NUREG/CR-5820
EGG-2666

Consequences of the Loss of the Residual Heat Removal Systems in Pressurized Water Reactors

Prepared by
L. W. Ward, W. Arcieri, C. Heath

Idaho National Engineering Laboratory
EG&G Idaho, Inc.

Prepared for
U.S. Nuclear Regulatory Commission

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grant publications, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

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 can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, for 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 the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG/CR-5820
EGG-2666

Consequences of the Loss of the Residual Heat Removal Systems in Pressurized Water Reactors

Manuscript Completed: April 1992
Date Published: May 1992

Prepared by
L. W. Ward, W. Arcieri, C. Heath

Idaho National Engineering Laboratory
Managed by the U.S. Department of Energy

EG&G Idaho, Inc.
Idaho Falls, ID 83415

Prepared for
Division of Systems Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN L1696
Under DOE Contract No. DE-AC07-76ID01570

ABSTRACT

During shutdown at Vogtle Unit 1 on March 20, 1990, the loss of vital ac power and the Residual Heat Removal System (RHRS) focused much attention on the need to evaluate system performance following such an event in light water reactor (LWR) facilities. The RELAP5/MOD3 transient, non-equilibrium system performance code and an alternate methodology were used to evaluate this scenario and to investigate the accident consequences and identify key phenomenological and system behaviors characterizing these events. To investigate thermal and hydraulic behavior following a loss of the RHRS, studies evaluated the use of the steam generators as an alternative for removing decay heat from mid loop operation. Additional studies investigated the effects of decay power and changes in reactor coolant system (RCS) water level on system behavior when attempting to use the steam generators for heat removal. Under these alternate heat removal conditions, analyses identified the time to core uncovery in the event a nozzle dam or temporary thimble tube seal fails. Other evaluations included assessing the impact of a loss of the RHRS with the reactor vessel internals in place and the upper head removed. For some plant designs, the flow restrictions through the upper internals may inhibit the downflow of water from the refuel pool cavity to the core during boiling, which could lead to the long-term uncovering of the fuel. Lastly, in the event boiling occurs in an open RCS, the impact of the addition of borated water for extended periods of time was investigated to identify the potential for boric acid precipitation.

FIN L1696—Task Order 1, Loss of RHRS

CONTENTS

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

The loss of vital ac power and the Residual Heat Removal System (RHRS) during shutdown at Vogtle Unit 1 has emphasized the need to evaluate system performance following such an event. The Nuclear Regulatory Commission Incident Investigation Team (IIT) reviewed the Vogtle Unit 1 event in NUREG-1410 and noted that alternate methods of residual heat removal, which would be required in the event the RHRS failed, need to be identified and made available for core cooling. The IIT also identified the need for emergency procedures to deal with an extended loss of the RHRS. Because there remains a lack of understanding of thermal-hydraulic behavior and nuclear steam supply system performance following loss of the RHRS, system performance was evaluated to identify the important phenomena characterizing these events.

To evaluate system performance following the loss of the RHRS, transient calculations were performed using RELAP5/MOD3 and an alternate methodology to predict primary and secondary thermal hydraulic behavior for a range of conditions during plant shutdown and refueling. Of particular interest were the conditions during shutdown when the steam generators could be used as an alternate means of decay heat removal in the event the RHRS failed. In a closed system, the steam generators could be used for decay heat removal by condensation of primary steam, usually referred to as a reflux cooling mode of heat transfer. Other conditions investigated included a loss of the RHRS during refilling of the refueling pool cavity in preparation for removal of the spent fuel.

The study was performed based on the conditions achieved following a normal cooldown of the RCS and after the RCS inventory has been reduced to very low levels. Operation with the RCS at low liquid inventory—where the liquid inventory is reduced to levels within the hot leg outlet piping—is commonly referred to as mid-loop operation. At mid-loop conditions, the steam generator manway entries can be removed in preparation for the maintenance and refueling activities. Because the primary system will contain nitrogen and air following opening of the primary system manways, the loss of the RHRS at this time presents two important considerations to recovering from the event. These considerations form the basis of the analyses contained in this report and include:

- If the integrity of the RCS temporary boundaries can be assured and the RCS vent paths and openings can be closed, use of the steam generators as an alternative means of heat removal may be a desirable strategy because such action would preclude the need to add water to the primary system once bulk boiling begins

- If RCS integrity cannot be assured, then a source of injection water will be needed to prevent core uncovery in the event boiling in the core develops. If boiling persists for an extended period of time, then the potential for precipitation of the boric acid in the reactor vessel needs to be avoided.

For the first consideration, noncondensible gases in the primary system at low pressure inhibit the condensation of steam in the steam generator primary tubes. Because RCS pressure is strongly influenced by this condensation process, an assessment of the resulting primary peak pressure is important, particularly for those plants with nozzle dams or temporary thimble seals in place. Nozzle dams and temporary thimble seals are designed to accommodate peak pressures in the approximate range of 40 to 50 psia (0.28 to 0.34 MPa), as documented in NUREG-1410. With air or nitrogen in the RCS, peak pressures could approach and potentially exceed the design limits of these temporary boundaries.

In regard to the second consideration above, should there be insufficient time to close the RCS, a source of injection water to the vessel would be required to prevent uncovery of the core once bulk boiling in the core has begun. The necessity of a borated water source to maintain core cooling, when core boiling persists for an extended period of time, will result in the accumulation of boric acid in the vessel. If the RHRS is restored following an extended period of boiling, use of the RHRS could reduce RCS fluid temperatures below the precipitation temperature of vessel boric acid-water mixture, thereby precipitating the boric acid in the vessel and RHRS lines. Restoration of the RHRS under such conditions may not ensure long-term cooling can be successfully reestablished and maintained for an extended period of time, because the activation of this system could cause precipitation in the RHRS lines preventing its further use for decay heat removal.

To assess the primary and secondary response following a loss of the RHRS, the RELAP5/MOD3 code and an alternate methodology were used to simulate the RCS and secondary system thermal hydraulic behavior. The alternate method consisted of a simplified transient nonequilibrium code that was used to provide an initial analysis to identify key phenomenological behaviors warranting further investigations using the RELAP5/MOD3 code. The alternate methodology also served as a means to study separate effects on the loss of the RHRS event and to check the RELAP5/MOD3 code predictions. Separate models for the prediction of boric acid concentration during bulk boiling in the vessel were also included in the alternate methodology to assess the potential for boric acid precipitation.

These methodologies were used to evaluate the consequences of the loss of the RHRS in pressurized water reactors (PWRs) utilizing U-tube and once-through steam generator designs. The studies were performed for the Westinghouse-designed H. B. Robinson and the Oconee nuclear plants built by Babcock and Wilcox (B&W). These analyses include investigations of the following events:

- Use of the steam generators as an alternate means of decay heat removal in the event the RHRS fails during mid-loop operation
- Consequences of a failure of a hot leg nozzle dam when the steam generators are used for decay heat removal
- Consequences of a failure of all of the temporary thimble tube seals
- Consequences of a loss of the RHRS with a safety injection line open to the containment from the discharge leg
- Consequences of a loss of the RHRS during refilling of the refueling pool cavity with the reactor vessel internals in place
- Consequences of the addition of borated water to reactor vessel following the initiation of boiling.

A summary of the key results and conclusions from these analyses is given as follows:

- Analyses of the loss of the RHRS from mid-loop operation for PWR plants with U-tube or once-through steam generators reveal that the RCS can achieve peak pressures in the range of 40 to 60 psia (0.28 to 0.41 MPa) when a single steam generator is used for decay heat removal. While RCS peak pressure is relatively insensitive to decay heat level or the time of loss of the RHRS following shutdown, a steam relief path with eventual auxiliary feedwater is needed on the secondary side of the steam generator to maintain a heat sink should boiling occur.
- Additional analyses of the use of the steam generators for decay heat removal show that RCS peak pressures approach 95 psia (0.66 MPa) if the initial RCS water levels are above the top elevation of the hot leg. At these higher water levels, the fluid expansion fills the steam generator tubes with sufficient liquid to prevent decay heat removal until pressures reach 95 psia (0.66 MPa) or until sufficient primary to secondary temperature difference is established. Peak RCS pressure may therefore be sensitive to the initial liquid level at the time of loss of the RHRS.
- Because RCS pressures near the design conditions for nozzle dams and temporary thimble seals can be achieved, the successful use of the steam generators as an

alternative decay heat removal mechanism is not assured. The loss of the RHRS with initial RCS water levels above the top of the hot leg further suggests that use of the steam generators as an alternate means of decay heat removal could result in sufficient pressure to challenge the integrity of all of the RCS temporary boundaries.

- Analyses of the inventory loss due to the failure of the RCS temporary boundaries (i.e., thimble tube seals, nozzle dams, etc.) or openings in the RCS, such as the safety injection line or reactor coolant pumps, demonstrate that core uncovery could occur from 15 to 90 minutes following the opening of the break.

- The analysis of the loss of the RHRS with the vessel internals in place suggests that, without intervention or recovery of heat removal capability, a partial uncovery of the core may occur for those Westinghouse-designed plants where the upper support plate flow holes have been sealed. With the vessel internals in place, the restricted flow through the upper support plate inhibits refueling pool cavity water flow to the vessel upper plenum and core regions, which could result in the long-term uncovery of the top of the core in the event boiling occurs. While core uncovery would not initiate for several hours, the potential for its occurrence emphasizes the need for procedures and alternate strategies to prevent boiling under these particular circumstances.

- Analyses of the loss of the RHRS with the reactor vessel internals (in place for those Westinghouse plants with the restricted flow through the upper support plate) also demonstrate that the refueling pool cavity water temperature is not a valid indication of the temperature in the core. Because the vessel coolant cannot readily circulate with the pool water, pool cavity water temperature should not be used to infer time to boiling in the vessel.

- Should boiling in the RCS occur, the addition of borated water into the reactor vessel to prevent core uncovery could result in the precipitation of the boric acid in the RHRS lines preventing its further use for decay heat removal. Precipitation following the restoration of the RHRS after four or more hours of boiling could occur when the refueling water storage tank is used as a source of cooling. Precipitation as early as two hours into the event could occur if a high concentrated source of water is used, such as the boric acid storage tank.

Based on the analyses presented in this report, it is important that studies be conducted on a plant-specific basis to properly evaluate the effectiveness of the proposed strategies and construct procedures to effectively deal with a loss of RHRS event. The analyses contained in this report should therefore be used only as guidance to identify those areas where further, more detailed, plant-specific analyses are needed to quantify the magnitude and timing of the key behaviors effecting a loss of the RHRS event.

Also the report suggests that given the consequences of loss of RHRS, the use of the steam generators as an alternate means of decay heat removal may not be assured, because peak pressures can approach and exceed the design pressure of the temporary boundaries in the RCS. In the unlikely event boiling occurs, the addition of borated water to the RCS may easily eliminate the short-term potential for core uncovery; however, the injection of borated water for extended periods of boiling represents a mechanism that could compromise decay heat removal and core coolability during the long term. More importantly, these analyses strongly suggest that specific procedures and strategies, in addition to careful attention to the instrumentation used to monitor the accident progression, are needed to successfully deal with the range of potential consequences associated with a loss of the RHRS event.

ACKNOWLEDGMENTS

The authors wish to express their appreciation to the USNRC Staff for their support to this work. In particular, the efforts of Mr. Warren Lyon of the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission (NRR/NRC) have provided many valuable insights to the understanding of many of the key phenomenological behaviors identified during the course of these studies. The comments and feedback from Mr. Mark Caruso of NRR/NRC were also appreciated.

NOMENCL RE

A_{sg} = steam generator heat transfer area [m^2, ft^2]

C = boron concentration [wt%]

C_o = initial vessel boron concentration [wt%]

C_p = air specific heat [Joule/kg-°C, Btu/lb-°F]

C_s = boron concentration of injection source [wt%]

$d_{o,i}$ = outside or inside tube diameter [m, ft]

F(t) = decay heat fraction [dimensionless]

h_{fg} = latent heat of vaporization [Joule/kg, Btu/lb]

h_i = enthalpy in region i where i = 1 or 2

h_p = primary coefficient [W/m^2-°C, Btu/hr-ft^2-°F]

h_s = secondary coefficient [W/m^2-°C, Btu/hr-ft^2-°F]

J = energy conversion factor [1.0, ft-lb/Btu]

k_w = wall conductivity [W/m-°C, Btu/hr-ft-°F]

L = average tube length [m, ft]

M_i = total mass in region i where i = 1 or 2 [kg, lb]

N = number of tubes

P = total pressure [MPa, lb/in^2]

P_o = initial core power [Joule/sec, Btu/sec]

Q_i = heat rate for region i, where i = 1 or 2 [Joule/sec, Btu/sec]

R = gas constant [cal/g mole-K, ft-lb/mole-°R]

t = time [sec]

T_i = temperature in region i where i = 1 or 2 [K, °F]

T_s = secondary temperature [K, °F]

U = overall heat transfer coefficient [W/m^2-°C, Btu/hr-ft^2-°F]

U_i = internal energy in region i where i = 1 or 2 [Joule, Btu]

V_{mix} = mixing volume [m^3, ft^3]

V_i = volume in region i where i = 1 or 2 [m^3 ft^3]

W_{in} = liquid flow into vessel [kg/sec, lb/sec]

w_k = mass flow rate into or out of region 1 [kg/sec, lb/sec]

w_n = mass flow rate of air into or out of region 2 [kg/sec, lb/sec]

Z_{cond} = condensation height [m, ft]

Greek Symbols

v_i = specific volume in region i where i = 1 or 2 [m^3/kg, ft^3/lb]

ρ = vessel liquid density [kg/m^3, lb/ft^3]

Subscripts

1 = lower liquid or steam-liquid region

2 = upper air region

i = region 1 or 2 index

k = flow related parameters for lower liquid region

n = flow related parameters for upper air region

Consequences of the Loss of the Residual Heat Removal Systems in Pressurized Water Reactors

1. INTRODUCTION

This report presents the results of the analyses of the consequences of a loss of the residual heat removal systems in PWR systems. Separate analyses were conducted for plants utilizing U-tube steam generators and those with once-through steam generator designs. Because of these differing steam generator designs, system behavior will differ following a loss of RHRS event.

This section contains the introductory and background information, a description of the types of events analyzed, and the method of analysis. Section 2 contains a detailed description of the RELAP5/MOD3 model followed by an explanation of the results for each of the events. Section 3 presents a description of the development of an alternate methodology that was used to provide a preliminary investigation into the key phenomenological behavior governing such an event in addition to providing a check for some of the RELAP5/MOD3 code results. Section 4 briefly discusses the review of test data pertinent to the phenomenological behaviors expected in a PWR following a loss of the RHRS, and Section 5 presents conclusions from this study.

