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International Agreement Report

Simulation of ROSA-2 Test-2 Experiment: Application to Nuclear Power Plant

Prepared by:
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ABSTRACT

The purpose of this work is to overview the results provided by a first approach to the simulation of an Intermediate Break Loss-Of-Coolant Accident (IBLOCA) in a 3-loop PWR Nuclear Power Plant (NPP) using the thermal-hydraulic code TRACE5 patch 2 and the Symbolic Nuclear Analysis Packages software (SNAP) version 2.1.2.

The IBLOCA transient applied to the standard PWR TRACE5 model is the Test 2 (IB-CL-03) handled at the Large Scale Test Facility (LSTF) in the frame of the OECD/NEA ROSA-2 Project. Test 2 simulates a 17% cold leg IBLOCA under the assumption of the single-failure of High Pressure Injection and Low Pressure Injection systems and total failure of the Auxiliary Feedwater.

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

The simulation results are provided throughout several graphs, where the main system variables, such as pressures, pressure vessel liquid levels and temperatures are shown. These results represent a contribution to assess the predictability of computer codes such as TRACE5.

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/FAP[1] 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 international collaborations. In the TH framework, most of these actions are either covering not

[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

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

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EXECUTIVE SUMMARY

The purpose of this work is to test the capability of the thermal-hydraulic code TRACE5 patch 2 in a first approach to the simulation of a cold leg Intermediate Break LOCA (IBLOCA) in a standard PWR plant of three loops. For this goal, a TRACE5 model of a standard 3-loop PWR has been adapted to reproduce the Test 2 transient, in the frame of the OECD/NEA ROSA-2 Project.

Test 2 (IB-CL-03) was conducted in the Large Scale Test Facility (LSTF) of the Japan Atomic Energy Agency (JAEA). LSTF is a Full Height Full Pressure (FHFP) facility reproducing Tsuruga unit II Nuclear Power Plant (W-type 4-loop PWR). It allows to preserve time, power and mass inventory as in the LSTF, because the fluid exhibits the same properties at full pressure.

Test 2 simulates a 17% cold leg IBLOCA using an upward break nozzle mounted with the cold leg inner surface. Single-failure of both High Pressure Injection and Low Pressure Injection systems and total failure of the Auxiliary Feedwater are assumed.

Results of the simulation using TRACE5 are shown in several graphs, including primary and secondary pressures, discharged inventory, primary mass flow rates, and collapsed liquid levels (in the pressure vessel, hot legs, steam generators U-tubes, etc.).

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
PORV	Pilot Operated Relief Valve
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
SI	Safety Injection
SNAP	Symbolic Nuclear Analysis Package
SRV	Safety Relief Valve
ST	Storage Tank

1 INTRODUCTION

Since the accident at Three Mile Island Unit-2 (TMI-2) in 1979, many test facilities have produced experimental data on Small Break Loss-Of-Coolant Accidents (SBLOCA) for code assessment and development. However, the experimental data for Intermediate Break Loss-Of-Coolant Accidents (IBLOCA) is quite limited although the thermal-hydraulic responses could differ between SBLOCA and IBLOCA. Because full-scale testing is usually impossible to perform, there is a need to test scaled models of prototype systems. With this aim, the Japan Atomic Energy Agency (JAEA) in the frame of the OECD/NEA ROSA-2 Project prepared detailed thermal-hydraulic data.

The purpose of this work is to test the capability of the thermal-hydraulic code TRACE5 patch 2 [1, 2] in the simulation of a cold leg IBLOCA in a standard 3-loop PWR plant (representative of Spanish NPPs). For this goal, a generic 3-loop PWR TRACE5 model has been adapted to reproduce the Test 2, conducted in the Large Scale Test Facility (LSTF) [4] in the frame of the OECD/NEA ROSA-2 Project.

Test 2 (IB-CL-03) [3] simulates a PWR 17% cold leg IBLOCA under the assumption of a single–failure of the High Pressure Injection (HPI) and the Low Pressure Injection (LPI) systems and the total failure of the Auxiliary Feedwater (AFW) system.

LSTF is a Full Height Full Pressure (FHFP) facility designed to simulate Tsuruga unit II NPP, which is 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 loops in LSTF. The initial 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.

A generic 3-loop PWR TRACE5 model has been adapted applying the power-to-volume scaling strategy to simulate the transient conducted in the LSTF. The simulation results are shown in different graphs, including primary and secondary pressures, discharged inventory, primary mass flow rates, liquid levels (in the pressure vessel, hot leg, steam generators U-tubes, etc.) and temperatures.

2 LSTF DESCRIPTION

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

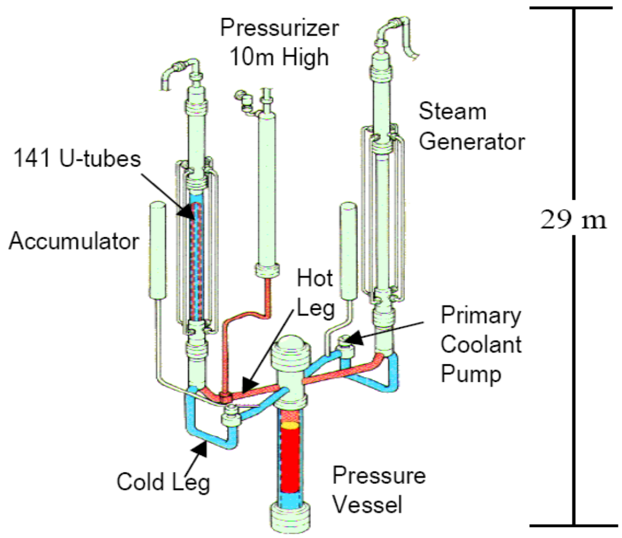

Figure 1 Schematic View of the LSTF Facility

The main features of LSTF are the following:

- Elevations: preserved.
- Volumes: scaled by 1/48 to the reference PWR.
- Flow area: scaled by 1/48 in the pressure vessel and by 1/24 in the steam generators. The flow area in the 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 the reference PWR.
- Fuel assembly: The total number of fuel rods is 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 loops include a primary coolant pump (PC) and 141 U-tubes for each steam generator. The secondary-coolant system includes a jet condenser (JC), a feed water pump (PF), the auxiliary feedwater pumps (PA) and the 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 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 includes 8 spray nozzles (of 3.4 mm inner-diameter) in 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 2 conducted in the LSTF simulates a 17% cold leg IBLOCA under the assumption of single-failure of the ECCS (HPI, AIS and LPI systems) and the total failure of the AFW. The complete control logic of the transient is listed in Table 1. The system pressures and the PCT evolution are shown in Figure 2. The break unit is connected to the cold leg and the orifice flow area corresponds to 17% of the volumetrically-scaled cross-sectional area of the reference PWR cold leg. The transient starts at time 0 with the break valve opening in the cold leg (in the loop without pressurizer) and increasing the rotational speed of the reactor coolant pumps. Few seconds later, the scram signal is generated. This signal produces the initiation of the core power decay curve, the initiation of the primary coolant pumps coastdown, the turbine trip, the closure of MSIV and the termination of the MFW.

When the primary pressure is lower than 12.27 MPa, the Safety Injection (SI) signal is generated. The HPI system is activated few seconds after the SI signal generation, only in the loop with pressurizer. The accumulators actuate when the primary pressure is lower than 4.5 MPa. The core power is automatically decreased by the LSTF Core Protection System when the maximum fuel rod surface temperature reaches 958 K. When the primary pressure is lower than 1.24 MPa, the LPI system actuates in the loop with pressurizer. Test 2 finishes with the break valve closure when primary and secondary pressures are stabilized.

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

Break.	Time zero
Generation of scram signal.	Primary pressure = 12.97 MPa.
Initiation of core power decay curve simulation.	Generation of scram signal.
Initiation of Primary Coolant Pump coastdown.	Generation of scram signal.
Turbine trip (closure of steam generators Main Steam Stop Valves).	Generation of scram signal.
Closure of Steam Generator Main Steam Isolation Valves (MSIVs).	Generation of scram signal.
Termination of Steam Generator Main Feedwater.	Generation of scram signal.
Generation of Safety Injection (SI) signal	Primary pressure = 12.27 MPa.
Initiation of High Pressure Injection (HPI) in loop w/pressurizer.	12 seconds after SI signal.
Initiation of the Accumulator Injection in loop w/pressurizer.	Primary pressure = 4.51 MPa.
Initiation of Low Pressure Injection (LPI) in loop w/pressurizer.	Primary pressure = 1.24 MPa.

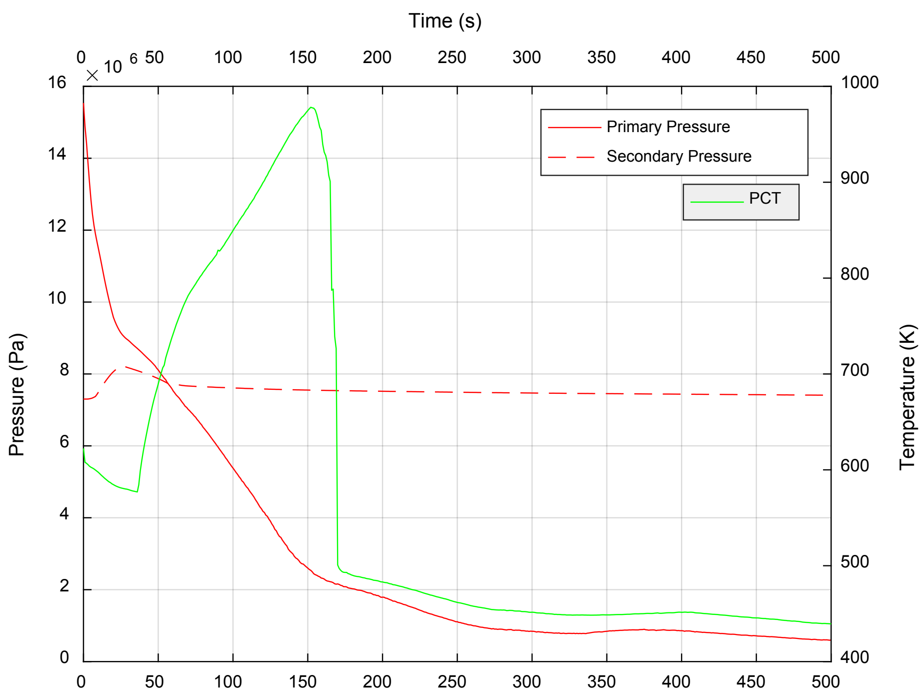

Figure 2 LSTF Experimental Data: Primary and Secondary Pressures and PCT

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. The levels 15 and 16 model the upper head. The control rod guide tubes (CRGTs) are modeled using 6 PIPEs: 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. In this case, the power has been characterized by means of a point kinetics model with constant initial reactivity and table-input reactivity as a function of time after a trip.

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 connection with the surge line of the pressurizer. A PIPE component is used to simulate the pressurizer. The pressurizer Relief Valves (RVs) and the Pilot Operated Relief Valve (PORV) are also considered.

The cold legs have been simulated with 3 PIPEs 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 connection with 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).

The HPI and LPI systems have been simulated with FILL components. The AIS is modeled using PIPE components 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 PIPE components, while a SEPD component has been used to model the steam separator. The relief and isolation valves have been modeled using VALVE components. The Relief Valve is linked to a BREAK component, while the Main Steam Isolation Valve is connected to a PIPE component, which simulates the steam line header.

The MFW and AFW are simulated using FILL components connected to a PIPE component. This PIPE is linked to the top of the downcomer. Figure 3 shows the nodalization of the standard 3-loop PWR TRACE5 model using the Symbolic Nuclear Analysis Package (SNAP) [5] software version 2.1.2.

