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ICSP-AS-BOUT-R

International Agreement Report

Assessment and Application of Blackout Transients at Asco Nuclear Power Plant with RELAP5/MOD2

Prepared by
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Office of Nuclear Regulatory Research
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Prepared as part of
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NOTICE

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ABSTRACT

The Asociación Nuclear Ascó has prepared a model of Ascó NPP using RELAP5/MOD2. This model, which include thermalhydraulics, kinetics and protection and controls, has been qualified in previous calculations of several actual plant transients.

The first part of the transient presented in this report is an actual black-out and one of the transients of the qualification process. The results are in agreement with plant data.

The second part of the transient is a hypothetical case. It consists in re-starting a primary pump and assume a new black-out.

The phenomenology prediction of this second part has been useful from the operation and safety point of view.

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EXECUTIVE SUMMARY

Ascó Nuclear Power Plant is a nuclear station with two PWR of 930 MWe of Westinghouse design.

The Thermalhydraulic analysis group of the Asociación Nuclear Ascó (ANA) has prepared a model of the plant using RELAP5/MOD2. This model includes thermalhydraulics, kinetics and protection and controls.

ANA's commitment with the International Code Assessment and Application Program (ICAP) is the participation with two cases.

One of the transients selected for this purpose is the "Assessment and application of Black-out Transients at Asco NPP with RELAP5/MOD2".

This transient has been chosen mainly because it is on-line with one of the most important goals of the Thermalhydraulics analysis group of ANA, this is to provide engineering support to plant operation.

After a successful assessment of an actual black-out transient, a hypothetical case is simulated in order to generate information about the application scenario (re-start and re-trip a Reactor Coolant Pump). The main conclusions of the analysis are the following:

- First part: close agreement between results and data.
- Second part: the primary system recovers natural circulation. Core mass-flow is always positive during all the transient.
- Relap5/Mod2 Asco Model is a valuable tool to analyze plant transients.

F O R E W O R D

This report has been prepared by Asociación Nuclear Ascó 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
- 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

In 1986, the Asociación Nuclear Ascó (ANA) created a group for plant and core thermal-hydraulic analysis. The objectives of the group are as follows:

1. Create and update core and plant thermal-hydraulic models based on best-estimate criteria.

2. Provide off-line engineering support to the different technical branches of ANA (i.e., technical services, reactor operation):

   a. Analyze operating events that result in event reports.

   b. Assess plant systems and/or equipment modifications as well as plant operating procedures and emergency instructions.

   c. Analyze plant behavior under incident or accident conditions in the abovementioned cases.

   d. Scenarios and core damage evaluation for probabilistic risk assessment.

3. Review final safety analysis report transients and accidents based on best-estimate criteria.

4. In the future and if appropriate, participate in Ascó individual plant examination.

The plant analysis activities developed so far include the following:

1. Implementation of RELAP5/MOD2 (Ref.1) cycle 36.05 in its IBM version in an IBM 4381 and 3090 and cycle 36.04 and its Control Data Corporation version in a Cyber 180/830. The results of both versions for the scenarios analyzed are in close agreement.

2. Thermal-hydraulic model of both the primary and secondary systems. /2/

3. Kinetic model specifically adapted to Ascó

4. Simulation of control and protection systems /3/.

5. Revision and detailed study of all start-up tests and every transient that has occurred in either unit. A total of 60 cases were studied. Because of the influence of plant dynamics and the quality and availability of plant data, six cases were selected to validate the complete plant model:

   a. Black-out

   b. Faulty pressurizer spray valve opening.

   c. Turbine trip without steam dump and secondary relief valves available.

   d. Loss of feedwater (LOFW)

   e. Turbine trip with all systems available

   f. Turbine power step

6. Simulation of the above six transients and adjustment of control parameters /4/, /5/, /6/.

7. Participation in the Internacional Code Assessment and Application Program with two cases.

8. Analysis of transients such as small-break-loss-of-coolant accident (SBLOCA), anticipated transient without scram (ATWS), and others for PRA studies.

The adjustment and qualification process is the first and most important part of plant analysis. Sufficiently accurate predictions with meaningful sets of measured data provide validation of both the model and the procedures to be used in the future to analyze various transient and accident scenarios of general interest such as SBLOCA and ATWS.

2. PLANT AND TRANSIENT DESCRIPTION

2.1 Plant Description

Ascó Nuclear Station is a nuclear power plant with two 930-Mwe Westinghouse Pressurized Water Reactors (PWR). The first criticalities were reached in June 1983 and September 1985, respectively, for units I an II. Today, both units are in their sixth and fourth cycles of normal operation.

The main characteristics of both units are given in Table 1. The core contains 157 fuel assemblies of (17x17 -25) fuel rods and the steam generators are typical ones with U-tubes and preheaters (model D-3). All other major components are standard Westinghouse components.

2.2 Plant Data Acquisition System

The plant data used in each assessment calculation is that produced by the plant process computer on the post-trip report. It types a value of each pre-selected variable every 10 seconds. The post-trip report of this particular transient is given in Annex I.

2.3 Transient description

The transient being analyzed is divided into two parts. The first one is an actual transient that took place on October, 14; 1987 and is used to assess RELAP5/Mod2 and the plant model capabilities. The second part of the transient can be considered as an aplication of the model since a hypothetical situation is analyzed.

The assessment part is described below: After an electrical power gird perturbation, the turbine and reactor tripped. Thirty-seven seconds later, a black-out took (loss of offsite power) place and the reactor coolant pumps (RCPs) tripped. The normal actuation of the emergency diesel generators allowed the correct safeguards sequence.

In Table 2 a description of the main events is presented.

3. MODEL DESCRIPTION

Figure 1 shows the model used to simulate the plant. It consits of 134 volumes, 146 junctions, 32 heat structures, and 259 control variables. The model includes the vessel, the three primary loops, the pressurizer, the three steam generator, the three secundary loops, and the steam lines. The turbine, condenser, and feedwater tank are modeled as time-dependent volumes.

The usual practice of implementing RELAP5 by homogenizing the multiple steam generator loops into two loops (one including the pressurizer) is not followed in this model because of the following reasons:

a) non - symmetric distribution of auxiliary feed-water among the three steam generators

b) Different length of the steam lines of each loop.

c) Different number of plugged tubes in each steam generator.

d) Non - symmetric transients (loss of feed water, steam generator tube break, small break in the primary circuit, one reactor coolant pump trip, and so on) that require modeling of different actuations or boundary conditions, in each loop.

e) Computer availability

3.1 Thermal - Hydraulic Model

a.- Primary System.

The model /2/ of the primary system includes the main components of the plant. The core is modeled by volume 120 and the proper heat structures. Volume 130 simulates the by-pass region between the core baffle and the core barrel. Volume 140 model the upper plenum and volumes 150 and 160 the vessel upper head. The three hot lines depart from the core upper plenum. Loop 3 (volume 410) is connected with the pressurizer through volume 510. The pressurizer is divided into two volumes.

The lower one (volume 520) is divided into five nodes. Heat structures, simulating the pressurizer actual heaters, are attached to the first two nodes. Volume 525 is a branch in order to model the junctions connecting the pressurizer with the safety and relief valves and with the spray system.

Volume 420 models the remaining of the hot leg. Volumes 430 and 440 simulate the water boxes of the steam generator. The primary side of the steam generator is modeled by volume 431 divided into nine nodes. Volumes 450, 465, 466 and 470 represent the cold leg. Volume 460 model the primary coolant pump, proper homologous curves, given by the vendor, have been used for this purpouse. Volume 468 models the Safety Injection System.

b.- Secondary System

The model of the secondary system starts with Time Dependent Volume 870 (Loop 3) that represent the feed-water going to the steam generator. Volume 871 models the Auxiliary Feed Water. The downcomer is simulated by means of volumes 800, 801, 822 and 825. The steam generator preheater is modeled by volumes 806 and 807, and the remaning of the tubes zone by volumes 808, 806, 809 and 810. The steam separator with volume 820. Volume 830, 840 and 850 model the steam dryer and the domo of the steam generator. The steam line starts at volume 880. Safety and relief valves (components 886 and 884) are connected to volume 881. Component Valve 885 models the isolation valve. Time Dependent Volumes 994 and 999 represent the free atmosphere. The steam is conducted throughout volume 883 to the steam-collector, volume 900. Finally Valves 906, 903 and 907 model the by-pass to condenser valve, and the turbine stop and control valves, respectively.

Proper heat structures are used to connect thermally the primary side of the steam generator with the secondary side. Actual values are used for all the variables except for the hydraulic diameter, heat transfer surface, and thermal conductivity of the tubes material where some changes were introduced in order to achieve the actual heat transfer rate without any change in primary average temperature and secondary pressure.

The data used to model volumes and junctions as well as heat structures were taken form plant design information /7/.

3.2 Kinetic Model

The kinetic model /3/ was prepared using the RELAP5/Mod2 space-independent reactor kinetics option with data from the ANA Nuclear Analysis Group. The model includes a scram table of reactivity versus time. The total control rod drop time is the actual value measured at plant. This table is activated by reactor trip.

The control model supplies the reactivity of the C and D control rod banks.

This control reactivity is added to the feedback reactivities calculated by the kinetic model from the data supplied for the specific burn-up condition of each transient.

3.3 Control and Protection Systems Model

The protection and control systems were modeled using RELAP5/MOD2 control blocks and following specific setpoint studies, logical diagrams and technical specifications of the plant /3/, /8/, /9/. The model includes the following systems:

a.- Reactor Trip System

The reactor can be tripped in the RELAP5/MOD2 model because of the following effects:

- Low primary pressure.
- High primary pressure.
- Low speed at any pump.
- High pressurizer level.
- High reactor power.
- Low level at any steam generator.
- Overtemperature.
- Overpower.
- Turbine trip.
- Safety injection.

In Figure 2 the logic of the reactor protection system is presented.

b.- High pressure Injection System

Using the following signals:

- Very low Average Temperature.
- Low steam generator pressure.
- High steam mass flow rate.
- Low primary pressure.
- Large pressure difference between S.G.

the logic of the safety injection system was reproduced. The massflow rate injected is modeled by means of the pumps characteristic curves.

c.- Turbine Trip and Control System

The position of the turbine control valve is controlled as a function of the difference between the Required Power and

Actual Power, with the proper control block to model the actual logic of the plant.

The Turbine Trip (closure of the turbine stop valve) is modeled. The signals of Safety Injection, very high steam generator level and Reactor trip, are used to trip the turbine.

Turbine run-back has not yet been modeled.

d.- Feed Water Control System

The feed water control system has been modeled as shown in Figure 3. The massflow rate calculated by the control system is injected by means of a time dependent junction.

The auxiliary feed water system is also included in the model.

e.- Pressurizer level and pressure control system

The model of the pressure control system actuates upon heaters, spray valve and charging pump. Pressurizer safety and relief valves and level control system are also simulated.

f.- Steam Dump control system

In Figure 4 the model used for the steam dump control system is represented.

g.- Average Temperature Control System

The average temperature control system modeled with RELAP5/MOD2 is shown in Figure 5. As can be observed this system controls both the primary average temperature and the primary-secondary power mistmach.

Other systems modeled are:

h.- Steam line Isolation logic.

i.- Main Feedwater Isolation logic.

j.- Safety and Relief valves of the secondary.

4.- STEADY STATE CALCULATION

A steady state calculation was performed with the plant at 100% rated condition. The objective is to obtain a stable condition to start transients.

In Table 3 a comparison between the model results and the plant data is given for the main plant variables. A complete description of the plant sensors and signals is given in ref. 10.

5.- ASSESSMENT: TRANSIENT CALCULATION AND COMPARISON VERSUS ACTUAL DATA

The plant transient took place on October, 14; 1987. As said before, the plant data acquisition system available at that time was only able to print-out a value of each variable preselected every 10 seconds. On the other hand, for this transient, only four variables were printed with enough accuracy to validate the RELAP5 model (see Table 4 and Table 5).

Table 6 shows a comparison of the cronology of the main events.

These variables are:

- loop 3 mass flow rate.
- Primary Average Temperature.
- Vessel Delta-Temperature.
- Primary pressure.

The comparison of model prediction versus actual values for these variables is given next:

a.- Primary Mass-Flow Rate

In Figure 6 the comparison between actual data and RELAP5/MOD2 is given.

