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Safety Analysis of WWER-440 Nuclear Power Plants: Potential Consequences of a Large Primary to Secondary System Leakage Accident

International Atomic Energy Agency

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FOREWORD

In 1990 the IAEA initiated a programme to assist the countries of central and eastern Europe and the former Soviet Union in evaluating the safety of their first generation WWER-440/230 nuclear power plants. The main objectives of the programme were to identify major design and operational safety issues; to establish international consensus on priorities for safety improvements; and to provide assistance in the review of the completeness and adequacy of safety improvement programmes.

The scope of the programme was extended in 1992 to include RBMK, WWER-440/213 and WWER-1000 plants in operation and under construction. The programme is complemented by national and regional technical cooperation projects.

The programme is pursued by means of plant specific safety review missions to assess the adequacy of design and operational practices; Assessment of Safety Significant Events Team (ASSET) reviews of operational performance; reviews of plant design, including seismic safety studies; and topical meetings on generic safety issues. Other components are follow-up safety missions to nuclear plants to check the status of implementation of IAEA recommendations; assessments of safety improvements implemented or proposed; peer reviews of safety studies, and training workshops. The IAEA is also maintaining a database on the technical safety issues identified for each plant and the status of implementation of safety improvements. An additional important element is the provision of assistance by the IAEA to strengthen regulatory authorities.

The programme implementation depends on voluntary extrabudgetary contributions from IAEA Member States and on financial support from the IAEA Regular Budget and the Technical Cooperation Fund.

For the extrabudgetary part, a Steering Committee provides co-ordination and guidance to the IAEA on technical matters and serves as a forum for exchange of information with the European Commission and with other international and financial organizations. The general scope and results of the programme are reviewed at relevant Technical Cooperation and advisory group meetings.

Guidance relevant to water moderated, water cooled WWER-type reactors and graphite moderated, boiling water RBMK type reactors has been developed within the IAEA's Extrabudgetary Programme on the Safety of WWER-and RBMK nuclear power plants. To a certain extent, accident analysis is also covered in several publications of the IAEA Safety Standards series, for example in the Safety Requirements on Safety of Nuclear Power Plants: Design (NS-R-1) and in the Safety Guide on Safety Assessment and Verification for Nuclear Power Plants (NS-G-1.2). Consistent with these publications, the IAEA also developed a Safety Report on Accident Analysis for Nuclear Power Plants (Safety Reports Series No. 23).

A Coordinated Research Project (CRP) on Assessment of the Interfaces between Neutronic, Thermohydraulic, Structural and Radiological Aspects in Accident Analysis was implemented from 2003 to 2005 to comprehensively evaluate a complex accident scenario within the framework of the IAEA subprogramme on Development of Safety Assessment Methods and Tools. Twelve organizations from eight Member States participated in the CRP. This report provides a summary of the CRP.

Results, recommendations and conclusions resulting from the IAEA programme are intended only to assist national decision makers who have the sole responsibility for the regulation and safe operation of their nuclear power plants. Moreover, they do not replace a comprehensive safety assessment which needs to be performed in the framework of the national licensing process.

The IAEA wishes to thank all the participants for their contributions to the CRP and to this publication. The IAEA officer responsible for this publication was Y. Makihara of the Division of Nuclear Installation Safety

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

One of the most important aims of safety analysis is to confirm that the design and operation of a nuclear power plant are capable of meeting established acceptance criteria. The essential part of accident analysis is currently performed by applying sophisticated computer code packages. Harmonization of the approaches adopted by different analysis teams is needed in order to minimize the impact of user effects on the results of the analysis.

Some of the accidents analysed are very complex, requiring a multidisciplinary approach and treatment. Management of the interfaces between different disciplines (neutronic, thermohydraulic, structural and radiological) is of crucial importance for acceptable results of an analysis. It is very important to ensure correct information transfer from one step of the analysis to another, but for many issues there is also a strong need for feedback from the other disciplines. An understanding of the uncertainties and limitations of the methods used for various aspects of the accident – and therefore active interaction of the experts from the different disciplines – is required to ensure the correct use and transfer of information.

Primary system to secondary system leakage accidents (PRISE) [1] are a typical example of such a complex case. For this particular case the different aspects to be studied are: 1) effectiveness of core cooling, 2) limitation of direct releases of radioactive material, 3) prevention of the accident from developing into a severe accident due to the release of core cooling water from the containment, 4) prevention of pressurized thermal shock (PTS) to the reactor pressure vessel, 5) prevention of inherent boron dilution due to backflow of pure water from the steam generators, and 6) possibility and consequences of steam/water hammers inside the steam generator collectors and in secondary steam lines. In general, reactor protection systems are designed to actuate automatically during the course of the PRISE, but depending on the scenario of the accident there are appropriate operator actions required to avoid more severe consequences Different specific computer codes in several subsequent steps or coupling of computer codes are needed for a comprehensive evaluation of the accidents.

A Coordinated Research Project (CRP) on Assessment of the Interfaces between Neutronic, Thermohydraulic, Structural and Radiological Aspects in Accident Analysis was established to comprehensively evaluate such a complex accident scenario within the framework of the IAEA subprogramme on Development of Safety Assessment Methods and Tools.

Large primary system to secondary system leakage accidents for water moderated, water cooled WWER-440 reactors were suggested as a reference case for the analyses. This category of accidents was selected because of their very complex nature, requiring consideration and treatment of interfaces between different disciplines. It is planned to obtain comprehensive information on these accidents, allowing an independent evaluation of their consequences and countermeasures.

1.2. OBJECTIVE

The objective of this report is to present the results of the CRP on Assessment of the Interfaces between Neutronic, Thermohydraulic, Structural and Radiological Aspects in Accident Analysis. The CRP did not aim to demonstrate the conservatism of the results, the compliance with the licensing limits or the acceptability of the selected transient scenarios, but rather to understand the capability of best estimate codes, to improve the understanding on how to treat interfaces between several disciplines and to identify the requirements for further improvements in this area.

Several IAEA publications [2, 3, 4, 5] have been developed to provide information on performing accident analysis of nuclear power plants. The CRP relates directly to ongoing IAEA activities in this area. It is intended to promote the use of these publications for complex multidisciplinary accident scenarios.

The CRP had two specific objectives. The first objective is to verify the IAEA publications relating to deterministic accident analysis, concentrating on the complex interfaces between various disciplines (neutronic, thermohydraulic, structural, radiological), and to research these interfaces with the aim of gathering information for the review and updating of these publications. At the same time, the CRP constitutes a vehicle for facilitating technology transfer in this area among Member States.

The second objective of this CRP is to contribute to the harmonization of approaches and to a better quality and consistency of accident analysis for nuclear power plants. Results of accident analysis affect several components of nuclear power plant safety such as plant design and various design modifications, the development and modification of operating procedures, operational flexibility and plant lifetime management.

1.3. STRUCTURE

Section 2 provides the methodology of the project, which was implemented under the framework of the CRP. It contains explanations of possible hazards associated with PRISE, specific tasks addressed by the CRP, conditions for the analyses (e.g. initial and boundary conditions, nuclear power plants and initiating events to be analysed, treatment of operator actions) and method of analyses. Section 3 consists of five subsections corresponding to the tasks addressed (i.e. release of radioactive material, pressurized thermal shock, boron dilution, mechanical integrity of secondary systems and severe accidents). Section 4 provides the results of uncertainty evaluations carried out by the participants of the CRP. This report concludes, in Section 5, with findings obtained from comprehensive analyses of PRISE and provides some recommendations to analysts who are in charge of accident analyses for WWER-and pressurized water reactors (PWRs).

Complementary to the main body of the report, detailed results of the analyses are provided in Annexes I and II on the attached CD.

2. METHOD OF EVALUATION

2.1. POSSIBLE HAZARDS EXPECTED AS RESULTS OF PRISE

PRISE represents a complex issue with several safety concerns identified. Possible hazards resulting from PRISE include:

— Releases of radioactive material to the environment through steam bypass station to atmosphere (BRU-A), steam generator (SG) safety valve (SV, containment by-pass), in particular if the BRU-A or SG SV remains stuck open, with uncontrolled coolant outflow into environment;

— Pressurized thermal shock due to primary system cooling down at high pressure after high pressure safety injection (HPSI) actuation and asymmetrical injection of cold water into reactor pressure vessel (RPV), in particular in the case of re-pressurization of the RPV following isolation of the accidental loop;
— Boron dilution due to backflow from the secondary side of non-borated water and resulting insertion of reactivity;
— Loss of integrity of secondary systems due to over-pressurization or due to secondary integrity endangered by pressure spikes in cases where primary liquid floods the secondary system,
— Potential of the event progressing into severe accidents;
— Long term cooling issues owing to the limited volume of emergency core cooling system (ECCS) tanks, as the ECCS water by-passing the hermetical compartment through the break cannot be transfered back into (ECCS) tanks.

The last hazard is mainly related to sufficient capacity of ECCS tanks in connection with appropriate and timely operator action, and will not be considered further as a separate issue. Five issues will be considered in more detail in the analysis. In addition, the problem of uncertainties associated with analysis of PRISE will be partially addressed.

There are several different categories of initiating events that can be considered as PRISE:

— Small leaks (single tube ruptures) – which can be compensated for by the makeup system, safety system actuation may be avoided, without any risk of affected SG overfilling;
— Medium breaks (multiple tube ruptures) – requiring automatic safety injection actuation, while affected SG overfilling may be avoided by proper operator action;
— Large breaks (SG collector breaks including cover lift-up as well as breaks at different locations), for which SG overfilling cannot be avoided.

Only large breaks (or multiple tube ruptures with a sufficiently large number of tubes ruptured) will be considered further as the most challenging ones. In particular, SG collector breaks including collector cover lifting-up will be analysed.

2.2. SPECIFIC TASKS ADDRESSED BY THE PROJECT

It has to be underlined that the project was not intended as an evaluation or demonstration of safety for any of the nuclear power plants considered. Its main objectives were intended to improve methodology for accident analysis. In accordance with the general and specific project objectives, the following specific tasks for analytical work were formulated:

— Identification of the dominant phenomena and scenarios from the point of view of different PRISE hazards;
— Identification and quantification of key parameters and assumptions influencing/aggravating consequences of PRISE and providing recommendations for conservative selection of data and assumptions for licensing type calculations;
— Performing sensitivity analysis regarding various hazards to initial and boundary conditions, for example to the selection of the damaged steam generator (loop

with/without pressurizer, emergency core cooling injection), break size and location, timing of operator action;

- Providing an indication of efficiency and prioritization of various strategies for operator action;
- Comparing control of the PRISE by various features available in different plant designs;
- Comparing results by referencing best estimate calculations and conservative calculations;
- Evaluating uncertainties of the selected PRISE calculations;
- Identifying cross-dependencies between various hazards (interactions between hazards).

2.3. ADDRESSING DIFFERENT HAZARDS BY THE PROJECT

In order to cover the whole complexity of PRISE, six task groups of specialists were established to address specifically the individual hazards and uncertainty evaluations, each of them consisting of one to five participating organizations. The task groups were as follows:

No.1 — Releases of radioactive material;
No.2 — Pressurized thermal shock;
No.3 — Boron dilution;
No.4 — Secondary circuit integrity;
No.5 — Potential of progressing to a severe accident;
No.6 — Uncertainty evaluations.

Due to limited time and resources, and also due to availability of the same plant-specific data for all the participants, individual organizations analysed different nuclear power plants with different design features, set-points, emergency operating procedures (EOPs), and often different assumptions used in the analyses. The analyses were performed by each organization individually and the results were shared, compared, discussed and adjusted in accordance with the comments from other participants. In order to ensure the consistency of the analyses performed by each organization as far as possible, initial and boundary conditions for the reference case calculations were determined as shown in Section 2.6.

Nevertheless it was believed that due to significant similarities in the design, this approach will not only provide sufficient information for updating the methodology, but will also lead to higher confidence in validity of the conclusions.