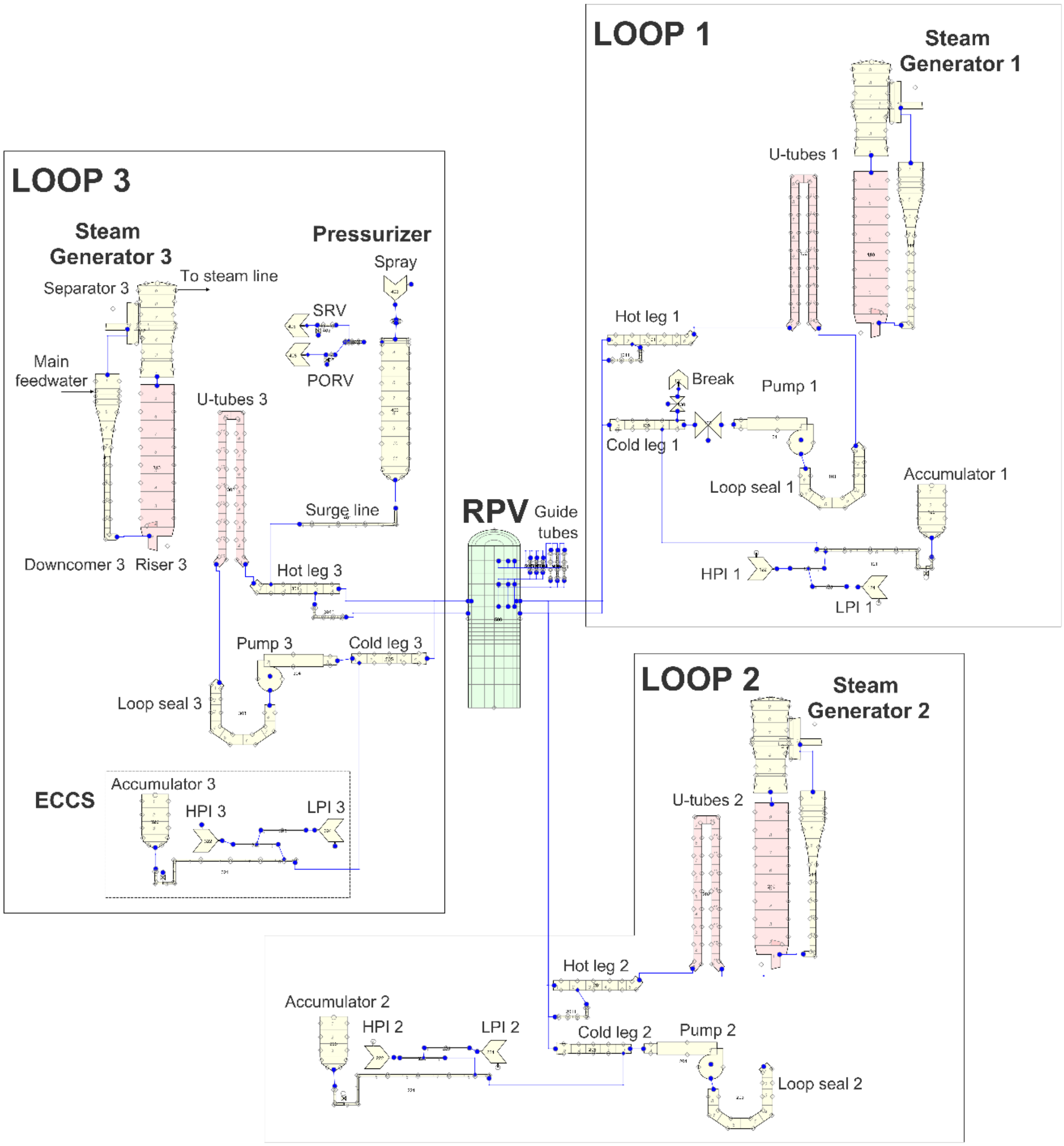

Figure 3 Standard 3-Loop PWR Model Nodalization

5 SCALING CONSIDERATIONS

In LOCA 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 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 IBLOCA in the cold 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:

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

Momentum equation:

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

Energy equation:

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

State equation:

$$\rho = \rho(h, p) \tag{4}$$

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):

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

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

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

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

$$\frac{\partial u_i{}^*}{\partial t^*} + u_j{}^* \frac{\partial u_i{}^*}{\partial x_j{}^*} = \frac{g d_0}{u_0^2} F_i{}^* - \frac{\Delta p_0}{\rho_0 u_0^2} \frac{1}{\rho^*} \frac{\partial p^*}{\partial x_i{}^*} - \frac{1}{\rho^*} \frac{\partial (\rho^* \overline{u_i'^* u_j'^*})}{\partial x_j{}^*} \tag{6}$$

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

$$\rho^* = \rho(h^*, p^*) \tag{8}$$

These equations contain other dimensionless parameters such as $\frac{u_0^2}{g d_0}, \frac{\Delta p_0}{\rho_0 u_0^2}$ 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.

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

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

$$\frac{\emptyset_{scaled-up}}{\emptyset_{LSTF}} = \frac{Q_{scaled-up}}{Q_{LSTF}} = \frac{V_{scaled-up}}{V_{LSTF}} = \frac{A_{scaled-up}}{A_{LSTF}} = K_v \tag{10}$$

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):

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

$$D = d \cdot K_v{}^{2/5} \tag{12}$$

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):

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

$$L = l \cdot K_v{}^{1/5} \tag{14}$$

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

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.

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 pressure of accumulators is the same in all cases. The break size considered in Test 2 has been scaled applying the power-to-volume criterion between LSTF, Tsuruga II and the 3-loop PWR NPP.

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

Parameter	LSTF	Reference 4-loop NPP	3-loop NPP	$K_{v\ (3\text{-loop/LSTF})}$
Primary pressure (MPa).	15.5	15.5	15.5	1
Core power (MW).	10	3423	2785	39.0 (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	
Break area: 17% cold leg flow area reference NPP (m^2).	$1.36\ 10^{-3}$	0.065	0.048	35.2
Accumulator pressure (MPa).	4.6	4.6	4.6	1

6 RESULTS AND DISCUSSION

6.1 Transient Conditions

The following assumptions are imposed to reproduce Test 2 in the NPP model:

1. Break size is 17% of the cold leg flow area.
2. An upward long break nozzle is located on top of the cold leg in the broken loop without pressurizer (loop 1).
3. Loss of off-site power concurrent with the scram.
4. HPI, AI and LPI systems are activated in intact loops.
5. Non-condensable gas inflow from accumulator tank may take place.
6. Total failure of AFW.

6.2 Steady-State

Table 3 shows the steady-state conditions achieved using TRACE5.

Table 3 Steady-State Conditions in the TRACE5 Model

Item	3-loop NPP TRACE5 model
Core Power (MW_t).	2785
Hot leg Fluid Temperature (K).	600
Cold leg Fluid Temperature (K).	569
Mass Flow Rate (kg/s).	4416
Pressurizer Pressure (MPa).	15.5
Pressurizer Liquid Level (m).	3.65
Accumulator System Pressure (MPa).	4.6
SG Secondary-side Pressure (MPa).	6.8
SG Secondary-side Liquid Level (m).	5.7
Steam Flow Rate (kg/s).	505.6
Main Feedwater Flow Rate (kg/s).	505.6
Main Feedwater Temperature (K).	490
Auxiliary Feedwater Temperature (K).	300

6.3 Transient

Table 4 lists the chronological sequence of events during the transient performed using the PWR TRACE5 model. In addition, in this section several graphs are shown with the main system variables obtained with the standard PWR TRACE5 model.

Table 4 Chronological Sequence of Events

Event	3-loop TRACE5 model Time (s)
Break valve open.	0
Scram signal.	5
Turbine trip and closure of steam generator MSIVs.	5
Initiation of a decrease in liquid level in steam generator U-tube.	5
Initiation of coastdown of primary coolant pumps.	5
Termination of steam generator Main Feedwater.	5
Open of steam generator Relief Valves.	5
Initiation of core power decay.	5
Initiation of HPI system in intact loops.	25
Primary pressure became lower than steam generator secondary pressure.	About 40
Initiation of AIS in intact loops.	About 100
Maximum fuel rod surface temperature.	130
End of the test.	300

6.4 System Pressures

In Figure 4, the simulated primary and secondary pressures are shown. As it can be seen, the primary pressure starts to decrease at time zero due to the loss of coolant through the break. When the primary pressure decreases below 15.92 MPa, the scram signal is produced. The MSIVs are closed, producing the increasing of the secondary pressure. Due to this increase, the steam generator RVs are opened to maintain the pressure almost constant. When the secondary pressure is lower than 6.8 MPa, the RVs are closed, and they are maintained closed till the end of the transient.

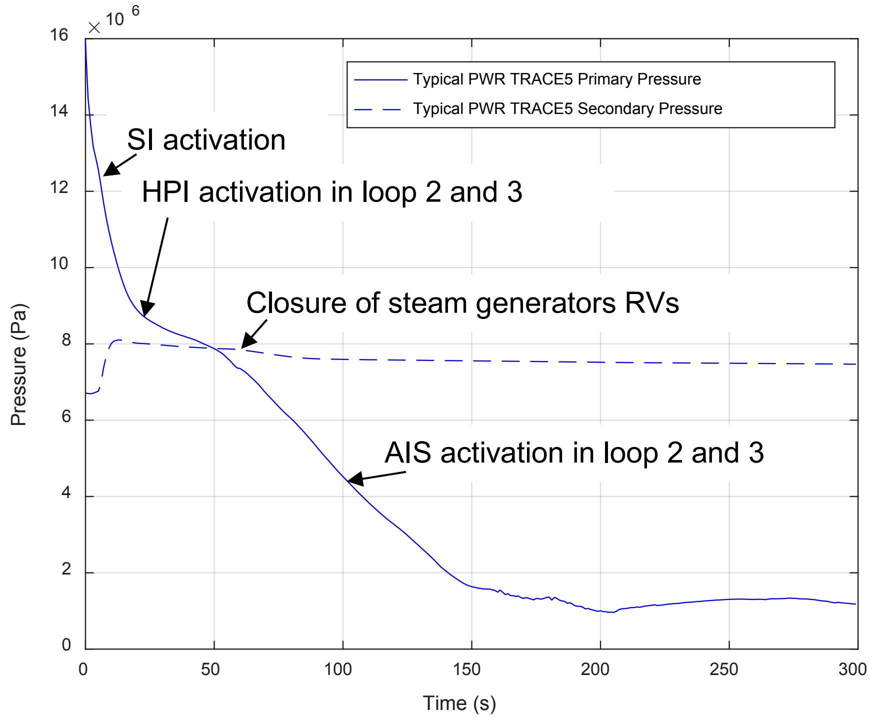

Figure 4 Primary and Secondary Pressures

The primary pressure soon becomes lower than the steam generator secondary pressure (at about 50 s). From this moment on, the steam generators no longer serve as a heat sink. The Accumulator Injection system is initiated at about 100 s, when the primary pressure is lower than 4.6 MPa. At this moment, the depressurization becomes effective due to steam condensation caused by the coolant injection in the cold legs. When the primary pressure is about 1 MPa, the Low Pressure Injection system is activated.

