SIMULATED AP1000 RESPONSE TO DESIGN BASIS SMALL-BREAK LOCA EVENTS IN APEX-1000 TEST FACILITY

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As part of the AP1000™[1)] pressurized water reactor design certification program, a series of integral systems tests of the nuclear steam supply system was performed at the APEX-1000 test facility at Oregon State University. These tests provided data necessary to validate Westinghouse safety analysis computer codes for AP1000 applications. In addition, the tests provided the opportunity to investigate the thermal-hydraulic phenomena expected to be important in AP1000 small-break loss of coolant accidents (SBLOCAs). The APEX-1000 facility is a 1/4-scale pressure and 1/4-scale height simulation of the AP1000 nuclear steam supply system and passive safety features. A series of eleven tests was performed in the APEX-1000 facility as part of a U.S. Department of Energy contract. In all, four SBLOCA tests representing a spectrum of break sizes and locations were simulated along with tests to study specific phenomena of interest. The focus of this paper is the SBLOCA tests. The key thermal-hydraulic phenomena simulated in the APEX-1000 tests, and the performance and interactions of the passive safety-related systems that can be investigated through the APEX-1000 facility, are emphasized. The APEX-1000 tests demonstrate that the AP1000 passive safety-related systems successfully combine to provide a continuous removal of core decay heat and the reactor core remains covered with considerable margin for all small-break LOCA events.

KEYWORDS : AP1000, APEX-1000, SBLOCA, Safety Analysis, Thermal-Hydraulic

1. INTRODUCTION

A wide spectrum of accident sequences was assessed for the AP1000 from a full spectrum of loss-of-coolant accidents (LOCA), to non-LOCA events. The AP1000 is designed to handle a large-break LOCA in a similar way to conventional active plants. The core is uncovered in blowdown and is reflooded and quenched by the accumulators. The non-LOCA events – such as loss of power, steam line and feed line breaks, and steam generator tube rupture events – are successfully mitigated by the passive core cooling system without the need for active systems, such as safety injection pumps. The spectrum of small-break LOCA (SBLOCA) events was also analyzed and was found to be successfully mitigated by the passive core cooling system. The U.S. Nuclear Regulatory Commission (U.S. NRC) agreed with this assessment, but required that additional SBLOCA tests be performed at the higher power level to demonstrate the safe operation of the plant, and to provide additional data to validate SBLOCA computer codes.

The APEX facility was used to provide data to support the AP600 design certification [1, 2]. For performing these SBLOCA tests, the APEX facility was modified to better reflect the larger AP1000 and was renamed the APEX-1000 facility. Tests were conducted to span a spectrum of SBLOCA events varying the location and size of the breaks. These tests represent a 1/4-scale height and a 1/4-scale pressure scaling of the AP1000 nuclear steam supply system and the passive safety systems. The data from these tests were used to address U.S. NRC concerns and to provide a basis for code validation.

Sections 2 through 5 briefly describe the AP1000 design, APEX-1000 facility, important thermal-hydraulic phenomena, and APEX-1000 test matrix. Section 6 discusses the test results in terms of the implications on the ability of the AP1000 to provide continued core cooling.

Because of the proprietary nature of these data, the results are presented in non-dimensional form.

[1)] AP1000 is a trademark of Westinghouse Electric Company LLC.

2. AP1000 DESCRIPTION

The AP1000 is a two-loop, 1000 MWe plant that uses the same basic design as the AP600 which received Final Design Approval by the US-NRC in 1998. The core diameter, reactor vessel diameter, containment diameter, and nuclear island footprint are all the same as for the AP600. The main difference between the AP600 and the AP1000 containment building is the building height, which was increased to add free volume and to accommodate larger components. Those component changes that impact the plant the most are the increase in size of the steam generator, from a Delta-75 design (6967 m^2 heat transfer area) to a Delta-125 design (11611 m^2 heat transfer area), and the larger capacity reactor coolant pump. An example of one of the added benefits of larger reactor coolant pumps is the higher inertia compared to that for the AP600. This higher inertia allows increased margin for departure from nucleate boiling (DNB) for loss-of-flow events by providing increased coastdown flow to the core after pump trip.

Unique to the Westinghouse advanced plant designs is the passive core cooling system shown in Fig. 1. The passive core cooling system consists of the following:

- Two full-pressure core makeup tanks that provide borated makeup water to the primary system at any pressure
- Two accumulators that provide borated water to the reactor vessel if the primary pressure falls below 4.83 MPa (700 psia)
- A passive residual heat removal heat exchanger (PRHR HX) composed of a C-shape tube bundle submerged inside the in-containment refueling water storage tank (IRWST), which can remove heat from the primary system at any pressure
- The automatic depressurization system (ADS), which comprises a set of valves connected to the pressurizer steam space and the two hot legs

These valves are opened sequentially to provide a controlled depressurization of the primary system. The first three stages of the ADS discharge to the IRWST

