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July 2 , 1981
L1L 197

Office of Nuclear Reactor Regulation
Attn: John F. Stolz, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Loft Test L3-1

As discussed in our letter of April 22, 1981 (L1L 120) in response to your letter of February 24, 1981, enclosed please find a copy of B&W Post Test Evaluation of LOFT Test L3-1. B&W's Post Test Analysis for Semiscale Test (S-07-10D) will be provided by July 15, 1981.

Sincerely,

H. D. Hukill

H. D. Hukill
Director, TMI-1

HDH:LWH:hh

Enclosure

cc: B. H. Grier
D. DiIanni

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B&W's Post Test Evaluation of
LOFT Test L3-1

Document No. 51-1125988-01

May 1981

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Prepared by

BABCOCK & WILCOX COMPANY

for

The Owners Group of Babcock & Wilcox

177 and 205 FA NSS Systems

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TABLE OF CONTENTS

1. INTRODUCTION

B&W's pretest predictions for LOFT Test L3-1 was submitted to the NRC by Reference 1. The test was designed to provide data for a PWR simulated 2.5 percent single ended break in the cold leg with ECCS injection limited to the intact loop. The break was large enough to cause system depressurization.

The test has been completed and the comparisons of pretest predictions to the test data have been published in Reference 2. Inspection of this comparison shows that certain discrepancies exist between the pretest predictions and the test data. Based on these discrepancies, Mr. R. W. Reid of the NRC sent a letter to all Babcock & Wilcox Licensees (Reference 3) requesting the submission of a post test evaluation of LOFT Test L3-1 and Semiscale Test S-07-10D. According to the letter, the objectives of this post test evaluation are:

1. Evaluate the code predictive capability using initial and boundary conditions consistent with the actual test data,

2. Identify code modifications and/or improvements necessary to predict the test data,

3. Assess whether any improvements and/or modifications necessary for code predictions to agree with test data should be incorporated in present ECCS small break evaluation models,

4. Identify shortcomings in the test facility, instrumentation, etc., and their impact on code prediction capability, and recommend improvements to the test facility, instrumentation, or test procedures to improve the verification process.

This report is in response to Mr. Reid's letter. It provides B&W's post test evaluation of LOFT Test L3-1 and addresses the above mentioned objectives. The post test evaluation of S-07-10D is provided in Reference 4.

2. SUMMARY

This report is a partial response to Mr. R. W. Reid's letter requesting the submission of a post test evaluation of LOFT Test L3-1 and Semiscale Test S-07-10D. This report provides the post test evaluation of the LOFT 3-1 Test.

LOFT Test L3-1 was a 2.5 percent single ended cold leg break with a nuclear core initially operated at a power of 50 MW. The pretest prediction of the test was performed using the CRAFT2 computer code and was submitted to the NRC by Reference 1.

The pretest prediction performed by B&W compares favorably with the test data. However, it is recognized that the comparison shows that cetain discrepancies exist between the pretest predictions and the test data. A detailed review of the results of L3-1 is performed and the sources of these discrepancies are identified. Qualitative justification is provided in this report demonstrating that if these sources of discrepancies were eliminated, the prediction would be in better agreement with the test data.

Relative to the objectives of the post test evaluation identified in Reference 3, the post test evaluation of LOFT Test L3-1 confirmed that the CRAFT2 computer code can predict the small break LOCA phenomena observed in the test, provided that adequate test conditions are provided. No code modifications are believed to be necessary for the prediction. In order to improve the verification process, it is suggested that a similar approach

to that utilized for the recent L3-6 prediction be employed. That approach consisted of setting up a "blind pretest" model. release of the test data, and then a post test evaluation where the changes from the "blind pretest" model must be justified.

3. DISCUSSION OF RESULTS

3.1 Pretest Prediction

Via Reference 1, B&W's pretest prediction of LOFT Test L3-1 was submitted to the NRC. A detailed comparison of the prediction to the actual data was performed by EG&G and is documented in Reference 2. As is shown by that report, the B&W prediction compares favorably to the experimental data.

LOFT Test L3-1 was a 2.5 percent single ended cold leg break with a nuclear core intially operated at a power of 50 MW. The major components of the LOFT facility are shown in Figure 1. The CRAFT2 noding diagram utilized for the L3-1 pretest prediction is shown in Figure 2 and described in Table 1.

The LOFT L3-1 system description and the initial conditions used for the pretest prediction were obtained from References 5 and 6. During the L3-1 test, some of the actual conditions were different than the initial conditions provided by EG&G before the test. These differences between the initial conditions and the actual conditions are described below:

1. In the test, the auxiliary feed pump was started at 75.0 seconds while in the B&W pretest prediction, the auxiliary feedwater was started at 60.0 seconds based on the initial conditions provided by EG&G.

2. The specified initial accumulator pressure in Reference 6 was 600 psig. The actual measured pressure of the accumulator during the test was 633 psig.

3. During the test, two leakage paths occurred, which were not identified to the participants who were attempting to predict the test. Hence, these leakage flow paths were not modeled in the B&W's pretest prediction model.

