Project Report

The LOBI Integral System Test Facility Experimental Programme

Carmelo Addabbo and Alessandro Annunziato

Commission of the European Communities, Joint Research Centre, Via E. Fermi 1, 21027 Ispra, Italy

Correspondence should be addressed to Carmelo Addabbo, carmelo.addabbo@ec.europa.eu

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The LOBI project has been carried out in the framework of the European Commission Reactor Safety Research Programme in close collaboration with institutional and/or industrial research organizations of EC member countries. The primary objective of the research programme was the generation of an experimental data base for the assessment of the predictive capabilities of thermal-hydraulic system codes used in pressurised water reactor safety analysis. Within this context, experiments have been conducted in the LOBI integral system test facility designed, constructed, and operated (1979–1991) at the Ispra Site of the Joint Research Centre. This paper provides a historical perspective and summarizes major achievements of the research programme which has represented an effective approach to international collaboration in the field of reactor safety research and development. Emphasis is also placed on knowledge management aspects of the acquired experimental data base and on related online open access/retrieval user functionalities.

1. Background

The LOBI project originated from a reactor safety research and development contract between the European Commission (EC) and the former Bundesminister für Forschung und Technologie (BMFT) of the Federal Republic of Germany. On the basis of contingent and perceived safety requirements, the BMFT decided in 1972 on the need of an integral system test facility for thermal-hydraulic investigations relevant to accident conditions in pressurized water reactors (PWRs) of German design.

A general international consensus of opinions emerged in the early 70s on the need to provide reliable methodologies for a realistic estimate of emergency-core-cooling system (ECCS) performance which was being questioned as large power reactors were being introduced. Due to the limited sophistication of safety codes and to the lack of relevant experimental data for assessing their predictive capabilities, sufficient conservatism was prescribed in the safety evaluations of design basis accidents (DBAs), such as loss-of-coolant accidents (LOCAs), in order to account for worst-case uncertainties. This entailed stringent licensing requirements and undesirable operational constraints on nuclear power plants.

Within this context, reactor safety research and development programmes were formulated at the international level to improve the understanding and modelling capabilities of the thermal-hydraulic phenomenologies governing the course of a LOCA or of any other anticipated abnormal occurrence in water cooled reactors. Emphasis was placed mainly on deterministic methodologies supported, as appropriate, by probabilistic risk assessment studies with the aim to better understand accident progression and to substantiate the request for the eventual relaxation of over-conservatism in some safety analysis acceptance criteria.

The European Commission has been engaged in nuclear safety research activities since the signing in 1957 of the treaties establishing its precursors; that is, the European Atomic Energy Community and the European Economic Community. In line with its institutional role, the action of the Commission has been mainly devoted to a systematic effort in promoting the harmonization of safety practices and methodologies among the member states.

On the basis of its tender of May 1973, the Joint Research Centre was charged by the German BMFT with a R&D contract which envisaged the design, the construction, and the operation of an integral system test facility for the investigation of thermal-hydraulic phenomenologies pertinent to

PWR large break LOCAs. The contractual agreement was signed in December 1973 and was then renegotiated in 1982 extending the original research objectives to the small break LOCA and Special Transients scenarios.

In its final configuration, the LOBI experimental data base consists of 70 tests (Table 2) covering a wide range of accident scenarios relevant to the safety analysis of PWRs.

2. Research Objectives

The LOBI research programme, as initially conceived, has been mainly oriented towards the generation of an experimental data base relevant to risk-dominant accidents and transients in PWRs. Specific research objectives included:

- (i) the identification and/or verification of basic phenomenologies governing the thermal-hydraulic response of an integral system test facility for a range of conditions relevant to LOCAs and special transients in PWRs,
- (ii) the generation of an experimental data base for the independent assessment of the predictive capabilities of large thermal-hydraulic system codes used in water reactor safety analysis.

The experimental programme has been supported by comprehensive code application and assessment activities. ATHLET (DRUFAN), CATHARE, RELAP4, RELAP5, and TRAC have been largely used either within JRC or by outside organizations for test design and test prediction calculations. Development and application of advanced two-phase flow measurement techniques to support the execution of the experimental programme have constituted an integral part of the overall research strategy. At the time, a considerable effort has also been devoted to the development of an IBM version of the RELAP5 code which, together with various model improvements introduced at the JRC, has been instrumental in enabling the code calculation capabilities of many organisations within and outside the EC.

The most stringent concern in the operation of nuclear power plants is to ensure that its inventory of radioactive material remains safely confined at all times, both during normal and off-normal conditions. In order to satisfy this compelling requirement, nuclear power plants have to comply with a set of design and construction standards, operational constraints, and safety regulations.

The many different safety features which are engineered in each nuclear power plant to compensate for any departure from normal operating conditions should an accident occur, must, irrespective of the cause, provide sufficient safety margins to prevent the release of radioactive material in excess of prescribed limits, and ensure the protection of the operators and the general public.

The emergency core cooling system (ECCS) is the most important safety feature of a water-cooled reactor. Its main purpose is to maintain the reactor in a coolable geometry in the event of loss of coolant caused by a break in the main cooling system or by the operation of the safety relief systems in the case of intact circuit faults. To meet this widely

accepted requirement, the ECCS should have the capabilities to

- (i) provide sufficient core cooling by removing both residual stored energy and decay heat at a rate such to limit the maximum fuel cladding temperature to a value less than the prescribed limit thereby preventing the cladding from losing its structural integrity,
- (ii) have sufficient capacity, diversity, reliability, and redundancy to provide core cooling under all conceived accident conditions.

The ECCS safety features of a pressurised water reactor consist of a set of independent subsystems which include the high pressure injection system (HPIS), the accumulator injection system (AIS), and the low pressure injection system (LPIS); each subsystem is characterised by redundancy of equipment and flow paths.

At present, PWR safety analysis is based on the evaluation of ECCS performance during design basis accidents (DBAs) which include large break LOCAs, small break LOCAs, and special transients resulting from intact circuit faults.

3. The Experimental Installation

The LOBI test facility was designed to represent a full-power high-pressure integral system test facility 1:700 scale model of a 4-loop, 1300 MWe PWR, Figure 1. It incorporates the essential features of a typical PWR primary and secondary cooling system. The test facility was commissioned in December 1979 and was operated until June 1982 in the MOD1 configuration for the investigation of large break LOCAs; it was then extensively modified into the MOD2 configuration which incorporates design and instrumentation features best suited for the characterization of phenomenologies relevant to small break LOCAs and special transients.

The measurement system comprises a total of about 470 measurement channels. It allows the measurement of all relevant thermohydraulic quantities at the boundaries (inlet and outlet) of each individual loop component and within the reactor pressure vessel model and steam generators. A process control system allows the simulation of time- or pressure-dependent parameters such as main coolant pump hydraulic behaviour, core decay heat release, and safety injection flow rates. A fast running data acquisition system complements the experimental installation.

