DESCRIPTION AND RESOLUTION OF ISSUES REGARDING THE DECEMBER 26, 1985 REACTOR TRIP

Prepared by the Staff
of the

SACRAMENTO MUNICIPAL UTILITY DISTRICT

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DESCRIPTION AND RESOLUTION OF ISSUES REGARDING
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I. INTRODUCTION

1. Overview of Event

Early in the morning of December 26, 1985, while operating at a steady 75%FP power, the reactor tripped on high pressure as a result of loss of main feedwater caused by an unexpected loss of DC power within the plant's Integrated Control System (ICS). The loss of ICS power caused the main feedwater pumps to reduce to minimum speed, causing Main Feedwater pressure to decrease which autostarted both AFW Pumps and permitted AFW control valves to operate. Concurrently, all other control devices receiving inputs from the ICS received a "zero" volts signal which is interpreted as a "50% demand" command since the ICS operates on a -10 to +10 vdc range corresponding to 0 to 100% demand. This resulted in several steam valves repositioning to the corresponding 50% open positions, and likewise, the Auxiliary Feedwater Control Valves opening to a mid-position. The effect upon the Nuclear Steam Supply system (NSS) was to provide a significant rate of heat removal via both steam discharge and the addition of cold Auxiliary Feedwater to the Once Through Steam Generators (OTSGs). These effects combined to cause a rapid reduction in the temperature and pressure in the Reactor Coolant System (RCS). The pressure reduction reached the point where the engineered safeguards equipment, the Safety Features Actuation System (SFAS), actuated bringing into service the High Pressure Injection (HPI) System (which had previously been operating on operator demand), the standby emergency Diesel Generators, the AFW bypass flow control valves, and isolation of the Reactor Building as it would in the event of a Loss of Coolant accident. The high rate of reactor coolant injection from HPI was effective in establishing the reactor coolant inventory needed at the reduced temperature caused by the overcooling. A detailed "Sequence of Events" is provided as Attachment 1.

During the event, all safeguards and related equipment performed as designed, although post-event analyses has identified enhancements to non-safeguards related equipment and procedures. The event did progress sufficiently to constitute an "Overcooling" event as a result of delays and complications in closing the open steam and feedwater control valves. Reactor Coolant Pumps (RCP's) remained online through the event.

The operating crew on duty had been augmented by an extra Senior Reactor Operator who was also a qualified Shift Supervisor. Early in the event he left the Control Room (this was appropriate, as he was not assigned a specific duty station) and assisted in the work of post-trip recovery and equipment operation. Upon returning to the

I. 1. (Continued)

Control Room, after nearly a half hour of vigorous exertion, he became "lightheaded" and had to lie down. This was diagnosed as hyperventilation and followup examinations have not identified any reason for concern as to the individual's fitness for duty.

During post-trip recovery activities, the suction to the Makeup Pump was inadvertently isolated. This led to the rapid destruction of the pump seals and the release of approximately twelve-hundred gallons of reactor coolant makeup water within the Auxiliary Building. The associated release of radioactive gases contained in the coolant constituted the primary source of radioactivity released during the event. An Unusual Event (as established by the facility Emergency Plan) was declared as a result of the SFAS actuation, which initiated the notification of the various emergency response agencies.

This event was defined by SMUD management to be unique and to require special study, analysis, and troubleshooting prior to authorizing a plant restart. This report details these studies and analyses, develops the lessons learned, provides the root causes, and describes the implementation of actions and changes to preclude recurrence of this event, while enhancing the ability to mitigate future transients at Rancho Seco.

2. SMUD Response to the Transient

In addition to collecting the various data available following an event of this type, the Plant Manager immediately instituted a deliberate and detailed process to govern the investigations and troubleshooting necessary to establish the root cause(s) of the event. This program was intended to systematically prepare the facility for return to power operation. Details of this program were defined and published as shown in Attachment 2.

In anticipation of industry interest in an overcooling event such as the one experienced, particularly at those facilities using an ICS, the District invited the Institute of Nuclear Power Operations (INPO), the Electric Power Research Institute (EPRI/NSAC), and the B&W Transient Assessment Team to assist with investigating the event and reporting it to the Utility industry. The INPO team developed a Significant Event Report (SER #3-86.) which was issued to the Industry on January 2, 1986.

EPRI/NSAC assisted in the evaluation of the Pressurized Thermal Shock (PTS) related questions and analysis as they pertain to the reactor vessel and potential effects upon its service life. These results are discussed in a following section, which confirms no Reactor Coolant System component degradation attributable to the transient.

I. 2. (Continued)

The B&W TAP Team assisted in compiling the sequence of events and in defining issues and areas which needed further investigation. The team played an important role in developing the troubleshooting plans for the ICS and the Maintenance Instructions (MIs) which implemented those plans. They also assisted in evaluating the procedural adequacy and operator response to the event.

3. NRC Incident Investigation Team (IIT)

Late on December 30, 1985, the District was notified that the NRC had elected to upgrade their involvement in the event investigations by replacing the augmented Region V investigative effort with an IIT charged with determining the:

1. Sequence of Events

2. Troubleshooting Action Plans
   to Determine Root Cause, and

3. Major Issues surrounding the event.

Interviews were held "on the Record" with the operating personnel involved in the event, its mitigation, and the District's investigative efforts.

An early effort established by SMUD, with concurrence by the NRC, was to develop the "Quarantined Equipment List", of that equipment whose failure influenced the event or operator efforts to stabilize the plant.

Details of the requirements for the process of troubleshooting and reporting of findings, relevant to the Quarantined Equipment, were developed and published in revisions to the District's investigation efforts. These were established by Attachment 2.

Once the IIT had completed their interviews, and agreed with the content of the developed Troubleshooting Action Plans, they left the site to write their report and await the results of the troubleshooting efforts. The Rancho Seco staff provided these results, as they became available, directly to the IIT and the NRC Site Resident Inspector.

4. Transient Analysis Organization

Attachment 2 established the organization and process for insuring that all available and potentially useful material necessary to gain a full understanding of the events of December 26th was obtained and utilized. It established an Action List which collected the developing issues and oversaw their resolution.

The implemented troubleshooting program identified four major milestones needed to resolve an issue. These are as follows:

I. 4. 1. Troubleshooting Action Plan

Following a description of the question, issue, or problem being investigated, a summary of information supporting the probable cause is developed. Included is a review of the components maintenance, surveillance testing, and modification history. From this body of information, the potential root cause(s) are developed and an outline of the troubleshooting plans to prove/disprove each is presented.

2. Engineering Report

This is a report of the results of the troubleshooting efforts and provides the conclusions and justification of the identified root cause.

3. Repair Action Plan

Once concurrence is obtained that the root cause(s) have been identified, then those steps required to repair, change, or modify the apparatus or procedure for return to service are developed. This step is imposed to insure that repairs/changes are properly coordinated and that troubleshooting is complete and sufficient to allow repair.

4. Action Item Closure Report

This report consolidates all of the developed information from the above efforts and completes the explanations and justification of root cause(s) of the item. It also provides for recommendations which will be useful in the development of lessons learned and programmatic improvements to guide in achieving excellence in operations and management while precluding recurrence.

Note, root cause, as used in this section is defined as the direct cause of the failure, malfunction, or discrepancy. It is not necessarily the programmatic or underlying cause that allowed the failure, malfunction, or discrepancy to occur. That "root cause" is determined by the Rancho Seco Incident Analysis Group. Section VII of this report discusses the Root Causes developed by the IAG.

For those items placed upon the SMUD NRC/IIT Quarantined Equipment List, or if so designated on the Action List (Attachment 3), the entire four phase program described above was implemented. Several items were added to the Action List for tracking and management which did not require the full program. In those cases, a Closure Report alone was required and used to document the scope of the analysis, or investigation, and to report the conclusions which resulted from analysis of the item.

I. 5. B&W Owners Group Regulatory Response Group (RRG)

Since the ICS is generic to the B&W designed NSS System, the NRC addressed its questions, which were of potentially generic interest to all B&W plants, to the RRG. This group met and/or communicated with the NRC, and in general, discussed issues which are beyond the scope of this report.

6. Region V Confirmatory Action Letter

On December 26, 1985, the NRC Region V Administrator sent to the District a Confirmatory Action Letter requiring that Root Cause Analysis of the event be done, and that the NRC be briefed on the root cause and provide the justification as to why the facility is ready to resume power operations. This same letter established the hold of repair work on equipment which malfunctioned pending evaluation by SMUD and NRC inspection teams. See Attachment 4.

NRC Region V has been briefed on the results of the Root Cause investigations and the justification for resumption of power operation.

II. SEQUENCE OF EVENTS

At the time of the event, the Bailey 855 Process Computer was out of service with problems in it's disk drive control circuits. As a result, it was unavailable for data collection and this source of sequence of events and plant process data was therefore unavailable. The IDADS (Interim Data Aquisition Display System) computer system was available and performed well, providing a comprehensive view of the sequence of events and plant conditions through the event. One consideration was that the analog data points that IDADS obtains from the Bailey 855 were not viable during this event, although much of the information was available from charts and recorders.

Extensive interviews with the operators supplemented their usual reports prepared following a reactor trip. While these reports do not have the high resolution time discrimination of computer data, they aid in gaining a full understanding of the actions leading to identifiable events and may provide the reasons for actions taken.

Attachment 1 is the Sequence of Events as developed by the Rancho Seco staff. The Sequence of Events issued by the NRC/IIT followed that developed by SMUD staff, but does include several additional observations and comments from the perspective of the NRC/IIT.

The INPO SER includes a summary of the Sequence of Events which did not benefit from the longer preparation and study period available to the Rancho Seco staff. This was appropriate given the early publication date of the SER.

Attachment 5 is the set of curves developed from the post-trip data base. These are useful for gaining a full understanding of the plant parameters surrounding the Sequence of Events.

III. ACTION LIST

In the very active period following the event, during which the issues were being identified and troubleshooting plans developed, daily meetings of those persons responsible for resolving the issues were held. The Action List served as the means of tracking the progress on a specific issue while providing a forum to consider the significance of the issues and need to expand the scope, or to address additional issues and concerns.

As a living document, the Action List went through numerous revisions. As a prerequisite to startup, or return to power operations, those items identified as "SU", i.e. "Startup Required", have been deemed to require "closing". Attachment 4 provides the Action List, current as of the date of issue of this report.

A discussion of each startup required item, or section on the Action List, is included in a following section of this report. Those discussions summarize the troubleshooting done, the identification of root causes, and the basis for determination of the effect upon return to power operations. A copy of the Action List is provided as Attachment 3.

IV. QUARANTINED EQUIPMENT LIST

The Region V Confirmatory Action Letter, Attachment 5, established the requirement to "... hold in abeyance any repair work planned on equipment that malfunctioned during the incident ...". In response to this requirement administrative controls were immediately placed upon maintenance and operation of that equipment which might in any way have been associated with the event. These broad boundaries were refined by the SMUD NRC/IIT to apply to the following equipment:

1. Integrated Control System (ICS)
2. Power to the ICS DC Power Supplies and Monitor
3. Main Steam Pressure Relief Valve, PSV-20544
4. Auxiliary Feedwater Flow Control Valves, FV-20527 and FV-20528.
5. Auxiliary Feedwater Manual Isolation Valves, FWS-063 and FWS-064.

For each of these items, except PSV-20544, detailed troubleshooting plans were developed and presented to the NRC/IIT. Discussion with the Rancho Seco staff resulted in concurrence to issue those Troubleshooting Action Plans for implementation via a detailed Maintenance Instruction (MI). These MIs were provided to the NRC Resident Inspector to allow for NRC monitoring and observation of the troubleshooting activities.

IV. QUARANTINED EQUIPMENT LIST (Continued)

In the case of PSV-20544 engineering evaluation concluded that the valve had operated properly. Since the study was not complete prior to the IIT leaving the site, it was agreed that the valve would stay on the list pending submittal of a "Closure Report" justifying its removal from the list. This was subsequently done.

The effect of having an item on the Quarantined Equipment List was that it's operation was restricted to safety needs or activities specified under the direction of the approved Troubleshooting Action Plan. No servicing, clearances, configuration changes, surveillance or other work was allowed. Work requests which might effect this equipment were held by scheduling. The equipment itself was identified by appropriate barriers and signs.

As of February 14, 1986, all plant equipment had been removed from quarantine, with three components removed and identified for subsequent analysis by an independent laboratory. These are the ICS Power Supply Monitor and the ICS Power Switches "S1 and S2". Replacement components meeting operational requirements have been installed in place of these removed items.

V. RESOLUTION OF ACTION LIST

This section will review each of the fourteen sections of the Action List. It will discuss the troubleshooting done, identify root causes found, establish the status of the item, and will evaluate its effect upon the return to power operation.

1. Post Trip Report (AP.28)

   There are numerous individual items identified in this section. Each will become an element of the final Trip Report, a document which compiles the technical basis for the event and establishes the historical record for its understanding. It is not intended to replicate the Trip Report herein; however, the items are briefly discussed below.

   1. Main Steam Line Analysis

      During the transient, OTSG A was overfilled with auxiliary feedwater. Approximately 11,000 gallons spilled into the "A" main steam line. Because of the large difference between the temperature of the steam line and auxiliary feedwater, there was a concern that high thermal stresses may have occurred. In addition, consideration was that the flooding may have caused "water hammer". Water hammer noises were heard in the Turbine Building sometime after the flooding had occurred..

      To resolve the thermal stress concern, a stress evaluation of the main steam line was performed using an augmented Class 2/3

V. 1. 1. (Continued)

fatigue analysis. This evaluation considered the loads imposed by thermal stratification caused by cold water in bottom half of the steam line and thermal gradient stresses caused by the thermal transient. Fatigue usage factors were calculated for two cycles of this event, plus operating basis earthquake (OBE), design basis earthquake (DBE) and 1000 cycles of pressure/temperature loads. The results of the stress evaluation indicate that the fatigue usage factors for the most critical components in the steam line are all below 0.3. compared to an allowable limit of 1.0.

To resolve the water hammer concerns, walkdowns of the "A" main steam line and the "A" main steam line bypass to the condenser were performed. A walkdown similar to those assembled for previous I&E Bulletin 79-14 walkdowns, packages consisting of check lists and pipe support drawings were assembled for each walkdown. Using the pipe support drawing as a reference, a check of the support configuration was performed. This as found configuration was then compared to the as found configuration of the I&E Bulletin 79-14 walkdowns. No visible evidence of water hammer was found on the "A" main steam line or the "A" main steamline bypass to the condenser as a result of the configuration comparison.

One anomaly was noted where the bypass line penetrates the Auxiliary Building wall. At this location, the sheet metal flashing covering the wall penetration was partially pulled from the wall. Since no apparent damage due to water hammer was identified, the pulled flashing may be due to thermal expansion of the bypass line. To ensure that water hammer did not occur in the area of the flashing, a detail examination of the welds and piping of the bypass line is being performed. This examination includes approximately fifteen feet of pipe in each direction from the penetration and encompasses several changes in piping direction. Any postulated water hammer in this area would have been limited to this run of pipe.

The results of the stress evaluations and the walkdowns indicate that the "A" main steam line and the "A" main steam bypass line to the condenser did not experience unacceptable stresses or water hammer. The follow-up verification described above for the bypass line will be completed prior to resumption of power operations.

Evaluation of a separate potential effect of water in the steam main, that water may have been injected into the AFW Pump Steam Turbine was evaluated. The result was that no water was injected. This was due to the fact that less than 13,000 gallons of water was spilled into the A-Steam line, and over 19,000 gallons is required to reach the line to the AFW steam turbine. The turbine continued to operate well throughout the event.

V. 1. 2. Minimum Pressurizer Level

Following a typical reactor trip, the RCS average temperature drops from 582°F to 545°F. This reduction in temperature corresponds to an increase in density of the cooling water which requires additional volume to keep the RCS flow path solid with subcooled water. The required mass of water is drawn from the Pressurizer until the level is reduced below normal, and then the Pressurizer Level Control Valve opens to provide makeup water from the Makeup/HPI System.

If the operator determines the rate of Pressurizer level decrease is greater than desired, other HPI Injection nozzles are opened, providing additional flow paths into the RCS. Additionally , HPI pumps are started to further augment the supply of RCS Makeup.

During the December 26 event, all of the preceeding occurred, as the operator observed the effects of overcooling being caused by the large quantities of Auxiliary Feedwater being provided to the steam generators. Normal post-trip reduction in Pressurizer Level was exceeded, and the level decreased to below the lower level taps on the Pressurizer. Although the level instruments indicate zero level, approximately 260 cubic feet of water remains in the pressurizer, including that in the surge line which connects the Pressurizer to the RCS piping. To obtain a full understanding of the actual RCS volume during the period when the Pressurizer was indicating offscale low, mass balance calculations were performed. These calculations included the effects of Makeup/HPI flow, letdown flow, RCP seal injection flow, and the temperature and pressure within the RCS. The results of this calculation show that for a few minutes the RCS may have withdrawn the water from the pressurizer and surge line which would indicate that steam voids may have formed for several minutes during the overcooling.

The degree of subcooling is important, for if subcooling is lost then the situation is changed as the Emergency Operating Procedures treat the loss of subcooling prior to any overcooling or other symptoms. During the December 26 event, approximately 80°F of subcooling existed during the period of interest, and this margin was being increased through the duration of the event. This fact alone demonstrates that the core was properly and effectively cooled during the event.

If at any time during this event the RCPs would have been lost or shut off, the steam voids in the system would not have been large enough to have impacted the natural circulation cooling process which would have been instituted. Previous operator training would have been appropriate to provide core cooling and re-establish pressurizer control via the injection of HPI water and heat removal via the OTSGs.

V. 1. 2. (Continued)

In summary, the fact that pressurizer level indicated offscale low, for approximately five minutes during the event, did not cause the operator difficulty or confound the recovery and control efforts which were in progress.

3. Pressurizer Heater Operation

As a result of the overcooling, water in the RCS contracted causing water from the Pressurizer to be drawn into the RCS. At an indicated level of 40 inches, a level switch interrupts the supply of electrical power to the Pressurizer Heaters. This is necessary, because the heater elements would be damaged if operated while not submerged in water. Since it was observed that Pressurizer Level went well below the heaters, it was prudent to verify that the protection device had functioned properly.

No direct recording or monitoring of this protective device exists. Thus the condition of the heater elements themselves was the only way to verify functioning of the protective interlock. Measurement of the individual heater element resistances showed no significant change from their historical values; thus, it was concluded that the interlock functioned as intended and that the heaters remain serviceable. Note that following the event, the plant was stabilized and no problems were experienced controlling RCS pressure using the Pressurizer Heaters.

4. Control Room Instruments Affected By Loss of ICS DC Power

A complete list of the Control Room instrumentation affected by ICS DC Power is provided as Attachment 7. This list was developed during the troubleshooting effort following a comprehensive series of tests and observations. A drawing study provided a similar list.

It had not been generally known that the Auxiliary Steam Reducing Station received its demand from a controller powered from the ICS. Since auxiliary steam is needed following a reactor trip for plant services, such as Condenser Air Ejectors and Turbine shaft seals, it was determined to retain this feature and manually throttle the supply when desired. A review of the effect this station had upon the overcooling showed that it was small, if any.

The Auxiliary Steam System normally obtains its steam from the main steam system following a turbine trip. In this case, with the demand set to 50%, excessive steam flow occurred which caused one of the two Auxiliary Steam relief valves to open until the main steam pressure had decreased and/or the load on Auxiliary Steam, (condenser hogging ejectors, turbine seals,

V. 1. 4. (Continued)

etc.) increased. This control station is not being modified or repowered pending a full study. In any case, the impact on RCS cooldown is limited.

Other than losing ICS control of the normal process devices (AFW Valves, Atmospheric Dump Valves (ADVs), Turbine Bypass Valves (TBVs), Main Feedwater (MFW) Pumps, Feedwater (FW) Values, etc.), it was also noted that the "Generator Frequency Error Indicator" and "Main Feedwater Loop Flow Recorders" also provide erroneous indications. These are acceptable conditions since:

a. The Reactor/Turbine trip terminates electrical generation, hence Frequency Error is meaningless.

b. Loss of ICS Power causes the Main FW Pumps to go to minimum speed, hence, Main Feedwater flow to the OTSGs has stopped. Alternate Main and Startup Feedwater flowmeters are located on the same panel which are not affected by the ICS, or its power supplies. In addition, the flow indicators move to the 50% flow position and any operator action based on this excessive indicated post-trip flow would be to trip MFW pumps, an appropriate action required by the "overcooling" Emergency Operating Procedure. Plant training materials are being revised to include this information.

5. Investigation of Report of Smoke Prior to Reactor Trip

This issue concerns Operators' reports of the smell of smoke in the controlled area shortly before the loss of ICS power. The purpose of the investigation was to clarify the Operators' observations and to determine the relationship between those observations and the loss of ICS Power, if any.

Although the scope of this investigation did not determine the source of the smelled smoke, it is concluded that this was not a factor contributing to the loss of ICS Power. Room 211, where the smoke was sensed, is isolated from any significant electrical power equipment. Two walls and a corridor separate the electrical equipment rooms (Rm 214 and Rm. 216) from the ventilation equipment. Furthermore, the electrical equipment rooms have separate heating and ventilation systems from the radwaste areas, which includes Room 211. Smoke sensed in Room 211 would not have originated in the electrical equipment rooms. The Operators said the smell was like that from wood or paper burning, but that it did not smell like either electrical equipment or insulation burning. They also noted that the smell went away as soon as they exited Room 211. Investigation of the room and adjacent areas showed no evidence of smoke or fire. Thus, the report of smelled smoke appears to be unrelated to the loss of the ICS power.

6. Primary to Secondary Leak Investigation

Prior to the event, the Condenser Air Ejector radiation monitor was indicating approximately 400 cpm of activity. This is a low value, however, it is above background and may be indicative of a very small primary to secondary leak.

During the event, review of the main steam line radiation monitors did not indicate any changes in activity which would be expected from a significant primary to secondary leak. Following the event it was deemed prudent to monitor the OTSGs, while in wet layup, to evaluate and quantify identifiable leakage. These results were inconclusive as some minute traces of primary radioisotopes were found, so a more sensitive technique was applied.

Helium leak detection did identify two very small leaks in the B-OTSG. Investigations, using eddy current techniques, are being used to characterize the leaks and to provide confidence that any incipient leaks in either OTSG are found. The size and number of leaks identified are commensurate with the amount of activity observed in the secondary plant prior to December 26.

7. Operations/Security Interface During Transients

During the event, one of the operators lost his security badge. He notified Security and they escorted him until a replacement badge was issued.

During the event the security area access card reading time at the Control Room door began to impact the timely dispatch, and reporting back, of operators as they established manual control of the various equipment outside the Control Room. The Shift Supervisor requested Security post a guard at the door, thus allowing it to remain open for the duration of the event. A security officer was provided.

These hinderences did not significantly affect the outcome of the event. Yet it is apparent that improvements could be made to ensure that Security is a part of event response and that effects of anticipatable occurrences, like misplaced badges, could be minimized by appropriate plans and procedures. Enhancements in the Rancho Seco Security Plan and Procedures are being developed which will improve the support provided by each organization to insure optimum response to plant emergencies and transients.

8. Main Steam Line Failure Logic (MSLFL)

Post trip review of the IDADS alarm printout showed that OTSG pressure was approximately 370 psig when MSLFL alarmed as actuated. The setpoint was 435 psig. In the ensuing analysis

V. 1. 8. (Continued)

and troubleshooting, it was found that the MSLFL is monitored by IDADS once per minute. During the period of interest, OTSG pressure was decreasing approximately 70 psig per minute. The conclusion is that the MSLFL operated as intended. The IDADS sample frequency for the parameter is going to be increased to improve its resolution.

Separate analysis determined that a higher MSLFL setpoint is desirable to insure that the condensate pumps will not begin feeding a low pressure OTSG. Engineering has developed, and Operations implemented, a new setpoint of 575 psig which will accomplish this purpose. It is important to note that MSLFL operates independently of the ICS and overrides any ICS demand signal, to the Main and Startup Feedwater Flow Control Valves, directly causeing them to close. MSLFL is independent of ICS and ICS power.

9. SPDS vs Strip Chart Recorder OTSG Operate Level

Several steps in the plant Emergency Operating Procedures (EOPs) have the operator taking specific actions based upon indicated OTSG Operate Level. An instrument of significant benefit to the operator, and utilized in this event, is the SPDS (Safety Parameter Display System). It is capable of an integrated presentation of several parameters simultaneously and allows the operator to observe trends in pressure and temperature, and anticipate developing conditions.

As a part of this SPDS display, a numerical value representing OTSG Operate Level is provided. Several of the operators reported that these values did not exceed 97% and 93% on the A-OTSG and B-OTSG respectively, yet other Control Room indications were that Operate Levels were in excess of 100%.

The SPDS uses the same uncompensated level source data as do the other Control Room instruments. There is a difference in the way temperature compensation occurs, the SPDS calculating it internally based upon the OTSG pressure and a steam tables algorithm. Simulations over the range of interest, using the algorithm showed good correlation with the other Control Room instruments, although the SPDS Operate Level tends to be 1 to 2% lower than comparable strip chart recorders and the computer.

It was concluded that, while the SPDS was likely indicating less than 100% Operate Level, when the level was actually higher, the maximum -2% error would not adversely impact operator actions directed by the EOPs since action is required upon exceeding 95%. The operator observations of SPDS Operate Level, during the event, were not benchmarked against other indications of level, with the result that no discrepancy could be verified. This study suggests that even with actual levels in the OTSGs

V. 1. 9. (Continued)

greater than 100%, the SPDS will indicate a value just under 100%. As discussed above, the operators were aware that OTSG Operate level was greater than 100%, hence the observation investigated here. The consequence of the slightly low reading does not adversely impact the proper and prompt implementation of the EOPs or mitigating action.

10. Transient Compared to USAR Design Basis

Loss of ICS Power is not a specific event considered within the plant design basis as reported in the facility Updated Safety Analysis Report (USAR). The plant design basis considers "worst case" scenarios for non-safety equipment.

Should any of the plant safety parameters be exceeded, the Reactor Protection System shuts down the reactor independently of the ICS.

Should the RCS pressure boundary integrity be challenged, or the containment pressure be excessive, the SFAS/ECCS (Emergency Core Cooling System) operates independently of the ICS to insure core integrity and isolation from the environment.

For comparison, the analyzed event Main Steam Line Failure, begins with a large uncontrolled steam release followed by a rapid cooldown. The pertinent parameters and results are:

USAR Chapter 14
Main Steam Line Failure Analysis

Assumptions and Results	12/26/85 Event
1% defective fuel rods.	<<1% defective fuel rods.
Complete severance of 36" Main Steam pipe. Duration of release, 4.5 hours.	6 code safety valves open for <1 minute. 2 ADVs and 4 TBVs 50% open for <10 minutes.
Steam line break between Reactor Building and a turbine stop valve.	Open valves located in the same portion of the main steam system.
490°F reached in 43 seconds.	490°F reached in 6 to 7 minutes.
10 gpm primary to secondary leakage.	<0.001 gpm - no measureable leakage.
Site area dose thyroid 12,700 mRem. Whole body 46 mRem.	Site area thyroid dose 0 mRem. Whole body 0.02 mRem.

V. 1. 10. (Continued)

A consequence of such a rapid cooldown is the likelihood of exceeding the Technical Specification limit restricting normal heatups/cooldowns to 100°F/hr. That specification was exceeded during the December 26 event. A full discussion of the significance of having exceeded this limit, and a related phenomenon, Pressurized Thermal Shock (PTS) is included in a following section. To summarize here, there was not excessive "usage" imposed on the RCS or the Reactor Vessel by this transient, and the Technical Specification limiting Nil-Ductility Temperature (NDT), was not challenged. The NSSS was not adversely challenged by the December 26 transient.

In conclusion, this event was within the bounds of the analyzed event, and its environmental impact was a tiny fraction of that resulting from the analyzed event.

2. Human Factors Analysis

The District's Human Factors program was applied to this event for the purpose of collecting the Operators' input and observations of the effectiveness of integrating the Control Room, plant equipment, procedures, and operator training and performance.

This process identified 26 items which were worthy of study. Of these, 12 had been previously identified in the Control Room Design review process, and the other 14 were on items beyond the scope of that review. A complete listing of the 26 items is provided as Attachment 6. Disposition of the individual recommendations is being handled by assigning appropriate priority and resources to each. Those which justify immediate resolution have been scheduled for resolution prior to startup.

3. Determine Cause/Corrective Action for ICS Power Failure

1. ICS AC Power Sources

Early studies of the potential causes of the loss of ICS DC power clearly eliminated the independent and redundant supply sources of 120v. AC, single phase, 60 Hertz, power as being involved. These studies looked at the performance of other equipment powered by these sources and how that equipment performed during the event. In addition, the Automatic Bus Transfer switch did not shift, meaning that its normal positioning to the vital bus had not been changed; hence, there had been no interruption in the normal supply from the vital bus. Attachment 8 schematically shows the ICS Power Supplies and Distribution.

V. 3. 2. ICS Equipment Investigations

Post event monitoring of the ICS, and the equipment it operates, shows no problems which would have created the need for tripping switches S1 and S2, through which AC power is fed to the ICS DC Power Supplies. This focused the development of Troubleshooting plans into the operation of the "Power Supply Monitor", the device which initiates the trip feature of the S1 and S2 switches.

A detailed "Troubleshooting Action Plan" was prepared which systematically investigated potential root causes without adversely effecting the "as-found" conditions, or ability to redirect the investigations should new potential root causes or considerations be identified. Implementation of this troubleshooting plan was accomplished, following NRC IIT concurrence, by use of very detailed Maintenance Instructions (MIs).

Early findings determined that the Power Supply Monitor trip setpoint was "drifting." This led to revision of the implementing MIs to better understand the phenomenon. This understanding was greatly aided by the receipt of a new Power Supply Monitor (PSM) which was setup in the laboratory where a full set of performance data could be obtained for comparison with the insitu "as-found" data available from the original Power Supply Monitor.

Significant to the event was the finding that small amounts of resistance, as little as one ohm, in series with the "sensed" voltage into the PSM could cause less than optimum performance of the device, i.e., "drifting," and could cause tripping, similar to that observed on the insitu PSM.

A full investigation into this effect and it's likely impact on an operating ICS, has been completed. The finding of the resistance at the input of the PSM affects performance, and led to troubleshooting the voltages and resistances within the ICS DC distribution cabling. The result was identification of several poorly made factory cable lug crimps on the +24 VDC supply to the PSM bus. One was actually so loose that it fell off of the wire when disconnected from the bus. An additional problem was traced to the PSM +24 VDC circuits, but this was not a contributing root cause as its effect would have been "no-trip, or delayed trip." A third finding was that the S1 and S2 "time delay before tripping" was approximately one-third the intended half second, making the S1 and S2 switches much more sensitive to transient voltage conditions.

V. 3. 2. (Continued)

The root cause for the loss of ICS event has been identified as the loose power connection on the bus ICS Cabinet 1 supplying power to the PSM. The factory bus wiring has subsequently been replaced with current standard wiring. Investigations into the PSM and it's sensitivity to input resistance are continuing, although this will be pursued as a B&W Owners Group issue.

The inservice PSM and the S1/S2 switches are being sent to an independent laboratory for a full analysis.

3. Procedure for Loss of ICS

At the time of the event, a Casualty Procedure to respond to Loss of ICS events did not exist. Procedures for repowering the power supplies were available, and the Emergency Operating Procedures (EOPs) were in place and sufficient to mitigate the consequences of a Loss of ICS. A review was accomplished for the purpose of:

a. Establish the "Failed" condition of equipment controlled by the ICS.
b. The desirability of "repowering" the ICS prior to post trip stabilizing of the NSS.
c. The appropriateness of the EOPs for mitigating and controlling Loss of ICS events.

From this review, it is apparent that changes to the EOPs which provide milestones, e.g., temperatures or pressures, to assure timely operator actions, would go far to preventing recurrence of consequences, such as overcooling, from happening. There is no need for a specific "Loss of ICS" EOP. Information obtained upon effects of repowering the ICS, which will aid the operator in the post-event troubleshooting and return to normal control, has been developed into an appropriate Loss of ICS Power Casualty Procedure.

4. Modifications to the ICS

Modifications have been developed which will enhance plant reliability in two ways by:

- Making ICS DC Power more reliable.

- Enhancing ability to mitigate Loss of ICS Power transients from Control Room.

.1 Within the ICS itself, the rewiring will ensure that system components receive full electrical power, and the PSM is being separately wired directly to the ICS Power Auctioneer Panel ±24 VDC buses it monitors. Failures as occurred on December 26 will be eliminated.

