



International Agreement Report

Simulation of ROSA-2 Test 3 Counterpart with TRACE5 - Application to Nuclear Power Plant

Prepared by:
S. Gallardo, A. Querol, Lorduy, and G. Verdu

Universitat Politecnica de Valencia
Instituto Universitario de Seguridad Industrial, Radiofisica y Medioambiental
Cami de Vera s/n
46022 Valencia, SPAIN

Kirk Tien, NRC Project Manager

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

Manuscript Completed: April 2018
Date Published: March 2019

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications 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 Library at 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 Publishing Office
Mail Stop IDCC
Washington, DC 20402-0001
Internet: bookstore.gpo.gov
Telephone: (202) 512-1800
Fax: (202) 512-2104

2. The National Technical Information Service

5301 Shawnee Road
Alexandria, VA 22312-0002
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
Multimedia, Graphics, and Storage &
Distribution 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 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, 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).

DISCLAIMER: This report was prepared under an international cooperative agreement for the exchange of technical information. Neither the U.S. Government nor any agency thereof, nor any employee, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this publication, or represents that its use by such third party would not infringe privately owned rights.



International Agreement Report

Simulation of ROSA-2 Test 3 Counterpart with TRACE5 - Application to Nuclear Power Plant

Prepared by:

S. Gallardo, A. Querol, Lorduy, and G. Verdu

Universitat Politecnica de Valencia
Instituto Universitario de Seguridad Industrial, Radiofisica y Medioambiental
Cami de Vera s/n
46022 Valencia, SPAIN

Kirk Tien, NRC Project Manager

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

Manuscript Completed: April 2018

Date Published: March 2019

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

**Published by
U.S. Nuclear Regulatory Commission**

ABSTRACT

This paper presents a first approach of a hot leg Small Break Loss-Of-Coolant Accident (SBLOCA) application to a Nuclear Power Plant (NPP). For this purpose, a standard 3-loop PWR NPP model by means of the thermal-hydraulic code TRACE5 patch 2 and Symbolic Nuclear Analysis Packages (SNAP) has been used.

The SBLOCA transient corresponds to the Test 3 (SB-HL-18) conducted at the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA) in the frame of the OECD/NEA ROSA-2 Project. LSTF is a Full-Height Full-Pressure (FHFP) facility designed to simulate the Tsuruga unit II NPP (a 4-loop W-type PWR) with volume and power scaled by a factor of 1/48. Test 3 simulates a PWR 1.5% hot leg SBLOCA with an assumption of the total failure of High Pressure Injection (HPI) system under two different pressure conditions.

Considering the assumptions of this test, some modifications have been performed in the 3-loop PWR TRACE5 model. The simulation results are provided throughout several graphs where the main system variables, such as primary and secondary pressures, collapsed liquid levels and temperatures are shown.

FOREWORD

Thermalhydraulic studies play a key role in nuclear safety. Important areas where the significance and relevance of TH knowledge, data bases, methods and tools maintain an essential prominence are among others:

- assessment of plant modifications (e.g., Technical Specifications, power uprates, etc.);
- analysis of actual transients, incidents and/or start-up tests;
- development and verification of Emergency Operating Procedures;
- providing some elements for the Probabilistic Safety Assessments (e.g., success criteria and available time for manual actions, and sequence delineation) and its applications within the risk informed regulation framework;
- training personnel (e.g., full scope and engineering simulators); and/or
- assessment of new designs.

For that reason, the history of the involvement in Thermalhydraulics of CSN, nuclear Spanish Industry as well as Spanish universities, is long. It dates back to mid 80's when the first serious talks about Spain participation in LOFT-OCDE and ICAP Programs took place. Since then, CSN has paved a long way through several periods of CAMP programs, promoting coordinated joint efforts with Spanish organizations within different periods of associated national programs (i.e., CAMP-España).

From the CSN perspective, we have largely achieved the objectives. Models of our plants are in place, and an infrastructure of national TH experts, models, complementary tools, as well as an ample set of applications, have been created. The main task now is to maintain the expertise, to consolidate it and to update the experience. We at the CSN are aware on the need of maintaining key infrastructures and expertise, and see CAMP program as a good and well consolidated example of international collaborative action implementing recommendations on this issue.

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 a continuous 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/FAP¹ 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 the beginning of this century 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 nuclear safety community during the coming decade. The different series of PKL, ROSA and ATLAS projects are under these premises.

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

¹ 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).

international collaborations. In 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, beyond design accidents), or enlarging code validation and qualification data bases incorporating new information (e.g., multi-dimensional aspects, non-condensable gas effects, passive components).

This NUREG/IA report is part of the Spanish contribution to CAMP focused on:

- Analysis, simulation and investigation of specific safety aspects of PKL2/OECD and ROSA2/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 the experiments and conducting the plant application with the last available versions of NRC TH codes (RELAP5 and/or TRACE). On the whole, CSN is seeking to assure and to maintain the capability of the national groups with experience in the thermalhydraulics analysis of accidents in the Spanish nuclear power plants. Nuclear safety needs have not decreased as the nuclear share of the nations grid is expected to be maintained if not increased during next years, with new plants in some countries, but also with older plants of higher power in most of the countries. This is the challenge that will require new ideas and a continued effort.

Rosario Velasco García, CSN Vice-President
Nuclear Safety Council (CSN) of Spain

TABLE OF CONTENTS

ABSTRACT	iii
FOREWORD	v
LIST OF FIGURES	ix
LIST OF TABLES	ix
EXECUTIVE SUMMARYxi
ACKNOWLEDGMENTS	xiii
ABBREVIATIONS AND ACRONYMS	xv
1 INTRODUCTION	1
2 LSTF DESCRIPTION	3
3 LSTF TRANSIENT	5
3.1 High-Pressure Phase ($P \geq$ about 5MPa).....	5
3.2 Conditioning Phase.....	5
3.3 Low-Pressure Transient Phase ($P \leq$ about 4.5 MPa).....	5
4 THE STANDARD 3-LOOP PWR MODEL	9
5 SCALING CONSIDERATIONS	11
6 RESULTS AND DISCUSSION	15
6.1 Experimental Conditions	15
6.1.1 Common Conditions to High- and low-pressure Transient Phases.....	15
6.1.2 Specific Conditions to High-Pressure Transient Phase.....	15
6.1.3 Specific Conditions to Low-Pressure Transient Phase.....	15
6.2 Steady-State Conditions	16
6.3 Transient.....	17
6.4 System Pressures.....	18
6.5 Break Mass Flow Rate	19
6.6 Vessel Collapsed Liquid Levels.....	20
6.7 Maximum Fuel Rod Surface Temperature.....	21
6.8 Hot and Cold Legs Liquid Levels.....	23
6.9 Emergency Core Cooling Systems Mass Flow Rates.....	24
6.10 U-tubes Collapsed Liquid Level	26
6.11 Secondary Liquid Level	27
6.12 Void Fraction	28
7 CONCLUSIONS	35
8 REFERENCES	37

LIST OF FIGURES

Figure 1	Schematic View of the LSTF Facility	3
Figure 2	LSTF Experimental Data: Primary and Secondary Pressures, CET and PCT	6
Figure 3	Standard 3-Loop PWR Model Nodalization	10
Figure 4	Primary and Secondary Pressures.....	18
Figure 5	Break Mass Flow Rate in 3-Loop PWR Plant	19
Figure 6	Core Collapsed Liquid Level.....	20
Figure 7	Downcomer Collapsed Liquid Level	21
Figure 8	Maximum Fuel Rod Surface and Core Exit Temperatures.....	22
Figure 9	Maximum Fuel Rod Surface Temperature Versus Core Exit Temperature	22
Figure 10	Hot Legs Collapsed Liquid Level in 3-Loop PWR	23
Figure 11	Cold Legs Collapsed Liquid Level in 3-Loop PWR Plant.....	24
Figure 12	High Pressure Injection System Mass Flow Rate in 3-Loop PWR Plant	25
Figure 13	Accumulator Injection System Mass Flow Rate in 3-Loop PWR	25
Figure 14	Low Pressure Injection System Mass Flow Rate in 3-Loop PWR Plant	26
Figure 15	SG U-Tubes Up-Flow Side Collapsed Liquid levels in 3-Loop PWR.....	27
Figure 16	SG Secondary-Side Collapsed Liquid Levels in 3-Loop PWR	28
Figure 17	Void Fraction in 3-Loop PWR Plant at 0 s.....	29
Figure 18	Void Fraction in 3-Loop PWR Plant when PCT Reaches 750 K and HPI Starts.....	30
Figure 19	Void Fraction in 3-Loop PWR Plant when Primary Pressure = 5 MPa	31
Figure 20	Void Fraction in 3-Loop PWR Plant when Second PCT Excursion is Produced.....	32
Figure 21	Void Fraction in 3-Loop PWR Plant at the End of the Transient.....	33

LIST OF TABLES

Table 1	Control Logic and Sequence of Major Events in the Experiment	7
Table 2	Main Characteristics of LSTF, 4-Loop NPP and 3-Loop NPP	13
Table 3	Steady-State Conditions.....	16
Table 4	Chronological Sequence of Events.....	17

EXECUTIVE SUMMARY

The OECD/NEA ROSA-2 project aimed to resolve Light Water Reactor (LWR) thermal-hydraulics safety issues using the Large Scale Test Facility (LSTF) at the Japan Atomic Energy Agency (JAEA). This project is based on experimental and analytical research of thermal hydraulic responses during operational/abnormal transients occurring in the LSTF, which reproduces Tsuruga unit II NPP (4-loop W-type PWR).

This report presents a first approach to simulate a Small Break LOCA (SBLOCA) in a standard 3-loop PWR nuclear power plant using the thermal-hydraulic code TRACE5 patch 2. This transient corresponds to the Test 3 (SB-HL-18) conducted at the LSTF in the frame of the OECD/NEA ROSA-2 Project. The transient simulates a PWR 1.5% hot leg SBLOCA using a sharp edge orifice upwardly mounted with the assumption of the total High Pressure Injection (HPI) system failure under two different pressure conditions. Furthermore, this experiment was used as a counterpart test between LSTF and Primarkreislauf Versuchsanlage facility (PKL). Due to the different working pressures of LSTF and PKL, this experiment is divided into three phases depending on the pressure: high-pressure (LSTF conditions), conditioning and low-pressure (PKL conditions).

An important goal of the test is to determine the relation between the Core Exit Temperature (CET) measured by thermocouples and the fuel rod surface temperature or peak cladding temperature (PCT) in both facilities (LSTF and PKL). The use of these facilities allows testing the effect of high and low-pressure conditions over the relation CET-PCT. On the other hand, the test permits to address scaling problems during the fast primary and secondary depressurizations, core uncovering, core boil-off, etc.

In this frame, Test 3 has been simulated using a standard 3-loop PWR NPP model with TRACE5. Simulation results are provided throughout several graphs including primary and secondary pressures, discharged inventory through the break, collapsed liquid levels (in the pressure vessel, hot and cold legs, U-tubes, etc.) and temperatures.

ACKNOWLEDGMENTS

This paper contains findings that were produced within the OECD-NEA ROSA-2 Project. The authors are grateful to the Management Board of the ROSA-2 Project for their consent to this publication, and thank the Spanish Nuclear Regulatory Body (CSN) and the Asociación Española de la Industria Eléctrica (UNESA) for the technical and financial support.

ABBREVIATIONS AND ACRONYMS

AFW	Auxiliary Feedwater
AIS	Accumulator Injection System
AM	Accident Management
BE	Best Estimate
CAMP	Code Assessment and Management Program
CET	Core Exit Temperature
CPU	Central Processing Unit
CRGT	Control Rod Guide Tubes
CSN	Nuclear Safety Council, Spain
DBE	Design Basis Event
ECCS	Emergency Core Cooling System
HPI	High Pressure Injection
IBLOCA	Intermediate Break Loss-Of-Coolant Accident
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
JC	Jet Condenser
LOCA	Loss-Of-Coolant Accident
LPI	Low Pressure Injection
LSTF	Large Scale Test Facility
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
NV	Normalized to the Steady State Value
PA	Auxiliary Feedwater Pump
PCT	Peak Cladding Temperature
PF	Feedwater Pump
PGIT	Primary Gravity Injection Tank
PJ	High Pressure Charging Pump
PL	High Pressure Injection Pump
PV	Pressure Vessel
PWR	Pressurized Water Reactor
PZR	Pressurizer
RHR	Residual Heat Removal
RV	Relief Valve
SBLOCA	Small Break Loss-Of-Coolant Accident
SG	Steam Generator
SGTR	Steam generator tube rupture
SI	Safety Injection
SNAP	Symbolic Nuclear Analysis Package
SRV	Safety Relief Valve
ST	Storage Tank

1 INTRODUCTION

Thermocouples are worldwide utilized as an important indicator to start an Accident Management (AM) action by detecting Core Exit Temperature (CET) excursion during Nuclear Power Plant (NPP) accidents. Nevertheless, a time delay between the detection of steam superheating by thermocouples and the CET excursion has been found in some tests performed in the frame of the OECD/NEA ROSA-2 Project. Results of these tests [1] have led a safety concern on the reliability of CET response to detect core uncovering and to start effective recovery actions. For this reason, the CET responses during core boil-off under similar conditions to those tests but with different conditions (break location and size) were selected as important safety concerns suitable for study in the OECD/NEA ROSA-2 Project.