4. No specific information about the steam flow control valve was provided in the initial conditions, e.g. flow area, the percent initially open, how fast the valve closed. No specification on leakage through the steam flow control valve was provided. Examination of more recent LOFT Tests, L3-7 and L3-6, indicate that there is some leakage through the main steam control valve when it is in its fully closed position (Reference 7).

   Based on conversations with EG&G personnel, the steam valve was assumed to be 75 percent open, and to close at a rate of 5 percent per second to the fully closed position. The effective steady-state flow area was calculated based on the Moody Critical Flow Model and area was changed as a function of secondary side pressure. No leakage through the valve was allowed when the valve was in its fully closed position.

As can be seen, there are several significant discrepancies between the initial conditions assumed in the analysis and the actual system conditions of the experiment. The impact of these differences on the prediction is described in the following section.

3.2 Post Test Evaluation

In this section, the comparison of B&W's pretest prediction with the test data is provided. Along with this comparison, the discrepancies between B&W's prediction and the test data are explained based upon the differences in assumed and actual system conditions described above.

3.2.1 Steam Generator Secondary Pressure

In this test, the steam generator secondary side pressure quickly rose after the reactor scram due to the closing of the steam flow control valve. In the test this valve did not open during the initial 1500 seconds of the transient. The auxiliary feed pump was started at 75 seconds cooling the mixture in the steam generator.

The steam generator secondary side pressure comparison of the pretest prediction and the test data is shown in Figure 3. It is obvious that the calculated secondary pressure exhibited the same trend as the test data; however, the pressure was underpredicted slightly for the first 250 seconds and then overpredicted from 400 seconds on.

As explained earlier, in B&W's pretest prediction, the steam flow control valve was assumed to be 75 percent open and to close at a rate of 5 percent per second to the fully closed position. The auxiliary feedwater was started at 60 seconds.

In the actual test, the auxiliary feedwater was started at 75 seconds. No specific information about the steam flow control valve performance is provided to compare it with the B&W's pretest prediction model assumption.

By modeling the steam flow control valve exactly as it behaved in the actual test, the steam generator pressure prediction is expected to improve. It is felt that the steam flow control valve was probably open less than 75 percent initially and thus, was fully closed earlier than calculated. The initiation of auxiliary feedwater at 75 seconds, as it happened in the test, is expected to bring the predictions in better agreement with the test data during the early part of the transient because the steam generator cooling will be delayed.

Examination of recent LOFT tests indicate that there is leakage through the steam flow control valve when it was in its fully closed position (Reference 7). In the B&W pretest prediction model, no leakage through steam flow control valve was allowed. It is expected that there was probably leakage in the L3-1 test. Inclusion of steam flow control valve leakage would result in a faster steam generator depressurization more typical of the experimental data.

3.2.2 Intact Loop Accumulator Pressure

The accumulator injection during the test was initiated at 634 seconds after the rupture. The measured initial pressure was 633 psig. The specified initial accumulator pressure in Reference 6 was 600 psig.

Figure 4 shows the comparison of the B&W's calculated accumulator pressure to the test data. As can be seen, B&W's predicted accumulator pressure response shows a trend similar to the test data indicating that the accumulator emptied at a rate similar to the experiment. The discrepancy prior to the initiation of the accumulator is due to the difference between the initial conditions given and the actual test conditions. B&W's pretest prediction resulted in the initiation of the accumulator flow at 660 seconds. The initiation of accumulator flow just due to the proper initial

accumulator pressure alone will result in $\sim$100 second earlier initiation of accumulator flow. Also, the expected improvement in the system pressure prediction by utilizing the actual test conditions in the pretest prediction model (as discussed in Section 3.2.4) will further improve the predicted time of accumulator actuation.

3.2.3 Break Flow

Figure 5 compares the leak flow rate of B&W's pretest prediction and the test data. The predicted break flow exhibits the trend of the data for the initial 50 seconds and from $\sim$500 seconds on. The pretest prediction over these time frames is within the range of the test data. The predicted break flow between 50 to 250 seconds is essentially constant and significantly higher than the test data. The predicted break flow then sharply drops and is slightly lower than the test data up to $\sim$500 seconds.

This discrepancy between the predicted break flow and the test data is primarily due to the leakage paths that existed in the test. During the test, leakage through the cold leg warm-up line, reflood assist bypass valves and from hot leg to downcomer occurred. These leakage paths were not identified to the participants who were attempting to model the test. Because of these leakage paths, there was a direct path for the hotter fluid of the system to be transported and mixed with the fluid at the break. Consequently, subcooled liquid discharge will end earlier and there will be a smoother transition through two phase flow to steam, which was typical of the actual experiment data.

