SIMULATION OF A SMALL COLD LEG BREAK EXPERIMENT ON PMK-2 TEST FACILITY USING THE CODES RELAP5 AND ATHLET

GY. ÉZSÖL, A. GUBA, and L. PERNECZKY *KFKI - Atomic Energy Research Institute, Budapest, Hungary*
E. KREPPER, H.-M. PRASSER, and F. SCHÄFER *FZR - Research Center Rossendorf Inc., Germany*

PUBLISHED IN NUCLEAR TECHNOLOGY VOL. 118 MAY 1997 PP. 162-174

Results of a small break loss of coolant accident experiment, conducted on the PMK-2 integral type test facility are presented. The experiment simulated a 1% break in the cold leg of a VVER-440-type reactor. The main phenomena of the experiment are discussed and in case of selected events a more detailed interpretation with the help of measured void fraction, obtained by a special measurement device is given. Two thermohydraulic computer codes, RELAP5 and ATHLET, are used for post test calculations. The aim of the presented calculations is to investigate the code capability for modeling natural circulation phenomena in VVER-440-type reactors. Therefore the results of the experiment and both calculations are compared. Both codes predict most of the transient events well, with the exception that RELAP5 fails to predict the dry-out-period in the core. In the experiment the hot and cold leg loop seal clearing is accompanied by natural circulation instabilities, which can be explained by means of the ATHLET calculation.

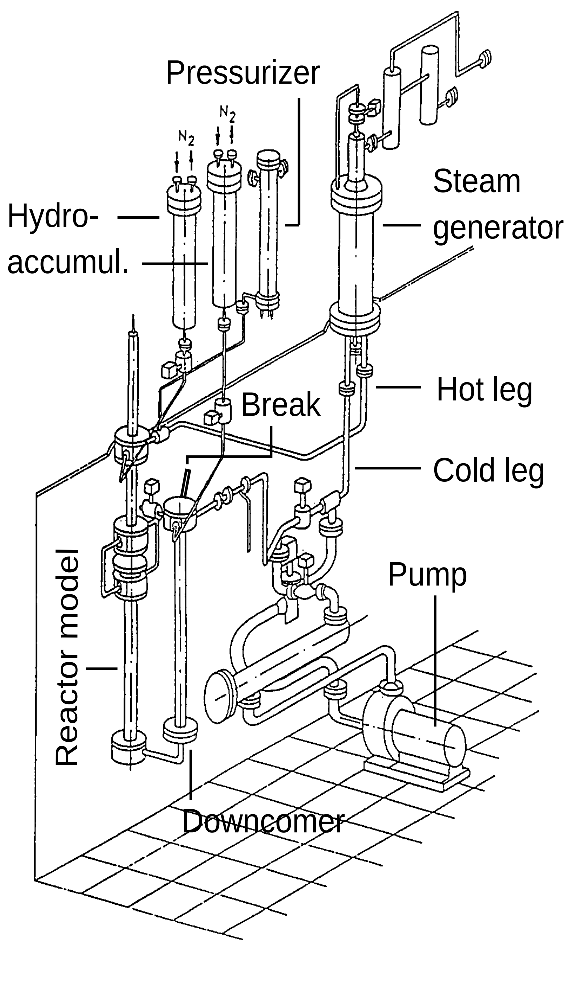

Fig.1: Axonometric view of PMK-2 facility

I. INTRODUCTION

An essential component of nuclear safety activities is the analysis of postulated accidents in nuclear power plants. Such analyses are usually carried out with complex thermohydraulic computer codes, which must be validated through the comparison of calculated results with experimental data. Computer codes like RELAP or ATHLET are developed for modeling western-type Nuclear Power Plants. To check the code capabilities for modeling an eastern-type reactor like VVER-440, pre- and post-test calculations of suitable experiments have to be performed. VVER-440-type reactors have a number of special features, e.g. horizontal steam generators and loop seals in both hot and cold legs. As a consequence of the differences, the

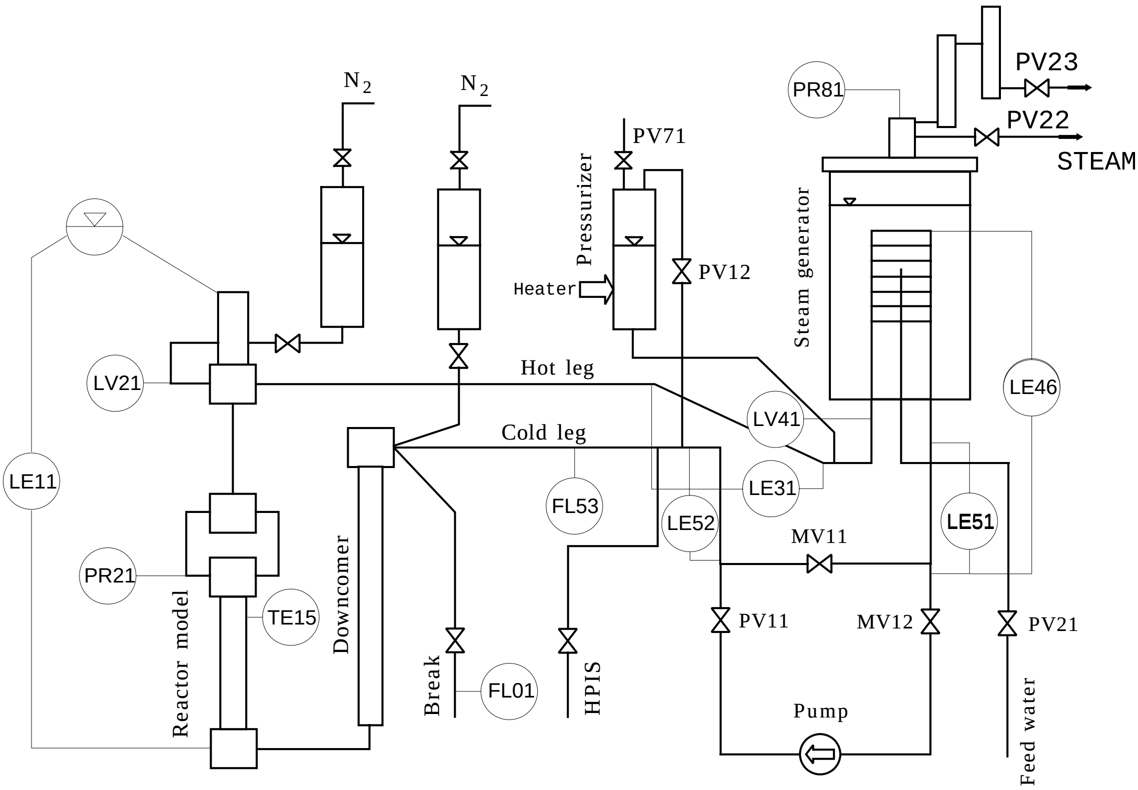

Fig.2: Block scheme of PMK-2 test facility with measurement positions

transient behaviour should be different from the usual reactor systems. The KFKI Atomic Energy Research Institute Budapest, Hungary designed and constructed the PMK-2 test facility, a downscaled model of the primary circuit of the VVER-440 type reactors of Paks Nuclear Power Plant (Fig.1). In the framework of the computer code assessment programme for the VVER-440 type Paks Nuclear Power Plant, a 1%-cold leg break experiment has been conducted on the PMK-2 integral type test facility. It was followed by calculations using RELAP5/Mod3.1 (Ref.1) and ATHLET Mod 1.1 Cycle A (Ref.2) in order to assess code capabilities.

This experiment was started from nominal operational parameters and it was considered that only the high pressure injection system (HPIS) is available and there is no injection from the safety injection tanks (SIT). The experiment was the repetition with improved data acquisition system of a test conducted in 1990 (Ref.3).

II. FACILITY DESCRIPTION

The PMK-2 test facility is a full-pressure, 1:2070 volume-scaled model of the Paks Nuclear Power Plant and designed mainly to investigate processes following small and medium size breaks in the primary circuit and to study the natural circulation behaviour of VVER-440 type reactor (Ref.4 and 7-9). The elevation ratio is kept 1:1.

A block scheme of the PMK-2 test facility is given in Fig.2. The six loops of the plant are modelled by a single active loop. The pump is installed in a bypass line. During steady state operation, valve MV11 is closed and circulation takes place through the bypass line. Pump trip modeling is achieved by controling the pump flow rate with the valve PV11. After pump coast down PV11 is fully closed, MV11 is opened and finally the bypass line is turned off from the loop by closing MV12. The core model consists of a 19-rod bundle with axially and radially uniform power distribution. The flow channel is made of ceramics (Fig.3). The fuel rods have an electrically heated length of 2.5m and a diameter of 9.1mm.

