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REPORT

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POST TEST ANALYSIS OF THE LOBI BT01 EXPERIMENT

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Budapest, December 1994

Z. Hózer, A. Takács :

Post Test Analysis of LOBI BT01 Experiment

ABSTRACT

~~The paper describes~~ the LOBI experimental facility and the BT01 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 secondary side break test performed on the PMK-2 facility.

Z. Hózer, A. Takács :

A LOBI BT01 kísérlet kiértékelése

KIVONAT

A cikk röviden ismerteti a LOBI kísérleti berendezést és a BT01 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 szekunder oldali töréses kísérlettel is.

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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 on 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 BT01 has been analyzed. This experiment represents a small break transient in the secondary side (steam line) followed by special conditions for the establishment of pressurized thermal shock and accident management procedures. The computational analysis was performed by the CATHARE code. The experimental behavior was compared to a secondary side (steam collector) break on 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]

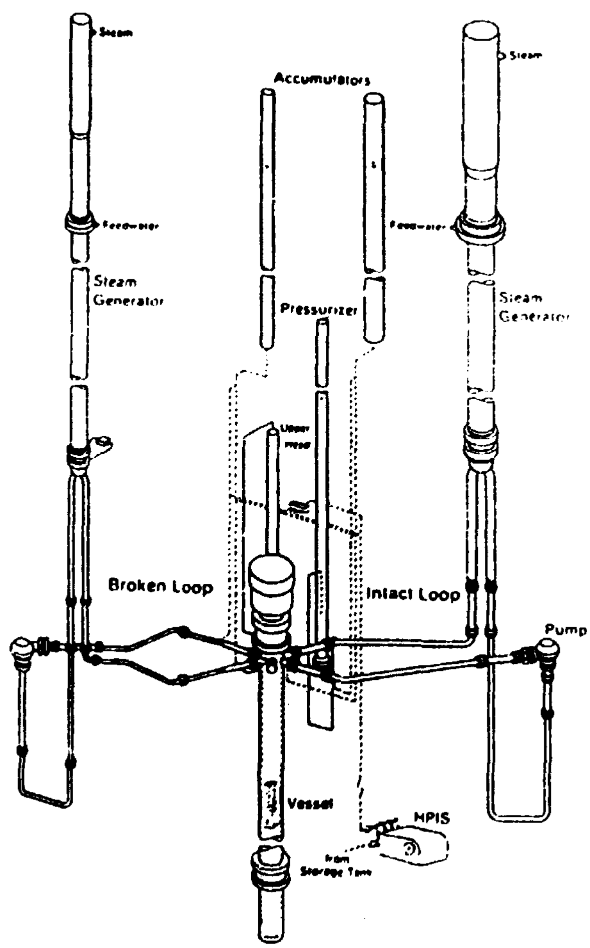


Fig. 1.

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

Table 2.
Setpoints and Sytem Configuration for Tests
LOBI MOD2 BT01 and PMK-2 MSCB

LOBI MOD2		PMK-2
DOEL 1-2 PWR 2 loops, 390 MWe	simulated power plant	VVER-440/213 6 loops, 440 MWe
BT01	test	MSCB
steam line break	test type	steam collector break
SL of IL SG	break position	on modelled SC
8.9mm	break diameter	9.5mm
to BL HL	PRZ connection	to HL
on during Phase 2	PRZ heaters	off
on during Phase 3	PRZ cooling coils	NE
at 16.5 MPa	PORV open	at 13.2 MPa
to UP and downcomer	UH connection	to UP
on at 11.7 MPa	HPIS	NA
NA	LPIS	on at 1.0 MPa
on at 4 MPa	HA	NA
on	AFW	on at 0.7 MPa
on	MCP seal water	NE
IL: off BL: on in Phase 2	MCP locked rotor res.	NE
open at 7.0 MPa	SG SRV	open at 5.3 MPa
atmosphere	containment	atmosphere

3. LOBI BT01 Test Description

Test BT-01 represents a special transient initiated by a small (10%) Steam Line Break (SLB). It was performed on the LOBI-MOD2 test facility on January 24, 1986.

Referring to PWR's secondary system faults, SLB transients are generally analyzed to verify the mitigative features of the engineered safety systems with respects to both plant integrity and environment protection.

The establishment of the overall experimental profile including also the test definition and specification was based on a reactor reference calculation for the DOEL 1-2 plant, a twin 2 loop 390 MWe PWR.

Test BT-01 was defined to cover the most important phenomenologies during overcooling transients. The overall experimental profile consists of three phases:

1. Steam Line Break (0-600s). The primary objective of this phase was to investigate the blowdown of the faulted SG, primary cooldown rate, pressurizer and surgeline behavior as well as heat transfer in SGs.
2. Establishment of pressurized thermal shock (PTS) conditions (600-2349s). Since SLBs belong to the class of overcooling transients and pose pressurized thermal shock to some power plants, a PTS scenario was conceived for the second phase of the test. This phase starts with tripping the pumps.
3. Accident mitigation based on optimized recovery guidelines (2349-6307s). Primary system depressurization using the pressurizer cooling system and in time, unaffected steamgenerator cooldown constituted the preferred procedure to recover and maintain and adequate degree of primary subcooling.

The system configuration and the specified setpoints for test BT-01 can be seen in Table 2, left column.

The break was assumed to occur in the steam line between the hypotetical flow restrictor and the main steam line isolation valve. It was simulated with a quick opening on-off valve coupled with an upstream orifice of 8.9 mm to scale the assumed break size of cca. 10% in the reference DOEL system. The break size was selected in order to produce a break flow which never would exceed the steady state value during the test thus preventing water carryover in the steam line. Regarding the pressurizer the setpoint of the simulated PORV is 16.5 MPa. The heaters are activated in the second phase of the test. The surge line is connected to the broken loop hot leg. The externally mounted upper head is connected to the upper plenum and the upper downcomer. An additional connection is applied to keep the fluid temperature at about the same value in the upper head and in the upper downcomer. The bypass flow through this duct is estimated between 2.4 and 3.7 % of the nominal core flow.

For the proper simulation of the locked pump hydraulic resistance a locked rotor simulator unit is installed at the outlet side of the single loop main coolant pump. This means a special valve which provides a flow area reduction to about 18 % of the normal full flow area. For operational reasons the orifice part of the valve is normally inserted cca. 4 s after pump coastdown to zero speed.

