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An approach to estimate radioactive release frequency from Ignalina RBMK-1500 reactor in Lithuania

Key-words

Ignalina nuclear power plant, nuclear safety, radioactive release, probabilistic safety assessment.

Słowa kluczowe

Elektrownia jądrowa Ignalina, bezpieczeństwo jądrowe, uwolnienie substancji radioaktywnych, probabilistyczna ocena bezpieczeństwa.

Summary

The paper presents an approach to estimate the radioactive release frequency from Ignalina nuclear power plant in Lithuania. The study was completed within the frame of Barselina project, initiated in 1991 as a multilateral co-operation between Lithuania, Russia and Sweden with the long-range objective to establish common perspectives and unified bases for assessment of severe accident risks and needs for remedial measures for the RBMK type reactors. The paper presents the study results and discusses a number of future development efforts.

1. Introduction

Probabilistic safety assessment (PSA) is a commonly used technique in nuclear industry to estimate risks from nuclear power plants. It is in general di-

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vided into three levels. The level 1 PSA takes into account all possible internal and external initiating events that could cause a reactor core damage and estimates the core damage frequency (CDF). The level 2 PSA takes level 1 results as an input and estimates frequency and magnitude of radioactive release outside the nuclear site. The level 3 PSA estimates public and environmental risks from the nuclear power plant. Presently, level 1 PSA methodology is the most developed and matured, while level 2 PSA may still contain a lot of epistemic uncertainties. The level 3 PSA studies are completed only for few plants worldwide and there is no much research being done in this area.

The Ignalina NPP operates two RBMK-1500 channel-type reactors, commissioned in 1984 and 1987 [1]. The first outline of Ignalina NPP RBMK-1500 reactor level 2 PSA study was completed in 1997 on the basis of available knowledge at that time. Since then updates have been made at the plant and also of the level 1 PSA, which motivated renewed efforts to elaborate level 2 PSA. This was performed in a joint project between Lithuania (INPP and LEI) and Sweden (SiP, ES-consult and RELCON) in "Accident management, Consequence Mitigation and PSA Level 2" project.

The general objectives of the "Accident management, Consequence Mitigation and PSA Level 2" project are to demonstrate plant features and accident management capabilities in a case of core damage. These general objectives can be further detailed as follows:

- Compilation of available knowledge about phenomena, release barriers and human actions structured in a PSA oriented way aiming at producing a description of the accident scenarios ending in a release of radioactivity.
- Identification of risk dominating sequences and their characteristics (frequency, release size, timing, release speed etc).
- Evaluation of operating procedures, emergency operating instructions, emergency plans and other available procedures with regard to the consequences resulting from a possible release of radioactivity.
- Prioritisation of continued work and research activities.
- Evaluation of uncertainties in assumptions and the model.

The objective of the PSA level 2 analysis is to expand the level 1 PSA into calculating preliminary source terms and their frequencies, *i.e.* the release of radioactive material to the environment for different types of accidents and to quantify the frequency for various source term levels.

The scope of the project is limited to the initiating events and operating modes described by the phase 5 level 1 model. The scope of this study is to expand the level 1 analysis in three directions:

1. Analysis of the confinement (accident localisation system) function in deterministic as well as probabilistic aspects.
2. Expansion of the level 1 analysis taking into account initiating events at a lower probability level, beyond the design basis.

3. Expansion of the number of hazard states and the calculation of source terms to a more refined classification of releases.

Still limitations in the scope exist, basically the same limitations that are given by the level 1 PSA. Hence, external initiating events, such as external flooding, earthquake and crashing aircraft, are excluded.

2. Interface Level 1/ Level2

2.1. Level 1 core hazard states

The following core hazard states after the initiating event were defined for the Level 1 PSA study of the Ignalina NPP:

– Safe state (S): No exceeding of Maximum Design Limits (MDL) or exceeding of Safety Operating 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 high pressure);
– Violation (V1, V2): exceeding of MDL in no more than 3 FC leading to or caused by 1-9 FC ruptures at high pressure;
– Damage (D): exceeding of MDL in 4-90 FC (accompanied by ruptures of no more than 9 FC at high pressure);
– Severe damage (A): exceeding of MDL in more than 90 FC or a rupture of more than 9 FC.

