NUREG/IA–0183

International Agreement Report

Analysis of the LOBI Experiment Test BT–56 Using the RELAP5/MOD3.2 Code

Prepared by

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U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

June 2000

Prepared as part of
The Agreement on Research Participation and Technical Exchange
under the International Code Application and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

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ABSTRACT

This document has been drawn up by UNIÓN ELÉCTRICA FENOSA, S.A. (UNIÓN FENOSA), which participates in the Code Application and Maintenance Project (CAMP) as a member of UNIDAD ELÉCTRICA, S.A. (UNESA). The document presents a comparison between experimental data from the plant and data obtained by simulation with the RELAP5/MOD3.2 code of the experiment BT-56 which took place on 3rd July 1990 at the Joint Research Centre of the Commission of the European Communities, located in Ispra (Italy).

The experiment originally scheduled for the LOBI-MOD2 facility was a Loss of Main Feedwater (LOFW), but multiple failures occurred, meaning that the evolution of the experiment deviated it from the scheduled sequence.

The simulation was performed using the RELAP5/MOD3.2 code, on a Digital AlphaServer 2000 4/200 computer and the operating system DIGITAL-UNIX.

The results show an acceptable agreement between the phenomenology observed during the experiment and that predicted by the code.

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EXECUTIVE SUMMARY

This paper presents the results obtained from simulation with the RELAP5/MOD3.2 code of the experiment BT-56, which took place at the LOBI-MOD2 experimental facility on 3rd July 1990. The experiment originally foreseen was the loss of main feedwater (LOFW), but multiple failures occurred which meant that the experiment deviated from the scheduled sequence. The sequence of events that finally characterized the transient was as follows:

- Loss of feedwater and turbine trip.

- Failure to open of the relief valve for the steam generator in the intact loop.

- Shutdown of the main coolant pumps, with loss of forced circulation in the primary.

- Failure of scram (disconnection of electrical core) up to 24 s.

- Rupture of primary system rupture disk, located on the downcomer.

Test BT-56 presented a sequence of particularly relevant thermalhydraulic events, as regards understanding of the interactions between components and assessment of the safety of Pressurized Water Reactors. The most outstanding phenomena that took place were as follows:

- Heating and pressurization of the primary system.

- Pressure mismatch between the primary system and the pressurizer.

- Core dryout at high pressure.

- Sudden depressurization of the primary after the break of the rupture disk (LOCA).

- Heat sink inversion.

- Reverse flow in the hot legs and the core, up to primary depressurization.

- Thermal stratification in the loops.

The results of the experimental simulation using the RELAP5/MOD3.2 code reflect an acceptable reproduction of the phenomenology and plant behaviour.

NOMENCLATURE

AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
AIS	Accumulator Injection System
ANM	Annular-Mist
BL	Broken Loop
CL	Cold Leg
ECCS	Emergency Core Cooling System
HL	Hot Leg
HPIS	High Pressure Injection System
IL	Intact Loop
JRC	Joint Research Centre (of the European Communities)
LOCA	Loss of Coolant Accident
LOFW	Loss of Feedwater
LPIS	Low Pressure Injection System
MCP	Main Coolant Pump
PL	Primary Loop
PZR	Pressurizer
RPV	Reactor Pressure Vessel
SG	Steam Generator
SL	Secondary Loop
SLG	Slug

1. INTRODUCTION

This report has been drawn up by UNIÓN FENOSA within the framework of the "Code Assessment and Maintenance Project" (CAMP). The objective of the report is to analyse the BT-56 experiment performed using the RELAP5/MOD3.2 code. The experiment was carried out at the LOBI experimental facility and originally consisted of a Loss of Feedwater transient (LOFW) with turbine trip. The experiment underwent an unexpected sequence of malfunctions, presenting a phenomenology of special interest for the verification of the codes habitually used for PWR reactor safety assessment.

The LOBI project (LWR Off-normal Behaviour Investigations) consisted of a high pressure experimental reactor, designed and operated at the Commission of the European Community's Joint Research Centre at Ispra (Italy).

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The reactor incorporates the essential primary components and secondary cooling systems of a four-loop PWR reactor, its design being scaled to obtain the same response during normal and off-normal operation. The construction of the reactor was commissioned in December 1979, and it started up in June 1982 with the MOD1 configuration for the research of large-break LOCA phenomena, subsequently being modified to the MOD2 configuration for the simulation of small-break LOCA's and special transients.

The main objectives of the LOBI experimental reactor were as follows:

- Identification and/or verification of the basic thermalhydraulic phenomenology postulated for LOCA (large or small-break) accident conditions in a PWR reactor, along with recovery procedures and procedures management in the event of an accident.

- Acquisition of an experimental database for the development and/or implementation of analytical and validation models independent of the thermalhydraulic codes used for PWR safety assessment.

The BT-56 experiment was performed as part of the experimental series corresponding to program B, carried out under the auspices of the Program of the European Commission's Reactor Safety Research Commission, with independent contributions made by many of the industries and institutional bodies of the European Community member countries.

2. DESCRIPTION OF THE LOBI-MOD2 FACILITY

The LOBI experimental reactor is an integral high-pressure system representing 1300 MWe KWU design PWR reactor, at a scale of approximately 1:712. The primary has two loops, one representing the three intact loops and the other representing the broken loop of the reference reactor. Each primary loop has a main coolant pump (MCP) and a steam generator (SG). The core is made up of 64 electrically heated rods arranged in a square matrix of 8 x 8 inside the reactor pressure vessel (RPV). The rated power is 5.28 MW. The facility also includes other components, such as the lower plenum, the upper plenum, the annular downcomer and the reactor head external assembly. The pressurizer is connected to the hot leg of the broken or intact loop, depending on the experiment to be performed. In test BT-56, it was connected to the hot leg of the broken loop.

The primary coolant system, which is shown schematically in Figure 2.2, operates under normal conditions of approximately 158 bar at a temperature between 294 and 326°C.

In configuration MOD2, the Emergency Core Cooling System (ECCS) may be replaced with a High Pressure Injection System (HPIS), by Accumulator Injection System (AIS) and by the Low Pressure Injection System (LPIS). The HPIS and LPIS systems may be connected to the cold and hot legs or to a combination of both. These systems did not actuate at any time during the BT-56 transient.

The heat generated in the core (5.28 MW) is removed from the primary loops via the secondary circuit, which is made up of a condenser, a cooler, the main feedwater pump and the auxiliary feedwater system (AFW). The operating conditions of the secondary coolant circuit are approximately 210°C for feedwater temperature and a pressure of 64.5 bar. The secondary cooling circuit is designed to operate at a pressure and temperature of 100 bar and 310°C, respectively. Figure 2.3 shows a schematic drawing of the secondary circuit.

The LOBI-MOD2 facility is entirely scaled to maintain similarity with the reference plant under normal and off-normal operating conditions, as regards thermalhydraulic phenomenology. The scaling required a coefficient of 3:1 between the thermal and hydraulic parameters of the intact and broken loop steam generators.

The broken loop steam generator (BL-SG) has 8 U-shaped tubes (+1 spare), and steady-state heat transfer at nominal power is 1.32 MW, while the steam generator in the intact loop (IL-SG) has 24 tubes (+1 spare) and a heat transfer of 3.96 MW under steady-state conditions at rated power. Figure 2.4 shows a schematic drawing of the steam generator in the LOBI-MOD2 intact loop.

Each steam generator consists of a single-piece cylindrical vessel, with an annular downcomer separated from the riser by means of a cylindrical sleeve supported by the tubesheet. Each generator has two steam separators. The U-shaped tubes are connected in a way as similar as possible to that used in the reference plant. There is a connection between the secondary and the primary water inlet or outlet at the tubesheet, in order to allow the steam generator tube rupture (SGTR) accident to be simulated.

The core is made up of 64 electrical resistances simulating 64 fuel elements, with hollow tubes having an active heating length of 3.9 metres, a diameter of 10.75 mm and a pitch of 14.3 mm. The thickness of the tube walls varies in 5 steps, in the heated zone (Fig. 2.1), in order to produce a cosine distribution in axial heat flux.

The measurement system consists of a total 470 measuring channels. This allows all of the relevant thermalhydraulic quantities to be measured in each of the components of the loops, vessel and steam generators.

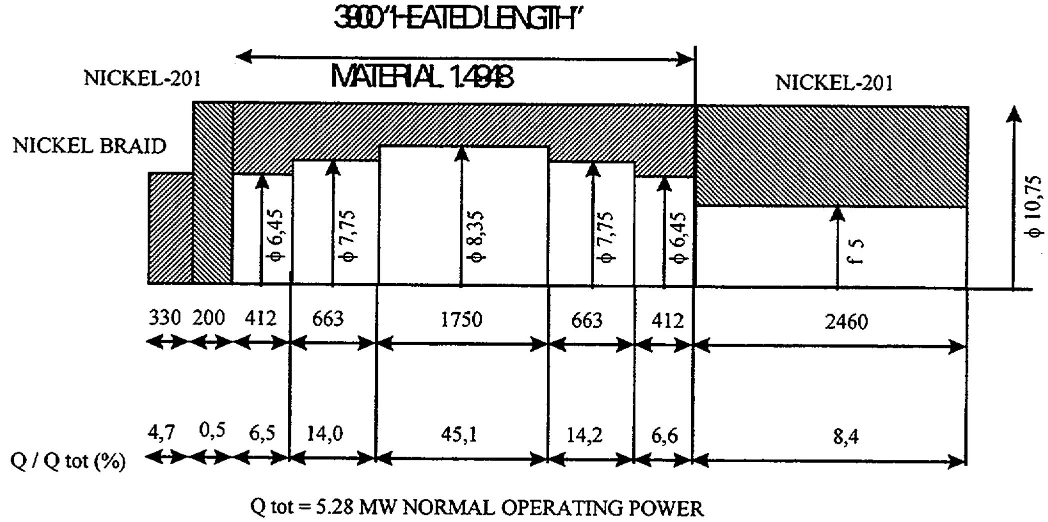

Figure 2.1. Heater rod (section and power distribution)

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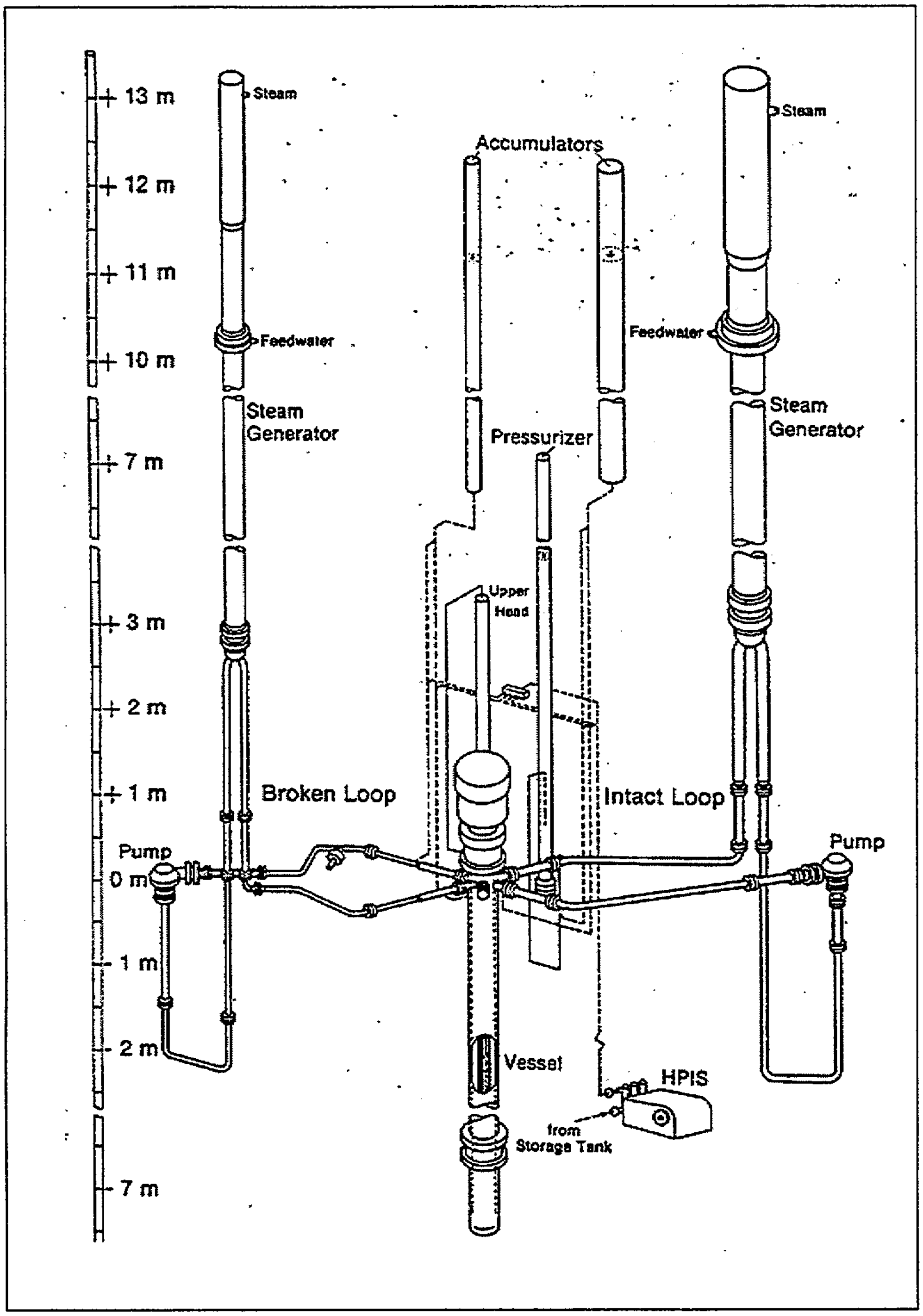

Figure 2.2. Schematic diagram of the primary system

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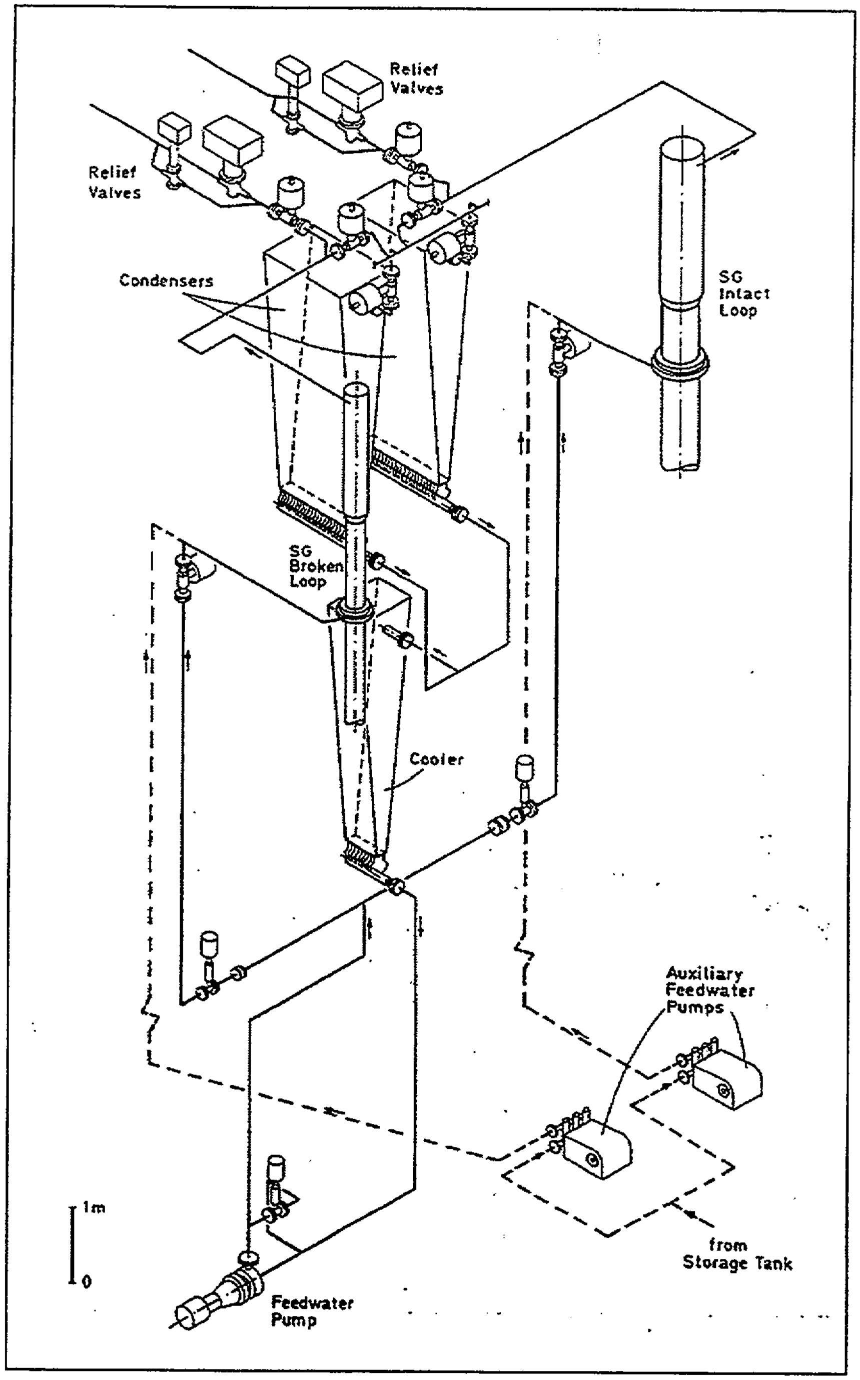

Figure 2.3. Schematic diagram of the secondary system

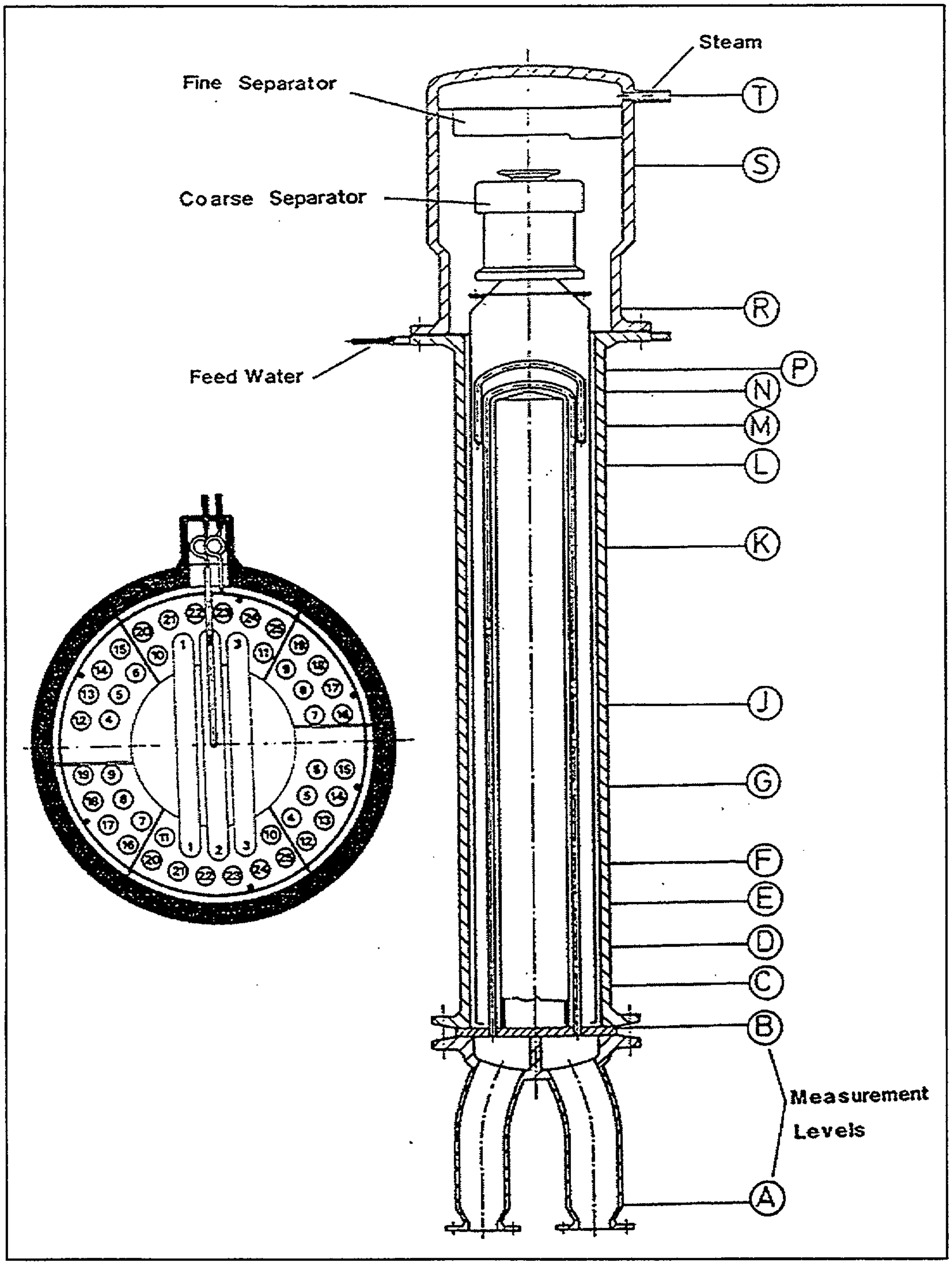

Figure 2.4. Intact loop steam generator

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3. DESCRIPTION OF THE EXPERIMENT

The BT-56 experiment represents a transient of multiple inadvertent failures developing on the basis of an originally specified LOFW type transient.

