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# ASSESSMENT OF TRAC-BD1/MOD1 USING FIST DATA*

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The TRAC-BD1/MOD1 code [1], developed at Idaho National Engineering Laboratory (INEL), has been assessed at Brookhaven National Laboratory (BNL) using some of the FIST Phase I experiments in support of the United States Nuclear Regulatory Commission's independent code assessment program. The TRAC-BD1/MOD1 code is an advanced best-estimate system code developed primarily to analyze postulated accidents and transients in BWR systems.

The FIST (Full Integral Simulation Test) facility [2] is a BWR safety test facility which was built to investigate small break LOCA and operational transients in BWRs and to complement earlier large break LOCA test results from TLTA (Two-Loop Test Apparatus). Particular attention was paid to eliminate the scaling compromises found in some of the earlier BWR test facilities. The FIST program is sponsored jointly by the NRC, Electric Power Research Institute (EPRI) and General Electric Company (GE).

The facility, as shown in Figure 1, is a full BWR height, integral test facility with volume scaling of 1/624 to the BWR/6 vessel and contains a single full-size BWR fuel bundle (electrically heated). It has all the prototypical components of a BWR/6. The flow areas and the fluid volumes in all regions are also closely scaled to 1/624. However, because of scaling difficulty, it has a cylindrical external downcomer connected to the main vessel. The test facility is capable of simulating large, intermediate and small break accidents, as well as many operational transients.

The FIST test program consists of two phases: the Phase I tests [3] were completed in 1983, and the Phase II tests in early 1985. In this study the TRAC-BD1/MOD1 code has been assessed with the Phase I tests. The FIST Phase I consists of eight matrix tests and two additional tests as listed in Table 1. Five of these tests were selected to be simulated. These were: a BWR/4 MSIV closure ATWS (Test 4PMC1), a BWR/6 small break LOCA without HPCS (6SB2C), a BWR/6 large break LOCA (6DBA1B), a BWR/6 small break LOCA with stuck open SRV (6SB1), and a BWR/6 main steam line break test (6MSB1). Simulations of four of the selected transients were completed, but simulation of Test 6SB1 was discontinued because of numerical difficulty.



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The "VESSEL" component of TRAC-BD1/MOD1 was used to represent the FIST facility. The VESSEL was nodalized with 12 axial levels, 2 radial rings, and 2 azimuthal sectors as shown in Figure 2.

Test 4PMC1 was a power transient simulation test for a BWR/4 with MSIV closure and without power scram. The transient calculation of this test was terminated at 400 seconds, since all the significant events occurred during this period and the rest of the transient was predictable. Table 2 compares the timing of the key events of the test and the calculation, which are generally in good agreement. However, there was a slight delay in some events in the calculation; the timing of reaching "Level 2" and, therefore, injection of RCIC and HPCI were delayed by 10 seconds. However, this delay did not appear to affect the transient significantly. Figure 3 compares the steam line flows of the test and calculation. The magnitude of the steam flow and the timing of the SRV openings and closings from the code calculations matched closely with the test results. The code predicted the peak pressure (Figure 4), and the magnitude and frequency of the pressure oscillation adequately (Figure 5). The collapsed downcomer water level was also well predicted by the code (Figure 6). Figures 7 and 8 compare the bundle void fraction of the test and the calculation at two different locations. They indicate that the code generally predicts a higher void fraction than the test void fraction. This may affect the core power if the power is calculated by neutron kinetics. The core was always covered and no rod heatup was observed in either the calculation or in the test.

Test 6SB2C was a small break test, simulating a BWR/6 recirculation line break of 0.05 $ft^2$. The High Pressure Core Spray (HPCS) was assumed to be unavailable. The MSIV was tripped when the downcomer water level reached "Level 1" and the Automatic Depressurization System (ADS) was activated with a 120 second delay. This test was also simulated by GE using TRACB02 (GE version of TRAC-BD1) [4]. Table 3 and Figures 9 through 12 compare the results of TRAC-BD1/MOD1 calculations with the test data and the results of GE calculations for Timing of Events, Pressure, Core Inlet Flow, Downcomer Water Level and Rod Temperature, respectively. They generally showed good agreement. However, in the calculation, Level 1 was reached about 10 seconds later and the depressurization after ADS activation was slightly slower than in the test. This resulted in about a 30-second delay in the initiation of Low Pressure Core Spray (LPCS) and Low Pressure Core Injection (LPCI). This, in turn, resulted in a delay of the rod heatup (Figure 12). However, the magnitude and duration of the heatup were well predicted by the code.

Test 6DBA1B was a large break test with a 200% recirculation line break for a BWR/6. Additionally, two LPCI pumps were assumed to fail. GE also simulated this test using TRACB02 and some of the available results of this calculation are compared with the test data and the results of the BNL calculation. In general, the code predicted the test results very well, as shown in Table 4 (Timing of Events) and Figures 13 (System Pressure), 14 (Core Inlet Flow), 15 (Intact Loop Jet Pump Flow), 16 (Broken Loop Jet Pump Flow), 17 (Intact Loop Jet Pump Mass), and 18 (Broken Loop Jet Pump Mass). However, it did not predict the bundle heatup as well, as shown in Figure 19. While the test results showed mild bundle heatup between 40 and 120 seconds, the code showed a shorter heatup period for both BNL (TRAC-BD1/MOD1) and GE (TRACB02) calculations. The magnitude of the bundle heatup in the test, however, was adequately predicted by the TRAC-BD1 calculation.

