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Attachment 1

D·R·A·F·T

AUXILIARY FEEDWATER SYSTEM
RELIABILITY ANALYSIS
FOR THE
DAVIS-BESSE NUCLEAR GENERATING STATION
UNIT NO. 1

By

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EXECUTIVE SUMMARY

The NRC has requested all operating plants with Babcock & Wilcox (B&W) designed reactors to consider means for upgrading the reliability of their Auxiliary Feedwater Systems (AFWS). As a part of the response to this request, Toledo Edison and the other B&W Owners Group utilities have requested B&W to perform a simplified reliability analysis of existing auxiliary feedwater systems. This draft report presents the results of that reliability study for the Davis-Besse AFWS.

The primary objective of this study was to evaluate Davis-Besse AFWS reliability (defined as "point unavailability") using an approach which would produce results comparable to those obtained by NRC staff analyses for Westinghouse and Combustion Engineering Plants. Another objective was to identify dominant failure contributors affecting system reliability.

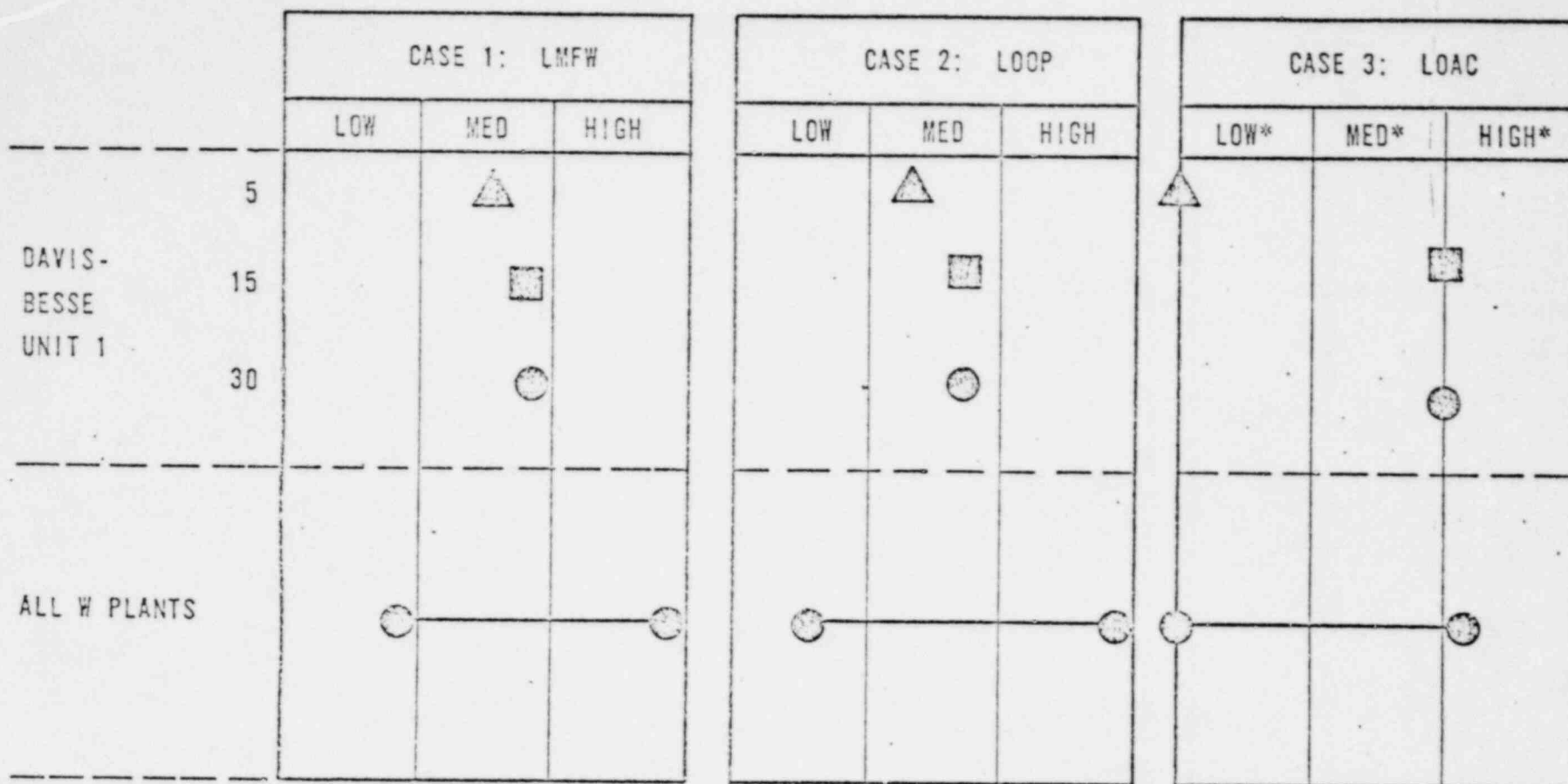
AFWS reliability was assessed for three cases: Loss of Main Feedwater (LMFW) with reactor trip, LMFW with Loss of Offsite Power (LMFW/LOOP) and LMFW with Loss of all AC power (LMFW/LOAC). System reliability was assessed by the construction and analysis of fault trees.

The results of this study are on the following page. These results indicate the Davis-Besse AFWS reliability, based on the reliabilities obtained by the NRC for Westinghouse plants, is medium for LMFW, medium for LMFW/LOOP, and low to high for LMFW/LOAC. For the LMFW/LOAC case, the AFWS is unavailable at 5 minutes because of the AC dependency of the AFWS motor-operated valves; however, for 15 and 30 minutes, the reliability improves because the operator can manually operate these valves.

Dominant failure contributors which were identified in this study include 1) the AC dependency of the motor-operated valves, and 2) system unavailability resulting from outages for preventive maintenance.

A similar study will be performed for each Owners Group utility and additional plant specific draft reports will be prepared. At the conclusion of the program, information contained in the plant specific reports will be collected and used to generate an AFWS reliability report comparing all B&W operating plants.

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▲ MISSION SUCCESS WITHIN 5 MINUTES

■ MISSION SUCCESS WITHIN 15 MINUTES

● MISSION SUCCESS WITHIN 30 MINUTES

● — ● RANGE OF W PLANTS

*THE SCALE FOR CASE 3 IS NOT THE SAME AS FOR CASES 1 & 2.

COMPARISON OF DAVIS-BESSE-1 AFWS RELIABILITY WITH NRC
RESULTS FOR W PLANTS

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1.0 Introduction

1.1 Background

This report presents the results of a reliability study for the Davis-Besse Unit 1 Auxiliary Feedwater System (AFWS). The NRC is conducting similar analyses for Westinghouse and Combustion Engineering plants. Preliminary results of the NRC study are available (Reference 1) and have been included in this report for comparison with the Davis-Besse 1 AFWS reliability. The approach employed in this study has been developed in close coordination with the NRC and is therefore expected to yield comparable results.

1.2 Objectives

The objectives of this study are:

- To perform a simplified analysis to assess the relative reliability of the Davis-Besse 1 AFWS. It is intended that the results of this analysis be directly comparable to those obtained by the NRC for Westinghouse and Combustion Engineering plants. This is assured by the use of the same evaluative technique, event scenarios, assumptions and reliability data used by the NRC.
- To identify, through the development of reliability-based insight, dominant failure contributors to AFWS unreliability.

1.3 Scope

The Davis-Besse 1 AFWS was analyzed as it existed on August 1, 1979. Three event scenarios were analyzed:

- Case 1 - Loss of Main Feedwater with Reactor Trip (LMFW).
- Case 2 - LMFW coincident with Loss of Offsite Power (LMFW/LOOP).
- Case 3 - LMFW coincident with Loss of all AC Power (LMFW/LOAC).

These event scenarios were taken as given; that is, postulated causes for these scenarios and the associated probabilities of their occurrences were not considered. Additionally, external common mode events (earthquakes, fires, etc.) and their effects were excluded from consideration.

For each of the three cases, system reliability as a function of time was evaluated.

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1.4 Analysis Technique

The evaluation of reliability for the AFWS was based primarily on the construction and analysis of fault trees. This technique encourages the development of insights which permit identification of the primary contributors to system unreliability. Application of this technique is described in detail in Section 3.1.

1.5 Assumptions and Criteria

Assumptions and criteria were made in consultation with the NRC staff and were selected to assure that the Davis-Besse 1 reliability evaluation results will be comparable to those obtained by the NRC for the Westinghouse and Combustion Engineering analyses.

- 1) Criterion for Mission Success - In order to evaluate the overall reliability contribution of system components, it is necessary to establish whether or not failure of those components will prevent successful accomplishment of the AFWS mission. Thus, it is necessary to explicitly define the criterion for mission success. The criterion adopted for this study was the attainment of flow from at least one pump to at least one steam generator. Mission success can be alternatively defined as at least one running pump with suction to a source of water and an open flow path to at least one generator without flow diversion.

System reliability was calculated at times of 5, 15, and 30 minutes to allow for a range of operator action. These times were specifically chosen because NRC-supplied operator reliability data for these times was available; however, these times are reasonable and consistent with LMFWR mitigation for B&W plants. In their study, the NRC staff has used steam generator dryout time as a criterion for successful AFWS initiation, and the 5 minute case represents a comparable result for B&W plants since auxiliary feedwater delivery within 5 minutes will prevent steam generator dryout. However, steam generator dryout itself does not imply serious consequences; a more appropriate criteria is the maintenance of adequate core cooling. Recent analyses (Reference 2) have shown that ^{with no RCS break} adequate core cooling can be maintained

without AFWS operation, providing that one makeup pump and the startup feed pump is operated and the PORV is opened within 30 minutes of loss of main feedwater.

- 2) Power Availability - The following assumptions were made regarding power availability:

LMFW - All AC and DC power was assumed available with a probability of 1.0.

LMFW/LOOP - One diesel generator was unavailable with a probability of 10^{-2} . The other generator was assumed available with a probability of 1.0.

LMFW/LOAC - DC and battery-backed AC were assumed available with a probability of 1.0.

- 3) NRC-Supplied Data - NRC-supplied unreliability data for hardware, operator actions and preventive maintenance were assumed valid and directly applicable. These data are listed in Appendix B.
- 4) Small Lines Ignored - Lines on the order of 1-inch were ignored as possible flow diversion paths.
- 5) Coupled Manual Actions - Manual initiation of valves with identical function was considered coupled. Such valves were assumed to be both opened manually or both not opened. The case in which one valve was opened and the other valve was left closed was not considered.
- 6) Degraded Failures - Degraded failures were not considered; that is, components were assumed to operate properly or were treated as failed.
- 7) Backup Water Sources - No credit or penalties were taken for the suction header connections to the startup feed pump, the fire water system or the deaerator storage tank.
- 8) Relief Valves - It was assumed that 2 relief valves stuck open would defeat the steam supply to the affected turbine until automatic or manual isolation of the associated steam generator is accomplished.

2.0 System Description

2.1 Overall Configuration

A diagram of the Davis-Besse 1 AFWS is presented in Figure 1. This safety grade system consists of two interconnected trains, capable of supplying auxiliary feedwater to either or both steam generators under automatic or manual initiation and control.

2.1.1 Suction

The primary water source for both AFWS trains consists of two interconnected condensate storage tanks. Each of the tanks has a capacity of 250,000 gallons; a combined reserve of 250,000 gallons is required by Technical Specifications for AFWS use. The water level in both tanks is indicated in the control room.

A single suction header conducts water from the tanks to the normally-open AC motor-operated pump suction valves, HV786 and HV790. This header contains several additional normally closed connections which were not addressed in this study.

An alternate water source for AFWS use is available from the service water system; details of this source are provided in Section 2.2.

2.1.2 Pumps and Discharge Cross-connections

Each train has a turbine-driven pump rated at 1050 gpm with a design recirculation flow of 250 gpm. Thus, each pump is capable of supplying 800 gpm against a steam generator pressure of 1050 psig (safety relief valve set pressure).

The pumps are protected against cavitation by pump suction pressure switches; the switches for each pump are interlocked with both the primary and cross-connect steam supply valves for the corresponding turbine (e.g., the pressure switch for AFWS pump 1 is interlocked with valves HV106 and HV106A). In the event that low suction pressure should occur for a pump, its steam supply valves will close, stopping steam flow to the pump turbine.

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Steam generator level is controlled by varying turbine speed. Turbine speed is controlled in part by the turbine governor valves ICS38A and ICS38B; operation of these valves is further described in Section 2.4.2. Auxiliary feedwater flow is also affected by the operation of speed-controlled valves HV360 and HV388. As turbine speed increases above 2800 RPM, these valves open; as turbine speed decreases below 1100 RPM, the valves close. Both speed-controlled valves are bypassed by restricting orifices which are capable of passing a flow of 200 gpm (nominal) when the valves are closed. For the purposes of this analysis, it was assumed that each speed-controlled valve must be open to obtain successful auxiliary feedwater delivery via the associated pump.

The primary discharge paths for the train 1 and train 2 pumps are through normally-closed AC motor-operated valves HV3870 (train 1) and HV3872 (train 2). An alternate cross-connection path, which permits each pump to feed the opposite steam generator, is available via normally closed AC motor-operated cross-connect valves HV3859 and HV3871. Automatic selection of the auxiliary feedwater flow path and opening of the associated valves is under the control of the safety grade Steam-Feedwater Rupture Control System (SFRCS) and is described in more detail in Section 2.4.1. The flow of auxiliary feedwater to each steam generator from either path must also pass through normally-open AC motor-operated steam generator isolation valves MV608 and MV599. These valves are also under control of the SFRCS.

2.1.3 Steam Supply for the AFWS Turbines

Steam supply for each turbine is available by opening a normally closed AC motor-operated valve (HV106 or HV107) in the primary steam path from the steam generator normally fed by the turbine pump. A cross-connect steam source is available from the opposite steam generator by opening another normally-closed AC motor-operated valve (HV106A and HV107A). Automatic operation of these steam supply valves is under control of the SFRCS, described elsewhere.

Other valves in the steam admission path include the turbine overspeed stop valve which trips closed on turbine overspeed (and which must be reset locally) and the turbine governor valve. The turbine governor valve controls turbine speed to control steam generator water level. Control for these valves is described in Section 2.4.2.

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Exhaust from the turbines passes through check valves into a common exhaust line and is vented to the atmosphere via an exhaust silencer.

2.1.4 Other System Features

Normal recirculation for each pump is provided by a 2-inch line containing locked-open valves, a check valve and a restricting orifice. This line returns "mini-flow" recirculation, 250 gpm, to a condensate storage tank crosstie or drain. Of more significance to reliability is the full flow test recirculation line containing locked closed valves AF21 and AF22, and AF23. If these valves are inadvertently left open in certain combinations, it is assumed that full pump flow could be diverted to the condensate storage tank or drain.

2.1.6 Valve Operation and Indication

With the exception of the turbine governor valves (hydraulic valves which make use of DC motor-operated speed changes), all other motor-operated valves in the AFWS are powered by 480 VAC. In the absence of AC power, these valves will remain "as is". All such valves are position indicated in the control room. The power for control and position indication for these valves is derived from the power for the valve motor operators.

Control switches for the AC motor-operated valves are provided in the control room. These switches can be used for manual operation of the AFWS valves whenever the SFRCS is not controlling those valves; however, these switches will not manually override an SFRCS control signal to the SFRCS controlled valves (HV106, HV106A, HV107, HV107A, HV3870, HV3869, HV3871, HV3872, MV608, MV599). Similarly, pressure switch signals to the interlocked valves cannot be remotely overridden (valves HV106, HV106A, HV107, HV107A, HV786, HV1382, HV790, HV1383). Manual control of the speed-controlled valves (HV360 and HV388) and the turbine governor valves (ICS38A and ICS38B) is available at all times.

2.2 Supporting Systems and Backup Water Source

The AFW turbines and turbine-driven pumps are self-contained entities without dependencies on secondary support systems. Circulation of lubricating oil for both the turbine and pump is shaft powered; lube oil cooling is provided by the pumped fluid.

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The only support system of significance is the backup water supply from the service water system. A simplified diagram of the portion of this system relating to the AFWS is shown in Figure 2. Water can be made available to both trains of the AFWS via normally-closed AC motor-operated alternate suction valves HV1382 (train 1) and HV1383 (train 2). This water is provided by three service water pumps which are on diesel generator-backed power. Normally two of these pumps are kept running at all times.

Automatic switchover to this backup water supply is initiated by the detection of low suction pressure by pressure switches immediately upstream of suction valves HV786 (train 1) and HV790 (train 2). These pressure switches will cause HV786 and HV790 to close and alternate suction valves HV1382 and HV1383 to open, thus making service water available to AFW pumps. (These switches will not cause switchover to occur in the event that HV786 or HV790 are inadvertently closed.)

2.3 Power Sources

A simplified diagram showing the AC power distribution for the AFWS components is provided in Figure 3. As shown, AC power for all AFWS components necessary to achieve auxiliary feedwater flow is derived from diesel generator-backed busses. Normally, (Case 1), power for these busses is obtained from the switch yard. However, in the event of LMFW/LOOP (Case 2), the diesel generators are automatically started and AFWS components will remain operable with no operator action required. As shown in the figure, train 1 valves (and the backup water supply for train 1, not shown) are powered by diesel generator 1, and, similarly, train 2 is powered by diesel generator 2. Thus, a failure of one diesel generator to start will not prohibit the initiation of auxiliary feedwater flow.

In the event of LMFW/LOAC (Case 3), automatic initiation of auxiliary feedwater flow will not occur because of the AC dependence of AFWS valves; manual actions, described in Section 2.5, are required.

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2.4 Instrumentation and Control

2.4.1 Initiation Logic

A functional logic diagram illustrating the means of AFWS initiation for train 1 is shown in Figure 4. This logic is a part of the SFRCS and is on battery-backed power. The diagram is greatly simplified and does not show actual hardware redundancies, the availability of manual initiation, and some interlocks.

As indicated in the figure, the train 1 AFW flow can be initiated by a low water level in either steam generator, low steam generator 2 pressure, loss of all four reactor coolant pumps, or high reverse differential pressure across main feedwater check valves. Among other things, this last condition will result from the loss of both main feedwater pumps, in which case reverse MFW ΔP signals will originate from both steam generators.