The horizontal steam generator is shown in Fig.4. It consists at the primary side of a hot and cold collector and 82 heat transfer tubes. For the injection of feed water at proper elevation a perforated tube is used. The secondary circuit is represented by the feed water and steam lines. On the secondary side of the steam generator the steam/ water volume ratio is kept constant. The main characteristics of the facility are given below.

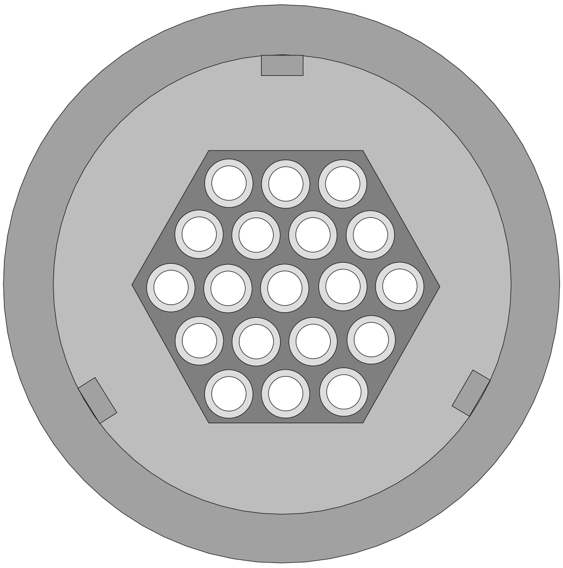

Fig. 3: Core model (cross section)

Reference NPP:

Paks Nuclear Power Plant
VVER-440/213 reactor - 6 loops
1375 MW th - hexagonal fuel arrangement

General scaling factor:

Power, volumes: 1:2070
Elevations: 1:1

Primary coolant system:

Pressure: 12.4 MPa
Core inlet temperature: 540 K
Core power: 664 kW
Nominal flow rate: 4.5 kg/s

Secondary coolant system:

Pressure 4.6 MPa
Feed water temperature: 493K
Nominal steam mass flow: 0.36 kg/s

Safety injection systems:

High pressure injection system (HPIS)
setpoint at 11.59 MPa + 37s
Low pressure injection system (LPIS)
setpoint at 1.04 MPa
Safety injection tanks (SIT)
setpoint at 6.01 MPa
Emergency feed water system

Measurement instrumentation:

Pressure (PR), differential pressure (DP)
Temperature (TE)
Level (LE), flow rate (FL)
Density (DE), local void fraction (LV)

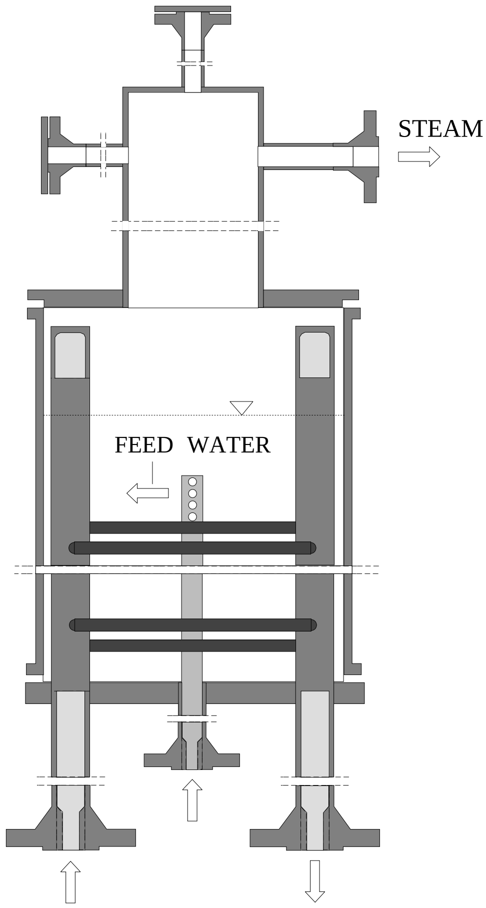

Fig. 4: Steam generator model

During the experiments, needle shaped conductivity probe devices, developed by the Research Center Rossendorf, were applied. The needle shaped conductivity probes are local void fraction sensors. Their function is based on the interruption of the electrical current flowing between the tip of the probe and the conducting fluid by the gas fraction. The void fraction is determined by integrating the time of the gas contact divided by the measuring time (Ref.5 and 6). The insulation tips of the Rossendorf needle probes are made from sintered Aluminium Oxide (Al_2O_3) ceramic (Fig.5), in order to withstand the high mechanical and corrosive loads during the test.

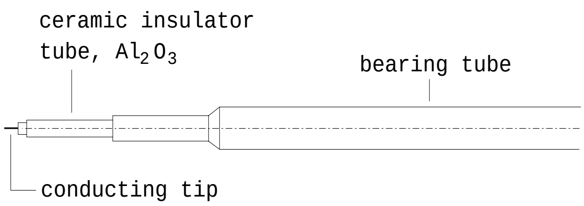

Fig.5: Needle shaped conductivity probe

III. EXPERIMENT DESCRIPTION

The test is characterized as follows (Ref.9). The break nozzle has a diameter of 1mm (modeling a 1% break in the Paks NPP) and is located on the upper head of the downcomer. The modeling of the HPIS flow corresponds to the case when only one of the three systems is available. The unavailability of the hydroaccumulator system is assumed. Transient is initiated by opening the break valve. The secondary side is isolated after starting the transient by closing valves PV21 and PV22. The initial steady state conditions for the test and the sequence of events during the course of transient are presented in Tab.1 and Tab.2.

IV. CALCULATIONS

IV.A. RELAP5 CALCULATION

The post-test RELAP5 calculations have been performed by use of the code version RELAP5/MOD3.1 (Ref.1) available in the framework of the international CAMP program of the US NRC and implemented at the KFKI Atomic Energy Research Institute on the IBM RISC-6000 type computer. The nodalization of the PMK-2 facility used for the calculation is shown in Fig.6. The nodalization scheme consists of 109 volumes including 12 time dependent volumes, 118 junctions including 5 time dependent junctions and 82 heat structures with a total number of 355 mesh points. This nodalization scheme is derived from the scheme used for IAEA-SPE-4 (Ref.4) analyses. The scheme considers break as a trip valve (618). To model both the steam generator relief valve and the safety valve trip valves were used (600, 605).

Several cross flow junctions have been used to model the most critical connections of the facility:

- cold leg - downcomer head,
- downcomer - vessel,
- UP 1 (230) - UP 2 (235),
- UP 2 (235) - UP 3 (240),
- UP 6 (246) - hot leg,
- steam generator secondary - at feedwater injection level.

The steady state control system for pressurizer pressure was used to achieve the desired initial conditions for the transient calculation. The end of the steady-state calculations was at 100 s process time.

The main parameters at the end of the steady-state calculation are presented in Tab.1. For the heat losses a convective boundary condition was calculated in all wall heat structures with a heat transfer coefficient of 5 W/m^2K. The value used for both subcooled and two-phase discharge coefficients of break junction (Ref.1) is 0.85. Loss coefficient in break junction (Ref.2) is 5.0.

An overview about the main occurences is given in Tab.2

IV.B. ATHLET-CALCULATION

The ATHLET calculations for the 1%-cold leg break experiment are performed at the Research Center Rossendorf on a Sun Workstation SPARC 10/40. For the calculations the thermohydraulic code ATHLET Mod 1.1 Cycle A (Ref.2) is used.

The complete PMK model consists of 104 control volumes, 109 junctions and 126 heat conduction volumes. The nodalization scheme is shown in Fig.7. In most control volumes, the flooding based drift model is applied. The wall friction is considered by using the Martinelli-Nelson friction model. To calculate the flow out of the break, a one dimensional four equations critical discharge model (CDR1D) is applied, which allows the consideration of thermal nonequilibrium. This model is available within an independent code. It calculates tables of critical mass fluxes and corresponding pressures and fluid densities at the break plane, depending on the fluid conditions in the upstream discharge control

