is a long time period between going beyond the design basis and the onset of a severe fuel damage. During this period, the core and MCC undergoes what is essentially a severe thermal-hydraulic transient. An active accident management performed by plant operators may terminate the accident before a severe fuel damage occurs.

The results of calculations with the integrated Level 1 and Level 2 PSA model are shown in Table 4. The total frequency of INPP3 (national emergency state) was estimated to be $4.6 \cdot 10^{-6}$ per year without AM interference and $1.9 \cdot 10^{-6}$ with accident management.

An intermediate release category INPP2 (local or regional emergency state) frequency was estimated to be $2 \cdot 10^{-4}$, which is a high figure for a release of this magnitude. The dominating sequence is a large LOCA in zone 4 with a successful core cooling. This frequency is not possible to reduce by AM actions. Despite the successful cooling, the release magnitude is high and occurs in a short time scale – such an accident could be rated as an INES 4-5. However, assumptions of a fission product release are extremely conservative, and it is believed that this accident should fall into the INPP1category, reducing the INPP2 frequency to the order of $1 \cdot 10^{-7}$.

The limited release category (INPP1) is dominated by the design sequence "rupture of a single pressure tube", which has a frequency of the order of $1 \cdot 10^{-2}$. This frequency is experience-based; a number of cases have occurred, e.g., in the Leningrad NPP. The other contributions come from sequences initiated by small LOCAs.

The long-term accident state is reduced with a factor of 10 with accident management. However, the large release frequency is only decreased with a factor of 2. The remaining contribution comes from short- and intermediate-term sequences. The success of the LOCA contribution to accident state could possibly be reduced by a more detailed study of conservative assumptions.

The ALS effectiveness calculations show that the major ALS function usefulness is reflected by the INPP2 frequency increase in case the ALS is not considered in the analysis. The ALS has no impact on the INPP1 frequency. For the INPP3 category, the ALS due to a possible transition to the accident state slightly increases the release frequency.

The Level 2 PSA study is considered to be the first approach for an RBMK-type reactor and contains a lot of uncertainties in many areas of analysis and quantification process. No explicit uncertainty analysis was performed at this stage of the study.

The study adopted a conservative approach to deal with uncertainties, and therefore the quantitative results (both release size and frequency) are estimated with a high level of conservatism, therefore the quantitative estimates could be interpreted as at least 95 percentile of the uncertainty distribution.

However, the following areas are of particular importance in reducing the conservatism and moving towards the best estimate values: ALS fragility evaluation; source term calculations; human action reliability during accident mitigation and quantification of phenomenologically severe accident events.

The results still clearly demonstrate the efficiency of improvements implemented in the plant, e.g., Diverse Shutdown System for the reactor scram, the new criteria for the reactor scram and Emergency Core Cooling System (ECCS) initiation from GDH low flow, and the new reliable main steam relief valves. These improvements have reduced the frequency of short- and intermediate-term accidents to a level of $8 \cdot 10^{-7}$.

## 3. LIST OF BEYOND DESIGN BASIS ACCIDENTS

The original RBMK design does not include analysis of BDBA, and there are no specific instructions or guidance for severe accident management. On the other hand, the nuclear safety regulations [4] in the Republic of Lithuania require that the reactor plant technical design shall contain provisions for controlling the beyond design basis accidents. For dealing with BDBA, the Operating Utility shall prepare, in accordance with design documentation, a special manual which must be adopted after consultations with the State Control and Supervisory Institutions [4]. The first step in this Severe Accident Management Guidelines (SAMG) process is compiling a list of beyond design accidents [5].

The BDBA list could be compiled using different approaches:

- based on deterministic principles,
- based on PSA approach.

In the deterministic approach, the accidents that are beyond the scope of the existing Emergency Operating Procedures (EOPs) are selected.

The PSA approach includes all the initiating events that can occur at a plant and the accident sequences that can occur as a consequence of these events, resulting in the core damage (or other endstates defined in the PSA). It also provides the frequency of each accident sequence. The BDBA list includes all important core damage and release sequences of the Level 1 and Level 2 Ignalina NPP PSA. To ensure the completeness of the BDBA list, each event tree sequence from

Table 5. **List of BDBA categories for internal full power events**

| No. | Description | Frequency |
|---|---|---|
| 1 | Rupture of MCC piping (diameter 100–900 mm) in reinforced leaktight compartments or GDH compartments or spaces below the reactor. No short-term ECCS | $8.49 \cdot 10^{-9}$ |
| 2 | Rupture of MCC piping or feedwater supply lines or steam lines (diameter 100–600 mm) outside the ALS compartments. No short-term ECCS | $7.08 \cdot 10^{-9}$ |
| 3 | Rupture of MCC piping (diameter 100–900 mm) in reinforced leaktight compartments or GDH compartments or spaces below the reactor. No ECCS, except hydro-accumulators | $3.68 \cdot 10^{-7}$ |
| 4 | Rupture of FC inside reactor cavity with ECCS failure except hydro-accumulators | $2.62 \cdot 10^{-7}$ |
| 5 | Rupture of MCC piping (diameter 50-100 mm) in reinforced leaktight compartments or GDH compartments or spaces below the reactor. ECCS failure, except hydro-accumulators | $9.47 \cdot 10^{-8}$ |
| 6 | Rupture of MCC piping or feedwater supply lines or steam lines outside the ALS compartments. ECCS failure, except hydro-accumulators | $2.46 \cdot 10^{-7}$ |
| 7 | Multiple FC rupture in the reactor cavity. Initiating events – local flow reduction leading FC failure. Transition to multiple FC rupture | $2 \cdot 10^{-9}$ |
| 8 | Rupture of MCC piping (diameter 100–600 mm) in reinforced leaktight compartments or GDH compartments or spaces below the reactor with local flow reduction. ECCS available | $4.98 \cdot 10^{-7}$ |
| 9 | Rupture of MCC piping or feedwater supply lines or steam lines (diameter de =100–600 mm) outside the ALS compartments with additional failures of check valves leading to local flow reduction. ECCS available | $2.65 \cdot 10^{-9}$ |
| 10 | Rupture of MCC piping (diameter 100–900 mm) in reinforced leaktight compartments or GDH compartments or spaces below the reactor, which leads to ALS failure (ALS failure leads to the failure of ECCS pumps, except the main feed water and auxiliary feed water pumps) | $7.82 \cdot 10^{-7}$ |
| 11 | Rupture of MCC piping or steam lines (diameter 100–600 mm) outside the ALS compartments with failure of reactor scram | $5.28 \cdot 10^{-8}$ |
| 12 | Transient with steam relief available (to condenser and/or to ALS). No ECCS, except hydro-accumulators | $3.14 \cdot 10^{-6}$ |
| 13 | Transient with steam relief not available | $3.05 \cdot 10^{-7}$ |
| 14 | GDH blockage + bypass line failure. ECCS available | $1.62 \cdot 10^{-8}$ |
| 15 | Blockage of a single fuel channel leads to FC rupture with transition to multiple FC rupture | $1.1 \cdot 10^{-7}$ |
| 16 | Transient with failure of reactor scram. MCPs are tripped | $5.12 \cdot 10^{-7}$ |
| 17 | Transient with failure of reactor scram. MCPs operate. DS failure due to overpressure (consequential LOCA) | $2.24 \cdot 10^{-8}$ |
| 18 | Transient with long-term failure to maintain sub-criticality after reactor shutdown | $1.67 \cdot 10^{-7}$ |