SFRCS logic is designed to isolate both main and auxiliary feedwater to a bad generator (e.g., a generator with a rupture in the associated steam piping) and automatically align AFW valves to feed the remaining good steam generator. This capability is reflected in the AFWS initiation logic shown in Figure 4. If steam generator 1 pressure is not low, then, upon AFW initiation, valves in the primary feedwater and steam supply paths between steam generator 1 and AFWS turbine/pump 1 are opened. If, however, steam generator 1 has low pressure and steam generator 2 does not, the cross-connect valves between AFW turbine/pump 1 and steam generator 2 are opened and the primary path valves are closed. SFRCS logic is designed so that actuation signals do not lock on; valve alignment is automatically changed to suit existing conditions. The logic shown in the figure is duplicated for train 2.

2.4.2 Control

After initiation, control of steam generator level is accomplished by varying turbine speed via control signals to the turbine governor valves. Signals for control come from level transmitters in each steam generator. Separate control hardware is provided for each train. This equipment is powered from battery-backed sources and is separate and independent from the Integrated Control System. Manual control of the turbine speed may be required depending on the steam generator level desired.

2.4.3 Instrumentation

AFWS instrumentation in the control room, in addition to the valve position indications previously described, includes:

- Level Indication for each condensate storage tank.
- AFW pump low suction pressure alarm for each pump.
- AFW pump discharge pressure for each pump.
- Steam generator outlet pressure.
- AFW flow indication obtained from flow measurement devices located upstream of the AFW steam generator isolation valves.
- Steam generator startup range levels.

All AFWS instrumentation is powered from battery-backed sources.

2.5 Operator Actions

Assuming no component failures have occurred and the system is correctly configured, no operator actions are required to achieve AFWS mission success in Cases 1 and 2. In Case 3, it will be necessary for the operators to manually open valves in the field. These valves are HV106, HV3870, and HV360 for train 1, and/or HV107, HV3872 and HV388 for train 2.

2.6 Testing

The AFW turbines and pumps are tested monthly, using the mini-flow recirculation line, to assure operability. Valve position for active valves are also checked monthly to assure correct positions. Level control equipment and pressure switch interlocks are tested monthly to assure operability.

Every 18 months, the AFWS is tested extensively. This testing includes a full flow test of the pumps recirculating via AF21 or AF22 to the condensate storage tanks, and a demonstration that valves actuate to the correct position and the pumps start on receiving an AFWS actuation signal. Level control and pressure switch interlock equipment is calibrated on an 18-month basis.

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2.7 Technical Specification Limitations

Technical Specifications require that both AFWS trains be operable; however, if one train is determined to be inoperable, 72 hours are allowed for restoring operability of the train. After that, the plant must be taken to hot shutdown within 12 hours.

Technical specifications also require that 250,000 gallons of water be reserved in the condensate storage tanks for AFWS use. Should this source become unavailable, continued operation for up to 7 days is permitted by relying on the service water system. In this case, operability of the service water system must be demonstrated every 12 hours.

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3.0 Reliability Evaluation

3.1 Fault Tree Technique

The Davis-Besse 1 AFWS reliability was evaluated by constructing and analyzing a fault tree. The fault tree developed during this study is contained in Appendix A. The top level event in this tree is failure to achieve mission success; from this point, the tree branches downward to a level of detail corresponding to NRC-supplied data. This level is generally indicated by basic event circles.

For construction of the first tier of the tree (page A-1), the AFWS components in each train were grouped into three categories - Suction, Pump and Discharge. System failure can result from Suction 1 - Suction 2, Pump 1 - Pump 2, or Discharge 1 - Discharge 2 failures or from failures within one train when the other train is out of service for preventive maintenance. The fault tree also accounts for system failures resulting from combination failures such as Pump 1 - Discharge 2 with the appropriate discharge cross-connection inoperable. All combinations considered are indicated by the first tier.

The techniques used in fault-tree construction and the symbols shown in Appendix A are consistent with those used in WASH-1400 (Reference 3). Following completion of the tree, hand calculations were performed to obtain system unavailability for 5, 15 and 30 minutes for each of the three event scenario cases.

3.2 Comparative Reliability Results

The results of the analysis are presented in Figure 5. Indicated in this figure are the system reliability results for each of the three cases and for each time 5, 15 and 30 minutes. The basic format for this figure, including the characterization of Low, Medium, and High reliability, was adopted from information presented by the NRC in Reference 1. Because the NRC-supplied input data were often unverified estimates of component and human reliability, absolute values of calculated system reliability must be de-emphasized; results have significance only when used on a relative basis for purposes of comparison. Accordingly, the intent of Figure 5 is to show the relative reliability standing of the Davis Besse 1 AFWS for each of the three cases and also to compare these results to the

NRC results for Westinghouse plants. The Westinghouse results and numerical values permitting construction of Figure 5 were all obtained from Reference 1. It should be noted that there is a scale change for the Case 3 results; reliability results for Case 3 cannot be cross-compared with Cases 1 and 2.

As shown in Figure 5, relative to Westinghouse, Davis-Besse 1 has medium reliability for Cases 1 and 2; Case 3 has low reliability for success in 5 minutes, but high reliability for success in 15-30 minutes. The underlying causes for these reliability results are described in Section 3.3

Some general observations may be made regarding the results in Figure 5. As the time for operator action increases from 5 to 30 minutes, the probability of mission success improves. Most of the improvement occurs between 5 and 15 minutes, reflecting a significant difference in the NRC-supplied operator reliability data for these times. On the other hand, there was little difference in the operator reliability data between 15 and 30 minutes and this is reflected in the system unavailability results. The small difference in the results for Cases 1 and 2 indicates the effect of using turbine-driven pumps in both trains as well as relatively small effect associated with the improbable loss of one diesel generator. The Case 3 results stem from the AC dependence of all AFWS motor-operated valves.

3.3 Dominant Failure Contributors

3.3.1 Case 1 - LMFW

This reliability evaluation did not reveal any outstanding design deficiencies for either Case 1 or 2 that would individually make a significant contribution to overall system unavailability. This results in part because of the excellent separation between the two trains of the AFWS. Dominant failure contributors, therefore, tend to represent combinations of the more probable random component failures.

The dominant failure contributor for Case 1 is turbine/pump failure in one train coupled with preventive maintenance related outage of the other train. The second most important contributor is turbine/pump failures involving both trains. Such combined failures could involve, for example,

the failures of a valve immediately upstream of one turbine (such as the turbine stop valve) coupled with a failure in the other train (such as a mechanical problem with the other pump).

3.3.2 Case 2 - LMFW/LOOP

The absence of offsite power has no effect on the dominant failure contributors. Loss of one diesel generator reduces overall system availability because the valves in one train can no longer be remotely operated. Nevertheless, system success will still depend largely on the same dominant failure contributors (i.e., component failures) for the other train as in Case 1.

3.3.3 Case 3 - LMFW/LOAC

The dominant failure contributor for Case 3 is the AC dependence of the AFWS motor-operated valves. The system will be unavailable for the 5-minute case. For the 15 and 30 minute cases, there is a very good chance that the operator will open the three valves needed to restore at least one AFWS train to operability. Operation of the system thereafter depends on battery-backed power sources and remaining failure contributors will resemble those of Cases 1 and 2.

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REFERENCES

- 1) "Auxiliary Feedwater Reliability Study", an NRC staff presentation to the ACRS at the ACRS meeting of July 26, 1979, 1717 "H" Street, Room 1046, Washington, D.C.
- 2) Attachment to Toledo Edison Company letter to the NRC
Serial No 517 dated June 15, 1979. Docket No 50-346.
- 3) WASH-1400 (NUREG-75/014), "Reactor Safety Study (Appendix II)," USNRC, October 1975.

Figure 1
DAVIS-BESSE-1 AFWS

POOR ORIGINAL

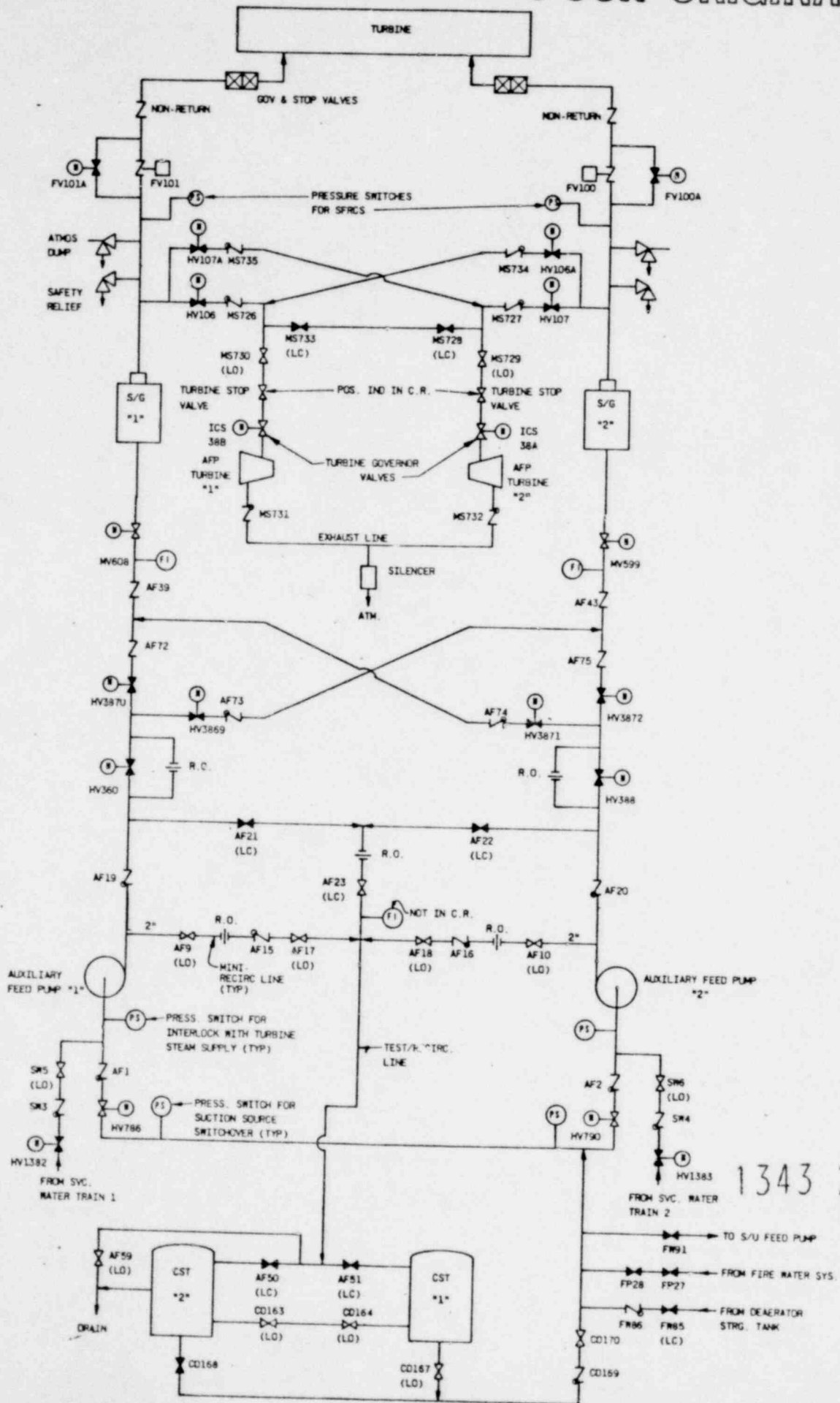
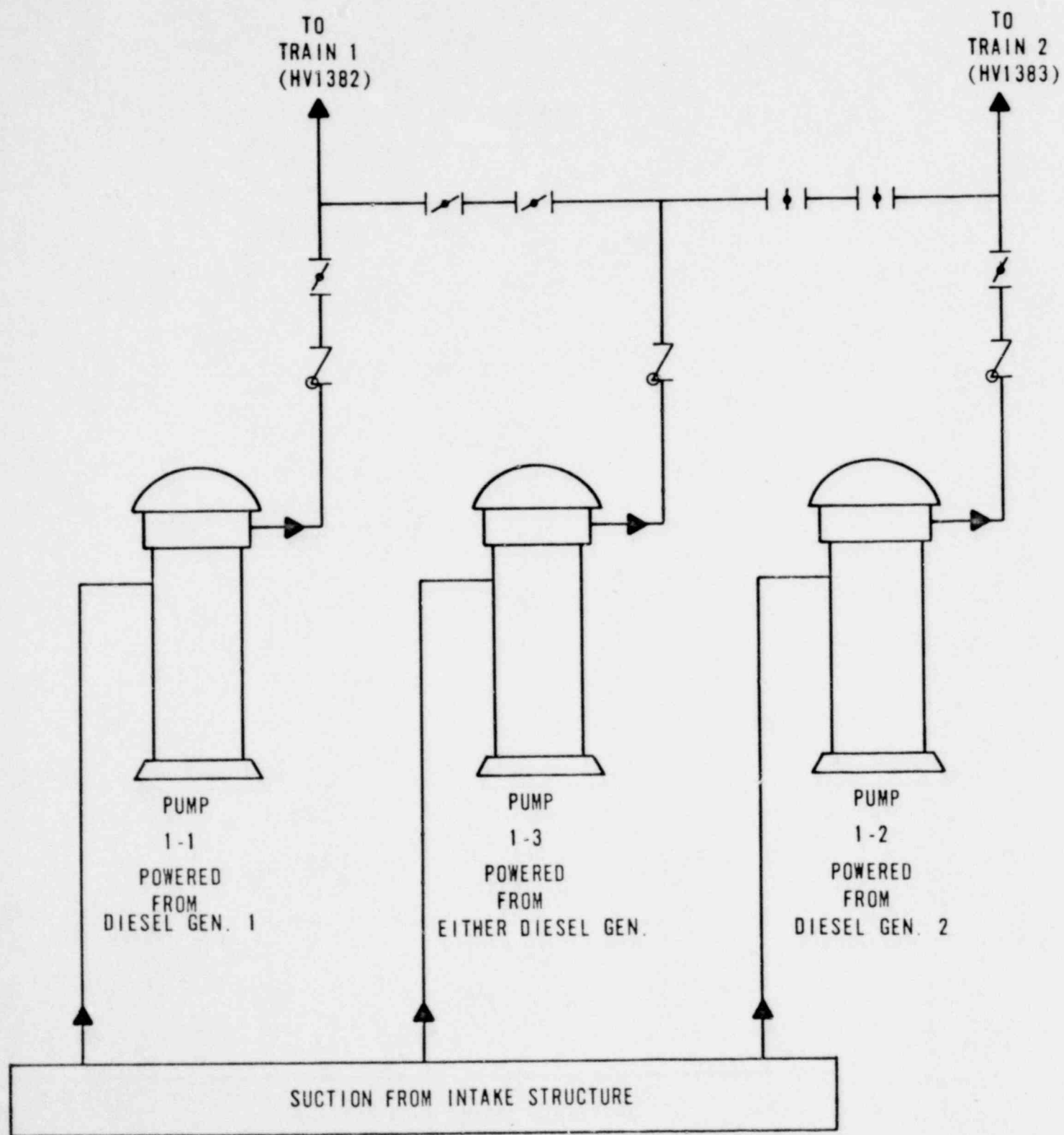
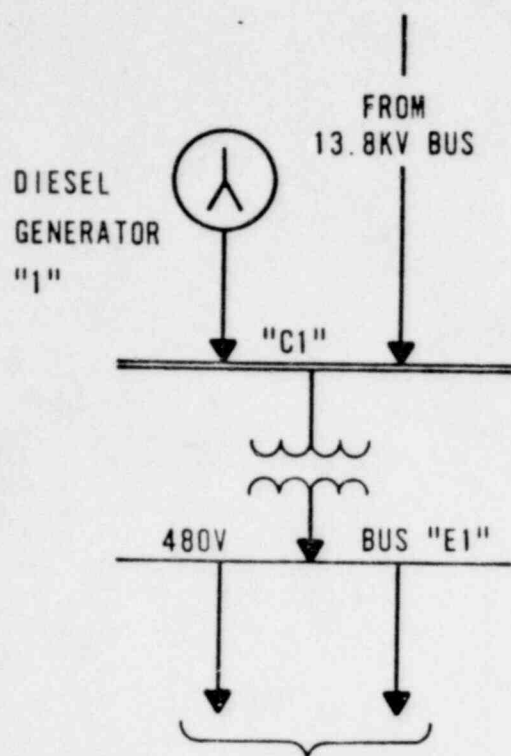


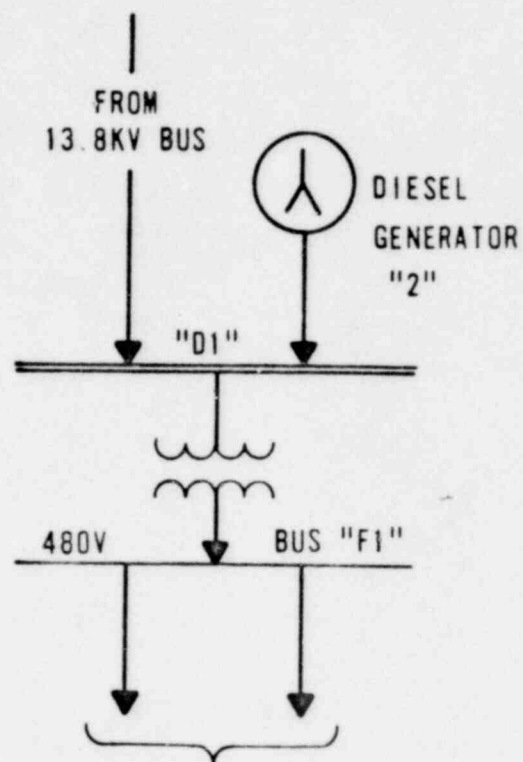
Figure 2 DAVIS-BESSE UNIT 1-ALTERNATE WATER SUPPLY (SERVICE WATER SYSTEM)





TO MCC'S, BATTERY
CHARGERS & INVERTERS
AFWS VALVES:

HV360
MV608
HV1382
HV786
HV3869
HV3870
HV106
HV106A



TO MCC'S, BATTERY
CHARGERS & INVERTERS
AFWS VALVES:

HV388
MV599
HV1383
HV790
HV3871
HV3872
HV107
HV107A

Figure 3 DAVIS-BESSE-1 AC POWER DISTRIBUTION TO AFWS VALVES

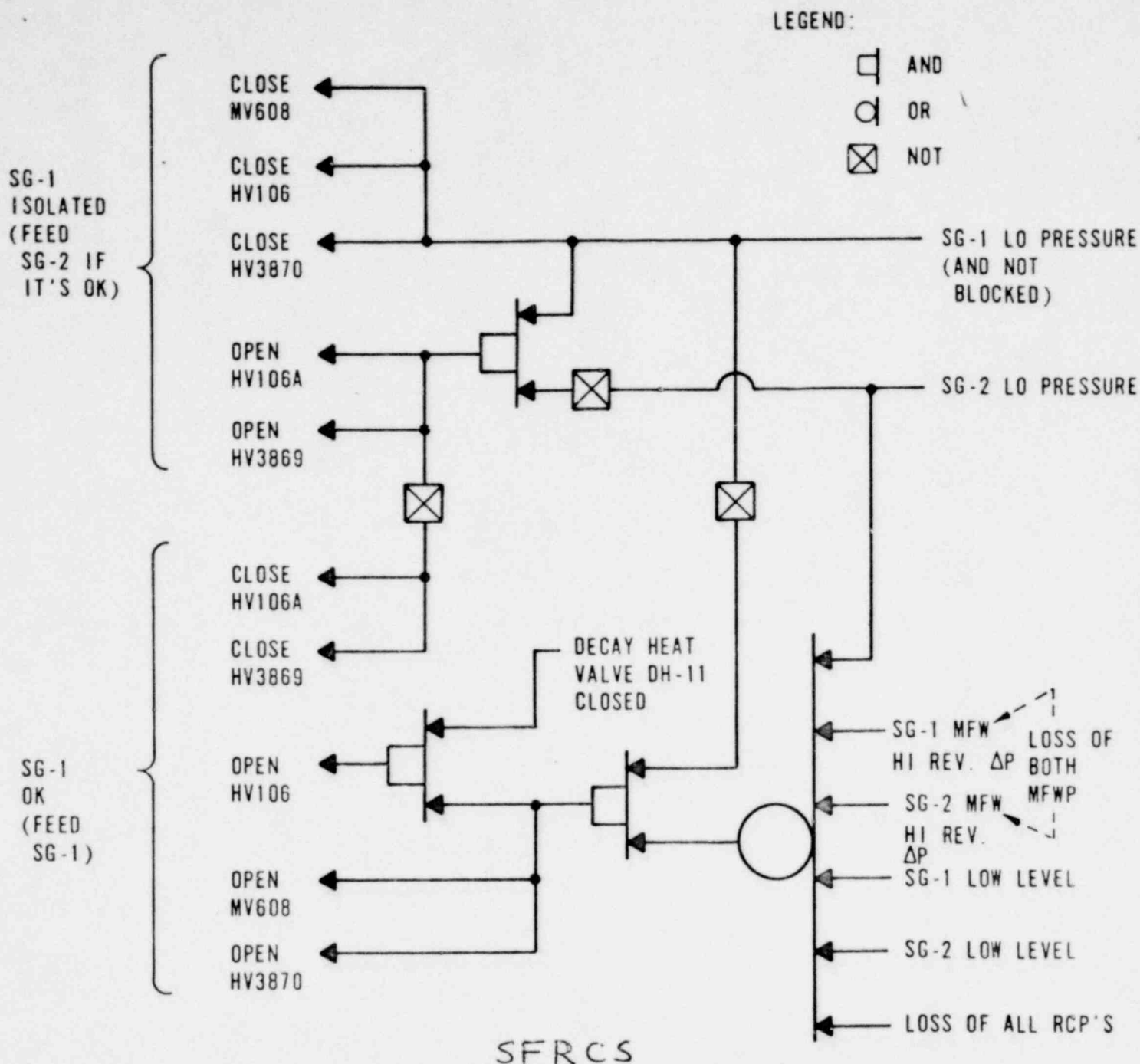


Figure 4 DAVIS-BESSE 1A FUNCTIONAL LOGIC
DIAGRAM FOR TRAIN 1 INITIATION AND S/G ISOLATION FOR AUXILIARY
FEEDWATER

NOTES 1. REDUNDANCIES, MANUAL INITIATION, AND INTERLOCKS NOT SHOWN.

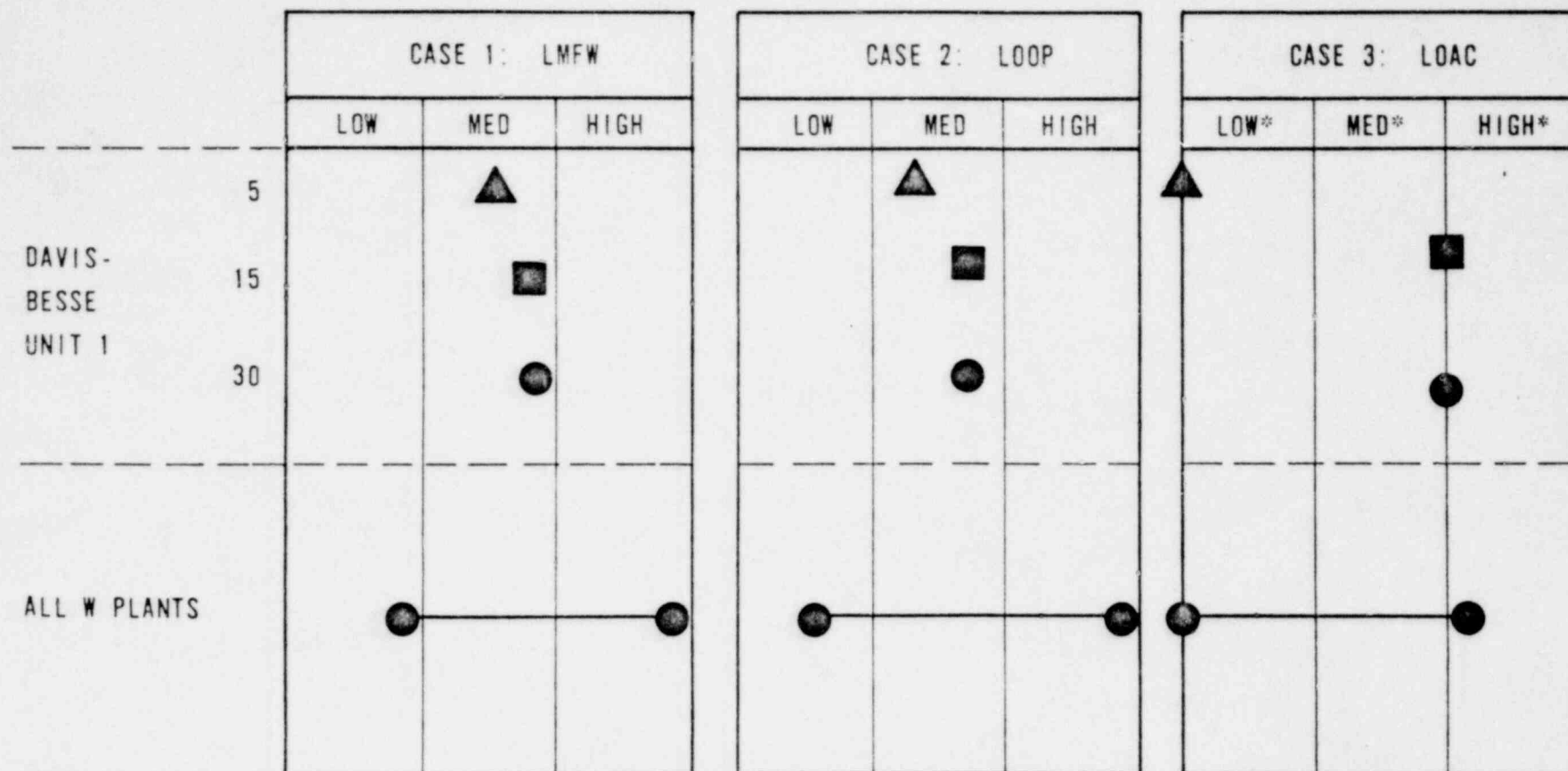
2. LOGIC DUPLICATED FOR TRAIN 2 WITH VALVE SUBSTITUTIONS:

TRAIN 1
HV106
HV106A
HV3870
HV3869
MV608
DH11

TRAIN 2
HV107
HV107A
HV3872
HV3871
MV599
DH12

3. VALVES ARE AVAILABLE FOR MANUAL CONTROL UNLESS THE SFRCS
HAS BEEN ACTUATED.

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▲ MISSION SUCCESS WITHIN 5 MINUTES

■ MISSION SUCCESS WITHIN 15 MINUTES

● MISSION SUCCESS WITHIN 30 MINUTES

●—● RANGE OF W PLANTS

*THE SCALE FOR CASE 3 IS NOT THE SAME AS FOR CASES 1 & 2.

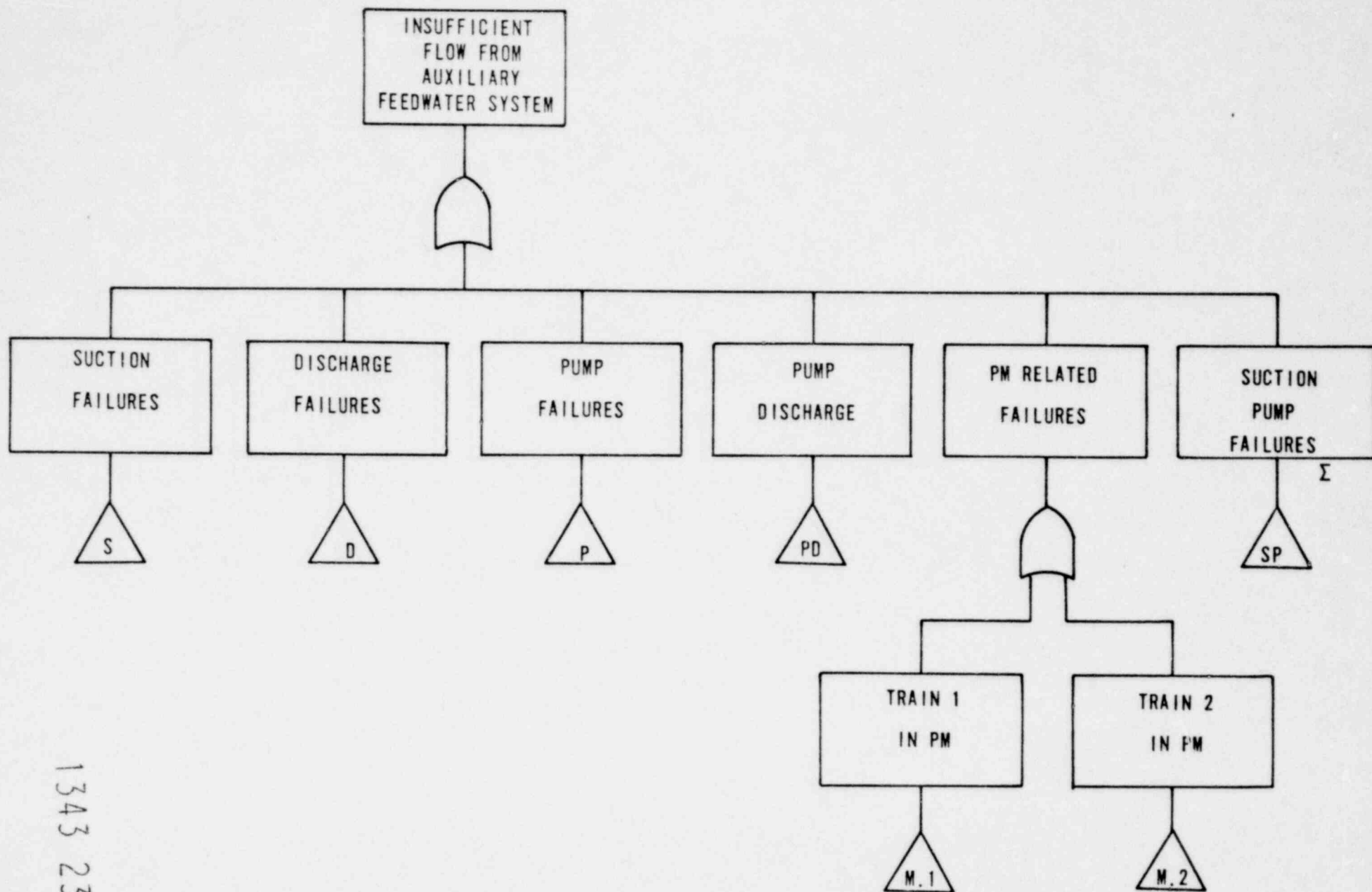
Figure 5 COMPARISON OF DAVIS-BESSE-1 AFWS RELIABILITY WITH NRC RESULTS FOR W PLANTS

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APPENDIX A

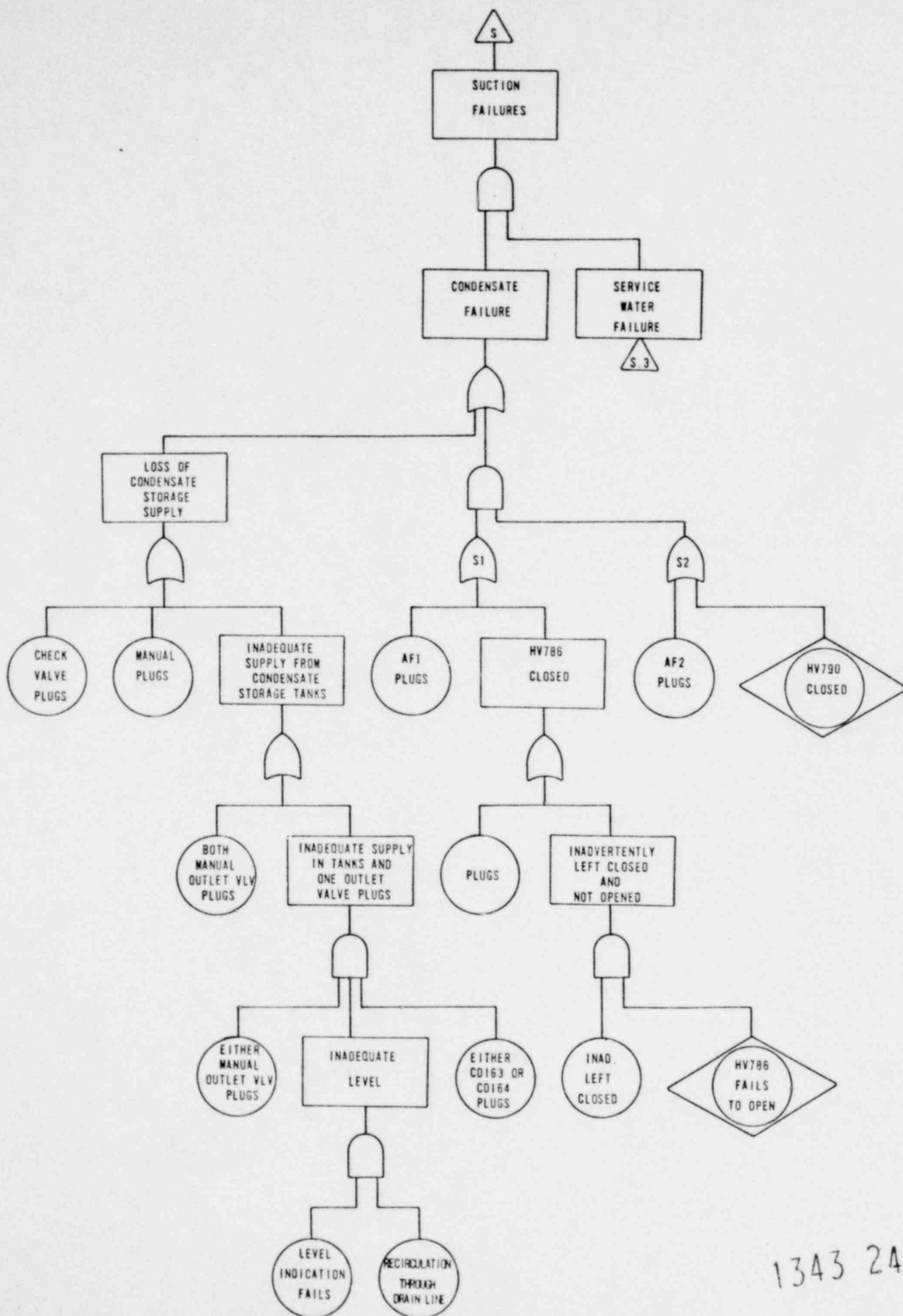
DAVIS-BESSE 1 FAULT TREE

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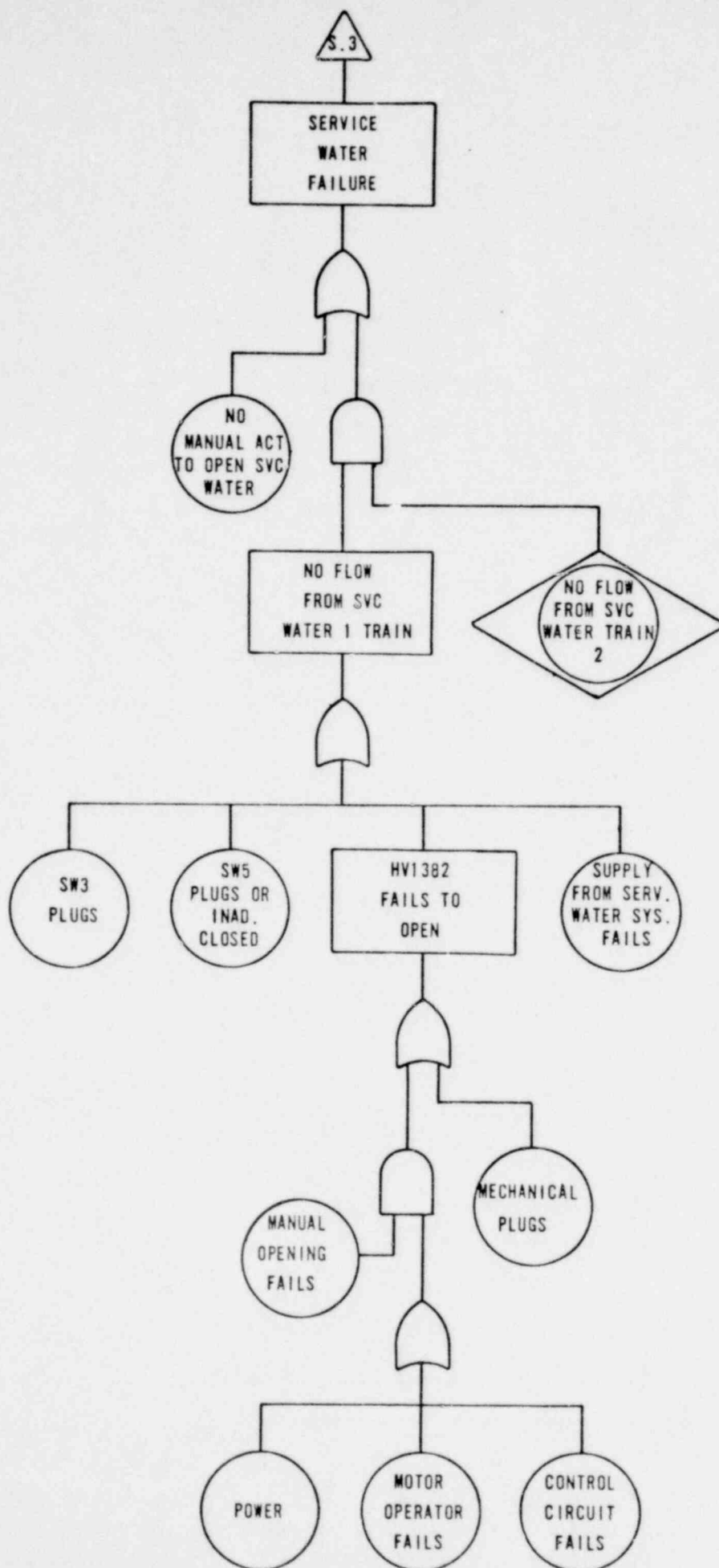


