NUREG/IA-0250

International Agreement Report

Simulation of the F2.1 Experiment at PKL Facility using RELAP5/MOD3

Prepared by:
S. Carlos, J.F. Villanueva, S. Martorell,
V. Serradell

Universidad Politécnica de Valencia
Cami de Vera s/n
46022 Valencia, SPAIN

A. Calvo, NRC Project Manager

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

February 2011

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

Published by
U.S. Nuclear Regulatory Commission

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402-0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161-0002
   www.ntis.gov
   1-800-553-6847 or, locally, 703-605-6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: U.S. Nuclear Regulatory Commission
Office of Administration
Publications Branch
Washington, DC 20555-0001
E-mail: DISTRIBUTION.RESOURCE@NRC.GOV
Facsimile: 301-415-2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036-8002
www.ansi.org
212-642-4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG-XXXX) or agency contractors (NUREG/CR-XXXX), (2) proceedings of conferences (NUREG/CP-XXXX), (3) reports resulting from international agreements (NUREG/IA-XXXX), (4) brochures (NUREG/BR-XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG-0750).

NUREG/IA-0250

International Agreement Report

Simulation of the F2.1 Experiment at PKL Facility using RELAP5/MOD3

Prepared by:
S. Carlos, J.F. Villanueva, S. Martorell,
V. Serradell

Universidad Politécnica de Valencia
Cami de Vera s/n
46022 Valencia, SPAIN

A. Calvo, NRC Project Manager

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

February 2011

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

Published by
U.S. Nuclear Regulatory Commission

ABSTRACT

When a nuclear power plant is in shutdown conditions for refuelling, the reactor coolant system water level is reduced. This situation is known as mid-loop operation, and the residual heat removal (RHR) system is used to remove the decay power heat generated in the reactor core.

In mid-loop conditions, some accidental situations may occur with not a negligible contribution to the plant risk, and all involve the loss of the RHR system. Thus, to better understand the thermal-hydraulic processes following the loss of the RHR during shutdown, transients of this kind have been simulated using best-estimate codes, comparing their results against experimental data taken from different integral test facilities. This paper focuses on the simulation, using the best estimate code RELAP5/Mod 3.3, of the experiment F2.1 conducted at the PKL facility, within the OECD/PKL project. This experiment consists of the loss of the RHR system when the plant is in mid-loop conditions for refuelling and with the primary circuit closed. In the experimental series F2.1the physical phenomena to investigate are the mechanisms of heat removal in presence of nitrogen and the deboration in critical parts of the primary system.

Two experiments belonging to this experimental series have been performed. The simulations present differences in the initial plant coolant inventory and temperature in the pressurizer, F2.1RUN1 and F2.1RUN2, to asses the influence of these differences in the transient evolution.

FOREWORD

Extensive knowledge and techniques have been produced and made available in the field of thermal-hydraulic responses during reactor transients and accidents, and major system computer codes have achieved a high degree of maturity through extensive qualification, assessment and validation processes. Best-estimate analysis methods are increasingly used in licensing, replacing the traditional conservative approaches. Such methods include an assessment of the uncertainty of their results that must be taken into account when the safety acceptance criteria for the licensing analysis are verified.

Traditional agreements between the Nuclear Regulatory Commission of the United States of America (USNRC) and the Consejo de Seguridad Nuclear of Spain (CSN) in the area of nuclear safety research have given access to CSN to the NRC-developed best estimate thermalhydraulic codes RELAP5, TRAC-P, TRAC-B, and currently TRACE. These complex tools, suitable state-of-the-art application of current two-phase flow fluid mechanics techniques to light water nuclear power plants, allow a realistic representation and simulation of thermalhydraulic phenomena at normal and incidental operation of NPP. Owe to the huge required resources, qualification of these codes have been performed through international cooperation programs. USNRC CAMP program (Code Applications and Maintenance Program) represents the international framework for verification and validation of NRC TH codes, allowing to:

- Share experience on code errors and inadequacies, cooperating in resolution of deficiencies and maintaining a single, internationally recognized code version.
- Share user experience on code scaling, applicability, and uncertainty studies.
- Share a well documented code assessment data base.
- Share experience on full scale power plant safety-related analyses performed with codes (analyses of operating reactors, advanced light water reactors, transients, risk-dominant sequences, and accident management and operator procedures-related studies).
- Maintain and improve user expertise and guidelines for code applications.

Since 1984, when the first LOFT agreement was settled down, CSN has been promoting coordinated joint efforts with Spanish organizations, such as UNESA (the association of Spanish electric energy industry) as well as universities and engineering companies, in the aim of assimilating, applying, improving and helping the international community in the validation of these TH simulation codes1, within different periods of the associated national programs (e.g., CAMP-España). As a result of these actions, there is currently in Spain a good collection of productive plant models as well as a good selection of national experts in the application of TH simulation tools, with adequate TH knowledge and suitable experience on their use.

Many experimental facilities have contributed to the today's availability of a large thermal-hydraulic database (both separated and integral effect tests). However there is continued need for additional experimental work and code development and verification, in areas where no emphasis have been made along the past. On the basis of the SESAR/FAP2 reports "Nuclear Safety Research in OECD Countries:Major Facilities and Programmes at Risk" (SESAR/FAP, 2001) and its 2007 updated version "Support Facilities for Existing and Advanced Reactors (SFEAR) NEA/CSNI/R(2007)6", CSNI is promoting since 2001 several collaborative international actions in the area of experimental TH research. These reports presented some findings and recommendations to the CSNI, to sustain an adequate level of research, identifying a number of experimental facilities and programmes of potential interest for present or future international collaboration within the safety community during the coming decade.

CSN, as Spanish representative in CSNI, is involved in some of these research activities, helping in this international support of facilities and in the establishment of a large network of international collaborations. In

1 It's worth to note the emphasis made in the application to actual NPP incidents.

2 SESAR/FAP is the Senior Group of Experts on Nuclear Safety Research Facilities and Programmes of NEA Committee on the Safety of Nuclear Installations (CSNI).

the TH framework, most of these actions are either covering not enough investigated safety issues and phenomena (e.g., boron dilution, low power and shutdown conditions), or enlarging code validation and qualification data bases incorporating new information (e.g., multi-dimensional aspects, non-condensable gas effects). In particular, CSN is currently participating in the PKL and ROSA programmes.

The PKL is an important integral test facility operated by of AREVA-NP in Erlangen (Germany), and designed to investigate thermal-hydraulic response of a four-loop Siemens designed PWR. Experiments performed during the PKL/OECD program have been focused on the issues:

- Boron dilution events after small-break loss of coolant accidents.
- Loss of residual heat removal during mid-loop operation (both with closed and open reactor coolant system.

ROSA/LSTF of Japan Atomic Energy Research Institute (JAERI) is an integral test facility designed to simulate a 1100 MWe four-loop Westinghouse-type PWR, by two loops at full-height and 1/48 volumetric scaling to better simulate thermal-hydraulic responses in large-scale components. The ROSA/OECD project has investigated issues in thermal-hydraulics analyses relevant to water reactor safety, focusing on the verification of models and simulation methods for complex phenomena that can occur during reactor transients and accidents such as:

- Temperature stratification and coolant mixing during ECCS coolant injection
- Water hammer-like phenomena
- ATWS
- Natural circulation with super-heated steam
- Primary cooling through SG depressurization
- Pressure vessel upper-head and bottom break LOCA

This overall CSN involvement in different international TH programmes has outlined the scope of the new period of CAMP-España activities focused on:

- Analysis, simulation and investigation of specific safety aspects of PKL/OECD and ROSA/OECD experiments.
- Analysis of applicability and/or extension of the results and knowledge acquired in these projects to the safety, operation or availability of the Spanish nuclear power plants.

Both objectives are carried out by simulating experiments and plant application with the last available versions of NRC TH codes (RELAP5 and TRACE). A CAMP in-kind contribution is aimed as end result of both types of studies.

Development of these activities, technically and financially supported by CSN, is being carried out by 5 different national research groups (Technical Universities of Madrid, Valencia and Cataluña). On the whole, CSN is seeking to assure and to maintain the capability of the national groups with experience in the thermal hydraulics analysis of accidents of the Spanish nuclear power plants.

Francisco Fernández Moreno, Commissioner
Consejo de Seguridad Nuclear (CSN)

CONTENTS

Figures

Tables

Page

EXECUTIVE SUMMARY

When a Pressurized Water Reactor is in shutdown conditions for refuelling, and to perform steam generator U-tubes and reactor coolant pump maintenance activities, the reactor coolant system water level is reduced to a height lower than the top of the hot leg pipe. Under these conditions, it is said that the plant is in mid-loop operation. In this mode of operation, the residual heat removal system is used to remove the decay power heat generated in the reactor core. Some accidental situations may occur in mid-loop conditions that have a significant contribution to the plant risk, and all involve the loss of the RHR system. In fact, the loss of RHR has been experienced several times in pressurized water reactor plants. For these reasons, the study of transients in mid-loop operation is of great interest to analyze the plant safety.

To better understand the thermal-hydraulic processes following the loss of the RHR during shutdown, transients of this kind have been simulated using best-estimate codes such as RELAP5 or CATHARE etc. Such codes have initially been developed to simulate full power operation conditions, which are different physical conditions from the ones faced in mid-loop operation mode. Thus, to assess the capability of best estimate codes in simulating the physical phenomena under mid-loop conditions it is necessary to compare the code calculations with data obtained from experiments simulating such type of conditions.

The work presented in this paper is focused on the simulation, using the best estimate code RELAP5/Mod 3.3, of the experiment F2.1 conducted at the PKL facility. The experiment F2.1 belongs to an experimental series established in the OECD/PKL program devoted to the study of boron dilution sequences and the effect of the primary coolant inventory in shutdown conditions, when the RHR system is lost and the plant is in mid-loop conditions for refuelling and with the primary circuit closed. F2.1 experiment is composed by three runs which differences in the initial plant conditions. In this work two runs have been simulated that differ in the amount of water inside the reactor coolant circuit and in the pressurizer temperature. Thus, in experiment F2.1 Run1 the primary level is initially fixed to ¾ loop and the pressurizer temperature ranges into 150-170ºC, while in experiment F2.1 Run2, the coolant inventory is reduced to the lower edge of the primary circuit and the pressurizer temperature ranges between 25-45 ºC.

From the results obtained in both simulations it can be concluded that, in general, the calculations reproduce the physical phenomena observed in the experimental data, although some differences are observed, especially in the reactor coolant circuit mass distribution. In addition when comparing the calculations obtained in F2.1 Run1 and F2.1 Run2, it is observed that small differences in the plant initial conditions influences the plant behaviour after the RHR is lost, in fact for both cases the plant reaches a stable situation but, as it happens in the experiment, the plant stabilization values are different in each case.

ACKNOWLEDGEMENTS

This work contains findings that were produced within the OECD-NEA/PKL Project. The authors are grateful to the Management Board of the PKL Project for their consent to this publication, and thank the Spanish Nuclear Regulatory Body (CSN) for the technical and financial support under the agreement STN/1388/05/748.

ABBREVIATIONS

ACC	Accumulator
ATWS	Anticipated transient without scram
C	Total number of volumes
CAMP	Code Assessment and Management Program
CL	Cold Leg
CPU	Execution time (s)
CSN	Consejo de Seguridad Nuclear (Spanish nuclear regulatory commission)
DT	Total number of time steps
ECCS	Emergency Core Cooling System
HL	Hot Leg
HPIS	High Pressure injection system
JAERI	Japan Atomic Energy Research Institute
kg	kilogram(s)
kg/s	kilograms per second
LBLOCA	Large Break Loss of Coolant Analysis
LPIS	Low Pressure injection system
m	meter(s)
Pa	Pascal
°C	degrees Celsius
°K	degrees Kelvin
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NEA	Nuclear Energy Agency
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
OECD	Organisation for Economic Cooperation and Development
ppm	parts per million
PWR	Pressurized Water Reactor
RCL	Reactor Coolant Line
RCS	Reactor Coolant System
RELAP	Reactor Excursion and Leak Analysis Program
RHR	Residual Heat Removal System
RT	Transient time (s)
s	second(s)
SBLOCA	Small Break Loss of Coolant Analysis
SG	Steam Generator
TH	Thermal-hydraulic
TS	Maximum time step (s)
UNESA	Asociación Española de la Industria Eléctrica

1. INTRODUCTION

When a Pressurized Water Reactor (PWR) is in shutdown conditions for refuelling, and to perform steam generator (SG) U-tubes and reactor coolant pump maintenance activities, the reactor coolant system water level is reduced to a height lower than the top of the hot leg pipe [1]. Under these conditions, it is said that the plant is in mid-loop operation. In this mode of operation, the residual heat removal (RHR) system is used to remove the decay power heat generated in the reactor core.