V. 3. 4. 1. (Continued)

New "human engineered" labels are being applieo to switches S1 and S2 to aid the operator in recognizing a tripped condition and in repowering the ICS, should it become necessary. While similar switches are used elsewhere in the plant, they are uniquely position indicated and controller under separate procedures.

An indicator light is being added to the Control Room operator's panel which will show when control is shifted away from the ICS, i.e., it illuminates upon Loss of ICS DC Control Power.

Study showed that there are hundreds of years of successful operating history on this type of PSM, and that this one is the only one to have tripped an ICS as a result of a failure. In addition, this failure was external to the PSM itself. For these reasons, the consideration of a two-out-of-three PSM arrangement has been abandoned. Should the ICS Power be again lost, for whatever reason, the modifications discussed below, in conjunction with the EOPs and Casualty Procedures, are sufficient to mitigate the consequences with no operator action outside the Control Room.

2. External to the ICS, modifications are being installed which will enable the Control Room operator to operate important valves independent of the ICS. This is being done by installing a relay which ICS AC Power energizes through PSM contacts. Upon loss of ICS power, the relay connects a secure battery backed power source into the circuit controlling the necessary valves. The majority of these circuits were installed as a response to Appendix R "Fire Protection" analyses and provide a high quality, separate, and diverse remote control without adversely effecting the Appendix R capability. The specific modifications are:

1. AFW Flow Control Valves

A new Hand/Auto controller, independent of the ICS, is provided for each Aux FW Valve. Located in the Control Room, it is adjacent to the existing AFW Flow indicators. New OTSG Startup Level indicators which are also being added. With the new Hand/Auto station in Auto, ICS controls the valve, except on loss of ICS Power, when the new H/A station automatically controls the valve to the pre-established initial flow rate set at a nominal 280 gpm per OTSG. Placing the H/A station in Hand allows manual control of the flow, independent of the ICS.

V. 3. 4. 2. 1. (Continued)

Still available are the existing controls which will cause the valve to go full open (on operator demand), provide ICS normal controls (when ICS is powered), or use of the local manual operator. Remaining also is the parallel and independent safety grade bypass valve operated by SFAS and controllable from the Control Room.

Attachment 9 provides a schematic view of the modified AFW Control Valve, independent of the ICS.

2. Atmospheric Dump Valves (ADVs)

There are three of these valves on each of the two main steam lines. Normally, only one is in service. This modification operates all inservice valves in parallel. It does so by interposing a selector switch in the Appendix R provided control scheme which is enabled when a loss of ICS Power is sensed. As a result, the valves stay in their normally closed positions, although the operator in the Control Room can select back to ICS at any time. If the ICS is not powered, then the valve will open to its 50% position. By this technique, steam pressure can be cycled about a nominal value. Similar and separate controls are provided at the shutdown panel, external to the Control Room, in addition to local controls, which allow adjusting the ADVs to intermediate positions at the operator's discretion. Attachment 13 provides a schematic view of the modified control circuits, independent of the ICS. Other methods of controlling these valves remain as before this modification.

3. Turbine Bypass Valves (TBVs)

There are two of these valves on each main steam line, and normally, all four are in service. This modification operates all four valves in parallel. It does so by interposing a selector switch in the Appendix R provided control scheme which is enabled when a loss of ICS Power is sensed. As a result, the valves stay in their normally closed positions, although the operator in the Control Room can select back to ICS at any time. If the ICS is not powered, then the valves will open to their 50% position. By this technique, steam pressure can be cycled about a nominal value. Separate controls are provided at the shutdown panel, external to the Control Room, which allow similar operation at the operator's discretion.

V. 3. 4. 2. 3. (Continued)

Attachment 11 provides a schematic view of the modified control circuits, independent of the ICS. Other methods of controlling these valves remain as before this modification.

4. Summary of ICS Modifications

These modifications will provide the operator, from within the Control Room, assurance that upon loss of ICS power the steam control valves will not fail open, that adequate auxiliary feedwater will be provided to the OTSGs, and that if desired, or necessary, controls independent of the ICS are available and sufficient for the timely mitigation of a transient.

The likelihood of inadvertently losing ICS DC Power has been reduced, and the ability to react to a loss of ICS Power enhanced. This results in fewer challenges to the plant safety systems and improved reliability.

4. Makeup Pump (P-236) Failure

1. Cause

The rapid overcooling early in the event caused the RCS pressure to decrease below the 1600 psig setpoint of the SFAS, which initiated HPI trains A and B. Since there was no loss of AC power to the Safeguards power buses, the Makeup pump (which is the normal source of high pressure reactor coolant for RCS volume control and RC Pump seals) remained in service. As a feature of the SFAS, the Reactor Building isolates on the assumption that a LOCA has occurred. Associated valving changes which occur are the opening of the suction valves from the BWST and closing the suction from the Makeup Tank. This has the effect of shifting the MU Pump suction from the MU Tank to the BWST in parallel with the A-HPI pump. Concurrently, the normal "mini-flow" from each of the HPI/MU Pumps is isolated from the Makeup Tank making full pump capacity available for delivery to the RCS.

.1 Once RCS requirements for HPI/MU had been met, the operators began to reestablish normal Makeup configuration. First, mini-flow was re-established to preclude damage to any pump as the pump flows were being throttled. This led to a rapid filling of the Makeup Tank, the receiver of the mini-flow, which caused the operator to then close the isolation valve from the Borated Water Storage Tank (BWST). This was done to cause the MU pump to draw from the MU Tank instead of the BWST. Omitted was the SFAS valve which had earlier automatically isolated the

V. 4. 1. .1 (Continued)

Makeup Tank suction. The result was failure of the pump approximately three minutes later due to lack of water flow. This in turn, led to approximately 1,200 gallons of MU Tank water being drained onto the pump room floor when the operator opened the MU Tank suction valve. Reactor Coolant Pump seal requirements were met from the available, and separate, HPI pump except for a 75 second period during switching of HPI pumps following the Makeup Pump failure. The design of the RCP seals is such as to successfully tolerate flow interruptions of this nature.

.2 Repairs and Modifications

On a short term basis, the pump failure does not impact the capability to resume power operation. Technical Specifications require two HPI pumps to be available, which there are, and either of them can provide the Makeup requirements. The Makeup Pump, in actuality, is an "installed spare."

The repair/replacement of the pump is on a longer schedule, as is consideration of alarms and/or protective interlocks which would preclude the likelihood of recurrence. A study was performed into the basis for the SFAS closure of the MU Tank suction/isolation valve. The study reconfirmed the isolation as necessary to prevent the gas binding of the MU/HPI pumps in a LOCA when the BWST level decreases. Options for providing the desired pump protection are still being developed.

5. Damage to Radiation Monitor R-15001

Following the SFAS by some eighty minutes was the receipt of a smoke alarm in the zone containing the radiation monitor which serves the Reactor Building during normal operation. Investigation by operators determined the source of the smoke to be the sample pump for radiation monitor R-15001. It was promptly shut down and the smoke soon cleared. Subsequent investigation found overheated sample pump seals caused by SFAS isolating the suction/discharge flow paths. This is a condition allowed by the existing design. Engineering is studying this radiation monitor for desirable or necessary changes, modifications, or possibly replacement. This study will include design arrangements to prevent this type of damage. This is not a startup requirement.

V. 6. RCS Overcooling

As a consequence of difficulties experienced in closing the Auxiliary Feedwater control valves, (which were demanded to 50% open on loss of ICS), and the resulting filling of the OTSGs, the RCS was rapidly cooled to below its normal post-trip temperature of 545°F.

The Rancho Seco Reactor Coolant System was designed to accommodate 240 normal cooldowns at 100 degrees F per hour. There have been several transients during which this cooldown rate was exceeded and therefore the Babcock & Wilcox Company was asked to determine the cumulative fatigue usage factor. This evaluation concluded that the allowable number of remaining cooldowns, at 100 degrees per hour, should be reduced from 240 to 235. Since only 31 cycles of this transient have been used to date, the reduction to 235 allowable cycles is expected to have no adverse impact on the current design life of 40 years.

Confusion over the interpretation of the meaning of 100 degree F per hour cooldown has prompted the District to request the calculation of a new pressure/temperature limit for use in the Rancho Seco Technical Specifications. This curve will be based on step decreases in temperature making it unnecessary to interpret the linear cooldown limit. In addition, a temperature will be specified, above which, any transient rate can be considered acceptable.

Very early in the event, the operators recognized the symptoms of overcooling and initiated actions necessary to restore normal post-trip conditions. The delay in terminating Auxiliary Feedwater flow to the steam generators took the RCS into the "Pressurized Thermal Shock" (PTS) region, identified on a operational curve showing desired operating parameters, in the cooldown procedure. This means that the pressure and temperature parameters of the transient were outside the bounds of the pressure and temperature parameters used in the pressurized thermal shock study in 1982. Other parameters of the RCS were well within the bounds of those parameters (e.g. end of life material properties, etc.) used in that study, BAW 1751.

The evaluation performed by the B&W Company showed that, for the entire transient duration, the allowable pressure was above 2750 psig, the design pressure of the RCS. During the cooldown transient, the pressure never exceeded 1700 psig. This demonstrates that the reactor vessel was not subjected to pressurized thermal shock as a result of this transient.

V. 6. (Continued)

An entirely independent analysis of the Reactor Vessel beltline was done by the Electric Power Research Institute (EPRI) using the draft ASME Section XI Appendix XX "Evaluation of Unanticipated Operational Transients." That analysis demonstrated adequate structural integrity for the Rancho Seco vessel as long as RCS pressure did not exceed the design pressure of 2500 psig, nor T_C-RT_{NDTS} less than 55°F. These requirements were met, the minimum T_C-RT_{NDTS} being 169°F, and the maximum pressure was less than 1700 psig.

7. Health Physics and Control of Contaminated Air and Water

Sources of potential release and/or exposure to Radioactive materials during this event were as follows:

* Secondary Plant Steam Relief to Atmosphere
* Secondary Plant Condenser Air Ejector
* Spilled Makeup Tank water in Auxiliary Building
* Flooding of Waste Gas Header with Makeup Tank/BWST water.

Each of these potential sources is discussed in turn.

.1 Secondary Plant Steam Relief to Atmosphere

Prior to the event the quantities of radioactivity in the Feedwater and Steam were just at or below detectable levels. As a quantifiable pre-existing primary to secondary leak did not exist, the steam release doses were determined by post-trip radioanalysis, which was based upon Cesium being the major contributor. This Cesium was residual in the secondary plant as a result of previous primary to secondary leaks. The secondary release averaged over a one hour period (actually the reliefs were open for less than ten minutes) which resulted in a source at the site boundary of 0.85 MPCs based upon insoluble Cesium as the major contributor. There was no radioiodine in this release.

.2 Secondary Plant Condenser Air Ejector

Any radioactive gases are collected from the secondary plant, in the condenser, and are discharged to atmosphere through the Auxiliary Building filtration equipment and monitors. Post event analysis of the charcoal cartridges, in service during the event, did not show any radioiodine. As these monitor both the Air Ejector and the Auxiliary Building Exhaust, it is concluded that no radioiodine was released and, therefore, there was no thyroid dose.

V. 7. .3 Spilled Makeup Tank Water in Auxiliary Building

When the Makeup Pump failed, and the Makeup Tank Discharge Valve was subsequently opened, approximately 1,200 gallons of Makeup Tank water spilled onto the floor of the Makeup Pump Room at the -20 foot level of the Auxiliary Building. This was a combination of filtered, cold reactor coolant, and BWST water. Radioactive Noble gases contained in the coolant were released into the Auxiliary Building/Exhaust system where they are monitored before being discharged to atmosphere. Total release of noble gases was determined to have resulted in a maximum 0.93 MPC at the site boundary. The whole body dose to a person at the highest downwind sector at the site boundary for the four hour length of the event was determined to be less than 0.02 mrem.

Approximately twenty minutes into the event, an Auxiliary Operator and an Equipment Attendant entered the Makeup Pump Room to access damage and isolate the Makeup Pump. Total time in the room was about six minutes each. Due to the perceived need to promptly isolate the Makeup Pump, and since Self Contained Breathing Apparatus was not readily available, the entry was made without respiratory protection. Subsequent whole body counting identified 1.0% and 0.3% Maximum Permissible Body Burdens for Silver-110M, while they received external exposures amounting to 20 and 10 mRem respectively. In addition, one individual had contamination to shoes, socks, and trousers, and the other had contamination on his shoes and hand only.

These releases and exposures were undesirable side effects of this event which did not significantly impact the health and safety of the public or the plant staff. Subsequent training, procedure changes, and equipment allocations have been implemented to preclude recurrence.

.4 Flooding of Waste Gas Header

When HPI/MU Pump "mini-flow" was re-established, as a first step in returning to normal lineups following the SFAS initiation, a consequence was the direction of as much as 300 gpm of BWST water into the Makeup Tank. As a result, the tank began filling which caused the operator to secure the A-HPI pump and close SFV-25003 from the BWST assuming that the Makeup Pump would then draw from the Makeup Tank. This action precipitated the damage to the Makeup Pump. Since the Makeup Tank discharge valve to the Makeup Pump was closed, the tank soon overfilled and discharged through its relief valve into the Flash Tank and through its vent into the Waste Gas Surge Tank and into the Waste Gas compressors. This entire system is designed to accommodate the accumulation of water and as a result no

V. 7. 4. (Continued)

significant damage was done. After the event, the water was drained and a complete Helium leak test of the system and its components and piping completed. A single small leak, external to one of the Waste Gas Compressors, was repaired by disassembly and reassembly. Only normal consumable parts were used. No damage was found. The leak likely preceded the event.

No unusual contribution to radioactive material release resulted from over filling of the Makeup Tank, and recent procedural changes are expected to preclude recurrence.

8. Emergency Plan

.1 Implementation

The initiation of SFAS, at a little over three minutes into the event, was noted by the Shift Supervisor/Emergency Coordinator as requiring declaration of an Unusual Event per the facility Emergency Plan. Fifteen minutes is allowed for the formal notifications to be initiated and this requirement was met.

Follow-up notifications were not performed with the result that the Counties expressed concern that there was "too much unconfirmed information to feel comfortable, yet too little information to make a decision."

Analysis of the implementation of the Emergency Plan in this instance has shown several enhancements which will aid the operators when confronted with complex and rapidly evolving conditions. The Emergency Plan supporting agencies and equipment were judged sufficient to provide timely communications, actions, and emergency support. Improvements in documentation were identified which will enhance the quality and timeliness of communications as required by the plan. The District is responding to the issues surrounding this activation of the Emergency Plan with additional training for the operators on Command and Control, in addition to retraining on the Emergency Plan, its recent improvements, and its effective implementation.

.2 Technical Support Center Fire Pre-Action Sprinkler System

A separate observation during this event was that, upon a Reactor/Turbine Trip, when the source of inplant power transfers from the Auxiliary to Startup Transformers, the momentary depowering of the controls to the fire pre-action sprinkler system for the Technical Support Center (TSC) occurs. This trips the sprinkler water supply valve and causes its pilot actuator to vent water. This water then spilled onto the TSC floor. Operations responded by resetting the sprinkler controls and placed plastic sheeting over equipment preventing any damage. Recent modifications have provided a suitable hard pipe drain to the control valve pilot while the electrical design is being studied to provide a method of precluding valve actuation when power transfers occur.

V. 9. Training

The large body of information developed during the study and troubleshooting done following the December 26 event, provided the opportunity and need to communicate the salient points to the plant staff. This was done by developing specific training lessons for implementation by the training department. The major issues and topics taught by that training are described below.

.1 Sequence of Events Training

The objective was to ensure that all plant operations personnel have the perspective of the cause of the event and the transient which ensued. The lessons were given in a two-part format. The first was a detailed review of the actual sequence of events and the occurrences which took place. The second lesson involved detailed discussions of the events and actions, with emphasis on the difficulties and problems which the operators faced while mitigating the event. This training was completed by mid-February.

.2 Plant Modifications

The modifications covered are those to the following components and systems: ADVs, TBVs, AFW flow control valves and ICS & NNI power supplies. The basic objectives for the training are:

- Describe the purpose of each modification.
- Describe the controls, interlocks, and operation of the modified system or component.
- Describe identified failure modes and conditions.
- Understand related procedure changes and requirements.

.3 Emergency Operating Procedure Changes

The plant EOPs are "Symptom Based" procedures based upon the B&W Owners' Group developed ATOG (Abnormal Transient Operating Guidelines). Primary focus of this approach is to address the symptoms of abnormal plant response typified by "overcooling," "undercooling," or "loss-of-subcooling." The symptoms of these situations are readily observable by the operator on available plant instrumentation. By freeing the operator from the constraints imposed by "event" procedures (e.g., must first identify which of the numerous ways "overcooling" can be caused, Loss of ICS, Small Break LOCA, Main Steam Line Failure, MFP Overspeed, etc., and then select appropriate response procedure), timely response and mitigation is accomplished by focusing activities to restore stable plant conditions which insures that the core is receiving proper cooling.

V. .9 .3 (Continued)

The "cause" of the "event" is then sought and corrected following the stabilization of the core cooling requirements, separate from the "Emergency" actions.

During the December 26 event the EOPs were utilized and applied to mitigate the event. Overcooling was not terminated in a timely manner due to lack of clear criteria for taking the action specified, to "...trip appropriate pumps." This condition has been rectified by developing "milestones" for the operator which will be effective in insuring that actions will be timely in accomplishing the EOP's purpose. As an example, on overcooling the procedure causes isolation of the services and loads upon the OTSGs. The operators were proceeding to do the necessary valve manipulations when difficulties and delays were experienced. The revised EOPs clearly state that if isolation is not effective and OTSG Operate Level exceeds 95%, or RCS Temperature is below 525°F or Pressurizer Level is less than 10 inches, then trip the pumps still supplying flow to the OTSGs." These changes are within the bases of ATOG and are procedural enhancements which will improve their implementation.

The basic objectives for the training will be:

- Description of changes.
- Basis for the specific changes.
- Effects on plant control.
- Dynamics of Response (Simulator Training)

.4 Loss of ICS Power Casualty Procedure

Such a procedure did not exist at the time of this event. A procedure has been wirtten and addresses those actions necessary to restore ICS power. Significantly, this is not intended to be attempted until the EOPs have stabilized plant conditions and then repowering will occur.

It is not appropriate to install a Loss of ICS Power procedure as an "Event" procedure for the reason that the fault may be within the ICS and it may not be available for repowering. The approach being taught is including the use of the new modifications being installed which provide power and controls (separate and independent from the ICS) in the Control Room which will insure that the demands from the ICS (during repowering) will not cause a subsequent transient.

V. .9 .4 (Continued)

The basic objectives for the training will be:

- Review the power supply system.
- Describe the purpose of the procedure.
- Describe the basis for actions.

.5 General Procedure Changes

Various Operating Procedures, Casualty Procedures, and Administrative Procedures are being changed as a result of the lessons learned from the transient. The training conducted on these changes will follow the following basic objectives:

- Describe the purpose of the procedure change.
- Describe the change.
- Describe the basis for the steps.

.6 Emergency Entry Into Areas of Unknown Radioactive Conditions

Following damage to the makeup pump during the event, operators entered the pump room to isolate the pump and assess damage. Evaluation of this entry identified the need for several changes in procedure, in addition to restatement of management policy. Training will address the changes made in the requirements and roles of health physics and operators, availability and use of protective equipment, and the procedures which govern work in areas of unknown radiological conditions. In addition, a health physics technician has been dedicated to supporting operations.

.7 Manual Valve Operation

This lecture will specifically cover the operation of the ADVs, TBVs, and AFW Valves. It will focus on management's policy that "valve wrenches" or "cheaters" will not be used on devices which include "mechanical advantage," such as linkages, screw jacks, and levers. The training will include:

- Description of basic components of the valve controls.
- Describe basic operation of the valves, both with air and manually.
- Describe how to manually operate the valves.
- Plant tour, operating the subject valves, as plant conditions allow.

V. 9. .8 Command and Control Training

Changes are being made to Administrative Procedures to revise the roles of the operators during normal and abnormal plant conditions, but this is only one part of the training. The Operations Department has also provided specific areas of control, mainly dealing with the Emergency Plan, that are to be covered. The training will be presented based on the following objectives.

- Define the line of command in normal and abnormal situations.

- Ensure the operators understand the philosophy behind the changes to the Administrative Procedures concerning their specific roles and duties.

- For the specific topics identified by the Operations Department:

  - Provide philosophy of change in control duties.
  - Provide specific guidance.
  - Discuss how to implement the guidelines.

The Shift Supervisor is to continue to maintain an overall perspective of the event, its mitigation, and compliance with administrative process. The Senior Control Room Operator will assume the responsibility to directly oversee the activities of the Control Room Operators. This will be accomplished by interacting with the control panel operators while monitoring the EOPs and directing the activities both in and outside of the Control Room.

The individual assigned as "communicator" for the Emergency Plan, will not be diverted from that assignment. This is to insure commitments to effectively implement the Emergency Plan are not delayed.

.9 Control Room Simulator

Rancho Seco operators train and practice EOP implementation on the Control Room simulator operated by the NSSS supplier, B&W. This simulator is modeled after Rancho Seco and as such does a good job of providing realistic dynamics for operator training. In the December 26 event, the makeup pump was damaged when the operator forgot to open the makeup tank suction when recovering from SFAS. The simulator is similar in this feature. A specific operator manual action is required to open that valve. It is not automatically interlocked. This feature will be again emphasized during the simulator training which is occurring prior to resumption of power operations.

V. 9. .9 (Continued)

The simulator training will include the following items:

- Emergency Operating Procedure (EOP) training including all necessary steps to terminate overcooling or OTSG overfill from any cause, including loss of ICS power.

- Changes to ADV, TBV, and AFW valve operation folloiwng a loss of ICS power.

- Command and control training including implementation of Emergency Plan when applicable. (Watchstanding Principles)

- Recovery from safety features actuation i.e., restoring normak makeup and letdown flow.

- Differences between the simulator and the facility (operator traps).

- HPI and AFW throttling and trip criteria.

- Pressurized thermal shock recovery actions.

- Cooldown rate interpretation and tracking.

- Conversion from AFW to MFW flow.

10. Operational Review

A detailed review of Operations activities, procedures, procedure adherence, training, plus operator comments and observations was done to ensure that lessons were learned and incorporated into plant procedures. A discussion of the major areas which resulted follows.

The station's Emergency Operating Procedures (EOPs) were developed and placed in use in accordance with the Abnormal Transient Operating Guidelines (ATOG) as developed by the B&W Owners' Group. These procedures are written to provide the basis for operator actions that are to prevent exceeding Technical Specification (TS) limits, or other critical limits imposed by manufacturers, such as the reactor vessel interim brittle fracture limits (IBF limits). Additionally these EOPs provide the basis for operator training on response actions required for, and philosophy of approach to, off-normal plant conditions and operations.

In this event, the overcooling was allowed to progress while the operators focused on closing the Auxiliary Feedwater Control Valves, rather than implementing the procedure step which would have shut down the associated feedwater pumps, and thereby controlled flow. This was a conscious decision on the part of the operators, and steps to preclude recurrence are discussed below.

V. 10. (Continued)

The station's Emergency Plan (EPs) was developed and placed in use in accordance with agreements with responsible regulatory and local agencies. These agreements establish guidelines to ensure specific operator actions, not associated with control of the plant, are taken during identified station emergencies. During the event, the Emergency Plan was not effectively implemented. The resolution is discussed below.

The station's Casualty Procedures (Cs) are provided to address recovery actions necessary when specific failures or events are known to have occurred. They are not implemented until the plant is stabilized by application of the EOPs. In this instance, improvements have been identified and implemented.

Operator performance and procedure adequacy were evaluated in the following areas:

1. Loss of ICS Power
2. Makeup Tank Overfill
3. Radiation Monitor R-15001
4. Manual Operation of AFW valves
5. Communications with/from Control Room
6. Fire Alarm Support Activities - Technical Support Center
7. Makeup Pump Damage
8. Loss of SRO (RO)
9. Emergency Plan(s)
10. OTSG Overfill

No attempt was made to prioritize these subevents in relation to significance of impact or influence on operator actions.

- Examination of the ten subevents identified the following common items for action. These are summarized below.

- The Emergency Operating Procedures (EOPs) are adequate and include all necessary steps to terminate overcooling or OTSG overfill from any cause, including loss of ICS power. The EOPs and ATOG do not contain plant parameter criteria (e.g., RCS temperature, Pressurizer Level, etc.) as to when to take certain steps, like tripping all feedwater pumps to meet the ATOG intent which is to minimize the overcooling. These criteria are being developed and implemented.

- Casualty procedures provide event oriented guidance to the operators as a followup and recovery following plant stabilization by the symptom oriented EOPs. The ICS power recovery and SFAS actuation recovery procedures have been implemented and training will be completed prior to return to power operation.

V. 10. (Continued)

- The Emergency Plan will be revised to provide more guidance in responding to multi-event emergencies. As written, the Emergency Plan adequately addresses single-event emergencies; however, in situations where the operator must cope with several competing priorities, the guidance to assign priorities and deal with the events is being provided.

- The plan is to be streamlined and simplified by reducing branching between procedures and reducing the number of forms or logs.

- Training programs are being reviewed for completeness inlight of the experience gained from this event. Areas being reviewed are related to operator knowledge and skills related to:

  - Integrated system operation
  - Manual operation of manual or power operated valves
  - Fire alarm response requirements
  - Emergency Radiation Control procedures and techniques
  - Integrated emergency plan response

11. System and Component Response

This section discusses individual items of equipment, or systems, which for one reason or another were deemed to require post-trip review.

.1 Auxiliary Steam Relief Valve PSV-36012A

The accelerometer monitor mounted on this valve indicated on IDADS that the valve was open following the Reactor Trip. Since this valve constitutes a steam (heat) load on the OTSGs, and thereby contributes to the overcooling, an analysis of the system and its operation was conducted. The results of the several investigations was that upon Loss of ICS DC control power, the Auxiliary Steam control station received a "50% Demand" signal. During power operation, this would not be observed as the Auxiliary steam demands are shared between Hot Reheat steam (from the Moisture Separator Reheaters - MSRs) and the Auxiliary steam supply, whichever is at the higher pressure. Following the trip, Auxiliary Steam needs decrease and with the control station set at 50% demand, the excess must be relieved, as happened here.

Its effect upon the overcooling is somewhat mitigated in that considerable pressure reduction occurs in providing steam through the controls to the Auxiliary Steam Header. As the main steam pressure decayed, in response to the cooldown, the point was reached at which 50% demand was not excess to the loads on the Auxiliary Steam system. This reduced pressure then allowed PSV-36012A to reseat and minimize the draw on main steam. The Auxiliary Boilers were soon placed into inservice, further reducing the demand for steam drawn from the OTSGs for plant use.

V. 11. .2 Difficulties in Manual Operation of Auxiliary Feedwater Control Valves FV-20527/8

Upon loss of ICS DC Control Power, these two valves received a "50% Demand" signal. Response to this demand brought the valves far enough open to effectively allow unthrottled flow of auxiliary feedwater into OTSGs A and B. These air diaphragm actuated valves are provided with side mounted manual operators capable of closing the valve against both the opening spring in the actuator and the hydrostatic load due to throttling water.

The valve controlling flow to the OTSG-B, FV-20528, was thought closed by the operator dispatched to close it, although it was still passing considerable water. A second operator observed that it was not fully closed and completed closing it by hand without difficulty.

The valve controlling flow to OTSG-A, FV-20527, was positioned in the closed direction by the operator using the manual device. He thought it was still partially open and proceeded to obtain a valve wrench to assist in getting the valve fully closed. Unknown to this operator was the fact that flow had stopped as the valve was fully closed. Use of the valve wrench then caused the two 5/16 inch dowel pins, which position the manual operator to the valve yoke, to be sheared off. At this point, the valve again responded to the 50% open demand and "popped" open. Uncontrolled flow to OTSG-A was re-established and the overcooling continued.

Detailed inspections of both manual operators showed signs of application of excessive manual closure forces, although indications on FV-20528 were minor in comparison and predated the damage done to the device on FV-20527. Both devices were found to be loosely mounted to the valve yokes, although this has not been identified as having had a significant effect on their operation. The dowel pins are the important load carrying and aligning components. Contributing to these problems is the difficulty which exists when trying to determine when the valve is "closed." There was no clear indication or index to indicate closed, nor is there a local indication of flow through the valve.

Repairs to these manual operators will involve installation of a new manual operator on FV-20527, and rebuilding the assembly for FV-20528. Operator training on valve wrench policy and proper manual operation of these devices will provide confidence that reoccurrence is unlikely.

Labels to clearly show the valve position are being installed to provide indication of position when manually operated.

V. 11. .3 Difficulty in Manual Operation of Auxiliary Feedwater Isolation Valve FWS-063.

When FV-20527 "popped" open, the operator attempted to stop auxiliary feedwater flow by closing the downstream, normally locked open, manual maintenance isolation valve, FWS-063. This was unsuccessful in that the valve was stuck open. The need to close this valve was soon eliminated by restoration of ICS power and return of its control functions to the Control Room. Troubleshooting subsequently identified a lack of lubrication and rusted yoke nut bearings. Reworking these components restored the valve to an operable condition.

As an element of the troubleshooting effort, the similar valve on the OTSG-B line, FWS-064, was inspected, as were all similar valves in service on the Auxiliary Feedwater System. All were found serviceable with only normal closing torque required to operate through their full travel.

Section V.11.8 discusses the preventive maintenance program for manual valves.

.4 Main Steam Relief Valve PSV-20544

Following a Reactor Trip on October 2, 1985, this valve was found to have simmered for some time prior to reseating. Subsequently, it was tested inplace as its lift setpoint was found to have drifted low. The indication of opening during the December 26 event was via the attached acoustic monitor, which showed the valve opening coincident with the Loss of ICS power, which had opened the Atmospheric Dump Valves. Since PSV-20544 and an ADV are both on the same main steam header, a test of the sensitivity of the installed monitor was run. This showed that the PSV-20544 monitor was extremely sensitive and would indicate "open" in sympathy with the ADV.

Independent analysis of the main steam header pressure versus time showed that at the time PSV-20544 first indicated open, steam pressure was dropping from its pre-trip steady-state value, and at no time had it challenged any of the main steam relief valves. This analysis determined it was appropriate to recalibrate the valve monitor and that PSV-20544 is serviceable for power operation.

.5 RCP Seal Injection Flow Interruptions

The first observed indication of problems with the Makeup Pump (results of operation with Makeup Tank isolated) was that RCP seal injection flow decreased when the B-HPI pump was shutdown. This was a result of all three HPI/MU Pumps discharging into a common header from which RCP seal injection flow is drawn. Since A-HPI was already shutdown, and the Makeup Pump isolated from a source of supply, shutting off the B-HPI would deprive

V. 11. .5 (Continued)

the RCP Seals of injection water. This anomaly occurred twice as the operators troubleshot the condition. Total time with reduced injection flow amounted to about 75 seconds. The RCP Seals are designed to operate indefinitely without injection if the RCP is running, and for 90 seconds if the RCP is shutdown. At this time, one RCP was shutdown and three were running. This analysis also looked at the detailed RCP and seal recorder data which confirmed the conclusion that seal performance or integrity were not compromised by the event and that the RCP Seals are suitable for returning to power operation.

.6 Feedwater Heater Shell-Side Relief Valves Opening

On October 2, 1985, the Reactor Tripped from low power, and a overcooling event resulted. The cause was attributed to overlapping setpoints (when tolerances are included) between the 4th Point pegging steam supply pressure and the associated shell side relief valve. This was resolved by developing and calibrating a new and higher relief valve setpoint.

On December 5, 1985, following a Reactor Trip, it was noted that the 4A Feedwater Heater Relief again opened. Isolation of the pegging steam supply prevented recurrence of the previously experienced overcooling. Detailed troubleshooting determined that the 4A and 4B pegging steam supply valves would overshoot their target setpoint by 10 to 20% when initially opened. Analysis was done to develop a new and lower setpoint which would provide the desired post-trip feedwater heating, yet accommodate the observed overshoot. It was this configuration which was in effect during the December 26 event. A report of steam venting from the 4th point feedwater heater relief valve caused the operators to again isolate pegging steam early in the event, to preclude its contributing to the overcooling. As a possible "repeat" event, this item was investigated.

Pre-existing work requests to investigate reports of steam leaking through the pegging steam control valves were activated. It was found that one control valve was seriously "steam cut" and required reworking. This confirmed the likely cause of the relief opening being that the pegging steam control valve had excessive seat leakage, even when closed. Such a condition would pass steam, and heat, into the feedwater heater even after feedwater flow through the heater has stopped, resulting in a buildup of pressure. This system is now serviceable and repeat relief valve opening is not expected.