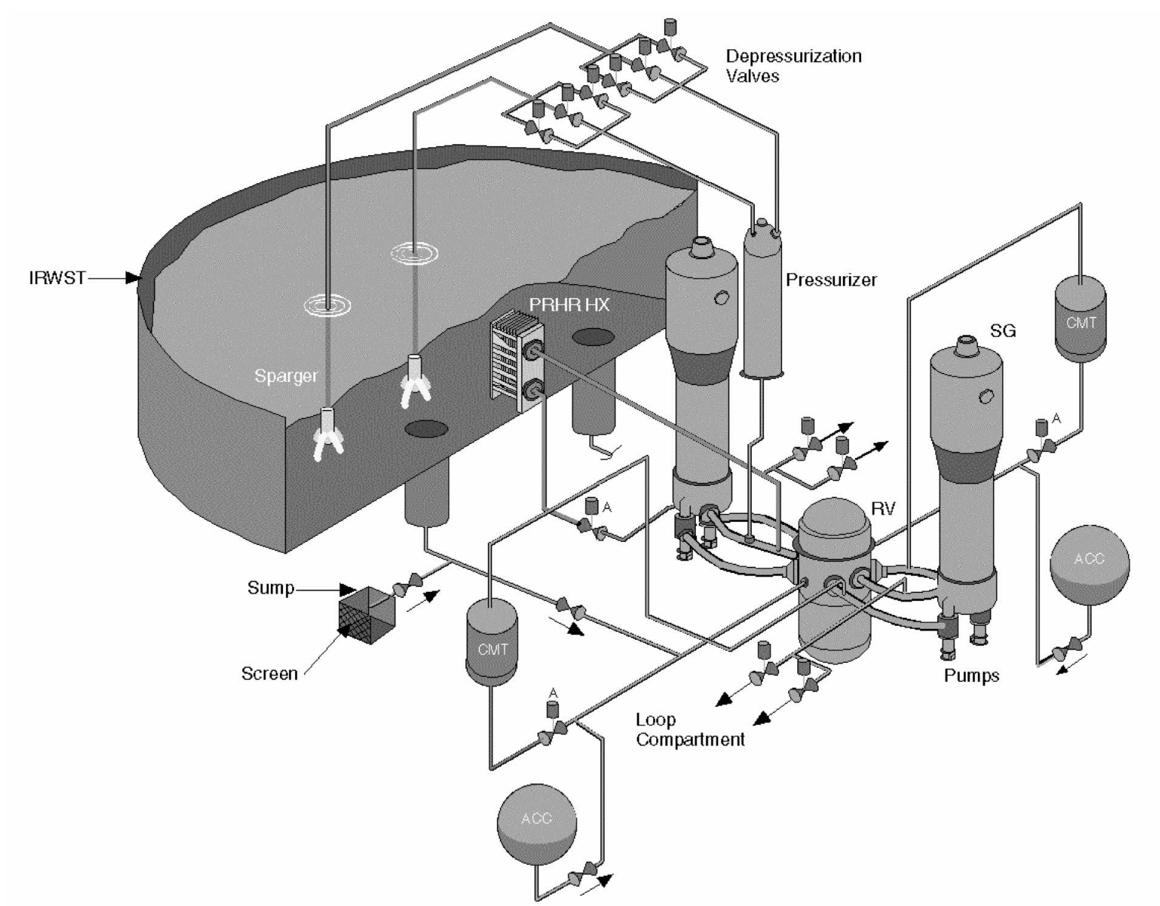

Fig. 1. AP1000 Passive Core Cooling System

where the steam is condensed, while the fourth stage discharges directly to the containment atmosphere.

- An IRWST that provides a large source of core cooling water, which drains by gravity after the ADS has reduced the primary system pressure to near the containment pressure
- A recirculation sump that collects water discharged from the primary system and steam that condenses within the containment

After the primary system is depressurized, and the gravity head becomes great enough, the water in the sump is recirculated to the primary system.

3. APEX-1000 FACILITY DESCRIPTION

The APEX-1000 test facility, shown in Fig. 2, is a 1/4-height, 1/2-time-scale reduced pressure integral systems facility [1]. Included in this facility are the following:

- *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, a pressurizer, and four reactor coolant pumps.
- *Passive safety systems*: This includes two core makeup tanks, two accumulators, a four-stage ADS, a PRHR HX, an IRWST, and portions of the lower containment compartments.
- *Balance of plant*: This includes a feedwater system, a nonsafety grade chemical and volume control system, and an active residual heat removal system.

All of the reactor coolant system components are constructed of stainless steel and are capable of consistent operation at 2.76 MPa (400 psia) while at the saturation temperatures. All primary system components are insulated to minimize heat loss.

Based on a review of the AP1000 design changes, a number of modifications were made to the APEX-1000 facility to best simulate AP1000 thermal-hydraulic behavior. Table 1 identifies the AP1000 changes and the

