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International Agreement Report

Assessment of RELAP5/MOD2 Against a Pressurizer Spray Valve Inadverted Fully Opening Transient and Recovery by Natural Circulation in Jose Cabrera Nuclear Station

Prepared by
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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

June 1993

Prepared as part of
The Agreement on Research Participation and Technical Exchange
under the International Thermal-Hydraulic Code Assessment
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NOTICE

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ABSTRACT

This document presents the comparison between the simulation results and the plant measurements of a real event that took place in JOSE CABRERA nuclear power plant in August 30th, 1984. The event was originated by the total, continuous and inadverted opening of the pressurizer spray valve PCV-400A.

JOSE CABRERA power plant is a single loop Westinghouse PWR belonging to UNION ELECTRICA FENOSA, S.A. (UNION FENOSA), an Spanish utility which participates in the International Code Assessment and Applications Program (ICAP) as a member of UNIDAD ELECTRICA, S.A. (UNESA). This is the second of its two contributions to the Program: the first one was an application case and this is an assessment one.

The simulation has been performed using the RELAP5/MOD2 cycle 36.04 code, running on a CDC CYBER 180/830 computer under NOS 2.5 operating system.

The main phenomena have been calculated correctly and some conclusions about the 3D characteristics of the condensation due to the spray and its simulation with a 1D tool have been got.

EXECUTIVE SUMMARY

This work shows the results of the analysis with RELAP5/MOD2 cycle 36.04 of a real event that took place in JOSE CABRERA nuclear power plant in August 30th, 1984. The event had its origen in a total, continuous and inadverted opening of the pressurizer spray valve PCV-400A. This is the second contribution of UNION ELECTRICA FENOSA, S.A. (UNION FENOSA) to ICAP.

JOSE CABRERA nuclear plant is a single loop Westinghouse PWR. A general purpose nodalization of JOSE CABRERA for RELAP5 has been used. This nodalization is being widely used in thermal-hydraulic applications and has given good results.

The inadverted opening of the spray valve caused a reactor coolant system depressurization, producing a reactor trip. The reactor coolant pump was manually stopped half an hour later in order to stop the spray flow. Afterwards, the cooling of the primary system was due to natural circulation. The safety injection system (9.7 MPa shut-off pressure) did not introduce water into the primary circuit because the RCS pressure was stabilized at 10.0 MPa.

The calculation purpose was not to study the control systems response, because they worked mainly in their manual mode, but to analyse the main thermal-hydraulic phenomena that appeared during the transient: steam condensation in the pressurizer due to the continuous spray flow and corresponding

RCS depressurization, primary cooling from the steam generator, reactor coolant pump trip effect, natural circulation, etc.

This objective was reached, the main phenomena were reproduced in the simulation and the discrepancies were justified. Main results of pressurizer separated effects experiments (MIT and NEPTUNUS) have been confirmed. In general, RELAP5/MOD2 code has performed properly and the host computer, CDC CYBER 180/830 has run well with the code with a good CPU time to real time ratio.

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F O R E W O R D

This report has been prepared by Unión Eléctrica Fenosa in the framework of the ICAP-UNESA Project.

The report represents one of the 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 Regulatory Commission (USNRC) in the form of Spanish contribution to the International Code Assessment and Applications Program (ICAP) of the USNRC 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
- EMPRESARIOS AGRUPADOS
- 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 developed by some organizations from several countries coordinated by the U.S. Nuclear Regulatory Commision (USNRC). Its purpose is to obtain a good vision of the USNRC thermal-hydraulic codes validity over the widest range of possible applications. The program generic interest derives from the extended diffusion and use of these computer codes.

The Spanish contributions to ICAP consist of different calculations, both assessment and application ones, with the RELAP5/MOD2, TRAC-PF1/MOD1 and TRAC-BF1 codes. The assessment cases compare simulations with real data, basically from commercial operating plants. The application cases study accidental situations, which prove the main thermal-hydraulic models of these codes in an exhaustive manner.

UNION ELECTRICA FENOSA, S.A. (UNION FENOSA) participates in ICAP as a member of UNIDAD ELECTRICA, S.A. (UNESA) contributing with two calculations: an application case (ref. 1) and an assessment case.

This report shows the assessment case, in which a real event in JOSE CABRERA nuclear power plant (NPP), a single-loop Westinghouse PWR belonging to UNION FENOSA, is studied. This event was caused by the total, continuous and inadverted locked opening of the pressurizer spray valve, producing the reactor trip. The reactor coolant pump (RCP) was manually stopped afterwards and this caused the cooling of the primary system by natural circulation without safety injection into the primary circuit.

RELAP5/MOD2-cycle 36.04 running on a CDC CYBER 180/830 computer under the NOS 2.5 operating system was used to analyse the selected plant transient.

The purpose of the calculation was not that of studying the interphase between the control systems, most of which were working in manual mode, and the thermohydraulic response of the plant, but that of reproducing the main event phenomena. The analysis has been based on the main thermal-hydraulic phenomena that the event presented: pressurizer steam condensation and corresponding reactor coolant system (RCS) depressurization, RCS cooling from the steam generator (SG), RCP trip effect, natural circulation, etc.

JOSE CABRERA NPP, where the event tooks place, is described in section 2, and the event itself in section 3. The model of the plant, used as RELAP5/MOD2 code input deck, is explained in section 4. The steady state results are commented in section 5. Section 6 shows the analysis results and their comparison with available plant data. Section 7 shows some interesting data on computer performance, and, finally, the main conclusions obtained are summarized in section 8.

2.- NUCLEAR STATION DESCRIPTION.

The analysis was done for JOSE CABRERA Nuclear Power Plant (fig. 2.1), a Westinghouse PWR commercial plant, sited in Zorita de los Canes (Guadalajara), belonging to UNION FENOSA, a Spanish utility (ref. 2). The plant had its first criticallity in 1968 and was the first nuclear station connected to the Spanish electrical grid.

The Reactor Coolant System (RCS) has only one loop that includes the cold leg, reactor pressure vessel, hot leg, pressurizer, steam generator tubes, cross-over leg, and reactor coolant pump. The Chemical and Volume Control System (CVCS) as well as the Residual Heat Removal System (RHRS) are also connected to the reactor coolant loop.

Nominal reactor power of 510 Mwt is generated by the reactor core that has a configuration of 69 fuel assemblies (14 x 14) with 2.40 m of active length. Reload fuel has an average enrichment of 3.60% in U-235. The plant nominal output electrical power is 160 Mwe with a frequency of 50 Hz.

The Emergency Core Cooling System (ECCS) 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 a jet pump taking water from the containment sump and feeding the injection pumps in the recirculation phase of a loss of coolant accident (LOCA).

There is one single steam generator that links thermally the primary and secondary systems. It includes the downcomer annulus, riser boiling chamber, separators and dryers.

In the secondary side the typical balance of plant (BOP) components are included (two 50% main feedwater pumps, steam line, safety (4) and steam-dump (3) valves, main steam isolation valve, turbine trip valves (2), main steam control valves (4), high (1) and low (1) pressure turbines, condenser, heaters (4), etc).

Main feedwater comes directly into the upper part of the downcomer without passing through any preheater section inside the steam generator, being previously heated through the heaters installed between the condenser and the steam generator. The circulation ratio in the secondary side of the steam generator is 1.96 at full power.

The SG auxiliary-emergency feedwater system includes one turbine driven and two motor operated pumps. Both subsystems take cold demineralized water from a tank and start their operation automatically. The turbine driven subsystem (AFWS) injects into the upper part of the downcomer, injection requiring the turbine operator intervention, by opening an isolation valve from the Control Room. The motor operated subsystem (EFWS) injects directly into the lower part of the SG tubes, no operator action being necessary to allow the subsystem injection. For this subsystem there is also the possibility of an optional injection into the upper part of the downcomer once the operator lines up properly the system.

The plant operates normally in automatic mode under the influence of the reactor control system that maintains the programmed coolant average temperature in the primary system by acting on the control bank B position.

The RCS pressure is controlled by the pressurizer pressure control system which acts on the PORV's, spray valve and heaters. The pressurizer level control system follows a level program as a function of the average temperature in the RCS by acting on the speed of one of the two charging pumps.

The steam generator level is controlled in operation by acting on the feedwater control valve based on the level error for the specified level setpoint, the feedwater flow and the steam flow.

Finally, there is a system to control the primary average temperature after the turbine trip by opening the steam-dump valves to the atmosphere (2) and to the condenser (1). This system can also work under pressure control by following a manually selected steam generator pressure setpoint.

The safety of the plant is guaranteed by the Reactor Protection System and the Emergency Safety Features.

3.- **EVENT DESCRIPTION**.

The event that was selected to be analysed and simulated with the RELAP5/MOD2 code is a reactor trip in JOSE CABRERA NPP, which happened at 13:40 h, August 30th, 1984. It was caused by the total, continuous and inadverted opening of the pressurizer spray valve PCV-400A that remained locked at its fully open position for more than half an hour. Until the beginning of the transient, the plant had been operating under automatic control with all the main parameters within normal ranges, at 96% thermal power.

The valve opening fault was not due to a control failure by the pressurizer pressure control system, but to a failure in its driving mechanism. At that time the only indication of the valve position in the Control Room, in percentage of demanding opening, came from the control system. Therefore, the indication the reactor operator could see during the depressurization event was 0% opening, preventing the correct diagnosis of the event. As a result of the evaluation by the NPP technical support center and in order to solve this problem in the future, a valve PCV-400A stem actual position indication was installed in the Control Room. This solution was considered acceptable by the Consejo de Seguridad Nuclear (Spanish regulatory commission).

When the primary pressure decrease following the spray valve opening was observed, the turbine operator tried to control it by means of a manual turbine load reduction. The pressurizer relief and spray valves indications were also checked showing 0% opening demanded by the pressurizer pressure control system.

The pressurizer level control system was set into its manual control mode, and the second charging pump was manually started to try to repressurize the primary circuit by partial refilling of the pressurizer.

As the RCS pressure went on decreasing, the low pressure reactor trip was produced at the programmed setpoint, so inducing a simultaneous turbine trip. After the turbine trip, the steam release through the steam-dump valves, which were working in their temperature control mode trying to reach the RCS no-load average temperature (275.0 °C), caused the primary pressure to go on decreasing very quickly. Therefore, the steam-dump control system was set into its pressure control mode by the reactor operator, and was manually controlled from that moment on.

The safety injection signal was automatically activated. From then on, the operation team continuously monitored the intermediate pressure safety injection pumps performance parameters, specially the driving motors currents and the ECCS lines flow. The ECCS flow indication was always 0.0 m^3/h, and the current of the motors was maintained at the normal value that is measured during the pumps maintenance monthly tests with the pumps working in the recirculation mode from/to the borated water tank. This fact demonstrated that cold water was never introduced into the vessel. This was also consistent with the fact that the minimum pressurizer pressure measured during the event was, according to the wide range recorder, 100.0 Kg/cm^2 (rel.), for a safety injection nominal shut-off pressure of 99.0 Kg/cm^2 (rel.).

When an auxiliar operator could go into the containment building, once its normal conditions were checked, he confirmed that the valve PCV-400A was fully open. A little bit later, and because it was impossible to close the valve either locally or from the Control Room, the RCP was tripped to finish the depressurization, by stopping the spray flow. By this operation, around half an hour after the beginning of the transient, the ECCS injection of cold borated water into the vessel downcomer was just prevented avoiding the potential risk of pressurized thermal shock to the reactor vessel. Some time later, the auxiliar operator could manually close this valve locally.

Throughout the incident, the operation team supervised the core behaviour by reading the core exit thermocouples temperatures, in order to control the subcooling margin that was always higher than 20.0 °C.

The minimum measured levels were 25% for the pressurizer and -80.0 cm for the steam generator.

About 40 minutes after the RCS depressurization was initiated, the event was already solved and the pressure was being recovered with natural circulation in the RCS, the fan coolers of the vessel head functioning, without any pressurizer level symptom of a steam bubble formation in the vessel head.

The lack of a process computer at the plant prevented knowing exactly the sequence of the event and the precise chronology of the operators intervention. Both had to be deduced from the several records and graphics available in the trip report, within the limits associated with their scales, speeds and precisions. In spite of these difficulties, this transient was selected as an assessment case because of its duration and the amplitude of the variation of the main thermal-hydraulic plant variables.

Figures 3.1 to 3.9 show the available plant records, on which this validation study is based. More information about this incident can be found in references 3 and 4.

4.- CODE INPUT MODEL DESCRIPTION.

For this assessment analysis, the RELAP5/MOD2/36.04 code (ref. 5) running 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, 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 JOSE CABRERA NPP nuclear steam supply system (NSSS) depicted in figure 4.1 is currently being used in the transient and safety analysis of the plant. It is a general purpose model developed specifically for JOSE CABRERA NPP in order to have a tool to allow the utility to do its own in-house 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 engineering procedures and simulation rules, has also been developed.

4.1.- Primary system nodalization.

The reactor core was divided into eight 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 modelled 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 order 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).

Heat structures to simulate both the pressurizer heaters and the heat losses have been modelled. Also the continuous spray mass flow rate of has been considered. There is an equilibrium among this continuous spray (0.08 Kg/s), the pressurizer heat losses (15 Kw) and the power generated by the proportional heaters power during the steady state (40 Kw).

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) carries 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 specific homologous curves 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 reference 6. The RCP speed coastdown after a trip has been fitted by comparison with plant reliable measurements of a specific test performed at hot zero power conditions. The adjusted parameter has been the internal friction torque and specially the TF_0 parameter of this torque in the RELAP5/MOD2 input deck. As a consequence of that, a good agreement has been got between measured and calculated mass flow rates (fig. 4.2 and 4.3).

The cold leg leading from the pump discharge to the vessel inlet was represented by two nodes (160, 165). The injection of the charging system was simulated by a time dependent volume (164) and a time dependent junction (163).

By applying the appropriate friction and form pressure loss coefficients, the thermal-hydraulic design reference loop pressure distribution, total pressure drop 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 tuning appropriated coefficients the referenced Westinghouse accumulator discharge mass 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, reactor core (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 SG 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 best estimate fuel temperature, coolant temperature and coolant density feed-back reactivity effects, has been selected for the active heat structures of the reactor core. Realistic data have also been used in the estimation of the decay heat which is obtained from the ANS-79 standard. The reactor control and protection systems based on the functional diagrams corresponding to the real gains and delays measured at the power station have been simulated too.

4.2.- Secondary system nodalization.

Feedwater was simulated as a time dependent junction (445), connected to the upper part of the downcomer, taking warm water from the time dependent volume (444) that represents the outlet of the 4^{th} heater. Feedwater temperature was simulated based on at power and post-trip 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 riser, 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 counterparts 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 so as to obtain 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 the appropriate friction and form loss coefficients in the natural circulation loop of the steam generator (400, 410, 450), with the highest resistance located in the downcomer/riser junction (455), the specified circulation ratio of 1.96 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 a plate of orifices 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 plate of orifices to the top of the downcomer.

When defining the scope of the plant model, this special emphasis in the simulation of the three separator stages, including the real definition of the draining ways, was 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, a 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 turbine trip valve (512), a single node (513), the main steam control 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 just 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 the appropriate friction and form loss coefficients, the reference secondary pressure distribution was met.

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

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

The automatic steam-dump system modulates the opening of the relief valve (532) to the condenser (time dependent volume 538) through the relief line (530, 534) and the opening of the relief valves (522 and 526) to the environment atmosphere (time dependent volumes 564 and 565) through their relief lines (524, 528), looking for the RCS no-load programmed average temperature. There is a common relief line (520) to the atmosphere and a general common relief line (518) from the main steam line. A valve (536) isolates the relief line to the condenser in case of loss of offsite A.C. power, protecting the condenser that would be unavailable under this circunstance.

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

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

The tuned 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 of the hydraulic diameter for the definition of the riser volumes.

A summary description of the model including concept, node number and type is given in Table 4.1.

4.3.- Trips and control variables.

The software of the reactor protection system has been modelled with a set of trips that forces the reactor scram when necessary.

A wide range of control variables has been defined to model the different control systems that work in the plant:

a) RCS average temperature control system:

Before the reactor trip, this system controls the rods bank insertion and withdrawal taking into account:

- the difference between the primary average temperature and the programmed one as a function of the turbine load determined by the pressure in the impulse chamber.

- the difference between the primary pressure and the reference one.

After the reactor trip, this system controls the steam dump from the steam generator to the atmosphere and to the condenser by following a primary temperature programme, as well as the stem position of the feedwater valve by following a RCS temperature hysteresis cicle.

b) Pressurizer pressure control system:

This system controls the primary pressure by acting on the spray, PORV's and heaters, looking for maintaining the reference pressure.

c) Pressurizer level control system:

This system controls the pressurizer level by acting on the speed of the CVCS charging pumps, looking for maintaining the programmed level as a function of the RCS average temperature.

d) Steam generator level control system:

This system controls the level of the steam generator by acting on the stem position of the feedwater control valve, looking for maintaining the constant reference SG level.

For an easier comparison with plant measurements in assessment cases, all the important Control Room instrumentation has been simulated with their actual units and delays (2.0 sec. for liquid temperature, 20.0 sec. for steam temperature, 1.0 sec. for liquid and steam mass flow rate, and 0.5 sec. for pressure).

4.4.- Simplifications.

In order to save CPU time in the specific transient analyses of this report, some simplifications have been done in the general purpose model to generate a reduced ad-hoc one (fig. 4.4), some control volumes and junctions being deleted (for instance, and due to the fact that this is a depressurization transient, the pressurizer relief lines, PORV's, safety lines and valves as well as the relief tank have been eliminated), the final nodalization having 73 control volumes, all of which remain as previously defined excepting the spray valve that has been removed and substituted by a constant mass flow rate TMDPJUN.

5.- 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 with high code running time and was not considered the more appropriate for getting the steady state.

To that aim, a stabilization system has been developed according to the logic of actuation represented in Table 5.1. In this way, the pressurizer pressure and level, primary coolant flow, primary average temperature, steam generator pressure and steam generator downcomer level have been fitted to the values measured in the plant.

This stabilization system was based on controllers that, by using properly selected gains, fitted the system parameters in such a way that the error signal (defined as the difference between the desired and the calculated value of each controlled variable) was reduced to zero as fast and stably as possible.

Reactor power and feedwater temperature were maintained constant as boundary conditions representing the nominal values corresponding to the normal operating condition. For this first calculation the kinetic model was not used.

After 346.5 sec. reactor time steady state calculation a fully stable condition for the controlled model has been got.

The final state has fitted 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.

Then, a second step eliminating the stabilization system and introducing the reactor kinetic model and the real reactor control and protection systems was carried out. The stable condition was reached inmediately. Table 5.2 gives a summary of the comparison between the main variables measured and calculated in the steady state simulation at full power.

Since the plant conditions at the beginning of the event were not exactly those corresponding to 100% power, but to 96%, a new steady state corresponding to these conditions was obtained with the same methodology as explained above. Because this steady state calculation started from the steady state at 100% power, the new desired conditions at 96% power were found very quickly. These new steady state results, which were used as the start point for the transient, are shown in table 5.3 compared to the measured values.

6.- TRANSIENT RESULTS AND COMPARISON WITH PLANT DATA.

The purpose of this analysis was to compare the response of the RELAP5/MOD2 code with the selected plant transient, focusing the attention on the following aspects of the RCS:

- depressurization due to the continuous spray,
- response to the cooldown from the steam generator,
- response to the RCP trip,
- natural circulation evolution.

As is regular practice in JOSE CABRERA NPP, the operation team worked most of the control systems. So these systems didn't perform according to their automatic logic.

The control systems behaviour during the transient up to the 2150.0 sec. studied in this simulation may be summarized as follows:

- RCS average temperature control system:

  It worked in automatic mode for the first 435.0 sec. in the transient (up to the reactor trip), reducing power to match the manual turbine load decrease.

- Pressurizer pressure control system:

  It worked in automatic mode during the whole transient. From 87.0 sec. on, all the pressurizer heaters worked at full power (300 Kw).

- Pressurizer level control system:

Before the reactor trip (435.0 sec.), the reactor operator started the second charging pump manually and, from then on, the whole system was manually operated to maintain the desired pressurizer level. Taking into account the operator intervention, the charging flow has been simulated as a TMDPJUN following the plant register data (fig. 3.6).

- Steam-dump control system:

Just after the turbine trip, the reactor operator switched the system from RCS average temperature control mode to SG pressure control mode, setting the adequate pressure setpoint to obtain the desired RCS average temperature. Taking into account that, because of the high capacity of the steam-dump valves, the SG pressure response reached in a few seconds the demanded one the measured SG pressure has been used as the reference for the simulation of the actuation of the control system.

- Steam generator level control system:

Once the turbine was tripped, it was manually controlled by the turbine operator to recover the steam generator downcomer level and to maintain it within an acceptable range.

In the calculation with RELAP5/MOD2 the automatic and manual modes of the control systems behaviour have been simulated, simplifying their modelation as much as possible, especially after the operators took the control of the systems. Anyway, as mentioned above, the main aim of the calculation was not to validate the control systems performance, as other plant transients would be more suitable for this purpose.

Therefore, the response of the pressurizer level and the steam generator pressure have been fitted with special care. Great attention has not been given to the steam generator level evolution, whose impact on other variables, specially in the primary circuit, is not very important as has been proved by sensitivity calculations.

The spray mass flow rate has been set as a TMDPJUN in order to obtain with the RELAP5/MOD2 models related to this phenomenon and specially with the condensation one, the same depressurization rate that was measured at the plant until the reactor trip. The assumption of a constant mass flow rate TMDPJUN model may be justified taking into account the following considerations:

- There is a constant geometry in the RCS between the discharge of the main coolant pump and the spray nozzle in the pressurizer.

- The failure of the spray valve forced a fully open position with constant area from the begining of the transient on.

- The density of the fluid in the cold leg was practically constant because of the small variation in the temperature and the reduced impact of the depressurization on it.

- A previous calculation of the same transient with TRAC-PF1/MOD1 (ref. 7) in which the spray valve was simulated with a VALVE component showed an almost constant calculated mass flow rate.

It has been shown that, with this hypothesis, the more suitable value of the equivalent mass flow rate at nominal conditions is 2.7 Kg/s, in contrast with the design mass flow rate of 4.8 Kg/s (reference documentation). It is important to point out that the real spray flow can not be measured in the plant and so it is not well defined but only referenced from the Westinghouse design data.

The discrepancy between both flows might be due to two different reasons:

- The three-dimensional (3D) characteristics of the condensation phenomenon caused by the spray (ref. 8):

  * Geometry of the nozzle and distance to the pressurizer level that may be divided into three spray zones: the continuous liquid zone, the break-up zone and the droplet zone.

  * Mean drop diameter, velocity, spray angle and break-up length as a function of the spray mass

flow rate.

* Form of the liquid sheet - characterized by the spray angle and the break-up length - that is responsible for the final form and efficiency of the spray.

* Break-up length of the liquid sheet into a swarm of fine droplets as a function of Weber and Jacob numbers. Due to the mechanism of aerodynamic instability, the produced drops are of very different sizes and fly at different velocities.

It may be summarized that in order to describe the spray, measurements of the shape and length of the liquid sheet and of the size and velocity distributions of the droplets are necessary.

- The limitation of the condensation correlations of the RELAP5/MOD2 code when analysing the condensation process only in one dimension (1D).

A similar 1D code (TRAC-PF1/MOD1) yielded a calculated spray mass flow rate of 2.9 Kg/s (ref. 7), in fact very close to the RELAP5/MOD2 value of 2.7 Kg/s, in such a way that it may be concluded that it is not a problem of models and correlations but of physical effects and/or uncertainty in the real mass flow rate.

The time sequence of the transient's more significant events is shown in table 6.1.

At time t=0.0 sec. the spray valve was completely open, and a RCS depressurization began due to the pressurizer steam cooling and condensation (fig. 6.1). This depressurization activated the pressure control system in such a way that, at time 87.0 sec., all the pressurizer heaters were working at full power and maintained this situation until the end of the simulation.

As the depressurization continued, the shift supervisor asked the turbine operator to decrease the turbine load in order to create a reactor/turbine mismatch to force an increase of the RCS temperature during the first moments in order to reduce the pressure decrease rate. He therefore throttled the turbine control valves, forcing a pressure increase in the steam generator.

The RCS average temperature control system started working when this temperature was higher than the system dead band, decreasing the nuclear power by means of control bank B insertion (fig. 6.2, 6.3 and 6.4). The throttling action produced an average temperature increase of between 0.5 and 1.0 °C in the primary circuit and a later temperature decrease by following automatically the program of average temperature in the RCS as a function of the turbine load. In this way the turbine load decreased and, therefore, the nuclear power decreased from 96%, the starting point, to 85% of the rated value.

The mentioned RCS average temperature increase produced a primary coolant swelling and the corresponding pressurizer level increase (fig. 6.5). The level control system, still performing in automatic mode, decreased the flow of the controlled CVCS charging pump. Afterwards, when the RCS average temperature decreased and the pressurizer level decreased consequently, the same pump automatically increased its flow. From this moment to the reactor trip, which occurs very soon, the second charging pump was manually started and the reactor operator tooks control of the pressurizer level. The manually started charging pump, injected its maximum flow of 240.0 l/min into the system.

The primary system depressurization continued until the low pressurizer pressure reactor trip set point of 125.5 Kg/cm^2 (rel.) was reached (fig. 6.1). As a result of the reactor trip, the turbine tripped and the steam-dump valves tried to get the no-load temperature in the RCS. Some seconds later, when the primary average temperature was lower than 280.0 °C, the main feedwater control valve began to close, in order to obtain a mass flow rate of 5% of the nominal one in 70.0 sec. time.