Coming to the test conditions and operation procedures firstly the initial and boundary conditions will be summarized.

3.1 Initial Conditions

The transient was started from low power conditions, hot zero power state. In steady state phase thermal equilibrium was maintained between the primary and secondary side and the core heating power was kept to be sufficient to compensate the overall heat loss to the ambience.

The initial conditions for test BT01 are shown in Table 4, left column.

3.2 Boundary Conditions

Boundary conditions for the first phase were generally scaled to the conditions to be expected in the reference system following an SLB event. In phases 2 and 3 they were set to correctly reproduce thermal-hydraulic phenomena. Boundary conditions for the three phases are summarized in Table 4.

Concerning the core heat power it was switched off after transient initiation so no nuclear thermal-hydraulic feedback was simulated.

The main coolant pumps operated at nominal steady state speed during phase 1. In phase 2 pumps coasted down to zero speed at 608 s. Both pumps were controlled to stop within 2 s. At 616 s the locked rotor resistence simulator was inserted in the broken loop. In phase 3 the main coolant pumps were restarted at nominal speed at 2506 s also the resistence simulator has been removed from the broken loop pump.

The main feedwater was used neither in steady state nor in accident phase. However the auxiliary feedwater system has been applied after initiating the SLB phase as main feed system to control the water level in the intact loop SG.

The HPIS was starting at a pressure of 11.7 MPa injecting into the intact cold leg. This set point corresponds to 431 s transient time and later on it follows the prescribed pressure-mass flow characteristics. During the second part of the transient a constant flow rate of cca. 0.02 kg/s was kept to refill the pressurizer level to cca. 3.8 m. In the third part the HPIS was operated as a make-up system to control the pressurizer level.

The hydroaccumulators were kept in stand-by conditions at a pressure of 4.0 MPa in the first period of the test. Then they were disabled.

The SG safety relief valve did not get in action through the test since the pressure never reached the setpoint of 7.0 MPa. The intact loop SG cooldown started during the accident mitigation phase as it was specified. This event happened when a subcooling of cca. 15 °C in primary intact hot leg was achieved.

Concerning the PORV it was expectedly not operating since the setpoint was at 16.5 MPa. The pressurizer heaters of cca. 20 kW was activated at 1204 s of the second phase and deactivated at 2349 s when initiating the third phase. The pressurizer cooling coils were cycled during the third phase to preserve primary system subcooling (flow rate: 0.4 kg/s, inlet fluid temperature 22 °C).

In the following the major test results will be reviewed including the scenario and also a brief analysis of the test will be presented.

3.3 Overall Experimental Profile

The sequence of main events occurring during the course of the transient are listed in Table 5.

The transient was initiated at time 0 s with the simulation of a small steam line break assumed to occur downstream of the reference reactor steam line flow restrictor. At this moment the core heating power was switched off and simultaneously auxiliary feedwater injection was initiated in both the affected and unaffected steam generators. Up to 4 s of the transient both steam generators boiled down. Thereafter following the isolation of the unaffected steam generator via the closure of the simulated closure of the simulated main steam line isolation valve, only the affected steam generator continued to depressurize. The primary system cooled down and the primary system pressure reached the setpoint of the high pressure ECC injection system which became operational at 431 s. At 524 transient seconds AFW injection to the affected steam generator was terminated in response to a low steam generator downcomer water level (cca. 1 m). During this initial part of the transient the main coolant pumps were kept in operation at nominal speed.

The initiating phase of Test BT-01 was terminated at 600 s; thereafter the transient progressed through a second phase which aimed at establishing thermal-hydraulic conditions of interest for the pressurized thermal shock issues. To maximize thermal stratification in cold legs and in the pressure vessel downcomer the main coolant pumps were tripped at the initiation of this phase; also to balance the flow resistance of the primary loops under natural circulation heat transport; one additional locked rotor resistance simulator was inserted at the outlet of the broken loop main coolant pump. To study the mixing mechanism in case of very low loop flow HPIS injection into the intact loop cold leg was delivered at a constant rate. At 1204 s the pressurizer heating system (i.e. 20 kW) was activated to ensure the preservation of a steam space on top of the pressurizer which was refilling. This caused a repressurization of the primary system. The AFW injection into the unaffected steam generator was terminated at 1612 s in response to the high steam generator downcomer water level. This phase of Test BT-01 ended at 2349 s when the pressurizer water level had reached a value of about 80% of nominal.

Test BT-01 was terminated with an accident mitigation phase. To establish required operating conditions the main coolant pumps were reactivated at the nominal speed and the pressurizer heaters were switched off. At 2534 s the pressurizer cooling system was activated to reduce primary system subcooling and at 4302 s, following the attainment of an adequate intact loop hot leg subcooling margin of cca. 15 °C, cooldown of the intact loop steam generator at a rate of cca. 100 °C/h was initiated. Thereafter the pressurizer cooling and heating systems were specified to be cycled to keep the subcooling margin in the range 10-15 °C. Fluid make-up to compensate the shrinkage caused by primary system cooling was ensured by the pump seal water injection and pressurizer level control. As it was previously mentioned, during the evolution of this phase a leak developed in the broken loop cold leg which called for the intervention of the HPIS to compensate the lost inventory which could not be compensated by pump seal water injection. This leak was responsible for the voiding in parts of the primary circuit and loss of subcooling margin. This phase was terminated at 6307 s.

