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International Agreement Report

RELAP5/MOD2 Analysis of a Postulated "Cold Leg SBLOCA" Simultaneous to a "Total Black-Out" Event in the José Cabrera Nuclear Station

Prepared by
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Office of Nuclear Regulatory Research
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Prepared as part of
The Agreement on Research Participation and Technical Exchange
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Madrid, Spain.

Luis Rebollo

ICAP REPORT

RELAP5/MOD2 ANALYSIS OF A POSTULATED "COLD LEG SBLOCA" SIMULTANEOUS TO A "TOTAL BLACK-OUT" EVENT IN JOSE CABRERA NUCLEAR STATION AS AN APPLICATION OF 'LESSONS LEARNED' FROM OECD-LOFT LP-SB-3 EXPERIMENT. DEVELOPMENT OF A MITIGATION PROCEDURE.

ABSTRACT

Several beyond-design bases cold leg small-break LOCA postulated scenarios based on the "lessons learned" in the OECD-LOFT LP-SB-3 experiment have been analyzed for the Westinghouse single loop Jose Cabrera Nuclear Power Plant belonging to the Spanish utility UNION ELECTRICA FENOSA, S.A.

The analysis has been done by the utility in the Thermal-Hydraulic & Accident Analysis Section of the Enginneering Department of the Nuclear Division.

The RELAP5/MOD2/36.04 code has been used on a CYBER 180/830 computer and the simulation includes the 6" RHRS charging line, the 2" pressurizer spray, and the 1.5" CVCS make-up line piping breaks.

The assumption of a "total black-out condition" coincident with the occurrence of the event has been made in order to consider a plant degraded condition with total active failure of the ECCS.

As a result of the analysis, estimates of the "time to core overheating startup" as well as an evaluation of alternate operator measures to mitigate the consequences of the event have been obtained.

Finally a proposal for improving the LOCA emergency operating procedure (E-1) has been suggested.

TABLE OF CONTENTS

EXECUTIVE SUMMARY

A RELAP5/MOD2/36.04 simulation has been conducted to evaluate the capability of the code to calculate long SBLOCA transients under degraded conditions for a commercial nuclear power plant.

This is an "application" case of the ICAP program and no measured data exist to make an "assessment" because it is not a real but a postulated scenario.

A full scope nodalization for the plant has been developed in order to be able to analize different kind of transients. A stabilization system that initialize the model at the desired conditions of pressurizer pressure and level, primary coolant average temperature, primary coolant mass flow rate, and steam generator downcomer level, has been designed. This system, although does not correspond to a real one, is very usefull to get the transient initial condition in a fast way. Depending on the case this initial condition can correspond to the nominal one or can take into account some deviations corresponding to the uncertainty and measurement dead band of the pressure, average temperature and thermal power channels. Once the desired condition has been got, the user has to delete the artificial system and incorporate the real control system.

Based on the "lessons learned" in the analysis and simulation of the OECD-LOFT LP-SB-3 experiment, several calculations corresponding to cold leg SBLOCA and simultaneous total black-out have been made. The total electrical failure declares unavailable every active system in the plant, including the emergency safety features.

Size and location of the break correspond to the real small pipes connected to the cold leg of the primary system:

- RHRS charging line .... (6")
- Pressurizer spray line .... (2")
- CVCS make-up line .... (1.5")

The postulated cases assume some conservative boundary conditions in the simulation of the decay heat, heat losses, break discharge coefficient, automatic steam dump, core power distribution, and turbine driven pump operation.

A complete set of cases without operator intervention were analized in order to have an estimation of the time margins for core overheating startup.

As in LP-SB-3 experiment the manual steam generator bleed has been demonstrated to be a proper recovery action in order to force the accumulator to discharge. This was the operator intervention in the LP-SB-3 experiment.

For accumulator self-discharge cases (intermediate LOCA) or once the operator has force the accumulator discharge (small break LOCA) a special procedure to regenerate the accumulator has been suggested. This is possible in this plant because the accumulator is installed outside of the containment building.

The safe response of the plant after the application of this mitigation procedure has been demonstrated for the complete set of cases and so a proposal for improving the present emergency operating procedure has been suggested.

The code ran on a CYBER 180/810 with a CPU time to reactor time ratio of 83, and on a CYBER 180/830 with a ratio of 44. The maximum time step selected was 0.05 s. using the semi-implicit numeric scheme. All the cases ran without special difficulties.

The good design and sizing of the major components of the plant have been demonstrated in this beyond-licensing bases analysis. Time requirement for operator intervention is always higher than the maximum estimated credible duration estimated of the "total black-out" event.

As a conclusion of the analysis the RELAP5/MOD2 code has been demonstrated to be appropriate to cover this kind of long term degraded transients. The suggested improvement of the present emergency operating procedure to mitigate the consequences of such an unlikely event has been demonstrated to be good enough to guarantee the safety of the plant.

FOREWORD

This report represents one of the assessment/application calculations submitted in fulfilment of the bilateral - agreement for cooperation in thermalhydraulic activities between the Consejo de Seguridad Nuclear of Spain (CSN) and the United States Nuclear Regulatoy Commission (US-NRC) in - the form of Spanish contribution to the International Code Assessment and Applications Program (ICAP) of the US-NRC whose main purpose is the validation of the TRAC and RELAP system codes.

The Consejo de Seguridad Nuclear has promoted a coordinated - Spanish Nuclear Industry effort (ICAP-SPAIN) aiming to - satisfy the requirements of this agreement and to improve the quality of the technical support groups at the Spanish - Utilities, Spanish Research Establishments, Regulatory Staff and Engineering Companies, for safety purposes.

This ICAP-SPAIN national program includes agreements between CSN and each of the following organizations:

- Unidad Eléctrica (UNESA)
- Unión Iberoamericana de Tecnología Eléctrica (UITESA)
- Empresa Nacional del Uranio (ENUSA)
- TECNATOM
- LOFT-ESPAÑA

The program is executed by 12 working groups and a generic code review group and is coordinated by the "Comité de Coordinación". This committee has approved the distribution of this document - for ICAP purposes.

1. **INTRODUCTION.**

The International Thermal-Hydraulic Code Assessment and Applications Program (ICAP) is being carried out by a number of countries and organizations and coordinated by the U.S. Nuclear Regulatory Commision (USNRC). Its purpose is to support the effort to obtain a considered view of the accuracy and validity of USNRC thermal-hydraulic codes over their range of applicability.

Contributions to ICAP of the Spanish utilities relate both to TRAC (PF1 and BD1) and RELAP5/MOD2, and include code assessments as well as code applications. Assessments will be done by comparing code results versus measured data in Spanish commercial power plants. Applications will be done for the old plants in wich the measurement recording system is not appropriate for an assessment comparison.

This report provides a summary of the principal research results of the OECD-LOFT LP-SB-3 scenario simulated for a commercial nuclear power station.

The major objective was to evaluate the performance of the plant under a postulated "cold leg SBLOCA" simultaneous to a "total black-out" event and to check the effectiveness of operator actions as steam generator bleed and accumulator refilling as a method to recover the degraded condition of a commercial PWR.

The participation in the international "Comparison Report for the OECD-LOFT LP-SB-3 experiment simulation" (Ref. 1 to 4) as well as the availability of the RELAP5/MOD2/36.04 code (Ref. 5), in the frame of the ICAP program, for the safety analysis of nuclear power plants encourage us to perform this application as part of a plant safety review based on the last-generation best-estimate codes.

The main objectives of the analysis were:

- to identify the available time margins for "core heatup startup",
- to get an estimation of the steam generator "bleed" effectiveness,
- to improve the knowledge of the phenomena associated to SBLOCA events and station behaviour under degraded conditions.

A description of the plant and the postulated transient is given in section 2. The nodalization is described in section 3 and the steady state calculation is reviewed in section 4. Transient results both without and with operator intervention as well as a proposal for improving the present LOCA emergency operating procedure is outlined in section 5. Run statistics are summarized in section 6 and the conclusions are given in section 6.

2. PLANT AND TRANSIENT DESCRIPTION.

2.1 NUCLEAR STATION DESCRIPTION.

The analysis was done for José Cabrera Nuclear Station (Fig. 2.1), a Westinghouse PWR Spanish commercial plant belonging to UNION ELECTRICA FENOSA, S.A. utility (Ref. 6). The plant had its first criticallity in 1968 and was the first nuclear station connected to the Spanish electrical grid.

As LOFT facility (Fig. 2.2) it has only one loop that includes a cold leg, reactor pressure vessel, hot leg, pressurizer, steam generator tubes, cross-over leg, and circulation pump. CVCS and RHRS systems are also connected to the reactor coolant loop.

Nominal reactor power is 510 thermal Mw representing a scale factor of 10 versus LOFT. The reactor core has a loading pattern of 69 fuel assemblies (14 x 14) with 2.40 m. of active lenght. Reload fuel has an average enrichment of 3.40 % in U-235. The plant nominal output electrical power is 160 Mwe.

The ECCS system connects directly to the downcomer of the reactor vessel and includes one accumulator, two intermediate pressure safety injection pumps taking borated water from the reload water storage tank, and two recirculation pumps and an ejector taking water from the containment sump and feeding the injection pumps in the recirculation phase of a LOCA.

In the secondary side the typical BOP components are included (two 50% main feedwater pumps, steam line, safety and relief valves, main steam isolation valve, turbine trip valve, main steam control valve , turbine, condenser, heaters, etc).

Main feedwater comes directly into the upper part of the downcomer without passing through any preheater section in the steam generator, being previously heated in four heaters installed between the condenser and the steam generator. The circulation ratio in the secondary side of the steam generator is 3.34 at full power.

The auxiliary-emergency feedwater system includes one turbine driven and two motor operated pumps. Both systems take cold water from a tank and start their operation automatically. The turbine driven subsystem injects into the upper part of the downcomer requiring from the operator the allowance to inject by opening an isolation valve. The motor operated subsystem injects into the lower part of the downcomer. No operator action is necessary to allow the system to inject into the steam generator. There is also the posibility of an optional injection into the upper part once the operator lineup properly the system.

The plant operates normally in automatic mode under de influence of the reactor control system that maintains the programmed coolant average temperature in the primary system acting on the control rod position. Reactivity changes due to fuel burnup are compensated through the cycle by the manual operator reduction of the boron concentration in the primary coolant.

The safety of the plant is guaranteed by the reactor protection system and the emergency safety features.

2.2 OECD-LOFT LP-SB-3 SCENARIO.

The postulated scenario was based on OECD-LOFT LP-SB-3 experiment (Ref. 1), conducted on March 5, 1984, that simulated a small break in the cold leg of a commercial 3 loop Westinghouse PWR without high pressure pumped ECC injection. The scenario was intended to cause an inadequate core cooling condition with a resulting fuel cladding temperature excursion. The transient was designed to be long enough to require operator intervention to recover the plant because the normal engineered safety features would be ineffective.

For a small break loss of coolant accident in a PWR in which the HPIS becomes unavailable and inadequate core cooling conditions result, a potential method of recovery consists of an operator-initiated steam generator cooldown. In this method, the operator actuates the plant's secondary steam relief capability and initiates auxiliary feedwater flow to the steam generators.

This action causes a very rapid cooldown and depressurization of the secondary coolant system, which in turn causes the primary coolant system to cool and depressurize due to the thermal connection through the U-tubes.

The operator continues this process, known as "steam generator feed and bleed", until the primary coolant system pressure has reached the point where the low pressure ECCS (accumulators and LPIS) can be used to cool the core. This is the first time in which the effectiveness of this recovery method in an integral test facility has been demonstrated.

The experiment was designed to produce the system conditions in LOFT which would achieve the following experiment objectives:

- Investigate core heat transfer characteristics when core uncovery occurs during relatively slow boil-off conditions at primary system pressures above the normal accumulator setpoint.

- Evaluate the effectiveness of the steam generator "feed and bleed" as a means of depressurizing and cooling a highly voided primary system at a high primary system pressure.

- Evaluate the effectiveness of accumulator injection in establishing core cooling in a highly voided system when a low pressure differential exists between the accumulator and the primary system.

- Provide data to assess the capability and conservatism of computer codes to predict the transient response of a long-term loss-of-coolant scenario.

The post-test analysis of the experiment (Ref. 2,3,4) demonstrates both the proper recovery of the facility and the excellent performance of the RELAP5 code.

2.3 SB-3 SCENARIO POSTULATED FOR THE NUCLEAR STATION.

Due to the redundancy and diversity of the electrical supply to the motors of the emergency safety features, the coincidence of a SBLOCA and the unavailability of the ECCS injection pumps requires in José Cabrera plant the simultaneity of the SBLOCA and a "total black-out" condition.

In loss-of-offsite power condition the diverse and redundant emergency power supply system guarantees the operation of the safety injection and emergency feedwater pumps. The only possibility for such a failure in the redundant SIS and EFWS is the external black-out with failure under demand in the emergency power supply system that includes two hydroelectric turbines and one Diesel generator. The probability of such an electric failure has been estimated as 4.7E-06 per year with a maximum credible duration of 20 minutes for the postulated "total black-out" condition.

In such an event the loss of power supply to the motors of the pumps forces the coastdown of the primary coolant flow and main feedwater flow.

The reactor would be directly tripped on RCP breakers opening due to black-out. The loss of offsite power supply would also directly trip the MFWS wich in turn would trip the turbine and would force a reactor trip on turbine trip.
Low reactor coolant flow, mismatch between feedwater and steam flow simultaneous to low steam generator downcomer level (NR), variable low pressurizer pressure, fixed low pressurizer pressure, low-low steam generator level (NR), and "S" signal on low pressurizer pressure would trip the reactor wich in turn would force the turbine trip.

The emergency safety features would be required as follows:

- ECCS pumps would be started on "S" signal due to low pressurizer pressure.

- EFWS pumps would be started on low-low level in the narrow range of the steam generator downcomer or "S" signal. Isolation valves would be opened on low level in the wide range in coincidence with reactor trip.

In the case of a "total black-out" condition both systems would be unavailable and the behavior of the plant would correspond to the SB-3 experiment in the LOFT facility.

The only difference would be that in the LOFT experiment the reactor coolant pumps were running during the first 1600 seconds while in the power plant simulation the reactor coolant pump trips due to black-out condition. Reactor coolant pump would also be tripped by the operator following the recommendations of the emergency operating instruction for LOCA (E-1).

The analysis intends to identify the time the operator has in order to initiate a recovery before a core heatup transient starts. It will also be analized if there is enough steam generator secondary side inventory in order to "bleed" the system.

The standard "feed and bleed" procedure used in LOFT would be reduced to the "bleed" action taken into account that the "feed" operation would not be possible in case of "total black-out" event because the power operated pumps (main and auxiliary-emergency feedwater systems) would not be active.

Due to conservative reasons, no credit is given to the existing turbine driven pump of the emergency feedwater supply system that would be available to "feed" the steam generator and guarantee its inventory.

The accumulator of the ECCS will be considered as available due to the fact that it is a passive element.

3. CODE INPUT MODEL DESCRIPTION.

For this application, the RELAP5/MOD2/36.04 code on a CYBER 180/830 computer under NOS 2.5 operating system has been used. NEW, RESTART and STRIP modes of operation were used for the steady state, transient and plotter applications respectively.

As the DISSPLA plotter package was not available in the company, the reading-writing POSTRIP (post-STRIP) program has been developed. This program reads from the "strip file" and writes a file adapted to the input of GRAPHS (a general purpose plotter program).

The RELAP5 model of José Cabrera nuclear steam supply system depicted in Fig. 3.1 is currently been used in the transient analysis of the nuclear power plant. It is a general purpose model developed specifically for José Cabrera nuclear station in order to have a tool to allow the utility to make its own safety analysis.

The nodalization comprised 124 control volumes or nodes, 15 of which are time dependent volumes, 133 junctions and 63 heat structures.

A transient and accident analysis methodology adapted to the use of the code, including enginneering procedures and simulation rules, has also been developed.

3.1 PRIMARY SYSTEM NODALIZATION.

The reactor core was divided into eigh vertical nodes: a six nodes pipe (209) representing the active core and two unheated inlet (211) and outlet (207) nodes respectively. The upper plenum (206) collects coolant coming from the core, from the core bypass (210) and from the head of the vessel.

The vessel has a lower (four nodes annulus 208) and an upper (204, 202) downcomer, a lower (212) and an upper (201) dome, and an upper plenum (203, 205). Three bypass ways for the coolant have been considered: core bypass (from 210 to 206), vessel head bypass (from 205 to 206), and cold leg - hot leg bypass (from 208 to 100). By applying appropriate loss coefficients, the specified flow distribution between core flow and each bypass flow was met.

The hot leg was divided into two nodes (100 and 105), the junction of wich corresponds to the surge line (three nodes pipe 300) connection.

The pressurizer was modeled by two pipe components: the two nodes upper one (312) and the six nodes lower one (310), the connection of wich corresponds to the spray junction (from 354 to 310). This nodalization was chosen in orden to allow the model to introduce coolant spray from the pump discharge (150) through the spray line (three nodes pipe 350, single volume 354) directly into the steam volume under the influence of the modulation of the spray control valve (352).

The pressurizer relief lines (322, 326), valves (324, 328) and collector (330), as well as the safety lines (314, 318) and valves (316, 320) have been simulated. The four nodes pipe common safety-relief collector (332) directs the steam discharges into the pressurizer relief tank that was simulated as a couple of volumes; the bottom one corresponds to the water part (334) and the top one corresponds to the steam-nitrogen part (336). The rupture disc was simulated by a valve (338) having the disc real section and an opening set-point equal to its rupture pressure. This valve allows the discharge of steam directly to the containment atmosphere simulated as a boundary condition (time dependent volume 340).

The primary side of the steam generator was modeled with an inlet plenum (110), the portion of the tubes in the "up" direction inside the tubeplate (115), the eight nodes pipe (120) representing the U-tubes, the portion of tubes in the "down" direction inside the tubeplate (125), and the outlet plenum (130).

The loop-seal between the steam generator outlet and the pump suction was simulated with three volumes corresponding to the "down" part (140), the "horizontal" part (142) and the "up" part (144).

The reactor coolant pump (150) was represented using the homologous curves specific for this plant obtained from Westinghouse. Two-phase factors from LOFT facility were used to simulate the degraded behavior under abnormal conditions of void fraction as an application of the conclusions of Ref. 7.

The cold leg leading from the pump discharge to the vessel inlet was represented by a couple of nodes (160, 165).

By applying appropriate loss coefficients, the specified loop pressure distribution and flow were achieved.

The emergency core cooling system was simulated by a couple of subsystems. The passive subsystem includes the accumulator (600), the discharge line (605), the isolation valve (610), the three nodes pipe discharge line (620), and the check valve (630). By tunning appropriated coefficients the referenced Westinghouse accumulator discharge flow rate, under LBLOCA conditions, was met. The active subsystem corresponds to the safety injection pumps, modeled as a time dependent junction (655) taking borated water from the reload water storage tank (time dependent volume 650) as a boundary condition. The injection flow has been defined as a function of the primary system back-pressure with conservative assumptions for the line pressure losses.

Heat structures for the accumulator, vessel, core with average and hot channels, hot leg, surge line, pressurizer, steam generator plena, U-tubes, loop seal, pump, spray line and cold leg, have been simulated. In the case of the steam generator plena, three different heat structures have been considered: one connecting each plenum with the containment atmosphere, one connecting both plena, and one connecting each plenum with the riser, simulating the tubeplate thermal structure.

The point kinetic model, including fuel temperature, coolant temperature and coolant density feed-back reactivity effects, has been selected for the active heat structures of the reactor core.

The reactor control and protection systems based on the functional diagrams corresponding to the real gains and delays measured in the power station have also been simulated.

Several control variables have been defined to calculate:

a) Pressurizer collapsed liquid level.
   Error in the pressurizer collapsed liquid level.
   Correction to the pressurizer level controller.
   Modulated make-up & let-down mass flow rate.

b) Reactor coolant system average temperature.
   Error in the reactor coolant system average temperature.
   Correction to the RCS average temperature controller.
   Modulated position of the steam control valve.

c) Cold leg volumetric flow rate.
   Error in the cold leg volumetric flow rate.
   Correction to the RCS flow controller.
   Modulated reactor coolant pump speed.

d) Steam generator downcomer collapsed liquid level.
   Error in the steam generator downcomer level.
   Correction to the steam generator downcomer level controller.
   Modulated feedwater mass flow rate.

e) Circulation rate in the secondary side of the steam generator.
   Steam generator riser collapsed liquid level.
   Total steam generator relief flow to the atmosphere.
   Total steam generator relief flow.
   Total steam generator safety flow.
   Total flow from the steam generator to the atmosphere.
   Integral discharge from the steam generator to the atmosphere.
   Turbine driven AFW steam consumption.
   Turbine driven AFW mass flow rate.
   Mismatch in the secondary system.

f) Core collapsed liquid level.
   Total ECCS mass flow rate.
   Integral accumulator mass flow rate.
   Integral safety injection mass flow rate.
   Integral ECCS mass flow rate.
   RCS delta temperature.

g) Reactor protection system functions.
   Reactor control system functions.

h) CPU rate (CPU time/Reactor time).

3.2 SECONDARY SYSTEM NODALIZATION.

Feedwater was simulated as a time dependent juntion (445), connected to the upper part of the downcomer, taking warm water from the time dependent volume (444). Feedwater temperature was simulated as a function of plant power from full power and part load operational data.

Turbine driven auxiliary feedwater pump was represented as a time dependent junction (449), connected to the upper part of the downcomer, taking cold water from a constant temperature time dependent volume (448).

Emergency feedwater motor pumps were simulated as a time dependent junction (457), connected to the lower part of the downcomer, taking cold water from a constant temperature time dependent volume (456).

The steam generator downcomer was simulated by a five nodes annulus (450). The single junction (455) connects the downcomer bottom to the riser inlet. The riser was represented by a five nodes pipe (400) with the same elevations as their corresponding in the downcomer. The first four are thermally connected to the primary system through the U-tube heat structure.

A non-ideal but nearly-real first separator (410) was simulated at the top of the riser with special detail in the carry-over and carry-under flow characteristics. To do that, a geometrical analysis of the real dimensions of the ciclonic pathways in the separators has been done obtaining the values of the VOVER (carry-over) and VUNDER (carry-under) parameters for the RELAP5 separator model. A connection (from 410 to 450) representing the separator draining paths has been provided. The separator bypass (440), connecting the downcomer and the steam dome, has been simulated.

By applying appropriate loss coefficients in the natural circulation loop of the steam generator (400, 410, 450), with the higest resistance located in the downcomer/riser junction (455), the specified circulation ratio of 3.34 has been met. Also, by adjusting the secondary side liquid inventory, the measured downcomer level has been obtained.

The steam node (420) corresponds to the volume between the ciclonic separator and the steam dryer. The dryer was simulated as a nearly-ideal second separator (424) which allowed nearly-only steam to escape upwards. The drain flow path (426) represents the real pipes that connect the steam dryer to the top of the downcomer.

The steam volume at the top of the steam generator dome has at its bottom an orifice plate that behaves as a separator and so it has been simulated as an ideal third separator (430) allowing only steam to escape upwards. The drain flow path (428) represents the real pipes that connect the orifice plate to the top of the downcomer.

This special emphasis in the simulation of the three separator stages, including the real definition of the draining ways, were considered to be important in the analysis of depressurizations of the secondary system due to steam line breaks. These draining pipes behave in such an event as a riser bypass leakage pathways for the inventory of the steam generator that leaves the downcomer without any cooling effect on the primary coolant through the riser/U-tubes thermal connection.

The steam line was divided in several parts: the four nodes pipe (500), two single nodes (502, 504), the main steam isolation valve (506), a single node (508), a three nodes pipe (510), the main steam control valve (512), a single node (513), the turbine trip valve (514) and the time dependent volume (516) that represents the turbine.

The turbine was simulated as a boundary condition selecting its constant back-pressure high enough to avoid critical flow in the main steam line valves.

The real characteristics and actuation logic of each valve have been modeled. Also, by using appropriate loss coefficients, the specified secondary pressure distribution was met.

The model includes the four safety valves (540, 544, 548 and 552) as well as their relief lines (542, 546, 550 and 554) to the environmental atmosphere simulated as constant time dependent volumes (560, 561, 562 and 563).

The steam consumption of the turbine driven pump have been simulated by time dependent junctions discharging to the environmental atmosphere (time dependent volumes 460 and 462) for the turbining (459) and turbining-pumping (461) modes of operation.

The automatic steam-dump system operates modulating the opening of the relief valve (532) to the condenser (time dependent volume 538) through the relief line (530, 534) and the opening of relief valves (522 and 526) to the environmental atmosphere (time dependent volumes 564 and 565) through their relief lines (524, 528), looking for the RCS hot zero power average temperature. There is a common relief line (520) to the atmosphere and a general common relief line (518) from the main steam line. The valve (536) isolates the relief line to the condenser in case of loss of offsite power, protecting the condenser that results unavailable under this circunstance.

Heat structures for the steam generator vessel and internals have been simulated.

A sensitivity calculation tunning the hydraulic diameter of the steam generator U-tubes/riser heat structure was done to fit the pressure in the secondary side. The explanation for needing this correction can be found in the substantial amount of crossflow created by the U-tubes support plates in the riser. The crossflow enhances the heat transfer considerably and is not taken into account in the ordinary heat transfer correlations.

The tunned hydraulic diameter corresponds to a value similar to the gap between tubes. This value was only used in the definition of the U-tubes/riser heat structure, maintaining the real geometric value for the hydraulic definition of the riser.

A summary description of the model including concept, node number and type is given in Table 3.1.

The main applications of the model are : FSAR accident analysis review, real plant transients simulation, evaluation of potencial changes in Technical Specifications, plant personnel training, support to PSA analysis, etc.

4. STEADY STATE CALCULATION.

The first step was to get a steady state condition representing the normal operation of the plant at full power. The aim was to get the desired stable condition with the minimum CPU time consumption. The simulated control system, that reproduces the real characteristics of the system in the plant, behaved slowly and was not considered the more appropriate for getting the steady state.

For this objective a stabilization system has been developed according to the logic of actuation represented in Table 4.1. In this way pressurizer pressure and level, primary coolant flow rate, primary average temperature and steam generator pressure and downcomer level are fitted to the values measured in the plant.

This stabilization system is based on proportional-integral controllers that, with the use of properly selected gains, fits the system reducing to zero, as fastly and stably as possible, the "error signal" defined as the difference between the desired and the calculated value of the controlled variable.

Reactor power and feedwater temperature are maintained constant as a boundary condition representing the nominal value corresponding to the normal operating condition. For this calculation the kinetic model was not used.

After a 346.5 s. reactor time null transient a stable condition for the controlled model has been got. The final state fits the desired full power measured operating conditions. The RELAP5/MOD2 code stopped the steady state calculation once the stability condition was accepted by its internal checking procedure. Main results are represented in Fig. SS.1 to SS.13 of Appendix SS as indicated in Table 4.2.

A second step eliminating the stabilization system and introducing the reactor kinetic model and the real reactor control and protection system was done getting inmediately the stable condition.

As a first application of the model, the simulation of full load and partial load (20% to 100%) conditions has been made. In general the results show a very good reproduction of the measures in the plant. Table 4.3 gives a summary of the comparison between the main variables measured and calculated in the steady state simulation at full power.

5. TRANSIENT CALCULATION.

For this SBLOCA application the general nodalization could have been simplified based on enginneering judgement eliminating those nodes that are considered irrelevant in a depressurization transient:

- Pressurizer relief and safety valves. (316, 320, 324, 328)
- Pressurizer relief lines. (314, 318, 322, 326, 330, 332)
- Pressurizer relief tank. (334, 336, 338, 340)

Elements neglected by the hypothesis of the transient analysis could also have been deleted:

- Safety injection pumps. (650, 655)
- EFWS turbine-driven and motor pumps. (448, 449, 456, 457, 459, 460, 461, 462)
- S.G. relief line to the condenser. (530, 532, 534, 536, 538)

However, in order to check the proper behavior of each component, the described standard nodalization has been used. Appropriate changes in the logic of actuation to avoid their automatic operation have been provided.

5.1 HYPOTHESIS OF THE TRANSIENT ANALYSIS.

A conservative analysis of the SB-3 scenario was done based on a best-estimate code and conservative transient boundary conditions as follows:

a) Decay heat generation rate.

ANS-5.1 (Oct-73) with a multiplier factor of 1.20 was used.

b) Heat losses.

No credit for the system-to-containment heat losses was considered. The primary coolant is the only heat sink for the primary thermal structures.

c) Break.

A guillotine break in a small pipe connected to the cold leg of the primary system with a discharge coefficient of 1.0 was simulated as a trip valve (163) connecting the 160-165 junction to the containment atmosphere (164) simulated as a constant pressure time dependent volume.

d) **Steam** dump system.

Due to conservativa reasons, no credit for the actuation of the automatic steam dump relief system from the secondary side to the atmosphere was considered and only safety valves were allowed to operate. This system would actuate automatically and would cool the primary system reducing the average temperature to the hot zero power set point.

e) Core power distribution.

An axial power distribution peaked at the top of the core with a peak to average factor of 1.77 and an axial offset of + 40% was simulated both in steady state and transient conditions.

f) Turbine driven pump.

Its operation would guarantee the inventory of the steam generator by controlling the downcomer level even under "black-out" condition. However no credit is given to this available subsystem of the AFWS because operator intervention to open an isolation valve is required. Due to the fact that this system is automatically activated, the steam consumption corresponding to the turbine operation turbining and not pumping has been considered.

g) Black-out.

A "total black-out" coincident with the break was considered. This forced the coastdown of the reactor coolant and main feedwater pumps. Power operated emergency safety features (ECCS and EFWS) were considered for this reason as unavailable.

Hypothesis "a" and "b" increase the heat source while "c" and "e" accelerate the core uncovery.

Hypothesis "d" increases secondary temperature due to the higher set-point of the secondary system safety valves in comparison with the relief valves. As a consequence of that the primary temperature is conservatively overpredicted.

Hypothesis "f" reduces the estimation of the steam generator secondary side inventory and downcomer level.

Hypothesis "g" has been done to reproduce the LOFT-SB-3 scenario of SBLOCA without ECCS pumps operation.

5.2 <u>TRANSIENT BEHAVIOUR WITHOUT OPERATOR INTERVENTION.</u> <u>(A)</u>

Three small break LOCA's corresponding to the three lines connected to the cold leg of the primary coolant system have been analized.

The corresponding break size for the postulated cases are:

a) case I : 6 inch diameter for the RHRS line.

b) case II : 2 inch diameter for the spray line.

c) case III: 1.5 inch diameter for the CVCS line.

A simulation without any operator intervention has been done in order to analize the passive behaviour of the plant for such kind of transients. Corresponding figures without operation intervention are identified in Appendix I, II, and III, with the letter <u>"A"</u> in brakets. The set of figures for each transient case is listed in Table 5.1. Variable identifications are listed in Table 5.2.

There are some common features for the three cases:

- depressurization of the primary system,
- automatic reactor trip changing from nominal power to decay heat generation,
- automatic turbine trip on reactor trip signal,
- "S" signal on low pressurizer pressure,
- primary coolant pump and main feedwater pump coastdown,
- subcooled, two phase and only steam discharge through the break,
- loss of inventory in the primary system with reduction in the core collapsed liquid level,
- core level recovery due to loop seal clearance,
- pressurization in the secondary system,
- void redistribution in the secondary side of the steam generator with reduction in the downcomer collapsed liquid level,
- core heat-up.

However there are some differences that should be pointed out:

a) <u>Case I(A), (6" RHRS)</u> :

The size of the break is big enough to depressurize directly the system under the accumulator set-point (Fig. 2). In fact, it cannot be considered as a small break but an intermediate one taking into account that the steam generator is not needed as heat sink during the transient.

The chronology of events is summarized in Table 5.3. and the main results are represented in the set of figures of Appendix I.

The scenario does not correspond to the specification of the experiment LP-SB-3 but the analysis was continued in order to identify the margin for core uncovery and heat-up.

The accumulator discharge starts at t = 190. s. (Fig. 7), initiating the recovery of inventory in the primary system (Fig. 9). Once the accumulator discharge finishes, a continuous mass depletion is predicted (Fig. 9) with the simultaneous reduction in the core collapsed liquid level (Fig. 19) that is responsible for the initiation of the cladding temperature excursion at t = 1480. s. (Fig. 26).

The potential reduction of the U-tubes heat transfer due to nitrogen discharge from the accumulator to the reactor coolant system was observed not to be very important in this case considering that the steam generator is not needed as heat sink during the transient.

In the secondary side a pressure increase without reaching the safety valves set-point is predicted (Fig. 2).

As there is not steam discharge through the safety valves to the atmosphere (Fig. 5, 6), the steam generator maintains its inventory (Fig. 13) and behaves as a heat source instead of a heat sink. This effect is clearly observed once the reactor coolant system depressurizes under the steam generator pressure (Fig. 2) when the decay heat generation rate (Fig. 1) becomes smaller than the energy release through the break.

b) Case II(A), (2" spray line) :

Its behaviour is similar to the observed in the experiment LP-SB-3, a stable pressure in the primary system (Fig. 2) above the accumulator set-point, coincident with a monotonic mass depletion (Fig. 9), being achieved.

The chronology of events is summarized in Table 5.4, and the main results are represented in the set of figures of Appendix II.

The pressure in the secondary system (Fig. 2) reaches the safety valve set-point at t = 110. s. From this time on the valve cycles (Fig. 5) in order to compensate the difference between the decay heat generation rate and the energy release through the break. Steam release from the steam generator to the atmosphere (Fig. 6) reduces continuously the steam generator inventory and the downcomer collapsed liquid level (Fig. 13).

At t = 1370. s. the safety valve stops cycling due to the fact that the decay heat generation rate (Fig. 1) becomes smaller than the energy release rate through the break. From this time on, a constant steam generator inventory is observed (Fig. 13) and a continuous cooldown (Fig. 22) and depressurization (Fig. 2) of the primary system is predicted driving the plant to the accumulator discharge set-point without any operator intervention.

The core heat-up begins at t = 1950. s. (Fig. 26) due to low core level (Fig. 19). This happens before the accumulator discharges. The depressurization continues and reaches the accumulator set-point at t = 2320. s.

A conservative analysis of the core heat-up without credit for the accumulator discharge has been done to analize the transient temperature of the cladding.

c) Case III(A), (1.5" CVCS) :

The primary system behaviour is similar to case II but slower.The chronology of events is summarized in Table 5.5. and the main results are represented in the set of figures of Appendix III.

A continuous discharge of steam from the safety valve of the secondary side (Fig. 5, 6) is predicted during the first 2580. s. From this time on the steam generator is not needed as heat sink and the safety valves remains closed because the energy release through the break is higher than the heat source from the core to the coolant.

At t = 2980. s the core begins to heat-up (Fig. 26) reducing the heat transfer coefficient to the coolant which, consequently, starts to cool (Fig. 22) and depressurize (Fig. 2), as a result of its energy reduction as a consecuence of the smaller core/coolant heat transfer with constant heat release through the break.

The accumulator set-point is reached at t = 3910. s. (Fig 7, 9). In this case the accumulator discharge has been considered looking for a realistic simulation of the transient cladding temperature during this event. Again the potential reduction of the U-tubes heat transfer due to nitrogen discharge from the accumulator to the reactor coolant system was observed not to be very important considering that the steam generator is not needed as heat sink from this time on.

5.3 TRANSIENT BEHAVIOUR WITH OPERATOR INTERVENTION. (B)

A simulation with operator intervention has been done in order to analize the passive behaviour of the plant for such kind of transients under the influence of a recovery procedure. Corresponding figures are identified in Appendix IV, V, and VI, with the letter "B" in brakets. The set of figures for each transient case is listed in Table 5.1. Variable identifications are listed in Table 5.2.

a) Cases II(B), (2" spray line), and III(B), (1.5" CVCS) :

The LP-SB-3 recovery procedure reduced to "bleed" of the steam generator, due the unavailability of the electrical "feed" system in "total black-out" condition, has been simulated in both cases.

A RELAP5/MOD2 restart analysis from the restart file previous to the core heat-up has been done by simulating the manual opening of the relief valves to the atmosphere. The "bleeding" action was delayed in both cases until the core heat-up started.

The chronology of events is summarized in Tables 5.7 and 5.8, and the main results are represented in the figures of Appendix V and Appendix VI respectively.

The steam generator downcomer level (Fig. 13) has been analized in both cases during the transient in order to guarantee that enough inventory is still available in the secondary system when the "S.G. bleed" operation is required to cool the plant. In both cases the results indicate that the "S.G. bleed" is possible (Fig. 5, 6) and effective.

The behaviour of the plant in the recovery operation is similar to the observed in the LOFT facility forcing the primary system cooling (Fig. 22) and depressurization (Fig. 2) and the discharge of the accumulator (Fig. 7, 9) with core quenching (Fig. 19) avoiding the core overheating (Fig. 26).

Both manual steam dump closure as well as accumulator isolation on low accumulator level have been simulated as operator actions. This intends to avoid the undesired nitrogen discharge from the accumulator to the primary system that could reduce the primary/secondary heat transfer characteristics. Anyway, as in both cases the steam generator is no more needed as heat sink from the accumulator discharge on, nitrogen injection was not considered to be dangerous.

The effectiveness of the operation and the correct sizing of the accumulator have been so demonstrated in both cases for such an event.

b) <u>Case I(B)</u> :

For case I(B) the "bleed" operation has no sense because, once the accumulator has discharged automatically at the beginning of the transient, it is not possible to discharge it again by bleeding the steam generator to the atmosphere.

In case of continuous "total black-out" condition, the only possibility for recovering the plant is to guarantee a minimum inventory in the primary system to avoid the core uncovery and heat-up.

Due to the unavailability of the ECCS injection pumps, the only way to inject water in the system is through the accumulator pathway.

Taking into account that the accumulator of José Cabrera nuclear power plant is installed outside of the containment building and that there is a margin to core heat-up startup of at least 25 minutes with the conservativa hypothesis and boundary conditions considered, it is possible to make it available again by :

- isolating the accumulator by closing the accumulator isolation valve,
- venting it,
- refilling it from the reload water storage tank by gravity circulation or gasoline pump operation,
- presurizing it with nitrogen,
- opening the isolation valve to force the discharge,
- monitoring the accumulator discharge in the control room by watching the accumulator level,
- isolating the accumulator discharge before its level was reduced to the bottom in order to avoid any nitrogen injection into the primary system that could reduce the heat transfer.

The chronology of events is summarized in Table 5.6, and the main results are represented in figures of Appendix IV.

In this way the inventory of the primary coolant system can b maintained (Fig. 9) without core uncovery (Fig. 19) avoidir the cladding temperature excursion (Fig. 26).

The proccess should be repeated in a cyclic way with intervals of 25 minutes until electrical supply is recovered and the pumps of safety injection system are available to mitigate the consecuences of the event.

This procedure is also needed in case II(B) and III(3) for long term cooling in case of continuous "total black-out" condition once the first discharge of the accumulator has been forced by the operator with the manual initiated "S.G. bleed" operation.

The effectiveness of the operation and the correct sizing of the accumulator have been so demonstrated in this case for such an event.

5.4 PROPOSAL FOR IMPROVING THE LOCA EMERGENCY OPERATING PROCEDURE.

The emergency operating procedures for José Cabrera nuclear power plant are based on a diagnosis of the accident (E-0) followed by specific procedures for the three design base accidents:

- Loss of coolant .... (E-1).
- Main steam line break .... (E-2).
- Steam generator tube rupture .... (E-3).

An improvement of the present E-1 procedure considering the out-of-the-standard-licencing-bases SB-3 scenario has been suggested (Fig 5.1).

The operator identifies the accident as a SBLOCA based on low primary pressure, low pressurizer level, high secondary pressure, and high containment pressure, temperature and radiation. If there is a simultaneous total electrical failure, known as "total black-out" condition, the operator has to identify if it is a tipe (I) or a tipe (II/III) case by watching the accumulator water level in the control room.

In case of accumulator level reduction in the first minutes the tipe (I) option shoud be applied. It is recommended to isolate, vent, refill, pressurize and line-up it again as soon as possible during the first 25 minutes. He must repeat this operation at regular intervals until electrical supply is recovered and ECCS pumps are available to guarantee the long term cooling following the present E-1 procedure.

In non discharge case the tipe (II/III) option should be applied. It is recommended to force the depressurization of the primary system below the accumulator set-point as soon as possible in the first 30 minutes. There are three ways of doing it:

- Indirect primary depressurization, by manual "bleeding" of the steam generator through the relief valves to the atmosphere (air supply to open the valves were normally available and, anyway, manual access to this valve outside the containment is guaranteed). This operation is recommended as a first solution.

- Direct primary depressurization, by opening the pressurizer relief valve (air supply to open the PORV is guaranteed). This is not recommended as a first solution because this operation reduces the inventory in the primary system and accelerate the core uncovery and heat-up. It also increases the radiation level in the containment building. The PORV opening behaves in the same way as an increase in the break area. This option should be used as a second solution only in case of impossibility for steam dump operation.

- Combined primary depressurization by simultaneous direct and indirect above operations.

The indirect operation, provided that enough level is available in the steam generator, would be a real direct application of the "lessons learned" from the analysis of OECD-LOFT LP-SB-3 experiment to the José Cabrera nuclear power plant.

Once the discharge is verified, the operator will proceed following the recommendations indicated above for the self-discharge case.

6. RUN STATISTICS.

Calculations started on a CYBER 180/810 computer that was transformed to 180/830 during the execution of the last set of cases. CPU time vs. reactor time plots have been provided for two cases:

- Case II(A), 2" break without operator intervention, on a CYBER 180/810 computer.

- Case I(A), 6" break without operator intervention, on a CYBER 180/830 computer.

A direct comparison of the CYBER 180/810 and 180/830 CPU time to reactor time ratio can be seen in Fig 6.1.

In table 6.1 a typical run statistics has been summarized for both cases. The table includes the "ICAP required number" that was calculated based on the transient time, the total number of active volumes in the model (time dependent volumes were not considered), the total number of time steps and the total CPU time.

A maximum time step of 0.05 s. and the semi-implicit option were selected for all the transients that ran without special difficulties. Mass error was always under the control of the code giving negligible values (Fig. 27) in comparison with the total mass of the system.

7. CONCLUSIONS.

RELAP5/MOD2 cycle 36.04 has been used to analize the SB-3 scenario in a commercial power plant. No major difficulties in the use of the code have been detected.

The good design and sizing of the major components of the plant have been demonstrated in this beyond-licensing bases analysis.

As a result, an improvement to the emergency operating procedure has been suggested. Calculated time requirement for operator intervention (25' in case I, 30' in case II and 50' in case III) is always higher than the maximum estimated credible duration for the "total black-out" (20').

As a conclusion of the analysis, the suggested improvement of the present emergency operating procedure to mitigate the consequences of such an unlikely event has been demostrated to be good enough to guarantee the safety of the plant.

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TABLES

TABLE 3.1 RELAP5/MOD2 model description (nodalization).

CONCEPT	NODE NUMBER	NODE TYPE
PRIMARY COOLANT SYSTEM		
Cold leg - pump discharge	160	branch
Break	163	valve
Containment	164	tm.dp.vol.
Cold leg - vessel inlet	165	branch
Vessel downcomer - lower part	208	annulus
Vessel bottom	212	branch
Core lower plenum	211	single vol.
Core inlet junction	243	single jun.
Reactor core	209	pipe
Core outlet junction	244	single jun.
Core upper plenum	207	single vol.
Core bypass	210	pipe
Reactor outlet	206	branch
Vessel downcomer junction	245	single jun.
Vessel downcomer - ECCS injection	204	annulus
Vessel downcomer junction	246	single jun.
Vessel downcomer - upper part	202	annulus
Vessel top	201	branch
Vessel upper-upper plenum	203	branch
Vessel upper plenum	205	branch
Hot leg - vessel outlet	100	branch
Hot leg - steam generator inlet	105	branch
S.G. inlet plenum	110	branch
S.G. tube inside the tube-plate (hot)	115	branch
S.G. tube connected to the riser	120	pipe
S.G. tube inside the tube-plate (cold)	125	branch
S.G. outlet plenum	130	branch
Cross over leg - S.G. outlet	140	branch
Cross over leg - intermediate	142	branch
Cross over leg - pump inlet	144	single vol.
Primary coolant pump	150	pump
Pressurizer surge line	300	pipe
Pressurizer inlet junction	301	single jun.
Pressurizer vessel (lower part)	310	pipe
Pressurizer junction	311	single jun.
Pressurizer vessel (upper part)	312	pipe
Pressurizer safety valve 1 inlet	314	branch
Pressurizer safety valve 1	316	valve
Pressurizer safety valve 2 inlet	318	branch
Pressurizer safety valve 2	320	valve
Pressurizer relief valve 1 inlet	322	branch
Pressurizer relief valve 1	324	valve
Pressurizer relief valve 2 inlet	326	branch
Pressurizer relief valve 2	328	valve
Pressurizer relief collector	330	branch
Pressurizer relief-safety collector	332	pipe

TABLE 3.1 RELAP5/MOD2 model description (nodalization). (Cont.)

CONCEPT	NODE NUMBER	NODE TYPE
Pressurizer relief tank (water volume)	334	branch
Pressurizer relief tank (steam volume)	336	branch
Pressurizer relief tank rupture disc	338	valve
Containment atmosphere	340	tm.dep.vol
Cold leg - spray line connection	349	single jun.
Pressurizer spray line (up)	350	pipe
Pressurizer spray valve	352	valve
Pressurizer spray line (down)	354	branch
Accumulator	600	accumulator
Accumulator discharge line	605	singl vol.
Accumulator isolation valve	610	valve
Accumulator discharge line	620	pipe
Accumulator discharge line check valve	630	valve
Reload water storage tank	650	tm.dep.vol.
Emergency core cooling system (pumps)	655	tm.dep.jun.
SECONDARY SYSTEM		
S.G. downcomer	450	annulus
S.G. downcomer - riser connection	455	single jun.
S.G. riser	400	pipe
S.G. separator	410	separator
S.G. separator bypass	440	branch
S.G. separator outlet steam volume	420	branch
S.G. first dryer	424	separator
S.G. first dryer draining pipe	426	branch
S.G. second dryer (orifice plate)	430	separator
S.G. second dryer draining pipe	428	branch
S.G. feedwater tank	444	tm.dep.vol
S.G. feedwater flow	445	tm.dep.jun.
AFWS turbine-driven pump FW2 suction tank	448	tm.dep.vol
AFWS turbine-driven pump FW2 junction	449	tm.dep.jun.
EFWS motor pump suction tank	456	tm.dep.vol.
EFWS motor pump junction	457	tm.dep.jun.
Steam line (S.G. outlet)	500	pipe
Steam line	502	branch
Steam line	504	branch
Steam line isolation valve	506	valve
Steam line	508	branch
Steam line	510	pipe
Turbine trip valve	512	valve
Steam line	513	singl vol.
Main steam control valve	514	valve
Turbine	516	tm.dep.vol.
S.G. safety valve 1	540	valve
S.G. safety valve 1 discharge line	542	branch
Environmental atmosphere	560	tm.dep.vol.

TABLE 3.1 RELAP5/MOD2 model description (nodalization). (Cont.)

CONCEPT	NODE NUMBER	NODE TYPE
S.G. safety valve 2	544	valve
S.G. safety valve 2 discharge line	546	branch
Environmental atmosphere	561	tm.dep.vol.
S.G. safety valve 3	548	valve
S.G. safety valve 3 discharge line	550	branch
Environmental atmosphere	562	tm.dep.vol.
S.G. safety valve 4	552	valve
S.G. safety valve 4 discharge line	554	branch
Environmental atmosphere	563	tm.dep.vol.
AFWS FW2 steam consumption (injecting)	459	tm.dep.jun.
Environmental atmosphere	460	tm.dep.vol.
AFWS FW2 steam consumption (not injecting)	461	tm.dep.jun.
Environmental atmosphere	462	tm.dep.vol.
Steam relief line	518	branch
Steam relief line to the atmosphere	520	single vol.
S.G. relief valve 1 (to thw atmosphere)	522	valve
S.G. relief valve 1 discharge line	524	branch
Environmental atmosphere	564	tm.dep.vol.
S.G. relief valve 2 (to the atmosphere)	526	valve
S.G. relief valve 2 discharge line	528	branch
Environmental atmosphere	565	tm.dep.vol.
S.G. relief line to the condenser	530	branch
S.G. relief valve 3 (to the condenser)	532	valve
S.G. relief valve 3 discharge line	534	single vol.
Condenser isolation valve	536	valve
Condenser	538	tm.dep.vol.

TABLE 4.1 Stabilization system actuation logic

VARIABLE TO BE FITTED	CONTROL SYSTEM ACTION
Primary pressure	Time dependent volume connected to the pressurizer steam dome
Primary coolant average temperature	Main steam control valve modulation
Primary system mass flow rate	Reactor coolant pump speed modulation
Pressurizer level	Make-up and let-down (CVCS) modulation
Steam generator downcomer level	Main feedwater modulation
Primary - secondary temperature difference. Steam generator pressure	Parametric analysis of the hydraulic diameter

TABLE 4.2. Description of the steady state analysis figures.

FIG	SUBJET
SS.1	Reactor power (w)
SS.2	Pressurizer pressure (Pa)
SS.3	Steam generator pressure (Pa)
SS.4	Primary coolant mass flow rate (Kg/s)
SS.5	Stabilization pump velocity (Rad/s)
SS.6	Pressurizer collapsed liquid level (%)
SS.7	Stabilization make-up mass flow rate (Kg/s)
SS.8	Stabilization let-down mass flow rate (Kg/s)
SS.9	Steam generator downcomer collapsed liquid level (m)
SS.10	Stabilization feedwater mass flow rate (Kg/s)
SS.11	Primary coolant average temperature (C)
SS.12	Stabilization main steam control valve mass flow rate (Kg/s)
SS.13	Steam generator circulation rate

TABLE 4.3 Steady state results at nominal conditions.

Variable	Units	JOSE CABRERA (measured)	RELAP5/MOD2 (calculated)
Reactor power	(Mw)	510.00	510.00
RCS average temperature	(ºK)	566.60	566.60
Pressurizer level	(%)	67.00	67.00
Pressurizer pressure	(MPa)	13.82	13.82
RCS mass flow rate	(Kg/s)	3605.00	3605.00
Reactor coolant pump speed	(rpm)	990.00	995.00
Steam generator pressure	(MPa)	4.63	4.63
Steam generator circulation rate	(-)	3.34	3.34
Steam generator collapsed liquid level	(m)	8.68	8.67
Steam flow rate	(Kg/s)	266.40	265.90
Feedwater temperature	(ºK)	477.00	477.00

TABLE 5.1. Description of the transient analysis figures for cases I(A&B), II(A&B), and III(A&B).

FIG	SUBJET
1	Reactor power (w)
2	Primary and secondary pressure (Pa)
3	Primary coolant mass flow rate (Kg/s)
4	Secondary system mass flow rates (Kg/s)
5	Steam generator relief and safety mass flow rates (Kg/s)
6	Integral steam mass from the S.G. to the atmosphere (Kg)
7	Break and ECCS mass flow rate (Kg/s)
8	Break void fraction (-)
9	Balance of mass in the primary system (Kg)
10	Pressurizer collapsed liquid level (%)
11	Liquid fraction in the "up" part of the steam generator tube
12	Liquid fraction in the "down" part of the steam generator tube
13	Steam generator downcomer collapsed liquid level (m)
14	Liquid fraction in the steam generator riser (-)
15	Liquid fraction in the steam generator downcomer (-)
16	Liquid fraction in the "down" part of the loop seal (-)
17	Liquid fraction in the "horizontal" part of the loop seal(-)
18	Liquid fraction in the "up" part of the loop seal (-)
19	Core collapsed liquid level (m)
20	Core liquid fractions (-)
21	Primary coolant densities (Kg/m3)
22	Primary coolant temperatures (K)
23	Primary and secondary temperatures (K)
24	Primary coolant average temperature (C)
25	Delta temperature in the primary coolant (C)
26	Average channel cladding temperatures (K)
27	Mass error (Kg)

TABLE 5.2. Variables identification in the transient figures.

KEYWORD		CONCEPT
P	312020000	Pressurizer pressure
P	100010000	Hot leg pressure
P	165010000	Cold leg pressure
P	430010000	Steam generator dome pressure
TEMPF	312010000	Pressurizer liquid temperature
TEMPF	100010000	Hot leg liquid temperature
TEMPF	110010000	Steam generator inlet temperature (primary)
TEMPF	130010000	Steam generator outlet temperature (primary)
TEMPF	165010000	Cold leg liquid temperature
TEMPF	211010000	Core lower plenum liquid temperature
TEMPF	207010000	Core upper plenum liquid temperature
TEMPF	430010000	Steam generator dome temperature
RHO	100010000	Hot leg density
RHO	142010000	Loop seal density (horizontal part)
RHO	165010000	Cold leg density
RHOFJ	163000000	Break flow density
MFLOWJ	100010000	Hot leg mass flow rate
MFLOWJ	163000000	Break mass flow rate
MFLOWJ	165010000	Cold leg mass flow rate
MFLOWJ	243000000	Core inlet mass flow rate
MFLOWJ	630000000	Accumulator check valve mass flow rate
MFLOWJ	655000000	Safety injection system mass flow rate
MFLOWJ	457000000	Steam generator AFWS mass flow rate
MFLOWJ	514000000	Steam flow to the turbine
MFLOWJ	522000000	Steam generator relief to the atmosphere mass flow rate (valve 1)
MFLOWJ	526000000	Steam generator relief to the atmosphere mass flow rate (valve 2)
MFLOWJ	536000000	Steam generator relief to the condenser mass flow rate
HTTEMP	209100110	Average channel cladding temperature node 1 (bottom)
HTTEMP	209100210	Average channel cladding temperature node 2
HTTEMP	209100310	Average channel cladding temperature node 3
HTTEMP	209100410	Average channel cladding temperature node 4
HTTEMP	209100510	Average channel cladding temperature node 5
HTTEMP	209100610	Average channel cladding temperature node 6 (top)
VOIDF	140010000	Liquid fraction in the loop seal (down)
VOIDF	142010000	Liquid fraction in the loop seal (horizontal)
VOIDF	144010000	Liquid fraction in the loop seal (up)
VOIDF	207010000	Liquid fraction in the core upper plenum

TABLE 5.2. Variables identification in the transient figures. (Cont.)

KEYWORD		CONCEPT
VOIDF	209010000	Liquid fraction in the core node 1 (bottom)
VOIDF	209020000	Liquid fraction in the core node 2
VOIDF	209030000	Liquid fraction in the core node 3
VOIDF	209040000	Liquid fraction in the core node 4
VOIDF	209050000	Liquid fraction in the core node 5
VOIDF	209060000	Liquid fraction in the core node 6 (top)
VOIDF	400010000	Liquid fraction in the SG riser node 1 (bottom)
VOIDF	400020000	Liquid fraction in the SG riser node 2
VOIDF	400030000	Liquid fraction in the SG riser node 3
VOIDF	400040000	Liquid fraction in the SG riser node 4
VOIDF	400050000	Liquid fraction in the SG riser node 5 (top)
VOIDF	450010000	Liquid fraction in the SG downcomer node 1 (top)
VOIDF	450020000	Liquid fraction in the SG downcomer node 2
VOIDF	450030000	Liquid fraction in the SG downcomer node 3
VOIDF	450040000	Liquid fraction in the SG downcomer node 4
VOIDF	450050000	Liquid fraction in the SG downcomer node 5 (bottom)
VOIDF	120010000	Liquid fraction in the SG tube (up)
VOIDF	120020000	Liquid fraction in the SG tube (up)
VOIDF	120030000	Liquid fraction in the SG tube (up)
VOIDF	120040000	Liquid fraction in the SG tube (up)
VOIDF	120050000	Liquid fraction in the SG tube (down)
VOIDF	120060000	Liquid fraction in the SG tube (down)
VOIDF	120070000	Liquid fraction in the SG tube (down)
VOIDF	120080000	Liquid fraction in the SG tube (down)
VOIDGJ	163000000	Break flow void fraction
PMPVEL	150	Primary coolant pump velocity
RKTPOW	0	Total reactor power
CPUTIME	0	CPU time
EMASS	0	Mass error
CNTRLVAR	2	Pressurizer collapsed liquid level
CNTRLVAR	3	S.G. downcomer collapsed liquid level
CNTRLVAR	4	Primary coolant average temperature
CNTRLVAR	19	S.G. riser collapsed liquid level
CNTRLVAR	22	S.G. safety valves total flow rate
CNTRLVAR	23	S.G. safety & relief total flow rate to the atmosphere
CNTRLVAR	24	Integral of the steam flow rate from the S.G. to the atmosphere
CNTRLVAR	25	Core collapsed liquid level
CNTRLVAR	26	Total ECCS flow rate
CNTRLVAR	29	Integral of the ECCS flow rate
CNTRLVAR	30	Delta temperature of the primary coolant

TABLE 5.2. Variables identification in the transient figures. (Cont.)

KEYWORD		CONCEPT
CNTRLVAR	38	Total S.G. feedwater flow rate
CNTRLVAR	43	Integral of the break flow rate
CNTRLVAR	45	Integral of the safety injection system (pumps) flow rate
CNTRLVAR	46	Inventory in the primary system

TABLE 5.3. Chronology of events for case I(A).

Event (Cold leg SBLOCA 6" without recovery)	Time (s)
Cold leg small break	0.
Black-out	0.
Reactor coolant pump start to coastdown	0.
Main feedwater pump start to coastdown	0.
S.G. turbine driven pump consumption starts	0.
Reactor trip signal on low primary flow	0.1
Turbine trip signal on black-out	0.5
Reactor trip signal on turbine trip	0.5
Reactor coolant pump (RCP) breakers opening	0.5
Reactor trip signal on RCP breakers opening	0.5
Reactor trip signal on variable low pressurizer pressure	1.35
Reactor trip signal on fixed low pressurizer pressure	2.1
Reactor trip signal on SG low level (NR) & mismatch	2.6
Safety injection signal on low pressurizer pressure	3.7
Reactor trip signal on "S" signal	3.7
Reactor trip signal on S.G. low low level (NR)	7.8
End of subcooled discharge	13.5
S.G. safety valve first opening	-
S.G. down tubes ends of draining	65.
S.G. up tubes ends of draining	90.
Loop seal clearance	105.
Primary pressure under secondary pressure	118.
Primary pressure under accumulator set-point	190.
S.G. turbine driven pump injection starts	-
S.G. motor pump injection starts	-
Safety injection system pump injection starts	-
Accumulator discharge starts	190.
Accumulator discharge ends	480.
Core heat-up startup	1480.
End of transient	2369.

TABLE 5.4. Chronology of events for case II(A).

Event (Cold leg SBLOCA 2" without recovery)	Time (s)
Cold leg small break	0.
Black-out	0.
Reactor coolant pump start to coastdown	0.
Main feedwater pump start to coastdown	0.
S.G. turbine driven pump consumption starts	0.
Turbine trip signal on black-out	0.5
Reactor trip signal on turbine trip	0.5
Reactor coolant pump (RCP) breakers opening	0.5
Reactor trip signal on RCP breakers opening	0.5
Reactor trip signal on low primary flow	1.15
Reactor trip signal on SG low level (NR) & mismatch	2.6
Reactor trip signal on variable low pressurizer pressure	2.8
Reactor trip signal on fixed low pressurizer pressure	5.7
Reactor trip signal on S.G. low low level (NR)	7.7
Safety injection signal on low pressurizer pressure	9.5
Reactor trip signal on "S" signal	9.5
S.G. safety valve first opening	110.
End of subcooled discharge	120.
S.G. down tubes ends of draining	520.
S.G. up tubes ends of draining	800.
Loop seal clearance	1300.
S.G. safety valve last closure	1370.
Primary pressure under secondary pressure	1460.
Core heat-up startup	1950.
Primary pressure under accumulator set-point	2320.
S.G. turbine driven pump injection starts	-
S.G. motor pumps injection starts	-
Safety injection system pumps injection starts	-
End of transient	2500.

TABLE 5.5. Chronology of events for case III(A).

Event (Cold leg SBLOCA 1.5" without recovery)	Time (s)
Cold leg small break	0.
Black-out	0.
Reactor coolant pump start to coastdown	0.
Main feedwater pump start to coastdown	0.
S.G. turbine driven pump consumption starts	0.
Turbine trip signal on black-out	0.5
Reactor trip signal on turbine trip	0.5
Reactor coolant pump (RCP) breakers opening	0.5
Reactor trip signal on RCP breakers opening	0.5
Reactor trip signal on low primary flow	1.25
Reactor trip signal on SG low level (NR) & mismatch ..	2.6
Reactor trip signal on S.G. low low level (NR)	7.65
Reactor trip signal on fixed low pressurizer pressure	8.6
Safety injection signal on low pressurizer pressure ..	14.85
Reactor trip signal on "S" signal	14.85
S.G. safety valve first opening	70.
End of subcooled discharge	405.
S.G. down tubes ends of draining	950.
S.G. up tubes ends of draining	1410.
Loop seal clearance	2510.
S.G. safety valve last closure	2580.
Core heat-up startup	2980.
Primary pressure under secondary pressure	3220.
Primary pressure under accumulator set-point	3910.
S.G. turbine driven pump injection starts	-
S.G. motor pumps injection starts	-
Safety injection system pumps injection starts	-
Accumulator discharge starts	3910.
Accumulator discharge ends	-
Core quenched	4360.
End of transient	4500.

TABLE 5.6. Chronology of events for case I(B).

Event (Cold leg SBLOCA 6" with recovery)	Time (s)
Cold leg small break	0.
Black-out	0.
Reactor coolant pump start to coastdown	0.
Main feedwater pump start to coastdown	0.
S.G. turbine driven pump consumption starts	0.
Reactor trip signal on low primary flow	0.1
Turbine trip signal on black-out	0.5
Reactor trip signal on turbine trip	0.5
Reactor coolant pump (RCP) breakers opening	0.5
Reactor trip signal on RCP breakers opening	0.5
Reactor trip signal on variable low pressurizer pressure	1.35
Reactor trip signal on fixed low pressurizer pressure	2.1
Reactor trip signal on SG low level (NR) & mismatch	2.6
Safety injection signal on low pressurizer pressure	3.7
Reactor trip signal on "S" signal	3.7
Reactor trip signal on S.G. low low level (NR)	7.8
End of subcooled discharge	13.5
S.G. safety valve first opening	-
S.G. down tubes ends of draining	65.
S.G. up tubes ends of draining	90.
Loop seal clearance	105.
Primary pressure under secondary pressure	118.
Primary pressure under accumulator set-point	190.
S.G. turbine driven pump injection starts	-
S.G. motor pumps injection starts	-
Safety injection system pumps injection starts	-
First accumulator discharge starts	190.
First accumulator discharge ends	480.
Second accumulator discharge starts	1500.
Second accumulator discharge ends	1625.
Core heat-up startup	3360.
End of transient	4400.

TABLE 5.7. Chronology of events for case II(B).

Event (Cold leg SBLOCA 2" with recovery)	Time (s)
Cold leg small break	0.
Black-out	0.
Reactor coolant pump start to coastdown	0.
Main feedwater pump start to coastdown	0.
S.G. turbine driven pump consumption starts	0.
Turbine trip signal on black-out	0.5
Reactor trip signal on turbine trip	0.5
Reactor coolant pump (RCP) breakers opening	0.5
Reactor trip signal on RCP breakers opening	0.5
Reactor trip signal on low primary flow	1.15
Reactor trip signal on SG low level (NR) & mismatch ..	2.6
Reactor trip signal on variable low pressurizer pressure	2.8
Reactor trip signal on fixed low pressurizer pressure	5.7
Reactor trip signal on S.G. low low level (NR)	7.7
Safety injection signal on low pressurizer pressure ..	9.5
Reactor trip signal on "S" signal	9.5
S.G. safety valve first opening	110.
End of subcooled discharge	120.
S.G. down tubes ends of draining	520.
S.G. up tubes ends of draining	800.
Loop seal clearance	1300.
S.G. safety valve last closure	1370.
Primary pressure under secondary pressure	1460.
Core heat-up startup	-
Manual S.G. "bleed" starts	1900.
Primary pressure under accumulator set-point	1960.
S.G. turbine driven pump injection starts	-
S.G. motor pumps injection starts	-
Safety injection system pumps injection starts	-
Accumulator discharge starts	1960.
Manual S.G. "bleed" ends due to low accumulator level	2200.
Accumulator discharge ends	2200.
End of transient	2300.

TABLE 5.8. Chronology of events for case III(B).

Event (Cold leg SBLOCA 1.5" with recovery)	Time (s)
Cold leg small break	0.
Black-out	0.
Reactor coolant pump start to coastdown	0.
Main feedwater pump start to coastdown	0.
S.G. turbine driven pump consumption starts	0.
Turbine trip signal on black-out	0.5
Reactor trip signal on turbine trip	0.5
Reactor coolant pump (RCP) breakers opening	0.5
Reactor trip signal on RCP breakers opening	0.5
Reactor trip signal on low primary flow	1.25
Reactor trip signal on SG low level (NR) & mismatch	2.6
Reactor trip signal on variable low pressurizer pressure	-
Reactor trip signal on S.G. low low level (NR)	7.65
Reactor trip signal on fixed low pressurizer pressure	8.6
Safety injection signal on low pressurizer pressure	14.85
Reactor trip signal on "S" signal	14.85
S.G. safety valve first opening	70.
End of subcooled discharge	405.
S.G. down tubes ends of draining	950.
S.G. up tubes ends of draining	1410.
Loop seal clearance	2510.
S.G. safety valve last closure	2580.
Core heat-up startup	2980.
Manual S.G. "bleed" starts	3000.
Primary pressure under accumulator set-point	3040.
Accumulator discharge starts	3040.
S.G. turbine driven pump injection starts	-
S.G. motor pumps injection starts	-
Safety injection system pumps injection starts	-
Accumulator discharge ends	-
Manual S.G. "bleed" ends due to low accumulator level	-
End of transient	3300.

TABLE 6.1. Run statistics.

CASE	II(A)	I(A)
COMPUTER	CYBER 180/810	CYBER 180/830
CPU TIME (s)	207918	104476
REACTOR TIME (s)	2500	2369
C (TOTAL NUMBER OF ACTIVES VOLUMES IN THE MODEL)	109	109
DT (TOTAL NUMBER OF TIME STEPS)	50000	47380
(CPU E+3) / (C x DT)	38.15	20.23
CPU TIME / REACTOR TIME	83	44

FIGURES

NOTE :

RELAP5/MOD2 figures for the steady state and transient calculations are listed in Tables 4.2 and 5.1.

FIG. 2.1 JOSE CABRERA NUCLEAR POWER PLANT REPRESENTATION.

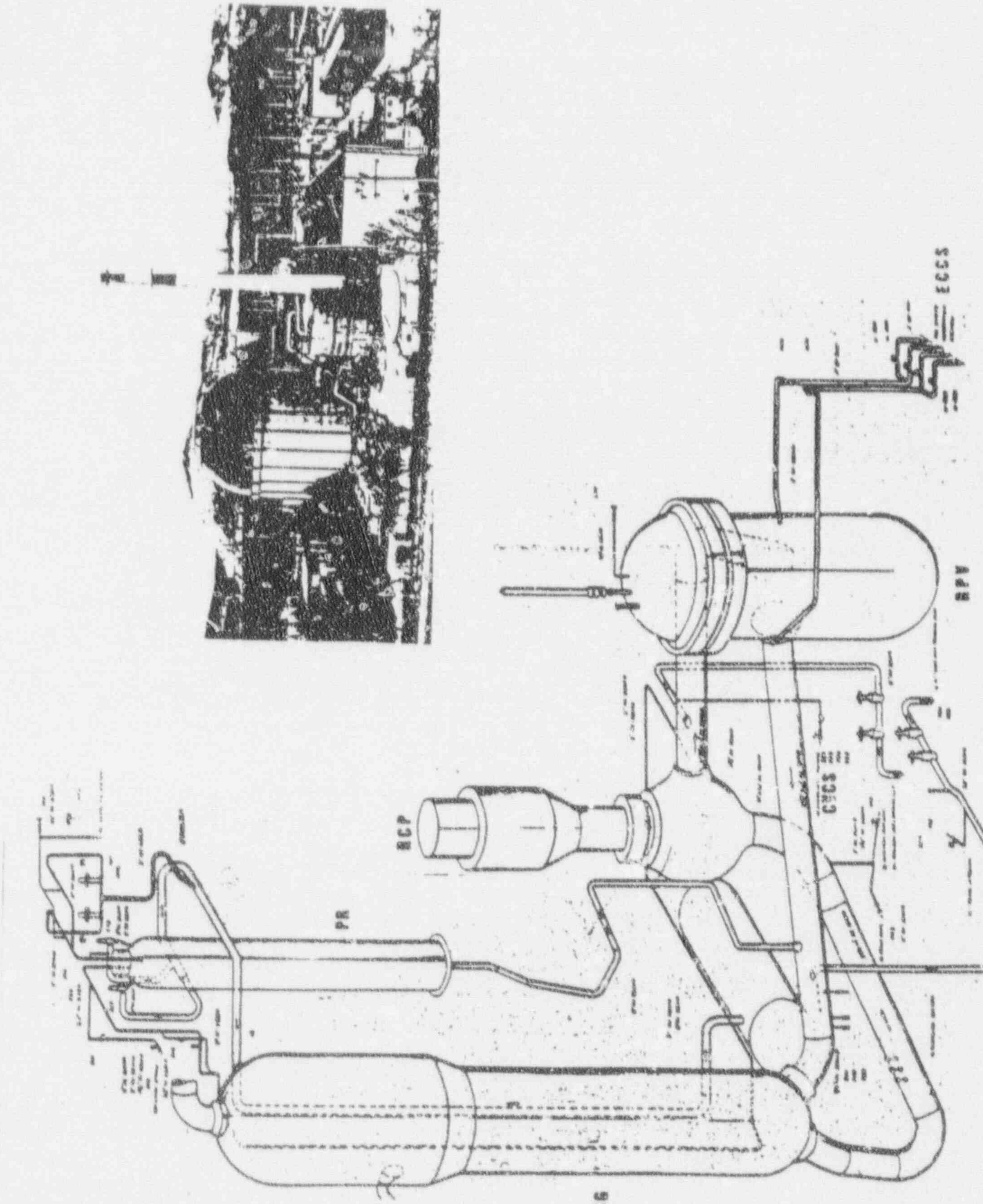
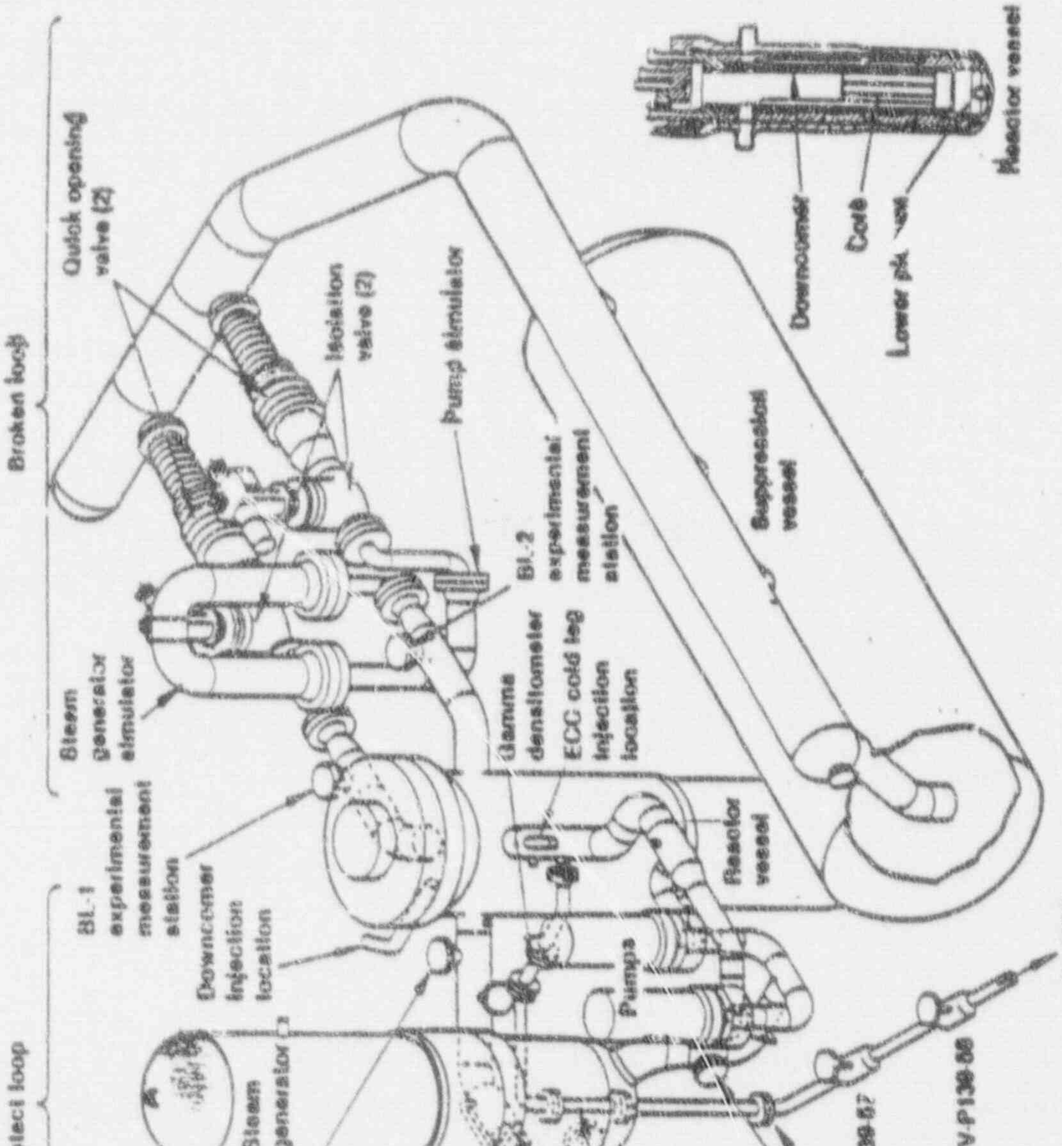


FIG. 2.2 LOFT FACILITY CONFIGURATION FOR LP-SB-3.



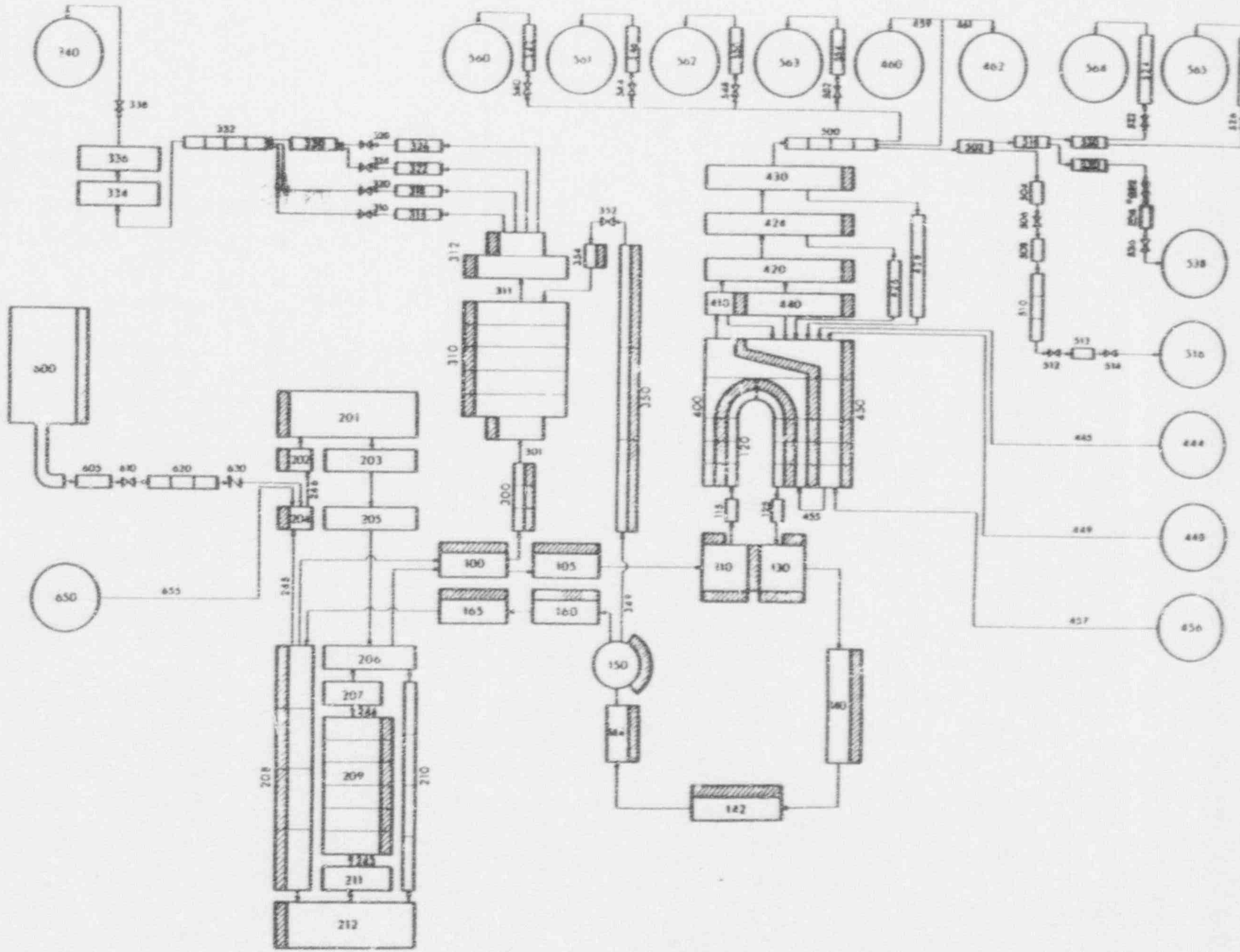
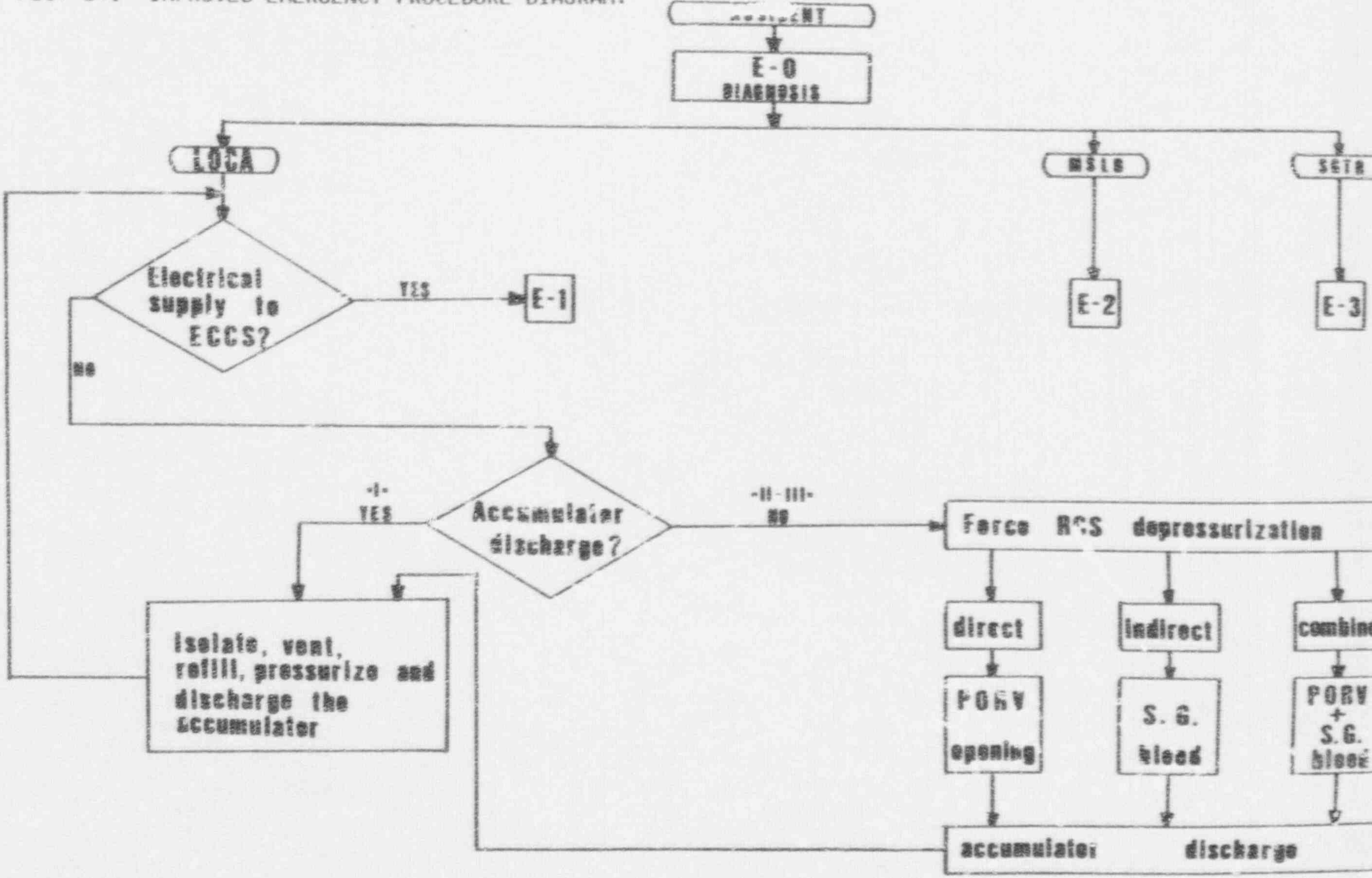


FIG. 3.1 JOSE CABRERA PLANT NODALIZATION.

FIG. 5.1 IMPROVED EMERGENCY PROCEDURE DIAGRAM.



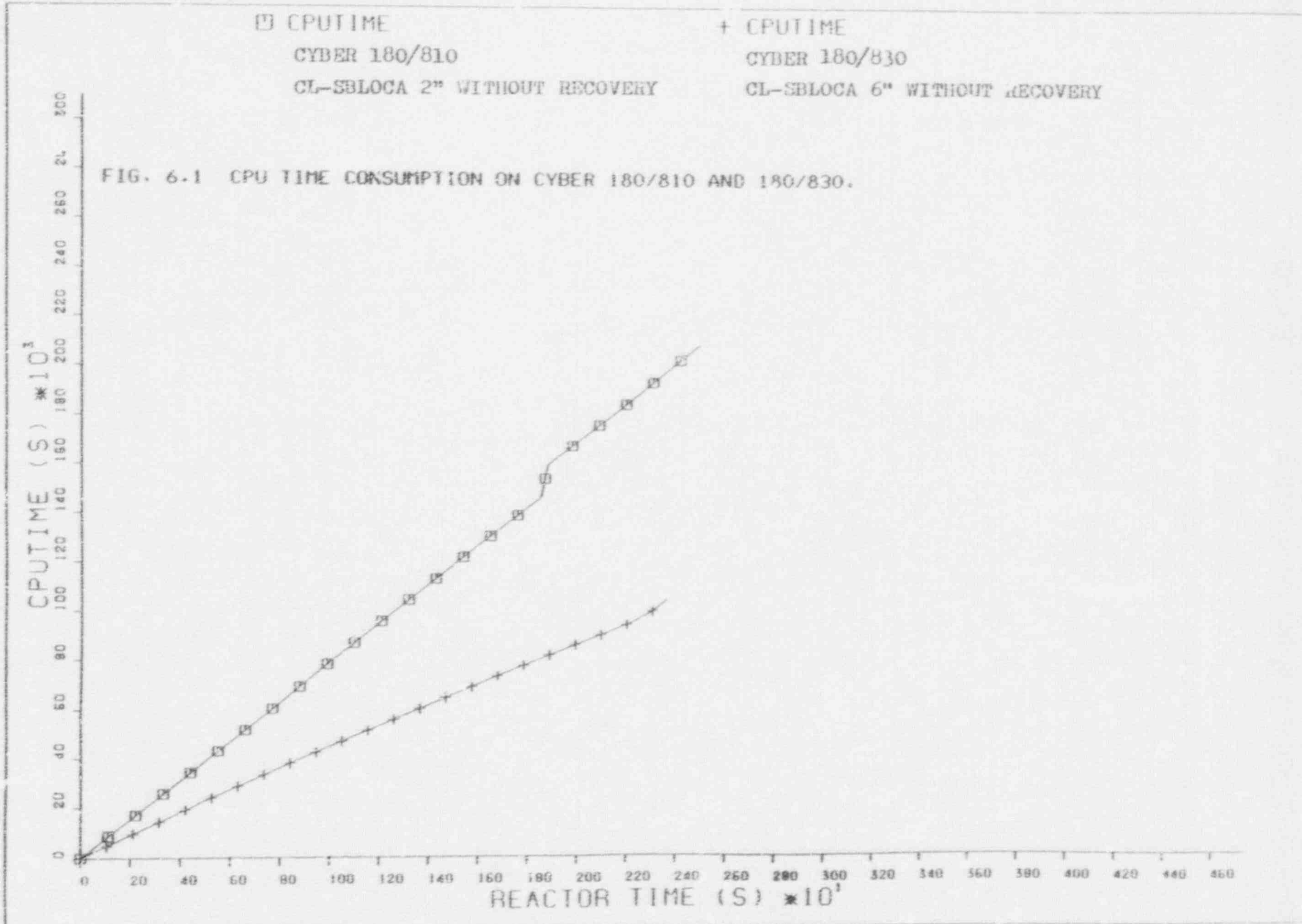


FIG. 6.1 CPU TIME CONSUMPTION ON CYBER 180/810 AND 180/830.

