Enclosure to
LD-82-078

Natural Circulation Cooldown
of
C-E System 80 NSSS

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1.0 INTRODUCTION

1.1 Purpose

This report is the result of a study performed to evaluate a natural circulation cooldown in C-E's System 80 NSSS. A full plant cooldown will not necessarily be required whenever forced RCS flow is not available. However, conditions may exist which could warrant cooldown and depressurization of the RCS and may require that this be accomplished in an expeditious manner. Such conditions will most likely result in formation of a steam void in the reactor vessel upper head. The purpose of this report is to assess the procedures and instrumentation available for this evolution as well as to evaluate the likelihood of circumstances that may lead to such a situation.

1.2 Scope

This report is applicable to the C-E System 80 NSSS as described in CESSAR (Reference 1). It encompasses those aspects of the cooldown which effect the NSSS and does not specifically address Balance of Plant (BOP) Systems. Only that portion of the cooldown from hot standby conditions to the establishment of the Shutdown Cooling System (SDCS) entry condition is examined. Procedures are described in this report for the purpose of identifying the effects of various design features, however, this document does not constitute a procedure or a procedure guideline.

1.3 Background

In June of 1980, the operators at St. Lucie Unit 1 conducted a natural circulation cooldown of the C-E supplied NSSS. During this cooldown rapid variations in pressurizer level were noticed and subsequently reported and evaluated (References 2 and 3). The variations in pressurizer level were due to steam void formation in the Reactor Vessel Upper Head (RVUH) during depressurization. This region of the NSSS is essentially stagnate during natural circulation and hence was not cooled with the remainder of the RCS prior to depressurization.

Because of a perceived lack of industry training and procedures and a belief of increased susceptibility to more serious accidents, the NRC in Generic Letter 81-21 (Reference 4) stated that future natural circulation cooldowns should be conducted in a manner that would prevent void formation in the RCS. Analysis of this (Reference 5) indicated that such restrictions would result in overall cooldown times in the order of 25 to 30 hours for St. Lucie. Additionally, analyses indicated that there were no significant structural concerns associated with voiding. C-E therefore stated its position that voiding does not constitute a significant operational problem and that a more expeditious forced cooldown of the RVUH could be completed by voiding and filling in a manner similar to that which occurred in the St. Lucie cooldown of June, 1980.

During a discussion with the NRC staff on this subject relative to CESSAR (Reference 6), C-E indicated that System 80 had a RVUH free volume that was over twice as great as St. Lucie 1. This design would result in cooldown times on the order of 50 to 60 hours if void formation was to be prevented. C-E restated its position that the operator should have the capability to conduct a plant cooldown in a more expeditious manner. C-E agreed to provide a description of the procedures and instrumentation involved and an assessment of the scenarios that could result in the conduct of such a cooldown.

1.4 Report Summary

This report describes the operator actions necessary to perform an expeditious natural circulation plant cooldown with the controlled formation of a RVUH steam void. A natural circulation cooldown would be required when the plant must be brought to hot shutdown and Reactor Coolant Pump (RCP) flow is not available. Initially, after all RCPs have been tripped, a stable hot standby condition is established. The need to perform a natural circulation plant cooldown to the SDCS entry temperature and pressure may then arise in certain situations if forced circulation cannot be restored (see Section 4.0).

The actual process to achieve hot shutdown will depend upon plant conditions and on the amount of condensate available to the operator to conduct a cooldown and depressurize the RCS. Essentially, two basic alternative paths exist. Each alternative requires the operator to initially perform an RCS cooldown to the SDCS entry temperature while maintaining RCS pressure relatively high (i.e. above RVUH saturation pressure) without exceeding Technical Specification limits. The process of depressurizing to the SDCS entry pressure is different for each alternative. If circumstances require the natural circulation depressurization process to be completed expeditiously, a method using the formation, expansion and collapsing of a RVUH steam void is recommended. Otherwise, a prolonged plant depressurization period is required. In this case, the RCS pressure is held sufficiently high for a specified length of time and then gradually reduced to avoid the formation of a RVUH steam bubble. This report deals primarily with the first alternative.

The instrumentation that is available in System 80 provides the operator with sufficient information to conduct such an expeditious natural circulation plant cooldown. Specific instruments to monitor and to control a RVUH steam void during a natural circulation cooldown include RVUH Level, RVUH Temperature and RVUH Subcooled Margin along with other plant instruments. This provides the operator with sufficient information to depressurize a System 80 plant to SDCS entry pressure and temperature under natural circulation flow conditions with the controlled formation of a RVUH steam void, if circumstances warrant it.

It should be noted however, that an expeditious natural circulation cooldown of a System 80 plant due to any one of the three possible

RCP trip events identified in this report would be highly unlikely. These events are a loss of Component Cooling Water (CCW) to the RCPs, a loss of offsite power (LOOP), and a SIAS on low pressurizer pressure. With a loss of CCW event the availability of Seal Injection flow to the RCPs will continue to provide overheating protection of the pump seals. Therefore, the immediate need to conduct a RCS cooldown to prevent degradation of the pump seals would not arise. If a prolonged LOOP event were to occur a limited supply of condensate may conceivably require an expeditious cooldown of the plant under natural circulation conditions. However, offsite power would in most cases be restored in a reasonable amount of time thereby detering any need of performing a plant cooldown. Finally, a SIAS on low pressurizer pressure is an infrequent occurrence with regard to any plant, particularly if its occurrence is due to a loss of coolant accident (LOCA). A LOCA will inevitably result in the formation a RVUH steam void at any plant and is not therefore unique to a System 80 plant design. If a SIAS occurrence is not due to a LOCA, procedures will permit the restart of the RCPs.

From this review then, C-E maintains its position that the operators of System 80 plants should have the option to conduct an expeditious natural circulation cooldown. Although this will result in void formation in RVUH, the procedures described in the report and the instrumentation available will provide for adequate control. C-E feels that this is a sufficiently infrequent occurrence that the procedures described do not present inordinate additional challenge to operators or plant systems.

2.0 DESCRIPTION OF PROCEDURE

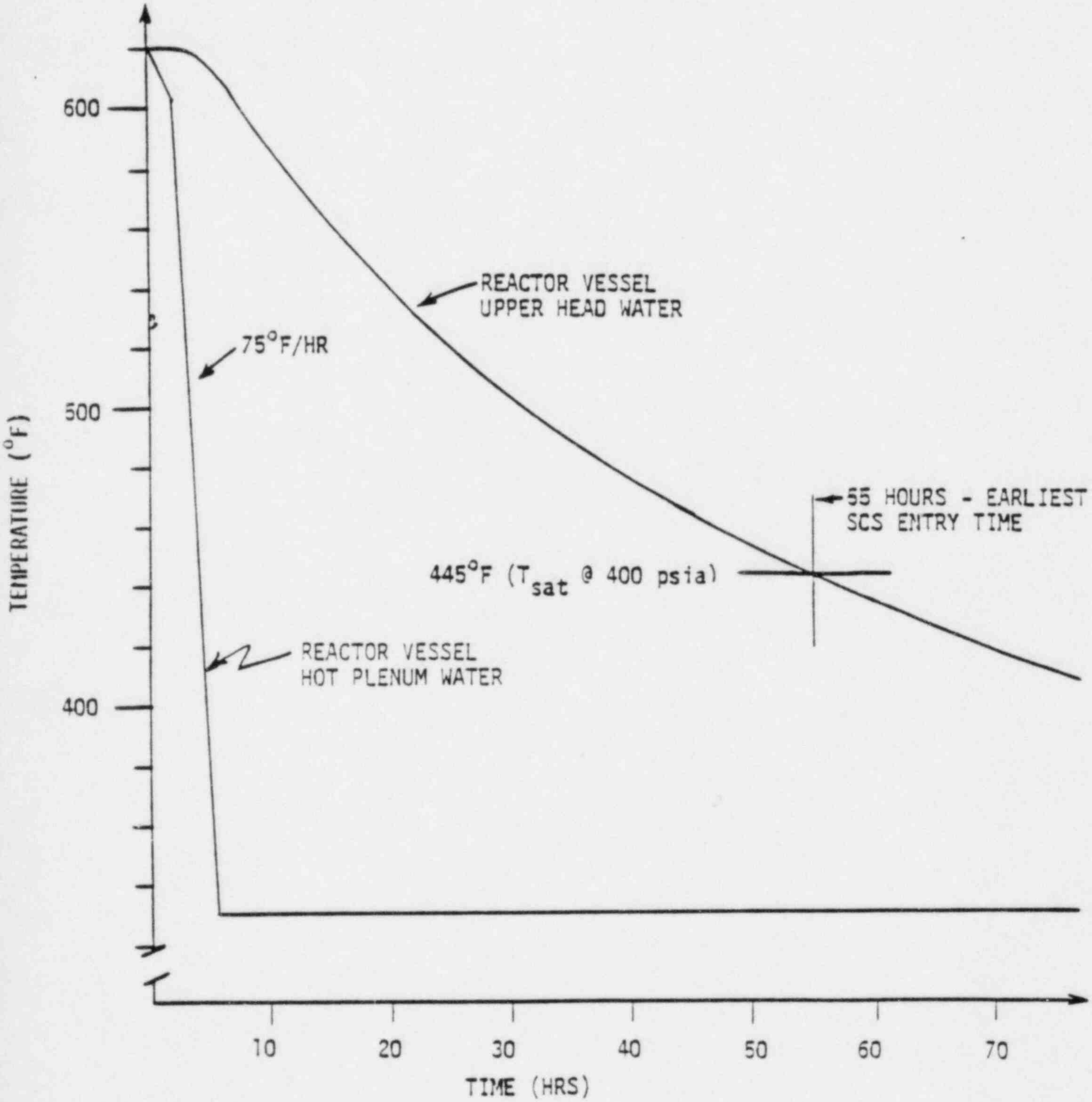
2.1 General

The most limiting factor in reducing RCS pressure from hot standby to SDCS initiation pressure is the amount of heat that the RVUH region will contain. During a natural circulation cooldown, the RCS loop is cooled by discharging secondary steam from the steam generators. However, the temperature of primary coolant in the upper head region of the reactor vessel (RV) will lag the loop temperature, as shown in Figure 2-1, because the flowrate through this region of the reactor may be very small or negligible without forced flow in the RCS. Thus the RVUH can contain a relatively stagnant volume of hot RCS coolant and metal. During primary depressurization, a condensible void could form in the RVUH by having the RCS pressure fall below the saturation pressure in this region. Maximizing the RCS cooldown rate and maintaining a relatively high RCS pressure will ensure maximum RVUH heat loss while preventing any possible void formation. Initially, the RCS cooldown rate should be maintained as high as possible without exceeding the maximum administratively controlled cooldown rate of 75°F per hour. Meanwhile, the operator should maintain the pressurizer pressure at normal operating pressure or as high as possible without exceeding Technical Specification pressure/temperature requirements by utilizing the pressurizer heaters and auxiliary spray. The operator must then depressurize the RCS to the SDCS entry pressure condition while facilitating RVUH cooling.

A specific method of cooling the RVUH by expanding and collapsing a RV steam dome bubble has been evaluated by C-E for use during natural circulation flow conditions. The analysis has shown that the RVUH cooldown rate can be accelerated by a RV head fill and drain process without any thermal stress or fatigue damage to reactor vessel components (Reference 5). This method allows the plant to be safely depressurized and the SDCS aligned in an

FIGURE 2-1

SYSTEM 80 REACTOR VESSEL UPPER HEAD TEMPERATURE DURING RCS NATURAL CIRCULATION COOLDOWN



expeditious manner. The procedures for performing a complete natural circulation cooldown to SDCS entry conditions with the controlled formation of a RVUH steam void is basically divided into four processes. First, steady state natural circulation conditions are established, then the RCS boron concentration is increased to cold shutdown requirements. The third process is the cooldown of the RCS to the SDCS entry temperature while the operator maintains a RCS pressure relatively high. The final process is depressurization of the RCS during which the operator implements a "forced cooling" method of reducing RVUH temperature by first voiding and then collapsing the void thereby refilling the RVUH with cooler RCS water. This process may be aided by the use of the Reactor Vessel Gas Vent System. Specific recommendations and a description of the procedures for performing this task are provided for in this section. Appendix A contains plotted results of a computer simulation of an expeditious natural circulation cooldown of the System 80 NSSS as outlined below.

2.2 Establishing Natural Circulation

The first part of the overall process to reach the SDCS entry temperature and pressure after a loss of forced circulation (LOFC) flow is the establishment of a steady-state natural circulation flow condition in the RCS. Once natural circulation has been established the operator is able to maintain the plant at hot standby and if necessary, continue to perform an expeditious natural circulation cooldown of the plant.

A LOFC event is characterized by reactor turbine and generator trips accompanied by negligible steam generator ΔP's and RCP ΔP's. Depending on the type of failure, there will also be "RCP trouble" alarms or abnormal RCP motor currents.

Immediately following the initial response of a LOFC event, certain operator actions must be performed in order to establish, maintain and verify a steady-state natural circulation flow at hot standby conditions. A general description of these follow-up actions are as follows.

1. Verify that the standard post-trip actions have been initiated.

2. Maintain the RCS within the acceptable post-accident pressure/temperature limits by

   a. Controlling RCS heat removal via the steam generators

   and

   b. Controlling RCS pressure using
      i) Pressurizer heaters and auxiliary spray
      ii) Charging and letdown
      iii) HPSI pumps

3. Maintain the RCS hot leg subcooled margin of at least 20°F + (inaccuracies).

4. Steam generator pressure should be controlled by the turbine bypass system. If condenser vacuum is lost, the turbine bypass system is not available, or if the MSIVs have closed, the atmospheric dump valves must be used to control steam generator pressure.

5. The Pressurizer Pressure Control System (PPCS) is verified to be automatically controlling or restoring RCS pressure. If not, pressurizer heaters or auxiliary spray are operated manually to control pressurizer pressure.

6. The Pressurizer Level Control System (PLCS) is verified to be automatically controlling or restoring pressurizer level. If

not, charging and letdown are operated manually to ensure pressurizer level is being maintained. Pressurizer level should normally be maintained at the normal shutdown reference level throughout the plant cooldown if a cooldown is necessary. If letdown is not available, pressurizer level may be allowed to vary over the full range of the pressurizer as long as care is taken not to go solid or uncover pressurizer heaters.

Verify, by the following indications, that natural circulation flow has been established within 5-15 minutes after all RCPs have tripped:

a. Loop ΔT (T_h - T_c) less than normal full power ΔT.
b. Cold leg temperatures constant or decreasing.
c. Hot leg temperatures stable or decreasing slowly (i.e., not steadily increasing).
d. No abnormal differences between T_h RTDs and core exit thermocouples.

If the RCS hot leg subcooled margin approaches or becomes less than 20°F + (inaccuracies) and/or natural circulation degradation is suspected, try to enhance natural circulation flow by:

a. Increasing turbine bypass or atmospheric steam dump flow to reduce RCS temperatures.
b. Increasing RCS pressure with pressurizer heaters or by operating safety injection or charging pumps.
c. Verifying adequate secondary water level.
d. Verifying adequate primary water inventory without any voids.
e. Verifying adequate subcooled margin.

9. Maintain the plant in a stabilized condition and evaluate the need for a plant cooldown based on plant conditions, auxiliary systems availability, and condensate inventory.

10. If required, conduct a plant cooldown to SDCS initiation conditions as addressed in the following sections.

2.3 RCS Boration

The second part of the natural circulation cooldown scenario is to borate the RCS in accordance with Technical Specifications requirements. Normally the RCS should be borated to the cold shutdown boron concentration level before starting a plant cooldown. However, normal boration procedures may not be available to change RCS boron concentration if the letdown system is not operable due to a LOOP event. In this case, pressurizer level will be increased by charging highly borated makeup water to raise the RCS boron concentration before commencing a plant cooldown. Shrink of RCS coolant during subsequent cooldown will allow for additional boration.

As the RCS boron concentration is changed, the auxiliary spray should be used to normalize the pressurizer and RCS boron concentration. This may require auxiliary spray flow beyond what is needed for depressurization. Pressurizer heaters should be used to offset the depressurization caused by intermittent auxiliary spray. An alternative to using spray flow for pressurizer boron control is to increase the RCS boron concentration higher than the required RCS cold shutdown concentration to account for a potential dilution from the pressurizer water. Thus, if a pressurizer outsurge occurs, mixing of the water from the pressurizer with the RCS loop water will not dilute the boron concentration below the cold shutdown concentration requirement.

Another region of reactor coolant which may contain less boron than the RCS loop water as cold shutdown boration progresses is the reactor vessel upper head areas. This potentially low borated water will be slowly flushed out throughout the natural circulation cooldown, quickly flushed out following subsequent start of a RCP, or forced out during formation of any RV upper head void.

After a cold shutdown boron concentration is attained in the RCS, makeup water added to the RCS during the cooldown should be at least the same boron concentration as in the RCS to prevent any dilution of RCS boron concentration.

The procedures necessary for RCS boration are summarized below.

1. Align the CVCS for boration of the RCS by lining up the available Charging Pumps to take suction from the Refueling Water Storage Tank via:

    a. The Boric Acid Makeup Pumps, <u>or</u>

    b. The gravity feed line.

2. With letdown, borate the RCS to the cold shutdown boron concentration. If letdown is <u>not</u> available, start the Charging Pumps and establish a pressurizer level of 1400 ft^3 (80% Indicated Level).

3. Commence cooldown when:

    a. An indicated pressurizer level of 80% has been attained with letdown <u>not</u> available, <u>or</u>

    b. The RCS has attained the cold shutdown boron concentration.

4. Continue to borate the RCS in accordance with Technical Specification requirements.

2.4 RCS Cooldown

Following boration, the next phase of the process is RCS temperature reduction. During the RCS cooldown phase, a maximum cooldown rate within Technical Specification limits is recommended to enhance the conductive cooling capability of the RVUH region. A large temperature difference between the RV head and the RCS coolant will provide a large thermal gradient and a greater heat transfer rate. Also during the cooldown phase, the operator should maintain the RCS pressure relatively high (i.e. above 1800 psia) within the acceptable pressure/temperature limits. This strategy will minimize the possibility of a void resulting from flashing in the stagnant RVUH region during cooldown. A RVUH void will not form until the RCS pressure is decreased to below the RVUH saturation pressure, regardless of the RCS cooldown rate used. The depressurization phase of the overall cooldown process to reach SDC is discussed in Section 2.5.

A general description of the procedures necessary to perform the RCS cooldown is presented below.

1. Commence an RCS cooldown by performing one of the following (listed in order of preference):

   a. If the condenser and turbine bypass system are available, commence the cooldown using the turbine bypass system and main or auxiliary feedwater.
   b. If the condenser or the turbine bypass system are not available, commence the cooldown using the atmospheric dump valves and main or auxiliary feedwater.

2. Establish a maximum cooldown rate in accordance with Technical Specification limit. Maintain the RCS pressure above 1800 psia and within the acceptable pressure/temperature limits.

3. Continuously verify natural circulation flow throughout the cooldown process using the following criteria:

    a. Loop ΔT (T_h - T_c) less than normal full power ΔT.
    b. Cold leg temperatures constant or decreasing.
    c. Hot leg temperatures constant or decreasing.
    d. No abnormal differences between T_h RTDs and core exit thermocouples.

4. If the RCS hot leg subcooled margin approaches or becomes less than 20°F + (inaccuracies) and/or natural circulation degradation is suspected, try to enhance natural circulation flow by:

    a. Increasing turbine bypass or atmospheric steam dump flow to reduce RCS temperatures.
    b. Increasing RCS pressure with pressurizer heaters or by operating safety injection or charging pumps.
    c. Verifying adequate secondary water level.
    d. Verifying adequate primary water inventory without any voids.
    e. Verifying adequate subcooled margin.

5. Maintain normal steam generator water level throughout the plant cooldown. Initiate the auxiliary feedwater system if the main feedwater system is not able to operate adequately.

6. Maintain the pressurizer level at or near 40% with the charging and letdown system throughout the plant cooldown if possible.

7. Once the RCS hot leg has reached below the SDCS entry temperature of 400°F evaluate the need to reach the SDCS entry pressure condition based on plant conditions, auxiliary systems availability, and condensate storage tank inventory.

8. If required, conduct a depressurization of the RCS as presented in the following section.

2.5 RCS Depressurization

The final part of the natural circulation cooldown process is the depressurization of the RCS. This method allows the plant to be safely depressurized and the RVUH to be cooled in a substantially shorter time. Specifically, the operator performs a RCS controlled depressurization allowing a RVUH steam void to develop. This condensible void in the RV can then be controlled by either lowering or raising system pressure allowing the void to either expand or shrink. During void expansion, hot stagnant water is forced down into the RCS natural circulation flow path and carried away. Subsequently system pressure is increased compressing the void and filling the RVUH upper head region with cool water. Hence, a quick method of removing heat from the RVUH region is provided by a controlled drain and fill process. Once the operator has completed several drain and fill process during RCS depressurization, the plant can enter SDC operation.

A general description of the procedures to be used for the depressurization method are summarized below.

1. A pressurizer level of between 35% and 50% must exist before commencing a RCS depressurization.

2. Commence a pressurizer cooldown and RCS depressurization by manually operating the auxiliary spray.

3. Maintain pressurizer cooldown rate within Technical Specification requirement.

4. Maintain at least 20°F + (inaccuracies) subcooled margin in the RCS loops based on T_H RTDs or core exit thermocouples.

5. Reset or bypass Engineered Safeguards Features and reduce safety injection tank pressures as required due to the decreasing primary and secondary pressures.

6. During the RCS depressurization, monitor for condensible void formation. Anticipate void formation at the saturation pressure corresponding to the indicated RVUH temperature Symptoms of void formation are:

   a. RVUH saturation margin of $\leq$ 0°F.
   b. Pressurizer level increases significantly greater than expected while operating auxiliary spray.
   c. Letdown flow unexpectedly greater than charging flow if the pressurizer level control system is in automatic.
   d. Void level formation as indicated by the reactor vessel level monitor.

7. When a condensible void formation in the RVUH is indicated perform the following:

   a. Continue the RCS depressurization allowing the RVUH to expand.
   b. Stop the depressurization when
      i. The pressurizer level has increased to 90% indicated level, <u>or</u>
      ii. The RVUH void level monitor indicates a minimum value of 16% (or 3 ft. above the Upper Guide Structur (UGS) Support Plate).
   c. Repressurize the RCS by energizing all available pressurizer heaters and/or commencing charging flow RCS loop. RV head vent may be opened to aid in void collapse.
   d. Stop the charging flow and deenergize the pressurize heaters when

    i. The pressurizer level stops decreasing and begins a normal steady increase due to charging or when the pressurizer level decreases, <u>and</u>
    ii. The RVUH void level has collapsed as indicated by the RV level monitoring system.

  e. Repeat the above four steps again for several drain and fill cooling cycles of the RVUH while monitoring RVUH temperature.
  f. If indication of a relatively constant void persists at the same system pressure, consider the void to contain noncondensible gases and follow the procedure for removing noncondensible gases.
  g. Resume the RCS cooldown and depressurization when RVUH temperature is below saturation temperature.

8. If void indication cannot be eliminated by implementing actions for condensible gas, consider the gases to be partially or completely noncondensible gases.

  a. Increase pressurizer pressure above the pressure where void symptoms were originally notice.
  b. Operate the RV head vent* as needed to eliminate the noncondensible gases.

9. Enter SDC.

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*RV head vent may also be operated to aid in cooling the RV head by venting during boration and cooldown provided sufficient RCS makeup is available from the charging system.

3.0 INSTRUMENTATION FOR MONITORING COOLDOWN

3.1 General

The purpose of this section is to demonstrate that the operator has sufficient instrumentation available to correctly respond to a natural circulation cooldown event with the controlled formation of a RVUH steam void. Specifically, a technical description of the available instrumentation and the monitoring systems, and the manner in which they are used by the operator, is provided.

Most instruments described in this section for use during a natural circulation cooldown event are the product of several evaluations performed by C-E on the response characteristics of Inadequate Core Cooling (ICC) detection methods. Although a natural circulation cooldown is not an ICC event, the process can still be monitored by the same ICC instruments thereby increasing the information available to the operators in this mode of operation. Results of the initial instrument studies are documented in the C-E Owners Group reports CEN-117 and CEN-125 with further studies documented in CEN-181 and CEN-185 (References 9 through 12). All studies provided detailed analyses of the existing instruments, as well as an investigation of the characteristics of selected new instruments. As a result of these evaluations and further design modifications performed by C-E, an ICC instrumentation sensor package is now dedicated to monitor a natural circulation cooldown event for System 80. The major purpose of this instrument package is to provide the operator with a continuous, unambiguous, easy-to-interpret indication of the hydraulic states within the RV. This instrumentation sensor package consists of the following components:

1) hot and cold leg Resistance Temperature Detectors (RTDs)

2) pressurizer pressure sensors

3) Core Exit Thermocouples (CETs)

4) Reactor Vessel Level Monitoring System (RVLMS) probes employing the Heated Junction Thermocouple (HJTC) concept.

All the above sensor inputs have been integrated into a processing, control, and display system used primarily for core heat removal safety functions. This system is referred to as the Accident Monitoring System (AMS) which consists of two major subsystems:

1. Critical Function Monitoring System (CFMS)

2. Qualified Safety Parameter Display System (QSPDS)

A functional overview of the AMS highlighting the above sensor inputs is shown in Figure 3-1 (this is described in Appendix B to CESSAR - Reference 1). The instrument sensors are input to the two channel QSPDS for processing and then transmitted to the CFMS for primary display and trending. The QSPDS also functions as a safety grade backup display to the CFMS for key safety parameters. Specifically, the AMS provides the operator with three major parameters which can be used to monitor a expeditious natural circulation cooldown event. These parameters include:

1. saturation margins

2. reactor vessel inventory/temperature above the core

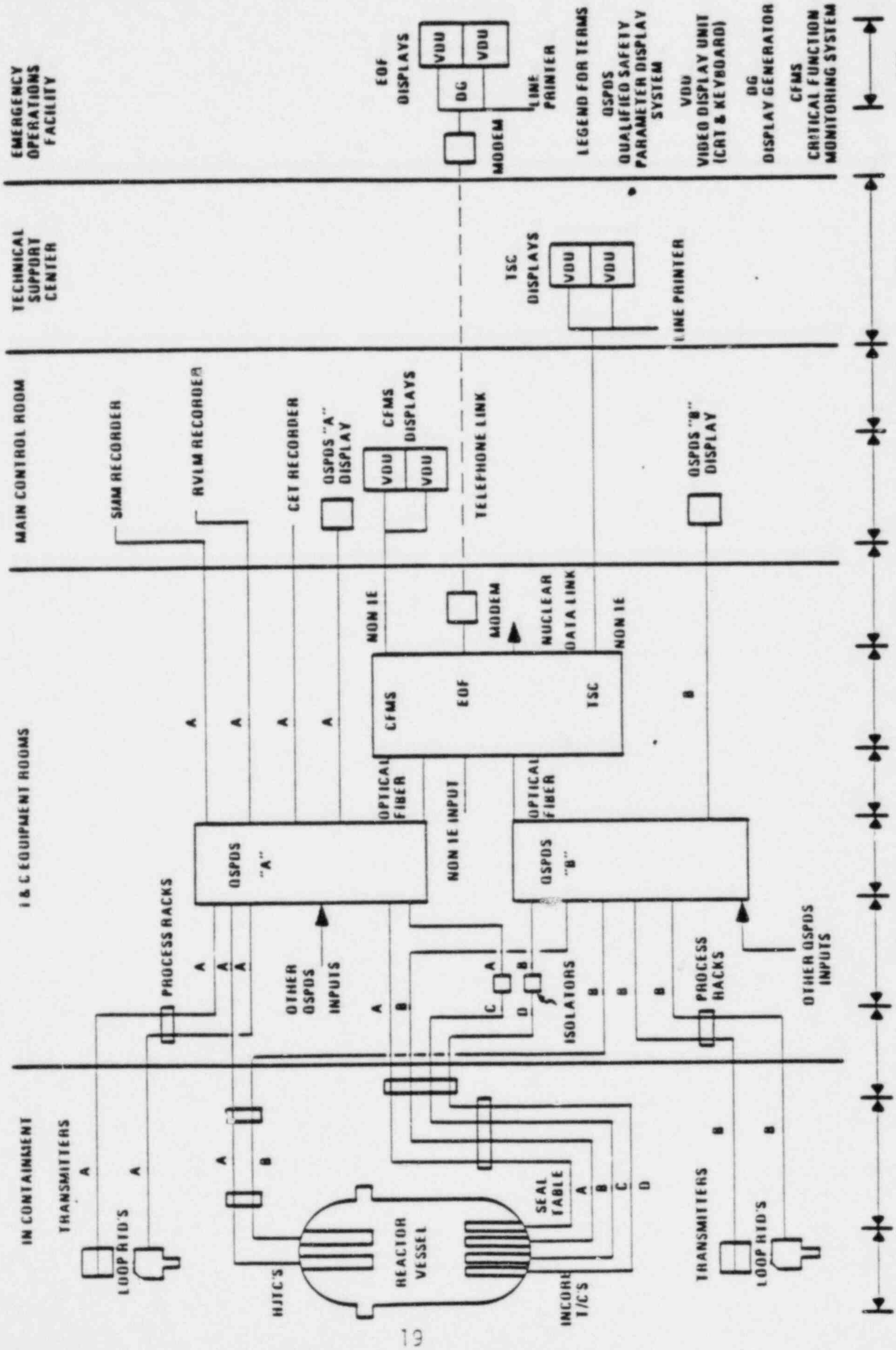
3. and core exit temperature.

