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RESULTS OF 15 YEARS EXPERIMENTS IN THE PMK-2 INTEGRAL-TYPE FACILITY FOR VVERs

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INTRODUCTION

The Paks Nuclear Power Plant is equipped with VVER-440/213-type reactors. Such plants are slightly different from PWRs of usual design and have a number of special features, viz.: 6-loop primary circuit, horizontal steam generators (SG), loop seal in hot and cold legs, safety injection tank (SIT) set-point pressure higher than secondary pressure, coolant from SITs is directly injected to the upper plenum and downcomer. As a consequence of the differences the transient behaviour of such a reactor system is different from the usual PWR system behaviour.

To study the transient behaviour of this type of NPPs, to perform experiments for this special pressurised water reactor design the PMK integral-type facility, the scaled down model of the Paks NPP was designed and constructed in the early 1980s. The PMK-NVH facility was put into operation in 1985. This was the first and the only full pressure integral-type facility for VVERs, therefore there was a high international interest for the test results applicable for computer code validation. The abbreviation PMK-2 is used after an upgrading in 1990.

Since the start-up of the facility altogether 48 experiments have been performed for groups of transients as follows: one- and two-phase natural circulation, loss of coolant accidents (LOCA), special plant transients and experiments in support of the accident management (AM) procedures. The results have been used for the validation of thermal-hydraulic system codes like ATHLET, CATHARE and RELAP5.

The PMK-NVH/PMK-2 facility was used for experiments of four "Standard Problem Exercises" as SPE-1, SPE-2, SPE-3 and SPE-4 of the International Atomic Energy Agency (IAEA) in the time interval of 1985 to 1995 primarily for SBLOCA-type transients. In the time interval of 1996 to 2000 several PHARE projects have been performed with the aim of obtaining additional experimental data to support among

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others the development and qualification of AM measures. Another group of measurements supported further needs of the safety improvements programme of the Paks NPP.

The paper will give a short description of the PMK-2 to facilitate the understanding of the experimental result, a summary of the different types of experiments with the evaluation of the results and the computer code validation.

SHORT DESCRIPTION OF THE PMK-2

The PMK-2 [1] is a scaled-down model of the Paks Nuclear Power Plant equipped with VVER-440/213-type reactors of Soviet design. It is a full pressure model of the plant with a volume and power scaling of 1:2070. Due to the importance of gravitational forces in both single- and two-phase flow the elevation ratio is 1:1 except for the lower plenum and pressuriser (PRZ). The six loops of the plant are modelled by a single active loop. The coolant is water under the same operating conditions as in the plant, so transients can be started from nominal operating conditions.

The core model consists of 19 electrically heated rods with uniform power distribution. In the core the heated length, spacer type and elevations, as well as the channel flow area are the same as in the Paks NPP. The main circulating pump of the PMK-2 serves to produce the nominal operating conditions and to simulate the flow coast-down following pump trip. The pump cannot be applied to two-phase conditions; therefore it is accommodated in a by-pass line. The flow coast-down is modelled by closing a control valve. For natural circulation the by-passed cold leg part is opened. The horizontal design of the VVER-440 steam generator is modelled by horizontal heat transfer tubes between hot and cold vertical collectors in the primary side. In the secondary side of the steam generator the steam/water volume ratio is maintained. From the emergency core cooling systems (ECCS) the four SITs of the Paks NPP are modelled by two vessels. They are connected to the downcomer and upper plenum similar to those of the reference system. The high and low-pressure injection systems (HPIS and LPIS) are modelled by the use of piston pumps.

In the first design of the PMK-NVH facility only the primary circuit of plant was modelled. This version was used until 1990. The PMK-2 facility is an upgraded version - first of all by addition of a controlled secondary heat removal system - extending the capability of the test loop in modelling transient processes evoked by initiating events in the secondary circuit.

Fig.1. Shows a simplified scheme of the PMK-2 with the locations of the measured parameters, e.g. PR21, TE15, etc., selected for this paper. For the abbreviations see Figs. 2 to 13. The reference level of 0.00 m and the levels of the main components of the facility are also shown in Fig.1.

PROJECTS FOR PMK EXPERIMENTS

Since 1985 various projects were launched and a wide range of different experiments have been performed. Groups of tests are as follows:

IAEA Standard Problem Exercises. Four experiments have been performed in the IAEA framework to provide a possibility to the interested international community for joint code validation exercises. In the 1st (SPE-1 [2]), 2nd (SPE-2 [3]), and 4th (SPE-4 [5]) Exercises tests were SBLOCA tests, with a cold leg break size of 7.4% with different availability of ECC systems. The 3rd (SPE-3 [4]) exercise was a VVER-specific case namely the opening of the steam generator (SG) hot collector cover. In the exercises 26 countries participated and an extensive validation of thermal-hydraulic system codes was performed.