CPUTIME CONSUMPTION ON CYBER 180/810 AND 160/830

APPENDIX SS : FIGURES OF THE STEADY STATE CALCULATION

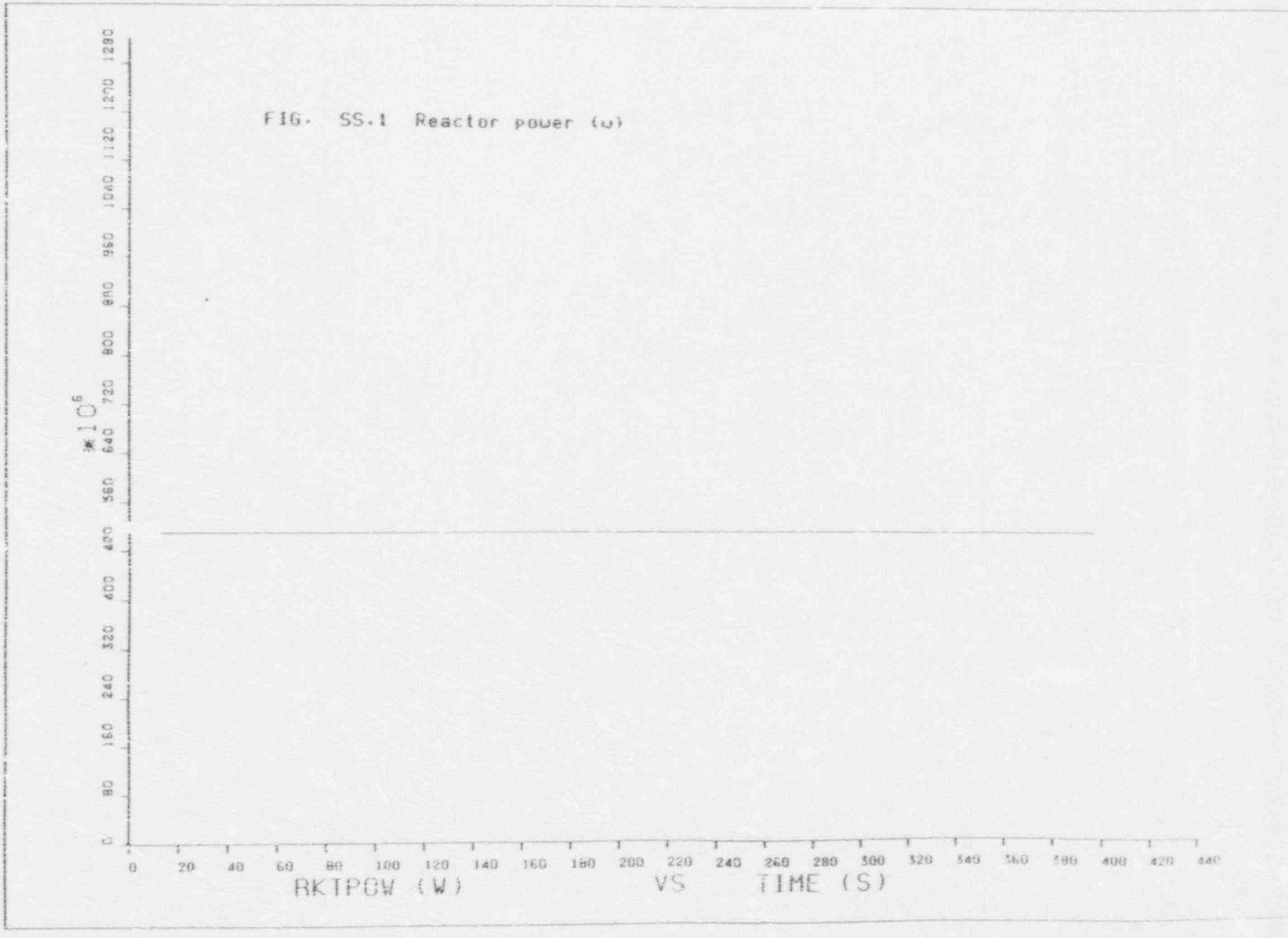


FIG. SS.1 Reactor power (w)

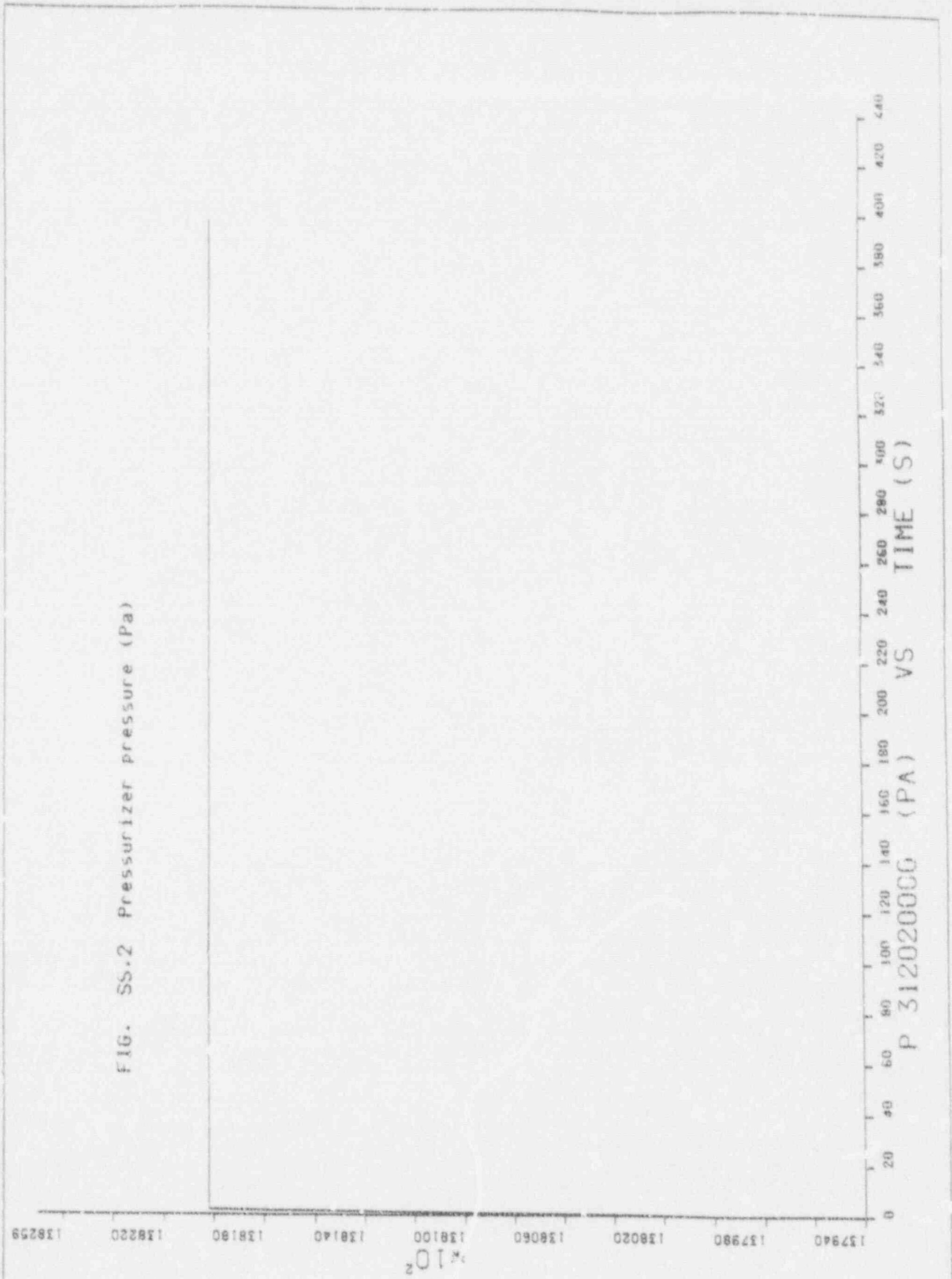


FIG. SS.2 Pressurizer pressure (Pa)

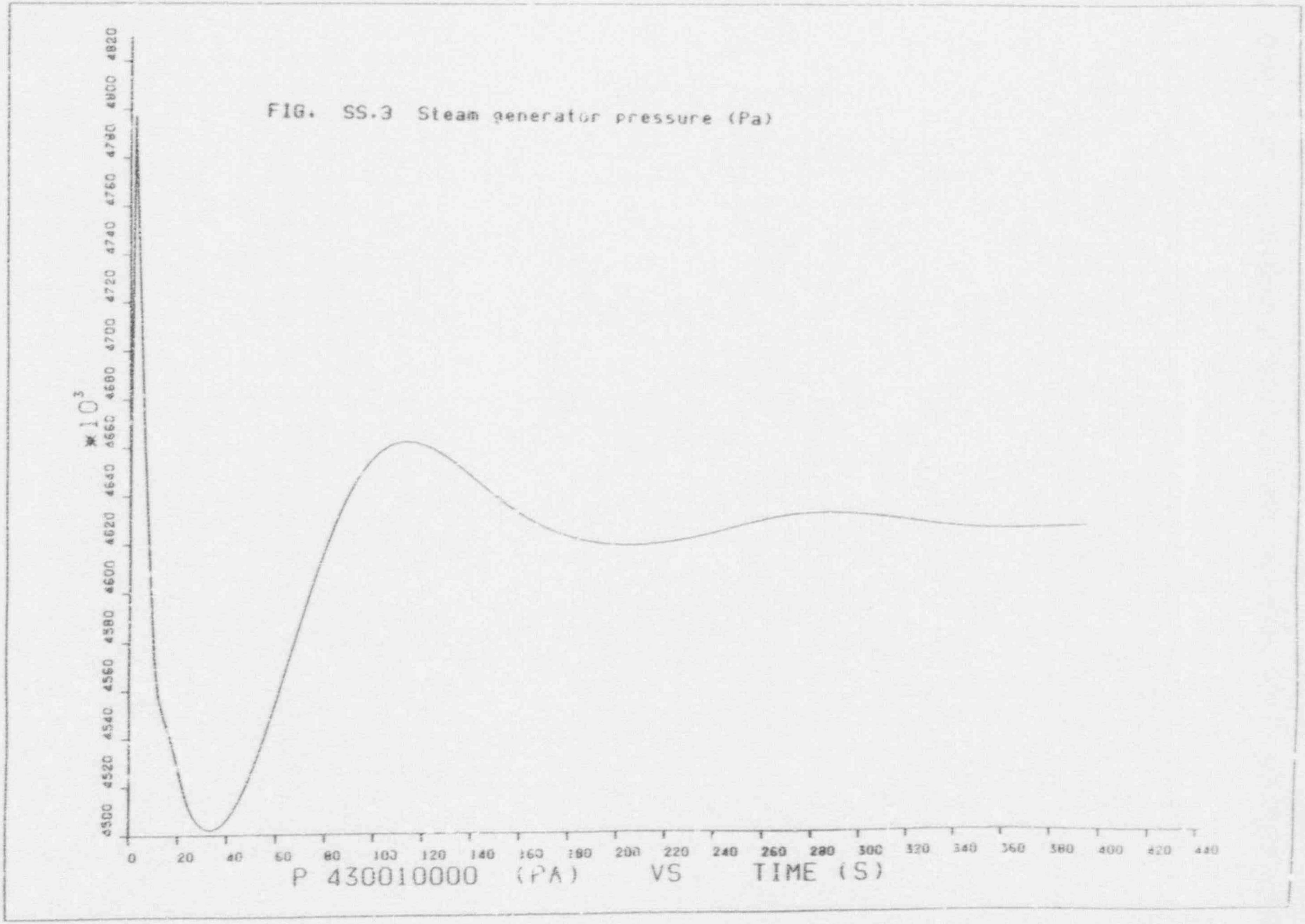


FIG. SS.3 Steam generator pressure (Pa)

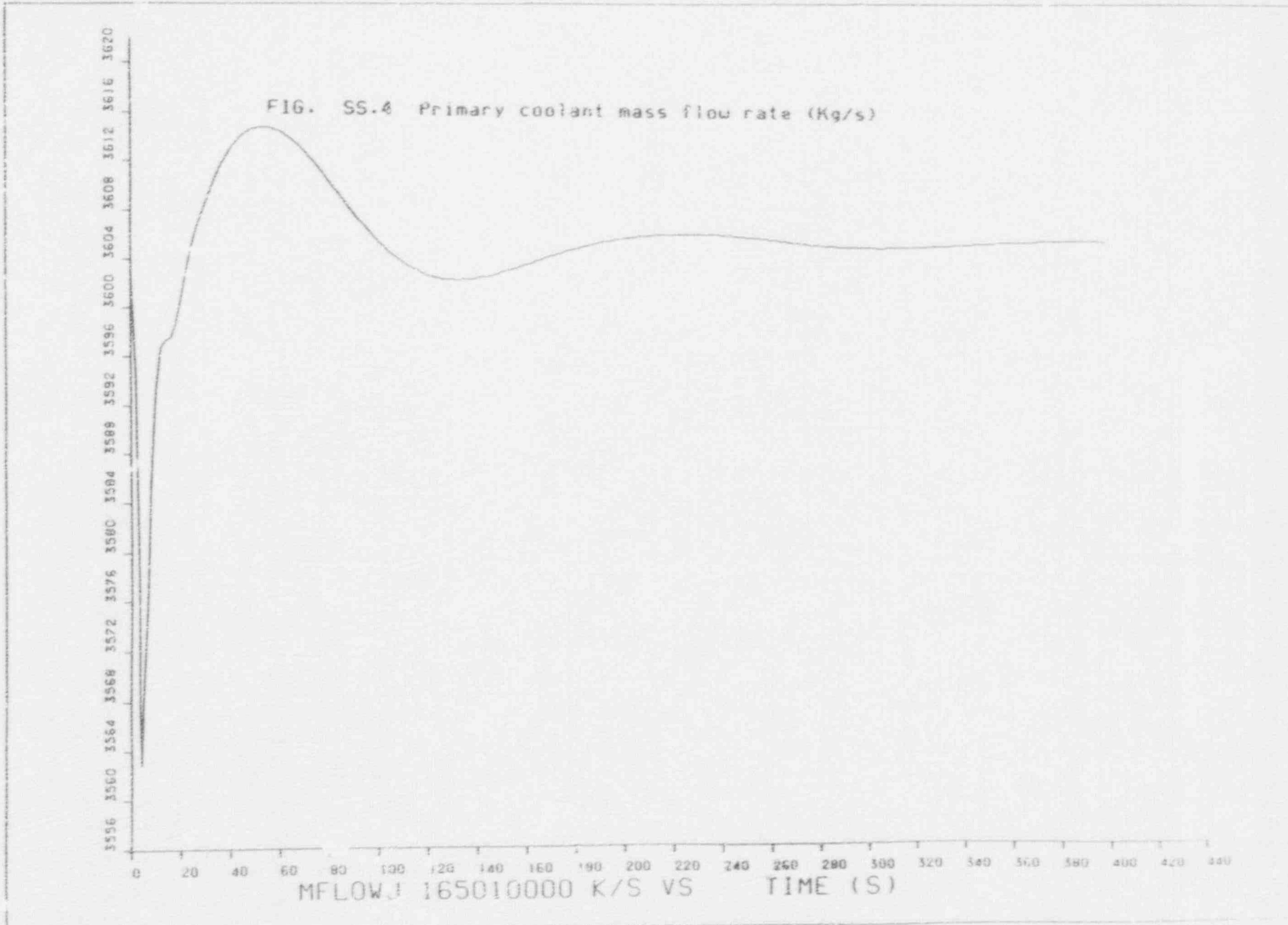


FIG. SS.4 Primary coolant mass flow rate (Kg/s)

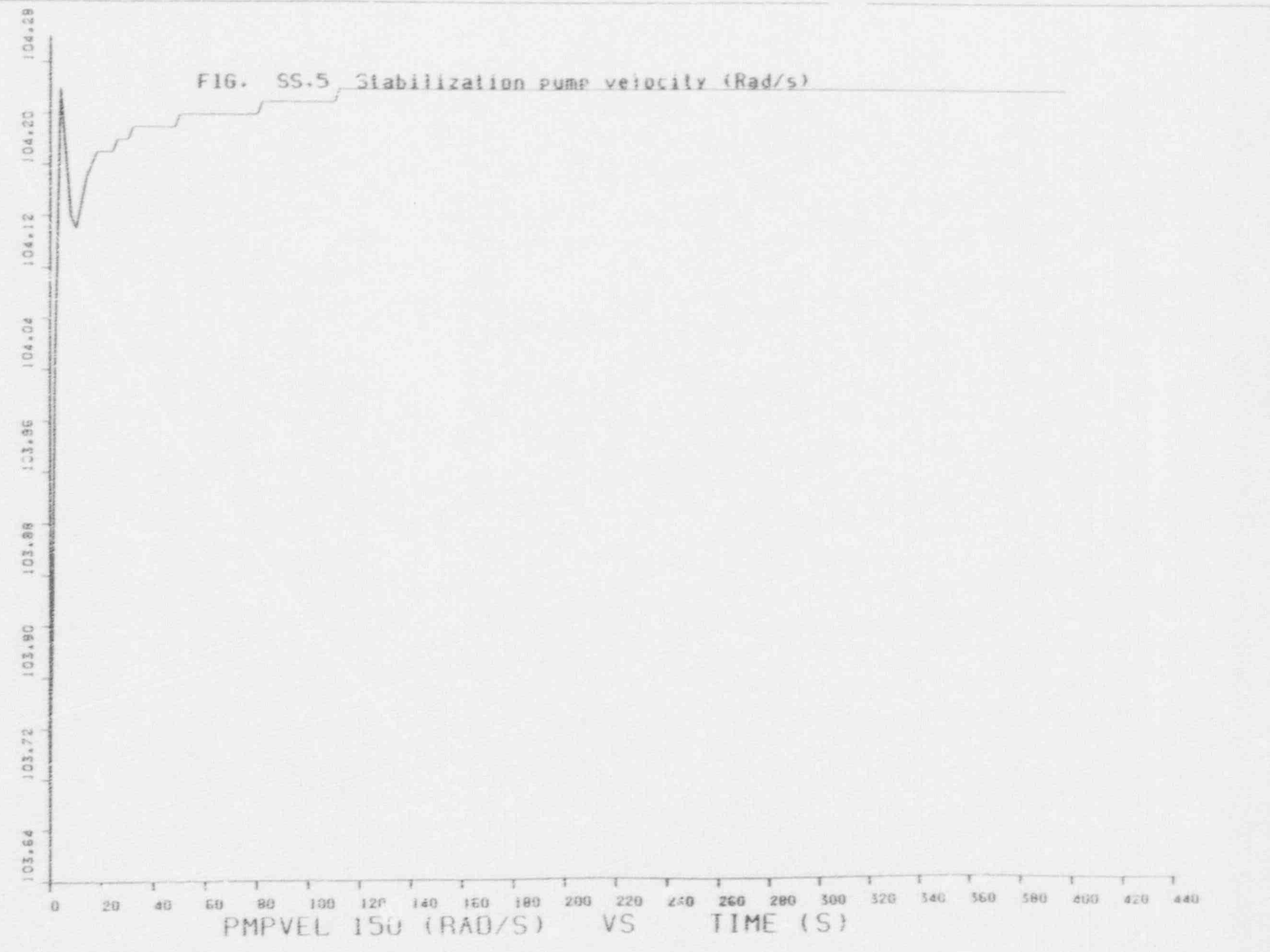


FIG. SS.5 Stabilization pump velocity (Rad/s)

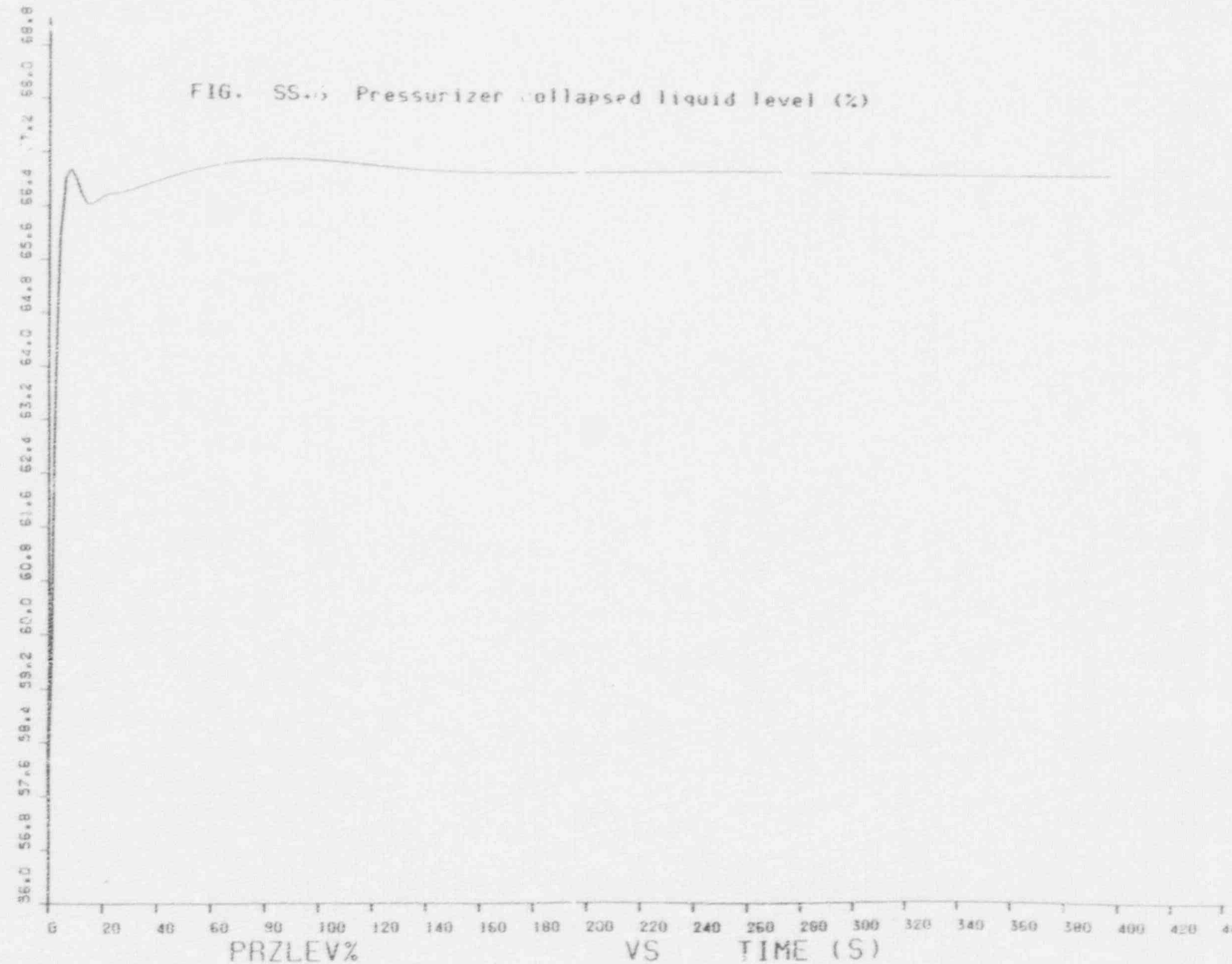


FIG. SS.5 Pressurizer collapsed liquid level (%)

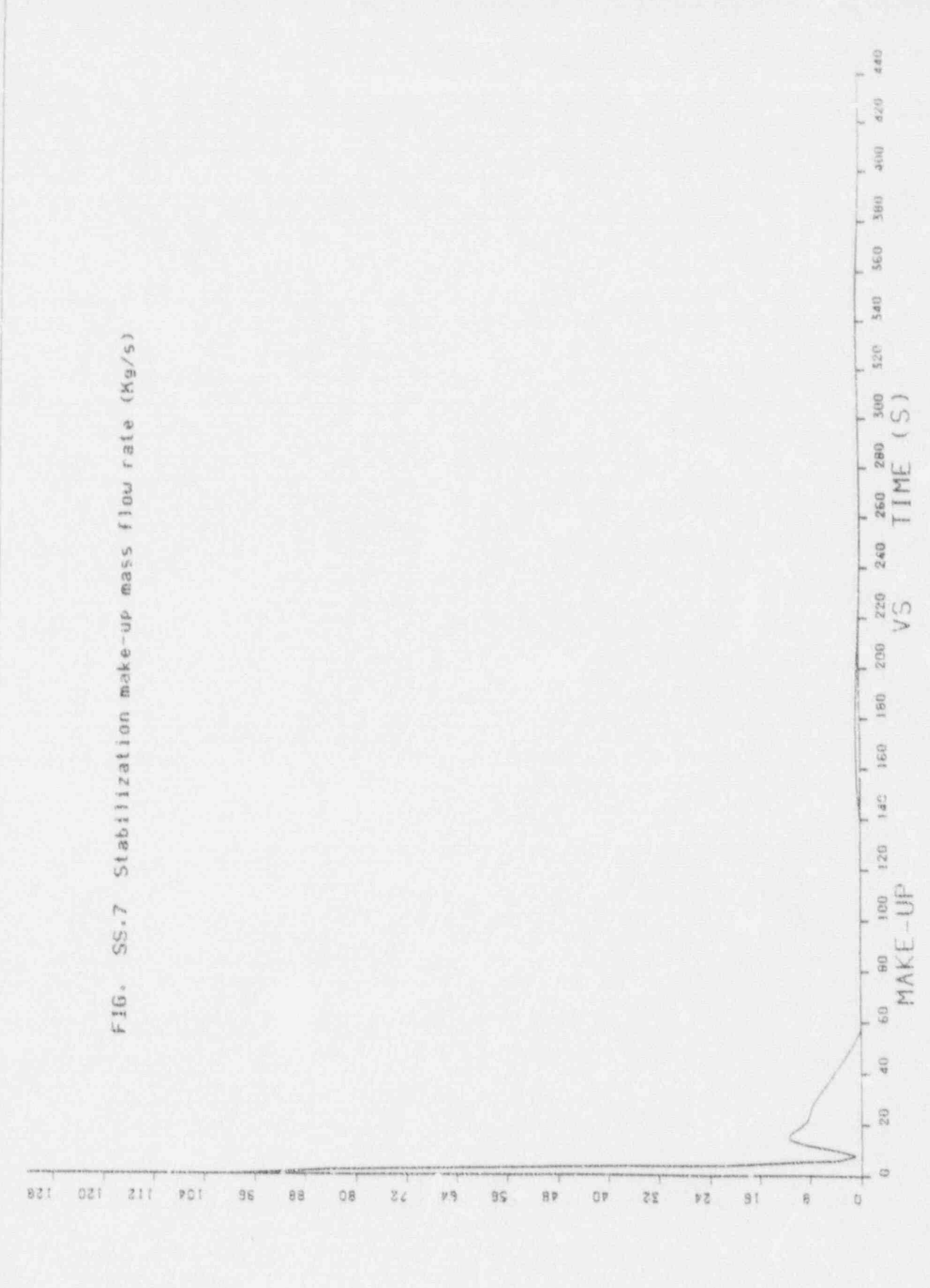


FIG. SS.7 Stabilization make-up mass flow rate (Kg/s)

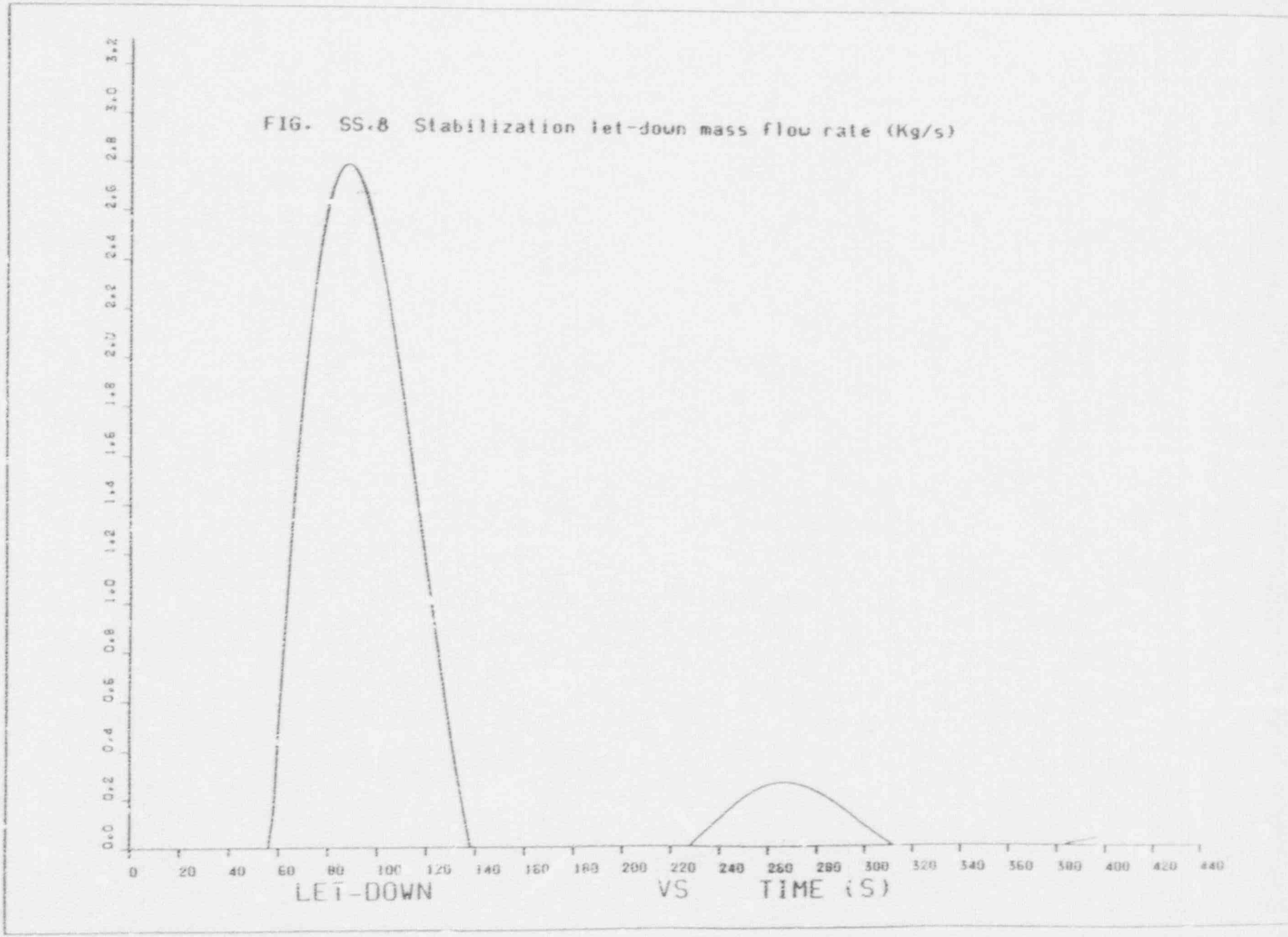


FIG. SS.8 Stabilization let-down mass flow rate (Kg/s)

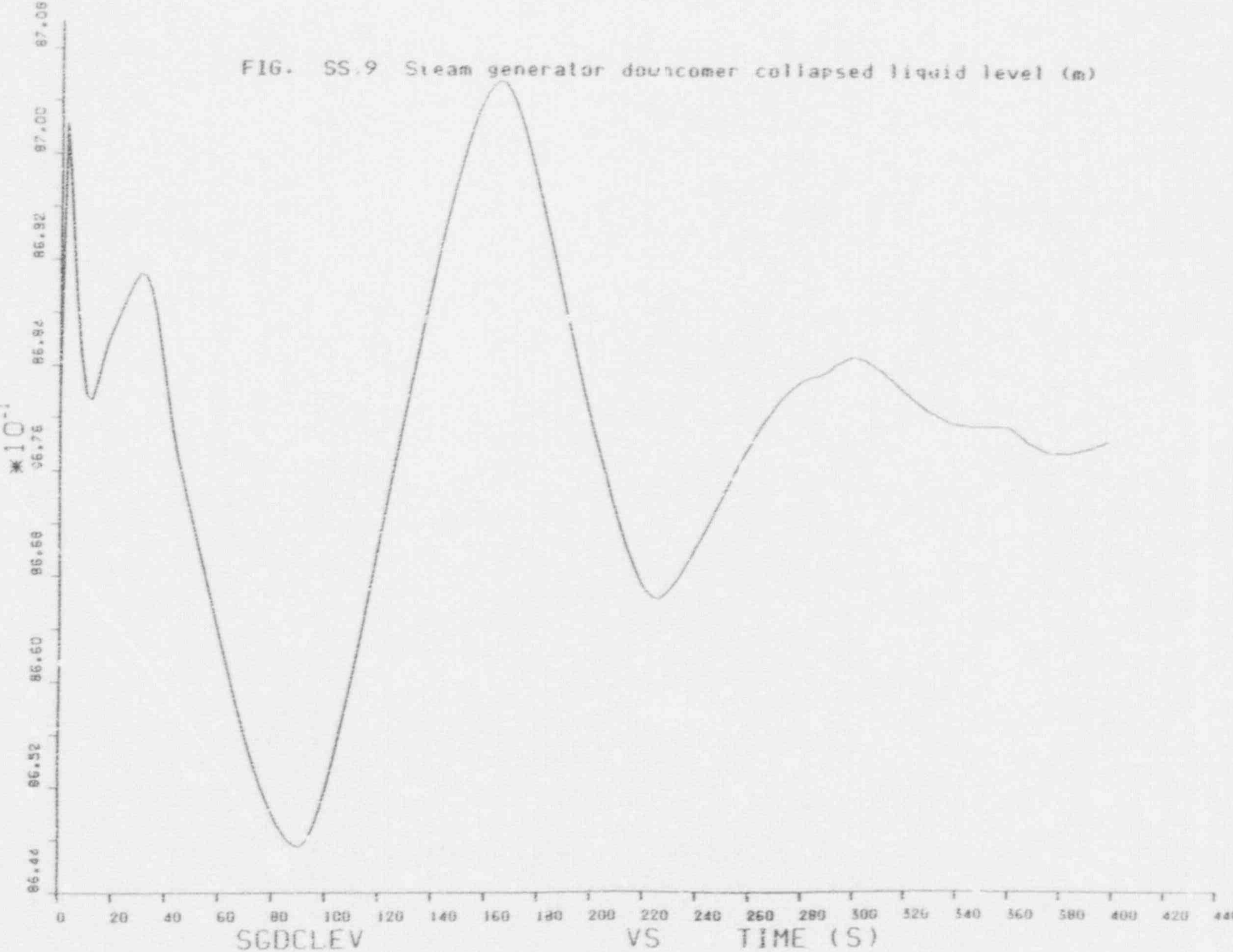


FIG. SS.9 Steam generator downcomer collapsed liquid level (m)

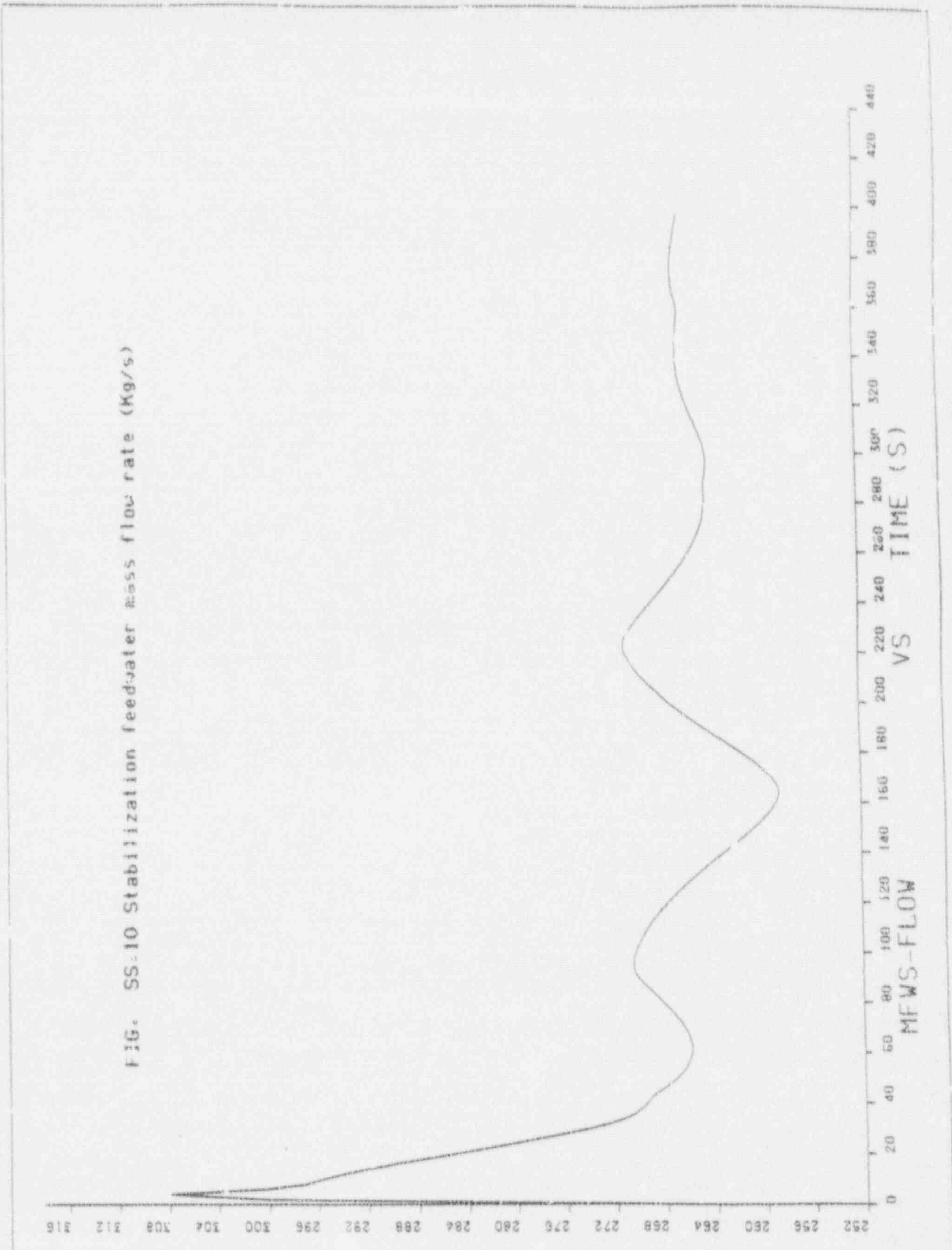


FIG. SS.10 Stabilization feedwater mass flow rate (Kg/s)

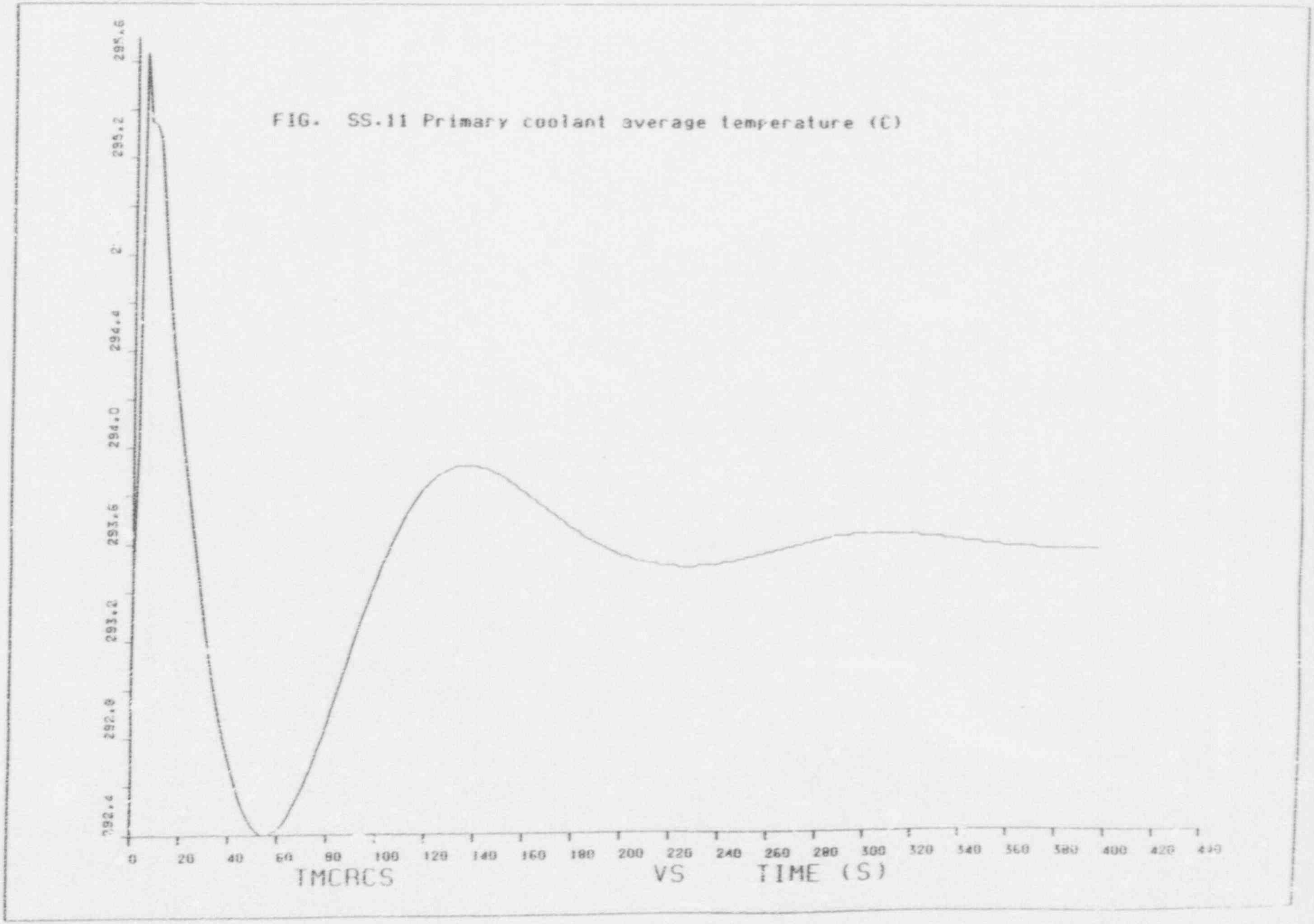


FIG. SS.11 Primary coolant average temperature (C)

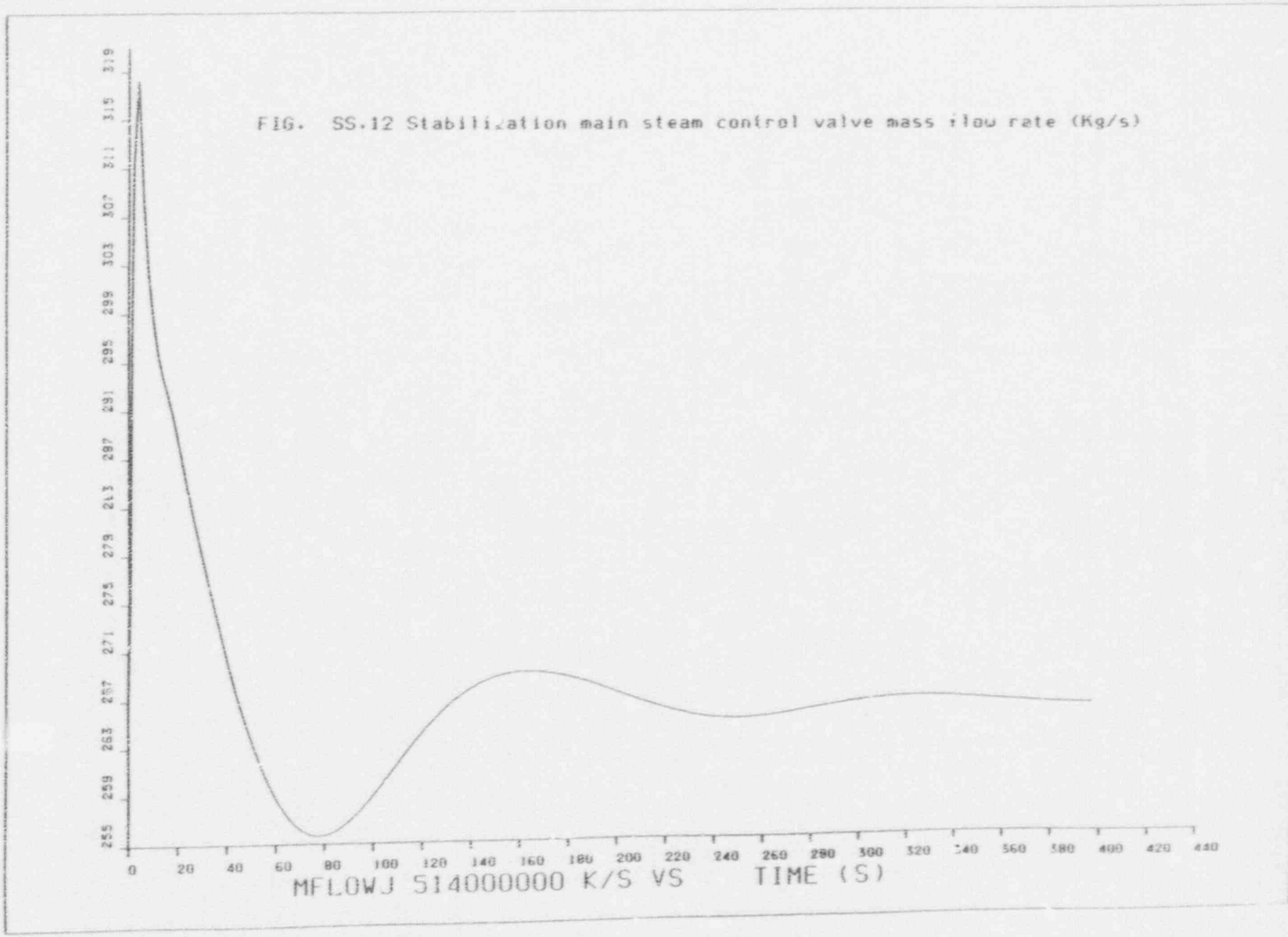


FIG. SS.12 Stabilization main steam control valve mass flow rate (Kg/s)

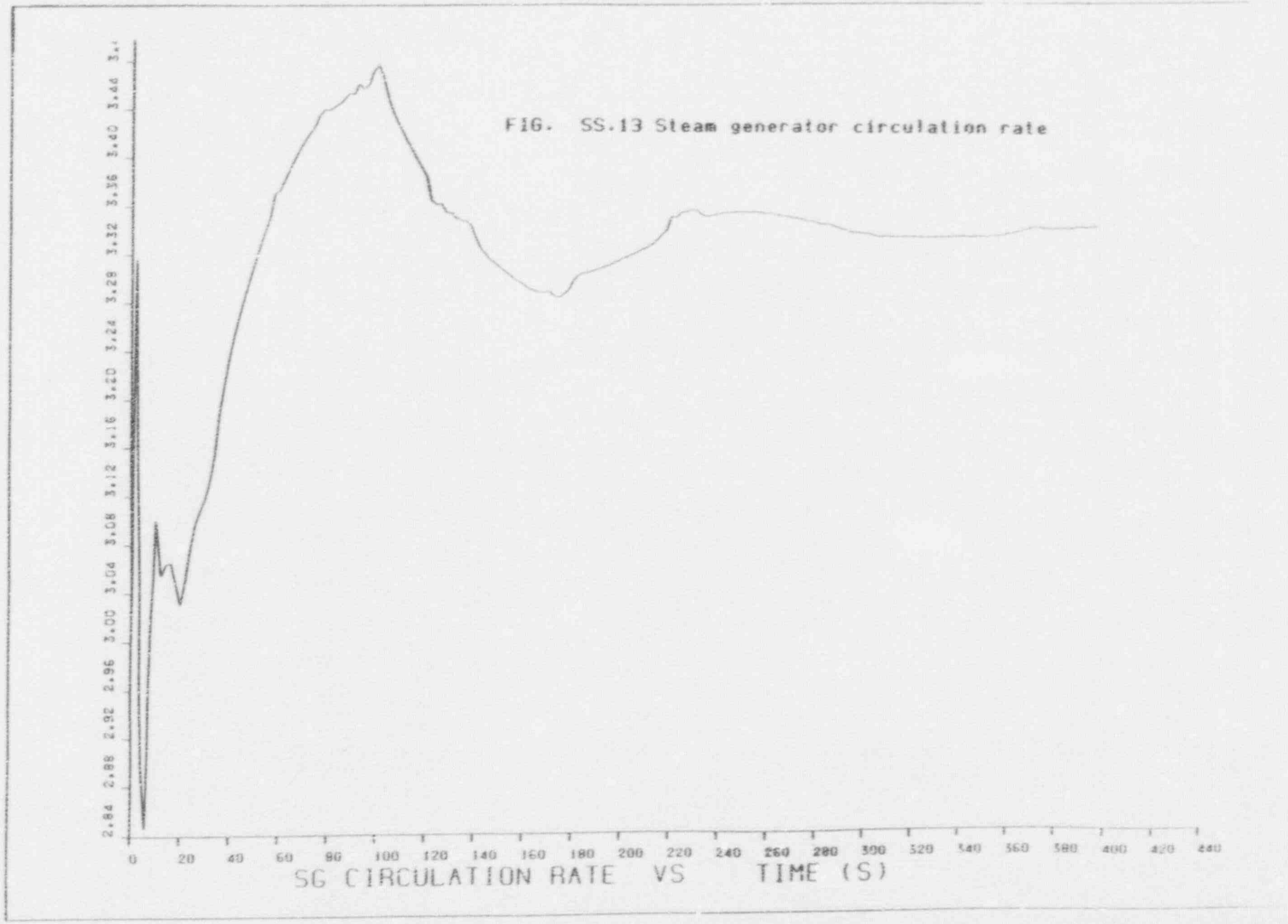


FIG. SS.13 Steam generator circulation rate

APPENDIX I : FIGURES OF CASE I(A), (6" WITHOUT RECOVERY)

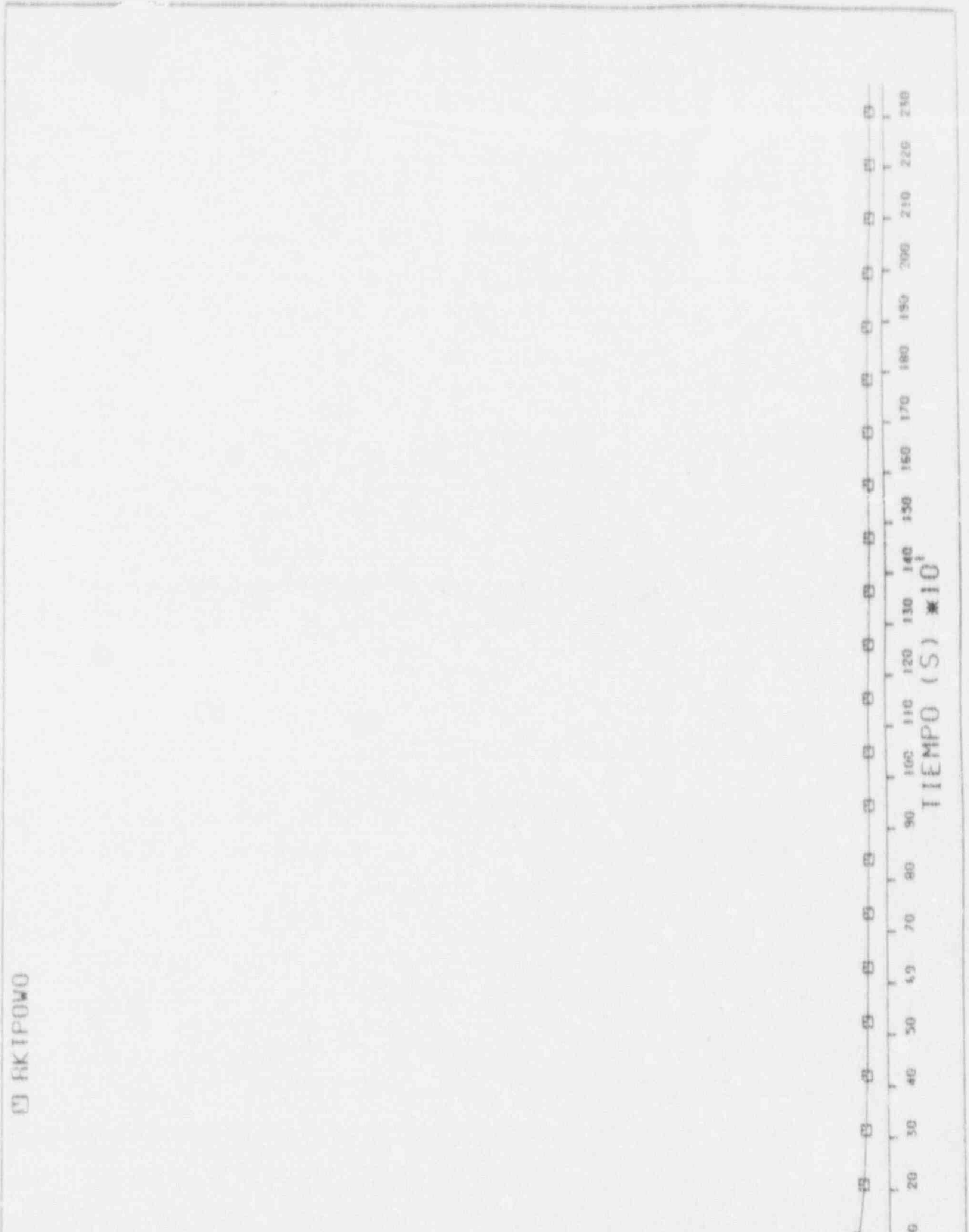

RKTPOWO
POTENCIA DEL REACTOR (W) *10^6
0 40 80 120 160 200 240 280 320 360 400 440 480 520 560 600
TIEMPO (S) *10^1
0 10 20 30 40 50 60 70 80 90 100 110 120 130 140 150 160 170 180 190 200 210 220 230
ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 1)

FIG. I (A) - 1

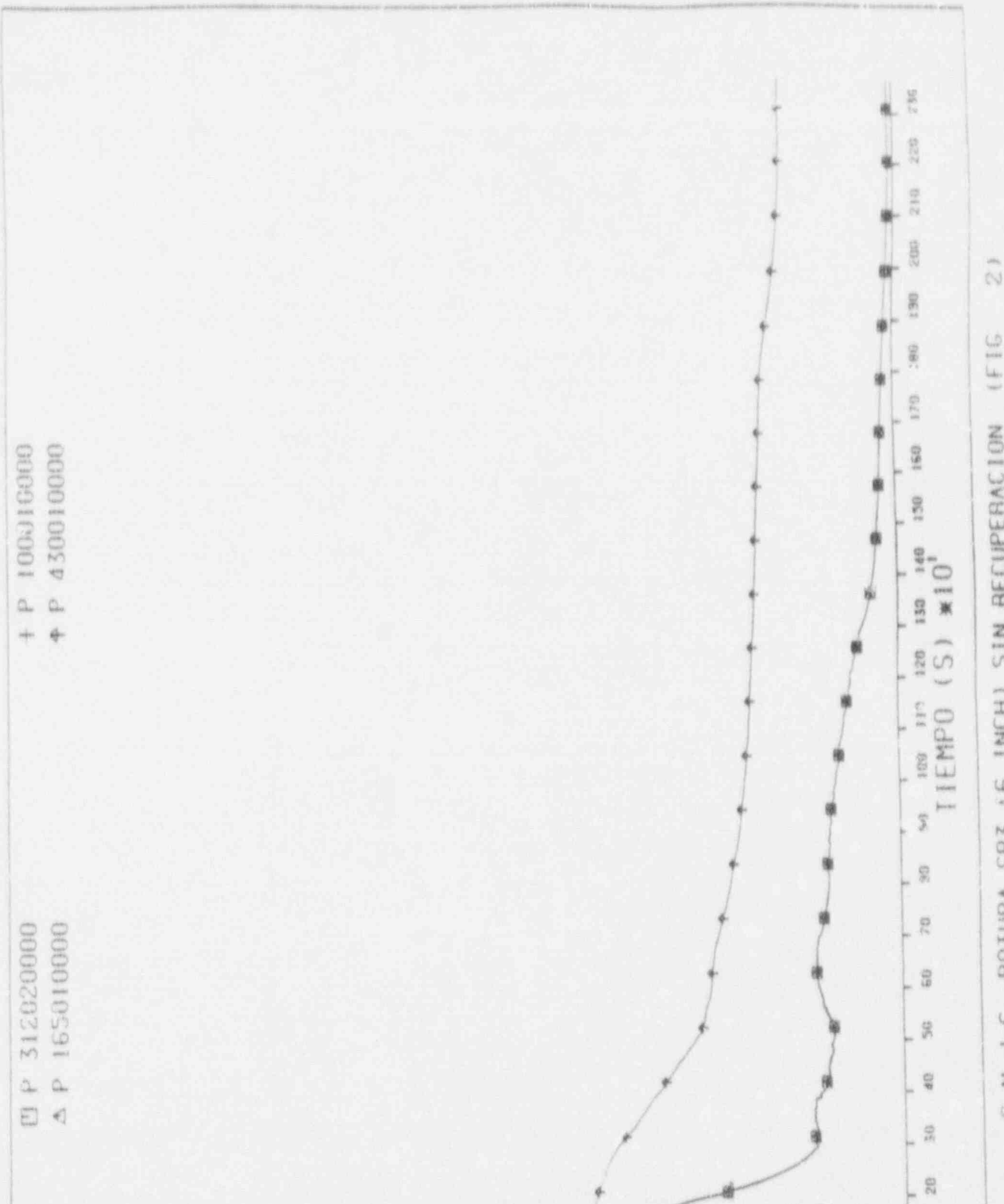

□ P 312020000
Δ P 165010000
+ P 100010000
◆ P 430010000
PRESION (PA) *10
TIEMPO (S) *10
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 2)

FIG. I (A) - 2

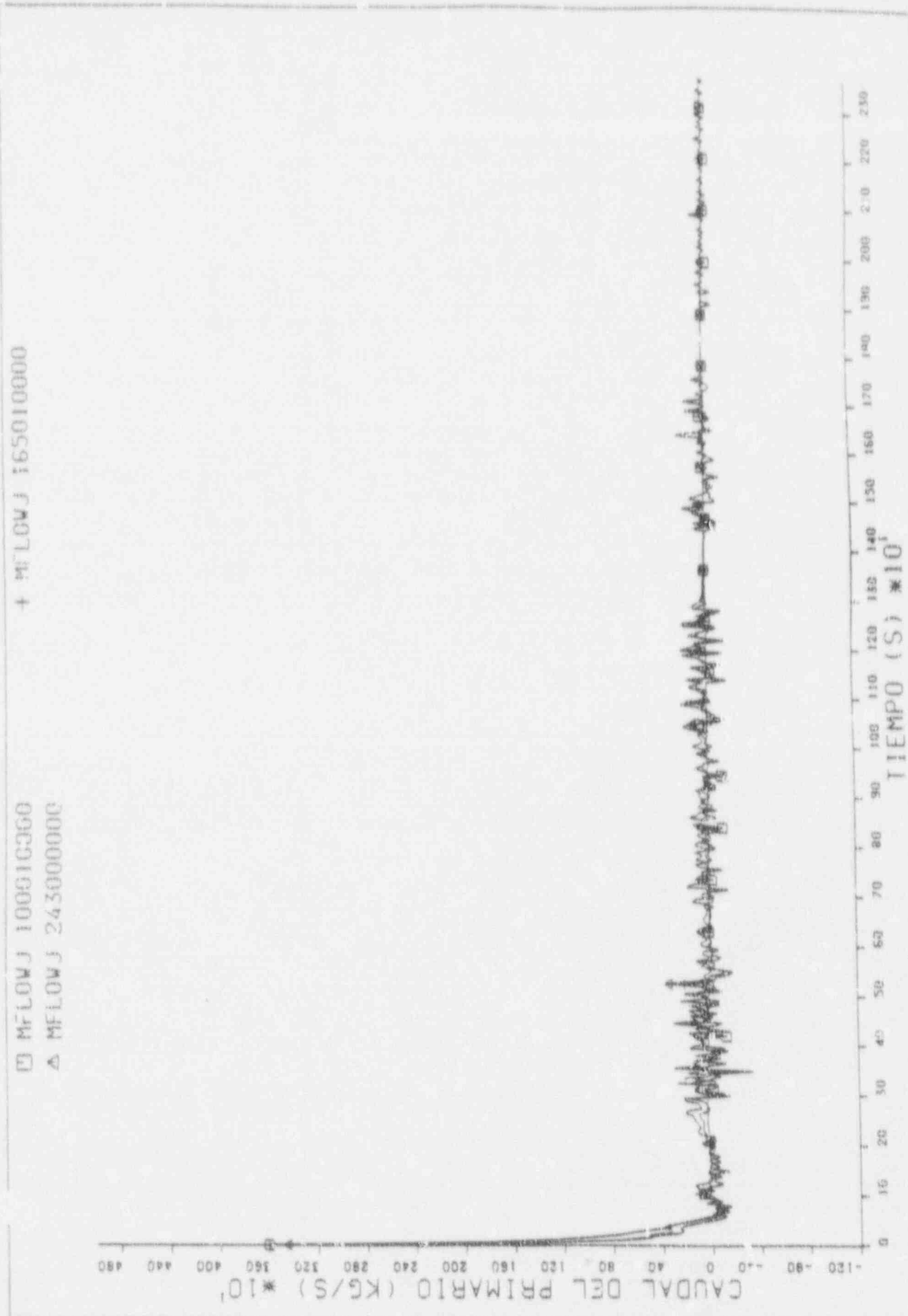

CAUDAL DEL PRIMARIO (KG/S) *10¹
TIEMPO (S) *10¹
C.N. J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 3)

FIG. I (A) - 3

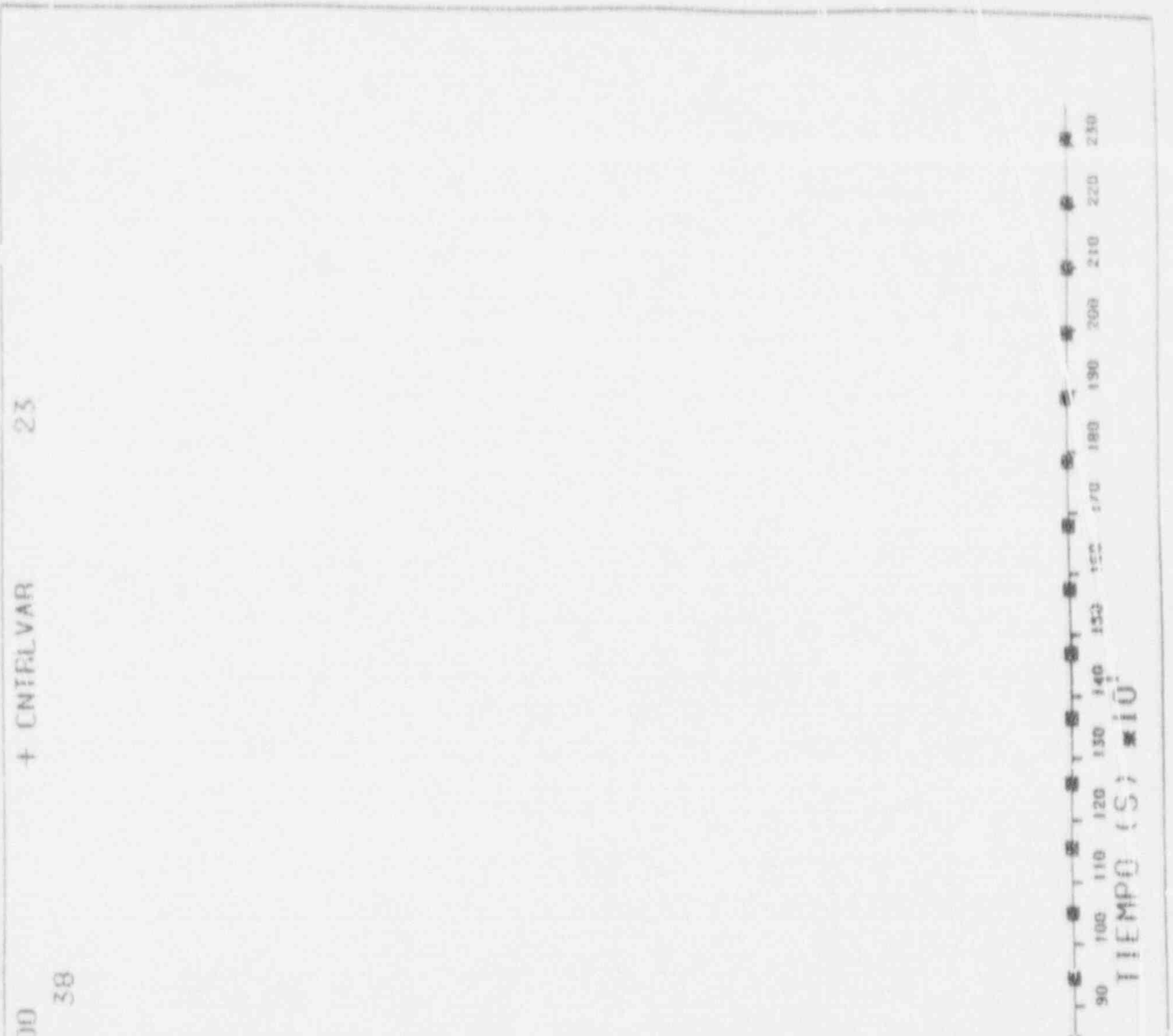
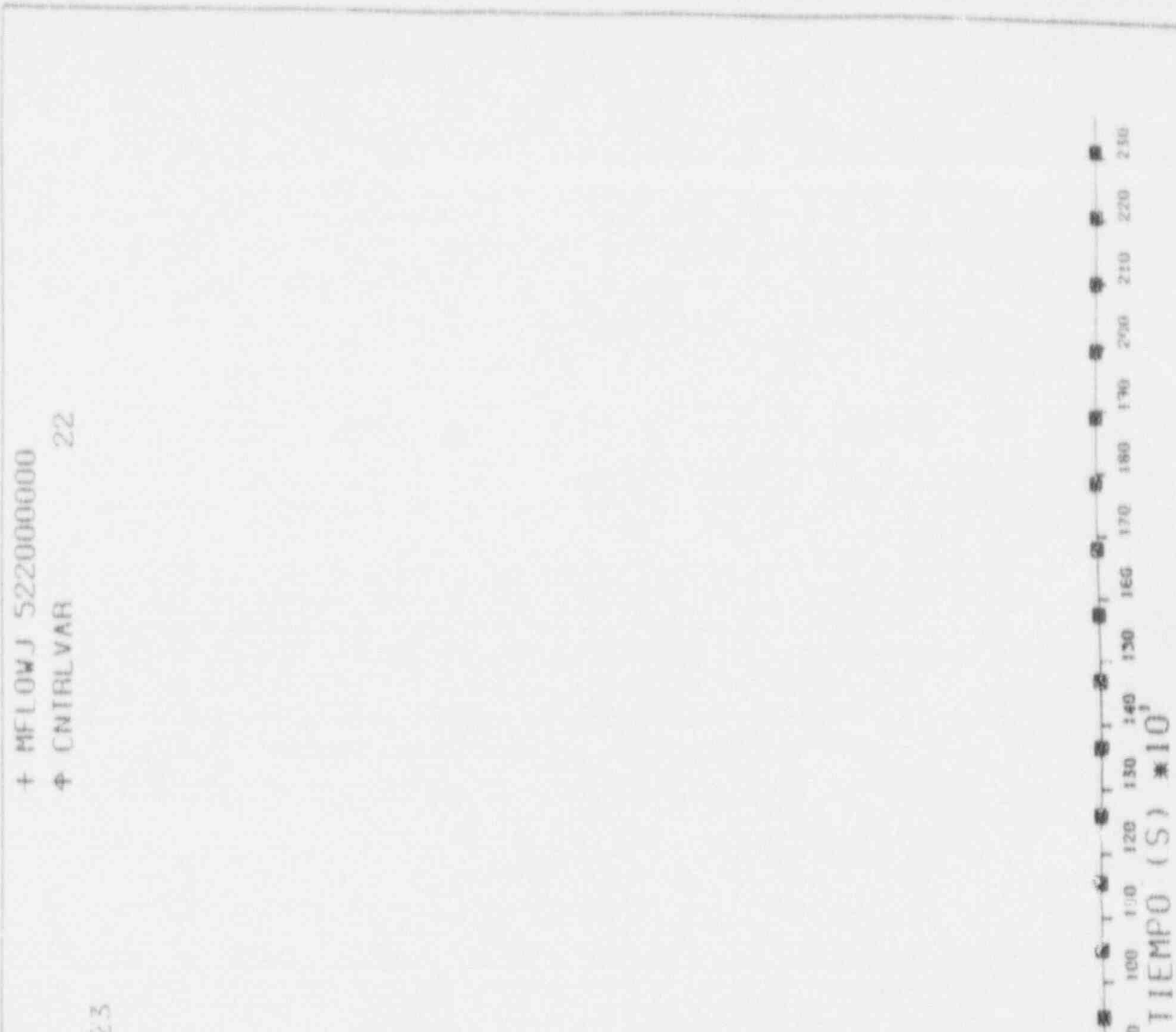
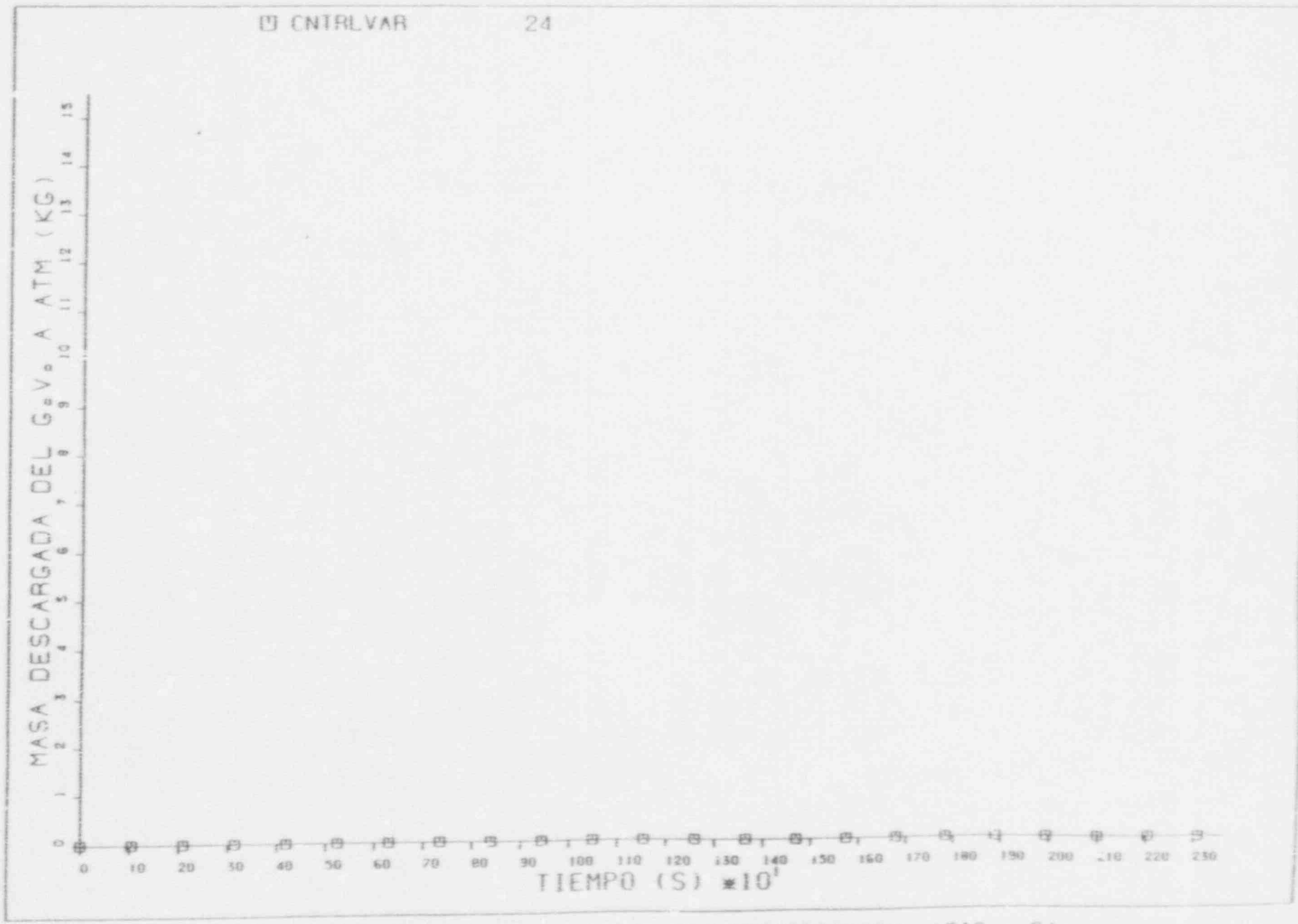

MFLOWJ 514000000
CNTRLVAR 38
+ CNTRLVAR 23
CAUDALES DE SECUNDARIO (KG/S)
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 4)

FIG. I (A) - 4



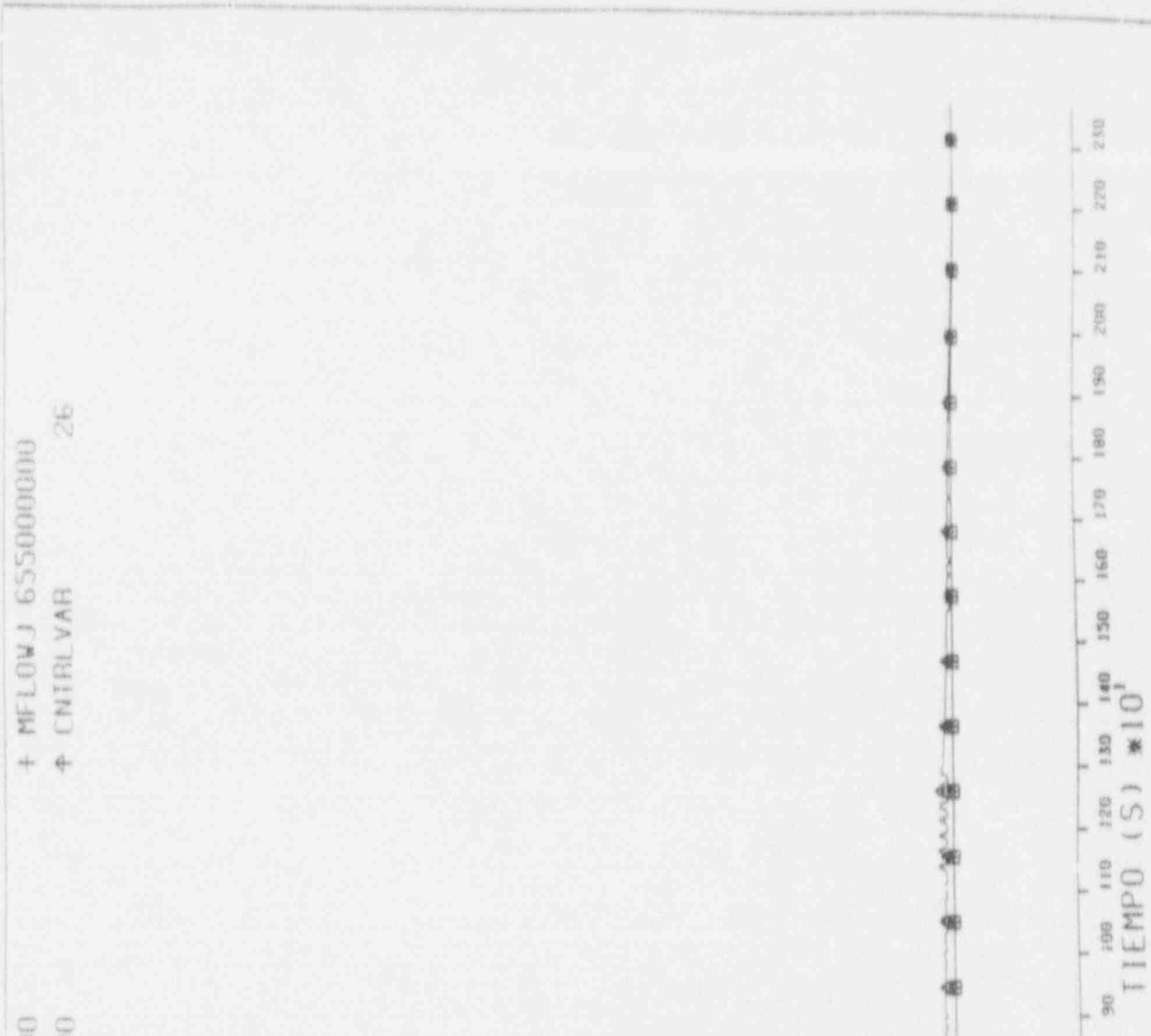
MFLOWJ 536000000
MFLOWJ 526000000
CNTRLVAR 23
MFLOWJ 522000000
CNTRLVAR 22
ALIVIO Y SEGURIDAD G.V. (KG/S)
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 5)

FIG. I (A) - 5



C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 6)

FIG. I (A) - 6



MFLOWJ 630000000
MFLOWJ 163000000
MFLOWJ 655000000
CNTRLVAR 26
CAUDAL DE ROTURA Y ECCS (KG/S)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 7)

FIG. I (A) - 7

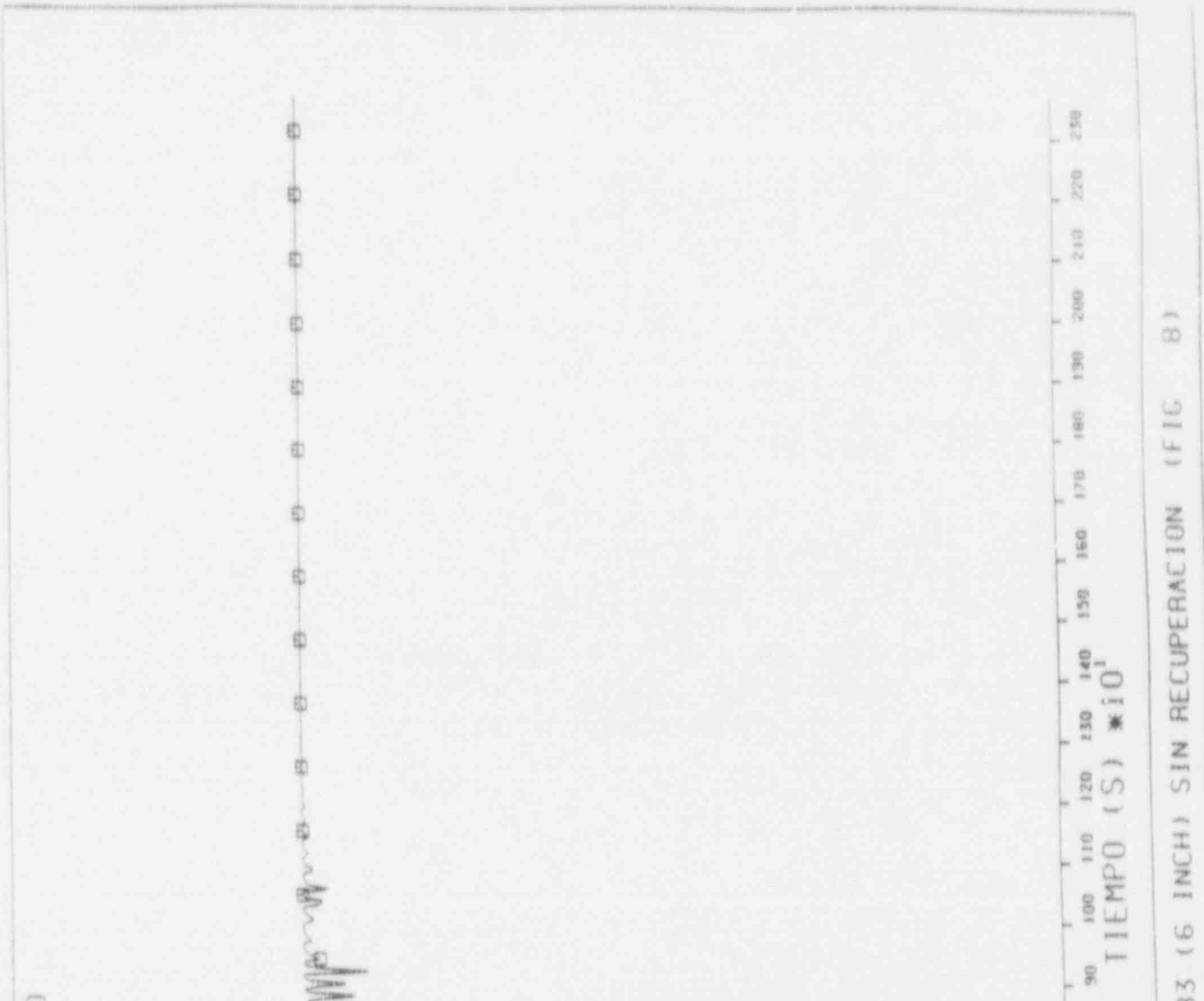

VOIDGJ 163000000
FRACCION DE VAPOR EN LA ROTURA (O/1)
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 8)

FIG. I (A) - 8

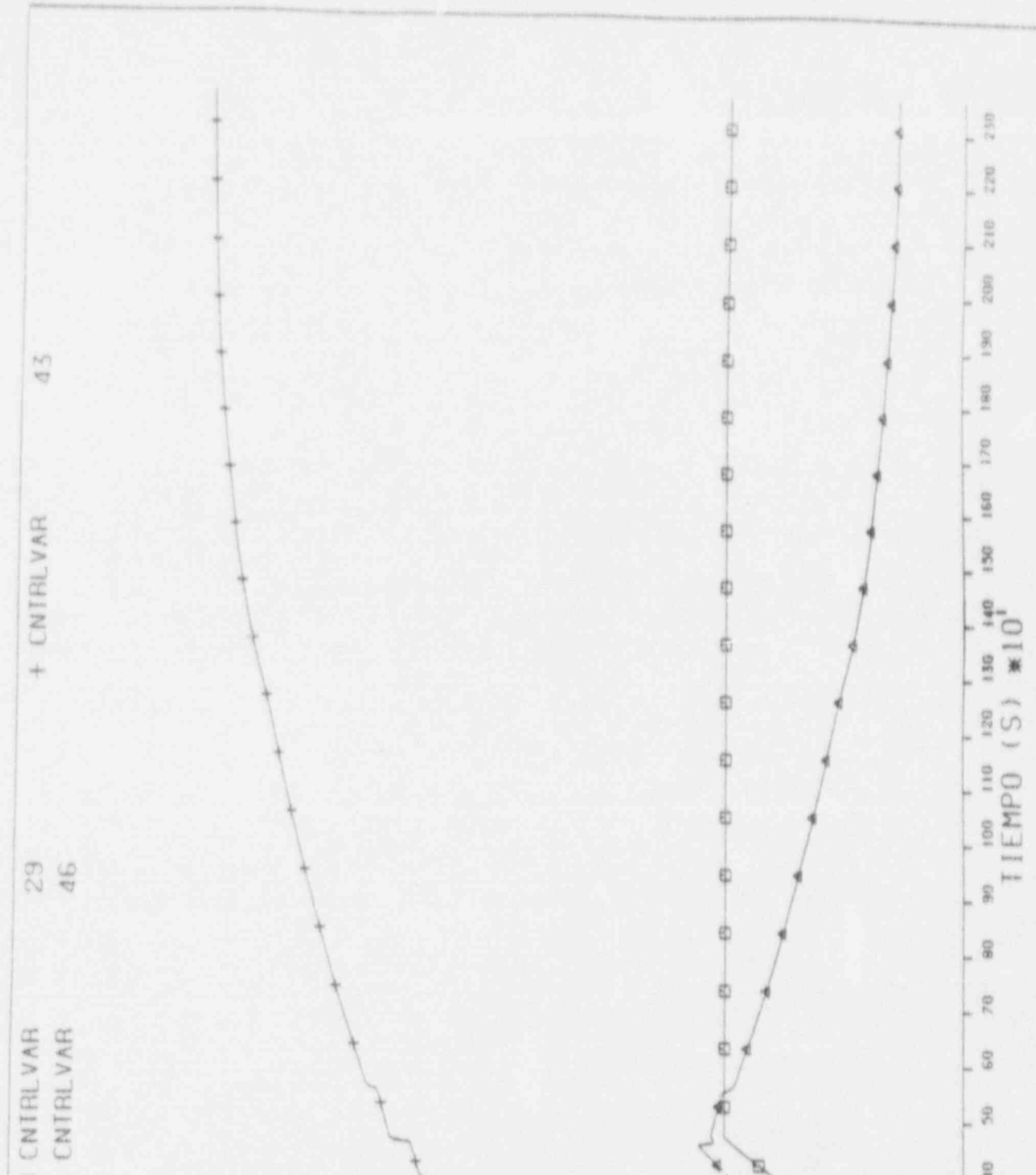

□ CNTRLVAR 29
△ CNTRLVAR 46
+ CNTRLVAR 43
BALANCE MASICO DEL PRIMARIO (KG) *10²
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 9)

FIG. I (A) - 9

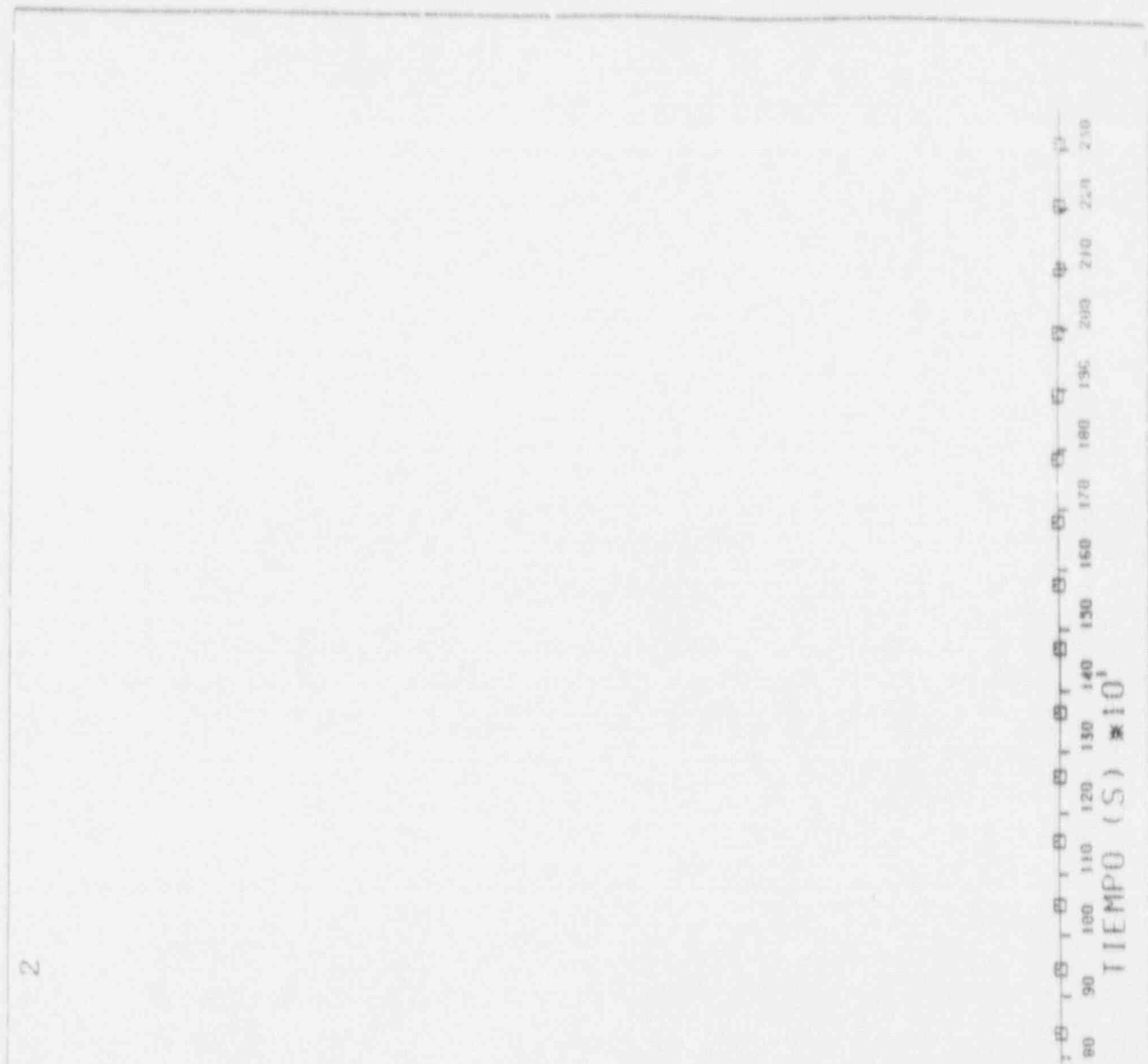

CNTRLVAH 2
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 10)
TIEMPO (S) *10[1]
NIVEL COLAPSADO EN PRESIONADOR (%)
0 5 10 15 20 25 30 35 40 45 50 55 60 65 70 75
0 10 20 30 40 50 60 70 80 90 100 110 120 130 140 150 160 170 180 190 200 210

FIG. I (A) - 10

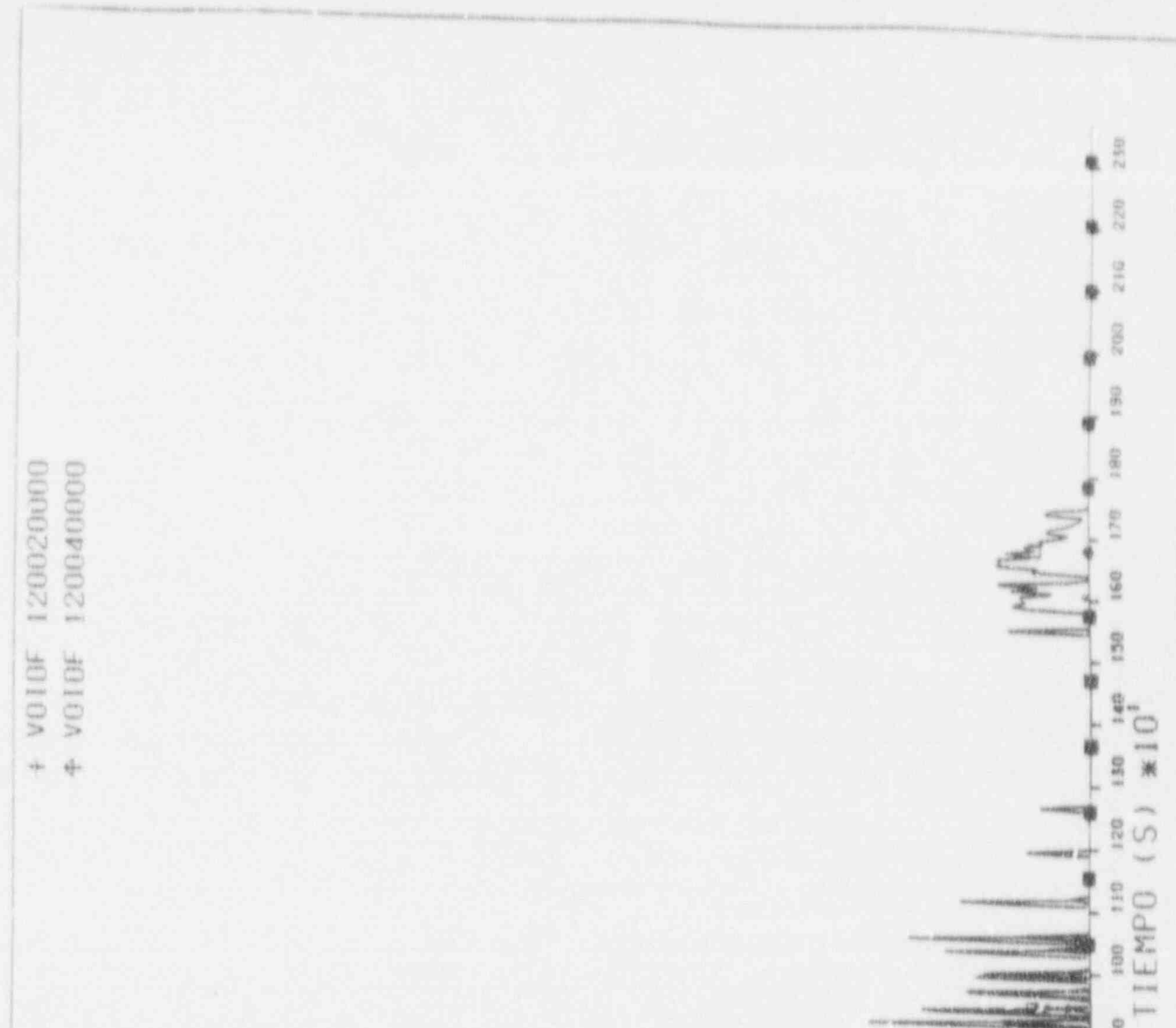
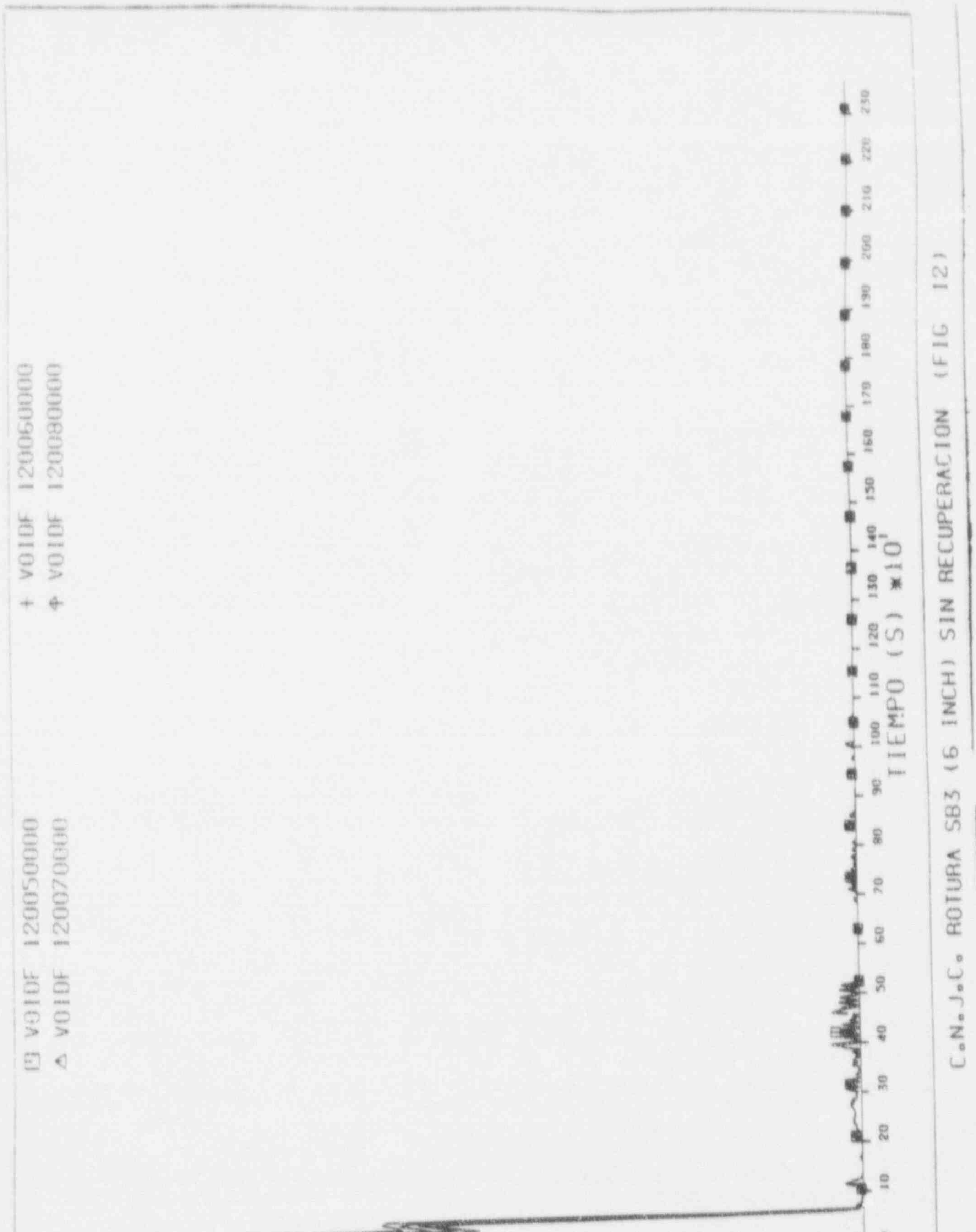
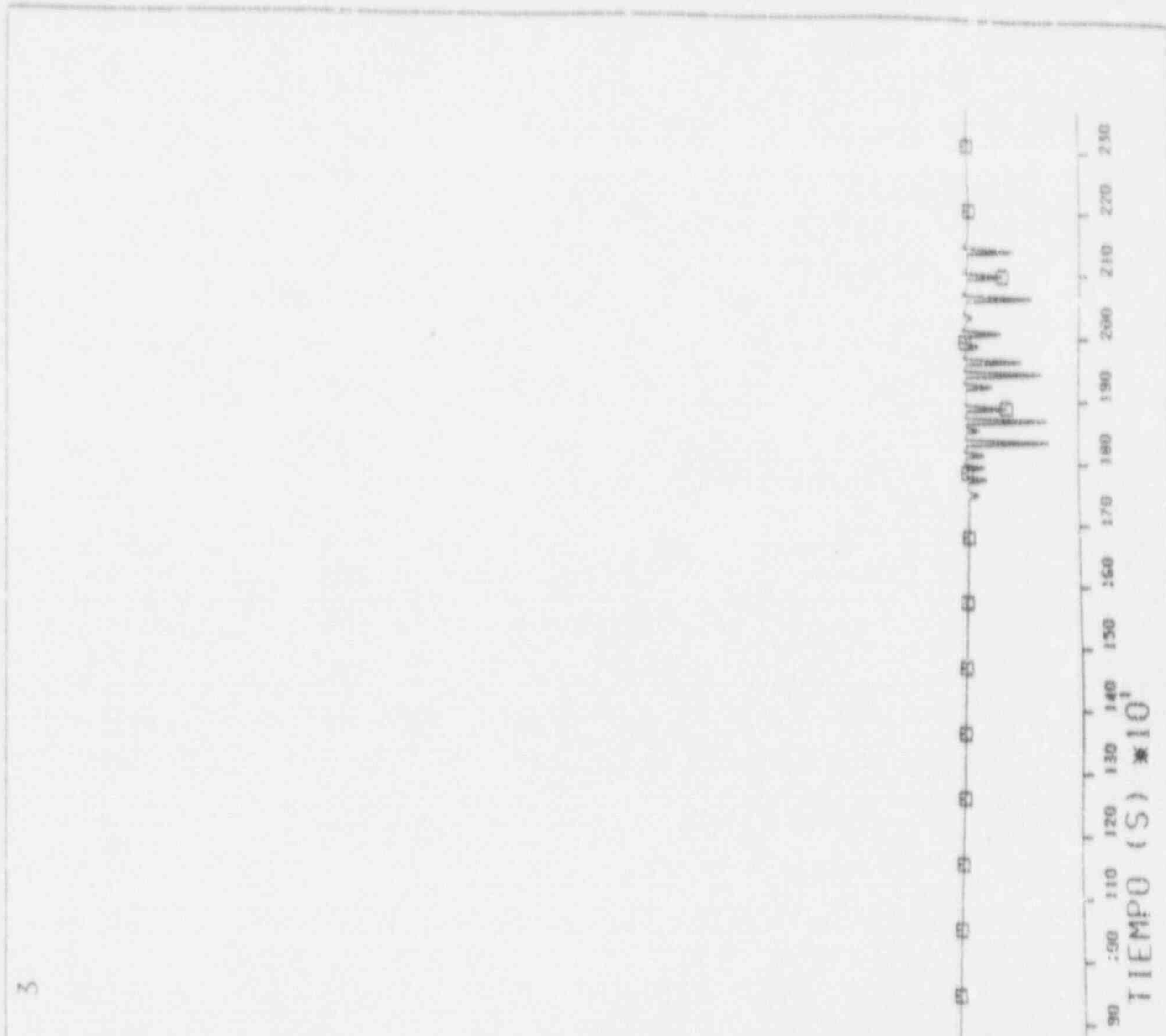