The AMS instrumentation systems associated with providing each of these key parameters is discussed in further detail in Subsections 3.2, 3.3 and 3.4 respectively.

Other instrumentation systems required for the monitoring of natural circulation cooldown are the pressurizer pressure indicator, pressurizer level indicator, and the hot and cold leg temperature

FIGURE 3-1

C-E ACCIDENT MONITORING SYSTEM



indicators using Resistance Temperature Detectors (RTDs). These items are discussed in Subsection 3.5.

3.2 Saturation Margin Monitor

The Saturation Margin Monitor (SMM) is a AMS feature which provides information to the reactor operator on the approach to and existence of saturation.

The SMM includes inputs from RCS cold and hot leg temperatures measured by RTDs, the temperature of the maximum of the top three Unheated Junction Thermocouples (UHJTC), and pressurizer pressure sensors. The UHJTC input comes from the output of the HJTCS processing units. In summary, the SMM sensor inputs and their associated ranges are as follows:

Input	Range
Pressurizer Pressure	0-4000 psia
Cold Leg Temperature	0-750°F
Hot Leg Temperature	0-750°F
Maximum UHJTC Temperature of Top Three Sensors (from HJTC Processing)	200-2300°F
Representative CET Temperature	200-2300°F

Using the above SMM inputs, the QSPDS processing equipment of the AMS will then perform the following functions:

1. Calculate the saturation margin

The saturation temperature is calculated from the minimum pressure input. The temperature, subcooled or superheat margin, is the difference between saturation temperature and the sensor temperature input. Three temperature presentations, subcooled or superheat margin, will be available. These are as follows:

   a. RCS saturation margin - temperature, saturation margin based on the difference between the saturation temperature and the maximum temperature from the RTDs in the hot and cold legs.

   b. RVUH saturation margin-temperature, saturation margin based on the difference between the saturation temperature and the UHJTC temperature (based on the maximum of the top three UHJTC).

   c. CET saturation margin-temperature, saturation margin based on the difference between the saturation temperature and the representation core exit temperature calculated from the CETs.

2. Process sensor outputs for determination of temperature saturation margin.

3. Provide an alarm output for an annunciator when temperature saturation margin reaches a preselected setpoint for RCS or upper head saturation margin. CET saturation margin is not alarmed to avoid possible spurious alarms.

Following the processing described above, the information listed below is then presented on the primary (CFMS) and backup (QSPDS) displays:

1. Temperature and pressure saturation margins for RCS, Upper Head, Core Exit Temperature.

2. Temperatures and pressure inputs.

During natural circulation cooldown, the information supplied by the SMM will be used to determine when, during the primary system depressurization, saturation temperature and pressure are reached. It is at this point where boil-off will occur and formation of a void in the RV upper head will begin.

3.3 Core Exit Thermocouples

The core exit thermocouples (CETs) is a In-Core Instrumentation (ICI) system used to monitor the coolant temperature at the core exit. The design of the System 80 ICI has been modified with improved Type K (Chromel-Alumel) thermocouples within each of the ICI detector assemblies. The CETs will be located inside the ICI support tubes, a few inches above the fuel alignment plate. The core locations of the ICI detector assemblies are shown in Figure 3-2.

The CETs have a usable temperature range from 200°F to up to 2300°F.

The following constitutes a description of the CET processing functions performed by the QSPDS:

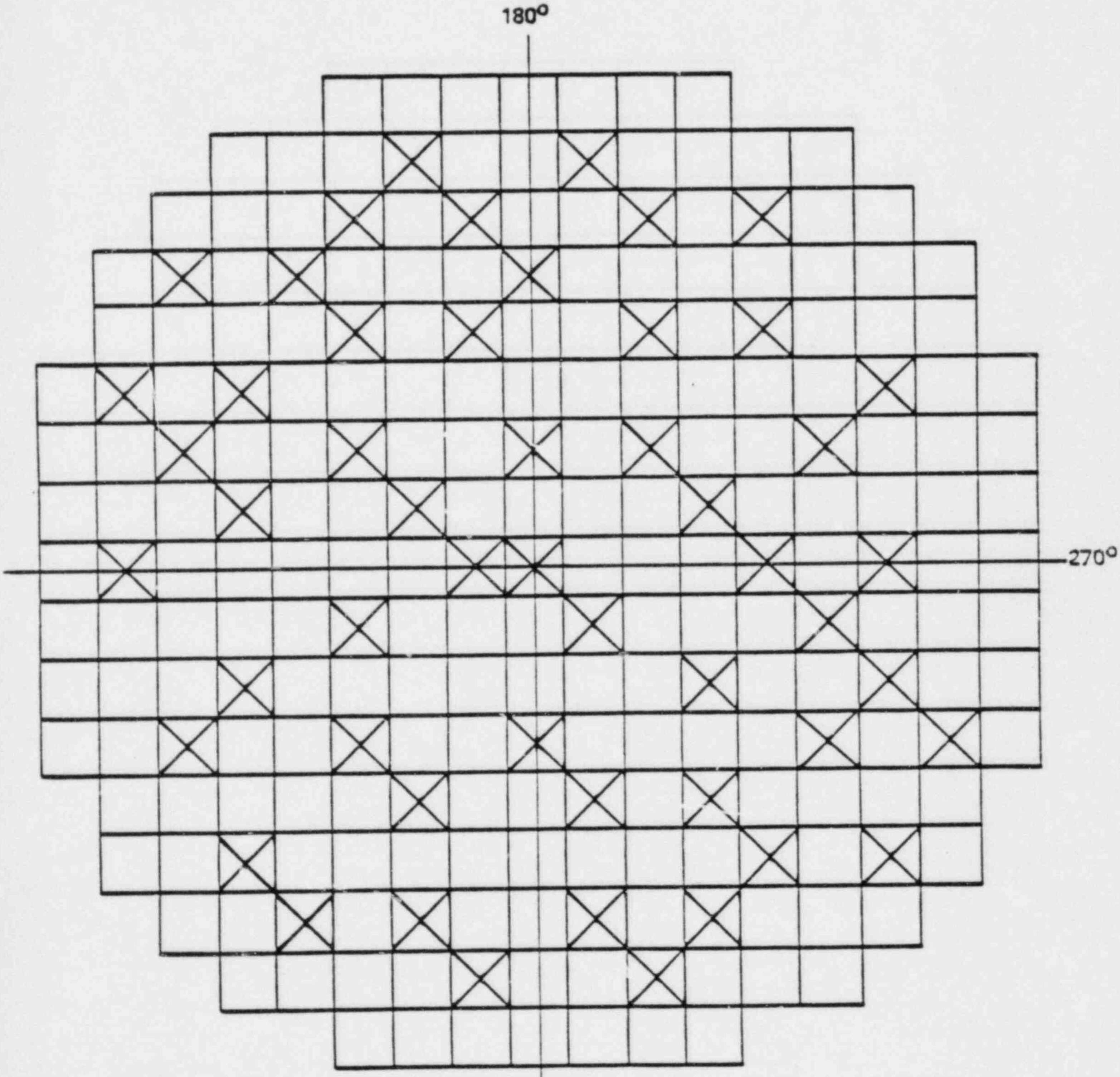
1. Process core exit thermocouple inputs for display.

2. Calculate a representative core exit temperature.

3. Provide an alarm output when temperature reaches a preselected value.

FIGURE 3-2

CORE LOCATION OF ICI DETECTOR ASSEMBLIES



4. Process CETs for display of CET temperature and superheat.

The display equipment of the CFMS will (at a minimum) be capable of trending:

1. A spatially oriented core map indicating the temperature at each of the CETs.
2. A selective reading of CET temperatures.
3. The representative core exit temperature.

The following additional information will be displayed on the QSPDS display system:

1. Representative core exit temperature.
2. A selective reading of the CET temperatures.
3. A listing of all core exit temperatures.

The purpose of the CETs during natural circulation cooldown is to determine the temperature of the primary coolant as it leaves the core. This is significant when the void is being collapsed, because the CETs will supply the operator with the temperature of the primary coolant that is filling the RV upper head.

3.4 Heated Junction Thermocouple Probe Assembly

The HJTC Probe Assembly measures reactor coolant liquid inventory above the fuel alignment plate using discrete HJTC sensors located at different levels within a separator tube ranging from the top of the fuel alignment plate to the reactor vessel head. The basic principle of operation is the detection of a temperature difference between adjacent heated and unheated thermocouples.

FIGURE 3-3

HJTC SENSOR-HJTC/SPLASH SHIELD

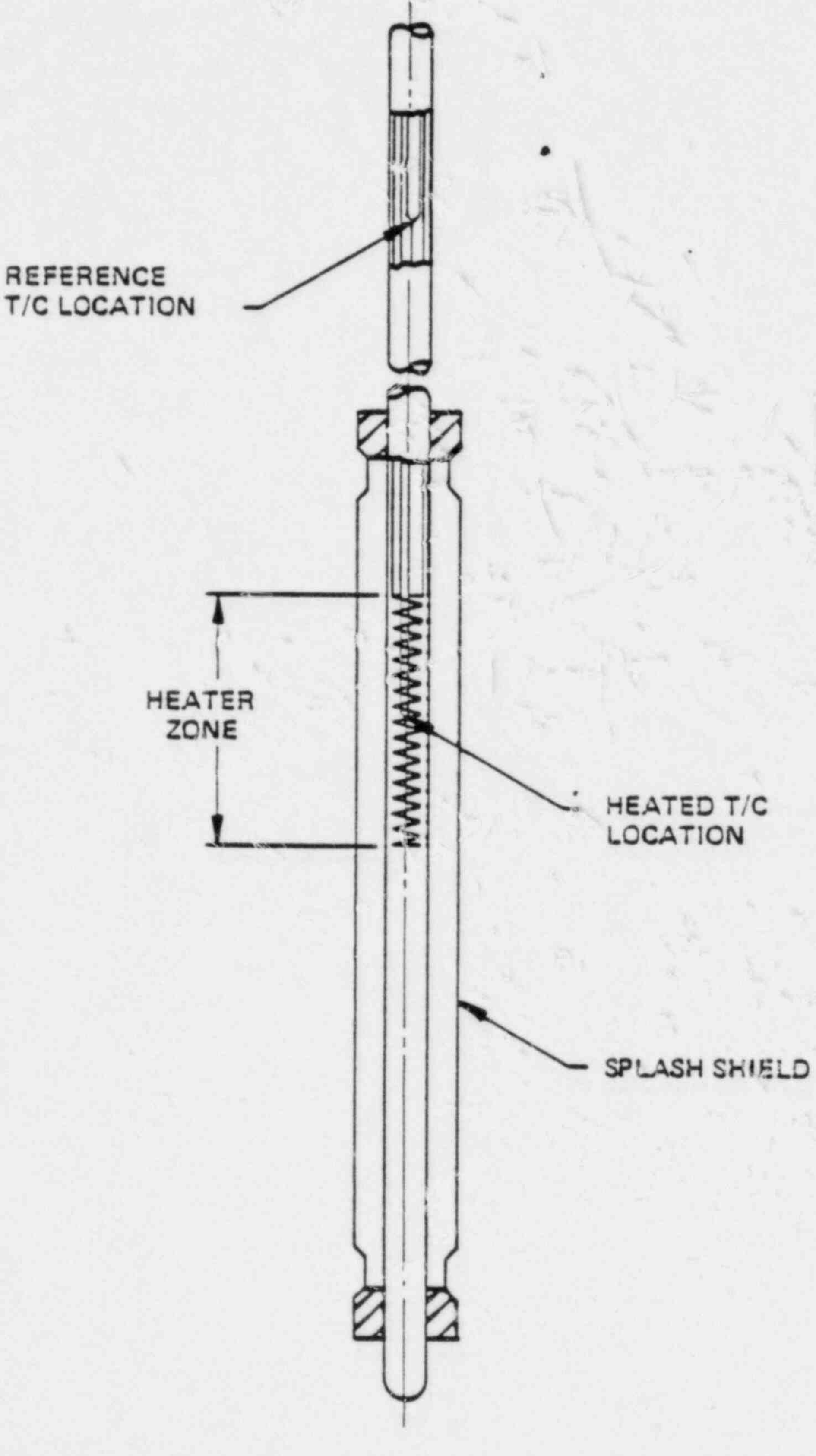


FIGURE 3-4
HEATED JUNCTION THERMOCOUPLE PROBE ASSEMBLY

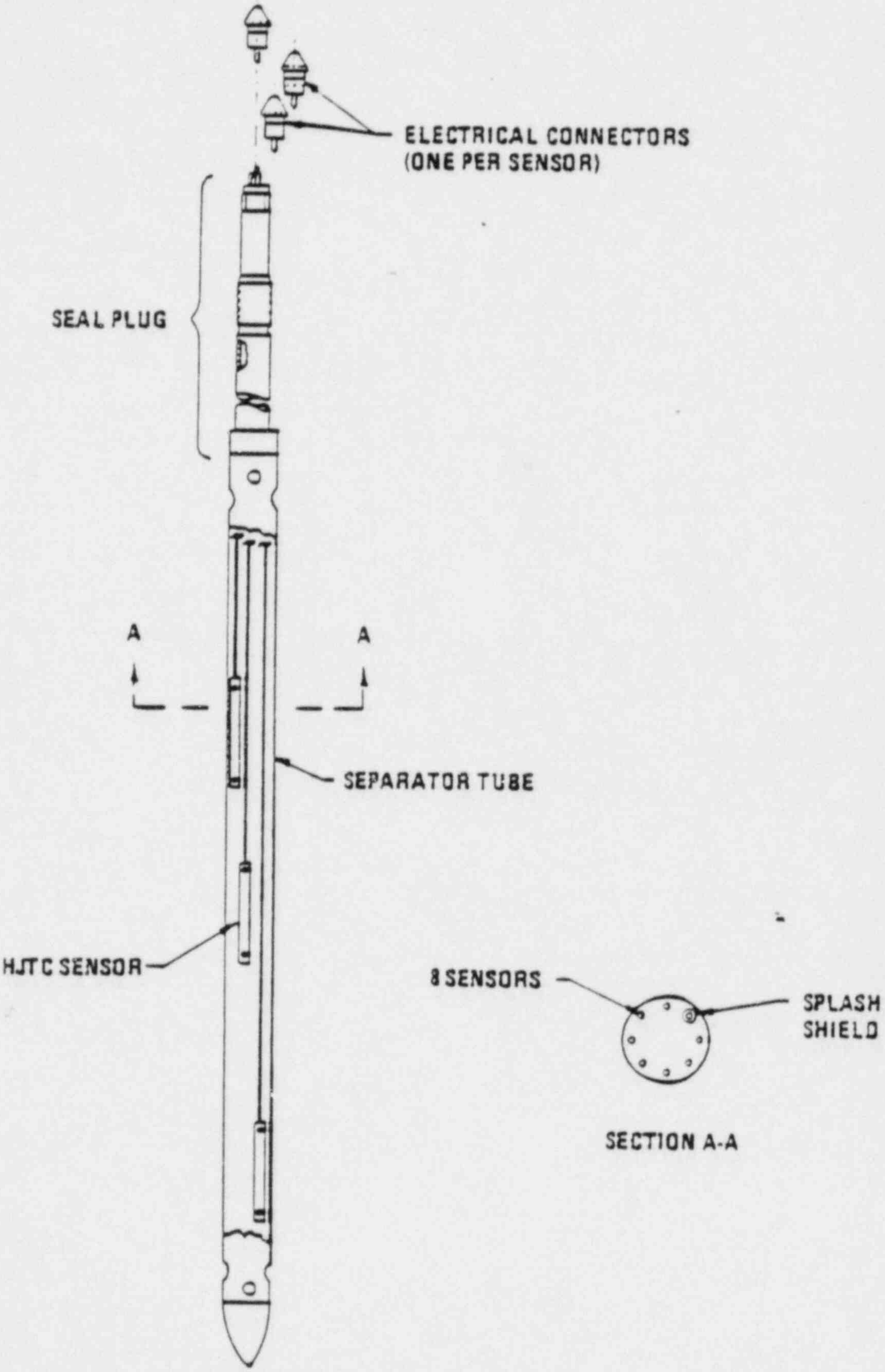


FIGURE 3-5

HJTC SENSOR AND SEPARATOR TUBE

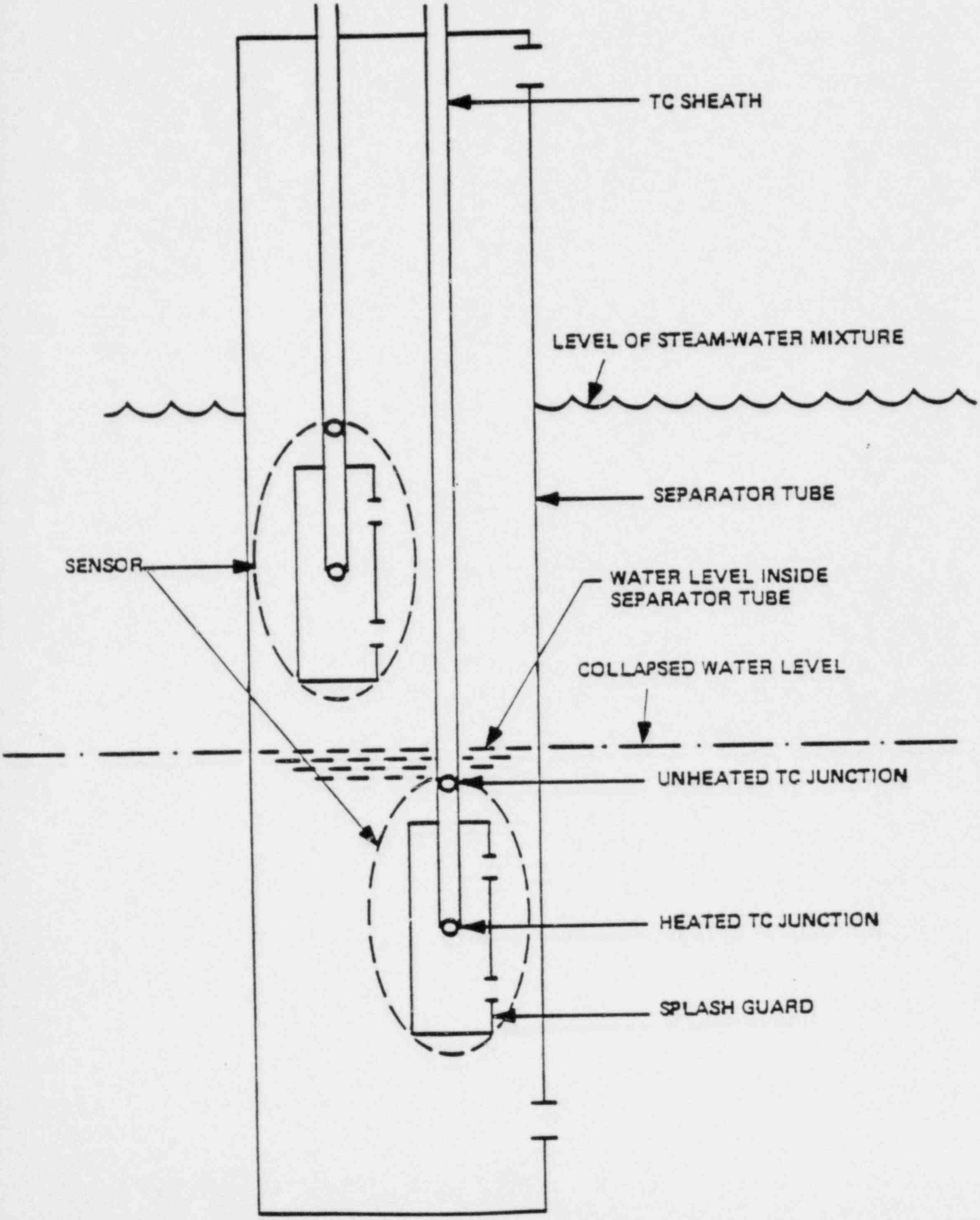
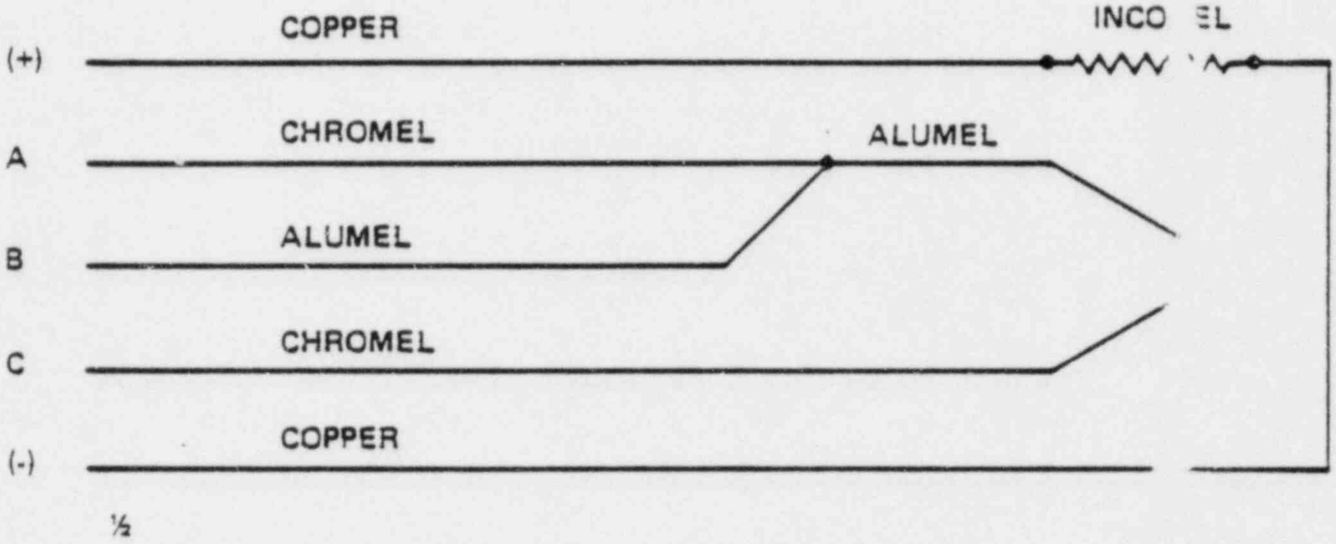


FIGURE 3-6

ELECTRICAL DIAGRAM OF HJTC



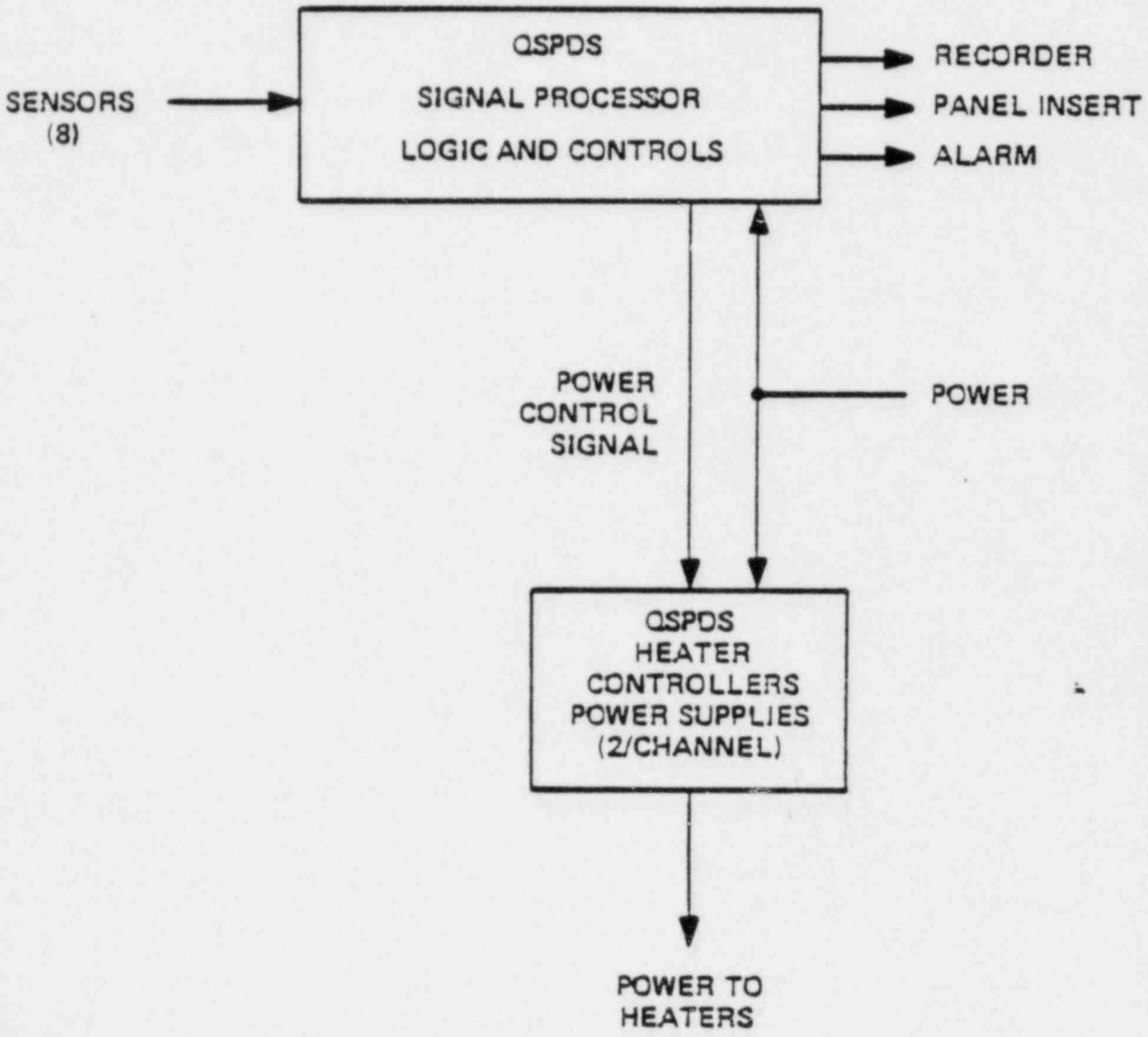
V (A - B) = ABSOLUTE TEMPERATURE, UNHEATED JUNCTION

V (C - B) = ABSOLUTE TEMPERATURE, HEATED JUNCTION

V (A - C) = DIFFERENTIAL TEMPERATURE

FIGURE 3-7

HJTC SYSTEM PROCESSING CONFIGURATION
(ONE CHANNEL SHOWN)



As pictured in Figure 3-3, the HJTC sensor consists of a Chromel-Alumel thermocouple near a heater (or heated junction) and another Chromel- Alumel thermocouple positioned away from the heater (or unheated junction). In a fluid with relatively good heat transfer properties, the temperature difference between the adjacent thermo- couples is small. In a fluid with relatively poor heat transfer properties, the temperature difference between the thermocouples is large.

Two probe assemblies are provided to allow two channels of HJTC instruments. Each HJTC probe assembly includes eight HJTC sensors, a separator tube, a seal plug, and electrical connectors (Figure 3-4). The eight HJTC sensors are electrically independent.

Two design features ensure proper operation under all thermal-hydraulic conditions. First, each HJTC is shielded to avoid overcooling due to direct water contact during two phase fluid conditions. The HJTC with the splash shield is referred to as the HJTC sensor (Figure 3-3). Second, a string of HJTC sensors is enclosed in a tube that separates the liquid and gas phases that surround it.

The separator tube (see Figure 3-5) creates a collapsed liquid level that the HJTC sensors measure. This collapsed liquid level is directly related to the average liquid fraction of the fluid in the reactor head volume above the fuel alignment plate. This mode of direct in-vessel sensing reduces spurious effects due to pressure, fluid properties, and non-homogeneities of the fluid medium. The string of HJTC sensors and the separator tube is referred to as the probe assembly. The probe assembly is housed in a stainless steel structure that protects it from flow loads.

Using the sensor measurements gathered from the HJTCs, the QSPDS processing equipment will perform the following functions:

1. Determine collapsed liquid level above core.

   The heated and unheated thermocouples in the HJTC are connected in such a way that absolute and differential temperature signals are available. This is shown in Figure 3-6. When liquid water surrounds the thermocouples, their temperature and voltage output are approximately equal. The voltage $V_{(A-C)}$, shown in Figure 3.6 is, therefore, approximately zero. In the absence of liquid, the thermocouple temperatures and output voltages become unequal, causing $V_{(A-C)}$ to rise. When $V_{(A-C)}$ of the individual HJTC rises above a predetermined setpoint, liquid inventory does not exist at this HJTC position.

2. Determine the maximum upper plenum/head fluid temperature of the top three unheated thermocouples for use as an output to the SMM calculation. (The temperature processing range is from 100°F to 2300°F.)

3. Process input signals to display collapsed liquid level and unheated junction thermocouple temperatures.

4. Provide an alarm output when any of the HJTC detects the absence of liquid level.

5. Provide control of heater power for proper HJTC output signal level. Figure 3-7 shows the design for one of the two channels which includes the heater controller power supplies.

The following information is displayed on the CFMS and QSPDS displays:

1. Percent liquid inventory level in the RV plenum between the fuel alignment plate and the Upper Guide Structure Support Plate derived from discrete HJTC positions.

