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NSD-NRC-96-4673
DCP/NRC0482
Docket No.: STN-52-003

March 26, 1996

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE NOTRUMP COMPUTER CODE

Dear Mr. Quay:

Enclosed is information in response to an NRC request for additional information on the NOTRUMP computer code. The enclosed information includes:

Enclosure 1 Westinghouse response to RAI 440.515 (Revision 1)

Please contact John C. Butler on (412) 374-5268 if you have any questions concerning this transmittal.

Brian A. McIntyre, Manager
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/nja

Enclosures

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Question 440.515

Re: NOTRUMP PVR FOR ~~CU~~ TESTS, LTCT-GSR-001, JULY 1995

The conclusions state that the NOTRUMP code "captures and accurately represents the key thermal hydraulic phenomena of importance for the AP600 small break LOCA." An important small break LOCA thermal hydraulic phenomenon is two-phase level swell; in particular the two-phase level in the inner vessel region containing the lower plenum, core, upper plenum, and upper head. Because there were no comparisons of the liquid level nor the two-phase level in the inner vessel, there is no assurance that the NOTRUMP code captures this important phenomenon. Based on the over predicted downcomer liquid level transient data and the fact that the upper head also prematurely drained in the NOTRUMP calculations, there is no assurance that the NOTRUMP code can adequately assess the potential for core uncovery for AP600 during small break LOCAs. Major changes have been made to the code bubble rise, drift flux, and level tracking models with no separate effects nor integral test comparisons (the OSU and SPES-2 test comparisons do not provide verification of the code ability to model level swell) provided to verify and validate the capabilities of the code to predict two-phase level swell. Until appropriate benchmark to level swell data can be provided, the NOTRUMP code's ability to accommodate two-phase level swell phenomena is an open issue. Candidate level swell test data for benchmarking NOTRUMP:

(1) THTF bundle uncovery tests(1,2,3,4) includes steady-state and transient bundle uncovery data where the code mixture, liquid level, and void distributions can be used to verify the code level swell and heatup models.

(2) The Containment System Experiments(5,6) provide level swell data from the simple blowdown of a vessel from side and bottom exit nozzles. Test B-10(5) provides pressure and level data for a bottom blowdown while tests B-50 through B-53(6) provide top blowdown level swell data.

(3) The G-2 test facility consists of a test vessel with a simulated core and includes bundle uncovery data(7) for a range of power levels and pressures down to and including atmospheric conditions. Please show the NOTRUMP code mixture levels, void distributions, steam and fuel rod temperatures for several of these tests covering a range of pressures and power levels. Please also provide the downcomer liquid level response for the tests chosen.

(4) The GE level swell data(8) provide level swell data for the top blowdown of a vessel with and without heat addition, presented in Section B.4.

(5) Additional GE level swell tests(9) were performed which also contains axial void distribution data.

REFERENCES

1. Anklam, T. M., Mills, R. J., White, M. D., "Experimental Investigations of Steady State Bundle Heat Transfer and Two-Phase Mixture Level Swell Under High Pressure Low Heat-Flux Conditions," Oak Ridge National Laboratory, NUREG/CR-2456, March, 1982.

2. "Experimental Investigation of Bundle Boiloff and Reflood Under High-Pressure Low Heat-Flux Conditions," NUREG/CR-2455 ORNL-5846, April, 1982.

3. "Heat Transfer Above the Two-Phase Mixture Level Under Core Uncovery Conditions in a 336-Rod Bundle," Westinghouse Electric Corp., EPRI NP-1692, Vol. 1, January, 1981.

4. Anklam, T. M., "ORNL Small Break LOCA Heat Transfer Test Series I: Rod Bundle Heat Transfer Analysis," Oak Ridge National Laboratory," NUREG/CR-2052, August, 1981.

5. "Experimental High Enthalpy Water Blowdown From a Simple Vessel Through a Bottom Outlet," Battelle Northwest, BNWL-1411, June, 1970.

6. "Coolant Blowdown Studies of a Reactor Simulator Vessel Containing a Perforated Sieve Plate Separator," Battelle Northwest, BNWL-1463, February, 1971.

7. "Heat Transfer Above the Two-phase Mixture Level Under Uncovery Conditions in a 336-rod Bundle," Westinghouse Electric Corporation, EPRI-1692, January 1981.

8. Slifer, B. C., "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, April 1971.

9. "BWR Refill-Reflood Program - Model Qualification Task Plan," EPRI Report No. EPRI NP-1527, October, 1991.

Response (Revision 1 - DRAFT):

~~For the OSU and SPES-2 tests, comparison plots of core and upper plenum levels are being provided in response to RAIs 440.492, 440.518, and 440.520. These plots show reasonable agreement between NOTRUMP and the tests. However, since the two-phase level in the vessel and core is ranked as a high in the final PIRT table for the AP600 small-break LOCA as given in the response to RAI 440.325, Westinghouse will perform analysis of the G-2 level swell experiments given in EPRI report EPRI-NP-1692. Tests will be simulated over the full pressure range given in the test data at different bundle powers. Comparisons of the mixture height as a function of time, void distributions, vapor temperatures, and heater rod temperatures will be provided. These separate effects experiments, along with the existing comparisons to the SPES and OSU tests will provide sufficient validation of the NOTRUMP level swell and drift flux models.~~

~~The results of the analysis of these G-2 experiments are scheduled to be provided to the NRC in early March, 1996. Westinghouse will meet with the NRC staff before this date to discuss progress on modeling these tests.~~

The conditions under which the NOTRUMP level swell and void fraction models are most important are low pressure. For most of the cold leg small break transients, the minimum reactor system and core/upper plenum inventory occurs at low pressures. Breaks of larger connecting lines such as the cold leg balance line or the direct vessel injection line can result in a low reactor coolant system inventory at higher pressures. However, for these larger breaks, the pressurized accumulators quickly inject and restore the reactor system inventory. The level swell

issues for the AP600 are more critical at the lower pressures when the inventory of the core makeup tanks is nearly depleted and the IRWST injection has not begun. This is a low pressure concern where the two-phase level swell and void fraction distribution is more difficult to predict due to the greater difference between the vapor and liquid phase densities. The suggested experiments have been evaluated as to the contribution they could make relative to the application of NOTRUMP to the AP600 small-break LOCA for the critical time periods in the transient.

The first experiment which was suggested was the ORNL THTF bundle uncovery tests (References 440.515-1 to 440.515-3 ), which includes transient bundle uncovery, void fraction, mixture level and heater rod heatup data. Two types of tests were conducted, boil-off and high pressure reflood tests under both steady state and transient conditions. The conclusions from the authors indicate that results of the transient mixture level swell tests are in reasonable agreement with the mixture level swell data from the steady-state tests (Reference 440.515-2). The test conditions for the steady-state boil-off tests are given in Table 3 from Reference 440.515-1. The test conditions for the transient depressurization and boil-off tests are given in Table 7 from Reference 440.515-2. The ORNL tests were specifically designed to provide data for the small break LOCA for current operating PWRs where core uncovery is calculated to occur at high pressure, typically above 500 psia as the loop seal in the reactor coolant cold legs vent and the accumulators are initiated to terminate the transient.

The AP600 has eliminated the loop seal by design, so there is a continual drainage of the reactor coolant inventory into the reactor vessel, particularly at high pressure. Therefore, for the AP600, there is no risk of core uncovery due to loop seal clearing occurring at higher pressure as calculated in conventional plants. The minimum core inventory is calculated to occur at lower pressures for the AP600 for the small break LOCA transients with the minimum mass inventory occuring at very low pressure for the smaller breaks. Figures 1 to 3 indicate the reactor system inventory, pressurizer pressure and the upper plenum and core mixture level for a two-inch break in the cold leg. Comparing the figures indicates that the minimum inventory occurs just before the IRWST injection when the system pressure is approximately 25 psia. The upper plenum and core mixture level indicate that substantial margin exists between the top of the core at 18.8 feet and the calculated mixture level. Similar behavior is shown in Figures 4 to 6 for a two-inch balance line break. Similar behavior has also been seen for all smaller cold leg and hot leg breaks. Larger breaks of attached piping such as the double-ended break of the direct vessel injection line (DEDVI) or the double-ended break of the cold leg balance line (DECLBL), cause more rapid depressurization and a minimum reactor system inventory at earlier time and higher pressure. Figures 7 to 12 show similar plots of the reactor system inventory, pressurizer pressure and the upper plenum and core mixture level for these two breaks. The minimum reactor system inventory occurs at approximately 275 seconds for the DEDVI break when the system pressure is approximately 400 psia. The reactor system inventory is rapidly recovered by the accumulator injection. For smaller breaks, the recovery is not so rapid and the system remains at a minimum inventory for longer periods of time. The level in the core and upper plenum indicate that there are periods later then 300 seconds, when the pressure is lower, and the mixture level is the same or lower. A similar behavior is seen for the DECLBL break where the minimum reactor coolant inventory is quickly recovered by the accumulator as shown in Figures 10 to 12.