In this frame, Test 3 transient (SB-HL-18) [2] defined as a counterpart to Test G7.1 at PKL (Primärkreisläufe Versuchsanlage)-2 Project [3] was designed to clarify the relation between the CET measured by thermocouples and fuel rod surface temperature during a hot leg Small Break LOCA (SBLOCA) and to address the scaling problems through observed thermal-hydraulic phenomena.

Test 3 (SB-CL-18) [1] conducted in the Large Scale Test Facility (LSTF) [4] of the Japan Atomic Energy Agency (JAEA) simulates a PWR 1.5 % hot leg SBLOCA with the assumption of total failure of High Pressure Injection (HPI) system, under two different pressure conditions as a counterpart to PKL-2 Project test.

LSTF is a Full Height Full Pressure (FHFP) facility designed to simulate the Tsuruga unit II Nuclear Power Plant, a 4-loop W-type PWR of 3423 MWt. The volumetric scaling factor is 1/48. The four primary loops of the reference PWR are represented by two equal-volume loops. The core power used to simulate the decay core power is 10 MW, corresponding to 14% of the 1/48 volumetrically scaled reference PWR rated power.

The purpose of this report is to perform a first approach of a SBLOCA application to a NPP using the thermal-hydraulic code TRACE5 patch 2 [5, 6] to simulate a standard 3-loop PWR (representative of Spanish NPP).

Results of the simulation with TRACE5 [5, 6] are shown in several graphs, including primary and secondary pressures, break mass flow rate, collapsed liquid levels (in the pressure vessel, hot and cold legs, U-tubes, etc.) and temperatures. In general, it has been tested that the most important physical phenomena are reproduced using the 3-loop PWR TRACE5 model.

2 LSTF DESCRIPTION

The LSTF simulates a PWR reactor, W-type, of four loops and 3423 MW of thermal power. It is characterized using prototypical-scaled components at full-height, 1/48 scaled volume and full-pressure conditions to the reference PWR (Tsuruga unit II NPP). The four primary loops of the reference PWR are represented by two equal-volume loops with a volume factor of 1/24. Figure 1 shows the scheme of the LSTF.

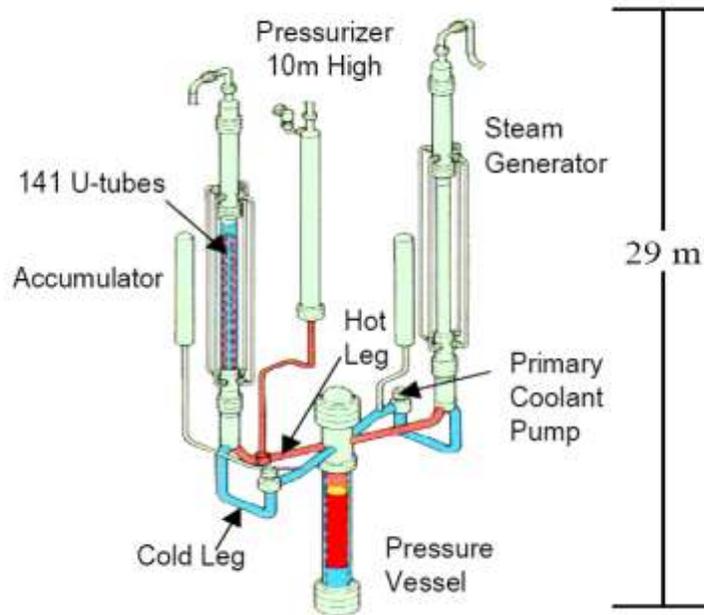


Figure 1 Schematic View of the LSTF Facility

The main LSTF features are:

- Elevations: preserved.
- Volumes: scaled by 1/48 to the reference PWR.
- Flow area: scaled by 1/48 in the pressurized vessel and by 1/24 in the steam generators (SG). The flow area in hot and cold legs is scaled to conserve the ratio of the length (L) to the square of pipe diameter (D) (L/\sqrt{D}).
- Core power: scaled by 1/48 and limited to 14% of the scaled core power of its reference PWR.
- Fuel assembly: designed to be the same than the reference PWR (17 x 17 fuel assembly). The total number of rods was scaled by 1/48. There are 1008 heated rods.

The primary coolant system of LSTF consists of the primary loop A with the pressurizer (PZR) and the symmetrical primary loop B. Both include a primary coolant pump (PC) and 141 U-tubes in each steam generator. On the other hand, the secondary-coolant system includes a jet

condenser (JC), a feed water pump (PF), the auxiliary feedwater pumps (PA) and two secondary systems with a related piping system.

The Emergency Core Cooling System (ECCS) consists of the following sub-systems: the high-pressure charging pump (PJ), the high-pressure injection pump (PH), the accumulators (ACC), the low-pressure injection pump (PL), the Residual Heat Removal (RHR) system and the primary gravity injection tank (PGIT). The primary coolant discharged from the primary system is stored in the break flow Storage Tank (ST).

The LSTF pressure vessel has five regions: the upper head located above the upper core support plate, the upper plenum situated between the upper core support plate and the upper core plate, the active core, the lower plenum and the downcomer annulus region that surrounds the core and the upper plenum. The LSTF vessel is structured with 8 spray nozzles (of 3.4 mm inner-diameter) at the upper head, and 8 Control Rod Guide Tubes (CRGTs) which lead the flow path between the upper head and the upper plenum.

3 LSTF TRANSIENT

Test 3 performed in the LSTF simulates a hot leg PWR SBLOCA 1.5% of the volumetrically-scaled cross-sectional area of the reference PWR (Tsuruga unit II) cold leg. In the LSTF, the break is located in the hot leg without pressurizer (Loop B). This transient is composed of three phases: high-pressure to meet the PWR pressure conditions, conditioning phase, and low-pressure to meet the PKL conditions. The main events produced during each transient phase are described in this section.

3.1 High-Pressure Transient Phase ($P \geq$ about 5 MPa)

The transient starts at time zero with the opening of the break valve in the hot leg of the loop without pressurizer and increasing the rotational speed of the coolant pumps.

A few seconds afterward, the scram signal is generated. This signal produces the initiation of the core power decay curve. This curve simulates the decay power of the fission products and actinides, the delayed neutron fission power and the stored heat release from the nuclear fuel rod. In addition, scram signal produces the initiation of the primary coolant pumps coastdown, turbine trip, closure of Main Steam Isolation Valves (MSIV) and termination of Main Feedwater (MFW). To protect the facility, LSTF Core Protection System automatically decreases the core power when the maximum fuel rod surface temperature reached 958 K.

Immediately after the maximum fuel rod surface temperature (PCT) reaches 750 K, the high-pressure coolant is injected into the pressure vessel upper plenum for core cooling to avoid subcooled water layer being formed at the vessel bottom. This phase is terminated when the primary pressure decreases to about 5 MPa and the break valve is closed.

3.2 Conditioning Phase

In the conditioning phase, the core power is manually changed to a constant value of 1.16 MW (equal to the PKL condition of 455 kW multiplied by 2.55, the scaling ratio of LSTF to PKL), and this value is kept till the end of the test. The primary mass inventory is recovered by the continuous high pressure coolant injection into the pressure vessel upper plenum. When the liquid level recovers up to the hot leg middle level, the HPI terminates.

The relief valves are fully opened in both steam generators (SGs) for secondary depressurization. The Auxiliary Feedwater (AFW) is then injected into the steam generators to avoid a significant liquid level drop. When the primary pressure decreases to about 3.9 MPa, the relief valves are closed and the AFW is terminated in both SGs. This phase finishes when the primary pressure reaches about 4.5 MPa.

3.3 Low-Pressure Transient Phase ($P \leq$ about 4.5 MPa)

In the low-pressure transient phase, the break valve is opened again. Due to the coolant loss from the break, the core uncover is produced. Immediately after the maximum CET measured by thermocouples reaches 623 K, the secondary depressurization is initiated by fully opening the relief valves of the SG as an AM action. The AFW is also injected into both SGs.

The accumulator system is initiated when the primary pressure decreases to 2.6 MPa and is terminated when the primary pressure reaches 1.2 MPa. The LPI system is actuated when the vessel lower plenum pressure decreases to 1 MPa. This phase finishes when a continuous core

cooling by the LPI system is confirmed. Figure 2 shows the experimental data obtained in LSTF for system pressures, CET and PCT excursions, considering the phases of the transient. The complete control logic of the experiment is listed in Table 1.

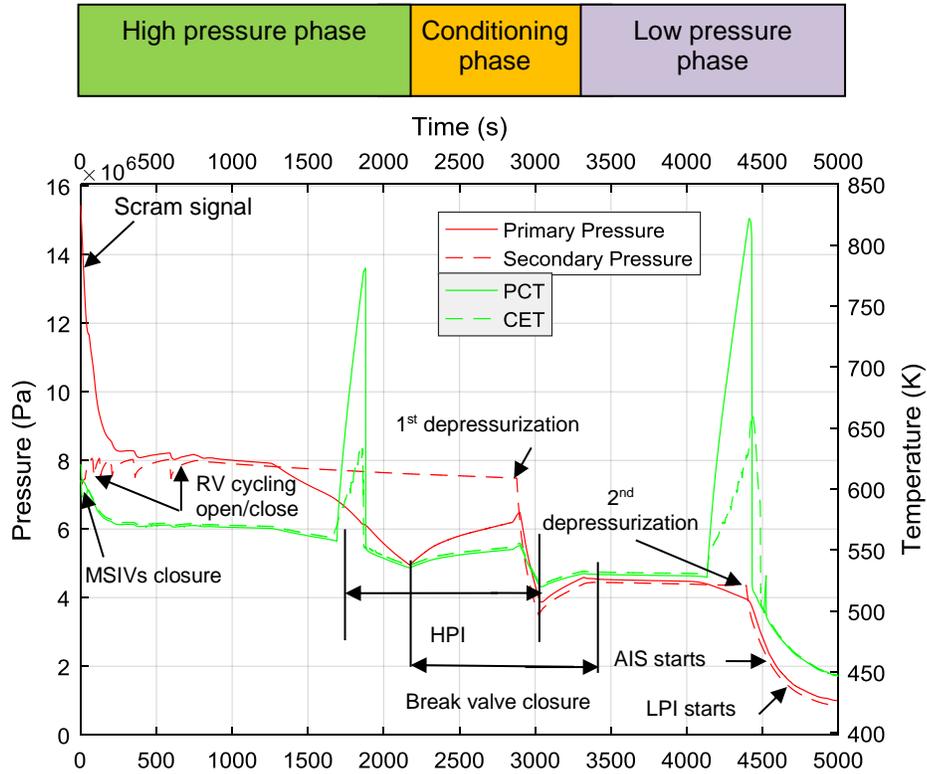


Figure 2 LSTF Experimental Data: Primary and Secondary Pressures, CET and PCT

Table 1 Control Logic and Sequence of Major Events in the Experiment

Break.	Time zero	High-pressure transient phase
Generation of scram signal.	Primary pressure = 12.97 MPa	
Pressurizer heater off.	Generation of scram signal or PZR liquid level below 2.3 m	
Initiation of core power decay curve.	Generation of scram signal	
Initiation of Primary Coolant Pump coastdown.	Generation of scram signal	
Turbine trip (closure of Steam Generators Main Steam Isolation Valve, MSIVs).	Generation of scram signal	
Closure of steam generators (SG) MSIVs.	Generation of scram signal.	
Termination of Steam Generators Main Feed Water (MFW).	Generation of scram signal.	
Generation of Safety Injection (SI) signal.	Primary pressure = 12.27 MPa	
Initiation of High Pressure Injection (HPI) into pressure vessel upper plenum.	Maximum PCT = 750 K	
Initiation of SG secondary-side depressurization by fully opening Relief Valves (RVs) in both loops as AM action.	Maximum CET = 623 K	Low-pressure transient phase
Initiation of Auxiliary Feedwater (AFW) in both loops.	Initiation of AM action.	
Initiation of Accumulator Injection System (AIS) in both loops.	Primary pressure = 2.6 MPa	
Termination of AIS in both loops.	Primary pressure = 1.2 MPa	
Initiation of Low Pressure Injection (LPI) system in both loops.	PV lower plenum pressure = 1 MPa	

4 THE STANDARD 3-LOOP PWR MODEL

The plant of reference used in this study is a standard 3-loop PWR with a thermal power of 2785 MWt. The TRACE5 model includes the pressure vessel, hot and cold legs, pressurizer (in loop 3), reactor coolant pumps, loop seals, HPI, AI and LPI systems, steam generators, etc.