2. Percent liquid inventory level in the RVUH region above the Upper Guide Structure Support Plate derived from discrete HJTC positions.

3. Eight discrete HJTC positions indicating liquid inventory above the fuel alignment plate.

4. Inputs from the HJTCS:

   a. Unheated junction temperature at the eight positions.
   b. Heated junction temperature at the eight positions.

   c. Differential junction temperature at the eight positions.

The data supplied by the HJTCs will be used to help determine the size of the void created in the RVUH region and indicate the level of the primary coolant in the RV.

3.5 Other RCS Instrumentation

A brief description of other vital instruments available to the operator during a natural circulation cooldown scenario is given below.

1. Pressurizer Pressure Indication

   The pressurizer pressure indicator has a maximum range of 15-3000 psia and provides input to the Plant Protection System (PPS). The data provided will be used to monitor the primary system pressure throughout the natural circulation cooldown.

2. Pressurizer Level Indication

   The pressurizer level indicator has two channels (hot and cold). Each channel has a maximum range of 0-100% (0-428 inches) and both provide level indication to the control room. The pressurizer level indication is used as another source of indication that a void is being formed. This is due to the fact that the water displaced by the void in the RV upper head region will be forced into the pressurizer, increasing its level. The level indicator also indicates when the void is collapsing.

3. Hot and Cold Leg Temperature Indicators

The RTDs measure the temperature of the primary coolant in the hot leg and the cold leg of the Reactor Coolant System (RCS). For each cold leg and hot leg a narrow and wide RTD temperature range exist for control room indication as follows:

RTD	Range °F
Hot Leg	375-675
Cold Leg	465-615
Hot Leg	50-750
Cold Leg	50-750

3.6 Conclusion

The instrumentation described in this section clearly shows that the operator will have sufficient information available to conduct and monitor a natural circulation cooldown event with the controlled formation of a RVUH steam void. Table 3-1, which follows, summarizes the operator use of this instrumentation for each procedural step encountered during a typical natural circulation cooldown scenario. Note that other relevant instruments and controls which the operator should refer to during this event are also included in the Table.

TABLE 3-1

Summary of Natural Circulation Cooldown: Procedural Steps and Relevant Instrumentation

PROCEDURAL STEP	INSTRUMENTATION
1. Ensure adequate natural circulation by dumping steam and starting auxiliary feed flow. Verify loop flow by observing the following indications (approximately 10 minutes after tripping the Reactor Coolant Pumps): a. Loop ΔT ($T_H - T_C$) less than normal full power ΔT. b. T_C constant or decreasing. c. T_H constant or decreasing. d. No abnormal differences between T_H RTD's and core exit thermocouples.	RCS Temperature (T_{cold}) RCS Temperature (T_{hot}) AFW Valve and Pump Controls Auxiliary Feedwater Flow CETs
2. Operate atmospheric dump valves or turbine bypass valves to dump steam from both steam generators, if available, or one steam generator if only one is available, to maintain steam generator pressure at approximately 1170 psia (T_{SAT} = 564°F) until the cooldown commences.	Steam Generator Pressure Atmospheric Dump Valve Control RCS Temperature (T_{cold})

TABLE 3-1 (Continued)

Summary of Natural Circulation Cooldown: Procedural Steps and Relevant Instrumentation

PROCEDURAL STEP	INSTRUMENTATION
3. Start or check running one of the Auxiliary Feed Pumps and restore (if necessary) and maintain steam generator level in the operating steam generator(s) in the range of the level instrumentation to keep the tube bundle covered. Operate only one Auxiliary Feed Pump.	Steam Generator Level Auxiliary Feedwater Flow AFW Valve and Pump Indication Controls
4. Establish and maintain RCS hot leg temperature at least 20°F below the saturation temperature corresponding to RCS pressure by: a. Operating pressurizer heaters and spray (main or auxiliary spray) to increase or maintain RCS pressure, and/or b. Reducing RCS loop temperatures by dumping steam.	Pressurizer Pressure Pressurizer Temperature RCS Temperature (T_{hot}) Subcooled Margin Monitor Backup Pressurizer Heaters Atmospheric Dump Valve Controls Auxiliary Spray and Charging Isolation Valve Controls
5. Align the CVCS for boration of the RCS by lining up the available Charging Pumps to take suction from the Refueling Water Storage Tank via the Boric Acid Makeup Pumps (if available) or the gravity feed line.	Charging and Boric Acid Makeup Pump Indication Controls Refueling Water Storage Tank Level

TABLE 3-1 (Continued)
Summary of Natural Circulation Cooldown:
Procedural Steps and Relevant Instrumentation

PROCEDURAL STEP	INSTRUMENTATION
6. With letdown borate the RCS to the cold shutdown boron concentration. If letdown is <u>not</u> available, start the Charging Pumps and establish a pressurizer level of 1400 ft^3 (80% Indicated Level).	Pressurizer Level Charging and Boric Acid Makeup Pump Indication and Controls
7. Operate auxiliary feedwater control valves to maintain level in the operating steam generator(s) in the indicating range and above the top of the tube bundle.	AFW Valve and Pump Controls AFW Flow Steam Generator Level RCS Temperature Pressurizer Level Condensate Storage Tank Level
8. Commence RCS cooldown by dumping steam from the operable steam generator(s) (preferably both) through the turbine bypass system (if condenser vacuum is being maintained) or the atmospheric steam dumps when: a. An indicated pressurizer level of 80% has been attained with letdown <u>not</u> available, <u>or</u> b. The RCS has attained the cold shutdown boron concentration.	Pressurizer Level Atmospheric Dump Valve Controls RCS Temperature (T_{cold}) Pressurizer Temperature
9. Continue to borate the RCS in accordance with Technical Specification requirements.	Charging and Boric Acid Makeup Pump Indication and Controls

TABLE 3-1 (Continued)
Summary of Natural Circulation Cooldown:
Procedural Steps and Relevant Instrumentation

PROCEDURAL STEP	INSTRUMENTATION
10. Maintain an RCS pressure above RVUH saturation pressure and within Technical Specification Limitations using pressurizer heaters and auxiliary spray.	Pressurizer Pressure Backup Pressurizer Heaters Auxiliary Spray Controls RVUH Temperature
11. Maintain charging flow to continue boration and makeup for shrinkage during the cooldown. Charging Pump suction is to be aligned to the Volume Control Tank if letdown is available or to the gravity feed path from the Refueling Water Storage Tank.	Charging and Boric Acid Makeup Pump Controls Volume Control Tank and Refueling Water Storage Tank Level Indicators
12. Charging flow may be reduced when pressurizer level is continuously increasing while cooling down at the maximum attainable rate. Do not exceed a maximum pressurizer indicated level of 80% (1400 $ft.^3$)	Charging Pump Indication and Controls Pressurizer Level
13. Maintain normal steam generator water level throughout the plant cooldown.	Steam Generator Level
14. When the RCS hot leg temperature has reached below the SDCS entry temperature of 400°F and the pressurizer level is between 35% and 50% commence a RCS depressurization using the auxiliary spray.	RCS Temperature (T_{hot}) Pressurizer Level Auxiliary Spray Controls

TABLE 3-1 (Continued)

Summary of Natural Circulation Cooldown: Procedural Steps and Relevant Instrumentation

PROCEDURAL STEP	INSTRUMENTATION
15. When RCS pressure reaches approximately 1850 psia reset the low pressurizer pressure trip setpoints in accordance with the Reactor Protective System Operating Procedure.	Pressurizer Pressure
16. When steam generator pressure reaches approximately 970 psia reset the low steam generator pressure trip setpoints in accordance with the Reactor Protective System Operating Procedure.	Steam Generator Pressure
17. When RCS pressure is reduced to 640 psia commence depressurization of the Safety Injection Tanks in accordance with the Safety Injection Tank Operating Procedure.	Pressurizer Pressure SIT Nitrogen Vent Valve Controls
18. When the RCS is 390 psia close the Safety Injection Tank isolation valves in accordance with the Safety Injection Tank Operating Procedure.	Pressurizer Pressure SIT Isolation Valve Control

TABLE 3-1 (Continued)

Summary of Natural Circulation Cooldown: Procedural Steps and Relevant Instrumentation

PROCEDURAL STEP	INSTRUMENTATION
19. During the RCS depressurization, monitor for condensible void formation. Anticipate void formation at the saturation pressure corresponding to the indicated RVUH temperature. Symptoms of void formation are: a. RVUH saturation margin of ≤ 0°F. b. Pressurizer level increases significantly greater than expected while operating auxiliary spray. c. Letdown flow unexpectedly greater then charging flow if the pressurizer level control system is in automatic. d. Void level formation as indicated by the reactor vessel level monitor.	Pressurizer Pressure RVUH Saturation Margin Pressurizer Level Letdown Flow Charging Flow RVUH Level
20. If a condensible void formation in the RVUH is indicated, continue the RCS depressurization allowing the void to expand. Stop the depressurization when: a. The pressurizer level has increased to 90% indicated level or b. The RVUH void level reaches a minimum value of 16% (or 3 ft. above the UGS support plate).	Pressurizer Level RVUH Level
21. Repressurize the RCS by energizing all available pressurizer heaters and commencing charging flow to the RCS.	Backup Pressurizer Heaters Charging Flow

TABLE 3-1 (Continued)
Summary of Natural Circulation Cooldown: Procedural Steps and Relevant Instrumentation

PROCEDURAL STEP	INSTRUMENTATION
22. Stop the charging flow and deenergize the pressurizer heaters when: a. The pressurizer level stops decreasing and begins a normal steady increase due to charging or when the pressurizer level decreases and b. The RVUH void level has collapsed as indicated by the RVUH level monitoring system.	Pressurizer Level RVUH Level
23. Repeat Steps 14, 19, 20 and 21 for several fill and drain cooling cycles of the RVUH. Monitor the RVUH temperature.	RVUH Temperature
24. If void indication cannot be eliminated by Steps 18 through 22, consider the gases to be partially or completely noncondensible gases. a. Increase pressurizer pressure above the pressure where void symptoms were originally noticed. b. Operate the RV head vent as needed to eliminate the noncondensible gases.	RV Heat Vent Controls Pressurizer Pressure RVUH Level

TABLE 3-1 (Continued)
Summary of Natural Circulation Cooldown: Procedural Steps and Relevant Instrumentation

PROCEDURAL STEP	INSTRUMENTATION
25. Resume the RCS cooldown and depressurization. If indications of a condensible void reappear repeat Steps 18 through 23.	RVUH Level RVUH Saturation Margin
26. When the RCS pressure is reduced to below 400 psia and the RCS temperature is below 400°F enter SDC.	Pressurizer Pressure RCS Temperature (T_{hot})

4.0 ASSESSMENT OF NATURAL CIRCULATION COOLDOWN SCENARIOS

4.1 General

As can be seen from the previous sections a natural circulation cooldown is a complex evolution that requires significant operator action over a lengthy period of time. It is not however, a frequently executed evolution. In fact, there have only been two full plant cooldown and depressurizations conducted at a C-E supplied NSSS. Additionally, C-E is aware of only 5 full and partial cooldowns at other PWRs. Based on this apparent infrequency and the relatively slow controlled progression of the cooldown it is felt that the operator actions described in previous sections are the most reasonable approach to cooldown under natural circulation conditions.

This section provides a study of all initiating events which can result in tripping all four RCPs and consequently put the plant into a natural circulation mode. Each event is examined relative to the need to cooldown to cold shutdown conditions. Tables 4-1 through 4-4 summarize results of a review of known operating experience.

All plant systems, components and operational guidelines surrounding the necessary operation of the RCPs and the criteria for stopping them while the plant is at power were reviewed. As a result of this review, only three event categories were determined to either manually or automatically trip the RCPs offline and establish a natural circulation condition in the RCS. The first of these events is a loss of off-site power (LOOP) where the pumps automatically trip since all electrical power to the RCPs and their associated subsystems are lost. The second of these events is a loss of all four RCPs due to a failure of component cooling water (CCW) to the RCPs. CCW to the RCPs is a vital subsystem needed to protect the integrity of the pump seals while the pumps are running. Accordingly following CCW failure, the operator must trip the pumps offline. The last event is a procedural requirement to stop all

RCPs on a SIAS due to low pressurizer pressure. The low pressurizer pressure is conjunction with the SIAS is an indication of a major loss of coolant accident (LOCA) where running the RCPs could cause more dire consequences. Therefore, the RCPs are required to be manually tripped but the associated subsystems will be maintained. In the sections below each of these events is discussed from the standpoint of the subsequent need to cooldown.

4.2 Loss of Off-Site Power

The causes of LOOP events may include weather related damage, natural phenomenon events, or system and component failures. For example, a LOOP event may result from a failure of all independent circuits between the regional grid network and the unit's switchyard or from a failure of all onsite transformers and/or buses which directly distribute off-site AC power to the plant. A complete failure of the regional grid network could also occur and result in a LOOP event.

Therefore, maintaining off-site power depends on the high reliability of design of on-site components, of transmission lines and of the regional grid network. These in turn strongly depend on the applicant and on the location of the plant itself.

In order to insure a high reliability of the off-site power system for System 80, the design, inspection, and testing of the system is to be in compliance with General Design Criteria 17 and 18 of 10CFR50, Appendix A and Regulatory Guide 1.32. This requires the redundancy and physical separation of all vital components such as transmission lines, startup transformer, and buses which supply and distribute off-site power to the plant. A typical electrical one-line diagram for a System 80 plant is shown in Figure 4-1.

If a LOOP event does occur while the plant is at power, the operator actions should be directed towards shutting down the plant and maintaining hot standby conditions using emergency on-site power supplied from the diesel generators. The plant is kept at hot standby until the electrical emergency is over. There is no inherent reason that a LOOP will require plant cooldown and depressurization. However, conceivably there may be extenuating circumstances that do require consideration of a plant cooldown. The first of these would be an event or failure independent from the loss of off-site power that requires cold shutdown conditions for repair. In most cases even this would not necessitate an immediate cooldown and the operator could wait until power is restored (see discussions relative to power restoration below) and RCPs can be restarted.

The second reason would be approach to a Technical Specification limit on condensate storage capacity for System 80. This limit is 300,000 gallons which provides a cooldown capability as indicated in Figure 4-2. However, all plants have significant backup to the required condensate supply. The Palo Verde Nuclear Generating Station for example has identified over 1,500,000 gallons of reserve feedwater which would provide over 100 hours of decay heat removal. In this regard it is significant to note that the duration of LOOP events is historically short. The restoration time assumed for WASH 1400 for example was 0.25 hours. More recently in a report submitted to the NRC (Reference 8), Florida Power and Light evaluated their system and determined a mean restoration time of 27 minutes. An EPRI report (Reference 14) provides an estimate of restoration for plants pooled according to eight geographical groups provided by the National Electric Reliability Council (NERC) regions. Results from this report which is shown in Table 4-5 show a median recovery time of no greater than one hour and twenty-four minutes for any one of the eight geographical regions.

Hence, it would appear extremely unlikely that a LOOP event would result in a natural circulation cooldown. In fact, of the 24 LOOP events known to C-E none have resulted in a full plant cooldown and depressurization.

4.3 Loss of CCW

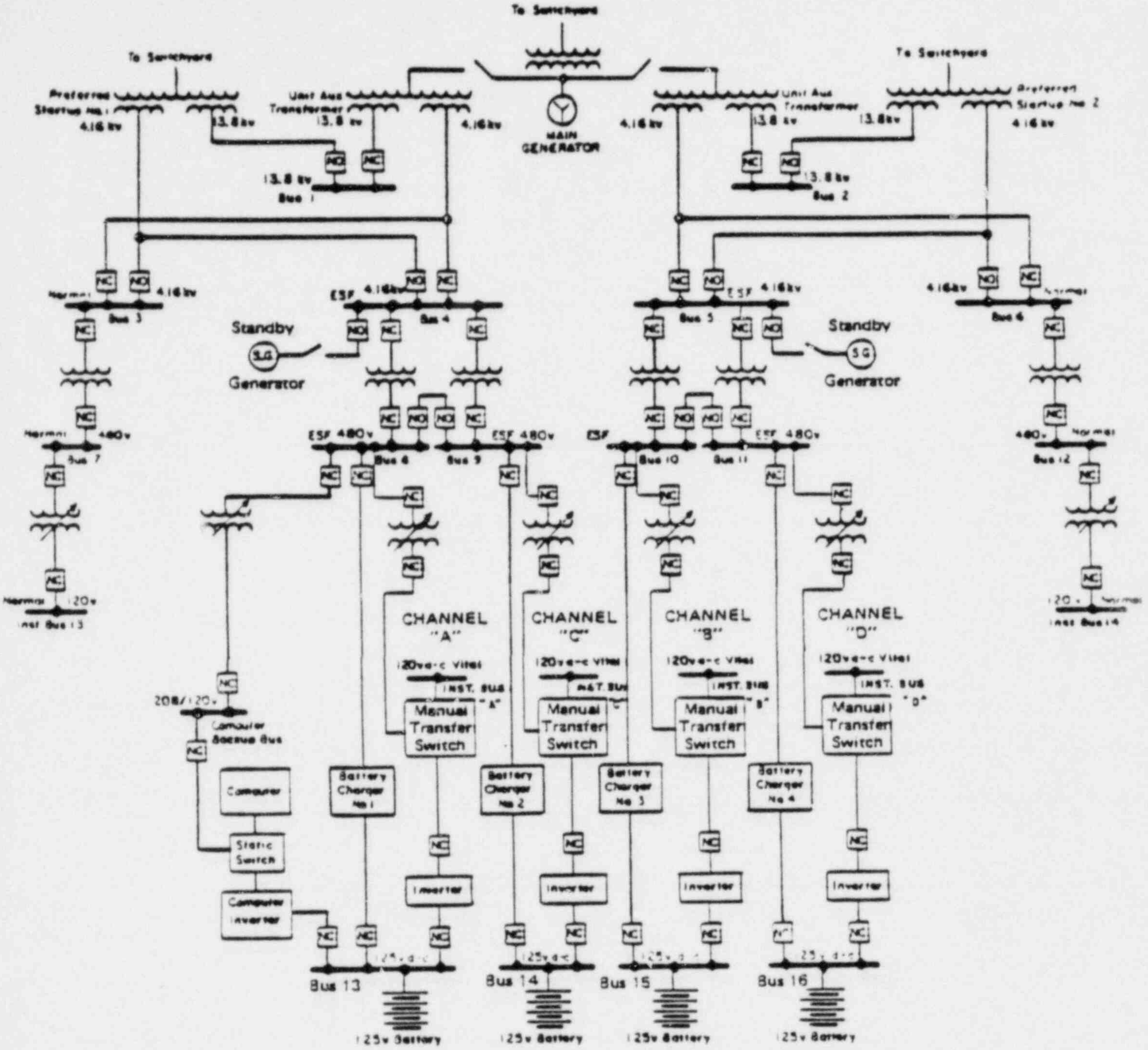
As stated earlier, there have been two natural circulation cooldown in a C-E supplied NSSS. Both of these were at St. Lucie Unit 1 and were caused by a loss of component cooling water to the RCP seals. In both cases, the CCW failure resulted in loss of cooling water to all RCPs. This then required the operator to stop the RCPs and necessitated a cooldown to inspect and (as it turned out) replace pump seals. There is however, a significant difference between the SYS 80 RCPs and the St. Lucie RCPs in that System 80 is supplied with seal injection from the charging system which will essentially eliminate any resulting seal damage from loss of CCW and relieve any necessity to cooldown and depressurize. System 80 RCP requirements relative to service systems are shown in Table 4-6.

4.4 Safety Injection Actuation Signal (SIAS)

As a result of post TMI small break LOCA analyses, operators are required by procedure to stop RCPs following an SIAS on low pressurizer pressure thus placing the plant in Natural Circulation. The operator must then determine the cause of the pressure drop. If there is not a LOCA, the RCPs can then be restarted. Since events resulting in a SIAS may depressurize the plant to the point where a void is formed in the RVUH, (e.g. Ginna) there is little operator action that can be done in these cases to prevent void formation. The response of the C-E plant to these type events when the RVUH voiding has been considered and reviewed and is documented in Reference 7.

FIGURE 4-1

TYPICAL ELECTRICAL ONE-LINE DIAGRAM FOR SYSTEM 80



BUS	EQUIPMENT
1	REACTOR COOLANT PUMP (2)
2	REACTOR COOLANT PUMP (2)
4	LOW PRESSURE SAFETY INJECTION PUMP HIGH PRESSURE SAFETY INJECTION PUMP
5	LOW PRESSURE SAFETY INJECTION PUMP HIGH PRESSURE SAFETY INJECTION PUMP
7	PRESSURIZER HEATERS (BACKUP AND PROPORTIONAL) MOTOR CONTROL CENTER CEDM MG SET
8	BORIC ACID PUMP BATTERY CHARGER CHARGING PUMP
9	PRESSURIZER HEATERS (BACK UP: 1 BANK, 150 KW) BATTERY CHARGER MOTOR CONTROL CENTER CHARGING PUMP

BUS	EQUIPMENT
10	PRESSURIZER HEATERS (BACK UP: 1 BANK, 150 KW) BATTERY CHARGER MOTOR CONTROL CENTER CHARGING PUMP
11	BORIC ACID PUMP BATTERY CHARGER
12	PRESSURIZER HEATERS (BACKUP AND PROPORTIONAL) MOTOR CONTROL CENTER CEDM MG SETS
13	RTSG CONTROL CIRCUIT VALVE CONTROL CIRCUITS MOTOR CONTROL CIRCUITS
14	RTSG CONTROL CIRCUIT VALVE CONTROL CIRCUITS MOTOR CONTROL CIRCUITS
15	RTSG CONTROL CIRCUIT VALVE CONTROL CIRCUITS MOTOR CONTROL CIRCUITS
16	RTSG CONTROL CIRCUIT VALVE CONTROL CIRCUITS MOTOR CONTROL CIRCUITS

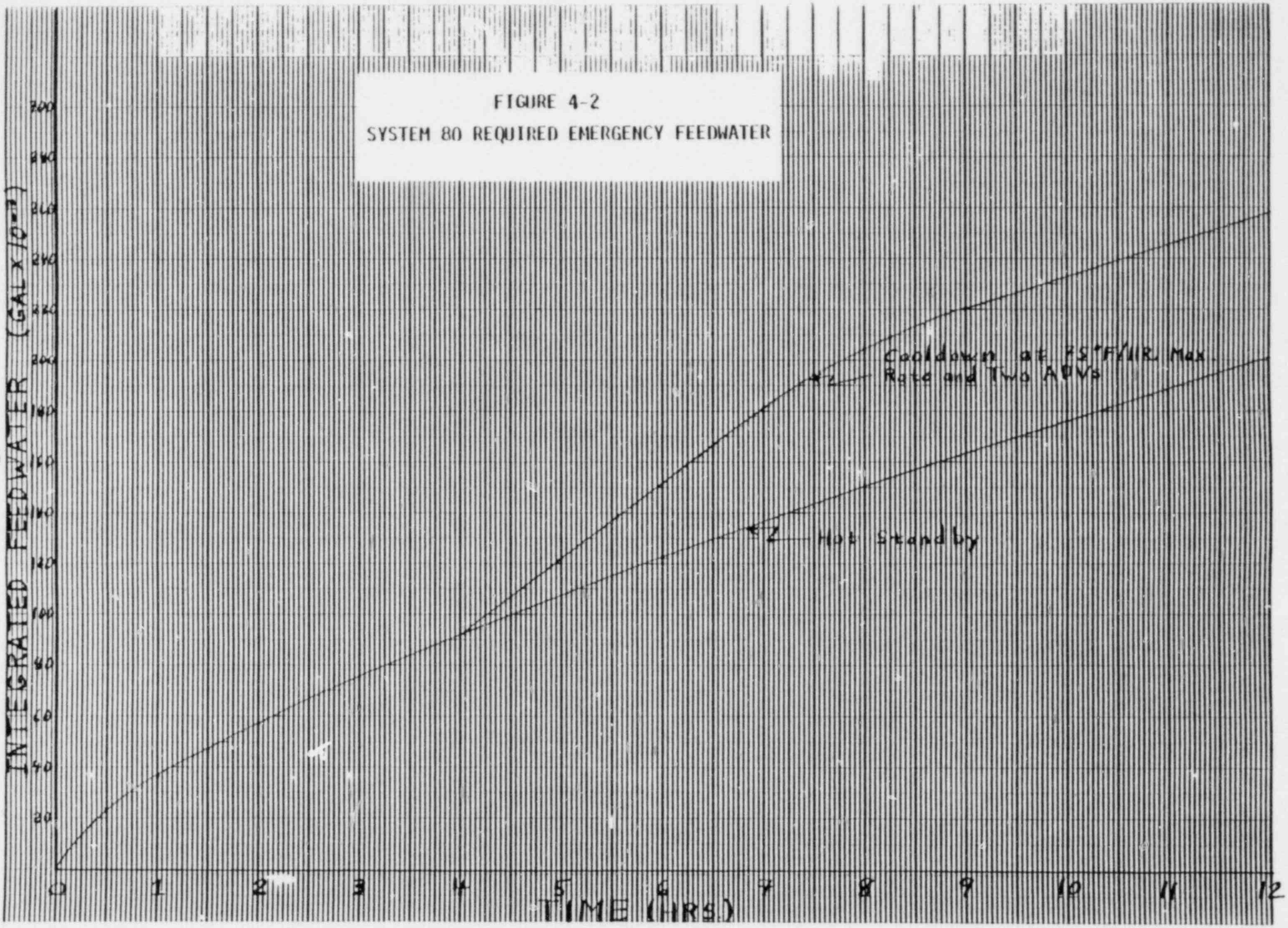


FIGURE 4-2

SYSTEM 80 REQUIRED EMERGENCY FEEDWATER

TABLE 4-1

Events Involving Natural Circulation But Without Extended Cooldown

(Data 1/68 - 12/81)

Event Cause	Number of Events	Occurrence Frequency (310 Plant Years)
Loss of Offsite Power (> 15 Minutes)	24	0.077/Plant Year
Loss of Component Cooling Water to RCP Seals	3	0.010/Plant Year
Manual Trip of RCP's Following Safety Injection	2	0.006/Plant Year

TABLE 4-2
Summary of Events Involving Natural Circulation But Without Extended Cooldown

Plant	Date	Description
Haddam Neck	4/68	Plant operating at 100% power with one of two incoming 115 KV lines out of service. Improper procedure was used to restore out-of-service line, resulting in loss of off-site power and a reactor/turbine trip. Off-site power not available for 25 minutes. Plant taken to hot standby.
Palisades	9/71	Faulty breaker failure relay led to loss of SYS KV line and loss of off-site power. Power restored in 56 minutes.
R. E. Ginna	10/73	One in-coming line was out-of-service a flashover and loss of off-site power occurred due to overload on the other lines. SIAS generated on excess cooldown. Power restored in 40 minutes.
Turkey Point 3	3/74	Grid instabilities led to loss of off-site power and station blackout. Power restored in approximately 30 minutes, unit taken to hot standby.
Turkey Point 4	3/74	Same as above.
Palisades	10/74	A faulty startup transformer differential current protection relay led to a loss of off-site power during the quarterly SIS test. Plant was in hot standby at the time. Power was restored in approximately 30 minutes.

TABLE 4-2 (Continued)
Summary of
Events Involving Natural Circulation
But Without Extended Cooldown

Plant	Date	Description
Oconee 2	1/74	A spurious signal actuated solid state breaker failure relays in the switchyard, resulting in total isolation of 230 KV switchyard. Unit tripping on loss-of-off-site power. Natural circulation cooling was established for 1 hour.
Millstone 2	8/76	Unit tripped on loss of off-site power during hurricane Belle. Unit placed in Mode 3 with natural circulation. Power restored in 24 hours.
Beaver Valley	12/76	138 KV bus differential relay tripped, interrupting power to 1A station transformer. Unit tripped on loss-of-off-site power. Unit maintained in hot standby. Power was restored in 38 minutes.
Turkey Point 3	5/77	Unit lost off-site power following a reactor trip. Natural circulation cooldown started. Power restored in 20 minutes and pumps restarted to complete cooldown.
St. Lucie 1	5/77	Following trip of Turkey Point 3, St. Lucie and a fossil unit were picking up the load when St. Lucie had a 50% load reduction. This lead to wide spread voltage fluctuation on the grid and a loss of off-site power at St. Lucie and Turkey Point. Power was restored in 20 minutes.

TABLE 4-2 (Continued)
Summary of
Events Involving Natural Circulation
But Without Extended Cooldown

Plant	Date	Description
Indian Point 3	5/77	A lightening strike lead to a loss-of-off-site power. Cooldown was started, but when off-site power was restored, pumps were restarted and unit returned to power.
Indian Point 3	7/77	A lightening strike to lead to loss-of-off-site power and a unit trip. Unit maintained in hot standby using natural circulation. Power restored in 6.28 hours.
Palisades	9/77	Bus was lost during electrical storm. Power restored in 4.76 hours. Unit maintained in hot standby.
Palisades	11/77	Off-site power was los when bus deenergized (cause unknown). Power restored in 3.5 hours. Unit maintained in hot standby.
Palisades	12/77	Offsite power was lost when bus was deenergized (cause unknown). Power restored in 1.5 hours. Unit maintained in hot standby.
Calvert Cliffs 2	4/78	Abnormal operation of various 500 KV breakers caused by the presence of an AC voltage in the DC control circuit for the breakers caused a loss of off-site power. Power restored in 5.48 hours.
Calvert Cliffs 1	4/78	As above.