Tests in EU PHARE projects. Ten different experiments have been performed in the framework of five PHARE projects to validate codes as ATHLET, CATHARE and RELAP5. Test types are as follows: inadvertent opening of pressuriser safety valve; rupture of pressuriser surge line [15]; LOCA from the primary to the secondary circuit with accident management (AM) actions [17]; small break LOCA with AM actions like primary and secondary bleed and feed [16].

National safety research projects. To give further support to the safety reassessment of the Paks NPP different national research projects were fulfilled. In this national framework altogether 26 PMK experiments have been performed.

Others cases. Seven tests were performed to study special processes and to make preparations for experiments mentioned above.

SELECTED RESULTS OF EXPERIMENTS AND CODE VALIDATION

Main groups of operational transients and accidents simulated are as follows:

One- and two-phase natural circulation. The study of the natural circulation processes is of great importance, because in off normal plant conditions the heat removal from the reactor occurs by single or two-phase natural circulation. The natural circulation in VVER reactor systems is also effected by the loop seal in the hot leg and the horizontal steam generator. 3 experiments have been performed to study the conditions of the one-phase natural circulation, to measure the two-phase characteristics at different primary mass inventories, to know the effect of non-condensable gases on the natural circulation and to measure the coolant inventory in the core when heat transfer crisis occurs.

Another measurements were the investigation of the possible disturbances of natural circulation in shutdown conditions of the reactor. In VVER systems the heat in these conditions is transferred to the SG by single-phase natural circulation with two loops. Four different disturbances have been tested as follows: gas in the upper plenum, gas in the collectors of the steam generator, partial closure of the main loop isolation valve, injection of cold water to the upper plenum (6 experiments).

Cold and hot leg break LOCA. In most of the experiments the break location was in the cold leg with a wide range of break sizes as 0.5%, 1.0%, 3.5%, 7.4% and 14.8%. The break size 7.4% was selected to study the effect of different ECCS configurations, the AM actions and presence of non-condensable gases. 9 experiments of this type have been performed. Hot leg break LOCA accidents were studied for 7.4% and 14.8% to compare the results with the relevant cold leg break

LOCA cases. In two types of other experiments, as the opening of the pressuriser safety valve and the break of the pressuriser surge line, the location is also in the hot part of the system [8], [9], [15].

Other tests. Altogether 10 experiments have been performed for different purposes with different initial events [9], [13]. Test types are as follows: loss of flow; total loss of feed water; total loss of off-site power; rupture of main steam collector, simulation of anticipated transient without scram (ATWS).

To show a few results examples are selected and given below

1% cold leg break. [6], [7] Results of two experiments are included: experiments without and with primary side bleed and feed. It was considered in each experiment that 1 HPIS is available and the secondary side is isolated at the beginning of the transient. The transient time for both experiments is 4000 s. The bleed using the pressuriser safety valve is initiated at 600 s. The most important information is shown in Fig. 2 where the coolant-collapsed level in the reactor model is presented for both without bleed "OBF" and with bleed "BF". The core is uncovered in the experiment without bleed (the elevation of the outlet section of the core is 3.5 m), while in the "BF" experiment the core is fully covered. The reason is the earlier opening of the cold leg loop seal that occurs at 1800 s in the "OBF" experiment and at 1500 s in the "BF" experiment. The prediction of the time variation of the coolant collapsed level by the RELAP code for the "BF" experiment is given in Fig. 3. The agreement is excellent.

SG collector cover opening. A few key parameters of the 3 PRISE tests from PHARE VVER01 project are compared in Figs. 4 to 6 [17]. The studied cases were: break of 3, 10 heat transfer tubes and the VVER specific event the lift-up of the SG collector cover. In Figures they are marked as "03", "10" and "CO".

The primary and secondary pressures are presented in Figs. 4 and 5, respectively. There are two AM actions: at 1000 s the pressuriser spray is initiated, at 2000 s the secondary side bleed is initiated resulting in a fast drop of pressure in the secondary side. In case of the rupture of 3 tubes the loss of coolant can be compensated by the ECCSs, while the case "10" and CO" show practically the same behaviour. The total masses leaked through the break are presented in Fig. 6. The break flow and consequently the total mass of coolant leaked are governed by the pressure difference between the primary and the secondary circuit.

Pressuriser surge line break. To study the pressuriser thermal-hydraulics two experiments, the transients following the opening of the safety valve and the break of the surge line have been tested in a PHARE project [14]. The surge line break is a relatively large leak in the hot leg with ECCS configuration as follows 2 SITs, 1 HPIS, 1 LPIS. Figure 8 shows the primary pressure history. Due to the large break size the pressure drops below 1.0 MPa at about 400 s and it practically stagnates until the end of the transient time. Measured and calculated break mass flow rates are presented in Fig. 8. The agreement is good enough. The reactor model is practically emptied at about 200 s, and then refilled again by the LPIS injection, which is initiated at 300 s (Fig. 9). As a consequence of the large coolant loss there is an extended dry out in the core (Fig.10). The prediction by ATHLET is good for the parameters selected for comparison.

