Florida Power
CORPORATION

October 10, 1979

File: 3-0-3-a-3

Mr. D. F. Ross, Jr.
Director
Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Identification and Resolution of Long-Term Generic Issues
Related to the Commission of May 1979

Dear Mr. Ross:

With reference to ATTACHMENT A TO ENCLOSURE 1, LISTING OF OUTSTANDING ITEMS RELATED TO B&W SMALL BREAK ANALYSIS, of your subject letter dated August 21, 1979, attached are:

1. Our response to items 1B, 2A, 4; and
2. Our interim responses to items 3C and 3D.

We will forward to you our responses to items 1A, 2B, 3A, and 3B as soon as the information is received from B&W.

B&W will forward, directly to you, responses to items 5 and 6.

Should you have any questions concerning this information, please contact this office.

Very truly yours,

FLORIDA POWER CORPORATION

G.C. Moore

G. C. Moore
Assistant Vice President
Power Production

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GCMekcT05(NRC8/21)DN87

STATE OF FLORIDA

COUNTY OF PINELLAS

G. C. Moore states that he is the Assistant Vice President, Power Production, of Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

G C Moore

G. C. Moore

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 10th day of October, 1979.

Catherine J. Cameron

Notary Public

Notary Public, State of Flor[illegible] at Large,
My Commission Expires: [illegible] 1983

CameronNotary 3(D12)

RESPONSE TO QUESTION 1B OF ATTACHMENT A TO D. F. ROSS
(NRC) LETTER DATED 8/21/79

Question

1B. Provide justification of relief and safety valve flow models used in the CRAFT2 code.

RESPONSE

The CRAFT2 code, which is documented in topical report BAW-10092, Rev. 2[1], does not have any special models for prediction of the fluid discharge through the relief and safety valves. Rather, they are modeled as leak paths from the pressurizer control volume to the containment. Thus, the Bernoulli (orifice) equation is used for subcooled discharge, while the Moody correlation is used for saturated steam or two-phase discharge. These models are the same as those used in B&W's ECCS Evaluation Model.[2] Since little information exists on the flowrate through pressurizer valves for subcooled or two-phase fluid conditions, it is impossible to ascertain the accuracy of this modeling technique. Since pressurizer leaks are inherently less severe than the breaks in the cold leg pump discharge piping analyzed to demonstrate compliance to 10 CFR 50.46, a truly realistic model for the discharge rates is not necessary. However, the modeling technique utilized is expected to reasonably approximate the discharge rates and their subsequent effect on the RCS.

System response to relief valve actuation has been analyzed and submitted to the Staff in Section 6 of the May 7, 1979[3], report. The cases specifically analyzed were:

1. A loss of main feedwater accident which results in actuation and a subsequent sticking open of the pressurizer relief valve was addressed. Offsite power was assumed to remain available and only one HPI train was used for emergency core cooling. This analysis is similar to the TMI-2 event that occurred on March 28, 1979, and demonstrated that if one HPI pump remained available, no core uncovery would have occurred.

2. A stuck open PORV assuming a loss of offsite power and only one HPI train available was analyzed. Results of this evaluation demonstrated that core uncovery would also not occur.

An additional analysis of the effect of a pressurizer break, which supplemented those presented in reference 3, was provided to the Staff in a letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), dated May 12, 1979[4]. That analysis examined the effect of the stuck open PORV case, Case 2, above, except the auxiliary feedwater system was assumed inoperable. The results of that evaluation showed that, even without auxiliary feedwater, one HPI pump can handle the accident, provided that realistic decay heat valves are utilized. In all of these evaluations, the PORV was modeled via a leak path representation in the CRAFT2 code. The orifice area of the PORV was modeled as the leak area (1.05 in.^{2}) and a discharge coefficient of 1.0 was utilized.

The method for modeling the PORV described above does result in a predicted steam flowrate, at the valve rated pressure, which is in excess of the design (rated)

RESPONSE (1B) (Continued)

flowrate. An alternative modeling approach is to use a discharge coefficient (C_D) which, at the valve rated pressure, would yield the valve rated flowrate. For the 177-FA plants, this is a C_D of approximately 0.85. For the first two cases described above, this modelling approach would result in a slower system depressurization and a slower discharge of the RCS inventory. Thus, the use of a C_D = 1.0, used in previous evaluations results, is a conservative assessment of the transient. For the third case, the use of a smaller C_D would result in a larger repressurization following the loss of the SG as a heat sink and the change in the discharge from steam to two-phase flow. However, use of a C_D of 0.85 would result in an inventory loss less than that calculated in reference 4 and no core uncovery would occur.

Besides the cases involving actuation of the pressurizer relief valves, the attached analyses were performed for a total loss of SG heat sink. In those evaluations, the pressurizer safety valves were exercised. To model these valves, the leak path representation was used with the leak path opening and closing at the opening setpoint of the valve. The valve area and C_D was chosen such that the rated flowrate for the valve would be simulated at the valve rated pressure. Because of the large relief capacity of the valve, the system pressure oscillated within a few psi of the valve setpoint and the valve was exercised intermittently. Thus, any discrepancies between the modeled and the actual relief capacity of the pressurizer safety valve is not expected to significantly alter the system response.

While there is little information available on the discharge rates through the pressurizer valves, it is also important to note the breaks in the pressurizer are bounded by breaks in the cold leg pump discharge piping. Pump discharge breaks are analyzed to show conformance of the ECCS to meet the criteria of 10 CFR 50.46. The reason that cold leg breaks bound breaks in the pressurizer was discussed in detail in reference 3. Therefore, it is not necessary to simulate the actual relief capacities of the pressurizer valves in order to demonstrate the ability of the ECCS to mitigate the consequences of a loss of RCS inventory through the valves within the criteria of 10 CFR 50.46.

REFERENCES

1 BAW-10092, Rev. 2, "CRAFT2 - FORTRAN Program for Digital Simulation of a Multinode Reactor Plant During LOCA", R. A. Hedrick, J. J. Cudlin, and R. C. Foltz, April 1975.

2 BAW-1010[illegible], Rev. 3, "B&W's ECCS Evaluation Model", B. M. Dun, et al., August 1977.

3 Letter J. H. Taylor (B&W) to R. J. Mattson, May 7, 1979, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177-FA Plant".

4 Letter J. H. Taylor (B&W) to R. J. Mattson, May 12, 1979.

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505

Telephone: (804) 384-5111

File: 595-7102-5

September 11, 1979

To: B&W 177 Owners Group
Technical Subcommittee on TMI-2 Incident Related Tasks

Subject: Complete Loss of Feedwater Transient

Gentlemen:

The attached is the results of analyses performed for a complete loss of feedwater for the 177-FA lowered-loop plants. This analysis completes Item No. 40 on the Owners Group Task List.

The analyses show that one HPI pump provides sufficient makeup to prevent core uncovery in the event of a complete loss of feedwater transient. Since no core uncovery occurs, the cladding temperatures would remain within a few degrees of the saturated fluid temperatures and, therefore, no cladding rupture or metal-water reaction occurs. Hence the criteria of 10CFR50.46 in satisfied for this transient.

Sincerely,

THE BABCOCK & WILCOX COMPANY

RB Davis

R. B. Davis
Product Manager

RBD/ryc

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SYSTEM RESPONSE TO TOTAL LOSS OF SG HEAT SINK

1. Introduction

An analysis of a complete loss of feedwater transient accident for the 177-FA lowered-loop plants has been conducted. The analysis was performed utilizing a "realistic" decay heat curve, and assumed that offsite power was lost and that the operator actuated one of the HPI systems at 1200 seconds.

2. Summary and Conclusions

An analysis of a complete loss of feedwater transient for the 177-FA lowered-loop plants has been performed utilizing a "realistic" decay heat curve. Consistent with the operating procedures, it was assumed that the operator would initiate the HPI system by 20 minutes. A single failure in the HPI system was included in the evaluation.

The analysis demonstrated that 1 HPI pump provided sufficient makeup to prevent core uncovery. The ultimate heat sink for this transient is the containment via energy release through the pressurizer safety valves. Since no core uncovery occurs, cladding temperatures would remain within a few degrees of the satuated fluid temperature and no cladding rupture nor metal-water reaction occurs. Thus, the criteria of 10 CFR 50.46 is satisfied for this transient.

3. Results of Analysis

3.1 Method

Since the system response for this transient is relatively quiescient, detailed noding of the primary system is not required, thus, the analysis in this report was performed using a six-node CRAFT model to develop the history of the reactor coolant system hydrodynamics. Figure 1 shows a schematic diagram of the model. Node 1 comprises the cold leg pump discharge piping, the reactor vessel (RV) downcomer, and the lower plenum of the RV. Node 2 represents the steam generator, primary side and the cold legs suction piping, while Node 3 represents the core, RV upper plenum, and the hot leg piping. Nodes 4, 5, and 6 of the model are used to simulate the pressurizer, containment, and the secondary side of the steam generators, respectively.

The assumptions used in the analysis are listed below:

1. The reactor is operating at 102% of the steady-state power level of 2772 MWt.
2. Loss of main feedwater flow to the steam generator occurs at time zero. The auxiliary feedwater systems are assumed not to operate.
3. Offsite power is not available.
4. The reactor trips on high pressure at 2300 psig.
5. No credit is taken for operation of the PORV.
6. The pressurizer safety valves start to open at the set pressure of 2500 psig. They are assumed to be full open at 103% of the set pressure.
7. The discharge rate through the code safety valves is calculated using the Bernoulli equation, for subcooled fluid discharging through the valve, and the Moody correlation, for two-phase or steam flow through the valve. The flow area utilized for the safety valves was chosen such that the Moody calculated discharge rate, for steam flow through the valve at the valve rated pressure, is equivalent to the design capacity of the valve.
8. Actuation of one HPI train, via operator action at 20 minutes, is assumed. A single failure is assumed which renders the other HPI train inoperable. Operator guidelines specify that, upon loss of SG heat sink, he should manually actuate all HPI trains.
9. In order to simulate a realistic decay heat curve, 1.0 times the 1971 ANS standard was utilized.

3.2 Results

Figures 2 through 5 show the transient system response for this accident. The following table presents key results of the analysis:

Sequence of events	Time, s
Loss of main feedwater, turbine trip, loss of offsite power (RC pumps coastdown)	0.
Reactor trips on high pressure	8.
SG side inventory boiled-off	100.0
Pressurizer goes solid	350.
Two pressurizer code safeties open	400.
Long term cooling estab. (based on 1 HPI)	8900.

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Figure 2 shows the core pressure transient. Following the simultaneous loss of main feedwater and offsite power, the fluid in the RCS expands due to decreasing heat transfer via the steam generator, and the RCS pressure increases. At 8 seconds, the high pressure trip setpoint (2300 psig) is reached, thus causing the reactor to scram. Pressure then starts to decrease due to contraction of the fluid in the RCS caused by the decrease in core power. At 100 seconds, the steam generator inventory has been boiled-off, which results in a loss of heat sink and a heatup of the RCS fluid, and repressurization of the system. At 350 seconds, the pressurizer becomes "solid," and the system pressure rapidly increases to the code safety valve set pressure of 2500 psig. The system pressure remains at this value for the remainder of the transient with the safety valves acting as a path to the ultimate heat sink of the RCS for this accident, i.e., the containment.