Since level swell and bubble size, is a strong function of the system pressure, the high pressure ORNL experiments are not typical of the conditions of interest for the AP600. Therefore, these tests do not provide the pertinant data needed for validation of the NOTRUMP level swell models for low pressure AP600 conditions.

The Containment System Experiments (References 440.515-4 and 440.515-5) were also suggested as data which could be used to validate the NOTRUMP level swell models for the AP600. These experiments are unheated vessel blowdown tests which were primarily designed to provide data on structure loads during blowdown. One of the concerns in using the test data is that the tests had a significantly large amount of dissolved nitrogen in the water. This was due to the method used to heat the water and establish the test conditions (see pages 5.25 to 5.29 from BNWL-1463). The two-phase froth height was measured in the experiments using a Time Domain Reflectometer (TDR) which gives the distance from the top of the vessel to the two-phase level. The report indicates that there is a temperature drift in the instrument which is not corrected in the data. Since the tests depressurize, the temperature of the fluid and tank will change to follow the saturation temperature, resulting in error in the indicated two-phase level (page 5.52, Figure 5.28, page 5.58). The available predictions do not compare very well to the data as seen in Figures 5.31 to 5.34. Also, the other data which are available are absolute pressures which can not be used to infer void distribution. We do not believe that these tests are worth analyzing due to the dissolved nitrogen effects, the lack of temperature calibration and drift of the TDR and the lack of other measurements.

The Westinghouse G-2 steady-state core uncovery experiments, Reference 440.515-6, were also suggested as tests which could be used to validate the NOTRUMP level swell models. The G-2 experiments were conducted on a 14-foot, 336 heater rod bundle, at several different power levels, and over a range of pressures which also include low pressures typical of the AP600. There were also differential pressure measurements which can be used to infer void fraction and the heatup of the heater rod thermocouples indicate the location of the two-phase froth level. We believe the G-2 experiments will give useful data which can be used to validate the NOTRUMP level models for the AP600. These tests will be added to the benchmark and validation experiments for the NOTRUMP code.

It was also suggested that the General Electric (GE) vessel blowdown level swell experiments be considered for the NOTRUMP validation (References 440.515-7,8). These level swell experiments also include pressure drop data which has been used by GE to calculate the void distribution in the tank as well as to determine the froth height of the mixture as the tank drains. The small vessel tests (Reference 440.515-8) are the most useful since they have longer blowdowns which last for hundreds of seconds, similar to a small break LOCA. The larger vessel tests were designed as large break LOCA simulations and are not as useful. The GE small vessel tests will be useful for the validation of the NOTRUMP level swell models and tests 8-21-1 and 8-25-1 will be added to the NOTRUMP code validation matrix.

During separate discussions, the NRC staff have suggested that Oregon State University (OSU) tests SB26 and SB28 (Reference 440.515-9) be added to the NOTRUMP validation matrix since these are the only OSU experiments in which the heater rod bundle became uncovered. These tests were specifically designed as "Beyond Design Basis" experiments in which additional equipment failures, beyond the limiting single failure, had to be assumed to force uncovery of the heater rod bundle. While the OSU tests do have a transient boil-off and bundle uncovery with the associated heater rod temperature heatup transient, the G-2 experiments capture the same thermal-hydraulic phenomena, on a full length bundle, as opposed to the OSU bundle which is only three feet in length. The uncovery behavior in the OSU experiments occurs at a near constant pressure in the same pressure range which the G-2 experiments provide data. Therefore, we believe that the G-2 bundle uncovery experiments provide a more complete data set for the boil-off and core uncovery which covers the range of conditions for the AP600. The addition of the OSU "Beyond Design Basis Tests" makes no additional contribution to the validation of the NOTRUMP level swell models.

A benchmark/thought problem (Reference 440.515-10) was also sent to Westinghouse by INEL, as a suggested problem for NOTRUMP. The problem examines the codes numerical predictions of single phase flow in two parallel pipes when fed from the same reservoir which are then recombined into a common resevior. The AP600 noding for NOTRUMP does not have such a noding configuration as given in the problem. Also the flow in NOTRUMP is two-phase, not single phase for much of the transient. Westinghouse has proposed eight (8) different benchmark/thought problems which are designed to examine in detail the specific two-phase or component models used in the NOTRUMP code. Since the small break transients are all two-phase, it is believed that the planned Westinghouse benchmark/thought problems will be very useful for the NOTRUMP review, while the suggested INEL problem, which may be of numerical interest, would be less useful. Therefore, we will concentrate on the planned benchmark/thought problems.

In conclusion, Westinghouse believes that the addition of the G-2 core uncovery experiments and the General Electric small vessel blowdown experiments to the planned separate effects and benchmarking NOTRUMP code validation plan will adquately address the issues on the level swell models.

References

440.515-1. Anklam, T.M. et al, "Experimental Investigations of Steady-State Bundle Heat Transfer and Two-Phase Mixture Level Swell Under High Pressure Low Heat Flux Conditions", NUREG/CR-2456, (1982).

440.515-2. Anklam, T.M., et al,"Experimental Investigation of Bundle Boiloff and reflood Under High Pressure Low Heat Flux Conditions", NUREG/CR-2455, ORNL-5846 (1982).

440.515-3. Anklam, T.M. "ORNL Small Break LOCA Heat Transfer Test Series 1, Rod Bundle heat Transfer Analysis", NUREG/CR-2052, (1981).

440.515-4. Allemann, R.T. et al, "Coolant Blowdown Studies of a Reactor Simulator Vessel containing a Perforated Sieve Plate Separator" BNWL-1463 (1971).

440.515-5. Allemann, R.T. et al, "Experimental High Enthalpy Water Blowdown From a Simple Vessel Through a Bottom Outlet", BNWL-1411 (1970).

440.515-6. Andreycheck,T.A. et al, "Heat Transfer Above the Two-Phase Mixture Level Under Core Uncovery Conditions in a 336-Rod Bundle", Westinghouse Electric Corporation, EPRI NP-1692, Vol 1 and 2 (1981).

440.515-7. Slifer, B.C. "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling reactors", NEDO-10329 (1971).

440.515-8. Findlay, L.A. and G.L. Sozzi, "BWR Refill-Reflood Program- Model Qualification Task Plan", NUREG/CR-1899, (1981).

440.515-9. Dumsday, C.L. "AP600 Low Pressure Systems Test at Oregon State University, Final Data Report", WCAP-14252, (1995).

440.515-10. Correspondence from L. Ward, INEL to L.E. Hochreiter. "Flow Anomaly Test Problem" Feb 2, 1996.

SSAR Revision: NONE

Table 3. Summary of uncovered-bundle heat transfer test conditions[a] *

| Test | System pressure [MPa (psia)] | Linear power/rod [kw/m (kw/ft)] | Mass flux [kg/m²·s (lbm/h·ft²) x 10⁻⁴] | Mixture level [m (ft)] | Steam cooling region [m (ft)] | Vapor Reynolds number (BOSCR)[b] | Vapor Reynolds number (EOSCR)[c] | Fractional heat loss | Heat transfer regime (BOSCR)[b,d] | Heat transfer regime (EOSCR)[c,d] |
|---|---|---|---|---|---|---|---|---|---|---|
| 3.09.10I | 4.5 (650) | 2.22 (0.68) | 29.7 (2.19) | 2.62 (8.6) | 3.02–3.62 (9.91–11.88) | 16,600 | 12,200 | 0.018 | FCT | FCT |
| 3.09.10J | 4.2 (610) | 1.07 (0.33) | 12.7 (0.94) | 2.47 (8.1) | 3.02–3.62 (9.91–11.88) | 6,700 | 5,000 | 0.052 | MCT | FCT |
| 3.09.10K | 4.0 (580) | 0.32 (0.10) | 3.1 (0.23) | 2.13 (7.0) | 2.42–3.62 (7.94–11.88) | 1,900 | 1,100 | 0.176 | MCT | FCL |
| 3.09.10L | 7.5 (1090) | 2.17 (0.66) | 29.1 (2.15) | 2.75 (9.0) | 3.02–3.62 (9.91–11.88) | 17,700 | 13,000 | 0.017 | FCT | FCT |
| 3.09.10M | 7.0 (1010) | 1.02 (0.31) | 12.6 (0.93) | 2.62 (8.6) | 3.02–3.62 (9.91–11.88) | 6,500 | 5,100 | 0.042 | MCT | MCT |
| 3.09.10N | 7.1 (1030) | 0.47 (0.14) | 4.6 (0.34) | 2.13 (7.0) | 2.42–3.62 (7.94–11.88) | 3,000 | 1,600 | 0.162 | MCT | MCTR |

[a] Numbers in this table have been rounded off. For precise listing of test conditions see Appendix B.

