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REPORT

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PHYSICS

BUDAPEST

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Post Test Analysis of LOBI BT17 Experiment

ABSTRACT

~~The paper describes~~ the LOBI experimental facility and the BT17 experiment. The computational analysis has been performed by the CATHARE thermal hydraulic system code. The results of calculations are in satisfactory agreement with the experimental values. A comparison has been made with a Loss-of-Feedwater test performed on the PMK-2 facility.

I.T. Farkas, Z. Hózer, A. Takács :
A LOBI BT17 kísérlet kiértékelése

KIVONAT

A cikk röviden ismerteti a LOBI kísérleti berendezést és a BT17 kísérletet. A folyamat számítógépes elemzése a CATHARE termohidraulikai rendszerkóddal készült. A kísérleti és számított adatok kielégítő egyezést mutatnak. A LOBI kísérletet összehasonlítottuk a PMK-2 berendezésen végzett tápvízvesztéses kísérlettel is.

Contents

1. Introduction

The LOBI Project has been carried out in the framework of the Comission of the European Communities Reactor Safety Research Programme. The experimental program has been performed at the LOBI full-power high pressure integral system test facility. This research programme generated an experimental data base for safety studies including the assessment of predictive capabilities of thermal-hydraulic system codes.

The KFKI Atomic Energy Research Institute has had access to the LOBI Data Base since the end of 1992 on the basis of a contract between CEC Joint Research Centre Ispra and KFKI AEKI. The experimental data are used for nuclear safety purposes in the following terms : 1. analysis of general phenomena of PWR transients, 2. use of LOBI know-how for experiment preparation, 3. validation of CATHARE thermal-hydraulic system code.

In the present paper LOBI test BT17 has been analyzed. This experiment represents a loss-of-feedwater transient with feed and bleed procedure. The computational analysis was performed with the CATHARE code. The experimental behavior was compared with a loss-of-feedwater test performed at the PMK-2 facility located in the KFKI AEKI.

2. The LOBI Facility

The LOBI-MOD2 test facility located at the Ispra Site of Joint Research Centre was designed to perform safety related experiments and study such DBA, ATWS, station blackout etc. accident situations that can happen in western design PWRs.

This full-power high-pressure integral test facility is a 1:712 scale model of the reference reactor (1300 MW PWR of KWU design). Beside considering the similarity of thermohydraulic behavior a power-to-volume scaling principle was used in the design of the facility to keep the preservation of the specific power input into the primary fluid. Also the elevations of the major components are identical in the reactor and the test facility to model properly the gravitational effects which play important role in natural circulation.

The core and steam generators flow- and heat transfer areas were chosen according to the scaling factor. The essential features of typical PWR primary and secondary cooling system are incorporated in the facility.

Concerning the primary side model the four loops of the PWR are represented by two loops of the test facility in such a way that the intact loop corresponds to three loops of the reactor. The total volume is 0.6 m^3. Each primary loop contains a main circulation pump and a steam generator. The centrifugal type pumps are operating at different speeds according to scaling.

The lower and upper plenum, the annular downcomer and the externally mounted upper head simulator are the main components of the reactor model. The primary cooling system operates at nominal PWR conditions: at 15.8 MPa primary pressure and 294/326 °C temperature.

The simulated core consists of a directly electrically heated 64 rod bundle arranged in an 8x8 square matrix inside the pressure vessel model. The nominal heating power is 5.3 MW and the heated length is 3.9 m. The outer diameter of the rods is 10.75 mm and the pitch is 14.3 mm. The wall thickness varies in 5 steps to give a cosine shaped axial heat flux distribution.

Regarding the secondary side heat is removed from the primary loops by the secondary cooling circuit containing a condenser and a cooler, the main feedwater pump and the auxiliary feedwater system. At normal conditions the feedwater temperature is about 210 °C and the pressure in the secondary side is 6.45 MPa.

Each steam generator consists of a single cylindrical pressure vessel with an annular downcomer separated from the riser region by a skirt tube. The broken loop SG and the intact loop SG contain 8 and 24 U-tubes respectively. The U-tubes are arranged in a circle within the riser region around an axially mounted filler tube. Feedwater is injected into the SG downcomer by a feed ring and flows downward to mix with the recirculating water coming from the fine and coarse separators. The hydraulic behaviour of the main coolant pumps and the core decay heat release can be regulated and studied through a process control system.

Approximately 470 test channels provide signals for the data acquisition system regarding the main thermohydraulic parameters at the inlet and outlet sections of each individual loop component, within the reactor pressure vessel model and the steam generators.

[1],[2],[3]

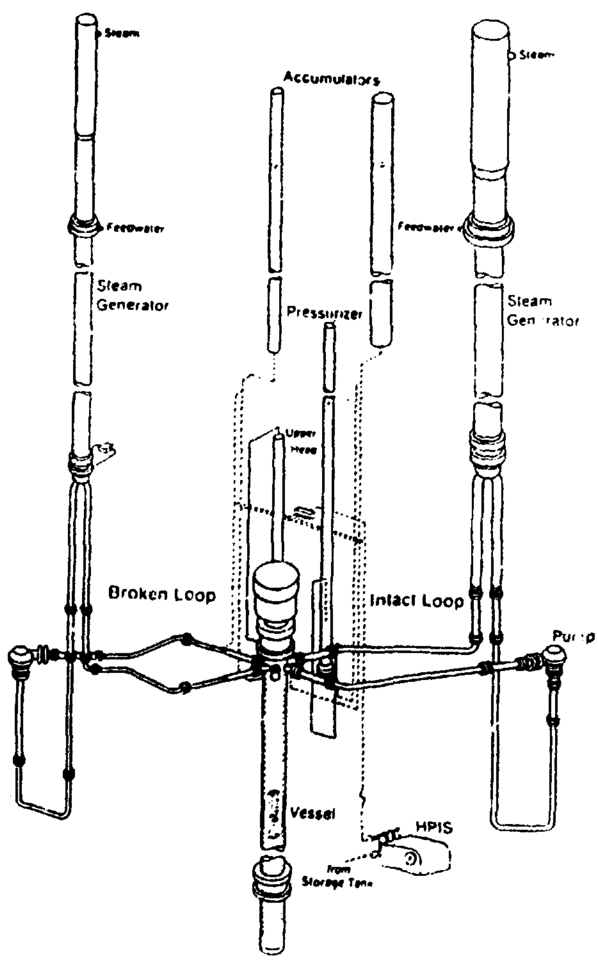


Fig. 1.