Cold leg breaks with primary and secondary bleed [16]. The unavailability of the high pressure injection system (HPIS) is considered. The question is whether the primary system pressure is dropping to the set-point pressure of the low pressure injection system, without HPIS. The break size is 7.4% and 3 SITs are available. The primary bleed is initiated by opening of the pressuriser safety valve. On the secondary side the bleed is actuated at 900 s using the BRU-A valve model. Due to the very deep core uncovery (Fig.12) there is an extended dry out in the core (Fig.13). As shown, the RELAP5/MOD3.2.2Gamma (SIEMENS calculation) predicts the relevant thermal-hydraulic phenomena such as the system behaviour, the break flow (Fig.11), the core uncovery and the heat transfer mechanisms sufficiently well.

CONCLUSIONS

Due to the specific features of the VVER-440/213-type reactors the transient behaviour of such a reactor system is different from the usual PWR system behaviour. To provide an experimental database for the transient behaviour of VVER systems the PMK integral-type facility, the scaled down model of the Paks NPP was designed and constructed in the early 1980s.

Since the start-up of the facility 48 experiments have been performed. It was confirmed through the experiments that the facility is a suitable tool for the computer code validation experiments and to the identification of basic thermal-hydraulic phenomena occurring during plant accidents. High international interest was shown by the four Standard Problem Exercises of the IAEA and by the projects financed by the EU-PHARE.

A wide range of small- and medium-size LOCA sequences have been studied to know the performance and effectiveness of ECC systems and to evaluate the thermal-hydraulic safety of the core. Extensive studies have been performed to investigate the one- and two-phase natural circulation, the effect of disturbances coming from the secondary circuit and to validate the effectiveness of accident management measures like bleed and feed. The VVER-specific case, the opening of the SG collector cover was also extensively investigated.

Examples given in the report show a few results of experiments and the results of calculation analyses performed for validation purposes of codes like RELAP5, ATHLET and CATHARE.

There are some other white spots in Cross Reference Matrices for VVER reactors [19] and, therefore, further experiments are planned to perform tests primarily in further support of accident management measures at low power states of plants to facilitate the improved safety management of VVER-440-type reactors.

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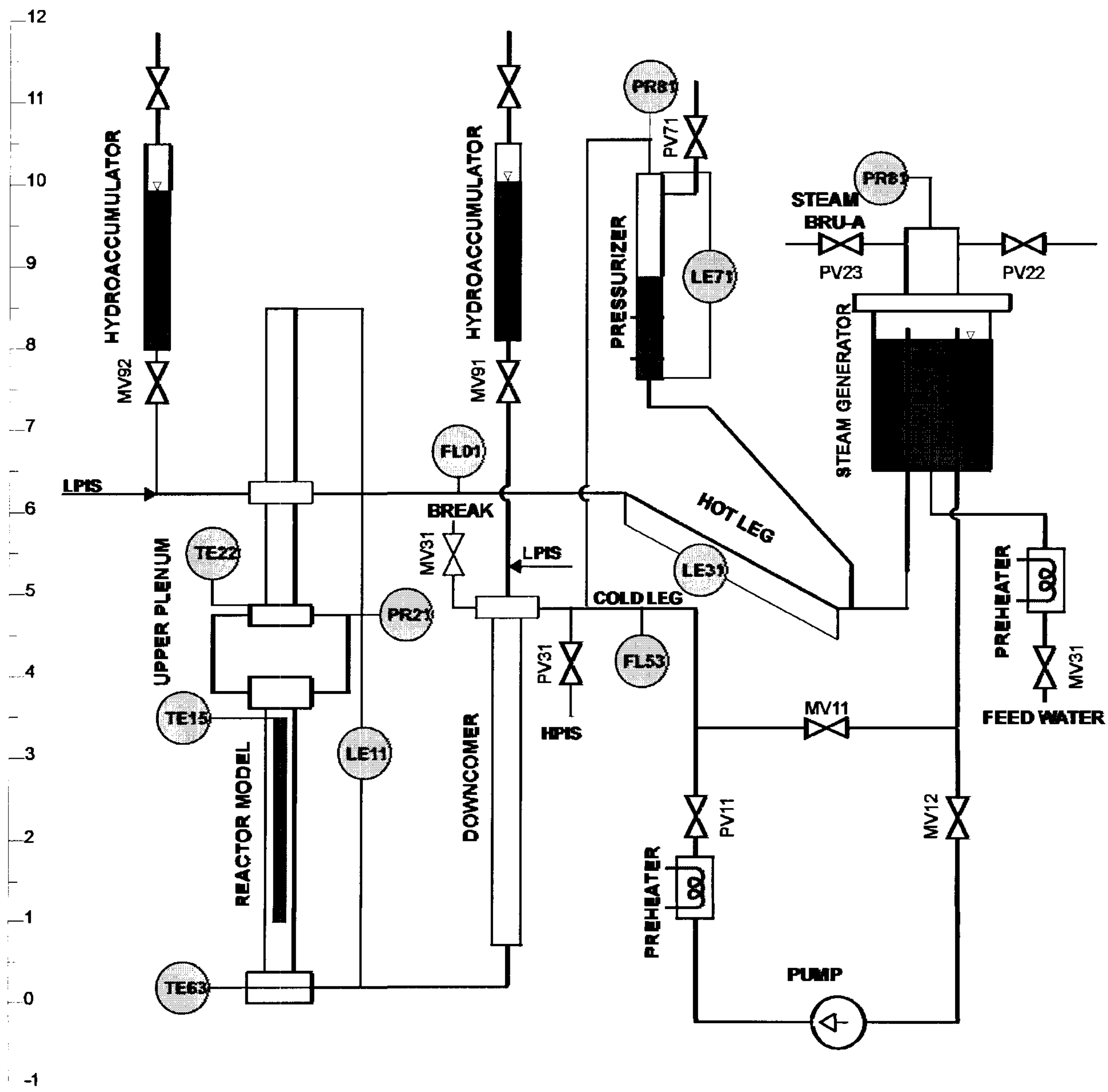