TABLE 4-2 (Continued)
Summary of
Events Involving Natural Circulation
But Without Extended Cooldown

Plant	Date	Description
Beaver Valley 1	7/78	A single phase short circuit on the main transformer lead to a loss of off-site power. Natural circulation cooldown started. One pump restarted to complete cooldown when power was restored 17 minutes later.
Davis Besse 1	10/79	When reclosing the generator output breaker following a unit trip, the J bus tripped causing a station blackout. Natural circulation maintained for 1.25 hours until power restored and one RCP restarted.
Crystal River 3	2/80	Loss of non-nuclear instrument bus leads to a unit scram, a pressure transient and a safety injection. RCPs were secured per procedure. Plant maintained in hot shutdown on natural circulation for 7.5 hours.
North Anna 1	5/80	With plant in hot standby, startup of one RCP caused loss of a vital bus. This initiated SIAS, and isolated CCW from the RCP seals. The RCP's tripped on loss of CCW.
Indian Point 2	6/80	A lightening strike on a transmission line caused loss of off-site power and a unit trip. Natural circulation established and maintained for 2.5 hours until power was restored and one pump was restarted.

TABLE 4-2 (Continued)

Summary of
Events Involving Natural Circulation
But Without Extended Cooldown

Plant	Date	Description
Arkansas Nuclear 1 Unit 1	6/80	A transmission line failure caused a partial loss of off-site power and a unit trip. Unit 1 was on natural circulation for 100 minutes in hot standby conditions.
Arkansas Nuclear 1 Unit 2	1/80	As above except natural circulation was maintained for 66 minutes. Unit then taken to cold shutdown on forced circulation.

TABLE 4-3

Summary of Extended Natural Circulation Cooldown Causes and Frequency

Cause	# of Occurrences 1968-Present	Frequency of Occurrences	# of Occurrences Involving RCS Voiding	RCS Voiding Frequency
Loss of Off-Site Power	1	6.5×10^{-3}/yr.	-	-
CCW System Malfunction	3	9.7×10^{-3}/yr.	2	6.5×10^{-3}/yr.
Manual RCP Trip on SI IAW IOE Bulletin 79-06C	3	9.7×10^{-3}/yr.	1	3.2×10^{-3}/yr.
Totals	7	2.3×10^{-2}/yr.	3	9.7×10^{-3}/yr.

TABLE 4-4
Summary of Extended Natural Circulation Cooldown Events at PWRs

St. Lucie-1 (802 MWe CE) (LER-11-26)	4/16/77	Reactor trip with subsequent loss of off-site power during grid transient caused by loss of turkey point units. Natural circulation cooldown to cold RHR conditions completed. RCP's not restarted when power was recover 1½ hours into transient, apparently to limit grid loading. Subsequent reviews of cooldown indicated void formation occurred.
St. Lucie-1 (802 MWe CE) (LER-80-30)	6/18/80	Steam leak shorting of a solenoid operated containment isolation valve caused loss of CCW to all RCP's. Natural circulation cooldown to cold RHR conditions performed. RCS voiding in head region occurred.
Prairie Island-1 (530 MWE W)		Steam generator tube rupture with subsequent safety injection. Operators secured RCP per NRC requirements. Natural circulation cooldown to cold RHR conditions performed. No voiding observed.

TABLE 4-4 (Continued)
Summary of Extended Natural Circulation Cooldown Events at PWRs

R. E. Ginna (490 MWe W) (SER-81-20/25)	1/25/82	Same as Prairie Island event.
H. B. Robinson (700 MWe W) (NPE-Y-A-40)	5/1/75	Catastrophic RCP seal failure caused significant oustage of RCS liquid in CCW System. CCW to all RCP's lost. Natural circulation to cold RHR conditions eventually performed. RCS voiding indicated, possibly in steam generators.
Salem-1 (1090 MWe W) (NPE-V-A-90)	10/78	RCP seal failure occurred in hot standby. RCP's secured due to seal cooling problems and leakage. Natural circulation cooldown to cold RHR conditions performed without indication of voiding.
Kewaunee-1 (535 MWe W)	1/80	Fault in reserve auxiliary transformers caused reactor trip and station blackout. Natural circulation cooldown to cold RHR conditions performed without evidence of RCS voiding.

TABLE 4-5
EPRI LOOP FREQUENCY AND RECOVERY TIME ESTIMATES
(EPRI Report EPRI-NP-2301)

REGIONAL COUNCIL	LOOP FREQUENCY (EVENTS/SITE YEAR)	MEDIAN RECOVERY TIME (HRS: MIN)
NPCC	.153	:19
MAAC	.061	1:24*
ECAR	.338	1:11
SERC	.046	1:24*
MAIN	.076	1:23
MARCA	.204	:29
SPP	.149	----**
WSCC	.090	:06

*MAAC and SERC aggregate estimate

**No recovery time data for SPP sites.

TABLE 4-6

SYSTEM 80 - REACTOR COOLANT PUMP OPERATING/LIMITS

PUMPS RUNNING:

	INCIDENT	OPERATING LIMITS	EFFECTS	
			PUMP SEALS	PUMP BEARINGS
1.	Loss of Component Cooling Water (CCW)	10 minutes max. (Bearings limit)	No damage, No inspection	No damage, No inspection
	Seal Injection Water (SIW) Available	30 minutes (Bearings limit) (CENPD-201A)	No damage No inspection, SIW protects seals	No effect on pump coastdown, bearing inspection recommended
2.	Loss of Seal Injection Water (SIW) CCW Available	No limit, restore SIW ASAP	No damage, No inspection, CCW protects seals	No damage, No inspection, CCW protects bearings
3.	Simultaneous Loss of CCW and SIW	3 minutes max. (seals limit)	No damage, No inspection	No damage, No inspection
	(Not a credible incident - SIW system on emergency power also seismic system)			
4.	Loss of AC Power Pumps on hot standby for 2 hours (No CCW)	Restore SIW within 20 minutes. (NUREG 0737)	No loss of function SIW protects seals	No effect on pump coastdown

5.0 REFERENCES

1. Combustion Engineering Standard Safety Analysis Report, Final Safety Analysis Report.
2. NRC Summary of Meeting with FP&L and C-E regarding St. Lucie Unit 1 Cooldown on Natural Circulation, Chris C. Nelson, June 25, 1980.
3. C-E Availability Data Program Info Bulletin 80-003A, June 20, 1980.
4. NRC Generic Letter 81-21, Natural Circulation Cooldown, D. G. Eisenhut, May 5, 1981.
5. C-E-NPSD-154, Natural Circulation Cooldown Task 430 Final Report, October 1981.
6. NRC Summary of February 10, 1982 Meeting Regarding Natural Circulation Cooldown, C. I. Grimes, February 18, 1982.
7. CEN 199, Effects of Vessel Heat Voiding During Transients and Accidents in C-E NSSS's, March 1982.
8. Florida Power and Light Letter L-82-203, Loss of AC Power, May 14, 1982.
9. CEN-117, Inadequate Core Cooling - A Response to NRC IE Bulletin 79-06C, Item 5 for Combustion Engineering Nuclear Steam Supply Systems, October 1979.
10. CEN-125, Input for Response to NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply Systems, December 1979.

11. CEN-181-P, Generic Responses to NRC Questions on the C-E Inadequate Core Cooling Instrumentation, September 1981.

12. CEN-185, Documentation of Inadequate Core Cooling Instrumentation for Combustion Engineering Nuclear Steam Supply Systems, September 1981.

13. CEN-128, Response of Combustion Engineering Nuclear Steam Supply System to Transients and Accidents, April 1980.

14. WASH 1400, (NUREG-75/014) Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, 1975.

15. EPRI-NP-2301, Loss of Off-Site Power at Nuclear Power Plants: Data and Analysis, March 1982.

APPENDIX A

LTC Analysis
of
Natural Circulation Cooldown

APPENDIX A

INTRODUCTION

The Long Term Cooling (LTC) computer code was used to model a System 80 plant natural circulation cooldown from hot standby to shutdown cooling entry conditions. The purpose of this simulation was to provide an estimate of the time necessary to conduct an expeditious natural circulation cooldown using the procedures outlined in this report.

LTC is a Combustion Engineering best estimate computer code; the code description has been submitted to the NRC via CEN-128 (Reference 13). The code has been used successfully to model similar transients in C-E plants. The Reactor Vessel Upper Head (RVUH) Model in LTC is essentially a generalized non-equilibrium pressurizer model with two "surge lines" to allow a simultaneous representation of flows into and out of the region. The hydraulic data used for this region was best estimate data which was lumped to allow the two flow paths to model many flow paths. However, the alignment key leakage flow (see Figure A-1), which is a cold leg flow directly into the upper part of the region, is not modelled. This is conservative for purposes of calculating RVUH cooldown times. Also, the RVUH metal is indirectly in contact with the relatively cold RCS water over many hours of the transient; this process is not modelled in LTC (i.e., node to node metal heat transfer) and it too represents a conservatism in the results obtained.

Results

The results of the LTC analysis are summarized in the attached plots. The assumptions used in the analysis are listed in Table A-1. The scenario investigated is as follows:

Time Frame	Times (hrs.)	Operator Actions	Notes
I	0 to 0.5	None	RCP tripped to start transient; natural circulation

Time Frame	Times (hrs.)	Operator Actions	Notes
I (Cont'd.)			established; Steam Generators steam through secondary safety valves; RCS temperatures stabilize.
II	0.5 to 1.1	Boration of RCS charging flow	Pressurizer filled to 80% level to provide inventory for RCS shrinkage in Time Frame III.
III	1.1 to 5.5	50°F/hr. controlled RCS cooldown	Pressure control via heaters, RCS cooled to below SDC emergency entry temperature.
IV	5.5 to 10.0	Controlled drain and fill of RVUH to cool RVUH to below 444°F (T_{sat} at SDC emergency entry pressure of 400 psia).	1) Drain of RVUH induced by auxiliary spray, halted at pressurizer level indication of 80%. 2) Pressure control regained by heaters. 3) Upon regaining pressure control, RVUH filled by increasing pressure in pressurizer. 4) Successive drain/fill cycles (i.e., 1) through 3) above) flush cool RCS water into RVUH region, cooling it below 444°F.

Details of the analysis for each time frame are discussed below:

Time Frame I (0 to 0.5 hours)

During this time frame it is assumed that there is no operator action. Early in the first half-hour, the RCPs trip and coast down. Within the first 30 minutes of the transient the plant achieves a stable natural circulation state. During the coastdown, the loop average temperature is dropping, so that pressurizer level and pressure also drop. (Note: The coastdown is barely discernable on the 0 to 10 hour plot). Following coastdown, the RCPs stop turning, which significantly lowers the core flow rate and, correspondingly, raises the core exit temperature. The temperature rises until the resulting density differences in the RCS increase the flowrate and stabilize the increasing core exit temperature. Upon stabilization, there is a gradual decrease in the core exit/hot leg temperature as decay heat continues to decrease. As decay heat "stabilizes" at about 1800 seconds, the hot leg temperature also stabilizes. The cold leg temperature during this time is essentially that associated with the SG secondary due to the long residence times of the natural circulation flow. This behavior is characteristic of C-E plants during natural circulation, and has been observed (e.g., St. Lucie 1 Natural Circulation cooldown of June 1980).

During this time, there is no voiding in the RCS or in the RVUH region. The pressurizer level and pressure peak 8 minutes into the transient, which as previously described is also the time when the hot leg temperature reaches its peak.

Time Frame II: 0.5 hours to 1.1 hours

During this time frame the operator leaves the plant in the stable natural circulation state achieved in the earlier time frame and borates the RCS prior to starting a controlled cooldown (Time Frame III). The effect of boration on RCS inventory is to fill the pressurizer. This is easily seen in Plot A-5 (pressurizer level). The effect of the charging on RCS temperatures is also seen in Plot A-10, where the second (lower) cold leg temperature represents the single cold leg associated with the charging flow. During this time, RCS pressure rises due to the pressurizer level increase, and auxiliary spray is used as required to control the pressure. The corresponding decreases (to zero) in charging flow represent the diversion of charging flow to auxiliary spray. Steam generator pressure is still controlled via the secondary safety valves and there is no RCS or RVUH voiding during this time.

Time Frame III: 1.1 hours to 5.5 hours

During this time frame, a 50°F/hr. controlled natural circulation cooldown of the RCS is implemented by steaming both SGs through the atmospheric dump valves. Pressurizer level drops as RCS shrinkage is greater than the contribution of one charging pump (this was expected, and is one of the reasons for raising RCS inventory during Time Frame II). The second charging pump is used as required to control pressurizer level in the 30% to 35% range; this was done in order to have maximum pressurizer capacity available to absorb RVUH voiding (during Time Frame IV) as well as to keep the pressurizer heaters covered.

RCS pressure control is adequately maintained during this time frame by the pressurizer heaters. The controlled 50°F/hr. RCS cooldown has lowered all RCS temperatures to values well below the SDC entry point of 400°F.

There is no RCS voiding or RVUH voiding. The RVUH temperature has conservatively changed little, and is still at about 600°F at the end of the RCS cooldown time. Heat conduction out of the RVUH region was not modelled; its effect would be minimal on the RVUH water temperature during these times even if it had been modelled.

Time Frame IV: 5.5 hours to 10.0 hours

During this time, the RVUH is deliberately partially voided (drained) and subsequently refilled (also partially) a total of 5 times. (For simplicity, RCS cooldown was halted, as there was no need for a further cooldown.) The voiding is deliberately induced by auxiliary spray, which depressurizes the RCS and RVUH until the saturation pressure corresponding to the RVUH water temperature is reached (about 1550 psia for the first void at 600°F). Since the RCS has already been cooled, and given that the depressurization is carefully controlled, the voiding induced by the depressurization is limited to the RVUH region. The process is controlled by stopping the spray when the pressurizer level reaches 80%. After stopping the spray, and hence stopping the RVUH voiding, the code then waits for the pressurizer heaters to remove the subcooling from the subcooled insurge due to the RVUH voiding. This typically takes about 1 hour, and represents the greater part of the time spent during the drain/fill process. This analysis represents a conservatively slow process of refilling and cooling the RVUH region. The very rapid effect of refilling the RVUH by switching charging flow from auxiliary spray to the loop as seen at St. Lucie 1 and the cooling effect of venting through the Reactor Gas Vent System were not credited. Upon regaining pressure, the RVUH is filled with cooled RCS water, which then cools the water remaining in the RVUH. The fill process is stopped when the pressurizer level drops to 30% or the RVUH is filled solid; as shown, the RVUH was never completely refilled due to RCS shrinkage as pressure control is regained.

As expected, the first void increase was the most dramatic, with a void of about 700 ft^3 of steam being created. The subsequent refill collapsed the void size to about 500 ft^3. Each subsequent drain and fill averaged about 400 ft^3 and 350 ft^3 respectively, so that the maximum void size at the end of the 5 cycles was about 1000 ft^3. Upon draining, the RVUH steam and water state is fully saturated as the voiding takes place. Upon refill, the steam is superheated due to compression and the water is cooled due to the mixing with the cool RCS water. When the fill process stops and the drain process is instantly started, the two temperatures again coalasce to the previously described saturated condition. At the end of the fifth cycle, both the RVUH water and steam are well below the temperature required to allow the RCS to enter SDC without further voiding of the RVUH. Further, even if such voiding were to take place, the temperature "spikes" induced in the hot legs from the voiding are minimal (~10°F) and there is no discernable effect on the cold leg temperatures as the slow natural circulation flow continues to equilibrate with the cool SG secondary temperatures.

We conclude that the drain and fill process can adequately cool the water in the RVUH region in approximately 5 hours once the RCS has been cooled when using the pressurizer heaters to slowly control the filling process. Pressurizer level control is maintained at all times, as is RCS subcooling. Moreover, the maximum void size is about 1000 ft^3 (out of about 2000 ft^3) so that there is no possibility of directly voiding into the hot leg or of disrupting the stable natural circulation flow process.

FIGURE A-1

CORE LEAKAGE FLOWS

Note: (6) is the Alignment
Key Leakage Flow

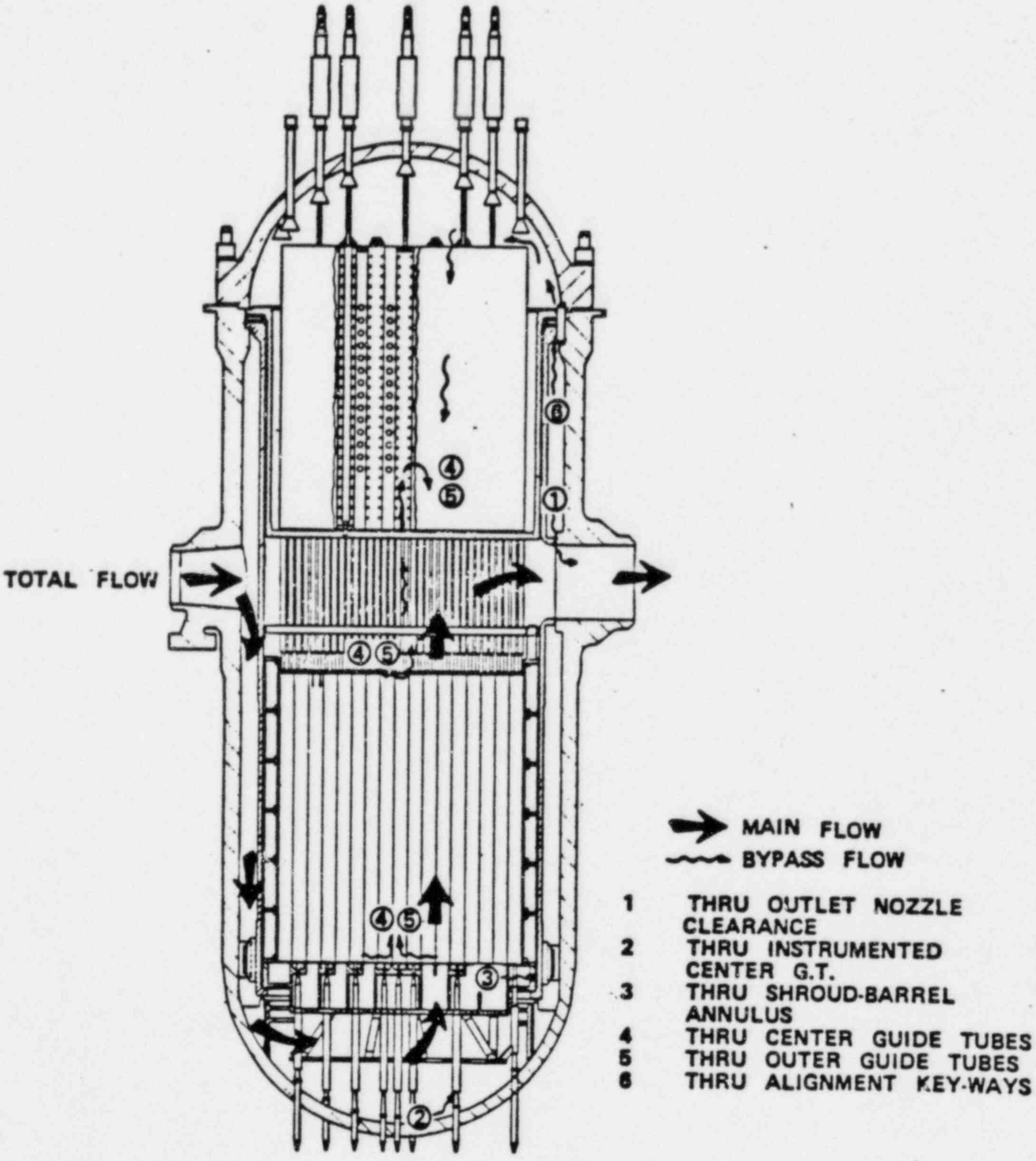


Table A-1

LTC Computer Code

Assumptions and Initial Conditions

Initial Power: 100%

Initial T_{HOT}: 620.2°F

Initial T_{COLD}: 564.35°F

Initial P_{SEC}: 1070.0 psia

Secondary Safety Valve Setpoint: 1265.0 psia

Atmospheric Dump Valve Area: 0.25 ft^2 per SG

Letdown: Not Available

Heater Capacity Assumed: 1300 KW

Charging Flow/Auxiliary Spray: Two pumps at 44 GPM each, flows at 120°F

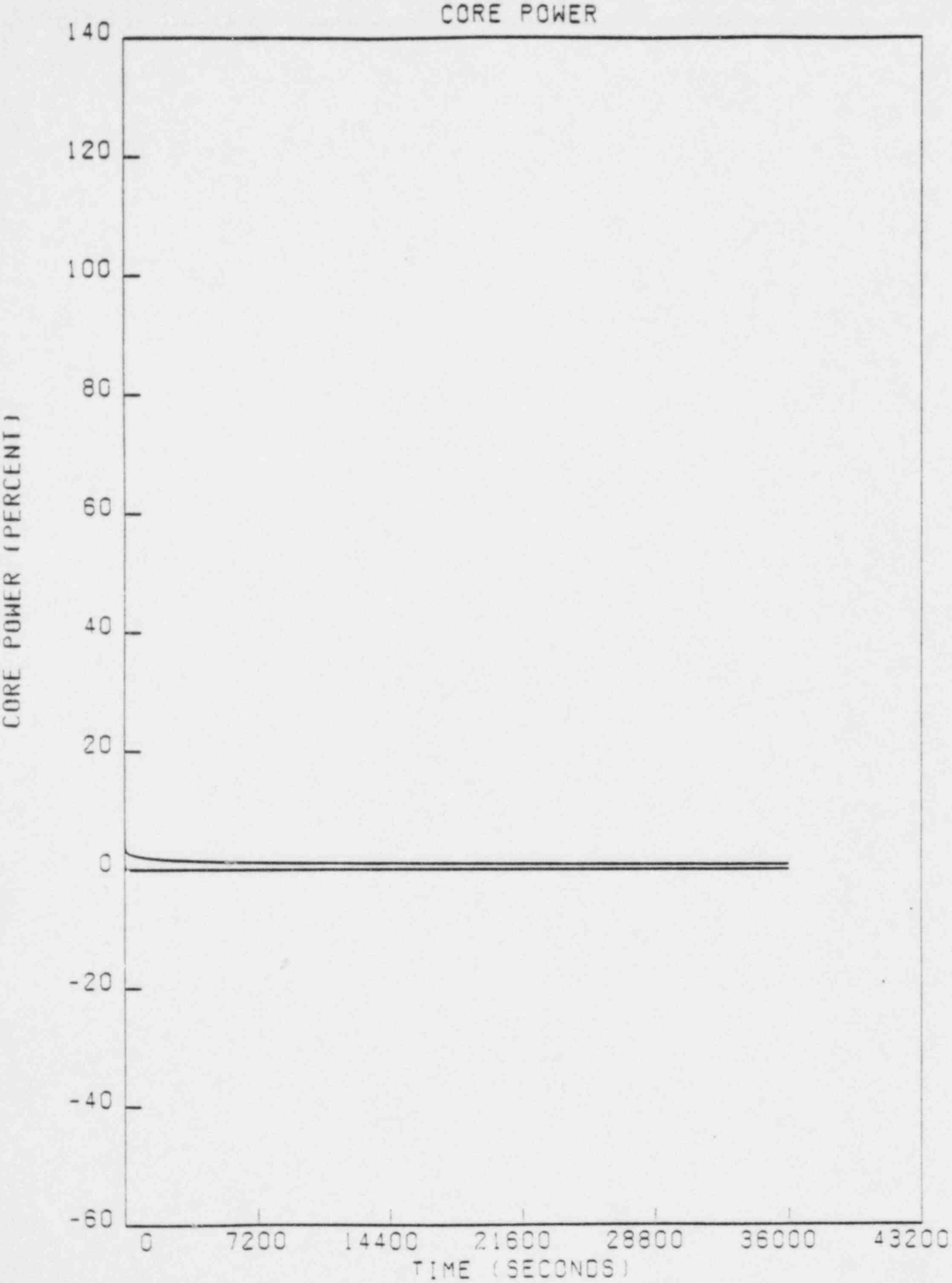
Controlled RCS Cooldown Rate: 50°F/hr.

RCPs: Tripped at t = 0.

APPENDIX A

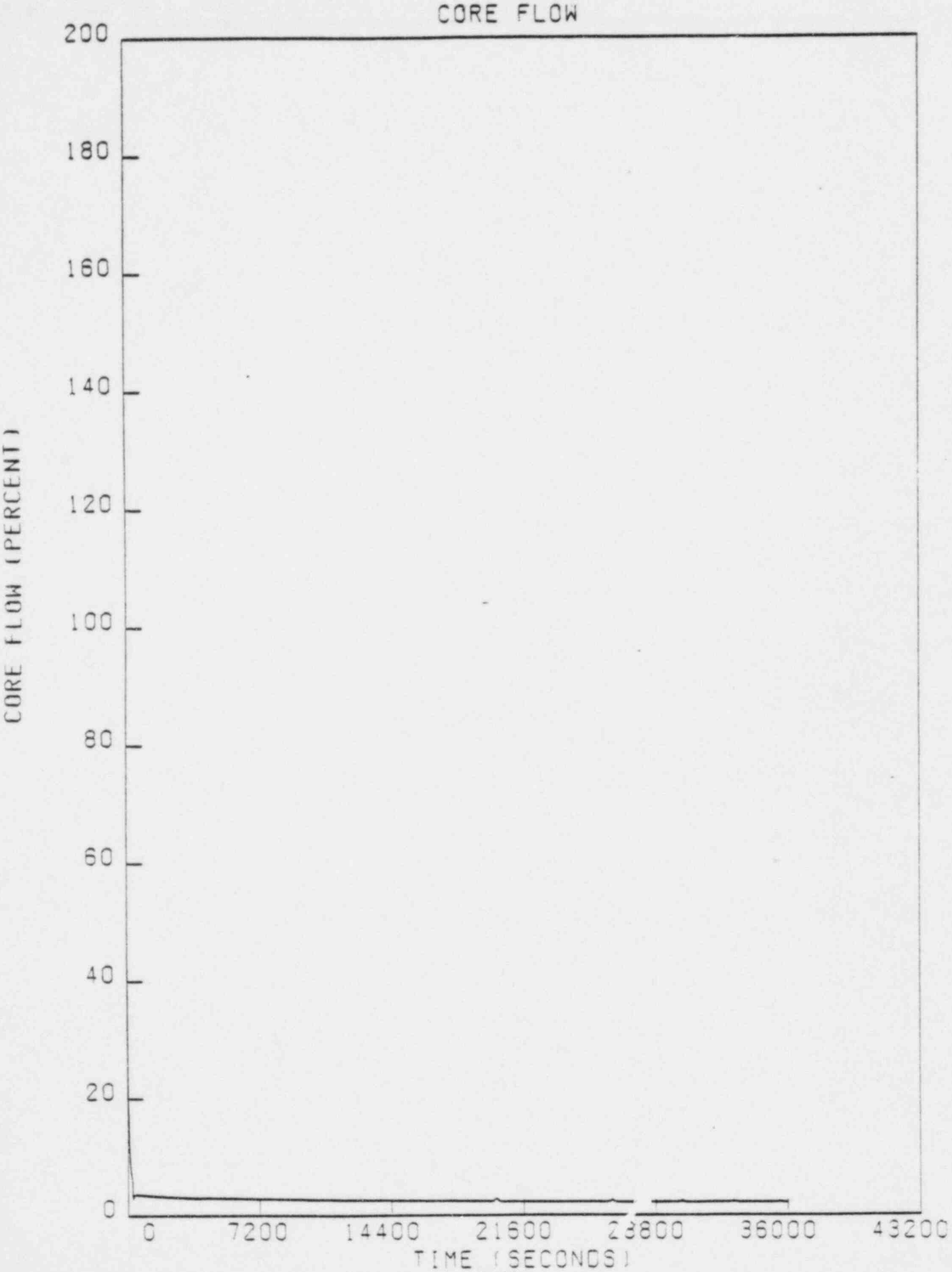
LIST OF LTC COMPUTER CODE PLOTS

PLOT A-1

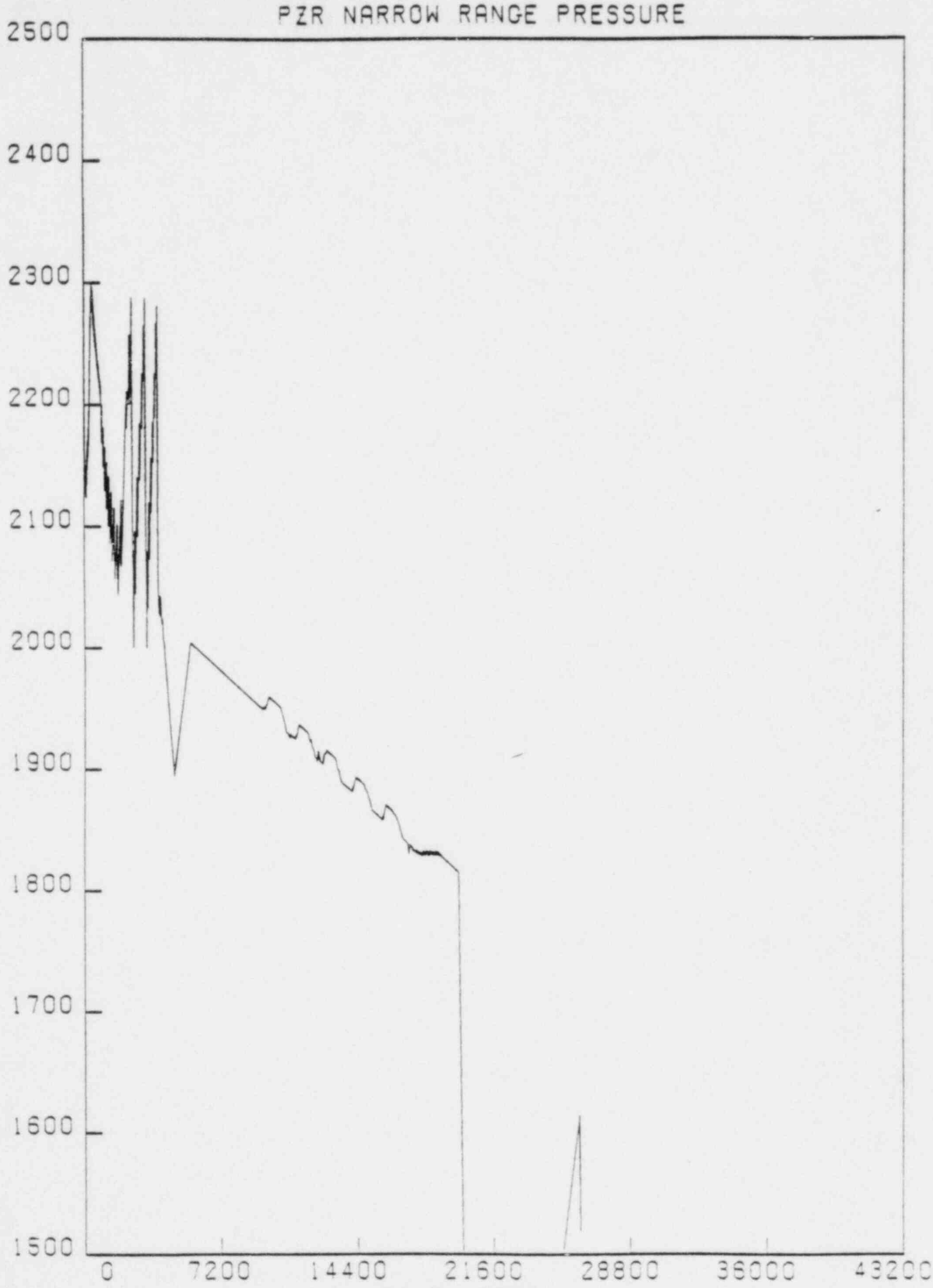


CORE POWER
CORE POWER (PERCENT)
140
120
100
80
60
40
20
0
-20
-40
-60
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