2.4. APPLICABILITY OF THE RESULTS

From the point of view of the approach adopted in analysis, at least two different types of analysis should generally be recognized: licensing type analyses, and analytical support for development of EOPs. These different analyses have different objectives, and approaches also differ accordingly.

The main objective of the licensing type safety analysis is to demonstrate for all plant states in a robust way that all safety requirements are met, i.e. that sufficient margins exist between real values of important parameters and their threshold values at which damage of the barriers against release of radioactive material would occur. In other words, the objective of the licensing type analyses is:

— Demonstration of the capability of safety systems to maintain fundamental safety functions,
— Demonstration of the fulfilment of specified acceptance criteria (different for different categories), which shall not be exceeded with sufficient margins.

A robust demonstration of safety with margins has to be ensured by using a conservative approach in the analysis. According to the IAEA Safety Guide on safety assessment and verification [6], use of best estimate codes is generally recommended for deterministic safety analysis. Two options are offered to demonstrate sufficient safety margins in using best estimate codes:

— The first option is use of the codes "in combination with a reasonably conservative selection of input data and a sufficient evaluation of the uncertainties of the results". In this statement, evaluation of uncertainties is meant more in a deterministic way: code-to-code comparisons, code-to-data comparisons and expert judgements in combination with sensitivity studies are considered as typical methods for estimation of uncertainties.
— The second option is use of the codes with realistic assumptions on initial and boundary conditions. However, for this option "an approach should be based on statistically combined uncertainties for plant conditions and code models to establish, with a specified high probability, that the calculated results do not exceed the acceptance criteria".

The objectives for emergency operating procedures (EOPs) supporting analysis are quite different:

— Choice of key symptoms and specification of set-points to initiate and to terminate operator action;
— Confirmation of choice, prioritization and optimization of strategies;
— Evaluation of capability of systems to perform intended functions;
— Specifications of environmental conditions for operation of instrumentation and nuclear power plant systems;
— Recommendations for equipment or instrumentation upgrades.

Best estimate approach (using best estimate computer codes with most likely input data for the code) is recommended for this type of analysis, aimed at predicting the most likely nuclear power plant behaviour.

The project contributed to improving methodologies for both types of analyses.

2.5. OVERVIEW OF ANALYSED NUCLEAR POWER PLANTS

The participants of the CRP agreed to select WWER-440/V213 as a basis for the analysis, since this type of reactor is currently in operation in most of the participants' countries. Nevertheless it has to be mentioned that in spite of the same name for the plants there are many design differences between various V213 units. In addition, in some analyses WWER-440/V230 and WWER-1000 units were considered. Specifically, the units considered in the calculations were as follows:

— Loviisa nuclear power plant, unit 1;
— Armenia nuclear power plant, unit 2;
— Kozloduy nuclear power plant, unit 3&4 and unit 5&6;
— Bohunice V-2, unit 1&2;
— Paks nuclear power plant unit 3;
— Dukovany nuclear power plant, units 1-4;
— Kola nuclear power plant, unit 3.

There are several plant as-built features, which are important for treatment of PRISE as follows:

— Primary loop main steam isolation valves (MSIVs) (leak isolation): not applicable for WWER-1000;
— Affected SG steam side isolation;
— SG safety valves (important for pressure control in the affected SG);
— Steam bypass stations to condenser (BRU-Ks) and BRU-As;
— Reactor coolant system (RCS) cooldown capabilities;
— RCS depressurization capabilities;
— ECCS pumps and delivery lines characteristics;
— Main feed water (FW) control and isolation system;
— Instrumentation and control (I&C) involved in diagnosis and mitigation of PRISE.

Not only these features, but also set-points governing the initiation and termination of the system action, and usually plant specific EOPs, are essential for the course and mitigation of the PRISE. In addition, in several plant safety upgrading programmes are under implementation aimed at improving the capacity for dealing with the PRISE. Examples of effective equipment backfits include improved steam lines integrity, pressurizer spray reliability, modifications of I&C logic to allow the operators to terminate the safety injection, monitoring of the subcooling margin, increased ECCS storage capacity and measures for the return of ECCS coolant from the secondary circuit back into the hermetic compartments of the RCS.

Specific design differences and given plant configuration should be carefully considered in the interpretation and use of the results of the analyses.

2.6. SPECIFICATION OF THE REFERENCE CASE AND VARIETY OF ANALYSES

In order to be able to compare the effects of different designs on the course of PRISE, a reference case has been agreed upon for analysis, to be performed in parallel by all participating organizations. In the reference case, conservative assumptions usually taken into account in licensing type accident analysis were not considered (i.e. single failure criterion, loss of offsite power, conservative initial plant operating conditions), since the work was directed towards understanding the realistic sequence of phenomena during the accident. Corresponding initial and boundary conditions for the reference case are specified in Table 1. It has to be understood that specific numerical values corresponding to individual items are frequently different for different plants due to the differences in design.

TABLE 1. INITIAL AND BOUNDARY CONDITIONS FOR REFERENCE CASE

Items	Conditions
Operating condition	Nominal full power operation
Initiating event	Sudden max. PRISE break opening at t_{start} = 0 s on the cold SG header with the shortest steam line
Control and protection system	Normal operation of reactor protection, engineered safety features actuation system, control and auxiliary systems assumed (availability and functioning of systems and components)
Emergency core cooling system (ECCS)	All high pressure ECCS available
Off-site power	Loss of off-site power is not assumed
Single failure	Single failure criterion is not considered
Secondary system operating condition	Nominal water level in all SGs
Decay heat	Decay heat corresponding to long-term operation
Operator actions	No operator action considered
Operation of turbine bypass valve (BRU-K)	BRU-Ks available up to water level increasing in condenser to allowable limit (for the cases without automatic isolation of broken SG)
Initial boron concentration	Initial boron concentration c_B = 0 g/kg
Calculation time	t_{end} = 2 hours – end of calculation

The reference case served also as a basis for parametric studies of selected parameters. These studies include:

- Break size and location, and its hydraulic characteristics;
- Initial conditions of the plant, including reactor power, initial coolant inventory, boron concentration and radioactivity in the RCS;
- Availability and functioning of systems (number of trains in operation, performance of the systems and time delays for their operation);
- Set-points and characteristics of the systems;
- Time for individual operator actions;
- Consideration of various single failures;
- Consideration of consequential failures of non-safety grade systems.

2.7. COMPUTER CODES USED

A variety of computer codes were used by the CRP participants, as shown in Table 2. The majority of the codes can be considered as best estimate codes. The MELCOR 1.8.5 code, a severe accident analysis code used by VEIKI, can also be considered as one of best estimate codes, since no core damage is expected during the transient and will be analyzed in the CRP.

TABLE 2. COMPUTER CODES USED BY THE CRP PARTICIPANTS

Task Group No. (TG)	TG1	TG2	TG3	TG4	TG5	TG6
ENPRO Consult Ltd., Bulgaria	RELAP5/Mod 3.3		RELAP5/Mod3.3			
University of Zagreb, Croatia						RELAP5/Mod3.3
Nuclear Research Institute Řež plc (NRI), Czech Republic	RELAP5	NEWMIX RELAP5 COSMOS FLUENT	ATHLET-DYN3D/3.2 RELAP5-3D FLUENT			RELAP5 ATHLET
Fortum Nuclear Services Ltd., Finland	APROS 5.03			APROS 5.03 RELAP 5, FPIPE, MOC		
VEIKI Institute for Electric Power, Hungary					MELCOR 1.8.5	
Paks Nuclear Power Co., Hungary			APROS 5.03			
University of Pisa, Italy			RELAP5			RELAP5
FSUE EDO "Gidropress", Russian Federation	DINAMIKA-97/ RELAP5/Mod 3.2					
Alexandrov Research Institute of Technology (NITI), Russian Federation	KORSAR					KORSAR
Nuclear Regulatory Authority of the Slovak Republic (ÚJD SR), Slovakia						ATHLET Mod 1.2 D RELAP5/mod3.3
VUJE Trnava,Inc.-Engineering, Design and Research Organization, Slovakia	RELAP5/Mod 3.2.2beta RTARC					
IVS Trnava, Slovakia	RELAP 5 – 3D					

The RELAP5/MOD3 series of codes are system thermohydraulic codes describing the behaviour of complex reactor systems including hydrodynamics, heat transfer, reactor kinetics, control systems and other components. Specific applications of the code have included simulations of transients in LWR systems, such as loss of coolant, anticipated transients without scram (ATWS), and operational transients such as loss of feedwater, loss of offsite power, station blackout and turbine trip. [7]

RELAP5-3D is the latest code version in the series of RELAP5 codes. It is a highly generic code that, in addition to calculating the behaviour of a reactor coolant system during a transient, can be used for simulation of a wide variety of thermohydraulic transients in both nuclear and non-nuclear systems involving mixtures of steam, water, non-condensable gas and solute. The RELAP5-3D version contains several important enhancements over previous versions of the code. The most prominent attribute that distinguishes the RELAP5-3D code from the previous versions is the fully integrated, multi-dimensional thermohydraulic and kinetic modeling capability.

ATHLET is a thermohydraulic computer code (ATHLET stands for analysis of thermohydraulics of leaks and transients) developed by GRS, describing behaviour of complex reactor systems including hydrodynamics, heat transfer, reactor kinetics, control systems and other components under normal and abnormal operational regimes as well as design basis accidents (DBAs). Depending on the application, models are available to account either for a detailed physical modelling or the requirement of a fast running code. The code structure is highly modular and allows for easy implementation of different models. [8]

Code KORSAR/V1.1 completed in 2000 in Russia is intended for the numerical modelling of the WWER-nuclear power plants and other thermohydraulic systems with water coolant. The area of KORSAR/V1.1 code application includes steady state and transient situations including DBAs and beyond design basis accidents (BDBAs). The thermohydraulic processes in the reactor with arbitrary topology and equipment composition can be simulated in the KORSAR code for the whole range of operational regimes.

DYN3D/M2 is a computer code for 3D calculation of reactivity insertion accidents (RIAs) in LWR cores with hexagonal fuel elements, including the initial steady state. The code was coupled with the ATHLET and RELAP system TH codes. [10]

DINAMIKA-97 computer code is intended for the analysis of transients relating to the operating states of a nuclear power plant including normal operating conditions and anticipated operational occurrences. It is a part of "Code package for thermohydraulic calculations of unsteady conditions" (TRAP-97) and is also used for calculation of accidents related to PRISE and breaks of the secondary side pipelines. DINAMIKA-97[1] has been verified on the basis of experimental data, results of preoperational work at nuclear power plant with WWER-1000-type reactors and solution of standard international problems.

2.8. IMPORTANCE OF OPERATOR ACTION

Since in general, reactor control and protection systems are not designed to deal with PRISE accidents automatically, early and appropriate operator actions are essential for successful control of PRISE. Diagnosis of PRISE is quite straight forward due to the unique symptoms:

- Radioactivity of steam in the main steam line and/or blowdown of the leaking SG;
- Decrease of the primary pressure and pressurizer level without pressure increase in the hermetic compartments;
- High level of water in the affected SG and misbalance between FW flow and steam production of this SG.

Objectives of the relevant design features and operator action are as follows:

- Ensuring 'short term' core cooling by means of ECCS;
- Minimization of the release of radioactive material to the environment in order to keep the radiological consequences within the acceptable limits;
- Ensuring 'long term' core cooling by limitation of the coolant release;

[1] DINAMIKA-97 was developed by EDO "Gidropress" and has been certified by Gosatomnadsor, Russian Federation (GAN-RF).

- Prevention of uncontrolled increase of reactor power due to backflow of unborated water from the SG secondary side; critical insertions to be prevented by a realistically low frequency of occurrence;
- Prevention of PTS on the reactor pressure vessel during shutdown of the plant.

The following operator actions are necessary for the treatment of PRISE:

- Isolation of the affected SG from the primary side by means of the main gate valves;
- Isolation of the affected SG from the secondary side;
- Reactor cooldown and depressurization;
- RCS water inventory control;
- Keeping of the reactor subcriticality margin.

Analytical support for EOPs is performed in order to identify proper symptioms for accident diagnosis, appropriate operator action and its timing, to demonstrate efficiency of the action and to specify optimum order of the action.

