

TRACE CODE VALIDATION FOR DVI LINE BREAK LOCA IN THE ATLAS FACILITY

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Abstract

The object of this work is the validation and assessment of the TRACE code using the scaled test ATLAS¹ facility in the context of the ISP-50². The ISP-50 was proposed by the OECD/NEA/CSNI due to its technical importance to develop a best-estimate safety analysis methodology for the DVI³ line break accidents. In particular, the experiment selected for ISP-50 models the 50% 6-inch break of a DVI line.

Introduction

In some of the advanced nuclear reactors⁴ (Generation III and III+) the DVI is employed instead of the Cold Leg Injection (CLI) as accident mitigation strategy, such as in the Korean APR1400. Since the thermal-hydraulics phenomena that come into play are different, especially the two-phase flow in the upper annulus downcomer, relevant models need to be implemented into safety analysis codes in order to predict correctly these thermal-hydraulics phenomena. For this reason, and also to increase the knowledge of the phenomena in play, in 2005 the integral test ATLAS facility was built by the KAERI⁵. After several experimental campaigns on various types of rupture of the DVI line [1], in 2008 the experiment SB-DVI-07 was chosen by the OECD/NEA/OCSI as a reference test for the 50th International Standard Problem (ISP-50). The test is a 50% 6-inch DVI break line and it is used in this work for the validation and assessment of TRACE V5.0 Patch02 with the objective to establish its capability limits in the simulation of a DVI line break LOCA⁶.

Facility description

The ATLAS facility has 2 loops, 4 cold legs and 2 hot legs as the reference APR1400 reactor and it is designed according the scaling method suggested by Ishii and Kataoka [2] to simulate the various test scenarios as realistically as possible. In particular, it is a half-height and 1/288-

¹ Advanced Thermal-hydraulic Test Loop for Accident Simulation

² International Standard Problem

³ Direct Vessel Injection

⁴ AP600, AP1000 and VVER-1000

⁵ Korea Atomic Energy Research Institute

⁶ Loss Of Coolant Accident

volume scaled test facility and the time for the scaled model is 2 times faster than the prototypical time. The major scaling parameters are summarized in Table 1.

Table 1: Major scaling parameters of the ATLAS facility [3].

Parameters	Scaling law	ATLAS design
Length	λ_{00}	$1/2$
Diameter	λ_{00}	$1/12$
Area	λ_{00}^2	$1/144$
Volume	λ_{00}^3	$1/288$
Core ΔT	$\Delta \lambda_{00}$	1
Velocity	$\lambda_{00}^{-1/2}$	$1/2$
Time	$\lambda_{00}^{1/2}$	$1/2$
Power/Volume	$\lambda_{00}^{-1/2}$	2
Heat Flux	$\lambda_{00}^{-1/2}$	2
Core Power	$\lambda_{00}^{1/2} \lambda_{00}^{-1/2}$	$1/203.6$
Flow rate	$\lambda_{00}^{1/2} \lambda_{00}^{-1/2}$	$1/203.6$
Pressure drop	λ_{00}	$1/2$

According to the APR1400 geometry, the ATLAS fluid system consists of:

- a primary system
- a secondary system
- a Safety Injection System (SIS)
- a break simulating system
- a containment simulating system,
- an auxiliary system.

The primary system includes a Reactor Vessel (RV), two hot legs, four cold legs, a pressurizer, four Reactor Coolant Pumps (RCPs), and two SGs. The secondary system is simply a circulating loop-type. The steam generated at two SGs is condensed in a direct condenser tank and the condensed feed-water (FW) is again injected to the SGs. The SIS incorporates most of the safety injection features of the APR1400, among them the most important are the four Safety Injection Tanks (SITs) and a high pressure Safety Injection Pump (SIP). The break simulation system consists of a quick opening valve, a break nozzle, a case holding the break nozzle and instruments (Figure 1). It is manufactured to have a scaled break flow during the test. The containment simulating system has a function of collecting the break flow and maintaining a specified back-pressure in order to simulate the containment. The schematic of the ATLAS facility for the actual DVI line break tests is showed in Figure 2.

