LBLOCA AND DVI LINE BREAK TESTS WITH THE ATLAS INTEGRAL FACILITY

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This paper summarizes the tests performed in the ATLAS facility during its first two years of operation (2007~2008). Two categories of tests have been performed successfully: (a) the reflood phase of the large-break loss-of-coolant accidents in a cold leg, and (b) the breaks in one of four direct vessel injection lines. Those tests contributed to understanding the unique thermal-hydraulic behavior, resolving the safety-related concerns and providing an evaluation of the safety analysis codes and methodology for the advanced pressurized water reactor, APR1400. Several important and interesting phenomena have been observed during the tests. In most cases, the ATLAS shows reasonable accident characteristics and conservative results compared with those predicted by one-dimensional safety analysis codes. A wide variety of small-break LOCA tests will be performed in 2009.

KEYWORDS : ATLAS, APR1400, OPR1000, Direct Vessel Injection, Integral Effect Test

1. INTRODUCTION

The ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) is a new large-scale thermal-hydraulic integral effect test facility for advanced pressurized water reactors (PWRs), APR1400[1] and OPR1000[2]. It can simulate a wide variety of accident and transient conditions including large- and small-break loss-of-coolant accidents (LOCAs). The information on the ATLAS program, major design characteristics, scoping analyses, commissioning tests, and some test results can be found in the literature [3-6].

The ATLAS facility started its operation at the end of 2006 after extensive commissioning work. In 2007, the major subject of the ATLAS tests was the behavior of APR1400 during the reflood phase of large-break loss-of-coolant accidents (LBLOCAs). Those LBLOCA tests are directly related to safety issues in the licensing of the APR1400. In 2008, the tests focused on breaks in one of four direct vessel injection (DVI) lines of the safety injection system. These tests are important to understand the unique behavior of APR1400 and to assess the safety analysis codes against these new types of accidents for which available experimental data are very limited. One of these DVI line break scenarios has recently been selected as International Standard Problem (ISP) No. 50 of the Committee on the Safety of Nuclear Installations, OECD Nuclear Energy Agency (OECD/NEA)[7].

This paper summarizes the overall test programs and important findings of these two types of tests: (a) the reflood phase of LBLOCAs, and (b) the DVI line breaks.

2. THE ATLAS FACILITY

2.1 The ATLAS Program

The ATLAS program was started in 1997 as one of the mid- and long-term nuclear R&D projects funded by the Korean government. It has progressed as follows:

- 1997~2001: development of the basic technology for integral thermal-hydraulic tests and conceptual design of the test facility
- 2002~2006: Basic design, detailed design, construction, and commissioning of the ATLAS facility
- 2007~2008: APR1400-specific tests (LBLOCAs and DVI line breaks)

The ATLAS program has been the representative R&D project in Korea in the area of nuclear reactor safety. The expected roles of the ATLAS include [3]:

(a) tests for safety issues and/or interests related to the new reactor APR1400;
(b) simulation of the transients and accident conditions of APR1400 and OPR1000 for an assessment & validation of system analysis codes;
(c) tests in support of future industrial needs, including the development of advanced reactors, optimization

Table 1. Scaling Parameters of the ATLAS

Parameters	Scaling law	ATLAS design
Length	l_{0R}	1/2
Diameter	d_{0R}	1/12
Area	d_{0R}^2	1/144
Volume	$l_{0R}d_{0R}^2$	1/288
Core ΔT	ΔT_{0R}	1
Velocity	$l_{0R}^{1/2}$	$1/\sqrt{2}$
Time	$l_{0R}^{1/2}$	$1/\sqrt{2}$
Power/Volume	$l_{0R}^{-1/2}$	$\sqrt{2}$
Heat flux	$l_{0R}^{-1/2}$	$\sqrt{2}$
Core power	$l_{0R}^{1/2}d_{0R}^2$	1/203.6
Flow rate	$l_{0R}^{1/2}d_{0R}^2$	1/203.6
Pressure drop	l_{0R}	1/2

of operating and emergency response procedures, and resolution of safety issues for operating reactors (if needed);
(d) international cooperative programs for improvement of nuclear safety worldwide.

2.2 The Facility

The ATLAS facility has the following characteristics [3]:
(a) 1/2-height & length, 1/288-volume, and full-pressure simulation of APR1400,
(b) a geometrical similarity to APR1400 including 2(hot legs) ×4(cold legs) reactor coolant loops, a direct vessel injection (DVI) for emergency core cooling water, integrated annular downcomer, etc.,
(c) incorporation of specific design characteristics of OPR1000 such as cold leg injection and low-pressure safety injection pumps,
(d) maximum 10% of scaled nominal core power using the three-level scaling methodology of Ishii and Kataoka [8],
(e) detailed measurement of loop behavior with ~1,250 instrumentations.

Tables 1 and 2 summarize the major scaling parameters and actual design parameters of the ATLAS facility against the APR1400 design.