Figure 3 shows the pressurizer mixture level response for this accident. The initial pressurizer response follows the same behavior as the system pressure transient. Due to the loss of heat sink at 100 seconds, the pressurizer starts to refill and becomes solid at 350 seconds. At 2180 seconds, the liquid level in the primary system falls below the surgeline entrance and steam passes into the pressurizer. Shortly thereafter, the pressurizer mixture level drops slightly and steam exits through the code safety valves. The pressurizer remains in this condition for the remainder of the transient.

The RCS liquid inventory, with the exclusion of the pressurizer, during this transient, is given in Figure 5. Makeup to the RCS was initiated at 1200 seconds via operator action to start one HPI train. The RCS reached saturated conditions at 1725 seconds, and rapidly starts decreasing in inventory. With the liquid level in the primary system falling below the pressurizer surgeline nozzle at 2180 seconds, the loss rate in system inventory slows. At 8900 seconds, the HPI flow exceeds the core boil-off and the system starts to refill. At no time does the core uncover. Thus, the cladding temperatures will be maintained within a few degrees of the saturated fluid temperature and no cladding ruptures nor metal-water reaction will occur.

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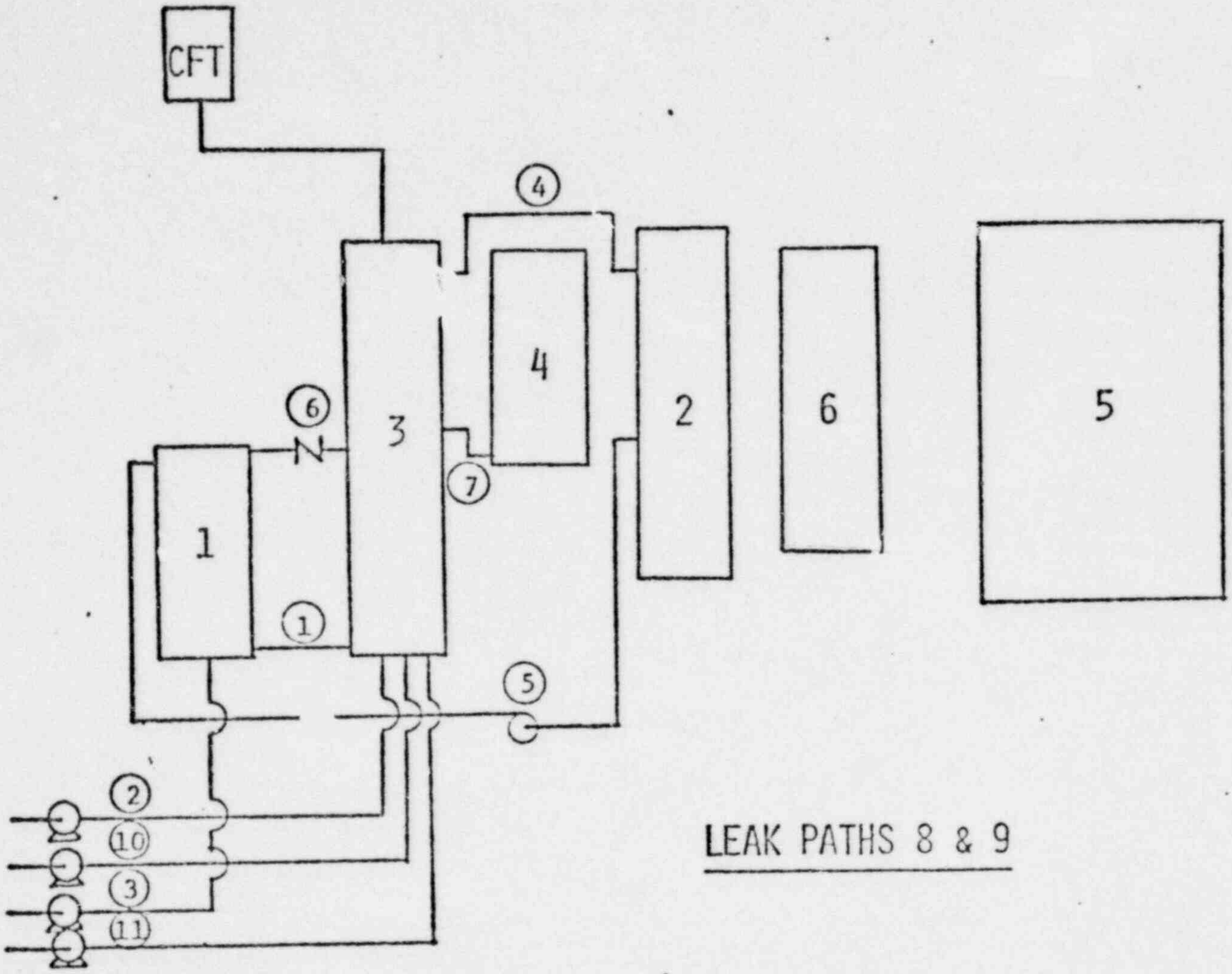
FIGURE 1

CRAFT2 NODING DIAGRAM FOR SMALL BREAKS

(6 NODE MODEL)



Node No.	Identification	Path No.	Identification
1	PD Piping, DC, LP	1	Core
2	Primary SG	2	LPI
3	Core, UP, Hot Legs	3,10,11	HPI
4	Pressurizer	4	Hot Legs
5	Containment	5	Pumps
6	Secondary SG	6	Vent Valve
		7	Pressurizer
		8,9	Leak & Return Path

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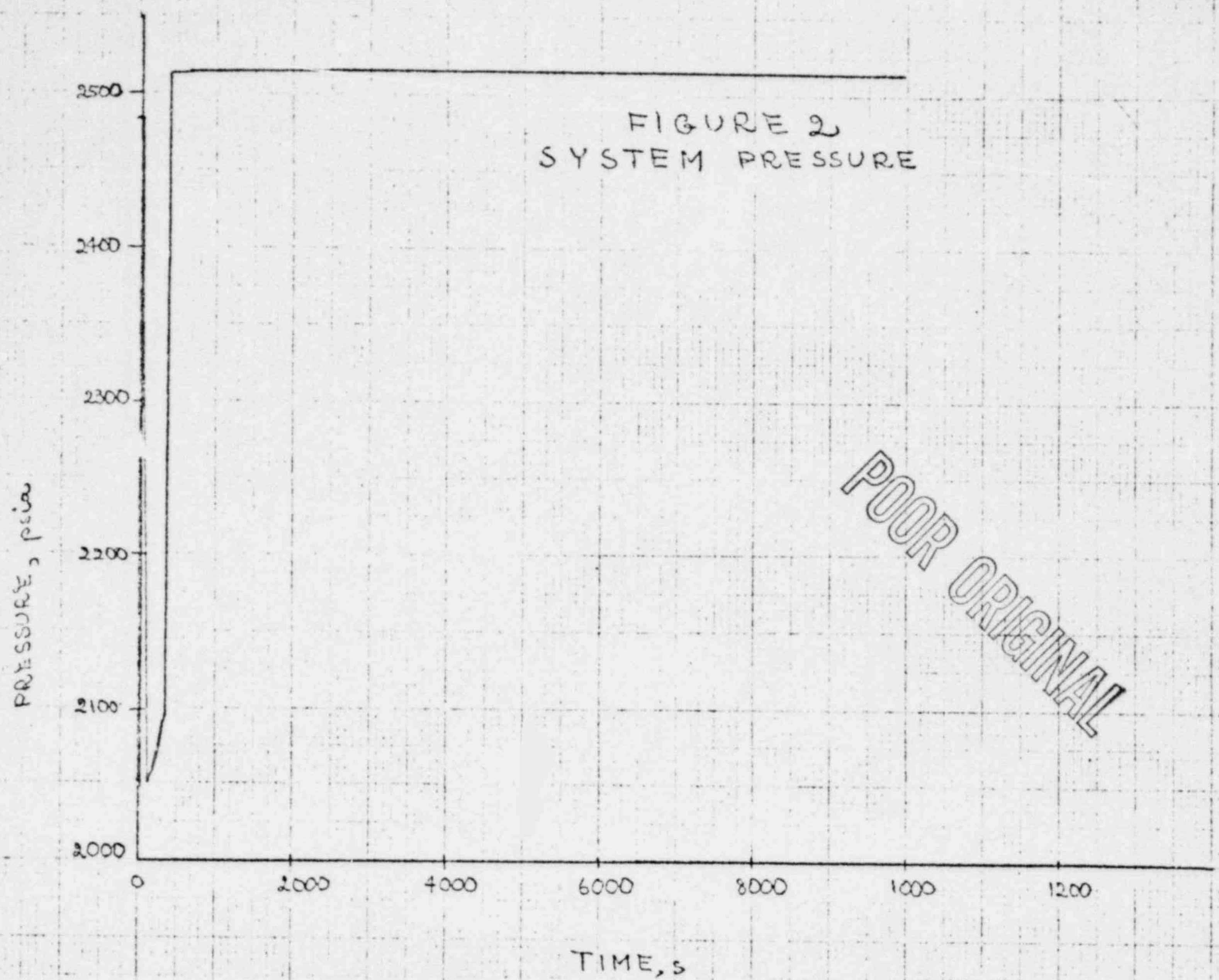