[b] BOSCR – beginning of steam-cooling region.

[c] BOSCR – end of steam-cooling region.

[d] Abbreviations are:
- FCT – forced-convection turbulent
- MCT – mixed-convection turbulent
- FCL – forced-convection laminar
- MCTR – mixed-convection transition to laminar

* Reference 440.515-1

Table 7. SBLOCA-II boiloff test operating conditions *

| Test | Duration (s) | Linear heat rate [kW/m·rod (kW/ft·rod)] | Depressurization rate [kPa/s (psi/s)] | Pressure range [MPa (psia)] | | Core height uncovered[a] (%) | | Peak rod surface temperature [K (°F)] |
|---|---|---|---|---|---|---|---|---|
| | | | | Desired | Observed | Desired | Actual | |
| 3.09.10T | 200 | 0.951 (0.29) | 11.58 (1.68) | 6.21–4.14 (900–600) | 5.93–3.72 (860–540) | 80 | 75 | 1088 (1500) |
| 3.09.10U | 65 | 1.94 (0.59) | 33.44 (4.85) | 7.93–6.21 (1150–900) | 8.14–5.86 (1180–850) | 70 | 91 | 994 (1330) |
| 3.09.10V | 135 | 0.656 (0.20)[b] | 17.79 (2.58) | 7.93–6.21 (1150–900) | 7.79–5.52 (1130–800) | 70 | 64 | 819 (1015) |
| 3.09.10W | 96 | 0.623 (0.19)[c] | 21.72 (3.15) | 7.93–6.21 (1150–900) | 7.86–5.86 (1140–850) | 70 | 42 | 705 (810) |
| 3.09.10X | 470 | 0.623 (0.19) | 0.765 (0.111) | 8.62 (1250) | 8.56–8.21 (1242–1190) | 60 | 75 | 1112 (1542) |

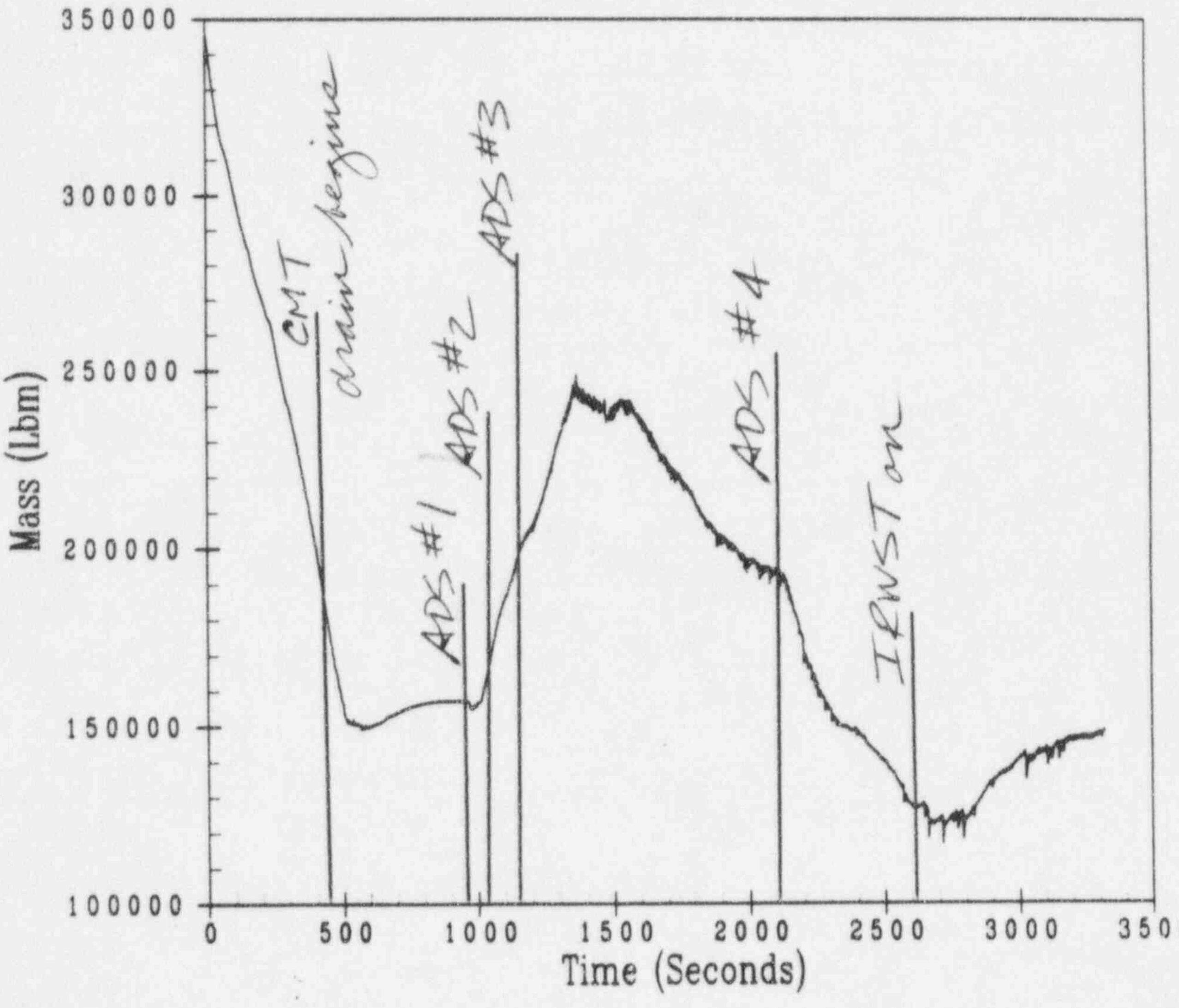
[a] 100% = 3.66 m.

[b] This test experienced a power reduction to 0.56 kW/m·rod (0.17 kW/ft·rod) at 47 s.

[c] This test experienced a power reduction to 0.33 kW/m·rod (0.10 kW/ft·rod) at 12 s.

* REFERENCE 440.515-2

Figure 15.6.5b-40 Two Inch Cold Leg Break In PRHR Loop
Reactor System Coolant Inventory