1.1 Background

The loss of vital ac power and the Residual Heat Removal System (RHRS) during shutdown at Vogtle 1 has emphasized the need to evaluate system performance following such an event. The NRC Incident Investigation Team (IIT) published their findings regarding the Vogtle Unit 1 event in NUREG-1410[1] and noted that alternate methods of residual heat removal need to be identified and made available for core cooling in the event of RHRS failure. The IIT also identified the need for emergency procedures to deal with an extended loss of the RHRS. Because there remains a lack of understanding of thermal hydraulic behavior and nuclear steam supply system performance following loss of the RHRS, analyses of system performance were conducted to develop a better understanding of the important phenomena affecting recovery from these events. To address these issues, transient thermal hydraulic studies on the consequences of the loss of RHRS were performed using the RELAP5/MOD3 code with the objective of assessing the primary and secondary system thermal and hydraulic performance.

Of particular importance is the need to assess RCS pressure and temperature response if the steam generators are used as an alternate means of decay heat removal from mid-loop operation. This means RCS heat removal is of interest following a loss of RHR because from this mode of plant operation, the RCS contains a noncondensible gas. Because RCS pressure will increase once boiling initiates following a loss of the RHRS and there may also be temporary boundaries maintaining RCS integrity (i.e., nozzle dams, temporary thimble seals), assessment of the thermal and hydraulic behavior in the RCS is needed to identify the peak pressure achieved during such events. Because RCS pressure can approach and potentially exceed the design pressure of these temporary boundaries, additional analyses were performed to investigate the consequences of their failure. Evaluations of the consequences of the loss of the RHRS with the vessel internals in place was also performed. These analyses were used, because for some Westinghouse designs, the restricted flow through the upper internals could preclude cooling water in the refueling cavity pool from entering the core, resulting in core uncovery following boiling in the RCS. Lastly, in the event the steam generators cannot be used for decay heat removal and borated water is added to the RCS following boiling, the potential for boric acid precipitation was also investigated. A more thorough description of these events and methods is presented in the following section.

1.2 Types of Events Analyzed

Transient calculations were performed to predict primary and secondary thermal hydraulic behavior following a loss of RHR event in a PWR system for plants utilizing U-tube and once-through steam generators designs. The events include:

- Evaluation of the use of one steam generator to remove decay heat following a loss of RHRS event from mid-loop operation
- Determination of the time to core uncovery in the event of failure of all of the temporary thimble seals in the instrument tubes while a steam generator is being used to remove decay heat
- Determination of the time to core uncovery following nozzle dam failure (blowout) in the hot leg connected to the steam generator (with the manway removed from inlet plenum during mid-loop operation)
- Determination of the time to core uncovery assuming loss of RHRS with a check valve assumed to be removed from a safety injection line
- Analysis of the consequences of loss of RHRS with the upper core support plate assumed to be in position during flooding of the refueling cavity
- Analysis of the rate of boric acid accumulation in the reactor vessel as a result of using borated water for core cooling should boiling initiate.

1.2.1 Use of the Steam Generators for Decay Heat Removal. The ability to remove decay heat through one steam generator following a loss of RHRS event during reduced inventory conditions, including mid-loop operation, represents an alternate means of decay heat removal which, if successful, would not require water addition to maintain the core covered with a two-phase mixture. The success of this strategy also requires a closed RCS and the assurance that the temporary RCS boundaries will not fail as a result of the pressurization. In this condition, the nozzle dams are assumed to have been installed in the hot and cold leg penetrations to all but one steam generator. To maximize the calculated RCS pressure response, only a single steam generator is used to remove decay heat following a loss of the RHRS. Should a loss of the RHRS occur, the peak pressure and temperature reached in the reactor coolant system is significant because the nozzle dams and other temporary boundaries must be able to withstand these conditions to prevent a loss of coolant accident. Failure of a hot leg nozzle dam, for example, would create a direct path from the RCS to the containment through an open steam generator manway. In this analysis, air was assumed present in all portions of the primary system above the liquid level located at the centerline elevation of the hot and cold legs.

1.2.2 Failure of the Temporary Thimble Tube Seals from Mid-loop Operation. Use of the steam generators to remove decay heat from the RCS could result in peak RCS pressures sufficient to cause failure of the temporary thimble tube seals used to isolate the instrument tubes. These thimble tube seals are used during plant outages while instrument maintenance is performed. A bounding analysis was performed to determine the time to core uncovery assuming all of the temporary thimble seals fail.

1.2.3 Failure of a Hot Leg Nozzle Dam from Mid-loop Operation. The time to core uncovery due to the overpressure failure of the hot leg nozzle dam with the manway removed from the steam generator inlet plenum was identified. Nozzle dam failure is arbitrarily assumed to occur once the RCS pressure reached 40 psia (0.28 MPa). For these conditions the reactor vessel head is assumed to be in position with air in all portions of the primary system above the centerline of the hot and cold legs.

1.2.4 Loss of the RHRS with a Check Valve Removed from the SI Line. As in the case of nozzle dam failure, the time to core uncovery assuming loss of the RHRS during the maintenance of a safety injection line check valve was also evaluated to identify the time that plant personnel have to restore cooling before core damage occurs with an open penetration in the cold leg piping. Again, the reactor head has not been removed and air is assumed to be in all portions of the primary system above the centerline of the hot and cold legs. All of the steam generators were assumed isolated with nozzle dams.

1.2.5 Loss of the RHRS with Vessel Internals In Place. The consequences of the loss of the RHRS with the vessel internals in place were analyzed. Such conditions would occur, for example, during the flooding of the refueling pool cavity in preparation for the fuel reloading. This event is important because of the possibility of core uncovery, should boiling occur after the RHRS is lost. The potential for water holdup at the support plate by the core decay heat generated steam could prevent the downflow of water to the core needed to maintain the core covered with a two-phase mixture. The total flow area through the upper core support plate, which represents the only path for communication of the refueling pool cavity water with the upper plenum and core, is approximately 2.3 ft^2 (0.21 m^2) for some Westinghouse plant designs. With this very small flow area, once bulk boiling in the reactor vessel initiates following a loss of the RHRS, the steam velocities through the plate may be sufficient to limit the downflow of water from the refueling pool. In these circumstances, core uncovery could occur resulting in the heatup and oxidation of the fuel. While temperatures are not expected to exceed 2200°F (1477 K), the potential for the long-term uncovery of the core suggests that strategies and procedures be developed to deal with the loss of the RHRS under these conditions.

1.2.6 Consequences of Borated Water Addition Following Boiling. The injection of borated water into the RCS would be necessary to maintain vessel inventory should boiling occur for conditions where the RCS cannot be sealed in a timely manner. Because borated water would be injected to prevent core uncovery, boric acid will accumulate in the vessel. The subsequent restoration of the RHRS could cause the boric acid to precipitate in the RHRS piping, preventing its further use for decay heat removal.

1.3 Method of Analysis

This section discusses the methods used to evaluate the consequences of the loss of the RHRS events discussed in the previous section.

1.3.1 RELAP5/MOD3 Computer Program. Transient thermal hydraulic analyses of the loss of RHRS were performed using the RELAP5/MOD3 (Version 5m5) computer program[2] executed on a DECstation 5000/Model 200 RISC Workstation. The RELAP5 code was used to assess performance for the H. B. Robinson plant (HBR-2), which is a three-loop Westinghouse PWR with a thermal power rating of 2300 MW(t), and the Oconee 2568 MW(t) plant, which utilizes a once-through steam generator design. These plants formed the basis for the RELAP5/MOD3 analyses discussed in Section 2.

1.3.2 Alternate Methodology. Independent transient analyses were performed utilizing the alternate methods developed by Ward.[3] For these analyses, a non-equilibrium transient system performance code was developed to simulate the RCS and secondary system thermal hydraulic behavior following a loss of the RHRS with air in the system. These calculations served as an alternate method to analyze and provide early insights into the consequences of a loss of the RHRS and a check of some the RELAP5/MOD3 code predictions. As part of these alternate methods, condensation models were also developed with the capability of assessing the degradation of steam condensation due to the noncondensible gas as the steam-air mixture enters the primary steam generator tubes.

In developing the alternate method, the transient fluid behavior in the primary and secondary systems was computed based on a mass and energy balance for these regions. In treating the behavior of the air-steam-water transient system behavior two phenomenological models were considered and included the following:

- Steam Compresses Air—Upon boiling, the steam is assumed to compress the air until the volume of the air was decreased to that necessary for steam to enter the primary tubes. A condensation height was computed with the condensation coefficient on the primary side based on pure steam flowing in the primary tubes. This model is referred to as the piston model.

- Steam Mixes with Air—Upon boiling, the steam mixes uniformly with the air. The resulting mass fraction of air was then used to compute the effective condensation heat transfer coefficient in the steam generator primary tubes. This model includes a methodology to compute the degradation in condensation heat transfer with a noncondensible gas present.

The air compression or piston model is considered more appropriate for assessing the consequences of a loss of the RHRS event from mid-loop operation with air in the RCS. This conclusion is based on an evaluation of several pertinent experiments that are discussed in more detail in Section 3. These experiments show that with a noncondensible gas above a steam-water mixture, the production of steam released from the two-phase mixture region acts as a piston compressing the air into the upper portions of the system. RELAP5 analyses of the loss of RHRS from mid-loop operation demonstrated that the code calculated behavior was consistent with the piston model. This behavior and the results of the RELAP5/MOD3 predictions using the steam generators as an alternate means of decay heat removal from mid-loop operation are discussed further in Section 2.1.

The models and resulting analyses for the prediction of boric acid concentration during bulk boiling in the vessel were also based on the work of Ward.[3] Section 2.6 provides a description of the models used to evaluate the potential for boron precipitation.

2. RESULTS OF THE LOSS OF RHRS ANALYSIS

The results of the application of RELAP5/MOD3 and the alternate methods to the loss of RHRS for the events described in Section 1 are presented in this section.

It should be emphasized that the analyses presented in this section were performed to identify some of the key phenomenological behaviors and plant hardware characteristics influencing the RCS and secondary transient thermal and hydraulic performance following a loss of the RHRS. Because of the wide variation in plant conditions during the shutdown modes of operation and given the multitude of equipment differences, these analyses should not be used directly to develop plant specific procedures. While the thermal hydraulic phenomena identified in this study are pertinent to understanding the general behavior for a range of loss for RHRS events, the timing and magnitude of the calculated system response parameters are expected to differ significantly among the various plant designs. The results of these analyses should therefore be used only as general guidance for identifying some of the areas and range of initial conditions warranting further plant-specific analyses to support development of plant-specific guidelines and procedures.

2.1 Use of Steam Generators for Decay Heat Removal

2.1.1 Model Description. The RELAP5/MOD3 model used to assess the consequences of a loss of RHRS event for the HBR-2 plant, which utilizes U-tube steam generators, is presented in Figures 1 through 3. This one-loop model containing the pressurizer and surge line was constructed by eliminating two of the three loops to simulate isolation of these components through the use of nozzle dams in the hot and cold leg piping. This elimination is made on the premise that only one steam generator is available for decay heat removal in the event the RHRS fails. The nozzle dams are also assumed to be installed at the entrance and exit to the steam generator inlet and outlet plenums. Also note that the hot leg piping was separated into two parallel components as shown in Figure 4. This nodalization was chosen to prevent the unrealistic accumulation of water in the steam generator active tube region caused by excessive entrainment by the steam penetrating the hot leg from the reactor vessel. The high interfacial drag between steam and water predicted by RELAP5 in the hot leg expelled unrealistically high amounts of water from this region toward the steam generators in the active loop, while also limiting the drainage of condensed and deentrained water toward the vessel from the steam generator tubes and inlet plenum. To more appropriately model this behavior, the hot leg and inlet plenum to the active steam generator was divided into two parallel components to allow drainage of the liquid from the steam generator and inlet plenum to the vessel upper plenum in the lower half of the hot leg, while steam exits the vessel toward the steam generator along the upper half. This choice of partitioning only limits the steam-water interaction and entrainment behavior to the upper portion of the hot leg piping. As discussed by Fletcher,[4] this modeling approach is considered appropriate for the fluid behavior expected from mid-loop operation with the liquid levels located at or below the hot leg centerline elevation. The work by Fletcher demonstrated that entrainment and carryover of the liquid in the hot legs would not be expected under these conditions should a loss of the RHRS occur no earlier than one day following shutdown from mid-loop operation.

The nodalization for the reactor vessel, primary loop, pressurizers and steam generators for the once-through steam generator analyses, based on the Oconee Plant, is presented in Figures 5 through 8.

Initial conditions for the loss of RHR analysis for the H. B. Robinson and Oconee plants are presented in Tables 1 and 2, respectively.

2.1.2 Analysis Results. This section presents the results of the analyses of the use of the steam

Table 1. Initial conditions for loss of RHRS analysis (H. B. Robinson Plant).

Parameter	Value
Power	2300 MW(t)
Decay heat fraction	0.5% (24 hours after shutdown)
Number of available steam generators	1
RCS liquid temperature	90°F (305 K)
Secondary liquid temperature	90°F (305 K)
Secondary liquid level	Top of active tubes
Secondary pressure	14.7 psia (0.101 MPa)
RCS liquid level	Center line of hot/cold legs
RCS pressure	14.7 psia (0.101 MPa)
RCS air temperature	90°F (305 K)
RCS air humidity	100%

generators to remove decay heat following a loss of the RHRS for the HBR-2 and Oconee plants. Analyses were performed for each of these plant types to investigate the impact of the different steam generator designs on transient thermal hydraulic behavior following a loss of the RHRS from mid-loop operation.

Table 2. Initial conditions for loss of RHR analysis (Oconee Plant).

Parameter	Value
Plant power	2568 MW(t)
Decay heat fraction	0.5% (24 hours after shutdown)
Number of available steam generators	1
RCS liquid temperature	90°F (305 K)
Secondary liquid temperature	90°F (305 K)
Secondary liquid level	Top of active tubes
Secondary pressure	14.7 psia (0.101 MPa)
RCS liquid level	Center line of hot/cold legs
RCS pressure	14.7 psia (0.101 MPa)
RCS air temperature	90°F (305 K)
RCS air humidity	100%

2.1.2.1 HBR-2, U-Tube Steam Generator Results. In evaluating the use of the steam generators for decay heat removal following a loss of the RHRS from mid-loop operation and other reduced inventory conditions for the HBR-2 plant, four cases were considered and include as follows:

- Case 1—The RCS is at mid-loop conditions with a water temperature of 90°F (305 K) and a liquid level at the hot leg centerline elevation. Air at 90°F (305 K) and 100% relative humidity is present in all volumes above the centerline of the hot and cold legs.