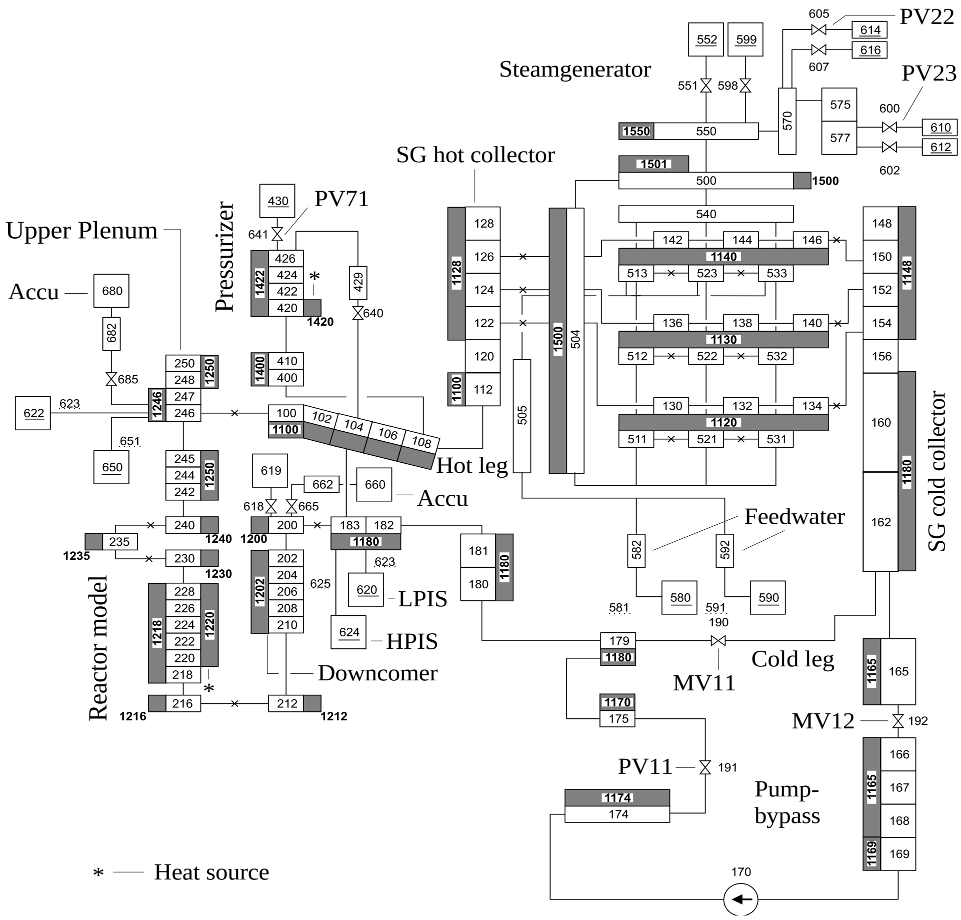

Fig.6: Nodalization scheme for RELAP5/MOD3.1

TABLE I: Measured and calculated initial conditions

Parameter	Experiment	RELAP	ATHLET
Pressure in upper plenum	12.43 MPa	12.46 MPa	12.42 MPa
Loop mass flow rate	5.10 kg/s	5.10 kg/s	5.13 kg/s
Core inlet temperature	536.4 K	540.4 K	538.5 K
Core outlet temperature	565.0 K	565.2 K	563.5 K
Core power	658.0 kW	658.1 kW	658.0 kW
Collapsed pressurizer level	9.02 m	9.08 m	9.03 m
Secondary side pressure	4.51 MPa	4.50 MPa	4.51 MPa
Collapsed steam generator level	7.83 m	8.06 m	8.12 m
Feedwater mass flow rate	0.348 kg/s	0.350 kg/s	0.350 kg/s
Feedwater inlet temperature	496.2 K	496.2 K	496.2 K

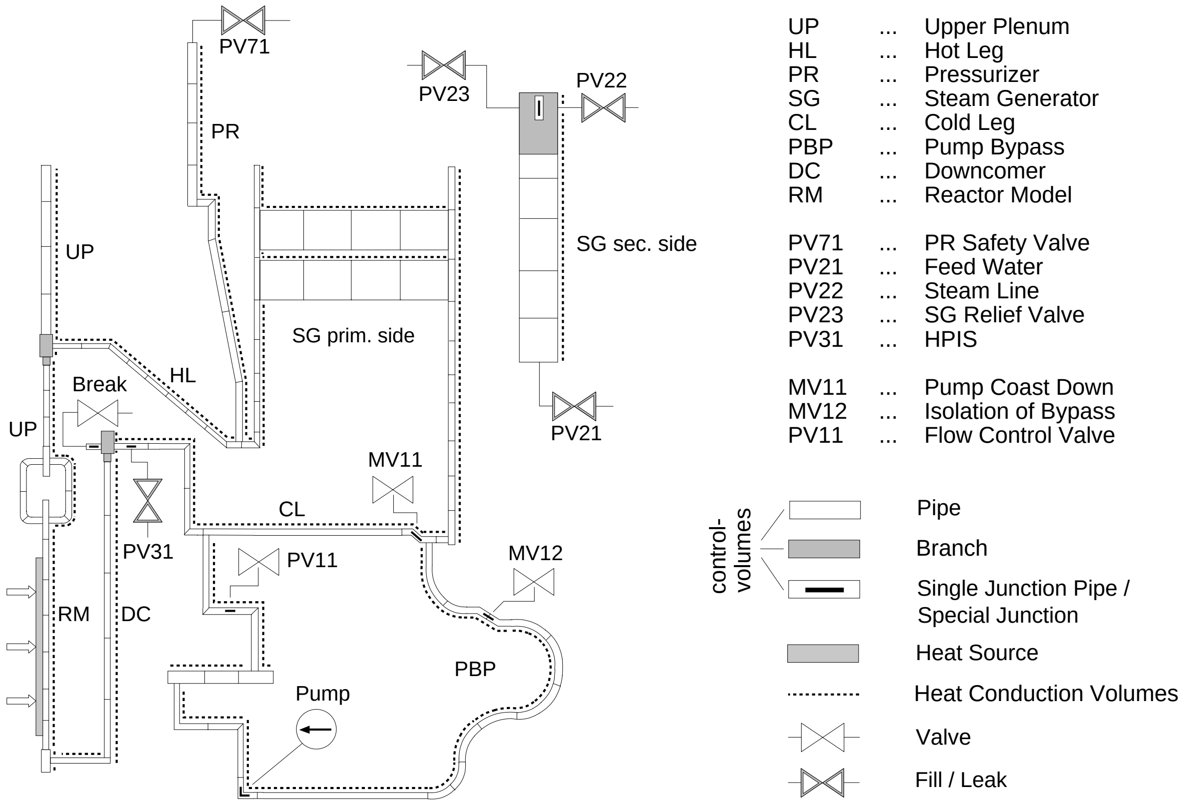

Fig.7: Nodalization scheme for the ATHLET-code

TABLE II: Measured and calculated occurences

Occurences	Experiment	RELAP	ATHLET
Break valve opens	0 s	0 s	0 s
Steam generator relief valve opens	41 s	24.8 s	38 s
Scram and HPIS flow initiated	65 s	63.6 s	57 s
Pump trip simulation initiated	74 s	75.3 s	80 s
Steam generator relief valve closes	150 s	109.8 s	165 s
Pressurizer empty	180 s	145 s	130 s
Level in upper plenum drops to hot-leg elevation	640 s	504 s	535 s
Hot-leg loop seal cleared	750 s	762 s	776 s
Core uncovery begins	1737 s	-	1815 s
Cold-leg loop seal cleared	1806 s	1765 s	1846 s
Test terminated at	3998 s	4000 s	4000 s

volume. The tables generated by the CDR1D model are used as a part of the input data file for the ATHLET code. The pump coast down is simulated by closing the valve PV11. To reproduce the correct mass flow in the loop, for the calculation an approximately linear decrease of the valve cross section during 150s is assumed. The given time-dependence of the pressure difference of the pump is considered in modeling the pump behaviour.

Different kinds of nodalizations have been tested for modeling the steam generator. Best results have been achieved by modeling the steam generator with two tube bundles for the primary side, including all 82 pipes of the steam generator model.

Before starting the transient, a steady state calculation at stationary boundary conditions is performed over 1000 seconds. The initiation of power scram, pump coast down and the start of high pressure injection system (HPIS) are controled by the primary pressure. The time-dependence of the reactor power is assumed according to the decay heat curve. The HPIS is modeled as a fill with a constant mass flow rate.

V. COMPARISON OF THE RESULTS

The measured and calculated parameters selected for this report are given in Fig.8-18. Fig.2 should be used for identification of the measurement positions.

The time-dependence of the primary pressure is shown in Fig.8. By opening the break valve, a fast decrease of the system pressure can be observed. This pressure decrease is accelerated due to the reactor power scram. During this time, a fast increase of the secondary pressure (Fig.9) can be observed, reaching the setpoint of the steam generator relief valve PV23. The decrease of the primary pressure is reduced by a lower heat transfer to secondary side, after closing the valve PV23. The calculations provide a good qualitative agreement to the primary pressure up to approximately t=200s. Deviations between the ATHLET calculation and the experiment are caused by the influence of modeling the pump coastdown. Due to the higher heat transfer from the primary to the secondary side, in the RELAP5 calculation the steam generator relief valve PV23 opens again for a short period at t=261 s. This discrepancy between experiment and RELAP5 calculation can be attributed to the heat transfer model, which is not validated for horizontal steam generators.