The hot and cold leg temperature decreased and, therefore, the RCS average temperature reduction (fig. 6.2, 6.3 and 6.4), produced a pressurizer level decrease (fig. 6.5), that was partially compensated by the charging pumps both working in their manual mode. In normal conditions, the level after the reactor trip would have automatically reached a value of 18% of the span. In the transient, under the manual influence of the reactor operator, a minimum level of 25% of the span was reached. It must be noted that the primary system pressure decreased sharper in the simulation than in the plant. That was because the pressurizer emptying and refilling caused a sharper depressurization and pressurization in the analytic results with RELAP5/MOD2 than observed, as was also demonstrated with the NEPTUNUS Y-05 pressurizer experiment and its subsequent study with this code (addendum A).

It has also been observed, as in other transients, a calculated steam generator level drop larger than measured in the plant after the turbine trip (fig. 6.7). This was due to the dynamic effects on the instrumentation, to the way the level in the RELAP5 model was defined (sum of all the downcomer cells liquid fraction multiplied by their heights), to the nodalization of the decreasing area annulus node (450-01) which was between the upper part and the lower part of the downcomer and finally to the uncertainty in the circulation ratio and feedwater flow after the reactor trip.

The feedwater flow evolution corresponding to the closing of the control valve from its nominal position to the 5% one has been considered to be linear, although it would be logical to assume that in the first phase of the closing transient the flow would decrease very slightly with the result of a value higher than simulated. So, in the analysis, the integrated mass introduced has been supposed to be lower than it would actually be at the plant (fig. 6.8). The feedwater temperature may also have some effect; the best estimate temperature evolution coming from other turbine trip analysis has been adopted.

Less than one minute after the turbine trip, the steam-dump control system was manually set into its SG pressure control mode. The reactor operator tuned the pressure setpoint to meet the desired RCS temperature evolution, which is equivalent to open the steam-dump valves when needed.

In the first phase of the steam-dump manual control the steam generator pressure is allowed to increase (fig. 6.6), with the subsequent RCS temperature and pressurizer level increase in the primary circuit. This fact, in combination with the additional pressurizer level increase as a result of the charging pumps manually working, forces a RCS repressurization which, for the same reason mentioned regarding the emptying after the reactor trip, is sharper in the simulation than observed in the plant (fig. 6.1).

After the partial refilling, the pressurizer level changes as a function of the primary average temperature, with small deviations due to manual and not very severe charging flow oscillations around the letdown flow value (fig. 6.5).

There is a phase in the simulation, between 800.0 sec. and 1150.0 sec., with a pressurizer pressure sharp decrease and later increase while the primary temperature and pressurizer level are increasing (fig. 6.1, 6.2 and 6.5). This tendency, apparently against the physic laws, appears in the plant records too, but in a smoother manner. It has been thought that this abnormal tendency could be due to local effects in the pressurizer, which had been refilled with subcooled water coming from the hot leg. This fact, together with the pressurizer water stratification and temperature axial gradient that forces the homogenization of the stratified water zones, the interphase heat transfer and the pressurizer walls temperature effects, justifies this pressure response.

The calculated depressurization that appears between 1150.0 and 1350.0 sec. is sharper than measured in the plant. This discrepancy has been considered to be due to a code malfunction as a consequence of the calculation of very cold water introduced in the node 6 of the pressurizer when the node 5 is just filled of water and the pressurizer level increasing continues. The very cold water temperature calculated at node 6, colder than the water at node 5 and colder than the previous calculated spray temperature at node 6 after its heating in the steam condensation proccess, results very near to the pressurizer inlet spray temperature. This non-physical-meaning code malfunction could correspond to the same error observed in the RELA5/MOD2 simulation of the MIT pressurizer test (addendum A) when the pressure evolution was abnormally calculated at the time the level was going from one node to the next one.

After this phase, from about 1350.0 sec., all the variables evolution meets the steam generator pressure evolution very well and with its effect on the primary temperature (fig. 6.6).

Between 1450.0 sec. and 1550.0 sec. a significant steam-dump manual depressurization in the steam generator is forced by the reactor operator (fig. 6.6), producing an RCS average temperature decrease (fig. 6.2). Approximately at that time, the system's lowest pressure is reached, about 100.0 Kg/cm^2 (rel.) according to the plant records. Anyway, this pressure was greater than the one at which the safety injection begins injecting water into the system because a value of 0.0 m^3/h safety injection flow was observed. In the simulation, the lowest pressure is about 2.0 Kg/cm^2 lower than measured and would have forced the ECCS water injection into the reactor pressure vessel if the corresponding TMDPJUN would not have been inhibited previously in the input deck.

At 1550.0 sec. the shift supervisor asked the reactor operator to trip the RCP once the problem has been clearly identified, in order to stop the spray flow. The main pump takes about 80.0 sec. to stop and, meanwhile, the spray flow decreases until it eventually stoped. Once the depressurization cause is finished and also due to the pump trip, the RCS begins increasing its pressure (fig. 6.1).

The hot leg temperature increases due to the RCP trip, but the cold leg temperature is not affected in the short term. This produces a primary average temperature and primary delta temperature increase (fig. 6.2 and 6.4).

At 1650.0 sec. the steam-dump is manually opened by the reactor operator in order to control the primary heating. The sharp cooling it produces can be seen more quickly and is sharper in the cold leg than in the hot leg, due to the low primary flow in natural circulation, and so, although the average temperature decreases, the delta temperature increases. This delta temperature increase, in combination with the one described above (the first one due to the hot leg temperature increasing, the second one due to the cold leg temperature decreasing), produces a very significant peak of the delta temperature, about 30.0 °C at the plant and 20.0 °C in the simulation (fig. 6.2, 6.3 and 6.4).

The difference between these two values has been concluded to be due in some way to different causes:

- A sharper or slightly shifted in time steam generator pressure evolution as a consequence of the turbine operator intervention.

- The records system reliability for such abrupt changes.

- The actual lack of hot leg axial temperature homogeneity, specially under low flow conditions.

- The temperature instrumentation locations in the upper part of the hot and cold legs for delta temperature calculation in the plant.

The last two reasons create a 3D problem beyond the 1D design bases of the RELAP5 code. This effect of axial thermal gradient in the hot leg combined with the instrumentation location has been previously observed in the plant, even under RCS nominal flow with the RCP running, when there is a pressurizer level decrease because of a malfunction of the CVCS and saturated water goes out of the pressurizer, is not mixed and flows through the upper part of the hot leg long enough to activate the reactor trip on "variable low pressurizer pressure" that is equivalent in this nuclear station to the OTAT reactor trip of a standard Westinghouse 3-loop PWR.

After that, stable natural circulation is intended. Of course, it is affected by the reactor operator manual action on steam generator pressure through the manual control of the steam-dump valves. The natural circulation primary delta temperature is established at about 12.0 °C at plant and 13.0 °C in the simulation (fig. 6.4). That is because the RELAP5 primary model has been fitted to the reference pressure drop that corresponds to the thermal design flow (75000. gpm), 8.5% lower than the actual flow. This actual flow is about 82000. gpm, but it is difficult to know it exactly. That is just a direct consequence of the overestimation in the pressure drop coefficients of the RCS based on the Westinghouse design references. The RCS nominal flow deviation justifies the small discrepancy observed in the simulation in the stabilized delta temperature under natural circulation: the RCS flow is lower and the delta temperature is higher than in the plant measurements.

The RELAP5/MOD2 simulation confirms the lack of a steam bubble formation in the vessel head due to the RCP trip under no-load RCS temperature and low pressure conditions.

The simulation finishes at 2150.0 sec., with the RCS pressure being recovered. This pressure is, at the end of the calculation, 106.0 Kg/cm^2 and it is increasing (fig. 6.1). Natural circulation is established, although it is affected by the manual depressurization of the steam generator performed by the reactor operator.

A good agreement between the code results and the data recorded in plant can be observed in all the transient phases, specially considering that all the interesting phenomena are clearly shown and the most significant discrepancies have been explained according to the RELAP5 models or the way some boundary conditions have been assumed. The higher primary temperature discrepancies are about 2.0-3.0 °C, and the greater RCS pressure discrepancies are about 5.0-6.0 Kg/cm^2. The worst agreement corresponds to the steam generator level, but even that variable shows about 30.0-35.0 cm. as the greatest discrepancy between plant and simulation data, which is considered acceptable.

Finally it is important to point out that the code assessment conclusion depends on the quality of the records. The script type thickness with the nominal recording speed of 1 inch/hour, which really is a little bit different from one record to another, produce an uncertainty of about 70.0 s. The uncertainty about the value of the variables depends on the scale of each one and is higher the higher the scale. Finally the chronology of events of the NPP trip report is just approximate. All these reasons make very difficult to describe the sequence of events with a greater accuracy.

Figures 6.1 to 6.9 show, in plant units and with the corresponding instrumentation delays, the comparison between the real data and those obtained in simulation. The remaining figures, from 6.10 to 6.25, have been attached to make the transient more easily understandable. Table 6.2 shows the variables identification for the transient figures.

7.- <u>RUN STATISTICS</u>.

During the first 1650.0 sec. of the transient, up to 100.0 sec. after the RCP trip, the maximum allowed time step was 0.5 sec. From this time to the end of the simulation, 500.0 sec. more, the maximum allowed time step was relaxed to 1.0 sec., as the Courant limit was less restrictive during this phase because of the low velocity of the fluid in the primary circuit.

Figure 7.1 shows the CPU time comsumption as a function of real time. Twice in the transient the integral CPU time goes to 0.0 sec. and corresponds to the time when a restart with some changes in the input file is made.

The run statistics summary is shown in table 7.1. The average CPU time (35172. sec.) to real time (2150.0 sec.) ratio was 16.36 to 1.

8.- CONCLUSIONS.

RELAP5/MOD2-cycle 36.04 has been used for an assessment case, comparing the analysis results with plant measurements of a real incident that took place in August 1984. The reference nuclear plant, JOSE CABRERA, has been nodalized and this nodalization is being used in a wide range of thermal-hydraulic applications. Both code and the model have worked friendly and properly.

The comparison with plant measurements has been difficult due to the lack of a process computer, the quality of the registers and, consequentely, the data uncertainties. Nevertheless, the selected transient has a big interest both generic (because of the thermalhydraulic phenomena that took place) and specific (behaviour of the single-loop nuclear power plant).

It has been possible to reproduce the main events in the transient and reach a good agreement between the simulation and the plant data. The steam generator level has been the variable in which a lower fitting effort has been done. It was proved that the highest discrepancy of 20.0-30.0 cm. has not a big impact on the other primary and secondary variables. Concerning RCS temperatures and pressure the highest discrepancies have been 2.0-3.0 ºC and 5.0-6.0 Kg/cm^2 respectively.

The spray mass flow rate has been tuned to fit a depressurization rate similar to the one in the plant before the reactor trip. The parameter was kept constant for the rest of the pump running time. The spray calculated value is lower than the reference design one. The same applies to TRAC-PF1/MOD1, for which the fitted value is close to the one of RELAP5. The discrepancy is due to the one-dimensional characteristic of RELAP5 studying a tri-dimensional phenomenon and mainly to the fact that the reference is a design value that can not be measured in the plant. However, with a well tuned spray equivalent mass flow rate, it is not neccessary a 3D pressurizer model to simulate its effect on the RCS pressure.

Some experimental results (NEPTUNUS Y-05 and MIT pressurizer tests) have been simulated to check the code. From their results with no uncertainty about the spray flow it may be concluded that the code underpredict the spray condensation efficiency. Based on this evaluation and the results of the plant simulation it might be concluded that the real spray flow in the plant is lower than the fitted one, so much lower than the design reference value. However, to support this conclusion, the previously mentioned 3D phenomena should be considered.

As in the simulation of the experiments, the plant RCS depressurization and pressurization due to pressurizer partial emptying and refilling were sharper in the analysis than in the plant measurements and also an abnormal pressure response appears when the pressurizer level changes from one node to the next one.

It is important to remark the big effect of some pressurizer local phenomena (such as cold water stratification with axial temperature gradient, heat transfer in the water/steam and spray/steam interphases, heat thansfer from/to the walls, heat slabs properties, heat losses to the containment atmosphere) on the RCS pressure. A detailed pressurizer nodalization is required to properly simulate these effects.

The response of the steam generator level immediately after the turbine trip is sharper in the simulation than in the plant. This might be due to the way of modelling the collapsed liquid level in the steam generator downcomer. The necessity of modelling the downcomer delta pressure instrumentation or, at least, of giving a good table of level as a function of the mass inventory, specially in the zones with complex geometry, has been detected.

In general, the primary parameters have followed very well the cooldown induced from the steam generator. The very high peak of delta temperature in the RCS after the main coolant pump trip has been obtained and also the stable delta temperature in natural circulation has been properly calculated without a vessel head steam bubble formation. This means that the friction and form pressure losses coefficients, fitted with the pump running, are good enough to reproduce the pressure losses in natural circulation with a Reynolds number much lower.

The final calculation has been carried out after several sensitivity analyses. In spite of the duration of the transient, these analyses have been possible because of the good CPU time to real time ratio, specially after the RCP trip when the CPU time consumption decreased a lot because the maximum allowed time step could be higher due to the lower restriction of the Courant limit. Also, the CDC CYBER 180/830 computer has performed successfully with RELAP5/MOD2.

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TABLE 4.1. **RELAP5/MOD2 model description (nodalization)**.

CONCEPT	NODE NUMBER	NODE TYPE
PRIMARY COOLANT SYSTEM		
Cold leg - pump discharge	160	branch
Charging flow (CVCS)	163	tm.dp.jun.
CVCS tank	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 4.1 **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 4.1 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 the 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 5.1 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 5.2 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	(%)	64.00	64.00
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	(-)	1.96	1.96
Steam generator collapsed liquid level	(cm)	0.0	0.0
Steam flow rate	(Kg/s)	266.40	265.90
Feedwater temperature	(ºK)	477.00	477.00

TABLE 5.3 Steady state results at 96.0%

Variable	Units	JOSE CABRERA (measured)	RELAP5/MOD2 (calculated)
Reactor power	(Mw)	490.00	490.00
RCS average temperature	(ºK)	565.80	565.70
Pressurizer level	(%)	62.00	61.90
Pressurizer pressure	(MPa)	13.82	13.82
RCS mass flow rate	(Kg/s)	3686.80	3686.80
Reactor coolant pump speed	(rpm)	990.00	995.00
Steam generator pressure	(MPa)	4.68	4.68
Steam generator circulation rate	(-)	1.96	1.96
Steam generator collapsed liquid level	(cm)	0.0	0.0
Steam flow rate	(Kg/s)	255.40	256.00
Feedwater temperature	(ºK)	477.00	477.00

TABLE 6.1. **Sequence of Events**

EVENT	TIME (s) calculated	TIME (s) measured
Spray valve PCV-400A opening	0.0	0.0
Pressurizer heaters at full power	87.0	120.0
Turbine load decrease begins	135.0	135.0
Reactor trip	435.0	440.0
Turbine trip	435.0	440.0
Safety injection signal	462.0	470.0
Manual reactor coolant pump trip	1550.0	1550.0
End of simulation	2150.0	

TABLE 6.2 **Variables identification in the transient figures**

CNTRLVAR	066	- PRZ PRESSURE (KG/CM2 REL)
CNTRLVAR	056	- RCS AVERAGE TEMPERATURE (C)
CNTRLVAR	054	- COLD LEG TEMPERATURE (C)
CNTRLVAR	058	- RCS DELTA TEMPERATURE (C)
CNTRLVAR	073	- PRZ LEVEL (% OF SPAN)
CNTRLVAR	076	- SG PRESSURE (KG/CM2 REL)
CNTRLVAR	083	- SG DOWNCOMER LEVEL (CM)
MFLOWJ	445000000	- FEEDWATER MASS FLOW RATE (KG/S)
MFLOWJ	430010000	- STEAM MASS FLOW RATE (KG/S)
CNTRLVAR	111	- CORE POWER (W)
MFLOWJ	425000000	- SPRAY MASS FLOW RATE (KG/S)
CNTRLVAR	100	- PRZ HEATERS POWER (W)
TEMPF	350030000	- SPRAY TEMPERATURE (K)
TEMPF	165010000	- COLD LEG TEMPERATURE (K)
TEMPF	100010000	- HOT LEG TEMPERATURE (K)
TEMPG	430010000	- SG DOME TEMPERATURE (K)
TEMPF	207010000	- CORE EXIT TEMPERATURE (K)
SATTEMP	207010000	- CORE EXIT SATURATION TEMPERATURE (K)
HTTEMP	209100110	- AVERAGE CHANNEL CLAD TEMP. NODE 1 (K)
HTTEMP	209100210	- AVERAGE CHANNEL CLAD TEMP. NODE 2 (K)
HTTEMP	209100310	- AVERAGE CHANNEL CLAD TEMP. NODE 3 (K)
HTTEMP	209100410	- AVERAGE CHANNEL CLAD TEMP. NODE 4 (K)
HTTEMP	209100510	- AVERAGE CHANNEL CLAD TEMP. NODE 5 (K)
HTTEMP	209100610	- AVERAGE CHANNEL CLAD TEMP. NODE 6 (K)
MFLOWJ	105010000	- RCS MASS FLOW RATE (KG/S)
TEMPF	310010000	- PRZ LIQUID TEMPERATURE NODE 1 (K)
TEMPG	310010000	- PRZ STEAM TEMPERATURE NODE 1 (K)
SATTEMP	310010000	- PRZ SATURATION TEMPERATURE NODE 1 (K)
HTTEMP	310100105	- PRZ WALL TEMPERATURE NODE 1 (K)
TEMPF	310020000	- PRZ LIQUID TEMPERATURE NODE 2 (K)
TEMPG	310020000	- PRZ STEAM TEMPERATURE NODE 2 (K)
SATTEMP	310020000	- PRZ SATURATION TEMPERATURE NODE 2 (K)
HTTEMP	310200105	- PRZ WALL TEMPERATURE NODE 2 (K)
TEMPF	310030000	- PRZ LIQUID TEMPERATURE NODE 3 (K)
TEMPG	310030000	- PRZ STEAM TEMPERATURE NODE 3 (K)
SATTEMP	310030000	- PRZ SATURATION TEMPERATURE NODE 3 (K)
HTTEMP	310200205	- PRZ WALL TEMPERATURE NODE 3 (K)
TEMPF	310040000	- PRZ LIQUID TEMPERATURE NODE 4 (K)
TEMPG	310040000	- PRZ STEAM TEMPERATURE NODE 4 (K)
SATTEMP	310040000	- PRZ SATURATION TEMPERATURE NODE 4 (K)
HTTEMP	310200305	- PRZ WALL TEMPERATURE NODE 4 (K)
TEMPF	310050000	- PRZ LIQUID TEMPERATURE NODE 5 (K)
TEMPG	310050000	- PRZ STEAM TEMPERATURE NODE 5 (K)
SATTEMP	310050000	- PRZ SATURATION TEMPERATURE NODE 5 (K)
HTTEMP	310200405	- PRZ WALL TEMPERATURE NODE 5 (K)
TEMPF	310060000	- PRZ LIQUID TEMPERATURE NODE 6 (K)
TEMPG	310060000	- PRZ STEAM TEMPERATURE NODE 6 (K)
SATTEMP	310060000	- PRZ SATURATION TEMPERATURE NODE 6 (K)
HTTEMP	310200505	- PRZ WALL TEMPERATURE NODE 6 (K)
TEMPF	312010000	- PRZ LIQUID TEMPERATURE NODE 7 (K)
TEMPG	312010000	- PRZ STEAM TEMPERATURE NODE 7 (K)
SATTEMP	312010000	- PRZ SATURATION TEMPERATURE NODE 7 (K)
HTTEMP	310200605	- PRZ WALL TEMPERATURE NODE 7 (K)
TEMPF	312020000	- PRZ LIQUID TEMPERATURE NODE 8 (K)
TEMPG	312020000	- PRZ STEAM TEMPERATURE NODE 8 (K)
SATTEMP	312020000	- PRZ SATURATION TEMPERATURE NODE 8 (K)
HTTEMP	312100105	- PRZ WALL TEMPERATURE NODE 8 (K)
VOIDG	310050000	- PRZ VOID FRACTION NODE 5
VOIDG	310060000	- PRZ VOID FRACTION NODE 6
CPUTIME	0	- CPU TIME (S)

TABLE 7.1 **Run Statistics**

Real time	RT	= 2150.0 seconds
CPU time	CPU	= 35172.0 seconds
Total number of volumes	C	= 73
Total number of time steps	DT	= 25961

$$(CPU \times 1000)/(C \times DT) = 18.5$$

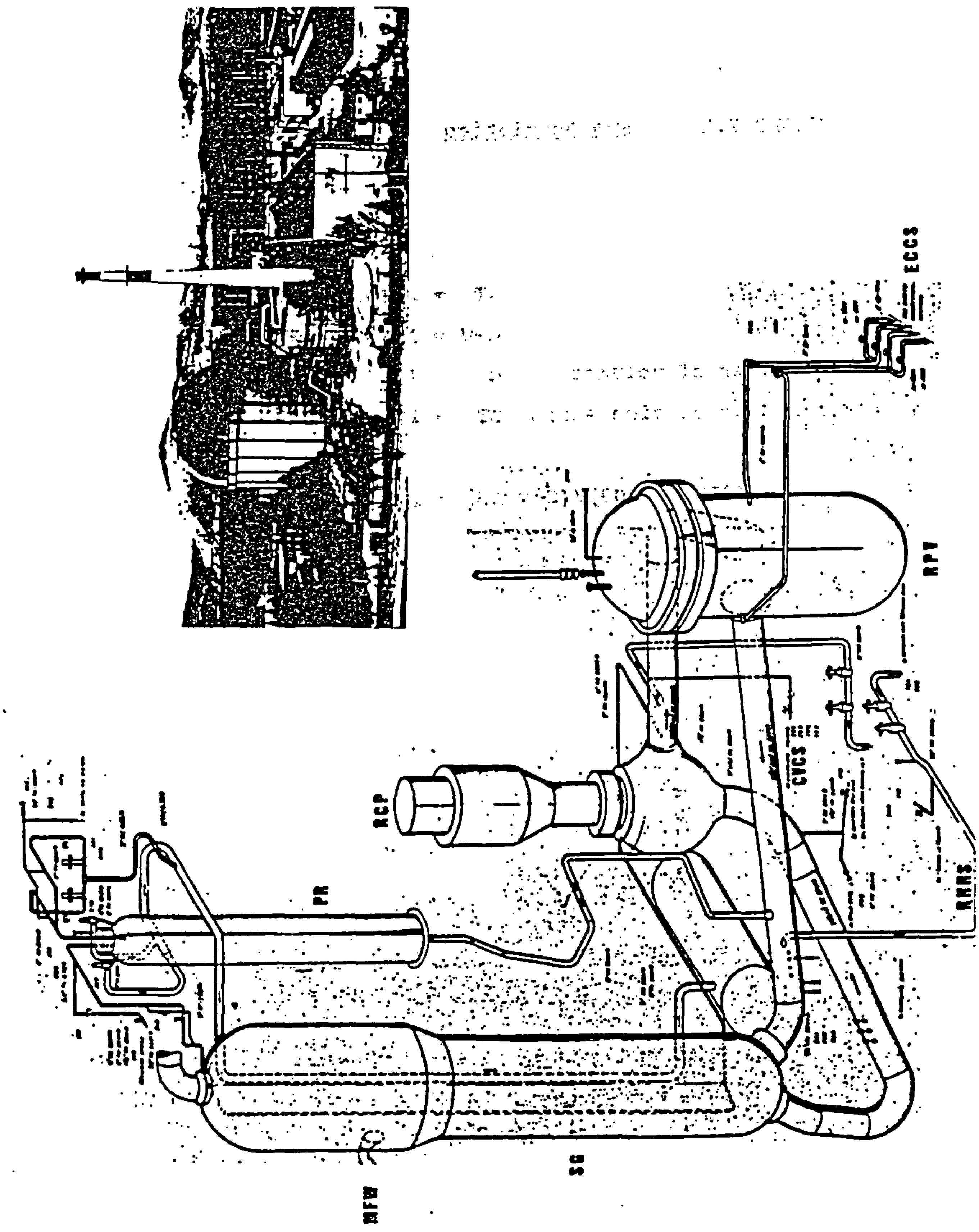


FIG. 2.1 JOSE CABRERA NUCLEAR POWER PLANT REPRESENTATION.

