ENCLOSURE 1

NOTICE OF VIOLATION

Duke Power Company
Catawba Unit 1

Docket No. 50-413
License No. NPF-35

During the Nuclear Regulatory Commission (NRC) inspection conducted on August 26, 1987, through September 25, 1987, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1987), the violation is listed below:

Technical Specification 3.7.1.2 requires at least three independent steam generator auxiliary feedwater pumps and associated flow paths to be operable. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pump to operable status within 72 hours or be in at least hot standby within the next 6 hours and in hot shutdown within the following 6 hours.

Technical Specification 6.8.1 requires that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A to Regulatory Guide 1.33, Revision 2. Catawba Nuclear Station Directive 3.1.15, Activities Affecting Station Operations or Operating Instructions, requires that permission be obtained from the Shift Supervisor or other Supervisor with operational control prior to removing from service an instrument or component that may affect unit operation.

Contrary to the above, pressure switch 1CAPS5131 was removed from service at some point between July 17, 1986, and July 6, 1987, by shutting its isolation valve without obtaining permission from the Shift Supervisor or other Supervisor with operational control. This caused the flow path from Auxiliary Feedwater Pump 1B to the 1C Steam Generator to be inoperable as it would have isolated under certain conditions and the licensee did not maintain the unit in an Operational Mode in which Technical Specification 3.7.1.2 did not apply.

This is a Severity Level IV Violation (Supplement I) applicable to Unit 1 only.

Pursuant to the provisions of 10 CFR 2.201, Duke Power Company is hereby required to submit a written statement or explanation to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector, Catawba, within 30 days of the date of the letter transmitting this

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PDR ADOCK 05000413
Q PDR

Notice. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include: (1) admission or denial of the violation, (2) the reason for the violation if admitted, (3) the corrective steps which have been taken and the results achieved, (4) the corrective steps which will be taken to avoid further violations, and (5) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending the response time. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken.

FOR THE NUCLEAR REGULATORY COMMISSION

for J. Nelson Grace
Regional Administrator

Dated at Atlanta, Georgia
this 01 day of Dec. 1987

ENCLOSURE 2

ENFORCEMENT CONFERENCE SUMMARY

On November 6, 1987, representatives of the Duke Power Company (DPC) met with the NRC at the NRC's request in the Region II office in Atlanta, Georgia. The conference was held to discuss the unauthorized isolation of an Auxiliary Feedwater (AFW) system pressure switch which rendered the AFW system unable to function as designed under certain conditions. A list of conference attendees is contained in Attachment 1.

Following opening remarks given by M. L. Ernst, NRC, Deputy Regional Administrator, DPC gave a presentation addressing the NRC concerns raised by the isolation of the pressure switch.

An outline of the DPC presentation is contained herein as Attachment 2.

The DPC presentation showed that while the AFW system would have operated in an off-normal configuration, the isolation of the pressure switch would not have prevented the system from performing its intended function under any postulated conditions.

This meeting served to enhance Region II's understanding of the issue and DPC's plan to prevent recurrence of similar problems. The NRC enforcement action concerning this issue is discussed in Enclosure 1.

Attachments:
1. List of Attendees
2. DPC Presentation Summary

ATTACHMENT 1

LIST OF ATTENDEES

U.S. Nuclear Regulatory Commission

M. L. Ernst, Deputy Regional Administrator
T. A. Peebles, Chief, Reactor Projects Section 2A, Division of Reactor Projects (DRP)
W. T. Orders, Senior Resident Inspector, McGuire
C. W. Hehl, Deputy Director, DPR
E. W. Merschoff, Deputy Director, Division of Reactor Safety (DRS)
G. R. Jenkins, Director, Enforcement Investigation Coordination Staff (EICS)
B. Uryc, Enforcement Coordinator
B. Bonser, Project Engineer
M. S. Lesser, Resident Inspector
D. Hood, Project Manager, NRR

Duke Power Company

D. Rains, Superintendent of Maintenance, McGuire
J. W. Hampton, Manager, Catawba Nuclear Station (CNS)
H. B. Barron, Superintendent of Operations, CNS
G. Smith, Superintendent of Maintenance, CNS
F. P. Schiffley, II, Licensing Engineer, CNS
G. E. Swindlehurst, Superintendent Design Engineer
E. O. McCraw, Compliance Engineer, McGuire
H. B. Tucker, Vice President, Nuclear Production
N. A. Rutherford, System Engineer, Licensing