The pressure vessel has been modeled using a 3-D VESSEL component available in TRACE5. The VESSEL component contains 16 axial levels, 4 radial rings and 4 azimuthal sectors. Three rings are used to simulate the upper head, upper plenum, active core and lower plenum of the pressure vessel. The fourth ring is used to simulate the downcomer annulus region. Lower plenum is located between the axial levels 1 and 2. Axial levels 3 to 10 model the active core. Upper plenum is simulated between the axial levels 11 and 14. Levels 15 and 16 model the upper head. The Control Rod Guide Tubes (CRGTs) are modeled using 6 PIPE components: 3 PIPEs allow the flow path between upper plenum and upper head (connect levels 13 to 16), while the other 3 connect the core exit (level 11) with the upper head (level 16).

The power is supplied to the active core using 12 Heat Structure components (HTSTR), which simulate the fuel rods present in the standard 3-loop PWR plant. The core includes 157 fuel assemblies with 17x17 lattice design. A POWER component has been used to manage the power from these HTSTRs.

The hot legs have been modeled with 3 PIPE components connected to the pressure vessel and the U-tubes. In loop 3, the hot leg has a cross flow connecting with the surge line of the pressurizer. A PIPE component is used to simulate the pressurizer. The pressurizer Relief Valves (RVs) and the Power-Operated Relief Valve (PORV) are also considered.

The cold legs have been simulated with 3 PIPE components linked to the pressure vessel, the reactor coolant pumps and the ECCS. In the broken loop (loop 1), the cold leg presents a cross flow connecting a VALVE component joined with a BREAK component to simulate the atmospheric conditions. In this valve, the choked flow coefficients have been fixed to default values (1.0 for subcooled and two-phase coefficients) following the recommendations of the code developers.

The HPI and LPI systems have been simulated with FILL components. The AIS is modeled using a PIPE component with “accumulator” option. The discharge lines are modeled with check VALVE components connected to the cold legs.

The reactor coolant pumps are simulated with PUMP components, considering specific head and torque homologous curves. The reactor coolant pumps are joined to the loop seal. The U-tubes of each steam generator have been modeled using a PIPE component. The secondary side consists of a boiler, a downcomer, a steam separator and the steam lines. The boiler and the downcomer have been simulated with a PIPE component each one, while a SEPD component has been used to model the steam separator. The relief and isolation valves have been modeled using VALVE components. The SG-RV is linked to a BREAK component, while the Main Steam Isolation Valve is connected to a PIPE component, which simulates the steam line header. This PIPE collects the steam from the three SG and conducts it to the turbine. The turbine is represented using two VALVE (turbine control and turbine stop valves) and BREAK components. MFW and AFW are simulated using FILL components connected to a PIPE component. This PIPE is linked to the top of the downcomer. Figure 2 shows the

nodalization of the standard 3-loop PWR TRACE5 model using the Symbolic Nuclear Analysis Package (SNAP) [5] software version 2.1.2.

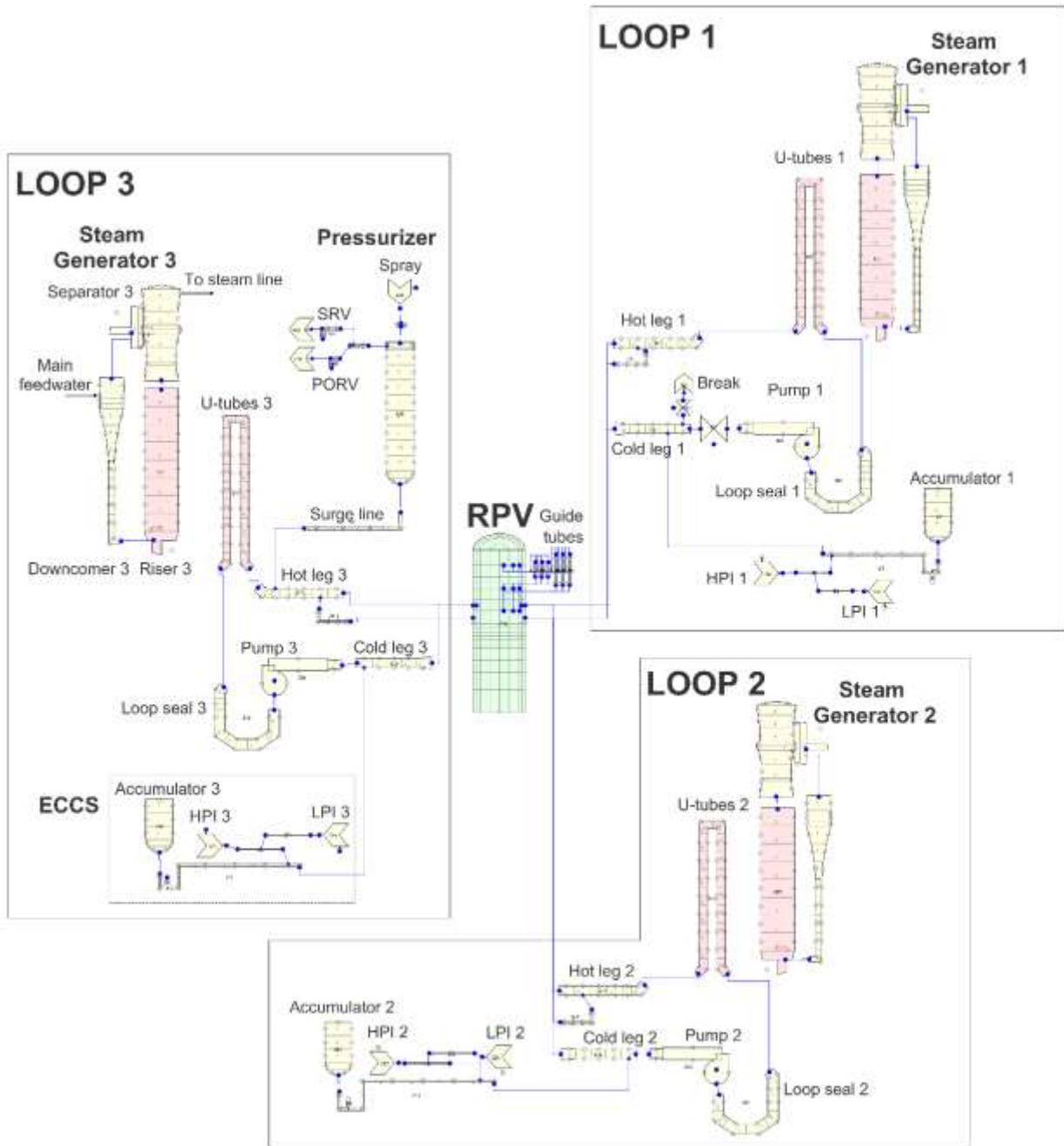


Figure 3 Standard 3-Loop PWR Model Nodalization

5 SCALING CONSIDERATIONS

In SBLOCA scenarios, the most important consideration of the scaling criteria is to preserve both the power and the coolant mass inventory during the transient. The power-to-volume scaling criterion is frequently used to preserve time, power and mass inventory in FHFP facilities regarding their reference NPPs because they have the same fluid properties at full pressure. To perform a power-to-volume analysis of the NPP model, scaling considerations should be made. In this frame, scaling factors between the LSTF and 3-loop NPP must be evaluated to assess the viability of the scaling analyses and can be applied to define the boundary conditions of the scaled model.

As LSTF is a FHFP facility of an actual NPP (Tsuruga unit II) and the scenario is an SBLOCA in the hot leg, the power-to-volume scaling criterion has been chosen to develop a scale-up TRACE5 model of LSTF. This criterion results from the application of conservation equations (1) to (4) under some requirements and implications:

Continuity equation:

$$\frac{\partial \rho}{\partial t} + \frac{\partial(\rho u_i)}{\partial x_i} = 0 \quad (1)$$

Momentum equation:

$$\frac{\partial u_i}{\partial t} + u_j \frac{\partial u_i}{\partial x_j} = F_i - \frac{1}{\rho} \frac{\partial p}{\partial x_i} - \frac{1}{\rho} \frac{\partial(\rho \overline{u'_i u'_j})}{\partial x_j} \quad (2)$$

Energy equation:

$$\rho \left(\frac{\partial h}{\partial t} + u_j \frac{\partial h}{\partial x_j} \right) = - \frac{\partial(\rho C_p \overline{u'_j T'})}{\partial x_j} + T \beta \left(\frac{\partial p}{\partial t} + u_j \frac{\partial p}{\partial x_j} \right) + \dot{q}^m \quad (3)$$

State equation:

$$\rho = \rho(h, p) \quad (4)$$

where ρ is the density, u is the velocity, x is the coordinate, t is the time, p is the pressure, F is the friction coefficient, h is the enthalpy, C_p is the specific heat, T is the temperature, β is the thermal expansion coefficient and \dot{q}^m is the power density.

Substituting the next dimensionless parameters (denoted by an asterisk) in equations (1) to (4):

$$x_i^* = \frac{x_i}{l_0}, \quad u_i^* = \frac{u_i}{u_0}, \quad t^* = \frac{t u_0}{l_0}, \quad F_i^* = \frac{F_i}{g}, \quad p^* = \frac{p}{\Delta p_0}, \quad \rho^* = \frac{\rho}{\rho_0}, \quad T^* = \frac{T}{\Delta T_0},$$

$$h^* = \frac{h}{C_p \Delta T_0}, \quad \beta^* = \beta \Delta T_0$$

gives a set of nondimensionalized equations (5) to (8):

$$\frac{\partial \rho^*}{\partial t^*} + \frac{\partial(\rho^* u_i^*)}{\partial x_i^*} = 0 \quad (5)$$

$$\frac{\partial u_i^*}{\partial t^*} + u_j^* \frac{\partial u_i^*}{\partial x_j^*} = \frac{gd_0}{u_0^2} F_i^* - \frac{\Delta p_0}{\rho_0 u_0^2} \frac{1}{\rho^*} \frac{\partial p^*}{\partial x_i^*} - \frac{1}{\rho^*} \frac{\partial(\rho^* \overline{u_i' u_j'^*})}{\partial x_j^*} \quad (6)$$

$$\frac{\partial h^*}{\partial t^*} + u_j^* \frac{\partial h^*}{\partial x_j^*} = \frac{\partial(\rho^* \overline{u_j' T'^*})}{\partial x_j} + \frac{\Delta p_0}{\rho_0 C_p \Delta T_0} \beta^* T^* \left(\frac{\partial p^*}{\partial t^*} + u_j^* \frac{\partial p^*}{\partial x_j^*} \right) + \frac{\dot{q}^m l_0}{\rho_0 u_0 C_p \Delta T_0} \quad (7)$$

$$\rho^* = \rho(h^*, p^*) \quad (8)$$

These equations contain other dimensionless parameters such as $\frac{u_0^2}{gd_0}$, $\frac{\Delta p_0}{\rho_0 u_0^2}$ and $\frac{\dot{q}^m l_0}{\rho_0 u_0 C_p \Delta T_0}$. The first two parameters are Froude and Euler numbers, respectively. The third is known as heat source number following Ishii and Kataoka terminology.

Power-to-volume scaling method requires that all dimensionless parameters in equations (5) to (8) have to be equal in the LSTF model and the scale-up LSTF model.

$$\frac{P_{scaled-up}}{P_{LSTF}} = \frac{\rho_{scaled-up}}{\rho_{LSTF}} = \frac{l_{scaled-up}}{l_{LSTF}} = \frac{t_{scaled-up}}{t_{LSTF}} = 1 \quad (9)$$

Considering that similarity between both systems has been achieved, the following power-to-volume relations are obtained:

$$\frac{\phi_{scaled-up}}{\phi_{LSTF}} = \frac{Q_{scaled-up}}{Q_{LSTF}} = \frac{V_{scaled-up}}{V_{LSTF}} = \frac{A_{scaled-up}}{A_{LSTF}} = K_v \quad (10)$$

being ϕ power, Q mass flow rate, V volume, A area and K_v the volumetric scaling factor.

Furthermore, in the scale-up model, the Froude number is conserved in horizontal components. It implies varying the diameter and length of these components. Trying to conserve the Froude number, from the scale-up mass flow rate calculated as Eq. (11), the scale-up diameter, D , can be obtained as Eq. (12):

$$U \frac{\pi \cdot D^2}{4} \cdot \rho = u \frac{\pi \cdot d^2}{4} \cdot \rho \cdot K_v \quad (11)$$

$$D = d \cdot K_v^{2/5} \quad (12)$$

where U is velocity in the scale-up model, ρ is the coolant density and d is LSTF diameter. Furthermore, from the volume equation Eq. (13) and trying to conserve the Froude number, lengths of the scale-up piping system are obtained as Eq. (14):

$$\frac{\pi \cdot D^2}{4} \cdot L = \frac{\pi \cdot d^2}{4} \cdot l \cdot K_v \quad (13)$$

$$L = l \cdot K_v^{1/5} \quad (14)$$

being l and L , LSTF and scale-up length, respectively.

