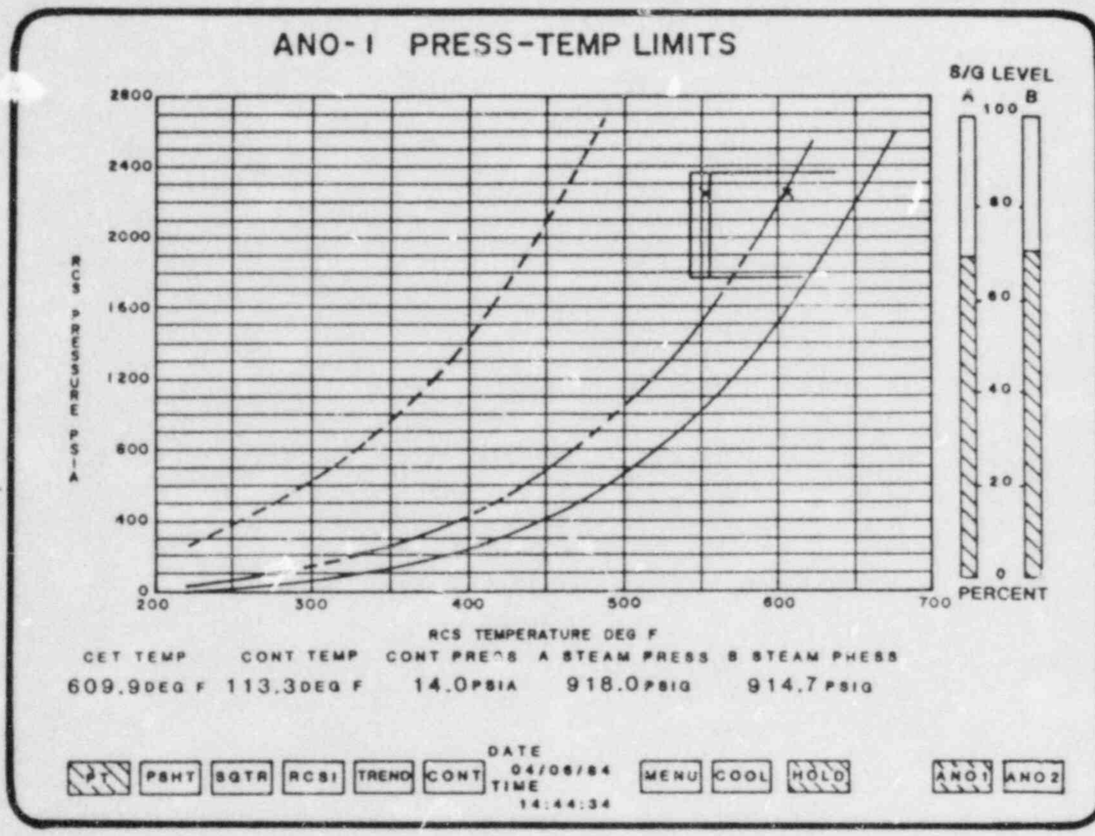
ARKANSAS POWER & LIGHT COMPANY
ARKANSAS NUCLEAR ONE
UNIT 1

SAFETY PARAMETER DISPLAY SYSTEM SAFETY ANALYSIS

JUNE 29, 1984



SAFETY ANALYSIS REPORT

FOR THE

ANO-1 SAFETY PARAMETER DISPLAY SYSTEM

INDEX

INTRODUCTION

The purpose of this report is to provide a written safety analysis for the Arkansas Nuclear One, Unit One (ANO-1) Safety Parameter Display System (SPDS). This document satisfies the requirement for a safety analysis specified in Section 4 of NUREG 0737, Supplement I.

In accordance with 10 CFR 50.59, implementation of the SPDS is a facility change and thus requires a written safety evaluation to provide the bases for the determination that the change does not involve an unreviewed safety question. In addition, NUREG 0737, Supplement I, specifically requires a written safety analysis describing the bases on which the selected SPDS parameters are sufficient. The SPDS parameters should be adequate to access the safety status of each critical safety function identified in NUREG 0737, Supplement 1, for a wide range of events, including symptoms of severe accidents.

The hardware changes to ANO-1 associated with the installation of the SPDS were implemented under a number of separate design change packages (DCPs). These changes were controlled by various engineering and administrative procedures. The design control procedure specifically addresses preparation of a safety determination for each DCP. This provides the basis for a pre-implementation determination of whether an unreviewed safety question exists with respect to the specific plant change. The safety determinations for the SPDS DCPs address the specific modifications contained in the package and provide the required safety evaluations. This procedure, therefore, ensures that proposed changes are adequately reviewed in accordance with the ANO Technical Specifications to determine whether the changes involve an unreviewed safety question or a change to the Technical Specifications, per 10 CFR 50.59 requirements.

The 10 CFR 50.59 safety evaluations have been documented as part of each DCP and none of the changes involved an unreviewed safety question. Furthermore, these DCPs have been reviewed by the Plant Safety Committee and the Safety Review Committee. As a result, this report focuses on the safety analysis for the selection of parameters.

The SPDS for ANO-1 was declared operational with existing parameters on January 9, 1984. Discussions in this analysis are of the existing capabilities as well as currently proposed modifications.

BACKGROUND

In response to the TMI-2 accident in 1979, the NRC began issuing "Lessons Learned" NUREGs and Regulatory Guides addressing improved emergency response capabilities for nuclear power plants. The accident and subsequent investigations demonstrated the opportunity for improving the presentation of plant and process information to reactor operators, especially during major transients. TMI-2 highlighted man/machine interface deficiencies because reactor operators were required to monitor and process such large amounts of data in order to ascertain the operating and safety status of the plant and take necessary actions. The available information in the control room was adequate, but was not presented in the most useful form, especially under stressful conditions.

To comply with the NRC's recommendations, various nuclear power industry organizations developed the concept of a safety console or panel to display plant safety information to the operator without subjecting them to "information overload". This concept was eventually refined into a safety parameter display system (SPDS) that would facilitate the assessment of plant safety by providing a set of predetermined color graphic displays which would be continuously updated with real time data to yield relevant, timely, accurate and unambiguous information.

The SPDS concept was to display a small but critical subset of the information already presented by control room instrumentation in order to minimize information overload. The ANO-1 Emergency Operating Procedure and the associated development documents were utilized as the basis for determining the critical parameters necessary to assess the plant safety status. These parameters were then combined and formated to provide the operators with a concise set of data to aid them in rapidly and reliably determining the safety status of the plant. Implicit in the EOP development program was the consideration of the five critical safety functions identified in NUREG-0737 Supplement I.

AP&L began the development of an SPDS in 1979 as part of an in-house initiated EOP upgrade program and expanded the development early in 1980, in response to the "Lessons Learned" NUREGs 0578 and 0585. Two Systems Engineering Laboratory (SEL) 32/77 computer systems were ordered in June 1980 to provide the capability to perform the required functions. NUREG 0737, issued in October 1980, required the implementation of the plant SPDS. NUREG 0737 referenced NUREG 0696 for use as the criteria for design of the SPDS and Technical Support Center (TSC)/Energy Response Facility (ERF) instrumentation systems. However, NUREG 0696 was not issued until March 1981, so several modifications were required to the original computer system design as a result of the new criteria. Supplement I of NUREG 0737 was issued in December 1982 to provide additional clarification to certain NUREG 0737 requirements. Supplement I also promoted an integrated approach for the implementation of the SPDS, upgraded Emergency Operating Procedure (EOP), control room design reviews, emergency response facilities and Regulatory Guide 1.97 instrumentation reviews. The present SPDS computer system is designed to meet the objectives of the above referenced NRC documents.

DESIGN OVERVIEW

Prior to TMI-2, emergency response was event-oriented. Each pre-defined event (plant transient) was covered by an abnormal operating procedure. The approach was straightforward: diagnose and recover, according to procedure. TMI-2 showed that this method was not always adequate.