[4],[5],[6]

Table 3.
Initial Conditions for Tests
LOBI MOD2 BT01 and PMK-2 MSCB

	LOBI BT-01	PMK-2 MSCB	Unit
Primary Pressure	15.46	12.84	MPa
Core Power	105.6	677.6	kW
IL mass flow	22.2	—	kg/s
IL vessel inlet temp.	283.9	—	°C
IL vessel outlet temp.	284.1	—	°C
BL mass flow	7.5	4.88	kg/s
BL vessel inlet temp.	283.1	266.8	°C
BL vessel outlet temp.	284.1	294.6	°C
PRZ water level	4.95	8.88	m
IL SG SL pressure	6.64	—	MPa
IL SG mass flow	0.055	—	kg/s
IL SG inlet temp.	130.	—	°C
IL SG outlet temp.	282.	—	°C
IL SG downcomer level	8.15	—	m
BL SG steam dome pressure	6.64	4.63	MPa
BL SG mass flow	0.1	0.36	kg/s
BL SG inlet temp.	122.	187.	°C
BL SG outlet temp.	282.	257	°C
BL SG downcomer level	8.15	—	m

Table 4.
Boundary Conditions for Tests
LOBI MOD2 BT01 and PMK-2 MSCB

	LCBI BT-01 Phase 1	LOBI BT-01 Phase 2	LOBI BT-01 Phase 3	PMK-2 MSCB
	10% SLB 0-600s	PTS 600-2349s	AM 2349-6307s	
MCP	on (100%)	off	on (100%)	coastdown at 26s
BL locked rotor	off	on at 616s	off	NE
MFW	off	off	off	off at 26s
AFW IL	on	off at 1612s	on at 5158s	NE
AFW BL	on at 0s off at 524s	off	off	on at 0.7 MPa
Power	0	0	0	off at 26s
HPIS	on at 431s	on(0.02kg/s)	off at 2349s on at 5110s off at 5817s on at 5945s	NA
LPIS	NA	NA	NA	on at 1.0MPa
HA	4.0MPa at stand-by	off	off	NA
SG SRV	7. MPa(NU)	7. MPa(NU)	open at 4302s	open at 375s
PRZ PORV	16.5 MPa(NU)	16.5 MPa(NU)	16.5 MPa(NU)	open 13.2 MPa close 12.8 MPa
PRZ heaters	off	on at 1204s	off	off
PRZ cooling	off	off	on at 2534s off at 4411s on at 5818s off at 5926s	NE

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 BT01 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. The break modelling was changed from the standard "BREAK" operator to a "TEE" with break boundary conditions. According to this modification the steam line scheme was changed from one 1D element to two 1D elements plus a TEE. The main elements of the nodalization scheme are presented in Fig. 2. The pressurizer walls were divided into three parts in order to have possibility for a better simulation of PRZ heaters and cooling coils.

The calculation was based on CATHARE V1.3E version. In this version some modifications have been made in order to improve results of earlier LOBI anlysises (e.g. thermal connection between two elements of the same circuit).

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. The manual actions (e.g. MCPs on and off) were activated at the same time as scheduled in the experiment.

The results of CATHARE calculation showed that the first phase (steam line break) was calculated in very good agreement with experimental data. The primary pressure and secondary pressure in both intact and broken steam generators are presented in Fig. 3. The calculated pressurizer level was also very close to the measured level (Fig. 4.). The calculated primary coolant temperatures showed that the broken loop hot leg temperature was higher due to the hot water coming from the pressurizer surgeline (Fig. 5.). Furthermore the calculated primary coolant temperatures and even the rod temperature were lower for approx. 10 °C at the end of phase 1 than the experimental values (Fig. 5-7.).

The second phase (PTS) proved to be more difficult from computational point of view concerning pump coast down and locked rotor resistance, for these actions resulted in very low coolant velocities (see Fig. 7-8.) and even in flow reversal in the intact loop for short periods. The coolant temperature stratification in the primary circuit observed in the experiment was pointed out in the calculation too (Fig. 10-13.). The average primary coolant temperature continued to decrease and at the end of phase 2 it was lower than the experimental values for approx. 15 °C. The lower coolant temperature caused lower pressurizer level as well (Fig. 14.).