Some accidental situations may occur in mid-loop conditions that have a significant contribution to the plant risk, and all involve the loss of the RHR system. The three major causes of a loss of the RHR system are: A loss of reactor coolant system (RCS) inventory, a loss of RHR flow and a loss of support systems [2]. Moreover, the loss of RHR has been experienced several times in pressurized water reactor (PWR) plants as, for example, in Diablo Canyon Unit 2 [3] and in Voltge Unit 1 [4]. The causes of the loss of the RHR in those plants were a failure in the RHR pump and a loss of off-site power, respectively.

For the reasons above exposed, the study of transients in mid-loop operation has become of great interest during the last decades [1, 5, 6, 7, 8, 9, 10, 11].

To better understand the thermal-hydraulic processes following the loss of the RHR during shutdown, transients of this kind have been simulated using best-estimate codes such as RELAP-5 [5, 6; 9; 10] or CATHARE [7] etc. Such codes have initially been developed to simulate full power operation conditions, which have different physical conditions from the ones one faces in mid-loop operation mode. Thus, to assess the capability of best estimate codes in simulating the physical phenomena under mid-loop conditions it is necessary to compare the code calculation with data obtained from experiments simulating such type of conditions.

Different experiments simulating transients in shutdown conditions have been conducted at different integral facilities, such as ROSA, BETHSY and PKL. This paper focuses on the simulation, using the best estimate code RELAP5/Mod 3.3 [12], of the experiment F2.1 conducted at the PKL facility. The experiment F2.1 belongs to an experimental series established in the OECD/PKL program devoted to the study of boron dilution sequences and the effect of the primary coolant inventory in shutdown conditions [11], when the RHR system is lost and the plant is in mid-loop conditions for refuelling and with the primary circuit closed.

The aim of the study presented in this paper is to determine the physical phenomena observed in experiment F2.1 that are well predicted by RELAP5-Mod3.3 code, as well as to identify those processes that cannot be reproduced with the used code and model.

In particular, in this experiment the physical phenomena to investigate are the mechanisms of heat removal in presence of nitrogen and the deboration in certain parts of the primary system.

The rest of the paper is organized as follows: The PKL facility is briefly described in Section 2. Section 3 is devoted to introduce both experiments F2.1RUN1 and F2.1RUN2, section 4 is devoted to describe the RELAP5/Mod 3.3 model for the PKL facility used to simulate the experiment. In section 5, the main results obtained from both simulations are presented and compared with the experimental data. Finally, the main conclusions of this study are summarized in section 6.

2. PKL FACILITY DESCRIPTION

The PKL test facility represents a typical 1300MWe Siemens/KWu design PWR with a volume and power scale of 1:145, while all the components height on the primary and secondary side correspond to real plant dimensions. It models the entire primary system and the relevant parts of the secondary side. In order to investigate the influence of non symmetrical boundary conditions on the system behaviour, PKL facility is equipped with four primary loops symmetrically arranged around the reactor pressurized vessel. Each loop contains a reactor coolant pump and a steam generator [11].

The facility also models all the important safety and auxiliary systems as eight accumulators, one in each of the hot legs and one in each of the cold legs, four independent injections from the high and low pressure injection system, the residual heat removal system and the pressure control in the presssurizer. Figure 1 shows an overview of PKL test facility.

Three experimental programs have been conducted at PKL facility. Programs PKL I and PKL II, focused on the study of Large Break Loss of Coolant Accidents (LBLOCAs) and Small Break Loss of Coolant Accidents (SBLOCAs) with the objective of best estimate codes test and validation. PKL III program started in 1986 [11], with the objective of studying different transients with and without LOCAs. The PKL tests results have also been used for preparation and verification of procedures described in the operating manuals and for answering questions posed by regulatory bodies.

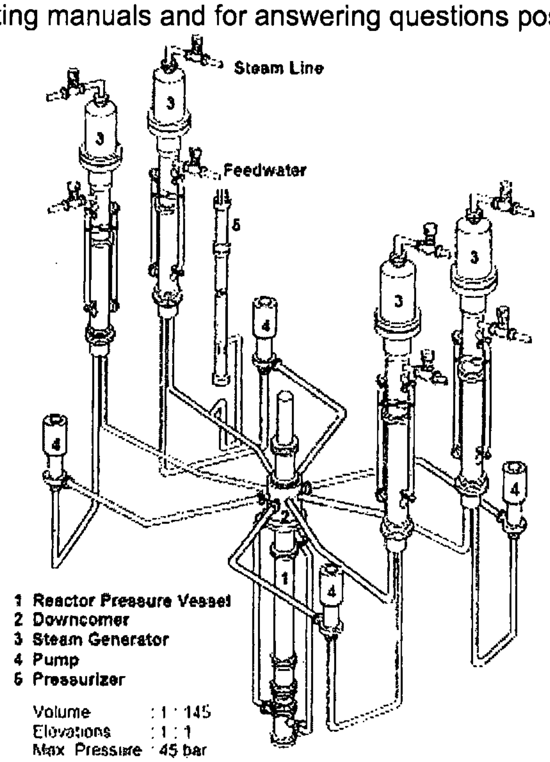

Figure 1. PKL facility.

In particular, PKL III F series include investigations on the inadvertent boron dilution events and on the effect of the primary coolant inventory in transients under shutdown conditions. The importance of the boron dilution events lies on the possibility that low borated water may enter into the reactor vessel, and this may lead to a local reactivity insertion event in the reactor core.

Moreover, the initial situation of the plant may affect the plant evolution towards a safe situation when the RHR is lost in mid-loop conditions. Thus, series F2.1 consist of three experiments with differences in the reactor coolant inventory and in the pressurizer initial conditions. In this work two of these experiments are simulated with different amount of water in the primary and different temperature in the pressurizer.

3. F2.1 TRANSIENT DESCRIPTION

In shutdown conditions, one of the most important initiating events is the loss of the RHR system. When this happens, other alternatives for heat removal should be available to cooldown the reactor core. One possibility consists of using the steam generators as heat sink, by means of using the cooling capacity of the reflux condensation. This phenomenon takes place in the U-tubes of the steam generators with the secondary side full of water. The vapour generated in the core comes into the U-tubes and transfers the heat to the steam generator secondary side. This vapour condenses in the U-tubes and comes back again into the reactor vessel cooling the system.

The experimental series F2.1 conducted in the PKL facility consists of the loss of the RHR system when the plant has been shut down for refuelling, with the primary circuit closed and partially filled. Two of the steam generators (SG1 and SG2) have their secondary sides full of water and the other two (SG3 and SG4) are full of nitrogen. This paper focuses on the simulation of experiment F2.1 Run1 and F2.1 Run2. Both experiments have differences in the plant initial conditions. In experiment F2.1 Run1 the primary level is initially fixed to ¾ loop and the pressurizer wall temperature ranges into 136-156°C (hot pressurizer) in the N_2 filled region, while in experiment F2.1 Run2, the coolant inventory is reduced to the lower edge of the primary circuit and the pressurizer wall temperature ranges between 30-39 °C (cold pressurizer) in the N_2 filled region. In table 1, the initial conditions of experiments F2.1 Run1 and F2.1 Run2 are shown.

Table 1. F2.1 Run1 and Run2 initial conditions.

Test (RUN)	PKL III F OECD-PKL	
	F2.1 RUN1	F2.1 RUN 2
Secondary side boundary conditions	2 SG filled with water, one ready for operation and activated to keep pressure at 2 bar	
Primary side: Upper head by-pass	0.5	0.5
Level (hot legs)	¾ loop	RCL lower edge
Temperature at core outlet (°C)	61	65
Pressurizer wall temperature (water-filled region) (°C)	65	46
Pressurizer wall temperature (N2-filled region)(°C)	136 – 156	30 – 39
Power (kW)	225	224

Under the conditions shown in table 1, both experiments start when the RHR fails. Due to the residual heat generated in the core there is a rise in the core temperature and the void formation in the core starts, with an associated increase in the primary pressure. Then, primary coolant comes out from the vessel towards the steam generator U-tubes, which act as heat sink. The heat transfer into the steam generators causes their secondary-sides to heat up leading to void formation in the water-filled secondary sides of steam generators 1 (SG1) and 2 (SG2), and the associated rise in the secondary side pressure.

When the pressure of SG1 secondary side reaches $2 \cdot 10^5$ Pa, the main steam control system is activated, and the pressure is maintained constant at this value during the rest of the transient. Also in SG1 the secondary side level is controlled and maintained at 12.2 m. In SG2, neither the pressure nor the level controls are activated, so the pressure rises following the evolution of the primary system pressure. There is no change in the pressures of SG3 and SG4 as these steam generators are empty and no heat is transferred through them, so in both cases the pressure remain constant during all the transient (see Figure 2 and 3). After the secondary side pressure control activation, most of the residual heat generated in the core is removed through SG1, and the plant reaches a stable condition.

Since this moment, the actions taken to mitigate the accident are different for experiment F2.1 RUN1 and RUN2. In experiment F2.1 RUN1, after the plant stabilization, water is injected into the primary system through the accumulators in the following sequence: Injection from accumulator in hot leg 1 (ACC HL1), injection from accumulator in cold leg 2 (ACC CL2), injection from accumulator in cold leg 3 (ACC CL3), and finally injection from accumulator in hot leg 4 (ACC HL4). In Figure 2 all the injections from the accumulators are represented.

After the injections, a depressurization in secondary side of the steam generator full of water but not available for operation, SG2, takes place, and the pressure in that zone is lowered to 10^5 Pa. Following the secondary side depressurization there is a primary side depressurization that lowers the pressure in the primary circuit to $6 \cdot 10^5$ Pa. This action allows the injection of the low pressure injection system (LPIS) and after that the RHR pumps are restarted.

Figure 2 shows the experimental evolution of the primary and secondary side pressures, with the main actions taking place during the transient, as the SG1 control activation, the injections from the accumulators, the secondary and primary side depressurization, the injection from the LPIS and the recovery of the RHR system.

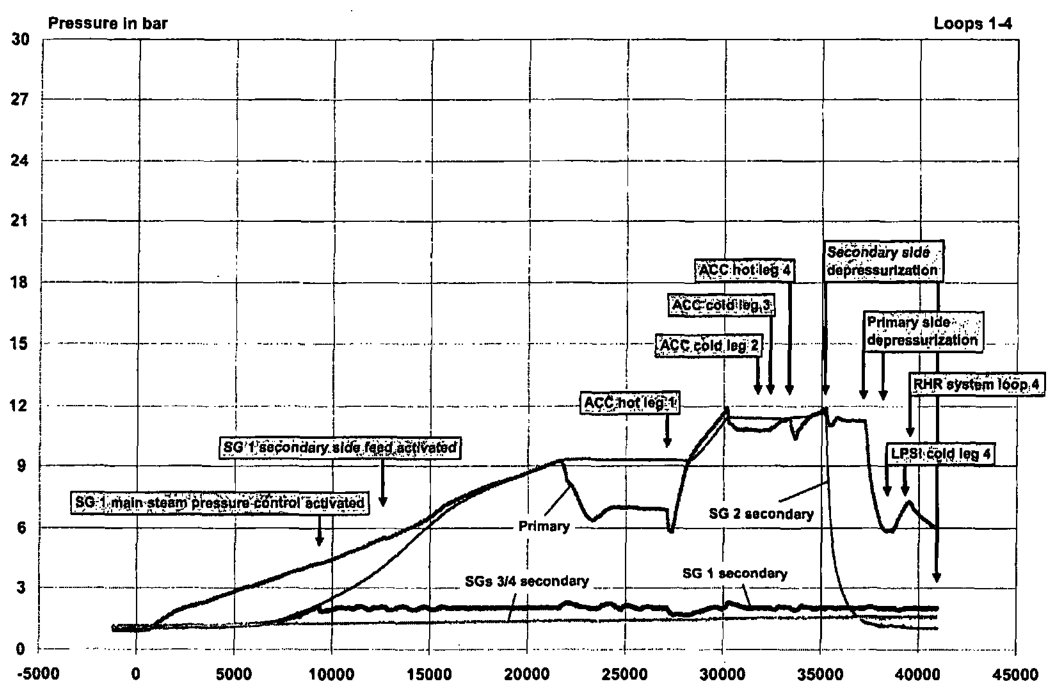

Figure 2. Experimental evolution of pressures experiment F2.1 RUN 1.

The first part of experiment F2.1 RUN2 is similar to F2.1 RUN1, there is an increase in the core temperature and void formation in the core, then a circulation of primary coolant towards the steam generators U-tubes is established and the heat is evacuated through the steam generators and the plant reaches at lower pressure than in Run1. In order to achieve the plant conditions to recover the RHR system, after the plant stabilization there is an injection from the low pressure injection system (LPIS) in loop 4, followed by the four injections through the accumulators in the same sequence as in the previous case. Once the injections from the accumulators have finished, there is a depressurization of the secondary side of SG2 followed by a depressurization of the primary circuit. Those actions are needed to allow a second injection from the LPIS in loop 4, followed by an injection from HPIS also in loop 4, before recovering the RHR system. Figure 3 shows the experimental evolution of the primary and secondary pressures and the different actions taken during the transient.