In complex systems like nuclear power plants, the operator must interpret and integrate a vast amount of information supplied by instrumentation and relate it to rule-based and knowledge-based training to determine the best recovery action. Because of information overload, however, the operator may not always be directing his attention to the parameters which are the most relevant to the disturbance.

Integration of key plant process information from different subsystems is difficult under normal conditions and even more difficult under stress. Control boards are often designed for control at the subsystem level, and are not necessarily conducive to overall plant response monitoring. The SPDS serves to integrate the appropriate information in a concise form by means of an on-line computer monitoring system with human-factored CRT displays.

In accordance with NUREG 0737, Supplement 1, the SPDS displays were integrated with the development of the upgraded ANO-1 Emergency Operating Procedure (EOP) to ensure compatibility. The new EOP is symptom-oriented. Instead of requiring diagnosis before action as the earlier emergency procedures, the new EOP allows the operator to respond to the symptoms of the event.

The SPDS was designed to assist the operator in implementing the upgraded EOP. The ANO-1 EOP was developed to achieve timely and accurate safety status assessment either with or without the SPDS. The design of the specific SPDS graphic displays correspond to specific sections of the upgraded EOP, so the SPDS complements the use of the upgraded EOP.

The SPDS design also includes consoles and displays in the on-site Technical Support Center (TSC) and off-site Emergency Operations Facility (EOF). This improves the operational aids available to the plant technical staff to assist them in evaluating transient conditions and providing guidance and direction to the operations staff. The TSC and EOF both have access to the large-scale data storage and retrieval capabilities of the SPDS to assist in event diagnosis and historical documentation.

The computer system selected by AP&L was chosen primarily because of its flexibility, reliability, and maintainability. Flexibility is needed to permit the incorporation of future modifications without unwarranted difficulty. The computer hardware selected is similar to existing hardware already in use at ANO. This hardware has been proven to be reliable. Furthermore, AP&L personnel have considerable experience in maintaining this equipment which should improve the overall reliability of the system.

The basic configuration of the SPDS consists of redundant data acquisition, processing and display devices. The SPDS computers access the necessary input parameters from sensors in ANO-1 and ANO-2, process these signals, and provide displays to each control room as well as to the TSC and EOF. The SPDS performs no plant control action, but serves as a human-engineered data display system to aid the operator in rapidly and reliably determining plant safety status. "Touch screen" controls are utilized on the color graphic CRTs to allow for rapid access to the information necessary to determine safety status of the plant. The SPDS design should provide enhanced capabilities for responding properly to both anticipated and unanticipated plant conditions.

Considerable effort has been expended during the initial design of the SPDS to incorporate human factors principles. In addition, the ANO operations staff has played a vital role in the SPDS design and implementation, to ensure that the system will be responsive to the needs of the operators during normal and emergency conditions. The primary "ATOG" display has been reviewed and studied by Sandia Labs during the Interim Reliability Evaluation Program and by a Honeywell study team for EPRI. These reviews concluded that the ANO-1 "ATOG" display is an excellent tool in providing the operator with information necessary for assessment of plant safety status. Furthermore, AP&L has included the SPDS in the scope of the Control Room Design Review (CRDR) program to formally evaluate the proper incorporation of human factors principles including equipment location, display formats and characteristics, operator interfaces, and compatibility with the EOPs.

The SPDS is designed to be isolated from electrical and electronic interference with equipment and sensors that are in use for safety systems. The SPDS design was reviewed with respect to IEEE Standard 384-1977, Section 6.2, and found to be in compliance with the isolation criteria. The specific methods of isolation include current transformers for the analog signals, and optical couplers and relays for the digital signals requiring interference isolation.

The ANO-1 SPDS was subjected to a verification and validation process to ensure that applicable requirements were met. The SPDS verification included a system requirements review and a design review based on the system requirements. Validation included testing and evaluation of the completed system, hardware and software, to ensure compliance with design, function, performance and interface requirements. This testing confirmed field input calibration; input source to computer point identification relationship; software programs for the acquisition, conversion, manipulation and display of data from field inputs; and proper operation of the central processors, related circuits and memory, and peripheral devices. The SPDS verification and validation process was performed and documented in accordance with NRC guidelines in NUREG-0737, Supplement I.

BASIS FOR DISPLAYS AND PARAMETER SELECTION

The Emergency Operating Procedure (EOP) for ANO-1 has its origin in the Babcock & Wilcox Abnormal Transient Operating Guideline (ATOG) program. From this program it was determined, following a reactor trip and verification of shutdown, that their are three symptoms of primary interest to a pressurized water reactor operator to prevent core and reactor coolant system damage: 1) inadequate subcooling of the primary system inventory, 2) inadequate primary to secondary heat transfer, and 3) excessive primary to secondary heat transfer. These symptoms are important for the following reasons:

1. Inadequate primary inventory subcooling: If the operator knows the primary fluids is in a liquid state, he is assured that it is available and capable of removing heat from the core. If subcooling is lost, these issues are in doubt, and he is therefore directed to make every effort to regain subcooling.

2. Inadequate primary to secondary heat transfer. This symptom addresses the heat transfer coupling across the steam generator. It describes the ability of the system to keep the flow of energy moving from the reactor coolant system to the ultimate heat sink.

3. Excessive primary to secondary heat transfer: In this case, the symptom is indicative of a secondary side malfunction (e.g., loss of steam pressure control or steam generator overfill). The heat transfer is again unbalanced and the operator's attention is directed toward generic actions to restore this balance.

The information required to identify and track these symptoms is available in all nuclear power plant control rooms and simply consists of reactor coolant system temperature, reactor coolant system pressure, steam generator pressure and access to a set of steam tables.

The ATOG pressure-temperature diagram (ATOG Diagram) was developed to provide the above described information to the plant operator in a timely fashion with little or no effort on his part. The ATOG Diagram is the top level display for the ANO-1 SPDS for this reason.

ATOG Pressure-Temperature (ATOG)

The ATOG diagram basic features are displayed in Figure 1. The static portion is a grid of RCS pressure versus RCS temperature with fixed curves showing the saturation line, a 50°F margin to saturation line and the RCS NDTT limits for normal operation. During power operation the Reactor Protection System (RPS) pressure-temperature trip limits are shown along with a normal transient window showing the minimum and maximum pressures and temperatures expected immediately following a trip. After a reactor trip, a small box will appear inside this window showing the expected pressure-temperature relationship for normal hot shutdown conditions. If no forced RCS flow is indicated, the single small box will be replaced by two boxes showing the expected T_{cold} and T_{hot} conditions during natural circulation at hot shutdown. The normal RCS NDTT limit will also be replaced by the natural circulation NDTT limits.

The dynamic elements of the display include a bar graph representation of steam generator levels, digital values of selected parameters displayed below the P-T grid, and points identifying T_{cold} and T_{hot} versus pressure. Following a reactor trip, the past plotted values of temperature versus pressure remain on the screen showing the trajectory which they are following. The values shown at the bottom of the screen are reactor building temperature and pressure, A and B steam generator pressures and the average of the five highest core exit thermocouple temperatures.

During inadequate core cooling situations, pressure versus temperature is plotted on a high range temperature scale. On this display, core exit temperatures versus pressure corresponding to 1400°F and 1800°F cladding temperatures are indicated. These curves and the high range temperature scale are displayed when the average of the five highest core exit temperatures indicates greater than 50°F superheat.

A typical plant response to a reactor trip is shown in Figure 2. T_{cold} should merge with T_{hot} as the decay heat rapidly drops. Both temperatures should then move toward normal hot shutdown pressure and temperature conditions. If they do not, a departure from normal is indicated. If they move outside a larger normal transient limit area, the definite need for operator action is indicated.

Each of the three basic symptoms discussed earlier leave their unique signature on the ATOG diagram as displayed in Figures 3, 4, and 5. Use of this tool enables an operator's priority to be fixed on controlling the plotted parameters within target bounds. If successful, he will be able to bring the reactor to a safe condition. This will be the case regardless of whether or not he has properly diagnosed (or diagnosed at all) the event which has occurred. However, use of the SPDS in conjunction with the EOP does not discourage an operator from diagnosing the cause of the transient. The SPDS and EOP are based on directing the operator to take proper actions without diagnosis or with misdiagnosis.