Test 6MSB1 was a main steamline break test, simulating a BWR/6 response with a double-ended break at the upstream of the flow limiter in one of the four main steamlines. This test was initially simulated using the correct break area as given in the test report. However, the calculation resulted in much larger break flow than in the test. Therefore, the calculation was repeated with reduced break area to match the break flow. Figure 20 compares the test break flow and the break flow calculated with the break area half of that of the test. Even with the reduced break flow, the calculated break flow was still substantially higher than the break flow in the test; yet the pressure did not decrease as fast in the calculation as in the test, as shown in Figure 21. This indicates that if the break flow was further reduced to match the test data, the pressure would be even higher than in the test. Since the pressure and mass inventory in the system are among the most important parameters determining other behavior in the reactor, the calculation was terminated at this point without further trials reducing the area. It appears that this inconsistent reactor behavior was caused by the faster increase of downcomer water level in the calculation than in the test due to the level swelling phenomenon and more liquid entrained through the break.

It appeared that the TRAC-BD1/MOD1 code adequately predicted the large and small break tests, and the MSIV closure ATWS test. However, it overpredicted the break flow in the main steamline break test. Furthermore, the code did not appear to be completely robust numerically as manifested by occasional failures and the need for restarting with small time steps. The code also needed some manipulation for geometric data such as cell length, area and/or hydraulic diameter around the "VALVE" components, which were used to simulate breaks and SRVs, to avoid taking excessively small time steps due to the material Courant limit. This difficulty was caused by the semi-implicit numerical scheme used in the code and is expected to be eliminated in the new code version (TRAC-BF1) with a SETS (Stability Enhancing Two Step) numerical method.

## REFERENCES


1. Taylor, D.D., et al., "TRAC-BD1/MOD1: An Advanced Computer Program for Boiling Water Reactor Transient Analysis," NUREG/CR-3633, April 1984.

2. Stephens, A.G., "BWR Full Integral Simulation Test Program," Contract No. NRC-4-76-215, NUREG/CR-2576, December 1982.

3. Hwang, W.S., et al., "BWR Full Integral Simulation Test (FIST) Phase I Test Results," Contract No. NRC-4-76-215, NUREG/CR-3711, November 1983.

4. Sutherland, W.A. and Alamgir, Md., "BWR Full Integral Simulation Test (FIST) Pretest Predictions With TRACB02," Proceedings of the Eleventh Water Reactor Safety Research Information Meeting, NUREG/CP-0048-Vol. 1, January 1984.

TABLE 1

FIST TESTS (PHASE 1)

| Test Number | Description | Initial Power | Available ECCS | Highlights |
|---|---|---|---|---|
| 6DBA1B | BWR/6 DBA | 5.05 | HPCS, LPCS LPCI (1) | core inventory depletion reduced due to full height jet pump, CCFL, PCT=710°F, reflood affected by stored heat. |
| 6SB2C | SB, W/O HPCS | 5.05 | LPCS, LPCI (3) | PCT = 925°F |
| 6SB1 | SB, STUCK SRV | 4.64 | LPCS, LPCI (3) | Responses similar to 6SB2C, PCT = 720°F. |
| 6MSB1 | MS LINE BREAK | 4.64 | HPCS, LPCS, LPCI (1) | CCFL, no core uncovery, no heatup |
| 6PNC1-1A<br>1-2B<br>1-3<br>1-4<br>1-5<br>1-6<br>1-7A | Natural Circ. | 0.5<br>1.0<br>1.5<br>2.0<br>2.5<br>3.0<br>2.0+SUB | N/A | Natural circulation flow affected by power and water level. Internal circulation flow observed. Responses similar to BWR analysis. |
| 6PMC1 | BWR/6 MSIV Clos. | 4.64 | All | Responses similar to BWR analysis, no core uncovery, no heatup. |
| 6PMC2A | BWR/6 MSIV Clos. (w/o HPCS) | 4.64 | RCIC, LPCS, LPCI (3) | Responses similar to BWR analysis, no core uncovery, no heatup. |
| 4PMC1 | BWR/4 MSIV Clos. | 4.35 | All | Responses similar to BWR analysis, no core uncovery, no heatup. |
| *6PMC2 | Separate Effect BWR/6 MSIV Clos. (w/o HPCS, 6PMC1 power) | 4.64 | RCIC, LPCS, LPCI (3) | Upper bundle uncovered and heatup due to high power, test terminated by bundle protection. |
| *6SB2B | SB, w/o HPCS | 5.05 | LPCS, LPCI (3) | Small ADS size, power off by bundle protection at 340 sec., PCT = 950°F. |

*Not matrix tests. Data are available in INEL data bank.

TABLE 2

TEST 4PMC1 MAJOR EVENTS TIMING

| EVENT | TEST | TRAC-BD1/MOD1 |
|---|---|---|
| Start of Programmed Power | 0 | 0 * |
| MSIV Closure | 2 | 2 * |
| Pump Trip | 3 | 3 * |
| First Opening of SRV | 3 | 3.1 |
| Maximum Pressure in Vessel | 4 | 8.0 |
| Feedwater Termination | | |
| Hot | 5 | 5 * |
| Cold | 8 | 8 * |
| Opening of All 5 SRVs | 5 | 5.5 |
| SRV Setting Switched to Low/Low Setting | 10 | 10 |
| Closing of 5th SRV | 20 | 18 |
| Recirculation Loops Isolation | 20 | 20 * |
| Closing of 4th SRV | 23 | 22 |
| Level 2 | 29 | 39 |
| Closing of 3rd SRV | 26 | 27 |
| Closing of 2nd SRV | 52 | 62 |
| RCIC and HPCI Initiation | 49 | 59 |
| Minimum Level | 49 | 70 |
| Calculation Terminated | - | 400 |
| Test Terminated | 1640 | - |

