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ANALISIS OF SURGE LINE MBLOCA SEQUENCES WITH HPSI FAILED

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Abstract

The main objective of OECD/NEA ROSA-2 Project is to analyze thermal-hydraulic issues relevant to light water reactors (LWR) safety by using Large Scale Test Facility (LSTF) . As a part of this project ROSA-2 test1 has been performed in LSTF during 2009. This test consists of a double ended guillotine break in the surge line, surge line loss-of-coolant accident (SL-LOCA), with high pressure safety injection, (HPSI) failed and reactor coolant pumps trip simultaneously to reactor scram. As part of the participation of Universidad Politécnica de Madrid (UPM) group in OECD/NEA ROSA-2 and Code Applications and Maintenance Program (CAMP) projects two tasks related with this test have been performed. In the first task the simulation of ROSA-2 test1 has been performed with TRAC/RELAP Advanced Computational Engine (TRACE) code. Furthermore, an analysis of similar sequences in a Westinghouse pressurized water reactor (PWR) has been carried out; within this analysis a wide range of break area size has been analyzed: from 0.0254 m to 0.2794 m (1" to 11"), and a sensitivity analysis of delay time in the beginning of manual depressurization in the secondary side has been performed. The results show that the sequences with intermediate break sizes, from 0.0508 m to 0.1016 m (2" to 4") have worse consequences in this kind of sequences.

Introduction

The main objective of OECD/NEA ROSA-2 Project is to analyze thermal-hydraulic issues relevant to light water reactors safety by using LSTF facility. As a part of this project ROSA-2 test1 has been performed in LSTF facility during 2009. This test consists of a double ended guillotine break in the surge line (SL-LOCA), with HPSI failure and reactor coolant pumps trip simultaneously to reactor scram. As part of the participation of Universidad Politécnica de Madrid group in OECD/NEA ROSA-2 and CAMP projects two tasks related with this test have been performed. In the first task the simulation of ROSA-2 test1 has been performed with TRACE code.

The second task is related to the analysis of similar sequences in a Westinghouse PWR. As an application to a commercial PWR an analysis of Surge line, small and medium break loss-of-coolant accident (SBLOCA and MBLOCA) with HPSI failed was performed (similar to ROSA-2 Test 1). These simulations have been carried out with a TRACE model of Almaraz Nuclear Power Plant (NPP), a Westinghouse PWR-3 loop. As a complementary task a sensitivity analysis of this sequence with respect to the break size and the beginning of secondary side depressurization has been also performed. The objective of this analysis is to check the effectiveness of secondary side cooling during a surge line SBLOCA/MBLOCA without High

Pressure Safety Injection (HPSI) available. This analysis has been performed by the damage domain methodology which is described in Section 2.

1. Simulation of OECD/NEA ROSA-2 test1

LSTF is a full-height, full-pressure and 1/48 volumetrically scaled simulator for a Westinghouse-type 4-loop (3423 MWt) PWR with primary and secondary coolant systems including an electrically-heated simulated core, Emergency Core Cooling Systems (ECCSs) and control systems for Accident Management (AM) actions, see Figure 1 for more details. The maximum core power of 10 MW is equivalent to 14% of the 1/48-scaled PWR rated power covering the scaled PWR decay heat after the scram.

The group of Universidad Politécnica de Madrid is working with TRACE model of ROSA/LSTF since February 2006. The development of the TRACE model of LSTF/ROSA is based on the TRAC-PF1 model delivered by Japan Atomic Energy Research Institution (JAERI) to the participants of OECD/NEA ROSA project.