Also time factors were determined: S – short term (0 - 2 min), I – intermediate term (2 - 60 min) and L – long term (1-24 hours). In fact, the Level 1 PSA core damage frequency (CDF) calculations combine the states ‘D’ and ‘A’ into the united core damage state.

2.2. Level 1 PSA adaptation for Level 2 purposes

For the Level 2 purposes more detailed consequences were required. For each level 1 accident sequence, the detailed consequence (AS, AI, AL, DS, DI, DL, VS, VI, VL) was assigned, consisting of damage severity and timing factors (e.g. AS means severe core damage in 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.

2.3. Identification of Plant Damage States (PDSs)

Totally 16 PDSs were defined. Each PDS was named by the following format: Y-Z – where Y represents Level 1 hazard state and Z – abbreviation of the Level 1 initiating event.

Seven PDSs were selected for the detailed APET (Accident Progression Event Tree) analysis, as 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 accident management (AM) scheme.

Table 1. List of analysed Plant Damage States for RBMK-1500 reactor
Tabela 1. Lista analizowanych stanów awaryjnych reaktora RBMK-1500

PDS	Freq.	Description	Comments
A-HP	2.6E-06	Accident in long term (AL) after transients and area events with high pressure	Accident mitigation possible. In case of successful mitigation it is possible to reduce core damage consequences.
D-LLZ1	3.4E-07	Damage in intermediate term after LLOCA in zone 1	Accident management actions are not possible due to short time window. APET analysis should be performed. Contribution from other LOCAs is negligible.
V-LLZ1	3.4E-07	Violation after LLOCA in zone 1	AM actions are not possible due to short time window. APET analysis should be performed.
V-MLZ1	1.7E-07	Violation after MLOCA in zone 1	AM actions are not possible due to short time window. APET analysis should be performed.
V-MLZ2	1.1E-06	Violation after MLOCA in zone 2	AM actions are not possible due to short time window. APET analysis should be performed.
S-LLZ1	4.0E-05	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 this PDS.
S-MLZ1	1.7E-04	Safe state after Medium 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 this PDS.
S-MLZ2	2.0E-05	Safe state after medium LOCA in zone 2	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 this PDS.

In Level 2 analysis for RBMK-1500 reactor, it is assumed that accident sequences after all initiating events resulting in severe core damage in short and intermediate terms (AS, AI) lead directly to the highest release category without accident localisation system (ALS) protection.

Accidents in long term (AL) were grouped into two PDSs, based on the pressure criteria in the primary system. Basically, after all LOCA (Loss of

Coolant Accident) events, pressure in the MCC (Main Circulation Circuit) will be low and for these events PDS – A-LP is defined. In case of transients and area 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 – rupture of several channels with the highest power. However, this scenario needs deterministic justification and further investigations. Long term cooling failure is subject to accident management actions, which should be justified by deterministic calculations. Deterministic investigations of the long-term cooling are further presented in brief.

The hazard state category 'Damage' is mainly subject to cut-off value screening criteria (10^{-8}). Only one PDS – D-LLZ1 is defined for the LLOCA in zone 1 resulting in DS and DI hazard states. Transients, blockages and area events do not contribute to the damage category.

The hazard state category 'Violation' is presented by several PDSs. The PDS - V-BLC corresponds to violation of the core due to a group distribution header blockage or channel rupture. The rest of the PDSs - V-LLZ1, V-MLZ1, V-MLZ2 and V-LZ4 are defined for the reactor core violation after large and medium LOCAs in zones 1, 2 and 4 respectively. V-SLZ123 is defined to analyze violation due to small LOCA in zones 1-3.

