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ICSP-GA-MSIV-T

International Agreement Report

Assessment of MSIV Full Closure for Santa Maria De Garoña Nuclear Power Plant Using TRAC-BF1 (G1J1)

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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CONTENTS

EXECUTIVE SUMMARY

An assessment of the first 60 seconds of a spurious Main Steam Isolation Valve (MSIV's) closure for Santa María de Garoña Nuclear Power Plant using TRAC-BF1 code is presented. Reasonable and realistic adjustments have been made in the model to improve its perfomance.

This work is part of the validation set for the TRAC model that it is being developed for wider use and allow to test the code capabilities.

As a result of the analysis, it is felt that TRAC-BF1 is capable of reproducing the plant behaviour with an acceptable degree of accuracy although better models are clearly needed, in addition to further noding work and code improvements. The code took almost 14000 sec. which makes a 1/230 calculation time to real time ratio.

For this transient a mechanistic separator model is needed. It will also help to cut down running costs if the vessel noding could have different number of cells at different heigths. Though not very important for this transient, the critical flow model will allow for realistic RV flow assumptions.

There are not guidelines available for separator modelling in transients. It has been found that a detailed noding in the separator region may be needed to represent steam-water interaction.

ABSTRACT

This document presents a spurious MSIV closure analysis for Santa María de Garoña Nuclear Power Plan describing the problems found when comparing calculated and real data.

The plant is a General Electric Boiling Water Reactor 3, containment type Mark I. It is operated by NUCLENOR, S.A. and was connected to the grid in 1971.

The analysis has been performed by the Applied Physics Department from the University of Cantabria and the Analysis and Operation Section from NUCLENOR, S.A. as a part of an agreedment for developing an engineering simulator of operational transients and accidents for Santa María de Garoña Power Plant.

The analysis was performed using the frozen version of TRAC-BF1 (G1J1) code and is the second of two NUCLENOR contributions to the International Code Applications and Assessment Program (ICAP).

The code was run in a Cyber 932 with operating system NOS/VE, property of NUCLENOR, S.A.. A programming effort was carried out in order to provide suitable graphics from the output file.

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FOREWORD

This report has been prepared by NUCLENOR 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.

I. INTRODUCTION

NUCLENOR, a Spanish Electrical Utility which owns the Santa María de Garoña Nuclear Power Plant made a decision in 1987 of carrying out a project to develop a specific engineering simulator of accidents and transients based on TRAC-BF1 and BWR-LTAS codes. In order to obtain the TRAC-BF1 code NUCLENOR joined ICAP-Spain program through UNESA under the compromise of carrying out one plant-specific application case and one plant-specific assessment case with the code.

NUCLENOR selected as an assesment case the analysis of the first 60 seconds a spurious full MSIV closure. The analysis, described in the present report, was carried out by the Applied Physics Department of University of Cantabria and the Analysis and Operations Section of NUCLENOR.

The transient consisted of a full MSIV closure at 100% power which actually took place in the plant, so it will help to time the model for further analysis.
The model was changed to improve its accuracy, but all the changes were made only if there was a physical support. The modelling accuracy had to trade off with the simplicity needed to keep down running costs.

A summary description of the Santa María de Garoña Nuclear Power Plant and the simulated accident is given in Section II. Section III describes the code input model nodalization. Calculation results and discusion of main phenomena are presented in Section IV. Run statistics are summarized in Section V and the conclusions are given in Section VI.

II. PLANT AND TRANSIENT DESCRIPTION

II.1 PLANT DESCRIPTION

Santa María de Garoña Nuclear Power Plant is a General Electric Boiling Water Reactor 3, with containment type Mark I. The plant is operated by NUCLENOR, which is a subsidiary of Iberduero, S.A. and Electra de Viesgo, S.A., and was connected to the grid in 1971. The plant is rated at 1380 Mw (thermal) and is located in the province of Burgos (Spain).

The nuclear boiler assembly consists of the reactor pressure vessel and internal reactor components such as the core structure, steam dryer assembly, fuel supports and control guide tubes.

The reactor core is made up of 400 fuel assemblies and 97 control rod blades. Each fuel assembly has 64 rods in a square array (8 x 8). Each control rod blade consists of sheathed cruciform array of vertical absorber rods made of boron carbide (B_4C). A complete description of relevant parameters of the plant is shown in Table II-1.

A Recirculation System, consisting of two external centrifugal recirculation pumps (variable speed) and twenty reactor vessel internal jet pumps, provides the core coolant flow to meet the plant thermal rating. This coolant consists of saturated water rejected from the steam separators and dryers that has mixed with subcooled feedwater entering the vessel at the steam separator elevation (9.33 m. above the downcomer bottom). The recirculation pumps suction exits the vessel at an elevation just above the shroud support ring, goes through the outside loop and re-enters the vessel through a riser pipe to become the driving flow for the jet pumps. The remainder of the coolant is thus entrained in the jet pump, mixing with the driven flow in the pump throat section. Flow then exits the jet pump via the diffuser section and is directed to the core inlet plenum.

Main Steam System consist of four lines that penetrate the reactor vessel and provide steam to the turbine at rated operating conditions. Three relief valves (RV's), two safety-relief (SRV's) valves discharging into the Suppression Pool, and seven Safety Valves (SV's) discharging into the Drywell, are installed on the steam lines. As well as, there are two isolation valves per line and one flow restrictor per steam line downstream RV's and SRV's.

The reactor vessel is inside an inerted containment called Drywell and the Supression Pool is connected to the drywell through vent pipes.

An Isolation Condenser System allows for condensating steam from the reactor vessel and returning the condensate to one of the recirculation loops, when the vessel is isolated from turbine and main condenser.

A Feedwater System made up for the inventory coming out of the vessel through main steam line by two out of three motor-driver pumps. A Feedwater Control System assures the water level in the downcomer remains between predetermined limits.

II.2 TRANSIENT DESCRIPTION

A spurious Main Steam Isolation Valve (MSIV's) closure signal occurs at time zero. The core was operating at 100% power and 89% core flow at the start of the transient. No relevant manual actions took place the Isolation Condenser System is not activated while Feedwater and Recirculation System continue in operation during the first 60 seconds.

The MSIV's closed completely in about 3 seconds. The SCRAM took place when the valves were at 90% position. The steam that was still coming out of the core caused a pressure increase till the RV's setpoints were reached; two of them opened for 3 seconds and the third one opened for 5 seconds. Even after the RV's closure the pressure kept decreasing due to condensation around the feedwater sparger and the separators.

The vessel level fell down because of void collapse until the feedwater flow recovered it.

The core flow varied because of voids redistribution throughout the vessel, getting larger as it moved from two-phase to single phase pressure drop. This helped to accelerate the vessel depresurization and cooling. The drive flow kept the same because of the flow controller being in manual mode. Figures II.1 though II.4 show the most important variables recorded from the process computer.The instrumentation precision is:

Dome pressure	0.5%
Core flow	3%
Downcomer level	0.5%
Feedwater flow	2%

III. CODE INPUT MODEL DESCRIPTION

The Transient Reactor Analysis Code (TRAC) is an advanced best estimate code for analyzing Light Water Reactor (LWR) accidents. The version for Boling Water Reactors was developed at INEL (Idaho) under the sponsorship of the Reactor Safety Research Division of the US Nuclear Regulatory Commission (NRC).

Specifically, TRAC-BF1/G1J1 was developed to analyze postulated transient and accidents in BWRs. It uses a full developed two fluid model with one-dimensional geometry, except for the vessel where the resolution is three dimensional. The two-fluid model, in conjuntion with a model of stratified flow regime, handles a countercurrent flow treatment in a best estimate way that the drift-flux model used in the past versions of TRAC-B.

Several different types of hydrodynamic components, including PIPE, VALVE, CHAN, VESSEL, PUMP TEE, JETPUMP, BREAK and FILL components, are used in TRAC-BF1. The BREAK and FILL components are used to impose thermalhydraulic boundary conditions. The other components can be used to represent different types of hardware such as a pipe, valve, fuel channel, reactor pressure vessel, jet pump, or pump. The user can node all the components, except BREAK and FILL components, with as many hydrodynamic cells as desired. The reactor vessel, both recirculation loops, and portions of the feedwater, steam, and safety systems were represented in the TRAC-BF1 model.

The development of TRAC-BF1 input deck for this analysis was based on data taken from drawings, and specific technical documents. (Ref. 1)

The model contains 35 components with 33 fluid juntions. Table III-1 includes a listing of all the TRAC components used in the model. Figure III-1 shows a schematic of the TRAC-BF1 model. The illustrated model represents Garoña for normal operation only. If should be noted that this is a generic model, not specifically related to this transient.

The reactor vessel has been modeled by the VESSEL component with the nodalization shown in Figure III--2. This nodalization is made up of ten axial levels, three radial segments and one azimuthal sector.

The radial segments correspond to the separator, surrounding region and downcomer region.Since only one average bundle is studied, two rings are actually used below the separator. Though the dryers skirt has a different inner diameter than the downcomer, no extra segment was considered. The difference is taken into account by volume and area fractions. This avoids large number of cells in the vessel model

The first axial level extends from the vessel bottom to the jetpumps discharge support ring. The second one goes from these support ring to the core bottom. The core is divided into two axial levels (3 and 4) of different lenght. The remaining axial levels 5 through 10, represent the steam dome, upper plenum, with the feedwater inlet at axial level 6 and the main steam line outlet at the axial level 9. The downcomer extends from axial level 2 to axial level 5.

Instead of individual steam separator and dryer components, a perfect separator option is used in axial level 8. This allows the vapor to continue upwards into the axial level 9 and the liquid to drain radially outward into the sorrounding region.

The guide tubes are not modelled and their volume has been substracted from the lower plenum.

Fuel bundle modelling is accomplished using one CHAN component simulating the active region of the 400 bundles.

The axial distribution of the CHAN components consists of 12 axial levels. The bypass flow path is modelled in the VESSEL component. Only an average rod group is modelled.

Axial power profile was taken from plant data recorded 2 days afterwards for the same core flow and power. Radial fuel rod dimensions represent beginning of life values. Reactor decay power history was based on the American Nuclear Society decay heat standart (ANS-5.1) built into the code. Although the control rods are not physically modeled, the negative reactivity insertion associated with the rods is accounted for a scram reactivity vs. time table derived from Reference 2.

Both recirculation loops have been modelled by six TRAC-BF1 component, neglecting the small differences between them.

