This testimony of Brian W. Sheron and Walton L. Jensen, Jr. presents the NRC Staff's response to the Appeal Board's Questions 2, 4, 5, 6, 7, 9, 10 and 11 (ALAB-708 at 43-44).

SUMMARY

Answer to Question 2: The use of high point vents to promote liquid natural circulation has not been experimentally or analytically studied by the NRC Staff. However, a qualitative assessment indicates that the vents may be useful to restore interrupted liquid natural circulation in two situations.

Answer to Question 4: Except for the nodes added to the description of the hot leg piping to allow better representation of steam formation and natural circulation interruption, the approved and the modified ECCS evaluation models are identical. Both models predict that the boiler-condenser process would be effective in removing decay heat.

Answer to Question 5: The Staff has reviewed and approved the B&W ECCS evaluation model and the CRAFT-2 computer code contained in the B&W ECCS evaluation model that calculates heat transfer, including heat transfer by the boiler-condenser process, between the primary and secondary sides of the steam generators. The Staff has concluded that the boiler-condenser process will remove sufficient decay heat to prevent core uncovery if at least one train of ECCS is available.

Answer to Question 6: Only breaks slightly smaller than 0.07 ft^2 in area must be analyzed to demonstrate compliance with 10 CFR 50.46 because breaks in that size range produce the highest peak cladding temperature and the greatest amount of core uncovery. Studies performed by B&W have shown that no core uncovery or clad heat-up occurs for breaks much smaller than about 0.07 ft^2 in area.

Answer to Question 7: EG&G - Idaho performed an analysis of a 0.01 ft^2 cold-leg break using RELAP5 to duplicate to the extent possible an analysis performed by B&W in which natural circulation was calculated to be lost and then reestablished in the boiler-condenser mode. The EG&G analysis did not show the boiler-condenser process to be established but did show that decay heat was removed by intermittant establishment of a two-phase "chugging" circulation. Because boiler-condenser natural circulation was not established in the first calculation a second calculation in which boundary conditions were imposed to force boiler-condenser natural circulation to occur was performed, and boiler-condenser natural circulation was calculated to be established at a decay heat removal rate greater than the decay heat production rate.

Answer to Question 9: Feed and bleed is neither relied upon nor needed to remove decay heat for design basis events at TMI-1.

Answer to Question 10: EG&G has compared the results of a post-test RELAP5 computer code analysis of Semiscale test S-SR-2 to the experimental data. RELAP-5 would have acceptably predicted the experimental data had HPI flow characteristics and secondary system heat losses been better accounted for.

Answer to Question 11: The results of a RELAP5 computer code analysis performed by EG&G demonstrate that feed and bleed will successfully provide core cooling at TMI-1. The results of a TRAC computer code analysis performed by LANL for an Oconee unit support that conclusion.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of METROPOLITAN EDISON COMPANY, ET AL. (Three Mile Island Nuclear Station, Unit No. 1)	))))))	Docket No. 50-289 (Restart - Design Issues)

NRC STAFF TESTIMONY OF BRIAN W. SHERON
AND WALTON L. JENSEN, JR. IN RESPONSE TO
APPEAL BOARD QUESTIONS 2, 4, 5, 6, 7, 9, 10 AND 11

Q.1 State your names and positions with the NRC.

A. Our names are Brian W. Sheron and Walton L. Jensen, Jr. Brian W. Sheron is Chief of, and Walton L. Jenson, Jr. is a Senior Nuclear Engineer in, the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. Statements of our professional qualifications were submitted in "NRC Staff Response to Appeal Board Order of October 15, 1982," dated October 25, 1982.

Q.2 What is the purpose of your testimony?

A. The purpose of our testimony is to address Appeal Board Questions 2, 4, 5, 6, 7, 9, 10 and 11. (ALAB-708 dated December 29, 1982). Those questions and our answers are as follows:

Question 2:
When and under what conditions such size vents would or would not be useful to promote liquid natural circulation, including reasons for the conclusions reached?

Answer: The use of high point vents to promote liquid natural circulation has not been evaluated either analytically or experimentally in detail by the staff. Therefore, the following discussion is a qualitative rather than a quantitative one and is subject to analytical and/or experimental confirmation.

High point vent operation to restore interrupted liquid natural circulation is expected to be useful in two situations. The first situation, and that for which the vents are to be installed, involves the removal of non-condensible gases that could become trapped at the top of the hot leg inverted U-bends as a result of an inadequate core cooling condition. The gas that is removed would have to be replaced by reactor system coolant or the pressure decrease associated with the venting would lower reactor system pressure to the saturation pressure at the prevailing coolant temperature and cause liquid coolant to flash to steam. If this steam, along with any steam produced by boiling in the core, were to accumulate at the hot leg high points, liquid natural circulation might not be readily restored.
The second situation is similar to the first but involves removing a steam bubble rather than non-condensible gas from the top of the hot legs . During the recovery period of a small break LOCA in which the primary system is refilling with subcooled water, or during the recovery from any transient event which results in steam bubble formation at the

top of the hot legs and in which the primary system is refilling with subcooled water, opening the high point vents would aid in the recovery of liquid natural circulation. By opening the vents, the steam would be removed and readily allow subcooled water to replace the volume occupied by the steam. If sufficient subcooling were to exist, then no flashing would be expected and the system would be expected to refill liquid solid and provide a liquid natural circulation flow path between the core and the steam generator.

There are some conditions under which operation of the hot leg high point vents is not expected to be useful. For example, during certain small break LOCAs, the initial net inventory loss from the reactor coolant system will result in steam voids at the system high points, including the top of the hot leg U-bends. These steam voids may interrupt natural circulation. If the hot leg high-point vents were to be opened in an attempt to restore liquid natural circulation, they would produce a local depressurization and could cause liquid coolant to flash to steam. This steam would rise to the top of the hot leg U-bends to replace the steam being vented. Liquid as well as steam might be expelled when the vents were opened and would further deplete reactor coolant. In general, during any conditions of reduced primary system inventory that results in uncovering the hot leg U-bends, liquid natural circulation will be interrupted and cannot be restored until primary system inventory is restored sufficiently to refill the hot-leg U-bends, opening of the hot leg vents for conditions of inadequate reactor system inventory would result in additional coolant loss and would not promote liquid natural circulation.

Question 4:

Whether the modified B&W ECCS evaluation model for small breaks that predicts the boiler-condenser process is an NRC approved code under Appendix K to 10CFR Part 50?

Answer: It is our understanding that the "modified B&W ECCS evaluation model" referred to by the Appeal Board is the one used by B&W to perform some of the calculations documented in Licensee Exhibit 5. In that exhibit the licensee presented the results of LOCA analyses for break sizes smaller than those which had been previously evaluated to demonstrate compliance with Appendix K to 10CFR 50.

ECCS evaluation models include computer codes, the model description of the reactor system and assumptions about the performance of the engineered safety features. The modified B&W ECCS evaluation model is identical to the NRC approved ECCS evaluation model with the exception that additional nodes have been added to the description of the hot leg piping to allow steam formation and natural circulation interruption to be better represented. The computer code (CRAFT-2) which is used in both the modified B&W ECCS evaluation model and the NRC approved ECCS evaluation model is the same. The CRAFT-2 computer code contains the equations and assumptions for heat transfer, including heat transfer by the boiler-condenser process, between the reactor system and the steam generator. Therefore the equations and assumptions dealing with the boiler condenser process which are utilized in the B&W modified evaluation model have been approved by the NRC under Appendix K to 10CFR50. It is only because of the additional noding that the modified evaluation model is not considered staff approved, since we originally approved a different nodal description. Exhibit 5 presented analyses of small break LOCAs utilizing both the modified B&W ECCS evaluation model and the NRC approved B&W ECCS evaluation model. Both models predicted that the boiler-condenser process would be effective in removing decay heat if a condensing surface were uncovered within the steam generators.

Question 5:

Whether the staff has reviewed the B&W Appendix K model to determine the ability of the code to calculate the effects of small breaks, including reliance upon boiler-condenser circulation?

Answer: The equations and assumptions dealing with heat transfer between the reactor system and the steam generators including heat transfer by the boiler-condenser process are contained in the CRAFT-2 computer code which is part of the B&W ECCS evaluation model. The B&W ECCS evaluation model and the CRAFT-2 code that is included in the model have been reviewed and approved by the NRC Staff. Following the TMI-2 accident, B&W performed a number of small break LOCA calculations for break sizes smaller than those which had been evaluated to demonstrate compliance with Appendix K to 10CFR50. These calculations indicated that for certain small break sizes, heat removal by the boiler-condenser process would be required to remove decay heat from the reactor system. The calculations were performed to provide a basis for revisions to small break LOCA emergency procedures. The staff did not re-review the equations and assumptions contained in the CRAFT-2 code at that time. The staff did perform audit calculations of small breaks in B&W designed plants using the RELAP-4 computer code. These calculations are documented in NUREG-0565. As did the B&W analyses, RELAP-4 predicted the boiler-condenser mode of natural circulation to be effective in removing decay heat and providing continued core cooling.

The staff has concluded that the heat transfer mechanisms involved in the boiler-condenser process are adequate to remove decay heat from the reactor system and will prevent core uncovery if at least one train of ECCS is operable. This conclusion is based on both the B&W CRAFT-2 calculations and the RELAP-4 audit calculations, as well as our evaluations of the heat transfer mechanisms involved in the process and discussed in commonly available heat transfer texts. Although detailed reactor coolant system behavior during the period of natural circulation interruption in the analysis of certain small break sizes is not well understood, the system must eventually drain down and a steam condensing surface in the steam generator would be exposed before the core could begin to be uncovered. Once a steam condensing surface were uncovered, boiler-condenser natural circulation would commence and depressurize the system so that the decreased break flow, along with the increased HPI flow, would result in a net inventory increase in the primary system before the core could begin to uncover. The staff has evaluated the mechanism involved in the boiler-condenser heat transfer process and has concluded that the condensing surface that would be available would be capable of removing all decay heat generated by the core if an adequate supply of feedwater were available.

Question 6:

Whether only breaks slightly smaller than $0.07ft^2$ must be analyzed?

Answer: To demonstrate compliance with the criteria set forth in 10CFR50.46, only breaks slightly smaller than 0.07 ft^2 and above must be analyzed for B&W designed reactors because B&W has shown that the limiting small break LOCA occurs for this break size. A postulated small break size of $0.07ft^2$ was calculated to produce the highest peak cladding temperature and the greatest amount of core uncovery.

Part I(C)(1) of Appendix K to 10CFR50 requires that ECCS performance analyses consider a spectrum of possible breaks. The purpose of this requirement is to ensure that the worst break size is analyzed. For small break LOCAs, compliance with this requirement has historically been demonstrated by selecting a limited spectrum of break sizes to determine the break size which produces the maximum amount and duration of core uncovery, and hence the highest cladding temperatures, amount of oxidation, etc.

Licensing analyses and sensitivity studies performed by B&W over the years have shown that for break sizes much less than about $.07ft^2$, no core uncovery is predicted to occur. Without core uncovery, the cladding will not heat up and will remain only slightly above the coolant saturation temperature since the core will be cooled by pool boiling.

Our confidence that core uncovery will not occur for breaks much less than about $0.07ft^2$ is based on the following considerations:

(1) Analyses performed to date by B&W and our own contractors do not predict any core uncovery for breaks much less than about $0.07ft^2$.

(2) The relative elevation of the condensing surface in the steam generators with respect to the top of the core is such that an ample condensing surface for steam condensation and decay heat removal will be exposed before the primary system inventory would drop below the top of the core. Because of the establishment of the condensing surface and consequential decay heat removal, the primary system will be depressurized so that safety injection flow can exceed break flow and replenish primary system inventory before the core can become uncovered .

(3) Vent valves which allow pressure equalization between the vessel upper plenum and the downcomer assure that the liquid level in the core cannot be significantly mismatched with the liquid level in the steam generators.

Question 7:

Confirmation (such as by means of detailed computational analysis or experimental testing) that boiler-condenser circulation flow will transport sufficient core decay heat to the steam generators to prevent core damage.

Answer: At present, there are no experimental data from a test facility geometrically similar to the B&W reactor design confirming the boiler condenser mode of natural circulation. However, we have recently received a commitment from the B&W Owners Group to enter into a cooperative program with the NRC to construct and operate a test facility at the Alliance Research Center (ARC). We anticipate data from this facility to become available in 1985. We also anticipate that some applicable data may become available from the single-loop GERDA facility, also at ARC, within a few months. The purpose of the testing is not to confirm the effectiveness of boiler condenser decay heat removal.

We are relying on detailed computational analyses being performed by both B&W, the NRC staff and our contractor EG&G, Idaho to demonstrate the efficacy of boiler condenser natural circulation.

In response to the Appeal Board's question, we asked EG&G, Idaho, under contract to and the direction of the NRC staff, to perform a small-break LOCA analysis for the TMI-1 plant using the RELAP5 computer code. This code is an advanced analysis code which accounts for both non-homogeneity and non-equilibrium.

The analysis performed for was an 0.01 ft^2 small break in the cold leg piping. The objective of this analysis was to duplicate, to the extent possible, an analysis performed by B&W and documented in Licensee's Exhibit 5 in which natural circulation was calculated to be lost and then reestablished in the boiler condenser mode.

The following initial conditions were assumed for the analysis:

- only one of the two HPI pumps was assumed operable
- the decay heat used was the draft 1973 ANS standard increased by 20 percent
- only one motor-driven auxiliary feedwater pump was assumed to be feeding

  both steam generators
- reactor coolant pumps were tripped at reactor trip.

Table 7-1 describes this calculated sequence of events.

The results of the analysis were that boiler-condenser natural circulation was not calculated to be established, but rather, decay heat was removed by intermittent establishment of a bubbly, two-phase "chugging" type circulation.