ATTACHMENT 2

DPC PRESENTATION SUMMARY

DUKE POWER COMPANY/NRC

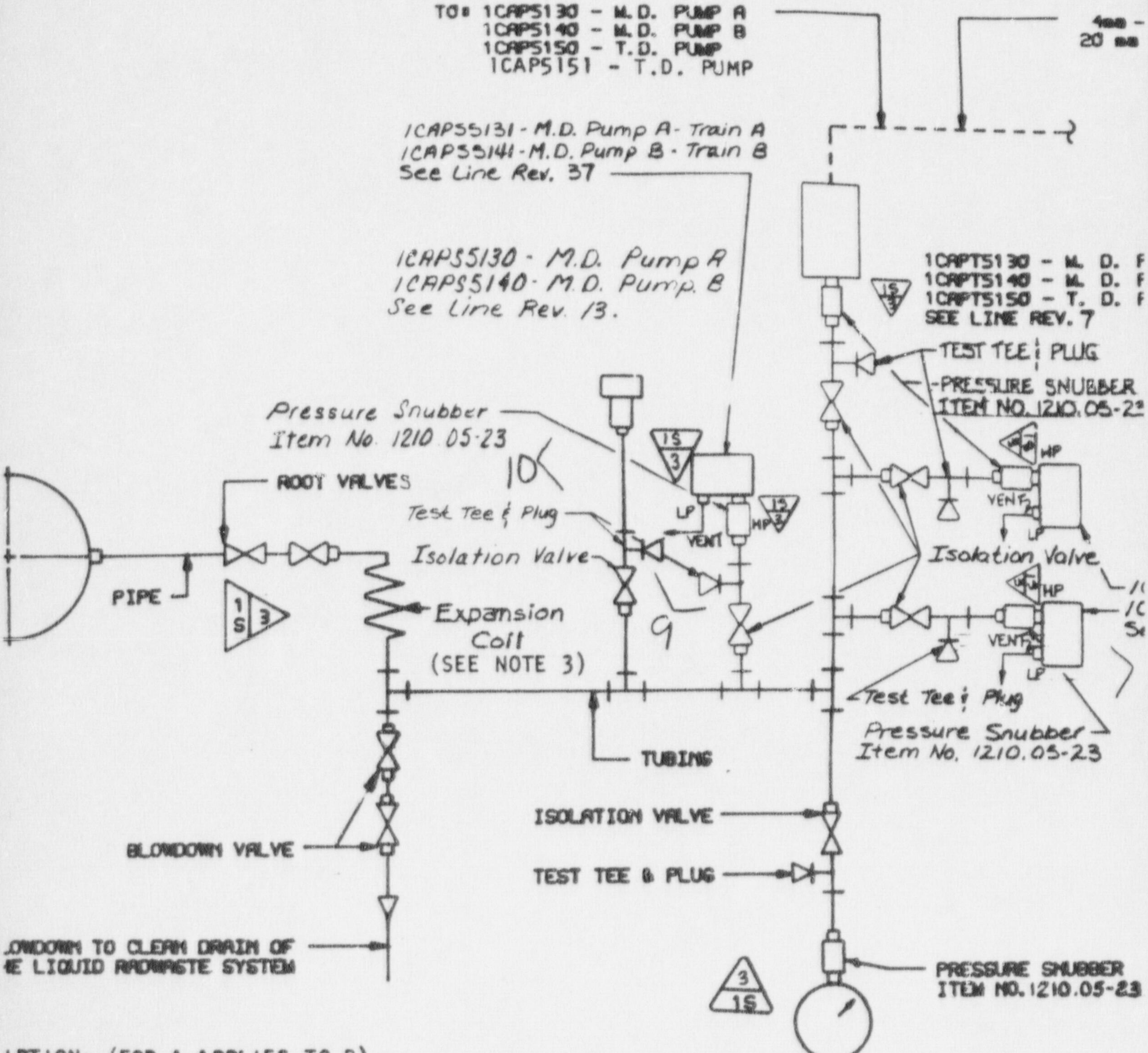
ENFORCEMENT CONFERENCE

NOVEMBER 6, 1987

1:00 PM

- CATAWBA EVENT

- PROBLEM DESCRIPTION	J. W. HAMPTON
- SYSTEM DESCRIPTION	H. B. BARRON
- SAFETY CONSEQUENCES	G. B. SWINDLEHURST
- ROOT CAUSE/CORRECTIVE ACTIONS	G. T. SMITH
- SUMMARY STATEMENT	J. W. HAMPTON



1CAPG5130 - M.D. PUMP A
1CAPG5140 - M.D. PUMP B
SEE LINE REV. 7

IPTION: (FOR A APPLIES TO B)

APG5130 MONITOR MOTOR-DRIVEN AUXILIARY FEEDWATER PUMP
SURE. 1CAPT5130 PROVIDES A SIGNAL TO REMOTE INDICATOR
5130 PROVIDES LOCAL INDICATION. 1CAPS5130 PROVIDES A
INPUT FOR RESPONSE TIME TESTING. IF AN AUTOMATIC AUXILIARY
SIGNAL IS RECEIVED AND LOW DISCHARGE PRESSURE IS INDICATED,
NITIATE A 30 SEC. TIME DELAY. IF AFTER THE DELAY LOW PRESSURE
ED, A SETPOINT MONITOR CLOSES VALVE 1CA46B. 1CAPT5150 MONITORS
UXILIARY FEEDWATER PUMP DISCHARGE PRESSURE AND PROVIDES SIGNALS
TORS 1CAP 5150 AND 1CAP 5151. A SETPOINT MONITOR PROVIDES COMPUTER
R RESPONSE TIME TESTING. 1CAPS5150 AND 1CAPS5151 MONITOR LOW TURBINE

9	ADDED VENTS
8	REV. PER D(
7	ADDED NOTE
6	Added Ex
5.	Revised

AUXILIARY FEEDWATER SYSTEM

Design Basis:

Assure sufficient feedwater supply to the steam generators to remove energy stored in the core and primary coolant in the event of a loss of normal feedwater.

Design Requirements:

Provide 491 gpm auxiliary feedwater flow to the S/G under the following conditions:

- Failure of any pump to start.
- No credit for flow to a faulted S/G.
- No operator action for thirty minutes
- S/G pressure at first safety setpoint plus 3% (1210 psig).

Postulated Conditions

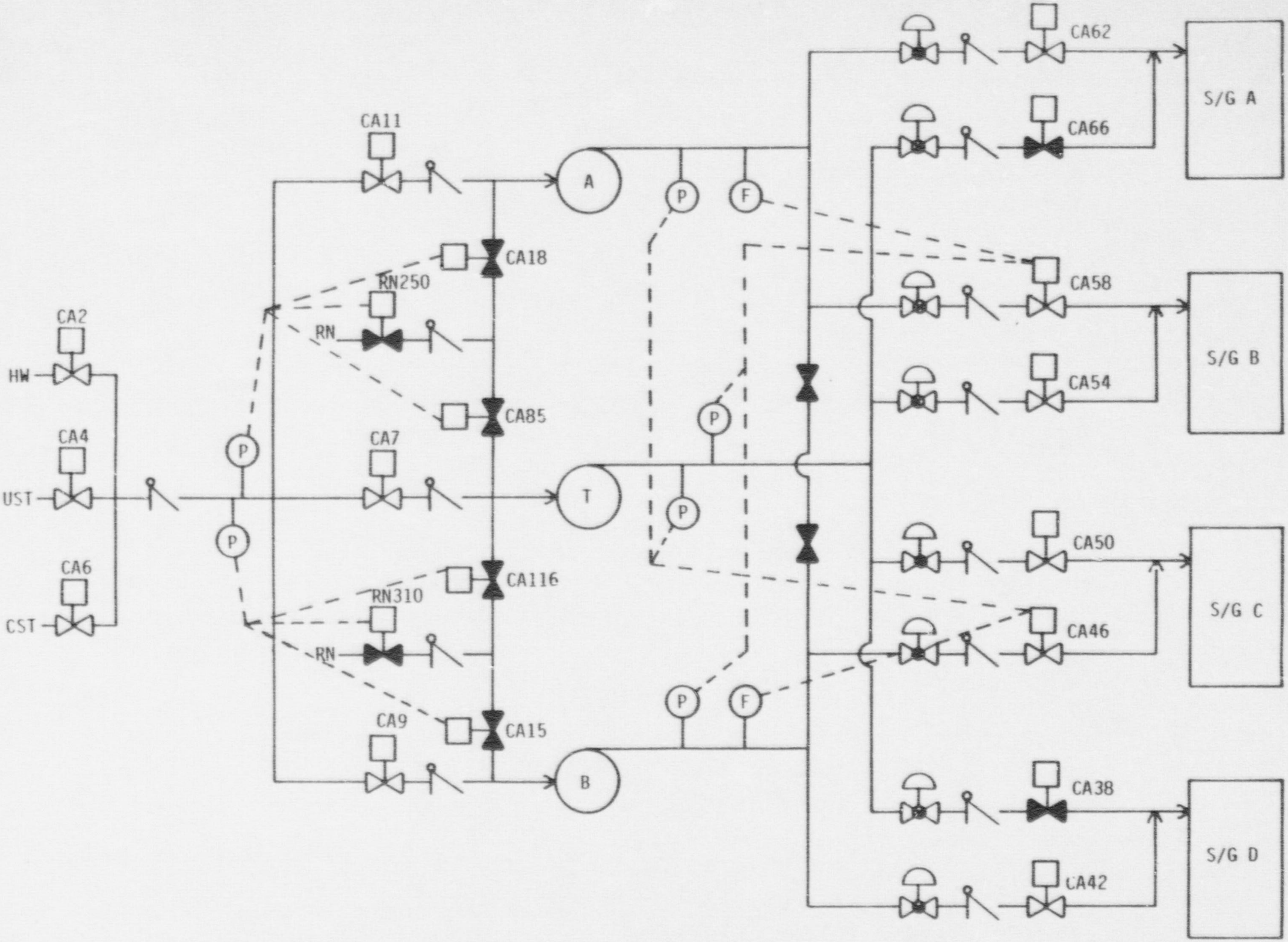
- 1CAPS5131 Isolated.
- Feedwater line rupture at A S/G.
- Turbine Driven pump start and successful run for thirty seconds, then failure.
- Loss of normal feedwater.