In B&W's pretest prediction model. these leakage flow paths were not modeled. Therefore, the predicted break flow during the first 250 seconds of the transient is subcooled liquid as no steam relief path to the break is available until the loop seal is cleared. The clearing of the loop seal will decrease the break flow sharply as a result of a sudden decrease in the density of the fluid at the break. The slightly underpredicted break flow between 250 to 500 seconds is due to the discharge model utilized in the prediction model and as a result of the earlier difference between the prediction and the test data. The Bernoulli-Moody Discharge Model with a discharge multiplier of 0.6 for subcooled and two-phase flow was utilized in the prediction. The discharge multiplier was changed to 0.9 when the break flow stabilized at pure steam. In order to predict the two-phase discharge flow precisely, the discharge multiplier should be changed as a function of break fluid quality. The expected transient of the test without the leakage flow paths was a long duration subcooled liquid discharge followed very quickly by a transition to steam. Hence the two-phase discharge was expected to occur only for a short time during the transient. Based on this, the discharge multiplier for the entire two-phase discharge flow was kept constant at 0.6. The discrepancy between the predicted break flow and the test data during 250 to 500 seconds is essentially due to the constant discharge multiplier of 0.6 for the entire two-phase flow and the earlier difference between the prediction and the test data.

In short, the predicted break flow compares reasonably well with the test data. The prediction is expected to improve significantly if the actual test conditions (leakage paths) and the more appropriate discharge model for the test are utilized in the prediction model.

3.2.4 System Pressure

In the test, the system pressure was characterized by a rapid depressurization to the system fluid saturation pressure. The depressurization rate then decreased until 300 seconds when the system liquid level decreased below the break orifice, allowing steam to escape and the system to depressurize more rapidly. From 1000-15000 seconds depressurization rate was decreased as a result of the liquid level in the broken loop increasing above the break orifice.

Figure 6 shows the comparison of B&W's pretest prediction with the test data. The prediction generally shows the same trend as the test data and is in good agreement with the test data. However, it is recognized that the pressure is slightly under predicted from 50 to 200 seconds and over predicted from 600 to 1100 seconds.

The initial underprediction of the pressure is mainly due to the fact that steam generator secondary side pressure was underpredicted during this time frame. During this period of time, the secondary side pressure controlled the primary side pressure. Thus, the expected improvement that could be obtained in the steam generator pressure would improve the primary system pressure comparison. The long term overprediction of system pressure is due to a combination of various factors. As mentioned earlier, the pretest prediction did not include the leakage paths. With the addition of these leakage paths, as explained in Section 3.2.3, the break flow prediction can be improved and this would improve the system pressure with the test data.

As discussed in Section 3.2.2, B&W's pretest prediction of accumulator initiation was later than the acutal test. This was partially a result of the difference between the initial condition given and the acutal test condition of the accumulator. With the initiation of the accumulator, steam will be condensed and hence a more rapid system depressurization will take place. The delayed predicted accumulator actuation contributed to the over prediction of the system pressure.

The steam generator secondary side became a heat source in B&W's pretest prediction at about 400 seconds. During the pretest prediction, no leakage through the steam flow control valve was allowed once it was in its fully closed position. If in the actual test, leakage through the steam valve existed once it was in its fully closed position, a portion of the secondary side energy would be released through the leakage and hence less energy would be added to the primary system. This also contributes to the over-prediction of the system pressure in B&W's pretest prediction model.

In short, B&W's pretest prediction of the system pressure compares reasonably well with the test data. The difference between the prediction and the test data is a result of the differences between the assumed and actual test conditions.

4. CONCLUSIONS

In general, B&W's pretest predictions of the L3-1 test are in good agreement with the test data. Review of the comparisons of B&W's pretest prediction and the test data by EG&G in Reference 2, reveals that the predictions exhibited the same trend as the data for all the parameters compared. It is recognized that the comparison shows that certain discrepancies exist between the predictions and the test data. The differences between the predictions and the data are due to differences in the assumed and actual system conditions. The qualitative analysis provided in this report demonstrates that incorporation of the actual experimental conditions would result in an improved prediction.

Relative to the objectives of the post test evaluation, as stated in the February 24, 1981 Reid letter (Reference 3), the following conclusions are drawn:

1. The evaluation of LOFT Test L3-1 confirmed that the CRAFT2 computer code is capable of predicting the small break LOCA phenomena observed in the test if the actual test conditions were utilized.

2. No code modifications are believed to be necessary to predict the test.

3. Since no code modifications are necessary to predict the test, no modifications to the present ECCS small break evaluation model have been identified from this experiment.

5. REFERENCES

1. Letter from J. H. Taylor to D. F. Ross, "B&W's Best Estimate Prediction of the LOFT L3-1 Nuclear Small Break Test Using the CRAFT2 Computer Code," December 13, 1979.

2. Czapary, L. S., "LOFT L3-1 Preliminary Comparison Report" EGG-CAAP-5255, dated September 1980.

3. Letter to All Babcock & Wilcox Licensees from R. W. Reid, Chief Operating Reactor Branch #4, Division of Licensing, February 24, 1981.

4. T. E. Geer, et.al., B&W's Post Test Analysis For Semiscale Test S-07-10D Doc. No. 86-1125888-00, May 20, 1981.

5. LOFT System and Test Description (5.5 ft. Nuclear Core 1 LOCES), NUREG/CR=0274, TREE-1208, Reeder, D. L. dated July 1978.