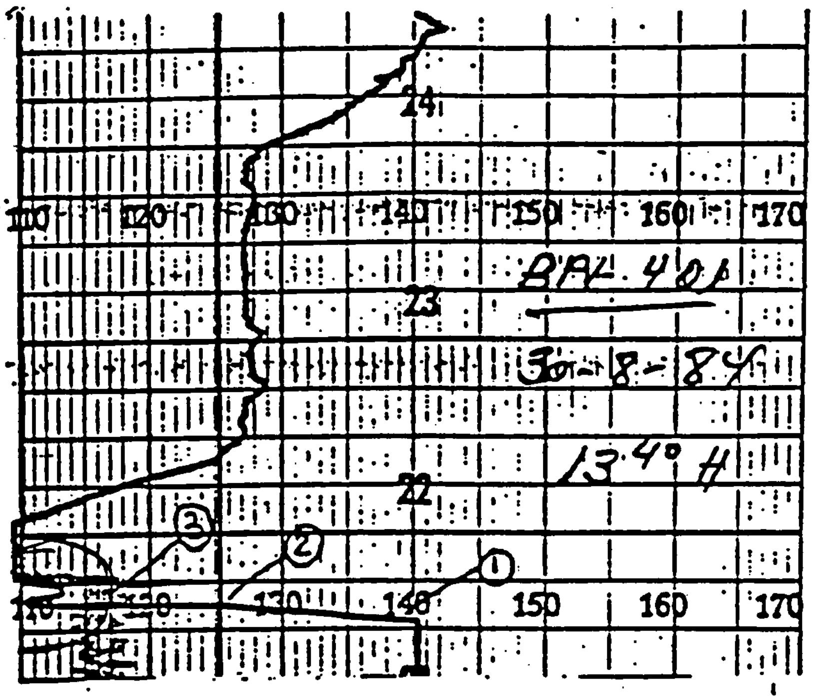


FIG. 3.1. PRESSURIZER PRESSURE (KG/CM2 REL, NARROW RANGE)

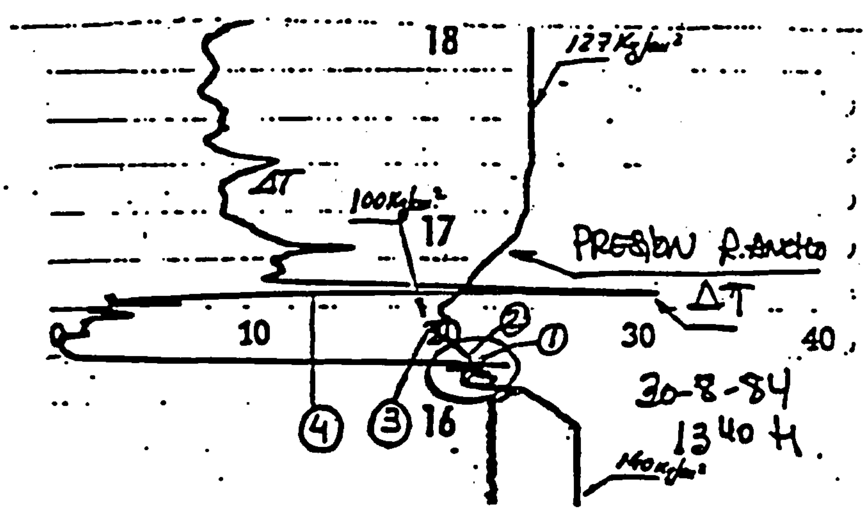


FIG 3.2. PRESSURIZER PRESSURE (KG/CM2 REL, WIDE RANGE)

RCS DELTA TEMPERATURE (C)

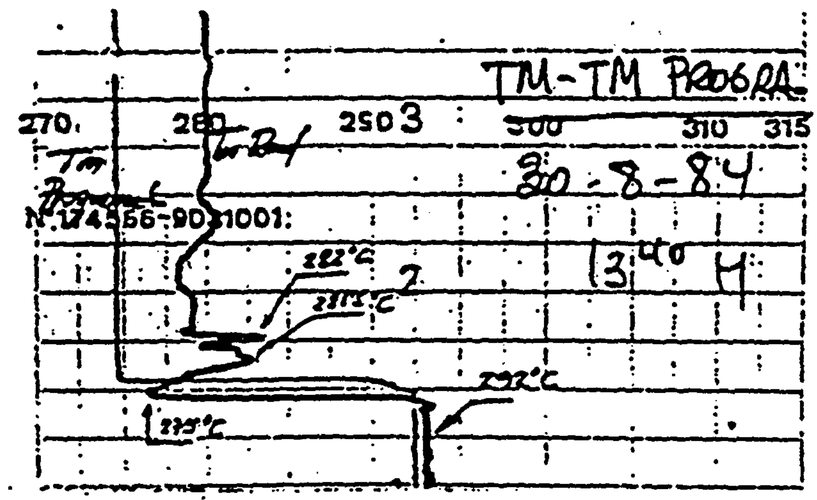


FIG. 3.3. RCS PROGRAMMED AVERAGE TEMPERATURE (C)

RCS AVERAGE TEMPERATURE (C)

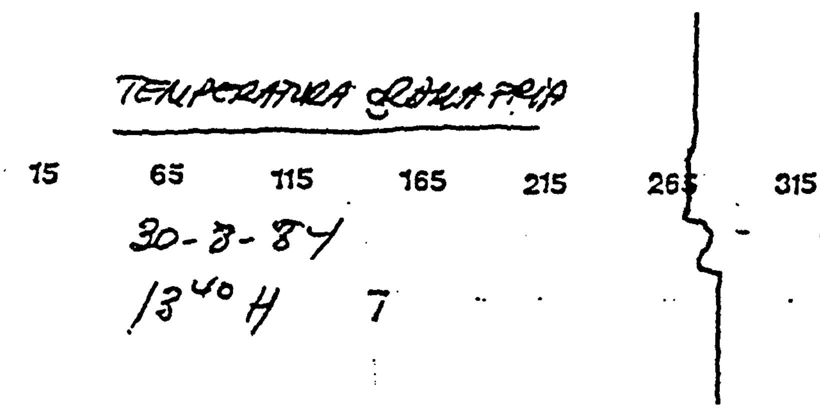


FIG 3.4. COLD LEG TEMPERATURE (C)

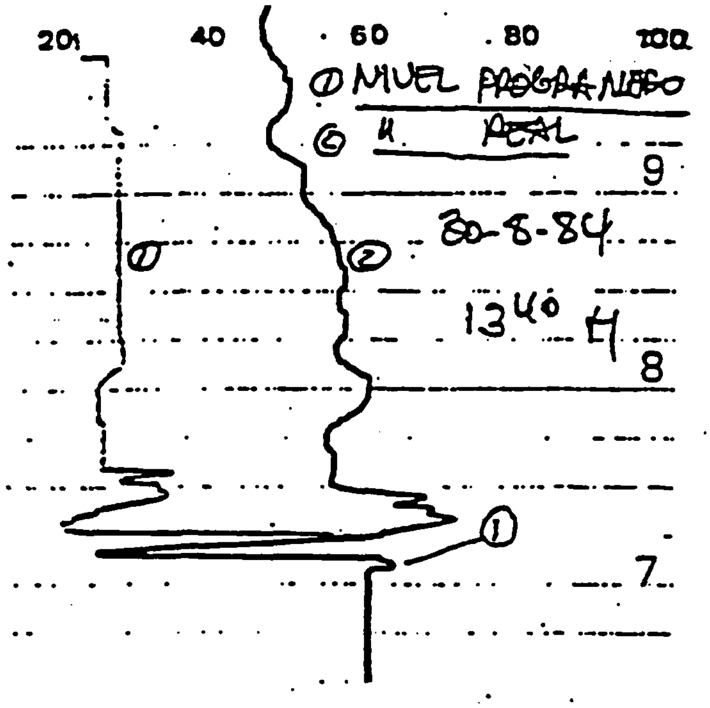


FIG. 3.5. PRESSURIZER PROGRAMMED LEVEL (% OF SPAN)

PRESSURIZER LEVEL (% OF SPAN)

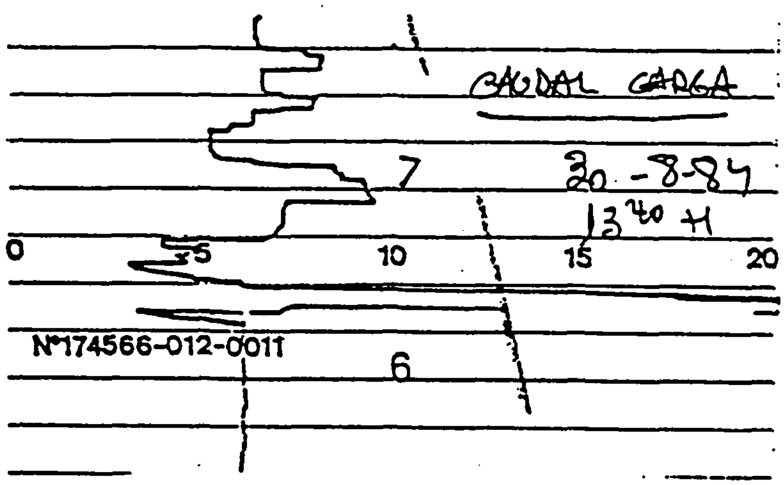


FIG. 3.6. RCS CHARGING FLOW (LPM)

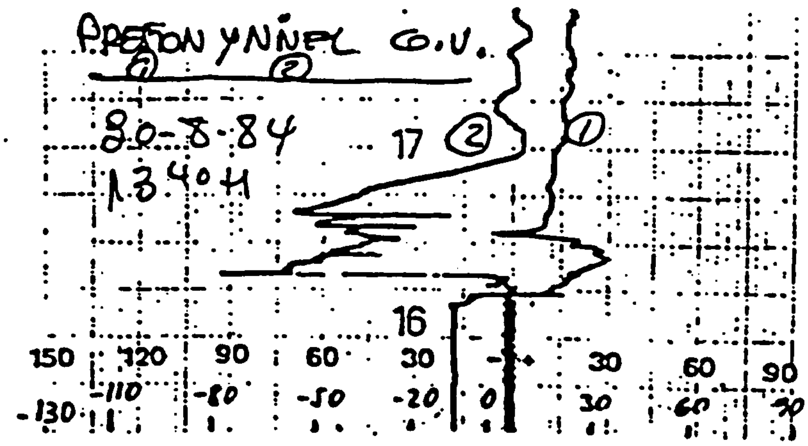


FIG. 3.7. STEAM GENERATOR PRESSURE (KG/CM2 REL)
STEAM GENERATOR DOWNCOMER LEVEL (CM)

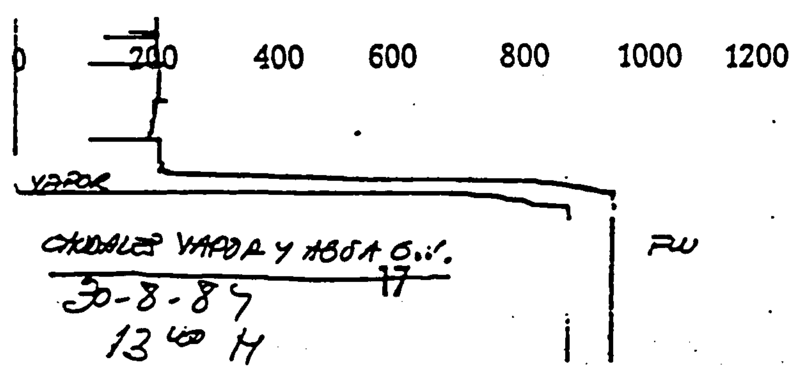


FIG. 3.8. FEEDWATER MASS FLOW RATE (TN/H)
STEAM MASS FLOW RATE (TN/H)

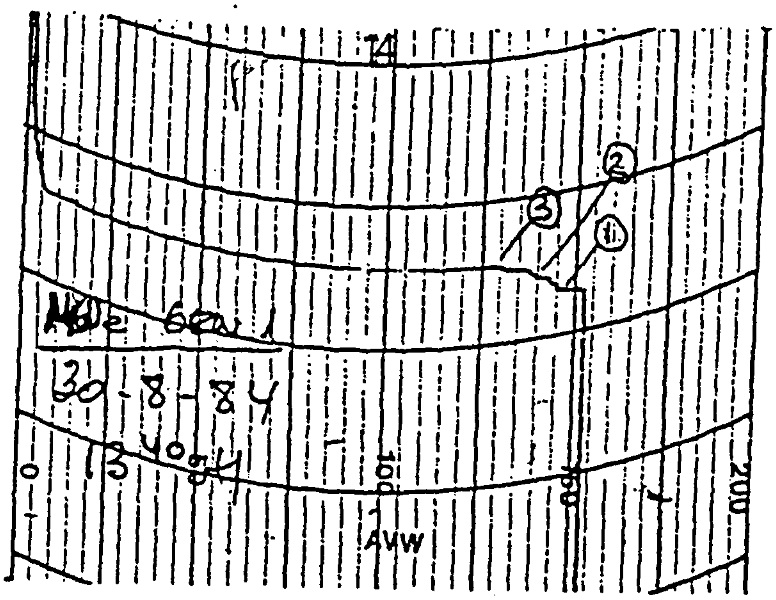


FIG. 3.9. TURBINE LOAD (MWe)

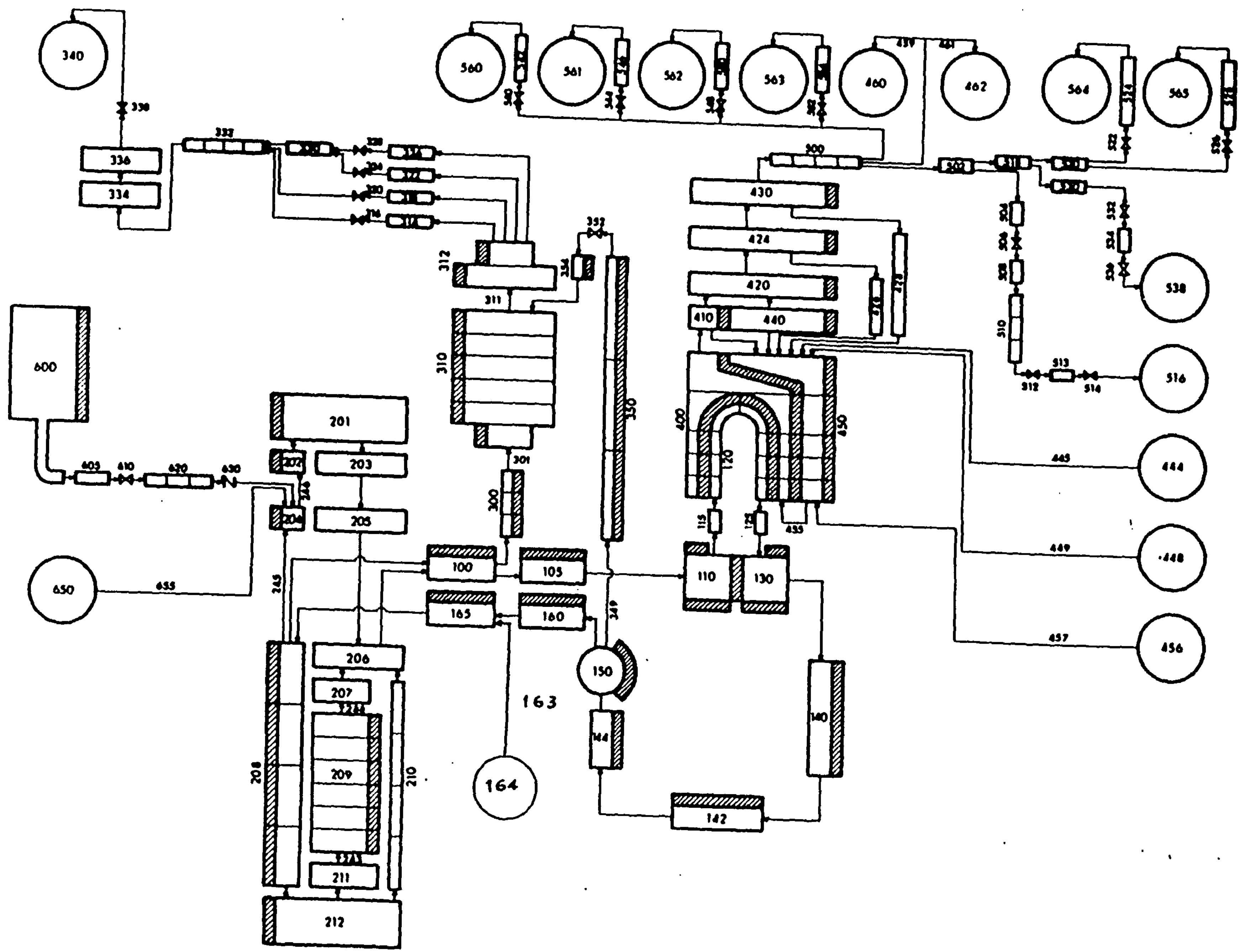


FIG. 4.1 JOSE CABRERA PLANT NODALIZATION

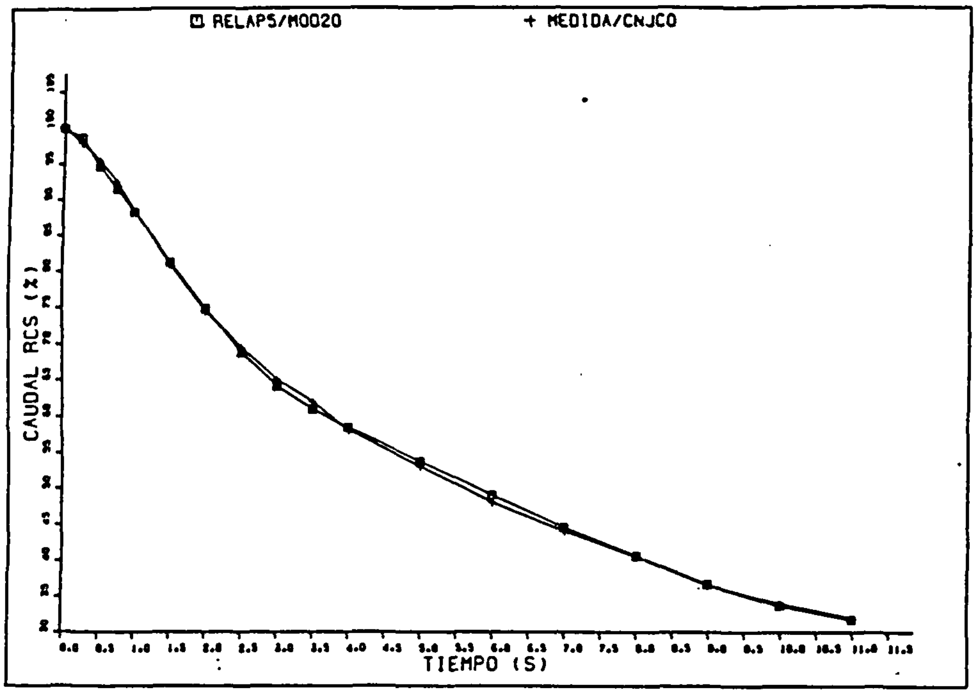


FIG. 4.2. RCP COASTDOWN. PERCENTAGE OF THE NOMINAL FLOW. SHORT-TERM.

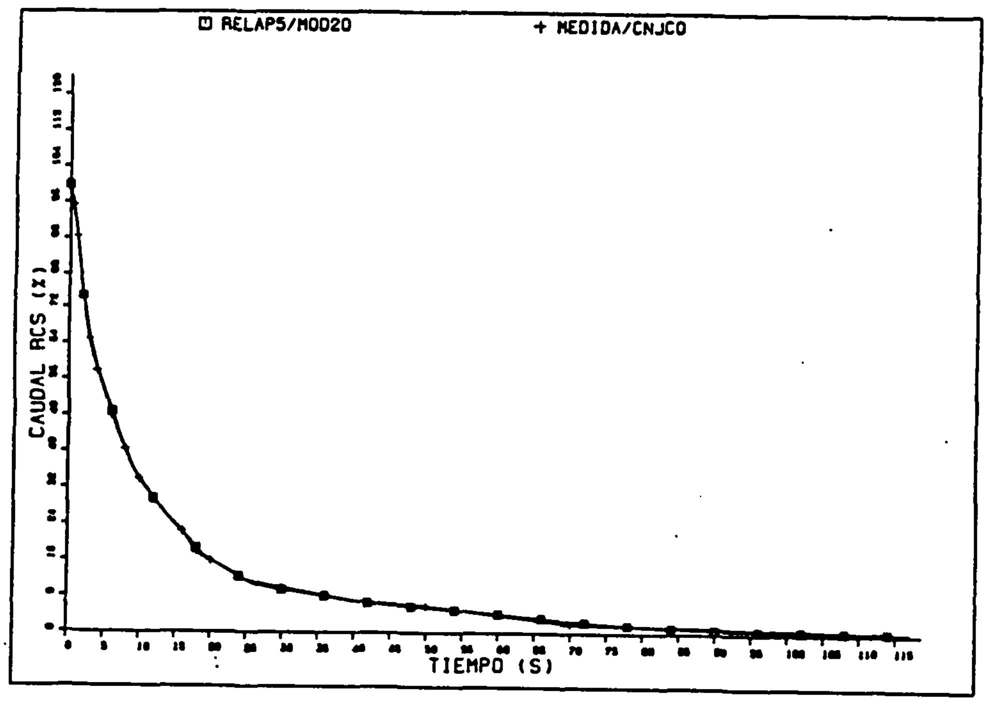


FIG. 4.3. RCP COASTDOWN. PERCENTAGE OF THE NOMINAL FLOW. LONG-TERM.