the Risk Spectrum PSA model for the Ignalina NPP with a frequency above a certain cut-off were collected. The selection of cutoff frequency allows to reduce the total number of PSA sequences that have to be reviewed. The cutoff frequency of $10^{-9}$ was selected. This criterion is more restrictive than the one often used when PSA sequences are reviewed for the identification of BDBAs. A higher cutoff of $10^{-8}$ per year is more typical.

Later, the groups of sequences were based on the sequences having the same event tree functional failures. For example, the transients were separated from LOCAs, transients with failure to shutdown were grouped in to one group and transients with a failure to maintain subcriticality into another group. The long-term failures were separated from short-term failures, the transients with ECCS operation from transients with ECCS failure, etc. The grouping of BDBA allows producing a BDBA list consisting of a reasonable number of BDBA sequence groups. The individual accident sequences within each group will have similar characteristics defined in terms of, for example, accident type (LOCA or transient), LOCA location, power level, ECCS status and fuel damage extent. Each of these groups is a BDBA. A frequency was assigned to each of these groups (it is simply the sum of the frequencies of the individual PSA sequences in the group).

The very small LOCA were excluded from the list, because the consequences of these events cannot be severe accidents (the feedwater supply can compensate the loss of coolant through the break). All events that do not lead to the core damage were excluded also. The detailed list of BDBAs grouped into Categories for internal full power reactor events is presented in Table 5.

In these BDBA categories, the failure of ECCS includes a failure of ECCS pumps to start automatically or by the operator or to start/run the Main Feedwater Pumps and Auxiliary Feedwater Pumps, or a failure to switch to a long-term water source (tanks or recirculation). The frequencies of transients with a failure of the reactor scram are taken from PSA and are based on a previous plant design. After implementation of the Diverse Shutdown System and Additional Holddown System at Unit 2, the frequency of the events related to a failure of the reactor scram or to a failure to maintain subcriticality would fall below $10^{-9}$ per year and could be excluded from the BDBA list.

## 4. DETERMINISTIC ANALYSIS OF BEYOND DESIGN BASIS ACCIDENTS

### 4.1. Analysis of processes in reactor core

On the basis of the vulnerability assessment and an understanding of accident behaviour, as well as of the plant's capabilities of coping with accidents, the next step is to develop accident management strategies. The objectives of the strategies are specified and related to the basic safety functions, e.g., to maintain in fuel rods and fuel channels heat removal by restoring core cooling, to protect the integrity of the MCC by the pressure control, to protect the integrity of the Reactor Cavity (RC) by maintaining pressure and temperature control in RC, to protect the ALS integrity by controlling the pressure in the ALS and to minimize radioactive releases if the ALS fails or a break occurs outside it.

The problems related with maintaining heat removal in the fuel rods and fuel channels and the protection of the integrity of MCC are discussed in this section. The initiating events were selected from the list of BDBA (Table 5). As is shown in the list, the highest CDF ($3.1\cdot10^{-6}$ per year) has the BDBA No. 12. This category contains a transient with ECCS failure, except hydro-accumulators (transients with failures of the core long-term cooling). The most likely initiating event which probably leads to the loss of long-term cooling is station blackout. The station blackout is the loss of normal electrical power supply for local needs with an additional failure to start up all six diesel generators. In the case of loss of electrical power supply the MCPs, pumps of the service water system and feedwater pumps are tripped. Due to failure of diesel generators the long-term subsystem of ECCS is unavailable, i.e. it is impossible to inject water to MCC using the design measures. Analysis [6, 7] showed that ~1.5 hours after the beginning of the accident the heat-up of fuel elements and FC walls starts (see Fig. 4) [8].

Figure 4 shows the behaviour of fuel claddings, fuel channels, and the graphite stack temperatures calculated using the RELAP5/SCDAPSIM code [9] in case of the RBMK-1500 reactor blackout. The fuel channels in RBMK reactors are made from zirconium-niobium alloy (Zr + 2.5% Nb). This alloy becomes plastic at temperature higher as 400 °C. If the pressure inside fuel channels remains near to nominal the FC fails due to ballooning. At the 7–8 MPa pressure in FC the temperature of fuel channel failure is ~650 °C [8]. The dangerous temperature limit increases by decrease of pressure inside FC. Thus, it was assumed during the modeling, that after ~4 hours from the beginning of accident, the operator opens one Steam Relief Valve (SRV) to discharge the steam. This allows to depressurize MCC and to prevent rupture of fuel channels.

After operator intervention the depressurization starts and the processes would continue at low pressure. When the fuel cladding temperature exceeds 800 °C, they would fail because of ballooning. The ballooning occurs because at that time the pressure in the MCC (outside fuel elements) is close to atmospheric while the pressure inside the fuel elements is high. After the temperature of fuel cladding exceeds 900 °C, the cladding oxidation starts. But the fast oxidation process starts only with the MCC depressurization because of SRV opening. It can be explained by the fact that after depressurization the coolant remaining in the cooling circuit starts boiling. The generated steam contacts the hot surfaces of fuel claddings and fuel channel walls, which makes favourable conditions for oxidation. As a result of steam–zirconium reaction hydrogen is generated (Fig. 5). The oxidation and hydrogen generation processes terminate after the pressure in MCC decreases down to atmospheric (Fig. 5). This indicates that there is no steam available in the MCC, thus the steam–zirconium reaction is impossible. The conditions for a fast oxidation of claddings and fuel channels made from a zirconium–niobium alloy are reached after the fuel cladding and fuel channel temperatures exceed 1000–1200 °C (~15 hours after the beginning of the accident). But the oxidation process is slow due to absence of steam in the MCC. Within these first 15 hours the water supply to the fuel channels is required for reactor cooldown. The supply of water in the later phases could lead to a fast steam–zirconium reaction and could accelerate the core damage process.