6. Information for L3-1 Standard Problem, Kaufman, N. C., Kau-203-79, dated September 28, 1979.

7. Modro, S. M & Condil, K. G., "Best Estimate Prediction for LOFT Nuclear Experiment L3-5/L3-5A," EGG-LOFT-5240, dated September 1980.

Table 1. Node Description

Node No.	Description
1	Downcomer annulus
2	Downcomer
3	Lower plenum
4	Core
5	Upper plenum & upper head
6	Pressurizer
7	Hot leg, intact loop
8	Steam generator, front half of primary side, intact loop
9	Steam generator, back half of primary side, intact loop
10	Steam generator outlet, cold leg piping, intact loop
11	Pump suction, intact loop
12	Pump discharge, intact loop
13	Hot leg plus half simulated SG, broken loop
14	Half simulated SG plus half pump, broken loop
15	Half pump plus piping, broken loop
16	Leak node
17	Suppression tank
18	Secondary side, intact loop

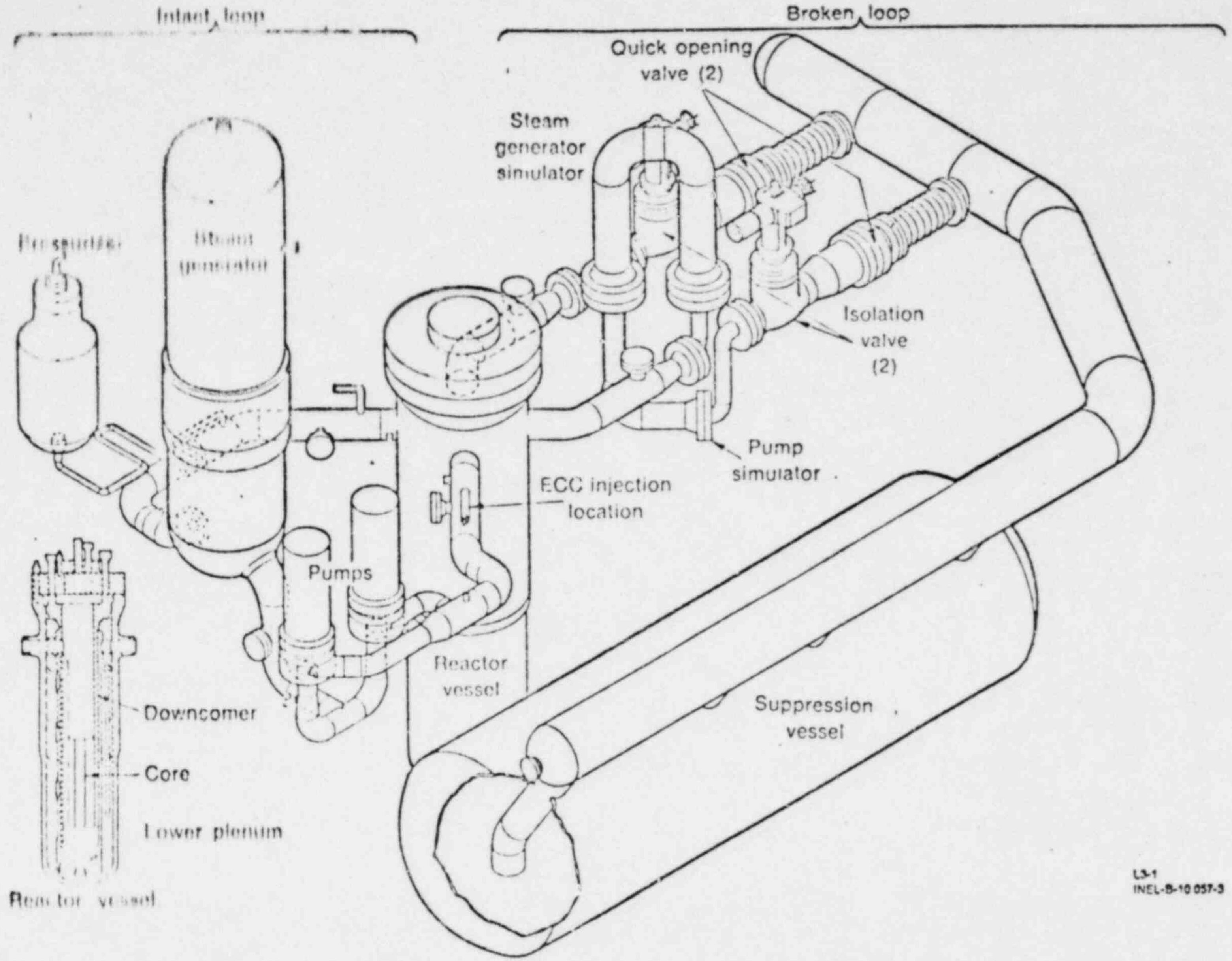
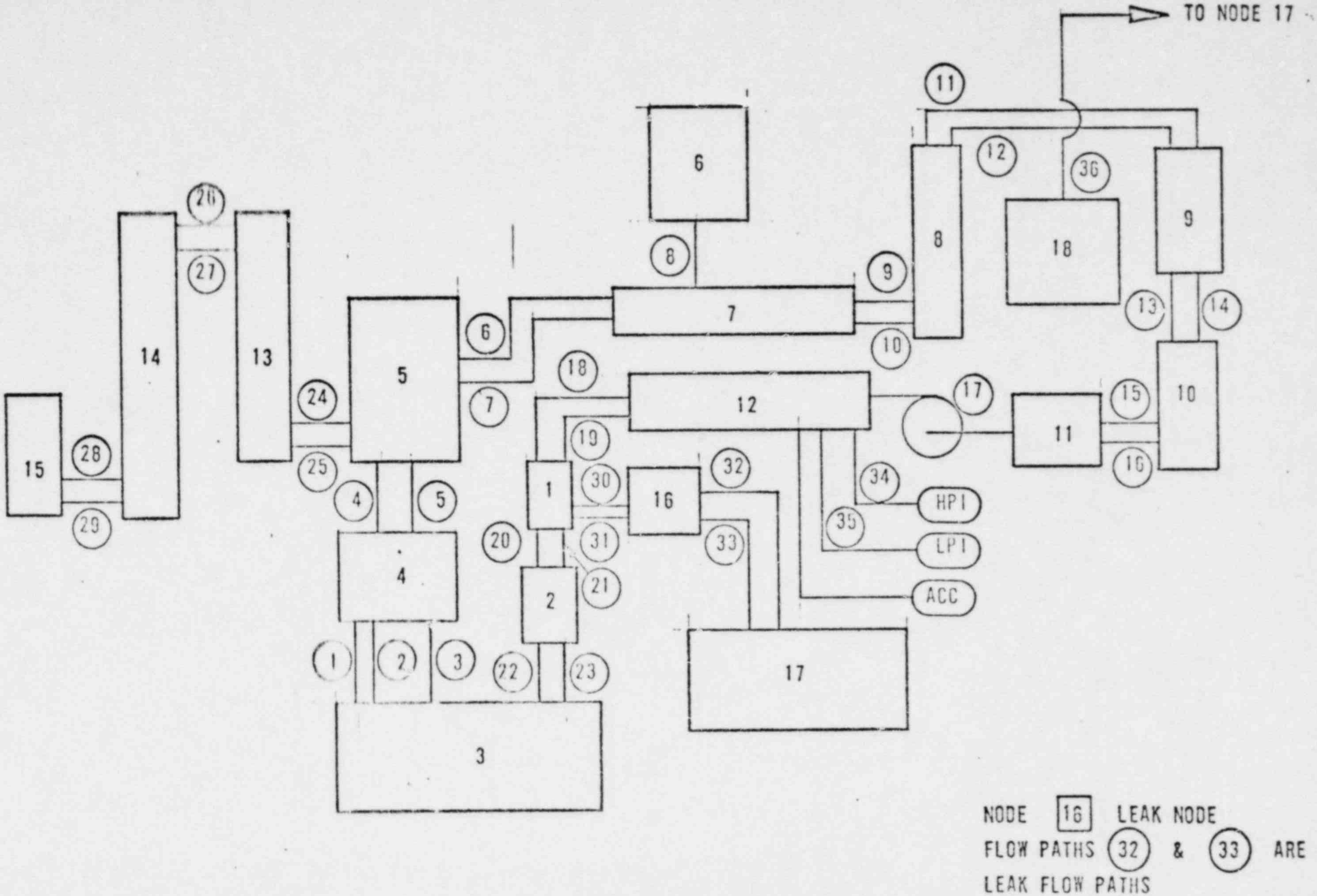
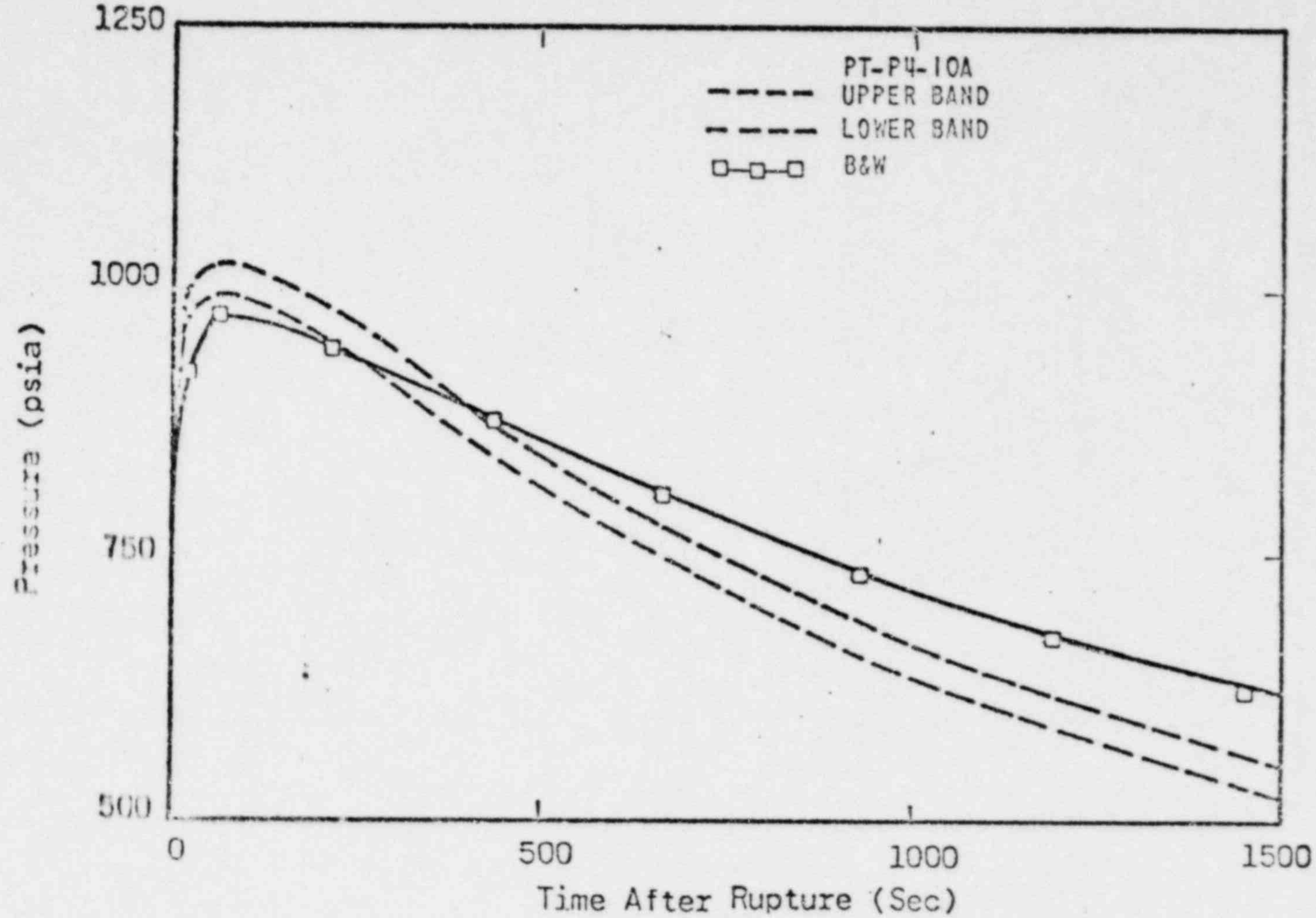