V. 11. .7 HPI/RCS Injection Valve SFV-23811 Position Indication

Following SFAS initiation during the event, the operator "balanced" HPI flow between the four injection nozzles. This is accomplished by switches controlling the SFAS controlled injection valves from the Control Room. While adjusting to achieve approximately 100 gpm through each nozzle, the "closed" indicator light illuminated for valve SFV-23811.

Post-event troubleshooting found the position switch adjusted to actuate at approximately two turns open. This position is commensurate with throttled flow of approximately 100 gpm. The switch was reset to about one half turn and the valve operation checked. In the as-found condition, the valve would still operate to the fully closed position. In a similar fashion, the valve operator "open" torque switch bypass will remain effective to allow the valve to operate through its full stroke with the new "closed" switch setting.

The above described discrepancy is independent of a previously resolved condition relating to the flow meters. Those meters have to be calibrated to compensate for operating pressure effects. The procedure which calibrates the flow meters accommodates this need and insures that the resulting indicated flow will be conservative. As described above, the flow was determined to have been "real," that is, the valve was not closed and the meter indicated 100 gpm. This condition was observed only on this valve; the other three performed as expected.

.8 Preventive Maintenance (PM) Program for Manual Isolation Valves

Troubleshooting of the Auxiliary Feedwater manual isolation valve, FWS-063, found its bearings dry of lubricant and in a condition suggesting it would benefit from periodic servicing. Investigations of five similar valves showed them all to be operable, although evidence of recent lubrication was missing.

Reviews of the existing preventive maintenance program shows that these valves are not on a PM schedule. Their need for servicing being determined by observation during frequent monitoring of plant equipment by operation staff.

In recognition of the desirability of having certain manual valves readily operable, the Nuclear Operations Manager has identified a list of approximately 100 valves which will be verified operable prior to resumption of power operation. These manual isolation valves will be characterized by their purpose and need to allow isolation of important active equipment such as pumps, valves, and heat exchangers. They will be selected to include both primary and secondary plant systems. Function, not

V. 11. .8 (Continued)

class, being the criteria. The program will involve actual stroking the valve, and where necessary, servicing with lubricants, packing, or adjustments. Statistics will be collected and evaluated to determine a summary status of valves in similar service.

Significant changes are underway with respect to the Preventive Maintenance Program at Rancho Seco. Staff is being added for the specific purpose of expanding the scope and quality of the program. Specific procedures detailing the PMs are being expanded to provide confidence in the operability of the PM'd equipment. This expanded PM program will include the above identified valves in addition to those already receiving periodic maintenance, and any which meet the criteria being developed for this program.

12. Quarantined Equipment List

This section of the Action List was used to provide quick reference to the scope and status of that equipment assigned to the Quarantined Equipment List, as discussed elsewhere in this report.

VI. PLANT MODIFICATIONS RESULTING FROM EVENT

The event of December 26th, was in reality two events. They were: A Reactor Trip, caused by a loose connection in the DC distribution wiring within the ICS; and an Overcooling, one cause of which was the inability of the operators to effectively manually control the Auxiliary Feedwater valves which "failed open" upon Loss of ICS DC power. Section V.3.4 of this report discussed the ICS related modifications in detail.

Review of the recommendations coming from the event shows they can be categorized as above. The following discussion looks at the minimum set of changes necessary to preclude recurrence of the initiating events while enhancing the operators ability to effectively mitigate the consequences of such events.

1. Loss of ICS DC Control Power

   a. Power Supply Monitor

      The Power Supply Monitor is sensitive to resistance added in series with the voltage it is monitoring. This is a condition peculiar to its design. To date over 160 service years have been accumulated on this type of device in Nuclear Power Plants. The situation at Rancho Seco on December 26, 1985, is the only time the peculiarity is suspected to have caused a Loss of ICS Power. Having replaced the power distribution wiring the likelihood of a recurrence is remote. On this basis, there is no need for a modification at this time.

   b. Redundant Power Monitors

      If installed in parallel with the one in the Rancho Seco ICS, they would have all initiated a trip condition, for the reason that they would have all been monitoring a degraded signal, combined with the peculiarity discussed in VI.1.a above.

      The one advantage a "2 out of 3" arrangement does have is that a random failure of a PSM would not thereby cause a trip, while a disadvantage is that they constitute new failure modes not presently existing. One such issue is "crosstalk," wherein a low voltage is sensed by one PSM causing it to trip. When tripped, the PSM current draw is reduced, which then unloads the source allowing voltage to increase. The result is that the PSM reacts and that a cycling can persist. Still, the history on these units is that they are suitably reliable. On this basis, such a modification is not desirable prior to resumption of power operations.

VI. 1. c. Time Delay on S1/S2 Trip

Testing showed that the time delay to trip interval was shorter then the maximum 0.5 second. Investigations with the Manufacturer continue to determine the appropriate value. The switches are being replaced with similar new switches to allow for independent accessment of switch characteristics. Adjustment of the delay to achieve the maximum value, is not practical with this type of switch.

2. Consequences of Loss of ICS AC or DC Control Power

a. Turbine Bypass Valves

Modifications are being installed in the control path to these valves which will cause them to "fail closed" upon Loss of ICS DC Control Power. The operator will be provided with a switch in the Control Room to allow cycling the valve as necessary to maintain the plant stable following loss of normal valve control.

b. Atmospheric Dump Valves

Modifications are being installed in the control path to these valves which will cause them to "fail closed" upon Loss of ICS DC Control Power. The operator will be provided with a switch in the Control Room to allow cycling the valves as necessary to maintain the plant stable following loss of normal valve control.

c. Auxiliary Feedwater Flow Control Valves

Control of these valves is being revised in a similar fashion to the TBVs and ADVs, except that, rather than "fail closed", they will "fail to setpoint." The "setpoint" has been selected to preclude the situation where the valves fail closed when flow is actually required. Operator action will be available to fully close the valve, or throttle to intermediate positions without interaction of the ICS, as deemed necessary.Operator action could of course open the parallel SFAS Auxiliary Feedwater Valves, but to preclude the requirement for operator action, the minimum position is preferred. A spectrum of AFW system criteria was reviewed and considered in the selection of this setpoint.

d. Auxiliary Steam Header Regulator

No modifications are necessary as the operator can isolate the main steam supply independent of the ICS from the Control Room, should that be desired. A future modification to power this control independently from the ICS is being considered. The extent of cooling available through this path is limited in any case and does not require prompt action.

VI. 3. EFIC Implementation

By letter dated August 15, 1980, the NRC identified a situation where failure of power supplies to NNI or ICS could result in ADVs opening to 50% open position. The District concurred with this scenario in its October 6, 1980 submittal. The District proposed to correct this ADV response as part of its EFIC (Emergency Feedwater Initiation and Control) AFW system upgrade. The design concept was presented to the NRC at a September 4, 1980 meeting.

Equipment delivery for EFIC was originally estimated to be in early 1982. When actually signed, the contract specified equipment delivery for April 1983. During the initial design review process, additional improvements to EFIC were identified. As a result of these design changes, the deliver schedule was adjusted to May 1984.

NUREG 0737 required AFW automatic initiation and flow indication (II.E.1.2.1 and II.E.1.2.2). The NRC issued Safety Evaluation Reports in January and September 1982. In October 1982, the District indicated that it would install interim safety grade AFW modifications and that EFIC was separate and beyond the AFW upgrade requirements of NUREG 0737. The District also submitted a new schedule for EFIC implementation showing completion by Cycle 7. This schedule was confirmed by the District in December 1982.

The District informed the NRC in April 1983 that the installation was tied to Control Room Design Review (CRDR) and RG 1.97 modifications. This was based on the need for an EFIC control panel in the Control Room that was compatible with the CRDR effort. Part of EFIC are the associated RG 1.97 instrumentation commitments for Rancho Seco; as a result it was necessary that EFIC be rescheduled for Cycle 8 (i.e., the next scheduled refueling).

In late 1983, the District implemented an Integrated Living Schedule to better control resources, scheduling of modifications, and enhanced operations at Rancho Seco. Since the AFW requirements of NUREG 0737 were previously completed, EFIC was considered a plant betterment. Using the Living Schedule, to prioritize the use of District resources, the District scheduled EFIC to be installed in two phases--Cycle 8 and Cycle 9. The Living Schedule process determined that other NUREG 0737 modifications, 10CFR50.49 - Environmental Qualification of Electrical Equipment, Appendix R - Fire Protection, Generic Letter 83-28 ATWS, and NUREG 0737 Supplement 1, items receive high priority which resulted in heavy commitment of District resources during the Cycle 7 outage.

VI. 3. (Continued)

It became clear in meeting the requirements of NUREG 0737, that the number of modifications imposed in Rancho Seco would exceed the electrical capacity of its existing emergency diesel generators. The District decided in 1980-81 to purchase two additional diesel generators to augment the existing system. The District originally planned the installation of these new generators during the Cycle 7 refueling outage. This schedule was compatible with the installation of the majority of the TMI modifications, as well as the implementation of EFIC. The diesels purchased were made by TDI and the District, as well as several other utilities, were forced into a major TDI generator requalification program as a result of design problems discovered on the Shoreham plant diesels. This requalification program required both time (several years) and resources to complete. The current schedule will have the diesels operational during the Cycle 8 refueling outage.

Since EFIC, and several other modifications, were tied to the installation of the diesels, the District was forced to defer implementation of EFIC. This delay also afforded the District time to take a closer look at EFIC as installed at CR-3 and ANO-1. Because of some initial startup and operational difficulties at these installations, the District decided on installing the indication portions of EFIC during Cycle 8. This would allow the operators to gain familiarity with the system during an operating cycle. Likewise, the District has been interfacing with the staff of ANO-1 to minimize any operational problems and benefit from the ANO experience, as the District's EFIC will closely resemble the ANO EFIC.

In October 1985, the District committed to accelerate implementation of EFIC. The District outlined the specifics of the EFIC implementation in a letter to the NRC dated January 17, 1986. This implementation will result in the majority of the EFIC actuation and control functions being operational at the completion of the Cycle 8 refueling outage.

4. Other Modifications

Many of the event related recommendations deal with potential modifications. These are beyond the requirements for startup and are being evaluated on their merits for subsequent incorporation.

VII. ROOT CAUSE(S)

The Rancho Seco Incident Analysis Group (IAG) has responsibility to determine the "root cause" of problems or events at the facility. This involves addressing the programmatic causes as well as the direct causes unique to specific occurrences.

VII. (Continued)

The IAG charter causes this effort to be separate from other efforts to seek and resolve causes. In the instance of the December 26 event, the IAG monitored the activities and findings of the Transient Analysis Organization, but used methods and resources to independently arrive at the "Root Cause(s)" of the event. These root causes are presented below. Although developed independently, their basis is supported by the material presented in this summary report.

The December 26 event was subdivided into five "event themes" for root cause determination. They are:

- Loss of ICS Power
- Rapid Cooldown
- Makeup Pump
- Health Physics
- Emergency Plan

Analyses of these event themes were then studied for the following considerations:

- Procedural Adequacy
- Unique Design Features
- Human Factor Consideration
- Related Issues -
  - Training
  - Preventive Maintenance
  - Personnel Access
  - Vendor Technical Analysis

The following discussion identifies the Root Causes and the contributory causes, as appropriate.

1. Loss of ICS Power

   Root Cause: Manufacturing Error

   A lug was improperly installed on a factory prepared wire. The resulting connection exhibited variable resistance which was on the input to the Power Supply Monitor. The resulting variable voltage lead to the PSM tripping when the ICS was still being supplied with nominal voltage and power. Corrective action involved installing new wiring and lugs.

   Contributory Causes:

   1. The Power Supply Monitor is sensitive to resistance in series with its voltage input. As little as one ohm was found to cause the trip point to increase. Approximately 5 ohms at the failure point was sufficient to cause the PSM to trip at its nominal operating voltage, 24 VDC. Corrective action involves wiring the PSM directly to the DC source bus rather than the end of the distribution bus.

VII. 1. 2. The S1 and S2 source switches were found to have short built-in time delay characteristics, approximately 0.15 second while the specification is for 0.5 second. This made them more sensitive to short term trip signals generated by the PSM. Corrective action is to obtain new switches.

2. Rapid Cooldown of NSS

<u>Root Cause:</u> Delay in Implementing Design Changes to Mitigate Effects of Loss of ICS Power

The susceptability to this event has been recognized for some time. In response, a Class I design modification called EFIC has been developed. Revisions in design criteria, delays in reaching scheduled refueling outages, equipment manufacturing problems and delays have compounded to delay its implementation. Interim corrective action is to install modifications which will provide the Control Room Operator with necessary controls powered independently of the ICS, while aggressive efforts are being made which will install EFIC at the next refueling.

Contributory Causes:

1. Procedures

   The overcooling procedure did not clearly identify criteria which would cause the operator to take second level actions to terminate the condition when the initial effort proved ineffective.

   Corrective action has been taken to incorporate this guidance into the EOPs.

   A casualty procedure to address Loss of ICS Power was not provided. A procedure for recovery from SFAS operations was not available. Corrective action is to provide these procedures and the training to effectively implement them.

2. Training could have compensated for the lack of the above features or procedures. Training which had been given was not able to compensate for those deficiencies. Corrective action; training developed to understand the event, revised policies, and new procedures will increase the awareness and knowledge of the operators.

VII. 3. Damaged Makeup Pump

Root Cause: Procedures

A procedure specific to restoration of normal equipment lineups following SFAS initiation was not available. Procedural references in the EOPs for insuring pump suction/discharge paths were not consistently included. Corrective action is to provide the missing procedure and add the appropriate caution steps.

Contributory Cause: Training

The operator was trained and aware of the consequence of operating this pump without suction. This event demonstrates that training alone may not always be sufficient to insure requirements will be remembered. Training should not be expected to compensate for lack of appropriate procedures. Corrective action; this event reinforces previous training with an example of the consequences. Retraining and new procedures are sufficient to preclude reoccurrences.

4. Health Physics Procedure Implementation

Root Cause: Human Performance

The perception of the individuals involved was that they were following appropriate guidelines for the conditions which prevailed and the directions they have been given. Corrective action is to clearly restate Administrative Policy that procedures will be followed and provide training on implementing procedures.

Contributory Causes:

1. Training

These individuals did not clearly understand their obligation to follow established procedures and to utilize the protective equipment and other personnel available to them. Corrective action is retraining of all operational personnel to assure that others may not harbor similar concepts and perceptions.

2. Imprecise Definition of HP Responsibilities, Authority, Duties

Administrative policy is being restated while policy is being changed to assign a HP Technician to Operations for the single purpose of supporting their activities.

VII. 5. Emergency Plan Communications

Root Cause: Training

Training was not sufficient to insure the Emergency Plan would be properly implemented during a plant transient. Priority of responsibility to that plan was not fully appreciated by some. Corrective action is to provide training on implementing command and control policy and obligation to assigned duties.

Contributory Causes:

1. Human Performance

The operators were faced with several independent "Emergencies." The overcooling, fire alarms, radiation alarms, damaged equipment, disabled operator, and failed equipment were the major ones. These diverse events challenge the ability to be effective and prioritize. Corrective action is to restate policy and process in the training program while developing training exercises which will better practice complex scenarios. The relationship between the roles of the Shift Supervisor/Emergency Coordinator and the Senior Reactor Operator are being defined and practices in plant simulator training.

2. Procedures

Complex branching within the Emergency Plan and its implementing procedures makes smooth implementation difficult. This event highlighted the situation and led to revisions to remove many of the obstacles. Corrective action, in addition to increased training, the necessary forms for an event are being packaged into a folder which will insure all are convenient and ready to use.

VIII. Conclusions

1. Safety Significance

The Loss of ICS Power initiated events which led to a Reactor/Turbine trip and a subsequent overcooling of the NSS. This report analysed the causes of those events and describes changes and modifications whose effects are to reduce the likelihood of subsequent losses of ICS power while significantly improving plant and operator response to such an occurrence. Neither the facility design bases, typified by the Main Steam Line Failure event, or the NSS components were stressed to the values assumed in the design for normal cooldowns.

Extensive study and analyses have not determined any unreviewed safety questions, while showing that the facility is ready to return to power operations with its reliability significantly enhanced. Challenges to safety systems should be correspondingly reduced.

VIII. 2. Appropriateness of Findings

Study and analysis of this event benefited from the systematic identification of issues, logically developed troubleshooting and careful investigations which substantiated or rejected the postulated root causes. Independent from the troubleshooting and repair activities, the Rancho Seco Incident Analysis Group developed the Root Cause(s) of the events. This summary report relates the collected information on each issue which is subsequently resolved in the root cause analysis. The correlation between the findings and the root causes demonstrates the appropriateness of the methods used and validates the results. This consistency provides confidence that the changes and modifications made will accomplish their intended purpose of precluding recurrence while enhancing reliability and safety.

3. Status of Resolution of Recommendations, Repairs, and Modifications

Each of the several investigations resulted in recommendations and proposed modifications to the facility, its procedures, or the training program. Those issues, which are necessary for effective plant operation, have been identified as required for startup and will be completed accordingly. Items of specific interest which will be completed are recapped below:

- Repair Hand Operators on AFW Control Valves
- Institute PM and exercise important Manual Isolation Valves
- Rewire the ICS Cabinet DC Power Distribution circuits
- Wire the ICS PSM direct to the supply bus
- Provide Controls, independent of the ICS, to cause valves to go to desired positions of Loss of ICS Power:
  - AFW Flow Control Valves
  - Atmospheric Dump Valve
  - Turbine Bypass Valves
- Provide permanent drain for TSC Fire Control Valve
- Revise EOPs to provide specific action points
- Provide Loss of ICS Casualty Procedure
- Incorporate Lessons Learned Procedure Changes
- Institute formal Command and Control concepts
- Provide Operator Simulator Training on event and procedure revisions

4. Effectiveness of Modifications and Training

The described modifications will provide the desired, and necessary, capability to control events such as loss of ICS power, from within the Control Room. This control is achieved independently of the ICS or its sources of power. With the modified controls, the transient resulting from Loss of ICS Power will be much less severe, even assuming no operator action.

VIII. 4. (Continued)

Training related to this event, and the proper and effective use of the modifications and revised procedures, will be demonstrated to be effective by observation of the operators at the simulator. Practice and experience so gained will enhance and reinforce the effectiveness of these changes. The result will be an improved ability to mitigate transients in a timely and effective manner.

5. Suitability for Restart

This report describes the investigations done, the results determined, and the actions taken to reduce the likelihood and consequences of loss of ICS power. The improvements and modifications made go beyond that single event to provide to the operator additional capability to ensure the timely mitigation of a wide variety of events. The applicability of the operators emergency operating procedures has been validated, while the specifics have been improved to enhance confidence and response.

Rancho Seco has been physically configured to provide similar control features and capability similar to its sister B&W plants. The achievement of this configuration provides a significant enhancement of safety and the degree of confidence necessary to support its return to power operation.

ATTACHMENT 1

SMUD SEQUENCE OF EVENTS

Rev 4
2-7-86
DW

SEQUENCE OF EVENTS KEY

TIMES

04:16:20	Clock time. The IDADS computer time was used as the official clock. All times have been normalized to the IDADS clock.
[07:23]	Time after initial event in minutes and seconds.
04:20:40?	Time is estimated to the nearest second. However, the associated event may have occurred a few seconds earlier or later.
04:16:?	Time is estimated to the nearest minute. The event has occurred sometime during this minute.
04:58?	This is the best estimate for the occurrence of this event. It could have occurred a few minutes earlier or later.

SOURCE ABBREVIATIONS

IDADS	from the Interim Data Acquisition and Display System computer.
OPS	from operator personal statement or control room logs.
FUI	from operator follow up interviews.
CALC	from an engineering calculation of pressures or flowrates.
SCRTY	from security log.
ECL	from Emergency Coordinator's log.
SCHRT	from control room or ventilation strip chart

CHRONOLOGICAL SEQUENCE OF EVENTS
12/26/85

INITIAL CONDITIONS

Unit operating at steady state power of 76%, 710MW(e)
Integrated Control System in full automatic
Bailey Computer out of service (one of the Control Room's two main computer systems)

TRANSIENT INITIATOR

TIME	SOURCE	EVENT/ACTION
04:13:47	IDADS	Loss of ICS is caused by the simultaneous de-energizing of all redundant ICS DC power supplies. All lights on the ICS stations go out. All Bailey station demands go to 50%. "ICS OR FAN POWER FAILURE" annunciator alarms.

SEQUENCE OF EVENTS

TIME	SOURCE	EVENT/ACTION
04:13:47 [00:00]	IDADS	"ICS OR FAN POWER FAILURE" annunciator alarms. Upon the loss of ICS DC power, all ICS demands went to midscale, corresponding to 0 volts. The startup and main feedwater valves closed to 50% because of this decrease in demand signal. The main feedwater pump speeds reduced to low speed stop, 2500 rpm. De-energization of the ICS DC power supplies caused the main feedwater block valves to close. Main feedwater flow to the steam generators decreased. The loss of ICS DC power also sent a demand to the Bailey AFW control valves, AOVs, and TBVs to open to 50% demand. The main turbine cannot respond to changes in steam header pressure because signals to change governor valve position originate in the ICS. Total steam flow increased. Steam generator pressures began to decrease. The reduction in feedwater flow had a greater effect than the extra steam flow. RCS pressure began to rise rapidly as the RCS heated up. The main steam to auxiliary steam pressure reducing station also failed to mid scale. Auxiliary steam pressure began to increase.
04:13:55?	OPS	Operators notice MFW flow decreasing rapidly and RCS pressure increasing. Operators manually open one of the pressurizer spray valves in an attempt to stop the RCS pressure increase.

04:14:01	IDADS	Main feedpump discharge pressure decreases to less than 850 psig, automatically starting the electric driven AFW pump, P-319. At this time, there was no main feed flow to the steam generators because steam pressure was higher than feedpump discharge pressure.
04:14:03 [00:16]	IDADS	Reactor trip on high RCS pressure. The turbine and generator trips are also initiated by the reactor trip. Operator closes pressurizer spray valve.
	OPS	Immediately upon reactor trip, transformer yard fire alarms, seismic trouble alarm, and SFP high temperature alarm are received on the main annunciator panels. A TSC fire system actuation alarm is received on the IDADS computer. An auxiliary steam relief valve lifts.
04:14:04	SCHRT IDADS	Momentum of RCS carries peak pressure to 2315psig. AFW flow begins to both OTSGs.
04:14:06	IDADS	AFW dual drive pump, P-318, autostarts on low main feedpump discharge pressure (850psig). RCS hot leg temperature reaches a peak of 606.5 F.
04:14:07?	OPS	Operators perform the actions of Emergency Procedure Section E.01. This included reducing letdown flow. Operators then proceed with Emergency Procedures Section E.02.
04:14:12	IDADS	Six OTSG code safety valves are lifting.
04:14:25	IDADS	Operators fully open "A" inject valve for more makeup addition to the RCS due to low pressurizer level in accordance with E.02, Vital System Status Verification.
04:14:26	IDADS	All OTSG code safeties have reseated.
04:14:48	OPS	Makeup Tank level decreasing rapidly due to high rate of makeup to RCS. Operator opened BWST suction valve on "A" side (SFV-25003).
04:15:04	IDADS	Operators start "B" HPI pump to increase reactor coolant inventory from BWST
04:15:18	IDADS	No level remains in OTSGs. AFW is removing heat.
04:15:30?	OPS	Operators sent to close AFW flow valves and place covers on MSR relief valves. The MSR's go into a vacuum following a turbine trip. The relief valves have been a source of major vacuum leaks in the past. There were no condenser vacuum problems during this event.

04:16:00?	OPS	Operators sent to close TBVs and ADVs. The TBVs and ADVs could have been shut from the Remote Shutdown Panel located two floors below the Control Room. However, the operators failed to remember this fact. Ranch Seco does not have Main Steam Isolation Valves (MSIV's).
04:16:02 16:04	IDADS	Operator trips both main feedpumps. Two operators verify AFW flowrates to each OTSG were greater than 800 gpm. Operators noted that both AFW pumps were running and that they had no control over main feedwater due to the loss of ICS.
04:16:14	IDADS	OTSG levels begin to increase.
04:16?	OPS	Operator secures pegging steam from Control Room to ensure that it would not contribute to the cooldown. A Control Room operator heard a steam relief valve blowing on the turbine deck. Pegging steam had caused feedwater heater relief valves to lift in the past. This had contributed to a recent overcooling event at Rancho Seco.
04:16:40	IDADS	RCS temperatures decrease below expected post trip value (550 F.) RCS pressure is 1670 psig.
04:16:57 [03:20]	IDADS	RCS pressure has decreased to 1600 psig. Pressurizer level is 15 inches. SFAS automatically initiates on low RCS pressure. "A", "B", "C", "D" HPI injection valves travel to prethrottled position. Selected SFAS equipment, including motor driven AFW pump P-319, trips and block loading of SFAS equipment begins. AFW SFAS valves travel full open. "A" and "B" DHR/LPI pumps autostart in their recirculation mode. Diesel generators autostart but do not close onto vital busses. There has been no loss of power to the vital 4160 volt busses. SFAS also actuates containment building isolation. The stripping of P-319 from its vital bus causes total AFW flow to decrease by half.
04:16:59	IDADS	"A" HPI pump autostarts from SFAS signal.
04:17:00	IDADS	Pressurizer level goes offscale low. Subcooling margin is 75 F. and increasing.

04:17?	OPS/ FUI	Operators unlocked the ICS cabinets. They found all DC power power supplies de-energized. The dual switches (S1-S2), which provide power to the supplies, were both found in the full down position and were assumed to be closed. The ABT, which supplies all ICS power (A.C. and DC), was found powered from its normal supply. An operator switches the ABT to its alternate supply, but ICS DC power does not return. The ABT is switched back to its normal supply.
04:17:10?	OPS	Operator closes AFW SFAS valves which were fully opened by the SFAS actuation.
		These valves are in parallel with the already open ICS controlled AFW valves.
04:17:15	IDADS	"A" and "B" CR/TSC Essential HVAC units start from the SFAS signal.
04:17:27	IDADS	Motor driven AFW pump P-319 is block loaded back onto its vital bus and immediately restarts. The dual drive AFW pump has been running continuously since it started on low feedpump discharge pressure a few moments after the loss of ICS power. AFW flow to both steam generators increases and is now in excess of 1000gpm to each steam generator.
04:18:58	IDADS	RCS temperature goes below 500F.
		One RCP should have been stopped at this time to avoid core lift concerns. An RCP was tripped at 04:28.
04:19:00 [05:13]	CALC	Pressurizer surge line empties. RCS pressure begins steeper decline. Subcooling margin drops 8°F to 77°F and then begins to increase. Water at vessel head is flashing to steam.
04:19:15	IDADS	Operators secured "A" train of CR/TSC Essential HVAC to reduce the ambient noise level in the Control Room.
04:20:01	IDADS	Auxiliary steam relief valve reseats and does not lift again.
		Main steam pressure was 550psig at this time. There was enough flow across the pressure reducing valve to reduce the auxiliary steam pressure below the relief valve setpoint.
04:20:20	IDADS	Steam generator pressures have decreased to 500psig. Main feedwater flow begins.
		At this pressure the running condensate pumps began to supply feedwater to the OTSGs through the idle Main Feedpumps. This added approximately 1000gpm to the feed rate of each OTSG for a little over two minutes.

04:20?	OPS	Operator sent computer technician to look at ICS power. He confirmed all DC power supplies were de-energized and the ABT had not transferred.
04:21:25	IDADS	RCS reaches minimum pressure of 1064psig. Operator is closing ADVs. The combination of water flashing in the head and the reduction in steam flow as the operator manually isolates the ADVs has stopped the RCS pressure decrease. HPI flow was now sufficient to keep up with the cooldown rate.
04:21:30 [07:43]	CALC/ OPS/ IDADS	Pressurizer surge line begins to refill. RCS pressure and subcooling margin are increasing. Formation of steam at the head has ceased. HPI has refilled all steam voids.
04:22:00	IDADS	Violated B&W recommended PTS curve for the reactor vessel.
04:22:40	OPS/ IDADS	Operator isolating TBVs. RCS pressure is rapidly increasing.
04:22:50	IDADS	Steam generator pressures have decreased to 435psig. Main steamline failure logic closes the startup and main feed valves. FW flow from the condensate pumps is stopped.
04:23:?	OPS	Local isolation of TBVs and ADVs is completed. ADVs were the first valves closed, followed by the TBVs.
04:23:10 [09:23]	OPS/ IDADS	"B" AFW control valve partially closed using handwheel. The operator thought he had completely closed the valve at this point. Feed flow to the "B" OTSG, however, has decreased by about 60%. This increased flow through the "A" AFW valve.
04:25:00	OPS/ IDADS	Operator begins to close "A" AFW control valve with its handwheel.
04:25:30	IDADS	Operator unisolates HPI pump SFAS recirc valves, opening the recirc path to the Makeup Tank.
04:26:15	IDADS	CR/TSC Essential HVAC train "B" is secured.
04:26:20	IDADS	Pressurizer level returns on scale.

04:26:22	FUI/ IDADS	"A" AFW valve closed. This stops AFW flow to the "A" OTSG. Operator believes the valve is only 80% closed, but cannot close it any further by hand. Leaves to locate valve wrench.
04:26:47	IDADS	Pressurizer level rising rapidly. Subcooling margin is 170°F. Operators start to throttle HPI injection valves to minimize RCS repressurization.
04:28:00	IDADS	Operators stopped the "C" RCP per core lift requirements. RCS temperature is 410°F. Flow from four RCP's at low RCS temperatures, may give excessive lifting force to the core components (fuel assemblies).
04:28:43	IDADS	RCS letdown flow is reestablished to help control pressurizer level. Letdown flow is directed to the Makeup Tank. Letdown flow can be directed to either the Makeup Tank or the Flash Tank. It is normally lined up to the Makeup Tank.
04:28:45	CALC/ IDADS	Makeup Tank level goes offscale high. Makeup Tank relief valve lifts and discharges to the Flash Tank.
04:28:59	IDADS	Operators stop "A" HPI pump. The RCS pressure has peaked at 1616psig. RCS temperature at this time is 422°F.
04:29:40 [15:53]	OPS/ IDADS IDADS	Operator uses valve wrench on "A" AFW valve. Manual operator is damaged. Valve reopens. Operator calls Control Room and is told to close downstream manual isolation valve FWS-063. AFW flowrate to the "A" OTSG is greater than 1300gpm, resuming the rapid cooldown.
04:29:45	IDADS	"C" and "D" HPI injection valves are closed to reduce the rate of increasing RCS pressure.

04:30	OPS	"A" side BWST suction valve, SFV-25003, is closed in an attempt to decrease Makeup Tank level. The operators assumed the running makeup pump would take a suction from the Makeup Tank if the BWST was isolated. They forgot that the Makeup Tank outlet valve (SFV-23508) closes on SFAS actuation. Their action isolated the suction of the Makeup pump, "A" HPI pump, and the "A" Decay Heat pump. (Note that the "A" HPI pump was not running at this time.) Operators shift letdown flow to Flash Tank. This action was taken to reduce the inflow to the Makeup Tank. HPI recirculation flow continued to the Makeup Tank. Shift Supervisor declares Unusual Event. A Senior Control Room Operator begins notification of the state, counties, and the NRC.
04:30:39	CALC	Makeup Tank relief valve begins to pass water. Flash Tank level begins to increase. The relief valve discharge is routed to the Flash Tank. The Flash Tank pumps start on high level but do not have the capacity to handle the letdown flow and "B" HPI pump recirculation flow. Flash Tank level began to slowly increase.
04:30:40	IDADS	Both OTSG operate levels off scale high. AFW flow continues to both OTSGs. "A": > 1500gpm, "B": 670gpm.
04:33:00	OPS/ IDADS	Started depressurizing RCS to return to condition outside PTS region using normal pressurizer spray.
04:33:20	OPS/ IDADS	Operator arrives at "B" AFW valve and finds it partially open. He closes it the rest of the way. Feedwater to the "B" OTSG has been stopped. This increases flow to the "A" OTSG to greater than 1700gpm.
04:33:40 [19:53]	IDADS	"A" OTSG is full up to the top of the steam shroud and begins to spill water into the steam annulus.
04:36?	OPS	Operator has attempted to close FWS-063 but it will not move, even with a valve wrench.
04:39:00	IDADS	RCS subcooling margin reaches peak of 201°F and begins to decline. RCS temperature was 390°F. RCS pressure was 1430psig. This is approximately 800psi into the PTS region.