The decay heat power level corresponding to one day after shutdown is assumed [0.5% power or 11.5 MW(t)].

- Case 2—Identical to Case 1 except that the decay heat power level corresponding to one week after shutdown is assumed [0.3% power or 7.13 MW(t)].

- Case 3—Identical to Case 1 except that initial RCS liquid level is at the top elevation of the hot and cold legs.

- Case 4—Identical to Case 1 except the initial RCS level is at the elevation of the reactor vessel flange.

It should be mentioned that decay heat at one day following shutdown was chosen as an upper limit for the analyses. More importantly, mid-loop operation would not be expected to occur until at least 48 hours following shutdown.

For all of the cases, the secondary side of the steam generator was initialized with water at a temperature of 90°F (305 K) and upon the initiation of secondary boiling, a vent path was assumed to be available to maintain secondary pressure and temperature near atmospheric conditions. Also, after secondary boiling initiates, the eventual addition of auxiliary or emergency feedwater will also be necessary to prevent loss of the secondary as a heat sink and an uncontrolled pressurization of the RCS.

In comparing Cases 1 and 2, the insensitivity of RCS peak pressure to the time of loss of decay heat removal following shutdown is illustrated, while comparison of Cases 1, 3, and 4 reflect the sensitivity of peak RCS pressure to the initial RCS water level. A discussion of the results of each case is presented below.

2.1.2.1.1 Case 1—Loss of the RHRS at One Day—Figure 9 presents the pressure transient in the primary side of the steam generator when the RHRS is lost at one day following shutdown. The peak pressure of about 41 psia (0.28 MPa) was achieved at about 7000 seconds after the loss of RHRS. The peak pressure is reached at the time that the water temperature (tempf) in the steam generator secondary reaches saturation (sattemp), shown in Figure 10.

Table 3 presents the distribution of air in the RCS as a function of time. These results show that the air is predicted to be displaced from the vessel, hot leg piping, and steam generator inlet plenum and accumulated in the steam generator active tubes and outlet plenum volumes, pressurizer and suction leg piping by the steam generated in the core. Boiling in the core region is initiated at about 1250 seconds after loss of RHRS as shown in Figure 11, and develops when the liquid in the upper plenum and top portion of the core regions achieve saturation (sattemp), indicated by the water temperature (tempf) in the uppermost core volume (114-06).

Following the initiation of bulk boiling in the vessel, a condensing surface is formed once the steam has compressed the air in the RCS to a volume less than that of the steam generator active tube region. As a result, the primary heat transfer rate increases and secondary side boiling is achieved. Once secondary side boiling occurs and the secondary temperature stabilizes, the primary system pressure stabilizes at about 7000 seconds into the event. At this time condensation in the generator occurs primarily in the first tube volume above the tube sheet, with some condensation also occurring in the next active tube volume. Figure 12 presents the vapor void fraction in the first two volumes of the steam generator primary tubes (408-01 and 408-02) just above the tube sheet; Figures 13 and 14 present the heat flux in these steam generator primary tube regions. The primary side heat transfer coefficients for these volumes are presented in Figures 15 and 16. Because of the low decay heat fraction and the limited mixing of the air and steam in these first two steam generator active tube volumes above the tube sheet, primary pressure stabilizes because the condensation coefficient of about 528 BTU/hr-ft^2-°F (3000 W/m^2-K) after 7000 seconds is sufficient to remove decay heat at the primary to secondary temperature difference of about 35°F (19 K), which develops during the later portion of the transient (See Figure 14). The primary to secondary

Table 3. Distribution of air in the RCS for RELAP5/MOD3 Case 1 (use of steam generators for decay heat removal).

Component (cmp)	Mass of air (lb)			
	Initial	3600 (s)	7200 (s)	9000 (s)
Downcomer (cmp 100)	7.7	12.4	10.7	10.3
Downcomer (cmp 102)	2.6	8.5	13.7	13.4
Downcomer (cmp 104)	0.0	6.6	10.5	10.4
Downcomer (cmp 106)	0.0	4.8	12.4	14.7
Upper plenum (cmp 120)	8.8	<0.1	<0.1	<0.1
Upper plenum (cmp 122)	34.0	1.9	<0.1	<0.1
Upper head (cmp 126)	35.2	44.2	<0.1	0.03
Hot leg piping (cmp 404, 405, 504, 505)	6.0	<0.1	<0.1	<0.1
Steam generator inlet plenum (cmp 406, 506)	7.8	<0.1	<0.1	<0.1
Steam generator tubes (cmp 408-01)	5.6	1.1	<0.1	<0.1
Steam generator tubes (cmp 408-02)	5.6	7.7	5.9	6.7
Steam generator tubes (cmp 408-03 to 408-08)	33.9	50.4	61.4	62.5
Steam generator outlet plenum (cmp 410)	7.9	13.0	7.1	6.7
Pressurizer/surge line (cmp 340, 341, 343)	91.4	87.9	85.5	63.9
Cold leg piping (cmp 412, 414, 416, 418, 420)	16.8	25.5	58.4	56.9

temperature difference is illustrated in Figure 17. At 7000 seconds into the transient, the mass flow toward the steam generator in the upper portion of the hot leg pipe is balanced by an equal amount of downflow from the steam generator toward the vessel in the lower part of the hot leg. Figure 18 illustrates the establishment of this quasi-steady state mass flow in the upper and lower portions of the hot leg. The high mass flow rates are attributed to the high hot leg steam velocities (which entrain the water in the upper portion of the pipe, while deentrained) and condensed water accumulated in the steam generator inlet plenum and tubes flows back toward the vessel in the lower section.

Figure 19 presents the noncondensible mass fraction in the U-bend of the steam generator active tubes while Figure 20 presents the accompanying vapor temperature for this event. The decreasing mass fraction in the air region demonstrates that some steam has entered the cold side of the steam generator active tube region. The secondary side temperature for this event is also presented in Figure 20.

Also of particular importance is the RELAP5 transient pressure response, which displays an oscillatory behavior once the peak pressure of about 41 psia (0.28 MPa) is achieved after about 7000 seconds (see Figure 9). These oscillations are a result of the changes in steam velocity at the entrance to the steam generator tubes, which affect the condensation coefficient and void fraction in the first tube region above the tube sheet. The transition from countercurrent two-phase flow to slug flow as liquid builds up and drains affects the pressure drop and heat transfer characteristics at the tube entrance. The subsequent effect on the steam velocities in the active tubes causes the heat transfer coefficient and void fraction to vary, producing the oscillations in RCS pressure once the RCS pressure stabilizes near the peak value after 7000 seconds.

The possibility of pressure oscillations may be particularly important because they suggest that while RCS peak pressure for the above reflux boiling conditions may not exceed nozzle dam failure conditions, the oscillatory pressure behavior should they persist with time, may be sufficient alone to eventually dislocate a dam. If such conditions are realistic, the success of reflux boiling to mitigate the consequences of loss of the RHRS may not be judged on peak RCS pressure alone but on the amplitude and period of the peak pressure oscillations that could result later in the event.

2.1.2.1.2 Case 2—Loss of the RHRS at Seven Days—Figure 21 presents the pressure response in the RCS when the RHRS is lost seven days following shutdown. The peak pressure of about 38 psia (0.26 MPa) is reached at about 12,000 seconds after the loss of RHRS occurs. The peak pressure for this case is also reached at the time that the steam generator secondary reaches saturation as indicated by the behavior of the secondary water temperature (tempf) (see Figure 22).

Table 4 presents the distribution of air in the RCS as a function of time. These results show that again most of the air is predicted to be purged from the vessel, hot leg piping, and steam generator inlet plenum once core boiling begins. In general, the steam displaces the air into the high points of the RCS or steam generator and pressurizer regions. Unlike Case 1 (loss of the RHRS at one day), some air remains in the upper head due to the lower steam velocity in this region from the much lower decay heat level. Boiling in the core region begins at about 2600 seconds after loss of RHRS as shown in Figure 23, when the water temperature (tempf) in the uppermost core volume (114-06) is compared to the saturation temperature (sattemp).

Figure 24 illustrates the void fraction in the first two volumes of the steam generator primary tubes (408-01 and 408-02) and again shows that condensation in this portion of the steam generator develops sufficiently to remove decay heat when primary pressure reaches 38 psia (0.26 MPa). The heat transfer coefficients for these volumes are presented in Figures 25 and 26, while the primary to secondary temperature difference in the volume just above the tube

Table 4. Distribution of air in the RCS for RELAP5/MOD3 Case 2 (use of steam generators for decay heat removal).

Component (cmp)	Mass of air (lb)			
	Initial	3600 (s)	7200 (s)	12600 (s)
Downcomer (cmp 100)	7.7	11.5	13.2	16.6
Downcomer (cmp 102)	2.6	7.7	8.7	12.3
Downcomer (cmp 104)	0.0	5.9	6.4	9.2
Downcomer (cmp 106)	0.0	3.1	3.1	9.4
Upper plenum (cmp 120)	8.8	<0.1	<0.1	<0.1
Upper plenum (cmp 122)	34.0	2.9	4.4	0.02
Upper head (cmp 126)	35.2	37.6	33.2	8.7
Hot leg piping (cmp 404, 405, 504, 505)	6.0	0.03	<0.1	<0.1
Steam generator inlet plenum (cmp 406, 506)	7.8	<0.1	<0.1	<0.1
Steam generator tubes (cmp 408-01)	5.6	4.9	3.8	0.2
Steam generator tubes (cmp 408-02)	5.6	8.1	7.7	6.7
Steam generator tubes (cmp 408-03 to 408-08)	33.9	49.7	49.9	48.9
Steam generator outlet plenum (cmp 410)	7.9	11.7	13.5	18.2
Pressurizer/surge line (cmp 340, 341, 343)	91.4	89.7	87.6	88.9
Cold leg piping (cmp 412, 414, 416, 418, 420)	16.8	22.2	25.5	37.7

sheet is given in Figure 27. The figures showing heat flux in the steam generator and mass flow in the hot leg are not shown because this behavior is indicative of Case 1.

Based on a comparison of the Case 1 and 2 results, the peak pressure achieved following the loss of RHR can be considered to be relatively insensitive to decay power because the bulk of the RCS pressurization is a direct result of the need to compress the air volume sufficiently to enable the creation of an adequate condensing surface inside the active tube region of the steam generator. The pressure required to compress the air to this condition is relatively independent of power because the bulk of the pressurization of the RCS is needed to compress the air into the steam generator and outlet plenum regions. Lower decay heat power levels simply delay the initiation of boiling and consequently delay the time the peak pressure, in the range 38 to 41 psia (0.26 to 0.28 MPa), is reached.

2.1.2.1.3 Case 3—Loss of the RHRS at One Day with the Liquid Level at the Top of the Hot Leg—Figure 28 presents the pressure response in the RCS when decay heat is lost at one day following shutdown. Unlike Case 1, with the initial level at the hot leg centerline, the initial RCS water level for Case 3 is increased to the top elevation of the hot leg piping. As shown in Figure 28, following the loss of the RHRS and initiation of bulk boiling in the RCS, the peak pressure of about 40 psia (0.28 MPa) is reached at about 7500 seconds. The peak pressure again is reached after the time the water in the steam generator secondary reaches saturation as shown by the secondary behavior of the water temperature (tempf) in Figure 29. Comparison with Case 1 where the initial water level was at mid-loop operation demonstrates that the peak pressure is not sensitive to initial water levels between the hot leg midplane and top elevations. This is not surprising because with this initial water level, fluid expansion and the entrainment of water from the hot legs is still insufficient to plug the entrance into the steam generators to prevent the development of a condensing surface.

Table 5 presents the distribution of air in the RCS as a function of time. These results show that most of the air is predicted to be swept by the steam into the steam generator from the reactor vessel, hot leg, and inlet plenum once core boiling begins. Boiling in the core region begins at about 1500 seconds after loss of RHRS (see Figure 30) and is based on the water temperature (tempf) in the uppermost core volume (114-06).

Figure 31 presents the vapor void fraction in the first two volumes of the steam generator primary tubes (408-01 and 408-02) where condensation of the steam occurs. The heat transfer coefficients for these volumes are presented on Figures 32 and 33 while the primary and secondary water temperatures for the first tube volume above the tube sheet is presented in Figure 34.

2.1.2.1.4 Case 4—Loss of the RHRS at One Day with the Liquid Level at the Vessel Flange—Figure 35 presents the pressure response in the RCS when decay heat is lost at one day following shutdown with an initial RCS water level at the elevation of the reactor vessel flange. The Case 1 initial water level was assumed to be at the elevation of the centerline of the hot leg while Case 3 assumed the level was located at the top of the hot leg. With the level at the vessel flange, a peak pressure of about 95 psia (0.66 MPa) was achieved at about 25,000 seconds after the loss of RHRS (see Figure 35). The peak pressure again is reached at the time well after the water in the steam generator secondary reaches saturation, as shown by the behavior of the water temperature (tempf) (see Figure 36). As shown in Figure 37, boiling in the RCS initiates at about 2000 seconds when the water temperature reaches saturation in the uppermost core volume.

Comparison of Case 4 with Cases 1 and 3, where the initial water levels were at mid-loop and the top elevation of the hot leg piping, respectively, demonstrates that for water levels above the top elevation of the hot leg, the peak pressure achieved for these conditions could result in pressures well above the nozzle dam and other temporary boundary design conditions. With the much higher initial water level, fluid expansion and

Table 5. Distribution of air in the RCS for RELAP5/MOD3 Case 3 (use of steam generators for decay heat removal).