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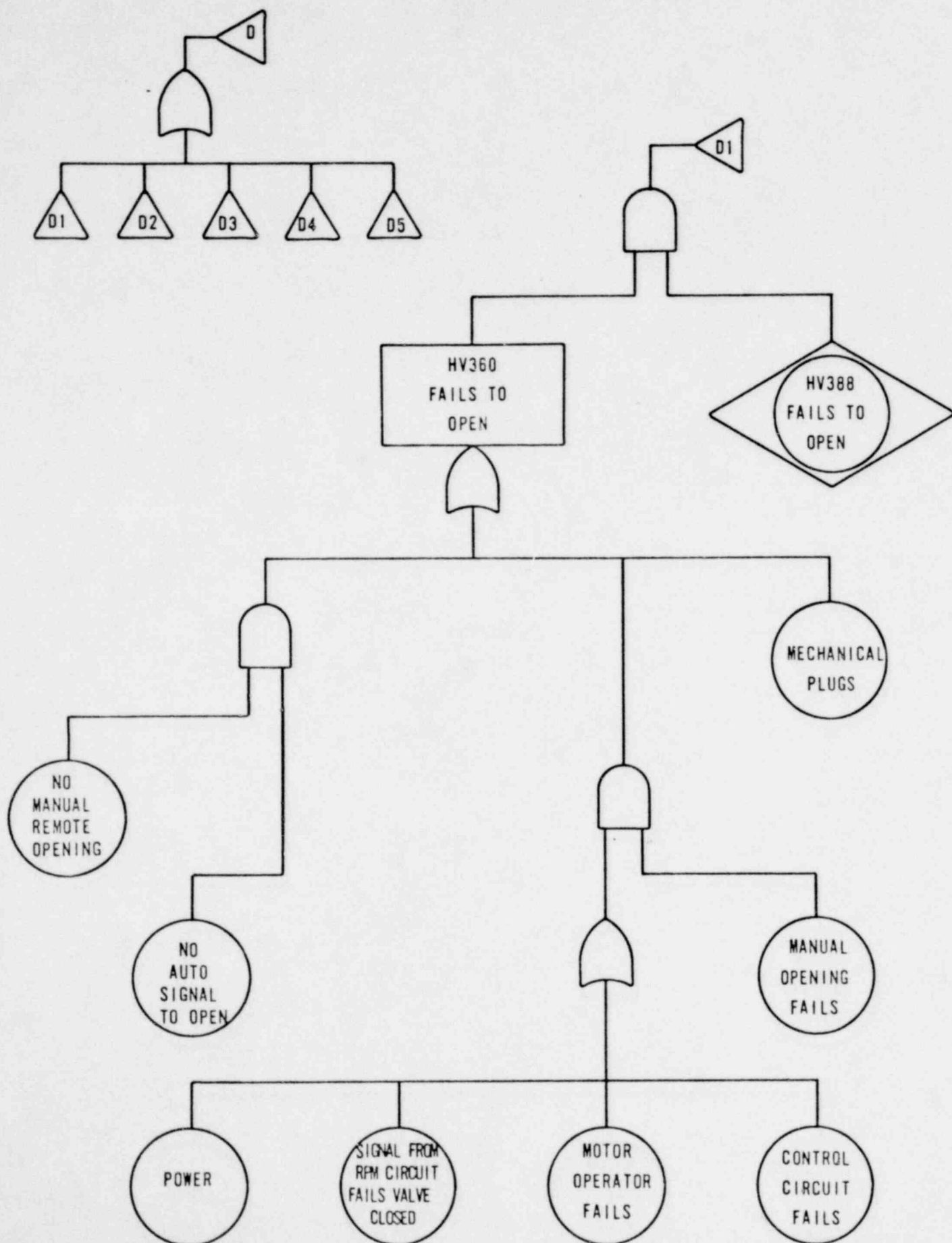
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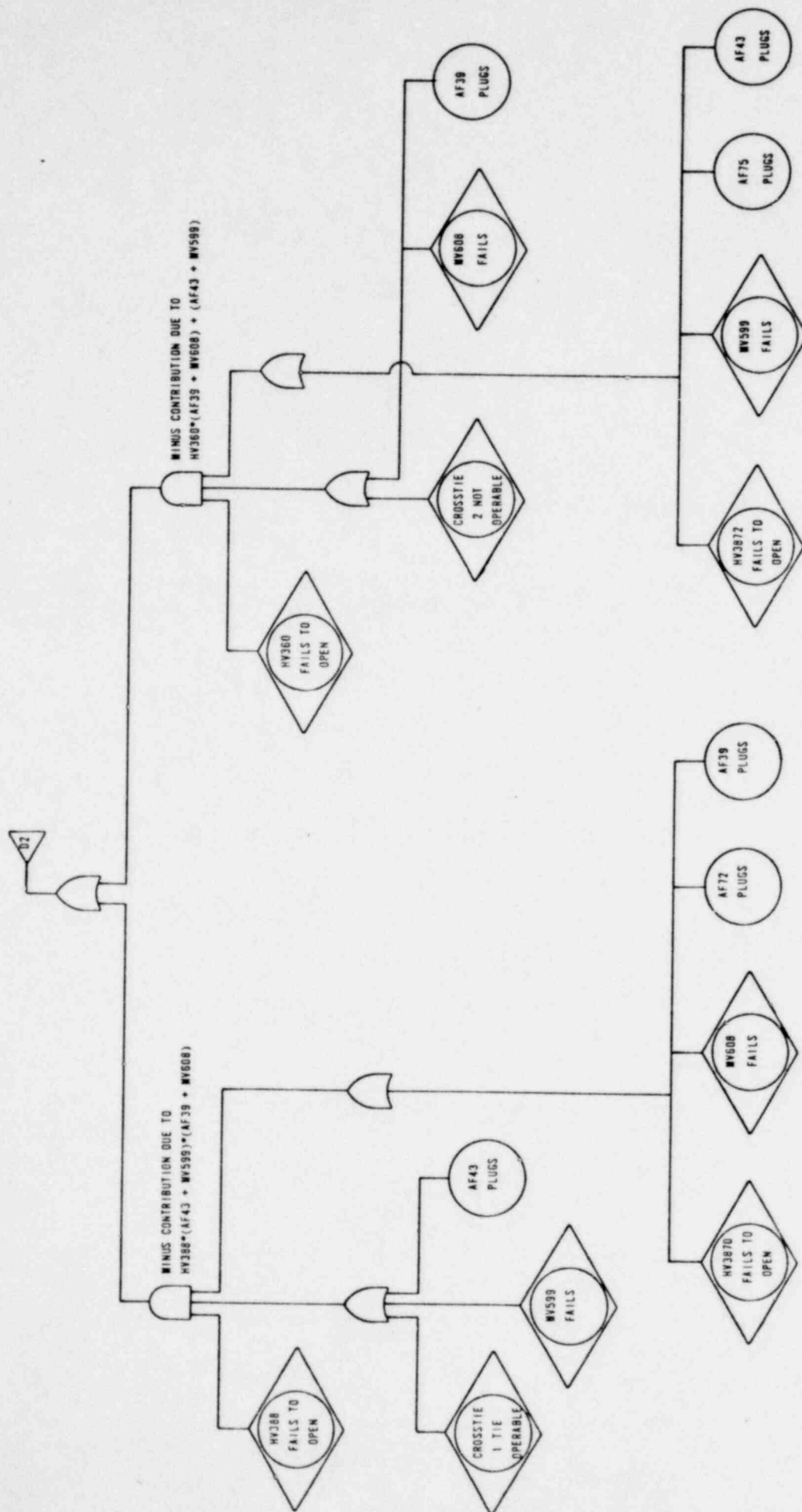
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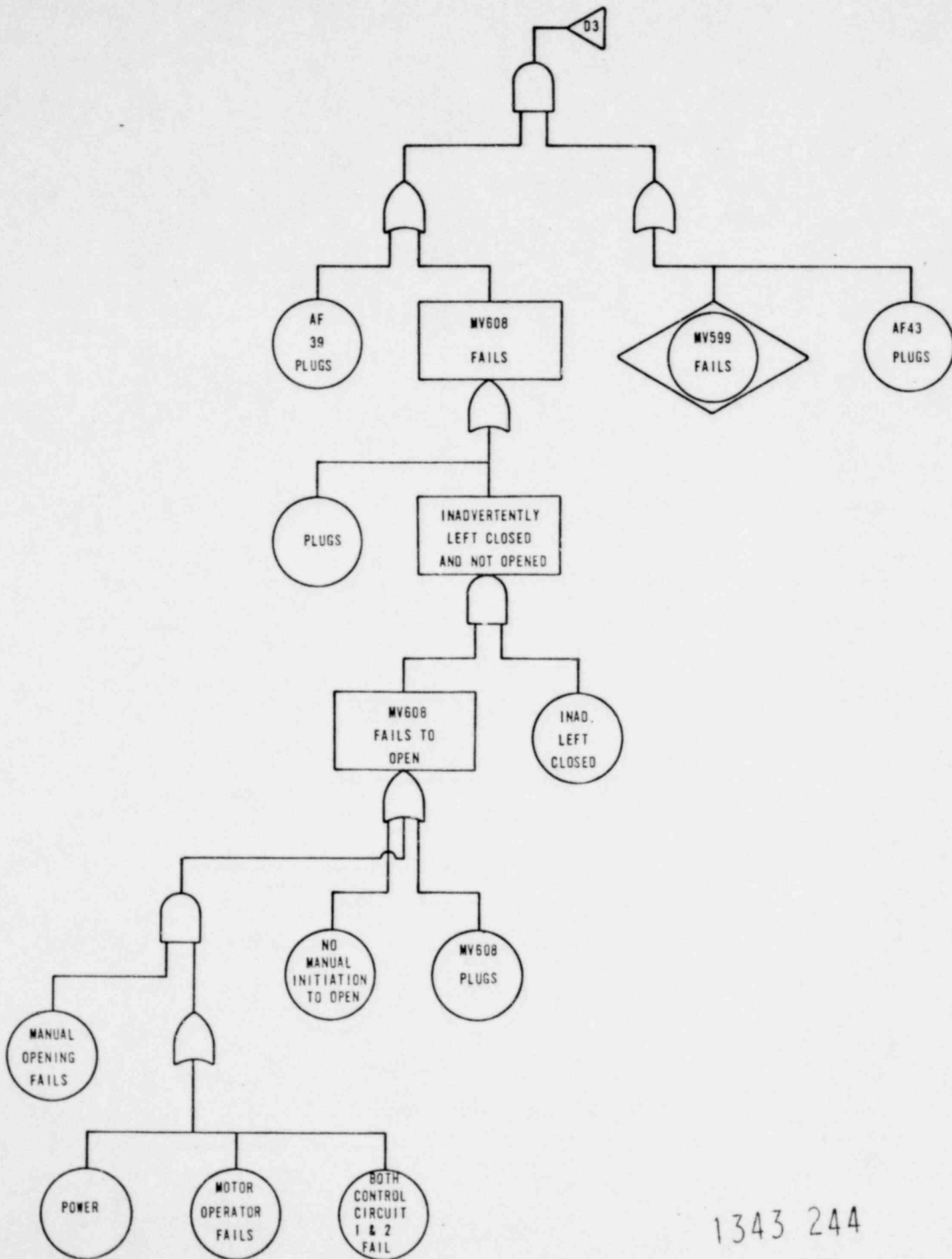
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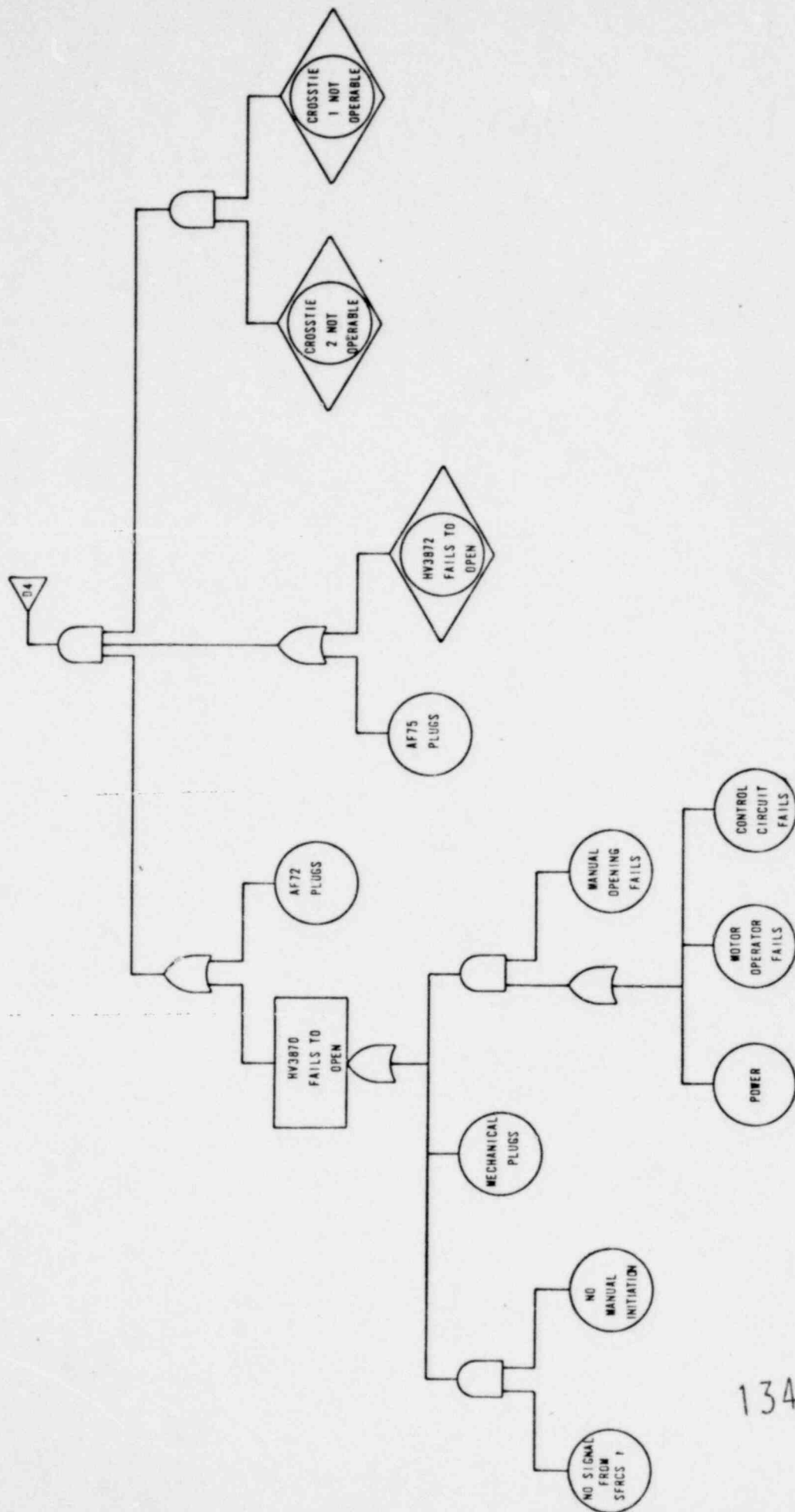
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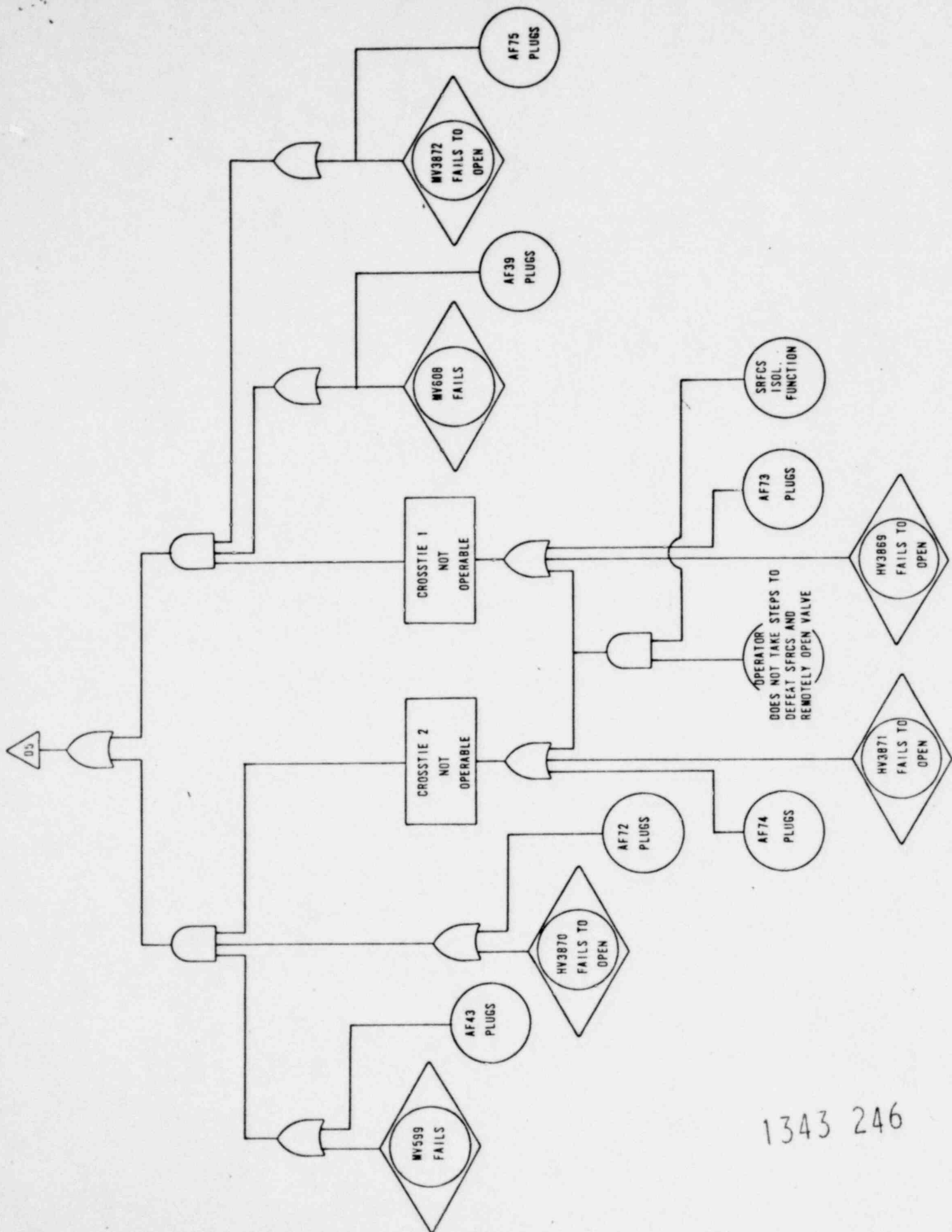
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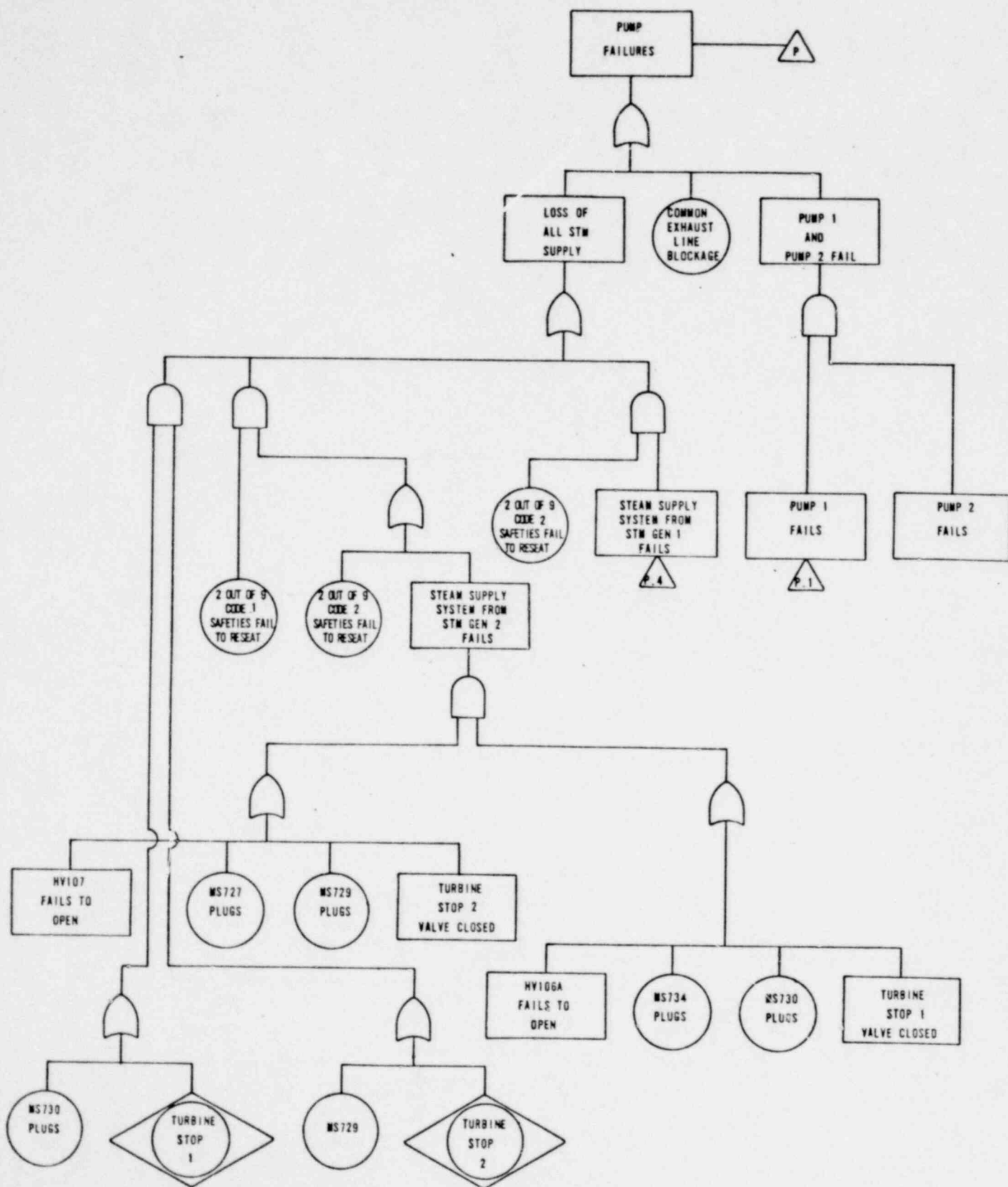
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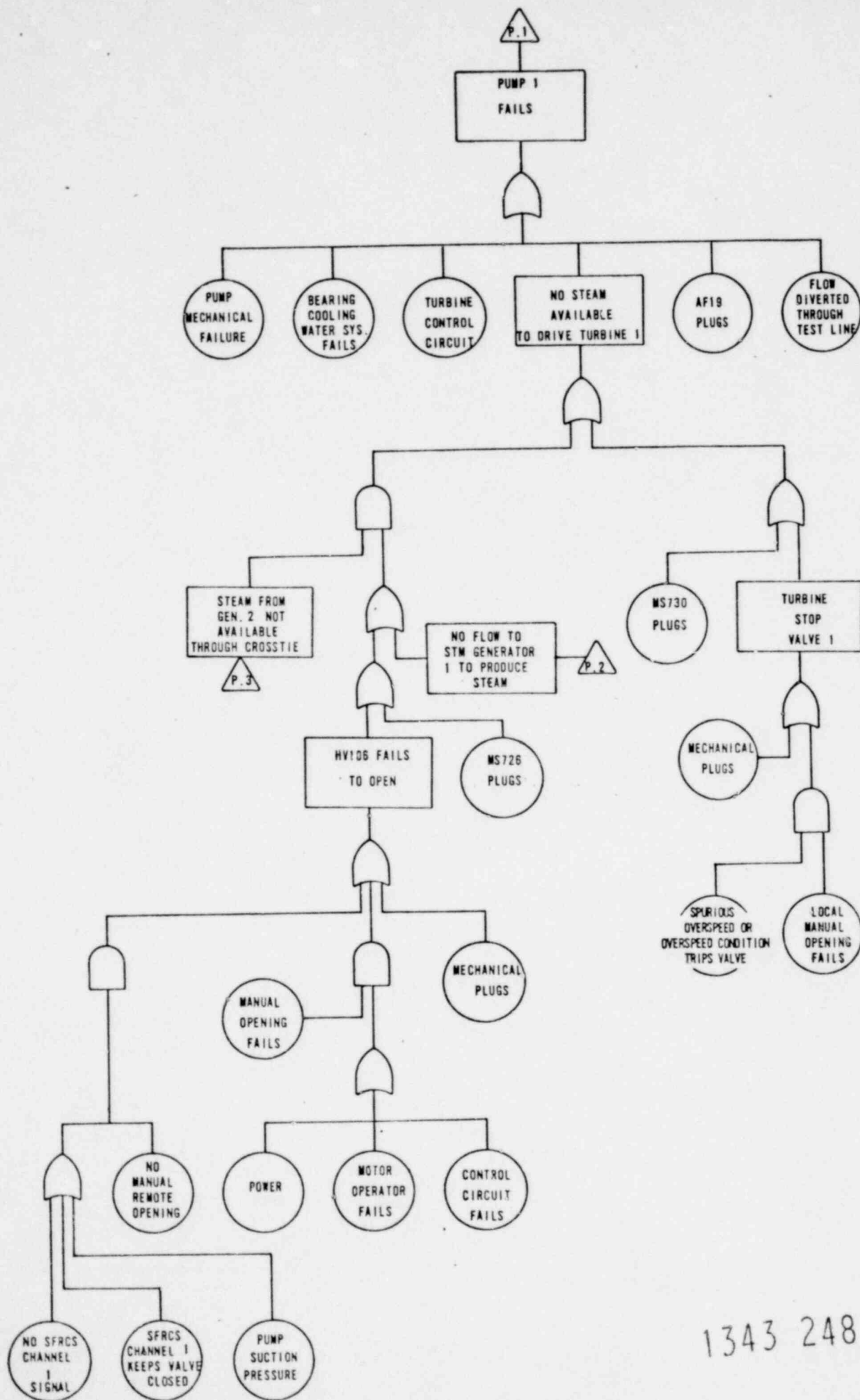
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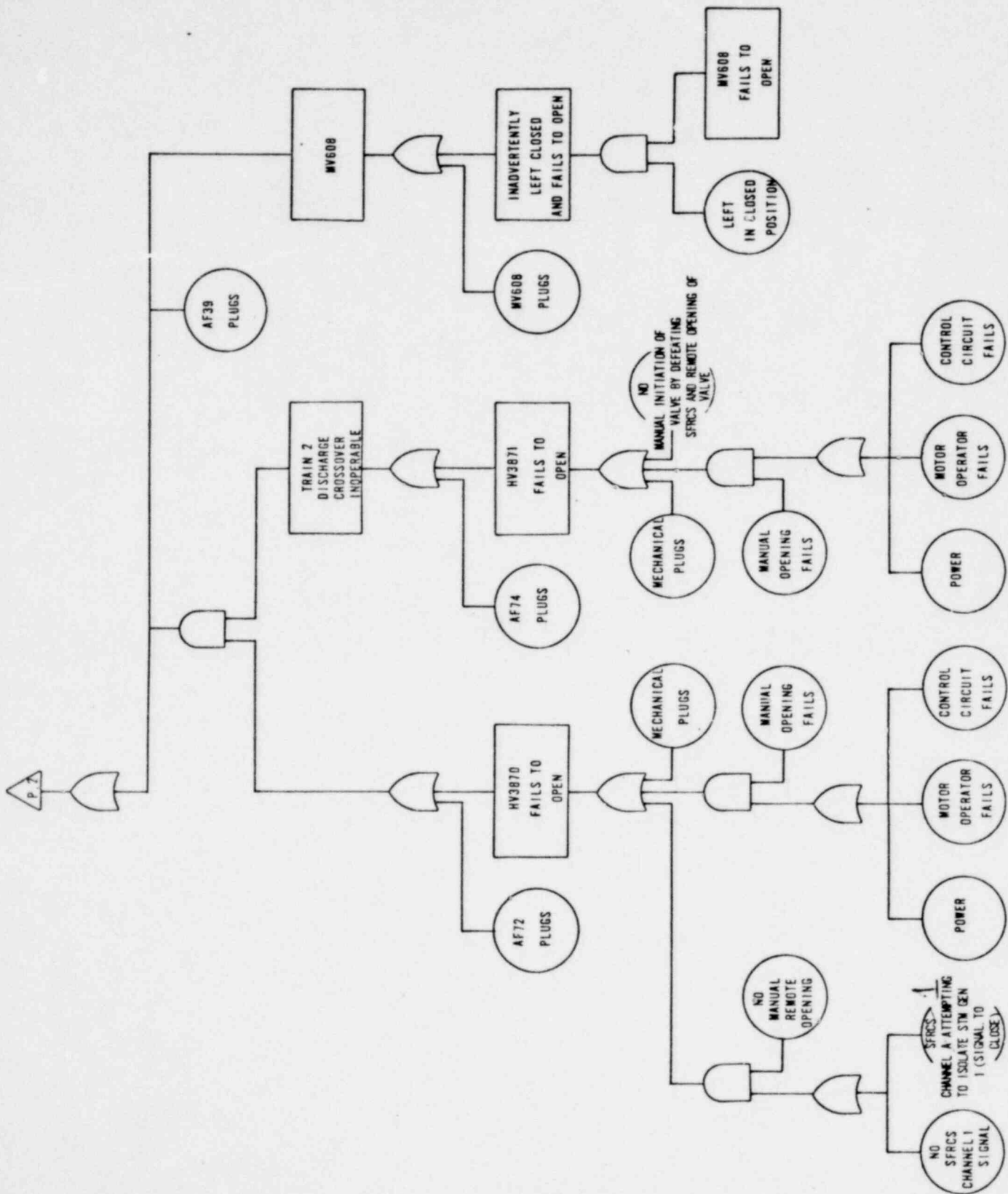
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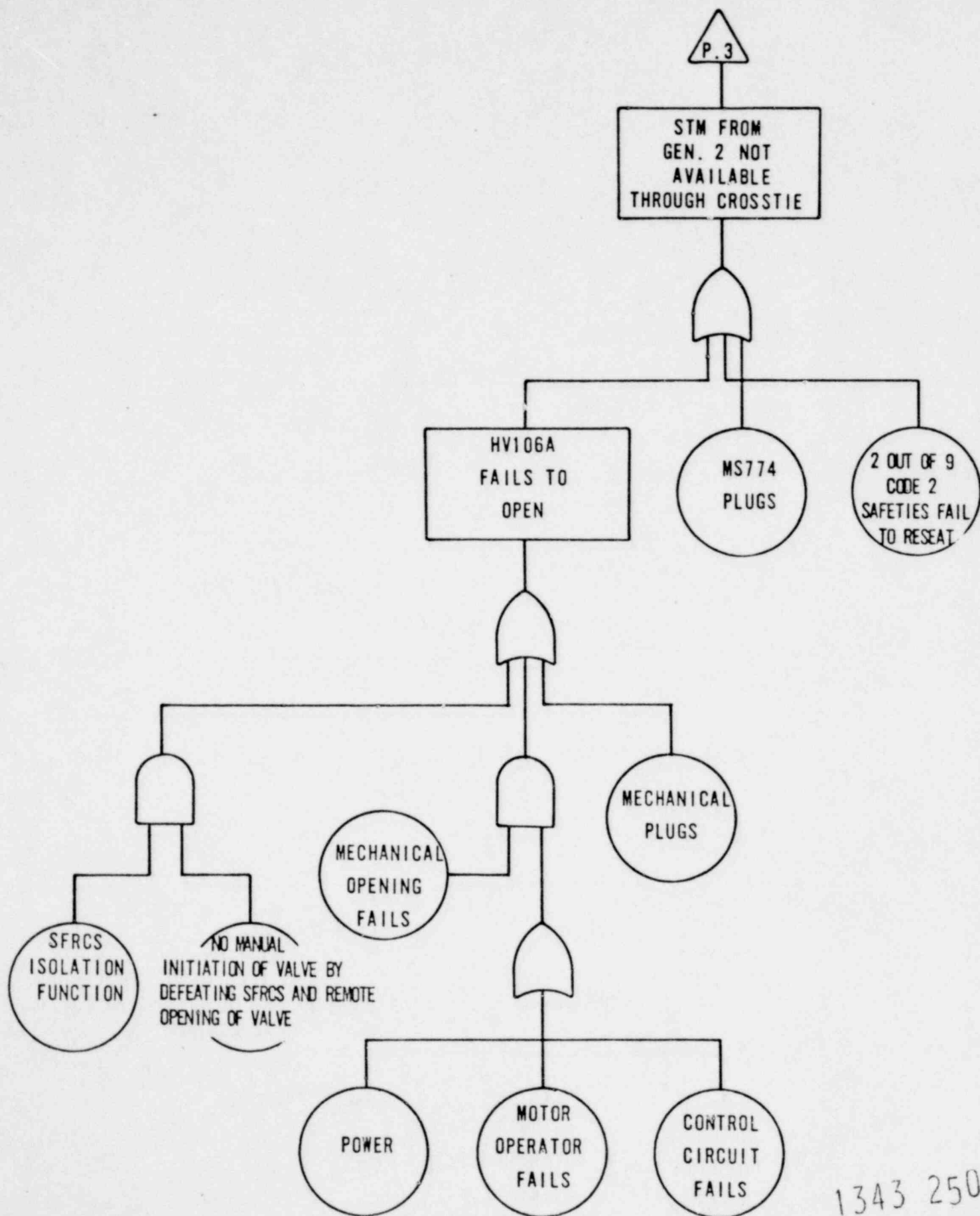
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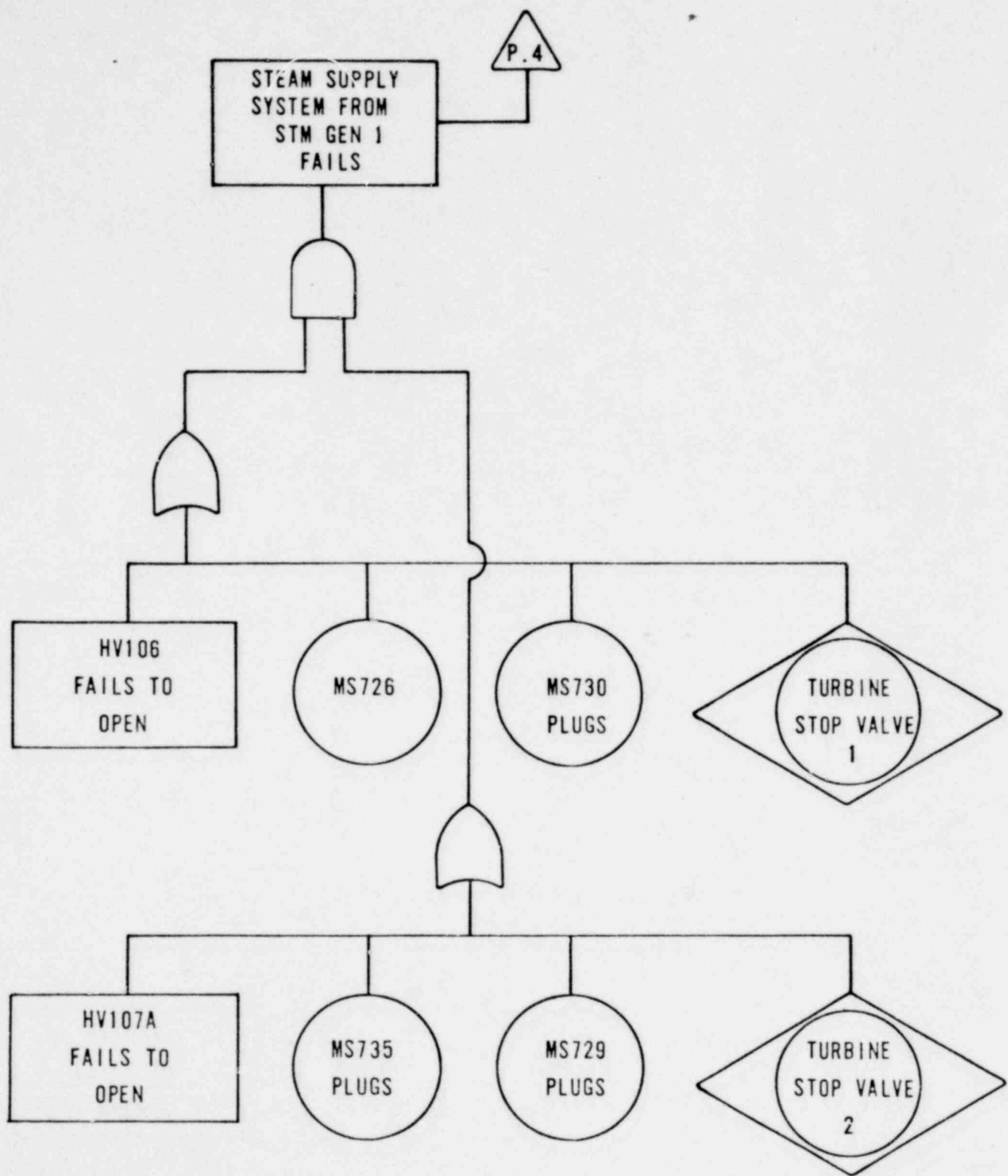
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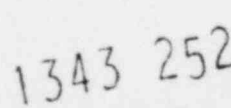
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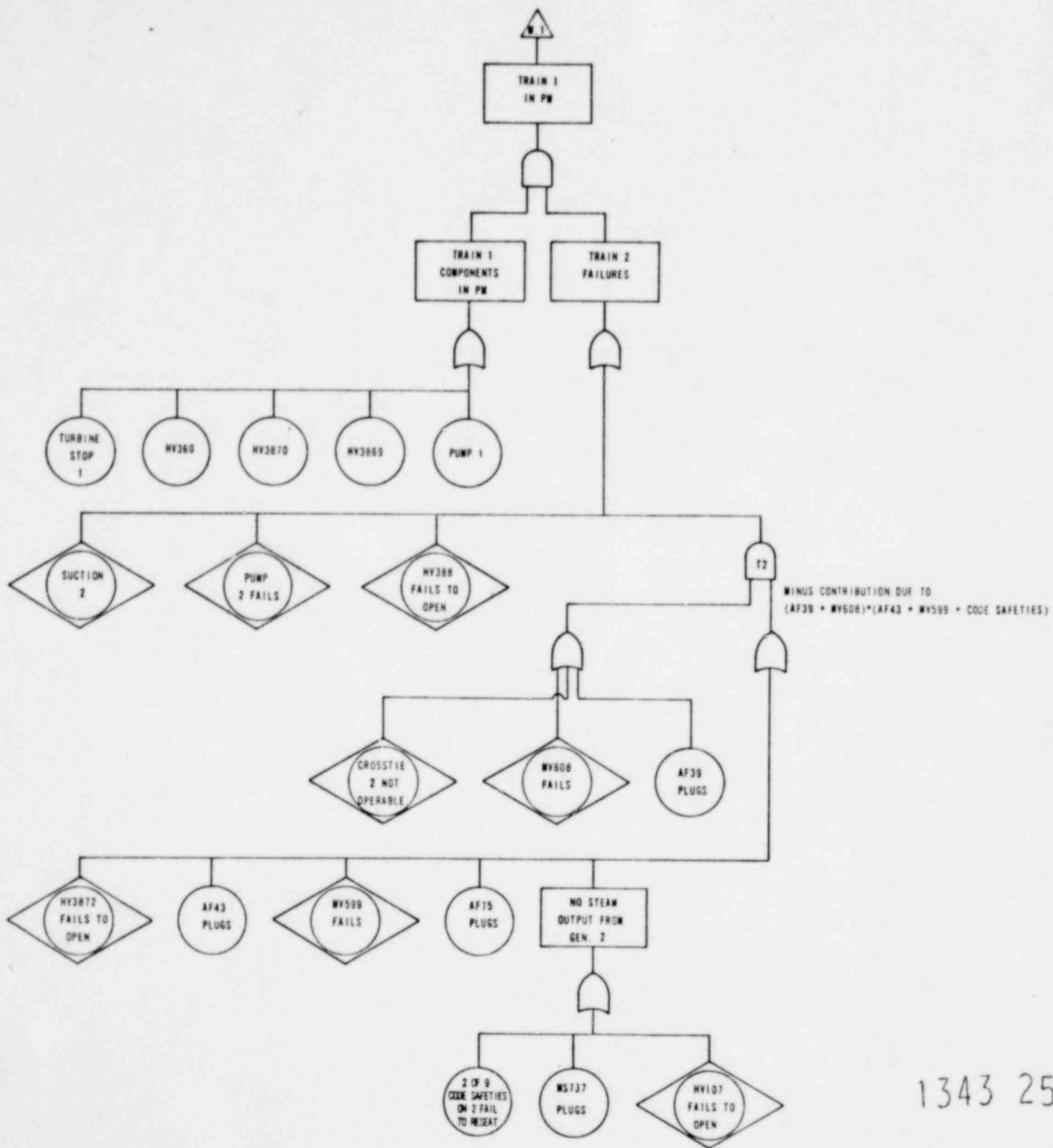