2.9. RELEVANT ACCEPTANCE CRITERIA

For licensing type calculations, acceptance criteria are specified either in terms of qualitative requirements or quantitative limits for selected parameters. General formulation of criteria as related to different accident hazards are presented in Table 3. These criteria are introduced mainly because they are important for identification of the key phenomena, parameters and assumptions. As already stated in this section, in this project they shall not be used for checking the acceptability of the design and/or any specific regime of a given nuclear power plant.

The criteria are usually expressed more specifically in terms of limiting values; examples of acceptance criteria for design basis accidents are as follows:

- Maximum centerline temperature less than fuel melting temperature, $T_{FUEL} < T_{MELT}$;
- Maximum radial average fuel pellet enthalpy below limiting value (burn-up dependent, fuel-design specific) $h_{FUEL} < h_{LIMIT}$;
- Peak cladding temperature $T_{CLAD} < 1204°C$ (1480°C);
- Maximum cladding local oxidation rate $\delta_{ZrO2} < 17\%$;
- For shutdown regimes with one of the barriers open, no coolant boiling, no fuel uncovery allowed;
- Maximum hydrogen generation $H_2 < 1\%$ H_{2TOTAL};
- Maximum RCS/secondary system pressure below 110% of design pressure (135% ATWS);
- No initiation of a brittle fracture or ductile failure from a postulated defect of the RPV for all anticipated operational occurrences and DBAs;
- Maximum containment pressure below design pressure;
- Maximum containment temperature less than the design specific limit value;
- Differential containment pressure within the given design limits;
- Calculated doses should be below the limits for design basis accident nationally defined.

TABLE 3. GENERAL ACCEPTANCE CRITERIA RELEVANT FOR THE CRP

Safety concerns	Acceptance criteria
Release of radioactive material	Calculated doses should be below the limits for a design basis accident assuming an event generated iodine spike, equilibrium iodine concentration for continued power operation and actual operational limits for primary and secondary coolant
Long term core coolability	Leak should be terminated before depleting ECCS resources assuming stuck in open position of SG safety valves (SV) or BRU-A
PTS	No initiation of brittle fracture or ductile failure from a postulated defect
Boron dilution	No boiling crisis as a consequence of core recriticality
Integrity of secondary system	Integrity of the secondary system should be maintained assuming static and dynamic loads resulted from PRISE and operator action
Transition into severe accident/severe accident	Probability of the transition should be extremely low/tolerable health effects for the population

For severe accidents, either probabilistic or deterministic criteria are used.

Probabilistic criteria (INSAG-12 targets):

- Existing plants: Core damage frequency less than 10^{-4} per plant operating year; Leak rate $<10^{-5}$ per reactor year;
- New plants: values further reduced ~10-times.

Deterministic criteria (country specific):

- No failure of the containment because of pressure and temperature loads;
- No immediate health effects for the population;
- For long-term effects the release limit for ^{137}Cs needs to be below a prescribed value (e.g. 100 Tbq) and all other nuclides together are not to cause larger danger after the time period specified (e.g. after three months).

2.10. LIMITATIONS OF THE METHODOLOGY ADOPTED

For interpretation and any use of the results and conclusions presented in this publication, it is important to have in mind the following limitations of the methodology adopted within the framework of the CRP:

- The main objective of the CRP was qualitative evaluation of the accident needed for methodological improvements and in no case for demonstration of nuclear power plant safety nor for identification of safety deficiencies of nuclear power plants considered;
- Computer codes and inputs used for the majority of cases were not qualified as normally required for licensing type calculations;
- Methodology for licensing analyses as formulated in the relevant IAEA Safety Standards was not followed strictly;
- Due to the large variety of the reactor designs, setpoints and EOPs, but also different assumptions used in the analysis, no direct comparison between the results is possible;
- Due to the reasons given above, it may be also misleading to extend the applicability of the results for other types of reactors without comparative analysis.

3. EVALUATION OF POSSIBLE HAZARDS

3.1. RESULTS RELATED TO RELEASE OF RADIOACTIVE MATERIAL

The most important results obtained by six organizations for the reference case are summarized in Table 4. It should be noted that the table provides the results of calculations that terminated at 7200 s from the initiation of the event. The predicted course of the transient was similar in all calculations. The secondary side of the steam generator (SG) is filled with water within ten minutes in the case of the damaged SG and within one hour in the case of the other SG. Exceptions are Loviisa and Paks nuclear power plants, where the damaged SG is isolated to prevent water flowing downstream from the main steam isolation valve (MSIV).(The steam line is not designed for water flow beyond the point of the residual heat removal system pipe line connection.)

The primary and secondary pressures are maintained by the HPSI system. Total releases of water are almost the same in all cases where similar assumptions were used. However, the total mass of released water is different depending on the assumptions adopted by each organization (e.g. break size, availability of the turbine generator bypass valve BRU-K).

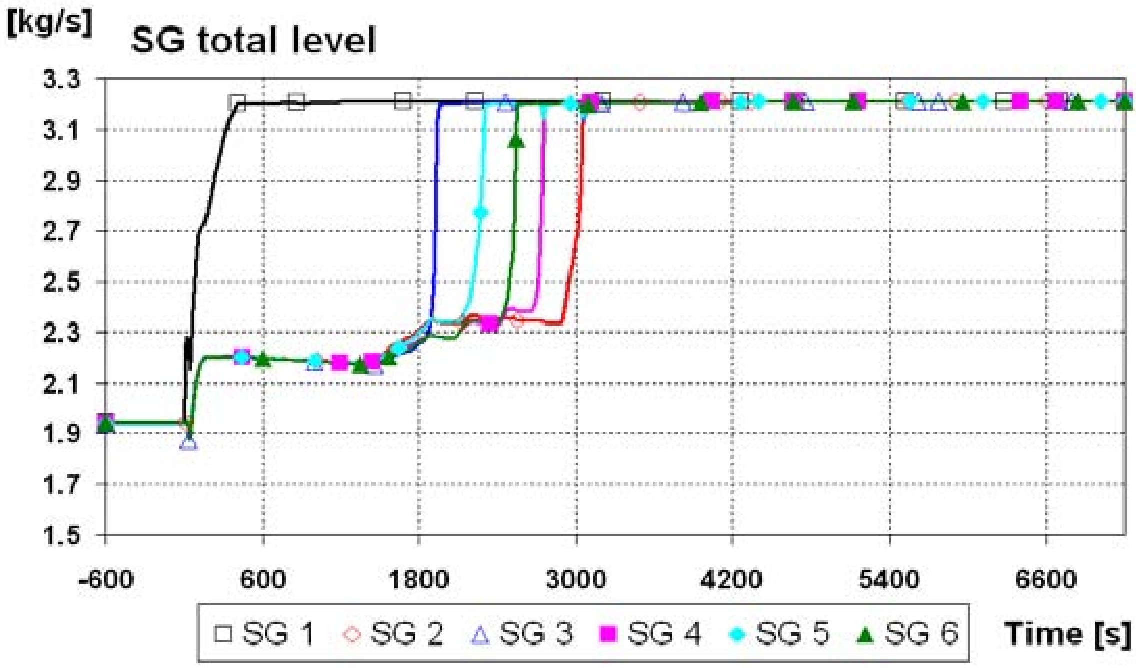
The studies performed by VUJE entailed selecting the input data, defining the break size and specifying the operator actions. These consisted of isolating the feedwater flow to the affected SG, tripping the main coolant pump associated with the defective SG, cooldown through BRU-K on intact loops, primary system depressurization and termination of the HPSI system.

The main results obtained are shown in Figure 1, which shows the comparison of the reference case with a conservative case with single failure (valve stuck open) of the steam dump valve (BRU-A) to the atmosphere:

- Coolant discharge to the atmosphere is about three times lower in the reference case;
- The secondary side of the failed SG would be filled with water (water solid) within ten minutes (both reference and conservative cases). Modification of the collector cover will delay the point at which failed SG water solid is reached by only a few minutes.

VUJE also performed a dose analysis for the conservative case with BRU-A stuck open (without operator action). Results showed that the releases to the atmosphere of radioactive material were smaller than the acceptance criteria after two hours. Further evaluation is necessary to determine the effects of the long term discharge of primary coolant and additional releases of radioactive material from the fuel.

Reference case



Conservative case

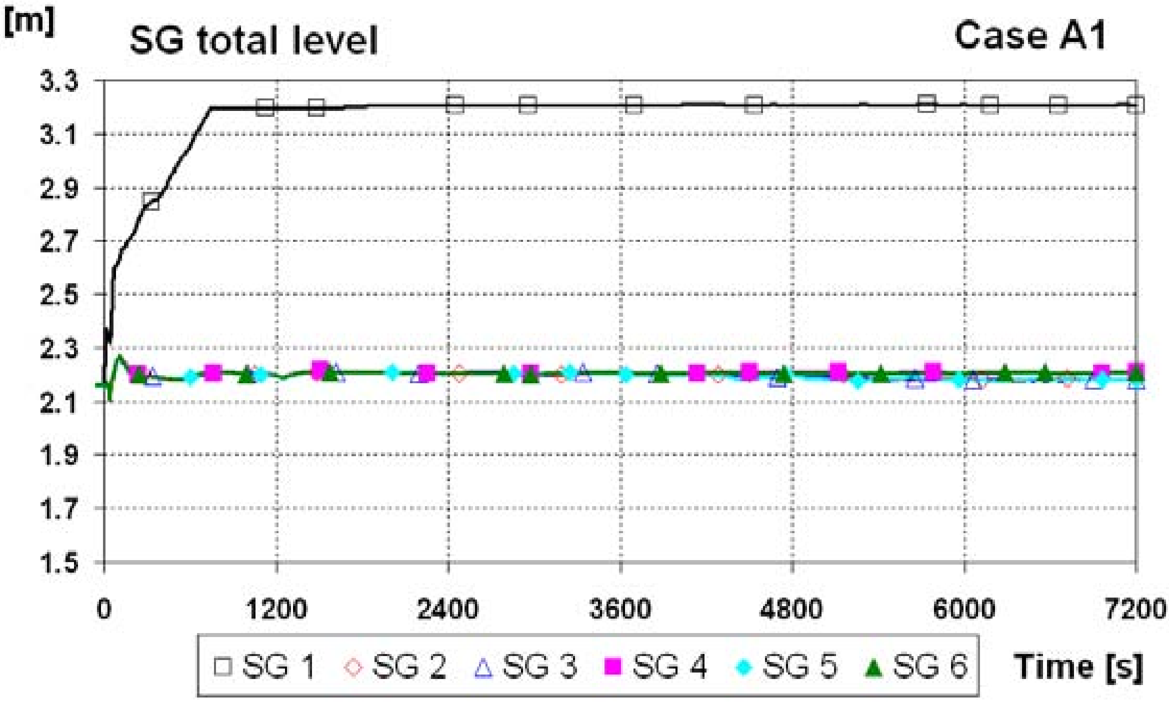


Fig.1. (1) Comparison of reference case and conservative case with BRU-A stuck open (VUJE).