The decreasing RCS mass inventory leads to boiling in the core after approximately t=600 s. This and the reduced heat removal to the steam generator secondary side results in an increase in the primary pressure. The hot leg loop seal level (LE31, Fig.10) begins to decrease. After reaching its minimum, the hot leg loop seal clearing takes place and steam generated in the core enters the steam generator hot collector (LV41, Fig.11). At the same time the level in the steam generator hot collector starts to drop. After that, both calculation and experiment show significant oscillations in the primary pressure, the levels, mass flow rate and void fractions with approximately the same time period, see Fig.11-13. The results of the ATHLET calculation make it possible to give an explanation for this kind of instabilities.

As a consequence of condensation in the steam generator inlet the primary pressure decreases. The steam flow from the reactor to the steam generator leads to an increase of the mass flow rate (FL53) and even the reactor level (LE11) increases. The rise of the reactor level leads to a decreasing void fraction at reactor outlet and hot leg and as a result there is less condensation in the steam generator. The phase-shift between void fraction at reactor outlet and steam generator inlet amounts to 180 degrees. As shown in Fig.12 and 13 the primary pressure reaches a local minimum and for a short period the mass flow rate (FL53) is negative. Because of limitations of the measurement device, the experimental mass flow rate could not show negative values. The calculation shows that there is a fluid mass flow directed from the steam generator inlet to the hot leg. This fluid mass flow and the rising water level in the reactor leads to a refilling of the hot leg loop seal from both ends. Once again, the primary pressure increases and the described process is repeated periodically. Due to the hot leg loop seal clearing, the primary pressure decreases after reaching a maximum. This effect is calculated very well by the ATHLET and the RELAP5 code (Fig.8).

During the experiment, an extended dry out period occurs in the core. This dry out phenomena connected with a high temperature excursion is calculated by ATHLET but not by RELAP. The decrease of the reactor level leads to a rising cladding temperature (TE15,

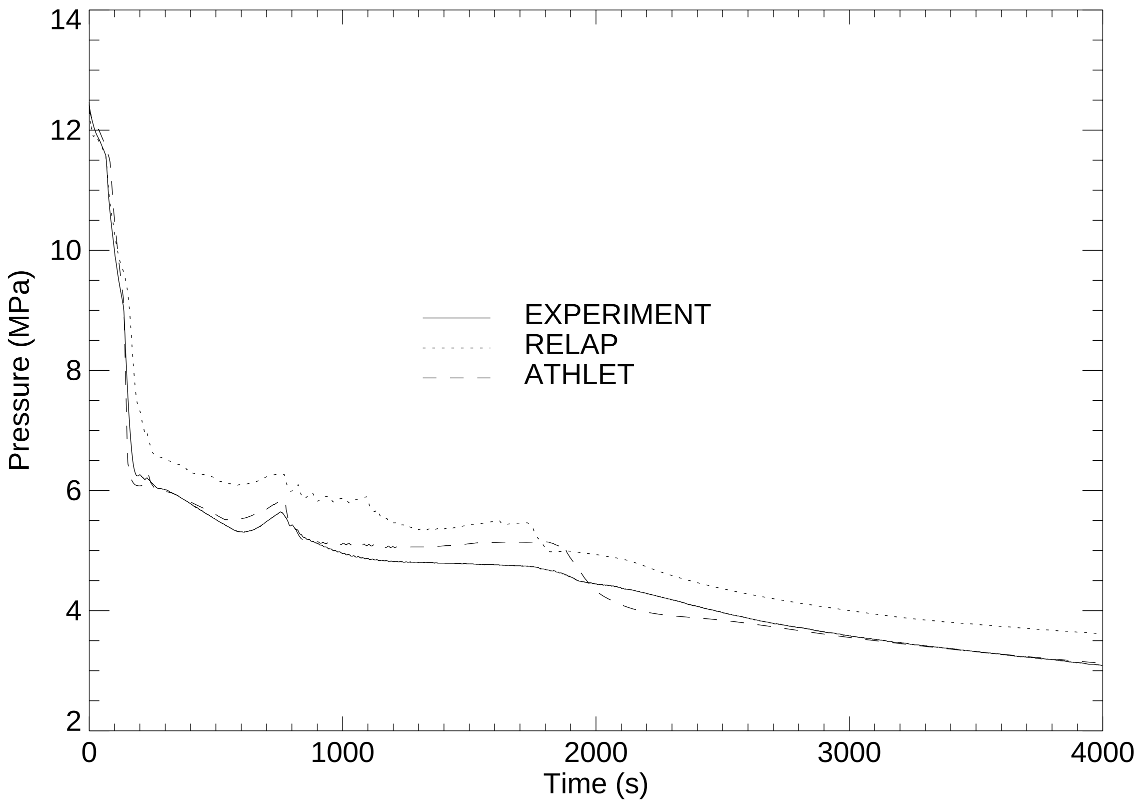

Fig.8: Pressure in primary circuit (PR21)

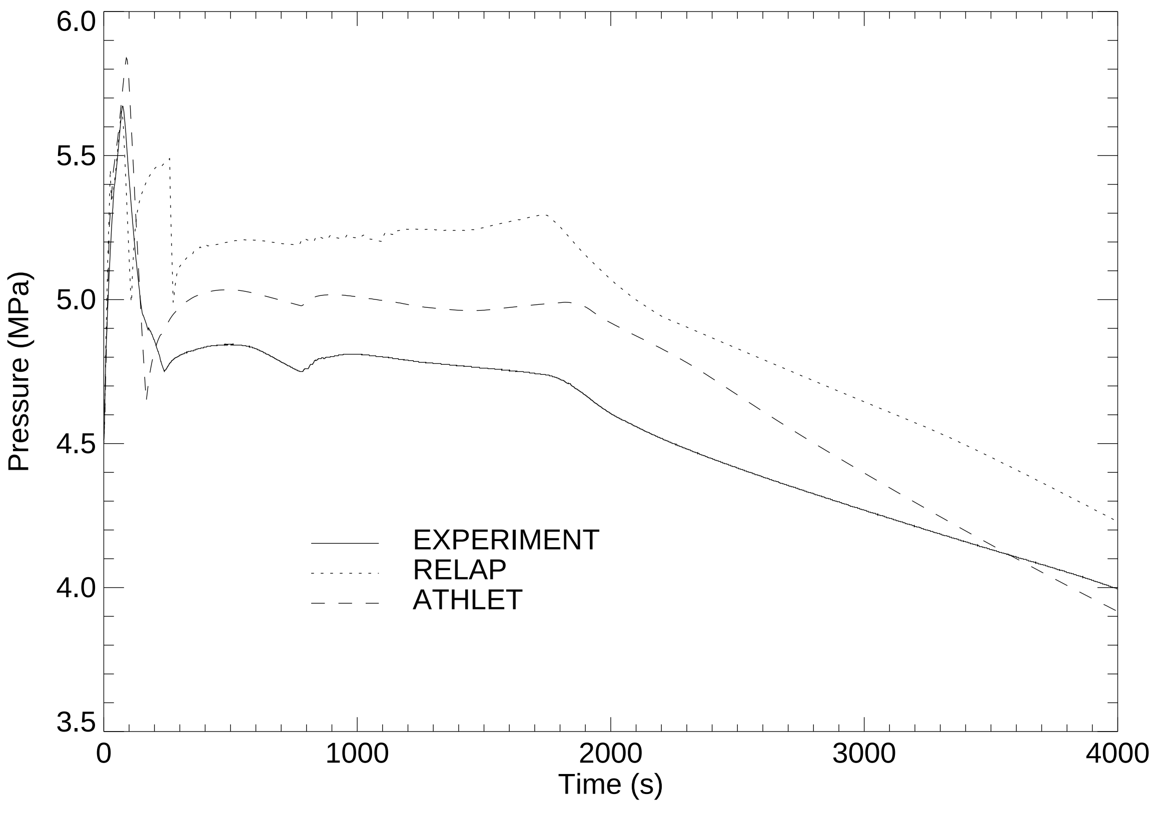

Fig.9: Pressure in secondary circuit (PR81)