The hazard state category 'safe' in Level 1 PSA was assumed as successful state of the reactor after the initiating event. However, this is not the case for all IEs in the Level 2 analysis. Safe state for transients and area events mean a safe state for both levels of analysis. But in case of LOCAs, although the safe state is reached, large amount of radioactive coolant is released to the ALS and can challenge structural integrity of the ALS. In case of ALS damage the reactor safety systems can be affected and even a accident state can be reached. APET analysis is essential for these PDSs.

It is assumed that a safe state after small LOCA in all zones, even in the zone 4, which is located outside the ALS, will end with the small release and environmental consequences. Five PDSs were defined for safe hazard category, depending on the location and size of the LOCAs.

3. Release categorisation and source term

Calculations of the radionuclides amount in the reactor core is performed by taking into account dependence on burn up (with 0.5, 5, 10, 15, 20, 24 MWd/kg) and applying a program sequence SAS-2 from the package SCALE 4.3. The amount of radionuclides accumulated in the reactor, and after 15 and 120 minutes after shutdown of the reactor is also calculated. The total amount of radionuclides (transuranium elements, fission products and corrosion elements) inside the reactor core was calculated by taking into account the real distribution of the fuel assemblies burn up and power of INPP Unit 2 and was estimated to be approximately 1.25E+08 Ci of ^{131}I. The amount of radionuclides in the gap between fuel and cladding was calculated using the NIKIET calculation results by their own developed RETI code.

The release of radioactive FP (Fission Product) from the fuel is governed by the fuel temperature and the number of damaged fuel elements. Only power generation and the residual power can generate the damaging fuel temperatures. Note that during Chernobyl accident, the power during initial phase of the accident was far above the nominal power – this could not occur in the current reactor design.

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

– Direct release: the release from confinement zone 4 or zone 5 to the environment without getting into the ALS or from zone 3 if the top metal structure of the reactor is lifted.
– Through ALS: the 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 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.

For the evaluation of the FP deposition in their transport way through the ALS compartments, Filter Factors were introduced. If two compartments (starting and ending) are assumed on FP flow path then the Filter Factor is defined as a ratio of the total amount of FP in the starting compartment and the total amount of FP in the ending compartment.

The filter factor is a very important parameter defining the release of radioactive FP to the environment and radiological impact to public. The use of filter factors in this study is generic and is relevant only in long term. In practice the filter factors vary with time and the variation is scenario specific. The retention of radioactivity in condensing pools, due to sprays and due to natural deposition is discussed and comparison is made with the Chernobyl accident. The values of the proposed filter factors are based on the experience of severe accident analysis for NPP with western type reactors and the experience of Chernobyl accident. The release estimate during Violation type of accidents is based on the event in Leningrad NPP unit 3 when the rupture of a fuel channel occurred. There is a very high filter factor showing efficient retention of FP in condensing pool and the ALS.

No calculations were performed to calculate the source term for different hazard states. The source terms are rough expert estimates.

On the basis of the PSA level 1 hazard states three release categories are defined. Each release category considers PSA level 1 hazard state and filter factor of the appropriate release path. Release categories are presented in Table 2.

Table 2. Release categories
Tabela 2. Kategorie uwolnień

	Release, % of core inventory	INES* scale
INPP1	<0.003	Low release INES 1-3
INPP2	0.003-0.2	Medium release INES 4-6
INPP3	>0.2	Large release INES 7

*International Nuclear Event Scale
of IAEA (International Atomic Energy Agency)

Low release is expected for INPP1 release category. This category comprises accident sequences of S, V and D hazard states with FP transport path Through ALS or S and V hazard states with FP transport path Limited ALS. For this category up to 3500 Ci of ^{131}I is released from the plant to the environment. The accident with such release could be rated as INES-3, which corresponds to small consequences and off-site protective actions are generally unlikely.

Large release is expected for release category INPP2. This category comprises such accident sequences of S and V hazard states with FP transport path Direct release and D hazard state with transport path – Limited ALS. For this category up to 34100 Ci of ^{131}I is released from the plant to the environment. The accident with such release could be rated as INES-5. As a result of the actual release, some protective measures will probably be required, for example, localized sheltering to minimize the likelihood of health effects.