The ATLAS is mainly used to simulate various accident and transient scenarios for APR1400 and OPR1000, which are in operation or under construction in Korea. The ATLAS can simulate a broad range of scenarios including the reflood phase of LBLOCAs, SBLOCA scenarios including DVI line breaks, steam generator tube ruptures, main steam line breaks, feedwater line breaks, mid-loop operation, etc. The simulation capability of the ATLAS

Table 2. Major Design Parameters of the ATLAS

Parameters	APR1400(P)	ATLAS(M)	M/P
Reactor			
Core flow rate, kg/s	20361.6	99.98	1/204
Core bypass flow, kg/s	630	3.09	1/204
Total press. drop, kPa	399.9	199.95	1/2
Total coolant vol., m^3	158.9	0.5483	1/289.9
Total height, m	13.94	6.19	1/2.25
Downcomer gap, mm	254.1	26.2	1/9.70
Fuel rod dia., mm	9.5	9.5	1.0
No. of fuel rods	56,876	396	1/144
Main Piping			
Hot leg ID, m	1.068	0.1288	1/8.283
Hot leg vol., m^3	5.741	0.02423	1/237
Cold leg ID, m	0.762	0.0873	1/8.73
Cold leg volume, m^3	3.2	0.01111	1/288
Int. leg ID, m	0.762	0.0873; 0.0669	1/8.73; 1/11.4
Int. leg vol., m^3	3.806	0.0172	1/221.3
Pressurizer			
Volume, m^3	67.96	0.2192	1/288
Inner diameter, m	2.438	0.2032	1/12
Height, m	15.15	7.556	1/2
Surge line ID, m	0.2573	0.0306; 0.018	1/8.38; 1/14.3
RCP			
Rated head, m	109.73	30	1/3.7
Rated flow, m^3/hr	27617.54	70	1/394.5
SG			
Tube ID, m	0.017	0.012	1/1.42
Tube aver. length, m	19.96	9.46	1/2.11
Heat transfer area, m^2	9251.2	44.8	1/206.5
No. of tubes	12559	175	1/71.8

for a LBLOCA, a DVI line break, and a main steam line break was evaluated by the best-estimate system code MARS during the design phase [4]. A recent OECD/NEA document [9] recognizes the ATLAS as one of the major facilities for thermal-hydraulic safety research related to existing and advanced PWRs.

3. THE LBLOCA REFLOOD TESTS

3.1 Test Objectives and Matrix

During the licensing review for the APR1400's Standard Design Approval, some concerns were raised on the thermal-hydraulic safety performance of the emergency core cooling system during LBLOCAs:

- overall validity of the safety analysis code and analysis methodology for the reflood phase of LBLOCAs
- the adverse effect of downcomer boiling during the late reflood phase of large-break LOCAs

Therefore, the first series of tests was focused on providing reliable integral and separate effect test data for a resolution of the above issues.

The LBLOCA reflood tests were performed for the double-ended guillotine break in a cold leg, and consisted of three phases. Phase-I tests were parametric effect tests for downcomer boiling during the late reflood period of the LBLOCA, while Phase-II tests were integral effect tests for the thermal-hydraulic phenomena in the downcomer and the core during the entire reflood period to provide the peak cladding temperature data for an evaluation of the safety analysis codes and corresponding licensing methodologies. In addition, a separate effect test was performed to provide data to help validate the reflood models of the existing codes, such as the RELAP5/MOD3 for core quench phenomenon under low flow rate ECC injection condition (Park *et al.*, 2008).

Park *et al.*[10] provide an overview of the LBLOCA reflood test program in 2007; only typical results are presented in this paper.

3.2 Test Conditions

3.2.1 Phase-I Tests

The Phase-I tests were focused on simulating the late reflood phase when downcomer boiling would be important based on APR1400 analyses. The late reflood period is started when the safety injection tank (SIT) injection ends at around 208 seconds after the initiation of a LBLOCA, and the quasi-steady state condition of the late reflood period is assumed to start at 250s after the initiation of the LBLOCA. The safety injection pump (SIP) flow rate of the APR1400 was estimated at about 65.2 kg/s from the MARS code, and thus 0.32 kg/s of ECC water should be delivered to the RPV through the DVI lines in the ATLAS.

3.2.2 Phase-II Tests

The Phase-II tests were focused on simulating the

Table 3. Test Conditions for the Major Phase-II Tests

Test ID	LB-CL-09	LB-CL-11	LB-CL-14
Decay curve	1.2*ANS73	1.02*ANS79	1.02*ANS79
CS pres.	0.1 MPa	0.2 MPa	Varying
Bypass valves	Fully open	Adjusted for APR1400	Adjusted for APR1400
Power	Uniform	Radial distribution	Radial distribution