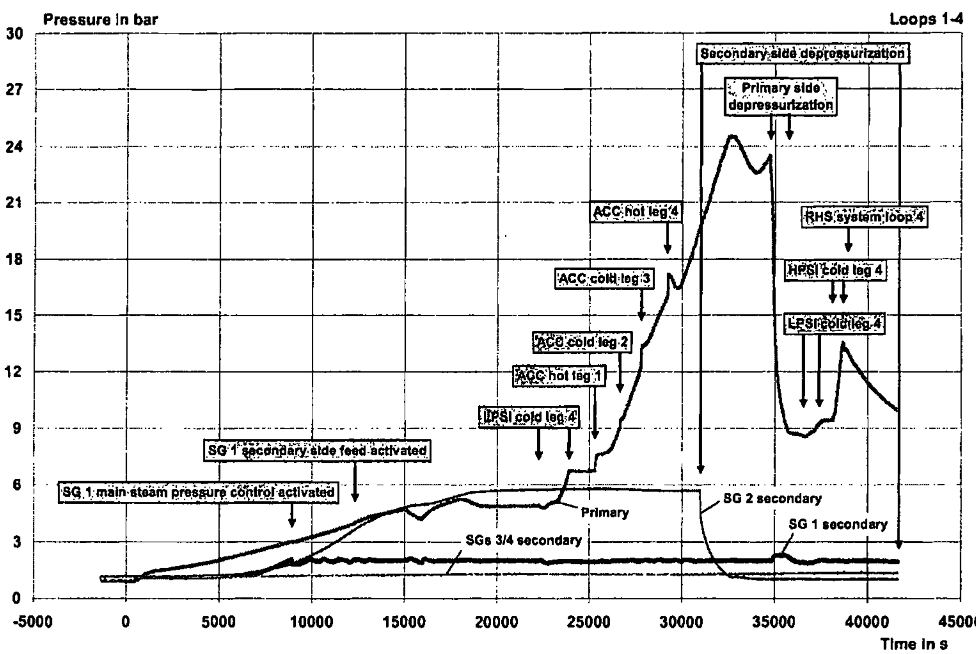

Figure 3. Experimental evolution of pressures experiment F2.1 RUN 2.

4. RELAP-5 MODEL OF PKL FACILITY

The transient simulation has been performed using RELAP5-Mod 3.3 code [12]. The RELAP-5 model used consists of 600 hydraulic volumes, 622 junctions and 512 heat structures. This model has been adapted to simulate shutdown conditions, from the PKL model provided by the facility [13]. Figure 4 outlines the nodalization used in the simulation.

The core is simulated using a pipe component of eight volumes. Six of these volumes contain the fuel rods, which are simulated using a *heat structure* component that generates the residual heat power established in Table 1. The vessel of the PKL facility has two external downcomers (see Figure 1) simulated in the RELAP model by means of two external pipes. The cold legs of all four loops are simulated using *pipe* and *branch* components, which are connected to two *branches*, volumes 232 and 234, which in turn are connected to the downcomer. The facility has four by-passes in the vessel upper head which have been collapsed in this model into two *branch* components, 223 and 225.

The four primary loops are modelled with a pump and a steam generator in each loop using *pipe, pump* and *branch* components. The U-tubes of the steam generators are lumped into three *pipe* components of different heights. The heat transfer between the primary and secondary systems is simulated using three *heat structures* one for each of the three *pipes* that simulate the steam generators U-tubes.

The different injections from the accumulators performed at the end of the transient (see Figure 2 and 3) have been simulated using an *accum* component connected to the loop by a valve. The locations of the injections from the accumulators are shown in Figure 4.

As the RHR, LPIS and HPIS inject in the same location of the facility, one injection in each clod leg (see figure 4), the model of these injections has been simplified by using *time dependent volumes* connected with *time dependent junctions* which establish the amount of water to be injected.

The failure of the RHR system is simulated in this model by the on set of the power in the core heat structure, which generates the residual heat power included in Table 1. In the same way, the restoration of the RHR system at the end of the transient is simulated by disconnecting the power in the core heat structure. A sensitivity analysis has been performed using TMDPJUN and TMPDVOL to simulate the RHR system operation, but no difference was observed in the results.

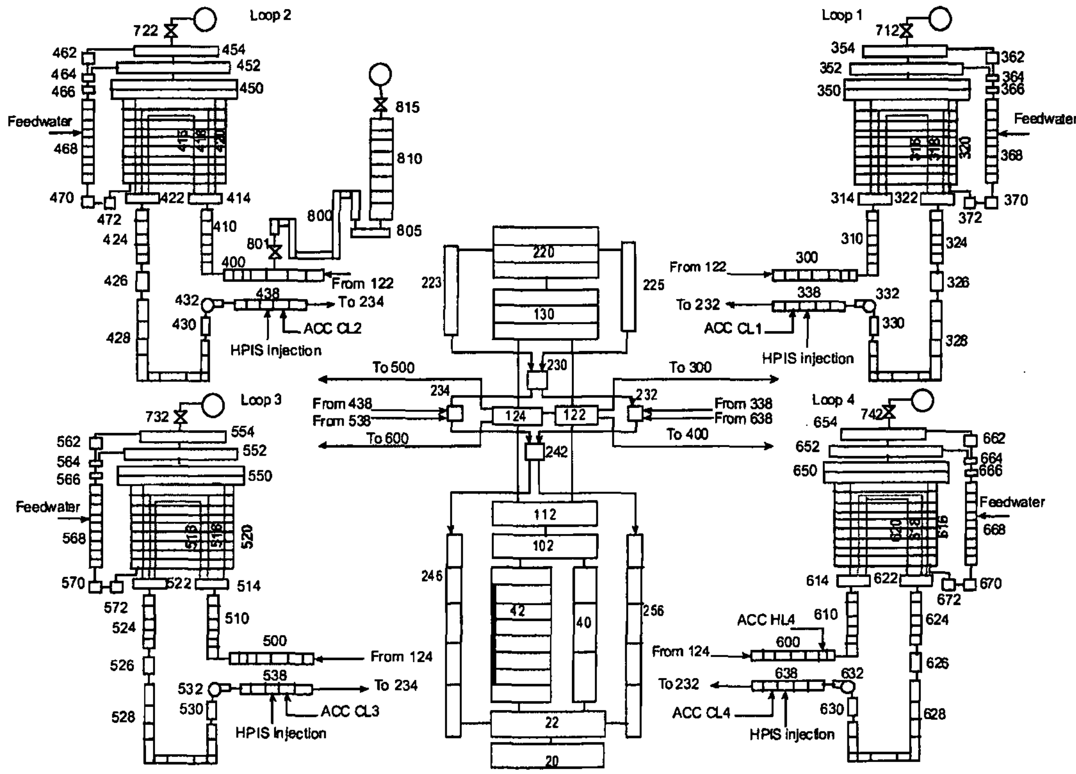

Figure 4. RELAP5 nodalization of PKL facility.

5. F2.1 SIMULATION RESULTS

This section is divided into two subsections, each one devoted to present the results obtained from the simulation using RELAP-5 of experiment F2.1 Run1 and Run2, respectively.

5.1. F2.1 Run1 simulation results

The first step in the simulation of F2.1 Run1 experiment is the calculation of the transient initial conditions shown in Table 1. The amount of water required in the primary circuit has been obtained by controlling the water level in the primary loops to ¾ of hot leg. In this phase, the RHR injections have been used to extract or supply the amount of water necessary to reach this level. The rest of the primary circuit is initialized as full of nitrogen.

The secondary side water levels of the steam generators in loops 1 and 2 are achieved using the secondary side feedwater injections (see Figure 4) to reach the initial value. The secondary sides of steam generator 3 and 4 are initialized full of nitrogen, as in the transient specification they are empty. Table 2 shows the values for the initial conditions obtained in the RELAP-5 calculations as compared with experimental data.

Table 2. Experimental and calculated initial conditions for F2.1 RUN1.

	Experiment specification	RELAP-5 initial condition RUN1
Primary: closed and filled with borated water up to ¾-loop, above filled with N_2		
Coolant inventory	¾ loop	7.71 m in vessel (¾loop)
Boron concentration	2200 ppm	2200 ppm
Pressure	0.93 bar	0.93 bar
Fluid temperature at core outlet	61°C	64°C
Pressurizer wall temperature (water-filled region)	65°C	65°C
Pressurizer wall temperature (N2-filled region)	136°C – 156°C	150°C
Loops flow conditions	No flow	0.0 kg/s
Pressurizer level	1.0 m.	1.0 m
Secondary: SG1-2 full of water, SG1 ready for operation, SG2 isolated, SG3-4 full of air and isolated		
Steam generators secondary pressure	aprox. 1 bar	1.14 / 1.12 / 1.05 / 0.99
SG1 and SG2 secondary temperature	58°C	58°C
SG1 and SG2 level	12.1 m.	12.1 m

Once the initial conditions are reached, the transient is initiated by the failure of the RHR system, which is simulated by connecting the heat structure in the core that generates the residual heat showed in Table 1.

Thus, when the RHR system fails the primary circuit starts to heat up until it reaches saturation conditions and void formation in the core begins. The two steam generators with the secondary filled with water, SG1 and SG2, act as heat sink.

Figure 5 shows the comparison between experimental data and RELAP-5 calculations of the water level in inlet side of SG1 U-tubes. The calculations performed by RELAP-5 provide a good prediction of the time at which water starts entering the U-tubes, although, in some periods of the transient the level calculated differs from the experimental data, e.g. around 15000 s.

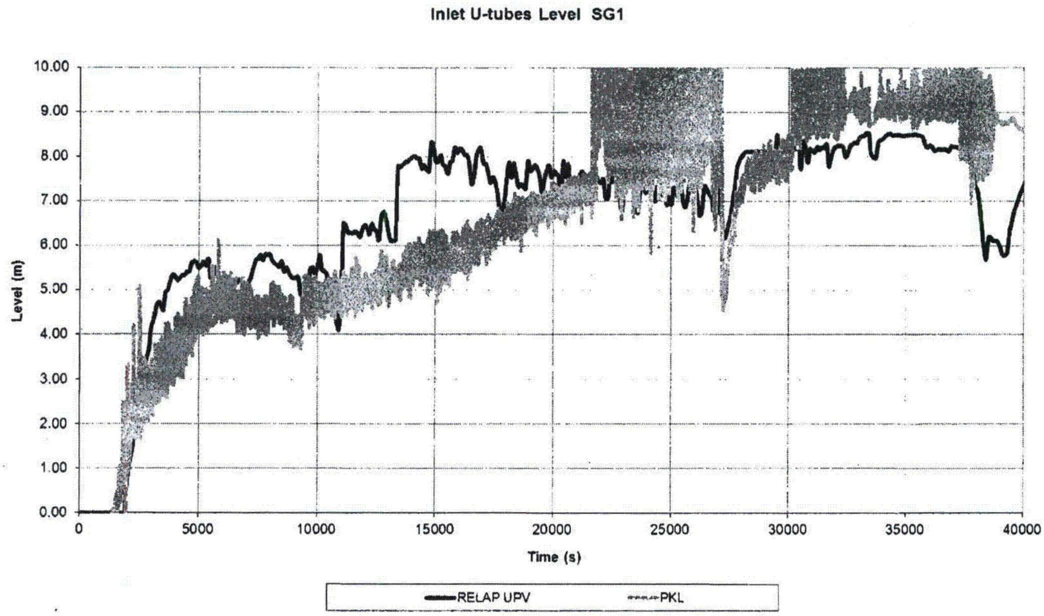

Figure 5. Water level inlet side of SG1 U-tubes.

Figure 6, shows the water level in the inlet side U-tubes of steam generator 2, which is the steam generator with the secondary side full of water but not available, i.e. neither pressure nor level control is activated. It can be observed that heat is transferred through this steam generator during the first 14000 seconds. At this moment pressure control in SG1 is activated and there is a water displacement from steam generator SG2, which empties, towards SG1, where the level increases (see Figure 5). At the end of the transient, safety injections produce a new increase in the water level in SG2 tubes.