From second 40 to 100 steady state at 100% rated power is assumed; however the plant was not stable during that period since the electrical network perturbation cause oscillations in turbine and reactor power. Since neither the actual cause that triggered the transient for the actual situation of the plant are perfectly known, due to lack of data, it was decided to trip the reactor and turbine from an 100% stable condition. Also during the 37 seconds between reactor and turbine trip and black-out, network perturbations caused some primary pumps velocity changes as can be observed in Figure 6. Once black-out took place the prediction of mass flow rate decay shows a close agreement with actual data. To achieve this agreement it was used the RELAP5/MOD2 Pump component adjusting accordingly the friction coefficcients.. The comparison ends at 280 seconds (180 seconds after reactor trip) because of lack of plant data beyond that point.

b.- Vessel Delta - Temperature

At Figure 7 the comparison between actual data and RELAP5/MOD2 prediction of this variable with and without signal treatment is given. As can be observed a slightly smaller decrease is obtain with the model during the period between reactor trip and black-out (100 to 137 seconds). It is due to the primary pumps behaviour during this time interval that induces an increase in primary mass flow (see Figure 6). Once black-out takes place the small difference between actual data and prediction is probably due to the difference in primary mass flow rate, and reactor residual power because of the grid perturbation that took place before reactor trip.

c.- Primary Average Temperature

Figure 8 shows actual data of this variable compared against code prediction with and without signal proccessing. A good comparison is obtained for most of the transient although at the latest part of it a growing mistmatch can be observed. It is probably due to reactor residual power and the actuation of the steam-dump at the early stages of the transient. Since there is not data of the steam-dump behaviour it was not possible to improve the agreement.

d.- Primary Pressure

Figure 9 shows the comparison between primary pressure data and RELAP5/MOD2 prediction. An almost constant mismatch is observed during the transient. The main reason, since a better agreement is obtained in average temperature, could be due to the pressurizer heaters actuation during the time from reactor trip to black-out. No data is available on heaters behaviour and its proper actuation is assumed in the model.

6.- APPLICATION: TRANSIENT DESCRIPTION

An interesting hypothetical case was suggested to study after the black-out transient was assessed.

The case basically consisted in re-starting a primary pump, once electrical power in recovered (500 seconds after reactor trip is assumed) and once if is at rated speed assume a new black-out. The objective being to know how long it will take to restore natural circulation throught-out the core since two-loop will have reverse mass flow rate at the time of the second black-out.

Table 7 shows the cronology of the main events of the application transient.

In figures 10 to 18 the main variables are given:

		PRIMARY VARIABLES
Fig.	10	Reactor Power
"	11	Primary Mass Flow Rates
"	12	Loop 3 Mass Flow Rate
"	13	Primary Pressure
"	14	Primary Average Temperature
"	15	Vessel Delta-Temperature
"	16	Secondary Pressure
"	17	Secondary Mass Flow Rates
"	18	Secondary Mass Flow Rates
"	19	Steam generator Narrow Range Level

The main conclusion of the analysis is that mass flow rates throught-out the core goes back to the natural circulation values previous to the Pump 2 transient without having negative mass-flow rate at any moment, and that natural circulation is restored in all loops after flow reversal in loop 1 and 3 (see Figures 11 and 12)

From Figure 15 it can be seen that the increase of temperature in the vessel returns to the values of the first part of the transient after the second black-out. The oscillation observed from second 900 to the end of the time analyzed is due to the small oscillation in mass flow rate after the second black-out (see Fig. 11 and 12)

Figures 14 and 13 show the evolution of primary average temperature and pressure. As can be observed after the second black-out temperature and pressure get stable values above those reached after the first one. It is due (see Figure 16 which represents secondary pressure) to the decrease in secondary pressure after the first black-out because of the actuation of the auxiliary feed water system which introduces cold (313 ºK) water above the steam generator water level so in the vapor region, causing condensation and then depressurization in the steam generator. After the second black-out the steam generator level (see Fig. 19) is such that no auxiliary feed-water is introduced into the system (see Fig. 17 and 18). Consequently no secondary side depressurization takes place resulting in a higher primary average temperature and pressure at the end of this application transient than in the assessment one.

7.- RUN STATISTICS

Calculations were carried out on a CYBER 180/830 NOS 2.5 property of Fundación Leonardo Torres Quevedo located at Santander - Spain.

RELAP5/Mod. 2 cycle 36.04 was used in all the calculations.

In Table 8 a typical run statistics is presented.

8. CONCLUSIONS

Both cases, the actual transient and the hypothetical one, have been simulated with RELAP5/MOD2 Ascó model.

The results of the actual transient are in agreement with plant data although only a limited number of variables were recorded.The nodalization used has allowed the correct simulation of the behaviour of the plan.

This calculation, along with those of the rest of transients of the qualification matrix, provides the validation of the model.

This transient provides an assessment of the RCP function that is valid for future predictions.

The simulation of signal treatment seems the correct one.

The Steam-dump actuation, although in this transient is truncated by the black-out, can be improved as long as predictions of primary temperature and pressure are higher than actual data.

The results of the hypothetical transient provide the improvement of the knowlegde of plant dynamics.

Inactive loops (1 and 3) get reverse flow after the re-start of pump 2.

Natural circulation is restored in all loops after the second black-out.

During all the transient, core mass-flow is always positive.

The hypothetical situation is as safe as the actual one.

The model of Asco using Relap5/Mod2 is a valuable tool to analyze plant transients and to provide engineering support to plant operation.

9.- REFERENCES

1) V. H. RAMSON and R.J. WAGNER, "RELAP5/MOD2: Code Manual, "EGG-SAAM-6377, EG&G Idaho, Inc. (Apr. 1984).

2) F. REVENTOS, J. SANCHEZ-BAPTISTA, and P. MORENO, "Modelo Termohidráulico de la C.N. Ascó", Asociación Nuclear Ascó (June 1987).

3) F. REVENTOS, J. SANCHEZ-BAPTISTA, and P. MORENO, "Modelo Termohidráulico y de protección y control de C.N. Ascó," Asociación Nuclear Ascó (May 1988).

4) F. REVENTOS, J. SANCHEZ-BAPTISTA, and P. MORENO, "Análisis de transitorios en A.N.A. con RELAP5/MOD2," presented at 14th Annual Mtg. Sociedad Nuclear Española, Marbella, Spain, October 26-28, 1988.

5) F. REVENTOS, J. SANCHEZ-BAPTISTA, P. MORENO, and A. PEREZ NAVAS, "Transient Analysis for ASCO Nuclear Power Plant Using RELAP5/MOD2," Proc. Ist Int. RELAP5 Users' Seminar, College Station Texas, January 31-February 2, 1989.

6) F. REVENTOS, J. SANCHEZ-BAPTISTA, A. PEREZ-NAVAS and P. MORENO. "Transient Analysis in the Ascó NPP using RELAP5/MOD2," Nuclear Technology, Volume 90, Number 3, pp 294-307. June 1990.

7) "Final Safety Analysis Report-Ascó I" (June 1983), "Final Safety Analysis Report-Ascó II" (Sep. 1985), and later revisions, Asociación Nuclear Ascó.

8) F. BALLERINI, "Setpoint Study Ascó Units I and 2," Westinghouse Electric Corporation (Mar. 1976).

9) "Precautions, Limitations and Setpoints of Ascó Units 1 and 2". E/PS/76/068. Rev. 11. Sep. 1988.

10) C. Simon, et al "Central Nuclear de Ascó Units 1 and 2. Emergency Recovery Guidelines Setpoint Values. Calculation and Methodology Appendix D. Instrumentation channels statistical calculation of Uncertainties". WENX 88-08. Rev. 2. October 1989.

TABLES

- Electrical power	930 Mwe
- Thermal reactor power	2686 Mwth
- Fuel	UO_2
- Number of assemblies	157
- Fuel rods per fuel assembly	(17x17 - 25) = 264
- Active length of fuel rods	3.657 m
- Outside diameter of fuel rods	4.75×10^{-3} m
- Cladding tube material	Zr -4
- Cladding tube wall thickness	0.655×10^{-3} m.
- Average linear heat generation rate	17.2 Kw/m.
- Absorber rods per control assembly	24
- Absorber material	Ag - In - Cd.
- Number of coolant loops	3
- Reactor operating pressure (pressurizer)	15.51 Mpa.
- Coolant Average Temperature	581.3 ºK
- Coolant flow rate	14287 Kg/s

Table 1.- Description of the main characteristics of Ascó I and II Nuclear Station. (1 of 3)

Steam Generator

- Type	Westinghouse D-3
- Number	3
- Height	20.6 m
- Diameter (Upper shell)	4.445 m
- Tube material	Inconel 600
- Average tube length	15.94 m
- Design pressure/temperature (steam plant side)	8.17 MPa/589 ºK
- Inner diameter of tubes	1.687×10^{-2} m.
- Outer diameter of tubes	1.905×10^{-2} m.

Reactor coolant Pumps

- Type	Westinghouse 93-DS
- Discharge head	86.25 m
- Design flow rate	5.928 m^3 /s
- Speed	155 rad/s

Table 1.- Description of the main characteristics of Ascó I and II Nuclear Station. (2 of 3)

Pressurizer

- Height	12.835 m
- Diameter (inner)	2.134 m
- Volume	39.64 m^3
- Operating saturation pressure	15.51 MPa
- Heating power of the heater rods	1.40 Mwe

Steam/Power Conversion Plant

- Feed Water flow rate	497.5 Kg/s/loop
- Main steam flow rate	1492 Kg/s
- Steam moisture at steam generator outlet	0.25%
- Feedwater temperature	497.05 ºK

Table 1 - Description of the main characteristics of Ascó I and II Nuclear Station. (3 of 3)

Second	Event
0.0	Turbine and Reactor Trip because of electrical power network perturbation.
0.4	Steam-Dump starts to open.
2.0	Steam Dump fully open.
20.0	Steam Dump starts to close.
37.0	Black-out.
47.0	Start-up of the Diesels.
92.0	Auxiliary Feed water available.

Table 2 - Main events that took place during the black-out.

VARIABLE	RELAP5/MOD2	PLANT DATA
PRIMARY MASS FLOW RATE (Kg/s)	14027	14287 *
CORE BY-PASS MASS FLOW RATE (%)	2.71	2.71 *
VESSEL DELTA-T (ºK)	33.24	33.26
REACTOR POWER (Mw)	2681	2686
PRIMARY PRESSURE (MPa)	15.50	15.51
PRIMARY AVERAGE TEMPERATURE (ºK)	581.1	581.3
RECIRCULATION RATIO	2.286	2.29 *
UP-STREAM FLOW RATIO IN THE S.G. PREHEATER	.517	.520 *
STEAM GENERATOR NARROW RANGE LEVEL	.66	.66
SECONDARY PRESSURE (MPa)	6.808	6.821
STEAM COLLECTOR PRESSURE (MPa)	6.705	6.724
STEAM MASS FLOW RATE (Kg/s)	1478.5	1492.0

* DESING DATA

Table 3 - Comparison between RELAP5/MOD2 values and actual date for steady state.

IDENTIFICATOR	DESCRIPTION
T0421A	Average Temperature measured in loop 2.
T0424A	Vessel Delta - Temperature measured in loop 2.
F0442A	Meassured Mass Flow Rate of loop 3.
P0499A	Pressure at loop 2, hot leg.

Table 4 - Description of plant data measurements.

VARIABLE	DESCRIPTION
P220010000	Pressure at the Hot leg of loop 2.
MFLOWJ 100030000	Mass Flow rate of loop 3, cold leg.
CNTRLVAR 5	Average Temperature.
CNTRLVAR 26	Proccessed Average Temperature.
CNTRLVAR 6	Vessel Delta-T (Hot minus cold leg Temperature).
CNTRLVAR 25	Processed Vessel Delta-T.
RKTPOW Ø	Nuclear Reactor Power.
MFLOWJ 105	Mass Flow Rate at the inlet of the lower Plenum.
MFJ10001	Mass Flow Rate of loop 1, cold leg.
MFJ10002	Mass Flow Rate of loop 2, cold leg.
P681010000	Secondary pressure at the steam line of loop 1.
MFJ906	Steam-Dump Mass Flow Rate.
MFJ675	Main Feed Water Mass Flow Rate of Loop 1.
MFJ672	Auxiliary Feed Water Mass Flow Rate of Loop 1.
MFJ772	Auxiliary Feed Water Mass Flow Rate of Loop 2.
CNTRLVAR 16	Steam Generator 1 narrow range water level.

Table 5 - Description of RELAP5/MOD2 variables.

Event	Time (s)	
	Plant	Relap
Turbine and Reactor Trip because of electrical power network perturbation.	100.0	100.0
Steam-Dump starts to open.	100.4	101.0
Back-up heaters actuation.	—	101.4
Steam Dump fully open.	102.0	104.0
Steam Dump starts to close.	120.0	113.0
Low average temperature signal.	—	118.3
Black-out.	137.0	137.0
Low primary flow.	—	138.2
Start-up of the Diesels.	147.0	147.0
Auxiliary Feed water motor pumps available. (Loops 1 and 3)	192.0	192.0

Table 6 - Comparison of the cronology of the main events.