Actual Capacity

- Immediate operator action available.
- 433 gpm supplied to 1D S/G at 1210 psig.
- Flows in excess of 491 supplied at lower S/G pressures.
- Capability to remove core and primary system heat without operator action to unisolate intact S/G.

CONCLUSION:

While the auxiliary feedwater system would have been operating in a configuration different from that described in the FSAR, i.e. supplying only one S/G and flowrate lower than assumed at 1210 psig, the isolation of pressure switch 1CA5131 would not have prevented the system from performing its intended function under any postulated conditions.



HW
UST
CST
CA2
CA4
CA6
P
P
CA11
CA7
CA9
RN250
RN310
RN
RN
CA18
CA85
CA116
CA15
A
T
B
P
F
P
P
P
F
CA62
CA66
CA58
CA54
CA50
CA46
CA38
CA42
S/G A
S/G B
S/G C
S/G D

SAFETY ANALYSIS CONSEQUENCES

ANALYSIS OF THE IMPACT OF A
DEGRADED AUXILIARY FEEDWATER SYSTEM
ON THE LIMITING FSAR TRANSIENT

- BACKGROUND
- WORST CASE SCENARIO
- ANALYSIS METHODOLOGY
- RESULTS
- CONCLUSIONS

BACKGROUND

- FSAR STATES THAT AT LEAST TWO STEAM GENERATORS CAN BE SUPPLIED WITH A MINIMUM OF 491 GPM OF AUXILIARY FEEDWATER

- MINIMUM FLOW WITH A NORMAL ALIGNMENT IS 499 GPM AT 1210 PSIG, BASED ON ONE MOTOR-DRIVEN PUMP SUPPLYING TWO STEAM GENERATORS

- DEGRADED SITUATION PROVIDES A MINIMUM OF ONE MOTOR-DRIVEN PUMP SUPPLYING ONE STEAM GENERATOR

SG PRES(PSIG)	FLOW(GPM)
1293	400
1210	433 (86% OF 499)
1168	450
1015	500
864	550
685	600
277	700

- LESS AUXILIARY FEEDWATER FLOW THAN ASSUMED IN THE FSAR CHAPTER 15 TRANSIENT AND ACCIDENT ANALYSES

WORST CASE SCENARIO

- AS IDENTIFIED IN NRC INSPECTION REPORT 50-413(414)/87-25, THE LIMITING SCENARIO IS A MAIN FEEDWATER LINE BREAK
  - ALL MAIN FEEDWATER LOST AT TIME ZERO
  - ALL FOUR STEAM GENERATORS BLOW DOWN UNTIL MSIV CLOSURE ON LOW STEAM LINE PRESSURE
  - ONE MOTOR-DRIVEN CA PUMP SUPPLIES ONE INTACT STEAM GENERATOR