*Boundary Condition

TABLE 3

FIST 6SB2C EVENTS

| EVENT | TEST | TRAC-BD1/MOD1 |
|---|---|---|
| Break Initiation | 0 | 0 * |
| Bundle Power Trip | 0 | 0 * |
| Jet Pump Trip | 0 | 0 * |
| Feedwater Trip | 0 | 0 * |
| Recirculation Loop Isolation | 20 | 20 * |
| Water Level Reached L1 | 75 | 80 |
| MSIV Closure | 77 | 82 |
| ADS Activation | 195 | 200 |
| Bundle Heatup Begins | 250 | 320 |
| Final Rod Rewet | 420 | 440 |
| LPCS Activation | 35 | 35 * |
| LPCS Injection | 310 | 340 |
| LPCI Activation | 35 | 35 * |
| LPCI Injection | 335 | 370 |

*Boundary Condition

TABLE 4

FIST 6DBA1B SEQUENCE OF EVENTS

| EVENT | TEST | TRAC-BD1/MOD1 |
|---|---|---|
| Blowdown Valves Open | 0.0 | 0.0 * |
| Bundle Power Decay Initiated | 0.1 | 0.1 |
| Bypass Flow Reverses | 1.0 | 4.5 |
| Jet Pump Suction Uncovers | 5.0 | 6.0 |
| Recirc. Suction Line Begins to Uncover | 8.0 | 7.0 |
| Lower Plenum Bulk Flashing | 11.5 | 12.0 |
| Guide Tube Flashing | 12.0 | 18.0 |
| Loop 1 Isolated | 13.0 | 13.0 * |
| HPCS Injection Begins | 27.0 | 27.0 * |
| LPCS, LPCI Activated | 35.0 | 35.0 * |
| LPCS Flow Begins | 64.0 | 65.0 |
| LPCI Flow Begins | 75.0 | 73.0 |
| Bypass/Guide Tube Region Begins to Refill | 115.0 | 120.0 |
| CCFL Break Down at Bypass Outlet | 115.0 | 100.0 |
| Bundle Begins to Refill | 125.0 | 110.0 |
| Bypass Region Refill | 125.0 | 140.0 |
| Bundle Reflood with Two-Phase Mixture | 125.0 | 130.0 |
| CCFL Breaks Down at Upper Tie Plate | 125.0 | 120.0 |

*Boundary Condition

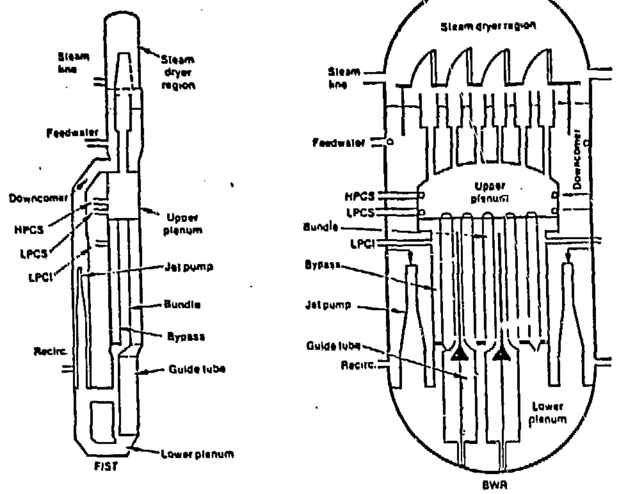


Figure 1. FIST and BWR Vessel Schematics.

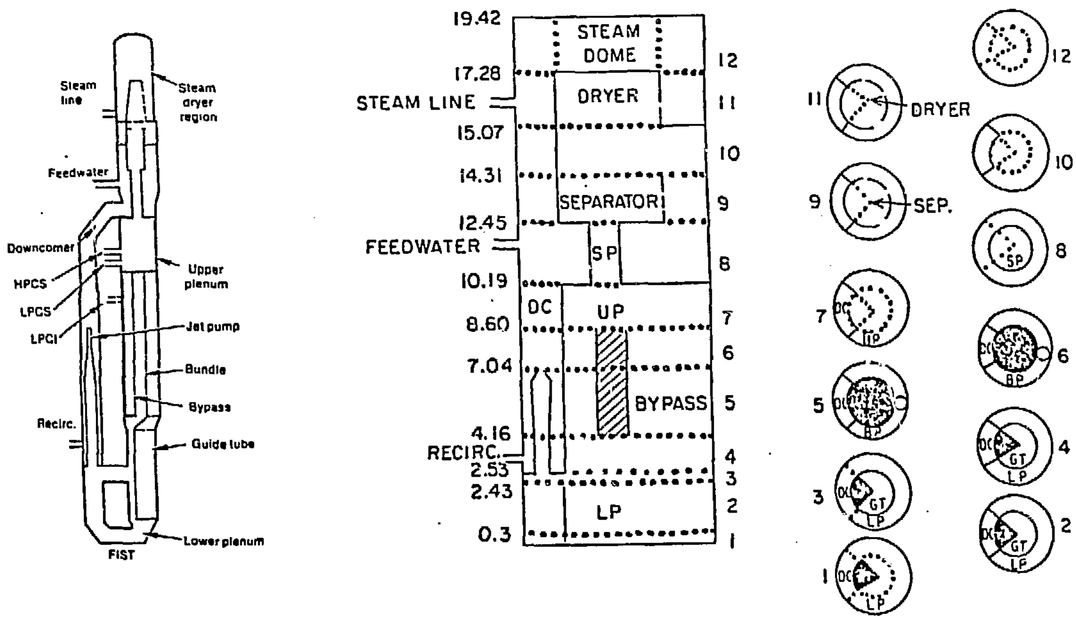


Figure 2. Nodalization of the FIST Vessel.