Event	Time (s)
Re-start pump 2	600. (*)
Pump 2 reaches nominal speed	610.
Re-trip pump 2	700. (*)
Loop 3 recovers positive flow	780.
Secondary refief valves start cycling	790.

(*) boundary conditions

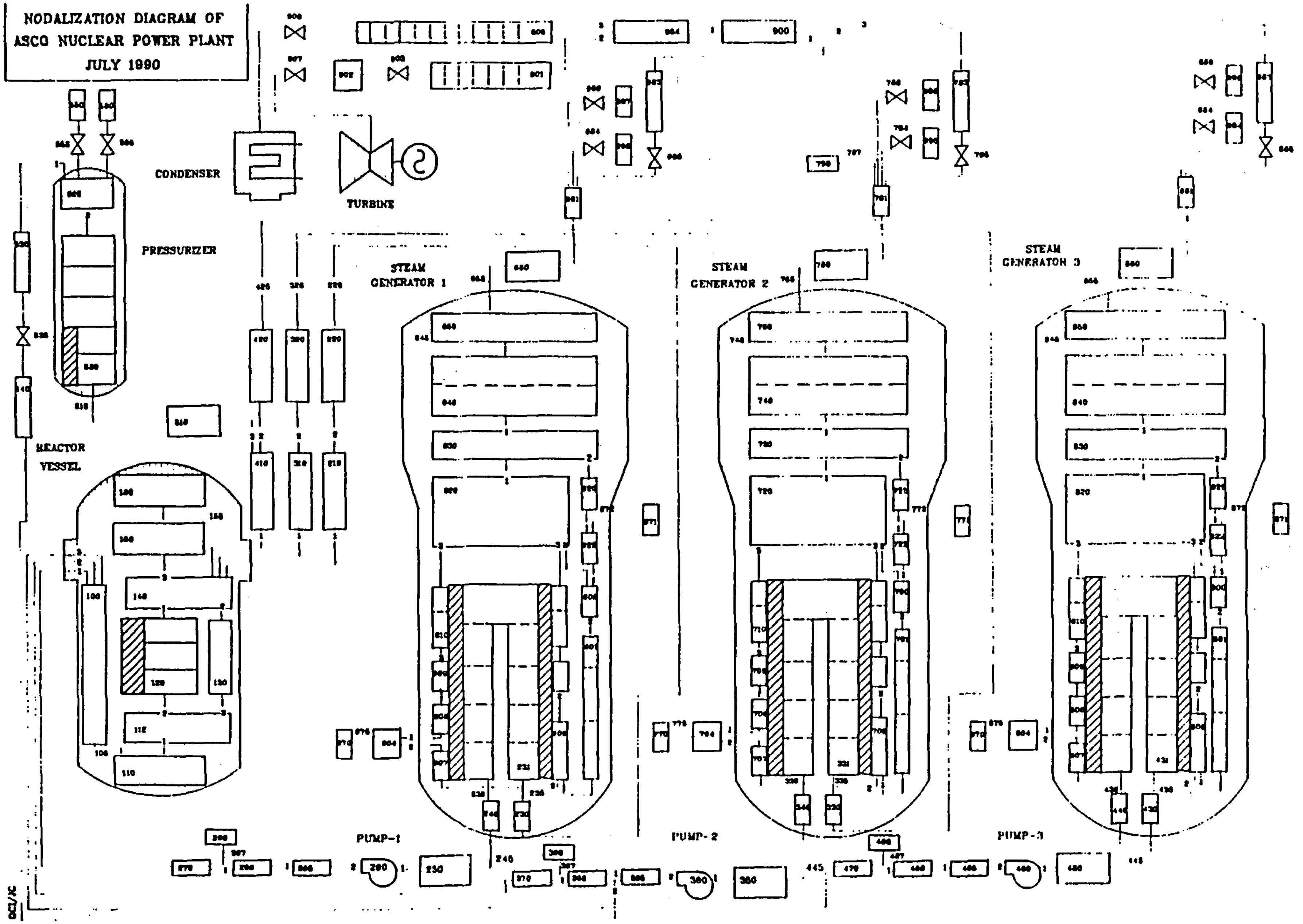
Table 7 - Cronology of the main events of the application transient.

COMPUTER ASSESSMENT	CYBER 180/830
TRANSIENT TIME (s)	180
CPUTIME (s)	14241
C (total number of actives volumes in the model)	173
DT (Total number of time steps)	4359
$\frac{\text{CPU} \times 1000}{\text{C} \times \text{DT}}$	18.884
CPUTIME/TRANSIENT TIME	79.11

Table 8 - Run Statistics.

FIGURES

FR/ac/mb/jv
IST00281
27.12.90


NODALIZATION DIAGRAM OF
ASCO NUCLEAR POWER PLANT
JULY 1990
CONDENSER
TURBINE
PRESSURIZER
REACTOR
VESSEL
STEAM
GENERATOR 1
STEAM
GENERATOR 2
STEAM
GENERATOR 3
PUMP-1
PUMP-2
PUMP-3