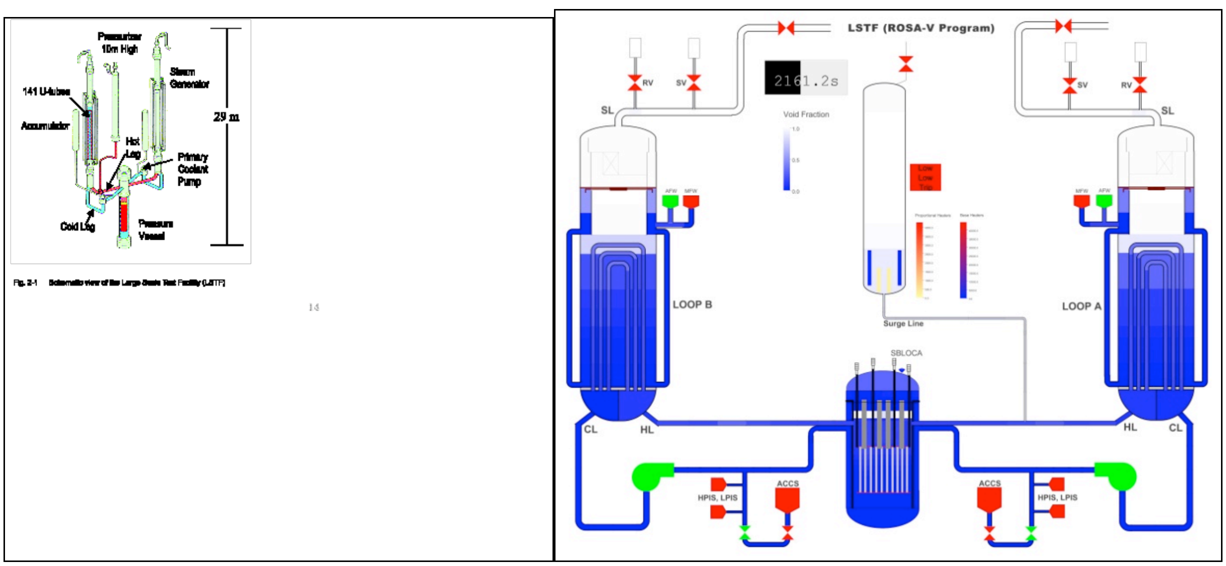

Figure 1 ROSA/LSTF Facility and Symbolic Nuclear Analysis Package (SNAP) mask.

As part of the participation of Universidad Politécnica de Madrid group in OECD/NEA ROSA-2 and CAMP projects the simulation of ROSA2 test1 has been performed with TRACE code. As it can be seen in Table 1 and Figures 2 and 3, the test was correctly simulated with the TRACE model. The primary and secondary pressures match fairly well with the experimental results until accumulators discharge. The results also show a good prediction of the core uncovering as well as the Peak Cladding Temperature (PCT). During the simulations it has been observed that PCT is quite sensitive to the actuation pressure of accumulators. The data are showed normalized because they are proprietary until 2012. Other groups participating in OECD/NEA ROSA-2 project have also simulated this test obtaining, in general, good results.

Event	Experimental (s)	TRACE (s)
Isolation of PZR	1940	1940
Break opening	2000	2000
Turbine trip (closure of stop valve)	2001	2001
Scram signal and Isolation of main steam line	2004	2004
Initiation of reactor coolant pump (RCP) coastdown and termination of main feedwater	2005	2005
Initiation core power decay	2020	2022
Initiation of accumulators discharge	2154	2160
Initiation of cladding temperature increase	2164	2165
PCT	2182	2175

