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Break location influence in pressure vessel SBLOCA scenarios

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ABSTRACT

The inspections performed in Davis Besse and the South Texas Project Unit-I reactors pointed out safety issues regarding the structural integrity of the Pressure Vessel (PV). In these inspections, two anomalies were found: a wall thinning and degradation in the PV upper head of the Davis Besse reactor and a small amount of residue around two instrument-tube penetration nozzles located in the PV lower plenum of the South Texas Project Unit-I reactor. The evolution of these defects could have resulted in Small Break Loss-Of-Coolant Accidents (SBLOCA) if they had not been detected in time. In this frame, the OECD/NEA considered the necessity to simulate these accidental sequences in the OECD/NEA ROSA Project using the Large Scale Test Facility (LSTF). This work is focused on simulating different hypothetical accidental scenarios in the PV using the thermal-hydraulic code TRACE5. These simulations allow studying the break localization influence in the transient and the effectiveness of the accident management (AM) actions considered mitigating the consequences of these hypothetical accidental scenarios.

Keywords: **SBLOCA, AM actions.**

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1. INTRODUCTION

Inspections performed at existing Nuclear Power Plants (NPP) have pointed out the possibility that some deficiencies in the Pressure Vessel (PV) could have initiated some accidental sequences. This is the case of the inspections performed in the Davis Besse reactor and the South Texas Project Unit-I reactor. In Davis Besse NPP (2002), a wall thinning and degradation in the reactor PV upper head was discovered as a result of circumferential cracking of a control rod drive mechanism penetration nozzle [1, 2]. In the South Texas Unit-I reactor, a small amount of residue was detected around two instrument-tube penetration nozzles located in the PV lower plenum [3]. In both cases, these anomalies could result in small Break Loss-Of-Coolant Accidents (SBLOCA), if they had not been detected in time.

The intergovernmental agency NEA (Nuclear Energy Agency), within the framework of the Organization for Economic Co-operation and Development (OECD), assists its member countries in maintaining and further developing all involved areas in nuclear safety through international cooperation, i.e., science, technology, environment and law. Thus, NEA promotes joint projects with the aim of achieving this purpose and facilitates coordination among countries. In this frame, the OECD/NEA considered the importance of studying the accidental sequence due to a small break in the upper head in the frame of the OECD/NEA ROSA Project, using the Large Scale Test Facility (LSTF). The LSTF is a full-height and 1/48 volumetrically scaled test facility of the Japan Atomic Energy Agency (JAEA) used for safety research and safety assessment of Light Water Reactors (LWR) [4].

Among the experimental series performed within the OECD/NEA ROSA Project, Test 6-1 [5, 6, 7, 8] reproduces an SBLOCA in the PV upper head with a break size equivalent to 1.9 % of the cold leg flow area. The total failure of the High Pressure Injection (HPI) system is assumed, and the Steam Generators (SG) secondary depressurization was performed as an Accident Management (AM) action.

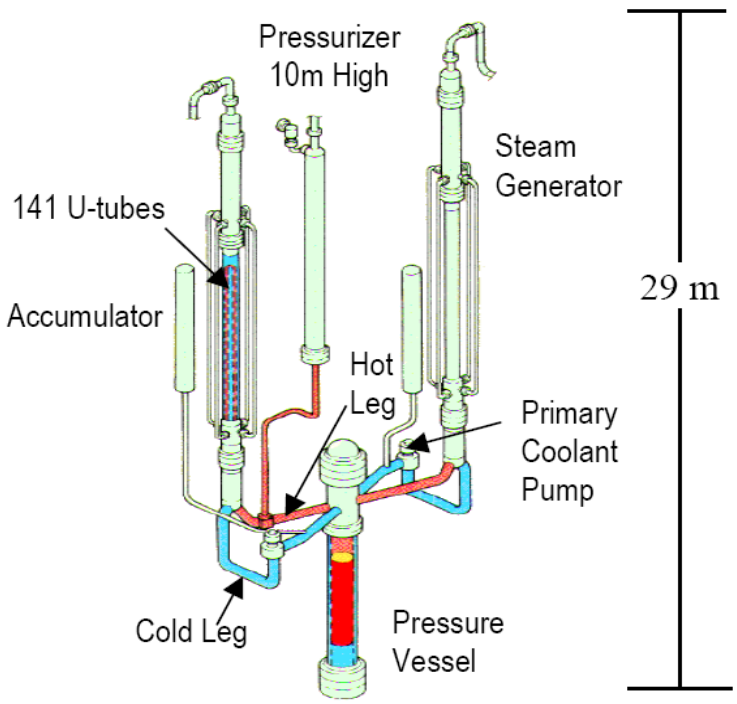
This work is focused on the study of the break localization influence. The thermal-hydraulic code TRACE5 has been used to develop an LSTF model. Different hypothetical accidental scenarios have been simulated. In these cases, the break size and the AM actions are the same, but the break localization has been changed from the upper head to the lower plenum and the downcomer. These simulations allow studying the break localization influence in facility response through the different thermal-hydraulic parameters evolution. In addition, it is compared the effectiveness of the AM action considered mitigating the consequences of these hypothetical accidental scenarios.

