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CONFIRMATORY TESTING INVESTIGATING ADVANCED PASSIVE PLANT THERMAL-HYDRAULICS

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ABSTRACT

Confirmatory integral tests sponsored by the US Nuclear Regulatory Commission (NRC) were conducted at APEX-AP1000, a Westinghouse AP1000 scaled test facility at Oregon State University, to study AP1000 passive safety system performance under simulated beyond design bases accidents. Of particular interest during these tests was the effect of multiple failures of the AP1000 Automatic Depressurization System (ADS) on safety system performance. Experimental results show that failure of 2 out of 4 ADS4 valves on the non-pressurizer side of the plant following a LOCA leads to core uncovery (based on the two-phase mixture level) in APEX-AP1000, since in-containment refueling water storage tank (IRWST) injection was significantly delayed. In contrast, failure of 2 out of 4 ADS4 valves on the pressurizer side of the plant during a LOCA did not lead to core uncovery following. Examination of ADS4 line flow qualities during these tests shows significant liquid entrainment and carry-over from the reactor vessel. Beyond design bases test NRC-AP1000-05 and Department of Energy sponsored design bases test DBA-02, both double-ended direct vessel injection (DEDVI) line break scenarios, were compared. The resulting comparison illustrated the sensitivity of the core hydraulics to ADS-4 valve performance. However, it should be noted that beyond design bases scenarios are highly unlikely due to the low probability of multiple, simultaneous valve failures. In conclusion, the results of these confirmatory tests supported the analysis of accident evaluations for AP1000 design certification. Additionally, review of test results and code calculations submitted by Westinghouse conclusively showed that no core uncovery or heat up would occur for design basis scenarios. The AP1000 design certification process is currently under rulemaking at the NRC and a final ruling is expected by December 2005.

1. INTRODUCTION

The Advanced Plant Experiment (APEX) is a unique thermal-hydraulic integral system test facility used to assess the performance of passive safety systems for the Westinghouse AP600 and AP1000 designs. Data from the APEX facility was used extensively as part of AP600 Design Certification, providing approximately 75 tests for use in code assessment and qualification of AP600 safety margin. To address performance specific to the AP1000 design, the APEX facility underwent significant modifications in 2002. Included in the facility modification were an increase in the maximum core power, a new pressurizer (PZR) and surge line, larger core makeup tanks (CMTs), larger diameter fourth stage automatic depressurization (ADS4) system piping, and decreased line resistances for the CMTs, ADS4, and Passive Residual Heat Removal (PRHR) heat exchanger.

Testing in the “APEX-AP1000” facility began in 2003, with several integral experiments sponsored by the U. S. Department of Energy (DOE) to investigate performance of AP1000 passive safety systems at design basis accident conditions. The NRC Office of Nuclear Regulatory Research (RES) also conducted several integral tests to

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explore beyond design basis performance and to provide confirmatory information on thermal-hydraulic processes to benchmark thermal-hydraulics codes. In particular, liquid entrainment in the hot leg and upper plenum was of significant interest because of the higher core power in AP1000 than in the AP600.

Table 1.1 lists the APEX-AP1000 confirmatory tests sponsored by the NRC. Most of the tests investigated Double-Ended DVI (DEDVI) line or cold leg breaks with multiple Automatic Depressurization System (ADS) valve failures. The US DOE sponsored numerous design bases tests; one of which (DBA-02) is used as base case in this paper for comparison with beyond design bases results for a DEDVI line break. For this paper, five NRC tests (NRC-AP1000-03, 05, 06, 11, and 02) are examined in detail.

2. EXPERIMENTAL METHODS

2.1 Scaling

The APEX-AP1000 scaling analysis by Reyes (2003) provided the basis for the following APEX test facility modifications: core decay power, the CMT and PZR volumes, the ADS4, IRWST, PRHR, and CMT line resistances, the in-containment refueling water storage tank (IRWST) and containment flood-up elevations, the ADS4 flow area, the PZR surge line diameter, the upper core plate and upper support plate flow areas and the upper plenum structures. The majority of the APEX-AP1000 scaling analysis was based on results from the original AP600 scaling analysis since the AP1000 is geometrically similar to the AP600 and the thermal-hydraulic phenomena studied fell within the purview of the original AP600 test program.

The following bullets summarize the results of the APEX-AP1000 scaling analysis:

- In general, tank volumes, flow areas, and line resistance scaling ratios remained the same from AP600 to AP1000;
- The PZR surge line was modified based on study by DiMarzo and Bessette (1999) to produce a better simulation of PZR draining;
- The Reactor Coolant System (RCS) depressurization scaling analysis found that the characteristic time ratios for AP1000 would be well matched in APEX-AP1000;
- The AP1000 core axial void fraction profile and the core averaged void fraction were shown to be preserved in APEX-AP1000;
- It was shown that the APEX-AP1000 test facility conservatively simulates upper plenum entrainment behavior when the two-phase mixture level in the reactor vessel is below the midpoint of the upper plenum.