Fig.1 PMK-2 facility

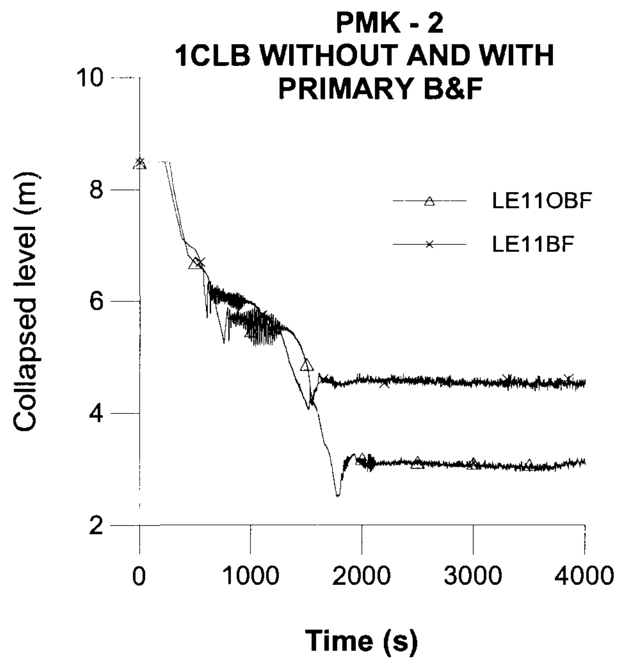

Fig. 2 Reactor model level

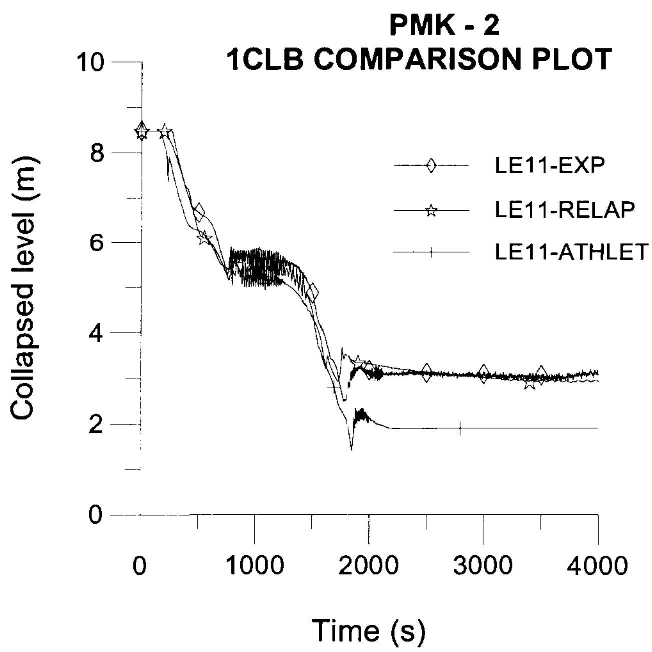

Fig. 3 Reactor model level (BF)