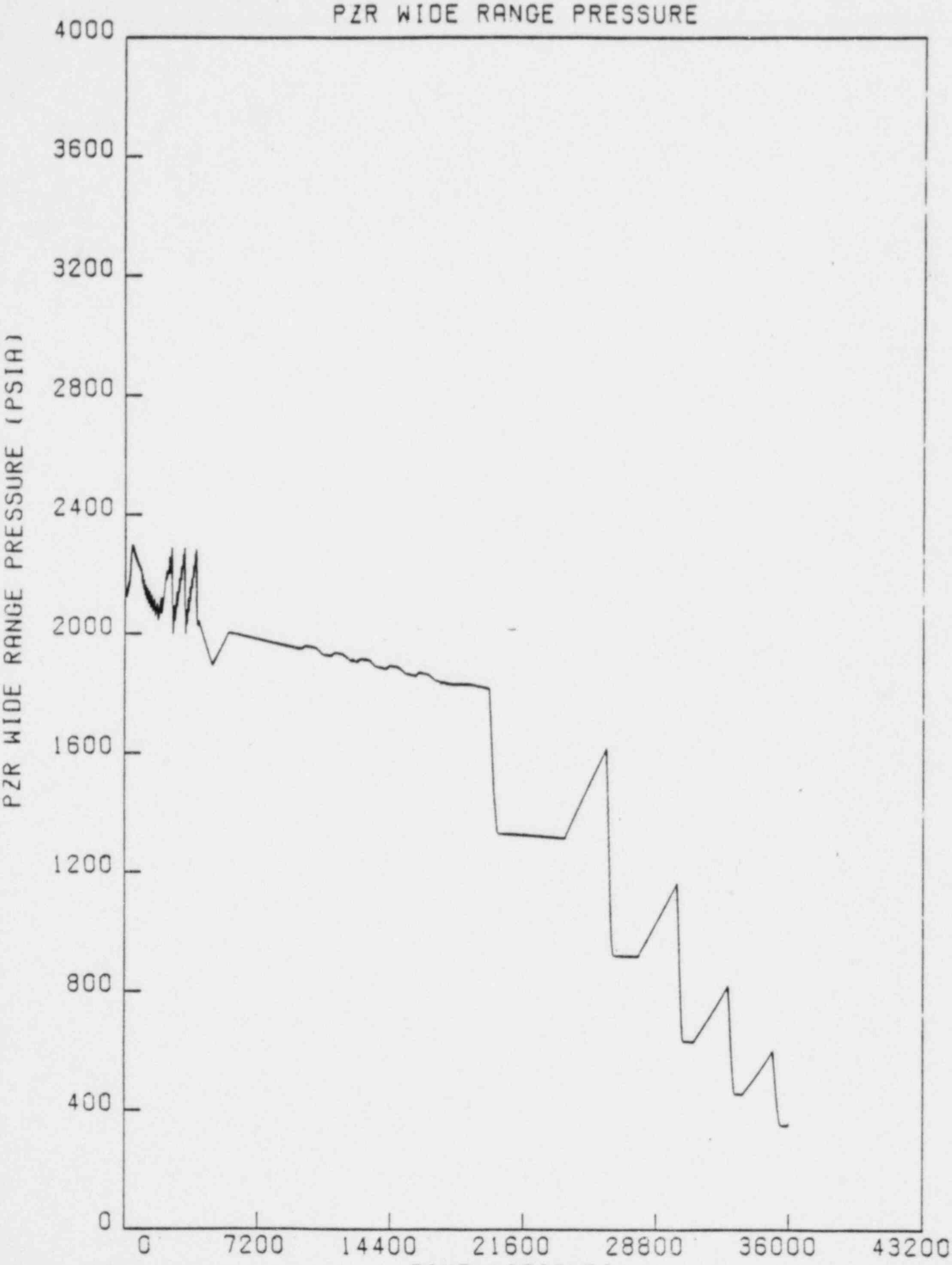
PLOT A-2


CORE FLOW
CORE FLOW (PERCENT)
200
180
160
140
120
100
80
60
40
20
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

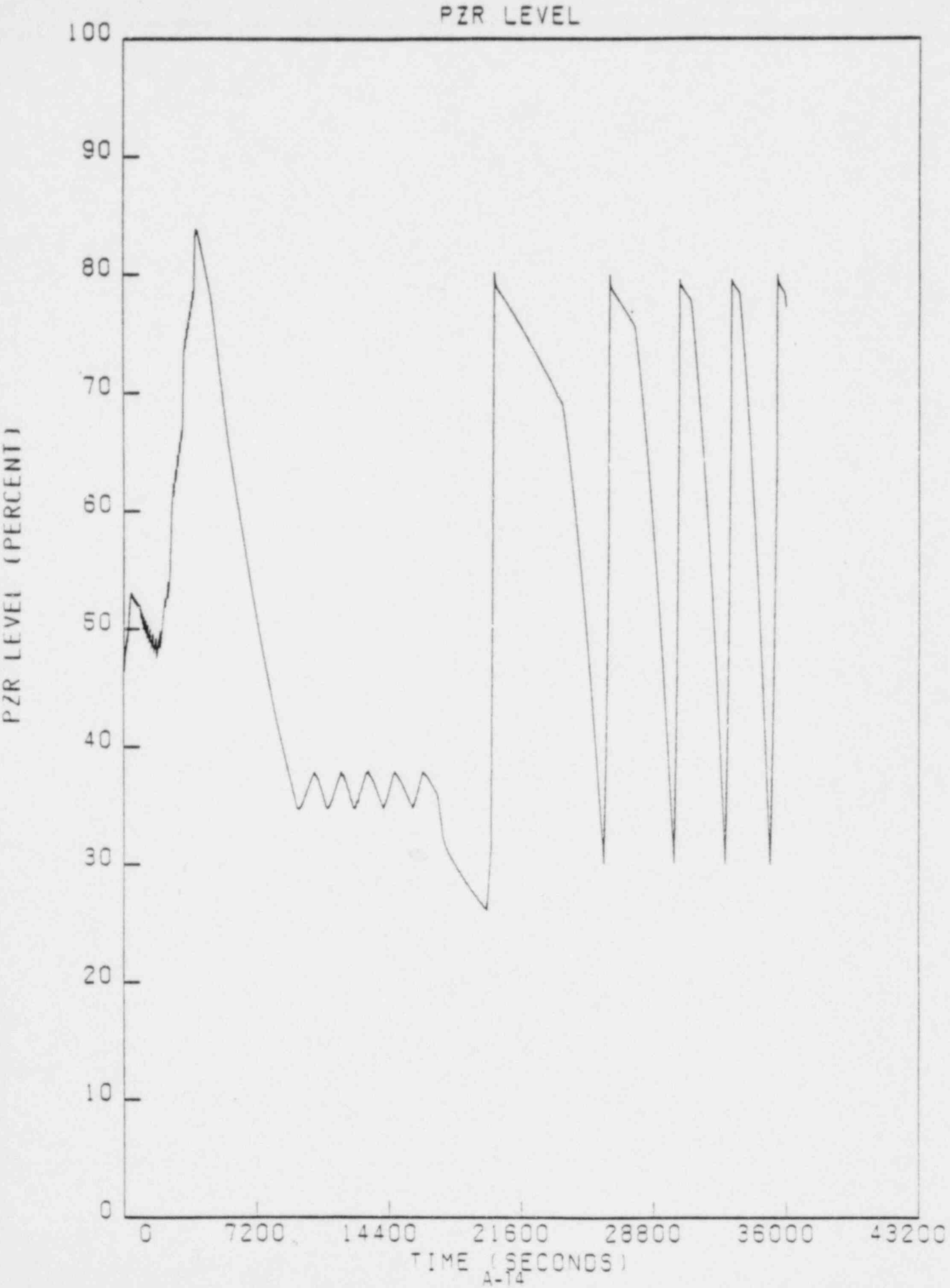
PLOT A-3


PZR NARROW RANGE PRESSURE
PZR NARROW RANGE PRESSURE (PSIA)
2500
2400
2300
2200
2100
2000
1900
1800
1700
1600
1500
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

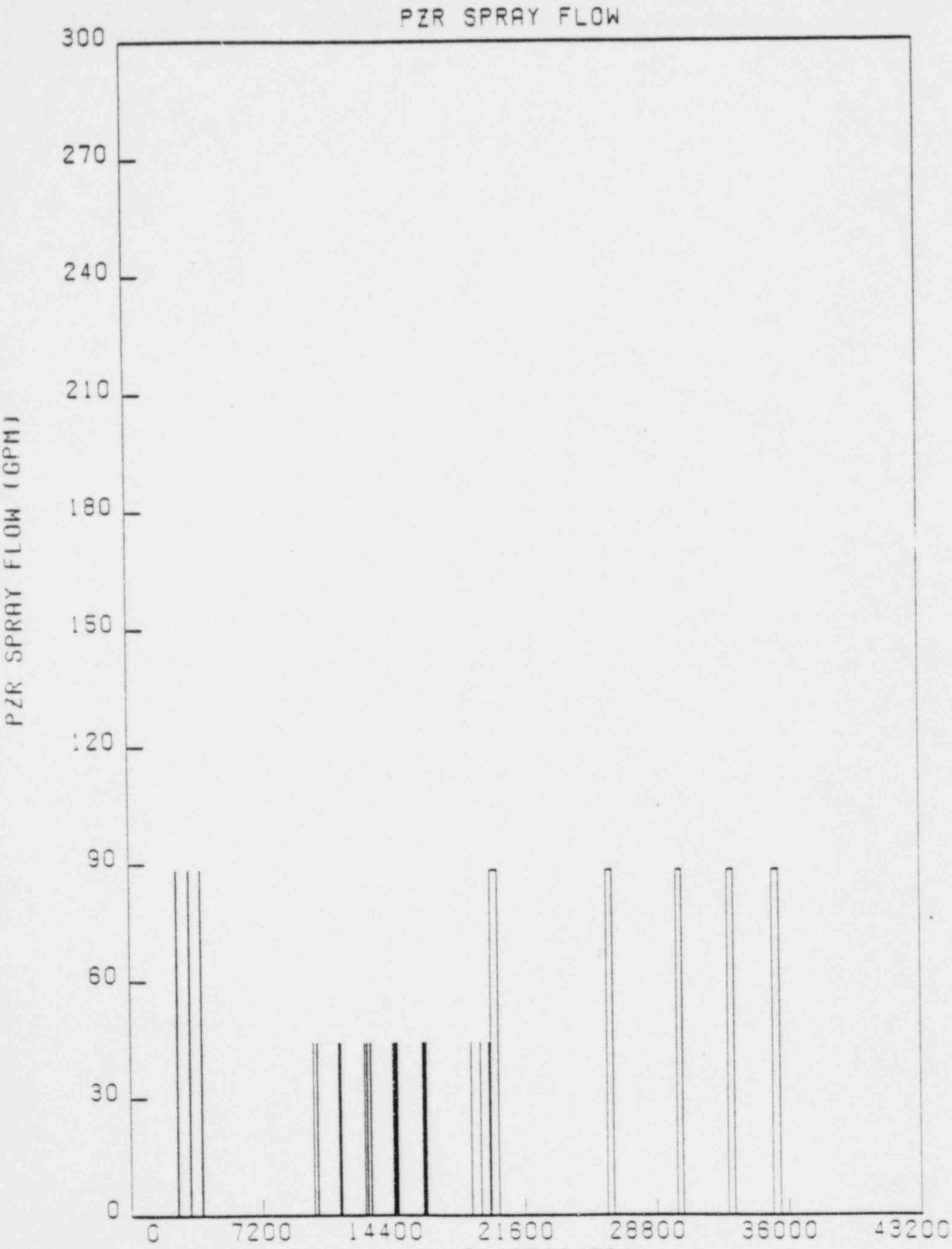
PLOT A-4


PZR WIDE RANGE PRESSURE
4000
3600
3200
2800
2400
2000
1600
1200
800
400
0
PZR WIDE RANGE PRESSURE (PSIA)
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

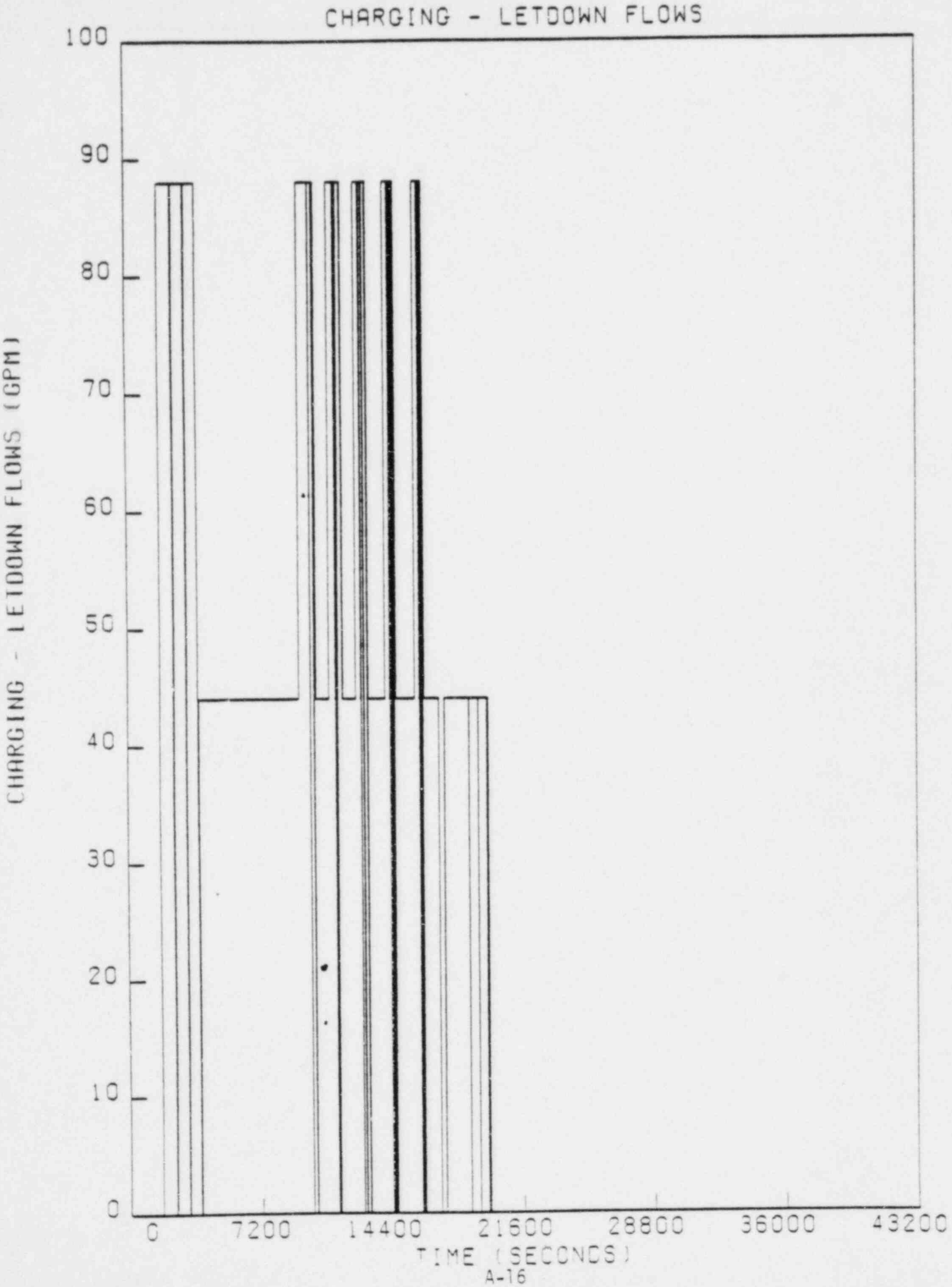
PLOT A-5


PZR LEVEL
PZR LEVEL (PERCENT)
100
90
80
70
60
50
40
30
20
10
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

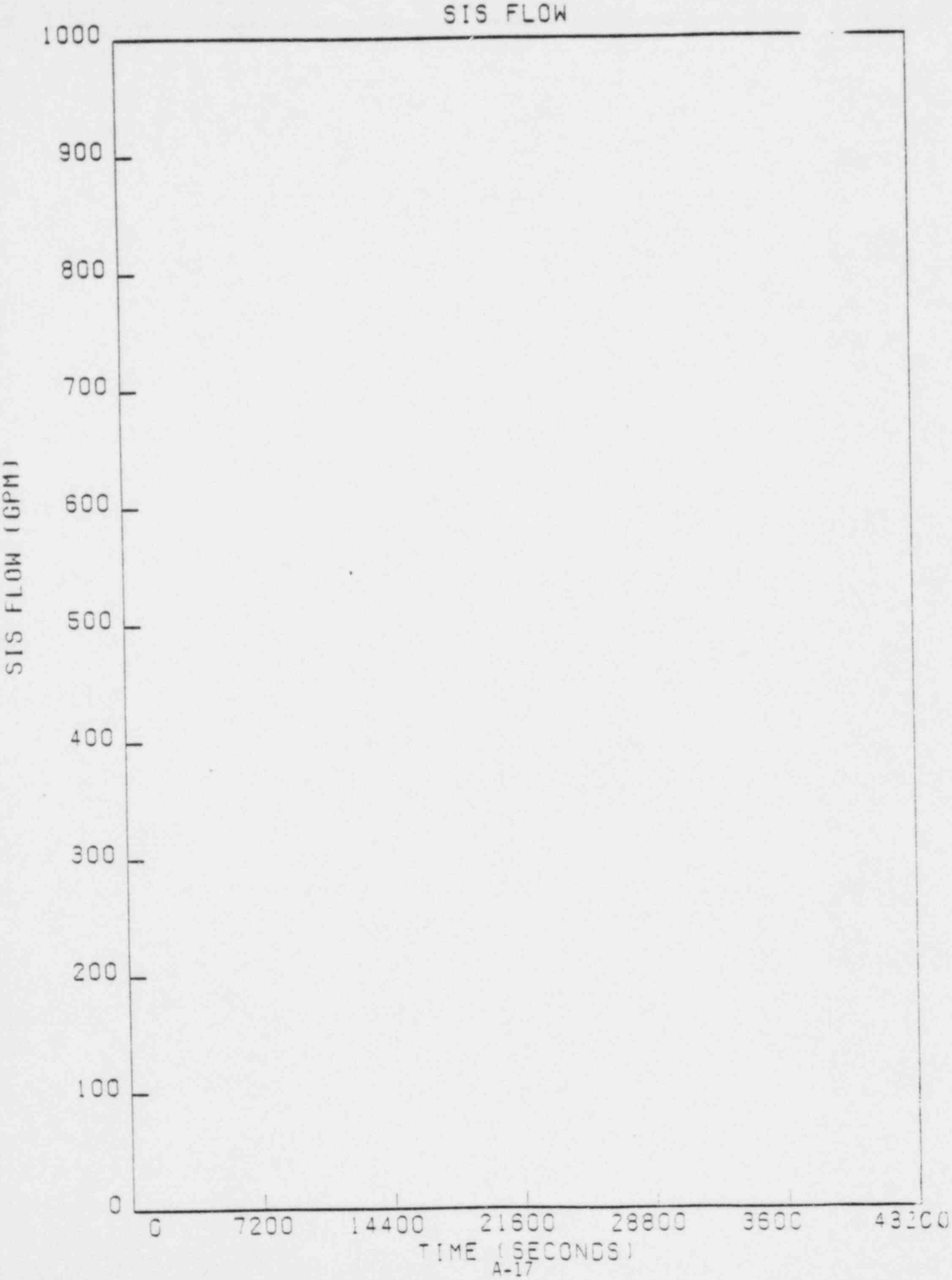
PLOT A-6


PZR SPRAY FLOW
300
270
240
210
180
150
120
90
60
30
0
PZR SPRAY FLOW (GPM)
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

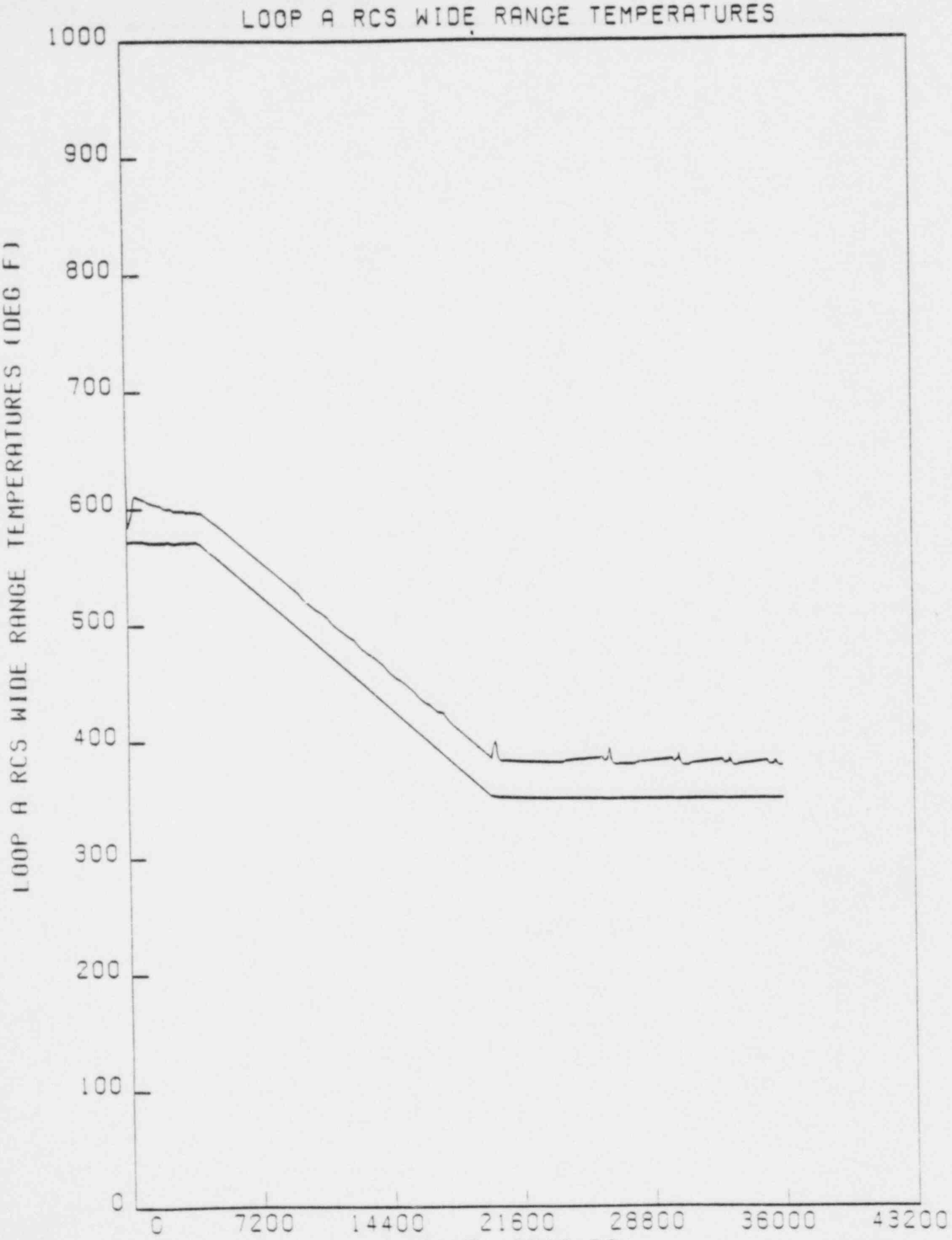
PLOT A-7


CHARGING - LETDOWN FLOWS
CHARGING - LETDOWN FLOWS (GPM)
100
90
80
70
60
50
40
30
20
10
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

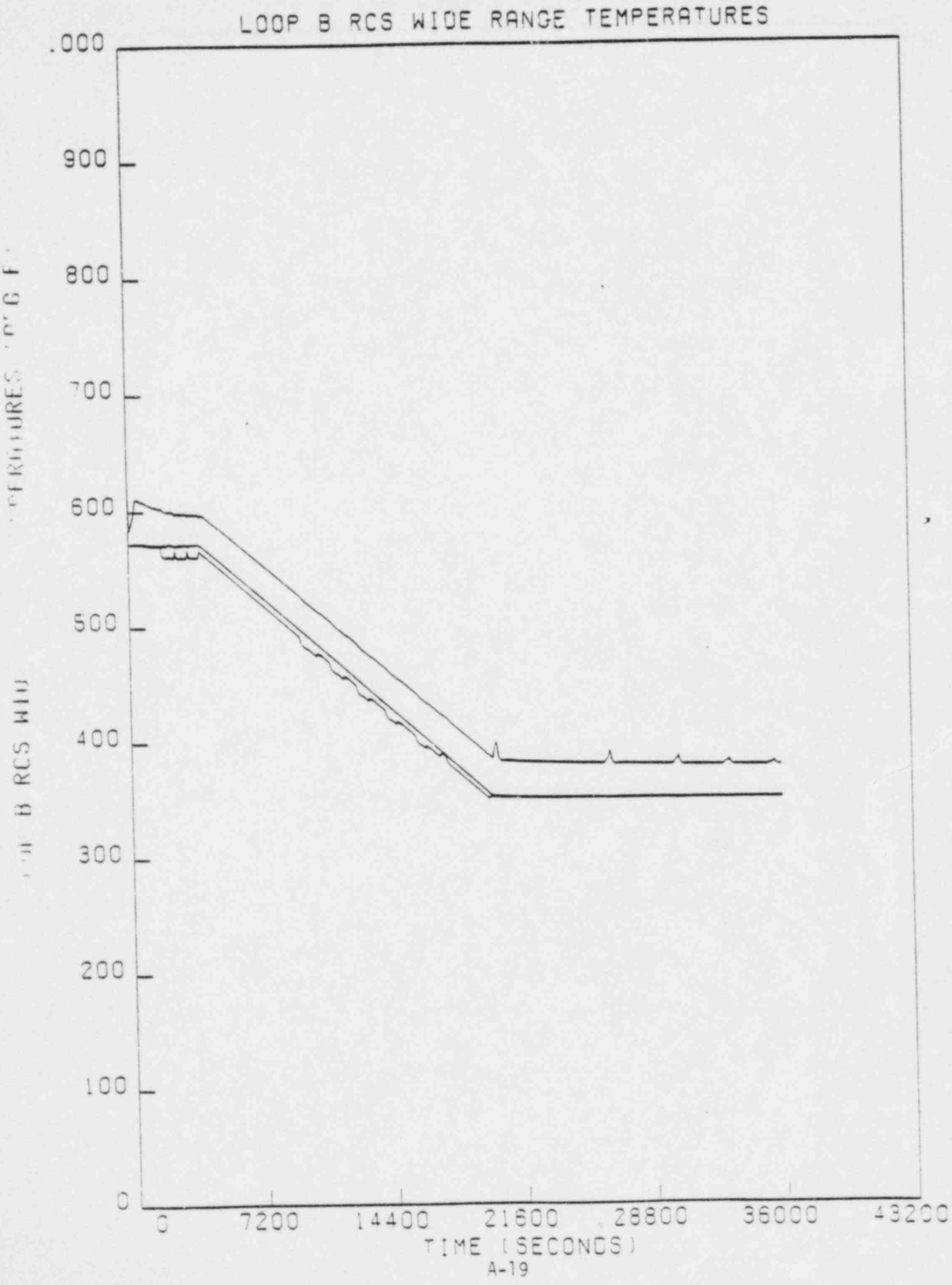
PLOT A-8


SIS FLOW
SIS FLOW (GPM)
1000
900
800
700
600
500
400
300
200
100
0
0
7200
14400
21600
28800
3600C
43200
TIME (SECONDS)