— Figure 1

Figure 15.6.5b-32 Two Inch Cold Leg Break In PRHR Loop
Pressurizer Pressure

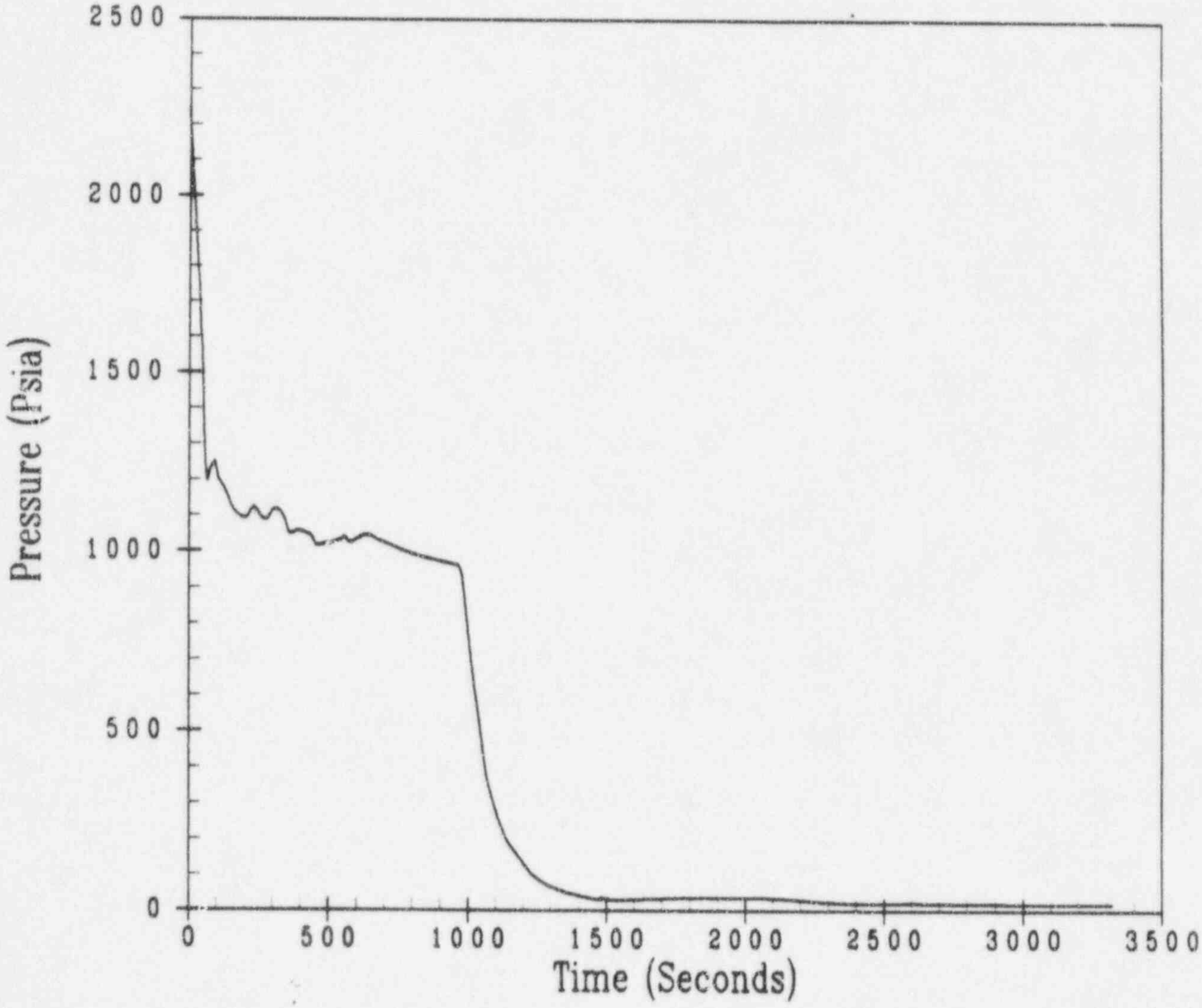
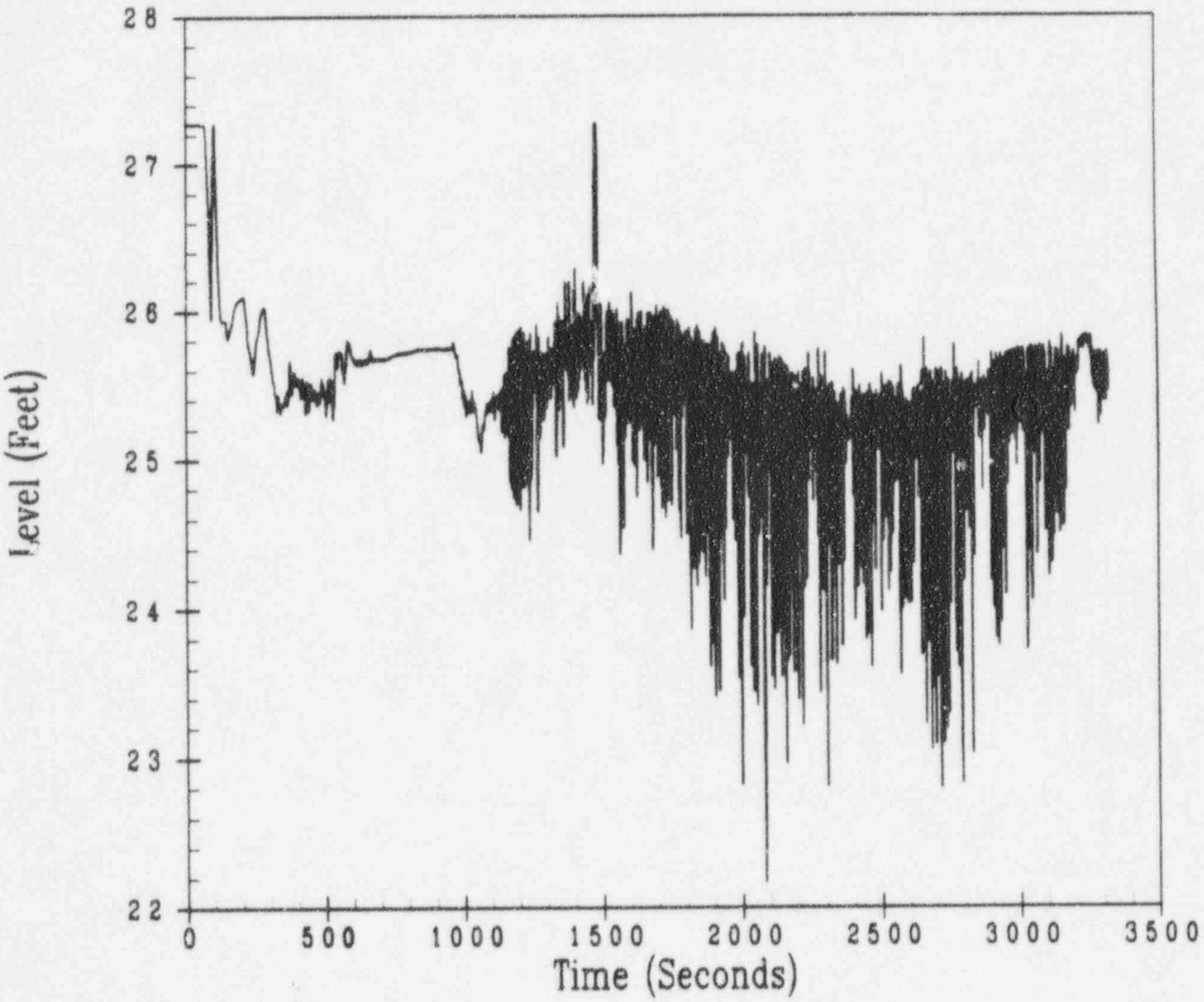



Figure 2

Figure 15.6.5b-41 Two Inch Cold Leg Break In PRHR Loop
Upper Plenum and Core Mixture Level




— Figure 3

Figure 15.6.5b-51 Two Inch Cold Leg Break In Balance Line Loop
Reactor System Coolant In entory

