NUREG/IA-0171

International Agreement Report

Simulation of LOCA 6” and LOCA 2” Transients in the RHR of a PWR Under Low Power Conditions Using RELAP5/MOD3.2

Prepared by
S. Martorell, S. Carlos, G. Verdú, V. Serradell, J. Rodenas, R. Miró ,C. Llopis*

*C.N. Vandellos II
Travessera de les Corts, 55, Lateral
08028 Barcelona

Department of Chemical and Nuclear Engineering
Polytechnic University of Valencia
Camino de Vera, 14
Spain

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

February 2000

Prepared as part of
The Agreement on Research Participation and Technical Exchange
under the International Code Application and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy,* of the *Code of Federal Regulations,* may be purchased from one of the following sources:

1. The Superintendent of Documents
   U.S. Government Printing Office
   P.O. Box 37082
   Washington, DC 20402–9328
   <http://www.access.gpo.gov/su_docs>
   202–512–1800
2. The National Technical Information Service
   Springfield, VA 22161–0002
   <http://www.ntis.gov>
   1–800–553–6847 or locally 703–605–6000

The NUREG series comprises (1) brochures (NUREG/BR-XXXX), (2) proceedings of conferences (NUREG/CP-XXXX), (3) reports resulting from international agreements (NUREG/IA-XXXX), (4) technical and administrative reports and books [(NUREG-XXXX) or (NUREG/CR-XXXX)], and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Office Directors' decisions under Section 2.206 of NRC's regulations (NUREG-XXXX).

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: Office of the Chief Information Officer
Reproduction and Distribution
Services Section
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001
E-mail: <DISTRIBUTION@nrc.gov>
Facsimile: 301–415–2289

A portion of NRC regulatory and technical information is available at NRC's World Wide Web site:

<http://www.nrc.gov>

After January 1, 2000, the public may electronically access NUREG-series publications and other NRC records in NRC's Agencywide Document Access and Management System (ADAMS), through the Public Electronic Reading Room (PERR), link <http://www.nrc.gov/NRC/ADAMS/index.html>.

Publicly released documents include, to name a few, NUREG-series reports; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigation reports; licensee event reports; and Commission papers and their attachments.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852–2738. These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
<http://www.ansi.org>
212–642–4900

NUREG/IA-0171

International Agreement Report

Simulation of LOCA 6” and LOCA 2” Transients in the RHR of a PWR Under Low Power Conditions Using RELAP5/MOD3.2

Prepared by
S. Martorell, S. Carlos, G. Verdú, V. Serradell, J. Rodenas, R. Miró ,C. Llopis*

*C.N. Vandellos II
Travessera de les Corts, 55, Lateral
08028 Barcelona

Department of Chemical and Nuclear Engineering
Polytechnic University of Valencia
Camino de Vera, 14
Spain

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

February 2000

Prepared as part of
The Agreement on Research Participation and Technical Exchange
under the International Code Application and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

ABSTRACT

The present study consists of the simulation of two loss of coolant accidents, LOCA 6" and LOCA 2", in one of the residual heat removal system (RHR) lines outside the containment, using the thermal-hydraulic code RELAP5/MOD3.2.

Both transients have been simulated on a typical three loop, Westinghouse design, pressurized water reactor plant working under shutdown conditions.

The study was focused on the simulation of the most important thermal-hydraulic parameters in order to check the validity of the success criteria assumed in the plant probabilistic safety analysis (PSA) under shutdown conditions. Also to analyze the code capability for simulating shutdown conditions was of interest in this study.

As a result of this study, it can be concluded that the main thermal-hydraulic plant features follow what is foreseen in the plant PSA, although it can not be assured that the values reached are the correct ones due to the lack of experimental data.

TABLE OF CONTENTS

LIST OF TABLES:

LIST OF FIGURES:

EXECUTIVE SUMMARY

In this paper the simulation of two loss of coolant accidents of 6" and 2" of diameter in one of the RHR lines of a three loop pressurized water reactor working under shutdown conditions using the RELAP5/3.2 thermal-hydraulic code are presented. Although this code was primarily developed to study loss-of-coolant (LOCA) accidents with the system working at full power, what means high pressure, this study is focused on studying those transients with the plant working under low power or shutdown conditions, what means low pressure.

The residual heat removal system is part of the emergency core cooling system (ECCS) in a nuclear reactor, which is responsible for removing the residual heat during low power and shutdown conditions. As the computer code RELAP5/MOD3.2 has already been validated for LOCA transients at full power conditions, it seemed of interest to analyze the capability and limitations of the code to simulate low power and shutdown conditions. Particularly, in order to investigate if RELAP5/MOD3.2 could reproduce the physical phenomena involved in a LOCA transient in one of the RHR lines when the plant works at low power or shutdown conditions.

The main objectives of this study are the verification of the success criteria foreseen in the plant probabilistic safety analysis (PSA), and the evaluation of the RELAP5/MOD3.2 code capability for simulating transients under the established conditions. And also, to investigate if RELAP5/MOD3.2 could reproduce the physical phenomena involved in a LOCA transient in one of the RHR lines when the plant works at low power or shutdown conditions.

The plant chosen for simulating the transient was Vandellos II, which is a three loop pressurized water reactor plant, designed by Westinghouse, of 2775 Mwt of nominal thermal power. The plant is initially working in Mode 5 with the main reactor coolant system (RCS) average temperature lower than 93.3°C, the main reactor coolant system (RCS) pressure close to 24 kg/cm^2, a pressurizer level of 25% and the steam generators in wet layout. This situation is known as cold standby.

The first transient simulated is a LOCA 6" in one of the lines of the RHR system outside the containment. It is supposed that inventory recovery is successful, which takes place by the charging lines through the normal charging way, being the maximum available time for human action 600 sec. after the break is initiated.

The second transient consists of a LOCA 2" in one of the lines of the RHR system. It is supposed that the inventory recovery by the charging pumps fails, both through the normal charging way and through the safety injection. Nevertheless, the break can be isolated; having 600 sec. available after the break is initiated to take this action. The residual heat is removed using a steam generator inventory as the final heat sink, without being necessary neither the main nor the auxiliary feedwater systems.

Neither for LOCA 6" nor for LOCA 2" transient any damage to the core is supposed in the plant PSA. Nevertheless, for LOCA 6" inventory of the refueling water storage tank needs to be recovered in order to assure long-term injection. For LOCA 2" the plant is expected to tend to a stable situation.

With regard to the initial conditions, in the original plant nodalization at full power conditions, some changes were realized to model the RHRS in order to allow recirculation mode, as it is demanded in the initial conditions.

From both transients developments we have obtained the evolutions of the most interesting thermal-hydraulic variables. The calculations were run in a CONVEX SPP 1000 owned by the University Polytechnic of Valencia. The CPU time consumed in both transient simulations has been considerable, due to the time steps needed for achieving a correct transient simulation, and to the large periods of time that had to be simulated in order to appreciate changes in the evolutions of the most important thermal-hydraulic parameters in order to analyze the events that take place.

Once the study was finished we concluded that the results of both transient simulations confirm, satisfactorily, the success criteria foreseen in the plant PSA, although experimental data are needed to assure the values obtained are the correct ones and whether the code works properly under these conditions.

1. INTRODUCTION

The RELAP5 is a thermal-hydraulic code widely used for studying transients in pressurized water reactors (PWR). This code was primarily developed to study loss-of-coolant (LOCA) accidents with the system working at full power, what means high pressure.

Here the study is focused on studying a loss-of-coolant accidents, but with different conditions from the above mentioned. In this case, the plant is working under low power or shutdown conditions, what means low pressure, with the residual heat removal (RHR) system working in recirculation mode.

The residual heat removal system is part of the emergency core cooling system (ECCS) in a nuclear reactor, which is required for removing the residual heat during low power and shutdown conditions.

It is of great interest to study those transients that cause a loss of the RHR system under these working conditions, and also to check the alternative ways foreseen to evacuate the residual heat generated.

As the computer code RELAP5/MOD3.2 version has already been validated for LOCA transients at full power conditions, it seemed of interest to analyze the capability and limitations of the code to simulate low power and shutdown conditions. Particularly, in order to investigate if RELAP5/MOD3.2 could reproduce the physical phenomena involved in a LOCA transient in one of the RHR lines when the plant works at low power or shutdown conditions.

The ultimate objective of this study is to validate, considering the results obtained and the limitations encountered, the success criteria established in the probabilistic safety analysis (PSA) of the plant, for such operational modes.

This study is part of the code applications and maintenance program (CAMP), and is focused on studying some transients described in the PSA for low power and low pressure. It has been conducted by the Chemical and Nuclear Engineering Department of the Polytechnic University of Valencia (UPV), in collaboration with the Consejo de Seguridad Nuclear (CSN) of Spain.

This document is organized as follows. In chapter two a brief plant description is presented, highlighting its most important features. The descriptions of the two transients simulated, LOCA 6" and LOCA 2", are also presented in this chapter. The input file used for reaching the transient initial conditions is explained in chapter three, together with other specific modifications needed in each transient development. Chapter four contains the results of both simulations, showing the evolutions versus time of some thermal-hydraulic variables considered as important. It also contains the discussions of the additional sensibility studies performed, and some recommendations on the aspects that should be subject of a further study. Chapter five gives information about the time steps utilized in both simulations and the time consumed in the transient developments. The conclusions derived from the whole study and the most important recommendations for other users are discussed in chapter six. References needed in the elaboration of this study are detailed in chapter seven. Finally, the tables and figures referenced in the text are shown at the end of the report.

2. PLANT AND TRANSIENT DESCRIPTION

2.1. PLANT DESCRIPTION

The plant chosen for simulating the transient was Vandellos II, which is a three loop pressurized water reactor plant, designed by Westinghouse, of 2775 Mwt of nominal thermal power. The plant is equipped with three U-tube steam generators without preheaters, and uses the seawater as final heat sink. The reactor vessel is cold head type. In table 1, the most important plant features are presented.

Figure 1 shows the nodalization diagram used in RELAP5/MOD3.2 for the plant under low power and low-pressure conditions. In this diagram there are three loops, with a steam generator each, a reactor vessel, and a pressurizer. The RHRS is also modeled, which includes the high-pressure coolant injection system (HPCIS) and the low-pressure coolant injection system (LPCIS), adapted for recirculation.

The RHR system consists of two lines, A and B, with their low pressure coolant injection part connected with loops 1 and 2 of the main reactor coolant system (RCS) and with the refueling water storage tank (RWST). When the plant is in Mode 5, the RHR system extracts water from the hot legs of loops 1 and 2, which is recirculated by the RHR pumps towards the heat exchangers, to the cold legs of the three loops. In Mode 5, under the conditions required for initiating the transients, the RHR system recirculates a total mass flow of 300 kg/sec., using both pumps.

2.2 TRANSIENTS DESCRIPTION

Both studies consist of simulating a break, no guillotine, of 6" and 2" diameters respectively, in one of the RHR system lines of a PWR outside the containment, with the plant working in Mode 5. This situation is known as cold standby, with the following characteristics:

- Main reactor coolant system (RCS) average temperature lower than 93.3°C.
- Main reactor coolant system (RCS) pressure close to 24 kg/cm^2.
- Pressurizer level 25%.
- Steam generators in wet layout.

In order to obtain the above mentioned values, it was necessary to modify the level and pressure controls, adjusting them to the new working conditions. The initial conditions for the most important variables are presented in table 2.

The procedure applied for LOCA 6" and LOCA 2" transient scenarios is the ARG-2 Westinghouse Shutdown LOCA, which main actions are:

- Symptom or entry condition: decrease in pressurizer level.
- RHR pumps trip.
- Chemical and volumetric control system (CVC) discharge isolation.
- Increase of charging flow or realignment of safety injection.
- Break isolation attempt.
- Refueling water storage tank (RWST) recovery.

The transient developments are different depending on the break produced, due to fact that the sequence of the events of the success criteria foreseen in the plant PSA under these conditions is different in each case.

Thus, when a LOCA 6" in the RHR system outside the containment is produced, it is supposed that inventory recovery is successful, which takes place by the charging lines through the normal charging way, being the maximum available time for human action 600 sec. after the break is initiated. It is also supposed that the break can not be isolated and that the refueling water storage tank needs to be recovered in order to assure long term injection.

After a LOCA 2", it is supposed that the inventory recovery with the charging pumps fails, both through the normal charging way and through the safety injection. Nevertheless, the break can be isolated; having 600 sec. available after the break is initiated to take this action. The residual heat is removed using a steam generator inventory as the final heat sink, without being necessary neither the main nor the auxiliary feedwater systems.

3. CODE INPUT AND MODEL DESCRIPTION.

3.1. SYSTEM DESCRIPTION

The simulations have been run on a typical three loop pressurized water reactor full scaled model, figure 1, in which the low and high pressure injection systems are also modeled, and the latter has been modified to allow recirculation.

It has been necessary to adapt the plant nodalization using RELAP5/MOD3.2 from initial available full power conditions, to low power conditions, as this was the previous state of the plant before starting the transients.

An important nodalization change has been the inclusion of the RHR system modification, so it could work in recirculation mode, since it is demanded in the initial conditions. By this reason, some volumes have been added, simulating the pipes that connect the hot legs of loops 1 and 2 of the primary system with the lines A (vols. 301, 303) and B (vols. 321, 323) of the low pressure injection system respectively. Among these added volumes are the RHR heat exchangers (vols. 308, 309, 328, 329), which are able to remove all the residual power generated, providing the appropriated temperature to the water being recirculated to the loops.

From the initial input deck for full power conditions, the volumes that simulate the turbine, the steam dump, the Main Steam Relieve Valves (MSRV) and the accumulators have been eliminated, since they are not used in the transients simulated, nor in other possible transients in low power conditions.

The most important change in the reactor coolant system (RCS) was made in the pumps (vols. 118, 148, 178), for adjusting them to the new working conditions, as the characteristic curve was not prepared for simulating them when they are stopped. The solution was found by modifying the suction and discharge loss coefficients adjusting them until the mass flow rate through the RCS was the same as the measured in plant.

At the secondary side of the steam generators the water injection through the main and auxiliary feedwater systems has been deactivated, although it has not been removed to allow that they could be required in the simulation of future transients under low power conditions.

In order to obtain reliable results from the transient developments, it is needed to simulate the break as accurately as possible. It has been rather complicated in this case since experimental data were not available to compare with. The breaks have been modeled as a TRIPVALVE (vol. 340) that connects one of the RHR lines with a TMDPVOL (vol. 334), which simulates the atmosphere. The valve diameters that simulates the breaks corresponds to 6" or 2" break respectively in a 12" sch 40 pipe.