LSTF represents a 4-loop PWR W-type (Tsuruga unit II NPP) [4] with a volume and power scale of 1/48, while the height of all the components corresponds to the real plant. Flow areas are scaled by 1/48 in the pressure vessel and by 1/24 in the steam generators. The flow area in hot and cold legs is scaled to conserve the ratio of the length (L) to the square of pipe diameter (D) (L/\sqrt{D}). Due to the LSTF is a Full Height Full Pressure facility, the time scale of physical phenomena is maintained. Considering the nominal power and volume of LSTF, Tsuruga unit II and the 3-loop NPP, the scaling ratios between them can be obtained. Table 2 summarizes the main characteristics of LSTF, 4-loop and 3-loop NPP models.

Table 2 Main Characteristics of LSTF, 4-Loop NPP and 3-Loop NPP

Parameter	LSTF	Reference 4-loop NPP	3-loop NPP	K_v (3-loop/LSTF)
Primary pressure (MPa).	15.5	15.5	15.5	1
Core power (MW).	10	3423	2785	39.0 (considering 14% of 2785 MW).
Number of loops.	2	4	3	
Number of fuel assemblies		193	157	
Fuel assembly array.		17x17	17x17	
Core height (m).	3.66	3.66	3.66	1
Vessel volume (m ³).	2.75	131.7	106	38.54
Number of fuel rods.	1008	50952	41447	41.11
Volume of pressurizer (m ³).	1.2	51.0	39.7	33.08
Pressurizer heaters power (kW).	124	1800	1400	
Average length U-tubes (m).	20.2	20.2	20.2	1
Hot leg inner diameter (m).	0.207	0.737	0.737	
Hot leg L/\sqrt{D} .	8.11	8.11	8.11	1
Cold leg inner diameter (m).	0.207	0.698	0.698	

In terms of pressure and heights, the relation between LSTF and its reference NPP is 1/1, so the initial value for pressures and temperatures in the 3-loop PWR model is also the same than in the experiment performed in LSTF. The working pressure of accumulators is the same in all cases.

Due to the LSTF is a full height full pressure facility, the time scale of physical phenomena must be maintained. However, in the 3-loop PWR TRACE5 model has differences in the reactor technology, which must be considered:

- Number of loops: LSTF simulates a 4-loop PWR collapsed on 2-loops with a volumetric scaling factor of 1/24, while the NPP is a 3-loop design.
- ECCS: In the LSTF, the ECCS consists of HPI, AIS and LPI per each loop. The 3-loop PWR TRACE5 model has been adapted to reproduce the same systems. In Test 3 reproduced in the LSTF a manual HPI injection is produced in the PV upper plenum. The AIS and LPI have been considered in the 3-loop PWR model. Mass flow rates have been scaled according to the power to volume factor between LSTF and 3-loop PWR NPP.
- Break: The break size reproduced in Test 3 has been scaled applying the power to volume strategy.

6 RESULTS AND DISCUSSION

6.1 Experimental Conditions

6.1.1 Common Conditions to High- and Low-Pressure Transient Phases

- 1) Break size is equivalent to 1.5% cold leg break by using a sharp-edge orifice upwardly mounted in the hot leg of loop B in the LSTF and in loop 1 in the standard PWR model.
- 2) Loss of off-site power concurrent with the scram.
- 3) Total failure of high pressure injection (HPI) system under two different pressure conditions.

6.1.2 Specific Conditions to High-Pressure Transient Phase

- 5) The HPI coolant injection is performed into the PV upper plenum for core cooling. It is done to avoid subcooled water layer being formed at the PV bottom immediately after the maximum fuel rod surface temperature reaches 750 K.

6.1.3 Specific Conditions to Low-Pressure Transient Phase

- 6) The SG secondary-side depressurization is initiated by fully opening the RVs of both SGs as AM action. Immediately later, the maximum core exit temperature measured by CETs reaches 623 K.
- 7) The AFW is initiated in both loops simultaneously with the AM action.
- 8) Accumulators (ACC) and low-pressure injection (LPI) systems are activated. The ACC system is terminated when primary pressure decreases to 1.2 MPa, that is slightly before the LPI actuation.

6.2 Steady-State Conditions

A steady-state case of 1000 s has been run using the 3-loop PWR model. The conditions achieved at the end of the steady state are listed in Table 3. In general, the simulation results for the main system variables are in good agreement with the experiment performed in the LSTF.

Table 3 Steady-State Conditions

Item	3-loop PWR model
Core Power (MW)	2785
Hot leg Fluid Temperature (K)	604
Cold leg Fluid Temperature (K)	570
Mass Flow Rate (kg/s)	4416
Pressurizer Pressure (MPa)	15.5
Pressurizer Liquid Level (m)	5
SG Secondary-side Pressure (MPa)	6.71
SG Secondary-side Liquid Level (m)	10.1
Steam Flow Rate (kg/s)	483.24
Main Feedwater Flow Rate (kg/s)	483.24

6.3 Transient

Table 4 lists the chronology of events during the transient performed in the NPP model.

Table 4 Chronological Sequence of Events

Event	3-loop model Time (s)	Phase
Break valve open.	0	High-pressure transient
Scram signal.	20	
Closure of SG MSIVs.	20	
Initiation of coastdown of primary coolant pumps.	20	
Termination of SG main feedwater.	20	
Initiation of core power decay.	20	
Primary pressure became lower than SG secondary side pressure.	1320	
Start of increase in fuel rod surface temperature.	1560	
Maximum fuel rod surface temperature reached 750 K.	1786	
Initiation of HPI into PV upper plenum.	1776	
Break valve closure.	2100	Conditioning
Termination of HPI system into PV upper plenum.	2600	
Initiation of SG secondary-side depressurization by fully opening RVs in both loops.	2796	
Termination of SG secondary-side depressurization.	2982	
Break valve open again.	3296	Low-pressure transient
Start of increase in fuel rod surface temperature.	4050	
Primary pressure became lower than SG secondary side pressure.	4167	
Maximum core exit temperature = 623 K.	4298	
Initiation of SG secondary-side depressurization by fully opening RVs in both loops as AM action.	4374	
Maximum fuel rod surface temperature.	4382	
Initiation of ACC (AIS) system in both loops.	4510	
Termination of ACC system in both loops.	4992	
Initiation of LPI system in both loops.	4678	

In this section, several graphs with the main variables of the system are shown. These are obtained with the 3-loop NPP model.

6.4 System Pressures

Primary and secondary pressures obtained with the 3-loop PWR model are presented in Figure 4. In the high-pressure phase (until 2170 s), the primary pressure starts to decrease at time zero when the break valve is opened causing the scram signal. It implies the closure of the MSIV and the primary coolant pumps coastdown. The SG secondary-side pressure rapidly increases after the closure of MSIVs. From this moment on, the secondary-side pressure starts to oscillate by opening and closing the RVs of SGs. The primary pressure becomes lower than the secondary-side pressure at about 1250 s, soon after the break flow turns into single-phase steam (Figure 5). When the primary pressure reaches about 5 MPa, the break valve closes and this phase ends.

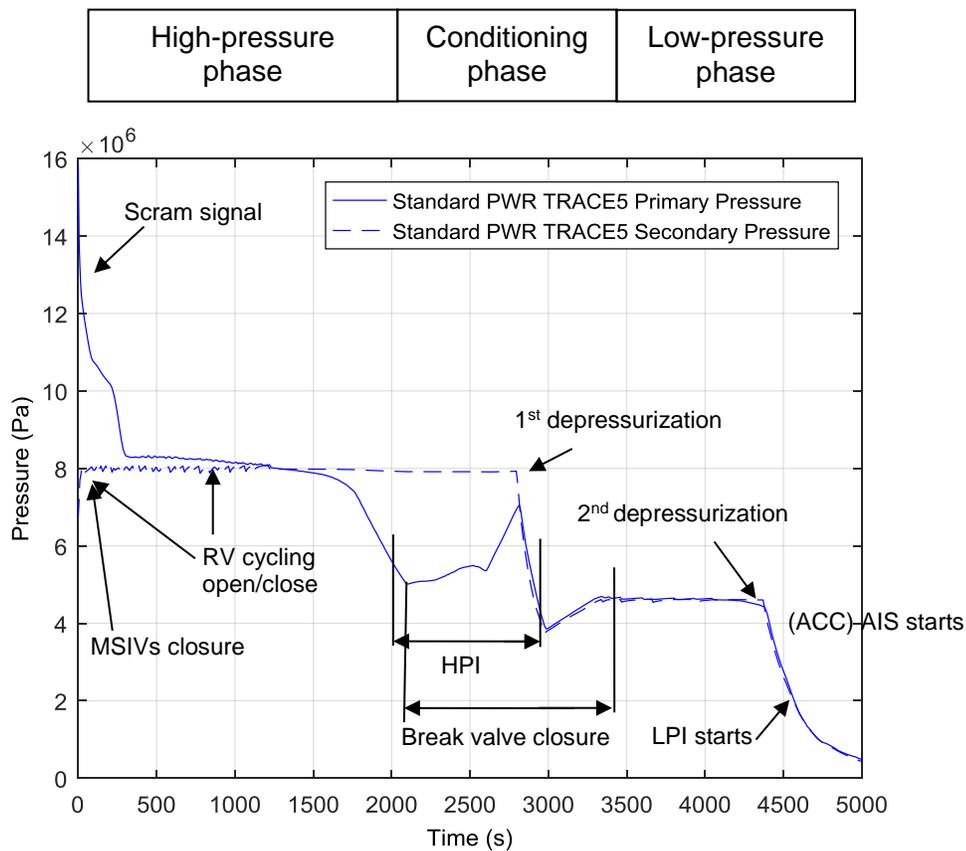


Figure 4 Primary and Secondary Pressures

In the conditioning phase (until 3400 s), the core power is fixed to a constant value until the end of the transient. Furthermore, the break valve is closed and the HPI system injects coolant into the PV upper plenum. In these conditions, the primary pressure increases up to 6.5 MPa (at 2900 s). At this time, hot leg levels reach mid-loop and the HPI injection ends. The secondary-side depressurization is produced and the AFW injection starts. Following the secondary pressure drop, the primary pressure decreases to 3.9 MPa. Under these conditions, depressurization is finished, the RVs are closed and the AFW injection ends. After that, the

primary and secondary pressures increase again up to about 4.5 MPa. The low-pressure phase starts by opening the break valve again. The primary pressure becomes lower than the secondary-side pressure at about 4100 s, shortly after the core boil-off begins. Immediately after, the maximum CET reaches 623 K (Figure 8) and secondary-side depressurization is initiated by fully opening the RVs as the AM action and continues until the end of the test. The primary pressure decreases after the AM action following the secondary-side pressure and results in the actuations of the ACC and LPI systems. In general, the 3-loop PWR TRACE5 model reproduces similar behavior than the experiment.

6.5 Break Mass Flow Rate

Figure 5 shows the mass flow rate through the break obtained with the model. As it can be seen, in the high-pressure phase, the break flow rate sharply decreases when the break flow turns from single-phase liquid to two-phase flow. At 1250 s the break flow turns from two-phase flow to single-phase vapor.

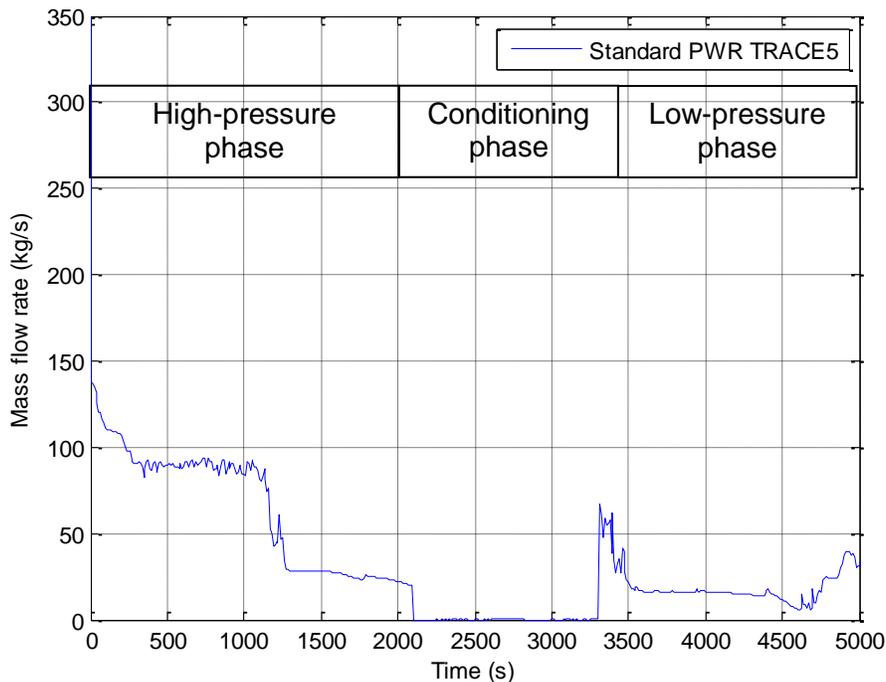


Figure 5 Break Mass Flow Rate in 3-Loop PWR Plant

The break mass flow rate obtained with TRACE5 has been adjusted using the discharge coefficients for single-phase liquid and for two-phase liquid vapor [5, 6] fixed to 1.0 (default value).