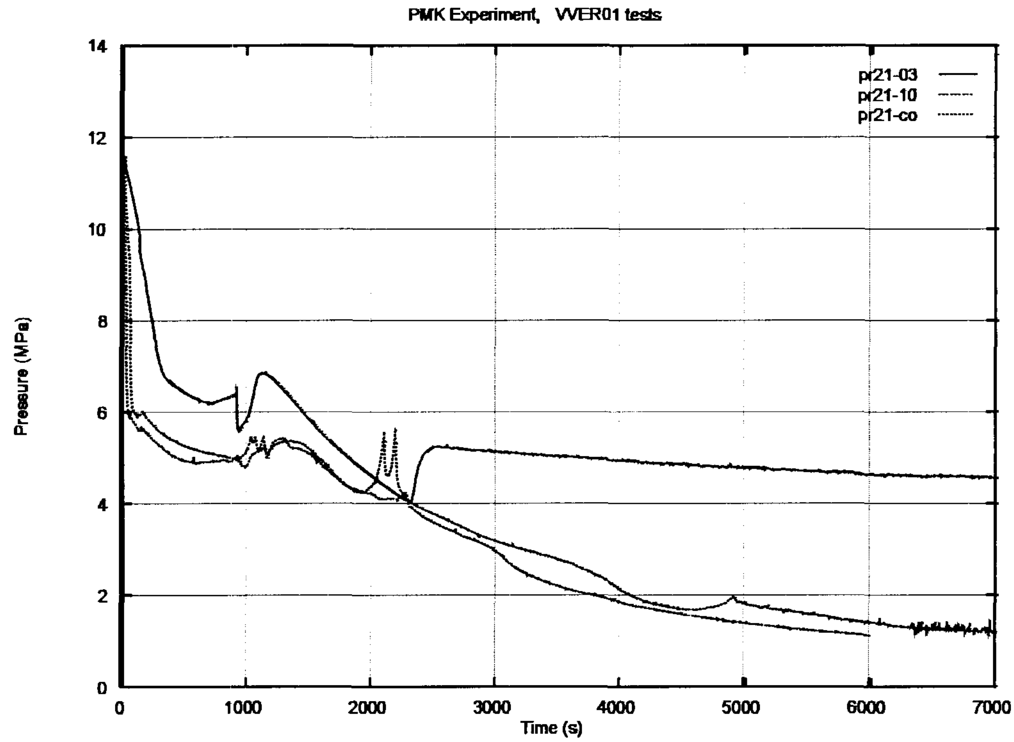

Fig. 4 Upper plenum pressure

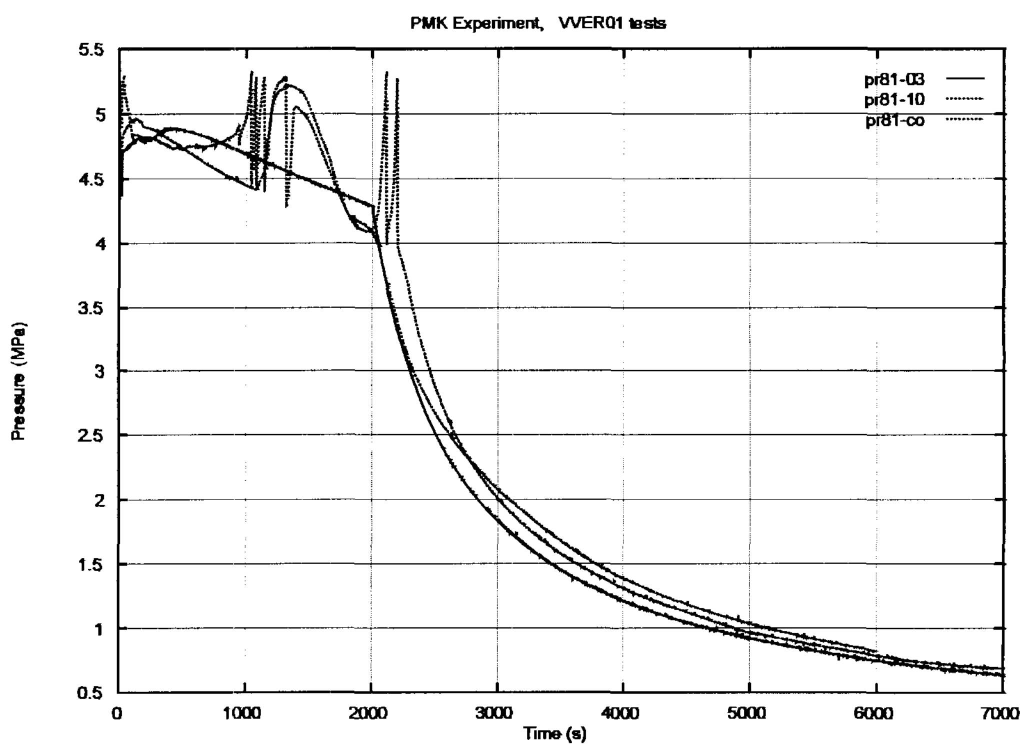

Fig. 5 Secondary pressure

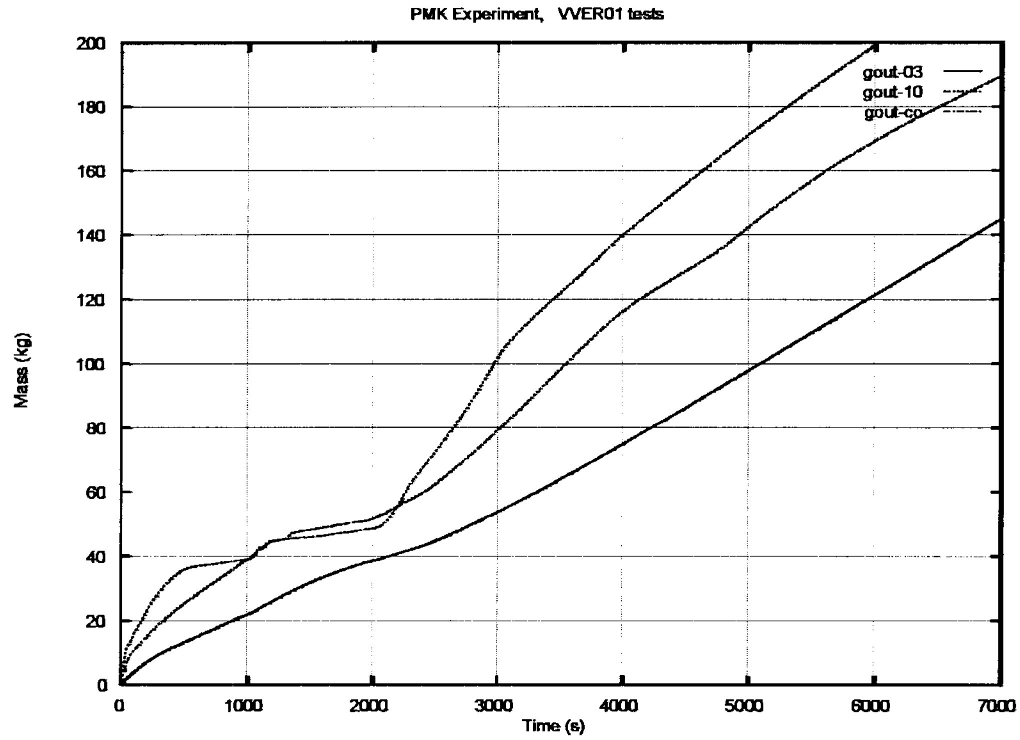