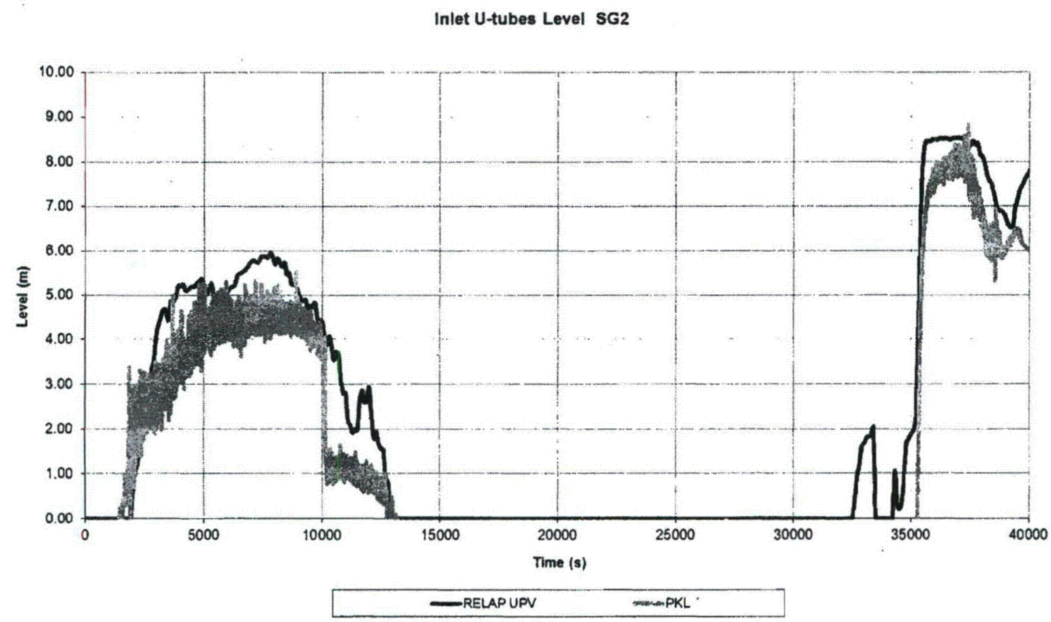

Figure 6. Water level in the U-tubes of steam generator 2.

The presence of water in the steam generators U-tubes that indicates there is heat transferred from the primary to the secondary side of the steam generators. Figure 7 represents the heat, calculated by RELAP5, which is transferred through both steam generators, SG1 and SG2, with the secondary side full of water. In this figure it can be observed that, when pressure and level control in SG1 is activated, the heat exchanged in SG2 decreases and SG1 becomes the only heat sink.

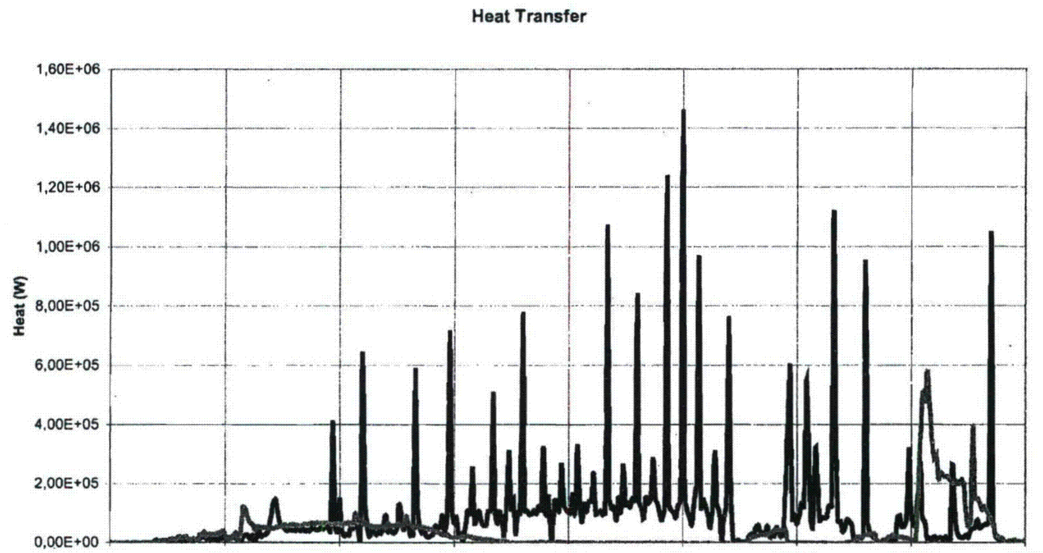

Figure 7. Calculated heat exchange through the steam generators.

The control pressure activation in the secondary side of SG1 is observed in Figure 8. This figure represents the evolution of the pressure calculated using RELAP5 and the experimental evolution for the SG1 secondary side pressure. It can be observed that, RELAP-5 calculation predicts a slight delay in reaching 2 10^5 Pa, which is the set point for pressure control activation, as compared with experimental data.

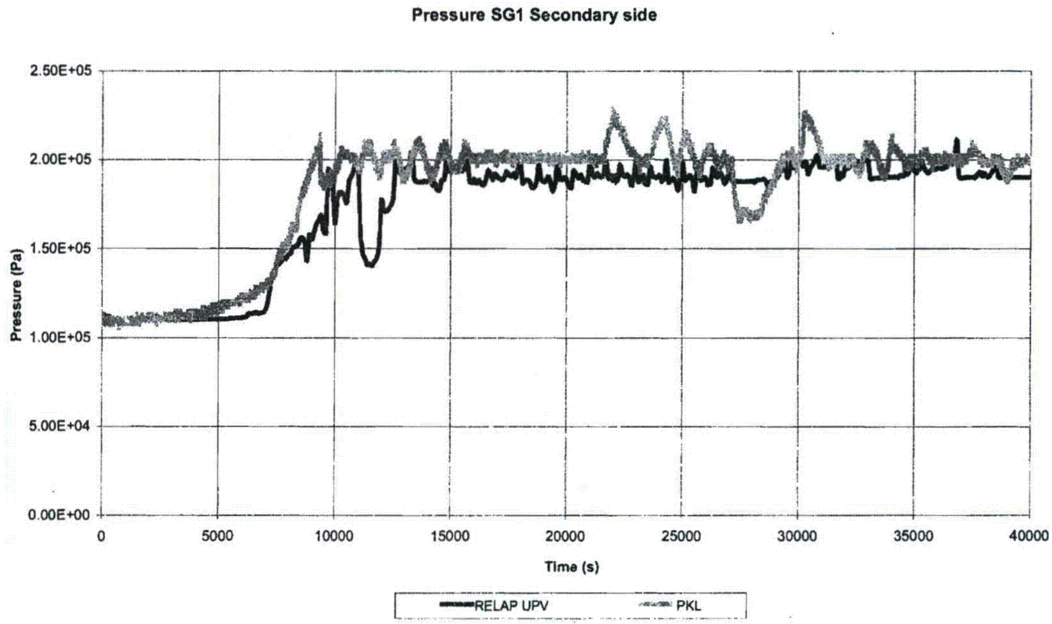

Figure 8. Secondary side pressure in steam generator 1.

The delay in the RELAP-5 pressure calculations is also observed in the evolution of the secondary side pressure of SG2, as shown in Figure 9. In this steam generator, there is no pressure control, so an increase in the secondary side pressure until the plant reaches a stable condition about $1.0 \cdot 10^6$ Pa is observed. The pressure starts to increase again when injections are activated, and in the experiment, the plant reaches another stabilization point around $1.2 \cdot 10^6$ Pa. However, in the calculations performed with RELAP5 the pressure does not reaches a stable value after the injections, but presents an increase until the secondary side depressurization, see figure 9.

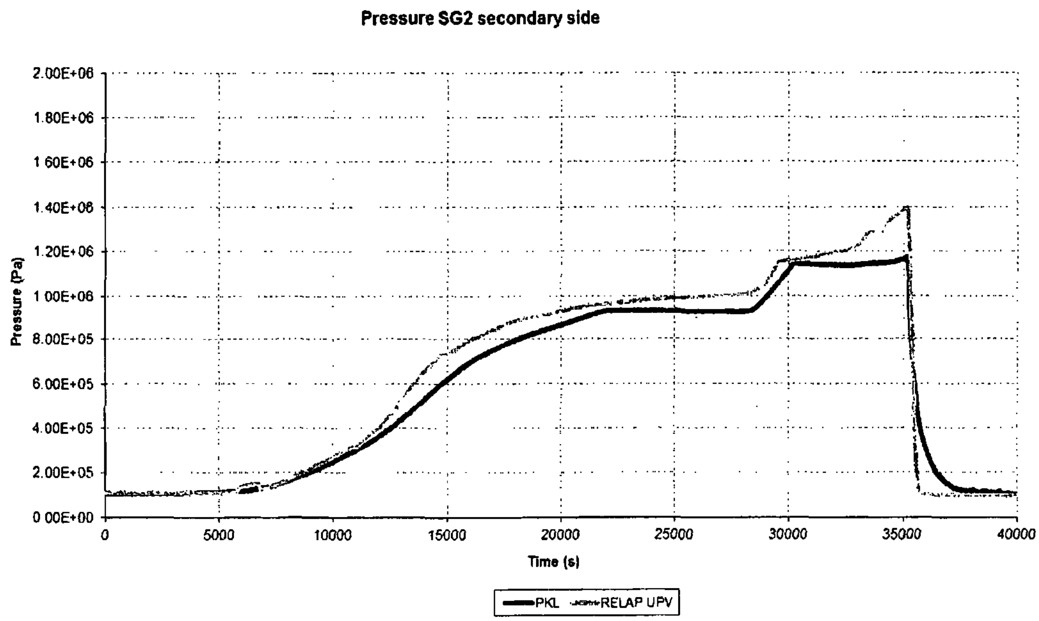

Figure 9. Secondary side pressure in steam generator 2.

In the primary circuit, the evolution of the pressure calculated by RELAP5 is quite similar to the experimental data, as it can be observed in Figure 10. This figure shows that it is possible to reach a stable situation of the plant using one steam generator as final heat sink, with a primary pressure around $7 \cdot 10^5$ Pa, which is quite close to the experimental value. The injections make the pressure rise until $1.00 \cdot 10^6$ Pa and finally the pressure of the system is lowered until $5 \cdot 10^5$ Pa when the RHR is recovered.

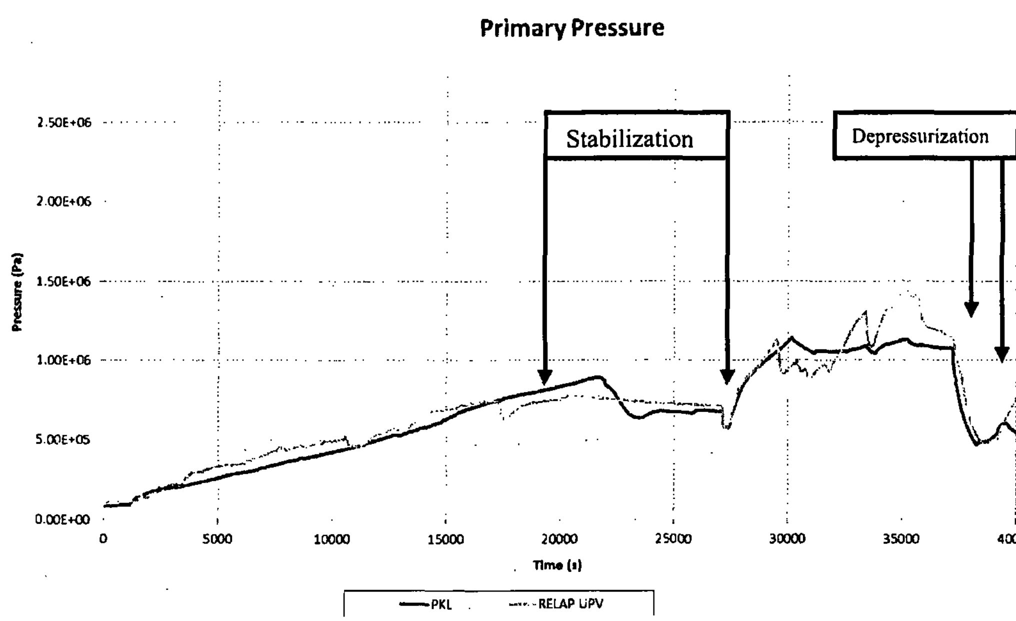

Figure 10. Primary pressure.

Regarding other important variables in the primary circuit, a difference between the mass inventory distribution calculated by RELAP5 and the experiment data is observed. Thus, figure 11 shows the reactor vessel level. This figure indicates that the mass inside the reactor vessel is similar for experimental and calculated data, although the calculations predict a lower level inside the reactor level, what means that more water is displaced from the core towards the reactor coolant circuit.