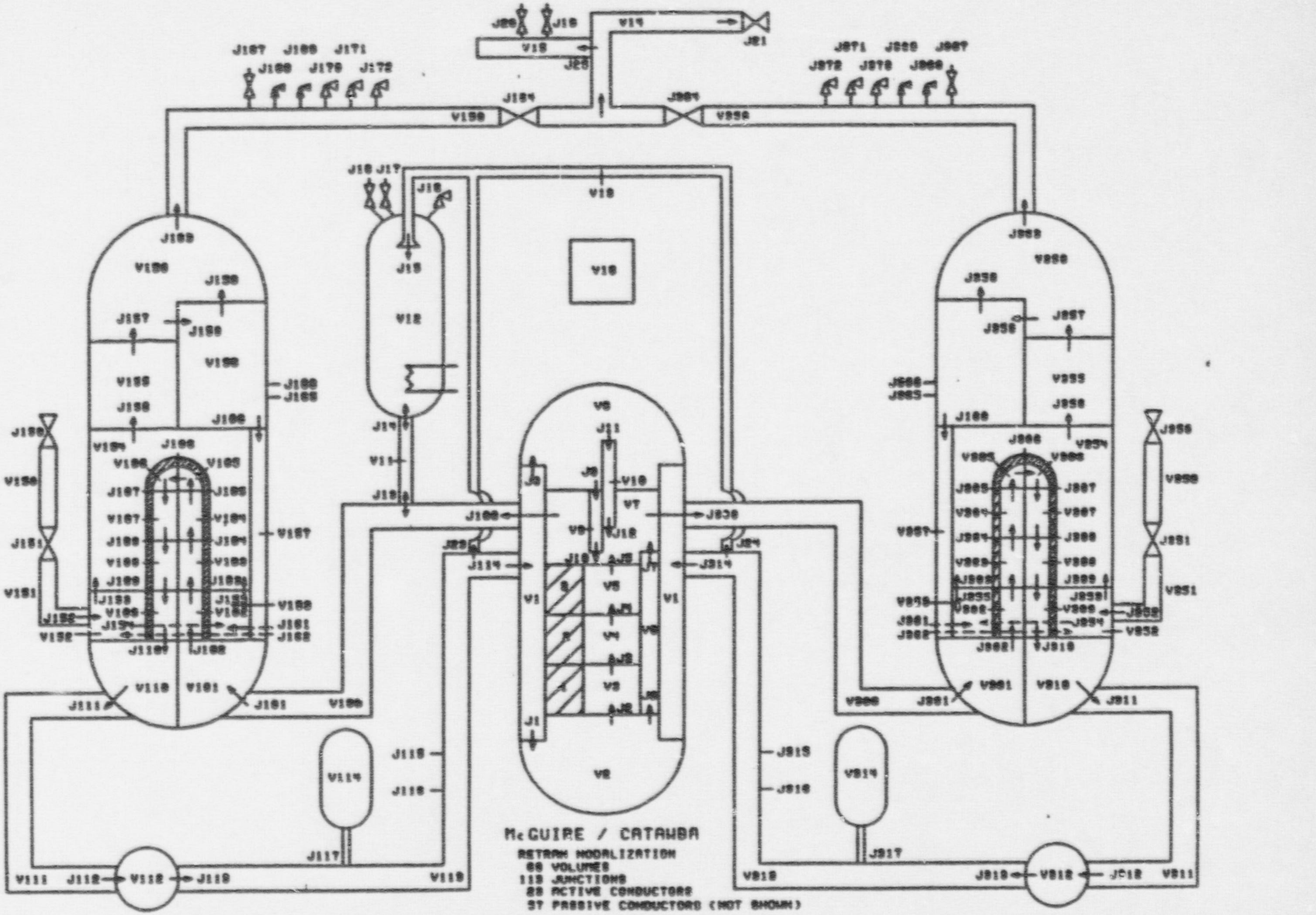
- NO OPERATOR ACTION FOR 30 MINUTES REGARDLESS OF CLEAR INDICATIONS OF SYMPTOMS AND EXISTING PROCEDURAL GUIDANCE
  - ISOLATE AUXILIARY FEEDWATER TO RUPTURED STEAM GENERATOR
  - MSIV CLOSURE ON RUPTURED STEAM GENERATOR
  - REALIGN AUXILIARY FEEDWATER TO INTACT STEAM GENERATOR
  - SAFETY INJECTION TERMINATION

ANALYSIS METHODOLOGY

- PLANT SPECIFIC SIMULATION OF THE WORST CASE SCENARIO USING A THREE-LOOP CATAWBA UNIT 1 RETRAN-02 MODEL

- BOUNDARY CONDITIONS

    - MAIN FEEDWATER LINE BREAK AT STEAM GENERATOR A
    - MINIMUM AUXILIARY FEEDWATER DELIVERED TO STEAM GENERATOR D
    - DECAY HEAT CORRESPONDING TO CONTINUOUS OPERATION OF CYCLE 2 AT FULL POWER UNTIL JULY 6, 1987 (195 EFPD).
    - NO CREDIT FOR HEAT TRANSFER DUE TO AUXILIARY FEEDWATER SUPPLIED TO THE RUPTURED STEAM GENERATOR (SIGNIFICANT CONSERVATISM)
    - SCENARIO SIMULATED FOR 30 MINUTES AT WHICH TIME OPERATOR INTERVENTION CAN BE CREDITED



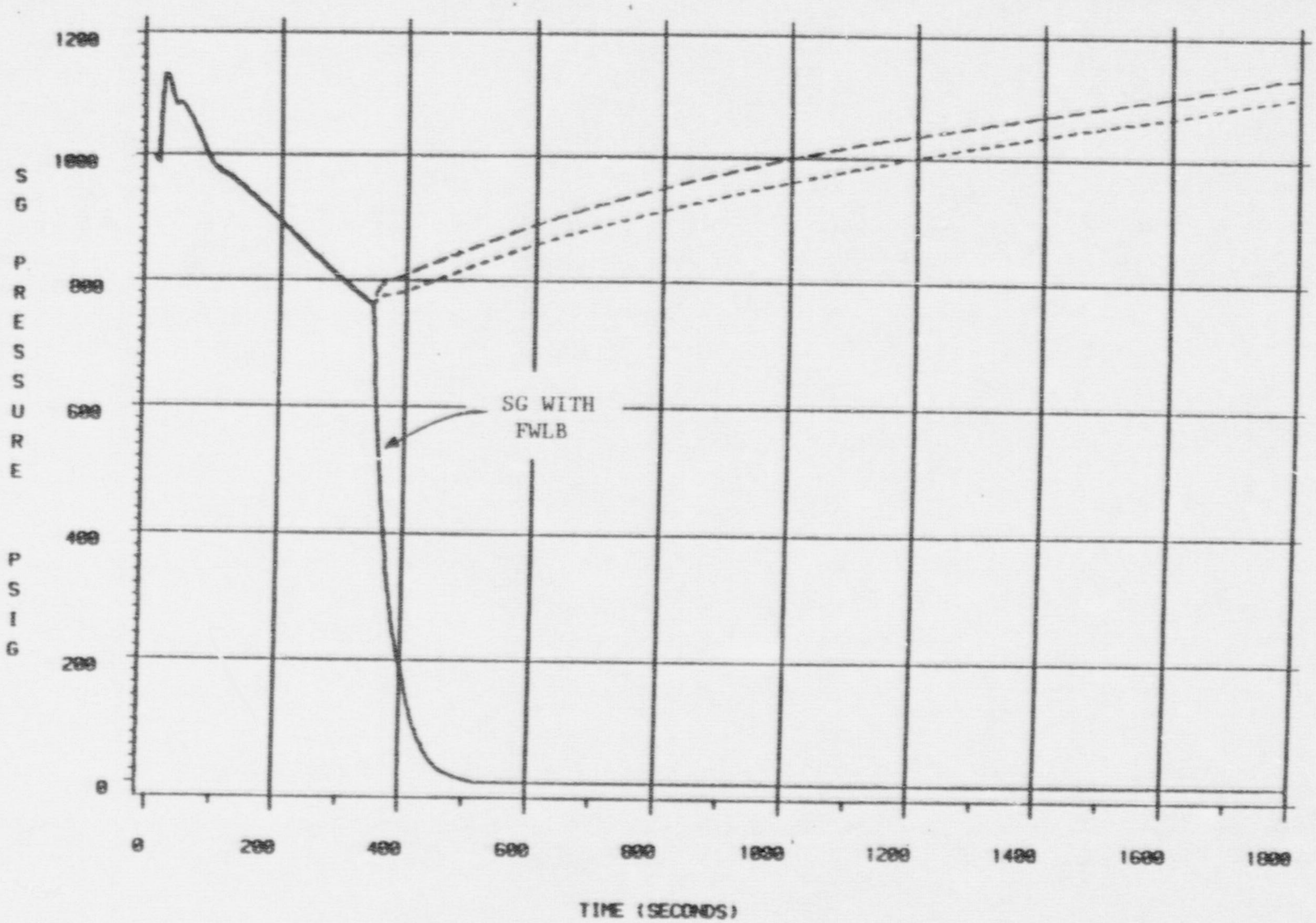
McGUIRE / CATAWBA
RETRAN NODALIZATION
68 VOLUMES
113 JUNCTIONS
28 ACTIVE CONDUCTORS
37 PASSIVE CONDUCTORS (NOT SHOWN)