Fig. 6 Integrated break mass

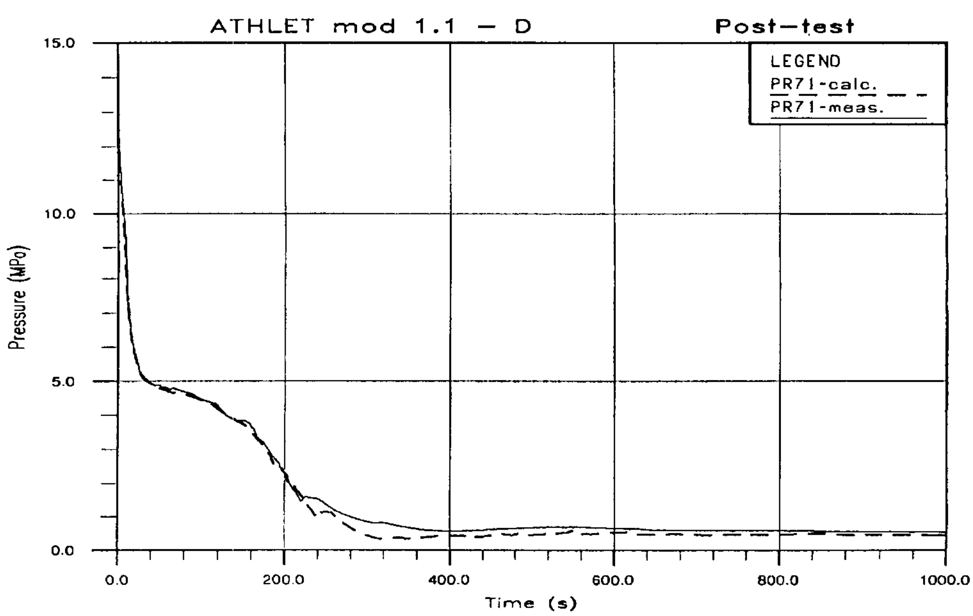

Fig. 7 Pressuriser pressure

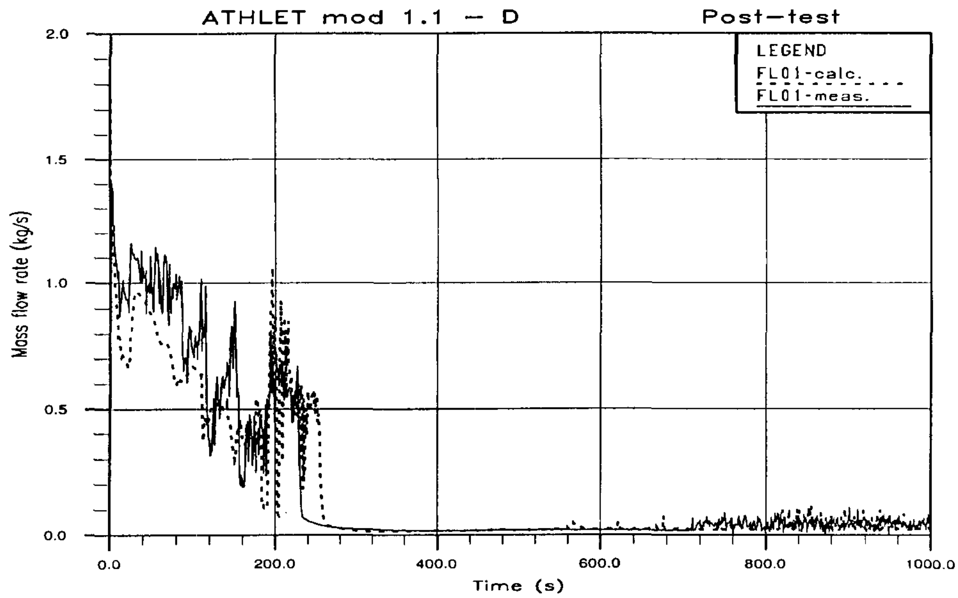

Fig. 8 Break mass flow rate