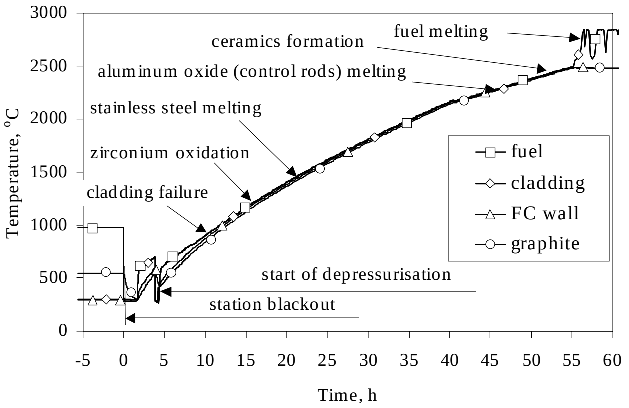


**Fig. 4.** Key events during station blackout

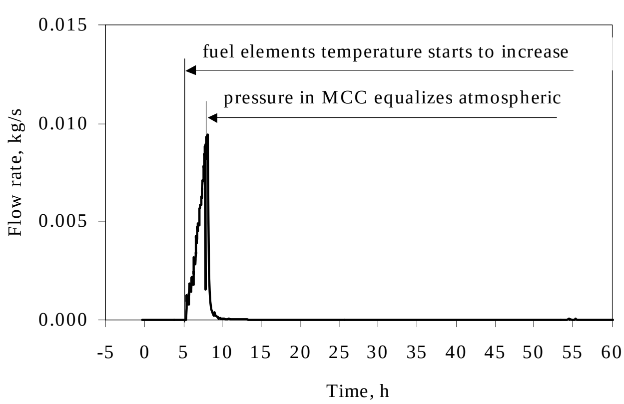


**Fig. 5.** Hydrogen generation rate in one intact average loaded FC

When the fuel claddings and fuel channel temperatures reach 1450 ˚C, the melting of stainless steel grids starts (Fig. 4). Probably that at the same time the fuel channels will fail. Due to the station blackout, the cooling of CPS channels fails as well, leading to heat-up control rods. At a temperature 1930–2050 ˚C and 2330 ˚C the melting of aluminum oxide (control rod claddings) and boron–carbide (control rod elements) starts. The formation of ceramics (U, Zr, $ZrO_2$) starts at 2600 ˚C.

The analysis performed using the RELAP5/SCDAPSIM code shows that the formation of ceramic and fuel melting (melting of $ZrO_2$ and $UO_2$) starts at a low pressure ~50 hours after the beginning of the accident at 2690 ˚C and 2850 ˚C, respectively (Fig. 4). The released melt is likely to have a significant superheat and so will be able to flow to a lower part of the fuel bundle, losing heat to the underlying clad by conduction and convection and to its surroundings mainly by radiation. After some distance the material will be immobilized and re-freeze. While the more volatile fission products are likely to have been released prior to melt relocation, the molten fuel will carry with it the major part of the fission products and thus a heat source. It may be anticipated that at some point the remaining $ZrO_2$ shells (and any undissolved $UO_2$) would collapse, forming a high temperature corium.

The core heat up process in RBMK-type reactors is rather slow because of two factors: 1) high thermal inertia of the graphite stack (there are ~1700 tons of graphite in the reactor), 2) the specific power per core volume of RBMK reactors is ~10 times lower compared to PWR and BWR reactors. The reactor cavity is surrounded by the cylindrical water tanks which play a role of biological shielding. In the case of such a severe accident these tanks will accumulate heat from the core. Thus, although the fuel melting process in reactor core starts, the metal structures which form the reactor cavity remain intact for a long time due to heat dissipation.

As is mentioned in the case of station blackout, there are no possibilities to supply water into the MCC using the design equipment, because due to the failure of diesel generators there is no power supply to the pumps. Alternative possibilities to cool the core were considered for the analysis:

• Direct water supply into the reactor cavity

• Ventilation of DS compartments

• Heat removal using the reactor cavity gas supply system

• Restoration of water supply into the control and protection system rods cooling circuit;

• Depressurization of MCC and water supply from non-regular means (low pressure sources).

The analysis presented in [7, 8] shows that the depressurization of MCC and the following water supply from non-regular means (low pressure sources) to the GDH in the case of the loss of long-term cooling gives considerably better results as compared with the other measures. It was recommended to inject water to the MCC from the hydro-accumulators, deaerators and artesian water supply which has the power supply independent of the plant. This way of reactor core cooling in emergency cases is recommended to include in the RBMK-1500 accident management program.

From the point of view of safety barriers, each fuel channel in an RBMK-type reactor corresponds to a pressure vessel of vessel-type reactors. Thus, the fuel channels are the most important element in the MCC. However, in case of BDBA the integrity of fuel channels could be challenged as they are not as strong as the pressure vessel. Water injection to the fuel channels is recommended as the most effective measure to cool the reactor core. However, during the water supply in overheated fuel channels of an RBMK-type reactor, two specific aspects should be considered:

• the oxidation of fuel claddings and fuel channels at temperatures above 1200 °C would accelerate the core degradation process;

• fast steam generation from the injected water, would increase the pressure in the fuel channels and lead to their rupture.

The first challenge, which is met in case of starting water supply into the overheated reactor core is the steam–zirconium reaction. A significant oxidation of zirconium starts at a temperature of 1000 ˚C and reaches a dangerous level at temperatures above 1200 ˚C. The steam–zirconium reaction is exothermic and at temperatures above 1200 ˚C would accelerate the core degradation process. A specific feature of the RBMK is that not only its fuel assemblies but also the fuel channels are made from a zirconium–niobium alloy. Thus, the mass of components which react with steam at a high temperature is significantly higher it RBMK-type reactors as compared with vessel-type reactors.

To evaluate the possibilities to cool down the overheated reactor core, a simple model of a single fuel channel was created using the RELAP / SCDAPSIM code [9]. The simplified RBMK-1500 model with a single fuel channel where the steam–zirconium reaction processes, fuel rod damage and other severe accident phenomena are modelled, and GDH and DS where the boundary conditions are set are described in [10]. In the present analysis, a close to atmospheric pressure in the fuel channel was assumed. Also, steam removal from the FC was modelled without any restrictions.