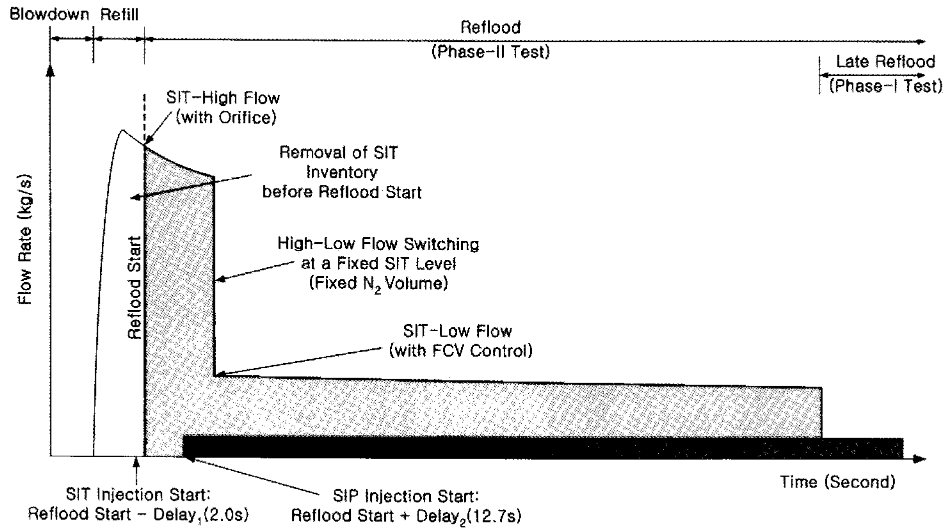


Fig. 1. Schematics of an ECC injection during a LBLOCA scenario for the APR1400

entire reflood phase starting at around 36 seconds after a reactor trip based on the APR1400 analyses. The effects of various parameters, including the decay power, the containment pressure, and the flow rate of the safety injection pumps were investigated. Major initial and boundary conditions for the Phase-II tests are summarized in Table 3.

According to the best-estimate evaluation of the LBLOCA scenario for APR1400, the reflood phase starts when the safety injection tanks (or accumulators) are still injecting a high flow of emergency core cooling water. The injection rate from the SITs was controlled to follow the characteristics of APR1400 as close as possible. Figure 1 illustrates the typical safety injection flow characteristics from the SITs and the safety injection pumps.

3.2.3 A Separate Effect Test

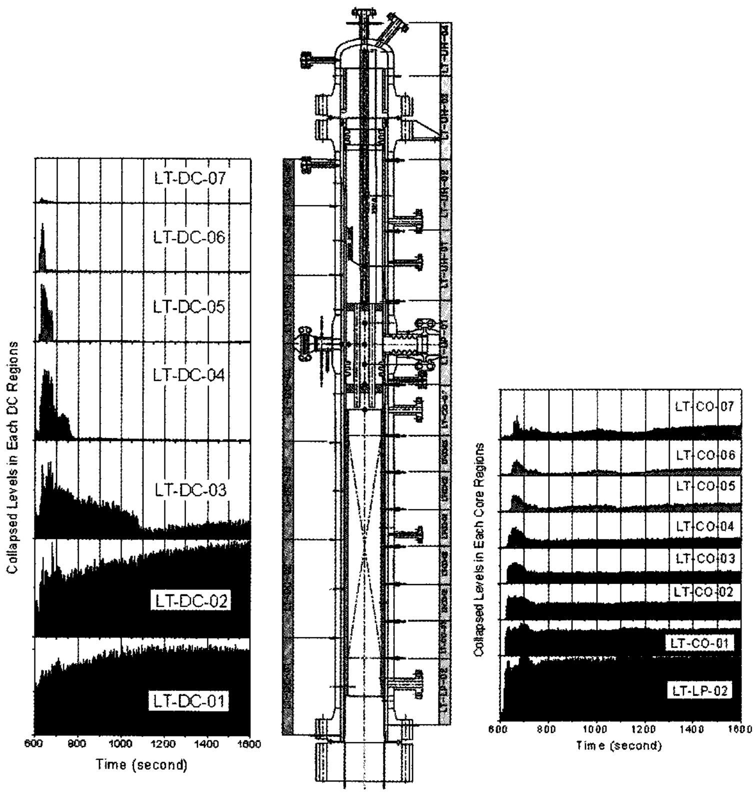
The experimental conditions for a separate effect test under a low reflooding rate condition were determined based on discussions with experts from the Korean industry. The present separate effect test could provide data peculiar to the APR1400, which includes the thermal-hydraulic phenomena of the direct vessel injection (DVI), the reverse heat transfer from the steam generator, and the steam binding effect throughout the primary loop. To reduce the pressure at the pump discharge line, the safety injection pumps supply ECC water through four SI lines, instead of two SI lines with the assumption of a single failure. The flow control valves were adjusted to provide a pre-determined flow rate of 0.3 kg/s each. The containment pressure was fixed at around 0.10 MPa and the initial outer wall temperature was determined to be 150°C in order to avoid downcomer boiling. For the heater power, 102% of the ANS-79 decay curve was adopted. The secondary pressure was set at around 5.0 MPa to consider the effects of reverse heat transfer and steam binding.

3.3 Test Results and Discussion

3.3.1 Phase-I Tests

Figure 2 illustrates the water distribution in the downcomer and core regions for the LB-CL-05 test. Even though the downcomer water level is maintained far below the cold leg level, the two-phase mixture level is maintained above the top of the core. A high void fraction is measured in the upper core region.

Figure 3 illustrates the variation of the maximum



[Downcomer] [Core]

Fig. 2. Sectional Void Fraction in Downcomer and Core

surface temperature of the core heater rods during the LB-CL-05 test. The heater rods were initially immersed in the nearly saturated water without any power and they were heated with a constant heat power of 715 kW during the test period. The experimental results illustrate the typical thermal-hydraulic trends expected to occur during the late reflood phase of the large-break LOCA. The maximum surface temperature of 309°C is much lower than that predicted by the MARS code (792°C).