INPP3 release category comprises the accident sequences of A and D hazard states with FP transport path Direct release. The external release comprises more than 50000 Ci of ^{131}I and accident with such release could be rated as INES-6 or INES-7. Under this magnitude of a release, protective measures such as sheltering and evacuation will be required, also there is a possibility of delayed health effects over a wide area.

4. Severe accident and APET analysis

The APET analysis of RBMK-1500 reactor is simplified and not typical in a sense that it does not include phenomenological events. The analysis of key processes in RBMK that potentially could lead to severe accidents showed the specific possibility and expediency of the simplified pragmatic approach that allows to avoid the detailed analysis of severe accident development. The important peculiarities of RBMK such as weak self-shutdown of the chain reaction, high heat capacity of the core, character of radial decay heat removal, etc.

make the investigation of high-temperature corium unnecessary and allow to remain mostly in the area near the boundary "undamaged and damaged core". No attempts were made to quantify any of the identified phenomenological events and processes.

In respect of severe accidents the channel type graphite moderated reactors RBMK differ significantly from the other water-cooled reactors like channel type heavy water (CANDU, ATR) or vessel type (PWR, BWR). Some of these differences are listed below:

- Online fuel reload puts specifics to the accidents during fuel reloading but allows practically to avoid reactivity reserve on burn-up;
- Cooling loops independent in water part and fully independent cooling of CPS (Control and Protection System) channels, which capacity is sufficient to compensate decay heat;
- Heat conductible, hot graphite of high heat capacity; heat transfer between fuel channels (FC) and graphite; consequently the low probability of the fuel melt due to decay heat;
- Possibility to obtain the largest effect of voiding on the reactivity (which is a little negative) by changing the volumes of fuel and moderator practically without change in the fuel burn-up. Unfortunately, it was positive in RBMK and caused the extent of the Chernobyl accident. After that the voiding coefficient at the operating units of RBMK was decreased to the level less than 1β by implementation of the additional absorbers and increase of the operating reactivity reserve from 30 to 45 manual control rods, however, neutron balance was degraded. For the newly constructed Unit 5 at Kursk NPP, the core was designed (change in volumes of fuel and moderator) in the way that voiding effect of reactivity is slightly negative;
- Reactor core consists of fuel assemblies, which are located in the channels. In case of severe accidents with fuel melting, fuel debris is expected to remain in the FC or lower water piping and should have a low probability of core-concrete interaction.
- Relatively large amount of water, steam and accumulated energy for 1 MW of power; slow change in steam pressure;
- Stressed channels and graphite have limited lifetime; oxidation, embrittlement, creep of FC and graphite, graphite sitting and the development of the stressed-deformed state and cracks; the possibility to change FC, based on gas gap criteria.
- The behavior (possible damage of fuel elements and/or other reactor elements) under fast cooling restoration of the overheated core (thermal shock) is still insufficiently investigated. It is expected that probability of fuel elements and fuel channels failure strongly depends on cooling rate.

The listed features of channel type graphite moderated reactors except the last helps with the problem of severe accident. Other features usually worsen the situation:

- ALS does not enclose the steam-water piping, steam-lines and drum separators (DS), untight reactor hall and DS compartments connected to this hall, large leakages of the MCC compartments;
- Limited strength of the reactor cavity (RC) and the MCC compartments in case of large LOCA;
- Positive voiding coefficient and the reactivity effect in case of FC and CPS channels dry out;
- One of the RBMK reactor features is a considerable amount of piping and Z-shape is frequently used. In such a peculiar geometry and at low coolant flow rates, complex steam separation, stratification and other phenomena are possible;
- ECCS (Emergency Core Cooling System) water supply beyond special check valves on the GDH (Group Distribution Header), flow stagnation possibility in case of MCC ruptures.