2. MATERIALS AND METHODS

2.1. Test facility description

The LSTF [4] simulates a PWR reactor, Westinghouse type, of four loops and 3423 MW of thermal power. The LSTF is a Full-Pressure Full-Height (FPFH) facility and 1/48 volumetrically scaled. The LSTF represents the four loops of the reference PWR using two equal-volume loops (1/24 volumetrically scaled). The schematic view of the LSTF is shown in Figure 1.

Figure 1: *Schematic view of the LSTF [4].*



The primary coolant system consists of the PV, the loop A with the pressurizer (PZR) and the loop B. The LSTF has an Emergency Core Cooling System (ECCS), which consists of the High Pressure Injection (HPI), Low Pressure Injection (LPI) and the Accumulators Injection (AIS) systems. The secondary system is simplified into two steam generators (SG) with the Main and the Auxiliary Feed Water systems (MFW and AFW, respectively) and a steam line.

The PV is composed of an upper head above the upper core support plate, the upper plenum between the upper core support plate and the upper core plate, the core, the lower plenum and the downcomer annulus region surrounding the core and the upper plenum. LSTF vessel has 8 upper head spray nozzles (of 3.4 mm inner diameter). Eight Control Rod Guide Tubes (CRGTs) form the flow path between the upper head and the upper plenum. The maximum LSTF core power is limited to 10 MW, which corresponds to 14% of the volumetrically scaled PWR core power. LSTF has 1008 heated rods in the active core, which reproduce the total number of the reference PWR scaled by 1/48. Each steam generator contains 141 U-tubes, which can be classified into different groups depending on their length. The U-tubes have an inner diameter of 19.6 mm and an outer diameter of 25.4 mm (with 2.9 mm wall thickness). The vessel, plenum, and riser of steam generators have an inner height of 19.840, 1.183 and 17.827 m, respectively. The downcomer is 14.101 m in height.