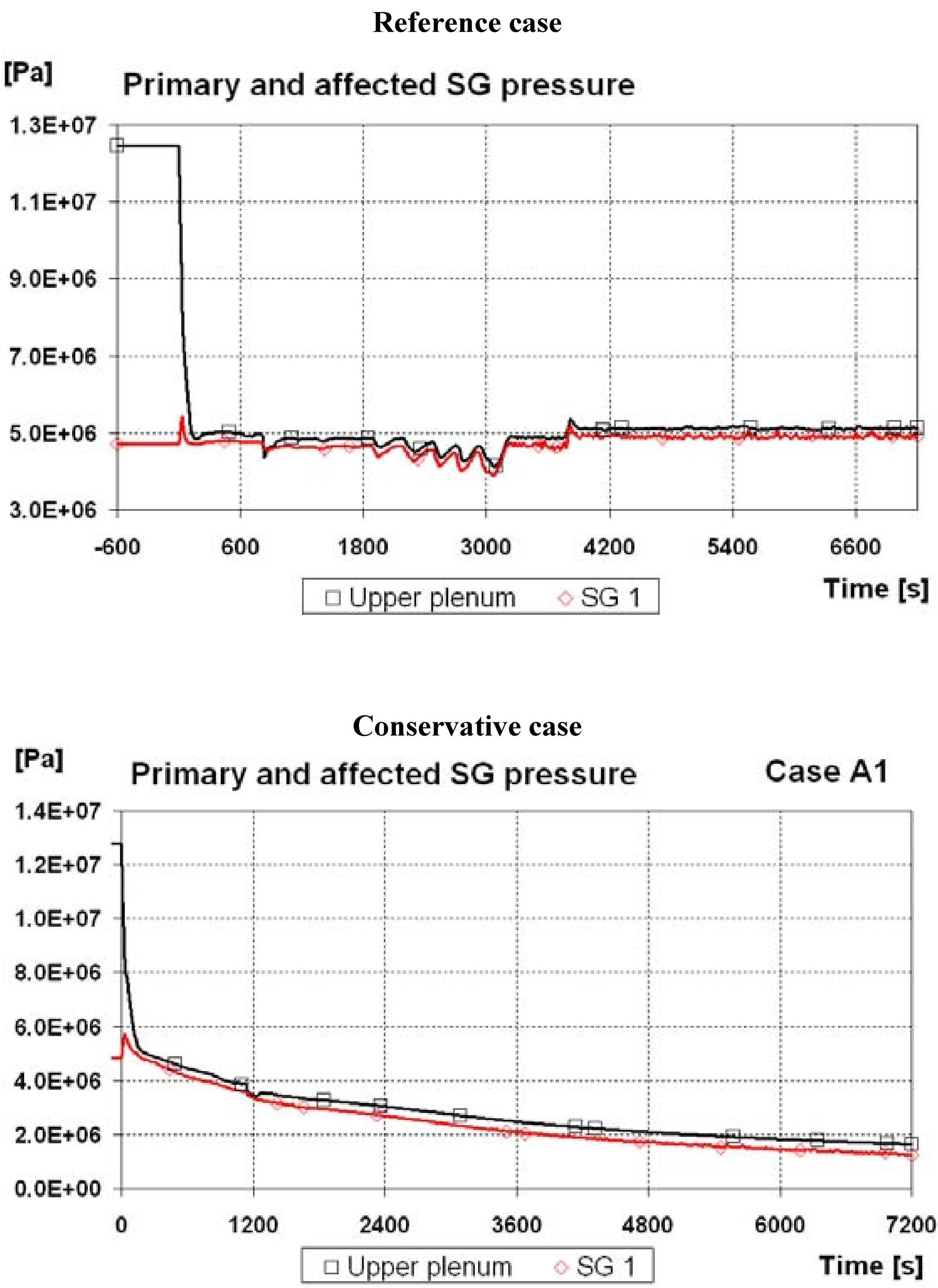
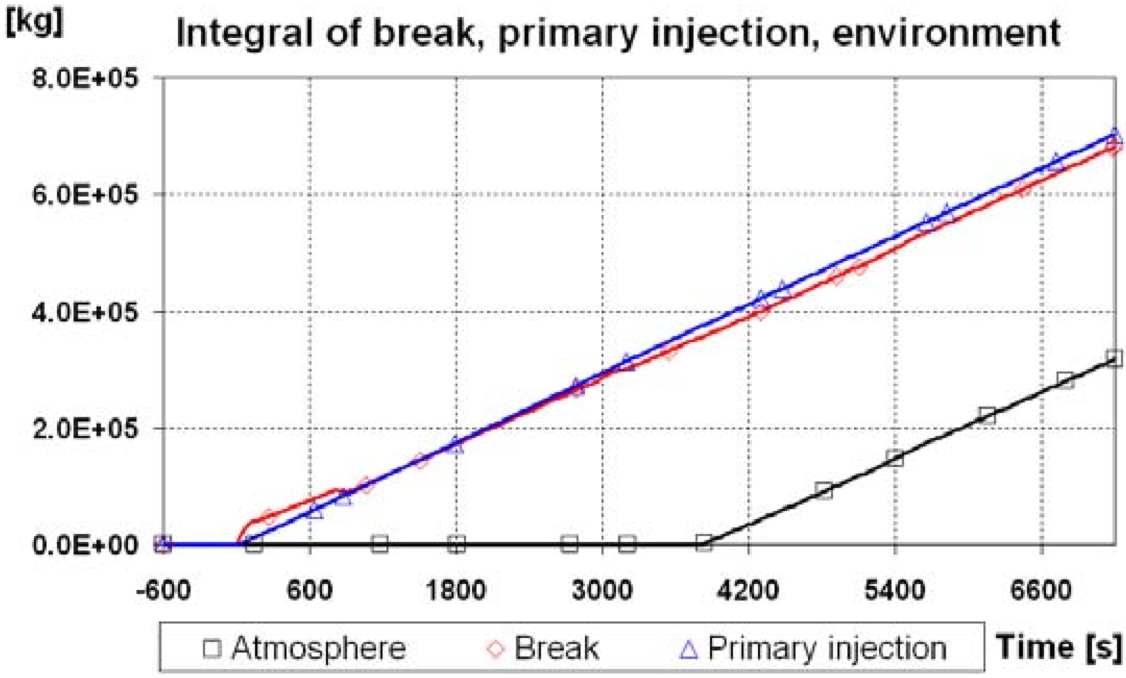


Fig.1. (2) Comparison of reference case and conservative case with BRU-A stuck open (VUJE).

Reference case



Conservative case

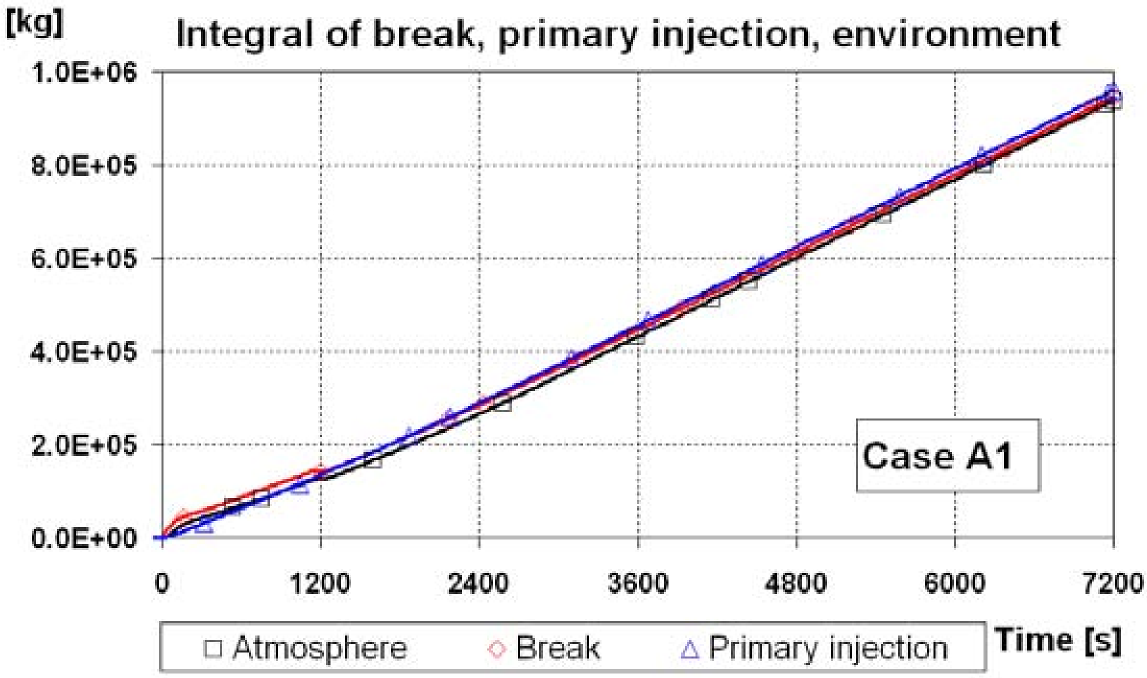


Fig.1. (3) Comparison of reference case and conservative case with BRU-A stuck open (VUJE).

TABLE 4. COMPARISON OF TYPICAL RESULTS FOR REFERENCE CASE

Organization	nuclear power plant	Computer Code	Break equivalent diameter [mm]	Break location	Integral break mass	Coolant discharged from ECCS tanks	Coolant mass in ECCS tanks	Coolant releases to condenser	Coolant releases to environment
VUJE	EBO V2	RELAP5 Mod 3.2.2	107	SG3 cold collector	682 t	702 t	413 t	[1] 56 t	317 t
			107		686 t	707 t	408 t	[2] 370.8t	0.0 t
			38		522 t	350 t	765 t	[1] 45 t	184 t
IVS	EBO V2	RELAP5 3D	107	SG3 cold collector	618	690	425	37 t	276 t
NRI Řež	EDU	RELAP5	120	SG1 cold collector	771 t	791 t	324 t	[2] 470 t	0 t
NITI	Kola 3	KORSAR 1.1	100	SG 3 cold collector	785 t	701 t	393 t	[2] 470 t	0 t
	Kola 3	DINAMIK A-97	100	SG3 cold collector	655 t	435 t	665 t	[2] 495 t	0 t
		RELAP5 Mod 3.2	100	SG3 cold collector	802 t	688 t	412 t	[2] 492 t	0 t
Fortum	Loviisa	APROS 5.03	44	SG 3 cold collector	542 t	395 t	505 t	0 t	435 t

[1] BRU-K closed due to high FW tank level.
[2] BRU-K available within the whole transient.

NITI investigated the effect of operator action, which included closure of the main gate valves (MGVs) and operation of BRU-K or BRU-A. The results, as shown in Table 5, indicate that the coolant discharge can be decreased through the closure of the MGVs and also that early operator action is effective in reducing the coolant discharge. It should be noted that the primary pressure will increase rapidly when the MGVs are closed while the HPSI pumps are in operation. This may challenge the safety issue of PTS. Furthermore, the decision to include the operation of BRU-K should be carefully considered since a two-phase flow entering the pipeline through BRU-K may cause water hammer in the pipe or loss of vacuum in the condenser if no separation of the broken side of the secondary circuit or isolation of the broken loop is made.

The case studies carried out by Gidropress investigated two different PRISE mitigation strategies:

a) Primary and secondary pressure equilibration below the opening set points of the secondary side safety devices. Actions consisted of isolation of the affected part of the main steam header (MSH) and isolation of BRU-A on the affected part of the MSH, restart of the main circulation pumps in the unaffected loops, primary system cooldown through BRU-A on the unaffected part of the MSH, primary depressurization by pressurizer (PRZ) spray, isolation of HPSI pumps.
b) Direct isolation of the broken SG from the rest of the primary system through closure of the MGVs, isolation of the damaged SG on the secondary side, HPSI isolation and reactor coolant system cooldown via BRU-A.

The results, as shown in Table 6, indicate that the coolant release to the atmosphere would be significantly decreased through most of the operator actions; however, the closure of MGVs is not so effective.

TABLE 5. COMPARISON OF COOLANT RELEASES (NITI)

Main gate valve op. (min)	N/A	10	10	10	15	30
BRU-A operation	Yes	Yes	Yes[3)]	Single failure	Yes	Yes
BRU-K operation	Yes	Yes[2)]	No	No	Yes[2)]	Yes[2)]
Coolant release via break (tons)	785	165	141	141	193	305
Coolant release[1)](tons)	315 (t)	167 (t)	62 (a)	127 (a)	155 (t)	163 (t)

[1)] (t): Released to turbine. (a): Released to atmosphere.
[2)] Primary system is cooled down by BRU-Ks not exceeding 30°C/h.
[3)] Primary system is cooled down by BRU-As not exceeding 30°C/h.