6.5 Break Mass Flow Rate

Figure 5 shows the mass flow rate through the break. As it can be seen, the change from liquid single-phase to two-phase flow is produced in a very short time after the break. The two-phase fluid regime is maintained until the hot legs are emptied. At this moment, the fluid regime changes to one-phase vapor. The change from two-phase to one phase vapor is produced at around 60 s. In order to adjust the break mass flow rate with TRACE5, the discharge coefficients for single-phase liquid and two-phase [2, 3] have been fixed to 1.0 (default value). These discharge coefficients and the application of the power-to-volume strategy to the break size provide similar results than in the experiment (Figure 25).

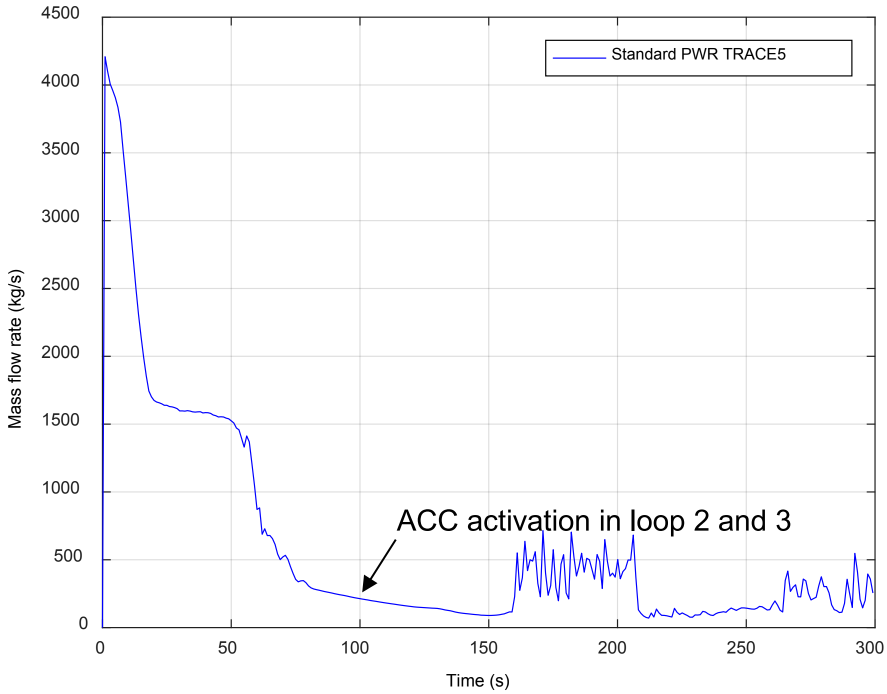

Figure 5 Break Mass Flow Rate

6.6 <u>Primary Loop Mass Flow Rates</u>

Figure 6 shows the primary mass flow rates obtained with the model. As it can be seen, the mass flow rates start to decrease simultaneously with the pumps coastdown. After the pumps coastdown only natural circulation occurs. There are no significant differences among loops, even considering the loop with the break.

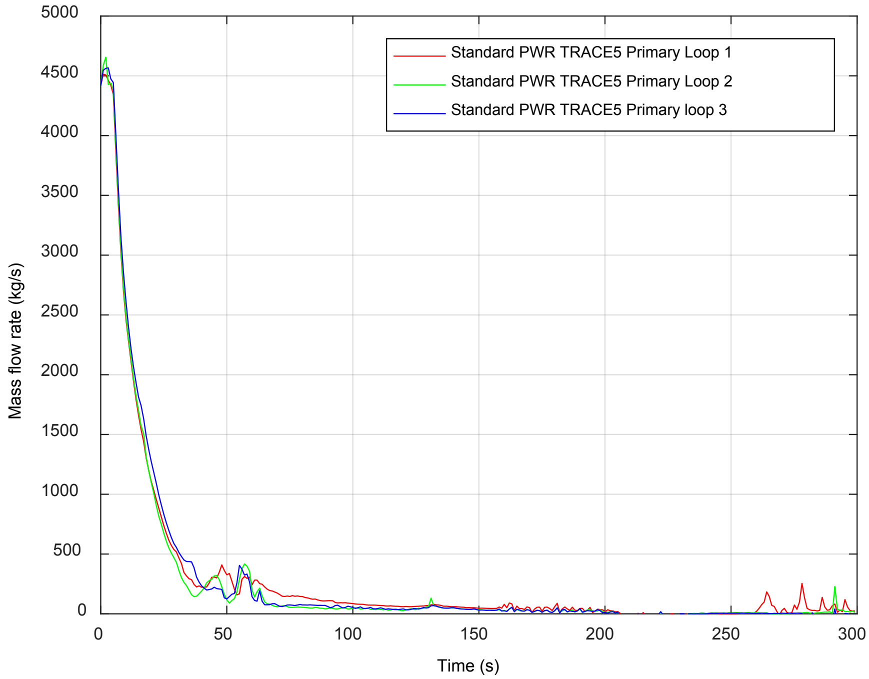

Figure 6 Primary Loop Mass Flow Rates

6.7 Pressure Vessel Collapsed Liquid Levels

The simulated pressure vessel collapsed liquid levels are shown in Figure 7. As it can be seen, the liquid level in the core sharply decreases after the break due to flashing of fluid because of a fast-primary depressurization, while in the downcomer, the minimum liquid level is achieved at 120 s. From this moment on, the refill of the active core is produced. The entrance of cold water injected in two of the cold legs produces the vapor condensation in the active core and its collapsed liquid level rapidly increases.

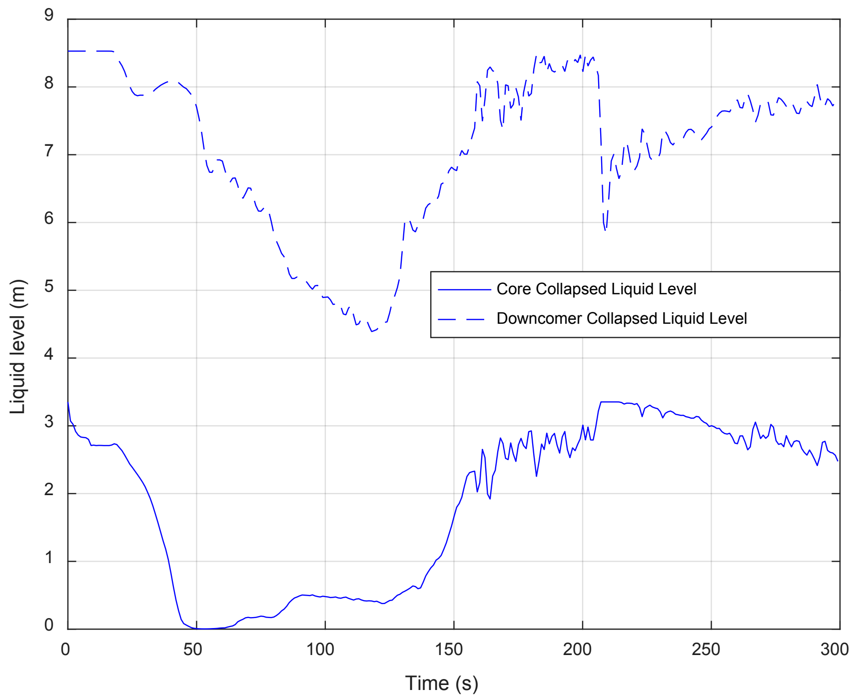

Figure 7 Core and Downcomer Collapsed Liquid Levels

6.8 Temperatures

Figure 8 shows the Maximum Peak Cladding (PCT) and the Core Exit Temperature (CET). As it can be seen, both temperatures start to increase when the core clearance takes place, at 40 s approximately. The maximum value of the PCT is reached at 130 s and the value is about 940 K. The CET excursion shows a similar behavior than the PCT, reaching its maximum value at a similar time.

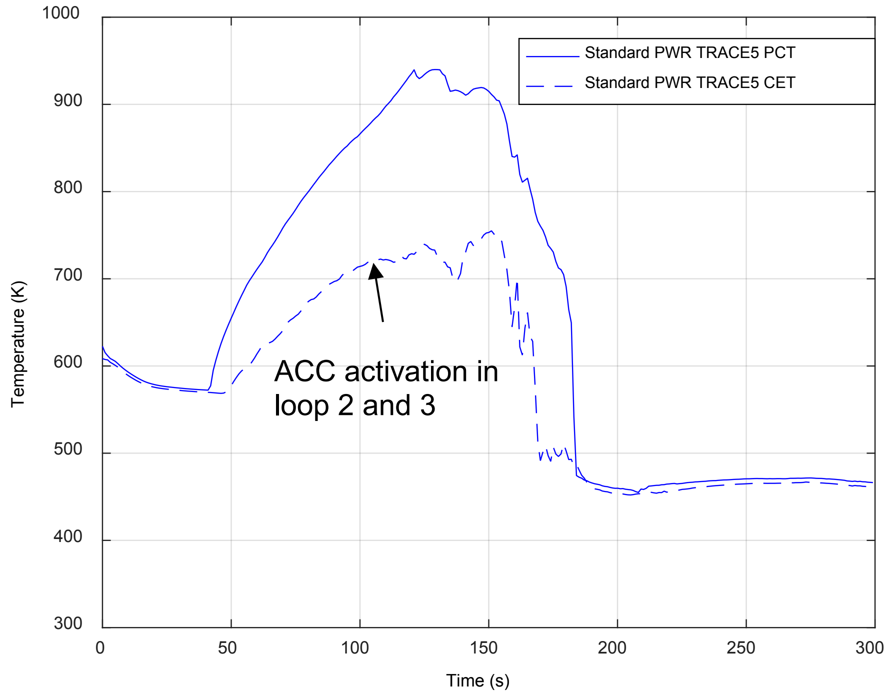

Figure 8 Maximum Fuel Rod Surface and Core Exit Temperatures

Figure 9 shows the cladding temperature at eight axial positions of a given fuel rod. In this case, a HTSTR component belonging to the second ring has been chosen due to the radial power distribution is maximum in it. The axial positions correspond to the eight axial levels in which is divided the active core. Furthermore, fuel rod temperatures for different HTSTRs (two of each ring) measured in the fifth axial level of the pressure vessel are shown in Figure 10. As it can be seen, the maximum fuel rod temperatures correspond to the HTSTR 51 and 83, which are in the second ring.