The accident states that basically predetermine the accident progression and consequences of severe accidents are the following:

- Severe accidents with the fast uncontrolled phase that causes major damage;
- Severe accidents with relatively slow heat up of the core (sometimes MCC as well) after the chain reaction shut down, coolant stagnation or the loss of water in the circuit (leak, steaming) under high or low pressures and different overheating levels;
- Severe accidents with the loss of water and the air ingress into the core and fuel oxidation;
- Severe accidents with the air ingress into the core and graphite oxidation;
- Damage of fuel elements and/or other reactor elements under fast cooling restoration of the overheated core (thermal shock).
- The key processes listed below are identified as important for RBMK severe accident scenarios:
- Fast acceleration – power burst;
- Fast thermal hydraulic phase of severe accident;
- Cooling and overheating of the fuel assembly and FC in case of voiding due to material misbalance;
- Heat sink, oxidation, fluidisation (shift) of long-term dry out fuel;
- Processes related with the oxidized, embrittled, damaged elements of the core and the reactor in case of fast cooling restoration (thermal shock);
- Two-phase thermal hydraulic processes in the complicated circuit (under low flow);
- FC behavior in case of contact with the bent fuel elements and graphite blocks;

- Estimation of the multiple FC failure due to failure of one FC;
- Steam generation, pressure increase in RC and graphite columns movement due to coolant release from one or several ruptured FC;
- RC failure due to steam generation;
- Processes in case of air ingress to RC and graphite oxidation;
- Hydrogen accumulation. Damage to structures in case of hydrogen explosion or large LOCA;
- Release, transport and thermal-chemical transmutations of FP.

Due to mentioned specifics of RBMK in most cases it is necessary and possible to avoid the transition to the area of large damage of the core. Level 1 PSA results of Ignalina NPP showed that the main risk contributors of 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 severe fuel damage. During this period core and reactor cooling system undergoes what is essentially a severe thermal-hydraulic transient. Active accident management performed by plant operators may terminate the accident before severe fuel damage occurs. One of the aims of accident management is to provide recommendations when and what kind of operator intervention is necessary. For example, during loss of long term cooling accidents the depressurisation of primary circuit should be fulfilled and core cooling should be recovered before overheating of core starts. The primary circuit depressurisation is possible in e.g. long-term blackout scenario because the steam discharge valves are equipped with hand drives for their opening. This enables to open them even in the case of low batteries. However, AM strategy should be developed and implemented.

The APET block diagrams are based on the schematic accident progression scenarios, which reflect a general idea that any kind of an accident (hazard state A) will lead to reactor building damage and direct release to the environment. In case of success, violation or damage after LOCA events, the ALS path is considered through the structural integrity, ALS isolation and pressure suppression functions. LOCA in zone 4 will also lead to direct release. This scheme does not treat long-term accidents that are analyzed as special case and AM can be applied.

The APET block diagram for the PDS – D-LLZ1 is shown in Figure 1 and is rather representative of the events analysed for other PDSs. Quantification of function events in APETs is supported by deterministic ALS fragility calculations. The APET considers reliability of ALS active systems: condenser tray cooling system (CTCS), spray availability and closure of venting valve (through fault trees). No equipment recovery was considered.