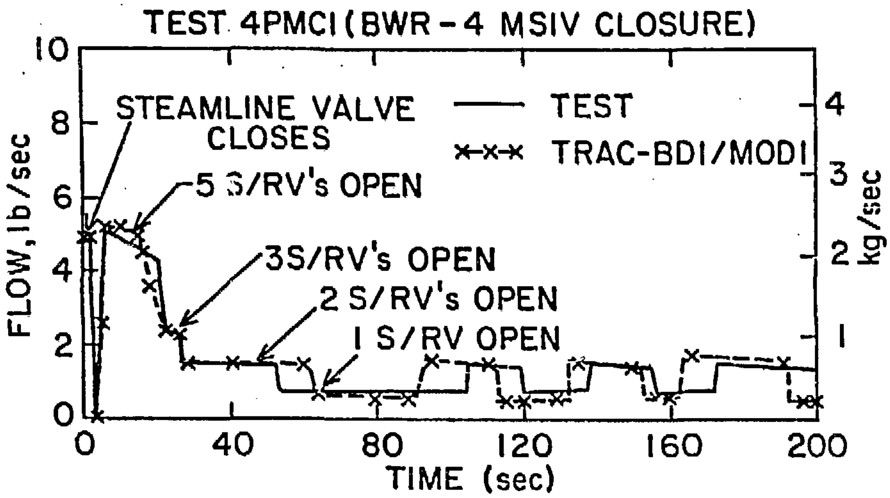


Figure 3. Steamline Flow.

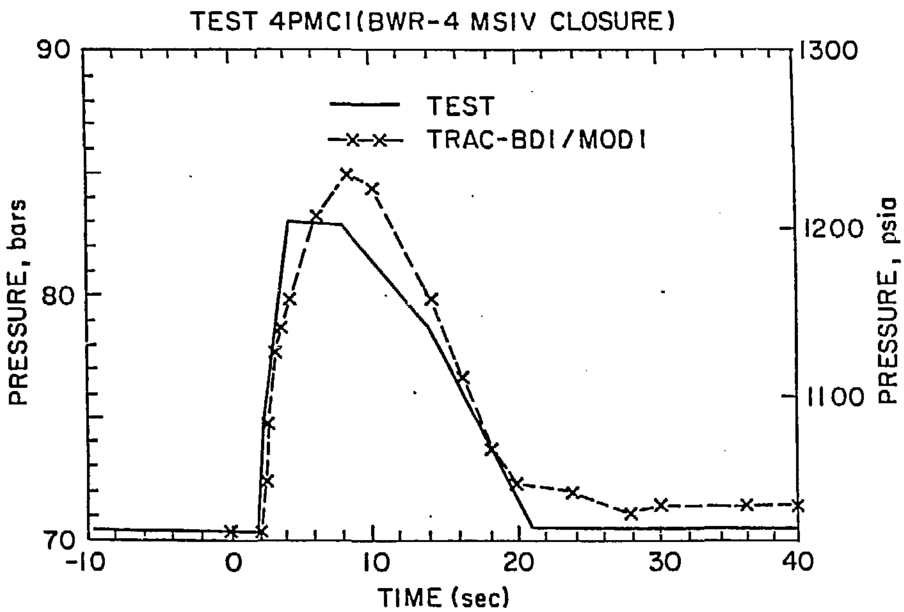


Figure 4. System Pressure.

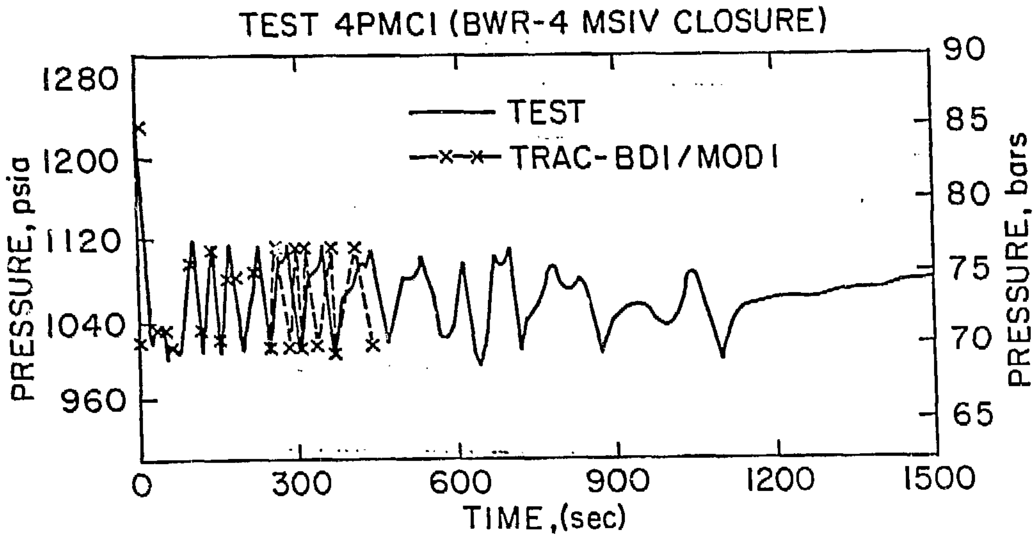


Figure 5. System Pressure.