The configuration of the system is shown in Table 3.1.

Pressurizer	• Surge line connected to the hot leg of the broken loop • PORV in control mode at 162 bar • SRV setpoint 167/164 bar, with an orifice of 2.44 mm
HPIS	• Disconnected
AIS	• Disconnected
AFW	• Actuation on low level in the SG downcomer and PCS temperature.
SG Control	• Pressure control during the initial part of Test BT-56 with a pressure increase rate of 0.07 bar/s • Following the Scram, foreseen at 59s, pressure control at 70 bar with SRV
MCP seal water	• Extraction of flow from upper plenum during steady-state phase, using the pressurizer level control system • Extraction of flow from lower plenum during the transient, using the seal water tank control system
Containment	• Atmospheric

Table 3.1. Configuration of the facility during the experiment

The sequence of events occurring during the experiment was as follows:

- Failure to open of the IL steam generator relief valve. Steam relief from the IL steam generator occurred across the BL steam generator relief valve. Isolation of triple-loop steam generator, with subsequent loss of heat transfer on closure of the steam line valves at 60 s. The closure sequence of the valves (foreseen and actual) is shown in Figure 3.1.

- Shutdown of the main coolant pumps (MCP's) due to loss of pressure in the seal system, with loss of forced circulation in the primary.
- Delayed core scram.
- Loss of primary inventory due to rupturing of the safety disk.

As a result of all the above, the sequence of thermalhydraulic phenomena was particularly relevant for understanding of the interaction between the components and validation of the codes used for the light water reactor safety assessment.

The most important thermalhydraulic phenomena that occurred were as follows:

- Primary pressurization and heating.
- Compensation of steam generator pressures due to their having the same steam relief path.
- Primary and pressurizer pressure mismatch.
- Dryout of the core at high pressure.
- Loss of primary inventory from the downcomer.

The sequence of events that followed during the BT-56 experiment is shown in Table 3.2.

Time (s)	Event
0	• Closure of condenser valve and of the feedwater valves. • IL steam generator SRV fails closed.

	• Partial loss of heat removal in the secondary.
9.70	• Primary SRV and PORV open at 170 bar.
12.2	• Closure of primary SRV and PORV.
16.3	• Opening of primary SRV and PORV.
17.1	• Main pump trip on low differential pressure in the seal water system.
19.0	• Primary pressure greater than 86 bar.
19.1	• Break of rupture disk (orifice of 24.7 mm).
19.8	• Core dryout.
20.5	• Closure of primary PORV and SRV.
23.5	• Manual scram of the core
33.0	• Core Rewet
60.0	• Steam generator isolation
100.	• End of experiment

Table 3.2. Sequence of main events during Test BT-56

Analysis of the transient may be divided into the following phases:

- Initial pressurization prior to pump shutdown.
- Overpressurization of the primary with the pumps shut down.
- Loss of inventory (LOCA) across the rupture disk.

3.1. Initial pressurization prior to pump shutdown

The steam generated in the SG's during the steady-state phase was collected a header and channelled to the condenser. At the beginning of the transient the steam generator feedwater valves closed. The feedwater flow in the intact loop decreased rapidly. The feedwater flow in the broken loop dropped to 8% of the rated value and remained at this value for approximately 36 seconds, at which moment it disappeared completely.

Closure of the feedwater valves also implied closure of the condenser valve (simulating a turbine trip), at the same time as opening of the steam generator relief valves. However, the valve on the intact loop steam generator did not open, as a result of which steam could be dumped via the relief valve on the broken loop steam generator. The scheduled opening procedure and the procedure actually performed are shown in Figure 3.1.

As a result of closure of the condenser valve and of failure of the steam dump, the secondary pressure and temperature increased, causing the temperature of the primary to increase also. This heating of the primary gave rise to an expansion in the pressurizer, causing the PORV and SRV setpoints to be reached.

The opening of these two valves served to control pressurizer pressure, as a result of which this pressure decreased until the valves closed.

Following the closure of the valves, the core was still active, so there was a second pressure increase which caused the valves to re-open. Pressurization of the primary, which in turn caused a reduction in the differential pressure between the seal water injection and the primary, caused the MCP's to trip as a result of the pump protection system. This protection system is equipped with brakes that stop the pumps immediately. In less than half a second the pumps stopped completely, forced circulation suddenly being lost. This effect of sudden shutdown may be seen in the fluid velocity graphics. The core continued at full power.

3.2. Primary overpressurization with the pumps shut down.

As a result of the shutdown of the pumps, the mass flow in the primary loops decreased rapidly. On the secondary side, the loss of forced flow caused a drop in pressure, as a result of the decrease in primary-secondary heat transfer. This led to a sudden increase in primary pressure, with the pressurizer valves open and discharging steam.

The flow across the pressurizer valves increased considerably. These valves were capable of controlling the pressure in the pressurizer, but not in the primary. Consequently, pressurizer pressure decreased and the pressure in the primary increased. This was due to the fact that the maximum flow along the surge line was lower than the maximum flow across the valves, as a result of which there was a mismatch between the pressurizer and primary pressures.

All the absolute pressure measuring instruments went off scale, due to the calibration range being limited to 17.4 MPa. The only exception to this were the pressure measurements in the pressurizer and the upper plenum, where there were wide range devices (with KDG extension).

3.3. Loss of inventory (LOCA) across the rupture disk.

The increase in primary pressure led to the break of the rupture disk, causing a loss of inventory (LOCA) at high pressure and with the core at full power.

The sudden depressurization that occurred as a result of the disk rupturing, in combination with the high temperatures in the core, caused the coolant to reach saturation temperature and to vaporize. The location of the break, along with the loss of forced circulation, caused flow reversal to occur in the hot legs and core. The phase change caused resistance coolant heat transfer to worsen drastically, as a result of which the temperature in the resistances rose sharply, a temperature of 645.7°C being reached in the central part of the core.

The manual scram of the core which occurred at 23.5 s (it was originally scheduled at 59 s) limited the maximum temperature reached in the resistances. Certain areas of the core, and the hot legs, underwent condensation and rewetting.

The pressurizer dried out at 40 s, causing an increase in volumetric flow along the surge line, due to phase change.

At 50 s reversal of the heat sink occurred. The primary pressure and temperature became lower than those in the secondary, as a result of which the secondary ceased to be the heat sink and began to transfer heat to the primary. From this moment onwards, there was stratification in the loops as a result of the presence of overheated steam and water in saturation conditions, as may be deduced from the measurements taken by the different thermocouples[1].

At 60 s the steam line isolation valves closed. This implied a disconnection between the two steam generators, which until that time had shared the only open relief valve. The intact loop lost all steam dump capacity and could be cooled only via the primary system. The BL steam generator was now able to release more steam, thus increasing its depressurization.

According to the scheduled sequence, steam generator dry out was foreseen at 100 s; however, due to a series of unforeseen events, this did not occur. The AFW did not actuate at any time during the sequence.

At the end of the loss of inventory phase, the primary circuit was completely depressurized, while the secondary continued to be pressurized.

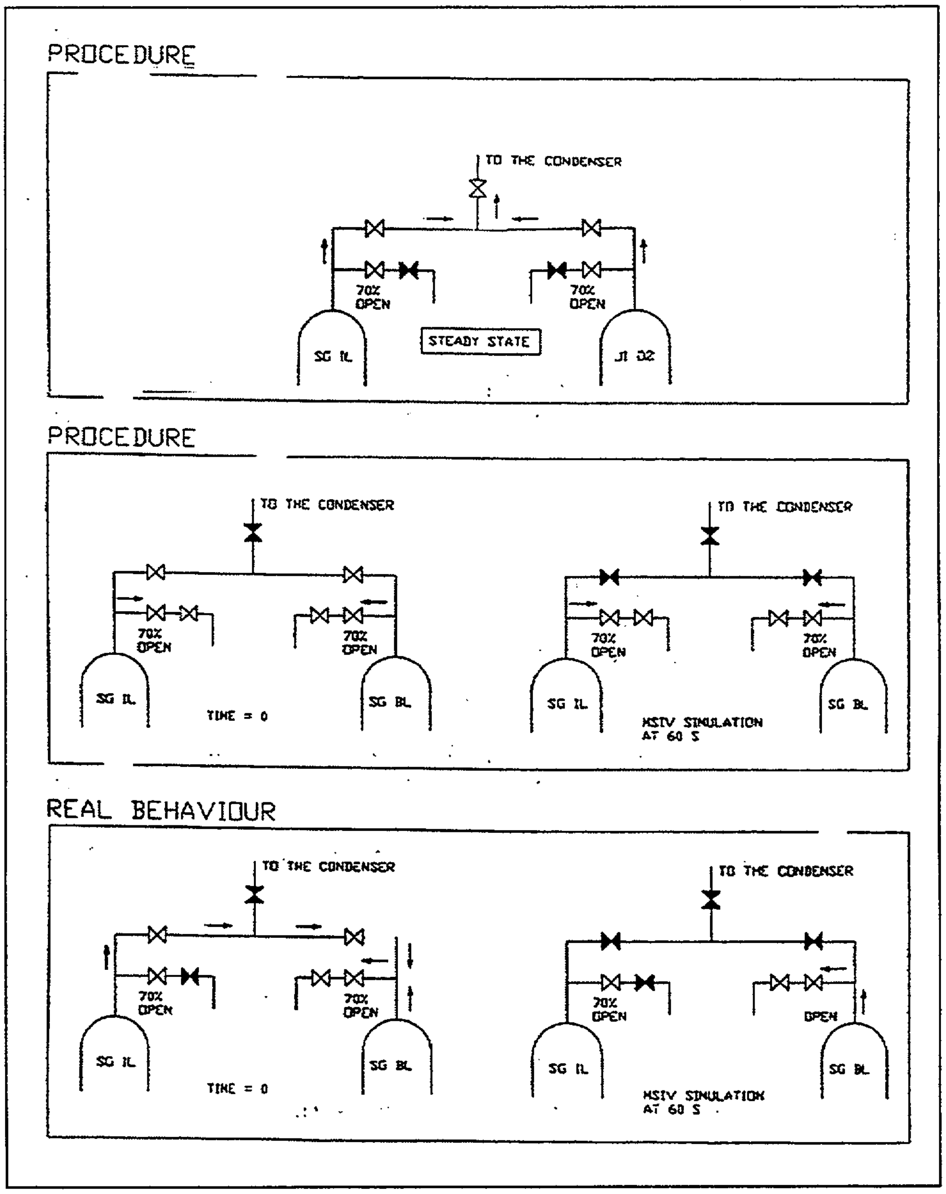

Figure 3.1. Schematic diagram of secondary operation during Test BT-56

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4. DESCRIPTION OF THE PLANT MODEL FOR RELAP5/MOD3.2

The RELAP5/MOD3.2 code was used for this analysis, running on a Digital AlphaServer 2000 4/200 computer under the DIGITAL UNIX operating system.

With a view to creating the data input file for the LOBI-MOD2 facility, Dr. C. Addabbo of the Joint Research Center (Ispra) was requested to supply the information available on Test BT-56, and the following was received:

- A disk containing the nodalization file for RELAP5/MOD2.5 plus another file with the nodalization for RELAP5/MOD3.0 of the LOBI-MOD2 reactor. These files were prepared for Test BT-17 (Loss of Feedwater Transient, LOFW).

- A disk containing the measures obtained during the BT-56 experiment.

- Reports on the experimental results for BT-56 (References /1/, /2/ and /3/). Subsequently, information was received on the control systems /4/ and on the characteristics of the main system valves /5/.

The plant nodalization used in the present analysis is shown in Figure 4.1. The nodalization is organized on the basis of various categories, depending on the number of components: primary system piping and components of the intact loops (series 100) and broken loop (series 200), pressurizer and surge line (series 400), reactor vessel (series 300), intact loop steam generator secondary side (series 500) and the same for the broken loop (series 600), emergency core cooling systems (series 700 and 800) and discharge lines and containment (series 900).

Nodes 363, 364, 365, 375, 370 and 380 correspond to the vessel head, which is an external assembly, as may be appreciated in Figure 4.2.

The starting point used consisted of the inputs for BT-17 for RELAP5/MOD2.5. This file incorporated the control systems and plant trips established for this experiment, including the following among others:

- Steam generator level control by FW control

- PRZR level control by removal from the upper plenum and make-up via the pump seals

- Adjustment of circulation ratio by means of valves 520 and 620

- Power generated by the electrical resistances in the core and the pressurizer heaters

- Pressurizer pressure control system (Cooling Coils) simulated by a control volume isolated during the transient (tmdpvol 430 and valve 425)

- Heat leakage throughout the system

Comparison of this file with the general model for LOBI-MOD2 /3/ showed a series of changes made at the JRC for the simulation of experiment BT-17:

- Pressurizer located on broken loop
- More detailed nodalization in the steam generators
- Replacement of safety injection for the accumulators
- Added volumes and junctions for the simulation of leakages

Although experiment BT-17 was very similar to BT-56, it was also a Loss of Feedwater scenario, there were significant differences between the two. This implied an in-depth review of the Ispra model for the LOBI-MOD2 facility and its modification on the basis of References /1/,/2/,/4/,/5/ and /7/, adapting it for the RELAP5/MOD3.2 version. The main modifications made to the original model were as follows:

- Initial temperature, pressure and flow conditions in the primary and secondary, under steady-state conditions (initialization).

- Differences in feedwater temperatures (bounding conditions).

- Activation and control of the pressurizer heaters and core electrical resistances (different sequence).

- Change in reference levels and circulation ratios in the steam generators under steady-state conditions.

- Adjustment of the by-pass of the upper part of the vessel and associated pressure loss coefficients.

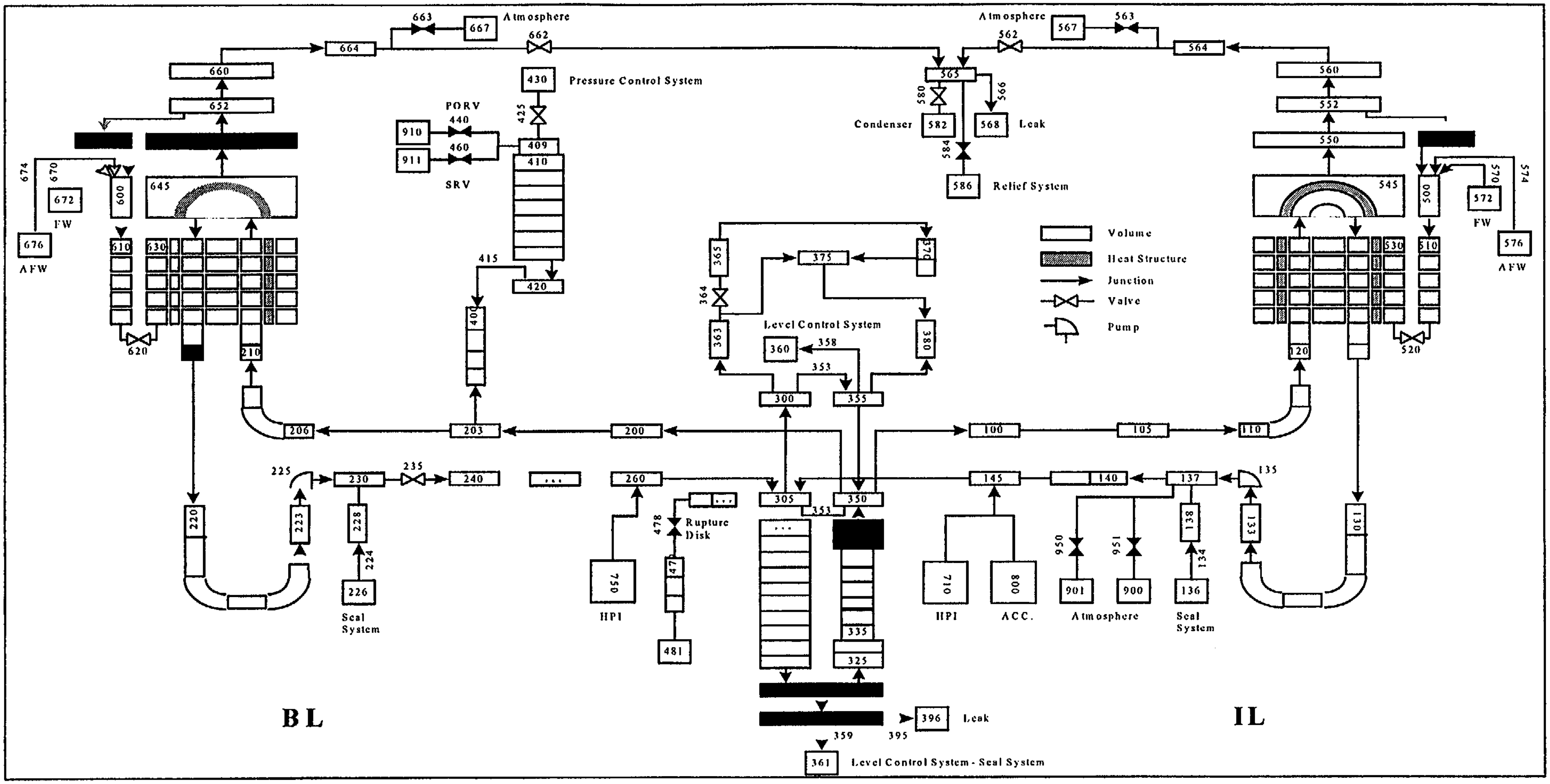
- Change in the area of the pressurizer relief and safety valves (PORV and SRV), pressure loss coefficients (/1/ and /5/).

- Adjustment of primary-secondary heat transfer by correcting the tube fouling factor (present in version 3.2 of the code). No steam generator tube bundle structure was incorporated, since the geometry of both the intact and broken loops was inadequate for simulation with this type of structure (few tubes, arranged in an annular configuration).

- Adjustment of the temperatures in the walls of the core resistances through correction of the fouling factor in the heat structures. No bundle structure has been incorporated for reasons similar to those presented above in relation to the steam generators (mesh of few tubes).

- Adjustment of pressure loss coefficients and discharge coefficients across the broken loop steam generator relief valve.
- Modifications to vessel head nodalization.
- Incorporation of detailed nodalization of the rupture disk and corresponding discharge line.
- Incorporation of steam line closure or isolation valves.
- Control of equipment operation in accordance with the transient sequence.