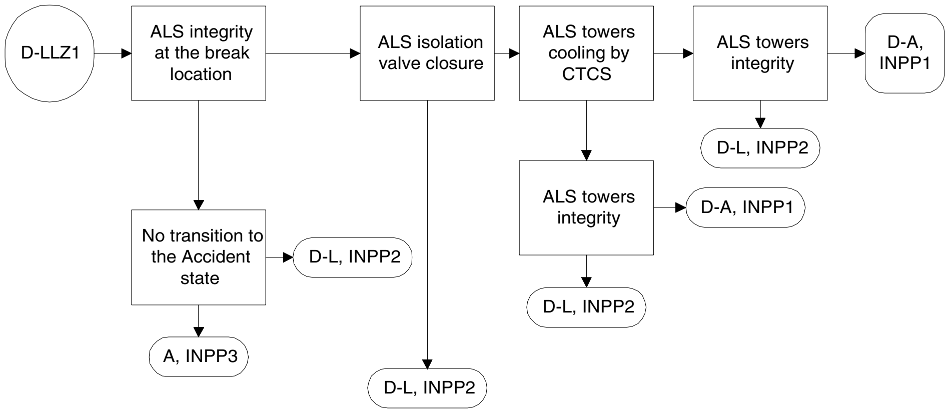

Fig. 1. APET block diagram for the PDS – D-LLZ1
Rys. 1. Schemat blokowy APET* dla PDS** – D-LLZ1
(*Accident Progression Event Tree – drzewo zdarzeń przebiegu awarii;
**Plant Damage States – stany awaryjne obiektu)

5. Containment fragility analysis

The ALS and reactor cavity are designed to perform containment function for RBMK-1500 reactors. The ALS belongs to the pressure suppression category of containments and is similar to BWR MARK II type containment. The ALS must ensure the protection of personnel and environment from radioactive contamination above the acceptable limits following any design basis accident (DBA). This is achieved by the following:
- steam discharged during the accident is condensed reducing the pressure in the compartments, and correspondingly, the release of radioactive materials;
- radioactive materials released during the accident are held in leaktight compartments until they can be decontaminated;
- in the first stage of an accident, clean air is released from the ALS wetwell;
- spray system is used to deposit fission products in ALS wetwell.

The ALS also serves as a water reservoir for emergency core cooling system.

High hydrogen concentrations can locally occur in some of the ALS compartments. The analyses have shown that flammable concentrations are avoided for DBA, but no evidence was if the hydrogen removal capacity is sufficient to cope with the maximum credible hydrogen production during beyond DBA.

Reactor cavity relief capacity has been improved in both units. According to RC integrity SAR calculations performed in 1996, the following reactor cavity capacity in terms of ruptured channels was estimated:
- 9 ± 4 at 7 MPa in the MCC
- 25 ± 9 at 3 MPa in the MCC.

One of the important verification issues is the flow and pressure distribution in the graphite stack during the cause of a multiple tube rupture accident and the impact this might have on mechanical load on tubes and RC head. More recent evaluation of reactor cavity capacity is presented in [2], which indicates even higher numbers: 15±4 at 7 MPa.

The integrity of the confinement is of importance for assessing the release paths and the possibility of dynamic effects following LOCA, which in turn may threaten safety systems and finally make the LOCA end in a core damage state. This scenario was considered as a negligible contributor to risk in the previous level 1 model. However, new lower CDF results for the updated PSA may identify this type of scenario as an important contributor.

The calculated results for the MDBA (Maximum Design Basis Accident) scenario with failed CTCS show that in the case of complete failure of the CTCS, the condenser tray water will reach saturation temperature and loose its pressure suppression function. However, even without the assistance of external cooling the pressure rise remains bounded. The energy removed by the very large heat capacities of the RBMK-1500 ALS water pools and concrete structures plus the energy required to heat the subcooled ECCS water to saturation, limits the absolute pressure rise below design values even if the CTCS failure is assumed [3].

The ALS integrity calculations [4, 5] are important for the estimate of ALS failure probabilities in case of LOCA, when transition from a success state into a damage or accident state is possible. Integrity failure followed by failure of the core cooling function may lead to accident as modeled by APETs.

The ALS integrity failure probabilities are based on 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 one.

Table 3. ALS failure probabilities
Tabela 3. Prawdopodobieństwa uszkodzeń ALS*

Event	Conditional probability
ALS failure in short term due to DBA in zone 1 or GDH rupture in zone 2	1.0E-02
ALS failure in short term due to medium LOCA in zone 2	1.0E-03
Conditional probability of safety system failure given events above	1.0
ALS failure in long term due to CTCS failure	1.0E-01
ALS failure in long term with CTCS available	1.0E-02

*Accident Localization System (System Lokalizowania Awarii)

6. Accident management strategy

Failure of long term cooling is identified as a major contributor to the core damage and eventually a radioactive release. Three long term cooling options were identified and evaluated by deterministic analyses. The main findings are:

– Air cooling via drum separators is not enough;
– Cooling via direct injection into graphite stack is not possible.
– The third analyzed option was to cool via depressurisation and injection of water from deaerators or from any other water source using the fire fighting system. The results showed this option could be applicable.