Following the break in the cold leg, the system began to drain since the break flow was primarily liquid and exceeded the HPI makeup flow. Once the top of the hot legs voided and natural circulation was lost, steam generated in the core flowed from the upper plenum through the vessel vent valves into the upper downcomer annulus and cold legs. The HPI injected into the cold legs was not sufficient to condense all of the steam being vented to the cold legs, and a steam "bubble" was calculated to form and grow in the cold legs. This steam bubble displaced water in the cold leg by forcing it into the steam generator. The water that was in the steam generator, however, was not circulating and was continuing to be cooled by heat transfer to the secondary. Thus, as water from the cold leg was pushed into the bottom of the steam generator, it forced the cool water in the generator back up into the downflow side of the hot leg and inverted U-bend. Once this cool water contacted the steam

TABLE 7-1

Sequence of Events

Time (sec) Since Start of Accident	Time (sec) Since Start of Steady State (shown on plot)	Event
0	400	.01 sq ft CL break occurs
37.52	437.5	Scram (based on P-T relationship ANS +20%, RCP trip, AFW initiates
38.53	438.5	Turbine stop valves close
41.92	441.92	Main Feed valves close
55	455	Prz. heaters off (low level)
108	508	HPI on (low prim pressure)
150	550	RCP Coastdown complete natural circ. begins
490	890	Loop A NC lost, Min. press.
700	1100	Loop B NC lost
1100	1500	Cyclic restart & stop of natural circulation
2010	2410	EFW terminated (SG=220")
4037	4437	Calc. over, stable primary pressure. HPI exceeds break flow, vessel level stable

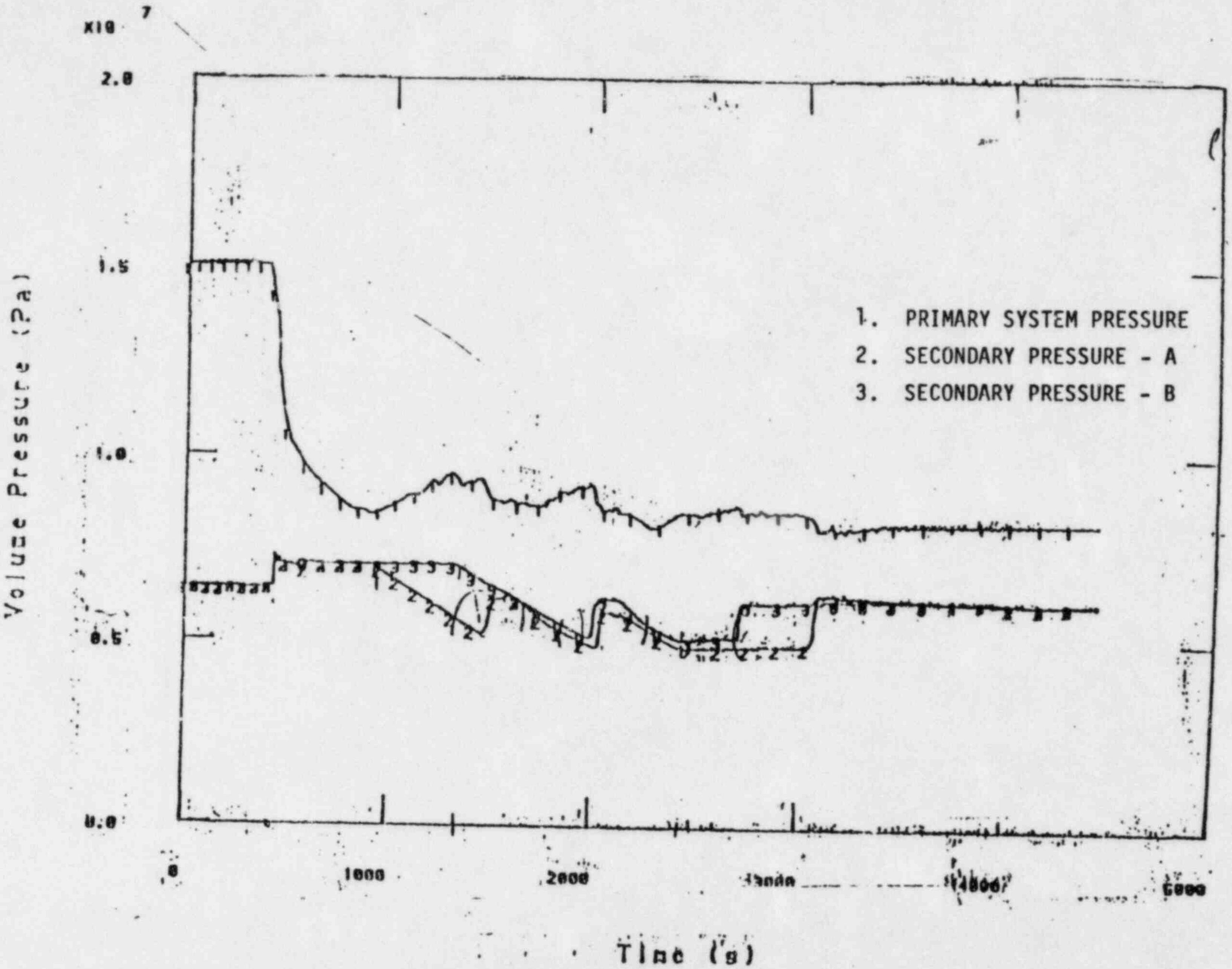
at the top of the U-bend, rapid condensation of the steam occurred, producing a local depressurization and allowed coolant from the hot leg to be forced over the U-bend and into the steam generator.

This behavior is best characterized as a "chugging" behavior and the sudden flow surge was calculated to "mix the system up," sweeping the steam bubble out of the cold leg and allowing the system to collapse back and again uncover the hot leg U-bend and interrupt natural circulation. The process is then calculated to repeat for several more cycles.

In Figures 7-1 through 7-4, key calculated parameters are shown.

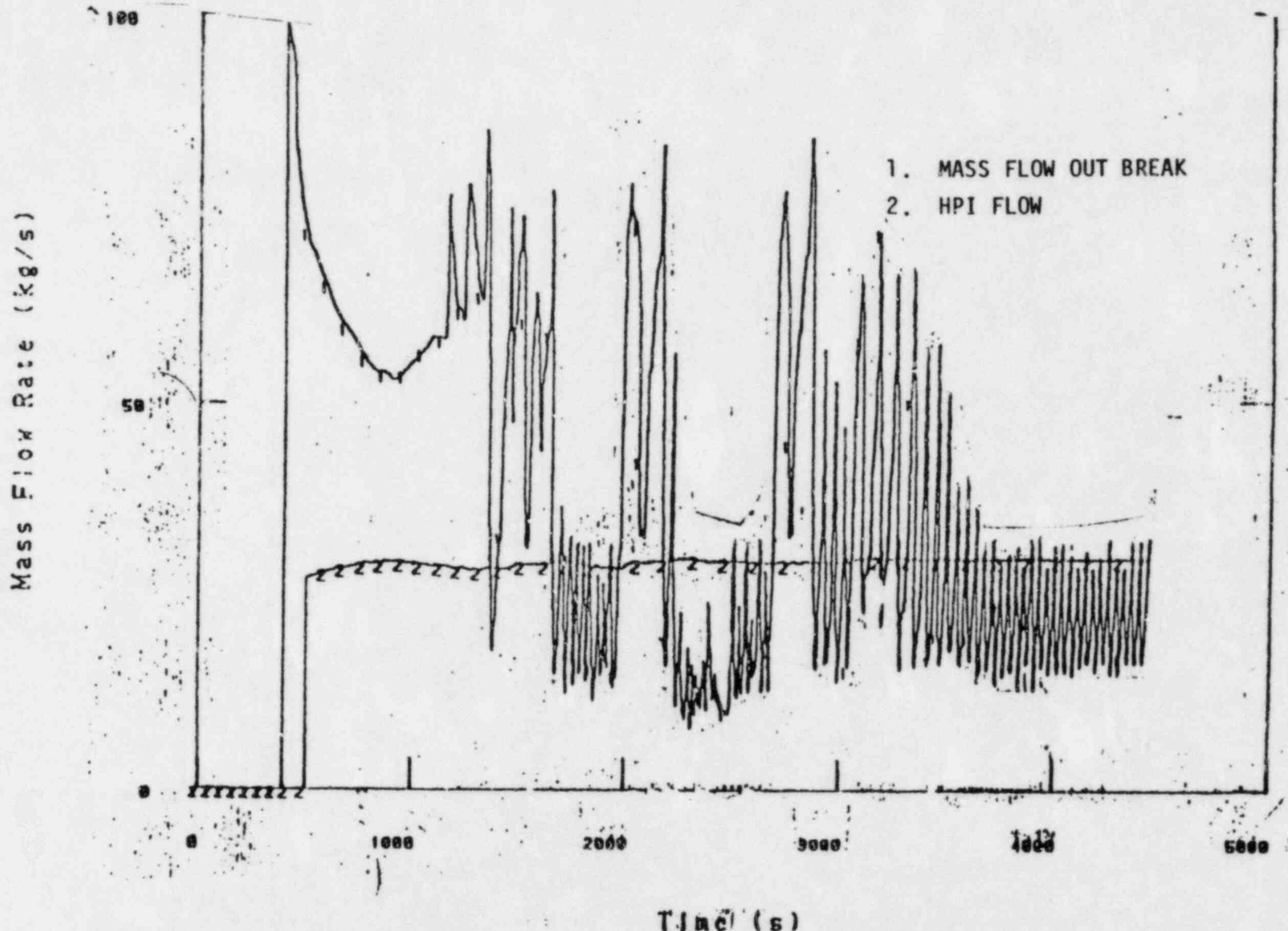
These results show a different system response from the B&W analysis results presented in Licensee's Exhibit 5. The reasons for the differences are not obvious but may result from differences in calculational assumptions. Nodalization is a possible reason for the differences. In addition, the power level in the B&W calculation was 12% lower than the EG&G calculation.

More recently, B&W has submitted a revision to their small break LOCA analysis model in response to TMI Action Plan Item II.K.3.30. These analyses show different system behavior from that predicted by the old model. These differences are believed to be primarily due to the use of a more realistic steam generator heat transfer model. The staff presently has the revised B&W model under review. Therefore, our