Table 1.1 NRC APEX-AP1000 Confirmatory Test Matrix (Beyond Design Basis)

Test Number	Failure(s)
NRC-AP1000-01	Double-Ended DVI (DEDVI) line break with failure of ADS1, 2, and 3.
NRC-AP1000-02	Mode 5 operation with loss of RNS cooling: No break; ADS1, 2, and 3 valves open; CMTs and ACCs unavailable; 3 out of 4 ADS4 valves unavailable.
NRC-AP1000-03	DEDVI line break with failure of 2 out of 4 ADS4 valves on the PZR side.
NRC-AP1000-04	Bottom of cold leg #4, 5.08 cm break with failure of 1 out of 4 ADS4 valves on non-PZR side. Containment sump degraded.
NRC-AP1000-05	DEDVI line break with failure of 2 out of 4 ADS4 valves on non-PZR side.
NRC-AP1000-06	Bottom of cold leg #4, 2.54 cm break with failure of 2 out of 4 ADS4 valves on non-PZR side.
NRC-AP1000-11	Station Blackout: No break; fail 1 out of 4 ADS4 valves on non-PZR side.

2.2 Facility Description

The Oregon State University (OSU) Department of Nuclear Engineering has modified its Advanced Plant Experiment (APEX) for assessing the AP1000 as shown in Figure 2.1 and described in detail by Abel (2003). APEX is a unique, world-class, thermal-hydraulic integral system test facility. The test facility is a one-fourth height, one-half time scale, reduced pressure integral systems facility. Three main system have been included in the facility:

- *Reactor Coolant System*. This includes an electrically heated 48-rod bundle core, a reactor vessel with internals, two hot legs, four cold legs, two 133 U-tube steam generators (SGs), a pressurizer (PZR), and four reactor coolant pumps.
- *Passive Safety Systems.* This includes two core makeup tanks (CMTs), two accumulators (ACCs), a four-stage ADS, a passive residual heat removal (PRHR) heat exchanger (HX), an in-containment refueling water storage tank (IRWST), and portions of the lower containment compartments.

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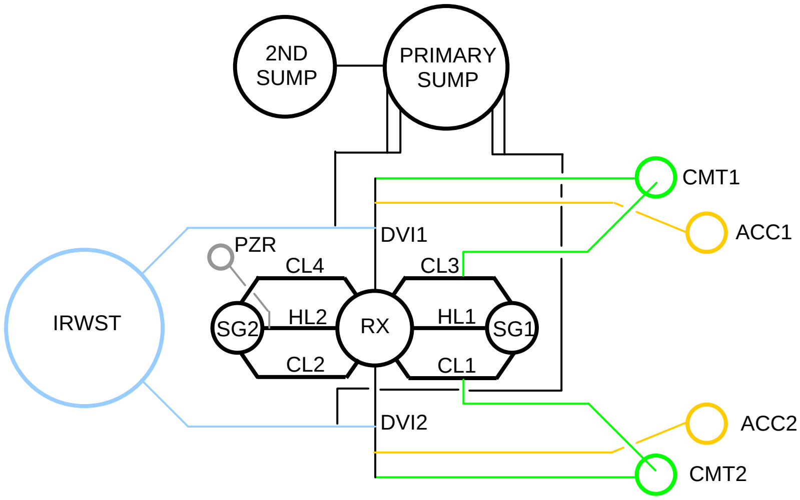

Figure 2.1 APEX-AP1000 Test Facility (Plan View)

- *Balance of Plant*. This includes a feedwater system, non safety grade Chemical Volume Control System (CVS) and an active Normal Residual Heat Removal System (RNS). The geometry of the interconnecting pipe routings was also duplicated.

3. BEYOND DESIGN BASES TEST RESULTS

One of the primary objectives of the beyond design bases tests sponsored by the NRC was to examine the effect of ADS valve failures on passive safety system performance. Figures 3.1 through 3.4 shows the location of the ADS valve failures (when applicable) and breaks for each test discussed below.

It should be noted that in the following tests descriptions, the core collapsed level is graphed to illustrate core thermal-hydraulics, since it is more stable than the core two-phase mixture level. However, in reality, the two-phase mixture level is the true indicator of core coolability. When the core is said to be "uncovered", it means that the two-phase mixture level dropping below the top of the heater rods.

3.1 NRC-AP1000-03

This test was a DEDVI line break with failure of 2 out of 4 ADS4 valves on the pressurizer side of the plant. Figure 3.1 shows the plant configuration for this test and plots of side 1 and 2 injection flows, along with reactor vessel collapsed liquid level and core temperature. The break on DVI1 was opened at time zero and CMT1 and ACC1 began injecting immediately. Shortly after, CMT2 and ACC2 began injecting. When CMT1 and ACC1 emptied at around time interval 1.5, a low core water level is observed. The ADS4-1 valves opened just after time interval 1.5 to reduce the plant pressure to allow for IRWST injection. IRWST1 injects immediately after ADS4-1 actuation, but most of this water was lost out of the DVI1 break. However, due to the drop in pressure, ACC2 and CMT2 injection flows increased.