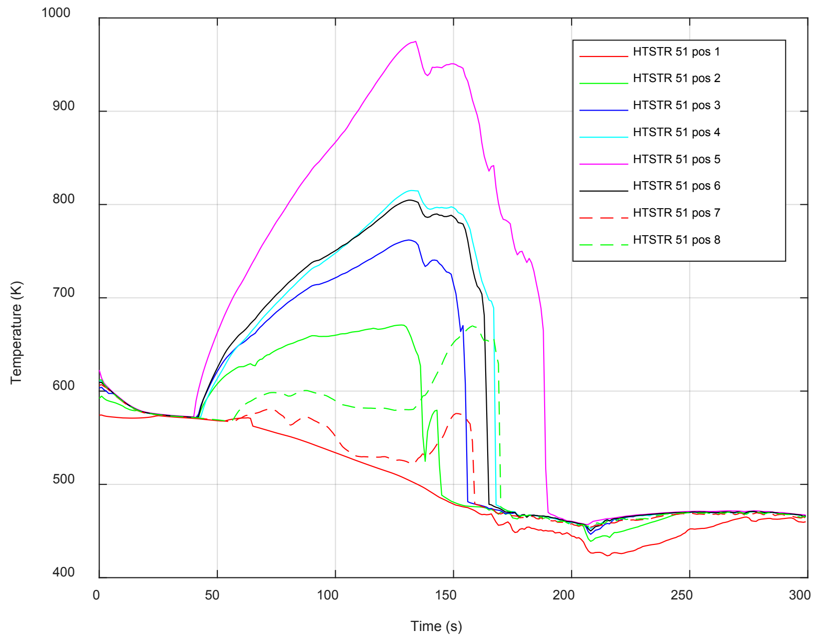

Figure 9 Fuel Rod Surface Temperature at Different Axial Positions

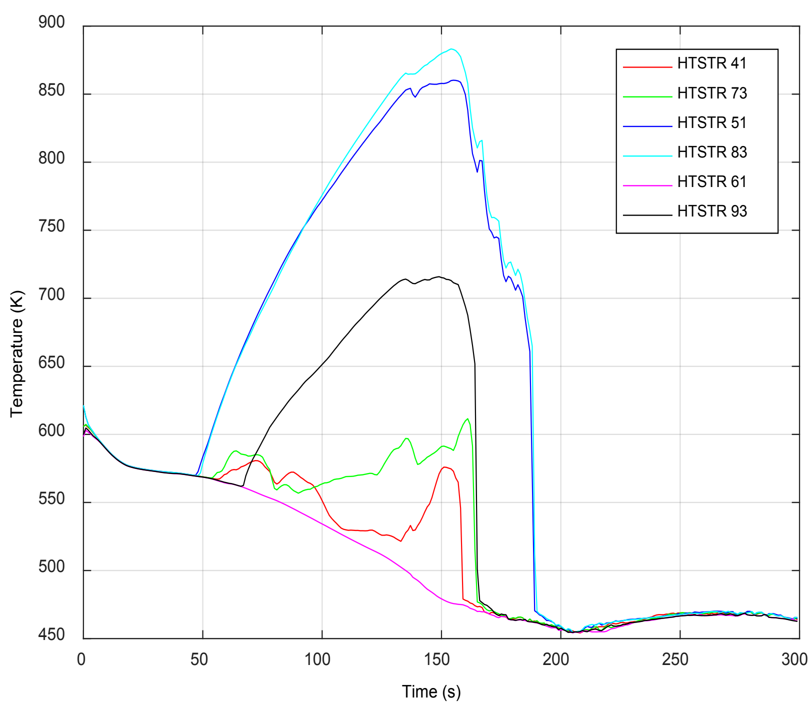

Figure 10 Fuel Rod Surface Temperature for Different HTSTR

6.9 Hot and Cold Legs Liquid Levels

Figure 11 and 12 show the collapsed liquid levels in the hot and cold legs, respectively. As it can be seen, liquid levels rapidly decrease due to the flashing produced by the fast primary depressurization just after the break. Regarding the cold leg liquid levels, it is observed that the three cold legs are emptied at around 70 s and at 120 s. When the activation of the AIS is produced, the water injected refills the three cold legs.

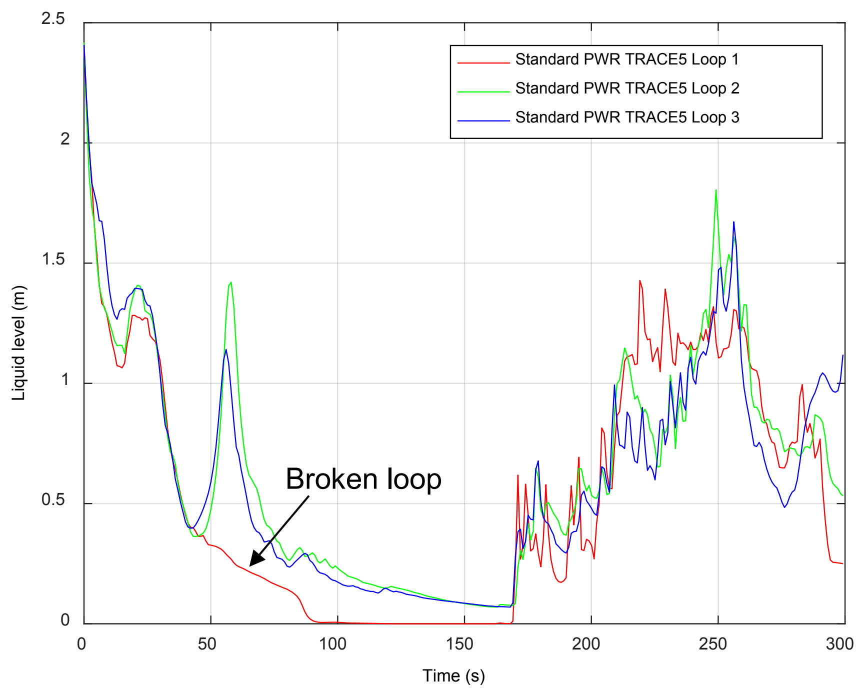

Figure 11 Hot Legs Collapsed Liquid Level

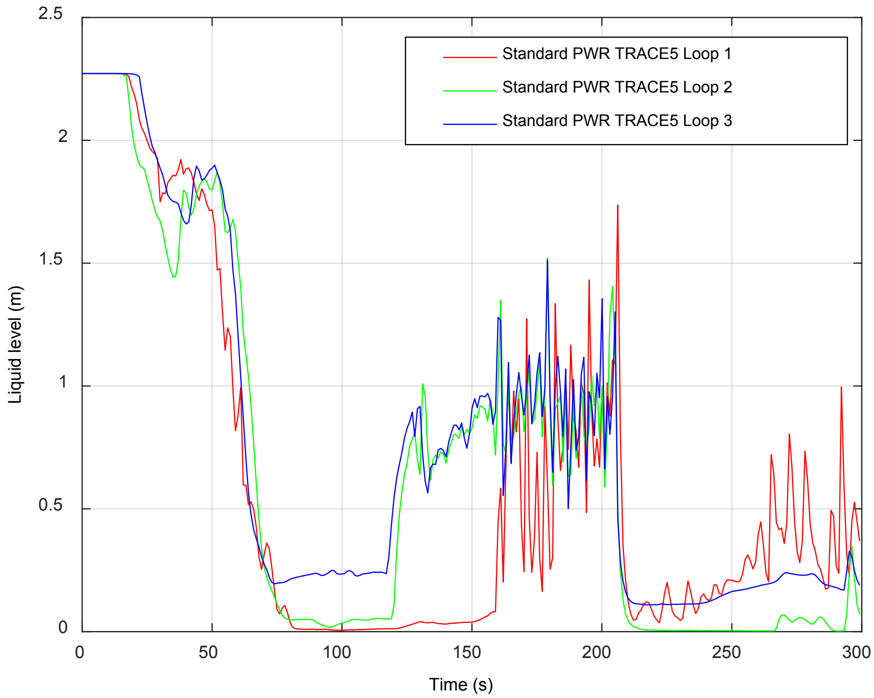

Figure 12 Cold Legs Collapsed Liquid Level

6.10 <u>U-Tubes Collapsed Liquid Level</u>

The collapsed liquid levels of the U-tubes are shown in Figure 13. As it can be seen, the U-tubes are emptied very early in the transient. At 70 s approximately, the U-tubes of loop 1 are completely empty, avoiding the heat transfer between the primary and secondary sides. However, in loops 2 and 3 the U-tubes emptying is delayed until 150 s, approximately.

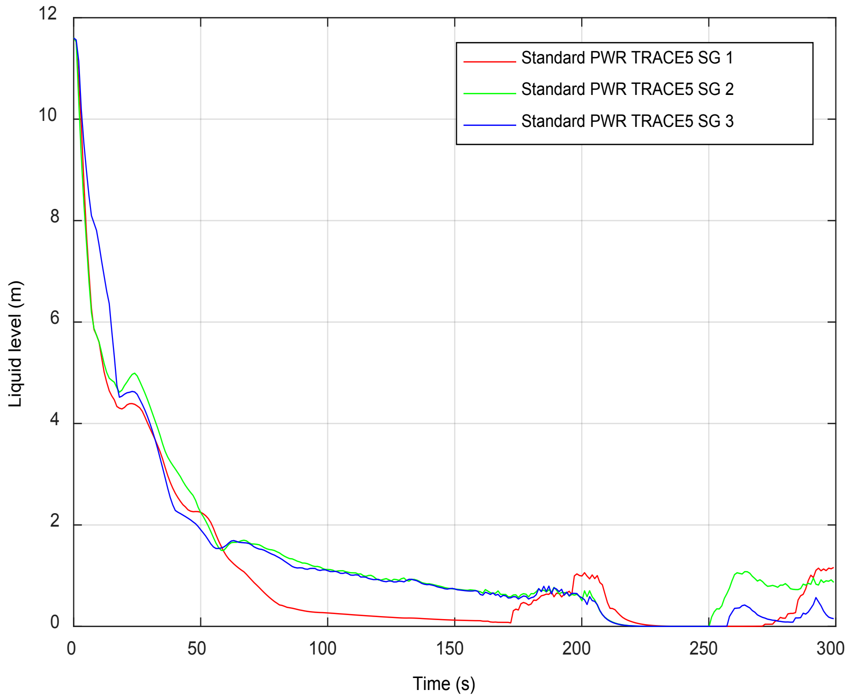

Figure 13 Steam Generator U-Tube Up-Flow Side Collapsed Liquid Levels

6.11 Study of the ECCS Actuation

In the LSTF experiment, the ECCS (HPI, AIS and LPI) are actuated only in the intact loop. However, in the 3-loop PWR TRACE5 model, it is modeled actuating in both intact loops and results obtained are similar to the experiment. In this section, the effect of the ECCS actuation in two or one intact loops is studied.

Figure 14 shows the pressures obtained in both cases (ECCS actuation in one or two intact loops). As it can be seen, there are no important differences between them.

Core and downcomer collapsed liquid levels are shown in Figures 15 and 16, respectively. As it can be seen, when the ECCS is actuated in two loops the core and the downcomer are refilled before.

Figure 17 shows the maximum PCT and the CET. As it is expected, the maximum values of both temperatures are reached when the ECCS is actuated only in one loop. Consequently, the drop of these temperatures is delayed in comparison to the case with the ECCS activated in two loops. Figures 18 and 19 show the accumulator injection flow rate and the High Pressure Injection flow rate, respectively in the PWR model.