To further enhance the operators' ability to assess the plants' response to transients and to more precisely monitor specific safety functions, additional displays were developed for the ANO-1 SPDS based on substantial operator and AP&L engineering input. These additional displays were carefully created to be used in conjunction with the ANO-1 EOP and will aid the operator in the implementation of this procedure as well as some select abnormal operating procedures. A description of these additional displays is provided below.

<u>Primary to Secondary Heat Transfer (PSHT)</u>

The primary to secondary heat transfer display was designed to provide more detailed historical data on some of the parameters shown on the ATOG diagram. Pages 1 and 2 of this display are shown in Figures 6 and 7, respectively.

Page one has trends for average core exit temperature, loop average hot leg temperature, loop average cold leg temperature and saturation temperature for steam generator pressure.

Page two displays a trend of feedwater flow, steam generator level and steam generator shell to tube temperature differential for both loops.

This display will be particularly valuable in natural circulation conditions. Verification of natural circulation can be readily accomplished by noting that core exit and hot leg temperatures are tracking and trending down. Also cold leg temperatures will be tracking the saturation temperature for that loop's steam generator pressure.

Reactivity Control (RHO)

The condition for entry into the EOP is a reactor trip, either manual or automatic. The reactivity control display will be provided to aid the operator in immediate verification that the reactor is indeed tripped and remains shut down. Intermediate and source range neutron flux are shown on this display along with indication of control rod index. A picture of this display is shown in Figure 8.

Steam Generator Tube Rupture (SGTR)

A steam generator tube rupture is treated as a unique event in both the Abnormal Transient Operating Guidelines and the EOP. This event is unique in that it has the potential for a direct release of radiation to the environment. Also, if the reactor does not trip initially, every effort is made to shut the plant down without a reactor trip and subsequent release of steam to the atmosphere through the main steam safeties. The steam generator tube rupture display, as shown in Figure 9, is designed to aid the operator in diagnosing a tube rupture, determining the affected generator, and cooling the plant down to a condition where the primary to secondary leakage can be terminated. This display shows trends of condenser off-gas radiation, steam generator shell to tube temperature differential and RCS average temperature.

RCS Inventory (RCSI)

The RCS inventory display, as shown in figure 10, is provided to aid the operator in assessment and maintenance of RCS inventory. Pressurizer level, RCS average temperature and pressurizer pressure are trended on the lower half of this display. On the top trend are reactor building pressure, reactor building radiation and reactor building temperature.

Trends shown on this display may also aid in the diagnosis of the initiating event. For example, the behavior of RCS temperature while pressurizer level and pressure are decreasing is the key to differentiating between a small loss of coolant accident and an

overcooling event. Both events will show decreasing pressure and pressurizer level; however, rapidly decreasing temperature would indicate an overcooling event while a constant or very slight decrease in temperature would indicate a loss of coolant.

Containment Conditions (CONT)

The containment conditions display is being developed to show reactor building parameters which may be useful in assessing and maintaining reactor building integrity. This display will provide trends of reactor building pressure, temperature, sump level, reactor building atmosphere hydrogen concentration and high-range radiation monitor indications. Several of these parameters are currently available in the SPDS data base and on other displays. The remainder are scheduled for addition during the next refueling outage.

Auxiliary Displays

In addition to the dedicated selector buttons on the screen for the primary SPDS displays described above, there is a dedicated button for a menu display from which auxiliary displays may be selected. Touching the menu button calls up a display showing a list of auxiliary displays with their associated selector buttons. When an "item" from the menu is selected, that display will appear and its label will be displayed in a blue backlit button at the bottom of the screen. Auxiliary displays include heatup and cooldown displays, the core exit thermocouple map, graphic trend displays, inadequate core cooling display and a pressure-temperature low range display. A one-line diagram of Engineered Safeguards (ES) electrical distribution system is planned when inputs are made available. Historical pressure versus temperature data from the previous 20 hours is saved and displayed on the heatup and cooldown displays when these displays are requested. These displays are not shown in this submittal.

As discussed above, the basis for the ANO-1 SPDS was the Emergency Operating Procedure development program. Implicit in this program was the consideration of the five critical safety functions identified in NUREG 0737 Supplement 1. Correlation between the previously described SPDS displays and the five critical safety functions are provided below:

1. Critical Safety Function - Reactivity Control

   SPDS Display on which function is assessed - Reactivity Control

   Parameters available - Reactivity Control
   - Intermediate range neutron flux
   - Source range neutron flux
   - Control rod index
   - Narrow range hot leg temperature

2. Critical Safety Function - Reactor core cooling and heat removal from the primary system

SPDS displays on which function is assessed - ATOG Diagram; Primary to Secondary Heat Transfer

Parameters available -

<u>ATOG Diagram</u>

- RCS pressure
- T_{hot} (loop average)
- T_{cold} (loop average)
- Core Exit Thermocouple (average of the five highest)
- Reactor building temperature (average)
- Reactor building pressure
- A/B steam generator pressures
- A/B steam generator levels
- Indication of saturation, 50° subcooled, and NDTT limit for given pressures and temperatures

<u>Primary to Secondary Heat Transfer</u>

- Core Exit Thermocouple (average)
- Hot leg temperatures (loop average)
- Cold leg temperatures (loop average)
- Steam generator saturation temperature (calculated from steam generator pressure)
- Steam generator levels
- Steam generator tube to shell temperature differential
- Feedwater flows to each generator

3.	Critical Safety Function -	Reactor coolant system integrity
	SPDS Displays on which - function is assessed	ATOG Diagram RCS Inventory
	Parameters available -	<u>ATOG Diagram</u> (See List Above) <u>RCS Inventory</u> •Pressurizer level •Pressurizer pressure •RCS average temperature •Reactor building pressure •Reactor building temperature •Reactor building radiation level
4.	Critical Safety Function -	Containment condition
	SPDS displays on which - function is assessed	ATOG Diagram Containment Conditions
	Parameter available -	<u>ATOG Diagram</u> (See List Above) <u>Containment Conditions</u> •Reactor building radiation •Reactor building hydrogen concentration •Reactor building temperature (average) •Reactor building pressure •Reactor building water level

5. Critical Safety Function - Radi vity Control

Comments: The four previous critical safety ions directly relate to one or more of the plant's barriers the release of radioactivity (i.e., fuel clad, RCS sure boundary and reactor building). The monitoring of these critical functions actually satisfies the goal of the safety function, radioactivity control. However, as ditional level of useful information to the operator concerni dioactivity control, offsite gaseous radioactivity release rmation has been included as desirable to be concisely ayed. This information may be used by control room personnel ckly assess the release data and make appropriate recom tions to offsite officials.

Effluent vent release data along with real meteorological data is input to the Gaseous Effluent Radia nitoring System (GERMS). The GERM system correlates this data de real time dose projections. GERMS output CRTs are independent of the SPDS but are located in the same facilities. This separate display technique is appropriate since the individuals who will actually be performing the dose assessment function are not the reactor operators who will be involved with the recovery of the plant. A detailed description of the GERM system and its capability was provided in our letter to Mr. John T. Collins dated October 21, 1982 (ØCAN1Ø8213).

The displays described above were designed to monitor both emergency and abnormal operations. Many of the parameters displayed are provided so that the operator can better understand the plant status, but are not essential for the monitoring of the five critical safety functions identified in NUREG 0737 Supplement 1. For these reasons the list of parameters described to monitor each critical safety function exceed the minimum set required. We believe the added information provided on the displays will enhance the operator's overall ability to understand the accident through the use of the SPDS and are thus justified.

Changes to the displays may be necessary in the future to further enhance the systems' capability. Such modifications will be properly reviewed prior to being implemented.

CONCLUSION

Based on the above discussion, AP&L has concluded that the ANO-1 SPDS design provides the control room operators with sufficient information to enable them to rapidly and reliably ascertain the safety status of the plant for a wide range of abnormal and emergency conditions. In addition, it has been concluded that the ANO-1 SPDS design provides sufficient information to be used with the upgraded emergency operating procedure to allow operator detection and mitigation of plant transients and accidents in a timely and accurate manner. Furthermore, the use of the SPDS will not mislead the operator and will not direct the operator to take improper actions.