FIGURE 2
SYSTEM PRESSURE

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FIGURE 3

PRESSURIZER MIXTURE LEVEL

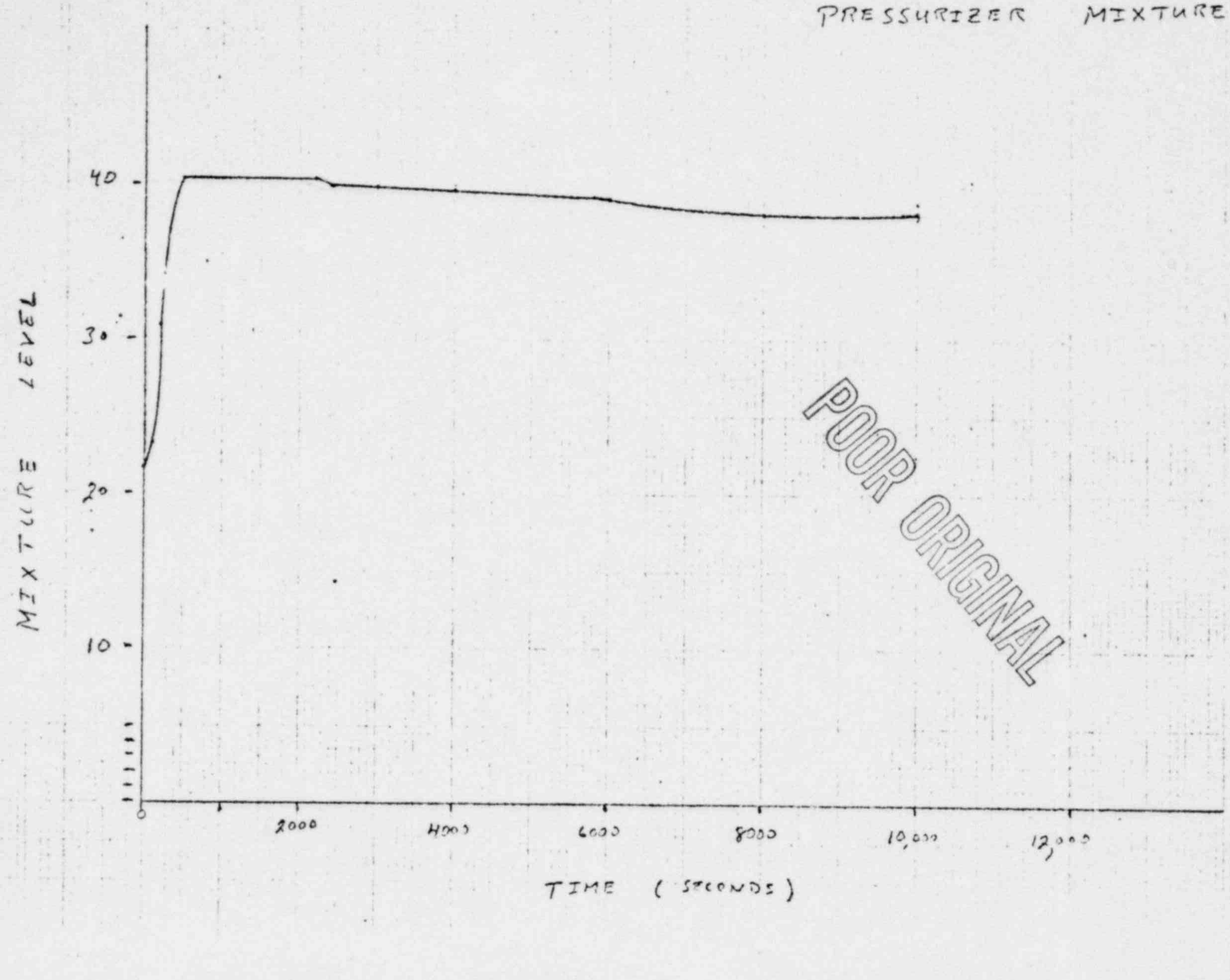

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FIGURE 4
INTEGRATED PRESSURIZER SAFETIES FLOW

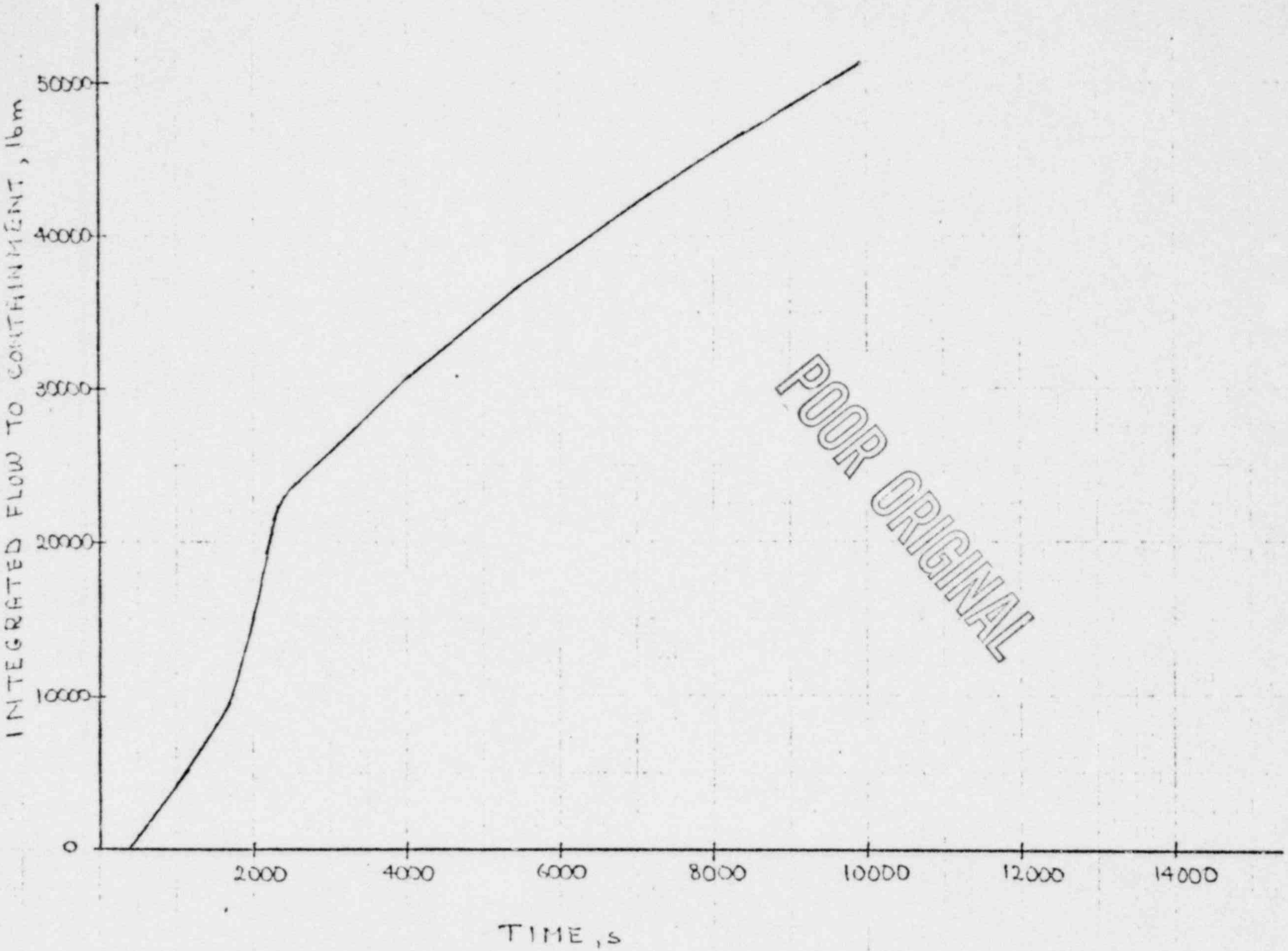
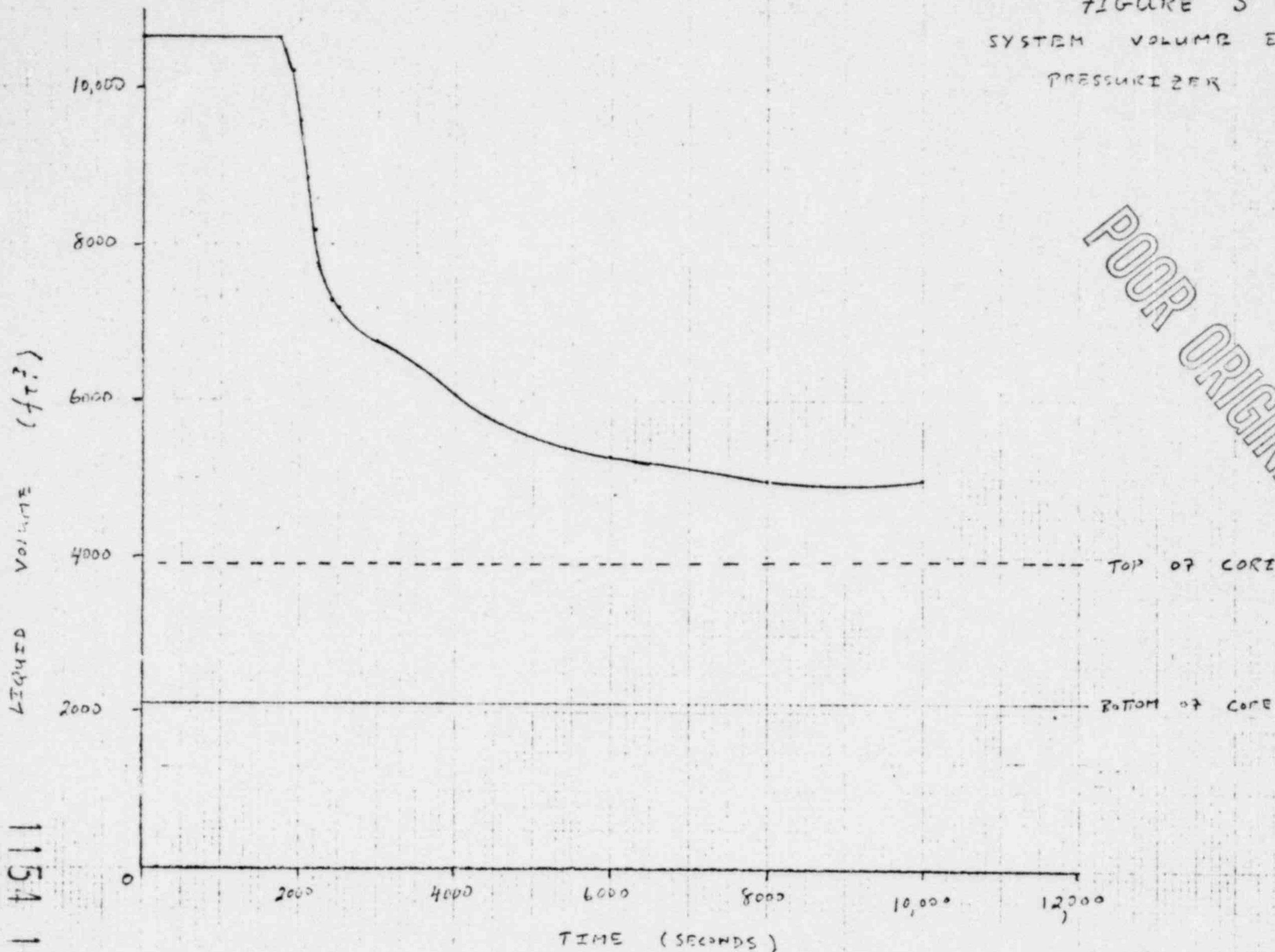


FIGURE 5

SYSTEM VOLUME EXCLUDING PRESSURIZER



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RESPONSE TO QUESTION 2A OF ATTACHMENT A TO D. F. ROSS (NRC) LETTER DATED 8/21/79

Question

2A. Provide justification that the 3 node steam generator model used in the CRAFT2 analysis of small breaks is adequate for the prediction of steam generator heat transfer.

RESPONSE

The B&W ECCS Evaluation Model[1] for small breaks, utillizes a three-node representation in the CRAFT2 simulation for the prediction of steam generator heat transfer following a small break. Two of the nodes, stacked vertically, are used to model the primary side of the once-through steam generator (OTSG). The upper node includes the hot leg piping, from the center on the 180° U-bends at the top of the vertical section of the hot leg to the SG upper head, the upper head of the SG, and the upper one-half of the tube region. The lower node simulates the lower one-half of the tube region. The third node is used to model the secondary side of the OTSG. To evaluate the suitability of this modeling technique, the unique characteristics of the OTSG and its effects on the small break transient must be examined. As is shown later, for small breaks evaluated with the auxiliary feedwater system operable, heat removal via the SG is not necessary for the worst case breaks, i.e., those that result in core uncovery, in order to successfully mitigate the transient. For the smaller breaks, heat removal via the SG is necessary. The three-node representation utilized appropriately models the heat transfer characteristics of the OTSG. For the smaller breaks, heat removal via the steam generator is necessary and the heat transfer characteristics of the OTSG must be appropriately considered. Although the three-node SG model does not rigorously account for the heat transfer process that will occur, it does provide a reasonable representation of the effects of these heat transfer processes in the OTSG. Since these smaller breaks exhibit large margin to core uncovery, the CRAFT2 SG model is adequate for demonstrating compliance to 10 CFR 50.46.

In performing small break evaluations, the CRAFT2[2] code is used to predict the hydrodynamic response of the primary system including the effect of SG heat transfer during the transient. The option 2 SG model, which is explained in detail in Section 2.6 of topical report BAW-10092, Rev.2, is utilized to predict heat flow in the SG. THe calculation progresses basically as follows:

1. Based upon the initial steady-state heat transfer characteristics of the OTSG and the initial primary and secondary fluid temperatures, an overall UA for each region of the SG is calculated.