The possibilities to cool down the overheated reactor core were evaluated by injecting different amounts of cold water into the fuel channel at different initial fuel cladding and fuel channel wall temperatures. The behaviour of the peak fuel cladding temperature in case of water supply with different flow rates starting at the initial temperature of 1000 ˚C and 1200 ˚C is presented in Figs. 6 and 7. The previous analysis shows that in long-term accident conditions with a loss of cooling, the temperatures of all core components (fuel, fuel claddings, fuel channel walls, graphite) are very similar.

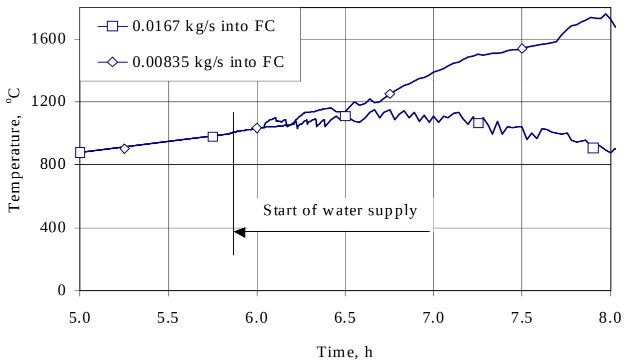


**Fig. 6.** Behaviour of peak fuel cladding temperature in case of water supply with different flow rates into single FC at initial temperature of 1000 ˚C

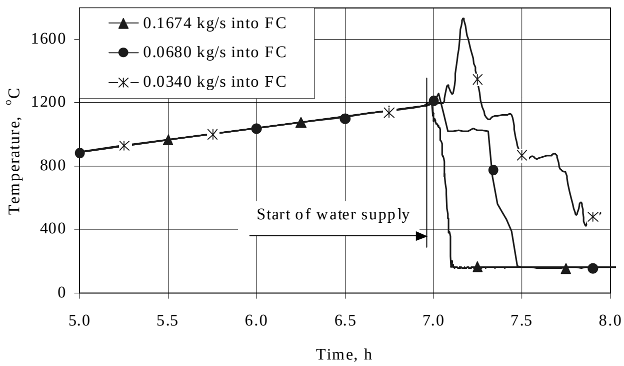


**Fig. 7.** Behaviour of peak fuel cladding temperature in case of water supply with different flow rates into single FC at initial temperature of 1200 ˚C

As is seen in Fig. 6, the water flow rate 0.0167 kg/s into a single FC at the initial temperature of 1000 ˚C allows to cool down the fuel assemblies and fuel channel walls. The decreased water supply with a flow rate 0.00835 kg/s into FC leads to temperature increase and the generation of a high amount of hydrogen. In this case, the peak temperature of 1750 ˚C is reached. Thus, water supply with a flow rate 0.00835 kg/s into FC aggravates the situation (in this situation it is better not to supply water). If water supply is started later when the initial fuel cladding and channel wall temperature are ~1200 ˚C (Fig. 7), the amount of supplied water should be higher. As is shown in the figures, in case of water supply with a flow rate of 0.1674 kg/s, the peak fuel cladding and channel wall temperatures start to decrease straightaway after starting water supply. This means that such amount of water allows to stop the steam–zirconium reaction and rapidly to cool down the fuel assembly and fuel channel wall.

If a smaller amount of water is supplied, the peak cladding and channel wall temperatures initially increase, later stabilizing at a level 1000–1100 ˚C for 10–15 minutes before starting to decrease (Fig. 7). A more detailed analysis shows that the top part of the fuel rods is overheated due to the exothermal steam–zirconium reaction. The heat from the top part of the rod is transferred to its bottom part, leading to fuel temperature stabilization just over 1000–1100 ˚C. The temperature increase in the top part of the fuel channel up to 1750 ˚C leads to fuel cladding damage, fuel fragmentation and flow blockage. Thus, to cool down the fuel assemblies and fuel channels in overheated conditions (t > 1200 ˚C), a high amount of supplied water should be used. In the opposite case – water supply with low flow rates – this will only speed up the core degradation process. Analysis performed using the RELAP/SCDAPSIM code shows that if the initial cladding and FC wall temperature is above 1250 ˚C, water supply with any flow rate is not recommended, because it leads to an uncontrolled steam–zirconium reaction and a damage of the reactor core components.

Based on this analysis, the minimum amount of water supplied into the reactor for cooling the core in respect to the steam–zirconium reaction was created (Fig. 8). The amount of supplied water was calculated, considering that there are in total 1661 fuel channels in the reactor core of the RBMK-1500. As is shown in this figure, if the fuel cladding temperature is below 1000 ˚C, water supply is not limited. The recommended water flow rate is 100 $m^3$/h. The supply of water with a lower flow rate allows to decrease the reactor overheating rate. In the temperature interval 1000–1250 ˚C, the flow rate of supplied water should be 100–1000 $m^3$/h (according to Fig. 8). The supply of water with a lower than prescribed flow rate leads to generation of hydrogen, temperature increase and is not recommended. At a temperature above 1250 ˚C, when the steam–zirconium reaction is fast, the supply of water is not recommended. This recommendation regarding water supply to the overheated RBMK reactor core is different in vessel-type reactors. In these reactors, due to a comparably smaller amount of zirconium, the supply of water is recommended irrespective of the fuel cladding temperature, because the positive effect of the cooling prevailing negative effect due to additional heat and hydrogen generation in this type reactors.

The second challenge in the case of starting water supply into the overheated reactor core is a sharp pressure increase in the fuel channels. Water supply into the overheated fuel channels, especially when the fuel

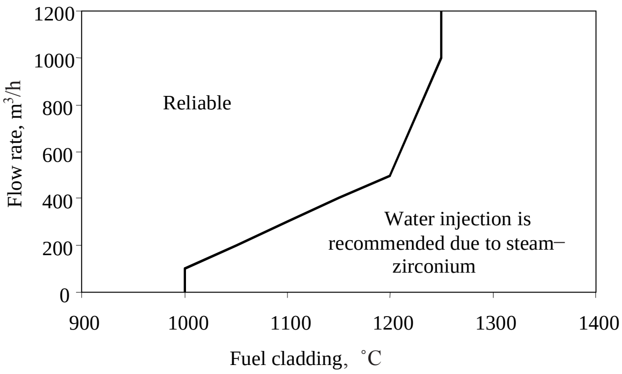


**Fig. 8.** Minimum supplied water flow rate for reactor cooling in respect to steam–zirconium reaction

cladding temperature is 1000 °C or higher, leads to generation of huge amounts of steam. There are long paths for steam evacuation from fuel channels through the steam relief valves, and due to fast steam generation the pressure increases in fuel channels. It was mentioned that the fuel channels in RBMK-type reactors are not so strong as the pressure vessel. The fuel channels in the RBMK can fail due to pressure increase at a high channel wall temperature. Thus, pressure increase in the fuel channel should not exceed 4 MPa (the pressure which can be supported by the graphite block [11]). The pressure peaks can be controlled by opening steam relief valves and removing the generated steam.