VOIDF 120010000
VOIDF 120020000
VOIDF 120030000
VOIDF 120040000
FRAC. DE LIQUIDO EN TUBO UP G.V. (O/I)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 11)

FIG. I (A) - 11



□ VOIDF 120050000
△ VOIDF 120070000
+ VOIDF 120060000
✧ VOIDF 120080000
FRAC. DE LIQUIDO EN TUBO DOWN G.V. (O/1)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 12)

FIG. I (A) - 12



CNTRLVAR 3
TIEMPO (S) *10¹
NIVEL EN DOWNCOMER DEL G.V. (M)
ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 13)

FIG. I (A) - 13

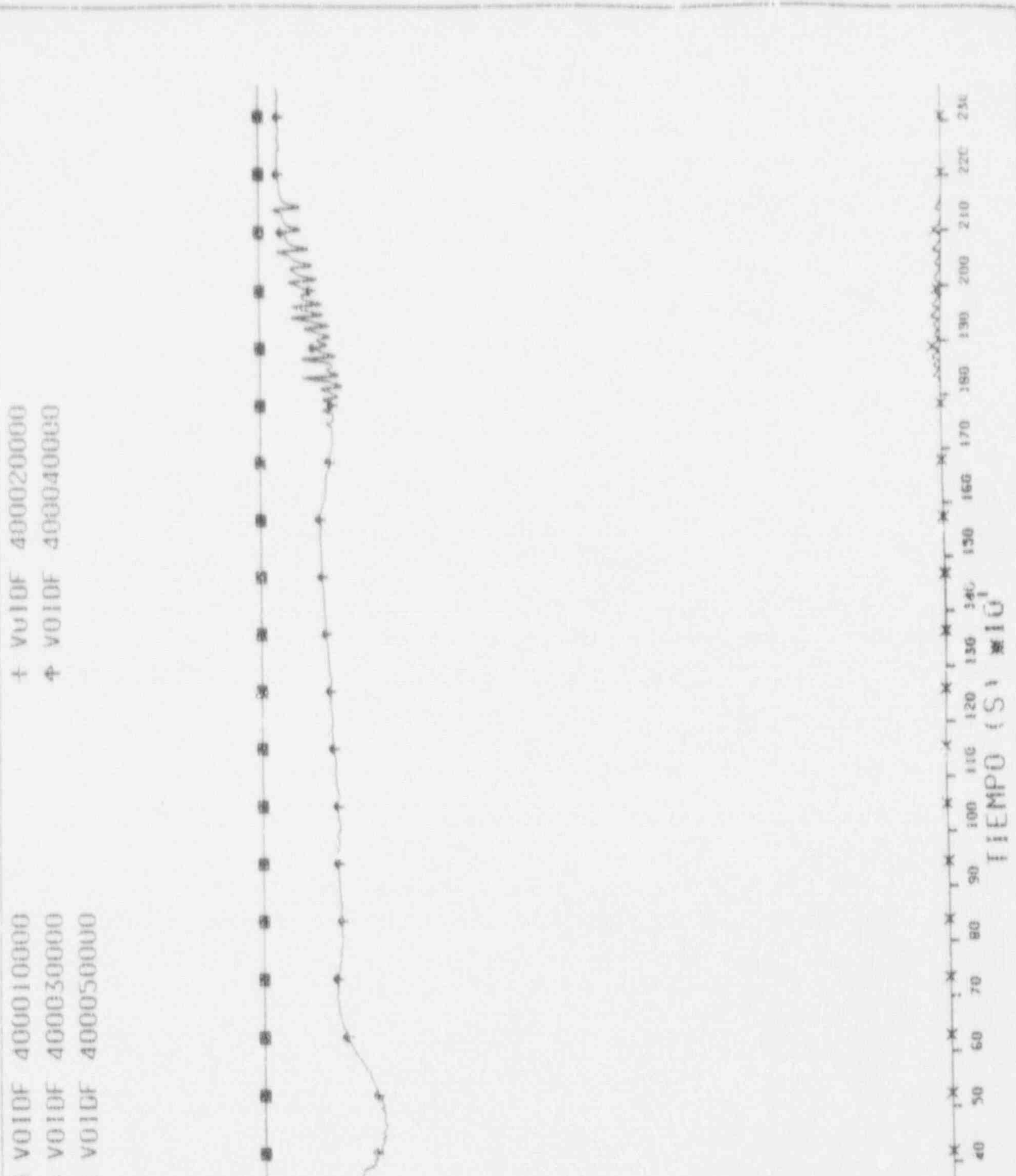
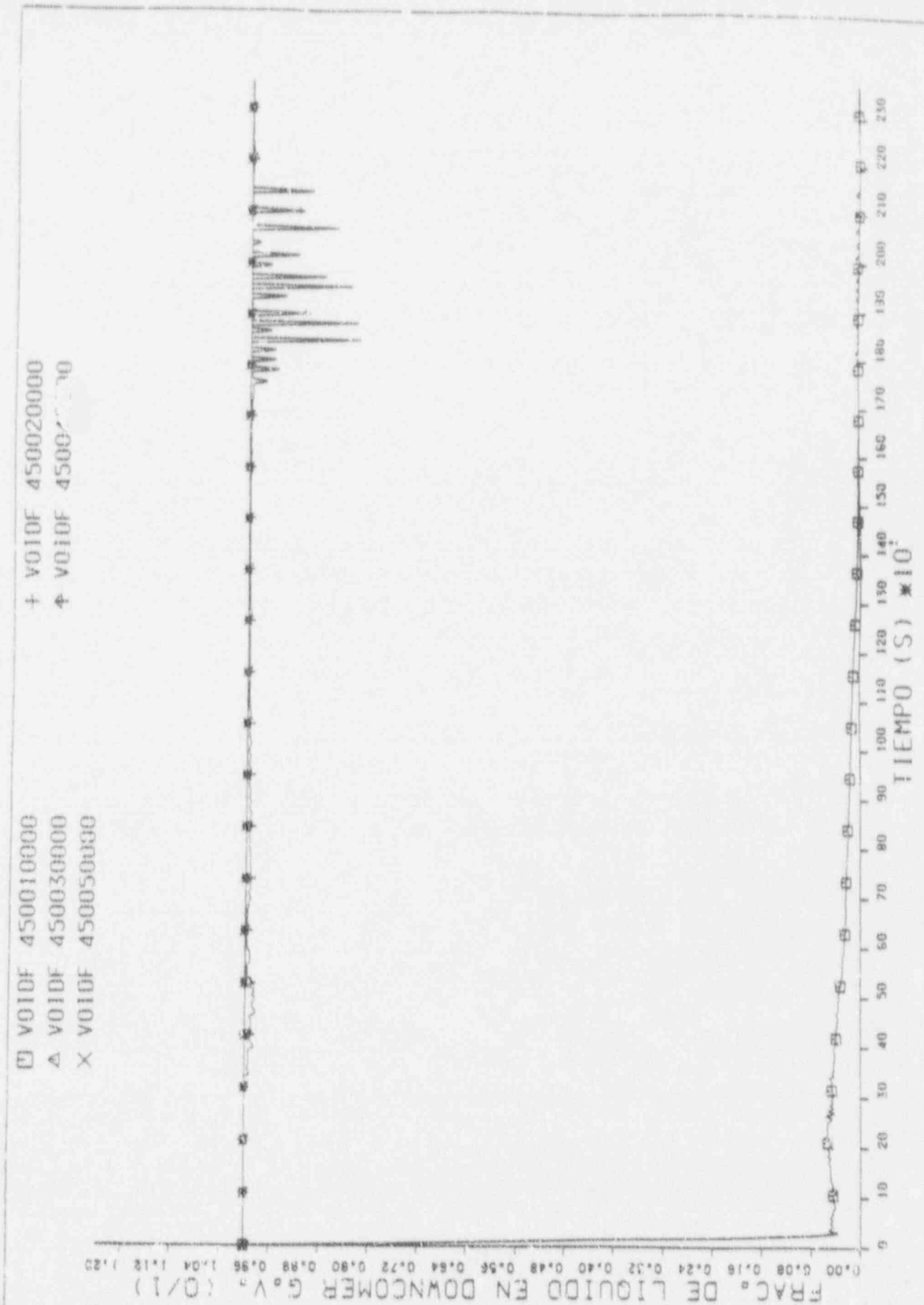

VOIDF 400010000
VOIDF 400030000
VOIDF 400050000
VOIDF 400020000
VOIDF 400040000
FRAC. DE LIQUIDO EN RISER G.V. (O/I)
TIEMPO (S) *10
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 14)

FIG. I (A) - 14



VOIDF 450010000
VOIDF 450030000
VOIDF 450050000
VOIDF 450020000
VOIDF 4500
FRAC. DE LIQUIDO EN DOWNCOMER G.V. (O/1)
TIEMPO (S) *10
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 15)

FIG. I (A) - 15

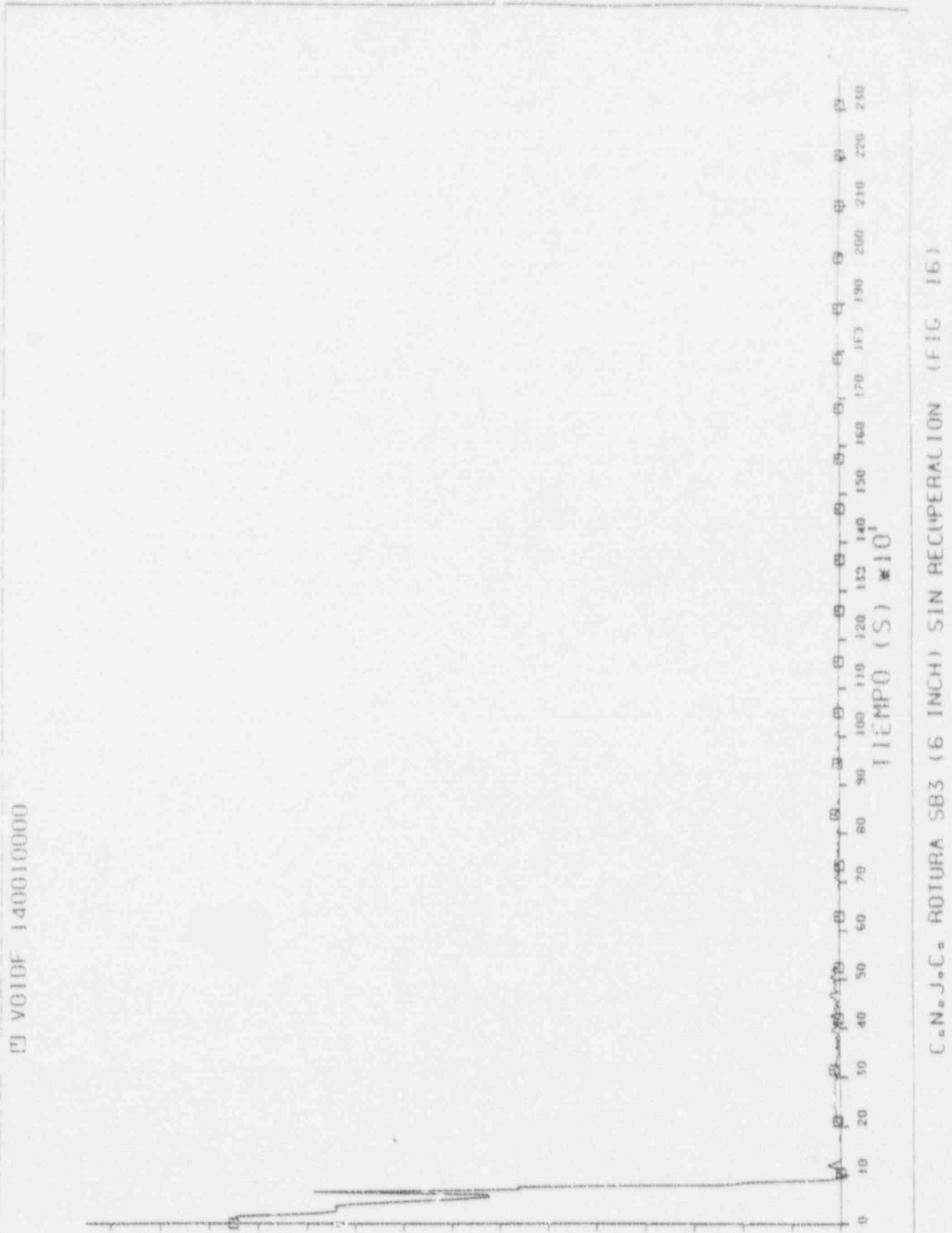

VOIDF 140010000
NIVEL COLAPSADO EN SELLO -140-
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 16)

FIG. I (A) - 16

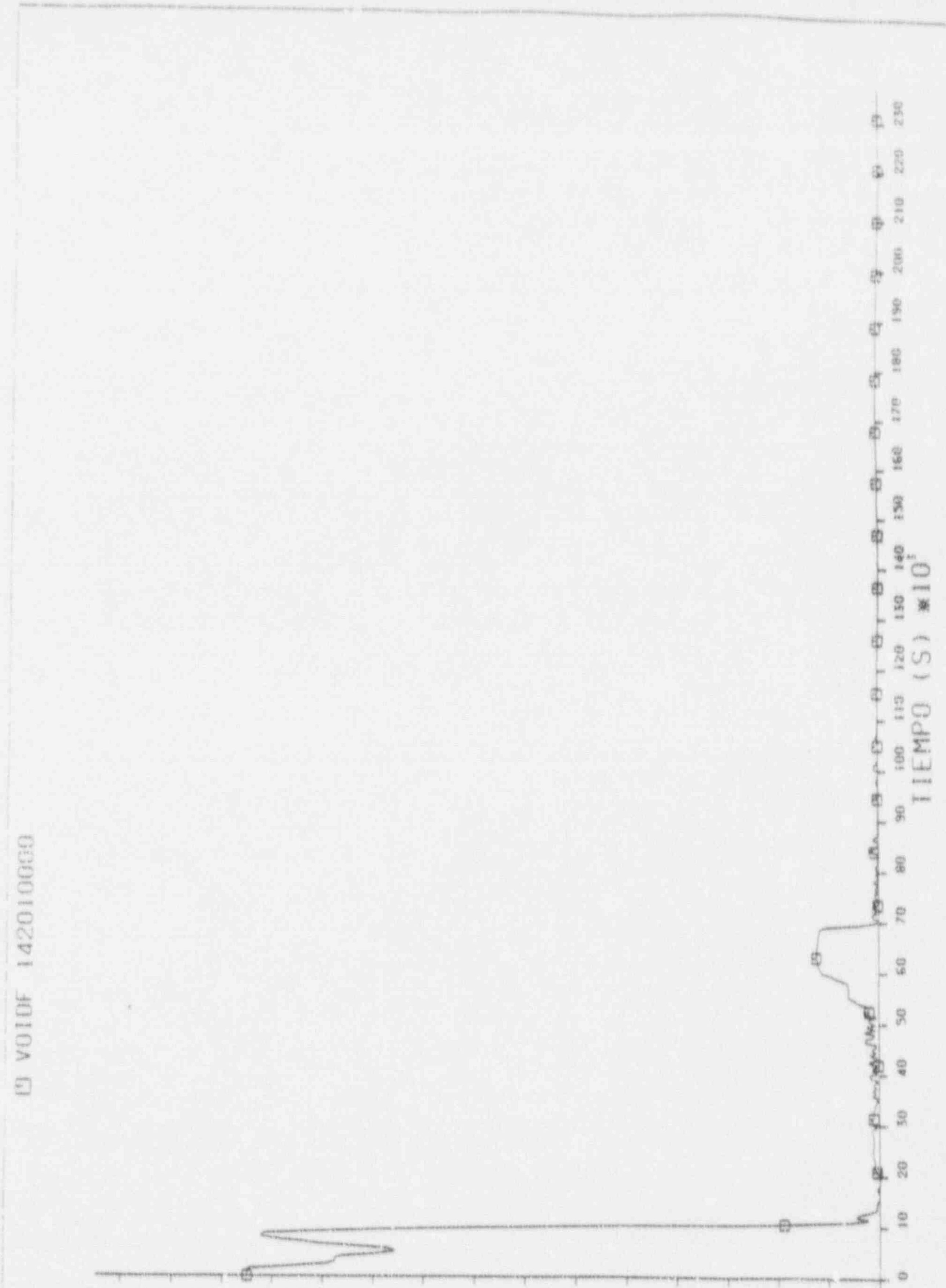
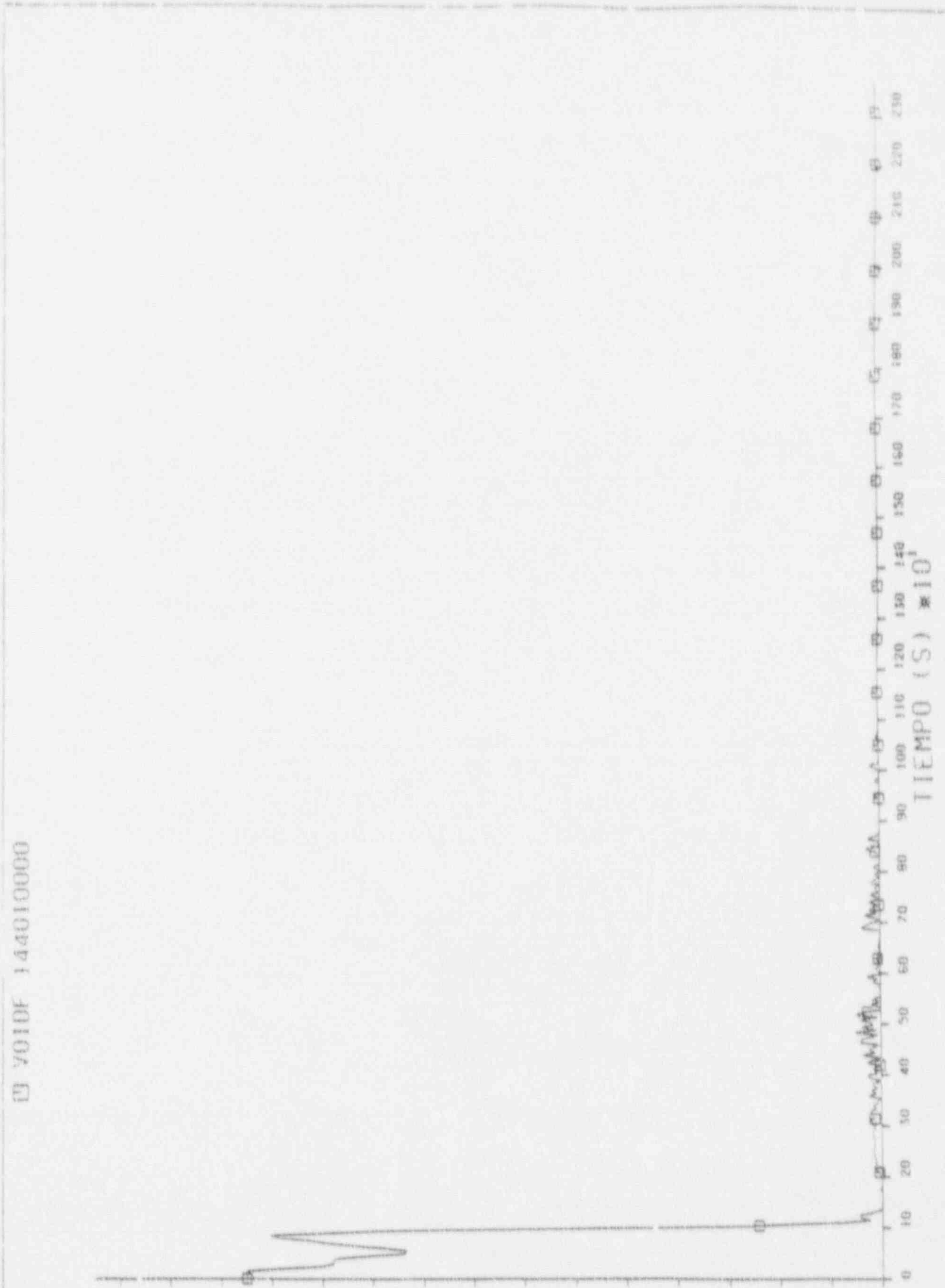
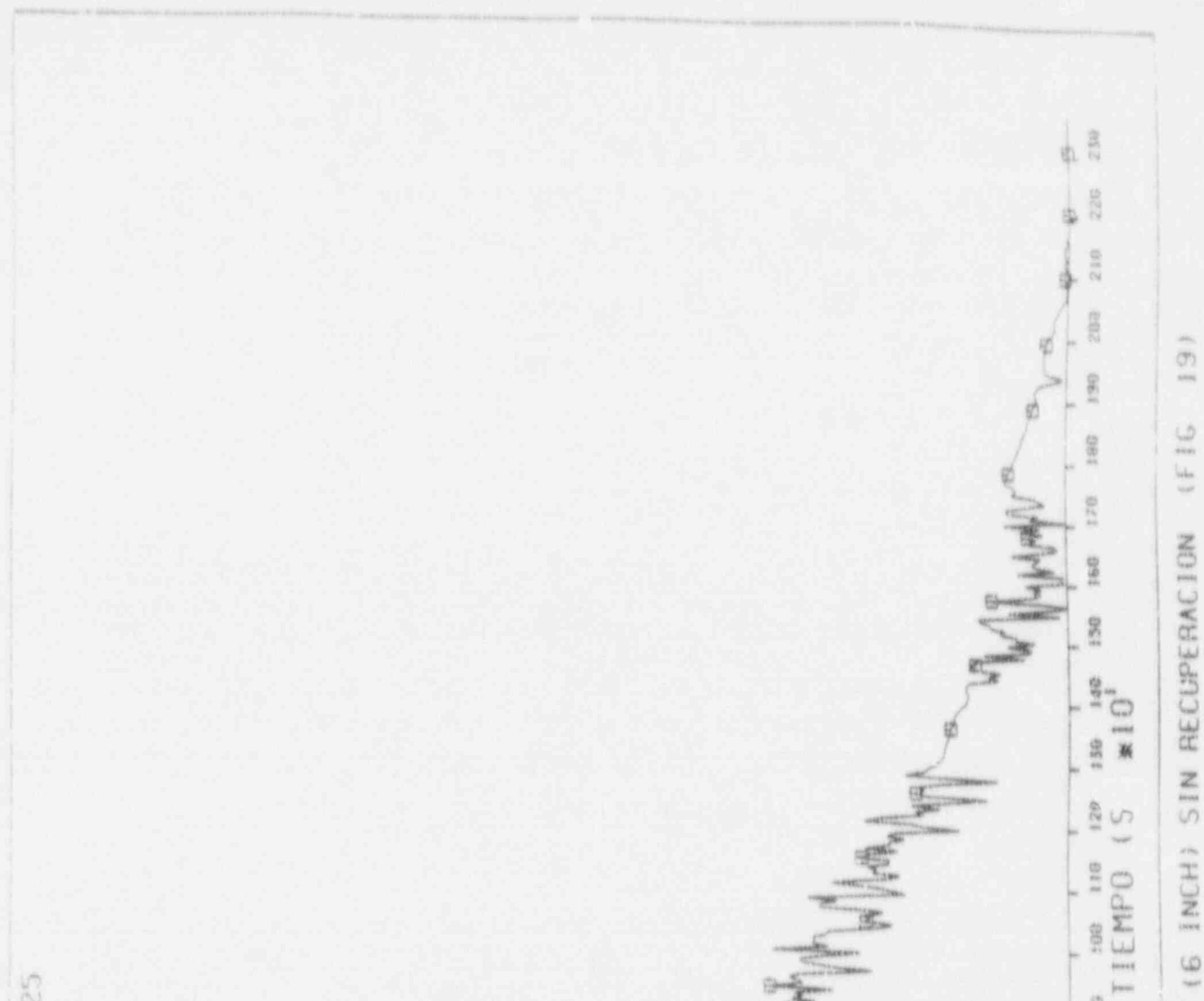

VOIDF 142010000
NIVEL COLAPSADO EN SELLO -142-
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 17)

FIG. I (A) - 17



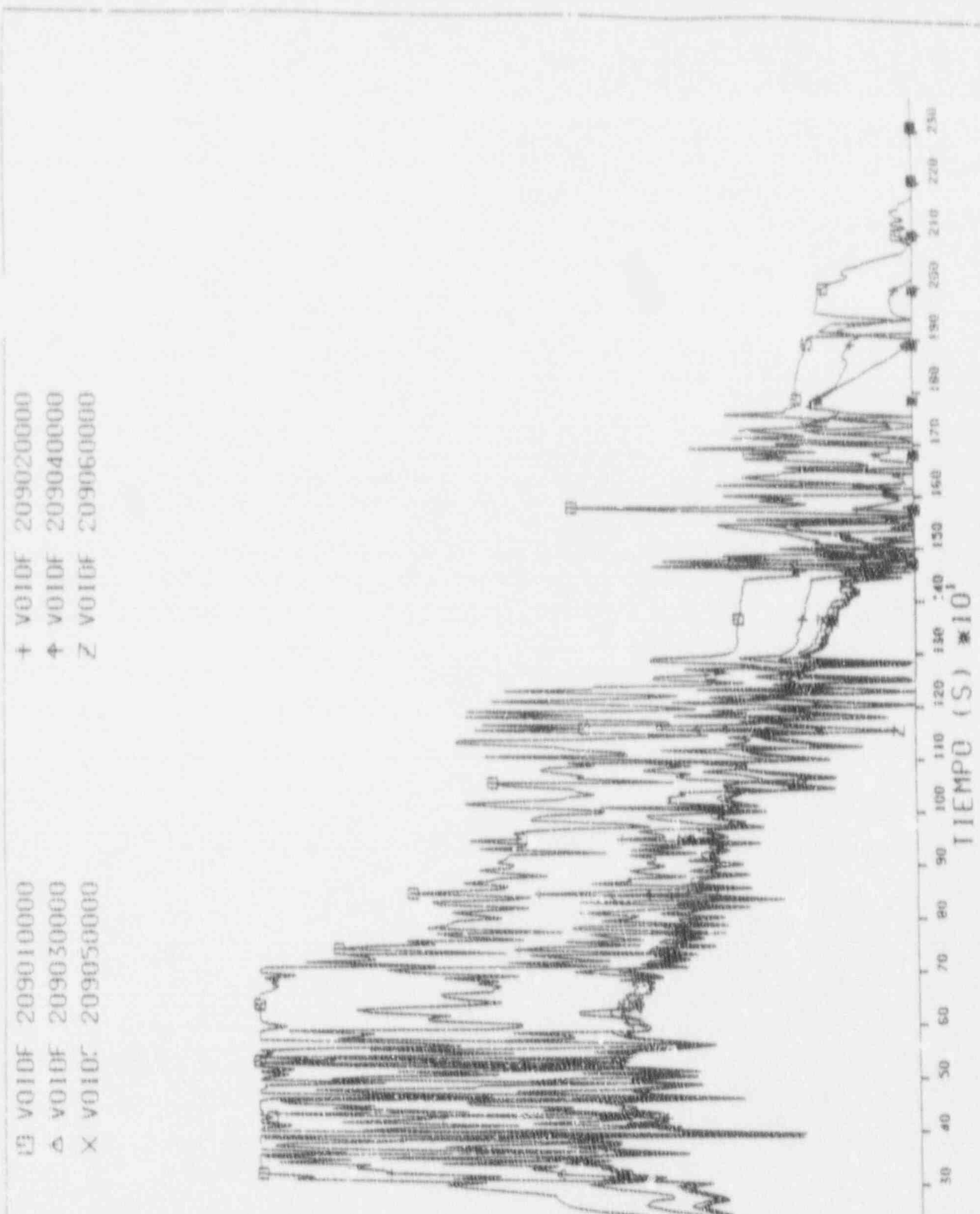
VOIDF 144010000
NIVEL COLAPSADO EN SELLO -144-
0.00 0.08 0.16 0.24 0.32 0.40 0.48 0.56 0.64 0.72 0.80 0.88 0.96 1.04 1.12 1.20
TIEMPO (S) *10¹
0 10 20 30 40 50 60 70 80 90 100 110 120 130 140 150 160 170 180 190 200 210 220 230
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 18)

FIG. I (A) - 18



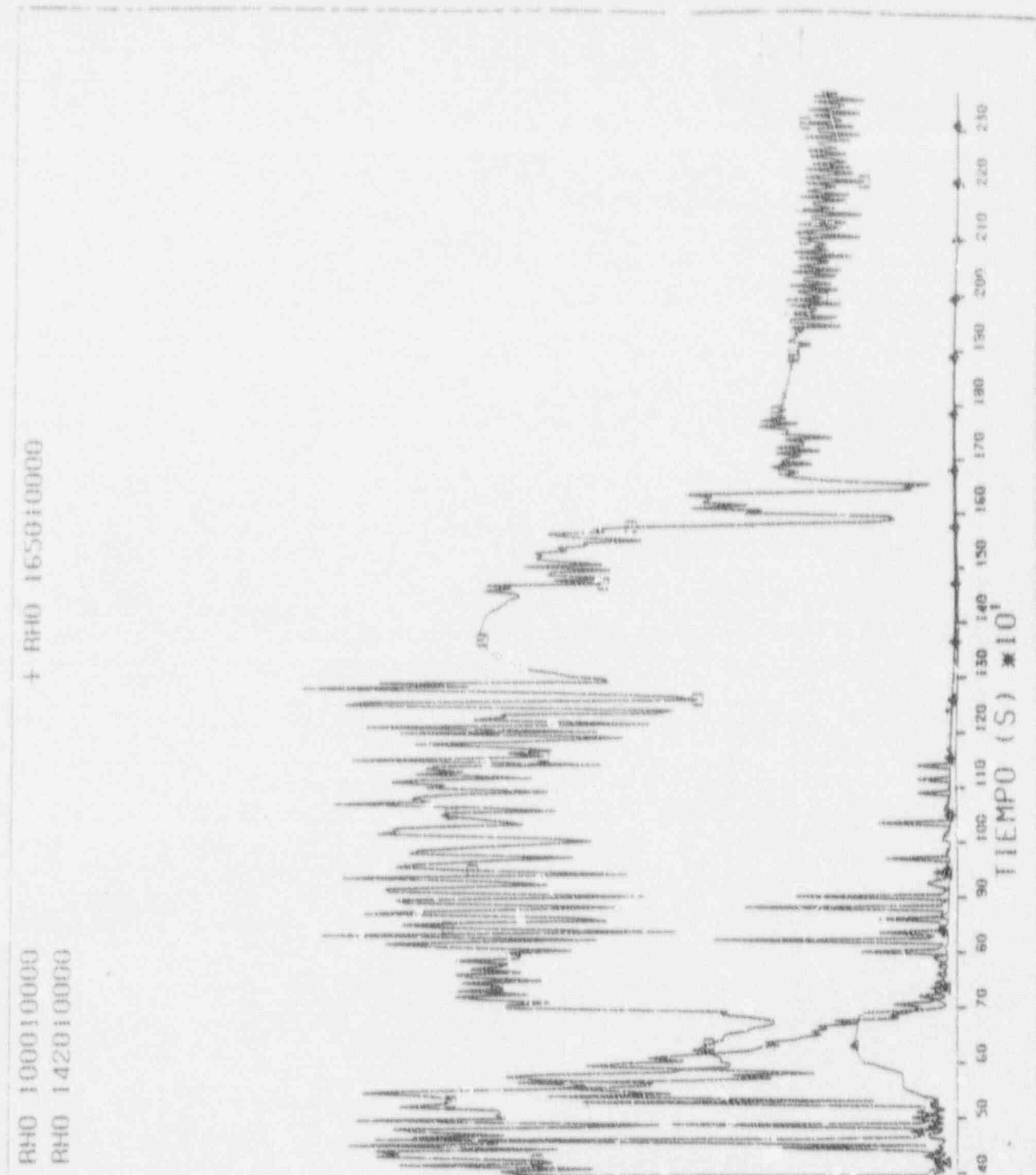
CNTRLVAR 25
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 19)
TIEMPO (S) *10¹
NIVEL COLAPSADO EN EL NUCLEO (M)
0 10 20 30 40 50 60 70 80 90 100 110 120 130 140 150 160 170 180 190 200 210 220 230
0.0 0.2 0.4 0.6 0.8 1.0 1.2 1.4 1.6 1.8 2.0 2.2 2.4 2.6 2.8 3.0

FIG. I (A) - 19



⊡ VOIDF 209010000
△ VOIDF 209030000
X VOIDF 209050000
+ VOIDF 209020000
◆ VOIDF 209040000
Z VOIDF 209060000
FRACCIONES DE LIQUIDO EN EL NUCLEO (O/I)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 20)

FIG. I (A) - 20



RHO 100010000
RHO 142010000
RHO 165010000
DENSIDADES DEL PRIMARIO (KG/M3)
TIEMPO (S) *10¹
C.N. J.C. ROTURA SB7 (6 INCH) SIN RECUPERACION (FIG 21)

FIG. I (A) - 21

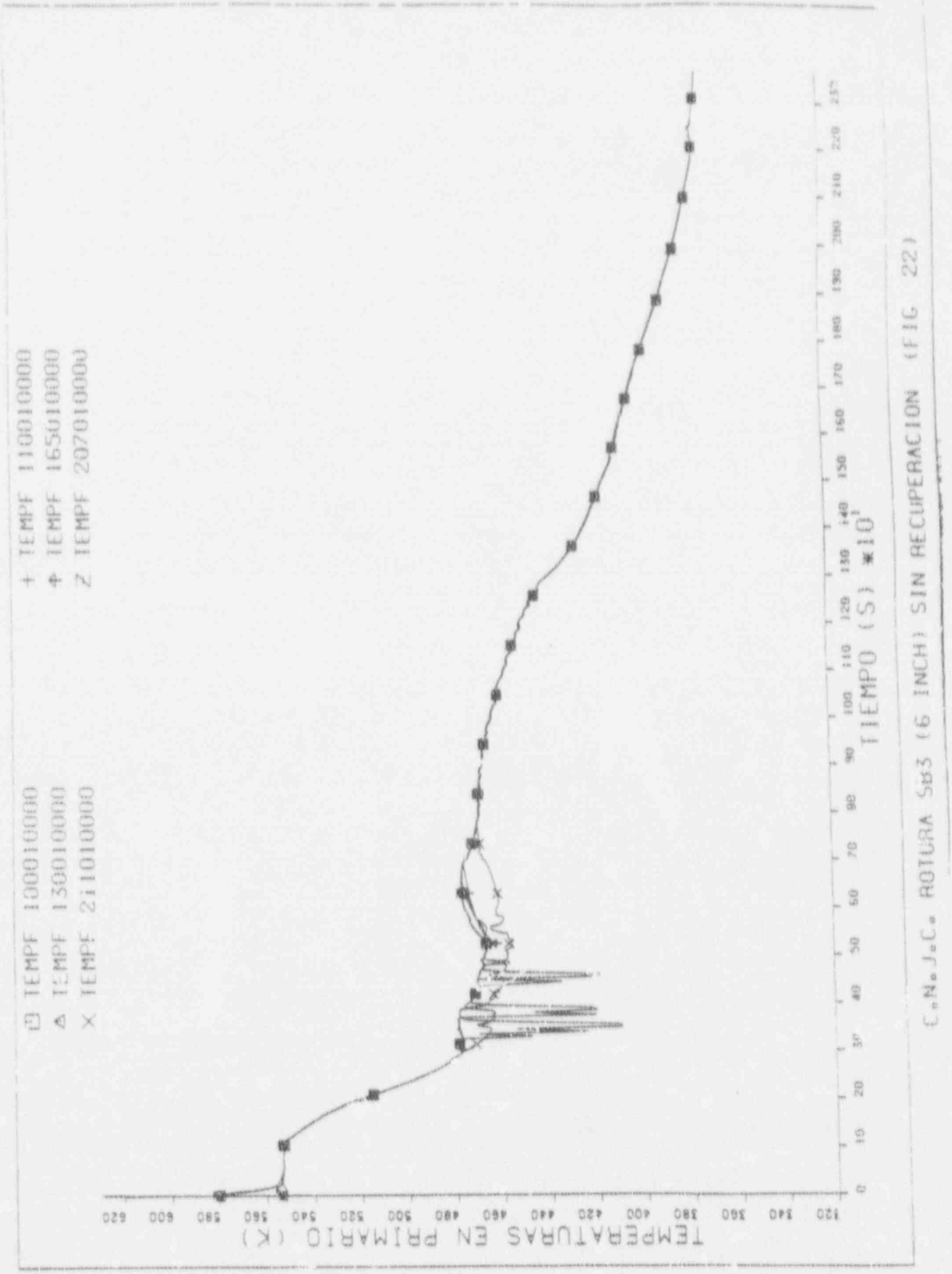

TEMPF 100010000
TEMPF 130010000
TEMPF 211010000
TEMPF 110010000
TEMPF 165010000
TEMPF 207010000
TEMPERATURAS EN PRIMARIO (K)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 22)

FIG. I (A) - 22

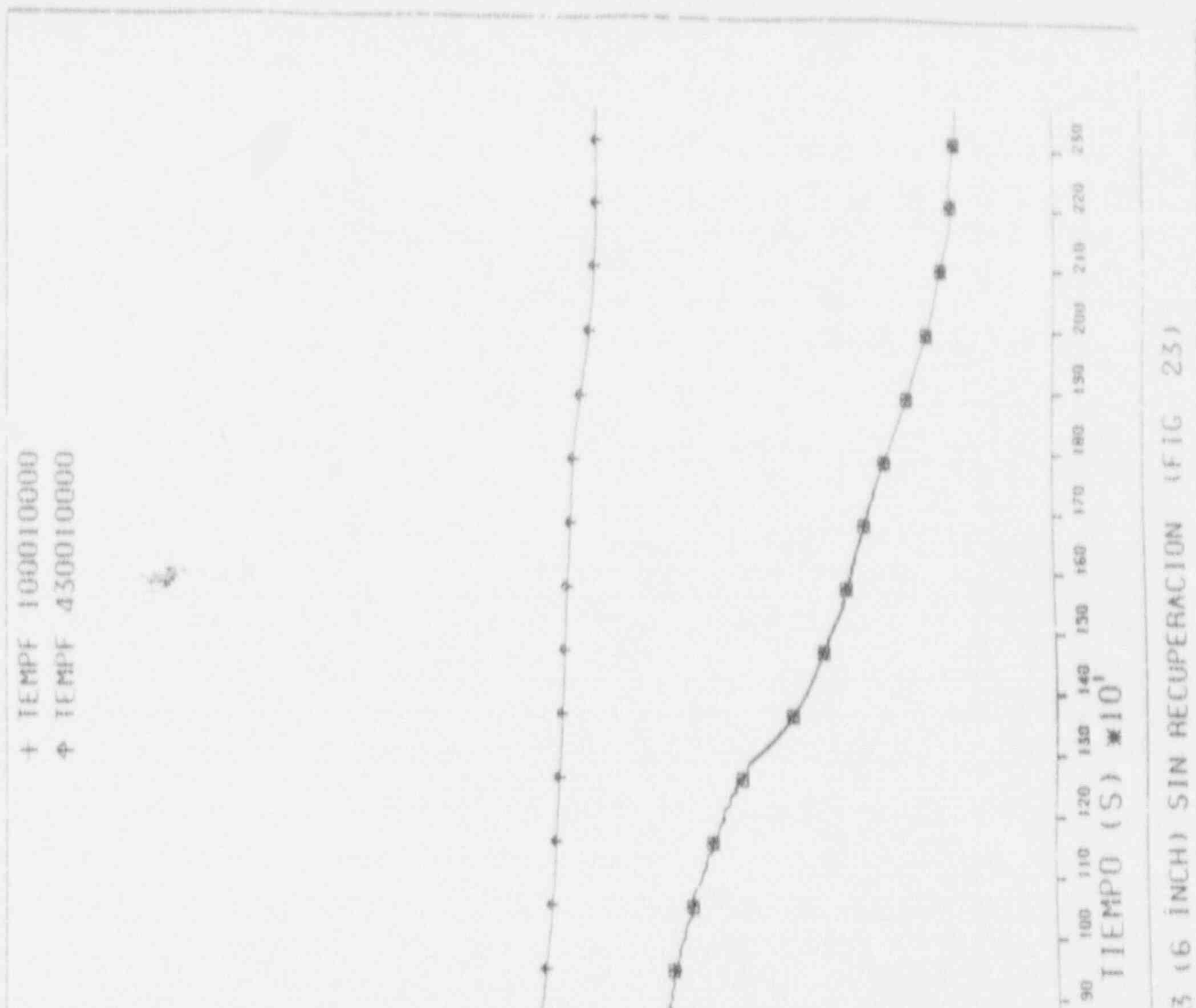

TEMPF 312010000
TEMPF 165010000
TEMPF 100010000
TEMPF 430010000
TIEMPO (S) *10¹
TEMPERATURAS EN PRIMARIO Y G.V. (K)
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 23)

FIG. I (A) - 23

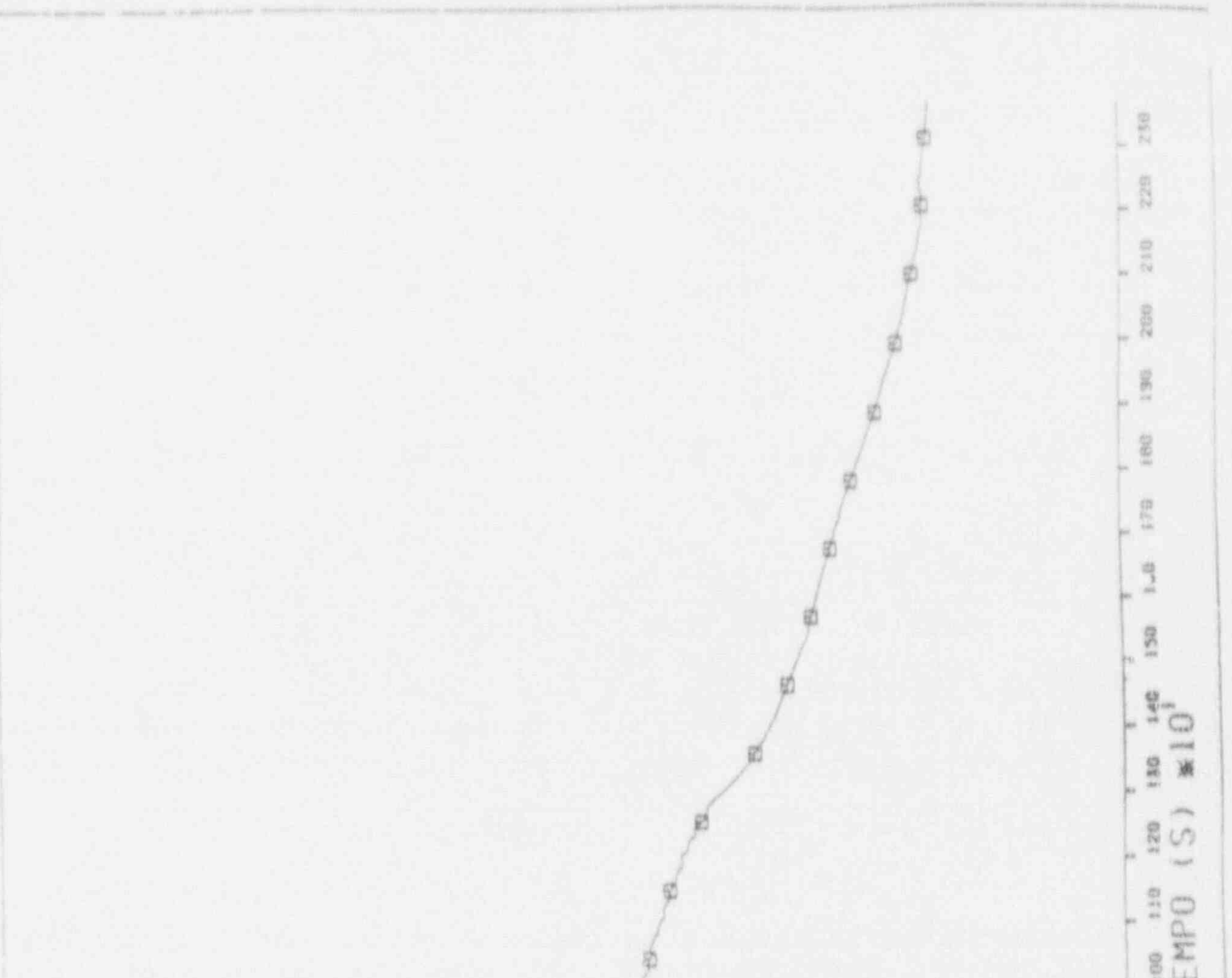

CNTRLVAR 4
TEMPERATURA MEDIA DEL PRIMARIO (C)
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (6 INCH; SIN RECUPERACION (FIG 24)

FIG. I (A) - 24

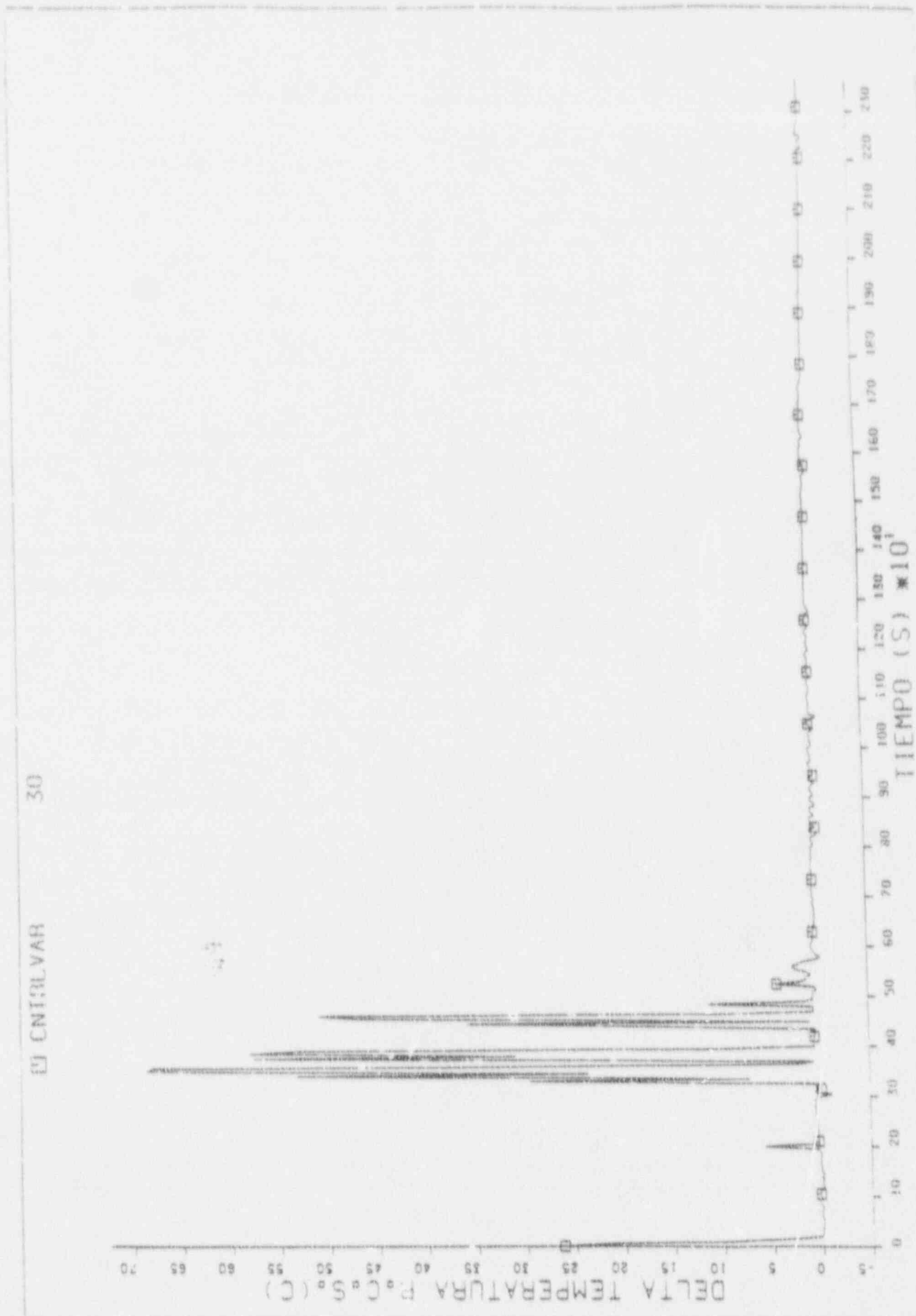
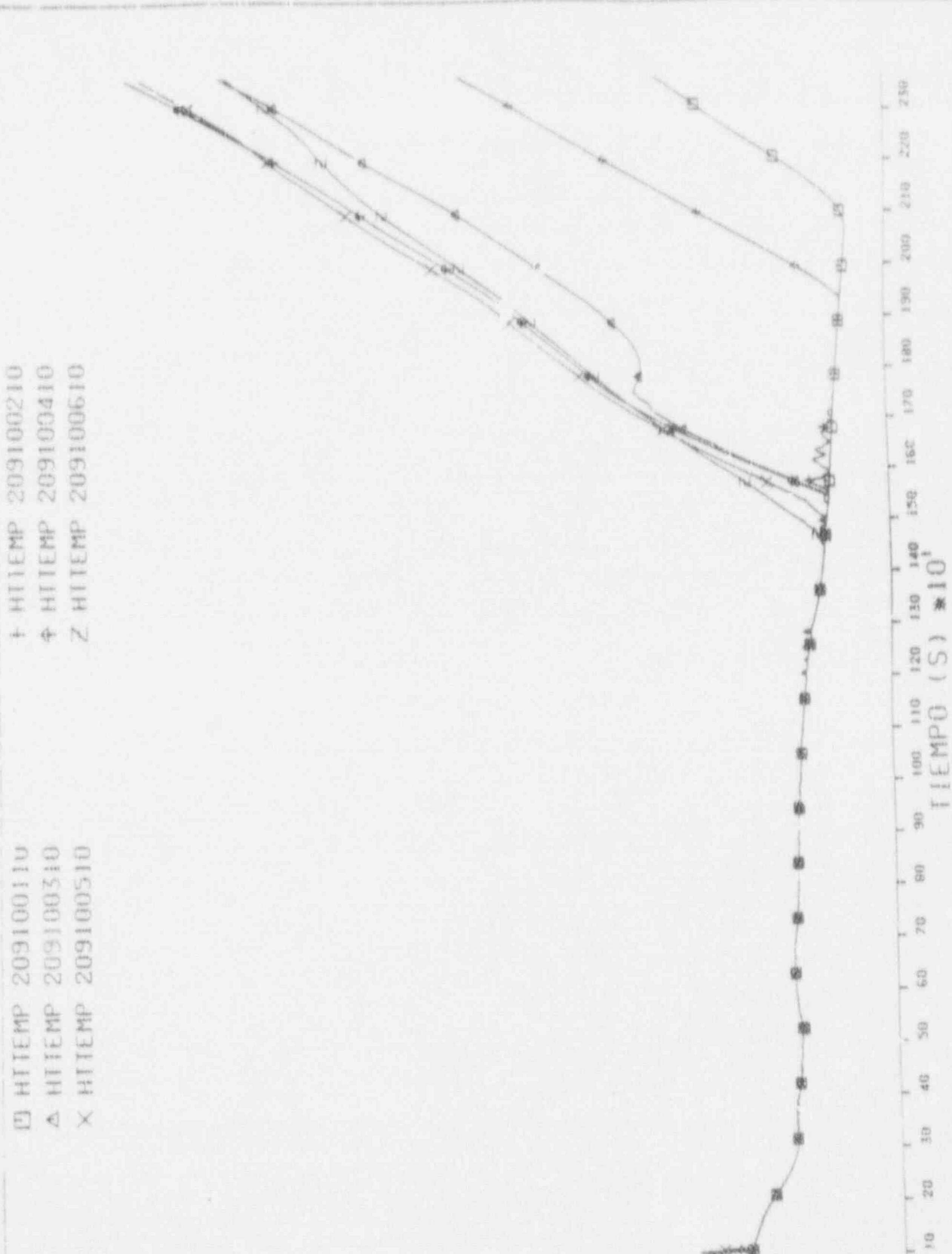
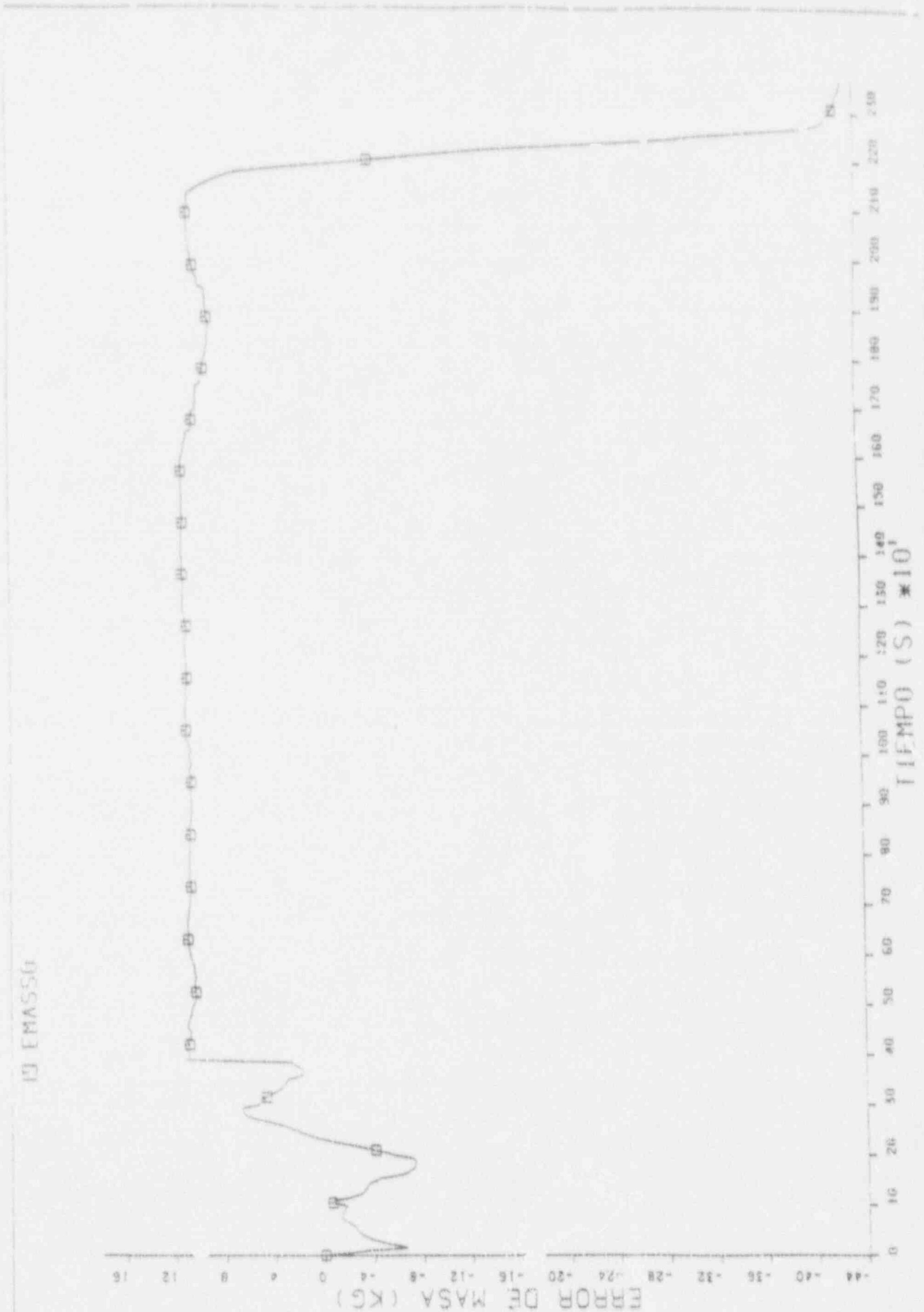

ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 25)
TIEMPO (S) *10^1
DELTA TEMPERATURA P.C.S.(C)

FIG. I (A) - 25



HTTEMP 209100110
HTTEMP 209100310
HTTEMP 209100510
HTTEMP 209100210
HTTEMP 209100410
HTTEMP 209100610
TIEMPO (S) *10[1]
TEMPERATURA VAINAS CANAL MEDIO (K)
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 26)

FIG. I (A) - 26



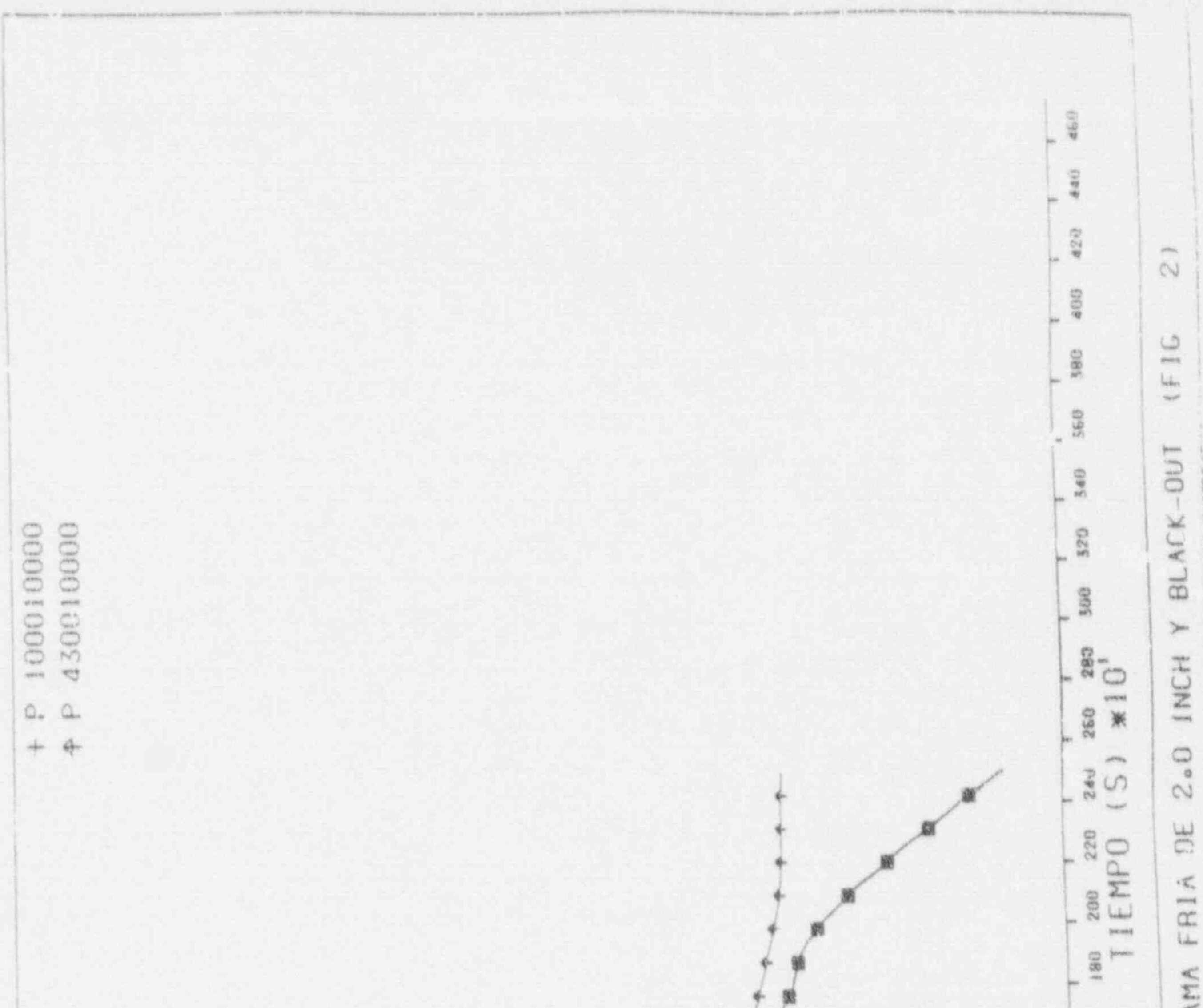
EMASSG
ERROR DE MASA (KG)
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (6 INCH) SIN RECUPERACION (FIG 27)

FIG. I (A) - 27

APPENDIX II : FIGURES OF CASE II(A), (2" WITHOUT RECOVERY)

FIG. II (A) - 1

C.N.J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 1)



□ P 312020000
△ P 165010000
+ P 100010000
✦ P 430010000
PRESION (PA) *10⁴
TIEMPO (S) *10¹
C.N.J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 2)

FIG. II (A) - 2

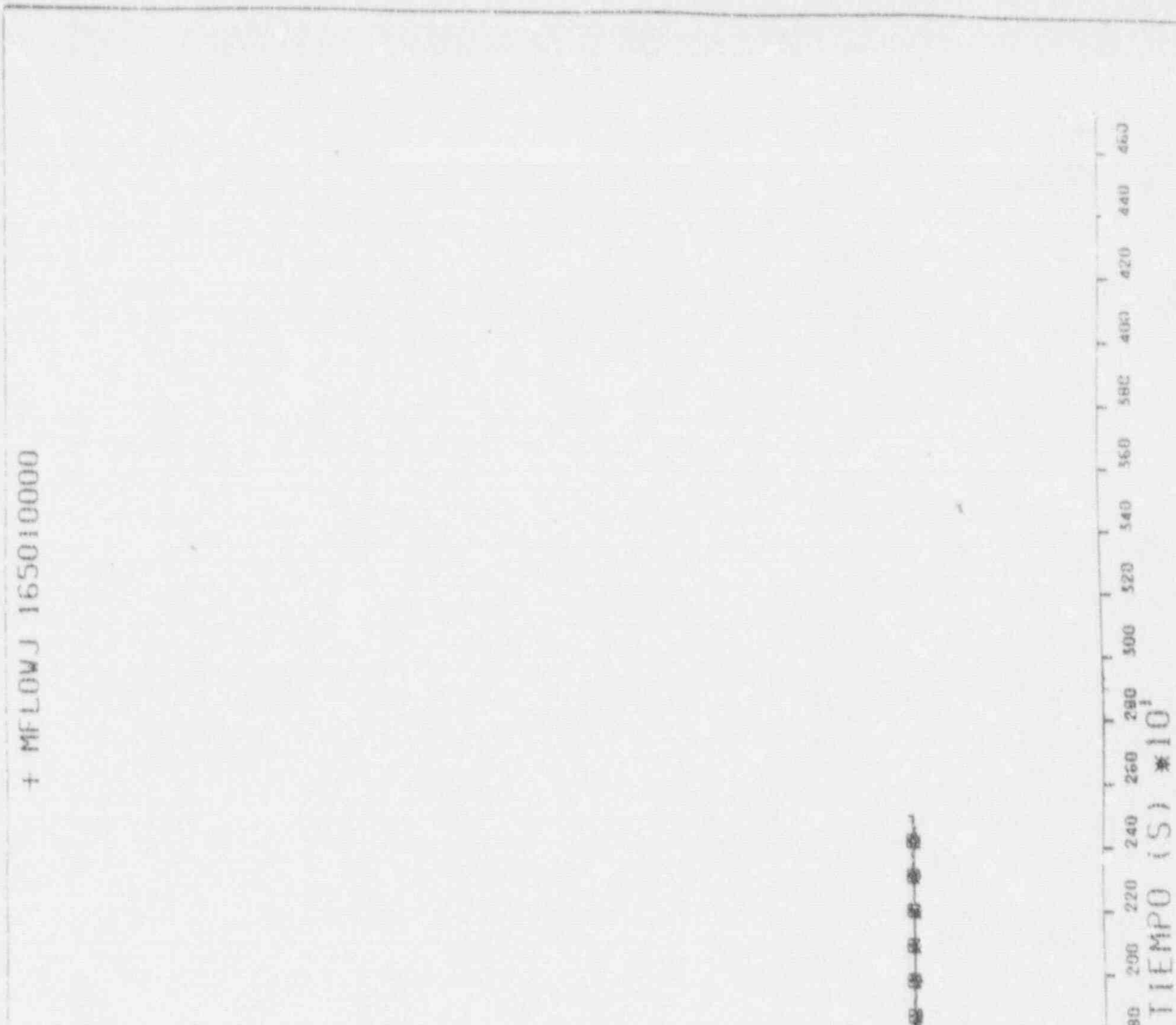
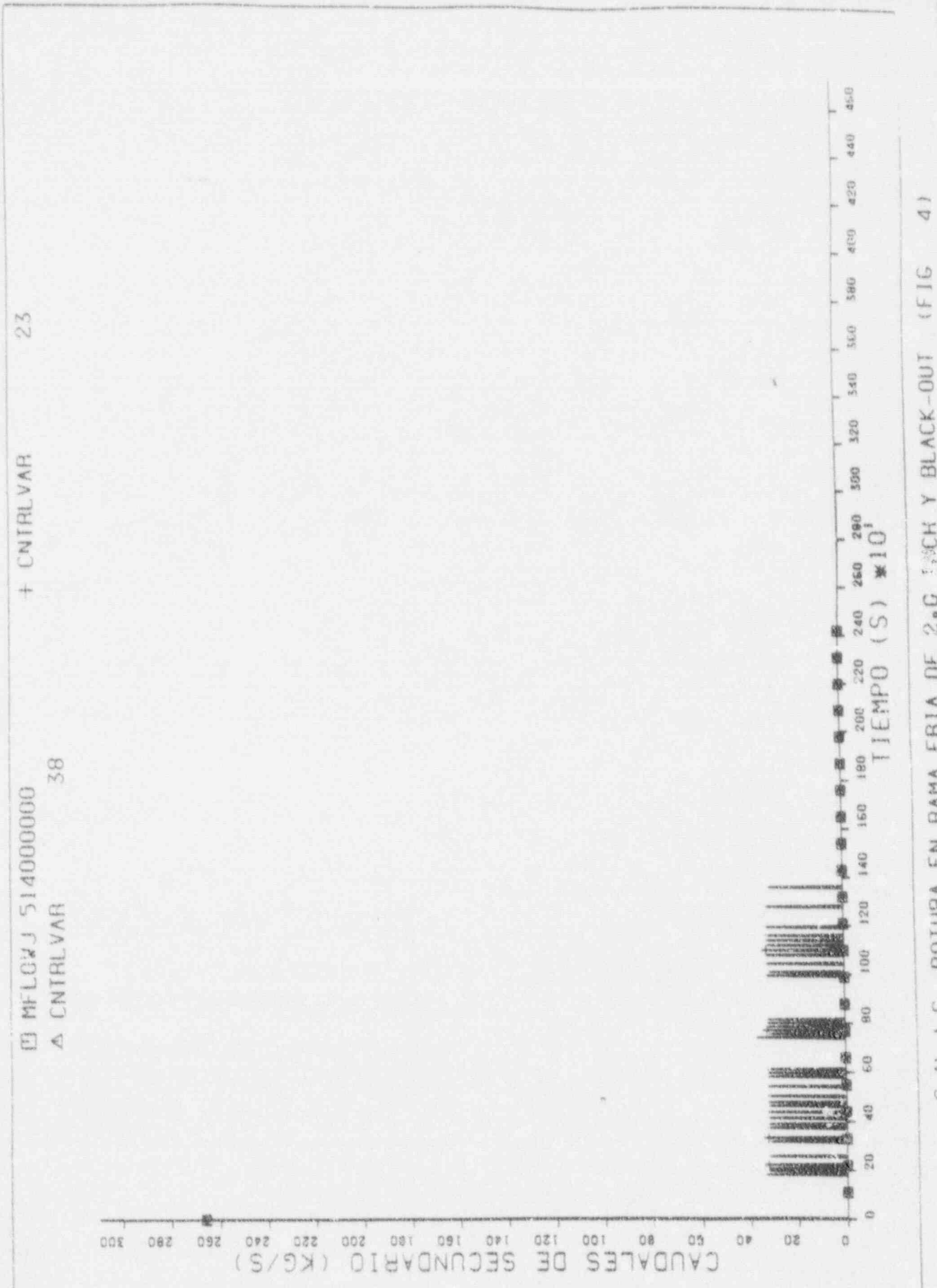

MFLOWJ 160010000
MFLOWJ 243000000
+ MFLOWJ 165010000
CAUDAL DEL PRIMARIO (KG/S) *10¹
TIEMPO (S) *10¹
ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 3)

FIG. II (A) - 3



MFLOWJ 514000000
CNTRLVAR 38
CNTRLVAR 23
CAUDALES DE SECUNDARIO (KG/S)
TIEMPO (S) *10¹
C.N.J.C. ROTURA EN RAMA FRIA DE 2.0
Y BLACK-OUT (FIG 4)

FIG. II (A) - 4

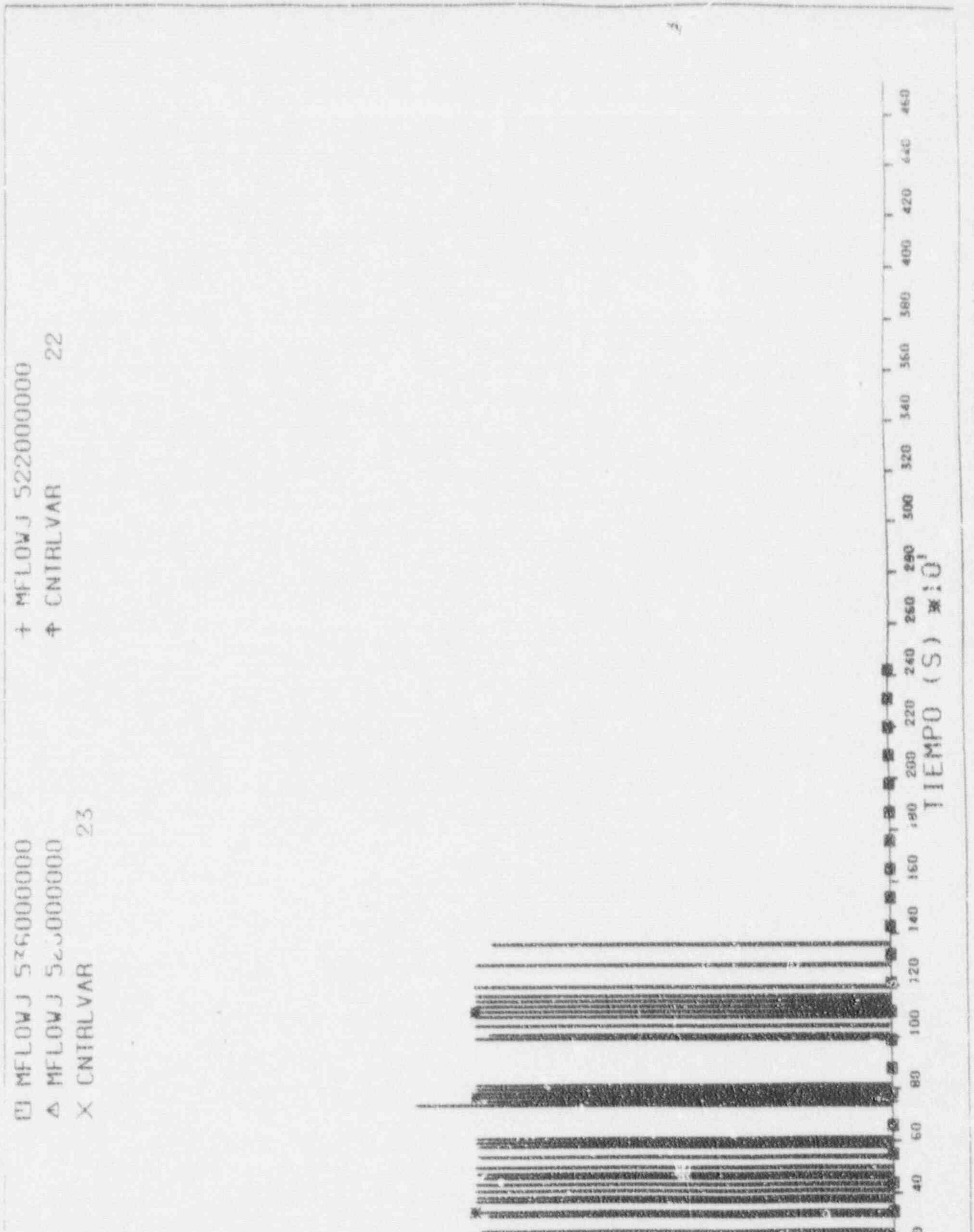

MFLOWJ
MFLOWJ
CNTRLVAR 23
MFLOWJ 522000000
CNTRLVAR 22
ALIVIO Y SEGURIDAD G.V. (KG/S)
0
4
8
12
16
20
24
28
32
36
40
44
48
52
56
60
0
20
40
60
80
100
120
140
160
180
200
220
240
260
280
300
320
340
360
380
400
420
440
460
TIEMPO (S) *10¹
C.N.J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 5)

FIG. II (A) - 5

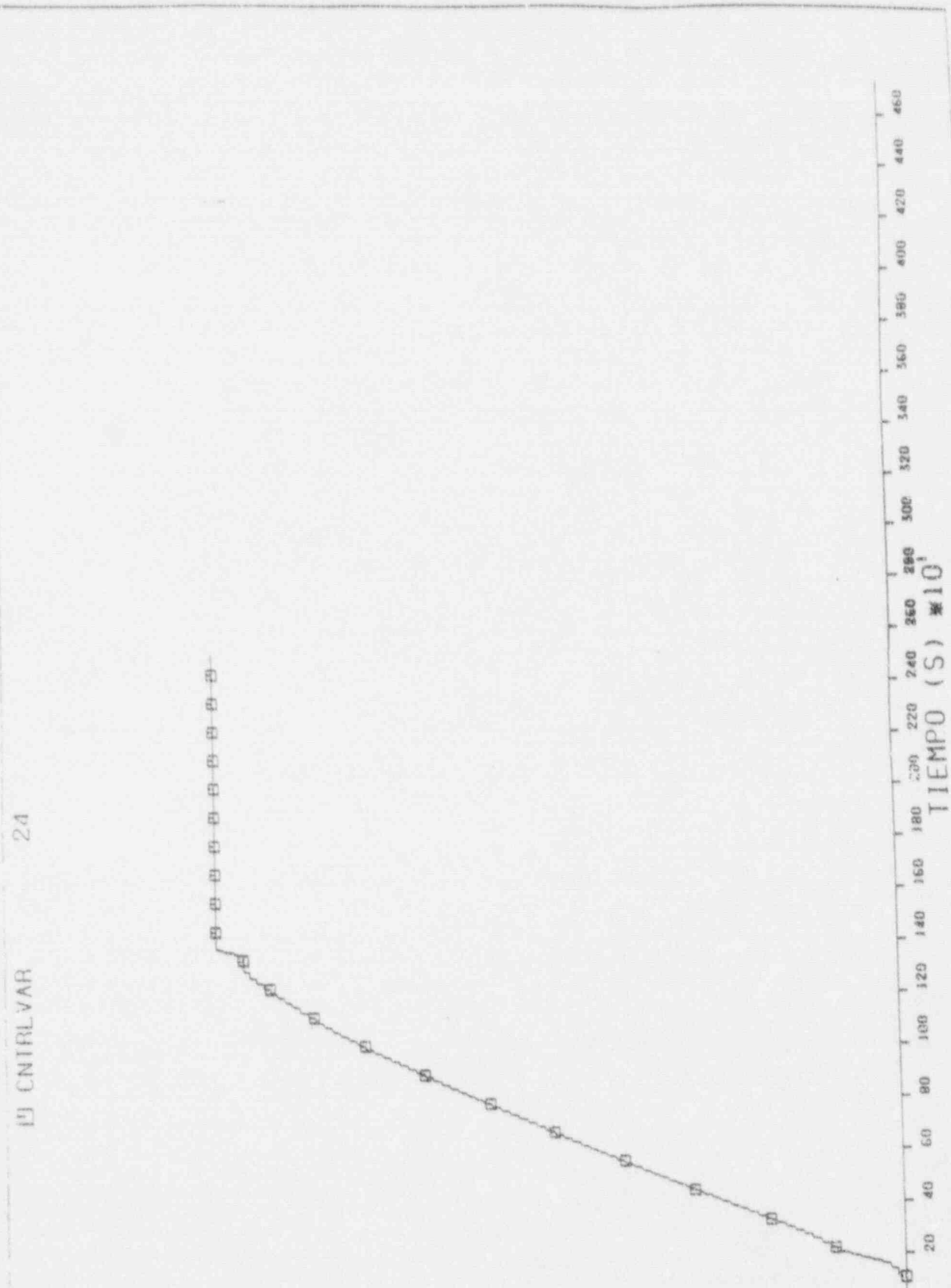
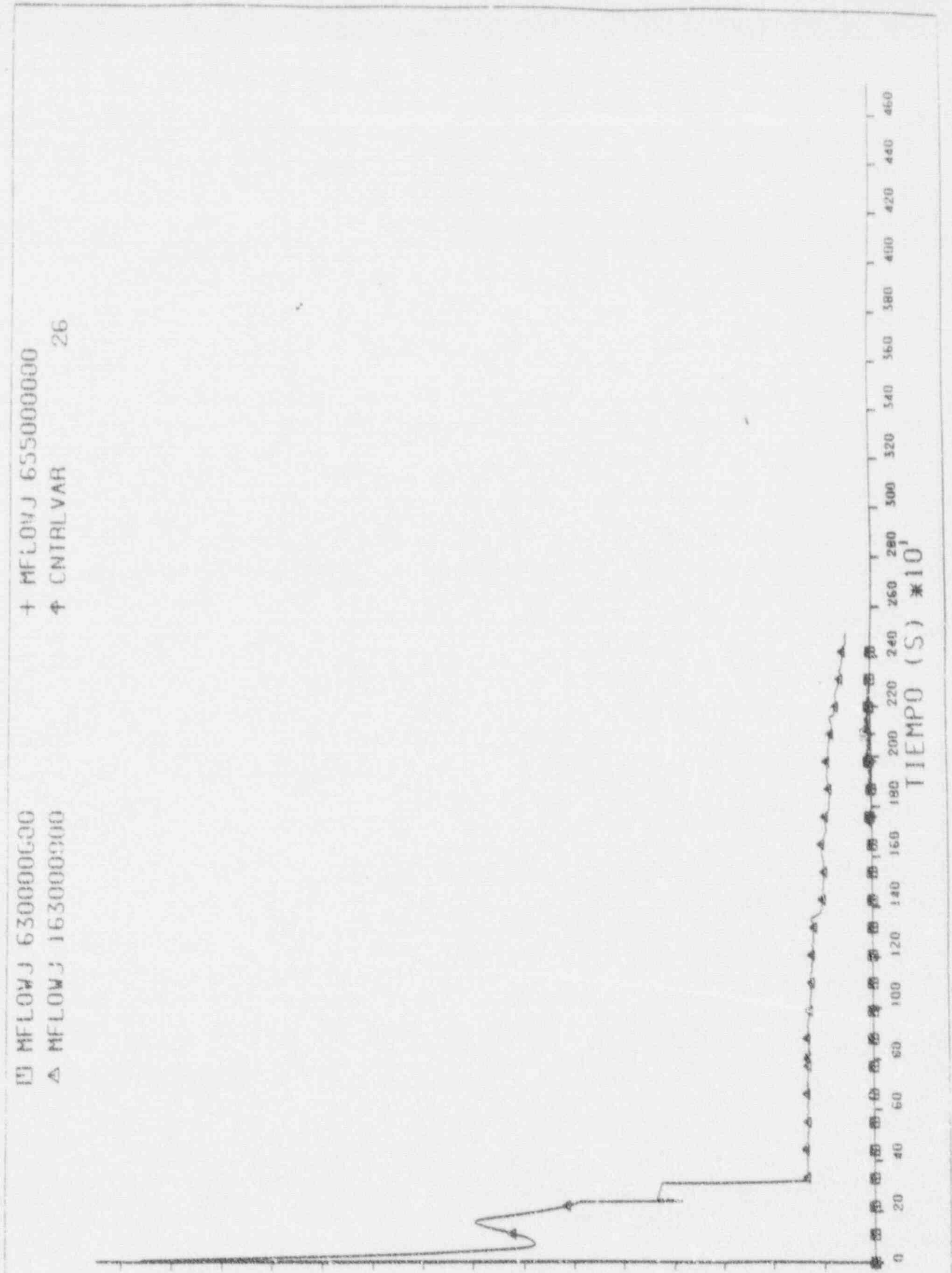
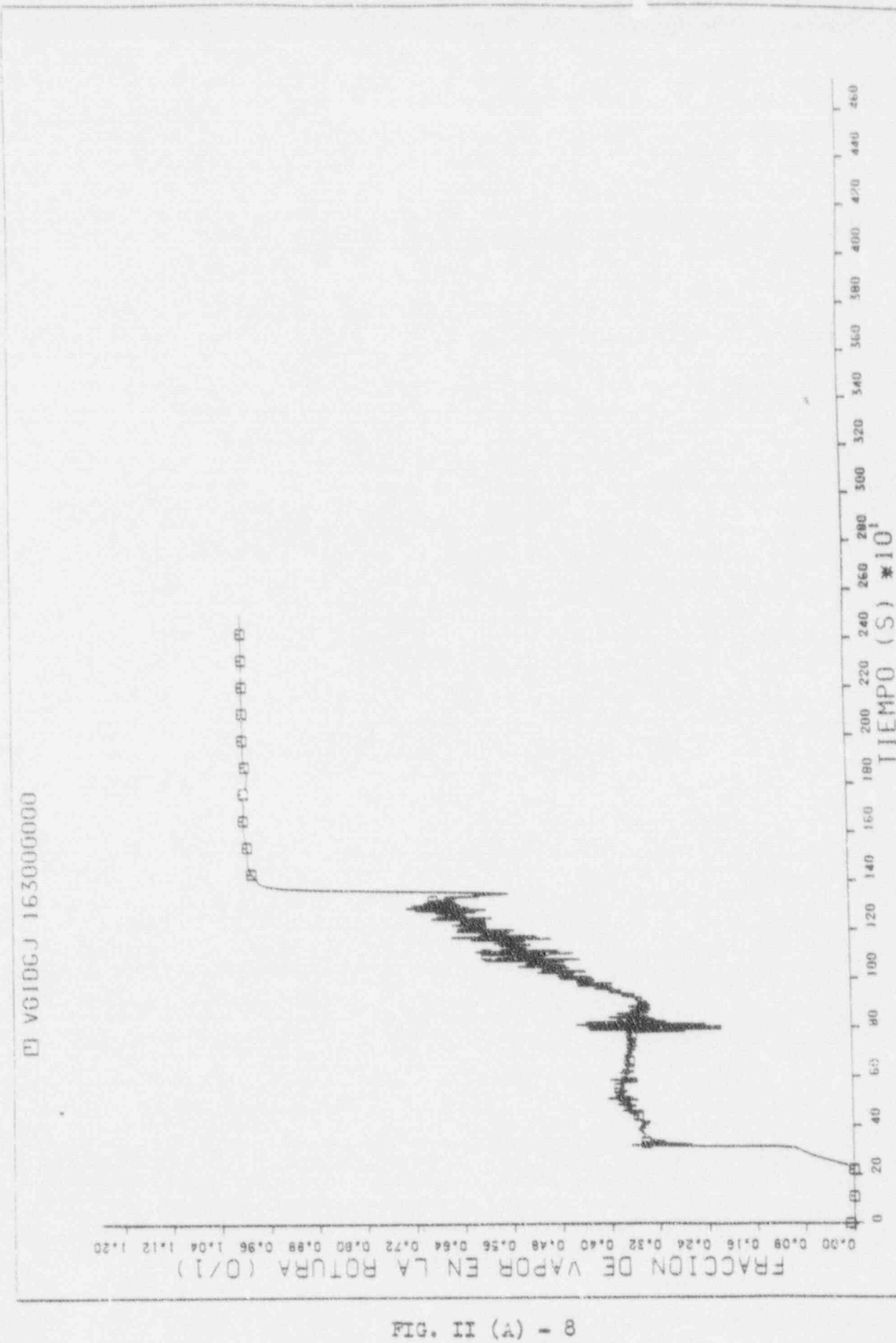

CNTRLVAR 24
TIEMPO (S) *10^1
MASA DESCARGADA DEL G.V. A ATM (KG) *10^1
C.N.J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 6)

FIG. II (A) - 6



MFLOWJ 630000000
MFLOWJ 163000000
MFLOWJ 655000000
CNTRLVAR 26
CAUDAL DE ROTURA Y ECCS (KG/S)
TIEMPO (S) *10¹
C.N. J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 7)

FIG. II (A) - 7

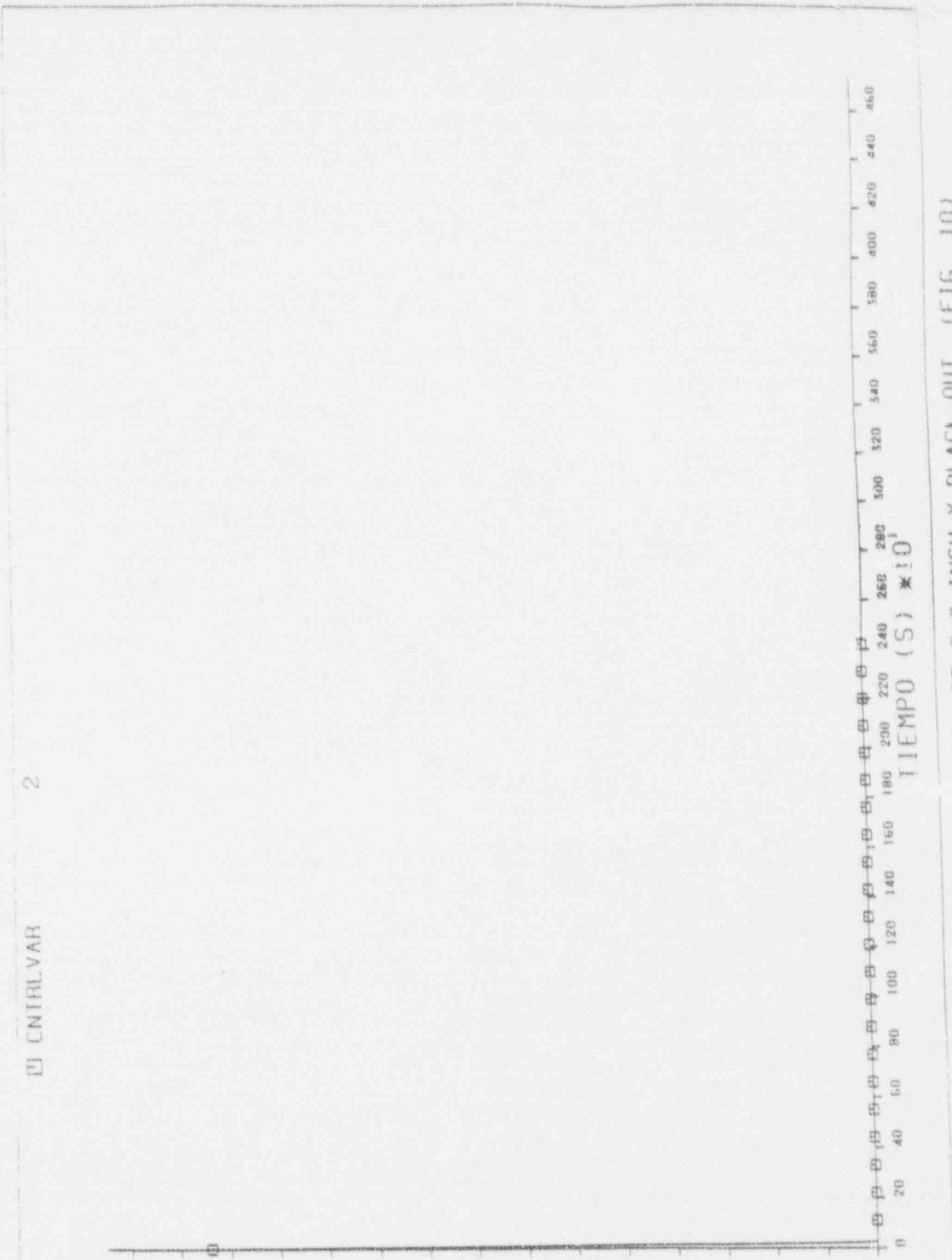


VGIDGJ 163000000
FRACCION DE VAPOR EN LA ROTURA (O/1)
TIEMPO (S) *10[1]
C.N.J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 8)