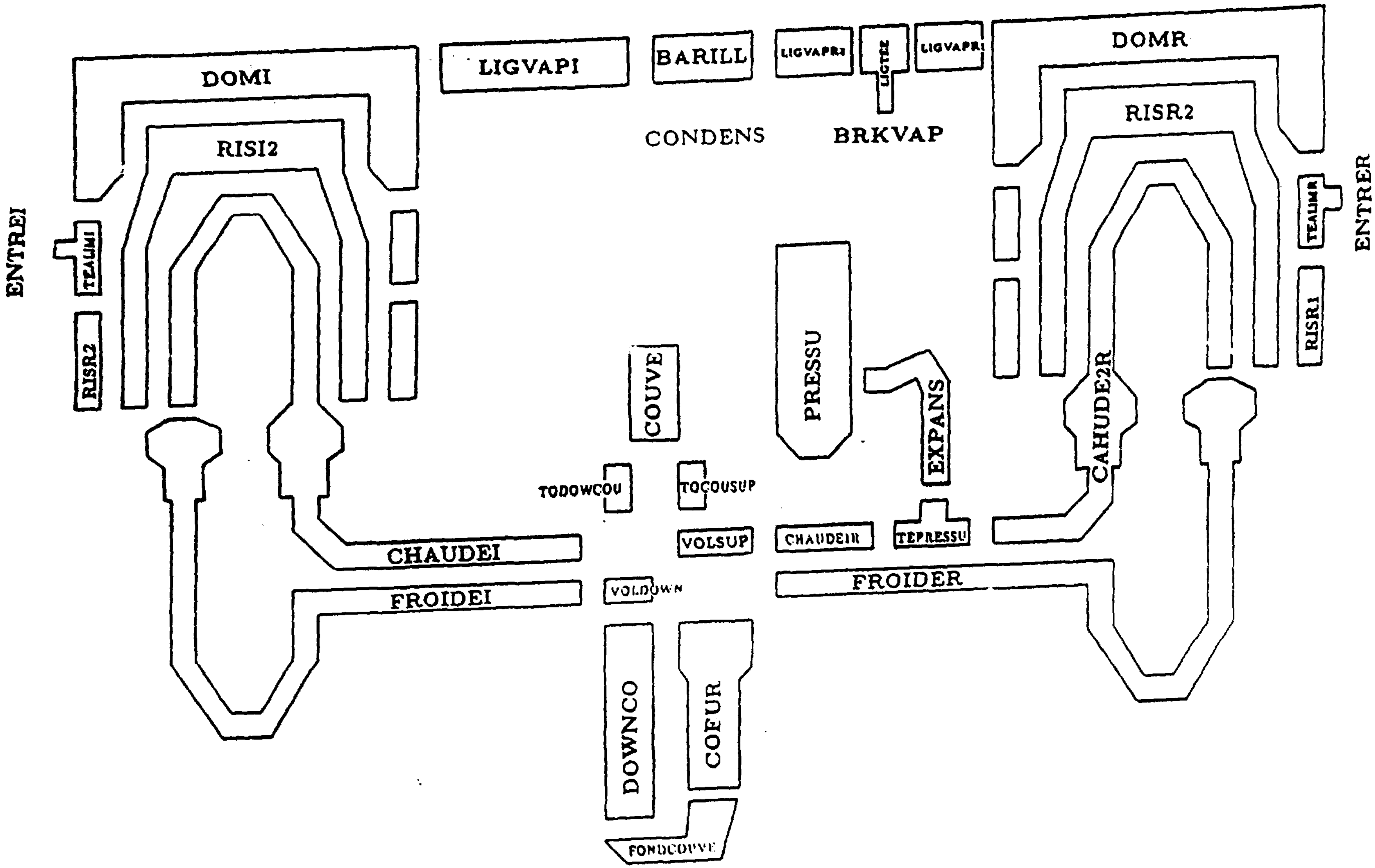


Fig. 2.
CATHARE Nodalization Scheme for LOBI BT01

Fig. 3.

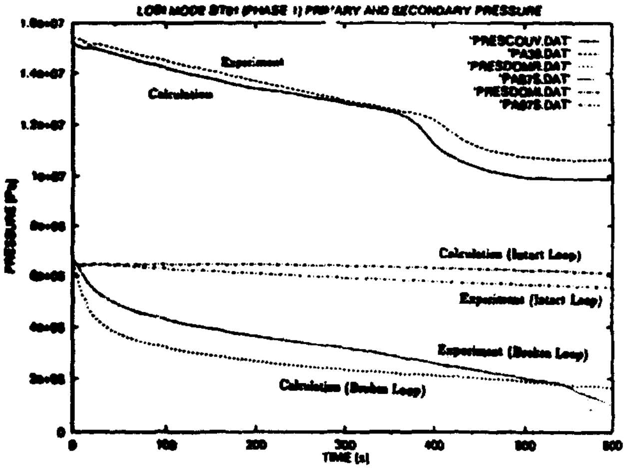


Fig. 4.

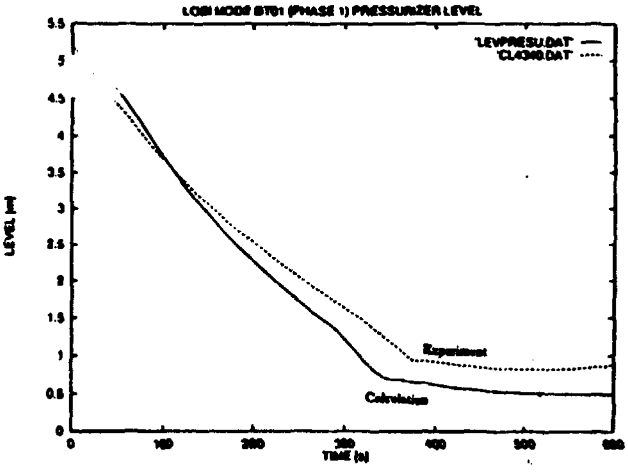


Fig. 5.

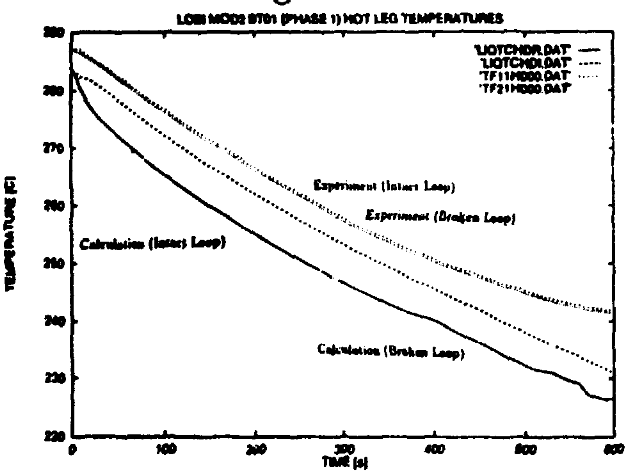


Fig. 6.

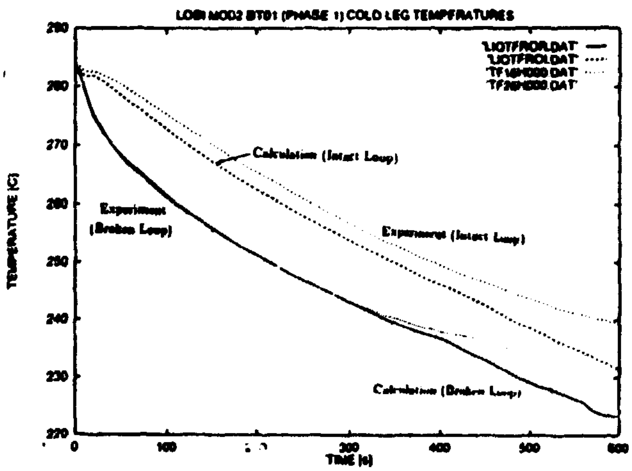


Fig. 7.

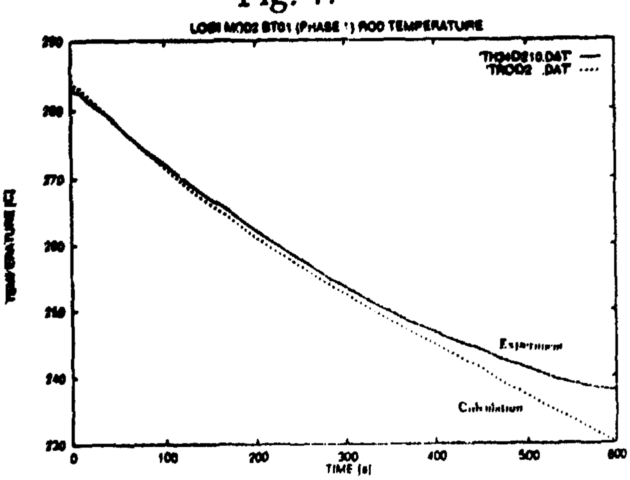


Fig. 8.