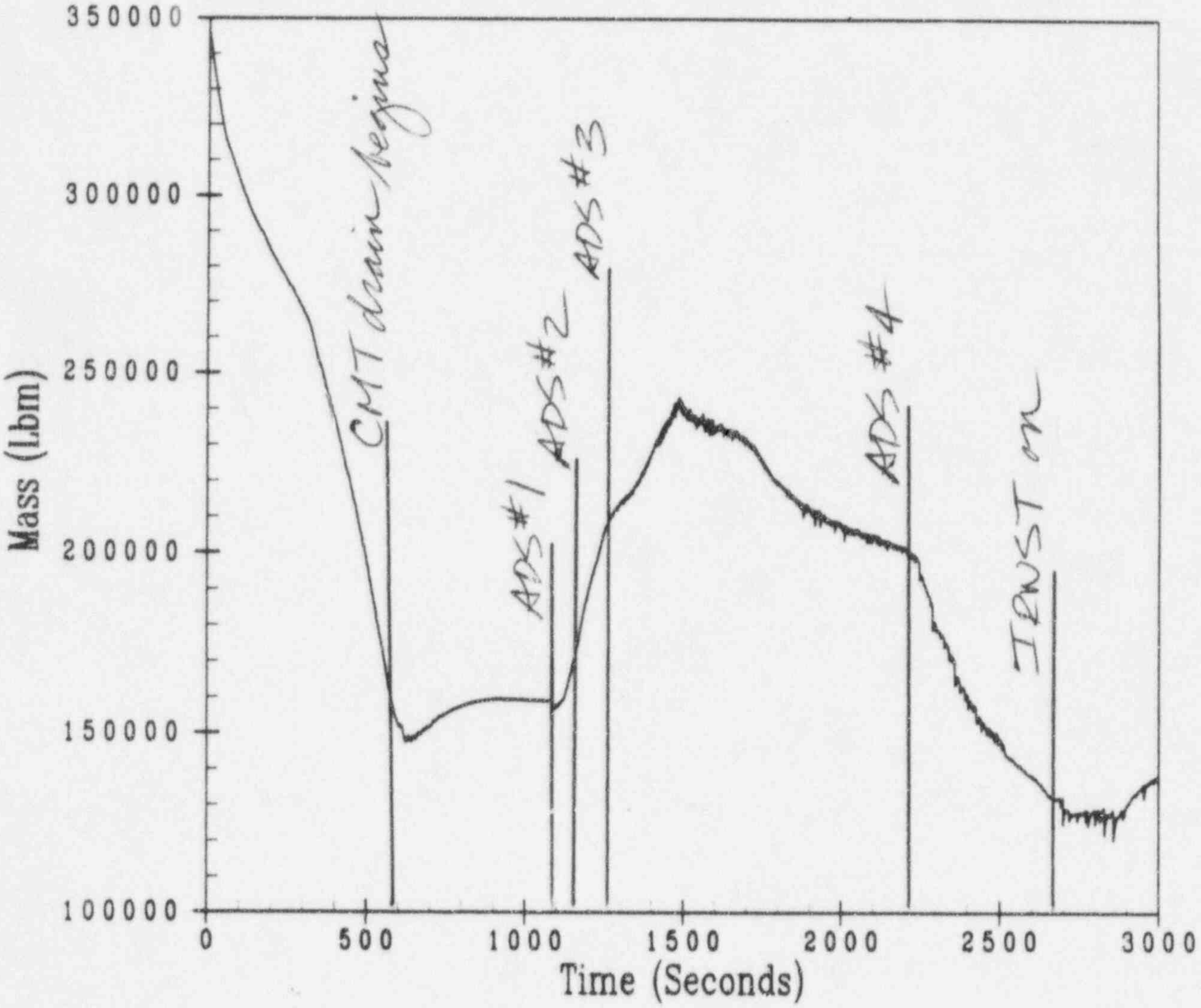
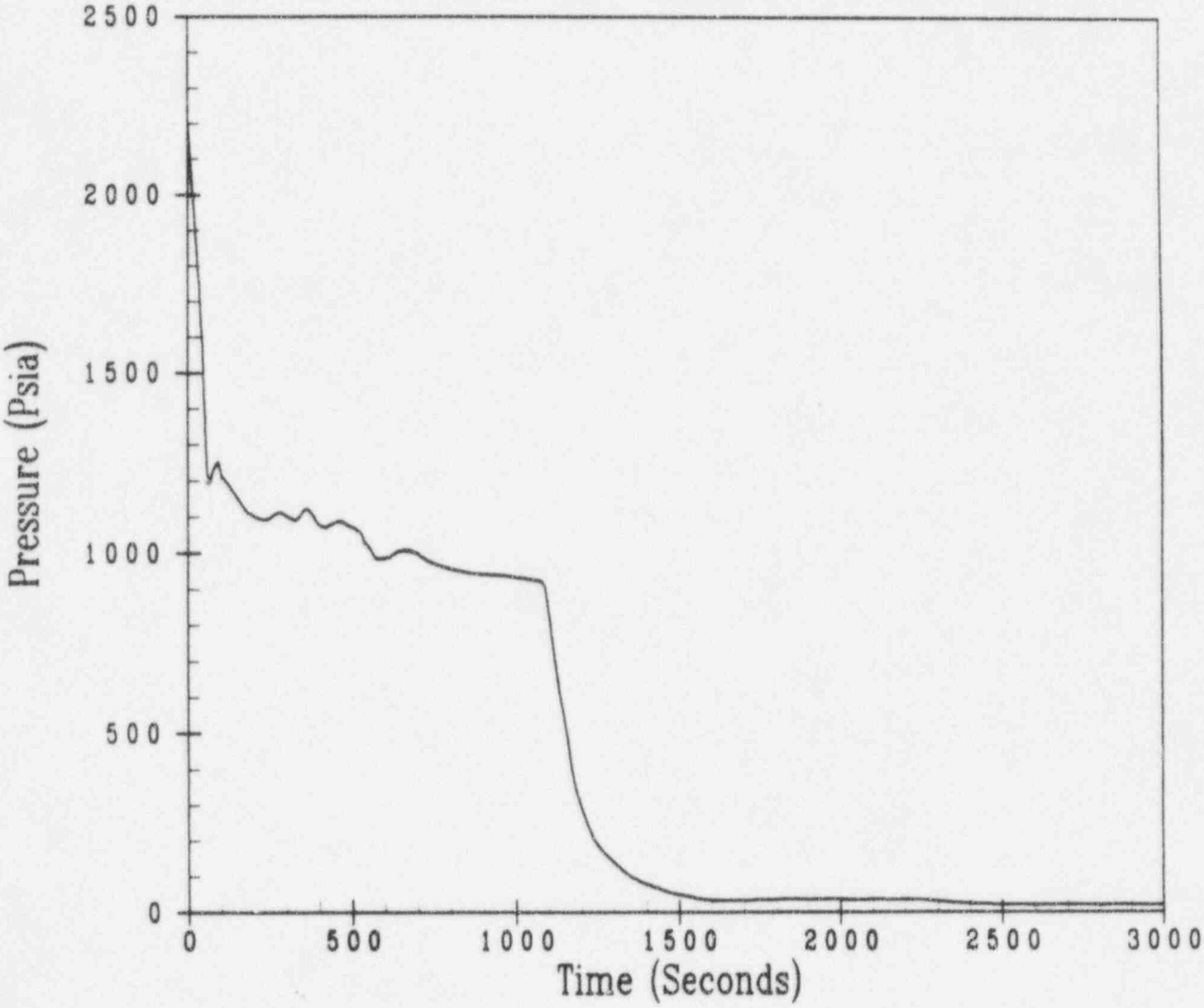



Figure 4

Figure 15.6.5b-43 Two Inch Cold Leg Break In Balance Line Loop
Pressurizer Pressure




— Figure 5

Figure 15.6.5b-52 Two Inch Cold Leg Break In Balance Line Loop
Upper Plenum and Core Mixture Level