2.2. LSTF model

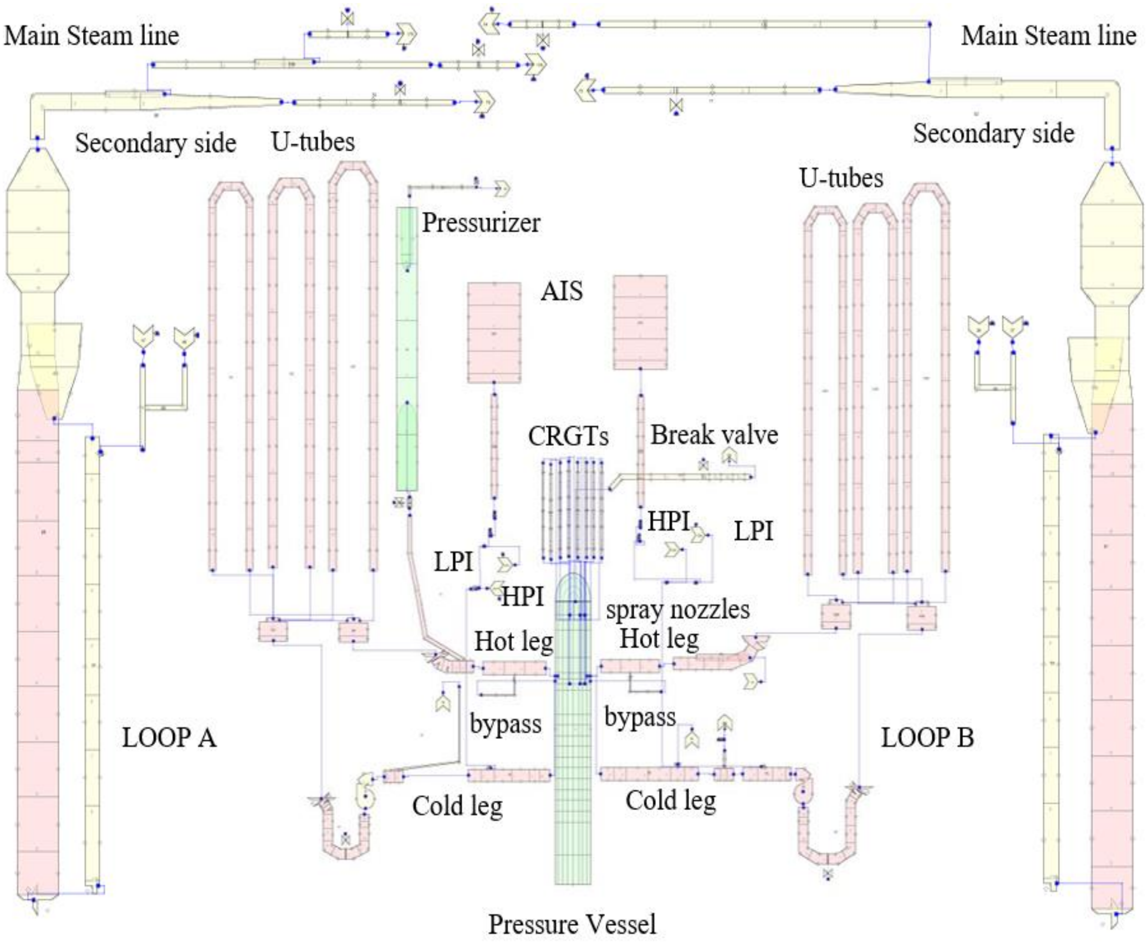
The LSTF has been reproduced using the thermal-hydraulic code TRACE5 [9, 10] developed by the U.S. Nuclear Regulatory Commission (USNRC). TRACE is the latest in a series of best-estimate reactor system codes for analyzing neutronic and thermal-hydraulic behavior in Light Water Reactors (LWR). 83 TRACE5 hydraulic components (8 BREAKs, 11FILLs, 23 PIPEs, 2 PUMPs, 1 PRIZER, 22 TEEs, 15 VALVEs and 1 VESSEL) have been used in the LSTF model [4].

The 3-D VESSEL nodalization is shown in Table 1. Besides, Figure 2 shows the nodalization of the LSTF model using the Symbolic Nuclear Analysis Package software (SNAP).

Table 1: 3-D VESSEL nodalization

3-D VESSEL nodalization	Levels
Lower plenum	Axial 1 and 2
Active core	Axial 3 to 11
Upper plenum	Axial 13 to 15
Upper head	Axial 17 to 19
Downcomer	Axial 1 to 19; 4th ring

Figure 2: *LSTF nodalization*



The discretization of the model is a determining factor in the results of the simulations. This fact led up in the past to perform different studies to define the distribution of nodes in the vessel or the U- tube bundles of the steam generators. Previous works [7, 8, 11, 12, 13] have served to verify the model by reproducing different ROSA experiments. Specifically, the capability of the model to correctly reproduce Test 6.1 has been confirmed, comparing simulated and experimental results. In addition, it is expected to be able to reproduce another relevant phenomenology in this type of transients.

The break unit consists of a VALVE component connected with a BREAK component, which simulates the discharge at atmospheric conditions. The geometric characteristics of the VALVE have been obtained from the orifice specifications of Test 6-1 [5].