- TEE 20: suction piping conected to the vessel that accounts for IC connection.
- VALVE 21: suction valves.
- PUMP 22: main pumps.
- VALVE 23: discharge valves.
- TEE 24: discharge piping accounting for the LPCI connection.
- JETPUMP 25: jet pumps.

The noding through the recirculation system is almost the minimum that takes into account all connections. The PUMP component represents only the fluid volume between the inlet and outlet nozzles of the pump. The internal friction is accounted for in the pump homologus curves.

Piping comprising the feedwater system, main steam line, and core emergency cooling system are also connected to the vessel. The feedwater line is modeled using a FILL component (FILL number 61) connected to the VESSEL at the level 6 into the ring 3 via one pipe (PIPE number 60). Also connected to this pipe is the HPCI (FILL number 62) by one leak path system.

The main steam lines are modelled as follows:

- TEE 50:	connection to the VESSEL and RV's.
- VALVE 51:	RV's & SRV's.
- TEE 52:	connection to SV's.
- VALVE 53:	SV's.
- VALVE 54:	MSIV'S.
- TEE 55:	connection of the bypass line.
- VALVE 69;	TCV's.
- VALVE 68:	Bypass valve.
- BREAK 58:	Turbine back pressure.
- BREAK 56:	Suppression pool pressure.
- BREAK 57:	Drywell pressure.
- BREAK 59:	Bypass back pressure.

The reactor point kinectics option was turned on in the Power Cards to calculate the core power rather than specifying it as a funtion of time after a trip. The information used is contained in Reference 2 to specify the programmed reactivity associated with the control rods and to specify the reactivity-feedback coefficient for changes in core-averaged fuel, coolant temperature, void fraction and core-average boron concentration.

The critical flow model was applied at the RV's, adjusting the area in order to match the flow measured during the startup tests. Trying the nominal area produced 30% more than the certified capacity, what was considered too high.

Control systems are modelled for the pressure regulator and trip logics.

Although a feedwater control model was available, it was not considered in the analysis because the performance of the control valves in the trasient resulted a hard task to model.

In order to run the transient, the MSIV trip was activated. The following boundary conditions were assumed:

a.- Decay Heat Generation rate was used the American Nuclear Society (ANS) 5.1 with a multiplier factor of 1.0.

b.- No consider heat losses to the drywell environment.

c.- Heat conduction through the vessel intermals is neglected except for the bundle wall.

'.- The backpressure at the turbine while the MSIV's were closing was assumed constant.

e.- The feedwater flow was input from plant data because the control valves were presumably locked for part of the transient, producing an abnormal response.

f.- Feedwater heaters inertia was assumed to be large enough as to avoid significant temperature changes.

The model has been developed by the analysis from simpler to more complex models, concerning the upper part of the vessel and the separator location. This is presented as a sensitivity analysis.

IV.. CALCULATION RESULTS

The calculation of the transient with TRAC-BF1/G1J1 has been developed in two steps. First of all, a steady state calculation was carried out. The second step consisted of the transient calculation.

The convergence criteria for the outer iteration (EPSO), and the vessel iteration (EPSI), were 1.0E-04, 1.0E-05, respectively. The maximum number of the outer iterations (OITMAX) was 8 and the VESSEL calculation was solved by direct inversion.

From the initial conditions described in Table IV-1, and the assumptions described in chapter III the transient calculation was run.

The chronology of events is summarized in the Table IV-2 and the main results are represented in the figures IV-1 through IV-5.

The comparison between TRAC and the plant is limited to the available plant data, that is, dome pressure, core flow, downcomer level and core plate Δp. It was also recorded the APRM's signal but since the SCRAM takes place in the first second and the data were taken every 5 seconds, there is no relevant information for this transient in that plot.

Figure IV-1 shows reactor pressure. The pressure peak is well represented. However, after the RV closure the code predicts a higher depresurization rate, although reaching similar minimum 10 seconds in advance. This is related to the condensing effect of the water flowing out of the separators, that is the main issue in this calculation.

The separator model used send the water downwards into the inner zone and the steam upwards into the upper zone. The separator model sends the water radially into the surrounding region and the steam upwards into the upper cell.

The separator model used have a good performance at rated conditions. After the SCRAM there is very little power in the core, but there is full core flow and the feedwater is making it cooler; as a result the core outlet quality gets close to 0. Figure IV-2 shows calculated mixing plenum void fraction. Simultaneously, due to the SCRAM a large level change takes place. These two facts are going to affect significantly the separator performance (carryover, carryunder, pressure drop). Obviously, it is different sending water under the water level and sending it upwards into the steam upper zone.

The separator model is ideal. No matter what the conditions are, there will not be water flow upwards or steam flow outwards, and this has not to occur necessarily at off rated conditions.

Since the pressure is dependent on the steam-water mixing process, no great accuracy can be expected with this separator model (other separators models in TRAC do not work properly).

A different analysis was made with a different cure pressure drop. The calculated pressure was closer to the data but the core flow was clearly lower.

Figure IV-3 shows core flow. The calculated results are good in general, but around 20 sec. the code produces lower flow not seen in the data. This is due to cavitation in the jet pump as can be seen in Figure IV-4, which shows the throat void fraction.

Figure IV-5 shows water level in the downcomer. TRAC results refer to collapsed level. The calculation is good in the first 40 seconds, but gets too high afterwards, with a final difference of about 0.5 m. This is believed to be caused by a lack of better modelling for the flow inside the dryer skirt, which is related to the separator and upper cell.

The main changes made, when developing the model, affected noding in the upper part of the vessel and separator location as well as pressure losses distribution. In the following a brief description of the process is given.

Figure IV-6 shows the results using a much simpler model obtained collapsing radial segments 2 and 3 into one and axial levels 7, 8 and 9 into one, with the separator located at level 6, that is, under level 9.33 m. It can be seen that as soon as this is covered the pressure rises again.

Figure IV-7 shows the results obtained using again 2 radial segments and collapsing levels 6 and 7 into one and 8 and 9 into another one.

The results show the same tendency. The separator discharge was under 9.83 m.

Figure IV-8 corresponds to a model where there are two radial segments and levels 7 and 8 are collapsed into one with the separator discharge in it.

V. RUN STATISTICS

TRAC includes a logic that may limit internally the time step, unless the user specifies maximum time step size. This logic is based on parameters such as the material Courant limit in the vessel, pressure rates, temperatures and void fractions through the system, and axial temperature gradients in fuel rods.

Figure V-1 shows the time steps as a function of real time. The RV's operation and the level drop seem to be the main couses of reduction.

The run statistics are shown in Table V-1. The real time/CPU time ratio was 1/229.

VI. CONCLUSIONS

A good degree of accuracy is attained for the pressure final value although the transient evolution is not very close to the data. Nevertheless, this result is surprisingly good for an ideal separator model, showing that a radial discharge at separators heigth causes water-steam mixing not too far from the actual process.

The accuracy in core flow and water level is not so good. Too fast depresurization will take jet pumps close to cavitation. Longer condensation will mean larger water inventory too. Anyway these parameters may also be affected by noding problems such as steam numerical diffusion into the downcomer or improper flow path representation from inside to outside of the dryer skirt.

In spite of all this TRAC model shows overall acceptable perfomance, which no doubt will be improved when a mechanistic separator is available.

By going through the process of improving the model it was found that the EXTRACT capability is not as powerful as needed. It should include the control mode.

VII. REFERENCES

1.- NUCLENOR, S.A. "Cuadernos de cálculo, NUCLE-01". Rev. 0, Febrero de 1989

2.- ENUSA. "Evaluación de seguridad de la recarga. C. N. Santa María de Garoña. Ciclo 15".

TABLE I I - 1

GAROÑA NPP DESIGN CARACTHERISTICS

REACTOR	
• Thermal Power (100%)	1380 Mw
• Vessel Pressure	70.3 Kg/cm^2
• Core Flow	21.77 x 10^6 Kg/hr
• Steam Flow	2.48 x 10^6 Kg/hr
• Feedwater Temperature	183ºC
CORE DIMENSIONS	
• Diameter	0.3683 m.
• Active Lenght	0.3658 m.
FUEL ELEMENTS	
• Number of Fuel Elements	400
• Rod Fuel Layout	8 x 8R and P8 x 8R
• Cladding	Zircaloy-2
• Fuel	UO_2
• Outter clad diameter	1.25 cm/ 1,23 cm
• Clad Thickness	0.086 cm/ 0,081 cm
• Channel	Zircaloy-4
CONTROL RODS	
• Number of Control Rods	97
• Shape	Cruciform
REACTOR VESSEL	
• Inner Diameter	4.775 m.
• Inner height	18.447 m.
• Design pressure	87.90 Kg/cm^2

TABLE I I - 1 (Cont.)

RECIRCULATION SYSTEM

• Location	Drywell
• Number of loops	2
• Loop diameter	61 cm.
• Nominal flow per pump	2,019 l/seg
• Number of jet pumps	20
• Jet pump Location	inside vessel

PRIMARY CONTAINMENT

• Type	Pressure Suppresion
• Drywell Design Pressure	62 psig.
• Suppresion Pool Design Pressure	62 psig.