The installation of the ANO-1 SPDS does not represent an unreviewed safety question or a change to the ANO-1 Technical Specification. In accordance with 10 CFR 10.59, the installation of the SPDS is being finalized and the safety evaluations are being documented as part of each Design Change Package.

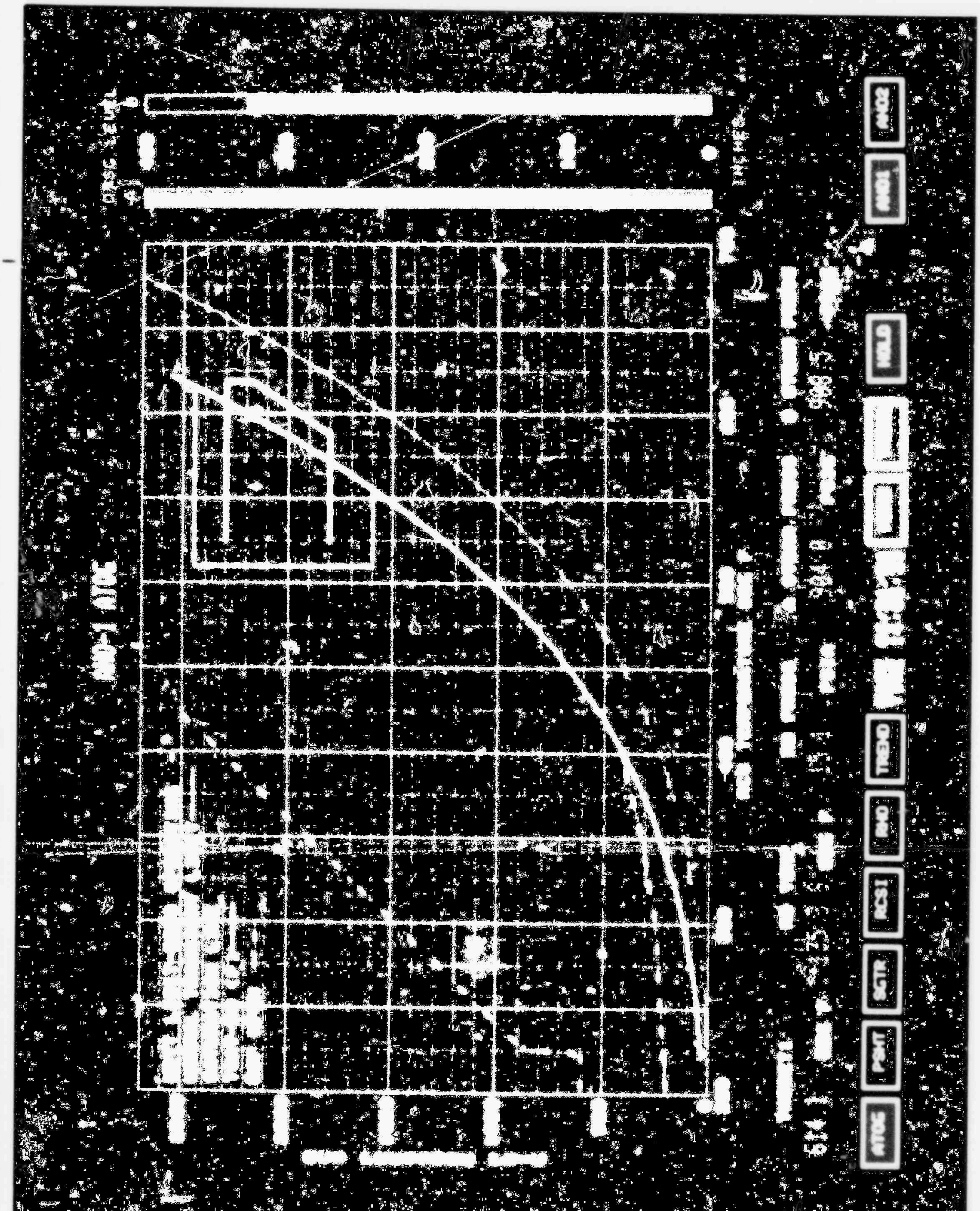


Figure 1. P-T Diagram

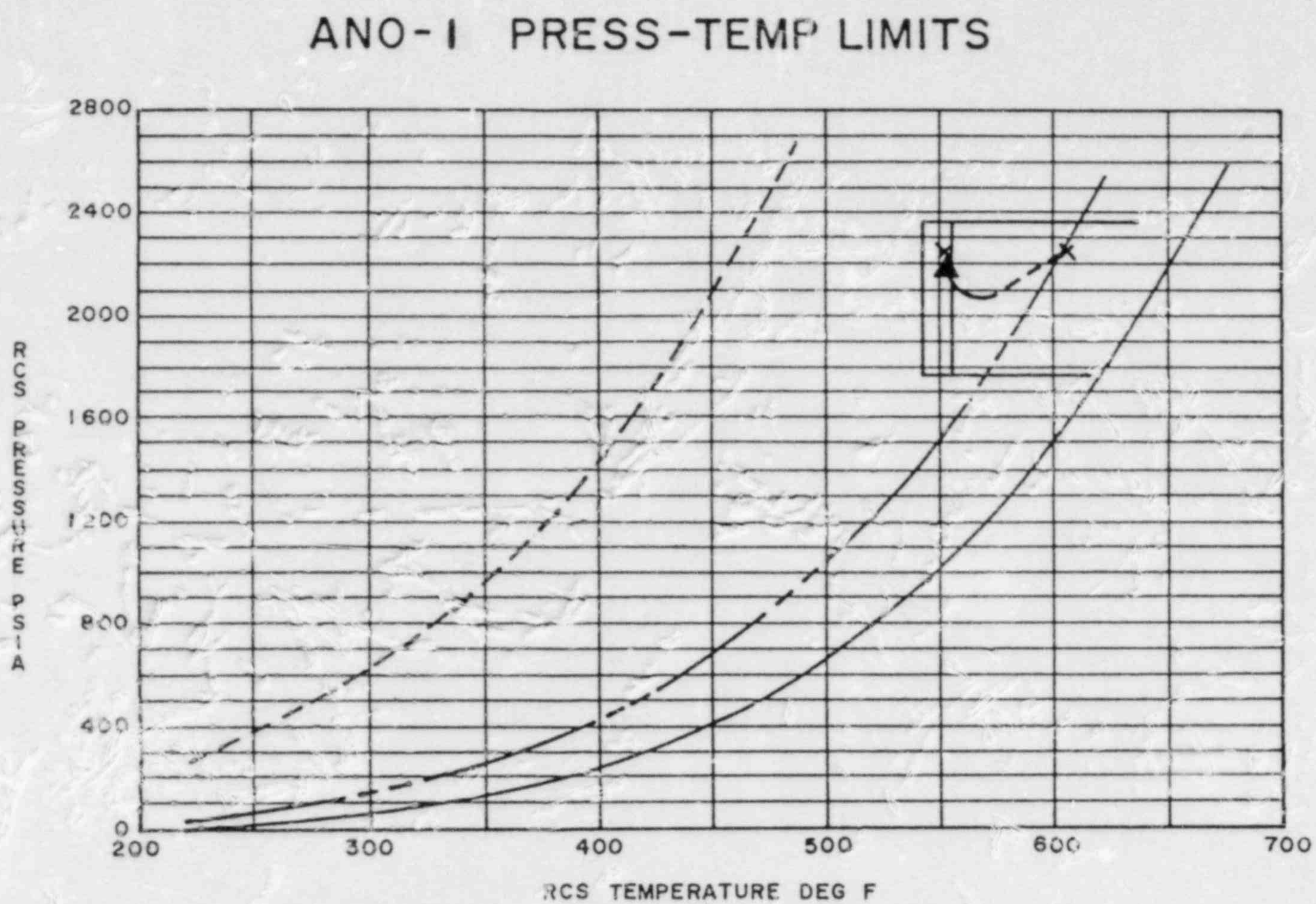


Figure 2. Typical Post-Trip Response

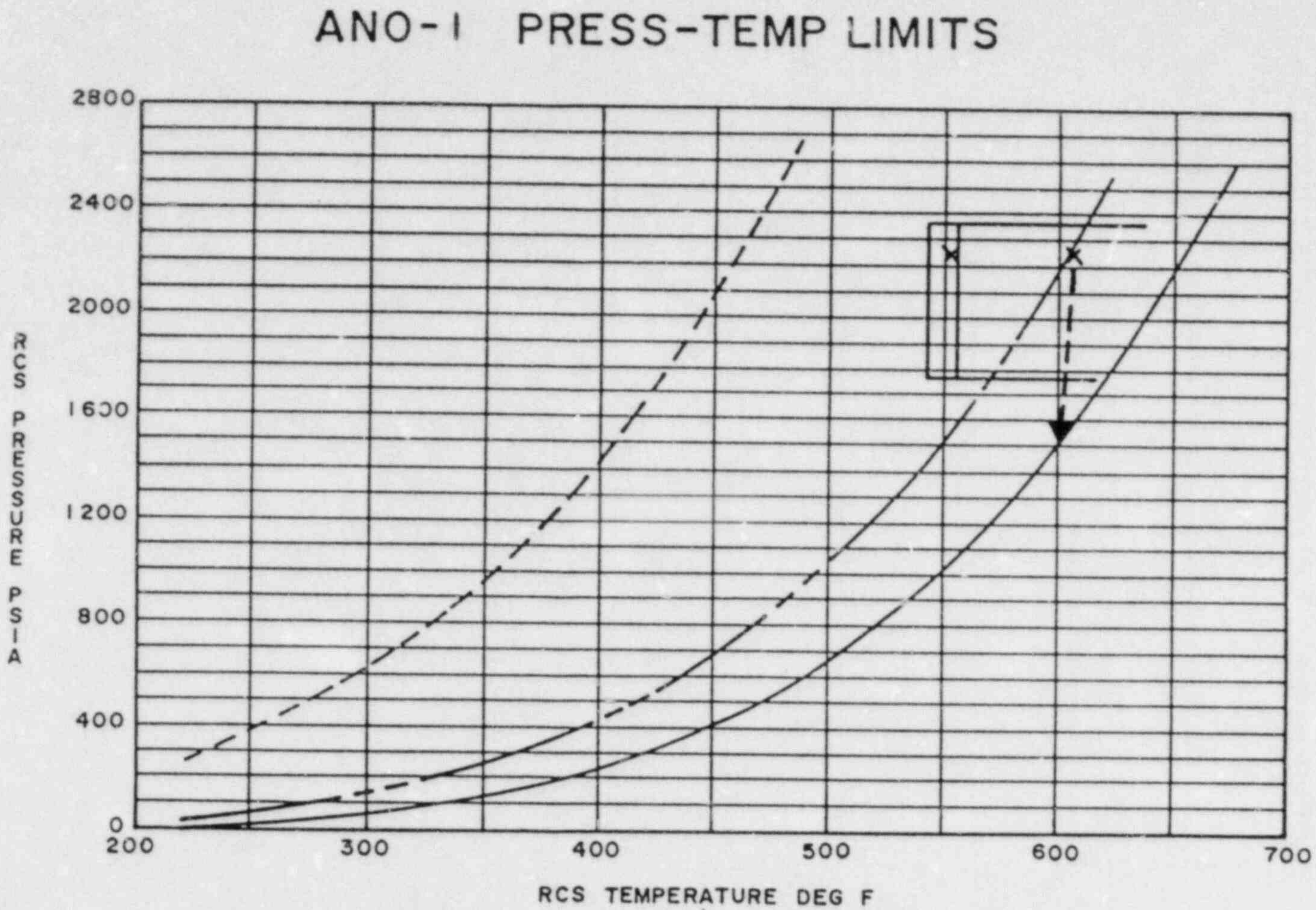


Figure 3. Inadequate Subcooling Margin: T_{hot} is not progressing toward its target value; in fact, it has rapidly dropped through the subcooled margin line. This condition is