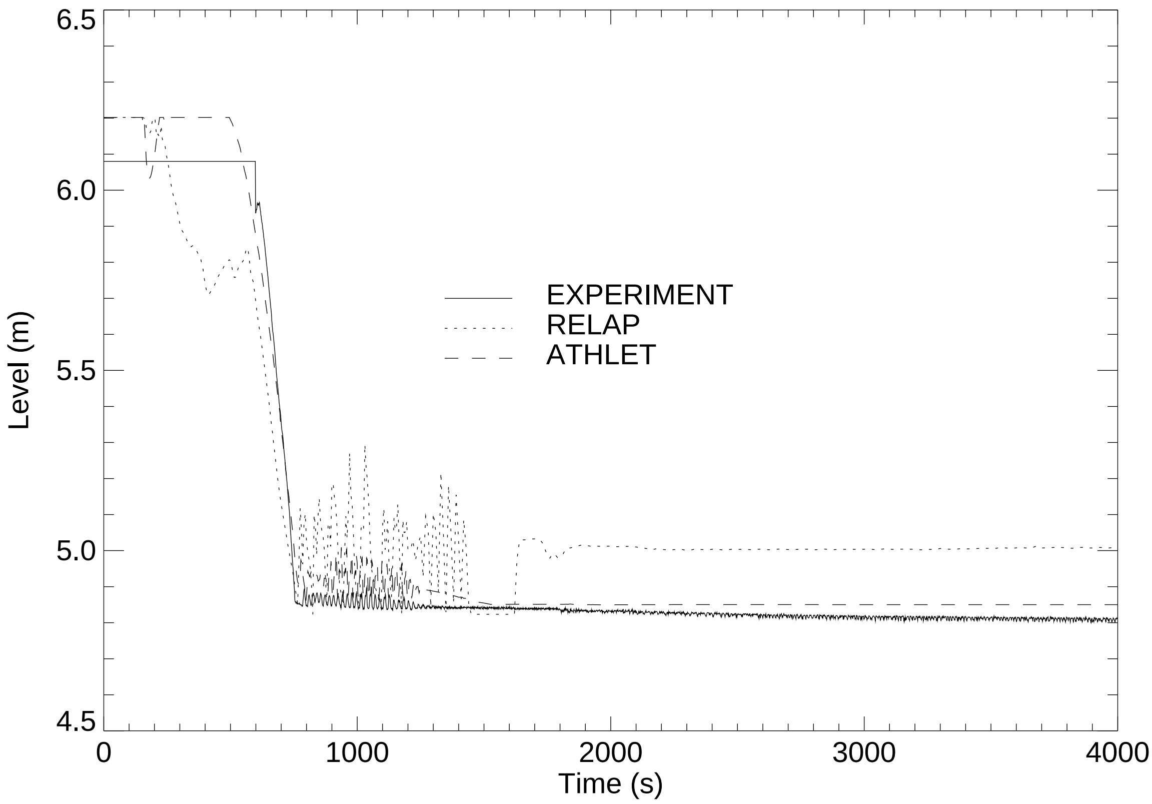

Fig.10: Level in the hot leg (LE31)