RV'S

• Number	3
• Setpoint	74.2, 74.9, 75.6 Kg/cm^2

TABLA III - 1

DESCRIPTION OF THE TRAC COMPONENTS

NUMBER	COMPONENT	DESCRIPTION
1	VESSEL	Vessel
20	TEE	Suction pipe of the recirculation loop
21	VALVE	Isolation valve of the recirculation loop
22	PUMP	Recirculation pump of the loop
23	VALVE	Isolation valve of the recirculation loop
24	TEE	Discharge pipe of the recirculation loop
25	JETPUMP	Jet pump of the recirculation loop
31	VALVE	IC Line
32	PIPE	IC Shell
40	CHAN	Average bundle
50	TEE	Main steam line from the vessel
52	TEE	Main steam line from the vessel
54	VALVE	Main steam isolation valve
58	BREAK	Turbine
60	PIPE	Feedwater pipe
61	FILL	Feedwater
62	FILL	High pressure core spray (HPCS)
70	PIPE	Pipe of the core spray
71	FILL	Low pressure core spray
55	TEE	Main steam line to the turbine

TABLA IV- 1

INITIAL CONDITIONS

PARAMETER	MEASURED VALUE	TRAC-BF1/G1J1
Reactor Vessel		
Total Core Power (100%), (Mw)	1380	1377
Downcomer Water Level, (m)	10.53	10.51
Steam Dome, (MPa)	6.96	6.96
Total Core Mass Flow, (kg/s)	5360	5432
Core Bypass , (kg/s)	—	555
Core Δp (MPa)	0.08	0.10
Recirculation Loop		
Speed Pump, (rad/s)	120	121
Temperature (K)	546	547
Drive flow (m^3/seg)	3.26	3.29
Feedwater System		
Feedwater Mass Flow, (kg/s)	684	685

TABLE IV-2

CHRONOLOGY OF EVENTS

Event	Time (seconds)	
	PLANT	TRAC
Closure Initiated	0.0	0.0
Scram Signal Generated	1.0	0.3
Low Water Level	2	2.8
Main Steam Isolation Valves Closed	3.0 - 4.0	4
High pressure peak	3 - 4	5
RV's Open	~ 4	3 - 5
RV's Closed	7 - 9	7 - 9

TABLA V-1

RUN STATISTICS

Real Time	RT= 60 seconds
CPU Time	CPU = 13764 seconds
Total number of volumes in the model	C= 119
Total number of time steps	DT= 1072