Table 1: Description of ROSA-2 Test 1. Chronology of major events and procedures

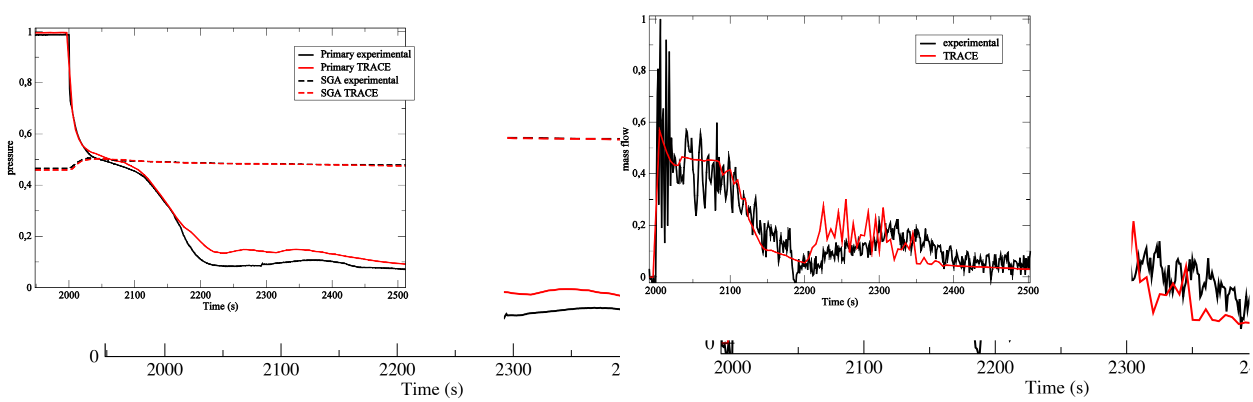

Figure 2 ROSA-2 test1. Primary/secondary pressures and discharge mass flow rate.

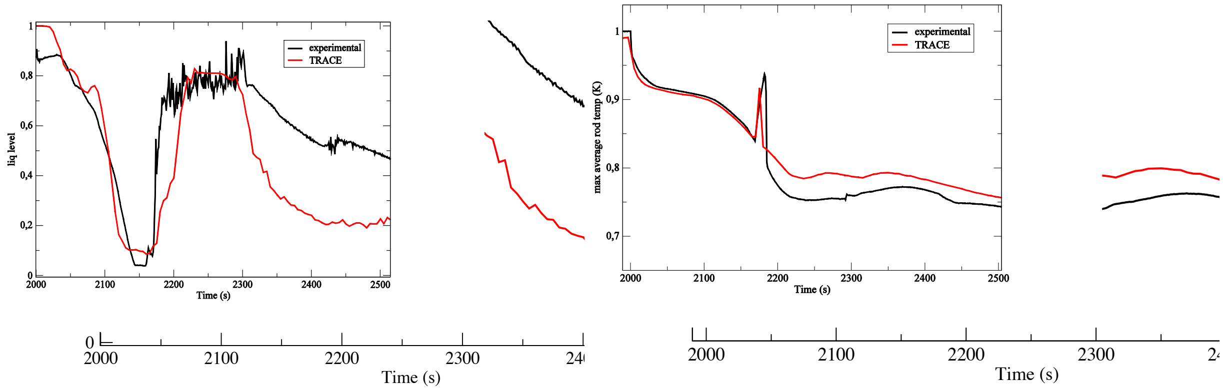

Figure 3 ROSA-2 test1. Downcomer level and PCT.

2. Simulation of similar sequences in a PWR Westinghouse

As an application to a commercial PWR an analysis of Surge line MBLOCA with HPSI failed was performed (similar to ROSA-2 Test 1). This analysis has two different stages:

1. Simulation of a similar sequence in a PWR Westinghouse.
2. Sensitivity analysis of this sequence with respect to the break size and the beginning of secondary side depressurization.

These simulations have been performed with a TRACE model of Almaraz NPP. This TRACE model has 255 thermal-hydraulic components (2 VESSEL, 73 PIPE, 43 TEE, 54 VALVE, 3 PUMP, 12 FILL, 33 BREAK, 32 HEAT STRUCTURE and 3 POWER component), 740 SIGNAL VARIABLES, 1671 CONTROL BLOCKS and 58 TRIPS, Figure 4.