TABLE 6. COOLANT RELEASED TO ATMOSPHERE FROM BRU-A ON AFFECTED LOSS OF OFFSITE POWER (GIDROPRESS*)*

PRISE mitigation strategy	N/A[(*)]	N/A	a)	a)	b)	b)
Operator action (min)	N/A	N/A	10	30	10	30
BRU-A stuck open	No	Yes	No	No	No	No
Coolant discharge (tons)	497	920	56	71	61	73

[(*)] Not applicable.

IVS performed a calculation to confirm the effect of the following operator actions: isolation of a failed steam generator on the secondary side via the isolation valve on the main steam header, cooldown through opening of BRU-K on the unaffected part of the secondary system and opening of the pressurizer relief valve (PRZ-RV). The results, as shown in Figure 2, indicate that the phenomenon of reverse flow from the secondary side to the primary side is observed after the PRZ-RV is opened. This may challenge the safety issue of boron dilution.

The operator actions in the case studies carried out by NRI involved the selection of input data and break locations. The results show that the coolant discharge to the atmosphere is significantly affected by the operator actions and that the break location does not have a significant influence on releases of radioactive material.

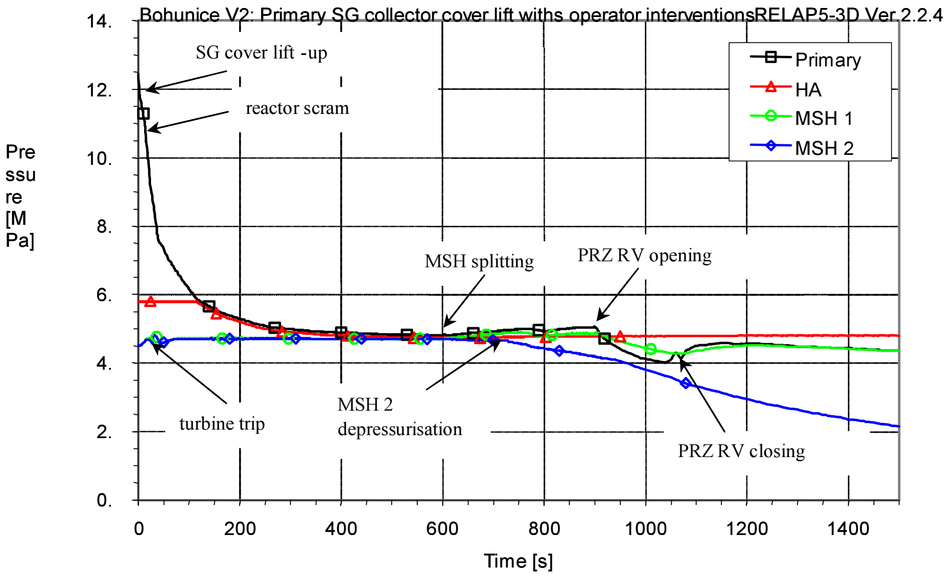
VUJE performed an evaluation of the radiological consequences for the early phase of the accident. Exposure of the population is calculated on the base of spatial and time-dependent air and ground concentrations of radioactive nuclides. The kinds of exposure considered in the calculation include external exposure due to the passing radioactive cloud, external exposure due to radioactive materials deposited on the ground and internal exposure due to inhalation of radioactive nuclides from the passing cloud and inhalation of the re-suspended radioactive nuclides from the ground.

The maximum calculated doses on the boundary of the exclusion area were as follows:

— Effective individual dose (in terms of time, 2 days = 0.245 mSv): 41 times lower than 10 mSv (dose level for introduction of sheltering);

— Effective individual dose (in terms of time, 7 days = 0.273 mSv): 183 times lower than 50 mSv (dose level for introduction of evacuation of all categories of inhabitants).
— Equivalent dose on thyroid of adults (in terms of time, 2 days = 2.37 mSv): 42 times less than 100 mSv (level for introduction of iodine prophylaxis for all categories of inhabitants);
— Effective individual dose (in terms of time, 1 year = 0.385 mSv): 2 times lower than 1 mSv (dose limit for individual inhabitants).

Pressure in primary, HA and secondary side.



Mass-flow-rate through prise leak.

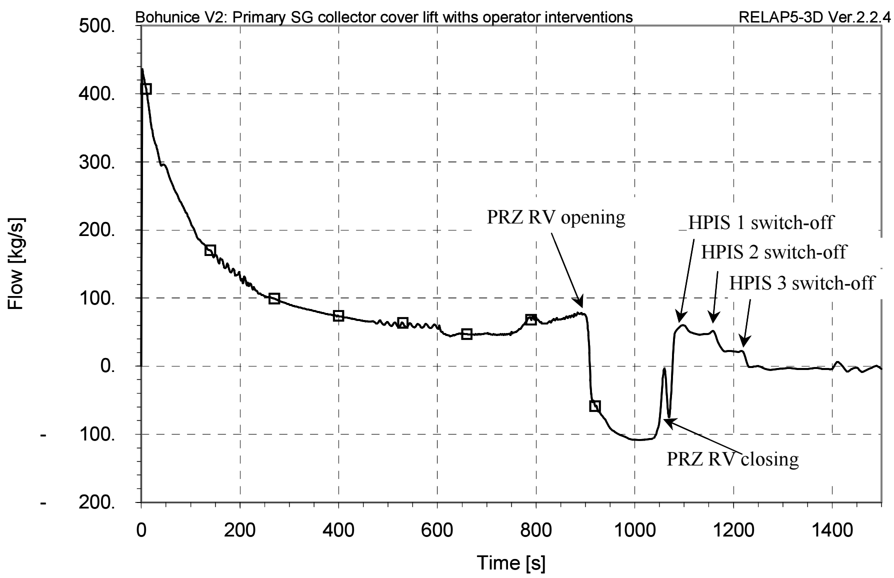


Fig.2. Pressure and break flow transients during PRISE (IVS).

In summary, it may be concluded from the results that PRISE leakages are not controlled by the automatic measures for WWER-440/V213 reactors. In the original design, the existing automatic means were designed to cope with a loss of coolant accident inside the confinement, but are not able to ensure final safe state during PRISE. They keep the emergency core cooling system (ECCS) flow at a maximum in order to maintain core cooling capability; this results in an increase in the PRISE leakage. Without operator action, a large PRISE leak results in the release of a substantial amount of radioactive material from the coolant into the environment. Over a longer time period the ECCS tanks could even become completely depleted and core heat-up may take place. However, due to the large ECCS water inventory and extended time margin available for operator actions, it is highly improbable that the accident would take such a course. Since one cannot rely on the isolation of the defective SG through the MGVs, the most commonly used strategy for PRISE mitigation is based on equilibration of the primary and secondary pressure below the opening set points of the BRU-A valves. The secondary side of the ruptured SG is prevailingly isolated through the MSIV but the disadvantage is that excessive pressure in the ruptured SG must be released into the atmosphere. On the other hand, closing of the MSIV will prevent the radioactive water from entering the intact SGs, which are thus available for primary system heat removal. Moreover, if the defective SG is not isolated and two-phase flow or liquid flow enters the steam line – which is not designed to take such loads – the line may break down and the effect is the same as a stuck open steam generator safety valve (SG SV). The situation may even deteriorate since the hundreds of tons of water in the turbine hall may result in other serious damage like the collapse of the control room roof if the steam line crosses the roof and the roof has not been designed for such water loads. Primary system cooldown and PRZ level recovery are a necessary condition for HPSI termination to support reactor coolant system depressurization. The main advantage of the alternative strategy with the main steam header split into two parts is that the BRU-K may be used – provided that AC power is available. In this case, a large PRISE leak can be managed without the direct release of radioactive material to the environment (although a certain small amount of radioactive material may be released via the condenser vacuum pumps). The main disadvantage of this method is that water from the defective SG penetrates into the affected part of the secondary system, which may endanger the integrity of the secondary system. Furthermore, only three SGs connected to the unaffected part of the main steam header are available for heat removal from the primary system.

3.2. RESULTS RELATING TO PRESSURIZED THERMAL SHOCK

During the course of a PRISE, following the actuation of emergency core cooling system (ECCS) signal, HPSI pumps inject cold water in the ECCS tank into the cold legs of the primary system. Since the reactor coolant pumps are tripped following to actuation of the ECCS signal, cold water from the HPSI pumps may not mix with hot water in the primary system and may cool down some part of the reactor vessel wall. Since the primary system pressure is maintained by high pressure injection, a possibility of PTS cannot be neglected.

The purposes of the PTS analysis were to identify dominant phenomena and scenarios, to develop recommendations for PTS analysis and to update the analysis methodology based on the results of the evaluation.

The analysis related to PTS, performed by NRI, was divided into three steps:

(1) Thermohydraulic analysis,
(2) Mixing analysis,
(3) Structural analysis.

In the thermohydraulic analysis with RELAP5 code, the conservative assumptions for the analysis were related to the worsening of the pressurized thermal shock. The computed alternatives focused on a different number of available emergency core cooling subsystems, initial power and different break points of the SG. The results for the most unfavourable cases correspond to the lifting off of SG2 (Loop No.2 with break and high pressure injection) cold collector cover at zero power, considering the maximum or minimum number of emergency core cooling subsystems. The NEWMIX code was used to analyse the coolant temperature distribution in the down comer. Typical results of mixing analysis, as shown in Figure 3, predicted that a large temperature difference vertically in the downcomer would be generated during the PRISE transient.

Calculations of thermal and stress field were carried out by the finite element method using the COSMOS/M code. Initial and boundary conditions were provided by the results of the RELAP5 and the NEWMIX analyses. Based on the results of the analysis, the maximum allowable value of the critical embrittlement temperature, T_K^a was calculated and compared with the critical embrittlement temperature, T_k. It was confirmed that in this case, the T_k value did not exceed the T_K^a.

In addition, NRI carried out the mixing analyses using the FLUENT computational fluid dynamics (CFD) code to calculate the temperature distribution in the downcomer after the start-up of an ECCS pump. The initial and boundary conditions were provided by the results of RELAP5 calculations. Figure 4 shows the temperatures at the horizontal cutting plane at weld 5/6 calculated with FLUENT and with NEWMIX. Results of the FLUENT calculation showed that the cold plume in the down comer was unstable if only one pump was in operation.

Fortum presented the results of PTS evaluation for Loviisa. Figure 5 shows that the Froude number in the cold leg after 300 s is lower than the criterion for thermal stratification during a large PRISE that is initiated in hot standby conditions. This means that thermal stratification in the cold leg is possible after 300 s and a cold plume formation in the downcomer can be expected.

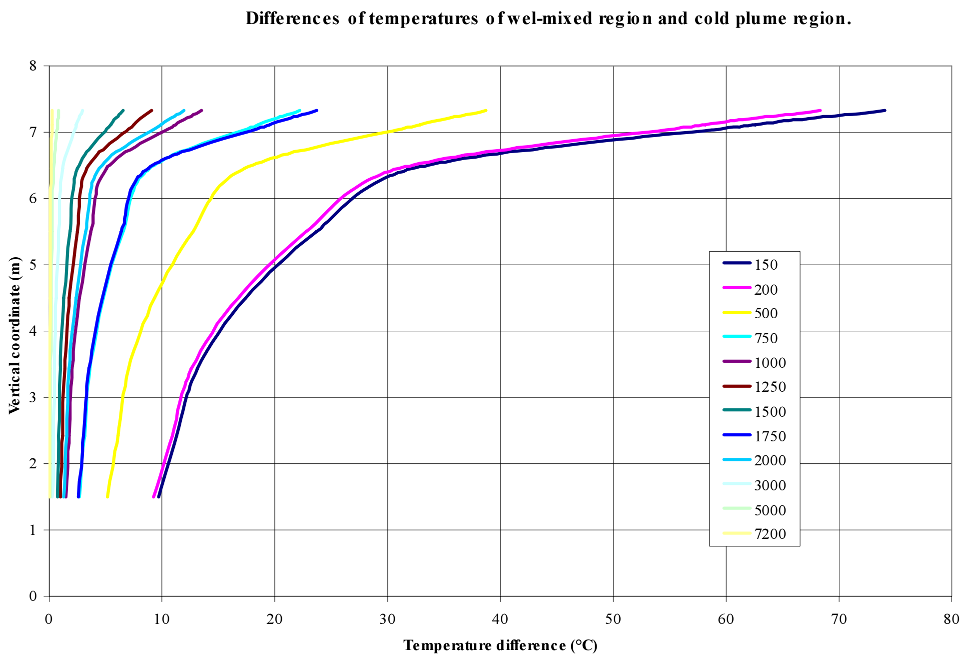


Fig 3. Temperature difference in downcomer (NRI).