04:40:00 [26:13]	OPS/ IDADS	ICS power is restored. Operator closed S1-S2 switches in ICS cabinet 3. Upon restoration of power, ADV demands went to 100%, TBV demands went to 0%, "A" AFW valve demand went to 100%, B AFW valve demand went to 0%. Operators reduced the 100% demand signals to 0%. The "A" AFW valve responded and closed. Operators regained control of ADVs, TBVs, and AFW flow control valves. The operators immediately reduced the demand signals to these valves to 0%. Since the ADVs, were held closed by the manual handjacks, they could not respond to their demand signals and did not lift. The "A" AFW control valve was closed by the operator action. All feedwater flow to both OTSGs was then stopped and the RCS began to heat up. The lowest RCS temperature of 386°F was reached and at this time RCS pressure (now 1413psig) is being reduced to achieve conditions outside the PTS region. The RCS had cooled down 196 degrees in 26 minutes.
04:40:10	IDADS	Minimum OTSG pressures reached. "A": 221psig, "B":202psig.
04:41?	OPS/ IDADS	Operator calls control room and informs them that FWS-063 is stuck open. Told to disengage the handjack from the "B" AFW control valve. Told other operators to unisolate TBVs.
04:41:10	IDADS	"A" OTSG level goes below steam shroud. Main steam flow is continuing to supply AFW pump P-318. An estimated 10,000 gallons of water spilled over the shroud and into the steamlines.
04:42:42	IDADS	Shutdown "B" HPI pump. Makeup pump continuing to run.
04:42:56	IDADS	Closed "A" and "3" injection valves. Pressurizer level is 130 inches and increasing. All sources of makeup water to the RCS have been closed except RCP seal injection.
04:43:30?	OPS	Operators noted loss of RCP injection flow. Seal flow slowly decreased as the "B" HPI pump coasted down. The operators were puzzled by the fact that seal flow was being lost with the Makeup Pump continuing to run. They were not yet aware of the Makeup Pump's isolated suction.
04:43:54	IDADS	Operator restarts "B" HPI pump to supply RCP seals. Seal flow returns to normal.
04:50:19	IDADS OPS	Operator stops "B" HPI pump again. The operators checked the RCP seal injection valves but could not find anything wrong with the valve lineup so the pump was shutdown again.

04:50:30	IDADS	Operators again notice loss of seal flow and restart "B" HPI pump.
04:52?	OPS	Senior operator collapses in front of control panel. He is moved to an office adjacent to the control room.
04:57	OPS	Started blowdown of both OTSGs to reduce level.
04:58	OPS	Both Emergency Diesel Generators are shutdown. They had run unloaded since the SFAS initiation.
04:58? [44?]	OPS	Operator in Control Room hears loud noise. The "MAKEUP/HPI PUMP LUBE OIL LOW PRESSURE" annunciator alarms. The operator notes that the Makeup Pump ammeter is reading only a fraction of normal running current. He realizes the pump has been damaged due to lack of suction. The pump became separated from the motor when the pump seized and broke the coupling between the speed changer and pump.
05:00:10	IDADS	Operator trips the Makeup Pump.
05:01:22	OPS/ IDADS	Operators open the Makeup Tank outlet valve (SFV-23508). Water from the tank spills out of the damaged Makeup pump seals and onto the pump room floor. Approximately 1200 gallons is spilled. The operators immediately deduce why the sharp drop in Makeup Tank level had occurred and reclose the outlet valve. The operators were aware that both the Makeup Pump and "A" HPI pump had their suctions isolated. These pumps share a common suction from the Makeup Tank and BWST. The "A" HPI pump had not been run since its suction was isolated. By taking this step, the operators were attempting to establish a supply path to the "A" HPI Pump.
05:05	IDADS OPS	Crossed out of B&W recommended PTS region; 3 hour soak is in progress. Ambulance is called for SRO.

05:08	OPS	Auxiliary Building Gas Radiation Monitors (R15002B and R15045) go into alarm.
		These instruments monitor air being exhausted from the Auxiliary Building basement. One of these rooms is the Makeup Pump room. The instrument is detecting gas from the mildly radioactive water spilled in this room.
05:09	IDADS	Both AFW pumps are stopped.
05:12:26 :12:40	IDADS	Both Decay Heat pumps are stopped.
		The "A" DHR pump had been running without a suction from the BWST since the closing of the BWST outlet valve (SFV-25003). The pump ran on its own recirculation and always had adequate NPSH. There was no damage to the pump.
05:13	OPS	Both Auxiliary radiation monitor alarms clear.
05:27?	OPS	Makeup Pump manual suction and discharge isolation valves are closed. This isolates the pump from the Makeup Tank/BWST suction header. The "A" HPI pump is now available for service, if needed.
05:29:04	IDADS	Operators stop the "A" RCP per procedure.
05:33 [79:]	SCHRT	Zone 20 fire alarm trips the Rad Waste Area Exhaust Fan (A-542A or A-542B). An interlock prevents restart of the fans until the fire alarm is manually reset.
		The SFAS initiation has closed the suction and discharge valves to the Reactor Building radiation monitor (R15001A&B). The compressor which normally pulls a sample from the Reactor Building overheats due to its lack of suction. The smoke from this compressor actuated the smoke detector in this room. The fans are located in an adjacent room within the same fire zone. If a fire is detected in either room, the fans are tripped. The fans draw air from the -20 and -47 foot areas of the Auxiliary Building, Spent Fuel Building and Chemistry labs. The Makeup Pump room is located in the -20 foot elevation of the Auxiliary Building.
05:35?	OPS	Control Room operators stop R15001A&B compressor.
05:38	SCHRT	Zone 20 fire alarm is reset and one of the exhaust fans is restarted. It runs for less than a minute and the fire alarm trips it once again.
05:40?	OPS	Main Steamline Failure Logic is inhibited. This action permits the normal feed flow pathway to the steam generators to be used.

05:48	OPS	TSC sprinkler system is isolated.
05:50	SCHRT	Zone 20 fire alarm is reset and one of the exhaust fans is restarted. It runs for only a few seconds when the fire alarm trips it again.
05:51?	CALC	Flash Tank begins to overflow into Waste Gas System.
05:54	OPS	Outside operator reports loss of security badge to Watch Commander.

The operator lost his security badge and radiation film badge around the time he was helping the licensed operator manually isolate the ADVs. He was busy assisting in the stabilization of the plant and did not report the loss until this time. He did not enter any radiologically controlled areas without his film badge.

05:59	SCHRT	Zone 20 fire alarm is reset and one of the exhaust fans is restarted. It runs for less than a minute and the fire alarm trips it once again.
06:01	SCHRT	Zone 20 fire alarm is reset and one of the exhaust fans is restarted. The smoke cleared from the area and the fan continues to run throughout the remainder of the shift.
06:02	OPS	Health Physics sample showed 1.5 MPC's of Xenon 133 and 135 in -20 foot level of Auxiliary Building.
06:04	OPS	Operator bypasses Safety Features signals.
06:11	OPS	"ICS OR FAN POWER FAILURE" alarms on main annunciator panel but immediately resets. There is no loss of power. No equipment response is noted.
06:14	OPS	"ICS OR FAN POWER FAILURE" alarms on main annunciator panel. ICS DC Power is lost. Operator immediately resets S1-S2 switches and ICS power is restored. Many ICS demands go to 100%. Operators reduce demands to 0% on ADVs, main and startup feedwater valves, and AFW control valves.

Operator response is quick enough to prevent another overcooling. Feedwater flow from the condensate pumps through the open startup feedwater valves added approximately 750 gallons of water to each steam generator before operators can complete the closing of the valves.

06:15	SCRTY	Security brings a spare visitor's security badge to the Control Room.

07:00	OPS	Senior operator released from hospital.
07:02	OPS	Site boundary release of .93 MPC from the Auxiliary Building stack is calculated. State OES is informed.
07:05?	OPS	Makeup Tank outlet valve is opened..
07:15	OPS	Started "A" HPI pump. Stopped "B" HPI pump. The "A" HPI pump took its suction from the Makeup Tank. Makeup Tank level began to decrease. This terminated the Flash Tank overflow into the Waste Gas System.
08:41	ECL	Unusual Event is terminated.

AUXILIARY BUILDING EXHAUST AIR UNITS (A-542A/B)
DECEMBER 26, 1985 TRANSIENT
ABRIDGED SEQUENCE OF EVENTS

TIME	EVENT	SOURCE
04:16:57	SFAS initiation (Rad Monitor R 15001 A/B isolated).	
04:43:50	Operator noted loss of RCP seal injection flow.(Loss of flow is indicative of Make Up Pump P-236 damage. Approximately 450 gallons of contaminated water from breached Make Up Pump spilled on floor in Auxiliary Building -20 ft. level.)	Operator personal statement
	About this time, an Equipment Attendant located in the Rad Monitor Room smelled smoke. He was dispatched to perform other duties before he could identify the source.	
05:03:49	Auxiliary Building stack hi rad alarm (caused by transport of airborne activity from Make Up Pump room).	Computer printout
05:10:09	Auxiliary Building stack hi rad alarm cleared.	Computer printout
05:36:04	Auxiliary Building Exhaust Air Handler (A-542B) tripped (probably tripped by smoke detector in Fire Alarm Zone 20). NOTE: R 15001 A/B is located in Fire Alarm Zone 20.	Computer printout
05:39:39	Attempted start of Auxiliary Building Exhaust Air Handler A (A-542 A) preempted by smoke detector.	Computer printout Operator personal statement.
05:40:28	Attempted restart of Auxiliary Building Exhaust Air Handler A(A-542A) preempted by smoke detector.	Computer printout, Operator personal statement.
05:47:	R 15001 A/B declared out of service--blower motor overheated.	Control Room log
05:59:07	Attempted restart of Auxiliary building Exhaust Air Handler A(A-542A) preempted by smoke detector.	Computer printout, Operator personal statement.
05:59:50	Attempted restart of Auxiliary Building Exhaust Air Handler A (A-542A) pre-empted by smoke dectector.	Computer printout, Operator personal statement.

AUXILIARY BUILDING EXHAUST AIR UNITS (A-542 A/B) (Continued)

TIME	EVENT	SOURCE
06:01:21	Attempted restart of Auxiliary Building Exhaust Air Handler A (A 542A) preempted by smoke detector.	Computer printout, Operator personal statement.
06:02:15	Started Auxiliary building Exhaust Air Handler	Computer printout

NOTE: A-542A/B Trip Devices

1. Smoke Detectors - automatic
2. Ground Fault - automatic
3. Control Room Pushbutton - manual

The Auxiliary building Air Handlers are not tripped by high radiation levels in the stack.

ATTACHMENT 2

GAC 85-1001R2

SACRAMENTO MUNICIPAL UTILITY DISTRICT

OFFICE MEMORANDUM

TO: J. V. McColligan   D. D. Whitney
R. W. Colombo   J. J. Field
S. J. Redeker

DATE: January 4, 1986
GAC 85-1001 Rev. 2

FROM: G. A. Coward *GA Coward*

SUBJECT: **TRANSIENT ANALYSIS ORGANIZATION, TROUBLESHOOTING, AND EQUIPMENT REPAIR FOLLOWING 12-26-85 TRANSIENT**

I am again establishing an organization for this analysis and startup similar to that used for other recent startups. The organization is outlined on the attached chart.

The troubleshooting and equipment repair guidelines used for the previous trips are to be used for this event and are repeated and expanded below. These guidelines apply to those Action List items marked with an asterisk with special requirements added for equipment on the Quarantined Equipment List. Other equipment may be added to the list by any of you; Dan Whitney will maintain and coordinate the master Action and Quarantine List.

It is important that we carefully plan and document all troubleshooting efforts that take place to fix the problems surrounding the event. An engineer is to be assigned to each specific problem, and will prepare an action plan for troubleshooting, evaluation, and correction of the problem. The Action List Coordinator will, as a minimum, approve all Action Plans. A format for the Action Plans is included. Specific equipment involved in the event has been placed on the Quarantined Equipment List which will require additional planning and monitoring during the troubleshooting and repair efforts, in addition to NRC Incident Investigation Team (IIT) review prior to commencement of any actual work. Information copies of MI's will be given to the IIT prior to beginning any troubleshooting.

The troubleshooting and investigative activity shall be preceded by an event evaluation and analysis to determine probable causes of failure or abnormal operation. The analysis and evaluation shall proceed as follows:

1. Collect and analyze known information/operational data for conditions prior to, during, and after the transient.

2. Review maintenance and surveillance/testing history, including recent modifications to relevant equipment, procedures, and operation.

P-1-601

3. Develop a summary of data, including 1 and 2 above, that support any proposed probable cause of failure or abnormal operation.

4. Based on above items, develop a probable root cause(s) of the problem.

5. Develop plans for testing the probable causes (i.e., checks, verifications, inspections, troubleshooting, etc.). In developing inspection and troubleshooting plans, care must be taken to insure, when possible, that the less likely causes remain testable. When planning troulbeshooting activity, try to simulate as closely as practical the actual conditions under which the system or component failued to operate properly.

6. Document the action above in the Troubleshooting Action Plan.

7. Prepare the MI's to implement the Troubleshooting Action Plan.

It is very important that our investigations do not in any way result in the loss of any information due to disturbances of components or systems prior to establishing the cause of the failure. Investigations are to be conducted in a logical, well thought-out, and documented manner. The Maintenance Instruction (MI) shall be used to implement the Action Plan, and where appropriate, to accomplish the investigation/repair, with special attention being given to insuring that current drawings and vendor manuals are used. If vendors or vendor representatives are involved, their use must be documented and their role explained.

Action Plans will be approved by the Action List Coordinator prior to any work being started to insure that the Troubleshooting Action Plans conform to these guidelines, and that the described Maintenance Instructions (MI's) are appropriate and effective in resolving the concern. The Action Plans will include, as appropriate, the following:

1. Asterisked equipment on the Action List is to be treated as QA Class 1 for purposes of documentation, inspection, purchasing, and control.

2. Troubleshooting and repair activities are to be accomplished on separate Work Requests. Separate Action Plans are to be prepared for troubleshooting and repair for equipment on the Quarantined List.

3. Document on the Work Request all "as found" conditions. Visually inspect and document any missing, loose or damaged components, noting positions, abnormal environmental conditions, leaks, cleanliness, fluid conditions, jumpers, abnormal circuits, etc. Describe the overall condition or appearance. Whenever possible, use photographs to document "as found" conditions, and retain samples of fluids and residues for analysis.

4. Should a change in scope or direction be warranted, then a revised Action Plan must be processed. No materials or components are to be shipped off site unless specifically called for by the Action Plan.

5. It is the Action List "responsible" lead person's responsibility to assure that the investigative actions are appropriate, sufficient, properly defined, documented, and that data is preserved. This person shall also approve MI's prior to their use and insure that a copy is provided to the Action List Coordinator.

QC is to be involved in the troubleshooting and repair process. By copy of this memo, I am asking Andy Schwieger and his Quality organization to support the entire process.

Upon completion of the work described in the Troubleshooting Action Plan, an engineering report on the cause(s) is to be prepared. This report is to justify and explain the cause(s), and include a description of the concern, statement of cause, and justification for this conclusion based upon the findings. A format for this report is attached. An Action Item Closure Report is to be developed and submitted to the Action List Coordinator following completion of both troubleshooting and repair activities for review and acceptance. A format for this report is attached.

In all cases, applicable procedures must be followed. The requirements of this memorandum must be communicated to all involved personnel to avoid any confusion or misunderstandings during this investigative and repair period.

Attachments (5)

cc:

N. Brock	R. Lawrence
H. Canter	V. Lewis
S. Crunk	C. Linkhart
R. Dieterich	B. Rausch
J. Eckhardt	R. J. Rodriguez
B. Fraser	L. G. Schwieger
J. Jewett	B. Spencer
L. R. Keilman	J. Sullivan
F. Kellie	R. Wichert

TROUBLESHOOTING ACTION PLAN
(Format, Rev. 2)

ACTION LIST ITEM NUMBER ______________________

ACTION LIST DESCRIPTION ______________________

____

QUARANTINED EQUIPMENT LIST ITEM NUMBER __________

RESPONSIBILITY OF ______________________

PREPARED BY ______________________ DATE __________

DESCRIPTION OF ISSUE:

(Describe purpose, problem, or reason for investigation)

SUMMARY OF INFORMATION SUPPORTING PROBABLE CAUSE:

(Include analysis/review of operational data, procedures, etc.)

REVIEW OF MAINTENANCE, SURVEILLANCE TESTING AND MODIFICATION HISTORY:

(Describe results of review.)

POTENTIAL ROOT CAUSE(S):

(Identify primary and alternative hypothesis(ses) as appropriate.)

OUTLINE OF TROUBLESHOOTING PLAN:

(Describe scope, content, and objectives of planned Maintenance Instructions sufficient to resolve potential root causes.)

APPROVED BY ______________________ DATE __________
Action List Coordinator - SMUD

RELEASED FOR IMPLEMENTATION BY ______________________ DATE __________
Action List Coordinator - SMUD

ENGINEERING REPORT
ON ACTION ITEM
(Format, Rev. 2)

ACTION LIST ITEM NUMBER ______________________

ACTION LIST DESCRIPTION ______________________________________

____

QUARANTINED EQUIPMENT LIST ITEM NUMBER ____________

RESPONSIBILITY OF ______________________

PREPARED BY ______________________________ DATE ____________

DESCRIPTION OF ISSUE:

SUMMARY OF INFORMATION SUPPORTING PROBABLE CAUSE:

(Similar to section in Troubleshooting Action Plan.)

REVIEW OF MAINTENANCE, SURVEILLANCE TESTING AND MODIFICATION HISTORY:

(Similar to section in Troubleshooting Action Plan.)

IDENTIFIED ROOT CAUSE:

(Present results of analysis and/or troubleshooting.)

CONCLUSIONS AND/OR JUSTIFICATION FOR ROOT CAUSE:

APPROVED BY ______________________________ DATE ____________
Action List Coordinator - SMUD

REPAIR ACTION PLAN
(Format, Rev. 2)

ACTION LIST ITEM NUMBER____________________

ACTION LIST DESCRIPTION____________________

____

QUARANTINED EQUIPMENT LIST ITEM NUMBER____________________

RESPONSIBILITY OF____________________

PREPARED BY____________________ DATE____________________

REPORT OF "AS FOUND" CONDITIONS FROM TROUBLESHOOTING:

(Describe findings and results of troubleshooting efforts to establish root cause(s).)

DESCRIPTION OF REQUIRED REPAIRS:

(Describe purpose, problem, or reason for repair actions.)

OUTLINE OF REPAIR PLAN:

(Describe nature of repairs, procedure changes, modifications, or changes necessary to return item to service.)

APPROVED BY____________________ DATE____________________
Action List Coordinator - SMUD

RELEASED FOR IMPLEMENTATION BY____________________ DATE____________________
Action List Coordinator - SMUD

ACTION ITEM CLOSURE REPORT
(Format, Rev. 2)

ACTION LIST ITEM NUMBER ____________________

ACTION LIST DESCRIPTION ____________________

____

QUARANTINED EQUIPMENT LIST ITEM NUMBER ____________________

RESPONSIBILITY OF ____________________

PREPARED BY ____________________ DATE ____________________

A. DESCRIPTION OF ISSUE/CONCERN:

B. INVESTIGATIONS DONE:

(Attach: 1. Troubleshooting Action Plan.)
(2. Engineering Report on Action Item.)
(3. Repair Action Plan.)

C. CONCLUSIONS/EXPLANATION (should include Root Cause):

(Recap of results from "B".)

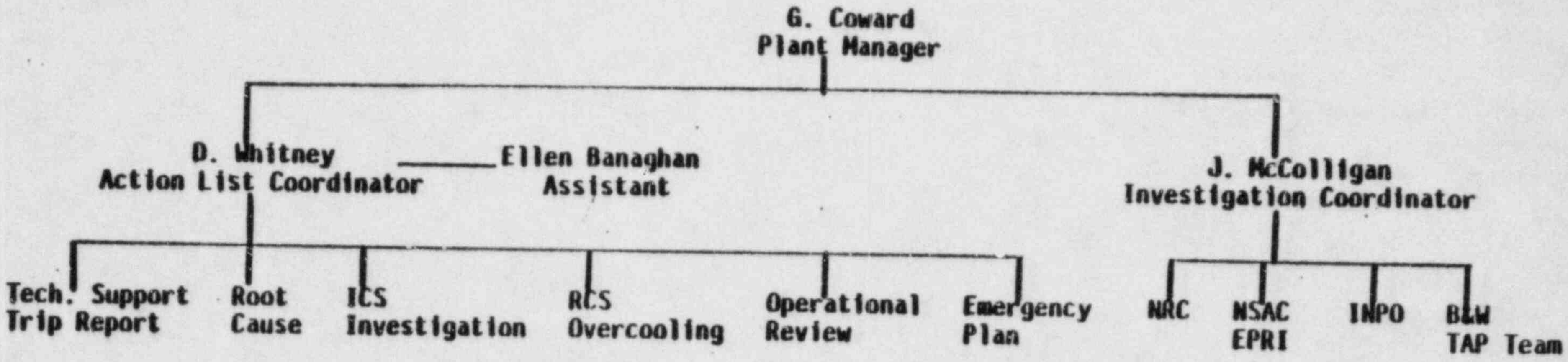
D. SHORT-TERM FOLLOWUP REQUIRED/RECOMMENDATIONS:

E. LONG-TERM RECOMMENDATIONS:

F. PROGRAMMATIC IMPLICATIONS:

REVIEWED AND ACCEPTED ____________________ DATE ____________________
Action List Coordinator - SMUD

TRANSIENT ANALYSIS ORGANIZATION



ATTACHMENT 3

ACTION LIST

LEGEND: SU = Startup Required
LT = Long Term
PE = Power Escalation
NA = Not Applicable
ST = Short Term

DECEMBER 26, 1985 TRANSIENT

- ACTION LIST -

**Closure Report Required to NRC
*Troubleshooting/Repair Guidelines Per GAC 85-1001 Apply

DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
1. Post Trip Report (AP.28)**	J. Field	ST	Compiling to PRC 2/22	TR No. 75	Grant Simmons Complete 2/15 Requested by NRC - IIT if available prior to 2/13.
a) TAP Team Assistance	R. Colombo	SU	Complete		
b) Sequence of Events	R. Wichert	SU	Complete		Rev. 3 issued 01-08-86.
				NOM 86-98	Transmitted Rev. 4 to NRC 2-7-86.
				NOM 86-99	Rev. 4 issued 02-07-86.
				NOM 86-112	Dist of Aux Bldg Exhaust 2-10-86.
				NOM 86-113	Transmittal of Aux Bldg Exhaust to NRC 2-10-86.
c) Aux FW System Initiation Response Report	G. Paptzun	SU	Complete		Approved 12-29-85.
d) Post Trip Shutdown Margin	G. Keney	SU	Complete		Always > 2 to 5 percent shutdown. Approved 12-31-85.
e) Aux FW Flow to OTSG Analysis	J. Field	SU	Closure Rpt Due 2/17		
f)** Main Steam Line Analysis	J. Field	SU			
1. Thermal Calc			Received 1/27		By Bechtel
2. Stress Calc			Received 1/27		By Bechtel
3. Closure Report			Due 2/17		Closure Report to cover walkdown

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DECEMBER 26, 1985 TRANSIENT

- ACTION LIST -

STATUS DATE 02-14-86
TIME 0800

**Closure Report Required to NRC
*Troubleshooting/Repair Guidelines Per GAC 85-1001 Apply

	DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
1. g)**	Minimum Pressurizer Level Calculation	J. Field	SU	Completion being Reviewed Due 2/18.	NOM 86-88	RV Head Bubble generated. B and W analysis complete 1-29-86. Closure Report issued 2-5-86.
					NOM 86-87	Transmittal to NRC, 2-5-86.
h)	"ICS Reliability Study"	R. Dieterich	SU	Complete		Input to RRG study.
i)**	Pressurizer Heater Operation	J. Field	SU	Complete	NOM 86-39	Troubleshooting plan approved 01-15-86, NOM 86-33. Heaters are OK.
j)**	Control Room Instruments which Fail on Loss of ICS Power	N. Brock	SU	Complete		Testing complete. Info to Ops due
					NOM 86-128	Closure Report Issued 2-13-86
					NOM 86-127	Transmittal to NRC 2-13-86.
k)**	Investigate report of SMOKE prior to event	E. Banaghan	SU	Complete	NOM 86-12	Closure Report approved 01-11-86. Revision due 2/7.
l)**	Primary to Secondary Leak Investigation	J. Field	SU	Closure Rpt. Due 2/25		Testing continuing.
m)	Security Interface	B. Spencer/D. Ross	ST	Closure Due 2/17		Complete actions per GAC 86-016

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1. m)	1. Review procedures to involve security in event.					SO to Integrate Ops/Security Revise AP 506.
	2. Response to Loss of Badges.					Badges in CASS
	3. Locked door control policy and procedures.					SO to Address
n)**	1. Main Steam Line Failure Logic	N. Brock	SU	Complete	NOM 86-107	Closure Report issued 2-8-86.
					NOM 86-114	Distribution to NRC. MSLFL worked properly.
	2. Change MSFL setpoint	V. Lewis	SU	Due 3/1/86	ECN R-0357	From 435.5 psig to 575 ± 5 psig. DBR in 3.f.5.
	3. Human Factors Improvements to MS Isolation Controls	V. Lewis	SU	Due 2/18		Paint/Tape/Labels to H2YS.
o)**	SPDS vs Strip Charts for OTSG operate level	J. Field	SU	Complete	NOM 86-92 NOM 86-91	To Distribution on 02-07-86. To NRC on 02-07-86.
p)**	Determine if transient was within USAR Design Basis	J. Field	SU	Complete	NOM 86-55	Requested by NRC/IIT telecon of 1-17-86. Issued to NRC 1-24-86.
					NOM 86-81	Distribution to Action List 02-01-86.

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STATUS DATE 02-14-86
TIME 0800

**Closure Report Required to NRC
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DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
1. q)** Discussion of no Operator action for 10 minutes on Loss of ICS	R. Dieterich	SU	Complete	NOM 86-83 NOM 86-82	Requested by NRC/IIT telecon of 1-31-86. To NRC 02-04-86. To Distribution.
r) Was Essential HVAC operable	J. Field	SU	Due 2/21		NRC Region V question of 02-03-86.
2. Human Factors/Root Cause Analysis					
a) Report	S. Crunk	ST	Closure Rpt. Due 2/15		Held up by ICS troubleshooting Closure Report forcast 2/24
b)** Human Factors Review	J. Jones	ST	Complete	NOM 86-48	Incorporate into Long Term Human Factors program. Issued 1-23-86 to NRC.
3. Determine Cause/Corrective Action for ICS Power failure.	N. Brock				
a)			Closed		Deleted. See 3c.
b)* ICS Power distribution investigation		SU	Complete	NOM 86-84	Removed buses from quarantine 02-04-86.

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DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
3. b) 1. C Inverter			Complete	NOM 86-37	Closure Report 01-16-86.
2. J Inverter and E and F buses			Complete	NOM 86-03	
NEW 3. Replace 30 amp breaker w/40 amp breaker on J Bus.			Closure Report due 2/21		
c)* ICS Equipment Investigation		SU	Complete	NOM 86-94 NOM 86-95 NOM 86-132	Engr Report to NRC 02-07-86. Distribution of Engr Report. Removed NRC Quarantine.
1. MI 1. Visual Inspections and Photographs			Complete		MI complete 1-21-86.
2. MI 2. Insitu Verification of Power Supply Setpoints			Awaiting Closure		MI complete 1-21-86. Outstanding NCR and MIDR.
3. MI 3. Load Carrying Capability of DC Power Supplies			Complete		MI complete 1-21-86.
4. MI 4. Insitu Verification of Power Supply Monitor Bus Voltage Setpoints			Awaiting Closure		MI complete 1-21-86. MIDR open pending closure of MI.2 MIDR. Identified drifting setpoint.
5.1 MI 5. Insitu Verification of S1 and S2 Trip Delay			Complete		0.12/0.15 sec vice 0.5 set Work complete 1-21.

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DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
3. c) 5.2 MI 5. ICS Repowering			Trblshooting	NOM 86-63 NOM 86-72	Revised Troubleshooting Plan to NRC 01-30-86. and NRC on 1-31-86.
6.1 MI 6. Bench Test Power Supply Monitor		SU	Awaiting Closure		MI being revised to include insitu testing.
6.2 MI-6. Remove and replace defective wiring for further troubleshooting	N. Brock	SU	Complete		Found defective crip/connection on +24VDC bus. Probable cause of ICS S1/S2 trip.
7. MI 7. Bench Test to Determine Hand/Auto Station			Awaiting Closure		Transferred to BWOG NRC Concurrence 1-26-86
d)* Repair Action Plan for ICS		SU	Due 2/18		In preparation.
e) Loss of ICS Procedure					
1. Equipment Input	N. Brock	SU	Complete	NOM 86-128	Recovery of ICS Power. ATOG review. RJR 86-19, Items 5 and 6. Closure Report issued 2-13-86
2. Procedure Development	B. Spencer	SU	Due 2/7		PRC Approval 2/5
3. ATOG Review of ICS	B. Ford	SU	Due 2/1		See item 10.a for action.

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3. f) Engineering review of proposed ICS Mods:					DBRs on all mods to be implemented requested by NRC-IIT.
1. BTU Limits	V. Lewis	ST	Complete	NOM 86-108	Engineering mod following receipt of B and W Analysis. Closure Report issued 2-8-86.
2. RCS Flow Noise	V. Lewis	ST	Complete	NOM 86-108	Engineering mod following receipt of B and W Analysis. Closure Report issued 2-8-86.
3. WR Recorder Alarms	V. Lewis	SU	Complete	ECN R-0245 NOM 86-109	ECNs released 1/27. Closure Report issued 2-8-86.
** 4. Modify ADV, TBV to fail closed on loss of ICS	V. Lewis	SU	Complete	ECN R-0357 Sub B NOM 86-90 NOM 86-129 NOM 86-130	Design Package by 01-27-86. DBR Info to NRC 2-7-86. Closure Rpt issued 2-13-86. Transmittal to NRC 2-13-86.
** 5. Provide remote manual control of AFW control valves independent of ICS	V. Lewis	SU	Complete	ECN R-0357 Sub A NOM 86-90 NOM 86-129 NOM 86-130	Consider including MFP speed and SU FW Valves. Design Package by 01-27-86. DBR Info to NRC 2-7-86. Closure Rpt. issued 2-13-86. Transmittal to NRC 2-13-86.

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3. f)** 6.	Evaluate multi-channel ICS/NNI power supply monitors to trip function	V. Lewis	SU	Closure due 2/14		Multi-channel monitors not appropriate. ECN will improve wiring.
** 7.	Evaluate ICS Trouble Annunciator and NNI	V. Lewis	SU	Revising Closure Rpt Due 2/14		Mgm't Review Draft 2/3.
** 8.	Provide protective fusing for AC/DC supplied loads	V. Lewis	SU	Complete	NOM 86-110 NOM 86-111	RJR 86-19, Item 2f. Closure Report: Transmitted to NRC 2-8-86. Distributed Closure Report 2-8-86.
** 9.	Investigate Signal Converson/NNI Interface	V. Lewis	SU	Investigating Report Due 2/14		
** 10.	Investigate ICS/NNI Power Supply Load current setpoint	V. Lewis	SU	Closure Due 2/21		Change 15 amp trip to 20 amp 1-27-86 memo from Williams. Process Standards change required. Formal calc package and ECN due 2/21.
11.	Remove Aux Steam Control from ICS	V. Lewis	ST	Closure Rpt. Due 2/21		Consider providing ± 24 VDC from another source.
g)	Can Loss of ICS Power cause failure to annunc. Loss of ICS Power	C. Linkhart	SU	Complete	NOM 86-133	RJR 86-19, Item 7. Closure Rpt. issued 2-14-86.

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DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
3. h) Motor Operated TBV isolation valves	V. Lewis	LT	Closure due 2/21		RJR 86-19, Item 8. Living Schedule Item.
4. MU Pump (P-236) failure					
a)* Repair/replace	M. Price	LT	Active	NOM 86-42	Troubleshooting Plan released for implementation 1-21-86.
b)* Cause	M. Price	SU	Closed	NOM 86-52	Issued 1-24-86.
c) 1. Revise SPs for running w/o MU Pump	J. Field	SU	Closure due 2/15		
2. Determine remaining service life on HPI motors.	V. Lewis	SU	Complete	NOM 86-79	Closure Report issued 02-01-86 4.8 years of continuous operation remain.
d)* Effect of SFV-25003 closure	J. Field	SU	Closure Rpt. Due 2/14	NOM 86-08	Troubleshooting Action Plan approved 01-10-86. Closed 02-07-86.
e) 1. Consider changing suction valve SFV-23508 from "Modulating" to "Seal-In."	V. Lewis	ST	Complete	WR 109713 NOM 86-80	Long Term change, requires Safety Analysis. Issued 02-01-86.