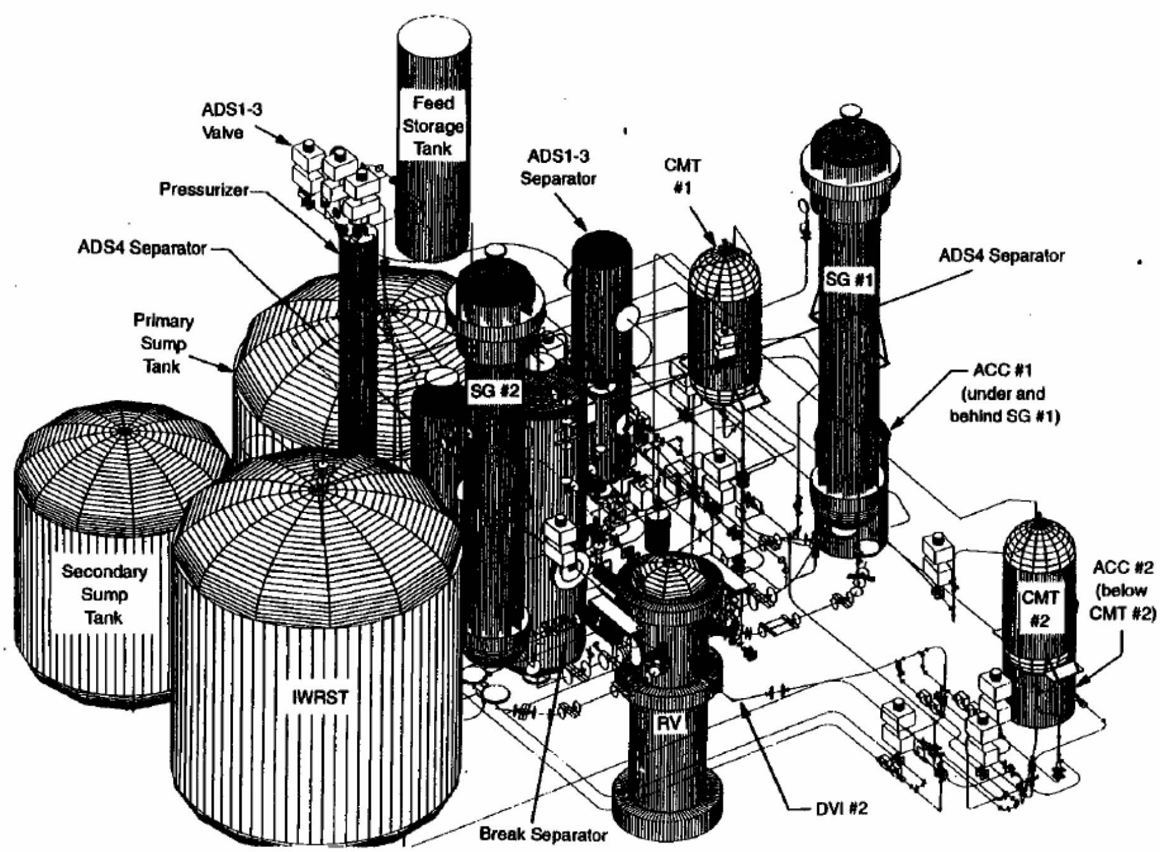

Fig. 2. APEX-1000 Test Facility

corresponding modification to the APEX-1000 facility. The reactor core power was increased from 600 kW to 1000 kW. Modifications were made to the pressurizer and surge line. The steam generators and reactor coolant pumps were not changed since the volume in the steam generator tubes is small compared to the remainder of the reactor coolant system, and the pumps are tripped at the start of the simulated accident.

The passive core cooling system was modified, including larger core makeup tanks, larger injection lines from the IRWST, and an increase in the line diameter for the fourth-stage valves ADS. Increases in the PRHR HX tube volume were not necessary since the heat exchanger was somewhat oversized for AP600. In addition, the first three stages of the ADS valves were not changed from the AP600 to the AP1000 designs and were not changed in APEX-1000.

Fig. 3 is a photograph of the reactor vessel and loop piping.

4. IMPORTANT THERMAL-HYDRAULIC PHENOMENA FOR AP1000 SBLOCAS

The SBLOCA transient can be subdivided into four different phases that characterize thermal-hydraulic phenomena. These are described in the following subsections.

4.1 Blowdown

This phase entails the initial depressurization from the primary system operating pressure to the steam generator secondary-side pressure, after which the pressure stabilizes. For the simulated SBLOCAs, the reactor is assumed to be initially operating at normal full-power, steady-state conditions. The break opens at time zero, and the pressurizer pressure begins to fall as mass is lost through the break. The depressurization rate is largely determined by critical two-phase flow through the

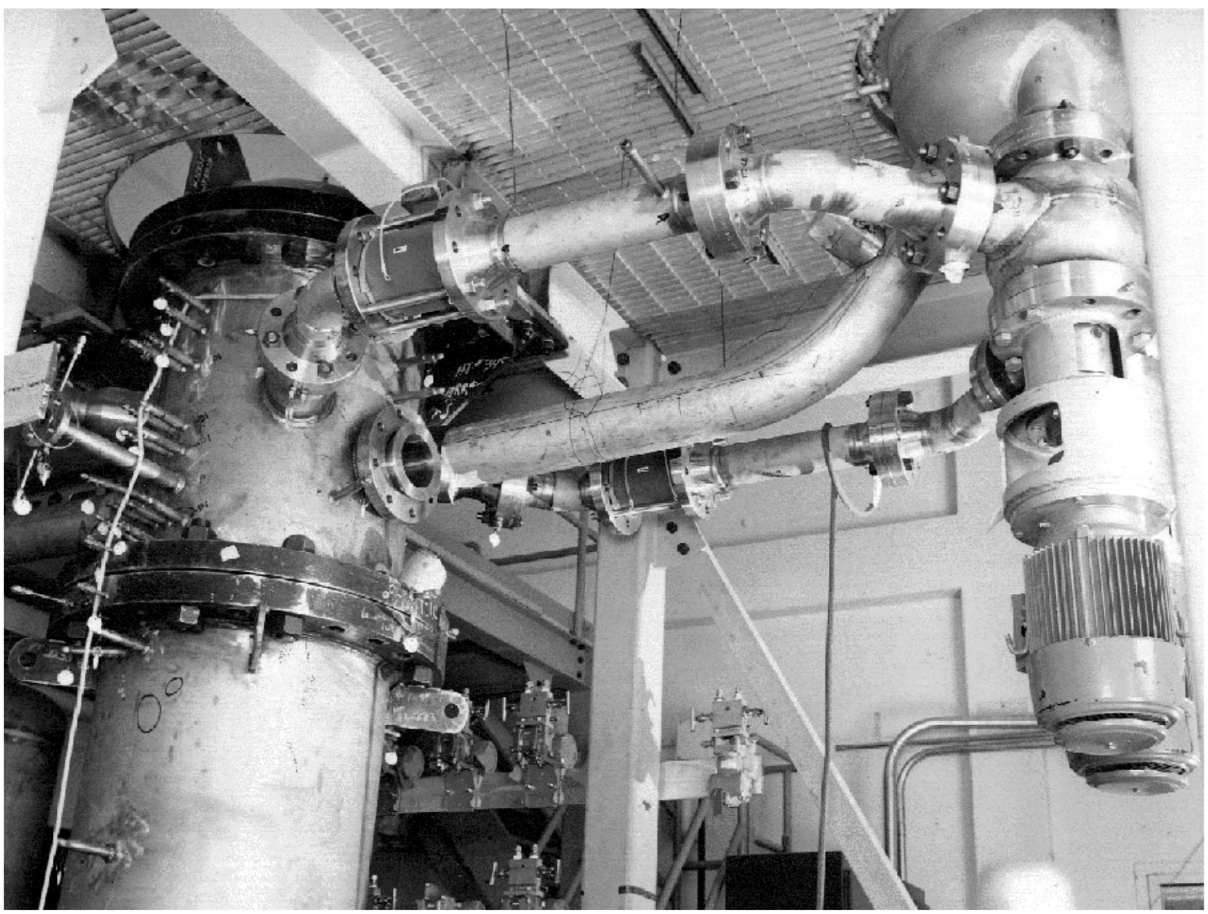

Fig. 3. APEX-1000 Test Facility Showing Reactor Vessel and Loop Piping

break. When the pressurizer pressure falls below the safety signal setpoint, the safety systems actuation (S) signal is issued, which causes the reactor to trip. The S-signal also causes the opening of the core makeup tanks and PRHR HX isolation valves. Once the residual fissions decrease, core power is defined by the decay heat model. The reactor coolant pumps trip after a short delay, and the rapid coastdown expected from the AP1000 canned motor reactor coolant pumps is simulated. After the pumps coast down, the primary reactor coolant system is cooled by natural circulation, with energy removed from the primary system by heatup of the steam generators, recirculation flow to the core makeup tanks, and fluid loss through the break. Stored energy from the metal in the primary system is transferred to the coolant. The liquid in the upper plenum and upper head may flash, and as the primary system pressure continues to fall, the upper head will begin to drain. Note that all of these phenomena are essentially the same for the AP1000 as they are for conventional pressurized water reactors.