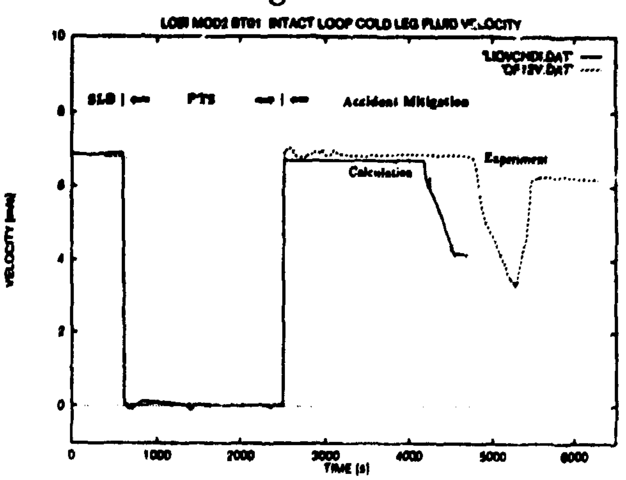


Fig. 9.

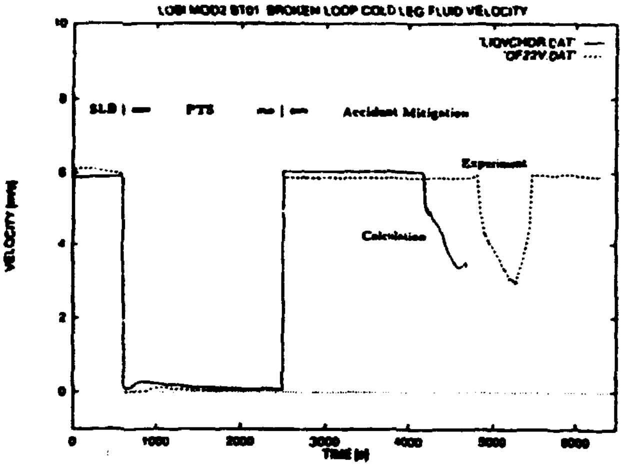


Fig. 10.

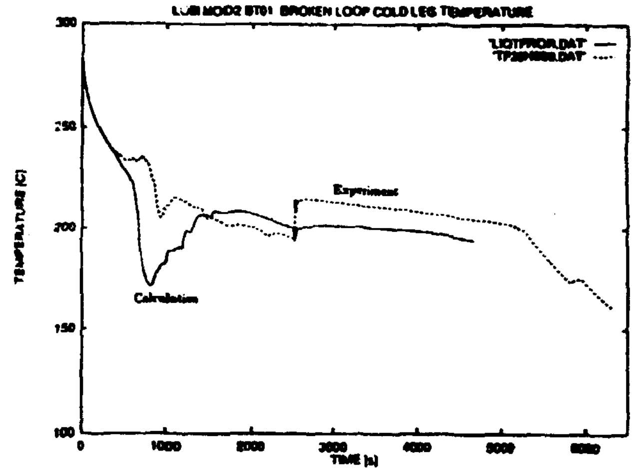


Fig. 11.

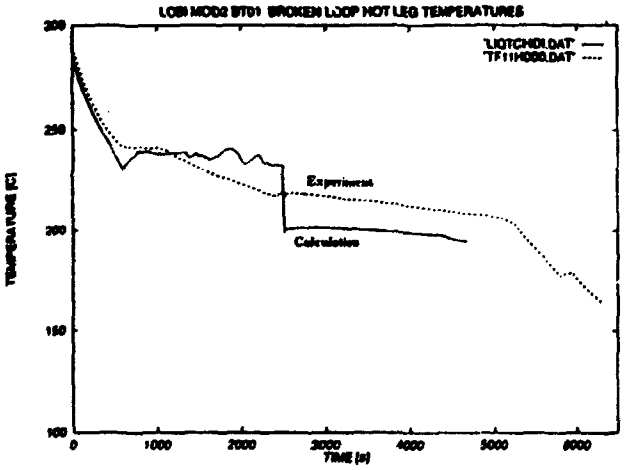


Fig. 12.

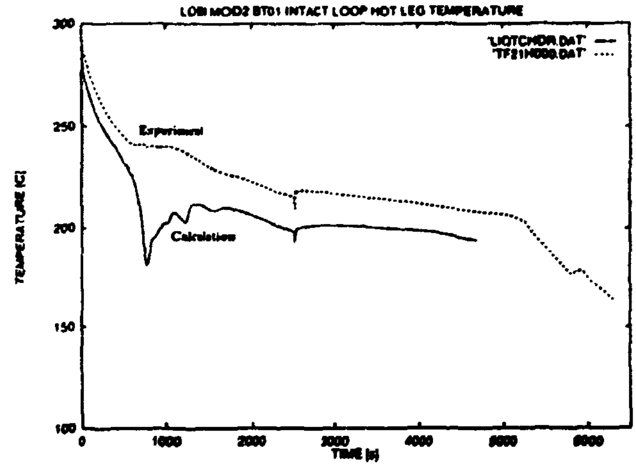


Fig. 13.

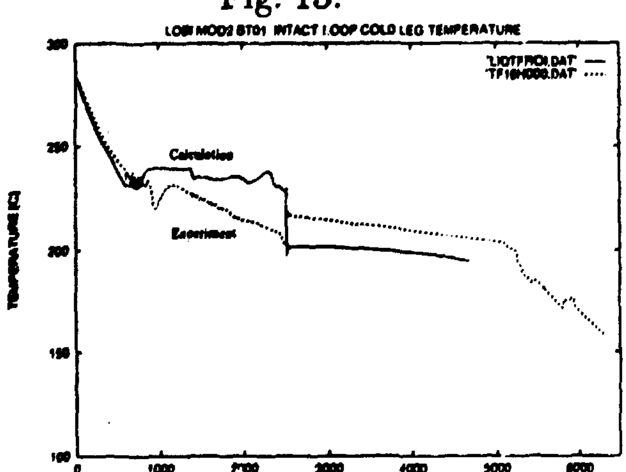
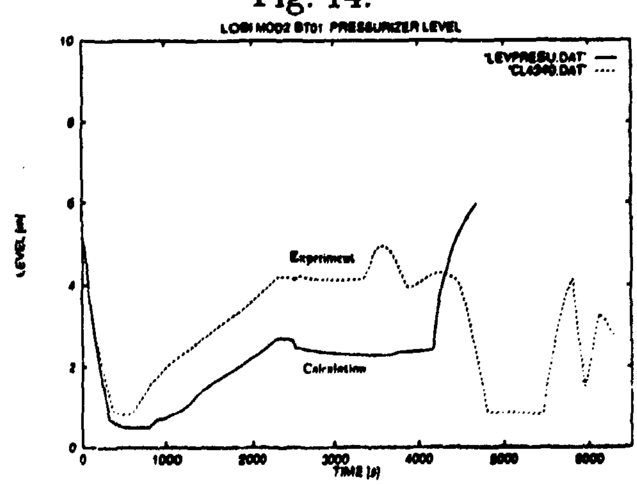


Fig. 14.