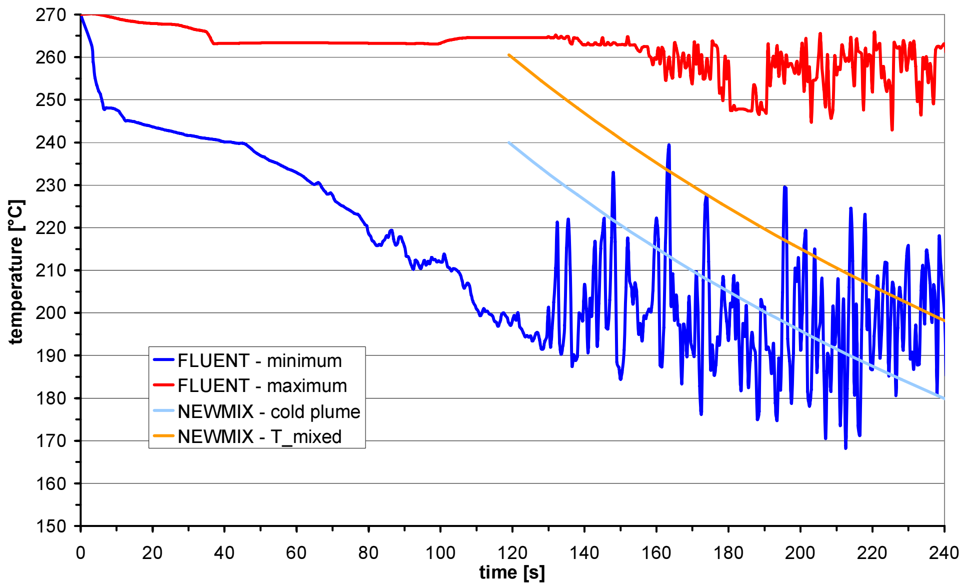


Fig. 4. Comparison of results of calculations performed with NEWMIX and FLUENT.

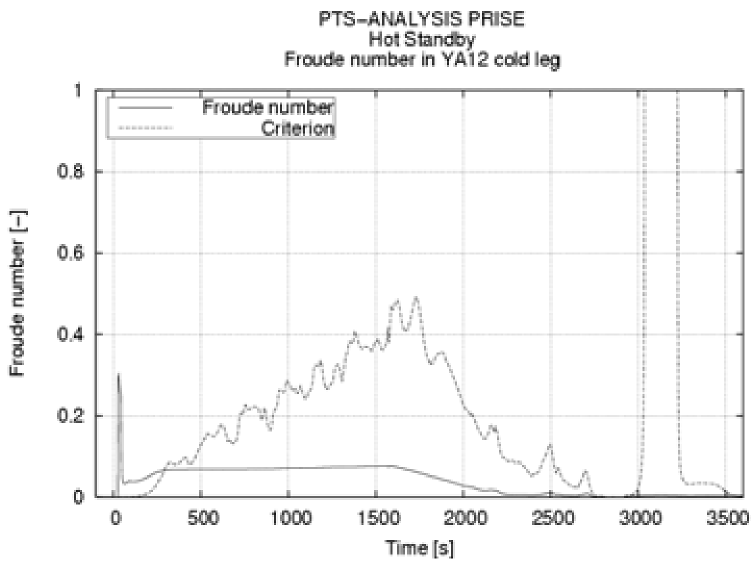


Fig. 5. Froude number vs.tTime (Fortum).

3.3. BORON DILUTION ANALYSIS

During the course of PRISE after the initiation of the ECCS signal and consequently starting the operation of HPSI pumps, operators are requested to stop HPSI pumps so as to avoid the over-pressurization of the secondary system and to start depressurization operation in order to bring the plant conditions to a steady state. With decreasing the primary system pressure, backflow from the secondary side to the primary side may introduce a slug of water with zero or low boron concentration into the primary system and the water coming into the core may result in a reactivity insertion.

The purpose of the boron dilution analysis is to establish a conservative but realistic way of analysis related to the operator actions after PRISE and to briefly assess the possibility of recriticality in a specific PRISE.

ENPRO analysed following two cases to investigate whether recriticality of the core would occur following the reverse flow from the secondary side after initiation of primary system depressurization.

Case (a): No loss of off-site power
Case (b): Loss of off-site power together with reactor scram

Initial and boundary conditions were selected so that the amount of reverse flow could be estimated conservatively. These included:

- The break size was equivalent to the SG collector cover lift-up accident,
- The break position was in the loop without HPSI so as to avoid potential mixing of the clean water flowing back from the break with borated water injected from ECCS,
- Single failure of one HPSI train so as to reduce injection of borated water,
- Early cool down so as to increase the backflow rate after starting cool down operation.

Typical results of case (a) are shown in Figure 6. Reverse flow through the break was calculated after the depressurization operation was started. The results typically indicated that the boron concentration increases rapidly after start-up of the HPSI pumps (about 1200 s) and decreases with the initiation of reverse flow from the secondary side, when the HPSI pumps are stopped and then becomes almost constant after the primary and secondary pressures are equalized. While the results of case (b) showed a similar course of accident, reverse flow after depressurization operation was smaller than case (a). It can be said that even if reverse flow is estimated conservatively, the decrease of boron concentration in the primary system is so small that recriticality cannot occur during a PRISE.

Paks performed an analysis to prove that by the time possible reverse flow occurred, the boron concentration in the damaged SG would already have increased due to operation of the HPSI and would be high enough to avoid any recriticality.

NRI carried out a particular type of bounding analysis in which it was assumed that $7.5m^3$ of deborated water entered one-third of the core and was replaced by pure water after 4.2 seconds. Results showed that the calculated departure from nucleate boiling ratio was higher than the acceptance criterion even though recriticality of the core occurred at the early stage of the calculation.

3.4. MECHANICAL INTEGRITY OF SECONDARY SYSTEM

Mechanical integrity of the secondary system is important during a PRISE. Loosing integrity is not favourable since e.g. if the steam line breaks in the turbine hall the consequences may be serious because the turbine hall is not designed for tremendous water leakage.

Because water flow into steam line cannot be prevented during especially a large PRISE leakage, the steam line must be designed in such a way that it will not loose integrity during water flow situation. Moreover, possibility of a water hammer phenomenon taking place in the steam line during PRISE must be taken into account since water hammer may result in very high force peaks.

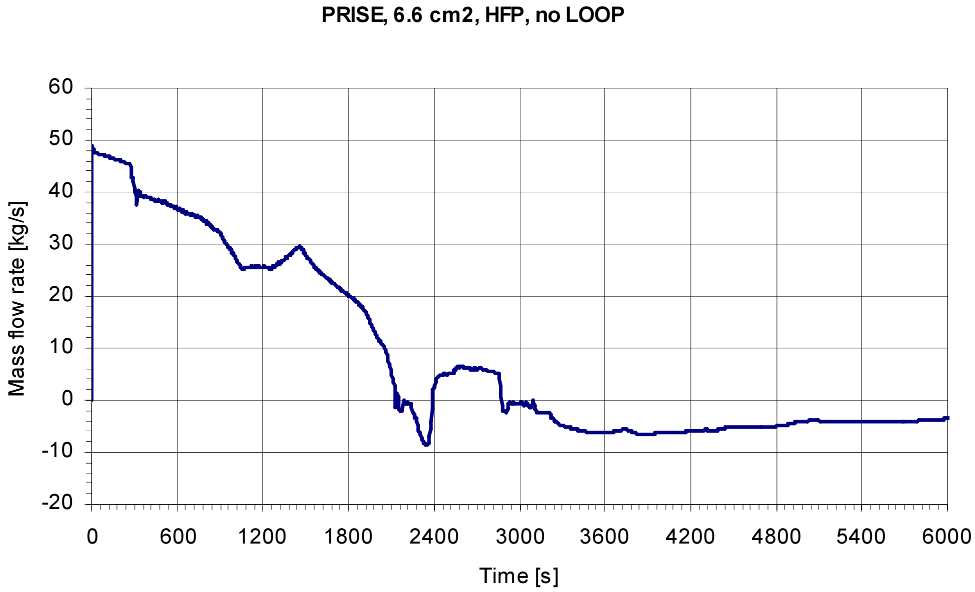


Fig.6 (a) Flow rate through the break (case (a))

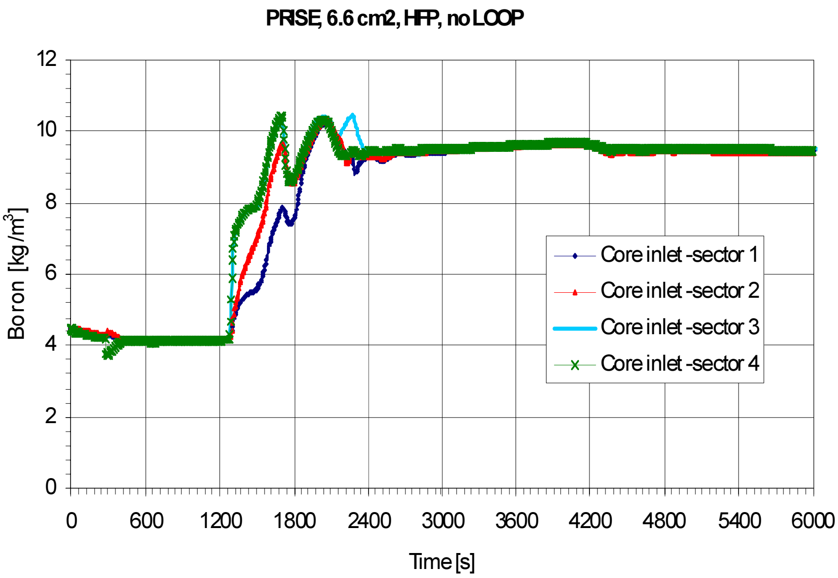


Fig.6 (b). Boron concentration vs. time (case (a)).

The most important parameter to calculate is the maximum effect of water hammer; i.e. maximum water front velocity when it hits e.g. to the isolation valve. When maximum force effect has been calculated, the steam line support structures can be designed in such a way that water hammer does not challenge the steam line integrity.

The purposes of the water hammer analysis are:

- Investigation of water slug impact to main steam isolation valve during large PRISE;
- Definition of hydraulic loads;

- Use of hydraulic analysis data for strength analysis;
- Recommendations for consideration of various phenomena, selection of modelling options and conservative input data.

An analysis related to the water hammer phenomenon during a large PRISE was performed by Fortum. The analysis was divided into three phases:

(1) Thermohydraulic analysis using a system analysis code like RELAP5. The secondary steam pipe between SG collector and MSIV was modelled. The purpose pf the analysis is to calculate the maximum water front inpact velocity as a function of initial volume of steam between the MSIV and the waterfront.
(2) Calculation of the force affecting the MSIV and the system. A computer code W_H_FORCE developed in FNS was used to calculate the impact force based on the results of thermohydraulic analysis.
(3) Analysis using the FPIPE or ABAQUS code to check the strength of the structure. The calculation model was extended from the closed valve to the first anchoring point.

The thermohydraulic analysis for the reference case predicted the maximum water front velocity at the moment of impact on the MSIV to be 9.4 m/s when the length of the steam volume is approximately 1.3 m (V = 0.2 m^3), as shown in Figure 7.

Figure 8 shows a typical result of water hammer analysis. For each steam line segment, the water hammer phenomenon causes a rather big force peak, which, however, is of very short duration. The result corresponds to the situation in extreme conditions. If there are, for example, non-condensable gases in the water or the impact takes place under two-phase conditions, the force is considerably smaller. This is because the speed of sound in a mixture of water and steam and/or non-condensable gas is substantially lower than in solid water.