Blowdown phase phenomena associated with the core makeup tanks and PRHR HX operation are unique to the AP1000. Once the core makeup tank isolation valves open on a safety systems actuation signal, the core makeup tanks inject cold borated water into the reactor coolant system through the direct vessel injection (DVI) lines as a result of gravity-driven recirculation. The core makeup tank volume injected is replaced with hot liquid from the cold leg, which circulates through the cold leg balance line. This hot liquid collects at the top of the core makeup tank. This phenomenon is referred to as the core makeup tank recirculation mode of operation. As a result of the core makeup tank injection, the downcomer fluid stays subcooled through the initial depressurization.

4.2 Natural Circulation

This is the time period from the stabilization of the primary pressure with the secondary-side pressure until ADS stage 1 is activated. The primary system is cooled by different modes of natural circulation; that is, single-phase natural circulation, two-phase natural circulation, and reflux condensation. Each cooling mode is dependent on the system mass inventory. As the mass is lost from the break, the primary system drains and the cooling evolves from single-phase to two-phase, to reflux condensation cooling.

During the natural circulation phase of the transient, the primary system exists in a quasi-steady-state condition with the secondary side. As the primary system continues to drain, the decay energy is removed by the flow through the break, steam generators, PRHR HX, and core makeup tanks that store energy from the recirculation flow. The steam generators in the AP1000 play a more limited role in the natural circulation cooling phase than for conventional plants because the primary tubes of the generator drain relatively early in the transient. The PRHR HX is activated early during an SBLOCA, and the IRWST becomes the primary heat sink for the reactor coolant system once the steam generator primary side tubes have drained. The continued removal of energy by the PRHR HX, the core makeup tanks, and the break causes the primary system to further depressurize. The core makeup tanks continue in the recirculation mode, providing heat removal and a small net injection to the primary system until the cold legs and balance lines drain. This allows vapor into the top of the core makeup tank volume after which core makeup tank draindown begins. The core makeup tank level eventually falls to the ADS actuation setpoint, which initiates the third phase of the AP1000 SBLOCA transient – the ADS phase.

4.3 Automatic Depressurization System (Stages 1, 2, 3, and 4)

Once the core makeup tanks drain to a specified level setpoint, the ADS stage 1 valve opens and the reactor system is depressurized through the ADS flow path connected to the top of the pressurizer, as well as the break. As the core makeup tanks continue to drain into the reactor vessel, additional ADS valves are opened on the pressurizer to enhance the depressurization of the system. When the core makeup tanks are almost completely drained, the fourth-stage ADS valves open directly from the hot legs and the system pressure is reduced to near-containment pressure.

Since ADS stage 1 actuation results in a reduction in pressure at the top of the pressurizer, the pressurizer two-phase fluid level increases markedly. Thus, pressurizer tank level and surge line phenomena are important factors in the depressurization behavior following ADS actuation. Fluid flashing occurs again in the reactor coolant system due to increased depressurization rate after the ADS is actuated.

Following ADS stage 1 actuation, ADS stages 2 and 3 activate in sequence via pre-set timers to provide a gradually increasing discharge area. Accumulator injection begins once the pressure drops below 4.83 MPa (700 psia). This reduces the discharge flow from the core makeup tank, which may be stopped temporarily due to increased backpressure in the DVI line, caused by the high accumulator flow rate. The core makeup tank drain rate, as well as DVI line and cold leg balance line behavior, directly affect the ADS stage 4 actuation since it is based upon the core makeup tank liquid level decreasing below a low-low setpoint value.

The flow through the ADS valves is the major factor in determining when the reactor coolant system has depressurized to the extent that the gravity injection of water from the IRWST can begin. Fourth-stage ADS performance is affected by the nature of flow in the hot legs, and can be affected by drainage from the pressurizer.

Successful operation of the ADS leads to the IRWST injection cooling phase of the AP1000 SBLOCA event.

4.4 Post-Automatic Depressurization System, IRWST Injection

Stable injection from the IRWST indicates the depressurization of the primary system to near containment pressure. Also, injection from the IRWST marks the end of the small-break transient and the beginning of the long-term cooling transient.

In the APEX-1000 tests, the final stage of the simulated SBLOCA is IRWST injection. At this point, the primary system is completely depressurized and the transient would continue into the long-term cooling phase of the accident. At the time of IRWST injection, the core makeup tank is either completely empty or nearly empty; therefore, core makeup tank phenomena have become relatively unimportant, whereas the IRWST gravity drain rate through the DVI lines is now significant. The hot leg flow phenomena including possible interaction from pressurizer draining, together with the ADS stage 4 flow, is also important. The break flow behavior has less impact than before since all the ADS flow paths are open. This provides a large area through which to vent fluid. At this time, the primary system pressure is nearly the same as that inside the containment building, and maintaining core coverage with liquid or a two-phase mixture becomes dependent upon the decay heat level and the IRWST flow.

It is possible for the core makeup tanks to empty before IRWST injection is established. This would occur if there is decreased venting through the ADS stage 4 due to a failure in one of the four valves. During this injection "gap," the core is cooled by boiling off the inventory in the reactor vessel. Eventually, the reactor coolant system pressure is reduced to the point where IRWST injection begins when the two-phase level is substantially above the top of the core.

Fig. 4 shows the pressure transient for a typical SBLOCA event. Superimposed on the graph is the relative timing of events as described in the previous sections.