RESULTS

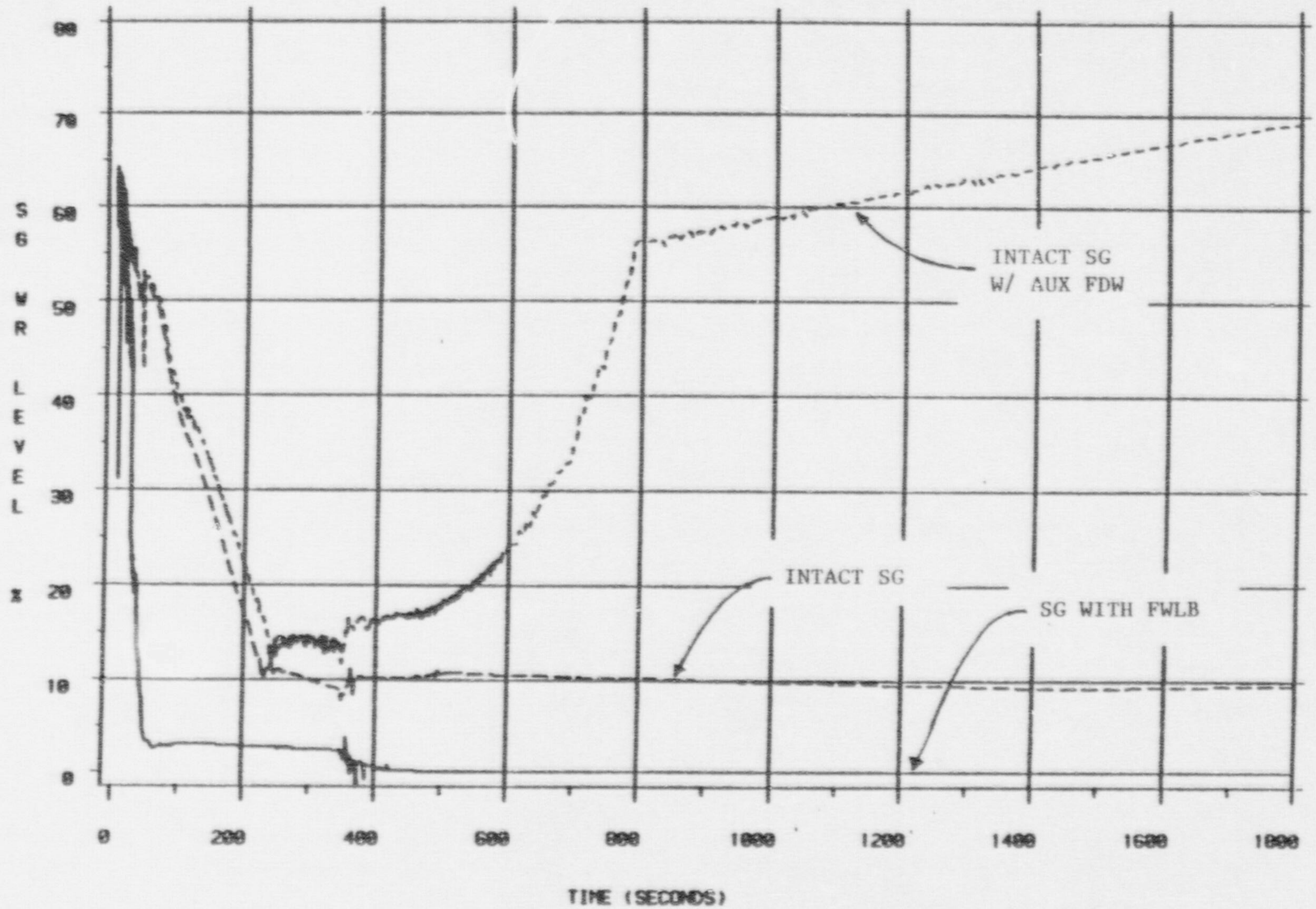
SEQUENCE OF EVENTS

TIME (SEC)	EVENT
0	FEEDWATER LINE BREAK TO SG A
2.7	PZR SPRAY ON
* 5.9	RX TRIP ON LOW-LOW SG LEVEL
6.0	TURBINE TRIP/STEAM DUMP OPENS
10.4	PZR SPRAY OFF
10.9	MOTOR-DRIVEN CA PUMP STARTS
12	PZR HEATERS ENERGIZE
14	ALL SG PORVs OPEN
15	STEAM DUMP BANK 3 CLOSED
26	STEAM DUMP BANK 2 CLOSED
29	ALL SG PORVs CLOSE
46	STEAM DUMP BANK 1 CLOSED
* 55	SG A BOILED DRY
* 173	SAFETY INJECTION ON LOW PZR PRESSURE
183	NV AND NI PUMPS STARTED
310	MINIMUM RCS PRESSURE = 1797 PSIG
* 344	STEAM LINE ISOLATION SIGNAL ON LOW SG PRESSURE
350	MINIMUM PZR LEVEL = 6%
370	MINIMUM T-AVE = 517^{O}F
559	PZR HEATERS OFF
571	PZR SPRAY ON
* 1615	PZR WATER SOLID
1624	PZR PORVs BEGIN CYCLING
1800	END OF SIMULATION