One of the most important characteristics of the valve is to adopt the abrupt area change model, using as discharge coefficients 1, 1.8, and 0.8 for subcooled, two phases and superheated regimen respectively, as it is suggested in the literature [3]. These values are more conservative than the ones suggested in the RELAP5/MOD3.2 code manual [12] [13], what will suppose an increase in the mass flow through the break. In fact, the change of the discharge coefficients will provide a more realistic value for the mass flow rate since the code underpredicts it for low quality flow, just as it occurs in our transient.

The choking model also presents several problems in simulating mass flows rates through breaks under low-pressure conditions. The root cause of this misfunction seems to be the inability of the code to unchoke a junction even though its velocity is below the critical one for the flow conditions [5]. This happens when nonhomogeneous flow is activated, since this model produces unrealistic low mass fluxes at low pressures. The current recommendation given in the RELAP5/MOD3.2 code manual [12] is to use homogeneous and choked flow for break junctions and other connections to the atmosphere, what will produce mass fluxes close to the homogeneous equilibrium critical flow model. For internal junctions it is suggested to invoke nonhomogeneous model with the choking model turned off.

Finally, we have to mention that the valve that simulates the break is governed by a trip. It is initially closed and opens when the trip is activated, in both transients time is the variable that activates the trip.

For LOCA 6" transient it was necessary to model the safety injection through the normal charging way (vols. 461, 462). Before the break is produced, a constant water mass flow rate is injected through the CVC system (vols. 455, 460). Once the transient is initiated this valve closes and, when the control variable that governs the inventory recovery is activated, the water begins to be injected through the time dependent junction. The mass flow rate injected depends on the RCS pressure.

3.2. INITIAL CONDITIONS.

A new input deck with the above modifications was built to simulate the plant state previous to both transients. It consisted of simulating the plant in Mode 5 with the following features:

- RCS average temperature lower than 93.3°C.
- RCS pressure about 24 kg/cm^2.
- Steam generators in wet layout.

Other data measured in plant were added to the requirements above exposed. Table 2 shows a list of the most important parameters calculated with RELAP5/MOD3.2 code and their comparison with the values required or measured in plant.

Once the initial conditions were reached and having checked that the plant was in a stable state, two new input decks were built considering the values of the variables at this point to perform the transient simulations.

3.3 LOCA 6" SIMULATION

In the LOCA 6" transient, it was necessary to activate the countercurrent flow limitation model due to the low RCS mass flow that leads to a misfunction of the code. This model was invoked only in the internal junctions where it was likely to occur, following the suggestions of RELAP5/MOD3.2 code manual [12] [13]. In this case it was activated in the core and core by-pass, the downcomer vessel, steam generator U-tubes, hot legs and the entrance of the vessel.

In all junctions the Wallis CCFL correlation was used by default. The values for the Wallis correlation parameters, m and c, were taken from the literature [4] [12], which have been obtained from the experimental data of other experiments with similar working conditions to our situation. Possibly, the correlation used is not the proper one, since it seems not to work properly at low-pressure situations [6]

After all the changes were made, the LOCA 6" transient was simulated following the sequence showed in figure 2, which describes the scenario foreseen in the plant PSA under these conditions.

Firstly, the plant has been maintained at the initial conditions for 50 sec., to assure that the plant was in the initial state. Then, the transient is initiated by opening the valve that simulates the break in the RHR outside the containment. Immediately after the break opens there is a fast decrease in the pressurizer level, which causes the activation of the trips that isolate the CVC system charge and discharge, stopping the RHR pumps isolating the RHR heat exchangers through their by-pass. Safety injection starts injecting water 600 sec after the break was produced, through the normal charging way. The break can not be isolated during the whole transient.

3.4. LOCA 2" SIMULATION.

LOCA 2" transient was simulated at indicated in figure 3, which agrees with the sequence of events foreseen in the plant PSA under such conditions.

In this case, the initial conditions were the same as for LOCA 6" transient, simulating 50 sec. before producing a 2" break in one of the RHR lines, outside the containment. Then, the break opens and the pressurizer level starts to decrease, but not so fast as in the previous case. When the pressurizer reaches a certain level the isolation of the RHR heat exchangers and the isolation of the CVC system charge and discharge trips are activated, and the RHR pumps are stopped. For this transient, it is supposed that the residual power generated is removed by the inventory of one steam generator, so the other two were isolated. In addition, no safety injection is allowed at any time.

4. RESULTS

From the transients developments the evolution of plant most important thermal-hydraulic variables have been obtained. Some of the most interesting evolutions are presented in this chapter.

4.1. LOCA 6" TRANSIENT RESULTS

For LOCA 6" transient 7300 sec. were simulated. During all this transient the core remains covered, as it can be seen in figure 4. This means that no damage to the core should be expected, as it is foreseen in the plant PSA.

The most critical temperatures are the core outlet and clad temperatures, shown in figure 5. In this figure it can observed that all the temperatures have a peak just when the break is produced, to decrease immediately after. Although safety injection starts at 650 sec., it has no effect on the core and clad temperatures until 1000 sec. when they begin to decrease. At the end of the transient the core temperatures tend to a stable value, which is approximately the temperature of the water injected. This result is also foreseen in the plant PSA, where it is supposed that the system will tend to a stable situation as long as water reservoir in the reactor water storage tank has to be recovered in a long term.

In figure 6, the mass flow rate through the break and the safety injection mass flow rate are presented. When the valve opens, there is a peak in the mass flow rate through the break that reaches 700 kg/sec. After this peak, there is a period of time in which the flow presents a smooth decrease. At about 900 sec. there is a sudden fall of the mass flow rate and it starts to fluctuate near the safety injection value. The safety injection mass flow rate has constant value during all the transient, since it depends on the primary system pressure change and, as can be seen in figure 7, the pressure remains almost at a constant value during all the time simulated after the break.

In figures 8 and 9 the hot and cold leg mass flow rates for each loop are represented. In both figures it can be observed that there are important mass flow rate fluctuations, not only when the break is produced, but during all the simulation.

In figure 10 the total mass inside the three primary loops is presented. It can be seen that the mass inside the loops present a sharp decrease at the beginning of the transient. At about 900 sec. the mass starts to recover due to the effect of the safety injection, which finally tend to keep stable.

4.2. LOCA 2" TRANSIENT RESULTS.

For LOCA 2" transient 300000 seconds were simulated. Under the conditions in which the plant is working the residual power is so low that a large period of time is needed to observe significant changes in the thermal-hydraulic variables. As it is foreseen in the plant PSA, no damage to the core should be expected, as it can be seen in figure 11 where the core level remains covered during all the transient, although there are some fluctuations.

In figure 12 the core outlet and clad temperatures are represented. All the temperatures rise from the beginning of the simulation until about 200000 seconds. At this time they remain at a constant value until the end of the transient simulation.

Pressures in the primary and secondary side of the steam generators, see figures 13 and 14 respectively, present the same shape as the temperatures. In this case, when the break is produced the RCS pressure, measured in the pressurizer, rises up until 200000 seconds. At this time it remains stable at 8.E+6 Pa, which is nearly the same pressure value as for the steam generator secondary side. It is also at this time when the level of the steam generator 2 starts to decrease, due to the relief valve in this steam generator opens, what means that this steam generator is removing the residual heat. Figure 15 presents the evolution of the level of the steam generator in which it can be seen that even at the end of the transient there is still liquid inside the secondary side.

As in the previous case, the hot and cold legs mass flow rates present high oscillations, figures 16 and 17, which only disappear when the time simulated is extremely high, and the plant tends to be stable.

Finally, figure 18 shows the pressurizer level evolution. When the valve, which simulates the break, opens the pressurizer level drops, and after the break is isolated it begins to recover until at 160000 sec., when the pressurizer is completely full.

4.3 SENSITIVITY STUDIES

Several sensitivity studies have been performed. In both transients the break was modeled as it is suggested in the RELAP5/MOD3.2 code manual [13], using the abrupt area change model. But, as it is not well known the actual mass flow rate through the break, it has been impossible to determine the suitable values for the discharge coefficients. The suggested values were tried, but in this case the flow through the break was extremely high in both transients, which led up to incoherent results.

The solution adopted was to model the break supposing crossflow in the valve with regard to the main flow direction in the RHR pipe. That is, the connection from the RHR line with the valve is supposed as a crossflow, and the connection between the valve and the time dependent volume that simulates the atmosphere is defined as a normal junction with an abrupt area change. In this case the results for the mass flow rate seemed to be more reliable than in the previous case.

In LOCA 2" transient two more sensitivity studies were performed. The first consisted on changing the maximum time step defined in the input deck, since in some transients the influence of time steps used in the calculations provoke differences in the values reached by the thermal-hydraulic parameters. In our case a maximum time step of 0.5 sec. was selected because the variables change slowly. A shorter simulation was run using 0.05 sec as maximum time step without observing any difference between both simulations.

Also in LOCA 2" a calculation was run activating the CCFL model, as it was made for LOCA 6" simulation, but in this transient the results obtained did not differ from those calculated without the possibility of invoking the CCFL model.

5. RUN STATISTICS

The calculations have been made in a CONVEX SPP 1000 owned by the UPV, using SPP-UX-3.1 as operating system and For77-HP as compiler. Table 3 presents the information about time steps used in each simulation and the CPU time consumed.

Figure 19 shows the CPU values versus time simulated for LOCA 6", and figure 20 presents the time step versus time simulated for the same transient.

As it was done for LOCA 6", figure 21 represents CPU time versus simulation time for LOCA 2" and in figure 22 the time steps versus simulation time.

6. CONCLUSIONS AND RECOMMENDATIONS

It has been not possible to compare the values of the results obtained in the simulation with experimental data, as no experiment is available under these conditions. Thus, although the thermal-hydraulic variable evolutions agree with the success criteria foreseen in the plant PSA, it can not be assured that the values they reach are the actual ones due to the lack of such experimental results.

Regarding the models implemented in the code, it was noted the misfunction of the code in simulating noncondensables presence, it was specially important in the LOCA 6" simulation due to the amount of noncondensable species that comes into the primary system through the break.

Thus, the presence of noncondensables produces an important increase of the time needed for calculations, because it was necessary to reduce the maximum time step required for the code to obtain a coherent solution. The introduction of noncondensables also produces a decrease in the transmission heat coefficients, specially in low flow conditions, just as it occurs in these studies.

Also, due to the lack of experimental data it can not be assured that the values used for the constants of the CCFL model are the most suitable ones, since the values encountered in others studies with similar working conditions were adopted. We can not either be completely sure that the Wallis correlation used was the most appropriated for our studies. In this case, a further study in which a comparison of the results obtained with different correlations seems to be necessary, and this study also requires experimental data in order to compare the results.

In the abrupt area change model, the values of the discharge coefficients used for the subcooled, two phases and superheated regimens, were taken from the literature for similar plant conditions, and although the results obtained seem to be quite reliable, nothing else can be assured.

So, the most important impediment for concluding that the results obtained with RELAP5/MOD3.2 are completely corrects is the lack of experimental data needed to assess the code capability under low power and low pressure conditions. It would be interesting the performance of experiments for these conditions to develop a more detailed study on the code capability.

7. REFERENCES

[1] Yassin A. Hassan and Sibashis S. Banerjee. "RELAP5/MOD3 simulation of the loss of the residual heat removal system during midloop operation experiment conducted at the ROSA-IV large scale test facility". Nuclear technology Vol. 108. November 1994.

[2] Sandra M. Sloan and Yassin Hassan. "Study of RELAP5/MOD2 and RELAP5/MOD3 predictions of a small break loss-of-.coolant accident simulation conducted at the ROSA-IV large scale test facility". Nuclear technology vol. 100. March 1995.

[3] Peter P. Cebull and Yassin A. Hassan. "Simulation of the IAEA's problem exercise small break loss-of-coolant accident using RELAP5/MOD3.1". Nuclear technology vol. 105. March 1994.

[4] Alessandro Annunziato. "Secondary-side feed-and-bleed effectiveness in pressurized water reactors". Nuclear technology vol. 105. March 1994.

[5] W.S. Yeung and J. Shirkov. "On the subcooled critical flow model in RELAP5/MOD3". Nuclear technology vol. 114. April 1997.

[6] Yuh-Ming Ferng and Chien-Hsiung Lee. "Assessment of the simulation capability of RELAP5/MOD3 compared with IIST tests for loss of the residual heat removal system during midloop operation". Nuclear technology vol. 116. October 1996.

[7] György Ézsöl, Atilla Guba et altri. "Simulation of a small cold-leg-break experiment at the PMK-2 test facility using the RELAP5 and ATHLET codes". Nuclear technology vol. 118. May 1997.

[8] Yuh-Ming Ferng and Chien-Hsiung Lee. "A comparison of RELAP5/MOD3 code with the IIST natural circulation experiments". Nuclear technology vol. 111. July 1995.

[9] Borut Mavko, Stojan Petelin, Oton Gortnar. "RELAP5 modelling of the Westinghouse model D4 steam generator". Nuclear technology vol. 101. February 1993.

[10] C Llopis, R Mendizábal, J. Pérez. "Assessment of RELAP5/MOD2 against a load rejection from 100% to 50% power in the Vandellós II nuclear power plant". ICAP assessment report. October 1991.

[11] Julián Rozalén, Carlos Llopis. "Modelación del primario y sistemas asociados de la C.N. Vandellós II con el código RELAP5/2.5". Julio 1994.

[12] "RELAP5/MOD3 code manual vol. 2: User's guide and input requirements". NUREG/CR-5535. June 1995.

[13] "RELAP5/MOD3 code manual vol. 5: User's guidelines". NUREG/CR-5535. June 1995.

[14] "RELAP5/MOD3 code manual vol. 4: Models and correlations". NUREG/CR-5535. June 1995.

Table 1: Main plant characteristics.