diagnosed as loss of adequate primary inventory subcooling, or simply "inadequate subcooling margin," and the procedure is written with directions to take care of inadequate subcooling margin.

ANO-1 PRESS-TEMP LIMITS

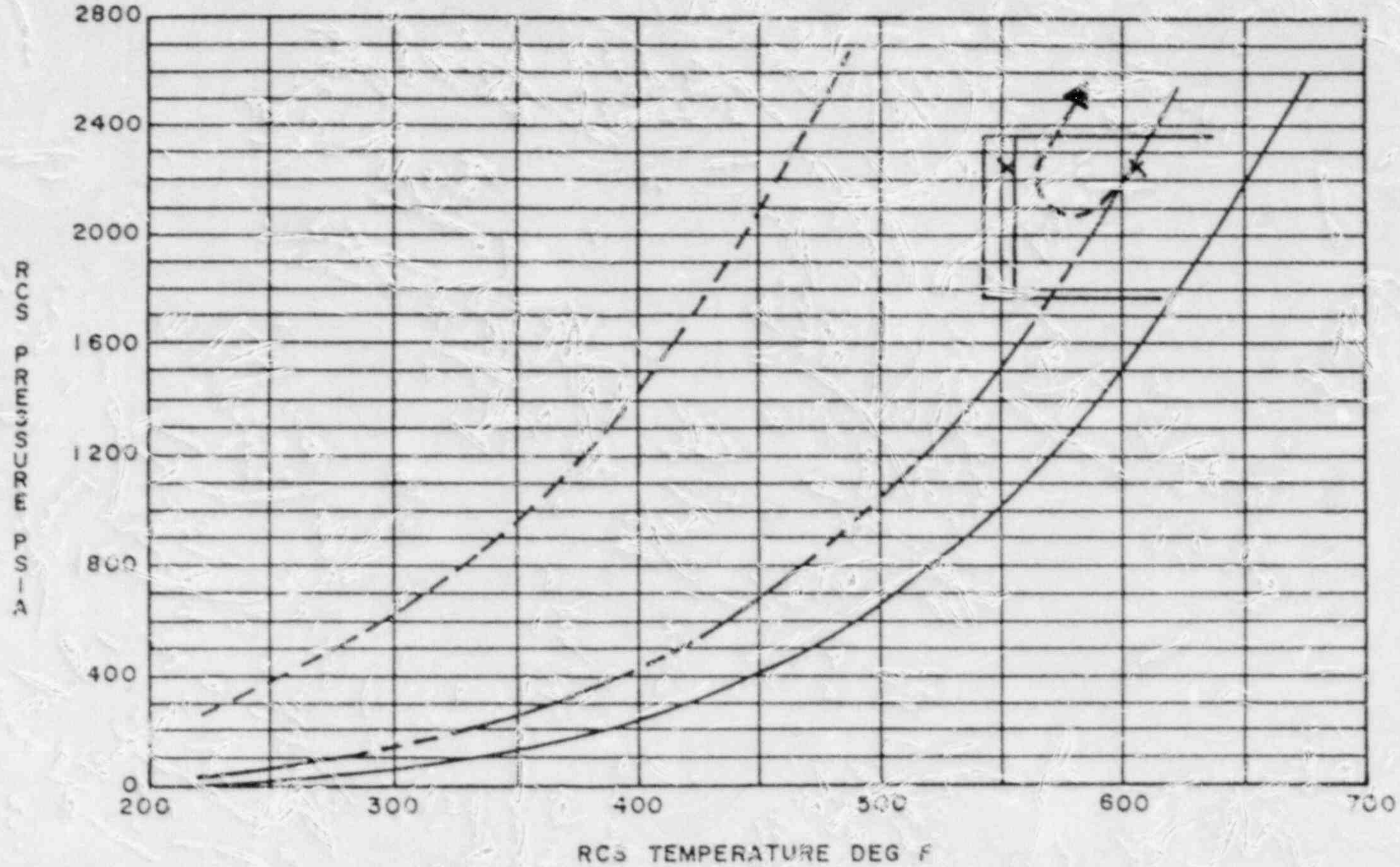


Figure 4. Loss of Primary-to-Secondary Heat Transfer: T_{hot} is increasing as steam generator T_{sat} is decreasing. A ΔT between the two is growing larger. The secondary is no longer removing heat and has lost coupling with the primary. This condition is diagnosed and treated as loss of (inadequate) primary-to-secondary heat transfer. SG T_{sat} is displayed as a digital readout below the P-T diagram.