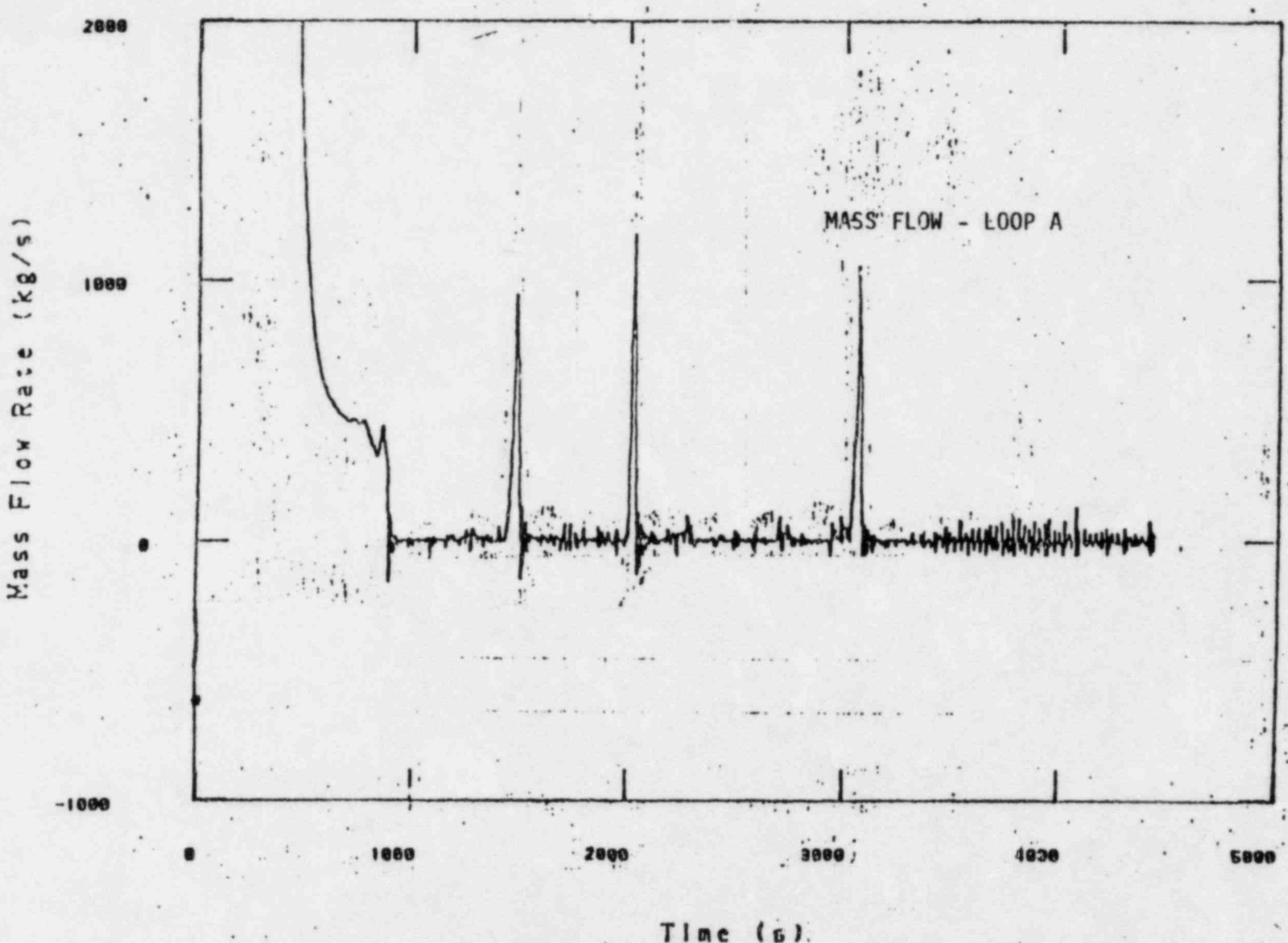
TMI-1 BOILER CONDENSER

Figure 7-1



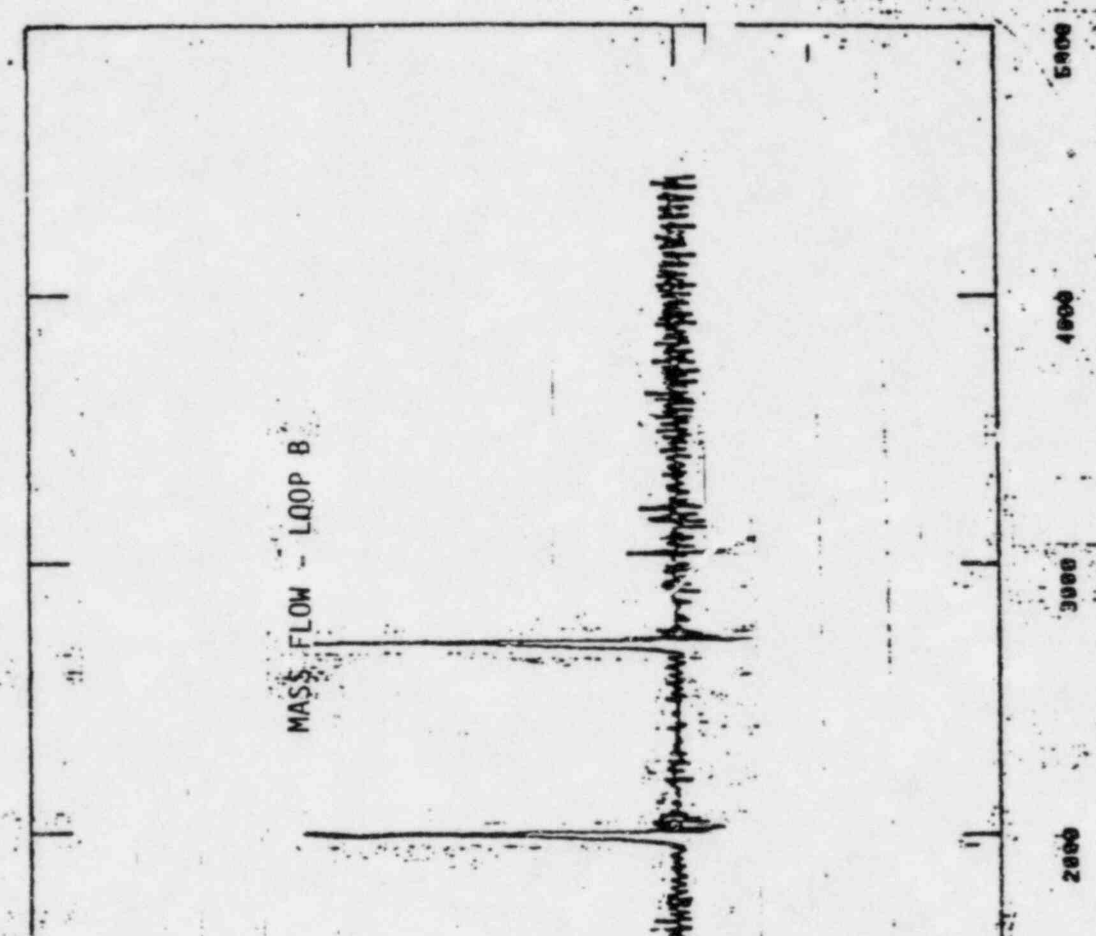
TMI-1 BØILER CØNDENSER

Figure 7-2



TMI-I BØILER CØNDENSER

Figure 7-3



TMI-1, BØILER CONDENSER

Figure 7.4

evaluation of this model revision is not yet complete. However, we recognize that the new model results do not predict any core uncovery.

The staff has long recognized the uncertainties in the behavior of the B&W plants during very small-break LOCAs due to uncertainties in the analysis models. However, we maintain that while these uncertainties affect our detailed understanding of plant performance, they do not change our conclusion that adequate core cooling will not be jeopardized for all small-break LOCAs within the licensing design basis.

Although the 0.01 ft² SBLOCA calculation described above did not result in stable boiler-condenser natural circulation, the staff performed a second calculation with RELAP5 for TMI-1 in which conditions necessary for establishing boiler-condenser natural circulation were imposed on the analysis through the scenario assumption.

A hypothetical transient was initiated in which all feedwater to the steam generators was assumed to be lost. The reactor was calculated to trip on high reactor system pressure after seven (7) seconds. Approximately 300 seconds later, the pressurizer safety valves began to cycle open. After 2000 seconds (33 min.), when the reactor coolant system was highly voided, one motor-driven AFW pump was assumed to be actuated. No ECCS flow was assumed. Figures 7-5 and 7-6 show the reactor system pressure and safety valve mass discharge respectively. As can be seen from these figures, establishment of AFW flow established boiler-condenser natural circulation. The decay heat removal rate was greater than the decay heat production rate as indicated by a sudden

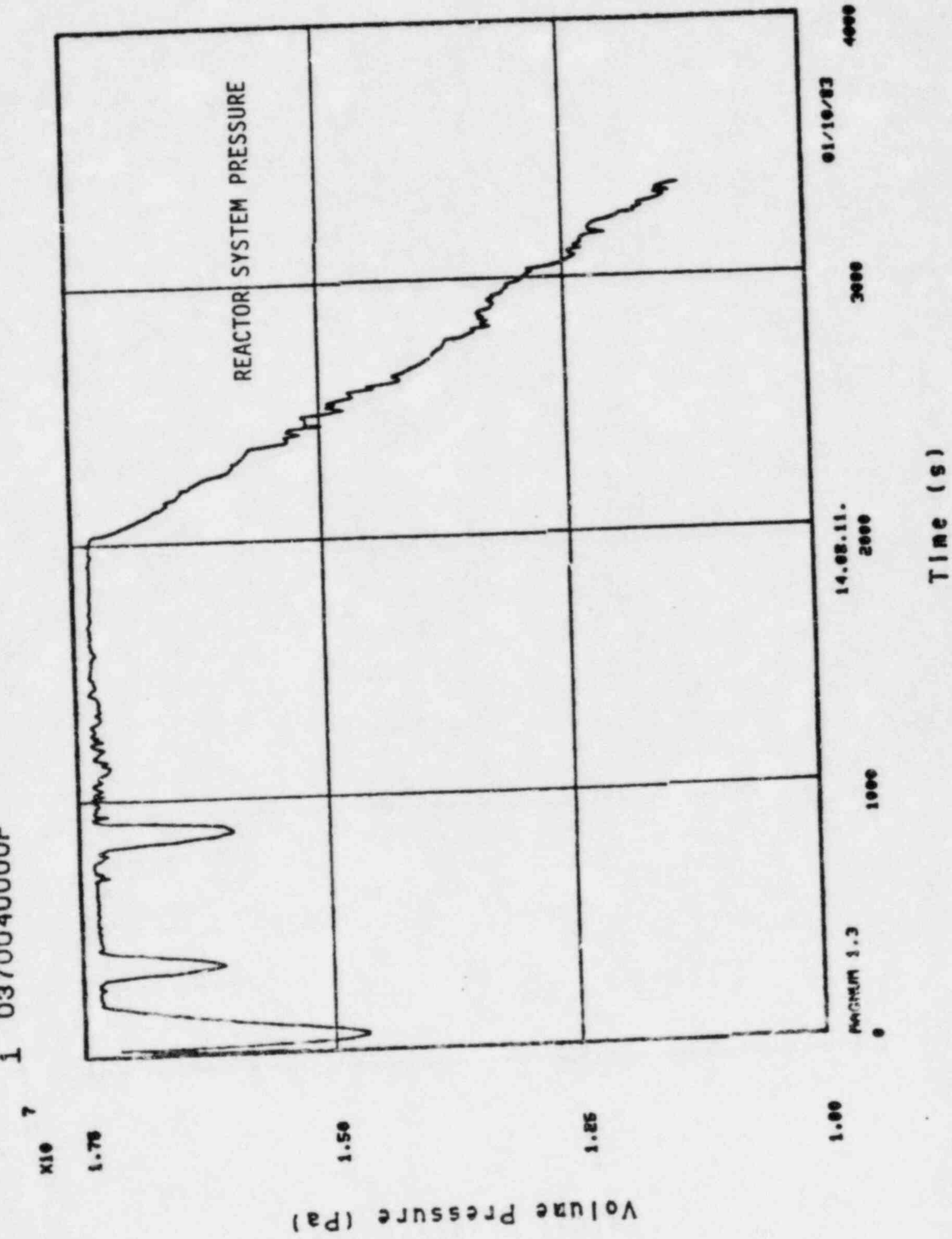

1 03700400000P
X10 7
1.75
1.50
1.25
1.00
Volume Pressure (Pa)
REACTOR SYSTEM PRESSURE
0
1000
2000
3000
4000
14.08.11.
01/10/83
Time (s)

Figure 7-5

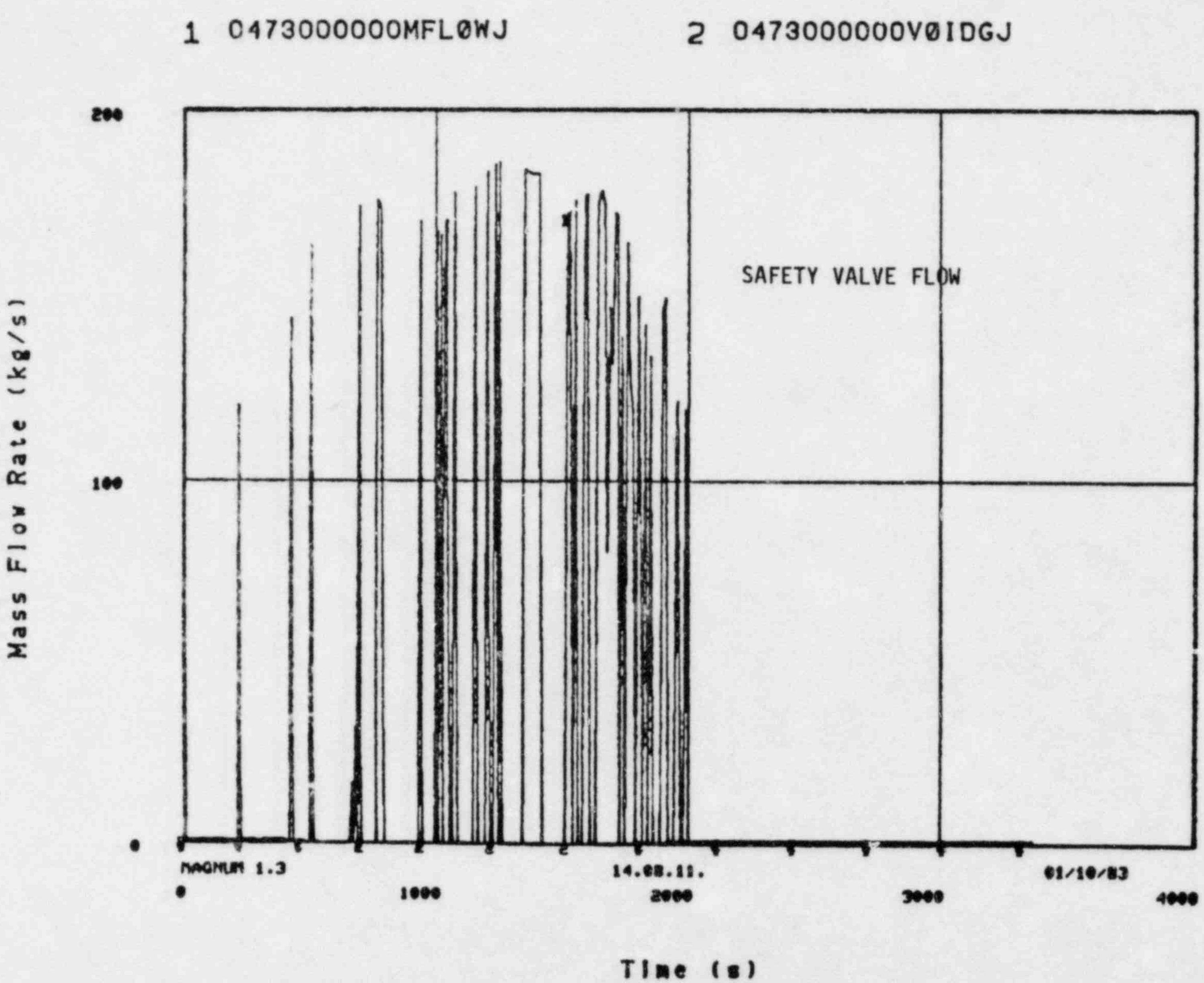

1 0473000000MFLØWJ
2 0473000000VØIDGJ
SAFETY VALVE FLOW
Mass Flow Rate (kg/s)
200
100
0
MAGNUM 1.3
14.08.11.
01/10/83
0
1000
2000
3000
4000
Time (s)

Figure 7-6

decrease in the reactor system pressure and closure of the pressurizer safety valves. The PORV was not modeled. Based on this calculation, the staff concludes that boiler-condenser natural circulation is an effective means for decay heat removal.

As an additional means of demonstrating the heat removal capability of the boiler-condenser mode of natural circulation, the staff evaluated the capability of the TMI-1 steam generators to remove decay heat in the boiler-condenser mode by performing a scoping heat transfer calculation using techniques discussed in commonly available heat transfer texts. The mechanism for boiler condenser heat transfer involves condensation of steam inside the steam generator tubes, heat conduction through the tube walls and boiling of feedwater on the outside of the tube walls. The heat transfer coefficients for all these mechanisms were determined to be high. For a temperature difference between the primary and secondary system of approximately 10°F an overall heat transfer coefficient of 515 BTU/hr-sqft-F was determined. The total heat transfer surface area of the TMI-1 steam generators is 236,020 ft². The resulting heat transfer rate would be 355 Mwt or 14% of full reactor power. B&W calculates that boiler-condenser natural circulation would not be established until at least 1500 sec. following a small break LOCA at which time the core decay heat power level would be only 2.5% of full power. Only 18% of the steam generator tube surface area would be required to remove 2.5% of full power. In the event of a small break LOCA, the operators at TMI-1 are instructed to raise the steam generator water level to 95% on the operating scale. This action would create a potentially large condensing surface above the top of the core equal to about 27% of the total tube surface area. Auxiliary feedwater enters the steam generators near the top of the tube bundle and in running down the steam generator tubes would create a still larger area for steam condensation above the core of about 62% of the total tube surface area. The staff concluded that an adequate fraction of the total steam generator surface area would be available to remove decay heat in the boiler condenser mode. The staff utilized the Nusselt equation for condensing heat transfer and the Jens and Lottes equation for boiling heat transfer. Both these equations have been determined to under predict experimentally measured heat transfer rates.