Thermal Reactor Power (Mwt)	2775.
Electrical Power (Mwe)	992.
Fuel	UO2
Number of assemblies	157
Number of coolant loops	3
Cladding tube material	Zircaloy 4
Absorber material	B4C + Ag-In-Cd
Reactor Operating Pressure (MPa)	15.4
Coolant Average Temperature Zero load (°K)	564.8
Coolant Average Temperature 100% load (°K)	582.3
Stem Generator	Westinghouse type F
Number of tubes in SG	5626
Total tube length (m)	98759.
Inner diameter tubes (m)	0.0156
Tube Material	Inconel
Pumps type	Westinghouse D 100
Discharge head of pumps (bar)	18.8
Design flow rate (m^3/sec))	6.156
Speed of pumps (rad/sec)	155.
Primary volume (m^3)	106.19
Pressurizer Volume (m^3)	39.65
Heating Power of the heaters rods (KW)	1400.
Maximum spray flow (kg/sec)	44.2
Steam mass flow rate at 100% (kg/sec)	1515.

Table 2: Initial values

Reference parameters	Problem Data	Obtained RELAP-5/3.2
Nuclear Power (%) - Residual Heat	0.05 (i)	0.05 (i)
RCS Pressure (MPa)	23	23
Pressurizer level (%)	25	25
RCS average temperature (°K) - loops 1/2/3	< 366.46	365.43 / 365.38 / 364.91
RCS hot leg temperature (°K) - loops1/2/3	--	366.01 / 365.98 / 364.94
RCS clod leg temperature (°K) - loops 1/2/3	--	364.85 / 364.88 / 364.89
Core outlet temperature (°K) - loops 1/2/3	--	366.02 / 366.02 / 364.94
Clad temperature (°K)	--	366.37
Core level (%)	--	6.6556
RCS mass flow (Kg/s) - loops 1/2/3	--	94.26 / 95.43 / 94.30
Primary GVs mass flow (Kg/s) - loops 1/2/3	--	-2.27 / -4.93 / -4.69
Secondary steam generator pressure (Mpa)- loops 1/2/3	--	0.081 / 0.0785 / 0.0802
Secondary GVs temperature (°K) - loops 1/2/3	--	366.15 / 366.15 / 366.15
Steam generator level (%) - loops 1/2/3	--	82.02 / 82.02 / 82.35
GVs Relief valves mass flow (Kg/s) - loops 1/2/3	--	0.0 / 0.0 / 0.0
RHR outlet pressure (MPa) - lines A/B	--	2.977 / 2.979
RHR inlet pressure (MPa) - lines A/B	--	2.50 / 2.50
RHR outlet temperature (°K) - lines A/B	--	364.87 / 364.84
RHR inlet temperature (°K) - lines A/B	--	366.11 / 366.00
RHR mass flow (Kg/s) - lines A/B	--	148.59 / 146.85
CVCS charge mass flow (Kg/s)	--	2.6
CVCS mass flow discharge (Kg/s)	--	2.6
Break mass flow (Kg/s)	--	0.0
Secondary RHR heat exchangers mass flow (Kg/s)-lines A/B	--	40.002 / 40.002

(i) Nominal power for 100% load is 2775.0 MW (1.3875 MW for 0.05%).

Table 3: Run statistics

	RT	CPU	TS	CPU/RT	C	DT	GT
LOCA 6"	4000.	13737.08	0.05	3.4343	252	119322	0.4605
	7000.	370944.41	0.001	52.9920	252	3149372	0.4674
	7200.	417045.81	0.0005	57.9230	252	3543058	0.4671
LOCA 2"	309000.	77400.	0.5	0.25	281	665346	0.4139

RT: Transient time (sec.)

CPU: Execution time (sec.)

TS: Maximum time step (sec.)

C: Total number of volumes

DT: Total number of time steps

GT: Grind time (msec.) $GT=(CPU*10^3/(C*DT))$

Figure1: Vandellos II nodalization diagram for RELAP5/MOD3.2 under low power conditions.

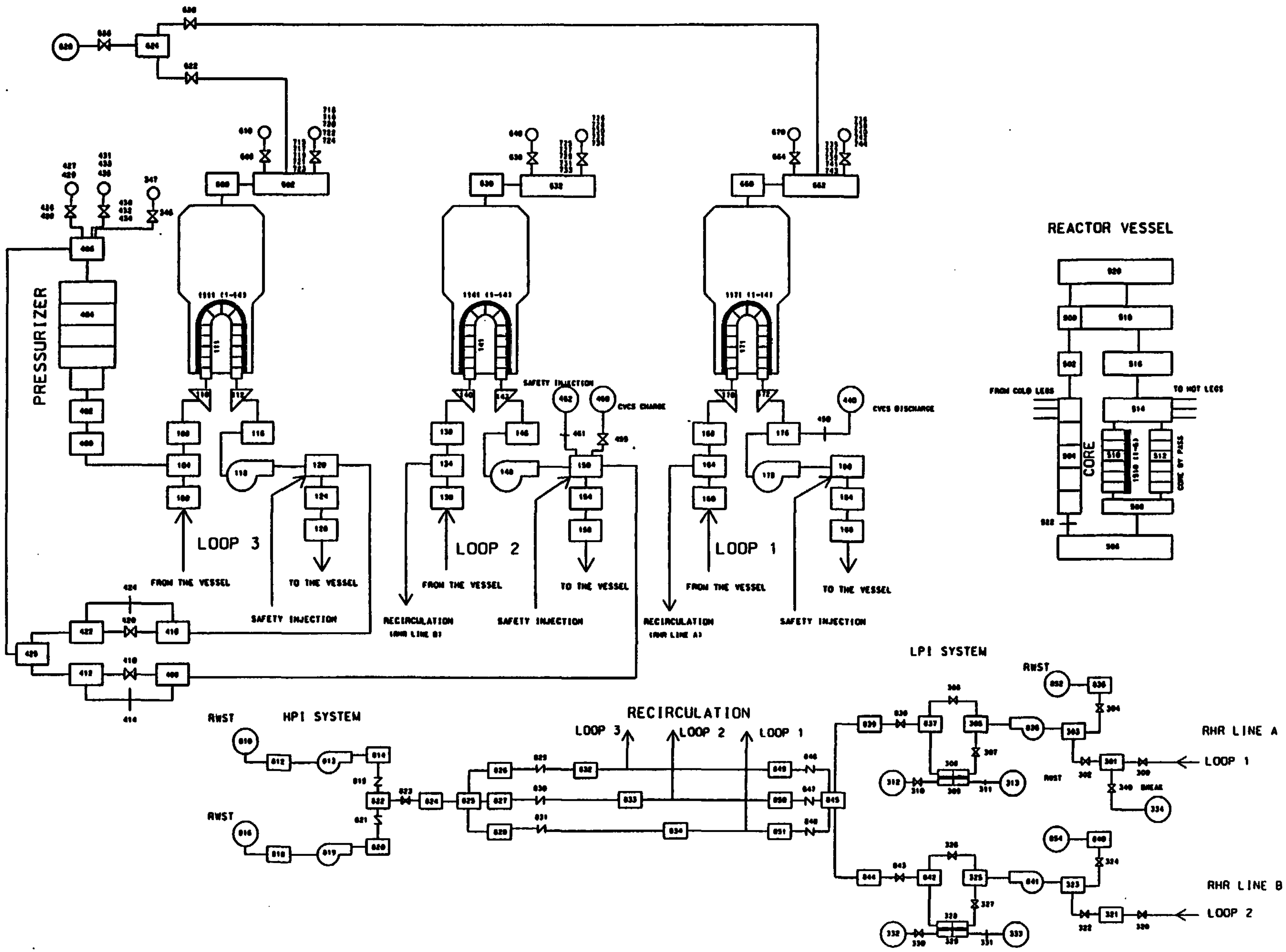


Figure 2: LOCA6" transient simulation.

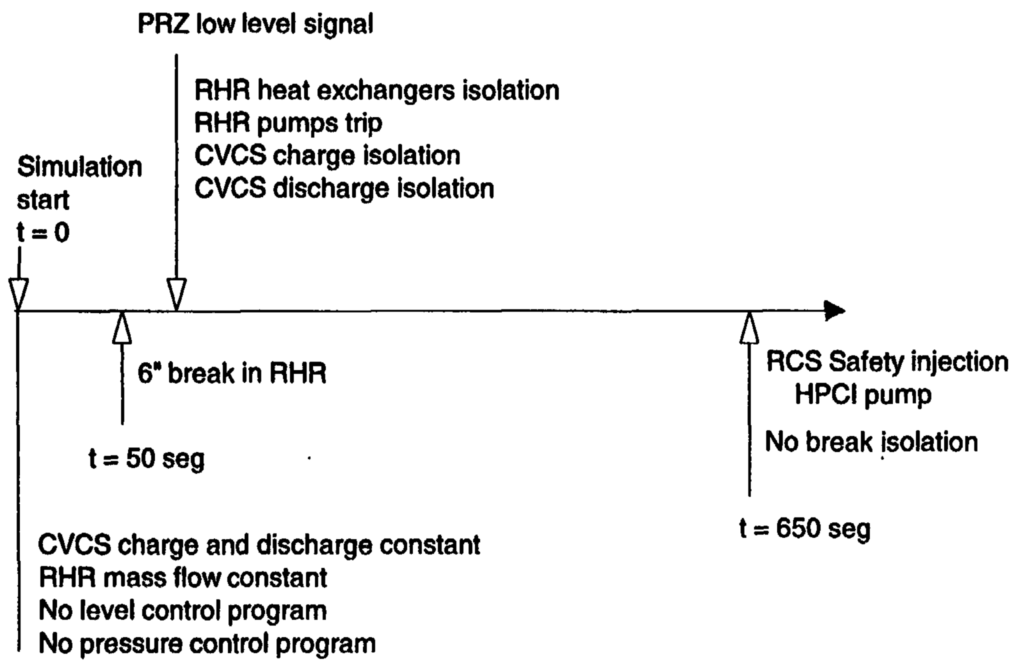
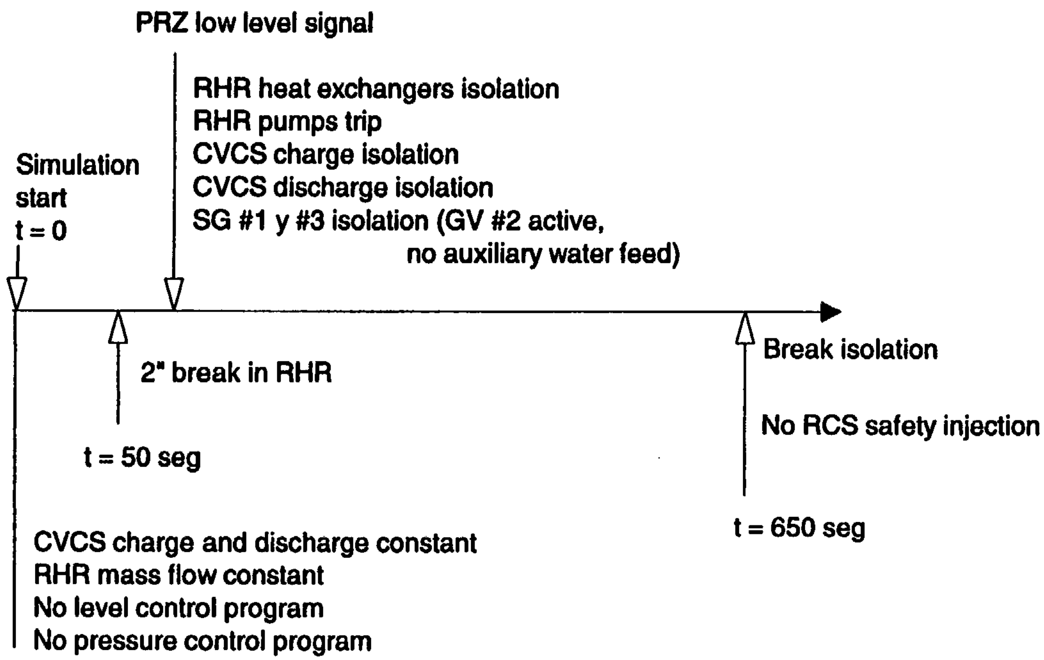
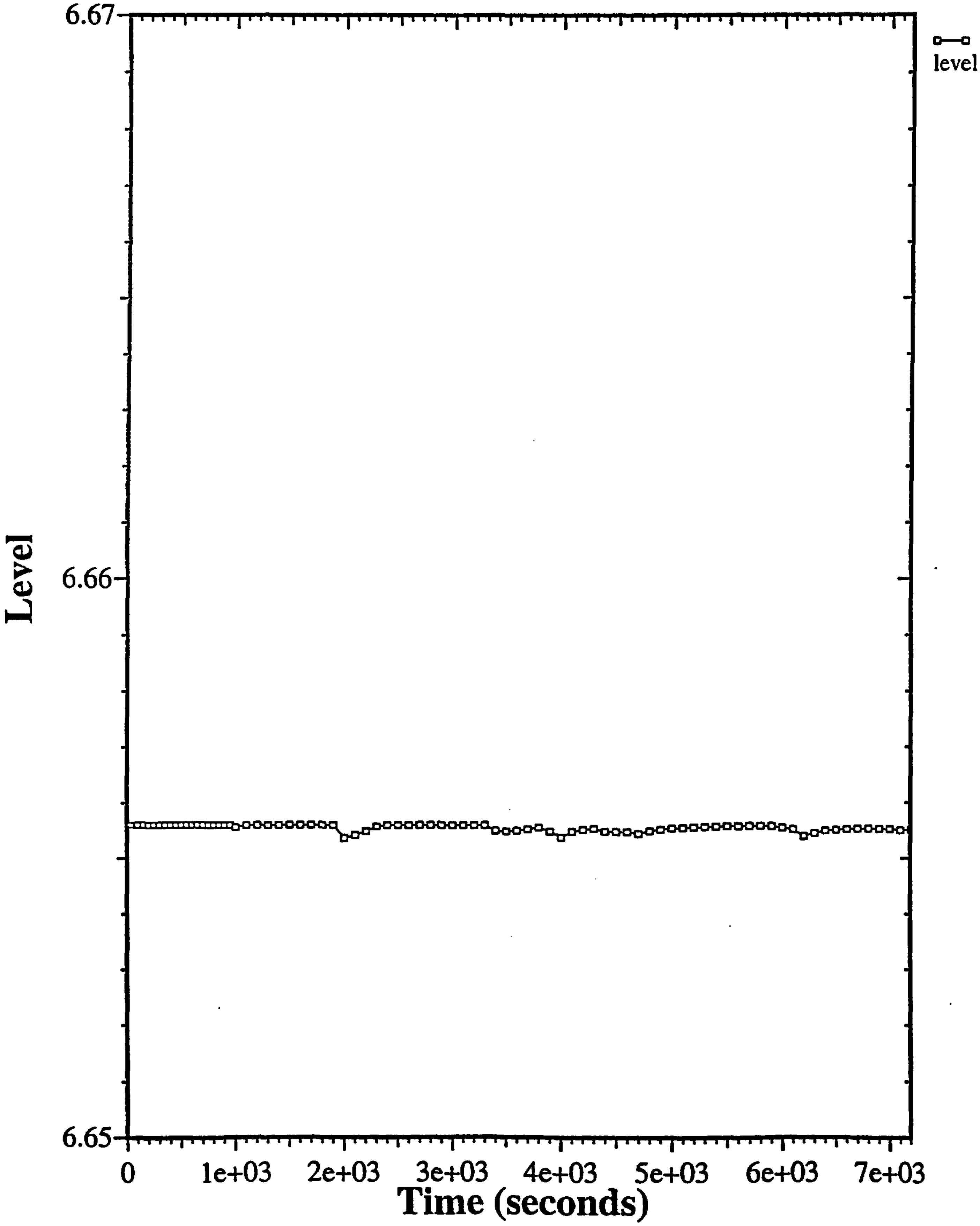


Figure 3: LOCA 2" transient simulation





6.67
6.66
6.65
Level
level
0
1e+03
2e+03
3e+03
4e+03
5e+03
6e+03
7e+03
Time (seconds)

Figure 5: Core outlet and clad temperatures for LOCA 6".

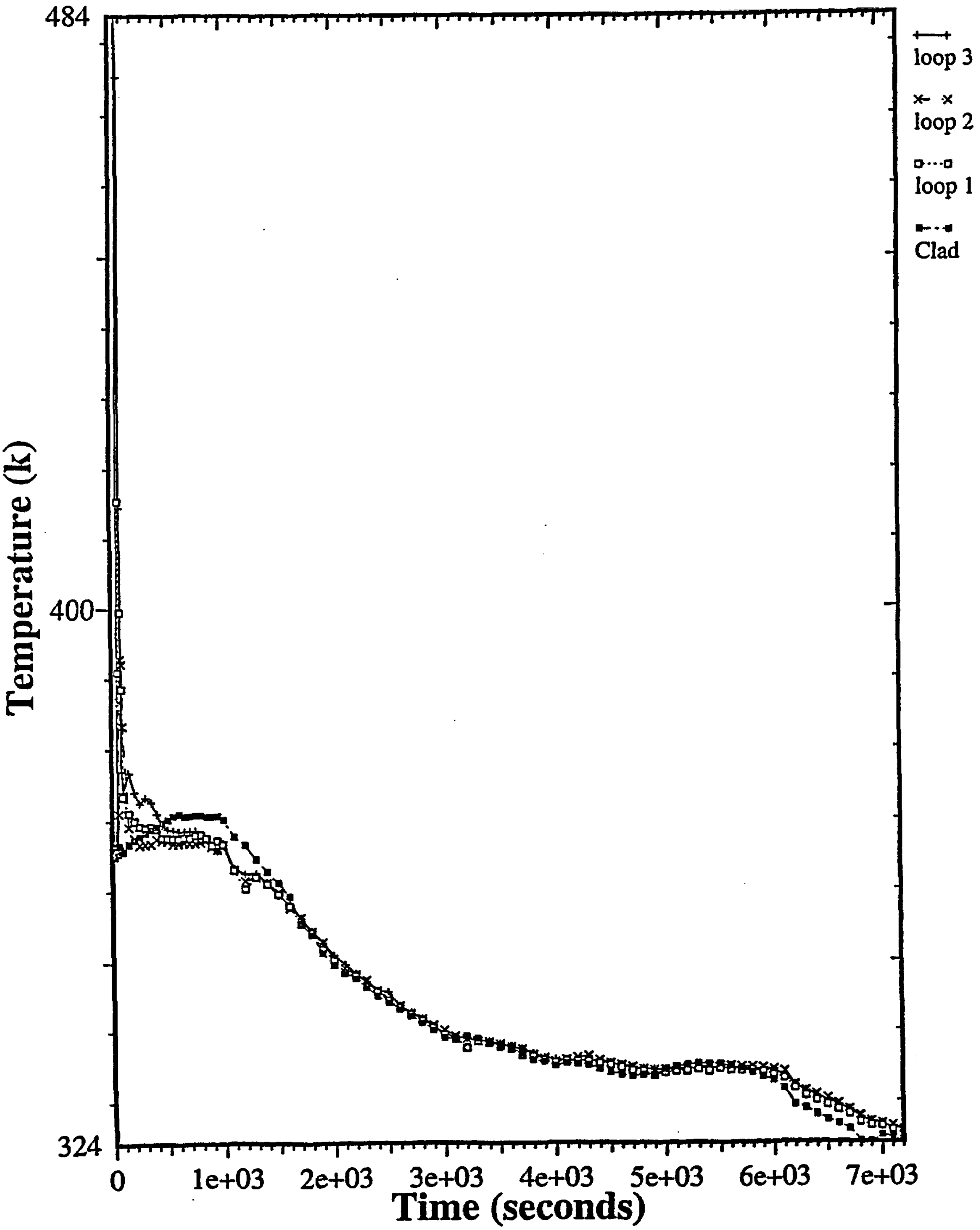


Figure 6: Break and CVC system mass flow rates for LOCA 6".

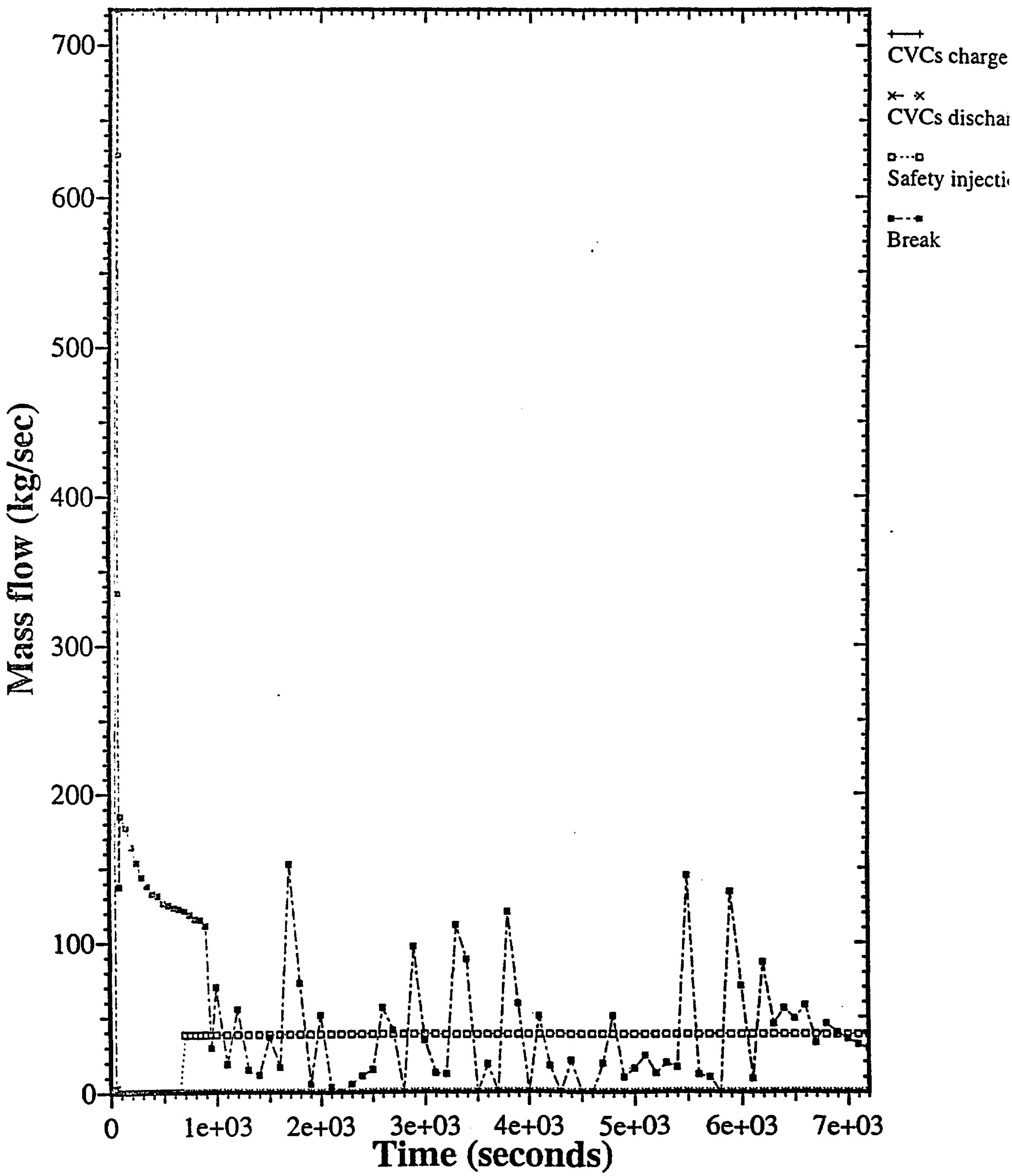


Figure 7: Reactor coolant system pressure for LOCA 6".

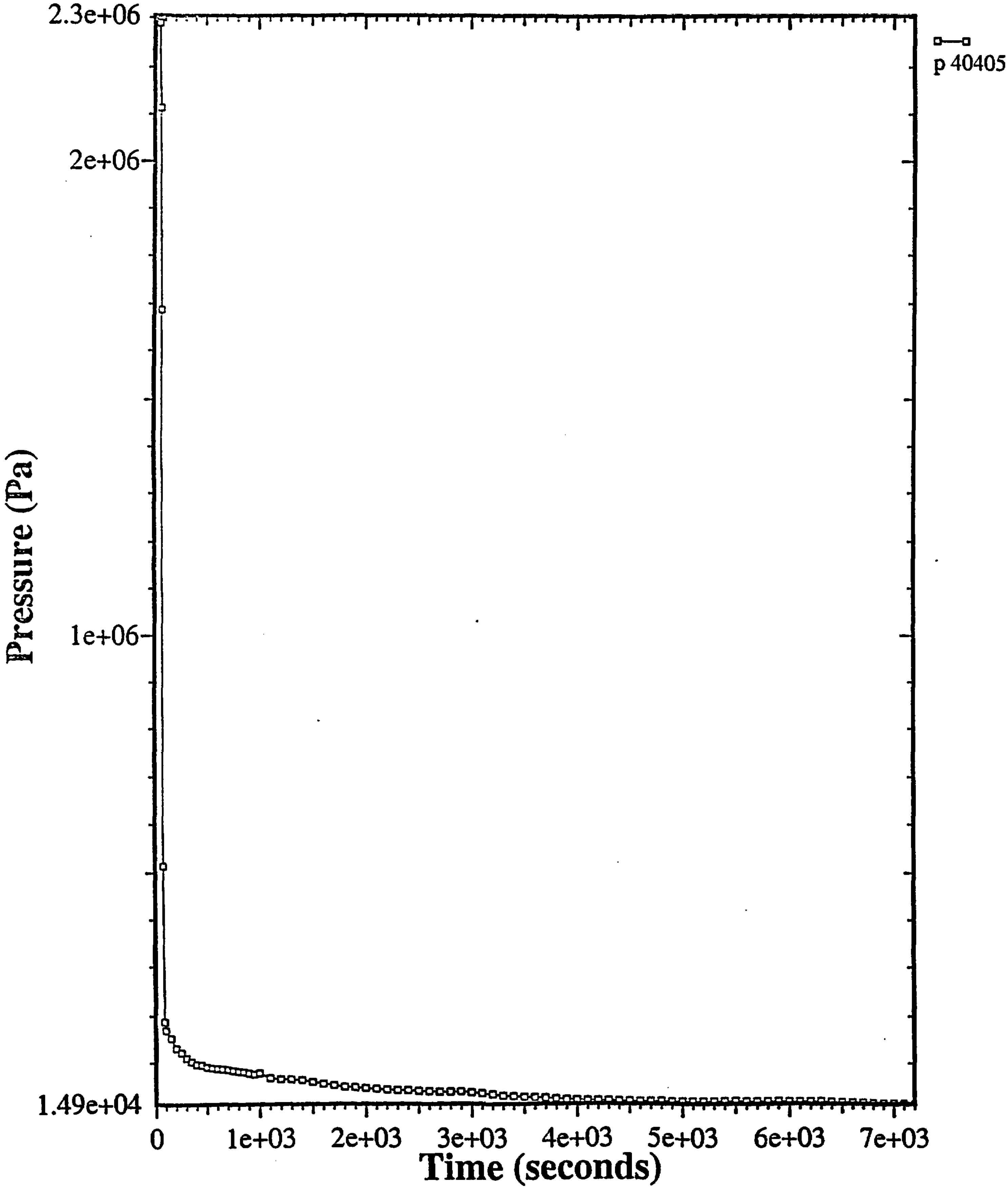


Figure 8: Hot legs mass flow rates for LOCA 6".

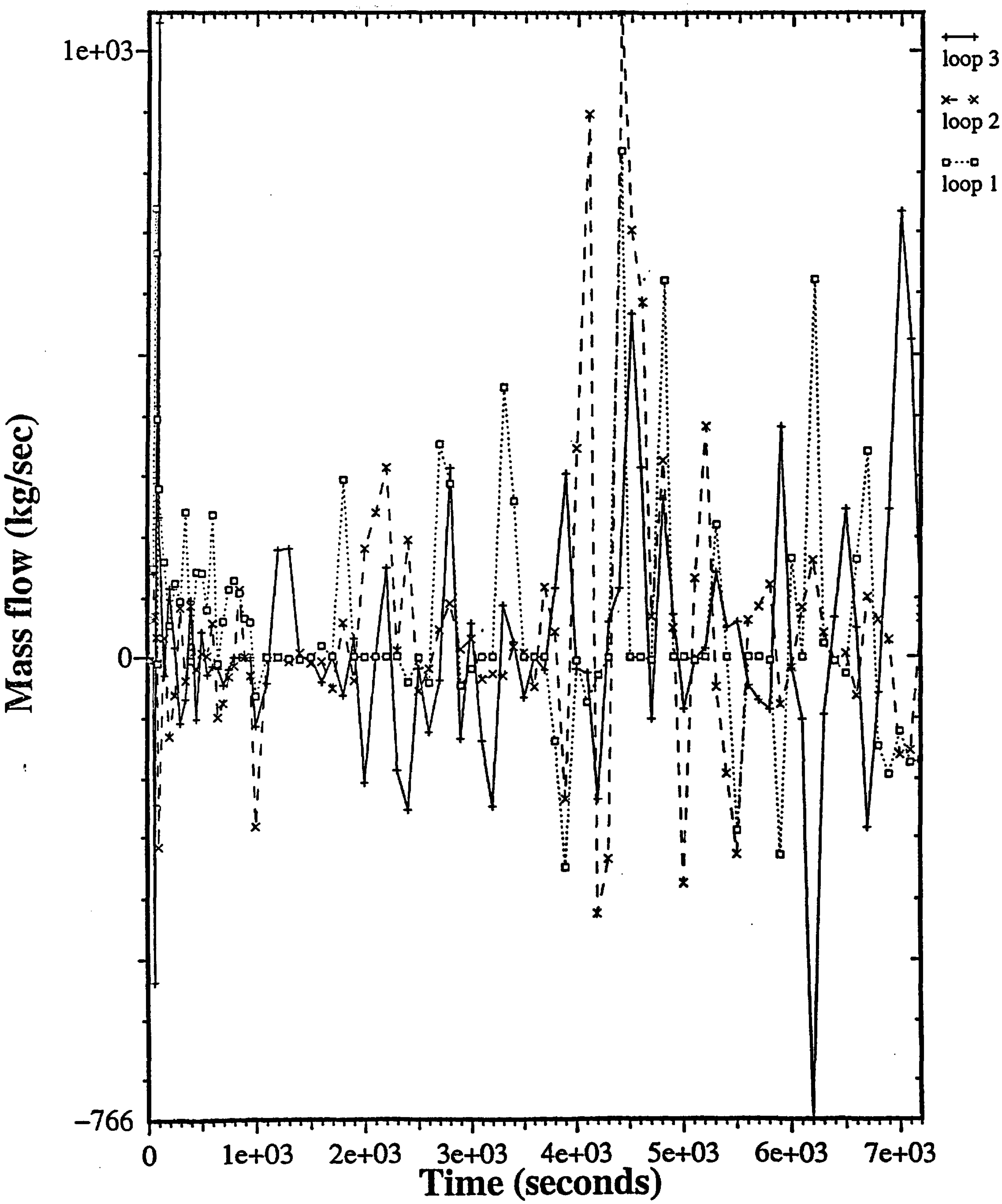


Figure 9: Cold legs mass flow rates for LOCA 6"

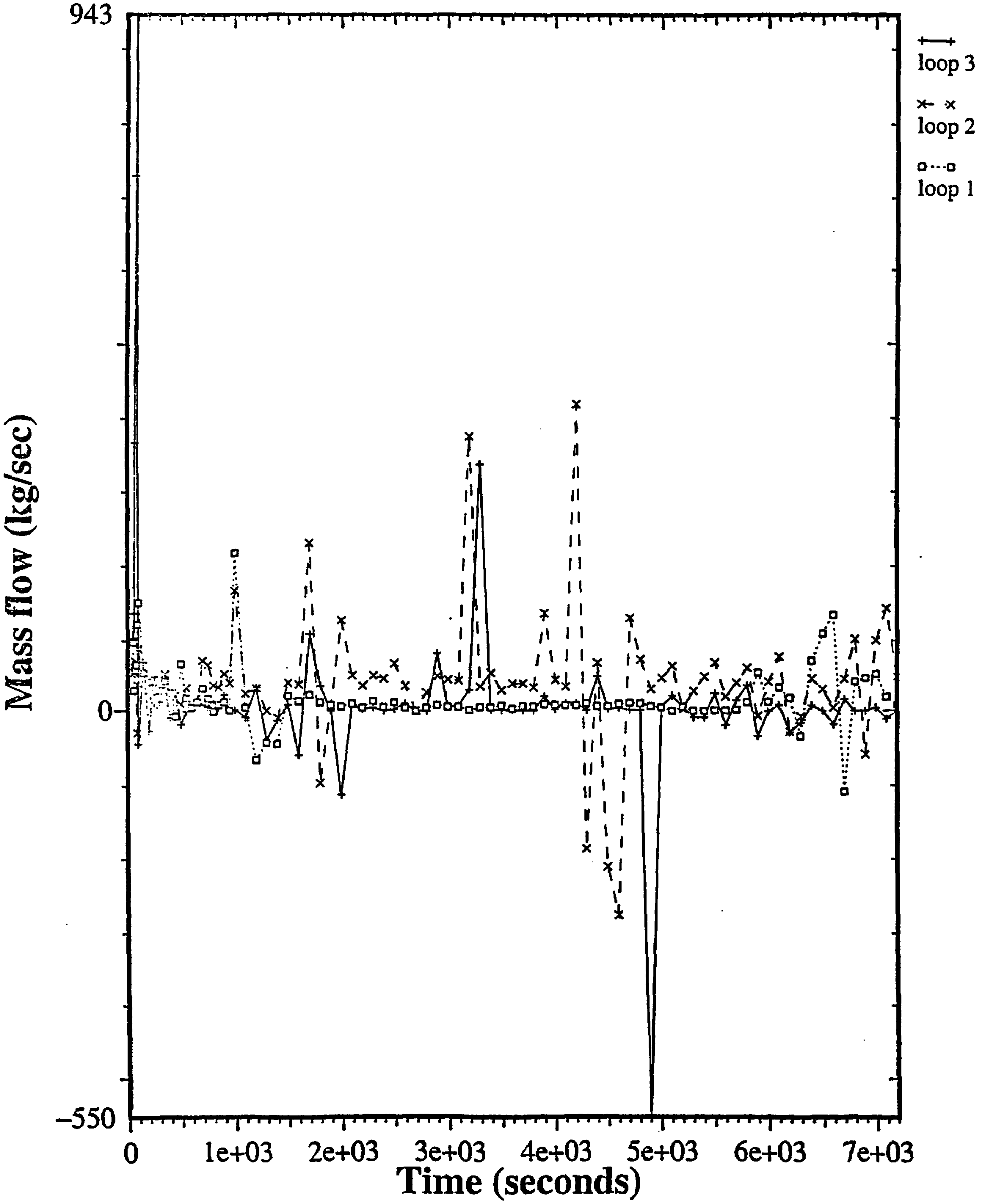


Figure 10: Reactor coolant system mass for LOCA 6".

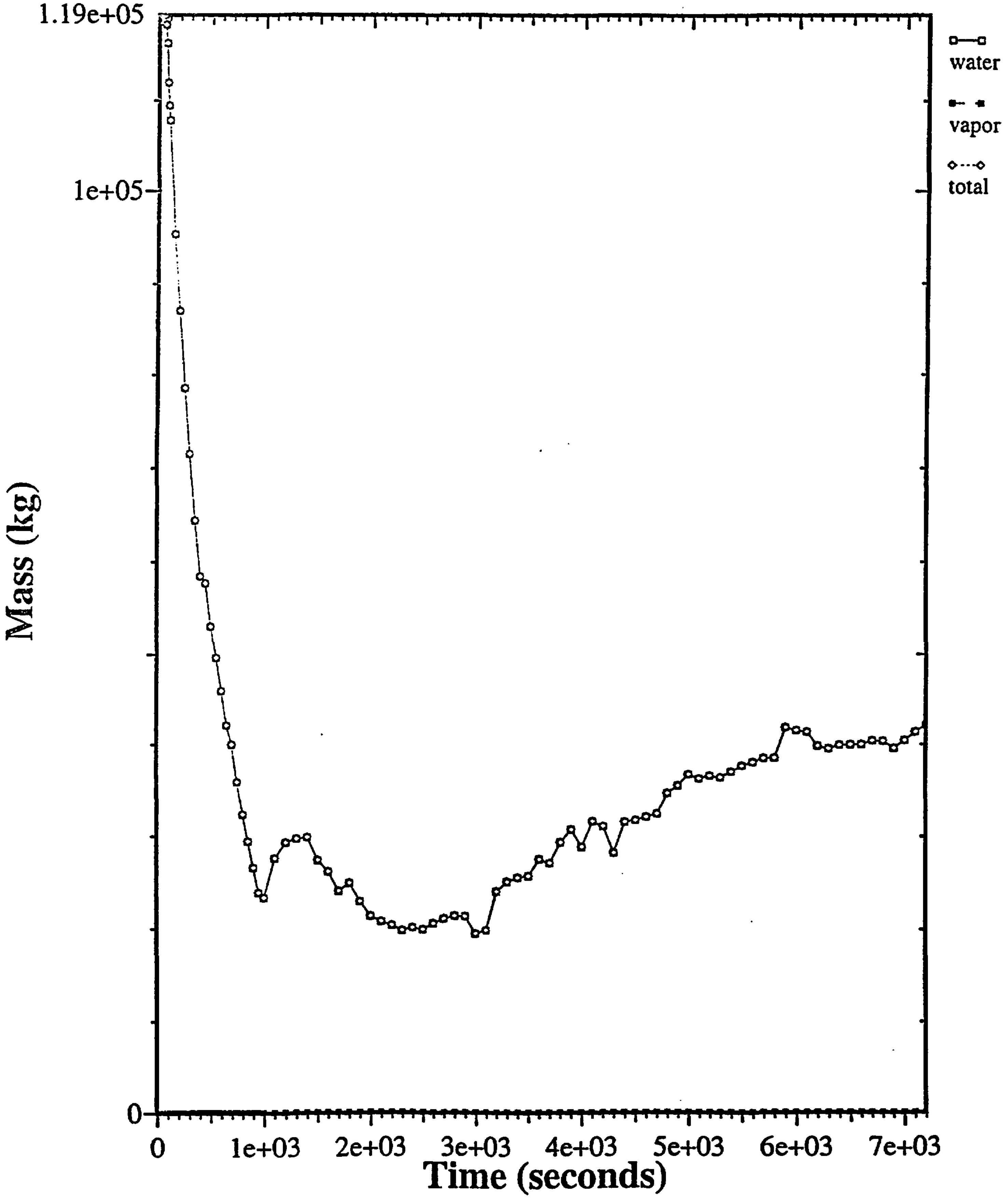


Figure 11: Core level for LOCA 2"

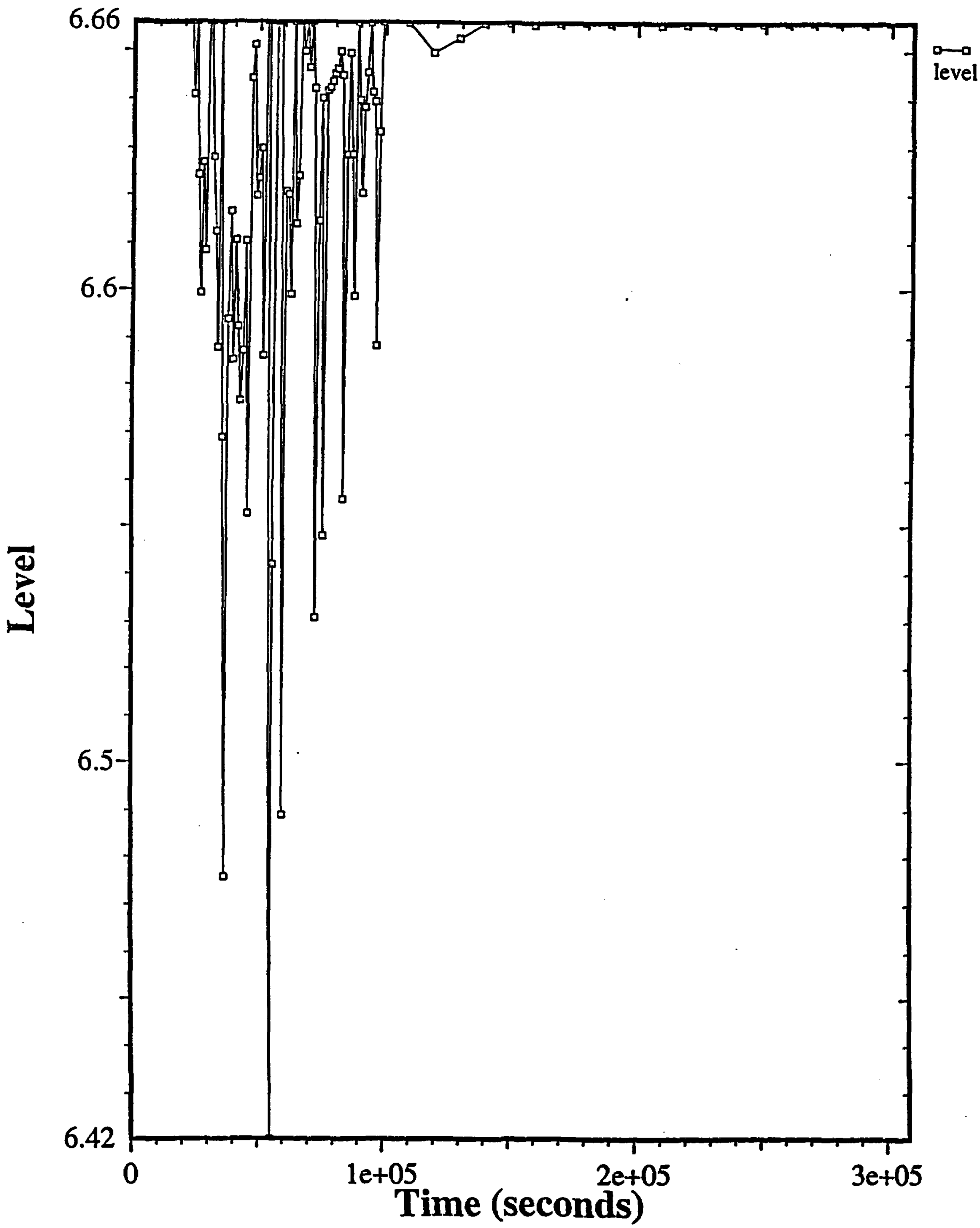


Figure 12: Core outlet and clad temperatures for LOCA 2".

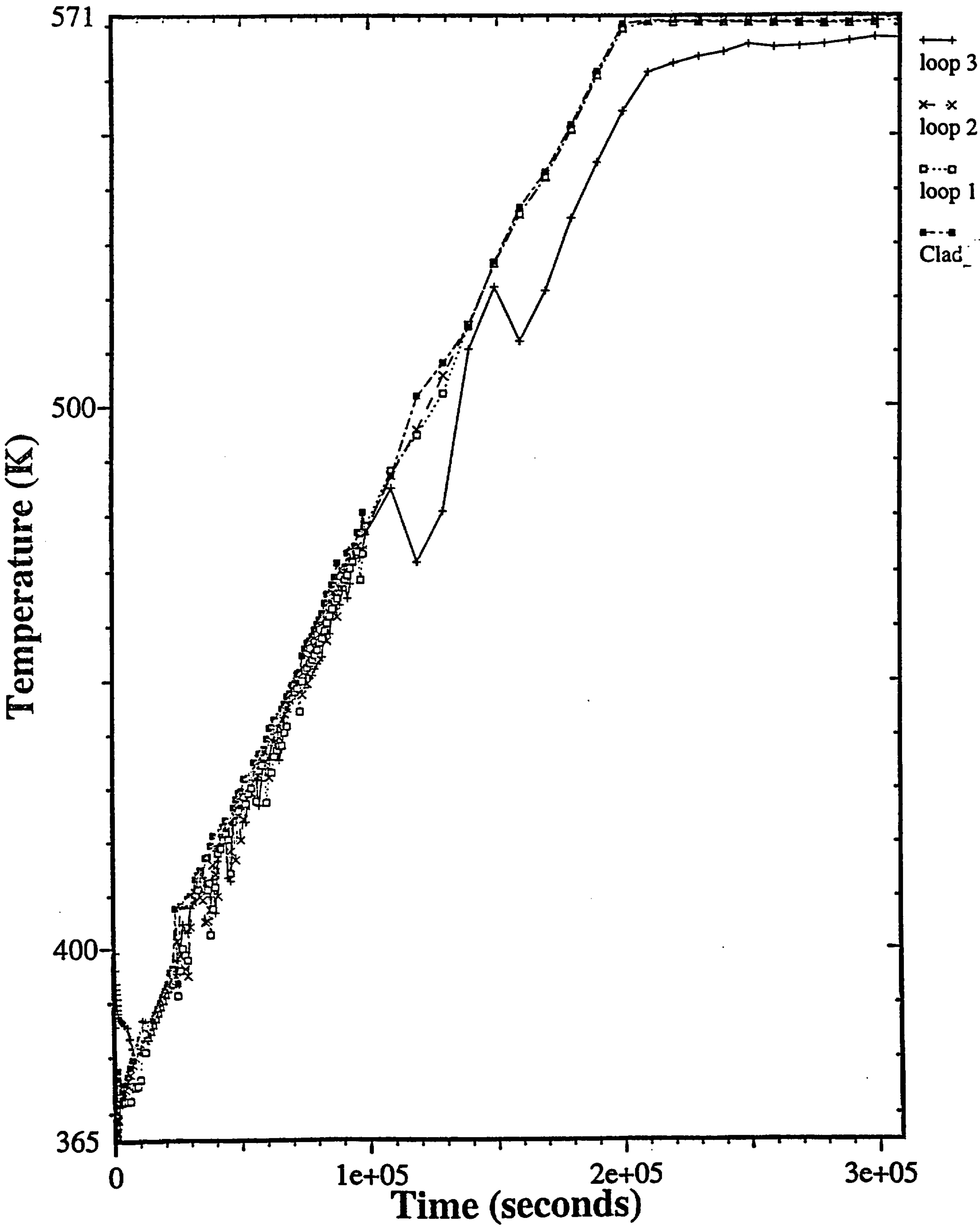


Figure 13: Reactor coolant system pressure for LOCA 2".

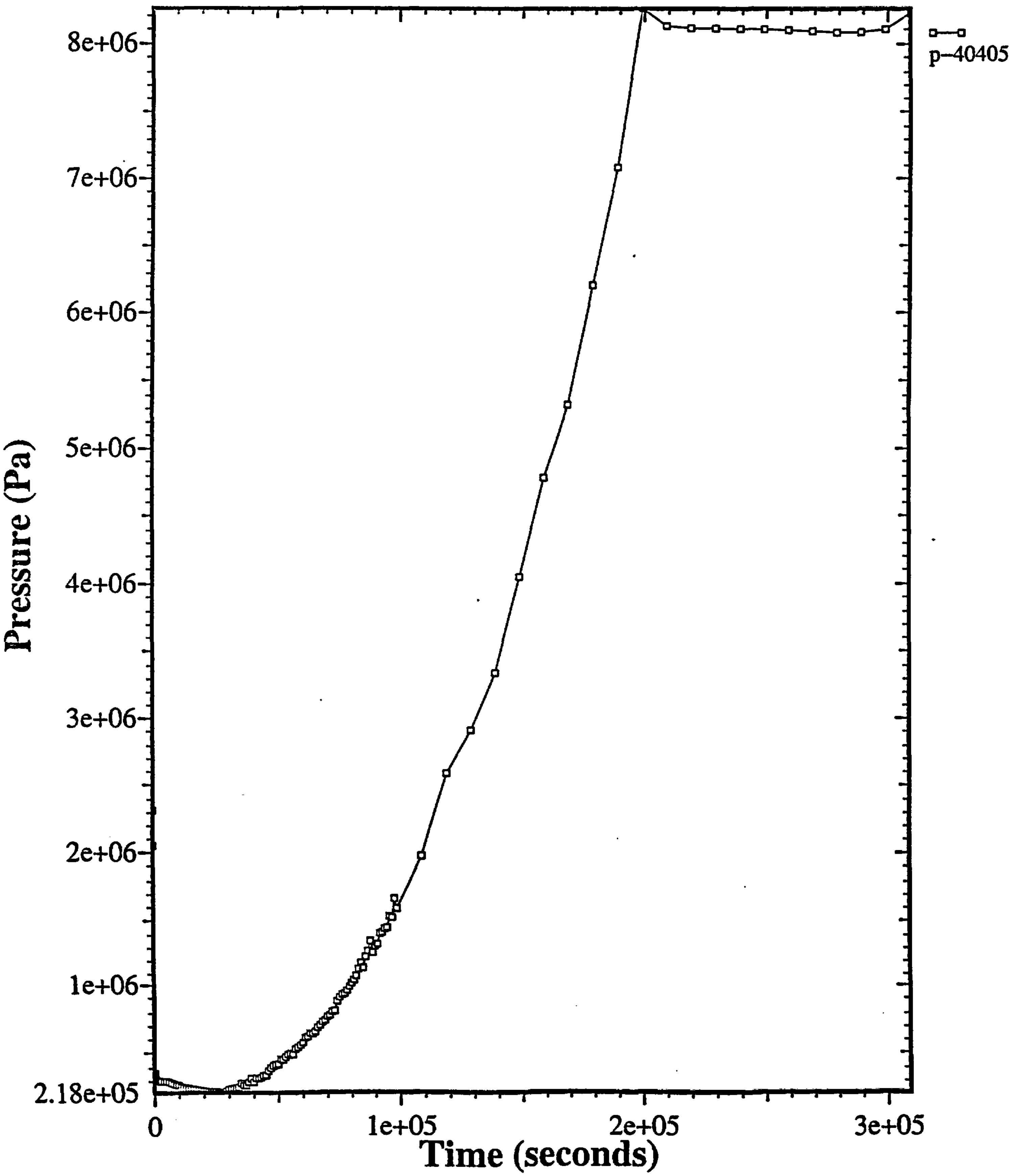


Figure 14: Steam generators pressures for LOCA 2".

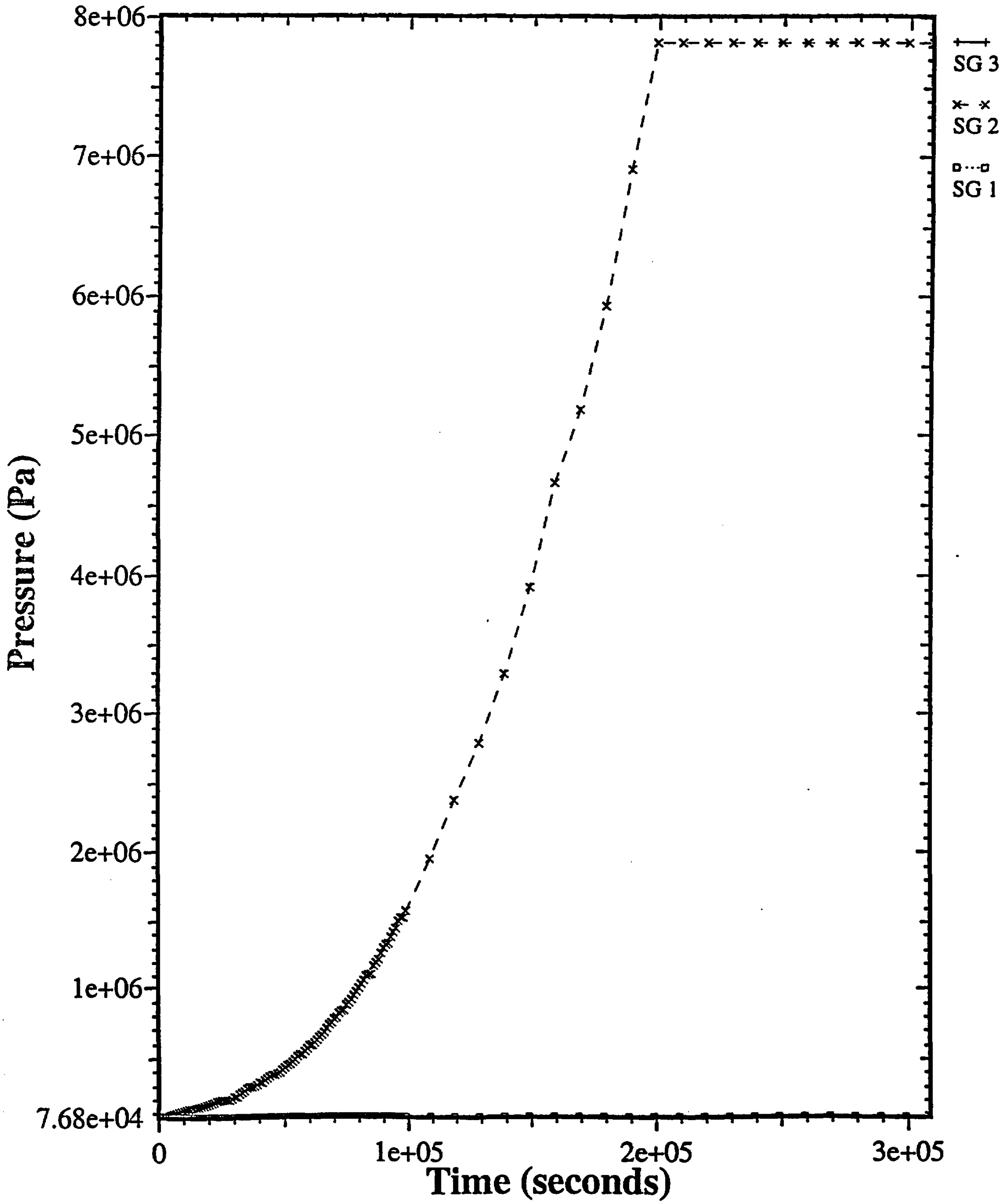


Figure 15: Steam generators levels for LOCA 2".

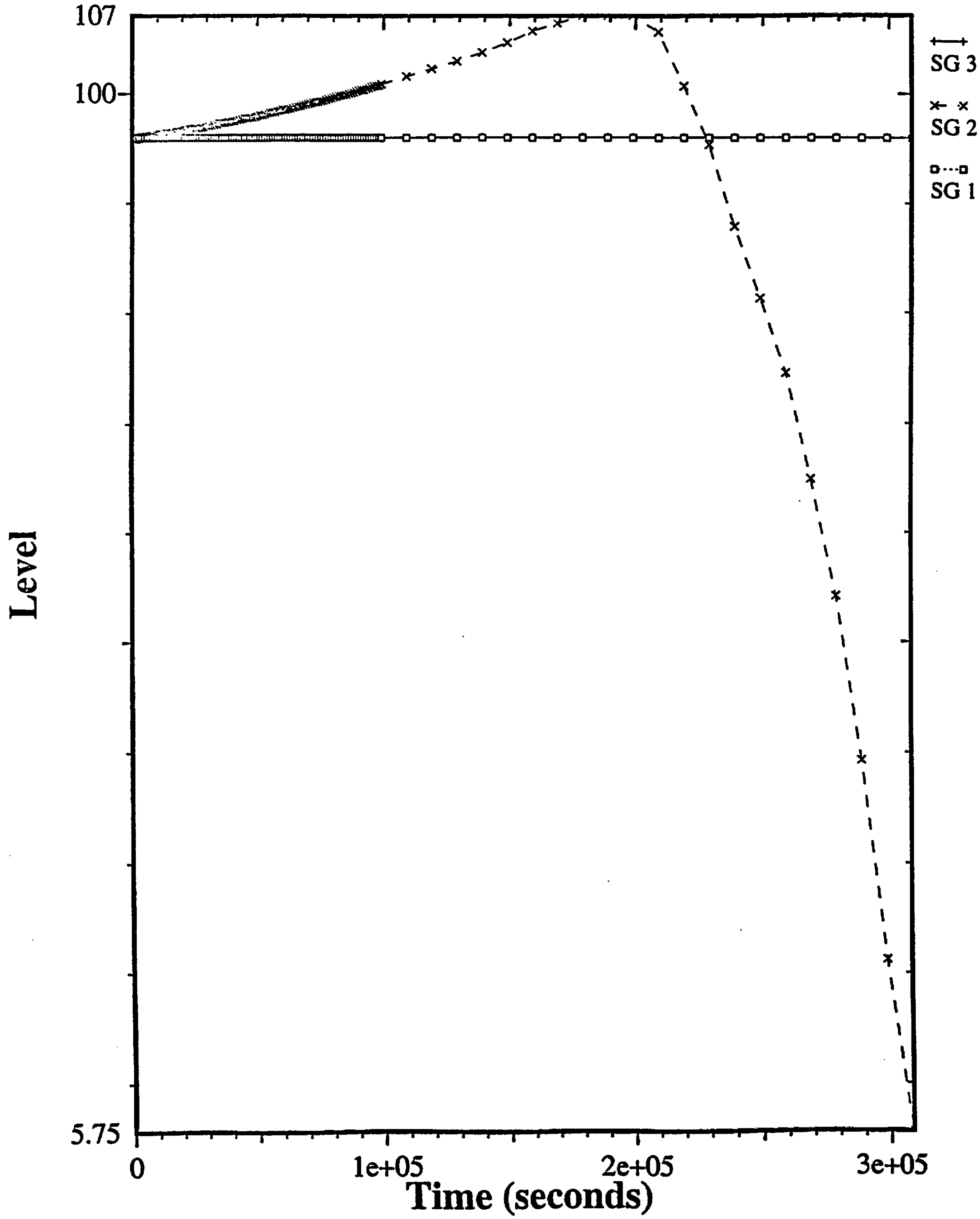


Figure 16: Hot legs mass flow rates for LOCA 2".

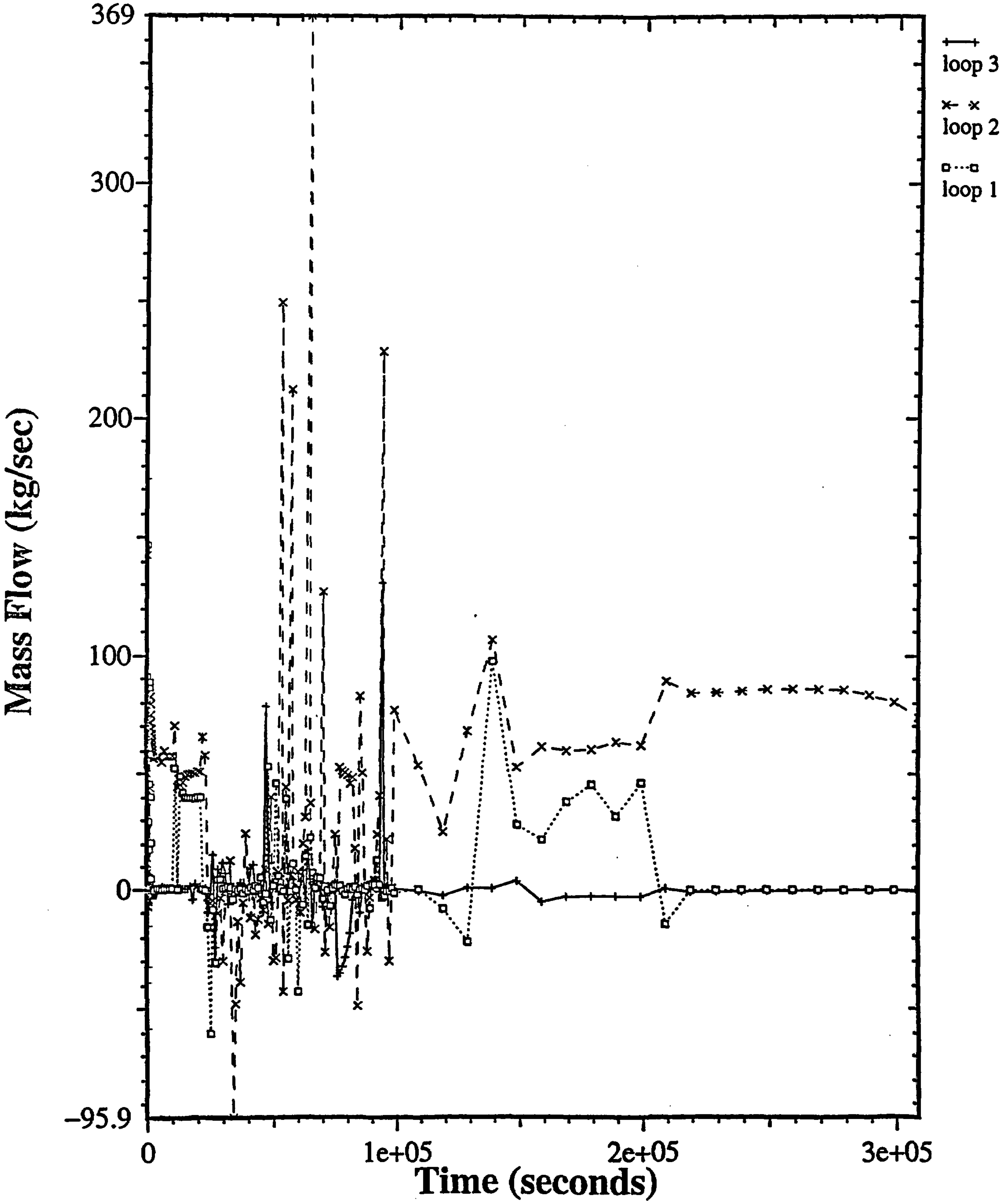


Figure 17: Cold legs mass flow rates for LOCA 2".

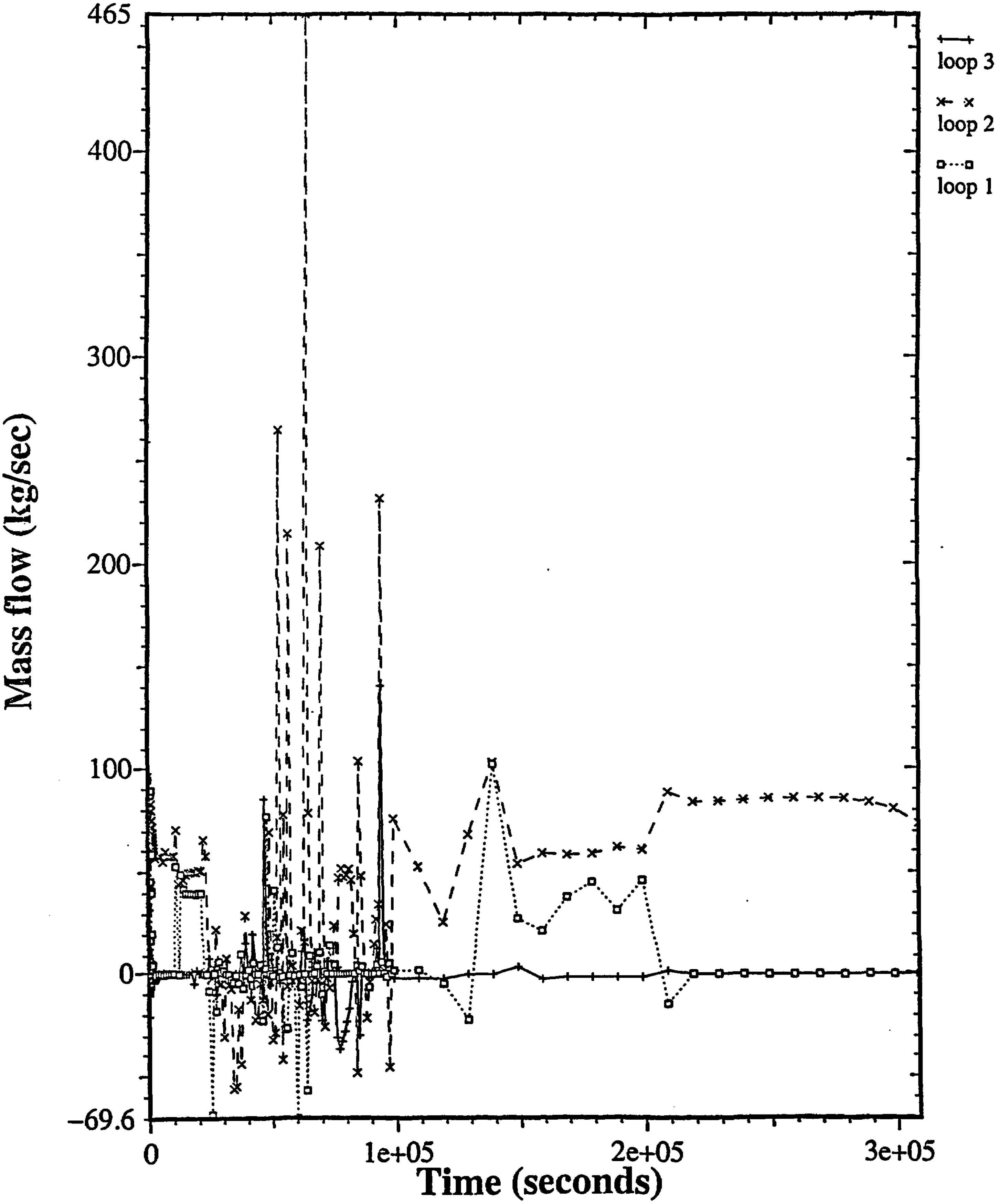


Figure 18: Pressurizer level for LOCA 2"

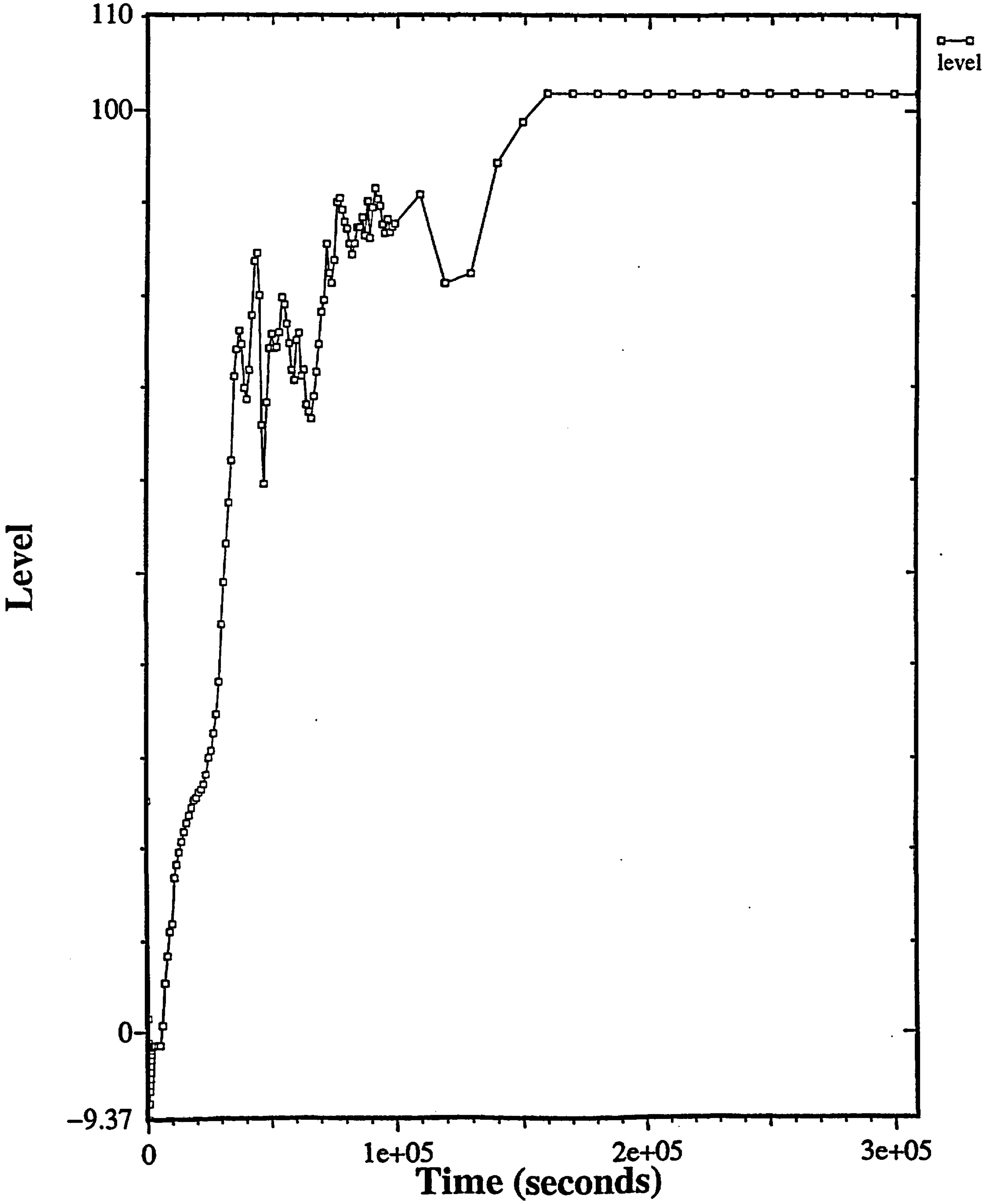


Figure 19: CPU time for LOCA 6".

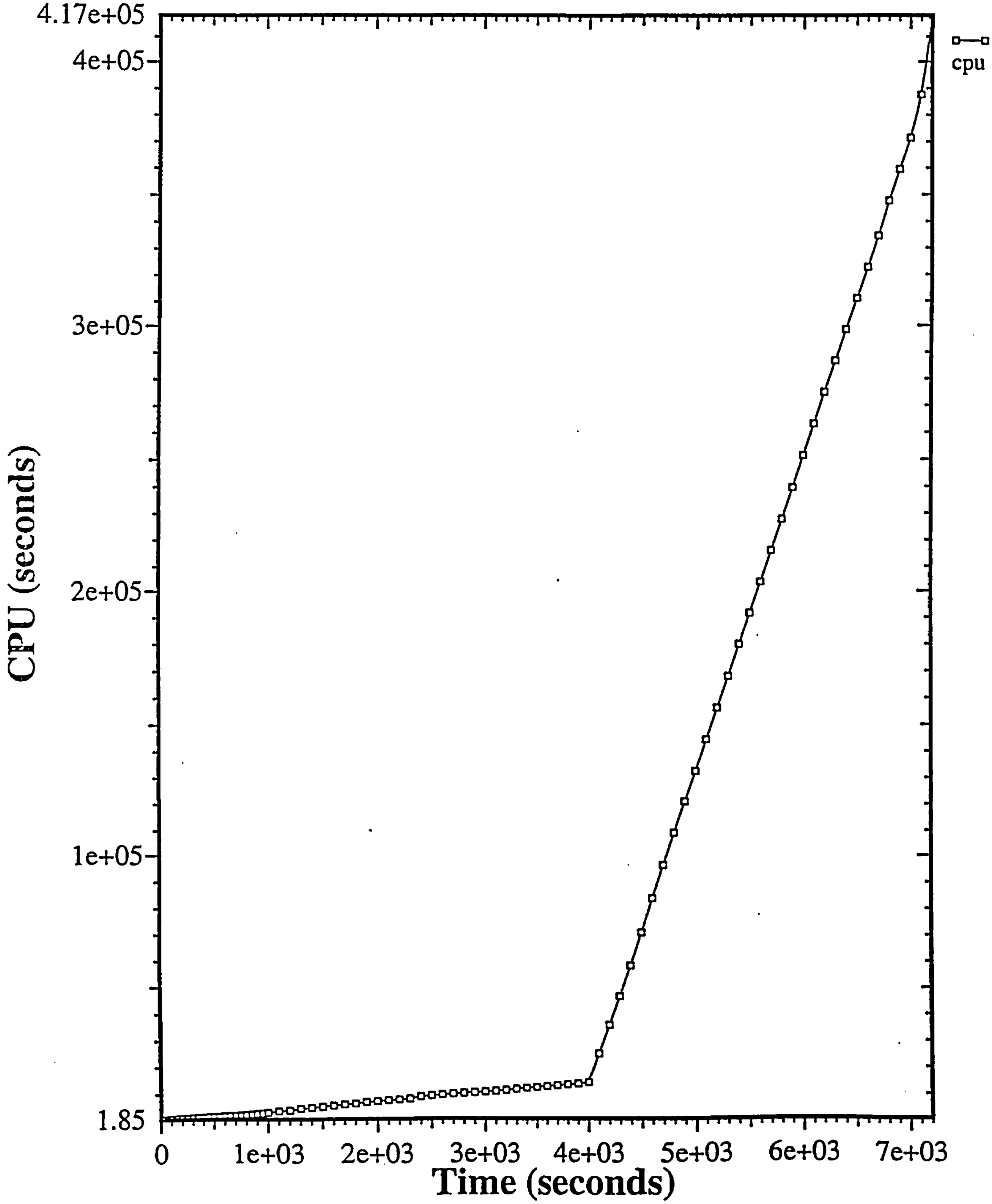


Figure 20: Time steps for LOCA 6".

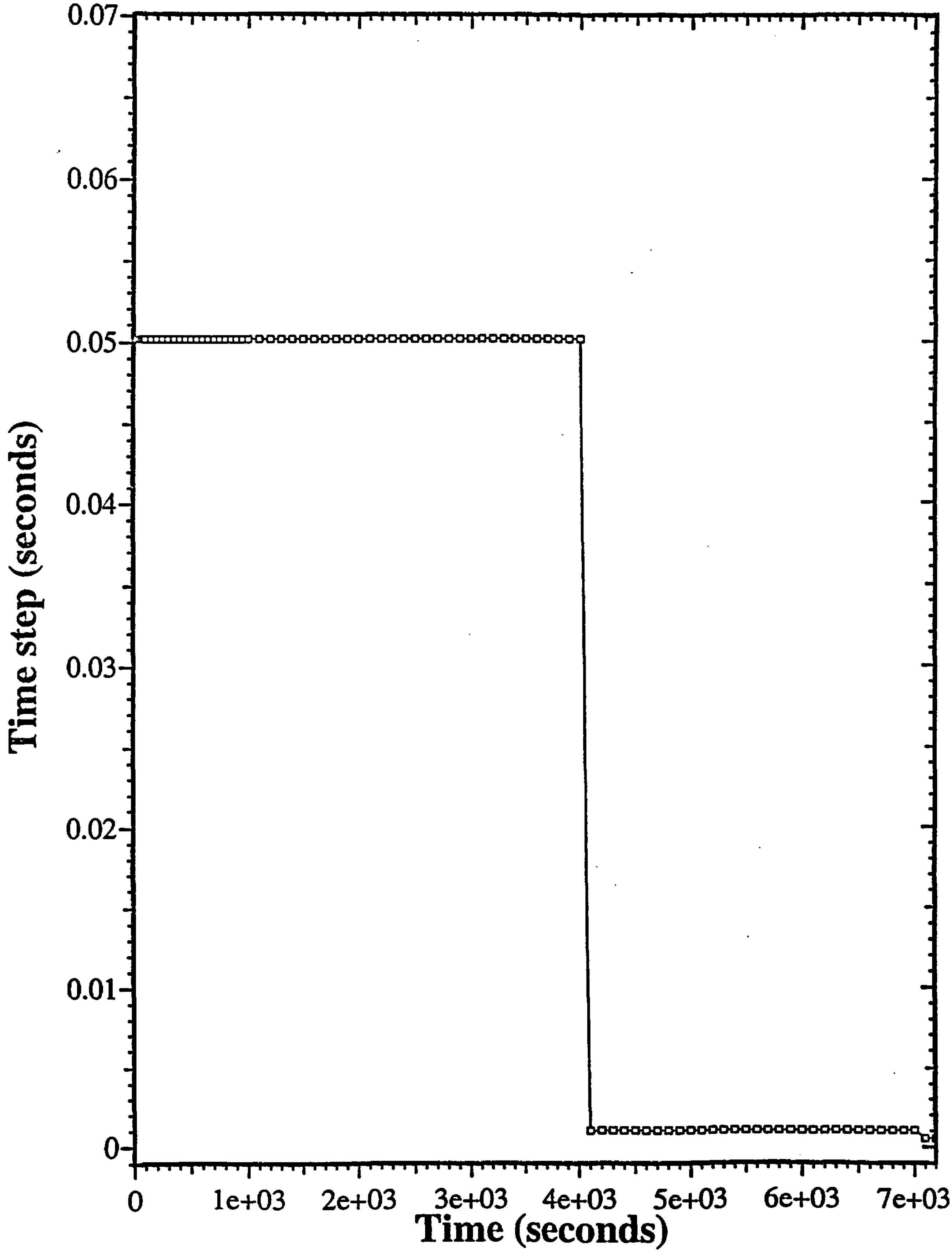


Figure 21: CPU time for LOCA 2".

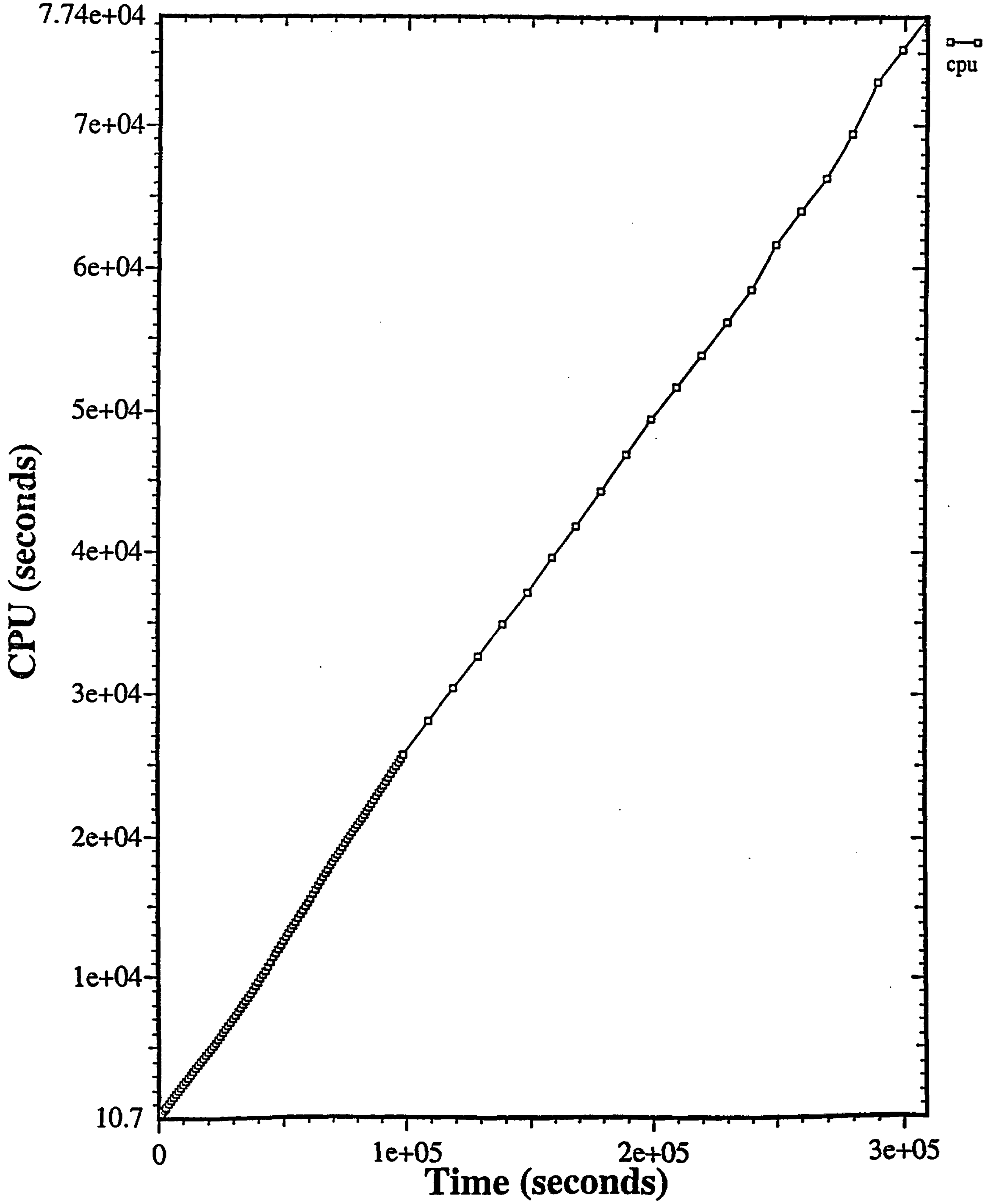
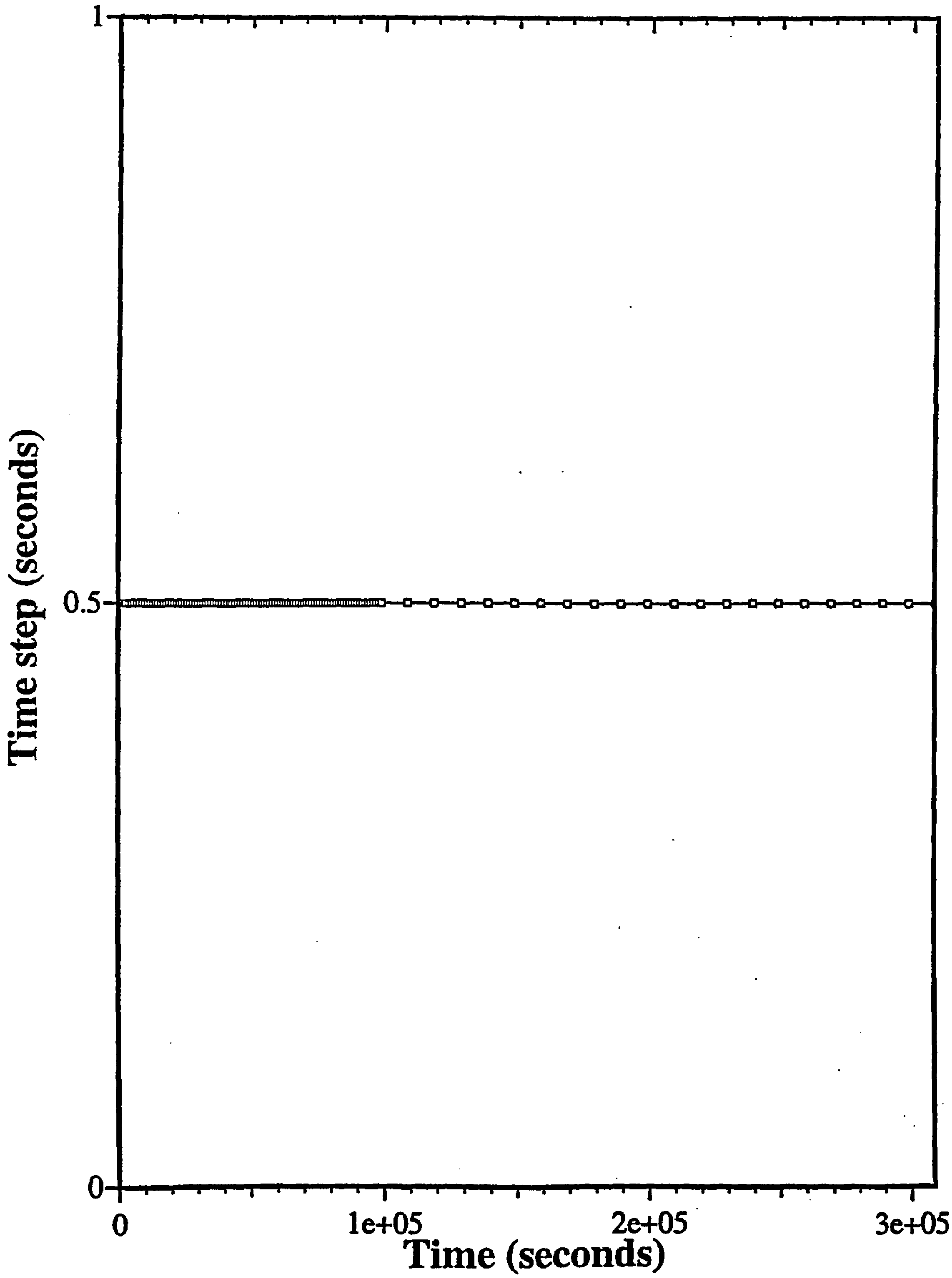


Figure 22: Time steps for LOCA 2"



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

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/IA-0171

2. TITLE AND SUBTITLE

Simulation of LOCA 6" and LOCA 2" Transients in the RHR of a PWR Under Low Power Conditions Using RELAP5/MOD3.2

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	2000

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

S. Martorell, S. Carlos, G. Verdu, V. Serradell, J. Rodenas, R. Miro, C. Llopis*

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

C. Vandellos II
Travessera de les Corts, 55, Lateral
08028 Barcelona

Department of Chemical and Nuclear Engineering
Polytechnic University of Valencia
Camino de Vera, 14
Spain

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of System Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The present study consists of the simulation of two loss of coolant accidents, LOCA 6" and LOCA 2", in one of the residual heat removal system (RHR) lines outside the containment using the thermal-hydraulic code RELAP5/MOD3.2. Both transients have been simulated on a typical three loop, Westinghouse design, pressurized water reactor plant working under shutdown conditions. The study was focused on the simulation of the most important thermal-hydraulic parameters in order to check the validity of the success criteria assumed in the plant probabilistic safety analysis (PSA) under shutdown conditions. Also to analyze the code capability for simulating shutdown conditions was of interest in this study. As a result of this study, it can be concluded that the main thermal-hydraulic plant features follow what is foreseen in the plant PSA, although it can not be assured that the values reached are the correct ones due to the lack of experimental data.

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

RELAP5/MOD3.2, LOCA 6", LOCA 2", PSA, low pressure, RHR

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE

Printed on recycled paper

Federal Recycling Program