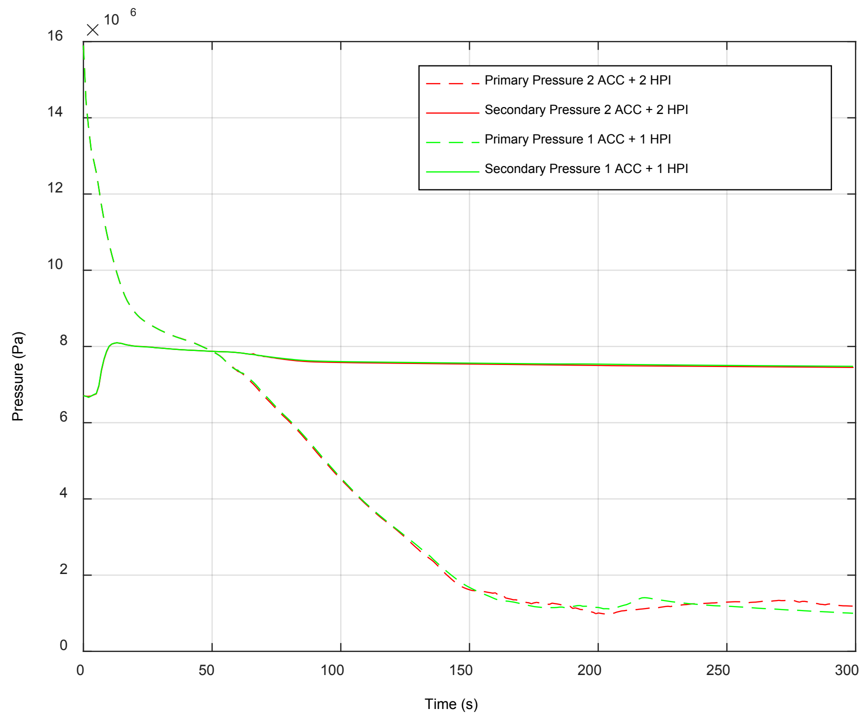

Figure 14 Primary and Secondary Pressures

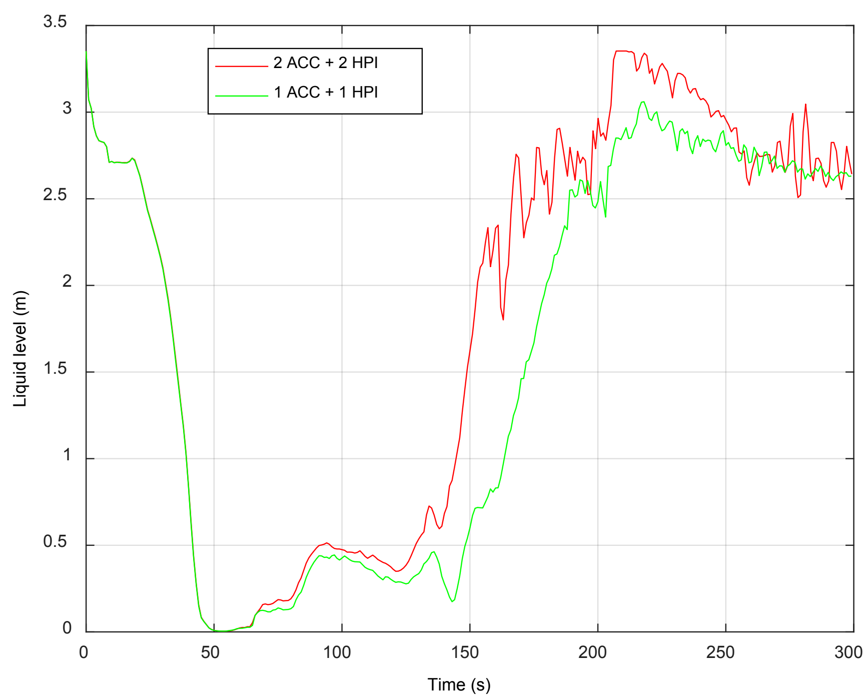

Figure 15 Core Collapsed Liquid Levels

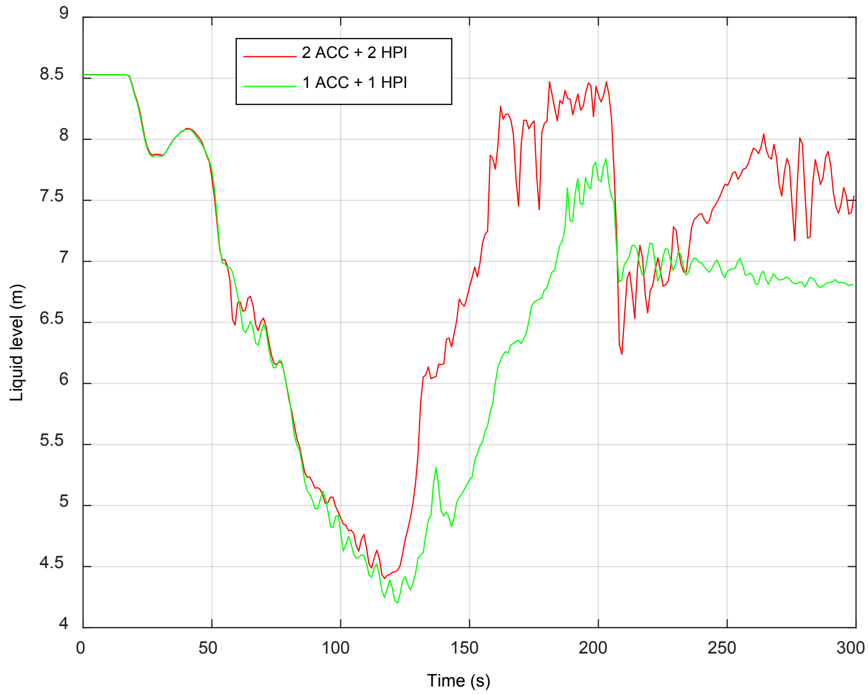

Figure 16 Downcomer Collapsed Liquid Levels

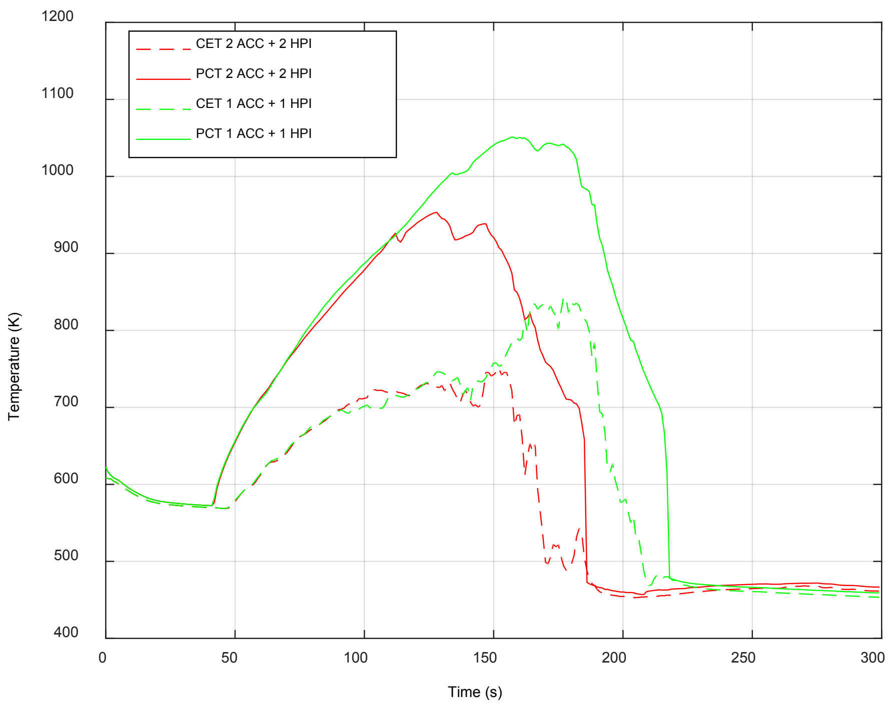

Figure 17 Maximum Fuel Rod Surface and Core Exit Temperatures

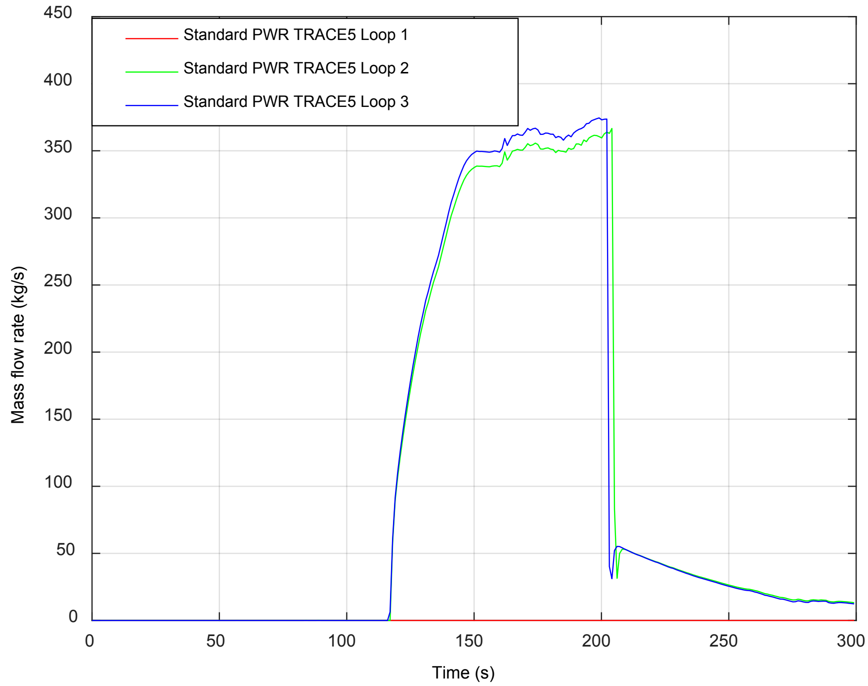

Figure 18 Accumulator Injection System Mass Flow Rate

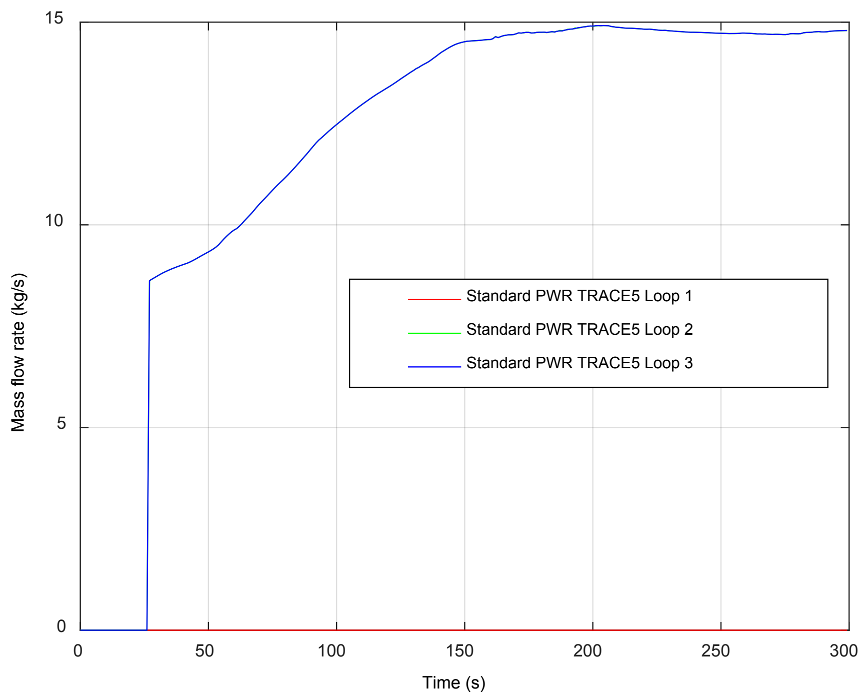

Figure 19 High Pressure Injection System Mass Flow Rate

6.12 Void Fraction

Figures 20, 21, 22 and 23 show the void fraction obtained at different moments during this transient. Firstly, Figure 22 shows the void fraction at the initiation of the test. As it can be seen, the primary system is full of liquid at this time. Then, the situation at 25 s is shown in Figure 21, when the pressurizer is empty. At this time, the liquid is located in the loop seals and in the bottom part of the pressure vessel. Figure 22 shows the situation when the accumulators of loops 2 and 3 are empty. As it can be seen, the loop seals are emptied, while the liquid is located in the core and in the pressure vessel bottom. Figure 23 shows the void fraction at the end of the transient.