It is apparent that this was not enough water to counteract boil-off due to decay heat as the core water level dropped. At around time interval 7 the plant reached a low enough pressure to allow for IRWST2 injection, which was enough to increase the core water level. Just before IRWST2 injection began, a small core temperature spike was observed, but was not large enough to cause a high temp heater rod excursion.

As water was injected into the core from the IRWST, the core level increased. Eventually, the IRWST ran out of water and the plant successfully entered sump recirculation for long term cooling. During the entire test, the core remained covered and adequately cooled (reactor vessel mixture level above the top of the core)

3.2 NRC-AP1000-05

This test was a DEDVI line break with failure of 2 out of 4 ADS4 valves on the non-pressurizer side of the plant. Figure 3.2 shows the plant configuration for this test and plots of side 1 and 2 injection flows, along with reactor vessel level and core temperature. The timing at the beginning of this test is very similar to NRC-AP1000-03.

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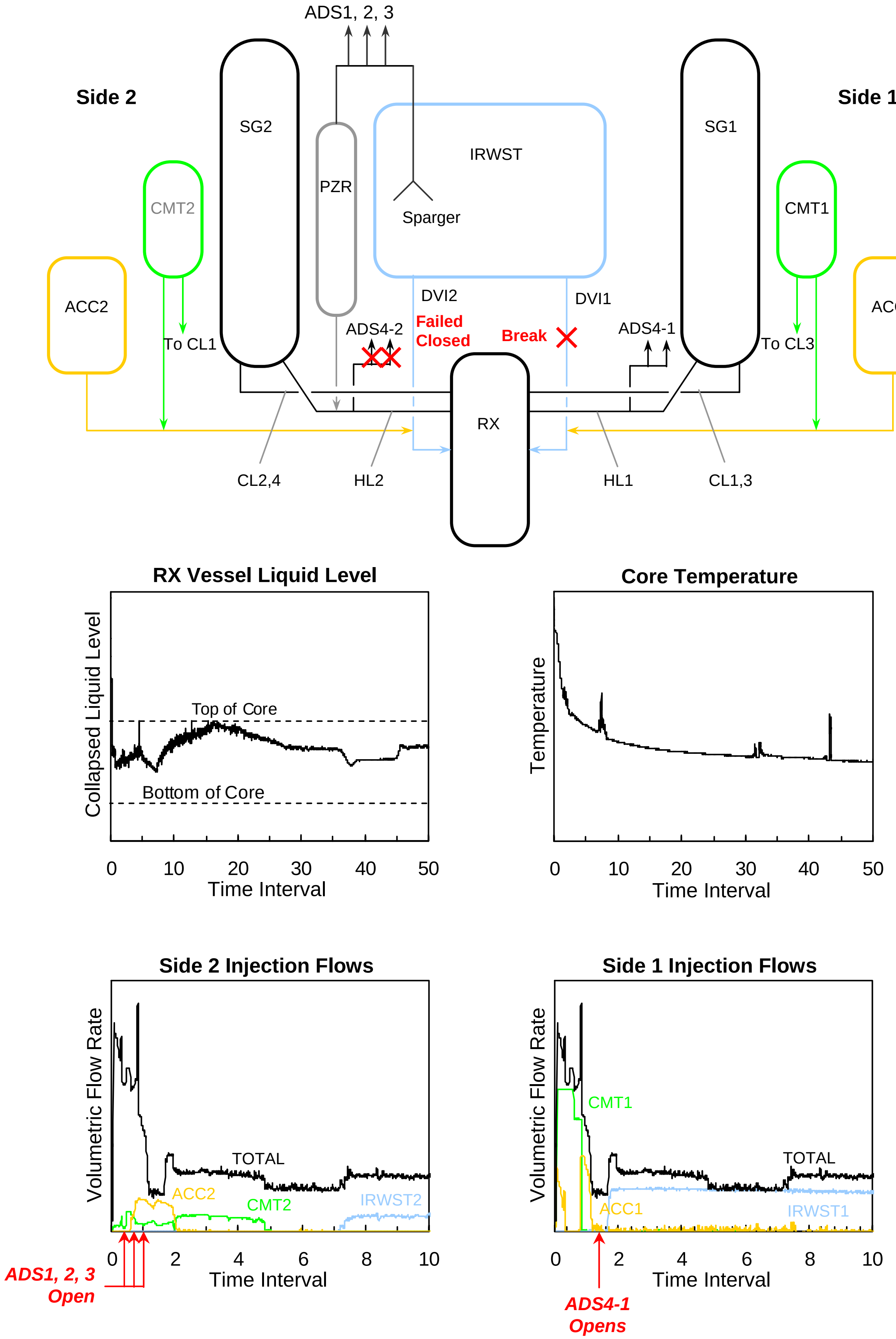

Figure 3.1 Test NRC-AP1000-03 passive safety system performance.