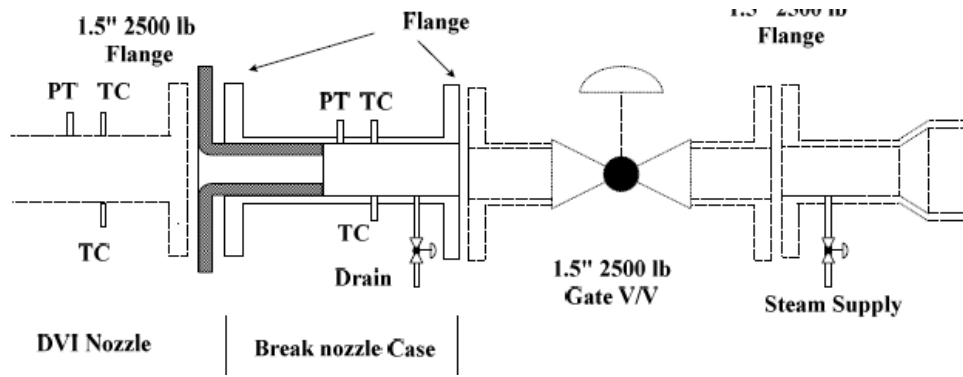


Figure 1: Configuration of the break simulation system for the DVI line break tests [3].

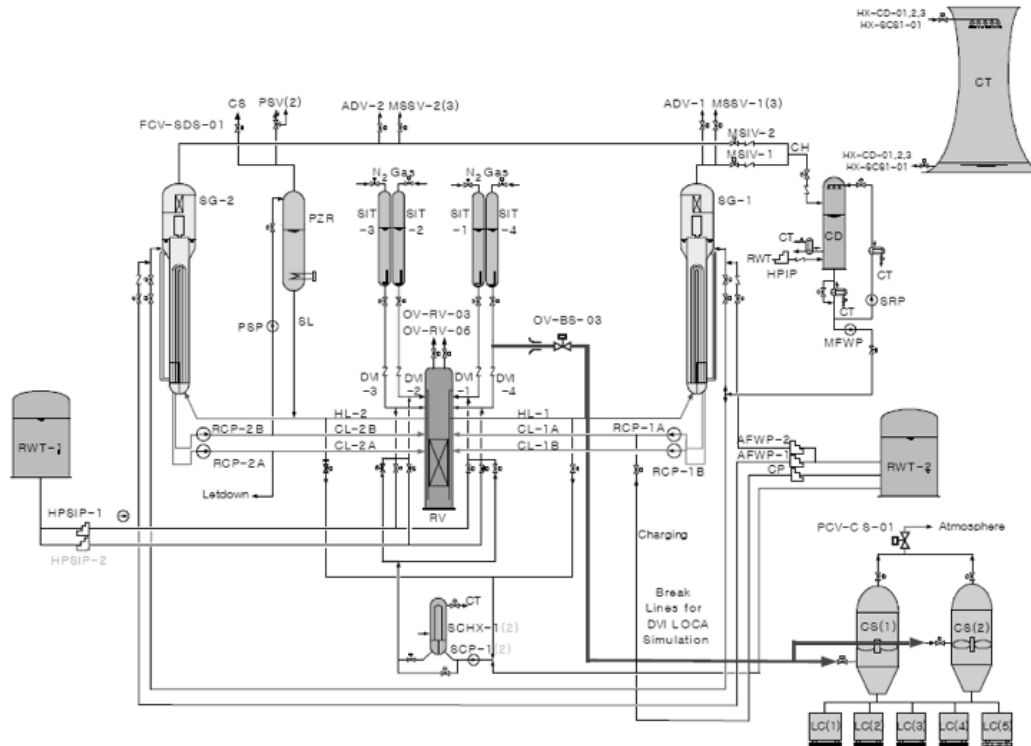


Figure 2 Schematics of the ATLAS for the DVI line break tests [3].

Experiment description

The ISP-50 test on a 50% DVI line break was performed in accordance with the test procedure described in the test specifications [3]. Based on the calculated sequence of events of the DVI line break for the APR1400 the initial and the boundary conditions for the present test were determined (Table 1).

Table 1: Comparison between reference reactor and ATLAS facility sequences.

Events	APR1400 (time, sec)	ATLAS (time, sec)	Description
Break open	0	0	
Low pressurizer pressure trip (LPP)	28.6=LPP		If pressurizer pressure < 10.72 MPa
Pressurizer heater trip	LPP+0.0 sec	LPP+0.0 sec	
Reactor scram & RCP trip	LPP+0.5 sec	LPP+0.35 sec	
Turbine isolation	LPP+0.1 sec	LPP+0.07 sec	
Main FW isolation	LPP+10 sec	LPP+7.07 sec	
Safety injection pump start	LPP+40 sec	LPP+28.28 sec	
Low upper down-comer pressure trip (LUDP)	LUDP	LUDP	If downcomer pressure < 4.03 MPa
Safety injection tank (SIT) start	LUDP+0.0 sec	LUDP+0.0 sec	
Low flow turndown of the SIT			If water level of the SIT is less than a specified set point