The LOBI MOD2 Test Facility

Table 1.

Main Characteristics of
LOBI MOD2 and PMK-2 Facilities

LOBI MOD2		PMK-2
KWU PWR 4 loops, 1300 MWe	reference reactor	VVER 440/213 6 loops, 440 MWe
yes	integral type	yes
2	number of loops	1
0.6 m^3	total volume	0.12 m^3
1:700	volumetric scale	1:2070
1:1	elevation scale	1:1
5.28 MW	core power	0.783 MW
3.9 m	heated length	2.5 m
64	number of rods	19
8x8(square)	rod bundle	hexagonal
10.75 mm	heater rod diameter	9.1 mm
14.3 mm	pitch	12.2 mm
elect. direct	heating	elect. direct
annular	downcomer	externally mounted cyl.
internal	upper plenum	external
external	upper head	external
centrifugal	MCPs	centrifugal
2	number of SGs	1
U-tube vertical	type of SG	horizontal
broken loop: 8 intact loop: 24	number of SG tubes	82
2	number of HAs	2
15.8 MPa	primary pressure	12.26 MPa
294/326 °C	core inlet/outlet temp.	267/297 °C
6.45 MPa	secondary pressure	4.53 MPa
CEC JRC Ispra	organization	KFKI AEKI Budapest

3. The LOBI BT17 Test

The BT-17 test simulated a loss of feedwater with an accident management procedure featuring the secondary side feed and bleed.

The experiment consisted in loss of heat sink with consequent pressurization and loss of primary mass through the pressurizer PORVs leading to core uncovery. The dryout had to be controlled by a deliberate secondary side depressurization with AFW injection at a pressure of 2.0 MPa in order to simulate the passive injection by feedwater tank flashing. The effect of the secondary feed and bleed - actuated in the transient when the core starts to uncover - was to determine a depressurization in the primary system by condensation in the SG U-tubes. As a consequence - draining of the water from the pressurizer should rewet the core.

3.1 Initial Conditions

The experiment started from nominal 100% conditions.

The main parameters of initial state are given in Table 2.

3.2 Boundary Conditions

After 37 s the scram is simulated with core power reduction and MSIV closure simulation.The decay heat continues to boil off the secondary side and 180 s after the level in one of the two SGs reached 2.24 m the main coolant pumps are stopped with a coast down of about 70 s. The loss of heat sink determines the primary system temperature and pressure increase. In order to limit the pressure increase the PORV is open and primary inventory is discharged. When the core level starts to decrease a dryout occurs and when the surface temperature reaches 400°C the secondary side blowdown is actuated in order to reduce the pressure and reach the actuation setpoint of 2.0 MPa for the auxiliary feedwater. No other action was pre-planned if the core temperature exceeded 600 °C. During the test this temperature was reached and therefore an ad-hoc emergency procedure was enabled to try to control the temperature rise. Within the specification of this procedure HPIS was activated in the final part of the test. In the test leak occured in the primary system. It is wotrh noting that during the first 500 s the valve failed to open, determining a positive inflow with the pump seal flowrate.

A short summary of boundary conditions is given in Table 3.

Table 2.
Initial Conditions for LOBI BT17
and PMK-2 LOFW Experiments

	LOBI BT-17	PMK-2 LOFW	Unit
Primary Pressure	15.9	12.3	MPa
Core Power	5190.	660.0	kW
IL mass flow	21.0	—	kg/s
IL vessel inlet temp.	296.	—	°C
IL vessel outlet temp.	326.	—	°C
BL mass flow	7.25	5.15	kg/s
BL vessel inlet temp.	296.	267.65	°C
BL vessel outlet temp.	328.	295.35	°C
PRZ water level	4.9	9.52	m
IL SG SL pressure	6.48	—	MPa
IL SG mass flow	1.98	—	kg/s
IL SG inlet temp.	208.	—	°C
IL SG outlet temp.	281.	—	°C
IL SG downcomer level	8.03	—	m
BL SG steam dome pressure	6.46	4.6	MPa
BL SG mass flow	0.71	0.36	kg/s
BL SG inlet temp.	205.	200.95	°C
BL SG outlet temp.	280.	258.	°C
BL SG level	8.40	8.19	m

Table 3.

Boundary Conditions LOBI BT17 and PMK-2 LOFW Experiments

	LOBI BT-17	PMK-2 LOFW
MCP	Start to coast down at 180 s after sec. side level was 2.24 m.	Start to coast down at p=8.6 MPa
MFW	off	off
AFW	Secondary pressure 2.1 MPa to start AFW injection	—
Power	Time to start heating power reduction at 37 s. Rod temperature to switch off 661°C	Scram at p=13.0 MPa
HPIS	Rod temperature to activate HPIS 611°C	NU
LPIS	NA	on at 0.7MPa
HA	NA	NU
SG SRV open	Rod temperature 402 °C	702 s
PRZ PORV	Open at 16.5 MPa primary pressure	Open at 14.2 MPa close at 12.6 MPa
Upper plenum valve	Rod temperature to open 596°C 1st time 544°C 2st time For closing:rewetting of the upper part of the core	—

3.3 Experimental Profile

Loss of feed water lead to rapid boil off of the steam generators. At 37 s the reactor trip was simulated and the secondary inventory decrease rate was reduced.The main coolant pumps were stopped at about 542 s. As the steam generators were emptying the primary system temperature continued to rise in order to compensate the reduced heat transfer area. As a consequence of the temperature increase the primary system pressure increased and at 1453 s the PORV started to cycle to keep the pressurizer pressure around 16.2 MPa. The continuous temperature increase resulted in reaching of saturated state 2942 s. Thereafter the pressurizer was completely filled and water started to be released by the PORV. As a consequence the primary inventory decreased and upper part of core was in dryout at 4935 s. At 5037 s the secondary side relief valve was latched open to induce a secondary blowdown and at 5133 s with a secondary side pressure of 2.1 MPa the AFW mass flow rate was activated. The objective of these actions was to cause a primary system depressurization through condensation in the U-tubes and outflow of water from the pressurizer into the primary system to rewet the core. However this did not occur because the water injected on the secondary side was evaporated on the downcomer walls before reaching the U-tubes thus preventing secondary to primary heat exchange. As a consequence the core surface temperature continued to rise reaching 600°C enabling an alternative Test Emergency Procedure (TEP) that was pı -defined for this test. The TEP aimed at an increase of primary system depressurization through the deliberate opening of a valve in the upper plenum. This action caused the water still in the pressurizer to be injected into the vessel. Thereafter the core was rewetted from the uppermost elevation. On the indication of core rewet the valve was closed. As the valve was closed the mass flow from the pressurizer was strongly reduced and therefore a second dryout occured. When the maximum core surface temperature reached about 600°C the valve was open again. However as soon as the pressurizer was completely empty the core temperature restarted to rise and the core power was switched off. The sequence of main events is given in Table 4.