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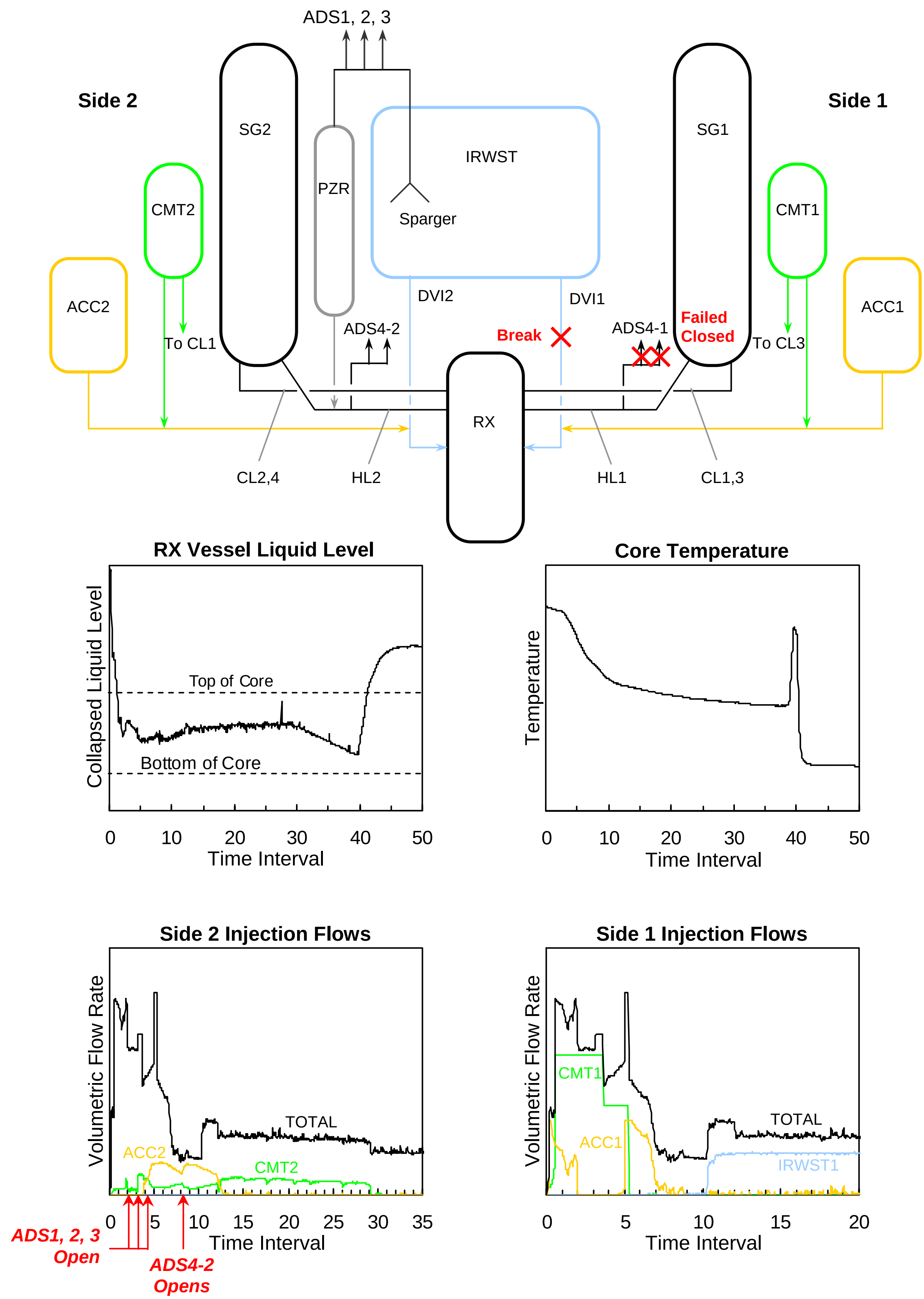

Figure 3.2 Test NRC-AP1000-05 passive safety system performance.

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In NRC-AP1000-05 when the CMT2 tank emptied, IRWST2 injection was significantly delayed. The core level began to decrease at around time interval 30 and never turned around. In addition, CMT2 injection took place over a much longer time period that in NRC-AP1000-03 when ADS4-1 valves were failed on the pressurizer side of the plant. This shows the sensitivity of the plant to the location of the ADS4 valve failure.

As the core water level decreased in NRC-AP1000-05, significant liquid carryover from the reactor vessel out the open ADS4 line was observed (ADS4-2 flow quality below 0.5) even when the vessel mixture level was below the bottom of the hot leg. This liquid carryover was due to pool entrainment at the surface of the two-phase mixture level due to decay heat boiling.