Question 9:

Whether and under what circumstances reliance on feed and bleed is necessary at TMI-1?

Answer: It is the staff's position that feed and bleed operation is neither relied upon nor necessary to remove decay heat for events within the design basis for TMI-1. The ability to remove decay heat by feed and bleed cooling is considered as a backup capability in the event of loss of secondary system heat removal capability (e.g., all feedwater to the steam generators is lost). Feed and bleed cooling involves using systems under conditions for which they were not specifically designed. Based on analyses performed by B&W and by the staff, we believe that there is a high probability that these systems will perform successfully, and we encourage operators to use all means available to maintain cooling of the core (including non-safety grade equipment) under emergency conditions. However, in our licensing review, we do not rely on these systems to perform a feed and bleed function in the context of a design basis for the plant. Instead, our reliance is on the auxiliary feedwater system for decay heat removal.

Question 10:

Results of the effort by EG&G to demonstrate the ability of the RELAP5 computer code to predict the results of Semiscale test S-SR-2.

Answer: The results of the EG&G effort to demonstrate the ability of the RELAP5 computer code to predict the results of Semiscale test S-SR-2 are documented in a report attached to a letter from P. North (EG&G) to

Mr. J. E. Solecki (DOE) entitled "Extension of Analysis of Primary Feed and Bleed Cooling in PWR Systems - PN-08-83," dated January 14, 1983.

In this report, EG&G compared the results of the RELAP5 post-test analysis of Semiscale test S-SR-2 to the experimental data, accounting for actual HPI flow characteristics and steam generator secondary side heat losses that occurred during the test.

These comparisons showed that RELAP5 was capable of predicting the data to within the accuracy of the experimental uncertainties. The calculations also showed that when the minimum system inventory results in a vessel liquid level in the vicinity of the top of the core, or below the top of the core, small uncertainties in the inventory calculation could produce significant uncertainties in the level calculation and, consequently, the degree to which core uncovery would be expected.

In Figures 10-1 through 10-5, comparisons of calculated to measured system pressure, PORV flow, HPI flow, and pressurizer and vessel collapsed liquid level are shown, respectively.

As can be seen in Figure 10-5, the vessel collapsed liquid level appears to be predicted quite well by RELAP5. However, in Figure 10-1, it can be seen that the code slightly overpredicted the system temporal pressure response. Because of this, the PORV flow (Figure 10-2) was slightly overpredicted and the HPI flow (Figure 10-3) underpredicted.

Since the effect of these differences between the calculated and measured mass inputs and outflows on the overall ability to predict temporal system inventory was not clear to the staff, EG&G, Idaho was requested to provide additional figures. In Figures 10-6 and 10-7, comparisons of the integrated HPI and PORV flows, respectively, are shown. As can be seen in Figure 10-6, the uncertainty in the integrated HPI flow is estimated at 2400 seconds to be

$$\frac{7\ \text{kg}}{37\ \text{kg}} \approx 19\%$$

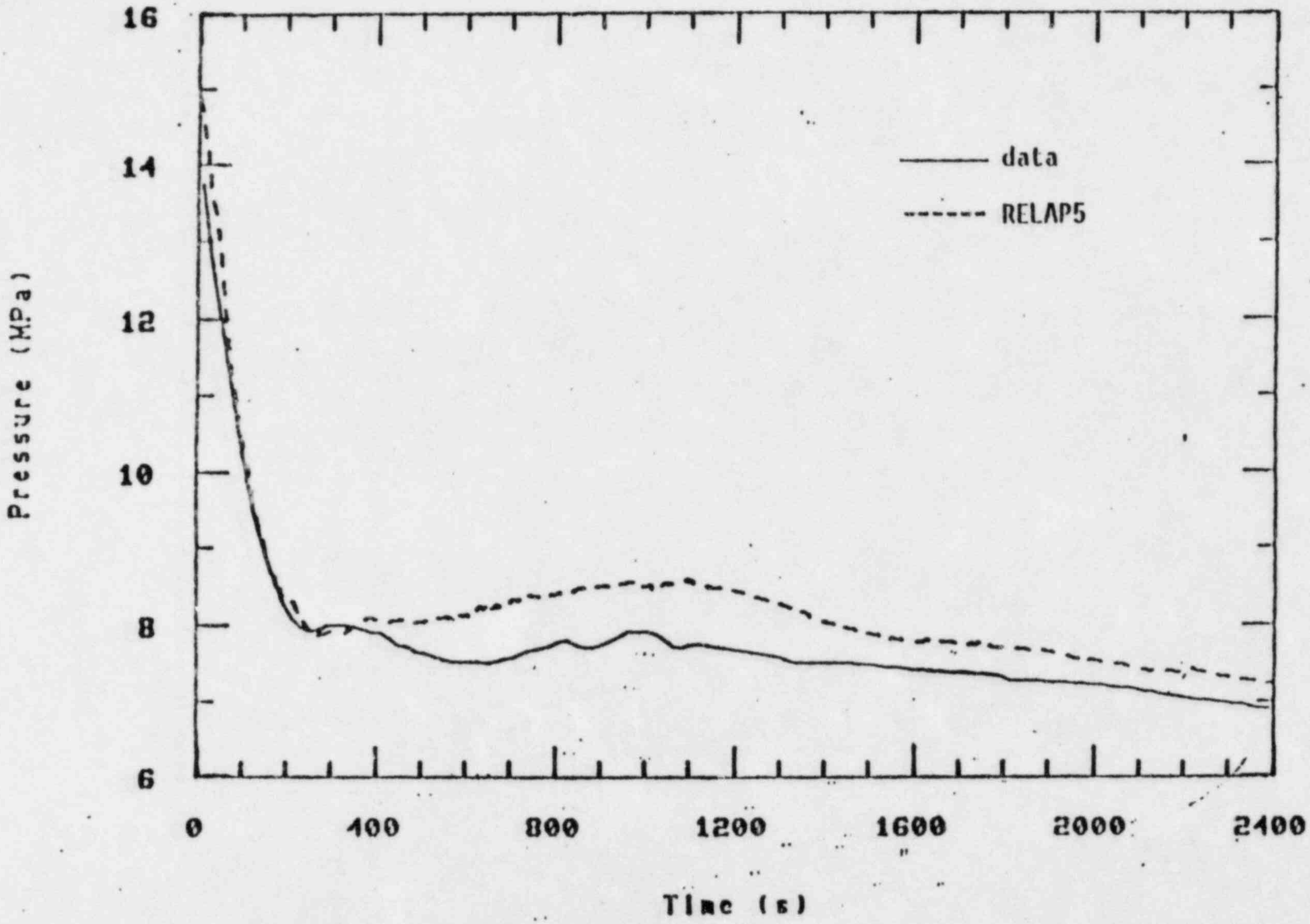
Similarly, from Figure 10-7, the uncertainty in the integrated PORV flow is estimated at 2400 seconds to be

$$\frac{12\ \text{kg}}{115\ \text{kg}} \approx 10\%$$

In Figure 10-8, measured to predicted net system mass is compared. As can be seen, at 2400 seconds, the uncertainty is estimated to be

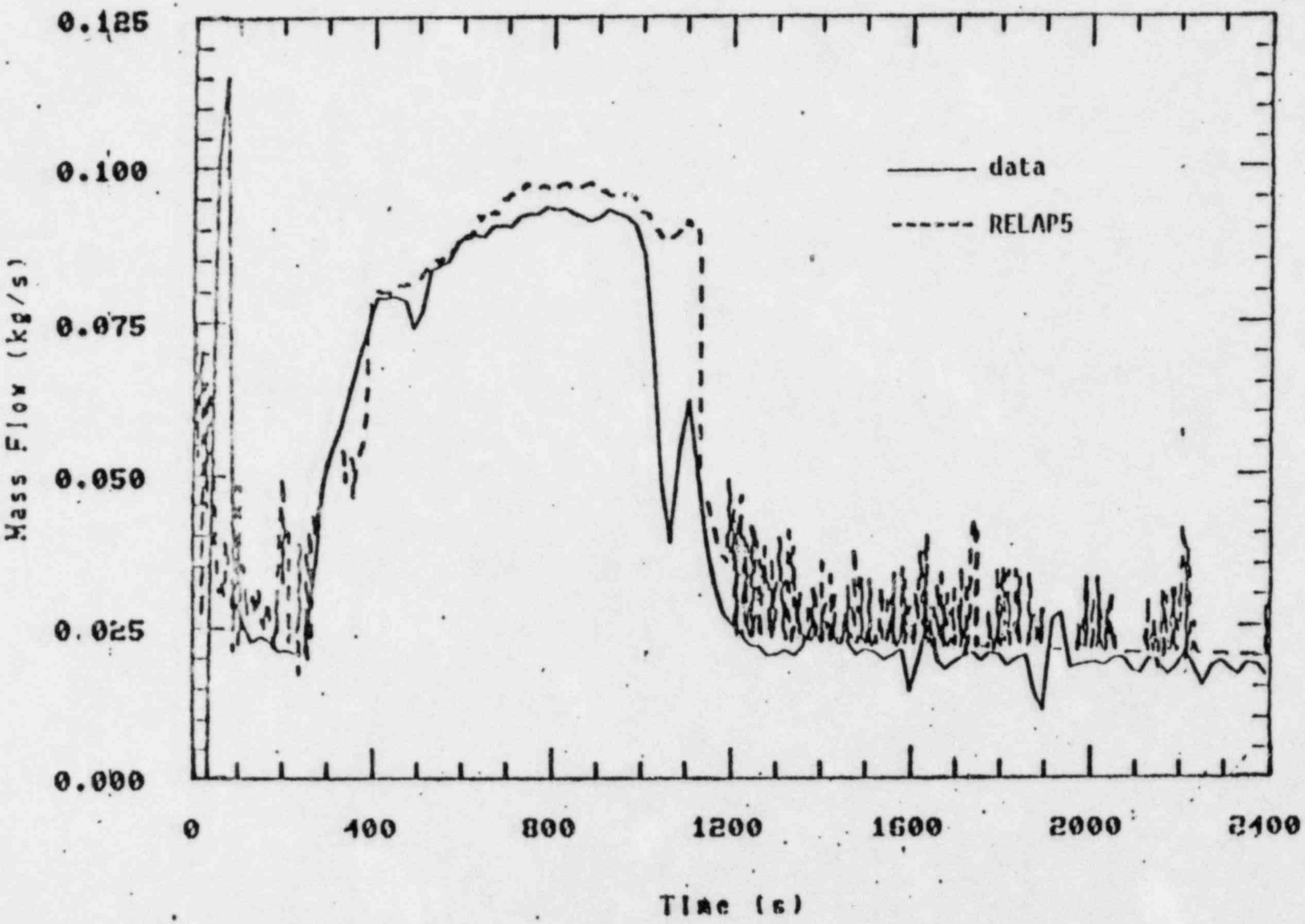
$$\frac{23\text{kg}}{113\ \text{kg}} \approx 20\%$$

Based on the above, we have concluded that RELAP5 has demonstrated its capability to correctly calculate Semiscale test S-SR-2, and that the discrepancies between the prediction and test previously noted have been satisfactorily accounted for by use of actual HPI flow characteristics and better steam generator secondary heat loss estimates. However, the uncertainty in the inventory calculation is such that it must be accounted for when reaching conclusions on the efficacy of feed and bleed cooling.



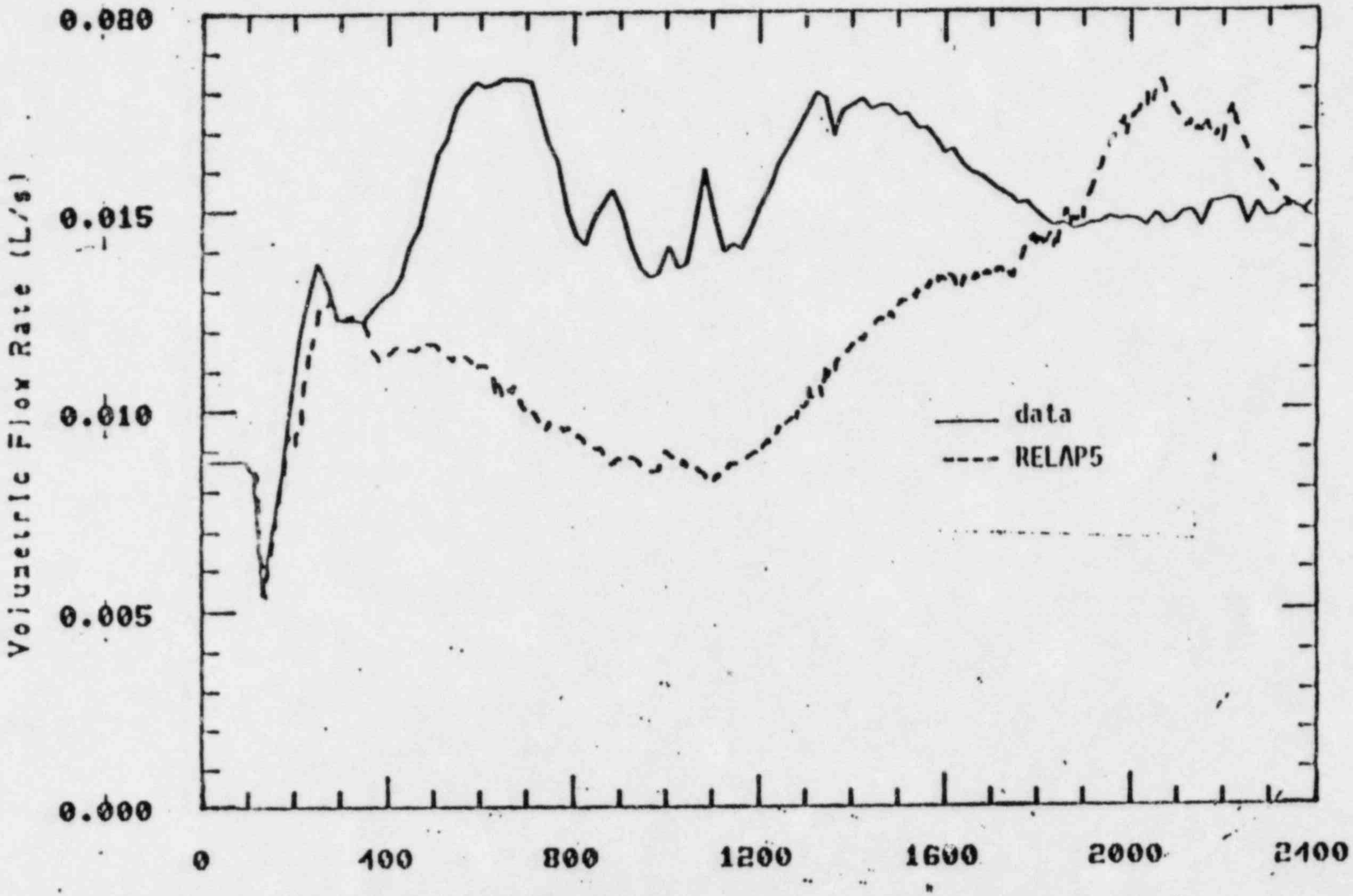
Comparison of measured and RELAP5 predicted upper plenum pressures. Test S-SR-2, point 3.