[4],[5],[6]

Table 4.
Main Events
of LOBI MOD2 BT17 Experiment

Event	Experiment time(s)	Calculation time(s) Case 1	Calculation time(s) Case 2
Loss of feedwater	0	0	0
Steam line closure Heating power to decrease	37	37	37
Pump coast down	542	542	542
PORV fist open	1453	2448	2448
PRZ full	3674	3980	3980
Core Dryout starts	4935	5115	4326
SG relief valve latched open	5307	5145	4351
AFW starts to inject	5133	5165	4367
UP valve open	5536	5293	4856
UP valve closed	5593	5359	5310
UP valve open 2nd	5689	5964	—
HPIS on	5836	—	5222
Power off	5861	—	5250

4. The CATHARE Code

The CATHARE code has been developed at C.E.N. Grenoble in the frame of a joint effort of CEA, EDF and FRAMATOME for system thermal-hydraulic studies in the field of nuclear safety.

The physical model of the CATHARE code is based on the *two-fluid* description of two-phase gas-liquid systems. This model is capable of handling non-equilibrium phenomena. Mechanical non-equilibrium is considered in the terms of :

- ⊕ Phase separation,
- ⊕ Stratification,
- ⊕ Co-current and counter-current flows,
- ⊕ Counter-current flow limitation.

Thermal non-equilibrium is taken into account in the following cases:

- ⊕ Critical flow,
- ⊕ Cold water injection,
- ⊕ Reflooding.

The mathematical description uses six basic equations of two-fluid model (two equations for mass balance, two for energy balance and two for momentum balance). One or two additional equations describe the transfer of noncondensible gases (hydrogen and nitrogen). The noncondensible gases are handled as perfect gases described by Dalton's law and being in mechanical and thermal equilibrium with steam. Hydrogen is produced during cladding oxidation while nitrogen can be injected from hydroaccumulators. Boron acid and fission product isotope transfer are described as well.

The wall heat transfer phenomena are represented by heat conduction model including wall-to-liquid and wall-to-gas heat transfer and describing boiling crisis and dryout phenomena. Radial heat conduction is used for multi-layer walls and for fuel rods. Radial and axial heat conduction is calculated for reflood transients making use of a 2D moving mesh in the vicinity of the quench front.

The system of basic equations is closed by a unique set of closure relations for the following terms:

- ⊕ Interfacial heat transfer:
  - ⊕ Condensation,
  - ⊕ Evaporation,
  - ⊕ Flashing.
- ⊕ Wall to fluid heat transfer:
  - ⊕ Heat exchange zone map,
  - ⊕ Wet wall heat flux,
  - ⊕ Dry wall heat flux,
  - ⊕ Critical heat flux,
  - ⊕ Minimum stable film temperature,
  - ⊕ Dryout criterion,
  - ⊕ Transition heat flux,
  - ⊕ Radiation heat transfer.
- ⊕ Stratification criterion,
- ⊕ Momentum transfer:
  - ⊕ Wall friction,
  - ⊕ Interfacial friction,
  - ⊕ Added mass.

- ⊕ Droplet diameter,
- ⊕ Interface velocity,
- ⊕ Special reflooding correlations.

The code uses five main elements for the geometrical presentation of the calculated facility:

- ⊕ 1D pipe (basic modul),
- ⊕ Volume (with two subvolumes and level calculation),
- ⊕ Tee (with three connecting junctions),
- ⊕ 1D pump,
- ⊕ 2D downcomer.

Several submodels have been developed for the special elements of nuclear power plants : point neutronic-kinetics, hydroaccumulator model, point pump model, point steam generator model, etc.

The code is able to simulate different kinds of boundary conditions as e.g.: Scram, pump trip, break opening, steam generator tube rupture, ECC injection, valve opening and closing, containment back pressure, etc.

The numerical solution is based on the finite difference approach. The finite difference equations are discretized implicitly on a staggered mesh considering donor cell averaging. The strongly non linear system of equations is solved by Newton-Raphson iterative method.

The code assessment process consists of two steps:

I. Verification against separate effect tests and,

II. Verification on integral loop tests.

More then 300 ***separate effect tests*** are calculated to validate the physical closure relations for each code version. The verification matrix on integral loops consists of 21 tests from LOBI, LOFT, LSTF, PKL facilities and the whole BETHSY experimental programme.

The latest version of CATHARE V1.3E has been implemented in KFKI AEKI and the present calculations have been performed making use of this version.

[7],[8],[9],[10],[11],[12],[13]

5. Post-Test Calculation of LOBI BT17 Experiment

The post-test analysis has been performed with the CATHARE code. The nodalization scheme has been developed originally by the CEA for JRC Ispra. In our calculation some modifications were made to that nodalization scheme. Special axial elements have been introduced for the modelling of normal and auxiliary feedwater injection (see Fig. 2.). The original input deck was developed for hot zero power initial conditions. Some changes were necessary in the singular pressure drop distribution on the secondary side and in the heat loss parameters on the primary side.

The calculation was based on CATHARE V1.3E version. The set-up of boundary was performed making use of different kinds of boundary conditions like sources, sinks, etc.. The signal of actions were considered as they were defined in the experiment (see Table 3.). The TEP valve closing condition was defined as rewetting of the upper part of the core in the experiment. In the calculation the TEP valve was closed by rod temperature 400°C which was in agreement with experimental data.