The steam generation and pressure increase in fuel channels due to water injection was modelled using the MCC model of RBMK-1500, developed employing the RELAP5 code. This model is presented in [7, 12, 13]. The generation of additional heat due to the exothermic steam–zirconium reaction was taken into account in this model. An example of reactor core cooling by depressurization of MCC and a start of water supply at a fuel cladding temperature of 1000 °C are presented below.

As it is shown in Fig. 9, the heatup of the core components in the case of station blackout in RBMK-1500 starts after ~1.5 h. Within this time span the pressure in the MCC is maintained by automatic operation of the SRV (Fig. 10). After 3.9 h from the reactor shutdown, when the calculated peak fuel cladding temperature in the fuel channel with an average initial power reaches 700 °C the action of the operator (manual opening of one SRV with a capacity of 720 t/h) is assumed. After this action, pressure decrease in MCC starts. It leads to a short-term decrease of temperatures. However, in ~50 minutes after opening the valve, a repeated heatup of the core components starts (Fig. 9). It was assumed in the modeling that at 6.5 h the operator closes the SRV to maintain the excess pressure in the MCC at 0.235 MPa (Fig. 11). When the peak fuel cladding temperature reaches 1000 °C, the injection of water into fuel channels through GDH with the total capacity 500 $m^3/h$ (0.0837 kg/s of water into each fuel channel) was assumed. As is shown in Fig. 8, the minimum amount of supplied water should be no less than 100 $m^3/h$ (0.0167 kg/s into each FC). At the same time, several SRVs with a total steam flow rate capacity of 2500 t/h through each valve at a nominal pressure (7 MPa) were opened

When the channel pipe wall is heated up at a high pressure in the MCC, it can be deformed in a radial direction (i.e. to be ballooned) up to a contact with the graphite block. It was assumed during the modelling that plastic deformations of FC start 4 h following the beginning of the accident because of a high pressure in FC and a high temperature of FC walls. The huge amount of heat accumulated in the graphite is transferred through fuel channels to the coolant. This leads to the generation of a significant amount of steam, which exceeds the amount of supplied water (Fig. 10). The generated steam is evacuated into condensing pools through the steam–water piping, steam lines and opened SRVs. Because the flow path for steam removal is very long, the pressure peak after starting the water supply appears in the fuel channels (Fig. 11). Because this pressure peak is below 4 MPa (pressure limit when the integrity of fuel channels is maintained due to support of graphite blocks), the fuel channels remain intact.

Based on the described analysis, the dependence of water injection rate into the MCC on the fuel cladding temperature and the number of open steam relief valves was revealed (Fig. 12).

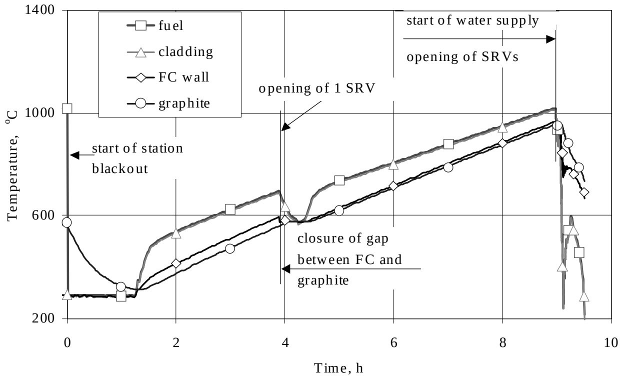


**Fig. 9.** Temperatures of fuel, fuel element cladding, FC walls and graphite

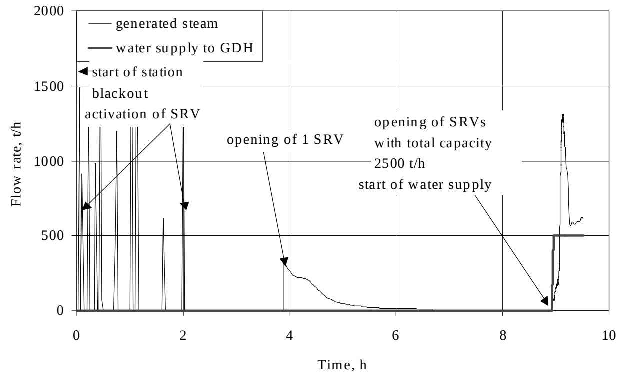


**Fig. 10.** Water flow rate into the reactor and flow rate of the steam discharged through SRV

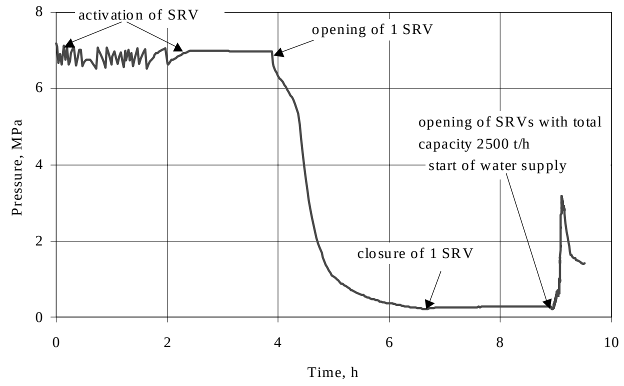


**Fig. 11.** Pressure inside FC

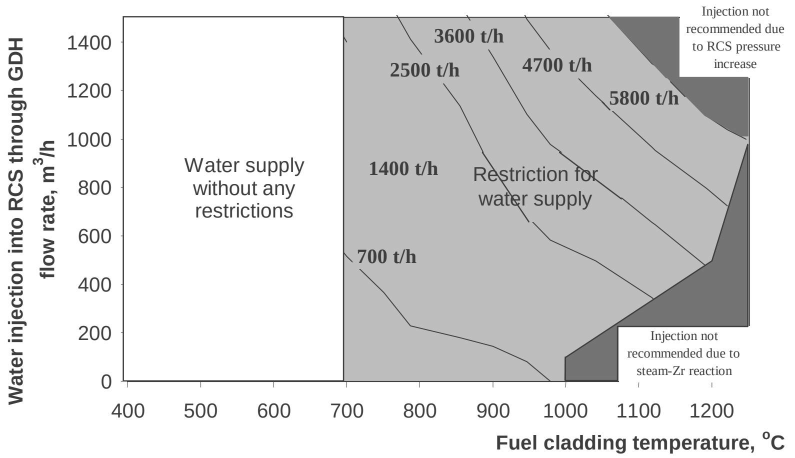


**Fig. 12.** Water injection flow rate to MCC depending on the fuel cladding temperature and number of open steam relief valves (700 t/h – 5800 t/h – steam flow rate through SRV at nominal pressure in DS)

There is no danger of FC break if the FC wall temperature is less than 650 ˚C (fuel cladding temperature <700 ˚C), thus in this temperature region the supply of water proceeds without any restrictions. As is seen in Fig. 12, in the temperature region 700–1250 ˚C it is necessary to pay attention to two conditions:

• the flow rate of water should ensure that increase of pressure in the FC does to break the channels (pressure in MCC should not exceed 4 MPa);

• the flow rate of water should be no less than that required to avoid steam–zirconium reaction.