5. APEX-1000 TEST PROGRAM

In addition to the extensive test program conducted during the AP600 design certification, it was determined that additional SBLOCA tests be performed for the AP1000 to provide additional data for safety analysis computer code validation. A series of tests was performed at the modified APEX-1000 facility to simulate design basis SBLOCA events [3]. The tests were performed under a U.S Department of Energy

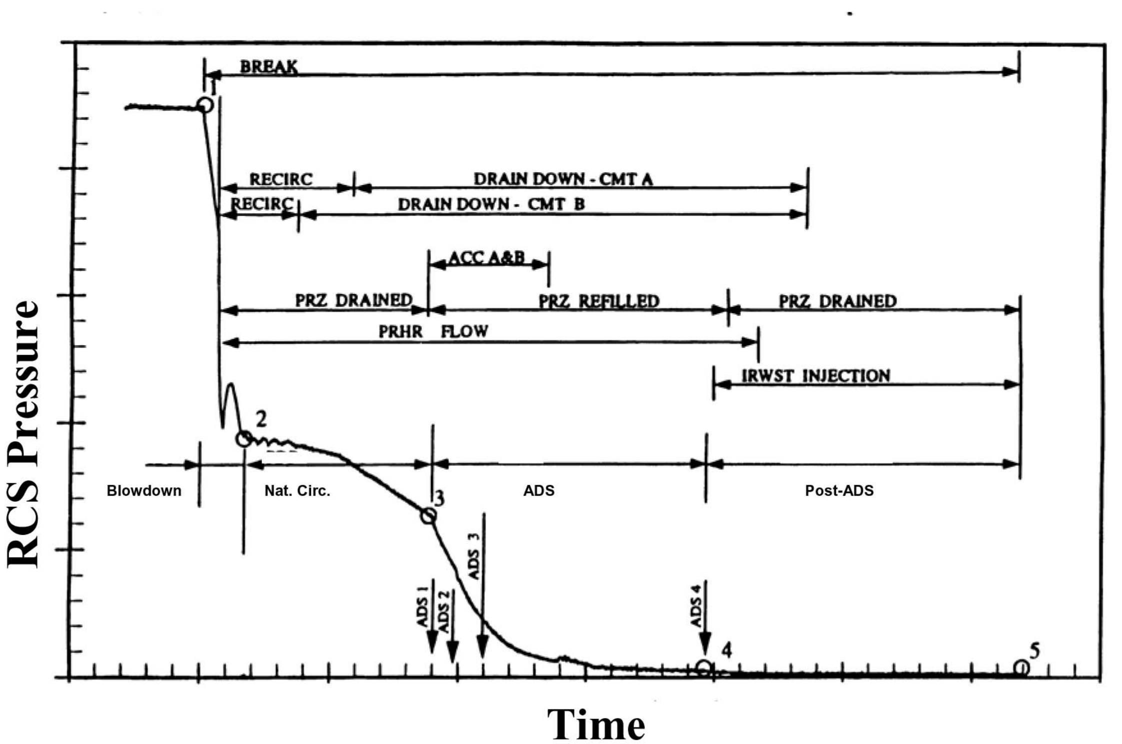

Fig. 4. AP1000 Typical SBLOCA Pressure Transient

program and included the following SBLOCA events:

- A double-ended break of the DVI line (first four tests)
- A 5.08 cm break in one of the cold legs
- A 1.27 cm break in one of the cold legs
- A "no-break" case with a spurious actuation of the ADS

Several other tests were conducted during the course of this program, but this discussion is focused on these design basis accident simulations. It will be determined whether the higher power will adversely affect the ability of the passive core cooling system to mitigate the accident and to achieve stable IRWST injection flow when the reactor coolant system is depressurized. For each test, a single failure of one of the safety components is assumed. The limiting single failure is one of the ADS Stage 4 valves. This limits the venting of the reactor coolant system and delays the onset of IRWST injection. Further, it was determined by numerical simulation that water draining from the pressurizer at the time of ADS-4 opening could interfere with the steam venting, and that the location of the single failure affected the reactor coolant system depressurization and timing of the start of IRWST injection. The matrix of design basis accident simulations at the APEX-1000 test facility is shown in Table 1.

6. RESULTS

Prior testing for the AP600 showed that the greatest challenge to the passive core cooling system is a DVI line break. Thus, the reference test for this program is the double-ended DVI break. Results are presented for this test, and these results are compared to other break sizes and locations. The timing of key events for the other tests is presented relative to this reference test, and comparisons are made of the reactor coolant system pressure, two-phase level in the core, and safety injection flow for the spectrum of breaks tested.

6.1 Results for Reference SBLOCA Test – Double-ended DVI Line Break

The AP600 test program identified the double-ended DVI line break as the limiting SBLOCA event. This is due, in part, to the relatively large size of the DVI line, which is a 20.3 cm line (17.3 cm inside diameter). In addition, the loss of one of the two DVI lines results in only half the safety injection sources being available for core cooling as the flow from the broken line is spilled to the containment.

The core makeup tank on the broken side drains relatively quickly resulting in an early actuation of the

Table 1. Matrix of AP1000 SBLOCA Simulations at APEX-1000

Test Number	Description	Initiating Event	Single Failure	Continuous Injection	IRWST Injection
DBA-01	DE DVI break	DE break in DVI 1	None	Yes	Yes
DBA-02	DE DVI break	DE break in DVI 1	1 of 4 ADS-4 valve (non-PZR loop)	CMT-IRWST gap	Yes
DBA-03	DE DVI break	DE break in DVI 1	1 of 4 ADS-4 valve (PZR loop)	Yes	Yes
DBA-03R	Repeat of DBA-03	DE break in DVI 1	1 of 4 ADS-4 valve (PZR loop)	Yes	Yes
DBA-04	5.04 cm break in CL	Break in cold leg 3	1 of 4 ADS-4 valve (non-PZR loop)	CMT-IRWST gap	Yes
DBA-05	Inadvertent ADS	ADS stage 1 opens	1 of 4 ADS-4 valve (non-PZR loop)	CMT-IRWST gap	Yes
DBA-06	1.27 cm break in CL	Break in cold leg 3	1 of 4 ADS-4 valve (non-PZR loop)	CMT-IRWST gap	Yes

Notes:
DE – double-ended
CL – cold leg
CMT – core makeup tank
PZR – pressurizer

ADS, and in turn, an early actuation of ADS stage 4. Actuating ADS stage 4 when the reactor decay power is relatively high presents a challenge since the higher steam velocities would tend to entrain more liquid and consequently reduce the venting capability. In addition, the pressurizer, which refills when the early stages of ADS are actuated, could have substantial liquid inventory when ADS stage 4 is actuated and could impede the venting of steam from the valves on the hot leg that contains the pressurizer.

Fig. 5 shows the reactor coolant system pressure for this event. The pressure decreases rapidly due to the large break size. Early actuation of the ADS further reduces the pressure until ADS stage 4 allows IRWST injection. This figure shows that the pressure decrease is gradual after ADS stage 4. Stable IRWST injection is established when the hydrostatic head in the injection line exceeds the pressure difference between the reactor coolant system and the containment. Restricted venting through the ADS stage 4 valves can delay the time of IRWST injection. Fig. 6 shows the injection flow for the three sources on the intact DVI line: core makeup tanks, accumulators, and IRWST. Note that the core makeup tank begins to inject immediately and continues injecting until the accumulator charging pressure is reached. The core makeup tank flow is suppressed as the accumulators inject because the two sources share a common injection line. As the accumulators empty, the core makeup tank flow resumes until the core makeup tank on the intact DVI line is drained.

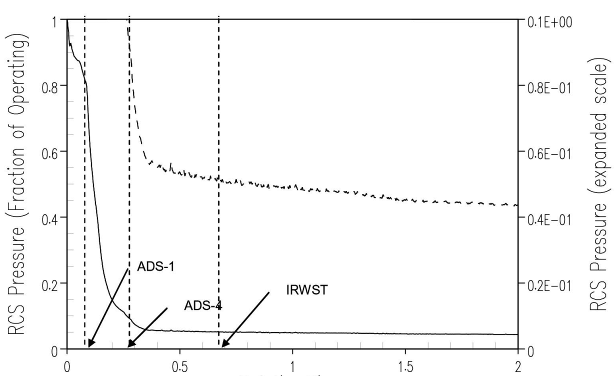

Fig. 5. Reactor Coolant System Pressure – Double-Ended DVI Break (DBA-02)

For this case (DBA-03R), with the single failure assumed on one of the two ADS stage 4 valves on the hot leg containing the pressurizer, there is no injection gap as IRWST flow begins before the core makeup tanks empty. However, for the case where the single failure is on the non-pressurizer hot leg (DBA-02), Fig. 7 shows an injection gap as the core makeup tanks empty before IRWST flow is established. This is due to the water inventory in the pressurizer, which is draining into the hot leg at the same time that steam is venting from the ADS stage 4 from the more open vent pathway (two of two ADS stage 4 valves available as opposed to one of two ADS stage 4 valves available on the non-pressurizer hot leg). Thus, the overall venting capability is reduced for the case where the single failure occurs on the non-pressurizer hot leg.

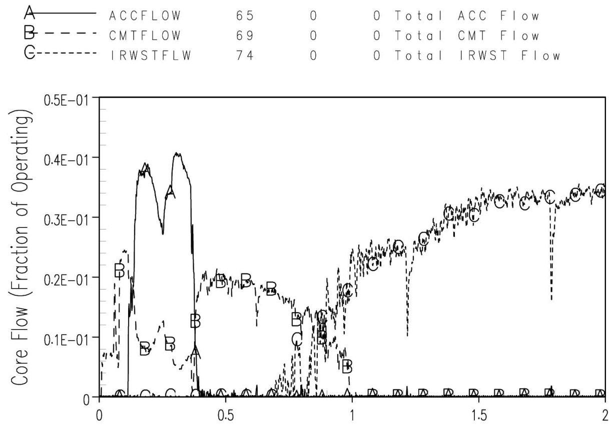

Fig. 6. Safety Injection Flow – Double-Ended DVI Break (DBA-03R, Failed ADS-4 on PZR Loop)

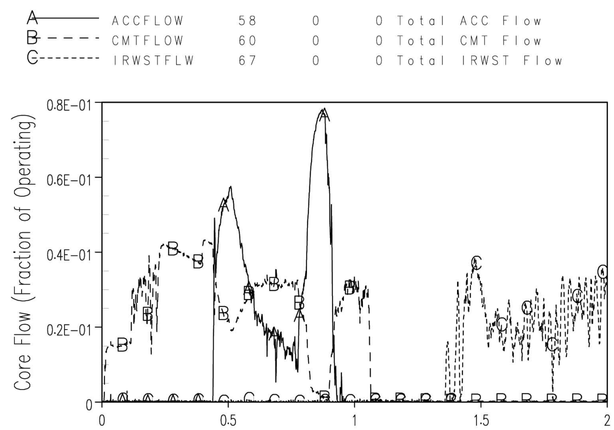

Fig. 7. Safety Injection Flow – 5.08 cm Cold Leg Break (DBA-02, Failed ADS-4 on Non-pressurizer Loop)

Fig. 8 shows the two-phase level in the reactor vessel for this event. The initial reduction due to the blowdown results in a minimum level in the upper plenum that is below the bottom of the hot leg, but well above the top of the core. There is no effect of the injection gap as the

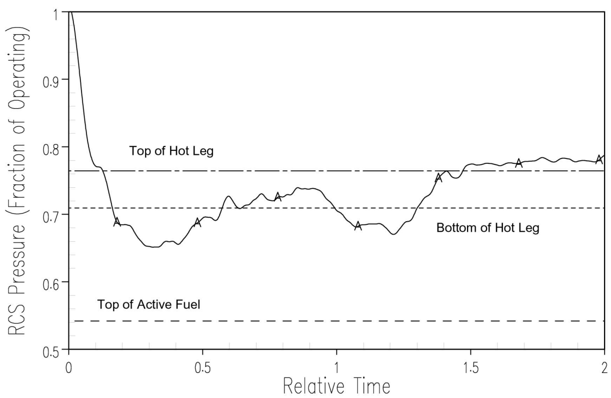

Fig. 8. Core Two-Phase Level – Double-Ended DVI Break (DBA-02)

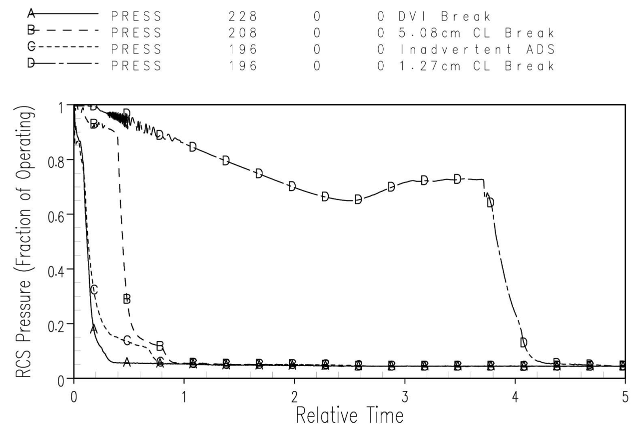

Fig. 9. Reactor Coolant System Pressure Response – Break Spectrum

two-phase level remains at the level of the hot leg. During the period of no injection, the core exit void fraction increases, which reduces the hydrostatic head in the upper plenum, hot leg, and ADS stage 4. This results in a higher steam fraction exiting the ADS stage 4 and more effective venting. Thus, the injection gap is self-limiting, and no drop in the two-phase level is observed. Even with one-half the makeup water sources available, the level recovers and remains in the vicinity of the hot leg for the remainder of the event, and stable IRWST injection is established.

6.2 Results for Other SBLOCA Tests

The biggest difference between the four SBLOCA tests is the timing of the events as shown in Table 2. This difference in the timing of the safety system actuation results in different reactor coolant system pressure response as shown in Fig. 9. As discussed in the previous section, the double-ended DVI is a relatively large break that depressurizes the reactor coolant system

Table 2. Comparison of Event Timing for SBLOCA Tests (Difference in Event Timing from Reference Test [DBA-03R] in Seconds)

	DVI	5.08 cm CL	1.27 cm CL	Inadvertent ADS
Start of test	0	0	0	0
S-signal	0	0	182	0
Core makeup tank/PRHR start	0	0	182	0
Reactor coolant pumps off	0	0	234	0
ACC start	0	326	3725	11
ACC end	0	540	3870	413
ADS-1	0	304	3628	-87
ADS-2	0	304	3628	-87
ADS-3	0	304	3628	-87
ADS-4	0	489	3745	367
Core makeup tank end	0	70	3505	-51
IRWST start	0	663	3817	540

Note:

ACC – accumulator

rapidly. In addition, the ADS is actuated early since the core makeup tank on the broken DVI drains quickly, resulting in further depressurization and early IRWST injection. The other tests follow a somewhat difference course.

For the cold leg breaks, the loss of inventory results in a safety injection signal due to low pressurizer pressure. The timing of this signal depends on the size of the break. The safety signal results in reactor trip, isolation of the steam generators, and opening of the isolation valves that allow the core makeup tanks and PRHR HX to begin operation. Decay heat is removed through the PRHR HX to the IRWST, and the core makeup tanks inject makeup water into the DVI lines as hot water from the cold legs is circulated via the balance lines. When the inventory falls to the level of the cold legs, steam enters the balance lines and the core makeup tanks begin to drain. The ADS is actuated on a low core makeup tank level, and the reactor coolant system is depressurized in a controlled manner. For the larger of the cold leg breaks (5.08 cm), the safety signal is generated quickly and the reactor coolant system pressure transient is comparable to the double-ended DVI break. In addition, both trains of safety injection sources are available for these events. Fig. 10 shows the safety injection flow for the 5.08 cm cold leg break. There is a significant injection gap after the core makeup tanks empty and before the IRWST injection begins due to the assumed location of the failed ADS stage 4 valve.

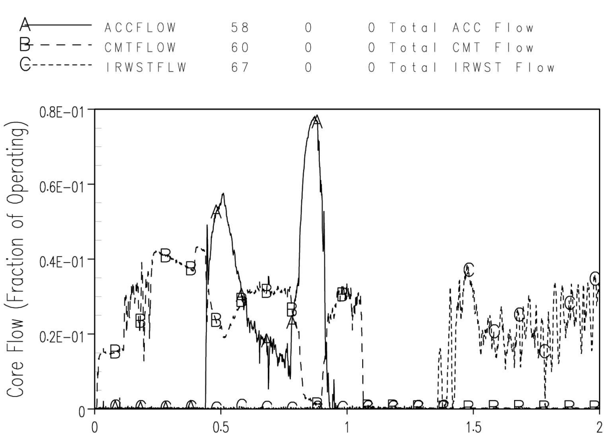

Fig. 10. Safety Injection Flow – 5.08 cm Cold Leg Break (Failed ADS-4 on Non-pressurizer Loop)

For the smaller cold leg break (1.27 cm), the depressurization is much slower and results in a significant delay in generating the safety signal. The core makeup tanks operate in the recirculation phase for a long time, and decay heat is removed by the PRHR HX. When the safety signal is first generated, insufficient heat is removed by the PRHR HX and the safety valves on the steam generator cycle to help remove the heat. The reactor coolant system pressure for this event remains relatively high for a long time since the temperature of the PRHR tubes must be high to effectively remove the decay heat. Finally, the reactor coolant system inventory is reduced, and the core makeup tanks drain, leading to ADS at a much later time. Once again, there is an injection gap between the emptying of the core makeup tanks and the beginning of IRWST injection as shown in Fig. 11.