FIG. II (A) - 8

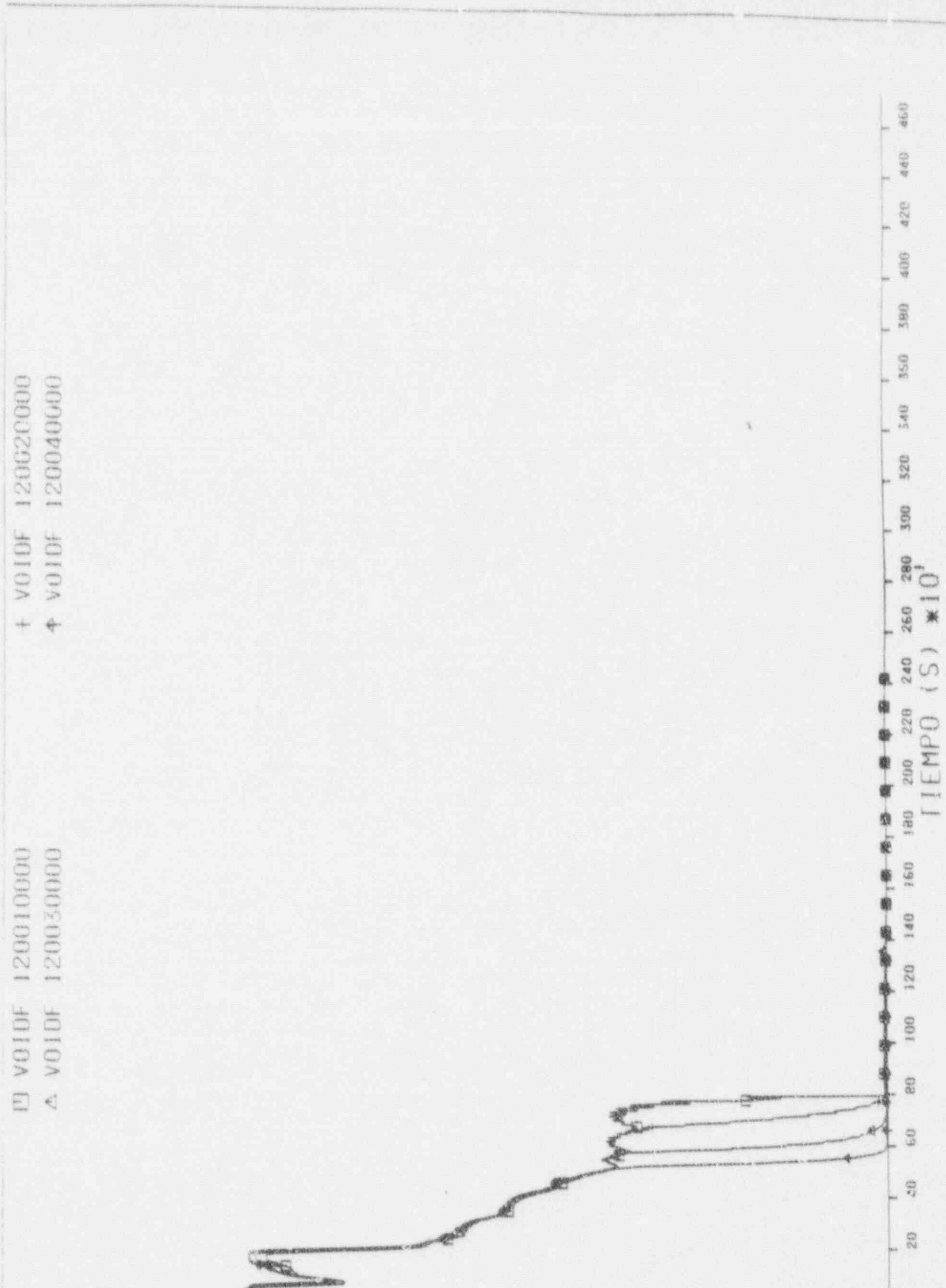

□ CNTRLVAR 29
△ CNTRLVAR 46
+ CNTRLVAR 43
BALANCE MASICO DEL PRIMARIO (KG) *10²
TIEMPO (S) *10¹
C.N. J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 9)

FIG. II (A) - 9



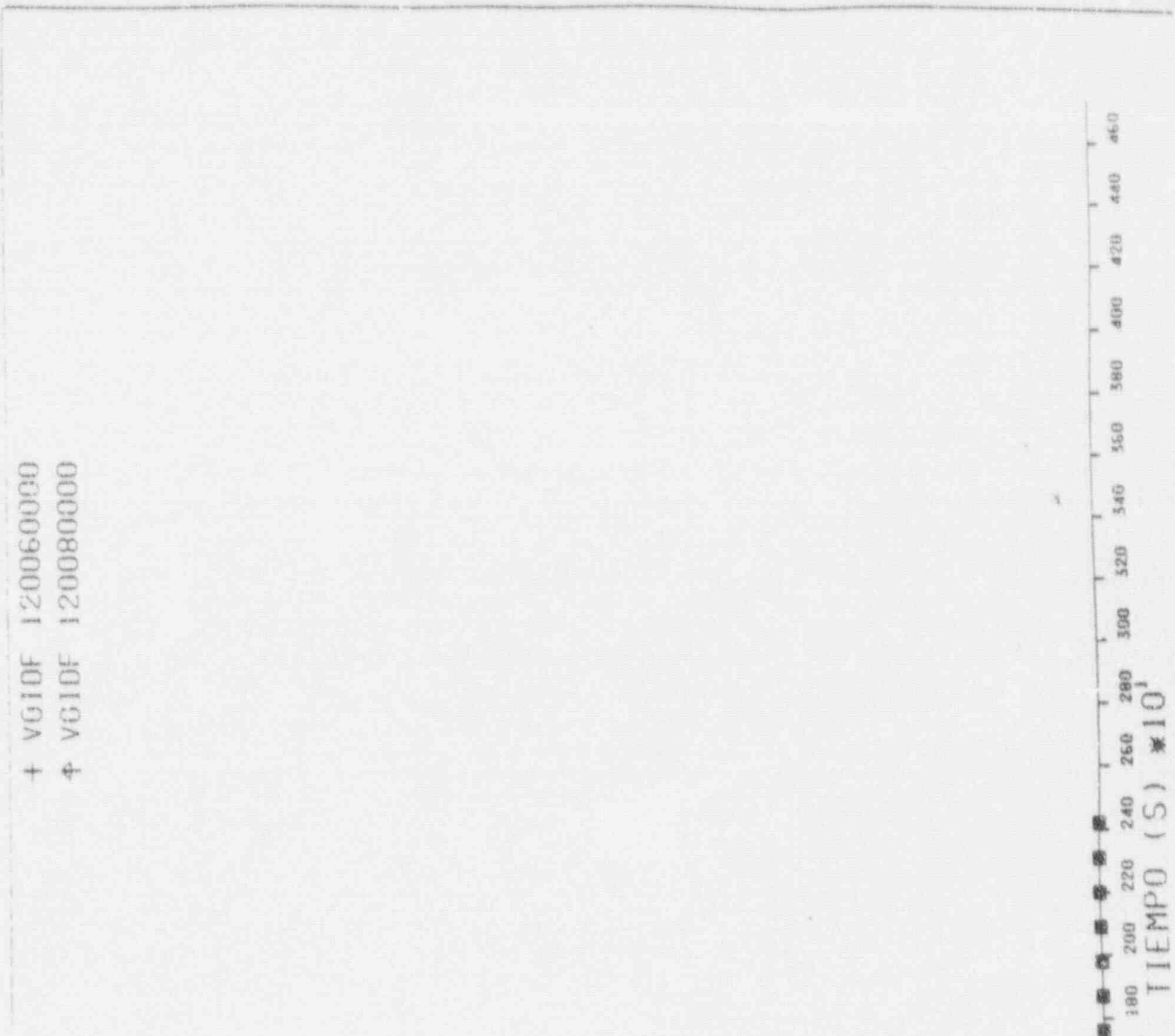
CNTRLVAR 2
NIVEL COLAPSADO EN PRESIONADOR (%)
TIEMPO (S) *10^1
C.N.J.C. ROTURA EN RA
FRIA DE 2.0 INCH Y BLACK-OUT (FIG 10)

FIG. II (A) - 10



VOIDF 120010000
VOIDF 120020000
VOIDF 120030000
VOIDF 120040000
TIEMPO (S) *10¹
FRAC. DE LIQUIDO EN TUBO UP G.V. (O/1)
ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT

FIG. II (A) - 11



VOIDF 120050000
VOIDF 120070000
VOIDF 120060000
VGIDF 120080000
TIEMPO (S) *10[1]
FRAC. DE LIQUIDO EN TUBO DOWN G.V. (O/1)
C.N.J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 12)

FIG. II (A) - 12

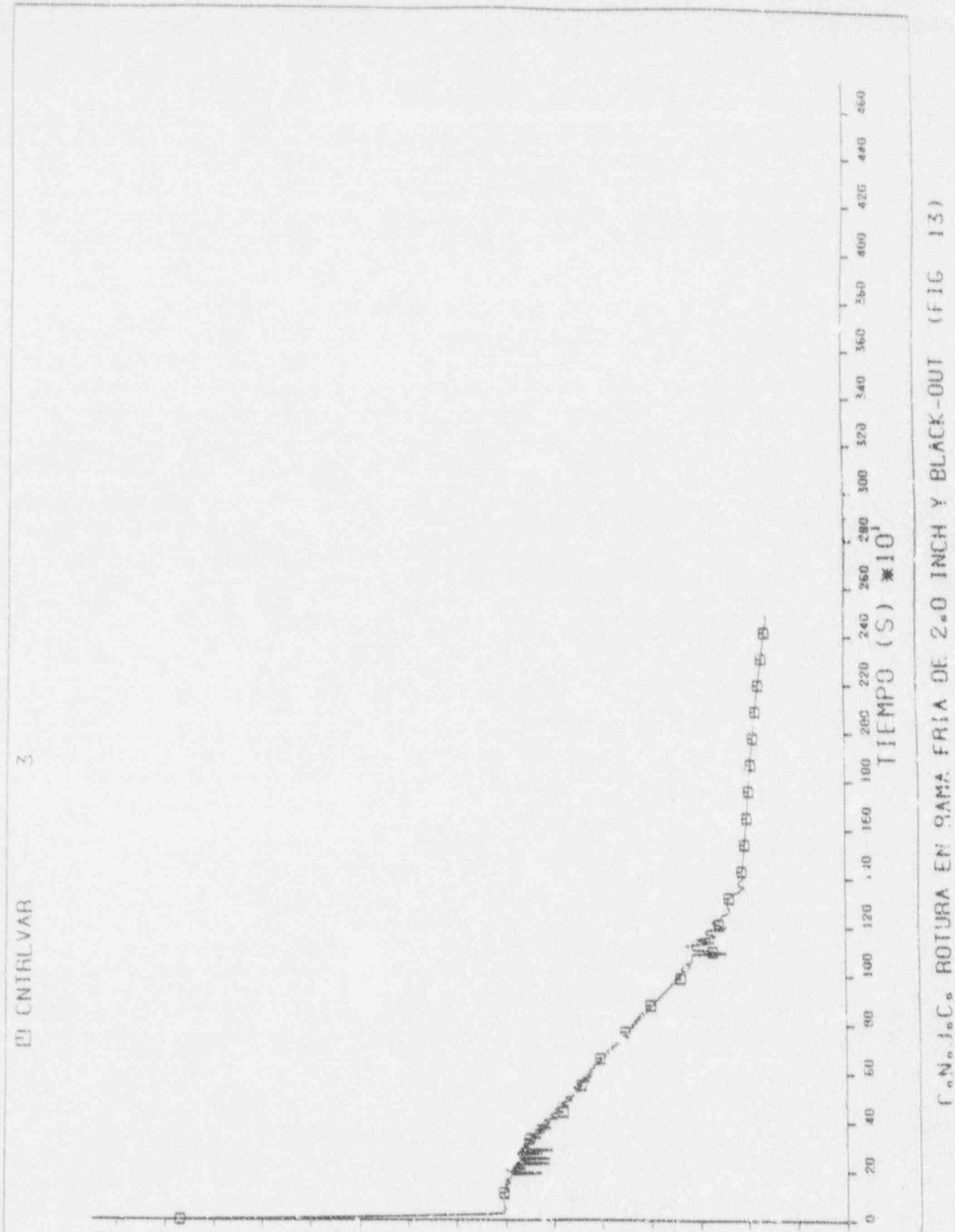

CNTRLVAR 3
C.N.J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 13)
TIEMPO (S) *10¹
NIVEL EN DOWNCOMER DEL G.V. (M)

FIG. II (A) - 13

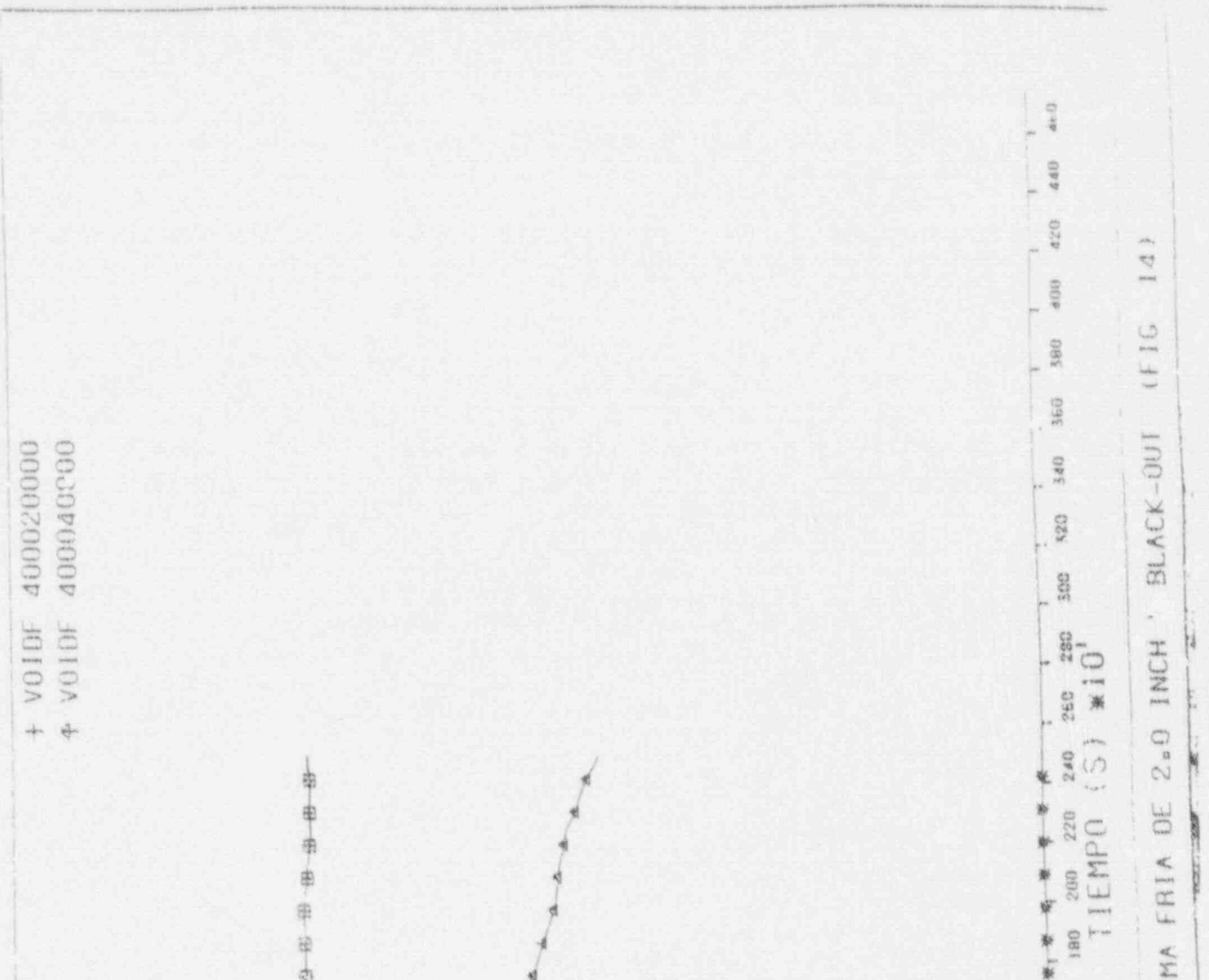
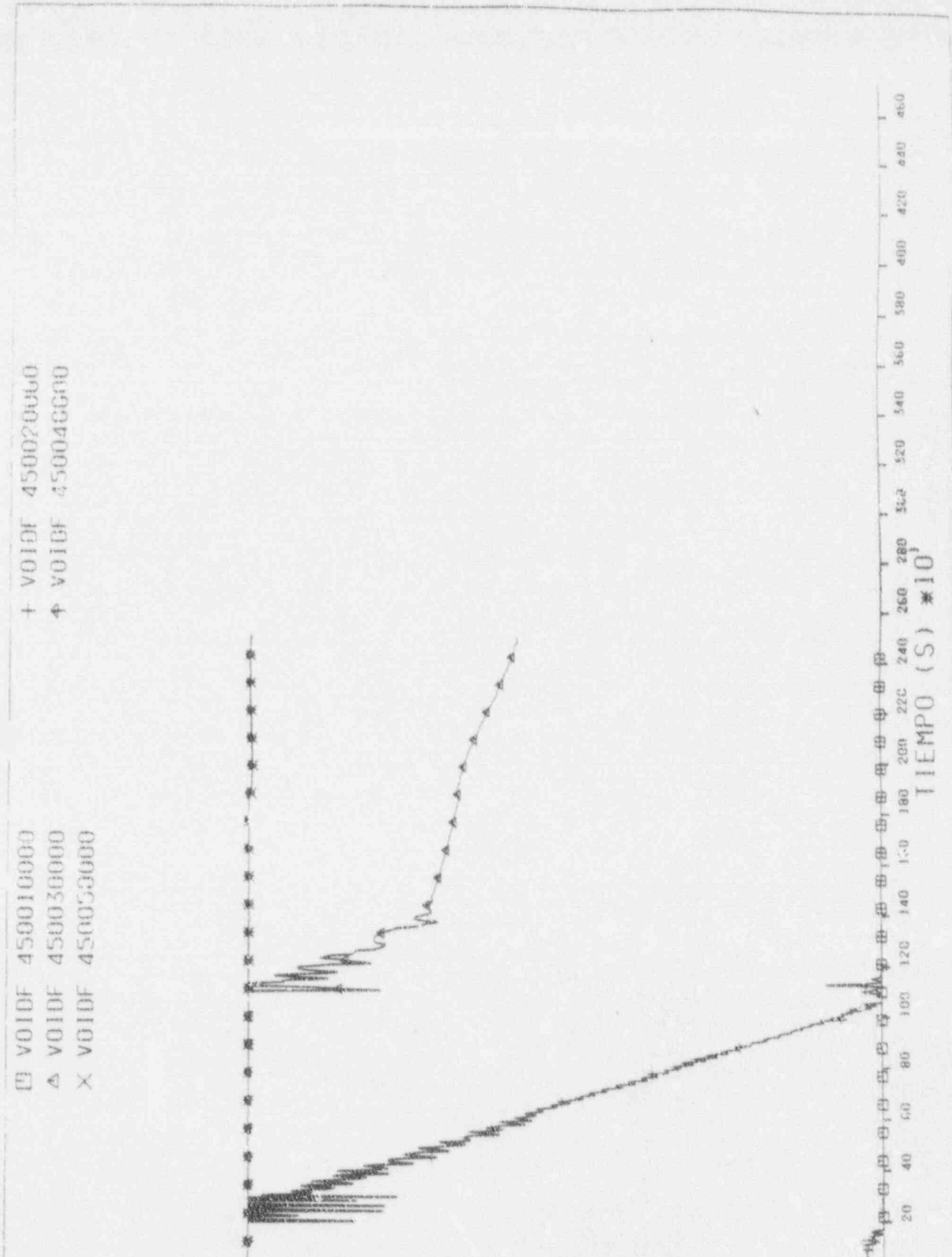

□ VOIDF 400010000
△ VOIDF 400030000
× VOIDF 400050000
+ VOIDF 400020000
◇ VOIDF 400040000
FRAC. DE LIQUIDO EN RISER G.V. (0/1)
0.00 0.08 0.16 0.24 0.32 0.40 0.48 0.56 0.64 0.72 0.80 0.88 0.96 1.04 1.12 1.20
0 20 40 60 80 100 120 140 160 180 200 220 240 260 280 300 320 340 360 380 400 420 440
TIEMPO (S) *10[1]
C.N.J.C. ROTURA EN RAMA FRIA DE 2.0 INCH ' BLACK-OUT (FIG 14)

FIG. II (A) - 14



VOIDF 450010000
VOIDF 450030000
VOIDF 450050000
VOIDF 450020000
VOIDF 450040000
TIEMPO (S) *10
FRAC. DE LIQUIDO EN DOWNCOMER G.V. (O/I)
C.N. J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 15)

FIG. II (A) - 15

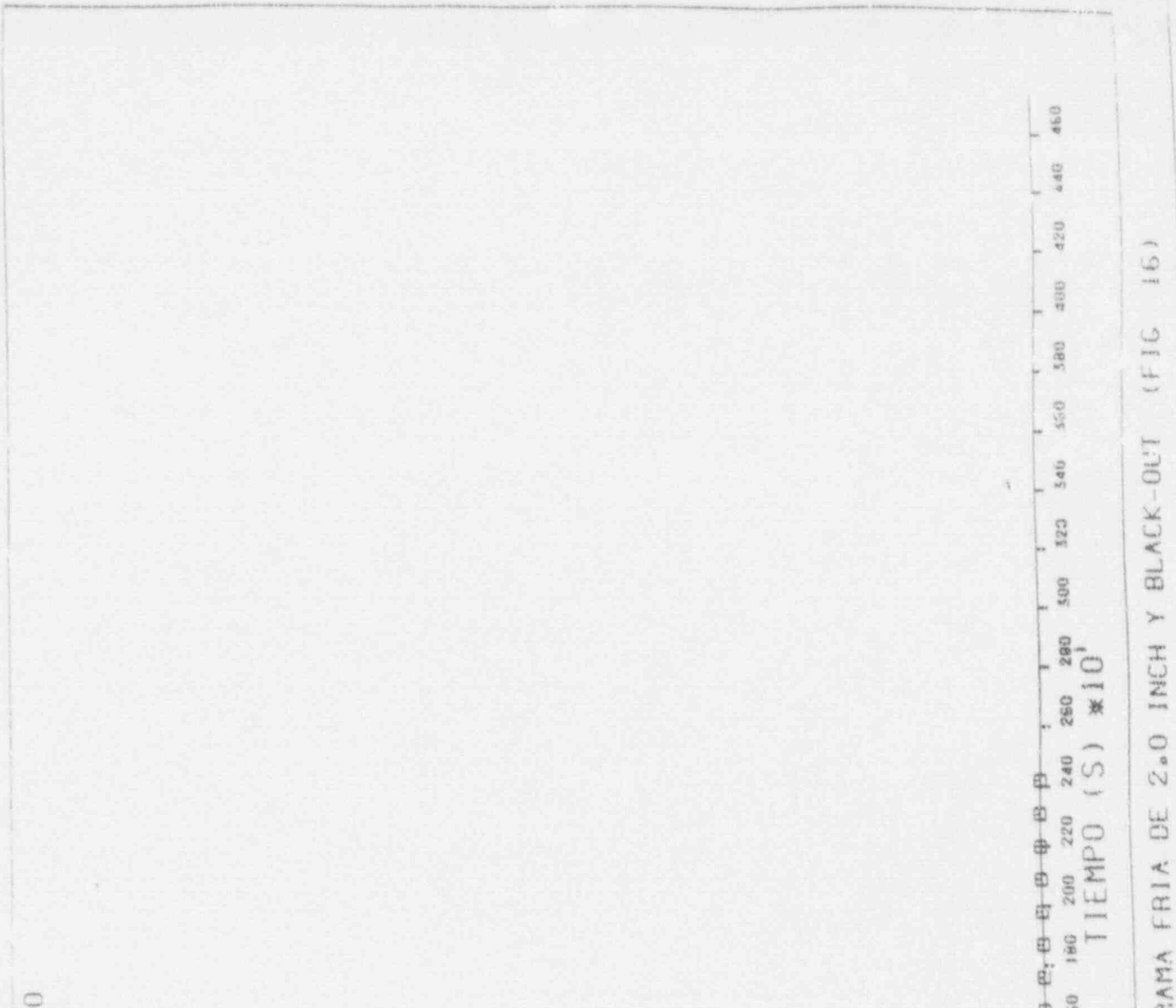

VOIDF 140010000
NIVEL COLAPSADO EN SELLO -140-
TIEMPO (S) *10^1
C.N.J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 16)

FIG. II (A) - 16

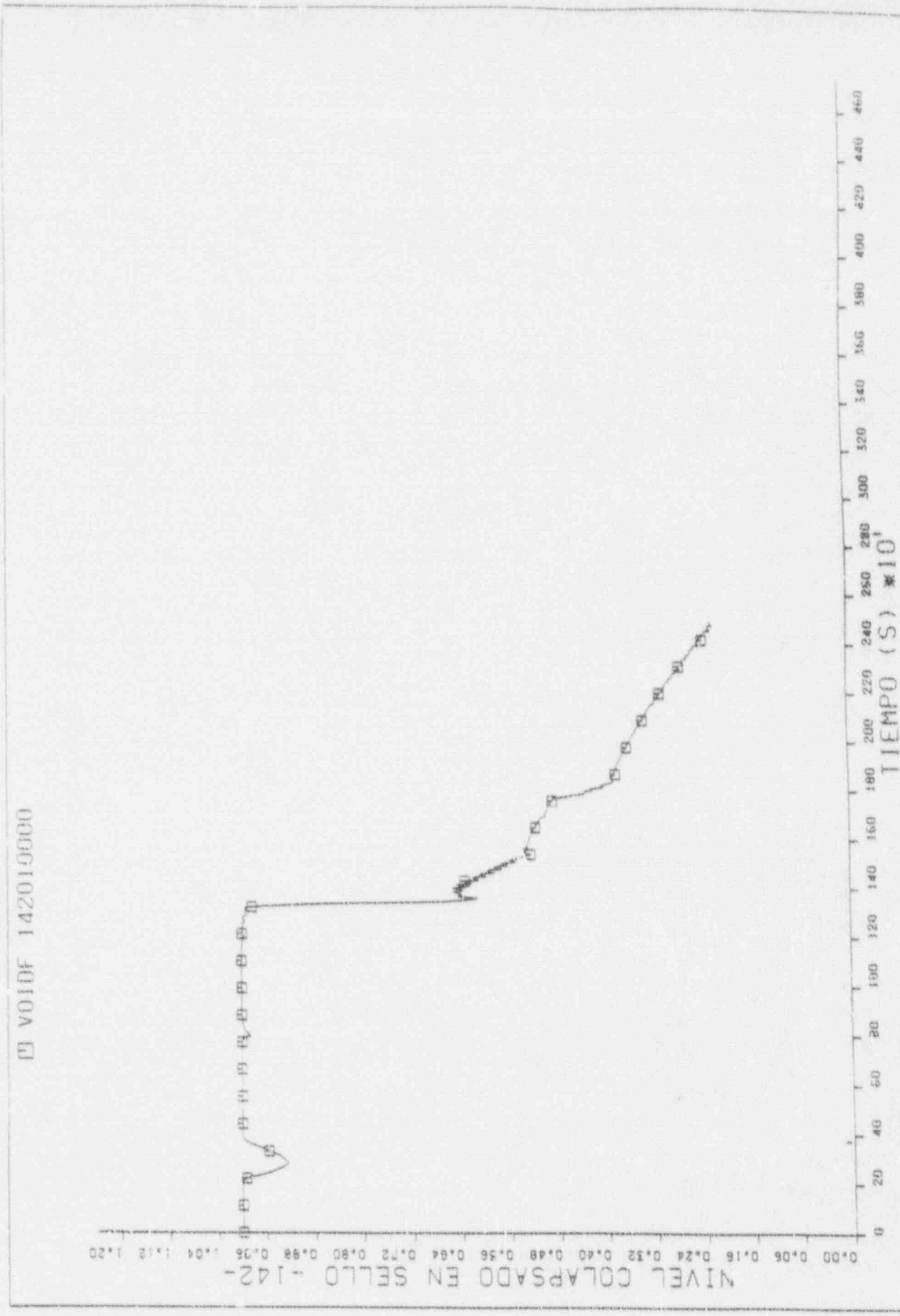

VOIDF 142010000
NIVEL COLAPSADO EN SELLO -142-
TIEMPO (S) *10^1
C.N. J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 17)

FIG. II (A) - 17

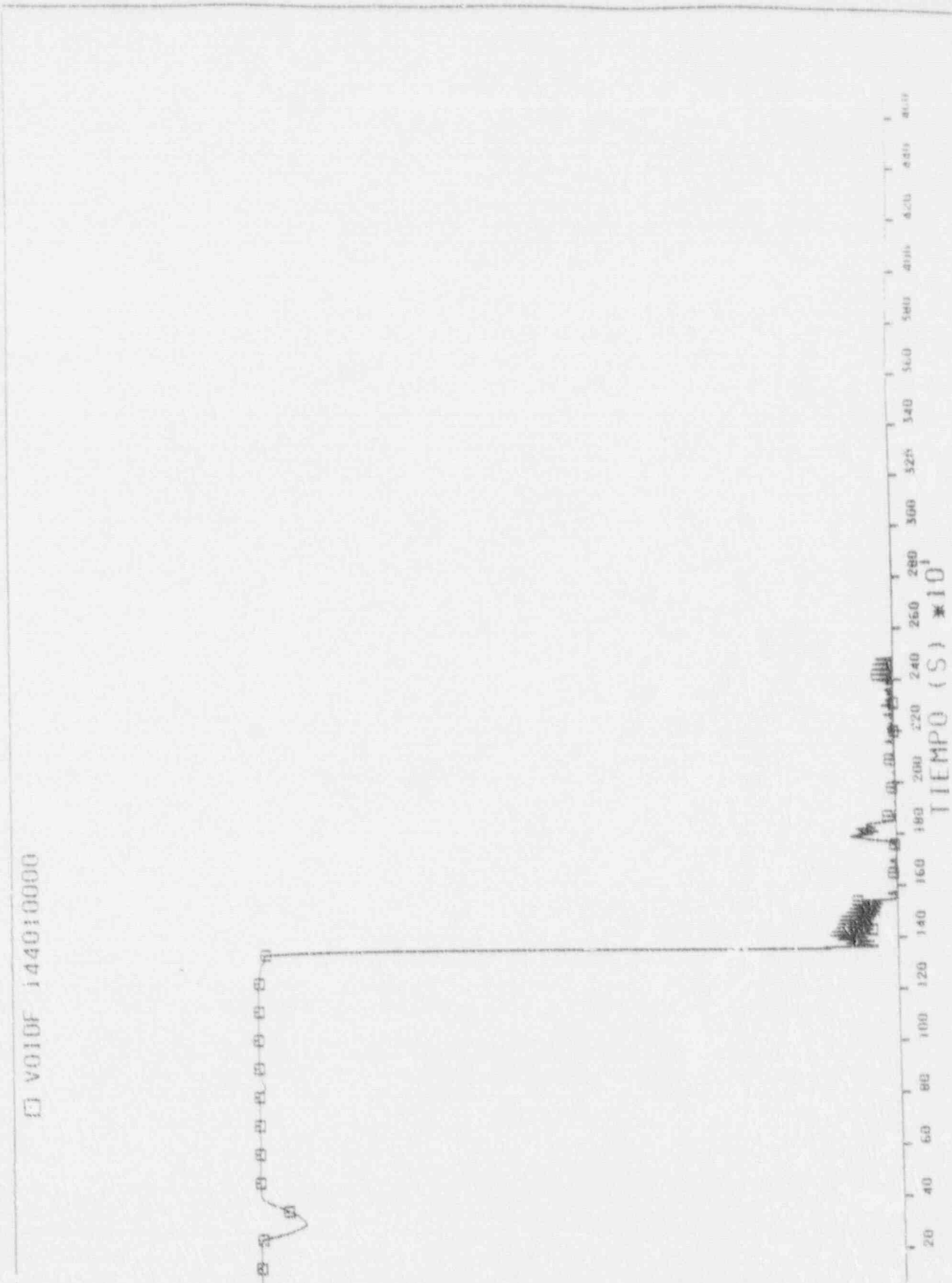


FIG. II (A) - 18

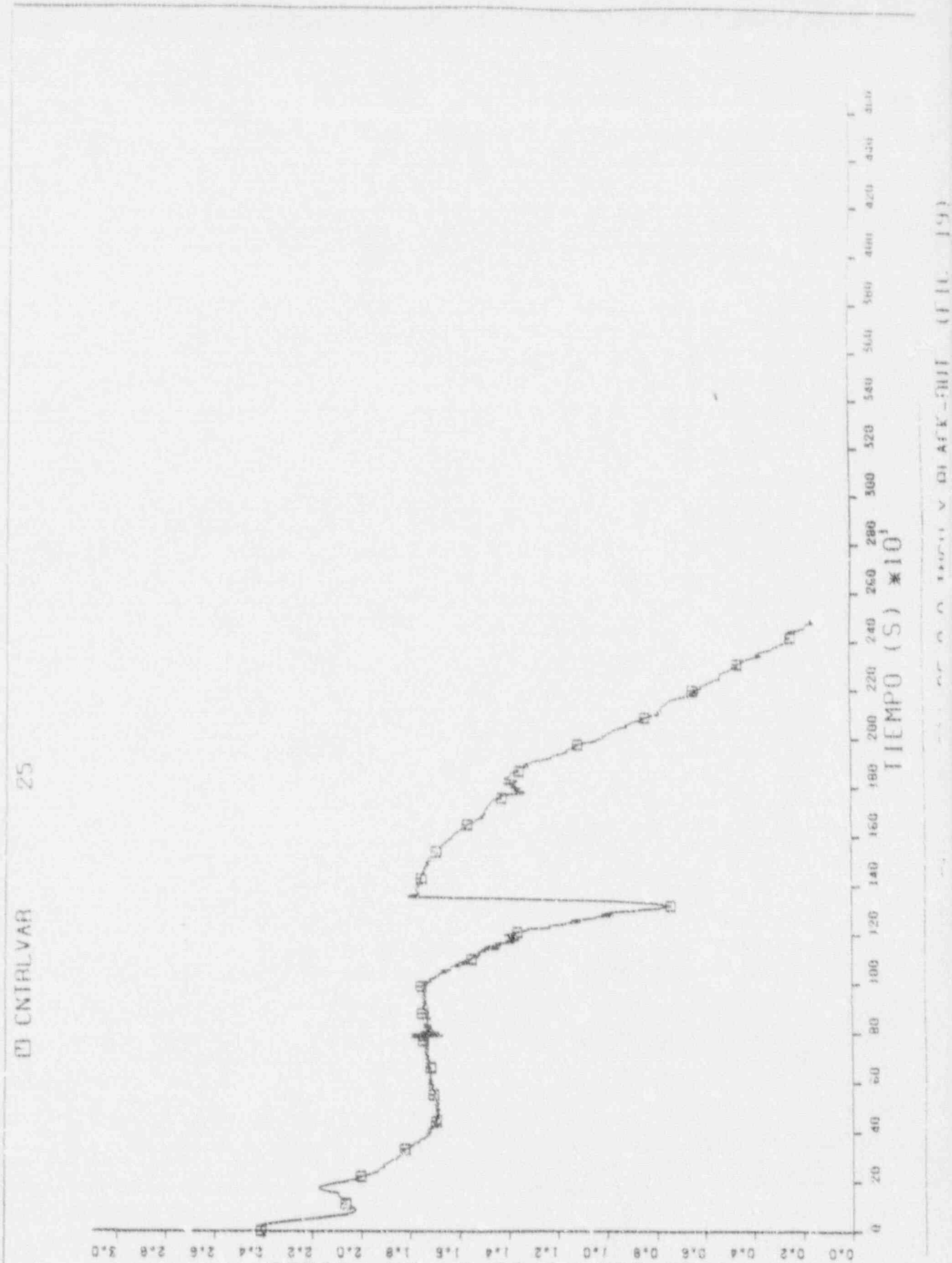
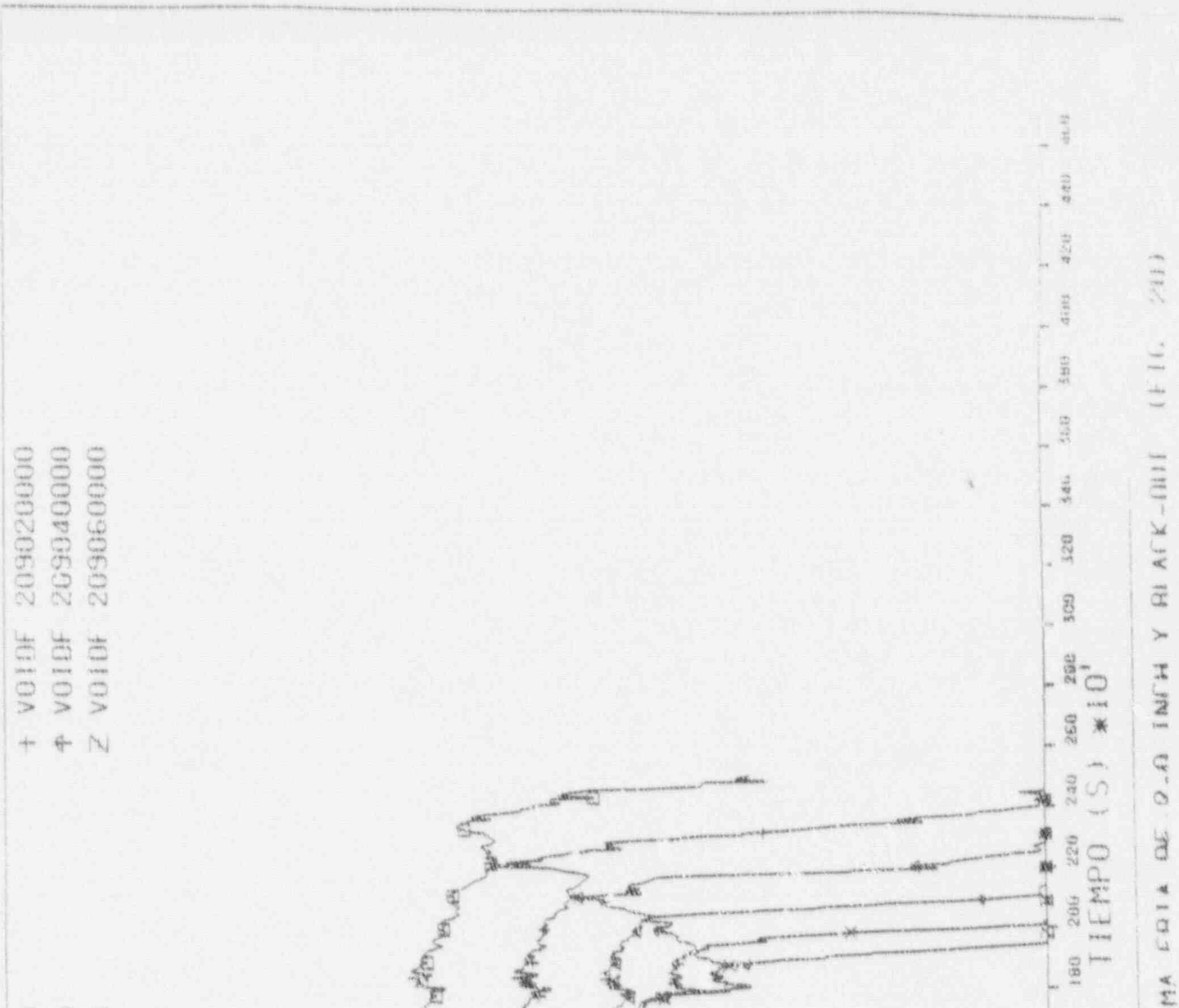


FIG. II (A) - 19



VOIDF 209010000
VOIDF 209030000
VOIDF 209050000
VOIDF 209020000
VOIDF 209040000
VOIDF 209060000
FRACCIONES DE LIQUIDO EN EL NUCLEO (O/I)
TIEMPO (S) *10¹

FIG. II (A) - 20

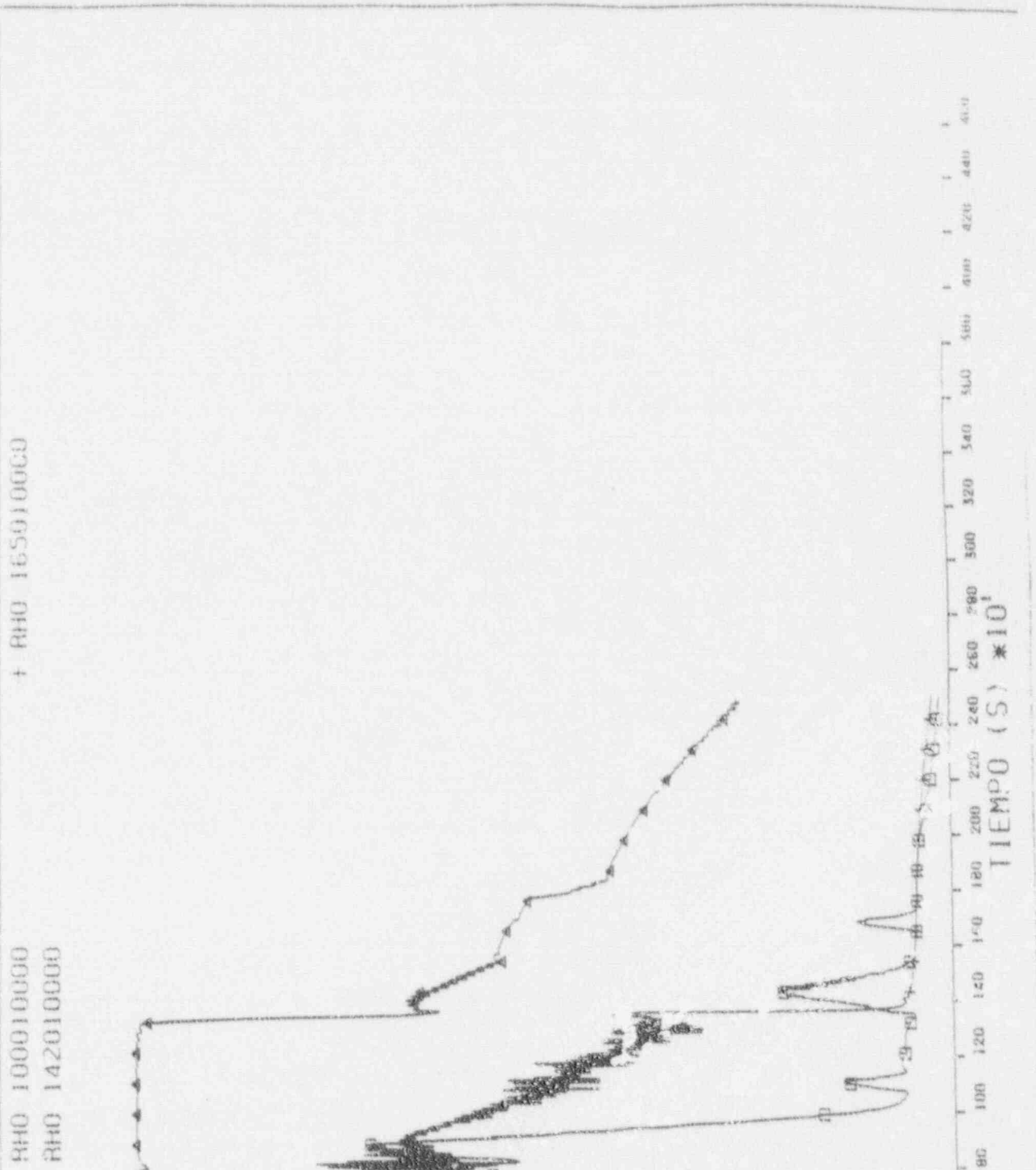

□ RHO 100010000
△ RHO 142010000
+ RHO 165010000
DENSIDADES DEL PRIMARIO (KG/M3)
TIEMPO (S) *10¹

FIG. II (A) - 21

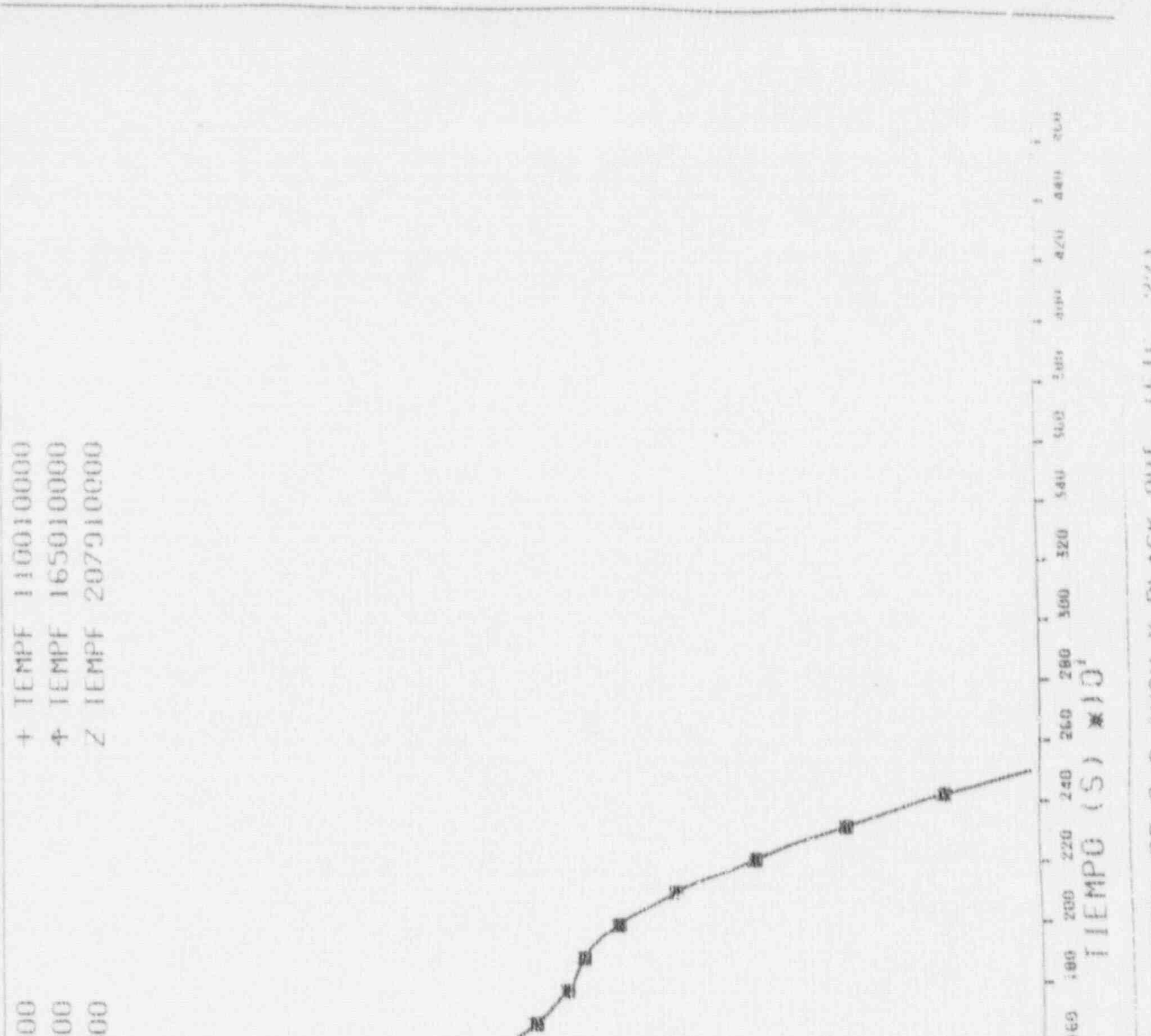

TEMPF 100010000
TEMPF 130010000
TEMPF 211010000
TEMPF 110010000
TEMPF 165010000
TEMPF 207910000
TEMPERATURAS EN PRIMARIO (K)
TIEMPO (S) *10^1
ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG 22)

FIG. II (A) - 22

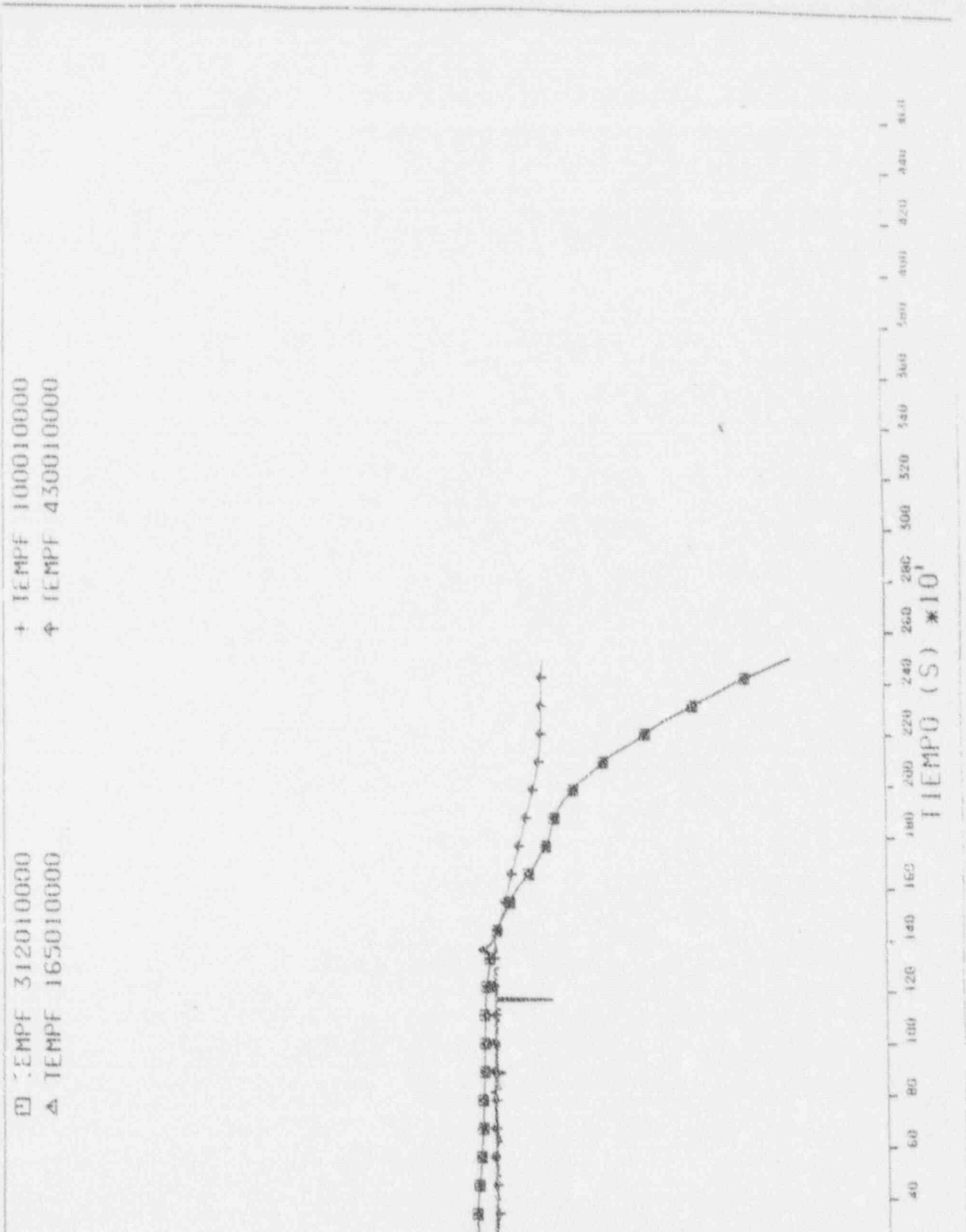

TEMPF 312010000
TEMPF 165010000
TEMPF 100010000
TEMPF 430010000
TIEMPO (S) *10¹
TEMPERATURAS EN PRIMARIO Y G.V. (K)

FIG. II (A) - 23

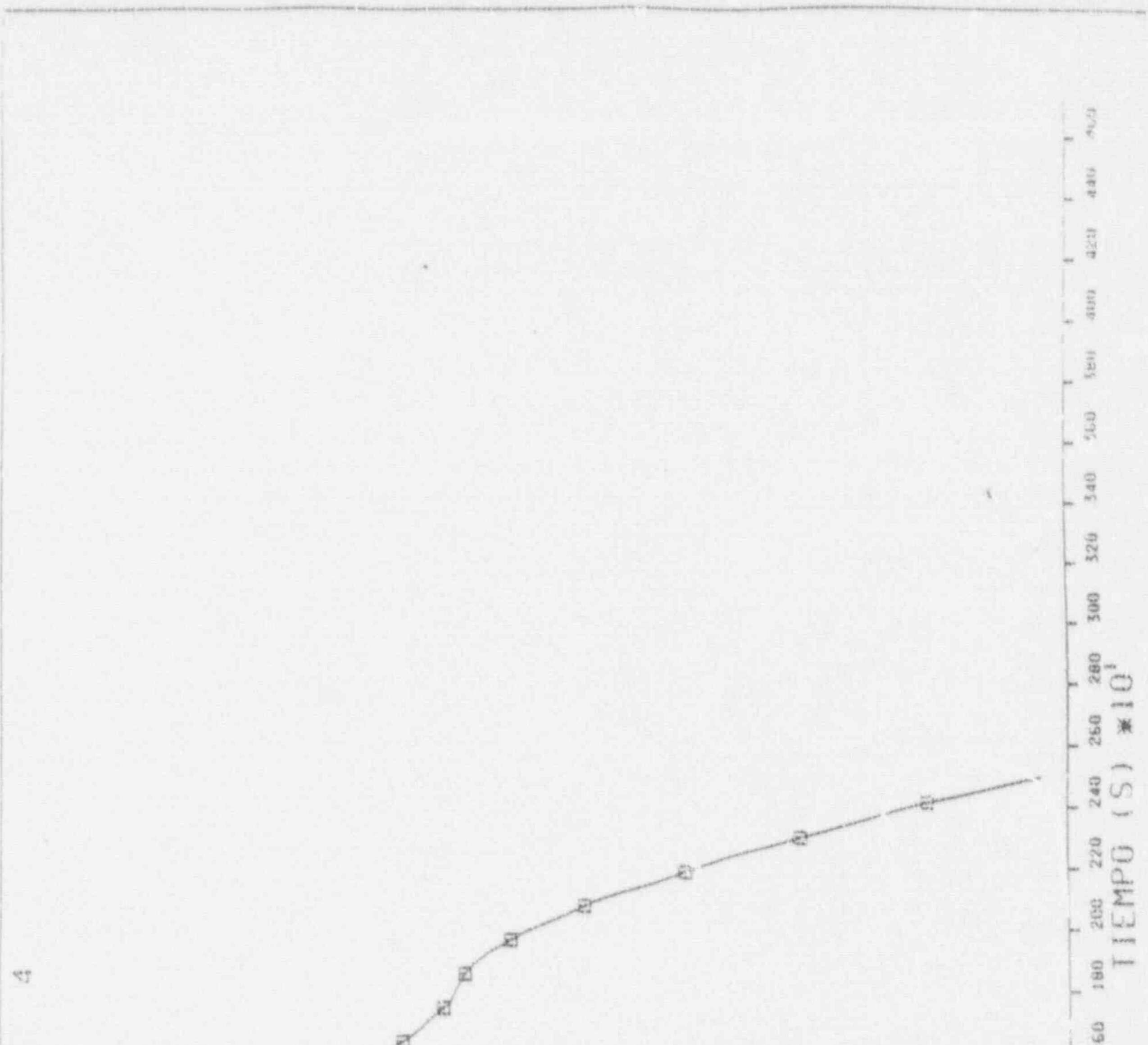

CNTRLVAR 4
TIEMPO (S) *10[1]
TEMPERATURA MEDIA DEL PRIMARIO (C)
C.N. J.C. ROTURA EN RAMA FRIA DE 2.0 INCH Y BLACK-OUT (FIG. 24)

FIG. II (A) - 24

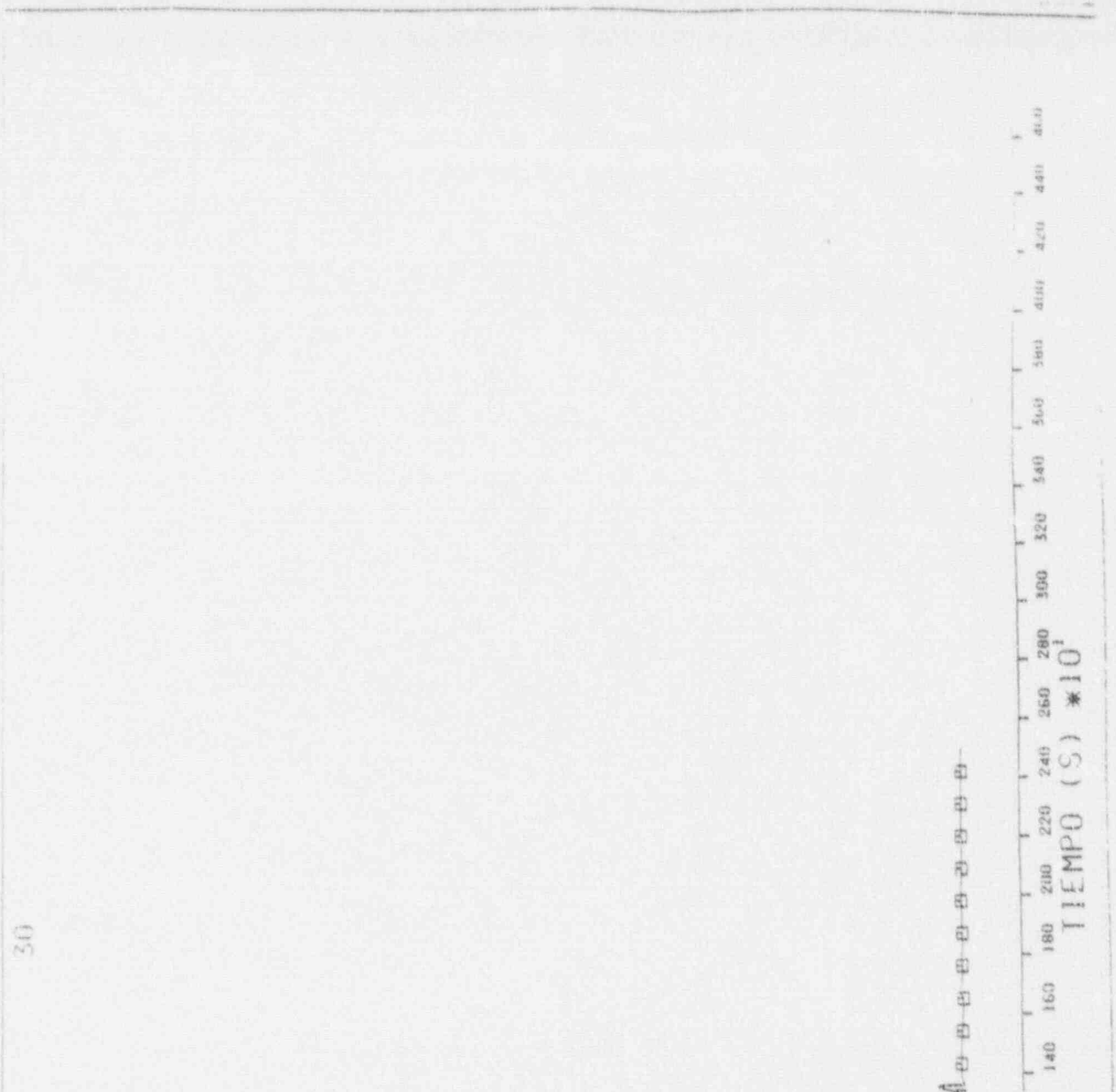


FIG. II (A) - 25

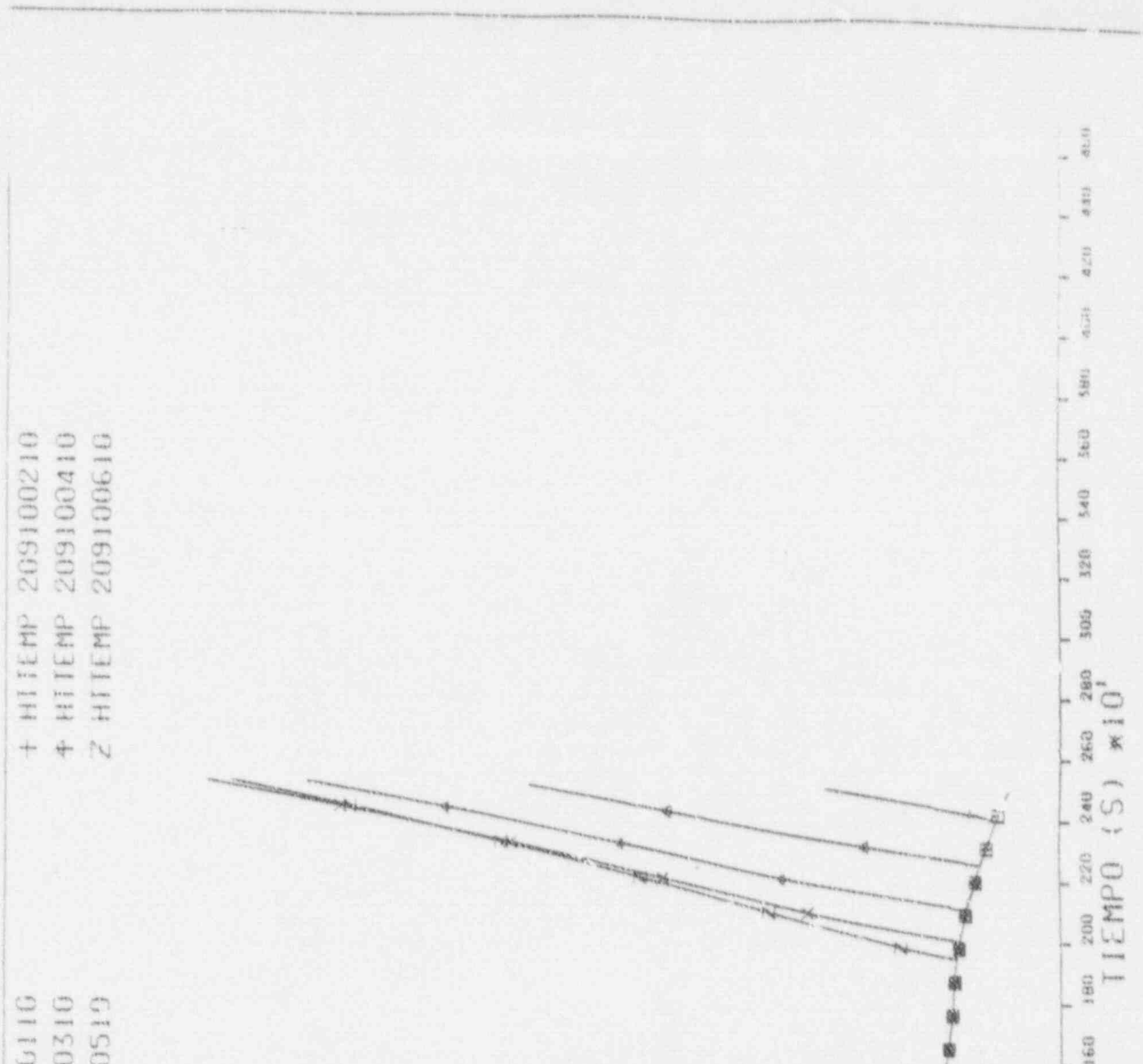

HTTEMP 209100110
HTTEMP 209100210
HTTEMP 209100310
HTTEMP 209100410
HTTEMP 209100510
HTTEMP 209100610
TIEMPO (S) *10[1]
TEMPERATURA VAINAS CANAL MEDIO (K)

FIG. II (A) - 26

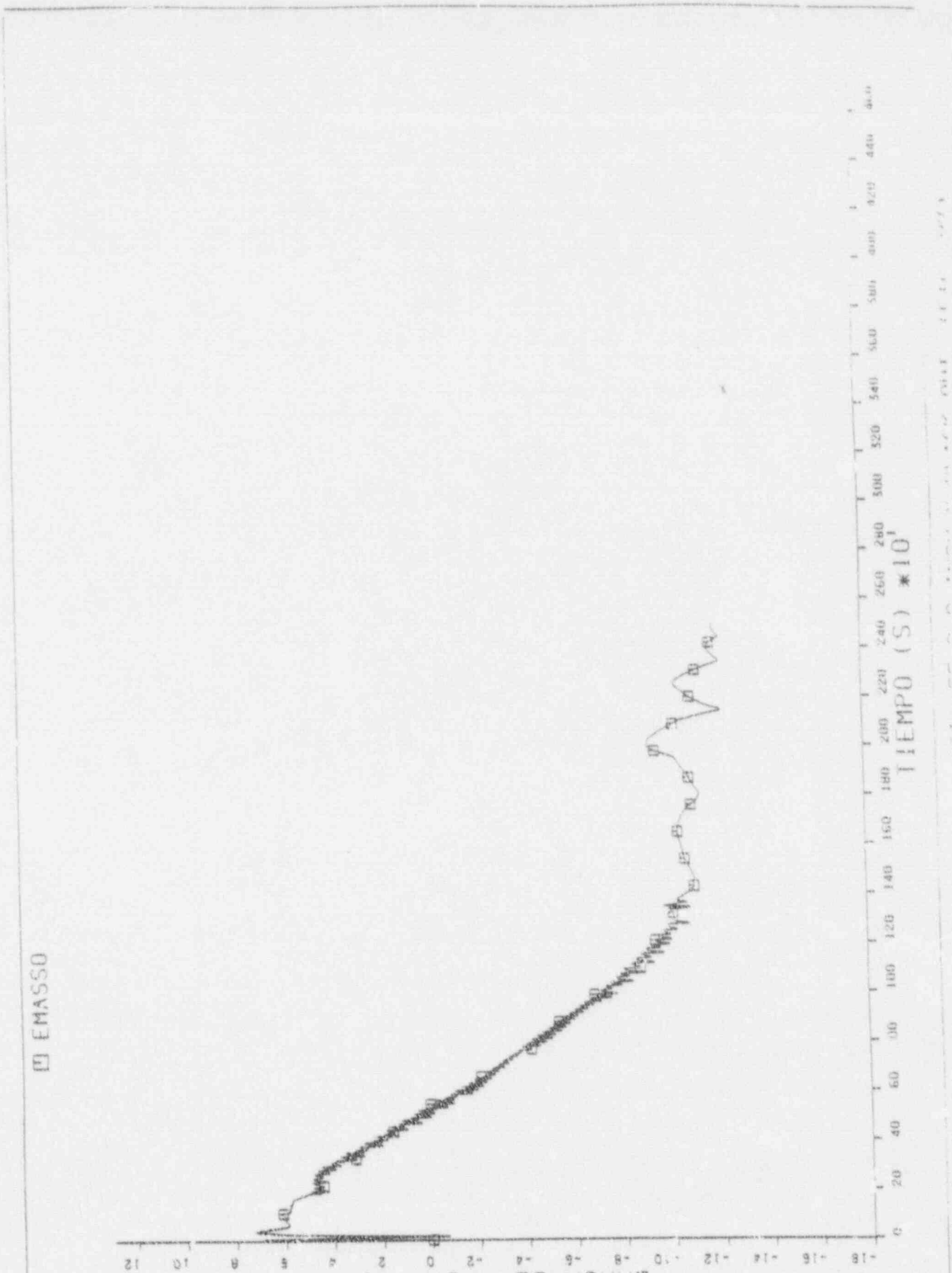


FIG. II (A) - 27

APPENDIX III : FIGURES OF CASE III(A), (1.5" WITHOUT RECOVERY)

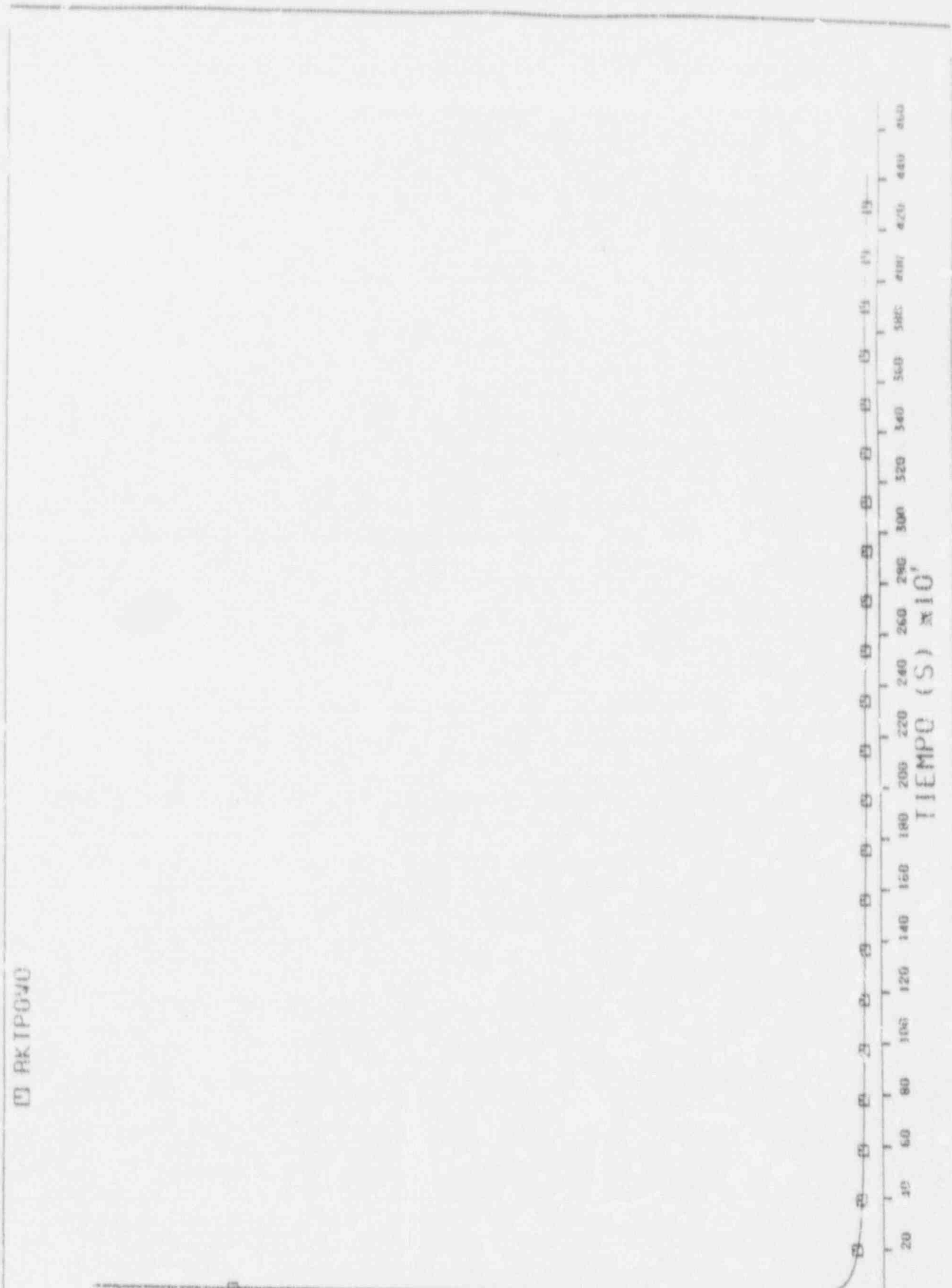
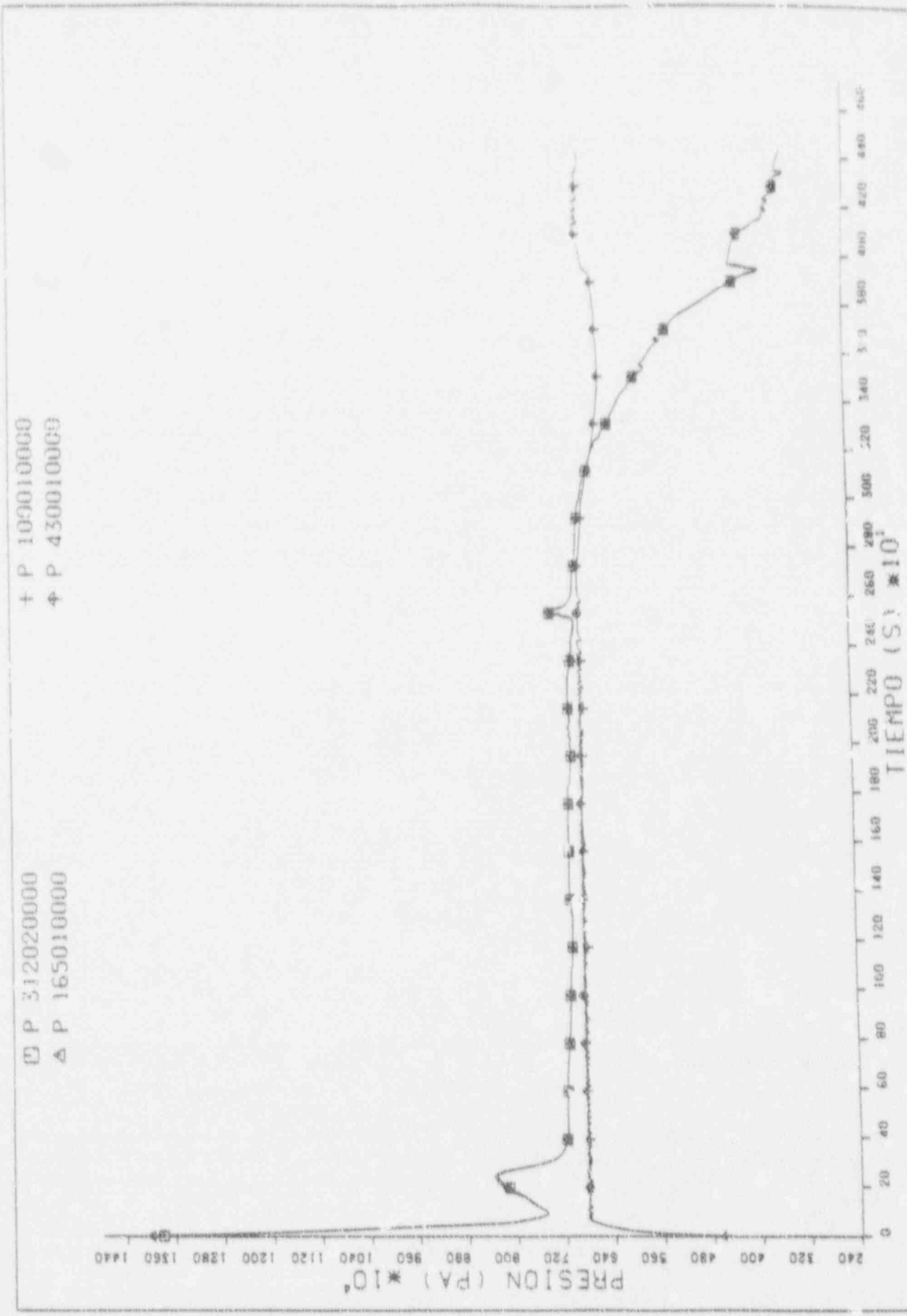
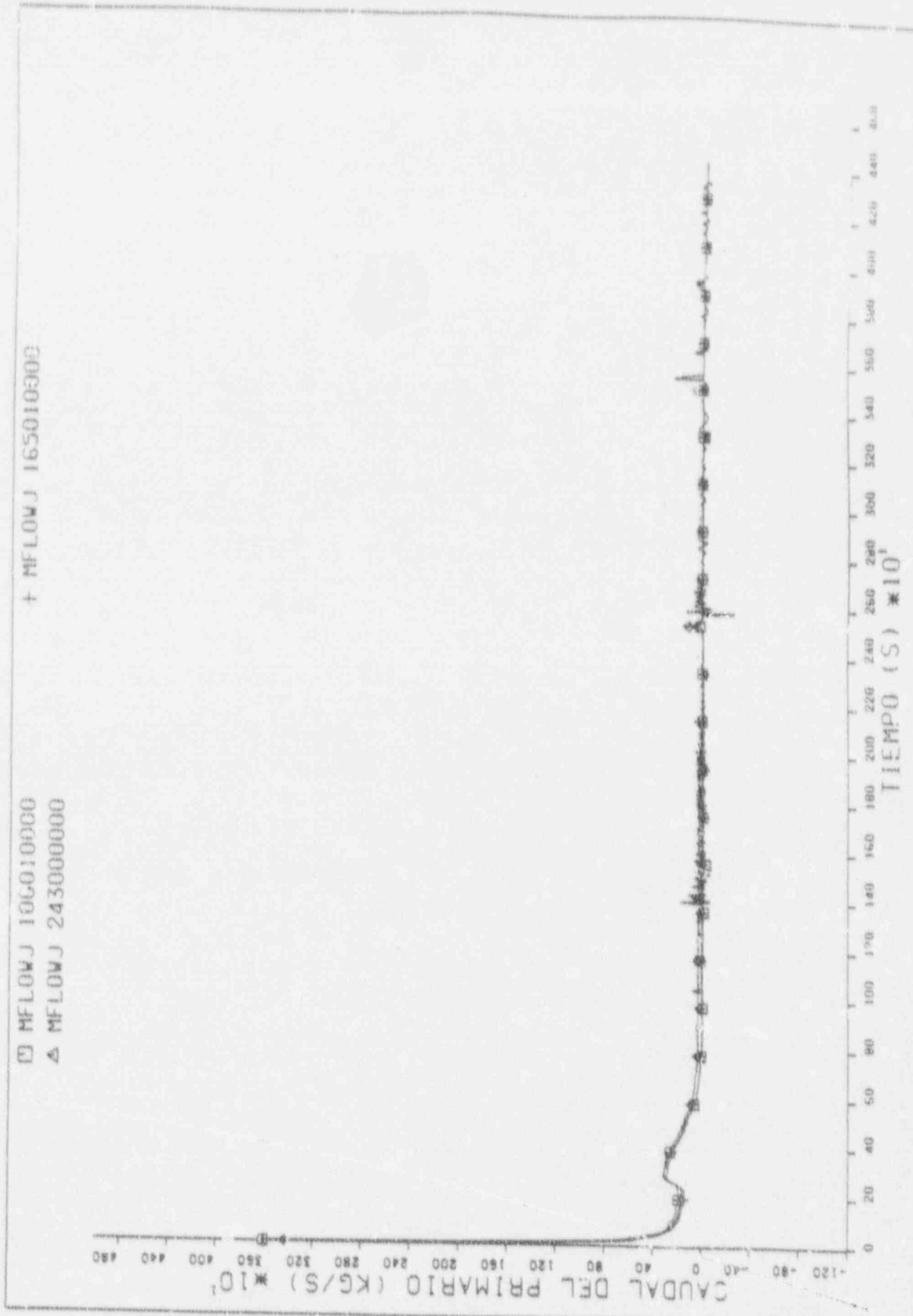


FIG. III (A) - 1



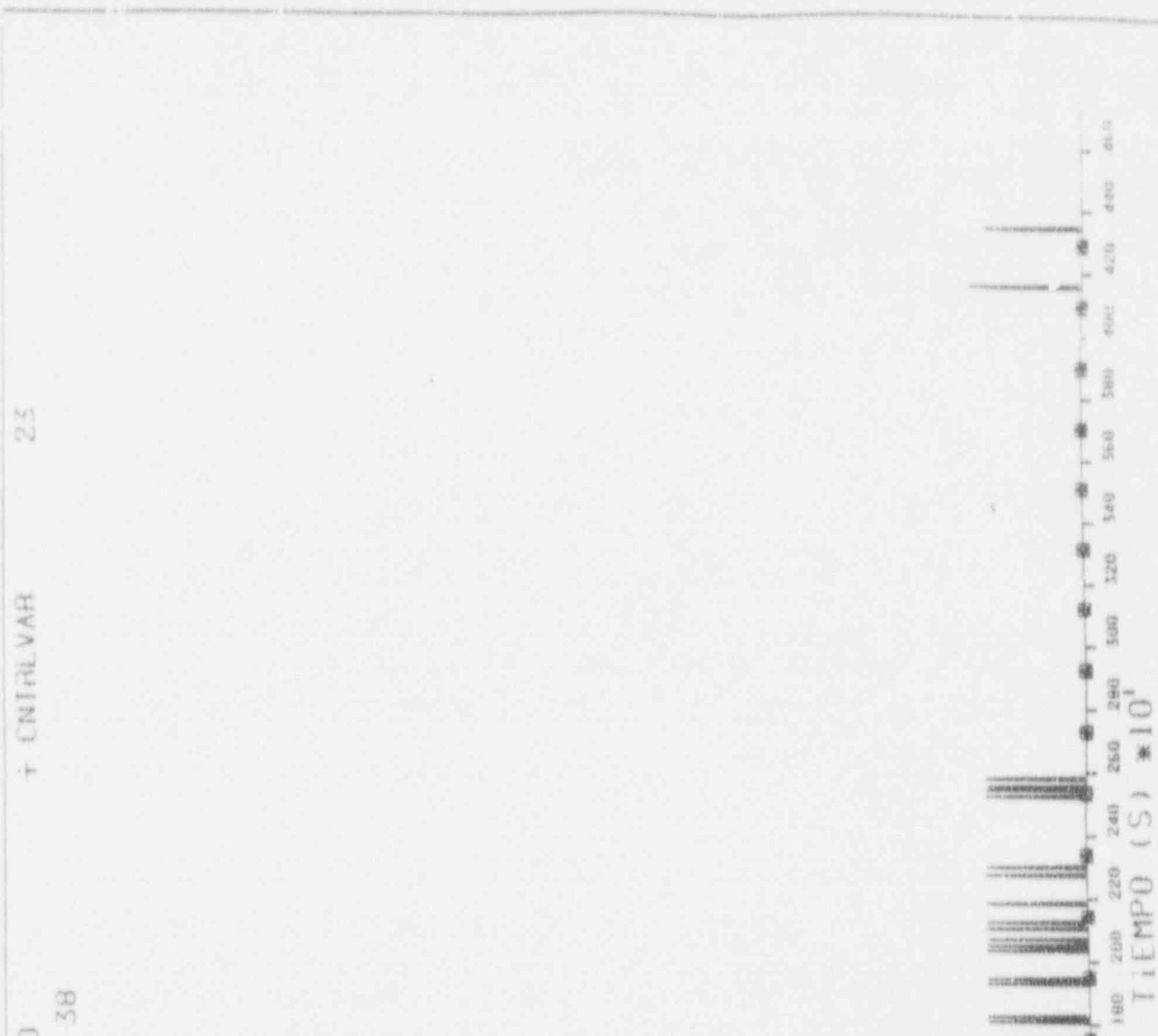
P 312020000
P 165010000
P 100010000
P 430010000
PRESION (PA) *10^4
TIEMPO (S) *10^1
ROTURA SB3 (1.5 INCH) SIN RECUPERACION

FIG. III (A) - 2



MFLOWJ 100010000
MFLOWJ 243000000
+ MFLOWJ 165010000
CAUDAL DEL PRIMARIO (KG/S) *10¹
TIEMPO (S) *10¹

FIG. III (A) - 3



MFLOWJ 514000000
CNTRLVAR 38
+ CNTRLVAR 23
CAUDALES DE SECUNDARIO (KG/S)
TIEMPO (S) *10^1

FIG. III (A) - 4

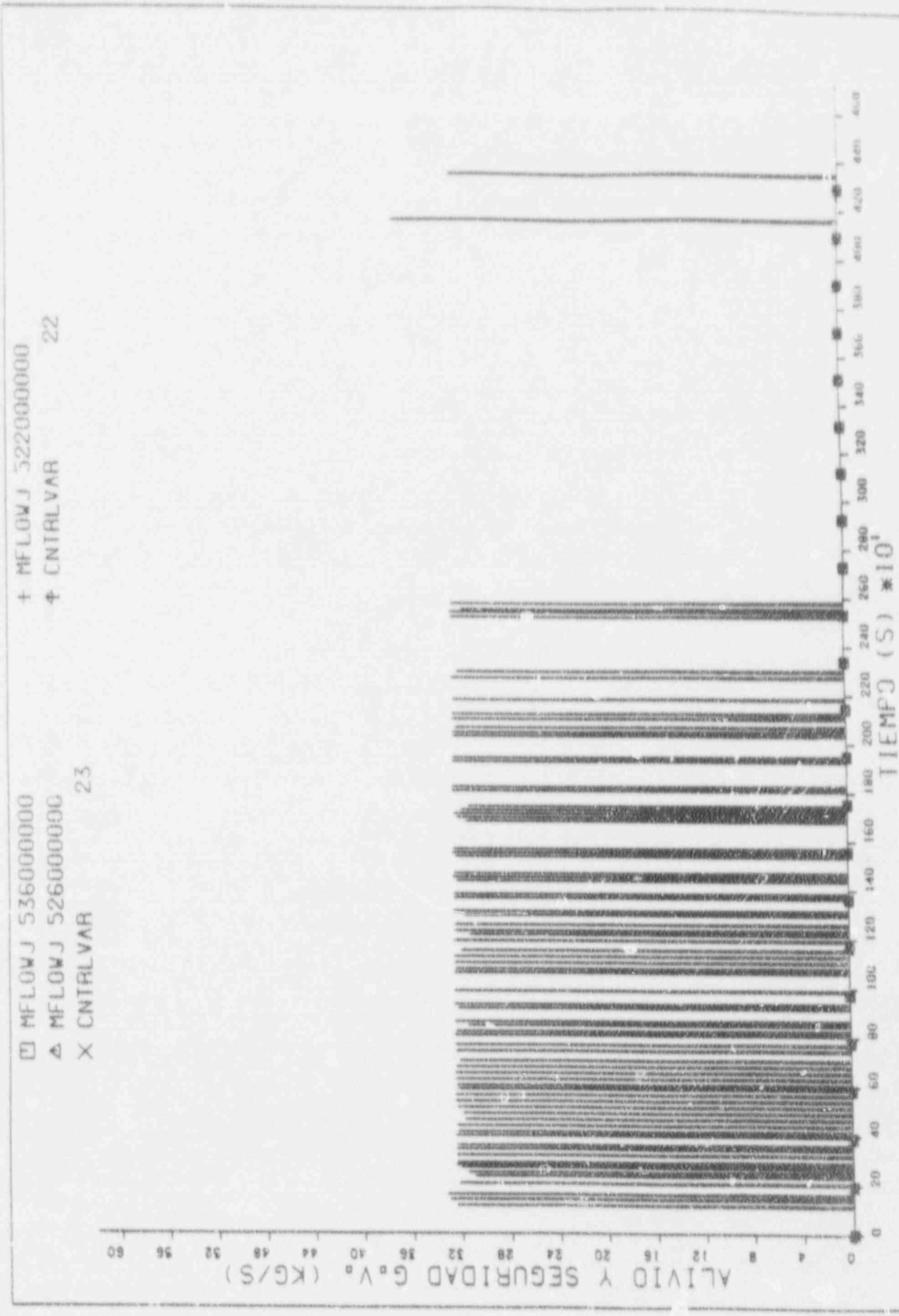

MFLOWJ 536000000
MFLOWJ 526000000
CNTRLVAR 23
MFLOWJ 322000000
CNTRLVAR 22
TIEMPO (S) *10^1
ALIVIO Y SEGURIDAD G.V. (KG/S)

FIG. III (A) - 5

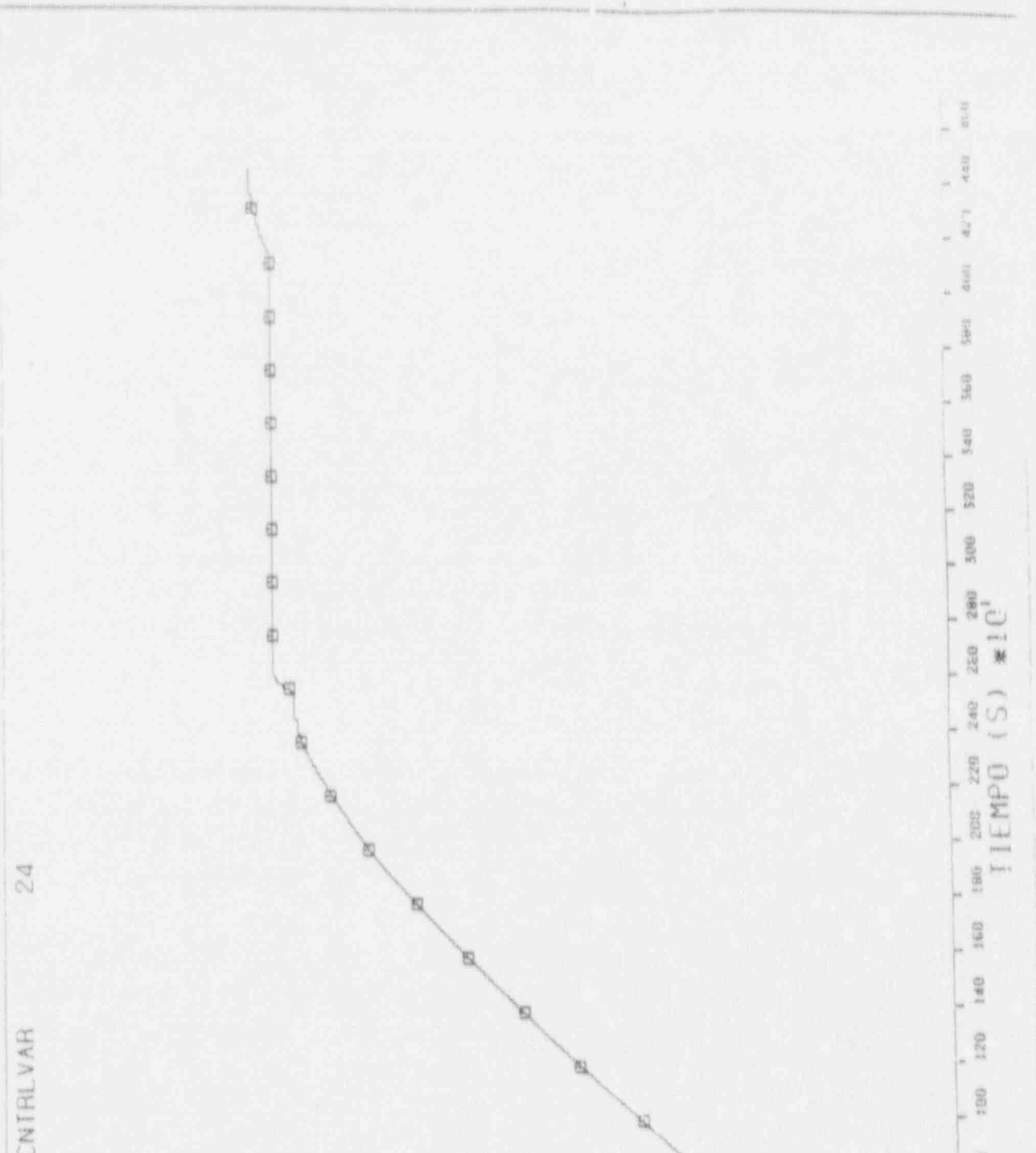
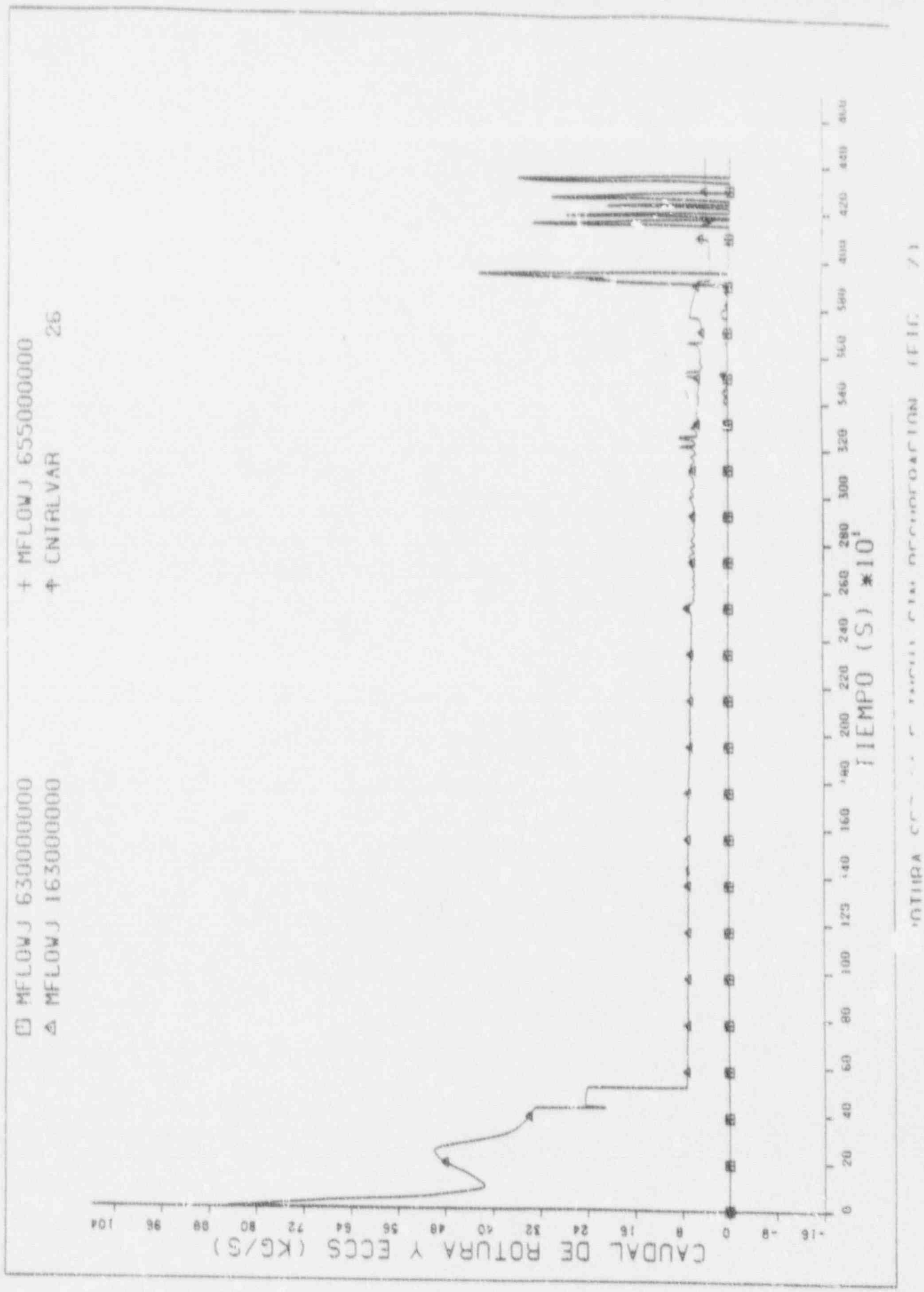