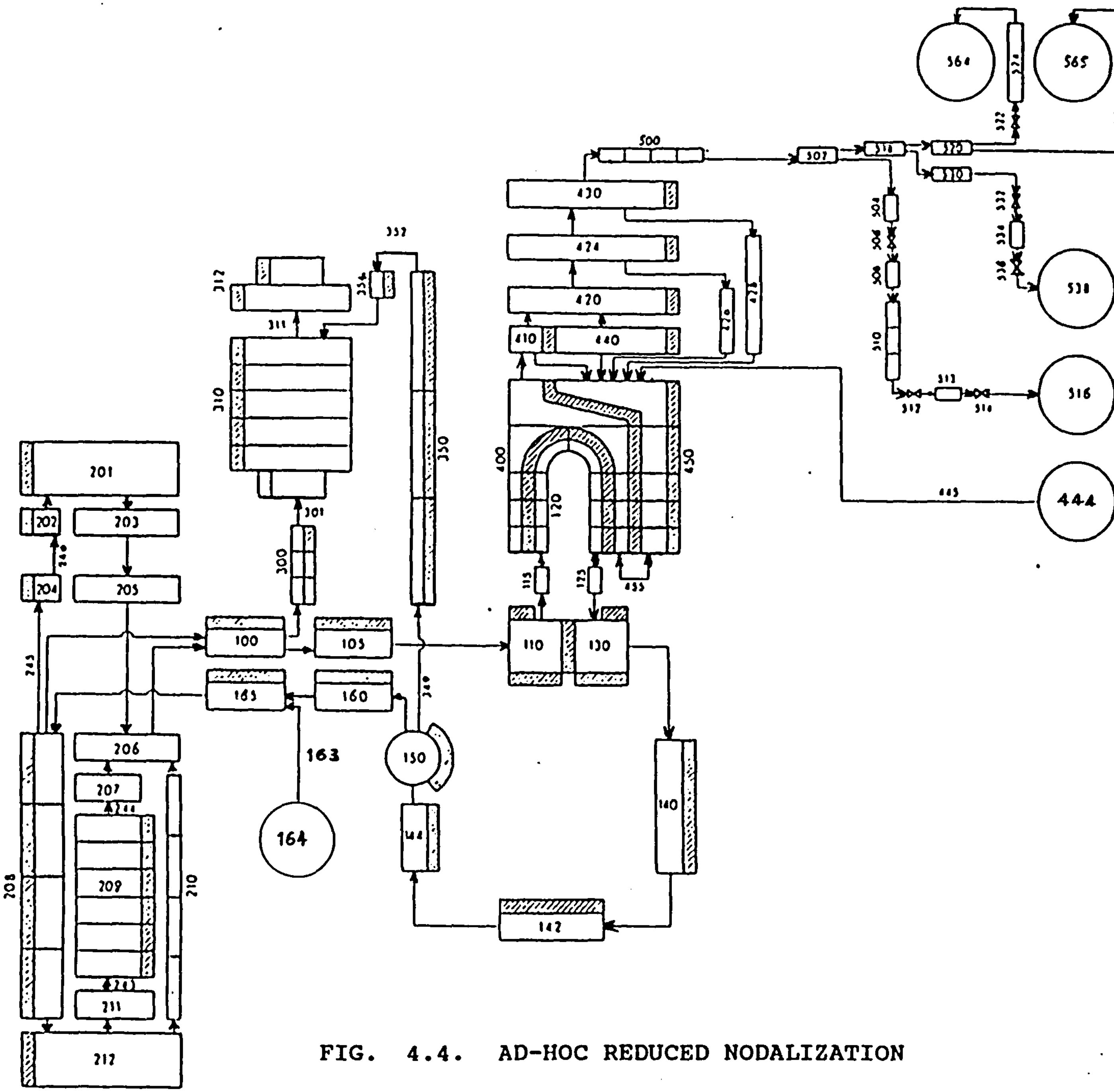
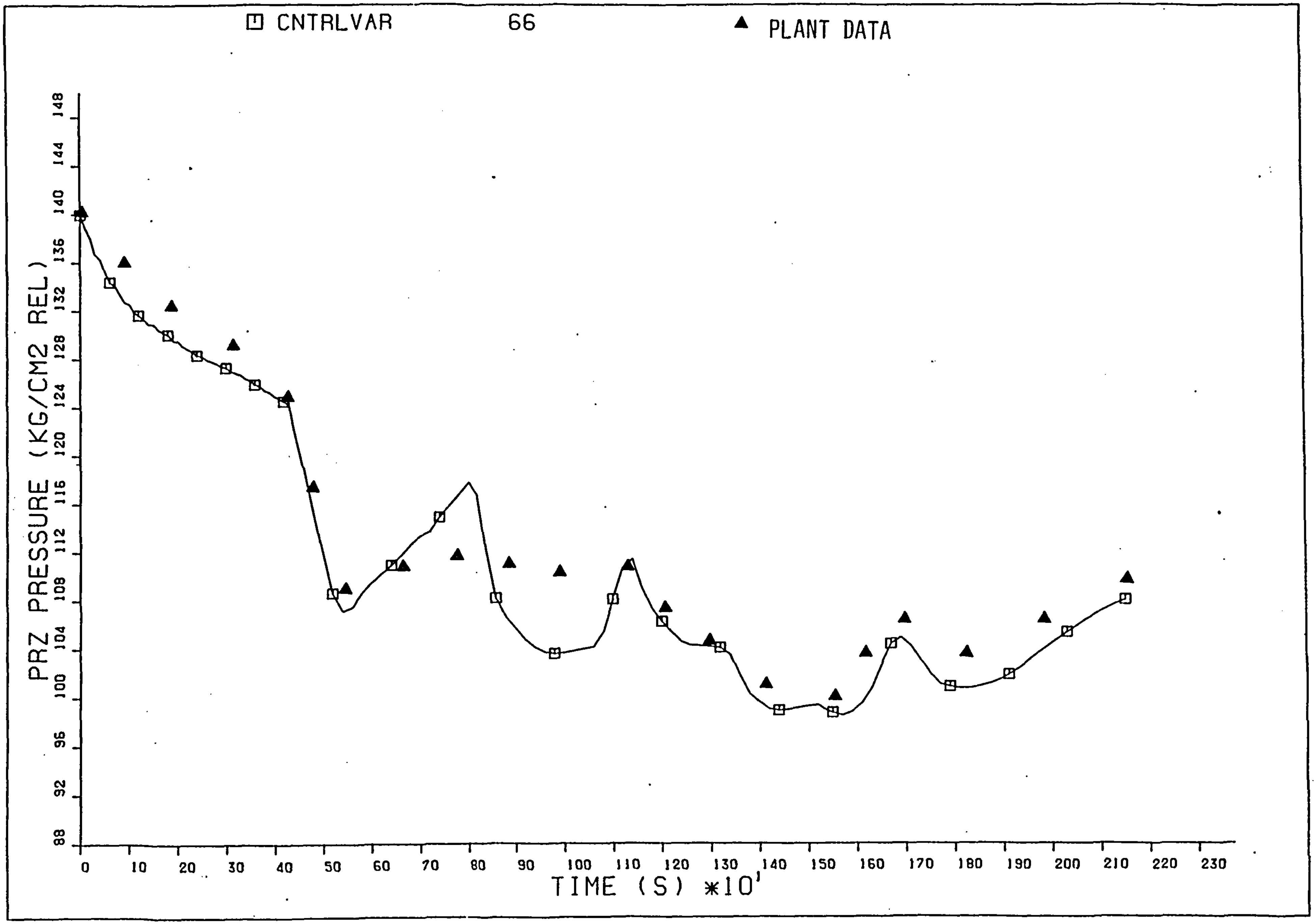
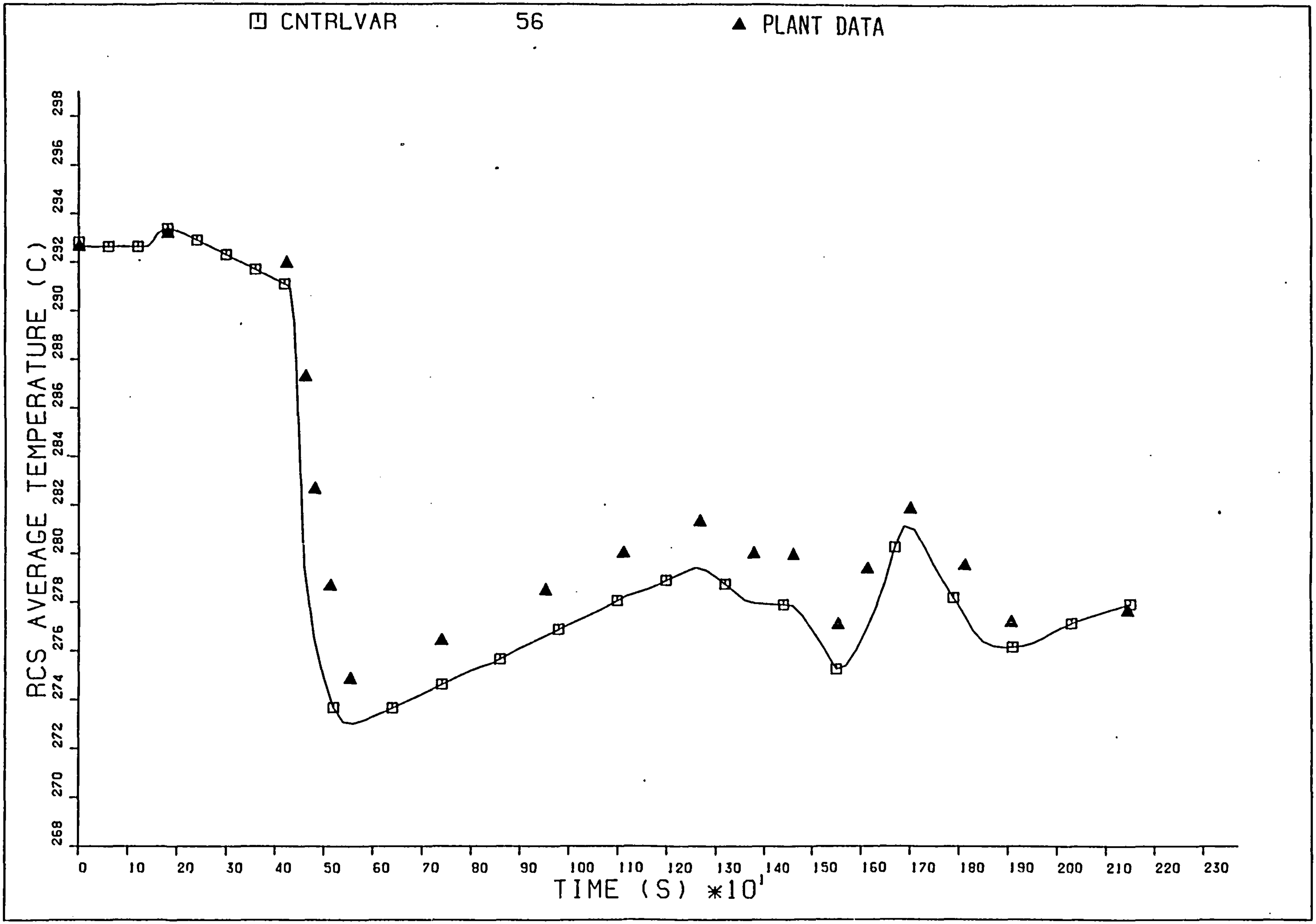


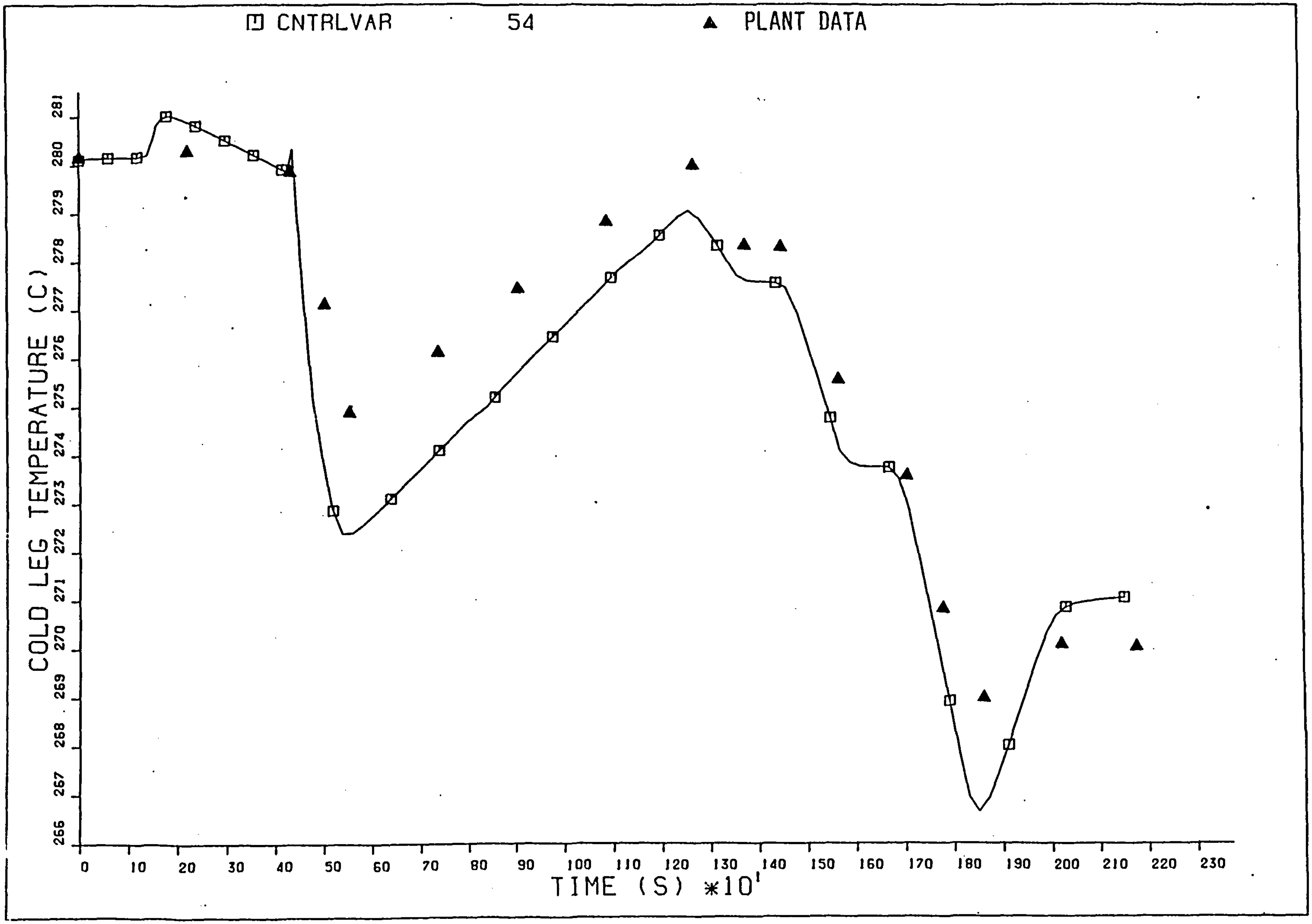
FIG. 4.4. AD-HOC REDUCED NODALIZATION



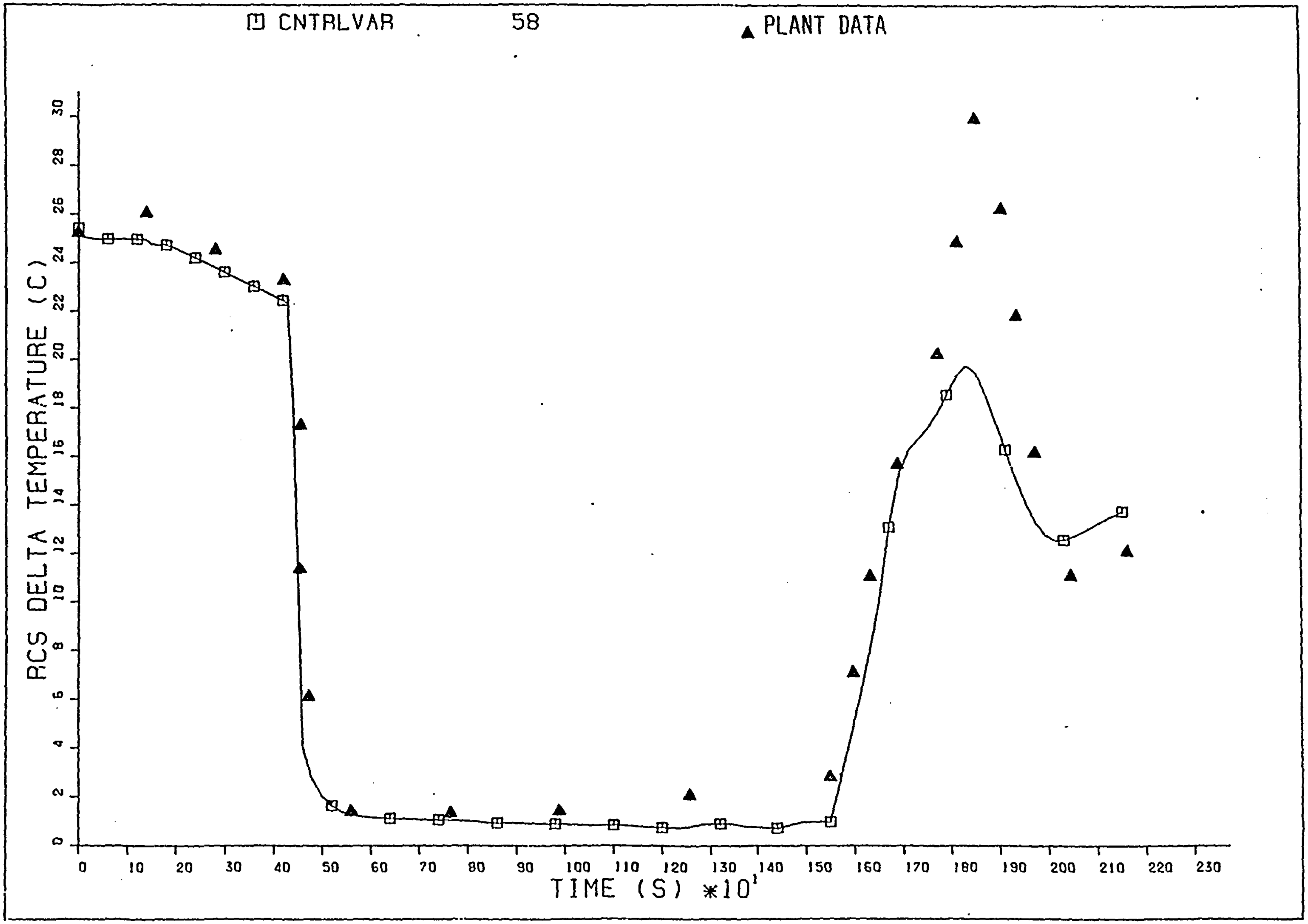
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.1)



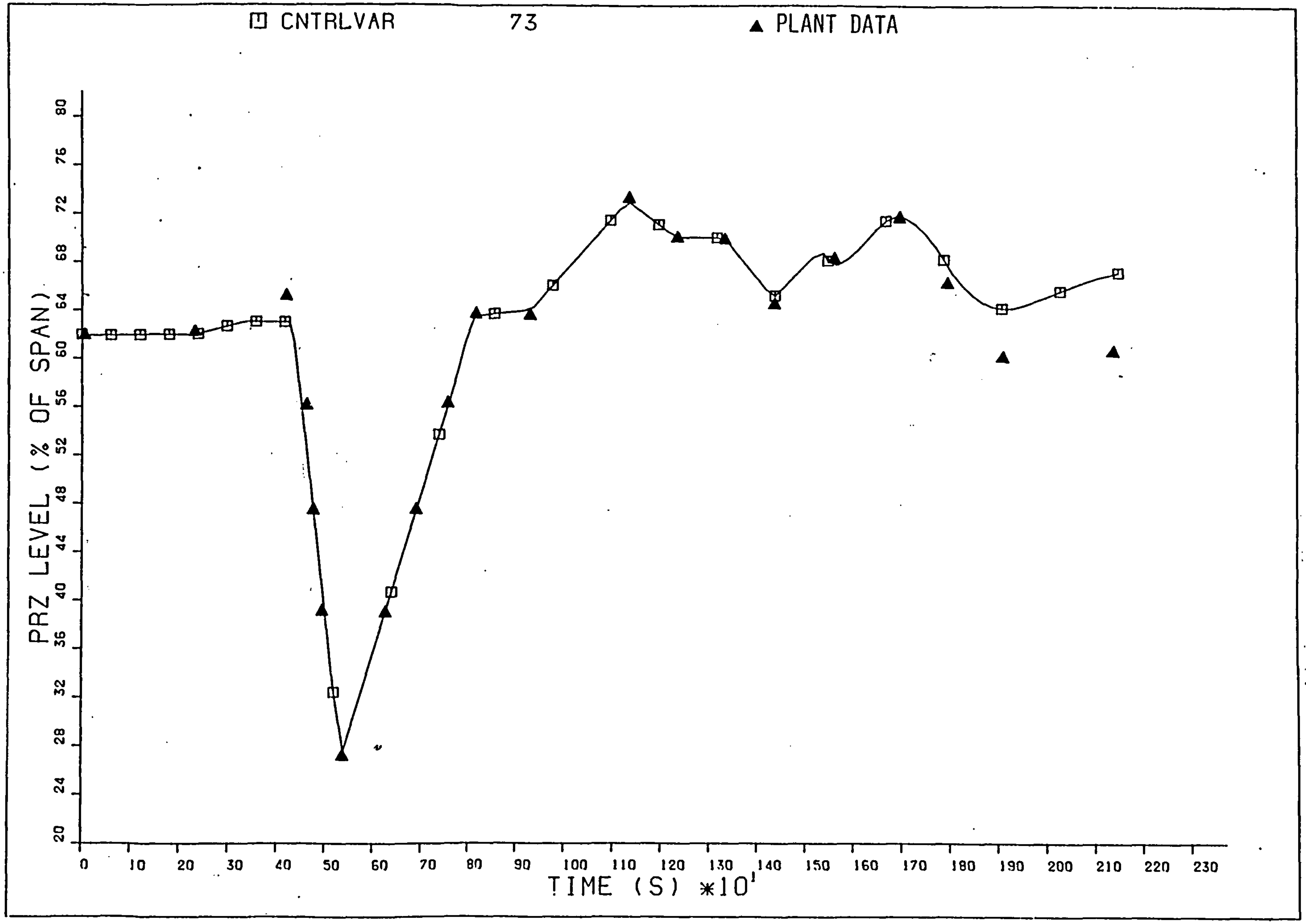
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.2)



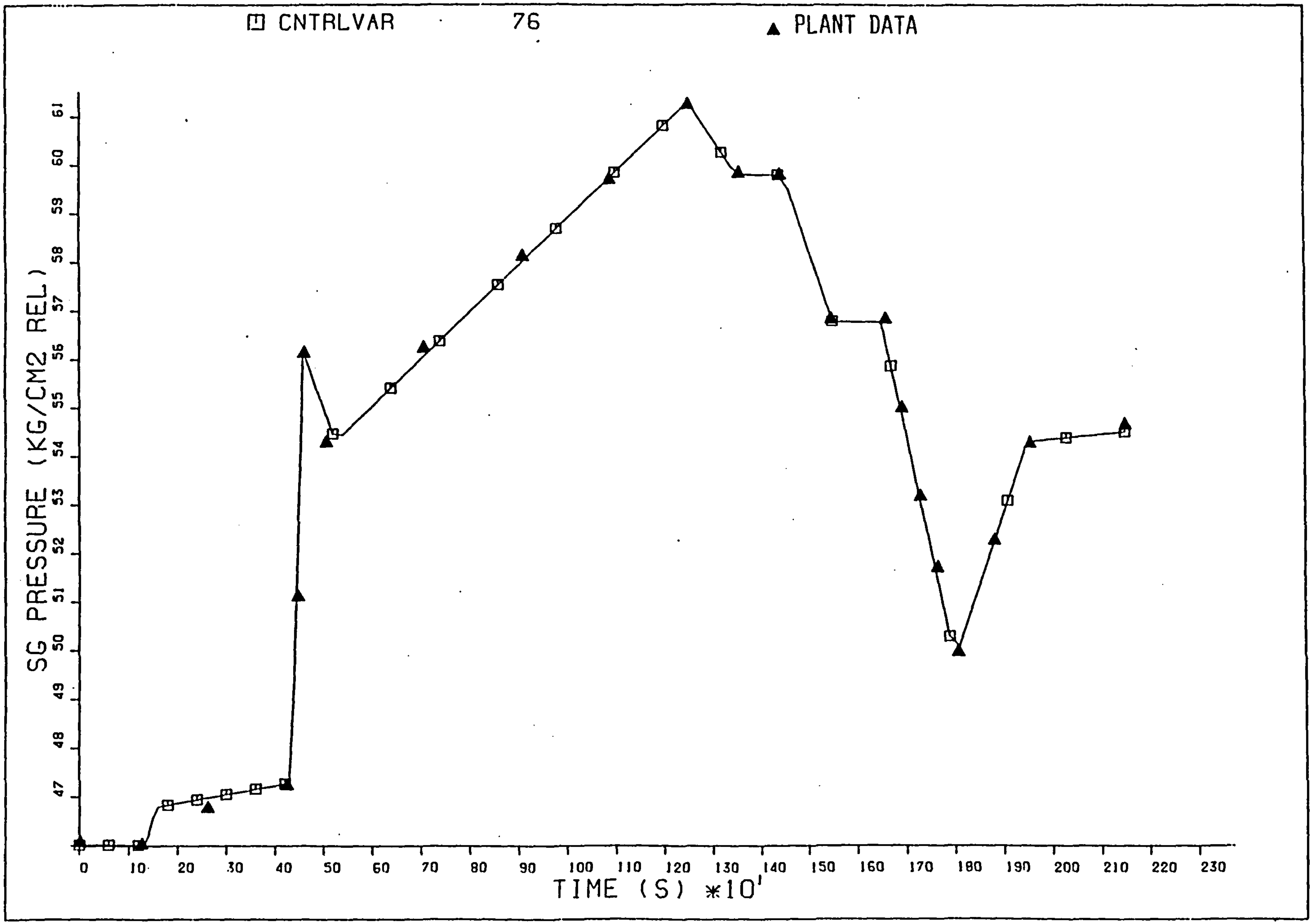
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6 3)



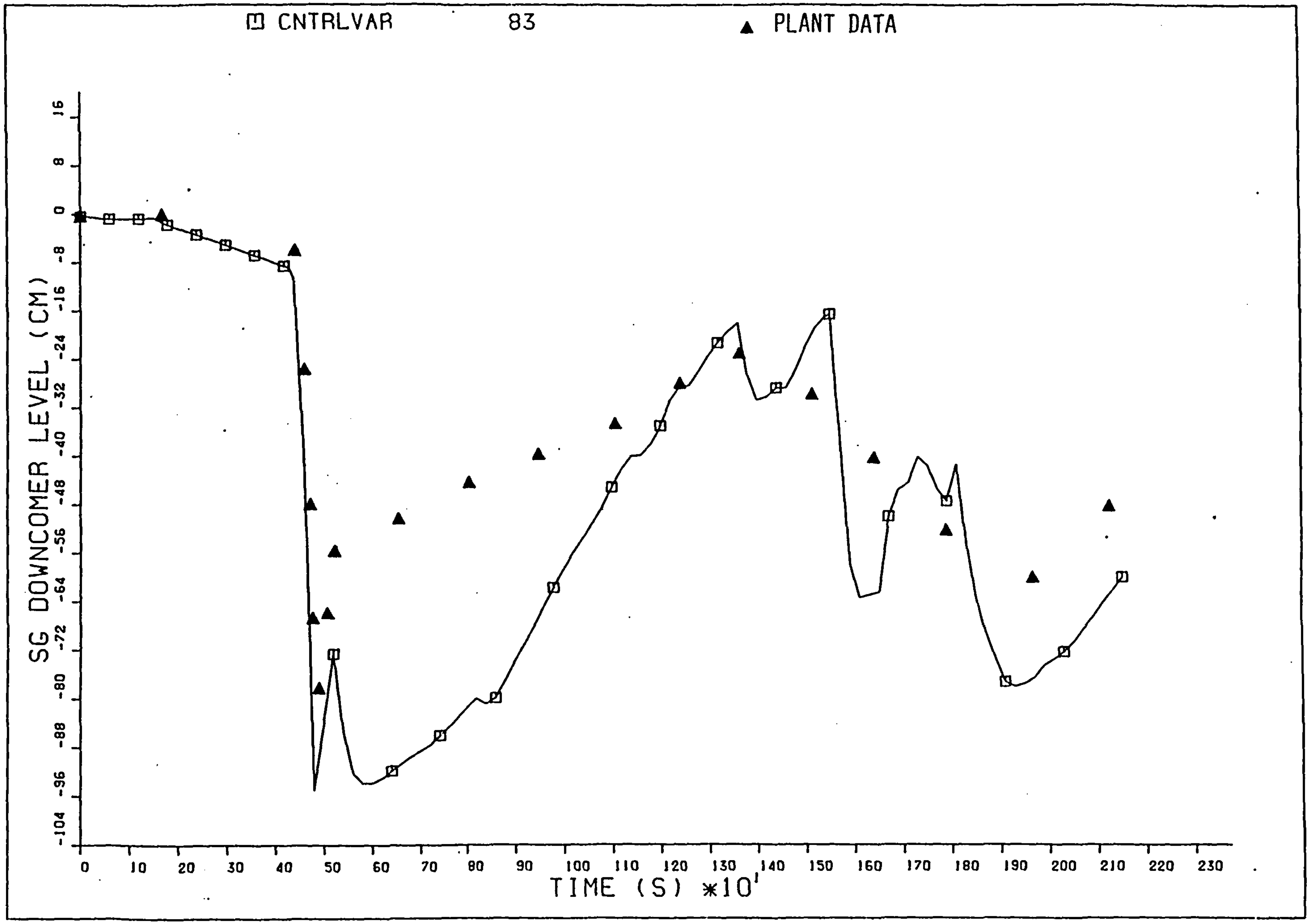
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.4)