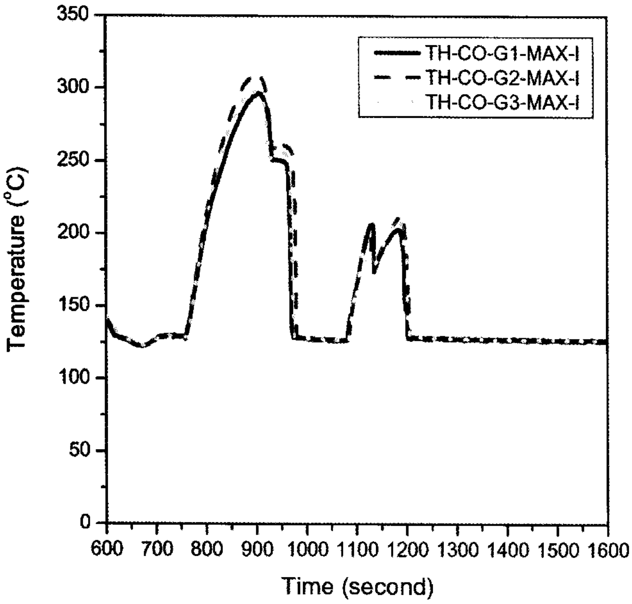


Fig. 3. Maximum Surface Temperature of the Heater Rods

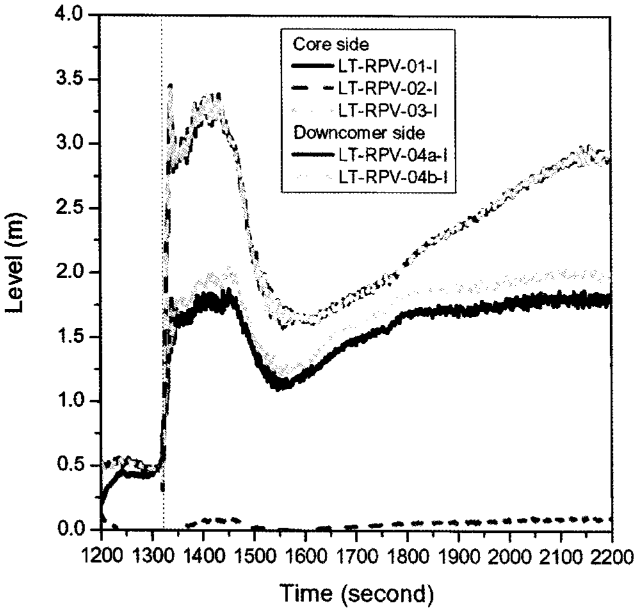


Fig. 4. Collapsed Water Levels for the LB-CL-14 Test

3.3.2 Phase-II Tests

Figure 4 illustrates the collapsed water level measured in the reactor core and in the downcomer for the LB-CL-14 test. It is found that the collapsed level in the downcomer is always higher than that in the core, which enables the supply of ECC water into the core.

Figure 5 illustrates the heater surface temperature distributions for the core heater group 1, which is located in the center region, for the LB-CL-14 test. A top quenching phenomenon was observed, which was more apparent in the center region (group 1) than in the outer region (group 3).

Table 4 presents the important test results for the major

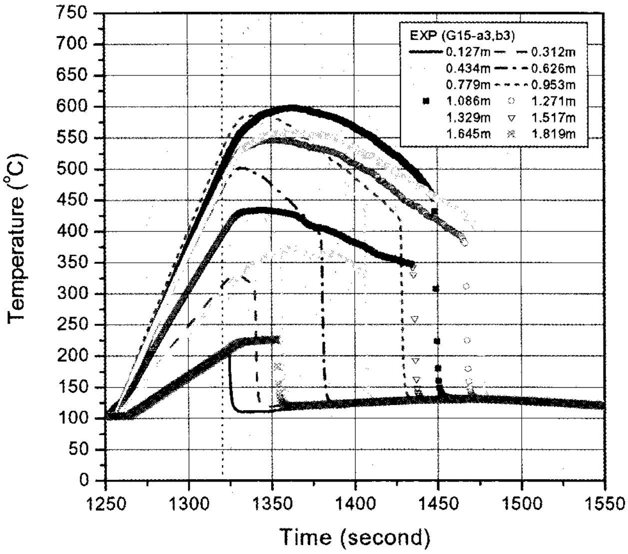


Fig. 5. Group-1 Heater Surface Temperature (LB-CL-14)

Table 4. Test Results for the Major Phase-II Scenarios

Test ID	LB-CL-09	LB-CL-11	LB-CL-14
Init. Power (kW)	1065	830.9	801.4
Power distribution	Uniform	Uniform	Radial
Avg. DC pres. (MPa)	0.15	0.21	0.21~0.11
Init. rod temp. (°C)	465	459	546
Max. rod temp. (°C)	722	558	615
Rewetting time (s)	654	218	171
Rewetting vel. (cm/s)	0.30	0.87	1.11