Q003 2



Atmosphere
Pressure Control System
PORV
SRV
Condenser
Leak
Relief System
FW
AFW
Volume
Heat Structure
Junction
Valve
Pump
Level Control System
Rupture Disk
Seal System
HPI
ACC.
Atmosphere
BL
IL
Leak
Level Control System - Seal System

Q004 2

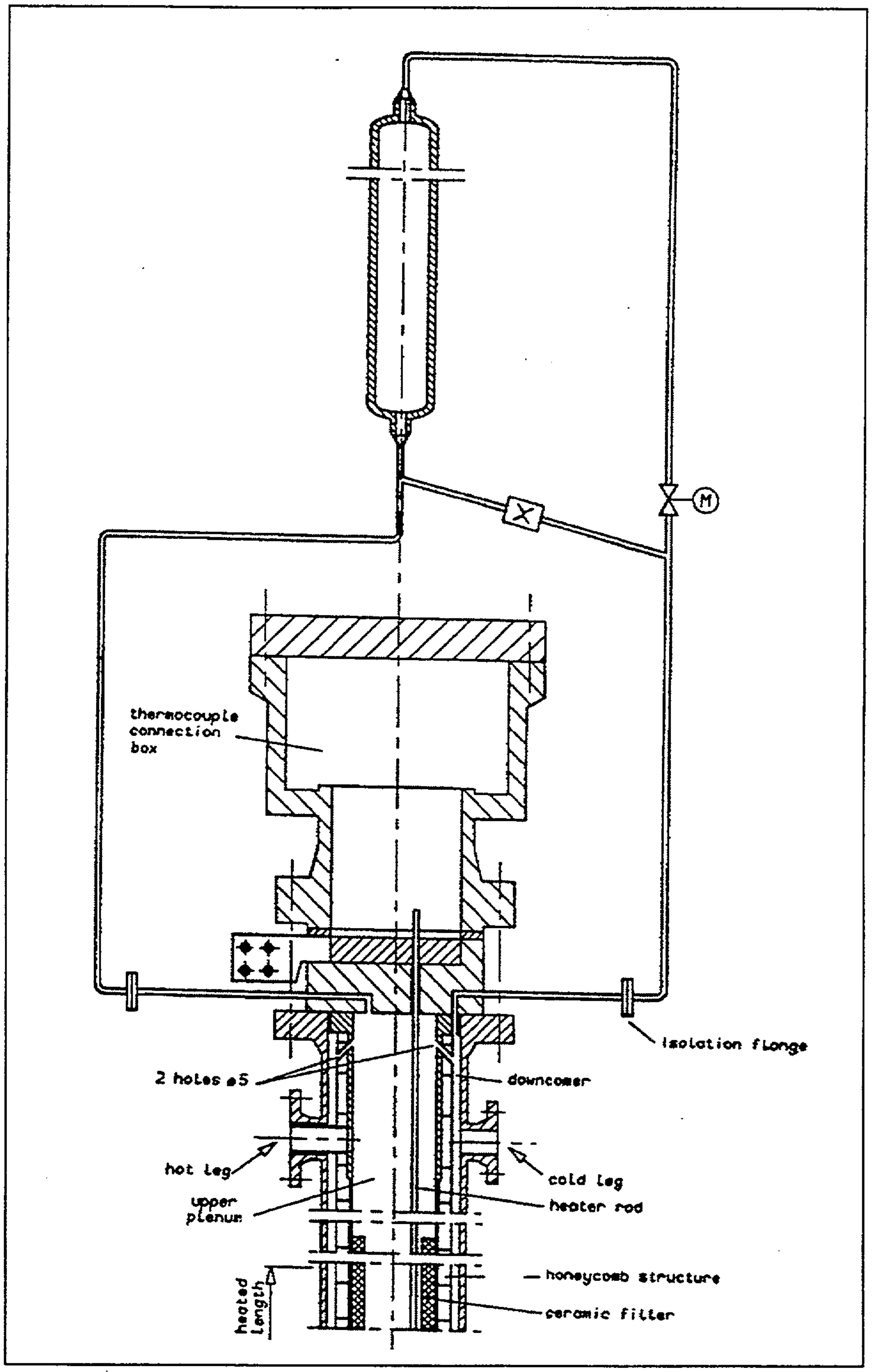


Figure 4.2. Reactor vessel head

Q003 2

5. STEADY-STATE SIMULATION

Adjustment of the steady state of the model developed for RELAP5/MOD3.2 of the LOBI-MOD2 reactor is the first step to correct simulation of the transient analyzed.

This adjustment is based on the information provided by the wide range of instrumentation with which the facility is equipped during the 20 seconds prior to initiation of the transient. During this period, the system was in steady-state conditions with slight oscillation of its values.

This section presents a brief analysis of the steady-state data for Test BT-56 /2/, along with the results obtained using the RELAP5 model following the adjustment.

5.1. Analysis of the Primary

In general there are few absolute pressure measuring devices in the system, although there are quite a number of differential pressure measurements, these having been used for adjustment of pressure losses under steady-state conditions.

The set of temperature measurements available is much more complete than that of absolute pressure. Comparison of the temperature measurements of the two loops shows that the hot leg temperatures are different, due to the presence of the by-pass in the vessel. According to the information obtained from reference/7/, there are three different by-pass paths:

- vessel head by-pass
- by-pass at the elevation of the lines due to nozzle losses
- by-pass due to the presence of 5mm orifices connecting the downcomer and the upper plenum.

From the experimental data /2/ it may be deduced that the effect of the by-pass mainly affects the hot leg of the intact loop. Thus, it is observed that hot leg of the broken loop shows the same temperature as the upper plenum, while the intact loop hot leg temperature is 4 °C lower. The nodalization and pressure losses in the upper part and the head of the vessel of the model for RELAP5/MOD3.2 were adjusted such that the effect of the by-pass was practically all on the intact loop, correct temperature simulation being obtained (as shown in Table 5.3.1).

Mention should be made also of the asymmetry of heat removal via the secondary, this being due to the fact that the steam generators are very different and that steam flow control is common to both. This control uses the cold leg temperatures of both loops weighed in terms of their flows, regulating the only steam exit valve to the condenser. All the above was taken into account in the simulation by means of the models of the steam generators and their turbine inlet control system.

Flows converge on the reactor vessel at different temperatures: the cold leg flows of both loops and the flow resulting from the vessel head by-pass. The lowest temperature is the head by-pass, which is some 4°C lower than the cold leg temperature. This means that the fluid at the inlet to the downcomer presents different temperatures, depending on the angle of measurement. The angles at which the instrumentation is located at different elevations varies but, as may be seen, the fluid is mixed as it flows along the downcomer. In view of the single dimension nodalization performed by RELAP5, the simulation cannot reproduce this type of effect.

The temperature in the walls of the core resistances is measured by means of thermocouples located at different elevations (levels). The adjustment of the temperatures in the walls of the core heat structures was obtained by adjusting the fouling factor for these structures. Determination of this parameter was based on the study attached in Appendix I.

5.2. Analysis of the Secondary

The pressure in the dome of the intact loop steam generator shows a slightly higher value than that in the broken loop, despite the two being connected to a common steam outlet. This is due to the different pressure losses presented by the steam lines, due in turn to the flows and diameters of the pipes being different. The pressure losses in the steam lines were adjusted in the model in order to take this effect into account. Mention should be made of the fact that during the last 20 seconds of the steady state the secondary pressure oscillated significantly.

Although the feedwater for both steam generators comes from one same tank, the experimental data show that the temperature of this water is different in the two generators, as was considered in the simulation. The steam outlet temperature for both loops was, however, practically identical. Furthermore, it should be pointed out that the steam temperature measured by thermocouples TF87S and TF97S for each of the loops is 2 °C higher than the saturation temperature corresponding to the pressure measured by the respective pressure measuring (PA87S and PA97S).

5.3. Steady-State Results

The steady state is the starting point for analysis of the transient. The calculation performed was for 500 s with an algorithm for time control implicit in heat transfer and hydrodynamics (flag 7). Table 5.3.1 shows a comparison between the plant measurements for Test BT-56 and the steady-state results calculated using RELAP5/MOD3.2. The values specified by JRC for Test BT-56 showed a slight variation with respect to those obtained during performance of the experiment.

	Specified	Test BT-56	RELAP5/MOD3.2	Units
Primary System				
Pressure in upper plenum	15.8	15.87	15.85	MPa
Core power	5.28	5.23	5.23	MW

Intact loop				
- Mass flow	21.0	20.9	20.9	Kg/s
- Vessel inlet temperature	294	296	295.56	°C
- Vessel outlet temperature	326	326	326.36	°C
Broken loop				
- Mass flow	7.00	7.30	7.30	Kg/s
- Vessel inlet temperature	294	296	296.96	°C
- Vessel outlet temperature	326	329	329.97	°C
Pressurizer				
- Liquid level	4.80	5.10	5.10	m
- Temperature	346	348	350.77	°C
MCP seal flow injection				
- Intact loop	-	0.01	0.015	Kg/s
- Broken loop	-	0.005	0.005	Kg/s
- Temperature	-	30	30	°C
Secondary System				
Intact SG loop				
- Steam pressure in dome	6.45	6.52	6.52	MPa
- Mass FW flow	2.00	2.00	1.96	Kg/s
- Inlet temperature	210	207	206.85	°C
- Outlet temperature	280	281	281.07	°C
- Level in downcomer	8.00	8.07	8.07	m
- Circulation ratio	-	6.40	6.40	-
Intact SG loop				
- Steam pressure in dome	6.45	6.51	6.52	MPa
- Mass FW flow	0.67	0.76	0.73	Kg/s
- Inlet temperature	210	204	204.05	°C
- Outlet temperature	280	281	281.04	°C
- Level in downcomer	8.40	8.45	8.45	m
- Circulation ratio	-	4.3	4.3	-

Table 5.3.1. Steady-state results

Q003 2

6. TRANSIENT SIMULATION

This section describes the results of simulation of the transient, explaining the evolution of the main variables, such as pressures, temperatures and densities.

The following sections contain information on the simulation performed and the results obtained, as well as on the statistics obtained from calculation CPU consumption.

6.1. Characteristics of Transient Simulation

The 500 s steady-state having been obtained with the model of the facility for RELAP5/MOD3.2, suitably adjusted, simulation of the accident sequence began. A 100 s restart transient type input was used for this new calculation, including the pertinent changes for adjustment of the transient.

- The feedwater flow in both steam generators was adjusted during the transient. For this purpose a flow table was introduced based on time for each SG, in accordance with the plant measurements. This implied cancellation of the feedwater flow control used for steady-state conditions.

- Closure of the turbine valve and opening of the broken loop steam generator relief valve were simulated by means of time-activated trips. The coefficients of pressure loss for the relief valve are those referenced by JRC /5/. The discharge coefficients across this valve were adjusted in order to fit steam flow.

- On the primary side, the dimensions of the PORV and the SRV were changed in accordance with the experiment documentation. The pressure loss coefficients were adjusted depending on the fraction of opening, in accordance with curve /1/. In order to make the valves more realistic, the decision was taken to simulate them by means of the "motorvalve" model, which allows opening time to be added. The PORV was assumed to be the slower of the valves, the opening time included being considerably longer than that of the SRV.

- Time-based tripping of the main pumps was simulated. The post-trip pump speed curve was simulated by introducing the pump revolutions per minute curve provided by JRC /2/ for the BT-56 experiment. The head losses associated with these pumps following shutdown were also simulated.

- As regards tripping of the electrical core, a minor decrease in power was simulated prior to the trip. As from this moment, the electrical power drops suddenly, reaching zero almost instantaneously. All the above was simulated using a power versus time curve, with the values measured at JRC for the BT-56 experiment /2/.

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- The rupture disk was simulated by means of an instantaneously opening valve (tripvalve). The rupture disk discharge coefficients were adjusted.

- The seal flows were simulated by means of time-based tables, in accordance with the measurements obtained during the BT-56 experiment.

- Extraction from the upper plenum was cancelled at the beginning of the transient, in accordance with the experiment documentation.

- The HPI and ACC systems were cancelled out, since they did not actuate during the transient.

- The pressurizer pressure control system (Cooling Coils) was cancelled out, since it did not actuate during the transient.

- Pressure loss coefficients were introduced for reversed flow (K reverse) at the vessel inlet from the hot legs, as well as at the inlet to the BL pump, in order to adjust the velocities in the loops.

- The pressure losses in the surge line were adjusted in order to correctly simulate the mismatch between the primary and pressurizer pressures.

- The transient 100s simulation was performed using the calculation algorithm implicit in heat transfer and hydrodynamics time (flag 7), this ensuring correct heat transfer.

6.2. Transient Simulation Results

The graphs included in Appendix II show a comparison between the evolution of Test BT-56 and the results of the transient simulation performed using RELAP5/MOD3.2. The transient lasted 100 s, as a result of which the last 20 seconds of steady-state conditions are represented in the graphs(480 s to 500 s) along with the 100 s of the transient (500 s to 600 s).

The main results of this comparison are as follows:

- Analysis of the graphs shows that the simulation of the transient using RELAP5/MOD3.2 presents good agreement as regards the evolution of the heating and the primary pressure peaks during the initial phase of the transient (fig. 1 to 4). The times obtained for opening of the valves and rupturing of the rupture disk coincide with those measured during Test BT-56.

- The depressurization of the primary following rupturing of the disk (after 19.1 s) was correctly simulated, although it may be observed that the primary pressure predicted by RELAP5 is slightly higher than that measured in the facility, this meaning that the primary becomes the heat sink 5 s later (fig 21). Nevertheless, the simulation presents a more pronounced decrease than that measured at the plant during the interval 55-70 s into the transient. This is due to an increase in the flow across the rupture disk (fig 41), caused by a change in phase (SLG to ANM) in the upper part of the downcomer (fig 42).

- The pressure mismatch and drying out of the pressurizer were correctly simulated, and the time taken for the pressure to converge with that of the primary agreed with the experimental values (fig 20).

- The effects of drying out of the primary and of flow reversal in the hot legs and core were correctly simulated (fig. 5 to 8). Mention should be made of the fact that the plant QF reference meters measure the velocities of two-phase, non-stratified flows and that they do not measure direction (this measure is always positive).

- The temperatures in the legs provided by the simulation are lower than the temperatures measured at the plant, as from 50 s into the transient (fig 9 to 12). This is due to the effect of the higher depressurization contemplated by the code for this interval.

- Following reversal of primary-secondary heat transfer, a temperature step is observed the broken leg hot leg (fig 11), following steam generator temperature. This effect is smoother in the simulation with RELAP5, due to lower heat transfer, and also occurs 5 s later.

- The thermal stratification that occurs in the legs as from 60 s into the transient, caused by the presence of superheated steam and liquid under saturation conditions, is qualitatively reproduced with RELAP5. The temperature obtained with RELAP5 for the superheated steam present in the hot legs (fig 9,11) shows a lower value than the experimental data. This effect could be due to the lower secondary to primary heat transfer in the simulation calculations.

- The core temperatures obtained with RELAP5 (fig 27 to 38) are lower as from 50 s into the transient, due fundamentally to the previously described hot leg temperature effect.

- The analysis of densities shows that the simulation presents a good adjustment in the legs (fig 23 to 26), although the core rewet shown is lower than that which occurs in the plant (fig 22).

- From analysis of the results it would appear to be the case that the secondary-primary heat transfer calculated by RELAP5 is lower than that actually occurring. This is due to the complexity of the phenomenon simulated, with the presence of steam inside the tubes and reversed flows in the hot legs. There is also high sensitivity to flow velocities in the legs.

Q003 2

- The evolution of the pressures (fig 19) and of the volumetric steam flow in the secondary (fig 17) were correctly simulated.

6.3. Run Statistics

The transient base case was performed on a Digital AlphaServer 2000 4/200 platform using the DIGITAL UNIX operating system.

The time elapse control algorithm used for the simulation with RELAP5/MOD3.2 is implicit in heat transfer and hydrodynamics (flag 7).

The maximum time step allowed was 0.05 s for the first 18 seconds of the transient and 0.001 s for the rest, although the simulation reduced this parameter as from 50 s. The CPU consumption time was 13773 seconds, giving a ratio of real time to CPU time of 138/1.

7. SENSITIVITY CALCULATIONS

Several sensitivity calculations were performed to study the importance of certain phenomena in the response of the model. The phenomena analyzed were: heat transfer in the steam generators and coefficients of discharge across the rupture disk and the BL steam generator relief valve. The graphs with the main results are included in Appendix II.

7.1. Coefficients of Discharge across the Rupture Disk

In this sensitivity calculation, an analysis was made of the response of the model to the coefficients of discharge. The rupture disk coefficients for the base case were adjusted as recommended for the code /8/.

Variations in the subcooled and superheated discharge coefficients causes slight changes in the evolution of primary pressure and temperature, but variations in the two-phase discharge coefficient produces significant effects. The sensitivity calculations included here consists of reducing the two-phase flow discharge coefficient:

Coefficients of discharge across rupture disk	Subcooled liquid	Two-phase flow	Superheated steam
Base Case (recommended)	0.8	1.2	1.0
Sensitivity	0.8	0.9	1.0

Variations are observed in the evolution of the primary pressure (fig. 43 vs fig.4) as a result of the change in the two-phase discharge coefficient (from 20 to 70 s), without any significant influence on the secondary (fig. 46 vs fig.19).

A significant effect is observed on the temperature of the electrical resistances in the core (fig. 44 vs fig. 30), with the maximum value reached increasing and cooldown delayed during the period of two-phase flow discharge. Differences are also observed in the cold leg temperatures (fig 45 vs fig. 12) after heat sink reversal.

7.2. Heat Transfer in the Steam Generators

The calculation performed consists of increase the heat transfer in the steam generators with respect to the base case. For this purpose, the value of the fouling factor was adjusted to 1.0 (steady-state and transient). The effect of fouling during reversal of the heat sink causes slight changes over heating of the steam in the legs (fig. 49 vs fig. 12) and the core (fig. 48 vs fig.30), without any significant influence on the secondary (fig. 50 vs fig. 19).

7.3. Coefficients of Discharge across the SG Relief Valve (BL)

In this sensitivity calculation, an analysis was made of the response of the model with varying coefficients of discharge across the BL steam generator relief valve. Variation in the subcooled and two-phase flow discharge coefficients causes slight effects on the secondary evolution, so the sensitivity calculations consists on change the steam flow discharge coefficient.

Coefficients of discharge across relief valves	Subcooled liquid	Two-phase flow	Superheated steam
Base Case	0.8	1.2	0.925
Sensitivity	0.8	1.2	1.1

The main differences are observed in the steam flow across the relief valve (fig. 53 vs fig. 17) and the pressure (fig. 54 vs fig. 19) in the steam generators, without any significant influence on the primary (fig 51, 52).

8. CONCLUSIONS

The LOBI-MOD2 reactor BT-56 experiment was unusual due to the multiple malfunctions that occurred during its performance, which caused a major deviation in the sequence with respect to the original approach.

The unexpected evolution of the sequence and the variety of events and phenomena that coincided made the result especially interesting for understanding of plant accident phenomenology, as well as for verification of the behaviour of the thermalhydraulic simulation codes.

The information obtained from the LOBI-MOD2 experimental facility has made it possible to adequately develop and adjust the model corresponding to the facility for RELAP5/MOD3.2. The calculations performed using the RELAP5 code have served to clear up some of the uncertainties encountered during transient analysis and development of the model, as well as to explain and confirm the phenomenology deduced from the experimental measures. In addition, the measures obtained from the facility have made it possible to verify the behaviour of the code when reproducing a transient as rapid and complex as the one dealt with here.

The simulation performed using RELAP5/MOD3.2 have faithfully reproduced the evolution of pressures as a result of closure of the condenser valve and shutdown of the primary pumps, as well as depressurization due to opening of the pressurizer valves and the break of the rupture disk.

The entire chain of sequences occurring during the experiment has also been correctly simulated: loss of forced circulation, rapid primary pressurization, depressurization of the secondary, high-pressure dryout and rewetting of the core. The time taken for the pressurizer to empty has been adequately calculated, as has the mismatch between the pressurizer and the primary circuit and subsequent convergence.

Likewise, RELAP5/MOD3.2 has acceptably reproduced heat transfer from the core and the evolution of temperatures in the walls of the rods. It should be observed that the three-dimensional temperature distribution effects in the core region cannot be simulated with RELAP5, since it is a single dimension model.

The phenomena that present the greatest uncertainty and a certain deviation with respect to what was deduced from the experimental data are: flow across the rupture disk (strongly influenced by the phase change occurring in the downcomer), primary-secondary transfer (with reversal of the heat sink) and condensation in the primary, causing core rewet.

A sensitivity analysis was performed on those parameters of the model showing uncertainties, with significant sensitivity observed in the evolution of pressures and temperatures in the core with respect to adjustment of the factors of discharge across the rupture disk. Mention should be made of the fact that the values assigned to the coefficients of discharge for better adjustment of the sequence are acceptable with respect to those referenced in the literature.

Judging by the results obtained, the RELAP5/MOD3.2 code demonstrates the capacity to satisfactorily reproduce a transient of these characteristics, showing adequate behaviour in simulation of the associated phenomenology.

9. REFERENCES

- /1/ "Quick look report on LOBI-MOD2 Test BT-56". Commission of the European Communities. JRC. No.: 4356. LQC 92-71. February 1994.

- /2/ "Experimental data report. LOBI-MOd2 Test BT-56". Commission of the European Communities. JRC. No.: 4352. LEC 93-73. December 1993.

- /3/ "RELAP5 Base Input Data for LOBI-MOD2". Commission of the European Communities. JRC. No.: 4030. LPC 85-18. February 1985.

- /4/ "LOBI-MOD2 Control Systems" Technical Note No. I. 06.C1.86.10, R. Wampach, February 1986

- /5/ Technical information on LOBI valves. 1613/308/87/BD-GFDS/bs. Referenced to TPTF-Meeting "LOBI Valves Characteristics". Ispra, 10/9/1987.

- /6/ "Test Specification LOBI-MOD2 Test BT-15/BT-16" C.Addabbo, LOBI. June 1, 1990.

- /7/ "Analysis of the Core Bypass in LOBI-MOD2 Test BC-04" Commission of the European Communities. JRC. No.: 4330. October 1990.

- /8/ "RELAP5/MOD3.2 Code Manual. Volumen V: User´s Guidelines". NUREG/CR-5535-V5. June 1995.