At about time interval 40, there was a large increase in the core temperature due to low reactor vessel water inventory, which caused the heater rods to automatically turn off. When electricity was cut from the heater rods, the core collapsed level immediately increased as boiling ceased and injection water began. No facility components were damaged.

3.3 NRC-AP1000-06

This test was a 5.08 cm break at the bottom of CL4 with a failure of 2 out of 4 ADS4 valves on the non-pressurizer side. It was apparent from NRC-AP1000-05 that this type of ADS4 failure can lead to core uncovery and so the purpose of this test was to determine if moving the break location from DVI1 to CL4 resulted in a similar core temperature excursion. Figure 3.3 shows the plant configuration for this test and plots of side 1 and 2 injection flows, along with reactor vessel level, core temperature, and primary system pressure.

As in previous tests, the break was opened at time zero. CMT1 and 2 began to inject immediately, but since the size of the break was relatively small, the pressure did not reach a low enough value to allow ACC injection until around time interval 15 when ADS1 and 2 actuated off the top of the PZR. After ADS1, 2 and 3 actuation, the total injection flow increased significantly until falling to zero by time interval 40.

Up to this point, IRWST injection has not begun. Without any additional safety injection water after time interval 40, the reactor vessel water level continued to drop due to pool entrainment. Eventually, since IRWST injection never happened, the core temperature rose due to boil off of primary liquid and a temperature spike occurred at around time interval 50 and the heater rods automatically shut off, thus ending the test.

3.4 NRC-AP1000-11

This test simulated a station blackout with a failure of 1 out of 4 ADS4 valves on the non-pressurizer side. Previous tests (NRC-AP1000-05 and 06) have shown that failure of ADS4 valves on the non-pressurizer side of the plant are the most challenging to the AP1000 passive safety systems.

The APEX-AP1000 control logic was modified to simulate a station blackout (loss of all AC power). The core operated at a reduced power (600 kW) and with a modified decay power curve to preserve integrated energy. The steam generator pressure operated relief valves (PORVs) were set to cycle between two relatively high pressures. When the steam generator water level decreased to a specified level due to PORV cycling, a “S” signal was sent to open CMT injection valves and the PRHR HX outlet valve. The ADS system was actuated at around time interval 68 based on a set time after initiation of the “S” signal. Figure 3.4 shows the plant configuration for this test and plots of side 1 and 2 injection flows, along with reactor vessel level and primary system pressure.

For the first several hours of the station blackout test, the PORVs cycled on and off. When the “S” signal was generated ADS1 actuated around time interval 68. The plant began to immediately depressurize and the ADS control logic depressurized the plant to allow passive safety system injection. The injection flow decreased very close to zero around time interval 70, but only for a short while until IRWST injection began. No temperature excursion was experienced and the core remain covered and adequately cooled throughout the entire test even with 1 out of 4 ADS4 failing to open on the non-pressurizer side.

3.5 NRC-AP1000-02

Test NRC-AP1000-02 investigated passive safety system performance during Mode 5 (Cold Shutdown) operation with a loss of the Normal Residual Heat Removal System (RNS). There was no break in this test, ADS1, 2, and 3 were always open, the SG secondary side was assumed drained, the CMTs and ACCs were unavailable, and 3 out of 4 ADS4 valves were unavailable or failed (one valve on the PZR side available). Power was increased to about 120 kW to simulate loss of RNS cooling. It should be noted that simultaneous multiple failures are extremely unlikely and that this test was designed to see if an extreme case could be found during Mode 5 operation that may lead to core uncovery.

Although interesting in design, test NRC-AP1000-02 did not lead to core uncovery. During