3.1. Mechanical Components. The test facility comprises two primary loops, the intact and the broken loop which represents, respectively, three loops and one loop of the reference PWR. Each primary loop contains a main coolant circulation pump and a steam generator. The simulated core consists of an electrically heated rod bundle arranged in a square matrix inside the pressure vessel model. The primary cooling system which is shown schematically in Figure 2 operates at normal PWR conditions, approximately, 158 bar and 294–326°C pressure and temperature, respectively.

FIGURE 1: Photographic view of the LOBI test facility.

FIGURE 2: Schematic view of the LOBI test facility.

Heat is removed from the primary loops by the secondary cooling system which contains a condenser and a cooler, the main feedwater pump, and the auxiliary feedwater system. Normal operating conditions of the secondary cooling system are 210°C feedwater temperature and 64.5 bar pressure.

3.1.1. Reactor Pressure Vessel Model. The reactor pressure vessel model comprises the pressure vessel, the core barrel tube, and the core simulator. Lower plenum, upper plenum,

FIGURE 3: Primary system pressure and heater rod temperature response to a large 200% cold leg break LOCA (Downcomer gap width 50 mm).

an annular downcomer, and an externally mounted upper head simulator are additional major components of the overall pressure vessel assembly. The reactor core is simulated by an electrically heated rod bundle consisting of 64 rods arranged in an 8×8 square matrix inside the flow shroud; heater rod bundle dimensions are reactor typical. The heater rods are directly heated hollow tubes (material 1.4948 DIN 17007), and the rod wall thickness within the heated length is varied in 5 steps to achieve a chopped cosine-shaped axial power distribution.

The upper part of the rod bundle which extends entirely into the upper plenum is formed by hollow nickel tubes connected to the upper power plate. The lower part is formed by nickel rods and flexible nickel braids which extend partially into the lower plenum where they connect to the lower-power connecting ring. The heat dissipated within these “unheated” regions amounts to about 14%.

Nine grid spacers of original design are placed along the heated length; five additional spacers are mounted in the upper unheated part of the rod bundle. Ceramic segments are arranged inside the core barrel tube forming a square flow shroud which extends over the heated length region of the rod bundle.

The upper head is simulated by an external vessel connected to the upper plenum and to the upper downcomer. Volume as well as height and relative elevations of the reference plant upper head are preserved. In the initial MOD1 version of the test facility, the annular downcomer formed by the pressure vessel and the core barrel tube had a gap width of 50 mm which was later decreased to 12 mm to better represent fluid volume distribution.

3.1.2. Steam Generators. The LOBI test facility contains two shell and inverted U-tube type steam generators having a geometrical configuration similar to that of the reference plant. In the MOD1 configuration, the steam generators were designed to preserve heat transfer capabilities without proper

simulation of secondary side fluid distribution. In the MOD2 configuration, the steam generators were designed with the aim to better represent thermal-hydraulic phenomenologies of interest in intact circuit faults.

The overall scaling ratio which required a capacity ratio of 3 : 1 between the intact and the broken loop steam generator led to a heat transfer exchange power of 3.96 MW (24 U-tubes + 1 installed spare) and 1.32 MW (8 U-tubes + 1 installed spare) for the intact and broken loop SG, respectively.

Each steam generator consists of a single cylindrical pressure vessel with an annular downcomer separated from the riser region by a skirt tube. This tube is supported above the tube plate and carries the coarse separator; a fine separator is arranged in the uppermost part of the steam dome. The U-tubes are arranged in a circle within the riser region around an axially mounted filler tube, with the U-bends crossing over one another above it. This design permits cross-flow between cocurrent and counter-current legs of the U-tubes over their entire length and heat and mass transfer between riser and downcomer to account for the recirculation characteristics of the prototypical system. An adjustable throttle device is installed at the lower end of the downcomer to properly set the recirculation ratios.

Feedwater is directed into the downcomer by a “J-nozzle” feed ring sparger and flows downward mixed with the recirculation water returned by the coarse and fine separators. The steam water mixture leaving the bundle region flows upward into the coarse separator where the moisture is partially removed by centrifugal separation and returned to the downcomer. Additional separation is attained in the fine, box-grid type separator from where saturated, practically dry steam flows into the outlet nozzle. On the primary side, fluid enters the U-tubes through an inlet chamber and flows first upward and then downward exchanging heat with the secondary fluid.

3.1.3. Main Coolant Pumps. The main coolant pumps of both loops are centrifugal type pumps having a specific speed of 29.2 (DIN). The two pumps are equal in size and are, therefore, operated at two different speeds such as to yield the two different steady-state mass flows of 21 Kg/s and 7 Kg/s for the intact and broken loop at the same pressure head. A special control and drive system allows variation of the pump speed in the forward and backward directions over a range of ± 8500 rpm. Since the locked rotor resistance of the MCPs is less than the equivalent reactor pump, there are provisions for the insertion of an additional flow resistance at the outlet of each pump. To account for the injection of the main coolant pump seal water, a closed loop seal water compensation system is installed.

3.1.4. Pressurizer. The pressurizer design is geometrically similar to that of the reference plant; however, it is scaled in volume but not in height. The surge line rises within the pressurizer and leaves it radially. The pressurizer is provided with normal and additional heaters; the spray system is simulated with cooling coils placed in the steam region. There

FIGURE 4: Primary system pressure and heater rod temperature response to a large 200% cold leg break LOCA (downcomer gap width 12 mm).

are provisions for connecting the surge line to either the intact or broken loop hot legs. The simulation of power-operated relief valves (PORVs) as well as safety relief valves (SRVs) is provided in the pressurizer relief line.

3.1.5. Primary Loop Pipework. The main coolant pipes connecting the major primary loop components have inner diameters of 73.7 mm and 46.1 mm for the intact and broken loop, respectively. Measurement inserts are installed at the inlet and outlet of each major component forming integral part of the main pipework. Since the main coolant pumps are equal in size with an inlet-outlet diameter of 65 mm, special cross-sections adapters are installed at the inlet-outlet of each pump to fit the main loop pipes.

3.1.6. Break Assembly. The break assembly consists of a T-shaped insert with the break orifice housed in a recess machined in the insert. The break assembly can be connected to the main coolant pipe at the selected break location; for example, cold leg, hot leg, or pump seal. Pressurizer breaks or inadvertent opening of the valves are simulated by an orifice inserted in the relief line. A proper connection can be established between primary and secondary systems at the tube plate elevation to simulate steam generator tube rupture (SGTR) sequences.

3.2. Safety Injection Systems. The LOBI-MOD2 emergency-core-cooling system (ECCS) comprises the high-pressure injection system (HPIS) and the accumulator injection system (AIS). As required, the low-pressure injection system (LPIS) could also be simulated. Provisions are made for cold leg, hot leg, or combined cold and hot leg ECC injection in both primary loops. In the MOD1 version of the test facility, only the accumulator system was simulated.

The HPIS water is supplied by a positive displacement pump driven by a variable speed motor. The pump is rated

FIGURE 5: Heater rod temperature and vessel-collapsed liquid level in a small 10% cold leg LOCA.

for a maximum flow of 0.39 Kg/s at a total head of 200 bars. A special speed control system provides appropriate flow regulation to match the reactor HPIS-expected performance. Properly designed and calibrated throttling devices are installed in the main injection line to provide the required partitioning (depending on particular simulation needs) of the injection rate between the broken and the intact loop.