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DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
4. e) 2. Consider Valve interlock modifications	V. Lewis	ST	Complete		Interlock M/U Tank discharge valve w/BWST valve per RJR 86-19, Item 2d.
				NOM 86-80	Issued 02-01-86
f) Consider reroute of HPI/MU miniflow to prevent flooding MU Tank	V. Lewis	LT	Complete	NOM 86-106	Closure Report distributed 2-8-86.
g) Reconsider Design Basis SFAS of MU Tank Outlet	R. Dieterich	LT	Complete	NOM 86-51	USAR Design Basis requires SFAS closure.
h) Review of IE Info Notice 85-94, Miniflow on Safeguards Pumps	V. Lewis	SU	Complete	NOM 86-78	Closure Report issued 02-01-86. Review shows not applicable.
5.** Radiation Monitor R-15001					
a) Repair	M. Price	SU	Complete	WR107890 WR107891	Mechanical Work Complete (Replaced Seals)
b) Design evaluation	V. Lewis	SU	Complete	NOM 86-38	Closure Report 01-18-86.
6. RCS Overcooling					
a) 1. Tech Spec review	R. Colombo	SU	Complete	RWC 85-804	Reportable, Cooldown exceeded Tech Spec by 108°F

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DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
6. a) 2. Overcooling Calculation	R. Dieterich	SU	Due 2/15	RWC 86-55	Requested Official calculation to be completed by Engineering.
b) Analysis/evaluation					
1. B+W Calculations	J. Field	SU			
a. Preliminary Evaluation			Complete	NOM 86-14	No adverse effects on NSS.
b. Core Lift Report			Complete	NOM 86-14	0 to 21 percent core lift. No adverse effects.
c. Fatigue Report			Complete		02-05-86.
2. NSAC Calculations	J. Field	SU	Complete	NOM 86-13	Determined no adverse effects.
3. Closure Report	J. Field	SU	Complete	NOM 86-117	Distributed Closure Rpt. 2-11-86.
				NOM 86-118	Transmitted Closure Rpt. to NRC 2-11-86.
7.** Health Physics					
a) Health Physics Aspects/ Radiological Evaluation	F. Kellie	SU	Complete	NOM 86-43	Report issued 1-21-86.
				NOM 86-62	Transmitted to NRC 1-28-86
				NOM 86-96	Distribution of Rev. 1 Closure Report.
				NOM 86-97	Transmitted Rev. 1 to NRC, 02-07-86.

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DESCRIPTION		RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
7.** b)* 1.	Flooding/Filling of Waste Gas Header/ Surge Tank	J. Field	SU	Complete	NOM 86-17	Helium leak test of WGS completed. Action Plan approved 01-13-86. Work complete.
					NOM 86-85 NOM 86-86	Closure Report 2-5-86. Transmittal to NRC-V, 2-5-86.
** 2.	Waste Gas Compressor Investigation	M. Price	SU	Complete	WR 107918	Investigation shows no damage.
					NOM 86-119 NOM 86-120	Transmitted Closure Rpt. to NRC 2-11-86. Distributed CR 2-11-86.
c)	Usefulness of RJR-013	V. Lewis	ST	Complete	NOM 86-50	Long term on Living Schedule. Interim trend points on H3TMI. 8 weeks lead time on PROMS.
					NOM 86-65	Transmitted to NRC-IIT 01-31-86.
8.	Emergency Plan					
a)	EP Activation Analysis	R. Myers	SU	Closure Rpt due 2/18		Troubleshooting Plan approved 01-14-86. Incorporating Redeker comments of 2/12.
** b)	TSC Fire Sprinkler Actuation	V. Lewis	SU	Closure due 2/14	NCR S-5060 ECN R-0197	Test of effected annunciators done 2/5.

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		DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
8.	c)	State Auto-notification upon SFAS	R. Myers	SU	Closure Rpt. due 2/18		System worked properly. Incorporating comments in Reports.
9.		Training					
	a)	Sequence of Events Awareness	J. Mau	SU	Complete		BWOG Item 5. RJR 86-19, Item 9. All operators received training by 1/24.
						NOM 86-102	Distributed Closure Report 2-8-86.
	b)	New Procedures and Mods	J. Mau	SU	Interim Complete		Identification and scheduling of training. RJR 86-19, Item 9. BWOG Item 1. Review must be documented.
						NOM 86-102	Interim Closure Report distributed 2-8-86. Final due 3-7-86 following completion of training.
						NOM 86-103	Transmitted to NRC 2-8-86.
	c)	Short Term Training	J. Mau	ST	Interim Complete		Event Training 1 week/shift required.
	d)	Simulator Training	J. Mau	SU	Interim Complete		RJR 86-19, Item 10. Simulator training Required.
	e)	Command and Control Training	J. Mau	SU	Interim Complete		Training simultaneous with that of Item 9b.

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10.	Operational Review					
	a) Procedure Adequacy	B. Ford	SU	Due 2/14		S. Redeker preparing Summary, due 2/14. Includes ATOG Review per item 3.e.3.
	b) Communications	B. Ford	SU	Due 2/14		
	c) Operator Performance	B. Ford	SU	Due 2/14		
	d) Local/Manual Valve Operations	B. Ford	SU	Due 2/14		
**	e) Effect of Loss of ICS during SFAS	V. Lewis	SU	Complete	NOM 86-124	RJR 86-19, Item 4. Closure Rpt. issued 2-12-86.
11**	System/Component Response					
	a)* Aux Steam, PSV-36012A	J. Field	SU	To be Revised Due 2/25	NOM 86-11 R1	Closure Report revised 1-17-86. Revision to address loss of ICS causing Aux Steam overflow.

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11**	b)* 1)	Failure investigation of Aux FW, FV-20527, FV-20528	J. Field	SU	Closure Rpt. Due 2/21	WR107855/856	TS Action Plans issued per NOM 86-06/01-10-86. Repair
						NOM 86-74	Action Plan for FV-20527/8 issued 1-21-86 per NOM 86-41.
						NOM 86-76	Distribution of Engr. Report 2/1.
						NOM 86-89	Removed from Q-List per NRC, 2-6-86.
						NOM 86-101	Distribution of Repair Plan, 2-8-86.
	2)	FWS-063, FWS-064	J. Field	SU	Closure Rpt. Due 2/21	NOM 86-75	FWS-063 had rusted bearings. Others OK.
						NOM 86-77	Distribution of Engr. Report 2/1.
						NOM 86-89	Removed from Q-List per NRC, 2-6-86.
						NOM 86-100	Distribution of Repair Plan, 2-8-86.
	c)*	Main Steam, PSV-20544	J. Field	SU	Complete	NOM 86-11 R1	Closure Report revised 01-17-86 to NRC on 1-21-86.
						NOM 86-70	Valve returned to Operations per NRC 1-30-86.
	d)	RCP Seal Injection Flow	J. Field	SU	Complete	NOM 86-07	No maintenance required.
						NOM 86-66	Transmitted to NRC-IIT 01-31-86.

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11** e)	DHP Lo Oil in Auto Feeder	M. Price	SU	Complete	NOM 86-09	No problems found. Closure Report 01-10-86.
					NOM 86-67	Transmitted Closure Report to NRC 2-8-86.
f)*	Feedwater Heater Shell Reliefs opened	J. Field	SU	Closure Rpt. being revised Due 2/17	NOM 86-40	Reworked control valve seats.
g)*	SFV-23811 Position Indication	J. Field	SU	Complete	NOM 86-60	Z9020 indicated "Closed" although F9003 continued to indicate flow. MI issued, due 2/3.
					NOM 86-104	Closure Report distributed, 2-8-86.
					NOM 86-105	Closure Report transmitted to NRC, 2-8-86.
h)	PM program for manual isolation valves	M. Price	SU			Mgm't Review complete 2-10-86
	1.) Determine List of Values to PM/Stroke for SU.	B. Spencer	SU	Complete		Issued to Maintenance 2-13-86.
	2.) PM Lists of 102 Values	Price/Spencer	SU	Scheduled Due 3/5		Maintenance/Ops Joint Project.
	3.) Enter into PM Program	M. Price	ST	Due 4/1/86		Follow-up Scheduling.
i)	HPI Flow Balance Test, STP-085.	J. Field	SU	Complete	NOM 86-71	Report sent to NRC-IIT per H. Bailey requested and sent 01-31-86.

NEW

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12. Quarantined Equipment List*					
a) PSV-20544, Main Steam Code Relief	J. Field	SU	Off List	NOM 86-11 R1	Closure Report, Rev. 1 to NRC 1-21-86.
				NOM 86-70	Returned to Operations per NRC 1-30-86.
b) FV-20527, A-AFW Control Valve	J. Field	SU	Off List		Hold on repairs pending NRC will tear down valve internals and replace hand jack.
				NOM 86-44	Engineering Report to NRC 1-21-86.
				NOM 86-74	Revised Engr. Report.
				NOM 86-41	Hold on repairs pending NRC concurrence.
				NOM 86-89	Removed from Q-List 2-6-86.
c) FV-20528, B-AFW Control Valve	J. Field	SU	Off List		Will rebuild hand jack.
				NOM 86-44	Engineering Report to NRC 1-21-86.
				NOM 86-74	Revised Engr Report.
				NOM 86-41	Hold on repairs pending NRC concurrence.
				NOM 86-89	Removed from Q-List 2-6-86.
d) FWS-063, A-AFW Manual Isolation Valve	J. Field	SU	Off List	NOM 86-75	Action Plan and MIs completed 1-21-86.
				NOM 86-89	Removed from Q-List 2-6-86.

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12.	e) FWS-064, B-AFW Manual Isolation Valve	J. Field	SU	Off List	NOM 86-75 NOM 86-89	Action Plan and MIs completed 1-21-86. Removed from Q-List 2-6-86.
	f) ICS associated Power Supplies	C. Linkhart	SU	Off List	NOM 86-84	Closure Report completed. See items 3.b.1/2. Returned buses C, E, F, and J to Operations 2-4-86.
	g) ICS Power Distribution	N. Brock	SU	OFF List	NOM 86-94 NOM 86-132	Troubleshooting complete. Engr Report to NRC 02-07-86. Removed from Q-List 2-14-86. Switches S1 and S2 separately Quarantined.
13.	Procedure Changes Tracking	J. Field	SU	Draft List Published 1/27 Update due 2/5		Steve Luke,maintain list of all related procedure changes.
14.	Carryover Items from October 2, 1985 Event Action List:					
	1. Item II: Root Cause Final Report	S. Crunk	ST	Pending Due 2/24	NA	Closure Required

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**Closure Report Required to NRC
*Troubleshooting/Repair Guidelines Per GAC 85-1001 Apply

	DESCRIPTION	RESPONSIBILITY	SCHEDULE	STATUS	WR No./NCR/ ETC.	COMMENTS
14. 2.	Item VIb: Check common failure possibility on controller linkages.	N. Brock	LT	Complete	NOM 86-121	Original Commitment 12-15-85. Distributed Closure Report 2-11-86.
3.	Item VIIIi: Correct/ upgrade documentation on turbine trip block.	M. Price	LT	Overdue Due 2/19	CCL 85-0856	Original Commitment 01-01-86.
4.	Item VIIIm: Design improvements to reduce MSR Relief Valve leakage	V. Lewis	LT	Complete		To Living Schedule by 01-30-86.
5.	Item Xb: Perform ICS tuning at power	N. Brock	PE	Needs Closure Due 2/9	NA	Original Commitment 12-01-85.
6.	Item XIIIj: P-319 Bearing Failure Root Cause Report	S. Crunk	LT	Pending Due 2/24	NA	Need Final Report
7.	Item XIVb: Loss of Aux. Steam Event Root Cause Report	S Crunk	LT	Pending Due 2/24	CCL 85-0862	Original Commitment 12-15-85
8.	Item XVIb:	D. Comstock	LT	Due 3-1-86	NA	Procedure Adequacy Lessons Learned
9.	Item V.f: Establish criteria for process setpoint determination	V. Lewis	LT	In Progress		NEP 50205.14 is in final review 2-5-86. Will close by 3-15-86.

ATTACHMENT 4

REGION V CONFIRMATORY ACTION LETTER

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION V
1450 MARIA LANE, SUITE 210
WALNUT CREEK, CALIFORNIA 94596

DEC 26 1985

Docket No. 50-312
License No. DPR-54

CONFIRMATORY ACTION LETTER

Sacramento Municipal Utility District
P.O. Box 15830
Sacramento, California 95813

Attention: Mr. R. J. Rodriguez

Subject: Return of Rancho Seco to Power Operations

Gentlemen:

In a Confirmatory Action Letter that I provided to you on December 26, 1985, I stated it was my understanding that prior to return of Rancho Seco to power operations, you will conduct a root cause analysis of the reactor trip which occurred on December 26, 1985. In addition, and prior to return to power, you will provide the NRC a briefing of your assessment of the root cause and your justification as to why the Rancho Seco facility is ready to resume power operations.

Based upon additional conversations we held on December 26, 1985, and upon the fact that the NRC is sending an Augmented Inspection Team to Rancho Seco to review the December 26, 1985 reactor trip and resulting cooldown, I further understand that you will hold in abeyance any repair work planned on equipment that malfunctioned during the incident, such as dismantling existing evidence, until the SMUD and NRC inspection teams have had an opportunity to evaluate this event. This restriction does not cover any repair needed to place systems and equipment in service that are required to monitor and maintain safe conditions of the facility, i.e., the containment radiation monitor that failed during the course of the event.

If my understanding concerning your proposed actions, as summarized above, is not correct, you should promptly notify this office in writing.

Sincerely,

B. H. Faulkenberry
for J. B. Martin
Regional Administrator

cc: L. G. Schwieger, SMUD
G. Coward, SMUD
State of CA

8512310280 1p

ATTACHMENT 5

POST TRIP PARAMETRIC DATA

GRAPHS FOR:

RANCHO SECO REACTOR TRIP

DECEMBER 26, 1985

January 3, 1986

TABLE OF CONTENTS

IDADS COMPUTER POINT IDENTIFICATIONS

PT ID	PT SOURCE	DESCRIPTION	UNITS	INSTRUMENT RANGE		COMMENTS
E0603	LoveJoy	VOLTS DC BFP A SPEED TO LOVEJOY (M5)	VOLTS DC	0.000	10.000	
E0614	LoveJoy	VOLTS DC BFP B SPEED TO LOVEJOY (M5)	VOLTS DC	0.000	10.000	
F1660	FT-31802	AFW FLOW TO OTSG A	GPM	0.00000	1300.0	
F1663	FT-31903	AFW FLOW TO OTSG B	GPM	0.00000	1300.0	
F9000	FT-23603	MAKEUP FLOW	GPM	0.00000	250.00	
F9001	FT-23805	HPI LOOP A FLOW	GPM	0.00000	600.00	
F9002	FT-23806	HPI LOOP B FLOW	GPM	0.00000	600.00	
F9003	FT-23807	HPI LOOP A FLOW	GPM	0.00000	600.00	
F9004	FT-23808	HPI LOOP B FLOW	GPM	0.00000	600.00	
F9600	FI-20563	MFW FLOW B	MBL/HR	0.00000	6.5000	
F9601	FI-20535	MFW FLOW A	MBL/HR	0.00000	6.5000	
L1805	LT-20503C	OTSG A SU RNG	IN H20	0.00000	250.00	UNCOMPENSATED
L1807	LT-20504C	OTSG B SU RNG	IN H20	0.00000	250.00	UNCOMPENSATED
L9000	LT-23502A	MAKEUP TANK LEVEL	IN H20	0.00000	100.00	UNCOMPENSATED
L9005	LT-21503A	PZR LEVEL	IN H20	0.00000	320.00	UNCOMPENSATED
L9801	LT-20501	OTSG A LEVEL FULL RNG	IN H20	0.00000	600.00	UNCOMPENSATED
L9802	LT-20502	OTSG B LEVEL FULL RNG	IN H20	0.00000	600.00	UNCOMPENSATED
P9000	PT-21050	RCS PRES LOOP B	PSIG	0.00000	3000.0	
P9300	PI-20520B	OTSG PRESSURE B	PSIG	0.00000	1200.0	
P9301	PI-20519B	OTSG PRESSURE A	PSIG	0.0000	1200.0	
T9010	TY-21031C	RCS LOOP B HOT LEG TEMP	DEG F	120.00	920.00	
T9016	TI-21024A	RCS LOOP B COLD LEG TEMP	DEG F	50.000	650.00	

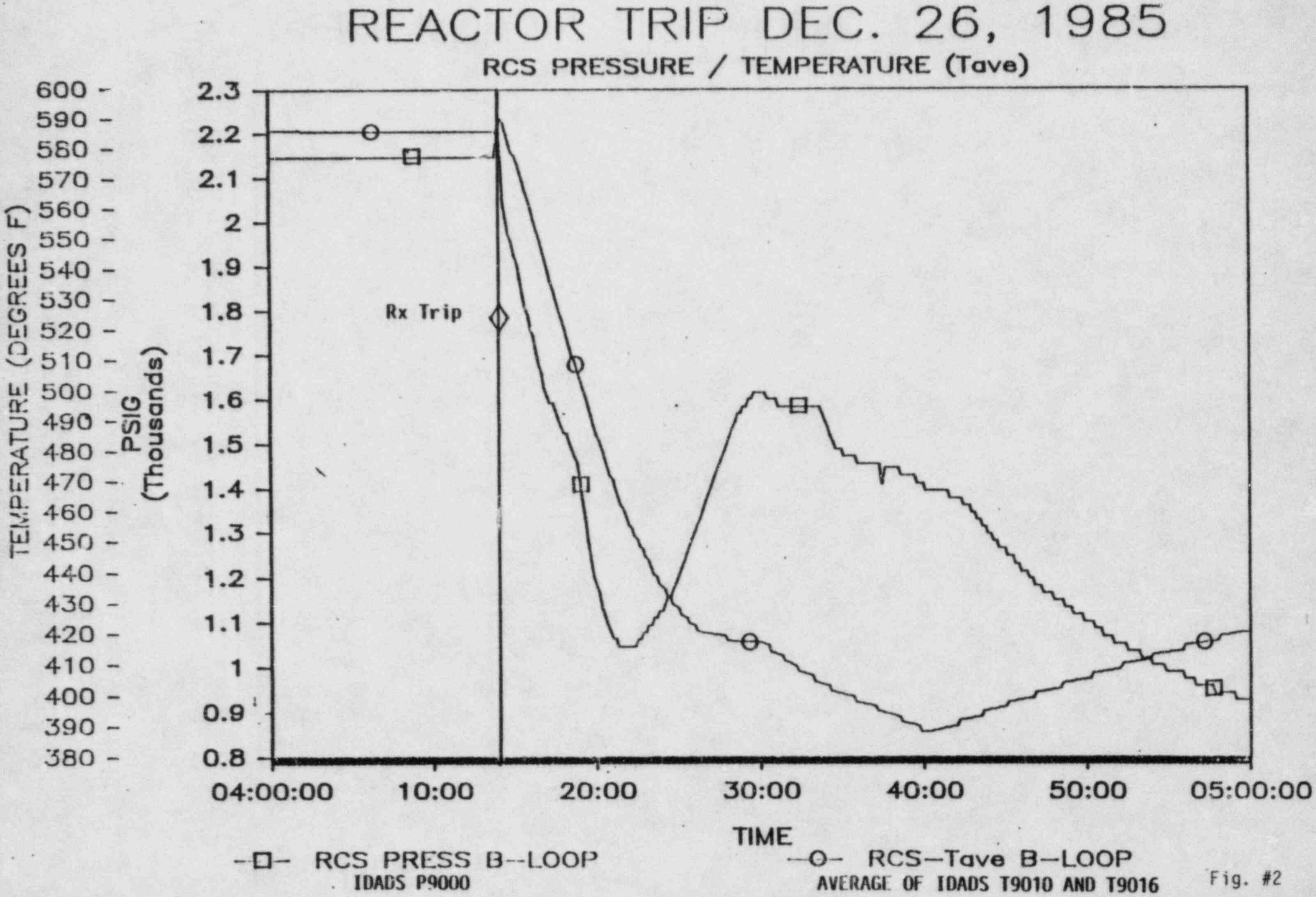

REACTOR TRIP DEC. 26, 1985
RCS PRESSURE / TEMPERATURE (Tave)
TEMPERATURE (DEGREES F)
PSIG (Thousands)
Rx Trip
TIME
04:00:00
10:00
20:00
30:00
40:00
50:00
05:00:00
-□- RCS PRESS B--LOOP
IDADS P9000
-O- RCS--Tave B--LOOP
AVERAGE OF IDADS T9010 AND T9016

Fig. #2

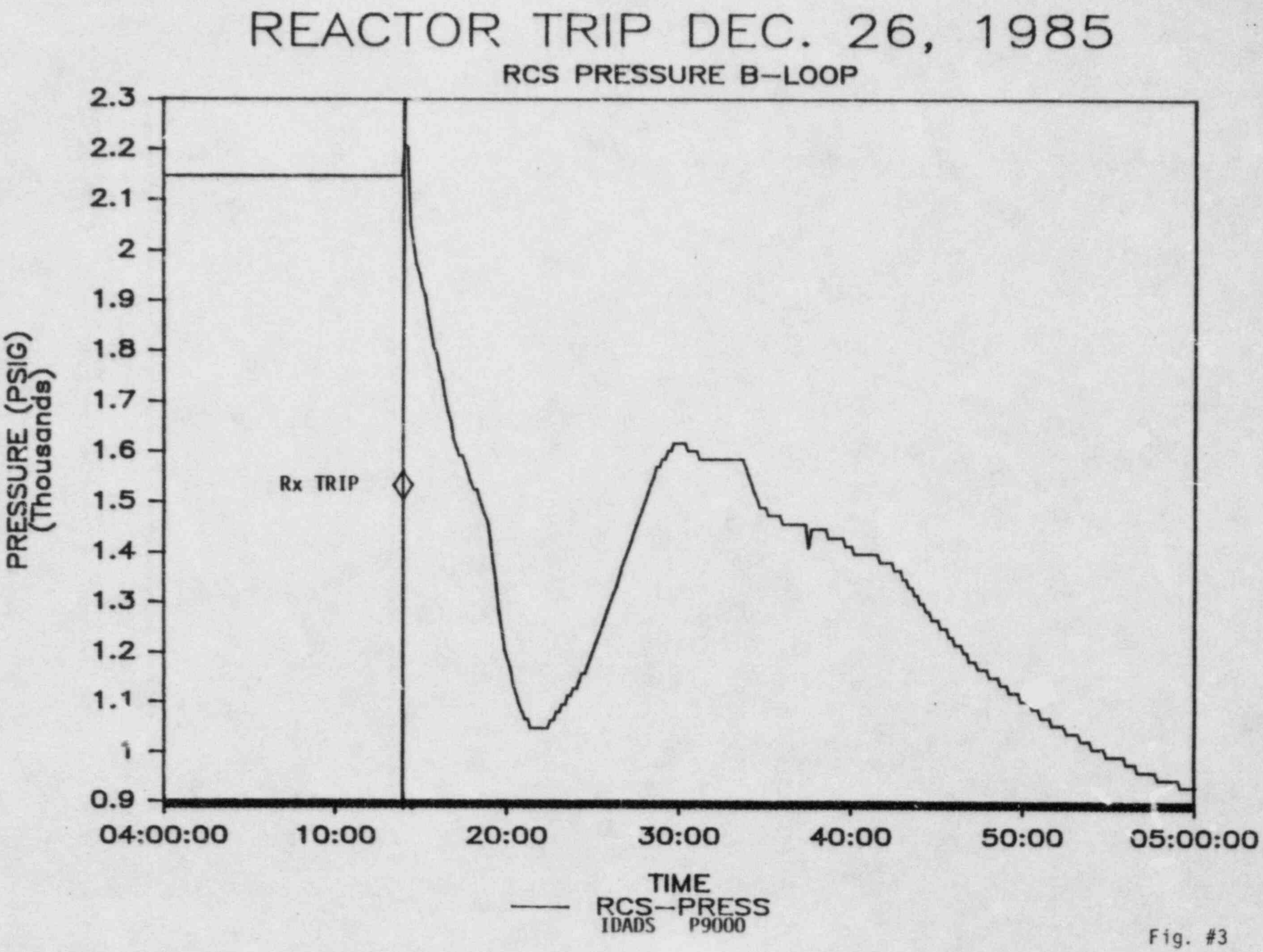

REACTOR TRIP DEC. 26, 1985
RCS PRESSURE B-LOOP
PRESSURE (PSIG)
(Thousands)
2.3
2.2
2.1
2
1.9
1.8
1.7
1.6
1.5
1.4
1.3
1.2
1.1
1
0.9
Rx TRIP
04:00:00
10:00
20:00
30:00
40:00
50:00
05:00:00
TIME
RCS-PRESS
IDADS P9000

Fig. #3

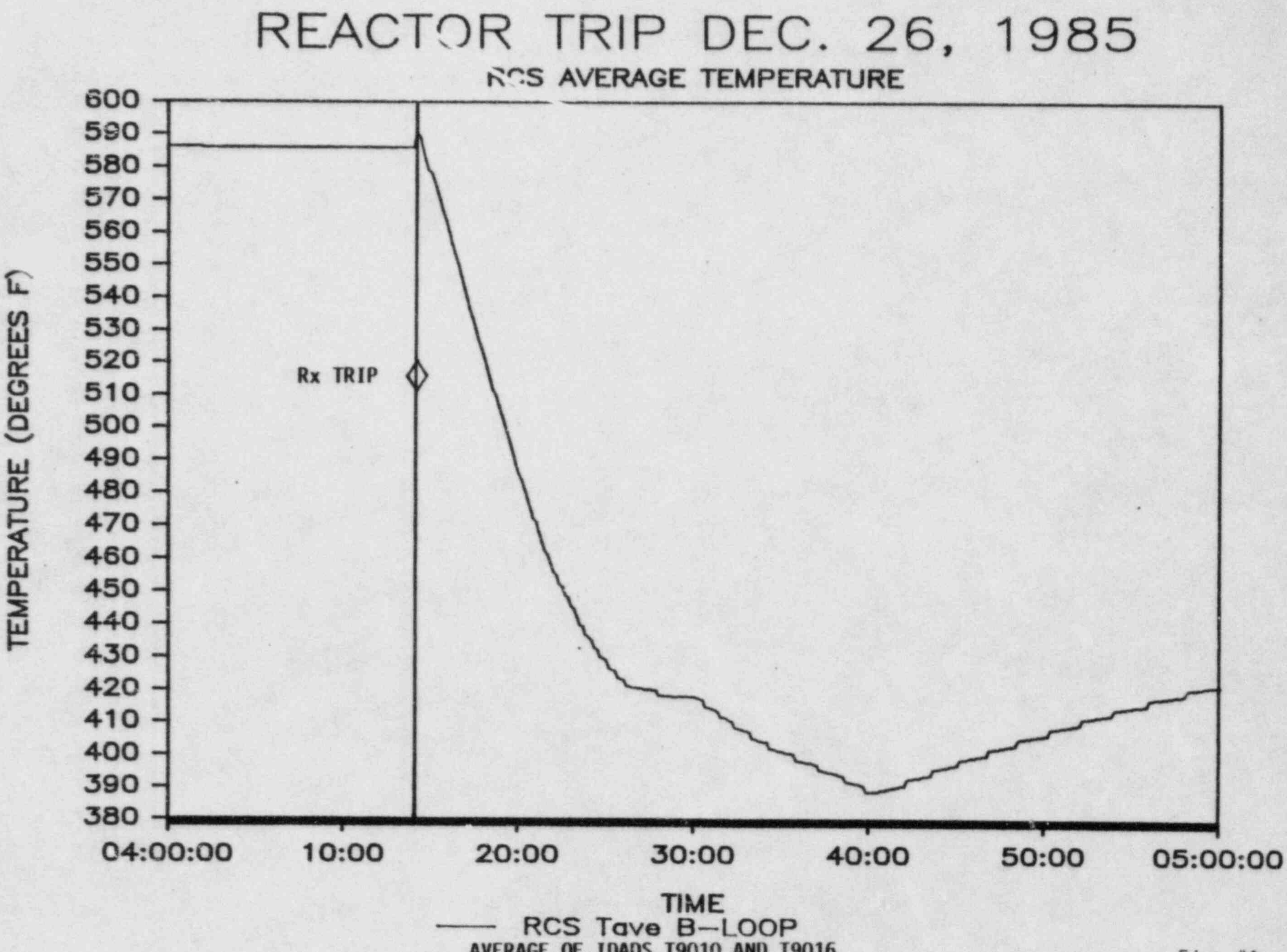
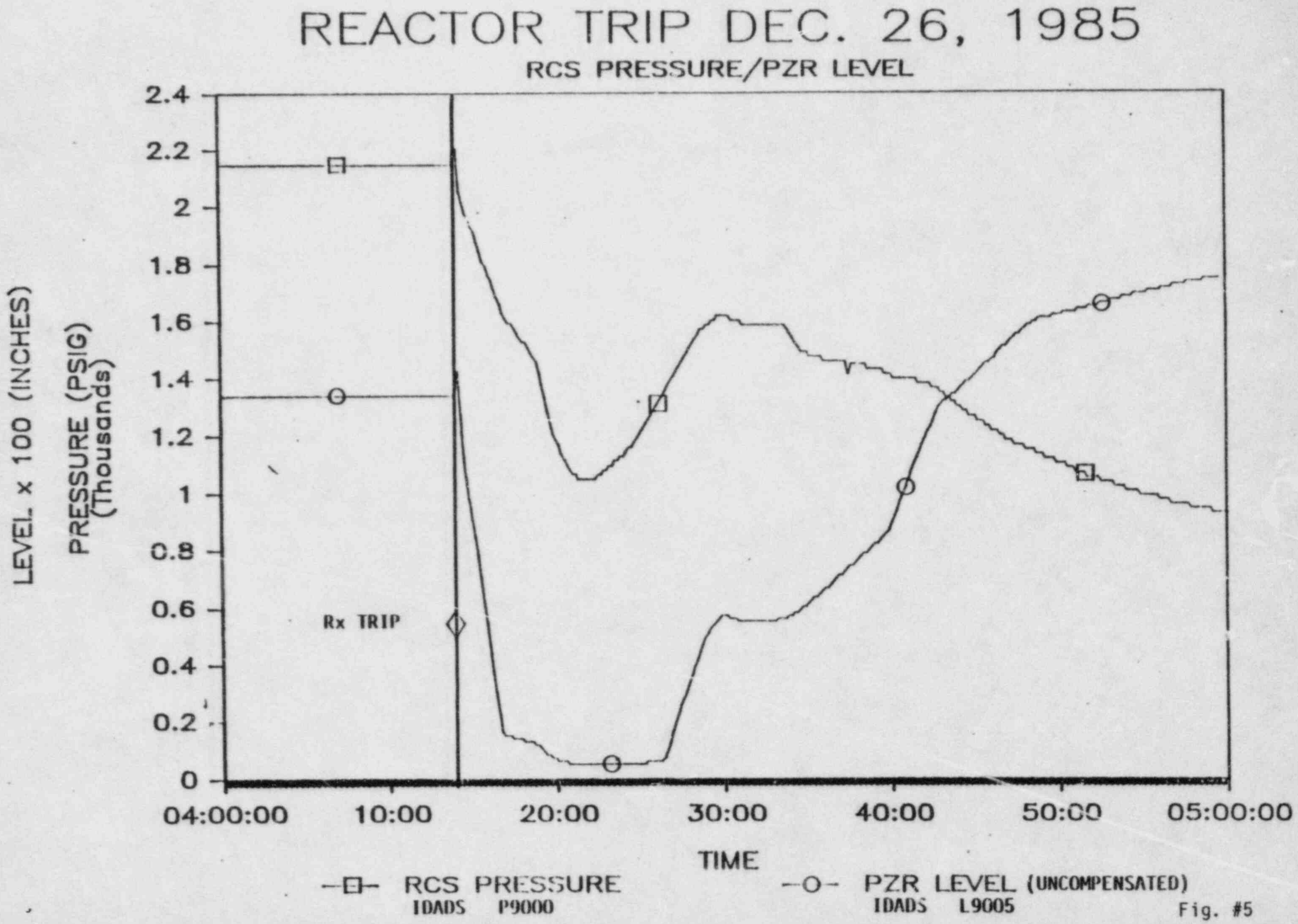
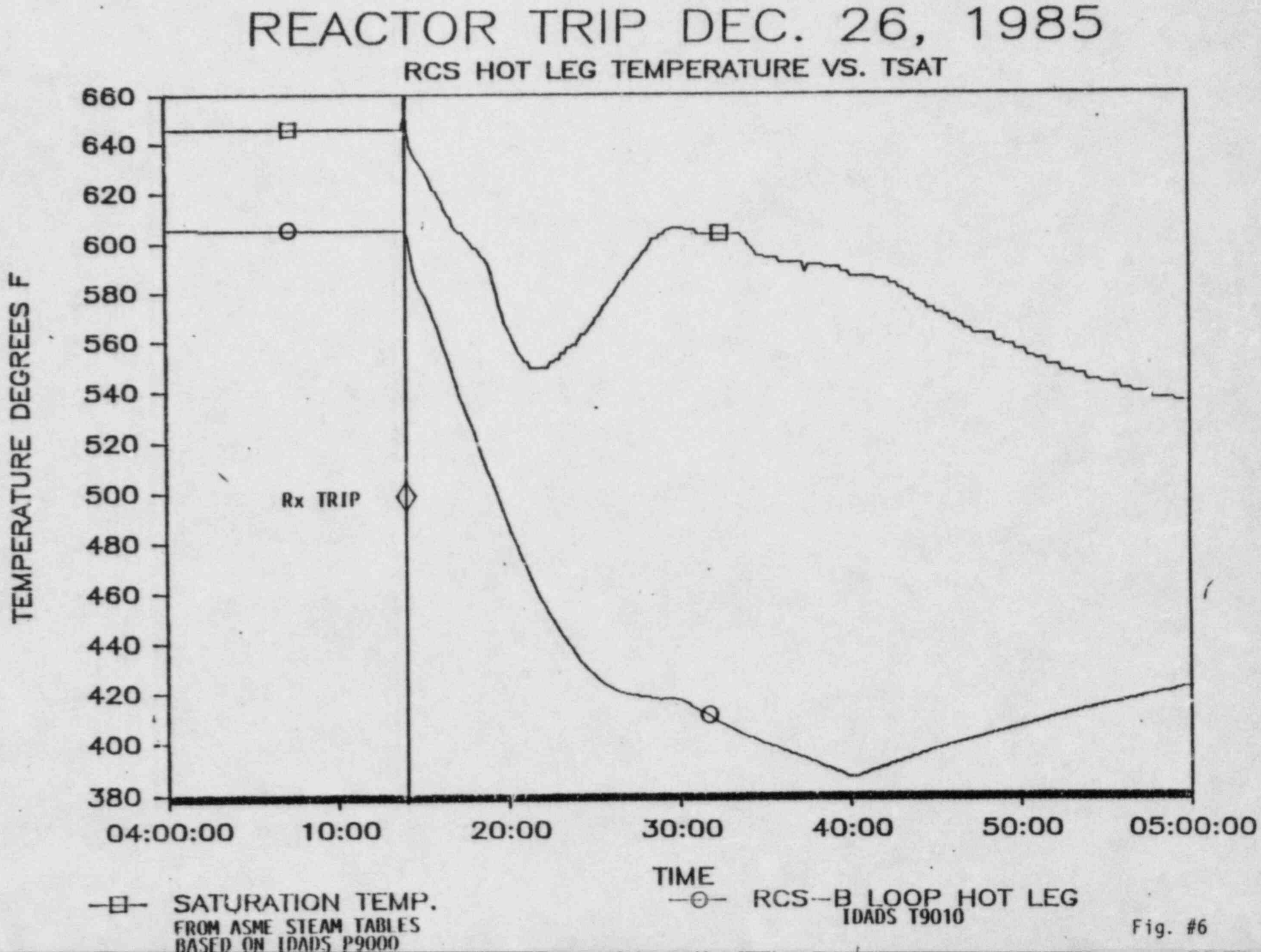