Figure 10-1



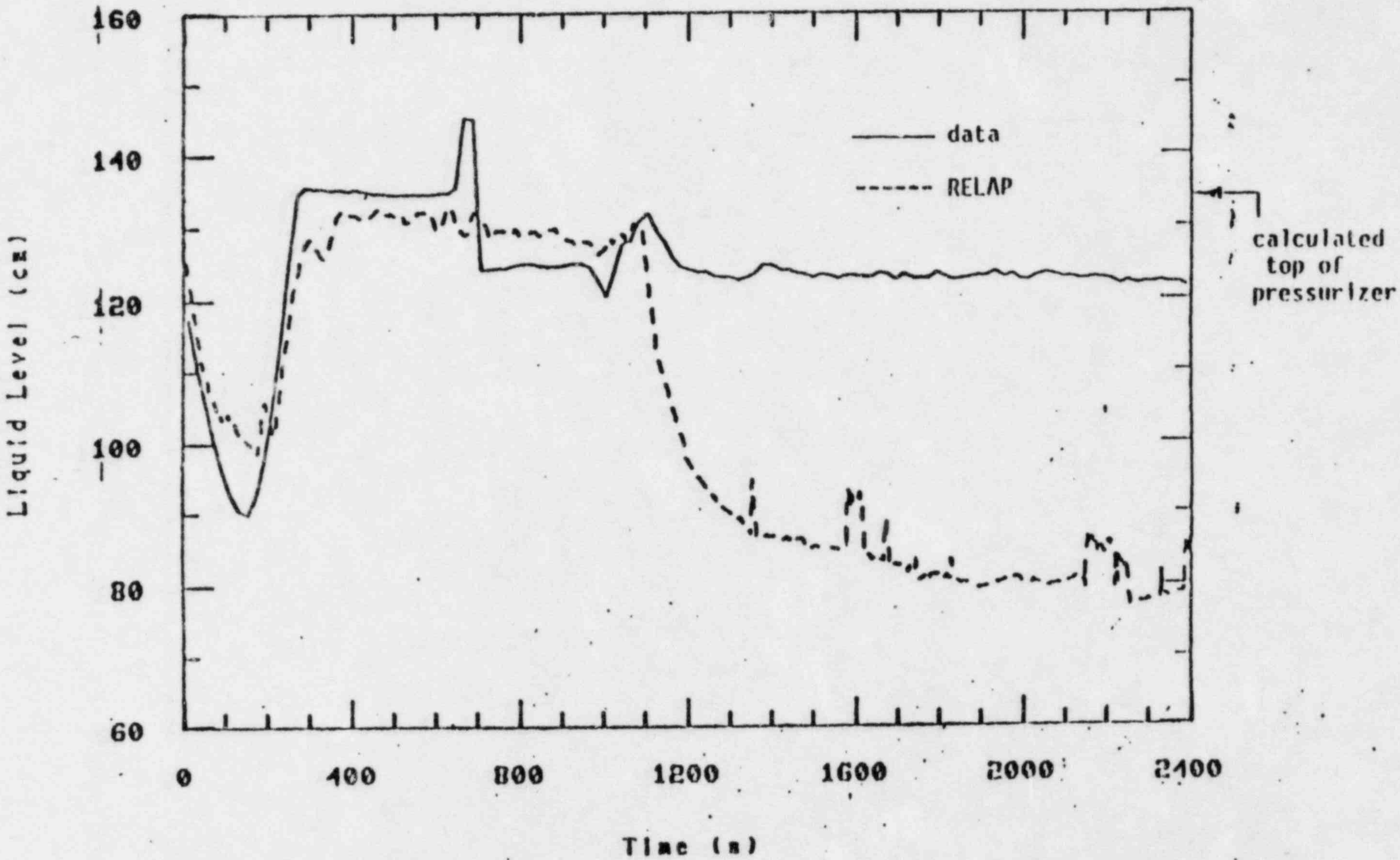
Comparison of measured and RELAP5 predicted PORV flow. Test S-SR-2, point 3.

Figure 10-2



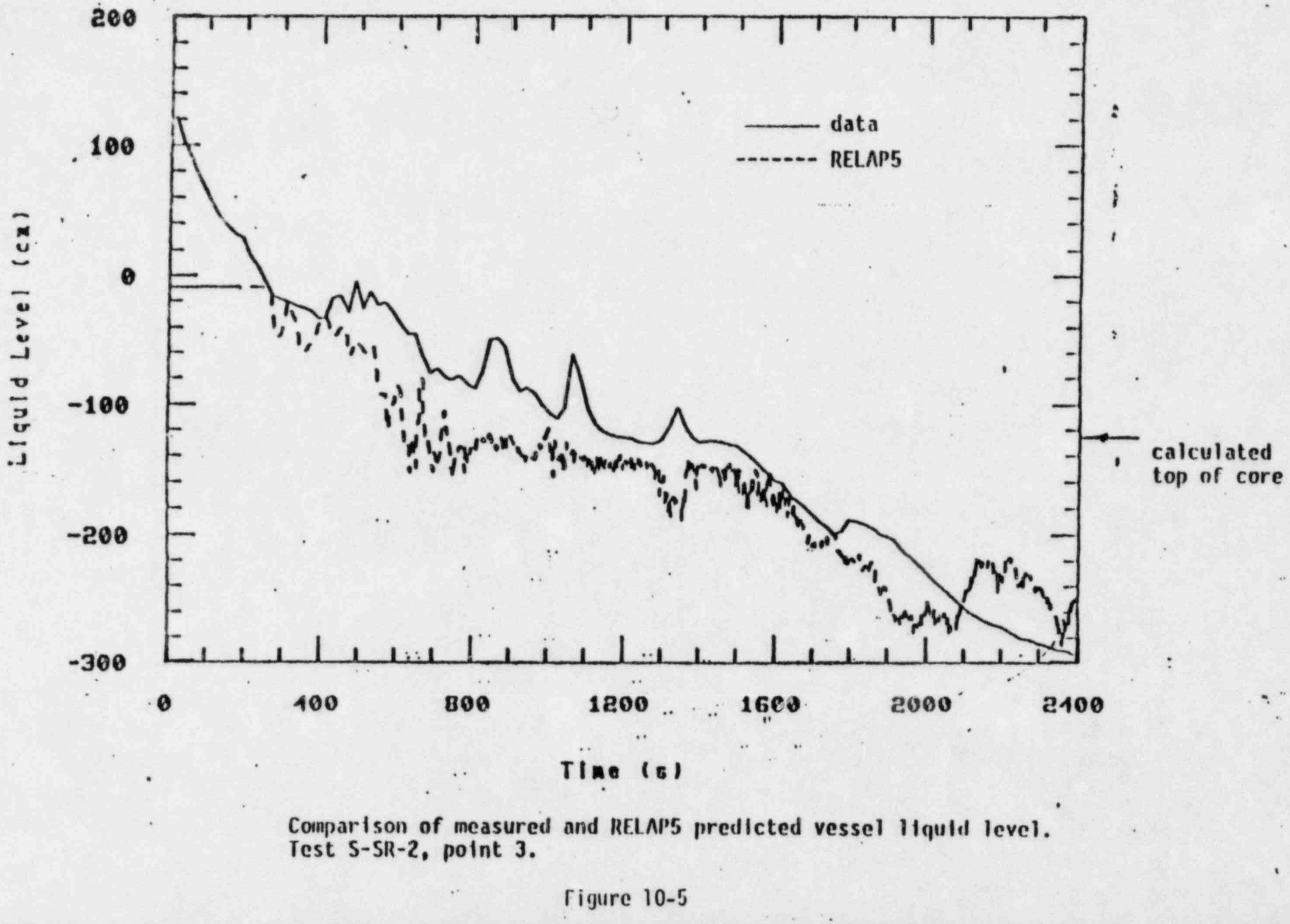
Comparison of measured and RELAP5 predicted HPIS flow.
Test S-SR-2, point 3.

Figure 10-3



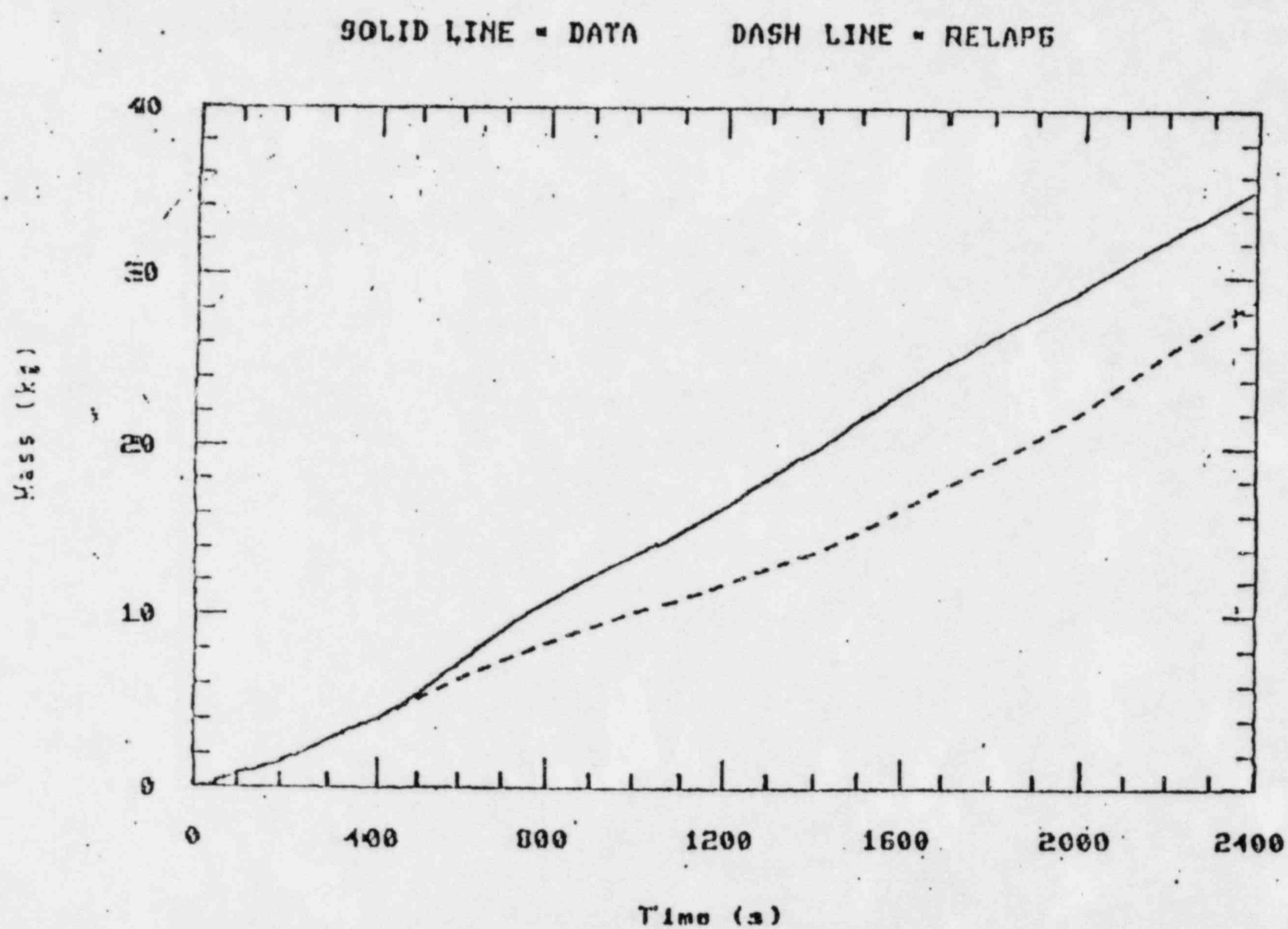
Comparison of measured and RELAP5 predicted pressurizer liquid level. Test S-SR-2, point 3.