The AIS is composed of two accumulators, one in each loop. The accumulator of the intact loop has three times the volume and water capacity of that of the broken loop. The total volume of each accumulator is scaled to that of the reference plant having one accumulator for each loop; gas space, water volume as well as height and elevations are preserved. Both accumulators are rated for a maximum pressure of 60 bar and a temperature of 50°C.

Additional safety injection systems consist of the volume control system (VCS) and of the auxiliary feedwater system (AFWS). The VCS consists of a feed pump and a water preheating system which allows the control of the injected water at the prevailing cold leg fluid temperature. Similarly, the AFWS consists of a feed pump and a water preheating system as the injected water is preheated to generally about 130°C to prevent thermal shocks in the SG feed line and J-nozzle feed ring.

3.3. Measurement System. The LOBI test facility in both the MOD1 and MOD2 configurations has been fitted with a comprehensive measurement system. Relevant thermal-hydraulic quantities were measured at the boundaries of each major component and within the reactor pressure vessel model and the steam generators.

Measurement in the primary loop pipework is performed within the measurement inserts at the inlet and outlet of each major component, that is, the reactor pressure vessel model, the steam generators, and the main coolant pumps. The inserts in the horizontal pipework are properly

instrumented in the lower and upper part of the flow cross-section to characterize eventual flow and thermal stratification phenomena. Fluid and wall temperatures, absolute and differential pressures, fluid velocities and density as well as flow direction indicators are generally provided at each measurement insert.

Measurement of temperatures, pressures, and differential pressures are extensively made along the downcomer, lower plenum, rod bundle section, and upper plenum flow paths. Fluid velocity and fluid density are measured at the rod bundle inlet box.

Each rod bundle is supplied with three thermocouples brazed into grooves of 0.8 mm depth and 10 mm length machined into the outer surface of the heater rod tubes and then led through the wall to the inside of the tubes; they leave the rods through the open upper end.

The LOBI-MOD2 steam generators are instrumented to provide a maximum of information on both the magnitude and location of the heat transfer process taking place between the primary and the secondary systems. In particular, the instrumentation is concentrated in the region of the lowest U-bend and immediately above the tube plate in order to detect changes in heat transfer regime.

Steam generators measurements include fluid temperatures, U-tube wall temperatures, and differential pressures on both the primary and secondary side. Primary side instrumentation is applied to two representative U-tubes in each steam generator; the highest U-tube is fitted with temperature sensors whereas differential pressures are measured in the shortest U-tube. Special Pitot tubes are installed in the downcomer of the steam generator secondary side to measure fluid velocities at three peripheral positions.

Measurements in the secondary system is concentrated in the feedwater and in the steam lines at the inlet and outlet of each steam generator. Feedwater line measurement includes fluid temperature and velocity whereas in the steam line fluid temperature and volumetric flow is measured. A special spool piece with low-density measurement capability may be mounted at the outlet of the broken loop steam generator for detecting eventual carry-over in those tests involving blowdown of the secondary side. The pressurizer and the surpline are provided with fluid temperature and differential pressure measurements and a full-flow turbine.

All ECC injection lines are provided with full-flow turbines and fluid temperature measurements. Mass flow measurement is also provided in the main coolant pump seal water injection lines. The break flow measurement system consists of a condenser-catch tank system. Initially, the high energy flow is condensed in the cold liquid pool contained in the catch tank; thereafter, long-term low-flow energy removal is ensured by a small 250 KW condenser. The break flow in its integral form is obtained by the incremental mass of the catch tank.

A special tailored data acquisition system based on state of art information technologies available at that time is used to record all measurement signals; the sampling rate can be varied up to 20000 samples/s as appropriate for the representation of the experimental information especially in fast evolving transients. Selected data are then read back from the

FIGURE 6: Display of collapsed liquid levels, heater rod temperature, and system pressures at 477 s into a 6% SBLOCA.

FIGURE 7: Primary system pressure and HPIS/PORV flow during a loss of feed water with bleed and feed.

FIGURE 8: Primary and secondary system pressures in a 0.4% steam generator tube rupture.

analog tape, converted into engineering units and stored on 300 Mbyte disk modules ready for normalization and correction.

The data acquired at very high sampling rate are discretised and reduced to about 1000 data points over the entire experiment time range; each reduced value is simply an average over the time interval between a discrete set of data points. This is required in order to have a manageable set of data for storage and further processing and analysis. The reduced data in its final form are stored on the corresponding normal-time experimental data tape. As appropriate, special selects are made for the analysis of experimental time intervals characterized by fast transients and stored on a related short-time data tape.

A real-time loop control and monitoring system provides defined set of time ordered operations to represent components response during the simulated transient. Control of main coolant pump speed, core heating power, high-pressure injection system flow rate and steam generator level is performed on a real-time basis in response to predefined trip signals.

4. Scaling Criteria

The LOBI test facility was scaled to preserve, insofar as possible or practical, similarity of thermal-hydraulic behaviour

The screenshot shows a web browser window titled "JOINT RESEARCH CENTRE - Microsoft Internet Explorer". The address bar shows "http://asa2.jrc.it/stresa/". The page is titled "LOBI Test Facility" and is part of the "JRC STRESA Database". It features a sidebar with navigation links like "STRESA Index", "Resources", "Facilities", "How to", "STRESA Statistics", "STRESA Documentation", "Links", and "Contacts". The main content area is divided into "Test Specific Documents" and "Facility general documents".

Test Specific Documents		
Test A1-06: 2A Break, Pump - Vessel, DC 12 mm		
EDR	L. Regel*, E. Ohlmer - "Experimental Data Report on Lobi Test A1-06" - 3952	A1-06-EDR.pdf
QLR	L. Piples, N. Granner* - "Quick Look Report on Lobi Test A1-06" - 3955	A1-06-QLR.pdf
DAT	Lobi digital data - "Base time range -7.0 112.0 s" -	ex06.dat

Facility general documents		
GEN	A. Annunziato - "Evaluation of Thermohydraulic Parameters for the Analysis of the Flow in Steam Generator U-Tubes" - 1.90.51	Annu1.GEN.pdf
GEN	A. Annunziato - "LOBI-MOD2 Steam Generator Analysis during Steady State" - 1.89.75	Annu2.GEN.pdf
GEN	A. Annunziato - "LOBI-MOD2 Accumulator System Behaviour during Small Break LOCA Experiments" - 1.89.51	Annu4.GEN.pdf
GEN	W. Kolar, L. Piples, H. Staedtke - "OECD-CSNI International Standard Problem No. 18 Comparison of the Prediction Calculations with the Experimental Results Volume 1: Participants AECL - LTB" - 1.06.C1.85.136	ISP18-Com1.GEN.pdf
GEN	W. Kolar, L. Piples, H. Staedtke - "OECD-CSNI International Standard Problem No. 18 Comparison of the Prediction Calculations with the Experimental Results Volume 2: Participants OECD - LTB" - 1.06.C1.85.136	ISP18-Com2.GEN.pdf

At the bottom left, there is a section "Accesses to STRESA" with a value of "401".