The results of the first CATHARE calculation showed satisfactory agreement with the experimental data concerning the main parameters as primary and secondary pressures, coolant temparatures, mass inventories (see Fig. 3-9.). In the beginning of the transient the coast down of MCPs was followed by a temporary pressure decrease in the primary circuit which was not observed in the experiment. In the calculation the heat removal from the primary circuit remained too intense at low coolant velocities after MCP stop. This lead to a delay in pressurizer level and pressure increase. The calculation described the main actions of the experiment (see Table 4.). The PORV leakage flow lead to rod temperature increase, the SG relief valves were opened by rod temperature signal. Then the further temperature increase lead to TEP valve opening, what resulted in rod temperature decrease for the first time, as the water accumulated in the pressurizer could enter the core (see Fig. 4.). At 400°C the TEP valve was closed and the rod temperature started to increase again and the TEP valve opened for the second time, but that was not enough to stop the core temperature increase. During the calculated period the HPIS system was not activated. In this calculation we found that the maximum core temperature was calculated not at the top of the core but in the middle of the core (see Fig. 10.). Furthermore the calculated rod temperatures in the middle of the core were much higher than the measured ones. For this reason another calculation was performed.

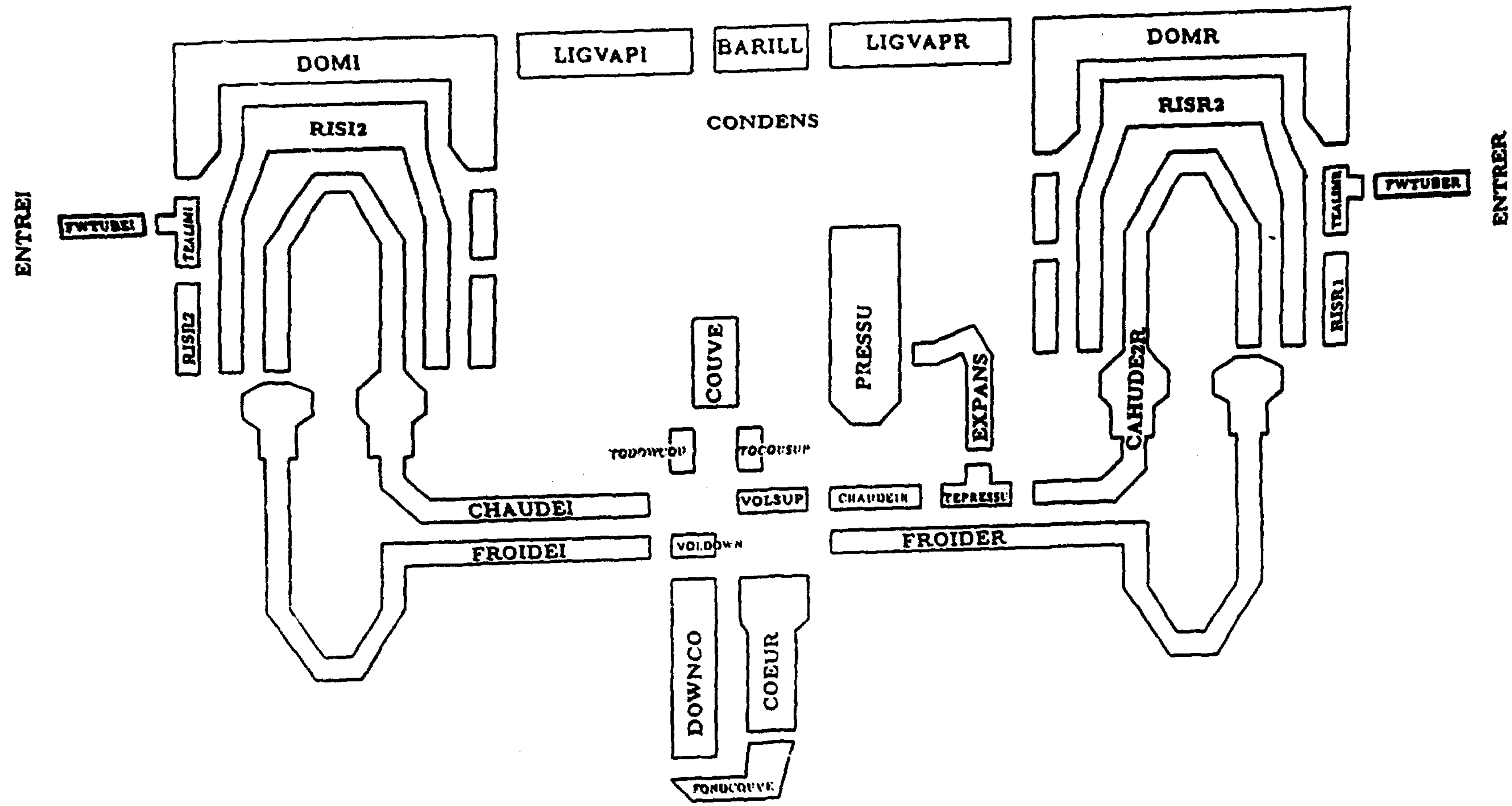


Fig. 2.
CATHARE Nodalization Scheme for LOBI BT17

Fig. 3.

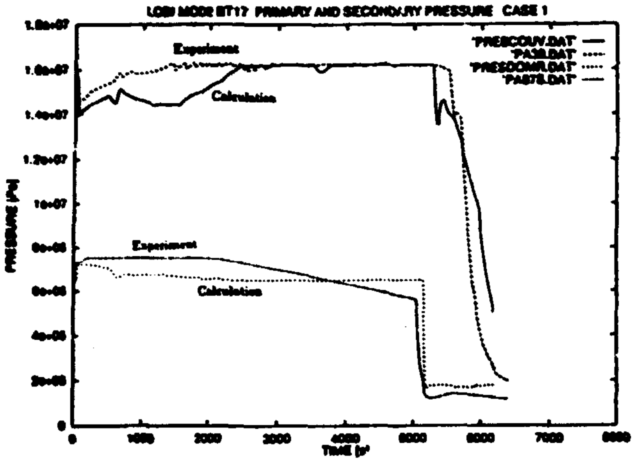


Fig. 4.

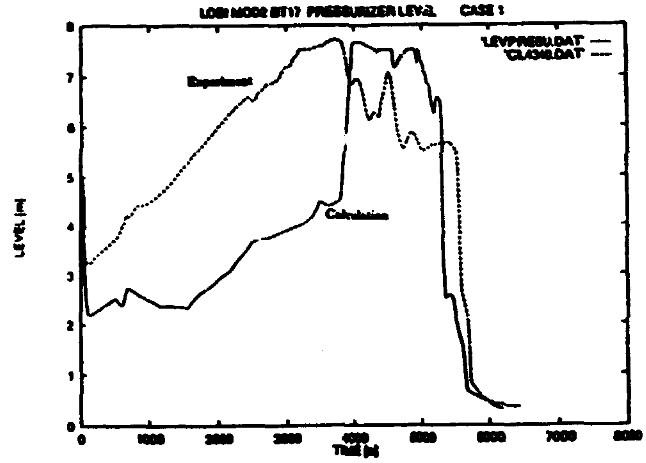


Fig. 5.