The pressure set points for actuation of major components were preserved as the same as the reference plant because the test was performed at the same pressure. Also, the temperature distribution along the primary loop was maintained the same as the reference plant. The secondary pressure of the ATLAS is slightly reduced to obtain a steady-state condition at 8% power level, because the ATLAS has a maximum power capacity of 8% of the scaled full power. The safety injection flow to the broken DVI-4 nozzle was not credited to obtain the minimum injection flow to the core. In addition, one train (two SIPs) was assumed to be disabled, so the safety injection flow by the SIP was provided only through the DVI-2 nozzle opposite to the broken DVI-4 nozzle. For the core power, a conservative 1973 ANS decay heat curve with a 1.2 multiplication factor was used in the transient calculation with a non-uniform cosine axial power distribution. In the DVI line break, the containment back-pressure does not affect the progression of this transient because a choking condition was maintained throughout the transient. Therefore, the containment back-pressure was not an important control parameter.

After a steady-state condition was achieved, the system was maintained for more than 10 minutes, and the transient test was commenced. First, data logging was initiated to log all measurement points in a steady-state condition. After the initial data logging was completed for about 200 s, the DVI line break test was initiated by opening a quick-opening break valve. When the pressurizer pressure reached a specified pressure of 10.72 MPa, the low pressurizer pressure (LPP) signal was automatically generated by embedded control logic. The heaters of the pressurizer and all tracing heaters in the primary system were tripped at the time of the LPP signal. The RCPs were automatically tripped with a time delay of 0.35 s after the LPP signal. The main steam and the main feed-water (FW) lines were isolated with a time delay of 0.1 s and 7.1 s after the LPP signal, respectively. Operation of the SIP was triggered by the LPP signal with a time delay of 28.3 s. When the down-comer pressure of the RPV became lower than the

specified pressure of 4.03 MPa, the SIT started to deliver the high safety injection flow to the RPV by fully opening the flow control valve. When the water level of the SIT reached a specified set point, the stem of the flow control valve was lowered to a specified position to supply a required low injection flow rate. When the water level of the SIT decreased to a specified empty set point, the flow control valve was fully closed for the nitrogen gas not to be injected into the RCS.

TRACE model and calculation

The post-test calculations have been performed using the NRC code TRACE V5.0 Patch02 on a current generation 3.00 GHz Intel CPU with Windows 7 operating system. Total CPU time for transient calculation is 25000 s, while the minimum time step used is 0.001 s.

The ATLAS facility model used for the calculation is shown on Figure 3 to Figure 6. The primary system is shown on Figure 3, the secondary system is shown on Figure 5. The break and ECCS nodalization is shown on Figure 4 and Figure 6, respectively.

