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International Agreement Report

Recirculation Suction Large Break LOCA Analysis of the Santa Maria De Garoña Nuclear Power Plant Using TRAC-BF1 (G1J1)

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

August 1992

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F O R E W O R D

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
- 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.

ABSTRACT

This document presents a recirculation suction large break loss of coolant accident analysis of Santa Maria de Garoña Nuclear Power Plant.

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 Chair of Nuclear Technology from the Polytechnical University of Madrid, 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 first of two NUCLENOR contributions to the International Code Applications and Assessment Program (ICAP).

The code was run in a CDC Cyber 830 with operating system NOS 2.6, property of NUCLENOR, S.A. Additionally the code was updated into a CDC Cyber 932 operating system NOS VE, in order to solve the problems related to central memory limitations found at early stages of the work. A programming effort was carried out in order to provide suitable graphics capabilities to the code.

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EXECUTIVE SUMMARY

A best estimate analysis of a Recirculation Suction Pipe Large Break Loss of Coolant Accident Analysis for Santa María de Garoña Nuclear Power Plant using TRAC-BF1 code is presented.

An investigation of the existing safety margins respect to the available Garoña licensing analysis based on GE SAFER-GESTR Methodology was the main objective of this work. Boundary and initital conditions of the analysis were consistent to those ones used in SAFER analysis.

A margin of 160°C in peak clad temperature (PCT) respect to the higher PCT calculated by SAFER code was obtained with TRAC-BF1.

As a result of the analysis, considerable knowlegde on LOCA phenomena and expertise in using TRAC-BF1 code has been obtained. It is believed that the developed Garoña LOCA model is qualified to performe best estimate analysis according to the state of the art of the current BWR LOCA technology.

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 application case the Analysis of a Large Break Loss of Coolant Accident. The analysis, described in the present report, was carried out by the Chair of Nuclear Technology, belonging to the Polytechnical University of Madrid, the Applied Physics Department of University of Cantabria and the Analysis and Operations Section of NUCLENOR.

The application analysis consisted a postulated large double-ended guillotine break in the piping on the suction side of a recirculation loop. It was intented to simulate the limiting LOCA case derived from the Garoña application of General Electric SAFER-GESTR Methodology in order to evaluate the existing safety margins in licensing.

Besides the postulated break, the following main hypothesis were assumed in order to use the same boundary conditions as SAFER calculation:

1. Loss of off-site power coincident with the LOCA initiation. So that recirculation pumps would coastdown and the Emergency Core Coolant Systems (ECCS) would be electrically supplied from Diesel Generators.

2. The worst single failure is also postulated to occur. In this case the failure of Low Pressure Coolant Injection (LPCI) System , remaining available the Core Spray (CS) System, High Pressure Coolant Injection (HPCI) System and Automatic Despresurization System (ADS).

Additionally, a programming effort was carried out in order to provide suitable graphics capabilities to the code.

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 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. 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.

The Garoña Emergency Core Cooling Systems, as shown in Fig. II-1, consist of following systems:

- High Pressure Cooling Injection System (HPCI)
- Automatic Depresurization System (ADS)
- Low Pressure Cooling Injection System (LPCI)
- Core Spary System (CS)

The HPCI System has a turbine-driven pump which injects water into the vessel, coming from Condensate Storage Tank first, and then from supression pool. Either high pressure in Drywell or low low level in the vessel signal produces automatic starting of the system.

The actuation of ADS consist of automatic opening of three relief valves, discharging steam from the vessel into the supression pool, when high drywell pressure and Low Low Level signal exist for two minutes. After the ADS actuation, the low pressure systems, LPCI and CS, are able to inject water into the vessel.

The LPCI System has four pumps and two interconnected hydraulic trains injecting in each recirculation loop. An electric logic exists for selecting the unbroken loop, so that permiting the entre flow the four pumps inject into those one, the water enters to the core form the bottom.

The CS System has two independent hydraulic trains, each one consisting of one pump which takes water from the supression pool and inject it directly on to the core through a nozzle sparger.

Both LPCI and CS starts automatically when either a drywell high pressure, or Low Low vessel level signal exists, and when the vessel pressure falls below 21kg/cm^2.

II.2 TRANSIENT DESCRIPTION

A full double-ended guillotine break in the piping on suction of a recirculation pump was postulated to occur at the time zero.

A loss-of-offsite power was assumed to occur simultaneously with the break. This caused a trip of the recirculation pumps, condensate and feedwater pumps and as a result isolation of main steam lines.

The safety system single failure criteria was applied to the admission valve in the LPCI line at the intact recirculation loop. The LPCI admission valve was assumed to fail closed which prevented any LPCI from reaching the reactor vessel. This single failure thus effectively failed the entire LPCI. Therefore the available emergency systems were ADS, HPCI and CS.

Provided that the main objective was to know the existing safety margin respect to the licensing calculations by SAFER-GESTR GE Methodology, the boundary conditions were intended to match as much as possible to those ones used in the mentioned method applied to Garoña.

According to that, the core was assumed to be operating at 104% rated thermal power at the start of the transient. As well as maximun average planer linear heat generation rate (MAPLHGR) of 12,5 kw/ft was assumed. Gap conductance and axial peaking factors for the high power bundle were consistent to those ones used in SAFER analysis.

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. All those different types of TRAC-BF1 components were used in the LOCA model of Garoña. 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 the analysis of the large break loss of coolant accident at the Santa María de Garoña Nuclear Power Plant was based on data taken from drawings (Ref. 2), and specific technical documents related with the nuclear fuel design (Ref. 3).

The model contains 33 components with 43 fluid junctions. Table III-1 includes a listing of all the TRAC components used in the model. Figure III-1 shows the identification number and the relative location of each component within the TRAC-BF1 model. The illustrated model represents Garoña for normal operation only. No broken or failed components are modeled in the Figure.

The reactor vessel has been modeled by the VESSEL component with the nodalization shown in Figures III-1 and III-2. This nodalization is made up of eight axial levels, four radial segments and one azimuthal sector.

The radial segments are distributed such that the inner three rings extend over the core region with the first ring containing 84 real channels, the second ring containing 240 real channels and, the third ring containing 76 real channels. The fourth radial ring models the reactor vessel downcomer.

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 four axial levels 5 through 8, represent the steam dome, upper plenum, with the CSs connections being made at axial level 5, the feedwater system at axial level 6 and the main steam line at the axial level 7. The downcomer extends from axial level 2 to axial level 7.

Instead of individual steam separator and dryer components, a perfect separator option is used in axial levels 6 and 7. This component allows the vapor for continuing upward into the axial level 8 and liquid draining radially outward into the downcommer region. Double-sided slab models account for the heat capacity and transmission within vessel internals.

Contained in the lower levels of the VESSEL, axial level 1 and axial level 2, the PIPE components number 2, 3 and 4 model the control rod guide tubes. These three pipes represent 97 real control guide tubes. The guide tubes were modeled from the top of the control rod drive housing to the core plate. The volume of the guide tubes and the vessel cells were computed assuming that the control rods were fully inserted into the core. The heat transfer between the fluid in the guide tubes and the lower plenum has been modeled.

Fuel bundle modeling is accomplished using four CHAN components simulating the 76 peripheral bundles by CHAN component number 36, 240 average bundles by CHAN component number 38, and 84 hot bundles by CHAN components numbers 40 and 43.

As shown in Figure III-3, the actual 8 x 8 rod array channel with two water rod is modeled by a geometrically identical 2 x 2 rod array CHAN component, with one water rod. All four CHAN components have this structure. The axial distribution of the CHAN components consist of 15 axial levels, 10 of them in the active core region, with a bypass flow path modeled by an explicit leak path. The total leak flow from all the CHAN components represent a 10% of the total flow through the core.

Axial and radial power profiles agree with data from General Electric Propietary Information (Ref. 5). 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) (Ref. 4) built into the code. Additionally, a decay and fission product neutron capture was modeled. 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 3.

The use of begining of life radial fuel rod dimensions yields the maximum size for the gap between the fuel and the cladding, since creep down, fuel craking, or fuel swelling is not considered. As a result of using the maximum gap size, the fuel stored energy is the maximun value for a given value of reactor power provided that the cladding has not previously been ballooned.

Each recirculation loop have been modeled by six TRAC-BF1 components. The intact loop (broken loop) is modeled by the PIPE component number 20 (number 10), representing the suction piping end connected to the vessel, the VALVE component number 21 (number 11), representing the isolation suction valve, the PUMP component number 22 (number 32) and the VALVE component number 23 (number 31), simulating the recirculation pump and the isolation discharge valve. Two PIPE components, numbers 24 and (numbers 41 and) take into account the five actual risers which supplie water into the jetpumps. Finally, the JETPUMP component number 25 (number 26) simulates the actual 10 jetpumps. The JETPUMP discharge is connected to the level 1, ring 4 of the vessel.

The suction pipes at the recirculation loop was represented by four nodes, which is a fine nodalization in order to follow better phenomena of critical flow. The PUMP component represents only the fluid volume between the inlet and outlet nozzles of the pump. The internal friction is accounted for 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 4 via one pipe (PIPE number 60). Also connected to this pipe is the HPCI (FILL number 62) by one leak path system.

The CS is simulated by two FILLs (FILL numbers 73 and 71) connected to two pipes (PIPE number 72 and 70). The simulation of two CS is due that the real plant has two pumps with small differences in their behavoir. The main steam line is modeled by two pipes (PIPE number 50 and 54) and the main steam isolation valve (VALVE number 52) cuts steam flow out of the vessel. The BREAK component number 58 provides a back pressure representative of the turbine.

The reactor point kinectics option was turned on 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 3 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 main option selected in the components was the critical flow model that was applied at the jet pump nozzle and the BREAK juntions in the broken loop. The counter current flow limiting (CCFL) model was applied at the side entry orifice of the bundle, at the upper tie plate and at the core bypass.

Several control blocks were added to the input in order to adjust the recirculation pump speed during the steady state calculation to yield a desired core flow and, as well as all the necessary control blocks to control the downcomer level.

The initial conditions predicted by TRAC-BF1/G1J1 for the analysis of the LOCA are compared with tipical measured data at plant in Table IV-1. Calculations produced stable initial conditions.

In order to run the transient, a modification consisting in two BREAK components in junction number 210, was introduced in the model. As well as 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.- A double-sided slab in the VESSEL component was available to simulate the cylindrical wall and at each guide tube cell the heat transfer between the fluid in the guide tubes and lower plenum is modeled.

d.- The backpressure at the double--ended break was assumed constant.

e.- Loss of off site power coincident with the LOCA initiation was postulated, so that recirculation pumps would coastdown and pumped ECC injection would be delayed a time equivalent to that needed for a starting diesel generators.

f.- Only the high pressure coolant injection system (HPCI) and the low pressure coolant spray (CS) are allowed.

g.- Peaking factors were consistent to those ones used in SAFER Analysis.

h.- Feedwater flow was assumed zero since the LOCA initiation.

IV. CALCULATION RESULTS

The calculation of the LOCA with TRAC-BF1/G1J1 has been developed in two steps. First of all, a steady state calculation was carried out at 104% of nominal thermal power. The second step consisted of the transient calculation.

The steady state calculation was also developed in two steps. A first run of 300 s real time with the built-in control system governing the level in the vessel, the recirculation pumps speed and the mass flow rate in the core. Secondly a null transient calculation including point kinetics was performed. After 100 seg., the simulation got a stable behaviour. Table IV-1 shows the values obtained at stable conditions and typical values measured at plant

The convergence criteria for the outer iteration (EPSO), the vessel iteration (EPSI), and the steady state calculation (EPSS) were 1.0E-04, 1.0E-04, 1.0E-04, respectively. The maximum number of the outer iterations (OITMAX) and VESSEL iterations (IITMAX) were 10 and 100 respectively.

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-18.

Calculated steam dome (level 8 in the model) pressure vs. time is shown in Figure IV-1. The pressure began to decrease following the break due to the fluid mass discharge through the two break sides. The main steam isolation valves closure produced a slighty recovery of pressure at 4.5 seconds. Once the valves had closed the pressure rate was being controlled by the core steam generation and the mass flow rate through the break sides. The uncovery of the recirculation suction line takes place at about 8.0 sec. As a result, pressure rate increase first, provided that steam is being discharged through both break sides, and then decrease due to the lower plenum flashing ocurring from instant 9.0 sec. until 11.0 sec. Thereafter the pressure continues decreasing until reaching the postulated constant drywell pressure.

The calculated reactor power vs time is shown in Figure IV-2. and IV-3 Figure IV-2 shows the reactor power during the first seconds. The decreasing core flow caused an increasing void fraction and consequently a decreasing reactor power. The scram was generated at 0.5 s due to the low water level signal. The control rod insertion began at the same time and the rods were fully inserted 2.9 s after, at that time the core power was all decay heat. According to that the power goes down slowly for the rest of the calculation (Fig. IV-3).

The calculated downcomer liquid level is shown in Figure IV-4. The level corresponds to a collapsed liquid level based on downcomer cells void fractions. The liquid level decreased quickly due to the loss of fluid through the break until the instant 8.0 of the transient. At that time level of liquid was 0.73 m above the jet pump supporting ring. The small increase in the level from the instant 9.0 to 12.5 s was caused mainly by the flashing in the lower plenum which forced liquid into the downcomer. There were two flow paths of liquid to the downcomer, the first one through the separator-dryer and the second one through the jet pump of the intact loop. A 1.23 m. level peak occurred at 11.05 sec., then the level went down and remained just above the downcomer bottom.

The calculated mass flow rates in the broken loop are shown in the Figure IV-5. The vessel-side break mass flow rate exceed the pump-side break mass flow from the onset of the transient. The difference between these mass flows are due to the different frictions in both paths coming out of the vessel. On the other hand, the flow rate out the break pump side was limited by the small area of the jet pump nozzles and hence was significatly smaller than the flow through the vessel-side break. At about 10.5 s of the transient, the mass flow rate out is increased as a result of the flashing of liquid in lower plenum. Due to the uncovery of the recirculation suction, once liquid mass was exhausted (at 13.5 sec) the mass flow rate drops again.

Critical flow was expected at both break sides and the jetpumps nozzles. As we know, if the pressure gradient applied to a fluid is continually increased, the corresponding increase in the mass flow will suddenly stop when the sonic limit of the fluid is reached. To model this phenomena, TRAC code has a homogeneus equilibrium model (HEM) and the Algamgir-Jone Lenhard correlation, and assumes that unhomogeneus or nonequilibrium process are not significant, with the exception of the Algamgir-Jones-Lienhard correlation which considers turbulence and nucleation in the break. These correlations are not adecuated when the stagnation state is in the region of subcooled liquid or very low quality.

The mass flow rate at the inlet and outlet of the low-powered channel (CHAN36), averaged channel (CHAN38) and high-powered channel (CHAN40) are shown in Figures IV-6 through IV-8, respectively. The channel inlet corresponds to the side-entry orifice, while the outlet corresponds to the upper tie plate. On the begining the mass flow rate inlet in all the channels decrease due to the trip of recirculation pump at 0 s and, by that reason the evolution of the mass flow rate follows the coastdown of the recirculation pumps. There was a small increase in the mass flow at 3.0 s, that was due to the close of the main isolation steam. There was a second surge of flow into the channels caused by the flashing of the liquid into the lower plenum which started at 8.0 s. The liquid at that time reached saturation and begun to flash into steam.

The Figure IV-9 shows the calculated pumps speed. In the event of a LOCA, the broken loop recirculation pump will be reversely driven by two phase fluid flowing back at high speed to the break pump-side. The coolant in the pressure vessel flashes due to the despressurization and passes through the jet pump nozzle into the discarge line forcing the pump to work as a turbine during the large part of the transient.