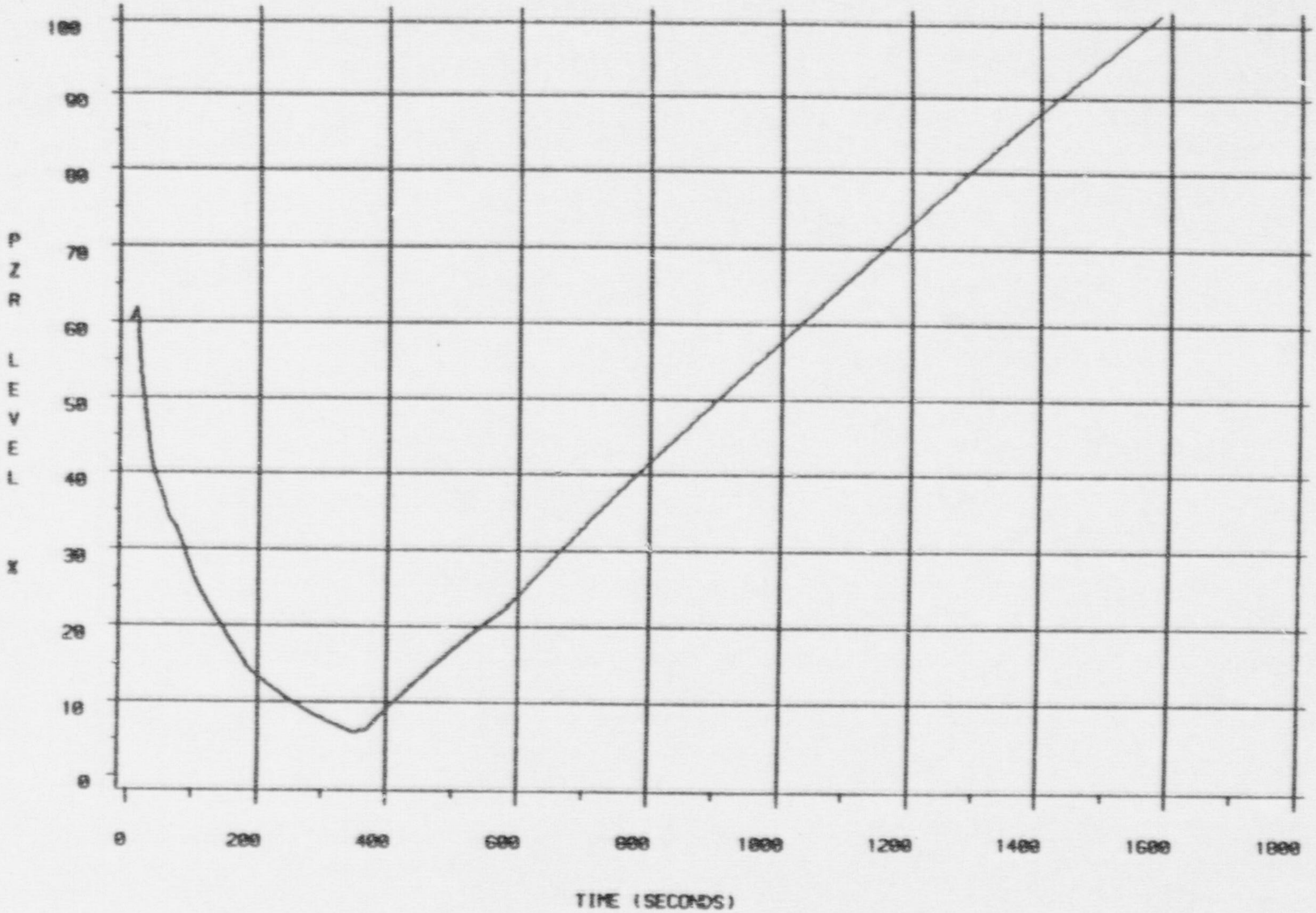
CNS FEEDWATER LINE BREAK WITH DEGRADED AUX FDW



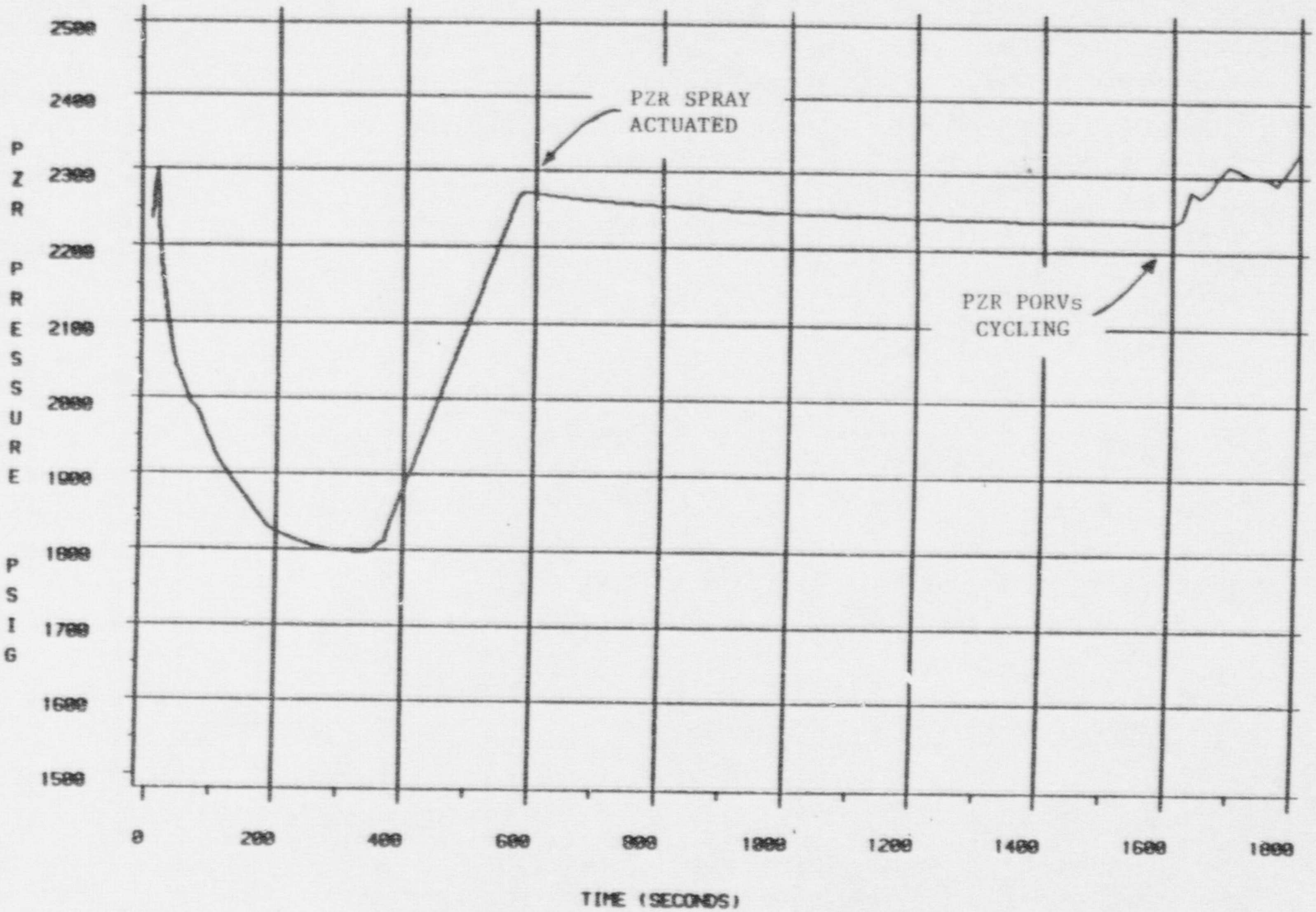
CNS FEEDWATER LINE BREAK WITH DEGRADED AUX FDW



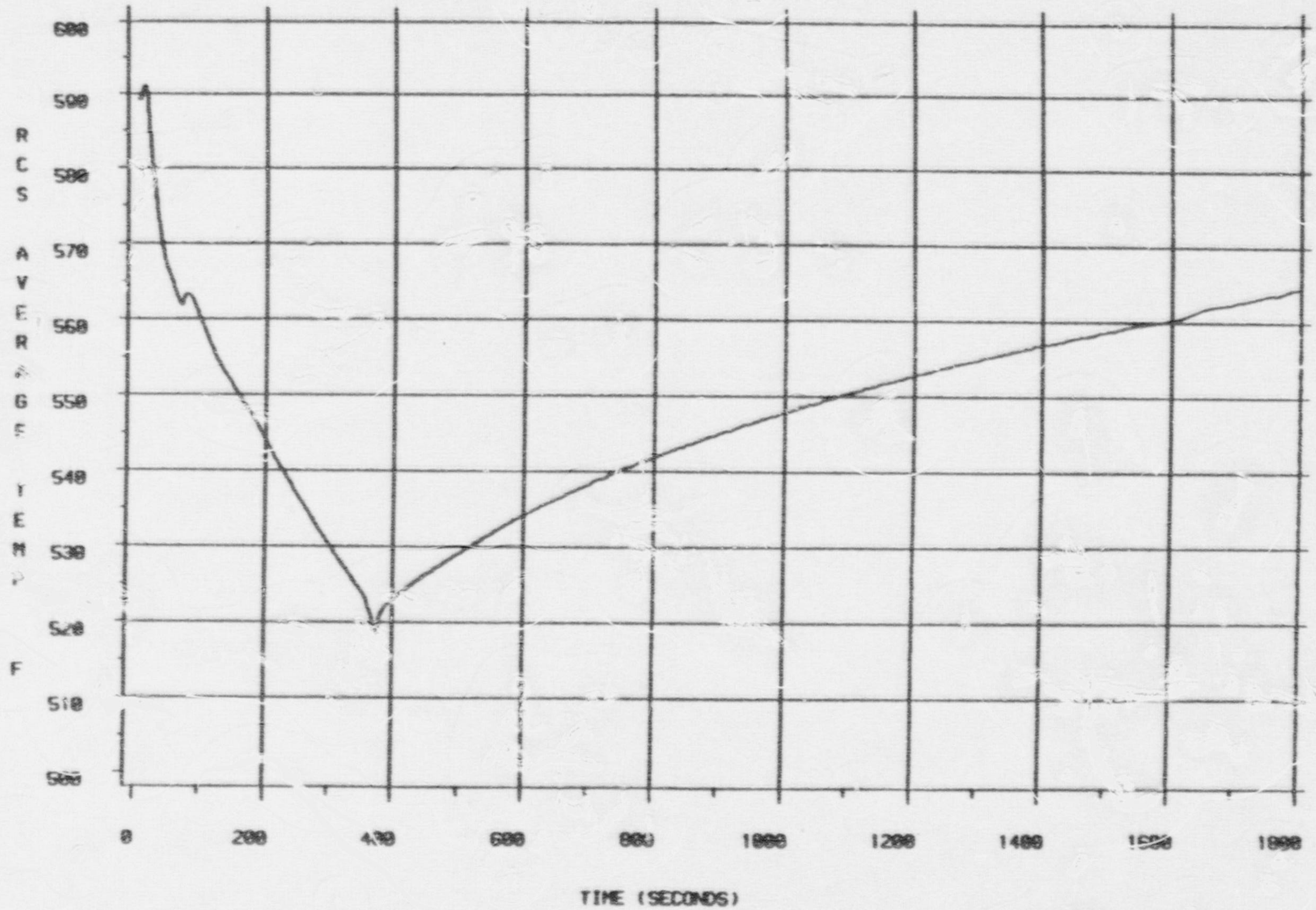
CNS FEEDWATER LINE BREAK WITH DEGRADED AUX FDW



CNS FEEDWATER LINE BREAK WITH DEGRADED AUX FDW



CNS FEEDWATER LINE BREAK WITH DEGRADED AUX FDW



CONCLUSION

- UNIT STATUS AT 30 MINUTES IS NOT SERIOUSLY DEGRADED
    - RCS IS WATER SOLID WITH CHARGING AND SAFETY INJECTION PUMPS ON
    - RCS T-AVE IS 565^{O}F (557^{O}F IS NORMAL POST-TRIP) AND IS INCREASING ~1.5^{O}F/MIN
    - AUX FDW REQUIREMENT TO MATCH DECAY HEAT AND REACTOR COOLANT PUMP HEAT IS 484 GPM
- OPERATOR INTERVENTION AT 30 MINUTES
    - REALIGN AUX FDW
    - MINIMUM AUX FDW FLOW AVAILABLE IS 499 GPM
    - TERMINATE SAFETY INJECTION
- UNIT RECOVERY AND STABLIZATION WITH OPERATOR ACTION AT 30 MINUTES WAS ASSURED

INSTRUMENT ISOLATIONS

NUMBER OF CASES: 6

VALVE TYPES:	DRAGON INSTRUMENT ISOLATION VALVE:	5
	DRAGON INSTRUMENT MANIFOLD VALVE:	1

ROOT CAUSES:	OTHER:	4
	INSTALLATION DEFICIENCY:	1
	PERSONNEL ERROR:	1

NUMBER REPORTABLE: 4
NUMBER NOT REPORTABLE: 2

INSTRUMENT ISOLATIONS

DATE: 12-6-86

VALVE TYPE: DRAGON

INSTRUMENT AFFECTED: 1NSPT5050, CONTAINMENT PRESSURE NARROW RANGE

METHOD OF DISCOVERY: OPERATIONS NOTED ABNORMAL PRESSURE READING ON PRESSURE GAUGE. IAE DISCOVERED INSTRUMENT ISOLATION VALVE CLOSED WHILE INVESTIGATING PRESSURE READING.