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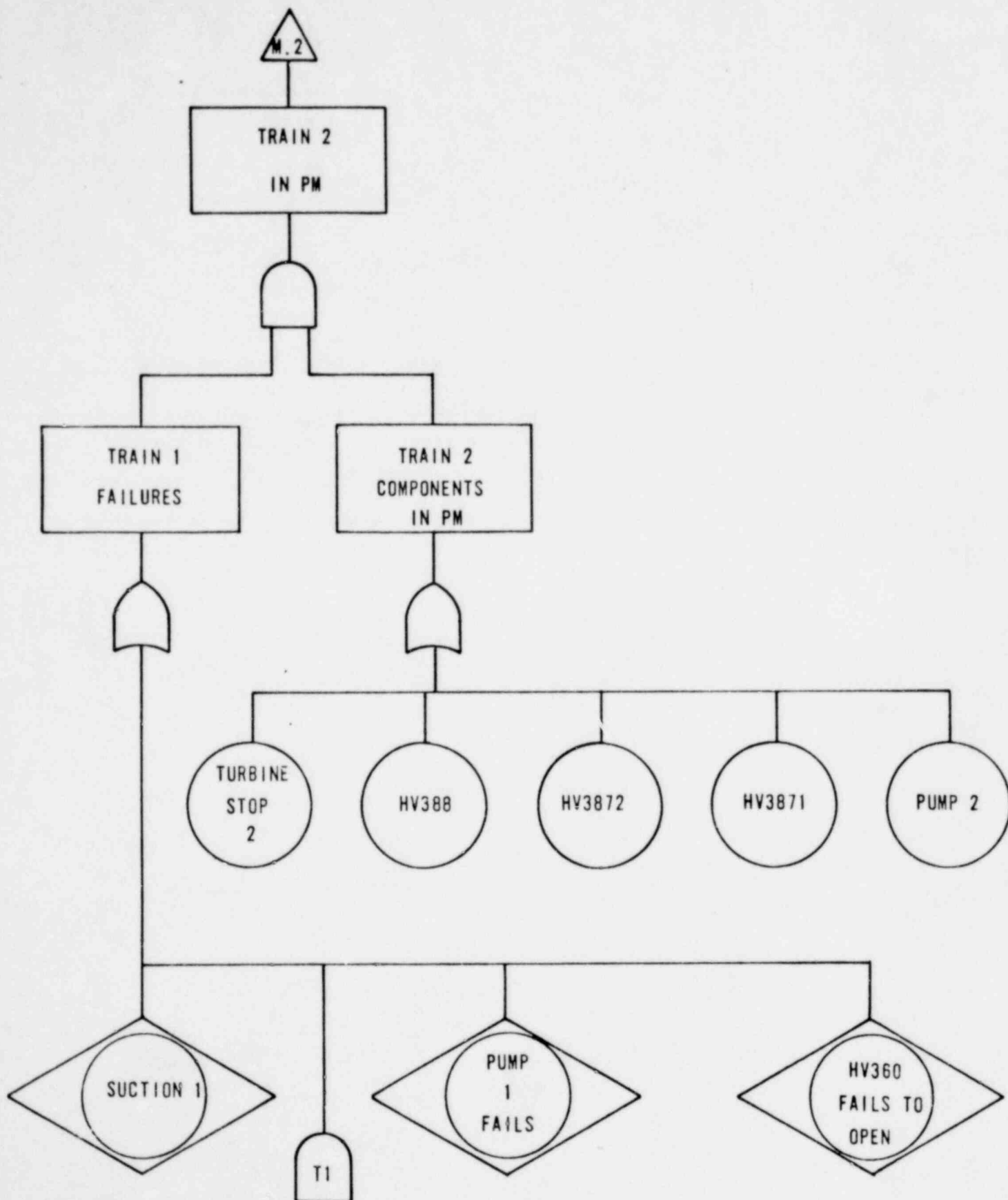