FIGURE 9: Typical STRESA data set for each LOBI test.

with respect to the reference plant; the German 1300 MWe BIBLIS B power station of Siemens/KWU design commissioned in 1976. It is worth noting that the first commercial nuclear power plant in Germany, the 17 MWe VAK KAHL station of the BWR type, was commissioned in 1960 and that the first unit of the PWR type, the 345 MWe KWO Obri- gheim station, was commissioned in 1968.

As general scaling principle, a power-to-volume scaling criterion was adopted in the design of the facility to ensure the preservation of the specific power input into the primary fluid, Table 1. To meet general scaling requirements, the test facility was designed to preserve, taking into account the selected 1 : 700 power to volume scaling ratio, the following main parameters:

- (i) core power to system volume ratio,
- (ii) volumes and relative volumes of individual components,
- (iii) rupture size to primary system volume ratio,
- (iv) pressure drop and temperature distribution along main flow paths,
- (v) height and elevation of major components, and
- (vi) core and steam generators heat transfer surfaces.

The elevation of the major components were maintained at full height with the exception of the pressurizer which, while preserving the total volume and the steam to liquid volume ratios, was somehow shortened to allow increased radial dimensions to accommodate the internal heaters. The core and

steam generators heat transfer and flow areas were matched to the scale factor. Strict adherence to the power-to-volume scale factor would have resulted in unacceptably high wall frictional pressure losses in the primary loop pipework which was appropriately shortened to increase the pipe diameter in order to match the expected pressure drop in the reference plant.

In the MOD2 configuration of the test facility, special emphasis was given to the scaling of the steam generator primary and secondary sides due to their importance on the thermal-hydraulic evolution of small break LOCAs and special transients. In particular, volume ratio, heat transfer surface-to-volume ratio, hydraulic resistances and elevations, and height of the shortest U-tube were preserved.

A major exception to the general scaling concept is the design of the reactor pressure vessel model annular downcomer. The test facility has been configured with a downcomer of two different gap widths. Initially, a downcomer gap of 50 mm was installed in the MOD1 configuration to prevent ECC bypass which is largely influenced by hot wall delay and counter-current flow limitation phenomena; this, however, resulted in a 6.3 times too large a downcomer volume and, as a consequence, in an atypical thermal-hydraulic system response during large break LOCAs. The downcomer gap width was later changed to 12 mm which again, was a technical compromise between a 7 mm volume-scaled and a 25 mm pressure drop-scaled downcomer.

The LOBI test facility, as any other scaled test facility, has inherent distortions with respect to the reference plant which may impair the typicality of some results. The power-to-volume scaling concept results in a design which exhibits

TABLE 1: LOBI test facility design and operational parameters.

Reference plant	Siemens-KWU	Biblis B
	Loops	4
	Power	1300 MWe
Primary system	Loops	2 (1:3)
	Total volume	0.6 m ³
	Scaling factor	1:700
Core	Power	5.28 MWe
	Length	3.9 m
	Number of rods	64
	Matrix	8 × 8 square
	Heater rod OD	10.75 mm
	Pitch	14.3 mm
	Electrical heating	Direct
Vessel downcomer	Configuration	Annular
	Gap width	MOD1: 50 and 12 mm
		MOD2: 12 mm
Steam generators	Type	U-Tubes
	Number of tubes	1-Loop SG: 8
		3-Loop SG: 24
Main coolant pumps	Type	Centrifugal
	Specific speed (DIN)	29.2
Operational conditions	Primary system pressure	15.8 MPa
	Core temperatures	294/326°C inlet/outlet
	Secondary system pressure	6.4 MPa
	SG temperatures	210/280°C inlet/outlet
Organisation	EC joint research centre	Ispira, Italy

a basically one-dimensional thermal-hydraulic response, components high surface area to fluid volume ratio, and large metal mass to fluid volume ratio. The structural stored energy and system heat losses are important contributors to distortions in those components, such as the reactor pressure vessel and steam generator downcomers, where the coupling between wall heat transfer and fluid flow is at time dominant.

System heat losses may significantly influence primary as well as secondary side energy removal especially during the long-term phase of a small break LOCA or intact circuit fault simulations. The LOBI test facility exhibits larger heat losses relatively to the reference plant due to design (higher surface area to volume ratio of fluid retaining components) and operation constraints (main coolant pump seal and instrument cooling); typically, heat losses in a full size plant account for about 0.05% of the nominal thermal power whereas in the LOBI test facility it accounts for about 1.5%.

All in all, the experimental results acquired in the LOBI test facility cannot be directly extrapolated to full-size plants; they provide, however, a reference data base for the understanding of governing thermal-hydraulic phenomenologies

and for the assessment of the predictive capabilities of system codes used in water reactor safety analysis.

5. Project Evolution

The LOBI project has evolved over a time period which has seen a very intensive international effort in reactor safety research and development. In the early '70s, the level of understanding of LOCA phenomenologies and the attendant computational capabilities were rather primitive. A very limited experimental data base from integral system tests conducted in an initial, very crude configuration of the USNRC semiscale test facility and early one-dimensional versions of system codes such as RELAP were available.

Since then, a very large experimental data base has been gathered in several integral system test facilities, and the computational capabilities have matured encompassing a number of best-estimate codes such as RELAP, TRAC, ATHLET, and CATHARE. In the meantime, two major reactor accidents occurred in TMI-2 (March 1979) and Chernobyl (April 1986) power plants which have had a significant impact on the deployment of nuclear reactor technologies.

5.1. Execution of the MOD1 Programme. The LOBI test facility became fully operational in December 1979 with the execution of the first 200% cold leg break LOCA experiment which was used for an international pretest prediction exercise (PREX). Early in the programme and in response to the TMI-2 accident which occurred in March 1979, new research priorities were formulated to emphasize small break LOCA and special transients tests in the follow-up programme.

After the execution of an initial test series and due to experimental evidences on the atypical influence of the large downcomer (and hence large vessel water inventory) on the system thermal-hydraulic response, the original 50 mm gap width downcomer was replaced with a downcomer having a smaller 12 mm gap width (and hence better scaled vessel water inventory), Figures 3 and 4. In the meantime, while the small downcomer was being procured, an interim test programme was carried out to assess test reproducibility, break geometry, and size effects on the course of a large break LOCA.

The experimental programme with the small 12 mm downcomer was initiated in March 1981 with test A1-66, a 200% cold leg break LOCA with cold leg ECC injection. The LOBI-MOD1 experimental programme was then concluded in June 1982. From December 1979 to June 1982, 28 experiments were carried out including 25 large break LOCA and 3 small break LOCA tests.

5.2. Execution of the MOD2 Programme. After extensive modifications to the test facility, the experimental programme was resumed in April 1984 with the facility in the MOD2 configuration. The first small break LOCA test, a 1% cold leg break LOCA which was used for the OECD-CSNI International Standard Problem 18, was performed in

TABLE 2: LOBI-MOD1 and LOBI-MOD2 test matrix.