Figure 10-4



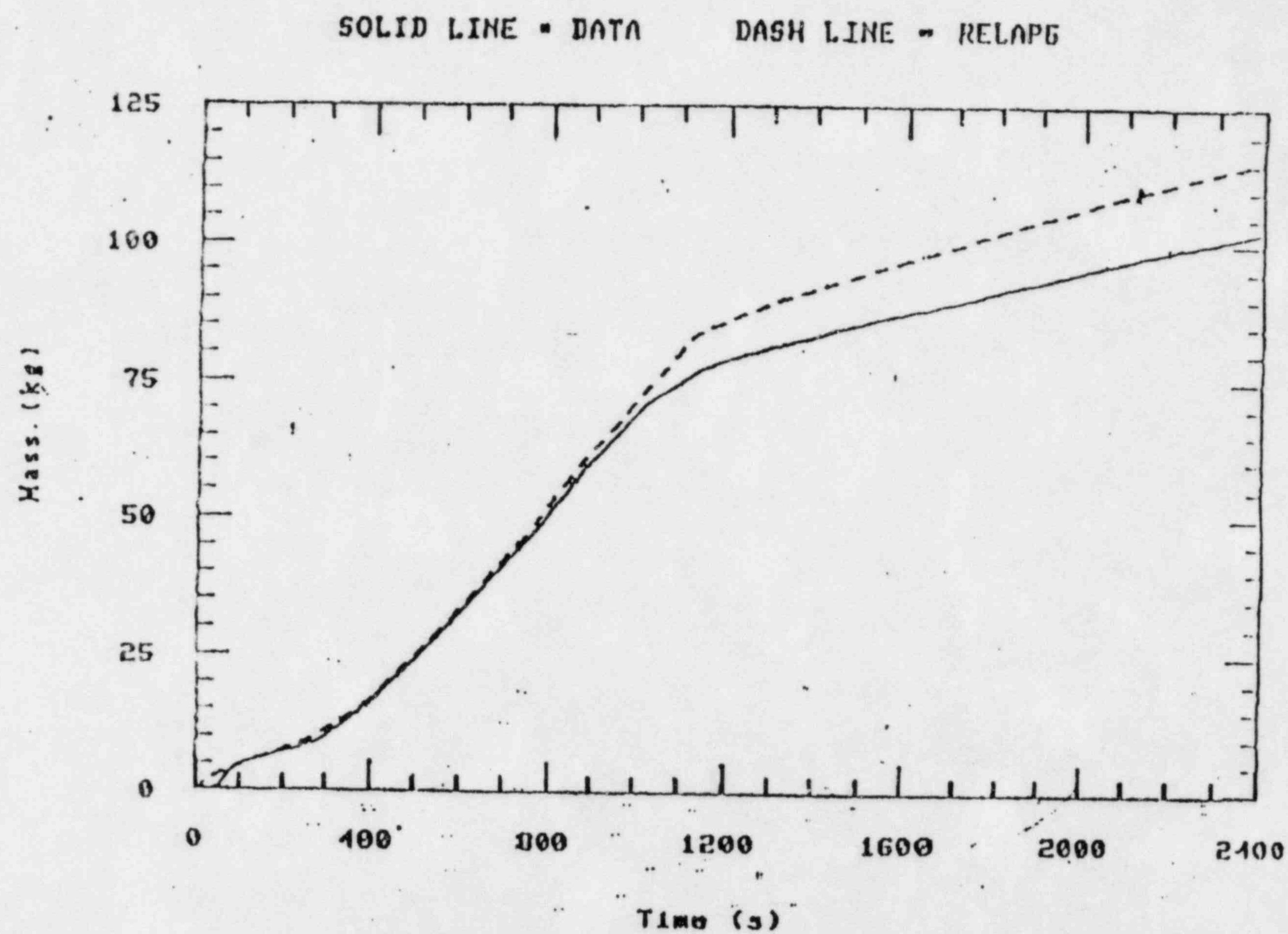
Comparison of measured and RELAP5 predicted vessel liquid level.
Test S-SR-2, point 3.

Figure 10-5

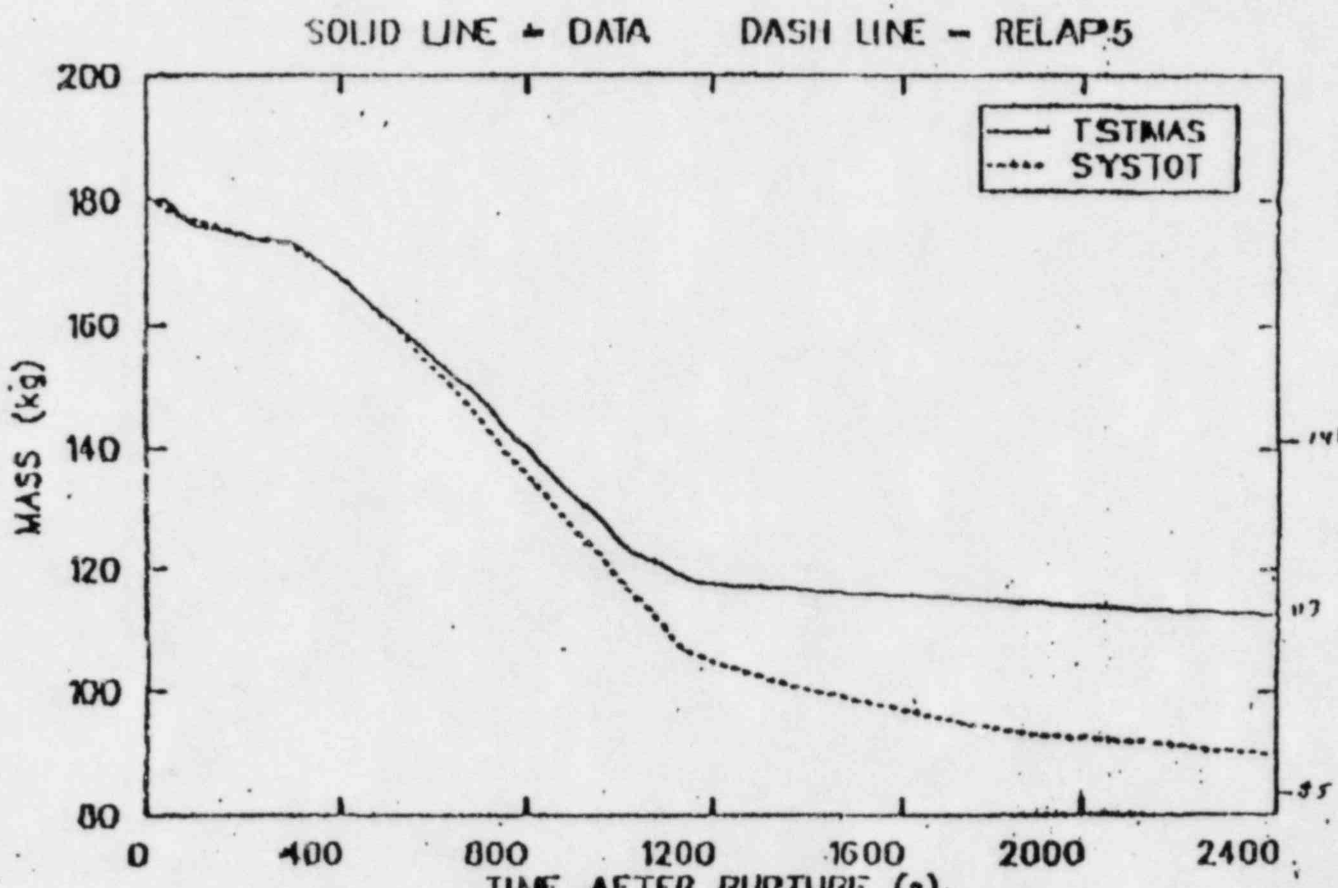


INTEGRATED HPIS FLOW RATE
TEST S-SR-2, POINT 3

Figure 10-6



INTEGRATED PORV FLOW
TEST S-SR-2, POINT 3
Figure 10-7



PRIMARY SYSTEM MASS
TEST S-SR-2, POINT 3

Figure 10-8

Question 11:

Results of a RELAP5-type analysis to determine whether feed and bleed will successfully provide core cooling at TMI-1.

Answer: In response to the Appeal Board request, the NRC staff asked EG&G, Idaho to perform a feed and bleed analysis for the TMI-1 reactor using the RELAP5 computer code. The analysis assumed a loss of all feedwater (both main and auxiliary) to the steam generators at 300 seconds. The reactor coolant pumps were also tripped at 300 seconds. The decay heat used was the draft 1973 ANS Standard. The HPI injection was delayed until 20 minutes after the event initiated and no credit was taken for the PORV (i.e., bleed only accomplished with safety valve). The results of the analysis are shown in Figures 11-1 through 11-5.

Figure 11-1 shows the hot leg pressure versus time. As can be seen, once the steam generators dry out, the system pressurizes to the safety valve setpoint and the safety valve then controls to cycle open and closed relieving sufficient fluid to remove the decay heat. Figure 11-2 shows the collapsed liquid level in the reactor vessel. In this calculation, the top of the active core is at a level of 238 inches. Because the collapsed level shown in Figure 11-2 does not drop below the top of the core, the core remains covered and well cooled. It should be noted that this is the collapsed liquid level. In reality, the water in the core will be boiling, and the voids distributed in the core will raise the effective level.

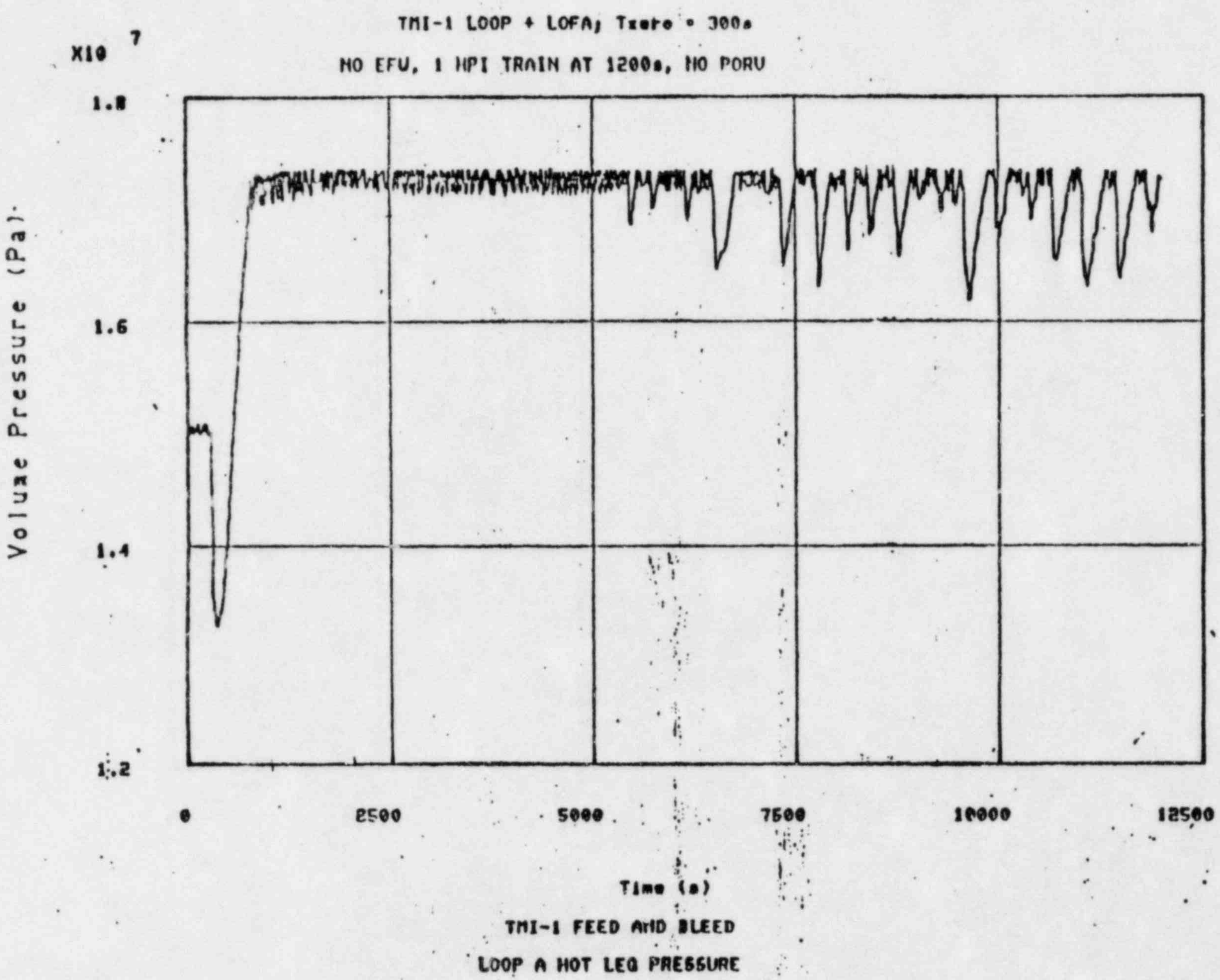

TMI-1 LOOP + LOFA; Tzero = 300s
NO EFW, 1 HPI TRAIN AT 1200s, NO PORV
X10 7
1.8
1.6
1.4
1.2
Volume Pressure (Pa)
0
2500
5000
7500
10000
12500
Time (s)
TMI-1 FEED AND BLEED
LOOP A HOT LEG PRESSURE