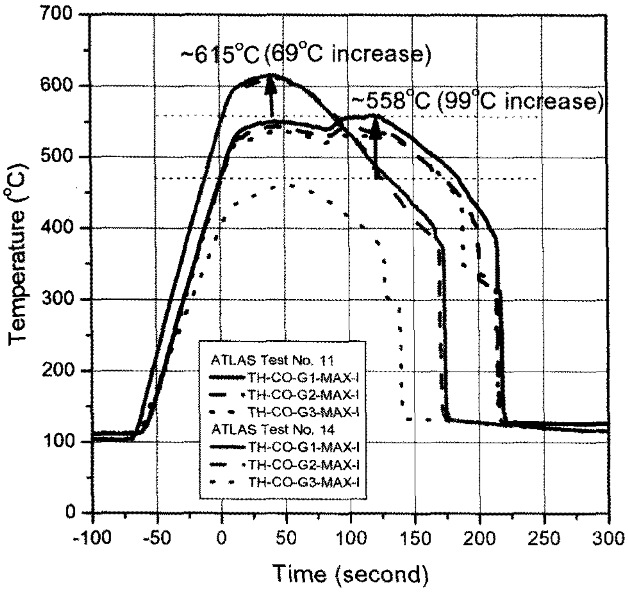


Fig. 6. Comparison of the Maximum Rod Surface Temperature between LB-CL-11 and LB-CL-14

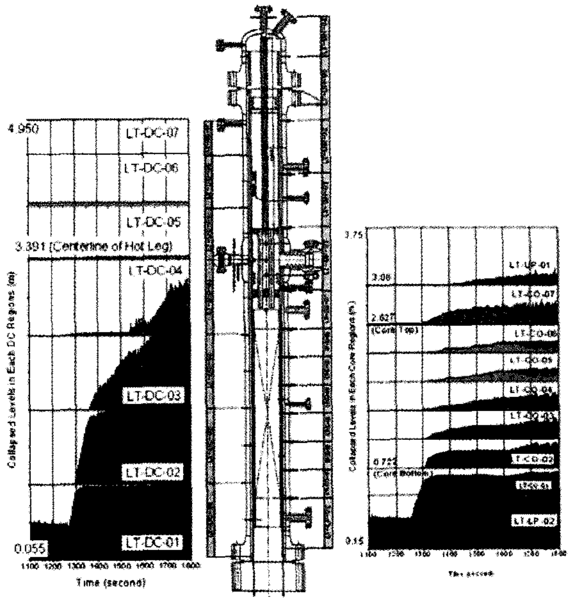


Fig. 7. Sectional Liquid Fraction in the Downcomer and Core

Phase-II test scenarios of Table 3. The peak cladding temperature is higher and the rewetting time is longer, as the decay power is higher and the containment back-pressure is lower. Figure 6 compares the maximum heater rod surface temperatures between the LB-CL-11 and LB-CL-14 tests. The LB-CL-11 test results are more conservative than those of the LB-CL-14 test, which considers a radial power distribution and a varying containment pressure more realistically.

3.3.3 A Separate Effect Test

Figure 7 illustrates the liquid fraction variations in the core and downcomer regions. After the test started with the initiation of the core power and SIP flow, the sectional liquid fractions in both the core and downcomer regions increased gradually. The cooling capability of the safety injection water overcomes the power given from the core heater and the stored energy from the reactor pressure vessel.

Figure 8 illustrates the maximum surface temperature variations of the core heater rods, showing a PCT of about 584°C. The entire rewetting process was finished by about 306 seconds after the reflood start time.

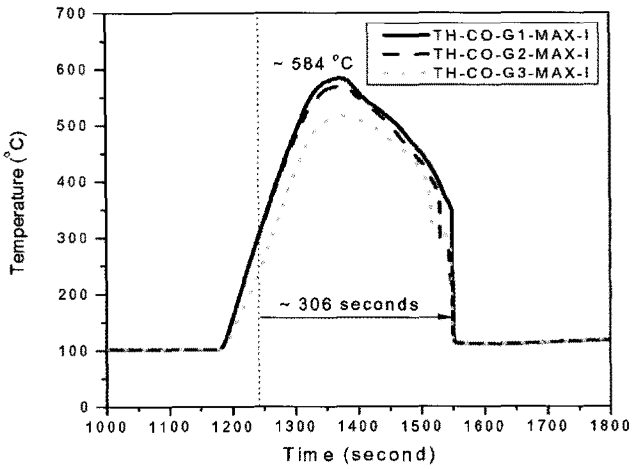


Fig. 8. Maximum Rod Surface Temperatures

4. THE DVI LINE BREAK TESTS

4.1 Importance of the DVI Line Break Tests

In APR1400, borated safety injection water is directly injected into the reactor vessel downcomer through DVI nozzles (8.5 in. ID), which are located 2.1 m above the cold leg, rather than through the cold legs as in most conventional PWRs. This new design feature has brought about a newly postulated accident, i.e., a break in a DVI line.

The postulated DVI line breaks are important accident scenarios for APR1400 from the viewpoint of its safety and operation; therefore, such accidents should be taken into account as SBLOCAs to ensure the safety of the

Table 5. Test Matrix for DVI Line Break Tests

Test ID	Break size		Test date	Remarks
	Nozzle ID (inch)	Break area ercentage (%)		
SB-DVI-03	8.5	100%	May 2008	Open to Korean org.
SB-DVI-04	4.25	25%	June 2008	Open to Korean org.
SB-DVI-05	1.9	5%	July 2008	Open to Korean org.
SB-DVI-06	6.0	50%	Aug. 2008	Hidden for an ISP exercise

APR1400. In the event of a DVI line break, the vapor generated in the core is introduced to the RPV downcomer through the hot legs, the steam generators, and the cold legs. Then, the vapor should pass through the upper part of the RPV downcomer to be discharged through the broken DVI nozzle. Therefore, the behavior of the two-phase flow in the upper downcomer is expected to be complicated, and relevant models need to be incorporated into the safety analysis codes in order to predict these thermal hydraulic phenomena correctly [11,12].