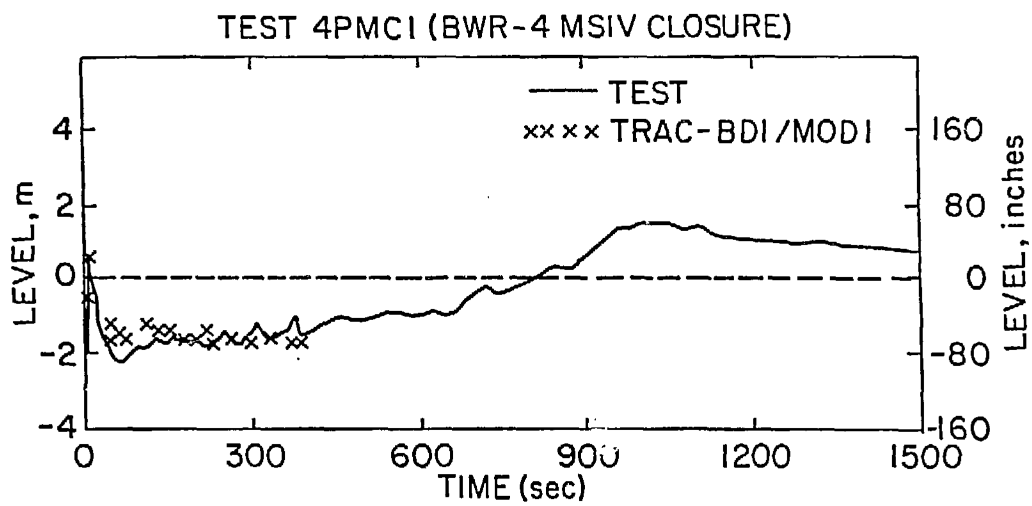


Figure 6. Downcomer Water Level.

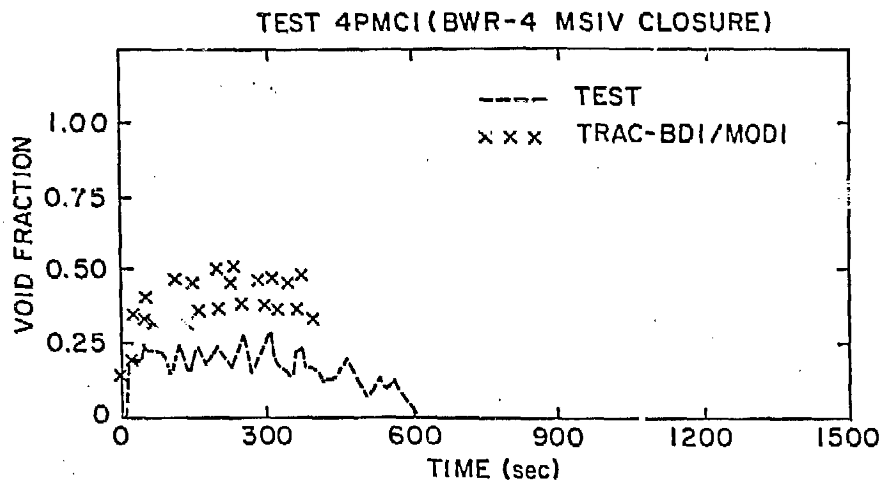


Figure 7. Bundle Nodal Void Fraction at Elev. 246" (37" Above the Bottom of the Bundle)

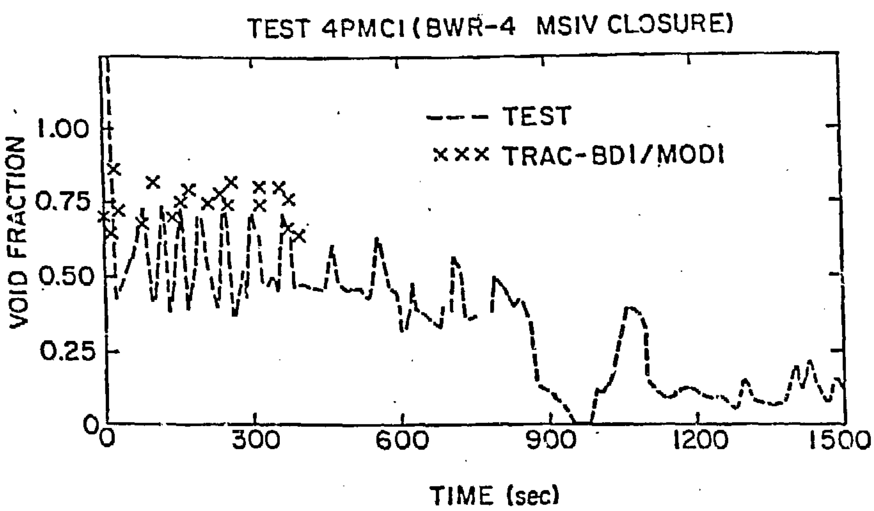


Figure 8. Bundle Nodal Void Fraction at the Top of Bundle.

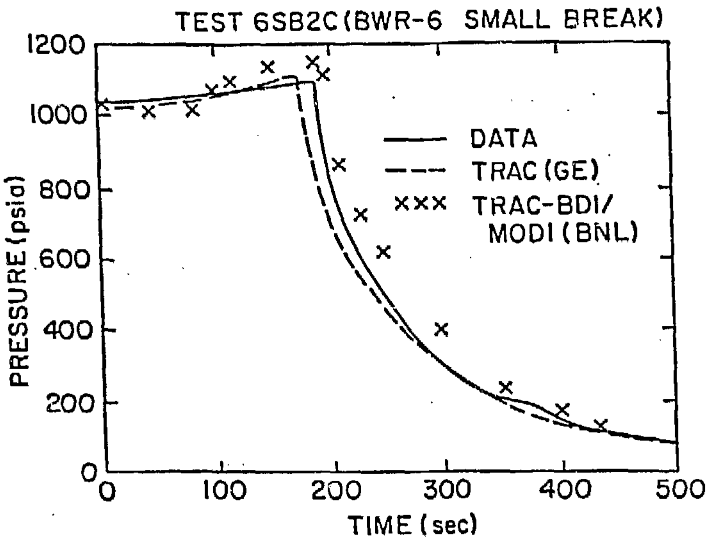


Figure 9. System Pressures.