Figure 1. LOFT major components.



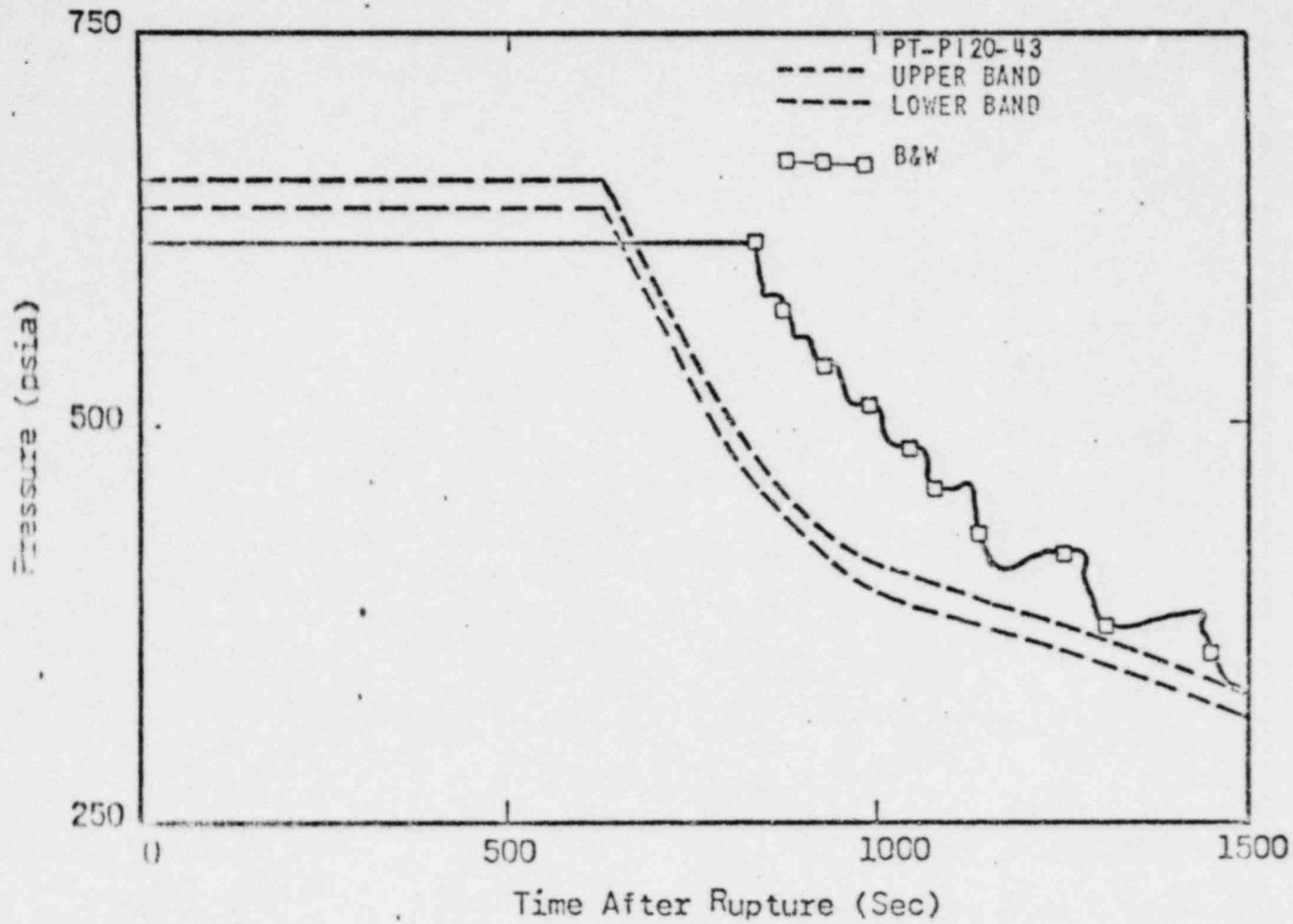
LOFT L3-1 NODING DIAGRAM

Figure 2



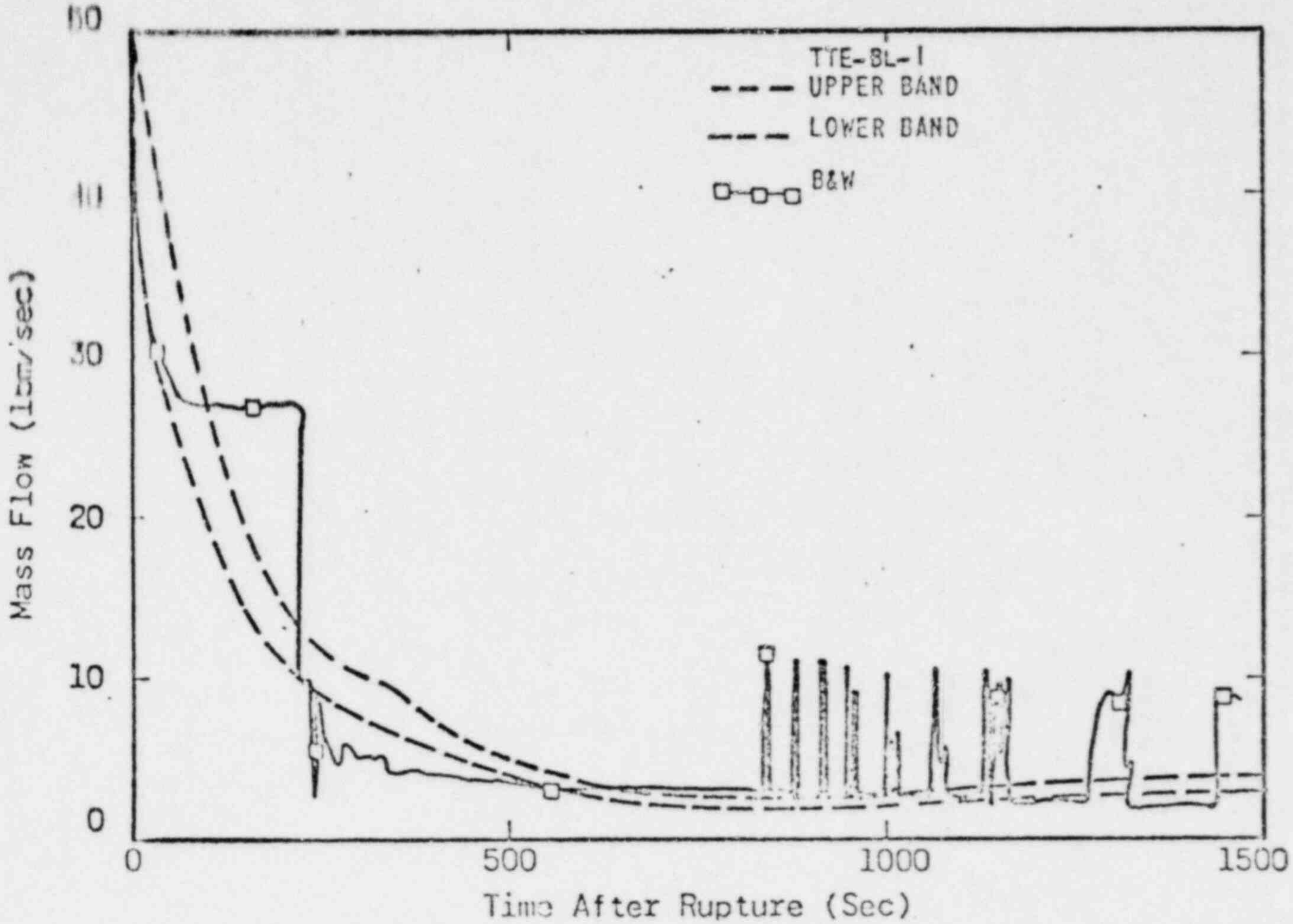
COMPARISON OF CALCULATIONS OF STEAM GENERATOR SECONDARY SIDE PRESSURE TO DATA

Figure 3