$(CPU \times 10^3)/(C \times DT) = 108$

Dome pressure

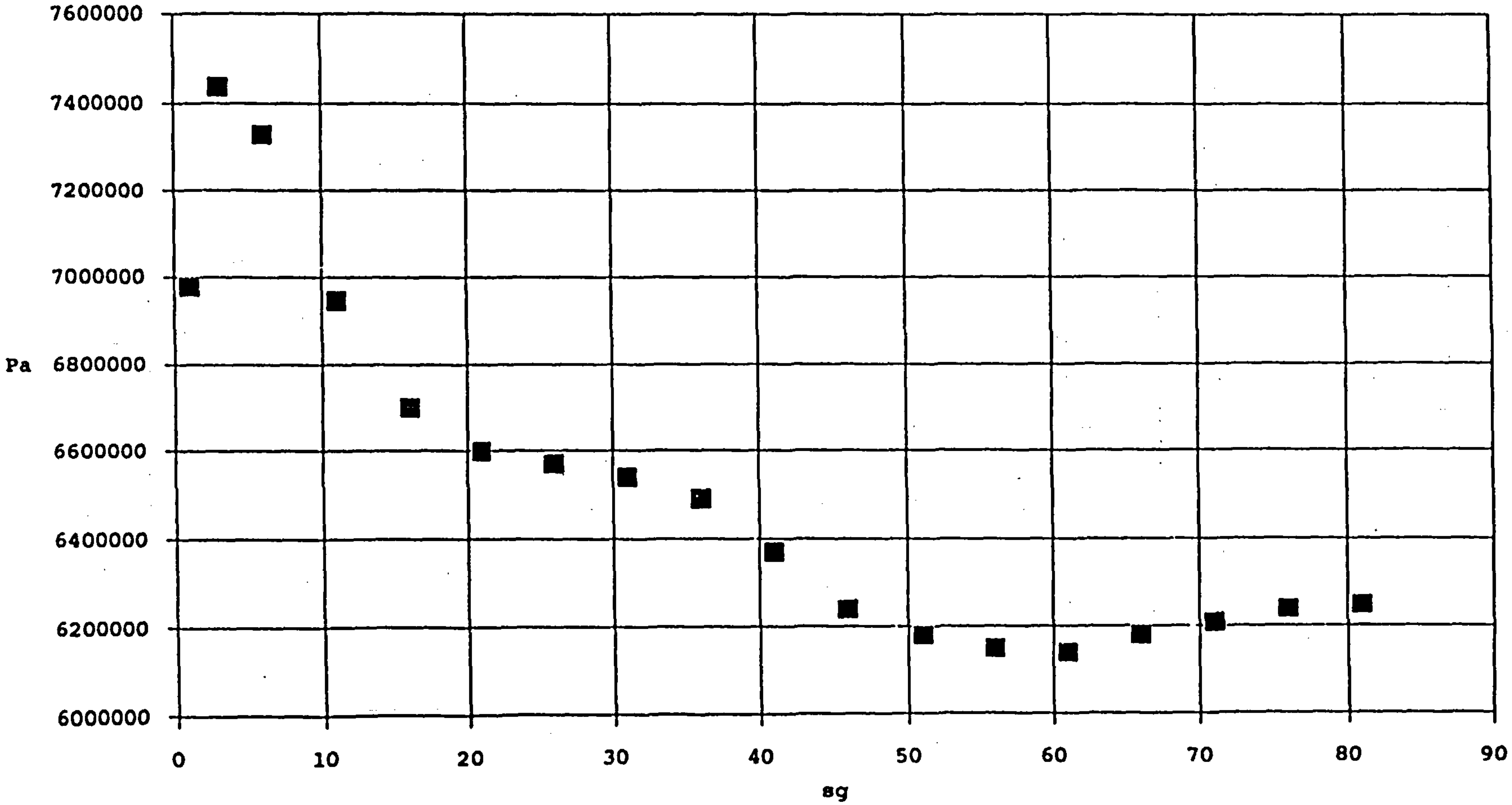


Figure II-1

Downcomer water level

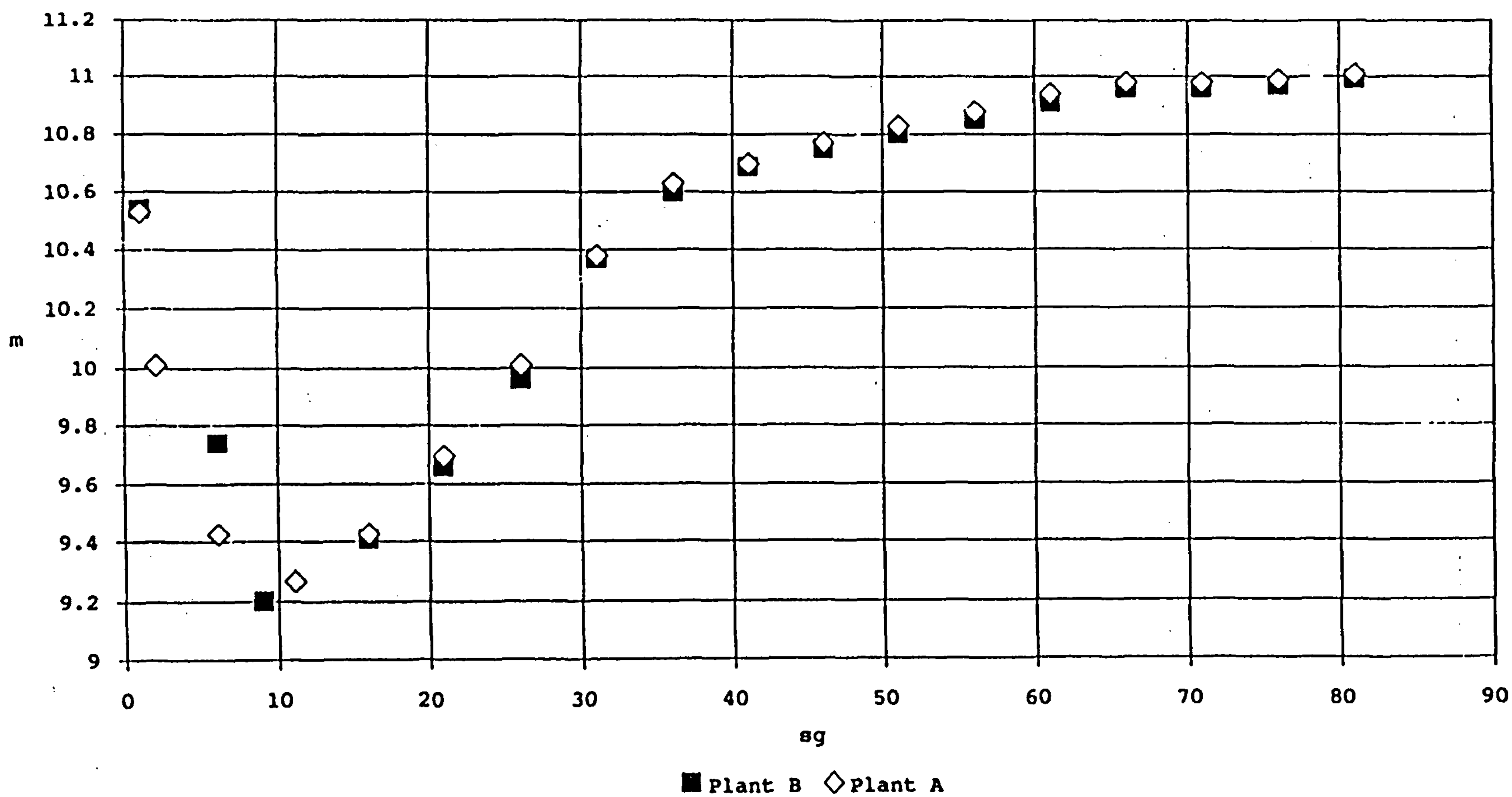


Figure II-2

Core flow

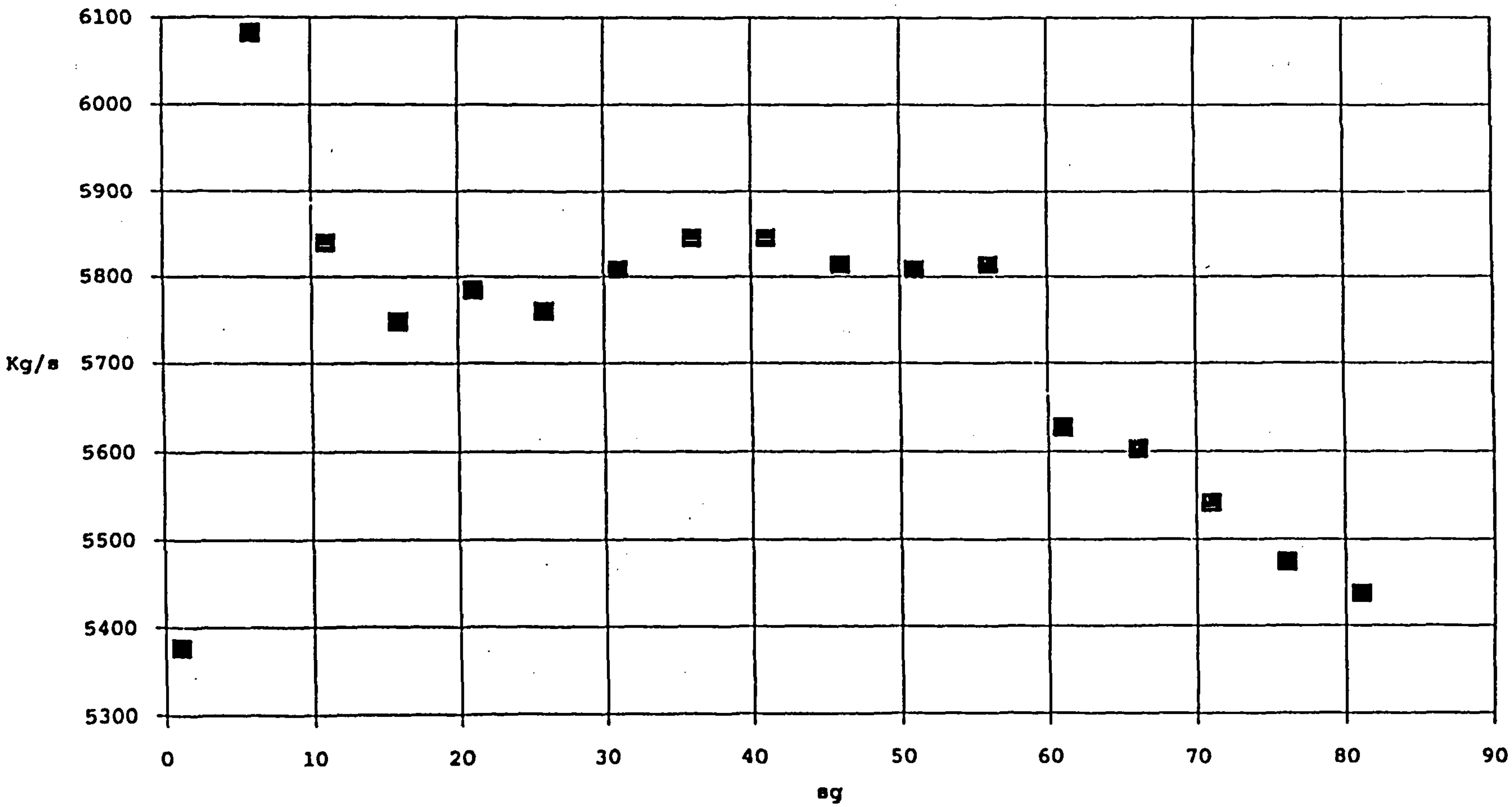


Figure II-3

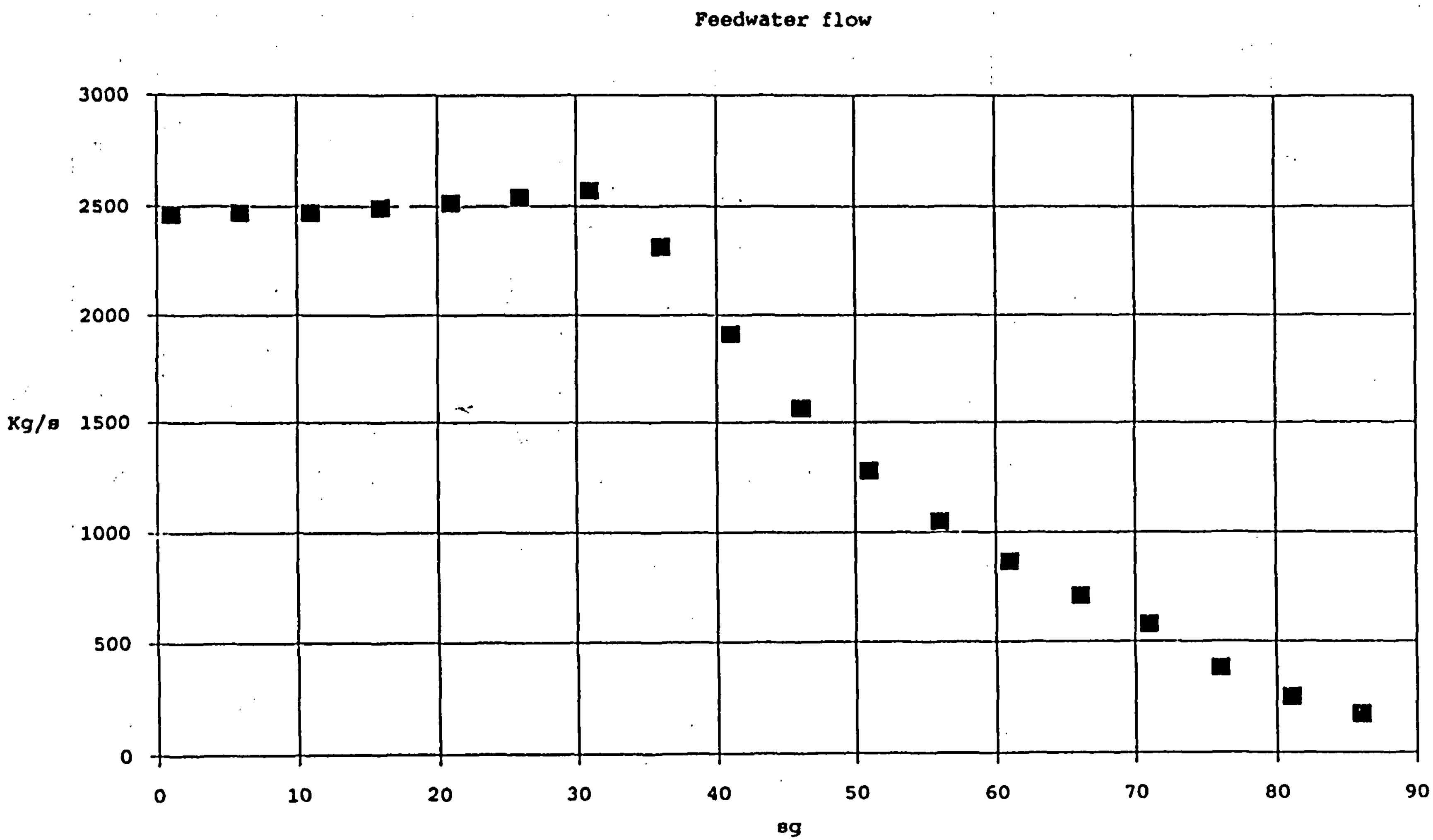

Feedwater flow
3000
2500
2000
1500
1000
500
0
Kg/s
0
10
20
30
40
50
60
70
80
90
sg

Figure II-4

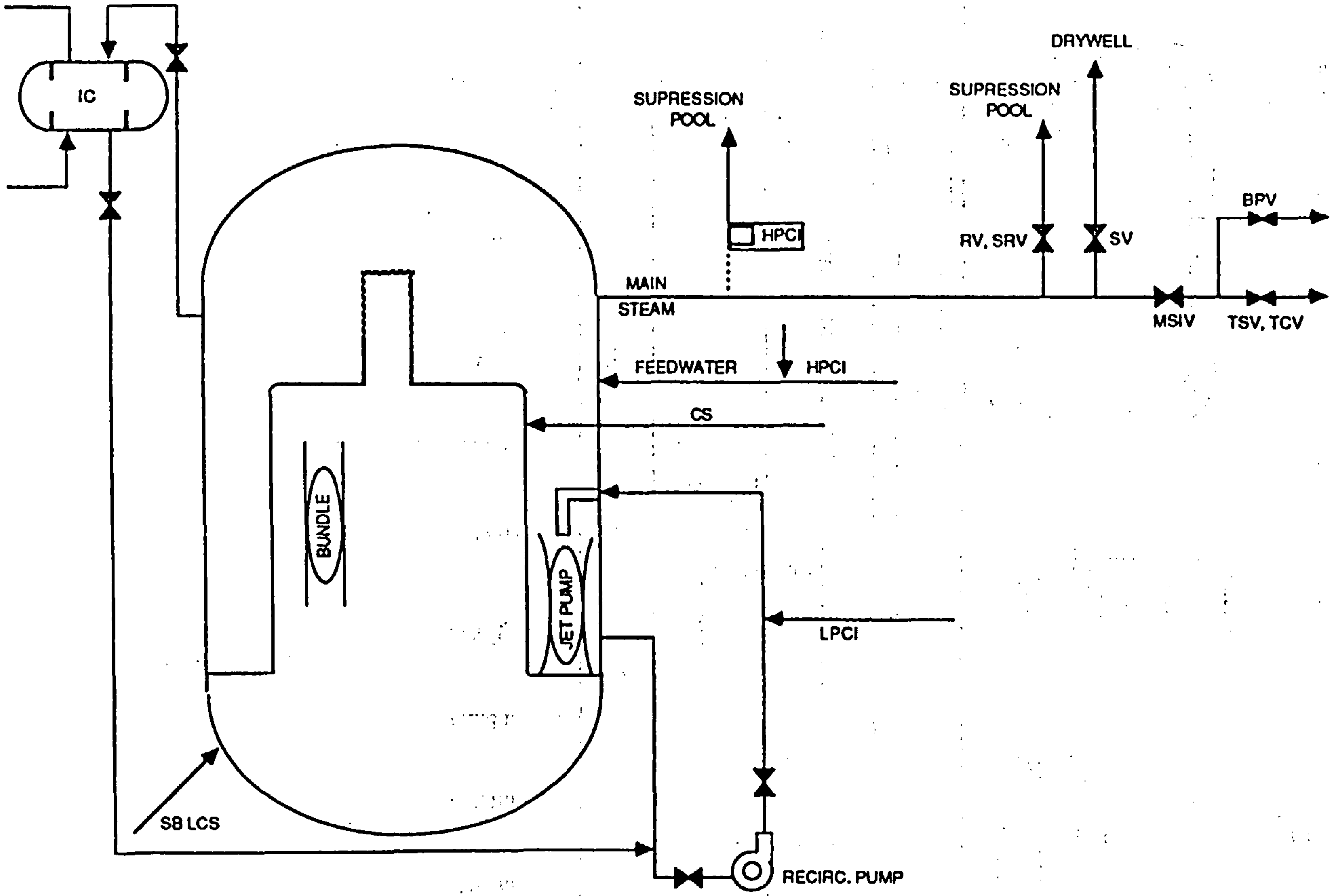

IC
SUPRESSION
POOL
HPCI
SUPRESSION
POOL
DRYWELL
RV, SRV
SV
BPV
MAIN
STEAM
MSIV
TSV, TCV
FEEDWATER
HPCI
CS
BUNDLE
JET PUMP
LPCI
SB LCS
RECIRC. PUMP