CNTRLVAR 24
TIEMPO (S) *10^1
MASA DESCARGADA DEL G.V. A ATM (KG) *10^2
ROTURA SB3 (1.5 INCH) SIN RECUPERACION

FIG. III (A) - 6



MFLOWJ 630000000
MFLOWJ 163000000
MFLOWJ 655000000
CNTRLVAR 26
CAUDAL DE ROTURA Y ECCS (KG/S)
TIEMPO (S) *10¹

FIG. III (A) - 7

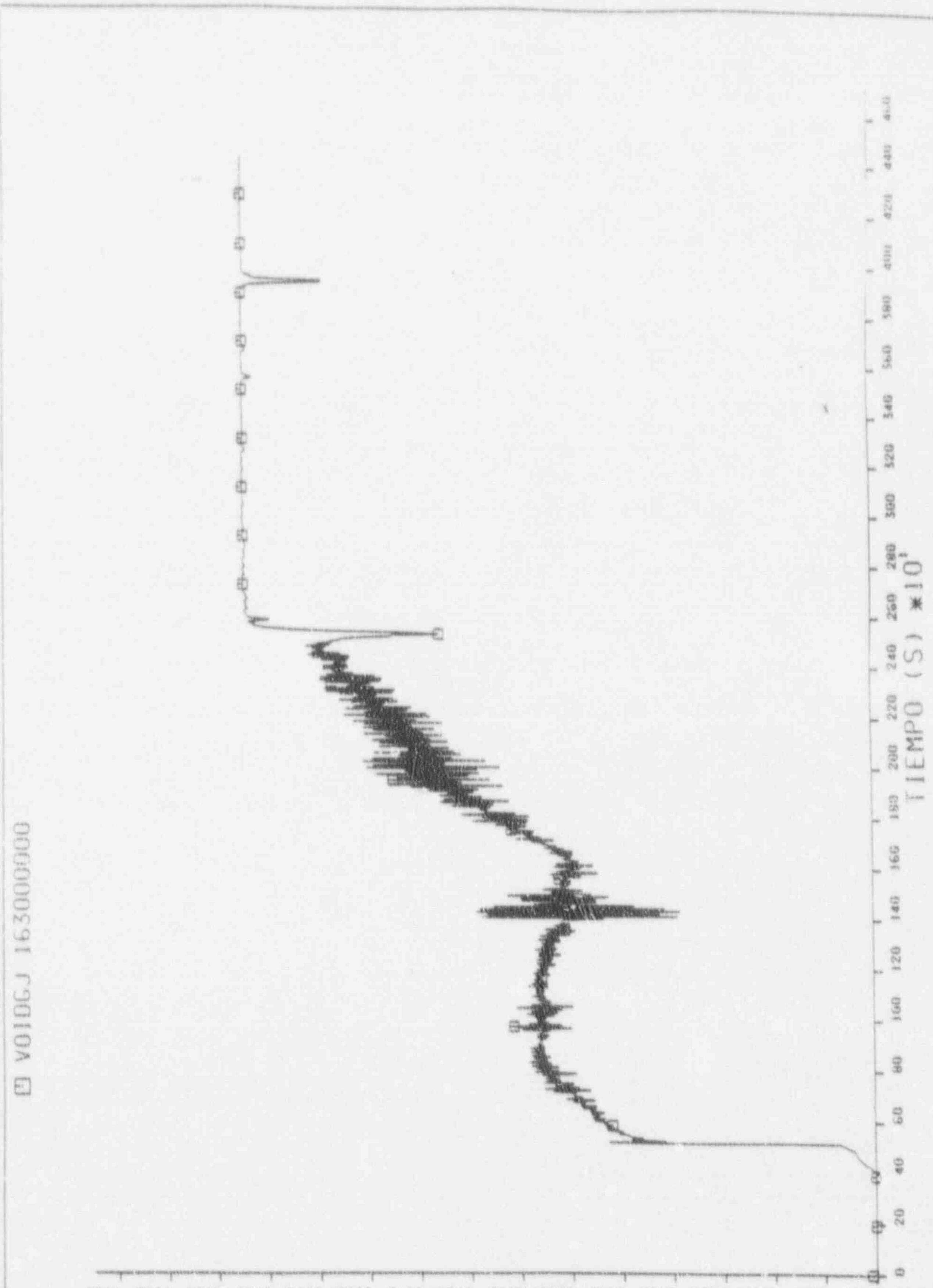

VOIDGJ 163000000
FRACCION DE VAPOR EN LA ROTURA (O/1)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (1.5 INCH) SIN RECUPERACION

FIG. III (A) - 8

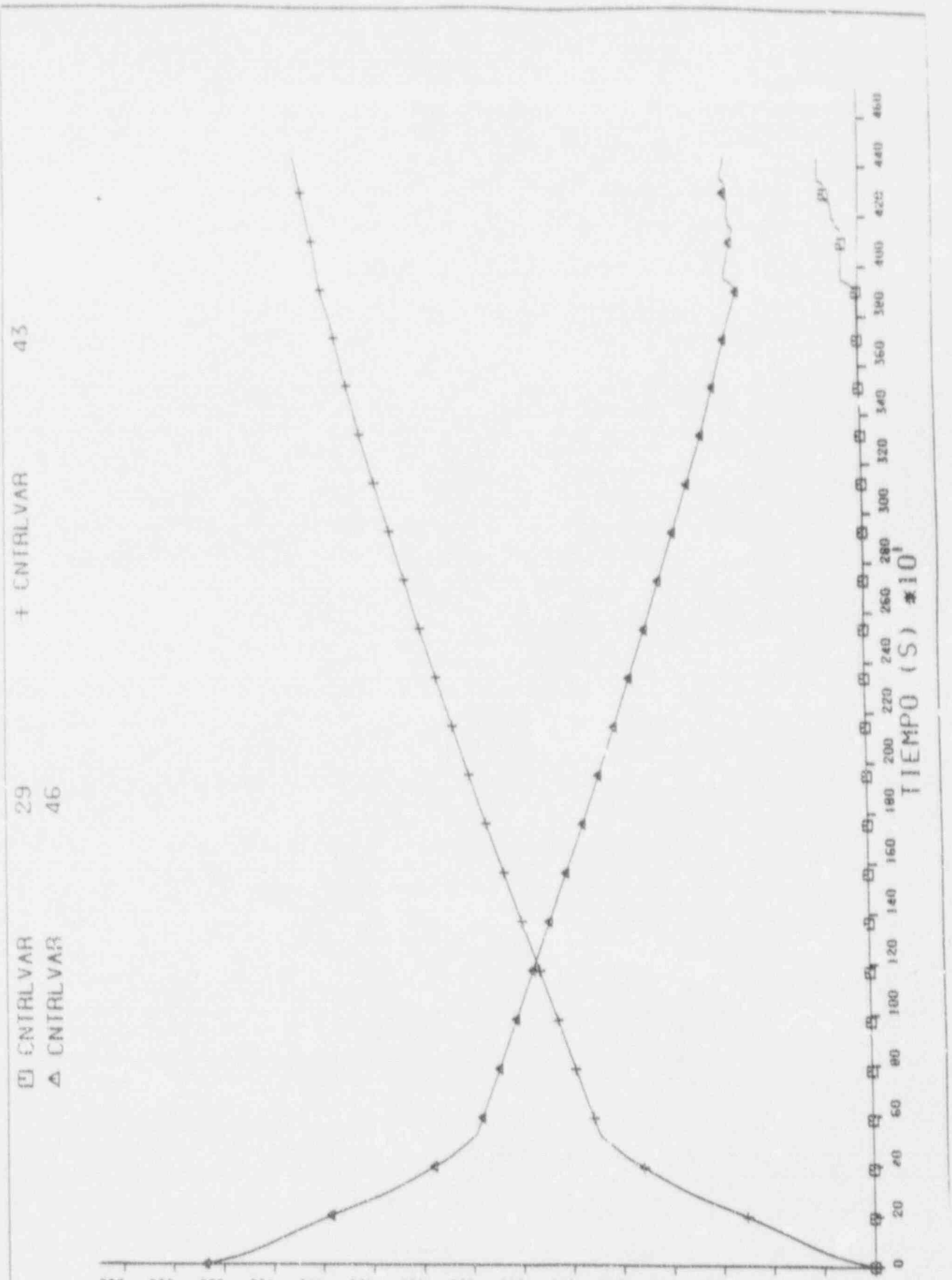

□ CNTRLVAR 29
△ CNTRLVAR 46
+ CNTRLVAR 43
TIEMPO (S) *10¹
BALANCE MASICO DEL PRIMARIO (KG) *10²

FIG. III (A) - 9

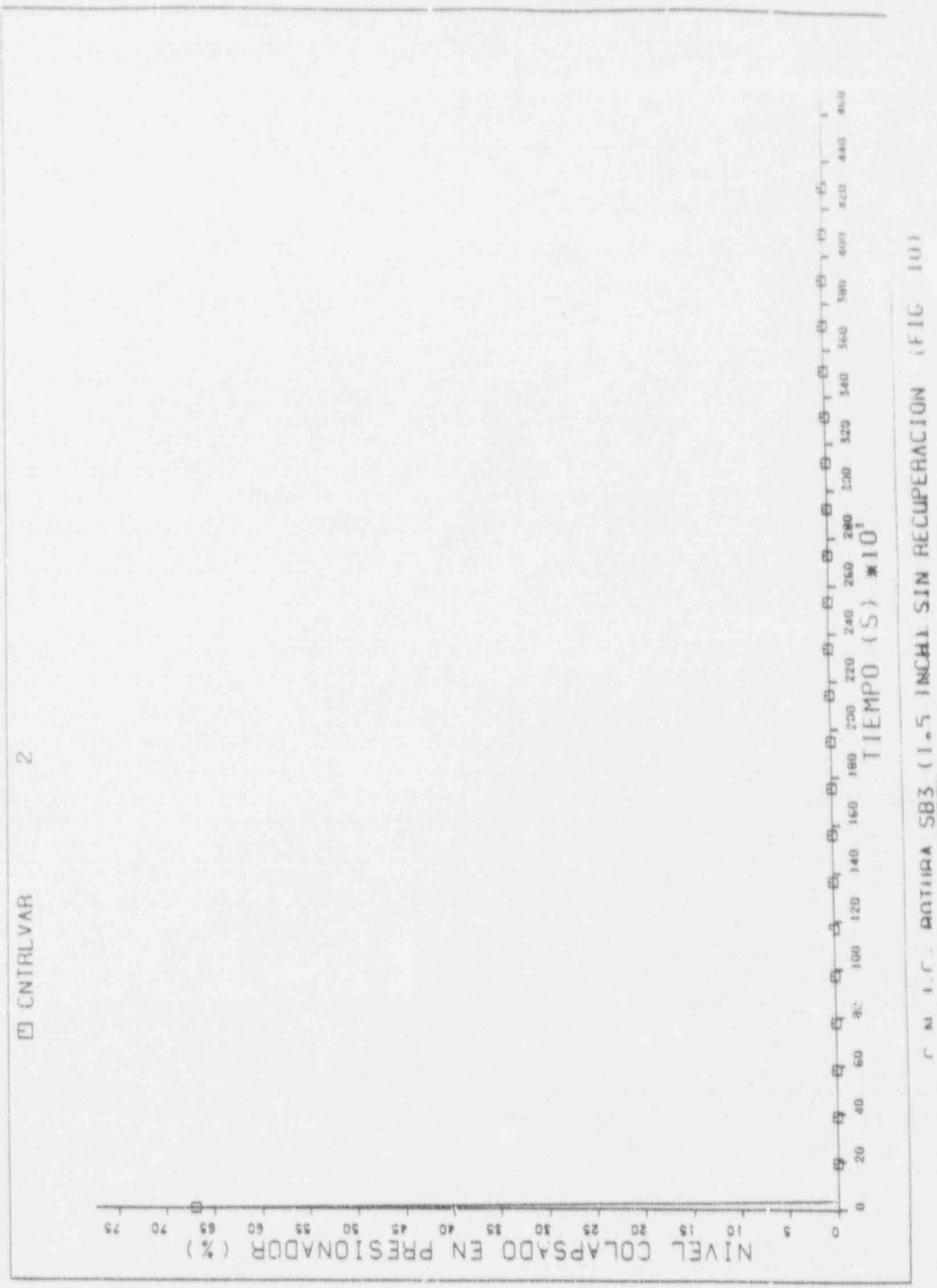
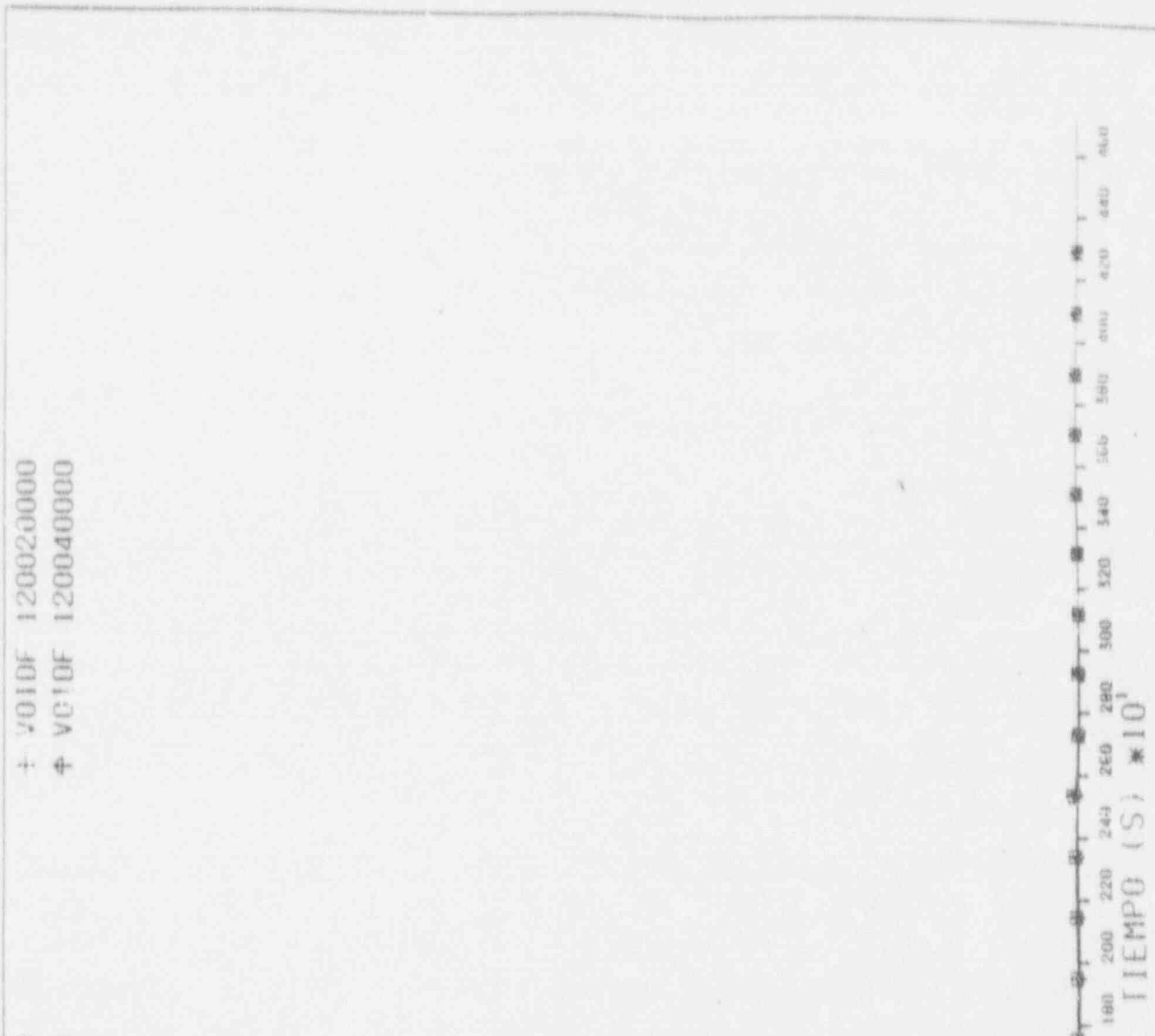

CNTRLVAR 2
TIEMPO (S) *10^1
NIVEL COLAPSADO EN PRESIONADOR (%)
ROTURA SB3 (1.5 INCH) SIN RECUPERACION (FIG 10)

FIG. III (A) - 10



VOIDF 120010000
VOIDF 120020000
VOIDF 120030000
VOIDF 120040000
TIEMPO (S) *10^1
FRAC. DE LIQUIDO EN TUBO UP G.V. (O/1)

FIG. XII (A) - 11

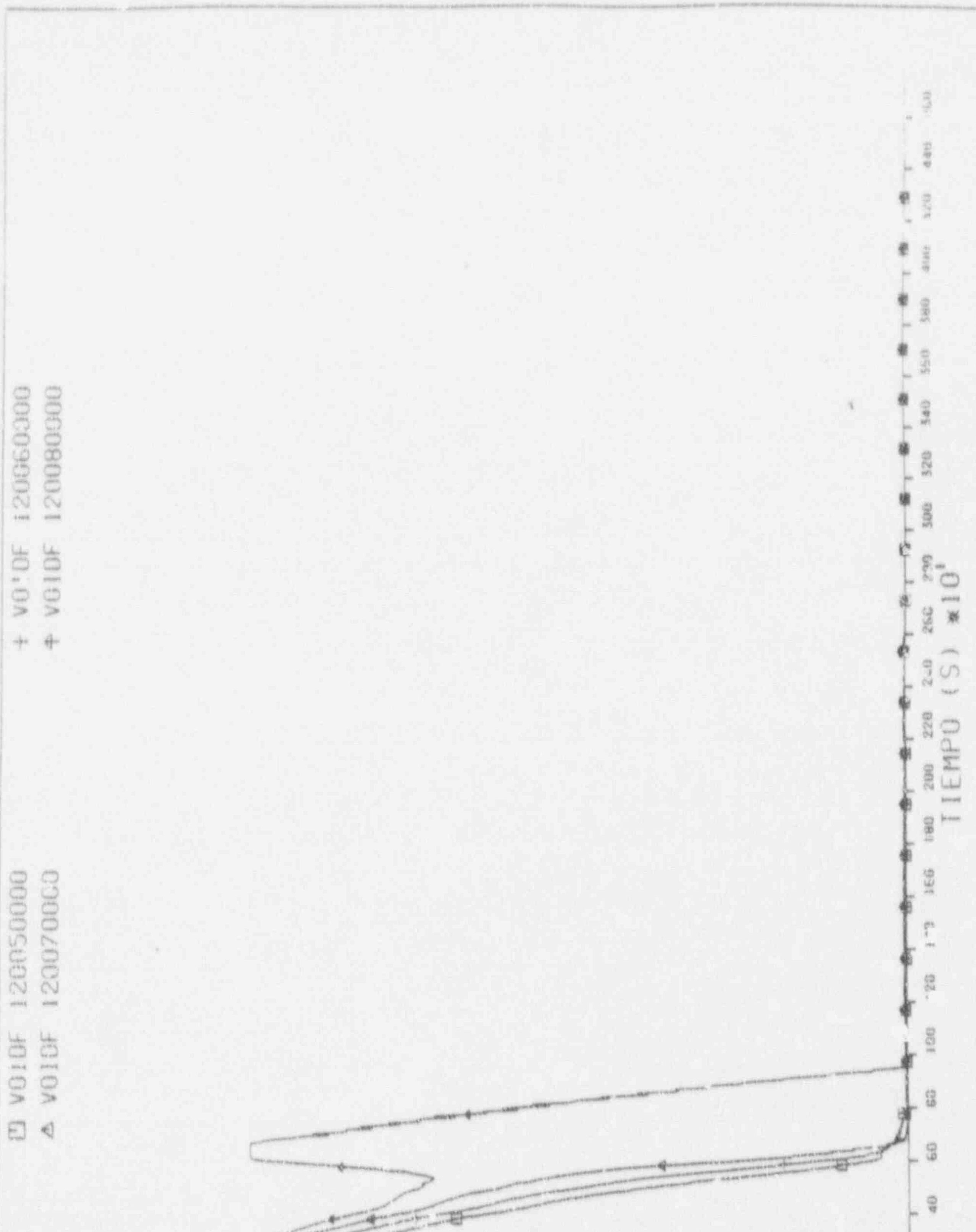
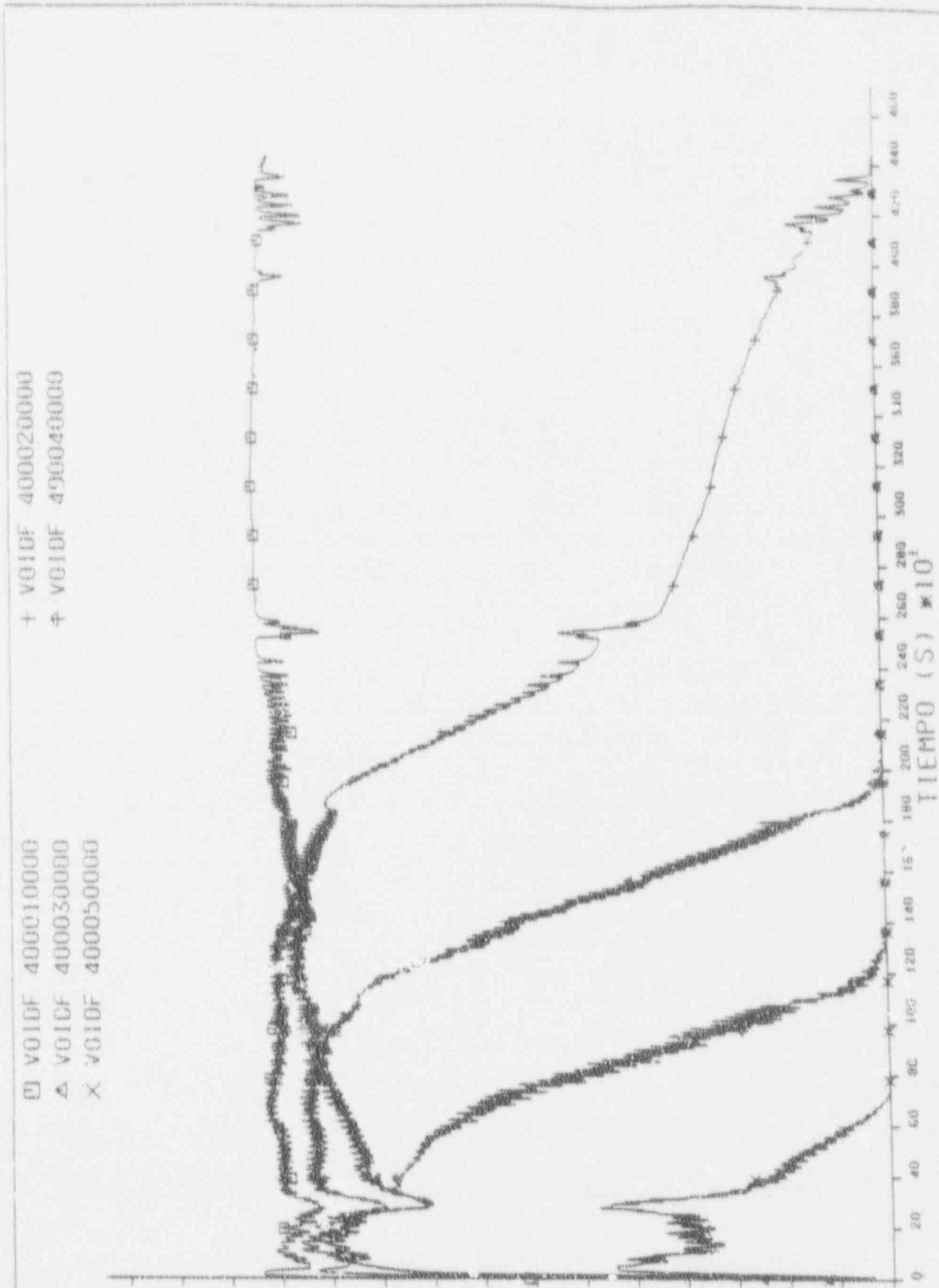

VOIDF 120050000
VOIDF 120070000
VOIDF 120060000
VOIDF 120080000
FRAC. DE LIQUIDO EN TUBO DOWN G.V. (O/1)
TIEMPO (S) *10[1]
ROTURA SB3 (1.5 INCH) SIN RECUPERACION (FIG 12)

FIG. III (A) - 12

FIG. III (A) -



□ VOIDF 400010000
△ VOIDF 400030000
X VOIDF 400050000
+ VOIDF 400020000
✦ VOIDF 490040000
FRAC. DE LIQUIDO EN RISER G.V. (O/I)
TIEMPO (S) *10¹
ROTURA SB3 (1.5 INCH) SIN RECUPERACION (FIG 14)

FIG. III (A) - 14

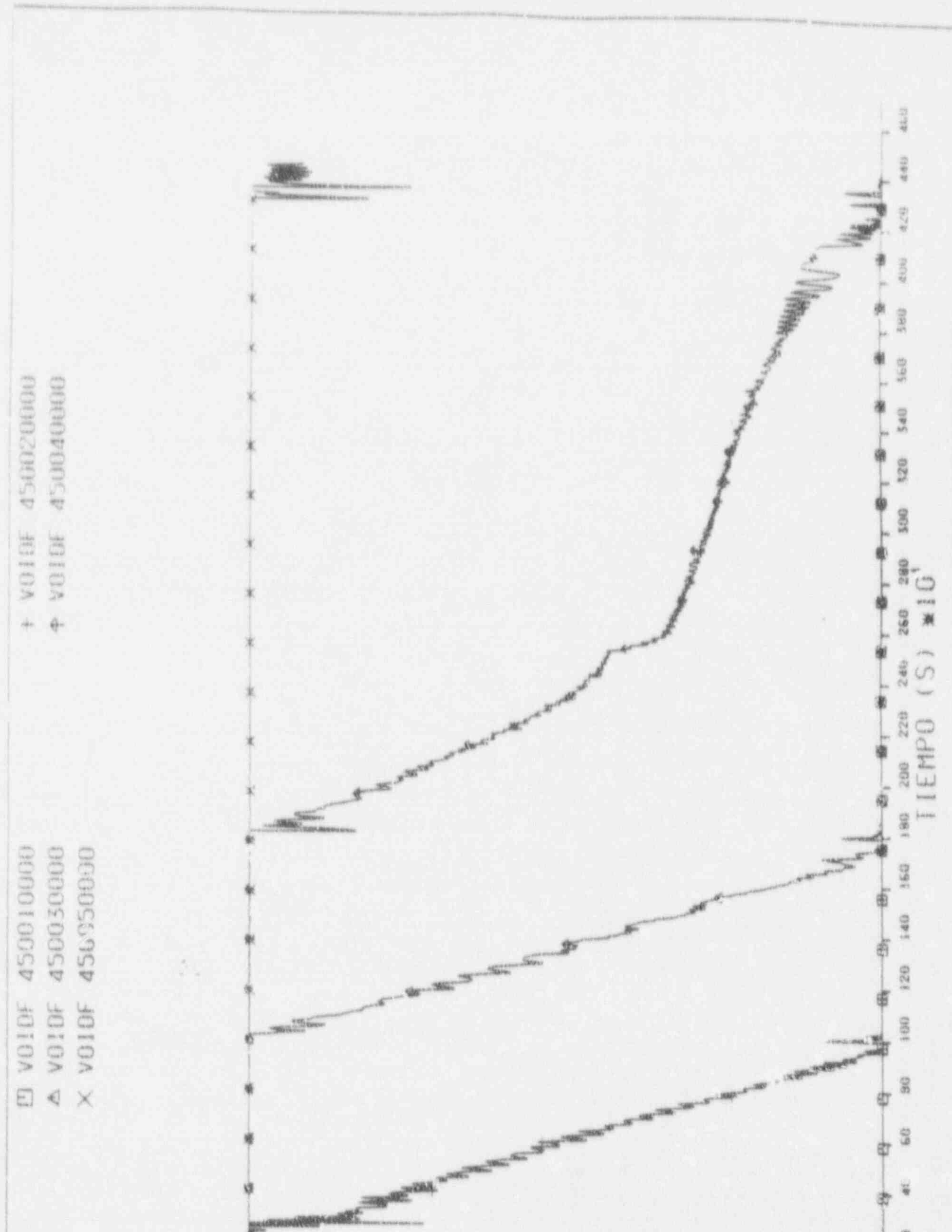

VOIDF 450010000
VOIDF 450030000
VOIDF 450050000
VOIDF 450020000
VOIDF 450040000
TIEMPO (S) *10^1
FRAC. DE LIQUIDO EN DOWNCOMER G.V. (0/1)

FIG. III (a) - 15

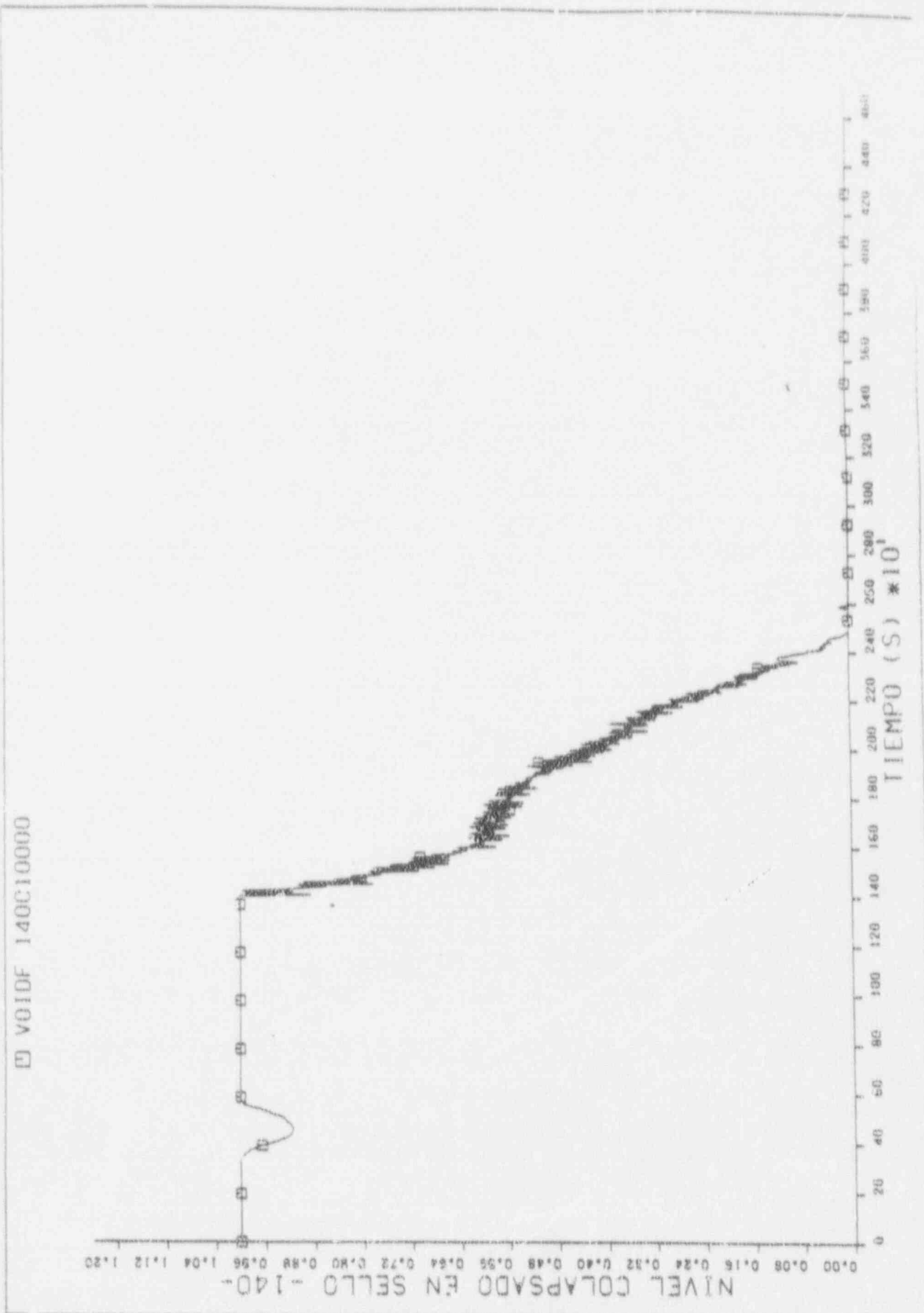


FIG. III (A) - 16

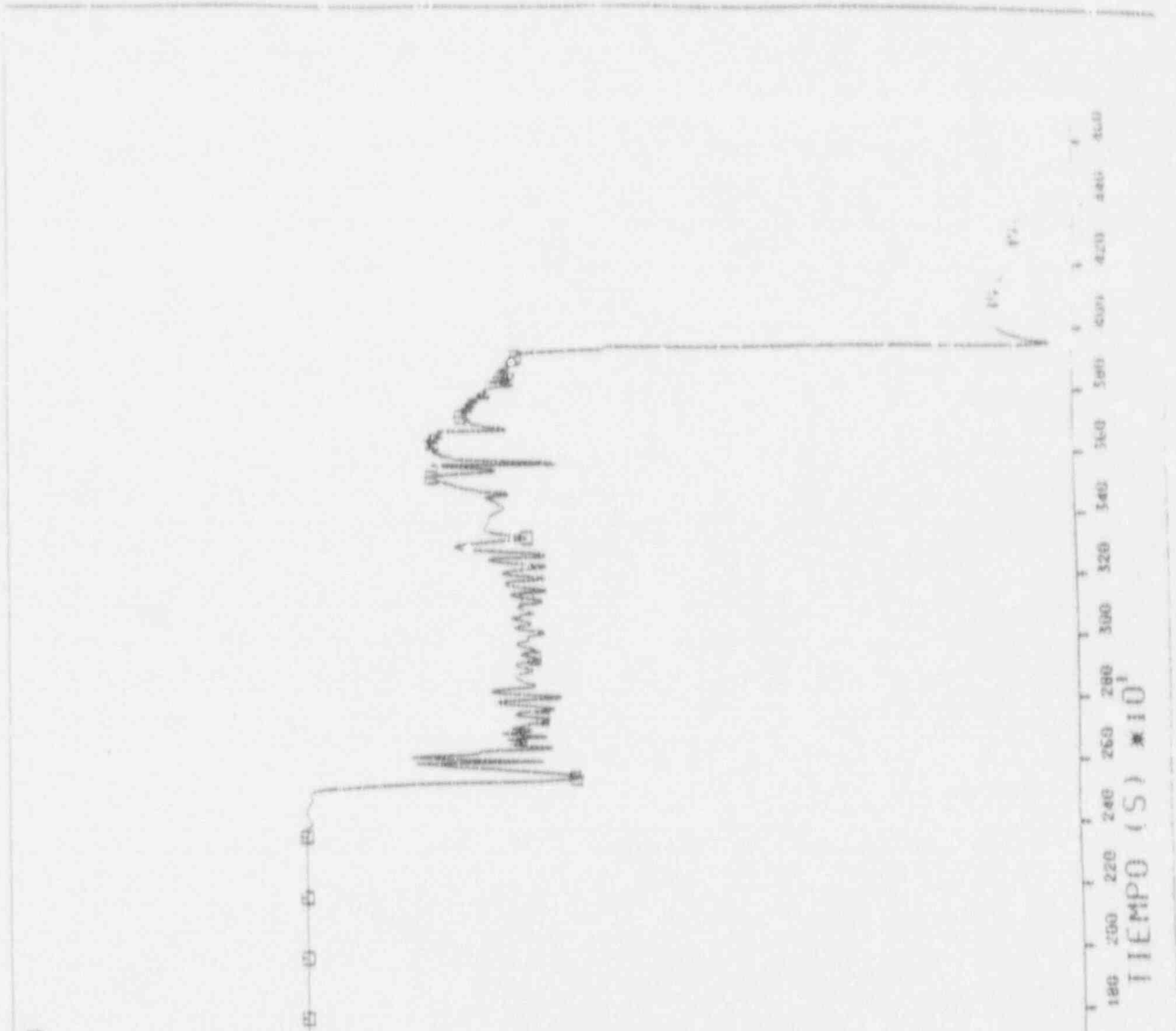


FIG. III (A) - 17

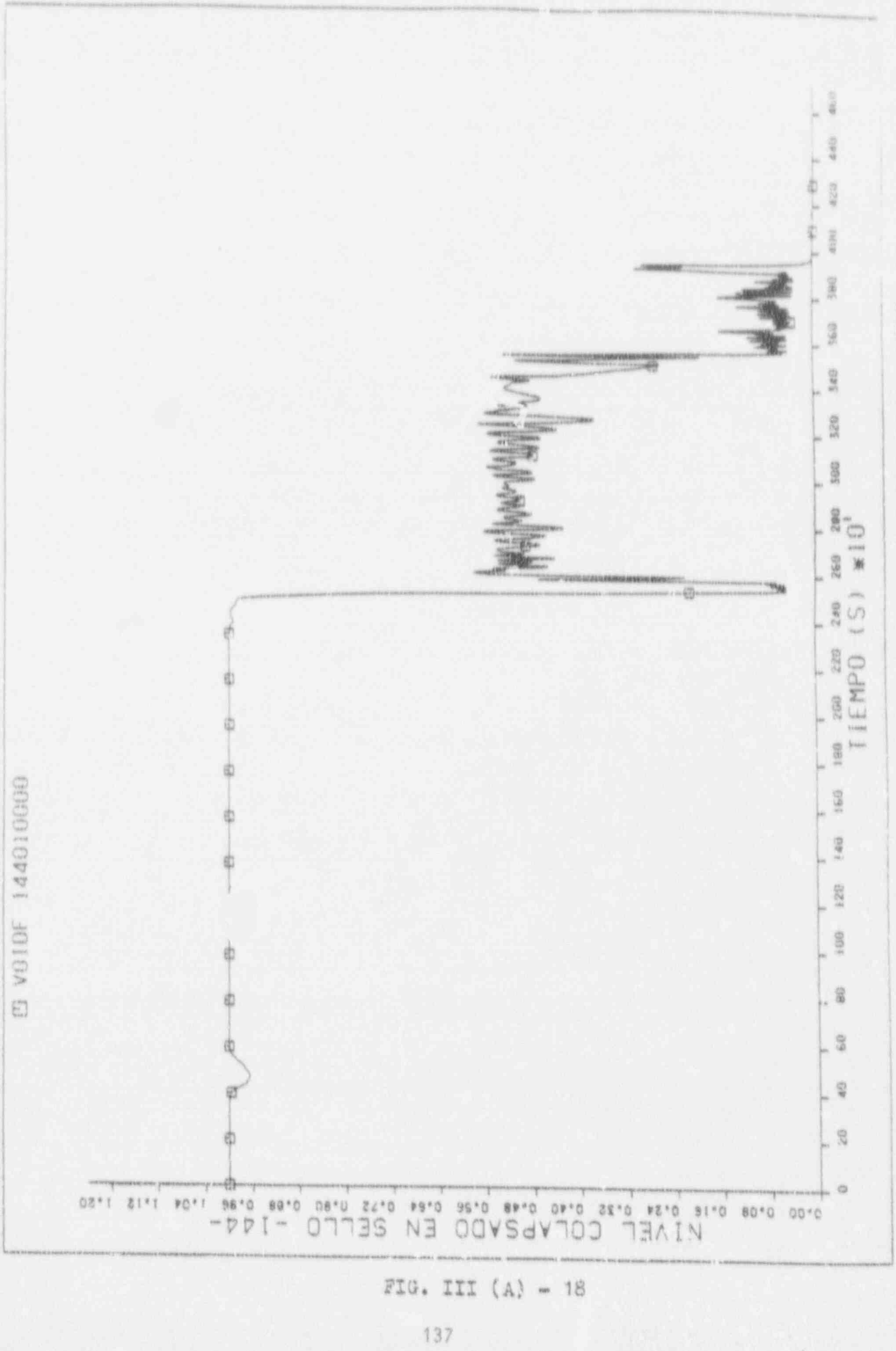

VOIDF 144010000
NIVEL COLAPSADO EN SELLO -144-
TIEMPO (S) *10
SB3 (1.5 INCH) SIN RECUPERACION (FIG 18)

FIG. III (A) - 18

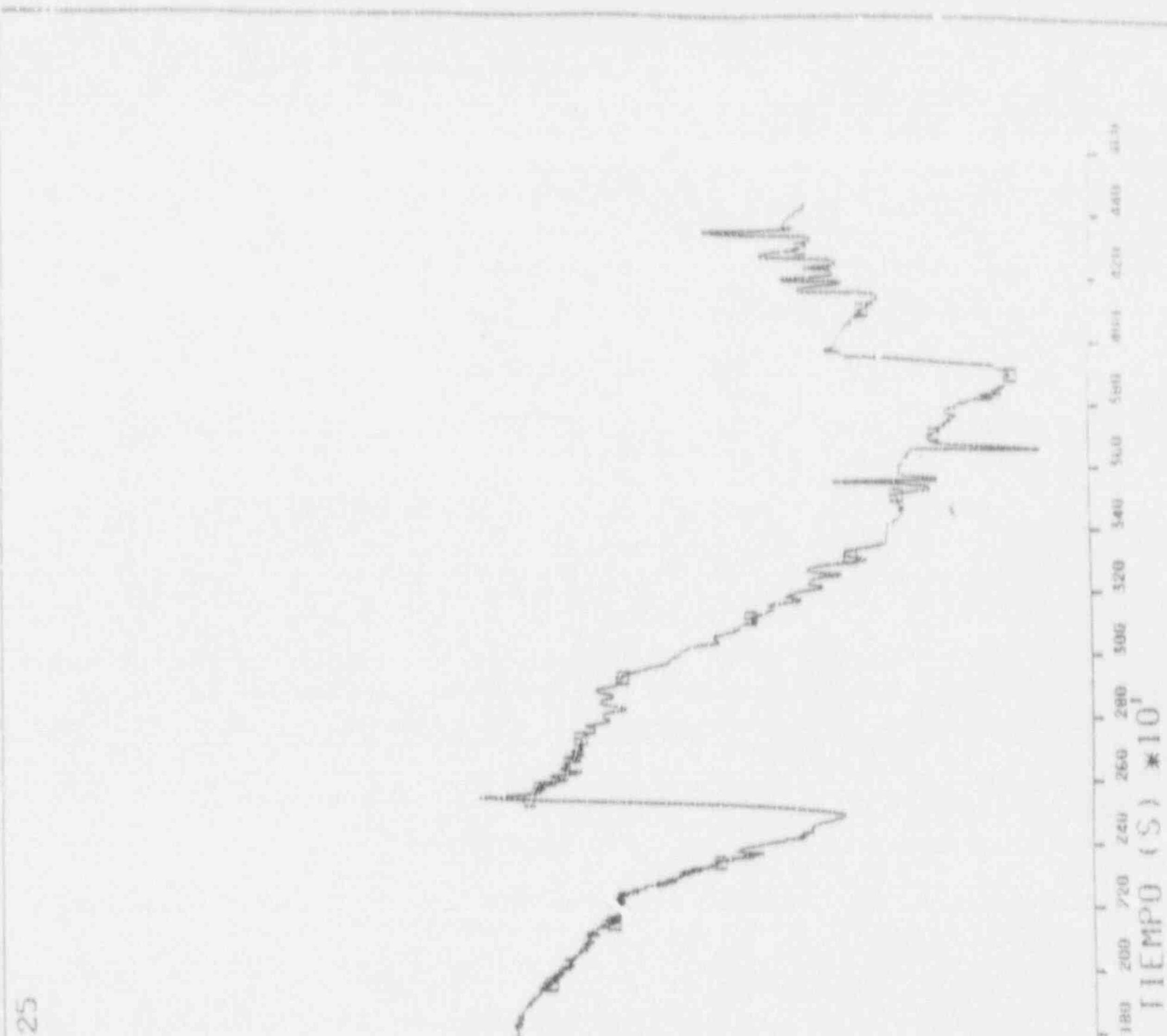


FIG. III (A) - 19

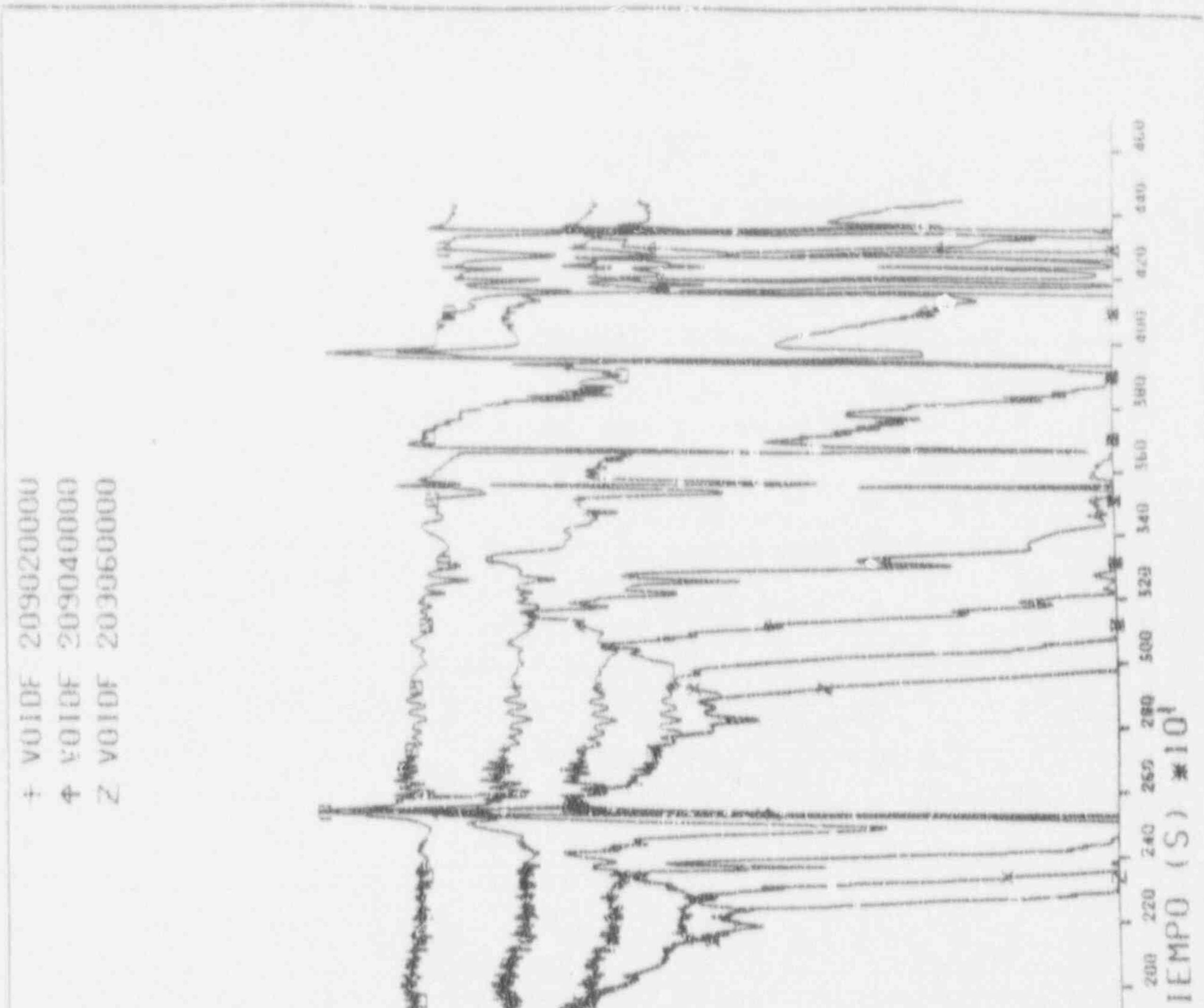

VOIDF 209010000
VOIDF 209030000
VOIDF 209050000
VOIDF 209020000
VOIDF 209040000
TIEMPO (S) *10
FRACCIONES DE LIQUIDO EN EL NUCLEO (O/1)
SIN RECUPERACION

FIG. III (A) - 20

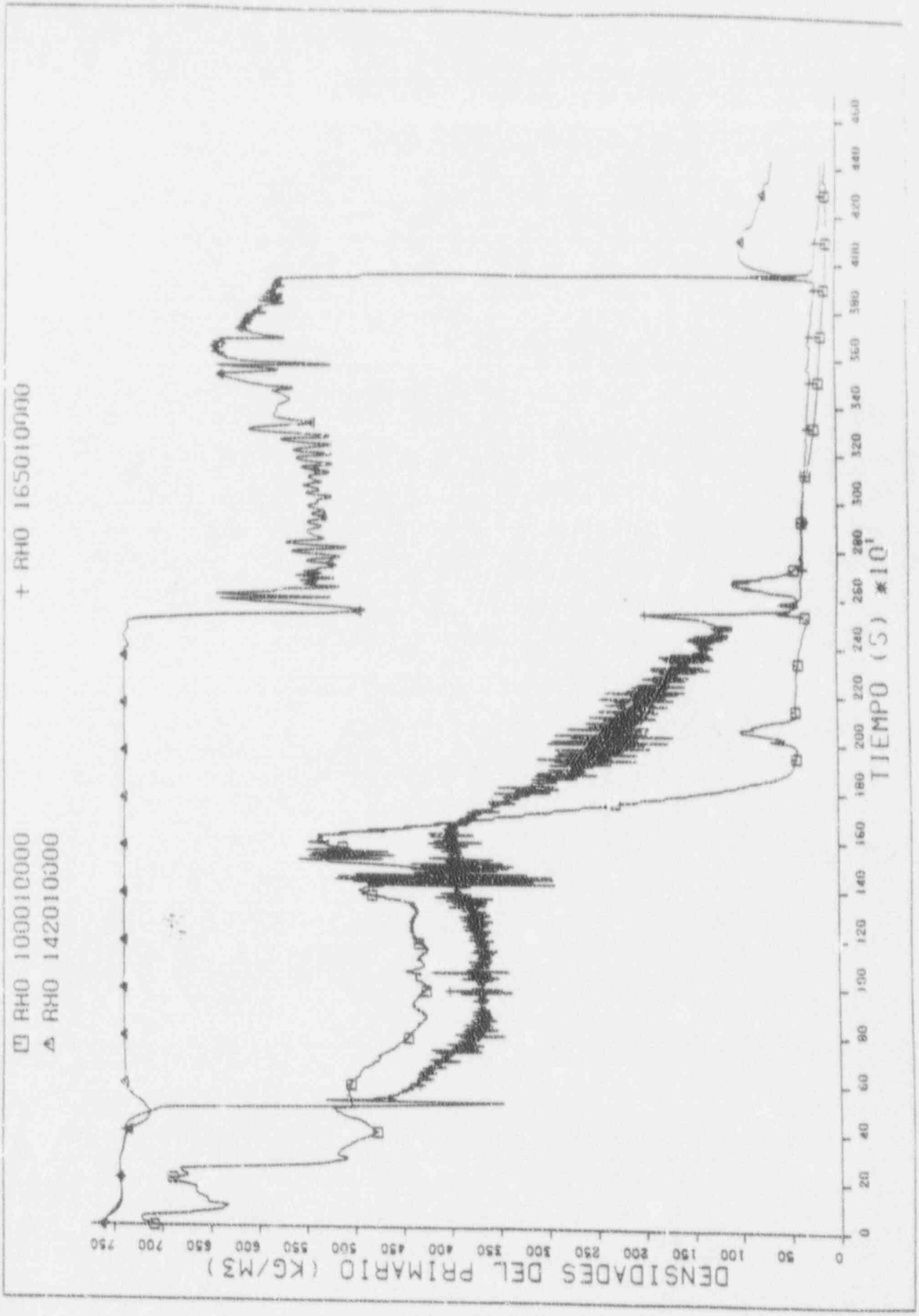

RHO 100010000
RHO 142010000
+ RHO 165010000
DENSIDADES DEL PRIMARIO (KG/M3)
TIEMPO (S) *10'

FIG. III (A) - 21

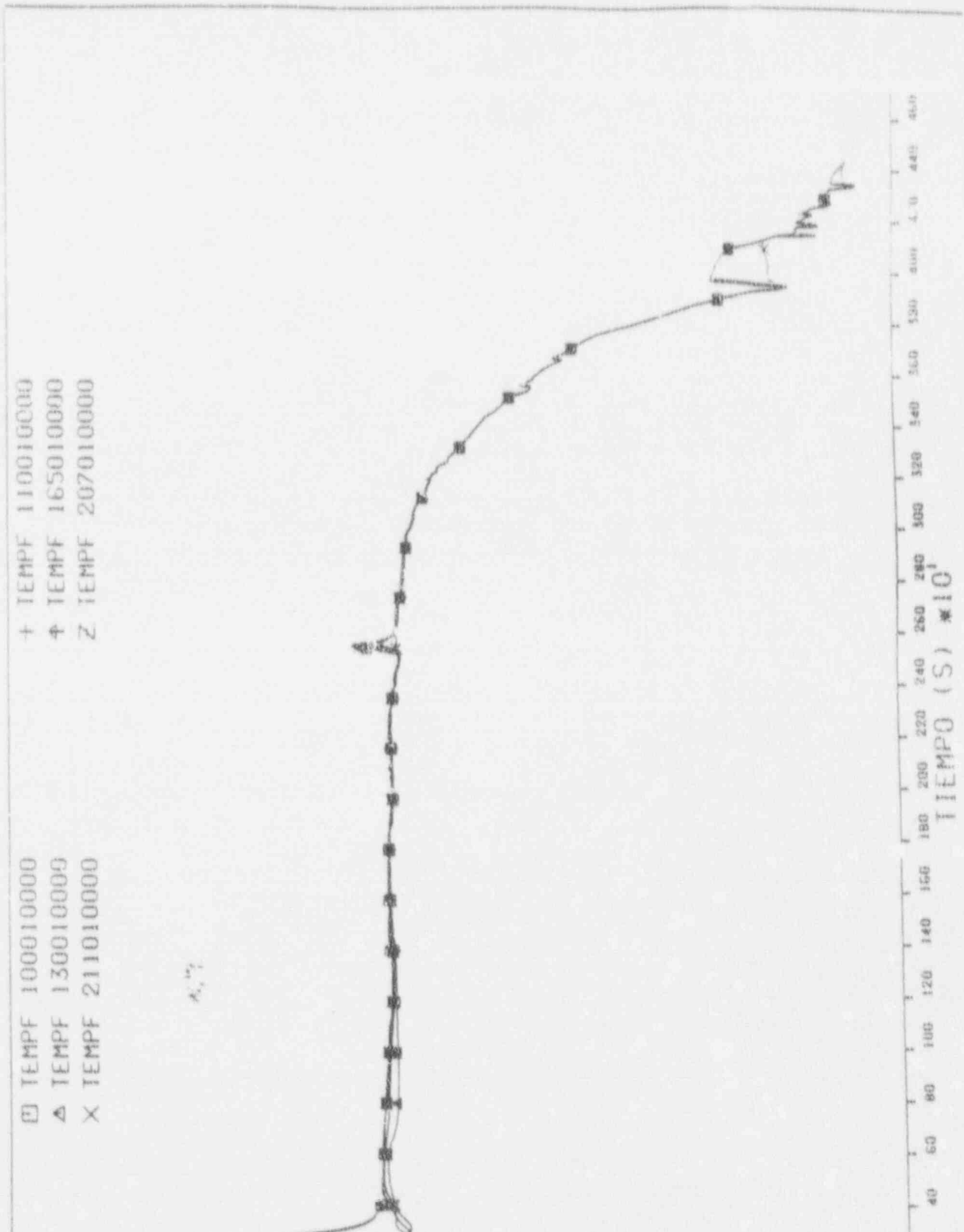

TEMPF 100010000
TEMPF 130010000
TEMPF 211010000
TEMPF 110010000
TEMPF 165010000
TEMPF 207010000
TEMPERATURAS EN PRIMARIO (K)
TIEMPO (S) *10¹

FIG. III (A) - 22

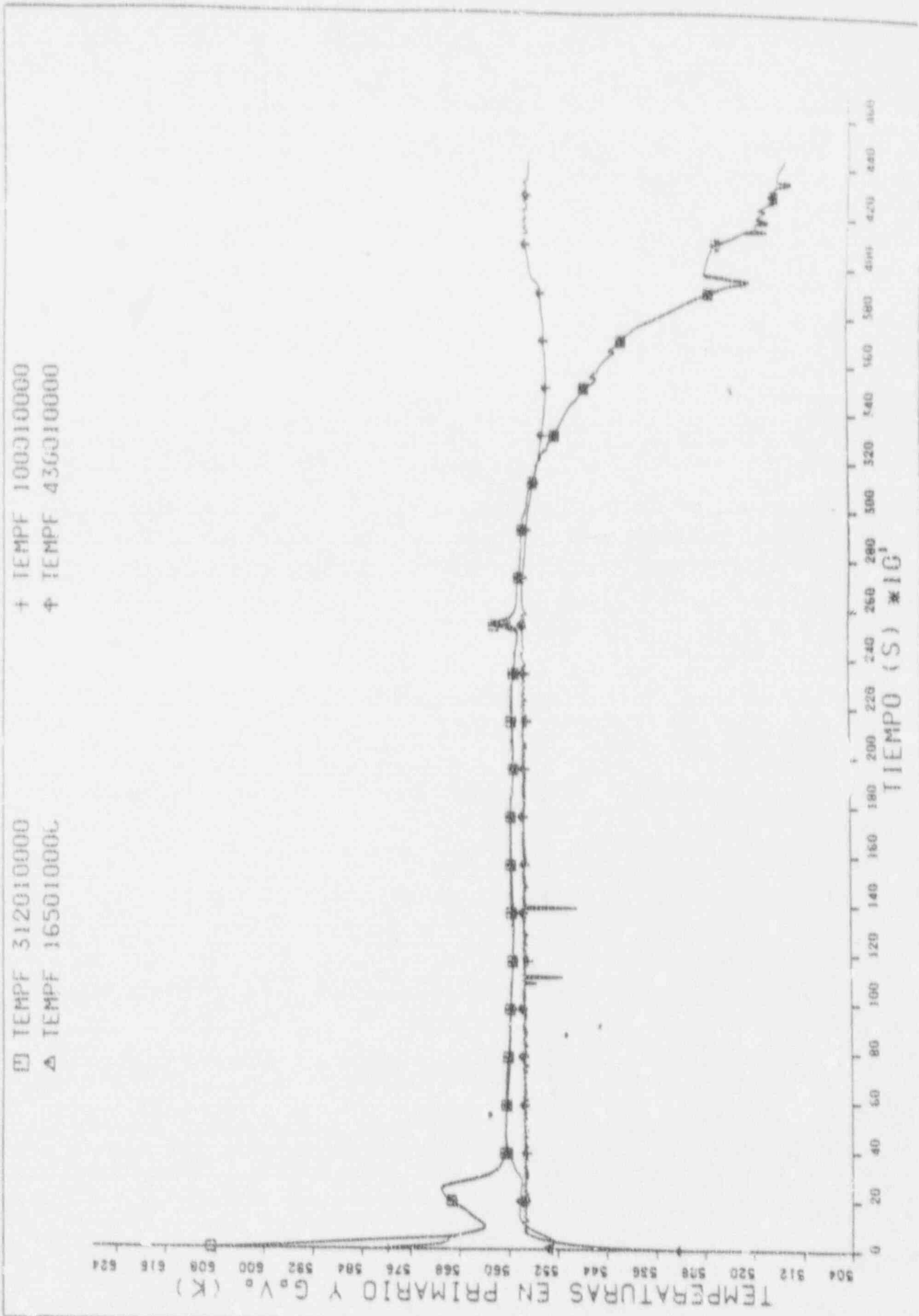

TEMPF 312010000
TEMPF 165010000
TEMPF 100010000
TEMPF 436010000
TIEMPO (S) *10¹
TEMPERATURAS EN PRIMARIO Y G.V. (K)

FIG. III (A) - 23

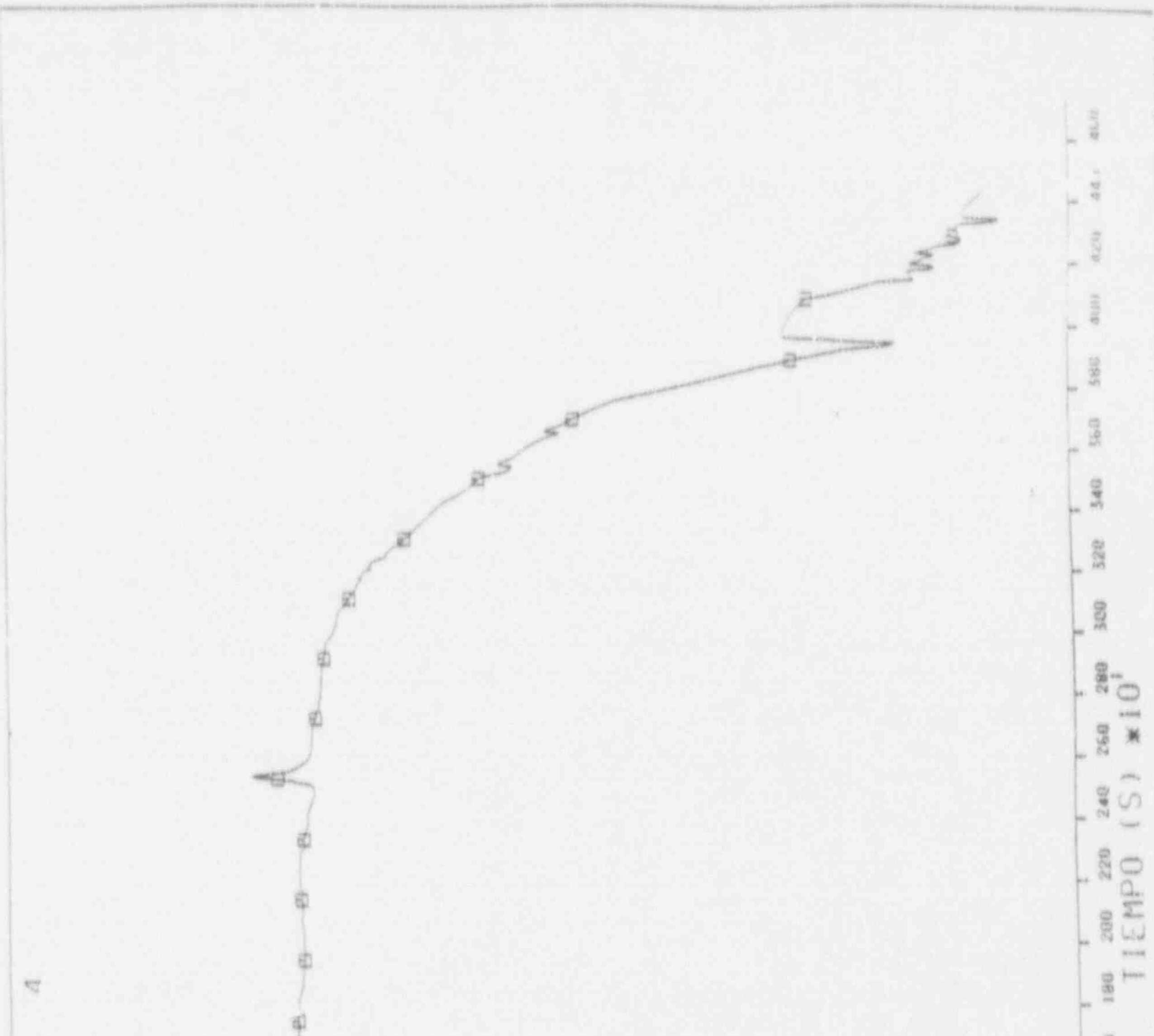

CNTRLVAR 4
TEMPERATURA MEDIA DEL PRIMARIO (C)
TIEMPO (S) *10¹
ROTURA SB3 (1.5 INCH) SIN RECUPERACION (FIG. 24)

FIG. III (A) - 24

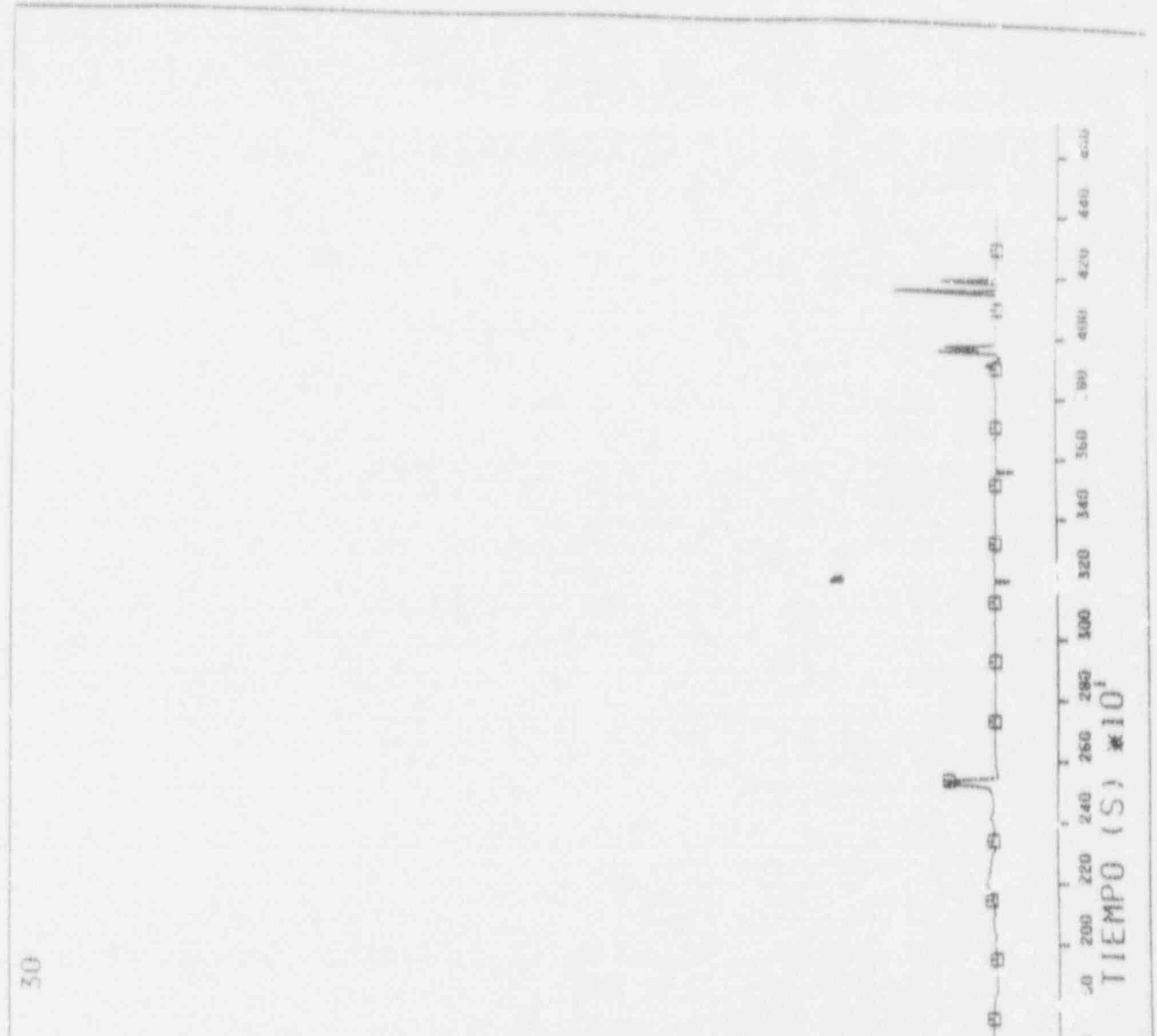
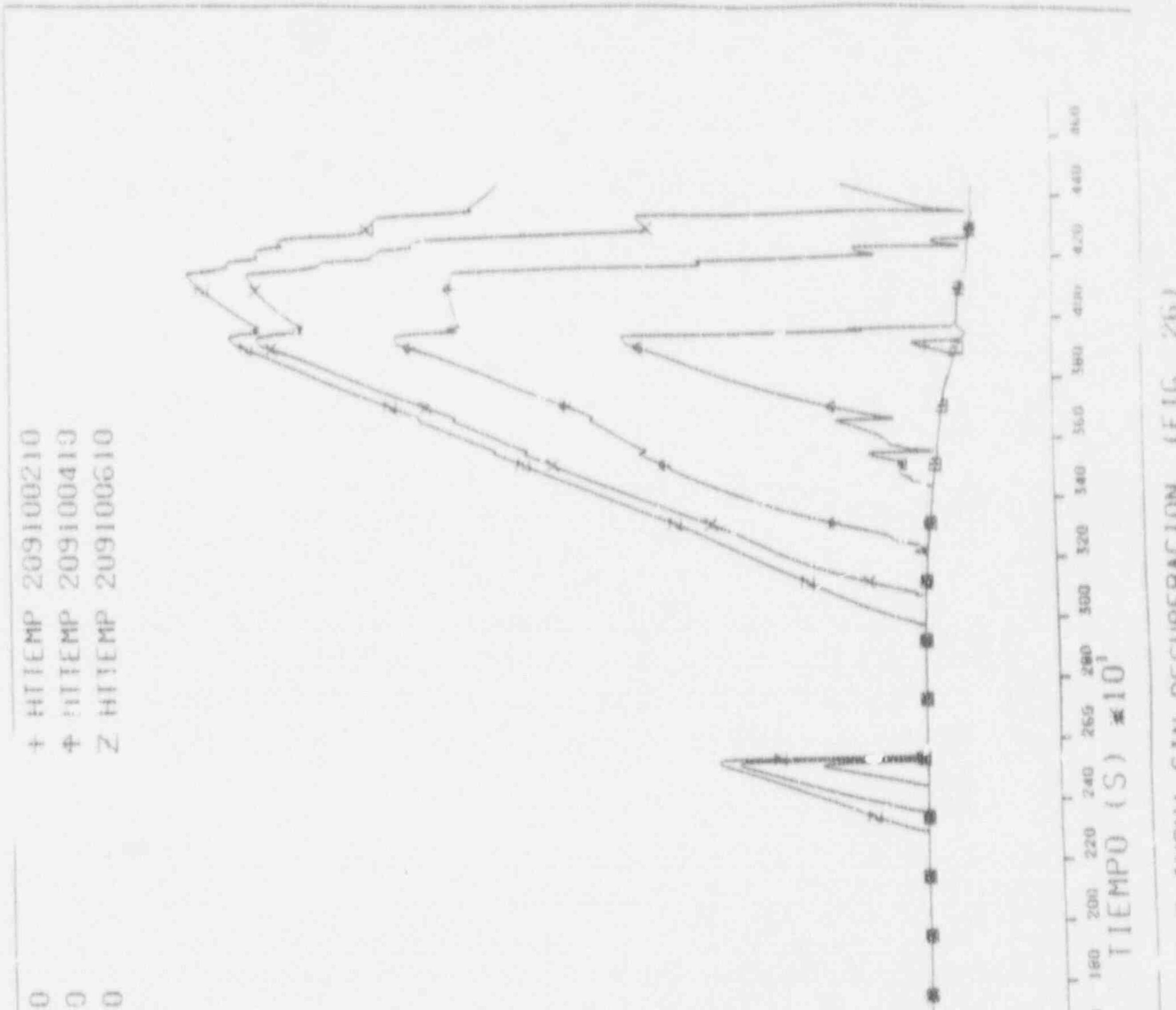


FIG. III (A) - 25



HTTEMP 209100110
HTTEMP 209100310
HTTEMP 209100510
HTTEMP 209100210
HTTEMP 209100410
HTTEMP 209100610
TEMPERATURA VAINAS CANAL MEDIO (K)
TIEMPO (S) *10¹
SIN RECUPERACION (FIG 26)

FIG. III (A) - 26

APPENDIX IV : FIGURES OF CASE I(B), (6" WITH RECOVERY)

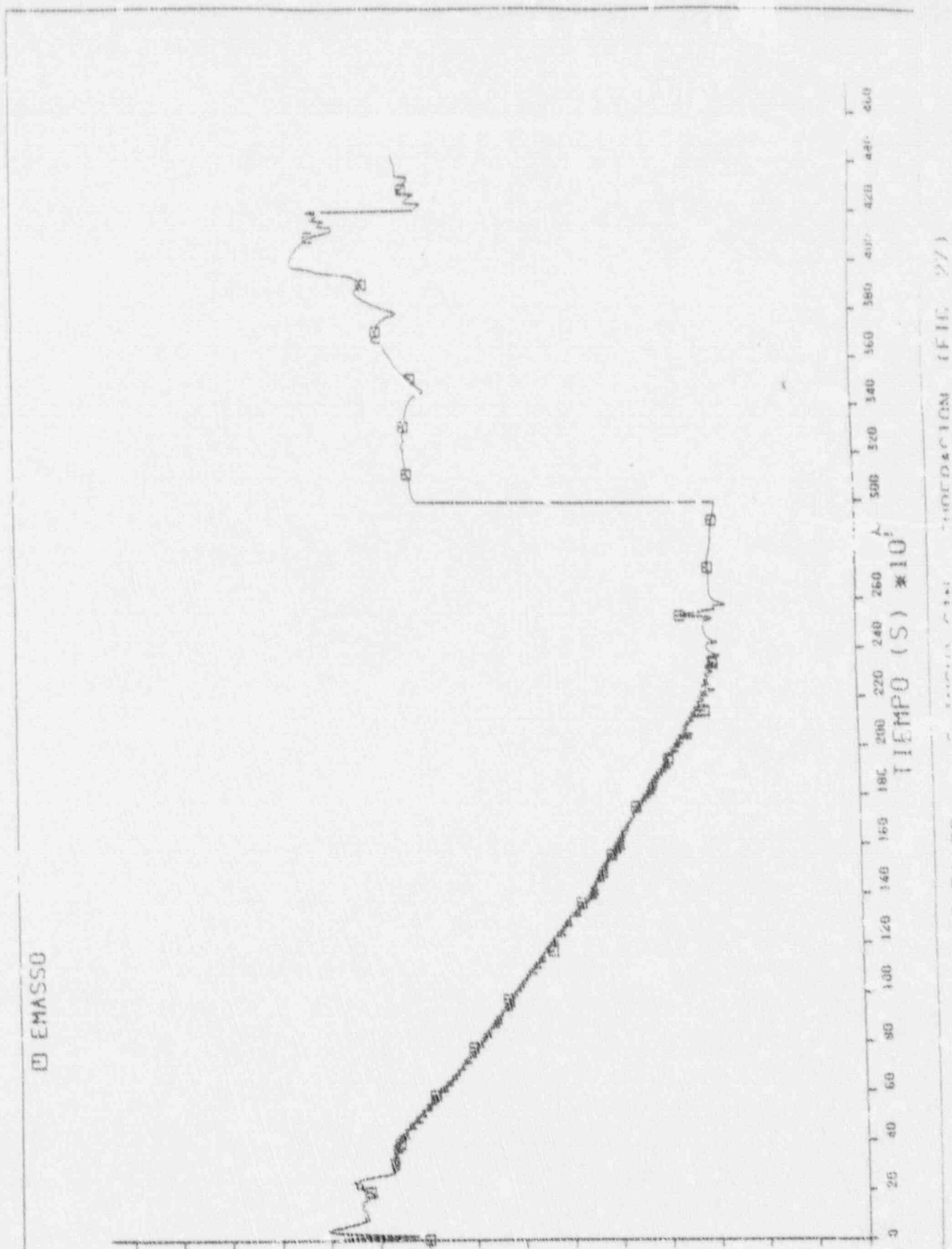


FIG. III (A) - 27

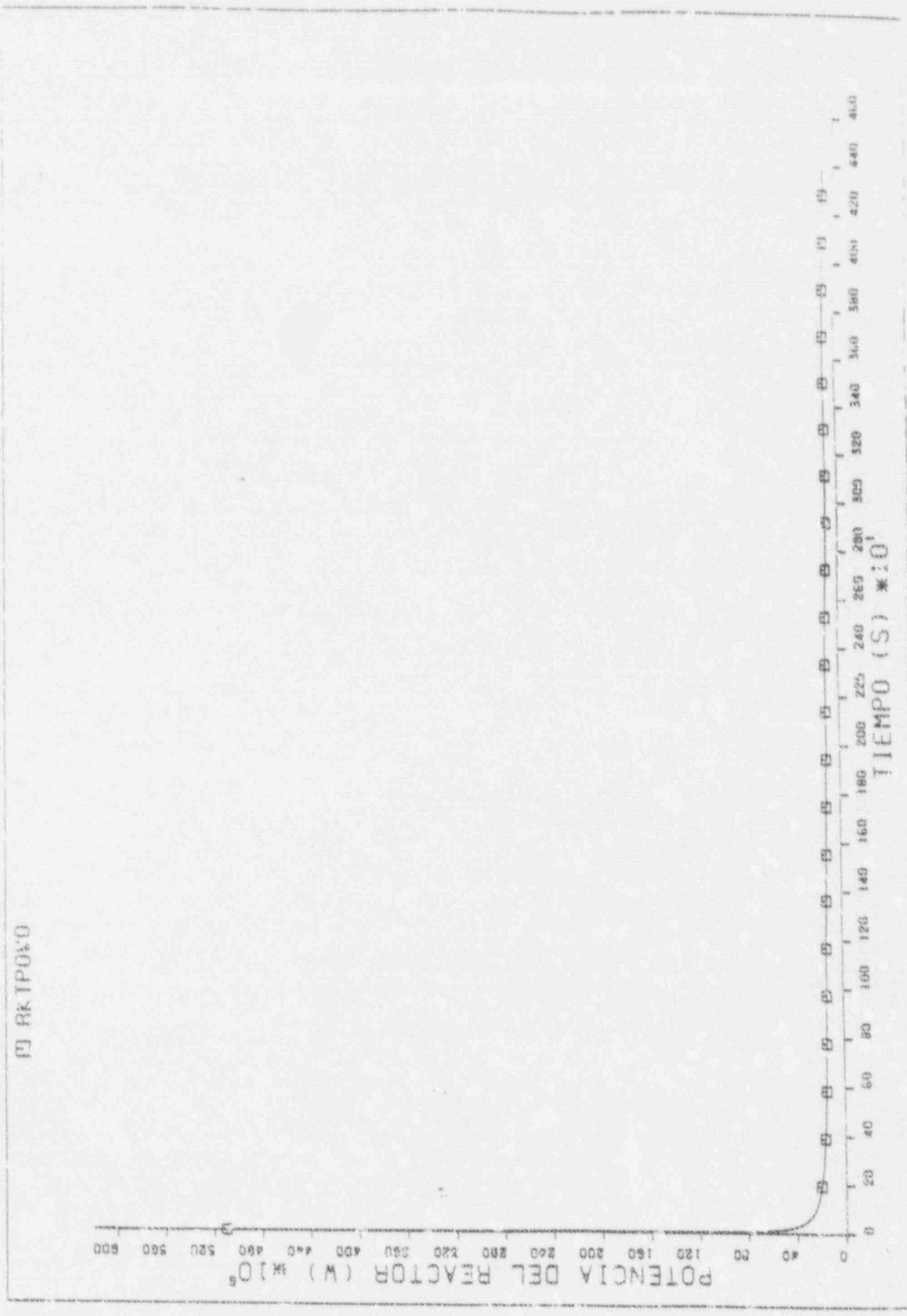
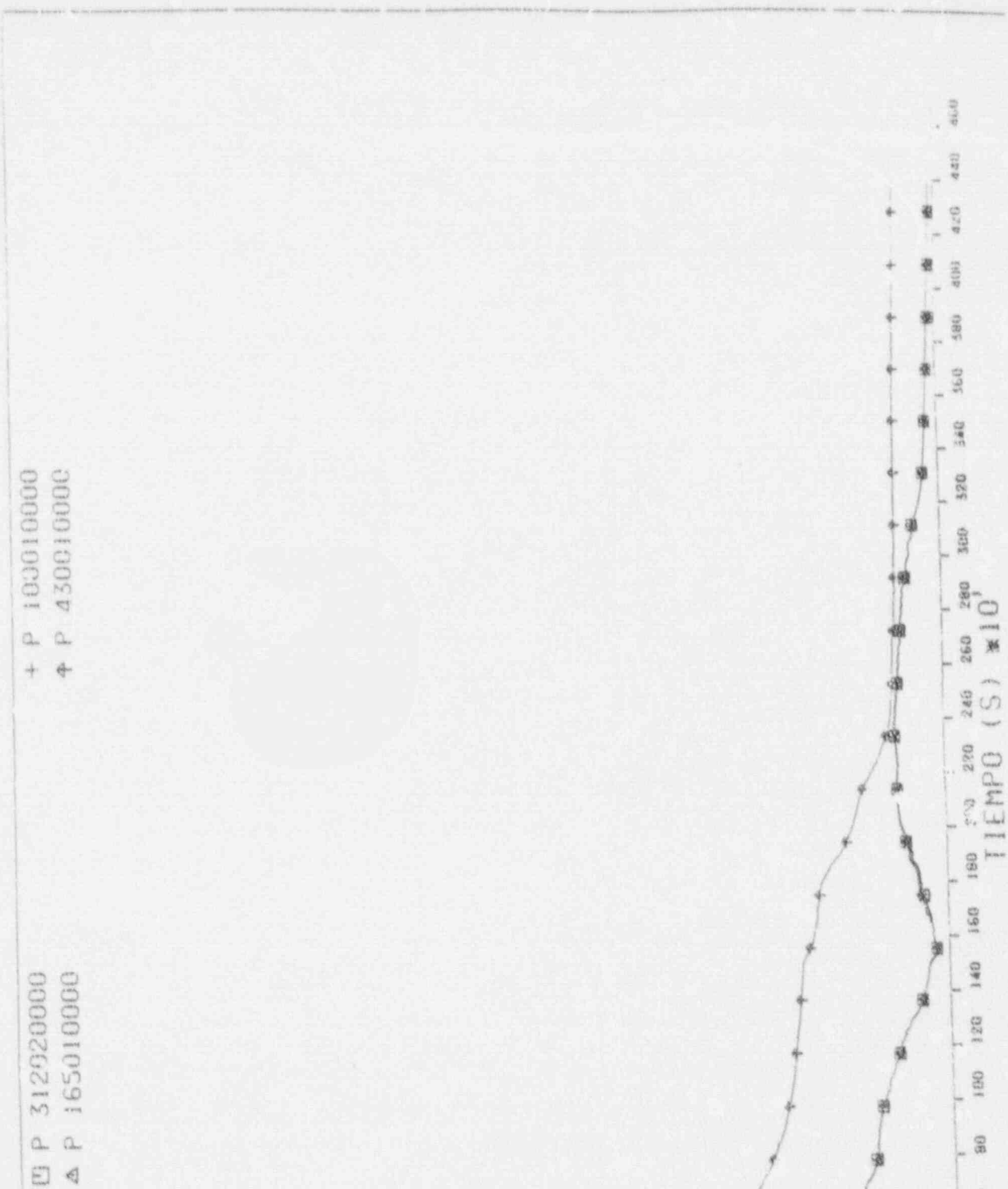


FIG. I (B) - 1



P 312020000
P 165010000
P 100010000
P 430010000
PRESION (PA) *10^5
TIEMPO (S) *10^1

FIG. I (B) - 2

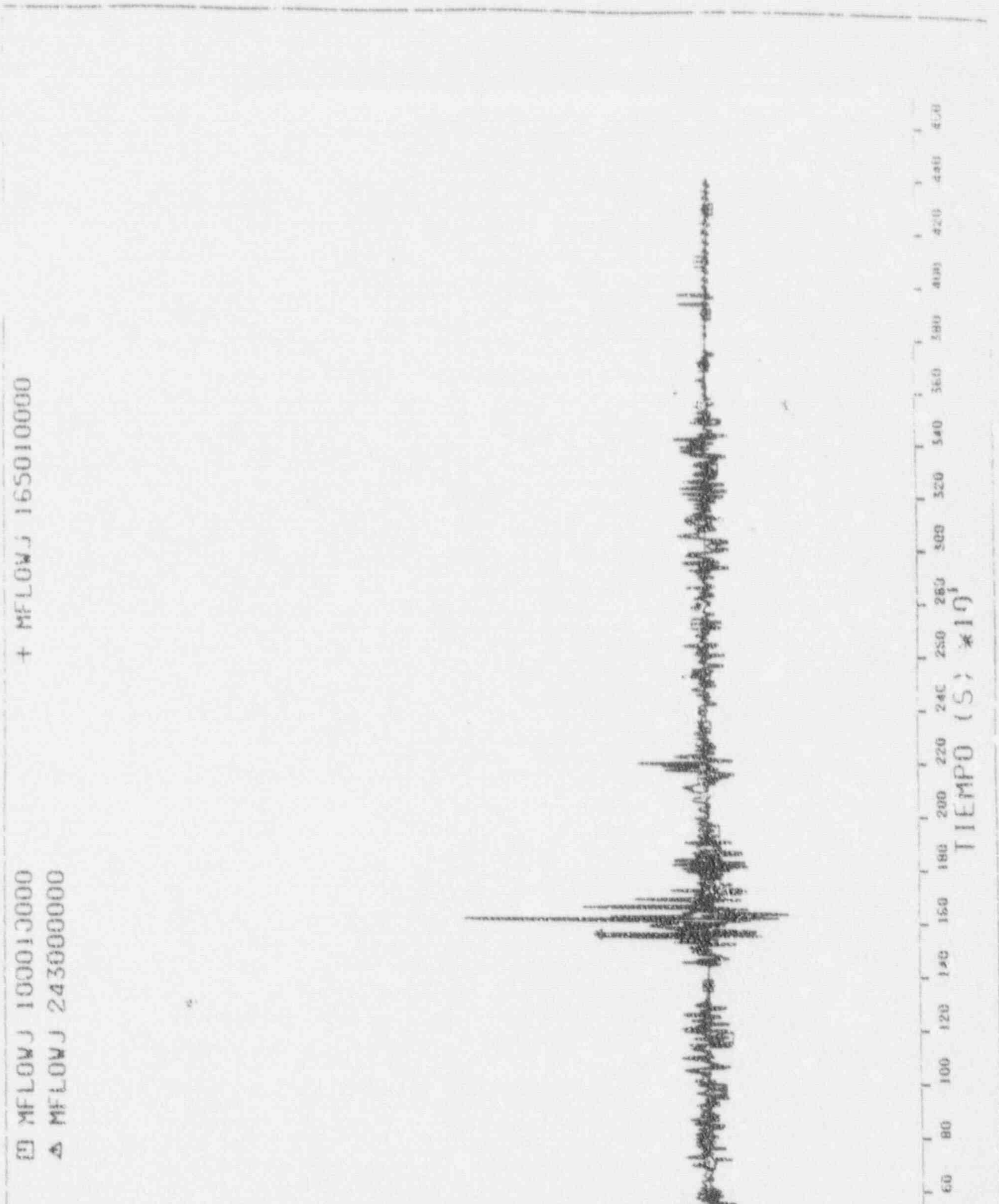

MFLOWJ 100010000
MFLOWJ 243000000
+ MFLOWJ 165010000
CAUDAL DEL PRIMARIO (KG/S) *10
TIEMPO (S) *10

FIG. I (B) - 3

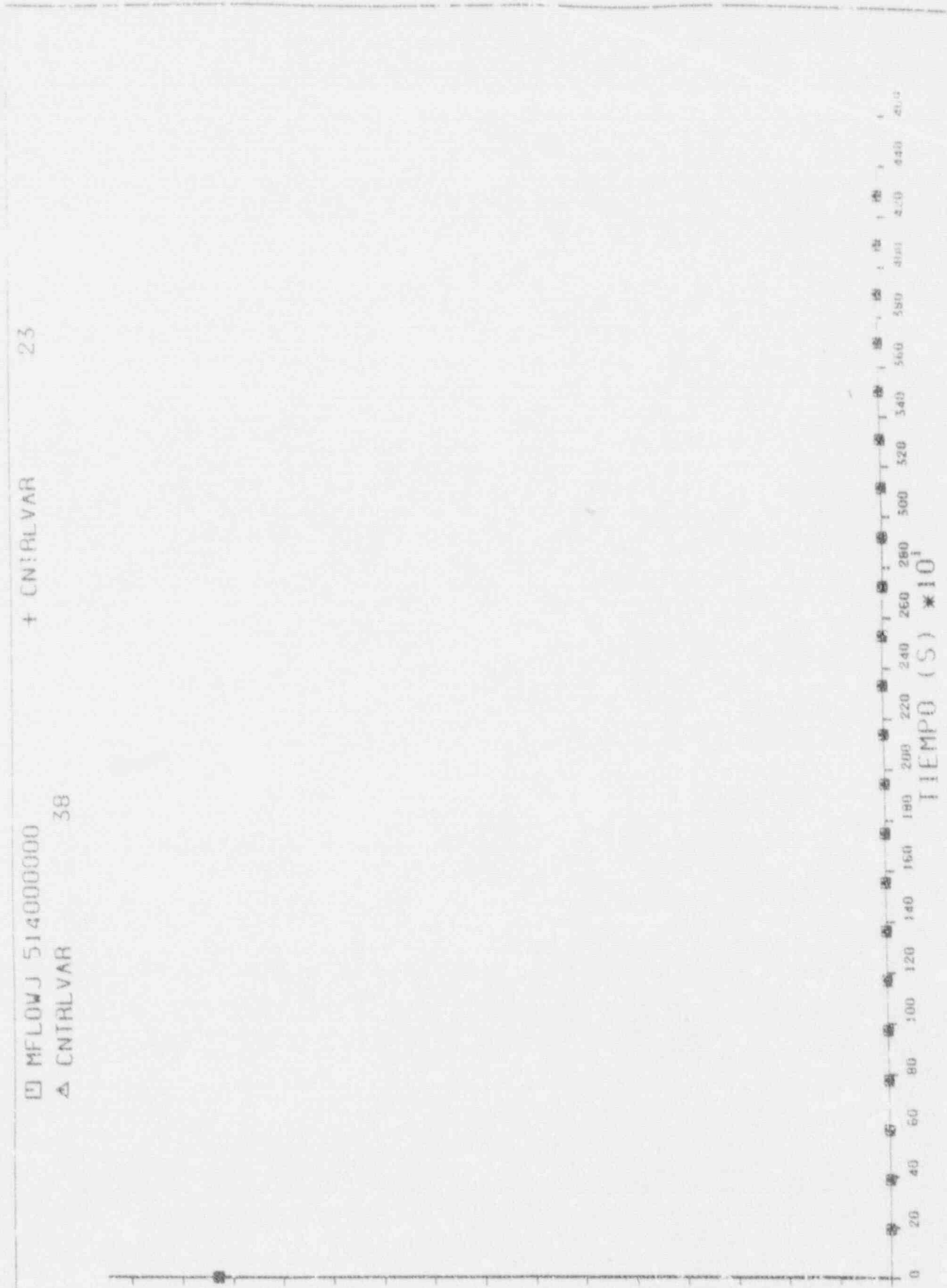

MFLOWJ 514000000
CNTRLVAR 38
+ CNTRLVAR 23
TIEMPO (S) *10^1
CAUDALES DE SECUNDARIO (KG/S)