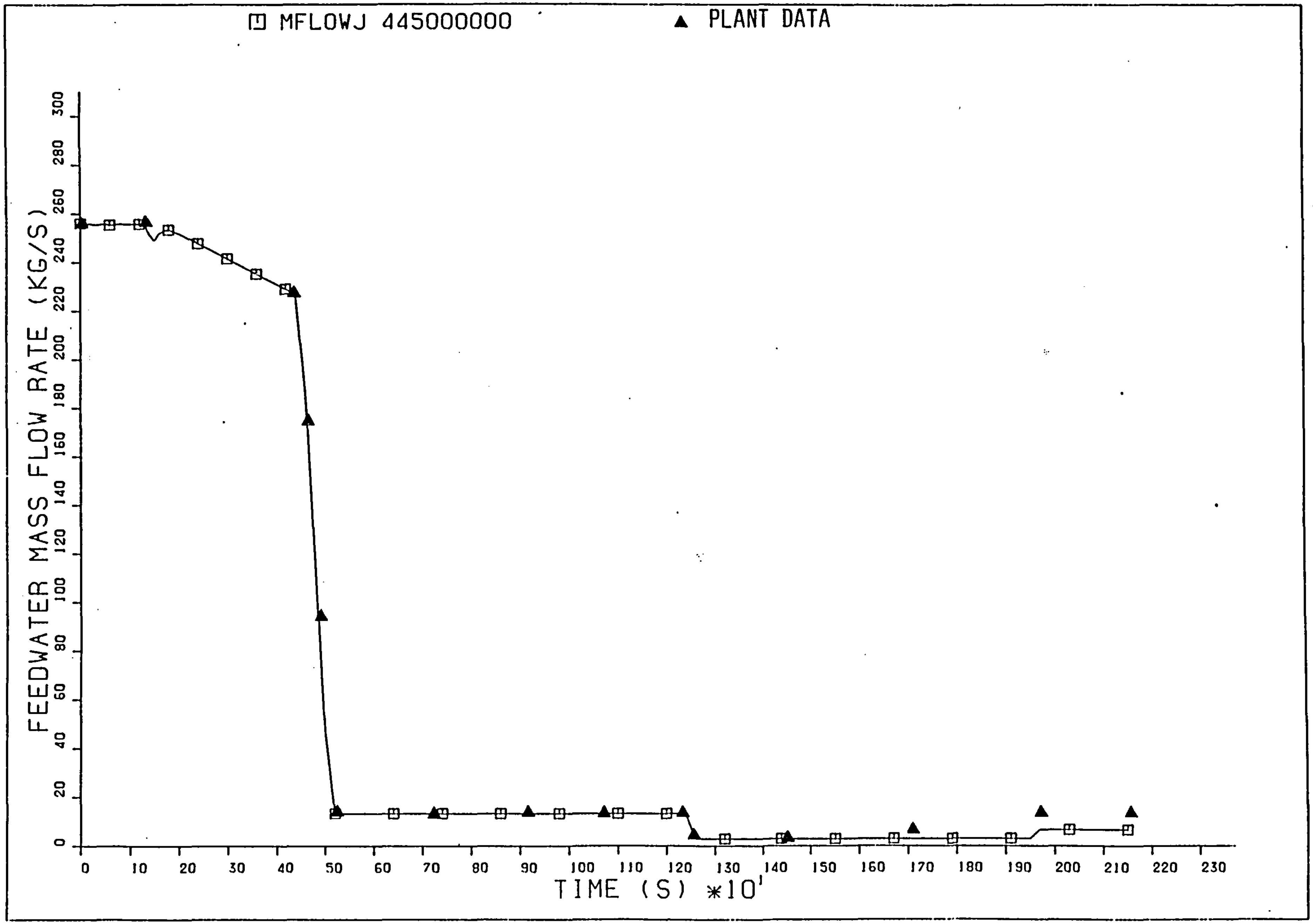
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.5)



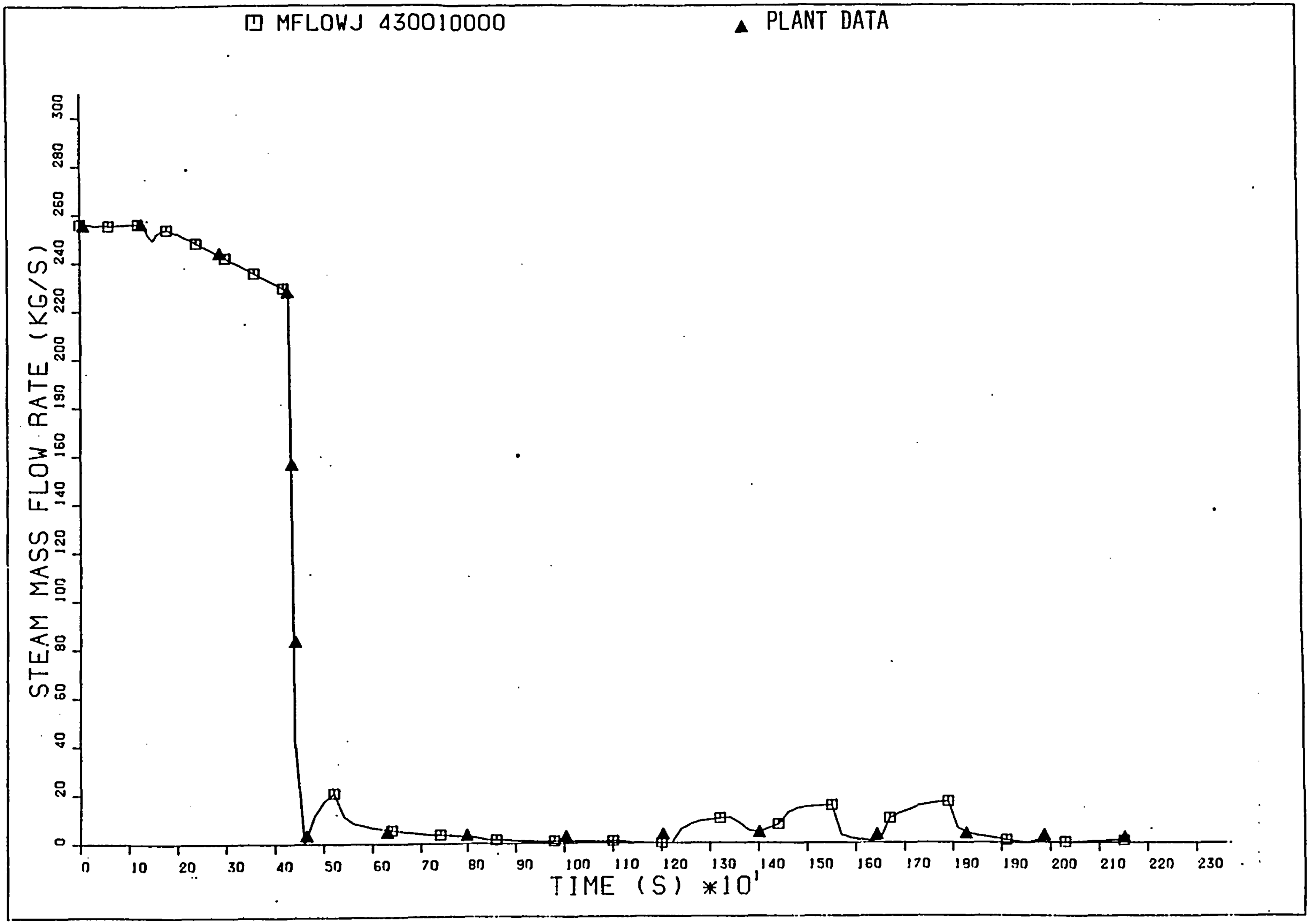
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.6)



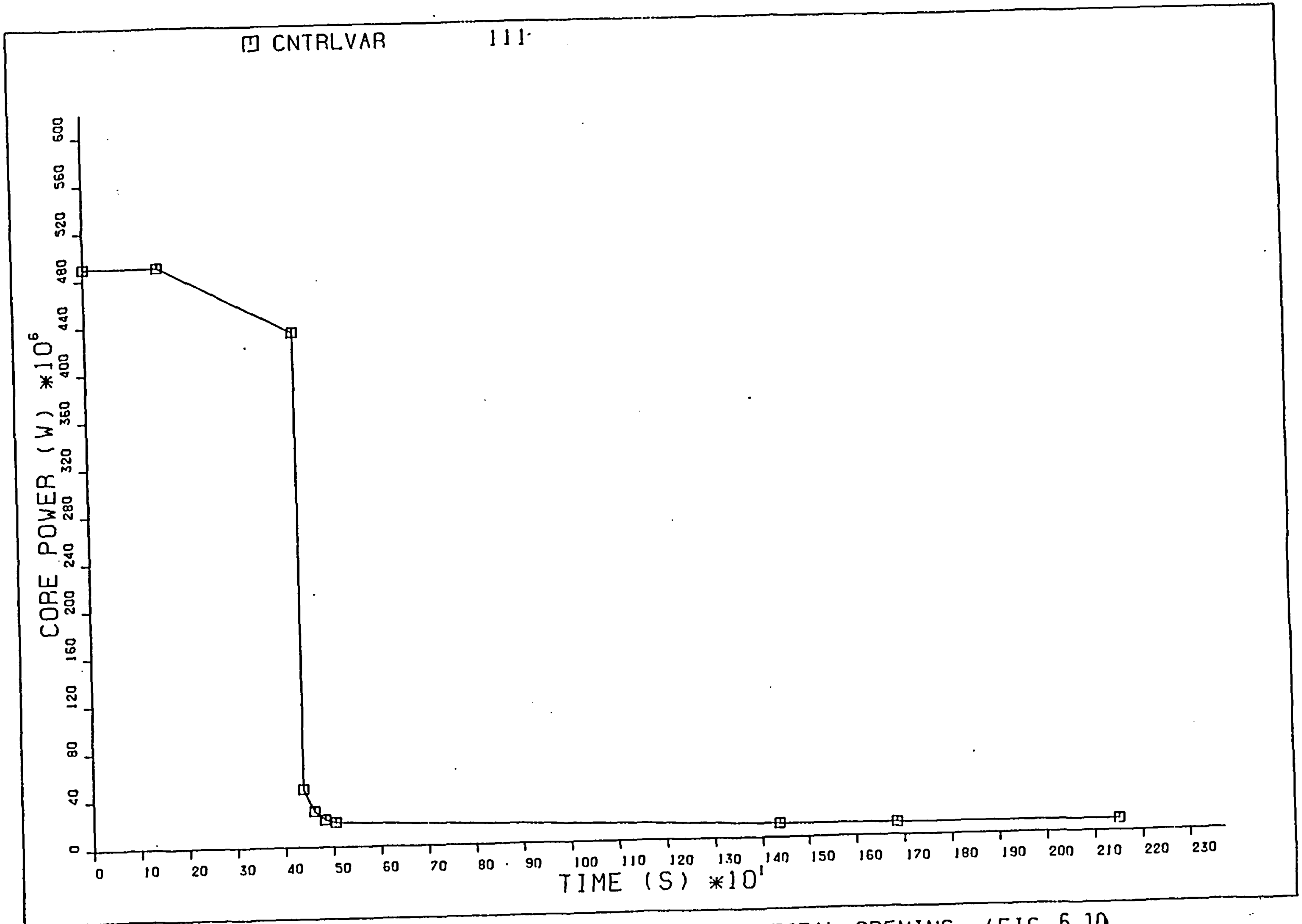
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.7)



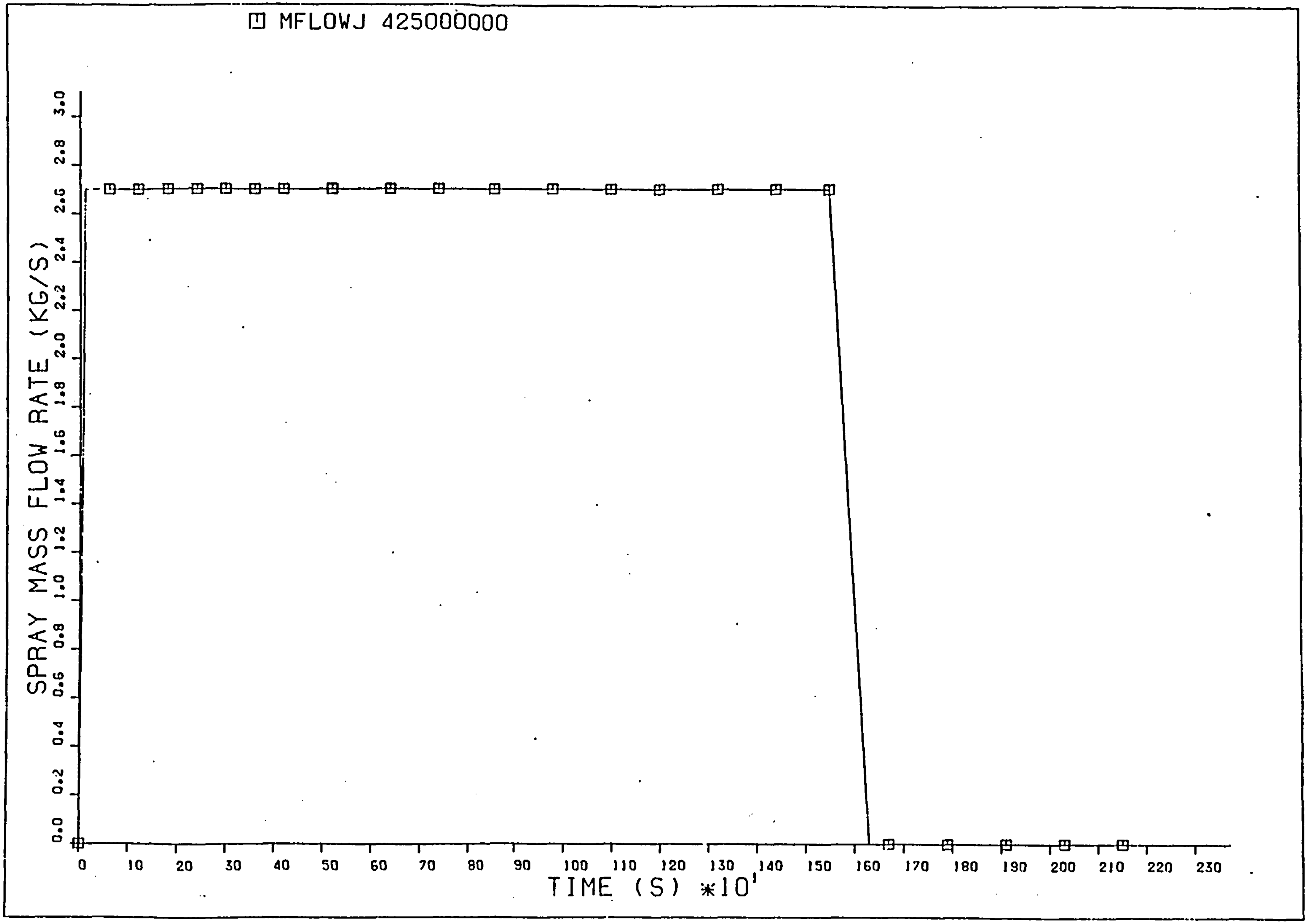
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.8)



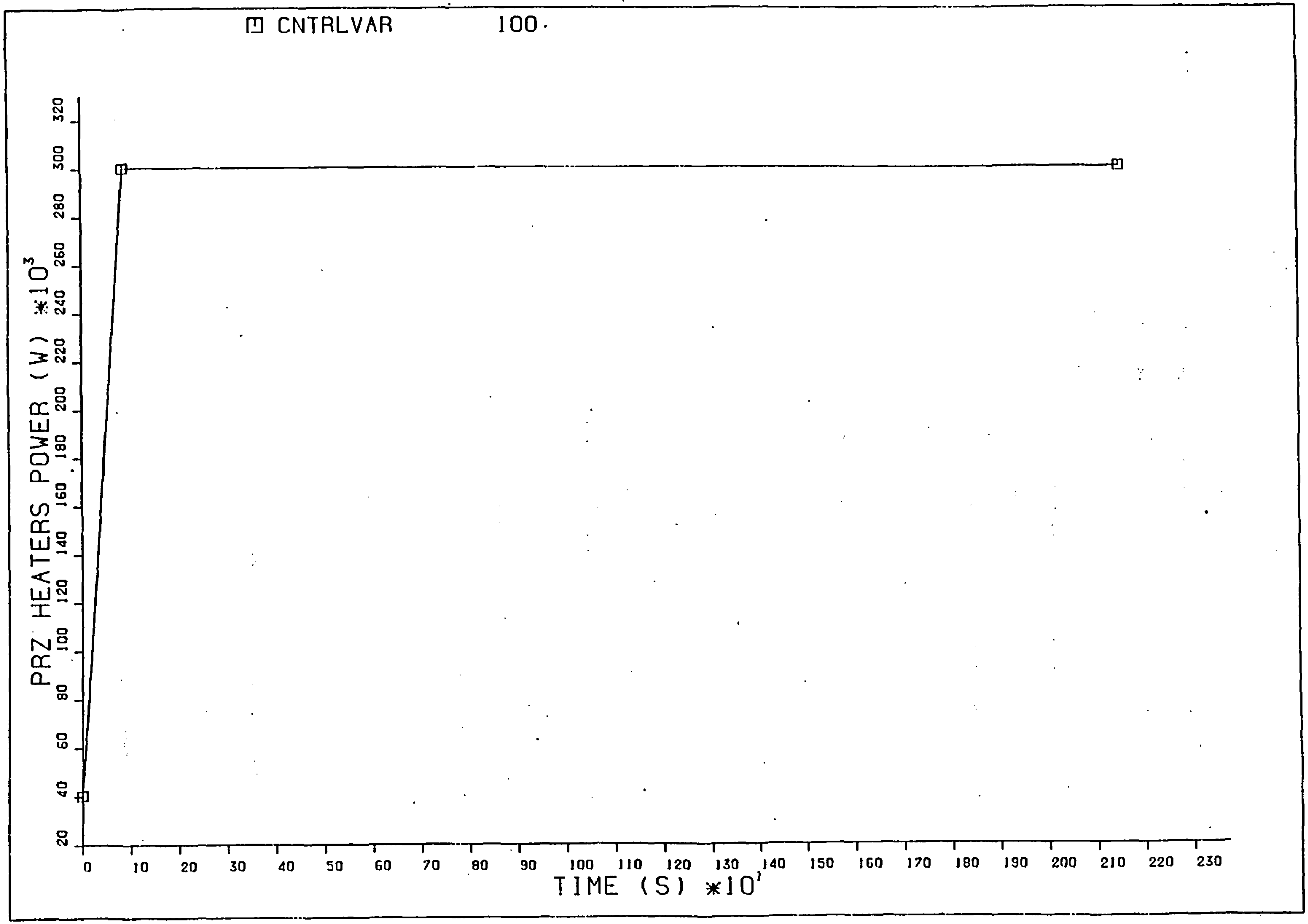
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.9)



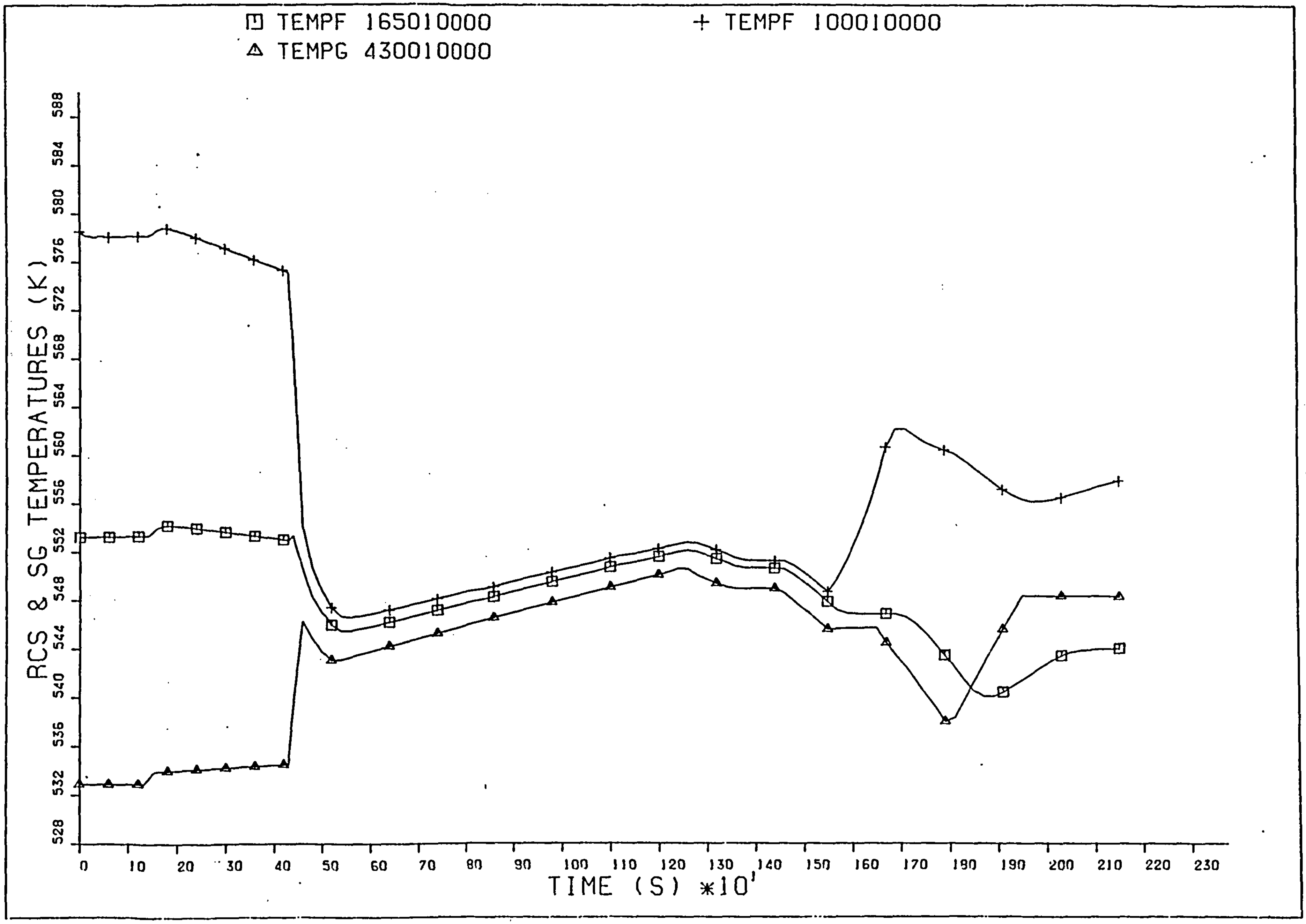
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.10)



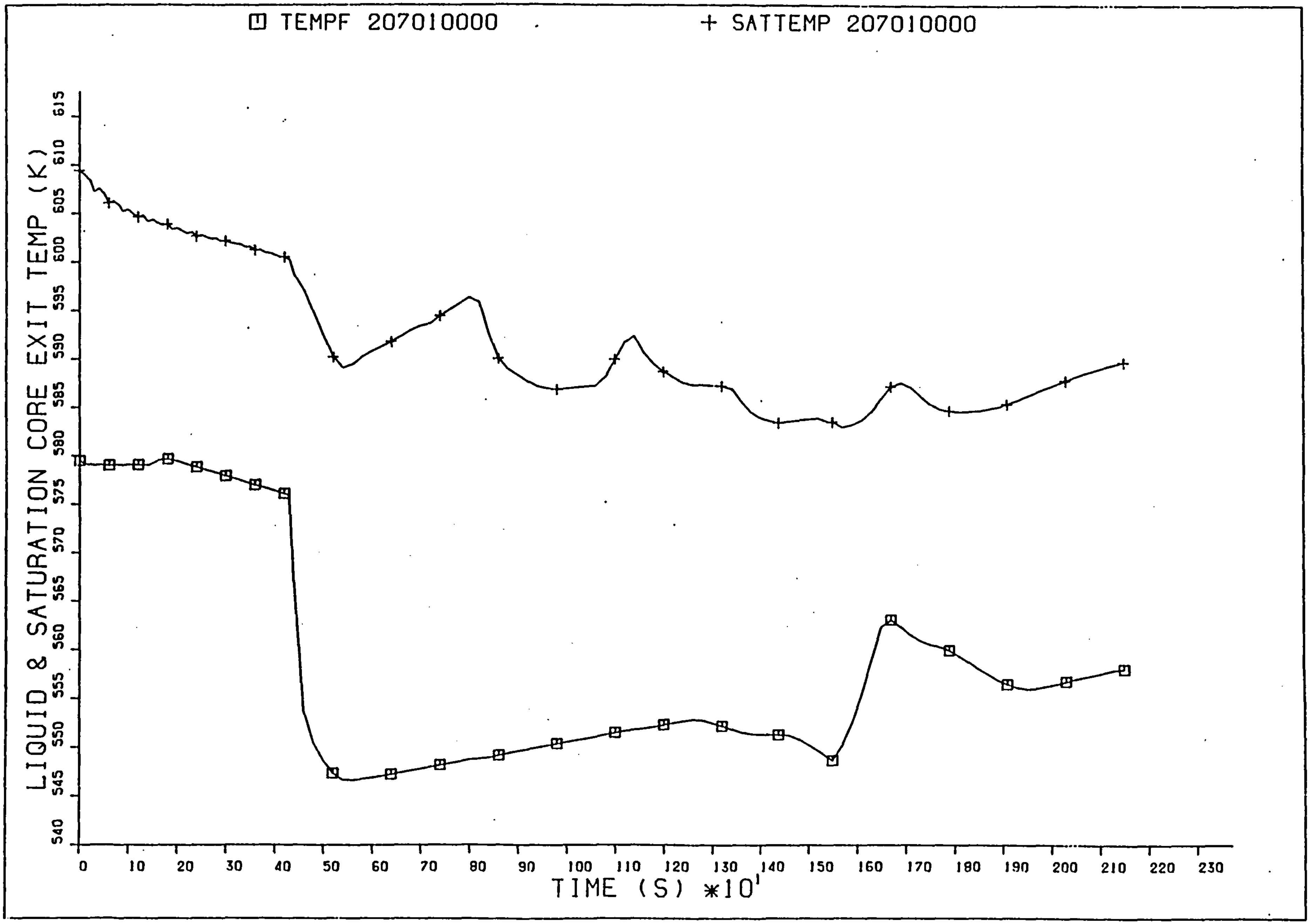
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG6.11)