Figure III - 1

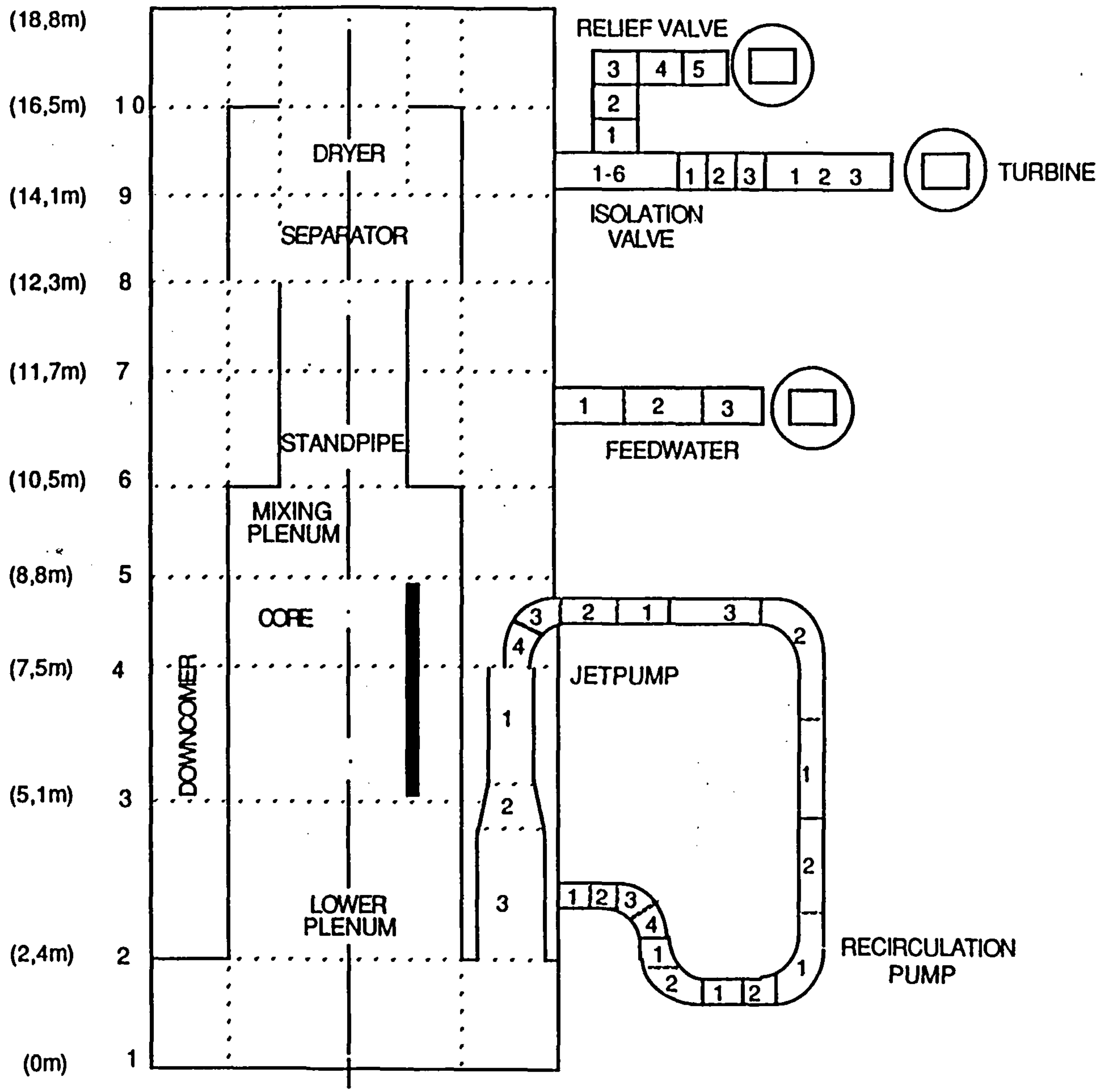


Figure III-2 TRAC-BF1 AXIAL NODALIZATION

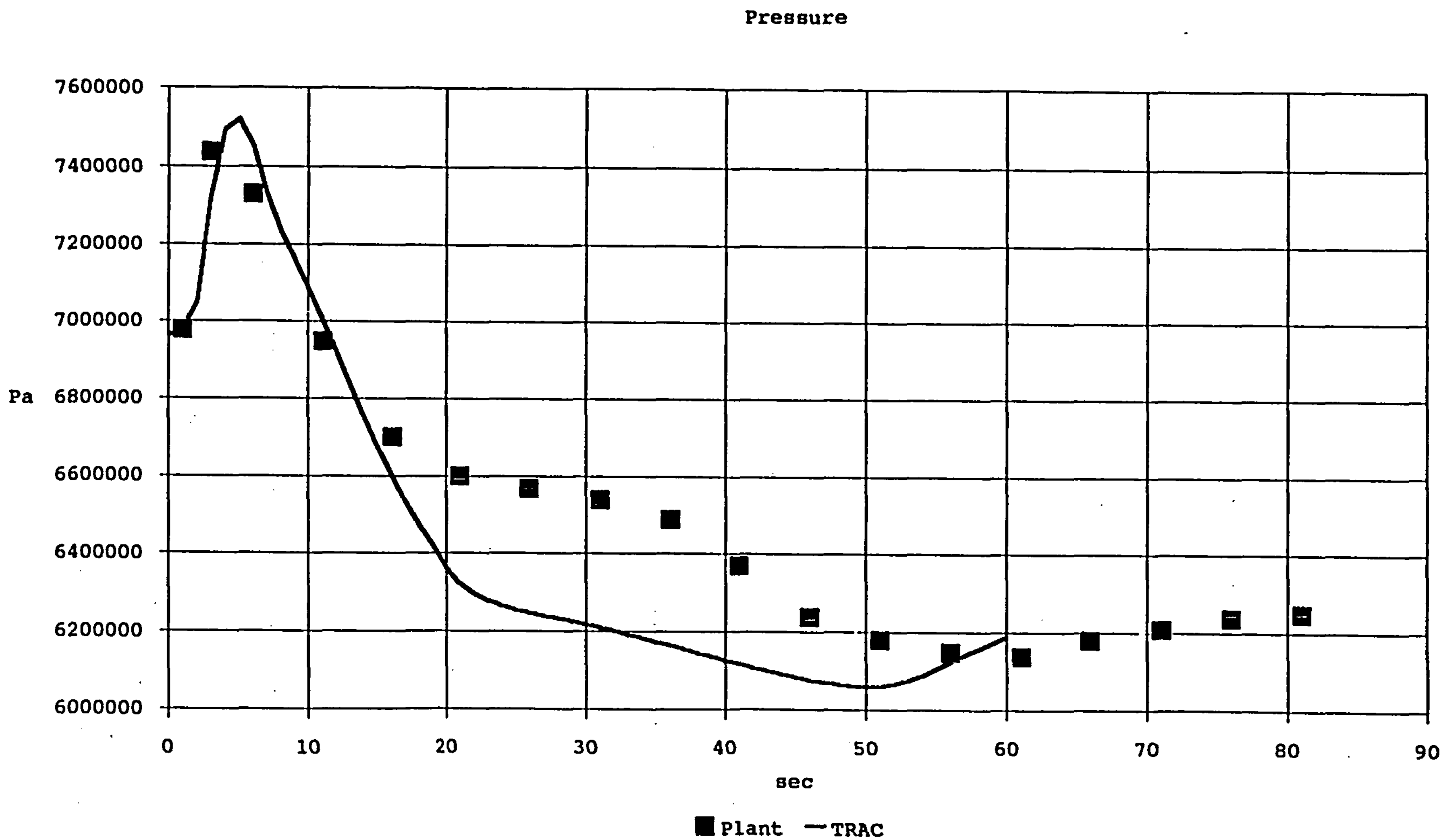

Pressure
7600000
7400000
7200000
7000000
6800000
6600000
6400000
6200000
6000000
Pa
0
10
20
30
40
50
60
70
80
90
sec
Plant
TRAC