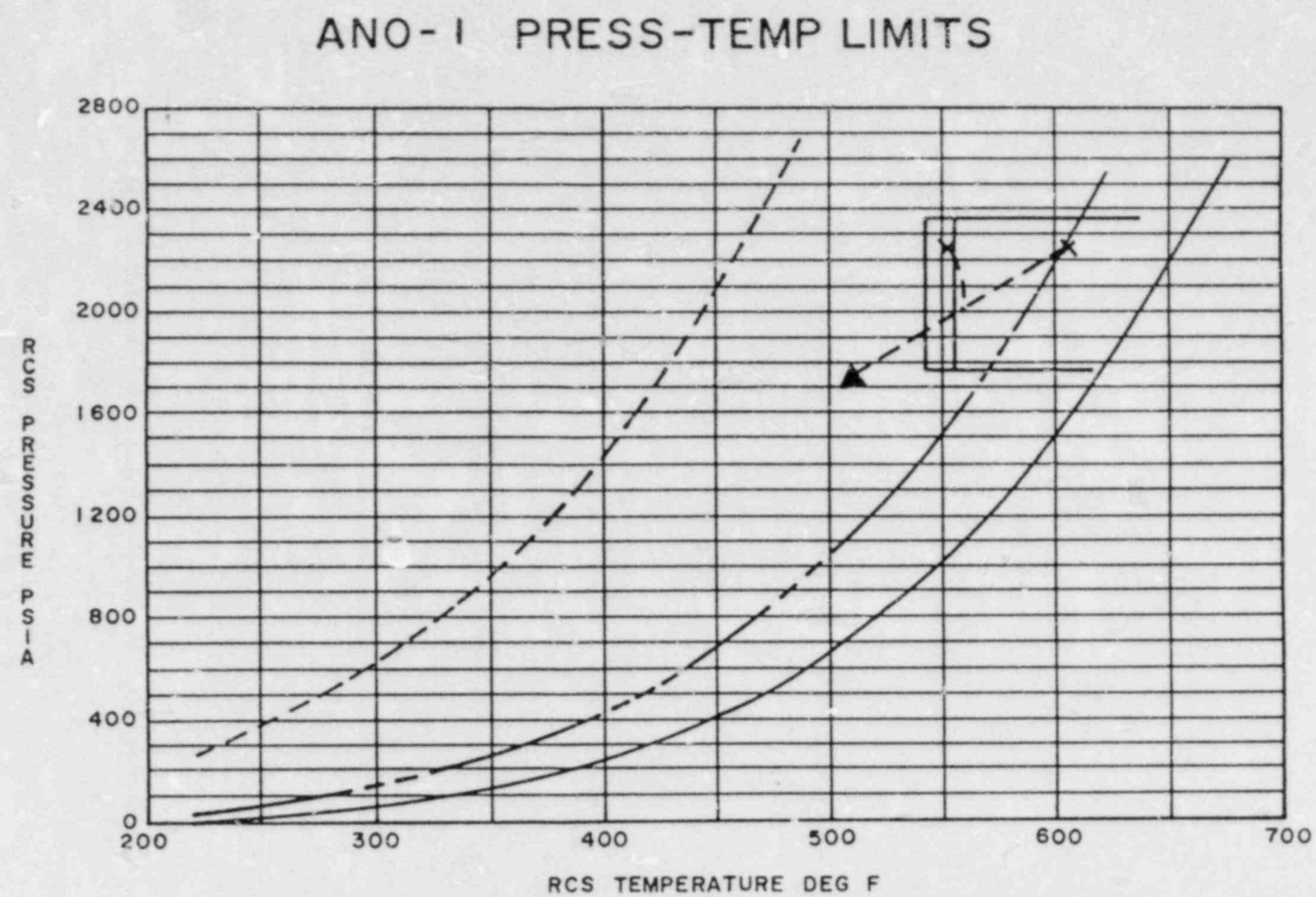


Figure 5. Excessive Primary-to-Secondary Heat Transfer: Steam generator T_{sat} has decreased below its established limit. T_{hot} and T_{cold} have reached equal values but both have gone out of the post-trip window following steam generator T_{sat}. This condition is diagnosed and treated as excessive primary-to-secondary heat transfer. SG Tsat is displayed as a digital readout below the P-T diagram.