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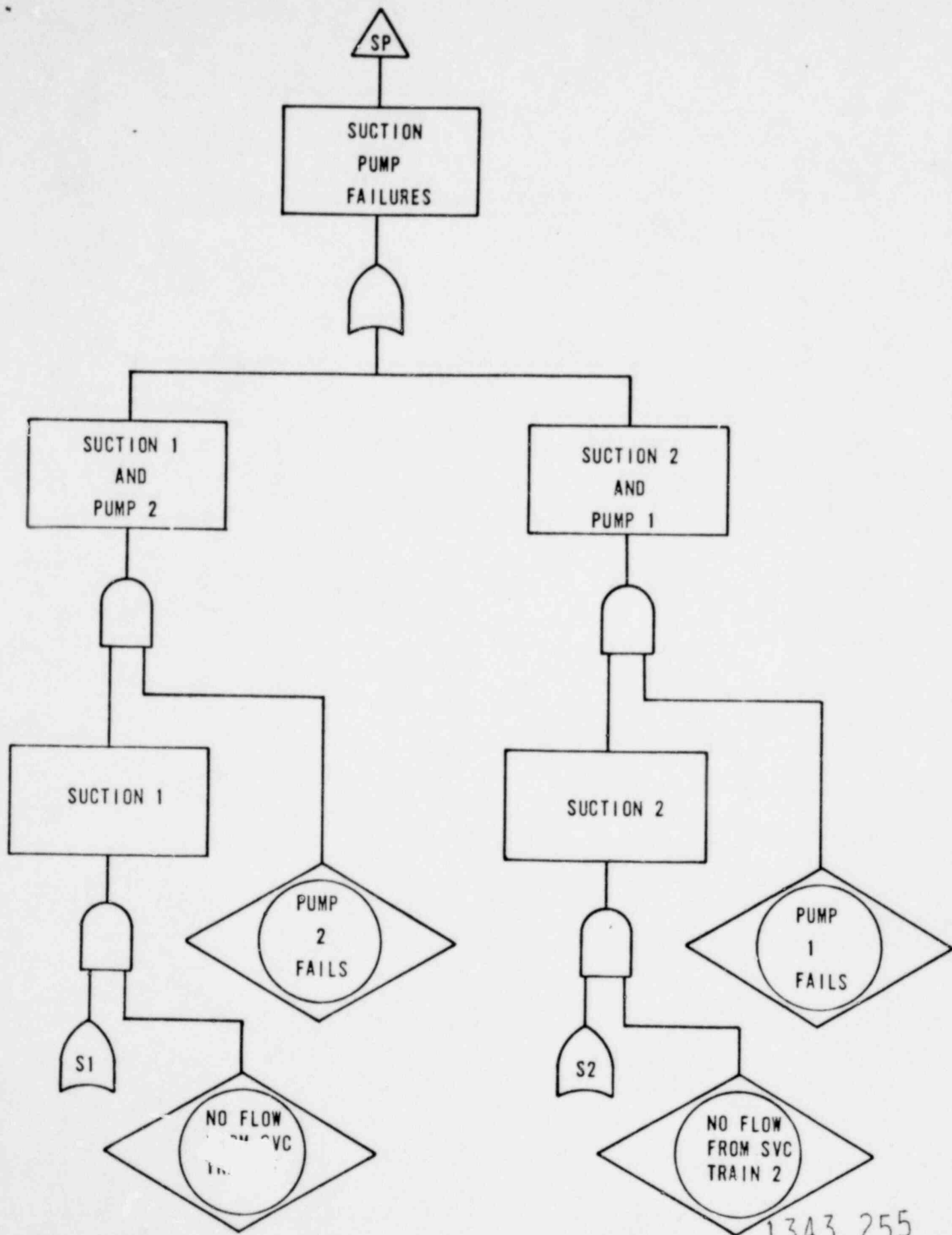




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APPENDIX B

NRC-SUPPLIED DATA USED FOR PURPOSES OF CONDUCTING A COMPARATIVE ASSESSMENT OF EXISTING AFWS DESIGNS & THEIR POTENTIAL RELIABILITIES

Point Value Estimate
of Probability of*
Failure on Demand

I. Component (Hardware) Failure Data

a. Valves:

Manual Valves (Plugged)	$\sim 1 \times 10^{-4}$
Check Valves	$\sim 1 \times 10^{-4}$
Motor Operated Valves	
• Mechanical Components	$\sim 1 \times 10^{-3}$
• Plugging Contribution	$\sim 1 \times 10^{-4}$
• Control Circuit (Local to Valve)	
w/Quarterly Tests	$\sim 6 \times 10^{-3}$
w/Monthly Tests	$\sim 2 \times 10^{-3}$

b. Pumps: (1 Pump)

Mechanical Components	$\sim 1 \times 10^{-3}$
Control Circuit	
• w/Quarterly Tests	$\sim 7 \times 10^{-3}$
• w/Monthly Tests	$\sim 4 \times 10^{-3}$

c. Actuation Logic

$\sim 7 \times 10^{-3}$

*Error factors of 3-10 (up and down) about such values are not unexpected for basic data uncertainties.

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II. Human Acts & Errors - Failure Data:

← Estimated Human Error/Failure Probabilities →
 ← Modifying Factors & Situations →

	With Valve Position Indication in Control Room		With Local Walk- Around & Double Check Procedures		w/o Either	
	Point Value Estimate	Est on Error Factor	Point Value Estimate	Est on Error Factor	Point Value Estimate	Est on Error Factor
A) <u>Acts & Errors of a Pre-Accident Nature</u>						
1. Valves mispositioned during test/maintenance.						
a) Specific single valve wrongly selected out of a population of valves during conduct of a test or maintenance act ("X" no. of valves in population at choice).	$\frac{1}{20} \times 10^{-2} \times \frac{1}{X}$	20	$\frac{1}{20} \times 10^{-2} \times \frac{1}{X}$	10	$10^{-2} \times \frac{1}{X}$	10
b) Inadvertently leaves correct valve in wrong position.	$\sim 5 \times 10^{-4}$	20	$\sim 5 \times 10^{-3}$	10	$\sim 10^{-2}$	10
2. More than one valve is affected (coupled errors).	$\sim 1 \times 10^{-4}$	20	$\sim 1 \times 10^{-3}$	10	$\sim 3 \times 10^{-3}$	10

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II. Human Acts & Errors - Failure Data (Cont'd):

← Estimated Human Error/Failure Probabilities →

	<u>Time Actuation Needed</u>	<u>Estimated Failure Prob. for Primary Operator to Actuate AFWS Components</u>
B) <u>Acts & Errors of a Post- Accident Nature</u>		
1. Manual actuation of AFWS from Control Room. Considering "non-dedicated" operator to actuate AFWS and possible backup actuation of AFWS.	~5 min. ~15 min. ~30 min.	$\sim 5 \times 10^{-2}$ $\sim 1 \times 10^{-2}$ $\sim 5 \times 10^{-3}$

III. Maintenance Outage Contribution

Maintenance outage for pumps and EMOVS:

$$Q_{\text{Maintenance}} \approx \frac{0.22 \text{ (#hours/maintenance act)}}{720}$$

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Question

Attachment 2

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. 2¹, 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 have been analyzed and submitted to the Staff in Section 6 of the May 7, 1979³, 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 uncover 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 uncover 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, 1979⁴. 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) 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 modeling 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 uncover would occur.

Besides the cases involving actuation of the pressurizer relief valves, analyses were performed for a total loss of SG heat sink and are provided in references 5 and 6. 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-10104, 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.
- 5 Letter from R.B. Davis to 177 Owner's Group, Subject: "Complete Loss of Feedwater Transient," September 11, 1979.
- 6 Letter from R.B. Davis to Mr. C.R. Domeck, Subject: "Complete Loss of Feedwater Transient on Davis-Besse," September 11, 1979.