Figure IV-1

Mixing plenum void fraction

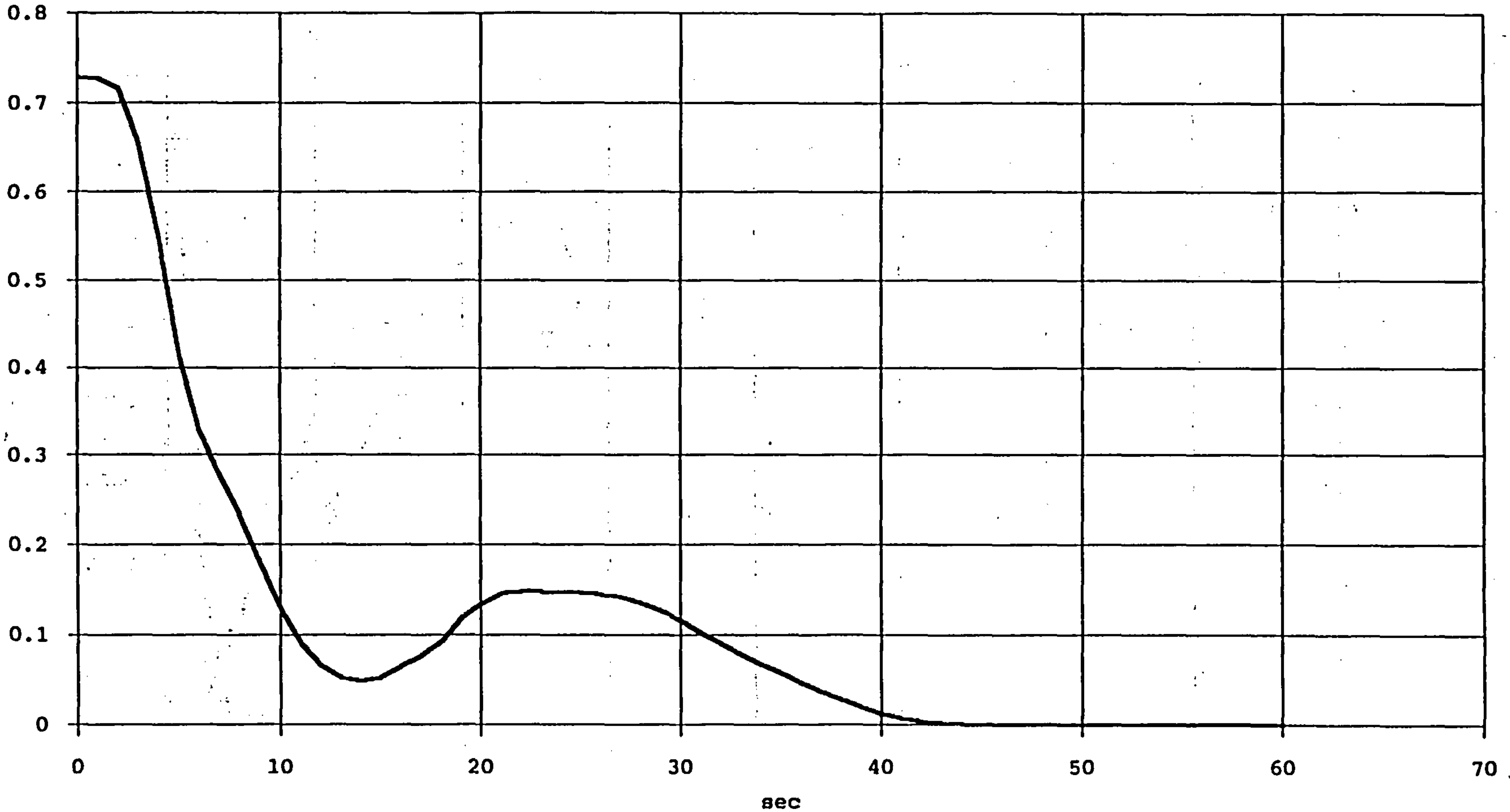


Figure IV-2

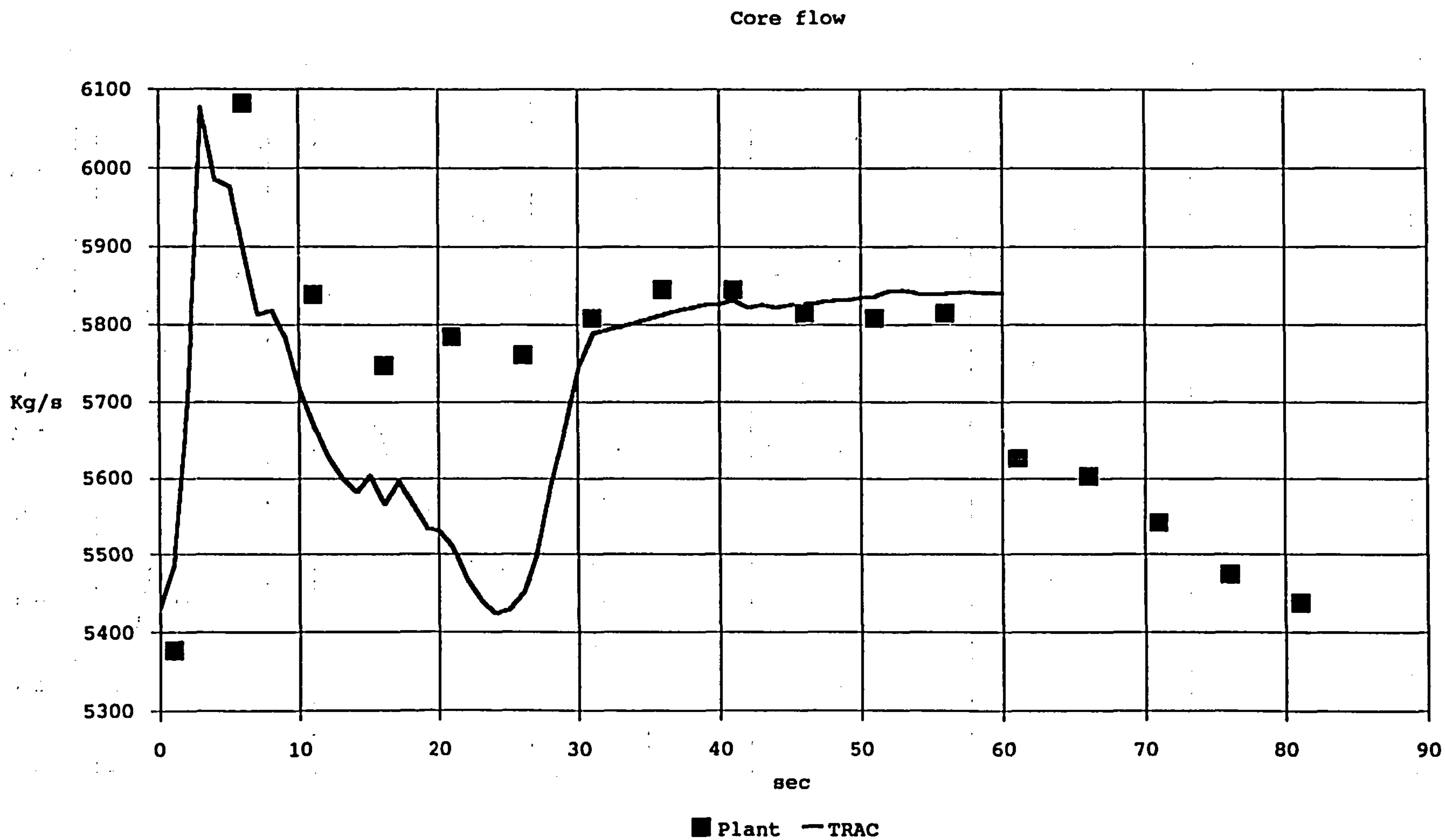

Core flow
6100
6000
5900
5800
5700
5600
5500
5400
5300
Kg/s
0
10
20
30
40
50
60
70
80
90
sec
Plant
TRAC

Figure IV-3

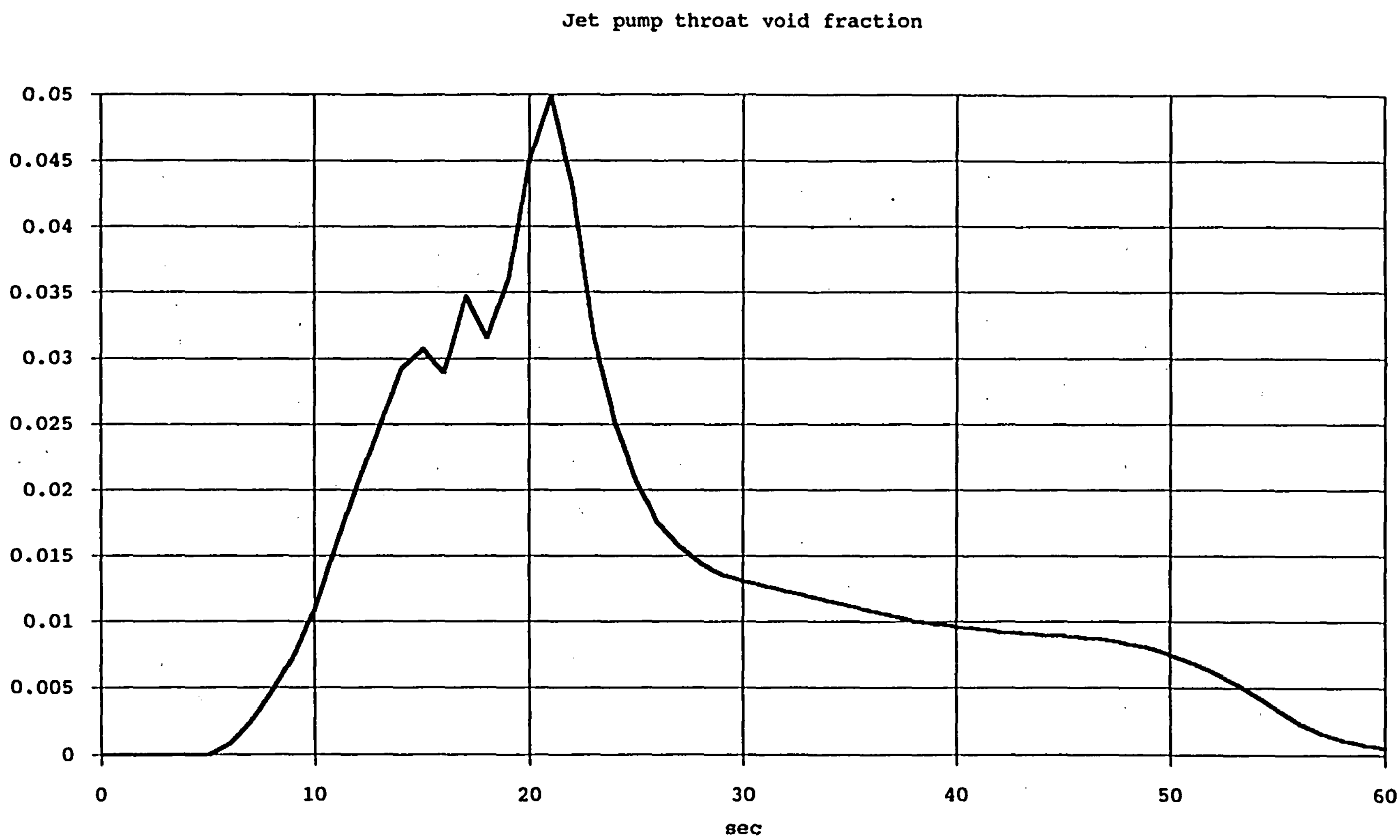
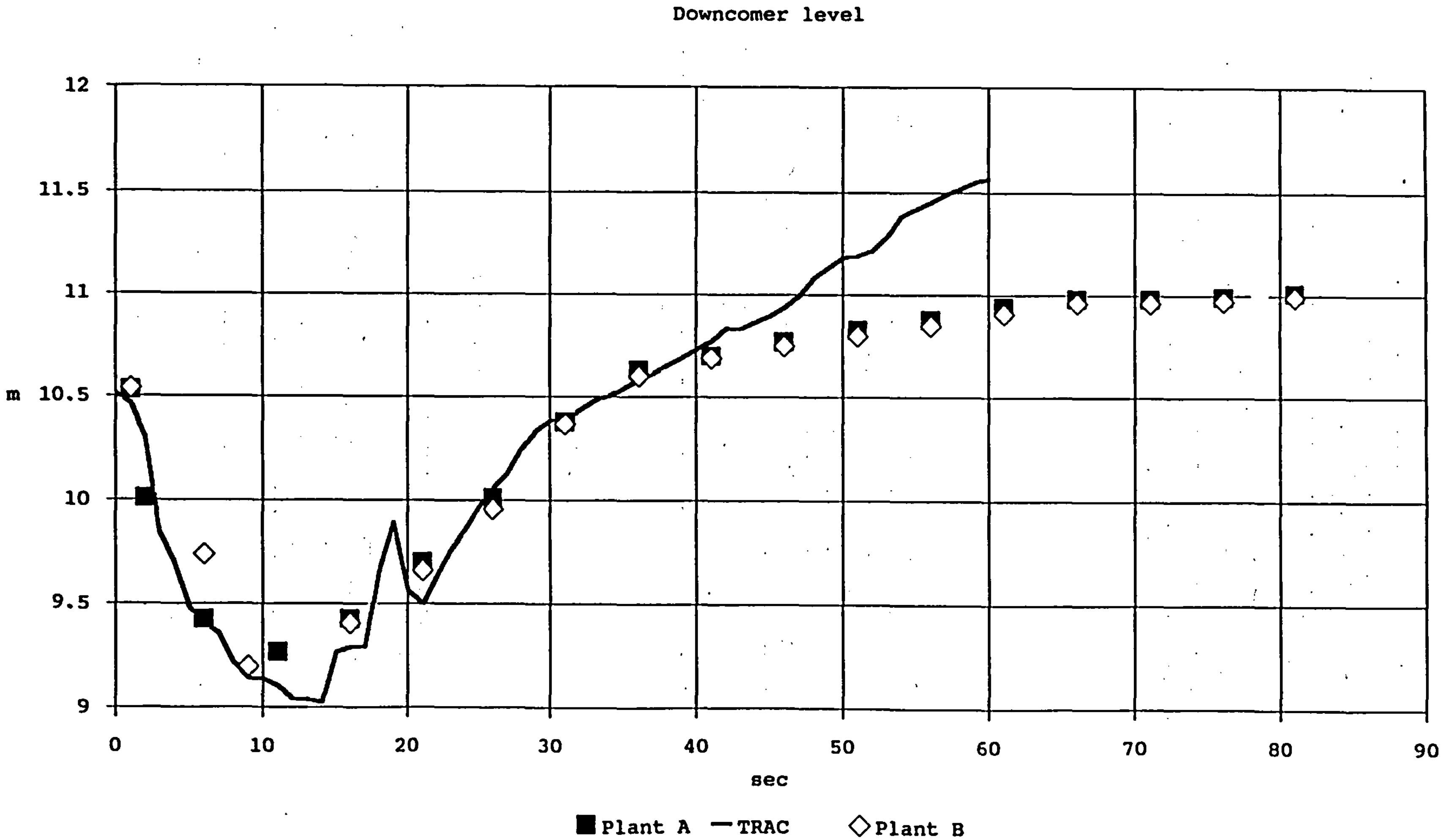