The primary pressure curves are shown in Fig. 15. The calculated value is close to the experimental one in phase 1, the agreement is also satisfactory in phases 2 and 3. In phase 3 the pressure decrease was faster than in the experiment and it resulted in voiding in the primary circuit. The primary liquid subcooling was lost at 4100 s (Fig. 16.), similar phenomena was observed in the experiment at 4800 s. The voiding in the calculation lead to pressurizer level increase (Fig. 14.).

The secondary pressure (Fig. 17.) was calculated with good accuracy during the whole transient. The broken loop side pressure curve followed the break flowrate curve (Fig. 18.). The calculation showed that there was no water carry over through the break.

In the third phase of the experiment a not quantified leak occured in the primary circuit. On the basis of total measured mass inventory this effect was estimated in the calculation but its accuracy may be questionable (Fig. 19.). The calculation was stopped at 4700 s.

Fig. 15.

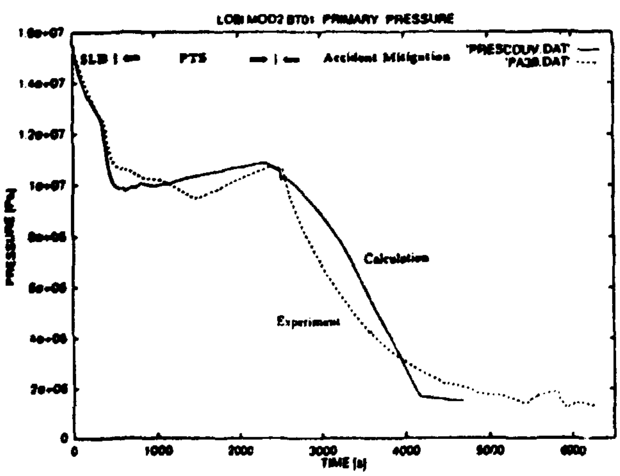


Fig. 16.

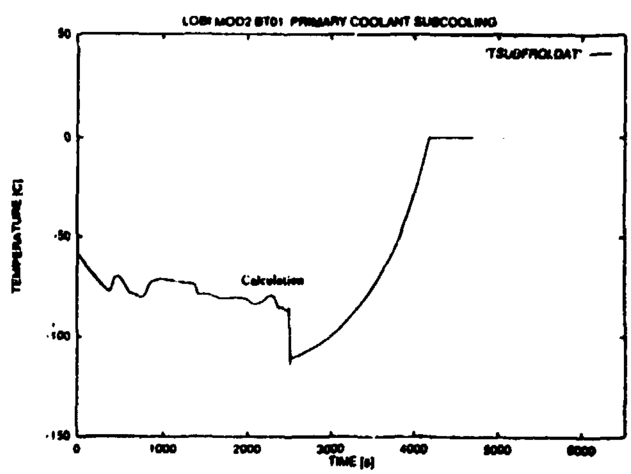


Fig. 17.

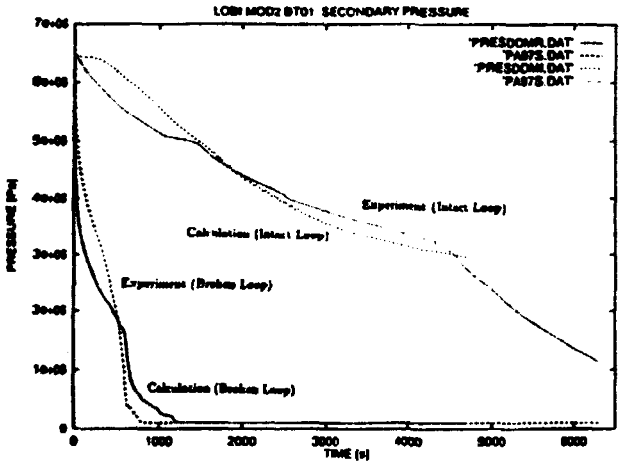


Fig. 18.

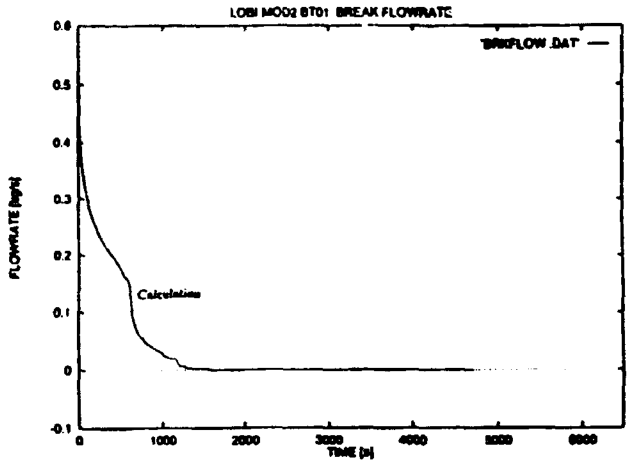


Fig. 19.

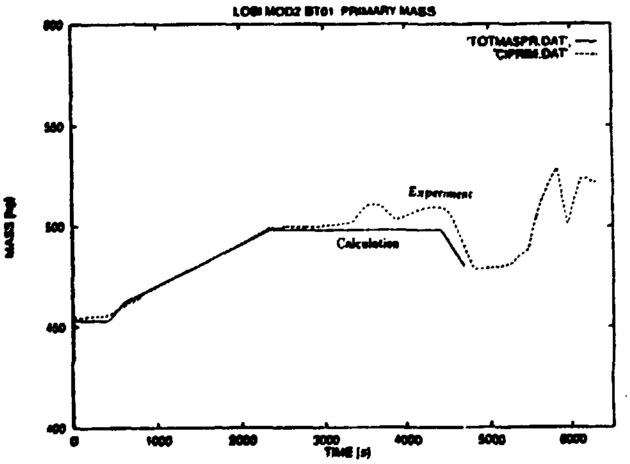


Fig. 20.