REACTOR TRIP DEC. 26, 1985
RCS AVERAGE TEMPERATURE
TEMPERATURE (DEGREES F)
600
590
580
570
560
550
540
530
520
510
500
490
480
470
460
450
440
430
420
410
400
390
380
Rx TRIP
04:00:00
10:00
20:00
30:00
40:00
50:00
05:00:00
TIME
RCS Tave B-LOOP
AVERAGE OF IDADS T9010 AND T9016

Fig. #4



REACTOR TRIP DEC. 26, 1985
RCS PRESSURE/PZR LEVEL
LEVEL x 100 (INCHES)
PRESSURE (PSIG)
(Thousands)
2.4
2.2
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
Rx TRIP
04:00:00
10:00
20:00
30:00
40:00
50:00
05:00:00
TIME
RCS PRESSURE
IDADS P9000
PZR LEVEL (UNCOMPENSATED)
IDADS L9005

Fig. #5



REACTOR TRIP DEC. 26, 1985
RCS HOT LEG TEMPERATURE VS. TSAT
TEMPERATURE DEGREES F
660
640
620
600
580
560
540
520
500
480
460
440
420
400
380
Rx TRIP
04:00:00
10:00
20:00
30:00
40:00
50:00
05:00:00
TIME
SATURATION TEMP.
FROM ASME STEAM TABLES
BASED ON IDADS P9000
RCS--B LOOP HOT LEG
IDADS T9010

Fig. #6

REACTOR TRIP DEC. 26, 1985

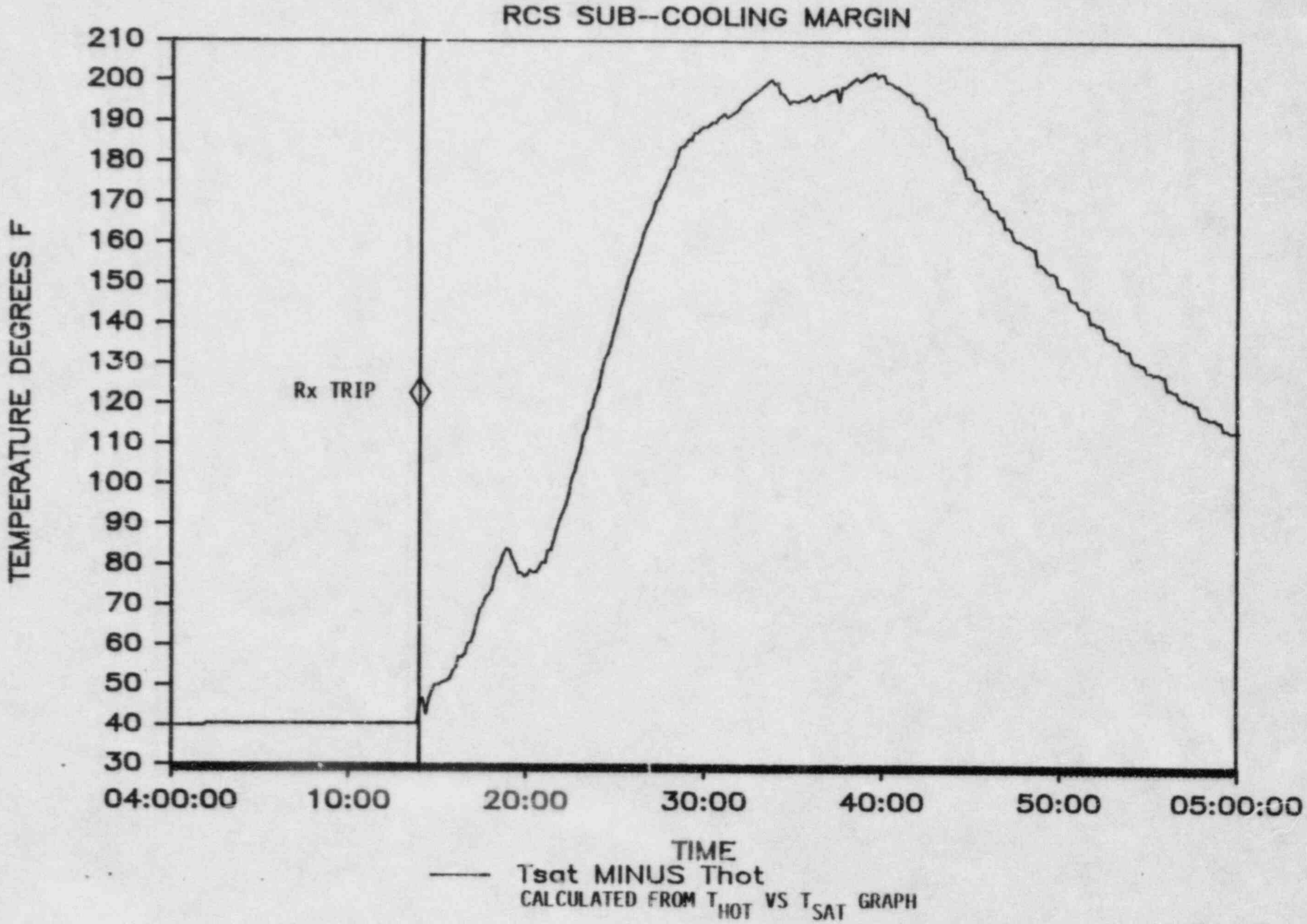


Fig. #7

REACTOR TRIP DEC. 26, 1985

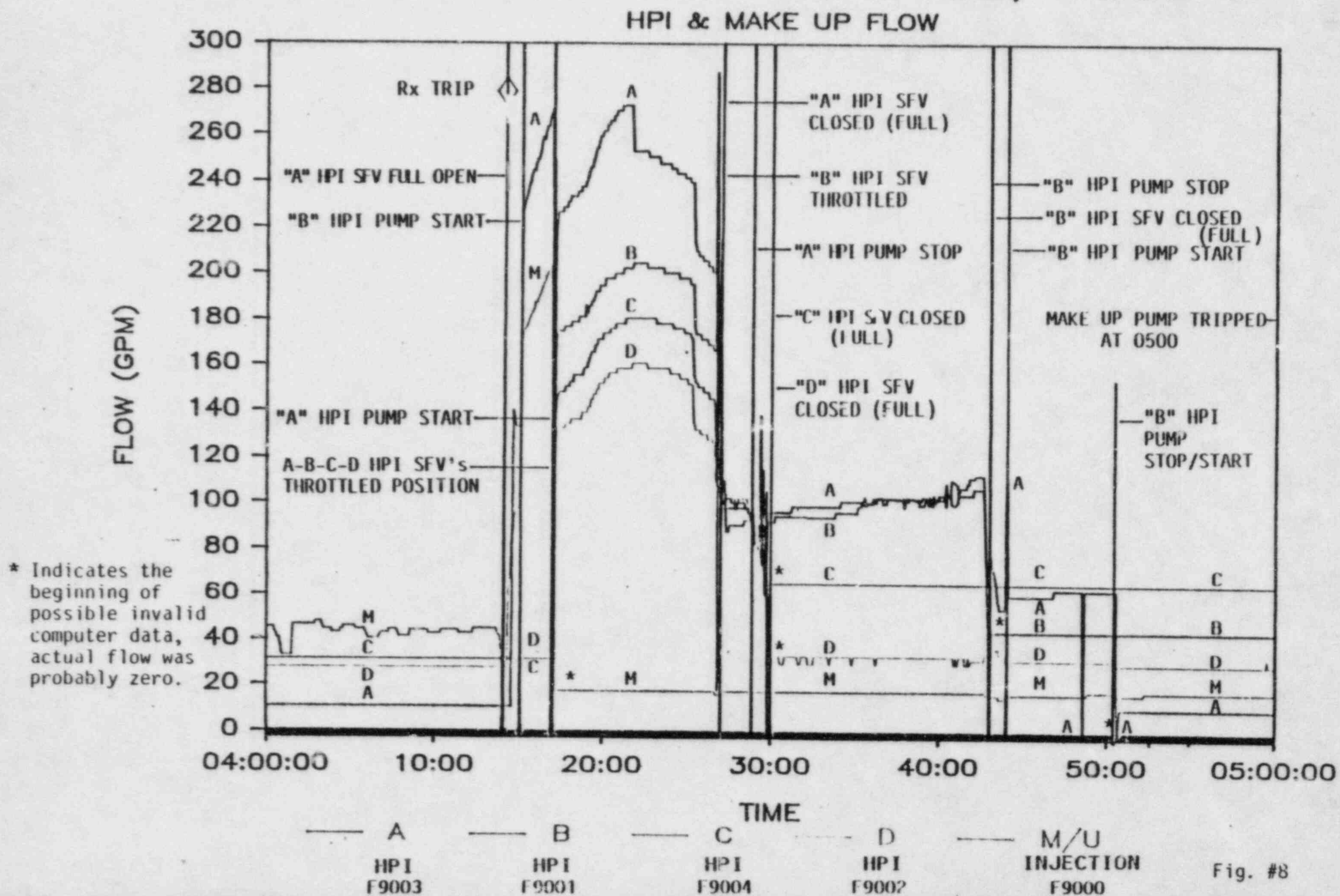


Fig. #8

REACTOR TRIP DEC. 26, 1985

TOTAL HPI & MAKE UP INJECTION

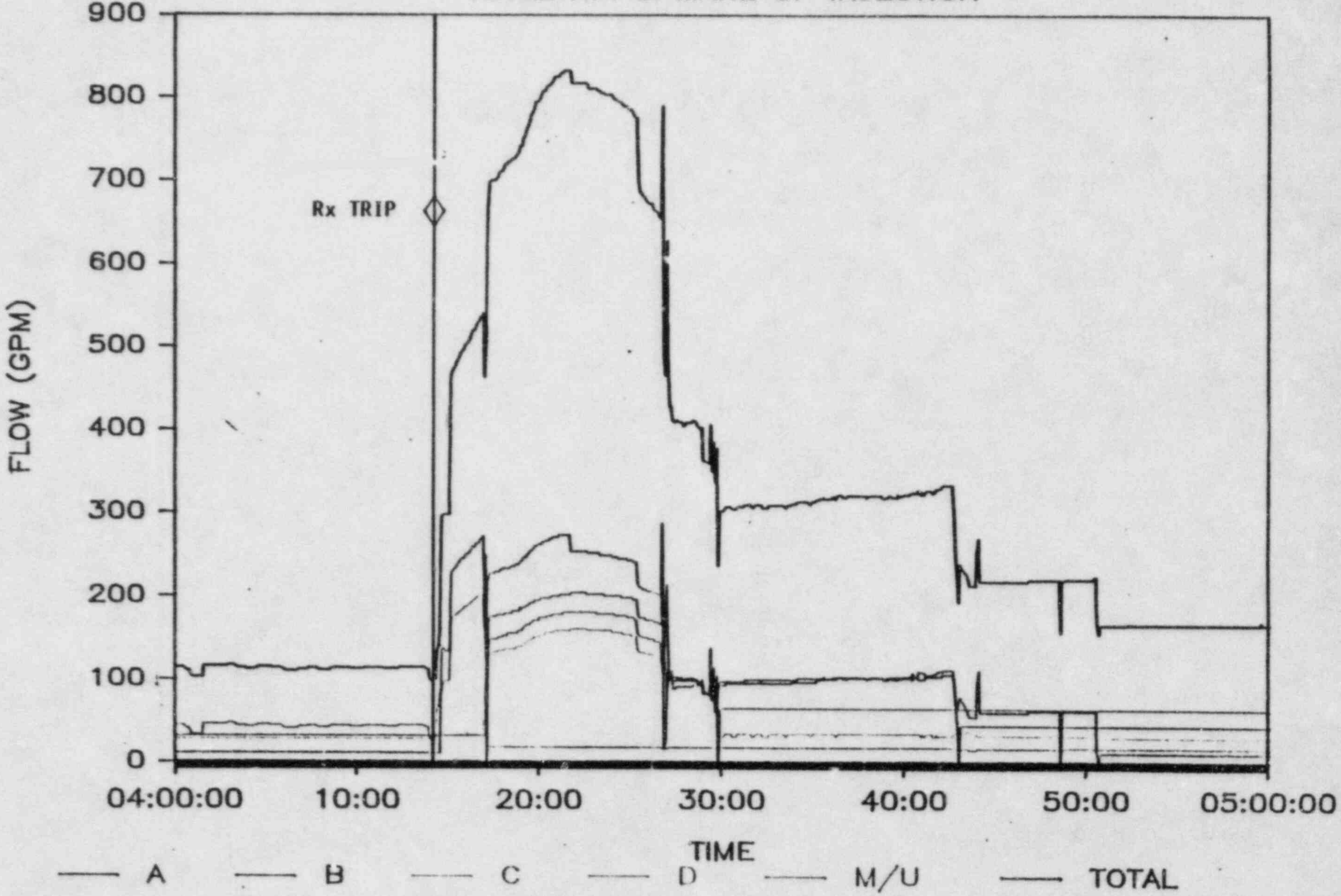


Fig. #9

HPI F9003 | HPI F9001 | HPI F9004 | HPI F9002 | INJECTION F9000 | INJECTION CALCULATED

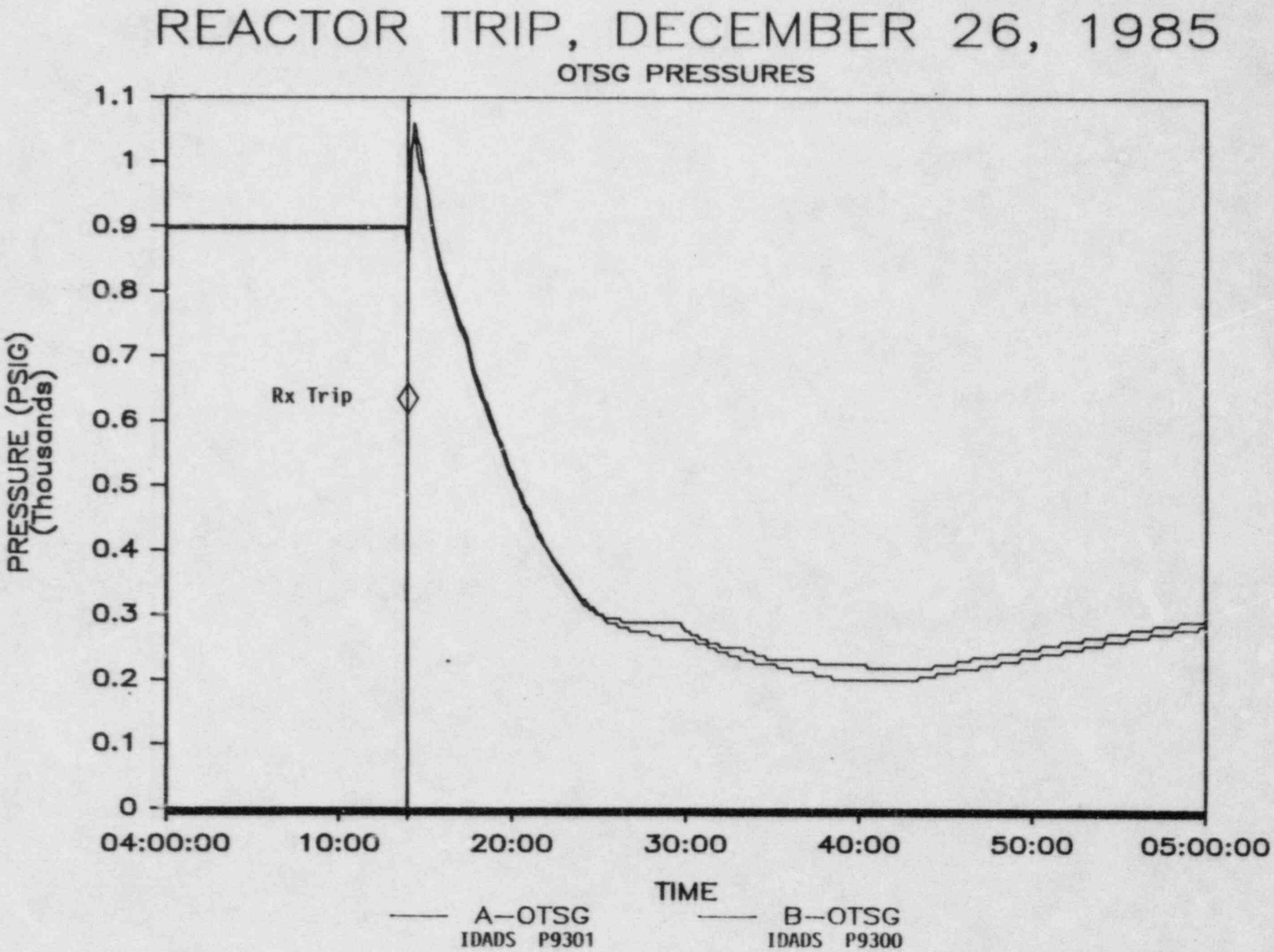

REACTOR TRIP, DECEMBER 26, 1985
OTSG PRESSURES
PRESSURE (PSIG)
(Thousands)
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
Rx Trip
04:00:00
10:00
20:00
30:00
40:00
50:00
05:00:00
TIME
A—OTSG
IDADS P9301
B—OTSG
IDADS P9300

Fig. #10

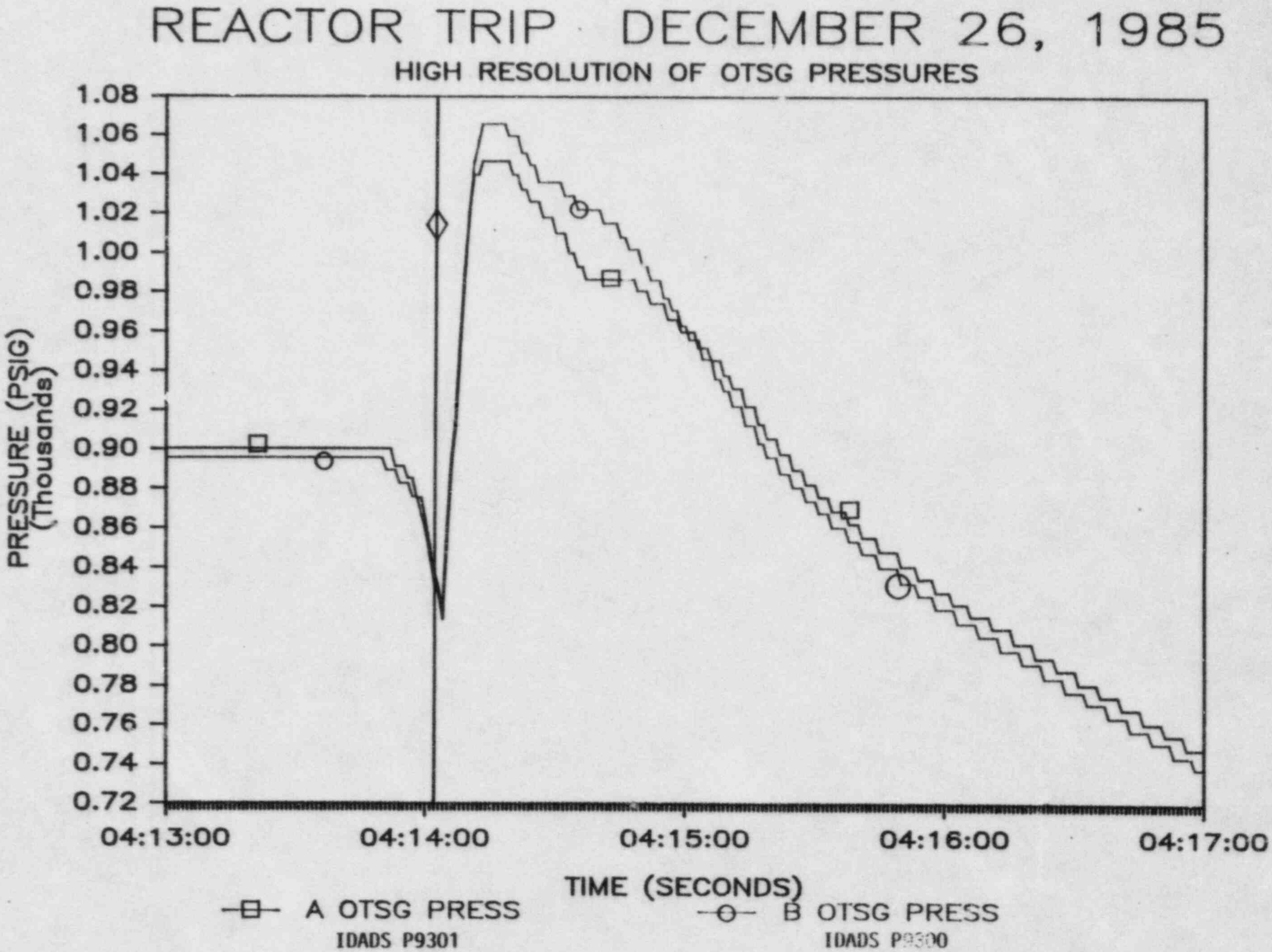

REACTOR TRIP DECEMBER 26, 1985
HIGH RESOLUTION OF OTSG PRESSURES
PRESSURE (PSIG) (Thousands)
1.08
1.06
1.04
1.02
1.00
0.98
0.96
0.94
0.92
0.90
0.88
0.86
0.84
0.82
0.80
0.78
0.76
0.74
0.72
04:13:00
04:14:00
04:15:00
04:16:00
04:17:00
TIME (SECONDS)
A OTSG PRESS
IDADS P9301
B OTSG PRESS
IDADS P9300

Fig. #11

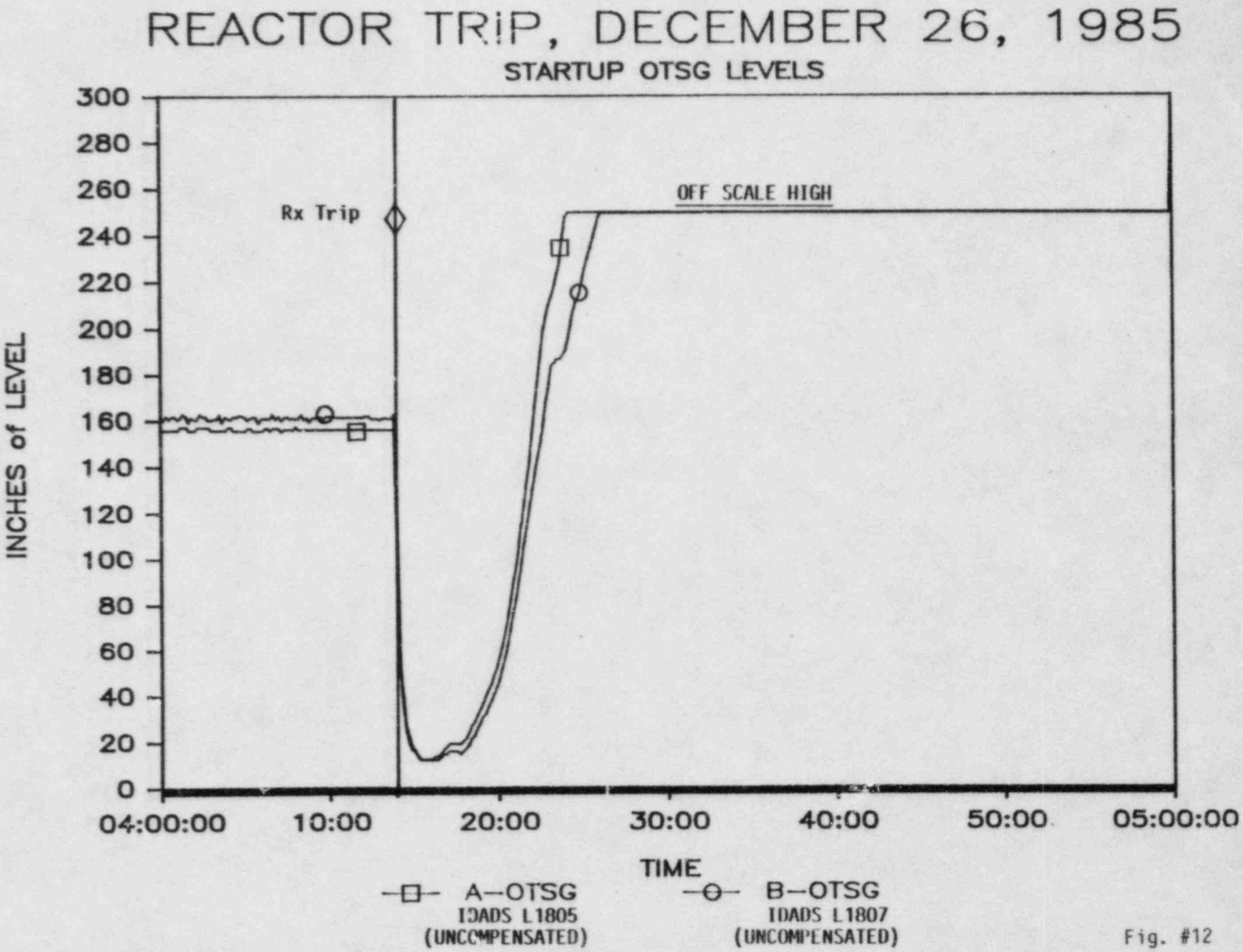

REACTOR TRIP, DECEMBER 26, 1985
STARTUP OTSG LEVELS
INCHES of LEVEL
300
280
260
240
220
200
180
160
140
120
100
80
60
40
20
0
Rx Trip
OFF SCALE HIGH
04:00:00
10:00
20:00
30:00
40:00
50:00
05:00:00
TIME
A—OTSG
IDADS L1805
(UNCOMPENSATED)
B—OTSG
IDADS L1807
(UNCOMPENSATED)