Jet pump throat void fraction
0.05
0.045
0.04
0.035
0.03
0.025
0.02
0.015
0.01
0.005
0
0
10
20
30
40
50
60
sec

Figure IV-4



Downcomer level
12
11.5
11
m 10.5
10
9.5
9
0
10
20
30
40
50
60
70
80
90
sec
Plant A
TRAC
Plant B

Figure IV-5

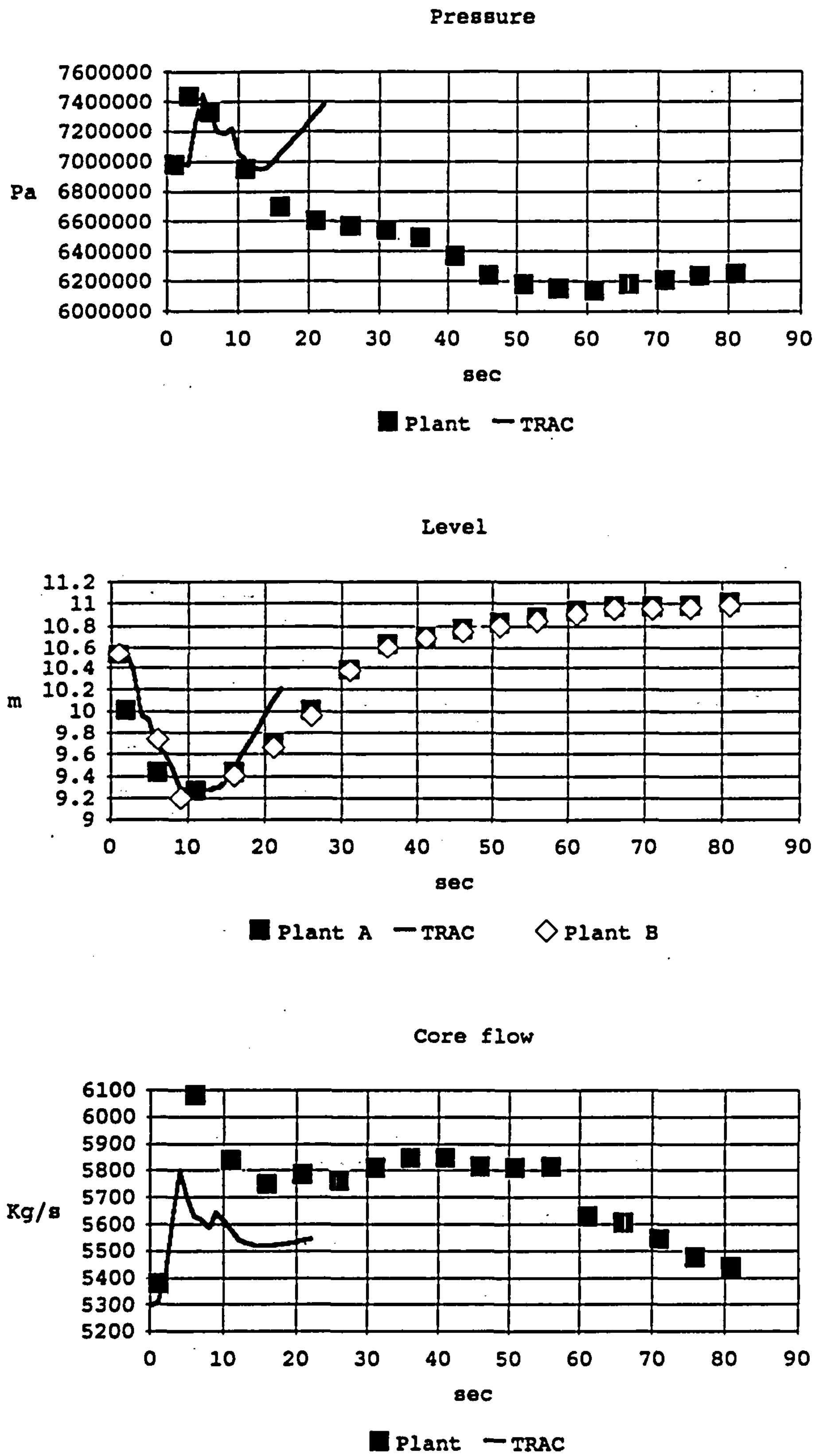

Pressure
Pa
7600000
7400000
7200000
7000000
6800000
6600000
6400000
6200000
6000000
0 10 20 30 40 50 60 70 80 90
sec
Plant
TRAC
Level
m
11.2
11
10.8
10.6
10.4
10.2
10
9.8
9.6
9.4
9.2
9
0 10 20 30 40 50 60 70 80 90
sec
Plant A
TRAC
Plant B
Core flow
Kg/s
6100
6000
5900
5800
5700
5600
5500
5400
5300
5200
0 10 20 30 40 50 60 70 80 90
sec
Plant
TRAC

Figure IV-6

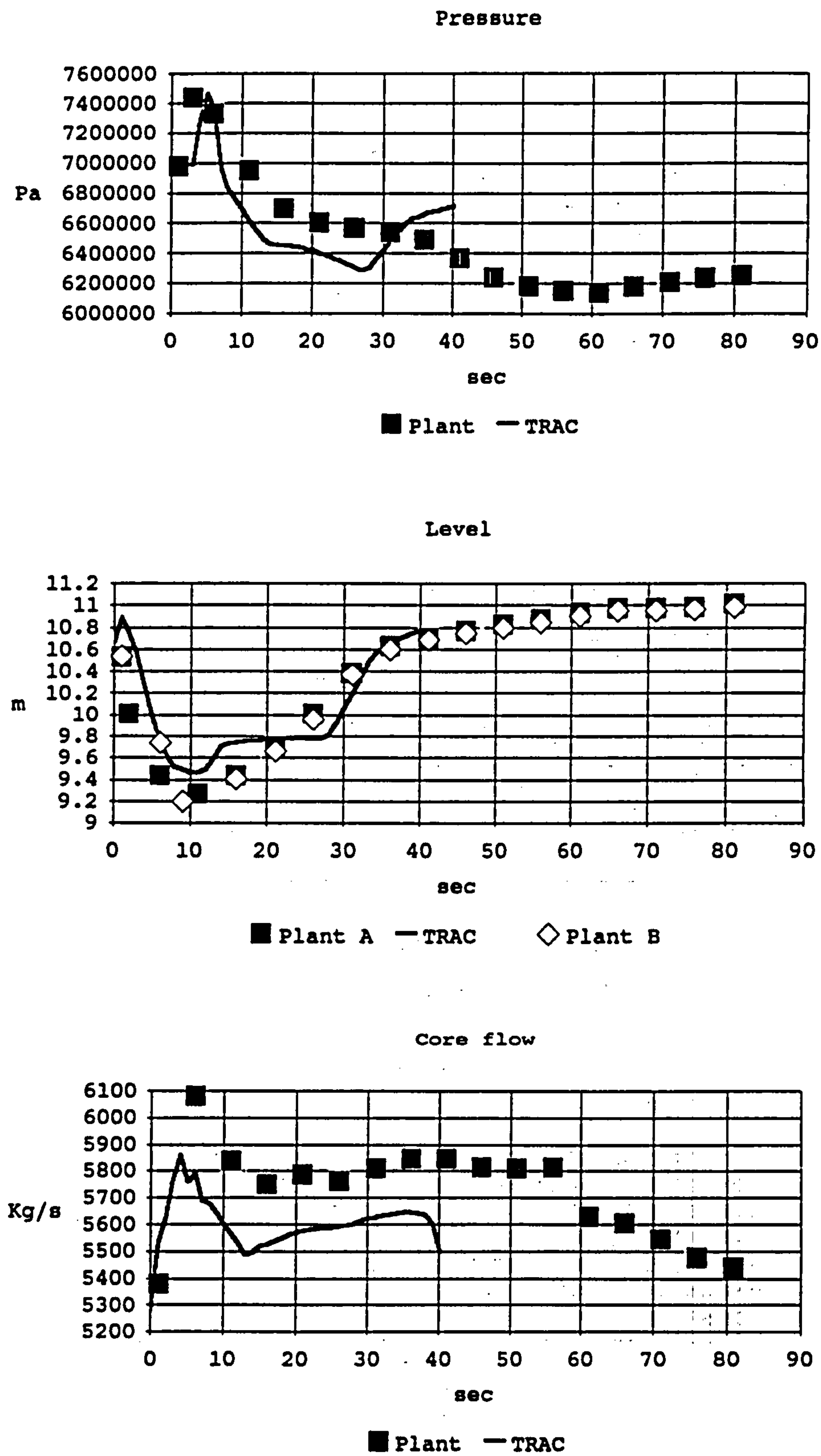

Pressure
Pa
7600000
7400000
7200000
7000000
6800000
6600000
6400000
6200000
6000000
0 10 20 30 40 50 60 70 80 90
sec
Plant
TRAC
Level
m
11.2
11
10.8
10.6
10.4
10.2
10
9.8
9.6
9.4
9.2
9
0 10 20 30 40 50 60 70 80 90
sec
Plant A
TRAC
Plant B
Core flow
Kg/s
6100
6000
5900
5800
5700
5600
5500
5400
5300
5200
0 10 20 30 40 50 60 70 80 90
sec
Plant
TRAC