For the inadvertent ADS case, the initial depressurization is very effective as steam is vented through the opened

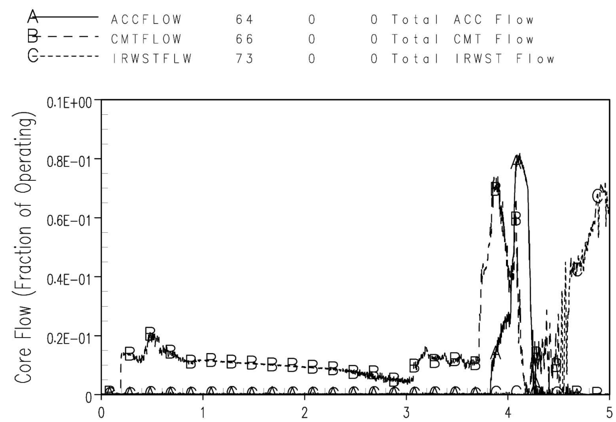

Fig. 11. Safety Injection Flow – 1.27 cm Cold Leg Break (Failed ADS-4 on Non-pressurizer Loop)

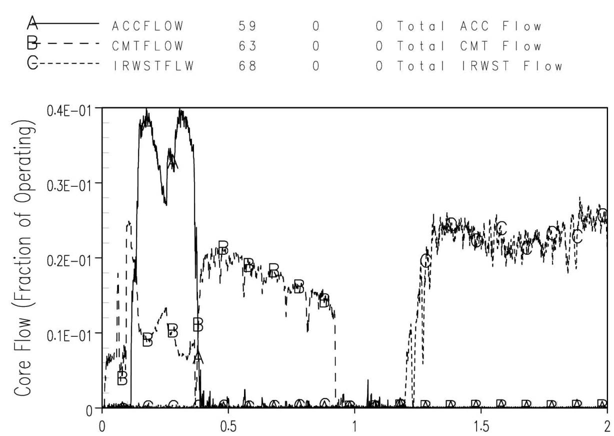

Fig. 12. Safety Injection Flow – Double-Ended DVI Break (Failed ADS-4 on Non-pressurizer Loop)

valve. The other valves open in sequence and the reactor coolant system pressure falls rapidly. Reactor coolant system inventory is generally higher for this scenario because the break is at the highest point in the reactor coolant system. The safety injection flow (Fig. 12) shows an injection gap between after the core makeup tanks empty due to the assumed single failure location.

Fig. 13 shows a comparison of the two-phase level in the reactor vessel for the four events. The minimum level occurs in the double-ended DVI simulation right after the initial depressurization. For each test, the level remains near the hot leg elevation with a large margin to core uncovery despite the gaps in injection flow.

Fig. 14 shows the integrated heat removed by the PRHR HX for the four events. The smallest break relies the most on the PRHR HX to remove decay heat. For all the tests, the PRHR becomes unimportant after the ADS is actuated.

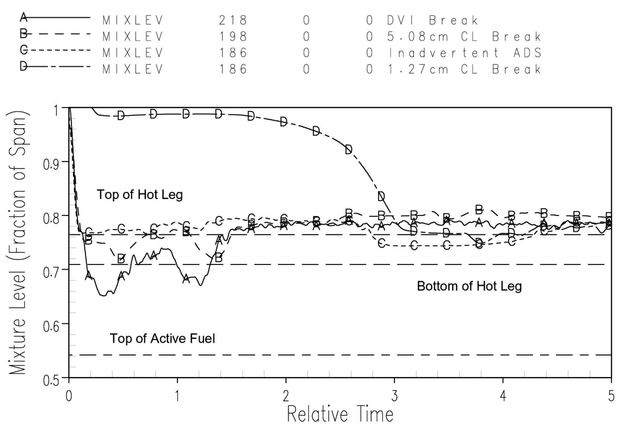

Fig. 13. Reactor Vessel Two-Phase Level – Break Spectrum

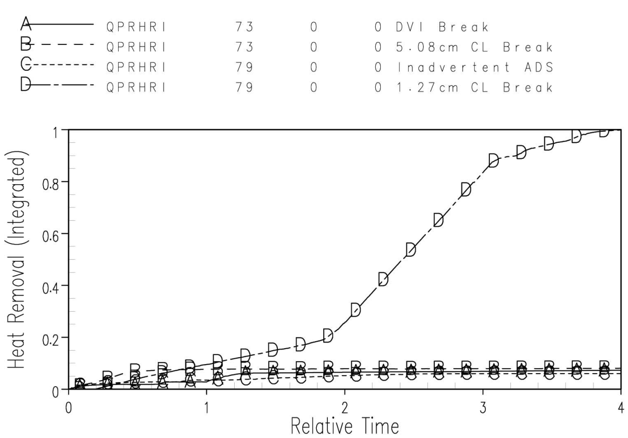

Fig. 14. Integrated Heat Removal by Passive Residual Heat Removal

7. CONCLUSIONS

The APEX-1000 simulations of the AP1000 design basis SBLOCA events have shown the following:

- The APEX-1000 facility adequately represents the important thermal-hydraulic phenomena expected in AP1000 accident analysis and is, therefore, representative of the expected AP1000 plant response to the simulated accidents. These tests can be used with confidence to validate AP1000 analysis computer codes.
- The AP1000 passive core cooling system provides excellent protection for the full spectrum of SBLOCA events. In all cases, the margin to core uncovery was maintained and fuel heatup did not occur. The two-phase level in the reactor vessel remains at or above the hot leg elevation.
- The location of the assumed single failure of one of the four ADS stage 4 valves was important when determining the passive core cooling system performance. Assuming the failure on the hot leg containing the pressurizer resulted in more rapid depressurization and earlier IRWST injection. Assuming the failure on the hot leg not containing the pressurizer resulted in interaction between the water draining from the pressurizer and the steam venting from the ADS stage 4 and less effective venting. These tests showed that the CMTs would empty before IRWST flow is established resulting in a gap in the injection flow. They also showed that the venting capacity increased as the void fraction in the upper plenum decreased, and that for all cases IRWST injection is established with large margins to core uncovery.
- Larger breaks result in earlier ADS actuation and rely less on the PRHR HX. Smaller breaks use a combination of the PRHR HX, CMT recirculation, and steam generator boiloff to remove decay heat after the steam generators are isolated. For all breaks, the PRHR HX becomes less important once the ADS is actuated.

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REFERENCES

[1] L.E. Hochreiter, et al., “AP600 Low Pressure Integral

System Tests at Oregon State University, Facility Description Report," WCAP-14136, Westinghouse Electric Company LLC, July 1994.
[2] R.F. Wright, et al., "AP600 Low Pressure Integral System Tests at Oregon State University, Test Analysis Report," WCAP-14293, Westinghouse Electric Company LLC, September 1995.
[3] R.F. Wright and J. Groome, "AP1000 Design Basis Event Simulation at the APEX-1000 Test Facility," Proceedings of ICAPP04, June 2004.