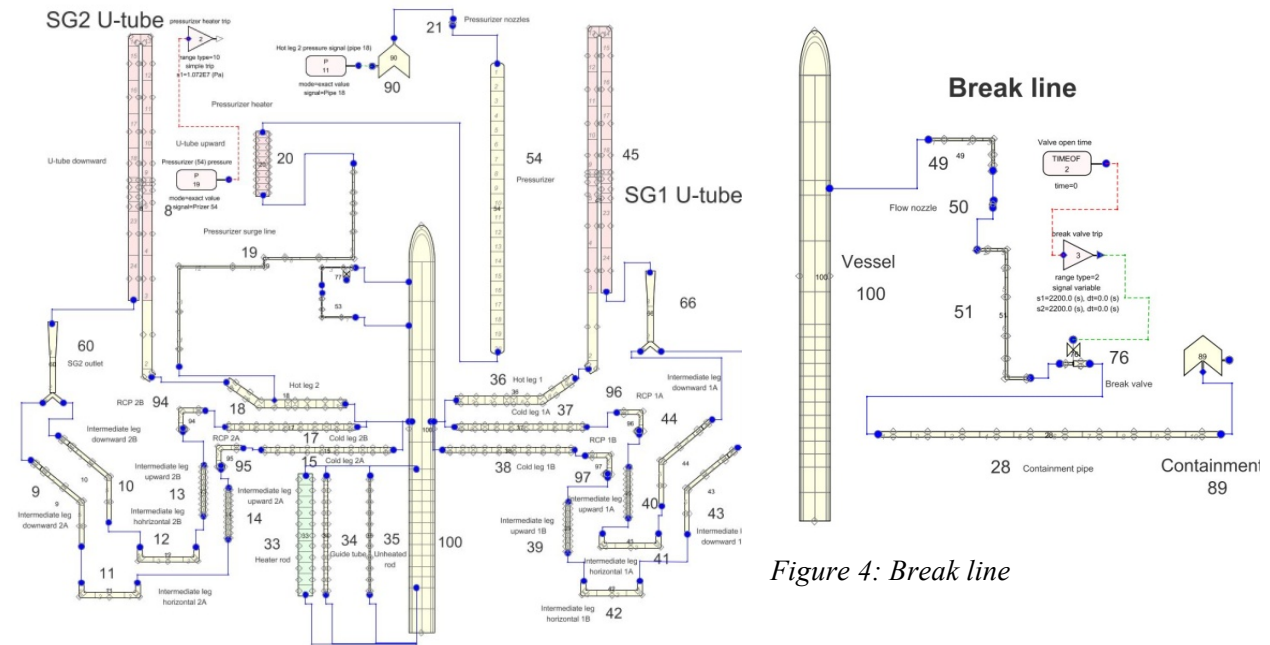


Figure 3: Primary system

Figure 4: Break line

The TRACE ATLAS model contains a total of 863 volumes (342 for the vessel, 521 for the remaining components) and 97 junctions (excluding internal junctions between volumes belonging to the same component). The number of heat structures is four: one heater (core) rod, two SGs exchange heat wall, and one pressurizer heater. The Reactor Pressure Vessel (RPV) is a 3D component with 19 axial cells, 3 radial cells, and 6 azimuthal cells, where the downcomer is

the external radial ring of the RPV. In all the model pipes the K-factors are set to 0.4 and 0.32 for every 45° and 15° elbows, respectively. In the RPV, k-factor is set to 10 in the flow area between Level 1 and Level 2, corresponding to the flow area between lower plenum and bottom of the core. In both the Steam Generator U-tubes (components 45 and 8) the CCFL model has been introduced.

Break line

The Break line has 20 volumes, while the break line is composed of 4 pipes and 1 valve (break valve). The Containment component is simulated imposing the pressure boundary conditions during the transient (BREAK component 89) (Figure 4). Abrupt area change and choking flow is activated for the break nozzle (component 50) inlet and outlet edge. Abrupt area change and choking flow is also activated for the edge facing the containment component (component 28). This pipe has also K-factor=10 distributed in every cell edge to obtain correct pressure drop along the break line (fitted to experimental data).

Emergency Core Cooling System (ECCS)

The ECC is composed of 165 volumes (Figure 6). The SIT tank is a pipe (component 22, 23, 24) composed of 12 cells. Every SIT line is composed of 2 pipes (component 71-46, 72-47, 73-48) with 22 volumes each. Abrupt area change and chock flow option are activated in the connection with the vessel component, in the bottom of the SIT and in the valve stem.