Recently, the SNUF (Seoul National University Facility) was utilized to simulate a DVI line break accident [13]. Researchers observed that the downcomer seal clearing phenomena had a dominant role in the decrease of the system pressure and increase of the coolant level of the core. However, the SNUF is a reduced-height and reduced-pressure facility that is operated at a pressure lower than 0.8MPa. It has limited simulation capability due to its low core power and the fact that the secondary system is simplified as a lumped boundary condition.

Unfortunately, there is a lack of publicly available integral effect test data for the DVI line breaks to be used for code assessment and improvement. The DVI line break tests with the ATLAS are expected to provide an important data set for international society.

4.2 Test Conditions

A series of sensitivity tests on DVI line break sizes are being carried out with the ATLAS to improve our poor knowledge level on the major thermal hydraulic phenomena related to DVI line breaks, as identified in the PIRT. So far, tests for four break sizes, 100%, 50%, 25%, and 5% of the cross section for a DVI nozzle have already been performed. A summary of the DVI line break tests performed so far is presented in Table 8. Among the tests defined in Table 6, the SB-DVI-06 corresponds to a critical case with a maximum break size of six inches where a core uncovery is not allowed according to the recommendation in the EPRI URD [14]. Figure 9 illustrates the location of the break and the safety injections (SITs and SIP) assumed for the DVI line break tests based on the single failure criterion.

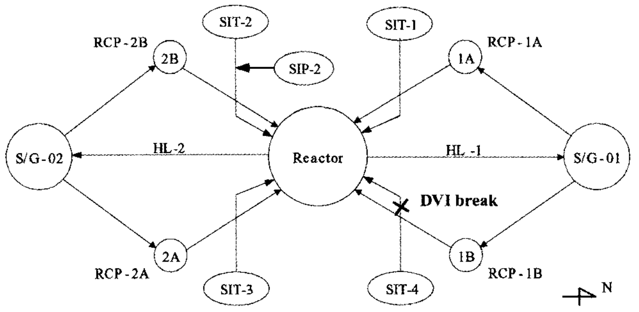


Fig. 9. Location of the Break and Safety Injection for DVI Line Break Tests

4.2 Test Results and Discussion

4.2.1 Pressure Trends

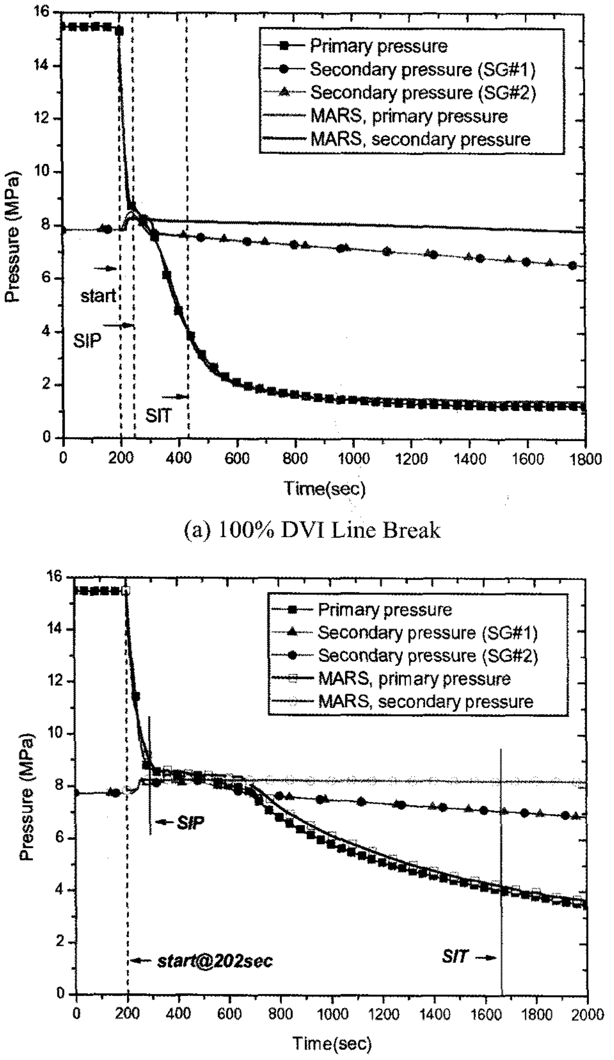
Figure 10 illustrates the variations of the primary and the secondary pressure during the DVI line breaks. On initiation of a break, the primary pressure decreases due to an abrupt loss of the inventory through the broken DVI-4 nozzle. The pressure decrease is similar to that observed in the cold leg small break LOCAs. The decreasing rate of the primary pressure is much faster for the 100% break, as expected. The trend of the primary pressure reveals a notable change when the SIP begins to operate; in particular, for the 25% break, it remains at a plateau region for about 400 seconds before starting to decrease again.

The MARS code was used to simulate the test conditions; the predicted trends are also plotted in the same figures. The predicted primary pressures are very consistent with the data but the secondary pressure is over-predicted. The reason for the inconsistency seems to be the heat loss of the SG to the environment or the distorted heat transfer characteristics of the U-tubes.