Component (cmp)	Mass of air (lb)			
	Initial	3600 (s)	7000 (s)	9000 (s)
Downcomer (cmp 100)	7.7	12.3	5.9	12.7
Downcomer (cmp 102)	0.0	8.7	12.6	12.3
Downcomer (cmp 104)	0.0	6.7	10.3	9.5
Downcomer (cmp 106)	0.0	2.4	9.3	17.6
Upper plenum (cmp 120)	0.0	0.0	<0.1	<0.1
Upper plenum (cmp 122)	34.0	4.4	<0.1	0.6
Upper head (cmp 126)	35.2	23.3	<0.1	1.6
Hot leg piping (cmp 404, 405, 504, 505)	4.8	0.0	<0.1	<0.1
Steam generator inlet plenum (cmp 406, 506)	7.8	0.0	<0.1	<0.1
Steam generator tubes (cmp 408-01)	5.6	0.9	<0.1	<0.1
Steam generator tubes (cmp 408-02)	5.6	8.3	5.5	6.4
Steam generator tubes (cmp 408-03 to 408-08)	33.9	52.7	60.5	50.6
Steam generator outlet plenum (cmp 410)	7.9	9.4	4.7	18.5
Pressurizer/surge line (cmp 340, 341, 343)	91.4	83.4	68.2	63.9
Cold leg piping (cmp 412, 414, 416, 418, 420)	0.6	21.2	41.9	21.6

level swell during the transient is sufficient alone to fill the entrance to the steam generator active tube region well above the tube sheet elevation. Figures 38 and 39 present the void fractions in the steam generator active tube region and show that all of the tube volume excluding the U-bend volumes (volumes 4 and 5) accumulate liquid after about 17,000 seconds into the event. In fact, as the RCS liquid heats up and expands, the flow rates and fluid velocities through the hot leg to the entrance to the steam generator tube region are initially too low to develop adequate steam generator heat removal to match decay heat generation. As a consequence, the RCS continues to pressurize until sufficient RCS pressure (and hence primary to secondary temperature difference) and fluid velocity in the steam generator is achieved to establish a heat removal rate that can accommodate the core decay heat generation. As shown in Figure 40, with the expanded time scale during the latter portion of this transient, the RCS has nearly ceased pressurizing as sufficient primary to secondary temperature difference and flow into the active tube region develops to match decay heat and slowly stabilize pressure near 95 psia (0.66 MPa).

The results of these analyses suggest that increasing the RCS liquid level above the top elevation of the hot legs could prevent the establishment of quasi-steady state reflux boiling heat removal capability at pressures below the failure limits of the temporary boundaries in the RCS.

Table 6 presents the distribution of air in the RCS as a function of time. For the earlier cases with the lower initial water levels, the higher air inventory in the RCS did not inhibit reflux boiling and peak RCS pressures remained in the range 38 to 41 psia (0.26 to 0.28 MPa). For Case 4 where the initial water level is at the reactor vessel flange, well above the hot leg elevation, peak RCS pressures approaching 95 psia (0.66 MPa) are predicted; the large inventory of subcooled liquid results in sufficient fluid expansion following a loss of the RHRS and plugs the steam generators with liquid.

The heat transfer coefficients for these volumes are presented in Figures 41 and 42. At 20,000 seconds (5.6 hrs) into the event, with the flow rate and resulting primary heat transfer coefficient established, the slowly increasing primary to secondary temperature difference gradually establishes a primary heat removal rate that can match the core decay heat generation rate. The primary and secondary temperatures are illustrated in Figure 43.

Although a natural circulation flow through the steam generators U-bend region to the cold side is not established during this event, the primary to secondary temperature difference, fluid velocity, and hence heat transfer coefficient in the first volume of the steam generator active tube region develops sufficiently to nearly stabilize RCS pressure at about 95 psia (0.66 MPa). Note that RCS pressure has not yet stabilized completely and displays approximately a one psi (0.01 MPa) increase over the last 5000 seconds of the event (see Figure 40). Although the RCS pressure continues to increase very slowly late in this event, the pressure is not expected to increase more than an additional 10 psi (0.07 MPa) because such an increase is expected to provide sufficient primary to secondary temperature difference to eventually stabilize RCS pressure.

2.1.2.1.5 Summary of Results for Cases 1 to 4—To summarize the results of the reflux boiling analyses, Figure 44 presents the RCS pressure profiles for Cases 1 and 2, illustrating the insensitivity of peak RCS pressure to time of the loss of the RHRS at one and seven days following shutdown. The relative insensitivity of peak RCS pressure to time of loss of RHRS, which dictates the decay heat power level, is due to the fact that approximately 39 psia (0.27 MPa) is required to compress the air in the RCS to a volume equal to the volume of the active tube and outlet plenum regions of the steam generator. The additional pressure increase needed to produce a condensing surface just above the tube sheet results in only a two to three psi (0.01 to 0.02 MPa) pressure increase. This result has also been confirmed and discussed in Reference 3.

Table 6. Distribution of air in the RCS for RELAP5/MOD3 Case 4 (use of steam generators for decay heat removal).

	Mass of air (lb)			
Component (cmp)	Initial	10800 (s)	18000 (s)	30000 (s)
Downcomer (cmp 100)	0.0	1.3	<0.1	2.2
Downcomer (cmp 102)	0.0	9.3	3.6	5.7
Downcomer (cmp 104)	0.0	5.3	7.2	5.6
Downcomer (cmp 106)	0.0	0.0	8.1	5.0
Upper plenum (cmp 120)	0.0	<0.1	<0.1	<0.1
Upper plenum (cmp 122)	0.0	<0.1	<0.1	0.9
Upper head (cmp 126)	35.2	0.2	<0.1	1.6
Hot leg piping (cmp 404, 405, 504, 505)	0.0	<0.1	<0.1	<0.1
Steam generator inlet plenum (cmp 406, 506)	0.0	<0.1	<0.1	<0.1
Steam generator tubes (cmp 408-01)	5.3	<0.1	<0.1	<0.1
Steam generator tubes (cmp 408-02)	5.6	0.1	<0.1	<0.1
Steam generator tubes (cmp 408-03 to 408-08)	33.5	46.7	46.3	50.5
Steam generator outlet plenum (cmp 410)	0.0	0.0	0.0	<0.1
Pressurizer/surge line (cmp 340, 341, 343)	89.6	19.4	15.4	14.9
Cold leg piping (cmp 412, 414, 416, 418, 420)	0.0	33.4	39.9	45.8

Figure 45 presents the sensitivity of the peak RCS pressure due to the variation in initial RCS water level for the condition where the RHRS is assumed lost at one day following shutdown. Clearly, for initial RCS water levels greater than the top elevation of the hot leg, peak RCS pressures greater than 95 psia (0.66 MPa) can be achieved. These results suggest that peak RCS pressure achieved following a loss of RHR event could increase dramatically when the initial water level in the system exceeds the top elevation of the hot leg. Peak pressures achieved from mid-loop operations were calculated in the range 38 to 41 psia (0.26 to 0.28 MPa) while pressures of 95 psia (0.66 MPa) may be possible when the initial water level is at the reactor vessel flange.

2.1.2.2 Oconee, Once-Through Steam Generator Behavior. In evaluating the use of the steam generators for decay heat removal following a loss of the RHRS from mid-loop operation for the Oconee plant, three cases were analyzed:

- Case 1—The RCS is at mid-loop conditions with a water temperature of 90°F (305 K) and a liquid level at the hot leg centerline elevation. Air at 90°F (305 K) and 100% relative humidity is present in all volumes above the centerline of the hot and cold legs. The decay heat power level corresponding to one day after shutdown is assumed [0.5% power or 12.8 MW(t)]. Also, the surge line was assumed to contain liquid. Both steam generators were assumed available for heat removal

- Case 2—This case is the same as Case 1 except the loop seal region of the surge line was assumed to be purged of liquid to allow air trapped in pressurizer to migrate into the RCS piping during the transient

- Case 3—This case is the same as Case 2 except only one steam generator is assumed available for heat removal.

Analyses were not performed at lower decay heat values because, based on the HBR-2 analyses presented above, peak RCS pressure for plants utilizing once-through designs is also expected to be relatively insensitive to decay power level or the time following shutdown of the loss of the RHRS. Also, studies were not performed to investigate the sensitivity of peak pressure to variations in RCS liquid level during reduced inventory operation for this plant type. The availability of emergency feedwater in addition to a secondary steam relief capability was assumed in these analyses. To maximize the heat removal capability of the steam generator, the secondary side was also assumed to be filled with water to an elevation near the inlet to the active tube region.

2.1.2.2.1 Case 1—Loss of the RHRS at One Day—Figure 46 presents the RCS pressure response when the RHRS is lost one day following shutdown. At about 8000 seconds, the RCS pressure response displays an oscillatory behavior that persists for the duration of the event. This behavior is a result of the secondary temperature behavior that shows an attendant oscillatory characteristic, which is due to the addition of emergency feedwater once secondary boiling begins. Figure 47 shows the secondary water temperature of the uppermost volumes (volumes 325 and 425) where the saturation temperature of 212°F is achieved at about 4000 seconds. The two-phase level is maintained in components 325 and 425 of the steam generator secondary and the remainder of the secondary side remains subcooled with the overall secondary behavior reflected in the temperature response for the next lowest secondary volume shown in Figures 48 and 49 for loops A and B, respectively.

Figures 48 and 49 display a rather erratic secondary temperature behavior that affects the primary to secondary temperature difference and therefore the primary heat removal rate. Because the feedwater is injected at the top of the secondary side, the non-uniform mixing in these lower regions produces the secondary temperature response, which in turn affects the primary heat removal and resultant RCS pressure. While there is uncertainty associated with the timing and magnitude of the calculated pressure oscillations, owing to the model or nodalization, emergency feedwater temperature, and operators actions, the importance of these results is that the secondary temperature and hence RCS pressure may be difficult to control or stabilize. The impact of the inability to stabilize RCS pressure may however be of little or no consequence because plants utilizing once-through steam generators do not generally employ hot leg nozzle dams or temporary thimble tube seals. Such concerns would only arise for those plant conditions where temporary RCS boundaries are employed.

Table 7 presents the distribution of air in the RCS as a function of time. These results show that the air is predicted to be displaced from the vessel, hot leg piping, and steam generator inlet plenum and accumulated in the steam generator active tubes and pressurizer. Once the air has been compressed into pressurizer and the steam generators, a condensing surface develops to limit the peak RCS pressure to approximately 55 psia (0.38 MPa).

The majority of the heat removal from the RCS occurs in the two uppermost primary volumes for the loop A steam generator. Figures 50 through 53 show the heat transfer coefficients for the loop A and B two uppermost primary volumes.

Figures 54 and 55 show the vapor void fractions for the loop A primary tube regions down to the active tube region containing the mid-loop liquid level. The noncondensible mass fractions for these regions are shown in Figures 56 through 58, showing that the air is compressed into the lower active tube regions just above the liquid level. These parameters for loop B are similar and are not presented.

The vapor void fraction and the noncondensible mass fraction in the pressurizer volumes are shown in Figures 59 and 60. The results show that air will migrate from the pressurizer to the primary system.

2.1.2.2.2 Case 2—Loss of the RHRS with the Surge Line Purged of Liquid—Analyses were performed to investigate the potential for the air in the pressurizer to enter the RCS hot leg piping and steam generators. Because the air is more dense than the steam, once steam has filled the hot legs, there is the potential for the air to propagate into the RCS piping. As the air enters the steam generators, RCS pressure could experience additional increases above the peak pressure achieved for Case 1 (loss of the RHRS at one day), because additional increases in pressure would be needed to develop a sufficient condensing surface in the steam generators to stabilize RCS pressure. Figure 61 presents the RELAP5/MOD3 model with the surge line separated into two parallel components. This nodalization was selected to allow air to exit through one portion of the surge line while steam could enter through the other.

The results of this case are similar to Case 1 (loss of the RHRS at one day). The small amounts of additional air that migrated into the loop piping and steam generators from the pressurizer did not appreciably affect the peak pressure (see Figure 62).

As in Case 1, the secondary water temperature in the uppermost secondary volume achieves saturation as shown in Figure 63 at about 4000 seconds. At this time emergency feedwater is introduced into both steam generators to maintain the secondary level at the top of the active tube region to maximize the heat removal. The two-phase level remains in the topmost steam generator secondary components 325 and 425. Again, most of secondary remains subcooled as reflected in the secondary water temperatures displayed in Figures 64 and 65.

Because the amount of additional air that migrates from the pressurizer is relatively small when compared to Case 1, the impact on the peak pressure is also small. The vapor void fraction and the noncondensible mass fraction in the pressurizer volumes are shown in Figures 66 and 67. Furthermore, Table 8 presents the air mass distribution in the RCS at the end of the event and comparison to Case 1 (see Table 7) shows that no appreciable amount of additional air exited the pressurizer for Case 2. Note that for Case 1, the model also predicted the migration of air from the pressurizer into the RCS.

Table 7. Distribution of air in the RCS for Oconee Case 1.

Component (cmp)	Mass of air (lb)		
	Initial	8100 (s)	18000 (s)
Downcomer (cmp 570)	0.0	0.4	0.3
Lower plenum (cmp 575, 505)	0.0	<0.1	<0.1
Core (cmp 510, 515)	0.0	<0.1	<0.1
Upper plenum (cmp 520, 525, 530, 535, 540, 545, 555, 560, 565)	46.3	17.4	12.8
Upper head (cmp 550)	33.4	<0.1	<0.1
Loop A hot leg piping (cmp 101, 105, 110, 113, 114)	33.6	<0.1	<0.1
Loop B hot leg piping (cmp 201, 205, 210, 213, 214)	33.6	<0.1	<0.1
Loop A SG inlet plenum (cmp 115)	19.9	<0.1	<0.1
Loop B SG inlet plenum (cmp 215)	19.9	1.4	<0.1
Loop A SG primary tubes (cmp 120, vol 1)	1.9	0.8	1.4
Loop A SG primary tubes (cmp 120, vol 2)	4.8	10.2	15.2
Loop A SG primary tubes (cmp 120, vol 3-10)	59.7	130.0	134.9
Loop B SG primary tubes (cmp 220, vol 1)	1.9	1.5	3.6
Loop B SG primary tubes (cmp 220, vol 2)	4.8	10.9	14.7
Loop B SG primary tubes (cmp 220, vol 3-10)	59.7	136.0	141.2
Loop A SG outlet plenum (cmp 125)	0.0	0.3	0.2
Loop B SG outlet plenum (cmp 225)	0.0	0.3	0.2
Loop A cold leg piping (cmp 130, 135, 140, 145, 150, 160, 165, 170, 175, 180)	18.4	23.5	24.0
Loop B cold leg piping (cmp 230, 235, 240, 245, 250, 260, 265, 270, 275, 280)	18.4	27.0	21.9
Surge line (cmp 600)	0.2	<0.1	<0.1
Pressurizer (cmp 610, 615)	97.9	110.6	110.9

Table 8. Air mass distribution in the RCS for Oconee Case 2.