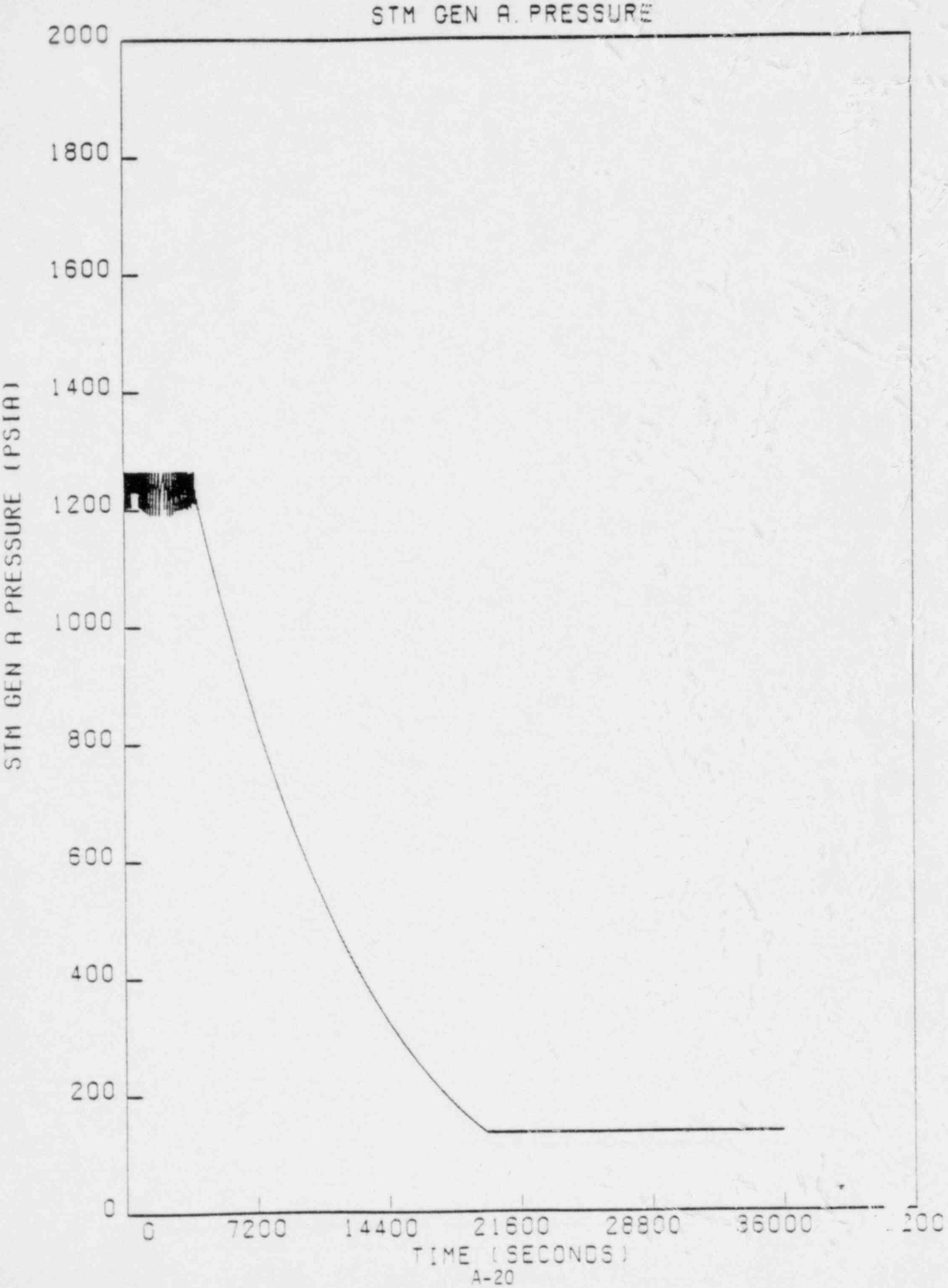
PLOT A-9


LOOP A RCS WIDE RANGE TEMPERATURES
LOOP A RCS WIDE RANGE TEMPERATURES (DEG F)
1000
900
800
700
600
500
400
300
200
100
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

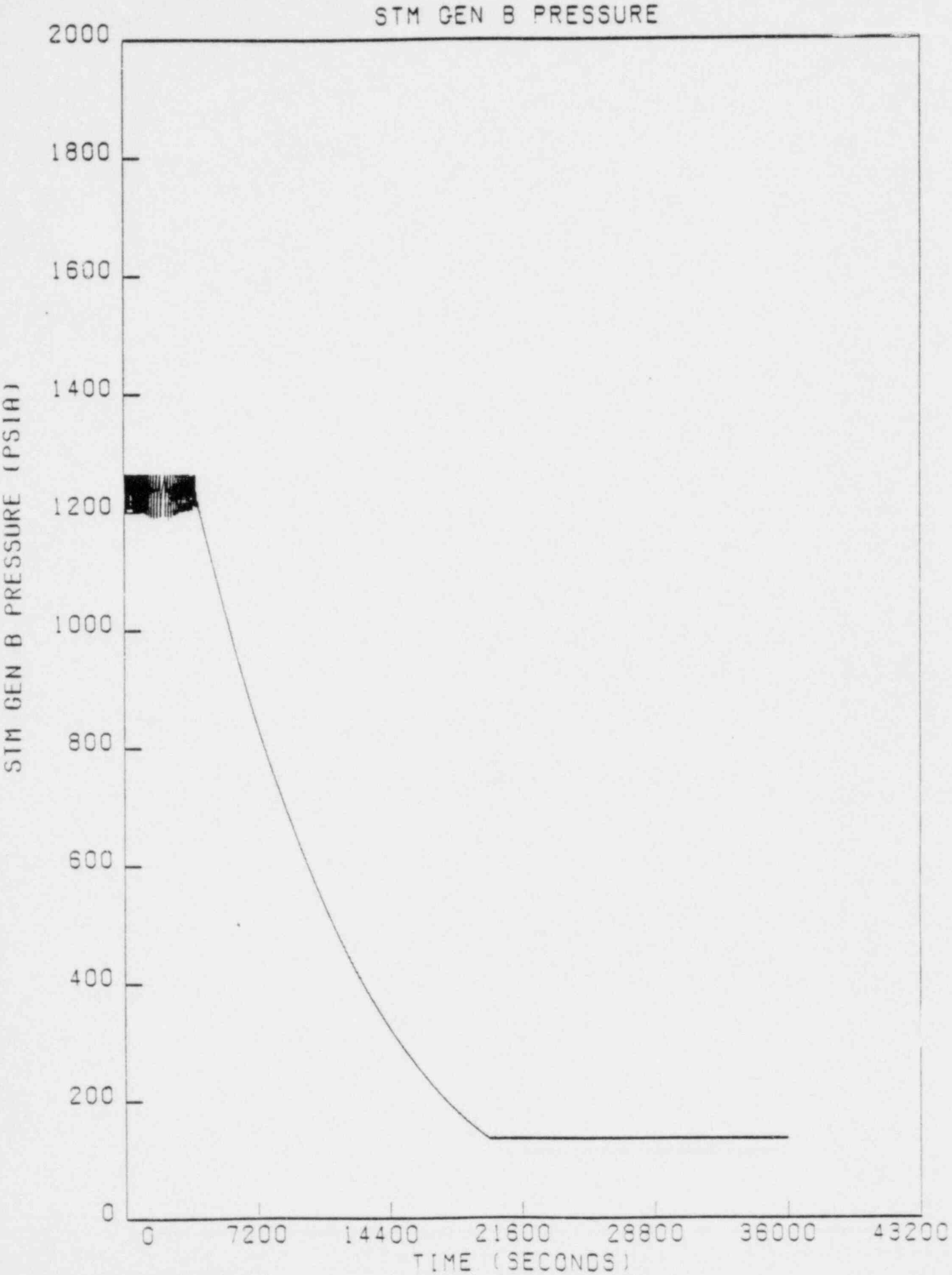
PLOT A-10


LOOP B RCS WIDE RANGE TEMPERATURES
.000
900
800
700
600
500
400
300
200
100
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)
B RCS WID
ERATURES (DEG F)

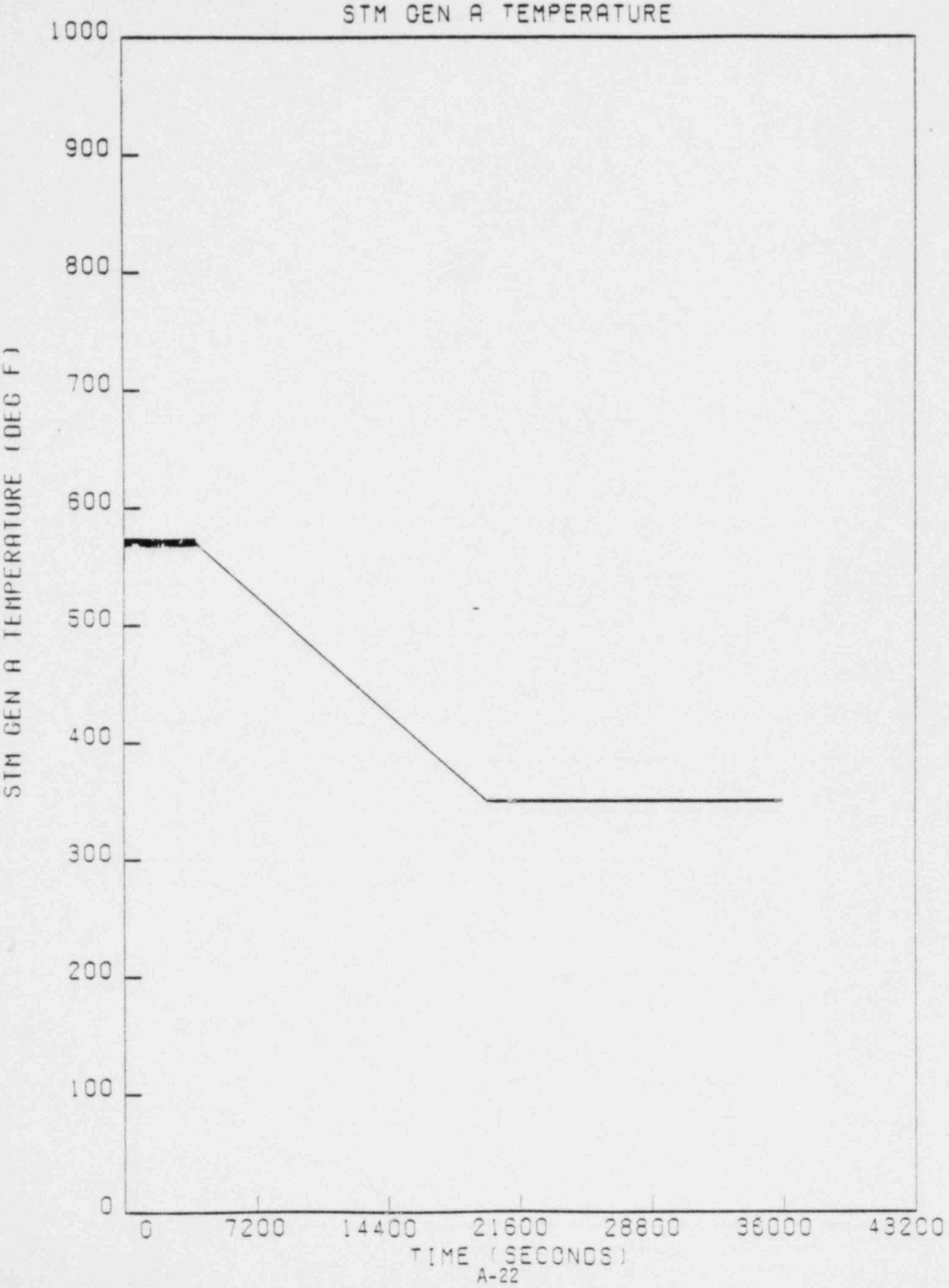
PLOT A-11


STM GEN A PRESSURE
STM GEN A PRESSURE (PSIA)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
7200
14400
21600
28800
36000
200
TIME (SECONDS)

PLOT A-12


STM GEN B PRESSURE
STM GEN B PRESSURE (PSIA)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

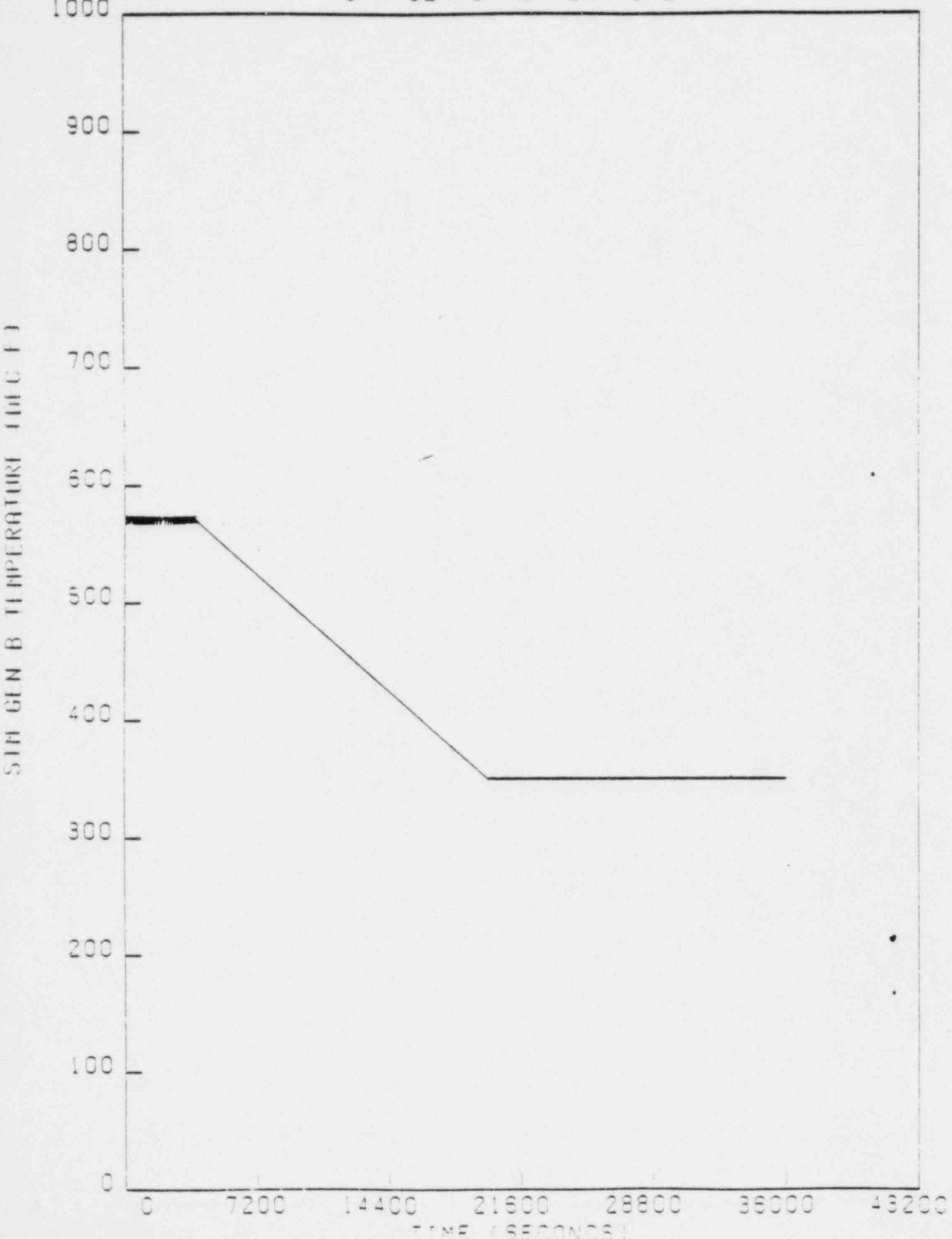
PLOT A-13



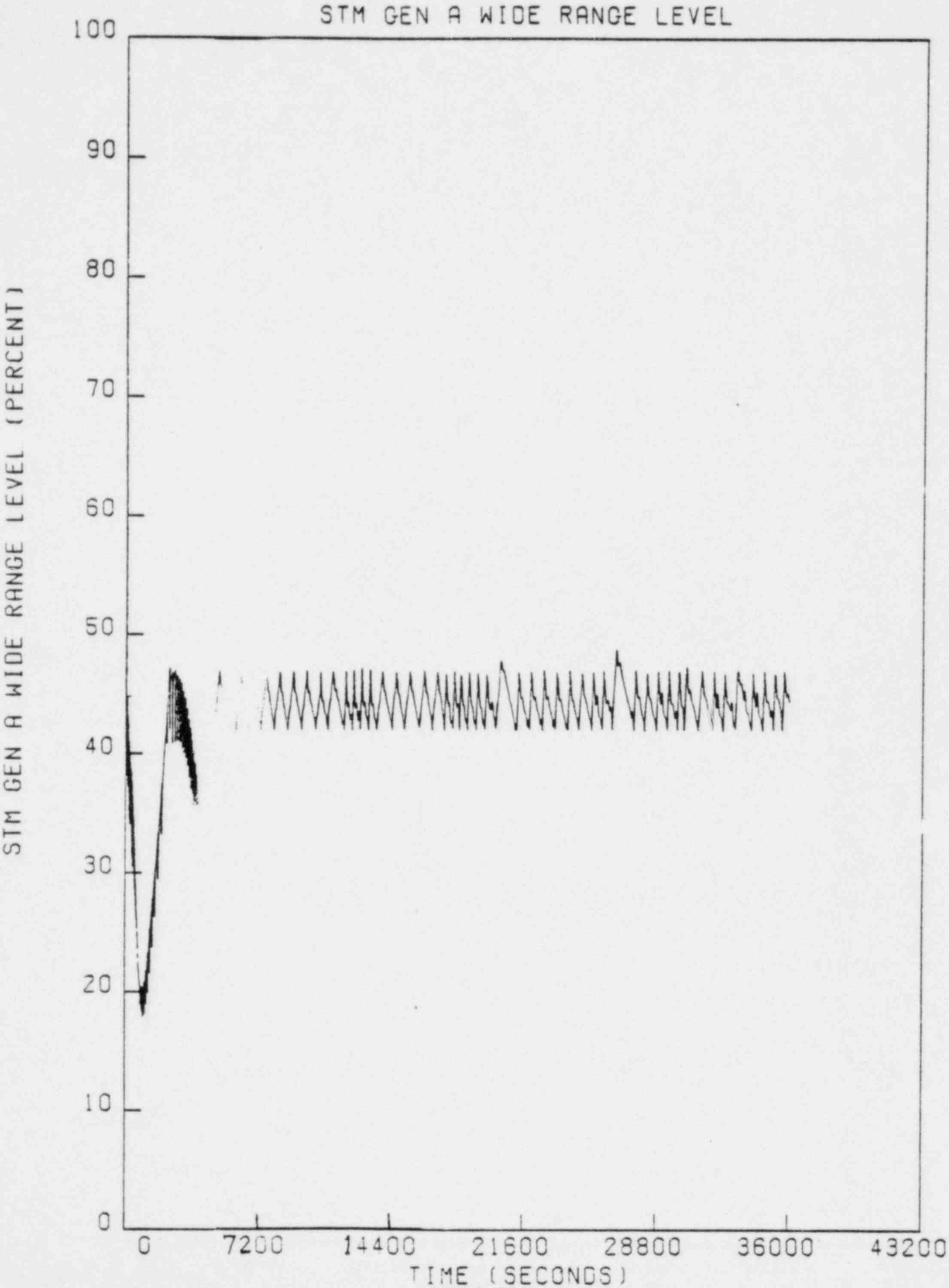
STM GEN A TEMPERATURE
STM GEN A TEMPERATURE (DEG F)
1000
900
800
700
600
500
400
300
200
100
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

PLOT A-14

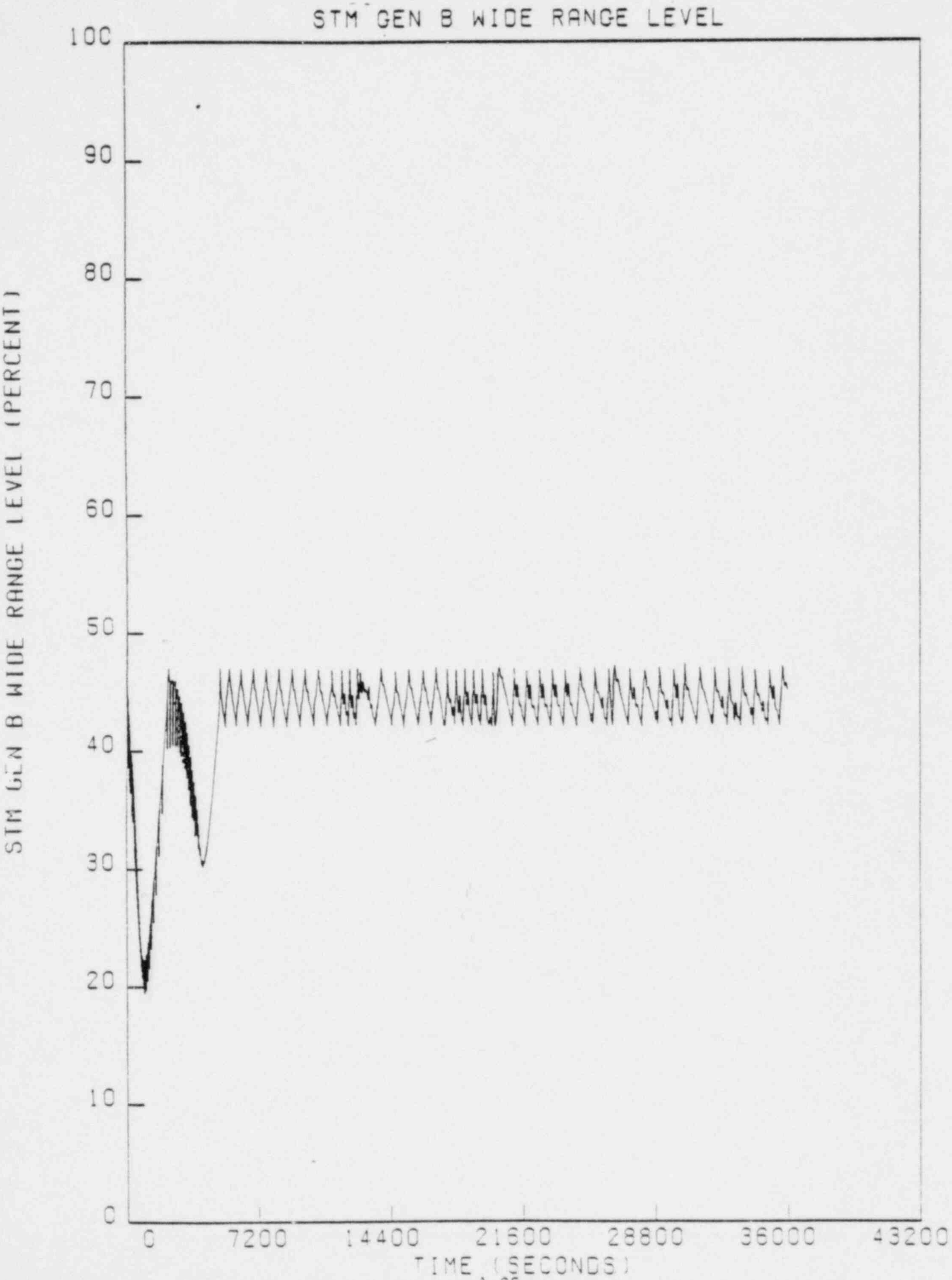
STM GEN B TEMPERATURE



PLOT A-15

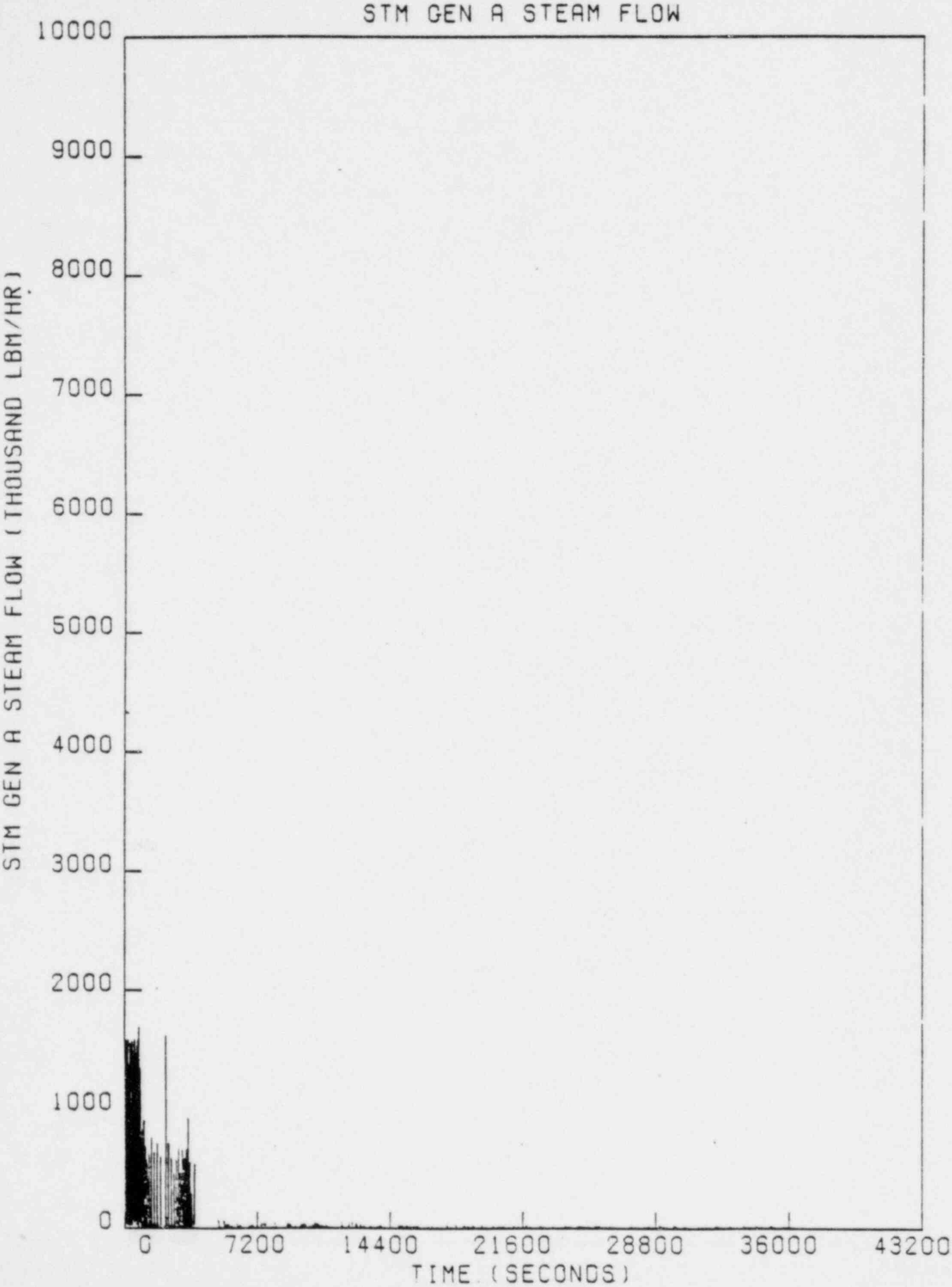

STM GEN A WIDE RANGE LEVEL
STM GEN A WIDE RANGE LEVEL (PERCENT)
100
90
80
70
60
50
40
30
20
10
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