4.2.2 Peak Cladding Temperature

Figure 11 compares the measured maximum cladding temperatures with the ones predicted by the MARS code.

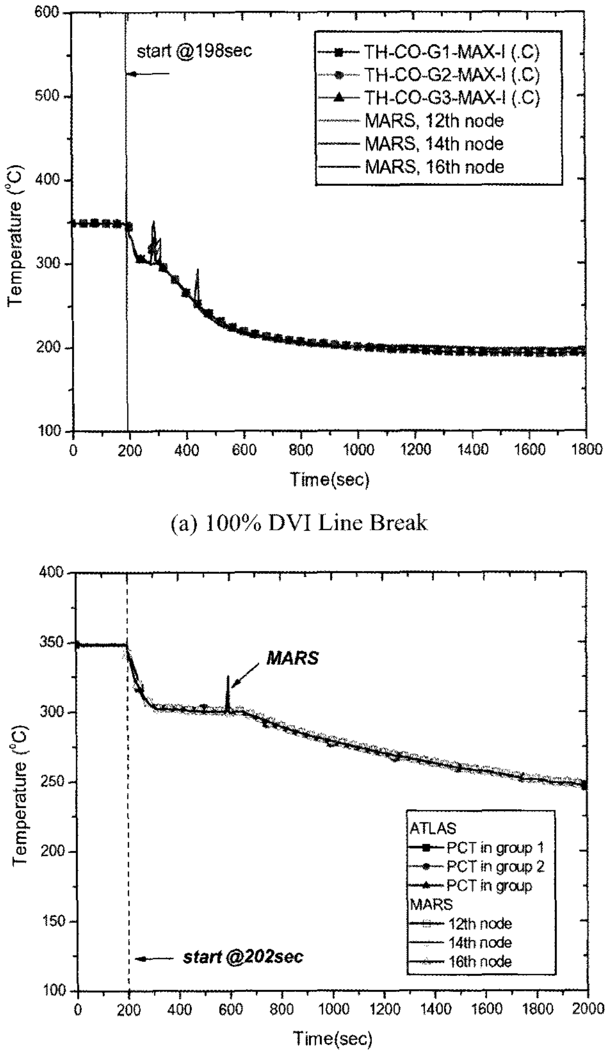
For the 100% break, a core uncovery and a temperature rise were observed in both tests and analysis. A PCT of



(a) 100% DVI Line Break

(b) 25% DVI Line Break

Fig. 10. Primary and Secondary Pressures



(a) 100% DVI Line Break

(b) 25% DVI Line Break

Fig. 11. Maximum Cladding Temperatures

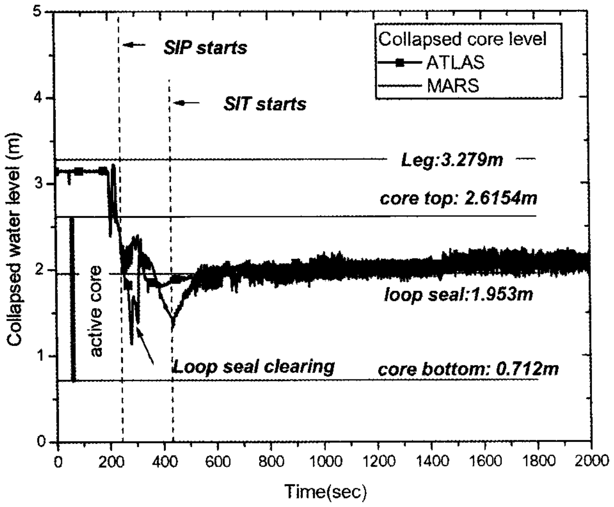
350°C was observed at around 300 seconds, while the MARS code predicted 293°C at around 400 seconds. The discrepancy is attributed to the fact that the MARS code predicts a higher core water level than the data before a loop seal clearing occurs. A more detailed description of the core level variation will be presented in the next section.

For the 25% break, a core uncovery was not observed in the test, while the MARS code predicted an uncovery with a PCT of 326°C at around 600 seconds. This discrepancy is attributed to the fact that the MARS code predicts a greater decrease in core water level than the data does before a loop seal clearing occurs. This is one of the findings about the thermal hydraulic phenomena near the loop seal clearing that were not properly predicted by the MARS code. This may be due to either an improper nodalization in the annulus downcomer region or a model deficiency related to the two-phase hydraulics in the upper annular downcomer of the reactor vessel.

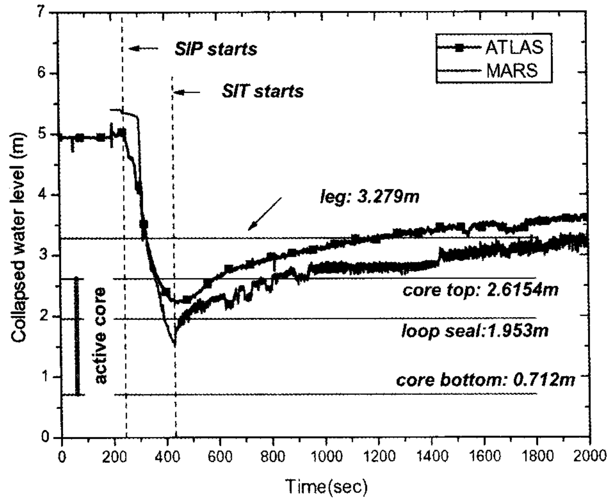
4.2.3 Collapsed Water Level in Core and Downcomer

Figure 12 compares the measured and predicted collapsed water levels in the reactor core and downcomer. The trends are quite different between the 100% break and the 25% break. The SIT injection plays a key role in recovering the water level for the 100% break. On the other hand, the SIP injection is sufficient to recover the water level for the 25% break. The MARS prediction reveals a similar trend in general.