Test	Date	Definition	Objectives	Phenomenologies
LOBI-MOD1: Downcomer width 50 mm				
A1-04	12.12.79	Germany	200% CL break LOCA ECC-AIS in CL, first test in power ascension series, 1.8 FPS, PREX test	DNB, early rewet, dryout, final rewet, system performance at low power
A1-01	29.01.80	Germany	200% CL break LOCA ECC-AIS in CL+ HL, second test in power ascension series, 3.0 FPS	DNB, early rewet, dryout, final rewet, oscillatory refill behaviour, DEGB partially communicative
A1-02	14.02.80	Germany	200% CL break LOCA ECC-AIS in CL + HL, third test in power ascension series, 8.2 FPS	DNB, early rewet, dryout, final rewet, oscillatory refill behaviour
A1-03	19.03.80	Germany	200% CL break LOCA ECC-AIS in CL + HL, fourth test in power ascension series, 8.8 FPS, power off from 3.2 s to 5 s	DNB, early rewet, dryout, final rewet, oscillatory refill behaviour
A1-04R	17.04.80	Germany	200% CL break LOCA ECC-AIS in CL, Counterpart to A1-04 at nominal power 10.2 FPS, baseline CL LOCA	DNB, early rewet, dryout no final rewet before power shutoff, no sustained oscillation during refill
A1-05	06.05.80	Germany	200% CL break LOCA ECC-AIS in CL + HL Counterpart to A1-04R, namely, AIS mode Enhanced AIS injection, 10.2 FPS	DNB, early rewet, dryout No effective improvement of AIS performance, oscillatory refill behaviour
SD-SL-01	04.06.80	Germany	10% CL break LOCA ECC-AIS in CL, first test of the small break LOCA scoping series, instrumentation response to slow transients Link between small and large break LOCAs	No degradation of core heat transfer
SD-SL-02	04.06.80	Germany	1% CL break LOCA ECC-AIS in CL, second test of the small break LOCA scoping series Secondary 100 K/h cooldown	Flow to break mainly from vessel side, flow from pump side impeded due to closure of break valve, no degradation of core heat transfer
SD-SL-03	24.09.80	Germany	0.4% CL break LOCA, ECC-AIS in CL, third test of the small break LOCA scoping series Secondary 100 K/h cooldown,	No degradation of core heat transfer, natural circulation and reflux condenser heat transport detected
A2-59	27.10.80	Germany	100% CL break LOCA ECC-AIS in CL, first test of interim matrix, system response to communicative break	DNB, early rewet, dryout, final rewet, clear bottom-up rewet trend, response similar to DEGB
B-101	26.11.80	France	2 × 50% CL break LOCA ECC-AIS in CL, second test of the interim matrix, influence of noncommunicative break configuration	DNB, early rewet Comparison of B-101 with A2-59 precluded by difference in power load
A2-55	19.01.81	Germany	50% CL break LOCA ECC-AIS in CL, third test of the interim matrix, system response to intermediate large break sizes	DNB and early rewet only at core upper levels, thereafter, effective core cooling prevailed over the whole core
A2-59R	11.02.81	Germany	100% CL break LOCA ECC-AIS in CL, fourth test of interim matrix, counterpart to A2-59 with respect to reproducibility	A2-59R and A2-59 system thermal-hydraulic response similar, MOD1 reproducibility confirmed

TABLE 2: Continued.

Test	Date	Definition	Objectives	Phenomenologies
B-R1M	17.03.81	Germany	25% CL break LOCA ECC-AIS in CL, fifth test of the interim matrix, system response to small large break sizes	Core thermal response followed prevailing system saturation temperature, no core thermal excursions tendency to loop seal formation
LOBI MOD1: Downcomer width 12 mm				
A1-66	03.07.81	Germany	200% CL break LOCA ECC-AIS in CL, baseline test, counterpart to A1-04R with respect to downcomer width	DNB, early rewet at core bottom and top dryout, no conclusive rewet observed, CCFL and hot wall delay effects in downcomer
A1-07	09.07.81	Germany	200% CL break LOCA ECC-AIS disabled System response with no ECCS Baseline test for ECCS injection mode	DNB, early rewet at core bottom and top ends, dryout, rod temperatures high after power shutoff
A1-06	21.07.81	Germany	00% CL break LOCA ECC-AIS in CL + HL System response to combined ECCS injection, Baseline test with respect to ECC injection mode	DNB, early rewet at core bottom and top ends, dryout, heater rod temperature turnaround after ECC injection starts, no clear definite rewet
A1-67	30.09.81	Germany	25% CL break LOCA ECC-AIS in CL + HL System response to intermediate break Break size test series	Delayed dryout at core mid and upper elevations, rewet, response tendency towards small break LOCA features, loop seal formation and clearout
A1-68	28.10.81	Germany	50% CL break LOCA ECC-AIS in CL + HL System response to intermediate break Break size test series	DNB, rewet, clear top-down rewet behaviour
A1-10A	25.11.81	Germany	200% HL break LOCA ECC-AIS in HL + CL System response to hot leg break Break location test series, Low core power	DNB, rewet only at core, mid, and upper elevations, hot leg ECC core penetration hindered by sustained positive core flow, CCFL at core exit
A1-10B	10.12.81	Germany	200% HL break LOCA ECC-AIS in HL + CL System response to hot leg break Break location test series Nominal core power	DNB, rewet limited at core upper elevations, higher peak cladding temperatures with respect to A1-10A, CCFL at core exit
A1-70	13.01.82	Germany	200% PS break LOCA ECC-AIS in CL + HL System response to pump suction break, Break location test series	DNB, rewet, less severe overall core heat transfer degradation with respect to similar cold and hot leg break DEGB-LOCA
A1-73	04.02.82	Germany	25% HL break LOCA AIS in CL + HL System response to hot leg small large break LOCA Break size and location test series	No core heat transfer degradation, heater rod temperatures at the prevailing system saturation temperature
A1-72	24.03.82	Germany	200% CL break LOCA ECC-AIS in CL + HL influence of pump operation mode, pump countdown delayed off, pump head simulation	DNB, early rewet, dryout, enhancement of initial recovery of positive core flow, lower peak cladding temperatures with respect to A1-06
A1-69	06.04.82	Germany	100% CL break LOCA ECC-AIS in CL + HL System response to intermediate break LOCA, break size effect test series	DNB, early rewet, dryout, final rewet, typical DEGB LOCA blowdown features, preferentially top-down rewet observed

TABLE 2: Continued.