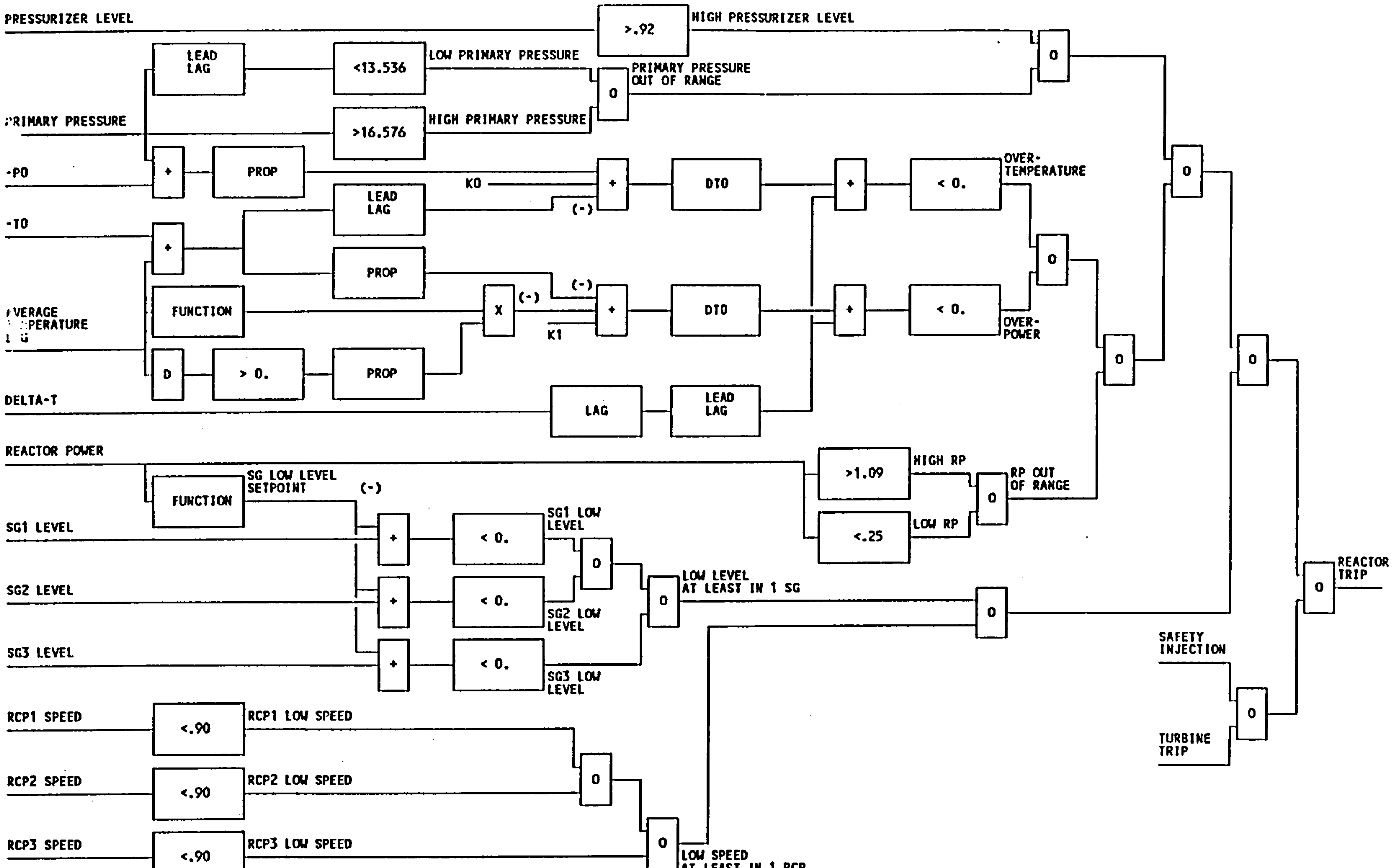


FIGURE 2. -LOGIC OF THE REACTOR PROTECTION SYSTEM

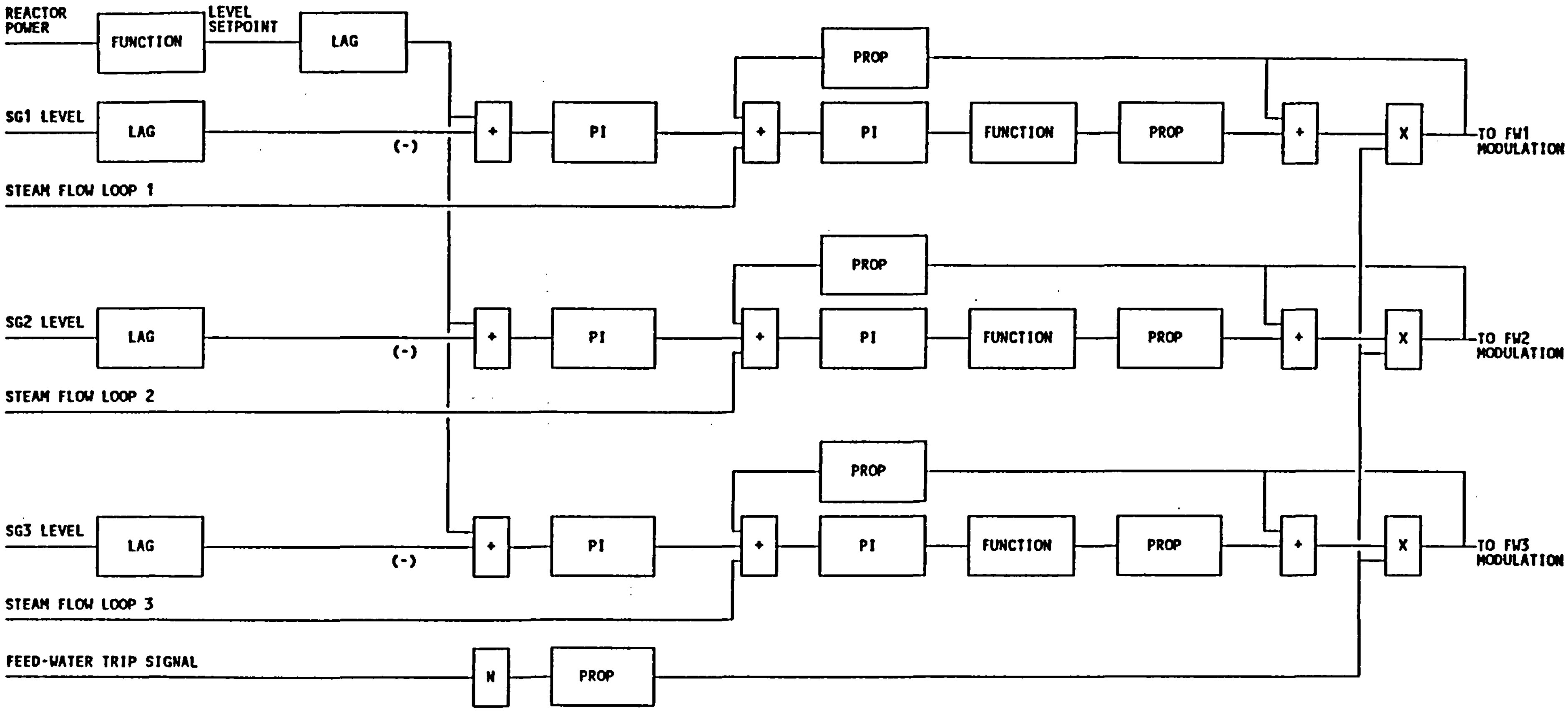


FIGURE 3. FEED-WATER CONTROL SYSTEM

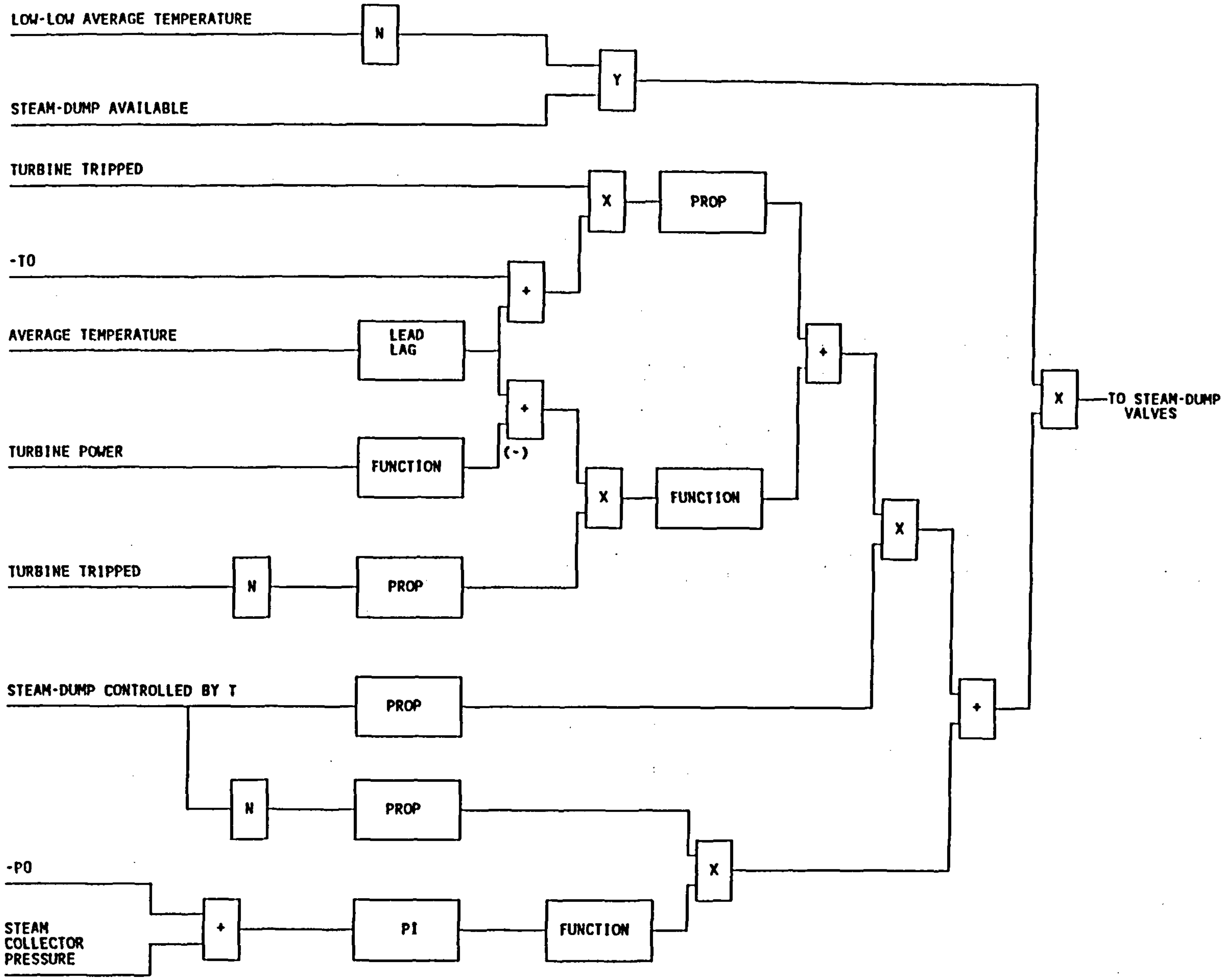


FIGURE. 4 STEAM-DUMP CONTROL SYSTEM

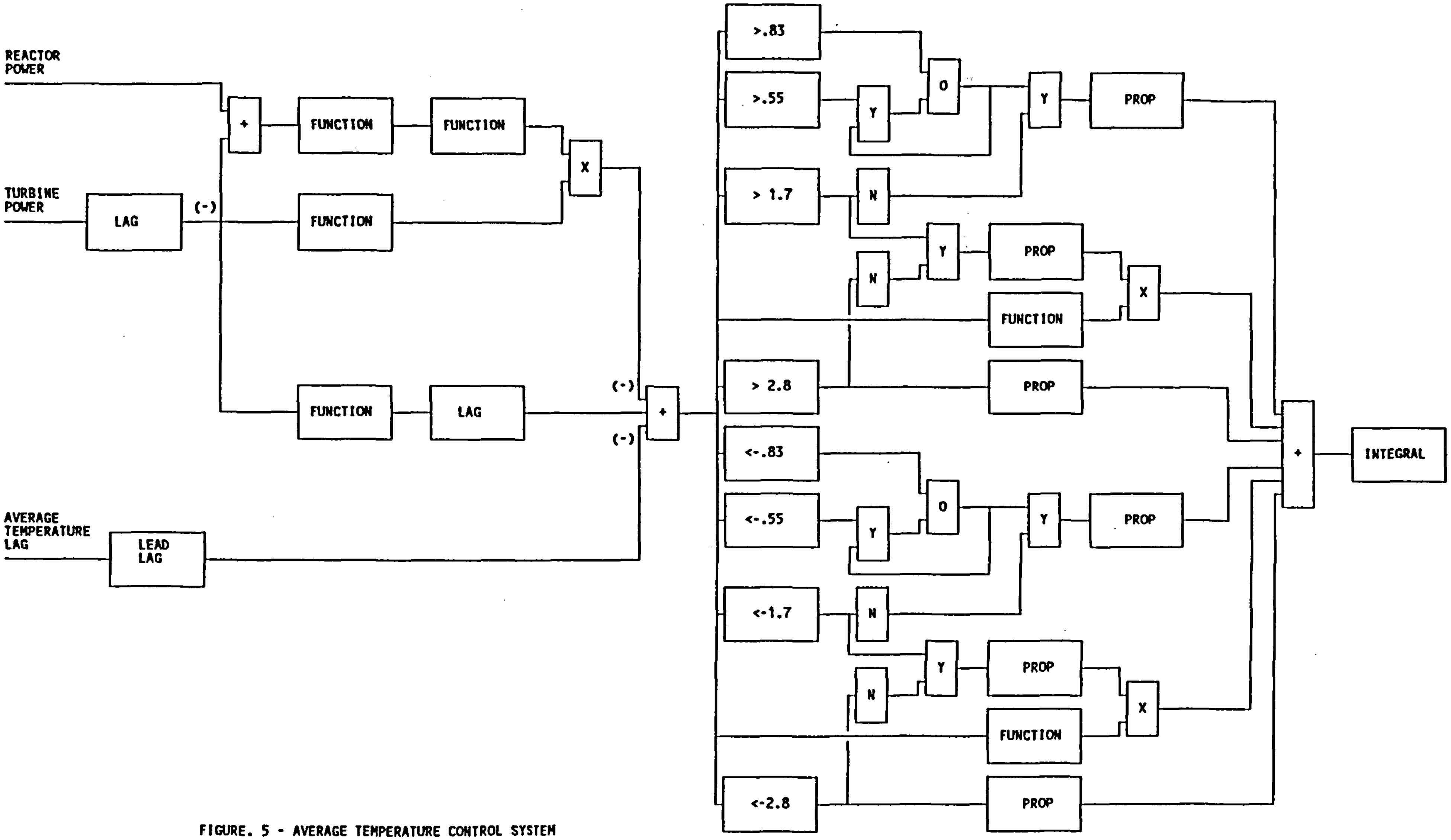
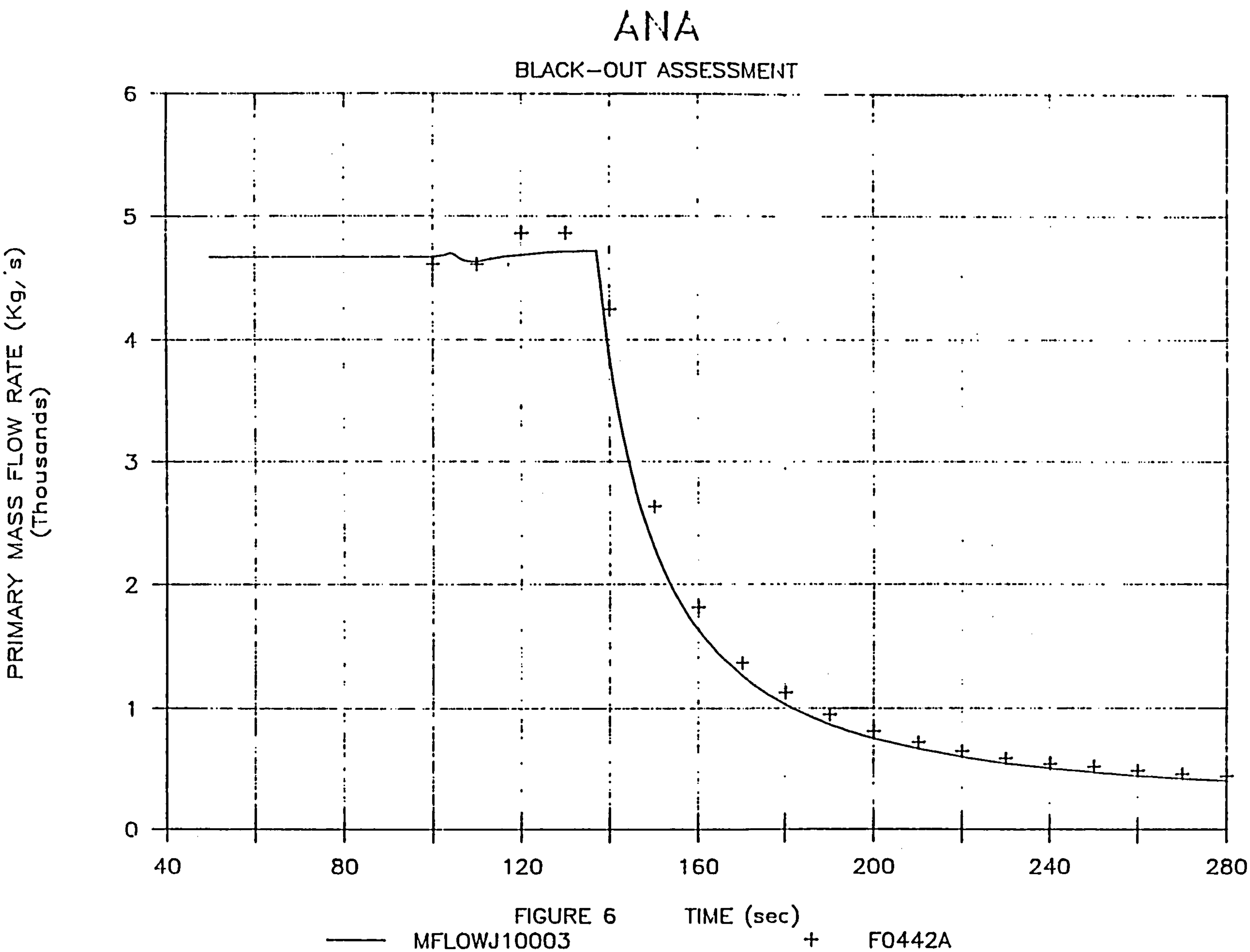


FIGURE. 5 - AVERAGE TEMPERATURE CONTROL SYSTEM



ANA
BLACK-OUT ASSESSMENT
PRIMARY MASS FLOW RATE (Kg/s)
(Thousands)
6
5
4
3
2
1
0
40
80
120
160
200
240
280
TIME (sec)
MFLOWJ10003
+ F0442A