Figure 11-1

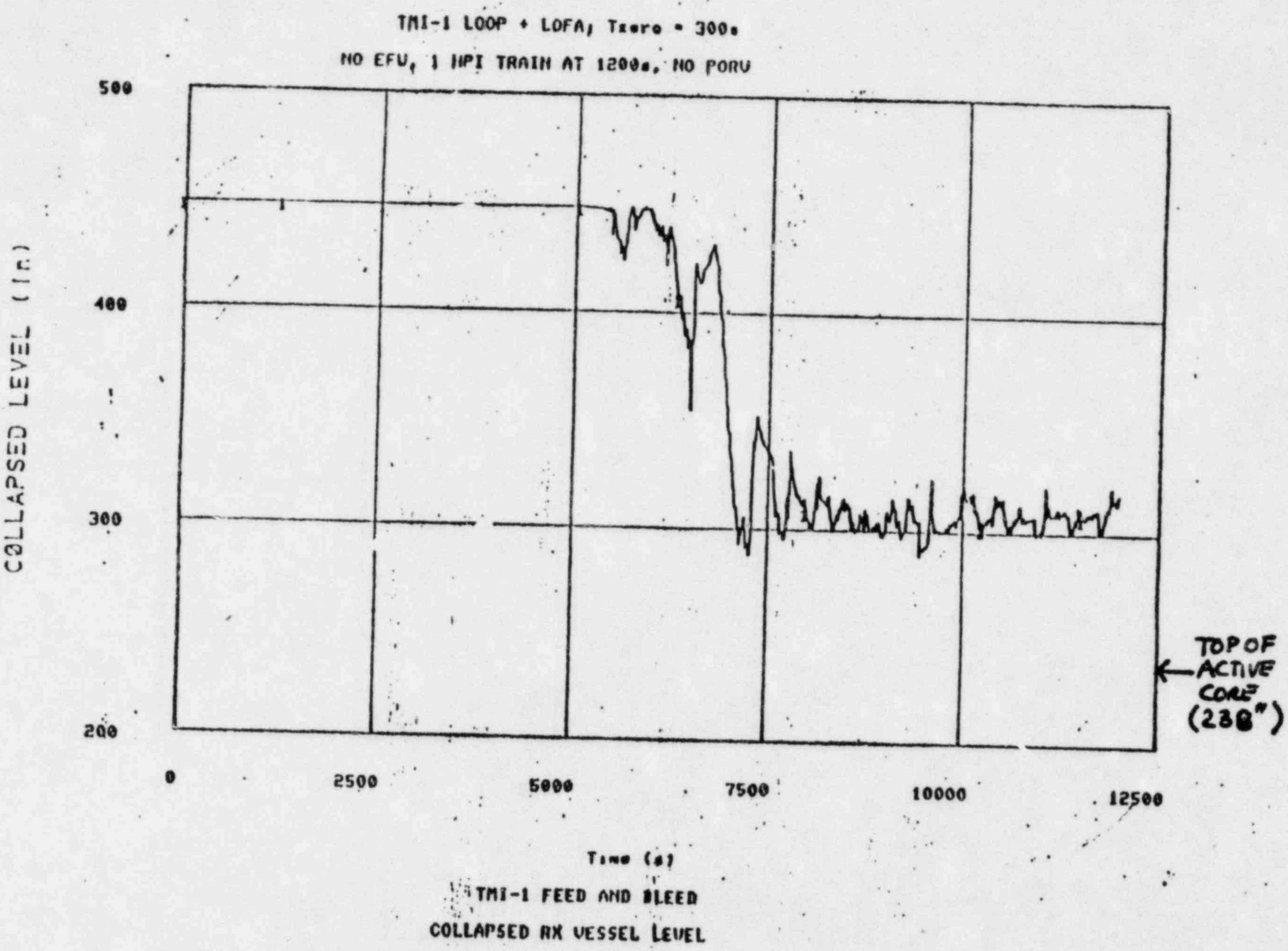

TMI-1 LOOP + LOFA; Tzero = 300s
NO EFW, 1 HPI TRAIN AT 1200s, NO PORV
COLLAPSED LEVEL (in)
500
400
300
200
0
2500
5000
7500
10000
12500
Time (s)
TOP OF ACTIVE CORE (238")
TMI-1 FEED AND BLEED
COLLAPSED RX VESSEL LEVEL

Figure 11-2

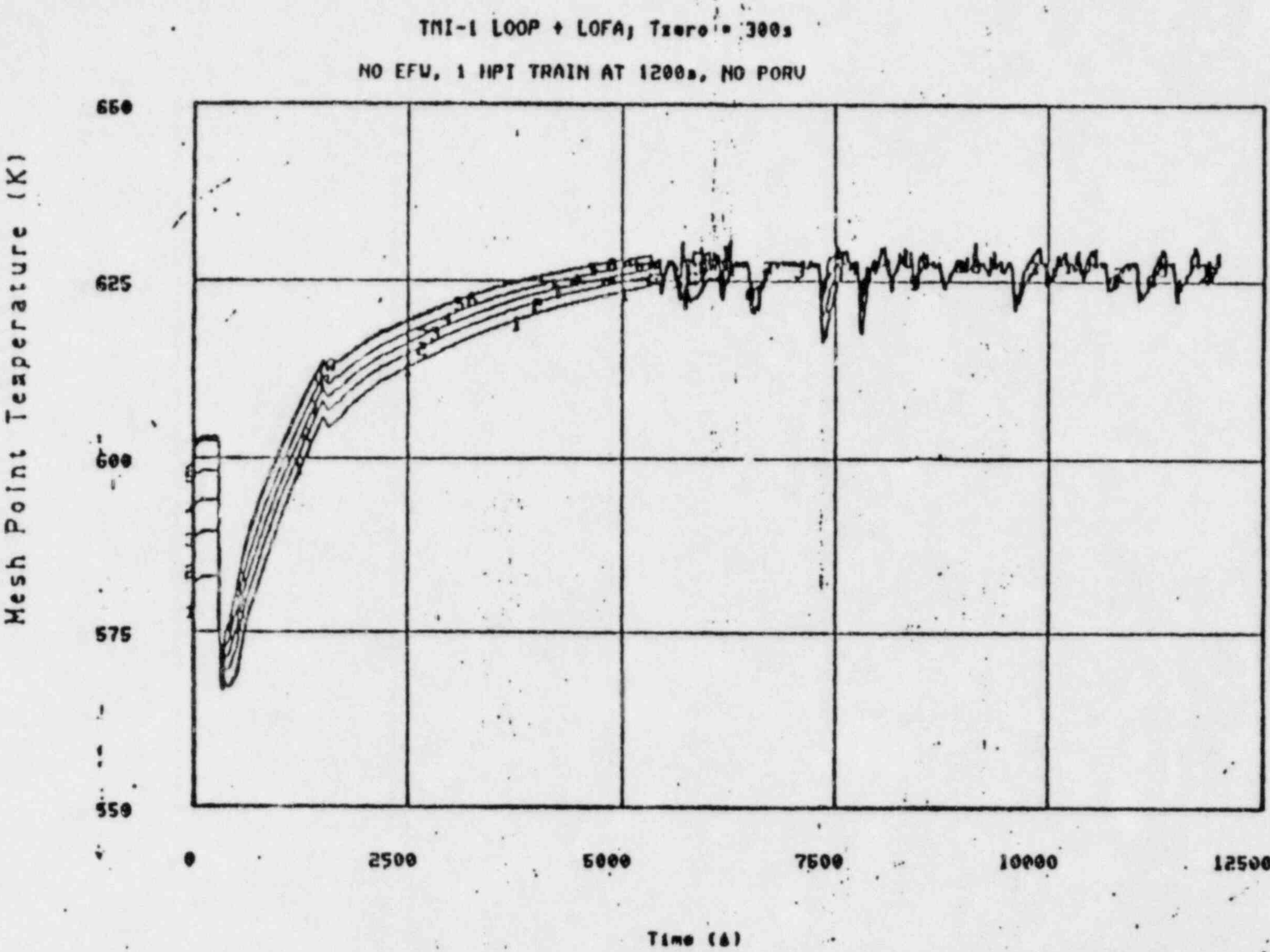
Figure 11-3 shows the calculated fuel cladding temperatures. As is seen, they remain slightly above the saturation temperature during the entire course of the event.

In Figure 11-4, the integrated mass flows both out of the system (safety valve) and into the system (HPI) are shown. In Figure 11-5, the net system mass flow out of the system is shown. This curve is obtained by subtracting the integrated HPI flow from the integrated safety valve flow. At approximately 9,000 seconds the net mass loss from the primary system ceases to increase and begins to decrease, indicating that the system is beginning to refill and recover.

As stated in response to Appeal Board Question 10, the results of Semiscale test S-SR-2 have shown that mass inventory uncertainty is important if the minimum system inventory results in liquid levels near the top of the core. To account for this uncertainty, we examined uncertainties in the HPI and safety valve flows, as well as uncertainties in the code calculation itself.

Safety Valve Flow Uncertainties

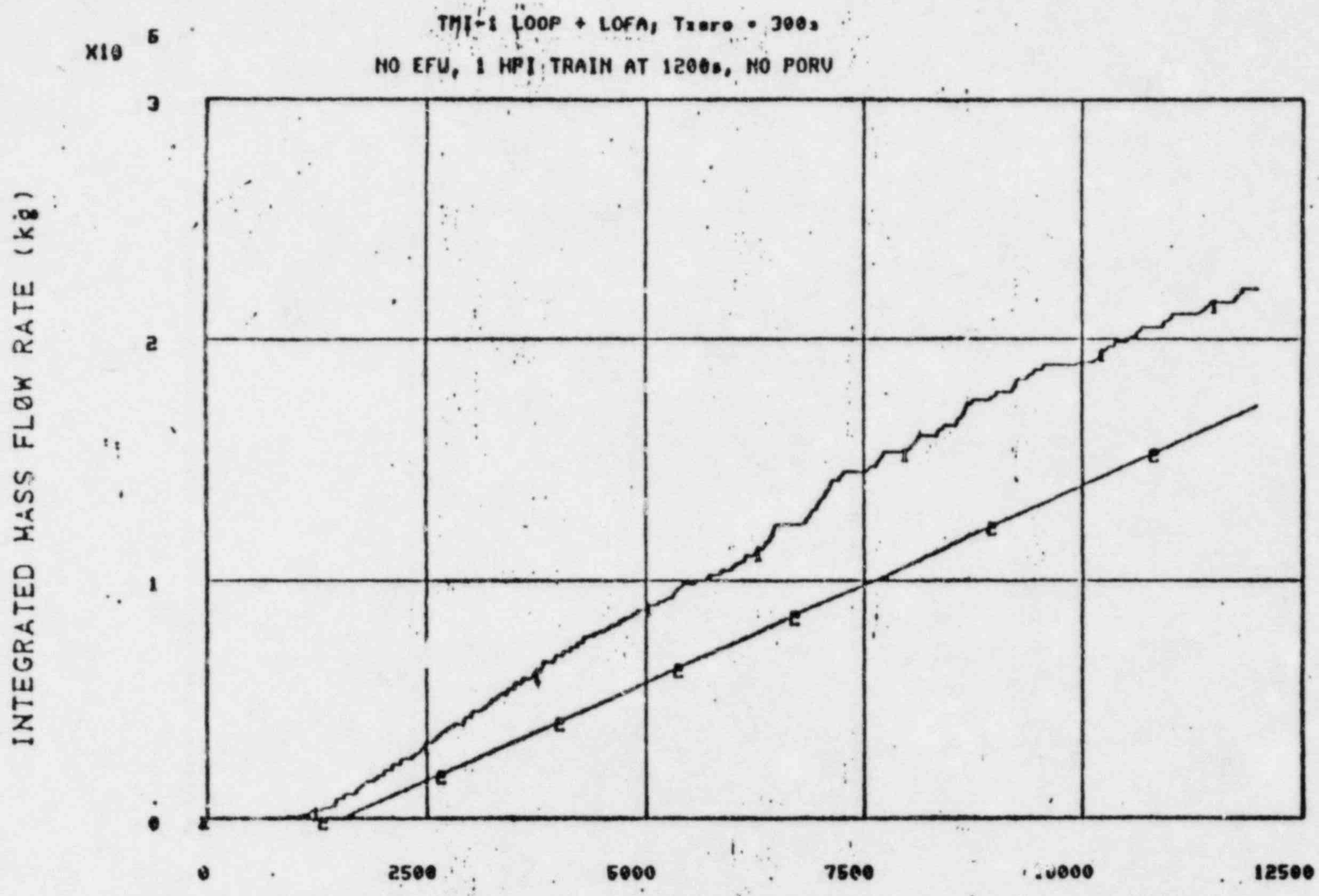
EG&G, Idaho reported that rated safety valve relief capacity, which is the relief capacity used in these analyses, is about 15 percent below the tested relief capacity for the Dresser- type safety valves used at TMI-1 for steam flow. The uncertainty in relief capacity is estimated at ±15 percent of the rated capacity for steam flow. Therefore, we conclude that the steam relieving capacity assumed in the RELAP5 analysis



TMI-1 FEED AND BLEED

CORE CLADDING TEMPERATURES

Figure 11-3

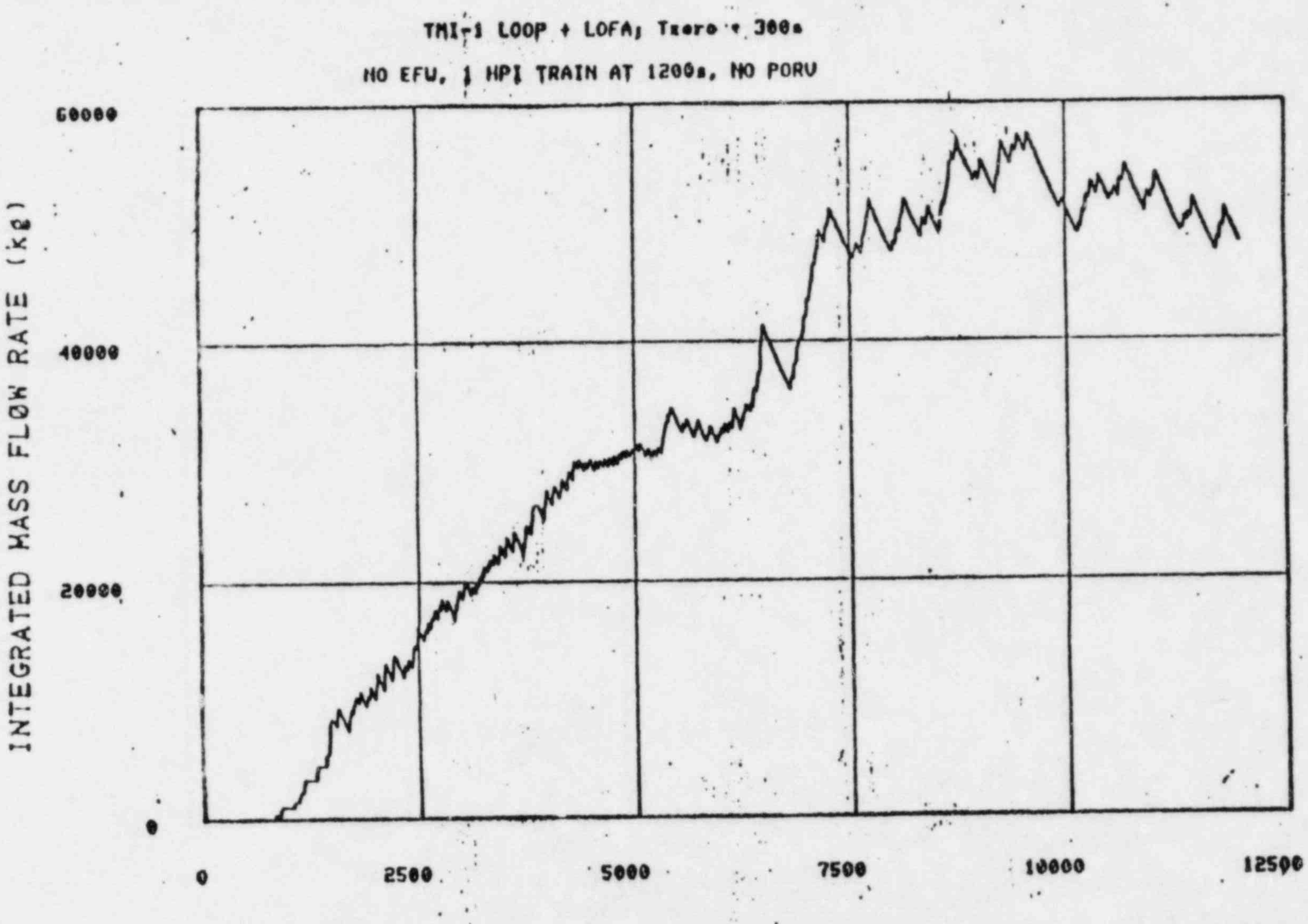


TMI-1 FEED AND BLEED

1 - INTEGRATED CODE SAFETY DISCHARGE

2 - INTEGRATED ECC/HPI INJECTION

Figure 11-4



TMI-1 FEED AND BLEED

MASS BALANCE

NET DISCHARGE MINUS NET INJECTION

Figure 11-5

conservatively accounts for uncertainties with respect to expected relieving capacity. We also point out that the safety valve was calculated to cycle open and closed. Hence it was not relieving at its rated capacity and any uncertainties in relieving capacity would be accommodated by changes in the cycling time.