2. The calculated steady-state UA can be modified by user-specified input options. These include an input multiplier table versus time, multiplied based on the primary side control voluime mixture height during the transient, and a multiplier for reverse heat transfer, i.e., heat flow from the secondary to the primary side of the SG.

3. Using the modified UA and the calculated primary and secondary side control volume temperatures, the amount of heat transferred is calculated.

RESPONSE (2A)(Continued)

In performing the small leak calculations for demonstrating compliance to 10 CFR 50.46 for the operating B&W plants, no input multiplier versus time is utilized, nor is the modification based on primary side mixture level used. However, a multiplier for reverse heat transfer of 0.1 is utilized. This multiplier and its basis is explained in the ECCS evaluation model topical report[1] and is utilized to reflect the change in heat transfer regime on the secondary side of the SG for reverse heat flow.

The OTSG design of the B&W designed operating NSSs allows use of a simplistic model for calculation of SG performance during a small LOCA transient. With the loss-of-offsite power, assumed in design calculations for small breaks, and the subsequent loss of main feedwater, the auxiliary feedwater system is actuated and will become operable in approximately 40 seconds and control the secondary side level. The auxiliary feedwater enters the SG very high, approximately 2 feet below the upper SG tube sheet, and is sprayed onto the tube bundles. Thus, heat transfer will occur in the upper portion of the SG independent of the actual level in the SG. The introduction of auxiliary feedwater to the SG has two effects on the small LOCA transient. First, it raises the thermal center in the SG during the natural circulation phase of the accident, which results in a continuation of circulation through the RCS, for some period of time, even while inventory is lost from the primary system. Later in the transient, after sufficient inventory has been lost from the system, circulation will be interrupted and the auxiliary feedwater, for a certain range of small breaks, will condense steam on the primary side of the SG; thereby maintaining the primary system pressure near the secondary side pressure. The analytical approach utilized for the small break evaluation is consistent with this performance of the auxiliary feedwater system.

It should be noted that between the time that circulation through the loops is lost and the time that the primary side SG level has dropped to the point where condensation heat transfer will occur, system repressurization can occur as heat removal via the SG will be lost. This phenomena occurs only for the very small sized small breaks in which the SG heat removal is necessary. If simulation of this repressurization phenomena of the very small breaks is desired, an additional node would be needed in the small break model in order to separate the hot leg and SG upper plenum volumes from the tube region. This will allow steam to accumulate in the upper regions of the RCS without being affected by heat removal that occurs in the steam generator. In the analyses presented in reference 5 for these smaller sized breaks, a model which included the additional node was utilized and showed that the repressurization phenomena does not result in core uncovery.

It is also important to note the role of the SG on the small break transient in order to evaluate the appropriateness of the SG model utilized in small break evaluations. Licensing calculations for the operating B&W units have previously been submitted to the Staff in references 3 and 4. These evaluations have shown that the worst case small breaks, i.e., breaks which result in core uncovery, occur for breaks in excess of 0.05 ft^2. As demonstrated in the May 7, 1979 report[5], SG heat removal is not necessary for breaks of this size. For smaller breaks, SG heat removal is necessary as the break alone is not sufficient to remove enough fluid volume and energy to depressurize the RCS. However, as demonstrated in reference 5, these breaks are of no consequence as the SG heat removal and the slower discharge rate for these breaks easily prevents core uncovery.

RESPONSE (2A)(Continued)

As demonstrated, the SG model utilized in the small break evaluation for the operating plants appropriately accounts for the effect of the spatial heat removal processes that will occur in the OTSG during a small break. It was also shown that the SG performance is not important for the worse case small breaks. Thus, the CRAFT2 SG model is adequate for demonstrating compliance of the ECCS to 10 CFR 50.46.

REFERENCES

1 BAW-10104, Rev. 3, "B&W's ECCS Evaluation Model", B. M. Dun, et al., August 1977.

2 BAW-10092, Rev. 2, "CRAFT2 - FORTRAN Program for Digital Simulation of a Multinode Reactor Plant During LOCA", R. A. Hedrick, J. J. Cudlin, and R. C. Foltz, April 1975.

3 Letter J. H. Taylor (B&W) to S. A. Varga (NRC), July 18, 1979.

4 BAW-10075A, Rev. 1, "Multinode ANalysis of Small Breaks for B&W's 177-Fuel Assembly Nuclear Plants With Raised Loop Arrangement and Internals Vent Valves", Babcock & Wilcox, March 1976.

5 "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177-Fuel Assembly Plant", Babcock & Wilcox, transmitted via letter from J. H. Taylor to R. J. Mattson, dated May 7, 1979.

RESPONSE TO QUESTION 3C OF ATTACHMENT A TO D. F. ROSS
(NRC) LETTER DATED 8/21/79

Question

3C. Describe any operator actions and/or emergency procedures necessary to preclude introduction of significant quantities of non-condensible gases into the primary system.

RESPONSE

Present provisions of the small break emergency procedures preclude the generation of significant quantities of non-condensible gases by preventing core uncovery. Therefore, no revision to these actions is necessary.

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RESPONSE TO QUESTION 3D OF ATTACHMENT A TO D. F. ROSS
(NRC) LETTER DATED 8/21/79

Question

3D. Describe operator actions to be taken in event of a significant accumulation of non-condensable gases in the primary system.

RESPONSE

The review and development of operator actions to be taken in the event of a significant accumulation of noncondensable gases in the primary system is incomplete pending finalization of small break impact studies. Preliminary investigations are centered upon (1) review of present provisions within the small break emergency procedures (e.g., restart of one or more RC pumps and/or remote manual operation of the PORV) which may be effective in minimizing the impact on system response, and (2) use of present plant features (e.g., pressurizer spray and venting and/or letdown to the makeup tank) to degass the reactor coolant in the long term following small breaks where a return to natural circulation is possible. A complete response to Question 3D will be provided by November 1, 1979.

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RESPONSE TO QUESTION 4 OF ATTACHMENT A TO D. F. ROSS
(NRC) LETTER DATED 8/21/79

Question

4. Provide a CRAFT-2 simulation for the first three hours of the TMI-2 accident. The first 20 minutes of this analysis was provided in the "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" (May 7, 1979). We require that the analysis be extended for a period of three hours in order to evaluate the ability of the CRAFT-2 code to evaluate the sequential reactor coolant pump trips and the subsequent period in which natural circulation was lost in the primary system. The analysis should include at least curves for the following paramaters: pressure, temperature, void fraction, and flow in the reactor coolant loops.

RESPONSE

See attached report, CRAFT2 SIMULATION OF THE MARCH 28, 1978 TMI-2 TRANSIENT.

CRAFT2 SIMULATION OF THE MARCH 28, 1978

TMI-2 TRANSIENT

I. INTRODUCTION

In the May 7, 19'9 "Blue Book" reports[1], a CRAFT2 simulation of the first hour of the TMI-2 transient was presented. That analysis has since been modified and updated to include more recent estimates of the net makeup to the RCS during the event. This report presents the results of the latest B&W CRAFT2 simulation of the TMI-2 event and covers approximately the first 2 hours and 20 minutes of the transient.

The small break ECCS evaluation model, which is described in topical report BAW-10104[2] and the July 18, 1978 letter report[3], was used,with some "best estimate" modifications, for the simulation. Actual TMI-2 data were combined with available information about the operator actions to determine estimates of the HPI and AFW injection times and flow rates.

The simulator results (described in detail in the "Results" section) show all the trends and very good comparisons to the actual plant data of system pressure, temperature and pressurizer level. The analysis also predicts the time for the start of core uncovery which is in reasonable agreement with the NSAC-1[4] report. Thus, the CRAFT2 code is shown to benchmark very well versus the TMI-2 data and is suitable for the performances of small break evaluations.

2. METHOD OF ANALYSIS

The CRAFT2 code which is documented in topical report BAW-10092[5], was used to simulate the TMI-2 reactor coolant system hydrodynamics. The model uses one node for the reactor building, two nodes for the secondary system, and 23 nodes to simulate the reactor coolant system, including four nodes for the pressurizer. A schematic diagram of the model is shown in Figure 1.

The analytical model used for this simulation is basically the same as B&W's ECCS evaluation model. However, certain input assumptions which differ from the evaluation model approach, were made in order to obtain a "best estimate" simulation. These assumptions are described below:

a. The initial core power level used in the model was 102% of 2772. However, following reactor trip, the fission product decay heat was adjusted to 98% power operation. The decay heat curve utilized

is 100% , instead of 120% required by Appendix K to 10 CFR 50, of the ANS 5.1 decay heat curve.

b. A loss of the main feedwater pumps, which is the initiating transient, was assumed at time zero. In order to account for potential draining of secondary side fluid from the steam generator downcomer into the tube region, a main feedwater coast-down of 10 seconds was utilized.

c. A turbine trip coincident with the loss of main feedwater was assumed. This results in the steam generator pressure being controlled by a combination of the turbine bypass valves, the atmospheric dump valves and the main steam safety valves. For the first 90 minutes, the turbine bypass valves control the secondary side pressure. In the simulation, these valves were set at 1025 psig.

d. The CRAFT2 input was adjusted to open the pilot-operated relief valve (PORV) at 8 seconds. This opening time had to be input, and the open valve simulated, since CRAFT2 code does not have models for the pressure relief systems of the RCS. Preliminary TMI-2 data was used to determine the PORV opening time. Present TMI-2 scenarios[4] indicate that the PORV actually opened at 3 seconds. As will be shown in the results section, if the CRAFT2 code had an explicit PORV model, it would have predicted the opening at 3 seconds.

e. The reactor scram was chosen to occur at 10 seconds based on preliminary TMI-2 data. Since the CRAFT2 code does not have provisions for a reactor trip on high pressure, this had to be simulated based on time.

f. The leak area utilized for the PORV is 1.05 in.^{2} and represents the orifice area of the valve. The Moody critical discharge correlation was utilized to predict the fluid lost through the PORV. For the first 4½ minutes of the simulation, a discharge coefficient (C_D) of 0.8 was used. For the remainder of the evaluation a C_D of 1.0 was employed.

g. Actuation of the High Pressure Injection System (HPI) was based on ESFAS signal of 1615 psia. This resulted in the actuation of the 2 HPI systems at 1 minute and 45 seconds into the transient, as opposed to 2 minutes and 2 seconds which was the make-up flow initiation time at TMI-2. Between 275 and 6100 seconds, the HPI flow was assumed to be throttled by the operator to an average flow of ≈ 34 gpm. This value is based on preliminary assessment on the <u>net</u> makeup flow to the RCS. No explicit modeling of letdown was used, only net flow was simulated. After 6100 seconds, an average net makeup (HPI) of 42 gpm was utilized.

h. A four-node pressurizer model was used in the evaluation in order to reduce instantaneous artificial condensation in the pressurizer. This phenomenon, which occurs when the subcooled reactor coolant fluid mixes with two-phase pressurizer fluid, results from the equilibrium model limitations of the code. This model is necessary only to predict the response of the RCS during the initial phase of the loss of main feedwater event. Also, the pressurizer surge line resistance was updated to reflect more realistically the TMI-2 surge line.

i. Steam Generator Modeling - The steam generator model was modified to account for the following phenomena:

   1. The overall heat transfer coefficient (between primary and secondary) was assumed to ramp to zero in one minute to account for the delayed auxiliary feedwater injection.

   2. Full heat transfer coefficient was reinstated at 500 seconds to account for the auxiliary feedwater injection after 8 minutes.