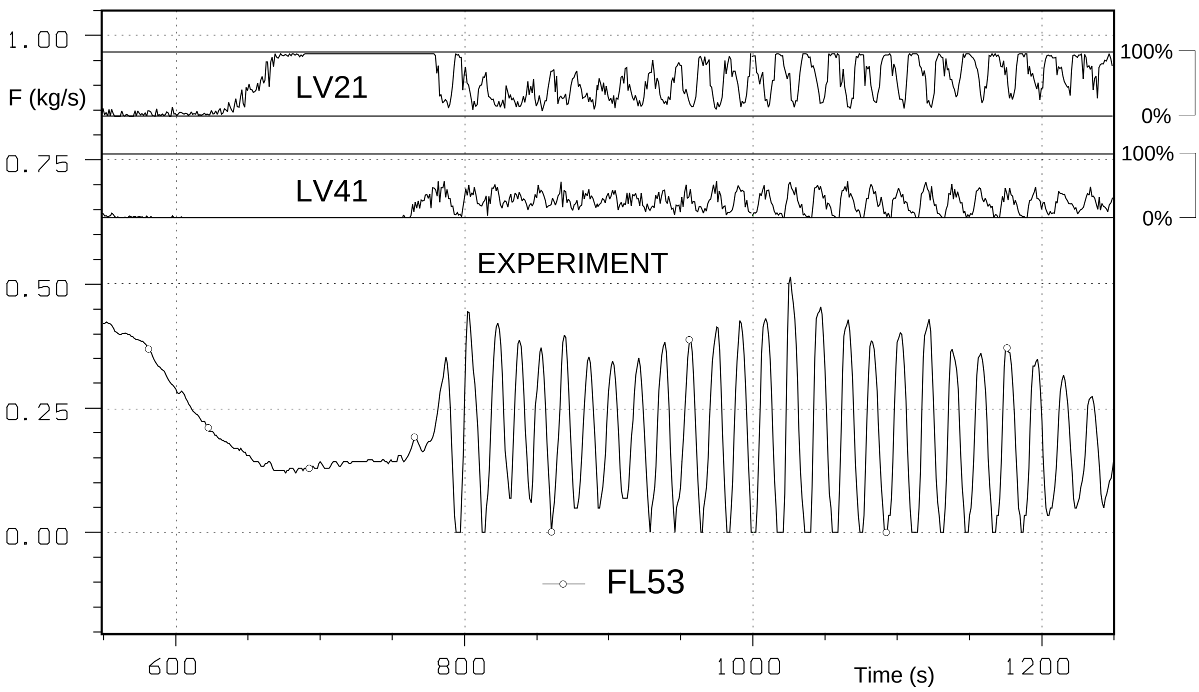

Fig.11: Oscillations in mass flow rate (FL53) an void fraction - experiment

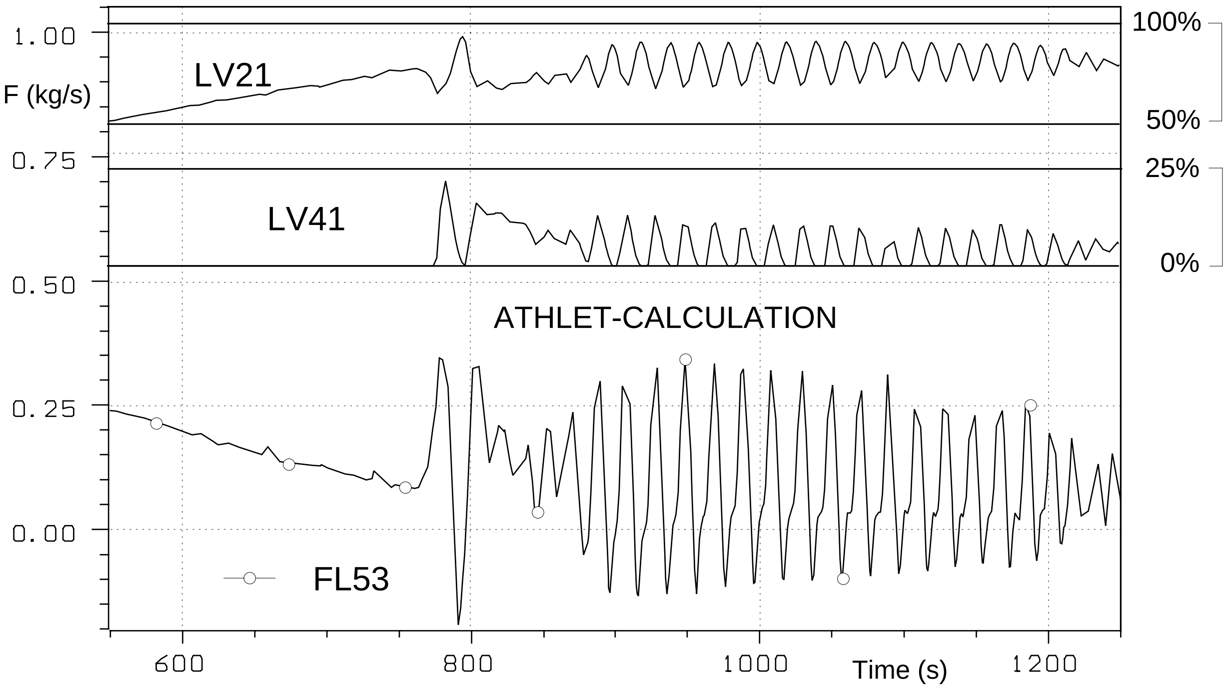

Fig.12: Oscillations in mass flow rate (FL53) and void fraction - calculation

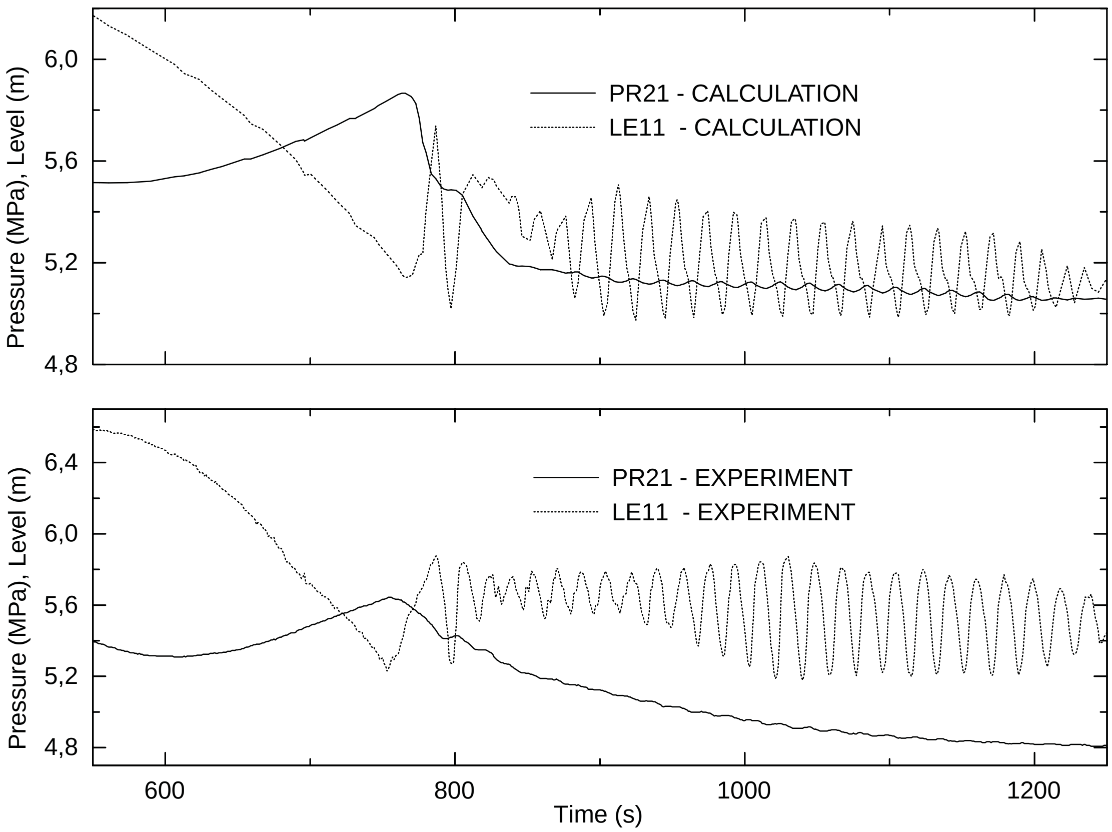

Fig.13: Oscillations in primary pressure (PR21) and reactor level (LE11)

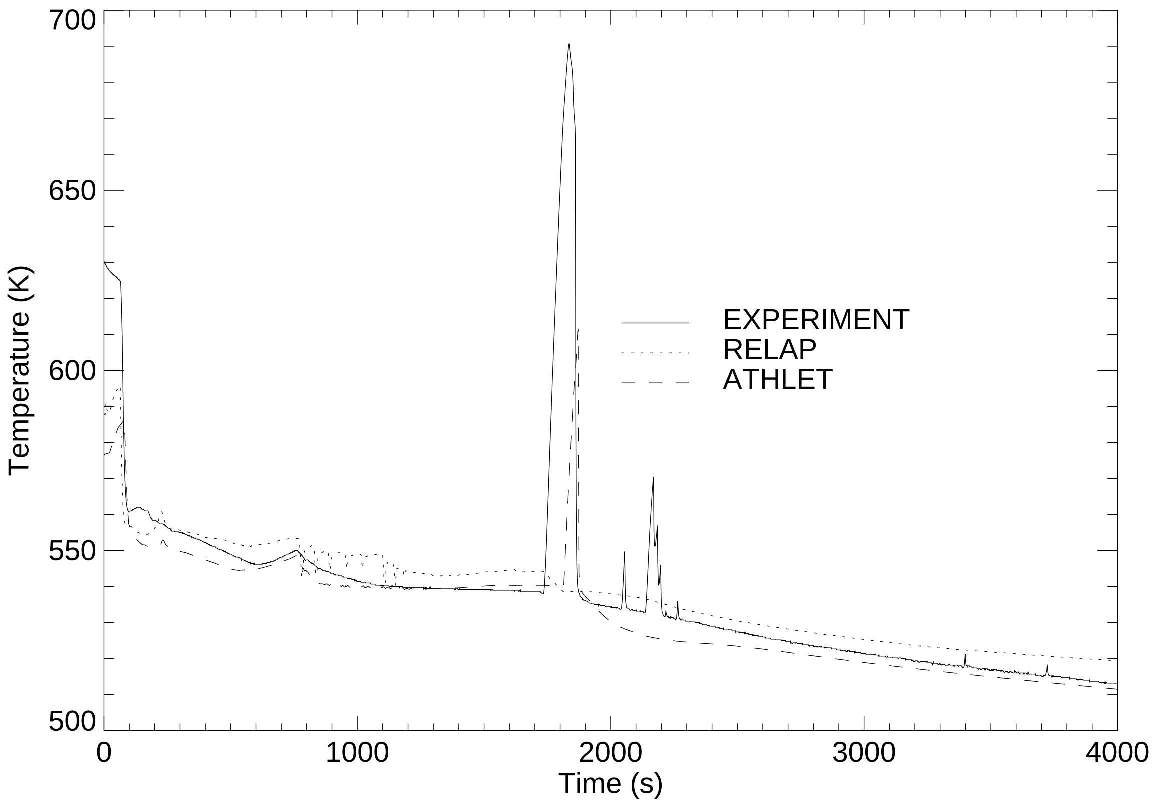

Fig.14: Cladding temperature (TE15)

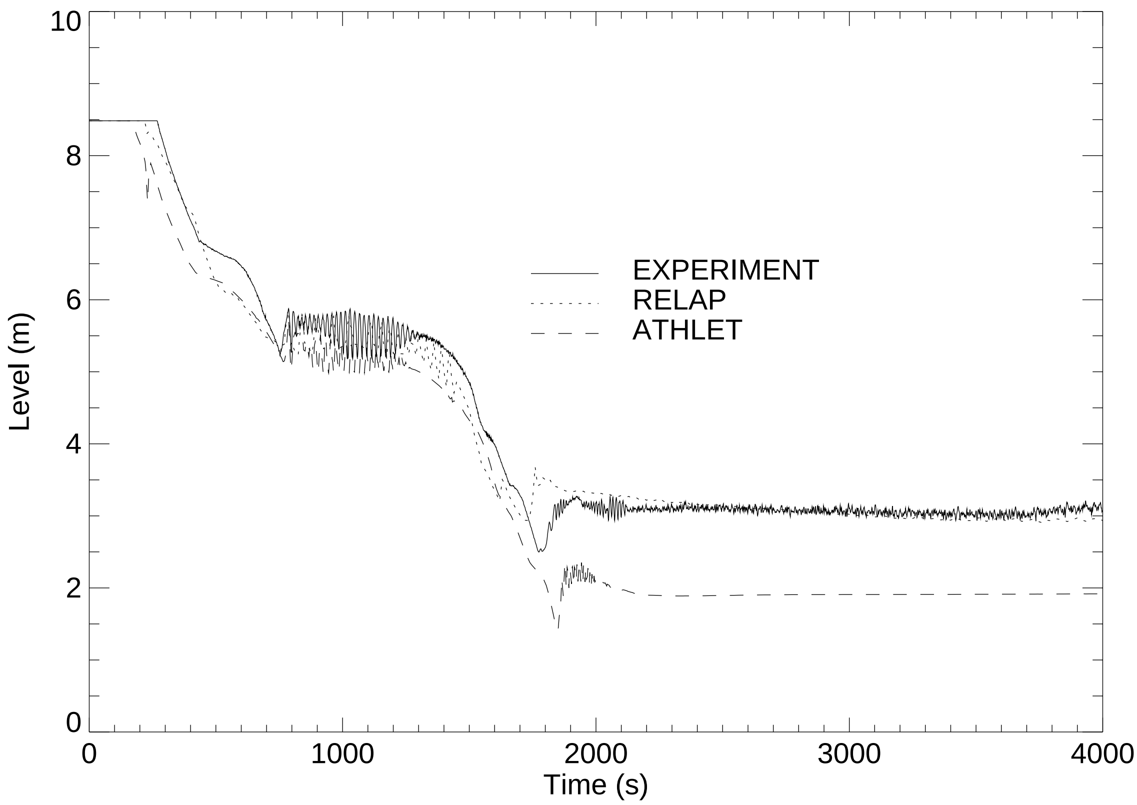

Fig.15: Level in reactor model (LE11)