FIGURE 6

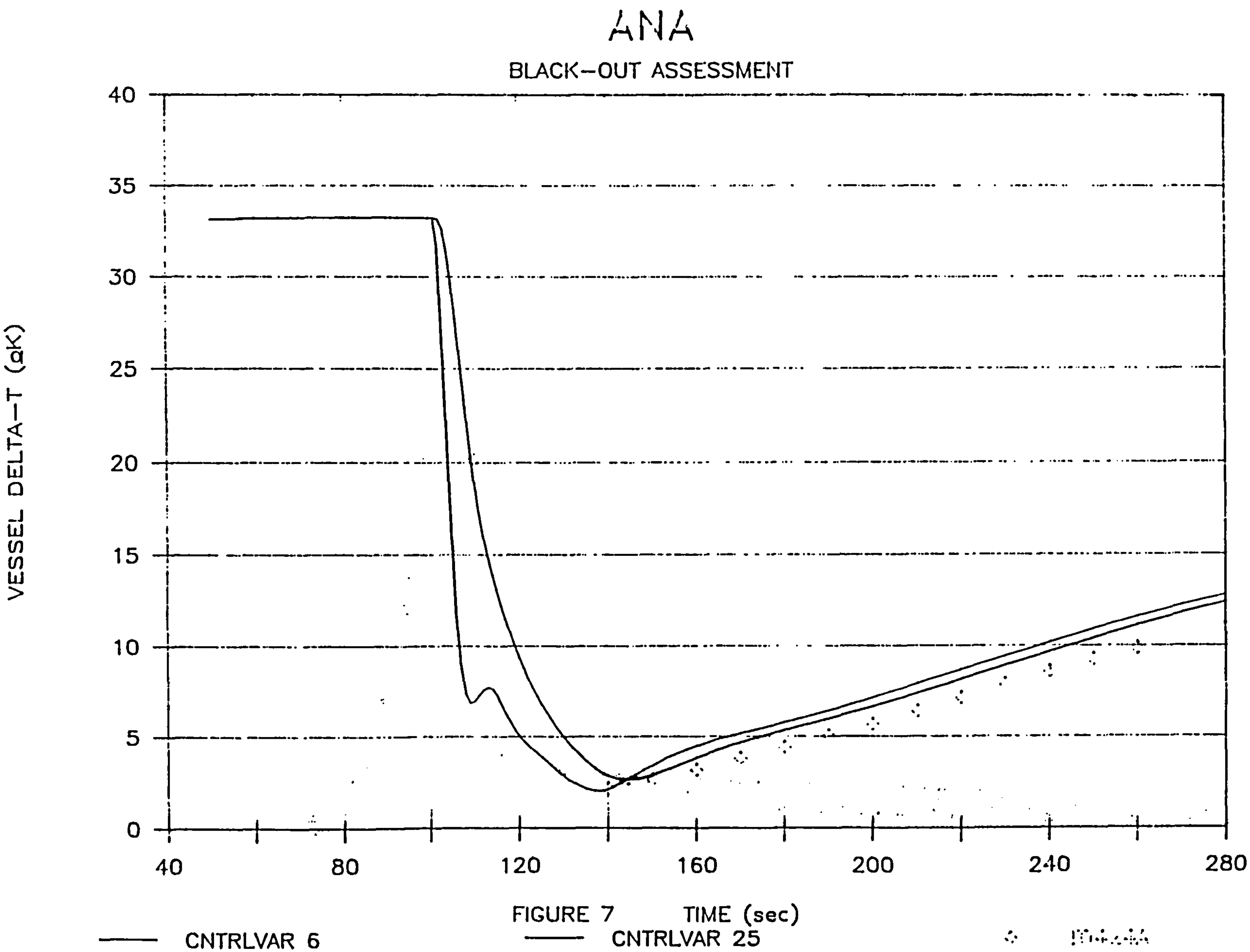
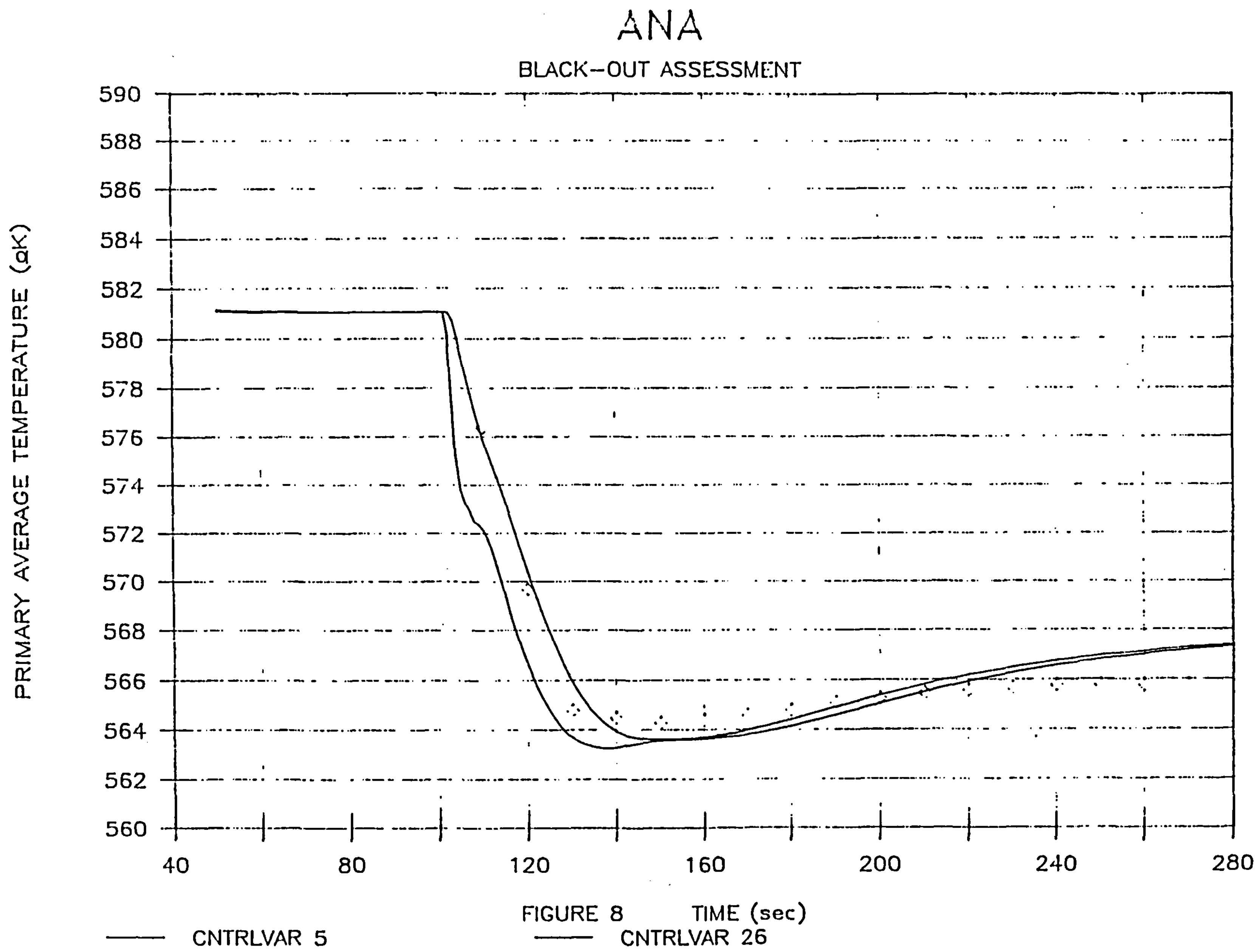
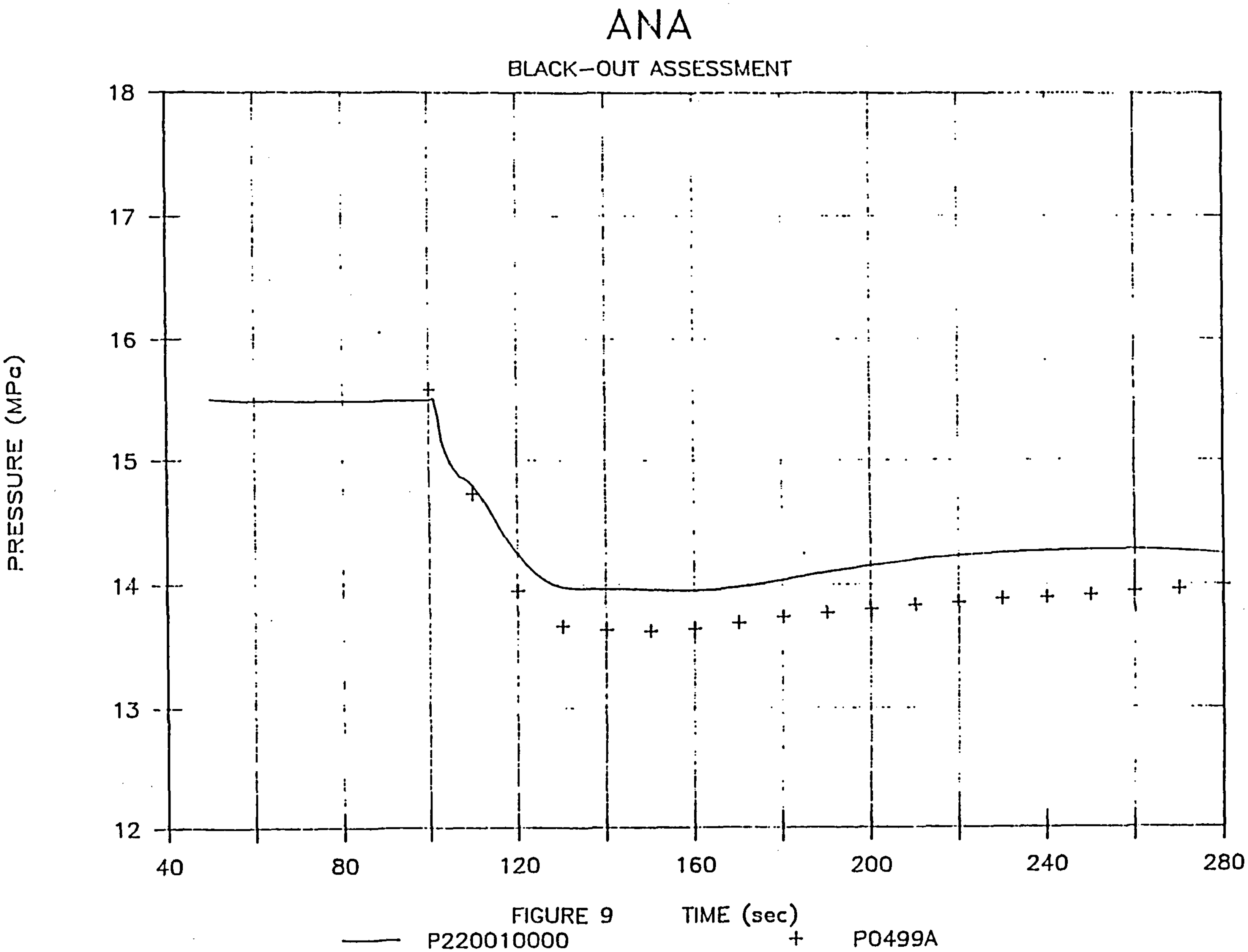

ANA
BLACK-OUT ASSESSMENT
VESSEL DELTA-T (aK)
40
35
30
25
20
15
10
5
0
40
80
120
160
200
240
280
TIME (sec)
CNTRLVAR 6
CNTRLVAR 25

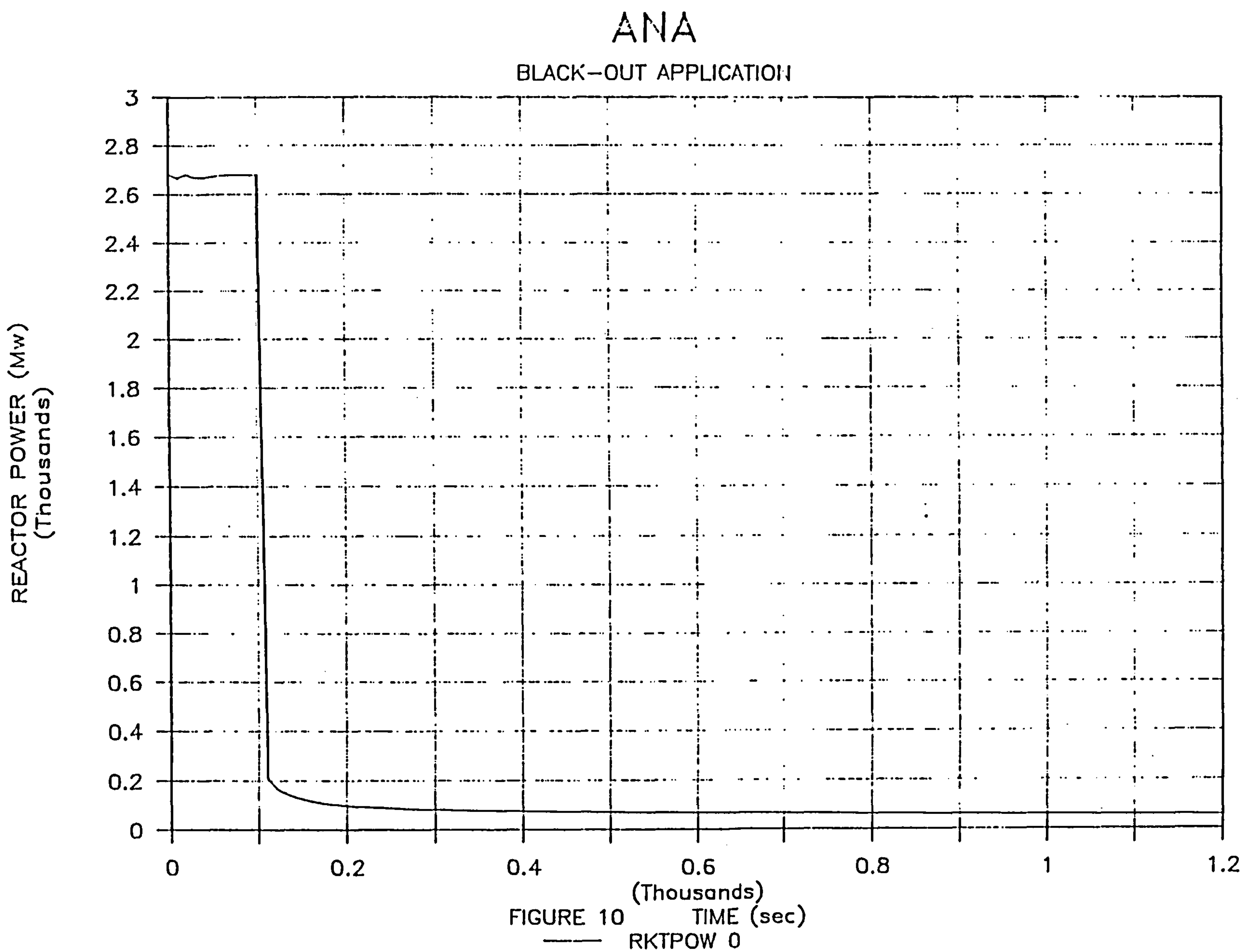
FIGURE 7


ANA
BLACK-OUT ASSESSMENT
PRIMARY AVERAGE TEMPERATURE (oK)
590
588
586
584
582
580
578
576
574
572
570
568
566
564
562
560
40
80
120
160
200
240
280
TIME (sec)
CNTRLVAR 5
CNTRLVAR 26
FIGURE 8