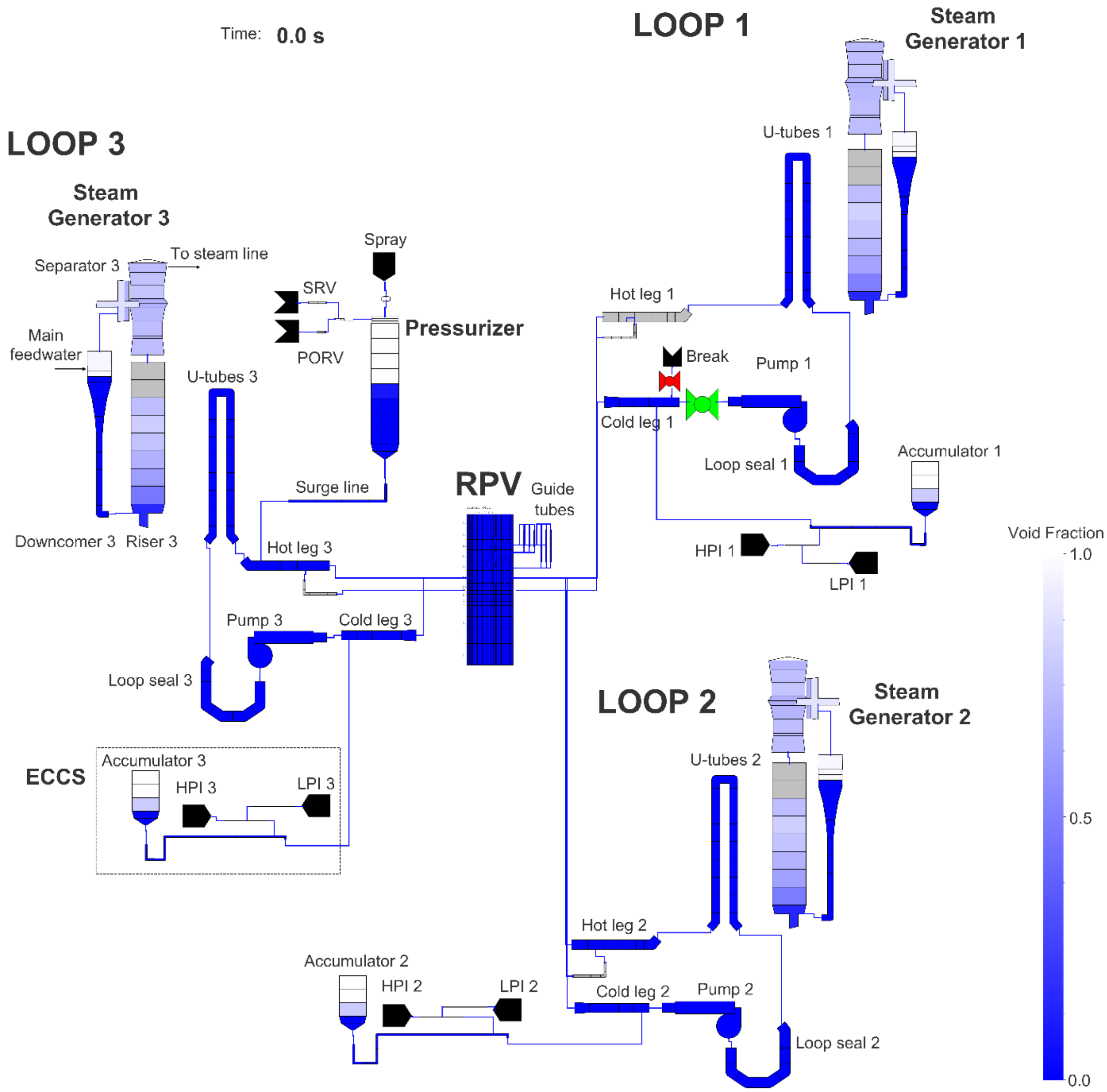

Figure 20 Void Fraction in 3-Loop PWR Plant at 0 s

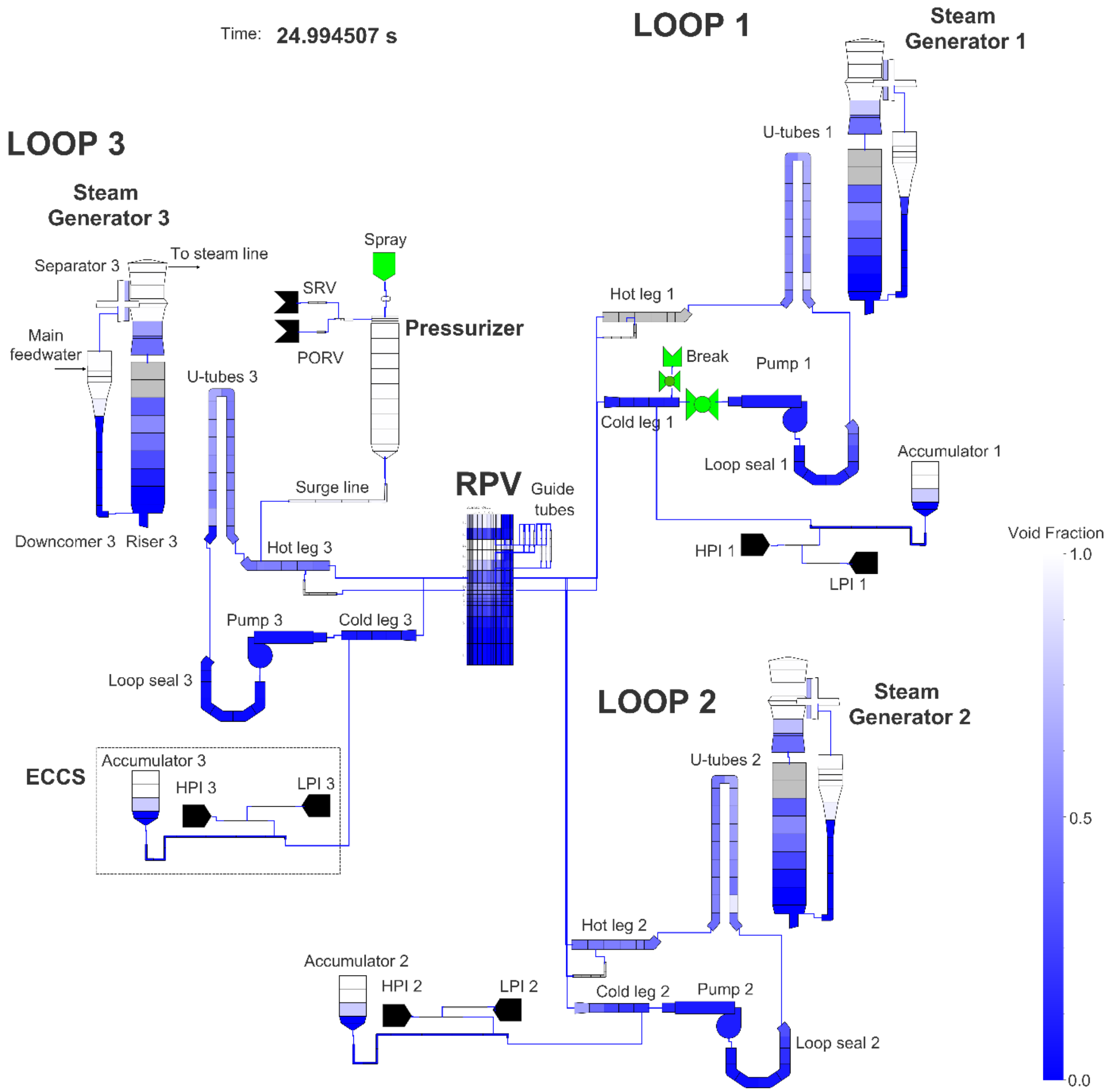

Figure 21 Void Fraction in 3-Loop PWR Plant at 25 s

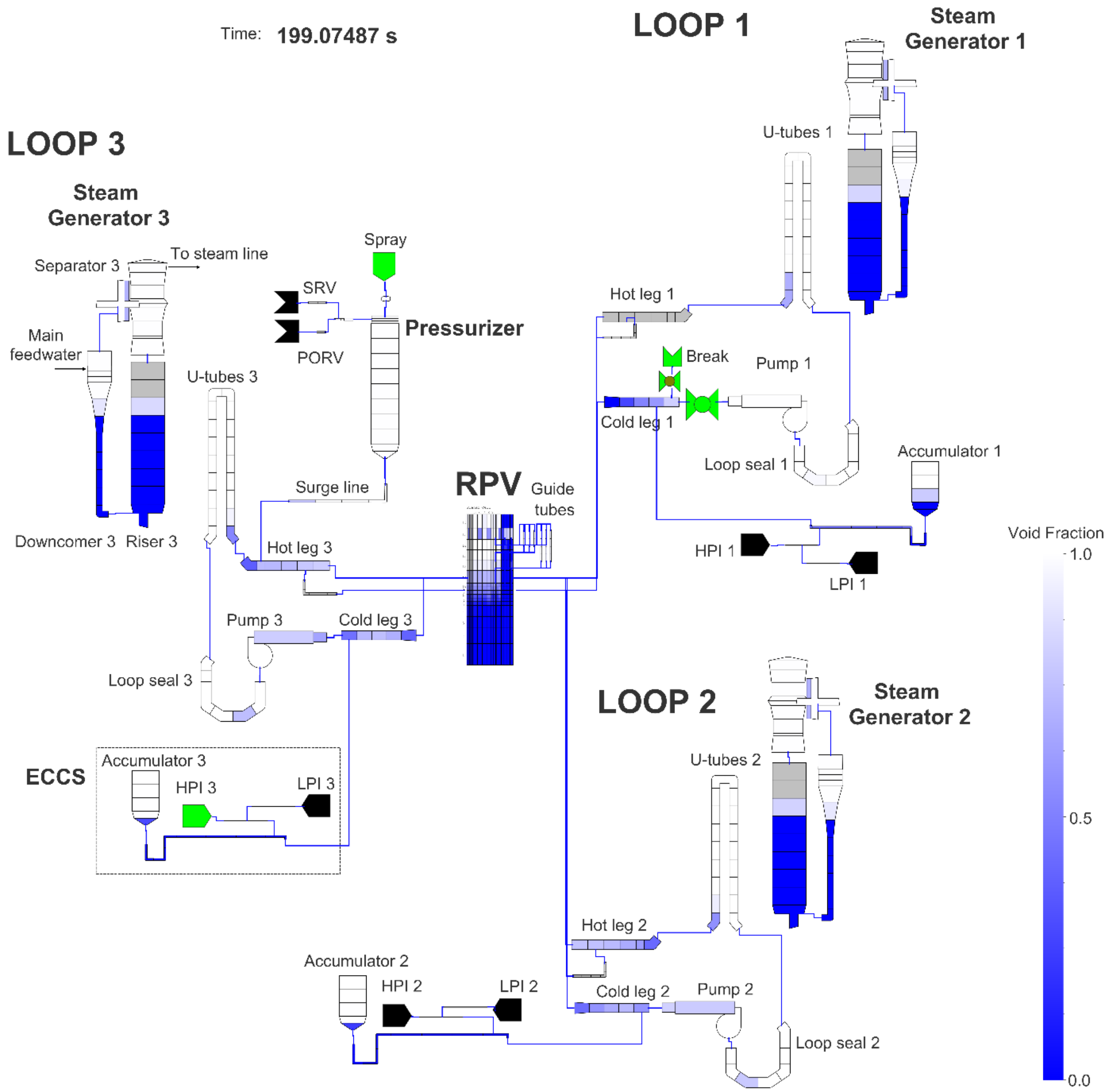

Figure 22 Void Fraction in 3-Loop PWR Plant at 200 s

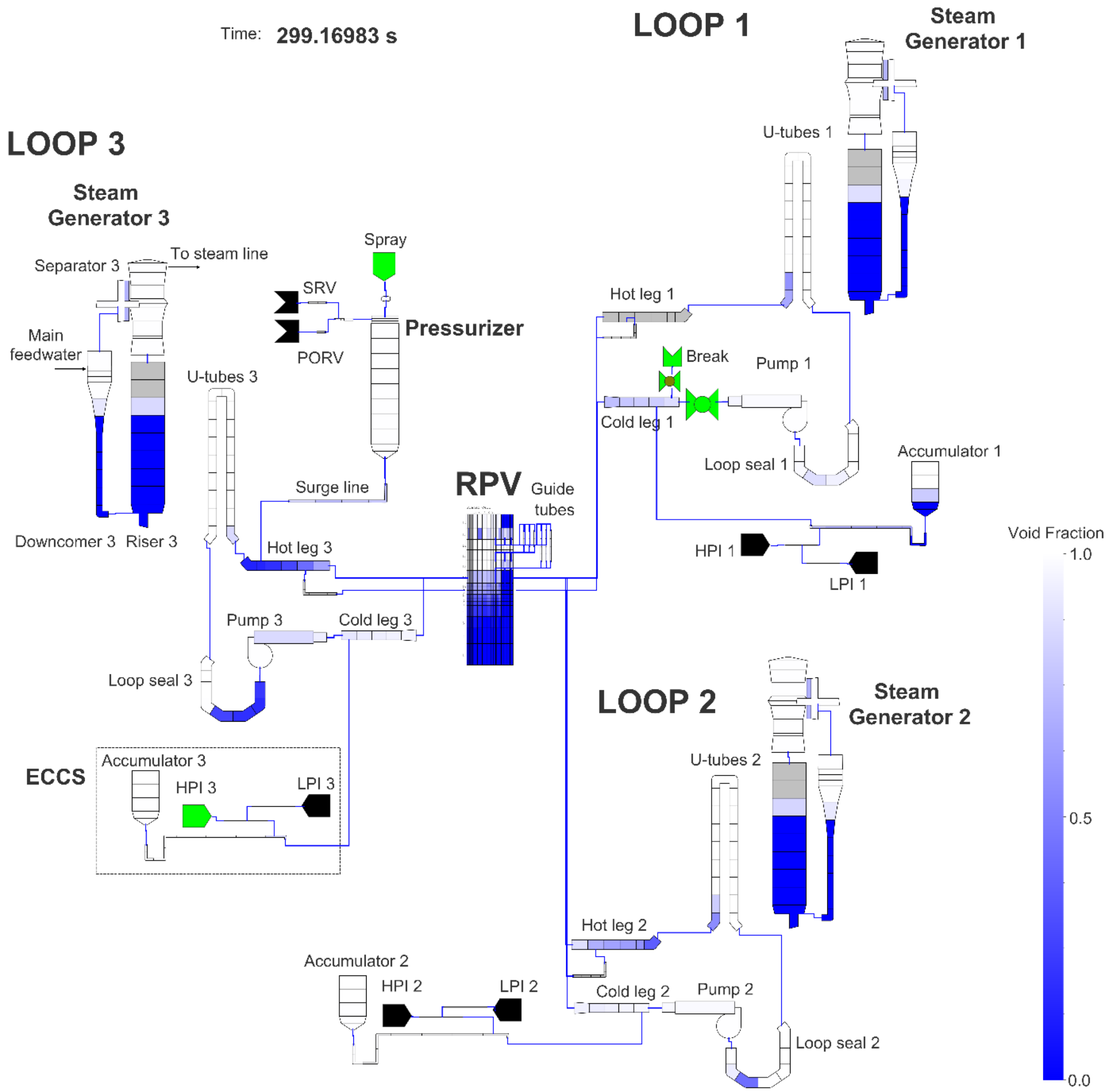

Figure 23 Void Fraction in 3-Loop PWR Plant at the End of the Transient

6.13 LSTF Model vs Standard PWR Model

In this section, a qualitative comparison between the experimental values (LSTF) and the simulation of both LSTF and Standard 3-loop PWR is shown.

There is a good agreement in timing between the experiment (LSTF) and the simulations of LSTF and the standard PWR. Nevertheless, some differences are found. Primary pressure (Figure 24) in the 3-loop model drops slightly earlier than in LSTF (experiment and simulation). This different behavior can be attributed to the different pump coast down used in the 3-loop and LSTF models. Pressure discrepancies can be also attributed to the different modelization of the break valve used in each case. In the LSTF model, the break has been simulated following the experiment conditions, while in the standard PWR model the break has been simulated as an orifice.