APPENDIX I: AVERAGE TEMPERATURES IN CORE HEAT STRUCTURES

Level - 1

No of thermocouples	TH32D301 (°C)	TH35D301 (°C)	AVERAGE (° C)
2	314	318	316

Level - 2

No of thermocouples	TH34D402 (°C)	TH35F402 (°C)	AVERAGE (° C)
2	323	322	322,5

Level - 3

No of thermocouples	TH32C103 (°C)	TH34E103 (°C)	AVERAGE (° C)
2	320	322	321

Level - 4

No of thermocouples	TH37C404 (°C)	TH37G404 (°C)	AVERAGE (° C)
2	326	327	326,5

Level - 5

No of thermocouples	TH35D105 (°C)	TH37F105 (°C)	AVERAGE (° C)
2	341	335	338

Level - 6

No of thermocouples	TH34D106 (°C)	TH35F106 (°C)	AVERAGE (° C)

2	344	346	345

Level - 7

No of thermocouples	TH32C407 (°C)	TH34E407 (°C)	TH37A407 (°C)	TH37E407 (°C)	AVERAGE (° C)
4	353	354	341	345	348,25

Level - 8

No of thermocouples	TH32A208 (°C)	TH32E208 (°C)	TH34G208 (°C)	TH35E208 (°C)	AVERAGE (° C)
4	342	349	352	355	349,5

Level - 9

No of thermocouples	TH31C209 (°C)	TH31G209 (°C)	TH34B209 (°C)	TH35D209 (°C)	TH37F209 (°C)	TH38C209 (°C)	AVERAGE (° C)
6	356	352	341	350	351	345	349

Level - 10

No of thermocouples	TH31A210 (°C)	TH32F210 (°C)	TH33G210 (°C)	TH34D210 (°C)	TH35B210 (°C)	TH36G210 (°C)	AVERAGE (° C)
6	349	352	352	355	343	356	351

Level - 11

No of thermocouples	TH31B311 (°C)	TH32C311 (°C)	TH33H311 (°C)	TH34E311 (°C)	TH37E311 (°C)	TH38F311 (°C)	AVERAGE (° C)
6	346	348	349	350	343	335	345

Level - 12

No of thermocouples	TH32E312 (°C)	TH34G312 (°C)	TH35A312 (°C)	TH35E312 (°C)	TH36B312 (°C)	TH38H312 (°C)	AVERAGE (° C)
6	351	351	335	349	345	332	344

Level - 17

No of thermocouples	TH33C217 (°C)
1	336

Level - 18

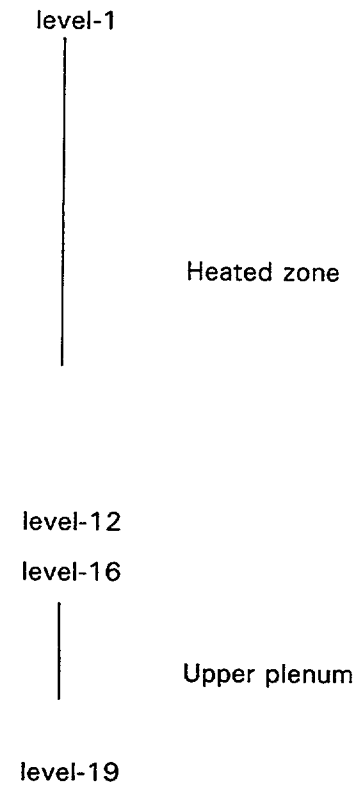
No of thermocouples	TH31D618 (°C)	TH35H518 (°C)	AVERAGE (° C)
2	335	338	336,5

Level - 19

No of thermocouples	TH33E119 (°C)	TH36D319 (°C)	AVERAGE (° C)
2	312	333	322,5

* Values of measurements in Test BT-56 LOBI-MOD2 /2/.

Length (m)	Avg, temp. (°C)	Measure level
0	316	Level - 1
0,16	322,5	Level - 2
0,44	321	Level - 3
0,66	326,5	Level - 4
1,16	338	Level - 5
1,66	345	Level - 6
2,16	348,25	Level - 7
2,66	349,5	Level - 8
3,16	349	Level - 9
3,38	351	Level - 10
3,66	345	Level - 11
3,82	344	Level - 12
4,71	336	Level - 17
5,51	336,5	Level - 18
6,17	322,5	Level - 19



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* The point of origin was taken at the first thermocouple level

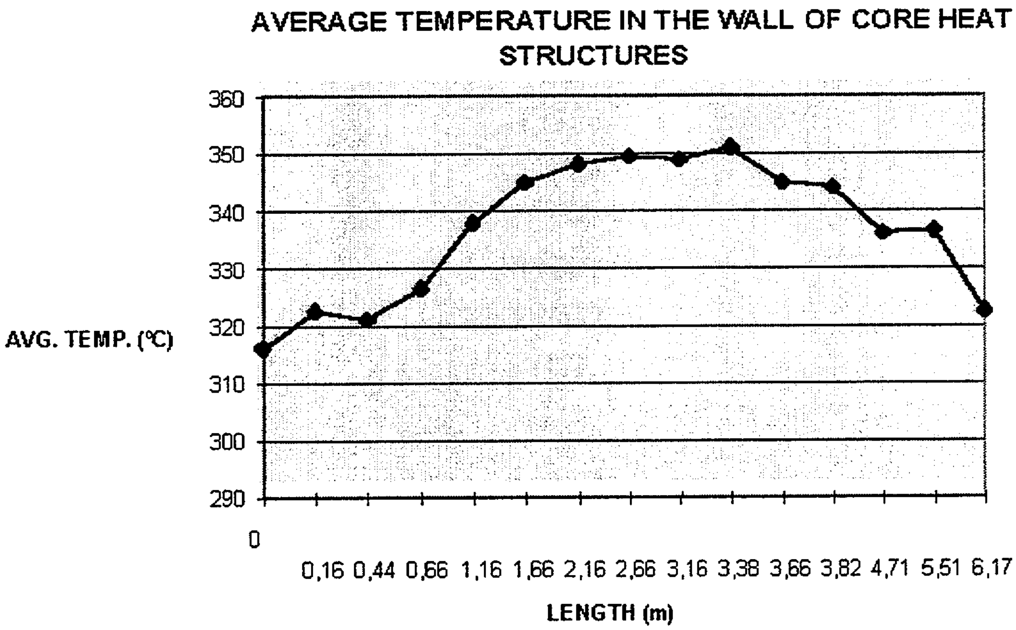

Figure A.II.1. Average temperature in the wall of core heat structures

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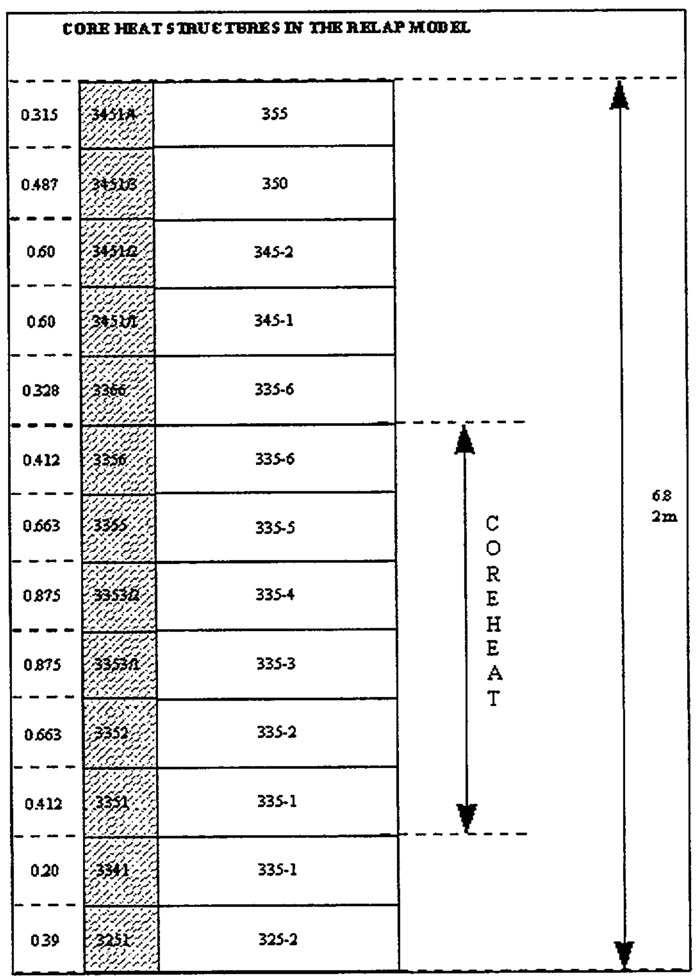

A.II.2. Core nodalization

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APPENDIX II: SIMULATIONS GRAPHICS

BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.1.– Pressure in IL Cold Leg

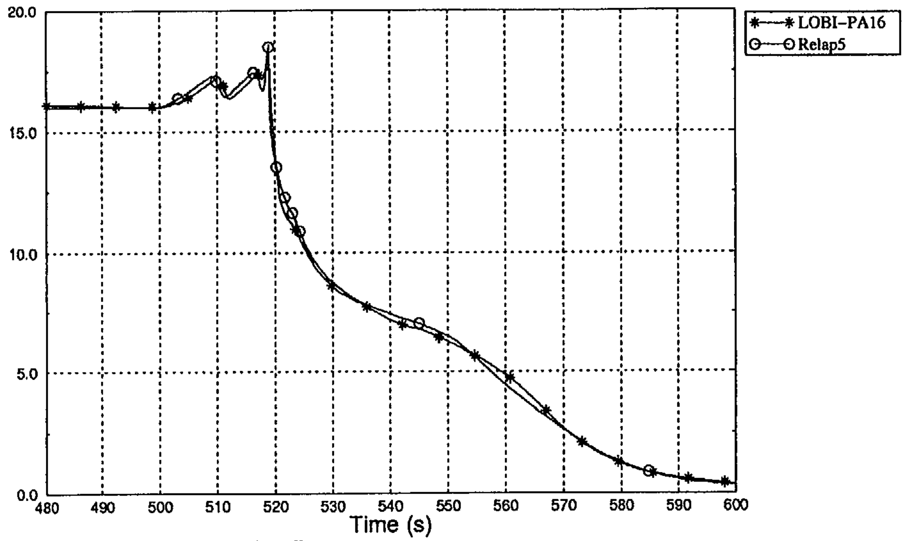


Figure A.II.2.– Pressure in BL Cold Leg

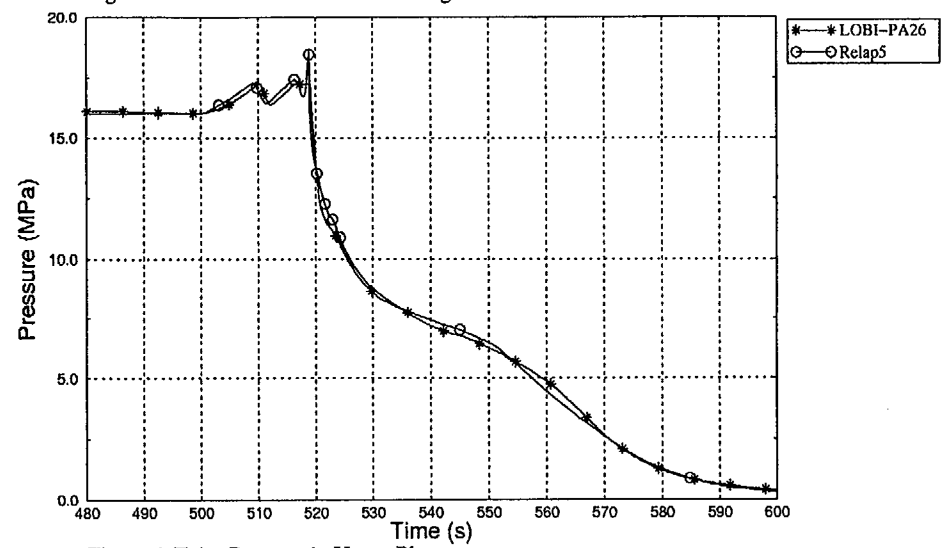


Figure A.II.3.– Pressurizer Pressure

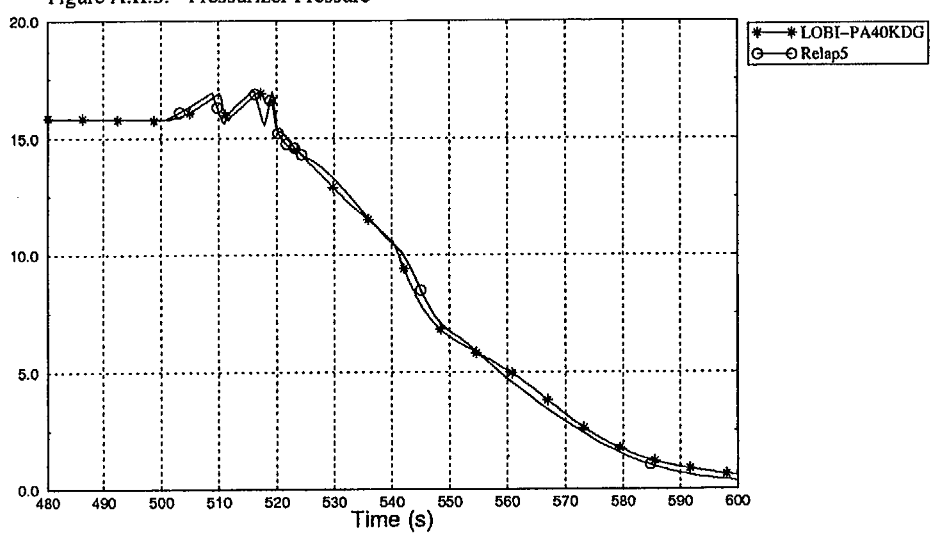
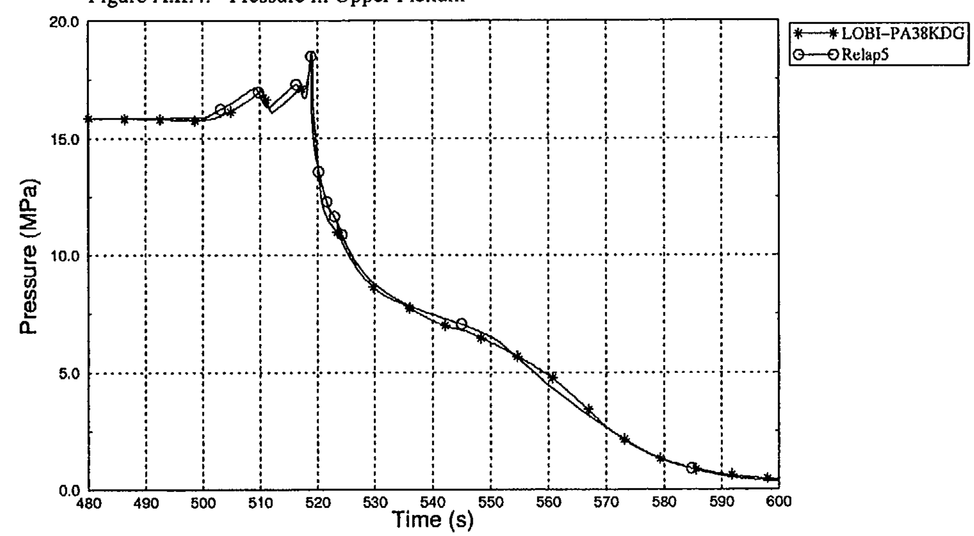


Figure A.II.4.– Pressure in Upper Plenum



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.5.– Flow Velocity in HL IL

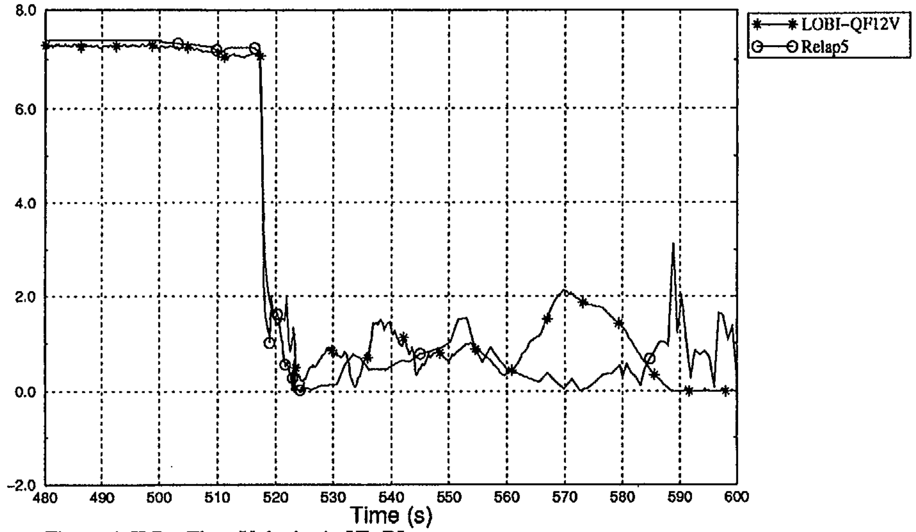


Figure A.II.6.– Flow Velocity in CL IL

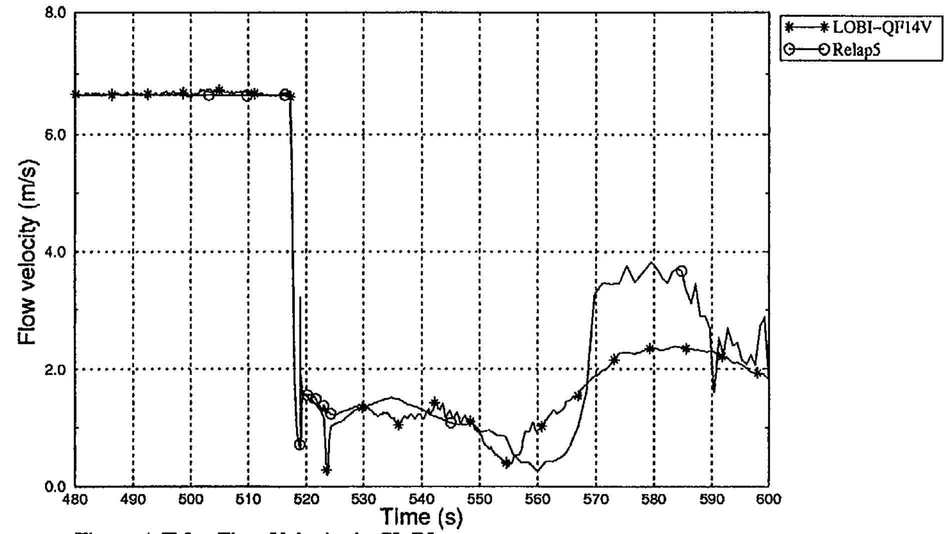


Figure A.II.7.– Flow Velocity in HL BL

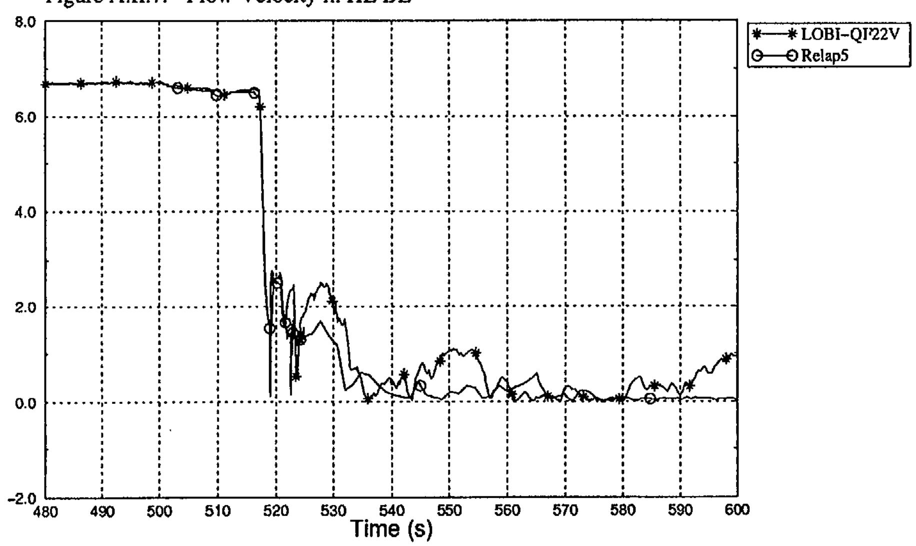
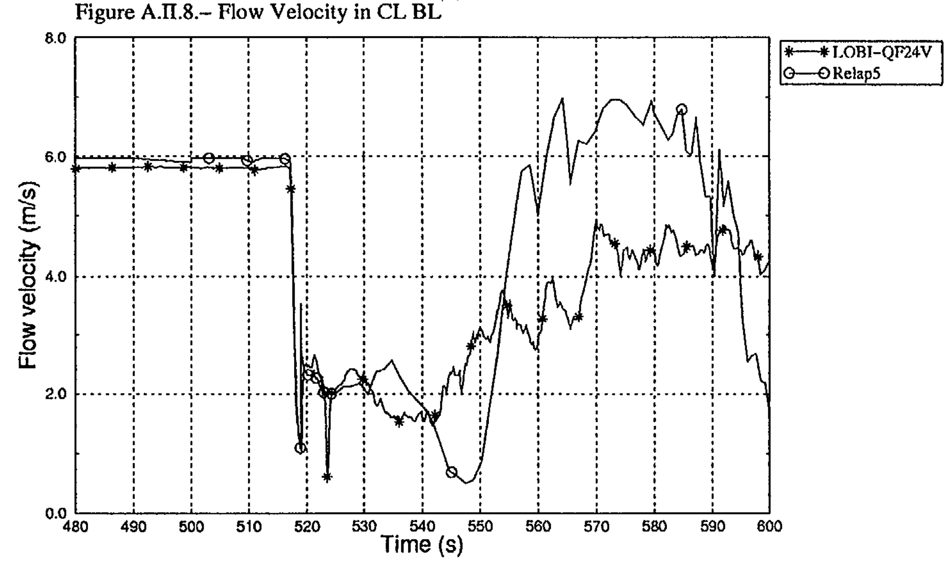