6.6 Vessel Collapsed Liquid Levels

Figures 6 and 7 show the collapsed liquid levels in the core and downcomer of the PV. As it can be seen, in the high-pressure phase the core liquid level starts to drop. The core uncovering takes place after the primary pressure becomes lower than the secondary-side pressure. The collapsed liquid level still drops to about 1/3 of the active core length until 1870 s, even after the initiation of high-pressure coolant injection. The whole core is quenched at 2100 s.

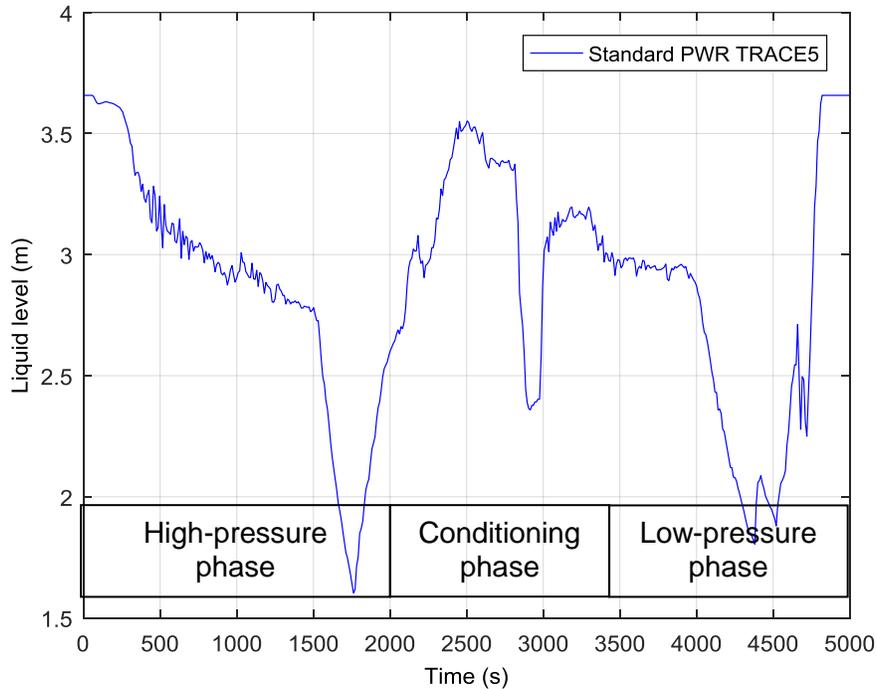


Figure 6 Core Collapsed Liquid Level

In the conditioning phase, the liquid level is recovered until the termination of high-pressure coolant injection. A second liquid level drop is produced in the core during the secondary-side depressurization from 2880 till 3028 s, suggesting that coolant flashing occurs. When the break valve is opened again at 3300 s (low-pressure phase), the core liquid level starts to drop at about 3910 s, and the core uncovering begins before the primary pressure becomes lower than the secondary-side pressure. The core liquid level starts to recover after the primary depressurization produced by the secondary-side depressurization (as the AM action starts immediately after the maximum core exit temperature reaches 623 K).

Regarding the downcomer, the liquid level gradually decreases between 400 and 1000 s and starts to drop again after about 1250 s. It starts to recover at 1910 s after the initiation of the high-pressure coolant injection. In the conditioning phase, a temporary liquid level drop happens as in the core, due to the secondary-side depressurization. In the low-pressure phase, the liquid level starts to steeply recover at 4530 s due to the AIS mass flow rate (Figure 7).

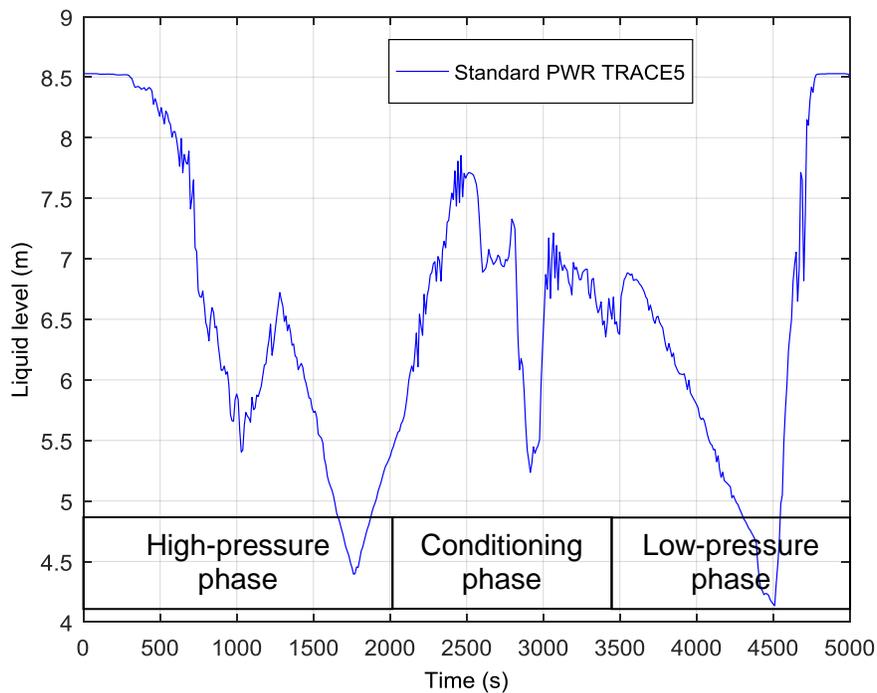


Figure 7 Downcomer Collapsed Liquid Level

6.7 Maximum Fuel Rod Surface Temperature

Figure 8 shows the Core Exit Temperature (CET) and the maximum Peak Cladding Temperature (PCT). In the high-pressure phase, CET and PCT start to increase at 1600 s. In the low-pressure phase, the initiation of the CET and PCT excursions are produced at about 4000 s. In both cases (high and low-pressure phases) a slight delay between the CET and the PCT excursions is observed. Figure 9 allows clarifying the relation between the CET and PCT.

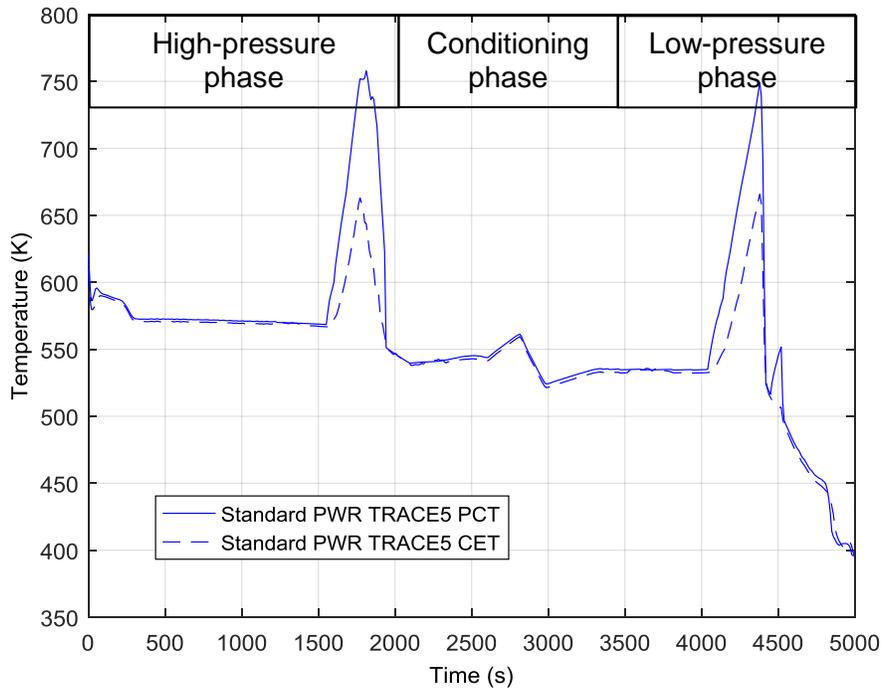


Figure 8 Maximum Fuel Rod Surface and Core Exit Temperatures

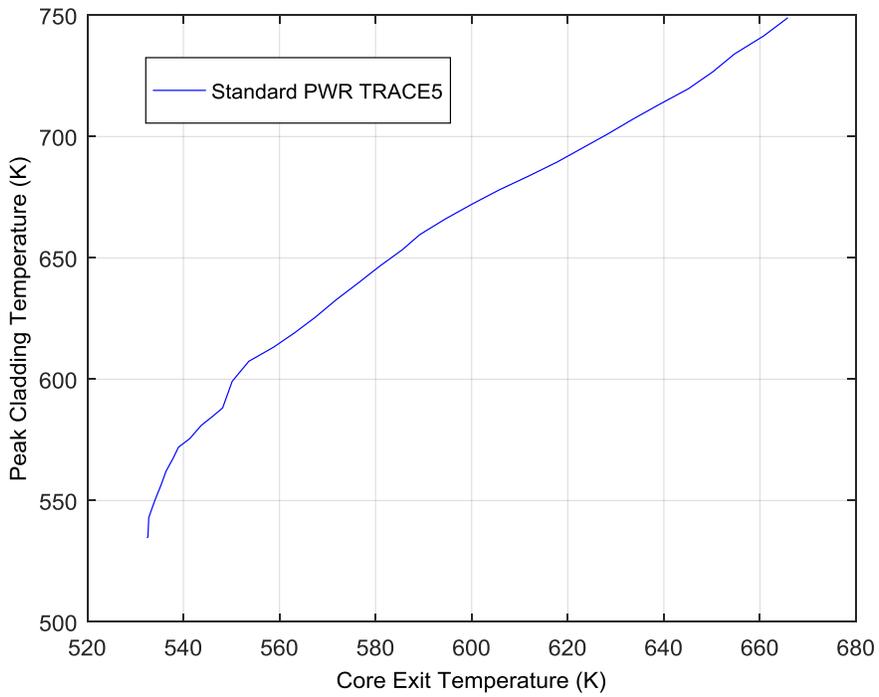


Figure 9 Maximum Fuel Rod Surface Temperature Versus Core Exit Temperature

6.8 Hot and Cold Legs Liquid Levels

Figure 10 shows the collapsed liquid level in the hot legs obtained with the model. As it can be seen, the liquid levels are similar in all the legs. In the high-pressure phase, the liquid level is kept at around 3/4 to 1/2 of the inner diameter until 1310 s when the break flow turns from two-phase flow to single-phase vapor and the primary pressure starts to decrease. The hot legs become empty at about 1400 s.

In the conditioning phase, the hot leg liquid levels are recovered and reach the middle level after the core refilling. A temporary level drop appears during the secondary-side depressurization, as it occurs in the PV liquid levels (Figures 6 and 7). In the low-pressure phase, the liquid level starts to decrease just after the break valve opening. The hot legs become empty of liquid at about 3600 s and start to recover the level at 4530 s due to the AIS coolant.

The cold leg liquid levels obtained with the model are shown in Figure 11. All the cold legs are almost empty at the same time, at about 1500 s. During the coolant injection by the AIS and LPI systems, the liquid level is recovered.