ANA
BLACK-OUT ASSESSMENT
PRESSURE (MPa)
18
17
16
15
14
13
12
40
80
120
160
200
240
280
TIME (sec)
P220010000
+
P0499A

FIGURE 9



ANA
BLACK-OUT APPLICATION
REACTOR POWER (Mw)
(Thousands)
3
2.8
2.6
2.4
2.2
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
RKTPOW 0

FIGURE 10

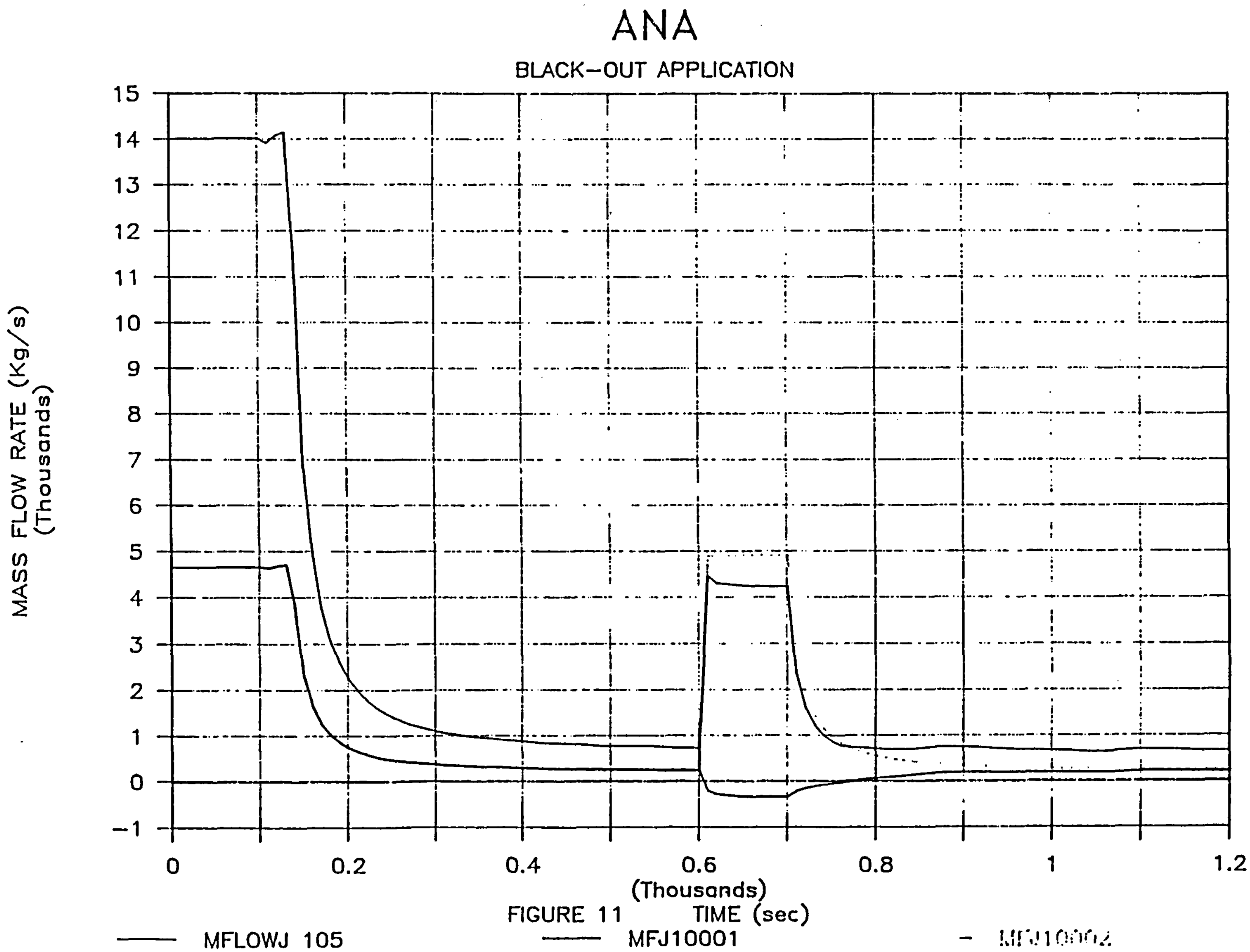

ANA
BLACK-OUT APPLICATION
MASS FLOW RATE (Kg/s)
(Thousands)
15
14
13
12
11
10
9
8
7
6
5
4
3
2
1
0
−1
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
MFLOWJ 105
MFJ10001
MFJ10002

FIGURE 11

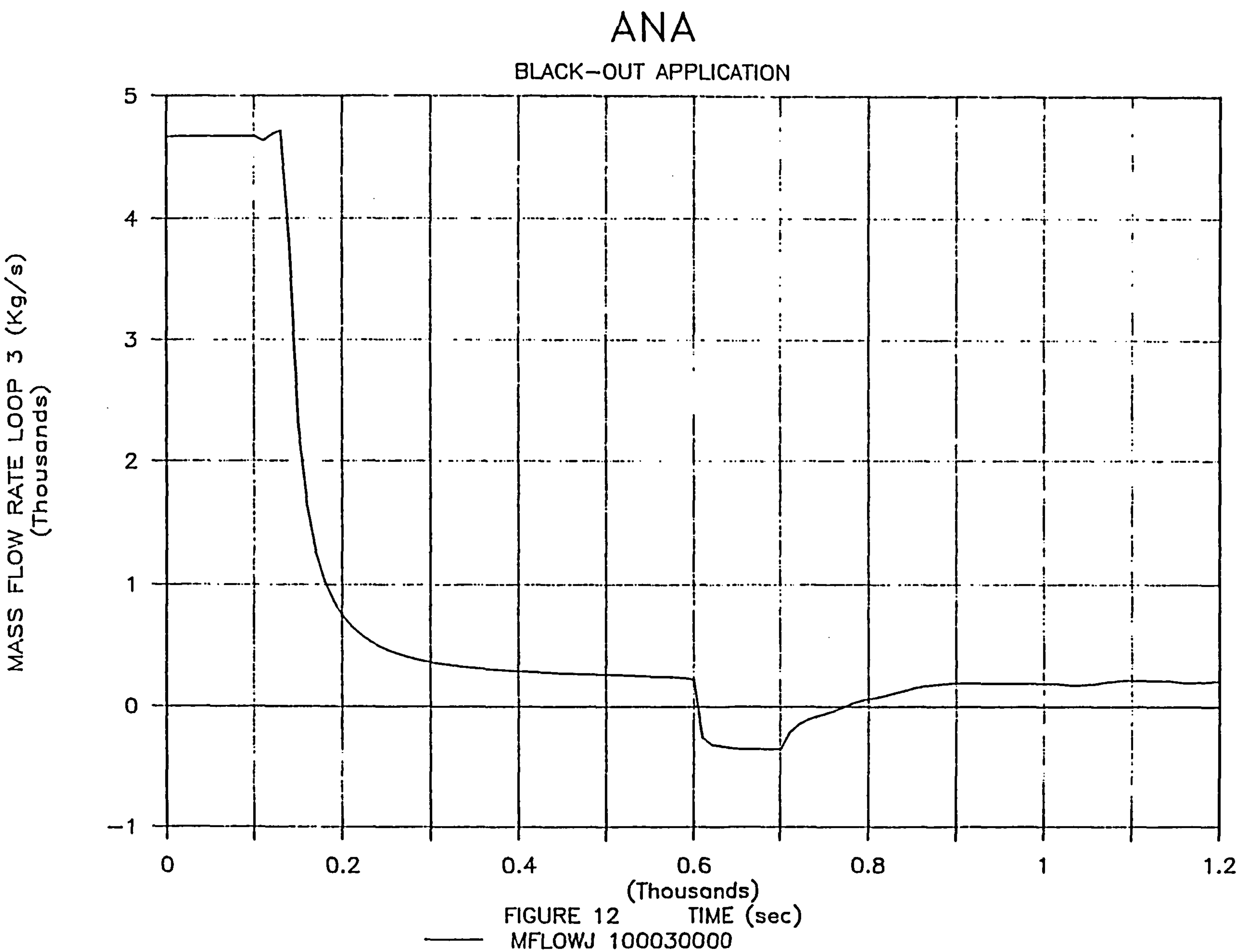

ANA
BLACK-OUT APPLICATION
MASS FLOW RATE LOOP 3 (Kg/s)
(Thousands)
5
4
3
2
1
0
-1
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
MFLOWJ 100030000

FIGURE 12

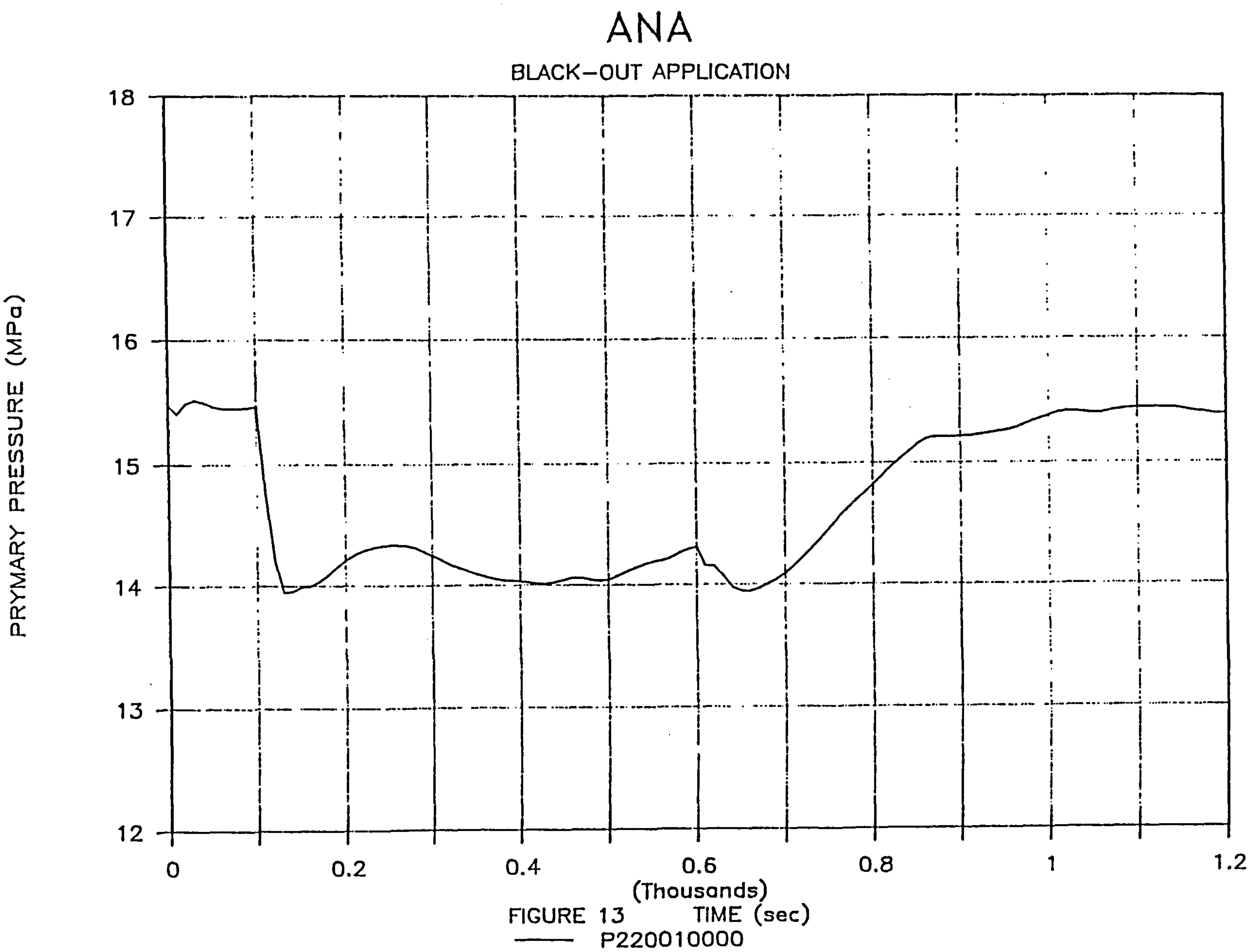

ANA
BLACK-OUT APPLICATION
PRYMARY PRESSURE (MPa)
18
17
16
15
14
13
12
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
P220010000