Component (cmp)	Mass of air (lb)		
	Initial	8100 (s)	25200 (s)
Downcomer (cmp 570)	0.0	3.7	<0.1
Lower plenum (cmp 575, 505)	0.0	<0.1	<0.1
Core (cmp 510, 515)	0.0	<0.1	<0.1
Upper plenum (cmp 520, 525, 530, 535, 540, 545, 555, 560, 565)	46.3	15.9	16.2
Upper head (cmp 550)	33.4	<0.1	<0.1
Loop A hot leg piping (cmp 101, 105, 110, 113, 114)	33.7	<0.1	<0.1
Loop B hot leg piping (cmp 201, 205, 210, 213, 214)	33.7	<0.1	<0.1
Loop A SG inlet plenum (cmp 115)	19.9	0.5	<0.1
Loop B SG inlet plenum (cmp 215)	19.9	0.3	2.1
Loop A SG primary tubes (cmp 120, vol 1)	1.9	1.9	0.9
Loop A SG primary tubes (cmp 120, vol 2)	4.8	12.3	12.6
Loop A SG primary tubes (cmp 120, vol 3-10)	59.7	122.0	142.3
Loop B SG primary tubes (cmp 220, vol 1)	1.9	2.9	2.8
Loop B SG primary tubes (cmp 220, vol 2)	4.8	12.7	13.0
Loop B SG primary tubes (cmp 220, vol 3-10)	59.7	130.3	146.2
Loop A SG outlet plenum (cmp 125)	0.0	0.4	0.2
Loop B SG outlet plenum (cmp 225)	0.0	0.4	0.3
Loop A cold leg piping (cmp 130, 135, 140, 145, 150, 160, 165, 170, 175, 180)	18.5	24.2	18.3
Loop B cold leg piping (cmp 230, 235, 240, 245, 250, 260, 265, 270, 275, 280)	18.5	23.7	18.3
Surge line (cmp 600, 700)	1.4	<0.1	<0.1
Pressurizer (cmp 605, 705, 610, 615)	104.0	134.2	134.1

2.1.2.2.3 Case 3—Loss of the RHRS with Heat Removal Through One Steam Generator—Analyses were performed to investigate the impact on peak pressure of using only one steam generator for decay heat removal. In this case, the steam generator heat transfer removal capability is halved once the loop B secondary boils dry.

Figure 68 presents the pressure transient in the RCS when the RHRS is lost one day following shutdown. As in the previous two cases, oscillations in RCS pressure are predicted to occur during the event. As shown in the figure, the peak RCS pressure of 75 psia (0.52 MPa) is achieved for this case. Note that at 30000 seconds into the event, RCS pressure is still increasing. Figure 69 presents the pressure rise over the last three hours of this event and shows about a 10 psi (0.07 MPa) increase, which indicates a very slow rate of pressurization. RCS pressure is not expected to increase more than an additional 10 psi (0.07 MPa) because this increase would produce a primary to secondary temperature difference sufficient to stabilize RCS pressure. As such, the analysis was not continued for this case.

The temperature in the secondary volumes are displayed in Figures 70 through 73 and demonstrates that the bulk of the secondary side reaches the saturation temperature of 212°F (373 K) at about 10,000 seconds into the event. The two-phase level is maintained in the uppermost secondary components 323 and 373.

As in the previous cases, the majority of the heat removal from the RCS occurs in the first two volumes of the primary steam generator tubes near the inlet. The heat transfer coefficients in these regions are presented in Figures 74 and 75.

The vapor void fraction and the noncondensible mass fraction in the pressurizer volumes are shown in Figures 76 and 77, respectively. The results show that the air migrates from the pressurizer into the primary loop.

2.2 Failure of Instrument Tube Thimble Seals

2.2.1 Model. The instrument tube thimble seal failure analysis was performed to determine the time that the core uncovers in the event of thimble seal failure in the instrument tubes when one steam generator is used to remove decay heat. The RELAP5/MOD3 model used for determining the time to core uncovery is based on the one-loop model described in Section 2.1 for the HBR-2 plant, while the instrument tube geometry was based on the Zion plant design. Although this particular analysis is not applicable to the HBR-2 plant design, the results of the analysis are of importance as a scoping evaluation to highlight the approximate timing of core uncovery following failure of all of the temporary thimble tube seals.

Because the purpose of this calculation is to identify the time to core uncovery, the number of volumes representing the core is increased from six to twelve volumes to better approximate the axial void profile in the core and better track the two-phase level in this region. Aside from the change in the number of core nodes, the vessel nodalization used in the initial HBR-2 model described in Section 2 was used. The one-loop primary and secondary side nodalization of Figures 2 through 3 applicable to the steam generator decay heat removal analysis above was also used for this analysis.

Thimble seal failure in the instrument tubes is assumed to occur when system pressure reached 35 psia (0.24 MPa). This value was chosen arbitrarily to investigate the consequences of failure of the thimble seals and may not reflect actual failure pressures for seals. This pressure does however, represent the maximum calculated pressure reached in the instrument tube at the seal table when decay heat removal by one steam generator is assumed. For this analysis, it is assumed that there are 58 thimble seals and all of these seals arbitrarily fail once the assumed failure pressure is achieved. The break flow area was based on an empty instrument tube with an inner diameter of 0.4 in. (0.0101 m), resulting in a total break area of 0.0506 ft^2 (0.0047 m^2). The failure location is assumed to be at the seal table at the elevation of the reactor vessel flange. The tubes are connected to the vessel at the bottom of the lower head and are collected at the seal table resulting in an elevation difference between these two locations of about 22.5 ft (6.86 m).

The RCS was initialized with water at 90°F at a level at the centerline of the hot and cold legs. Air at 90°F and 100% relative humidity is present in all volumes above the centerline of the hot and cold legs. The decay heat power level corresponding to one day after shutdown was assumed for this analysis [11.5 MW(t)].

2.2.2 Results. The assumed thimble seal failure pressure was predicted to occur at about 5900 seconds after loss of RHRS occurs (see Figure 78). The approach to core uncovery at about 6700 seconds after loss of RHRS, is indicated by the increase in void fraction to 1.0 in the upper plenum region just above the core (see Figure 79). Figure 80 presents the void fraction in the top three core volumes showing core uncovery at about 7000 seconds. These results suggest that while there is 1-1/2 hours to achieve the assumed failure pressure of 40 psia (0.28 MPa), there is less than 15 minutes to core uncovery once the temporary thimble seals fail. Clearly, this result is an upper bound because the time to uncovery for those plants with smaller tube areas or fewer temporary thimble tubes in use could be significantly delayed beyond that calculated for this prescribed sequence.

The break mass flow rate is shown in Figure 81, while Figure 82 presents the break flow void fraction.

2.3 Failure of Nozzle Dams

2.3.1 Model. The nozzle dam failure analysis was performed to determine the time that the core uncovers assuming a failure of the hot leg nozzle dam occurs in the steam generator with a manway cover removed from the inlet plenum during mid-loop operation. For this analysis, it was assumed that nozzle dams have been installed in all hot and cold legs, resulting in isolation of all steam generators from the RCS.

The RELAP5/MOD3 model used for determining the time of core uncovery when nozzle dam failure occurs is based on the HBR-2 plant model described in Section 2 except that the steam generators are removed from the model because it is assumed that nozzle dams are installed in all hot legs resulting in isolation of the steam generators from the RCS. Because the reactor coolant pumps are not operating under these conditions, each pump was replaced by a branch component utilizing two junctions corresponding to the pump inlet and outlet. A forward loss coefficient was used in the junctions to simulate a locked rotor condition. The pressurizer and surge line components are also retained.

Nozzle dam failure in the instrument tubes is assumed to occur when system pressure reached 40 psia (0.28 MPa). The assumed failure location is at the hot leg entry into the steam generator inlet plenum in Loop A. Because the steam generator manway is removed, there is a direct flow path to the containment upon nozzle dam failure. The manway diameter is 16 in. (0.4 m).

In this analysis, the RCS was initialized with water at 90°F at the level corresponding to the centerline of the hot and cold legs. Air at 90°F and 100% relative humidity was assumed present in all volumes above the RCS level. Two cases were analyzed and included a decay heat power level corresponding to one day after shutdown [11.5 MW(t)] and seven days after shutdown [7.13 MW(t)]. The results of these analyses are discussed in the following sections.

2.3.2 Results.

- Case 1—Nozzle dam failure at one day following shutdown

Nozzle dam failure was assumed to occur when RCS pressure reached 40 psia (0.28 MPa). Failure was predicted to occur at about 2000 seconds after loss of RHRS (see Figure 83). The approach to core uncovery is indicated by the transient void fraction in the upper plenum volume just above the core, where complete voiding occurs at about 4000 seconds into the event (see Figure 84). Core uncovery occurs at about 4000 seconds after loss of RHRS, as indicated by the increase in void fraction in the top three core volumes (see Figure 85). Core uncovery following nozzle dam failure occurs rather quickly (i.e., in about 30 minutes) because the steam velocities in the hot leg are sufficient to entrain and carry liquid from the hot legs and vessel out the open manway in the steam generator. Uncovery is therefore earlier in this event than if boiloff alone is credited for reducing the liquid inventory above the top elevation of the core.

The break mass flow rate and void fraction are shown in Figures 86 and 87, respectively.

- Case 2—Nozzle dam failure at seven days following shutdown

An RCS pressure of 40 psia (0.28 MPa) and an assumed nozzle dam failure was achieved at about 3900 seconds after loss of RHRS at seven days following shutdown (see Figure 88). The void fraction in the upper plenum volume just above the core is shown in Figure 89. Core uncovery was predicted to occur at about 9000 seconds after the loss of the RHRS, as indicated by the increase in void fraction in the top three core volumes (see Figure 90). Because of the much lower decay heat power levels, the later uncovery time of about one hour following the nozzle dam failure is attributed to the much lower liquid entrainment and boiloff rates experienced for these conditions.

The break mass flow rate is shown in Figure 91 while Figure 92 presents the break flow void fraction.

2.4 Safety Injection System Line Open Due to Check Valve Maintenance

2.4.1 Model. This analysis was performed to determine the time of core uncovery assuming a check valve has been removed from the safety injection (SI) system line connected to the cold leg during mid-loop operation. With the open SI line, there is a direct flowpath from the cold leg at the reactor coolant pump discharge to the containment. For this analysis, it was also assumed that nozzle dams have been installed in all hot legs, resulting in isolation of all steam generators from the RCS.

The RELAP5/MOD3 model used for this analysis is the same as that described for the nozzle dam failure analysis with a break path in the discharge pipe to the reactor coolant pump added. The SI line is modelled as a pipe component connected to the cold leg in Loop C. The diameter of this line is 8.5 in. (0.22 m). The length of the line is assumed to be 5.0 ft (1.52 m) from the top elevation of the cold leg at the reactor coolant pump discharge leg to the point where the check valve is removed. The line is assumed to extend vertically from the cold leg. The vertical distance from the top of the core to the location of the removed check valve was assumed to be approximately 9 ft (2.7 m).

The initial RCS water temperature of 90°F (305 K) was assumed with the RCS level located at the centerline of the hot and cold legs. Air at 90°F (305 K) and 100% relative humidity was assumed present in all volumes above the centerline of the hot and cold legs. The decay heat power level corresponding to one day after shutdown was assumed for this analysis [11.5 MWt].

2.4.2 Results. Figure 93 presents the RCS pressure transient. Voiding in the upper plenum region is illustrated in Figure 94 indicating the approach to uncovery as the void fractions increases to a value of 1.0 (indicating pure steam) toward the latter portion of the event. Core uncovery was predicted to initiate at about 4000 seconds after loss of RHR occurs as indicated by the void fraction in the top three core nodes (see Figure 95). As shown in Figure 95, bulk boiling in the core is initiated at about 2000 seconds at which time fluid is expelled from the opened SI line as indicated by the break flow shown in Figure 96. While core uncovery does not occur until about 4000 seconds, RCS fluid expulsion to the containment begins at 2000 seconds.

The break flow void fraction is presented in Figure 97.

To illustrate the transient response to changes in the initial core outlet temperature, wall heat transfer, and vessel upper plenum to annulus leakage path area, an additional calculation was performed for this event with the following modified assumptions:

- The upper plenum and hot leg initial fluid temperature was increased from 90°F (305 K) to 140°F (333 K)
- The leakage paths connecting the upper plenum and upper head regions of the reactor vessel to the annulus were eliminated
- The wall heat structures in the RCS were eliminated so that no heat could be absorbed by the RCS vessel and loop piping components.

With the these assumptions, the loss of the RHRS analysis with the open SI line was reanalyzed with the resulting RCS pressure response given in Figure 98. Core uncovery was calculated to occur at about 600 seconds into the event as illustrated by the void fraction in the core (see Figure 99). The void fraction in the upper plenum volume just above the core is shown in Figure 100. Comparison with the preceding case shows the above assumptions can have a significant impact on the results. These results illustrate the sensitivity of the transient response to changes in the leakage paths between the upper plenum and annulus regions of the reactor vessel. These leakage paths can significantly delay core uncovery by slowing the initial pressurization as a result of venting the core generated steam into the annulus and out the open SI line. Wall heat effects act to condense steam, further reducing the pressurization rate, while the lower initial temperature delays the initiation of boiling. The importance of these results is that plant behavior can be very sensitive to what may appear to be small differences in the design characteristics and initial conditions. Moreover, the use of conservative assumptions to bound the behavior for a class of plants may place unnecessary restrictions on the strategies and procedures to be developed, not only for this particular sequence, but the other events evaluated in this report as well.

2.5 Consequences of Loss of the RHRS with Upper Core Support Plate in Position

2.5.1 Model. The consequences of loss of the RHRS with the upper core support plate in position was evaluated because of the possibility of core uncovery during the refilling of the refueling pool cavity during preparation for removal of the spent fuel. The analysis was performed to investigate the potential for holdup of the water above the support plate by steam generated in the core. If the steam velocities reach the flooding limit for countercurrent flow through the extremely small flow area through the plate, core uncovery could occur. Figure 101 presents the HBR-2 vessel showing the location of the upper support plate, while Figure 102 illustrates the vessel and refueling pool cavity for a typical dry containment.

The RELAP5/MOD3 model was used for analyzing the flow characteristics through the upper core support plate (see Figure 103). To better accommodate single and two-phase natural circulation in the vessel during this event, the upper plenum and core regions were represented with three radial regions. As noted in Figure 103, the core was divided into three radial regions consisting of the center hot region, a middle annular region, and an outer low power region. This radial nodalization was then applied to the upper plenum region between the top of the core and upper support plate. The refueling pool and containment volumes were also added where the pool was assumed to contain liquid to a depth of 25 ft (7.62 m) above the upper core support plate (see Figure 103). The depth of water above the top of the core is about 35 ft (10.7 m).

It is also important to note that the flow nozzles connecting the reactor vessel head with the upper annulus were also included in the model. These flow paths consist of 16 flow nozzles that penetrate the barrel with a combined flow area of about 0.02 ft^2 (0.0018 m^2). The total flow area through the upper core support plate for the HBR-2 plant is 2.3 ft^2 (0.214 m^2). This total flow area consists of flow paths directly through the plate with a combined area of 1.3 ft^2 (0.121 m^2) and the flow area from the guide tubes through the plate with a combined area of 1.0 ft^2 (0.093 m^2). Because some plants may have sealed the flow leakage paths directly through the plate, leaving only the flow paths at the exit to the guide tubes, two cases were evaluated and include:

- Case 1—Loss of the RHRS with the flow paths through the plate at the guide tubes and the flow paths directly through the plate assumed to be open. This results in a total flow area of 2.3 ft^2 (0.214 m^2) through the upper support plate
- Case 2—Same as Case 1 except the flow paths directly through the plate are assumed to be sealed. This results in a total flow area of 1.0 ft^2 (0.093m^2) and consists of only the leakage paths through the plate just above the guide tubes.

The initial RCS water temperature for these analyses were assumed to be 90°F (305 K) in the RCS and the refueling pool cavity. The hot and cold leg nozzle dams were assumed to be in place while no vent paths were available in the pressurizer. The decay heat power level corresponding to one day after shutdown [11.5 MW(t)] was also assumed. The results of the analyses are discussed below.

2.5.2 Results

- Case 1—Flow holes directly through the upper support plate open

The RELAP5/MOD3 analysis, with the upper support plate assumed to remain in position during refilling of the refuel pool cavity, predicted that the core will remain covered with a two-phase mixture. This conclusion is supported by the results presented in Figure 104 which shows the void fraction in the upper plenum region just above the top of the core. The void fraction of about 20% in the upper plenum just above the top of the core demonstrates the core remains covered with a two-phase mixture during this event. Figure 105 presents the void fraction in the volume just below the support plate. The variation in void fraction just below the plate (between 80 to 90%) indicates that nearly complete voiding occurs in the region just below the plate.

The liquid temperatures in the upper plenum and refueling pool cavity above the upper support plate are presented in Figure 106. Note that the refueling pool cavity temperature is not a good indication of the temperatures in the upper plenum and core regions that achieve saturation early in the event. This is attributed to the limited flow circulation between the pool water and the upper plenum as a result of the small leakage paths through the support plate.

Although the flow between the refueling pool cavity and upper plenum is restricted through the plate, once boiling initiates, steam generated in the core is vented through the support plate while liquid downflow through the flow nozzles and outer region guide tubes are sufficient to maintain the core covered with a two-phase mixture throughout the event. The liquid downflow through the annulus flow nozzles and guide tubes matches the core decay heat steaming rate during this event thereby preventing long-term uncovery of the core.

RCS pressure during this event remains at about 25 psia (0.17 MPa) throughout the transient and is not presented.

- Case 2—Flow holes directly through the plate are sealed

With the flow leakage paths sealed through the plate, interaction of the fluid in the upper plenum with the refueling pool cavity is limited to only the flow through the plate at the exit of the guide tubes. As a consequence, once boiling begins, steam relief from the vessel upper plenum can only be achieved through the guide tube flow leakage paths through the upper support plate.

As in Case 1 (flowholes directly through the Upper Support Plate open), the core and upper plenum temperatures display a near adiabatic heatup because of the restricted flow through the plate (see Figure 107). As shown in Figure 107, the cavity refueling water temperatures exhibit a much slower rate of increase in temperature. The significance of this result is that, as in Case 1 above, the refueling cavity temperature is not a valid indication of the core and upper plenum temperature during a loss of the RHRS because of the flow restrictions in the upper internals. Figure 107 also shows that the pool displays a lower rate of heatup as hot fluid from the vessel expands into the pool, however, the temperature in the refueling cavity does not reflect the actual time at which boiling would occur in the core and upper plenum.

Figure 108 shows that the buildup of steam pressure in the upper plenum and intermittent relief through the guide tubes causes oscillations in the vessel pressure once boiling occurs. Again, as in the first case with the flow holes opened, core uncovery is not predicted to occur during the first six hours of this event because the intermittent buildup of pressure and relief of steam through the guide tubes allows a periodic voiding and refilling of the upper plenum with subcooled water from the cavity through both the guide tubes and upper annulus nozzles. Because there is no direct flow interaction between the upper plenum through the plate with the flow leakage paths closed, pressure increases in this region causing a substantial voiding of the upper plenum. The pressure increases in the upper plenum to depress the two-phase level to an elevation just below the bottom elevation of the guide tubes, which allows steam to vent through this region. Before core uncovery would occur during the early portion of this event however, steam relief through the guide tubes must reduce the vessel pressure and allow subcooled water to once again enter the upper plenum and core, causing the cyclic voiding and refilling of the upper plenum. This pressure buildup and relief is marked by the occurrence of the cyclic pressure pulses displayed in Figure 108. The temperature in the upper plenum, shown in Figure 109, also displays an oscillatory behavior during the early portion of the event as the upper plenum voids and refills during the transient. While core uncovery does not occur until six hours into the transient, a potential concern is that the pressure pulses over an extended period of time could cause a nozzle dam or temporary thimble seal to eventually fail. If such a failure occurred, the refueling pool cavity water would drain through the open manway or other temporary boundaries such as the thimble seals into the containment.

Figure 110 provides data on the void distribution in the top four volumes of the core and shows that core uncovery occurs six hours into the event. The core average clad temperature in the top four volumes in the core is shown in Figure 111 while the channel steam temperature responses in these regions of the core are shown in Figure 112. Cladding temperatures are calculated to be <1500°F (1088 K), however operation under such conditions for extended periods of time could cause excessive oxidation of the fuel and eventual failure of the cladding. The core two-phase level is located about 4.5 ft (1.37 m) below the top of uncovery at the end of this transient, allowing core uncovery to slowly increase for this event because no liquid downflow occurs through the guide tubes; the steam velocities in this region exceed the 53 ft/s (16 m/s) velocity required to hold up the liquid in the pool cavity. The liquid downflow through the annulus flow nozzles is also unable to prevent uncovery at this time because the leakage paths from the upper plenum through the hot leg penetration in the barrel allows pressurization of the upper annulus which prevents liquid downflow into this region. As a consequence, some additional core uncovery is expected for this case. The core is expected to continue to uncover until the steaming rate is reduced sufficiently to allow the downflow of liquid, from the refuel pool through either the annulus flow nozzles and/or the guide tube leakage paths through the plate to stabilize the core two phase level.

The importance of these results is that the analyses suggest the potential for core uncovery for plants with upper internal packages with small leakage path areas [i.e., approximately 2.0 ft^2

(0.186 m^2) and smaller] between the upper plenum and upper head regions. These findings suggest that the leakage through the upper internals could vary significantly upon removal of the upper head and given the wide variation in plant internal designs, core uncovery may be minimal or precluded for some plants. In the event these leakage paths are limited to the smaller flow areas, the analyses show that several hours are required before core uncovery is predicted, allowing sufficient time to take the appropriate actions to prevent boiling and core uncovery for such an event. However, in view of the uncertainty in these leakage paths and given the many different plant designs and operating conditions, the results of these analyses should not be used to develop plant-specific procedures. Rather, they should be used as a guide to identify the areas where more plant-specific analyses could be conducted to quantify both the timing and magnitude of the calculated response parameters.

In view of these analyses, the following important conclusions can be summarized regarding the loss of the RHRS with the internals in place:

- The heatup rate of the coolant in the core and upper plenum is not reflected in the refueling cavity water temperature behavior because the flow restrictions through the upper support plate do not promote sufficient natural circulation between the core and cavity regions.
- Significant voiding can occur in the upper plenum and for those plants with limited flow through the upper support plate, there is a potential for core uncovery. Pressure oscillations may also occur which, over an extended period of time, suggest that the integrity of the temporary RCS boundaries could be challenged.
- Plant-specific analyses should be conducted to assess the impact of RCS vents to preclude core uncovery for plants with low leakage through the upper internals. For example, plant-specific analyses could show that vent paths in the pressurizer may be sufficient to prevent core uncovery following a loss of the RHRS during flooding of the refuel pool.

In view of these considerations, the following recommendations are made:

- For those plants incorporating upper support plates with limited flow leakage paths, the upper support plate should be removed (if possible) as soon as the head has been removed.
- If the support plate must remain in position during filling of the refueling pool cavity, then procedures should be in place to quickly remove the plate before boiling in the vessel would occur.
- During filling of the refueling cavity with the plate in position, a means of measuring the upper plenum or core exit coolant temperature should also be available. Refueling cavity pool temperature is not a good indication of the upper plenum and core temperatures following a loss of the RHRS under these conditions.
- All possible steps should be taken to prevent boiling with the vessel internals in place. For example, both RHR systems could be kept operable if the internals are in place.
- As discussed in Section 2.6, should boiling occur for two hours or more, the addition of borated water will keep the core covered and cooled, however, the subsequent operation of the RHRS could cause an inadvertent precipitation of boric acid in the core and/or RHR piping. Procedures for adding water to the core during boiling should be considered, however, the prevention of the boric acid precipitation also needs to be addressed.

2.6 Consequences of Borated Water Addition to the Reactor Vessel

If the steam generators cannot be used for decay heat removal, the eventual bulk boiling of the liquid in the core and subsequent injection of borated water for core cooling will cause boric acid to accumulate in the reactor vessel. A method to compute the buildup of boric acid following an extended period of boiling was developed where the time rate of change in boric acid concentration in the reactor vessel can be expressed as (see nomenclature for definition of terms)

$$\frac{d}{dt}C = \frac{W_{in}}{\rho V_{mix}}C_s \tag{1}$$

Solving Equation (1) given that the initial boric acid concentration in the vessel is C_0, the boric acid concentration in the vessel at any time, t, is simply

$$C(t) = C_o + \frac{W_{in}t}{\rho V_{mix}}C_s \tag{2}$$

where the mixing volume, V_{mix}, is taken as the volume of the lower plenum, core and upper plenum regions containing liquid.

The steaming rate in the core establishes the rate of accumulation of boric acid in the vessel. That is, the rate at which borated water enters the core is equal to the core steaming or boil-off rate occurring from decay heat generation. The accumulated liquid at time, t, in the vessel in response to the core decay heat boiloff is defined as $W_{in}t$ in Equation (2), and can also be expressed as

$$W_{in}t = \frac{P_o}{h_{fg}}\int_0^t F(t)dt \tag{3}$$

Figure 113 presents the boric acid concentration versus time following loss of the RHRS. The RHRS was assumed to be lost one day following shutdown while the mixing volume was based on the liquid volume in lower plenum, core and upper plenum, only. Analyses were performed using two sources of injection: a low concentrate source, such as the refueling water storage tank, (RWST) with a concentration of 2200 ppm and a high concentrate source, such as the boric acid storage tank, with a concentration of approximately 22,000 ppm.

If boiling is allowed to persist for six hours or more using the RWST as a source of injection, boric acid concentrations in excess of 6% could result. In the event higher concentrate water is used for core cooling, such as the much higher boric acid content of the boric acid storage tanks (BAST), the rate of accumulation of boric acid would be much more rapid than that illustrated in Figure 113. The BASTs have water concentrations as much as 10 times that of the RWST, and would therefore more rapidly increase the boric acid content in the vessel. If the source of injection is the RWST, Figure 113 shows that if the RHRS is restored and decay heat removal is reestablished, the reduction in the RCS liquid temperature to values below the precipitation temperature will precipitate the boric acid in the vessel and piping of the shutdown cooling system. Figure 113 also shows the precipitation temperature versus the boric acid concentration during the event and shows that if RCS temperature is reduced below 100°F (311 K) after about six hours, precipitation could occur when the source of injection water is a low concentrate source, such as that for the RWST. In the event a high concentrate source of water is used, such as that of the BAST with concentrations of 22000 ppm and higher, precipitation conditions could be achieved in less than two hours as noted in Figure 114. Figure 114 shows that upon drainage of the BAST, the injection source is switched to the RWST at about three hours. The lower boron concentration of the RWST results in a lower rate of increase in boron concentration in the vessel for the remainder of the event.

The results of these calculations indicate that following a loss of the RHRS, should boiling in the reactor vessel occur over an extended period of time, the reestablishment of the RHRS for decay heat removal should not be attempted until the boric acid concentration in the vessel is reduced to levels below the precipitation temperature corresponding to the desired operating temperature of the shutdown cooling system.

These results further suggest that caution in using a source of high concentrate boric acid for core cooling should be exercised, because the addition of a high concentrate source could initiate a very early precipitation if boiling cannot be terminated. Also, should the injection water be switched to a colder source, the addition of cold water to the vessel after an extended boiling period could reduce the vessel fluid temperature and also cause an inadvertent precipitation.

The buildup of boric acid following a loss of the RHRS, when the reactor vessel internals have been removed and the refuel pool above the vessel has been flooded to normal refueling levels, is not expected. While such concerns would be applicable to those conditions where the refuel pool is being filled with the vessel internals in place, the potential for boric acid buildup and its consequences suggest that operation at reduced refuel pool levels should not be sustained for long periods of time without contingency or emergency procedures to address a loss of the RHRS. More importantly, in view of the potential consequences, procedures and euidance should be put in place that emphasize the prevention or quick termination of boiling.

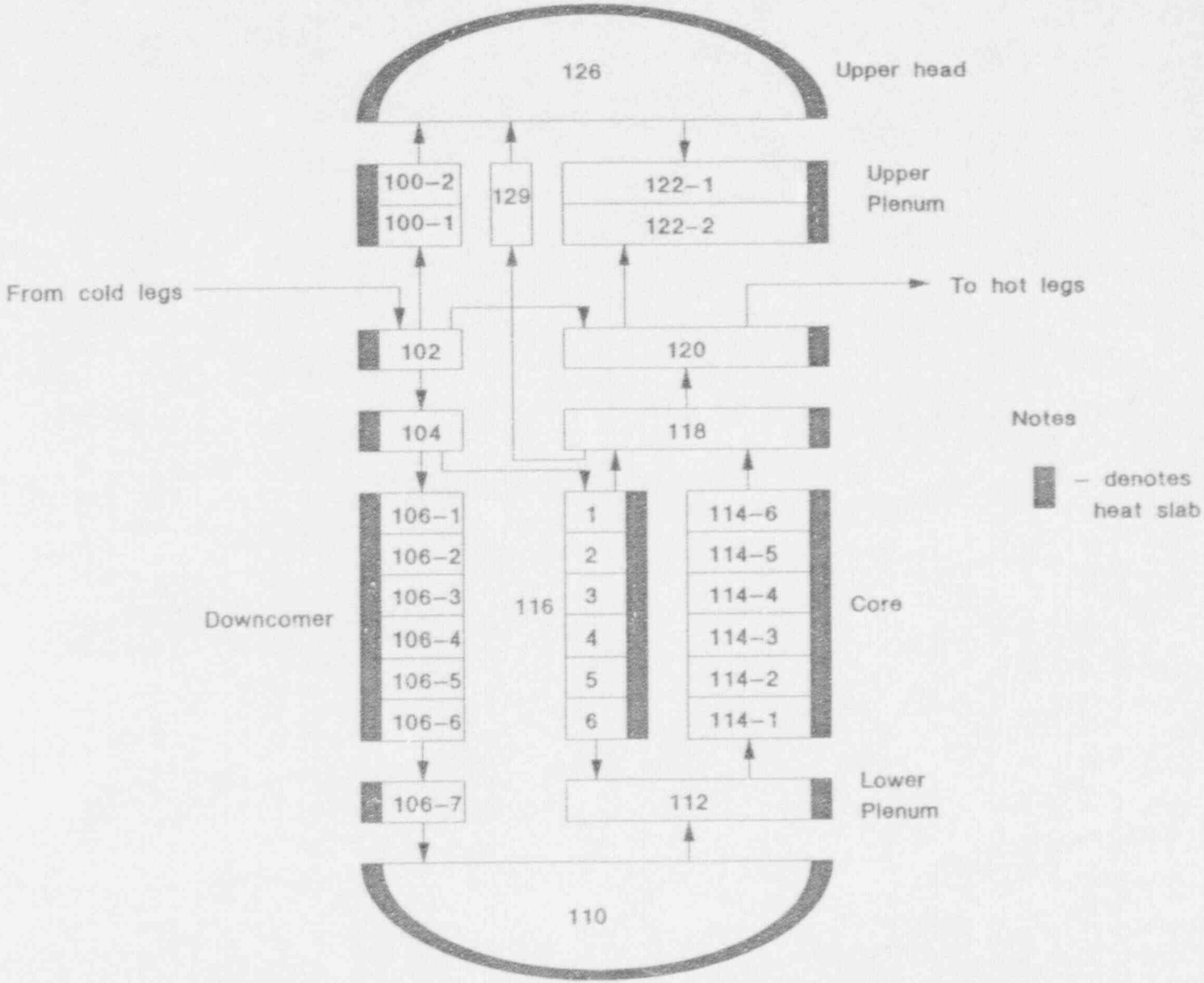

Figure 1. Nodalization of the reactor vessel for the U-tube steam generator analysis.

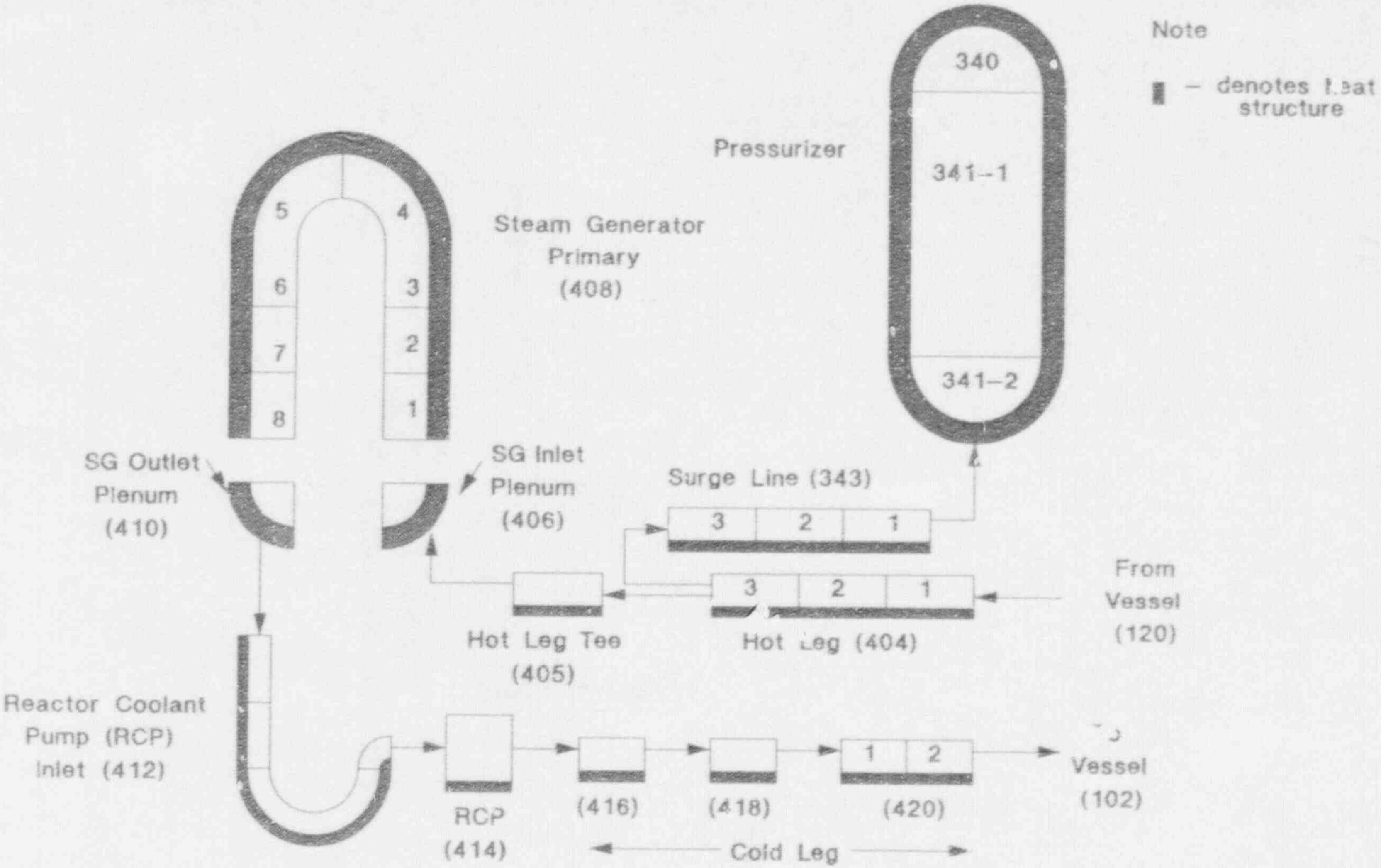

Figure 2. Nodalization of primary Loop C for the U-tube steam generator analysis.

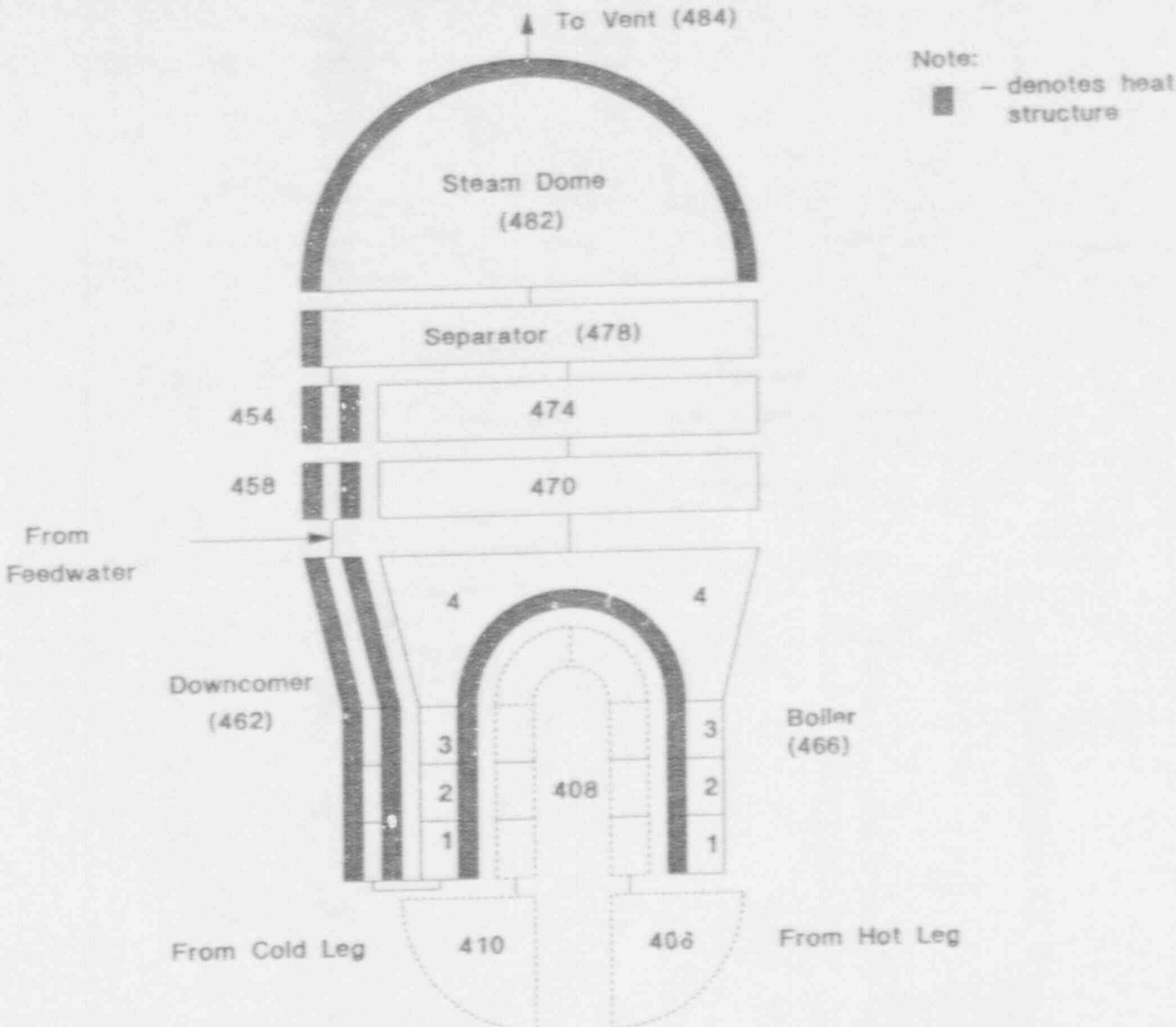

Figure 3. Secondary nodalization for U-tube steam generator analysis.

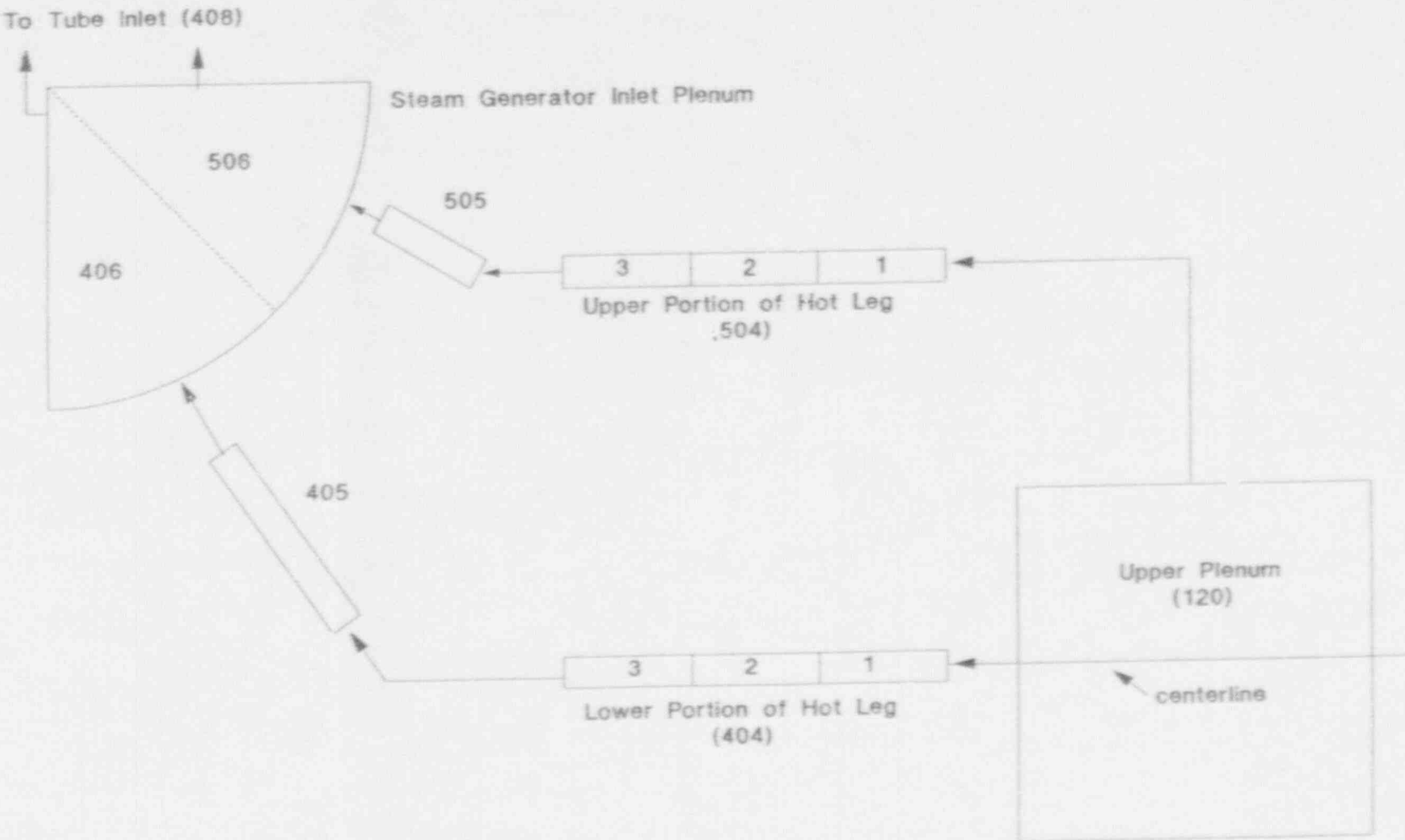

Figure 4. Dual hot leg nodalization for U-tube steam generator analysis.