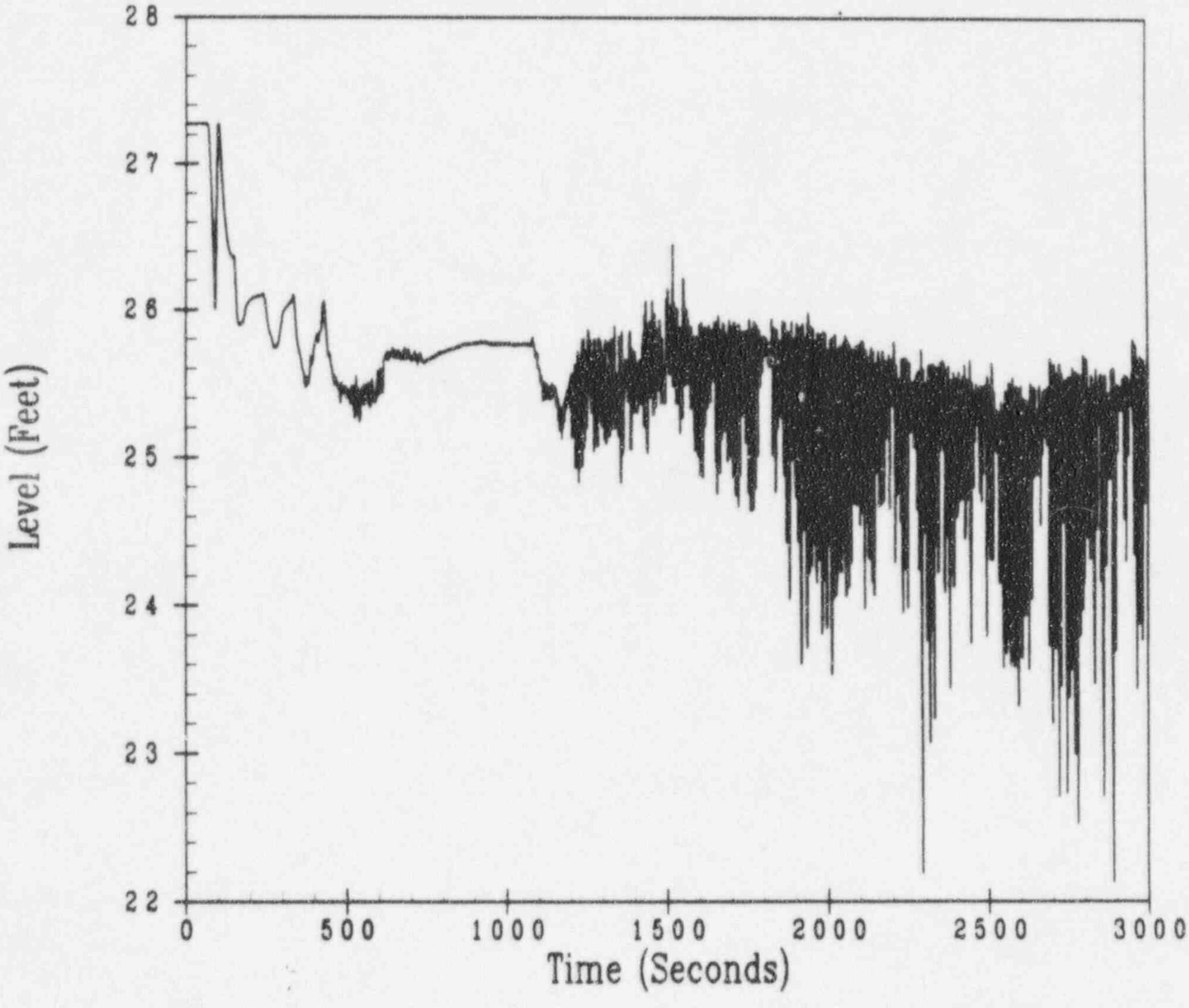



Figure 6

Figure 15.6.5b-29 Double Ended DVI Line Break
Reactor System Coolant Inventory

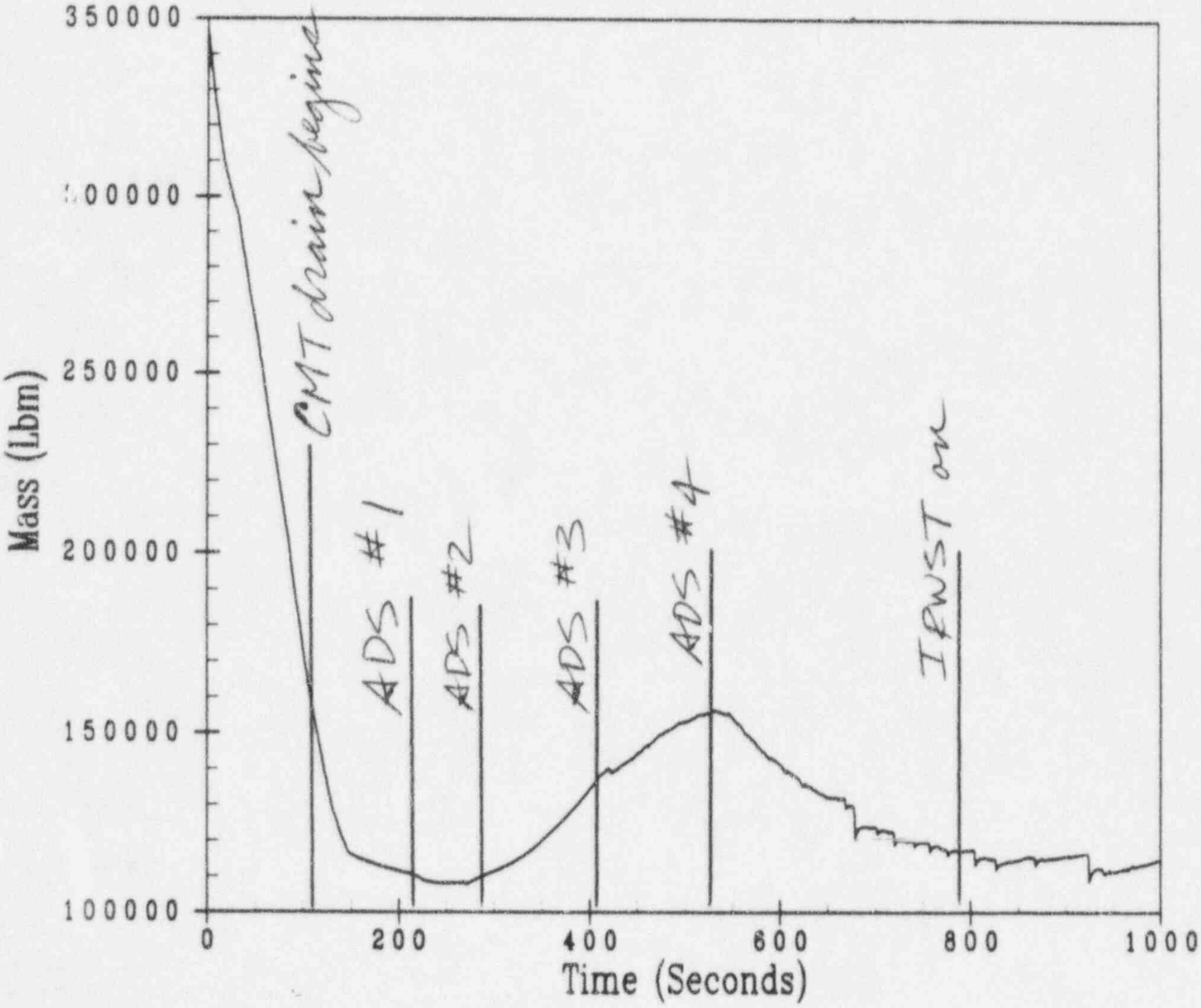



Figure 7

Figure 15.6.5b-14 Double Ended DVI Line Break
Pressurizer Pressure

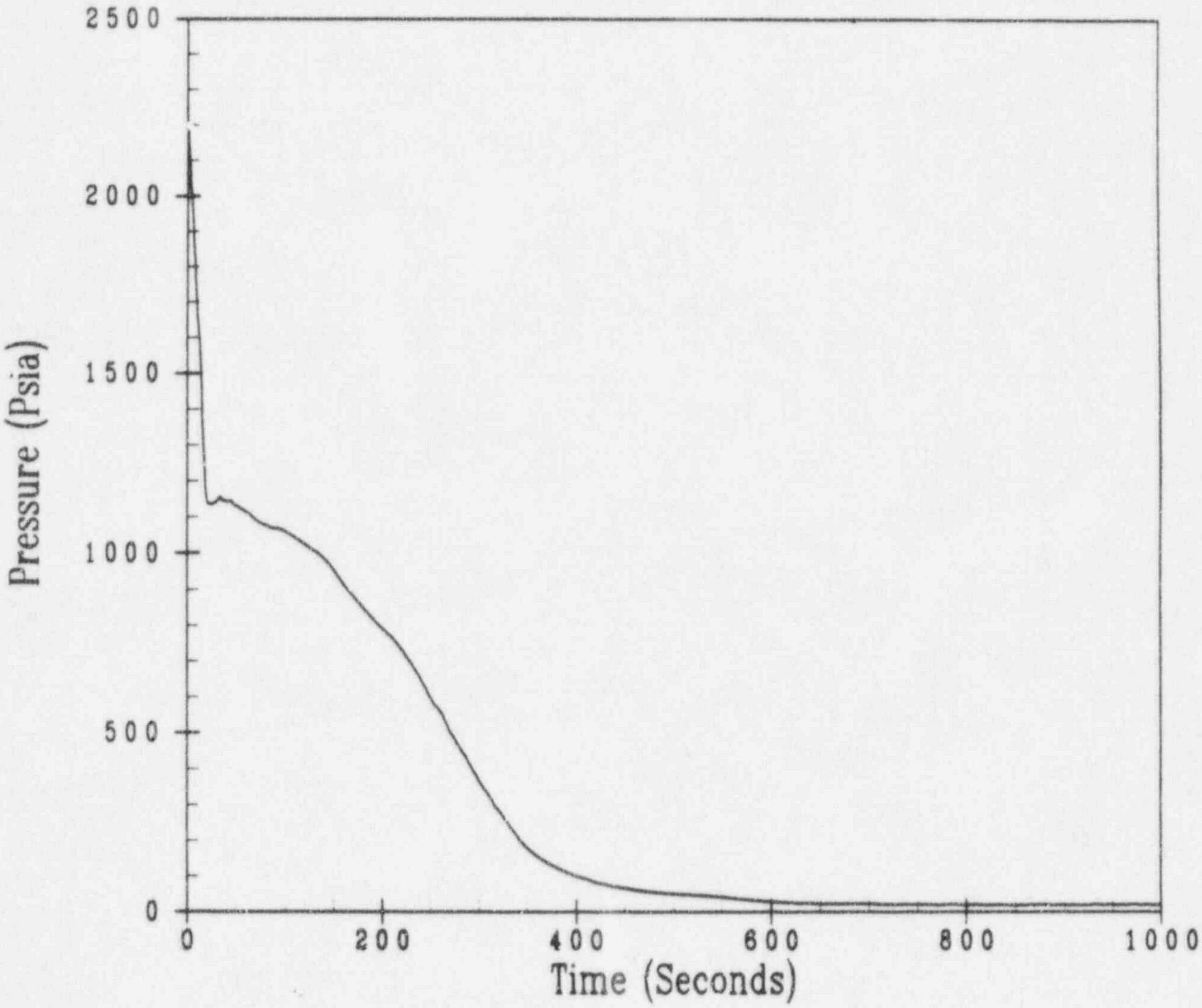



Figure 8

Figure 15.6.5b-17 Double Ended DVI Line Break
Upper Plenum and Core Mixture Level