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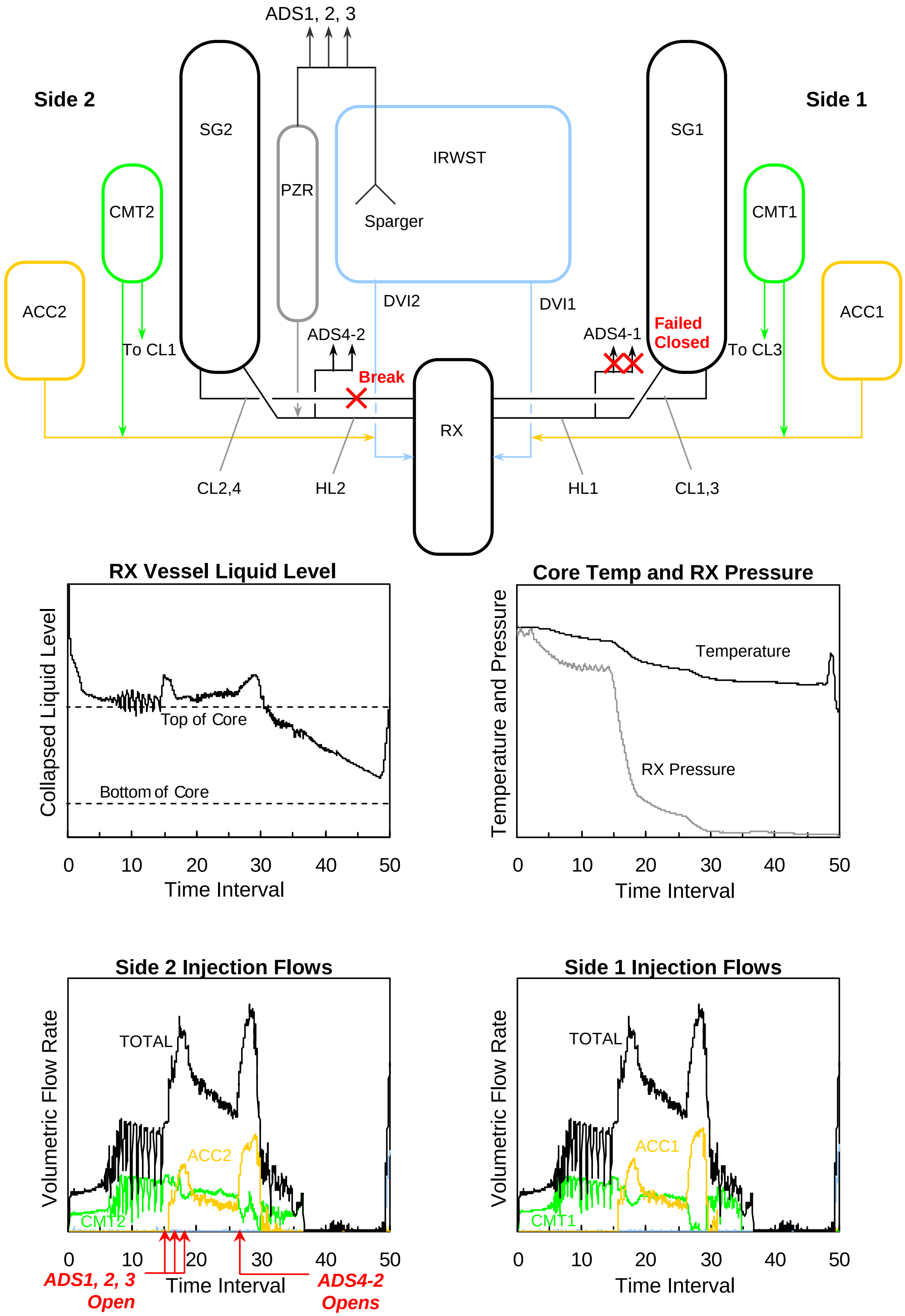

Figure 3.3 Test NRC-AP1000-06 passive safety system performance.

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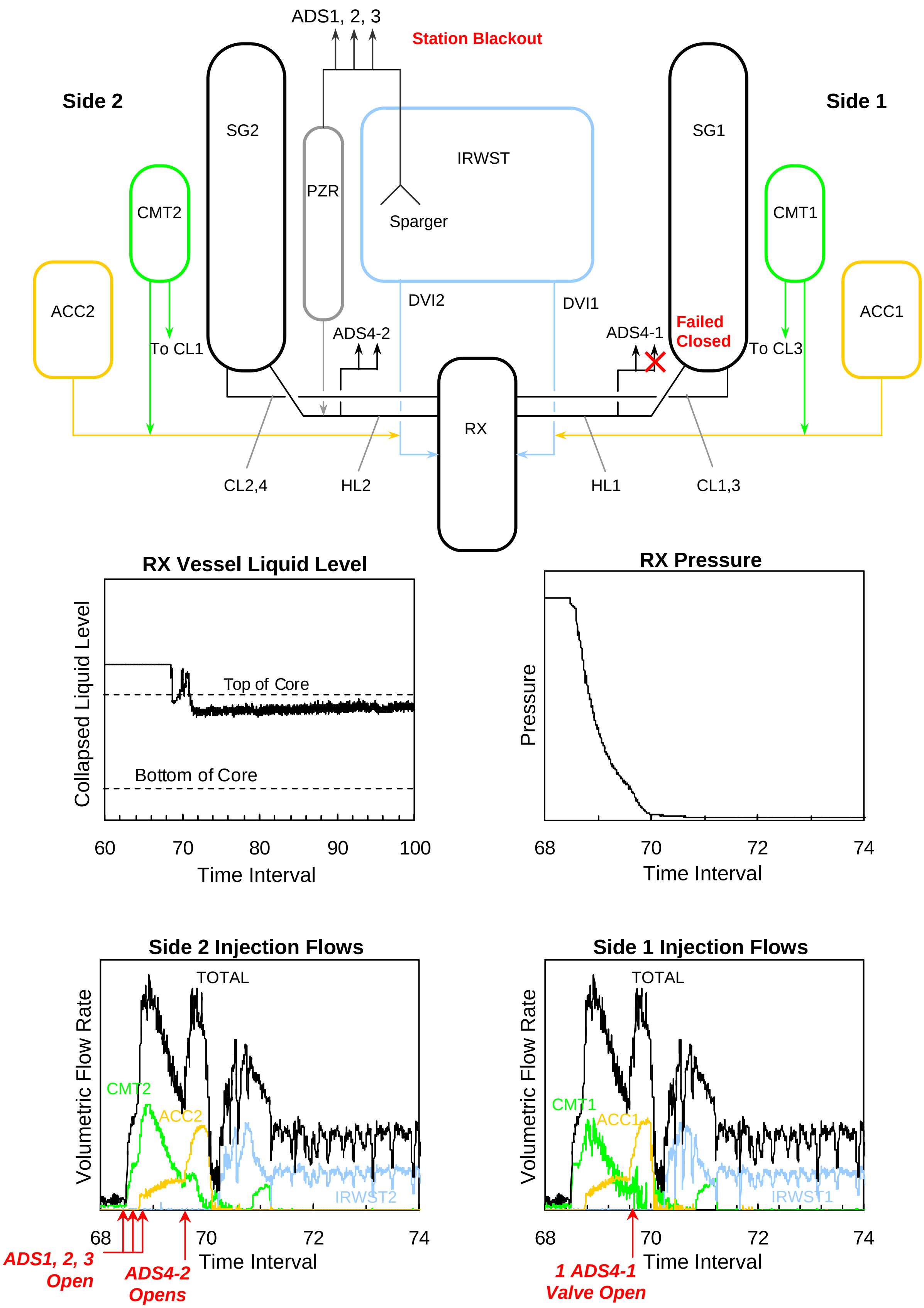