These specific restrictions must be followed in the case of loss of long-term cooling if the MCC integrity has to be maintained. If the pressure in FC does not increase under water supply into the channels (in case of large breaks of MCC elements at a not isolated leak), the flow rate of supplied water can be higher than that specified in Fig. 12. In this case, Fig. 8 should be used.

### 4.2. Analysis of processes in confinement

At the Ignalina NPP, there is a special system of hermetic compartments, which performs a function of confinement; usually it is referred to as the Accident Localisation System. The ALS is a pressure suppression type confinement, i.e. accident-generated steam is condensed in special condensing pools. A detailed description of the ALS is presented in [14].

The following compartments represent the confinement of the Ignalina NPP:

• Compartments before the condensing pools:

– Reinforced-leaktight Compartments and Bottom Steam Reception Chambers in both ALS towers, with total volume of 20600 m$^3$;

– Compartments of Group Distribution Headers, the volume 4200 m$^3$;

– Top Steam Reception Chambers (TSRC) in both ALS towers, 280 m$^3$ each;

– Reactor Cavity together with the Reactor Cavity Venting System (RCVS), volume 335 m$^3$.

• Compartments behind the condensing pools with the total volume of 14410 m$^3$ in each ALS tower.

Condensing pools of the Ignalina NPP confinement are located in two almost identical ALS towers (five pools in each tower positioned vertically). Bottom pools (1–4 level) are activated in case of LOCA in reinforced-leaktight compartments or in GDH compartments or in case of multiple fuel channel rupture, when membrane safety devices of RCVS open to reinforced-leaktight compartments. The fifth pool is activated in case of SRV opening or fuel channel rupture in the reactor cavity. The total mass of water maintained in the 1–4 pools of each tower is 1400 m$^3$. In the fifth pool the water reserve of 330 m$^3$ is maintained.

Usually, the containments of NPPs with vessel-type reactors have only one condensing pool ~7 m deep. This pool is used for ECCS pumps as well. The Ignalina NPP has several condensing pools located one above another. This structure is similar to the confinement of VVER-440 reactors. The Steam Distribution Devices (SDD) at the Ignalina NPP differ from the other NPPs:

• the distance from the edge of the SDD vent pipe is less (only ~0.1 m);

• the water level in condensing pools is 0.95–1.05 m and the submergence of SDD vent pipes is ~1 m and at the Ignalina NPP it is less.

The Condensing Tray Cooling System (CTCS) is used to cool water in condensing pools. The CTCS takes water from the Hot Condensate Chamber (HCC), cools it in the heat exchangers and supplies back to the condensing pools. Water from the pools overflows back to the HCC. The CTCS simultaneously supplies water to sprays located at the top of an ALS tower. The CTCS supplies water either to 1–4 condensing pool, or to the fifth pool. In case of a multiple fuel channel rupture it supplies water to all pools simultaneously. The heat exchangers are cooled by service water.

The major feature of ALS at the Ignalina NPP and the difference from the other type containments is that in the initial phase of the accident the clean air, which initially fills the compartments after the condensing pools, is released to the environment. This feature allows a less strength of the building structures. The design pressures of the confinement parts are:

• reinforced-leaktight compartments – 400 kPa (absolute);

• bottom steam reception chamber – 200 kPa (absolute);

• compartments after the condensing pools – 180 kPa (absolute);

• GDH compartments and TSRC – 180 kPa (absolute);

• reactor cavity – 314 kPa (absolute).

The usual design pressures for the other pressure suppression type containments are ~5 bar (500 kPa), for the "dry" containments design pressure being ~7 bar (700 kPa) [15].

During the normal operation, several ventilation systems provide the cooling of the ALS atmosphere to maintain a temperature of 50 ˚C in the compartments before the condensing pools and 35 ˚C in the compartments behind the condensing pools. The exhaust ventilation systems maintain a slight underpressure in the compartments to avoid any release of radioactive fission products to the environment. These ventilation systems are equipped with iodine and aerosol filters. In case of a LOCA inside the ALS compartments, all ventilation fans are stopped and the double hermetic valves on the ventilation lines are closed automatically.

In order to investigate the ALS response in case of station blackout, analysis was performed using the COCOSYS code. COCOSYS is a lumped-parameter code for the comprehensive simulation of all relevant phenomena, processes and plant states during severe accidents in the containment of light water reactors, also covering the design basis accidents [16].

The results of the performed analysis are presented in Figs. 13–15 The initial pressure peak in the TSRC is observed in a few seconds after the first opening of the SRV. This peak is just 130 kPa, i.e. it is much below the design pressure of 180 kPa. After this initial peak, the pressure in the TSRC decreases to ~111 kPa and remains almost constant. The pressure in the compartments behind the condensing pools reaches a level of the tip-up hatches opening (102 kPa) and remains almost constant. The pressure difference between the TSRC and the compartments behind the condensing pools corresponds to the level of vent pipe submergence in the fifth condensing pool, i.e. ~1 m of water column (~10 kPa). Such low pressure and such pressure difference remains until the ALS is isolated from the environment by flooding the last section of the Gas Delay Chamber (GDC).