FIG. I (B) - 4

⊡ MFLOWJ 536000000
△ MFLOWJ 526000000
× CNTRLVAR 23
+ MFLOWJ 522000000
◇ CNTRLVAR 22
ALIVIO Y SEGURIDAD G.V. (KG/S)
TIEMPO (S) ×10¹

FIG. I (B) - 3

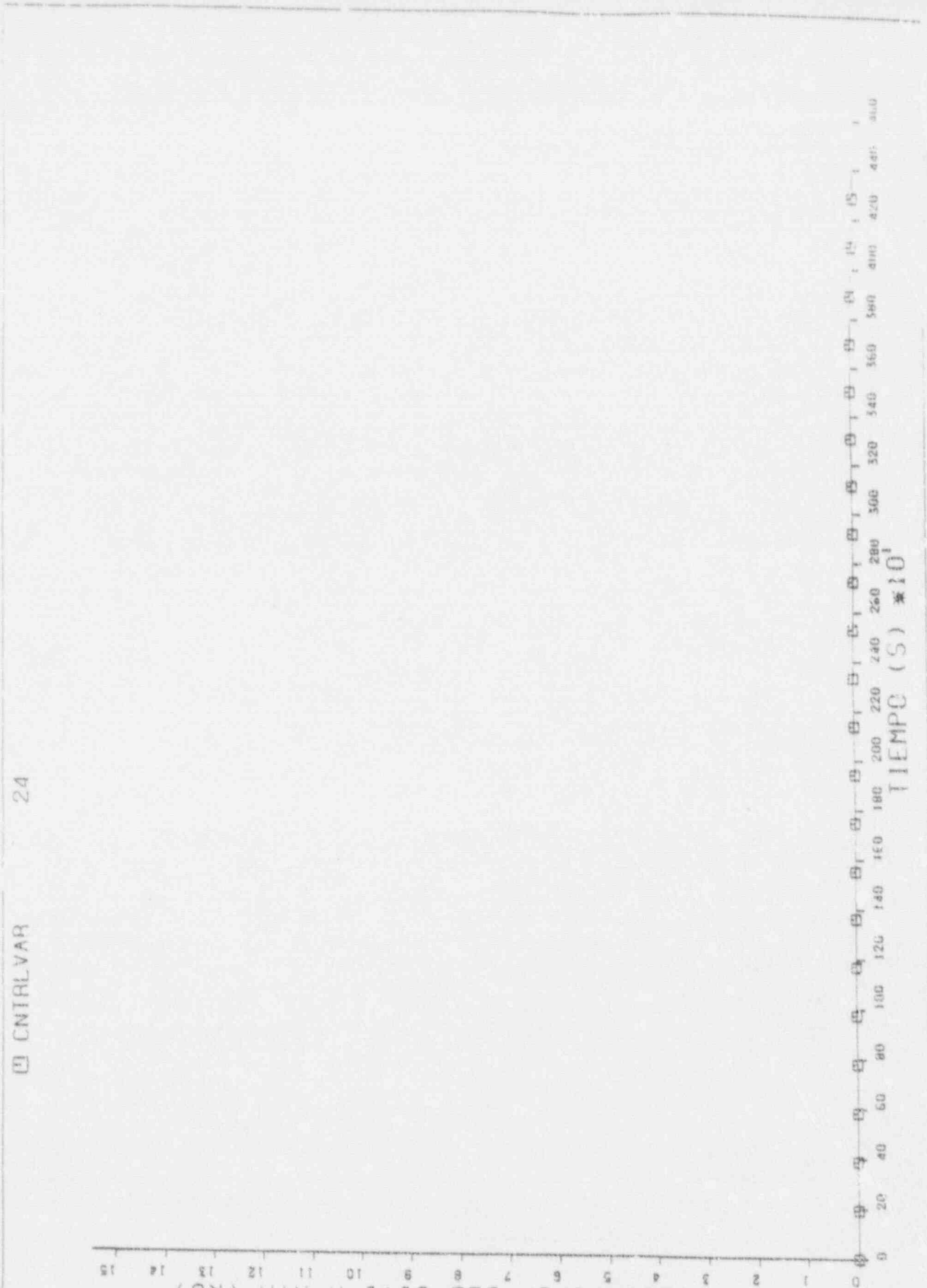


FIG. I (B) - 6

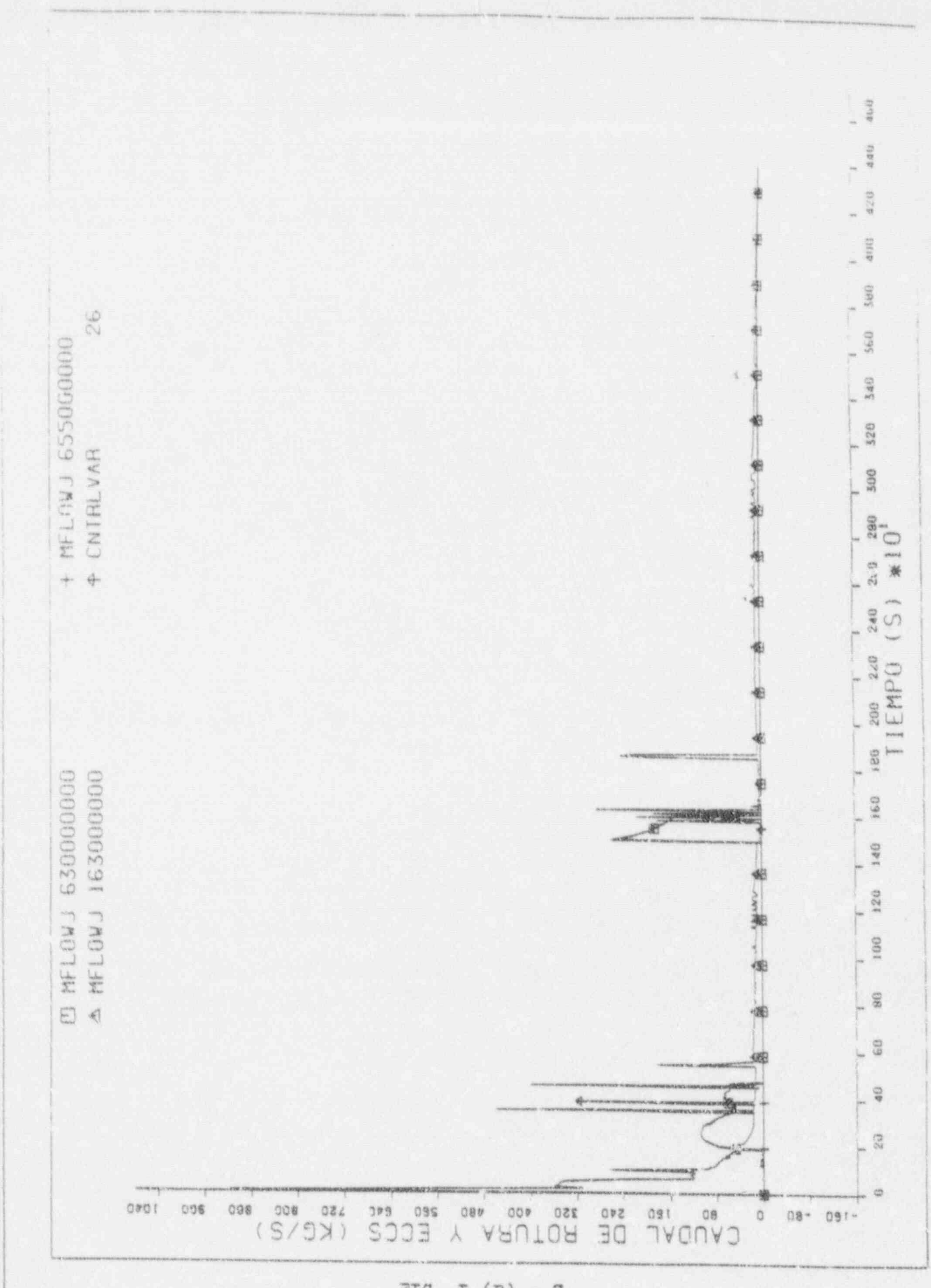

MFLOWJ 630000000
MFLOWJ 163000000
MFLOWJ 655000000
CNTRLVAR 26
TIEMPO (S) *10^1
CAUDAL DE ROTURA Y ECCS (KG/S)

FIG. I (B) - 7

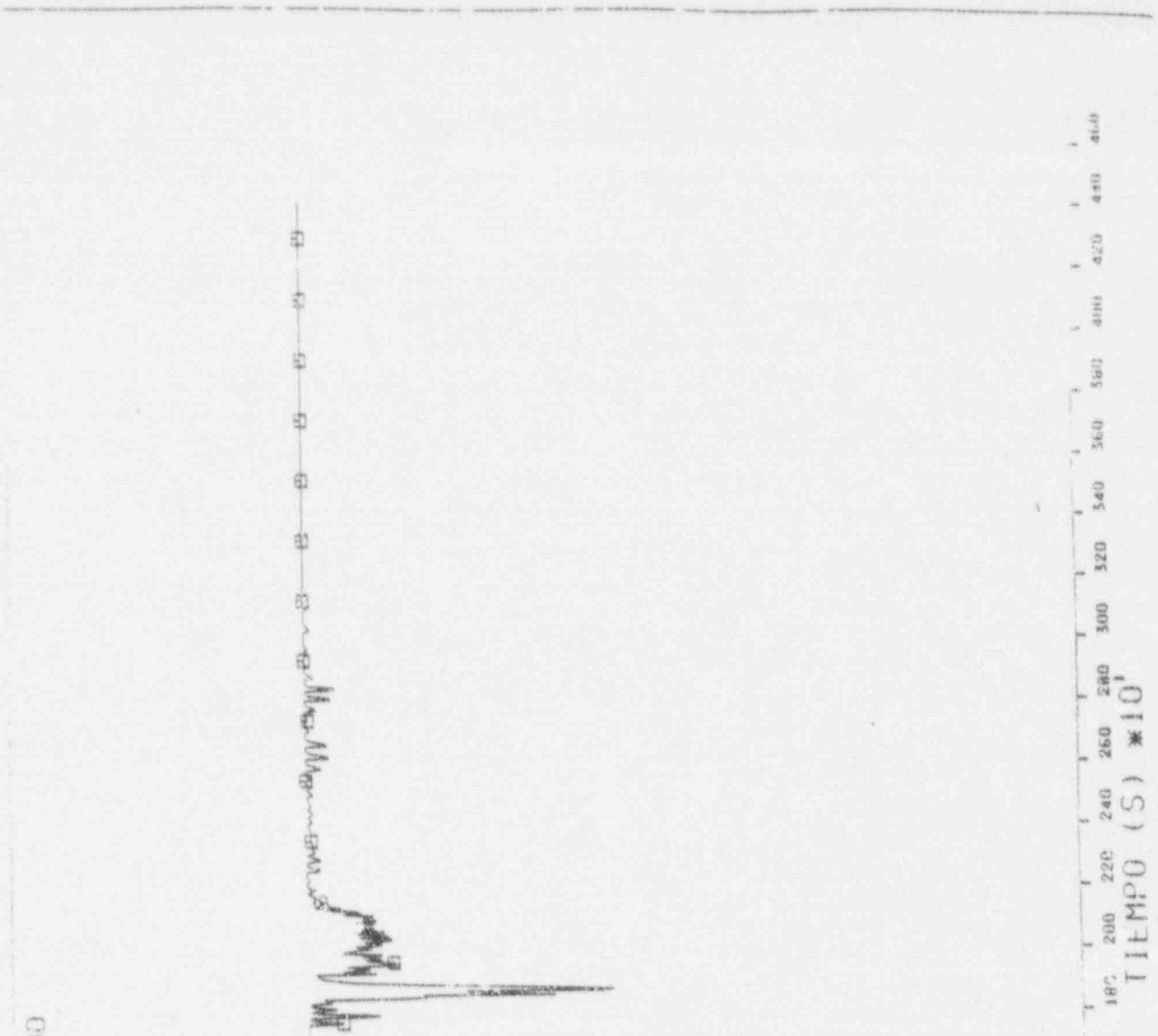

VOIDGJ 163000000
TIEMPO (S) *10'
FRACCION DE VAPOR EN LA ROTURA (O/1)
ROTURA SB3 (6 INCH) CON RECUPERACION

FIG. I (B) - 8

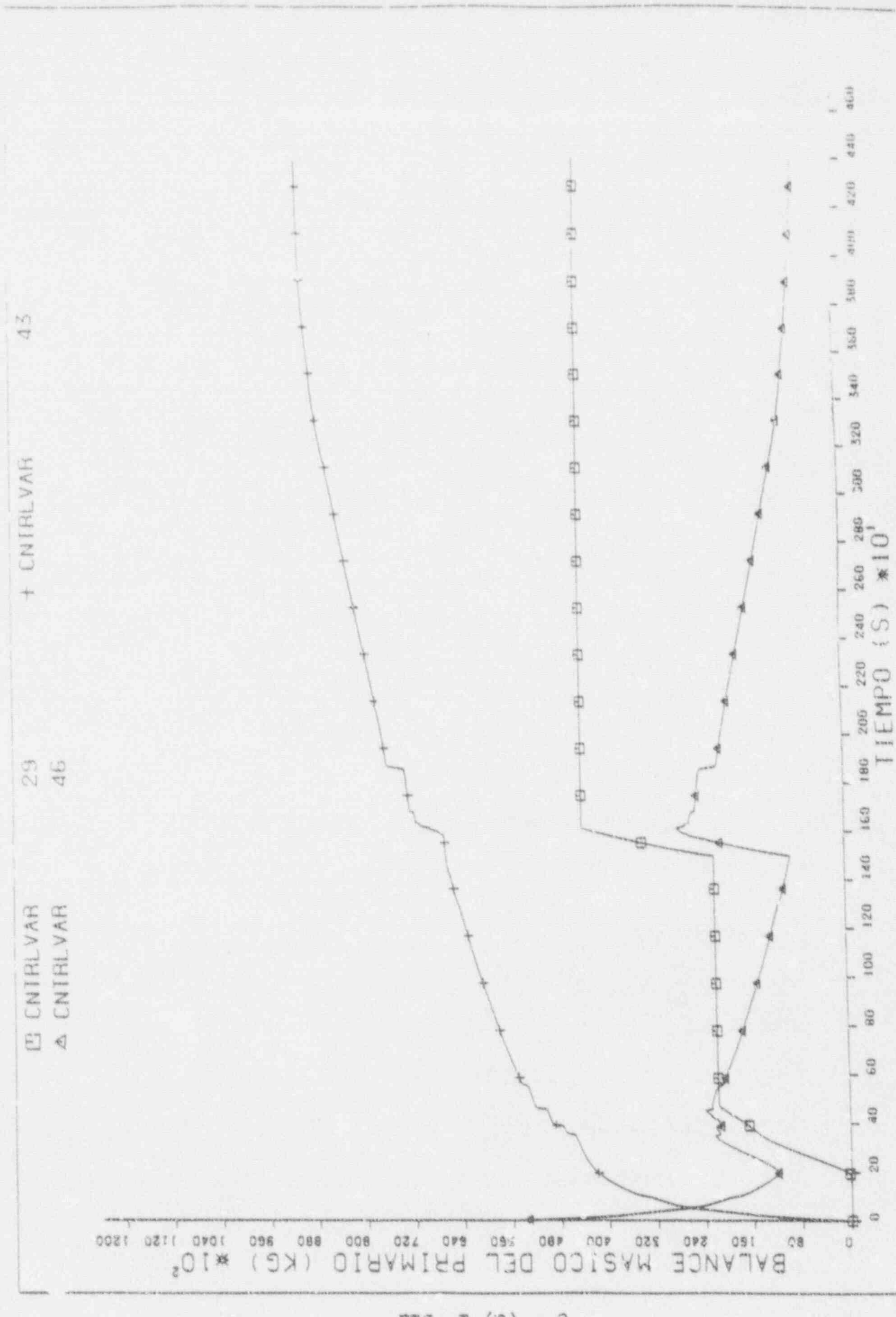
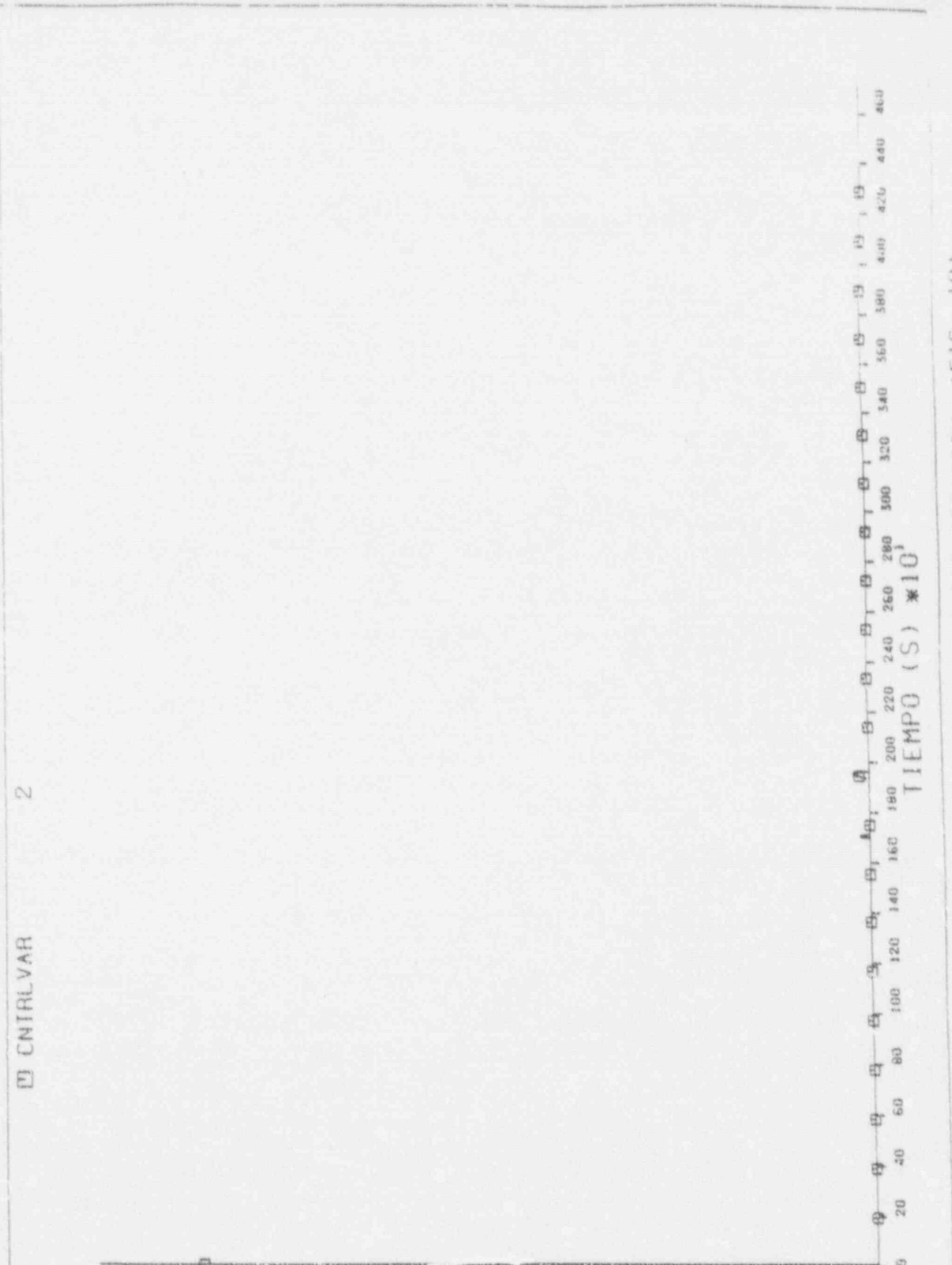
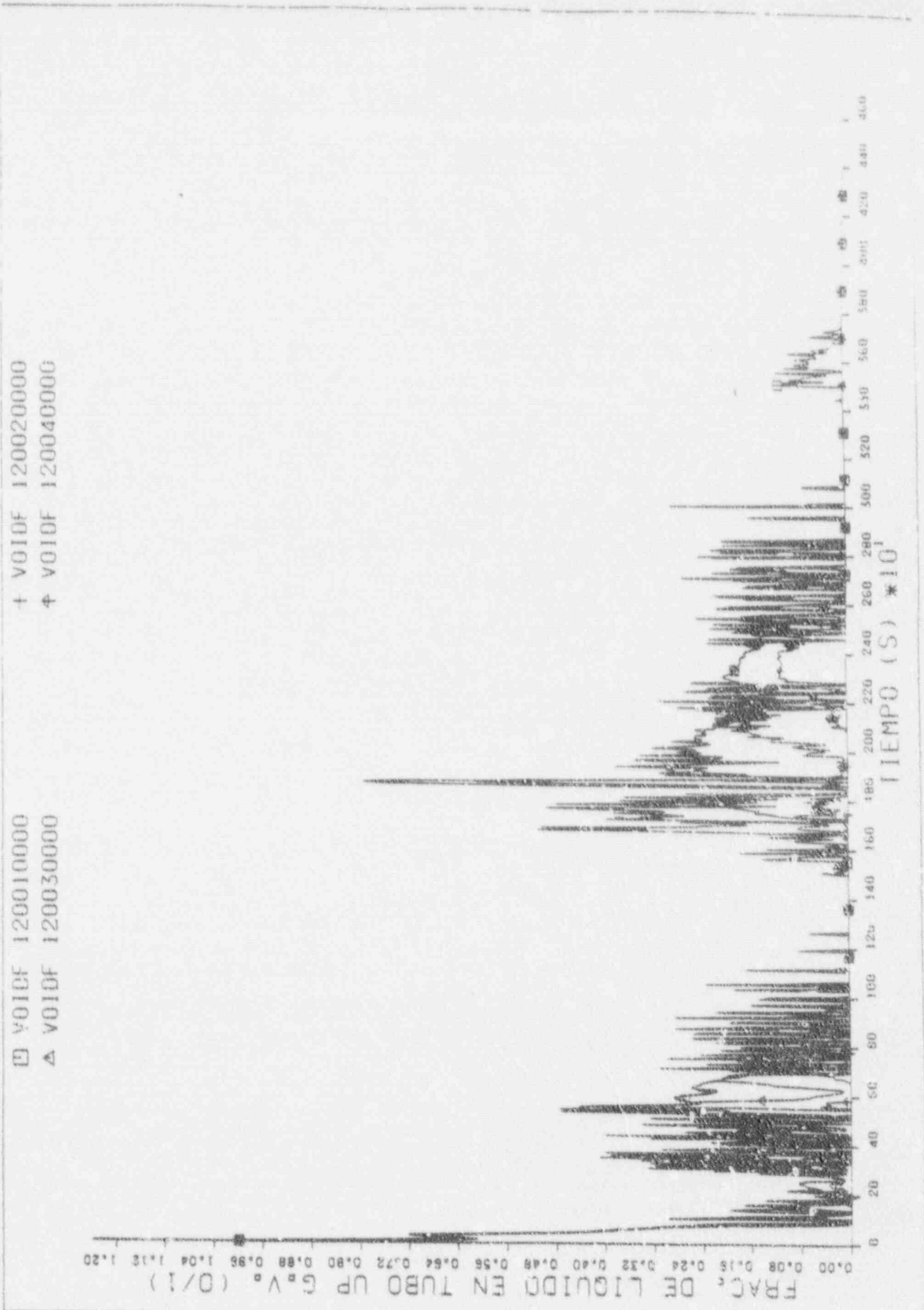

CNTRLVAR 29
CNTRLVAR 46
+ CNTRLVAR 43
TIEMPO (S) *10^1
BALANCE MASICO DEL PRIMARIO (KG) *10^2

FIG. I (3) - 9



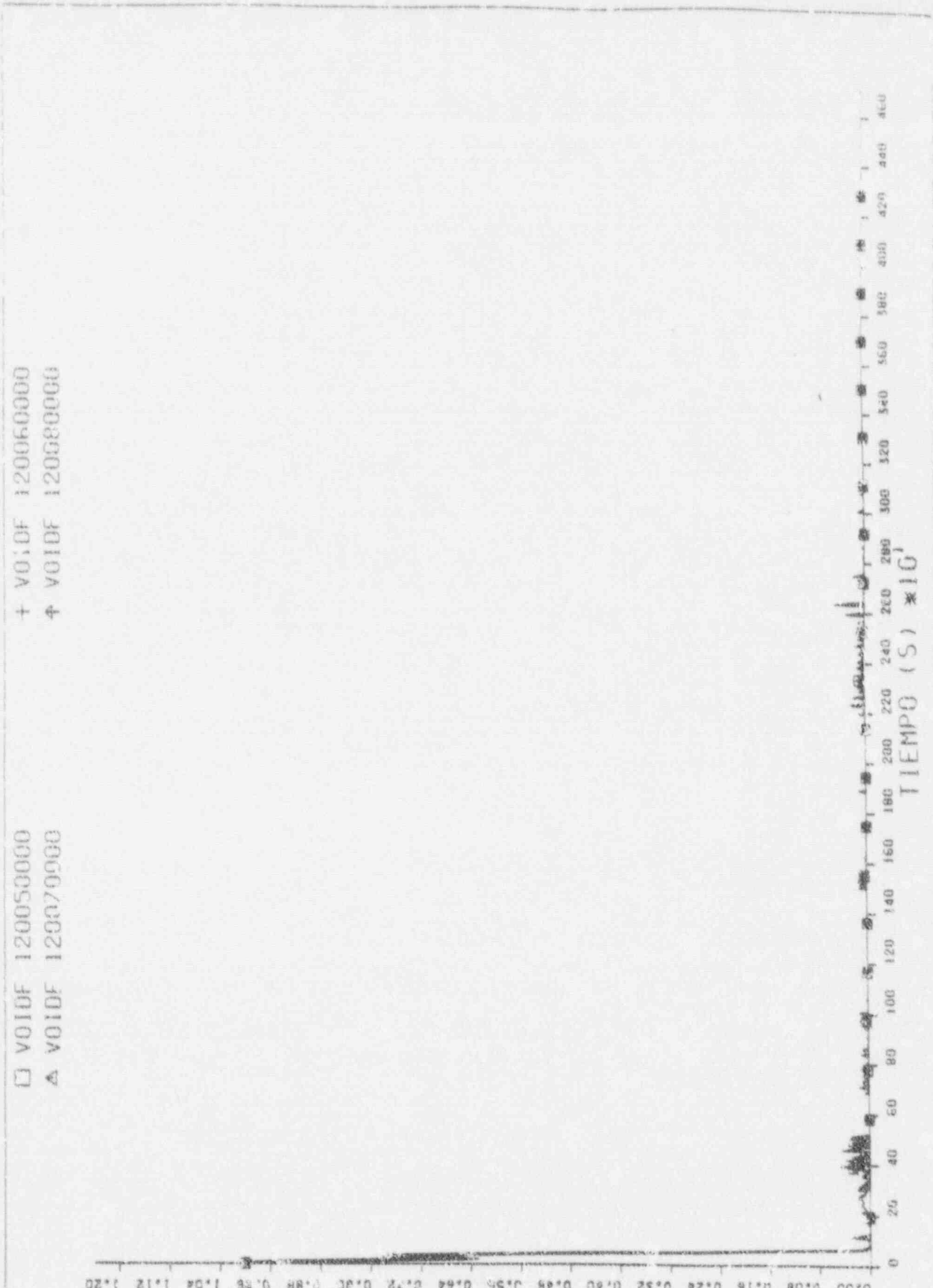
CNTRLVAR 2
NIVEL COLAPSADO EN PRESIONADOR (%)
0 5 10 15 20 25 30 35 40 45 50 55 60 65 70 75
TIEMPO (S) *10¹
0 20 40 60 80 100 120 140 160 180 200 220 240 260 280 300 320 340 360 380 400 420 440 460
ROTURA SB3 (6 INCH) CON RECUPERACION (FIG 10)

FIG. I (B) - 10



VOIDF 120010000
VOIDF 120020000
VOIDF 120030000
VOIDF 120040000
TIEMPO (S) *10¹
FRAC. DE LIQUIDO EN TUBO UP G.V. (O/1)

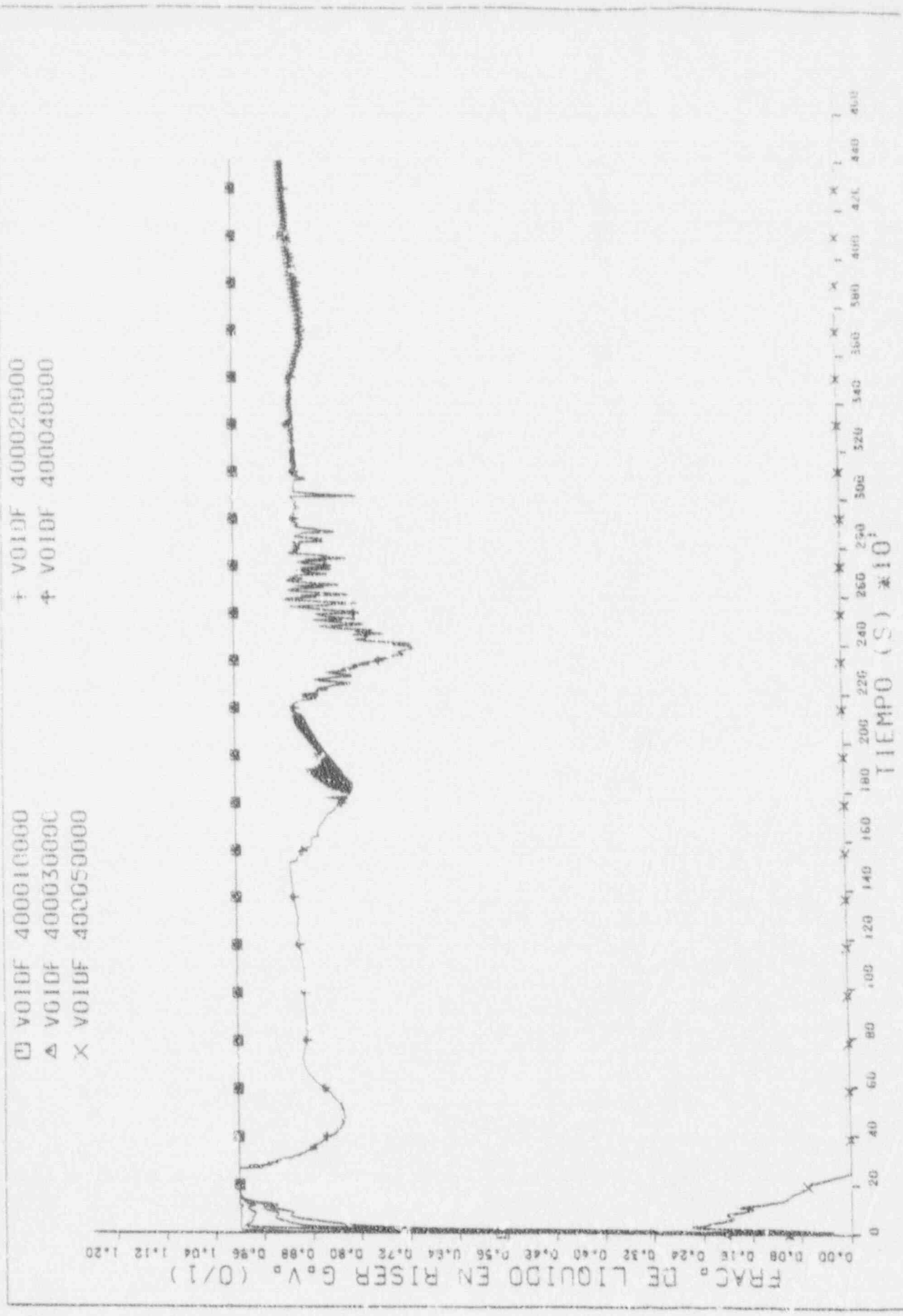
FIG. I (B) - 11



VOIDF 120050000
VOIDF 120060000
VOIDF 120070000
VOIDF 120080000
TIEMPO (S) *10'
FRAC. DE LIQUIDO EN TUBO DOWN G.V. (0/1)
ROTURA SB3 (6 INCH) CON RECUPERACION (FIG 12)

FIG. X (B) - 12

FIG. I (B) - 13



VOIDF 400010000
VOIDF 400030000
VOIDF 400050000
VOIDF 400020000
VOIDF 400040000
FRAC. DE LIQUIDO EN RISER G.V. (0/1)
TIEMPO (S) *10¹

FIG. I (B) - 14

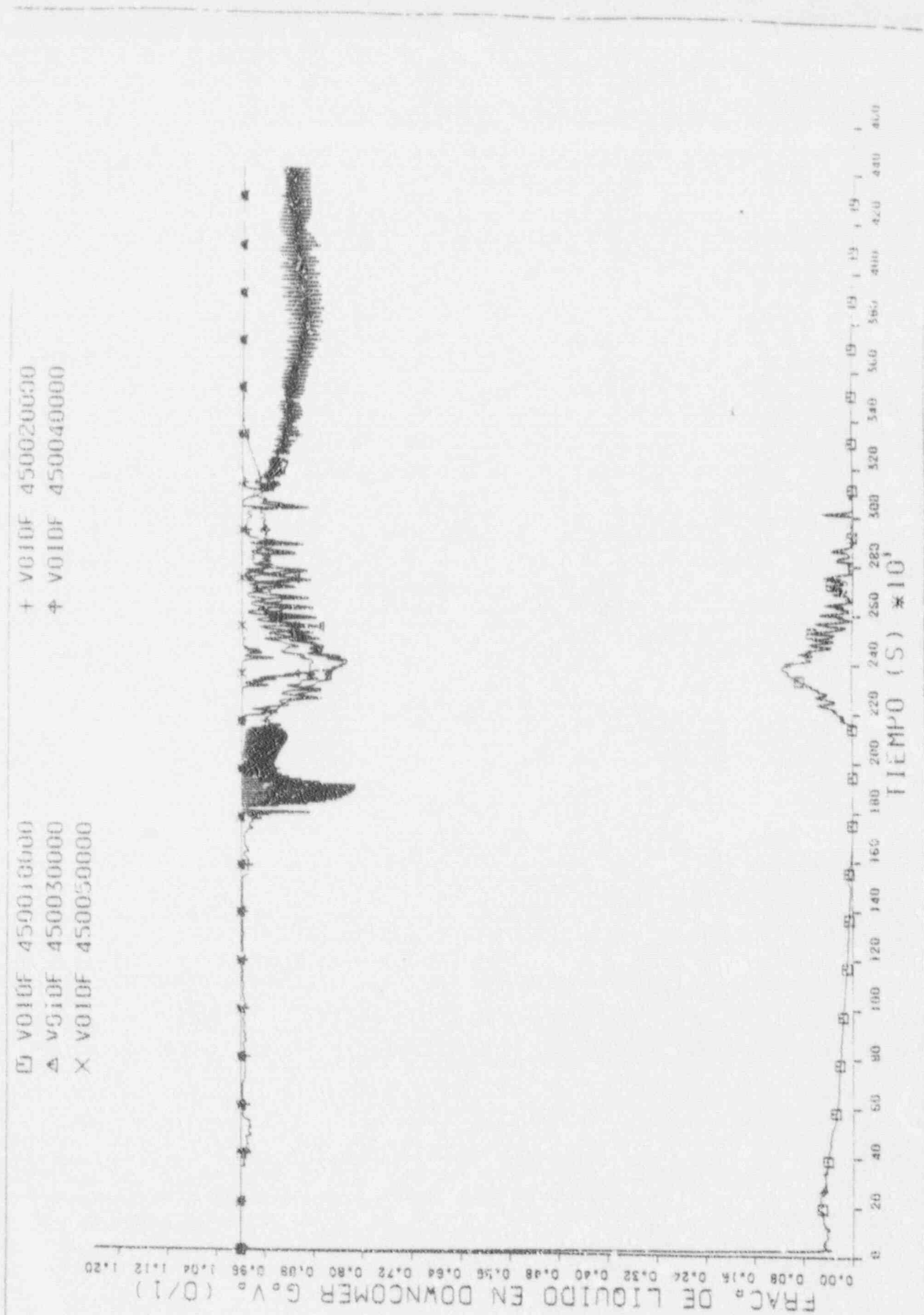

VOIDF 450010000
VOIDF 450030000
VOIDF 450050000
VOIDF 450020000
VOIDF 450040000
FRAC. DE LIQUIDO EN DOWNCOMER G.V. (0/1)
TIEMPO (S) *10'

FIG. T (B) - 15

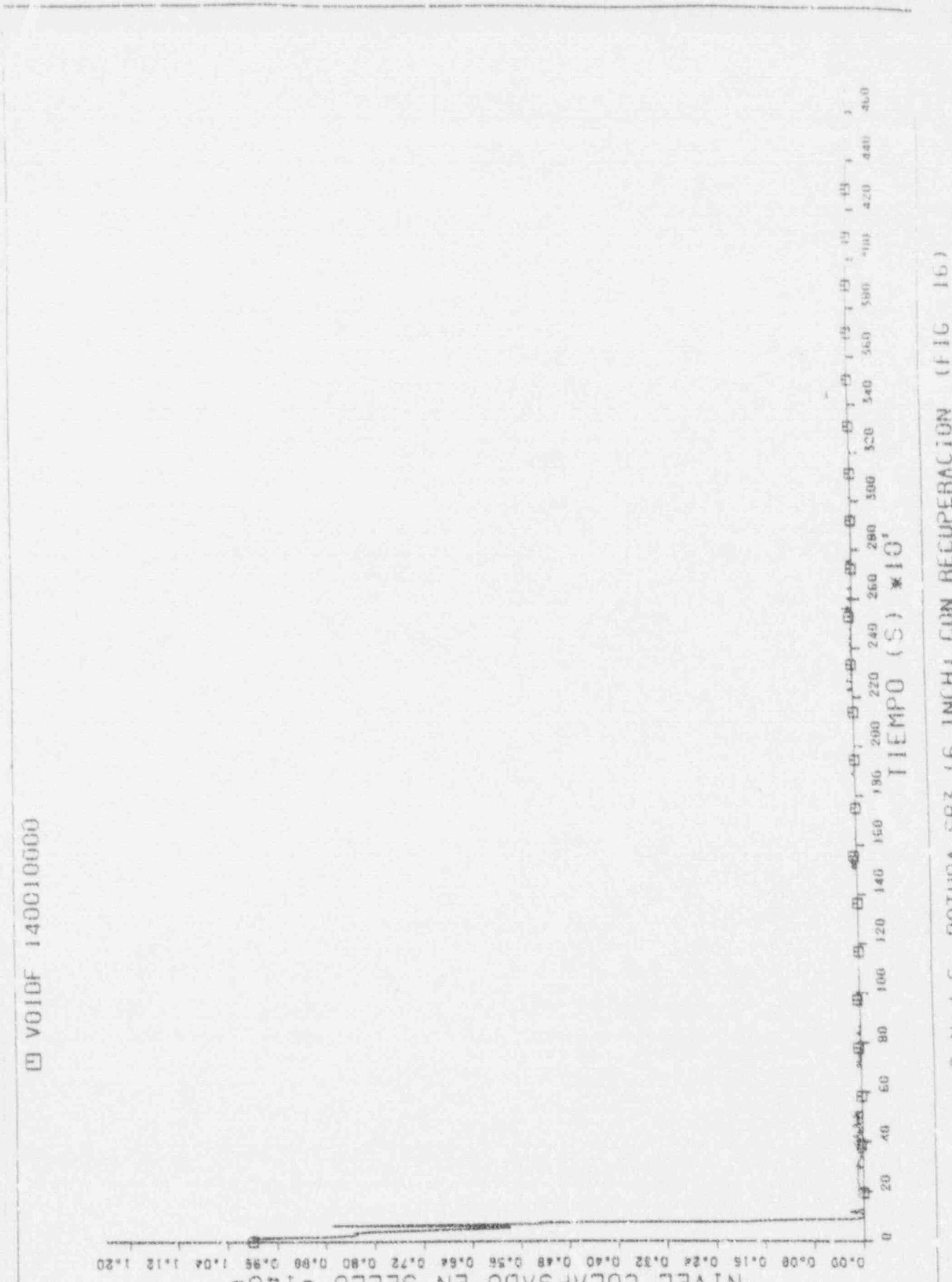

VOIDF 140010000
NIVEL COLAPSADO EN SELLO -140-
TIEMPO (S) *10'
ROTURA SB3 (6 INCH) CON RECUPERACION (FIG 16)

FIG. I (B) - 16

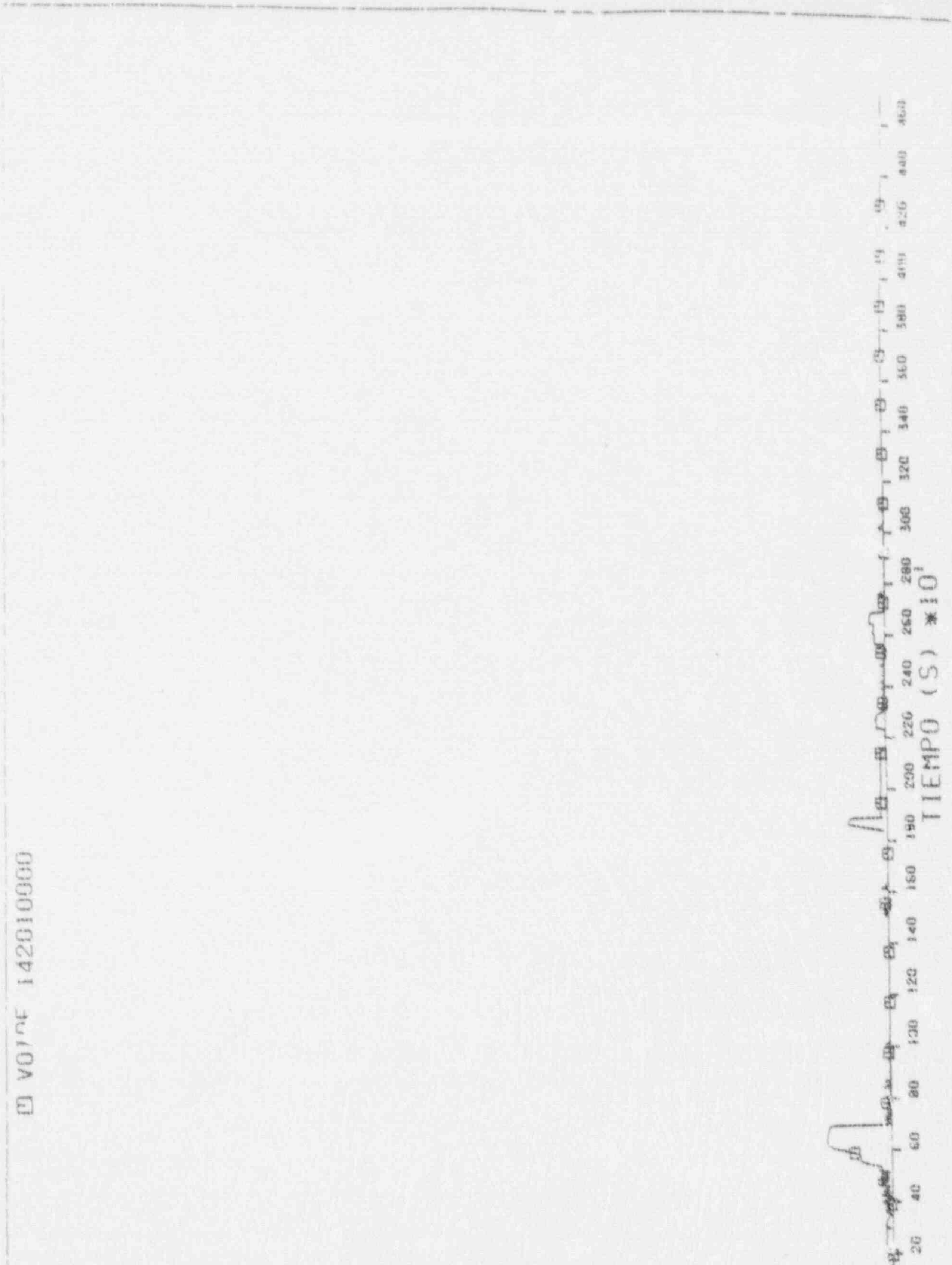


FIG. I (B) - 17

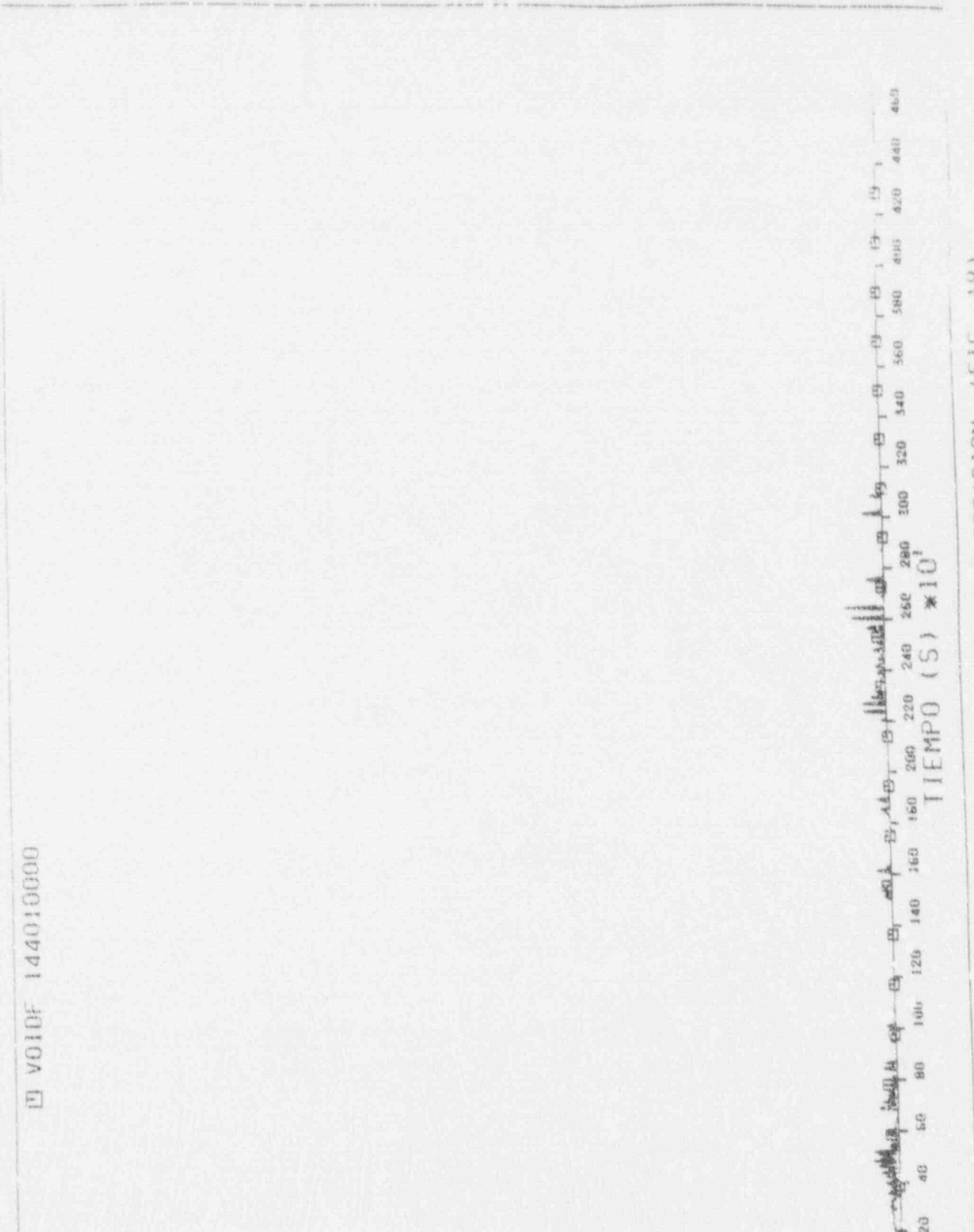


FIG. I (F) - 18

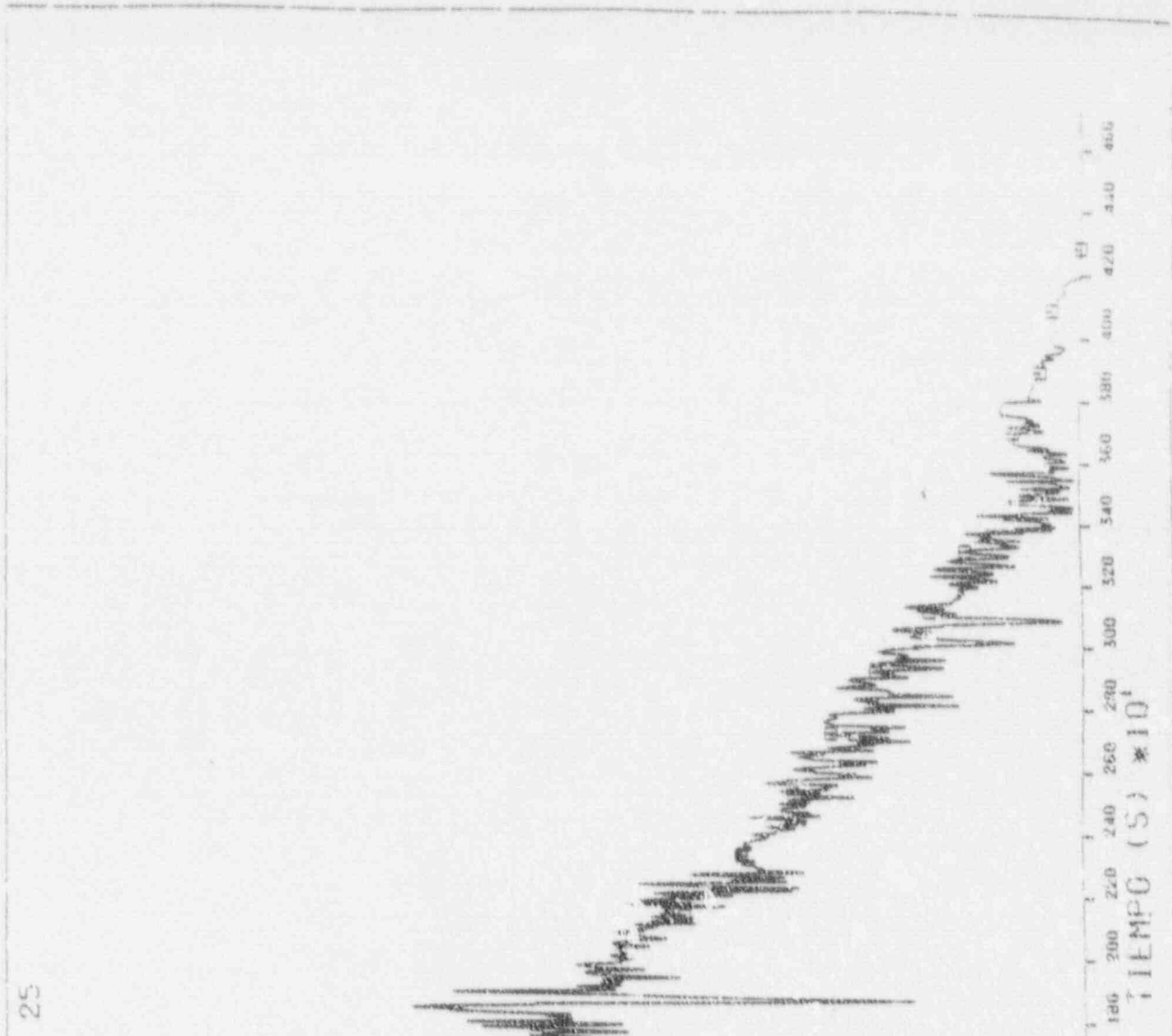


FIG. I (3) - 19

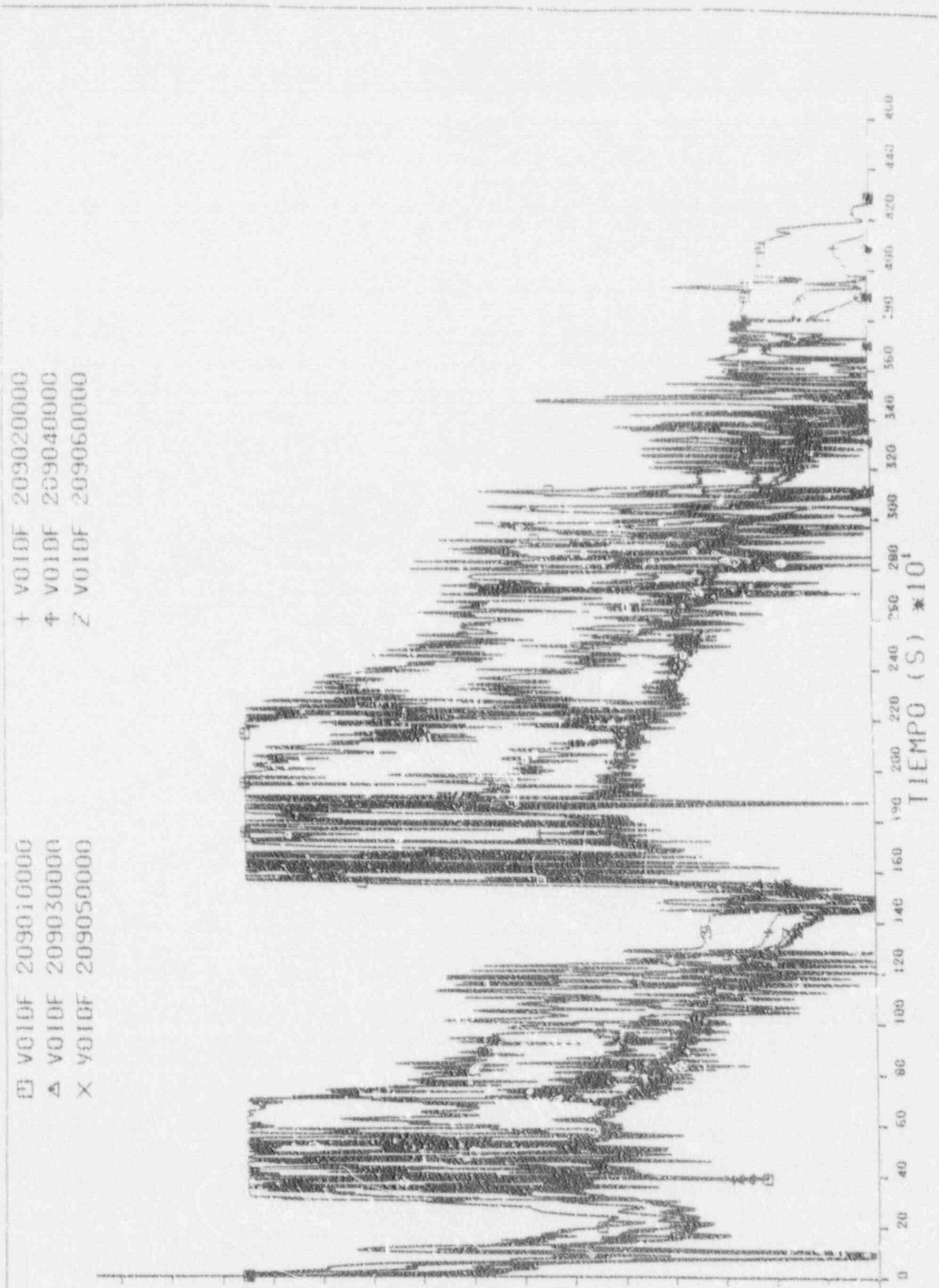

VOIDF 209010000
VOIDF 209020000
VOIDF 209030000
VOIDF 209040000
VOIDF 209050000
VOIDF 209060000
TIEMPO (S) *10¹
FRACCIONES DE LIQUIDO EN EL NUCLEO (O/1)

FIG. I (B) - 20

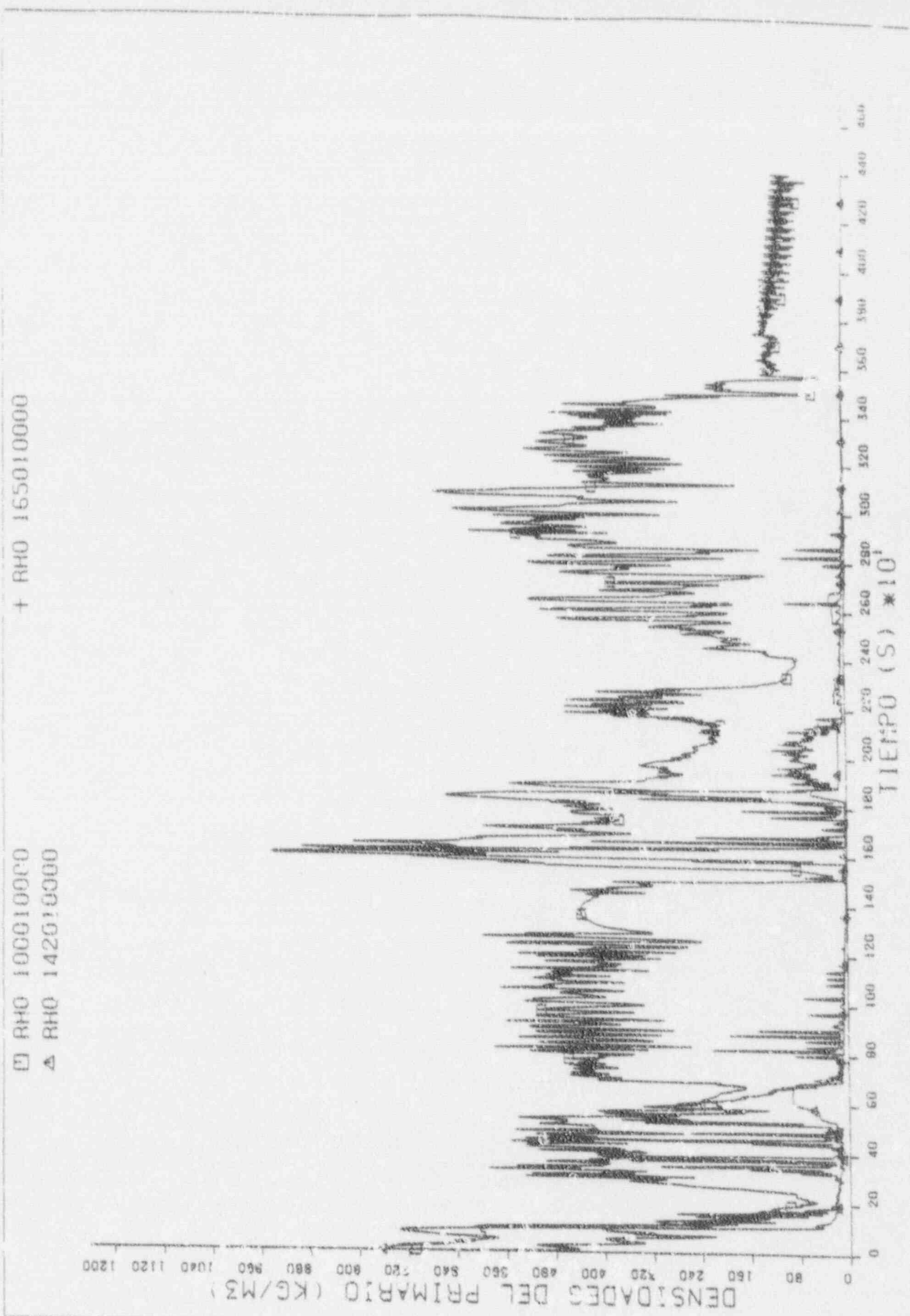

□ RHO 100010000
△ RHO 142010000
+ RHO 165010000
DENSIDADES DEL PRIMARIO (KG/M3)
TIEMPO (S) *10^1

FIG. I (B) - 21

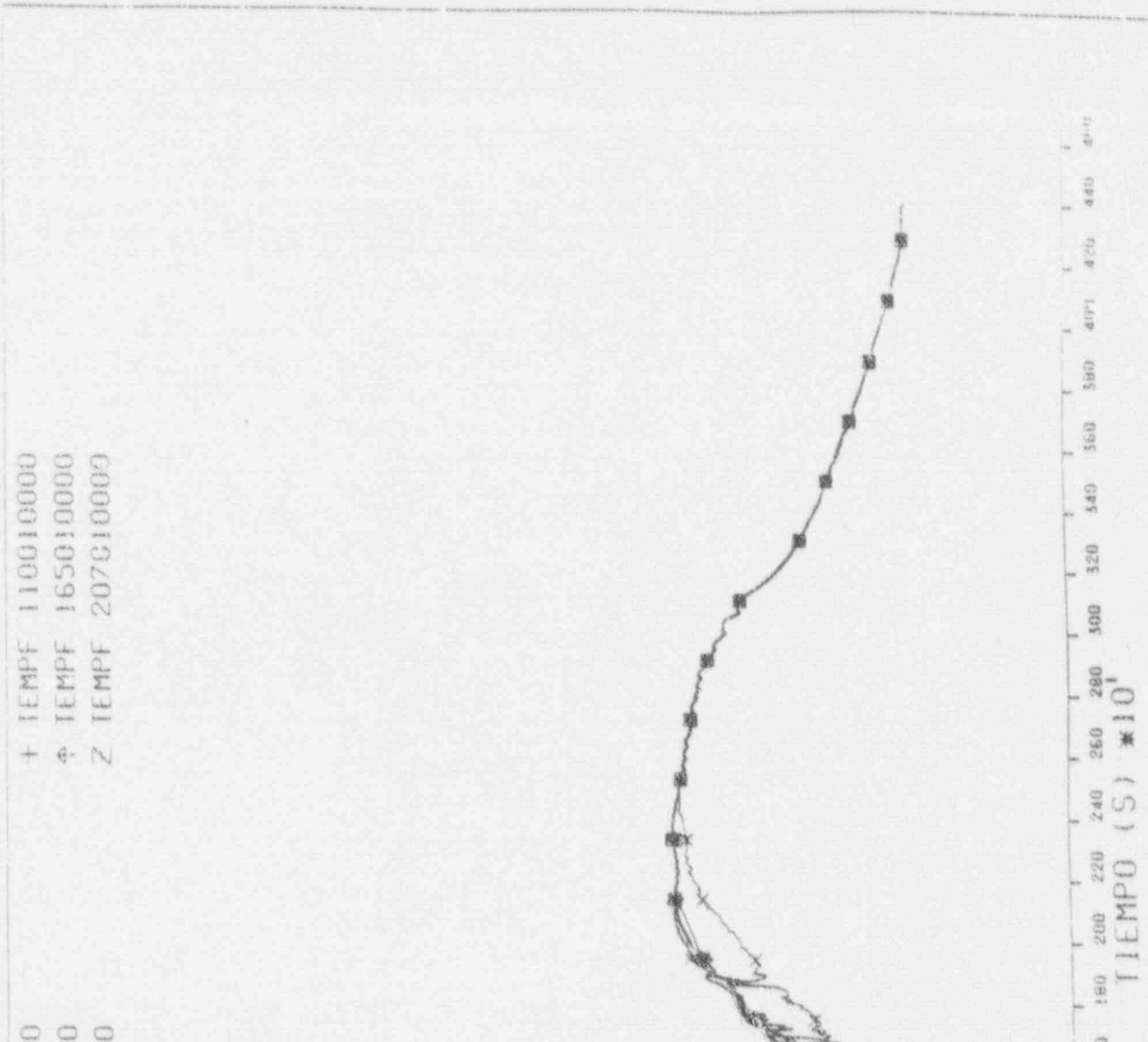

TEMPF 100010000
TEMPF 130010000
TEMPF 211010000
TEMPF 110010000
TEMPF 165010000
TEMPF 207010000
TIEMPO (S) *10^1
TEMPERATURAS EN PRIMARIO (K)
ROTURA SB3 (6 INCH) CON RECUPERACION (FIG 22)

FIG. I (B) - 22

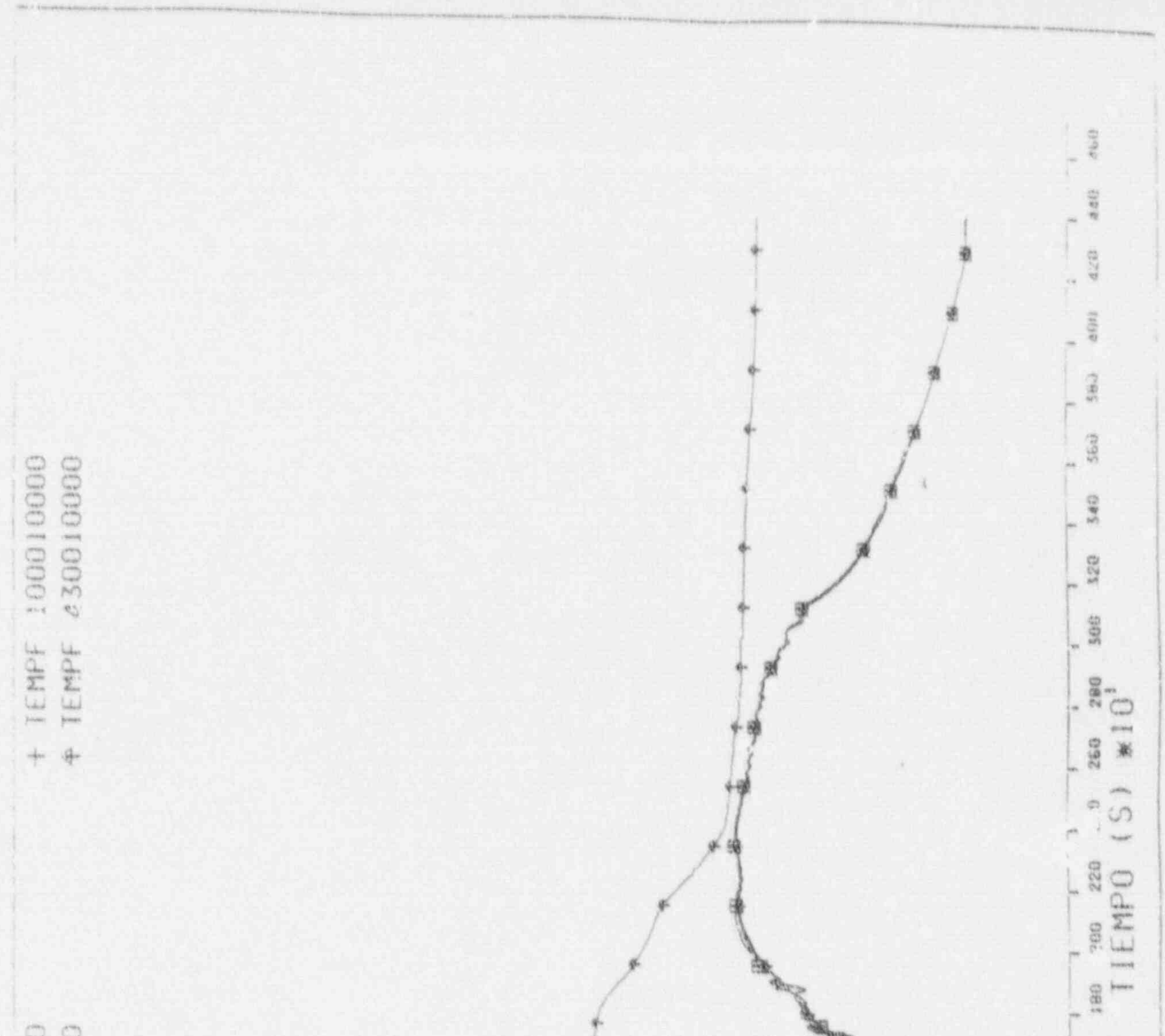

TEMPF 312010000
TEMPF 165010000
TEMPF 100010000
TEMPF 430010000
TIEMPO (S) *10¹
TEMPERATURAS EN PRIMARIO Y G.V. (K)

FIG. I (3) - 23

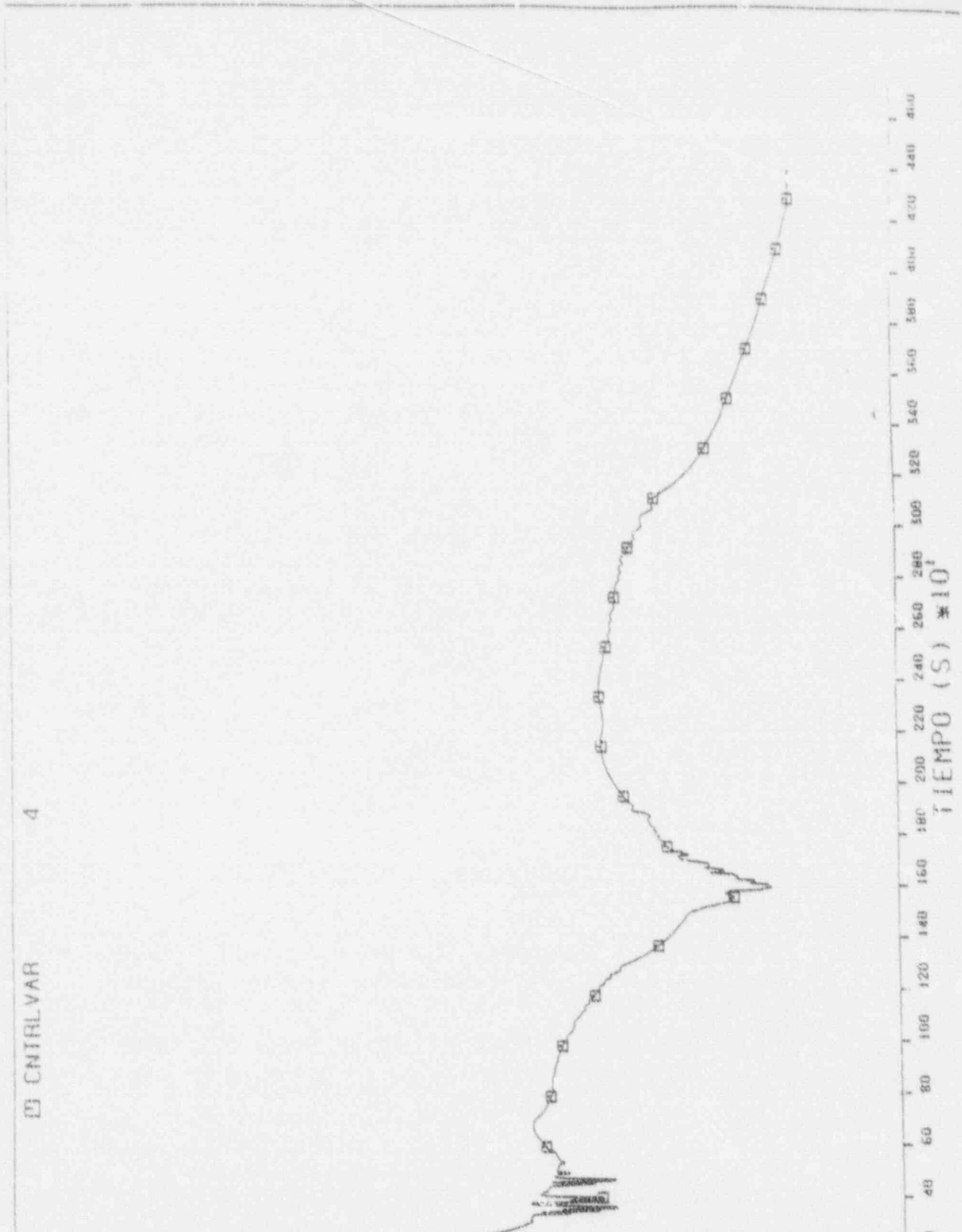


FIG. I (B) - 24

FIG. I (B) - 25

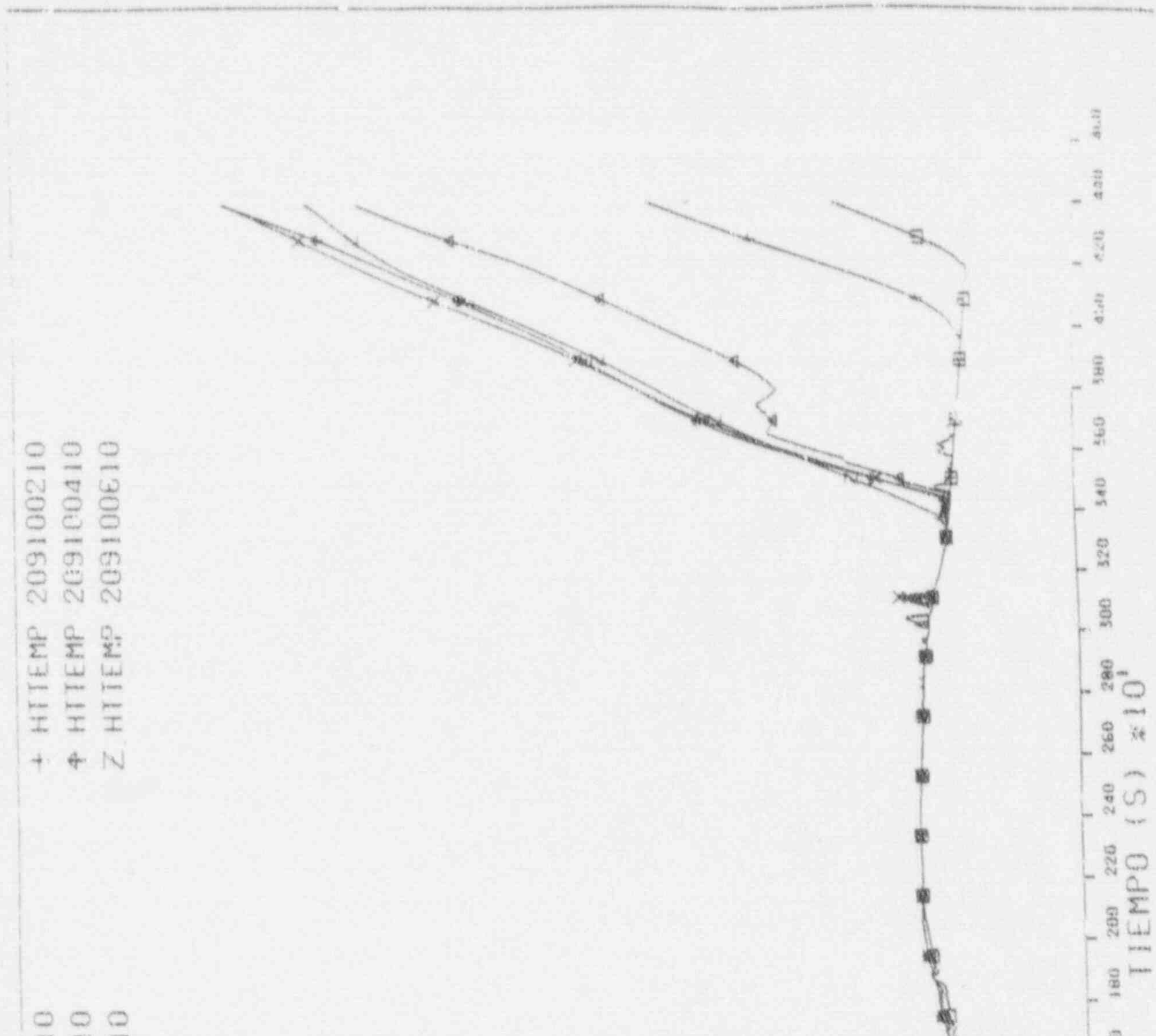

HTTEMP 209100110
HTTEMP 209100310
HTTEMP 209100510
HTTEMP 209100210
HTTEMP 209100410
HTTEMP 209100610
TEMPERATURA VAINAS CANAL MEDIO (K)
TIEMPO (S) *10[1]
CON RECUPERACION

FIG. I (B) - 26

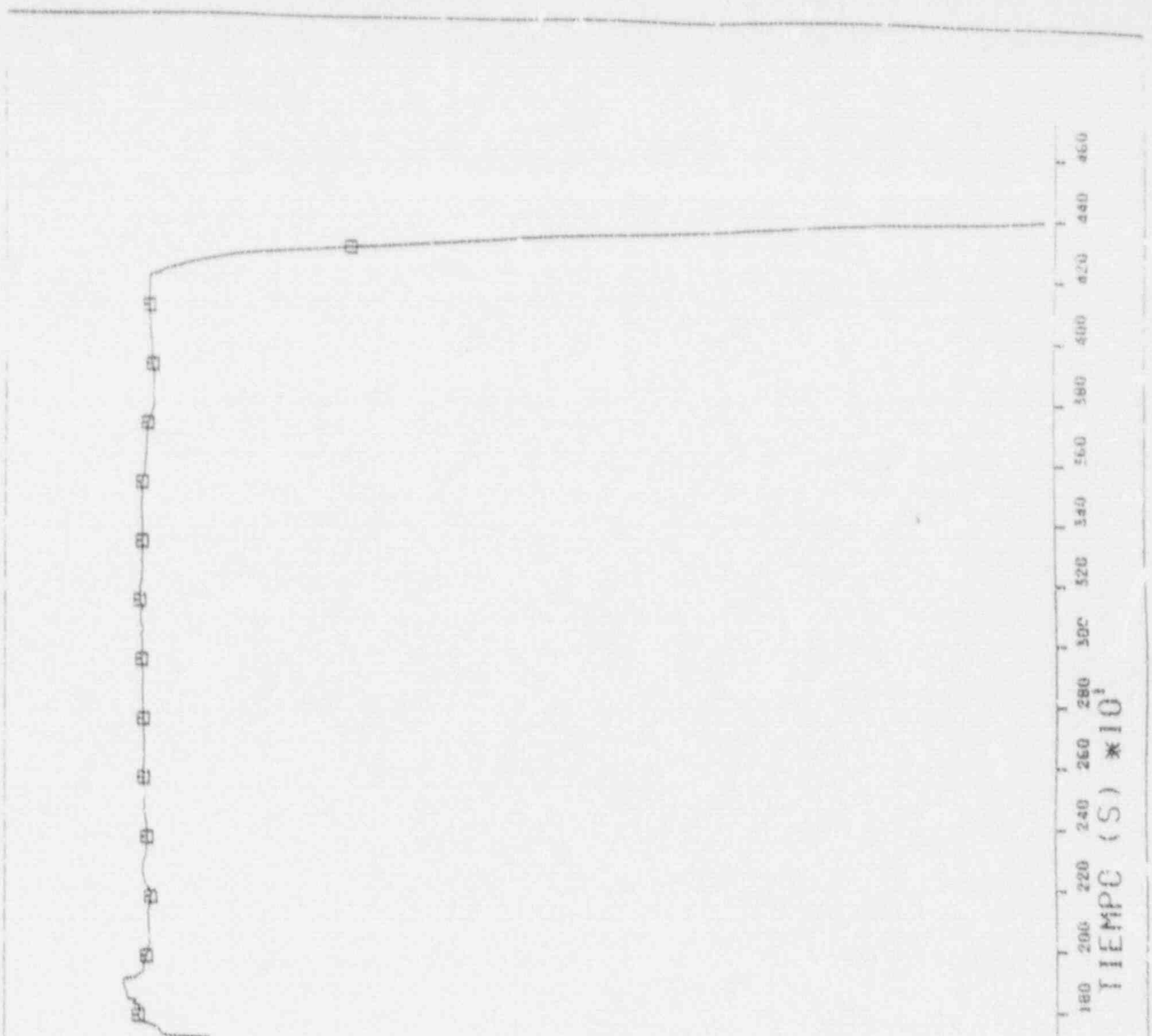

EMASSO
TIEMPO (S) *10^1
ERROR DE MASA (KG)
0
20
40
60
80
100
120
140
160
180
200
220
240
260
280
300
320
340
360
380
400
420
440
460
23
18
13
8
3
-2
-7
-12
-17
-22
-27
-32
-37
-42
-47
-52

FIG. I (B) - 27

APPENDIX V : FIGURES OF CASE II(B), (2" WITH RECOVERY)

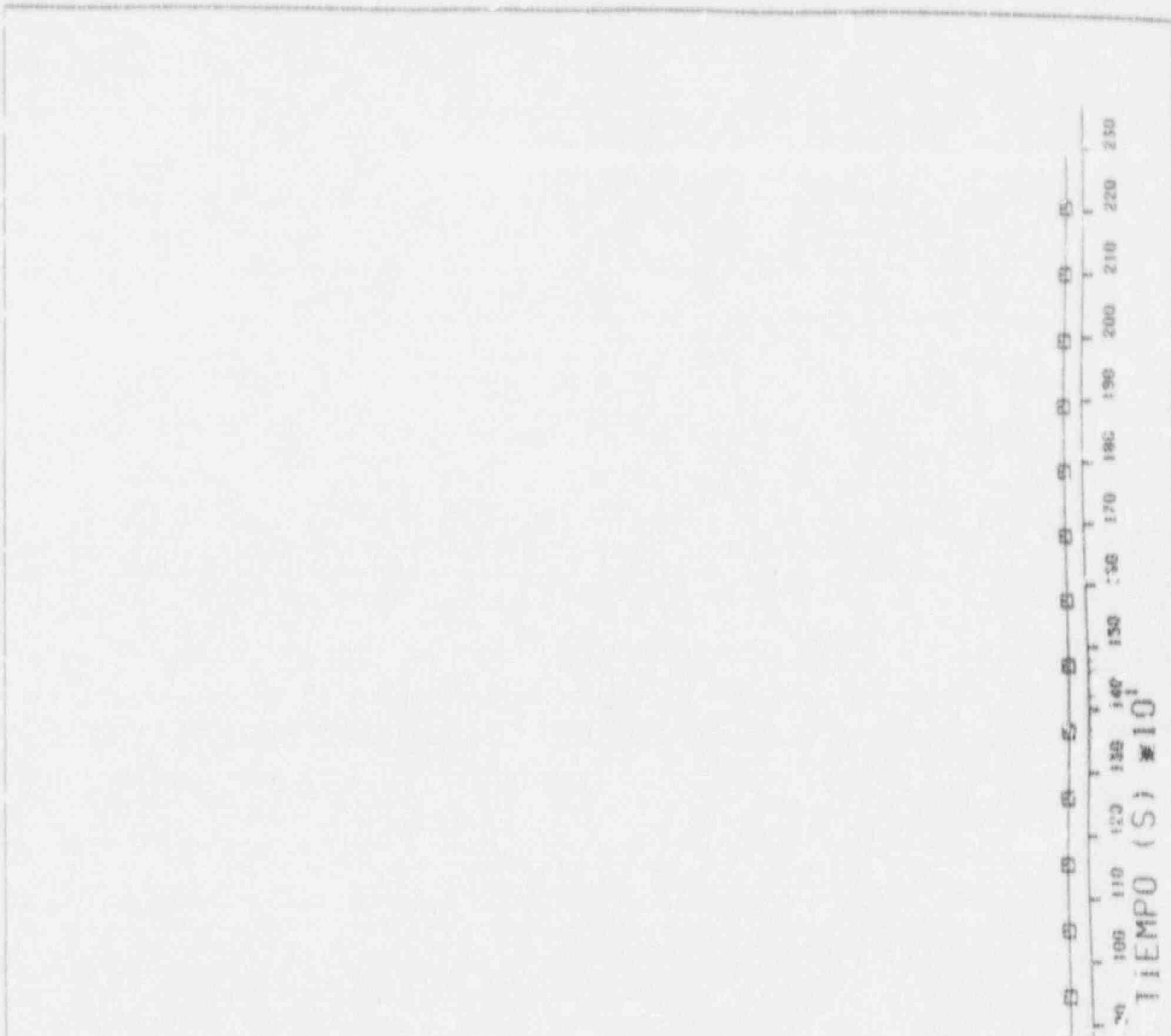
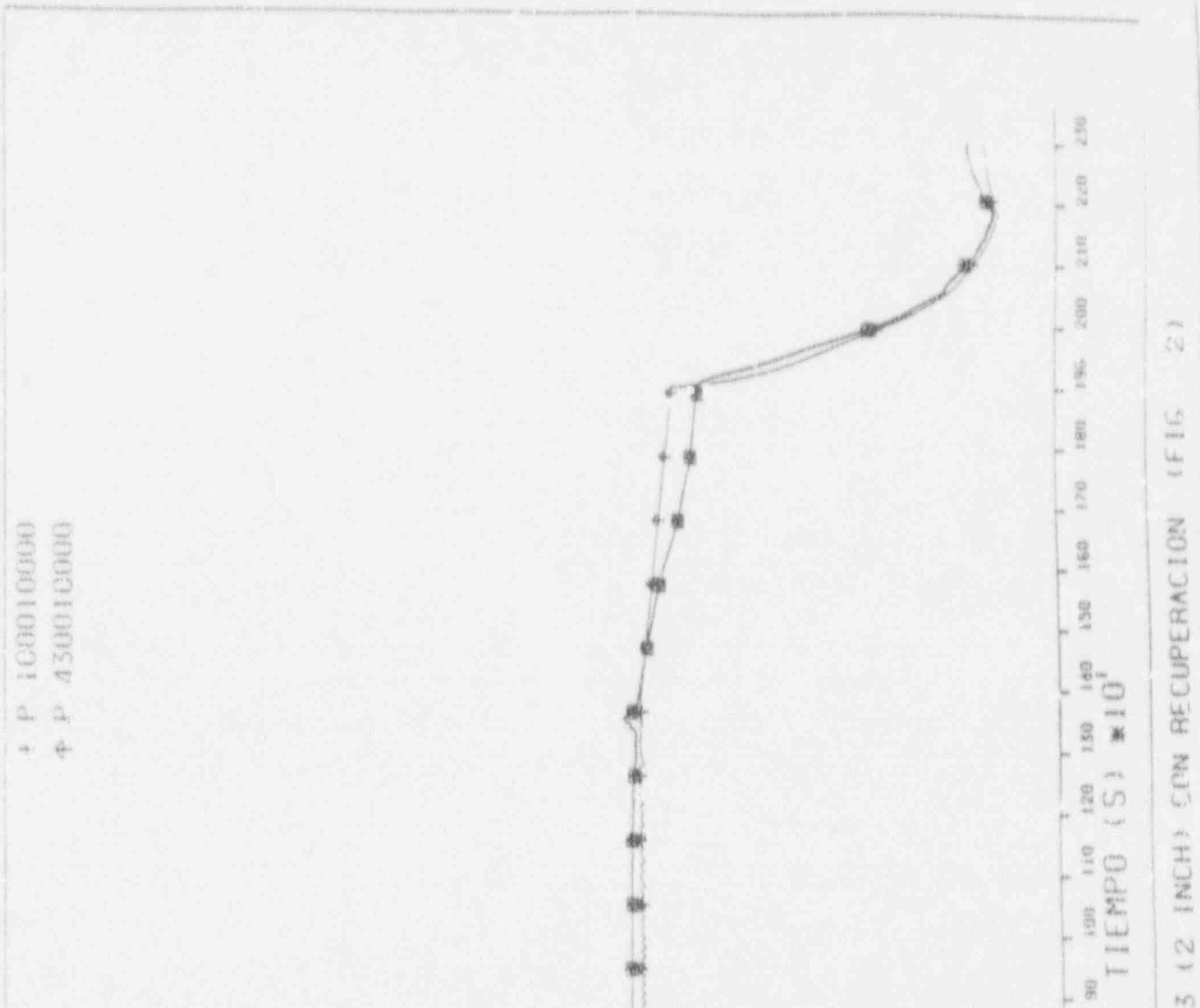

RKTPOWN
POTENCIA DEL REACTOR (W) *10^6
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 1)

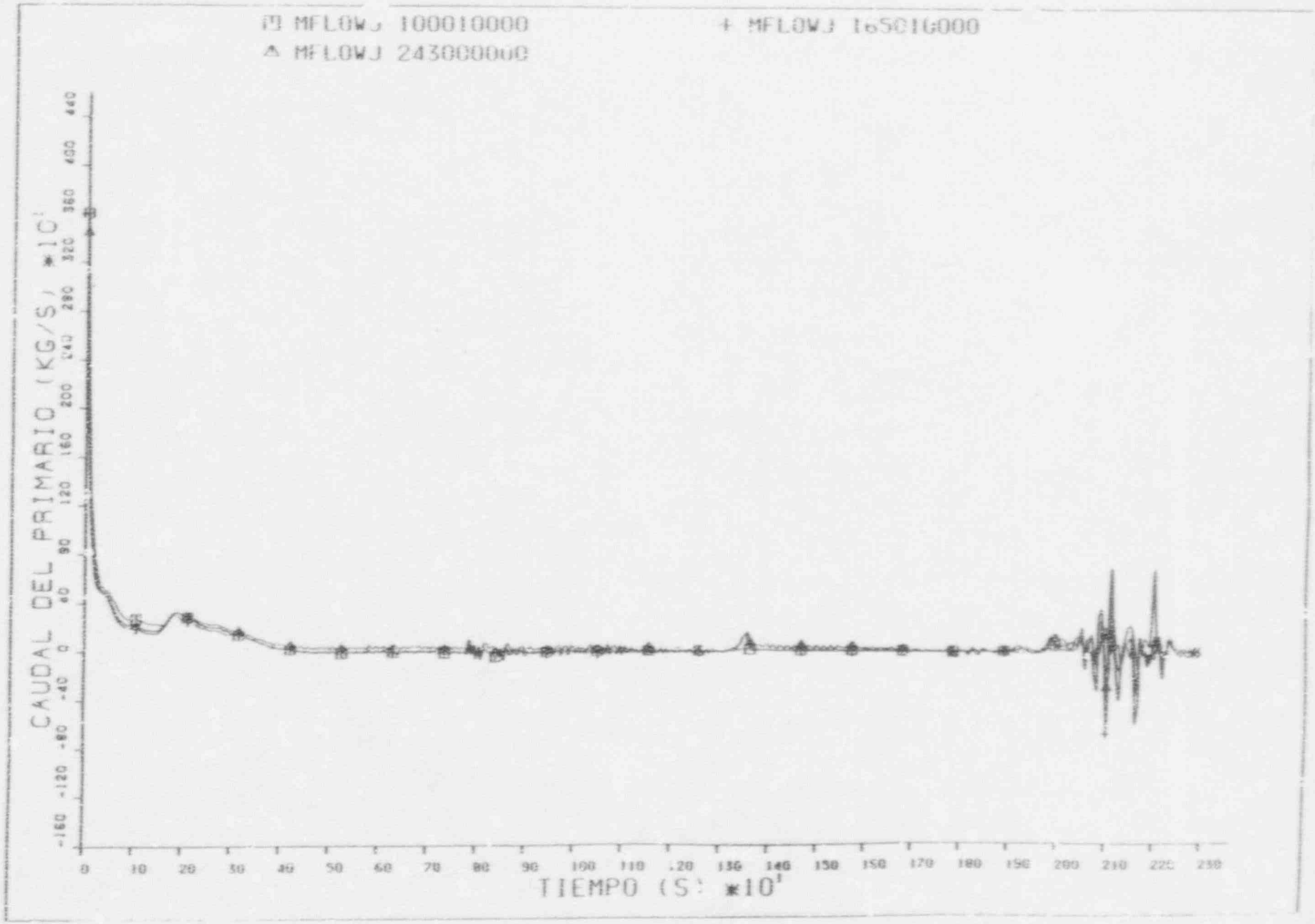
FIG. II (B) - 1



P 312020000
P 165010000
P 100010000
P 430010000
TIEMPO (S) *10
PRESION (PA) *10
C.N.J.C. ROTURA SG3 (2 INCH)