Test	Date	Definition	Objectives	Phenomenologies
A1-74	21.04.82	Germany	200% CL break LOCA ECC-AIS in CL + HL System response to ECCS injection in both intact and broken loop, counterpart to A1-72	DNB, early rewet, dryout, no discernible impact on overall system response from the addition of ECC injection into broken loop
B-222	05.05.82	France	100% CL break LOCA ECC-AIS in CL Noncommunicative ($2 \times 50\%$) CL break configuration, counterpart to B-101 with respect to downcomer size	DNB, early rewet, dryout, limited effectiveness of cold leg ECCS injection
B-302	16.0.82	Italy	100% HL break LOCA ECC-AIS in CL Noncommunicative ($2 \times 50\%$) HL break configuration	Dryout at core mid and upper elevations, rewet, positive core flow throughout the whole transient, enhanced refill and effective core cooling
LOBI-MOD2: Downcomer width 12 mm, shell and U-tube steam generators				
A1-76	12.04.84	Germany	SG performance under primary forced circulation, variation of secondary inventory, and core power: (i) flooding of SG coarse separator at nominal core power (ii) flooding of SG coarse separator at 50% core power and reduced secondary water level (iii) boiloff of SG SS at 10% core power	Data on coarse and fine separator efficiency, variation of recirculation ratios, void distribution in the SG riser region, and degradation of SG heat transfer
A2-81	27.09.84	Germany	1% CL break LOCA ECC- HPIS in CL, AIS off Secondary cooldown at 100 K/h First test of the small break LOCA test series, OECD International Standard problem 18 (ISP 18)	Coupling of primary and secondary systems depressurisation, 2 phase NC and reflux condenser heat transport, flow separation and stratification in horizontal pipes liquid holdup in hot legs and SG U-tubes
A1-82	28.09.84	Germany	1% CL break LOCA ECC-HPIS in HL, AIS in HL + CL Secondary cooldown at 100 K/h Counterpart to A2-81 relatively to HPIS location	Coupling of primary and secondary systems, 2 phase NC and reflux condenser heat transport, low subcooling in pressure vessel downcomer, reduced ECC bypass to the break
A1-78	24.10.84	Germany	2% CL break LOCA ECC-HPIS in HL, AIS in HL + CL Secondary cooldown at 100 K/h Test of the break size effect test series	Decoupling of primary and secondary systems, reverse SG heat transfer, voiding in SG U-tubes and liquid holdup in hot legs, loop seal formation, and core liquid level depression
A2-77A	28.11.84	Germany	Characterization of NC and reflux condenser heat transport mechanisms at a primary system pressure of 90 bar and 70 bar, (i) 90 bar: 1 and 2 phase NC and reflux condenser (ii) 70 bar: 2 phase NC and reflux condenser	NC heat transport mechanisms characterized as function of primary system mass inventory, minimum mass inventory of c. 45% at c. 3% of core power to sustain effective reflux heat transport and prevent core heat transfer degradation, oscillatory transition from 2 phase NC to reflux
A1-83	19.12.84	Germany	10% CL break LOCA ECC-HPIS in HL, AIS in HL + CL Secondary cooldown at 100 K/h, larger of the small break size effect test series	Decoupling of primary and secondary systems, early voiding of SG U-tubes and hot legs, initial core dryout and rewet coupled to loop seal formation and clearout, second core dryout, and rewet coupled to mass inventory boiloff and AIS injection

TABLE 2: Continued.

Test	Date	Definition	Objectives	Phenomenologies
A2-90	27.03.85	Germany	LONOP-ATWS or "SBO" Anticipated transient caused by loss of offsite and normal onsite electrical power with failure to SCRAM (i) boiloff of SG secondary system down to a level of c 1 m above tube plate, (ii) SG refill and cooldown at 100 K/h	Pressure increase in primary and secondary systems, fluid discharge from pressurizer PORV and SG SRV, pressurizer insurge/outsurge, SG heat transfer degradation, recovery of primary to secondary heat transfer, 1 and 2 phase NC
A1-85	07.05.85	Germany	0.4% pressuriser break LOCA ECC-HPIS in HL, AIS in HL + CL Secondary cooldown at 100 K/h, test of the break location effect series	Coupling of primary and secondary systems depressurisation, pressurizer insurge, primary system overfeeding
BL-00	03.07.85	France	0.4% CL break LOCA ECC- HPIS in CL Secondary cooldown at 57 K/h First test of the EC B test matrix	Primary and secondary systems thermally coupled, liquid holdup in SG U-tubes, stratification in horizontal pipework, Thermal nonequilibrium downstream ECC injection points, 2 phase and reflux condenser heat transport, primary overfeeding and refill, no core dryout
A1-84	14.10.85	Germany	10% HL break LOCA ECC-HPIS in HL, AIS in CL + HL Secondary cooldown at 100 K/h Test of the break location effect test series, counterpart to A1-83	Decoupling of primary and secondary systems depressurisation, early voiding of SG U-tubes and hot legs, holdup and CCFL at core exit, ECC penetration and flow channelling
BT-00	30.11.85	UK	LOFW with primary feed and bleed (i) loss of main feedwater and SG boildown to c. 1 m (ii) loss of auxiliary feedwater and SG dryout (iii) long-term cooldown via primary feed and bleed	SG boil-off and heat transfer degradation, PORV and SRV fluid discharge, pressurizer insurge/outsurge, PORV flow compensation via HPIS flow, primary system refill, verification of <i>Feed and Bleed</i> as an accident management procedure
BT-01	24.01.86	Belgium	10% SLB with pressurised thermal shock (PTS) and plant recovery procedure (i) small steam line break transient (ii) establishment of PTS conditions (iii) accident mitigation and recovery	SG secondary blowdown and heat transfer, primary system cooldown rate, pressurizer insurge/outsurge, downcomer temperature stratification, primary depressurization via pressuriser PORV cooling system and mass inventory control via HPIS injection
BL-02	22.03.86	UK	3% CL break LOCA ECC-HPIS in CL, AIS in CL SCS cooldown at 56 K/h Test of the break size effect test series	Primary and secondary systems decoupled, SG heat transfer reversed, formation and clearout of loop seal, no core dryout
A1-79	15.05.86	Germany	1% CL break LOCA HPIS in HL, AIS off Secondary cooldown at 100 K/h Effect of high (4/4) HPIS injection rate	Coupling of primary and secondary systems cooldown, primary system overfeeding, NC heat transport hindered by condensation in hot legs and upper plenum induced by high HPIS rate
A1-88	11.06.86	Germany	0.4% CL break LOCA ECC-HPIS in HL, AIS off SCS cooldown at 100 K/h in IL-SG Asymmetric cooldown of secondary system	Primary system pressure coupled to isolated SG, pressurizing effect of isolated SG, primary system depressurization, break flow compensated by HPIS flow

TABLE 2: Continued.