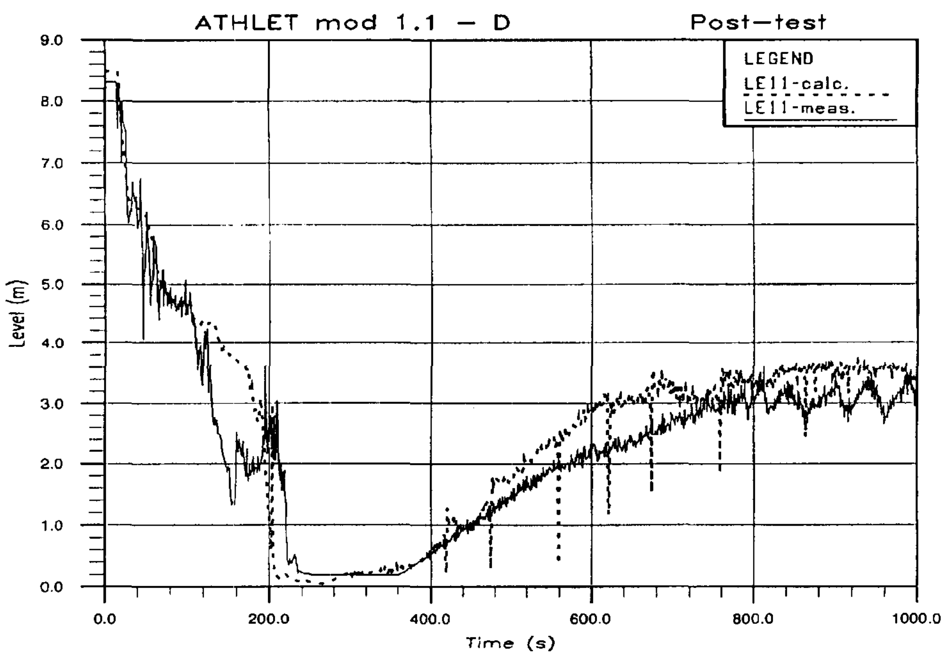

Fig. 9 Coolant level in reactor model

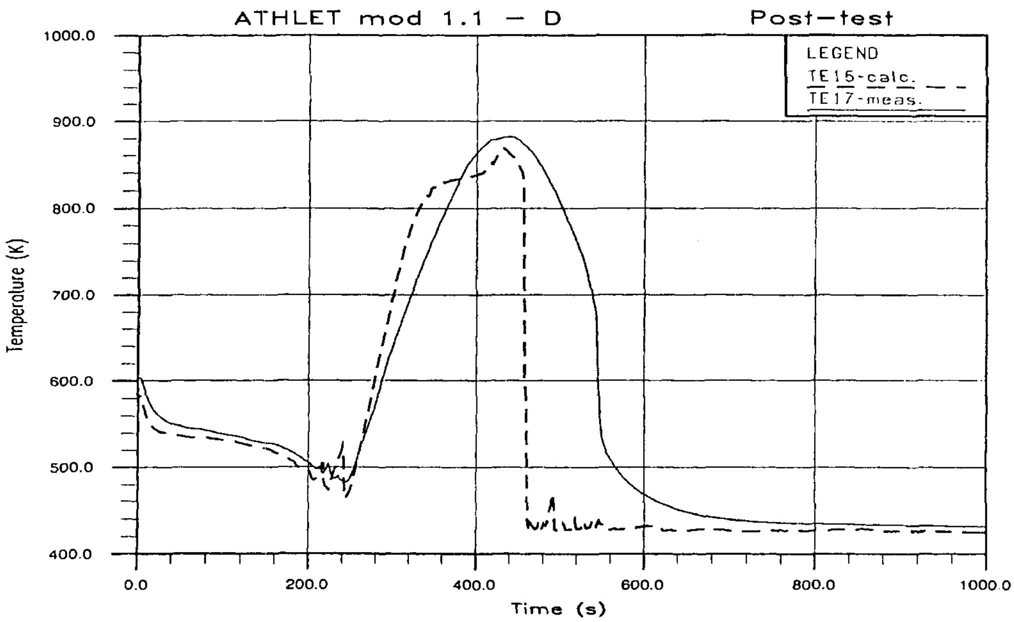

Fig. 10 Fuel rod surface temperature

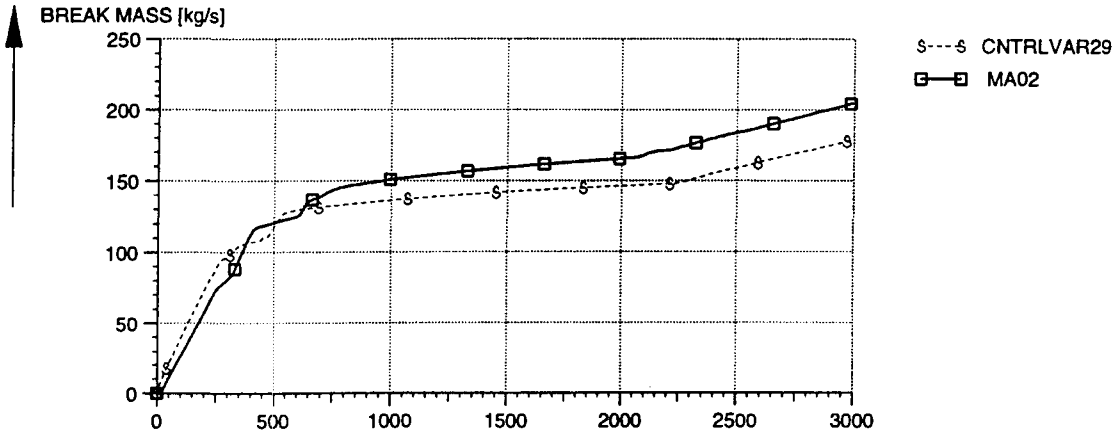