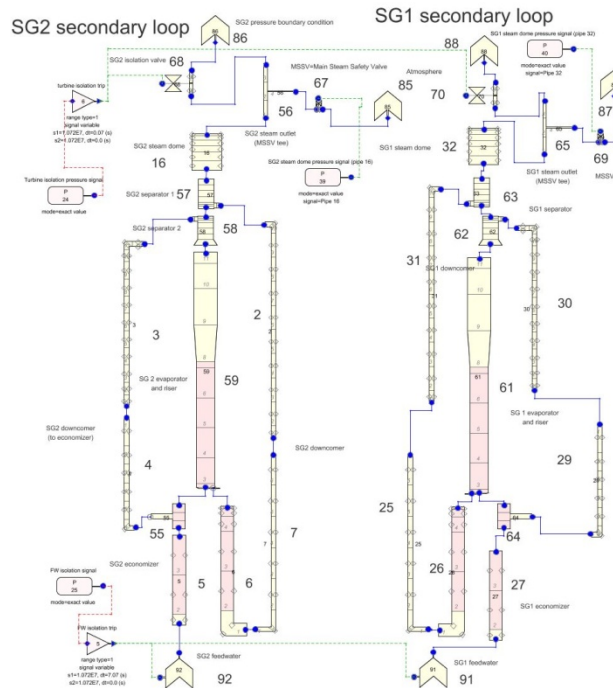


Figure 5: Secondary system

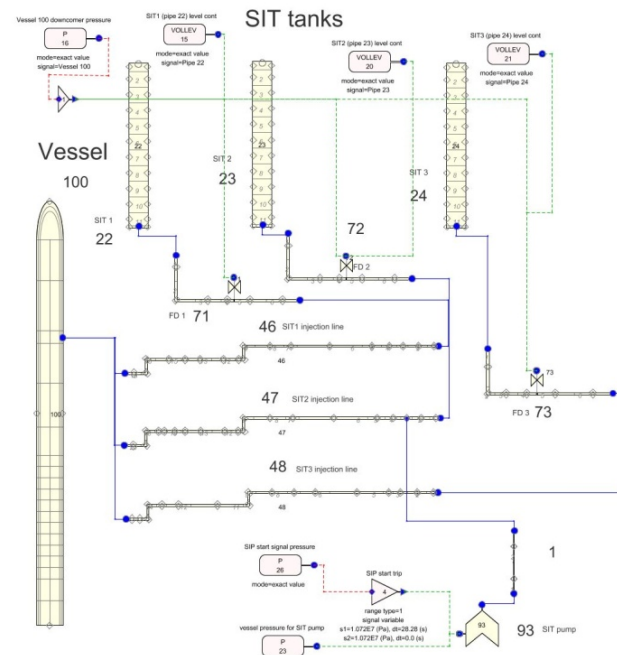


Figure 6: ECCS

Comparison of the results

Table 1 compares the main quantities initial conditions. The largest disagreements are for the flow rates, in the water levels and in the heat losses. The model flow rate in the primary system is lower in order to obtain correct fluid temperature and pressure, which were measured more precisely than the mass flow rates⁷. This explains also the differences in the RCP speeds. The higher FW flow rate entering the SG economizer is needed to simulate the heat losses in the primary and secondary circuit, which are not explicitly simulated in the model. This also explains the disagreements in the heat removal from each SG: it must be higher to maintain the temperature closer to the experimental value. The SG water levels are far off because SG model geometry had to be modified to obtain better model convergence.

The absolute percentage difference between experiment and calculation values was calculated using the following formula:

$$\frac{|\text{Calculation} - \text{Experiment}|}{\text{Calculation}} \times 100\%$$

⁷ Based on communication with ATLAS experimentalists The mass flow uncertainty reaches values of 10%

Table 1: Measured and calculated initial conditions.

Variables	Experiment	Calculation	Differences
Primary system			
Core power [W]	1.636E+06	1.636E+06	0.0%
Pressure [Pa]	1.56E+07	1.534E+07	1.7%
Core inlet temp [K]	563.2	562.9	0.1%
Core exit temp [K]	598.2	598.3	0.0%
CL flow rate [kg/s]	2.2	1.958	11.0%
Bypass flow [kg/s]	0	0	0.0%
PZR level [m]	3.32	3.23	2.6%
RCP speed [rpm]	18.7	21.5	14.9%
Secondary system			
Pressure [Pa]	7.83E+06	7.83E+06	0.0%
Steam temperature [K]	566.2	566.7	0.1%
FW temperature [K]	505.9	508.0	0.4%
FW flow (ECO) [kg/s]	0.431/0.435	0.473/0.473	9.7/8.7%
FW flow (DC) [kg/s]	0	0	0.0%
SG level [m]	2.03/1.97	3.44/3.43	69.5/74.1%
Heat removal per each SG [W]	7.80E+02	8.17E+02	4.8%
ECCS			
SIT pressure [Pa]	4.23E+06	4.21E+06	0.5%
SIT temperature [K]	323.15	323.42	0.1%
SIT level [%]	95	95	0.0%
RWT temperature [K]	323.15	323.15	0.0%
Containment			
Pressure [Pa]	1.01E+05	1.03E+05	1.6%

Timing of the main transient events is summarized in Table 2. All the events related to the safety injection system actuation occur later in the TRACE model than in the experiment until the SIP-2 injection starts. However, the SIT actuation occurs later in the model. This is essentially due to the difference in pressure trends in the pressurizer because it controls all the safety injection trips in the model and in the ATLAS facility.

This is easily understandable from Figure 7 where the main pressure trends in the facility are plotted. Until 500 seconds the pressure in the model is higher than in the experiment, while afterward the trend reverses. The difference in the pressure trend also affect the differences in the break accumulate mass (Figure 9) and the active SIP flow rate (Figure 10). In fact, because the pressure decreases faster in the TRACE model after 500 s, less water flows outside the primary system through the break and more water is injected by the SIP in the primary system. The SIP flow rate is inversely proportional to the pressure in the primary system because the pump is

centrifugal. As a direct consequence, there is a slightly disagreements in the primary total mass: in the TRACE model there is more water.

Table 2: Measured and calculated occurrences

Event	DAS ⁸ time (s)	TRACE time (s)	Remarks
Data logging starts	-203,0	-2200,0	
Break valve open	0,0	0	
MMSV of SG-2 open (1st opening)	24,0	41	PT-SGSD2-01>8.1 MPa
MMSV of SG-1 open (1st opening)	24,0	41	PT-SGSD1-01>8.1 MPa
Low pressurizer pressure trip (LLP)	25,0	38.08	PT-SGSD1-01<10.7214 MPa
Pressurizer heater OFF	25,0	38.08	LPP + 0.0 sec
Main steam isolation	25,0	38.08	LPP + 0.1 sec
RCPs trip	25,0	38.08	LPP + 0.35 sec
Main feed water isolation	32,0	44,09	LPP + 7.0 sec
Core power starts to decay	33,0	32,00	
SIP-2 injection	54,0	66.14	
MSSV of SG-2 open (2nd opening)	57,0	67	PT-SGSD2-01>8.1 MPa
MSSV of SG-1 open (2nd opening)	62,0	69	PT-SGSD1-01>8.1 MPa
MSSV of SG-2 open (3rd opening)	113,0	104	PT-SGSD2-01>8.1 MPa
MSSV of SG-1 open (3rd opening)	118,0	101	PT-SGSD1-01>8.1 MPa
1st loop seal clearing occurs	190,0	423	Only in loop 1A/1B
SIT actuation (high flow)	468,0	435,89	PT-DC-01<4.03 MPa
2nd loop seal clearing occurs	1236,0	-	Loop 2B
SIT low flow conversion	-	-	did not occur
Test stops	2933,0	3300	