Generally the pump would rarely work under two-phase flow conditions. As a result, there are very few data on pump behaviour under these conditions. Built-in code data from Semiscales experiments has been used in the analysis. The TRAC calculations for the other recirculation pump, the unbroken loop, show a high reversal velocity in the first instants of the transient probably due to the considered low pump inertia (Fig. IV-9).

The calculated high pressure coolant injection (HPCI) and low pressure core sprays (CSs) flow rates are shown in Figure IV-10. The HPCI system was activated by a low-low downcomer level signal which ocurred at 1.3 s, from this time a delay of 27 s to reach full flow was taken into account. HPCI was on until 52.08 s at that point the pressure into the vessel had reached the set point of low pressure HPCI trip, so that the HPCI was turned off. HPCI flow was 186.3 kg/s and temperature was 338.55 K.

The low pressure core spray (LPCS) system was actived 31 sec. after the initiation signal took place (at 8,34 m downcomer level) due to the delay of emergency diesel. The evolution of the mass flow rates of the LPCSs followed the derease of the pressure into the system. LPCS flow was 200 kg/s per loop and the spray temperature was 340 K. It was found that the injection water created a two-phase pool across the top of the entire core, while draining into the fuel bundles.

The void fraction in the two levels of lower plenum is shown in Figure IV-11. The void fraction increases rapidly at about 8 s, when the lower plenum began flashing. The void fraction continued to increase until maximum value that was calculated in the level 1 at the second 35 and in the level 2 at the second 50. The lower plenum reflooding began at 54.6 s of the transient, in the level 1. It was caused primarily by the HPCI water until instant 52.0 s of the transient. All this water went into the lower plenum through the downcommer. The contribution from low pressure core spray began at 33.0 s adding subcooled liquid to the upper plenum region above the core. One part of this liquid drains into the lower plenum through the core bypass region and from the fuel bundles through the side entry orifices.

Correct evaluation of dryout or critical heat flux (CHF) is of importance when predicting nuclear reactor core behaviour in a loss of coolant accident. The location of dryout within a bundle determines to a large extent whether rod integrity will be maintained. The heat transfer coefficients and the power generated at hottest channel determine the peak cladding temperature. Figures IV-12 and IV-13 show the heat transfer coefficients for the hottest channel (CHAN40 or CHAN43) for liquid and vapor. During the first seconds of the transient the calculated heat transfer coefficient to liquid was relatively large because the rods were generally calculated to be in nucleate boiling. Large nucleate boiling HTCs are predicted for the whole core until the core inlet flow reverse and fluid quality in the bundles increase. The main heat transfer mechanism was nucleate boiling and transition boiling along the rods during the first portion of the transient.

Heat transfer liquid coefficient dropped to aproximately 282 kw/(m)2K between the instants 31.0 and about at 70.0 s due to the fact the upper half of the active fuel regions was voided and therefore, the critical heat flux was reached. The main heat transfer mechanism during that period was convection to vapor, so that the vapor heat transfer coefficient increased.

The time evolution of the rod temperature before and after spray initiation are represented in Figure IV-14. That Figure shows three curves of temperatures of the hot rod in the hot channel. The points considered in the rod were the nodes at the top, the middle and the bottom. In general, the curves may be divided into four stages. In the first stage, the temperature goes down to saturation during first instants due to the reversal flow through the channel and the flashing of the lower plenum, mainly affecting the lower nodes of the rods. This phase occurred during the 25 first instants of the transient.

During the second stage, before the begining of spraying, the wall temperature rises regulary with the time. At that time of the transient rods are practically dry and all the decay heat is used in raising the temperature of the rods and the fluid (steam and droplets). The void fraction in the hot channel is shown in the Figure IV-15. During this stage the channel has an averaged void fraction higher than 0.9, except the first nodes that only have a void fraction of 0.5. Therefore, the whole channel was dry during that period. This phase occured from instant 25.0 to 33.0 of the transient.

The third stage starts with spraying initiation. When water is sprayed onto the central channel, massive heat removal does not occur inmediately over the entire surface of central levels. Upper levels are being wetted by the water film and droplets, forcing the temperature to lower, while the central and lower levels are still dry and not yet effectively cooled. The lowest level remains near saturation along the transient. In the central levels, cooling is induced by steam and falling droplets of water. Consequently, an extremely small heat transfer (Fig. IV-12 and IV-13) exists along this phase. The code calculated the maximun peak of temperature (PCT) 694.5 K at the instant 77.1 s of the transient. From instants 68 to 71 of the transient a first cooling of the central nodes occured. Therefore the cooling is becoming more efective. The Figure IV-15 shows a small decrease of the void fraction at upper levels first, and then at central levels. That was a consecuence of the phenomena of counter current flow which is model by the code. This phenomena is locate between the upper tie plate and the upper plenum. Figure IV-16 shows several oscillations of the void fraction that represent the drainage of the channel. This phenomena became more important in the next stage.

Last stage is characterized by a rapid heat removal and rod temperature drop to saturation, because the rods surface are covered by a water film. The evolution of the quench front is a primary function of the counter current flow in each channel. TRAC-BF1/G1J1 models this phenomena by the correlation of Kutateladze which is based on experimental data from General Electric . The CCFL breakdown occurs at about 63 s, hence upper plenum void fraction had a strong oscillation and the top channel void fraction decreased (Fig. IV-15 and IV-16). After that, the fluid temperature increase again due to CCFL model. The effect of condensation in the upper plenum is not included in this model, but it is known that the interfacial heat transfer in the condensation regime affects strongly to the CCFL break down, after subcooled spray initiation, (Ref. 6)

As a conclusion maximum PCTs of 694.5 K, 598.3 K and 543.2 K were reached at 77 s, 52 s and 43 seconds respectively in the hot powered channel, averaged powered channel and peripherial channel (Fig. IV-17). The CS flows down through the upper tie plate was sufficient to turn the cladding temperature around, even though the liquid downflow was limited by the CCFL phenomena. (Fig. IV-20). The fuel rods were effectively quenched at 122 s when the cladding temperature dropped to within 5 K around saturation temperature. Fig. IV-18 shown the channels collapsed liquid level.

Fig. IV-19 shows the Rod Temperature evolution during first seconds.

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. Since about second 45 the evolution of time steps show sharp oscillations. Several changes of heat transfer and flow regimes took place forcing the code to select smaller time steps.

The run statistics are shown in Table V-1. The real time/CPU time ratio was 1/3352.

VI. CONCLUSIONS

A considerable margin of 160°C in peak clad temperature respect to the so-called nominal SAFER calculations has been obtained using TRAC-BF1/G1J1 code. However, it is recognized that some of the calculation assumptions could still be considered conservative.

A Critical Heat Flux is not reached within the first phase of the transient and as a result there is not first peak in PCT.

Identified uncertainty areas will require additional sensitivy analysis in order to fully qualified the Garoña LOCA model. Two-phase flow through recirculation pumps and leakage flows in lower plenum, guide tubes, channel and bypass regions are two of these areas.

It is believed that the current model covers the main LOCA phenomena addressed by licensing methodologies in a best estimate way.

The Counter Current Flow has been identified to be the key phenomena in the analyzed LOCA scenario. The liquid downflow through the upper tie plate into the channel was the main contributor to core cooling.

Therefore uncertainties related to this phenomena should be addressed (i.e. by sensitivy analysis) in order to properly evaluate this particular LOCA scenario.

VII. REFERENCES

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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	0.4775 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.

TABLA III - 1

DESCRIPTION OF THE TRAC COMPONENTS

NUMBER	COMPONENT	DESCRIPTION
1	VESSEL	Vessel
2	PIPE	Central guide tube
3	PIPE	Average guide tube
4	PIPE	Peripherical guide tube
10	PIPE	Suction pipe of the recirculation loop I
11	VALVE	Isolation valve of the recirculation loop I
20	PIPE	Suction pipe of the recirculation loop II
21	VALVE	Isolation valve of the recirculation loop II
22	PUMP	Recirculation pump of the loop I
23	VALVE	Isolation valve of the recirculation loop II
24	PIPE	Discharge pipe of the recirculation loop II
25	JETPUMP	Jet pump of the recirculation loop II
26	JETPUMP	Jet pump of the recirculation loop I
31	VALVE	Isolated valve of the recirculation loop II
32	PUMP	Recirculation pump of the loop II
36	CHAN	Penpheral bundle
38	CHAN	Average bundle
40	CHAN	Hot bundle 1
41	PIPE	Discharge pipe of the recirculation loop I
43	CHAN	Hot bundle 2
50	PIPE	Main steam line from the vessel
52	PIPE	Main steam line to the turbine
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 1
71	FILL	Low pressure core spray 1
72	PIPE	Pipe of the core spray 2
73	FILL	Low pressure core spray

TABLA I V- 1

INITIAL CONDITIONS

PARAMETER	MEASURED VALUE	TRAC-BF1/G1J1
Reactor Vessel		
Total Core Power (104%), (Mw)	—	1435.00
Downcomer Water Level, (m)	10.58	10.58
Steam Dome, (MPa)	7.00-7.03	7.01
Total Core Mass Flow, (kg/s)	5,900	6,000
Core Bypass (10%), (kg/s)	—	598.40
Recirculation Loop		
Speed Pump, (rad/s)	135	130.6
Feedwater System		
Feedwater Mass Flow, (kg/s)	—	714.00
Main Steam Line		
Steam Mass Flow, (kg/s)	—	714.00

TABLE IV-2

CHRONOLOGY OF EVENTS

Event	Time (seconds)
Break Initiated	0.0
Loss of Offsite Power	0.0
Trip of Feedwater System	0.0
Recirculation Pump Coastdown Started	0.0
Scram Signal Generated	0.5
Low-Low Water Level	1.3
Control Rods Fully Inserted	2.9
Main Steam Isolation Valves Closed	3.0
Jet Pump Drive Uncovered	5.2
Lower Plenum Flashing	10.0
Recirculation Loop Suction Uncovered	10.5
HPCI Flow Started	28.5
Dryout at the Peak Power Zone	24.2
CS Flow Started	33.0
Lower Plenum Refill Started	54.6
HPCI stopped	52.0
Reflood Initiated	61.2

TABLE IV-2 (Cont.)

Maximum Cladding Temperature reached	77.0
Peripherial Bundle Quenched	47.8
Average Bundles Quenched	61.5
Hot Bundles Quenched	82.3

TABLA V-1

RUN STATISTICS

Real Time	RT= 120 seconds
CPU Time	CPU = 402,084 seconds
Total number of volumes in the model	C= 168
Total number of time steps	DT= 197,543