Figure 3.4 NRC-AP1000-11 passive safety system performance.

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the first part of the ADS4-2 blowdown, the reactor vessel collapsed level dropped very low, but injection flow from the CMTs and ACCs were able to quickly raise the reactor vessel water level and a temperature excursion did not occur.

4. COMPARISONS AND DISCUSSION

It is interesting to compare design bases test DBA-02 to beyond design bases test NRC-AP1000-05. Both tests simulated a DEDVI line break with ADS4 valve failure on the non-pressurizer side of the plant. The only difference between the two tests were that in the DBA-02, only 1 out of 4 ADS4 valves were failed, while in NRC-AP1000-05 two ADS4 valves were failed closed. Figure 4.1 shows a comparison of reactor vessel collapsed level for DBA-02 and NRC-AP1000-05. As previously shown in Figure 3.2 the core uncovered in NRC-AP1000-05 due to delay of IRWST injection. This delay is illustrated in Figure 4.2, where at around time interval 28, IRWST injection began in DBA-02, but in NRC-05, IRWST injection only began after the heater rods tripped on high temperature at around time interval 32. It is also important to note that there existed a significant amount of time in DBA-02 (between time interval 23 and 28) when the core did not receive any injection flow as shown in Figure 4.2. During this period, core cooling was maintained by boil off of the existing vessel inventory.

Figure 4.3 is a plot of PZR water level for both tests. At around time interval 10, the PZR was able to drain as the plant depressurized in DBA-02, providing additional makeup water to the core, but in NRC-AP1000-05, the PZR wasn't able to drain until the heater rods were turned off at around time interval 30.

Examination of the ADS4-2 flow quality (Figure 4.4) for both tests shows significant liquid carryover from the reactor vessel and entrainment from the hot leg out the ADS4-2 line. Even after the two-phase mixture dropped below the bottom of the hot leg in NRC-AP1000-05 (time interval 25 in Figure 4.1) the ADS4-2 quality was still relatively low (~0.6). The primary mechanism by which water left the reactor vessel at low vessel liquid levels was due to pool entrainment. Droplets from the two-phase reactor vessel mixture level were entrained by the steam, passed through the upper internals (with some droplets de-entraining), and entered the hot leg.

From there, some droplets struck the hot leg pipe walls and created an annular flow of liquid out the ADS4 line or were de-entrained into the hot leg liquid level and entrained out the vertical ADS4 line by a mechanism of liquid entrainment in horizontal pipes with vertical-up branches. In DBA-02, there was still one operational ADS4 valve on the non-pressurizer side of the plant, providing another outlet for the steam. In NRC-05, all of the steam exited the ADS4-2 line and the DEDVI break. Hence, a higher flow quality was observed in NRC-AP1000-05 than compared to DBA-02.