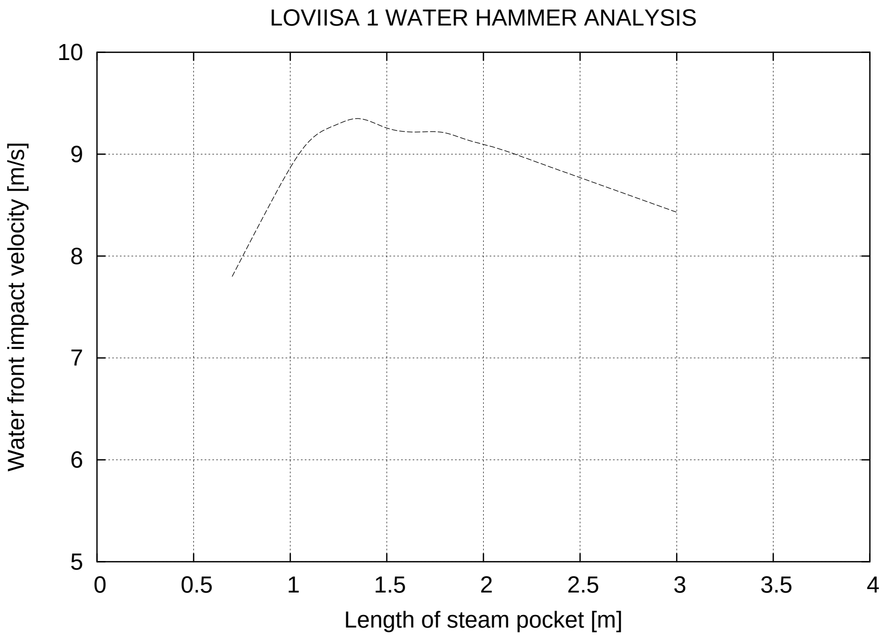


Fig.7. Water front velocity vs. length of steam pocket (Fortum).

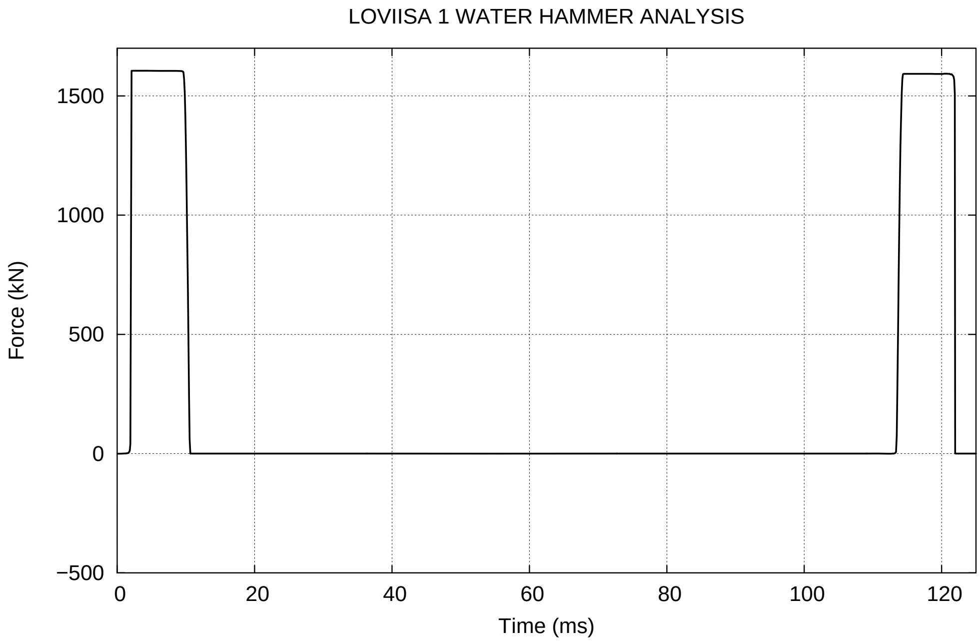


Fig. 8. Water hammer force vs. time calculated with the W_H_FORCE code (Fortum).

The situation considered in the analysis assumed that several phenomena take place during the sequence and also at a particular moment. Initially, the steam generator safety valve must remain in an open position and later on the valve must close at the right moment. When the pressure increases, the steam pocket must vanish precisely at the right moment as a result of fast steam condensation. The probability for this sequence is obviously very small. On the other hand, if the steam line can withstand the forces engendered by this kind of sequence, the structure is definitely strong enough to survive all PRISE .

Best estimate type analysis to calculate very rapid steam condensation in the steam line would be a very complicated undertaking. Since system analysis codes are not able to calculate this phenomenon, only a conservative approach can be applied.

3.5. RESULTS RELATED TO SEVERE ACCIDENT ANALYSIS

Severe accident analysis was performed by VEIKI to determine the thermohydraulic behaviour and the source term of the plant during an unmitigated severe accident sequence initiated by a large primary to secondary break. In the analysis it was assumed that the ECCS was not available and that the SG safety valve was stuck open upon its first opening.

The analyses were performed using the MELCORE 1.8.5. Since the time to severe accident strongly depends on operator actions, even if the ECCS was not available, following three cases with variation of operator actions were considered:

- **Base Case-** SG manway cover lift-up (break) with one of SG SVs stuck open after first opening

- **Case 1 (PRZ Open Case)**: same as Base Case but the operator opens the PRZ Safety Valve to containment.

- **Case 2 (Secondary cooling Case)**: same as Base Case but the primary system is depressurized through the secondary systems with intact SGs by opening the steam dump to atmosphere (by two BRUA valves) on the loop.

Table 7 shows the summary of the results. It will take at least 13.5 hours until reactor vessel failure occurs even if no ECCS actuation and no operator actions were assumed. This could be extended to a few days through appropriate operator actions, not including the resumption of the ECCS.

The results of analysis of the release of radioactive material are given in Figure 9. Results indicate that although the operator action with secondary side cooling can significantly prolong the initiation of severe accident, release of radioactive material to environment cannot significantly be decreased, once core damage starts.

TABLE 7. TIME OF REACTOR VESSEL FAILURE IN CASE OF PRISE WITH SG SV STUCK OPEN (VEIKI)

Conditions	Time of RV Failure (h)
Base case	13.5
Case 1	19.5
Case 2	77

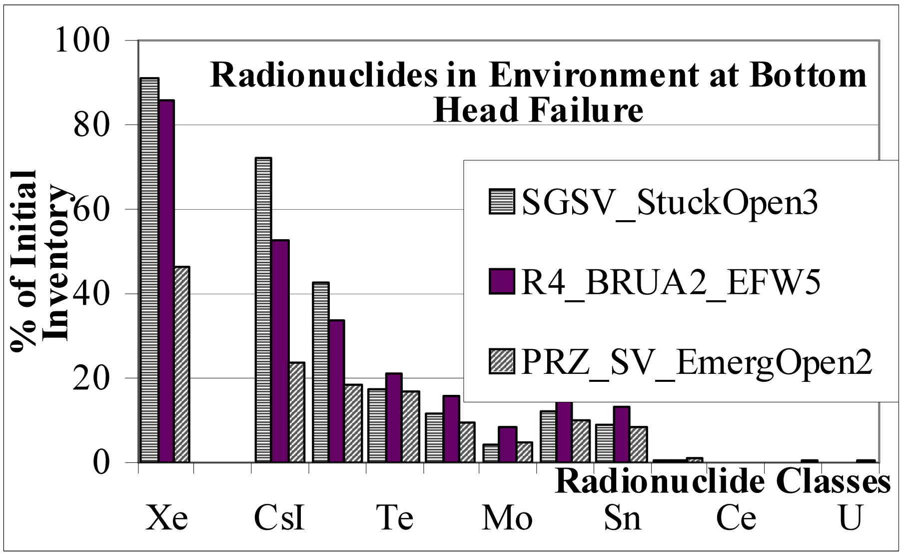


Figure 9 Fission product release to the environment during the G manway cover lift-up accident, d=38mm without and with operator actions, as a percentage of initial inventory.

4. UNCERTAINTY EVALUATION

The objectives of the application of uncertainty methods to the analysis of the PRISE event can be summarized as follows:

a) To establish a connection between the uncertainty in the prediction of the key parameters and the values of the acceptance criteria, thus supporting the evaluation of the radiological impact and other hazards of PRISE;

b) To show the possibility of supporting improvements in the design of emergency and accident management procedures;

c) To demonstrate the applicability of the method used for such kinds of transients.

TABLE 8. SUMMARY OF RESULTS FROM UNCERTAINTY ANALYSIS AND SENSITIVITY STUDIES

–			INSTITUTIONS					
No	QUANTITY or ITEM		ÚJD	NITI		NRI		UNIPI*/ UNIZA**
1	nuclear power plant	–	Bohunice	Kola		Dukovany		Metzamor
2	CODE		ATHLET	KORSAR		ATHLET		RELAP5
3	UNC. METHOD		GRS method	GRS method		GRS method		CIAU
4	SENS. STUDY		– from Unc.	– supporting – from Unc.		– from Unc.		– supporting
5	OTHER STUDY		– ITF analysis – k_v-scaled	–		–		–
	1) SUPPORTING SENSITIVITY STUDY							
6	BL isolated at 10 min	Sens. in BFI at 7000 s (tons)		25				
7		Sens. in MRE at 7000 s (tons)		127				
8	W / W/0 op. action	Sens. in BFI at 7000 s (tons)	–	715		–		
9		Sens. in MRE at 7000 s (tons)		560				
10	2 to 6 pumps on	Sens. in BFI at 7000 s (tons)						15
	2) APPLICATION OF UNCERTAINTY STUDY							
11			–	c)	e)	1)	2)	–
12	No. of input uncertain parameters		34	16	16	3	7	–
13	No. of code runs		100	100	100	100	100	1
14	Error in BFI at 1800 or 1000* s (%/ton)		15/30	50/ 70		10/ 10*	20/ 20*	15/ 20
15	Error in MRE at 1800 or 1000* s (%/ton)		15/25	–/ 140		20/ 8*	60/ 25*	
16	Error in BFI at 7000 s (%/ton)		–	50/ 70	5/ 40		5/ 40	20/ 60
17	Error in MRE at 7000 s (%/ton)		–	90/ 100	25/ 160		40/ 200	
18	Time error at 7000 s (s)		–	–	–	–	–	1000
	3) SENSITIVITY FROM UNCERTAINTY							
19	Most relevant parameter		HPSI flow Break BRU-A LC	Break SG level Decay power BRU-A setp		Reactor power Decay power Break		

* University of Pisa.
** University of Zagreb.

BFI = Break flow integral. MRE = Mass release to the environment
Error = maximum of| reference case – upper /lower Uncertainty bound| .
Sens. = Sensitivity = maximum of| sensisitity calculations. – reference calculations| .
ITF = Integral test facility. Setp = Set point.
LC = Loss coefficient. Unc. = Uncertainty.

The summary of the results obtained from the use of uncertainty methods at ÚJD, NITI, NRI and the Universities of Pisa and Zagreb can be found in Table 8. In this connection, the following should be noted:

- Four different nuclear power plants (all WWER-440 types) were considered in the uncertainty application.
- All the transient scenarios are caused by PRISE leakage, but the sequence of events and main calculation assumptions are not discussed.
- Three applications of the method developed by Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) (whereby in two cases the institution involved performed two applications, thus bringing the overall number of applications of the GRS method to five) and one application of CIAU (code with internal assessment of uncertainty) can be distinguished.
- All applications of the GRS method were completed based on 100 code runs, but the number of input uncertain parameters varied between 3 and 34.
- In two cases the duration of the transient was limited to 1800 s (the ÚJD application) and to 7000 s (the first case of the NRI applications) and also in the remaining cases the considered duration was 7000 s (the second case of the NRI applications). Furthermore, the proposed applicable duration for the uncertainty evaluation in the NRI study is about 4000 s (owing to the failure of some code runs).
- The three most relevant parameters in NRI's case are the three input parameters selected for the first application and the seven for the second application.
- The application of the uncertainty methods was completed without fixing the targets. Not even the relevant output parameters in relation to which uncertainty should be evaluated were fixed in advance. On the one hand, this gives freedom to the users of the method, on the other, however, it does not permit the capabilities of the methods to be fully exploited.