$(CPU \times 10^3)/C \times DT = 12.1$

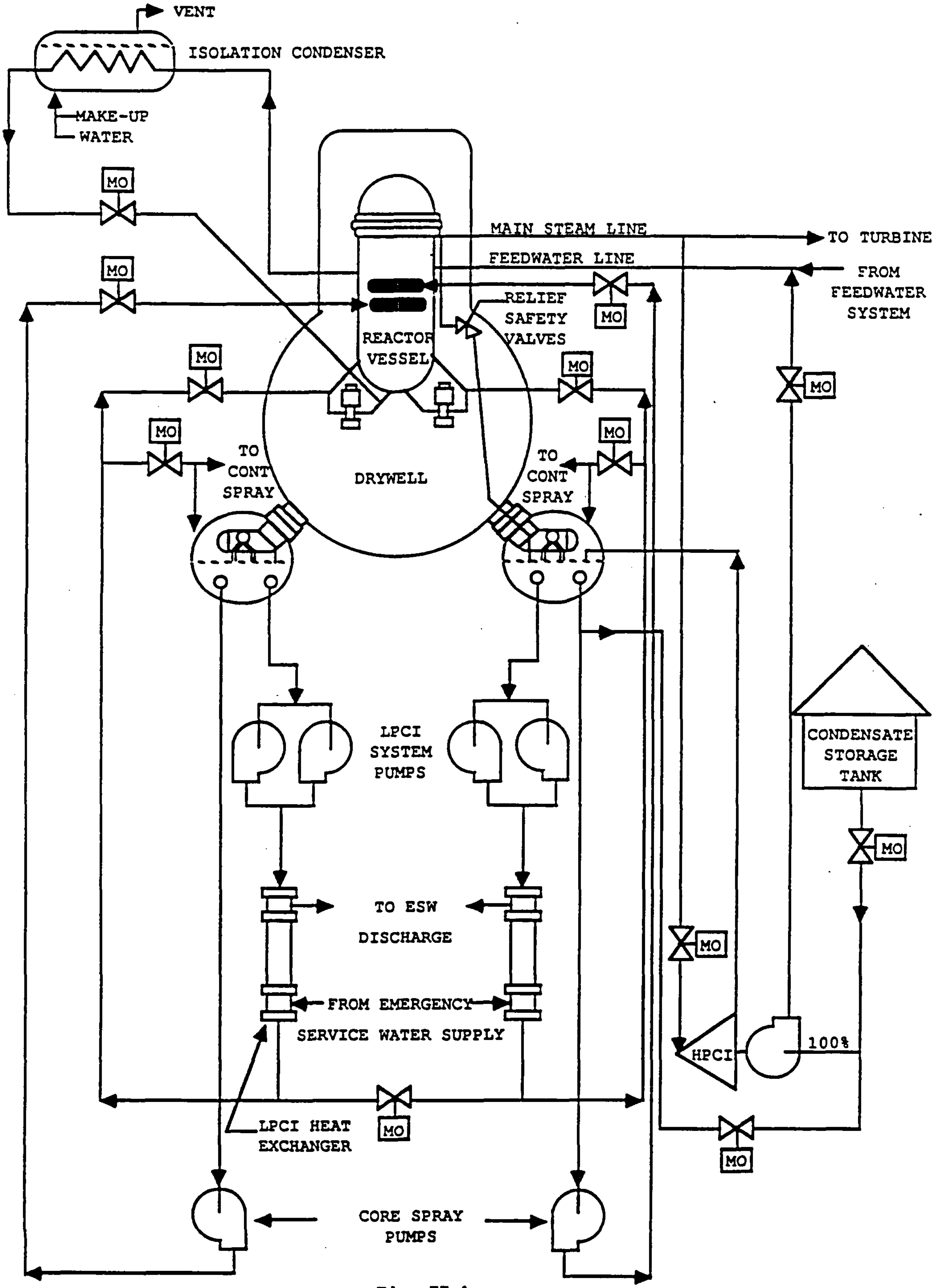


Fig. II-1
GAROÑA EMERGENCY CORE COOLING SYSTEMS AND ISOLATION CONDENSER SYSTEM

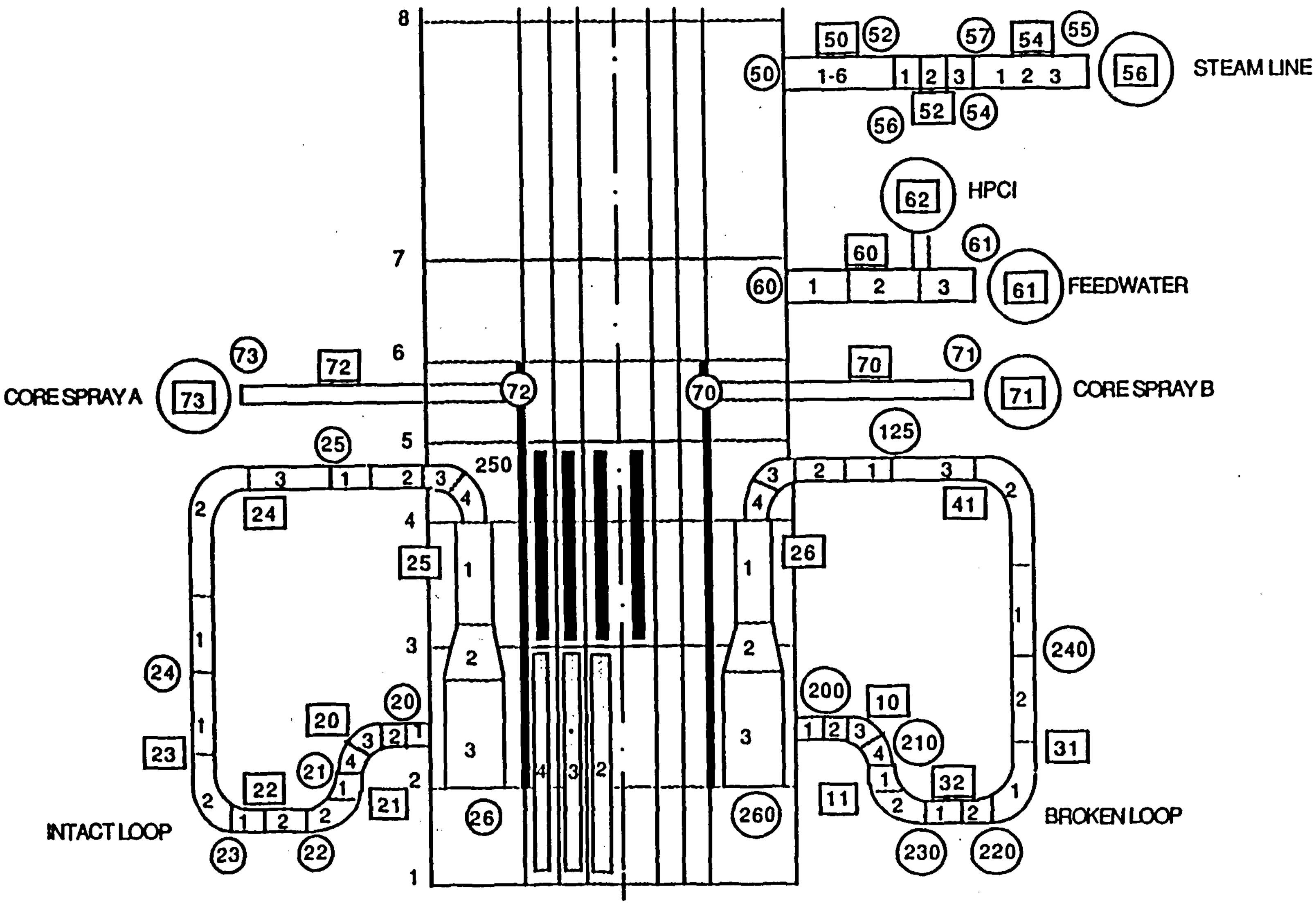


Figure III-1 TRAC-BF1 AXIAL NODALIZATION

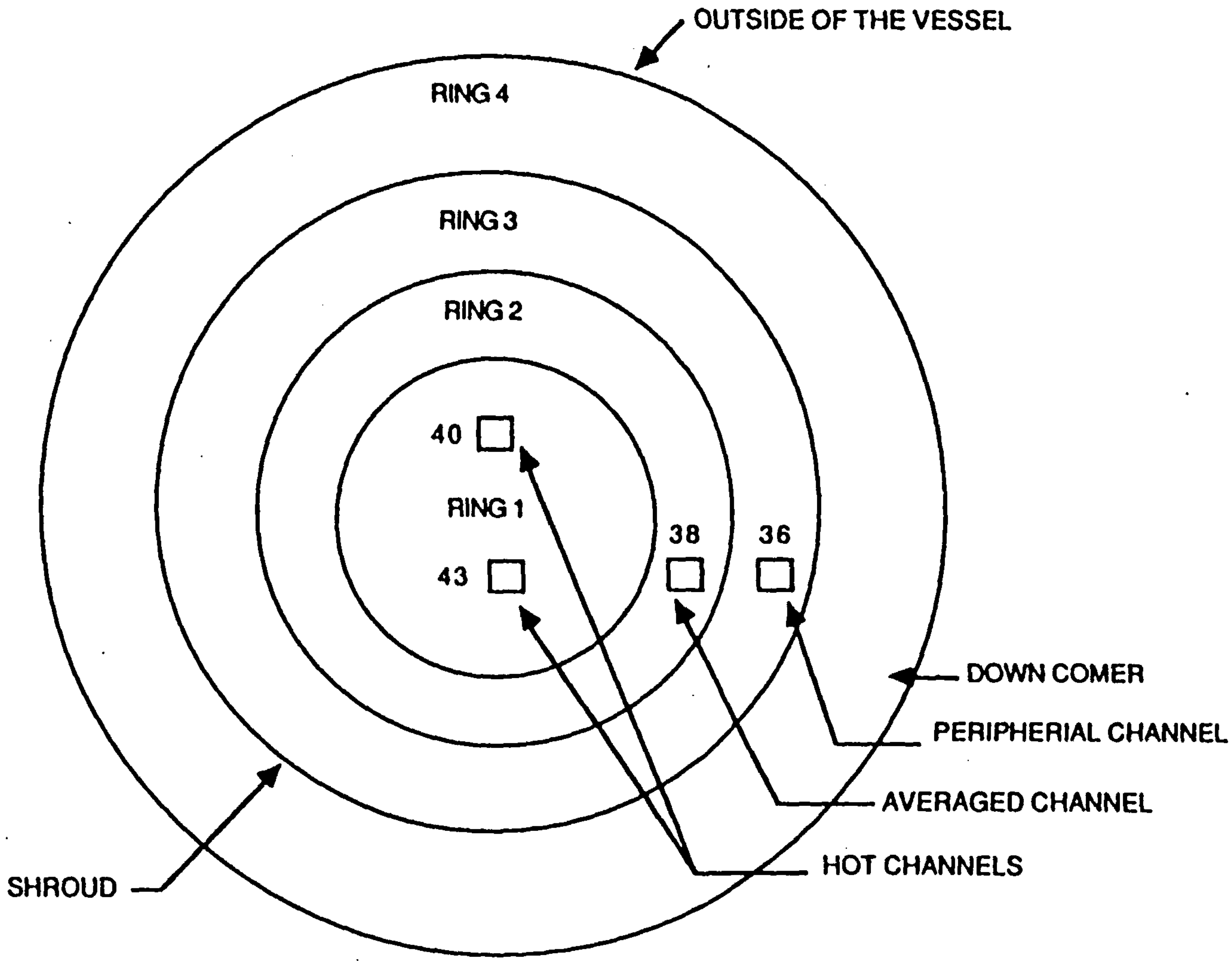
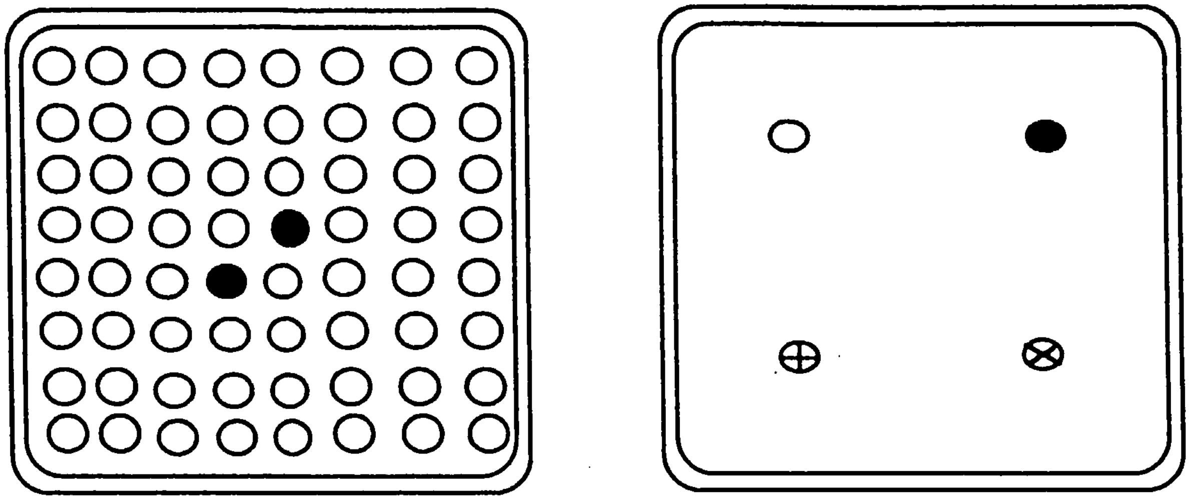


Figure III-2 TRAC-BF1 RADIAL NODALIZATION



WATER ROD

FUEL ROD

P8 X 8R

ACTUAL FUEL ELEMENT

WATER ROD

HOT FUEL ROD

AVERAGED FUEL ROD

COLD FUEL ROD

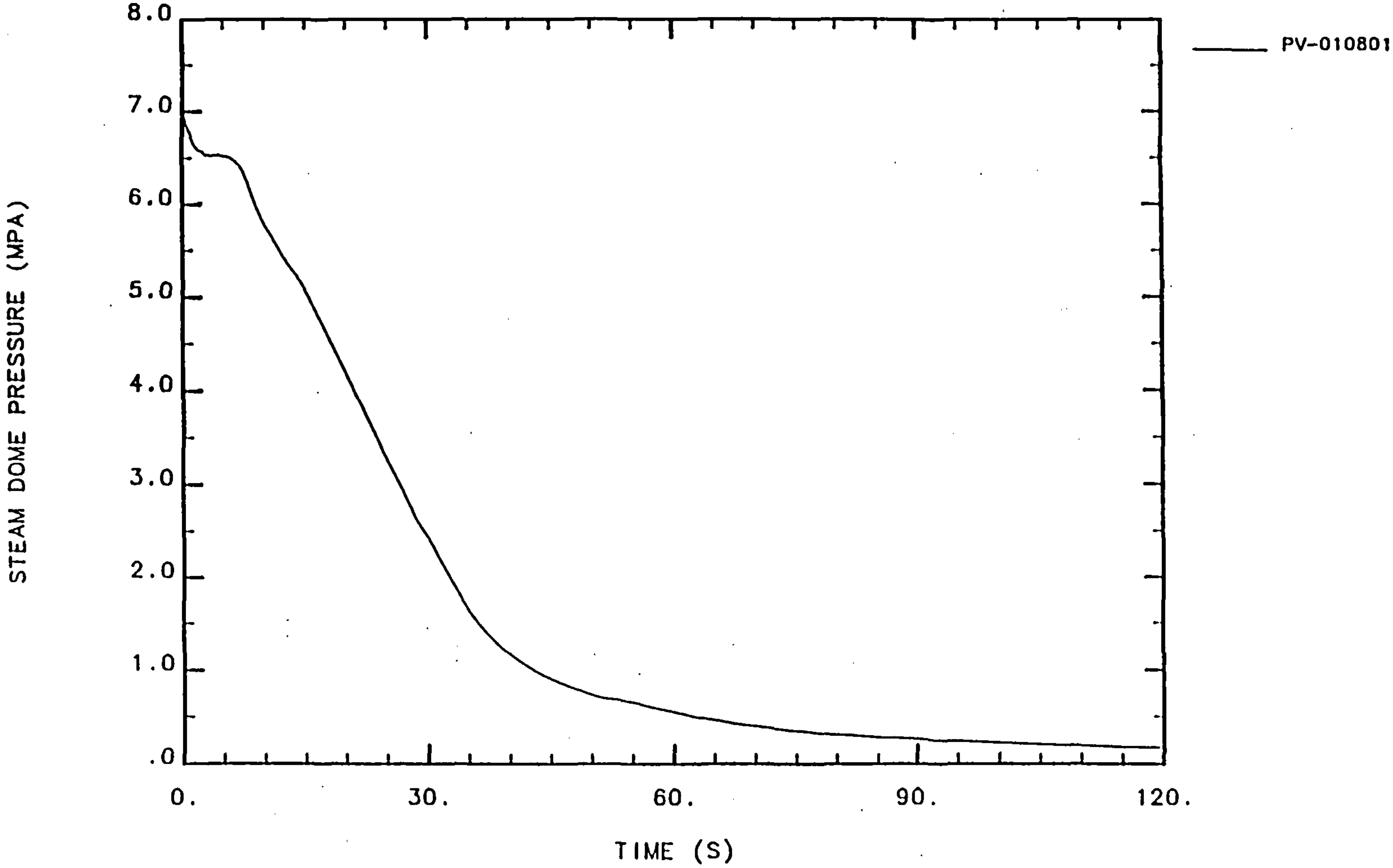
P8 X 8R

FUEL ELEMENT MODEL

Figure III-3 FUEL ELEMENT MODEL

SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90

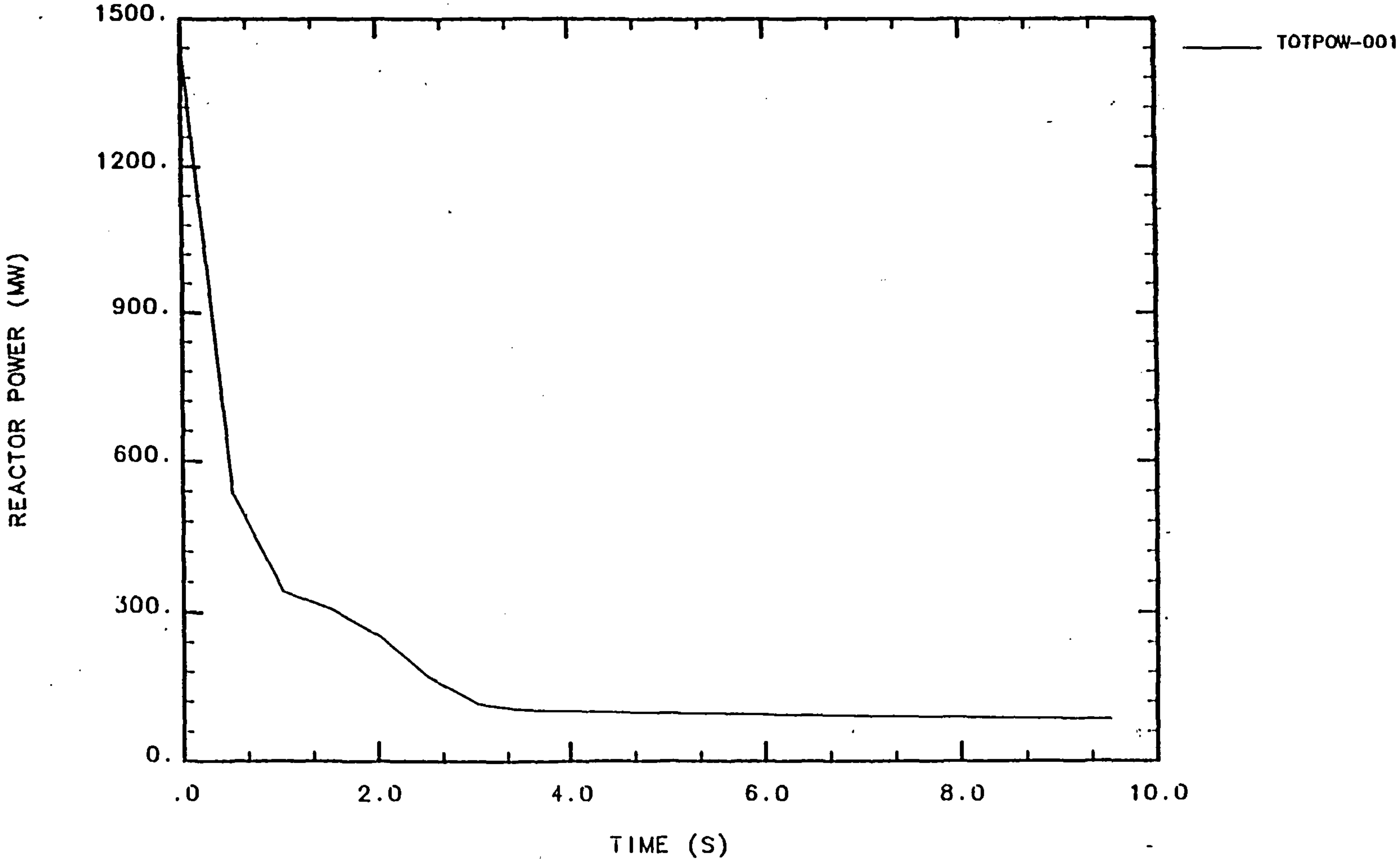


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-1

SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90

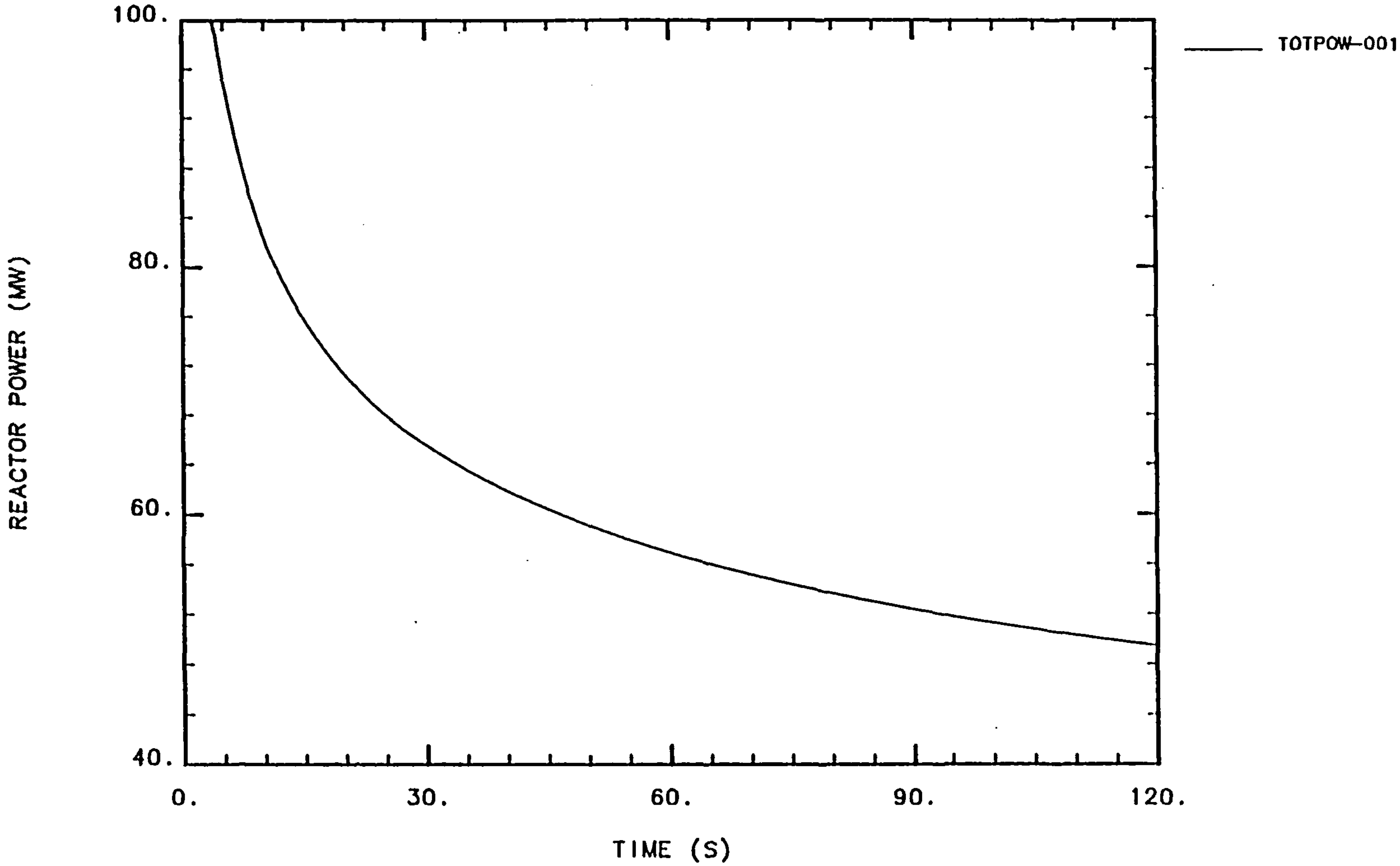


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-2

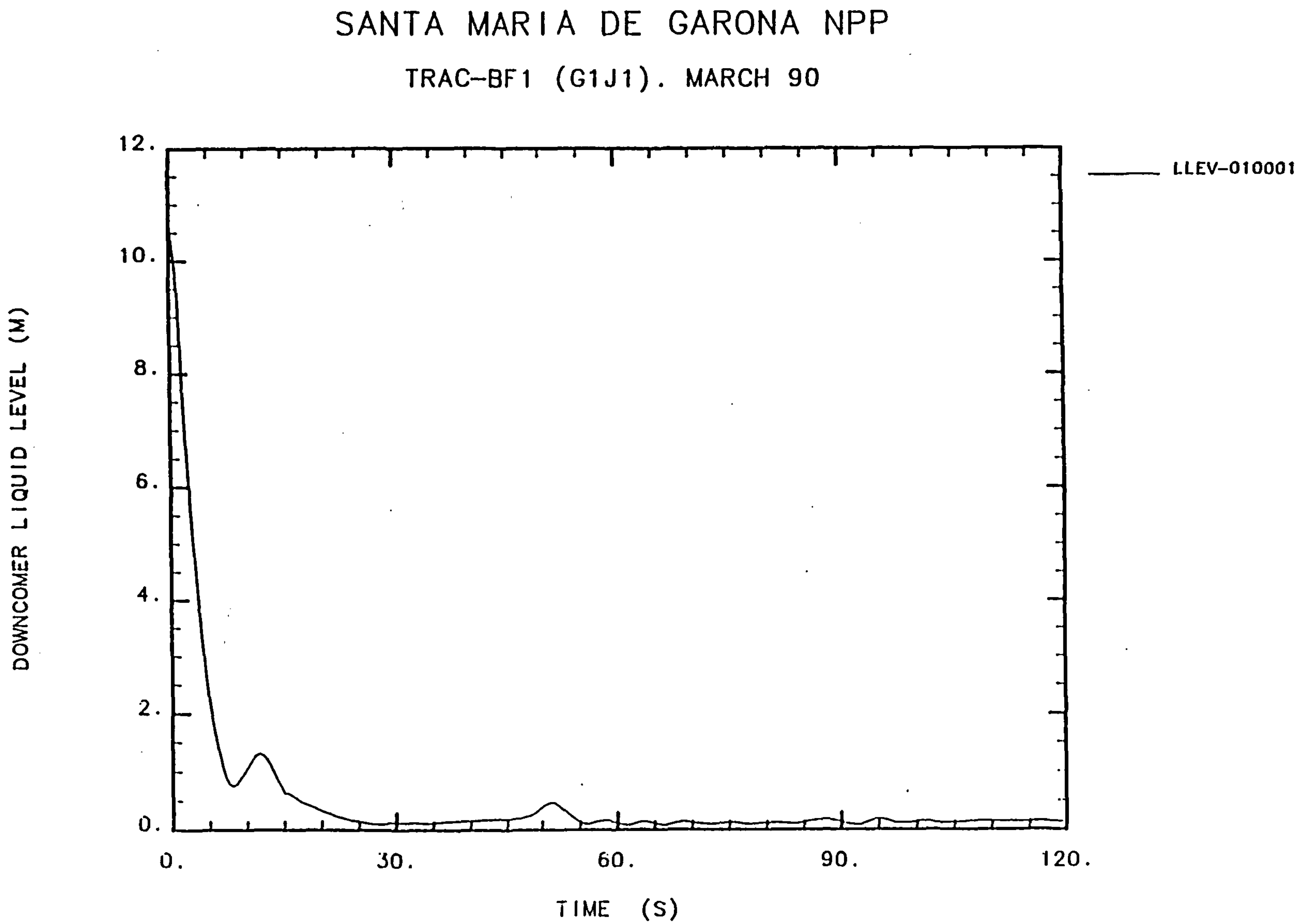
SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90



RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-3

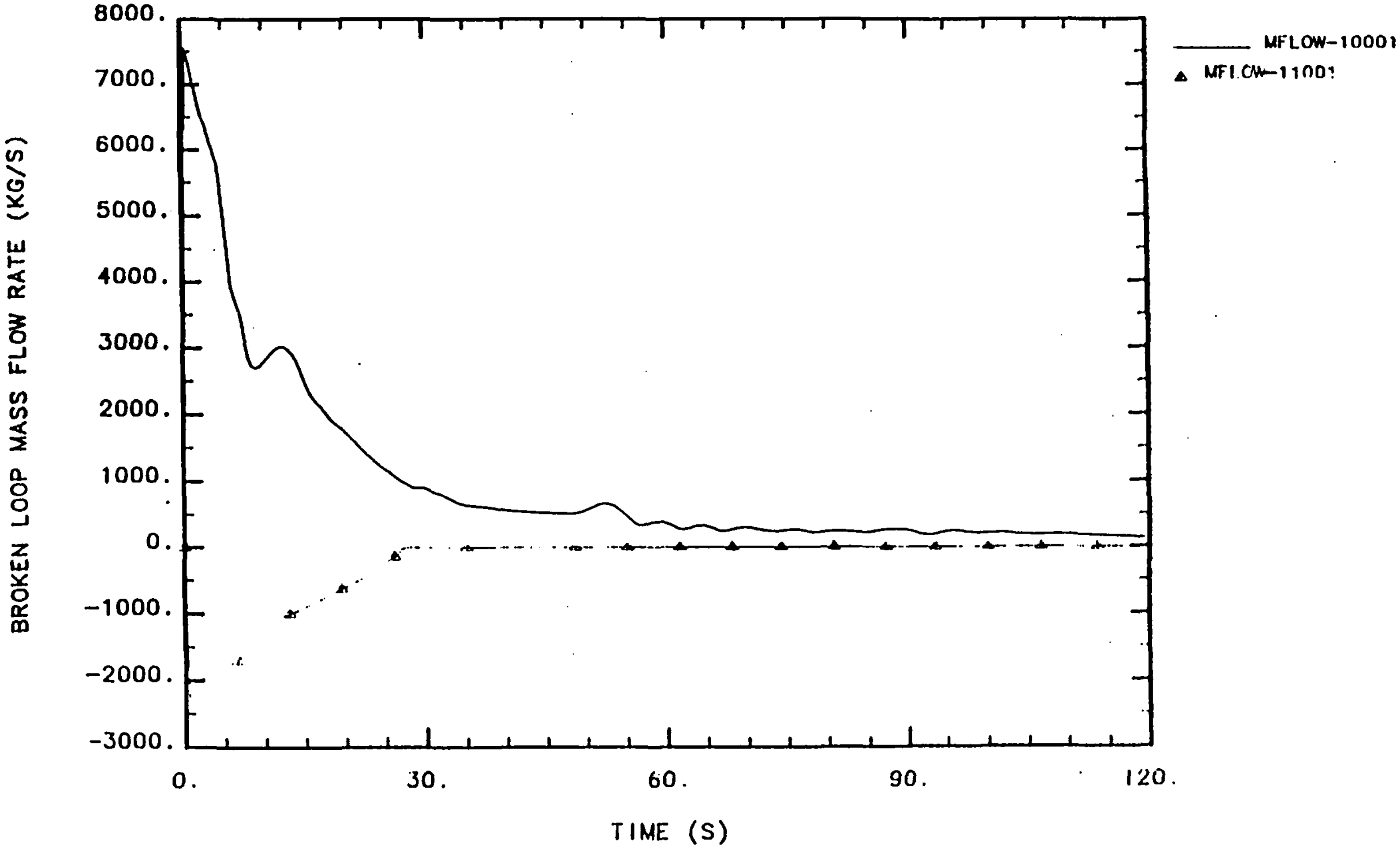


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-4

SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90

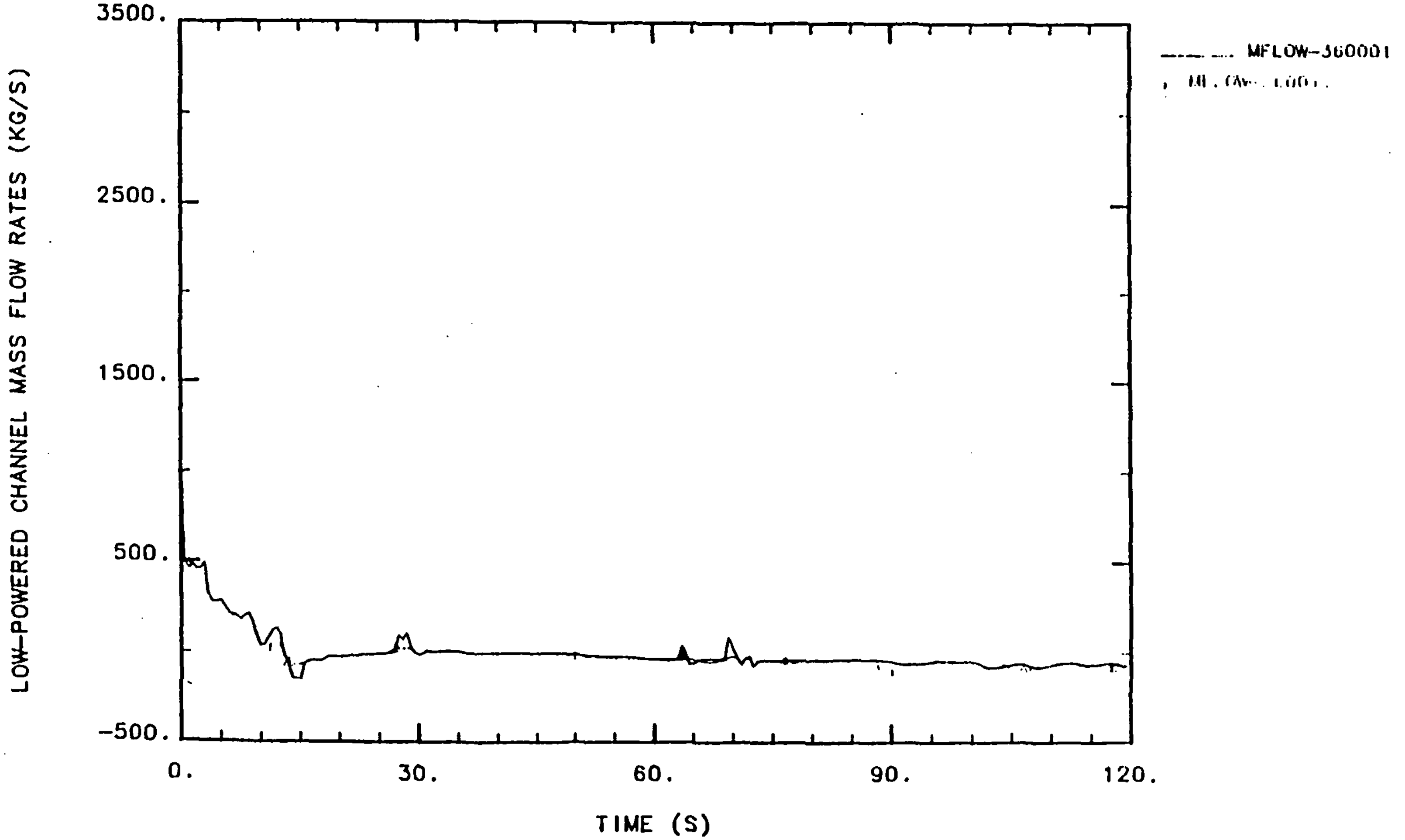


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-5

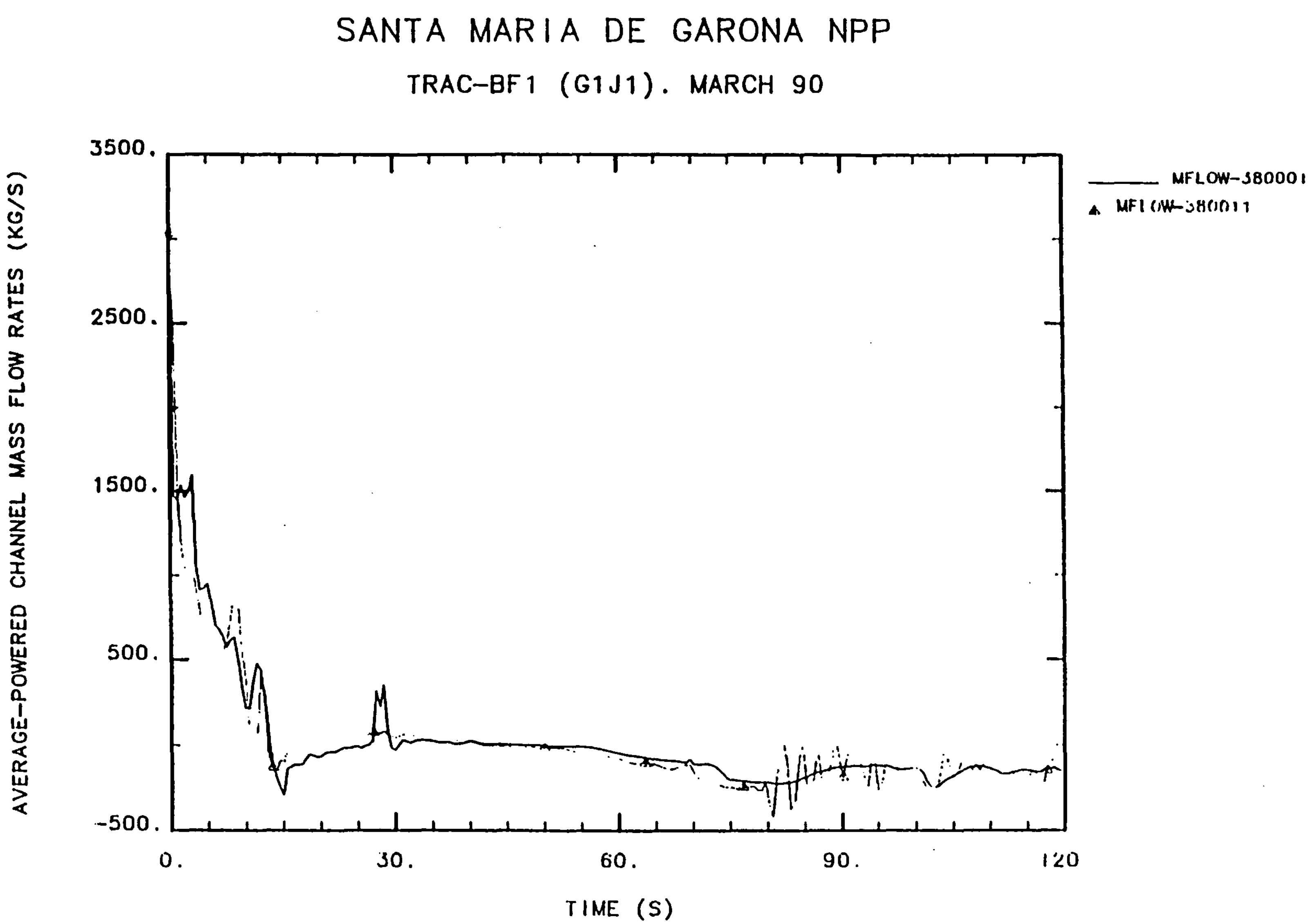
SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90



RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-6

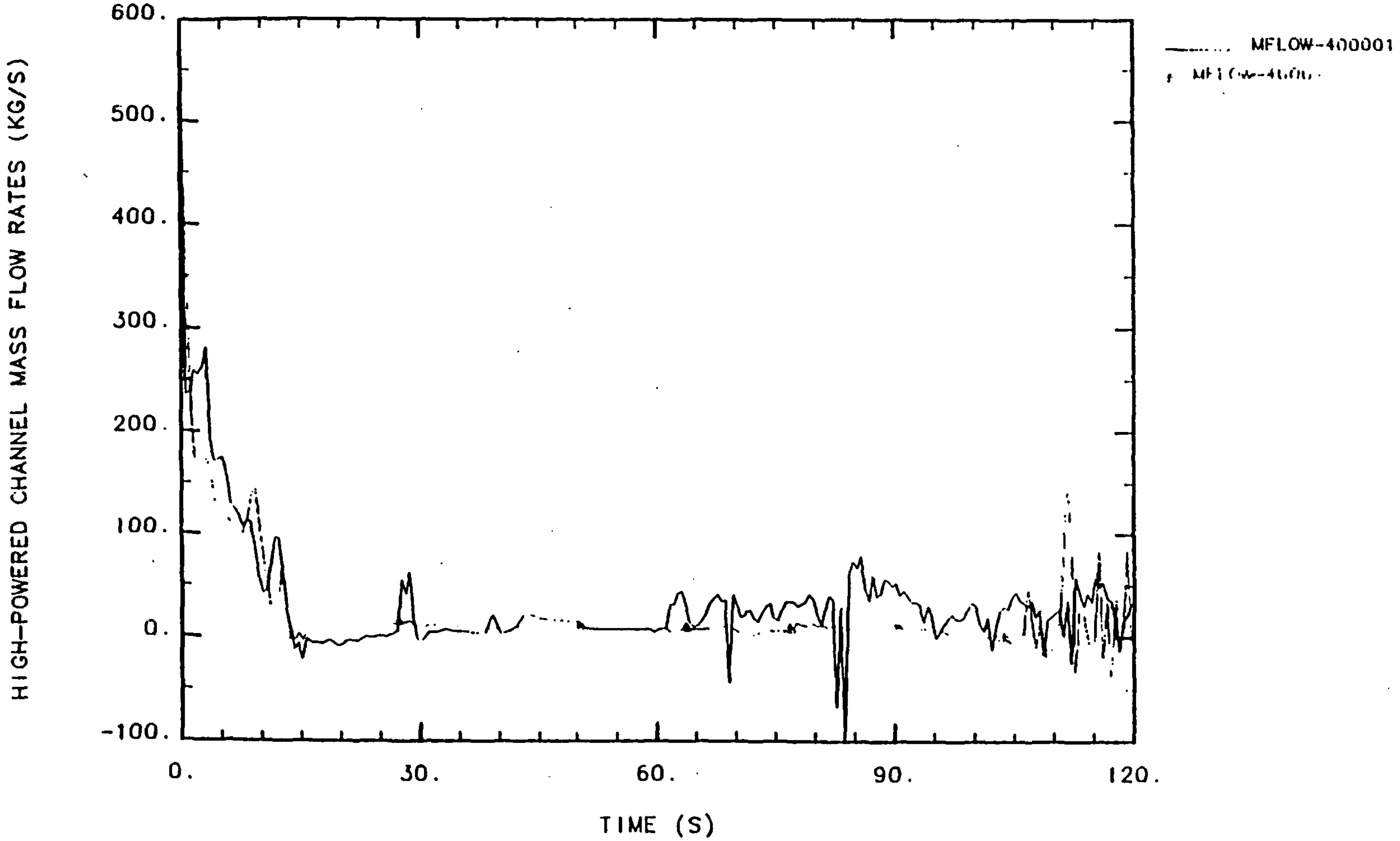


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-7

SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90

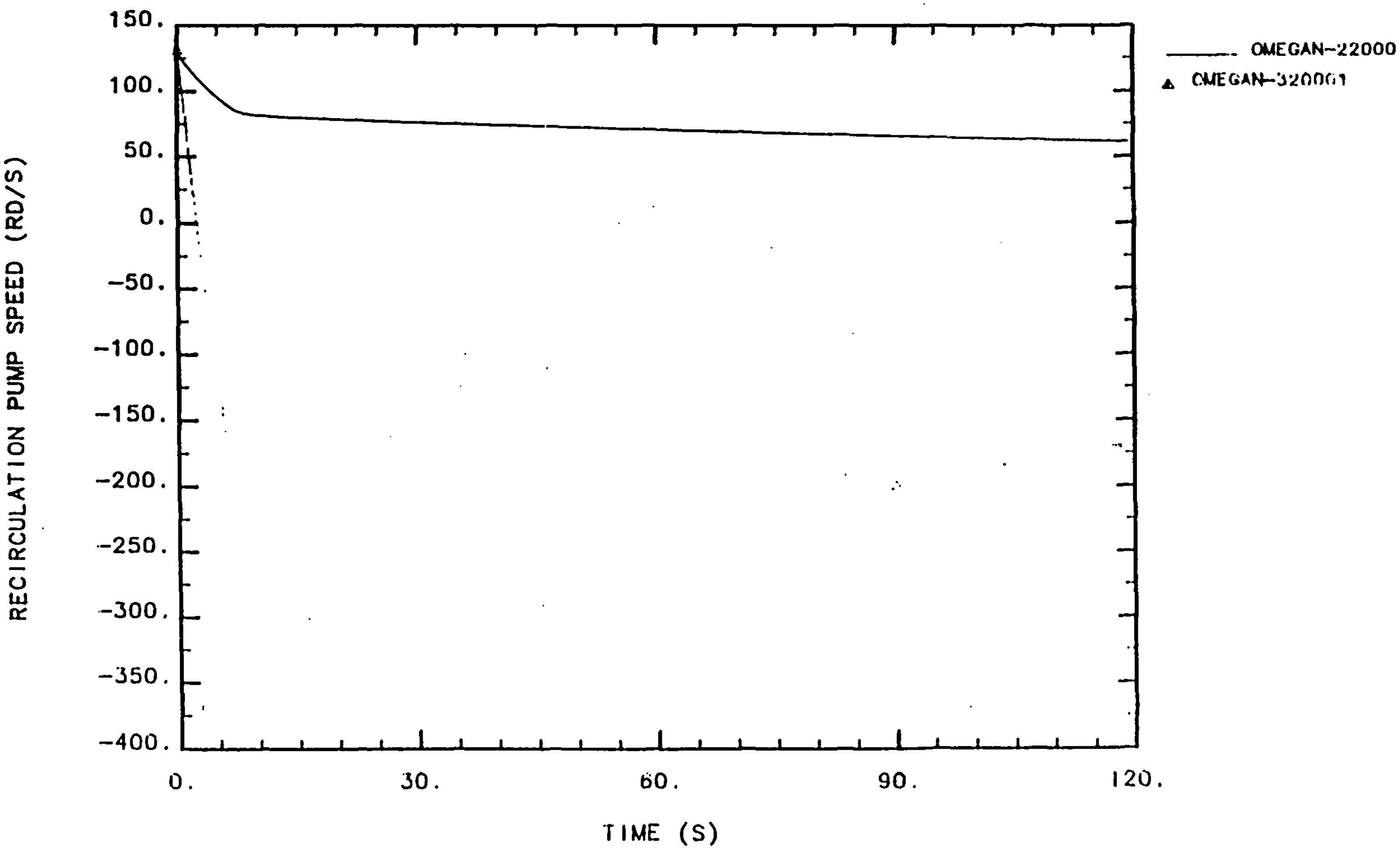


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-8

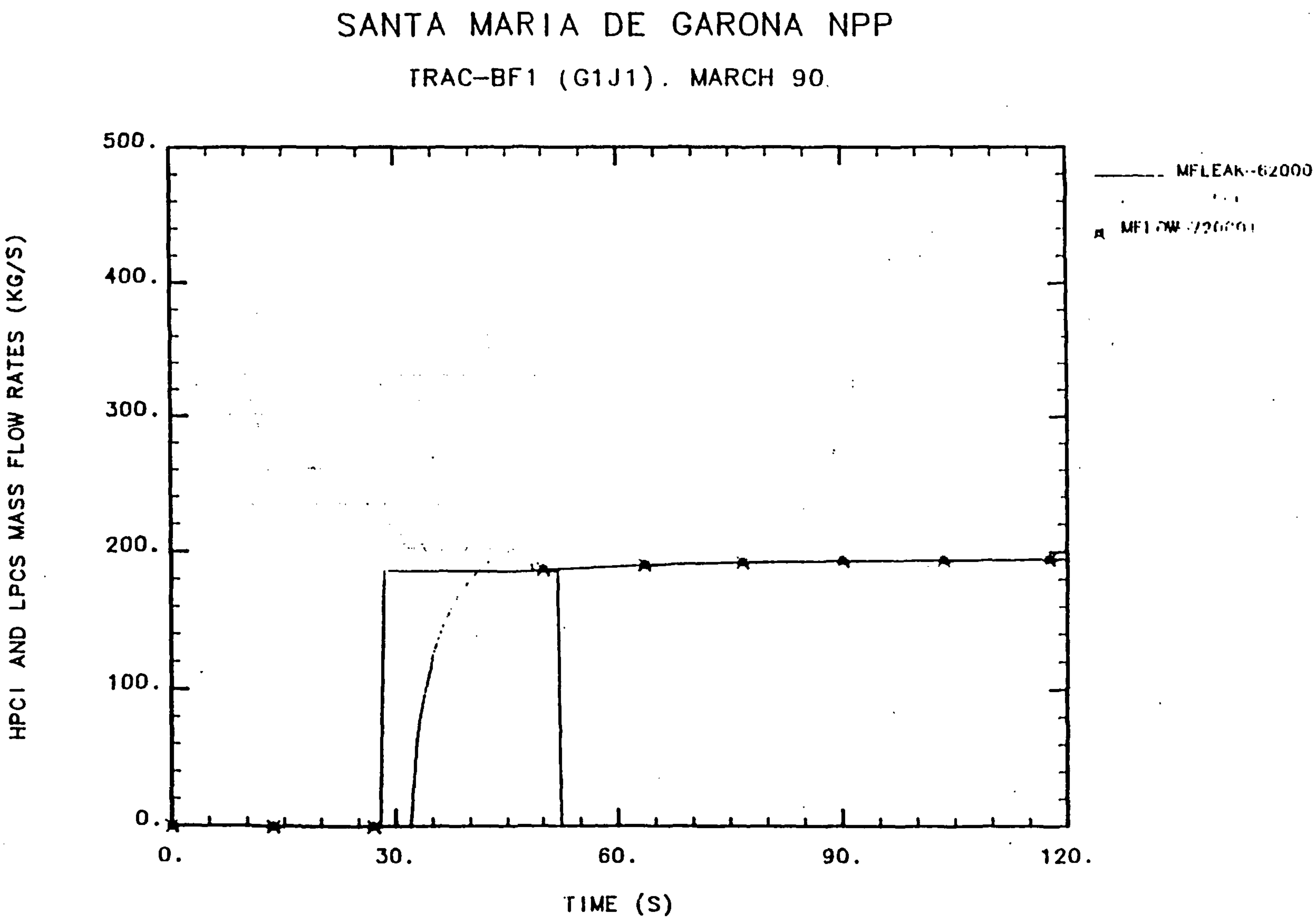
SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90



RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-9

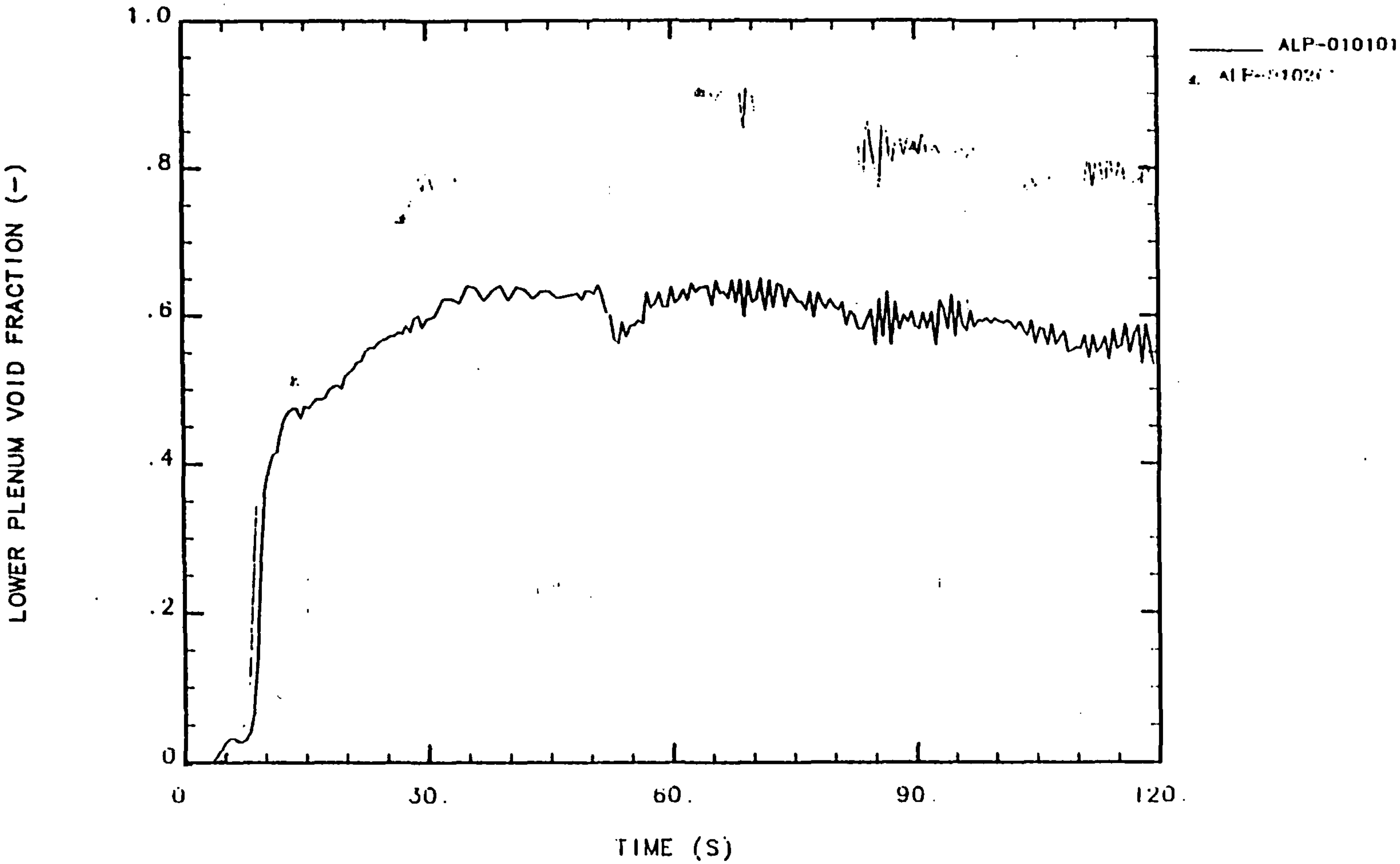


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-10

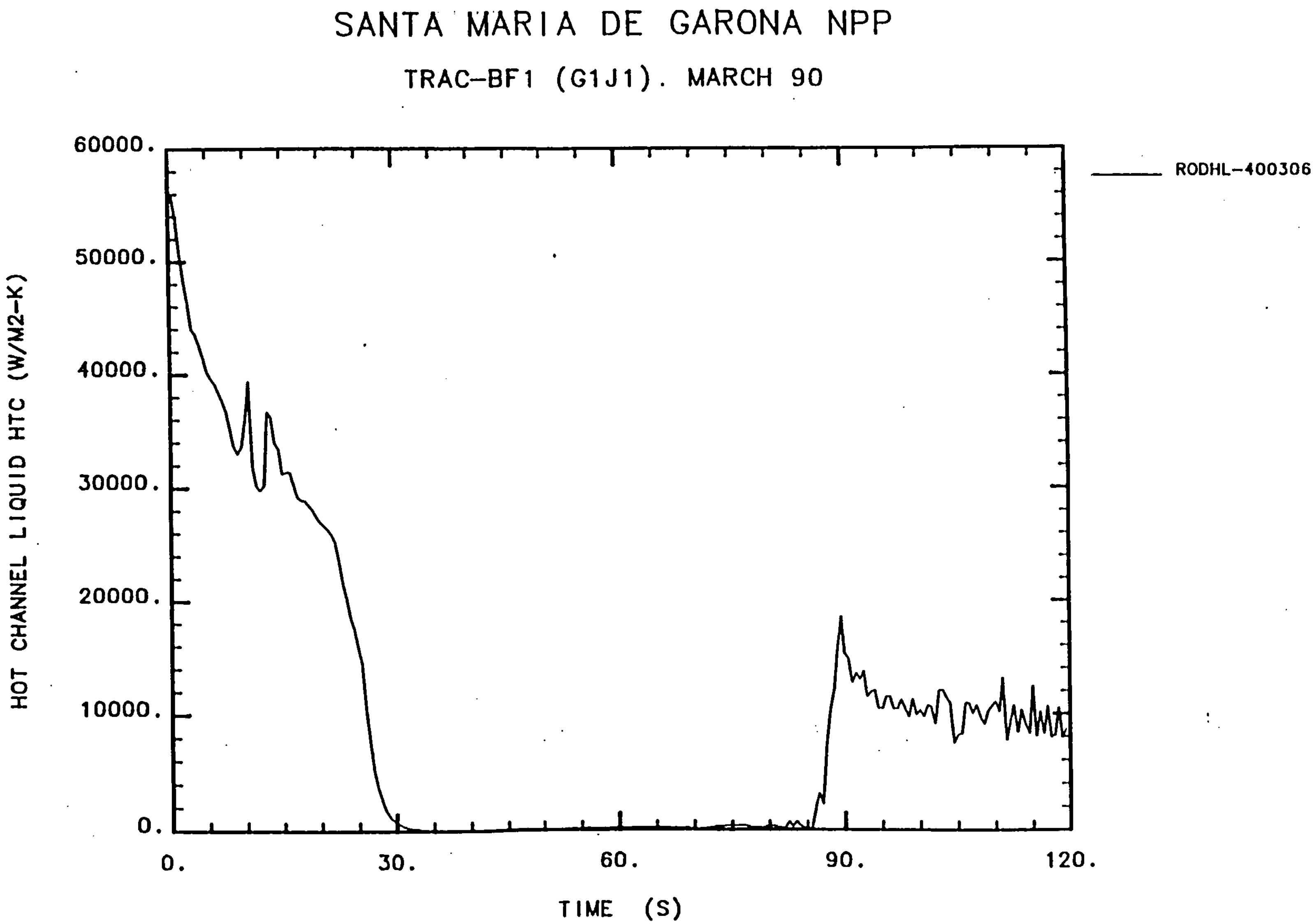
SANTA MARIA DE GARONA NPP

TRAC-BFI (G1J1). MARCH 90



RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-11

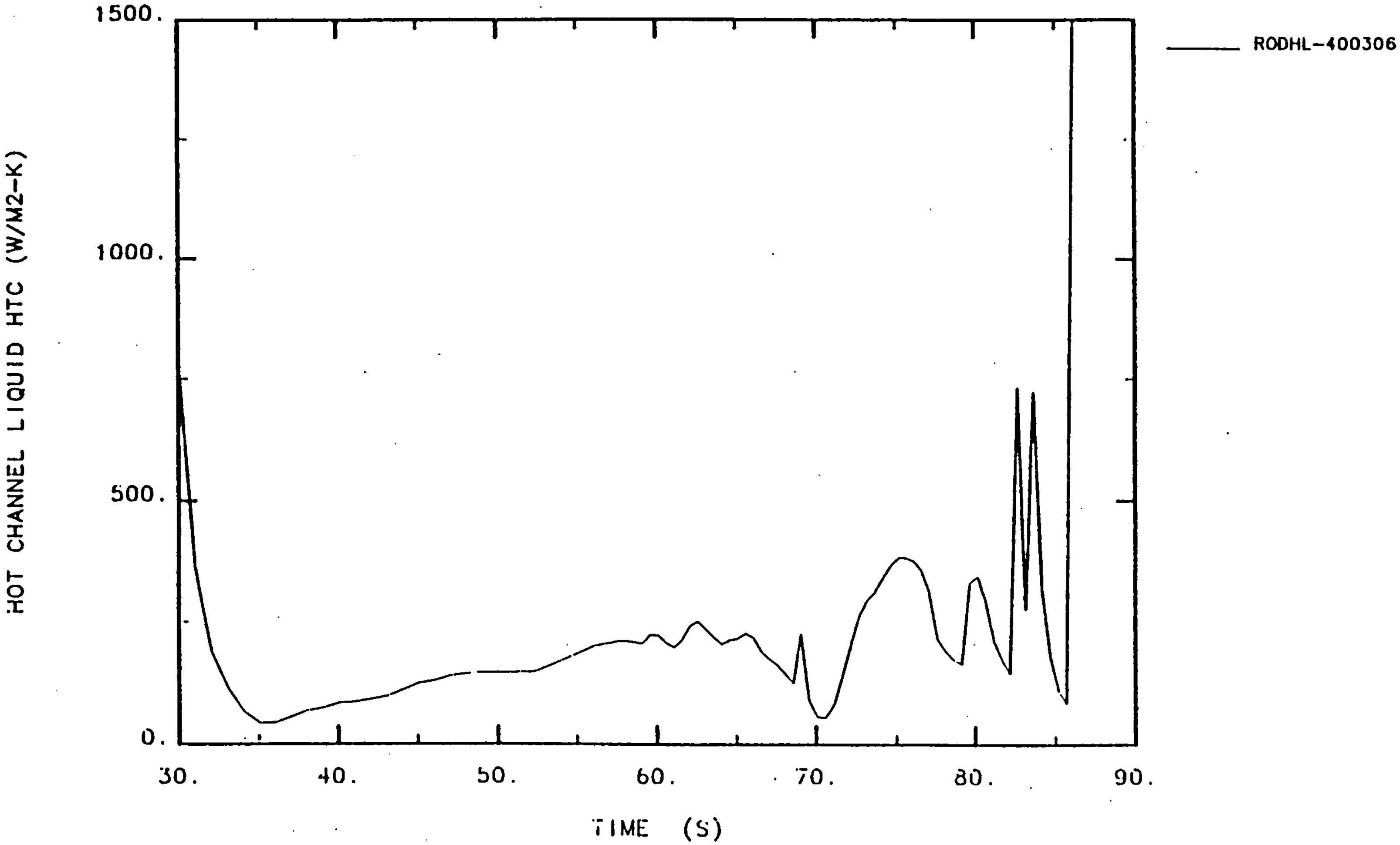


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-12.a

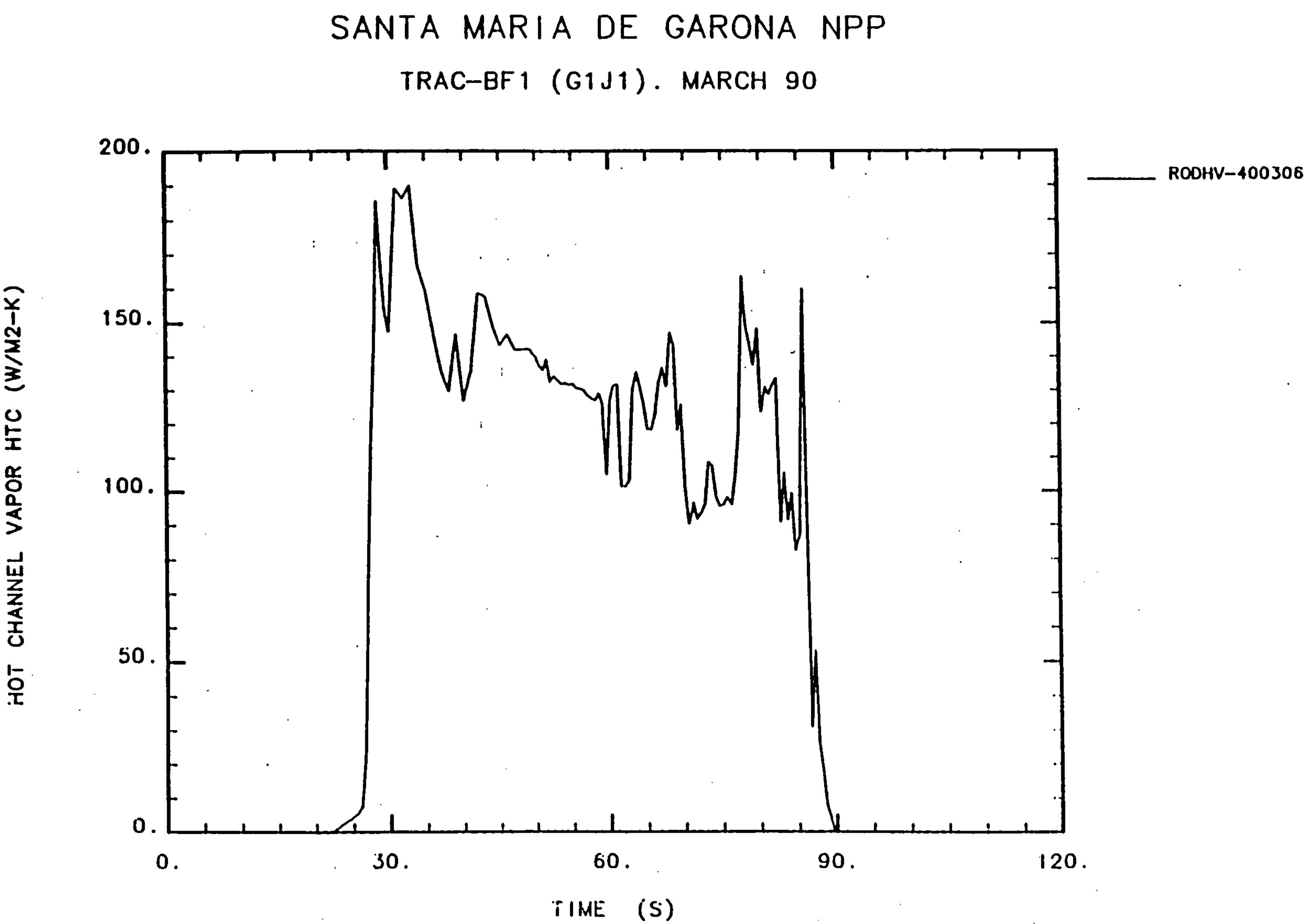
SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90



RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-1.[illegible] b

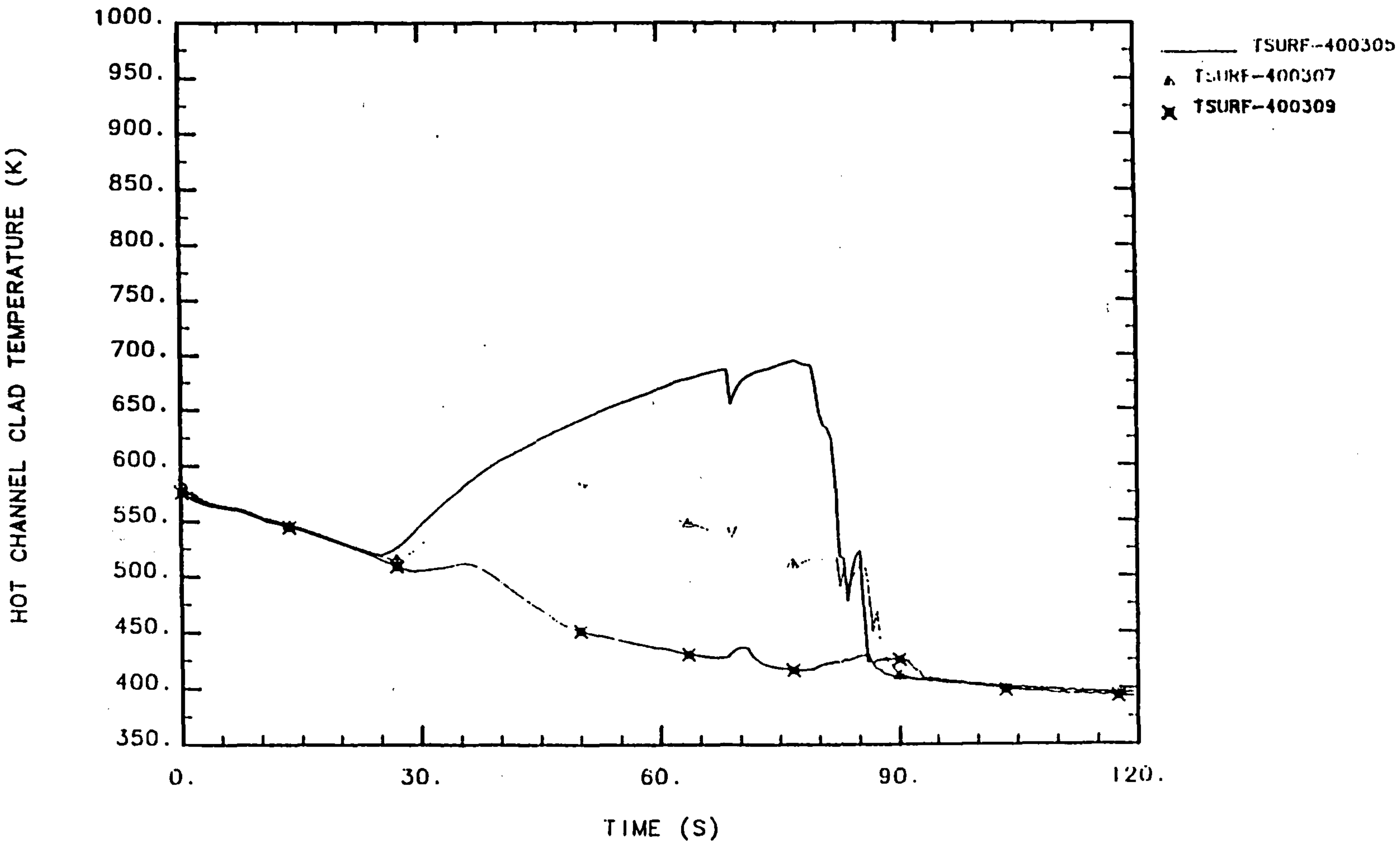


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-13

SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90

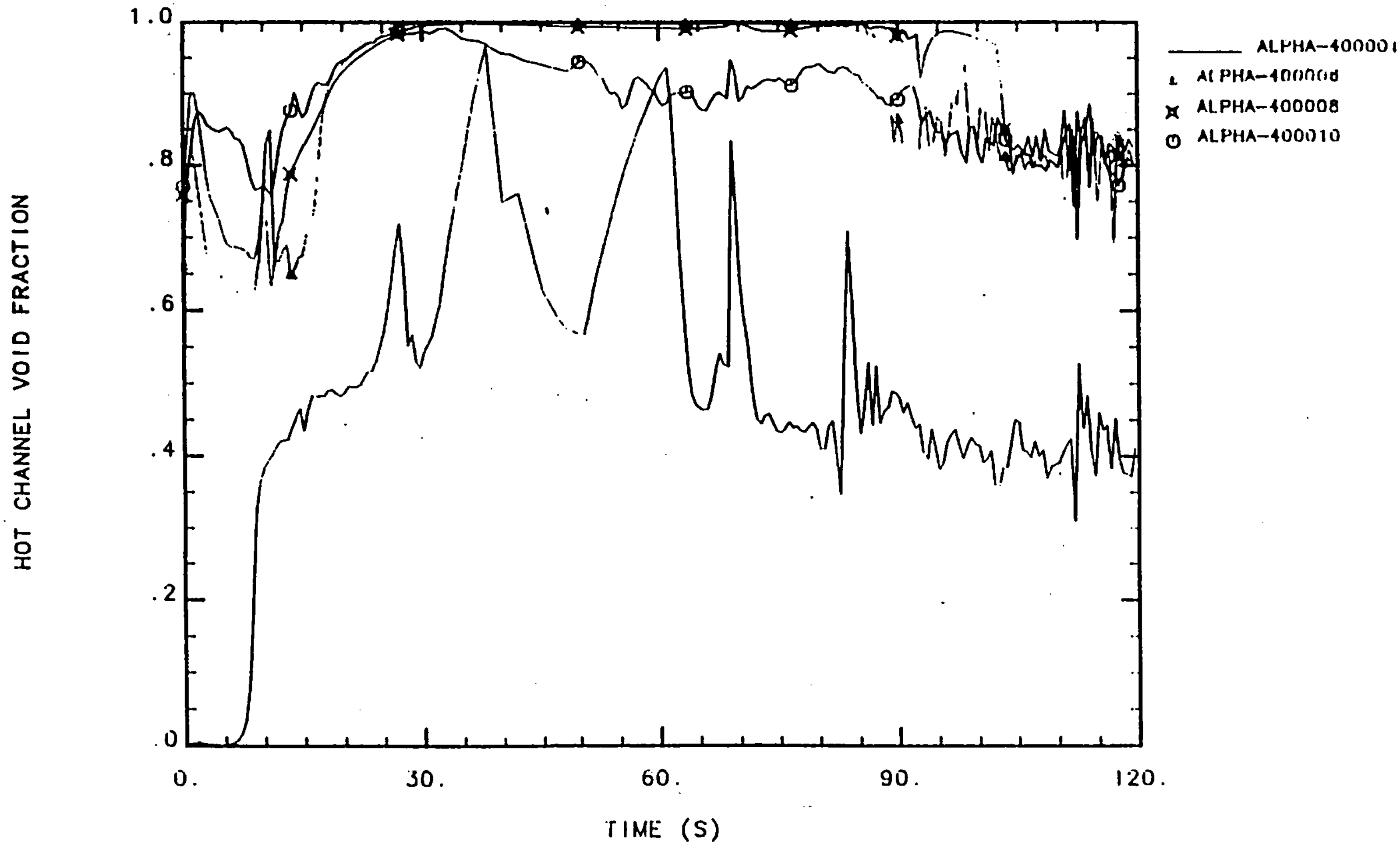


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-14

SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90

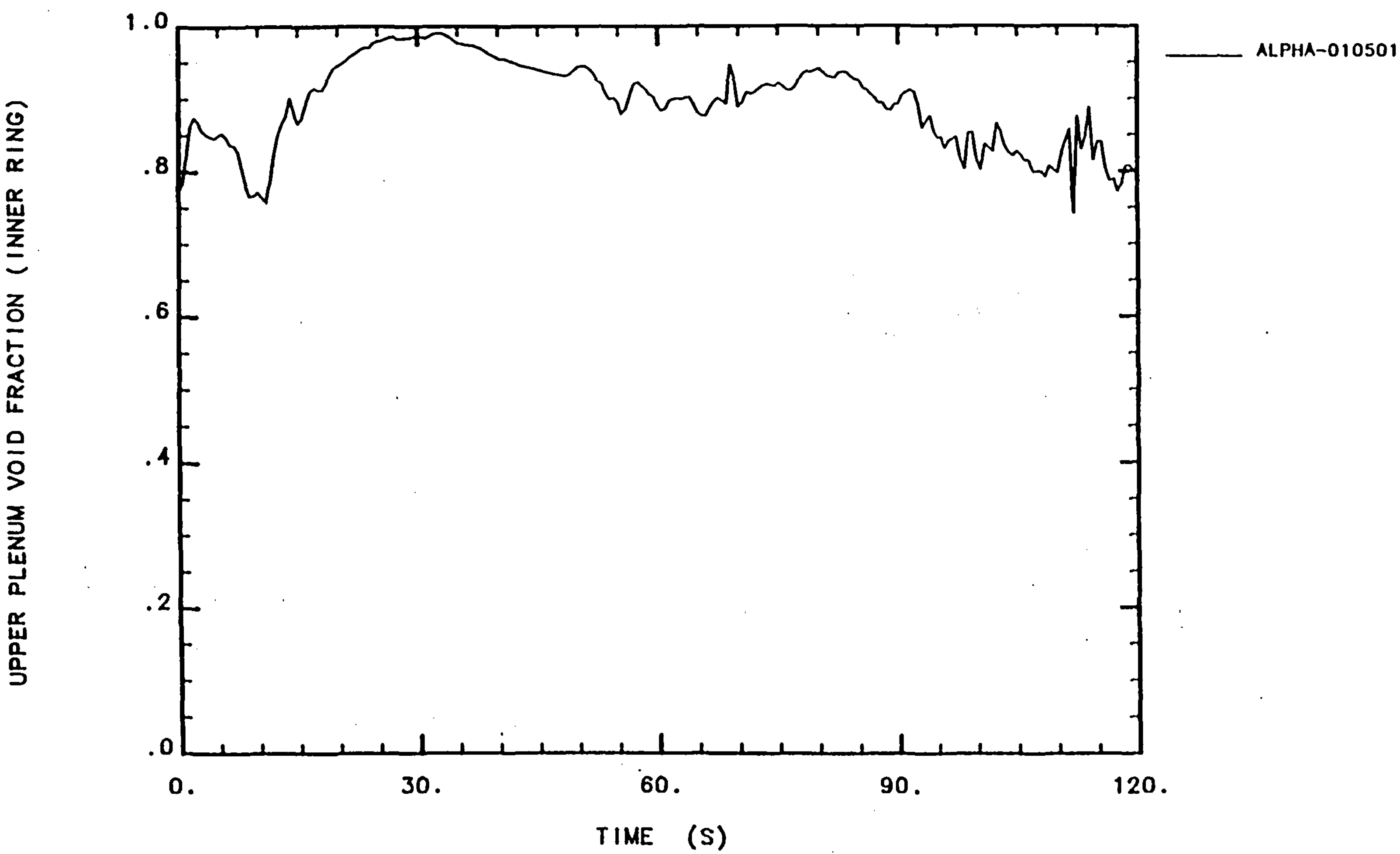


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-15

SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90

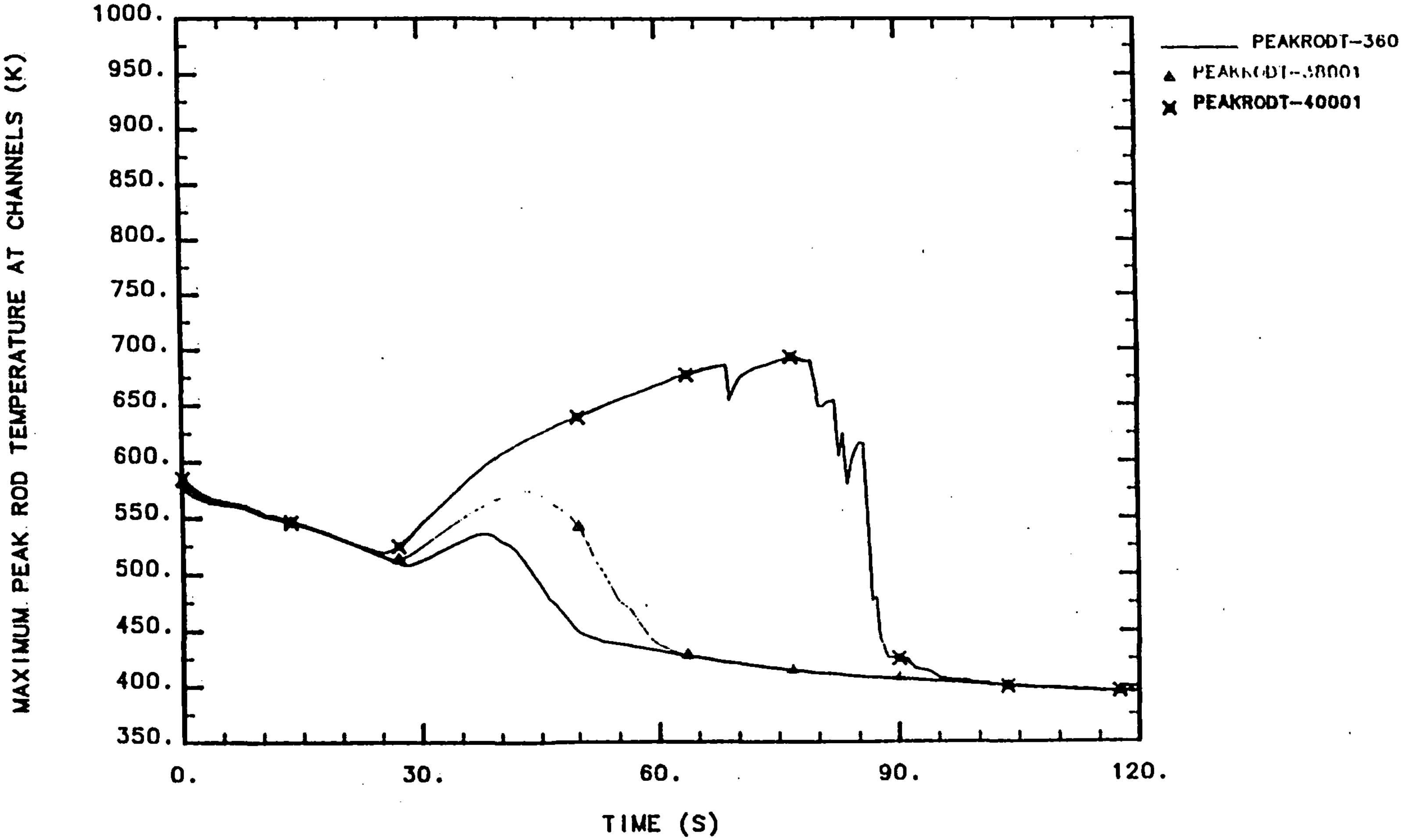


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-16

SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90

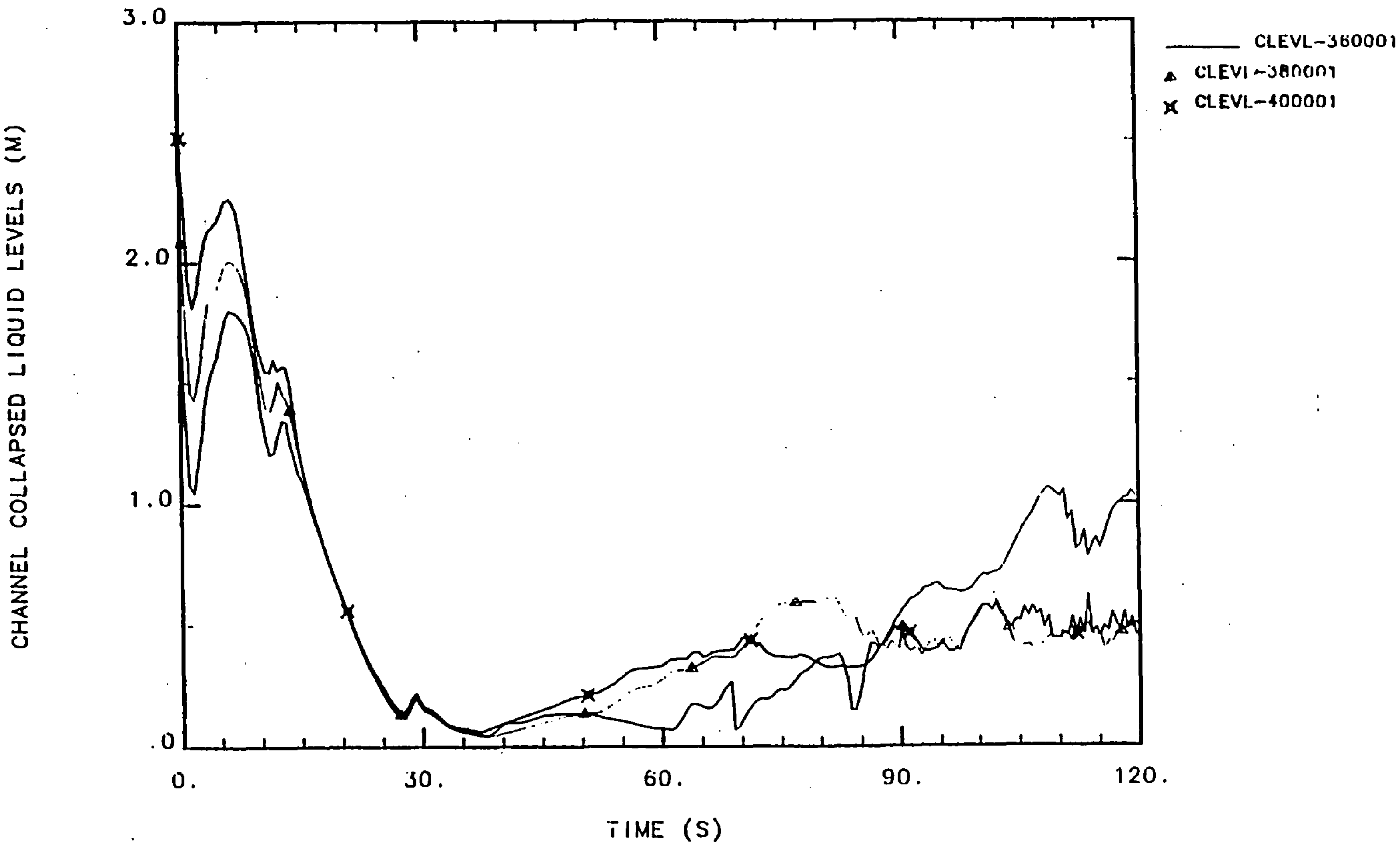


RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-17

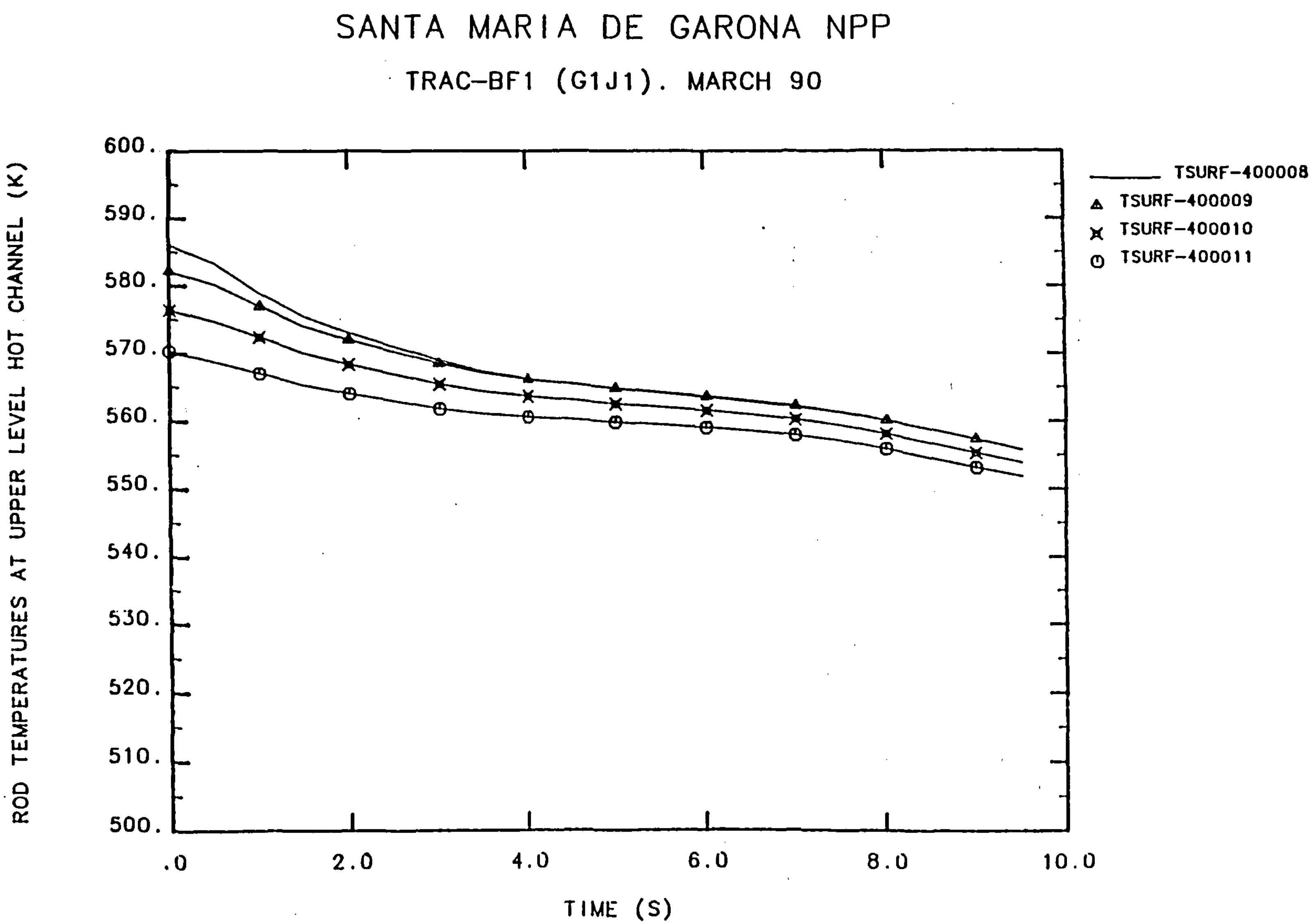
SANTA MARIA DE GARONA NPP

TRAC-BF1 (G1J1). MARCH 90



RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. IV-18



RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

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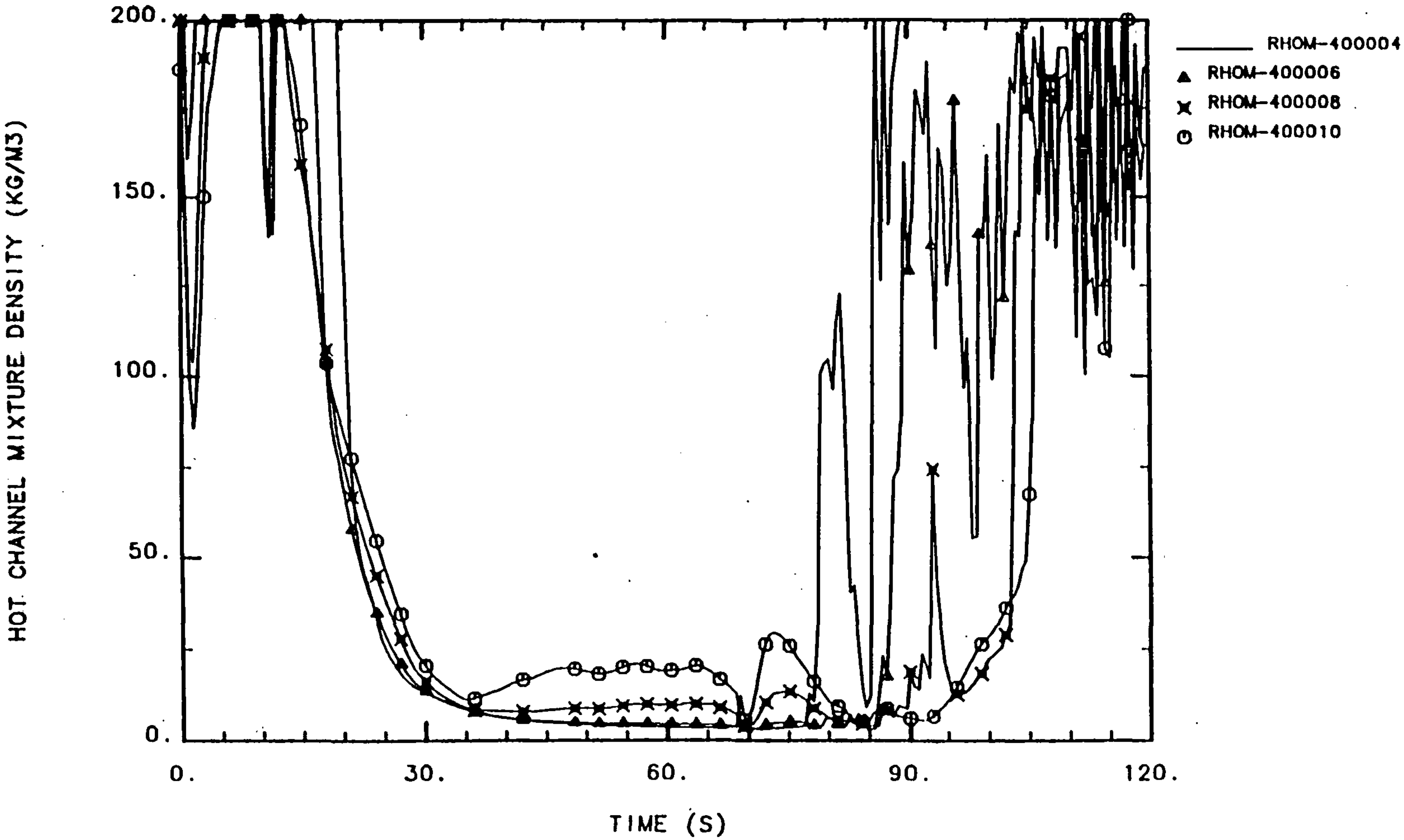
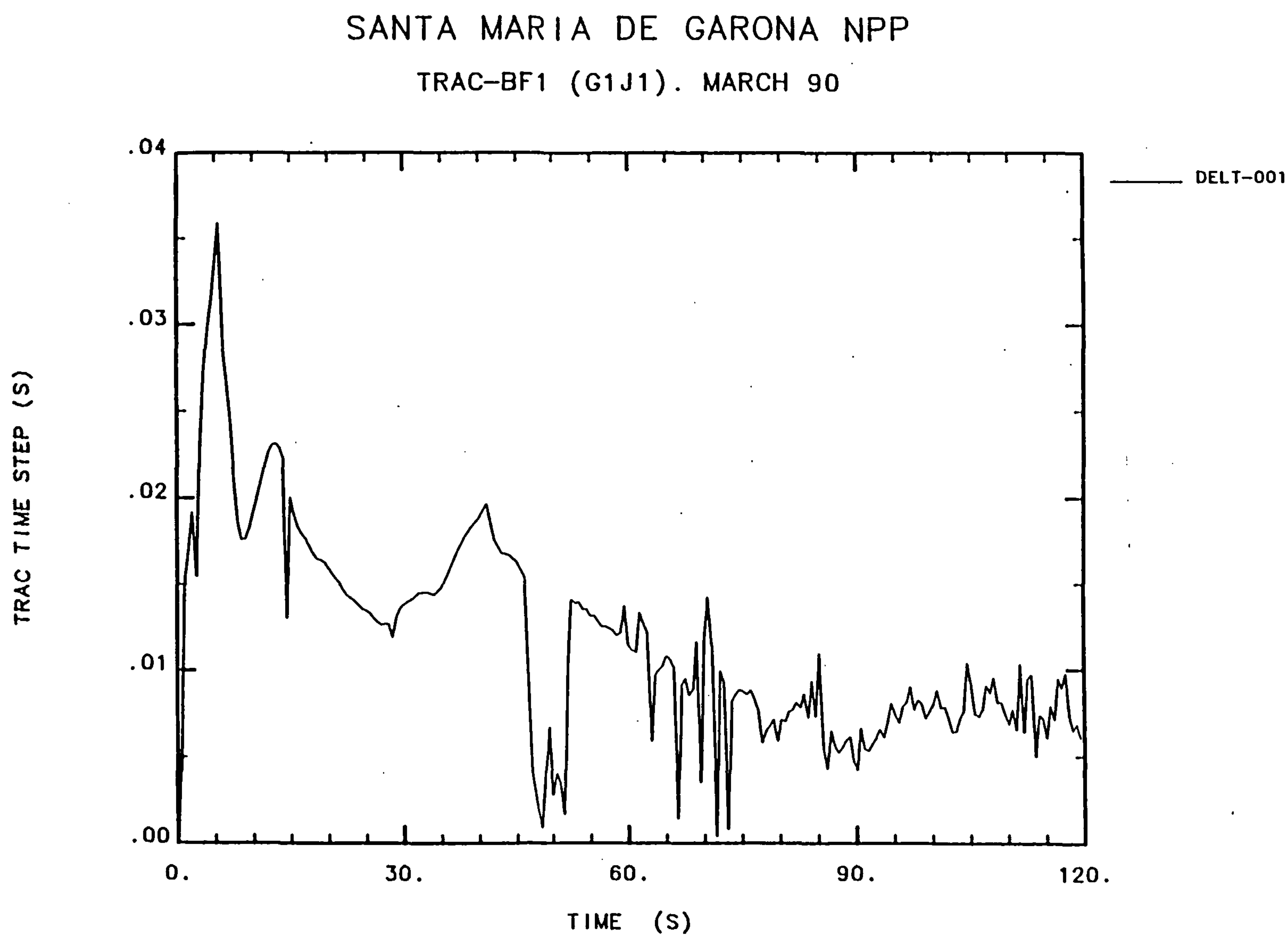


Fig. IV-20

RECIRCULATION PUMP SUCTION LARGE BREAK LOCA



RECIRCULATION PUMP SUCTION LARGE BREAK LOCA

Fig. V-1

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

U.S. NUCLEAR REGULATORY COMMISSION

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A best estimate analysis of a recirculation suction pipe large break loss of coolant accident analysis for Santa Maria De Garona nuclear power plant using TRAC-BF1 code is presented.

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