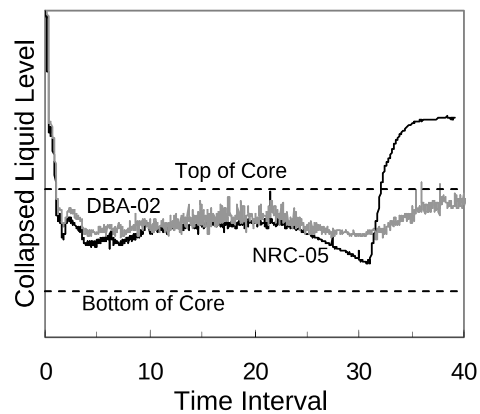

Figure 4.1 Reactor vessel collapsed liquid level

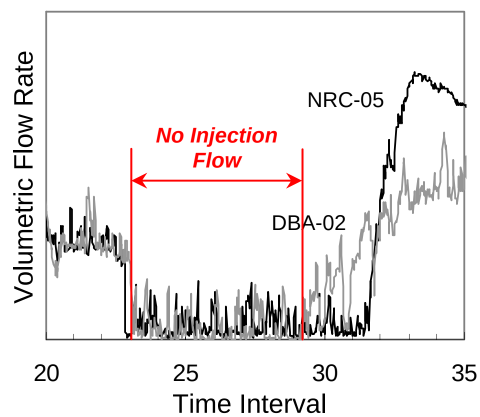

Figure 4.2 Total injection flow rate: ACCs, CMTs, and IRWST2.

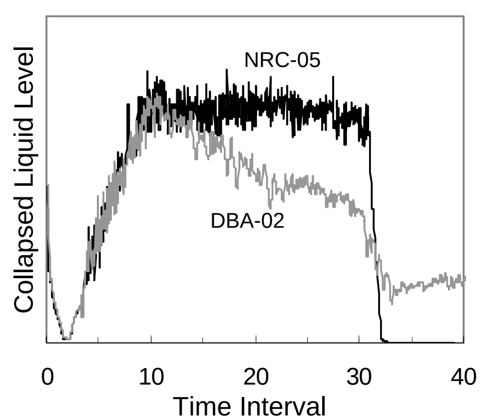

Figure 4.3 PZR collapsed liquid level.

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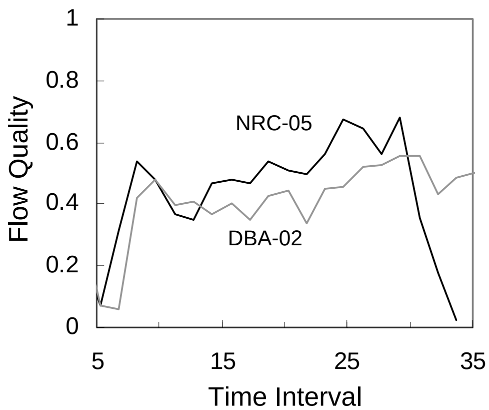

Figure 4.4 ADS4-2 flow quality.

5. CONCLUSIONS

The APEX test facility at Oregon State University was modified based on a comprehensive scaling analysis by Reyes (2003) to be representative of AP1000 prototypic conditions. The NRC sponsored numerous beyond design bases confirmatory integral tests to support AP1000 design certification activities. The main focus of these tests was to examine the effect of ADS valve failures on passive safety system performance during simulated LOCAs (DEDVIs and cold leg breaks). Tests NRC-AP1000-03, 05, 06, 11, and 02 were discussed in detail. In addition, NRC-AP1000-05 results were compared to a DOE sponsored design bases tests (DBA-02). Based on these results, the following conclusions can be made:

1. The APEX-AP1000 tests confirm significant liquid entrainment and carryover of water to the ADS4 system during and after ADS4 actuation. High liquid carryover to the ADS4 should also be expected in the AP1000. Thus, thermal-hydraulic codes used to analyze the AP1000 must adequately predict or bound upper plenum and hot leg entrainment;
2. Test results from NRC-AP1000-03, 05, and DBA-02 show that failure of ADS4 valves on the non-pressurizer side of the system results in a greater delay in IRWST injection than failure of ADS4 valves on the pressurizer side of the plant;
3. Failure of two out of four ADS4 valves on the non-pressurizer side of the plant was found to produce core uncovery (based on the two-phase mixture level) during simulated DEDVI line and cold leg breaks;
4. Review of the APEX-AP1000 test results and code calculations submitted by Westinghouse conclusively show that the core remains cooled and heat up is not experienced for design basis scenarios. The only core heat up that was experienced was for select beyond design basis tests (NRC-AP1000-05 and 06).

The analysis performed by RES in support of AP1000 design certification provided the staff with a clear and comprehensive understanding of AP1000 passive safety system performance and enabled the staff to address initial concerns of increased liquid entrainment and carryover and ADS4 valve sensitivity. In addition, the experimental AEPX-AP1000 data can be used to benchmark thermal-hydraulics codes. Currently, the AP1000 design certification is under final rulemaking at the United States Nuclear Regulatory Commission and a final ruling is expected by December 2005.

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