FIG. II (B) - 2

FIG. II (B) - 3



C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 3)

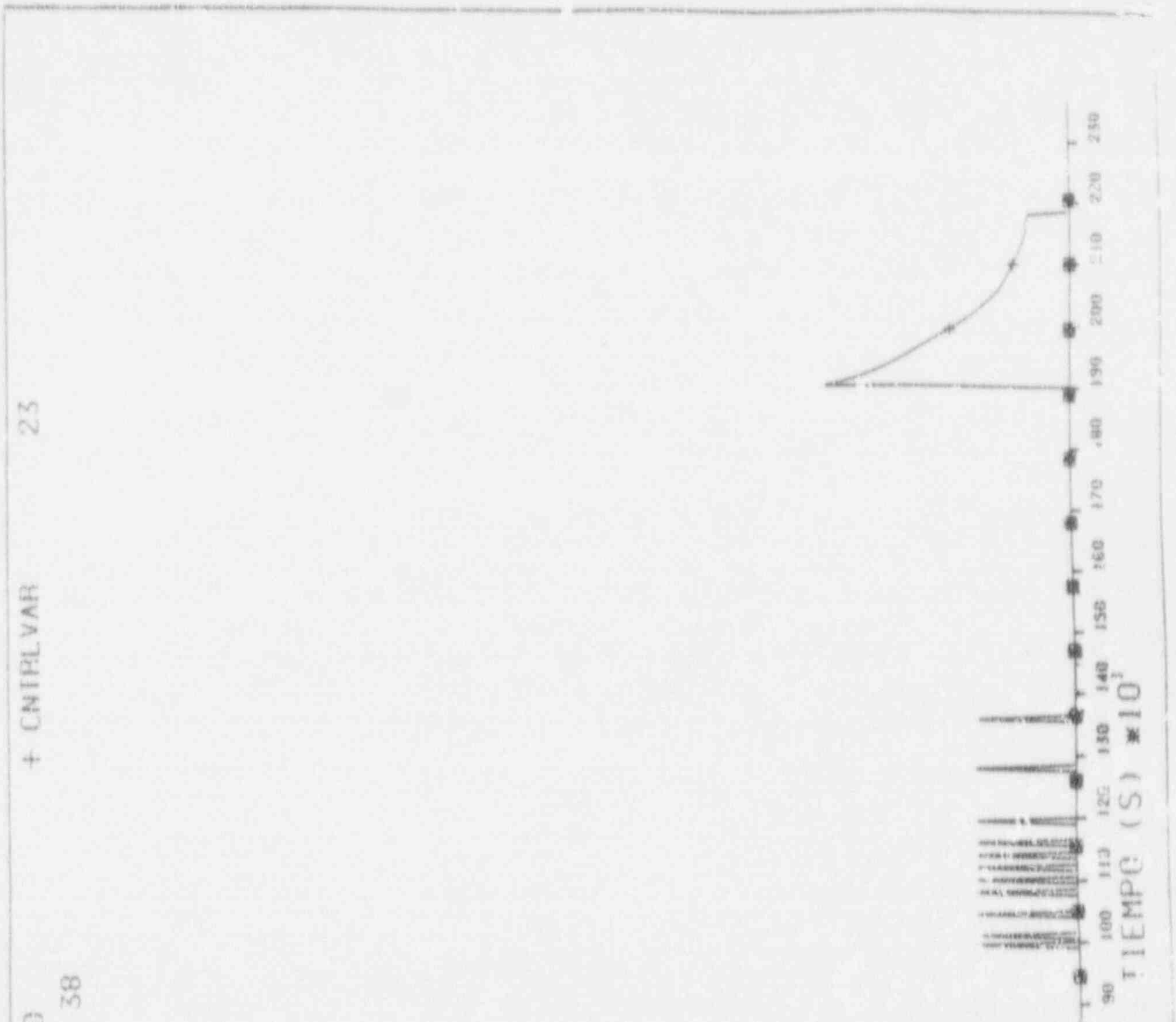
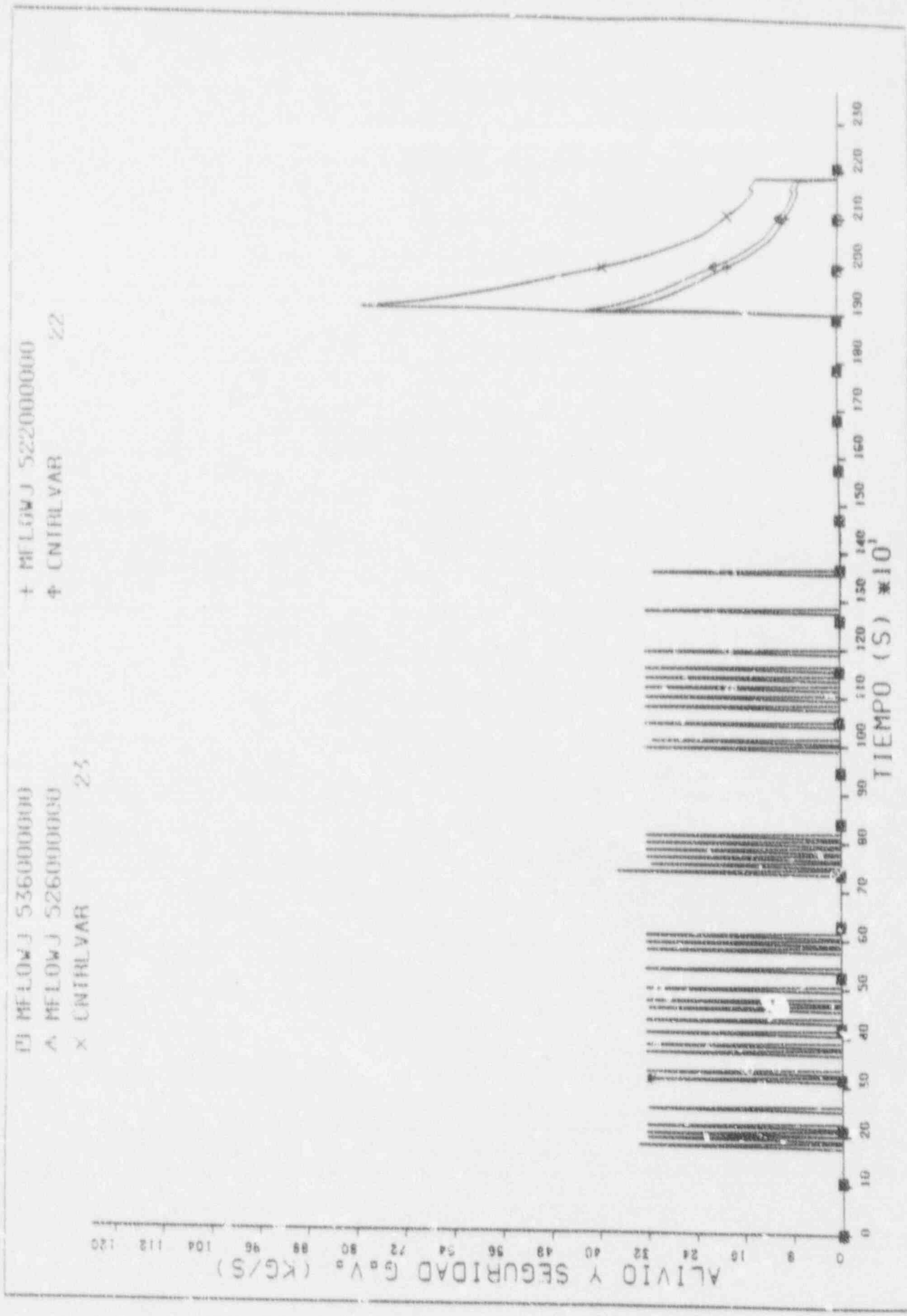

MFLOWJ 5:40000000
CNTRLVAR 38
+ CNTRLVAR 23
CAUDALES DE SECUNDARIO (KG/S)
TIEMPO (S) *10
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 4)

FIG. II (B) - 4



MFLOWJ 536000000
MFLOWJ 526000000
CNTRLVAR 23
MFLOWJ 522000000
CNTRLVAR 22
ALIVIO Y SEGURIDAD G.V. (KG/S)
TIEMPO (S) *10^1
C.N. I.C. ROTURA SR3 (2 INCH) CON RECUPERACION (FIG 5)

FIG. II (B) - 5

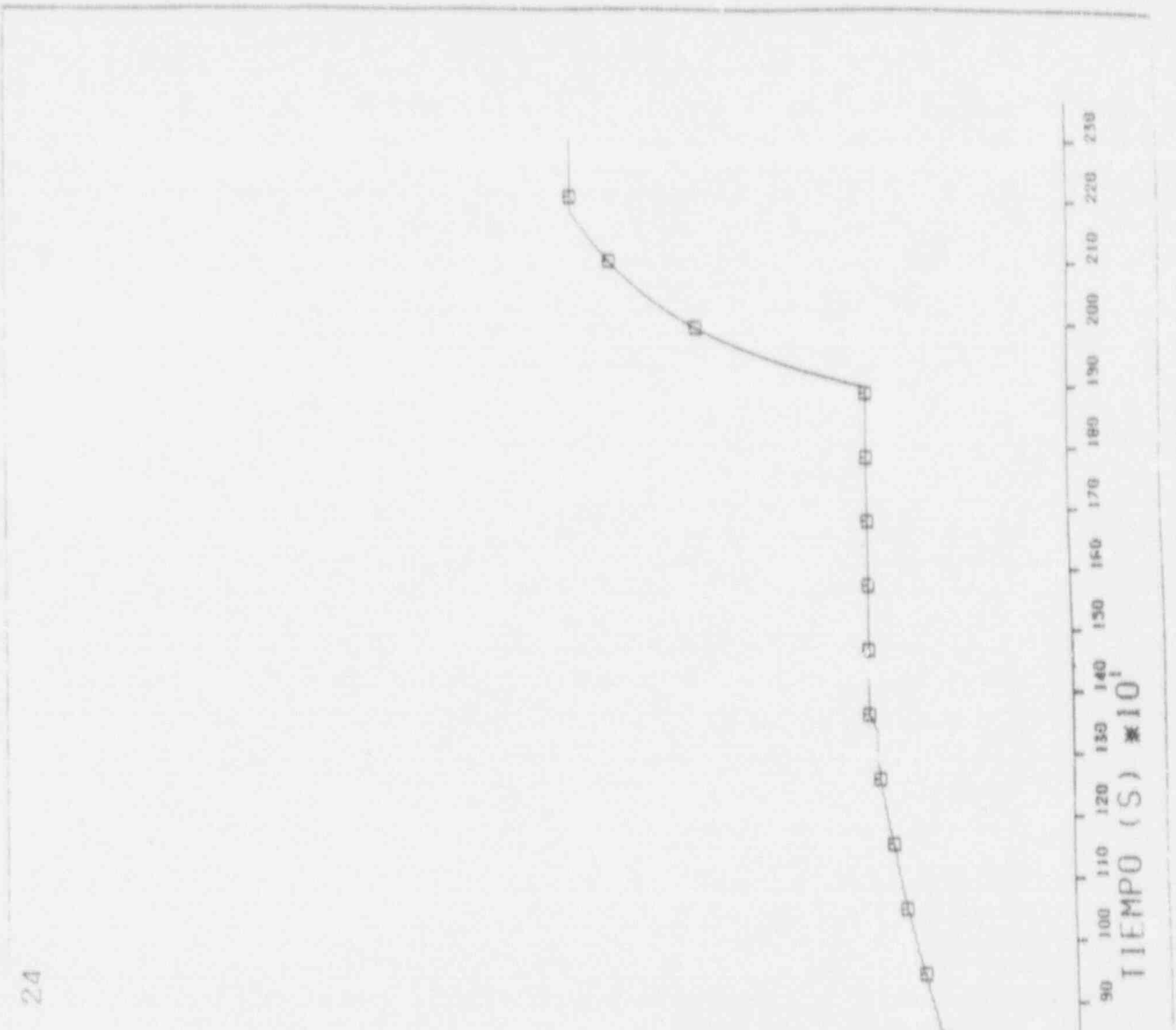
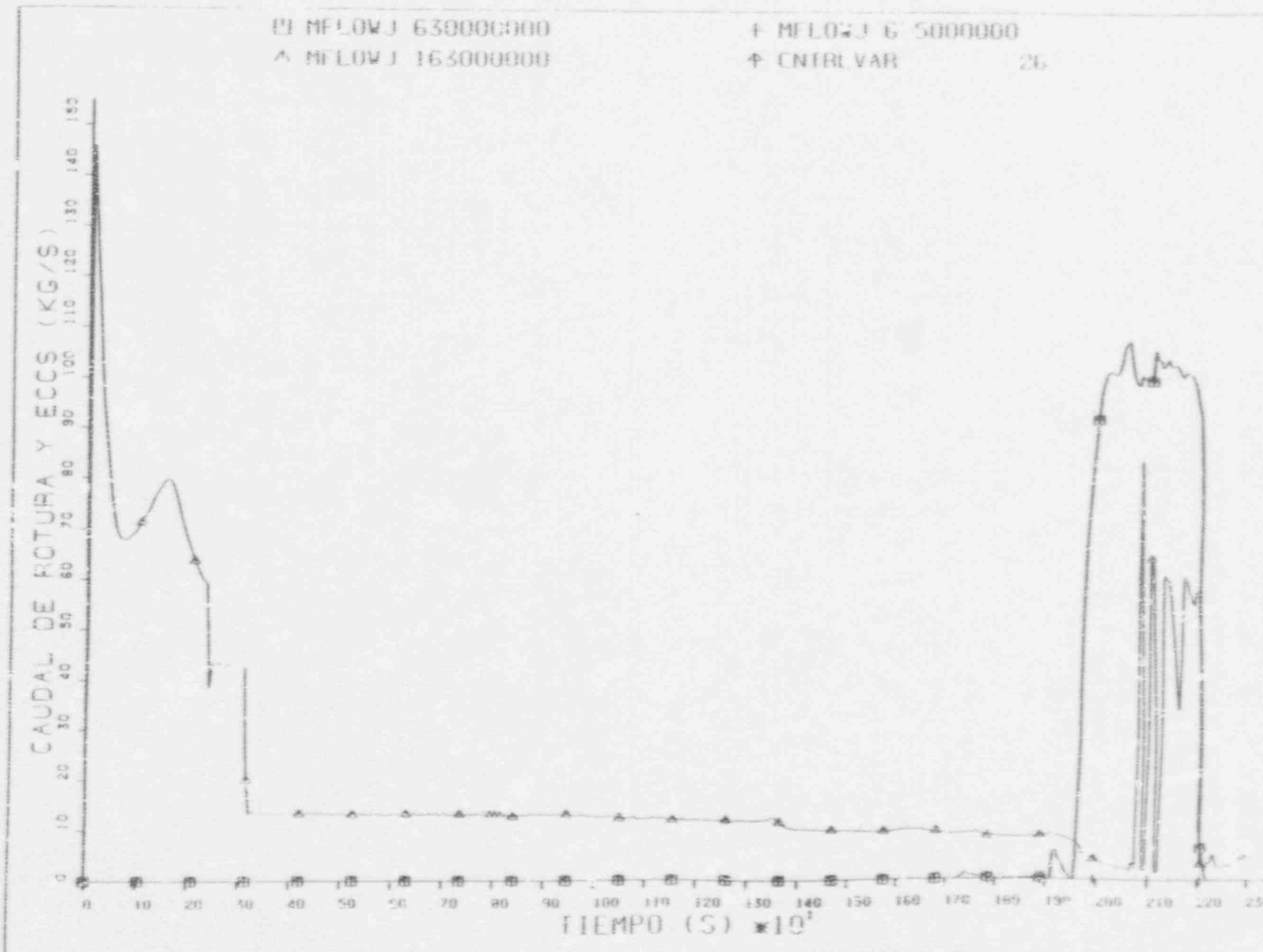

19 CNTRLVAR 24
MASA DESCARGADA DEL G.V. A ATM (KG) *10²
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 6)

FIG. II (B) - 6



C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 7)

FIG. II (B) - 7

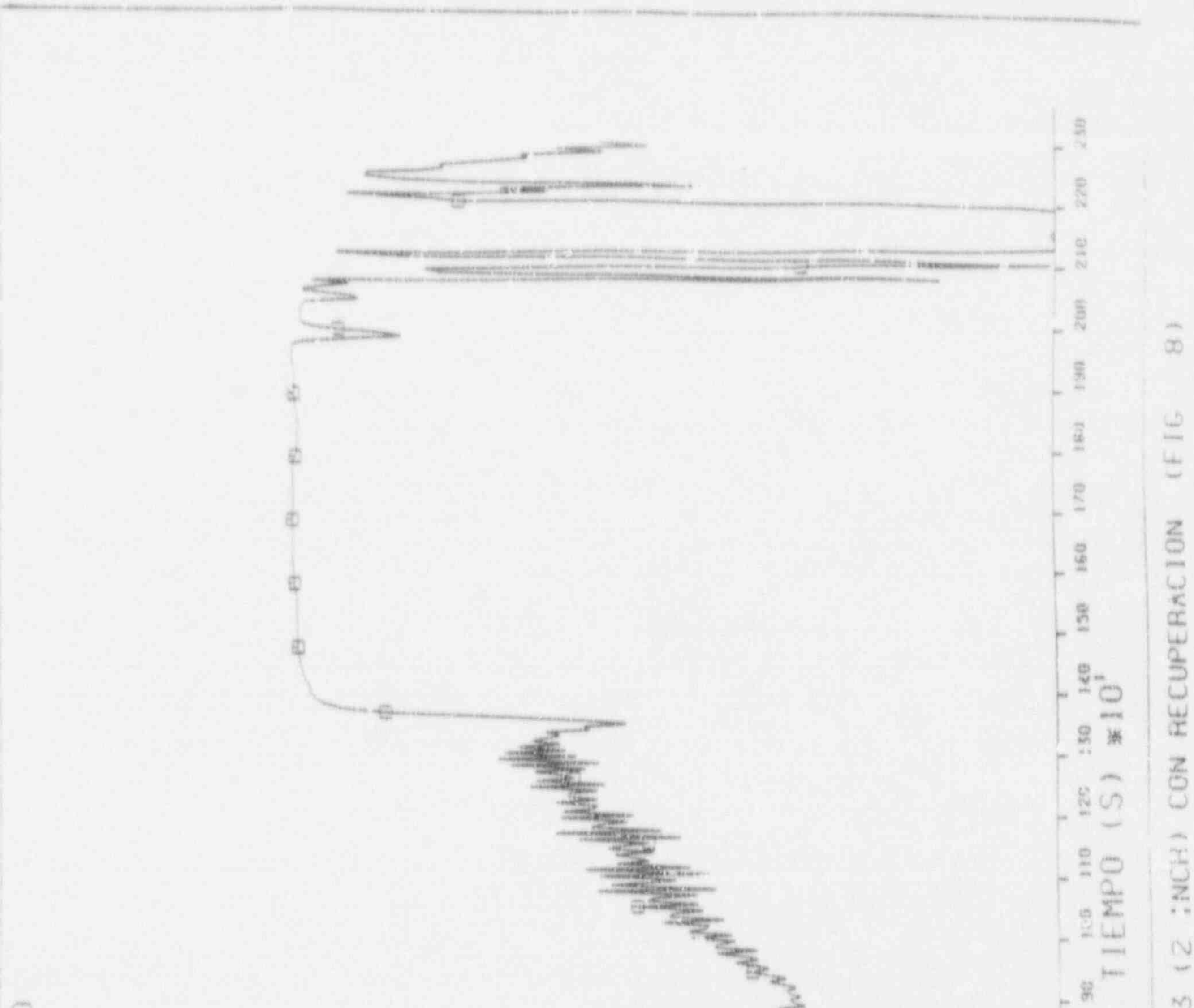
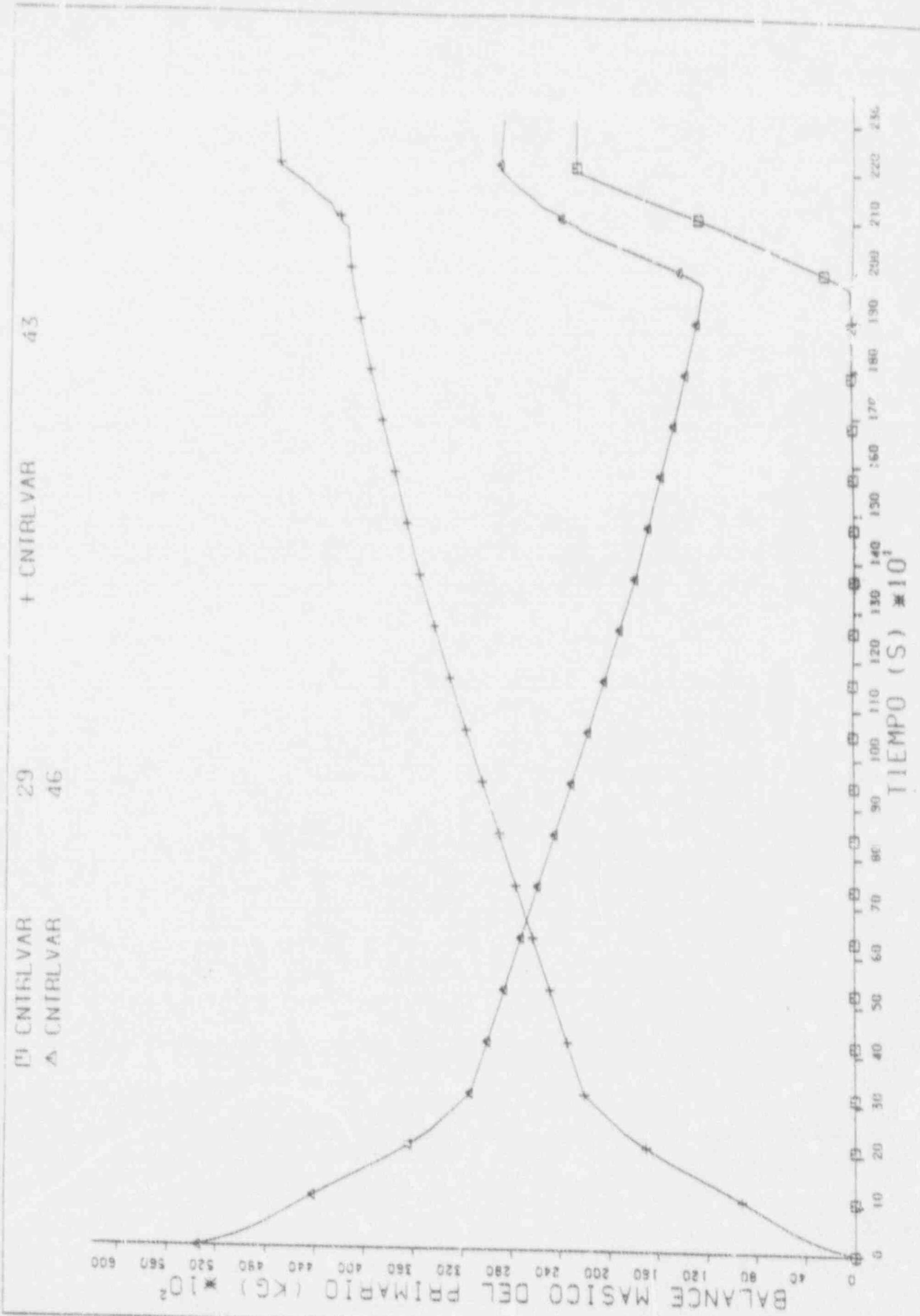

VOIDGJ 163000000
FRACCION DE VAPOR EN LA ROTURA (O/I)
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (2 :NCH) CON RECUPERACION (FIG 8)

FIG. II (B) - 8



□ CNTRLVAR 29
△ CNTRLVAR 46
+ CNTRLVAR 43
BALANCE MASICO DEL PRIMARIO (KG) *10²
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 9)

FIG. II (B) - 9

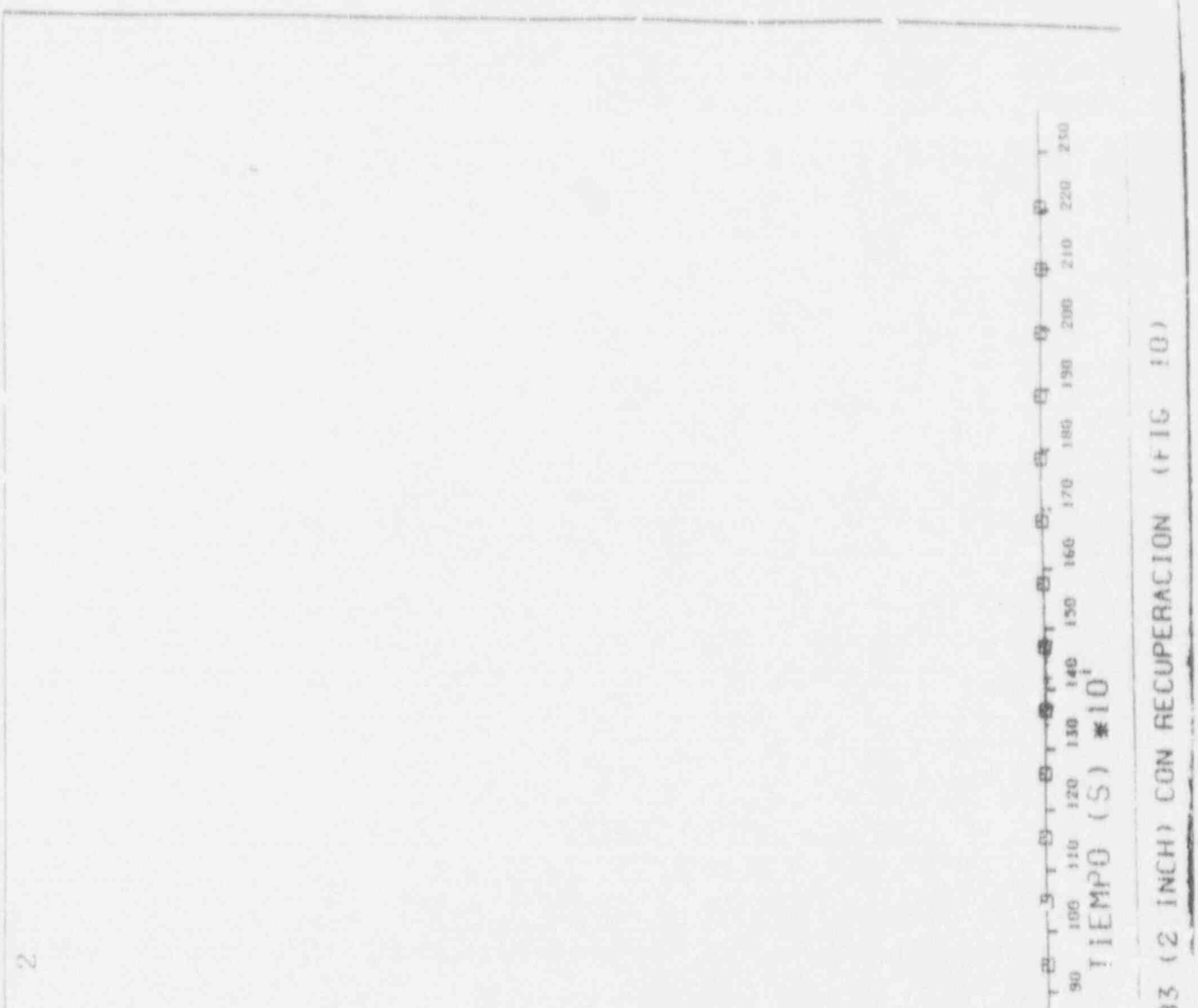
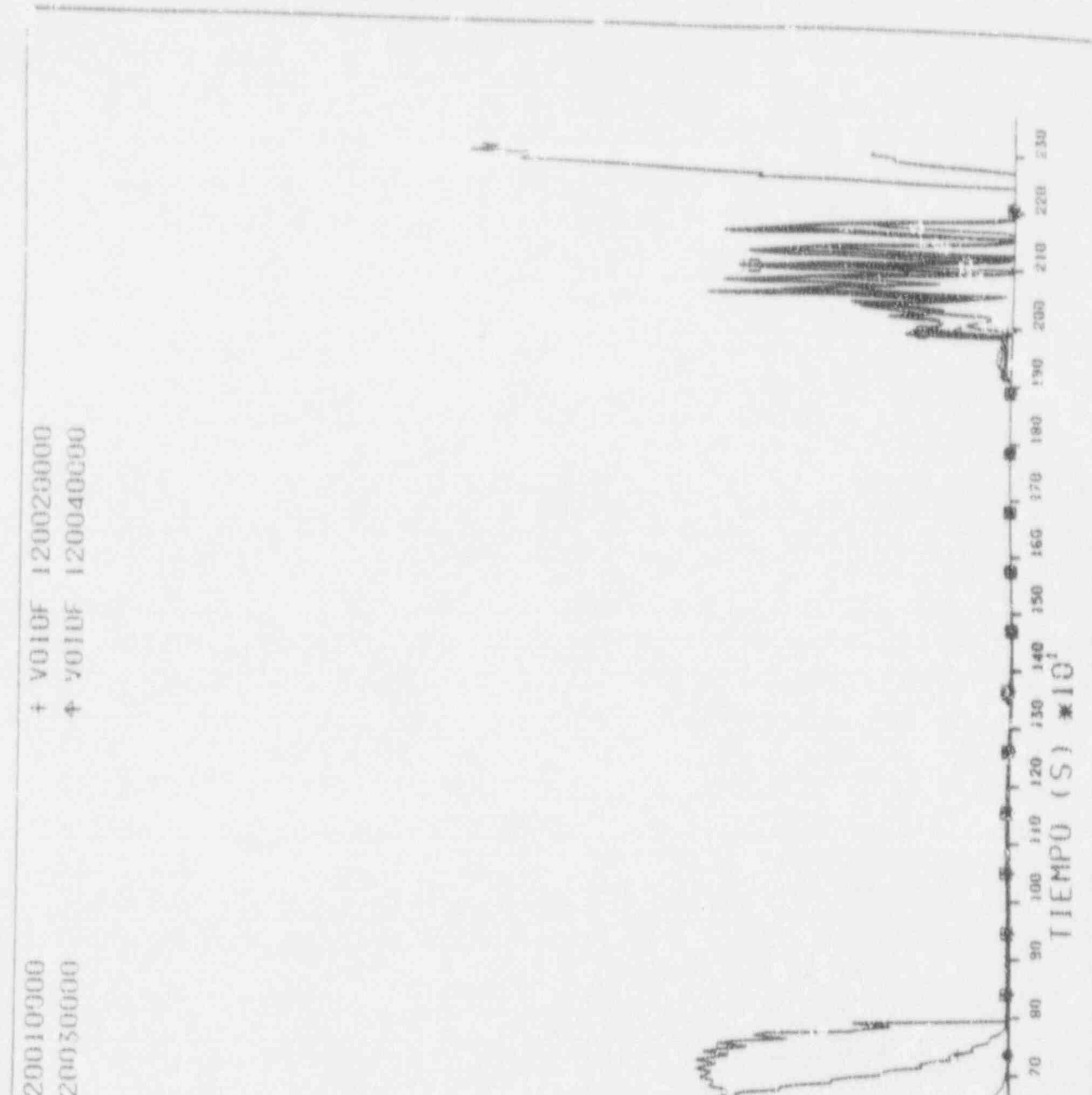

CNTRLVAR 2
TIEMPO (S) *10^1
NIVEL COLAPSADO EN PRESIONADOR (%)
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 10)

FIG. II (B) - 10



VOIDF 120010000
VOIDF 120030000
VOIDF 120020000
VOIDF 120040000
TIEMPO (S) *10^1
FRAC. DE LIQUIDO EN TUBO UP G.V. (O/1)
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 11)

FIG. II (B) - 11

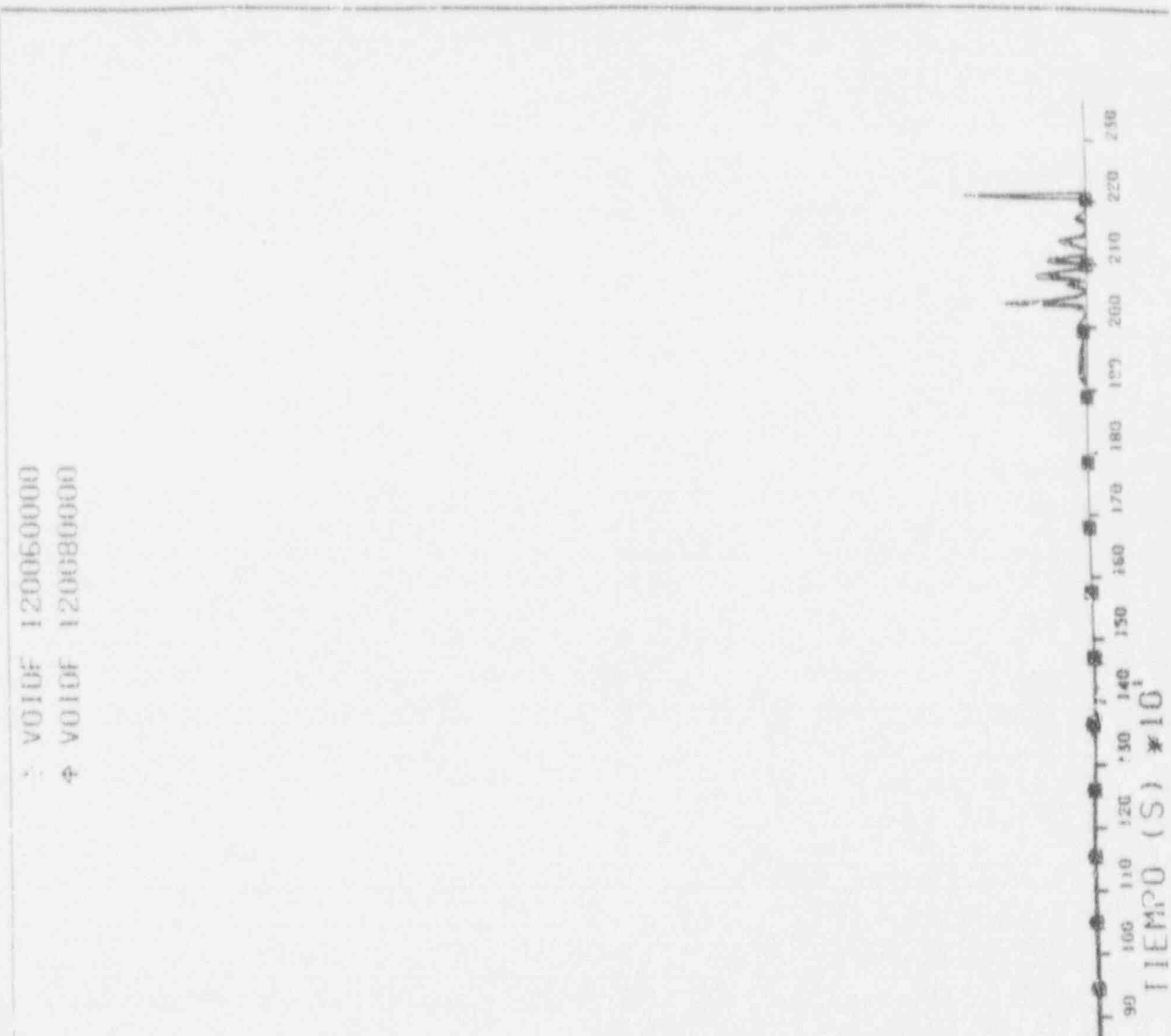

VOIDF 120050000
VOIDF 120060000
VOIDF 120070000
VOIDF 120080000
TIEMPO (S) *10¹
FRAC. DE LIQUIDO EN TUBO DOWN G.V. (0/1)
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 12)

FIG. II (B) - 12

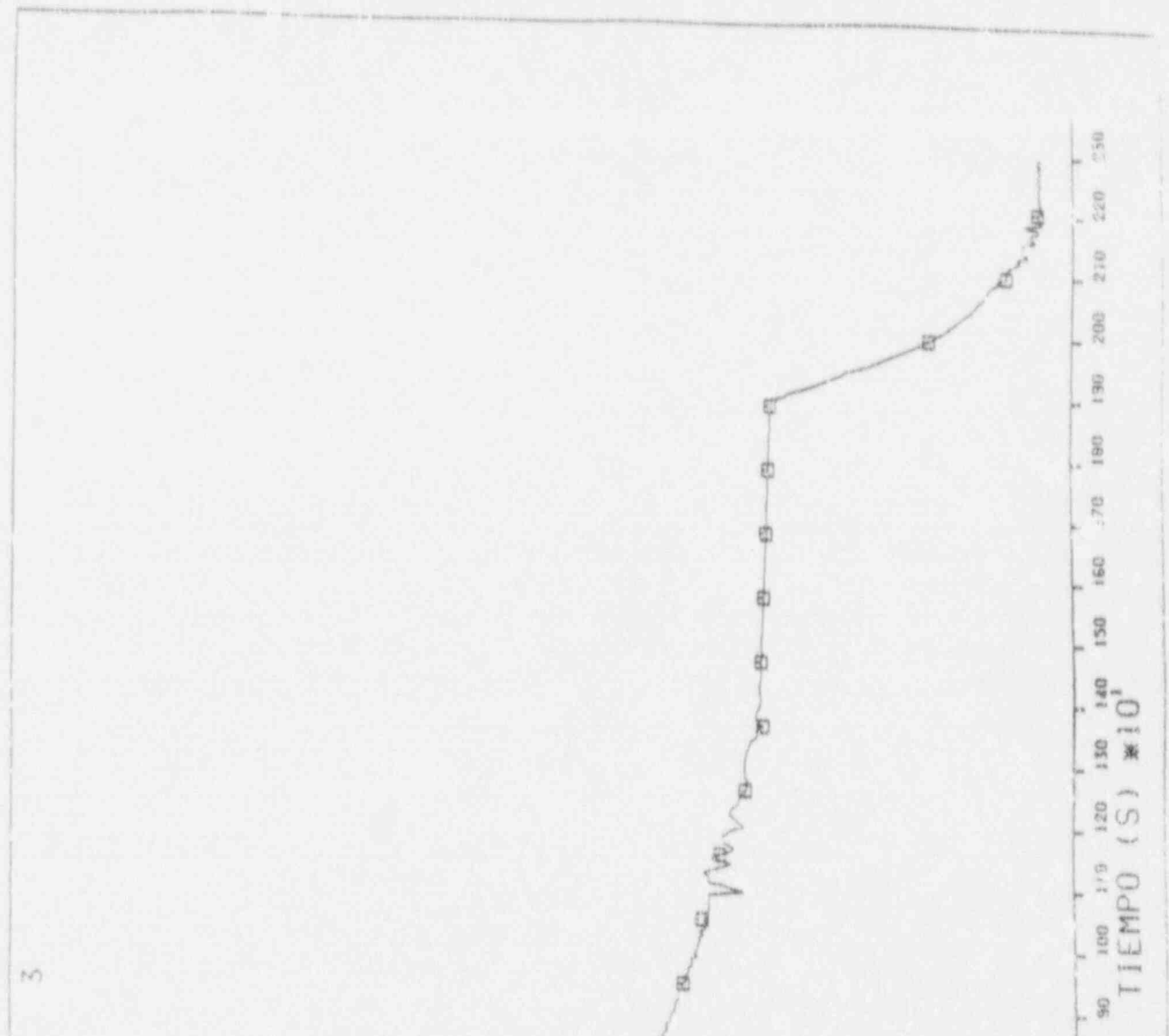

CNTRLVAR 3
TIEMPO (S) *10^1
NIVEL EN DOWNCOMER DEL G.V. (M)
ROTURA SRT (2 INCH) CON RECUPERACION (FIG 13)

FIG. II (B) - 13

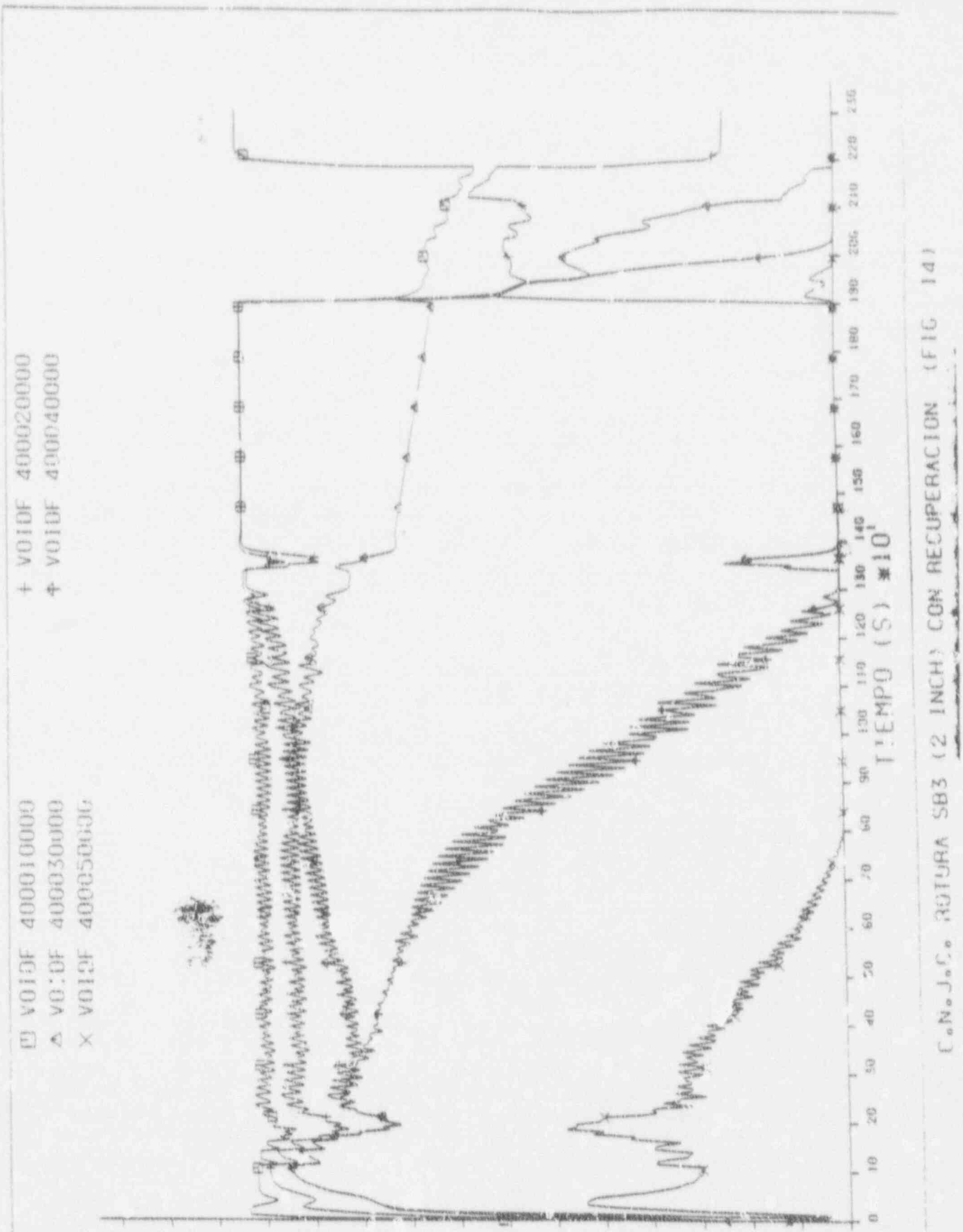
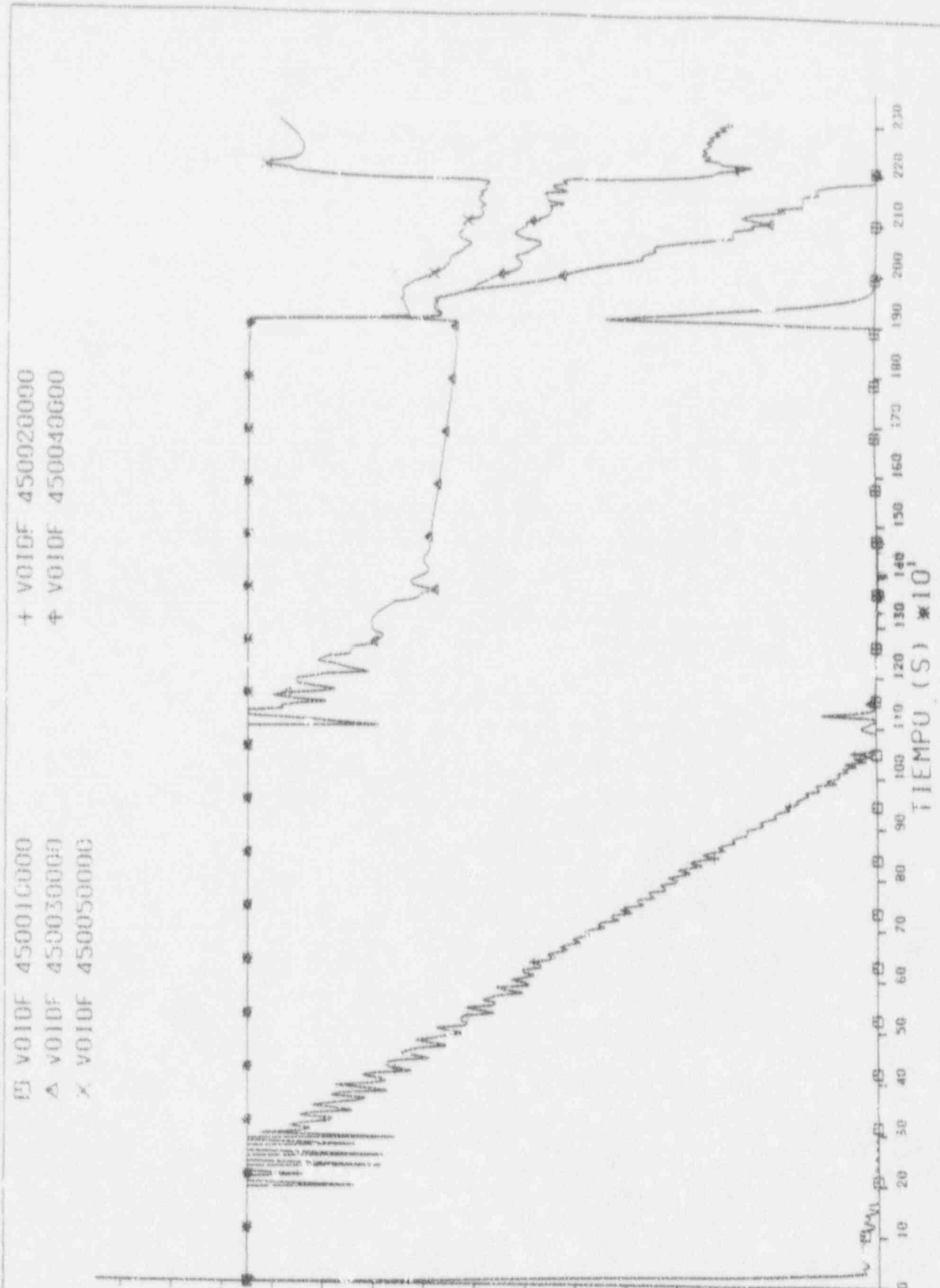

VOIDF 400010000
VOIDF 400030000
VOIDF 400050000
VOIDF 400020000
VOIDF 400040000
FRAC. DE LIQUIDO EN RISER G.V. (C/1)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 14)

FIG. II (F) - 14



VOIDF 450010000
VOIDF 450030000
VOIDF 450050000
VOIDF 450020000
VOIDF 450040000
FRAC. DE LIQUIDO EN DOWNCOMER G.V. (0/1)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 15)

FIG. II (b) - 15

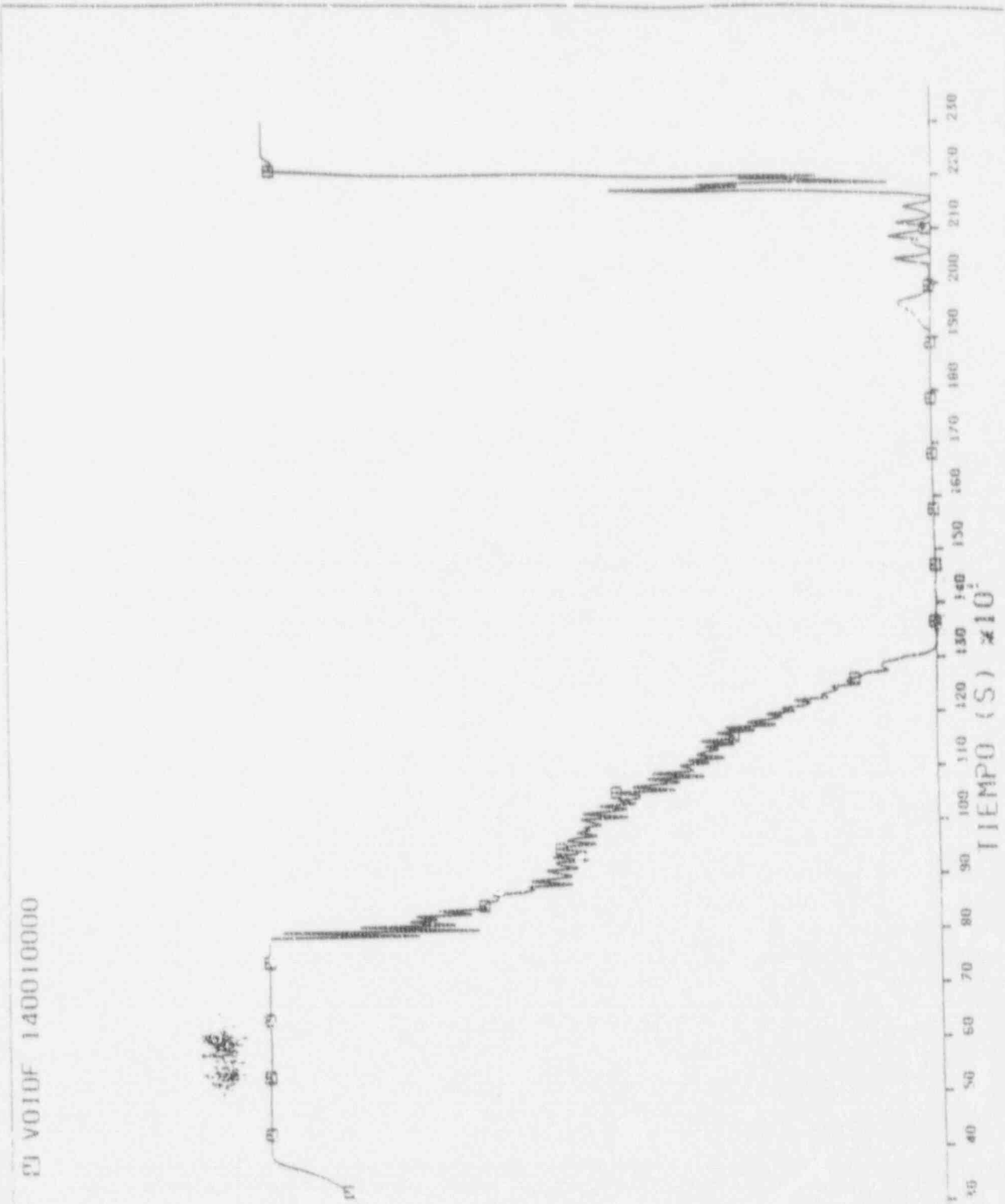

VOIDF 140010000
NIVEL COLAPSADO EN SELLO -140-
TIEMPO (S) x10
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 16)

FIG. II (B) - 16

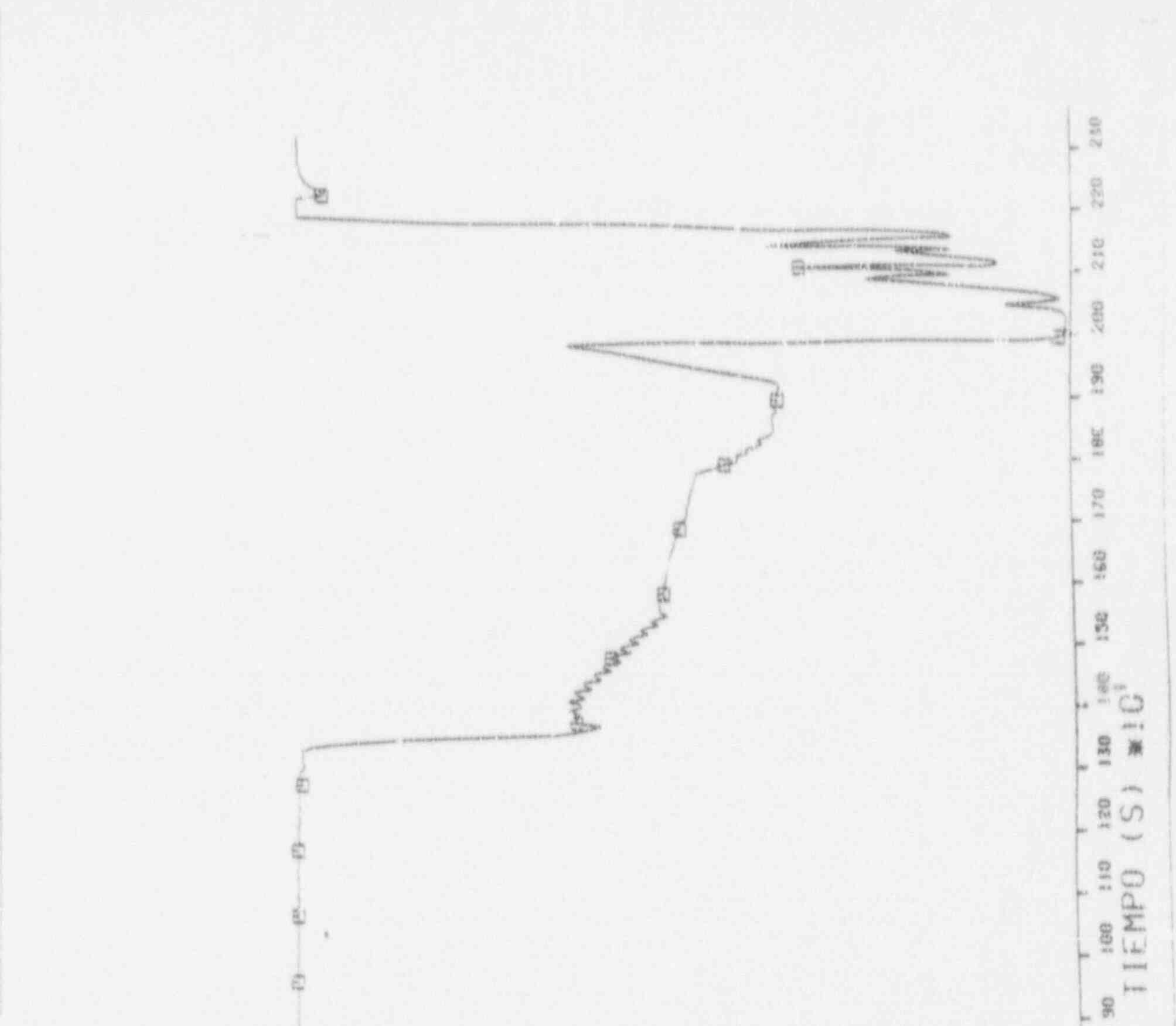
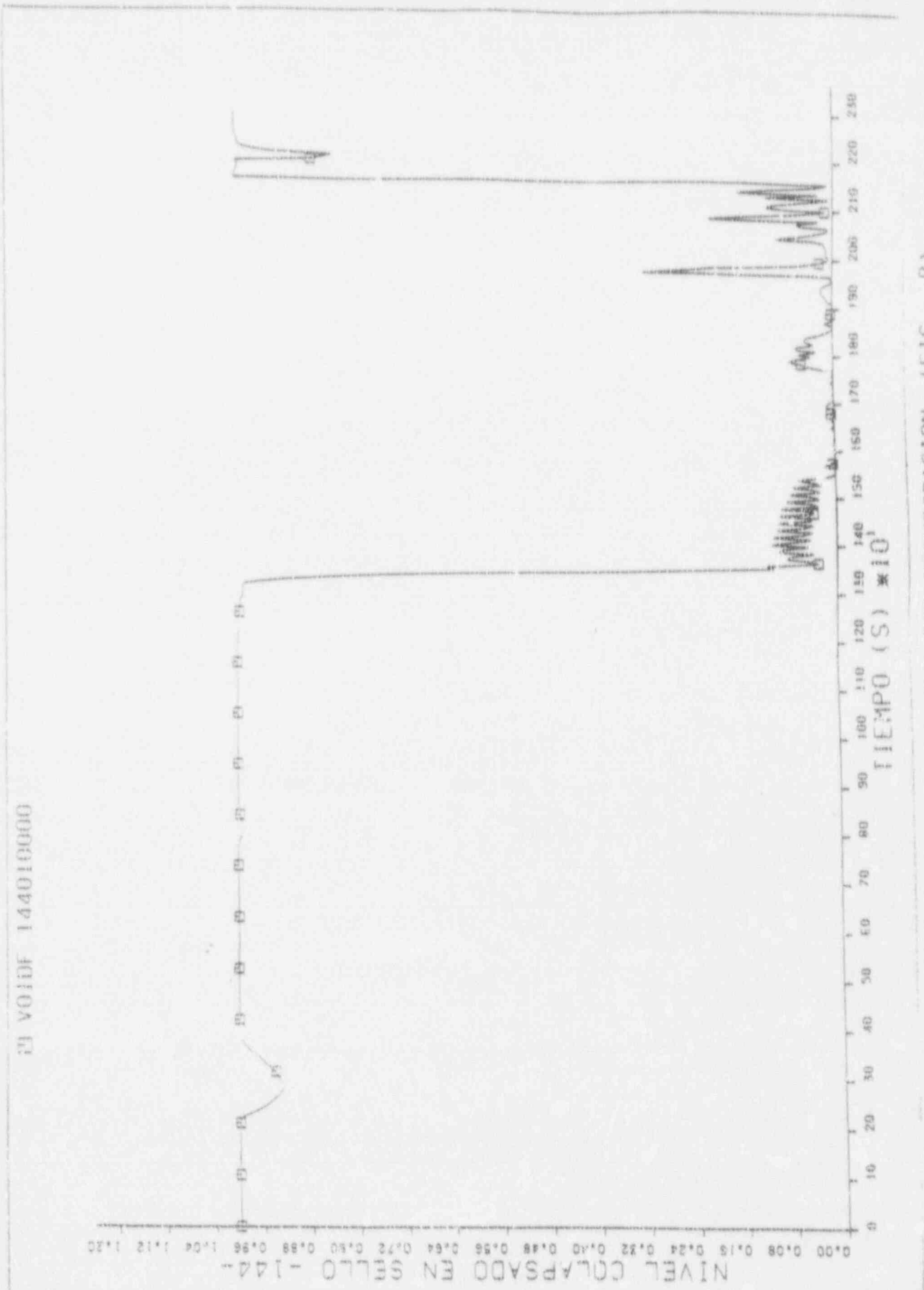

□ VOIDF 142010000
NIVEL COLAPSADO EN SELLO -142-
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 17)

FIG. II (B) - 17



VOIDF 144010000
NIVEL COLAPSADO EN SELLO -144-
TIEMPO (S) *10¹
C.N. J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG .8)

FIG. II (B) - 18

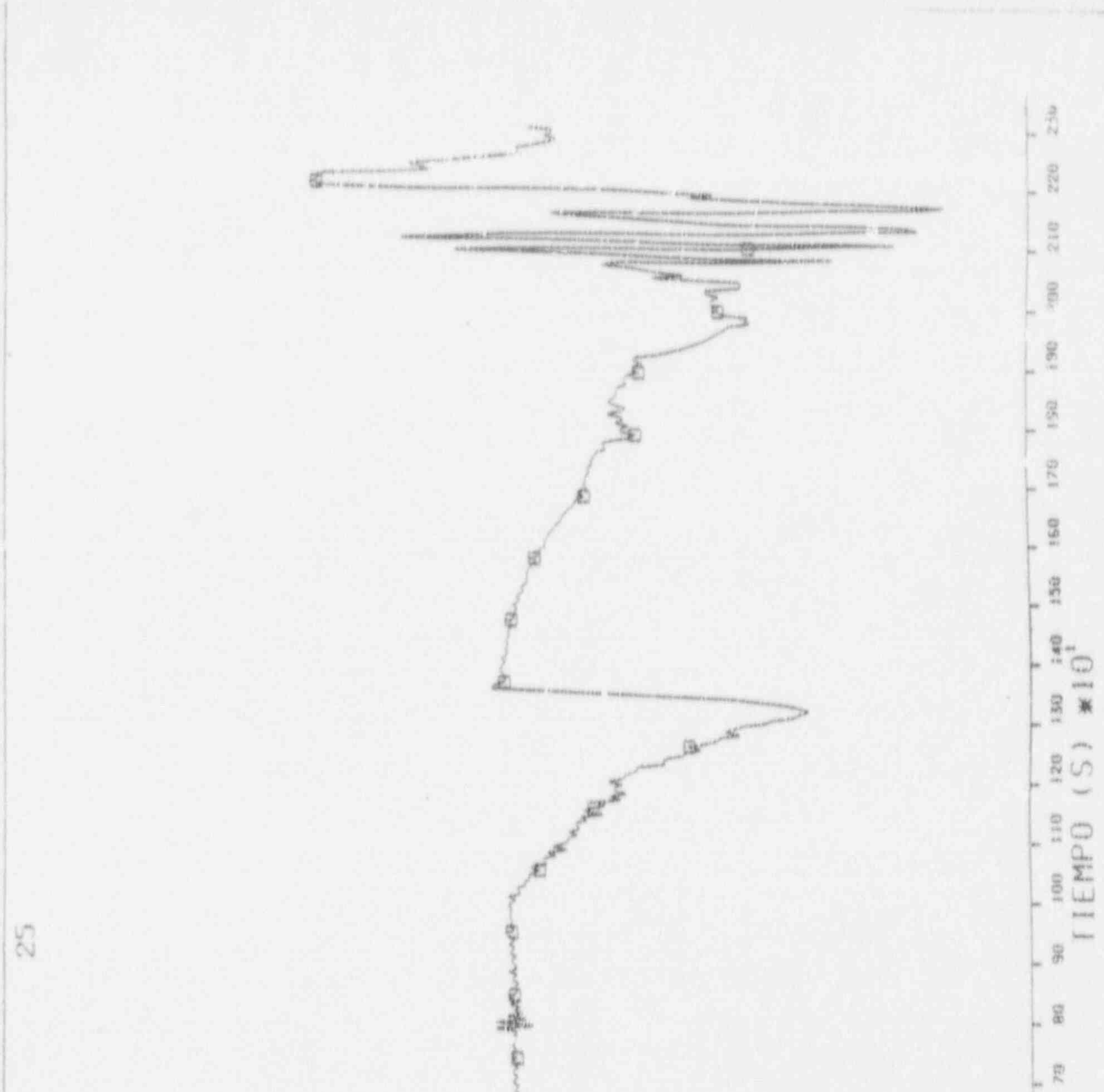
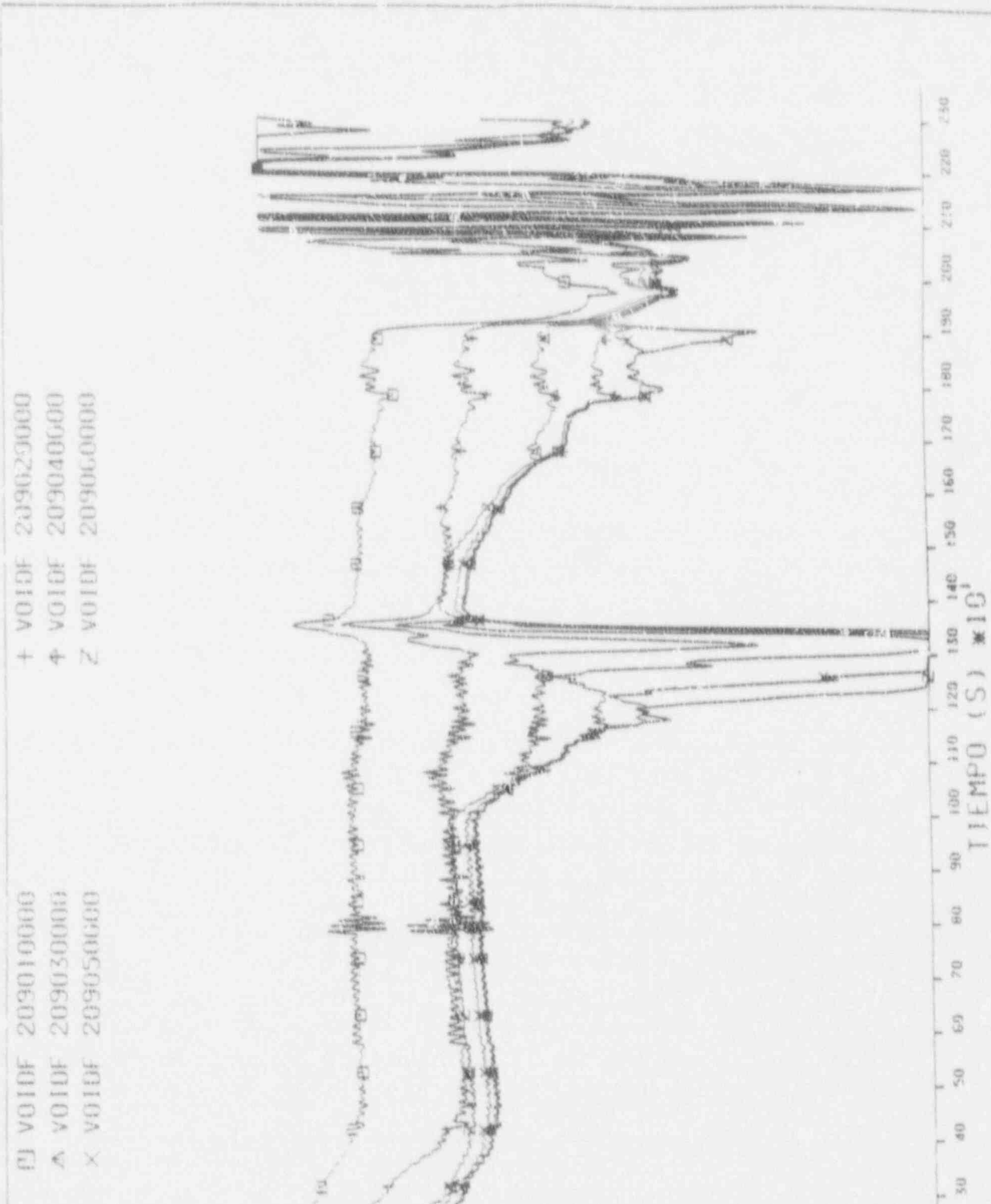

CNTRLVAR 25
NIVEL COLAPSADO EN EL NUCLEO (M)
TIEMPO (S) *10^1
ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 19)

FIG. II (B) - 19



□ VOIDF 209010000
△ VOIDF 209030000
× VOIDF 209050000
+ VOIDF 209020000
◇ VOIDF 209040000
Z VOIDF 209060000
FRACCIONES DE LIQUIDO EN EL NUCLEO (0/1)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 23)

FIG. II (B) - 20


□ RHO 160010000
△ RHO 142010000
+ RHO 165010000
DENSIDADES DEL PRIMARIO (KG/M3)
TIEMPO (S) ×10^1
ROTURA SPZ (2 INCH) CON RECUPERACION (FIG 21)

FIG. II (B) - 21

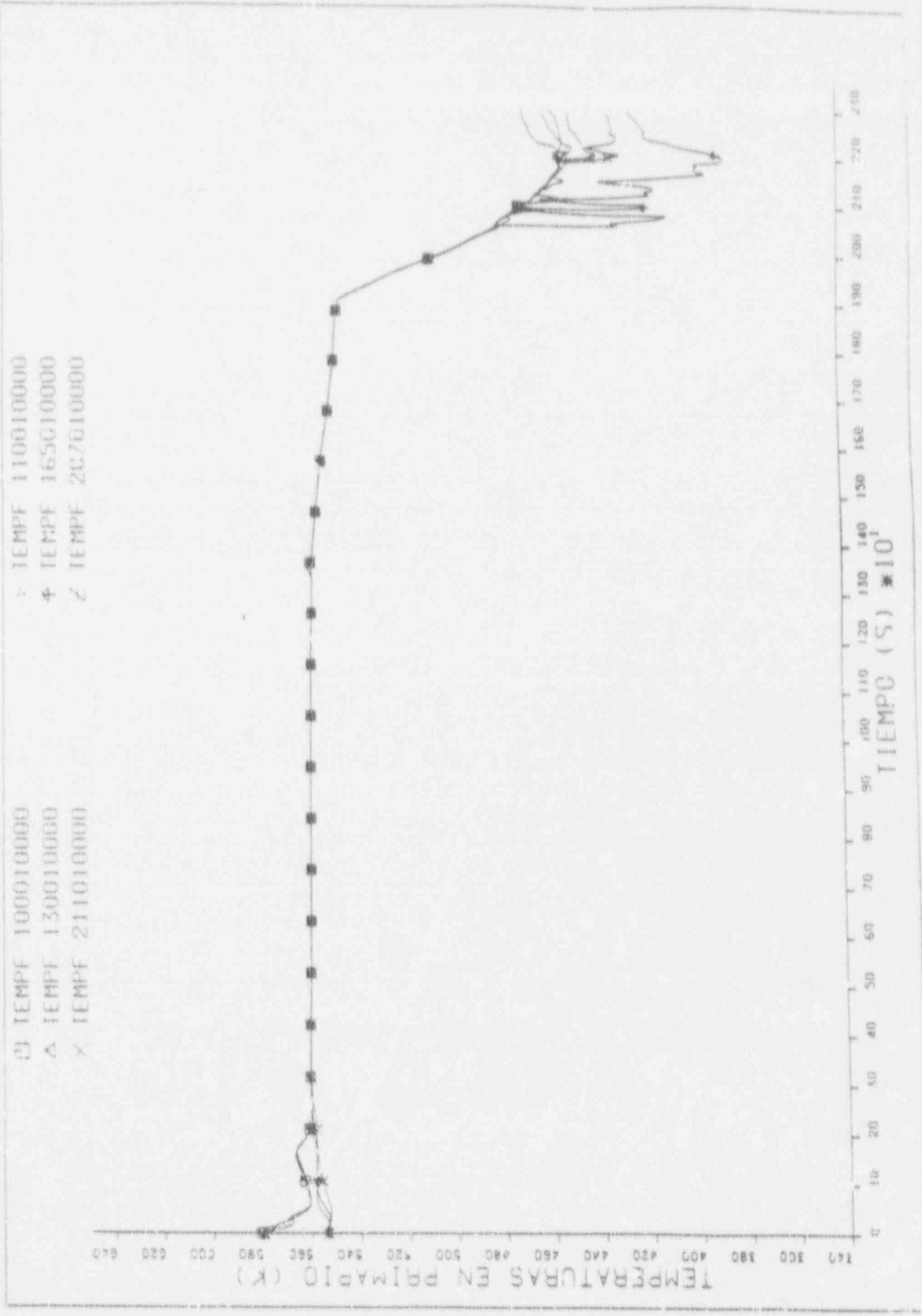

TEMPF 100010000
TEMPF 130010000
TEMPF 211010000
TEMPF 110010000
TEMPF 165010000
TEMPF 207010000
TIEMPO (S) *10^1
TEMPERATURAS EN PRIMARIO (K)
ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 22)

FIG. II (B) - 22

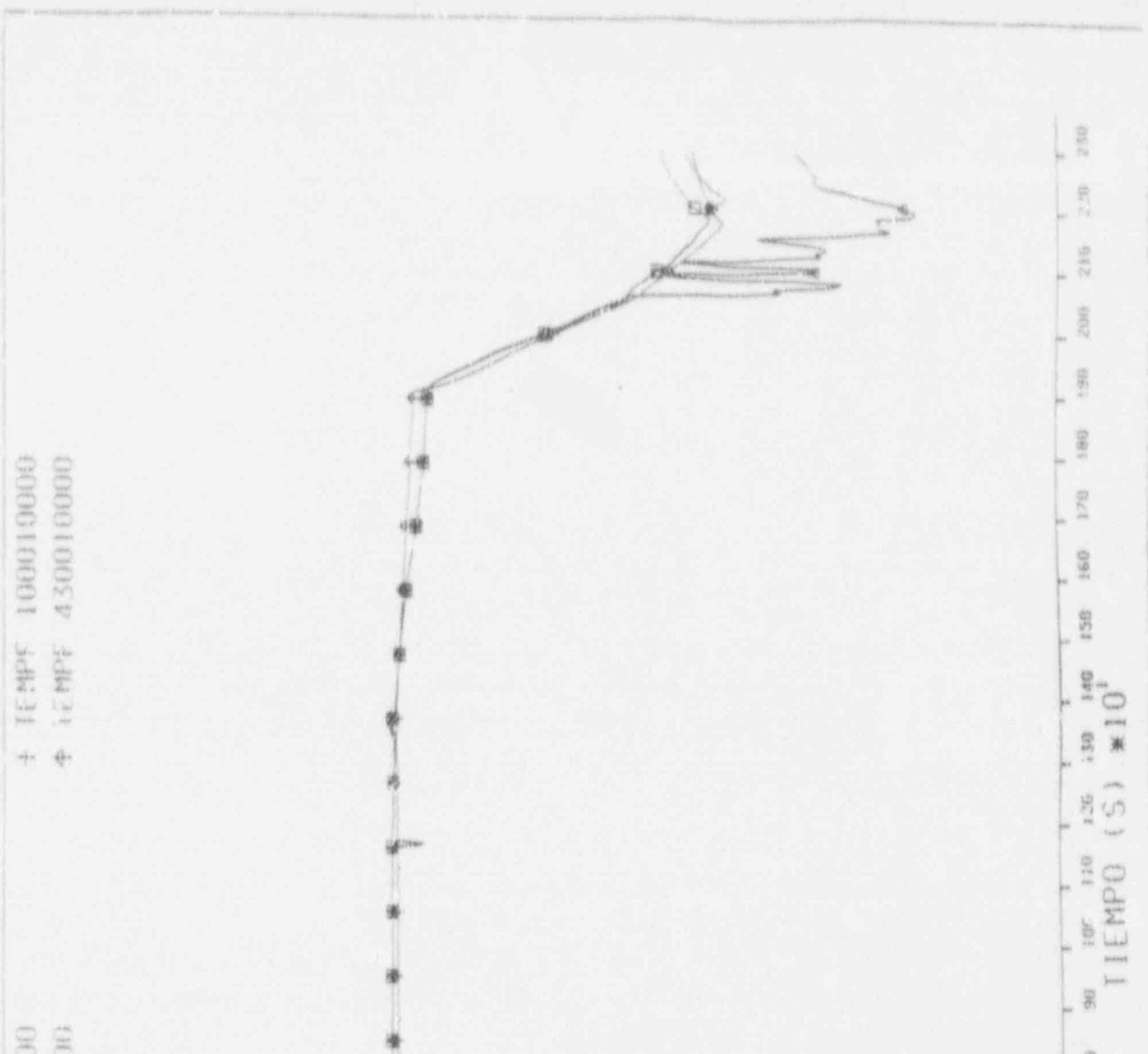
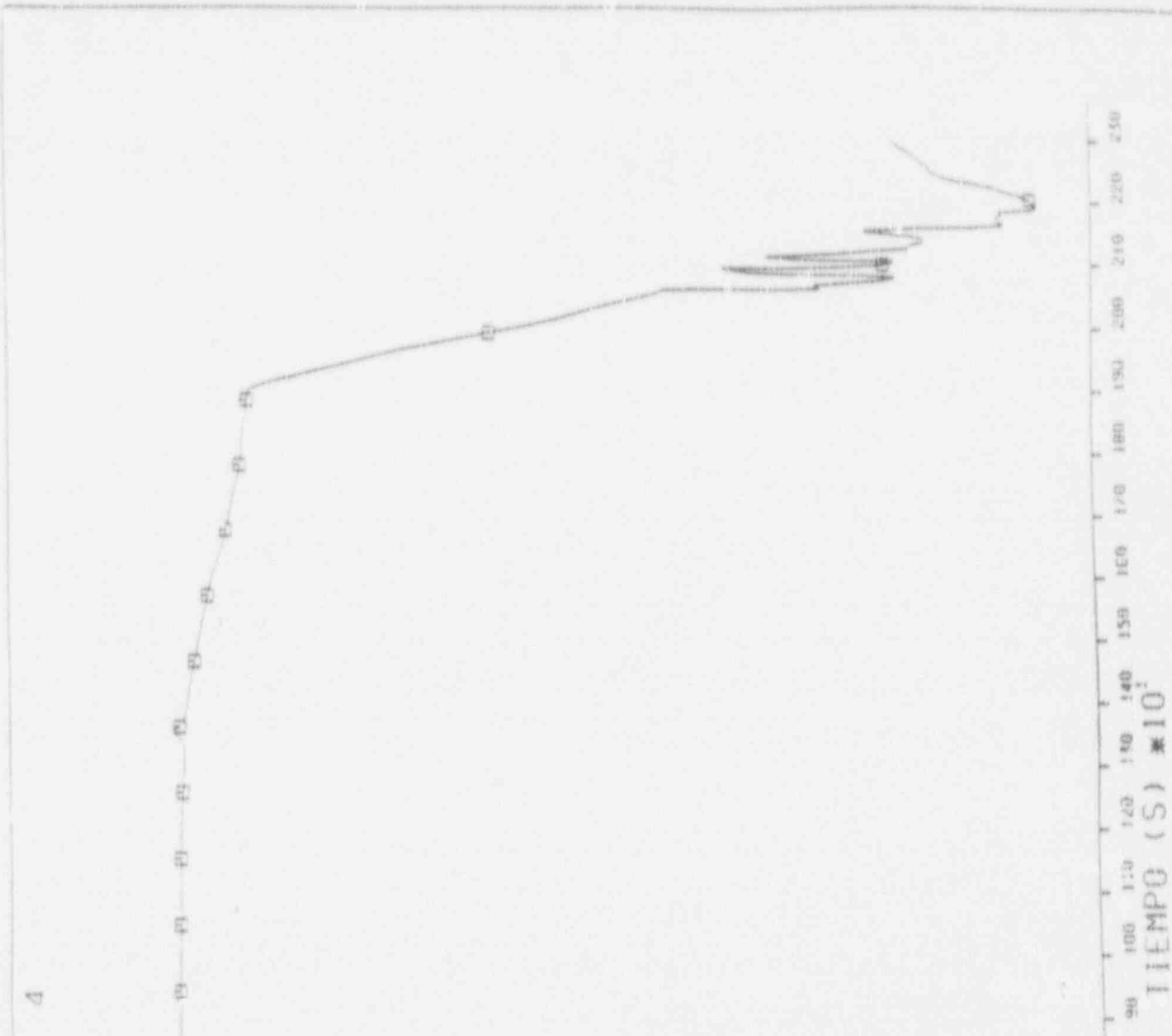

□ TEMPF 312910000
△ TEMPF 165010000
+ TEMPF 100010000
◆ TEMPF 430010000
TIEMPO (S) *10^1
TEMPERATURAS EN PRIMARIO Y G.V. (K)
ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 23)

FIG. II (3) - 23



CNTRLVAR 4
TEMPERATURA MEDIA DEL PRIMARIO (C
TIEMPO (S) *10¹
C.N.J.C. ROTURA SG3 (2 INCH) CON RECUPERACION (FIG 24)

FIG. II (B) - 24

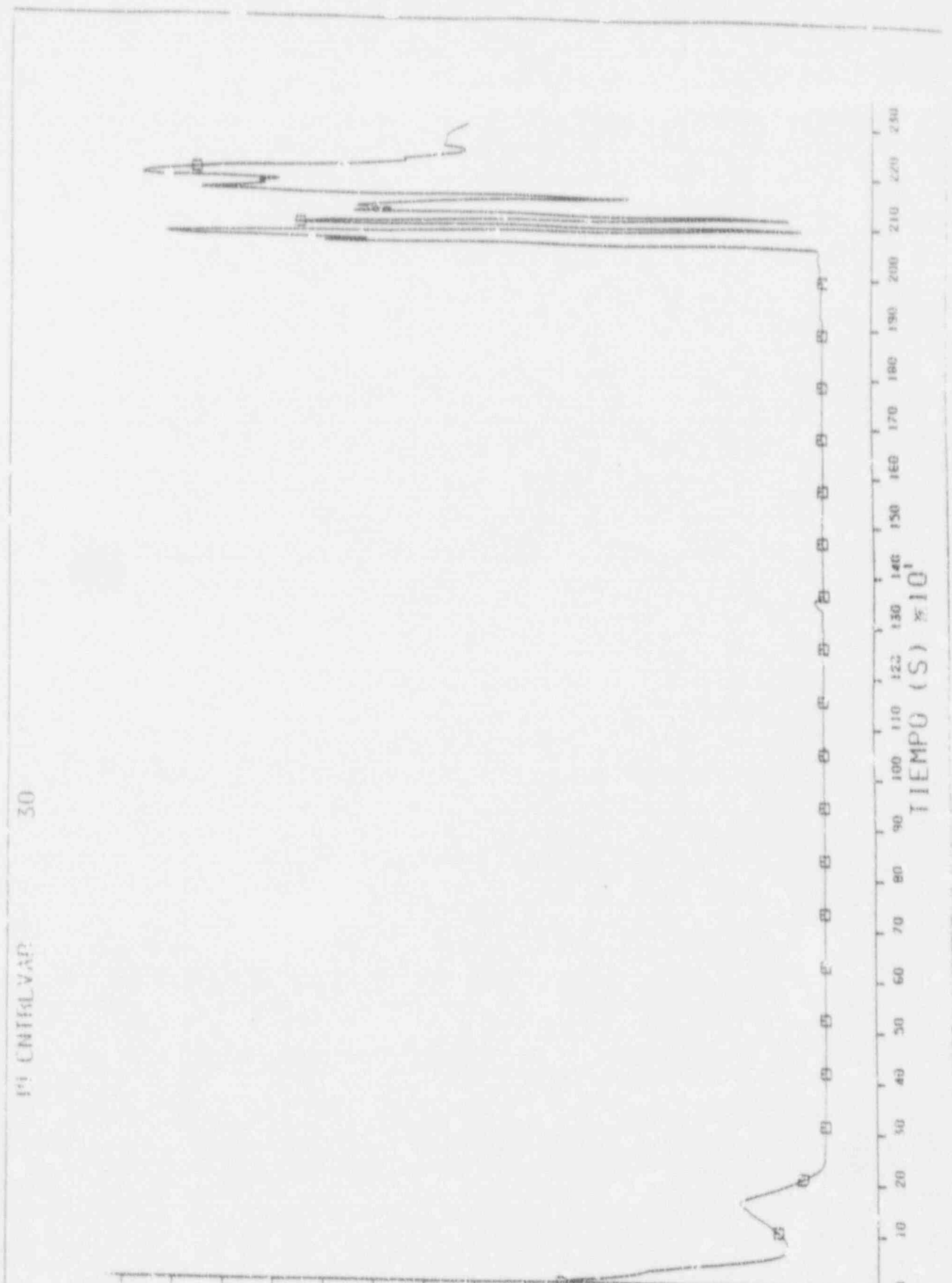

PI CNTRLVAR 30
DELTA TEMPERATURA R.C.S.(C)
TIEMPO (S) x10^1
C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 25)

FIG. II (B) - 25

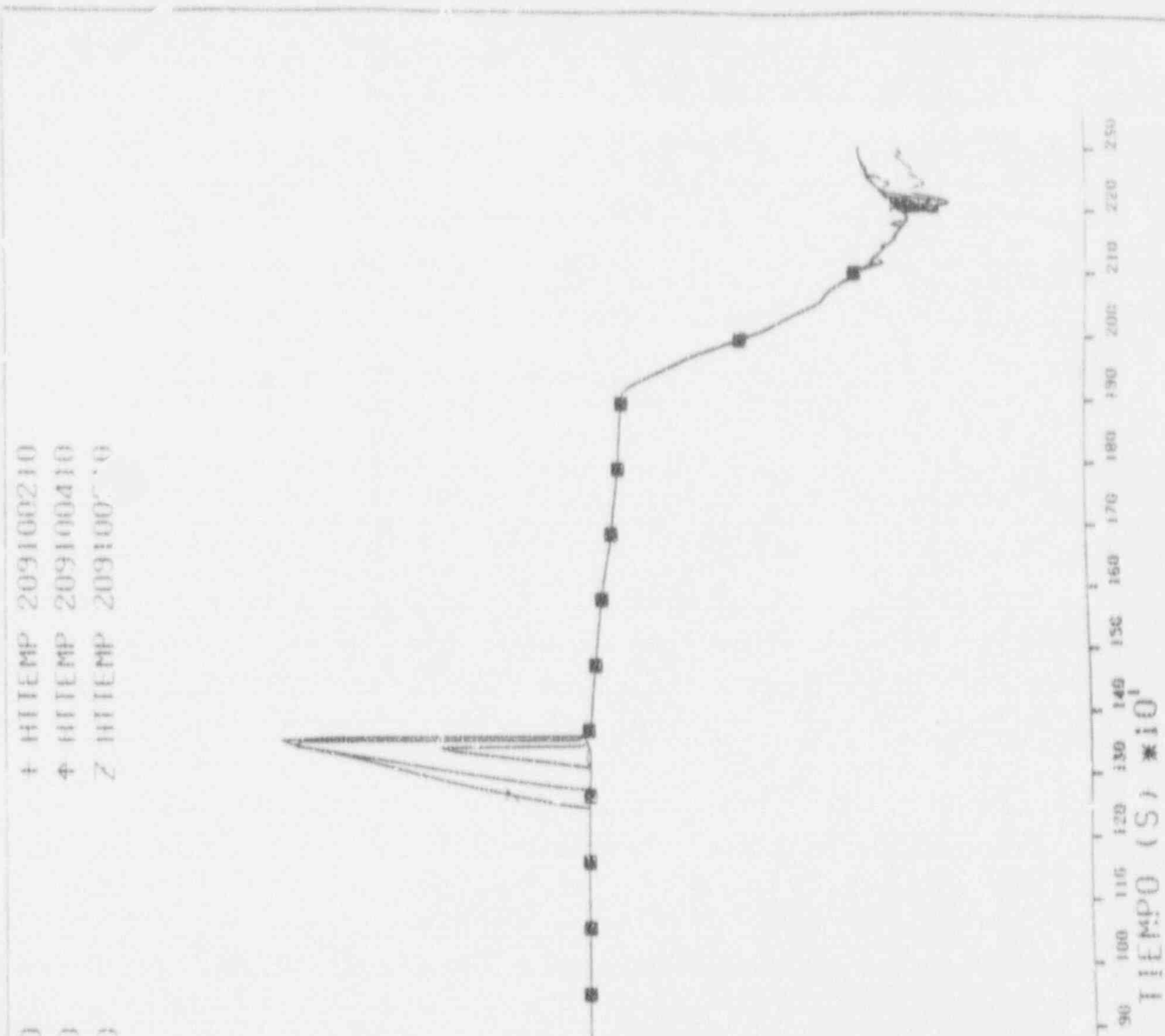

HTTEMP 209100110
HTTEMP 209100310
HTTEMP 209100510
HTTEMP 209100210
HTTEMP 209100410
TEMPERATURA VAINAS CANAL MEDIO (K)
TIEMPO (S) *10^1
C.N.J.C. ROTURA SBS (2 INCH) CON RECUPERACION (FIG 26)

FIG. II (B) - 26

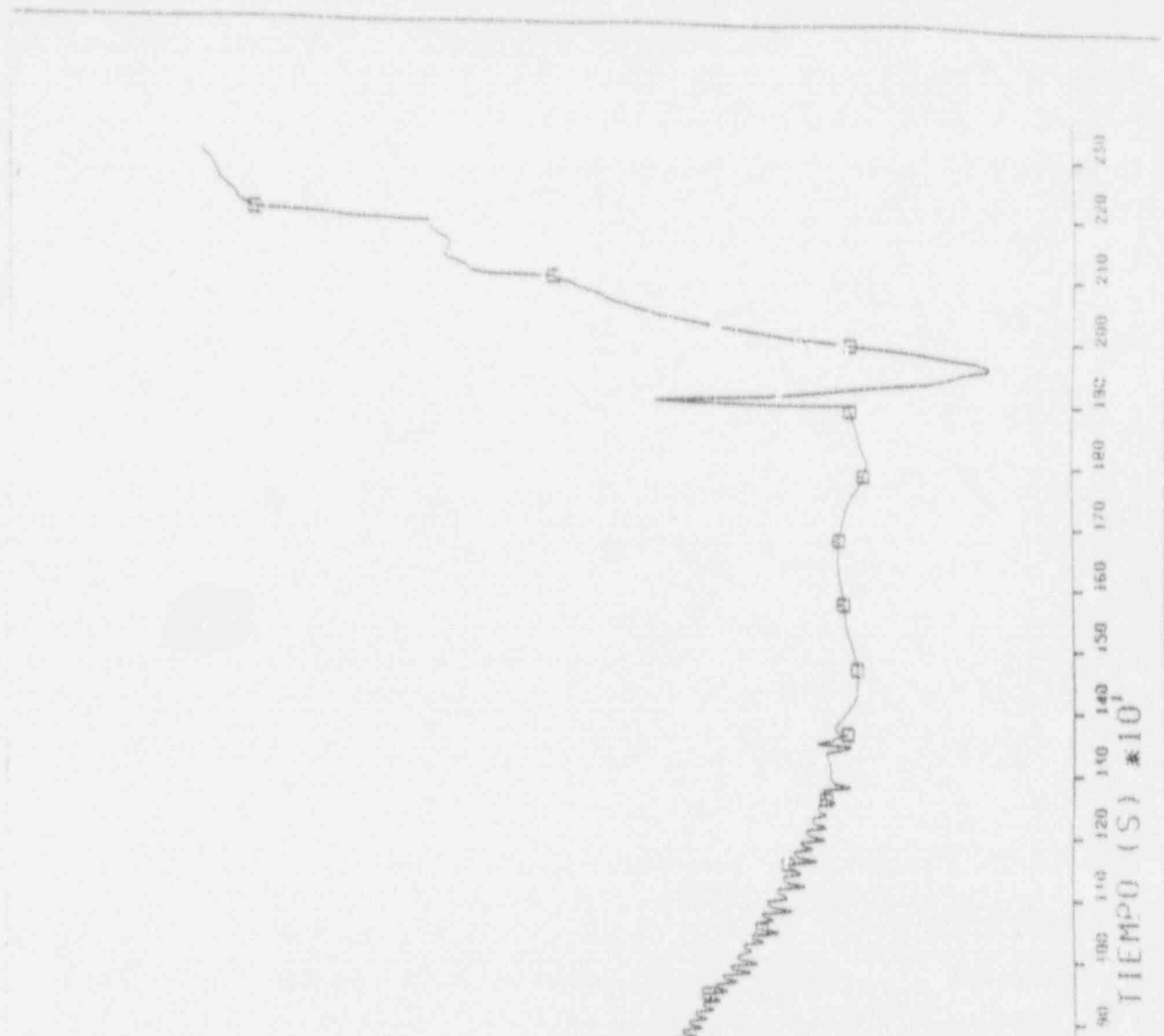
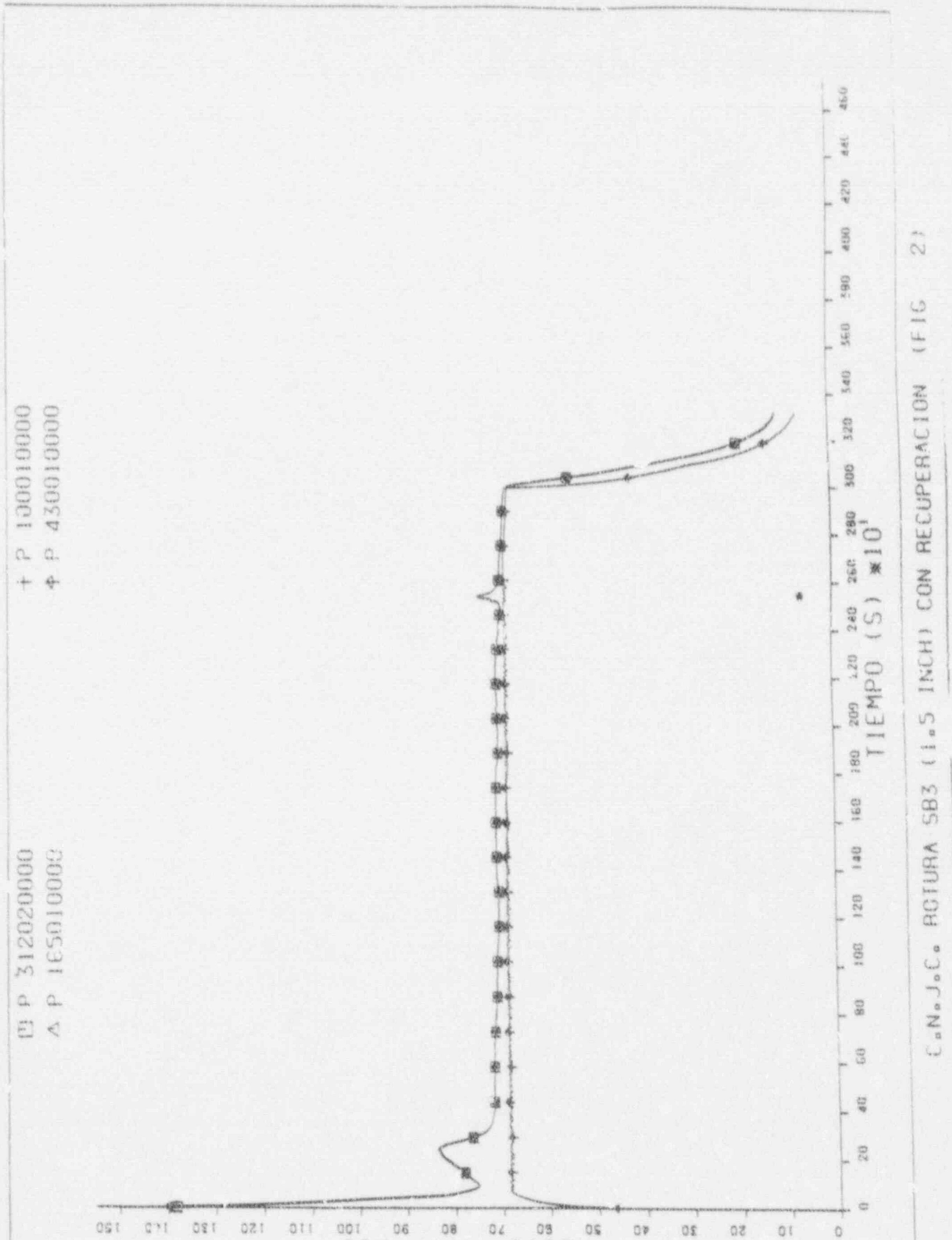