The TRACE model is able to predict the experimental core heatup, but the value is higher and it also occurs later (dash line and blue line respectively in Figure 8). The delay in the heatup is probably due to the higher liquid level trend in the circuit, especially in the core (Figure 11). In the experiment the core liquid level remains around 2 meters leaving the core partially uncovered, while in the TRACE model the core remains covered except for a short period. However, the level trend in the downcomer is better modeled (Figure 12). There is only a little delay in the rise of the level in the TRACE model due to the delay in the SIT activation. The liquid level disagreements are again due to the differences in the primary total mass balance. The core heat up is higher in the model because the k-factors in the RPV bottom are too high. They should be decreased to match better the experimental results.

⁸ Data Acquisition System

The TRACE model is not able to properly simulate the loop seal clearing phenomenon and its effects, such as the increase in the core level and the corresponding decrease in the downcomer level when loop seal clearing occurs. This normally happens in the reactors using CLI safety features [4] and the same it is inferred to happen in this test too. While in the model the 1st loop seal clearing is perfectly noticeable looking at the collapsed water level trend in the core active region (Figure 11), the same does not happen for the 2nd loop seal clearing.

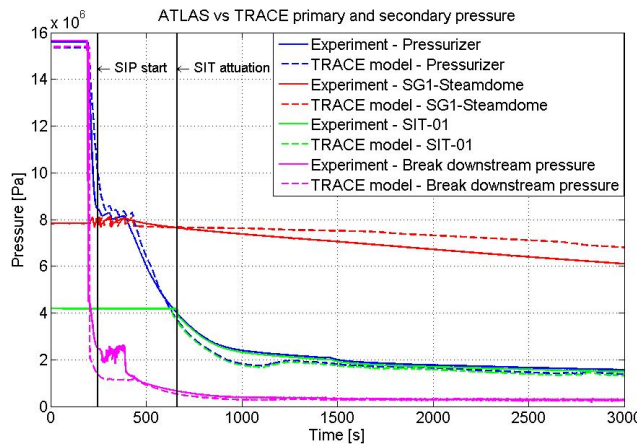


Figure 7: Main pressure trends in the primary and secondary circuit.

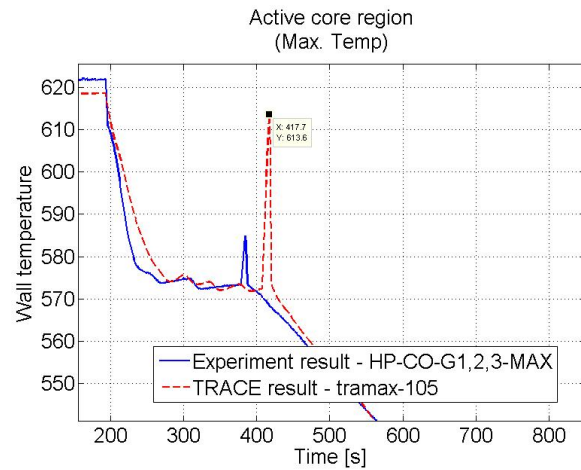


Figure 8: Enlargement of the Max active core region temperature.

Another evidence of the TRACE model deficiency in the loop seal simulation is showed in Figure 13. While in the experiment the 1st loop seal clearing occurs in loop 1 and only after some seconds it is followed by a 2nd loop seal clearing in the branch B of the loop 2, in the TRACE model the loop seal clearing occurs in all the loops at the same time (around 420 seconds). Furthermore, the level in the leg 1B rises again, followed by the leg 2A, which is the opposite of the measured trend.