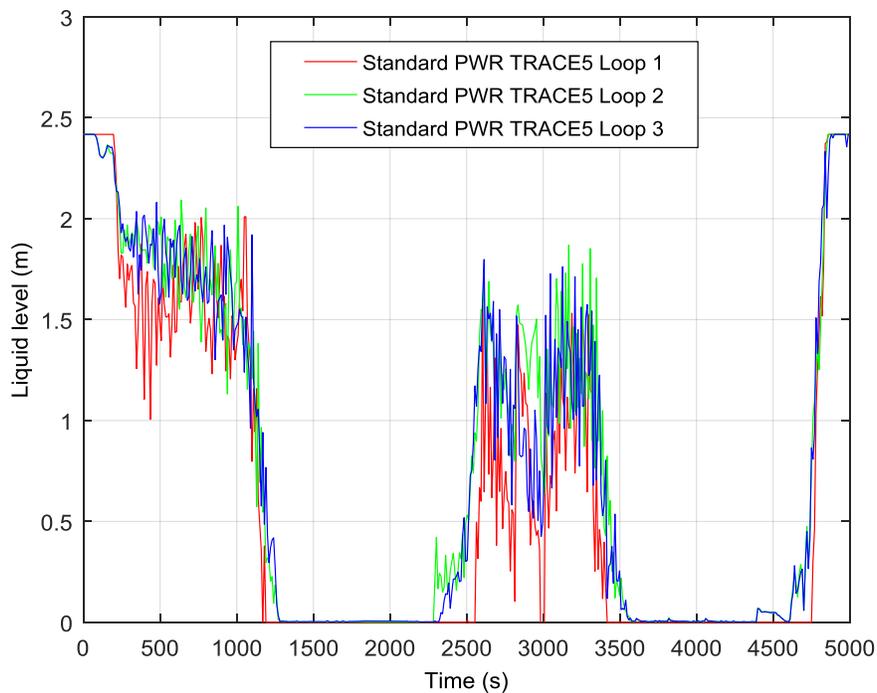


Figure 10 Hot Legs Collapsed Liquid Level in 3-Loop PWR

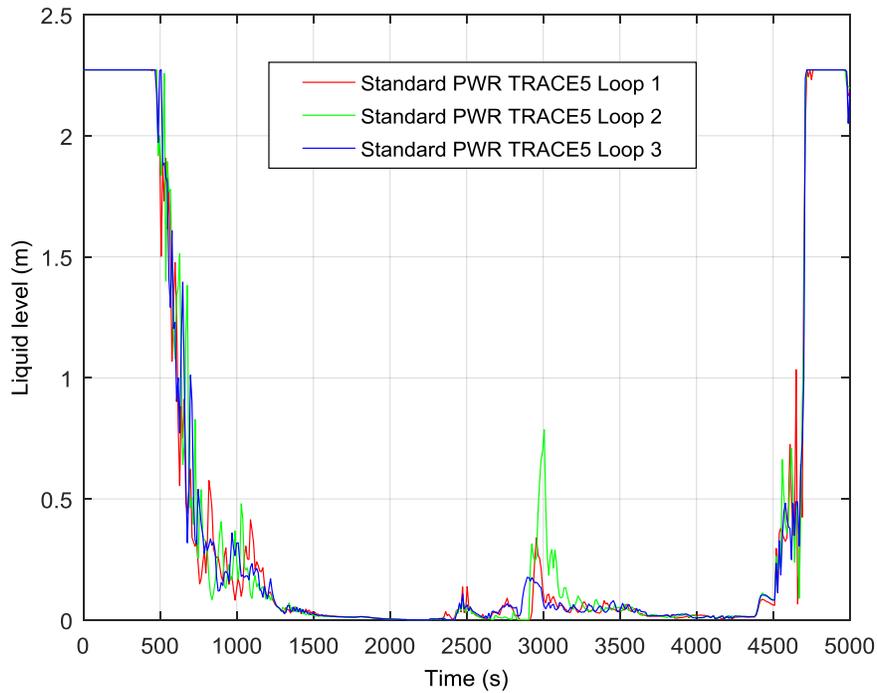


Figure 11 Cold Legs Collapsed Liquid Level in 3-Loop PWR Plant

6.9 Emergency Core Cooling Systems Mass Flow Rates

The high-pressure coolant injection starts immediately after the maximum fuel rod surface temperature reaches 750 K (Figure 8). The HPI is terminated when the hot leg liquid level is recovered to around the middle level. Figure 12 shows the high-pressure injection mass flow rate using the volumetric scaling factor. The AIS mass flow rate is initiated at about 4500 s in the three loops and is terminated in all the loops when the primary pressure decreases to 1.2 MPa (Figure 13). When the primary pressure is lower than 1 MPa, the LPI system is activated in the three loops (Figure 14).

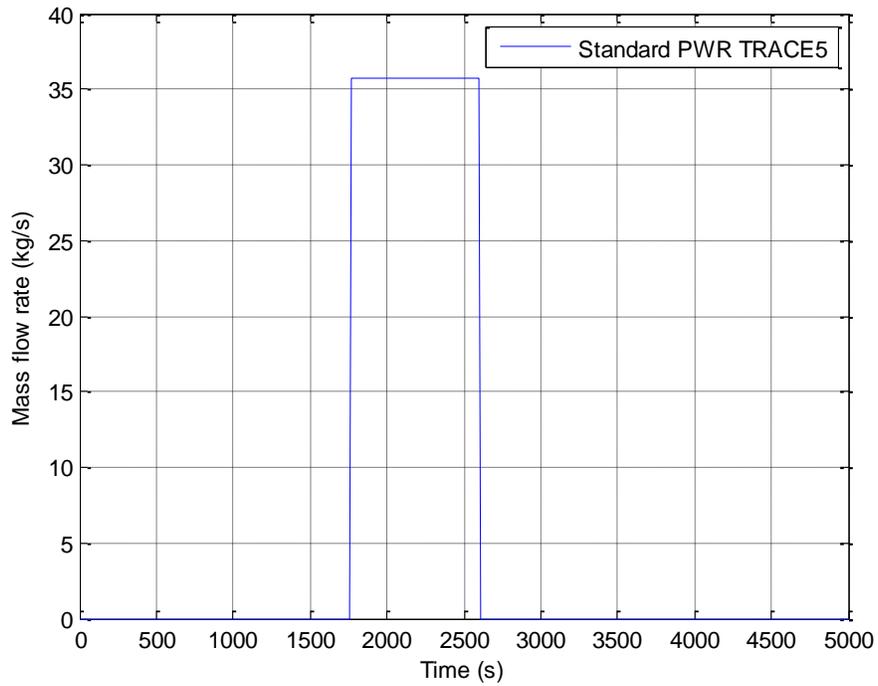


Figure 12 High Pressure Injection System Mass Flow Rate in 3-Loop PWR Plant

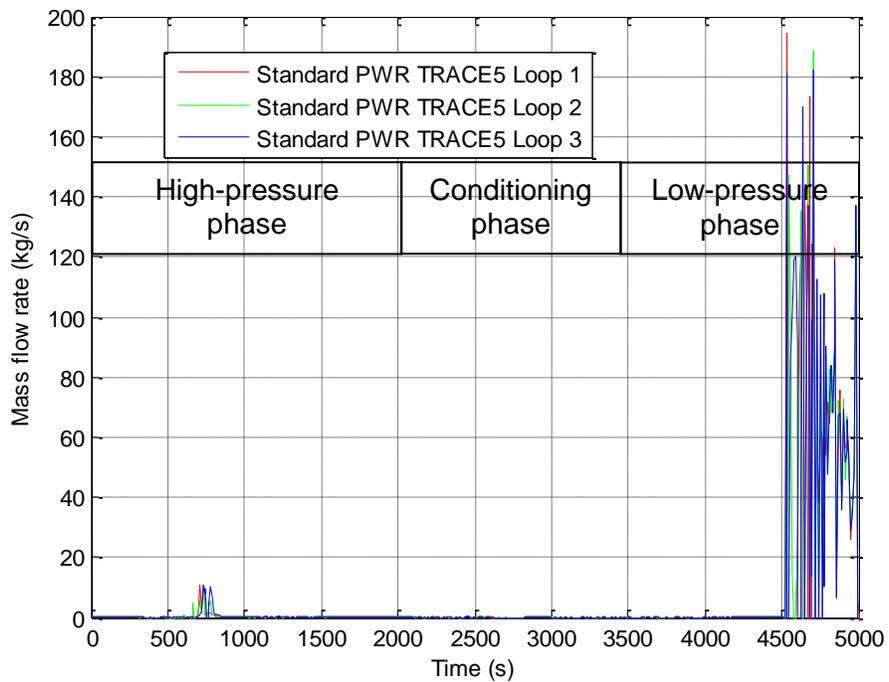


Figure 13 Accumulator Injection System Mass Flow Rate in 3-Loop PWR

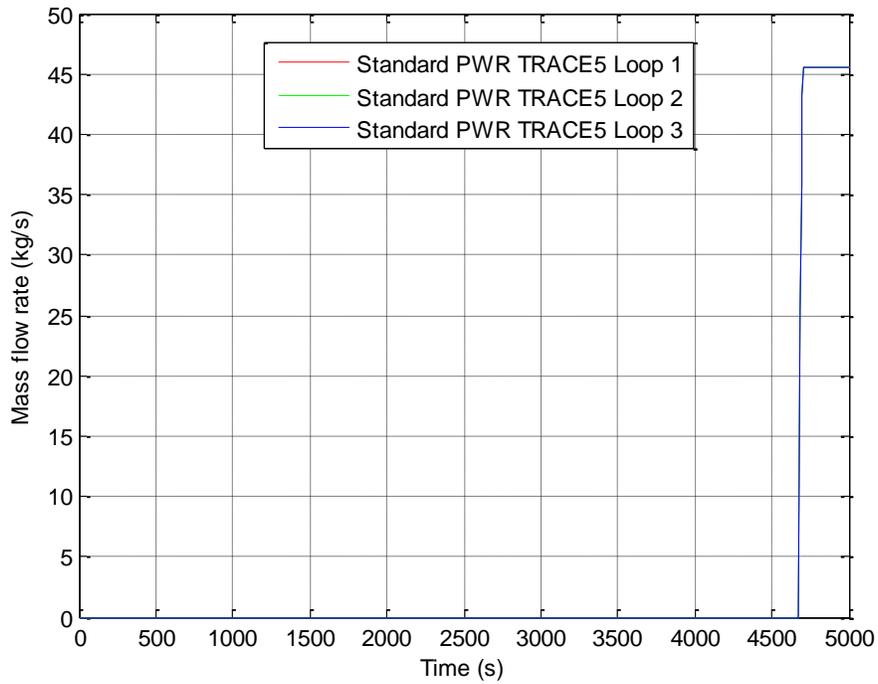


Figure 14 Low Pressure Injection System Mass Flow Rate in 3-Loop PWR Plant

6.10 U-tubes Collapsed Liquid Level

The U-tube collapsed liquid levels are shown in Figure 15. As it can be observed, the U-tubes of three SGs are emptied at 1100 following the same trend than in the experiment. During the second opening of the break valve, a sudden increase of the U-tubes liquid level is observed in the three SGs. Finally, at 4800 s, a second increasing of liquid level is observed due to the injection of coolant from the AIS and the LPIS.

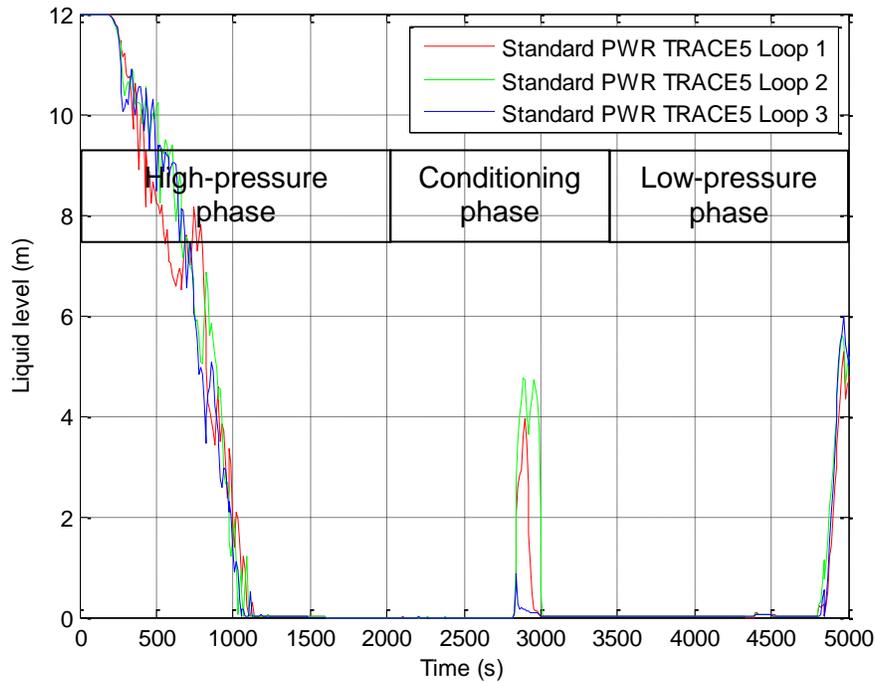


Figure 15 SG U-Tubes Up-Flow Side Collapsed Liquid Levels in 3-Loop PWR

6.11 Secondary Liquid Level

Figure 16 shows the SG secondary-side collapsed liquid levels. In the high-pressure phase (0 to 2170 s), the liquid levels start to increase due to the MSIVs closure. In the conditioning phase (2170 to 3300 s), the liquid levels are maintained constant between 9.5 and 9.8, which is enough to cover the long U-tubes, until 2880 s, when the secondary-side depressurization begins. Due to the secondary-side depressurization, the liquid levels drop. In the low-pressure phase (3300 to 5000 s), the liquid levels start to decrease again at 4394 s when the second secondary-side depressurization is started.