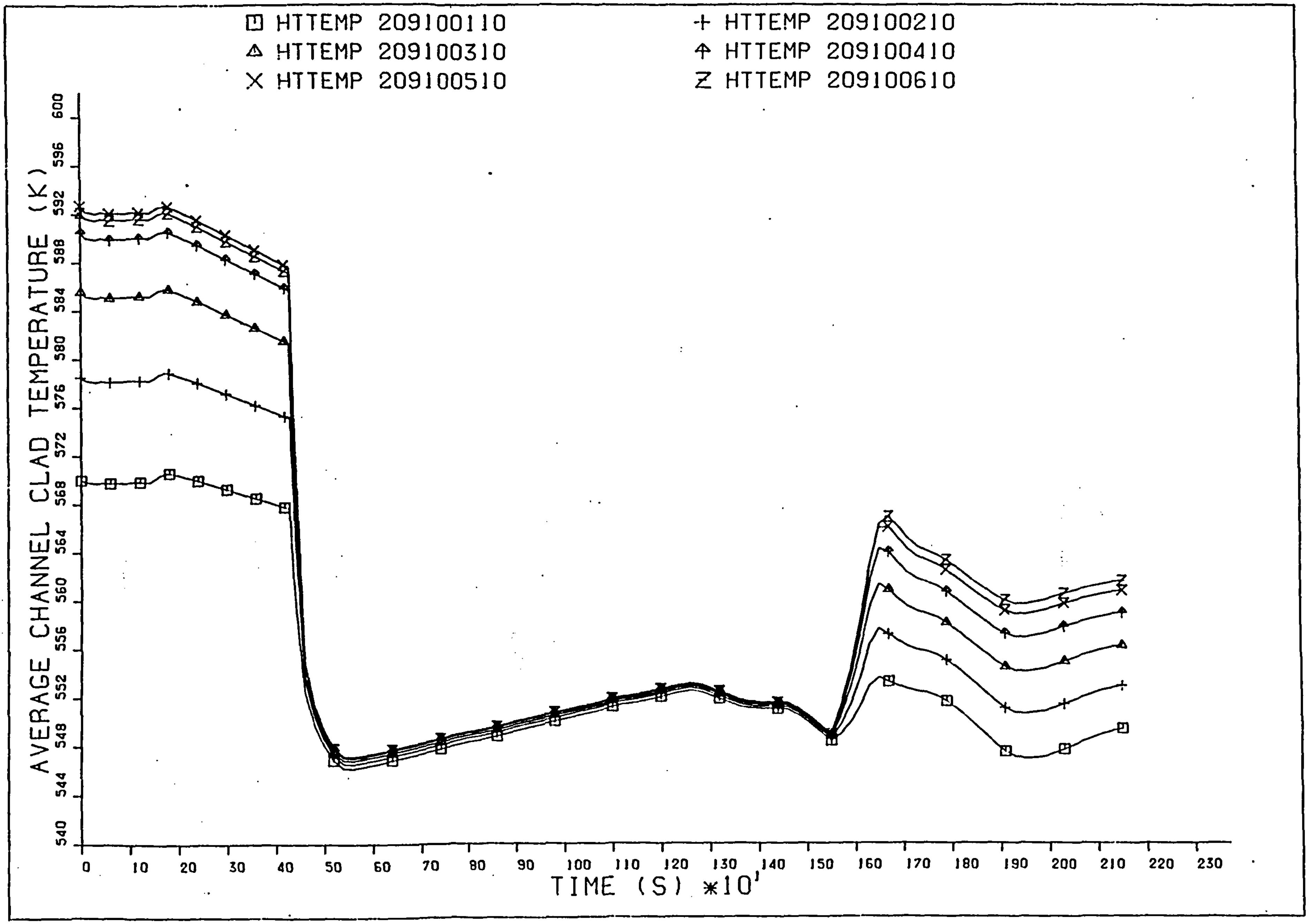
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.12)



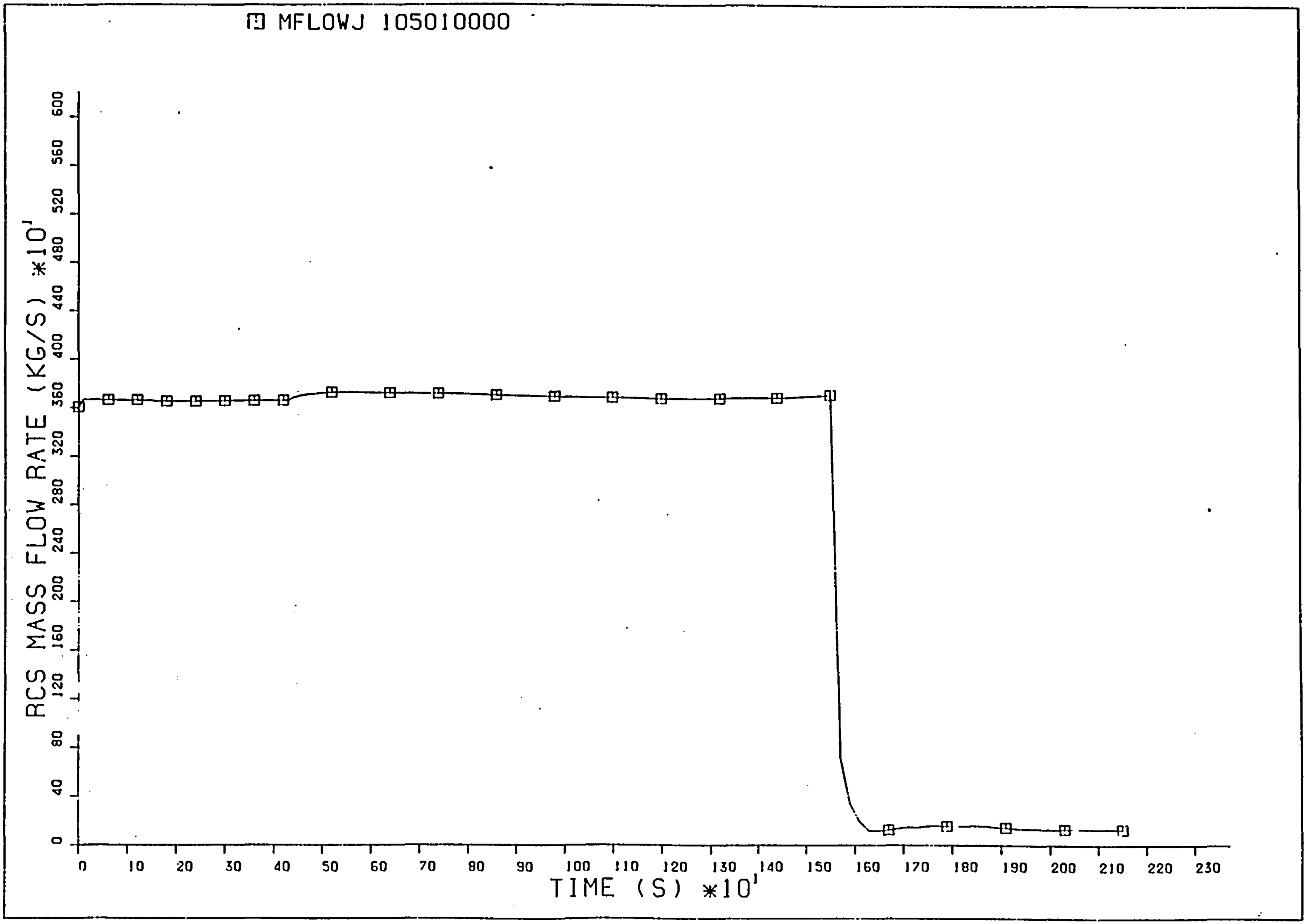
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.13)



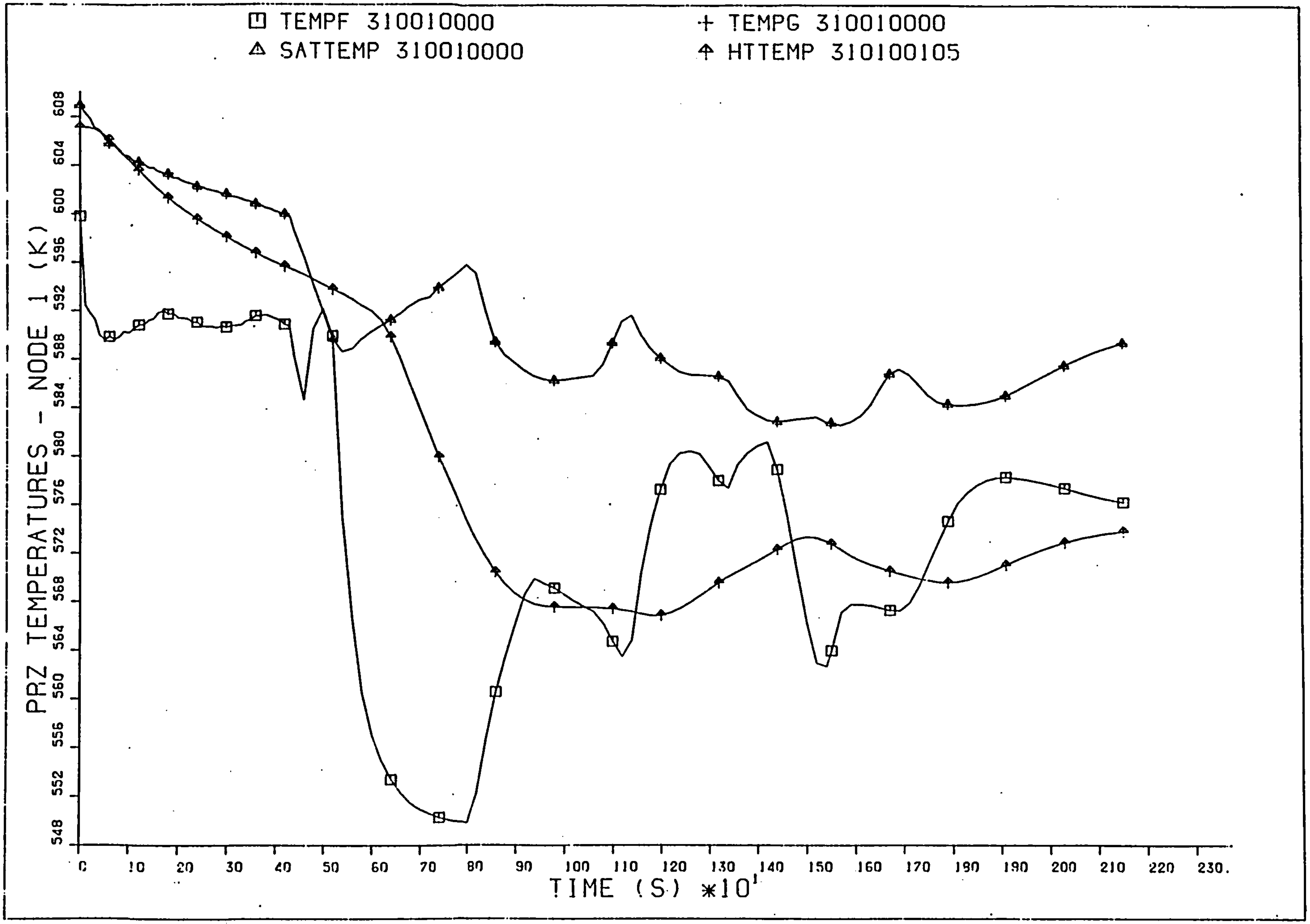
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.14)



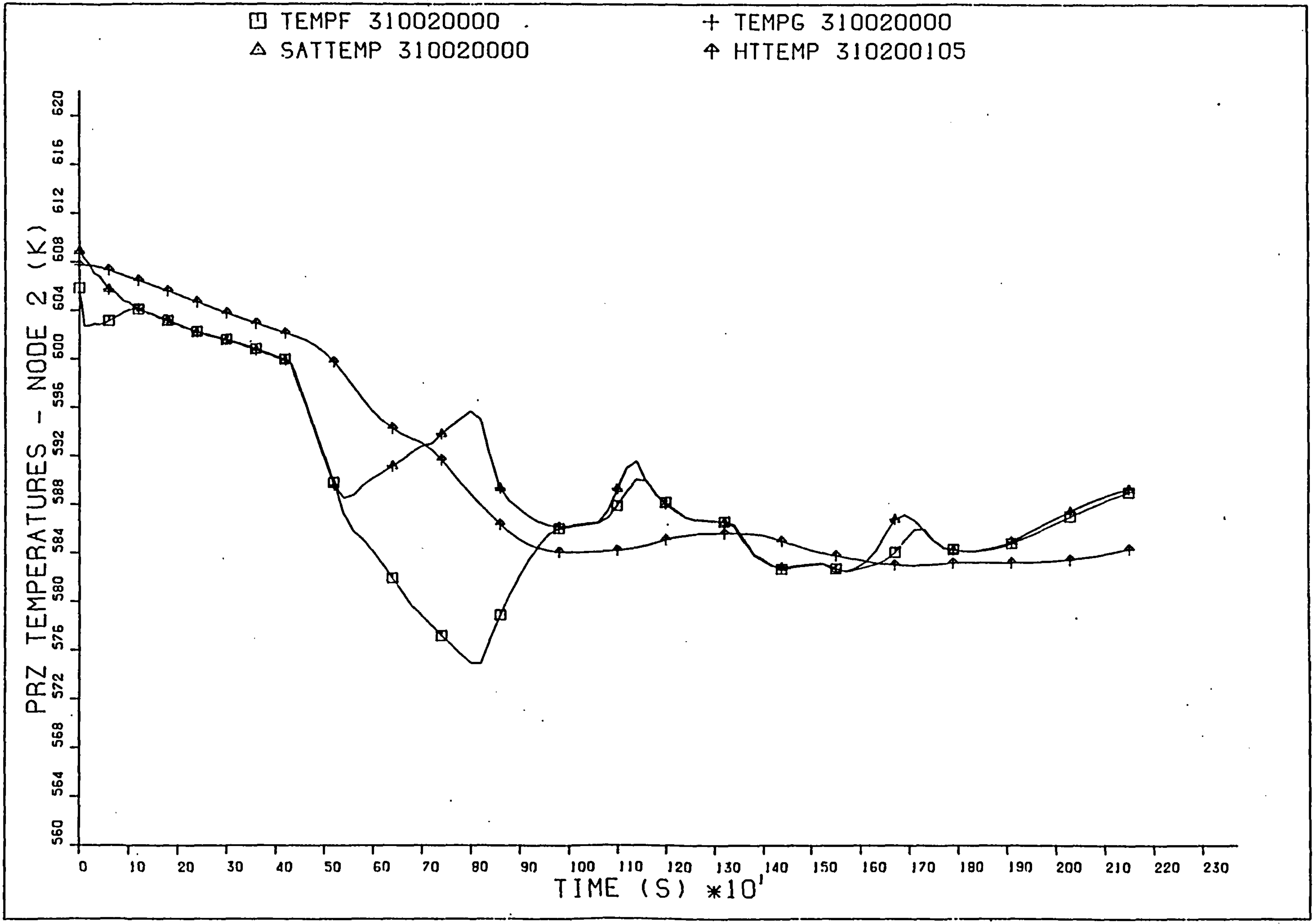
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.15)



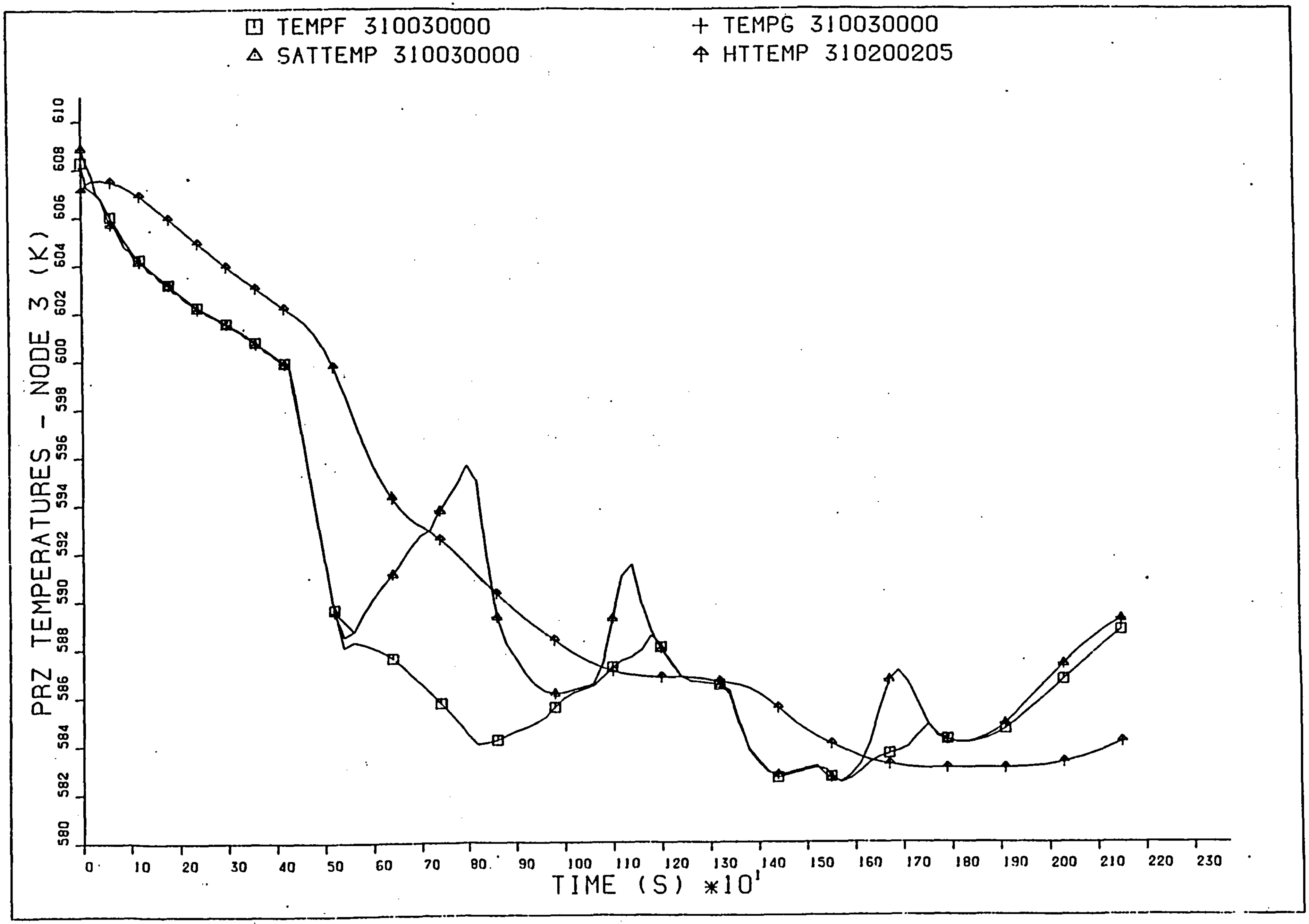
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.16)



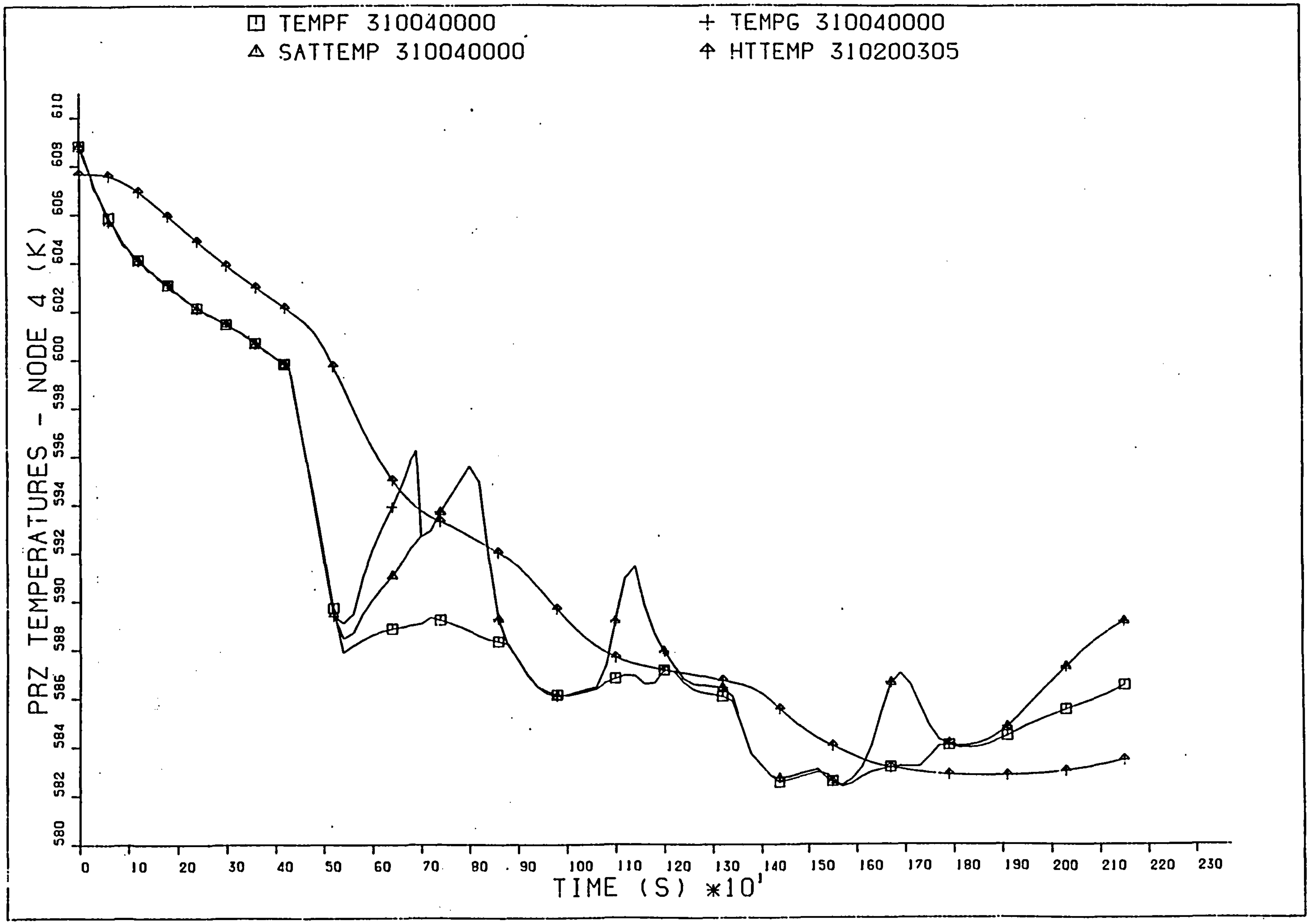
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.17)



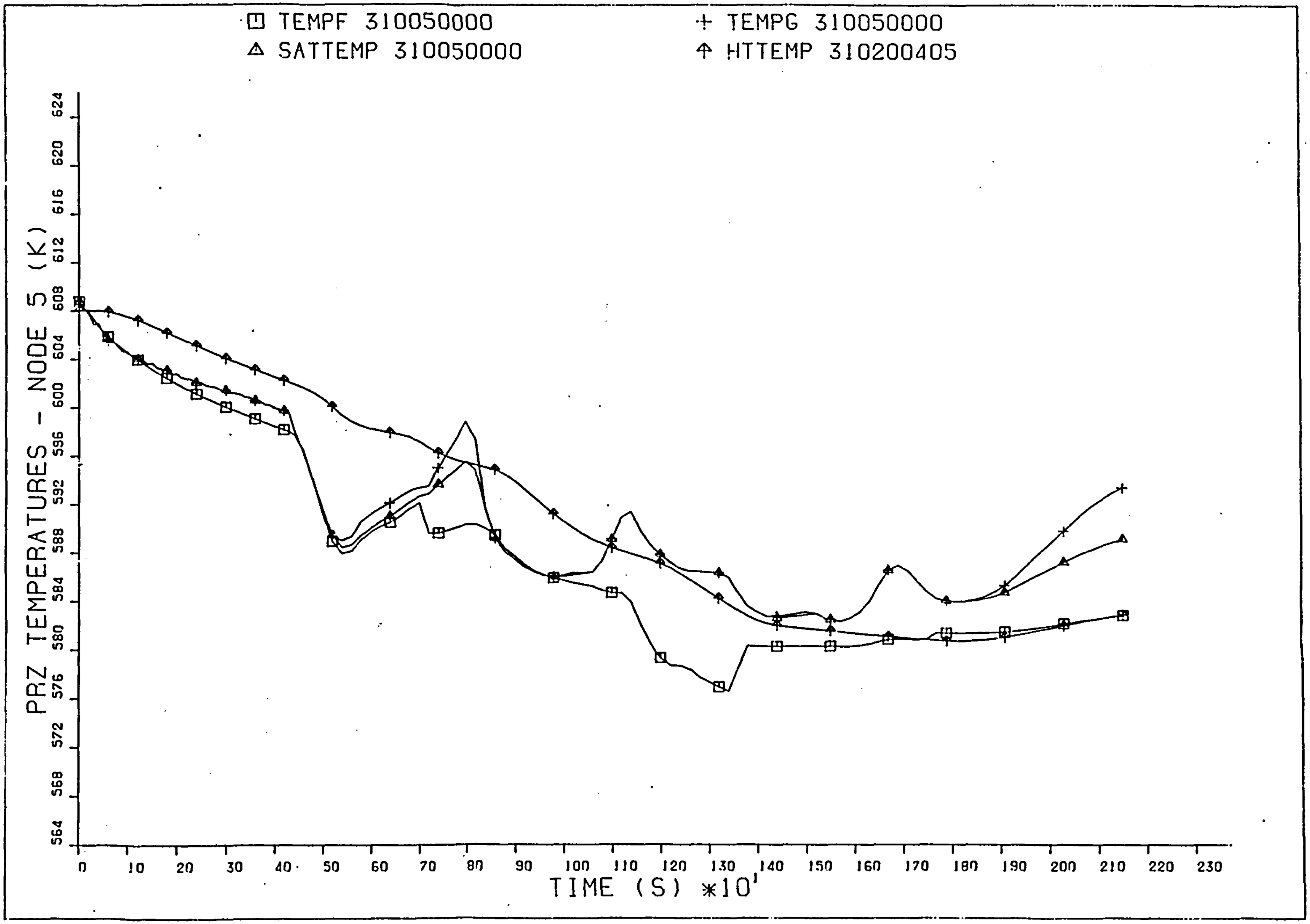
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.18)