   3. Auxiliary feedwater was initiated at 500 seconds with half of the design AFW flow capacity and with the SG level controlled to 3 feet. With the reactor coolant pumps on, AFW is controlled by the ICS to 3 feet.

   4. Steam Generator B was assumed to be isolated at 1 hour and 41 minutes based upon preliminary TMI-2 data. This was simulated by setting the heat transfer coefficient across the B steam generator to zero.

5. The auxiliary feedwater control level was manually raised to 50% on the operating range at 1 hour and 45 minutes into the transient due to the loss of the RC pumps.

6. The main steam safety valves were modeled to open at 5400 seconds, and the feedwater flow was increased at 6100 seconds. This was done to simulate the steam generator A depressurization following the operator's attempt to increase feedwater flow to steam generator A at about 1 hour and 34 minutes.

j. The RC pumps in the B loop were tripped at 4400 seconds; the A loop RC pumps were tripped at 6060 seconds. These values are consistent with the TMI-2 data.

Table 1 provides a comparison of the assumed times for various system actuations and operator actions to the NSAC scenario. As shown, the values utilized are reasonable compared to the actual performance during the TMI-2 transient.

3. RESULTS

3.1 System Pressure

Figures 2 and 3 compare the reactor coolant system pressure calculated by CRAFT2 to the TMI-2 data. Following the loss of main feedwater, the pressure in the RCS rose sharply due to the decreased heat removal across the SG. As shown by Figure 2, the CRAFT2 prediction overpredicts the pressure during this phase of the accident due to the delayed opening of the PORV 3 seconds in the transient versus 8 seconds for the CRAFT2 simulation, and the delayed reactor trip, 8 seconds for the transient versus 10 seconds for the simulation. If the CRAFT code had an explicit model for the PORV, opening of the valve would have been consistent with the data and a better comparison would have been obtained. After the reactor tripped, the RC pressure decreased. The calculated pressure drops below the actual data after 20 seconds. This is apparently caused by the 10 second main feedwater coastdown employed in the simulation overpredicting the drainage of secondary downcomer fluid to the SG. After the SG dries out, approximately one minute, the difference between the prediction and the data decreases.

Approximately 5 minutes into the transient, the fluid in the hot leg flashed due to the depressurization of the RCS and the system pressure increased. As indicated on Figure 3, the CRAFT2 code properly predicts the system repressurization time, but overpredicts the actual pressure. The overprediction of system pressure is probably caused by the assumed net makeup to the RCS during this time period. Although the HPI was throttled to a net makeup of 34 gpm during this time interval in the simulation, between 4:58 and 6:58, the NSAC scenario of events indicate that the letdown flow was in excess of 160 gpm. Thus, it is quite probable that there was a decrease in inventory in the RCS due to the high letdown over this time period.

At 8 minutes and 18 seconds, auxiliary feedwater flow was readmitted to the SG and primary system pressure decreased (Figure 3) to approximately 1100 psig and was maintained at that value up to approximately one hour and 20 minutes.

As shown by Figure 3, the CRAFT2 prediction is greater over this period by about 100 psi. The coolant pressure was measured in the hot leg during the accident; the predicted system pressure shown is the core pressure. The actual predicted hot leg pressure is about 60 psi lower than the predicted core pressure. Also, the pressure in the secondary side was held in the CRAFT2

simulation at 1025 psig, while the measured value was 1000 psig, resulting in an additional deviation. Thus, the CRAFT2 prediction reasonably follows the transient behavior over this period when the deviations are considered. It should also be noted that the primary system pressure during this phase of the transient is basically controlled by the SG. The CRAFT2 prediction did not demonstrate fluctuations in system pressure during this period as the secondary pressure of the SG is assumed to be regulated at 1025 psig. The plant data shows that the secondary side SG pressure was not held constant over this period, but fluctuated.

At one hour and 34 minutes, the RCS pressure dropped due to an apparent attempt by the operator to increase feedwater to the A SG. The analysis attempted to simulate the depressurization effect of the increased auxiliary feedwater flow by opening the relief valves at 5400 seconds and increasing the auxiliary feedwater flow at 6100 seconds. This modeling technique was utilized as little information is available on the actual auxiliary feedwater flow delivered to the SG during this period. As shown by Figure 3, this resulted in an underprediction of the primary system pressure until 7500 seconds and an overprediction for the remainder of the transient analyzed.

3.2 Pressurizer Level

A comparison of the CRAFT2 predicted pressurizer level to the TMI-2 data is provided in Figure 4. As shown, there are two pressurizer level predictions given in the figure. The first, entitled mixture level - CRAFT, is the calculated mixture level within the pressurizer. The second, entitled instrumentation reading - CRAFT, is the calculated liquid level that would be "seen" within the pressurizer level taps and is directly comparable to the TMI-2 data.

The initial pressurizer response and comparison to the loss of main feedwater event (first 4 minutes of the transient) is not easily discernable in Figure 4. It was, however, discussed in the May 7, 1979, report. During this phase of the accident, the pressurizer level responded in a similar manner as the system pressure. Also, the comparison of the predicted to the actual response of the pressurizer level is similar. That is, the rise in pressurizer level during the first 10 seconds is overpredicted and the pressurizer level after reactor trip is underpredicted. The reasons for this are the same as those discussed previously in section 3.1.

The significant aspect of this comparison is the predicted mixture level response to the predicted instrument reading response during the transient. As shown by Figure 4, the predicted instrument response and the measured response are in good agreement throughout the simulation. However, as shown by the figure, although the instrument reading is on scale for portions of the first 101 minutes of the transient, the actual predicted mixture level after 6 minutes is at the top of the pressurizer. Thus, a two-phase mixture exited through the valve during this entire period. After 101 minutes, only steam was entering the pressurizer through the surge line (note that the RC pumps have been tripped), and the pressurizer mixture had reached a sufficient void fraction to allow for phase separation at the top of the mixture and only steam started to flow out.

3.3 System Flow

Figure 5 shows a comparison of the predicted and transient loop flows. As shown, the predicted flow rates do not match well with the actual data. This disagreement is caused by two factors. First, loop flow was measured by Gentillis tubes, which are calibrated based on single phase flow. Their actual performance during two-phase flow is unknown. Secondly, performance of the RC pumps with two-phase flow is not well understood. In performing the evaluation, a two-phase pump degradation multiplier based on the semiscale pump tests was utilized. This multiplier results in a sharp decrease in pump head once any significant voiding is calculated at the pump inlet. As shown, at 55 minutes, the loop flow sharply decreased due to this effect. Although the agreement is not excellent, the pump flow calculation does not appear to have significantly affected the simulation.

3.4 Hot and Cold Leg Temperatures

Figures 6 and 7 show a comparison of the predicted versus actual response of the hot and cold leg temperature measurements during the transient. After 5 minutes and up to the time the core started to uncover, the RCS was saturated, and the fluid temperature comparison has the same deviations previously discussed in section 3.1.

After the core starts to uncover, which occurs at approximately 110 minutes, the hot leg temperature measurement indicated superheated steam (Figure 6). However, the CRAFT2 prediction does not exhibit this behavior. This is due to the one-node representation of the core and the equilibrium assumption of the

CF node. As long as fluid is predicted to remain within the core node, regardless of the actual amount of core uncovery, the one-node representation calculates the exiting steam temperature to be saturated. However, the actual physical process results in saturated steam at the top of the core mixture level. This steam superheats as it receives energy from the uncovered portion of the fuel pins. A multinode representation of the core would be necessary to predict the hot leg temperature response during this period.

3.5 System Void Fraction

The average system void fraction evolution for the primary system, excluding the pressurizer, is shown on Figure 9. Due to the continued loss of RCS inventory through the PORV and the inadequate net makeup to the RCS, the system void fraction increases almost linearly from 10 until 101 minutes into the transient. At 101 minutes all the RC pumps have been tripped. At this time, the RCS liquid inventory is distributed as follows; the RV is filled to slightly above the top of the core; the loop seal in the B loop is full; the A loop has very little inventory. During the subsequent 30 minutes, the RV inventory is boiled-off and the steam is condensed by the A loop steam generator. Because of the lowered loop design, this inventory remains trapped within the A loop pump suction piping and the steam generator. During this period of time, the core becomes uncovered. Thus, since the process is a redistribution of water within the RCS with the only fluid loss being steam vented through the PORV, the system average void fraction does not change significantly.

3.6 Core Mixture Level

The calculated core misture level for the transient is given in Figure 8. As shown, no core uncovery was calculated while the RC pumps were operating. However, closely following the termination of the RC pump flow, the level in the core decreased. Core uncovery was calculated to start occurring at 105 minutes into the transient. This compares reasonably well with the NSAC prediction of approximately 103 minutes. Thus, the calculated loss rate through the PORV and the net makeup to the RCS must be in reasonable agreement with the actual behavior during the TMI-2 incident.

As shown by Figure 8, the core was predicted to totally uncover. However, this result occurs due to the insufficient spatial detail in the core region. The simulation assumes that <u>all</u> core heat is removed and deposited in the fluid. This results in an overprediction of the core boil-off once the core is uncovered. A more detailed core model is necessary in order to predict how much core heat is deposited in the liquid region for subsequent boil-off of the

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core liquid and how much energy is used to superheat the steam. However, since the simulation was basically made to determine how the core uncovery occurred, the refined core model was deemed unnecessary.

4. CONCLUSIONS

As demonstrated, the CRAFT2 code simulation predicts reasonably well the system behavior during the first 2 hours and 20 minutes of the TMI-2 transient. In fact, the core uncovery time is predicted within a few minutes of the inferred core level response given in the NSAC report. Therefore, it is apparent that the net makeup to the RCS was very low (approximately 34 gpm) over this period which resulted in uncovery of the core and subsequent core damage. Also, it is shown that the CRAFT2 code is able to predict the system hydrodynamics during a small LOCA and is suitable for licensing calculations.

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4 "Analysis of Three Mile Island, Unit 2 Accident," NSAC-1, July, 1979.

5 R.A. Hedrick, J.J. Cudlin, and R.C. Foltz, "CRAFT2 - FORTRAN Program for Digital Simulation of a Multinode Reactor Plant During Loss-of-Coolant," BAW-10092, Rev. 2, Babcock & Wilcox, April 1975.