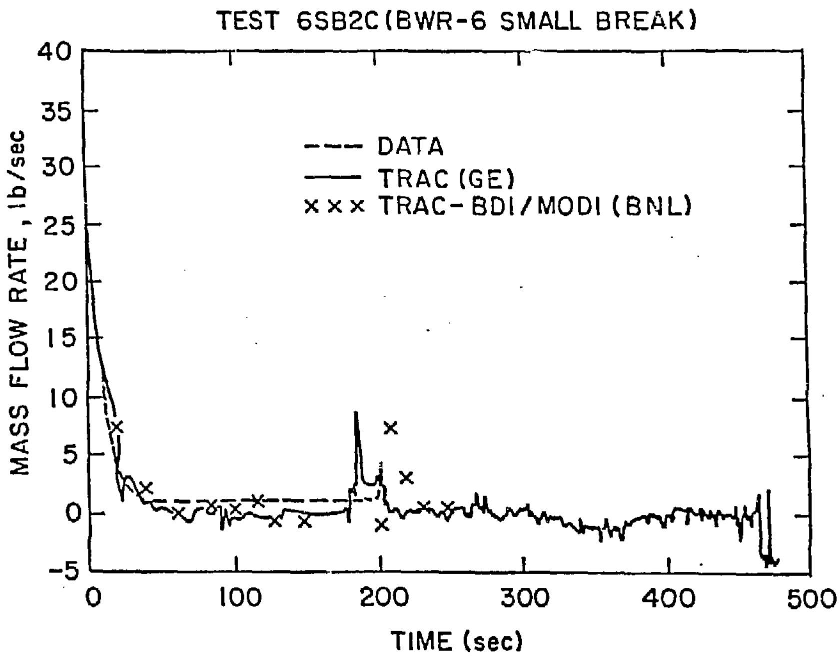


Figure 10. Core Inlet Flow.

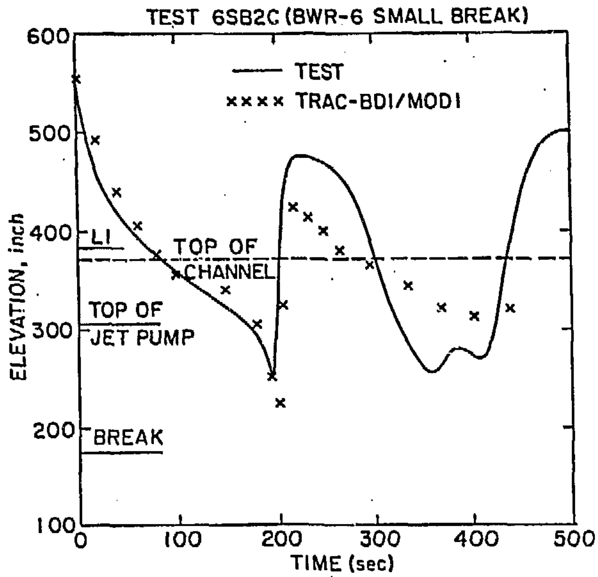


Figure 11. Downcomer Water Level.

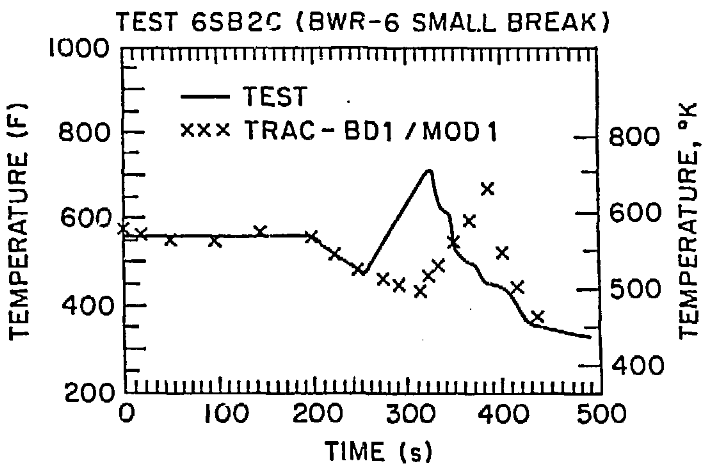


Figure 12. Rod Temperature (Elev. 117").

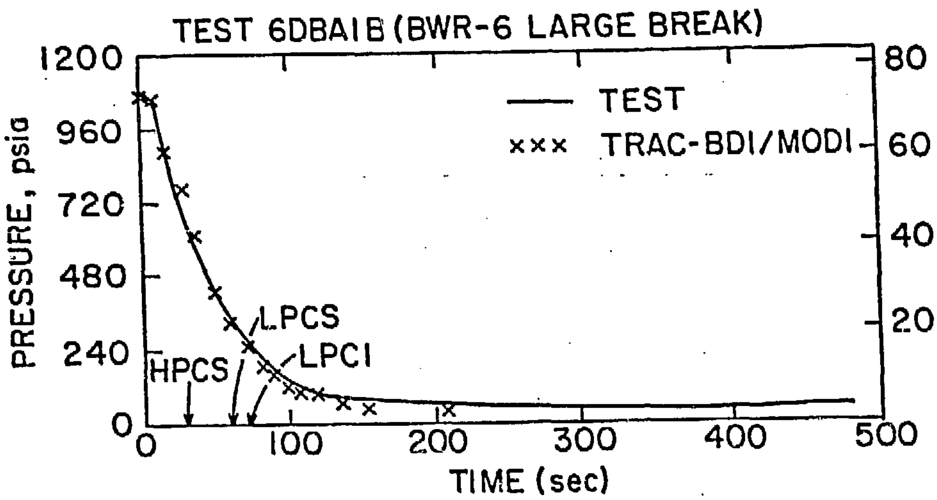


Figure 13. System Pressure.