The key results, derived considering the integral of break mass flow (from primary to secondary, BFI = break flow integral) and the overall coolant mass released to the environment (MRE = mass release to the environment) as the reference outcomes from the analysis, can be summarized as follows:

a) The selection of the transient boundary conditions, including operator actions, has a far larger impact upon the reference outcomes than the impact of uncertainty.

b) In relation to the above, best estimate evaluation of the results produces 0 or 622 tons of coolant released to the environment, depending upon operator actions. Similarly, the mass of coolant exchanged through the PRISE break varies between 859 and 141 tons depending upon operator action and other transient assumptions.

c) There is a convergence of results in relation to the absolute value of the error in predicting BFI and MRE. However, the error (relative) related to the reference (best estimate) prediction is different.

d) The predicted errors in the selected quantities are strongly a function of the transient, comparison between NITI cases c) and e)

e) The comparison of data in rows 14 and 15 with data in rows 16 and 17 shows that the absolute errors for BFI and MRE do not increase with time.

f) An error in the time of occurrence of events of the order of 1000 s at 7000 s of transient (actual) time, i.e. of the order of 15% of the actual time, is predicted by CIAU.

g) There is little agreement as to the most influential input uncertainty parameters for the study: these have been identified explicitly in the cases of ÚJD and NITI and are the three input uncertain parameters selected for NRI's Case 1. However, break flow appears relevant from the three analyses. The little agreement on influential parameters might result from the fact that each participant used different codes, different models and different input data.

5. CONCLUSIONS

Although PRISE events have been widely studied and the safety concerns as consequences of the event have been specified in the literature [1, 10, 11], guidance focusing on how to treat the interfaces between several disciplines has not been developed.

A broad scope investigation of PRISE events in WWER-440 reactors was carried out by ten institutions interested in the deterministic safety analysis of those reactors.

Twelve different codes were used and tens of code-runs were performed. Attention was focused on the following topics:

— **Release of radioactive material:** A consequence of the PRISE is a bypass of the system for confinement by the radioactive coolant in the primary system caused by the planned opening of the BRU-A valves. Therefore, a key issue for the analysis is quantification of the unavoidable radiological impact to the environment.
— **Pressurized thermal shock:** Depending on the PRISE break area and other boundary conditions primarily relating to operator actions, the transient may evolve at high pressure with cold water in the primary system following actuation of the ECCS or fast cooling of the steam generators. This creates the potential for PTS.
— **Boron dilution:** Flow reversal at the break, primarily a consequence of EOPs or of related operator actions, causes boron dilution in the primary system and the need to investigate potential re-criticality phenomena.
— **Mechanical integrity of secondary system:** Liquid filling of the steam generator, including the steam lines, causes the potential for water hammer and consequent structural loads that may damage components like valves or cause pipe breaks, e.g. additional steam generator tubes or steam lines.
— **Severe accident:** A non bounded PRISE evolves into a severe accident, as any other event, if the ECCSs do not work properly. The evaluation of system scenarios is relevant to mitigate the consequences of such an unlikely event.
— **Uncertainty evaluation:** Computational tools are used for any of the above topics. The results from any qualified numerical tool are affected by errors having different origins. These errors may significantly impact the results and need proper evaluation by suitable uncertainty methods.

At the basis of all the topics is the analysis of the system thermohydraulics and the transient behaviour of the nuclear power plant performed by best estimate codes. However, no effort was made to demonstrate the conservatism of any result or the compliance with licensing limits. Similarly, no effort was made to demonstrate the acceptability of the selected transient scenarios from the point of view of the probabilistic safety assessment (i.e. no consideration was made of the probability of occurrence of the events concerned). The reasons for the investigations can be summarized as follows:

a) To understand the interactions among topics and the phenomena and safety aspects involved;
b) To demonstrate the capability of coupled codes or to identify the requirements for improvements in the area.

Conclusions are derived hereafter for the thermodynamic system analysis and for each of the topics listed above and are provided hereafter together with recommendations for analysts in charge of accident analysis of WWER-reactors.

Thermohydraulic system analysis

PRISE phenomena are widely studied in the literature and several experiments have been performed in scaled facilities. Results of these studies generally show that, excluding the case of severe accidents, the core is predicted to remain covered during the entire course of the event, provided the ECCSs operate within their design conditions. Therefore, no special challenge is placed on the code models. However,

— Modelling of break region and related flow (both direct and reverse) prediction requires specific attention by the code user; proper values of pressure drop coefficients must be implemented considering that critical and non-critical (Bernoulli) flow conditions occur at the break;
— In cases when conditions are created for water hammer phenomena, proper noding features need to be added to ensure that related phenomena are predicted by the code.

The performed calculations show that, without proper operator actions, the PRISE event causes the ECCS pump to act as tool for transferring the reservoir tank liquid from inside the confinement system to the environment, i.e. throughout the break and the presumably stuck-open or cycling BRU-A valves. Suitable EOP are designed to prevent this conditions and their optimization constitutes one of the targets for the thermohydraulic analysis.

According with the IAEA requirements, the analyses documented within the present framework have been performed by best estimate codes either using conservative input conditions or realistic input conditions supported by uncertainty study (even though demonstration of compliance with licensing limits was not part of the study, as already mentioned).

Release of radioactive material

All pressurized water reactors up to the third generation, i.e. including WWER-440, are characterized by containment bypass in case a leakage occurs, e.g. PRISE, between the primary and secondary sides of the steam generators. The objective of the related analyses is twofold:

— To show that releases of radioactive material are within the licensing limits;
— To minimize the radiological impact to the environment.

Within the framework of the latter activities, it has been shown that operator intervention is essential. Namely, the depressurization of intact steam generators should be planned as soon as possible after the accident diagnosis. In contrast, the isolation from the primary side of steam generators is not a recommended countermeasure. In this framework, consideration must be given to the issues of PTS and boron dilution as outlined below.

From the point of view of the selection of reasonable boundary conditions for the analyses aimed at calculating the radiological impact, the recommendation is to follow the State specific and the reactor specific guidelines to account for radioactivity transport from the primary coolant and from the 'failed' pins to the environment, including the 'spike effect'.

Pressurized thermal shock

The PRISE does not constitute the most challenging scenario for PTS in WWER-440. Nevertheless a PTS study is recommended for the PRISE. The potential for PTS conditions may derive from the (too) fast depressurization of intact steam generators as well as from large flows of ECCS water. Therefore, the intact steam generator depressurization rate and ECCS flow rate should be limited by PTS related considerations.

From the point of view of (PTS) analyses, single and two-phase PTS conditions should be distinguished. In the former case CFD codes properly coupled with structural mechanics codes should be used to conduct reference (bounding) studies taking into account that a systematic application of coupled CFD structural mechanics codes is impractical owing to the necessary computational resources.

Boron dilution

The problem occurs as a consequence of flow reversal at the break. Flow reversal can be generated in the attempt to minimize releases of radioactive material to the environment. This constitutes a clear example that shows how an improvement in one of the topics (e.g. radiological impact) has a negative effect on another topic (boron dilution). The recommendation here is to develop nuclear power plant specific and scenario specific procedures in such a way that minimum flow reversal occurs and/or alternatively that flow reversal is associated with suitable (preventive) re-boration of primary system coolant. This might imply a preventive estimation of the PRISE leakage rate (e.g. based upon the rate of change of pressure or of level in secondary side and/or in the pressurizer).

From the analysis point of view, and in case the effect of boron dilution has to be evaluated upon the core reactivity, CFD analyses are recommended for predicting the mixing in the reactor pressure vessel, as well as the use of suitable three-dimensional neutron kinetics codes for the core. The use of point kinetics may lead to non-conservative results.

Mechanical integrity of secondary system

The steam generator integrity is challenged by high pressure or by water hammer phenomena as a consequence of overfilling. The operation of discharge valves BRU-A, BRU-K and safety valves are required to prevent the first risk, while the second risk is required to be excluded by suitable EOPs and supporting analyses.

The potential for overfilling is generated in case of an uncontrolled PRISE. The high primary system pressure and the (unlimited) operation of ECCS pumps may cause overfill of the broken steam generator.

Three recommendations can be issued:

- The overfill of steam generator must be avoided at any stage of the PRISE accident: this condition has the potential to create other breaks in the steam generator tubes or in the pipelines connected with steam generator and to impair the working conditions of relevant components such as relief valves: BRU-A stuck open is an expected consequence of overfilling with consequent water hammer.
- If the overfill condition is calculated by a system code, caution must be taken in evaluating the results. Water hammer conditions, i.e. a local pressure spike, can be reliably predicted only if a suitable nodalization is set-up: the problem of numerical diffusion may impact greatly on the results.
- Structural mechanical calculations should be carried out to evaluate the consequences of local pressure spikes. This requires specific computational tools not considered within the present framework.

Severe accident

Again PRISE does not constitute the reference or the most dangerous scenario for severe accidents in nuclear power plants and in WWER-440 reactors. Nevertheless, a) the direct flow path core-to-environment (through the break and the relief valves) established following PRISE and, b) the operation of ECCS pumps that, without proper operator actions, work to empty water storage tanks, suggested the consideration of the severe accident evolution within the performed PRISE analyses.

The severe accident analyses within the PRISE should have two main targets:

a) To fix time limits for operator actions,
b) To identify the time at which substantial degradation of the core occurs.

The third important target, i.e. to calculate the system scenario following extended core damage, should be carried out by specific codes that may not have the same qualification level of thermohydraulic system codes widely adopted within the present framework. The calculations, in this case, must address items like optimization of severe accident management guidelines and mitigation of releases of radioactive material to the environment.

Uncertainty evaluation

Uncertainty methods have been used for addressing the errors expected from the application of thermohydraulic system codes to PRISE analyses. Key attention was devoted to the evaluation of the error in predicting the mass inventory values released through the break and from the BRU-A to the environment, even though uncertainties in the prediction of other parameters like pressure and temperatures were also computed. Sensitivity studies were also performed. The key results can be summarized as follows:

a) The selection of the most relevant accident scenario is a prerequisite for performing an uncertainty analysis: namely, hypotheses about operator actions affect the results to an extent much larger than the error expected from the application of uncertainty methods. In this connection, the realism of the selected scenario in terms of probabilistic safety analysis should be evaluated as well.
b) Two uncertainty methods were applied, the method developed by the GRS the CIAU method: both demonstrated their maturity in the pioneering application to PRISE

conditions. However, available resources (man-months and time) did not allow the full exploitation of the capabilities of uncertainty methods.

c) Keeping in mind the above, the studies performed showed an error in the prediction of mass released to the environment, at about two hours from the transient beginning, of the order of 40% of the best estimate value. The time error in predicting the occurrence of a selected event was estimated to be of the order of +/- 15% of the physical transient time.

The recommendation here is to consider carefully the step a) above and to apply the uncertainty method to optimize the EOPs and operator interventions: on the one hand conservative calculations can be carried out to demonstrate compliance with regulatory thresholds (e.g. maximum releases to the environment), on the other hand uncertainty method applications are necessary to demonstrate whether one EOP is better than another.

ABBREVIATIONS

BRU-A	Steam bypass station to atmosphere
BRU-K	Steam bypass station to condenser
CFD	Computational fluid dynamics
CIAU	Code with the capability of internal assessment of uncertainty
ECCS	Emergency core cooling system
EOP	Emergency operating procedure
FW	Feed water
HPSI	High pressure safety injection
I&C	Instrumentation and control
MGV	Main gate valve
MSH	Main steam head
MSIV	Main steam isolation valve
PRISE	Primary to secondary system leakage accident
PRZ	Pressurizer
PTS	Pressurized thermal shock
RCS	Reactor coolant system
RPV	Reactor pressure vessel
SG	Steam generator
SV	Safety valve

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