Fig. 11 Break mass (MA02)

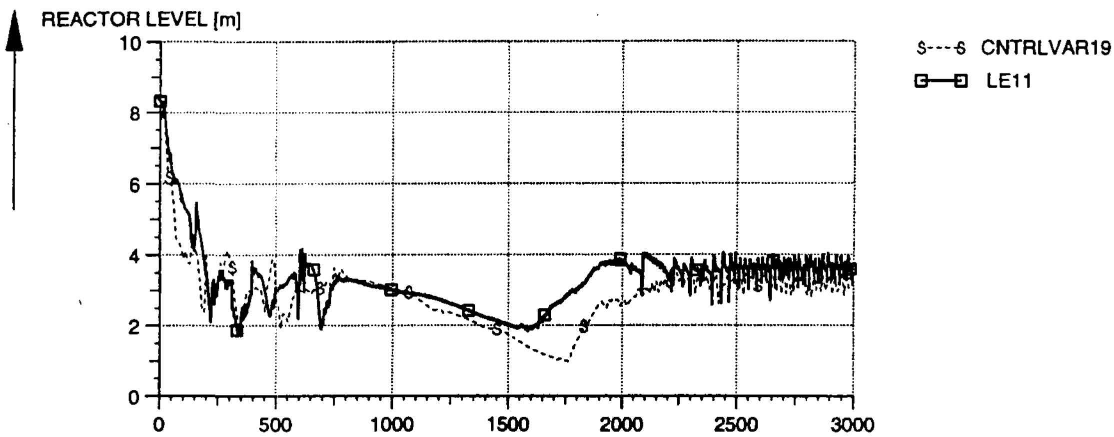

Fig. 12 Coolant level in the reactor model (LE11)

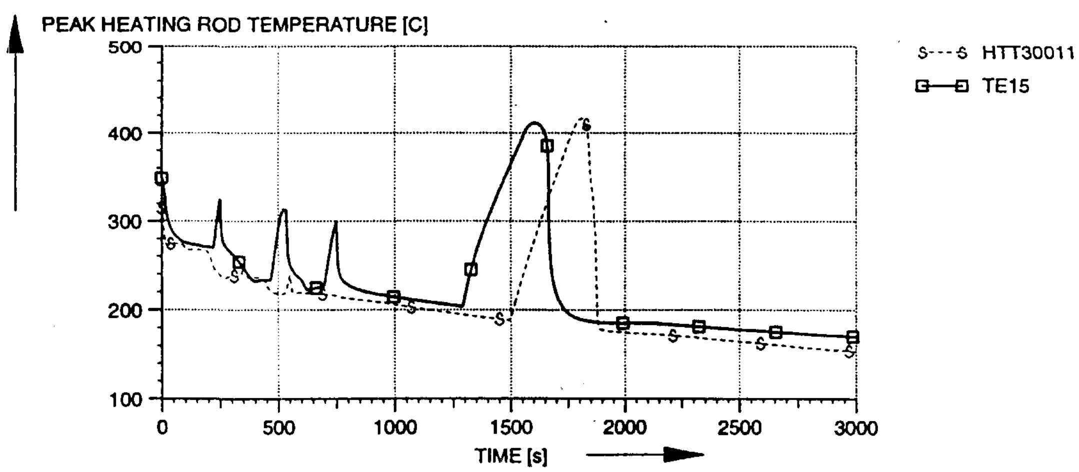

Fig. 13 Rod temperature (TE15)