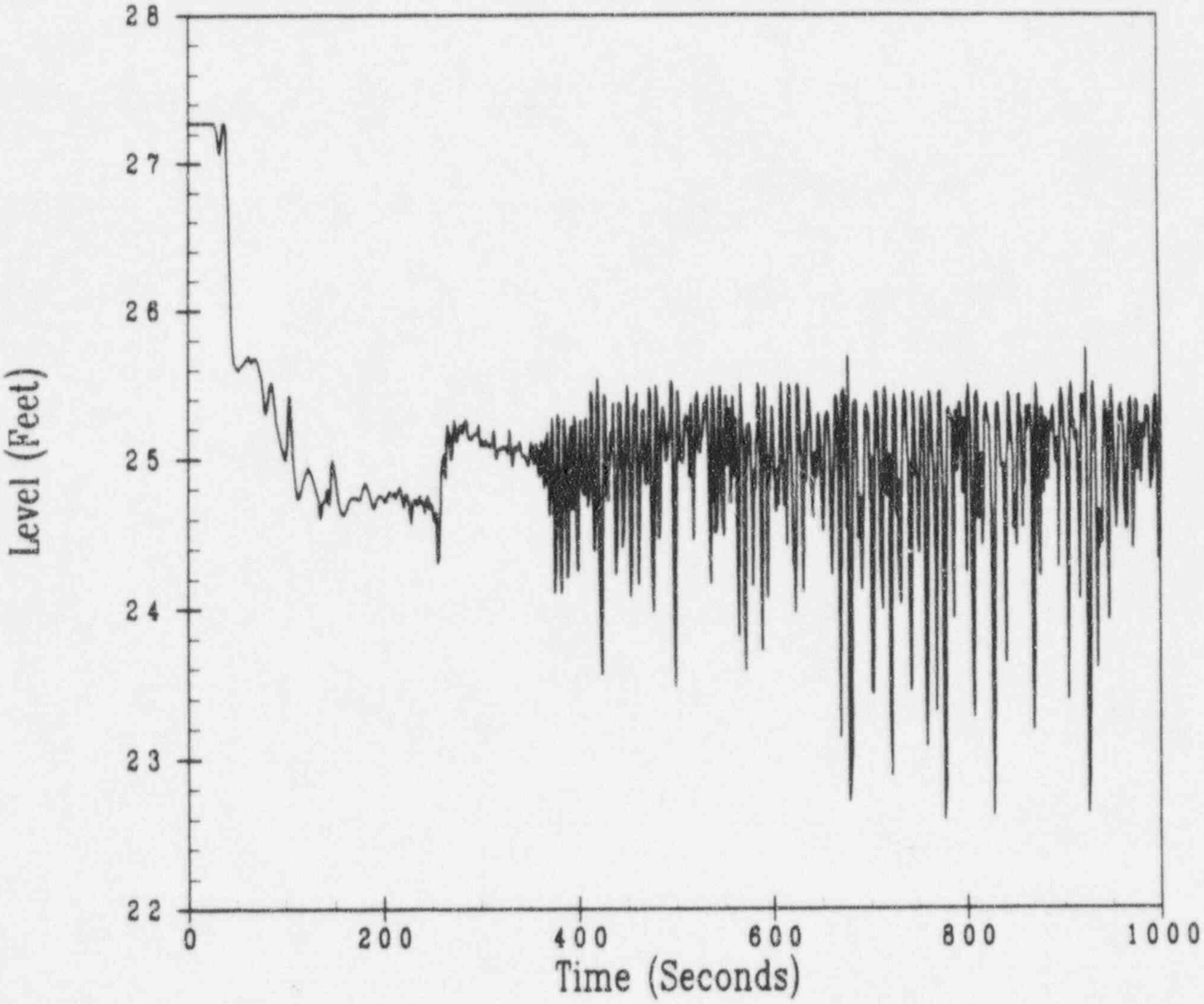



Figure 9

Figure 15.6.5b-66 Double Ended Balance Line Break
Reactor System Coolant Inventory