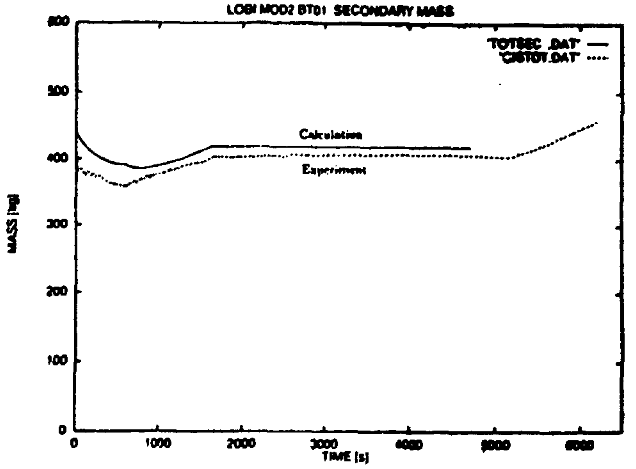
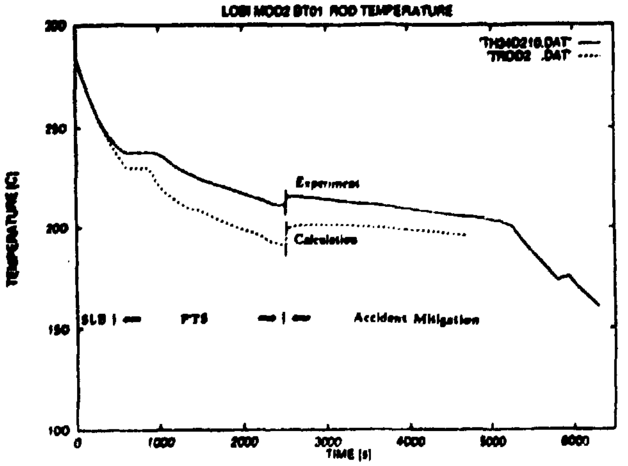


Fig. 21.



6. Comparison of LOBI BT01 and PMK-2 MSCB 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. 22.

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.

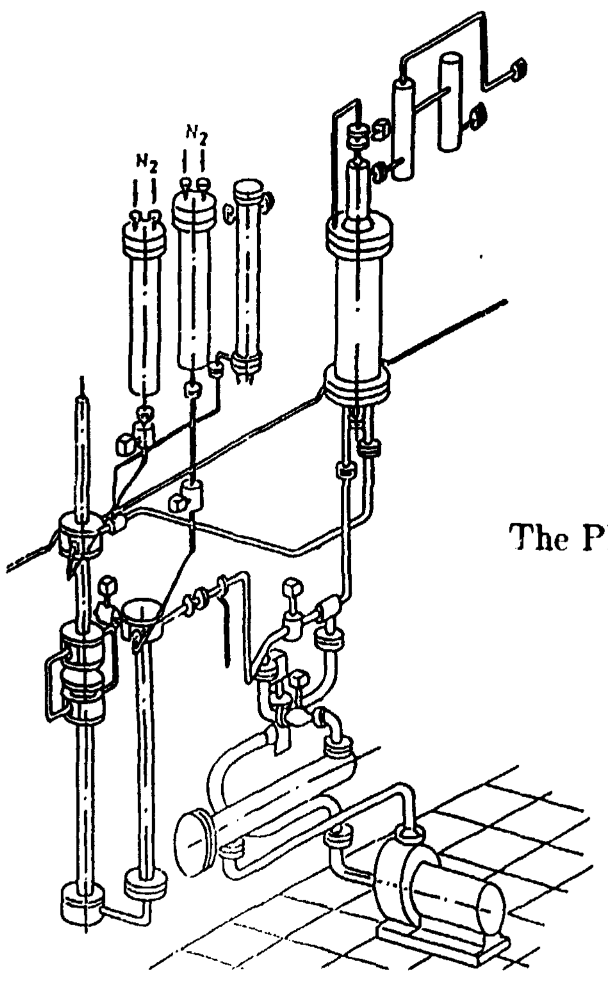


Fig. 22.
The PMK-2 Test Facility

6.2. The PMK-2 MSCB Experiment

At the PMK-2 facility a main steam collector break (MSCB) experiment had been performed. This is the only secondary side break experiment completed on VVER related integral test facilities. For this reason it seemed to be interesting to make a comparison of this PMK-2 test with LOBI BT01 test which represents a secondary side break as well.

The MSCB transient was started from nominal conditions (100% power). The break was located on the steam collector which is placed after the main steam isolation valve (MSIV). The transient started at 0 time. The signal of secondary pressure gradient change initiated the following actions: reactor scram, MCP coast down, secondary side isolation by MSIV. The primary pressure increase lead to the opening PORV which was open for 10 s. In the same time secondary side bleed procedure was started. Between 8000-11000 s some parts of the primary circuit reached saturated state and the the pressurizer filled up. Secondary side feed started at 10850 s. In this experiment the HPIS and hydro-accumulators were not activated. The LPIS system was activated in later phase of the experiment by low value of primary side pressure. The primary and secondary pressures and temperatures decreased monotonely during the main part of the transient (see Fig. 23-26.).

Fig. 23.

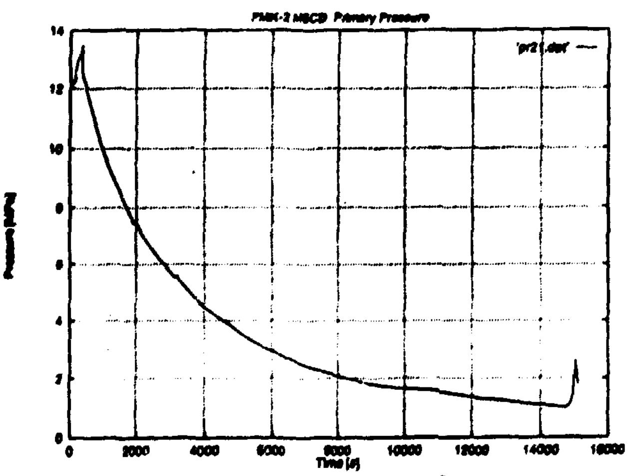


Fig. 24.