Figure A.II.8.– Flow Velocity in CL BL



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.9.– Temperature in HL IL

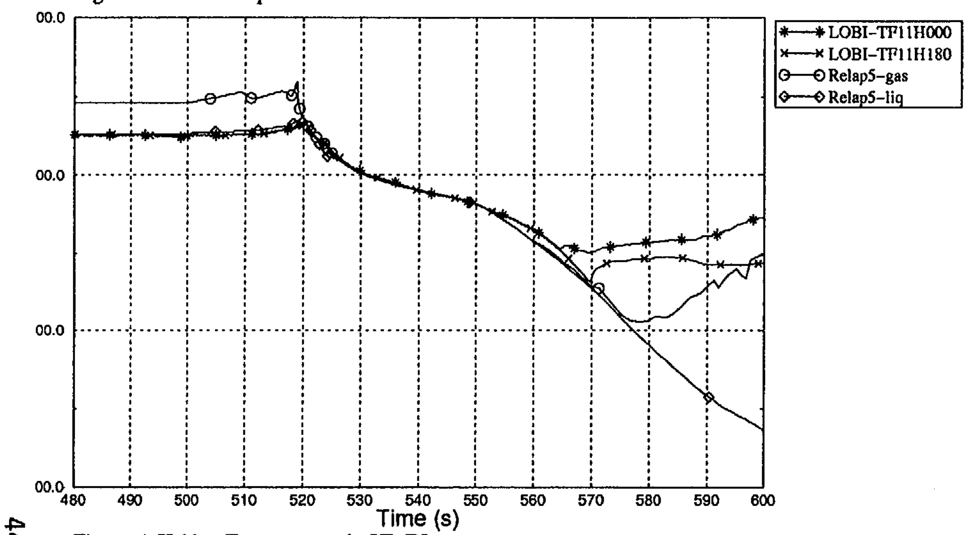


Figure A.II.10.– Temperature in CL IL

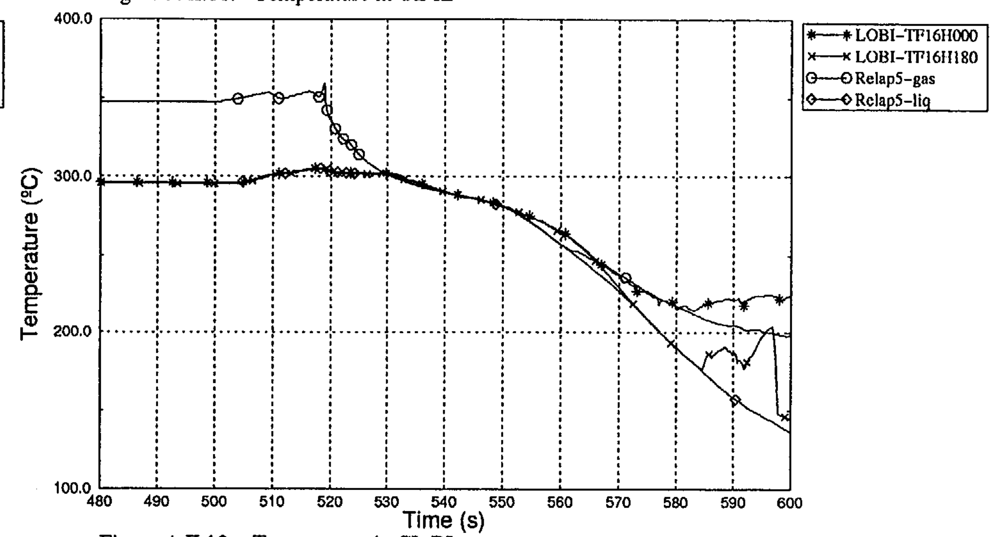


Figure A.II.11.– Temperature in HL BL

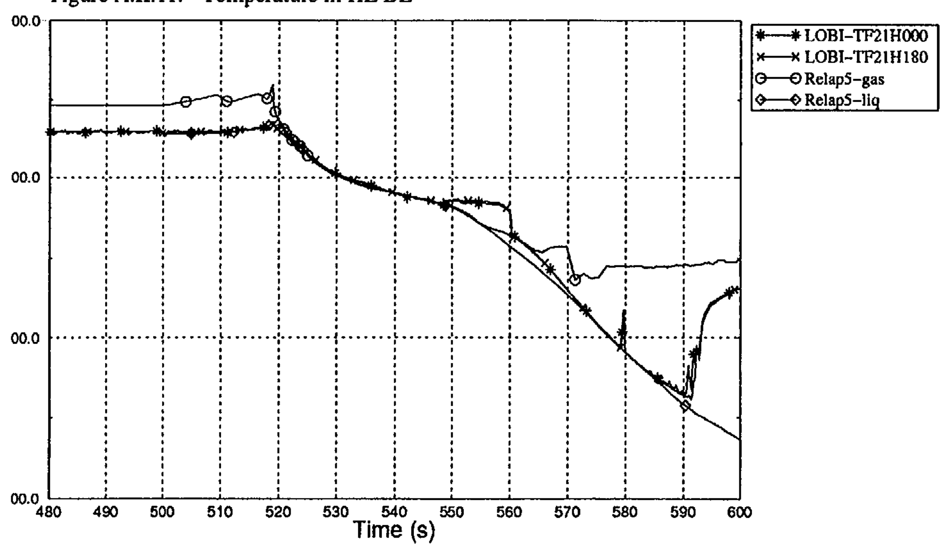
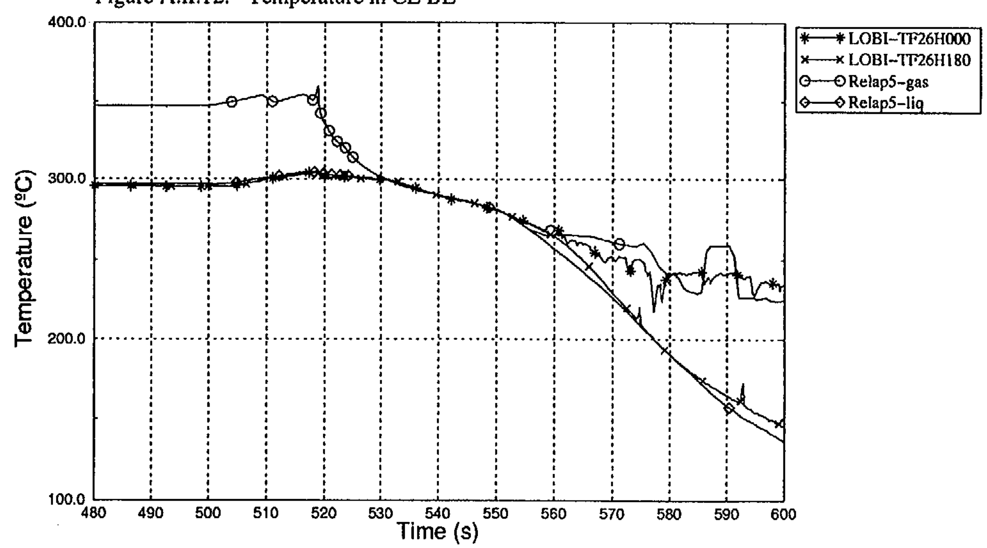


Figure A.II.12.– Temperature in CL BL



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.13.– Core Inlet Temperature

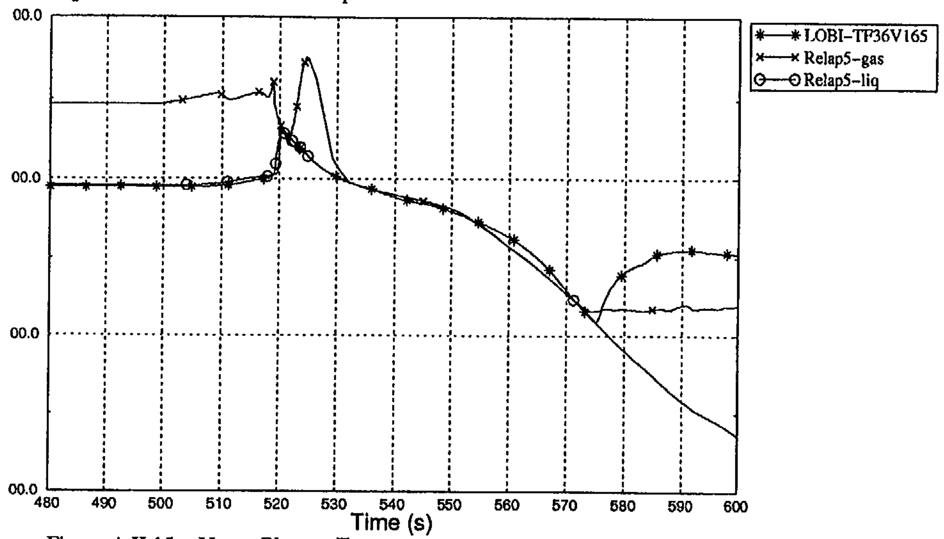


Figure A.II.14.– Core Outlet Temperature

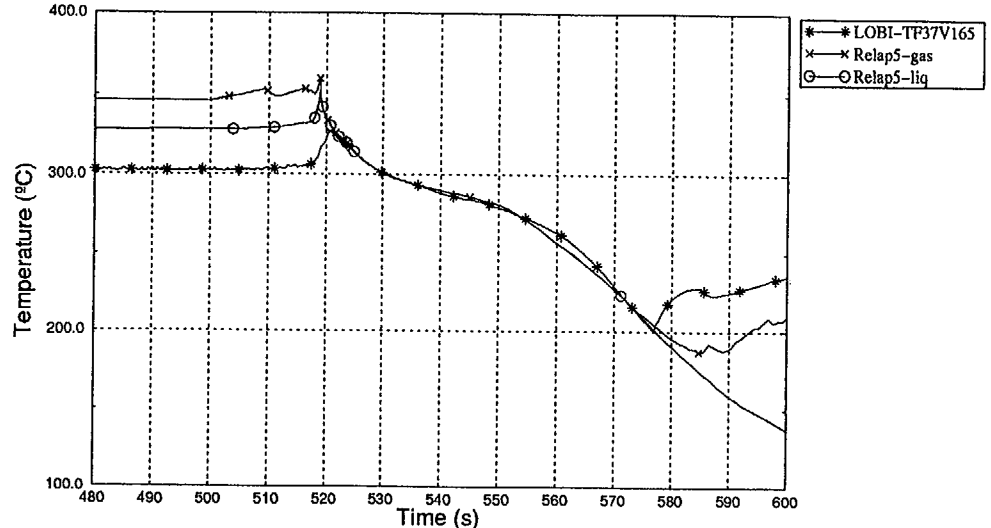


Figure A.II.15.– Upper Plenum Temperature

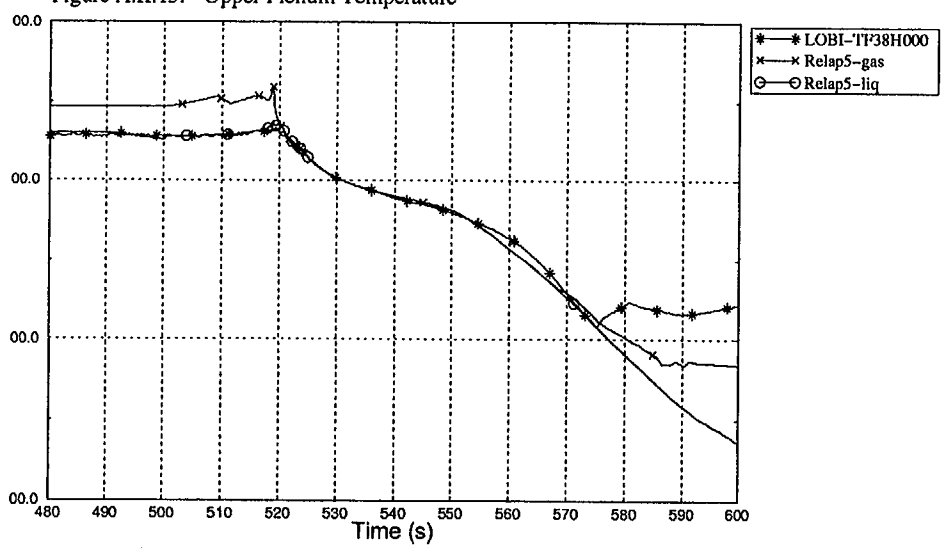
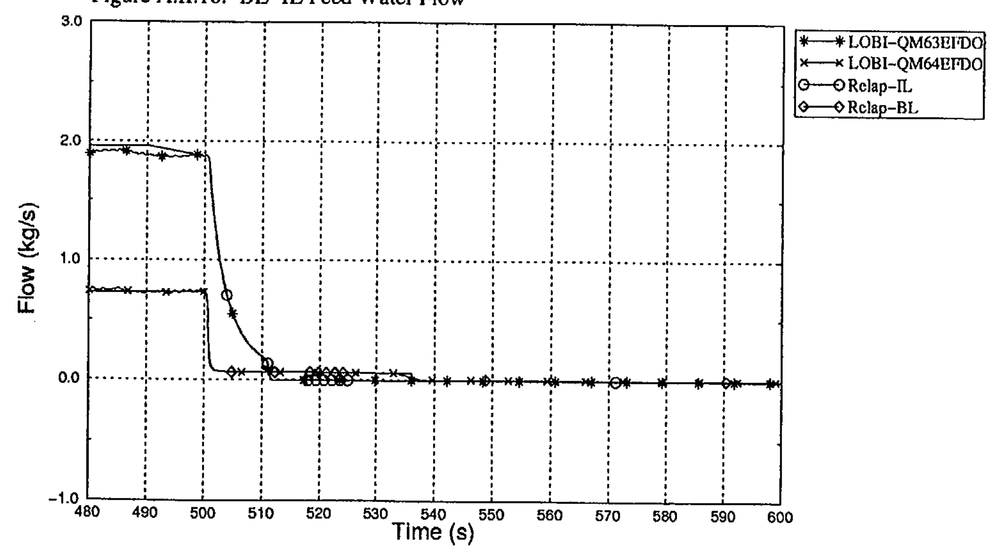


Figure A.II.16.–BL–IL Feed Water Flow



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.17.– IL–BL Steam Volumetric Flow

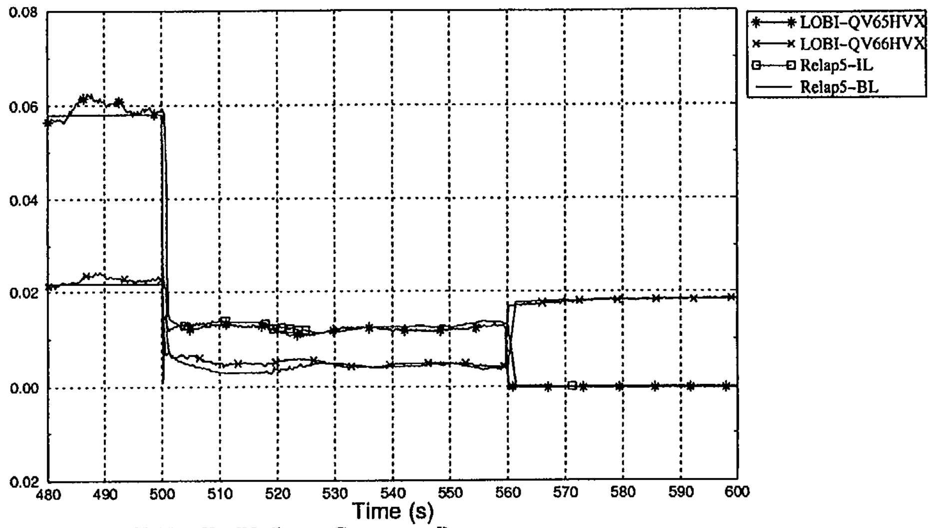


Figure A.II.18.– Volumetric Flow in Surge-Line

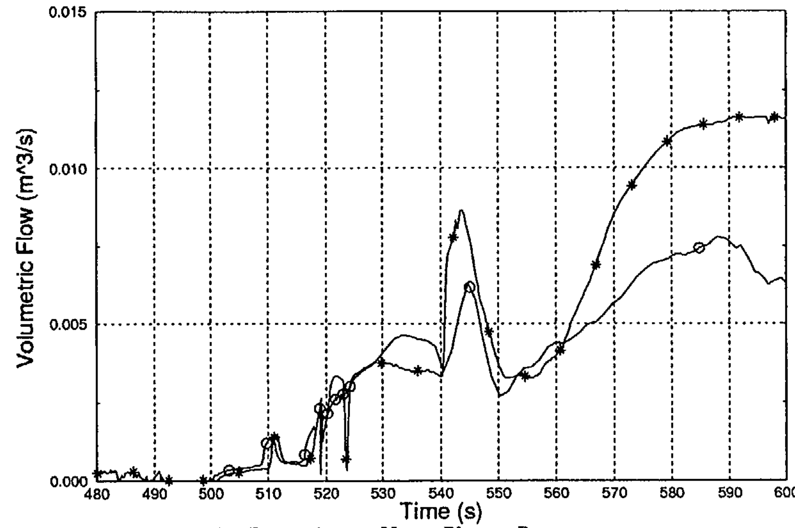


Figure A.II.19.– IL–BL Steam Generators Pressure

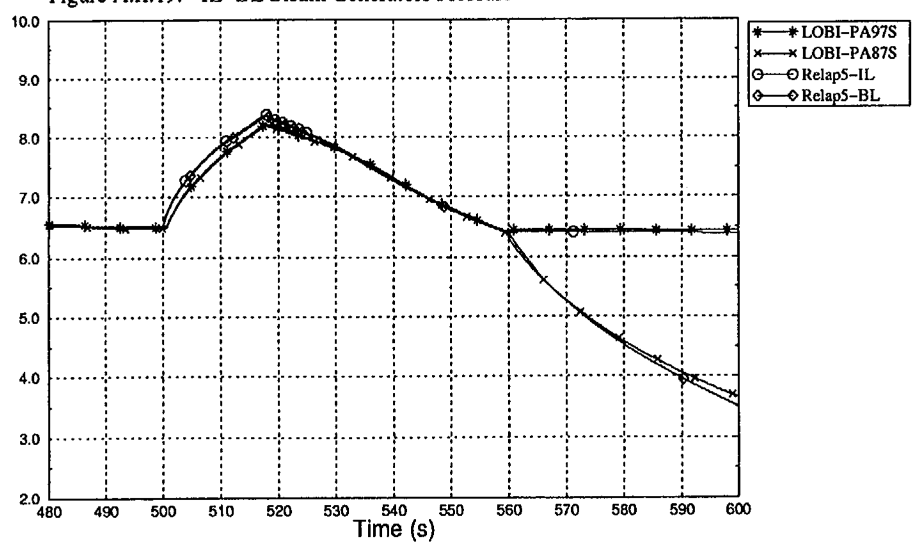
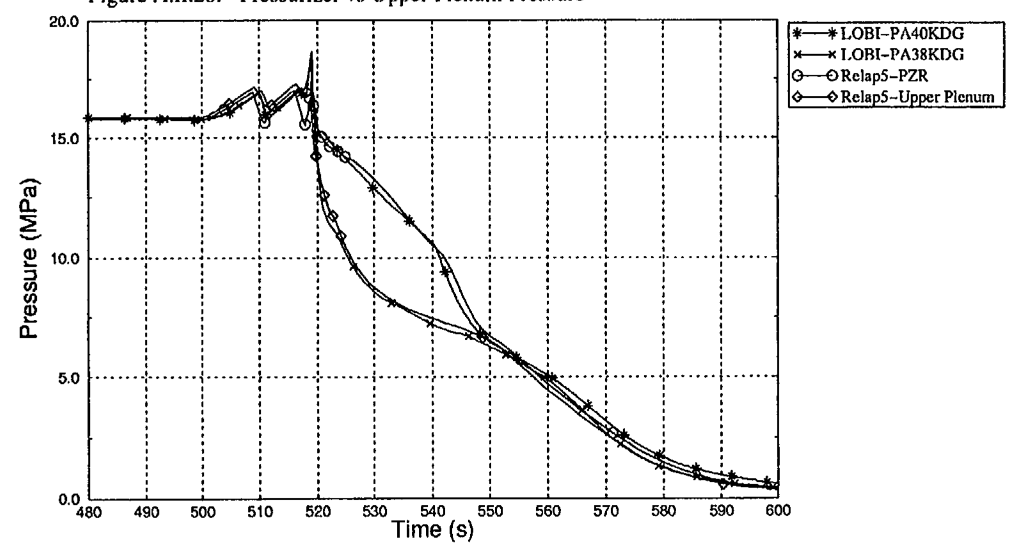