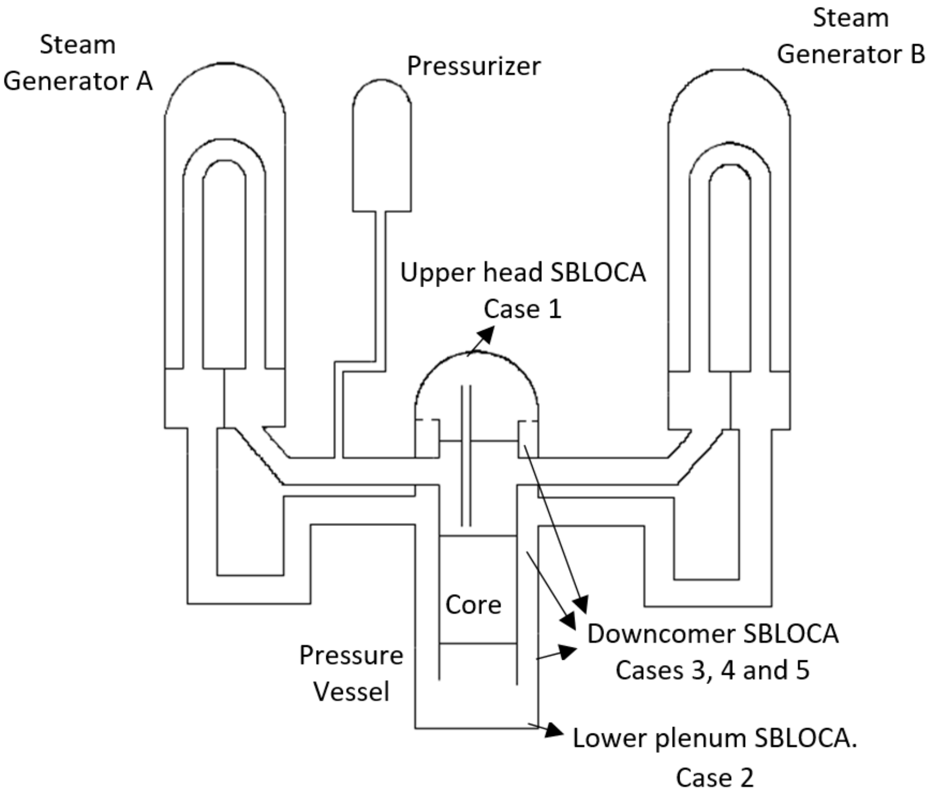
In this work, some hypothetical scenarios have been tested, in which the break size and the AM actions are the same as in Test 6-1, but the break localization has been changed from the upper head to the lower plenum and the downcomer. In the downcomer, three different positions have been considered. The hypothetical cases considered are summarized in Table 2.

Table 2: Cases simulated

Case	Location and nodalization
Case 1	SBLOCA Upper head. Case base Test 6-1 VALVE connected to axial level 20 of 3D-VESSEL.
Case 2	SBLOCA Lower plenum VALVE connected to axial level 1 of 3D-VESSEL.
Case 3	SBLOCA Downcomer VALVE connected to axial level 3 of 3D-VESSEL.
Case 4	SBLOCA Downcomer VALVE connected to axial level 10 of 3D-VESSEL
Case 5	SBLOCA Downcomer VALVE connected to axial level 17 of 3D-VESSEL.

Figure 3 shows the break localizations analyzed in this work.

Figure 3: *Break localizations simulated in LSTF (1.9 % break size)*



2.3. Test description

The experiment starts with the break valve opening. When the primary pressure drops to 12.97 MPa, the scram signal is generated producing the core power decay, the primary Reactor Coolant Pump (RCP) coastdown, the Main Feed Water (MFW) termination and the closure of the Main Steam Isolation Valves (MSIVs). The primary pressure continues dropping reaching the Safety Injection (SI) signal set point. Immediately after, the Auxiliary Feed Water (AFW) injection is activated. After that, the primary pressure continues decreasing near the secondary one, remaining slightly above it since then. From the MSIVs closure, the Relief Valves (RV) in both SG begin opening and closing to maintain the secondary pressure. During this time, the secondary side keeps removing heat from the primary system while the natural circulation is still on. Once the U-tubes are empty and natural circulation ends, the primary pressure begins to fall below the secondary one, which is stabilized. When the Core Exit Temperature (CET) reaches 623 K, depressurization of the secondary side is initiated as AM action by fully opening the RVs of both SG. When the primary pressure reaches 4.51 MPa, the Accumulator Injection System (AIS) is activated. The Low Pressure Injection (LPI) system is actuated when the pressure in the PV lower plenum is 1.23 MPa. The transient finishes when the system pressures are stabilized. The control logic is listed in Table 3.