The third option was analyzed quantitatively as part of APET. Block diagram of the accident management scheme is shown in Figure 2.

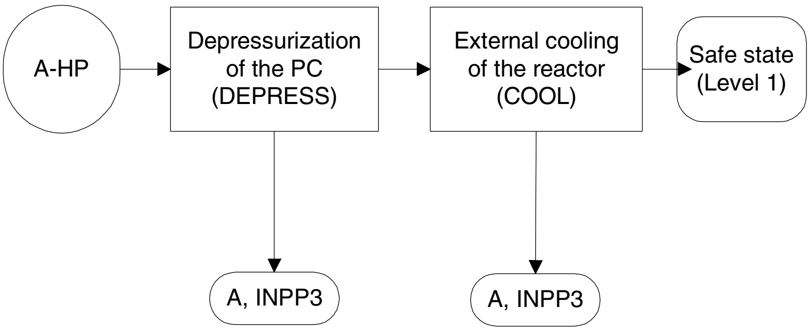

Fig. 2. AM block diagram for the PDS – A-HP
Rys. 2. Schemat blokowy AM* dla PDS** – A-HP
(*Accident Management – zarządzanie awaryjne;
**Plant Damage States – stany awaryjne obiektu)

7. Release estimation

Integrated Level 1 and Level 2 PSA model was developed to estimate the release frequency using Risk Spectrum software.

Large release category (INPP3) includes accidents with extreme release due to complete collapse of the core (national emergency state). The total frequency was estimated to be 4.6E-06 per year without AM interference and 1.9E-06 with accident management. The dominating contributors are:

– Short and intermediate term accident sequences contribution is 8.0E-07 and may not be affected by AM measures. These sequences include ATWS (Anticipated Transient without Scram), large LOCAs and blockage and some intermediate term transients. All contributions are numerically very small.
– Long term high-pressure accident scenarios give the dominant contribution, 3E-06, to the INPP3 release category. This value could be reduced by a fac-

tor of ten by AM procedures. These procedures are being implemented into corresponding EOPs (Emergency Operating Procedures).

– The third important contributor is the confinement integrity failure following a large LOCA. It is assumed that ALS failure leads to core damage and consequently INPP3 release category.

Intermediate release category INPP2 (local or regional emergency state) frequency was estimated to be 2E-04, which is a high figure for a release of this magnitude. The dominating sequence is a large LOCA in zone 4 with successful core cooling. This frequency is not possible to reduce by AM actions. Despite of successful cooling, release magnitude is high and occurs in a short time scale – such an accident could be rated as INES 4-5. However, the assumptions of fission product release are extremely conservative[4] and it is believed that this accident should fall into category INPP1, reducing INPP2 frequency to the order of 1E-07.

Limited release category (INPP1) is dominated by the design sequence "rupture of a single pressure tube", which has the frequency in the order of 1E-02. This frequency is experience-based, a number of cases have occurred, e.g. in Leningrad NPP. Other contributions come from sequences initiated by small LOCAs.

The summary results indicating ALS and accident management measures effectiveness are shown in Table 4.

Table 4. Release frequency without ALS *(with ALS)*
Tabela 4. Częstość uwolnienia bez ALS (z ALS)

	Freq., no AM*	Freq., AM included
INPP3	6.2E-06 (*6.3E-06*)	3.7E-06 (*3.8E-06*)
INPP2	1.5E-02 (*2.1E-04*)	1.5E-02 (*2.1E-04*)
INPP1	0 (*1.5E-02*)	0 (*1.5E-02*)

*Accident Management (zarządzanie awaryjne)

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 the short term and intermediate term sequences. The success LOCA contribution to accident state could possibly be reduced by more detailed study of conservative assumptions.