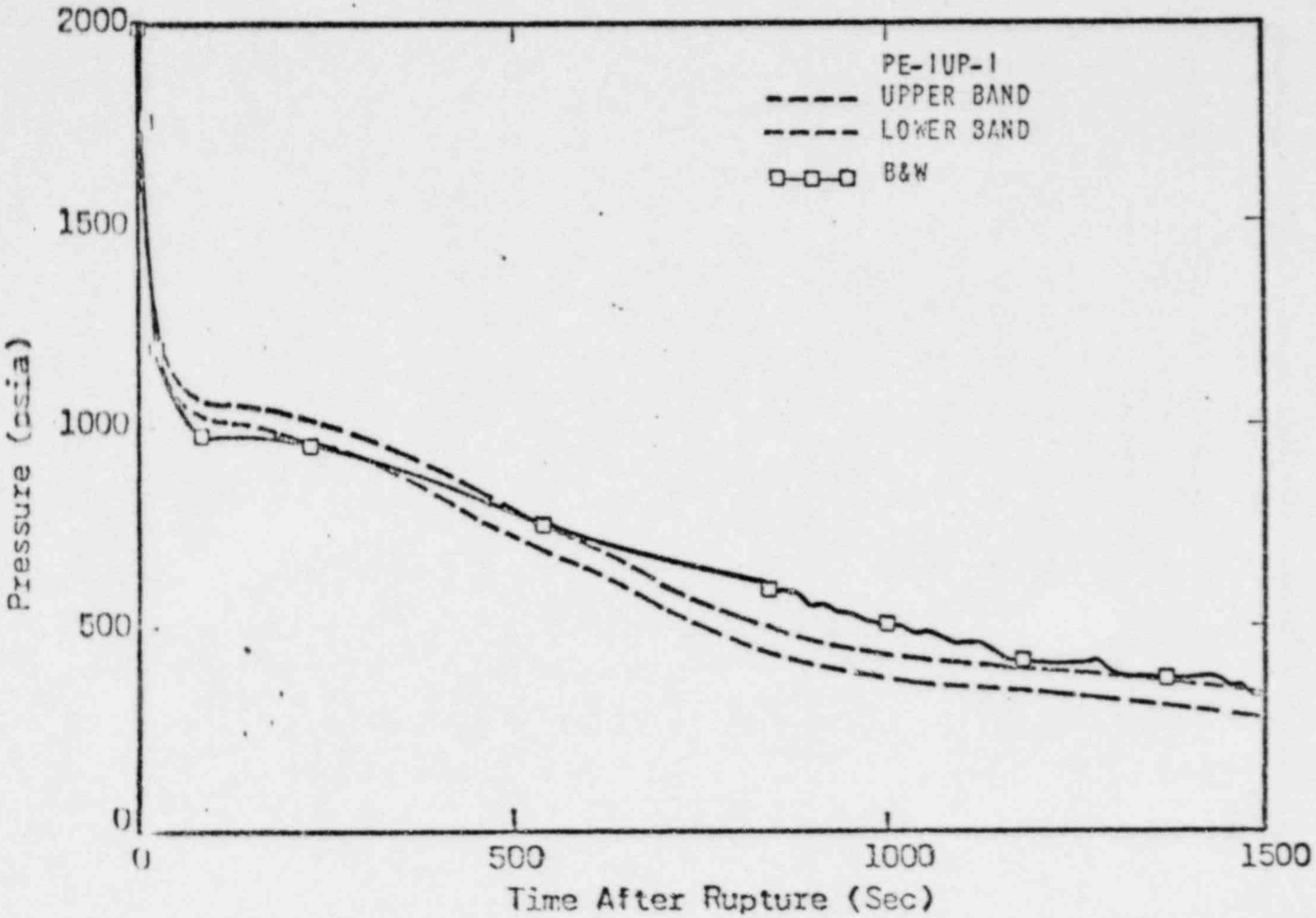
COMPARISON OF B&W CALCULATIONS OF INTACT LOOP ACCUMULATOR PRESSURE TO DATA

Figure 4



COMPARISON OF B&W CALCULATIONS OF BREAK MASS FLOW TO DATA

Figure 5



COMPARISON OF CALCULATIONS OF UPPER PLENUM PRESSURE TO DATA

Figure 6