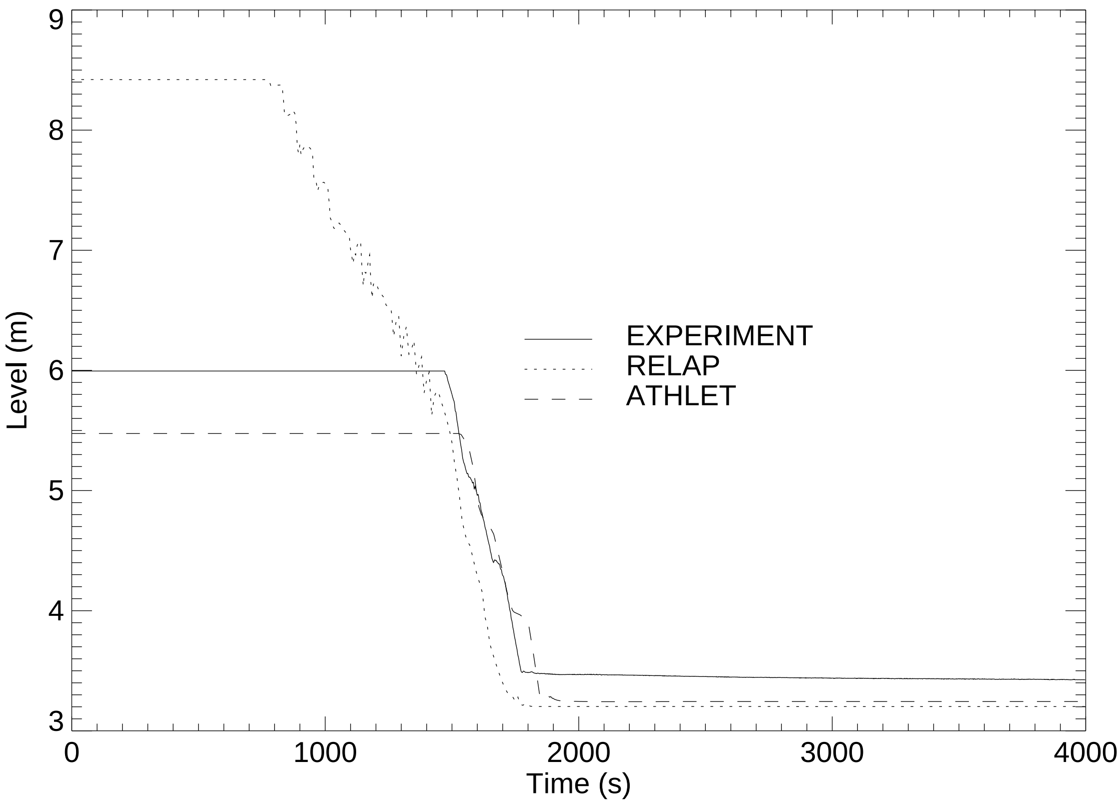

Fig.16: Level in cold leg steam generator side (LE51, RELAP5 includes LE46)

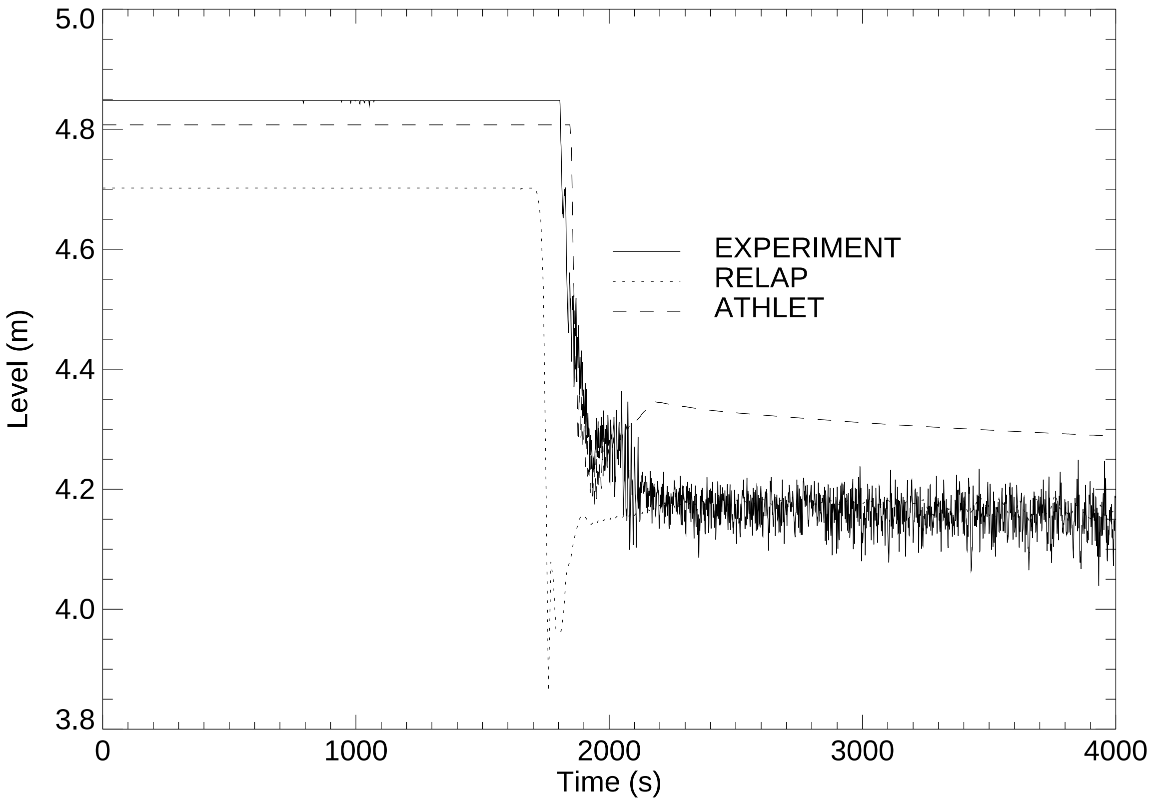

Fig.17: Level in cold leg reactor side (LE52)

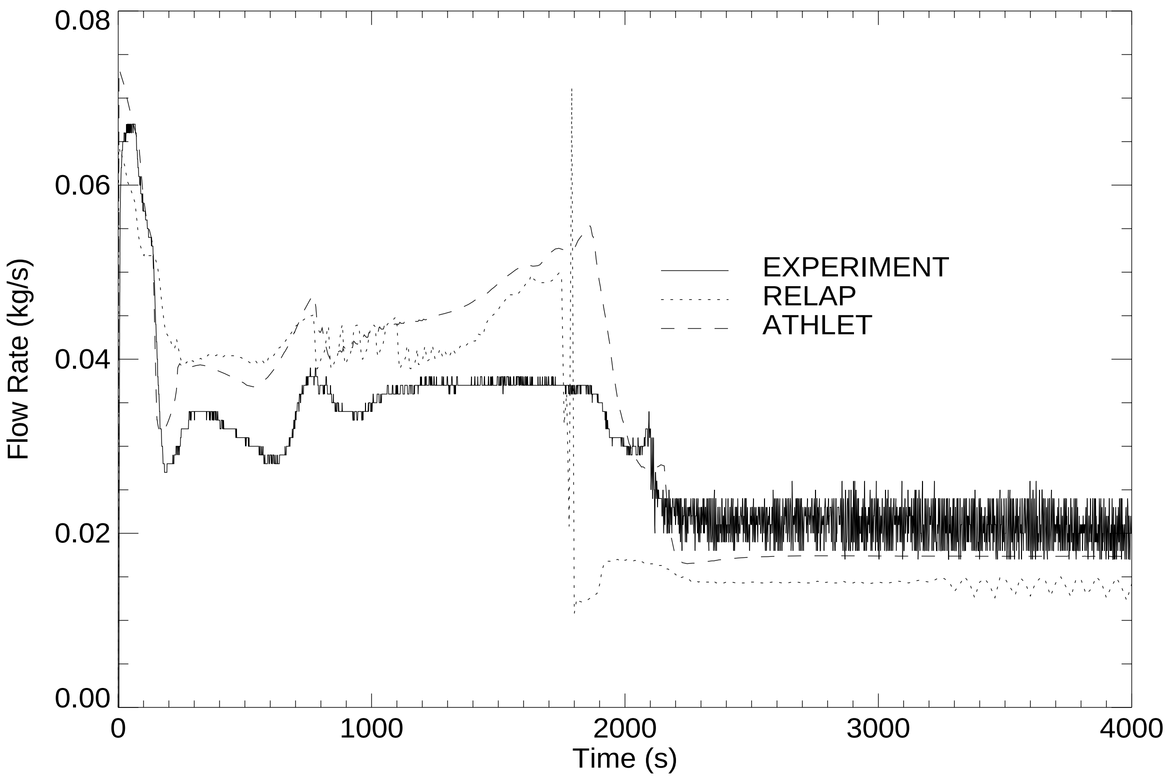

Fig.18: Break mass flow rate (FL01)

Fig.14) from 540K up to approximately 690K. The failure to predict dry-out also has been observed in other simulations using the RELAP5 code (Ref.12). Perhaps it is a general problem, caused by deficiencies in the heat transfer model.