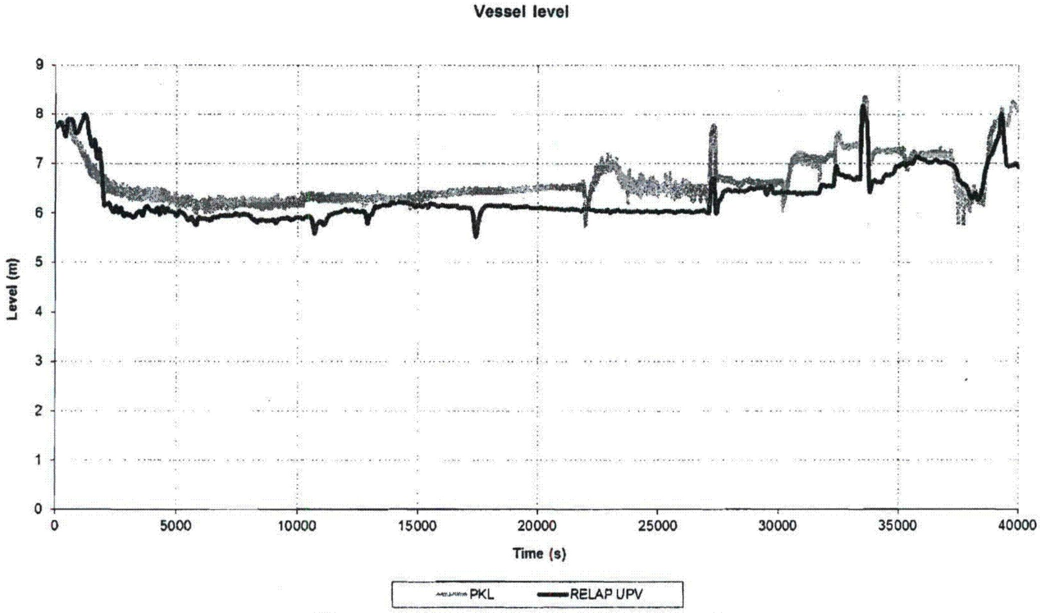

Figure 11. Reactor vessel level.

The water comes out from the vessel towards the steam generators U-tubes, see figures 5 and 6, and towards the pressurizer. Figure 12 shows the water level inside the pressurizer and great differences between the experimental data and RELAP-5 calculation can be observed. In both cases, water starts entering the pressurizer, however, after about 2500 seconds the level in the pressurizer calculated by RELAP-5 presents a larger increase than the experimental measurement. So, more water is displaced towards this volume in the calculation with respect to the experimental measure. This situation is maintained for about 17000 seconds after which the level is stabilized, but the value reached differs in 7.4 meters between RELAP-5 simulation and experimental data.

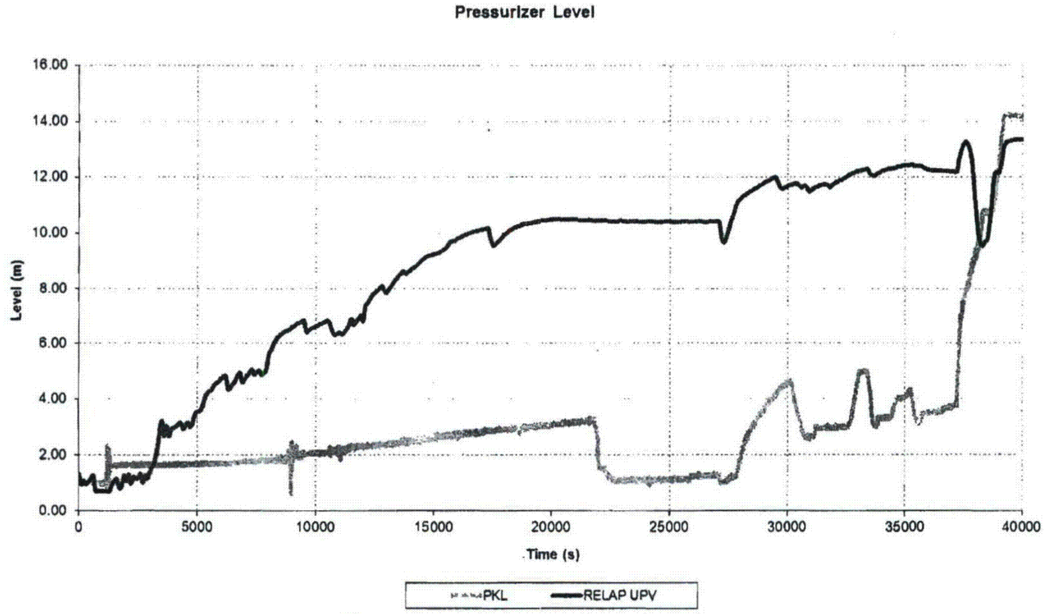

Figure 12. Pressurizer level.

To analyse the boron dilution process during this transient, the boron concentration in the loop seal one, that is, in the lowest part of the loop one, has been followed. Figure 13 shows the evolution of the boron concentration in this part of the facility. In this figure it can be observed that the calculation predicts a delay in the decrease of the boron concentration.

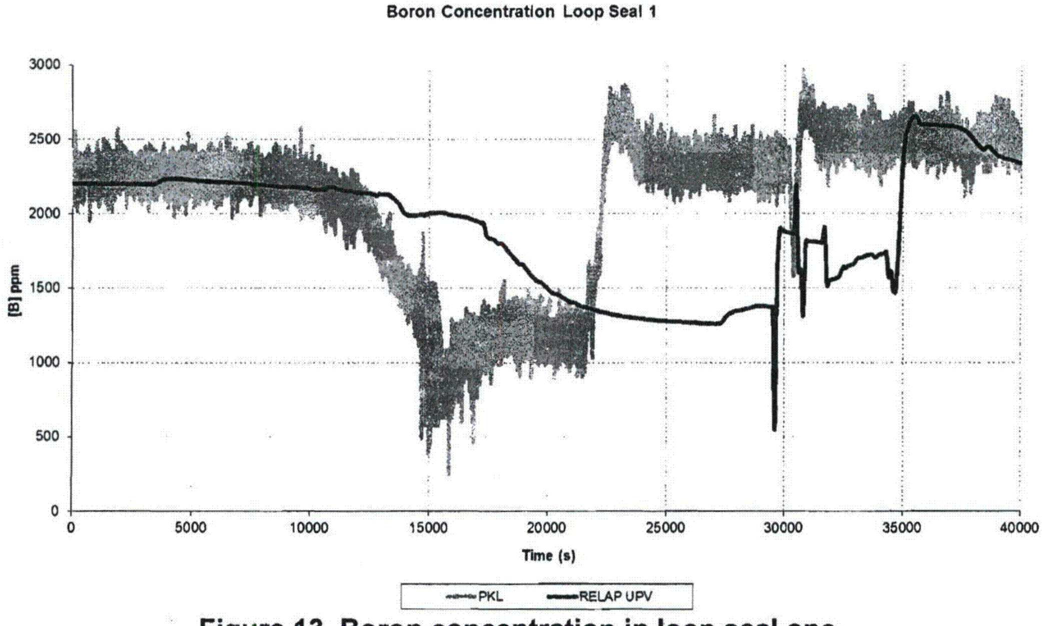

Figure 13. Boron concentration in loop seal one.

This drop in the boron concentration is due to the mass flow coming from the inlet to the outlet of the steam generators U-tubes. The vapour that reaches the top of the U-tubes condenses in the outlet side of the tubes. As this condensate is boron free, it causes the drop in the boron concentration when it reaches the loop seal, see Figure 13.

5.2. F2.1 Run 2 simulation results

As in the previous case, the first step in the F2.1 Run 2 simulation is the calculation of the transient initial conditions. In this case the primary circuit is full of water up to the lower edge of the reactor coolant line (RCL), and the wall temperature in the pressurizer ranges in the interval 45-65°C. Table 3 shows the initial conditions for F2.1 Run 2 transient obtained with RELAP-5 calculation as compared with experimental data.

Table 3. F2.1 RUN2 initial conditions.

	Experiment specification	RELAP-5 initial condition RUN 2
Primary closed and filled with borated water up to lower edge of RCL, above filled with N_2		
Coolant inventory	Lower edge of hot legs	7.70 m in vessel.
Boron concentration	2200 ppm	2200 ppm
Pressure	0.93 bar	0.93 bar
Fluid temperature at core outlet	61°C	67°C
Pressurizer wall temperature (water-filled region)	46°C	45°C
Pressurizer wall temperature (N2-filled region)	30°C – 39°C	39°C
Pressurizer level	1.0 m.	1.0 m
Secondary: SG1-2 full of water, GV1 ready for operation, SG2 isolated, SG3-4 full of air and isolated		
Steam generators secondary pressure	aprox. 1 bar	1.14/1.12/1.05/0.99 bar
SG1 and SG2 secondary temperature	48°C	48°C
SG1 and SG2 level	12.1 m.	12.1 m

Once the initial conditions are reached, the transient is initiated by the failure of the RHR system, which is simulated by connecting the heat structure in the core that generates the residual heat established in Table 1. Thus, when the RHR system fails the primary circuit starts to heat up until it reaches saturation conditions and void formation in the core begins. The two steam generators with the secondary filled with water, SG1 and SG2, act as heat sink.

Figure 14 and figure 15 show the comparison between experimental data and RELAP-5 calculations of the water level in inlet side of SG1 and SG2 U-tubes, respectively.

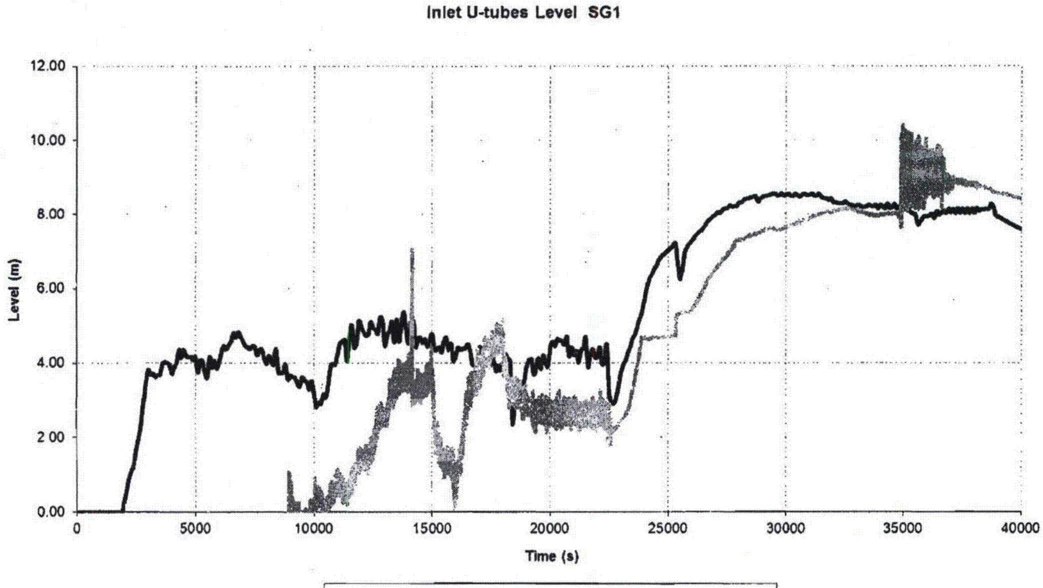

Figure 14. Inlet side of SG1 U-tubes

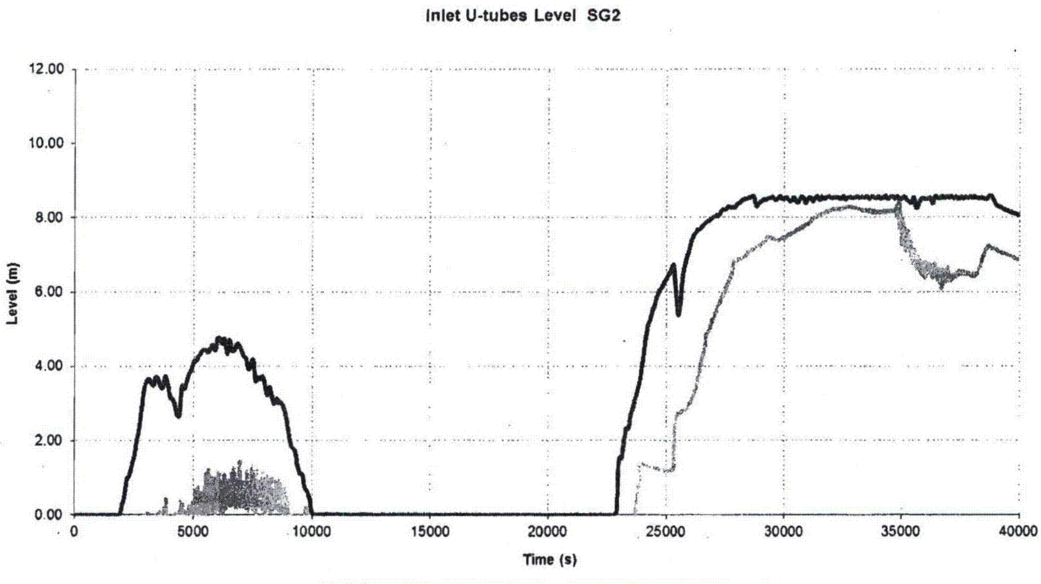

Figure 15. Inlet side of SG2 U-tubes

One can realize that great differences exist between the results obtained with RELAP-5 and experimental data. As shown in Figure 14, RELAP-5 predicts that water enters into the inlet side of SG1 at 2000 s. after the transient is initiated, while the experimental data indicates there is no flow inside the SG1 U-tubes until around 9000 s. The same occurs in SG2, see figure 15, while RELAP-

5 calculations predict that 2000 s after the RHR is lost there is an amount of water in the SG2 U-tubes, while in the experiment there is no water level in this part until 4000 s. In fact, differences exist not only in the time when water starts SG1 and SG2 U-tubes, but also in the amount of water present in the U-tubes, which would affect heat transfer between primary and secondary sides, and consequently primary system pressurization.