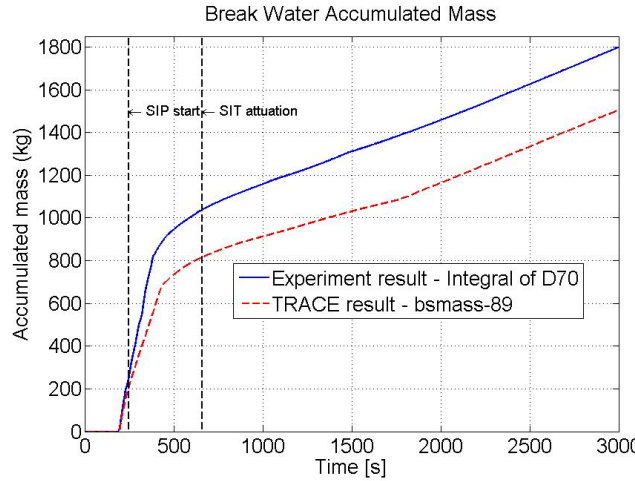


Figure 9: Break Water Accumulated Mass.

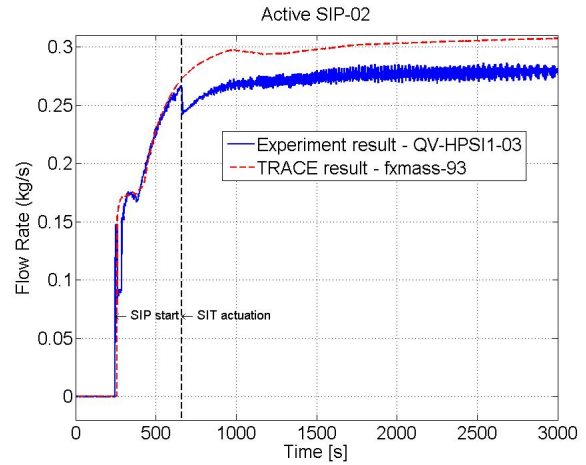


Figure 10: Accumulated break flow rate

The disagreement is probably due to the Counter Current Flow Limiting (CCFL) model, which is activated in the primary circuit but only in the Steam generator U-tubes, and to the mismatch in the primary system total mass balance. The CCFL is an important phenomenon every time there are liquid and vapor flowing in counter current in a vertical pipe.

The asymmetry in the CWL model trend (loop seal clearing occurs in all the legs at more or less the same instant, while in the experiment they occur before in both the legs 1, and then only in the leg 2A) could be imputable to some downcomer bypass flows present in the real experiment that are not well simulated in the TRACE model.