C.N.J.C. ROTURA SB3 (2 INCH) CON RECUPERACION (FIG 27)
TIEMPO (S) *10
ERROR DE MASA (KG)

FIG. II (B) - 27

APPENDIX VI : FIGURES OF CASE III(B), (1.5" WITH RECOVERY)


RKTPOW
POTENCIA DEL REACTOR (W) *10^6
TIEMPO (S) *10^1
ROTURA SB3 (1.5 INCH) CON RECUPERACION

FIG. III (B) - 1



P 312020000
P 165010000
P 100010000
P 430010000
PRESION (PA) *10^5
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 2)

FIG. III (B) - 2

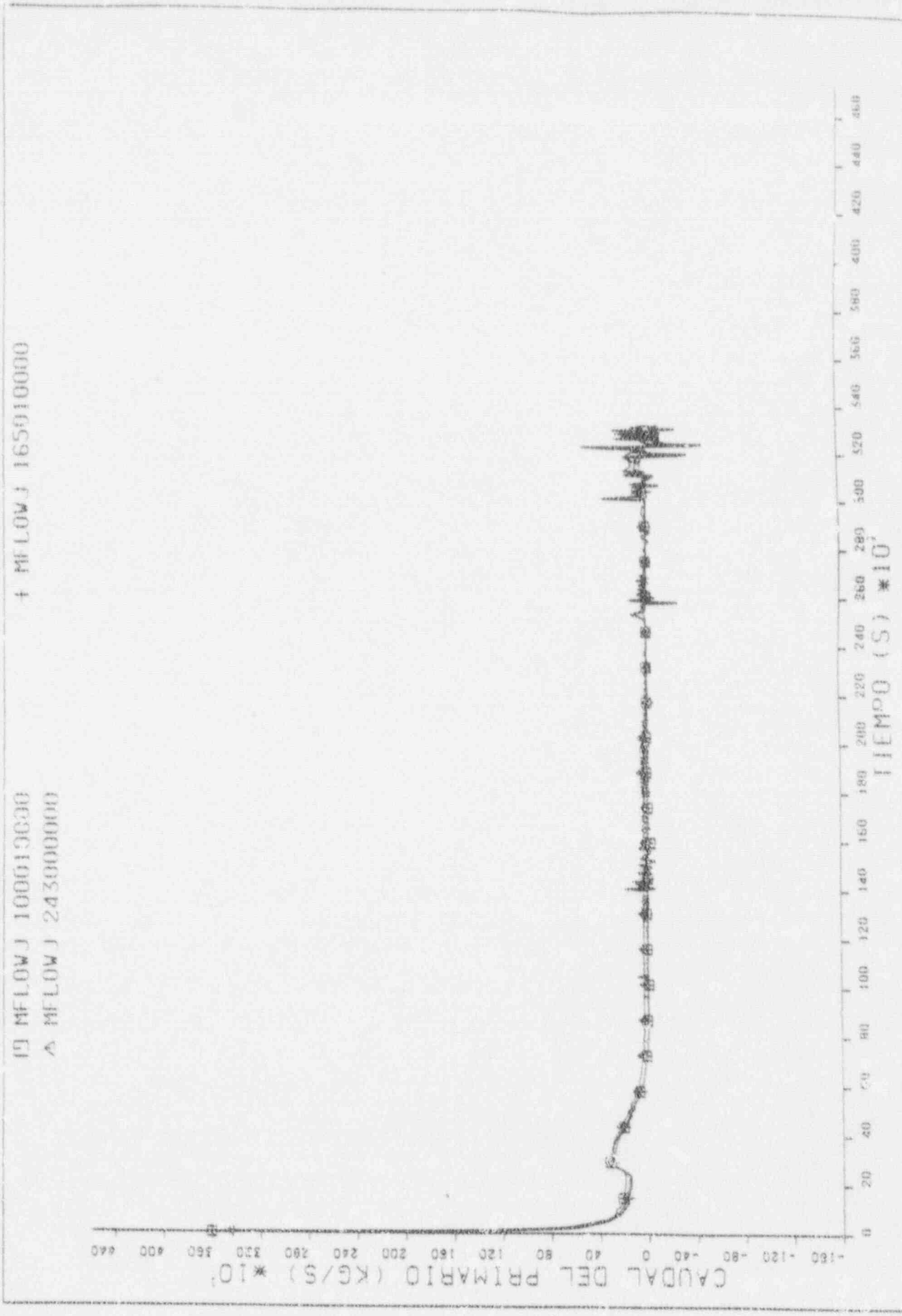

MFLOW J 100010000
MFLOW J 243000000
MFLOW J 165010000
TIEMPO (S) *10'
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 3)
CAUDAL DEL PRIMARIO (KG/S) *10'

FIG. III (B) - 3


MFLOWJ 514000000
CNTRLVAR 38
+ CNTRLVAR 23
CAUDALES DE SECUNDARIO (KG/S)
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 4)

FIG. III (3) - 4

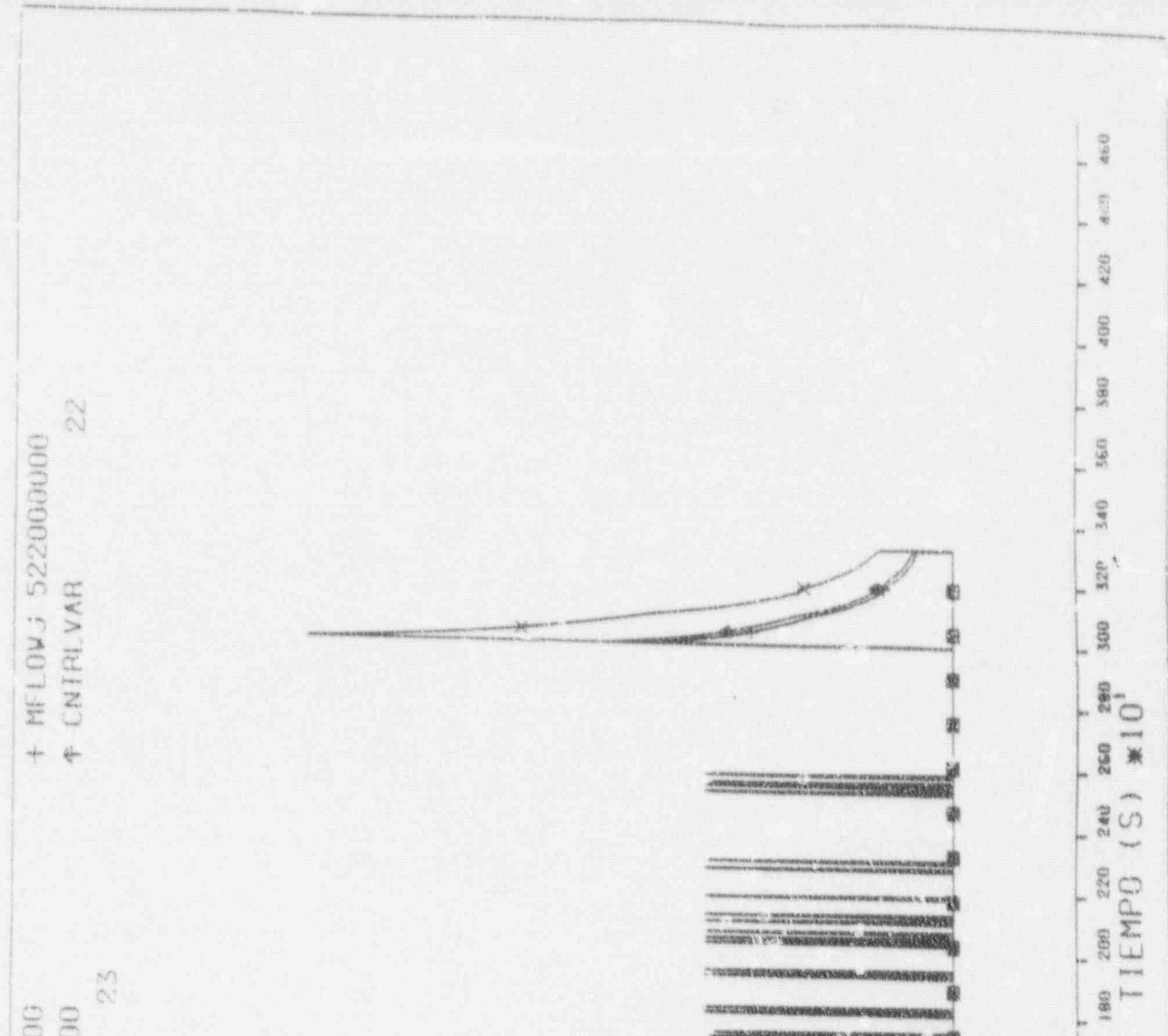
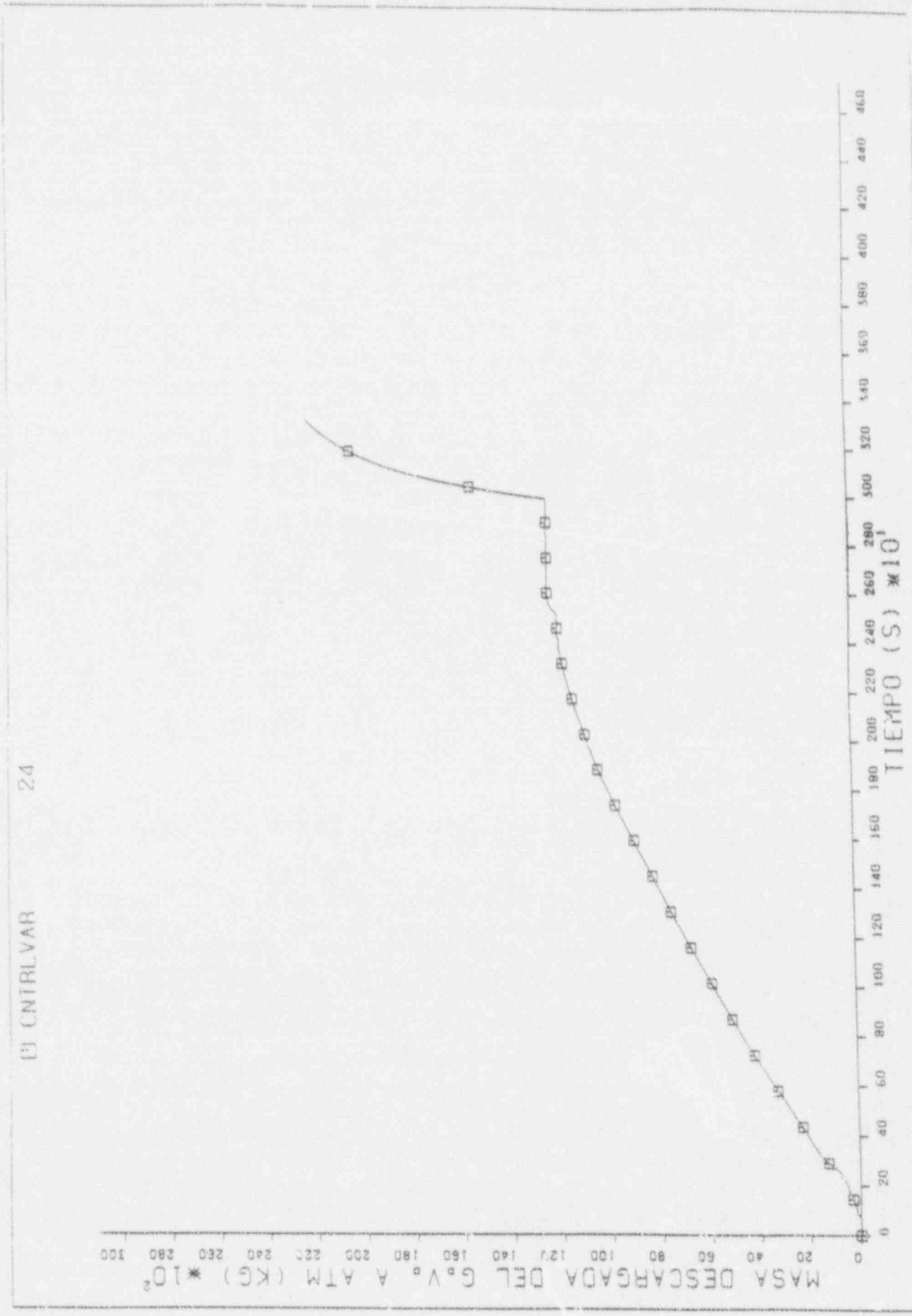

□ MFLOWJ 536000000
△ MFLOWJ 526000000
X CNTRLVAR 23
+ MFLOWJ 522000000
↑ CNTRLVAR 22
ALIVIO Y SEGURIDAD G.V. (KG/S)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 5)

FIG. III (3) - 5



□ CNTRLVAR 24
TIEMPO (S) *10¹
MASA DESCARGADA DEL G.V. A ATM (KG) *10²
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 6)

FIG. III (B) - 6

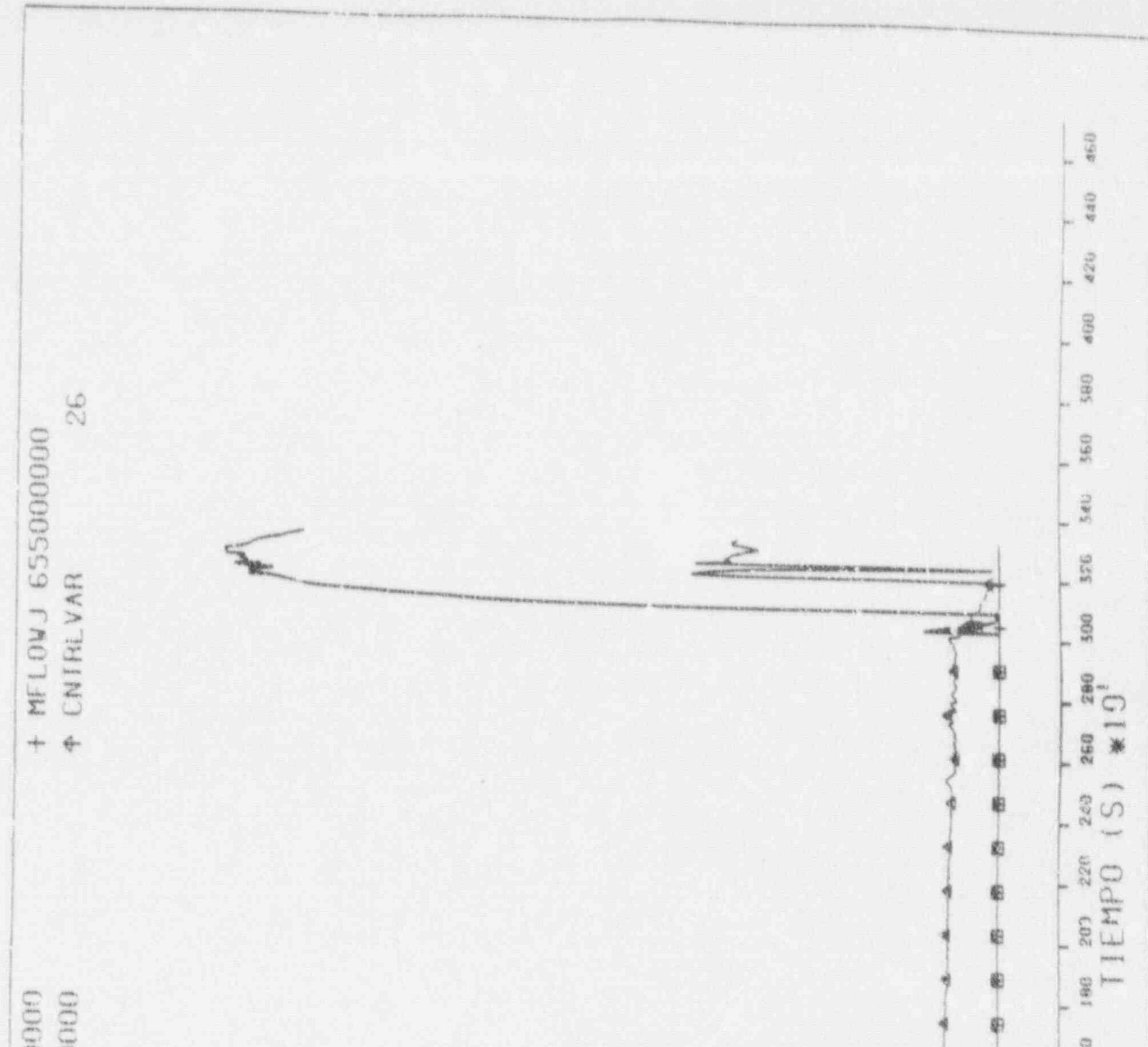

MFLOWJ 630000000
MFLOWJ 163000000
MFLOWJ 655000000
CNTRLVAR 26
CAUDAL DE ROTURA Y ECCS (KG/S)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 7)

FIG. III (B) - 7

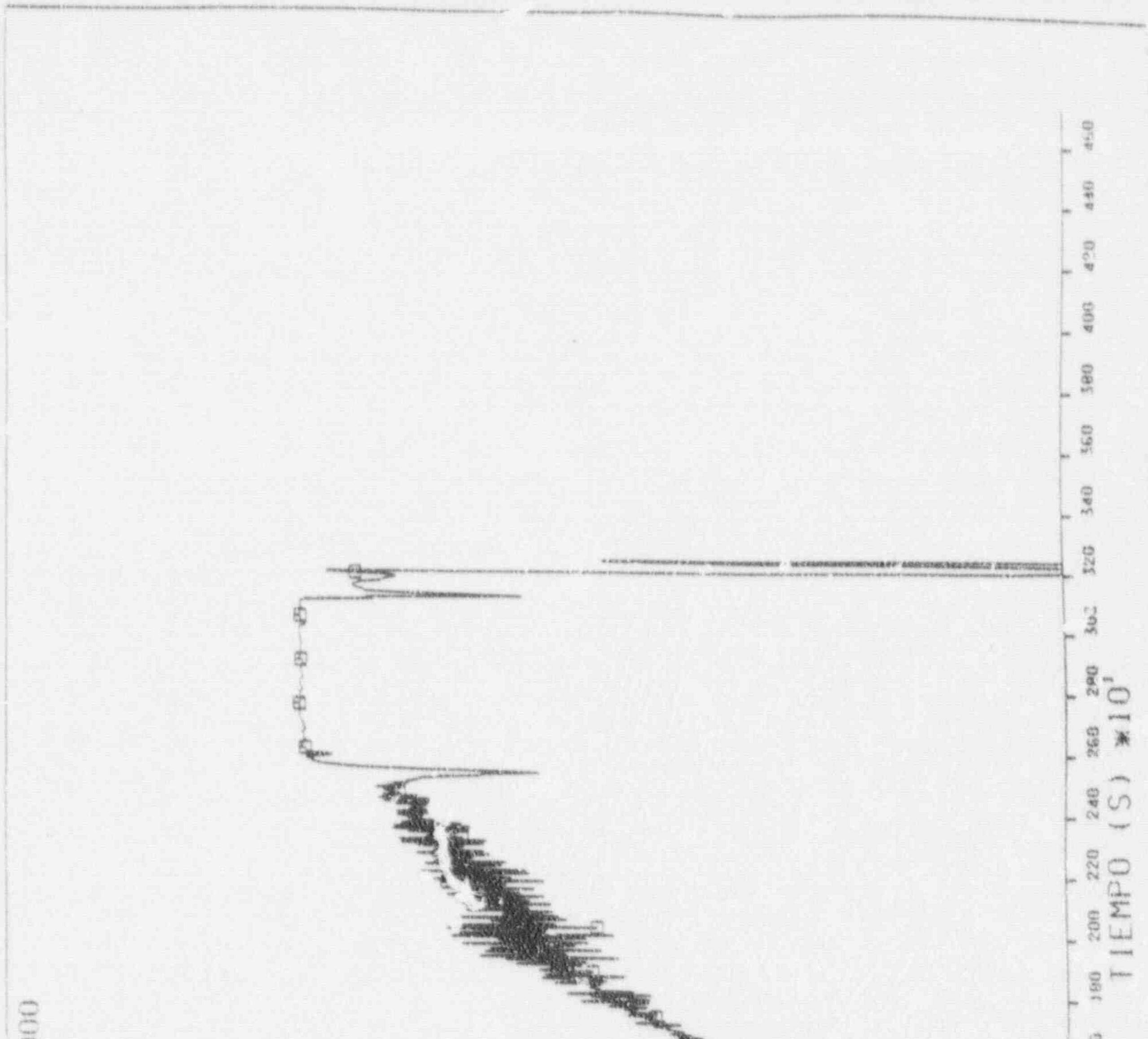
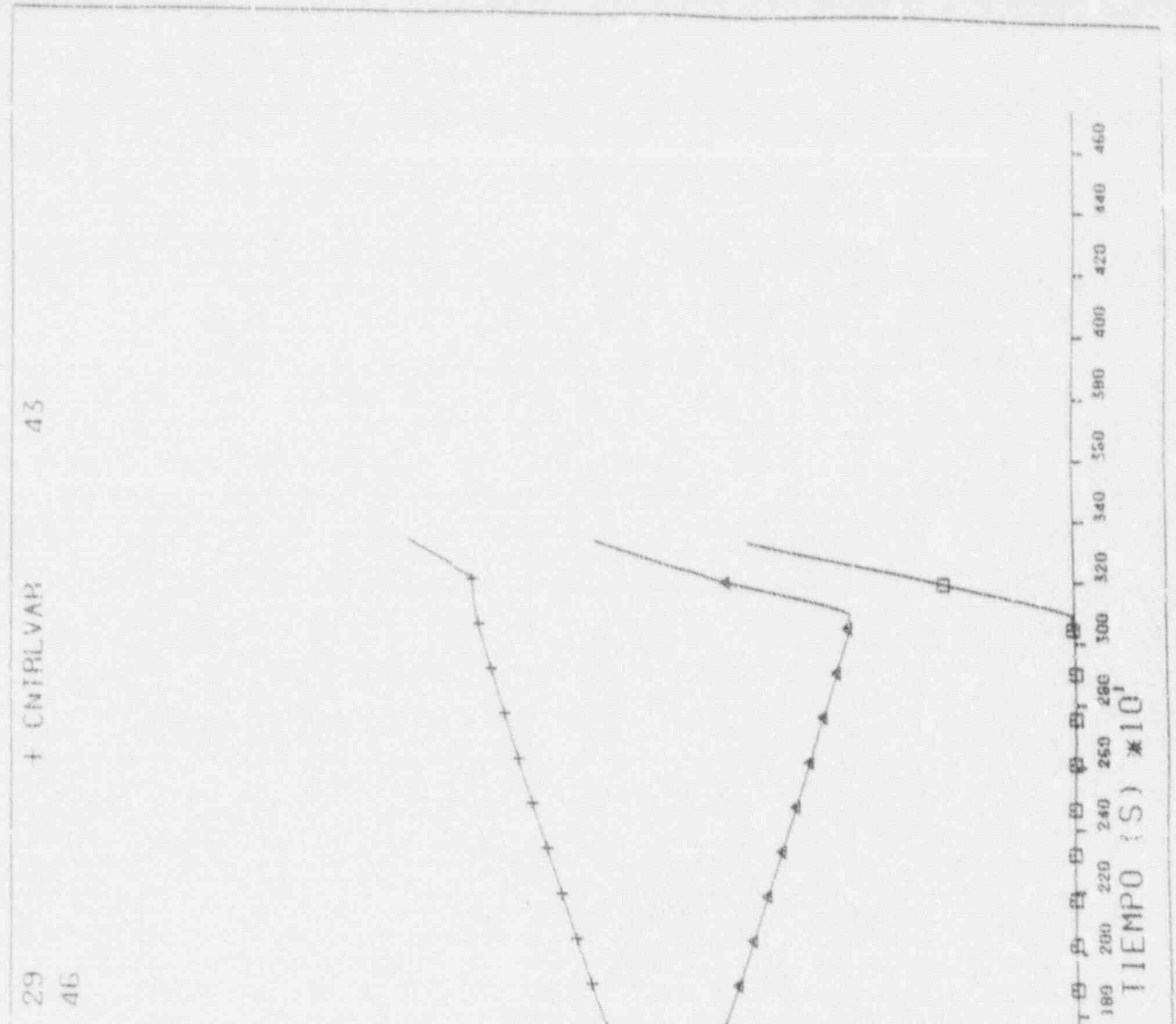
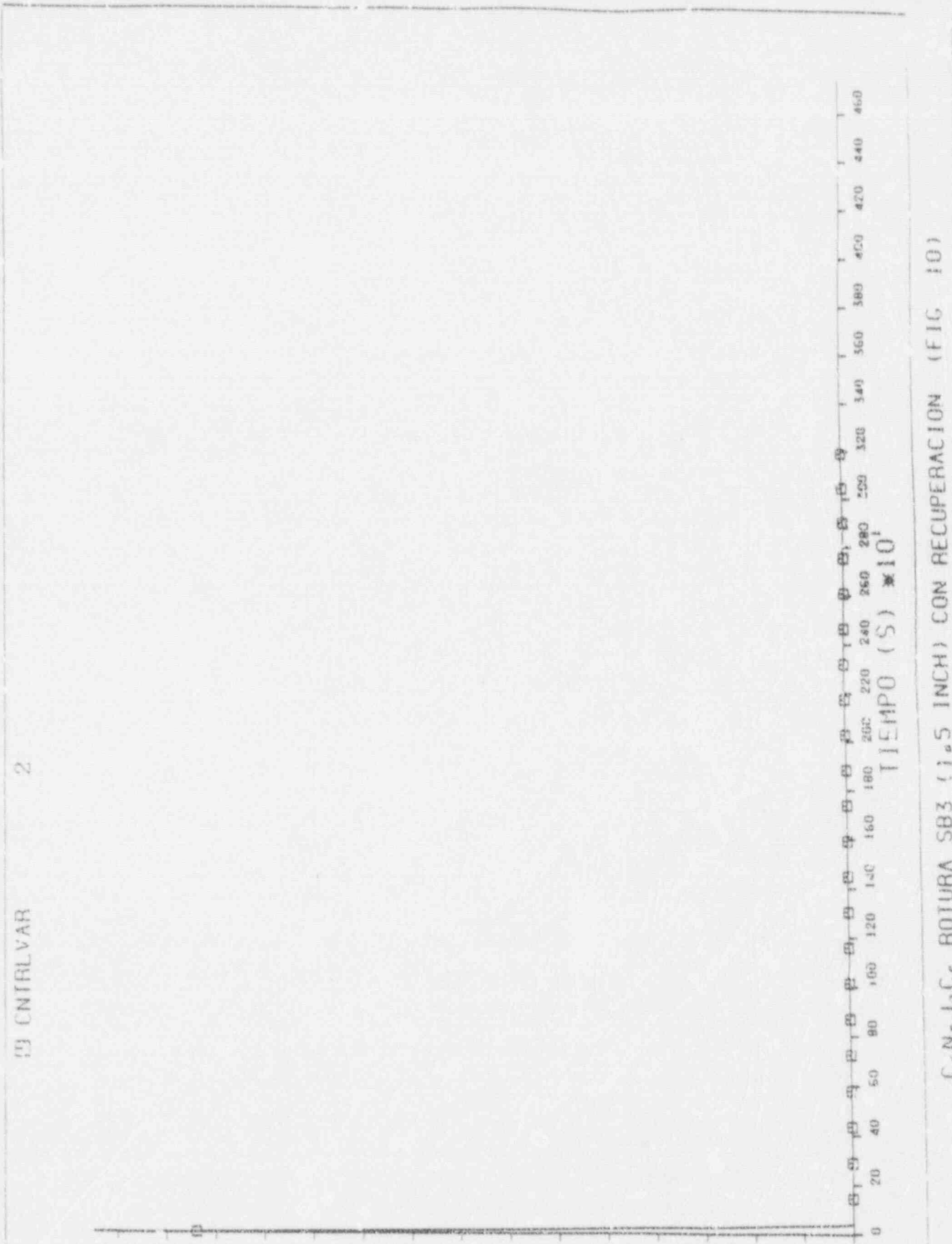

VOIDGJ 163000000
FRACCION DE VAPOR EN LA ROTURA (O/1)
TIEMPO (S) *10
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 8)

FIG. III (B) - 8



□ CNTRLVAR 29
△ CNTRLVAR 46
+ CNTRLVAR 43
BALANCE MASICO DEL PRIMARIO (KG) *10²
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 9)

FIG. III (B) - 9



CNTRLVAR 2
TIEMPO (S) *10[1]
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 10)
NIVEL COLAPSADO EN PRESIONADOR (%)

FIG. III (B) - 10

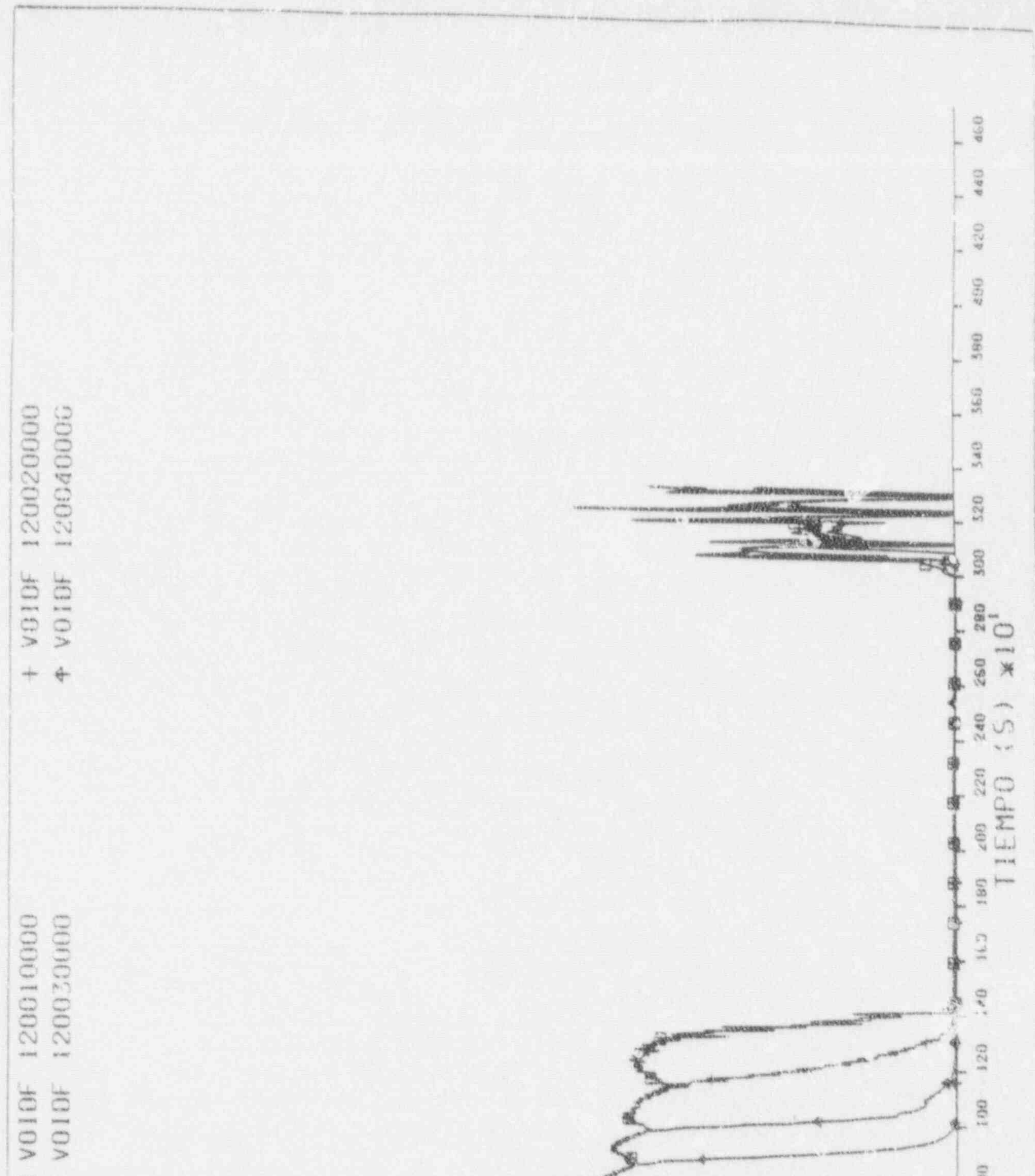

VOIDF 120010000
VOIDF 120020000
VOIDF 120030000
VOIDF 120040000
FRAC. DE LIQUIDO EN TUBO UP G.V. (O/1)
TIEMPO (S) *10¹
ROTURA SB3 (1.5 INCH) CON RECUPERACION

FIG. III (B) - 11

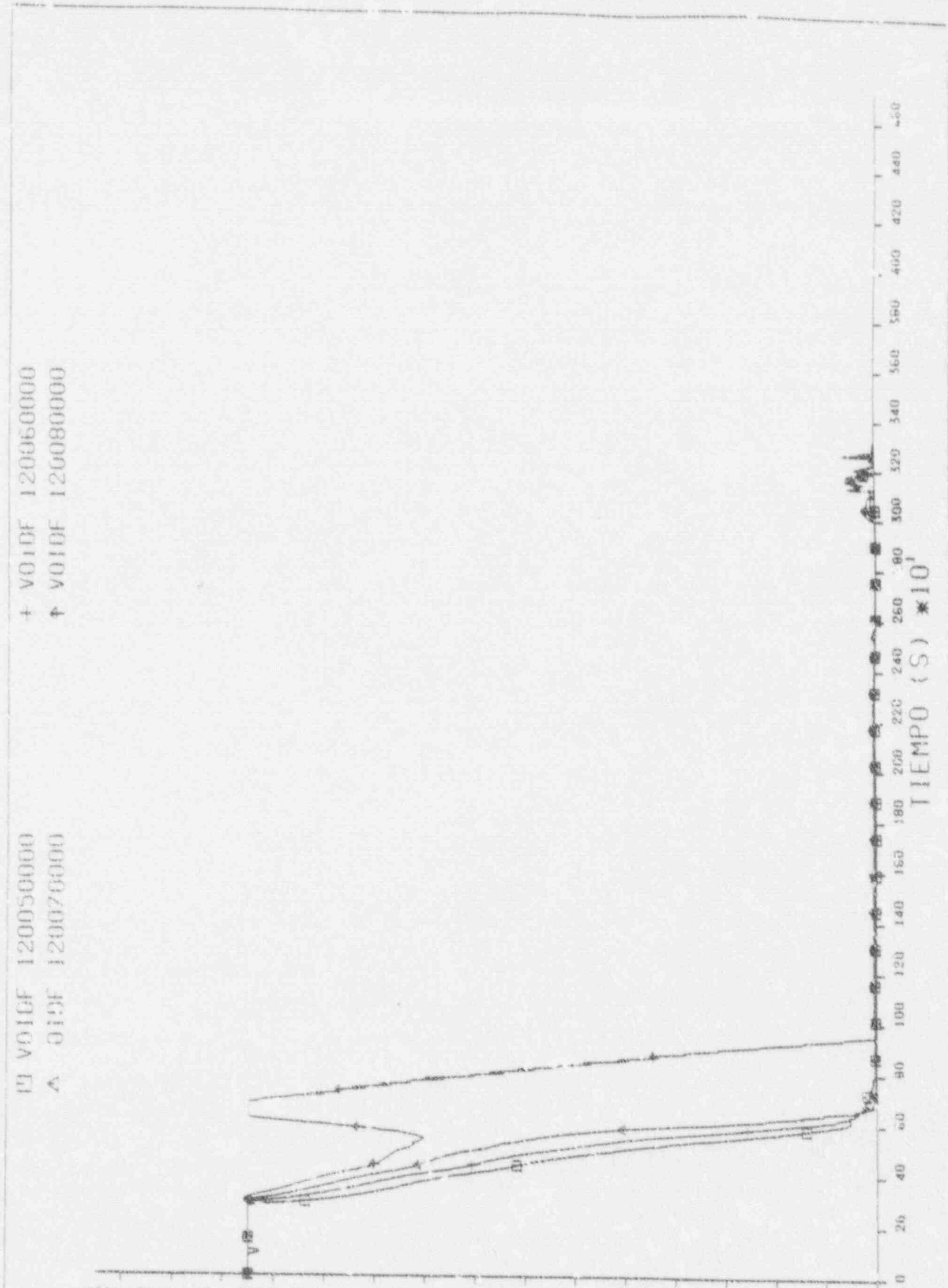

VOIDF 120050000
VOIDF 120060000
VOIDF 120070000
VOIDF 120080000
FRAC. DE LIQUIDO EN TUBO DOWN G.V. (O/1)
TIEMPO (S) *10^1
C.N. J.C. ROTURA SBS (1.5 INCH) CON RECUPERACION (FIG 12)

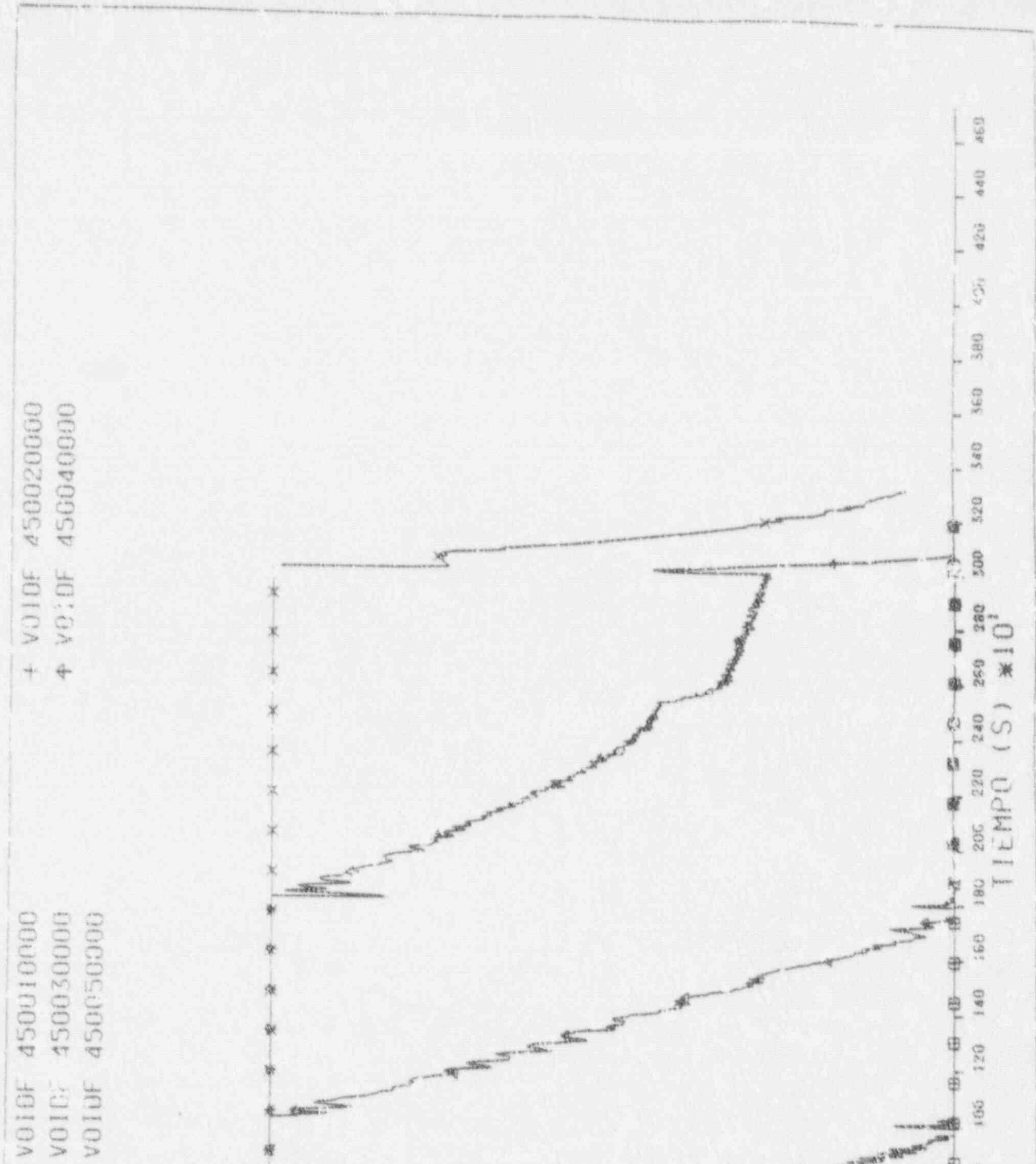
FIG. III (B) - 12


CNTRLVAR 3
NIVEL EN DOWNCOMER DEL G.V. ( )
TIEMPO (S) *10
ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 13)

FIG. III (B) - 13


□ VOIDF 400010000
△ VOIDF 400030000
× VOIDF 400050000
+ VOIDF 430020000
◇ VOIDF 400040000
FRAC. DE LIQUIDO EN RISER G.V. (O/1)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 14)

FIG. III (B) - 14



VOIDF 450010000
VOIDF 450030000
VOIDF 450050000
VOIDF 450020000
VOIDF 450040000
FRAC. DE LIQUIDO EN DOWNCOMER G.V. (O/1)
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 15)

FIG. III (B) - 15

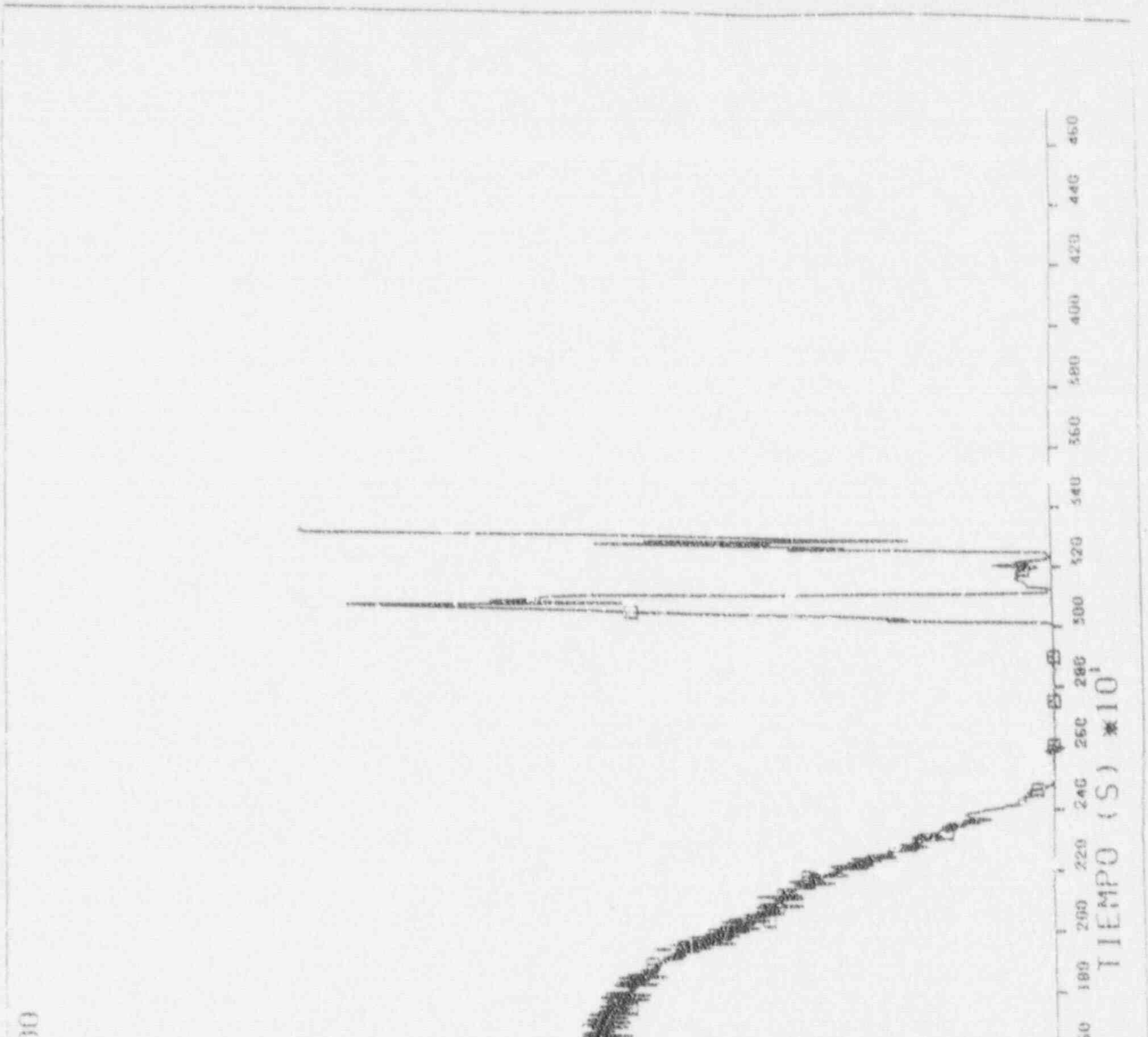

VOIDF 140010000
NIVEL COLAPSADO EN SELLO -140-
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 16)

FIG. III (3) - 16

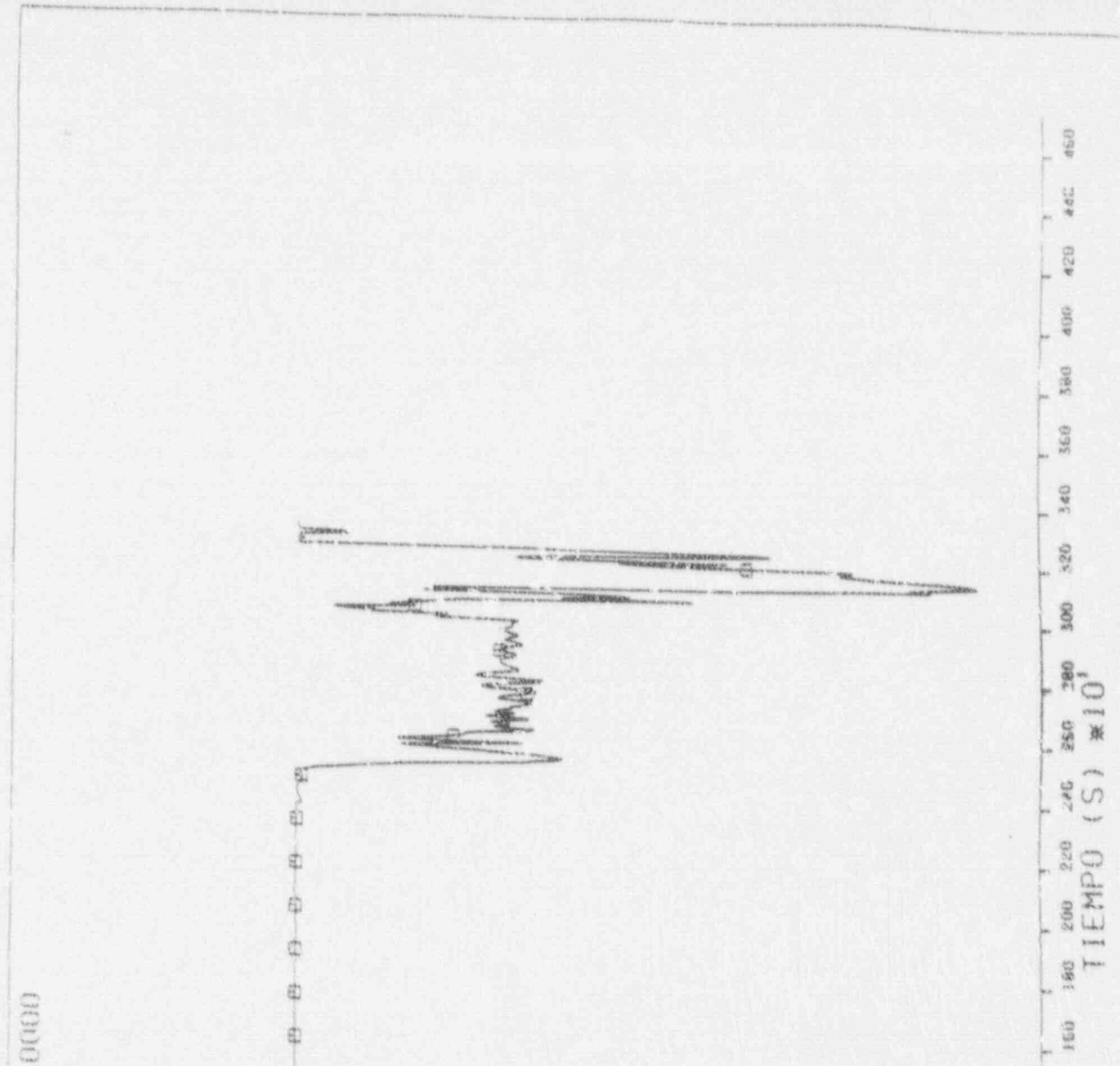
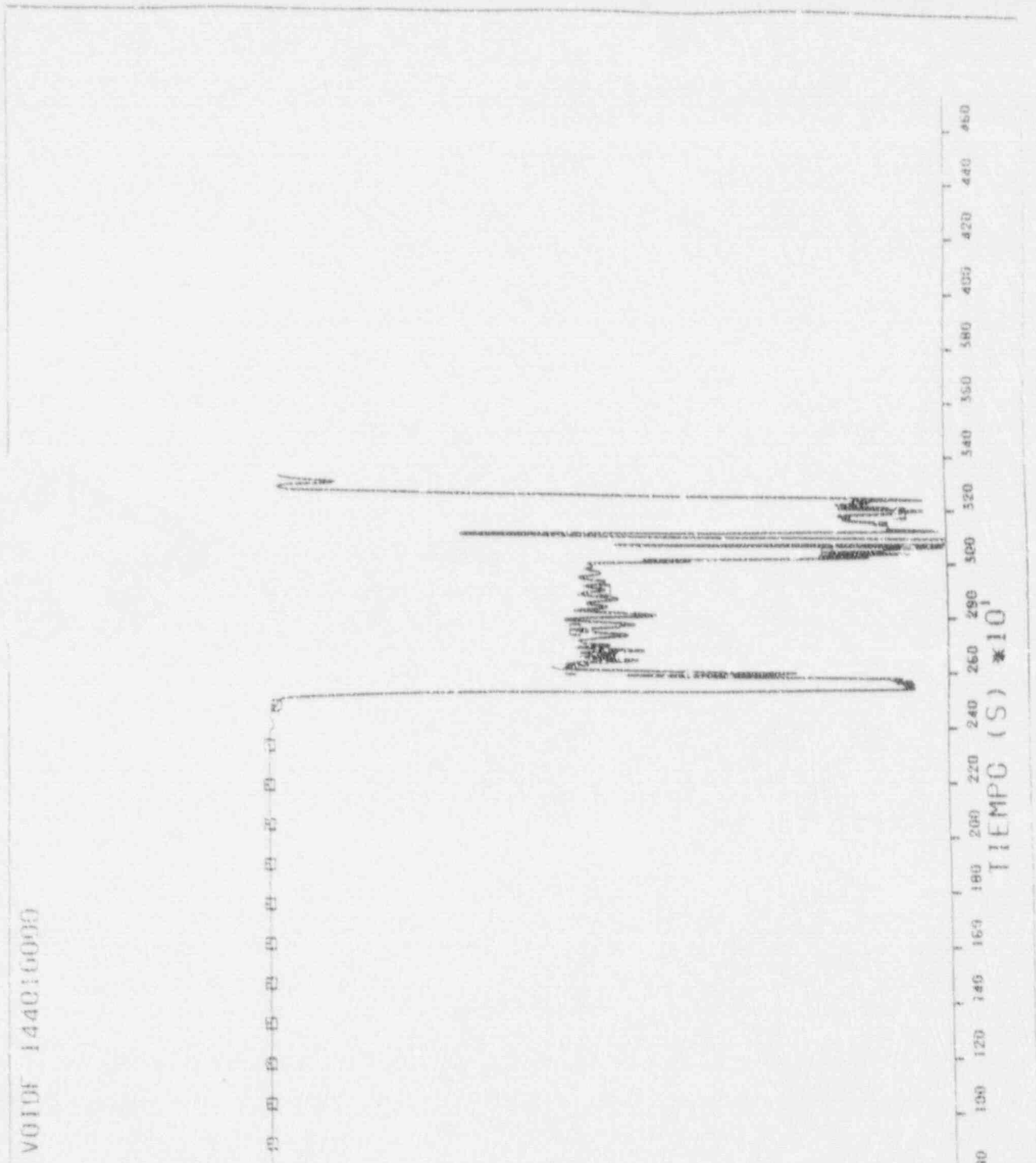

VOIDF 142010000
NIVEL COLAPSADO EN SELLO -142-
TIEMPO (S) *10¹
ROTURA SRI (1.5 INCH) CON RECUPERACION (FIG 17)

FIG. III (B) - 17



13 VOIDF 144010000
NIVEL COLAPSADO EN SELLO -144-
TIEMPO (S) *10^1
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 18)

FIG. III (3) - 18

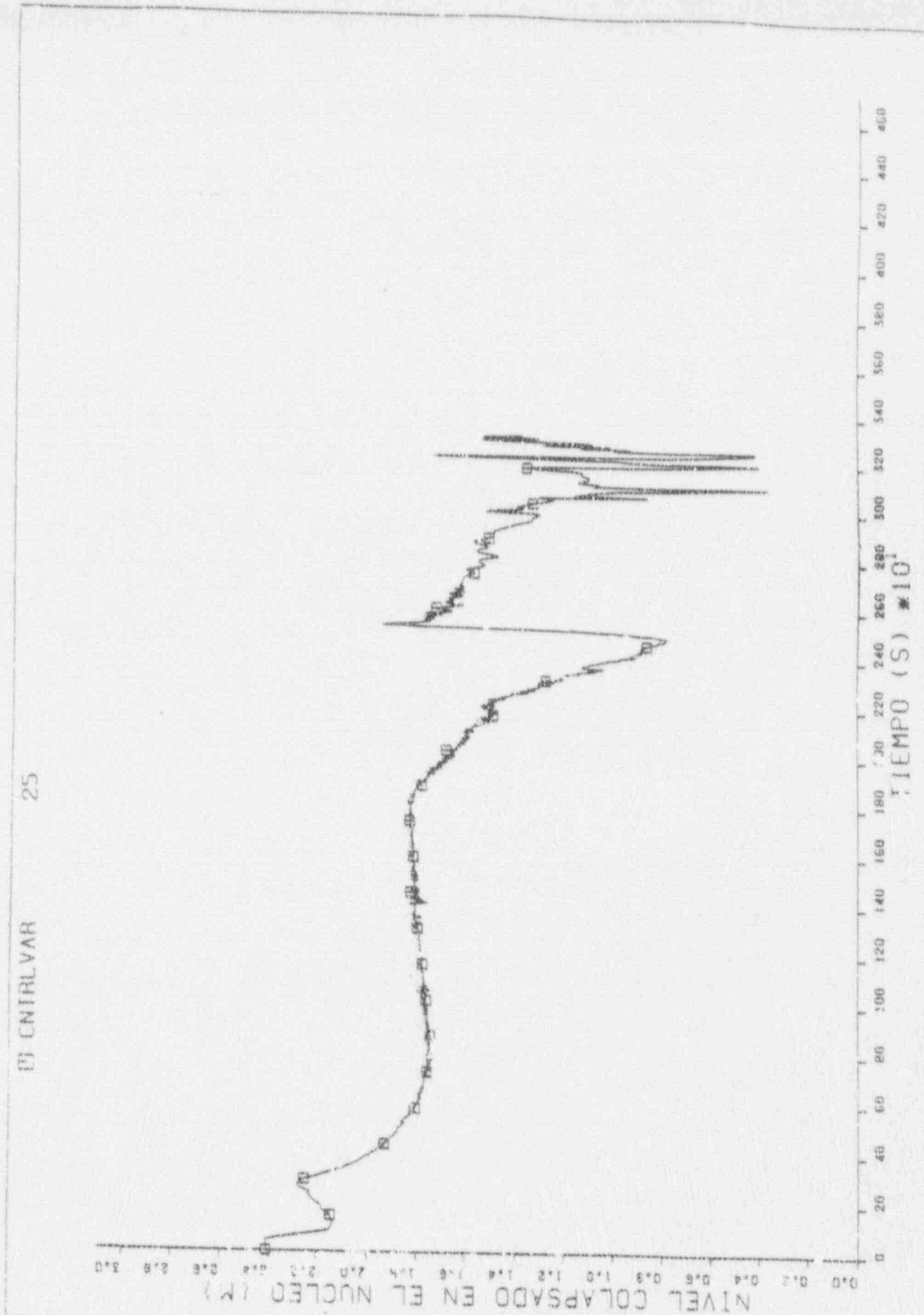
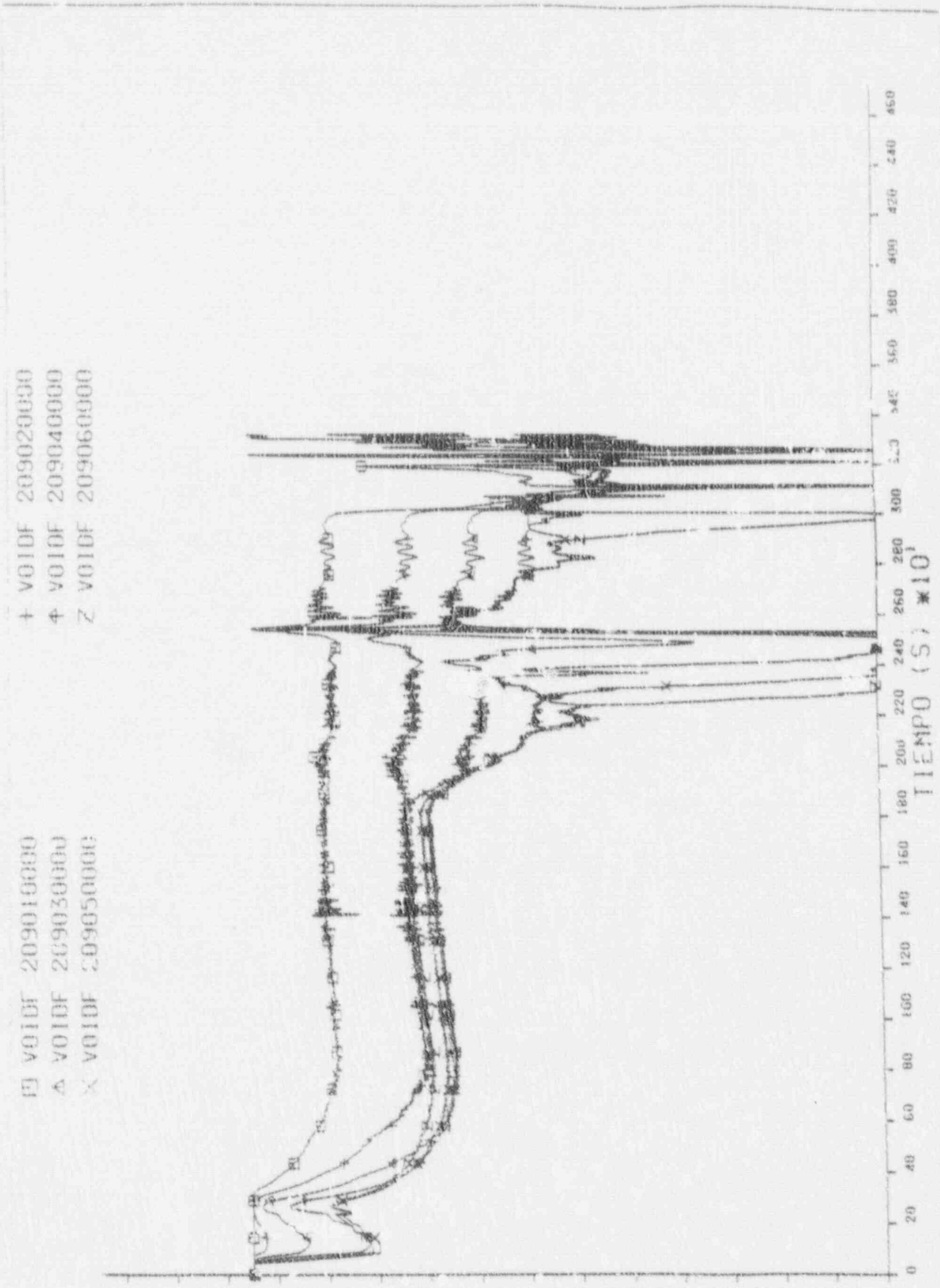
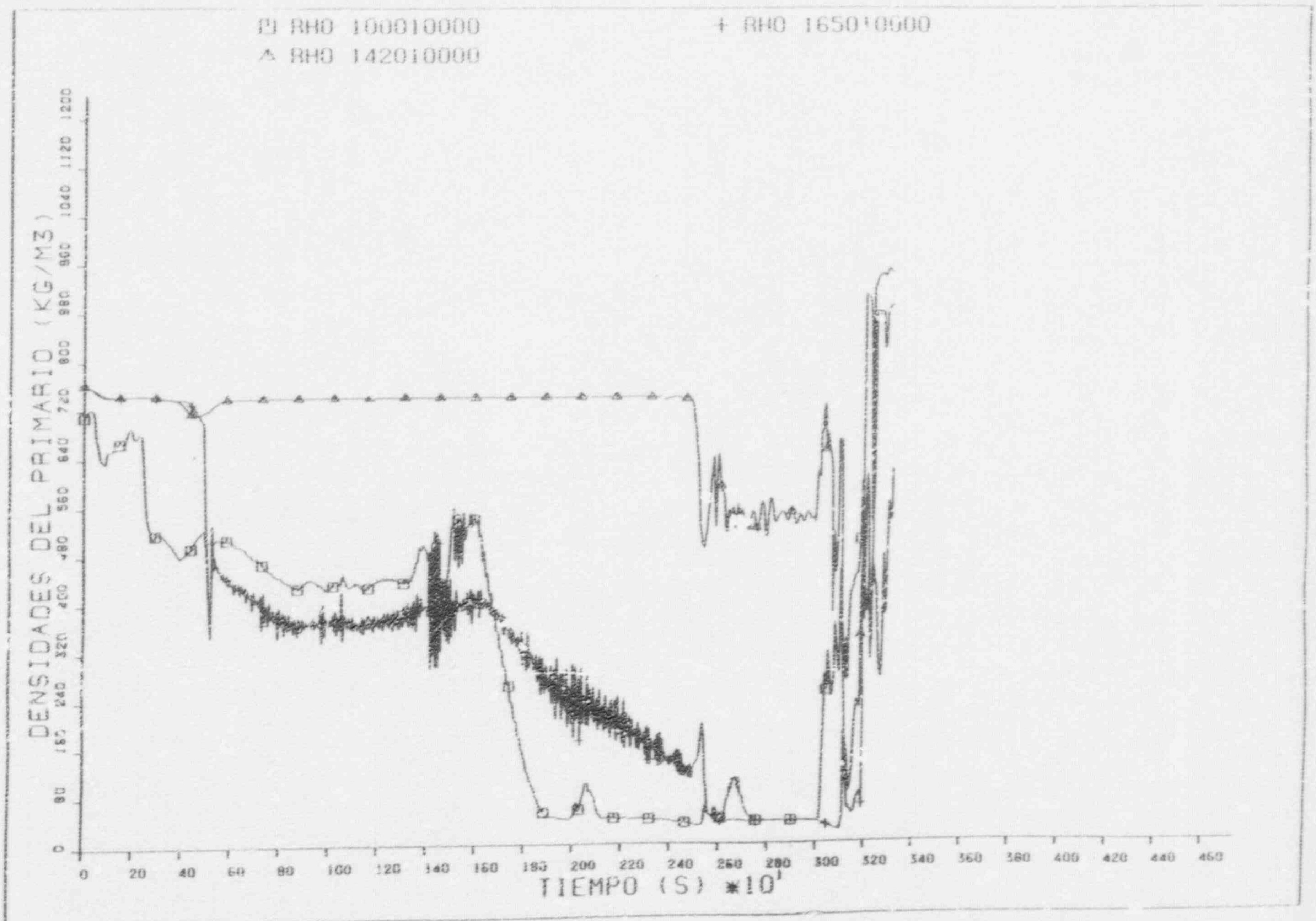

CNTRLVAR 25
ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 19)
TIEMPO (S) *10
NIVEL COLAPSADO EN EL NUCLEO (M)

FIG. III (B) - 19



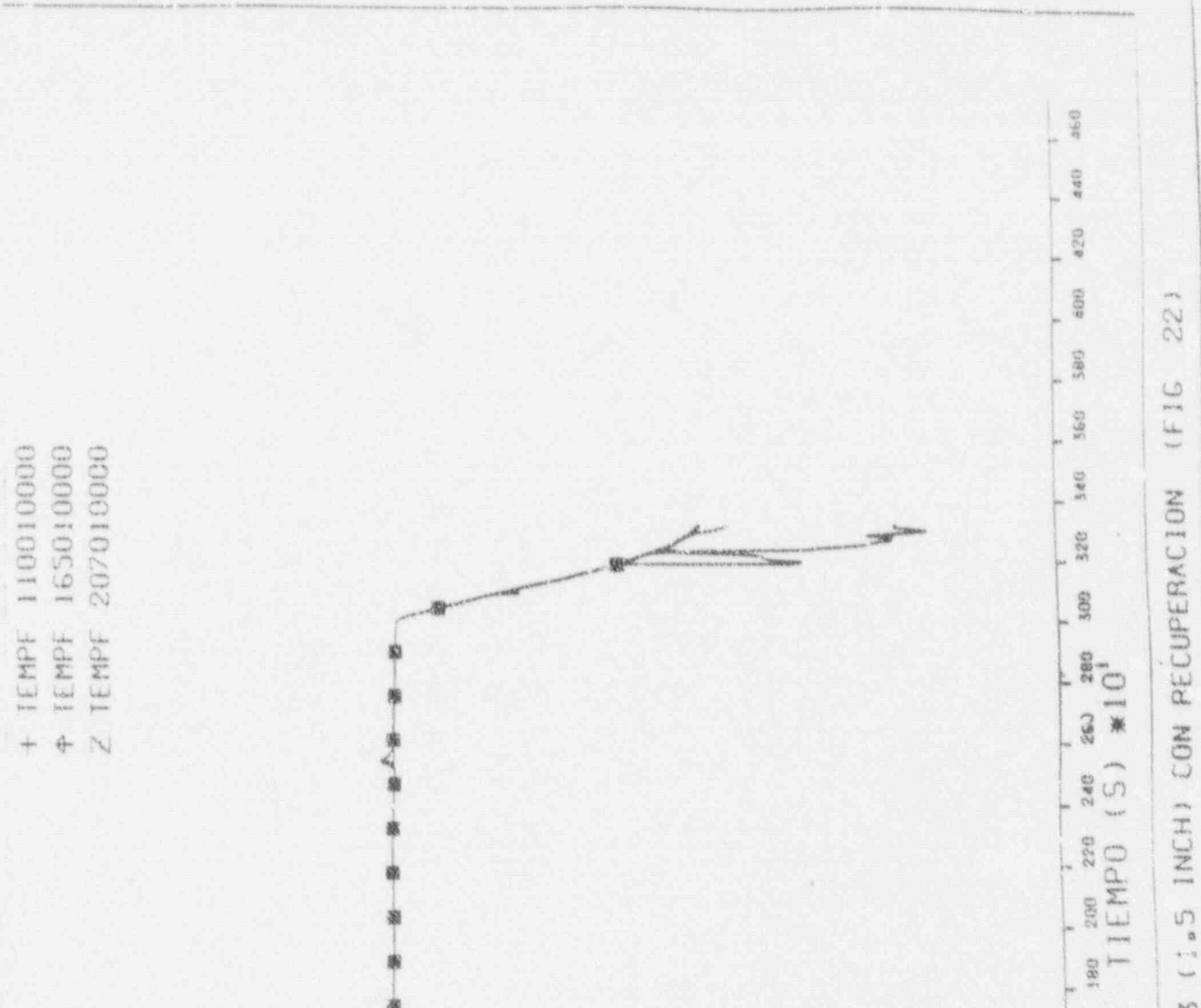
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 20)
FRACCIONES DE LIQUIDO EN EL NUCLEO (O/1)
TIEMPO (S) *10
VOIDF 209010000
VOIDF 209030000
VOIDF 209050000
VOIDF 209020000
VOIDF 209040000
VOIDF 209060000

FIG. III (E) - 20



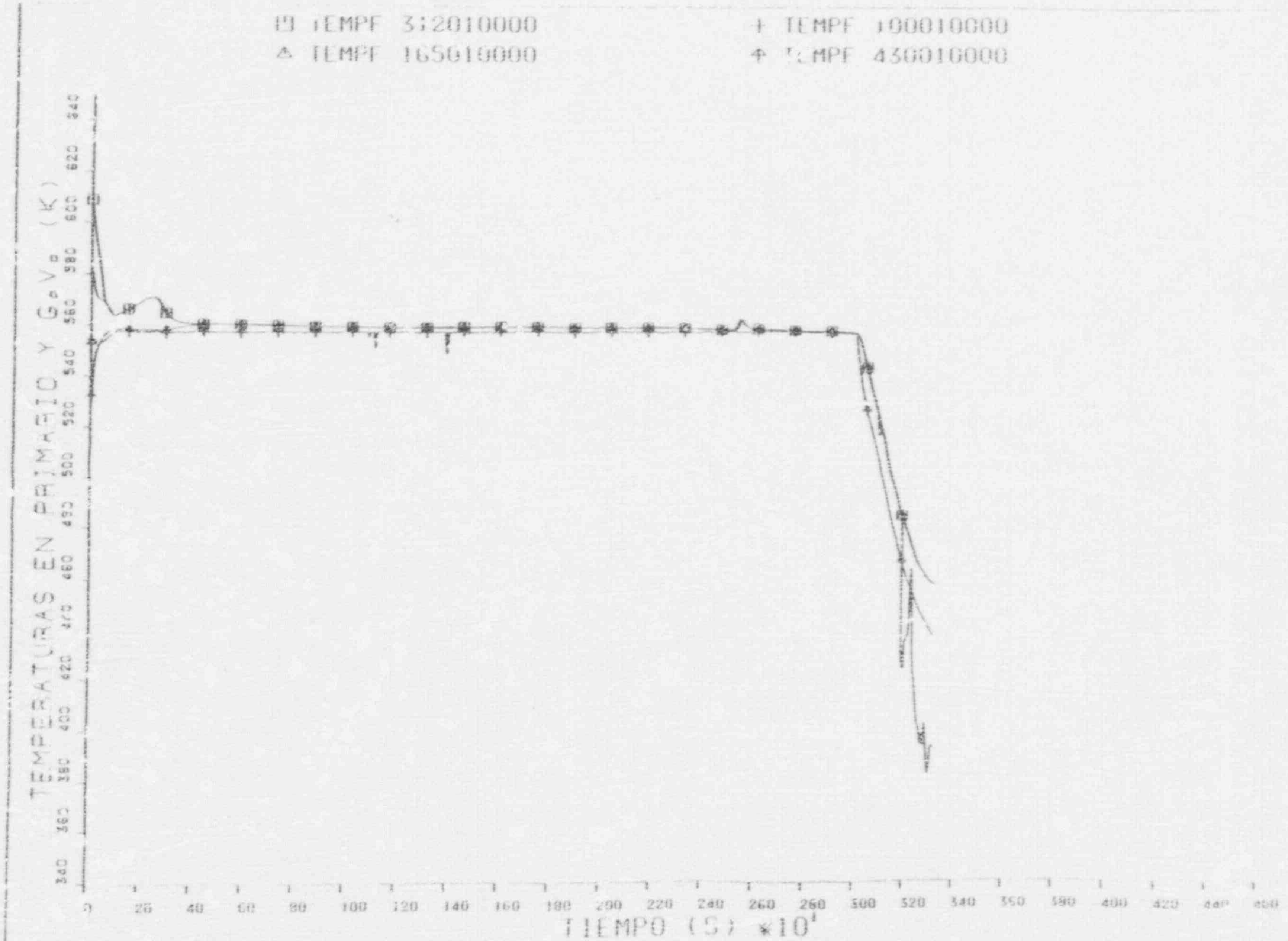
C.N. L.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 21)

FIG. III (B) - 21



□ TEMPF 100010000
△ TEMPF 130010000
× TEMPF 211010000
+ TEMPF 110010000
◆ TEMPF 165010000
Z TEMPF 207010000
TEMPERATURAS EN PRIMARIO (K)
TIEMPO (S) *10¹
C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 22)

FIG. III (B) - 22



C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 23)

FIG. III (C) - 23

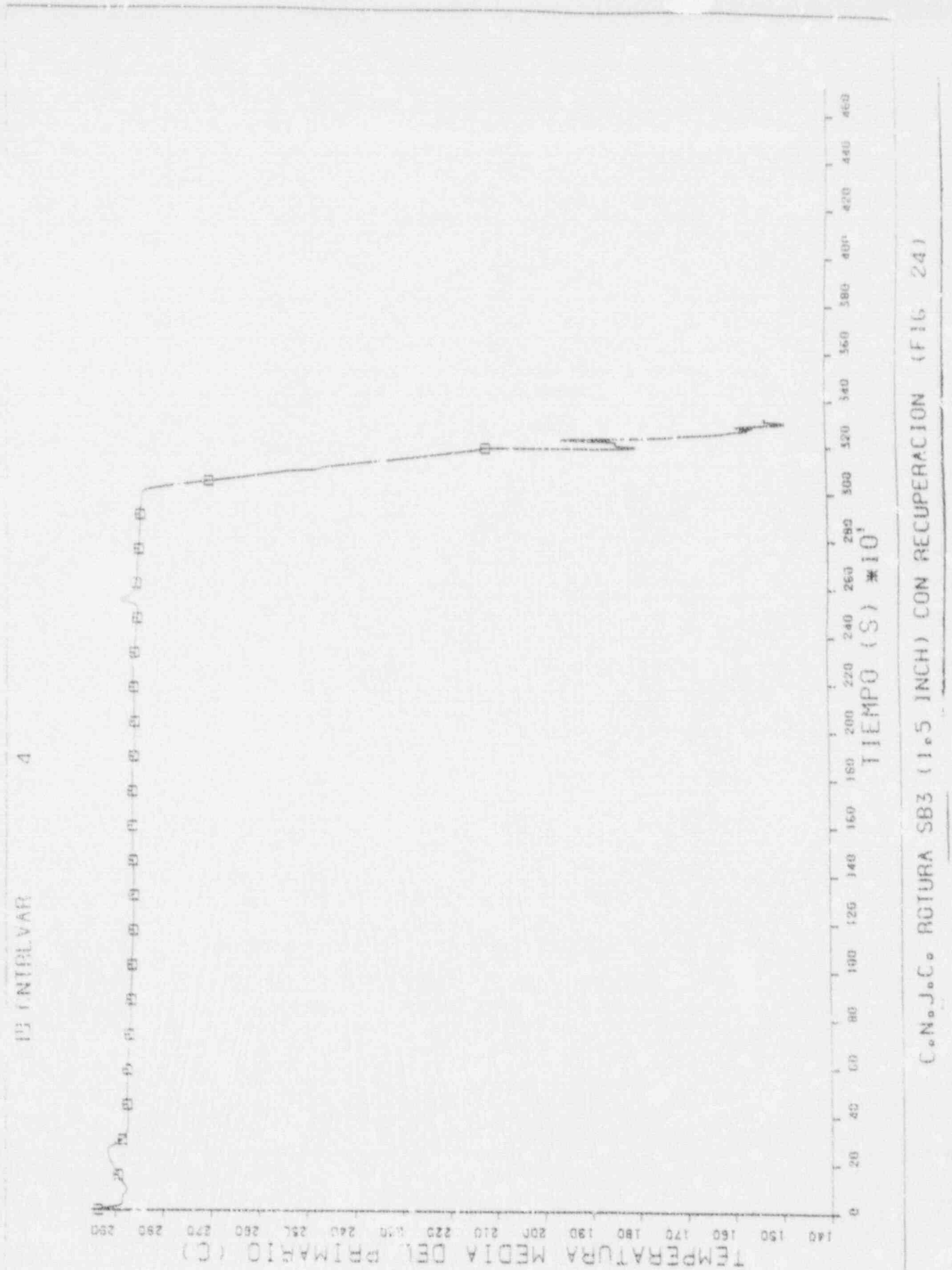
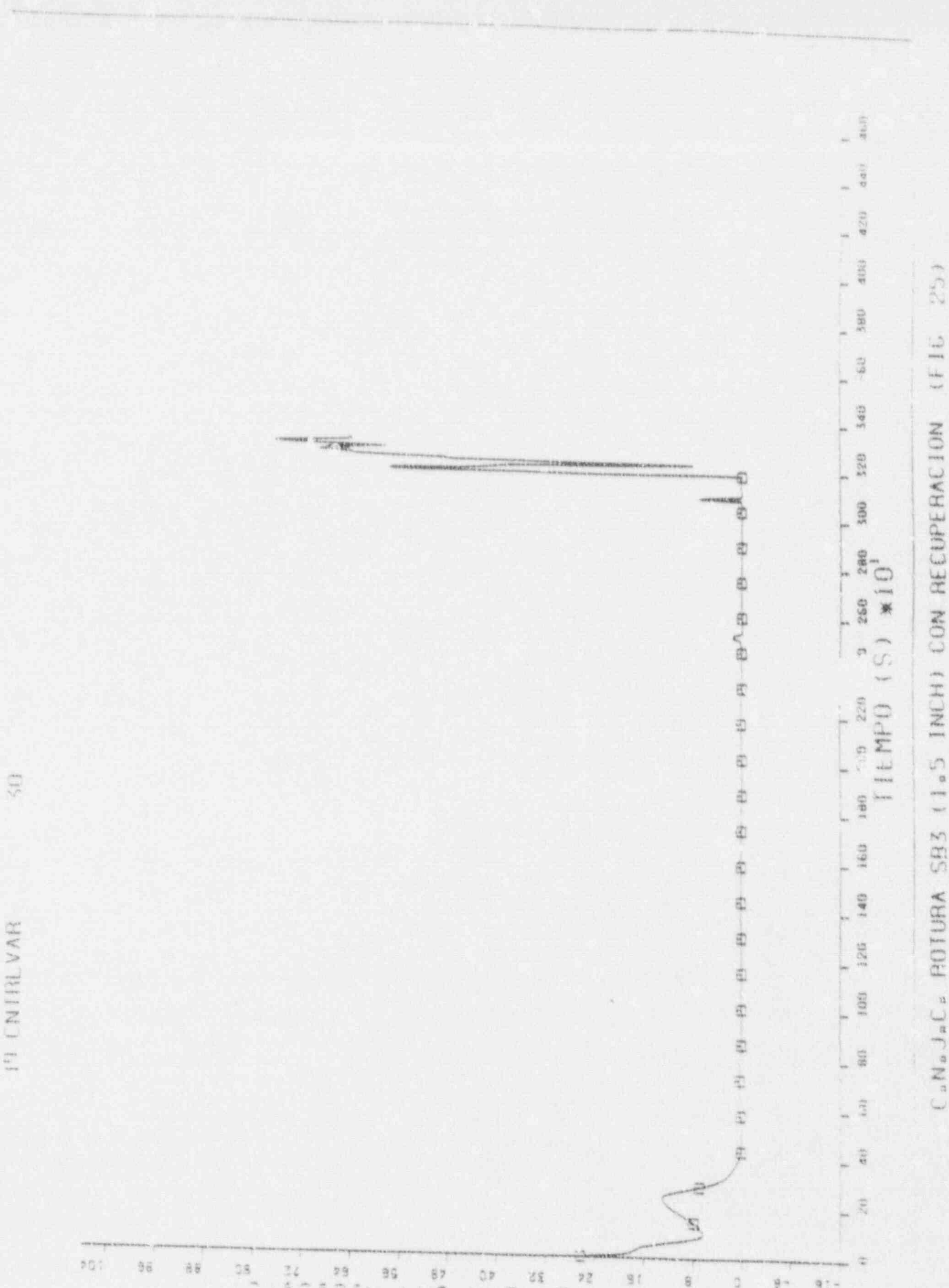

C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 24)
TIEMPO (S) *10^1
TEMPERATURA MEDIA DEL PRIMARIO (C)
IB INTRLVAR 4

FIG. III (B) - 24



C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 25)
TIEMPO (S) *10¹
DELTA TEMPERATURA R.C.S.(C)
CNTRLVAR 30

FIG. III (3) - 25

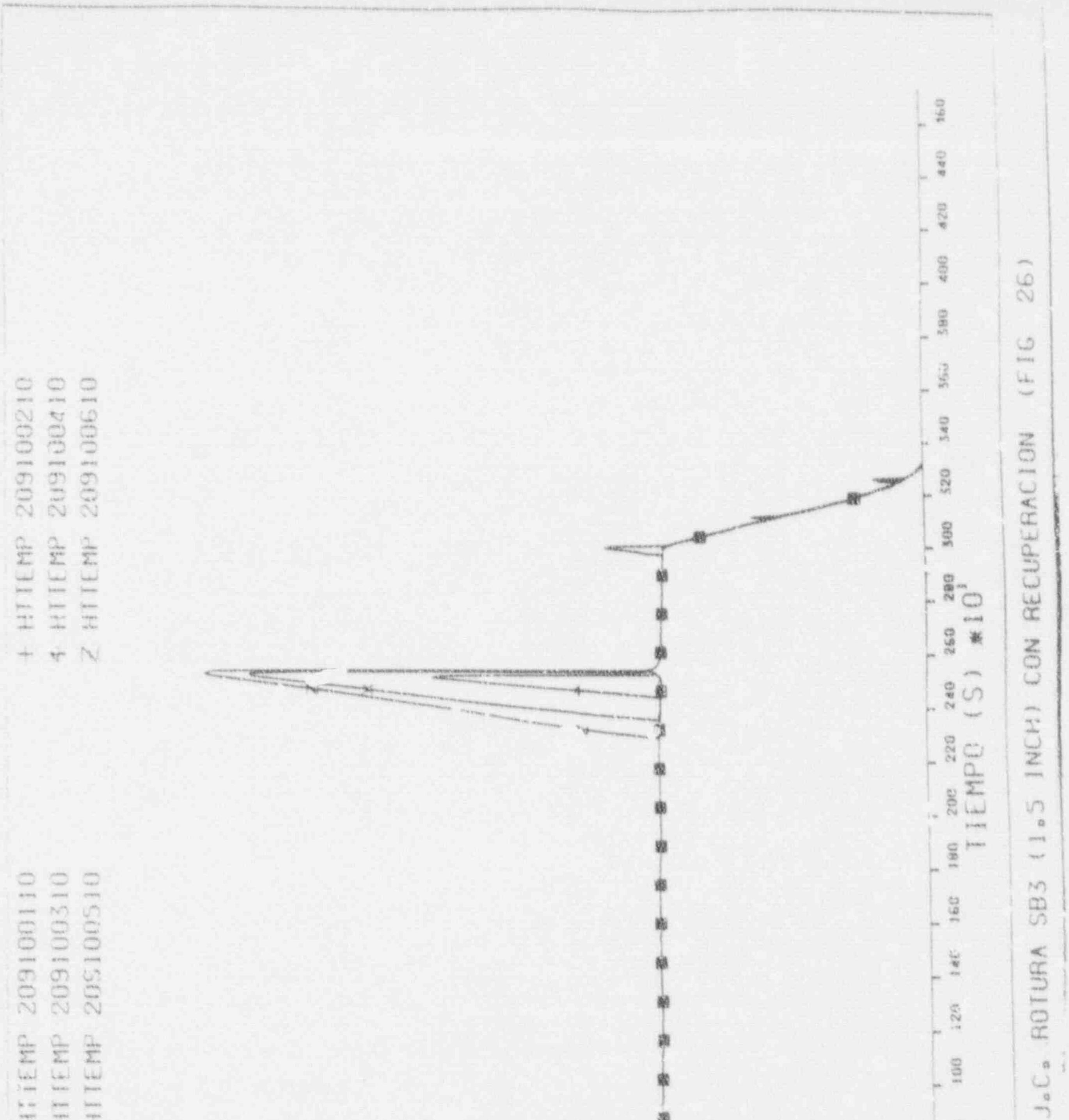
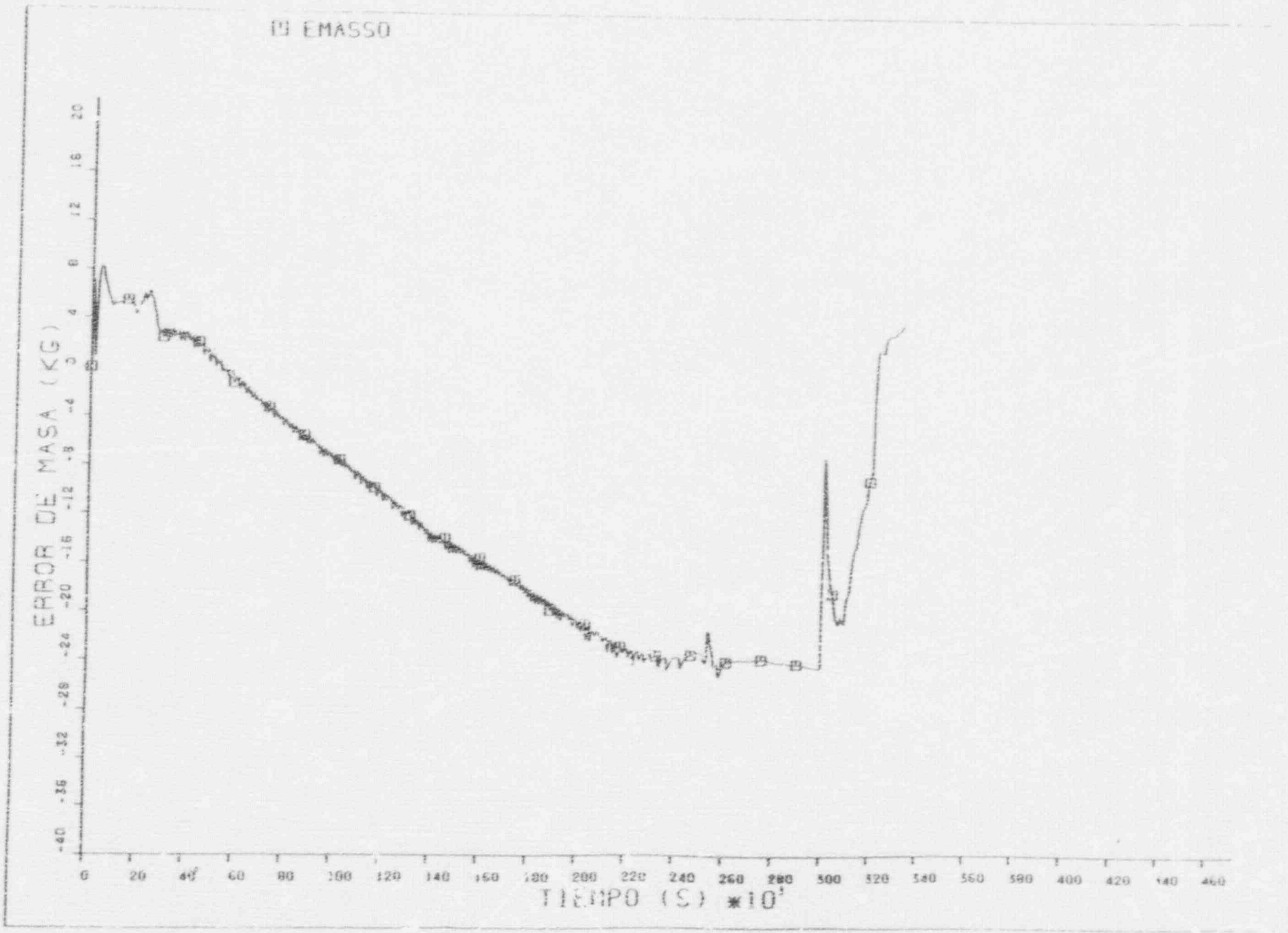

HTTEMP 209100110
HTTEMP 209100310
HTTEMP 205100510
HTTEMP 209100210
HTTEMP 209100410
HTTEMP 209100610
TEMPERATURA VAINAS CANAL MEDIO (K)
TIEMPO (S) *10^1
C.N. J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 26)

FIG. III (3) - 26



C.N.J.C. ROTURA SB3 (1.5 INCH) CON RECUPERACION (FIG 27)

FIG. III (B) - 27

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The RELAP5/MOD2/36.04 code has been used on a CYBER 180/830 computer and the simulation includes the 6" RHRS charging line, the 2" pressurizer spray, and the 1.5" CVCS make-up line piping breaks. The assumption of a "total black-out condition" coincident with the occurrence of the event has been made in order to consider a plant degraded condition with total active failure of ECCS. As a result of the analysis, estimates of the "time to core overheating startup" as well as an evaluation of alternate operator measures to mitigate the consequences of the event have been obtained. Finally, a proposal for improving the LOCA emergency operating procedure (E-1) has been suggested.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

ICAP, Postulated, LOCA, Cold Leg Small-Break, Lessons, OECD, LOFT, LP-SB-3, CYBER 180/830 Computer

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