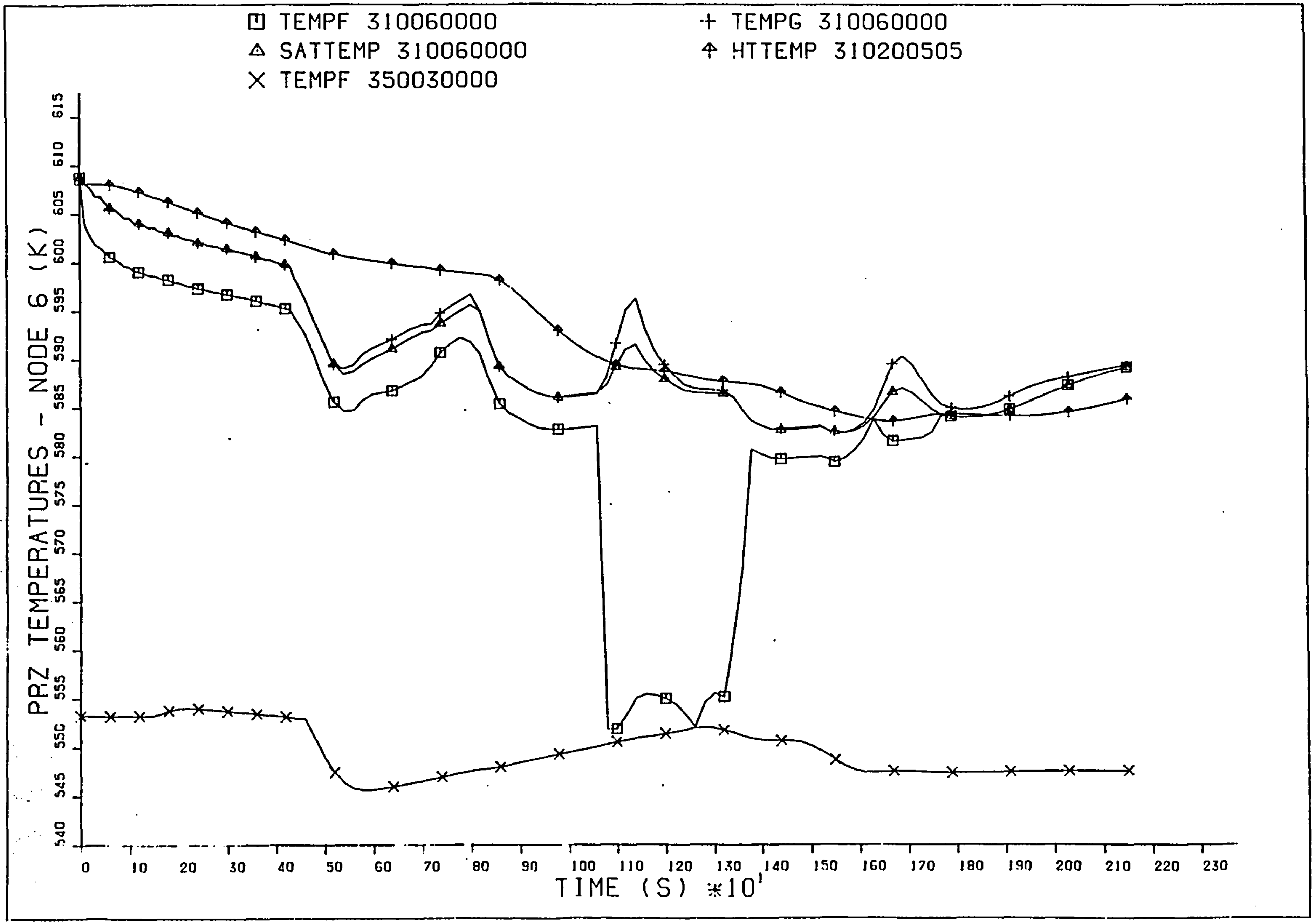
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.19)



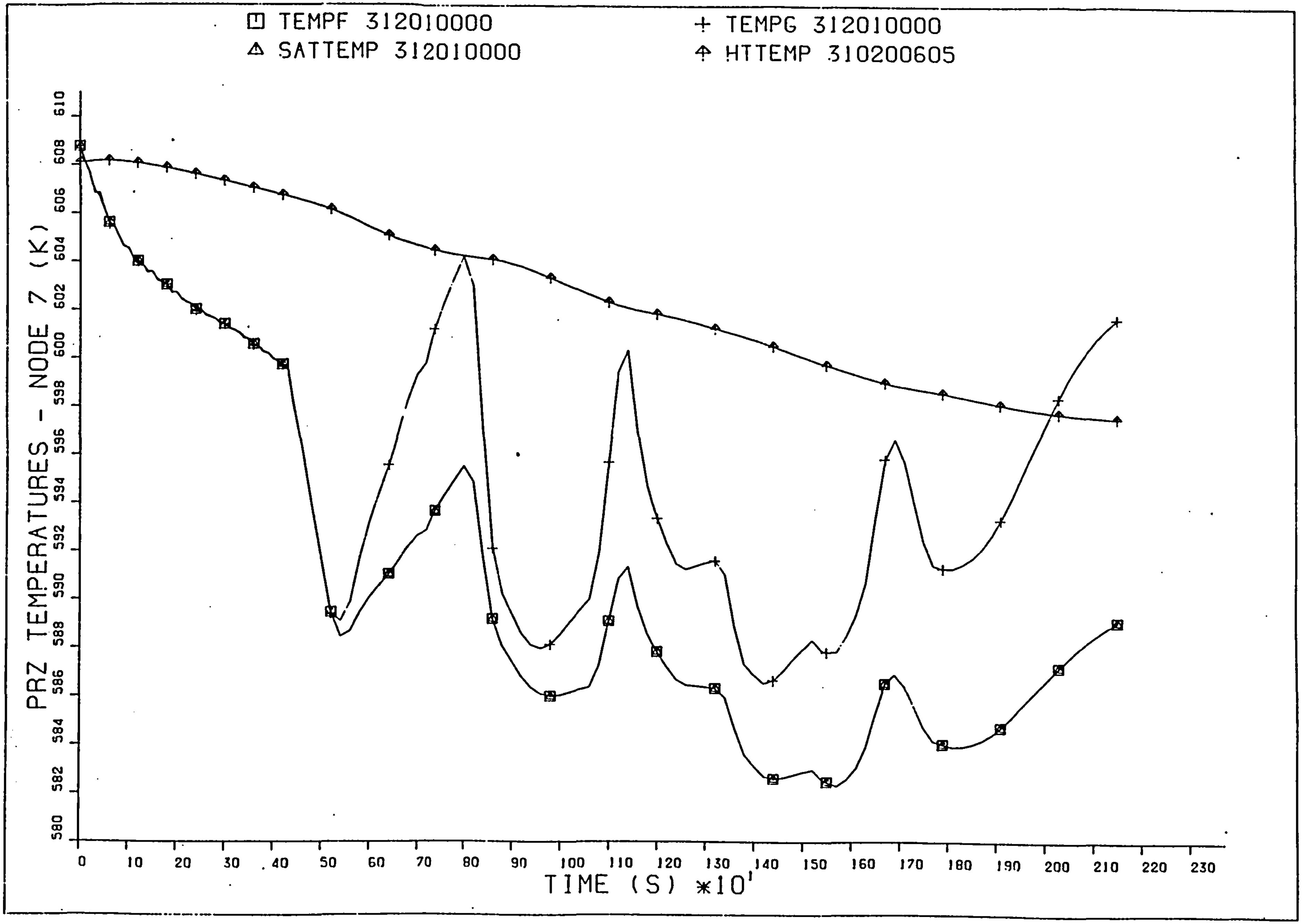
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.20)



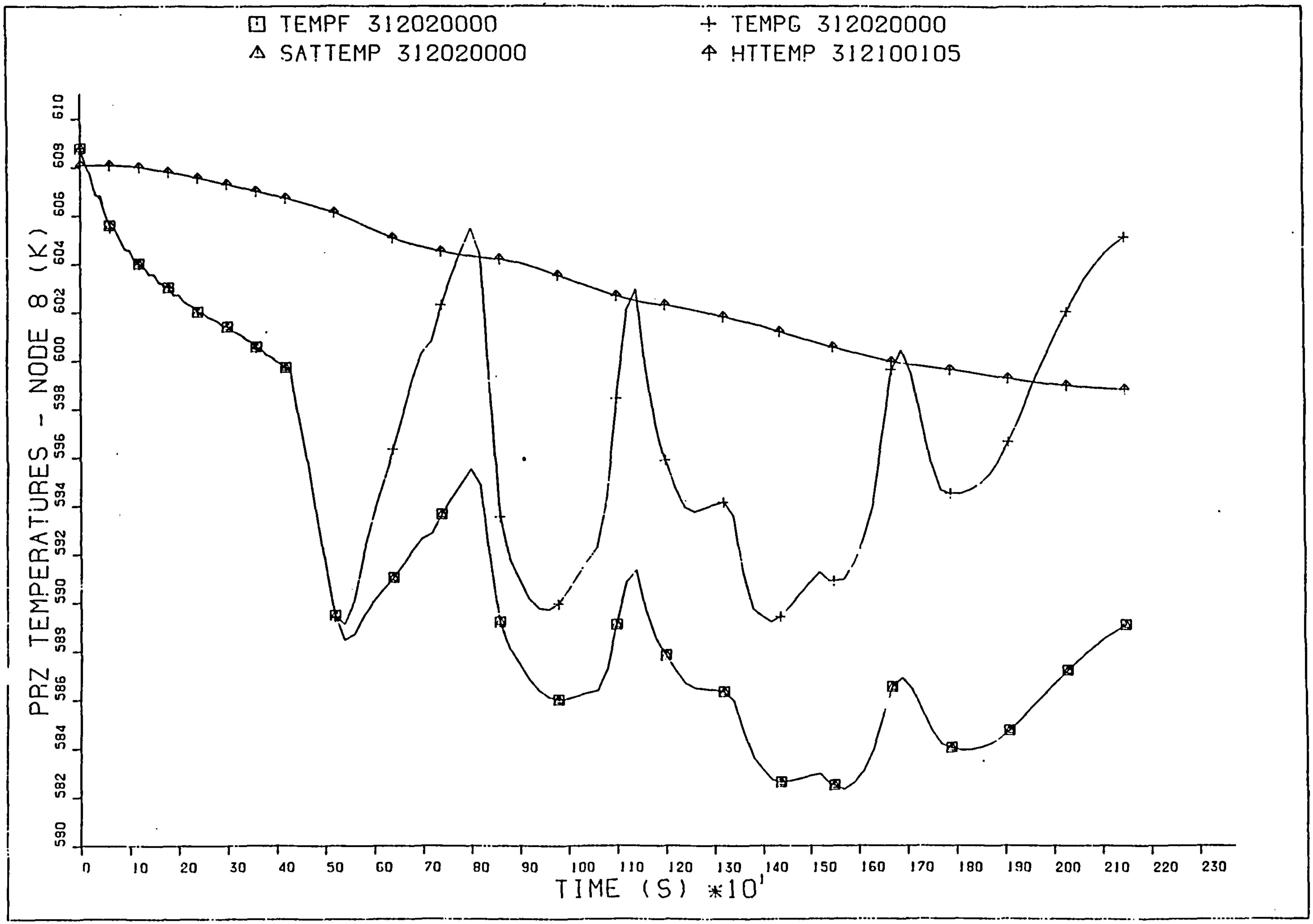
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.21)



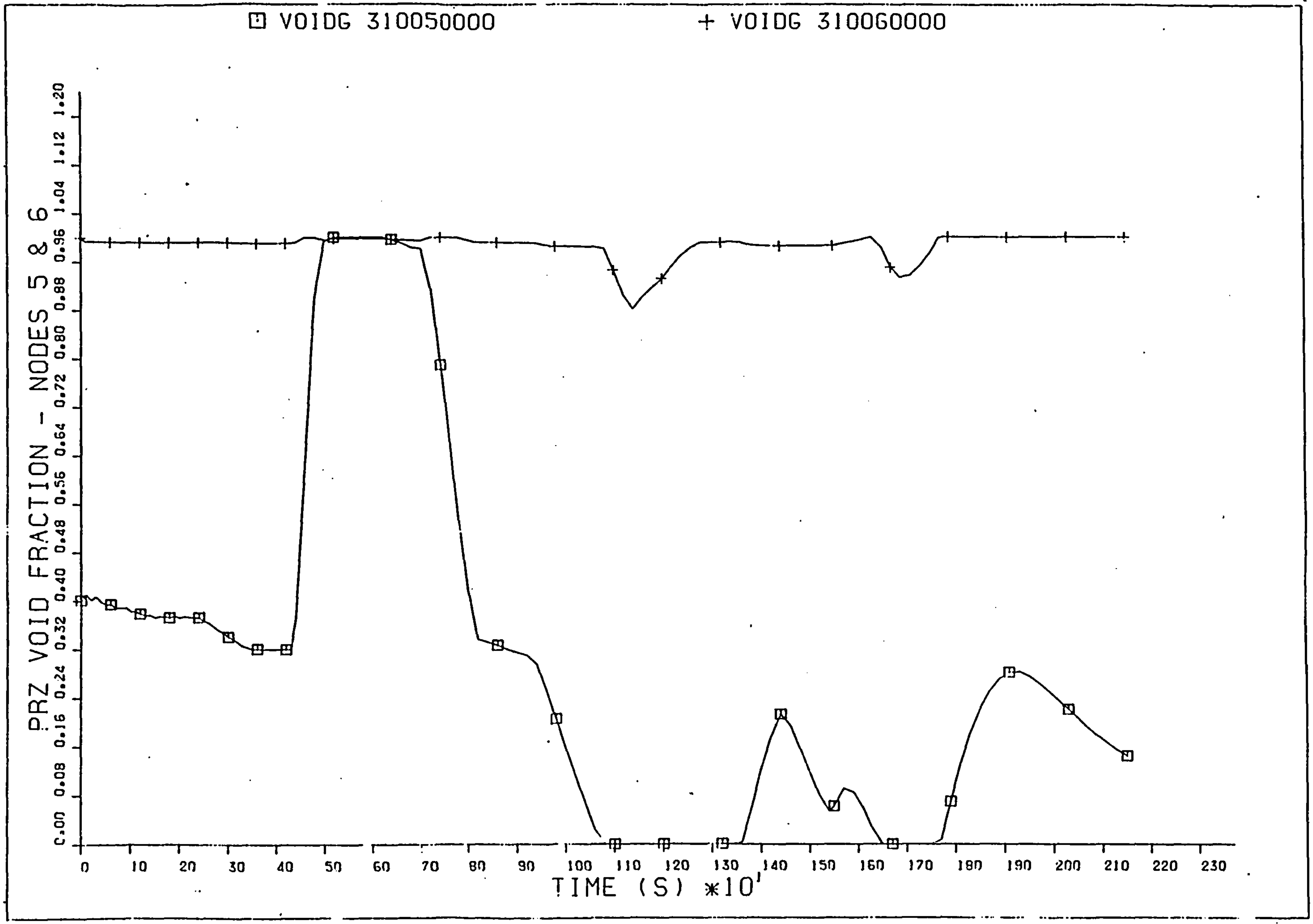
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.22)



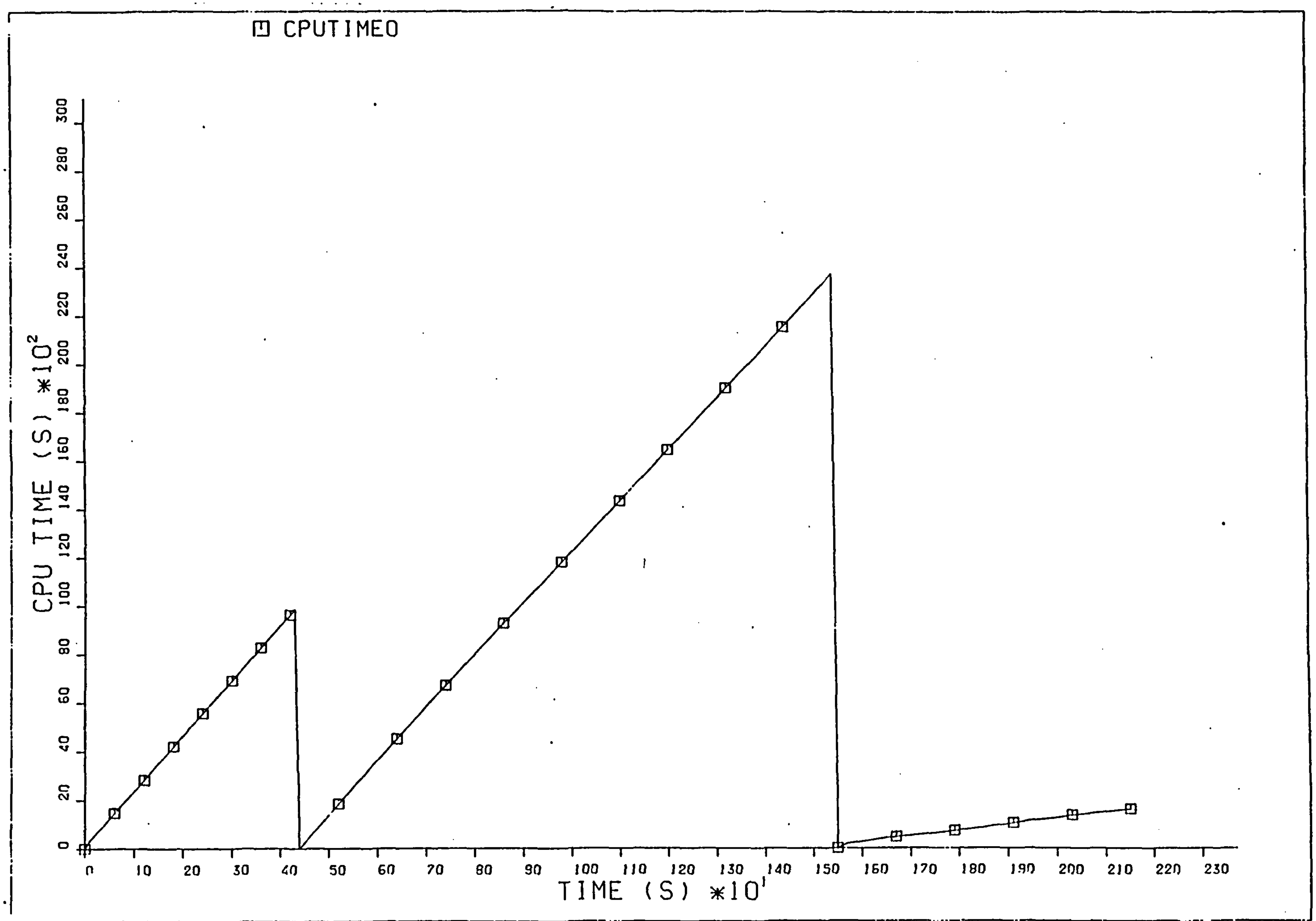
UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG6.23)



UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.24)



UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 6.25)



UNION-FENOSA ICAP INADVERTED PRZ SPRAY TOTAL OPENING (FIG 7.1)

ADDENDUM A

ASSESSMENT OF RELAP5/MOD2 AGAINST PRESSURIZER SEPARATED EFFECTS EXPERIMENTS

ASSESSMENT OF RELAP5/MOD2 AGAINST PRESSURIZER SEPARATED EFFECTS EXPERIMENTS

To evaluate the RELAP5/MOD2 response to pressurizer filling transients and the interphases impact, two pressurizer separated effects experiments have been selected. These experiments were conducted at the NEPTUNUS (Holland) and MIT (USA) experimental facilities.

The local pressurizer performance is emphasized in these concerns because it presents the wall/steam, liquid-steam and spray/steam interphases and, consequently, it generates the global pressure value to be transmitted throughout the system.

Before discussing the mentioned separated effects experiments, the most significant pressurizer parameters have been studied. These parameters are:

a) Geometry:

Height, diameter, spray valve position, wall thickness and material properties.

It may be pointed out that the volumes of these pressurizers are 0.032 m^3 (MIT) and 1.26 m^3 (NEPTUNUS) to confront with the 9.0 m^3 in the pressurizer of JOSE CABRERA NPP.

b) Operating conditions:

Pressure and level in steady state operation (saturation equilibrium) prior to each experiment.

c) Interphases:

Liquid-steam, liquid-wall, steam-wall and steam-spray.

d) Boundary conditions:

Heat losses, heaters power, continuous and controlled spray flow, surge line flow, relief and safety valves.

After an evaluation based on the definition of the "importance" functions for the interphases, it may be concluded that, due to the scale of the experimental pressurizers with respect to the one of JOSE CABRERA NPP, the "importance" of the interphases is amplified by a factor of between 1.65 and 3.48 for the NEPTUNUS pressurizer, and between 90.0 and 398.0 for the MIT pressurizer.

Simulation of these pressurizer tests is considered appropriate since, if with such an amplification the code provides an acceptable response, the scalability to JOSE CABRERA NPP will be guaranteed and the uncertainty minimized.

1.- MIT-A EXPERIMENT SIMULATION.

The pressurizer is a small one of 0.032 m^3 (fig A.1) set up in the laboratories of the Massachusetts Institute of Technology (MIT), USA, operating much below the reference reactor nominal values. The experiment started from a thermal saturated equilibrium condition at a pressure of 6.1+5 Pa and a fixed level of 0.35 m.

The experiment consisted of a partial filling with cold water (43 °C) up to a final level of 0.86 m. with spray and heaters out of operation. Pressure increased due to compression of the steam bubble that became overheated, thus producing heat transfer to the subcooled liquid interphase, wall condensation and heat losses to the environment. Coolant stratification was also observed during the test.

A system model for the RELAP5/MOD2 code has been generated based on the available drawings, processing the geometry as well as metalic structures, environmental losses, and surge line flow.

The following conclusions of the RELAP5/MOD2 post-test simulation may be emphasized:

- Pressure simulation is overestimated in case the wall effect and environmental losses are not simulated. A code malfunction with sudden pressure decrease whenever the level goes just from one node to the next one has been detected (fig. A.2).

- Steam temperature simulation is very deficient in case the instrumentation time delay constant is not simulated (fig. A.3).

- The response is very sensitive to the conductivity and specific heat of the steel wall.

- The evolution of the liquid, steam and wall temperatures are correct, both at the lower and upper part of the pressurizer vessel (fig. A.4 y A.5).

- Final pressure simulation has been satisfactorily calculated (fig. A.6).

- It becomes necessary to provide the model with a sufficient number of nodes in the liquid phase, to allow for a correct simulation of the coolant thermal stratification and axial gradient.

After the appropriate simulation of the MIT-A experiment, it may be concluded that RELAP5/MOD2 correctly predicts the liquid thermal stratification, the steam compression due to the filling of the pressurizer, the condensation in the wall and in the steam-liquid interphase, and the environmental losses, thus generating satisfactory evolutions of level, temperatures and pressure in pressurizer slow filling transients.

FIG. A.1: MIT pressurizer test. Facility diagram and RELAP5 model.

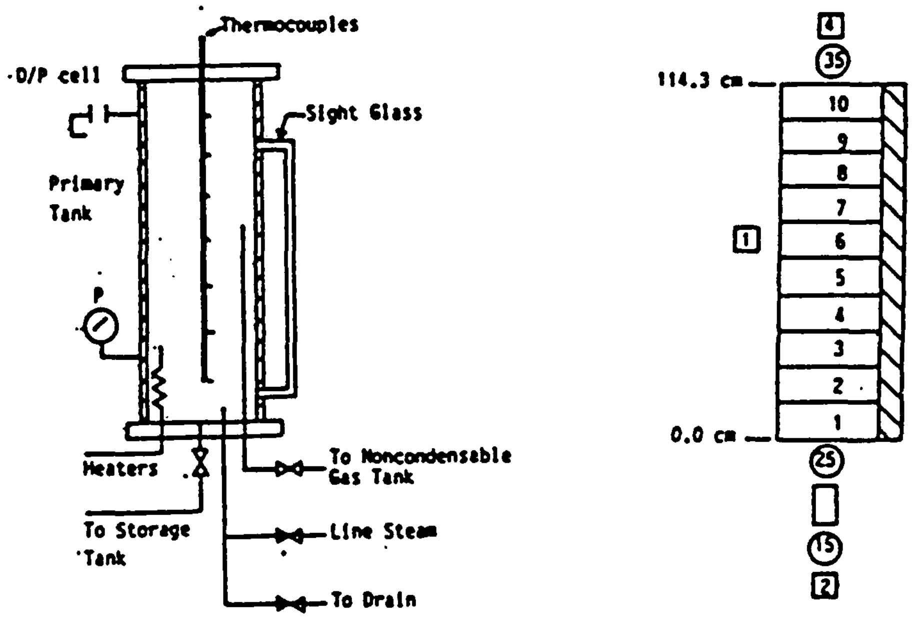


FIG. A.2: MIT pressurizer test. Effect of wall simulation on the RELAP5 calculated system pressure.