FIGURE 13

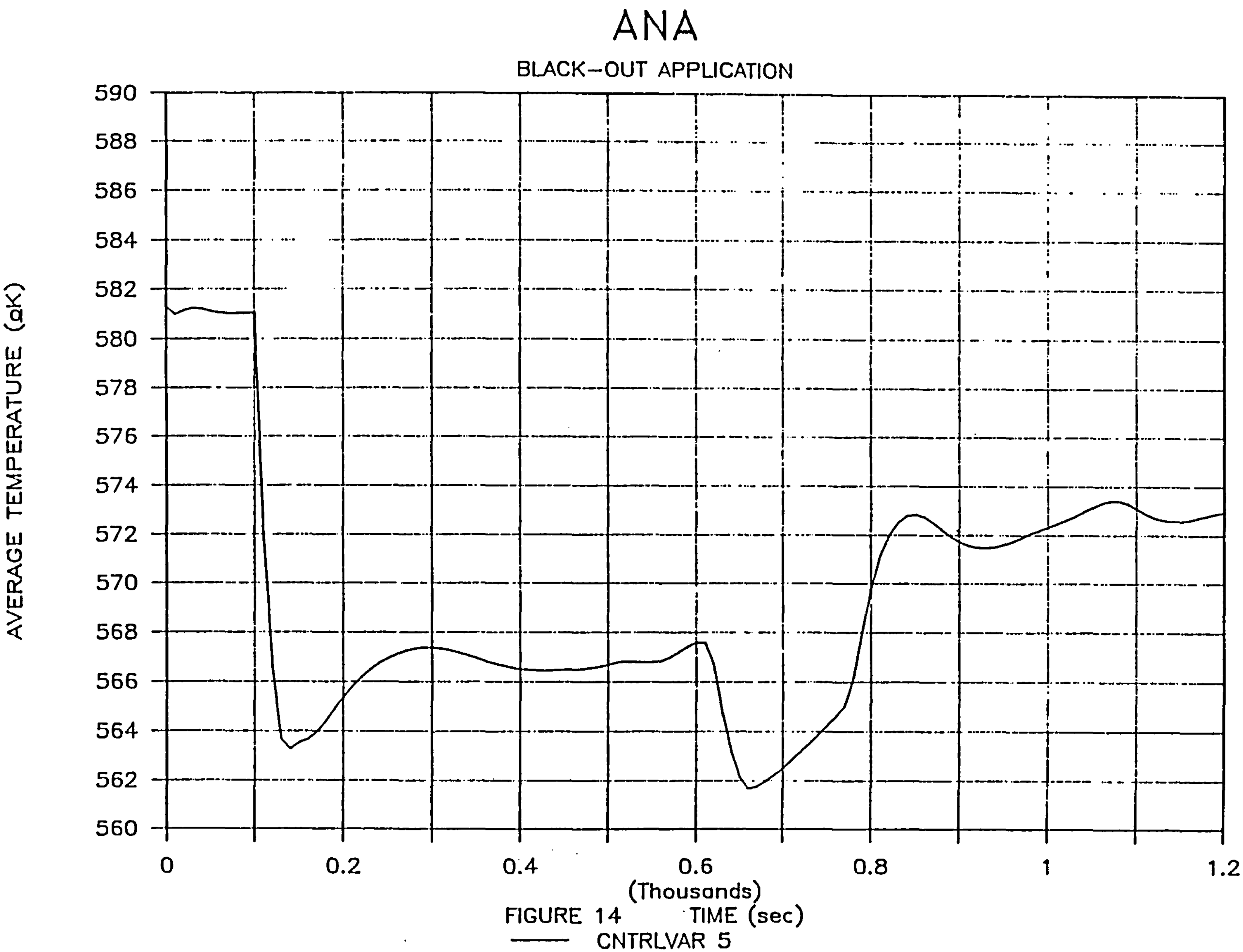

ANA
BLACK-OUT APPLICATION
AVERAGE TEMPERATURE (oK)
590
588
586
584
582
580
578
576
574
572
570
568
566
564
562
560
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
CNTRLVAR 5

FIGURE 14

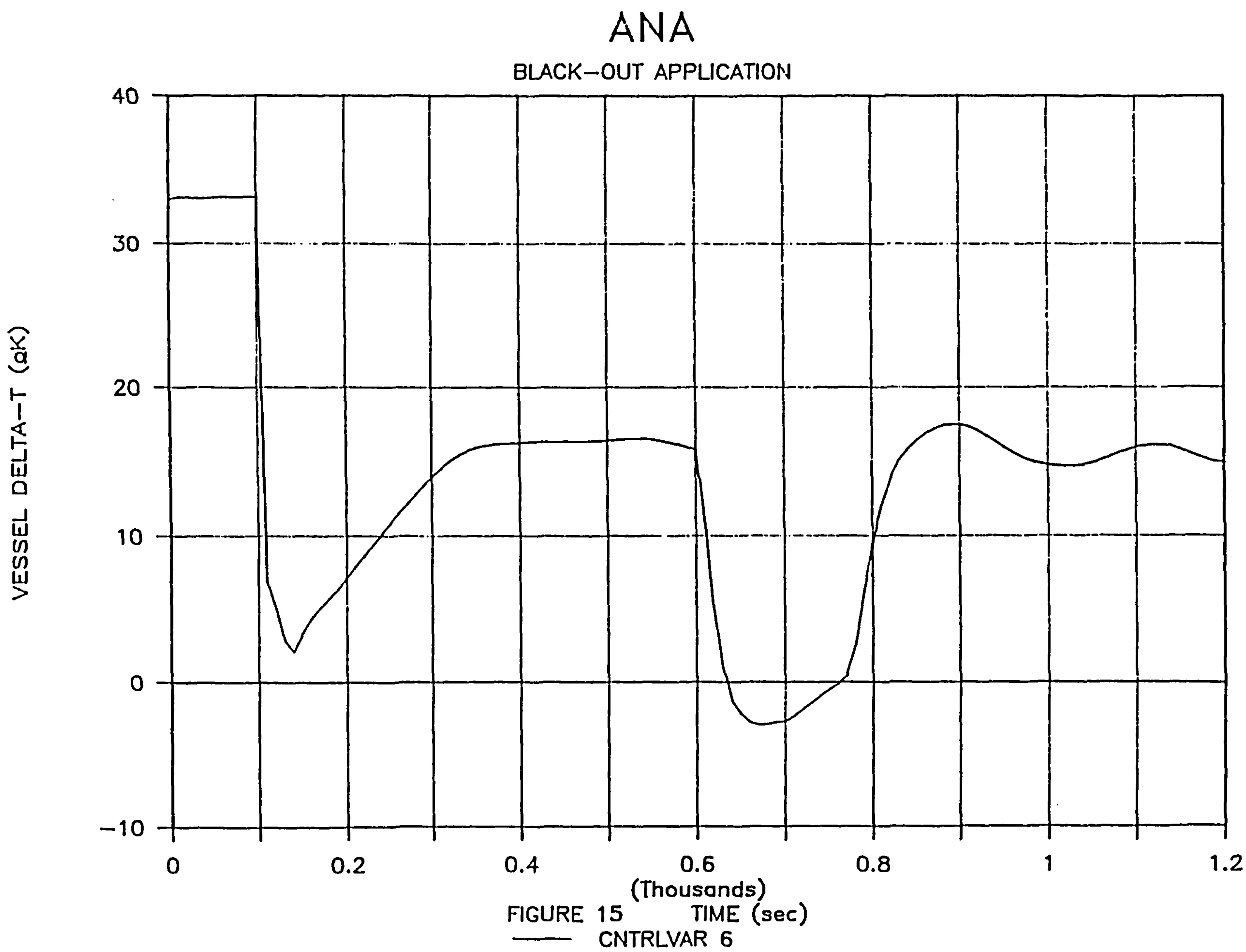

ANA
BLACK-OUT APPLICATION
VESSEL DELTA-T (ąK)
40
30
20
10
0
-10
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
CNTRLVAR 6

FIGURE 15

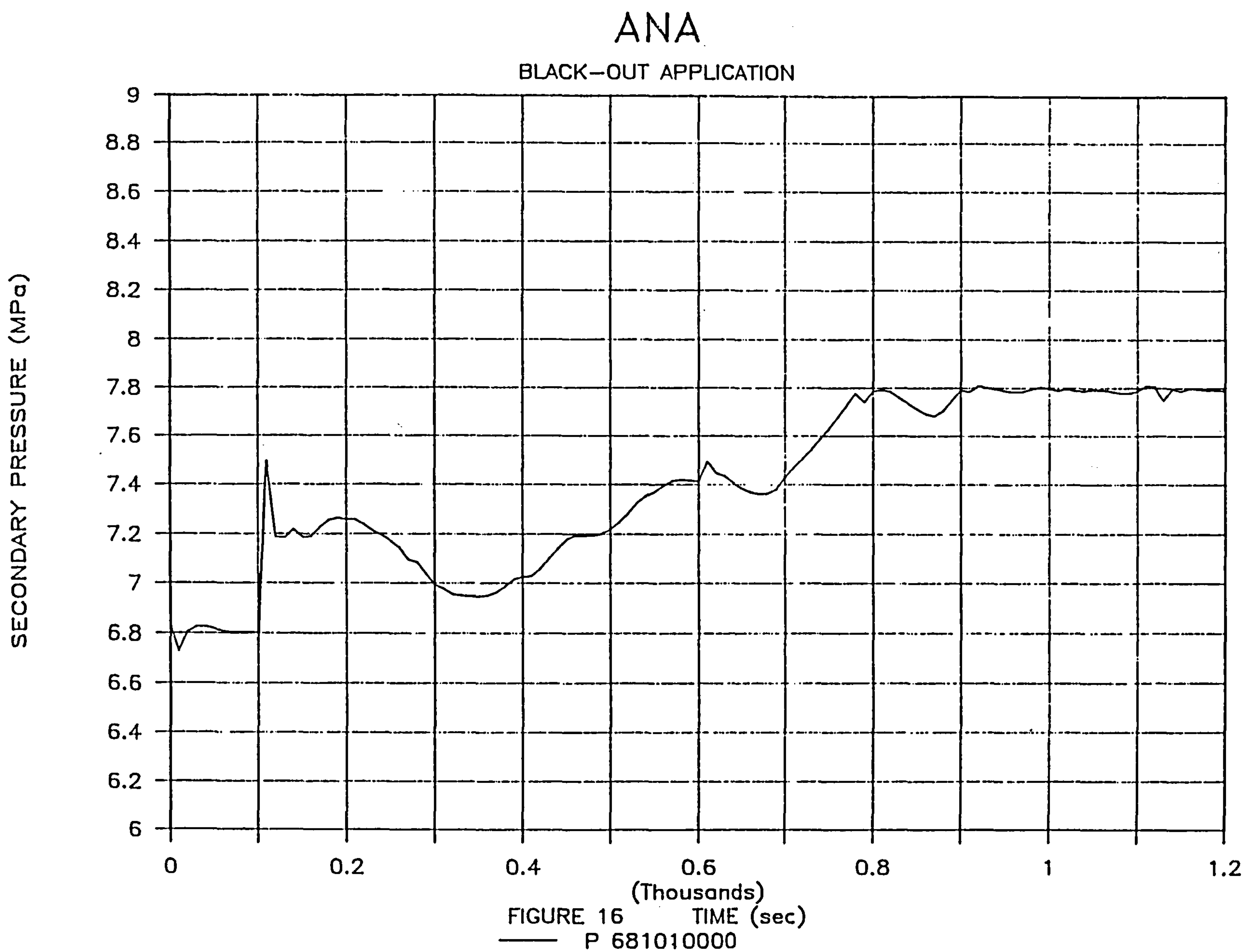

ANA
BLACK-OUT APPLICATION
SECONDARY PRESSURE (MPa)
9
8.8
8.6
8.4
8.2
8
7.8
7.6
7.4
7.2
7
6.8
6.6
6.4
6.2
6
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
P 681010000