Figure IV-7

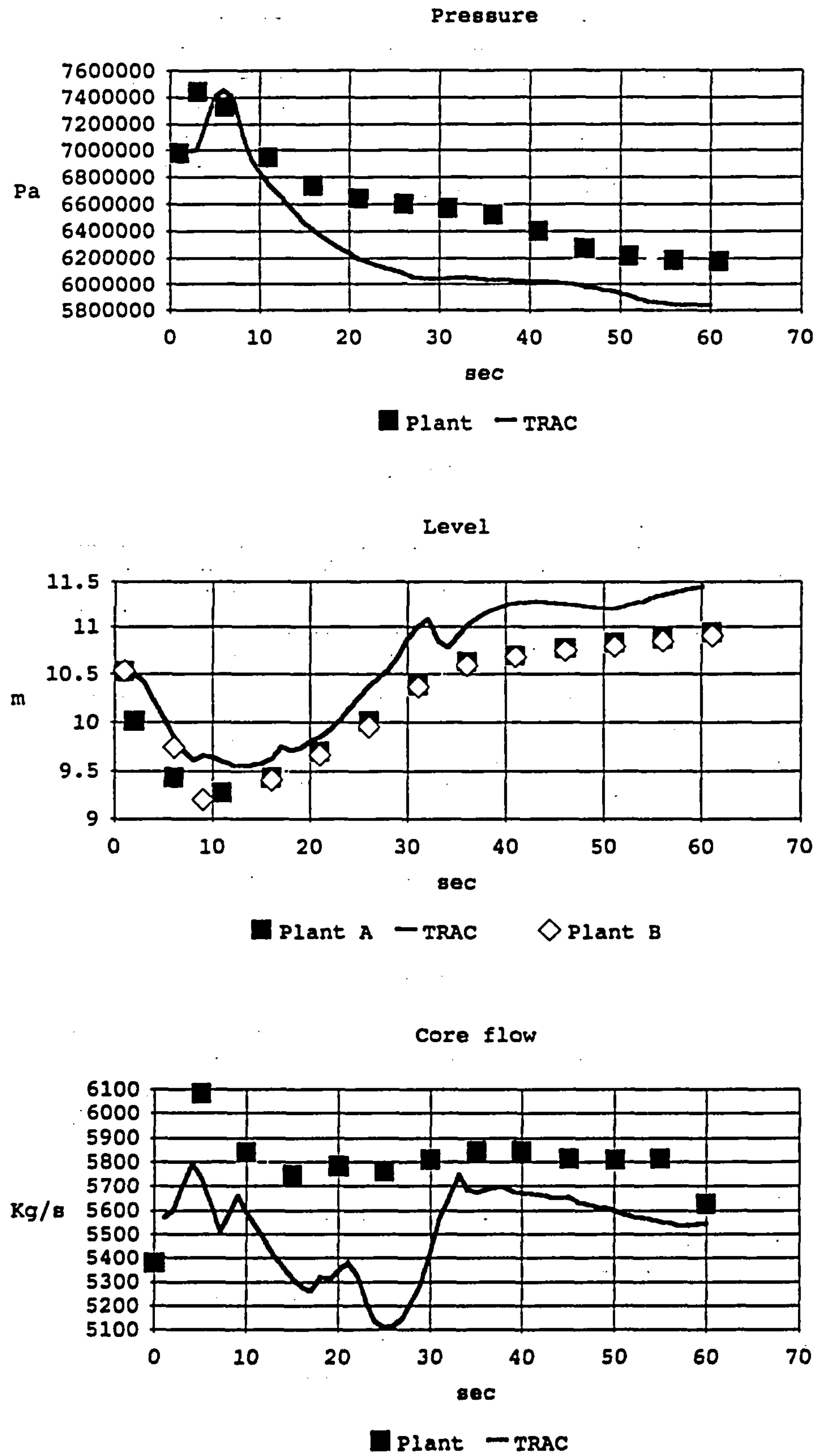

Pressure
7600000
7400000
7200000
7000000
6800000
6600000
6400000
6200000
6000000
5800000
Pa
0
10
20
30
40
50
60
70
sec
Plant
TRAC
Level
11.5
11
10.5
10
9.5
9
m
0
10
20
30
40
50
60
70
sec
Plant A
TRAC
Plant B
Core flow
6100
6000
5900
5800
5700
5600
5500
5400
5300
5200
5100
Kg/s
0
10
20
30
40
50
60
70
sec
Plant
TRAC

Figure IV-8

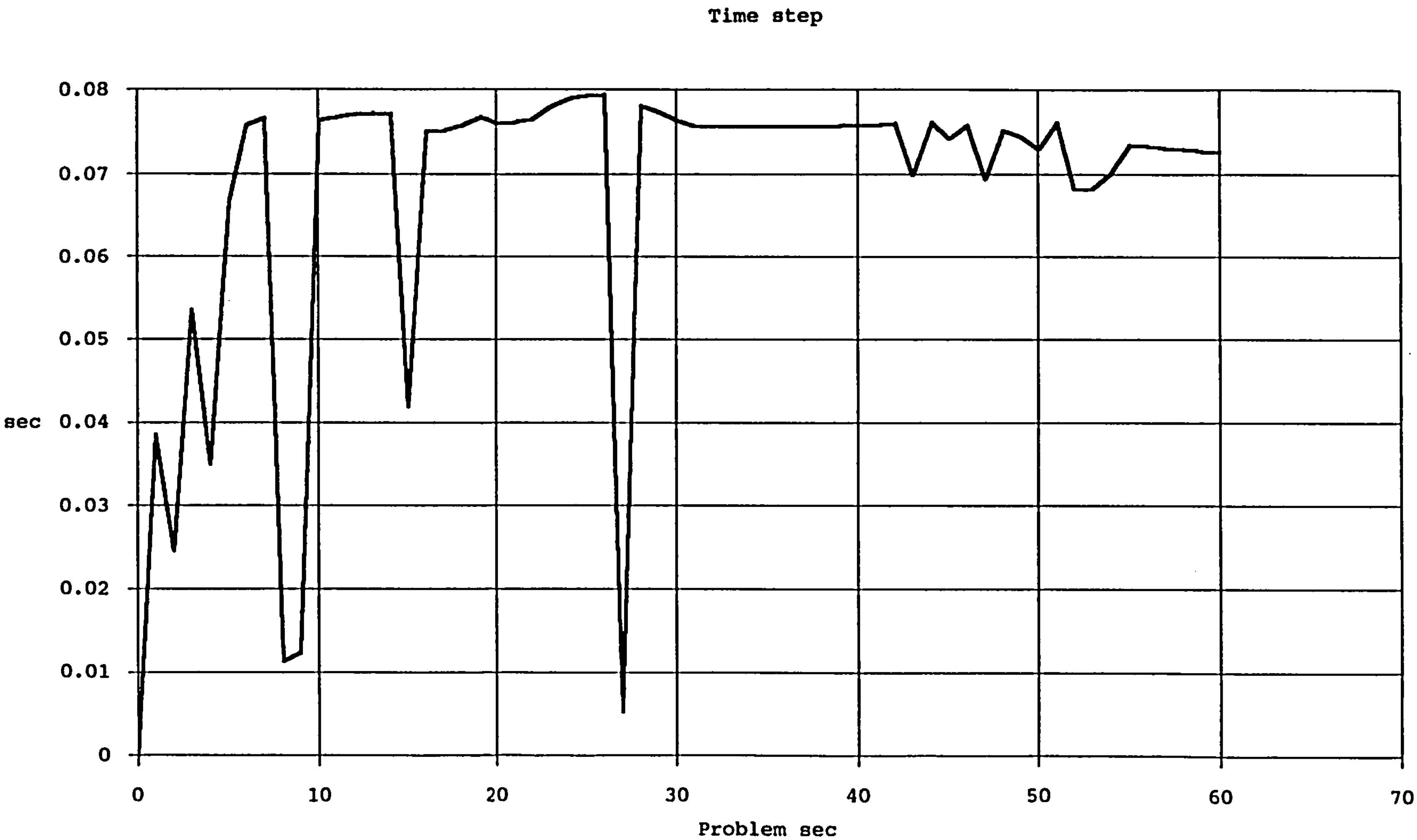

Time step
0.08
0.07
0.06
0.05
sec 0.04
0.03
0.02
0.01
0
0
10
20
30
40
50
60
70
Problem sec

Figure V-1

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U.S. NUCLEAR REGULATORY COMMISSION

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(See instructions on the reverse)

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

An assessment of the first 60 seconds of a spurious Main Steam Insolation Valve (MSIV's) closure for Santa Maria Garoña Nuclear Power Plant using TRAC-BF1 code is presented. Reasonable and realistic adjustments have been made in the model to improve its performance.

This work is part of the validation set for the TRAC model that is being developed for wider use and allow to test the code capabilities.

As a result of the analysis, it is felt that TRAC-BF1 is capable of reproducing the plant behavior with an acceptable degree of accuracy although better models are clearly needed, in addition to further noding work and code improvements. The code took almost 14000 sec. which makes a 1/230 calculation time to real time ratio.

For this transient a mechanistic separator model is needed. It will also help to cut down running costs if the vessel noding could have different number of cells at different heights. Though not very important for this transient, the critical flow model will allow for realistic RV flow assumptions.

There are not guidelines available for separator modelling in transients. It has been found that a detailed noding in the separator region may be needed to represent steam-water interaction.

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

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