The accumulator system is initiated at about 120 s in LSTF (experiment and simulation) and about 100 s in the 3-loop PWR model, when the primary pressure is lower than 4.51 MPa. At this moment depressurization becomes effective due to steam condensation caused by the coolant injection in the cold legs. Finally, when primary pressure is about 1 MPa, Low Injection System is activated.

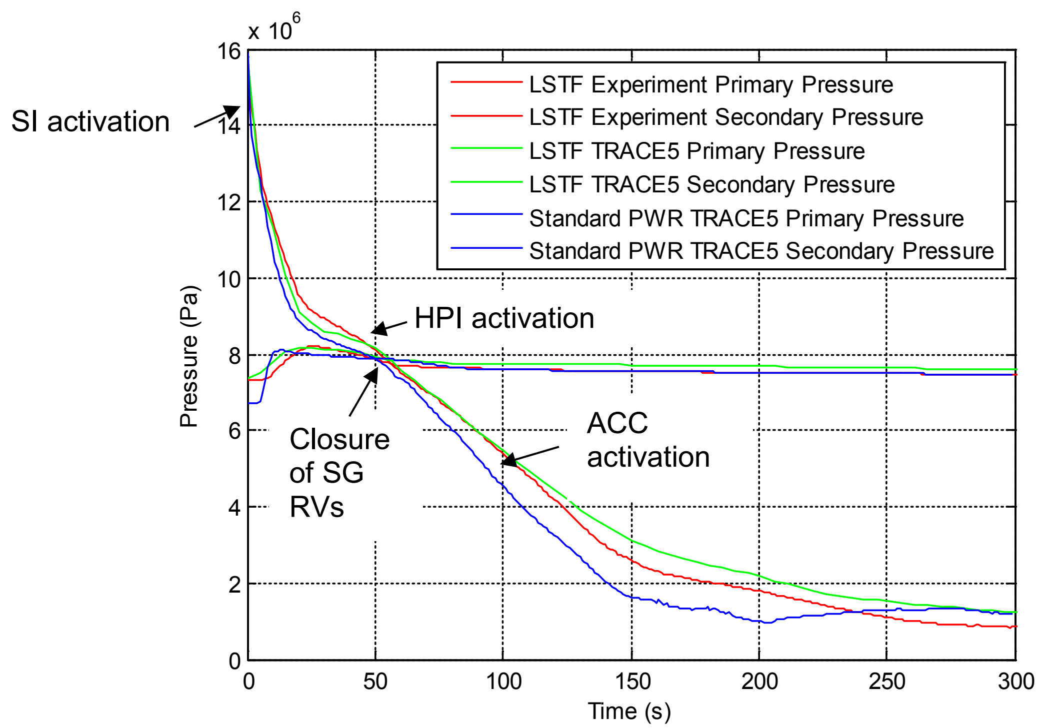

Figure 24 Primary and Secondary Pressures

Figure 25 shows the comparison between the mass flow rates through the break obtained in the LSTF and in the standard PWR divided by the volumetric factor scaling. As it can be seen, the break mass flow rate obtained in the standard PWR divided by the volumetric scaling factor shows a good agreement with the mass flow rate obtained in the LSTF.

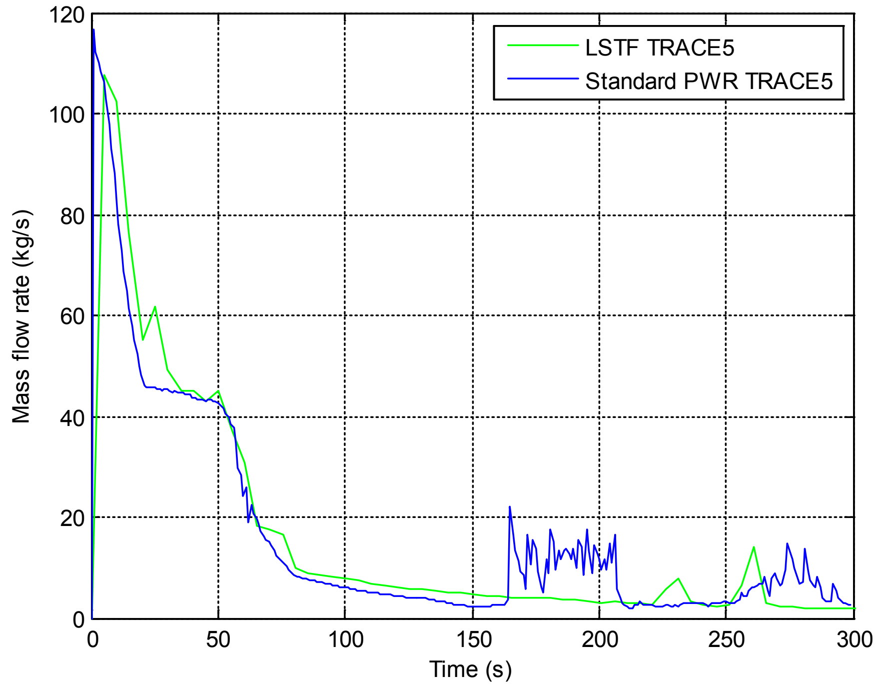

Figure 25 Break Mass Flow Rate Obtained In LSTF and in the 3-Loop PWR Plant Scaled

Figures 26 and 27 show the core and downcomer collapse liquid levels of LSTF (experiment and model) and the standard PWR model, respectively. As it can be seen, the general trend is similar in three cases, although in the standard PWR plant model the liquid level is recovered slightly rather than the LSTF (experiment and model). These discrepancies can be due to the mass flow rate injected by the AIS system in the standard PWR model.