PLOT A-16



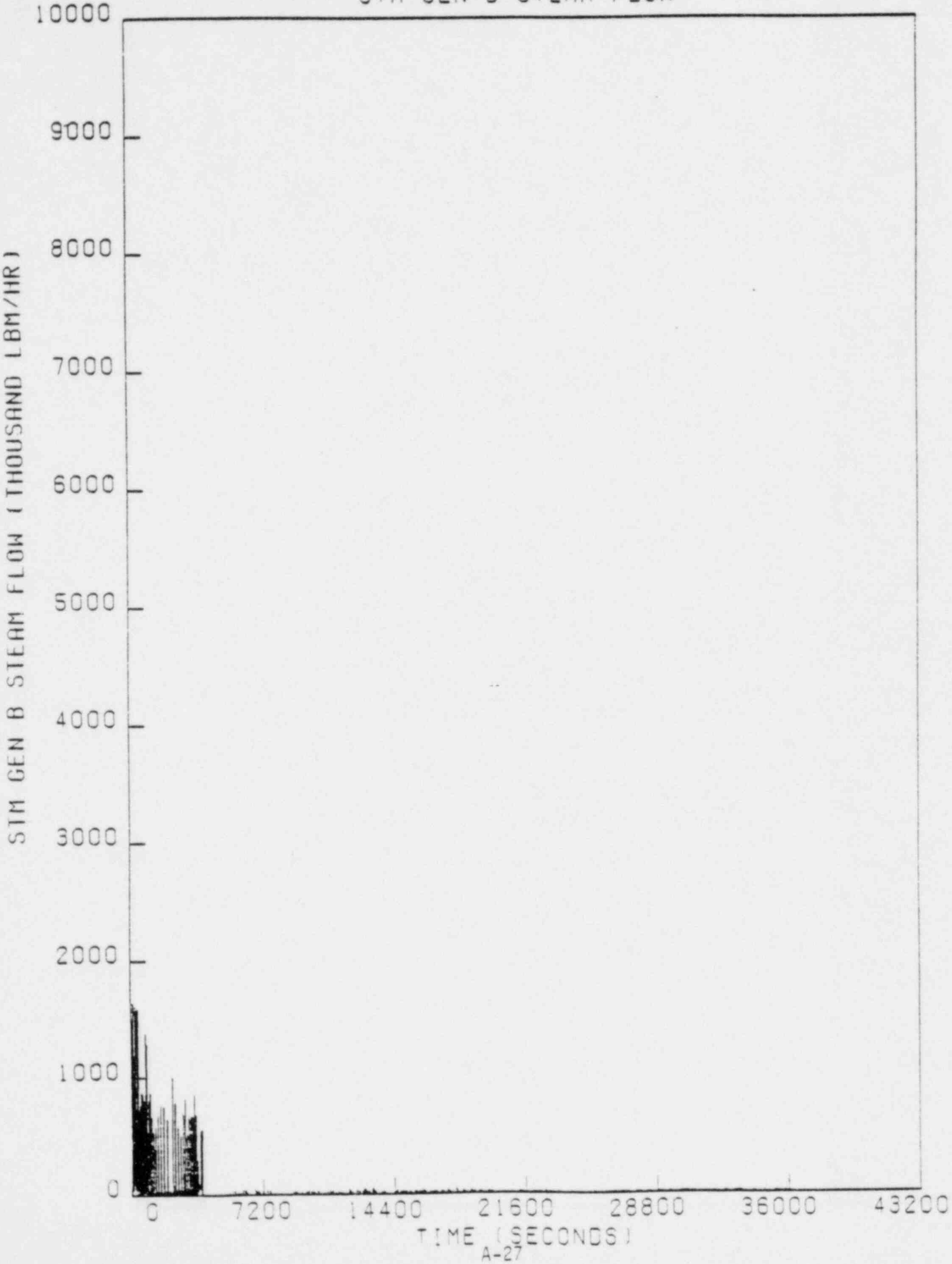
STM GEN B WIDE RANGE LEVEL
STM GEN B WIDE RANGE LEVEL (PERCENT)
100
90
80
70
60
50
40
30
20
10
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

PLOT A-17

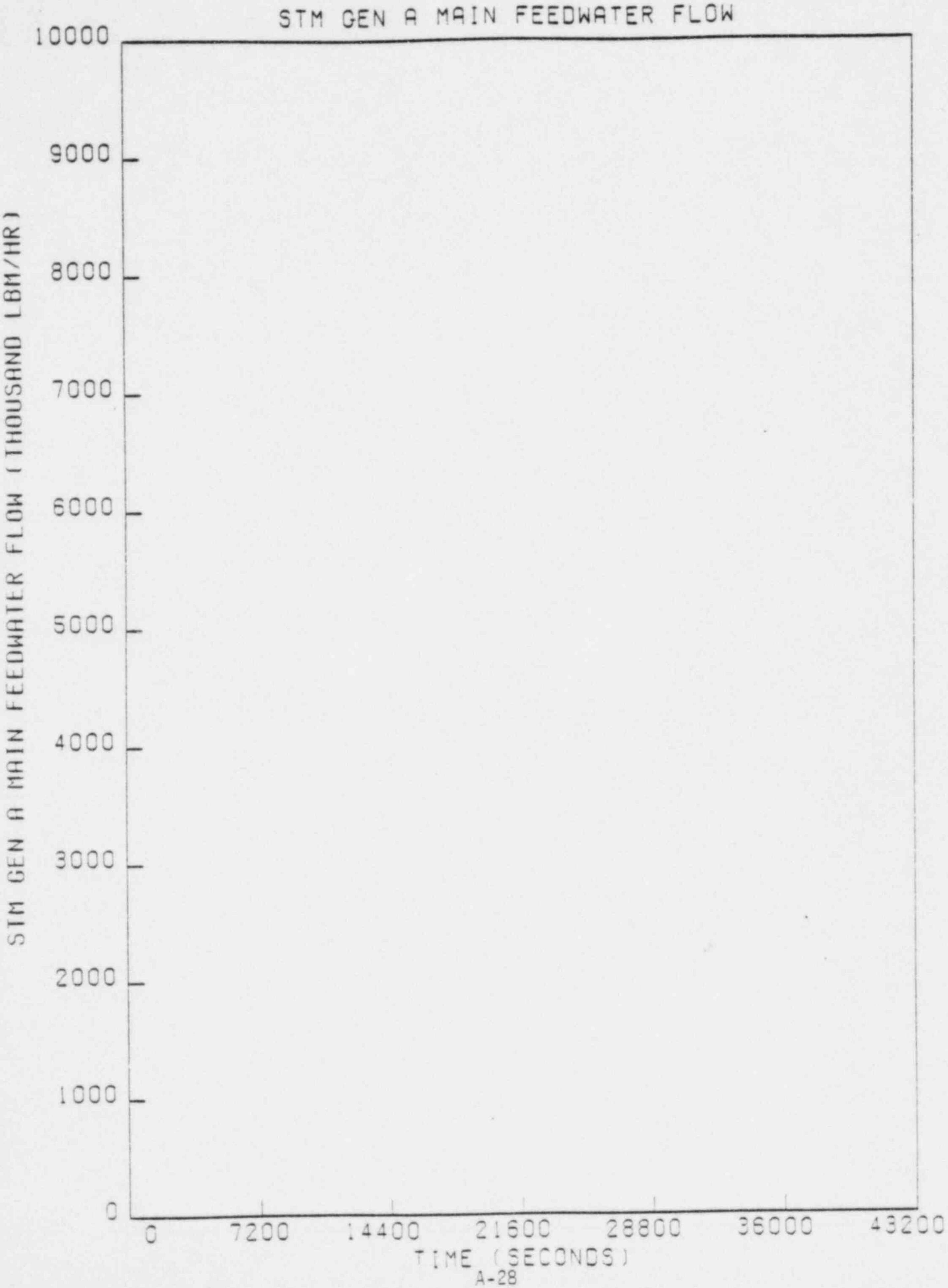

STM GEN A STEAM FLOW
STM GEN A STEAM FLOW (THOUSAND LBM/HR)
10000
9000
8000
7000
6000
5000
4000
3000
2000
1000
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

PLOT A-18

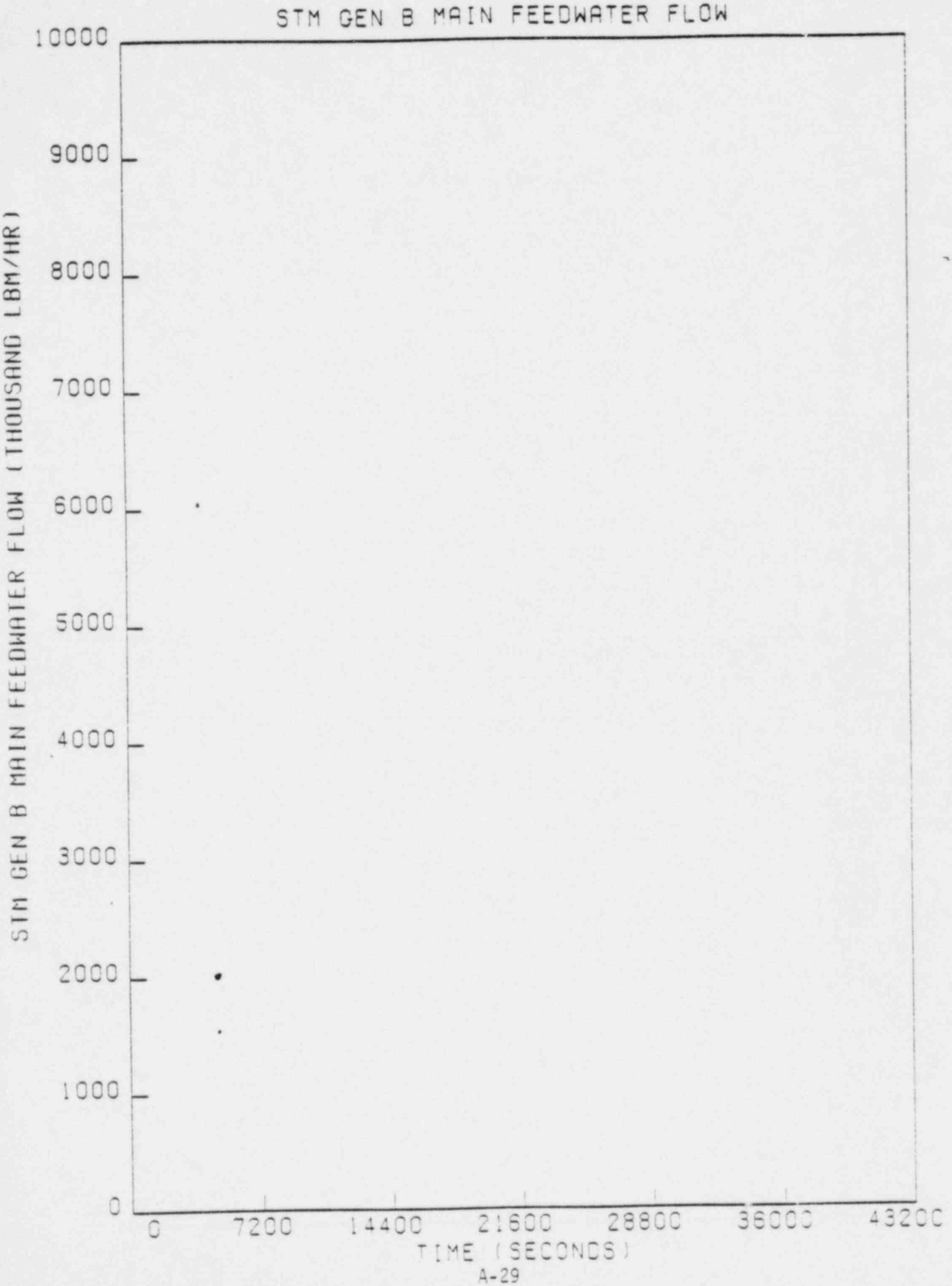
STM GEN B STEAM FLOW



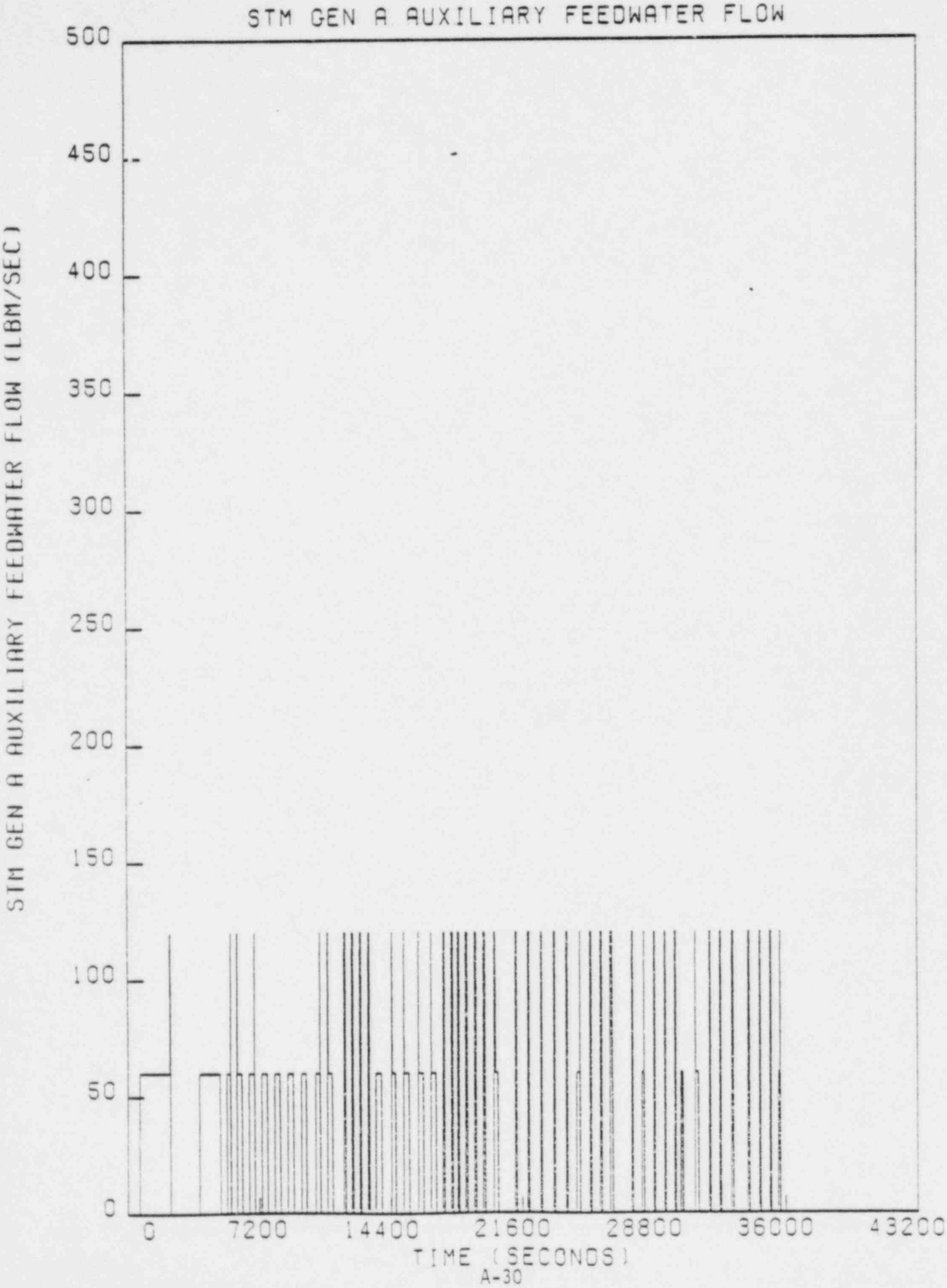
PLOT A-19



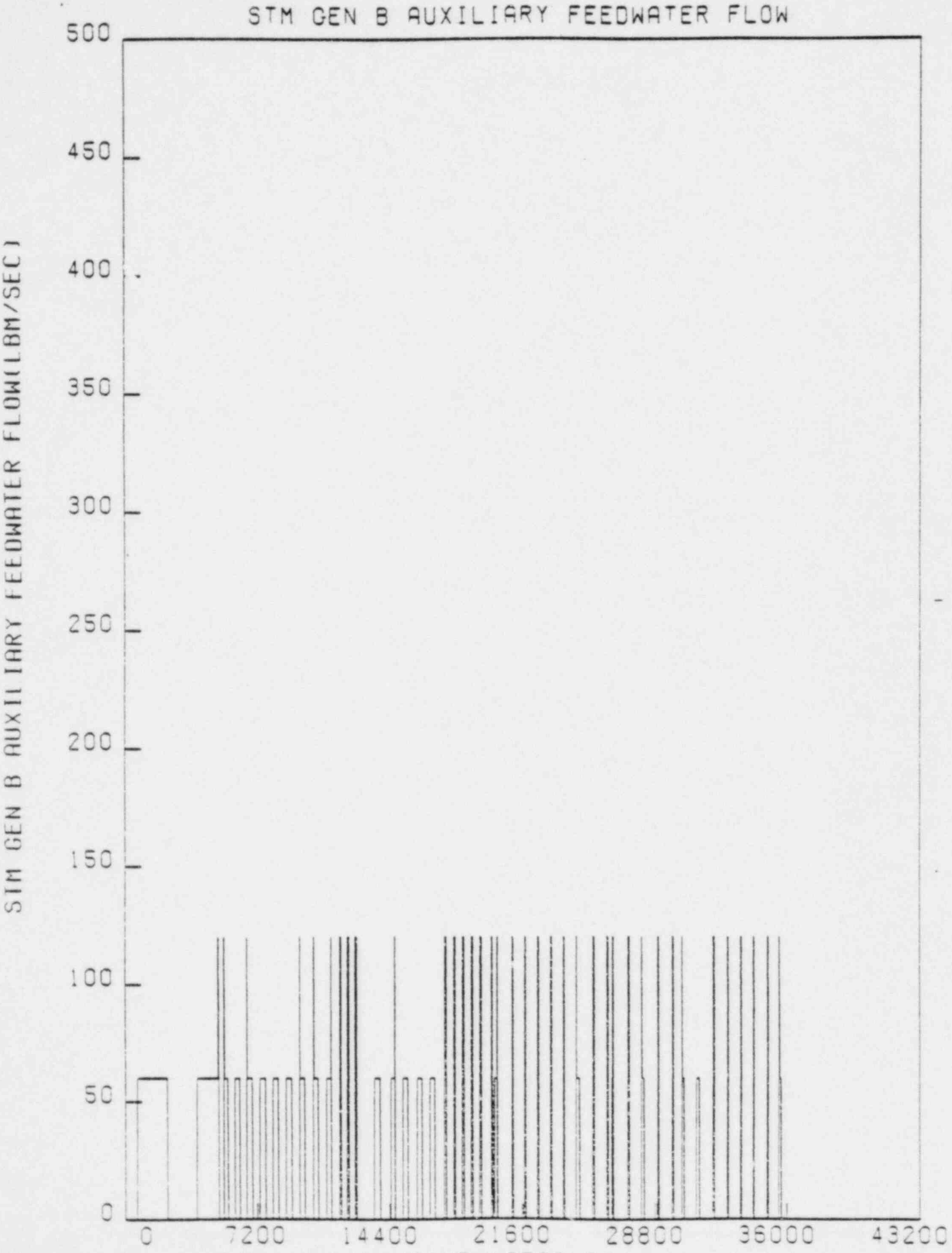
STM GEN A MAIN FEEDWATER FLOW
STM GEN A MAIN FEEDWATER FLOW (THOUSAND LBM/HR)
10000
9000
8000
7000
6000
5000
4000
3000
2000
1000
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)


STM GEN B MAIN FEEDWATER FLOW
STM GEN B MAIN FEEDWATER FLOW (THOUSAND LBM/HR)
10000
9000
8000
7000
6000
5000
4000
3000
2000
1000
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

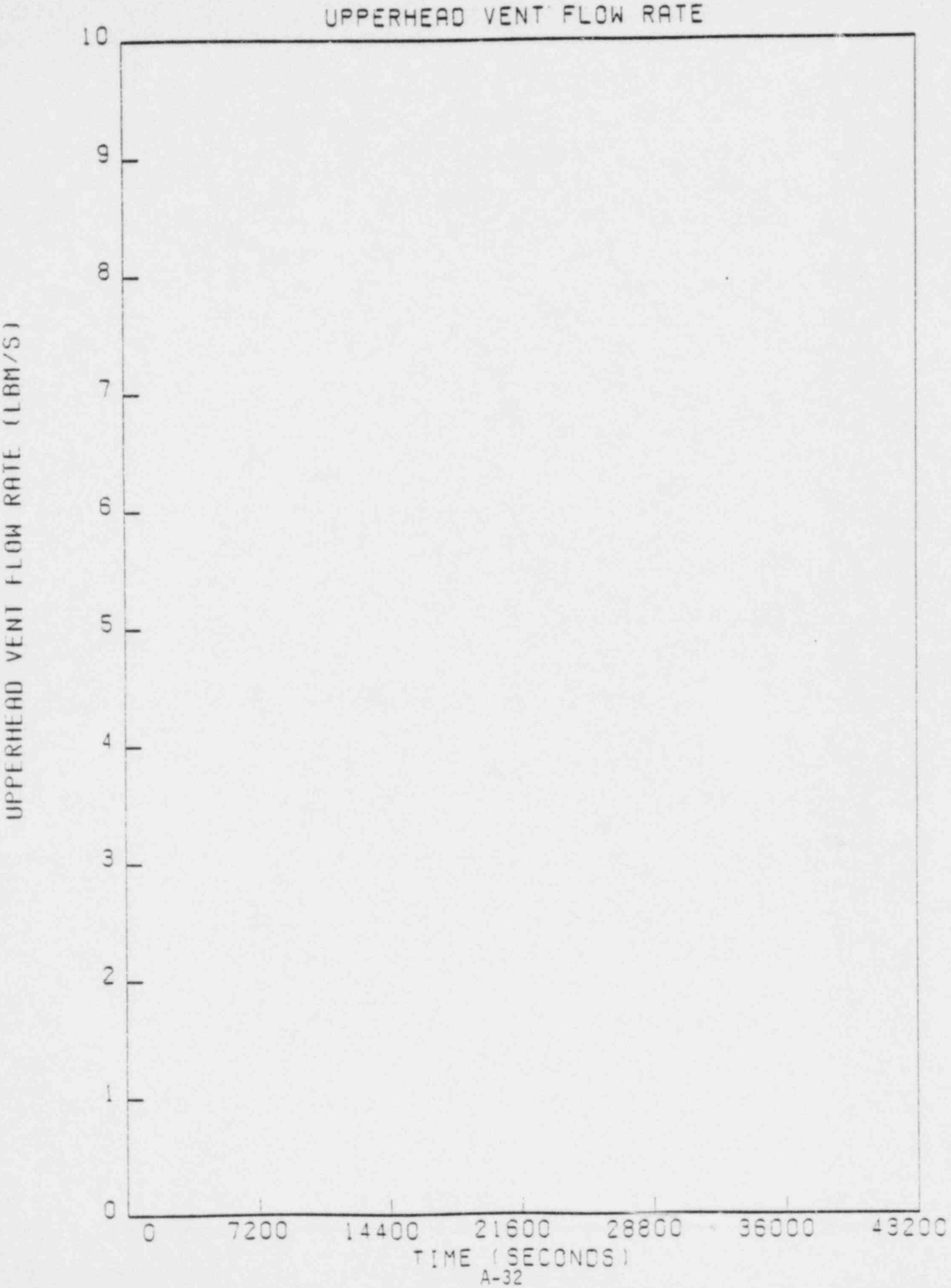
PLOT A-21


STM GEN A AUXILIARY FEEDWATER FLOW
STM GEN A AUXILIARY FEEDWATER FLOW (LBM/SEC)
500
450
400
350
300
250
200
150
100
50
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

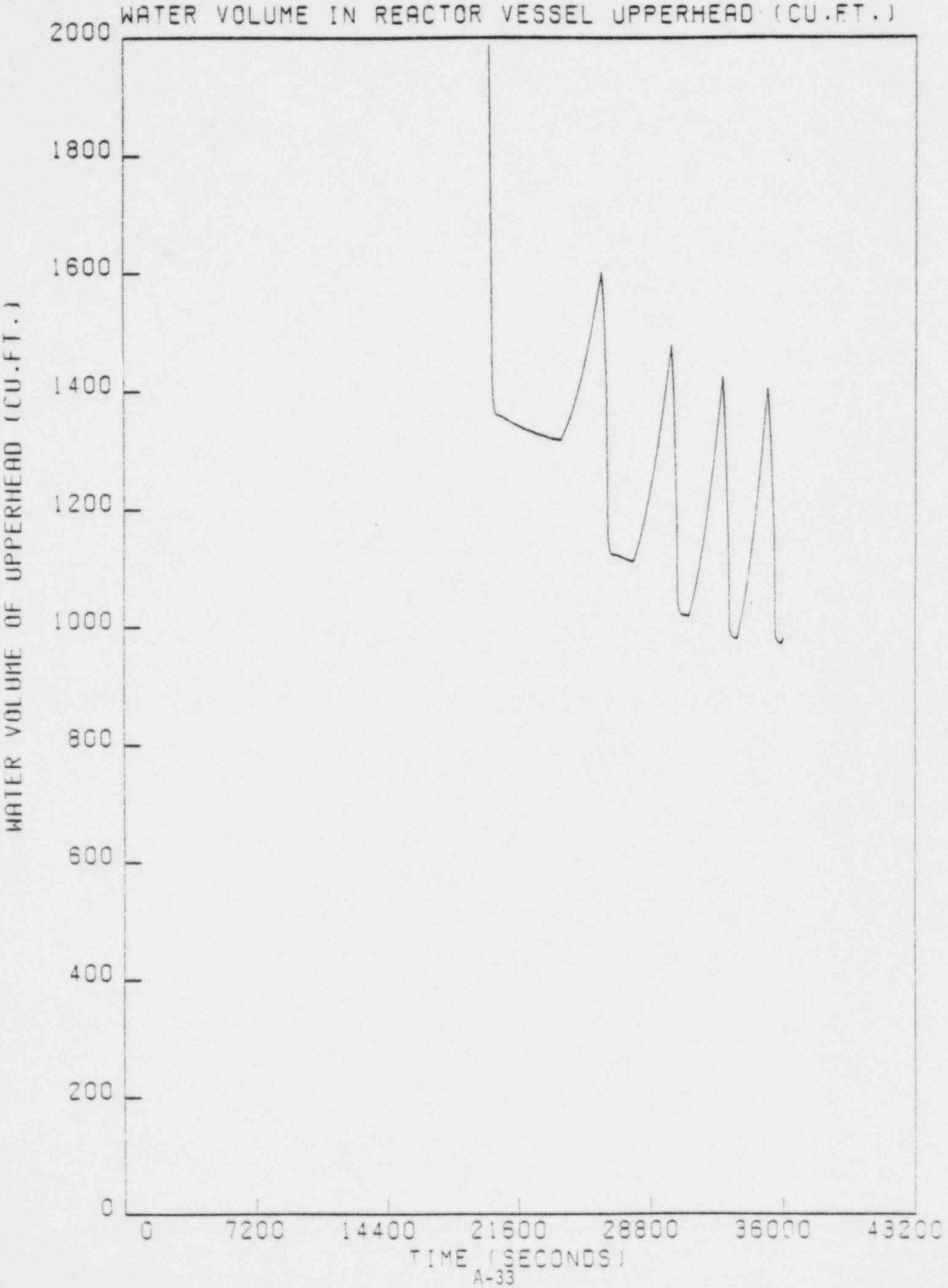
PLOT A-22


STM GEN B AUXILIARY FEEDWATER FLOW
STM GEN B AUXILIARY FEEDWATER FLOW(LBM/SEC)
500
450
400
350
300
250
200
150
100
50
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

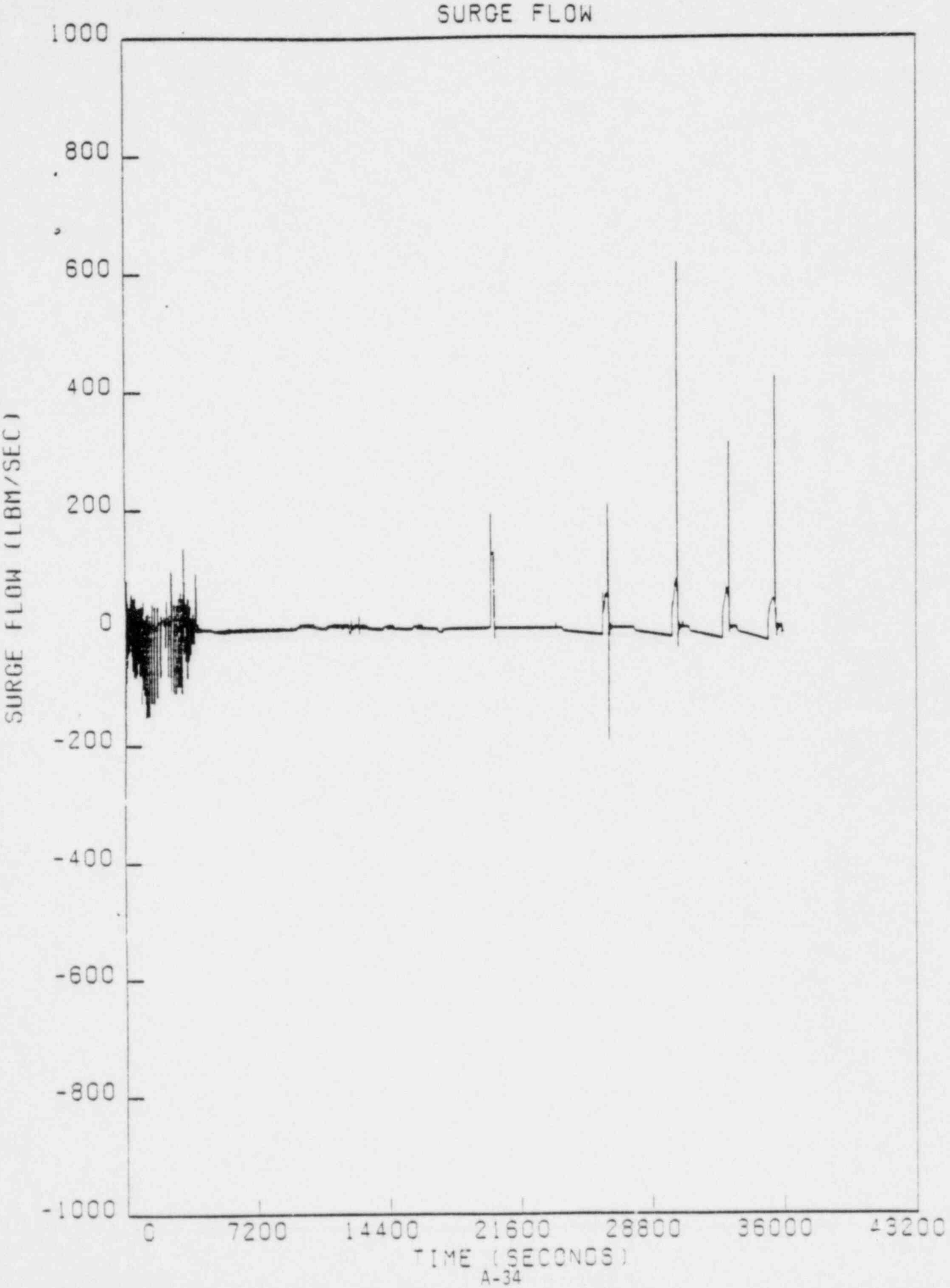
PLOT A-23


UPPERHEAD VENT FLOW RATE
UPPERHEAD VENT FLOW RATE (LBM/S)
10
9
8
7
6
5
4
3
2
1
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