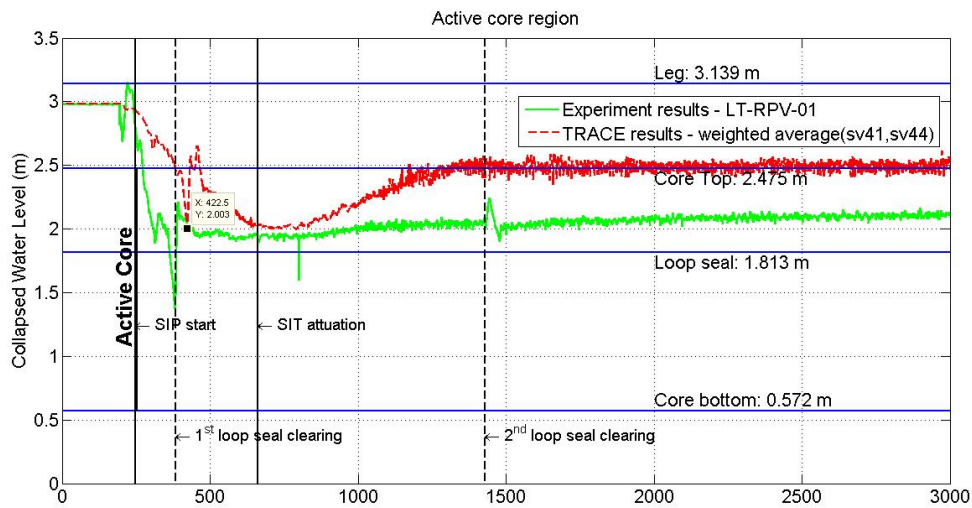


Figure 11: Measured vs. TRACE collapsed water level of core region

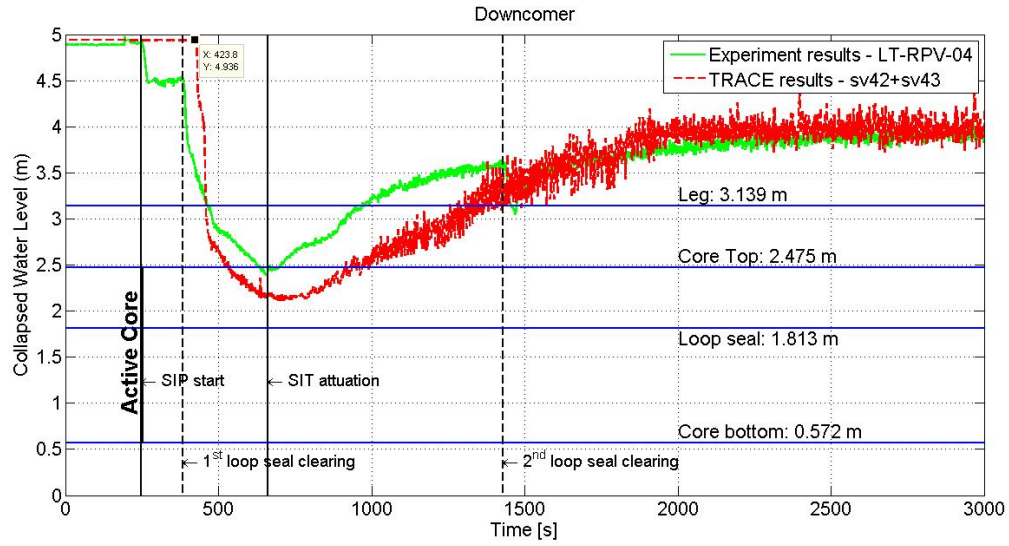


Figure 12: Measured vs. TRACE collapsed water level of the downcomer

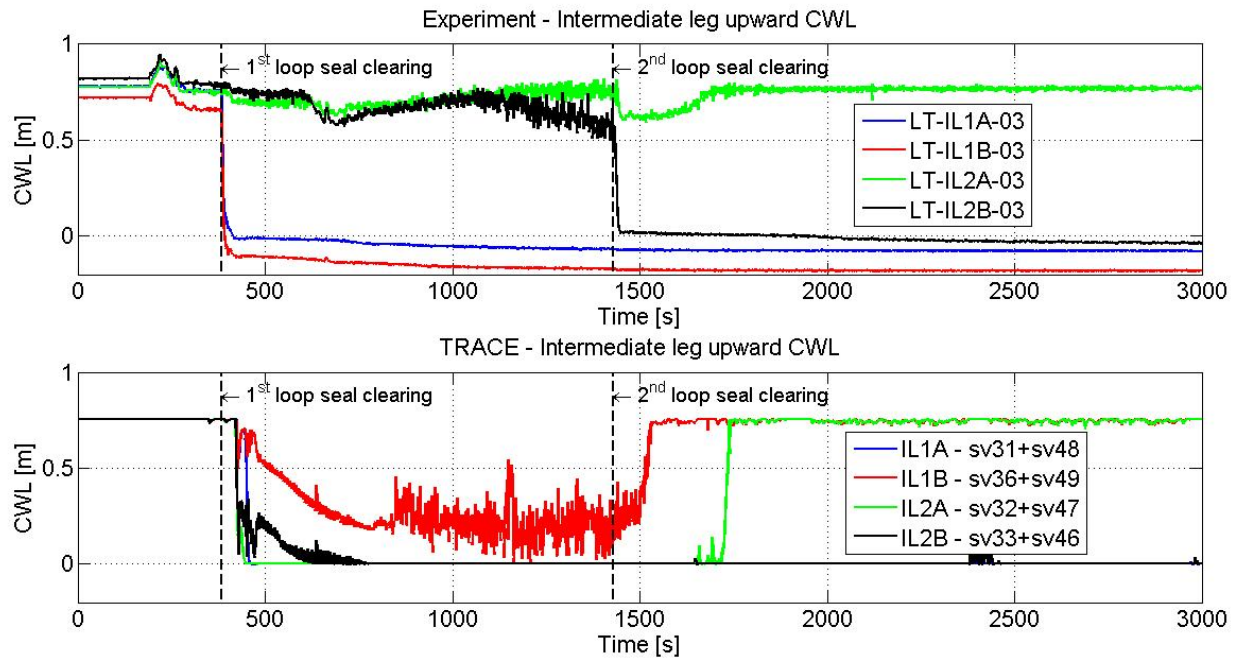


Figure 13: Intermediate leg upward CWL

Conclusions

The study of the 50% (6-inch) DVI break (experiment SB-DVI-07), reference test for the ISP-50, was used for the verification of the best-estimate thermo-hydraulic code TRACE.

Generally, the code is capable of calculating the main phenomena of the experiment, with the exception that the used model fails to simulate properly the 2nd loop seal clearing, while it is perfectly able to determine the core heatup. Even so, the calculated results show a good agreement with the measured data. It should be emphasized that the differences between the experiment and TRACE results in case of the loop seal clearing are not a code limitation. Rather, it is a modeling problem and further investigations are required to study such phenomena, especially in the direction of taking into account the CCFL phenomenon in all the primary circuit and not only in the SG U-tube components.

Acknowledgments

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References

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