FIGURE 16

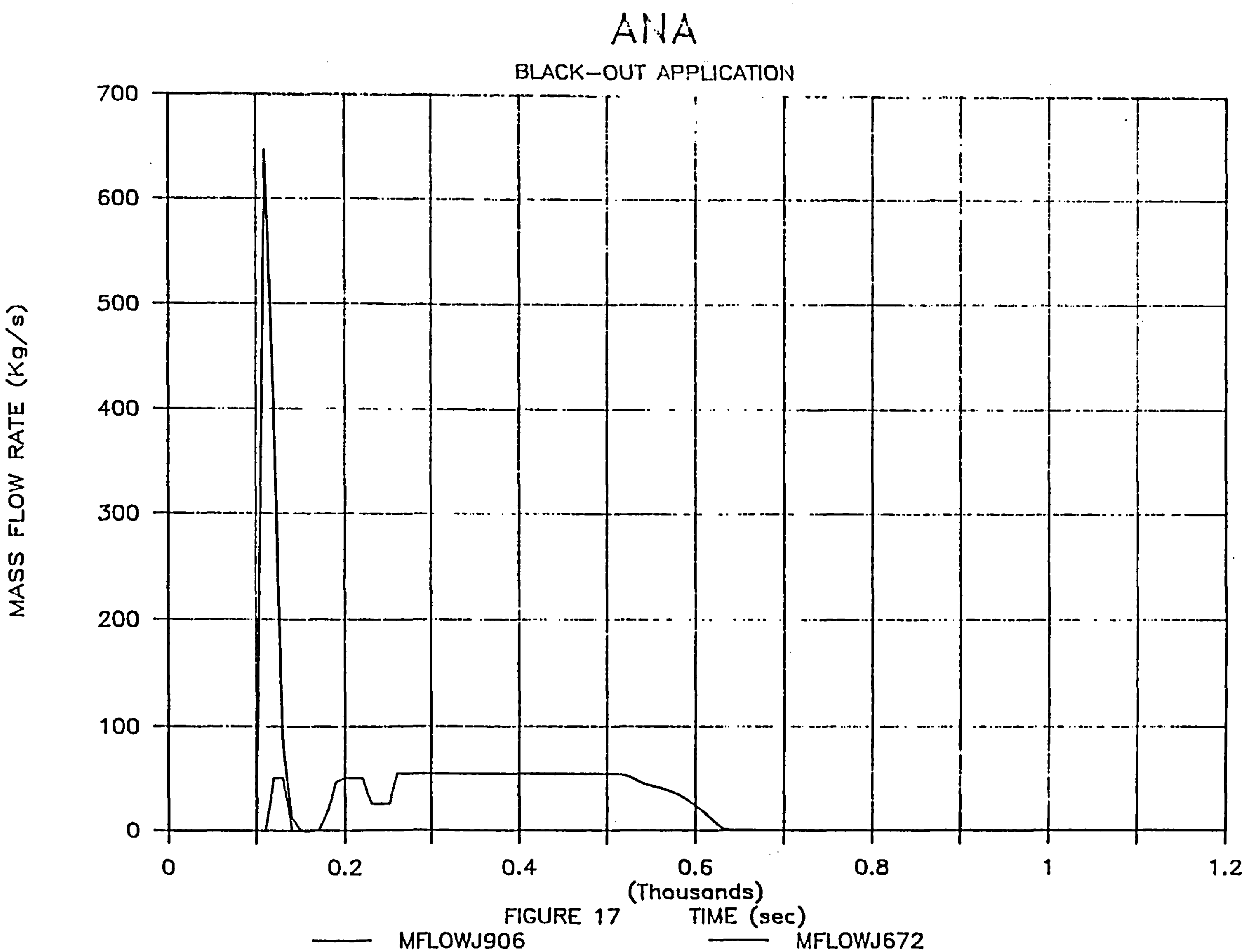
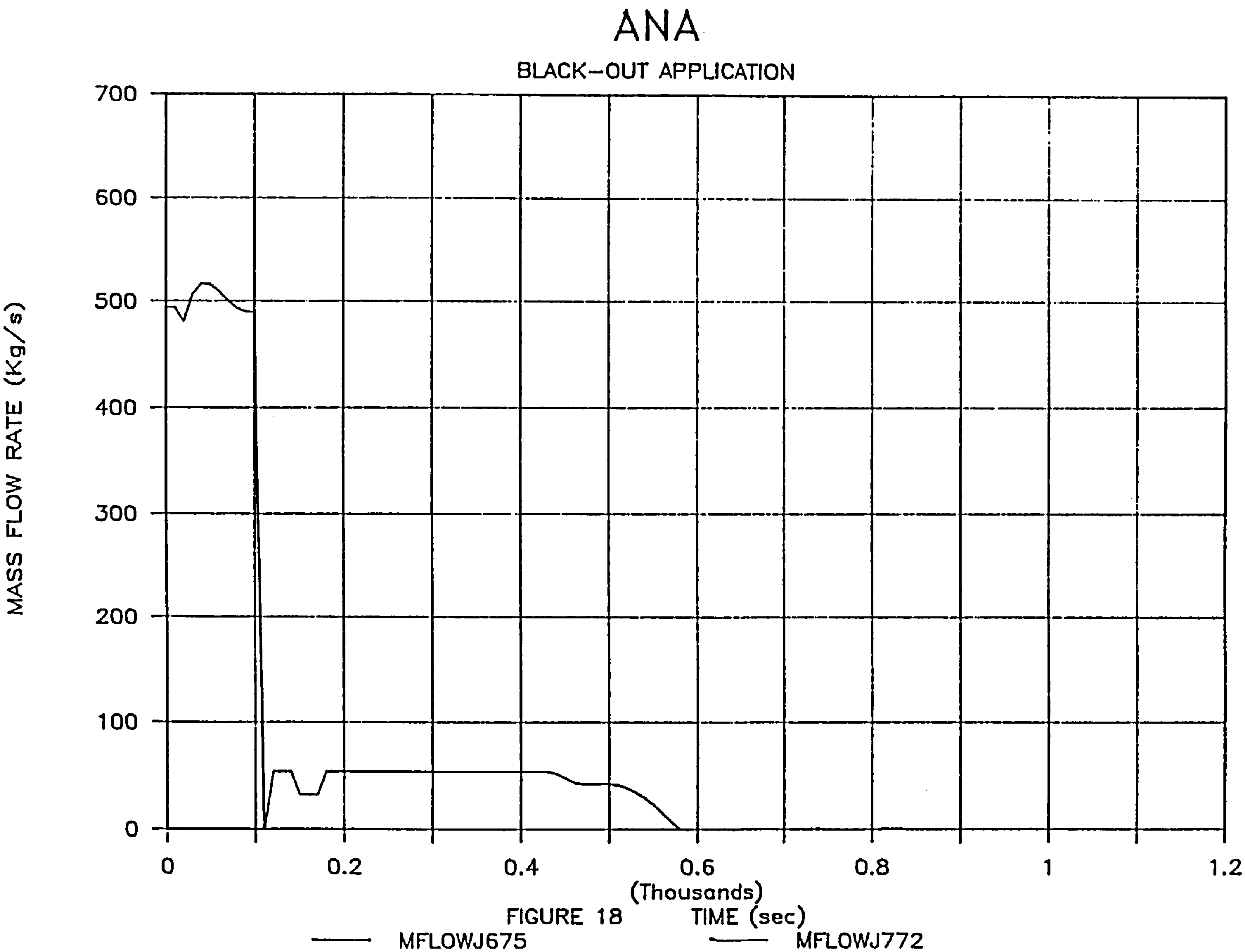

ANA
BLACK-OUT APPLICATION
700
600
500
400
300
200
100
0
MASS FLOW RATE (Kg/s)
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
MFLOWJ906
MFLOWJ672

FIGURE 17


ANA
BLACK-OUT APPLICATION
700
600
500
400
300
200
100
0
MASS FLOW RATE (Kg/s)
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
MFLOWJ675
MFLOWJ772
FIGURE 18

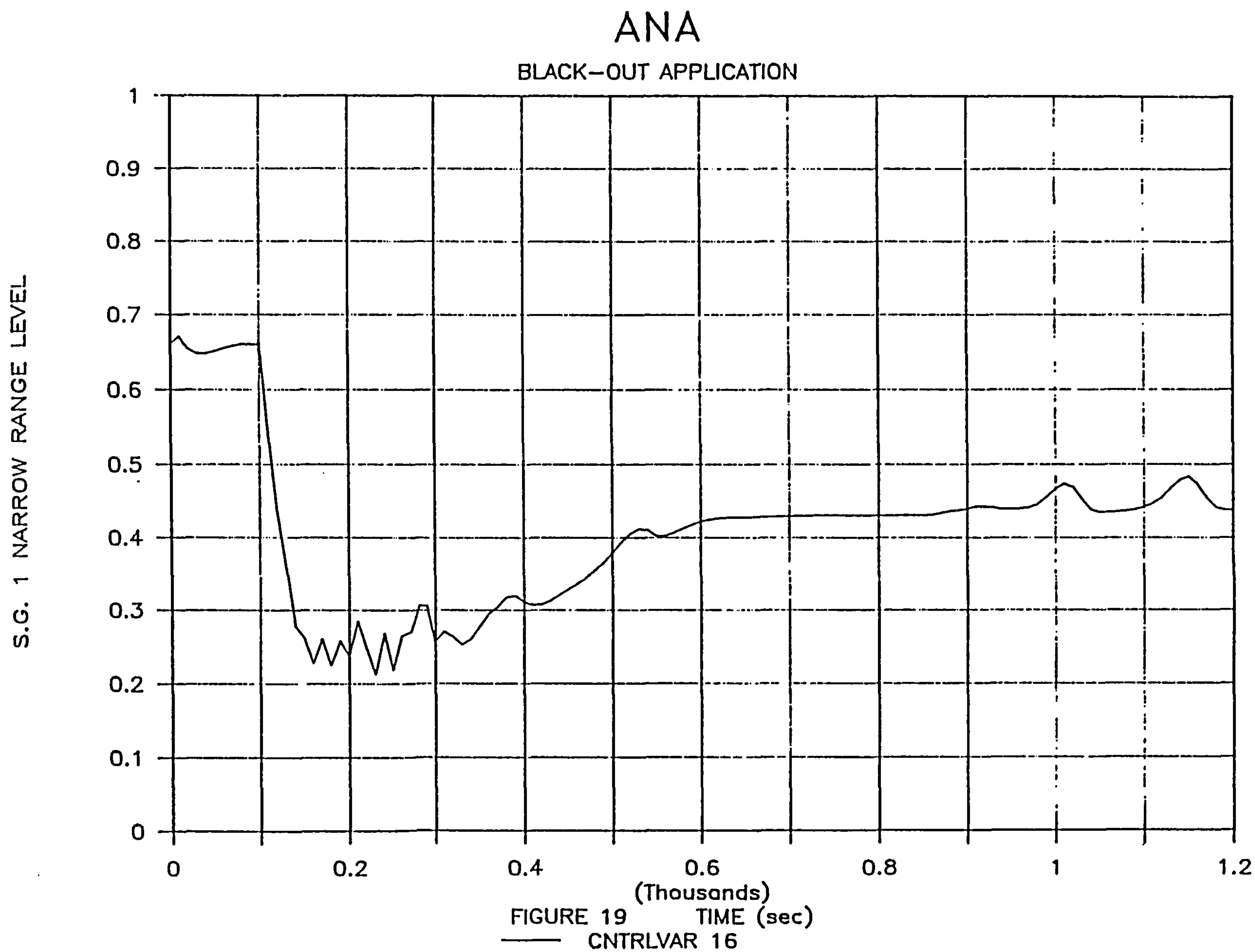

ANA
BLACK-OUT APPLICATION
S.G. 1 NARROW RANGE LEVEL
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
0.2
0.4
0.6
0.8
1
1.2
(Thousands)
TIME (sec)
CNTRLVAR 16

FIGURE 19

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

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Spain

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The Asociacion Nuclear Ascó has prepared a model of Ascó NPP using RELAP5/MOD2. This model, which include thermalhyraulics, kinetics and protection and controls, has been qualified in previous calculations of several actual plant transients.

The first part of the transient presented in this report is an actual black-out and one of the transients of the qualification process. The results are in agreement with plant data.

The second part of the transient is a hypothetical case. It consists in re-starting a primary pump and assume a new black-out.

The phenomenology prediction of this second part has been useful from the operation and safety point of view.

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

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NPP with RELAP5/MOD2
ASCO
Blackout

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