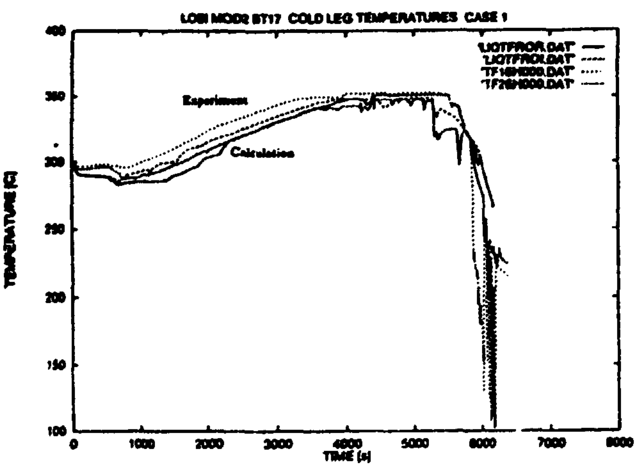


Fig. 6.

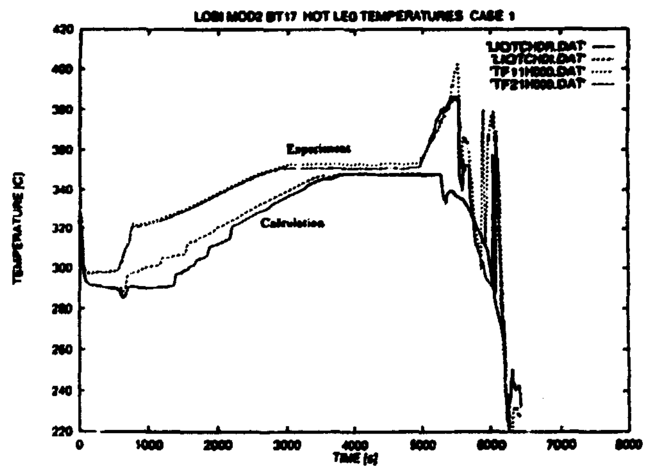


Fig. 7.

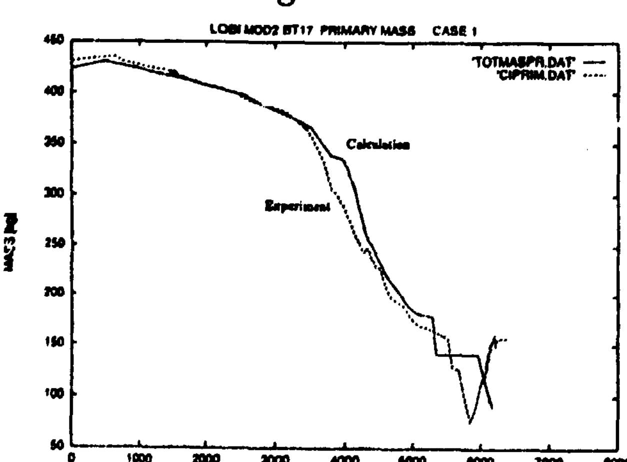


Fig. 8.

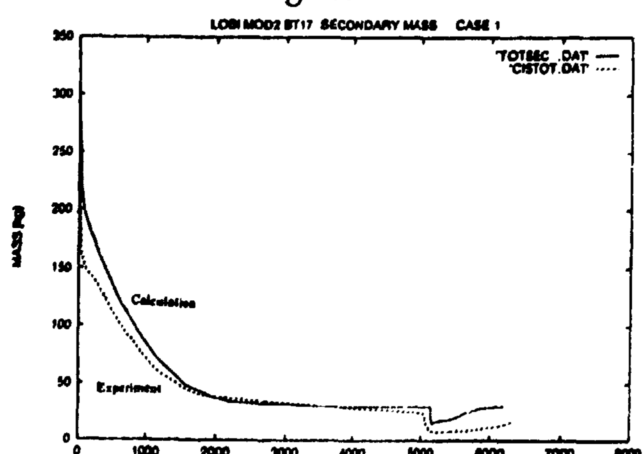


Fig. 9.

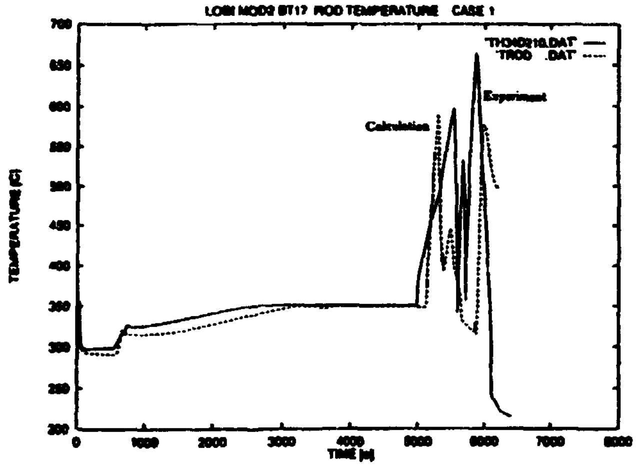


Fig. 10.

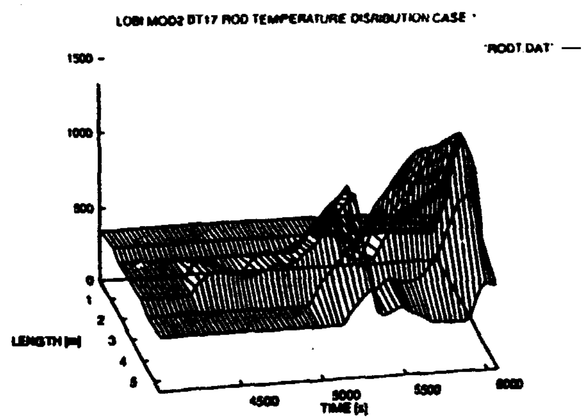


Fig. 11.