[4] Possibly a factor of ~50.

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

8. Uncertainty of the results

The Level 2 PSA study is considered to be the first approach for RBMK type reactor and contains a lot of uncertainties in many areas of analysis and quantification process. The explicit uncertainty analysis was not performed at this stage of the study.

The study adopted 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 conservatism and moving towards the best estimate values:

1. ALS fragility evaluation. ALS structural failure probability estimate is based on the deterministic maximum pressure resistance and pressure peak values under MDBA calculations in combination with expert opinion and is assumed to be largely conservative. Uncertainty analysis of deterministic load and resistance calculations could provide a rational tool to estimate structural failure probability. Further research is still needed in this area, but some useful results were already achieved. This approach could also be applicable for other type of containment structures.
2. Source term calculations. Current estimates are based on expert opinion and generic knowledge on radioactive releases, including Chernobyl accident releases. However, Chernobyl release should be interpreted as a very maximum release quantity and plant-specific source term calculations for release paths are necessary.
3. Human action reliability during accident mitigation. Current estimates are conservative expert opinions. Detailed human reliability analysis would be required to justify estimates lower than presently used probability values in the range of 0.1. As one of the arguments that could be used to justify lower HEP is approval of the symptom-based operating instructions.
4. Quantification of phenomenological severe accident events. Due to lack of knowledge on RBMK severe accidents, only qualitative evaluation of phenomenological severe accident events was performed. However, extensive research is needed in this area to get reliable quantitative estimates.

9. Results analysis, plant safety improvements and conclusions

The results of the level 2 analysis indicate figures that are slightly higher than the level 1 results. This is because in the level 1 analysis successful cooling after a LOCA has not been analysed with respect to the confinement function. In level 2 analysis failure of the confinement and consequential damage to the core cooling are taken into account.

The confinement function carried out by ALS does not form a complete barrier, and hence the level 2 analysis can only contribute to limited reduction of release risk. As a number of the level 1 core melt scenarios can not be mitigated by the confinement function, the release frequencies must be expected to lie less than one decade below core damage frequencies.

The lack of separation between different barriers also make the uncertainty assessment more difficult, as frequencies for all dominating scenarios are in the outskirts of what can be seen as reliable figures in PSA techniques, at the level of 1E-07.

The results still clearly demonstrate the efficiency of improvements implemented in the plant, e.g. new DAZ (additional protection) system for reactor scram, the new criteria for reactor scram and ECCS initiation from GDH low flow and new reliable main steam relief valves. These improvements have reduced the frequency of short and intermediate term accidents to a level of 8E-07. The diverse shutdown system is planned to be implemented in Ignalina NPP Unit 2 in 2006, which will further reduce the core damage possibilities and reduce the uncertainties in its estimates.

As a result of Level 2 PSA project, EOPs are being modified (expected to come in force in middle 2005) to include the accident management procedures in case of long term cooling failures under high pressure for transients and area events.

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Podejście do zagadnienia estymacji częstości uwolnień substancji radioaktywnych z reaktora RBMK-1500 w Ignalinie na Litwie

S t r e s z c z e n i e

W artykule przedstawiono podejście do zagadnienia estymacji częstości uwolnień substancji radioaktywnych z elektrowni jądrowej Ignalina na Litwie. Prace zostały wykonane w ramach projektu Barselina, zapoczątkowanego w 1991 roku jako wielostronna współpraca między Litwą, Rosją i Szwecją, mającego na celu ustanowienie wspólnych perspektyw i zunifikowanych baz do oceny ryzyka poważnych awarii oraz potrzeb dotyczących środków zaradczych dla reaktorów typu RBMK. W artykule przedstawiono wyniki dotychczasowych badań oraz omówiono przyszłe działania w tym zakresie.