The section is flooded either by the signal of pressure increase in the reinforced-leaktight compartments or the signal of radiation level increase in the GDC. In case of station blackout, steam is released to ALS towers and the reinforced-leaktight compartments are not affected, i.e. there could be no pressure increase signal early in the accident. Thus, the only signal for ALS isolation is the radiation level increase in the GDC. But there is a large uncertainty when this signal will be generated, because it would depend on the radiation level of the coolant in the MCC. Thus, this signal could be generated early in the accident or only after some hours when the fuel cladding ruptures occur. In our analysis, we assumed that the signal is generated early in the accident and the ALS is isolated from the environment within 300 s after the start of the transient. After the ALS is isolated from the environment, the pressure in it starts rising due to the fact that the condensing pools are not able to condense the accident-generated steam and reach the boiling temperature, because the CTCS is unavailable due to the loss of power supply (see Fig. 14). After the tipup hatches are closed, the pressure in the ALS towers starts increasing fast. The pressure increase in the ALS towers causes a water flow from the four lower condensing pools to the BSRC and reinforced leaktight compartments (see water level in Fig. 15). This water flow to the BSRC causes the pressure increase in the reinforced-leaktight compartments (curve PBB9) and later, when the lower condensing pools are emptied, the pressure in the whole ALS is nearly the same. After the emptying of the condensing pools the pressure in the ALS compartments starts increasing simultaneously, and after 864 s the blow-down hatches in the ALS tower are damaged, i.e. the pressure reaches 180 kPa. This means that if the safety systems related to ALS are failed, then the ALS integrity would be violated due to overpressure in less than 15 minutes.

The calculated water temperature in the fifth condensing pools of both ALS towers is presented in Fig. 14. The temperature in the fifth pool of the right ALS

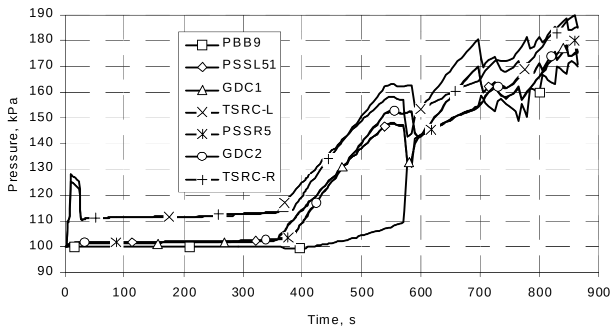


**Fig. 13.** Pressure in the ALS compartments

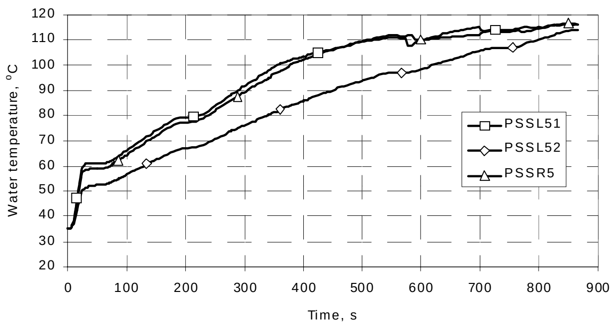


**Fig. 14.** Water temperature in the condensing pools

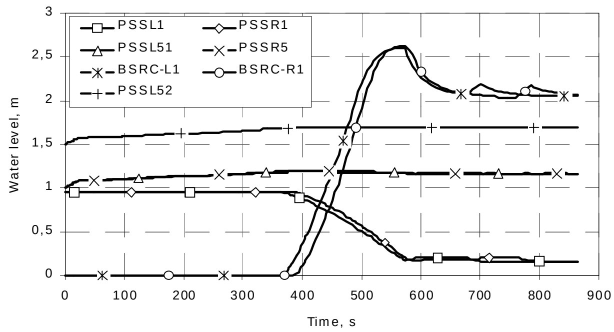


**Fig. 15.** Water level in the condensing pools and BSRC

tower (curve PSSR5) increases at almost the same rate as in the parts of the fifth pool in the left ALS tower (curve PSSL51). Another part of the left ALS tower (curve PSSL52) heats up slower, because it contained more water and a less number of steam distribution devices connect this part of the pool with the TSRC. This part of the condensing pool is deeper, because it is used for the condensation of steam coming from the reactor cavity in the case of the fuel channel rupture.

After 300 s, when the tip-up hatches are closed, the water temperature in the fifth condensing pool of the right ALS tower and in the parts of the pool in the left tower reaches 90 ˚C, i.e the pool is close to boiling. After 400 s these water volumes start boiling. The water temperature in the deeper part of the pool in the left ALS tower is ~15 ˚C lower, but this difference decreases and all the water in the fifth condensing pool of both ALS towers starts boiling, and thus the pressure suppression function of the condensing pools is lost.

The water level in different water volumes in the ALS is shown in Fig. 15. The water level in the fifth condensing pools increases due to released steam condensation to the level of the overflow gaps and remains constant during the whole transient sequence. The water level in the node PSSL52 is higher, because it represents the deeper part of the fifth condensing pool in the left ALS tower. The water surface level in both parts of the pool is the same. The water level in the lower condensing pools remains at the initial level until the pressure increase in the compartments behind the condensing pools remains small, i.e. corresponds to an opening pressure of tip-up hatches. But when the ALS is isolated from the environment and the condensing pool becomes boiling, the pressure in the Gas Delay Chamber increases and pushes the water from the lower condensing pools to the BSRC. This process starts in ~370 s, and after 570 s the lower condensing pools are empty. The water from the lower condensing pools through the steam distribution devices flows to the BSRC, and when the water level in the BSRC increases to 2 m the water starts overflowing to the reinforced leaktight compartments.

Results of the present study imply that if in the case of blackout the ALS is isolated from the environment early in the accident, the integrity of the ALS would be lost, i.e. blow-down hatches in the right ALS tower would rupture after 864 s (~14 min). In case of blackout the steam is released from the MCC to both ALS towers simultaneously, and the behaviour of pressure and temperature in both towers is almost the same. Nevertheless, it should be noted that the isolation of the ALS towers from the environment is independent, i.e. only a tower with an increased level of radiation is isolated. Thus, if only one tower is isolated from the environment, the steam released to this tower would increase the pressure in it and push the water from the lower condensing pools and then would enter the lower condensing pools of the opposite (not isolated) tower where the steam would be condensed and radioactive fission products would be scrubbed. Hence it follows that in order to reduce the radiation release to the environment there should be developed a strategy to enable discharging steam from the MCC to one ALS tower which would be isolated from the environment. The other tower would be not isolated until the level of radiation in this tower is less than required for isolation. Such configuration would allow to make the flow path of radioactive fission products longer and to utilise the mass of water in the lower condensing pools. Such approach would decrease the possibility of radiation release to the environment.

## 5. CONCLUSIONS

The paper presents a discussion of the probabilistic and deterministic analyses of the BDBA in the RBMK-1500 and the role of PSA in the development of strategies for SAMG.

The results of PSA-1 and PSA-2 for the Ignalina NPP, the impact of external events on the Ignalina NPP safety are discussed.