EG&G also examined the uncertainty in the liquid relieving capacity of the safety valve. They reported that the safety valve liquid flow calculated by RELAP5 is an average of 9 percent above the measured flow, and the uncertainty on this value is ±15 percent. However, the flow discharge area was sized to 15 percent smaller in the analyses. This leads to a estimated uncertainty range in liquid relief valve flow of

$$-15 + 15 + 9 = +9\%$$

$$-15 - 15 + 9 = -21\%$$

Thus, the liquid flow uncertainty is biased in a conservative direction.

We have not examined in detail the uncertainty in two-phase flow through safety or relief valves. However, as the following discussion illustrates, we believe that the uncertainty in either liquid or two-phase safety valve flow does not play an important role in whether or not TMI-1 can feed and bleed.

Following a loss of all feedwater and subsequent dryout of the steam generators, the primary system coolant begins to heat up from the core decay heat and expands, expelling steam from the pressurizer steam space. Once the steam is expelled, the safety valve flow transitions to liquid discharge. The primary system coolant continues to heat up until the saturation temperature corresponding to the safety valve setpoint pressure is reached. At this time, boiling in the core begins, and the steam generated rises and accumulates primarily at in the top of the vessel, displacing liquid which is forced out of the vessel, into the hot legs, then out through the pressurizer safety valve. This process continues until the mass loss through the safety valves causes the primary system to void extensively in the upper regions so the hot leg void is large and that mostly steam can enter the pressurizer surge line and exit the safety valve.

During the period in which the safety valve flow is liquid, the flow exceeds the HPI makeup capacity and a net system inventory loss is occurring. Only when the hot leg void fraction becomes significantly large does the safety valve flow transition to steam and only at that time does the HPI flow begin to exceed the safety valve flow. In other words, for feed and bleed to be a viable means of decay heat removal, the safety valve flow must be almost essentially steam flow, not liquid or two-phase. This means that the primary system void fraction must be high in the upper portion of the vessel and the hot leg pipes so that steam can enter the hot legs and the surge line. One question that does arise is that although steam from the core now has a direct path to the surge line, the pressurizer can have a significant quantity of liquid remaining in it, unable to drain due to counter-current flow limits. It is conceivable that the steam entering the surge line could entrain this residual liquid in the pressurizer as it rises to the safety valve entrance and still result in a two-phase discharge for a limited period of time until after hot leg uncovery all of the liquid in the pressurizer was finally entrained and discharged. We have examined the Semiscale test S-SR-2 data and conclude that this is not the case. The Semiscale data shows that once steam was able to enter the pressurizer surge line, the relief valve discharge quickly transitioned to steam flow, with very little entrainment of the residual liquid.

The conclusion reached is that in order for feed and bleed to be effective, the safety valve discharge must be steam. Therefore, only the steam flow uncertainty would affect the ability to feed and bleed, and the liquid or two-phase safety valve flow uncertainties have only a minor influence on the efficacy of feed and bleed at TMI-1.

HPI Flow Uncertainty

The HPI pump flow uncertainty was stated to EG&G, Idaho by GPU to be known to within ±5 gallons per minutes. Compared to the total HPI flow from one pump at the safety valve relief pressure (254 gpm), this represents a small uncertainty (less than 5 percent) and is acceptable.

Code Uncertainty

To account for code uncertainty, we examined the effect of assuming a 25 percent uncertainty in the calculated vessel inventory at the time of minimum inventory. If the remaining vessel inventory at the time of minimum system inventory is reduced by 25%, we estimate that although the collapsed liquid level would drop approximately two feet below the top of the core, the mixture level, assuming a similar axial void distribution in the core, would remain well above the top of the core.

Our conclusion is that our calculation of feed and bleed for TMI-1 demonstrates that the core will remain covered and adequately cooled. Our confidence in this conclusion is supported by the uncertainty evaluation discussed above, and the supplemental calculation we had performed by Los Alamos National Laboratory, using the advanced TRAC computer code and which is described below.

At the request of the NRR staff, the Los Alamos National Laboratory (LANL), under contract to and direction of the NRC's Office of Nuclear Regulatory Research, performed a feed and bleed analysis on an Oconee reactor. This reactor is essentially the same as the TMI-1 reactor and the power level used was 2568 MWth, which is 33 MWth higher than the TMI-1 power level. In Figure 11-6, the net mass loss versus time is shown. As can be seen, at about 8000 seconds the net mass loss stops. In Figure 11-7, the fuel rod temperature is shown. The temperature is calculated not to go significantly above the saturation temperature.

In summary, the LANL analysis shows that feed and bleed will provide adequate core cooling.

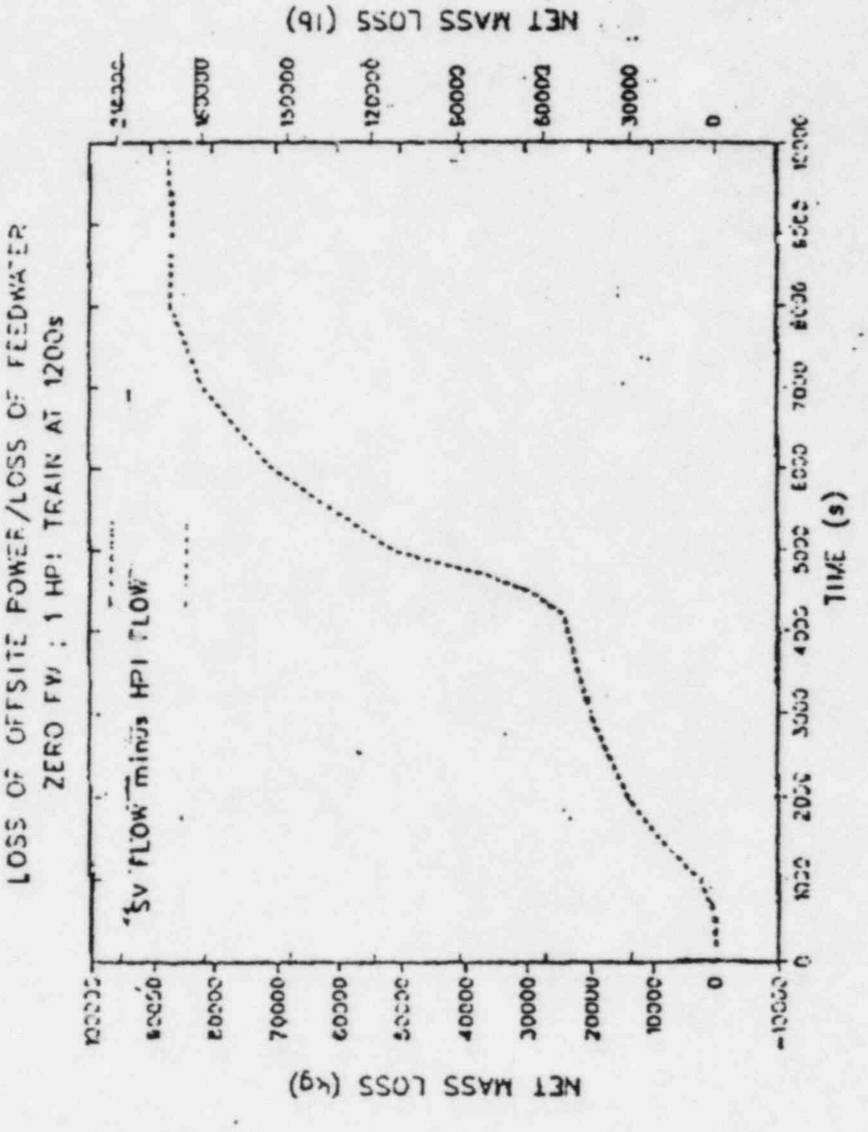

LOSS OF OFFSITE POWER/LOSS OF FEEDWATER
ZERO FW ; 1 HPI TRAIN AT 1200s
SV FLOW minus HPI FLOW
NET MASS LOSS (kg)
NET MASS LOSS (lb)
TIME (s)
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
10000
-10000
10000
20000
30000
40000
50000
60000
70000
80000
90000
100000
30000
60000
90000
120000
150000
180000
210000

Figure II-6

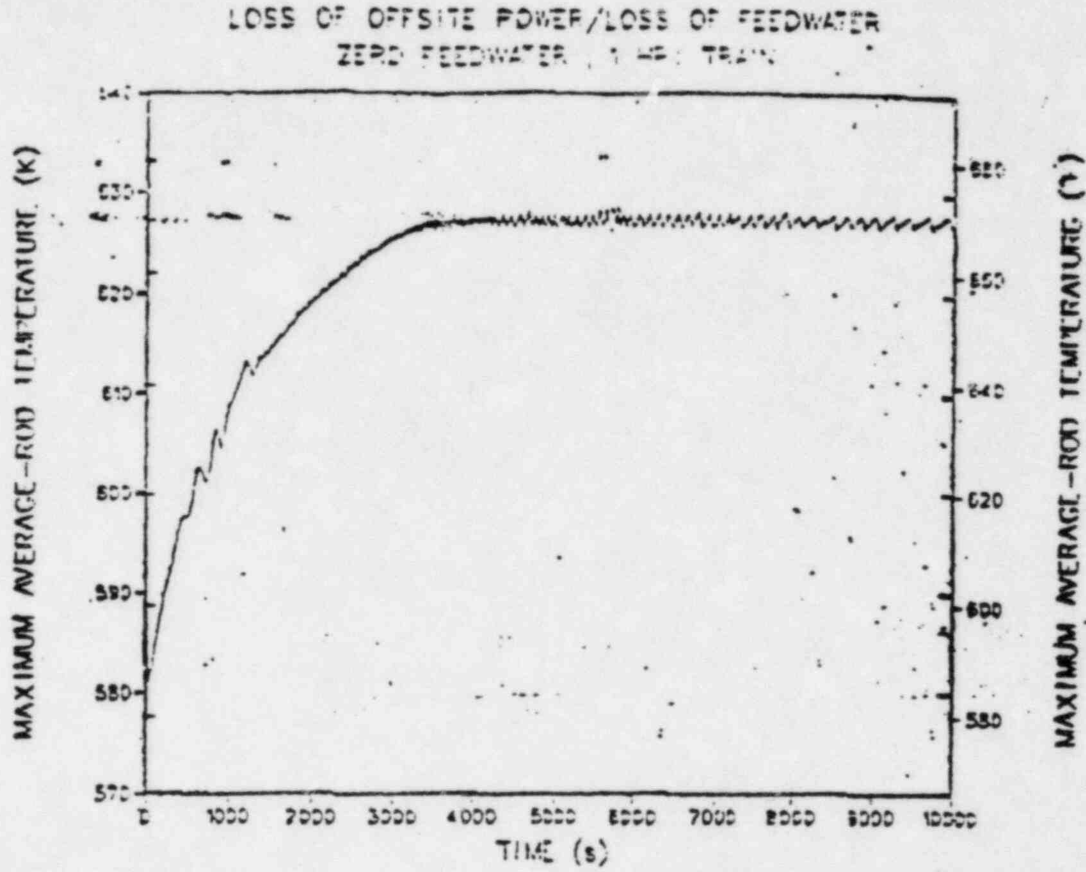

LOSS OF OFFSITE POWER/LOSS OF FEEDWATER
ZERO FEEDWATER, 1 HPI TRAIN
MAXIMUM AVERAGE-ROD TEMPERATURE (K)
MAXIMUM AVERAGE-ROD TEMPERATURE (°F)
TIME (s)
570
580
590
600
610
620
630
640
580
600
620
640
660
680
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
10000

Figure 11-7

This testimony of Jared S. Wermiel presents the NRC Staff's response to the Appeal Board's Question 8. (ALAB-708 at 43).

The purpose of this testimony is to clarify the safety-grade status of the EFW system functions and components that are necessary to cope with design basis events at TMI-1.

Summary

The flow control valve function and the condensate storage tank level indication function are not presently safety-grade for all design basis events. In addition, portions of the EFW system piping and controls have not been shown to be capable of withstanding a safe shutdown earthquake. Actions necessary to upgrade all EFW system functions and components to safety-grade status are expected to be completed by startup following the first refueling after restart.

Of the EFW system functions that are not safety grade for all design basis events the flow control valve function is the only one necessary to cope with a loss of main feedwater and a small break LOCA. Manual action can be taken to restore EFW flow in the event of a failure of the ICS that leaves both flow control valves closed.