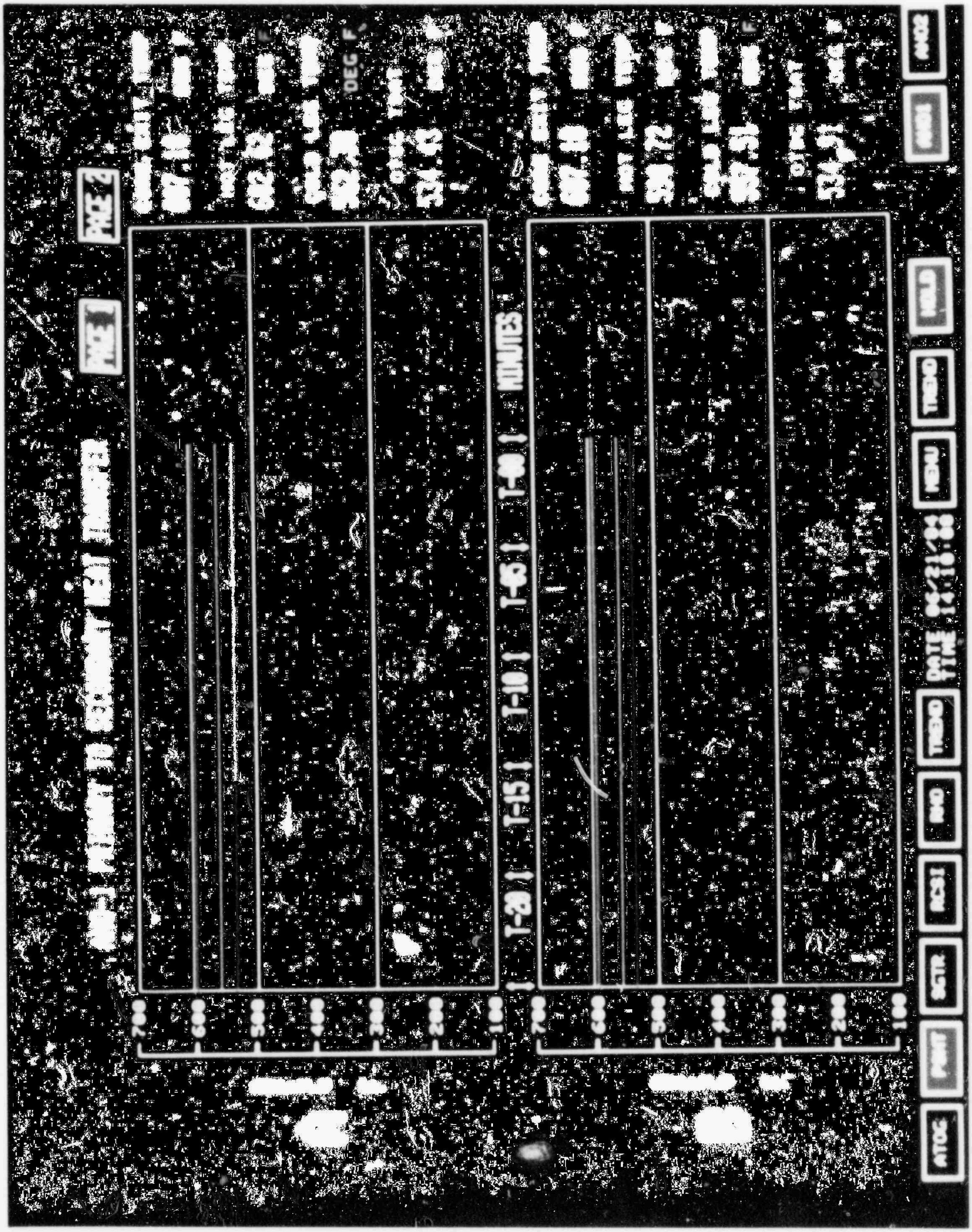


Figure 6. Primary to Secondary Heat Transfer - Page 1

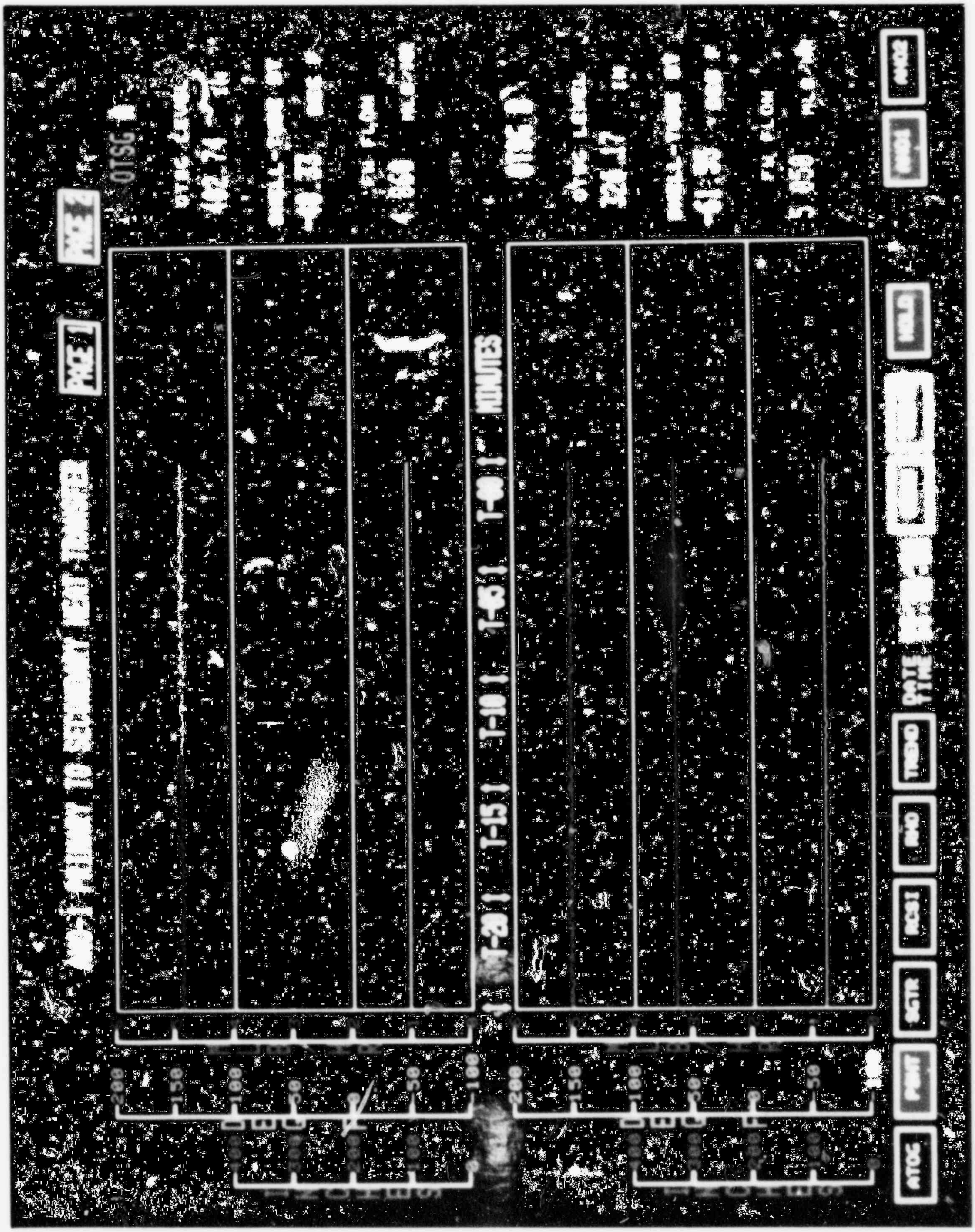


Figure 7. Primary to Secondary Heat Transfer - Page 2

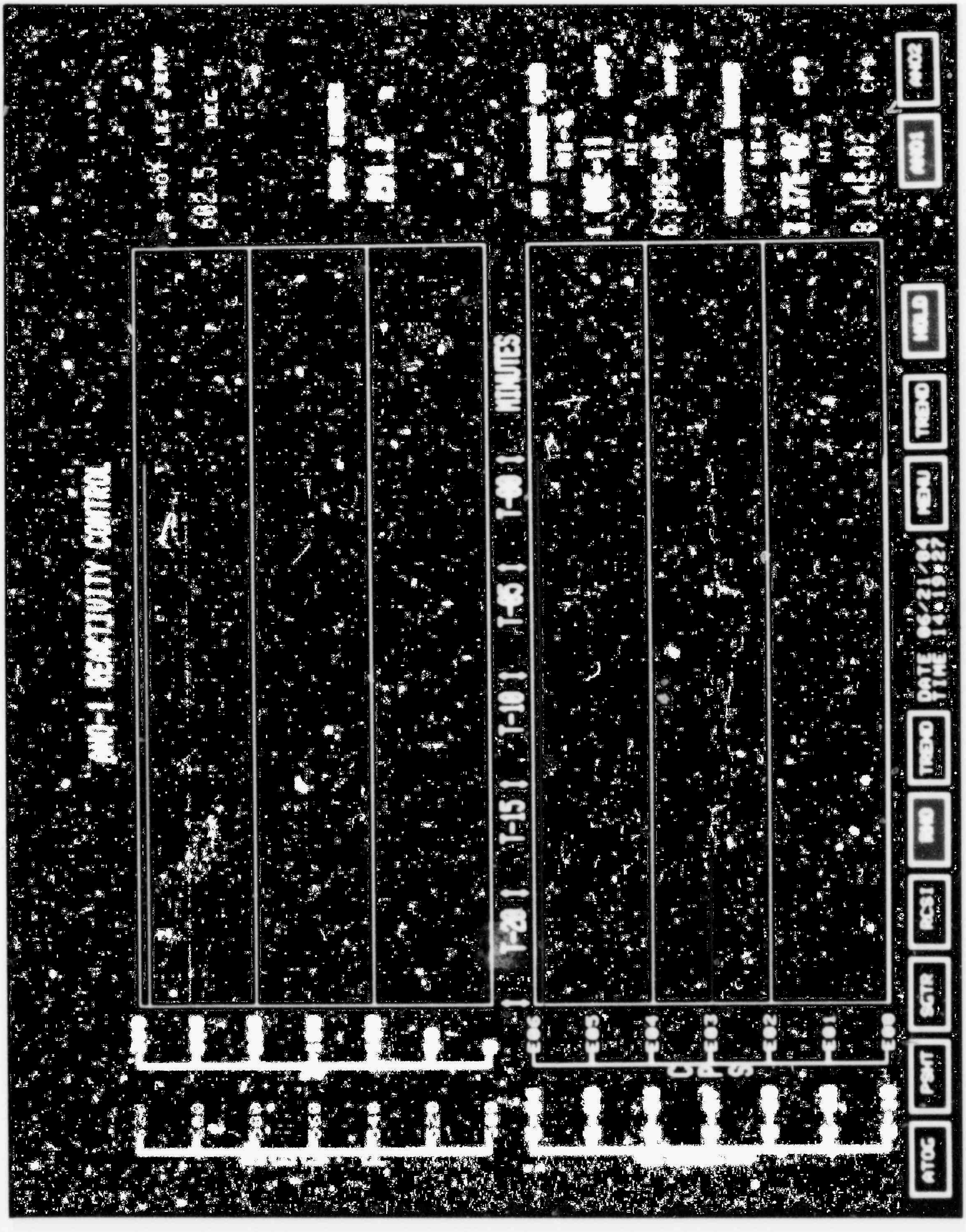


Figure 8. Reactivity Control

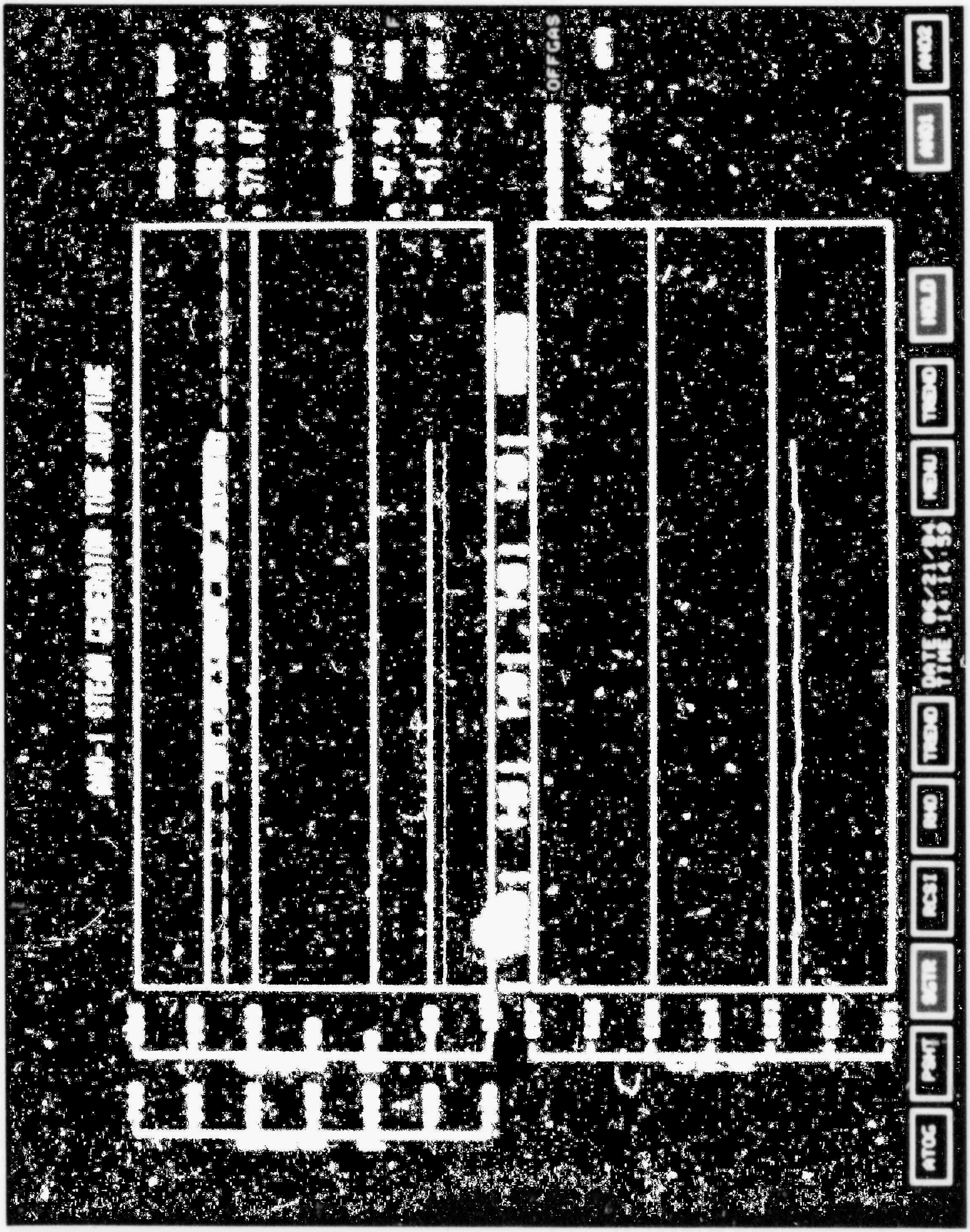


Figure 9. Steam Generator Tube Rupture

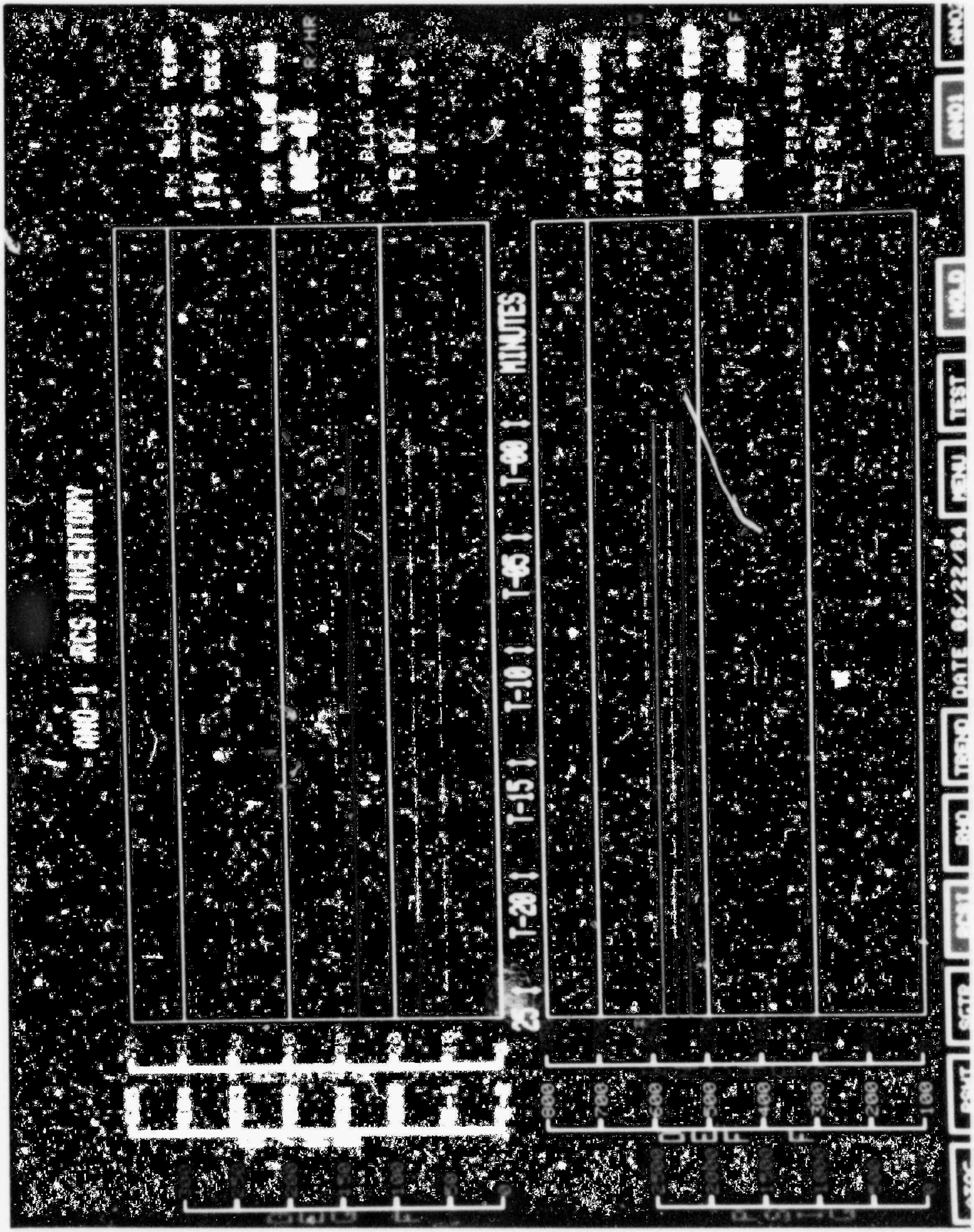


Figure 10. RCS Inventory