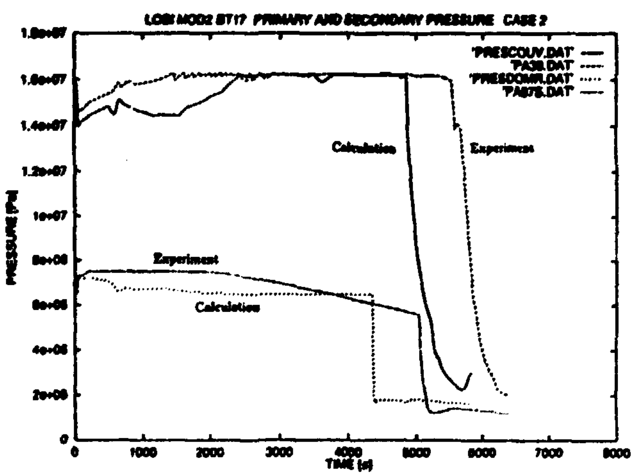


Fig. 12.

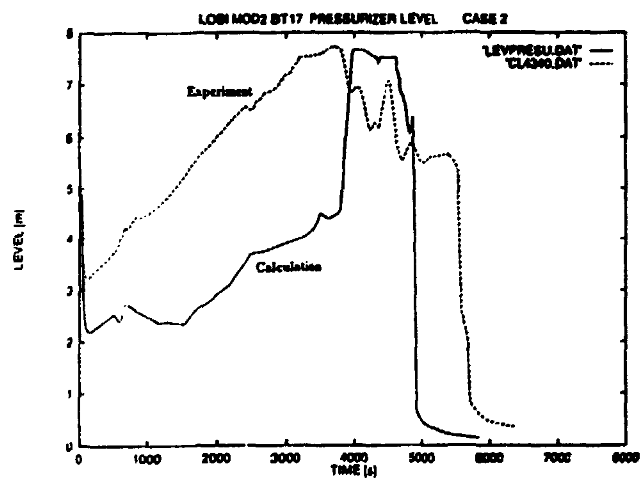


Fig. 13.

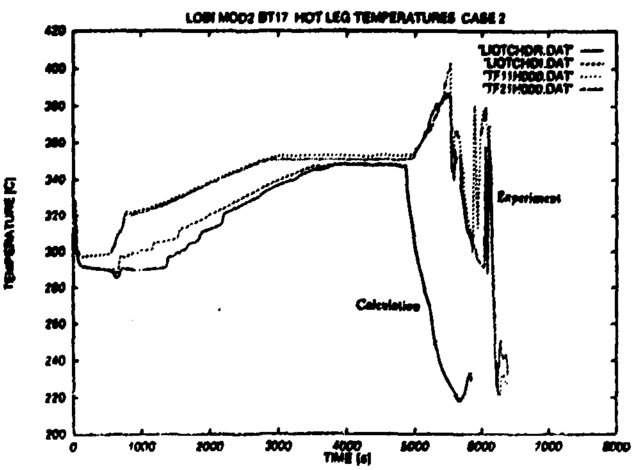
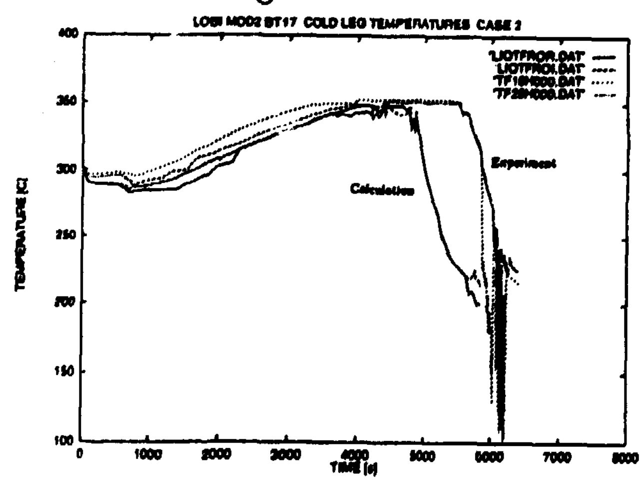


Fig. 14.



In the second calculation the regulation of TEP opening/closing, SG RV opening, HPIS injection, etc. were activated by the maximum rod temperature not by the temperature of the top of the core as that was defined in the first case. The beginning of the transient did not differ in the second case from the first calculation but the opening of SG relief valve, TEP valve and activation of HPIS injection happened earlier (see Table 4. and Fig. 11-17.). This calculation also showed that the late activation of secondary side procedure and the primary side TEP valve were not able to prevent the heated rods from dryout. The calculation pointed out that in this case even the HPIS injection was not able to cool down the rods and the core power had to be switched off at 661°C rod temperature.

In a third calculation the pressurizer volume (Fig. 2.) PRESSU and the axial surgeline element EXPANS were replaced with a common axial element. Since in the test the PORV opening/closing played an important role it was supposed that the pressurizer modelling with a more detailed axial element could increase the precisity of the calculation. However no significant difference was observed between the first and the third calculations.

Fig. 15.

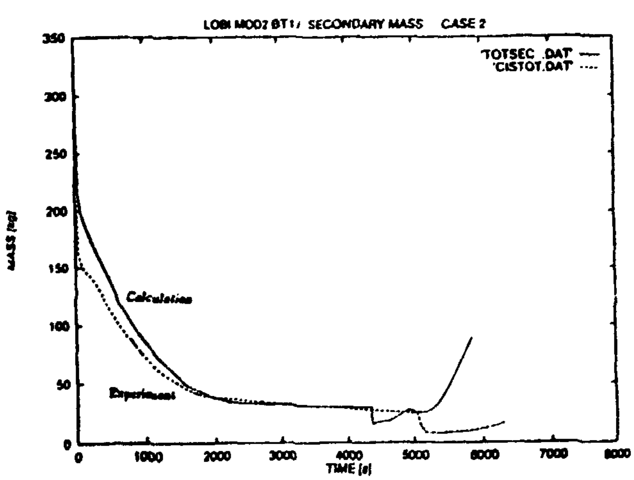


Fig. 16.