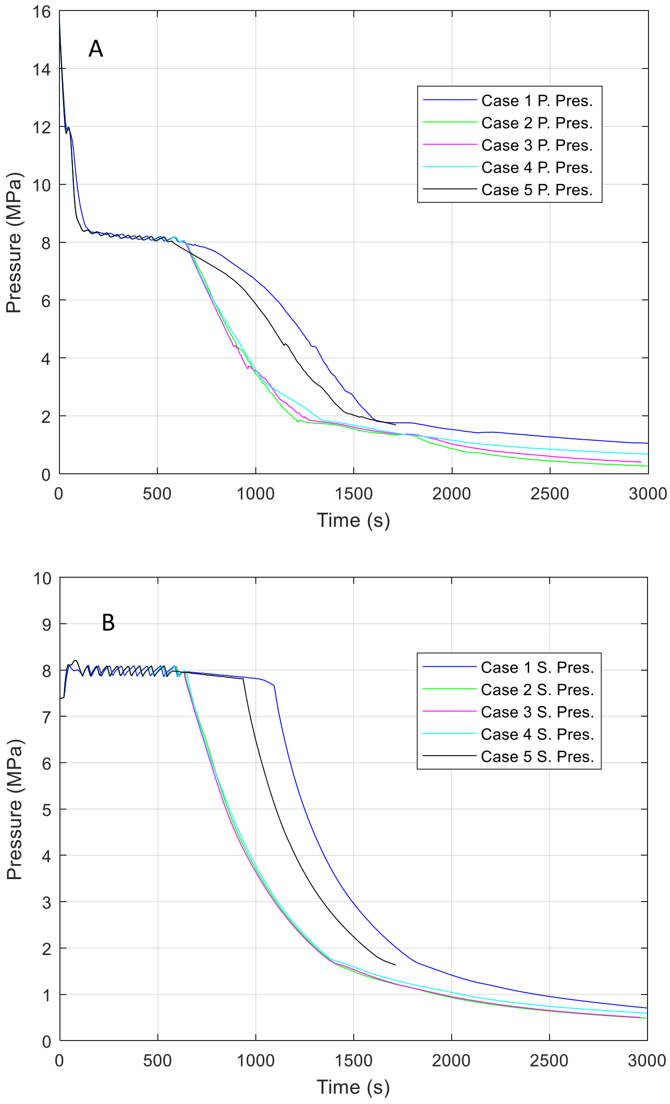
Table3: Chronology of the main events

Event	Signal
Break valve open	Time zero
Scram signal	Primary pressure = 12.97 MPa
Pressurizer heater off	Scram signal
Core power decay curve	Scram signal
Primary coolant pump coastdown	Scram signal
Closure of MSIV	Scram signal
End of MFW	Scram signal
Safety Injection (SI) signal	Primary pressure = 12.27 MPa
Start AFW	SI signal
SG depressurization as AM	CET reaches 623 K
AIS	Primary pressure = 4.51 MPa
LPI	PV lower plenum pressure = 1.24 MPa

3. Results and discussion

In this section, the simulation results of the hypothetical scenarios are described. Figure 4 shows the primary and secondary pressures obtained in each case.

Figure 4: *Pressures A) Primary system B) Secondary system.*



As can be seen, during the first part of the transient the system pressures are similar in all the cases. The initial primary pressure drop is produced within 250 s in all the tests. Then, the primary pressures closely follow the secondary ones, until 650 s, when the secondary depressurization is produced (Case 2, Case 3 and Case 4). In Case 5, this secondary depressurization is produced at 950 s. In all the cases, the depressurization is advanced in comparison to the case base (upper head LOCA).

Figure 5 and Figure 6 show the mass flow rate and the discharged inventory through the break, respectively. The break flow rate decreases as the flow changes from subcooled liquid to two-phase flow and then to single-phase vapor. These flow-quality transitions are shown in the graph of the discharged inventory. As can be seen, the time at which the flow-quality transitions occurred is greatly dependent on the break localization, in agreement with the results obtained by other authors [14].

The break flow turns from subcooled liquid to two-phase flow at 65 s in all the cases, except in Case 1. In this case, the flow transition is some seconds advanced since at the highest levels of the facility steam is formed before. Regarding the change from two-phase flow to single-phase vapor (at 650 s, approximately), more differences are observed. In general, the tendency is the flow changes earlier for breaks located at a higher elevation. Furthermore, when the break is located at higher elevations, the mass flow rate discharged through the break is lower. It can be seen in Table 4, which shows the discharged inventory through the break for different cases at 1500 s.

Table 4: Discharged inventory through the break at 1500 s

Case	Discharged inventory (kg)	Core level (m)
Case 1	4610	2.8
Case 2	6420	0
Case 3	6010	1.3
Case 4	5680	2.95
Case 5	4670	2.9

The higher break mass flow rate produces a larger inventory loss in the vessel, advancing slightly the Core Exit Temperature (CET) and Peak Cladding Temperature (PCT) excursions in the Case 5 and with a more remarkable effect in the Cases 2, 3 and 4. Consequently, the AM action is activated earlier, and the secondary side depressurization is advanced.