Test	Date	Definition	Objectives	Phenomenologies
BL-01	20.09.86	Germany	5% CL break LOCA ECC-HPIS in HL, AIS on secondary cooldown at 100 K/h Test of the break size effect test series	decoupling of primary and secondary systems, SG reverse heat transfer, clearout of intact loop seal, liquid holdup in HL
BC-01	18.10.86	JRC	SG secondary mass inventory determination LOBI-MOD2 characterization test	SG mass inventory determined at various power levels, relationship of SG mass versus downcomer water level determined
BC-02	26.11.86	JRC	SG heat losses determination LOBI-MOD2 characterization test	SG heat losses determined via (i) steady-state method balancing core power (ii) cool-down method SG heat losses unacceptably high: ILSG—24 kW, BLSG—18 kW, need improvement of thermal insulation
BL-21	24.01.87	Italy	steam generator tube rupture (SGTR) break Size 0.4%, ascending side Intentional PCS depressurization through PORV and AIS actuation as recovery procedure	Break and PCS depressurization, natural circulation and reflux condenser heat transport, core uncovery and dryout, PORV discharge, AIS actuation and core rewet
BL-12	19.02.87	France	1% CL break LOCA ECC-HPIS off, AIS in CL SCS cooldown off, system response with degraded safety systems	core uncovery and dryout at high PCS pressure, phase separation and stratification, thermal nonequilibrium downstream AIS location, loop seal formation and clearout, core rewet
BT-02	09.05.87	France	LOFW + LOAF (loss of main and auxiliary feedwater) PCS bleed and feed as recovery procedure	SG boiloff and heat transfer degradation, PCS heatup and pressurization, pressurizer insurge/outsurge, PORV discharge and HPIS compensation, recovery of PCS inventory
BT-12	17.06.87	UK	steam line break (SLB) Break size 100% (orifice limited) SCS break size effect and location test series	Faulted SG depressurization, break flow and steam line carryover, faulted SG heat extraction, reverse heat transfer in unaffected SG, pressurizer behaviour, PCS overcooling and thermal stratification
A1-91	26.09.87	Germany	1% CL break LOCA ECC-HPIS in HL, AIS off Secondary cooldown at 100 K/h Effect of low (1/4) HPIS injection rate	PCS and SCS thermal coupling, core thermal response with reduced HPIS capacity, core liquid level depression, loop seal formation, no core dryout at reduced HPIS capacity
BT-03	24.10.87	Italy	loss of feed water-anticipated transient without SCRAM (LOFW-ATWS) PCS passive recovery procedure	PCS heat up and pressurization, PORV and SRV discharge, voiding and refill of SG, pressurizer insurge/outsurge, intentional PCS depressurization and AIS actuation, core dryout
A1-92	30.11.87	Germany	Characterization of NC and reflux condenser heat transport at a primary pressure of 40 bar, 1 and 2 phase NC and reflux condenser, Counterpart to PKL test AC.1	NC and reflux condenser heat transport mechanisms as function of primary mass inventory, minimum mass inventory to sustain reflux condenser and prevent core heat transfer degradation c. 45%, rather stable transition from 2 phase NC to reflux condenser

TABLE 2: Continued.

Test	Date	Definition	Objectives	Phenomenologies
BL-16	19.03.88	Germany	0.4% small break LOCA ECC- HPIS in HL, AIS off SCS cooldown in BLSG at 100 K/h, MCPs off and restart	Asymmetric SCS cooldown, pressurizing effect of isolated SG, thermal homogenisation and fluid redistribution following MCPs restart
BC-03	15.04.88	JRC	SG heat losses determination LOBI-MOD2 characterization test	Measurement of SG heat losses after improvement of thermal insulation (ref.: BC-02) ILSG: 6.8 kW, BLSG: 5.0 kW
A1-93	30.04.88	Germany	2% CL break LOCA, HPIS off, AIS on secondary system cooldown at 100 K/h, accident management procedure, pressurizer PORV on high core heater rod temperature	Decoupling of primary and secondary systems, loop seal formation and core level depression, core dryout and AIS injection, enhancement of primary depressurization and AIS actuation through intentional opening of pressurizer PORVs
A1-94	27.05.88	Germany	4% CL break LOCA at 40 bar, HPIS off, AIS on, OBI counterpart to PKL test, secondary system cooldown at 100 K/h, AIS on at high core heater rod temperature	Core uncover and dryout, AIS injection and core rewetting, verification of PKL-III pressure scaling concept
BC-04	07.02.89	JRC	Core bypass flow measurement LOBI-MOD2 characterization test	Determination of upper plenum to upper downcomer flow bypass: c. 3% of core flow
BL-30	15.04.89	JRC	5% CL break LOCA, HPIS in CL, AIS in CL, SCS cooldown at 100 K/h, test of the break size effect test series	Primary depressurization at a moderate rate, primary and secondary systems thermally decoupled, loop seal formation and clearout
BL-22	17.06.89	Belgium	SGTR: Steam generator tube rupture (0.4%) accident initiation and mitigation phases	Break flow, overfilling of affected SG and SRV discharge, autostabilizing mechanism for break flow when affected SG level reaches U bend elevation, verification of emergency operating procedures
A1-87	11.11.89	Germany	Cooldown transient, LOBI counterpart to PKL-III test	1 phase NC under saturated conditions, upper head steam bubble formation and propagation, SG heat transfer
BT-04	0.02.90	France	Cooldown transient under asymmetric conditions, BLSG isolated, ILSG cooldown at 56 K/h	Reverse heat transfer in isolated SG, pressurizing effect of isolated SG, flow reversal in secondary of isolated SG
BL-34	22.03.90	JRC	6% CL break LOCA, HPIS off, AIS on at 40 bar, initial conditions scaled to low power (10%), SCS cooldown disabled, counter part test to BETHSY, LSTF and SPES	Sequence of 3 core thermal excursions (1) dryout and rewet due to loop seal formation and clearout, (2) dryout due to boiloff and rewet due to AIS, (3) dryout due to depletion of AIS injection and rewet due to LPIS injection
BL-44	26.04.90	JRC	6% CL Break LOCA, HPIS off, AIS on at 40 bar, initial conditions scaled to full power (100%), SCS cooldown disabled, Counter Part Test to BL-34 (full power-low power)	Phenomenologist similar to BL-34, sequence of 3 core dryout/rewet phases, first dryout less extensive due to less pronounced core liquid level depression
BT-56	03.07.90	UK	Multiple failure transient evolving from an original LOFW: isolation of ILSG, MCP power off, SCRAM failure, PCS blowdown through upper plenum due to rupture of the safety disk	PCS heat up and pressurization, pressurizer surge/outsurge, primary and secondary systems decoupled, dryout at high pressure, blowdown of the PCS through a relatively large break in upper plenum represented by the rupture disk opening

TABLE 2: Continued.