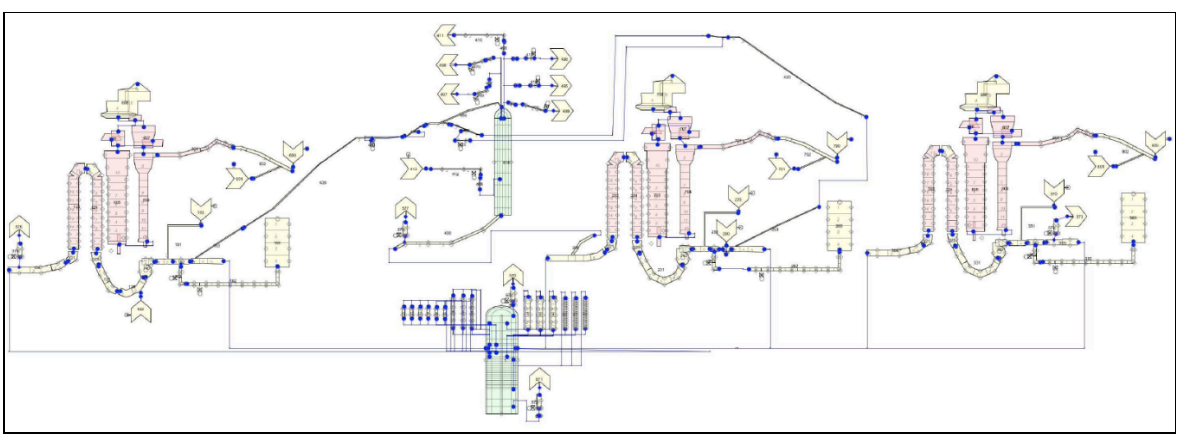

Figure 4 Simplified scheme of the Almaraz NPP TRACE model

In the first stage of the analysis several conditions have been imposed in Almaraz NPP model,

- LOCA in surge line with an equivalent break size with respect to ROSA-2 Test 1.
- No HPSI available (similar to ROSA-2 Test 1).

- RCP trip at the beginning of the transient (similar to ROSA-2 Test 1).
- Three accumulators available.
- Mean Steam Isolation Valve (MSIV) closed by high pressure inside containment.
- One train of Low Pressure Safety Injection (LPSI) is available.
- Several discharge coefficients (for liquid and two phases flow) have been tested in order to obtain an uncertainty band (from 0.75 to1.0 and from 1.0 to 1.1).

The results obtained from the simulations of Almaraz NPP, show a similar primary and secondary pressure transient with respect to ROSA-2 test 1, see Figure 5. However, there is no core uncovering in the simulation of Almaraz NPP, as it can be observed in the results of PCT, Figure 5.

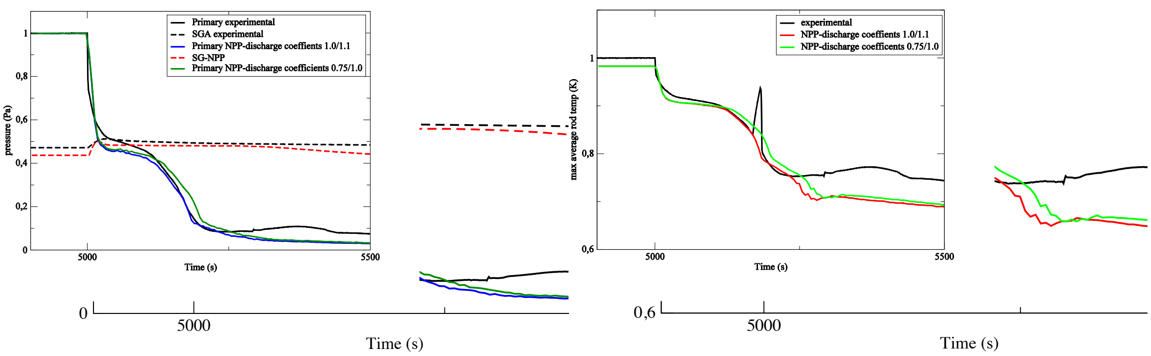

Figure 5 ROSA-2 test1. Primary/secondary pressures and PCT. Comparison between ROSA/LSTF and Almaraz NPP

As complementary task was decided to perform a wide sensitivity analysis with respect to two parameters:

- Beginning of secondary side depressurization (called as S(t)) to cool (55 K/h as cooling rate) and depressurize primary side.