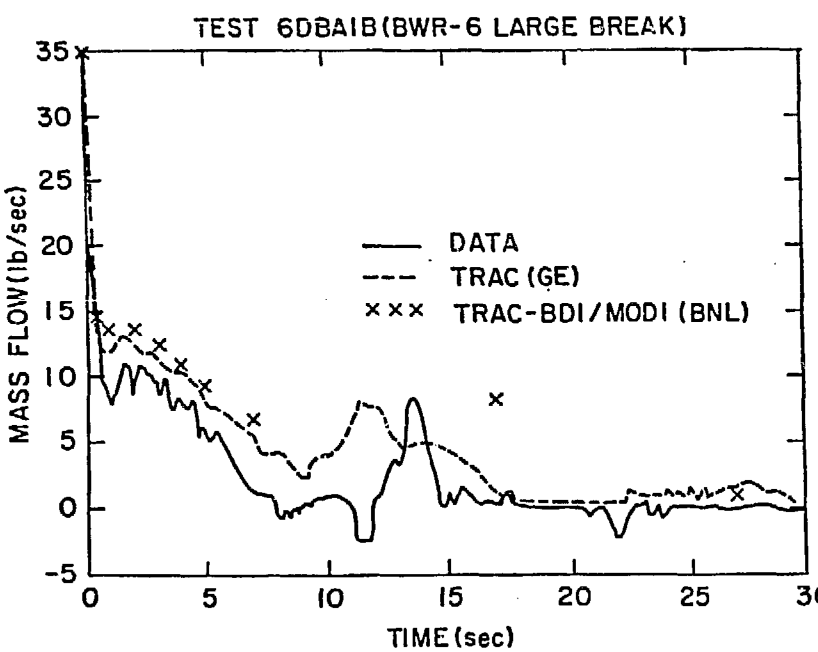


Figure 14. Core Inlet Flow.

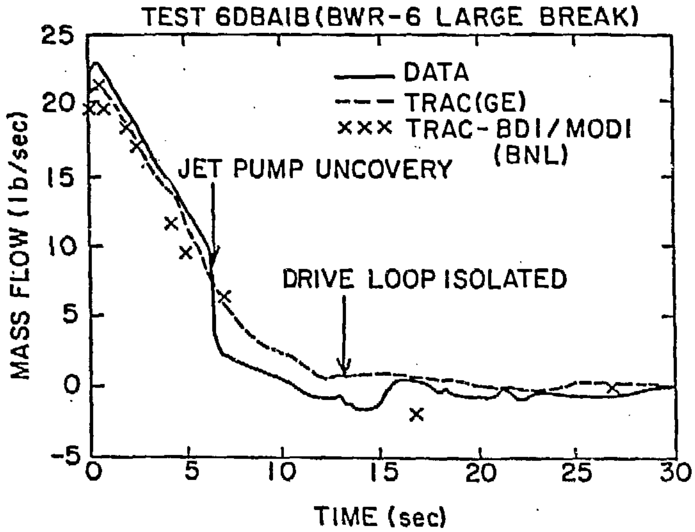


Figure 15. Intact Loop Jet Pump Flow.

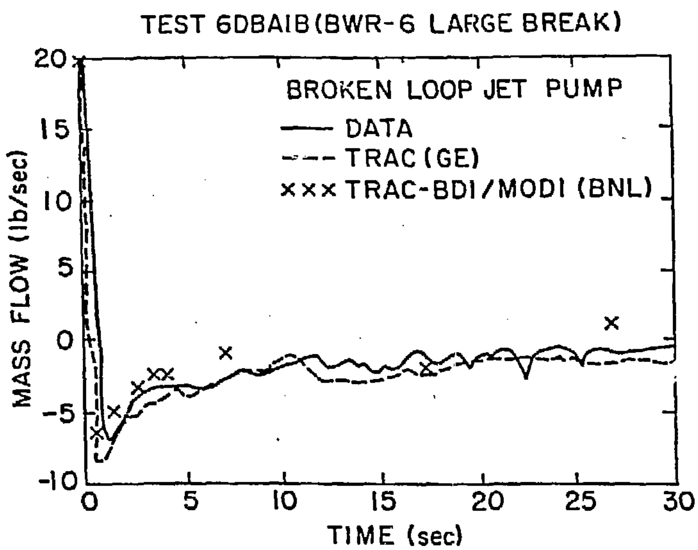


Figure 16. Broken Loop Jet Pump Flow.

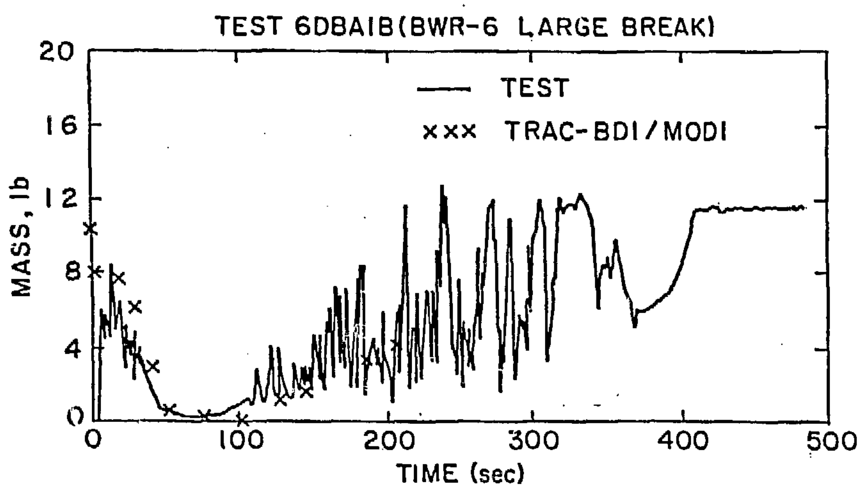


Figure 17. Intact Loop Jet Pump Mass.