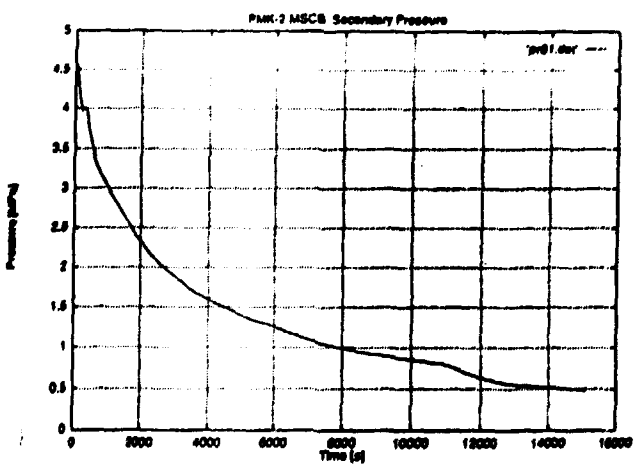


Fig. 25.

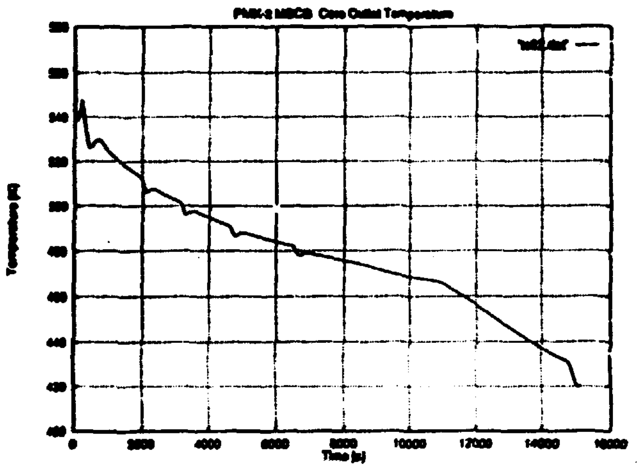
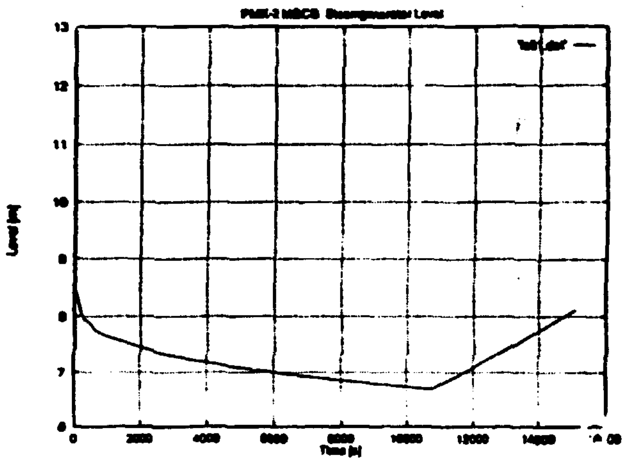


Fig. 26.



6.3. Comparison of LOBI BT01 and PMK-2 MSCB

Comparing the LOBI BT01 and PMK-2 MSCB experiments one can see two main differences:

a, The initial conditions were diffe-rent (hot zero power for LOBI and nominal power for PMK-2).

a, The break location made possible the isolation of secondary side in PMK-2 experiment while in the case of LOBI the break was pla-ced before the isolation valve and so the break was active during the whole transient and only the in-tact loop's secondary side could be isolated.

Furthermore the LOBI BT01 test consisted of three phases with several operator actions including pump coast down and start-up. The second phase aimed at the establishment of PTS conditions and in the third phase ac-cident mitigation procedures were con-sidered. In the PMK-2 experiment the break played a role only in the first 26 seconds and after the isolation practi-cally the technological systems with re-duced ECC capabilities (no HPIS and hydroaccumulators) were tested.

A list of main events of LOBI BT01 and PMK-2 MSCB tests is given in Table 5.

[14]

Table 5.
Main Events of LOBI MOD2 BT01
and PMK-2 MSCB Transients

Time	LOBI MOD2 BT01	PMK-2 MSCB	Time
0	start of SLB simulation core power off PRZ heaters off AFW injection on	Break open	0
4	Isolation of IL SG		
15	Heat transfer in intact SG reverses		
		Core power off MCP coastdown initiated Break closure Isolation of sec. side	26
370	Pressurizer surgeline uncovers		
		PRZ SV open Sec. bleed on	375
431	HPIS on		
		PRZ SV closed	385
524	AFW injection in affected SG off		
550	Affected SG dry		
599	PCS pressure cca. 108 bar		
600	HPIS flow rate set at constant value		
608	MCPs coastdown initiated		
612	MCPs coastdown completed		
616	BL MCP locked rotor resistance on		
1204	PRZ heating on		
1500	PCS repressurizes		
1612	AFW injection in IL SG off		

2348	Pressurizer water level > 3.8 m HPIS off PCS pressure cca. 107 bar		
2349	PRZ heating off		
2505	BL MCP locked rotor resistance off		
2506	MCPs restarted		
2534	Pressurizer cooling system on		
4302	PCS hot leg subcooling cca. 15 K Cooldown of IL SG initiated		
4411	Pressurizer cooling off		
4800	Loss of PCS subcooling		
5110	HPIS on		
5158	AFW injection in intact loop SG on		
5817	Pressurizer water level> 3.8 m HPIS off		
5818	Pressurizer cooling on		
5926	Pressurizer cooling off		
5945	HPIS on		
6307	End of Test		
		Level in reactor model	9250
		Sec. feed on	10850
		LPIS on	14700
		End of test	15100

7. CONCLUSIONS

LOBI-MOD2 Test BT01 provided a wide basis for the analysis of complex thermohydraulic transients occuring after a steam line break in PWRs. 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 well the first phase of the transient, but some problems were observed during the second and third phases of the test.

The comparison of LOBI steam line break and PMK-2 main steam collector break showed that in spite of the fact that both tests belong to secondary side break transients the behavior of VVER and the western design PWR related integral test facilities were totally different.

ACKNOWLEDGEMENT

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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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