It seems there is an inconsistency between the experimental data supplied by the organizers and the plant behaviour. For example, observing the evolution of the secondary side pressure of SG1 and SG2, see figures 16 and 17, it can be seen that the pressure on the secondary side rises similarly in both simulation and experimental data, due to the heat exchanged between the steam generators primary and the secondary sides, and some water is necessary to produce this exchange. Moreover, figure 16 shows the activation of the pressure control in SG1, which demonstrates with regard to this physical variable, that the calculations predict quite accurately the experimental data. So to achieve the appropriate conditions of pressure in the secondary side there must be a heat exchange as predicted by RELAP-5. Note also that activation of the pressure control in SG1 starts about 8000 s (Figure 16) while water level in the SG1 U-tubes starts about 9000 s. (figure 14).

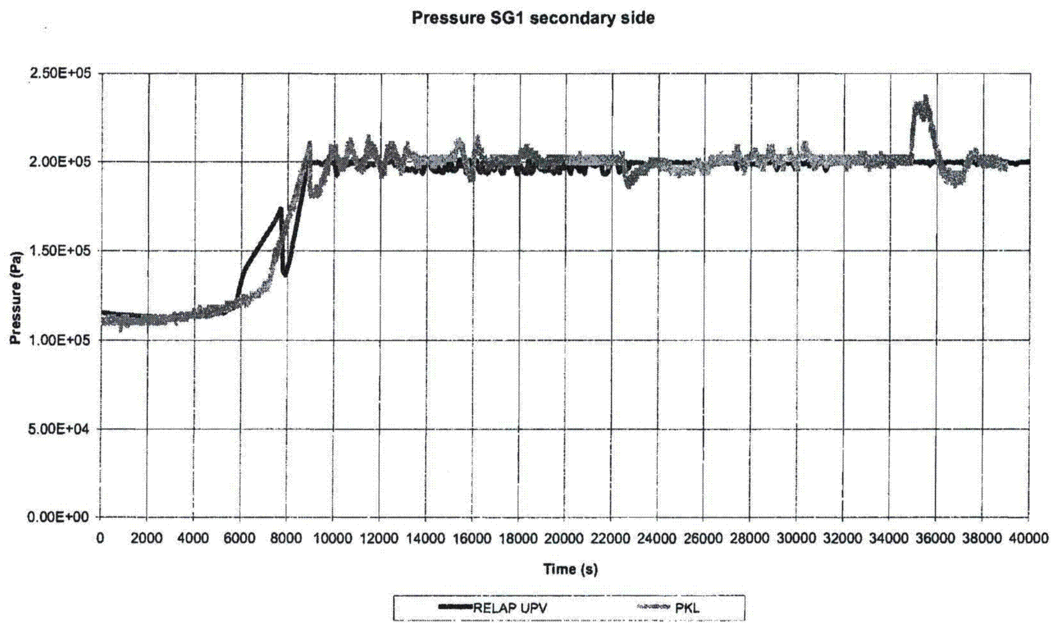

Figure 16. Pressure in the SG1 secondary side

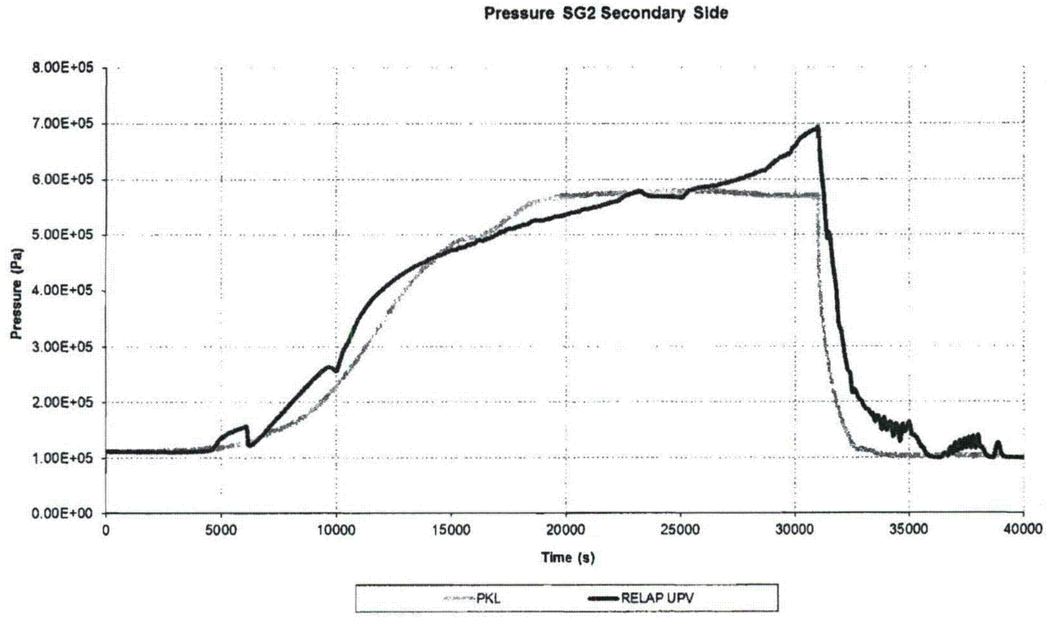

Figure 17: Pressure secondary side SG2

Regarding the heat exchanged, Figure 18 shows the amount of heat exchanged in SG1 and SG2 calculated by RELAP-5. The heat exchanged explains the pressurization of the steam generators secondary sides, as it happens in experiment F2.1 Run1.
Also in this experiment when the pressure control in SG1 is activated, see figure 16, all the residual heat generated in the core is removed through SG1, see figure 18.

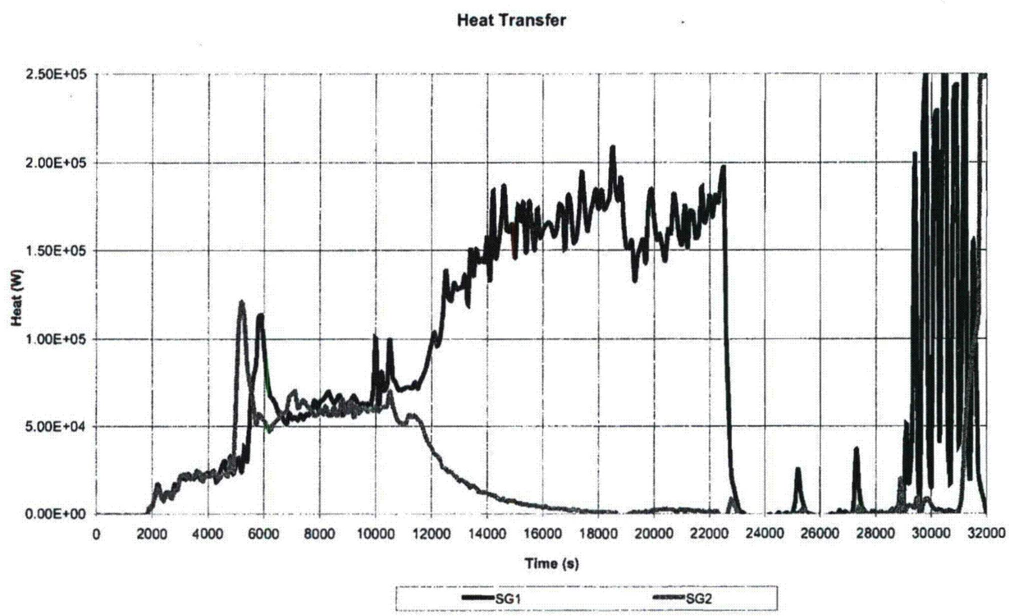

Figure 18. Calculated heat exchange through the steam generators.

In the primary circuit, when the RHR is lost there is a rise in the primary pressure until the plant reaches a stable situation around 5.00 10^5 Pa, as shown in figure 19. This situation is maintained until the injection from the LPIS at 22350 s. It is also shown that there is no difference between RELAP-5 simulation results and experimental data until LPIS injection. The pressure is higher when the injection are produced due to the mass distribution in the primary circuit calculated by RELAP-5, which predicts more water inside the pressurizer.

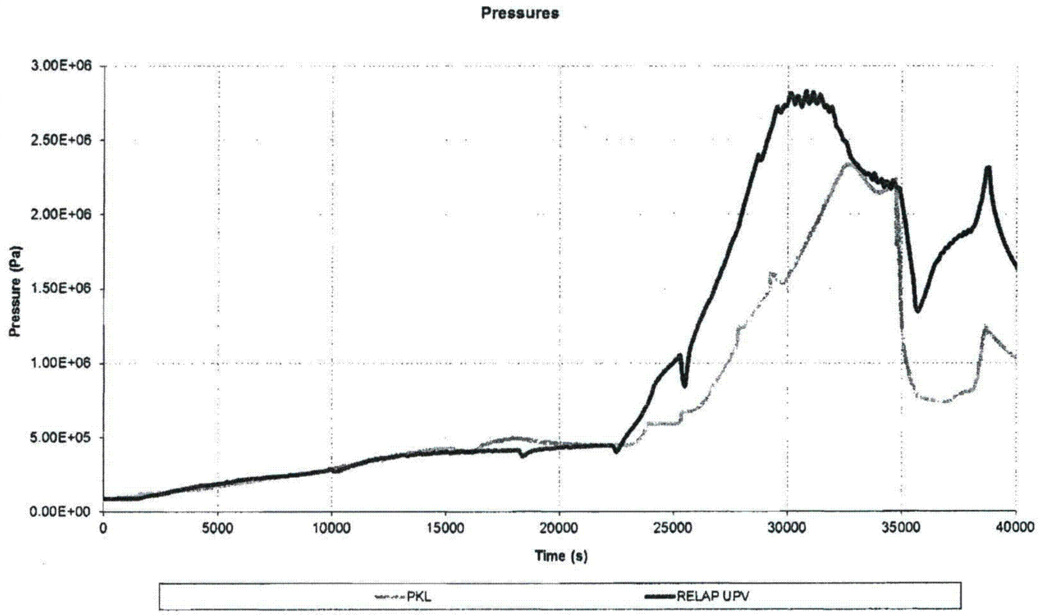

Figure 19. Primary pressure evolution

After the LPIS injection, safety injections through the different accumulators start, in the same sequence as in the previous case, what raise the pressure in the primary circuit. In order to produce the second injection from the LPIS, the pressure of the system must be lowered. The secondary side depressurization begins at 31022 s, see figure 17. The primary depressurization starts at 34750 s, see figure 19, and the injection from the LPIS is produced at 36600 s. Finally, at 38090 s an injection from the HPIS is produced.

At 38810 s., once all the injections have ceased it is possible to restart the RHR. Figure 20 shows the vessel level during the transient evolution. In this figure it is observed that the experimental level reaches more than 10 m but in the calculations performed by RELAP-5 the level in the primary circuit never reaches this value. This can be explained by the different mass distribution observed between the experimental data and the RELAP-5 calculations. Thus, the calculation predicts more water entering into the pressurizer that the experimental data as can be observed in figure 21, where the water level inside the pressurizer is represented. Moreover, the amount of water injected through the accumulators is lower in the simulation than in the experimental data, due to the pressure in the primary circuit calculated by RELAP-5 is higher than in the experiment as can be observed in figure 19.