Figure 5: *Mass flow rate through the break*

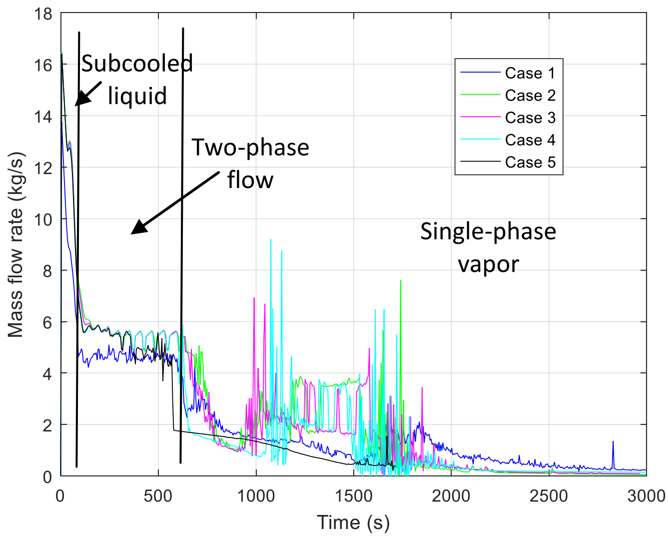


Figure 6: Discharged inventory

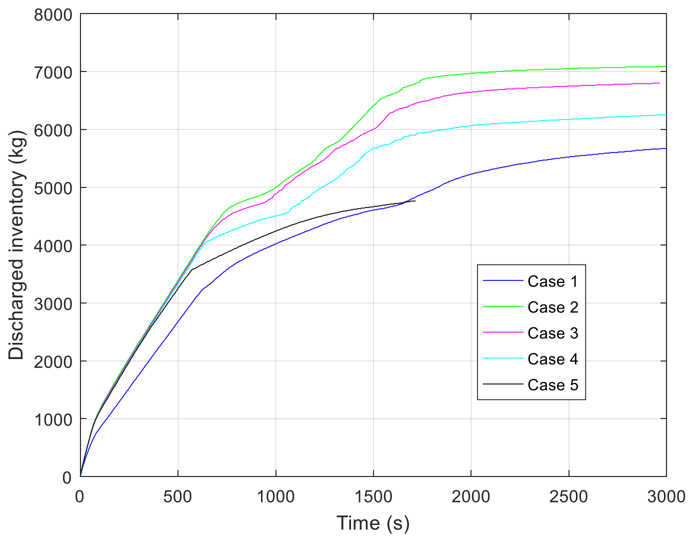
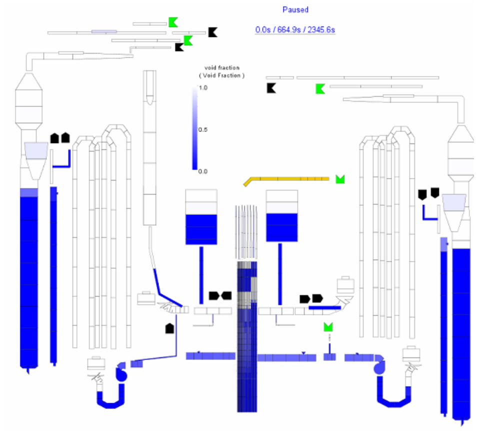
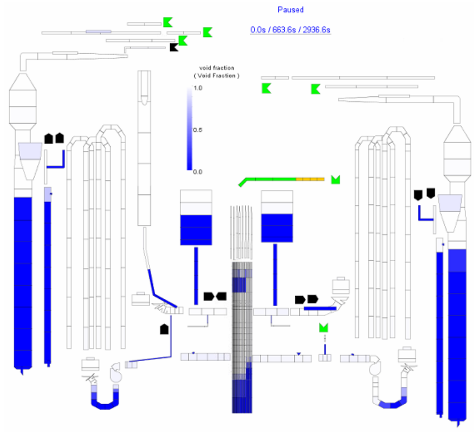


Figure 7 shows the void fraction at 650 s, at which the change from two-phase flow to single-phase vapor is produced.