Figure A.II.20.– Pressurizer vs Upper Plenum Pressure



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.21.– Upper Plenum Vs SG BL Pressure

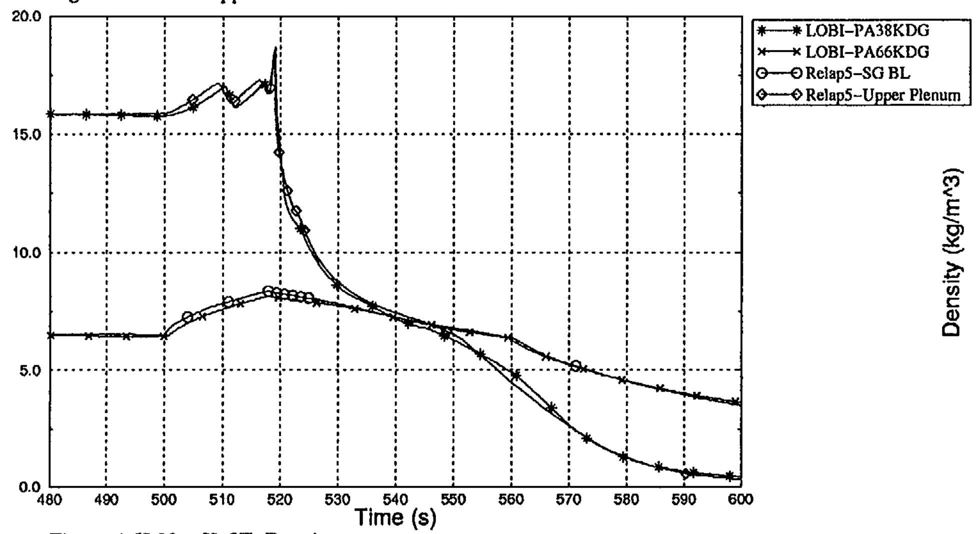


Figure A.II.22.– Core inlet Density

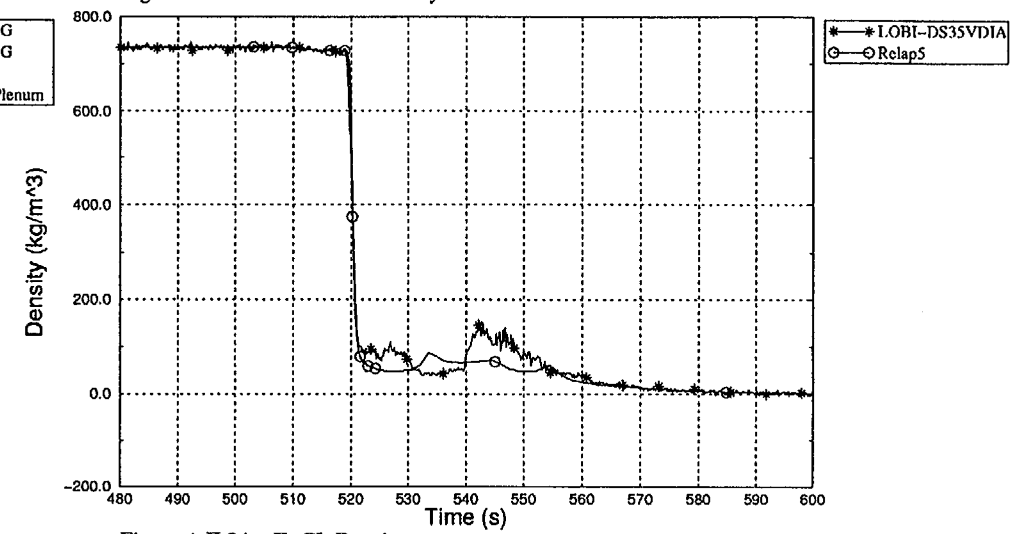


Figure A.II.23.– IL HL Density

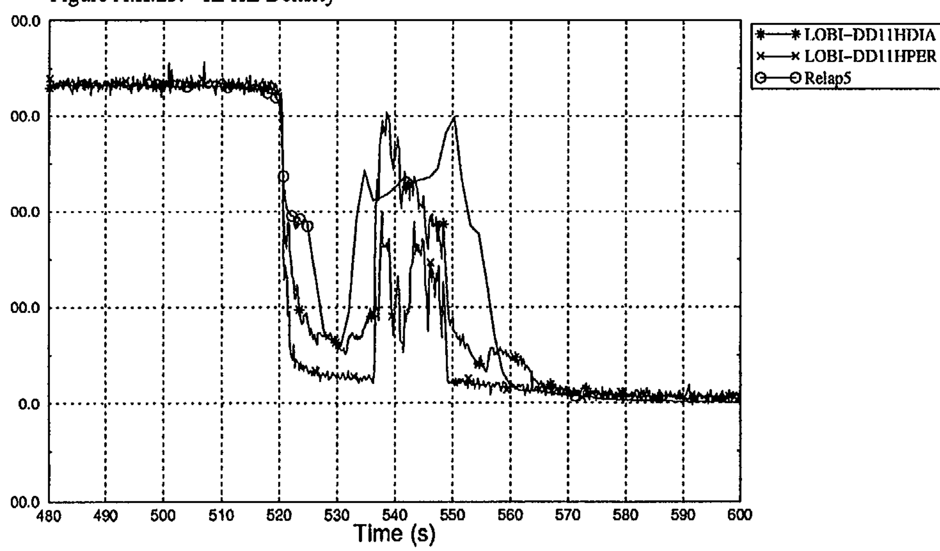
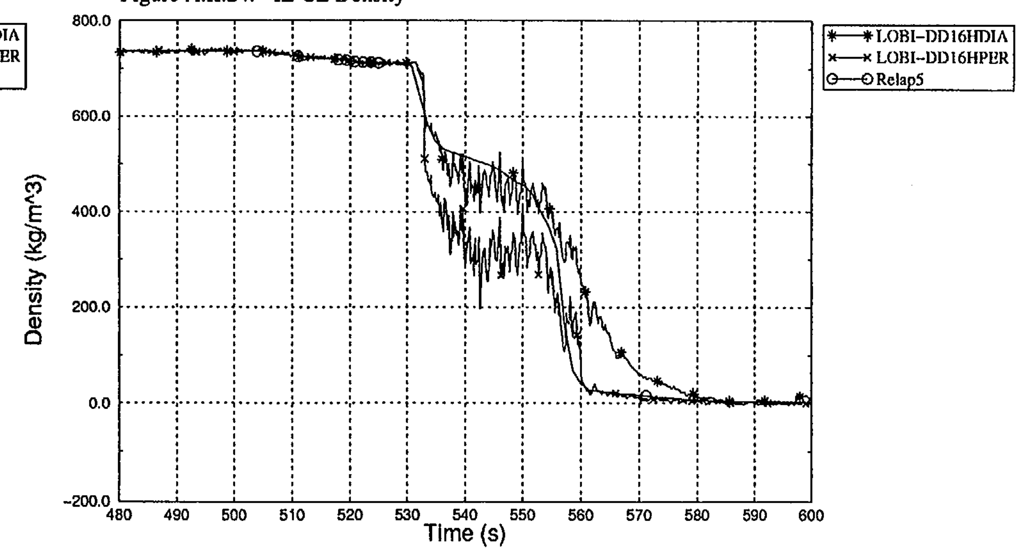


Figure A.II.24.– IL CL Density



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.25.– BL HL Density

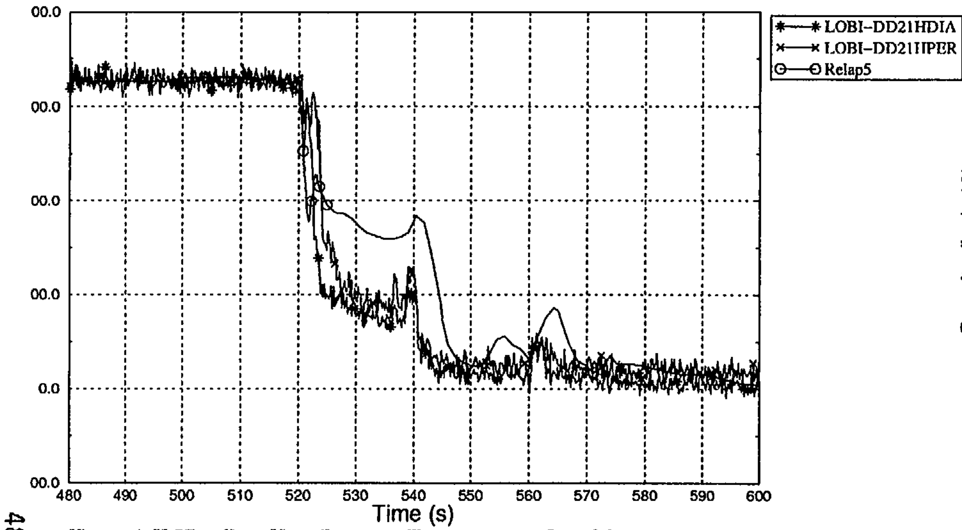


Figure A.II.26.– BL CL Density

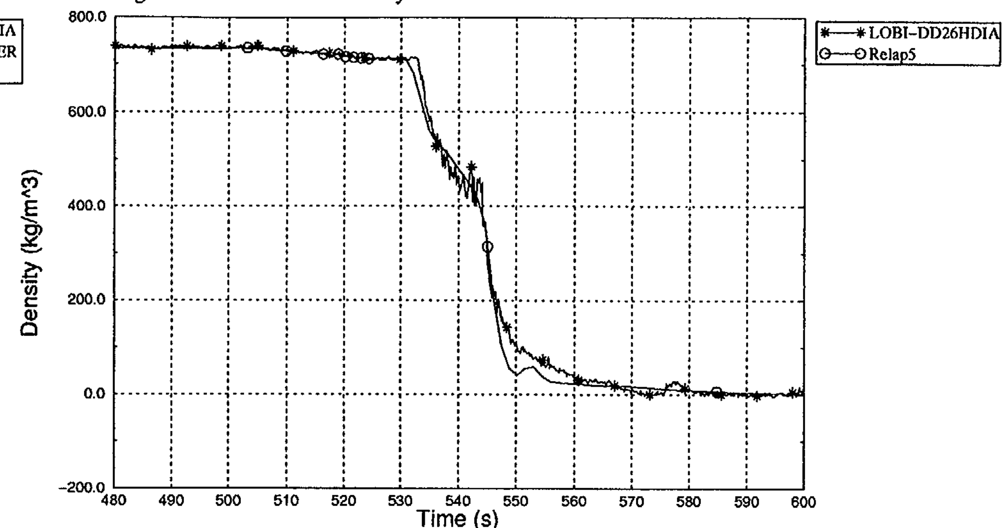


Figure A.II.27.– Core Heat Structure Temperature – Level 1

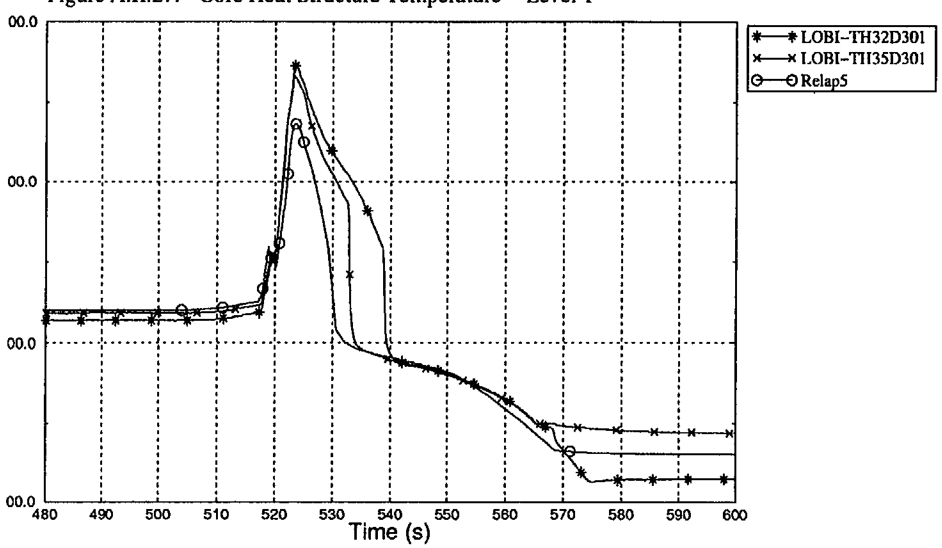
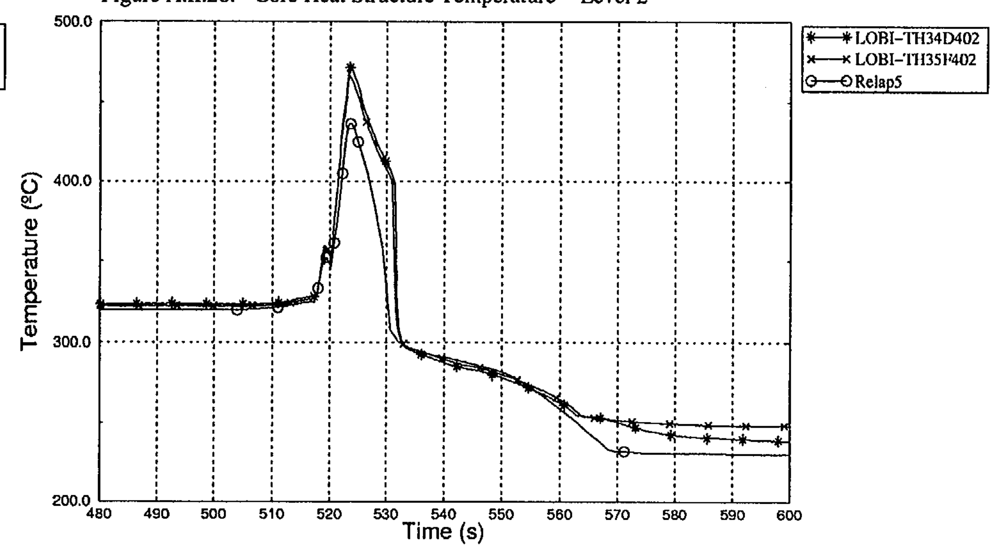


Figure A.II.28.– Core Heat Structure Temperature – Level 2



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.29.– Core Heat Structure Temperature – Level 3

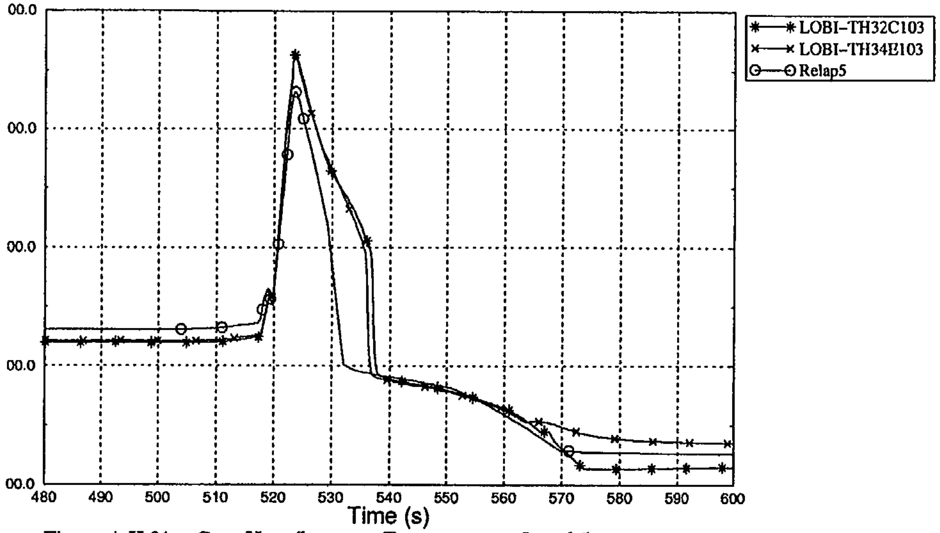


Figure A.II.30.– Core Heat Structure Temperature – Level 4

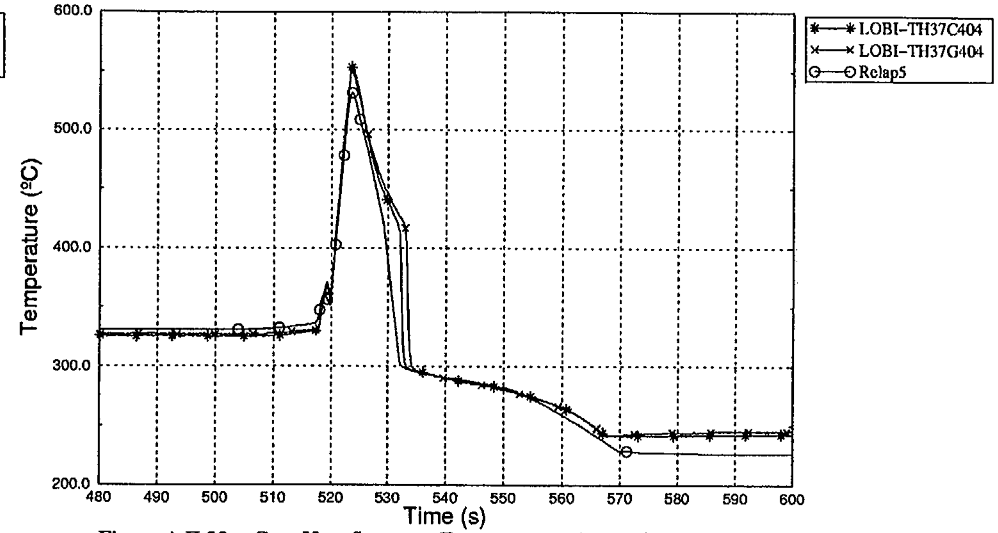


Figure A.II.31.– Core Heat Structure Temperature – Level 5

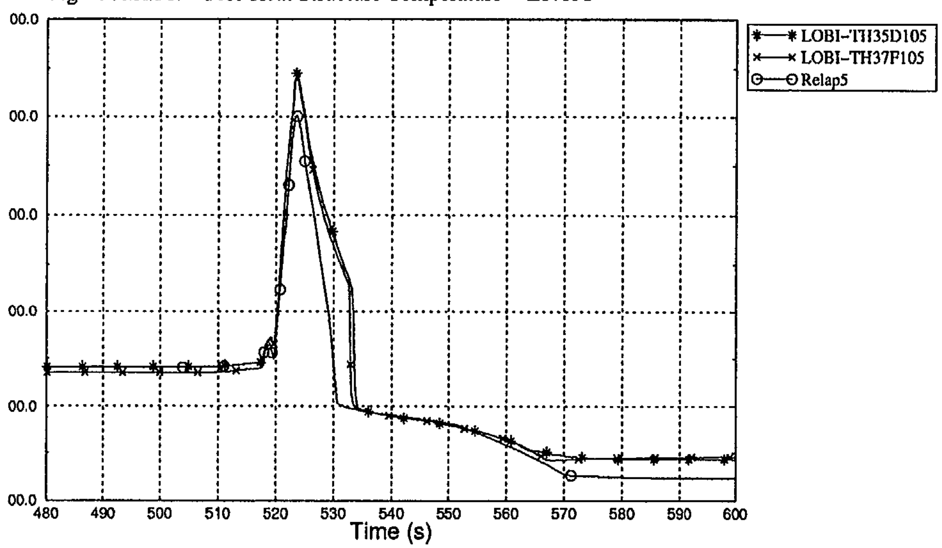
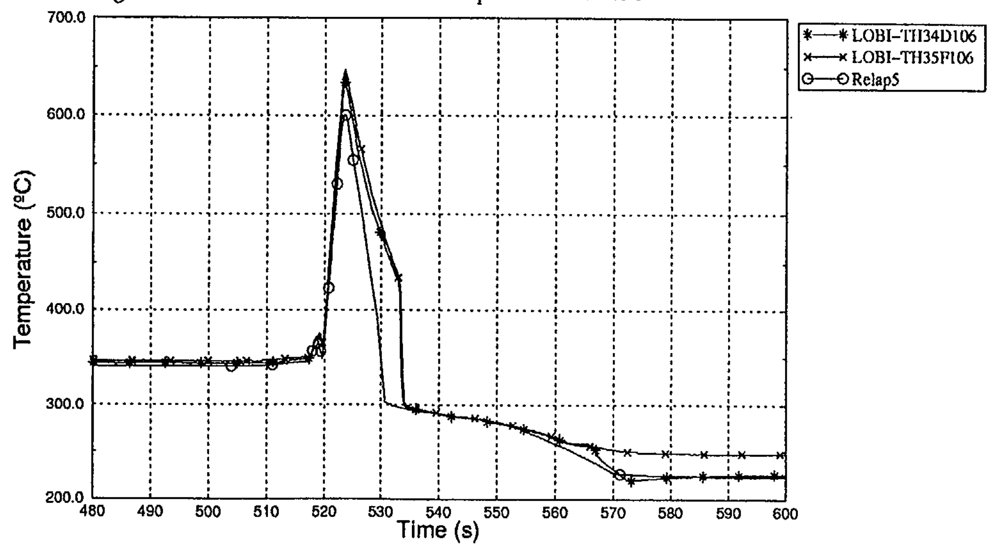