Fig. #12

REACTOR TRIP DEC. 26, 1985

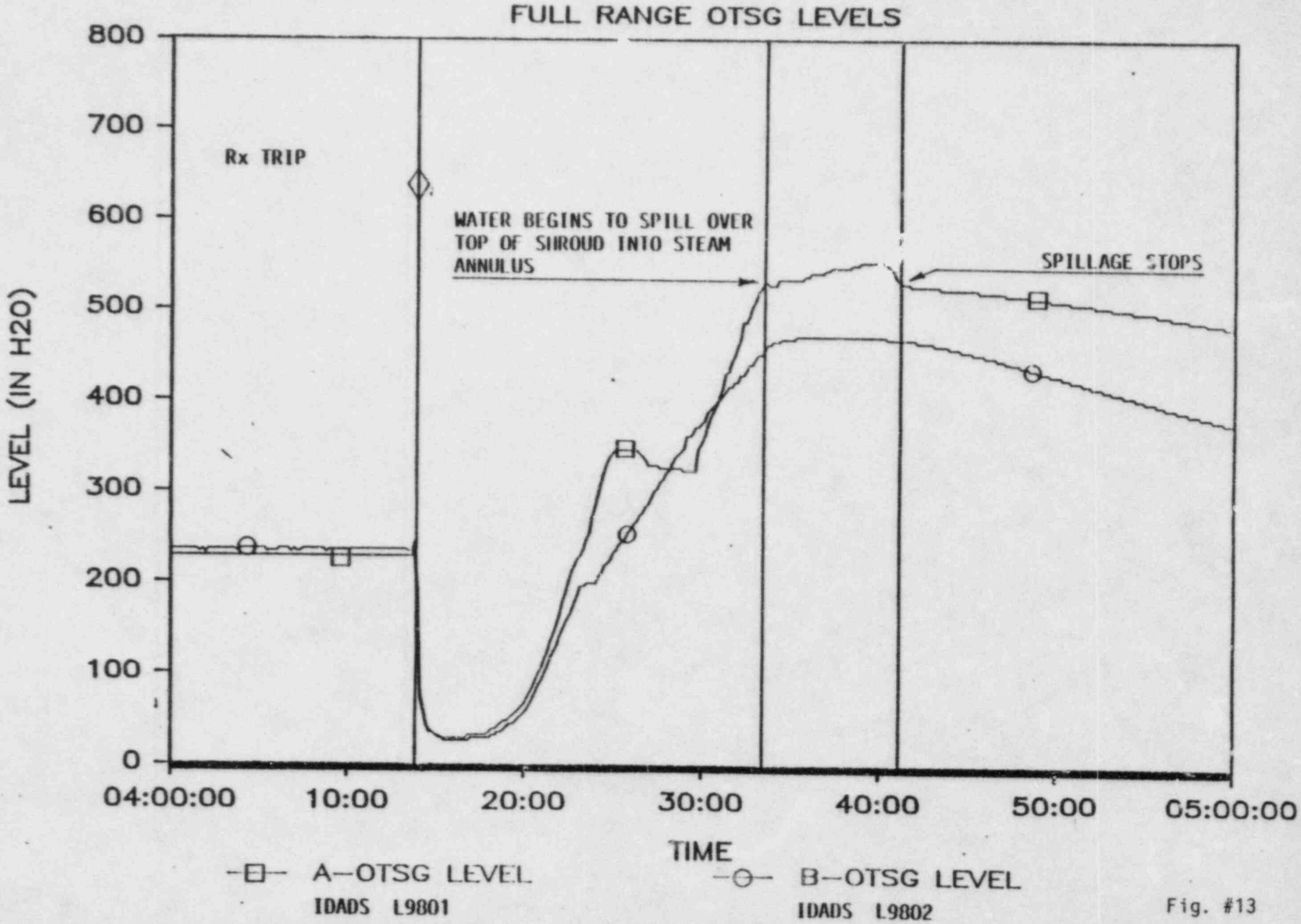
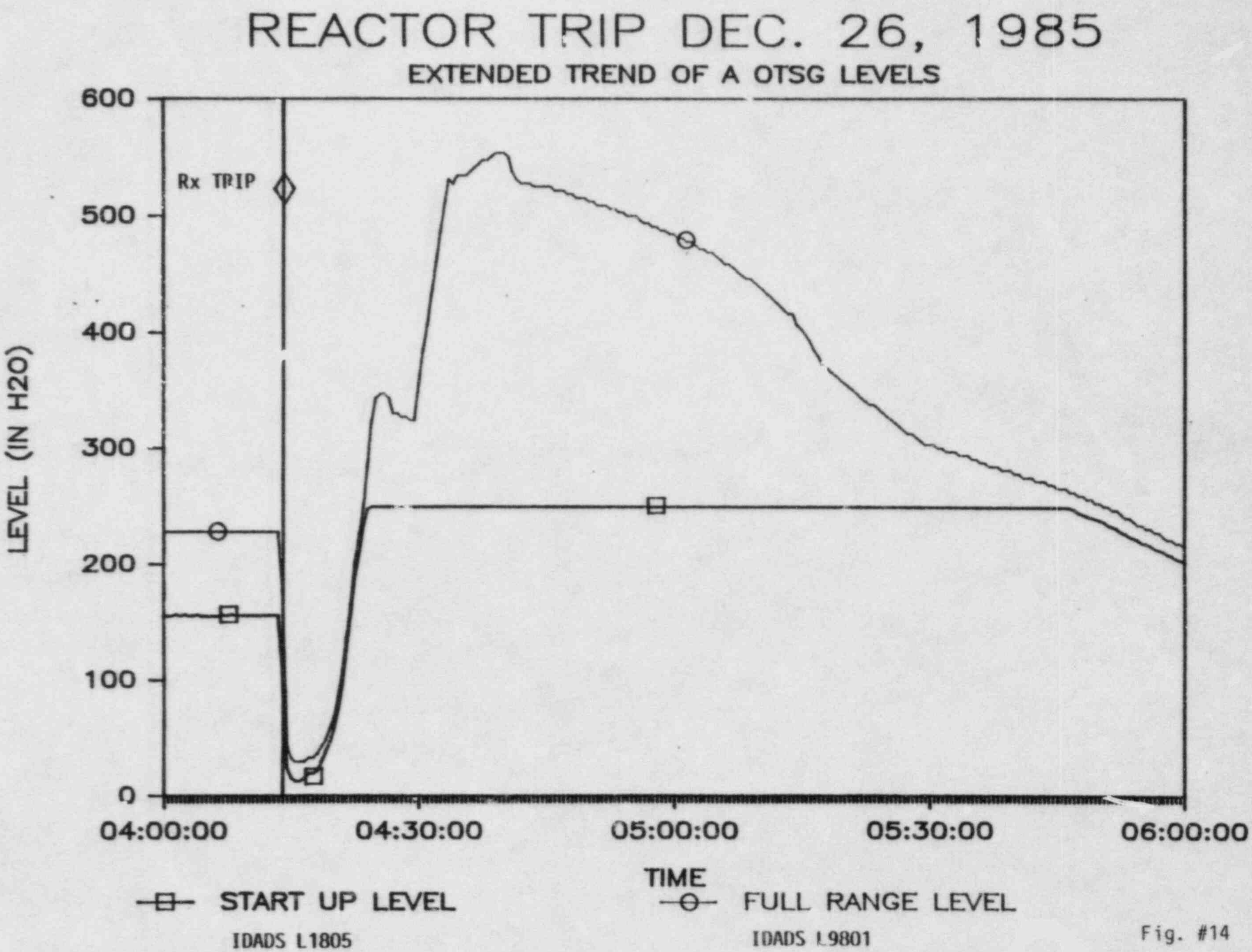


Fig. #13



REACTOR TRIP DEC. 26, 1985
EXTENDED TREND OF A OTSG LEVELS
LEVEL (IN H2O)
600
500
400
300
200
100
0
Rx TRIP
04:00:00
04:30:00
05:00:00
05:30:00
06:00:00
TIME
START UP LEVEL
IDADS L1805
FULL RANGE LEVEL
IDADS L9801

Fig. #14

REACTOR TRIP DEC. 26, 1985

EXTENDED TREND OF B OTSG LEVELS

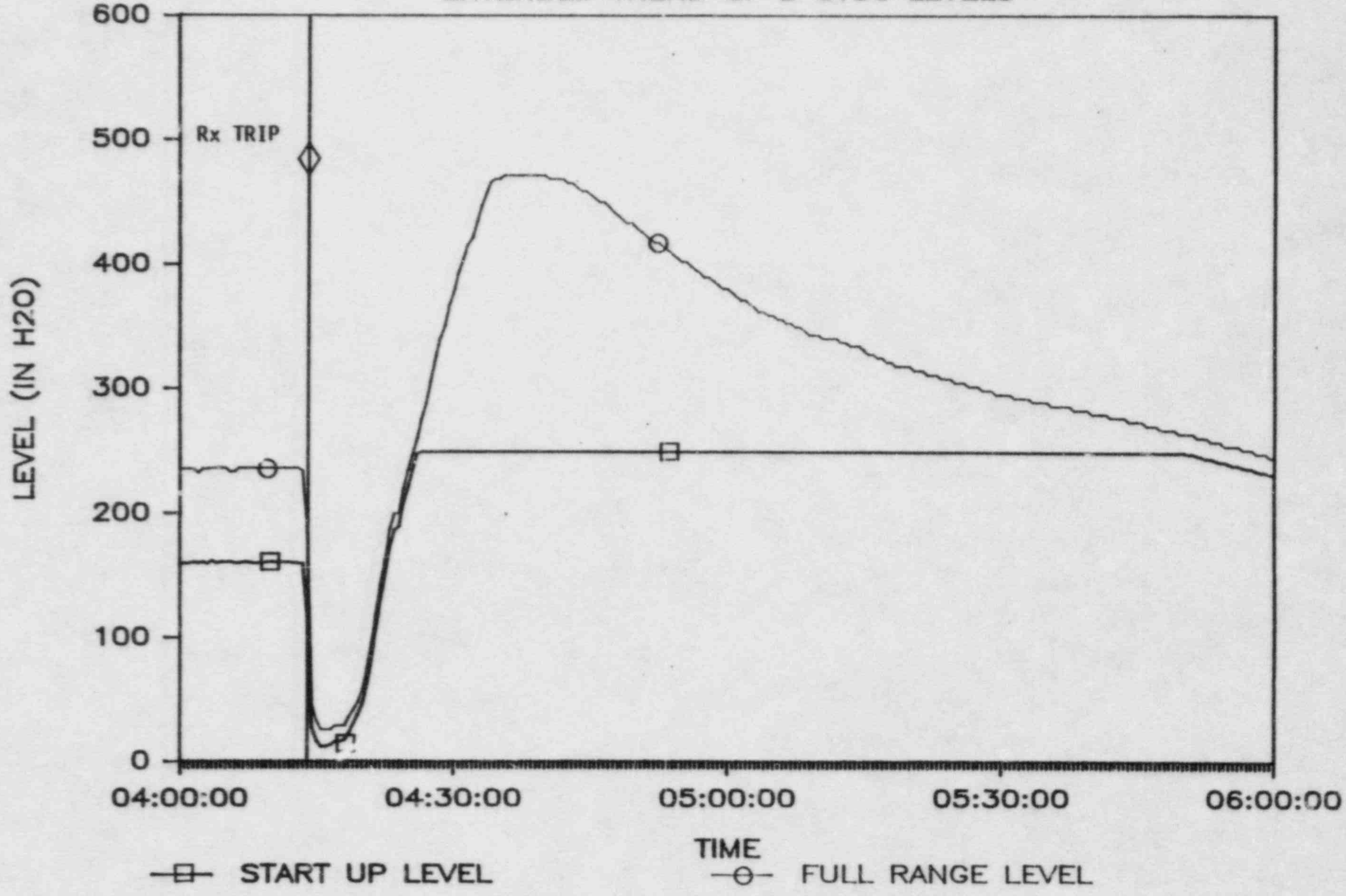


Fig. #15

REACTOR TRIP, DECEMBER 26, 1985

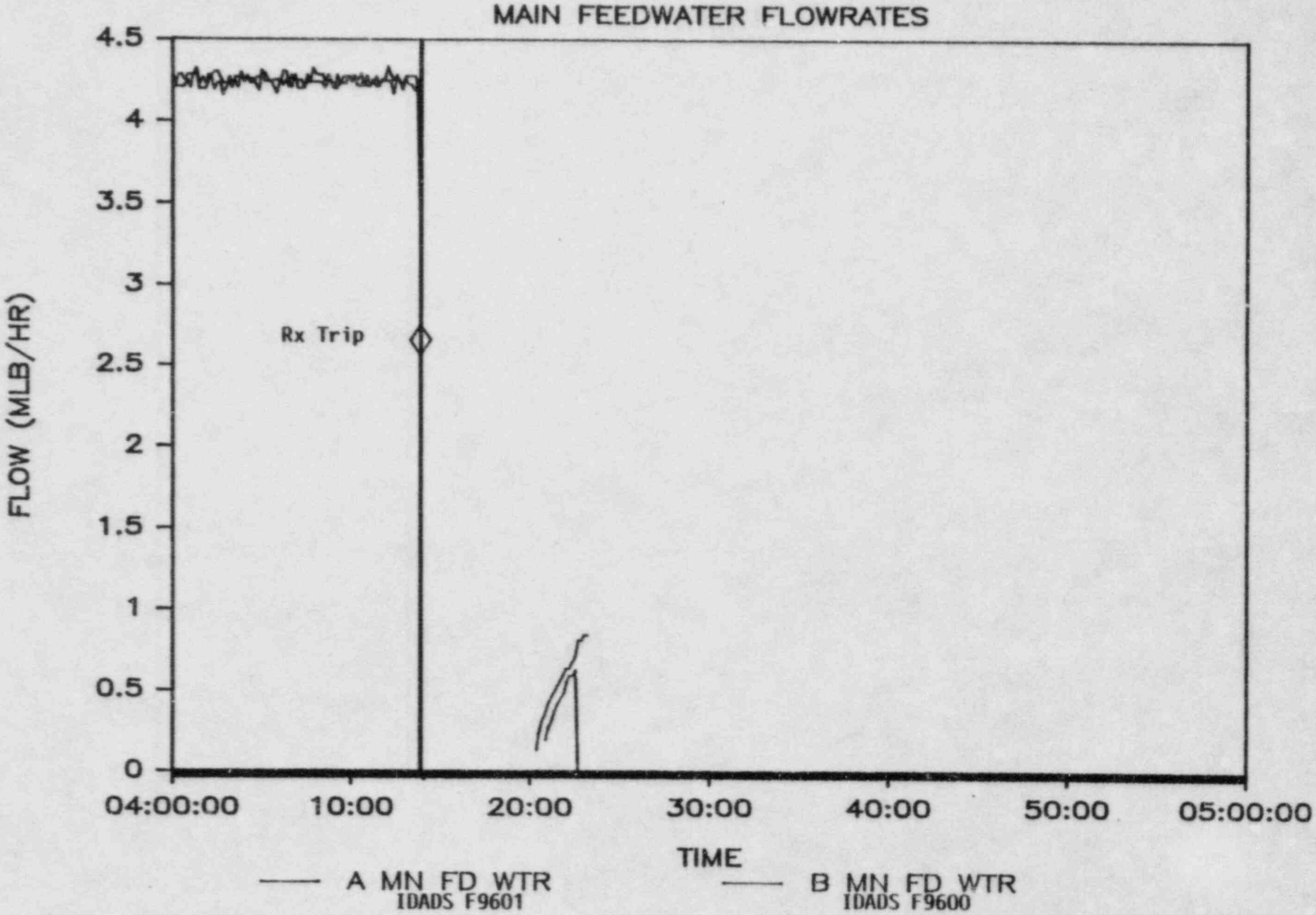
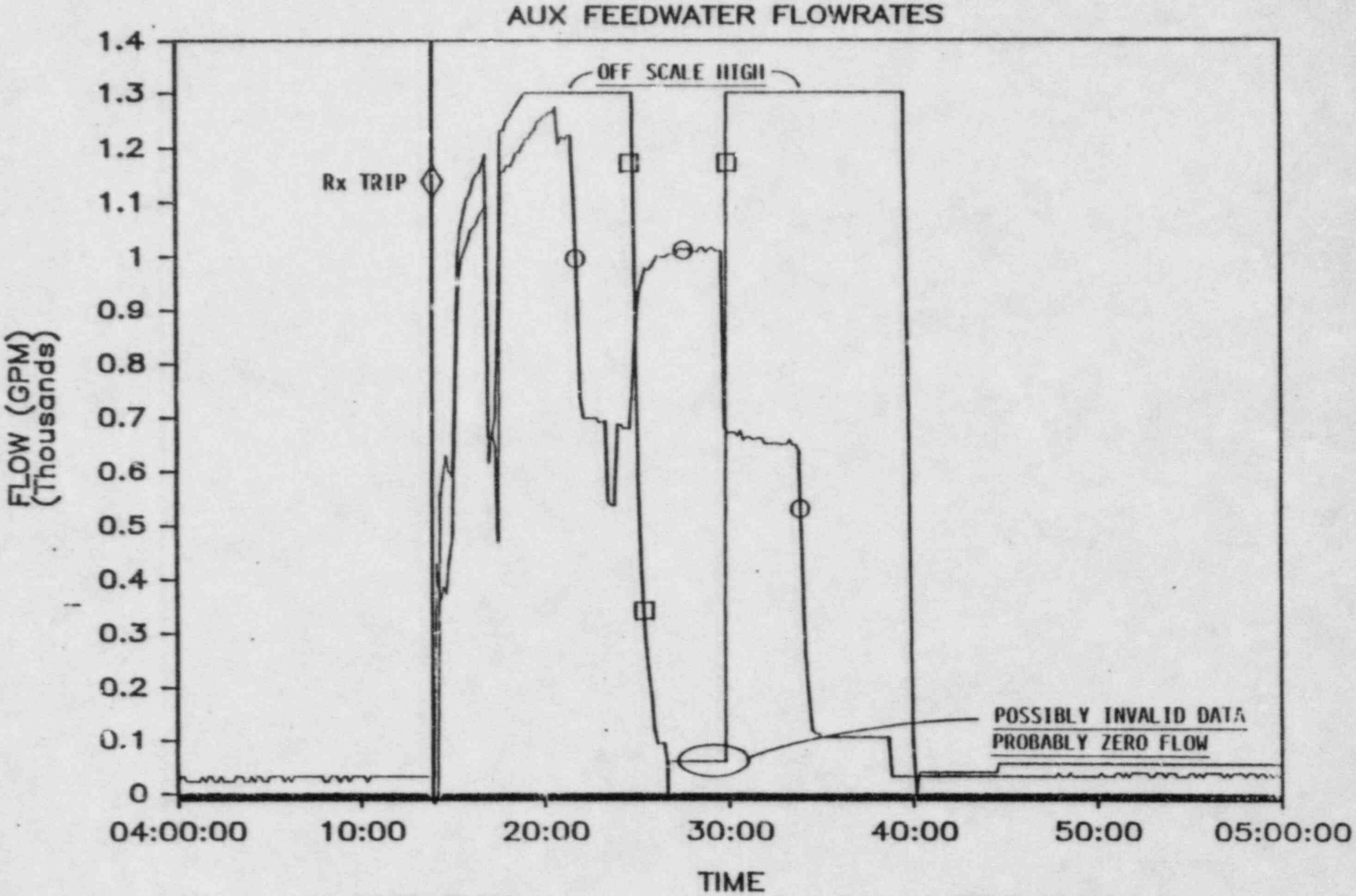


Fig. #16

REACTOR TRIP, DECEMBER 26, 1985



A AUX FEED FLOW — B AUX FEED FLOW

IDADS F1660 — IDADS F1663

Fig. #17

REACTOR TRIP DEC. 26, 1985

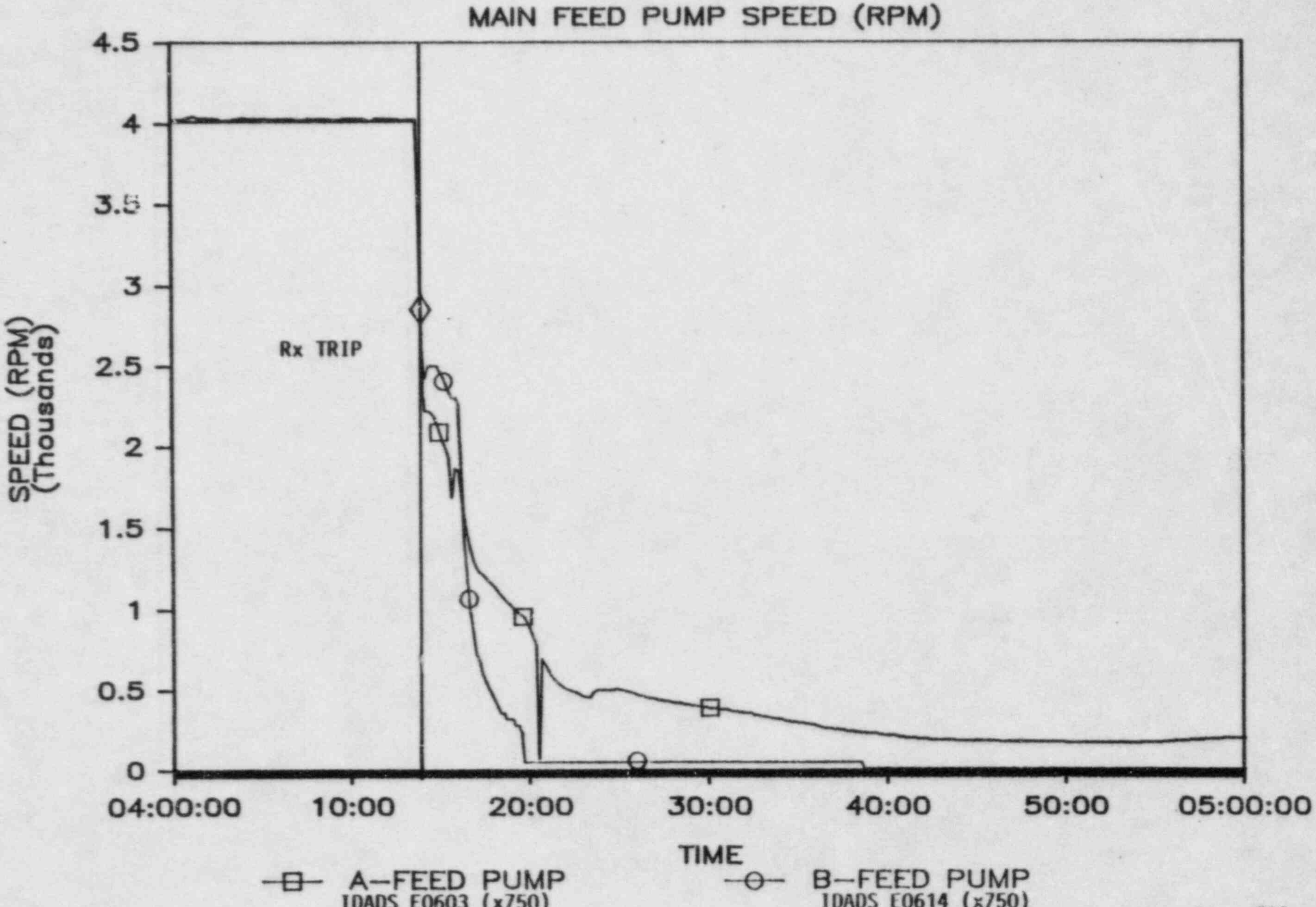


Fig. #18

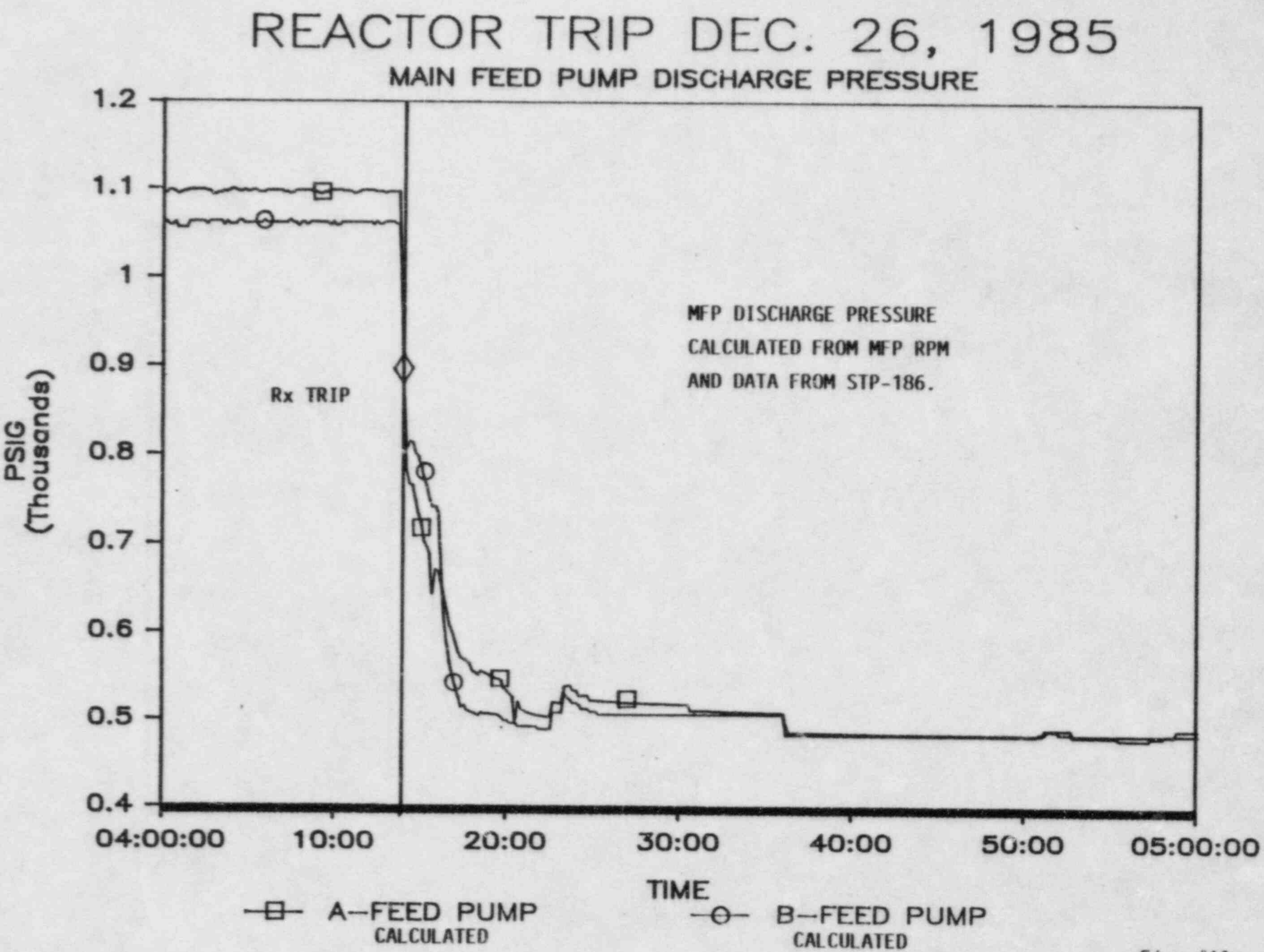
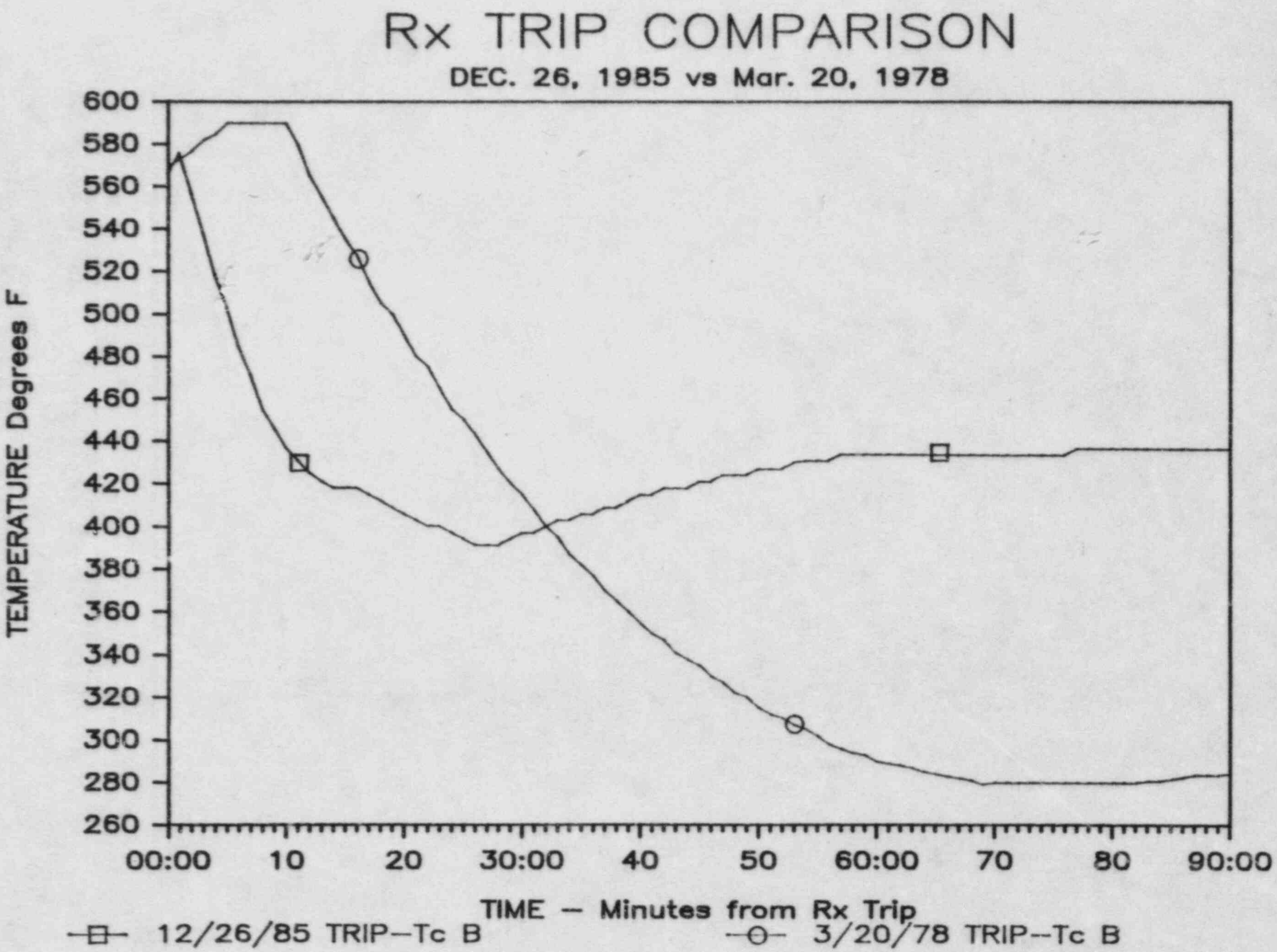

REACTOR TRIP DEC. 26, 1985
MAIN FEED PUMP DISCHARGE PRESSURE
PSIG (Thousands)
1.2
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
Rx TRIP
MFP DISCHARGE PRESSURE
CALCULATED FROM MFP RPM
AND DATA FROM STP-186.
04:00:00
10:00
20:00
30:00
40:00
50:00
05:00:00
TIME
A–FEED PUMP CALCULATED
B–FEED PUMP CALCULATED

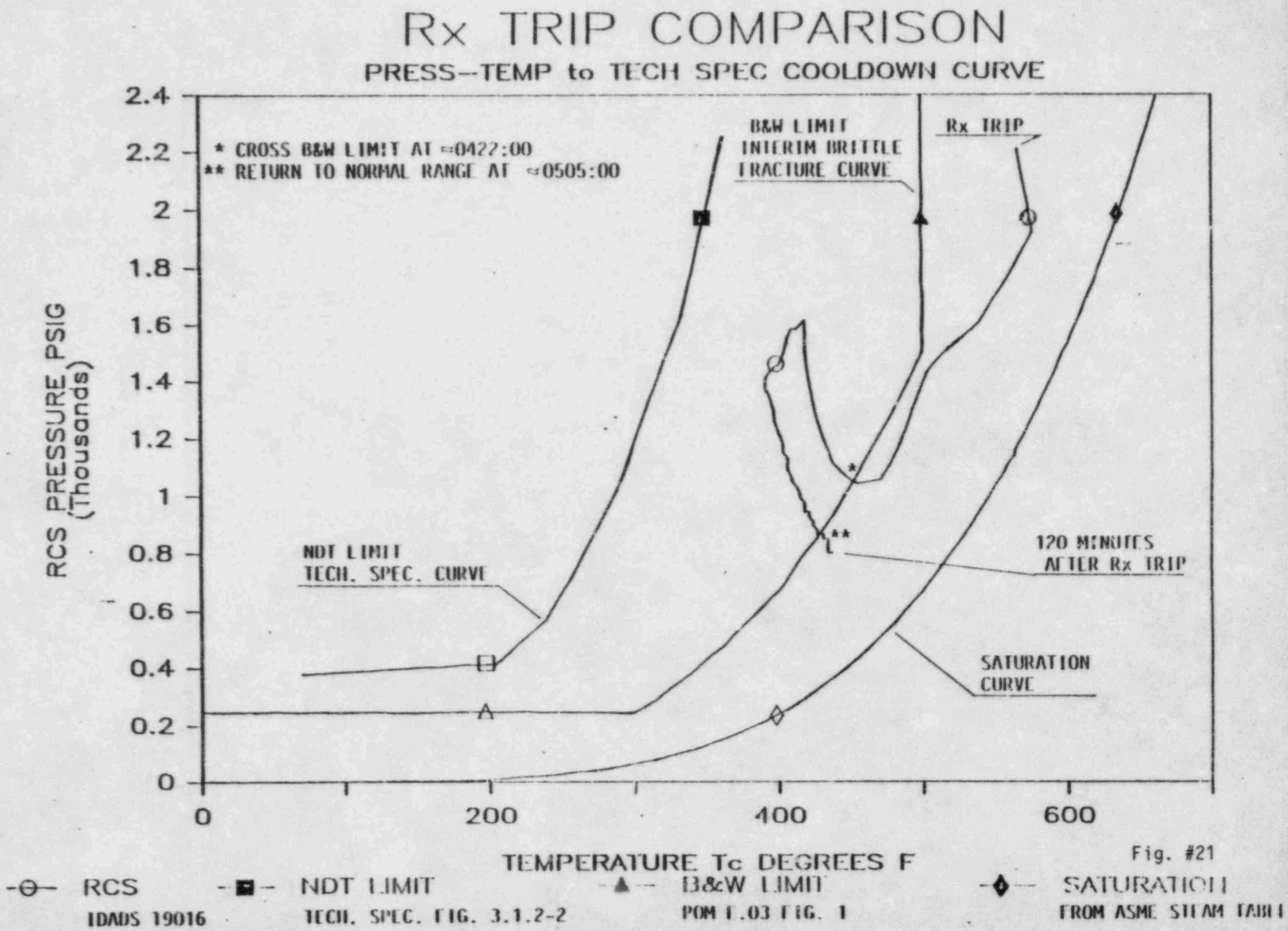
Fig. #19



IDADS T 9016

B & W REPORT BAW-1514

Fig. #20



Rx TRIP COMPARISON
PRESS--TEMP to TECH SPEC COOLDOWN CURVE
* CROSS B&W LIMIT AT ≈0422:00
** RETURN TO NORMAL RANGE AT ≈0505:00
B&W LIMIT
INTERIM BRITTLE
FRACTURE CURVE
Rx TRIP
NDT LIMIT
TECH. SPEC. CURVE
120 MINUTES
AFTER Rx TRIP
SATURATION
CURVE
RCS PRESSURE PSIG
(Thousands)
2.4
2.2
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
200
400
600
TEMPERATURE Tc DEGREES F
-Θ- RCS
IDADS 19016
-■- NDT LIMIT
TECH. SPEC. FIG. 3.1.2-2
-▲- B&W LIMIT
POM F.03 FIG. 1
-◆- SATURATION
FROM ASME STEAM TABLE

Fig. #21

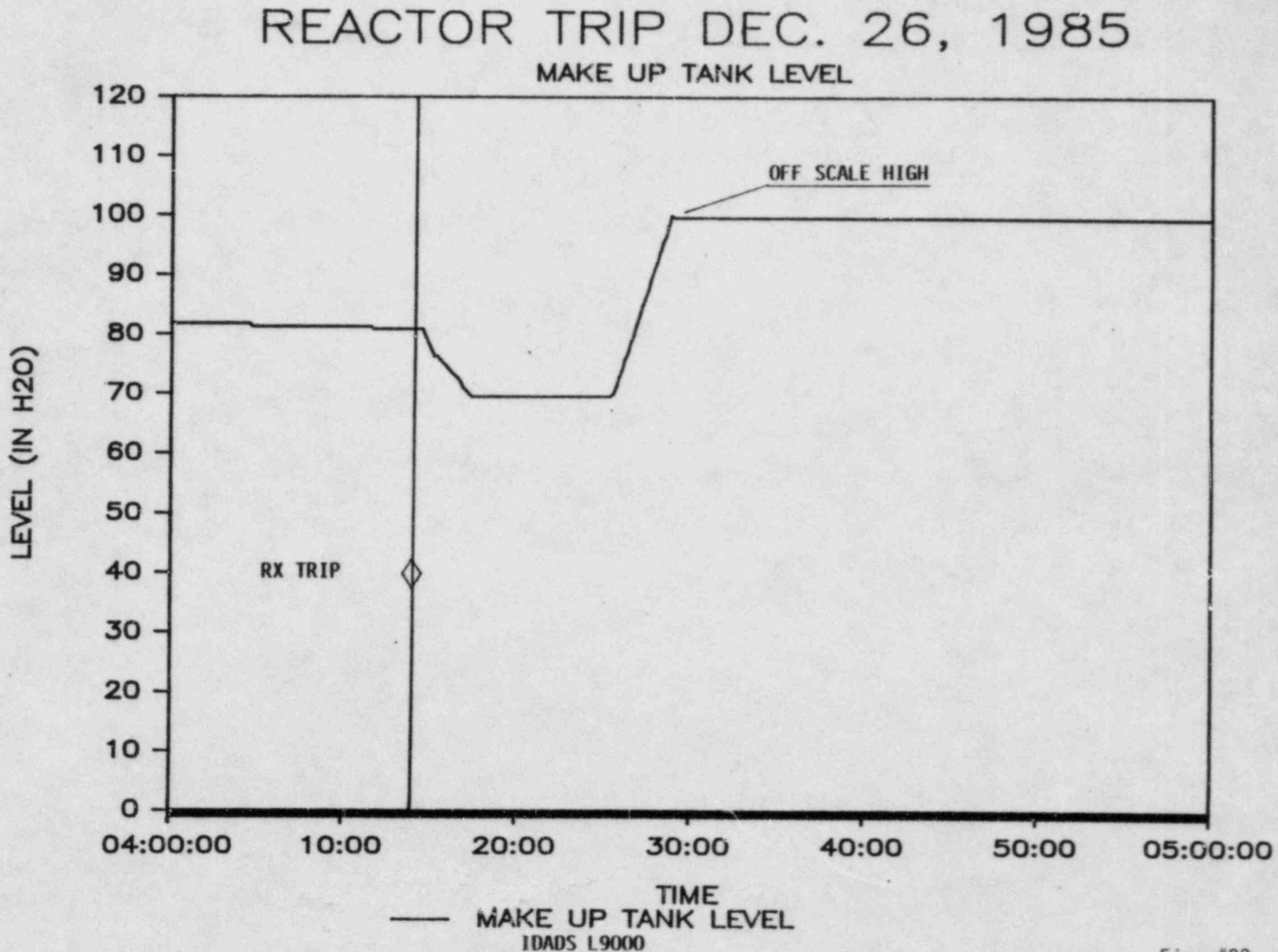

REACTOR TRIP DEC. 26, 1985
MAKE UP TANK LEVEL
120
110
100
90
80
70
60
50
40
30
20
10
0
LEVEL (IN H2O)
OFF SCALE HIGH
RX TRIP
04:00:00
10:00
20:00
30:00
40:00
50:00
05:00:00
TIME
MAKE UP TANK LEVEL
IDADS L9000

Fig. #22

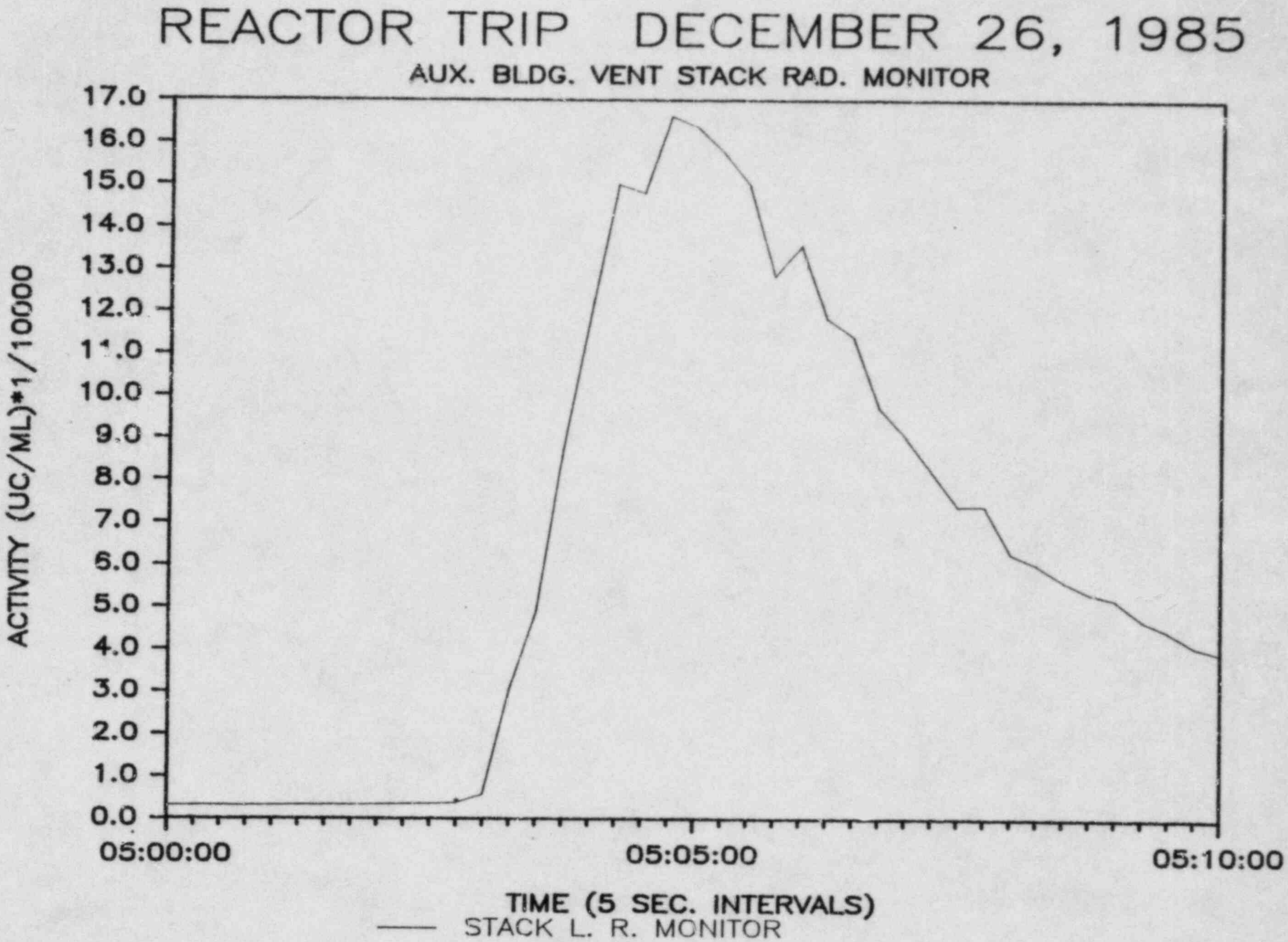

REACTOR TRIP DECEMBER 26, 1985
AUX. BLDG. VENT STACK RAD. MONITOR
ACTIVITY (UC/ML)*1/10000
17.0
16.0
15.0
14.0
13.0
12.0
11.0
10.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0.0
05:00:00
05:05:00
05:10:00
TIME (5 SEC. INTERVALS)
STACK L. R. MONITOR

Fig. #23

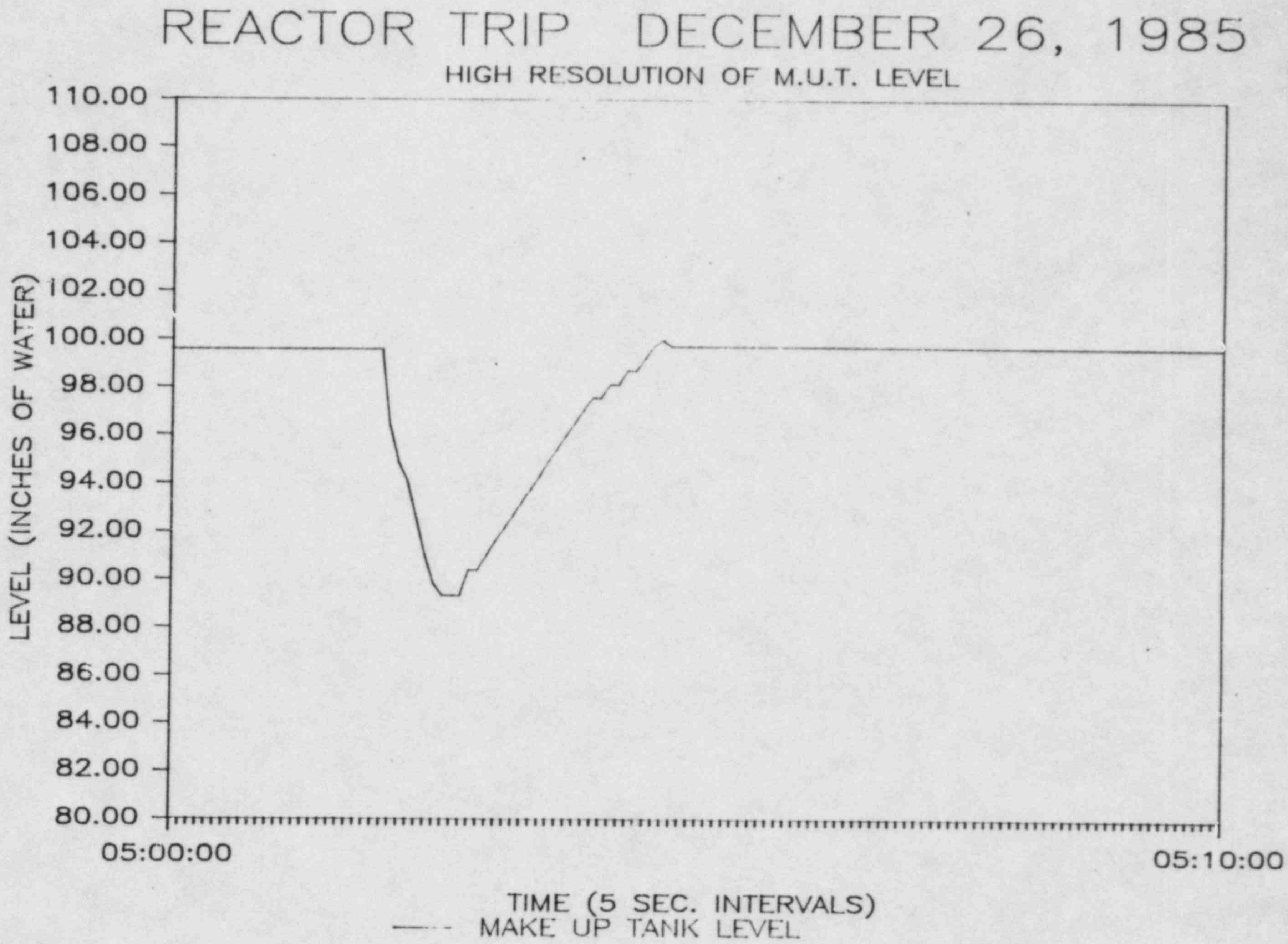

REACTOR TRIP DECEMBER 26, 1985
HIGH RESOLUTION OF M.U.T. LEVEL
LEVEL (INCHES OF WATER)
110.00
108.00
106.00
104.00
102.00
100.00
98.00
96.00
94.00
92.00
90.00
88.00
86.00
84.00
82.00
80.00
05:00:00
05:10:00
TIME (5 SEC. INTERVALS)
MAKE UP TANK LEVEL

Fig. #24

ATTACHMENT 6

HUMAN FACTORS ITEMS

SUMMARY OF HUMAN FACTORS OBSERVATIONS

OF THE 26 DECEMBER 1985 TRIP

Item Number	Addressed by HEO Number*	Control Room Related	HFE Observations	Disposition
1	GN-0-0324	Yes	Excessive noise in Control Room	Reduce noise level from essential HVAC
2		No	Time Delays in accessing valves	Analyze accessibility of valves
3	GN-S-0464 GN-S-0466 GN-S-0468 GN-S-0469	No	Poorly Located emergency/ protective equipment	Add more different sizes of equipment Clearly mark equipment sizes Provide maps indicating location of equipment Locate lead sealed lockers with equipment Develop procedure to analyze usefulness of equipment after each trip
4		Yes	Time delays thru Control Room door	Improve Control Room Security interface
5		Yes	Difficulty monitoring plant status while on red phone	Install long cord on NRC red phone
6		Yes	Spurious ringing on NRC phone	Repair system to alleviate
7	GN-S-0293	Yes	Poor communication between Control Room & plant	Provide new wireless communication system
8	AN-0-0225	Yes	No MSLF alarm	Provide alarm
9		No	Poor local valve status indication	Provide accurate valve status indication, including highly visible labels
10		Yes	No Control Room control of ICS components upon ICS failure	Investigate providing redundant controllers on ICS components
11	PS-0-0348 PS-W-0474	Yes	Difficulty tracking status of Pressurizer Relief Tank	Replace indicator with trend recorder

*HEO's referenced are from the submitted CRDR Summary Report.

Item Number	Addressed by HEO Number*	Control Room Related	HFE Observations	Disposition
12		Yes	Inconsistent indication between SPDS & OTSG level trend recorders	Modify systems to improve consistency
13		Yes	Insufficient status information on diesel generators	Provide important parameter status in the Control Room
14	GN-0-0278 GN-0-0305 XX-S-0214 PS-S-0212 PS-S-0209 XX-S-0213 SS-S-0215 GN-S-0208	Yes	Unreliable/unreadable recorders	Replace trend & impact recorders
15	GN-S-0068 GN-S-0074 GN-S-0075	Yes	Due to loss of power, inconsistency between feedwater flow meters and trend recorders	Label all indicators with power source and train
16		No	Poor labeling of breakers S1/S2	Relabel
17		No	Difficult to use elementary drawings	Improve nomenclature & provide operator training on elementaries
18		Yes	Confusing alarms on IDADS	Improve operator/IDADS interface
19	SF-V-0396 GN-S-0258	Yes	AFW components scattered throughout Control Room	consolidate with EFIC
20	GN-W-0513 GN-0-0283 GN-S-0253	Yes	Delays in gaining control of SFAS valves	Relabel lenses
21		No	EOP's were vague	Modify EOP's
22		No	Damage to pump	Investigate feasibility of adding low suction trips to important pumps
23		No	Inadequate personnel support	Provide 24 hr support
24	RI-V-0392	Yes	Difficulties recognizing loss of power	Yellow band failure point on indicators
25		No	No information on feedwater heater reliefs	Investigate providing relief information
26	MR-S-0493 thru MR-S-0511	Yes	Radiation information scattered throughout Control Room	Centrally located within the RM-11

* HEO'S referenced are from the submitted CRDR Summary Report.

ATTACHMENT 7

CONTROL ROOM EQUIPMENT AFFECTED BY
LOSS OF ICS DC POWER

The following list compiles all of the Control Room equipment affected by the loss of ICS power.

1. <u>Hand/Auto Stations</u>

   Unit Demand
   Steam Generator/Reactor
   Reactor Demand
   Steam Generator ΔTc
   Turbine Header Pressure Setpoint

"A" Feedwater Demand	"B" Feedwater Demand
"A" Feedwater Pump Speed	"B" Feedwater Pump Speed
"A" Main Feedwater Valve	"B" Main Feedwater Valve
"A" Startup Feedwater Valve	"B" Startup Feedwater Valve
"A" Turbine Bypass Valves	"B" Turbine Bypass Valves
"A" Atmospheric Exhaust Valves	"B" Atmospheric Exhaust Valves
"A" Auxiliary Feedwater Valve	"B" Auxiliary Feedwater Valve

   Auxiliary Steam Reducing Station

2. <u>Indications</u>

"A" Loop Feedwater Flow Recorder	"B" Loop Feedwater Flow Recorder

   Generator Frequency (H1SS)

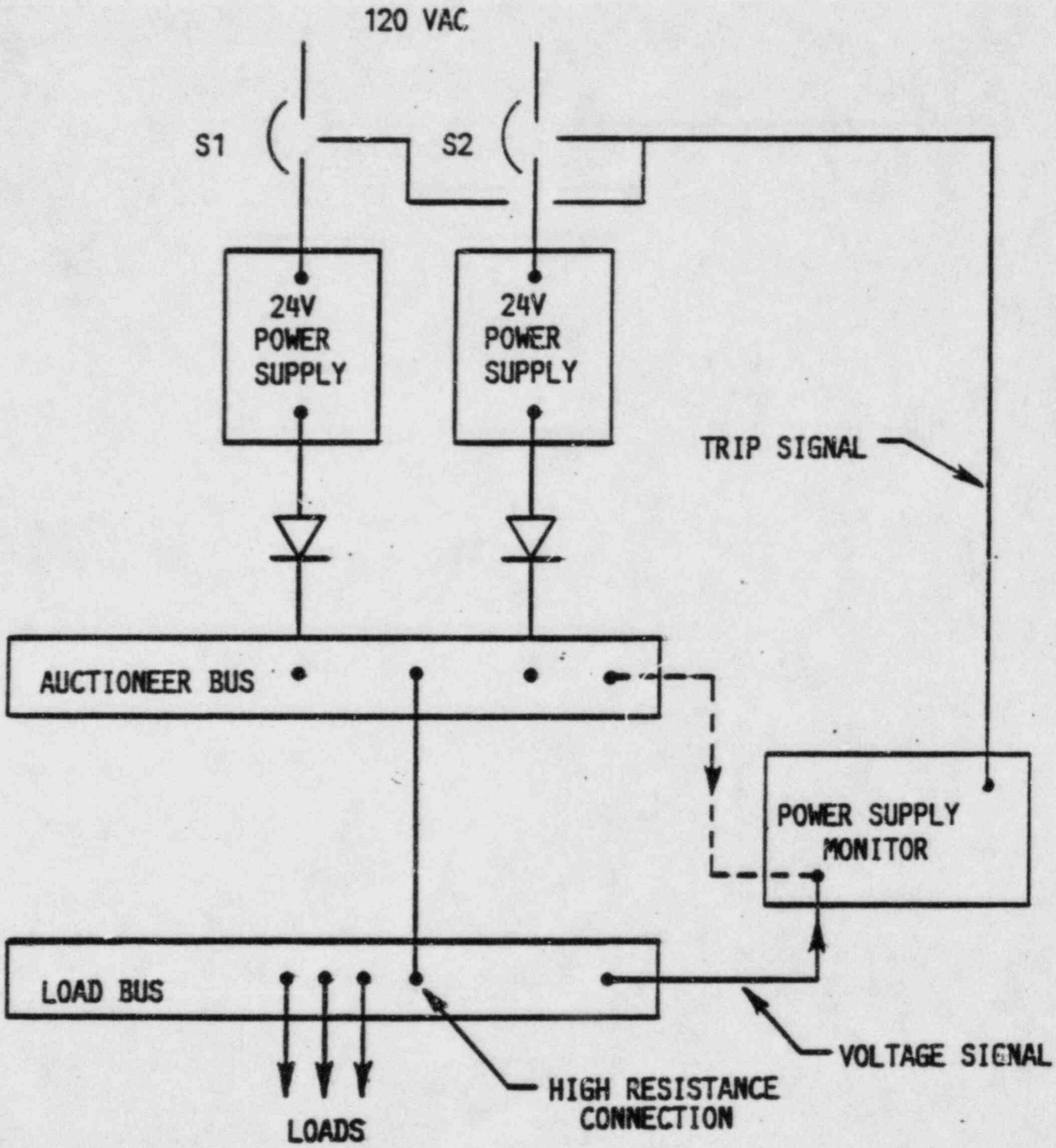
   ICS Indicator Lights on Load Control Panel H-1RI (above Diamond Panel)

3. <u>Annunciation (H2PSB)</u>

   #64 ICS or Fan Failure

ATTACHMENT 8

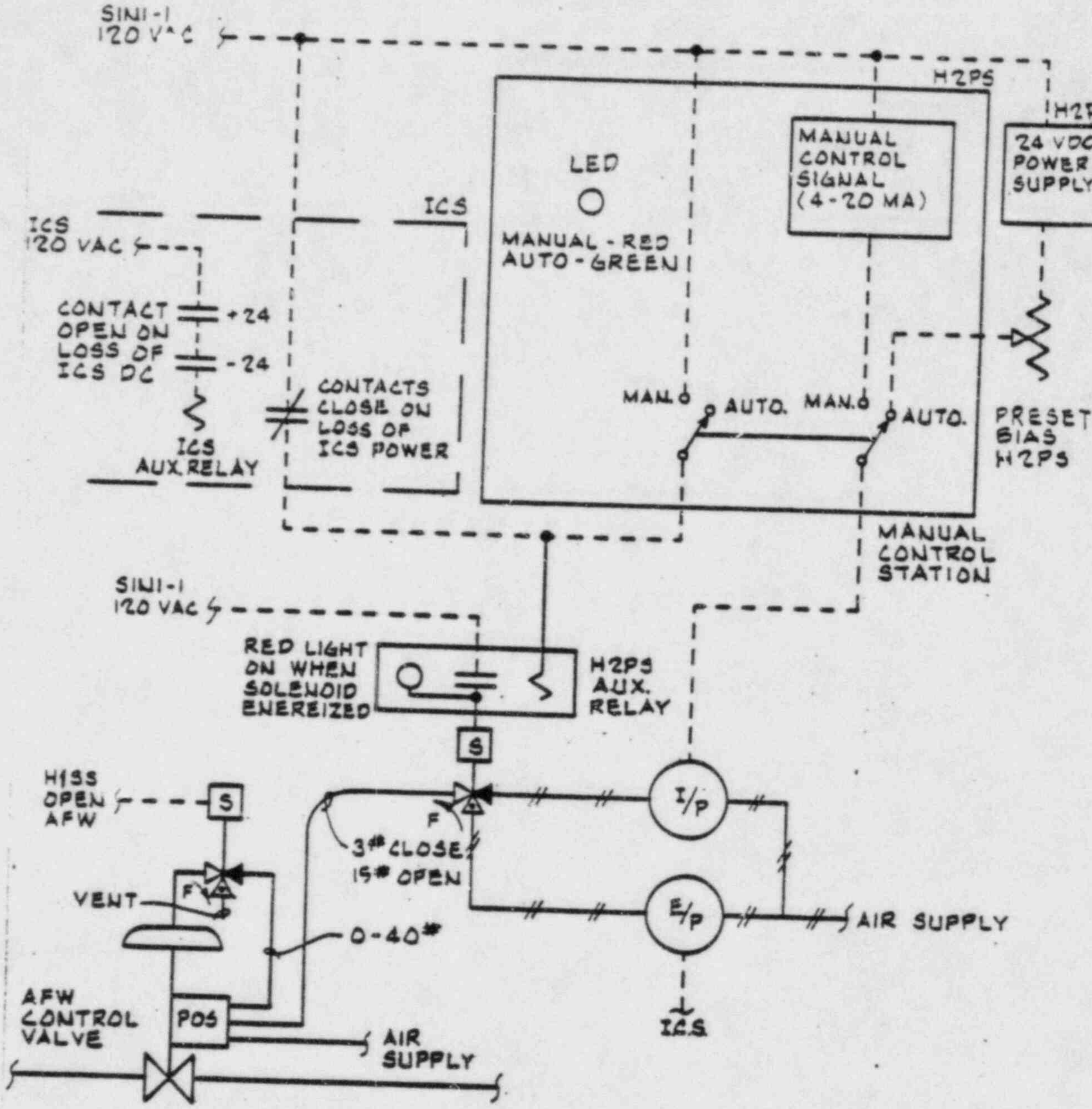
ICS POWER SUPPLIES AND DISTRIBUTION



ICS POWER DISTRIBUTION

ATTACHMENT 9

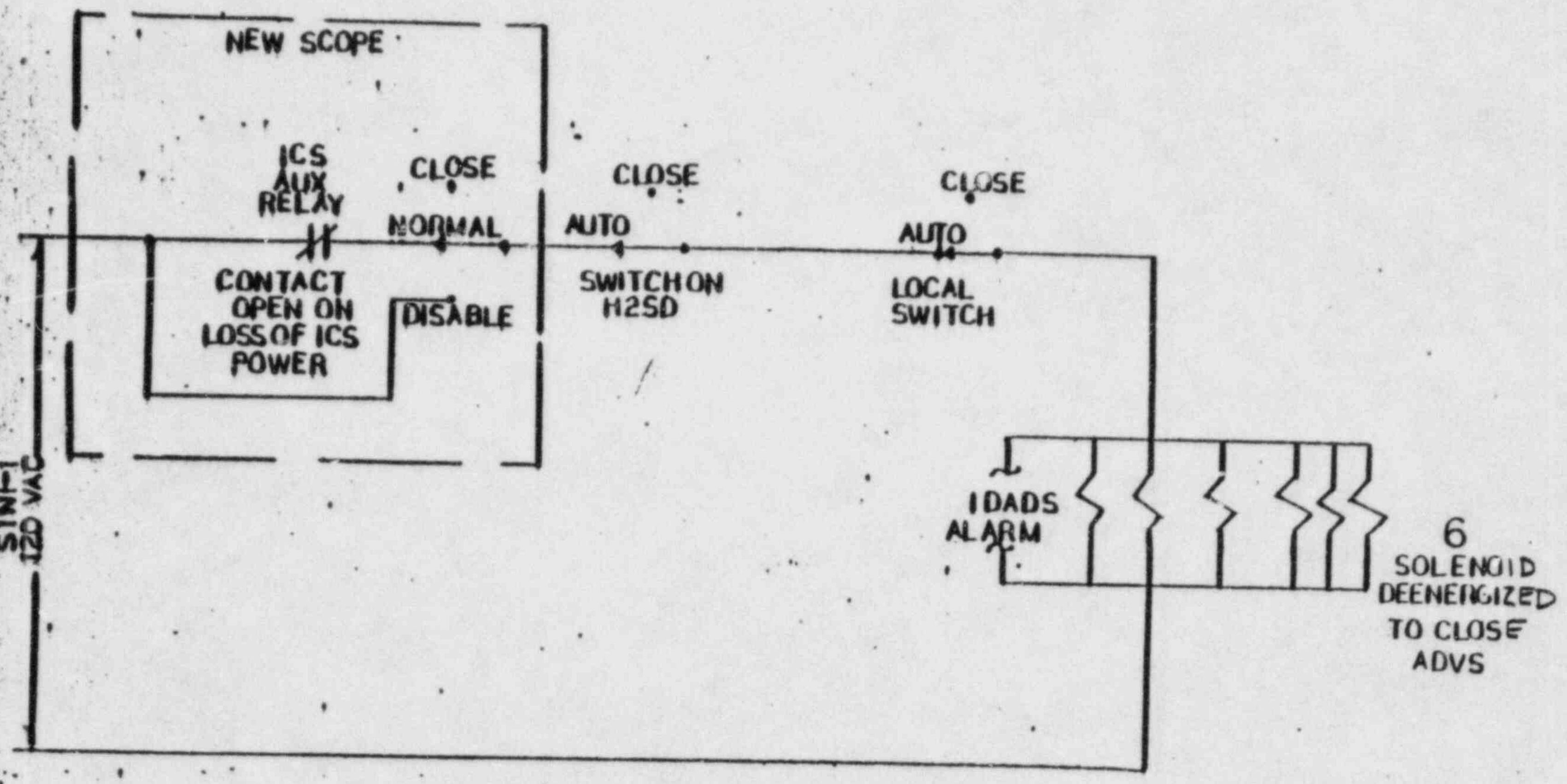
AFW CONTROL INDEPENDENT OF ICS



AUXILIARY FEEDWATER CONTROL
INDEPENDENT OF I.C.S.

ATTACHMENT 10

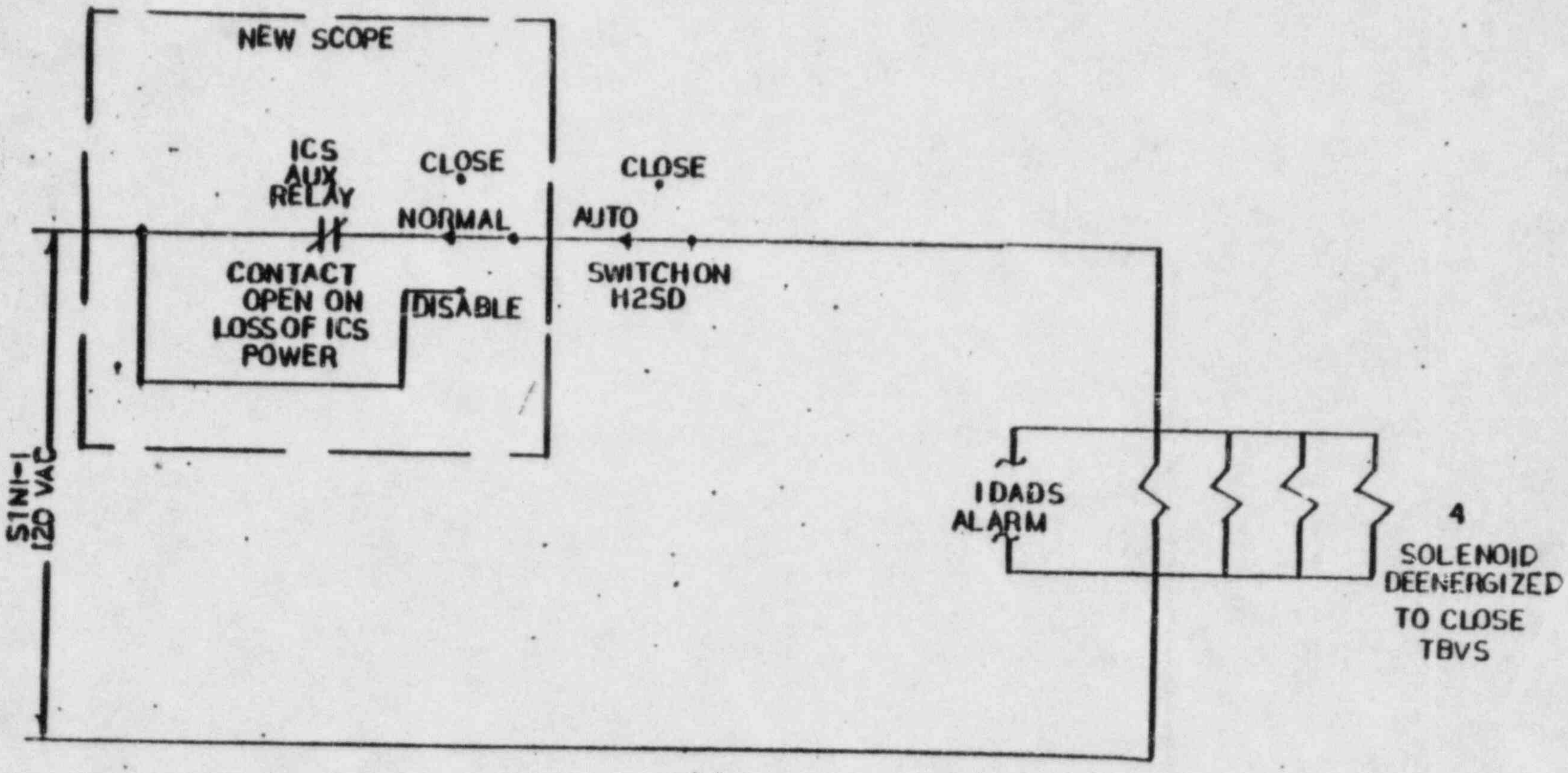
ADV CONTROL INDEPENDENT OF ICS



ADV MANUAL CONTROL

ATTACHMENT 11

TBV CONTROL INDEPENDENT OF ICS



TBV MANUAL CONTROL