ROOT CAUSE: OTHER

REPORTABLE: YES

CORRECTIVE ACTION: RETURNED TO OPERABLE STATUS

INSTRUMENT ISOLATIONS

DATE: 4-24-87

VALVE TYPE: DRAGON

INSTRUMENT AFFECTED: 1NSPT5040, CONTAINMENT PRESSURE NARROW RANGE

METHOD OF DISCOVERY: OPERATIONS NOTED ABNORMAL PRESSURE READING ON PRESSURE GAUGE. IAE DISCOVERED INSTRUMENT ISOLATION CLOSED AND VALVE HANDLE LOOSE (SLIPPING) ON VALVE STEM.

ROOT CAUSE: INSTALLATION DEFICIENCY

REPORTABLE: YES

CORRECTIVE ACTION: ALL CONTAINMENT PRESSURE (BOTH UNITS) ISOLATION VAVLE HANDLES WERE INSPECTED (3 OF ≈ 30 WERE LOOSE) AND CORRECTED. THE CALIBRATION PROCEDURES WERE REVISED TO POSITIVELY VERIFY RETURN TO SERVICE. ALL IAE AND PERFORMANCE TECHNICIANS WERE TRAINED ON THIS INCIDENT AND THE DRAGON VALVE HANDLE PROBLEM.

INSTRUMENT ISOLATIONS

DATE: 7-7-87

VALVE TYPE: DRAGON

INSTRUMENT AFFECTED: 1CAPS5131 PRESSURE SWITCH TO VALVE OPERATOR 1CA46B

METHOD OF DISCOVERY: AFTER A MANUAL REACTOR TRIP, TURBINE DRIVEN AUXILIARY FEEDPUMP AUTO STARTED. VALVE 1CA46B AUTO CLOSED UNEXPECTEDLY. OPS INITIATED A WORK REQUEST TO INVESTIGATE/REPAIR. IAE FOUND 1CAPS5131 ISOLATION VALVE CLOSED.

ROOT CAUSE: OTHER

REPORTABLE: YES

CORRECTIVE ACTION:
- CALIBRATION CHECK OF 1CAPS5131/RETURNED TO SERVICE
- ADD THIS AND OTHER SIMILAR ISOLATION VALVES TO THE INSTRUMENT STARTUP CHECKLIST FOR THE AUXILIARY FEEDWATER SYSTEM. (BOTH UNITS)
- REVIEWED ALL CA PRESSURE SWITCH CALIBRATION DOCUMENTATION

INSTRUMENT ISOLATIONS

DATE: 9-11-87

VALVE TYPE: DRAGON MANIFOLD

INSTRUMENT AFFECTED: 1CAFT5090 AUXILIARY FEEDWATER FLOW TRANSMITTER

ROOT CAUSE: PERSONNEL ERROR

REPORTABLE: NO

CORRECTIVE ACTION: THE ERROR WAS IDENTIFIED BY IAE ENGINEERING STAFF. THE PERSONNEL INVOLVED WERE INTERVIEWED AND DISCIPLINARY NOTICES WERE ISSUED TO TWO IAE TECHNICIANS FOR FAILURE TO FOLLOW PROCEDURE

INSTRUMENT ISOLATIONS

DATE: 9-15-87

VALVE TYPE: DRAGON

INSTRUMENT AFFECTED: 2NSPT5380 CONTAINMENT PRESSURE WIDE RANGE

METHOD OF DISCOVERY: IAE CONDUCTING SURVEILLANCE PROCEDURE PER SCHEDULE FOUND VALVE CLOSED.

ROOT CAUSE: OTHER

REPORTABLE: YES

CORRECTIVE ACTION: IAE ISSUED WORK REQUESTS TO VERIFY INSTRUMENT VALVE POSITIONS. UNIT II PROCEDURES WERE STARTED ON 9-16-87 AND COMPLETED ON 9-30-87 ON THE FOLLOWING SYSTEMS: CA, CF, FW, KC, NC, ND, NI, NS, NV, NW, RN, AND SM. UNIT I SYSTEMS WERE STARTED PRIOR TO THE REFUELING OUTAGE WITH CA AND NS COMPLETED ON 9-30-87. ALL UNIT I ABOVE LISTED SYSTEMS WILL BE VERIFIED PRIOR TO RESTART AT OUTAGE COMPLETION.

INSTRUMENT ISOLATIONS

DATE: 9-21-87

VALVE TYPE: DRAGON

INSTRUMENT AFFECTED: 2KCFT5810

METHOD OF DISCOVERY: IAE CONDUCTING SYSTEM STATUS REVIEW

ROOT CAUSE: OTHER

REPORTABLE: NO - NON SAFETY RELATED; NO SAFETY FUNCTION

CORRECTIVE ACTION: NONE

CATAWBA NUCLEAR STATION
Corrective Actions for Inadvertent Instrument Valve Mispositioning

I. Prevention Of Inadvertent Mispositioning Of Instrument Valves

- Training of IAE personnel on instrument valve operation. Completed Sept. 24, 1987.
- Training of Performance personnel on instrument valve operation. Completed Nov. 4, 1987.

II. Efforts Implemented To Detect Mispositioned Instrument Valves

- Upgrade instrument valve checklists to verify proper valve position after an outage. Complete Dec. 1, 1987
- Use of performance test procedures to identify questionable as-found instrument valve position. Completed Nov. 1, 1987
- Procedure upgrade for returning instruments to service. Completed Nov. 2, 1987

III. Programmatic Efforts Underway To Ensure Surveillance/Calibration Of Technical Specification / Safety-Related Instruments

- Calibration program review and maintenance procedure revisions.
  Complete Dec. 31, 1987

- Nuclear Station Modification review process.
  Complete April 1988