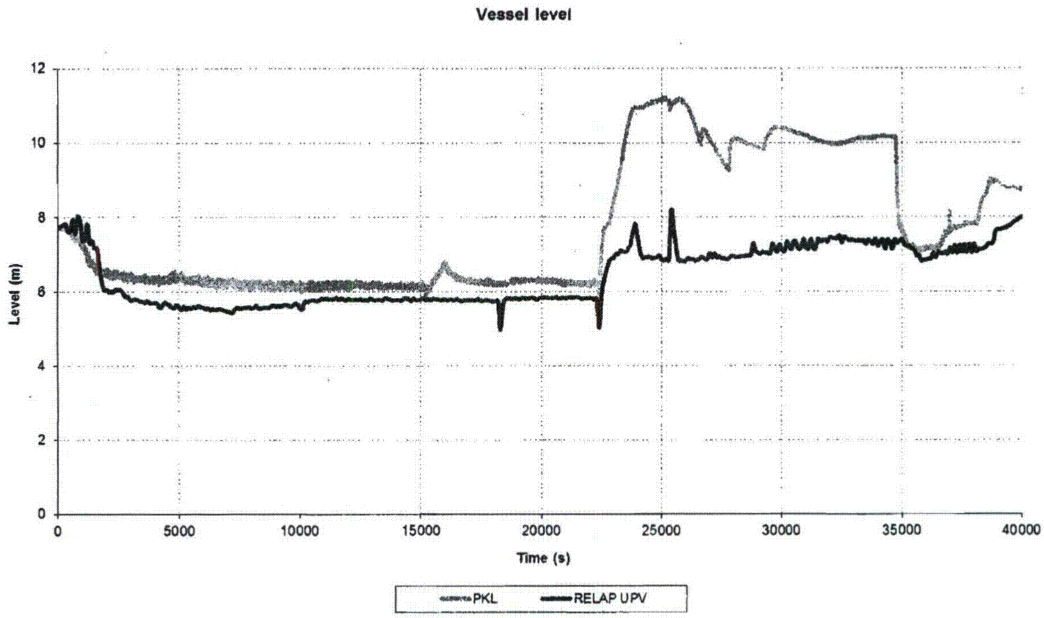

Figure 20 Primary level

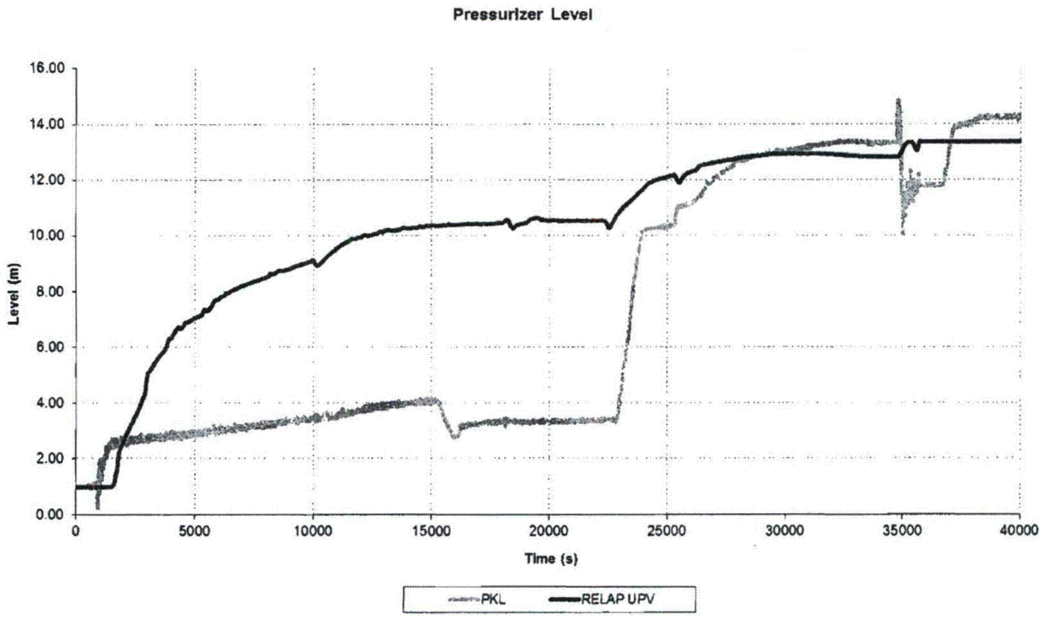

Figure 21. Pressurizer level

In order to analyze the boron dilution in the primary circuit, the boron concentration on the loop seal one has been tracked. Figure 22 shows the evolution of the boron concentration at the loop seal of loop 1. In this figure it can be observed that RELAP5 calculations predict a decrease until the time the injections start. Since this moment there is an increase in the boron concentration in this part of the installation with the maximum reached when the RHR is recovered. The sharp decrease observed in the experimental data, after 15200 s is not reproduced by the calculations, which shows instead a smooth decrease.

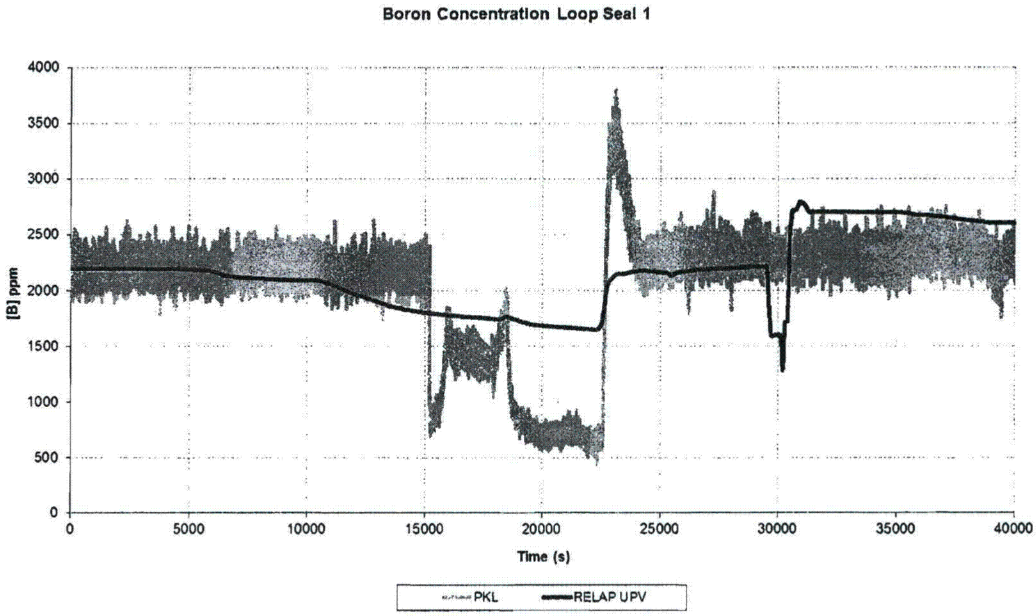

Figure 22. Boron concentration at the loop seal 1

5.3. Comparison of F2.1 Run 1 and Run 2 plant evolution simulation.

One of the objectives of this work is focused on analysing the influence of the plant initial conditions on the plant behaviour after the RHR is lost. In this case, the two runs simulated differ in the amount of water in the reactor coolant circuit and in the temperature of the pressurizer. The effect of the variation in the initial plant situation can only be analyzed during the first part of the transient, until the plant reaches a stable situation, as after this time, different actions are taken in each case to mitigate the transient consequences.

In this section, the evolution along the first part of the transient of some important variables, which show the difference in the plant behaviour, are presented. The figures show the evolutions of the calculations performed using RELAP5 for F2.1 Run1 and F2.1 Run2 respectively.

Thus, Figure 23 shows the evolution of the primary pressure for F2.1 Run1 and F2.1 Run 2. As it can be observed in this figure for both cases the plant reaches a stable condition, but the value of the primary pressure at which the plant reaches a stable condition is different depending on the initial conditions considered. Thus, with the primary full of water up to ¾ of loop and hot pressurizer the stabilization pressure is higher than supposing a lower level of water in the primary circuit with a cold pressurizer.

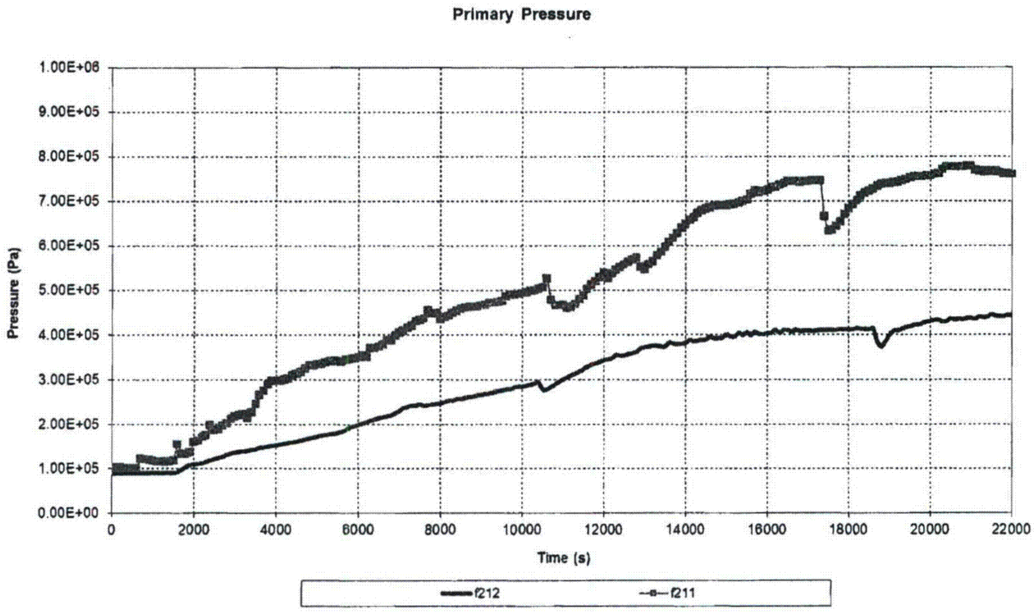

Figure 23. F2.1Run1 and Run2 primary pressure calculation.

Regarding the mass distribution inside the primary circuit, in both cases RELAP5 predicts a different behaviour than in their experiments as explained in the previous sections. In fact, the same behaviour is observed in both Runs, for which RELAP5 calculations predict a higher level of water in this part of the system, see figures 12 and 21. Figure 24, shows the water level inside the pressurizer that reaches 10m for Run1 and Run2, which corresponds to the PZR full of water. The only difference observed in the evolution of the water level inside the pressurizer is that for F2.1Run2, primary level up to the RCL edge and cold pressurizer, the water starts entering the pressurizer earlier than in F2.1Run1.

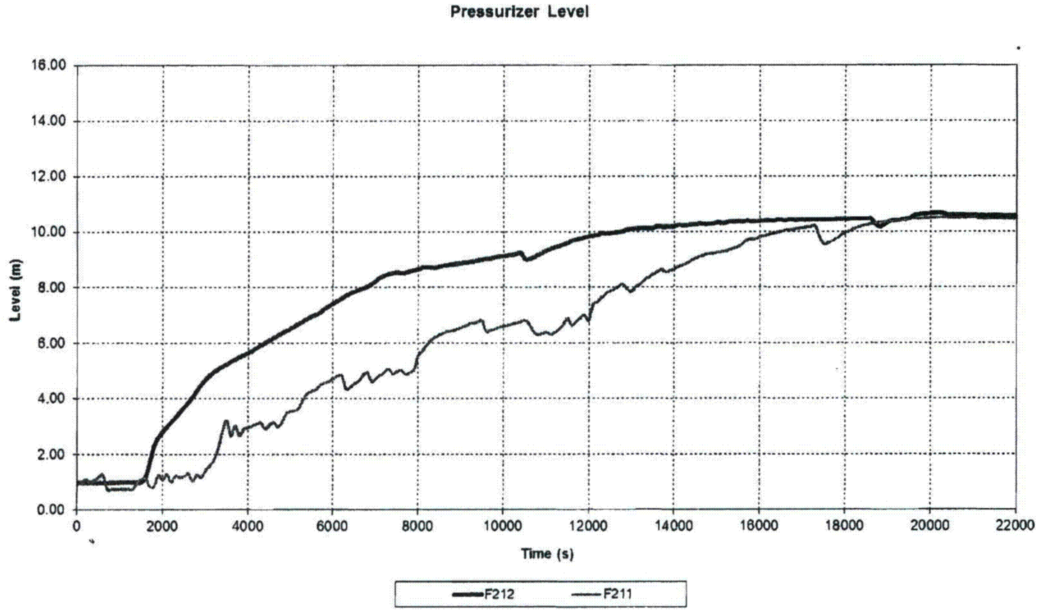

Figure 24. F2.1Run1 and Run2 pressurizer level calculation

The evolution of the water level in the pressurizer agrees with the behaviour of the water inside the reactor vessel in each case. Thus, figure 25 shows the evolution of the level of water inside the vessel for F2.1Run1 and F2.1Run2 calculations. In this figure it can be seen a slight advancement in the decrease of the level inside the vessel and the final level reached is lower than in the calculations of F2.1Run1. So, the water coming out of the vessel comes into the pressurizer earlier in Run2 than in Run1.