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 Model¹ for small breaks utilizes 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-bend 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 uncover, 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 3-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 uncover, the CRAFT2 SG model is adequate for demonstrating compliance to 10 CFR 50.46.

In performing small break evaluations, the CRAFT2² 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 volume 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.

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 report¹ 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 SC 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 uncover.

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 uncover, occur for breaks in excess of 0.05 ft^2 . As demonstrated in the May 7, 1979 report⁵, 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 uncover.

As demonstrated, the SG model utilized in the small break evaluations 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 "B&W's ECCS Evaluation Model," BAW-10104, Rev. 3, Babcock & Wilcox, August 1977.
- 2 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.
- 3 Letter, J.H. Taylor (B&W) to S.A. Varga (NRC), July 18, 1979.
- 4 "Multinode Analysis of Small Breaks for B&W's 177-Fuel Assembly Nuclear Plants With Raised Loop Arrangement and Internals Vent Valves," BAW-10075A, Rev. 1, 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.

CRAFT2 SIMULATION OF THE MARCH 28, 1978
TMI-2 TRANSIENT

Docket No. 50-346
License No. NPF-3
Serial No. 556
November 13, 1979

Attachment 4

I. INTRODUCTION

In the May 7, 1979 "Blue Book" reports¹, 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-10104² and the July 18, 1978 letter report³, 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 uncover which is in reasonable agreement with the NSAC-1⁴ 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-10092⁵, 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:

- 1343 266
- 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 scenarios⁴ 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 net 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.

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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

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CRAFT code. As long as fluid is predicted to remain within the core node, regardless of the actual amount of core uncover, 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 mixture level for the transient is given in Figure 8. As shown, no core uncover 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 uncover 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.

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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 all 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

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

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.

core liquid and how much energy is used to superheat the steam. However, since the simulation was basically made to determine how the core uncover 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 uncover 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 uncover 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.

REFERENCES

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- 3 Letter J.H. Taylor (B&W) to S.A. Varga (NRC), July 18, 1978.
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- 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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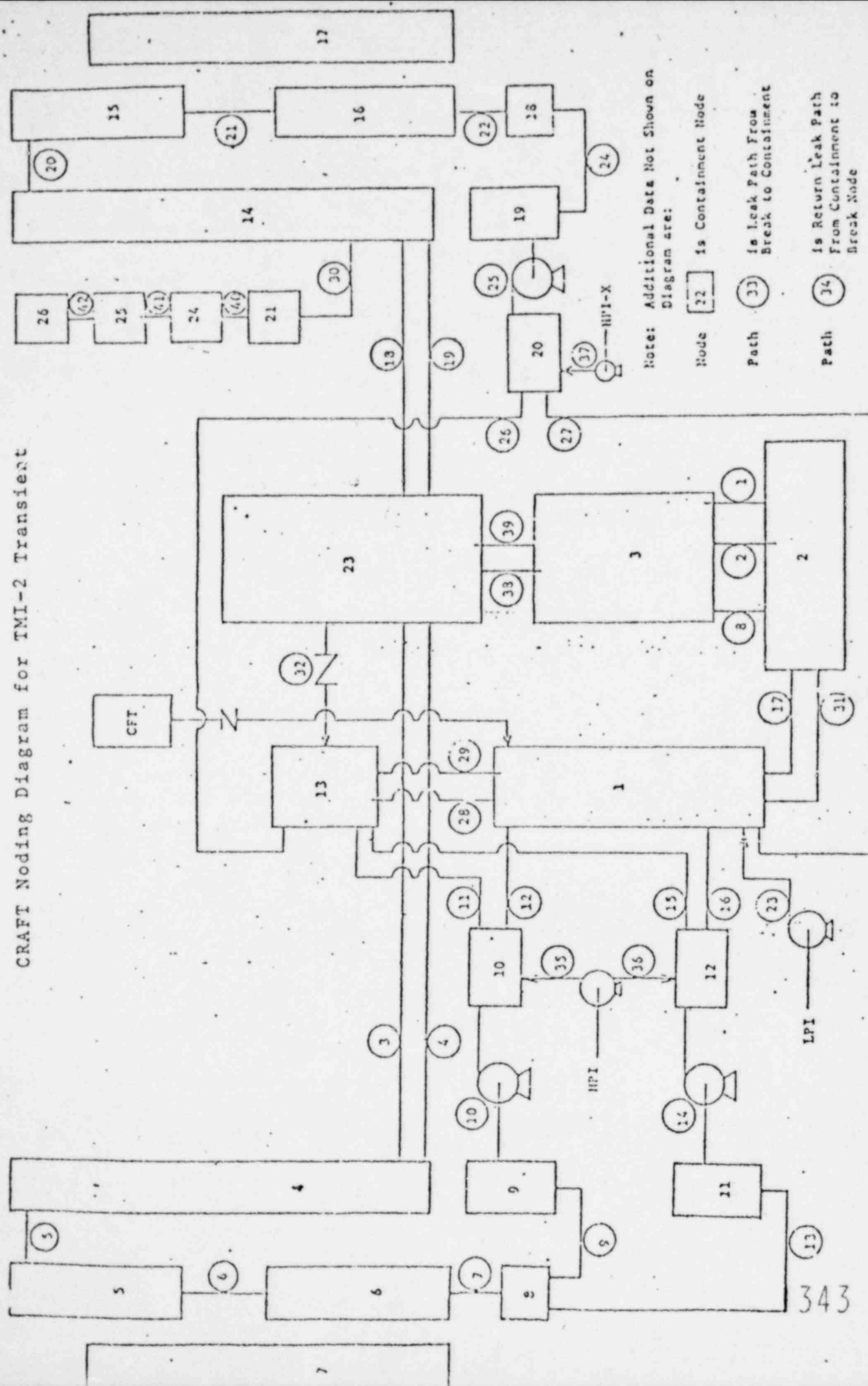
Table 1. Comparison of CRAFT2 Assumption
to NSAC Scenario

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



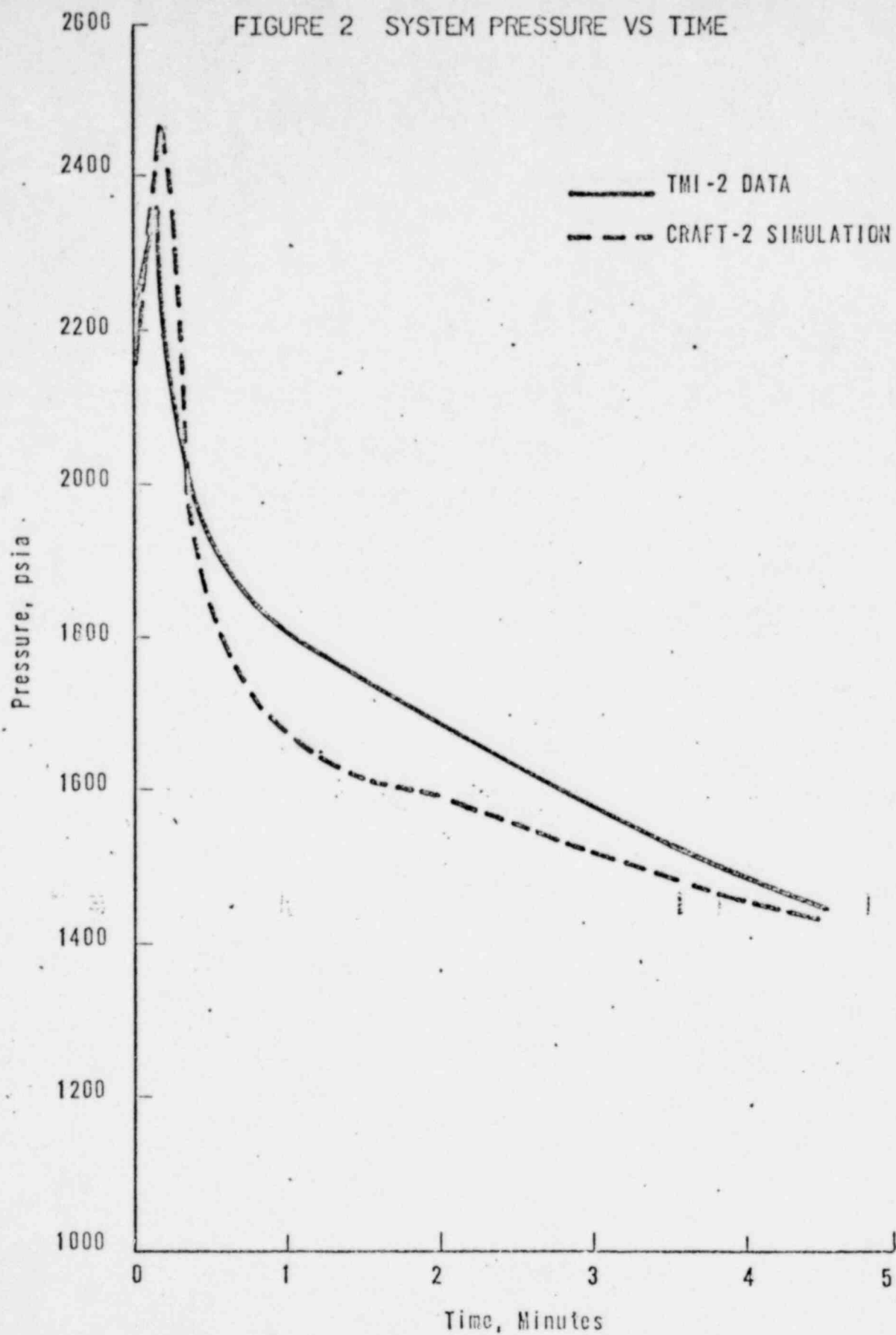
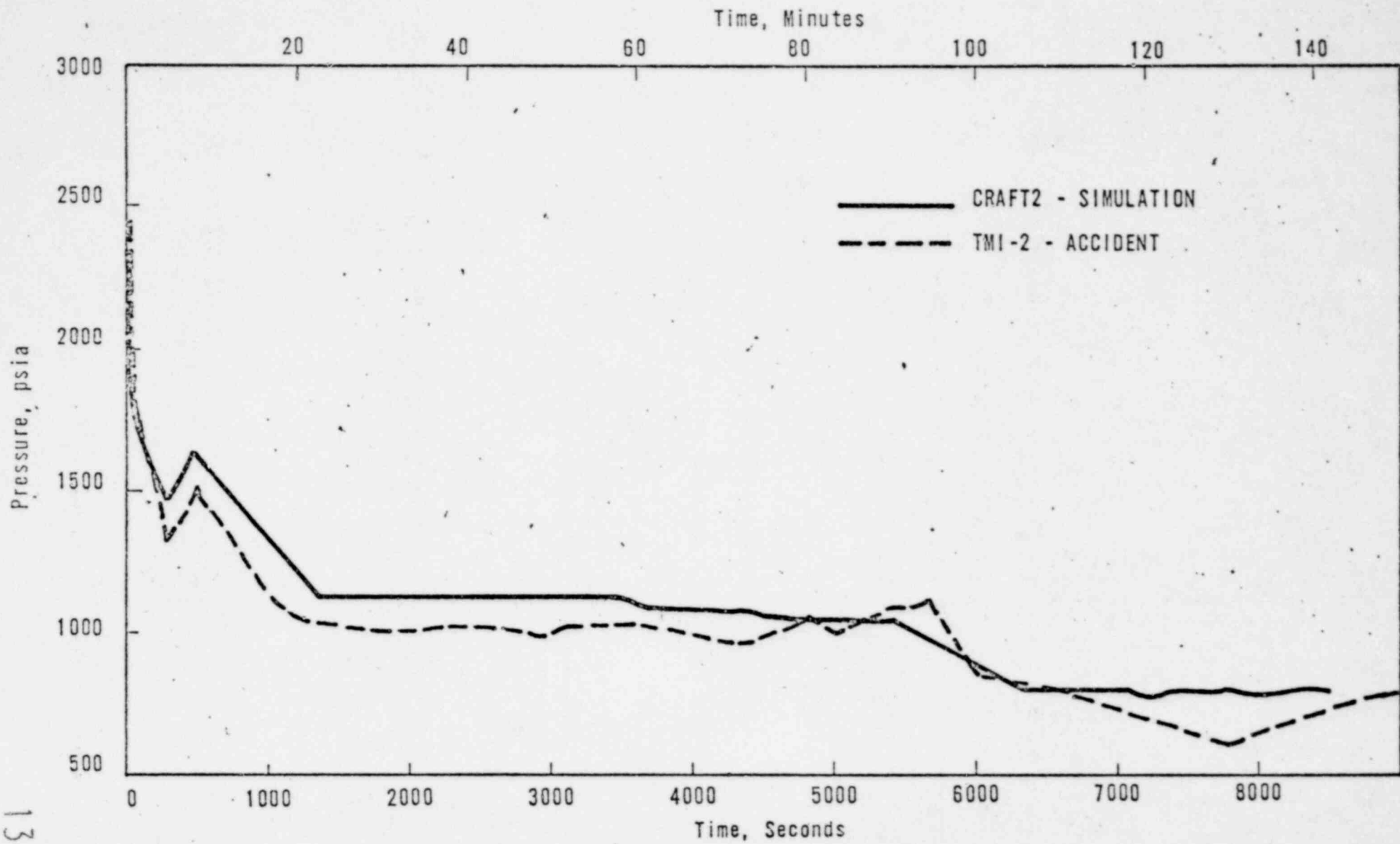
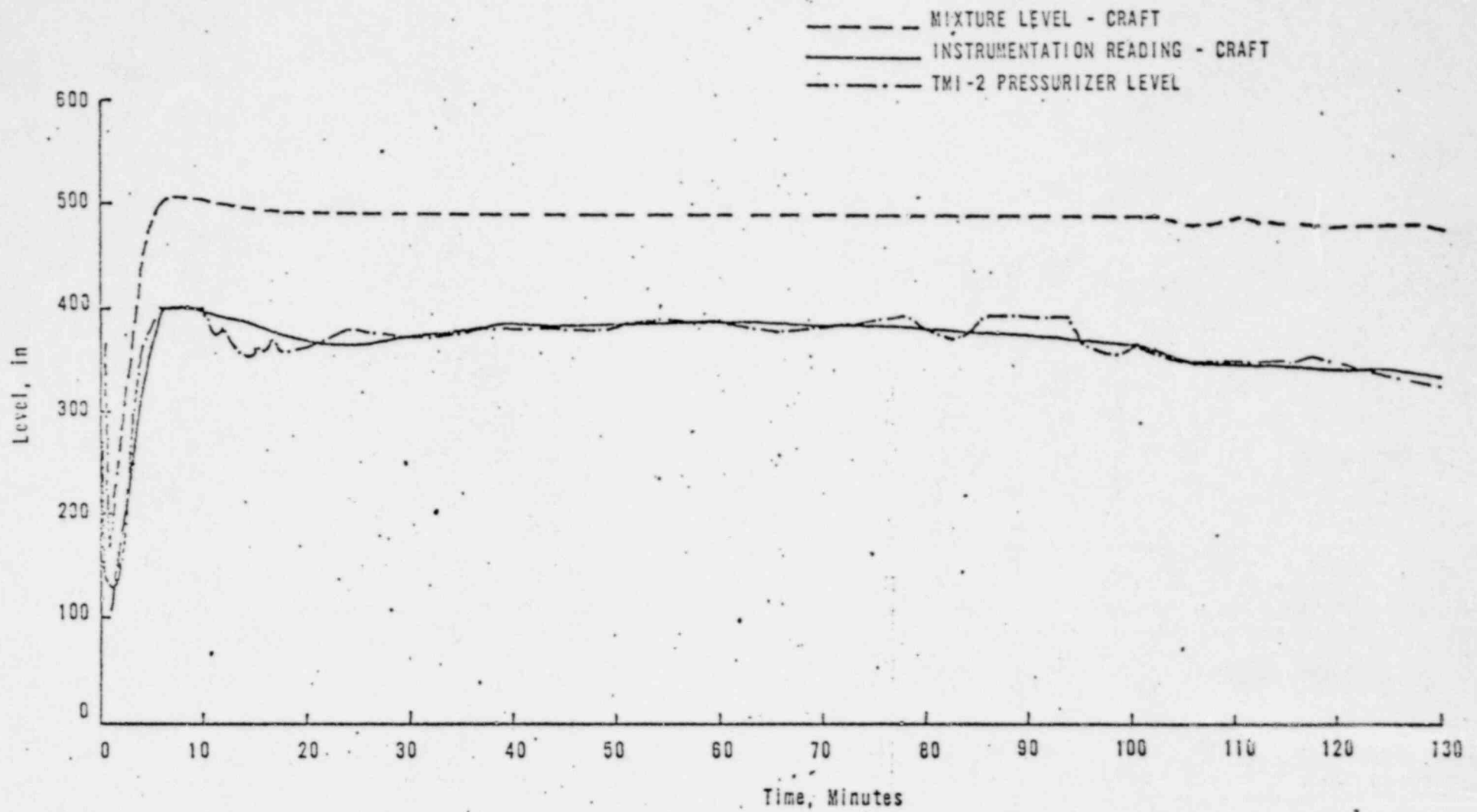