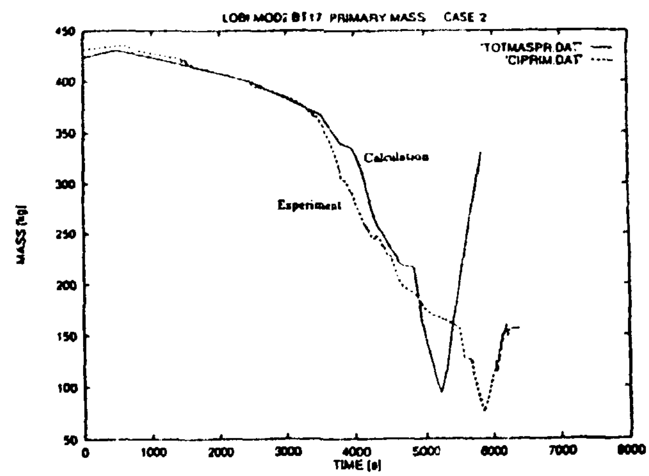
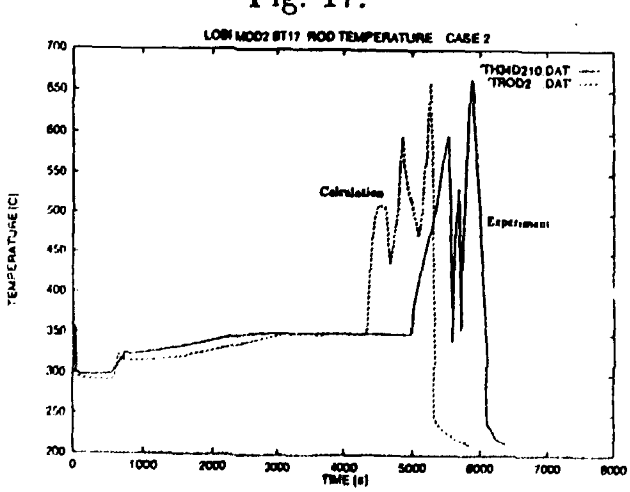


Fig. 17.



6. Comparison of LOBI BT17 and PMK-2 LOFW Experiments

6.1. PMK-2 Facility

The PMK-NVH (since 1991 PMK-2) is the basic tool of experimental activities in the field of PWR system thermal-hydraulics in Hungary. The PMK-2 facility is a full-power high pressure integral test facility representing a 1:2070 scale model of the Paks VVER-440/213 nuclear power plant's primary circuit. The scheme of the facility is shown in Fig. 18.

The elevation ratio is kept 1:1 in the facility except the lower plenum and pressurizer. On the secondary side of steam generators the steam/water ratio is kept. The most typical VVER specific features as horizontal steam generators, loop seals in both cold and hot legs, hexagonal core assemblies, spacers are considered in the design of the facility. The PMK-2 facility is equipped with hydroaccumulators, high and low pressure injection systems as well. The main characteristics of the PMK-2 facility are summarized in the right column of Table 1.

Originally the facility was designed for the investigation of small break-loss-of-coolant accidents but later the scope of experiments covered the following cases:

- Small and medium break LOCAs,
- Plant transients,
- Accident management transients. [14],[15]

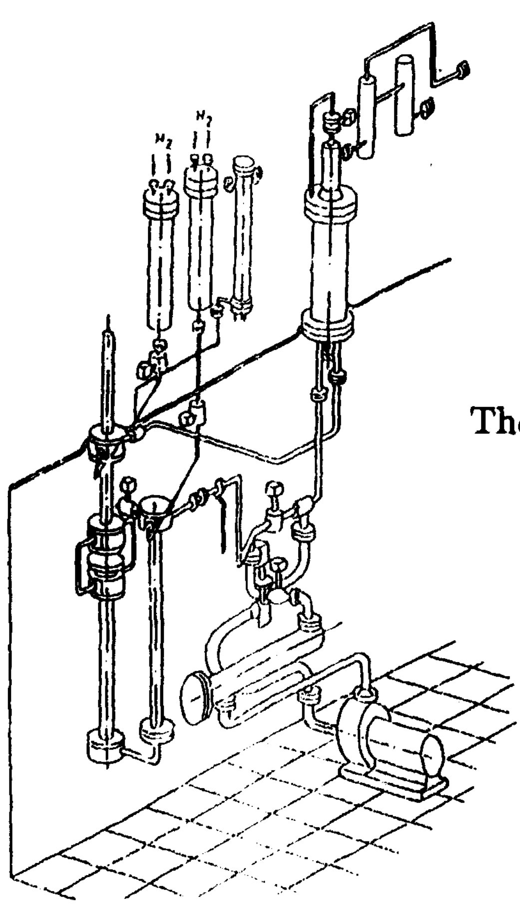


Fig. 18.
The PMK-2 Test Facility

6.2 PMK-2 Loss of Feedwater Experiment

On the PMK-2 facility a total loss of feedwater (LOFW) test had been performed with secondary side feed and bleed procedure. This experiment had a similar sequence of events as LOBI BT17. For this reason it seemed to be interesting to compare this PMK-2 test with BT17, which was a loss of feedwater test with secondary feed and bleed, too.

The LOFW test was started from nominal conditions (100% power). The parameters of initial conditions are listed in Table 2. together with LOBI BT17 initial conditions. The transient was initiated at 0 time with closing feedwater injection. At the same time the secondary side isolation started (closing MSIV) due to a signal for stop of feedwater pump. The secondary pressure increase lead to the SG relief valve opening in 95 s. The coast down of MCP started at 532 s. The secondary bleed was activated by manual opening of SG relief valve at 702 s experiment time. The secondary side bleed started later : at 2247 s. The LPIS system came into action in the later phase of the experiment at 10200 s.

In this test nor HPIS neither hydroaccumulators were used. A summary of boundary conditions is given in Table 3. Some selected measured parameters are showen in Fig..19-22.

[16]