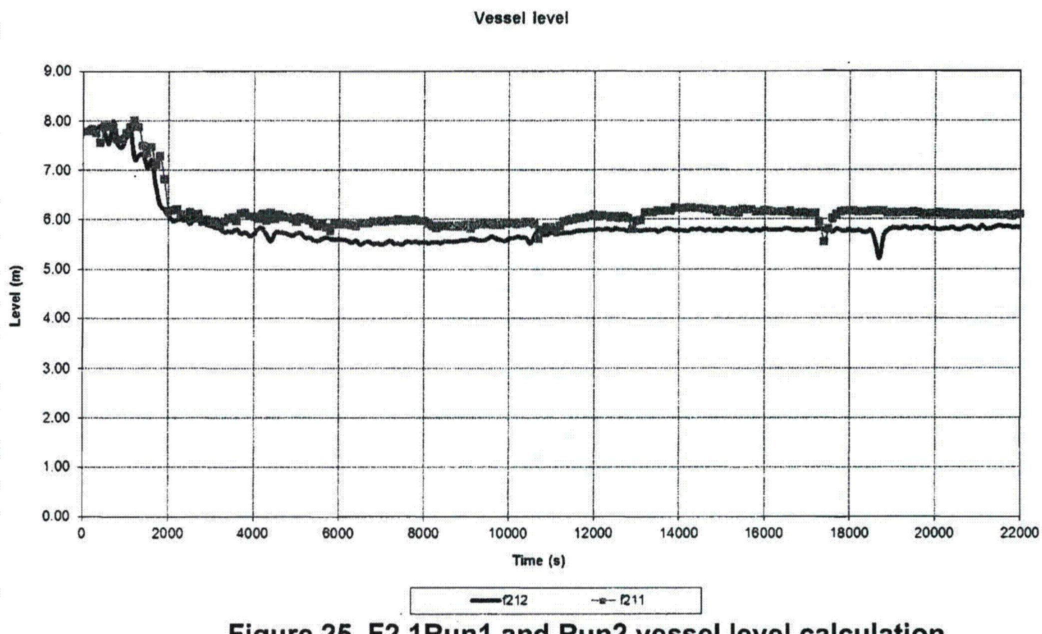

Figure 25. F2.1Run1 and Run2 vessel level calculation

Finally, figure 26 presents the results obtained using RELAP-5 for the boron concentration evolution in the loop seal 1. For this variable, the results obtained in the calculations performed are quite similar in both situations. As shown in figure 26, the boron concentration is maintained for both cases until 10000 s., and after this time the boron concentration presents a smooth decrease in both cases.

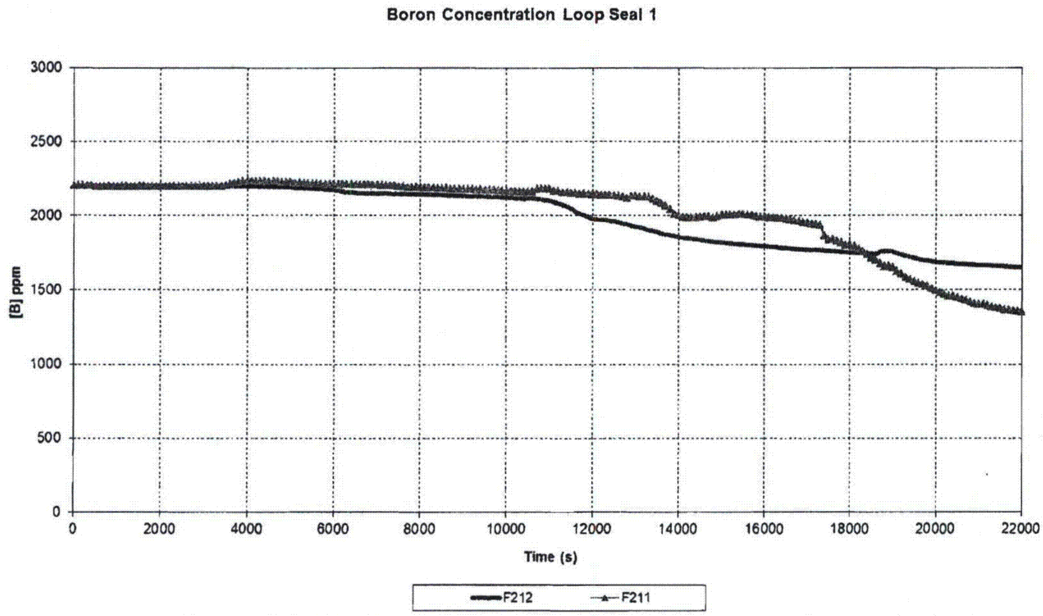

Figure 26. F2.1Run1 and Run2 boron concentration calculation

6. RUN STATISTICS

The calculations of F2.1 Run 1 and 2 have been performed using a server of Cluster IBM 1350 with a biprocessor Intel Xeon with the following characteristics:

- x335 2.40GHz/100MHz/512KB L2, 512MB Memory, 331W, HS Open bay.
- x335 Processor 2.4GHz/512KB Upgrade.
- 1GB PC2100 CL2.5 ECC DDR SDRAM RDIMM.
- 18.2GB 10K-RPM ULTRA 160 SCSI Hot-Swap SL HDD
- Remote Supervisor Adaptor

In tables 4 and 5 there are exposed the relevant parameters of the run statistics of the simulation of experiment F2.1 RUN1 and RUN2, respectively.

Table 4: Run Statistics F2.1 RUN 1

	RT	CPU	TS	CPU/RT	C	DT	GT
Steady-state	2800	10186.00	0.005	3.6379	600	101682	0.1670
Pressure control	9700	135953.61	0.01	14.0158	600	1412054	0.1605
Primary stabilization	18500	279384.75	0.005	15.1021	600	3234595	0.1440
First injection	27120	418987.17	0.005	15.4494	600	4954747	0.1409
End of transient	40940	647007.42	0.005	15.8038	600	7835986	0.1376

Table 5: Run Statistics F2.1 RUN 2

	RT	CPU	TS	CPU/RT	C	DT	GT
Steady-state	2800	8490.7998	0.005	3.0324	600	100543	0.1407
Pressure control	9400	117548.69	0.01	12.505	600	1317185	0.1487
Primary stabilization	14000	175739.52	0.01	12.553	600	1946681	0.1505
First injection	22350	356435.66	0.005	15.948	600	3657256	0.1624
End of transient	41650	621093.51	0.005	14.912	600	6667797	0.1552

RT: Transient time (s)
CPU: Execution time (s)
TS: Maximum time step (s)
C: Total number of volumes
DT: Total number of time steps
GT: GT = $(CPU*10^3/(C*DT))$

7. CONCLUSIONS

This work is focused on the simulation of a RHR failure with the plant in shutdown conditions and the primary circuit closed. In this conditions two experiments have been performed to asses the effect of the amount of water inside the primary circuit and the pressurizer initial conditions on the plant behaviour.

From the results presented in the previous sections it can be observed that in any case the plant reaches a stable situation when the residual heat sink is evacuated through the steam generators. The new situation reached maintains the plant integrity and the injections produced at the end of the transient assure the plant safety during the entire transient.

The results obtained in F2.1Run1 simulation agree with the experimental data although a significant difference in the mass distribution in the primary circuit is found, as can be observed in the pressurizer, U-tubes and vessel levels. Regarding the boron dilution in the loop seal, the simulation predicts a delay in the evolution of the boron concentration in this part of the facility. A similar result is obtained from the F2.1Run2 simulation when comparing the levels inside the primary, what implies that again in this calculation the mass distribution inside the primary circuit is different from the experimental data. And, also in F2.1Run2 the boron concentration in the loop seal differs from the experimental data as the simulation is delayed and there is a sharp deboration in the experimental data that is not predicted by the calculation.

The comparison of the result obtained with different plant initial conditions, shows that the primary level and the temperature conditions in the pressurizer affect the plant behaviour. In any case a stable plant situation is reached, and it is always maintained under safe conditions, but there is a difference in the stabilization situation depending on the initial situation supposed. However, the influence of each initial condition separately on the plant behaviour cannot be analysed as the experimental data is performed varying both parameters simultaneously.

8. REFERENCES

1. Seo J.K. and Park G.C. 2000. Return momentum effect on water level distribution during *mid-loop operations*. Nucl. Eng. Des., 202, 97-108.

2. Seul K.W., Bang Y.S., Kim H.J. 2000. Mitigation measures following a loss-of-residual-heat-removal event during shutdown. Nuclear Technology, 132, 152-165.

3. NUREG-1269, 1987. Loss of residual heat removal system, Diablo Canyon Unit 2, April 10, 1987. U.S. Nuclear Regulatory Commission.

4. NUREG-1410, 1990. Loss of vital AC power and the residual heat removal system during midloop operations at Vogtle Unit 1 on March 20,1990. U.S. Nuclear Regulatory Commission.

5. Hassan Y.A., RAJA L.L. 1993. Simulation of loss of RHR during midloop operations and the *role of steam generators in decay heat removal using the RELAP5/Mod3 code.* Nuclear Technology, 103, 310- 318.

6. Hassan Y.A., Banerjee S.S. 1994. RELAP5/Mod3 simulation of the loss of the residual heat removal system during a midloop operation experiment conducted at the ROSA-IV large scale test facility. Nuclear Technology, 118, 191-206.

7. Hassan YA, Troshko AA. 1997. Simulation of loss of the residual heat removal system of BETHSY integral test facility using CATHARE thermal-hydraulic code. Nuclear Technology, 119,1, 29-37.

8. Seul K.W. Bang Y.S. Kim H.J. 1998. Plant behaviour following a loss-of-residual-heat-removal event during shutdown conditions. Nuclear Technology, 126, 265-277.

9. Choi C.J., Nakamura H. 1997. RELAP5/Mod3 analysis of a ROSA –IV/LSTF loss of RHR experiment with a 5% cold leg break. Ann. Nucl. Energy, 24, 4, 275-285.

10. Ferng Y. Ma S. 1996. Investigation of system responses of the Maanshan nuclear power plant to the loss of the heat removal during midloop operation using RELAP5/Mod3 simulation. Nuclear Technology, 116, 160-172.

11. Umminger, K., Mandl, R. and Wegner, R. 2002. Restart of Natural circulation in a PWR-PKL test results and s-RELAP5 calculations. Nucl. Eng. Des., 39-50.

12. NUREG-5535, 2001. RELAP5/MOD3.3 code manual. Volume II: User's guide and input requirements. December 2001. U.S. Nuclear Regulatory Commission.

13. FRAMATOME ANP, 2002. PKL III: RELAP-5/Mod3 Input-Model. NGES1/2002/en/0059

NRC FORM 335
(9-2004)
NRCMD 3.7

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-0250

2. TITLE AND SUBTITLE
Simulation of the F2.1 Experiment at PKL Facility using RELAP5/Mod3

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	2011

4. FIN OR GRANT NUMBER

5. AUTHOR(S)
S. Carlos, J.F. Villanueva, S. Martorell, V. Serradell

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.)*

Universidad Politécnica de Valencia
Camí de Vera s/n
46022 Valencia, 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
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*
When a nuclear power plant is in shutdown conditions for refuelling, the reactor coolant system water level is reduced. This situation is known as mid-loop operation, and the residual heat removal (RHR) system is used to remove the decay power heat generated in the reactor core.

In mid-loop conditions, some accidental situations may occur with not a negligible contribution to the plant risk, and all involve the loss of the RHR system. Thus, to better understand the thermal-hydraulic processes following the loss of the RHR during shutdown, transients of this kind have been simulated using best-estimate codes, comparing their results against experimental data taken from different integral test facilities. This paper focuses on the simulation, using the best estimate code RELAP5/Mod 3.3, of the experiment F2.1 conducted at the PKL facility, within the OECD/PKL project. This experiment consists of the loss of the RHR system when the plant is in mid-loop conditions for refuelling and with the primary circuit closed. In the experimental series F2.1the physical phenomena to investigate are the mechanisms of heat removal in presence of nitrogen and the deboration in critical parts of the primary system.

The simulations present differences in the initial plant coolant inventory and temperature in the pressurizer, F2.1RUN1 and F2.1RUN2, to asses the influence of these differences in the transient evolution.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*
Residual heat removal (RHR) system
Nuclear Regulatory Commission of the United States of America (USNRC)
Consejo de Seguridad Nuclear of Spain (CSN)
thermalhydraulic codes
TRACE
RELAP5
TRAC-P
TRAC-B
Code Application Maintenance Program (CAMP)
UNESA (the association of Spanish electric energy industry)

13. AVAILABILITY STATEMENT
unlimited

14. SECURITY CLASSIFICATION
(This Page)
unclassified
(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE