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TRAC-PF1 Code Verification With Data from the OTIS Test Facility

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Abstract

A computer code (TRAC-PF1/MOD1) developed for predicting transient thermal and hydraulic integral nuclear steam supply system (NSSS) response was benchmarked. Post-small break loss-of-coolant accident (LOCA) data from a scaled, experimental facility, designated the Once-Through Integral System (OTIS), were obtained for the Babcock & Wilcox NSSS and compared to TRAC predictions. The OTIS tests provided a challenging small break LOCA data set for TRAC verification. The major phases of a small break LOCA observed in the OTIS tests included pressurizer draining and loop saturation, intermittent reactor coolant system circulation, boiler-condenser mode, and the initial stages of refill. The TRAC code was successful in predicting OTIS loop conditions (system pressures and temperatures) after modification of the steam generator model. In particular, the code predicted both pool and auxiliary-feedwater initiated boiler-condenser mode heat transfer.

Introduction

A best-estimate, multi-dimensional, nonequilibrium, thermal-hydraulics computer code has been developed for the NRC at Los Alamos. The computer code is designed to predict small break LOCAs and other nuclear plant transients. The most recent version of the code, TRAC-PF1/MOD1, was used in this study. TRAC was used to predict overall integral system behavior during a small break LOCA transient in a test facility simulating the important features of a Babcock & Wilcox (B&W) designed nuclear steam supply system (NSSS). The Once-Through Integral System (OTIS) test facility was a single-loop facility with a plant to model power scale factor of 1686. The OTIS Program was cooperatively funded by the Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), B&W, and the B&W Owners Group (owners of the B&W NSSS).

The TRAC predictions are compared to data from two OTIS tests. The first test was the OTIS nominal test which provided a baseline for comparison of subsequent off-nominal OTIS tests. Test initialization and initiation were aimed at obtaining plant-similar conditions at the time of loop fluid saturation in a small break LOCA. The second test, the steam generator characteristics test, was initialized identically to the nominal test. Upon test initiation, the steam generator secondary control was altered from the nominal test. The two tests demonstrated the same mode of primary-to-secondary heat transfer; namely, steam generator condensation or boiler-condenser mode (BCM), although differing in the way in which this heat transfer was initiated. Results are presented from the TRAC predictions and from the two mentioned OTIS tests.

OTIS Facility Description

The OTIS facility was designed to provide a test facility for separate effect and integral system studies of the natural circulation phases of a small break LOCA. The facility was a single hot leg and a single cold leg (1 x 1 loop) representation of the B&W raised loop NSSS. The general arrangement of the major OTIS components and systems is shown in Figure 1. The loop consisted of one 19-tube once-through steam generator (OTSG), a simulated reactor, a pressurizer, a single hot leg, and a single cold leg. Reactor decay heat was simulated by electrical heaters in the reactor vessel capable of 180 KW (614,000 Btu/hr) or about 8% scaled power. Other loop components included a reactor vessel vent valve, pressurizer power-operated relief valve, a hot leg high-point vent, and a reactor vessel upper head vent. Auxiliary systems were available for scaled high pressure injection, controlled primary leaks in five loop locations, and a secondary forced circulation system for providing auxiliary feedwater to the steam generator.

The loop configuration was governed by scaling considerations. The major scaling criteria, in order of priority, were loop elevations, post-small break LOCA flow phenomena, component volumes, and irrecoverable pressure losses. Full elevation scaling was incorporated in OTIS to maintain representative scaled natural circulation flow rates. Natural circulation is important for core cooling during a small break LOCA in a B&W nuclear steam supply system.

The flow characteristics in the hot leg were of particular interest. During the course of a small break LOCA, the hot leg fluid may be subcooled liquid, saturated liquid, saturated steam, superheated steam, or a two-phase steam-liquid mixture. Froude number scaling was used to preserve the characteristic flow conditions in the OTIS hot leg. This scaling resulted in a 6.60 cm (2.6 inch) inside diameter hot leg that was twice the diameter obtained by purely volumetric scaling.

The reactor vessel, pressurizer, steam generator, and cold leg were predominantly volume scaled. The OTIS power and volume scale factor originated with the size of the model steam generator. The model steam generator contained 19 full-length and plant-typical tubes, which represented the 32,026 tubes in the two steam generators in the raised loop nuclear steam supply system. Therefore, the power and volume scale factor for OTIS was 1/1686. The OTIS primary loop volume was 0.286 cubic meters (10.1 cubic feet) compared to an ideally scaled volume of 0.241 cubic meters (8.5 cubic feet). The excess loop volume over the ideally scaled volume resulted predominantly from the larger hot leg diameter.

Prototypical loop irrecoverable resistance was maintained to preserve the correct scaled, natural circulation flow rates. The primary piping irrecoverable pressure loss was low; therefore, a fixed plate orifice was installed in the cold leg pipe at the outlet of the steam generator, and another near the bottom of the reactor vessel downcomer. These orifices provided prototypical loop resistance in the overall loop as well as in the subloop involving the reactor, reactor vessel vent valve, and the downcomer.

An inherent atypicality experienced in scaled test facilities is higher piping heat losses than is experienced in the plant. The higher model heat losses result from the larger piping surface area to enclosed fluid volume ratio and instrument penetrations in the model. Active and passive insulation were used on the hot leg, pressurizer, and reactor vessel upper head to minimize the impact of the higher heat losses. The rest of the loop was covered with passive insulation.

OTIS had approximately 250 instruments to measure the thermal and hydraulic conditions. These instruments were interfaced to a computer-controlled, high-speed data acquisition system that provided the capability to record all instrument readings in 25 milliseconds as often as every 5 seconds. The OTIS instrumentation provided:

- Pressure and differential pressure measurements,
- Thermocouple and resistance temperature detector measurements of fluid and metal temperatures,
- Fluid level and phase indication by optical viewports, fluid conductivity probes, and differential pressures, and
- Pitot tube, head flowmeters, and turbine meters for flow rate measurements.

Mass fluxes into and out of the facility were controlled and measured at the facility boundaries.

Code Description and OTIS Nodal Representation

A version of the Transient Reactor Analysis Code (TRAC) denoted TRAC-PF1/MOD1 is a best-estimate, multi-dimensional, nonequilibrium, thermal-hydraulics computer code developed for the NRC at Los Alamos. This code has a full two-fluid model in all the one-dimensional components. The two-fluid model, in conjunction with a stratified-flow regime model, improves handling of countercurrent flow as compared to the drift-flux model used in the previous TRAC versions. Also, significant improvements have been made to the constitutive relationships, heat transfer models, the trip logic, and the input procedures.

The OTIS test facility was modeled with TRAC one-dimensional components since all OTIS components have large length-to-diameter ratios. OTIS was modeled with 26 TRAC components that have been subdivided into 148 computational cells. A TRAC component schematic of the OTIS test facility is shown in Figure 2.

The once-through steam generator was noded coarsely for these calculations to decrease computational costs. The steam generator model uses parallel flow channels to simulate the flow of primary fluid through the 19 tubes of the steam generator. Parallel channels allow one to separate the tubes that are wetted and unwetted when using the minimum-wetting auxiliary feedwater nozzle.

The TRAC model incorporated the following provisions to simulate the OTIS tests described in this paper:

- Adiabatic boundaries at the exterior walls of the components that were guard heated,
- Primary system leak located at the bottom of the cold leg pump suction piping,
- High pressure injection located in the cold leg downstream of the simulated pump discharge region,
- Steam generator auxiliary feedwater injected just below the upper tubesheet into the secondary side,
- Reactor vessel vent valve located between the reactor vessel upper head and upper downcomer was fully operational during the transient,
- Orifice plates in the cold leg pump suction pipe and downcomer were modeled with appropriate loss coefficients,
- High point vent installed at the top of the hot leg U-bend, and
- Level controller combined with auxiliary feedwater injection controlled the steam generator secondary liquid level at prespecified setpoints.

The initial and boundary conditions were specified with TRAC fill and break components. Break components were used to specify the steam generator secondary pressure and the atmospheric boundary for the cold leg suction leak. The steam generator auxiliary feedwater flow and the cold leg high pressure injection flow were controlled by fill components.

Test Descriptions

Two OTIS tests were examined in detail for TRAC code verification. These include the OTIS nominal test, Test 220100, and the steam generator characteristics test, Test 220402. The initial conditions and operator actions were duplicated in both tests, however, the steam generator secondary control was varied. The initial loop conditions for the tests were defined to simulate plant conditions at approximately 1-1/2 minutes after a reactor trip and assumed that the reactor coolant pump had coasted down and the boundary systems (high pressure injection and auxiliary feedwater) were available and activated. The resulting OTIS initial conditions were 3.7% scaled core power (where 1% power is 21.4 KW, 73,000 BTU/hr), reactor coolant system in subcooled natural circulation, primary pressure at 15.17 MPa (2200 psia), and the pressurizer level set at 3.05 m (10 feet). The secondary side was initialized with a 1.39 m (4.6 feet) level in the steam generator, with AFW injected at the upper elevation at approximately 310.9K (100°F), and with secondary steam pressure controlled at 8.15 MPa (1182 psia) to obtain primary hot leg fluid temperatures of approximately 594K (610°F). During loop initialization, high pressure injection and leaks were not used.

Initiation of these tests was performed in two steps. These steps were defined to obtain plant similar conditions at the time of loop fluid saturation. The first step was performed by opening the designated leak and the second was performed at the point of pressurizer inventory depletion. When the pressurizer liquid height reached approximately 0.61 m (2 feet), high pressure injection was activated to the cold leg discharge piping, a core power ramp simulating decay power from 1-1/2 minutes was initiated, and the auxiliary feedwater control characteristics were changed. For Test 220100, the auxiliary feedwater was injected to obtain a 0.91 m (3 feet) per minute level increase until a 11.6 m (38 feet) control level was reached. For Test 220402, auxiliary feedwater was injected at full capacity until a secondary control level of 3.20 m (10.5 feet) was reached. In both tests, when the secondary steam generator level reached the control level, the operator initiated a secondary depressurization from approximately 6.90 MPa (1000 psia) to obtain a 28K (50°F) per hour secondary cooldown.

Four small break LOCA phases were predicted and observed during these OTIS tests. These included pressurizer draining and loop saturation, intermittent reactor coolant system circulation, boiler-condenser mode, and the initial stages of refill. For Test 220402, the lower steam generator secondary level resulted in a later occurrence of the boiler-condenser mode of energy transfer. In Test 220402, the occurrence of the boiler-condenser mode with definite lack of a condensing surface in the steam generator pool region confirmed the high auxiliary feedwater boiler-condenser mode of heat transfer. This pair of OTIS tests provides a challenging exercise of the TRAC predictive capabilities.

The TRAC calculations were performed in a manner similar to the actual test procedures. A steady-state calculation was completed to establish the desired steady-state conditions during the natural circulation mode. The transient calculation was initiated from the final steady-state conditions and proceeded in a systematic fashion. The initial conditions for the two OTIS calculations presented in this paper are identical, thus, a single TRAC steady-state calculation was performed to obtain the proper initial conditions for the transient predictions. A 1000 second calculation was completed to ensure steady-state conditions were established.

Comparisons

In this section, a description and comparison is presented for each of the observed transient phases for Tests 220100 and 220402. Both tests experienced pressurizer draining and loop saturation, intermittent circulation and interruption, boiler-condenser mode, and the initial stages of refill. The TRAC predictions will be presented for primary and secondary pressures for each of the tests.

Test 220100

A comparison of calculated and measured test events for Test 220100 is listed in Table 1. In general, the correspondence between the calculated and measured values is very good. The primary system response to many of the events given in Table 1 is best illustrated by the behavior of the primary system pressure.

The calculated primary system pressure is compared with the experimental data in Figure 3 and is seen to compare quite well. The more important events have been annotated on the figure as well as the calculated steam generator secondary pressure. The opening of the leak valve in the cold leg suction piping resulted in the rapid primary system depressurization shown in Figure 3. At 228 seconds, the depressurization rate decreased sharply when the fluid in the hot leg became saturated. Prior to the saturation of the primary fluid, the core power decay ramp, cold leg high pressure injection flow, and auxiliary feedwater flow were initiated at 150 seconds. The sharp drop in the secondary side pressure was caused by continuous flow of auxiliary feedwater during the filling of the steam generator which condensed a significant amount of steam on the secondary side. During this period, the small pressure oscillations in the primary side were condensation induced, resulting from the interaction of high pressure injection fluid in the cold leg and steam flowing through the reactor vessel vent valve from the reactor vessel upper plenum. This was followed by a larger pressure increase when the primary flow was interrupted and the energy removal rate through the leak and steam generator was less than the energy added to the primary via core power and high pressure injection fluid.

The primary pressure remained fairly constant near 10.1 MPa (1465 psia) from 1620 to 2580 seconds as the primary system energy removal and addition rates were nearly equal. The sudden decrease in primary pressure beginning at 2580 seconds corresponded to the initiation of steam generator pool boiler-condenser mode of primary-to-secondary heat transfer. This increased heat transfer occurs as the steam generator primary level drops below the secondary pool, thus, exposing surface area for primary steam condensation. The high rate of energy removal from the primary to the secondary side of the steam generator resulting from the pool boiler-condenser mode phenomena lasted for approximately 840 seconds and resulted in a primary pressure decrease of about 5.2 MPa (750 psia). At the end of boiler-condenser mode, the primary pressure became coupled to the secondary pressure which was near 5.2 MPa (750 psia) and remained coupled for the duration of the calculation. The refill of the primary system began at 3300 seconds as the high pressure injection flow rate exceeded leak flow rate. The pool boiler-condenser mode weakened as the steam generator primary level increased. This weakening resulted in a significant decrease in primary system depressurization.

Test 220402

The steam generator characteristics test, Test 220402, was used to characterize the effects of the steam generator secondary level and level control on post-small break LOCA transients. This test started from the same initial conditions used for Test 220100. The core power decay, high pressure injection flow, leak size, leak location, and steam generator secondary pressure control were also the same as those used in Test 220100.

Upon test initiation, the secondary level was increased from 1.52 m (5.0 feet) to 3.20 m (10.5 feet) at full auxiliary feedwater capacity. After the secondary level reached the 3.20 m (10.5 feet) elevation, the facility level controller was used to maintain the level at 3.05 m (10.0 feet). But during the test, the level controller malfunctioned and the operator was forced to intervene and manually control the secondary level. In order to compensate for the resulting level variations, the measured level was used as a boundary condition for the TRAC calculation.

The predicted results for Test 220402 do not compare with the data as well as did those for Test 220100 because of imprecise modeling of the auxiliary feedwater flow and liquid level control to obtain the desired level. Table 2 shows a comparison of calculated and measured occurrences of certain key events. The comparison between the calculated and measured times is satisfactory for the first 2000 seconds of the transient. The major difference was the time high-elevation auxiliary feedwater induced boiler-condenser mode heat transfer begins, i.e., 3000 seconds predicted by TRAC and 3940 seconds indicated by the experimental data. For auxiliary feedwater induced boiler-condensor mode, steam generator primary steam condensation occurs due to injection of colder auxiliary feedwater near the top of the steam generator rather than being caused by a primary level below the secondary pool as in Test 220100.

The primary and secondary pressures are compared in Figure 4. The early commencement of predicted boiler-condenser mode and the change in the depressurization rate at 4300 seconds are shown in the figure. Otherwise, the agreement between the calculated and measured primary pressure was very good. The effects of the high-elevation auxiliary feedwater injection was significant in this test because of the increased condensing area that resulted from a low primary-side collapsed liquid level. Therefore, the timing and rate of auxiliary feewater injection were very important parameters in accurately predicting the high-elevation auxiliary feedwater boiler-condenser mode phenomena.

Conclusions

The TRAC-PF1/MOD1 post-test predictions of OTIS Tests 220100 and 220402 have shown that TRAC can simulate with reasonable accuracy the major phases of a small break LOCA as observed in these two tests. The draining of the pressurizer and loop saturation, interruption of natural circulation, primary pressure oscillations caused by steam/water mixing in the cold leg, pool and high-elevation auxiliary feedwater boiler-condenser mode, and the initial stages of primary refill were successfully predicted. The ability to adequately predict the mass distribution in the system was primarily responsible for the overall good predictions of the phenomena observed in the tests.

OTIS Tests 220100 and 220402 were important tests for TRAC verification purposes as these tests provided two different sets of boiler-condenser mode heat transfer data that are expected to be applicable for analysis of Babcock & Wilcox reactor systems. Successful simulation of these tests have given confidence in using TRAC as an analytic tool to investigate the thermal-hydraulic behavior of Babcock & Wilcox type test facilities and reactor systems.

Table 1
Comparison of Key Test Events for OTIS Test 220100

Event	Time (Seconds) Calculated	Time (Seconds) Measured
Cold leg suction leak initiated	0	0
Commence core power decay ramp, high pressure injection flow and auxiliary feedwater flow	150	151
Hot leg U-bend fluid saturated	228	205
Initial primary flow interruption	220	212
Steam generator secondary on level control	852	785
High pressure injection fluid reaches the cold leg leak region	1080	800
Complete interruption of natural circulation	1441	1030
Beginning of pool boiler-condenser mode	2580	2540
Refill of primary system begins	3300	3200
Pool boiler-condenser mode terminated	4020	3980

Table 2
Comparison of Key Test Events for OTIS Test 220402

Event	Time (Seconds) Calculated	Time (Seconds) Measured
Cold leg suction leak initiated	0	0
Commence core power decay ramp, high pressure injection flow and auxiliary feedwater flow	150	115
Hot leg U-bend fluid saturated	204	185
Steam generator secondary on band control	348	324
Initial primary flow interruption	206	204
Complete interruption of natural circulation	810	715
Beginning of high-elevation auxiliary feedwater boiler-condenser mode	3000	3940
Refill of primary system begins	4878	4578

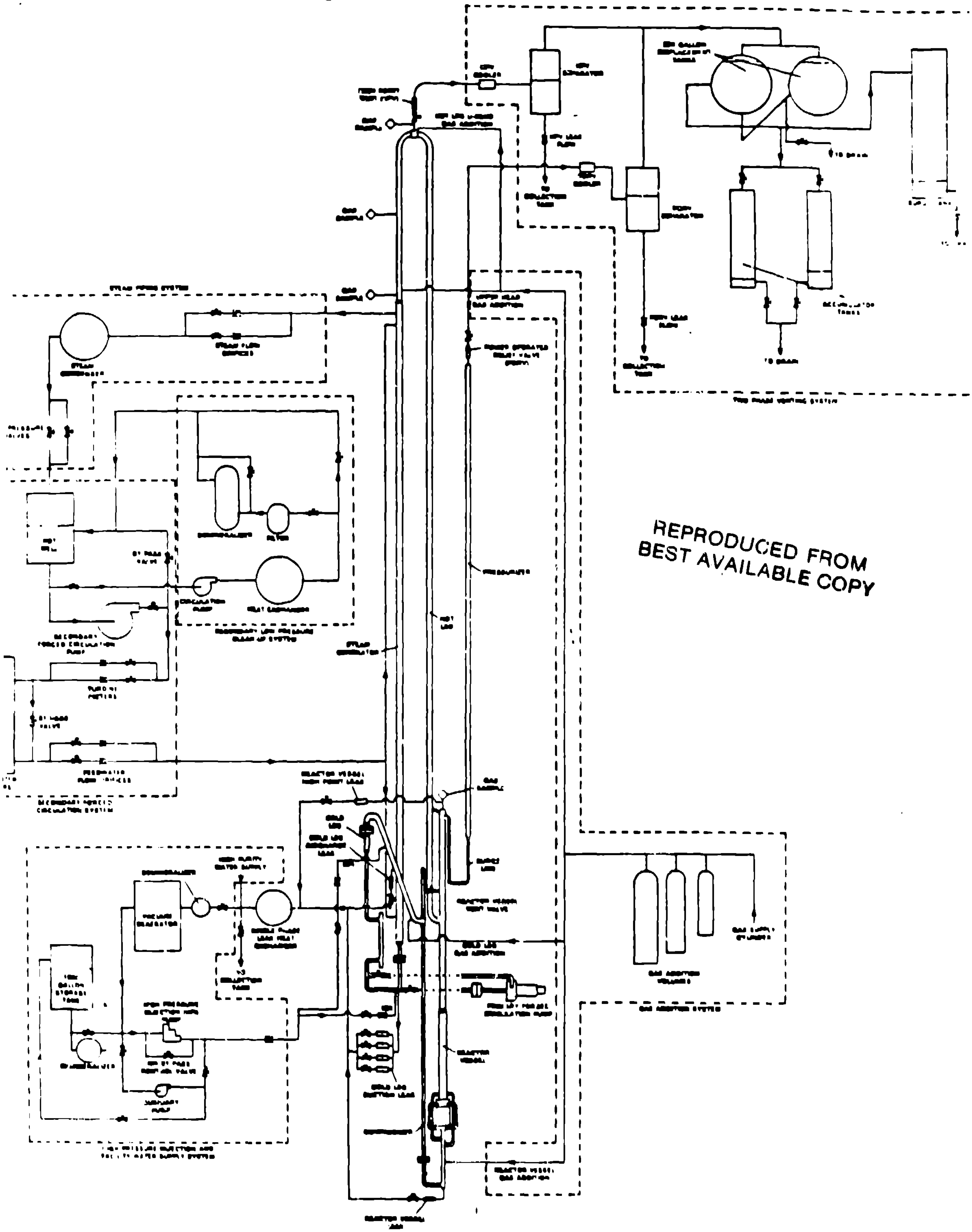

Figure 1. OTIS Test Facility

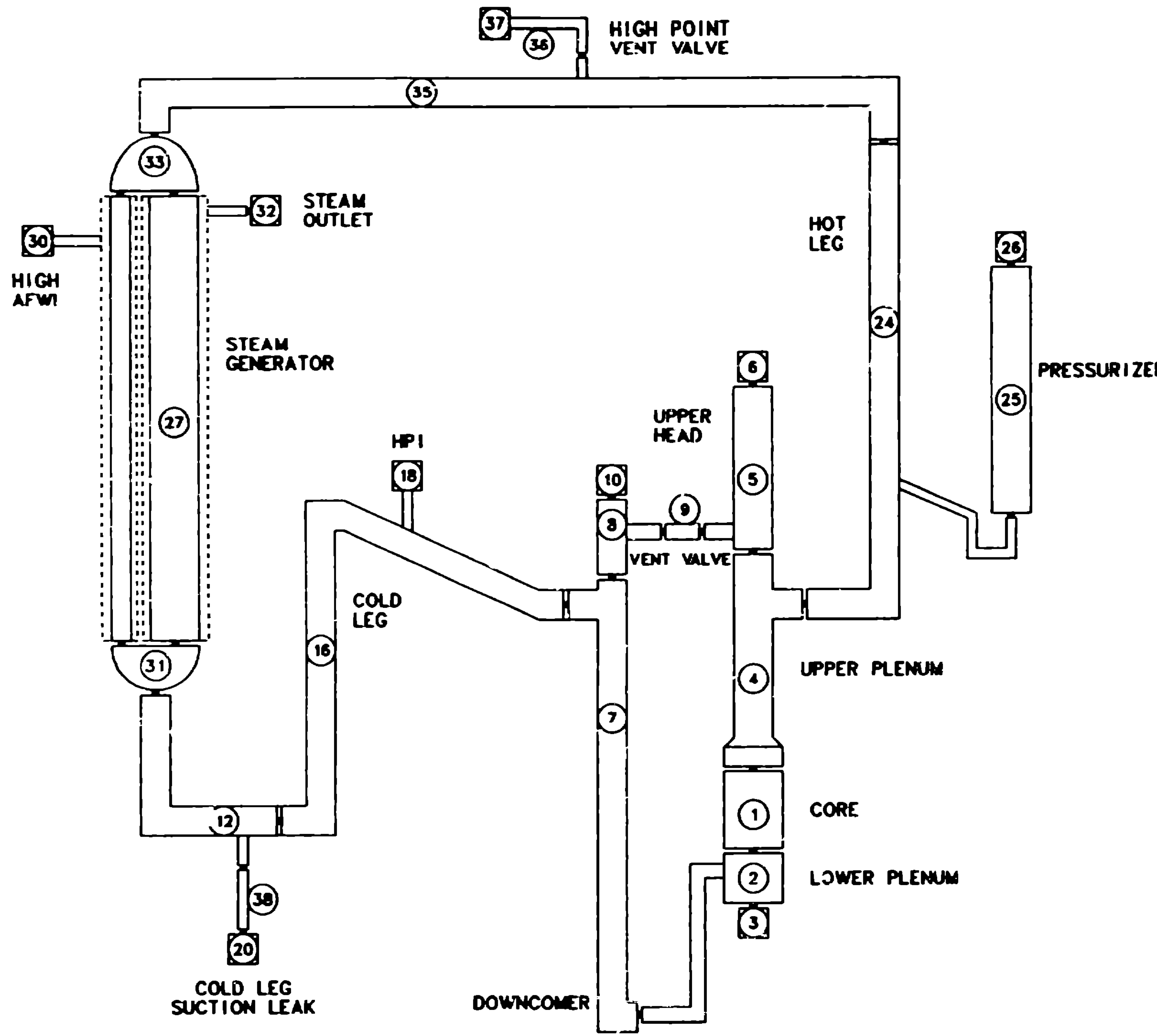

Figure 2. OTIS Nodal Representation

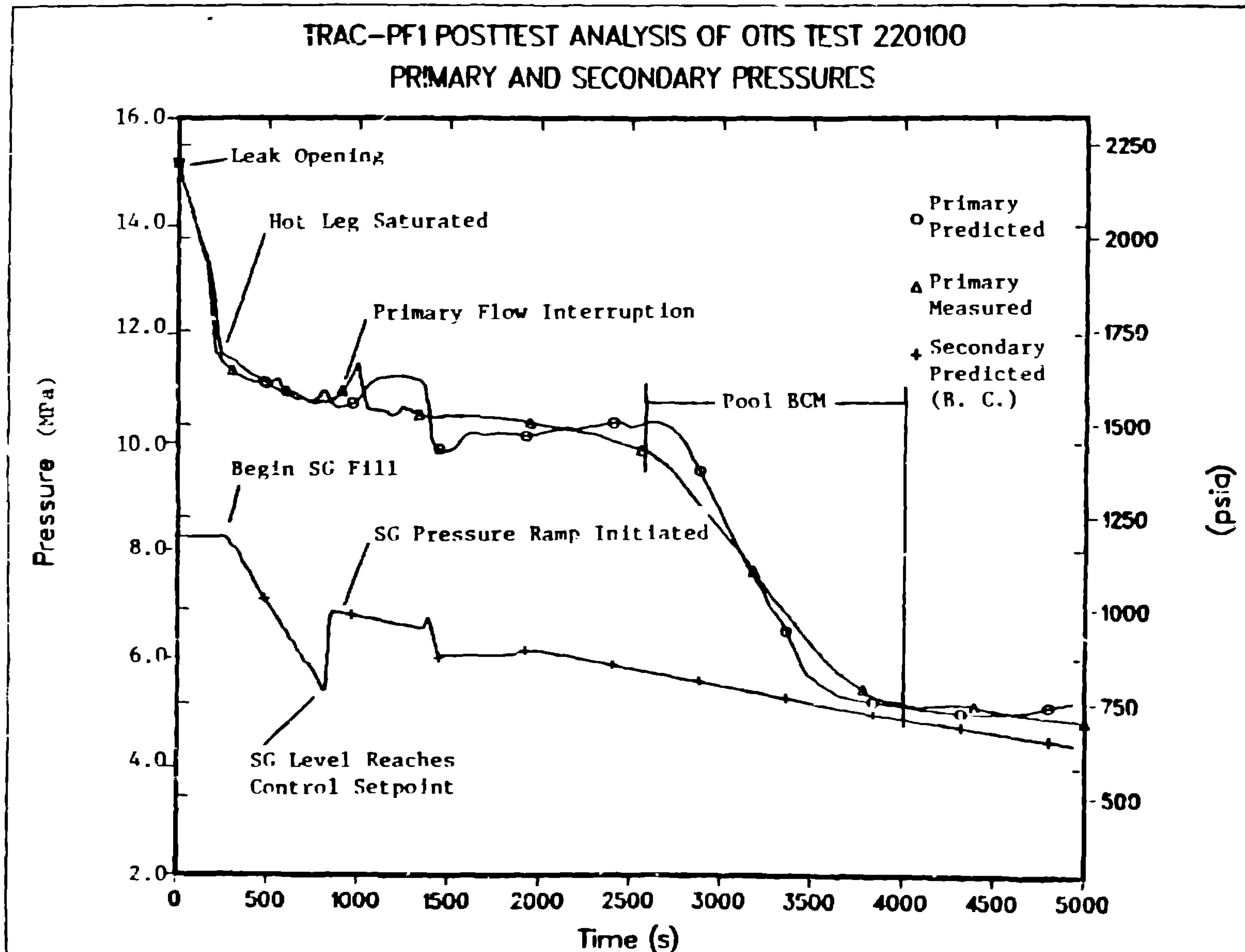

Figure 3. Primary and Secondary Pressure Comparisons for Test 220100

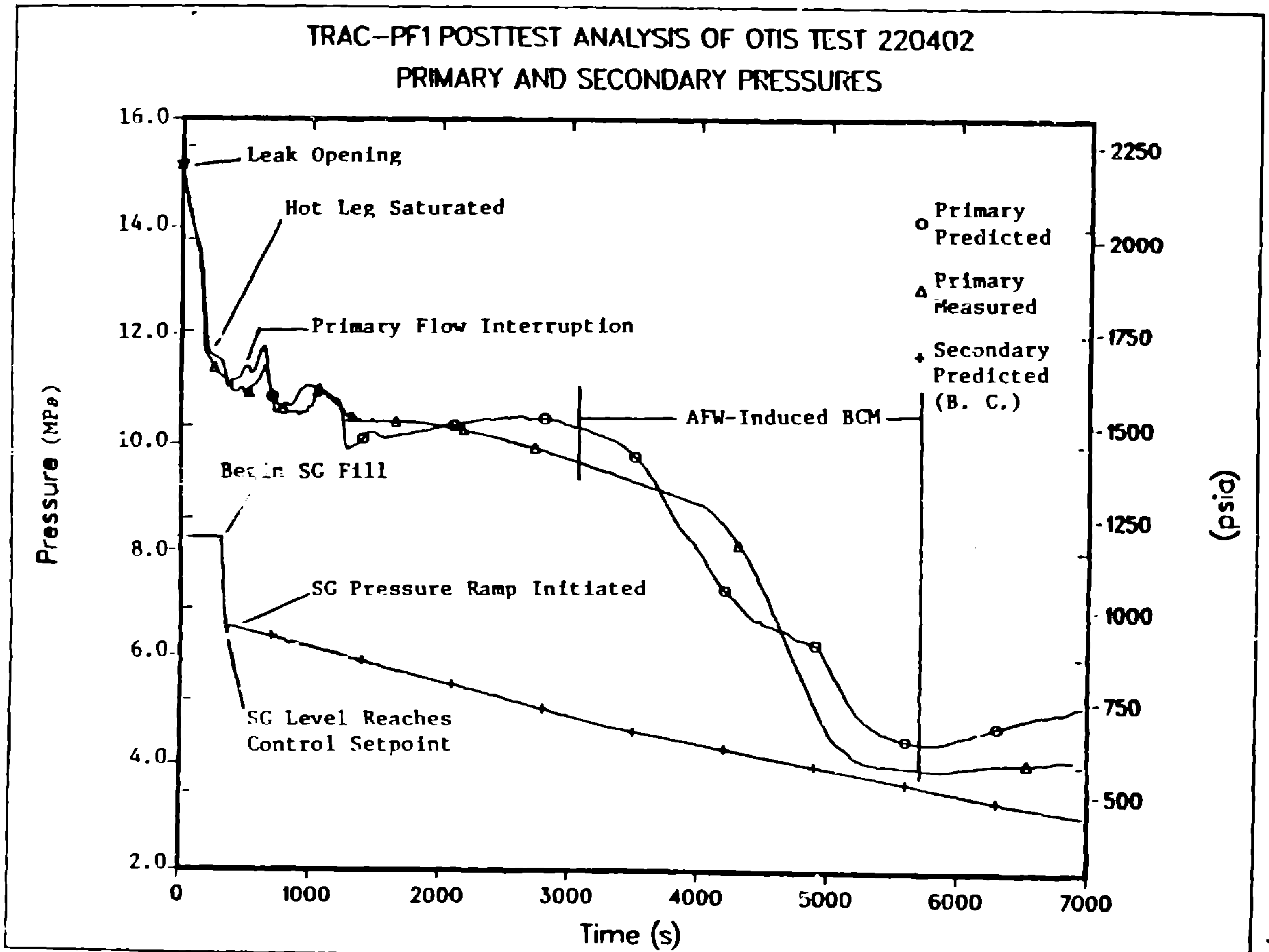

Figure 4. Primary and Secondary Pressure Comparison for Test 220402