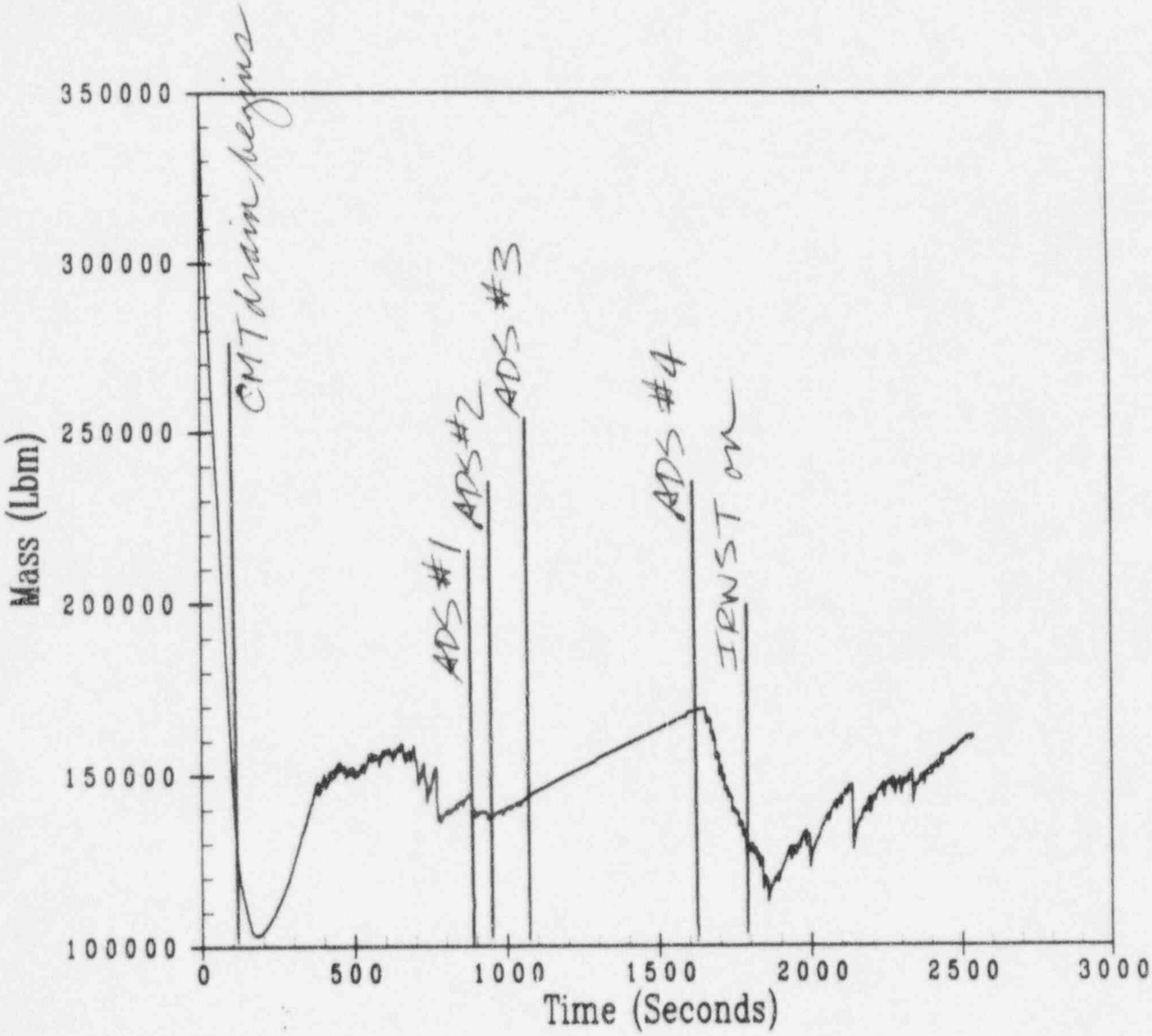



Figure 10

Figure 15.6.5b-57 Double Ended Balance Line Break
Downcomer Pressure

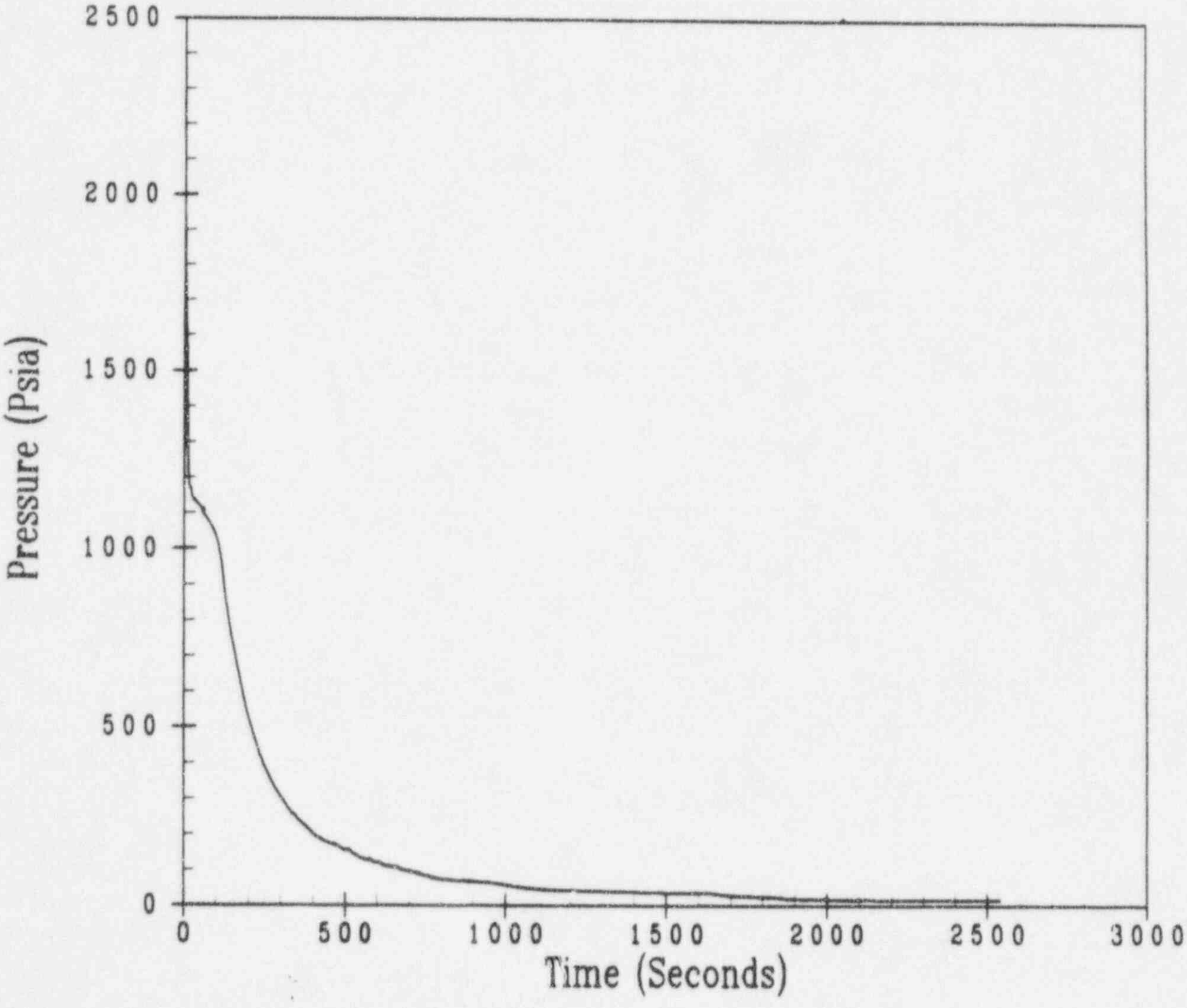



Figure 11

Figure 15.6.5b-58 Double Ended Balance Line Break
Upper Plenum and Core Mixture Level

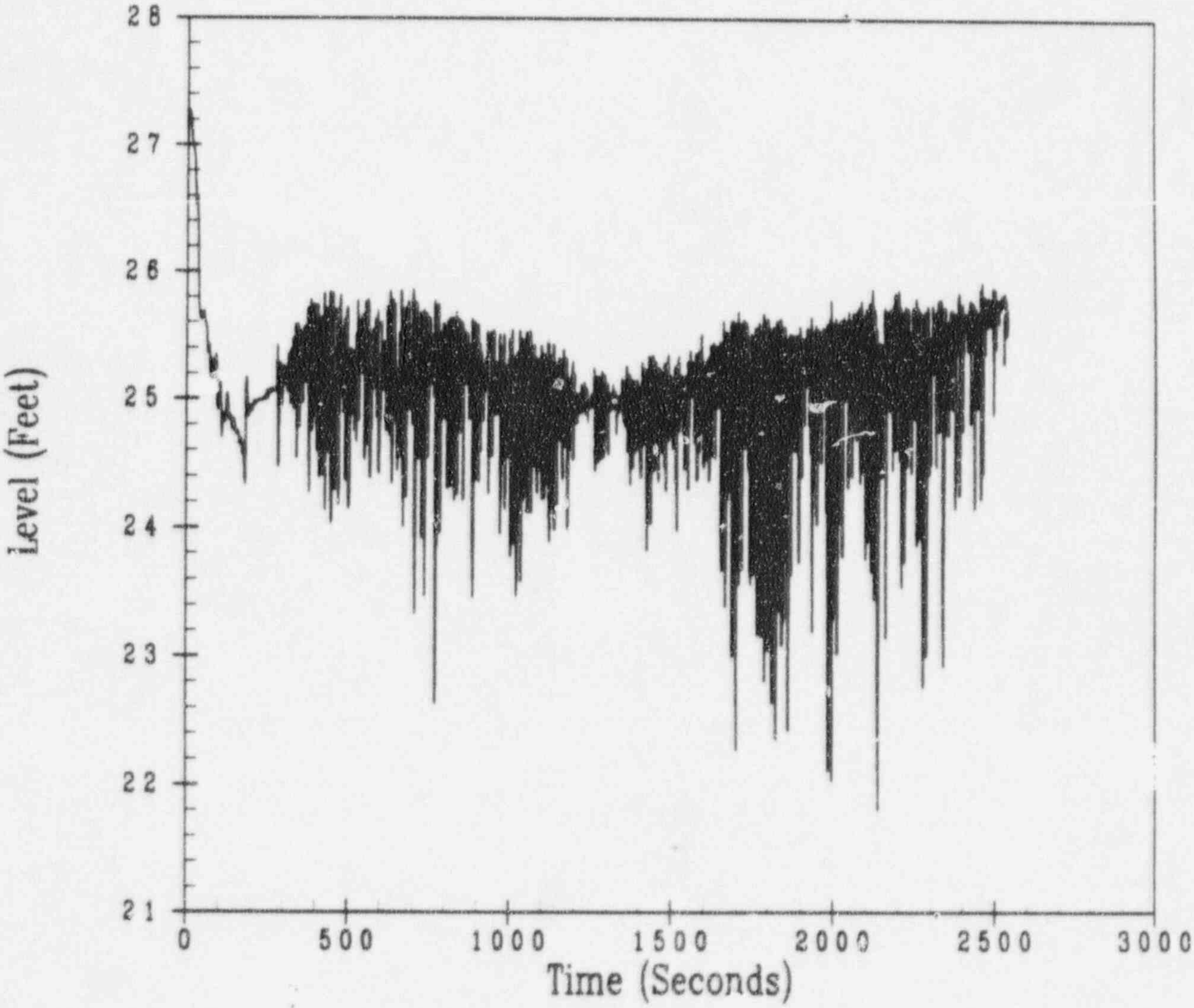



Figure 12