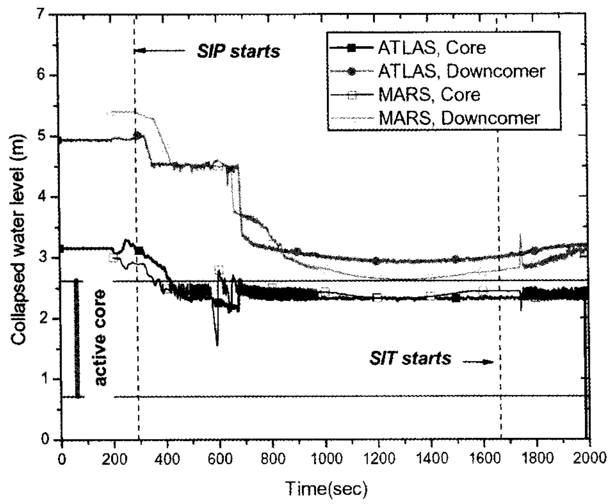
The core and downcomer water levels are related to the behavior of the loop seals. This should be investigated further for a better understanding of the phenomena.



(a) Core Level at 100% Break

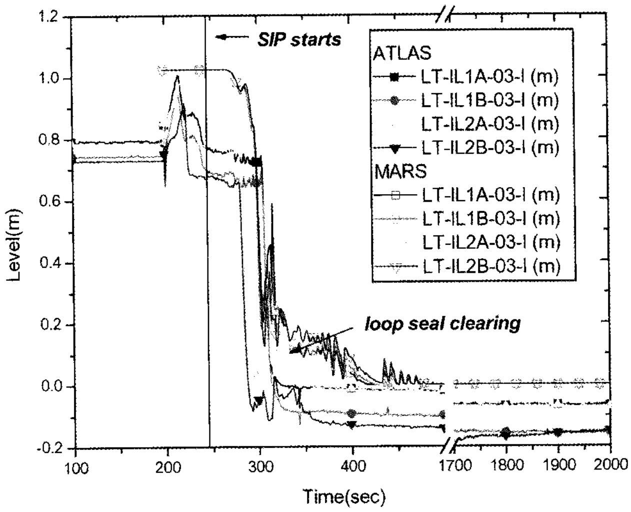


(b) Downcomer Level at 100% Break

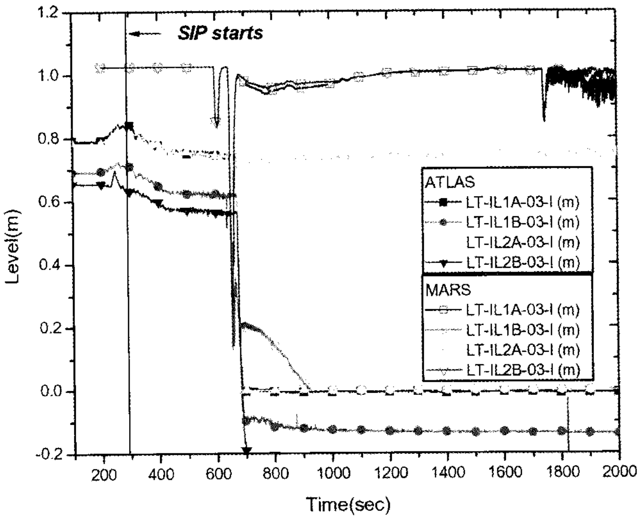


(c) Core and Downcomer Level at 25% Break

Fig. 12. Collapsed Water Levels in the Core and Downcomer



(a) 100% DVI Line Break



(b) 25% DVI Line Break

Fig. 13. Collapsed Water Level in the Loop Seals (Vertical Section)

4.2.4 Loop Seal Clearing

The blockage of the primary coolant loop with water filled at the intermediate legs, denoted as "loop seal," is expected to have a significant influence on the core cooling phenomena during a DVI line break accident. The core cooling capability is considered to be degraded during the loop seal period because the core water level is considerably decreased. When the loop seal is cleared, the core water level recovers rapidly and the primary pressure falls below that of the secondary side pressure. Consequently, the direction of the heat transfer is reversed and the steam generator begins to supply heat to the primary side.

Figure 13 illustrates the variation of the collapsed water level of the vertical intermediate legs. The loop seal was cleared at around 300 seconds. The prediction by the MARS code is also very consistent with the present data. It can be noted that the formation and clearing of the loop seals are repeated for the 25% break.

5. CONCLUSION

The ATLAS facility has been operated successfully according to the original test plan since the end of 2006. The LBLOCA and DVI line break tests have provided unique and valuable data bases for the APR1400; data are very useful for resolving safety concerns, understanding plant behavior, and assessing the safety analysis codes. Some of the data have been already transferred to the industry and regulators for an application to the safety assessment of the APR1400.

An extensive test plan for the next few years has already been established. In addition, an OECD ISP exercise has started in early 2009.

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