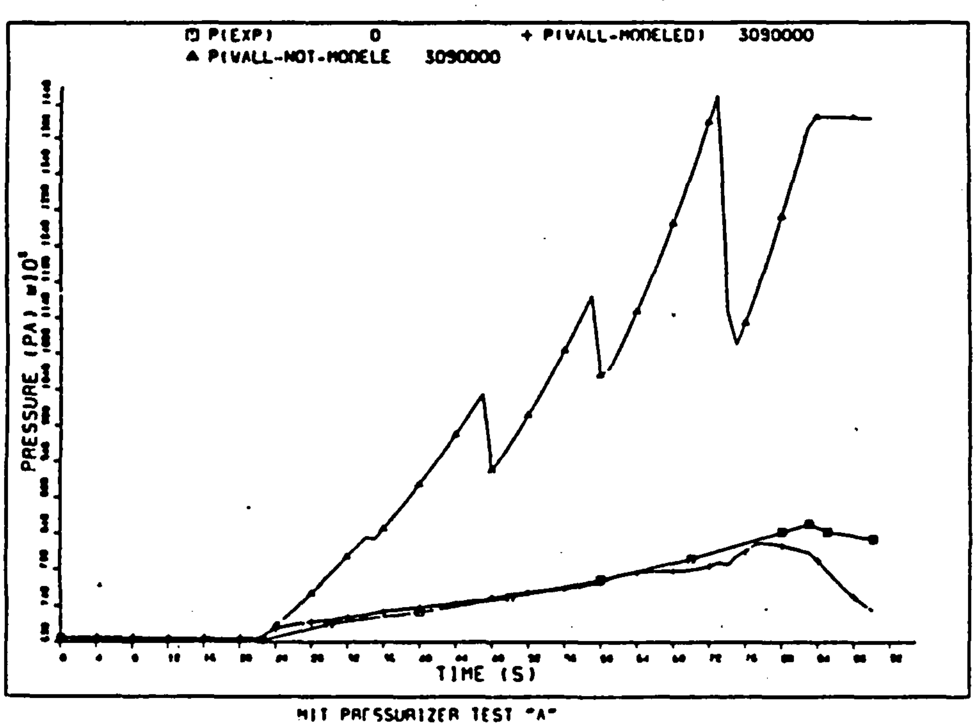


FIG. A.3: MIT. Effect of the steam temperature instrumentation lag on the RELAP5 calculated steam temperature.

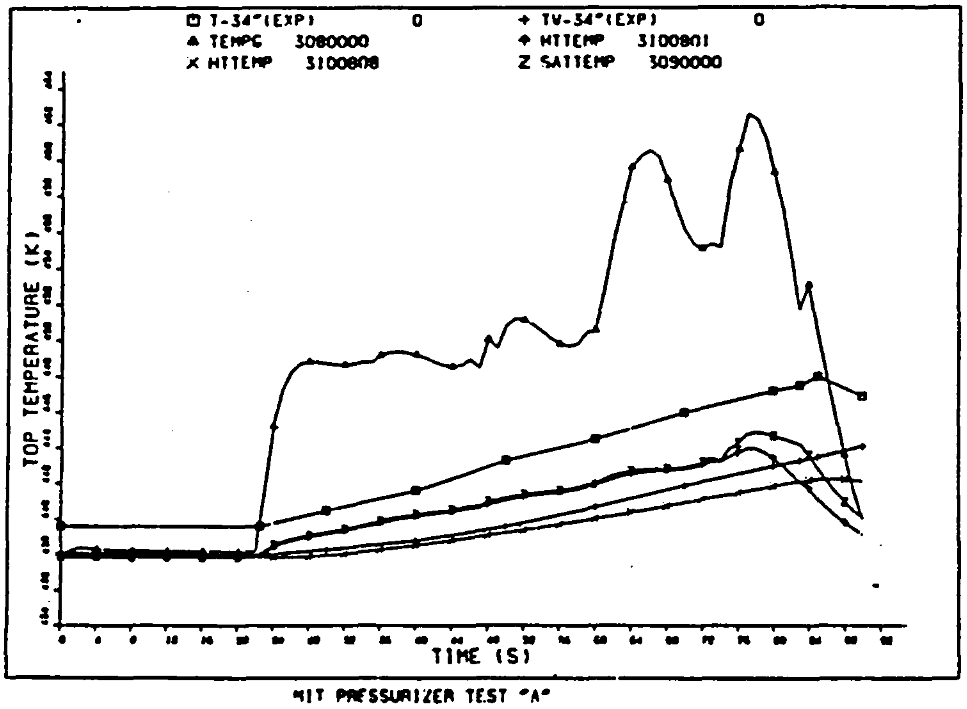


FIG. A.4: MIT. RELAP5 simulation of the wall and liquid temperature in the lower part of the pressurizer.

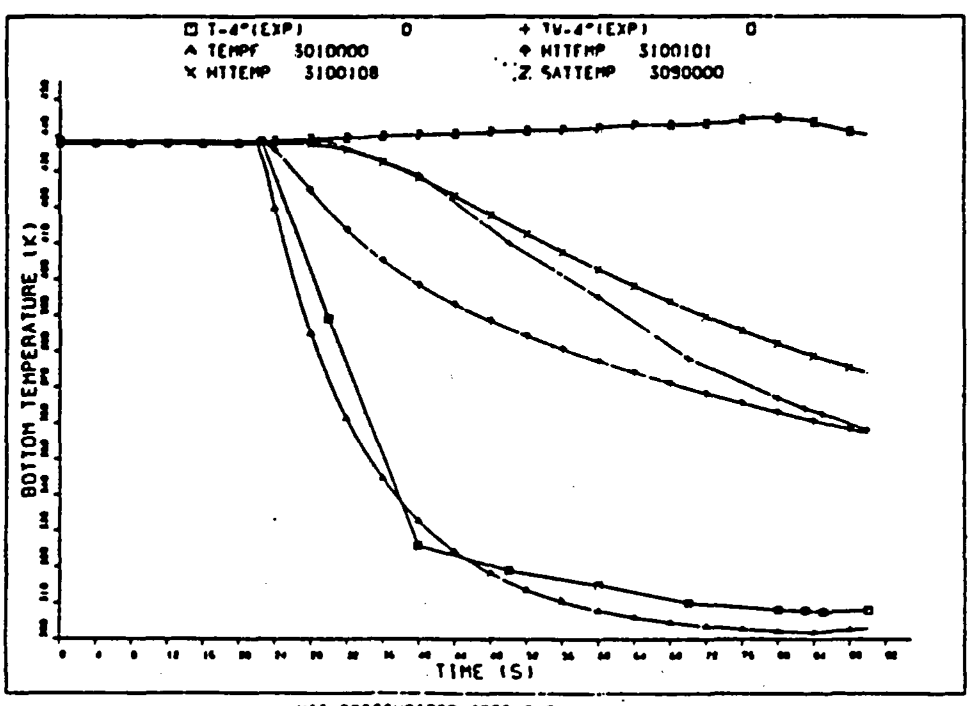


FIG. A.5: MIT. RELAP5 simulation of the wall and liquid temperature in the upper part of the pressurizer.

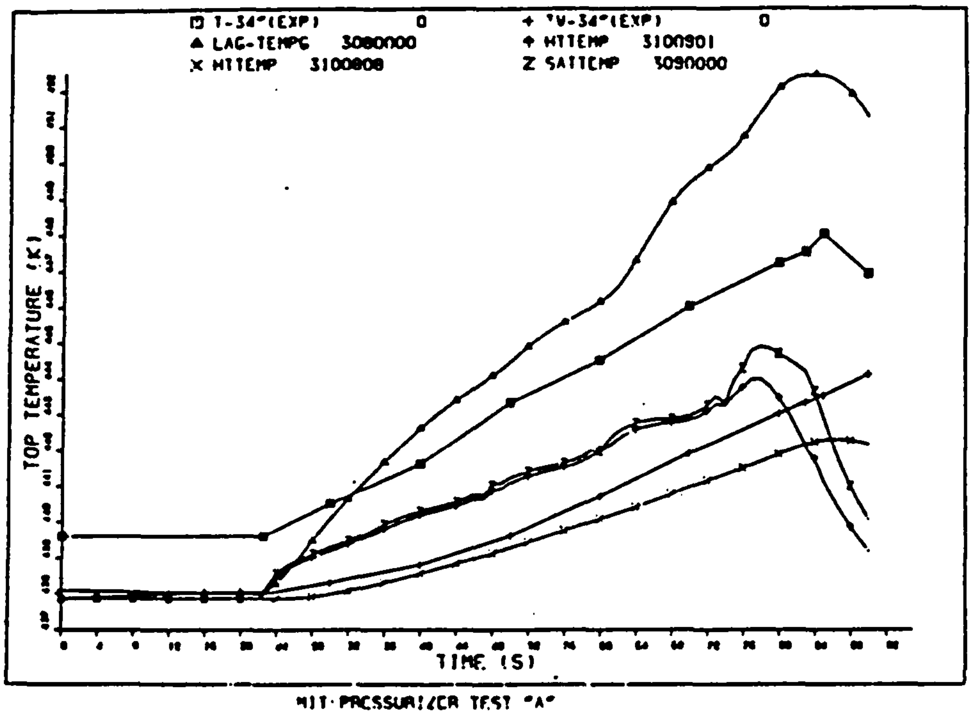
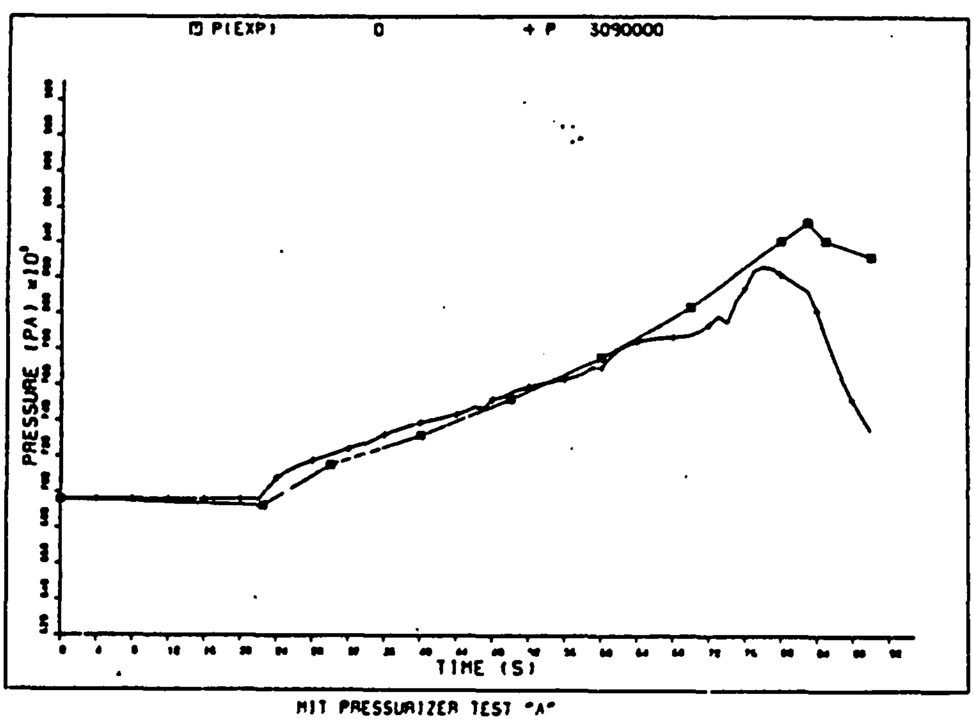


FIG. A.6: MIT. RELAP5 calculated system pressure.



2.- NEPTUNUS Y-05 EXPERIMENT SIMULATION.

The pressurizer has 1.26 m^3 of volume (fig. A.7), is located at the Delft University (Holland) and operates at conditions similar to those in an actual power reactor. The Y-05 experiment started from a thermal saturated equilibrium condition at a pressure of 126.0+5 Pa, an electrical heating power of 17. Kw and a fixed level of 1.12 m.

The experiment consisted of successive filling and emptying of the pressurizer as a result of cyclic oscillations (fig. A.8) of the lower surge line make up flow rate at constant temperature and of the upper spray flow rate at variable temperature with the heating power kept constant.

Level oscillations with regard to the reference level were observed, as well as pressure oscillations as a function of the combined effect of the following trends:

- pressure increase due to steam volume decrease associated to rising level.

- pressure increase due to heaters effect.

- pressure decrease due to steam volume increase associated to dropping level.

- pressure decrease due to steam condensation in the wall and in the water and spray interphases.

A system model for the RELAP5/MOD2 code has been generated based on the available drawings, processing the geometry as well as heaters, metalic structures, environmental losses, and surge line and spray flow.

The following conclusions of the RELAP5/MOD2 post-test simulation may be emphasized:

- Pressure and steam temperature increase rates are predicted conservatively in excess during filling, with amplification of pressure peaks (fig. A.9).

- Pressure decrease rate due to emptying is predicted in excess, with satisfactory fitting of pressure valley points (fig. A.9).

- A filling impact larger than the spray impact is conservatively predicted, in opposition to what was experimentally observed (fig. A.9).

- Presence of overheated steam in filling is correctly predicted as well as saturated steam in emptying. Experimentally observed level oscillations are predicted as well (fig. A.10).

- The difference between calculated and measured pressure peaks increases along the various transient cycles (fig. A.9) due to the increasing delay between surge line and spray flows (fig A.8).

- The heat transfer in the interphase with the subcooled spray is underestimated. Predicted depressurization rate due to spray condensation is lower than measured.

- It is convenient to fit the nodalization so that the pressurizer average level is in a node center.

- Performance of RELAP5/MOD2 matches TRAC-PF1/MOD1 as illustrated in NUREG/CR-3919.

After the NEPTUNUS Y-05 experiment simulation it may be concluded that RELAP5/MOD2 satisfactorily predicts, with conservative trend, the liquid thermal stratification, steam compression due to pressurizer filling, wall, steam-liquid and steam-spray interphase condensation, and environmental losses, thus generating a satisfactory response to level, temperature and pressure evolutions in pressurizer fast transients.

FIG. A.7: NEPTUNUS Y-05 test. Facility diagram and RELAP5 model.

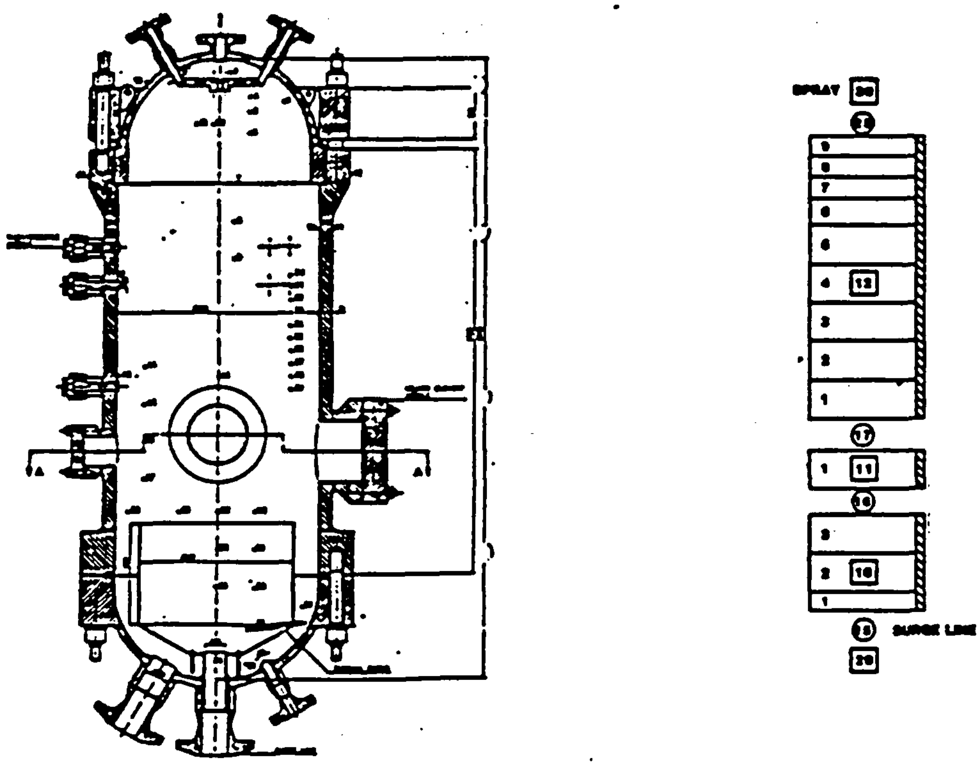
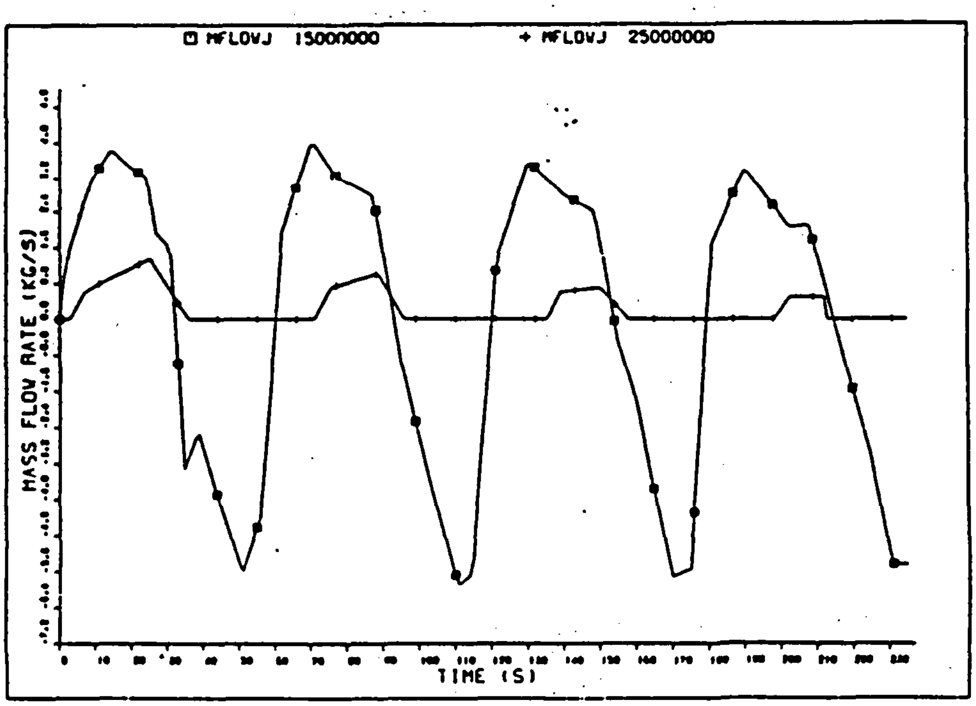


FIG. A.8: NEPTUNUS Y-05 test. RELAP5 simulation of the surge line and spray mass flow rates.



NEPTUNUS PRESSURIZER TEST Y-05

FIG. A.9: NEPTUNUS Y-05 test. RELAP5 calculated system pressure.

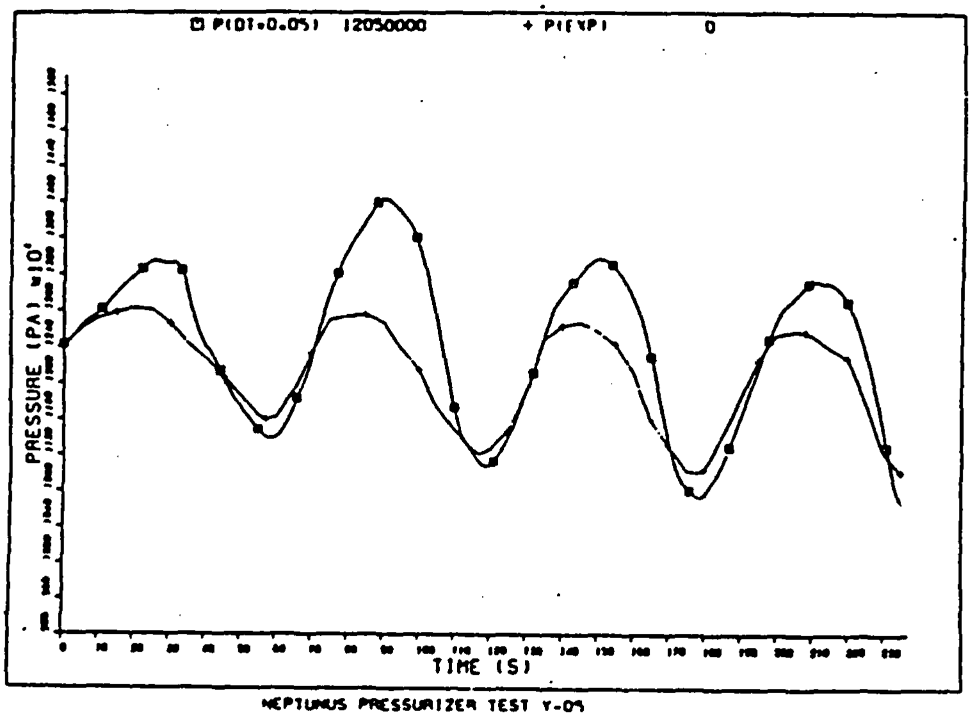
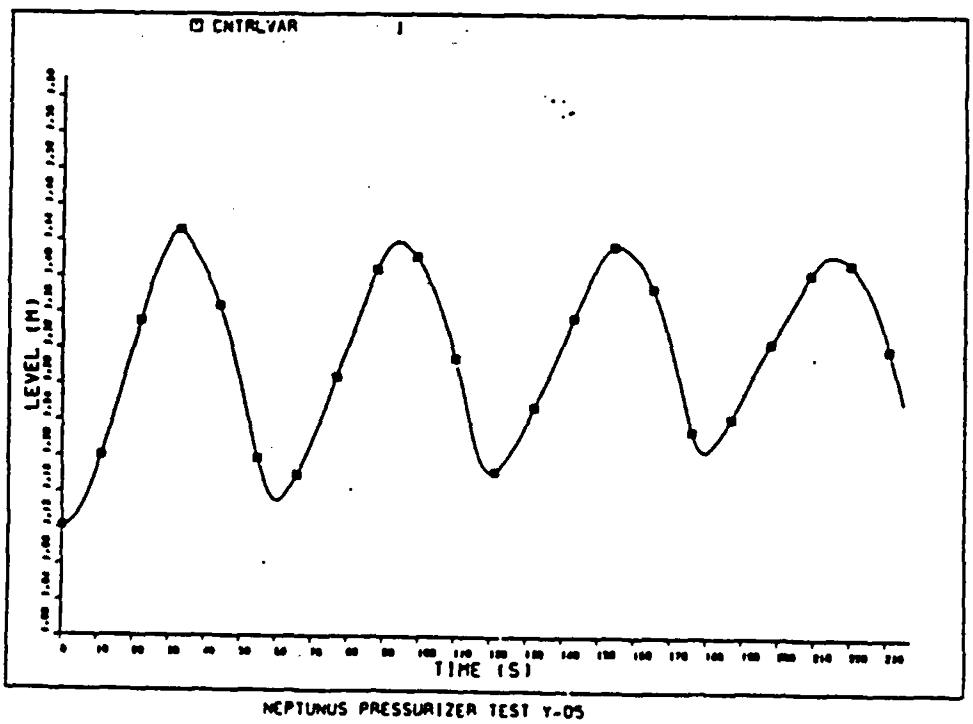


FIG. A.10: NEPTUNUS Y-05 test. RELAP5 calculated pressurizer level.



3.- ASSESSMENT CONCLUSIONS OF RELAP5/MOD2 WITH REGARD TO PRESSURIZER SEPARATED EFFECTS TRANSIENTS.

Both studies of pressurizer separated effects, with regard to the performance of RELAP5/MOD2 and its applications to pressurizer pressure slow and fast transients in commercial reactors, lead to the following conclusions:

- It is necessary a correct simulation of the thermal properties of the wall material as well as the environmental thermal losses.

- Simulation of the instrumentation time delay constant is fundamental in order to compare steam temperature calculated and measured values.

- Time steps of 0.05 seg. are adequate.

- Prediction of both liquid/wall and steam/wall heat transfer is correct.

- Prediction of axial thermal gradient in the wall as well as in the water is correct.

- Condensation in the steam/spray interphase is underestimated.

- Filling pressurization and emptying depressurization rates are overestimated as a result of an underestimation in processing condensation and vaporization rates in the liquid-steam interphase.

- RELAP5/MOD2 performance is similar to the observed in TRAC-PF1/MOD1 simulation.

- Because the pressurizer interphases "importance" function is lower in JOSE CABRERA NPP than in MIT and NEPTUNUS experimental facilities, the observed impact in these simulations will be reduced when applied to the real plant case.

This set of conclusions positively qualifies RELAP5/MOD2 code to be applied for pressurizer filling pressurization as well as for emptying and spray condensation depressurization both for slow and fast transients simulation in the case of a real nuclear power station.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This document presents the comparison between the simulation results and the plant measurements of a real event that took place in Jose Cabrera nuclear power plant in August 30, 1984. The event was originated by the local, continuous and inadverted opening of the pressurizer spray valve PCV-400A.

Jose Cabrera power plant is a single loop Westinghouse PWR belonging to UNION ELECTRICA FENOSA, S.A. (UNION FENOSA), a Spanish utility which participates in the International Code Assessment and Applications Program (ICAP) as a member of UNIDAD ELECTRICA, S.A. (UNESA). This is the second of its two contributions to the Program: The first one was an application case and this is an assessment one. The simulation has been performed using the RELAP5/MOD2 cycle 36.04 code, running on a CDC CYBER 180/830 computer under NOS 2.5. operating system.

The main phenomena have been calculated correctly and some conclusions about the 3D characteristics of the condensation due to the spray and its simulation with a 1D tool have been got.

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