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Table 1. Comparison of CRAFT2 Assumption to NSAC Scenario

Event	Time, hrs: min: sec NSAC	CRAFT2
Loss of feedwater flow/turbine trip	0:0	0:0
PORV opens	0:03	0:08
Reactor trip	0:08	0:10
HPIs actuated	2:02	1:45
HPI throttled	4:38	4:35
Auxiliary feedwater block valves opened	8:18	8:20
Reactor coolant pump 2B stopped	1:13:29	1:13:33
Reactor coolant pump 1B stopped	1:13:42	1:13:33
Steam generator B isolated	1:42:00	1:41:40
SG A level raised to 50% on operate range	1:40:00	1:41:40
Reactor coolant pump 2A stopped	1:40:37	1:41:00
Reactor coolant pump 1A stopped	1:40:45	1:41:00

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FIGURE 1

CRAFT Noding Diagram for TMI-2 Transient

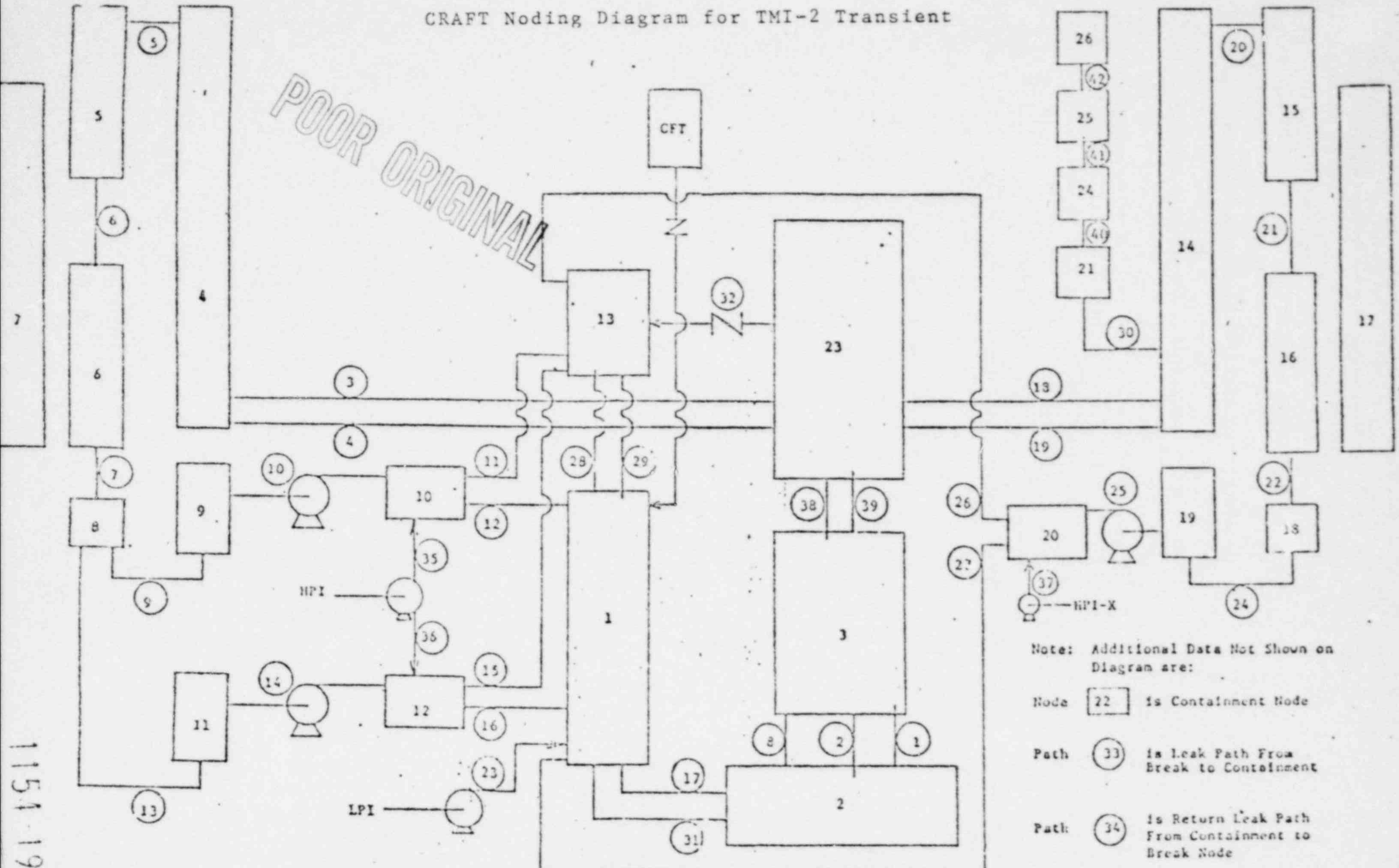

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FIGURE 2 SYSTEM PRESSURE VS TIME

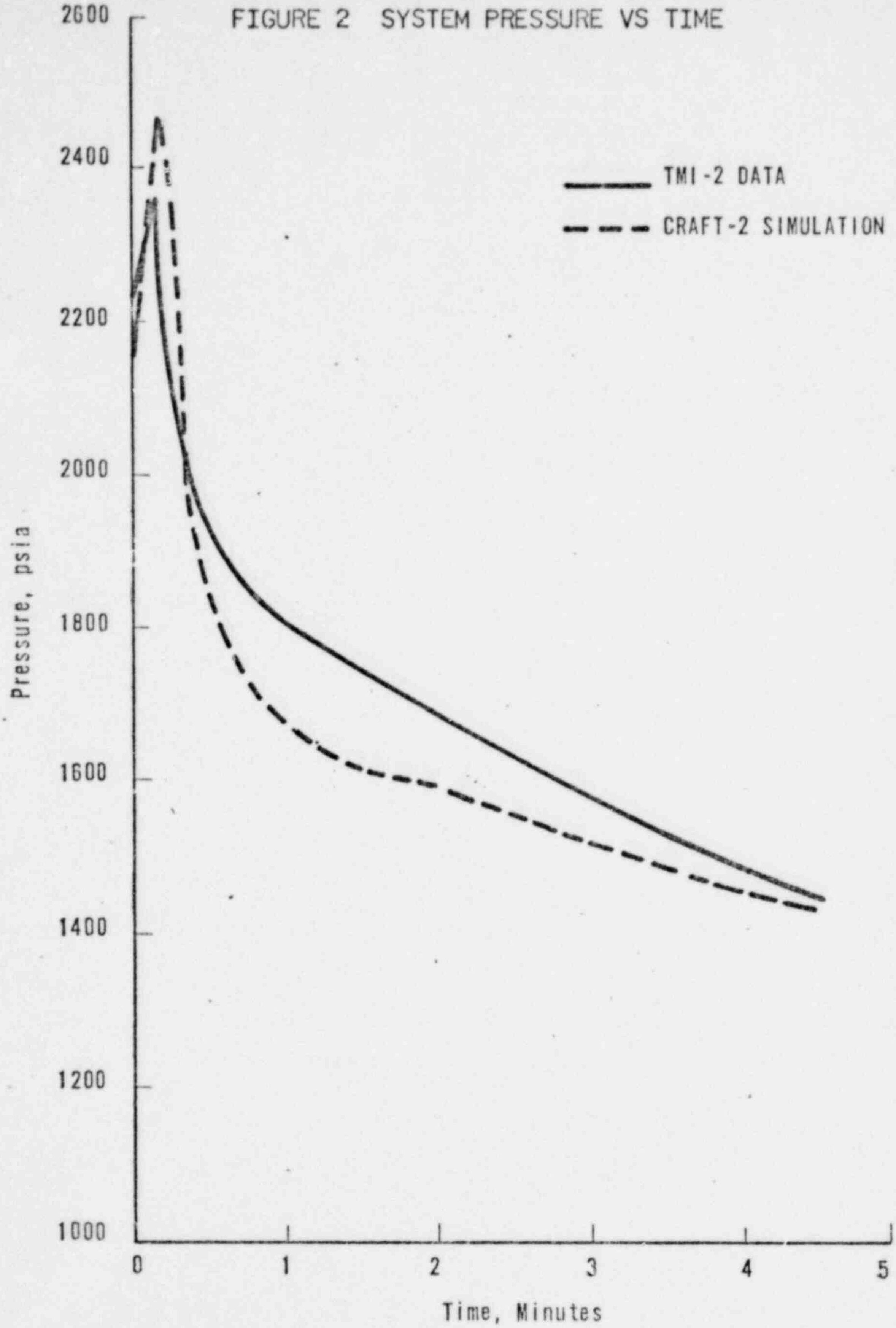

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FIGURE 3 REACTOR COOLANT PRESSURE