Test	Date	Definition	Objectives	Phenomenologies
BT-15/16	22.11.90	UK	LOFW: loss of feed water with MCP on (BT-15), SG boiloff and refill with MCPs off (BT-16)	BT-15: SG boiloff and heat transfer degradation with MCPs on, primary system heat up, reestablishment of SG heat transfer following AFW actuation, BT-16: SG boiloff and heat transfer degradation with MCPs off, natural circulation in PCS, SG refill
BT-17	07.02.91	Germany	LOFW; loss of feed water and secondary feed and bleed recovery procedure Intentional PCS depressurization from upper plenum	SG boiloff, PCS heat up and pressurization, pressurizer insurge and PORV discharge, SCS blowdown and PCS depressurization due to condensation in SG U-tubes, PCS depressurization from upper plenum, pressurizer outsurge, core dryout and partial rewet
BT-06	21.03.91	France	FLB: feed line break (10%) MCPs on, AFW on in ILSG MCPs off and asymmetric natural circulation in PCS with a voided SG	Blowdown and heat transfer from faulted SG, break flow fed from different flow paths, residual heat removal by unaffected SG, pressurizer insurge/outsurge
BL-40	16.05.91	Spain	SGTR: steam generator tube rupture in 1-loop PWR (Jose' Cabrera NPP), E-3 emergency recovery procedures	Break flow, PCS natural circulation, PCS depressurization and control of subcooling margin and pressurizer level cycling PORV and HPIS flow
BL-06	21.06.91	UK/France	1% CL break LOCA, HPIS off, AIS on at 40 bar, SCS cooldown as in BL-12, MCPs on, effect of MCP on/off issue	PCS depressurization with MCPs on, core dryout and rewet, PCS pressure stagnation, termination of AIS injection, depressurization from PORV and actuation of LPIS

ATWS: anticipated transient without scram; AIS: accumulator injection system; BL: broken loop; CL: cold leg; CCFL: counter current flow limitation; DEGB: double ended guillotine break; DNB: departure from nucleate boiling; ECC: emergency core cooling system; FLB: feed line break; FPS: full power seconds; HL: hot leg; HPIS: high pressure injection system; IL: intact loop; LPIS: low pressure injection system; LOCA: loss of coolant accident; LOFW: loss of feed water; LONOP: loss of offsite and normal onsite power; MCP: main coolant pump; NC: natural circulation; PCS: primary cooling system; PORV: power operated relief valve; SCS: secondary cooling system; SRV: safety relief valve; SGTR: steam generator tube rupture; SLB: steam line break; SBO: station break out.

September 1984 and the first special transient test case (A2-90) simulating a "Station Blackout" transient was performed in March 1985. The first MOD2 test of the Community programme, a 0.4% cold leg break LOCA specified by the French representatives in the LOCA Programme Task Force was executed in July 1985.

As the LOBI-MOD2 experimental programme was evolving, the catastrophic Chernobyl accident took place in April 1986; this, however, due to the peculiarities of the RBMK reactor type accident had no significant impact on the LOBI-established research priorities.

The last experiment of the BMFT contractual programme was executed in November 1989 with the termination of the CEC-BMFT contract in December 1989. Thereafter, the experimental programme was exclusively dedicated to the execution of tests from the Community matrix. With the execution in June 1991 of test BL-06, a 1% cold leg break LOCA designed to address the pump on-off issue, the execution of the envisaged LOBI experimental programme was successfully concluded. Significant signatures of small break LOCA tests and special transients are depicted in following Figures 5, 6, 7, and 8.

5.3. International Collaboration. The international context in which the LOBI research programme has been carried out has offered an opportunity for a close collaboration among delegates of national research laboratories. It has also provided an independent forum for the exchange of concerns and expertise among the participants contributing thus to the harmonization of national views on reactor safety-related matters.

5.3.1. Counter Part Test Programme. Large system codes used in reactor safety analysis are generally benchmarked against experimental data from scaled integral system or separate effect test facilities; comparison of code-predictive capabilities of simulated accidents or transients in full-size plants would be desirable, but this is, clearly, prohibitive for obvious economic and practical considerations. Controversy, thus, arises when the predictive capability of a system code is scaled up from a small size test facility to the full-size real plant.

It is, therefore, desirable although not strictly required to assess code predictive capabilities against a set of data obtained from different scale test facilities under similar initial and boundary conditions. This, to a certain extent, would

decouple the assessment process from physical assumptions emphasizing, instead, the relevance of the geometrical scaling parameters especially on the qualitative rather than quantitative evolution of the thermal-hydraulic phenomenologies.

Within this context, a number of tests of the MOD1 and MOD2 experimental programmes were defined and executed as counter part to similar tests performed in other test facilities such as Semiscale, PKL, BETHSY, LSTF, and SPES. The collaboration has also been extended to posttest analysis of the results.

5.3.2. International Standard Problems Prediction Exercises. The very first test of the LOBI-MOD1 experimental programme, test A1-04, was used for a special type of blind standard problem exercises, the LOBI preprediction exercise (PREX); 16 participants from various EC member states and the USA submitted calculations using a number of system codes.

The first small break LOCA test of the LOBI-MOD2 experimental programme, test A2-81, was designated by the Committee on the Safety of Nuclear Installations (CSNI) of the Organization for Economic Cooperation and Development (OECD) as International Standard problem 18 (ISP-18); 27 participants from European and North American organizations provided prediction calculations with 12 codes or code versions.

6. Management of the Experimental Data Base

Management of research data is an issue being debated at the national and international levels by industrial, institutional, professional, and academic research organisations; specifically, preservation and archive as well as access and retrieval of research data are seen as underlying principles of good scientific practices especially in the governance of publicly funded research programmes.

The European Science Foundation (ESF) has addressed the issue of data accumulation, handling and storage in its policy briefing on Good Scientific Practices In Research and Scholarship [1] recommending:

data are produced at all stages in experimental research and scholarship. Data sets are an important resource, which enable later verification of scientific interpretation and verification. They may also be the starting point for further studies. It is vital, therefore, that all primary and secondary data are stored in a secure and accessible form.

Archival and dissemination of research data has also been addressed in the Berlin Declaration [2] promoted by the Max-Planck Gesellschaft (MPG) together with representatives from several international research organisations who have recommended:

our mission of disseminating knowledge is only half complete if the information is not widely and readily available to society. New possibilities of knowledge dissemination not only through the

classical form but also and increasingly through the open access paradigm via the Internet have to be supported.

Since the inception of the LOBI project, it was anticipated that a proper data analysis and documentation management system would be of fundamental importance in order to ensure that over time the investment of public resources would be beneficial for the nuclear community at large. In view of the continuous advancement of computer hardware and software technologies that are making storage/retrieval techniques rapidly obsolete, particular attention has been placed on conversion of old format onto new media and machines enabling potential users to access and retrieve the data. Currently, the LOBI analytical and experimental data base can be accessed via Internet at <http://asa2.jrc.it/>. Each experimental data set comprises the digital data file, the Quick look Report (QLR), the Experimental Data Report (EDR) and as appropriate the Test Prediction Report (TPR). As appropriate AVI or MPEG files showing temporal evolution of collapsed liquid levels are also available. The overall data base is complemented by Test Facility Description Reports and components as built drawings (Figure 9).

7. Conclusions

The LOBI Project has represented an important contribution to reactors thermal-hydraulic safety research. A comprehensive data base relevant to the understanding of governing phenomenologies expected in PWR accident conditions and to the assessment of system codes used in water reactors safety analysis has been provided. As structured, the LOBI project has represented an effective approach to international collaboration in the field of reactor safety research and development. In addition, the EC-JRC context, in which the research programme has been carried out, has provided an independent forum for a systematic exchange of technical and scientific information among experts from EC member states and for sharing best practices in the field of water-cooled reactor safety analysis.

References

- [1] European Science Foundation, "Policy briefing on good scientific practice in research and scholarship," 2000.
- [2] Max-Planck-Gesellschaft, "Open access to knowledge in the sciences and humanities," 2003.