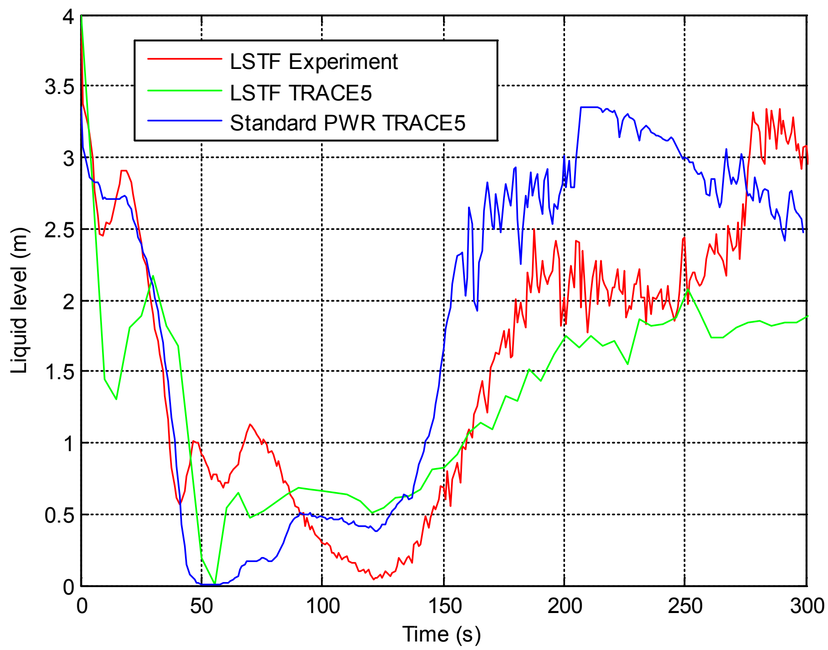

Figure 26 Core Collapsed Liquid Levels

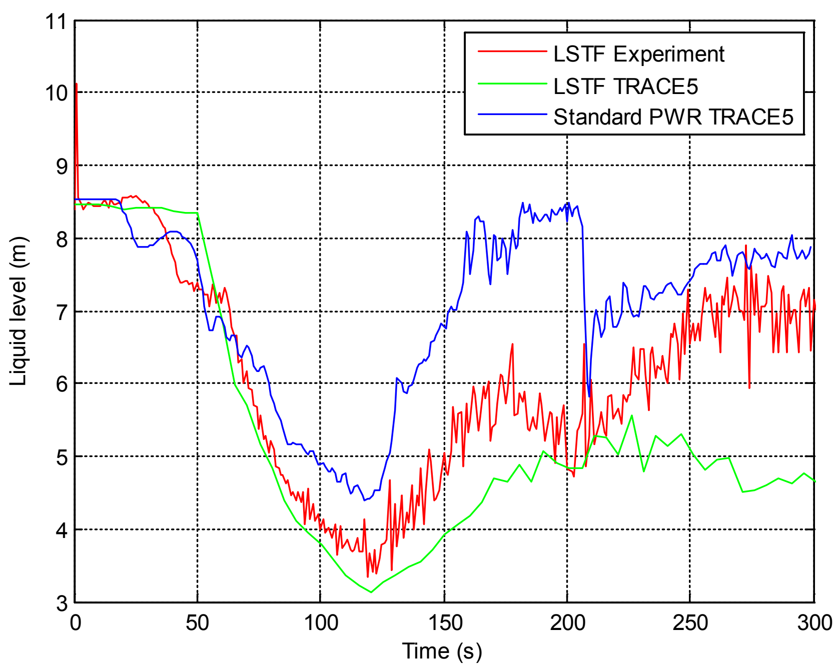

Figure 27 Downcomer Collapsed Liquid Levels

Regarding the temperatures of the system, in Figure 28 the maximum peak cladding (PCT) and the core exit temperatures (CET) are shown for the three cases considered (experiment, LSTF and standard PWR simulations). Both temperatures start to increase when the core clearance takes place, at 40 s approximately, in the experiment and in the standard PWR TRACE5 model. In the LSTF TRACE5 model, this increase is delayed some seconds. The maximum value of the PCT in the standard PWR simulation is reached at 130 s and the value is about 940 K. CET shows a similar behavior than the PCT, reaching its maximum value at a similar time. As it can be seen, standard PWR temperatures obtained using TRACE5 are very similar to the experimental data.

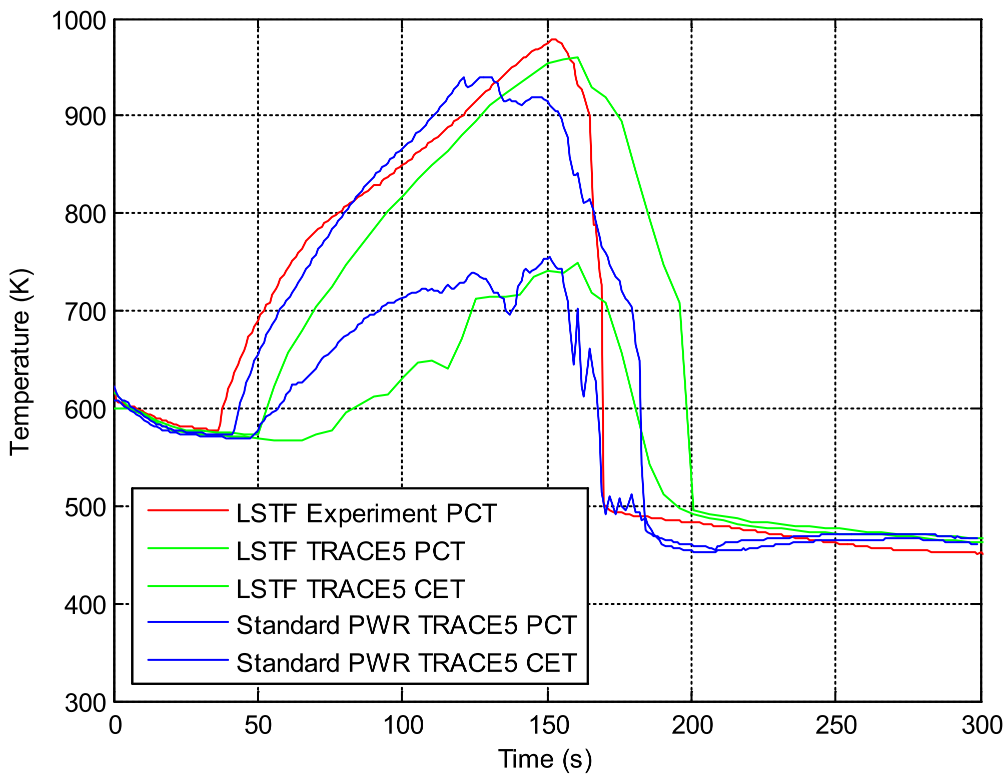

Figure 28 Maximum Fuel Rod Surface and Core Exit Temperatures

Figures 29 and 30 show the comparison between hot and cold legs collapsed liquid levels achieved in the experiment and in both simulations (LSTF and 3-loop PWR). As it can be seen, the refill in hot legs of the standard PWR is very similar to the LSTF. Regarding the cold leg, differences are more important, due to the discrepancies in the mass flow rate of the accumulator injection system.

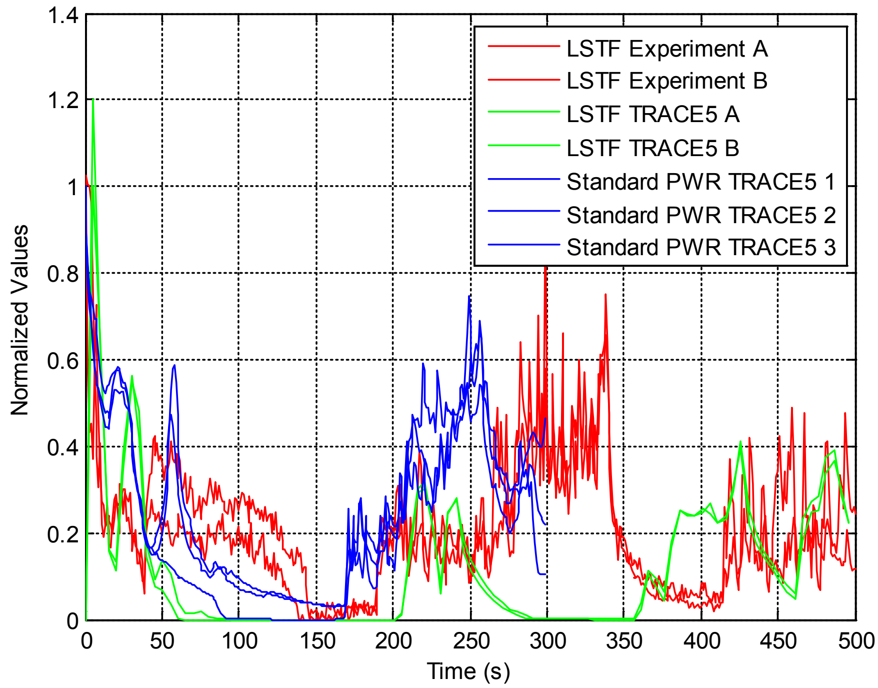

Figure 29 Hot Legs Collapsed Liquid Levels

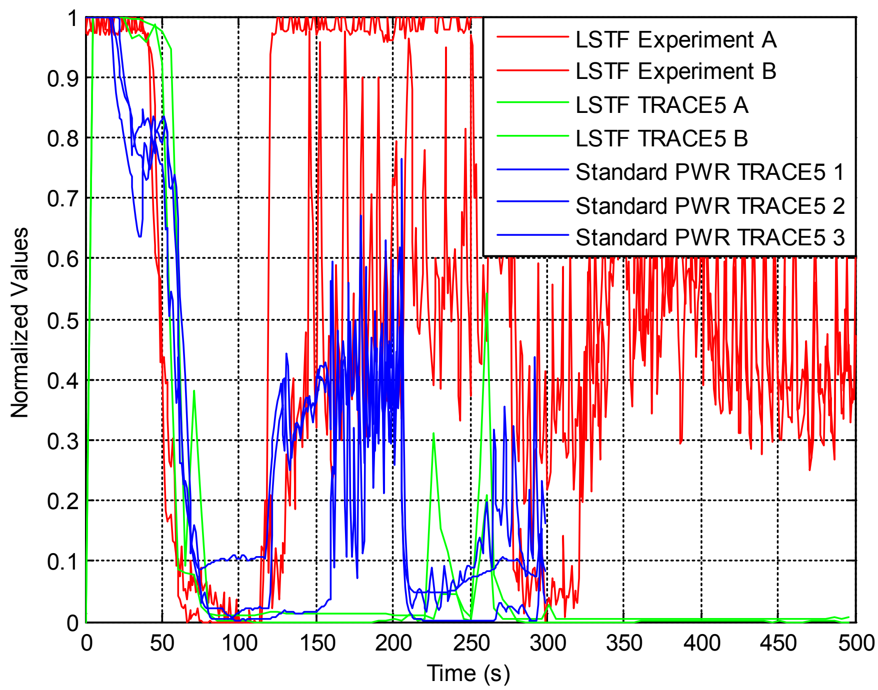

Figure 30 Cold Legs Collapsed Liquid Levels

7 CONCLUSIONS

This work presents results corresponding to the simulation of an Intermediate Break Loss-Of-Coolant Accident (IBLOCA) in the cold leg applied to a standard 3-loop PWR Nuclear Power Plant (NPP). The capability of the thermal-hydraulic code TRACE5 patch 2 in the simulation of a Cold Leg IBLOCA in a NPP has been tested. The experiment has been adapted to this case using the power-to-volume scaling methodology.

In general, TRACE5 can reproduce the significant events during an IBLOCA using the 3-loop PWR model. The simulation results are consistent with the experimental data obtained in the IBLOCA transient reproduced in the LSTF.

Although differences appear in design and scaling among the LSTF and the standard PWR TRACE5 model, it has been tested that using a volumetric scaling criterion the important physical phenomena are common to the LSTF and the 3-loop PWR TRACE5 model. The volumetric scaling criterion has been used in the break size and the ECCS injection rates (HPI, AIS and LPI). It has been observed that the mass flow rate through the break obtained in the 3-loop PWR divided by the volumetric scaling factor fits very well the break mass flow rate obtained in the LSTF.

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K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The purpose of this work is to overview the results provided by a first approach to the simulation of an Intermediate Break Loss-Of-Coolant Accident (IBLOCA) in a 3-loop PWR Nuclear Power Plant (NPP) using the thermal-hydraulic code TRACE5 patch 2 and the Symbolic Nuclear Analysis Packages software (SNAP) version 2.1.2.

The IBLOCA transient applied to the standard PWR TRACE5 model is the Test 2 (IB-CL-03) handled at the Large Scale Test Facility (LSTF) in the frame of the OECD/NEA ROSA-2 Project. Test 2 simulates a 17% cold leg IBLOCA under the assumption of the single-failure of High Pressure Injection and Low Pressure Injection systems and total failure of the Auxiliary Feedwater.

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

The simulation results are provided throughout several graphs, where the main system variables, such as pressures, pressure vessel liquid levels and temperatures are shown. These results represent a contribution to assess the predictability of computer codes such as TRACE5

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Accident Management (AM)
Committee on the Safety of Nuclear Installations (CSNI)
Consensus 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)

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