Figure A.II.32.– Core Heat Structure Temperature – Level 6



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.33.– Core Heat Structure Temperature – Level 7

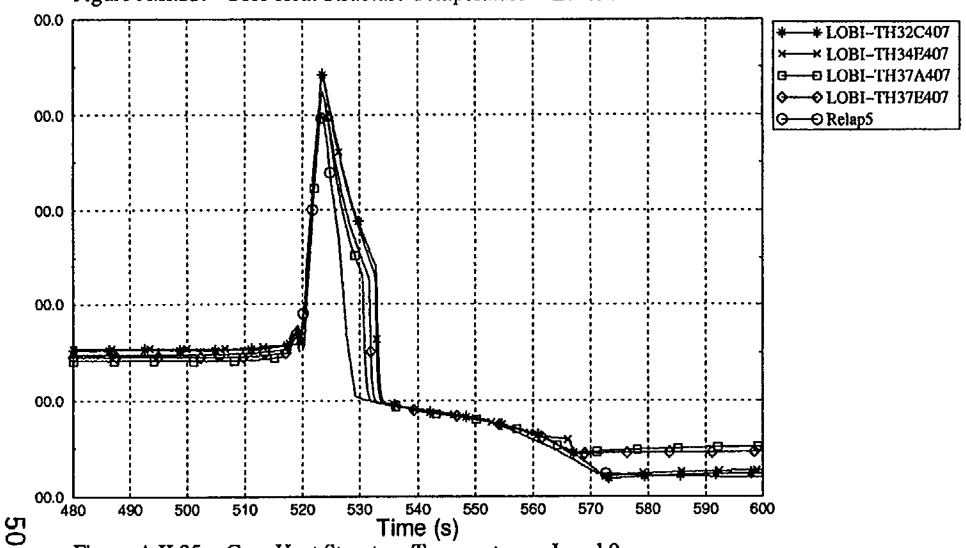


Figure A.II.34.– Core Heat Structure Temperature – Level 8

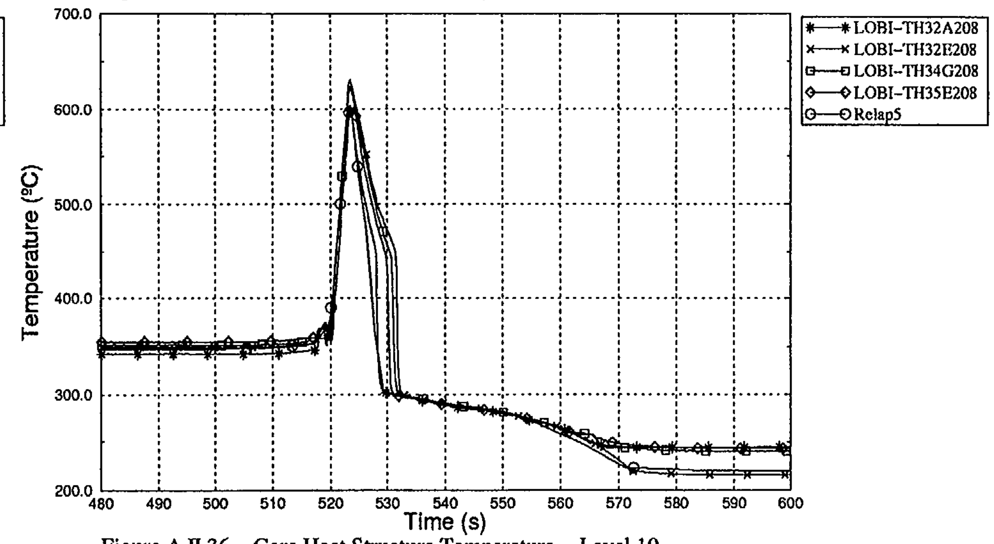


Figure A.II.35.– Core Heat Structure Temperature – Level 9

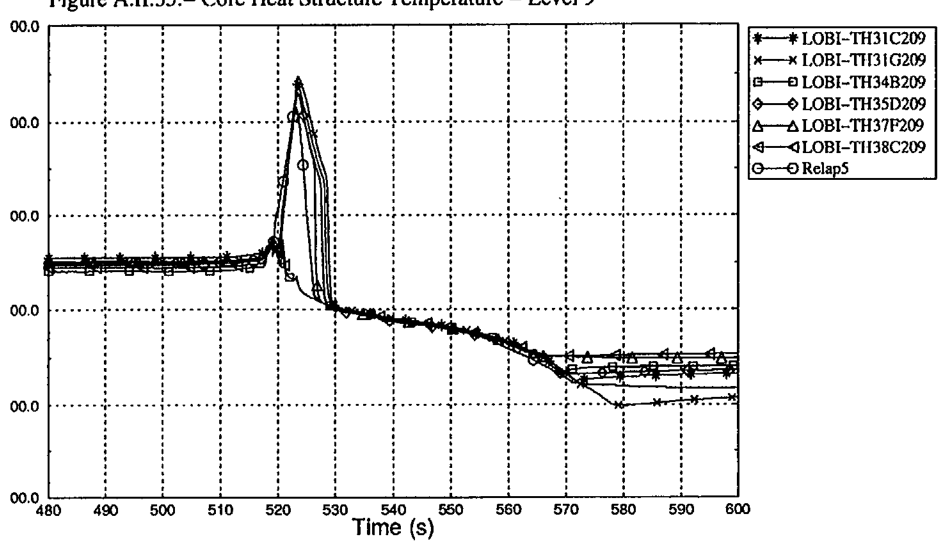
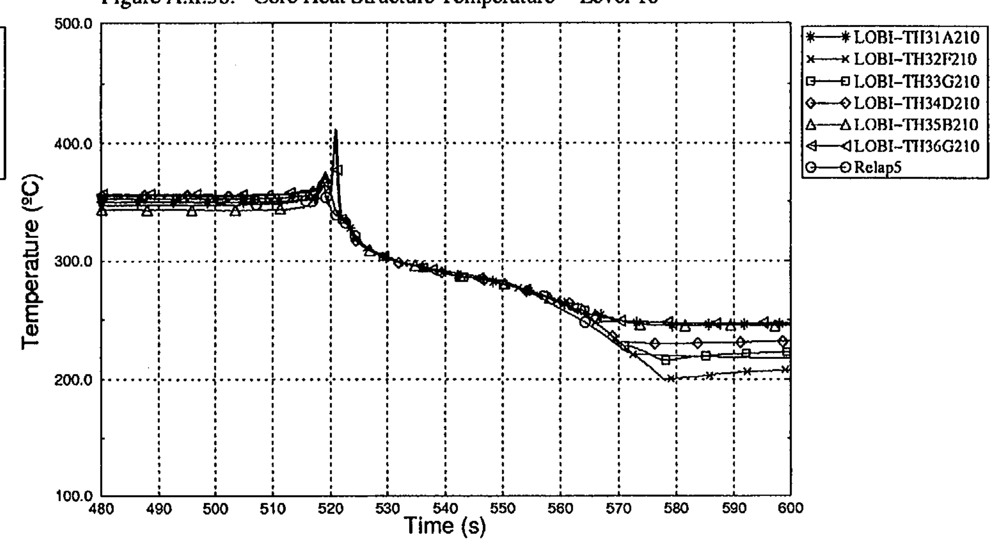


Figure A.II.36.– Core Heat Structure Temperature – Level 10



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.37.– Core Heat Structure Temperature – Level 11

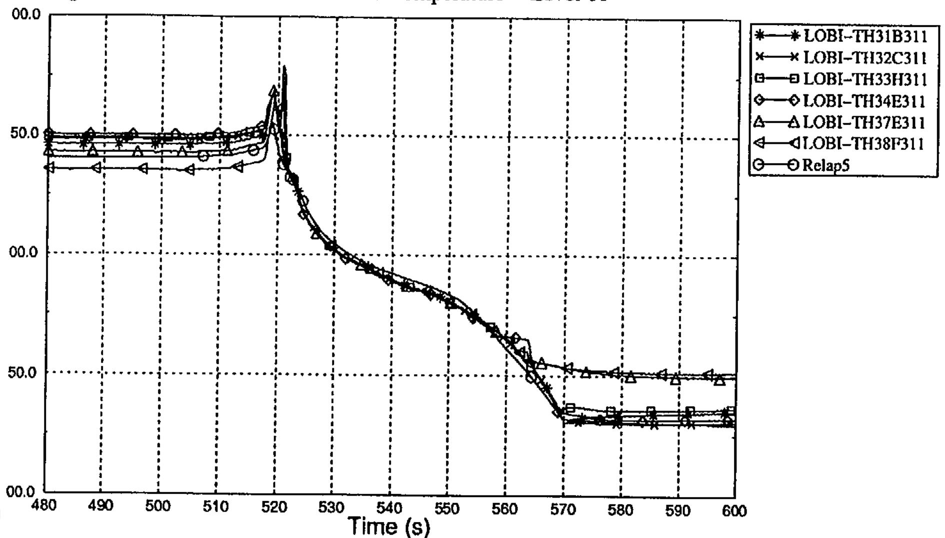


Figure A.II.38.– Core Heat Structure Temperature – Level 12

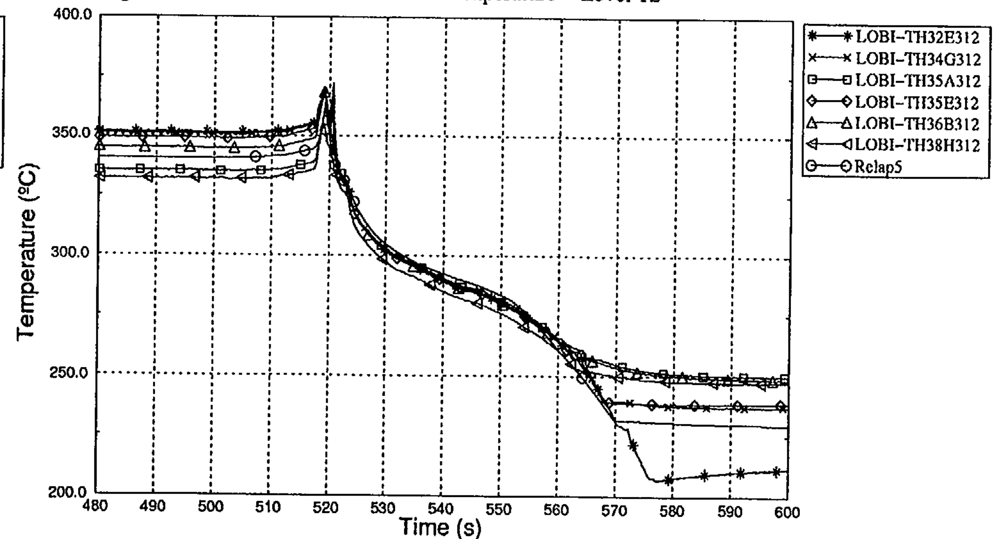


Figure A.II.39.– Pressurizer Collapsed Level

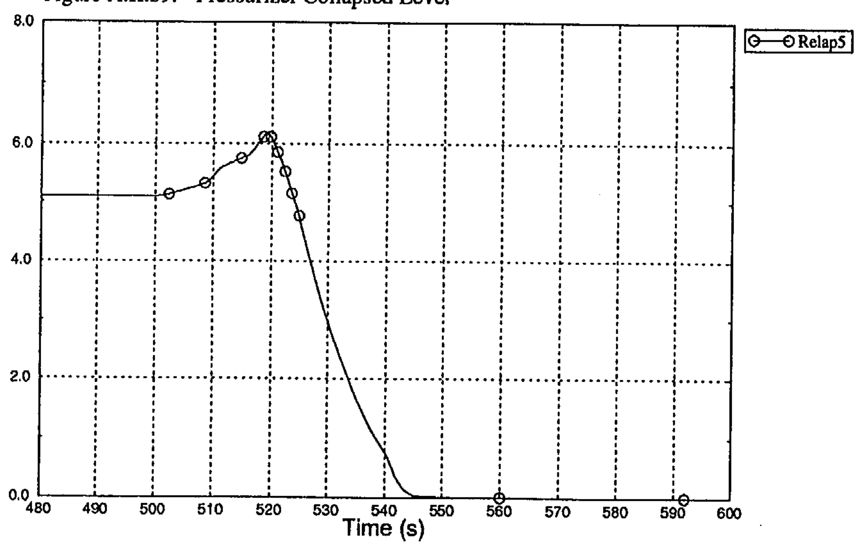
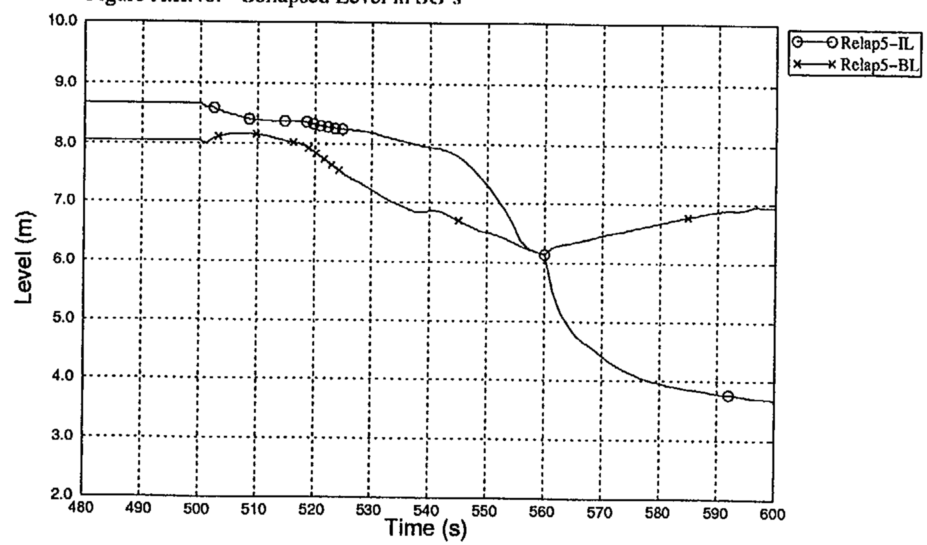


Figure A.II.40.– Collapsed Level in SG's



BT56 Transient Comparison LOBI/RELAP5. Base case

Figure A.II.41.– Rupture Disk Flow

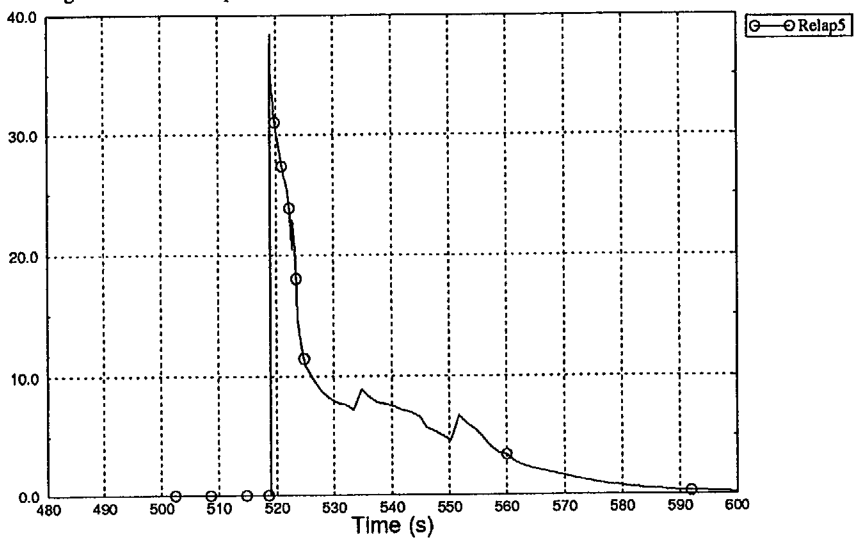
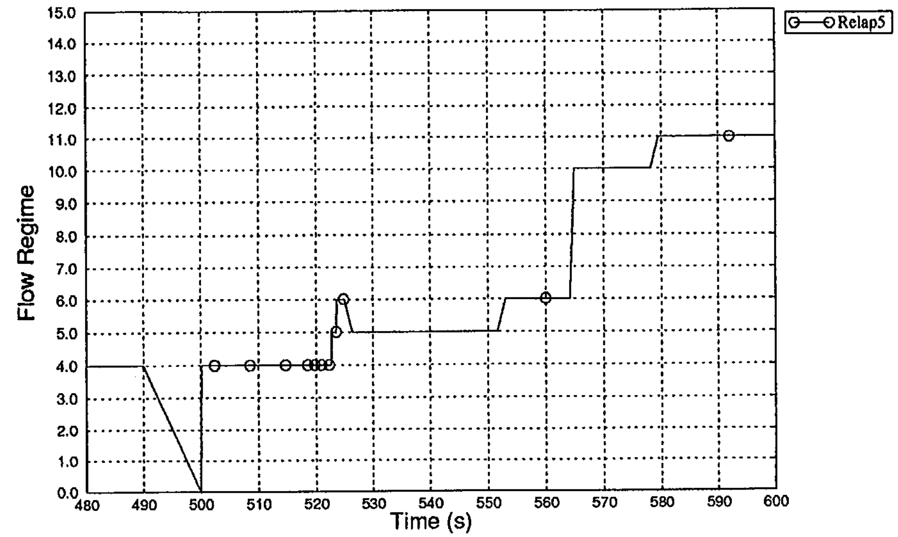