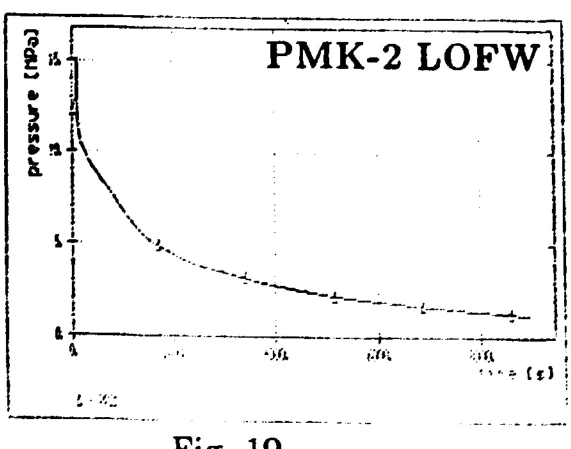


Fig. 19.
Primary Pressure

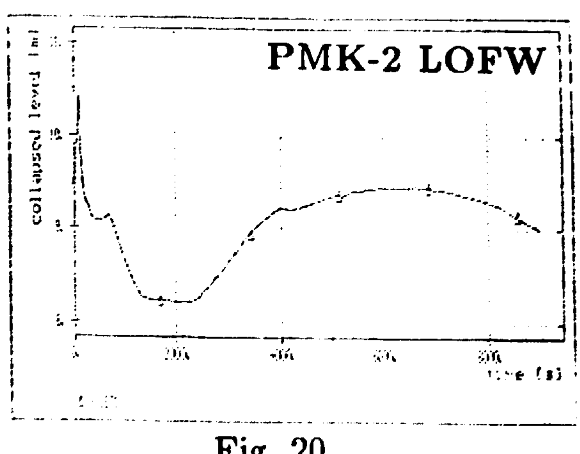


Fig. 20.
Pressurizer Collapsed level

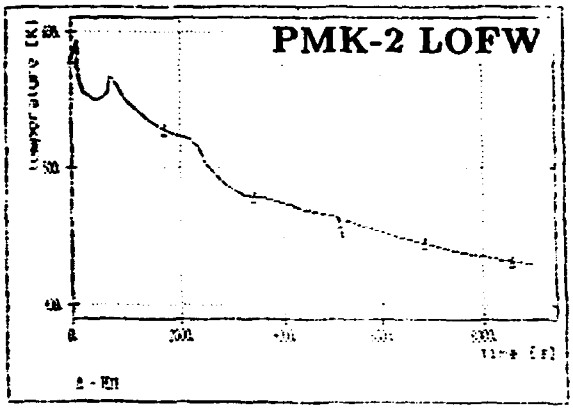


Fig. 21.
Rod Temperature

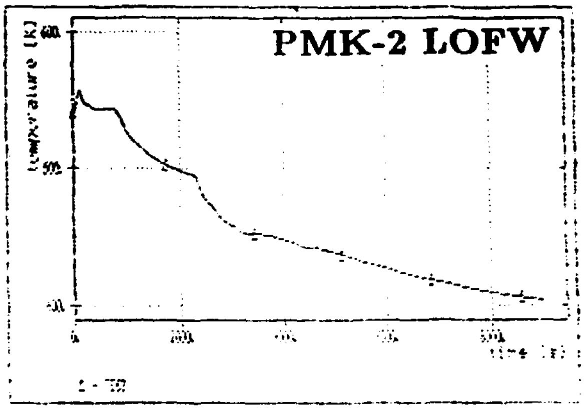


Fig. 22.
Core Inlet Temperature

6.3. Comparison of LOBI BT17 and PMK-2 LOFW

Comparing the LOBI BT17 and PMK-2 LOFW tests the following main differences can be observed. In spite of the similar sequence of events the measured parameters are absolutely different. In the case of LOBI test the loss of feedwater event lead quickly to the decrease of heat removal from the primary side and that resulted in primary pressure increase and PORV opening. In the case of PMK-2 the large thermal inertia of water masses in the SG secondary side were able to remove continously the heat produced in the core simulator from the primary side. The scram action was enough to provide pressure decrease in the primary side of the PMK facility. In the case of LOBI the secondary side was not completely isolated just the steam flow was reduced. In the case of PMK-2 test the MSIV isolated completely the secondary side and the SG RV had to handle the high secondary pressure. In the PMK-2 experiment the secondary side feed and bleed started much earlier, than in the LOBI test and this procedure was able to maintain the monotone cooldown of the primary system. In the LOBI experiment the secondary side feed and bleed started only when the core dryout started and the procedure was not able to stop the cladding temperature increase. In the PMK-2 experiment the PORV was not opened at all. In the LOBI experiment the loss of primary water through the PORV finally lead to the dryout of the upper part of the core. A summary of main events of LOBI BT17 and PMK-2 LOFW are listed in Table 5.

Table 5.
Sequence of Main Events for
LOBI BT17 and PMK-2 LOFW Experiments

Event	LOBI BT17	PMK LOFW
Loss of feedwater	0	0
Steam line closure	37	0
Core power to decrease	37	95
SG RV open	—	54
SG RV close	—	169
Pump coast down	542	532
PORV first open	1453	—
Core dryout starts	4935	—
Secondary bleed start	5307	702
Secondary feed start	5133	2247
HPIS on	5836	—
LPIS on	—	10200

7. CONCLUSIONS

LOBI-MOD2 Test BT-17 provided a wide basis for the analysis of loss of feedwater transients with secondary feed and bleed procedure. The experimental results were used for the understanding of basic phenomenologies and verification of the CATHARE code. The computational analysis showed that the CATHARE code was capable to simulate the present transient but some problems were also observed. The comparison of LOBI and PMK-2 loss of feedwater tests pointed out the basic differences between the VVER and the western design PWR related integral test facilities.

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ABBREVIATIONS

ACCORD - Assistance of the Community in Cooperation in Research and Development
AFW — auxiliary feedwater
AEKI - Atomenergia Kutató Intézet
AM — accident mitigation
ATWS - Anticipated Transient without Scram
BMFT - Bundesminister für Forschung und Technologie
BL — broken loop
CATHARE - Code for Analysis of Thermal-Hydarilucs during an Accident of Reactor and Safety Evaluation
CEC - Comission of the European Communities
CL — cold leg
EC - European Community
ECC - Emergency Core Cooling
HA — hydroaccumulator
HL — hot leg
HPIS — high pessure injection system
IL — intact loop
JRC - Joint Research Center
LOBI - Loop Blowdown Investigation
LOCA - Loss of Coolant Accident
LOFW - Loss of Feedwater
LPIS — low pressure injection system
MCP - Main coolant pump
MFW — main feedwater
MSCB — main steam collector break
NA — specified Not Available during the sequence
NU — not existing
NU — not used
O/C — open/close
OD/ID — outer/inner diameter
PCS - Primary Cooling System
PHARE - Poland and Hungary : Assistance to the Reconstruction of the Economy
PMK - Paks Modell Kísérlet
PORV — pressure open relief valve

PRZ — pressurizer
PS — primary side
PTS — pressurized thermal shock
PWR - Pressurized Water Reactor
RV — relief valve
SG - Steam generator
SGTR - Steam Generator Tube Rupture
SL — steam line
SLB — steam line break
SS — secondary side
SV — safety valve
UH — upper head
UP — upper plenum

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