Different possibilities to cool down the core of the RBMK-1500 reactor in cases of BDBA were analysed. The station blackout case was used as an example of BDBA, because the probability of such accident is of the top of the BDBA list. From a few possible measures, the most effective is depressurization of the MCC and water supply from non-regular low pressure sources (the so-called “bleed and feed” strategy). The challenges that are met when injecting water to the overheated core are used (steam–zirconium reaction, pressure increase in fuel channels due to a fast steam generation process).

Also, the processes inside the RBMK-1500 confinement were discussed. Analysis of these processes and their challenges are presented. An important proposal for the accident management strategy, which would enhance the capabilities of ALS to prevent release of radioactive fission products, is presented.

### Nomenclature

| | |
|---|---|
| AFSS | Auxiliary Feedwater Supply System |
| ALS | Accident Localisation System |
| AM | Accident Management |
| APET | Accident Progression Event Trees |
| ATWS | Anticipated Transient without Scram |
| BDBA | Beyond Design Basis Accident |
| BSRC | Bottom Steam Reception Chamber |
| BWR | Boiling Water Reactor |
| CCI | Common Cause Initiator |
| CDF | Core Damage Frequency |
| CPS | Control Protection System |
| CTCS | Condenser Tray Cooling System |
| DS | Drum Separator |
| ECCS | Emergency Core Cooling System |
| EOP | Emergency Operating Procedure |

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|---|---|
| EPSS | Emergency Power Supply System |
| FC | Fuel Channel |
| GDH | Group Distribution Header |
| HCC | Hot Condensate Chamber |
| IAEA | International Atomic Energy Agency |
| ICC | Intermediate Cooling Circuit |
| IE | Initiating Event |
| INPP | Ignalina Nuclear Power Plant |
| LOCA | Loss-of-Coolant Accident |
| MCC | Main Cooling Circuit |
| MCP | Main Cooling Pump |
| MDL | Maximum Design Limits |
| NPP | Nuclear Power Plant |
| PDS | Plant Damage States |
| PWR | Pressurised Water Reactor |
| PSA | Probabilistic Safety Assessment |
| RBMK | Russian abbreviation for "Large-power channel-type reactor" |
| RC | Reactor Cavity |
| RCVS | Reactor Cavity Venting System |
| SAMG | Severe Accident Management Guidelines |
| SDD | Steam Distribution Device |
| SRV | Steam Relief Valve |
| SOL | Safe Operation Limits |
| TSRC | Top Steam Reception Chamber |

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**Eugenijus Ušpuras, Algirdas Kaliatka, Juozas Augutis, Sigitas Rimkevičius, Egidijus Urbonavičius, Vytis Kopustinskas**

**TIKIMYBINĖ IR DETERMINISTINĖ NEPROJEKTINIŲ AVARIJŲ RBMK-1500 REAKTORIUOSE ANALIZĖ**

S a n t r a u k a

Projektinės avarijos RBMK-1500 yra detaliai išnagrinėtos. Remiantis šių analizių rezultatais sukurtos ir įdiegtos papildomos Ignalinos AE saugą garantuojančios priemonės ir atlikta daugelis modifikacijų. Toliau keliant jėgainės saugos lygį reikia sukurti avarinių situacijų instrukcijas, įgalinančias valdyti neprojektinių avarijų eigą, siekiant išvengti aktyviosios zonos pažeidimų arba sušvelninti sunkiųjų avarijų pasekmes. Siekiant sukurti tokias instrukcijas reikia sukaupti ir susisteminti informaciją apie neprojektines avarijas galinčius sukelti pradinius įvykius, tokių avarijų metu vykstančius reiškinius bei galimybes valdyti avarijos eigą. Šiai informacijai sukaupti reikėtų atlikti daugybės scenarijų deterministinius skaičiavimus, tačiau panaudojus pirmojo ir antrojo lygio tikimybinės saugos analizės rezultatus galima identifikuoti svarbiausius.

Straipsnyje pateikti Ignalinos AE pirmojo ir antrojo lygio tikimybinio saugos įvertinimo rezultatai ir parodyta, kaip tikimybinių tyrimų rezultatai buvo panaudoti sudarant neprojektinių avarijų sąrašą. Aptarti svarbiausi reiškiniai valdant neprojektines avarijas RBMK tipo jėgainėse. Parodyta, kokie procesai vyksta perkaitusioje reaktoriaus aktyviojoje zonoje, ir aptartos rekomenduojamos tokios būklės reaktoriaus aušinimo priemonės. Taip pat aptarti reiškiniai, vykstantys neprojektinių avarijų metu reaktorių ir aušinimo kontūrą gaubiančiose patalpose, ir rekomenduojamos priemonės tiems procesams valdyti (vandenilio koncentracijos mažinimas, radioaktyviųjų skilimo produktų surišimas bei nusodinimas ir pan.).

**Raktažodžiai**: neprojektinė avarija, RBMK reaktorius, avarijų valdymas, tikimybinė saugos analizė

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## ВЕРОЯТНОСТНЫЙ И ДЕТЕРМИНИСТИЧЕСКИЙ АНАЛИЗ ЗАПРОЕКТНЫХ АВАРИЙ ДЛЯ РЕАКТОРОВ РБМК-1500

Резюме

В настоящее время проектные аварии для реакторов РБМК-1500 довольно детально исследованы. По результатам таких исследований разработаны и внедрены дополнительные средства, обеспечивающие безопасность Игналинской АЭС, также внедрен ряд модификаций. Для дальнейшего повышения безопасности АЭС следует разработать инструкции, которые позволили бы управлять запроектными авариями, т. е. предотвратить повреждения активной зоны реактора или смягчить последствия тяжелых аварий. Во-первых, необходимо собрать и систематизировать информацию об исходных событиях, которые могут вызвать запроектные аварии, о возникающих при этом явлениях и о возможностях управлять авариями. Для сбора такой информации потребовалось бы множество детерминистических расчетов, но на основании результатов вероятностного анализа первого и второго уровней можно определить наиболее важные из них. В статье представлены результаты вероятностного анализа первого и второго уровней, показано, как они были использованы при разработке перечня запроектных аварий.

Рассмотрены явления, наиболее важные при управлении запроектными авариями на АЭС с реакторами РБМК. Показано, какие процессы протекают в перегретой активной зоне, обсуждены способы охлаждения реактора, рекомендуемые в такой ситуации. Описаны явления во время запроектных аварий в контуре охлаждения реактора и в защитной оболочке, а также средства, рекомендуемые для управления этими процессами (понижение концентрации водорода, осаждение продуктов деления и т. д.).

**Ключевые слова:** запроектная авария, реактор РБМК, управление авариями, вероятностный анализ безопасности