The last significant event is the cold leg loop seal clearing, caused by decreasing levels due to continuous liquid leakage from the break. In the experiment, at about t=1500s the level of the cold leg steam generator side (LE51, Fig.16) starts to decrease and drops down to a minimum. After reaching this minimum the cold leg loop seal clearing takes place and the level in the cold leg reactor side starts to drop (LE52, Fig.17). The reactor level reaches it's minimum (LE11, Fig.15) and by the steam flow out of the steam generator, fluid from the cold leg flows directly to the core. The reactor level rises again and so the dry out period is limited. In the ATHLET calculation the reactor level (LE11) reaches a lower minimum. In this way the dry out period can be modeled by the ATHLET code. In the calculation, the dry out occurs at t=1815s instead of t=1740s in the experiment. For this reason the dry-out-period lasts only 60 seconds and the maximum cladding temperature is lower than in the experiment. Although level LE11 reaches a very low minimum, a dry out in the cladding temperatures is only calculated in the upper part of the core.

During the cold leg loop seal clearing in both calculations a sharp pressure decrease can be observed, caused by condensation effects during partial refilling of the core. As was seen in LE11 (Fig.15), after the cold leg loop seal clearing a similar type of oscillations like after the hot leg loop seal clearing can be observed in the experiment and also in the ATHLET calculation. The oscillations in levels LE11, LE52 and also in the downcomer level are induced by the cold leg loop seal clearing and can be explained by periodically fluctuations of the fluid mass between cold leg and reactor model. This oscillations are not calculated by RELAP5.

Up to the end of the experiment there is practically a balance between the mass flow rate out of the break and the HPIS mass flow rate. Measured and calculated break flows are presented in Fig.18. The primary pressure decreases slowly and the reactor level stagnates approximately at a constant value. After the hot leg loop seal clearing, the mass flow rate in the loop is practically zero, except the time period of oscillations.

VI. CONCLUSIONS

The study of the 1% cold leg break experiment is one part of the cooperation between the KFKI Atomic Energy Research Institute, Hungary and the Research Center Rossendorf, Germany.

The experiment conducted at the PMK-2 test facility in Budapest is used for the verification of thermohydraulic computer codes. Generally both the ATHLET and RELAP5 codes are capable of calculating all main phenomena of the experiment, with the exception that RELAP5 fails to calculate the dry-out-period in the core and the oscillations after the cold leg loop seal clearing. The calculated results show a good agreement with the measured data. Especially effects, typical for VVER-440 reactors, are calculated very well. It should be outlined, that the differences between experiment and RELAP5 calculation in case of the dry-out-period and the oscillations are not a general code limitation. Rather it is more a modeling problem and further investigations are required to study such phenomena.

For a better understanding of the experimental results, the local void fraction sensors, developed by the Research Center Rossendorf, are very useful. The sensors provide more detailed information about evaporation, condensation and other two-phase flow phenomena.

Further experiments are intended to investigate the code capabilities, i.e. a 1% cold leg break experiment with primary bleed and a 1% cold leg break experiment with hydroaccumulator injection.

REFERENCES

1. "RELAP5/MOD3 Code Manual," Vol. 1-5, NUREG/CR-5535, EGG-2596 (1990-1992)

2. "ATHLET MOD1.1 - CYCLE A. Input Data Description," Gesellschaft für Anlagen- und Reaktorsicherheit mbH Germany (February 1993)

3. L. PERNECZKY, G. ÉZSÖL, L. SZABADOS, "1% Cold Leg SBLOCA Analysis on PMK-NVH Facility," Central Research Institute for Physics, Budapest (1990)

4. "Simulation of a loss of coolant accident without high pressure injection but with secondary side bleed and feed," Report of the IAEA Technical Co-operation Project RER/9/004 on Evaluation of Safety Aspects of WWER-440 Model 213 Nuclear Power Plants. IAEA-TECDOC-848, Vienna (November 1995)

5. H.-M. PRASSER, L. KÜPPERS, R. MAY, "Conductivity Probes for Two-Phase Flow Pattern Determination During Emergency Core Cooling (ECC) Injection Experiments at the COCO Facility (PHDR)," Proceedings of the 1. OECD (NEA) CSNI - Specialist Meeting on Instrumentation to Manage Severe Accidents, Cologne, Germany, July 1992, NEA/CNSI/R(92)11, pp 273-289

6. H.-M. PRASSER, W. ZIPPE, D. BALDAUF, L. SZABADOS, G. ÉZSÖL, G. BARANYAI, I. NAGY, "Two-Phase Flow Behaviour during a Medium Size Cold Leg LOCA Test on PMK-II (SPE-4)," Annual Meeting on Nuclear Technology, Stuttgart, Germany, 1994, Proc. pp 77-80

7. "Simulation of a Loss of Coolant Accident," Results of a Standard Problem Exercise on the Simulation of a LOCA, IAEA-TECDOC-425. Vienna, 1987

8. "Simulation of a Loss of Coolant Accident with Hydroaccumulator Injection," Results of the Second Standard Problem Exercise on the Simulation of a LOCA. IAEA-TECDOC-477. Vienna, 1988

9. G. ÉZSÖL, A. GUBA, L. PERNECZKY, H.-M. PRASSER, F. SCHÄFER, E. KREPPER, "1% Cold leg break experiment on PMK-2 - Test results and computer code analysis," Forschungszentrum Rossendorf, FZR-76, March 1995

10. G. ÉZSÖL, A. GUBA, H.-M. PRASSER, F. SCHÄFER, "Small cold leg break experiment on PMK-2," Annual Meeting on Nuclear Technology, Nürnberg, Germany, 1995, Proc. pp 119-122

11. F. SCHÄFER, E. KREPPER, "Rechnungen zum 1%-Leck an der Versuchsanlage PMK-2 mit dem Code ATHLET," Annual Meeting on Nuclear Technology, Nürnberg, Germany, 1995, Proc. pp 79-82

12. P. P. CEBULL, Y. A. HASSAN, "Simulation of the IAEA's Fourth Standard Problem Exercise Small-Break Loss-of-Coolant Accident using RELAP5/MOD3.1," Nucl. Technol., **109**, 327 (1995)

György Ézsöl (BS and MS, physics, Eötvös Lóránd University of Science, Budapest, Hungary, 1973; PhD, nuclear engineering, Eötvös Lóránd University of Science, Budapest, 1976) is on the Department of Thermalhydraulics at KFKI - Atomic Energy Research Institute, Budapest. His interest include experimental (PMK-2) and computational research of nuclear reactor safety of VVER-440 type NPPs.

Attila Guba (BS and MS, mechanical engineering, Technical University of Budapest, Hungary, 1994) is on the Department of Thermalhydraulics at KFKI - Atomic Energy Research Institute, Budapest. His interest include simulation methods (NPA) and experiments (PMK-2) for nuclear reactor safety of VVER-440 type NPPs.

László Perneczky (BS and MS, mechanical engineering and thermal energetics, Technical University of Budapest, Hungary, 1961; MS, electrical engineering and process control, Technical University of Budapest, 1967; PhD - numerical methods and control, TU Budapest, 1972) is on the Department of Thermalhydraulics at KFKI - Atomic Energy Research Institute, Budapest. His interest include computational (RELAP5) and experimental (PMK-2) investigation of nuclear reactor safety of VVER-440 type NPPs.

Eckhard Krepper (BS and MS, physics, University of Leipzig, Germany, 1974; PhD, nuclear engineering, Central Institute for Nuclear Physics Rossendorf, Germany, 1989) is a senior assistant in the Institute of Safety Research at Research Center Rossendorf Inc., Germany. His background includes nuclear safety analyses with complex thermal-hydraulic computer codes and modeling of thermal-hydraulic processes with 3-dim. CFD-codes. He has participated in international code assessment programs and several International Standard Problems.

Horst-Michael Prasser (BS and MS, nuclear reactor thermophysics, Moscow Energetics Institute, Russia, 1980; PhD, nuclear engineering, College of Advanced Technology in Zittau, Germany, 1984) is head of the Department of Accident Analysis in the Institute of Safety Research at Research Center Rossendorf Inc., Germany. His current interests include fluid dynamic investigations of safety related processes in chemical and nuclear industry, two-phase flow instrumentation, experiments and computer simulations.

Frank Schäfer (BS, mechanical engineering, 1989, and MS, nuclear engineering, 1991, College of Advanced Technology in Zittau, Germany) is currently a PhD candidate in nuclear engineering at Technical University of Dresden, Germany. He is a research assistant in the Institute of Safety Research at Research Center Rossendorf Inc., Germany. His interests include thermal-hydraulic code assessment, safety analyses for VVER-440 reactors and natural circulation instabilities in PWRs under SBLOCA conditions.