FIGURE 3 REACTOR COOLANT PRESSURE



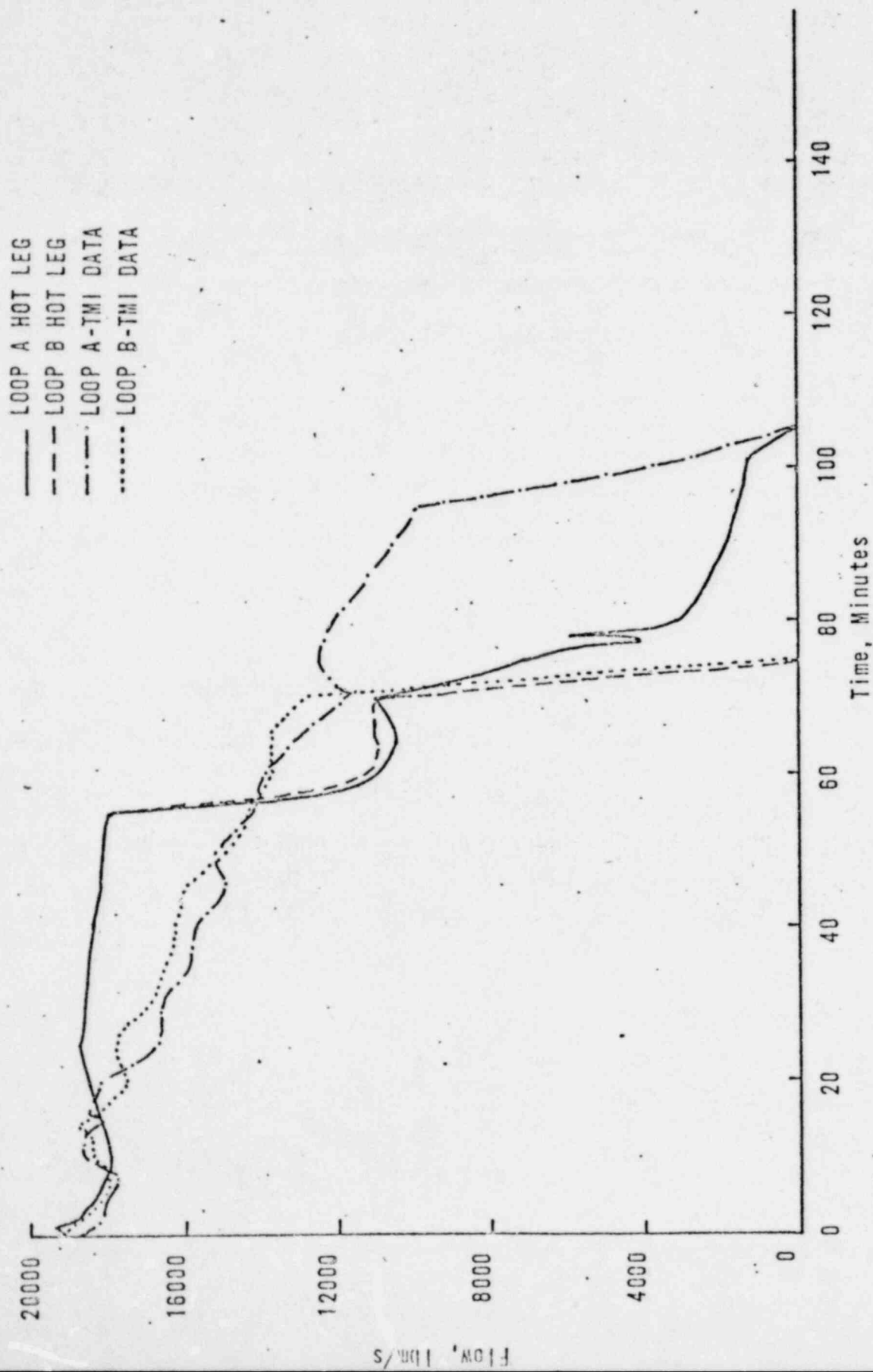
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FIGURE 4 PRESSURIZER LEVEL



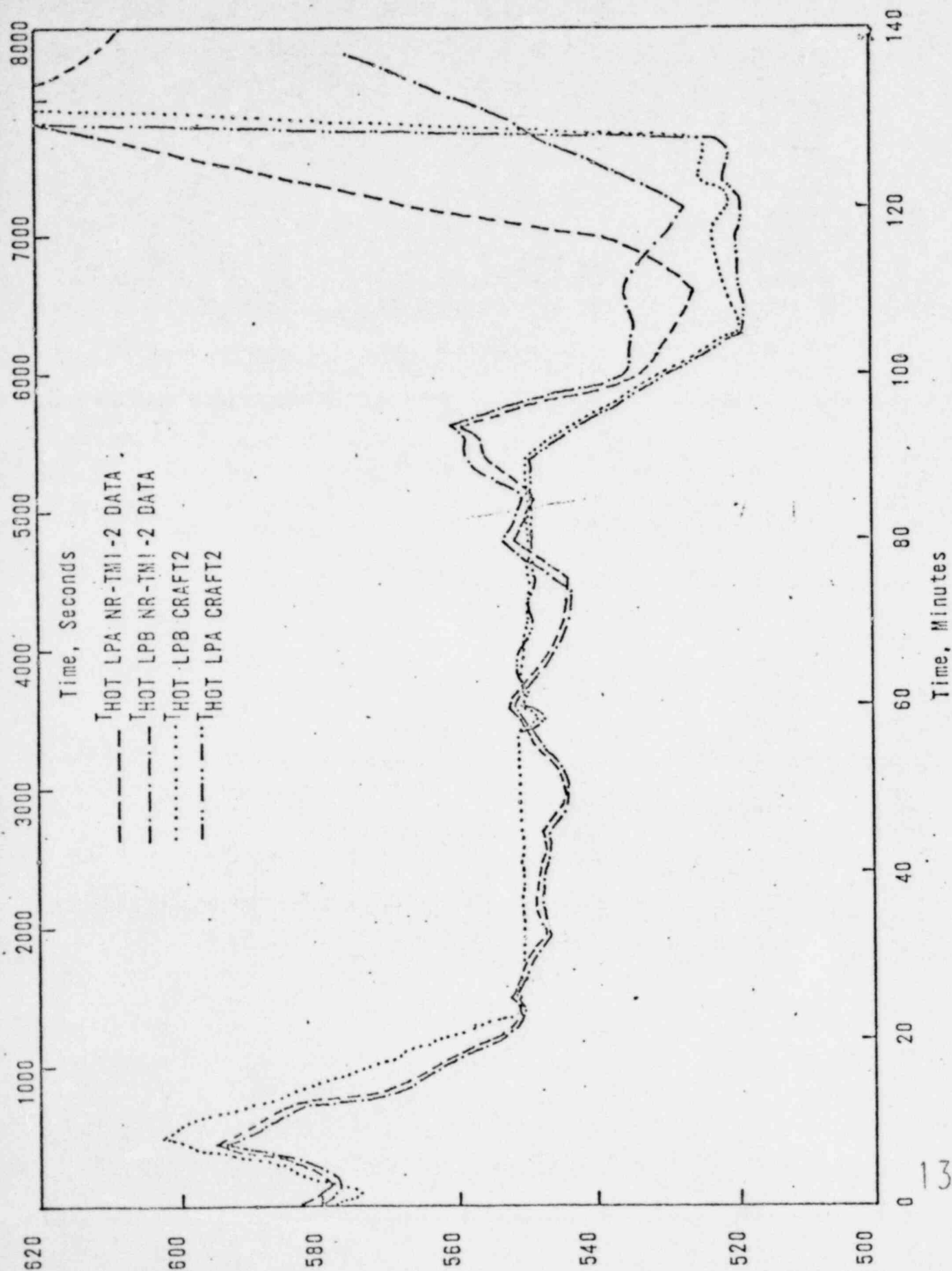
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FIGURE 5. HOT LEG FLOW



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FIGURE 6 HOT LEG TEMPERATURE



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FIGURE 7 COLD LEG TEMPERATURE

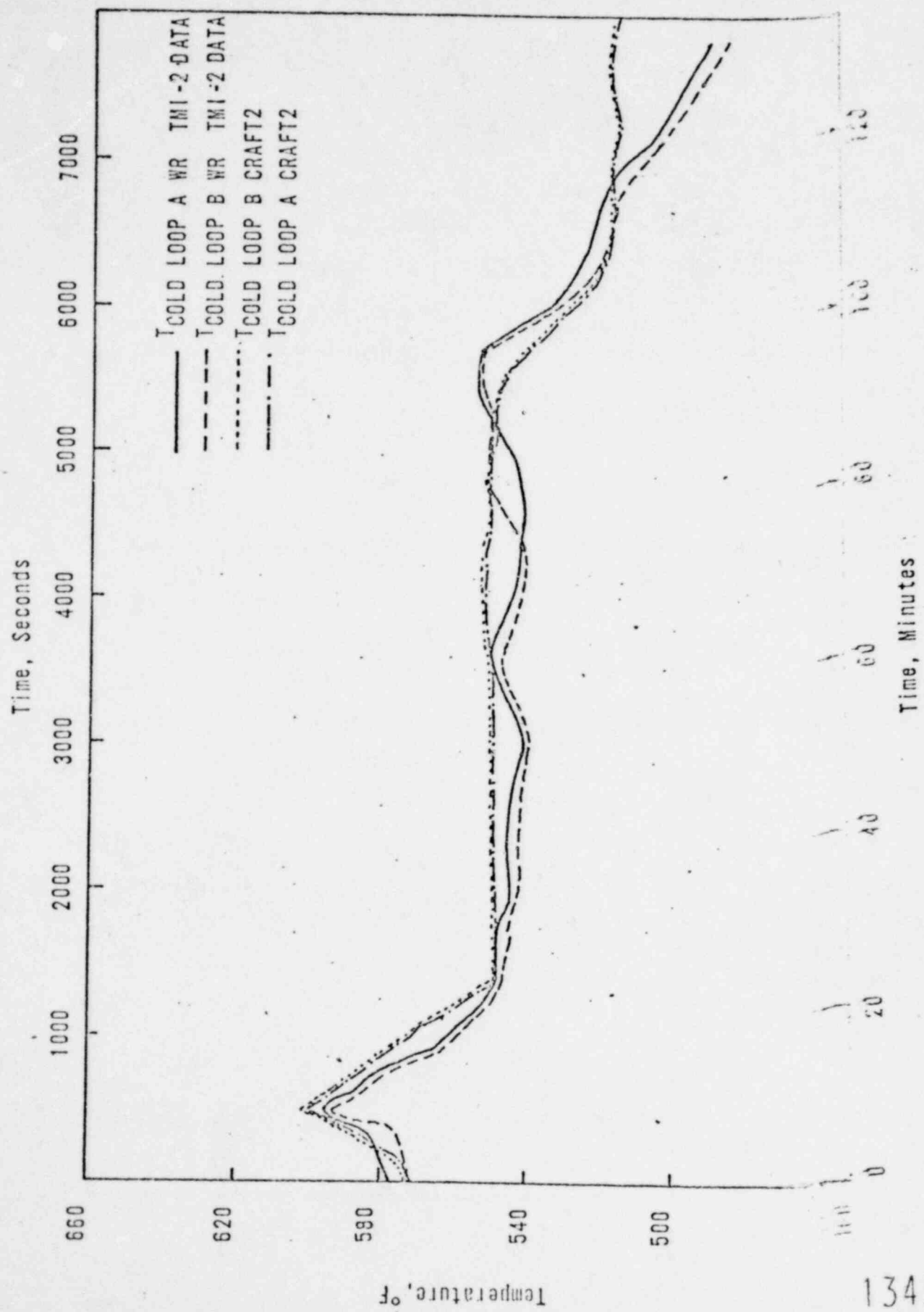
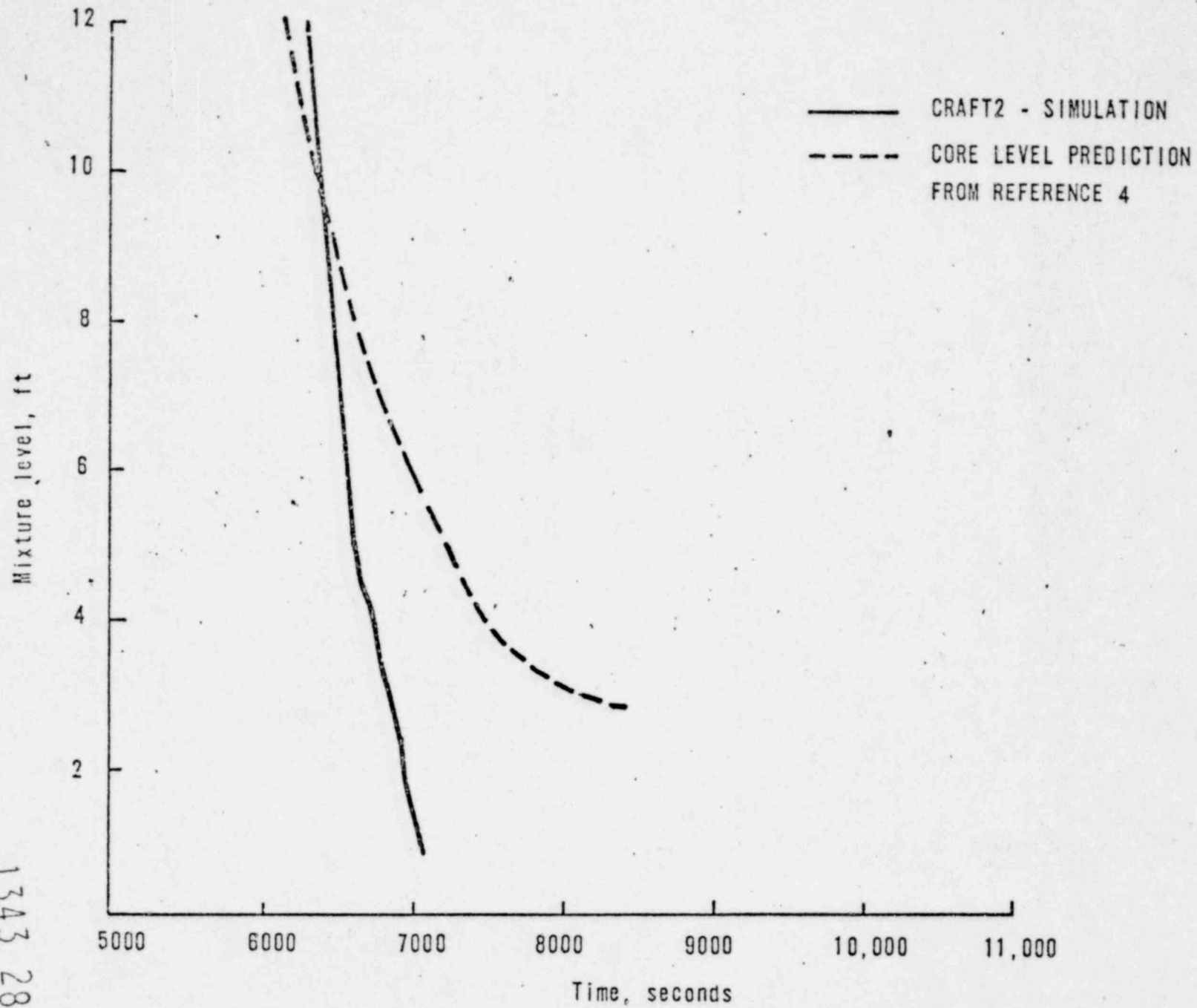
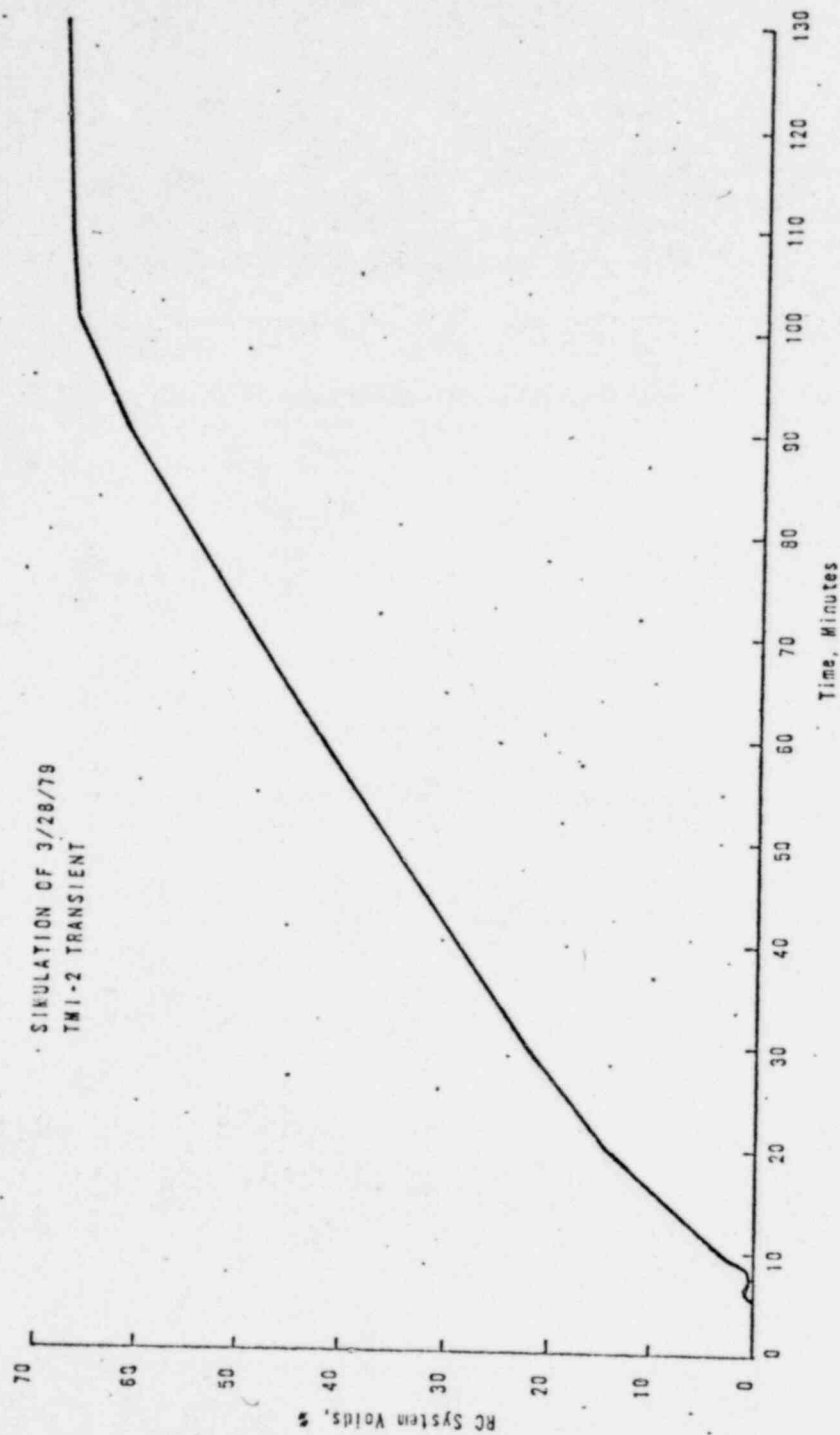


FIGURE 8 CORE MIXTURE LEVEL



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



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