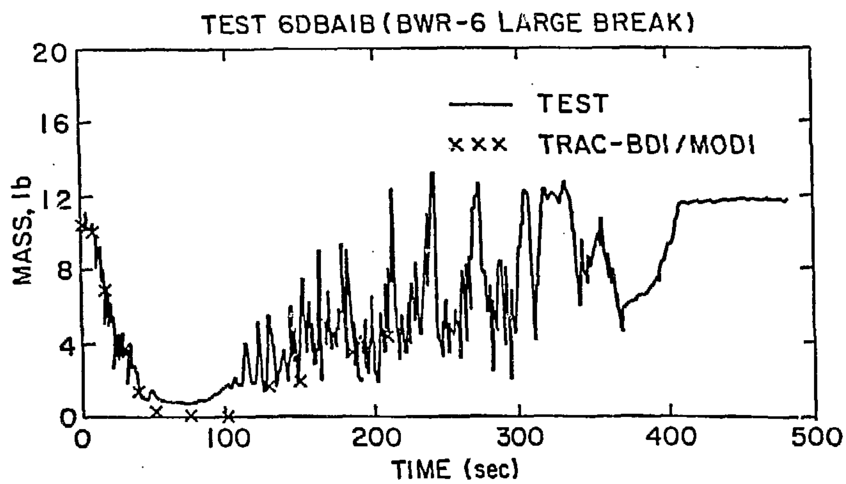


Figure 18. Broken Loop Jet Pump Mass.

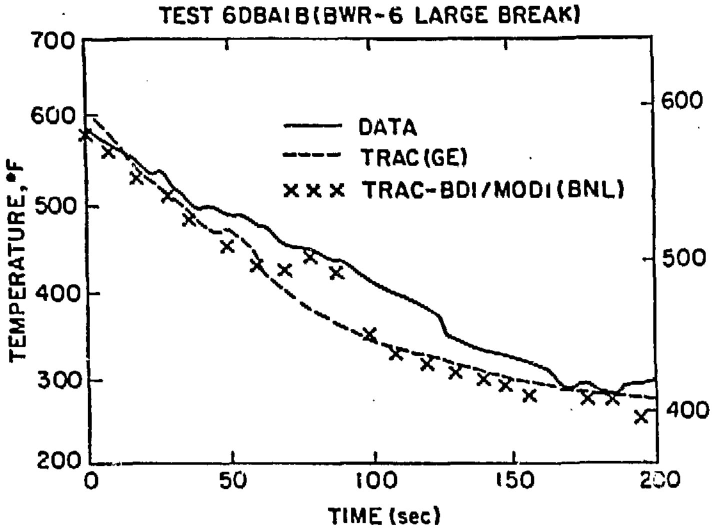


Figure 19. Bundle Mid-Plane Average Temperatures.

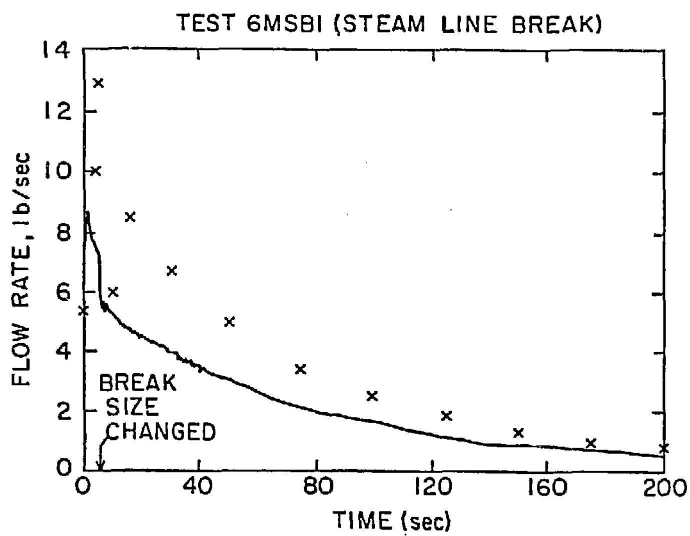


Figure 20. Steam Line FLow.

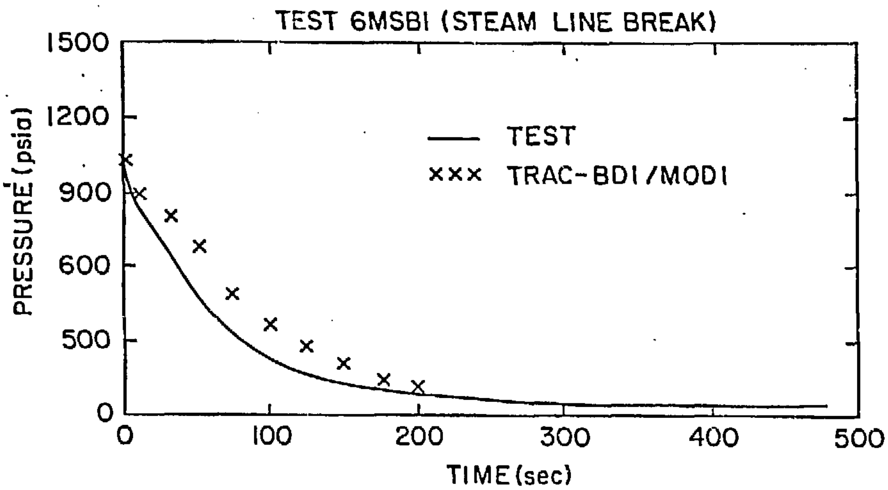


Figure 21. System Pressure.