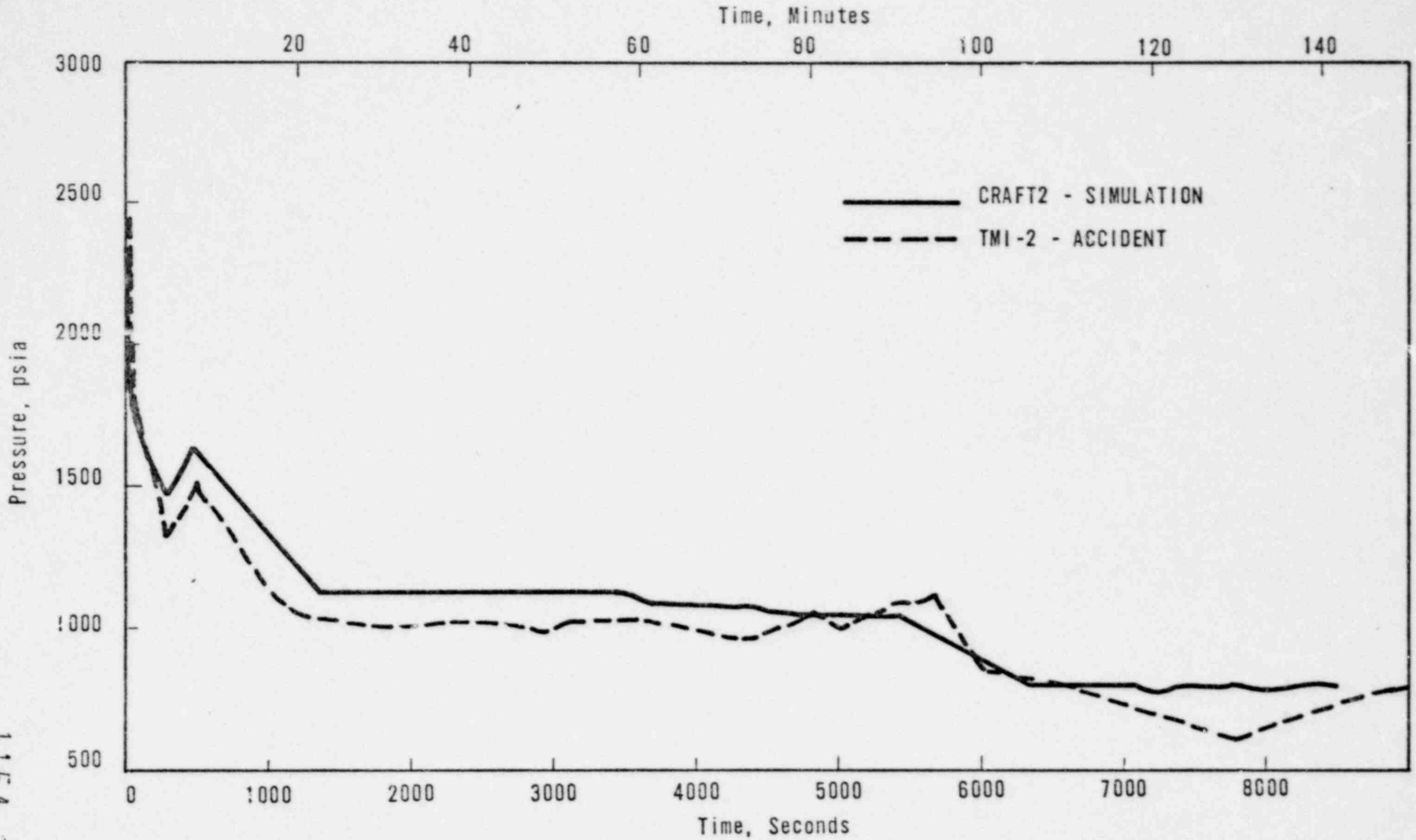

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FIGURE 4 PRESSURIZER LEVEL

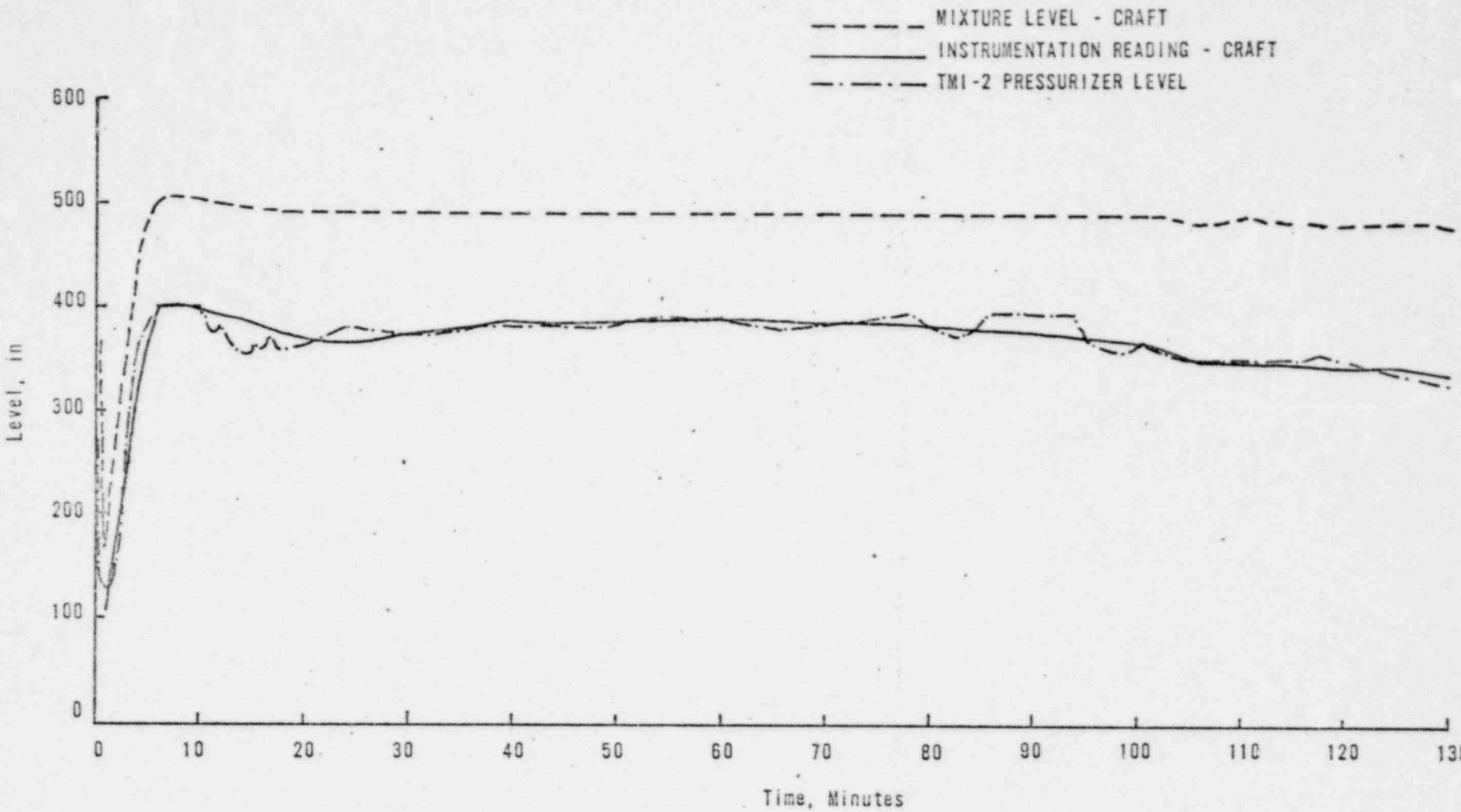

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FIGURE 5. HOT LEG FLOW

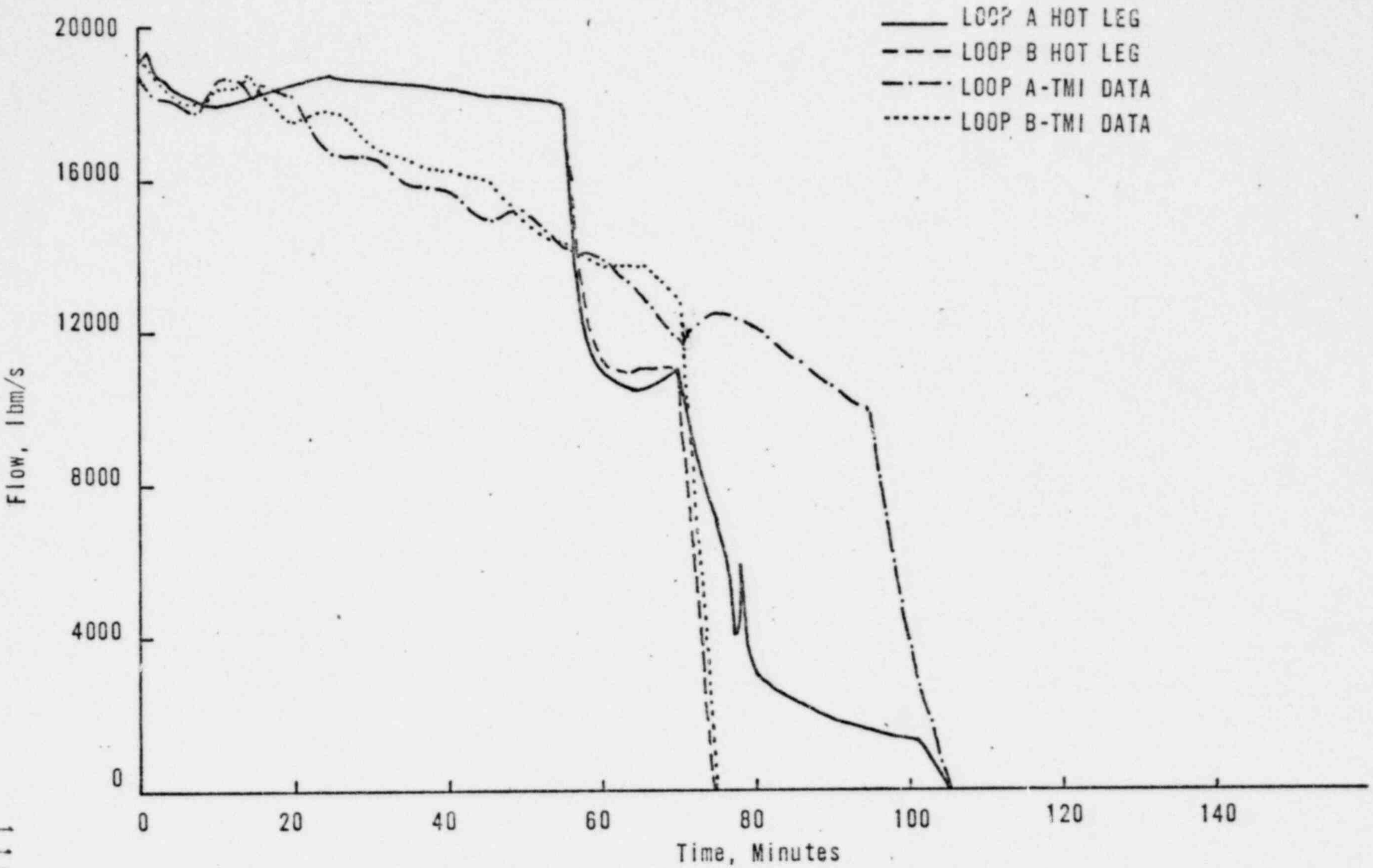

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FIGURE 6 HOT L[illegible] TEMPERATURE

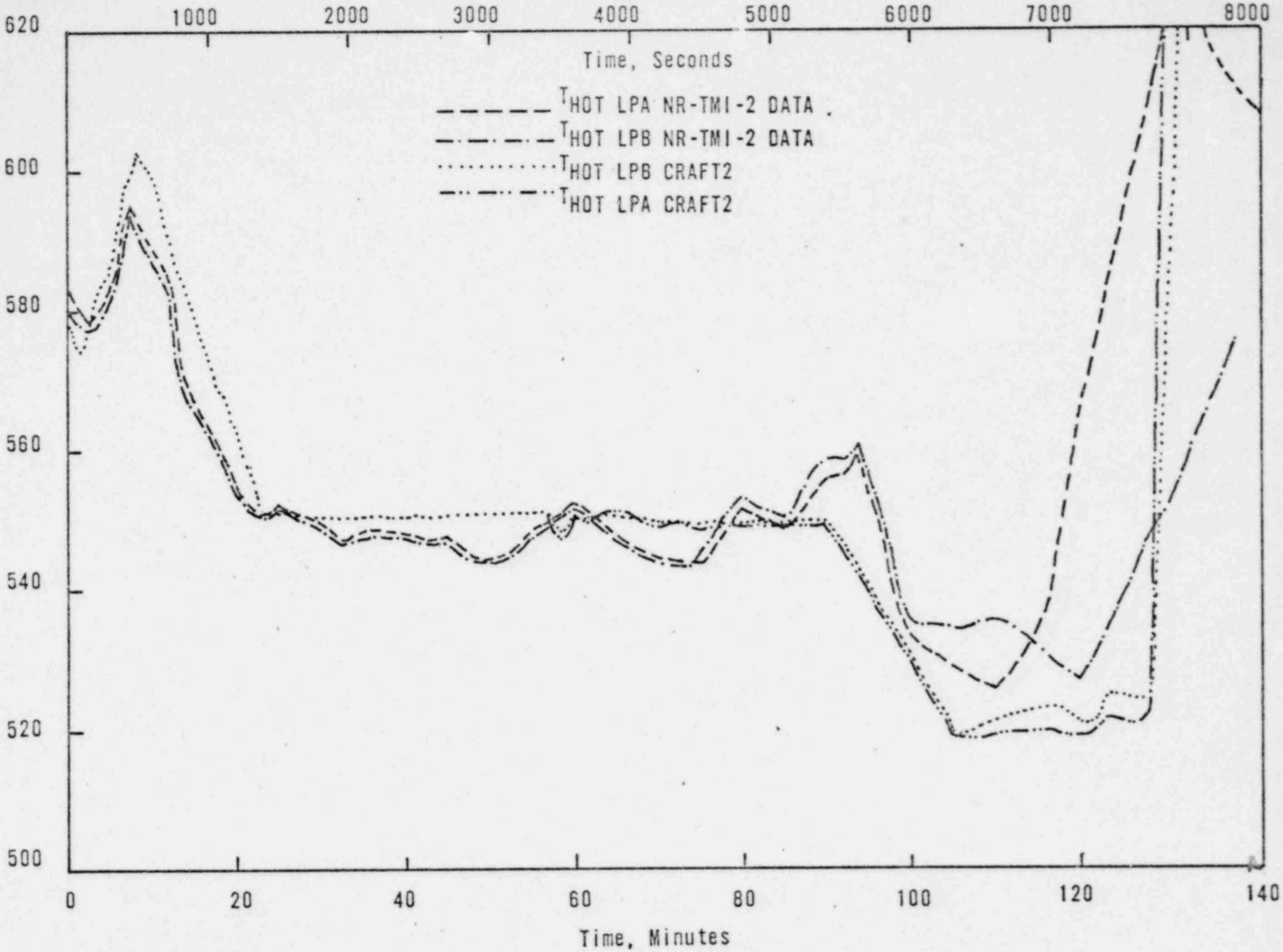

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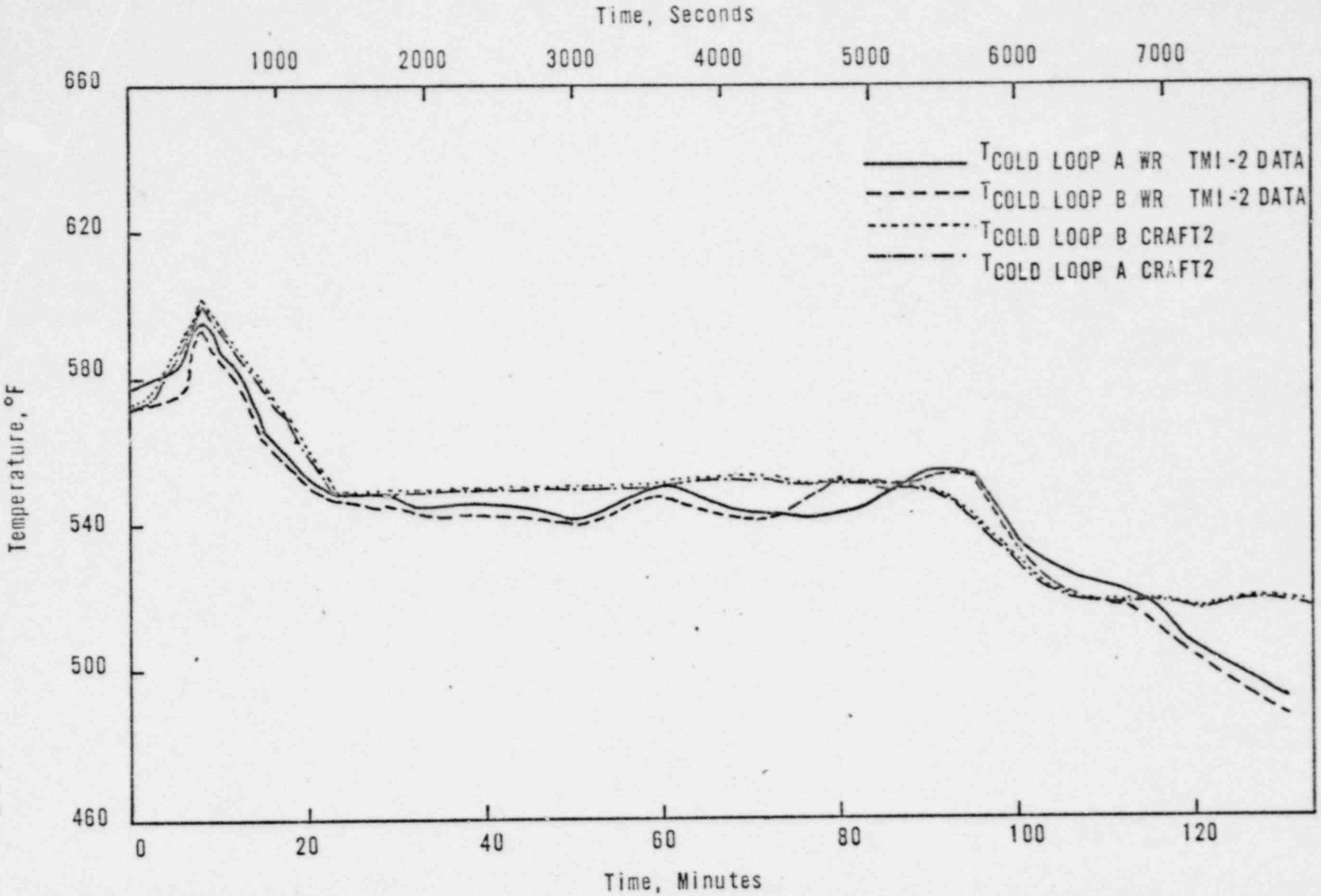
FIGURE 7 COLD LEG TEMPERATURE



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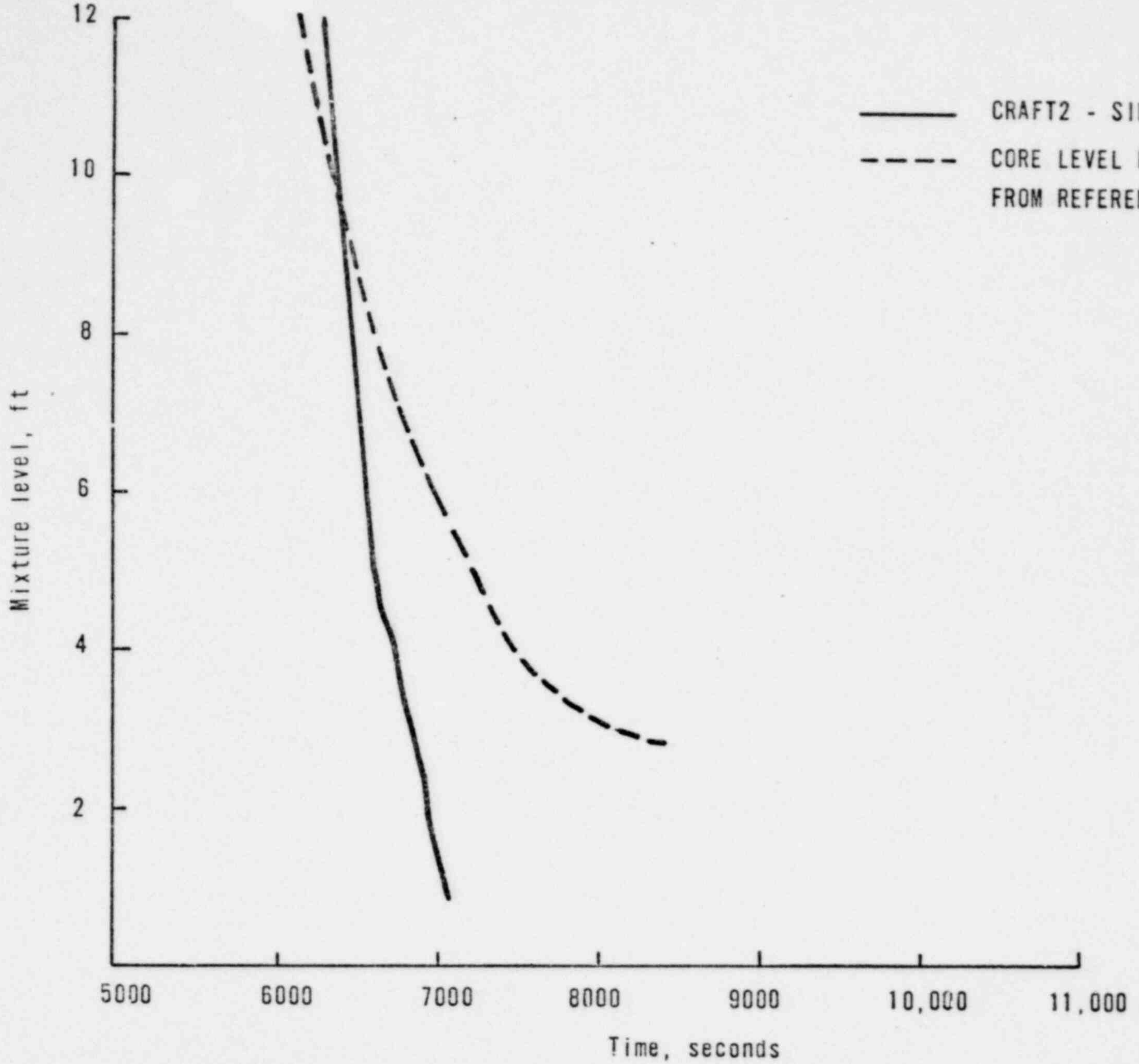

FIGURE 8 CORE MIXTURE LEVEL

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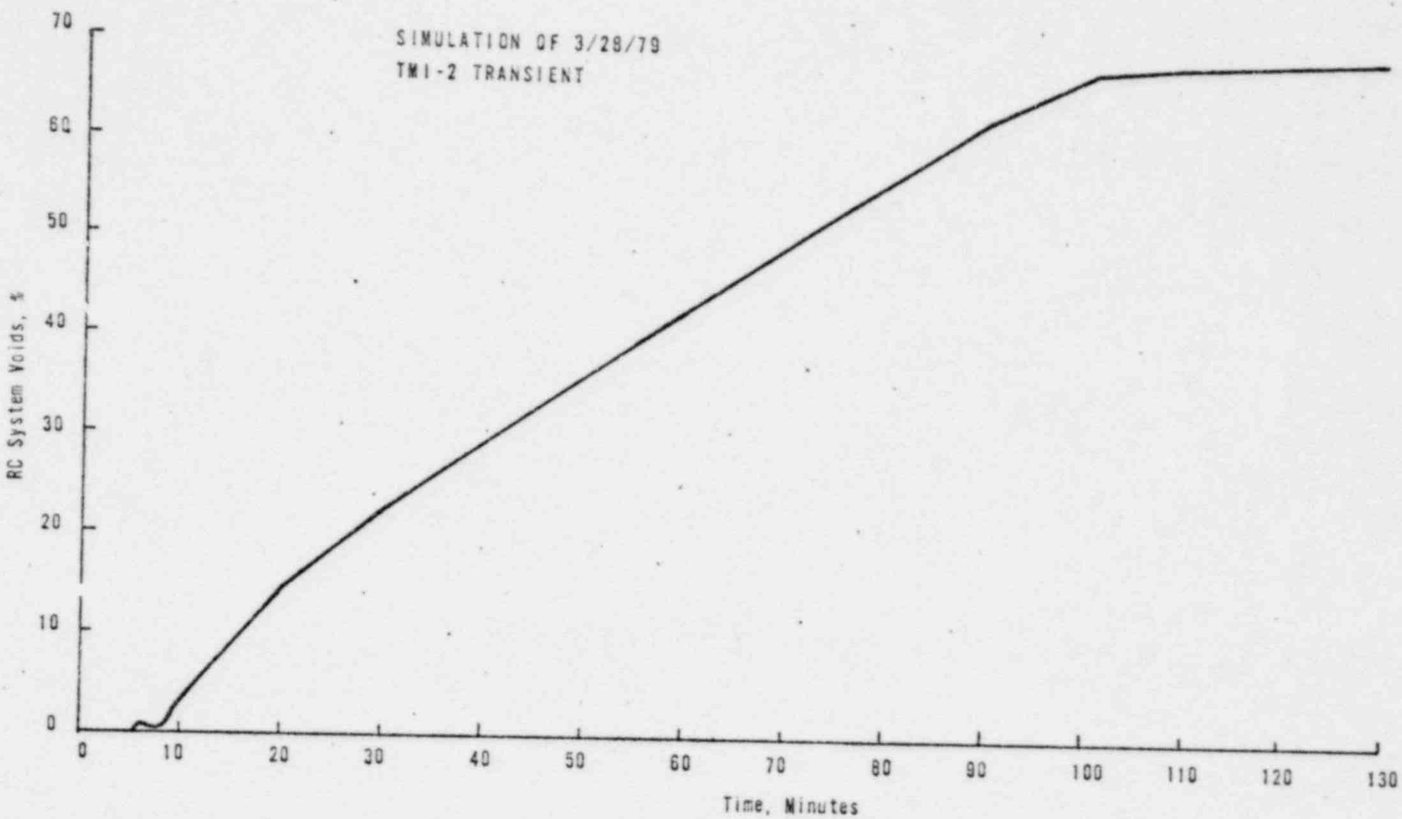
FIGURE 9 SYSTEM VOIDS VS. TIME USING BEST ESTIMATE HPI FLOWS



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