Figure A.II.42.– Rupture Disk Flow Regime



BT56 Transient Comparison LOBI/RELAP5. Sensitivity 1

Figure A.II.43– Pressure in Upper Plenum

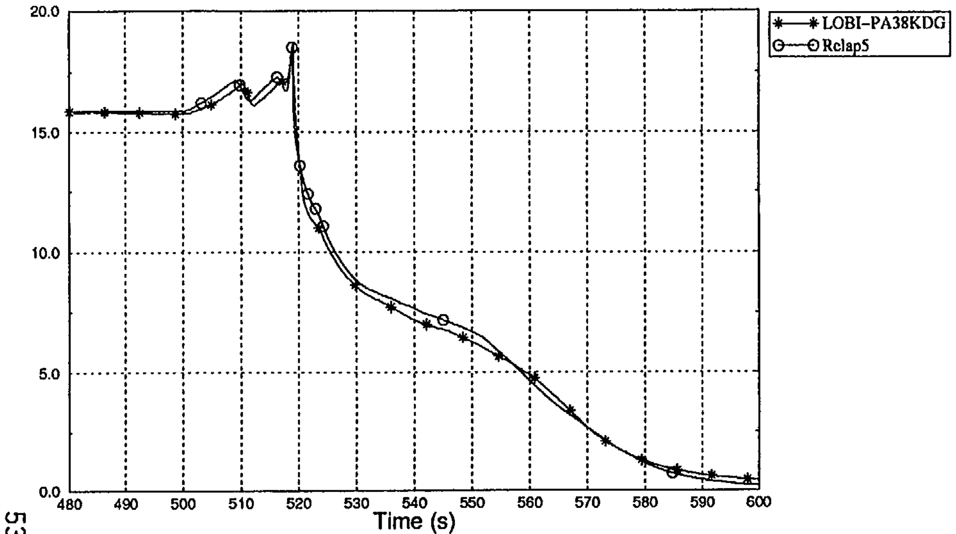


Figure A.II.44– Core Heat Structure Temperature – Level 4

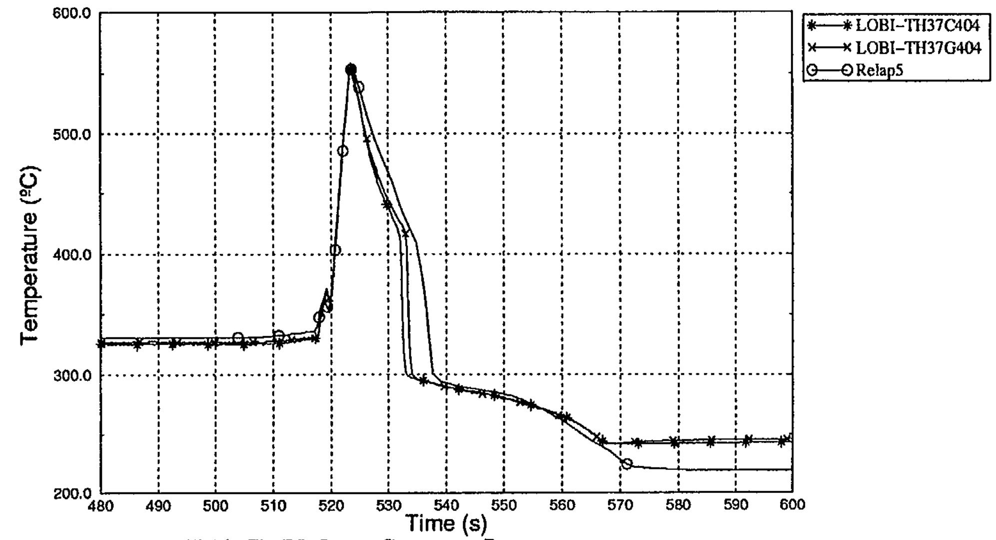


Figure A.II.45.– Temperature in CL BL

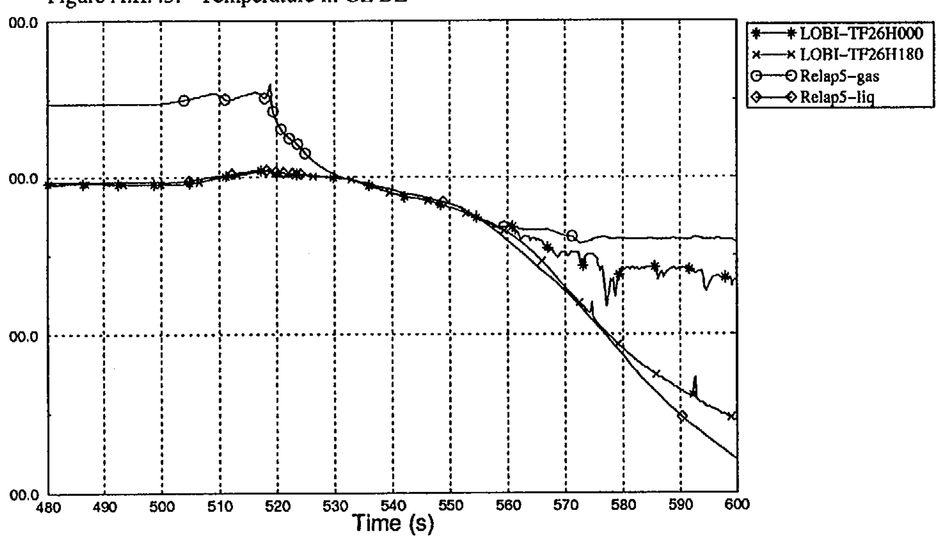
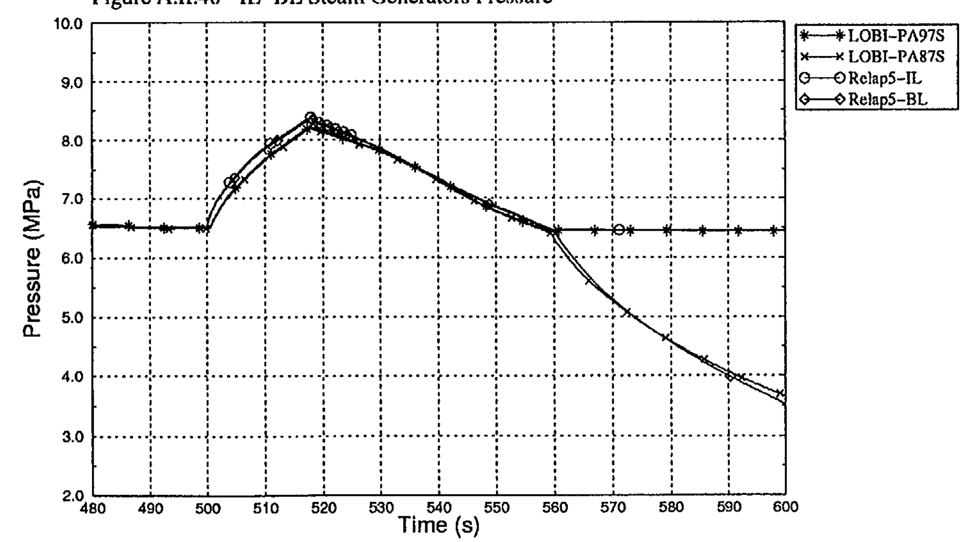


Figure A.II.46– IL–BL Steam Generators Pressure



BT56 Transient Comparison LOBI/RELAP5. Sensitivity 2

Figure A.II.47– Pressure in Upper Plenum

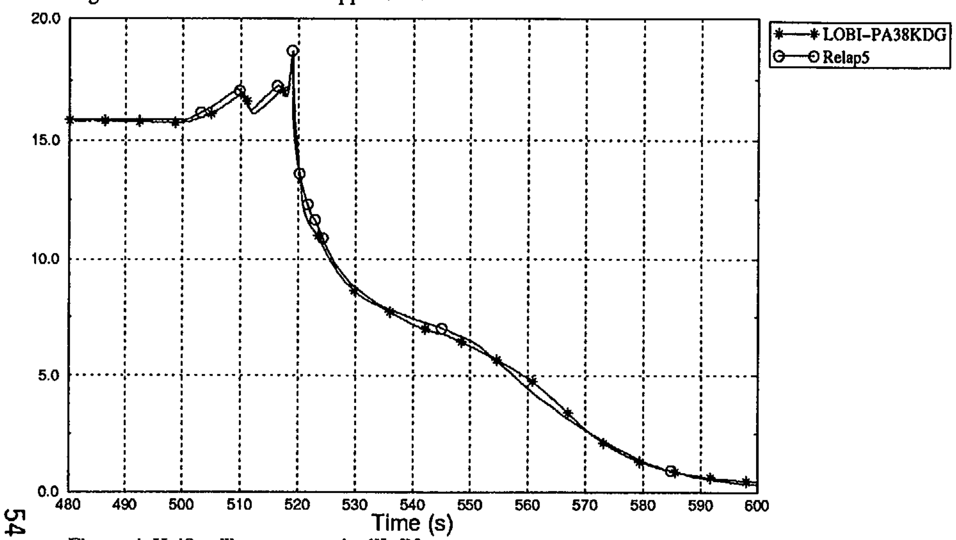


Figure A.II.48– Core Heat Structure Temperature – Level 4

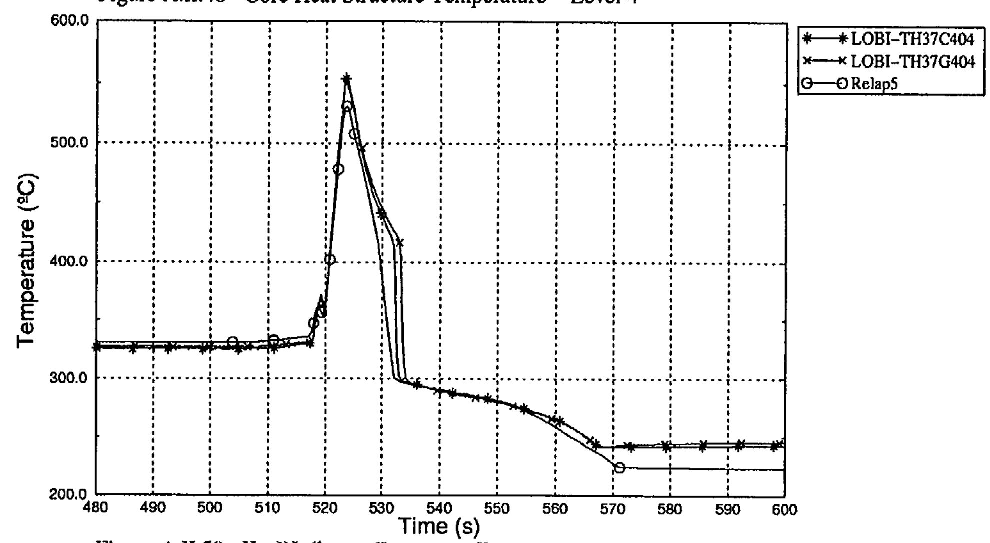


Figure A.II.49.– Temperature in CL BL

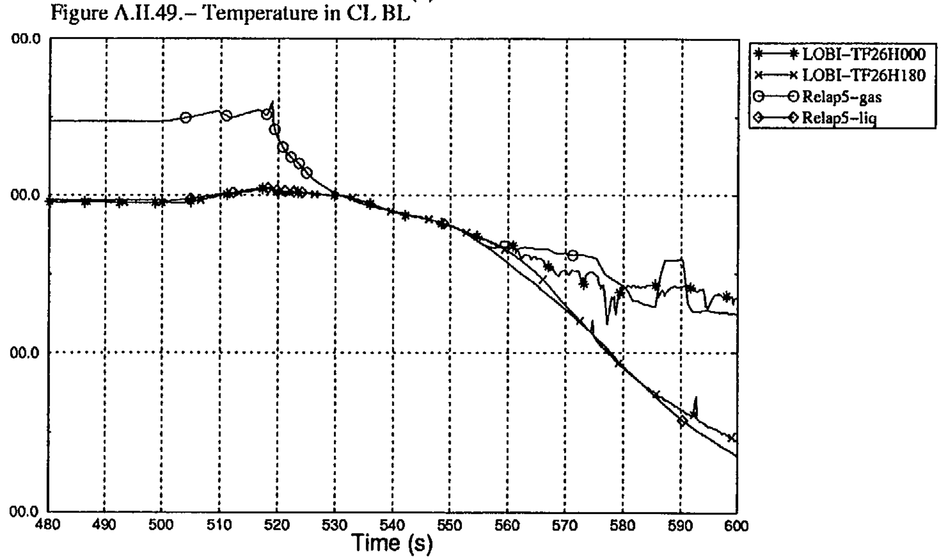
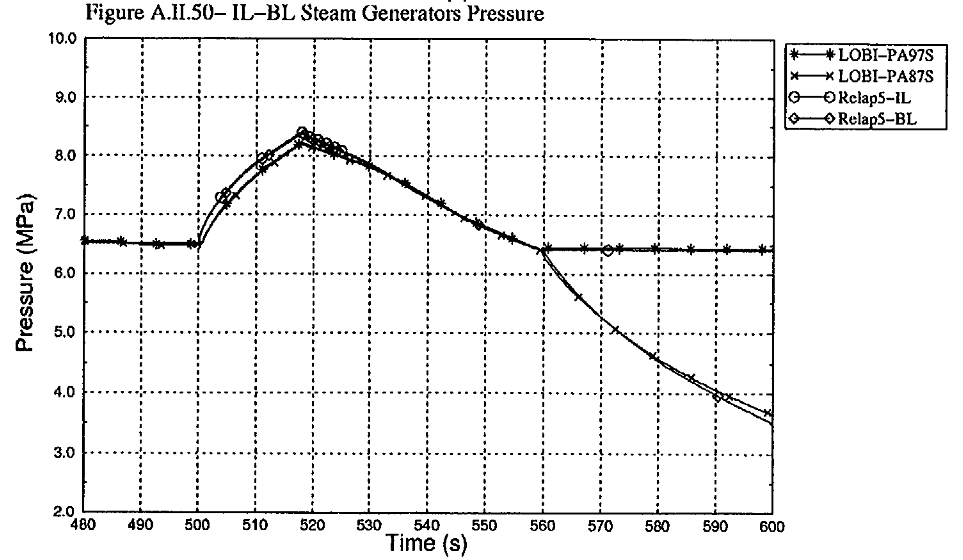


Figure A.II.50– IL–BL Steam Generators Pressure



BT56 Transient Comparison LOBI/RELAP5. Sensitivity 3

Figure A.II.51– Pressure in Upper Plenum

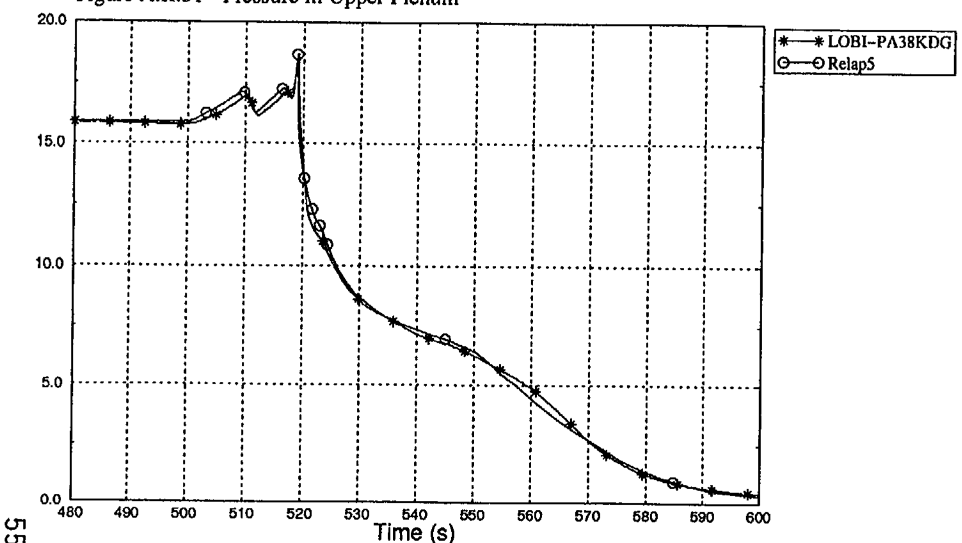


Figure A.II.52– Core Heat Structure Temperature – Level 4

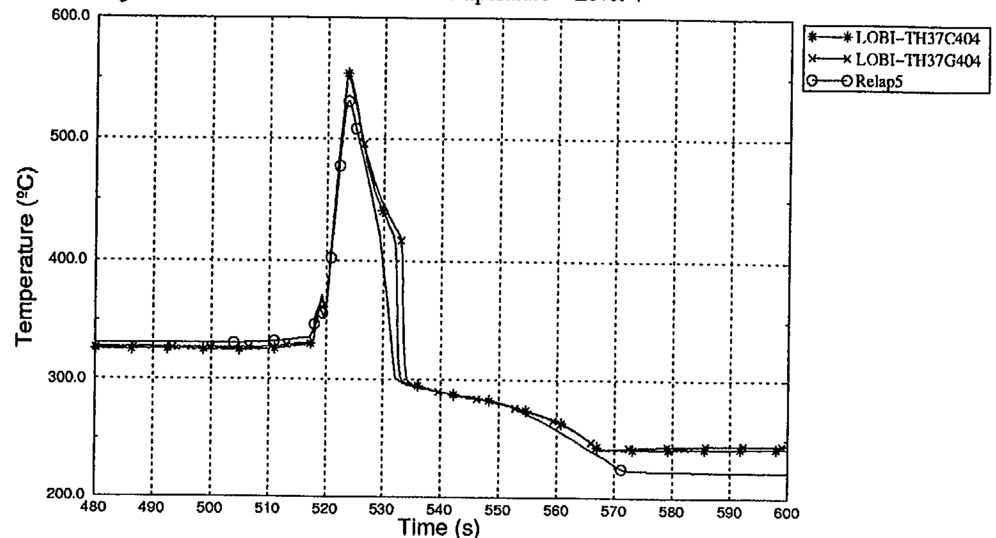


Figure A.II.53– IL–BL Steam Volumetric Flow

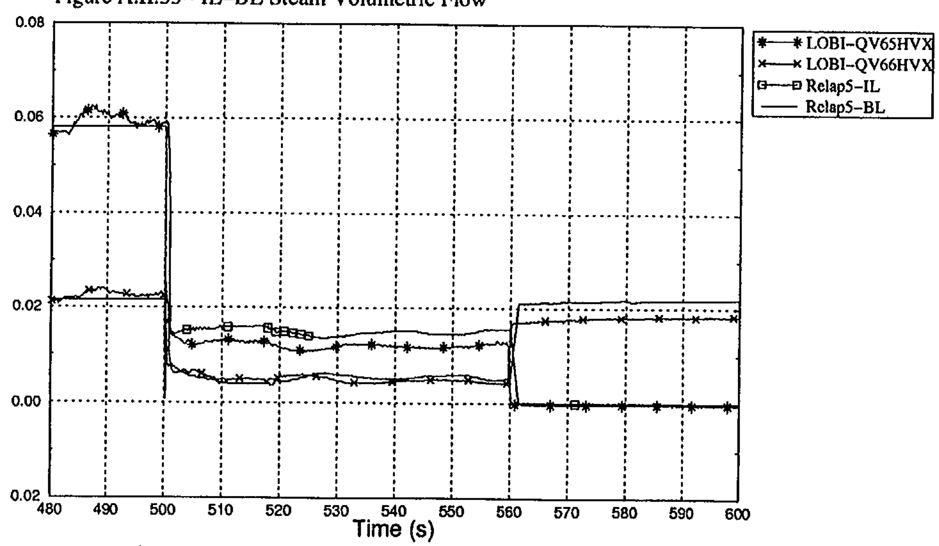
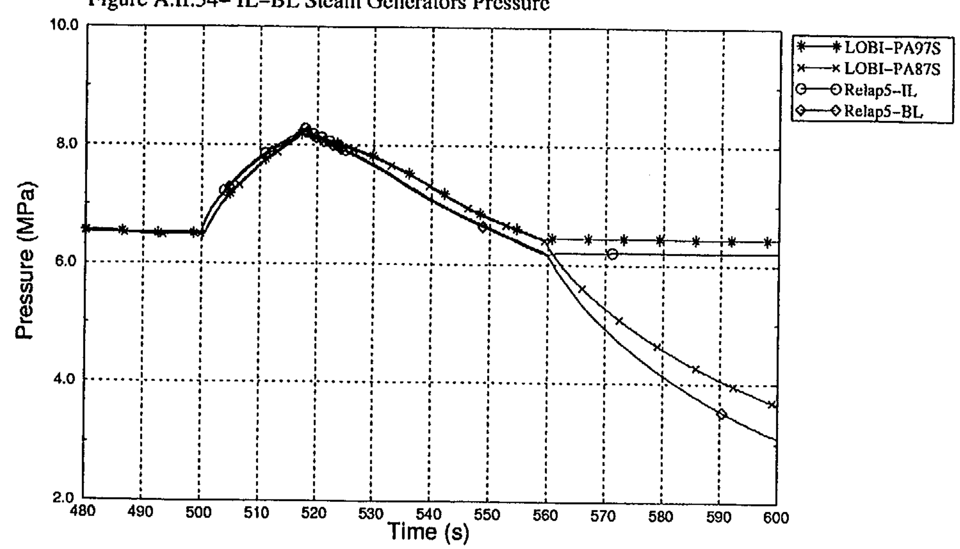


Figure A.II.54– IL–BL Steam Generators Pressure



NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/IA-0183

2. TITLE AND SUBTITLE

Analysis of the LOBI Experiment Test BT-56 Using the RELAP5/MOD3.2 Code

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	2000

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

J. Blanco, E. Moralo, R. Sanjuan, C. Gomez

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Union Fenosa Generacion S.A.
Central Nuclear Jose Cabrera
Madrid
SPAIN

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This document has been drawn up by UNION ELECTRICA FENOSA, S.A. (UNION FENOSA), which participates in the Code Application and Maintenance Project (CAMP) as a member of UNIDAD ELECTRICA, S.A. (UNESA) . The document presents a comparison between experimental data from the plant and data obtained by simulation with the RELAP5/MOD3.2 code of the experiment BT-56 which took place on 3rd July 1990 at the Joint Research Centre of the Commission of the European Communities, located in Ispra (Italy). The experiment originally scheduled for the LOBI-MOD2 facility was a Loss of Main Feedwater (LOFW), but multiple failures occurred, meaning that the evolution of the experiment deviated it from the scheduled sequence. The simulation was performed using the RELAP5/MOD3.2 code, on a Digital AlphaServer 2000 4/200 computer and the operating system DIGITAL-UNIX. The results show an acceptable agreement between the phenomenology observed during the experiment and that predicted by the code.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

RELAP5/MOD3
LOBI

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)
unclassified

(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

1975 25 2000
years