PLOT A-24

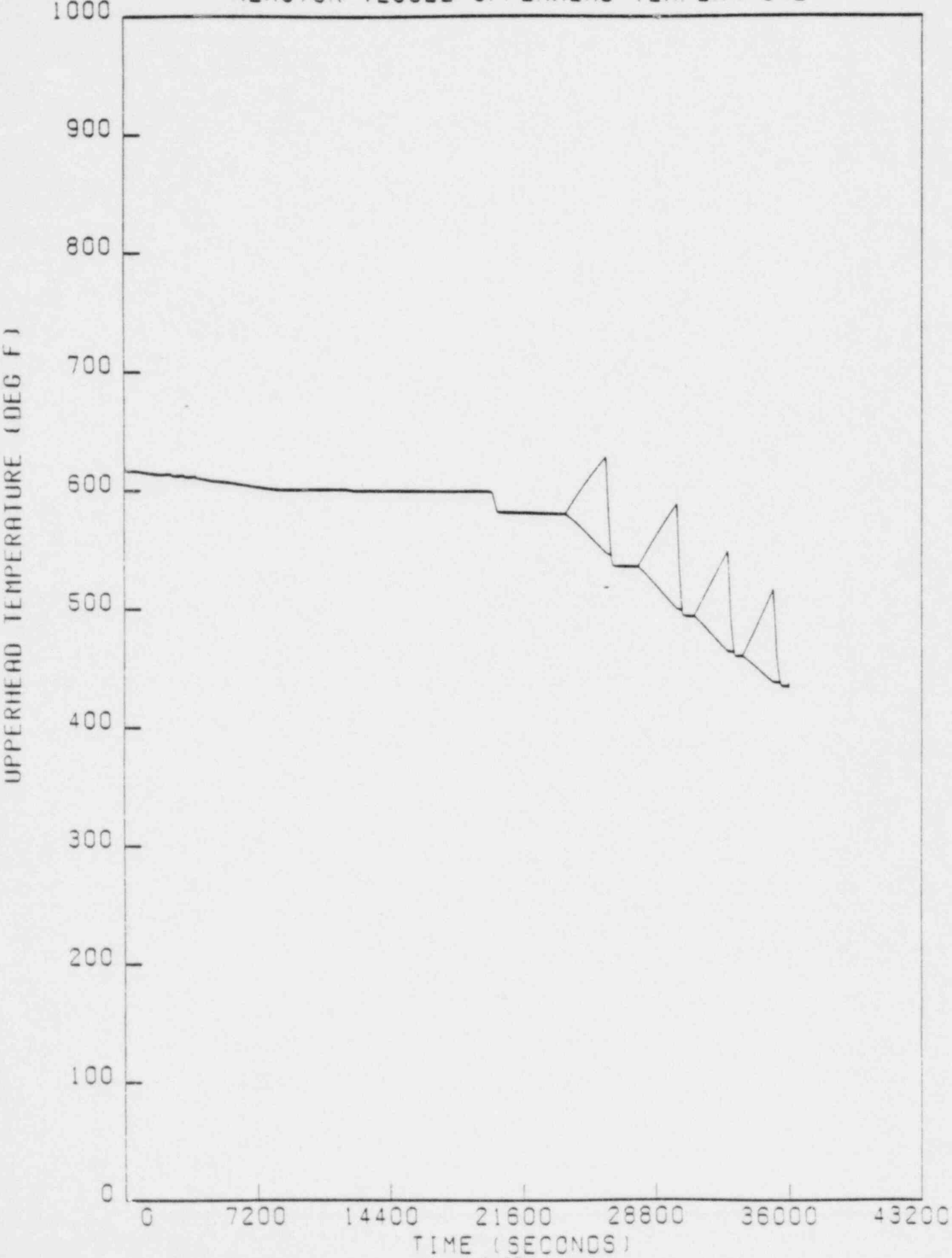

WATER VOLUME IN REACTOR VESSEL UPPERHEAD (CU.FT.)
WATER VOLUME OF UPPERHEAD (CU.FT.)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

PLOT A-25

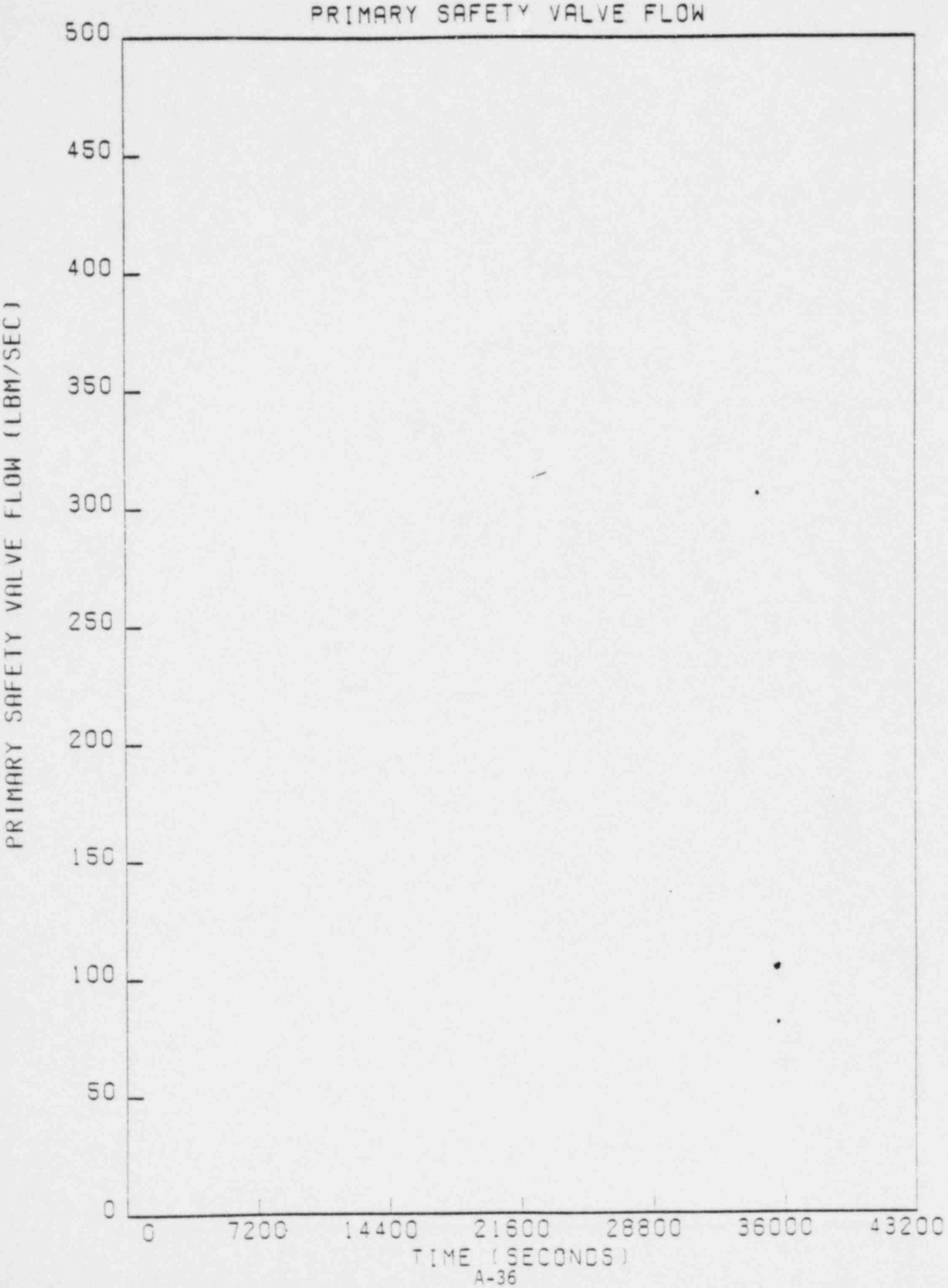

SURGE FLOW
1000
800
600
400
200
0
-200
-400
-600
-800
-1000
SURGE FLOW (LBM/SEC)
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

PLOT A-26

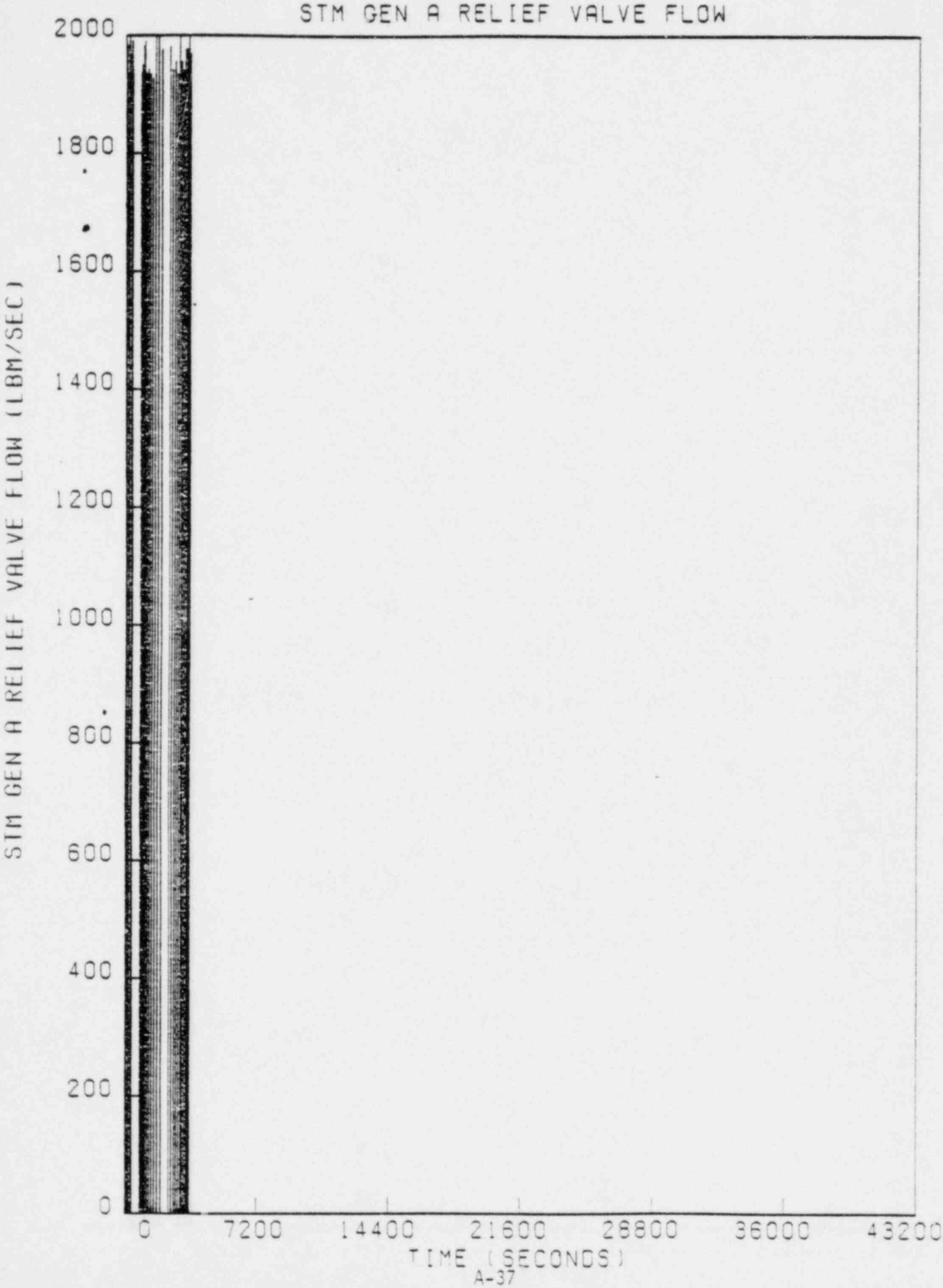
REACTOR VESSEL UPPERHEAD TEMPERATURE



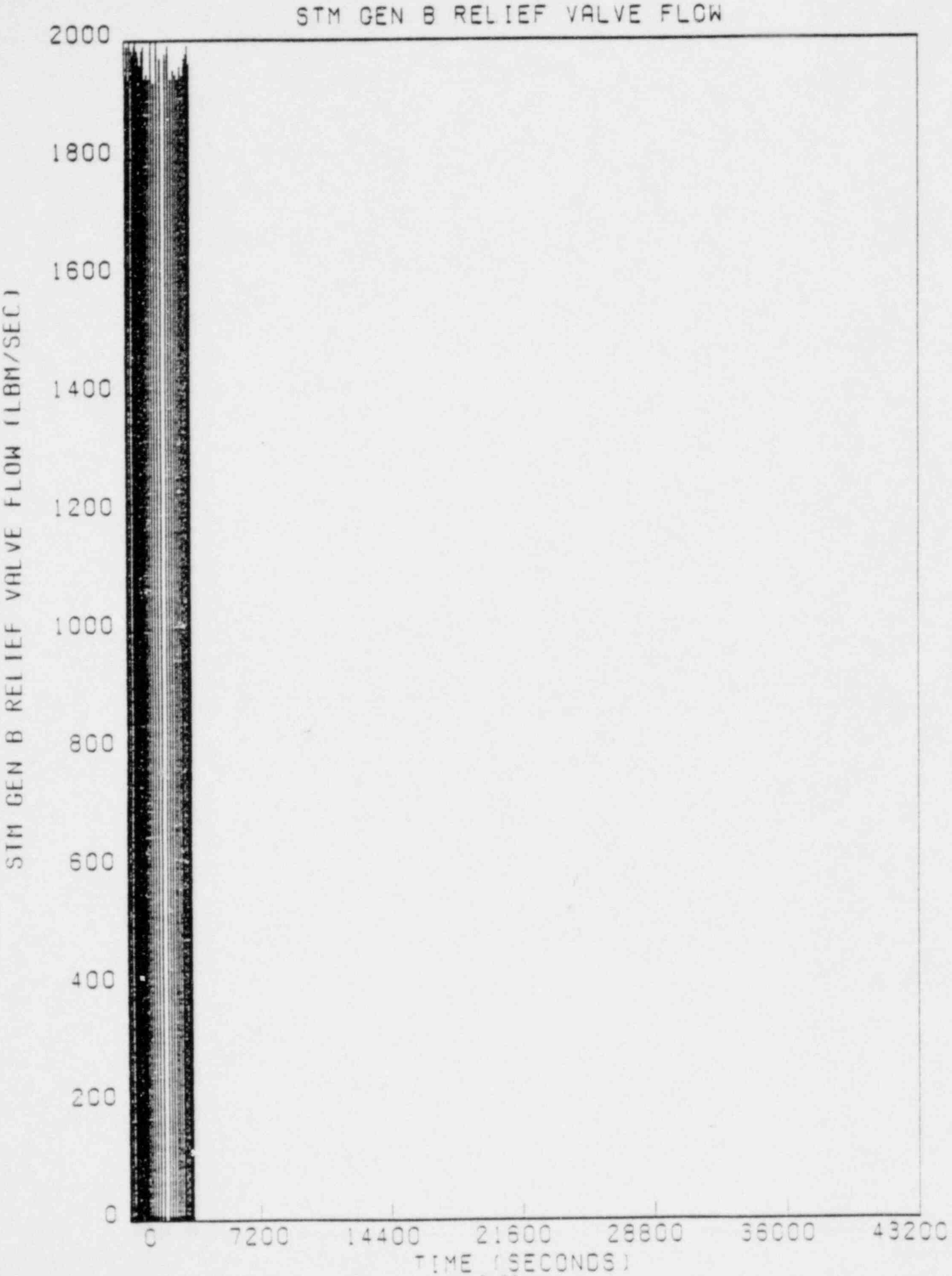
PLOT A-27


PRIMARY SAFETY VALVE FLOW
PRIMARY SAFETY VALVE FLOW (LBM/SEC)
500
450
400
350
300
250
200
150
100
50
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

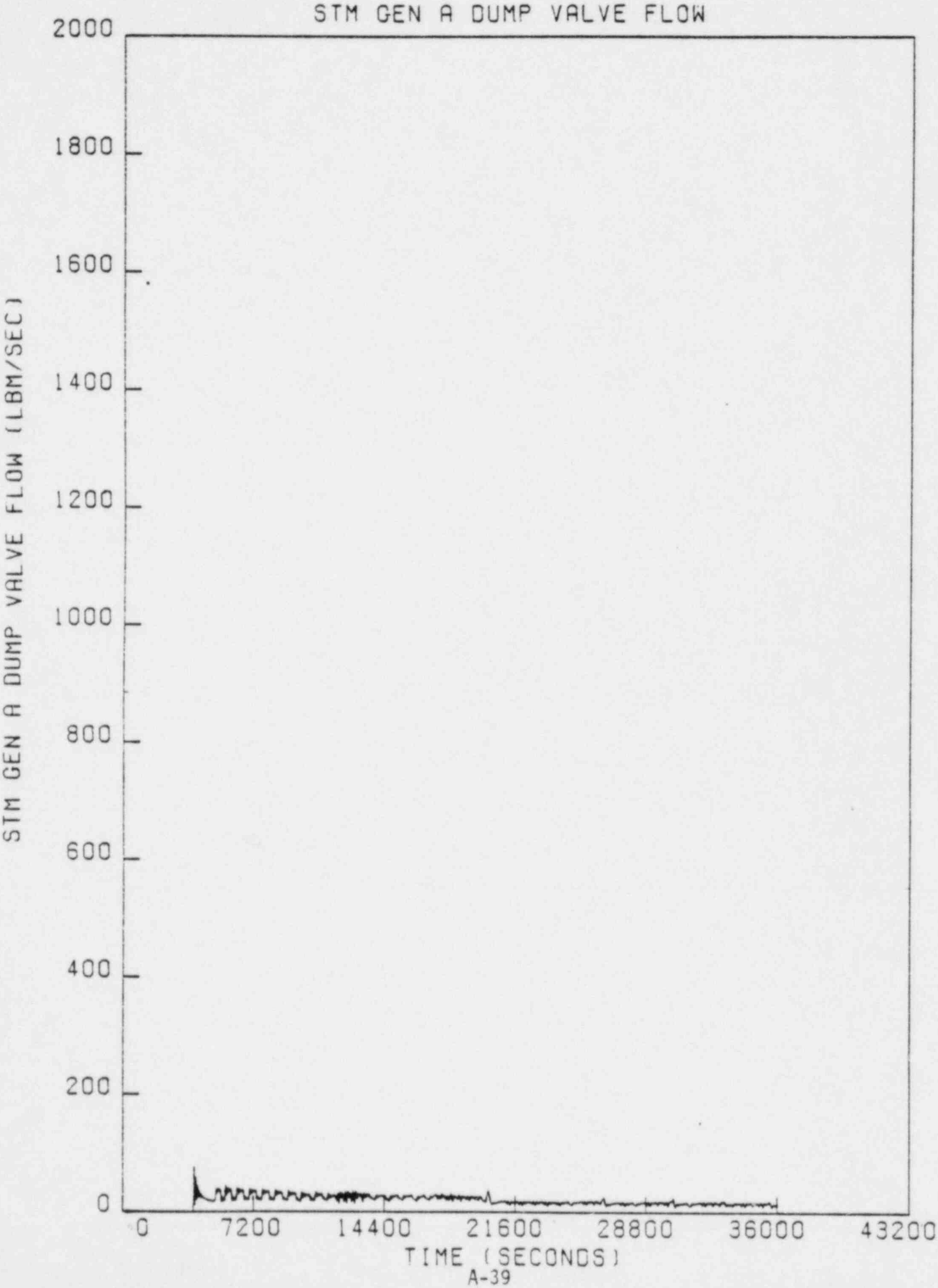
PLOT A-28


STM GEN A RELIEF VALVE FLOW
STM GEN A RELIEF VALVE FLOW (LBM/SEC)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

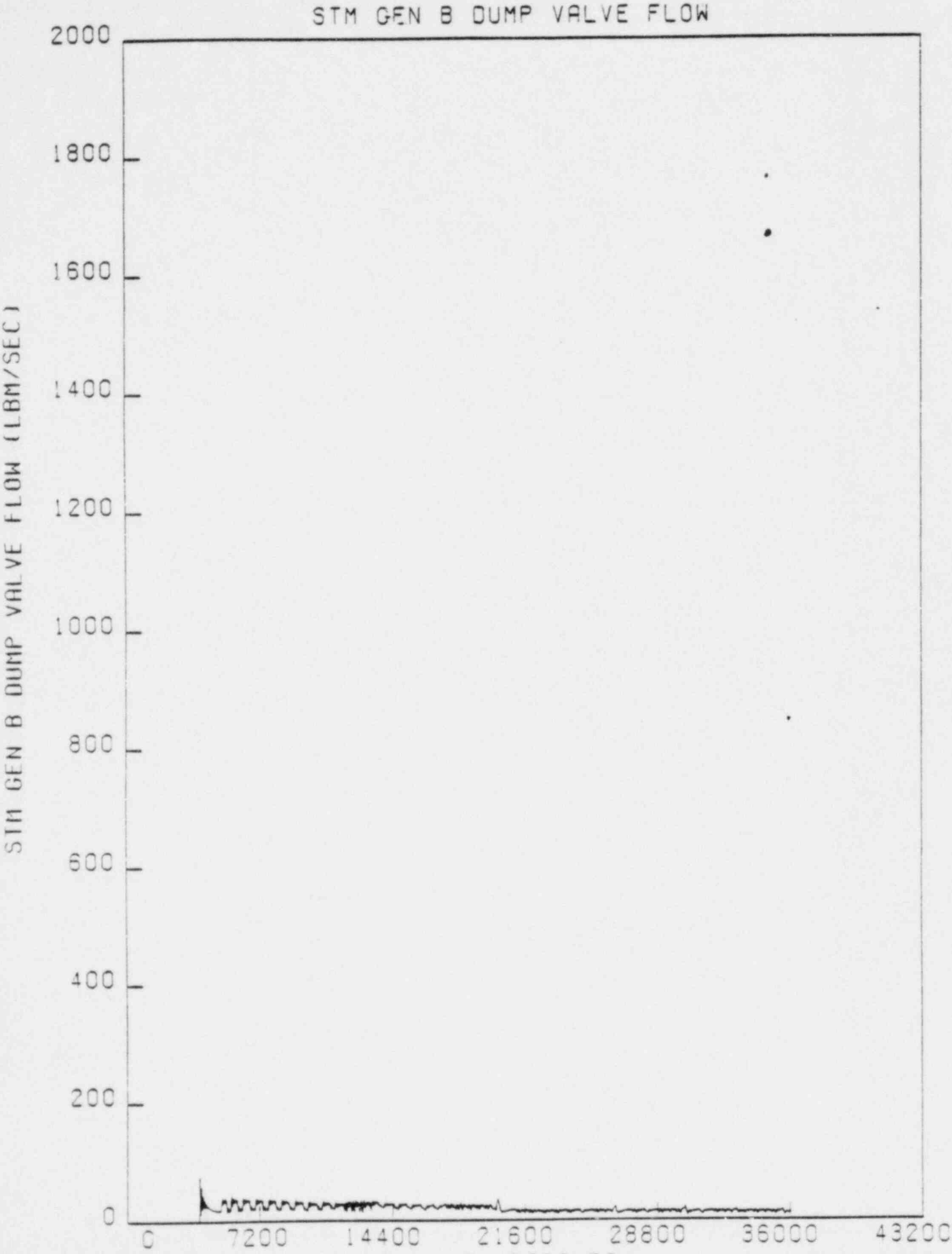
PLOT A-29


STM GEN B RELIEF VALVE FLOW
STM GEN B RELIEF VALVE FLOW (LBM/SEC)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

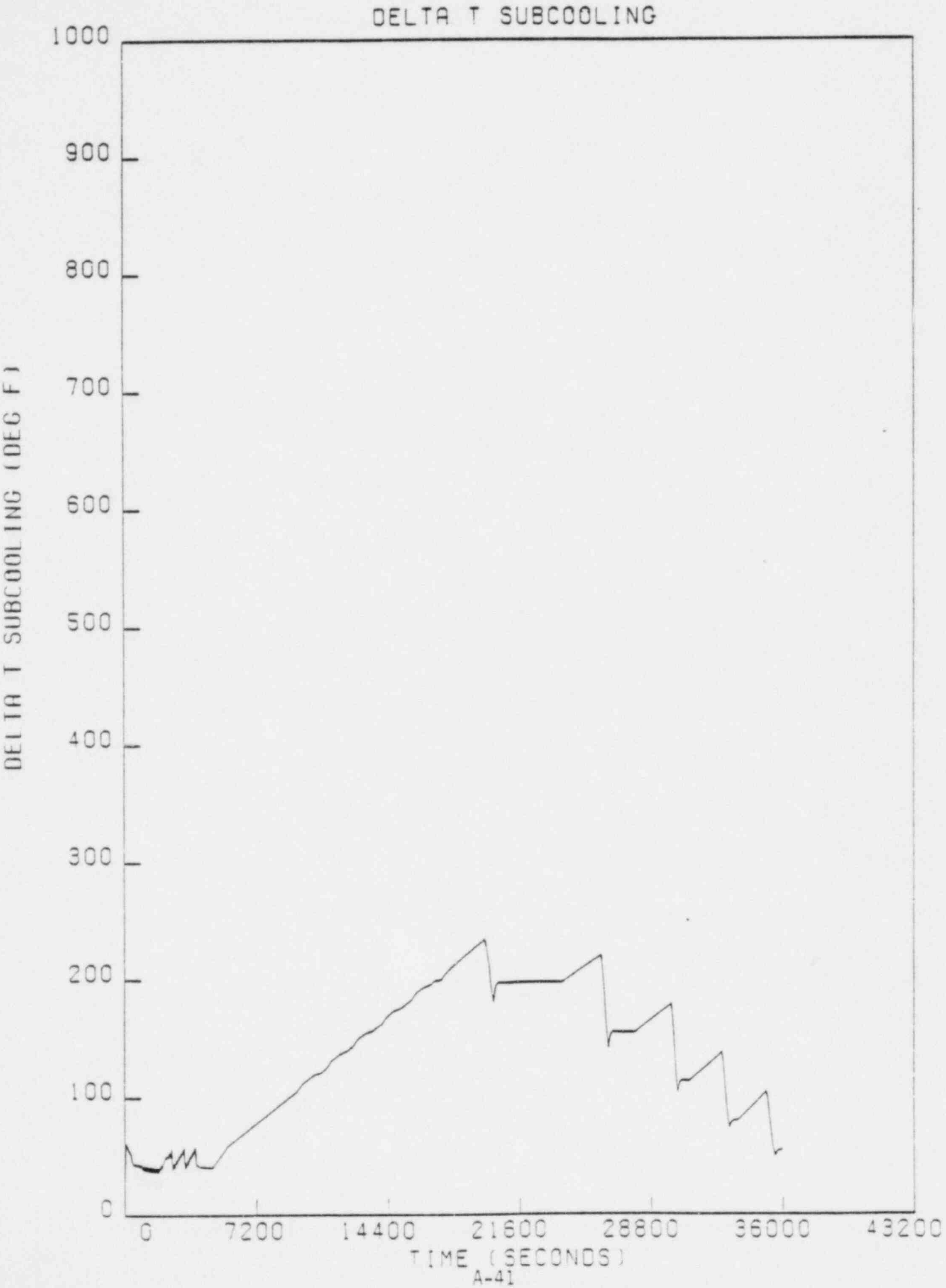
PLOT A-30


STM GEN A DUMP VALVE FLOW
STM GEN A DUMP VALVE FLOW (LBM/SEC)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

PLOT A-31


STM GEN B DUMP VALVE FLOW
STM GEN B DUMP VALVE FLOW (LBM/SEC)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)

PLOT A-32


DELTA T SUBCOOLING
1000
900
800
700
600
500
400
300
200
100
0
DELTA T SUBCOOLING (DEG F)
0
7200
14400
21600
28800
36000
43200
TIME (SECONDS)