- Break area: A wide range of break area has been analyzed, from 0.0254 m to 0.2794 m (1" to 11") diameter.

The objective of this analysis is to check the impact of the break size and the effectiveness of secondary side cooling during a surge line SBLOCA/MBLOCA without HPSI available. To perform this analysis has been selected the damage domain methodology. The damage domain (DD) is defined as the region of the space of parameters of interest that results in damage. The UPM group has applied extensively this methodology in several projects, for more details see Izquierdo (2008a), Hortal (2010), Ibáñez (2010) and Queral (2011).

The calculation process is described below (48 cases were simulated with TRACE code):

1. Failure of S header (no manual depressurization in secondary side) is assumed. A transient (path) is simulated for each considered break diameter, see figure 6, the horizontal line labelled as 1477K (2200 F) represents the acceptance criteria for PCT. In these sequences, damage will arrive at a certain time (t0), which sets the minimum time for the beginning of manual depressurization. Starting depressurization later than t0 time

is not useful to avoid damage. These time points form the line of Previous Damage (PD) shown in Fig. 8.

2. Furthermore, analyzing the evolution of the average temperature in the reactor coolant for the paths corresponding to PD line, it is possible to check from which time the manual depressurization is inefficient, i.e. the cooling rate due to depressurization is higher than 55 Kelvin per hour without human actions. Therefore the results of PD could be extended until the inefficiency depressurization line, see in Fig. 8.

3. As shown in Fig. 7 and 8, a set of paths are simulated with different times for the beginning of depressurization, always below the efficiency line. Some of the paths exceed the damage condition (red diamond) while other paths do not reach it (green circle).

4. With these results, it is possible to obtain the DD for this sequence, see lower part of Fig.8, by connecting the first depressurization time that leads to damage for each break size. In this DD it can be noted that there are damage paths with accumulator demand and others without it. This difference must be taken into account to calculate probabilities or frequencies.

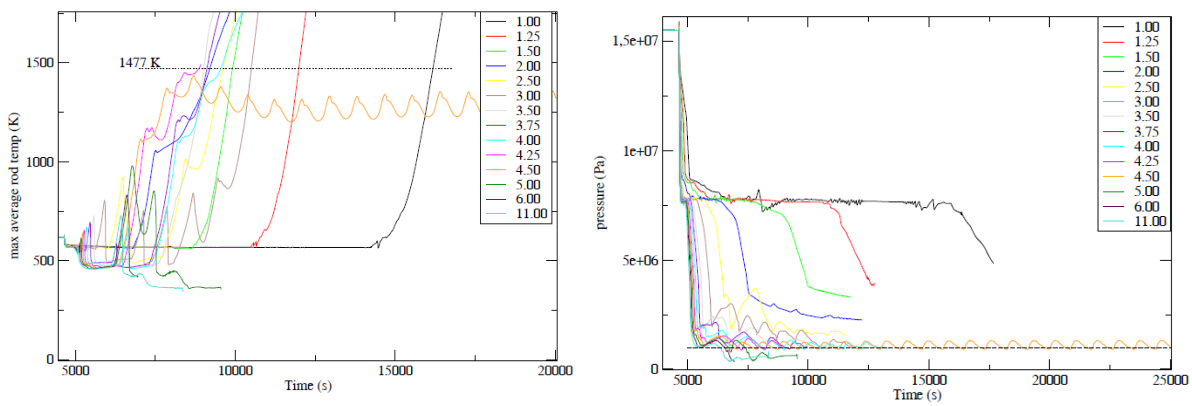

Figure 6 PCT and primary pressure. Sensitivity analysis with respect to the break size. No secondary side depressurization. TRACE code.

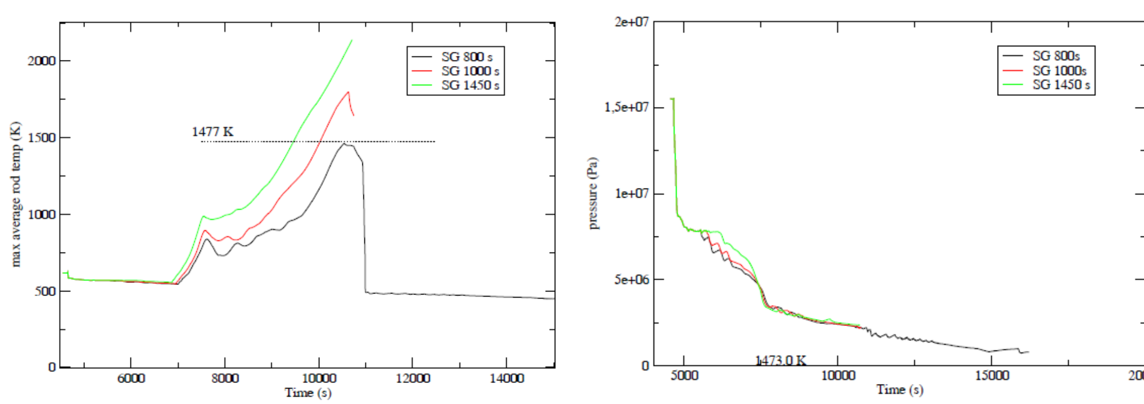

Figure 7 PCT and primary pressure. Sensitivity analysis with respect to the beginning of secondary side depressurization. Break diameter: 0.0508m (2”). TRACE code.

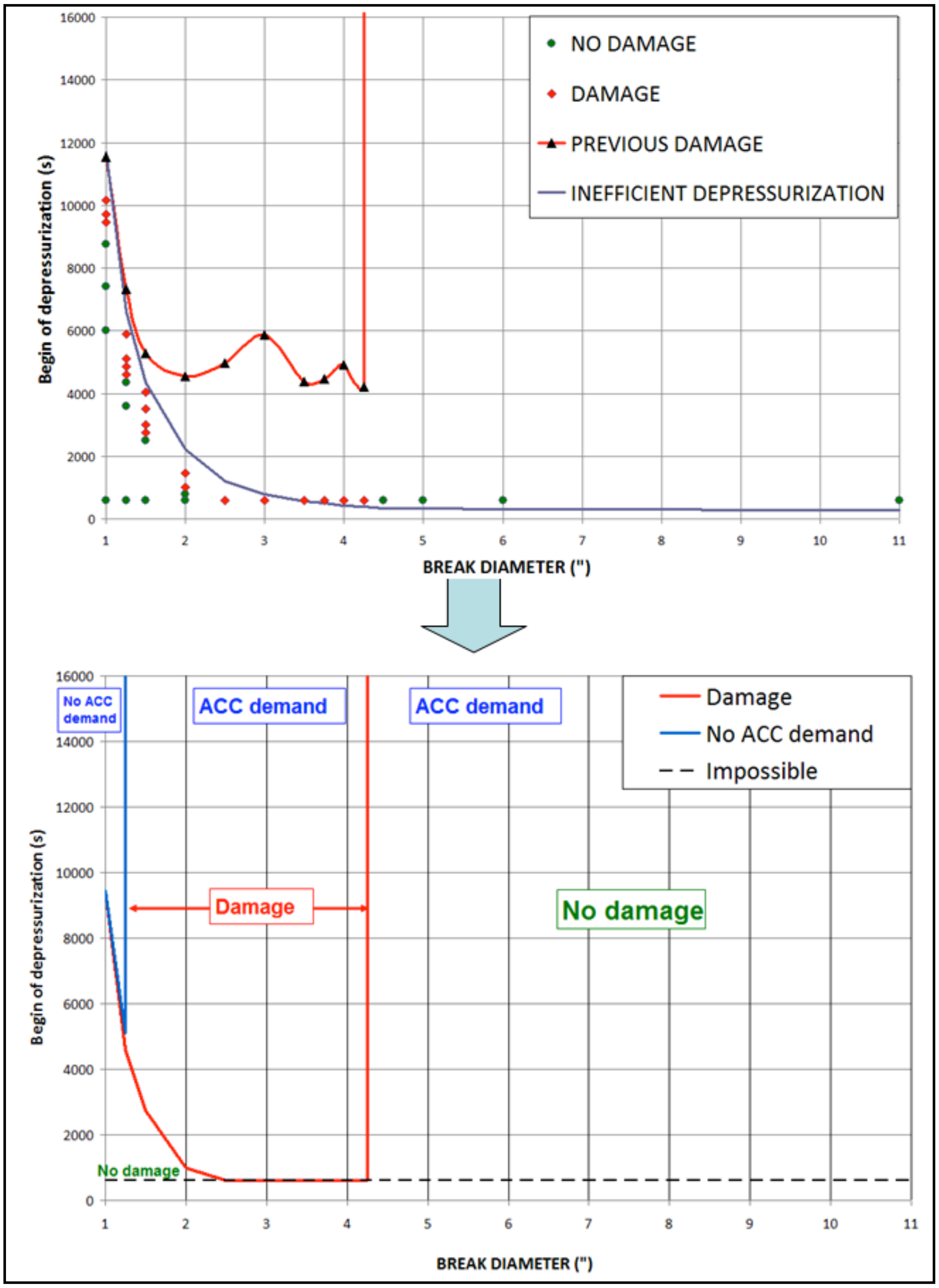

Figure 8 Procedure for obtaining the DD of surge line MBLOCA with HPSI failed. TRACE code.

The results show that the sequences with intermediate break sizes, from 0.0508 m to 0.1016 m (2" to 4") have worse consequences for this kind of sequences. In SBLOCA sequences, from 0.0254 m to 0.0508 m (1" to 2") there is an available time in order to avoid core damage. However in MBLOCA (without HPSI and with RCP trip) this manual action is not useful.

3. Conclusions

The comparison between ROSA-2 test 1 and the simulations performed in TRACE code show that the primary and secondary pressures match fairly well with the experimental results until accumulators discharge. The results also give a good prediction of the core uncovering, as well as the PCT, being this one quite sensitive to the actuation pressure of accumulators.

On the other hand, the results obtained from the simulations of Almaraz NPP show a similar primary and secondary pressure transient with respect to ROSA-2 test 1 but it must be noted that there is no core uncovering in the simulation of Almaraz NPP. Nevertheless, TRACE code has showed a good capability for simulating similar sequences to ROSA-2 test 1.

In reference to the complementary task (sensitivity analysis with respect to the beginning of secondary side depressurization and break area) the results show that the sequences with intermediate break sizes, from 0.0508 m to 0.1016 m (2" to 4") have worse consequences for this kind of sequences. In SBLOCA sequences, from 0.0254 m to 0.0508 m (1" to 2") there is an available time in order to avoid core damage. However in MBLOCA (without HPSI and with RCP trip) this manual action is not useful. Furthermore, it is necessary to check the RCP trip hypothesis in the analysis, because in actual sequences, and taking into account Westinghouse procedures, there is not RCP trip (in the Emergency Operating Procedure E-1 (LOCA sequence) the primary pumps are not tripped if there is not HPIS mass flow). It is important to analyze cases without RCP trip and delay trip.

The damage domain methodology is an useful tool to analyze the sequences in nuclear power plants. This kind of analysis is part of a more wide methodology called Integrated Safety Analysis Methodology (ISA) which has been developed by the Modelization and Simulation area (MOSI) of Spanish Nuclear Safety Council (Consejo de Seguridad Nuclear, CSN).

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