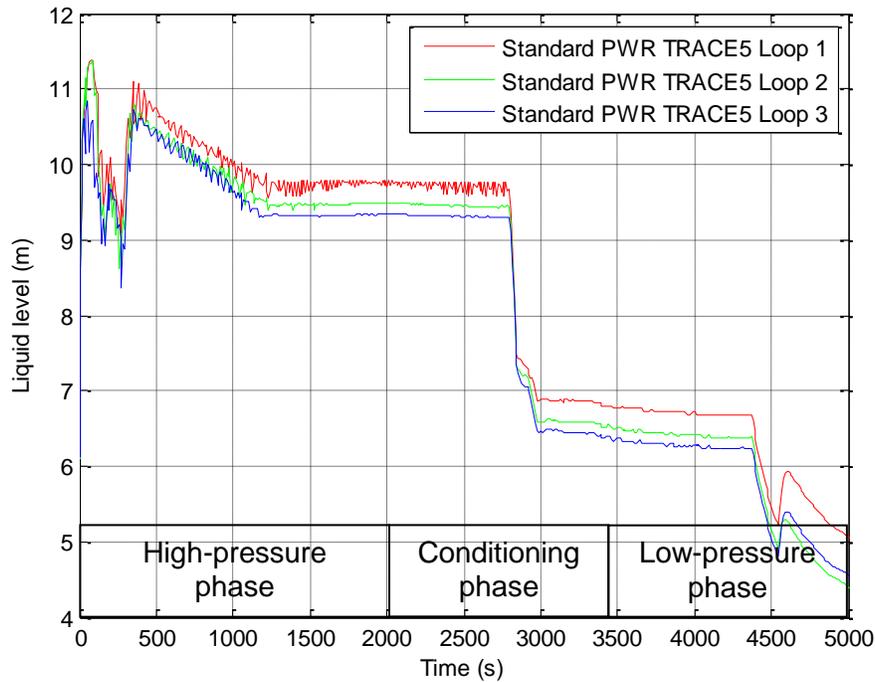


Figure 16 SG Secondary-Side Collapsed Liquid Levels in 3-Loop PWR

6.12 Void Fraction

Figures 17, 18, 19, 20 and 21 show the void fraction achieved along the transient when important events happen using the TRACE5 model. Figure 17 shows the void fraction at the start of the test. At this time, primary and secondary-sides are full of liquid. In Figure 18, it is shown the void fraction when PCT reaches 750 K and HPI starts. Figure 19 shows the void fraction when the primary pressure drops to 5 MPa and the break valve is closed. The situation when the second maximum of the PCT is reached is shown in Figure 20. As it can be seen, in these figures the situation is similar: the pressurizer is empty, and the liquid is located in loop seals, accumulators, the bottom part of the PV and SGs. Figure 21 shows the void fraction at the end of the transient. At this time, the SGs are almost empty and the AIS is completely empty.

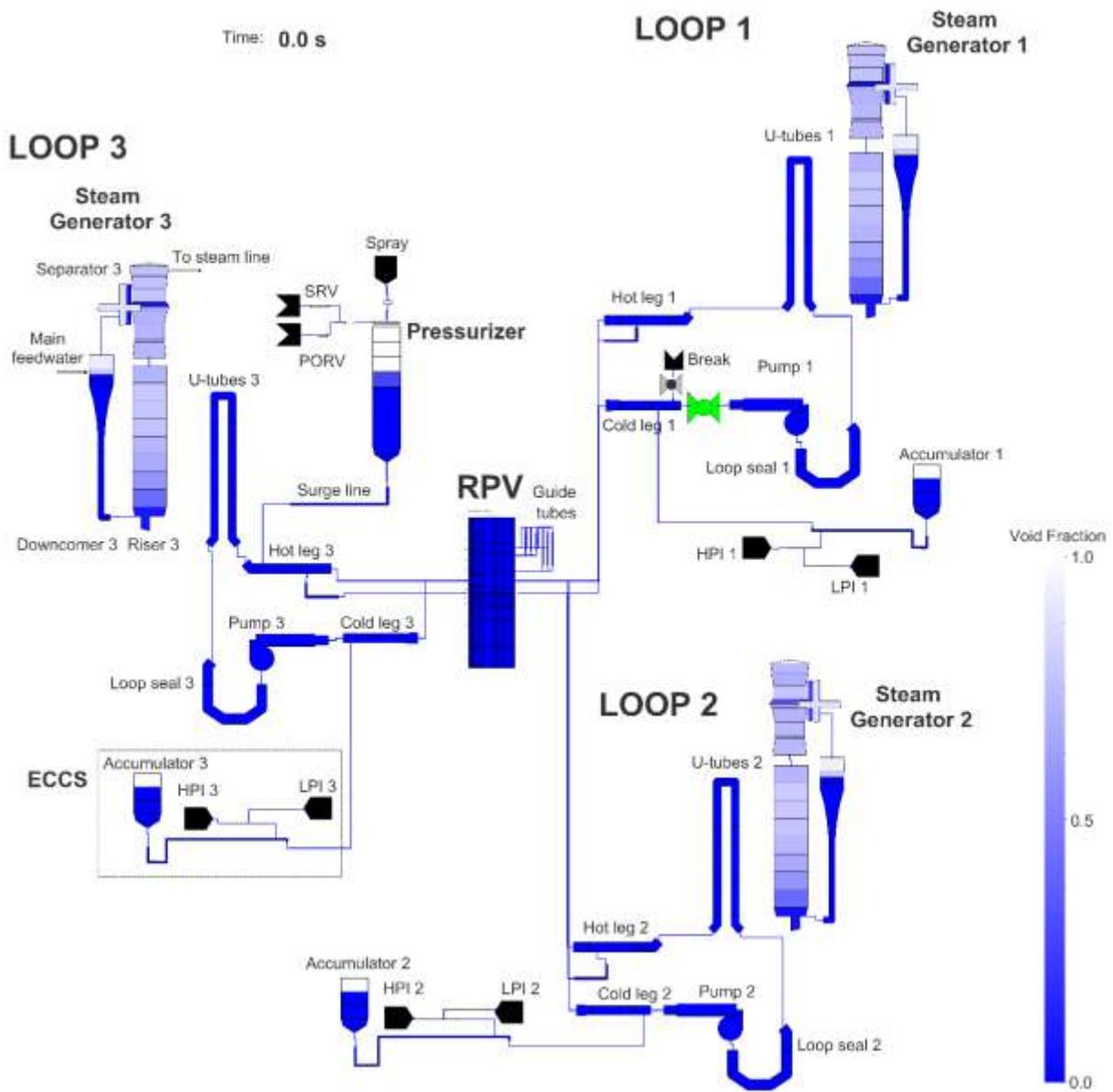


Figure 17 Void Fraction in 3-Loop PWR Plant at 0 s

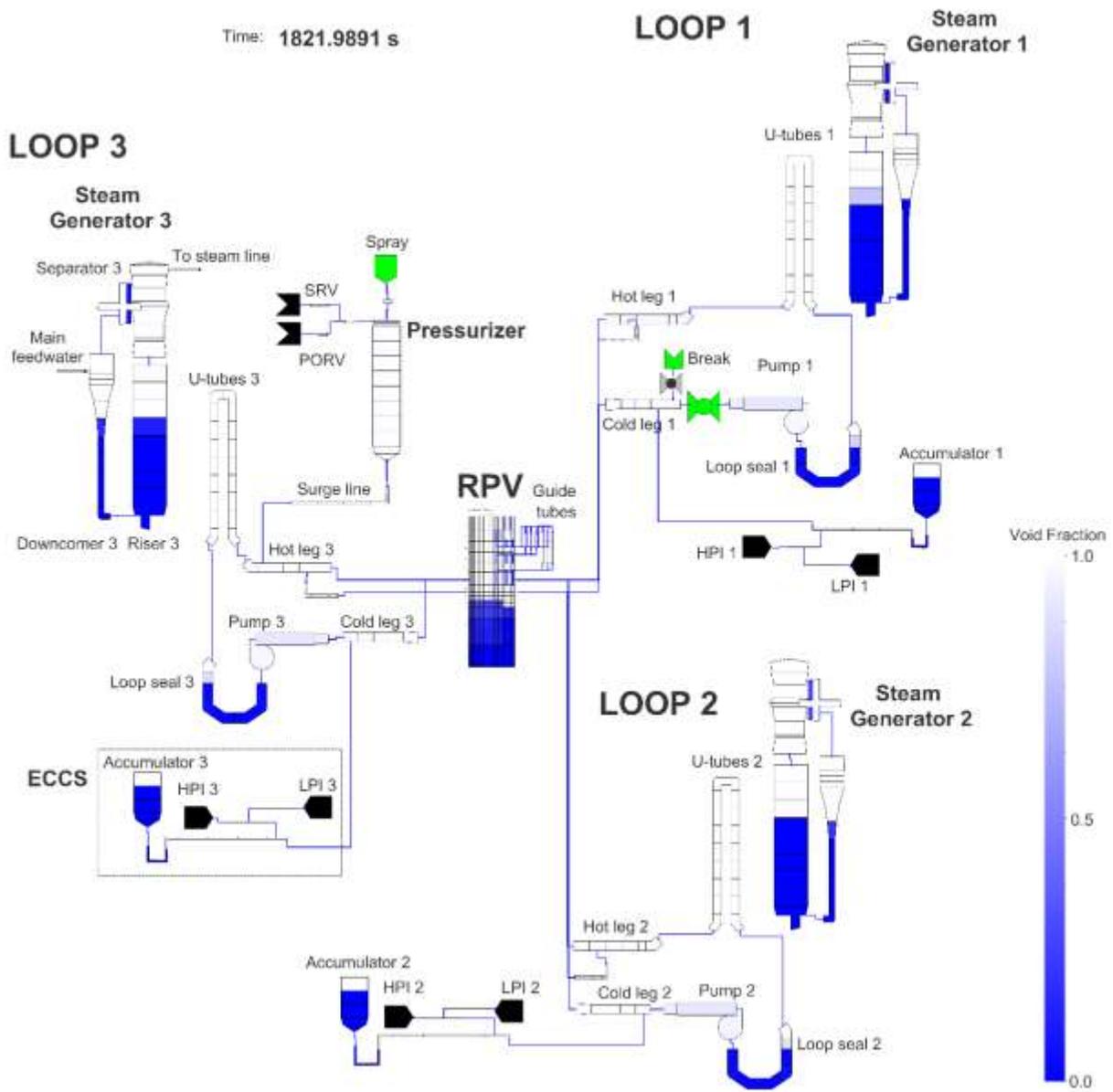


Figure 18 Void Fraction in 3-Loop PWR Plant when PCT Reaches 750 K and HPI Starts