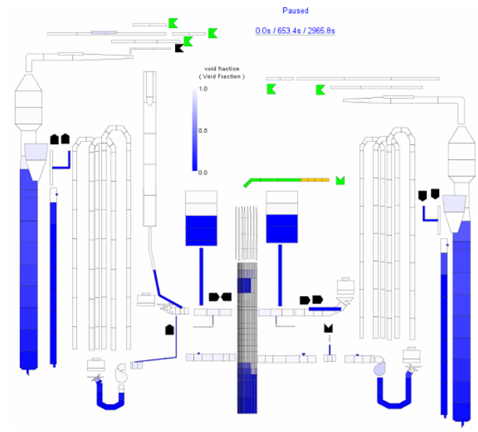
Figure 7: *Void fraction at 650 s for the different cases*



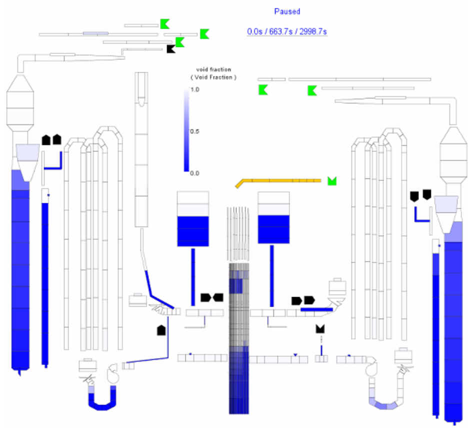
Case 1



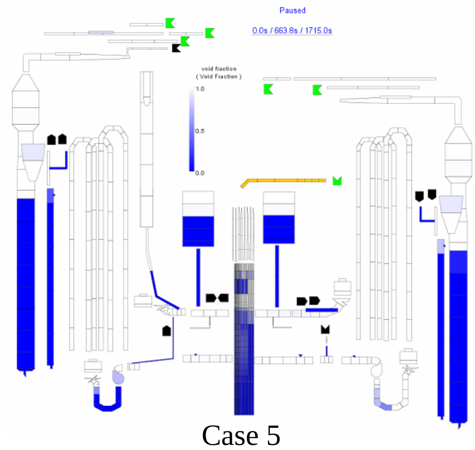
Case 2



Case 3



Case 4



Case 5

As can be seen, these results agree with other thermalhydraulic behaviors. At the highest break positions (Case 1 and Case 5), the change from liquid to two-phase and vapor flow is anticipated. This reduces the inventory discharged and the amount of liquid that remains in the vessel at any given time. On the contrary, at 650 s the liquid level in the vessel is lower in Cases 2, 3 and 4 and the core is already uncovered due to a greater loss of coolant.

Figures 8 and 9 show the maximum Peak Clad Temperature (PCT) and the Core Exit Temperature (CET), respectively. As it was expected, CET and PCT excursions for Case 2, Case 3 and 4 are started at 600 s approximately, reaching the AM action set point (623 K) before the other cases. In Case 5, the temperature excursions are produced at 850 s, while in the upper head SBLOCA (case 1) these excursions start at 1000 s.

The AM action used to mitigate the accidental consequences is not enough and the temperatures slightly drop but then, continue to rise in Case 2 and 3. In these cases, the core is not refilled and it could produce core damages. In Case 4, a second temperature excursion is produced around 1850 s, producing a new core emptying. These effects are observed in Figure 10, where the core collapsed liquid levels are shown. As can be seen, the core in Cases 2, 3 and 4 is not refilled. In these cases, the AIS is not enough and the HPI activation is necessary to avoid the core uncover.