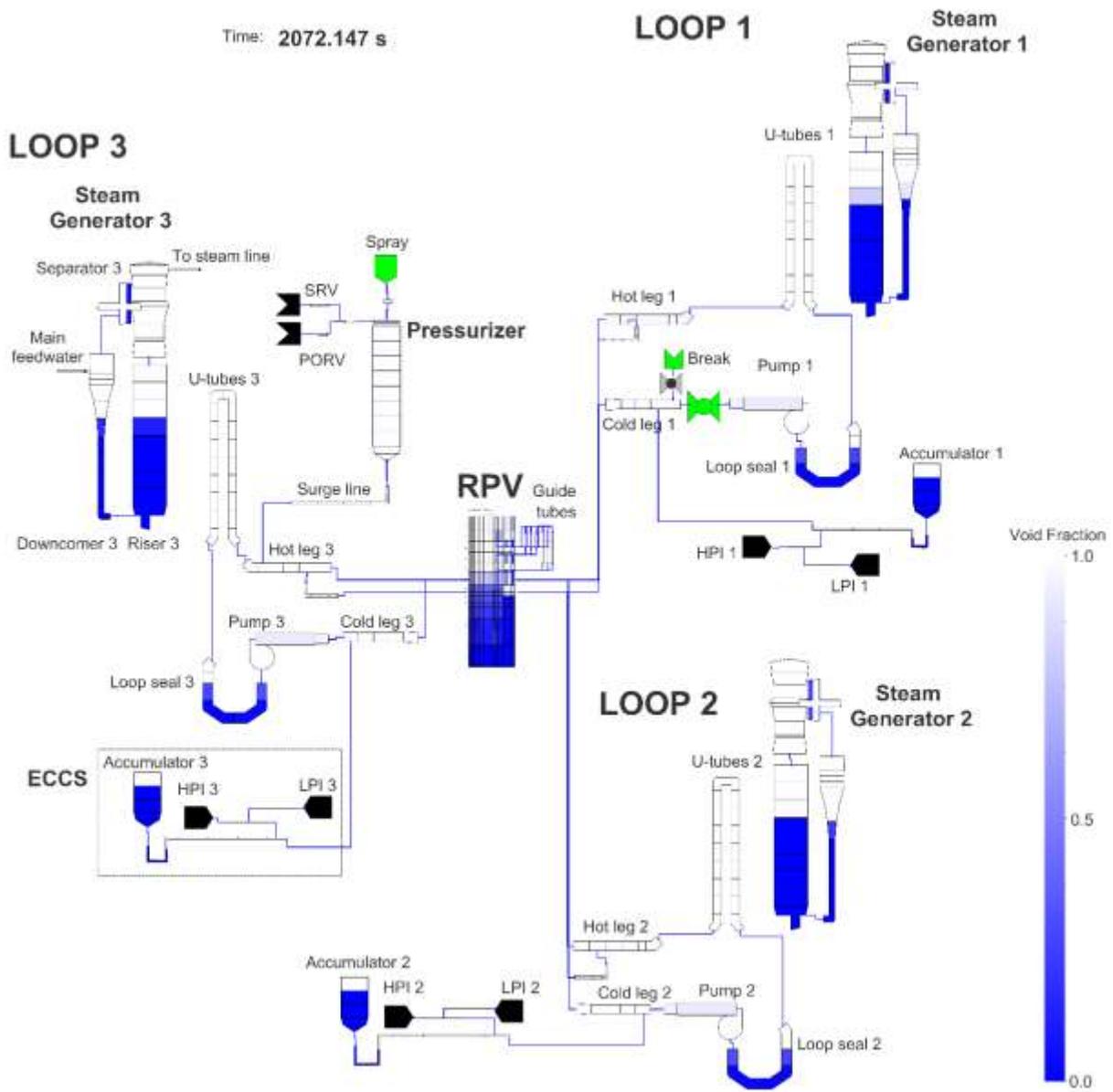


Figure 19 Void Fraction in 3-Loop PWR Plant when Primary Pressure = 5 MPa

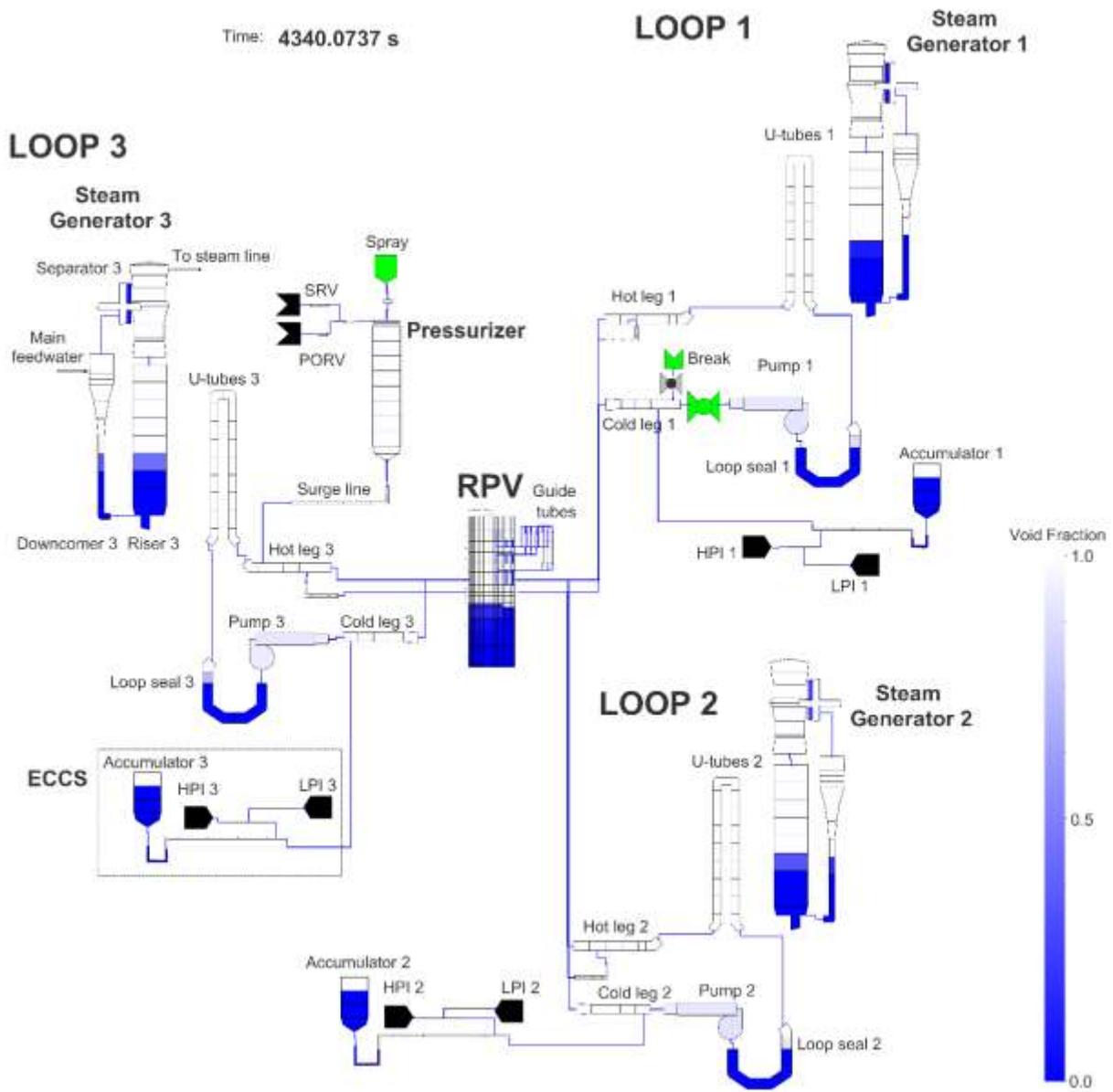


Figure 20 Void Fraction in 3-Loop PWR Plant when Second PCT Excursion is Produced

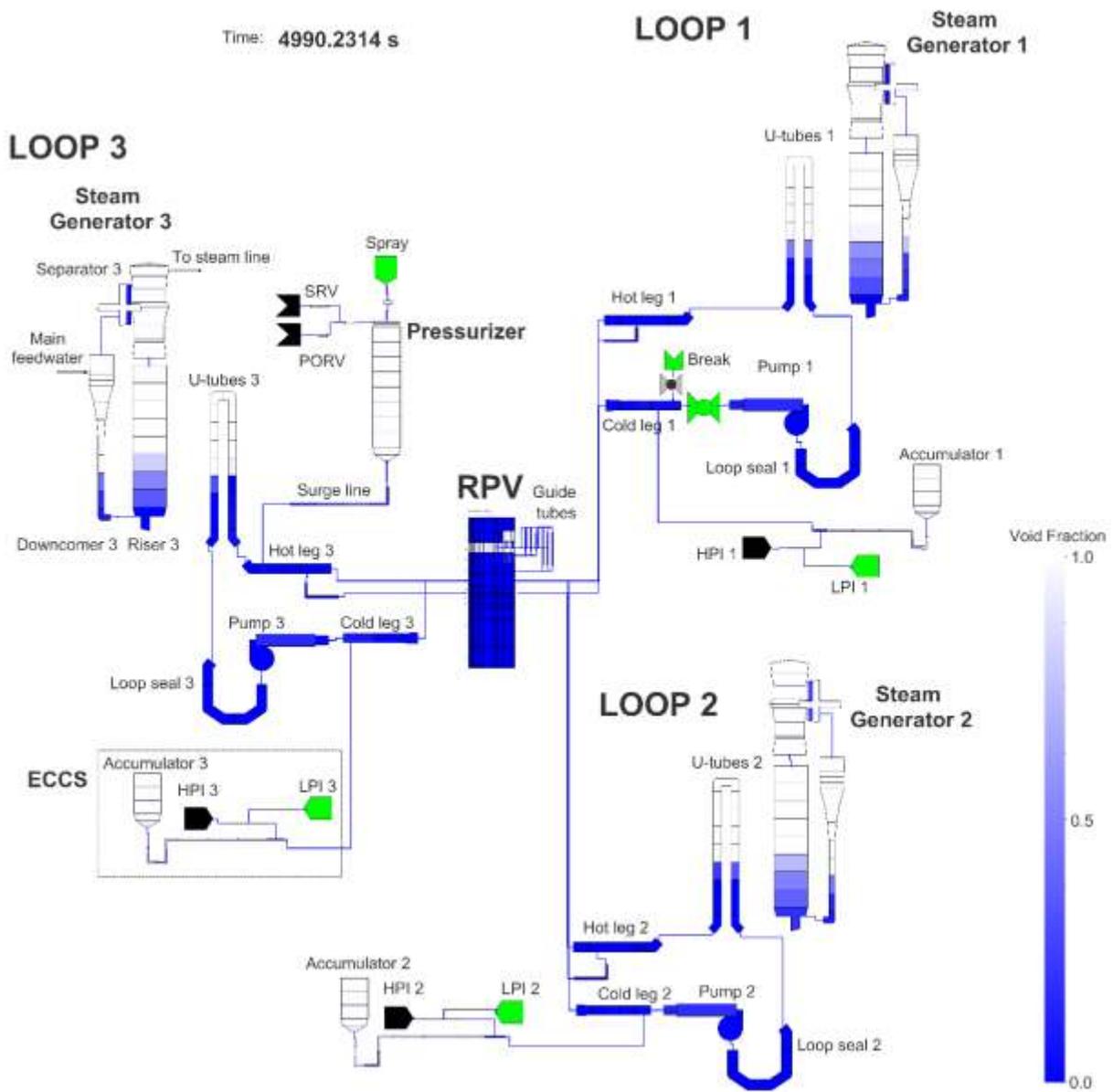


Figure 21 Void Fraction in 3-Loop PWR Plant at the End of the Transient

7 CONCLUSIONS

This work presents the simulation of a Small Break Loss-Of-Coolant Accident (SBLOCA) in the hot leg applied to a standard 3-loop PWR Nuclear Power Plant (NPP) using the thermal-hydraulic code TRACE5. This experiment, Test 3, was conducted in the Large Scale Test Facility (LSTF) in the frame of the OECD/NEA ROSA-2 Project.

A 3-loop PWR TRACE5 model has been adapted to the transient conditions applying the power-to-volume strategy. It can be done because the LSTF is a Full-Height Full-Pressure facility, which reproduces a 4-loop W-type PWR. It allows to preserve time, power and mass inventory as in the LSTF, because the fluid exhibits the same properties at full pressure.

The simulation results show that the important physical phenomena occurred in this transient during different phases (high-pressure, conditioning phase and low-pressure) are reproduced in the 3-loop PWR TRACE5 model.

8 REFERENCES

1. Thermohydraulic Safety Research Group, Nuclear Safety Research Center, Final Data Report of ROSA/LSTF Test 6-1 (1.9 % Pressure Vessel upper-head Small break LOCA Experiment SB-PV-09), Japan Atomic Energy Agency, JAEA (2010).
2. Thermohydraulic Safety Research Group, Nuclear Safety Research Center, Final Data Report of ROSA-2/LSTF Test 3 (Counterpart Test to PKL SB-HL-18), Japan Atomic Energy Agency, JAEA (2010).
3. H. Kremin, et al., "Description of the PKL III Test Facility". Framatom ANP GmbH, Private Communication, (2001).
4. The ROSA-V Group, ROSA-V Large Scale Test Facility (LSTF) System Description for the 3rd and 4th Simulated Fuel Assemblies, JAERI-Tech, Japan (2003).
5. Nuclear Regulatory Commission, Division of Risk Assessment and Special Projects. Office of Nuclear Regulatory Research. U. S Nuclear Regulatory Commission, TRACE V5.0. Theory Manual. Field Equations, Solution Methods and Physical Models, Nuclear Regulatory Commission, U.S. (2007).
6. Nuclear Regulatory Commission, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, TRACE V5.0. User's manual. Volume 1: Input Specification, Nuclear Regulatory Commission, U. S (2007).
7. Nuclear Regulatory Commission and Applied Programming Technology, Symbolic Nuclear Analysis Package (SNAP) (2007).
8. Kumamaru, H., Tasaka, K., 1990. Recalculation of Simulated Post-Scram Core Power Decay Curve for Use in ROSA-IV/LSTF Experiments on PWR Small-Break LOCAs and Transients. Report JAERI-M 90-142, Japan Atomic Energy Research Institute.

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

NUREG/IA-0512

2. TITLE AND SUBTITLE

Simulation of ROSA-2 Test 3 Counterpart with TRACE5. Application to Nuclear Power Plant

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	2019

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

S. Gallardo, A. Querol, M. Lorduy, G. Verdu

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.)

Universitat Politècnica de València
Instituto Universitario de Seguridad Industrial, Radiofísica y Medioambiental
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, D.C. 20555-0001

10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

This paper presents a first approach of a hot leg Small Break Loss-Of-Coolant Accident (SBLOCA) application to a Nuclear Power Plant (NPP). For this purpose, a standard 3-loop PWR NPP model by means of the thermal-hydraulic code TRACE5 patch 2 and Symbolic Nuclear Analysis Packages (SNAP) has been used. The SBLOCA transient corresponds to the Test 3 (SB-HL-18) conducted at the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA) in the frame of the OECD/NEA ROSA-2 Project. LSTF is a Full-Height Full-Pressure (FHFP) facility designed to simulate the Tsuruga unit II NPP (a 4-loop W-type PWR) with volume and power scaled by a factor of 1/48. Test 3 simulates a PWR 1.5% hot leg SBLOCA with an assumption of the total failure of High Pressure Injection (HPI) system under two different pressure conditions. Considering the assumptions of this test, some modifications have been performed in the 3-loop PWR TRACE5 model. The simulation results are provided throughout several graphs where the main system variables, such as primary and secondary pressures, collapsed liquid levels and temperatures are shown.

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

Accident Management (AM)
Committee on the Safety of Nuclear Installations (CSNI)
Consejo de Seguridad Nuclear (Nuclear Safety Council, Spain, CSN)
Large Scale Test Facility (LSTF)
High Pressure Charging Pump (PJ)
High Pressure Injection Pump (PL)
Intermediate Break Loss-Of-Coolant-Accident (IBLOCA)

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE



Federal Recycling Program



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS



NUREG/IA-0512

Simulation of ROSA-2 Test 3 Counterpart with TRACE5 - Application to Nuclear Power Plant

March 2019