Figure 8: *Maximum Peak Cladding temperature*

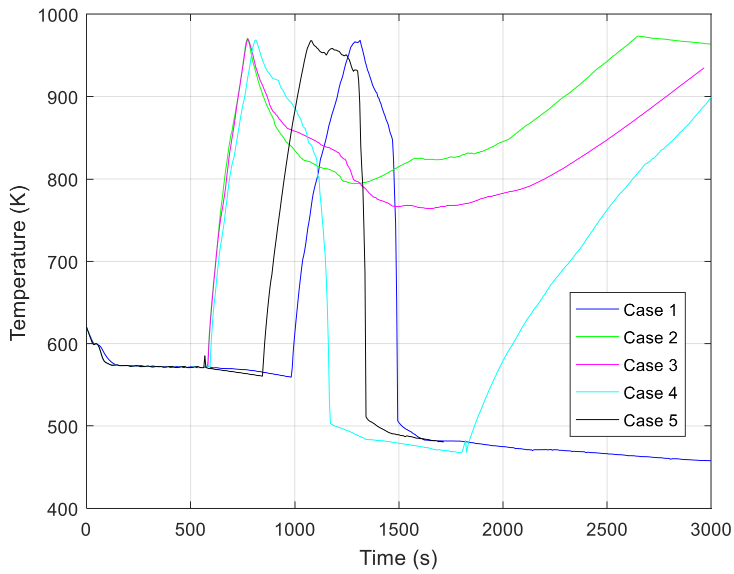


Figure 9: *Core Exit temperature*

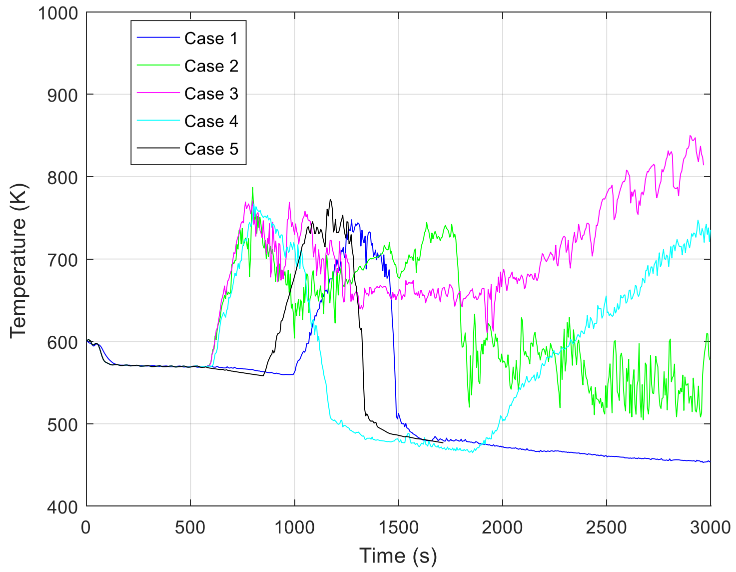
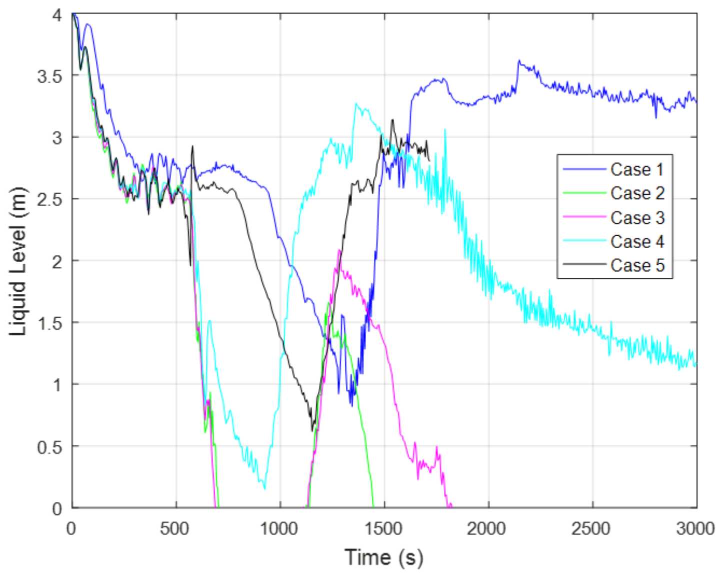


Figure 10: *Core collapsed liquid level*



4. CONCLUSION

Different SBLOCA tests have been simulated with an LSTF model using the thermal-hydraulic code TRACE5 and varying the break localization from the upper head to the lower plenum and the downcomer. Furthermore, in the downcomer three different elevations have been considered. The break size and the control logic of Test 6-1 in the frame of the OECD/NEA ROSA Project has been maintained in all the cases, assuming the total failure of the High Pressure Injection (HPI) system.

Results show that when the break is located in the lower plenum and the lower part of the downcomer, the temperatures do not decrease. The core is not refilled and it could produce core damages. However, when the break is located in the medium part of the downcomer, a second temperature excursion is produced. From the results, it can be stated that the AM actions defined in Test 6-1 are only effective for SBLOCA located in the upper part of the pressure vessel.

Regarding the break flow-quality transitions, results show that the time at which transitions occurred is greatly dependent on the break localization.

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