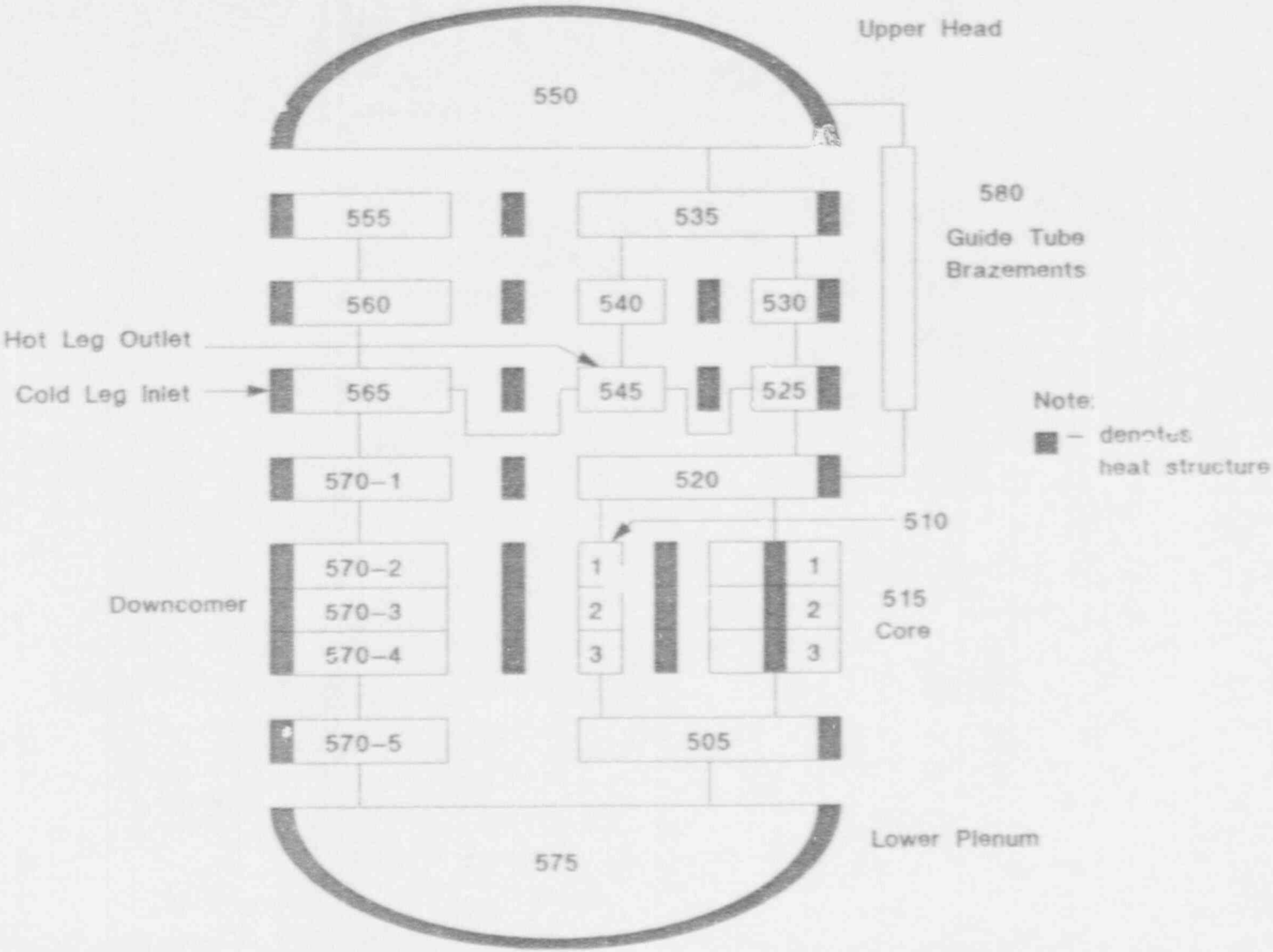

Figure 5. RELAP5/MOD3 vessel model for once-through steam generator analysis.

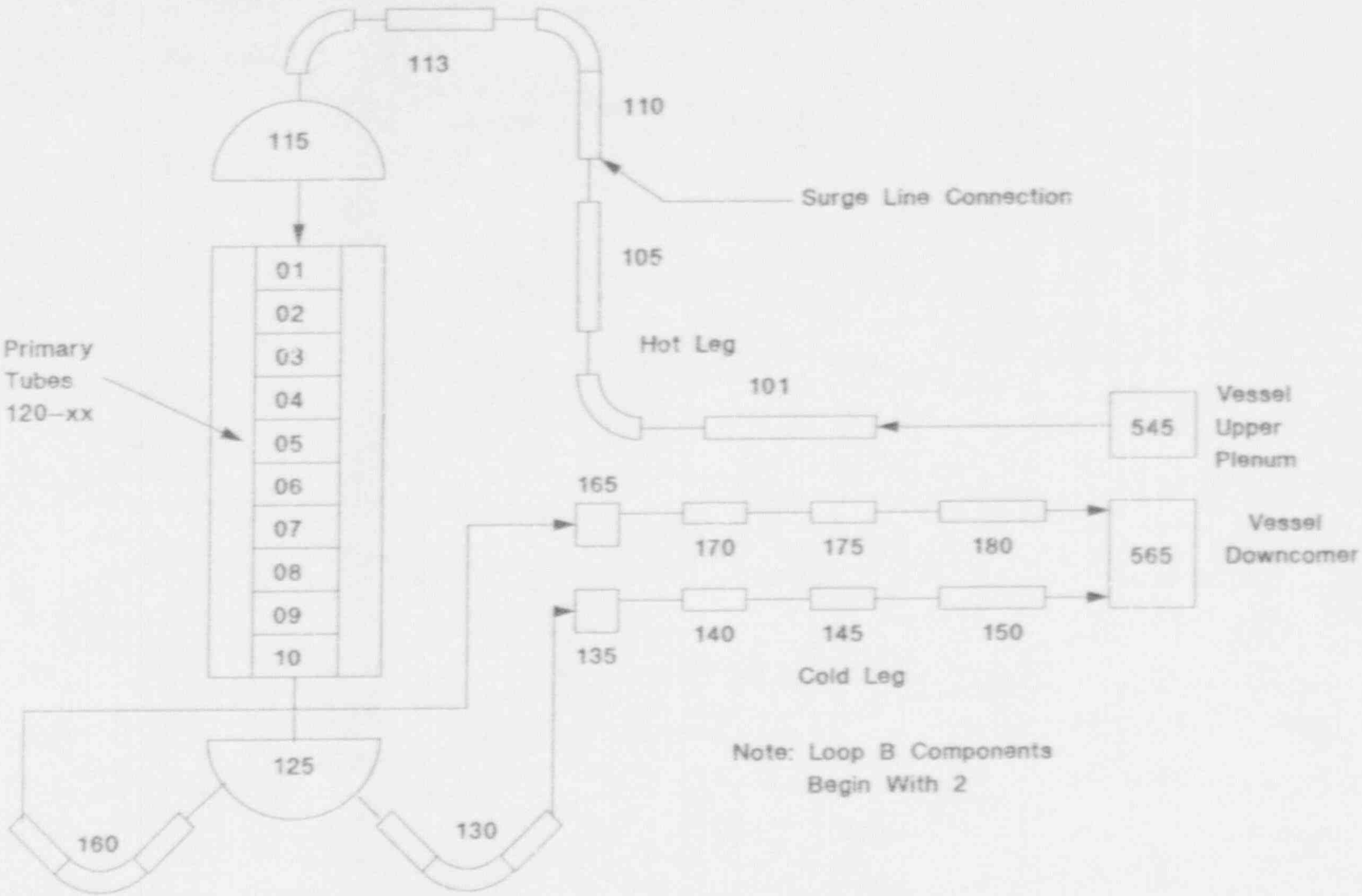

Figure 6. RELAP5/MOD3 primary loop model for once-through steam generator analysis.

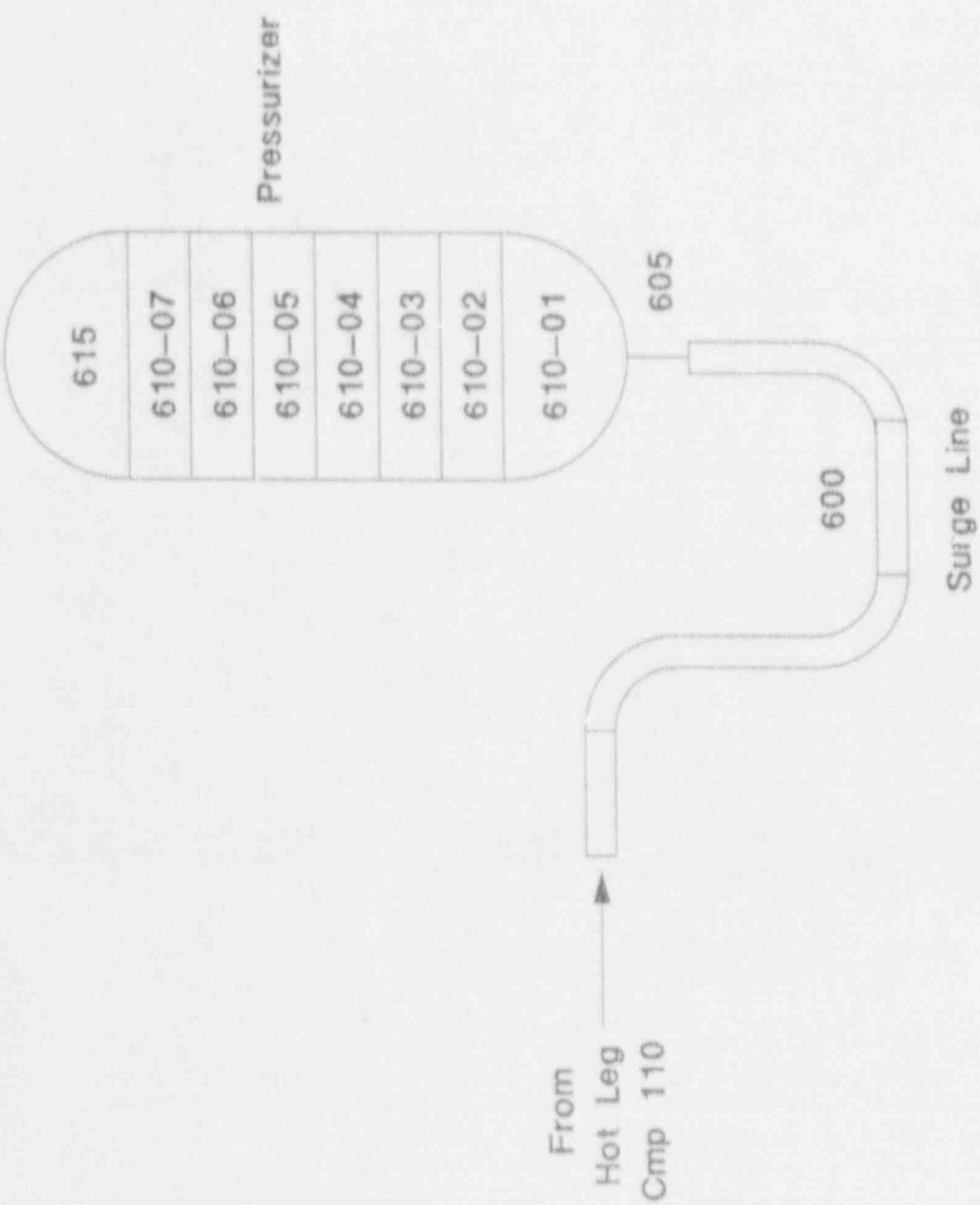

Figure 7. RELAP5/MOD3 pressurizer model for once-through steam generator analysis.

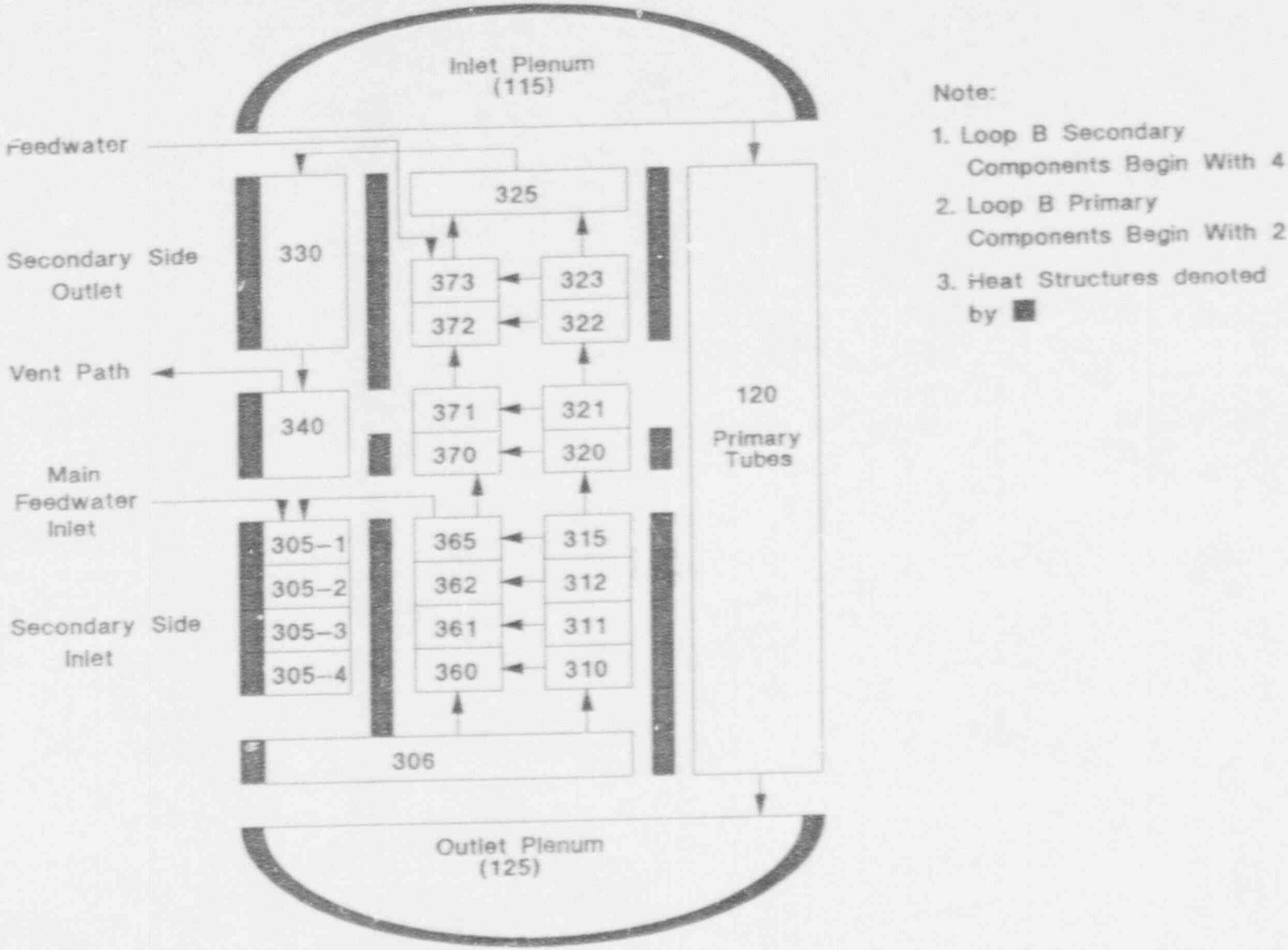

Figure 8. RELAP5/MOD3 secondary model for once-through steam generator analysis.

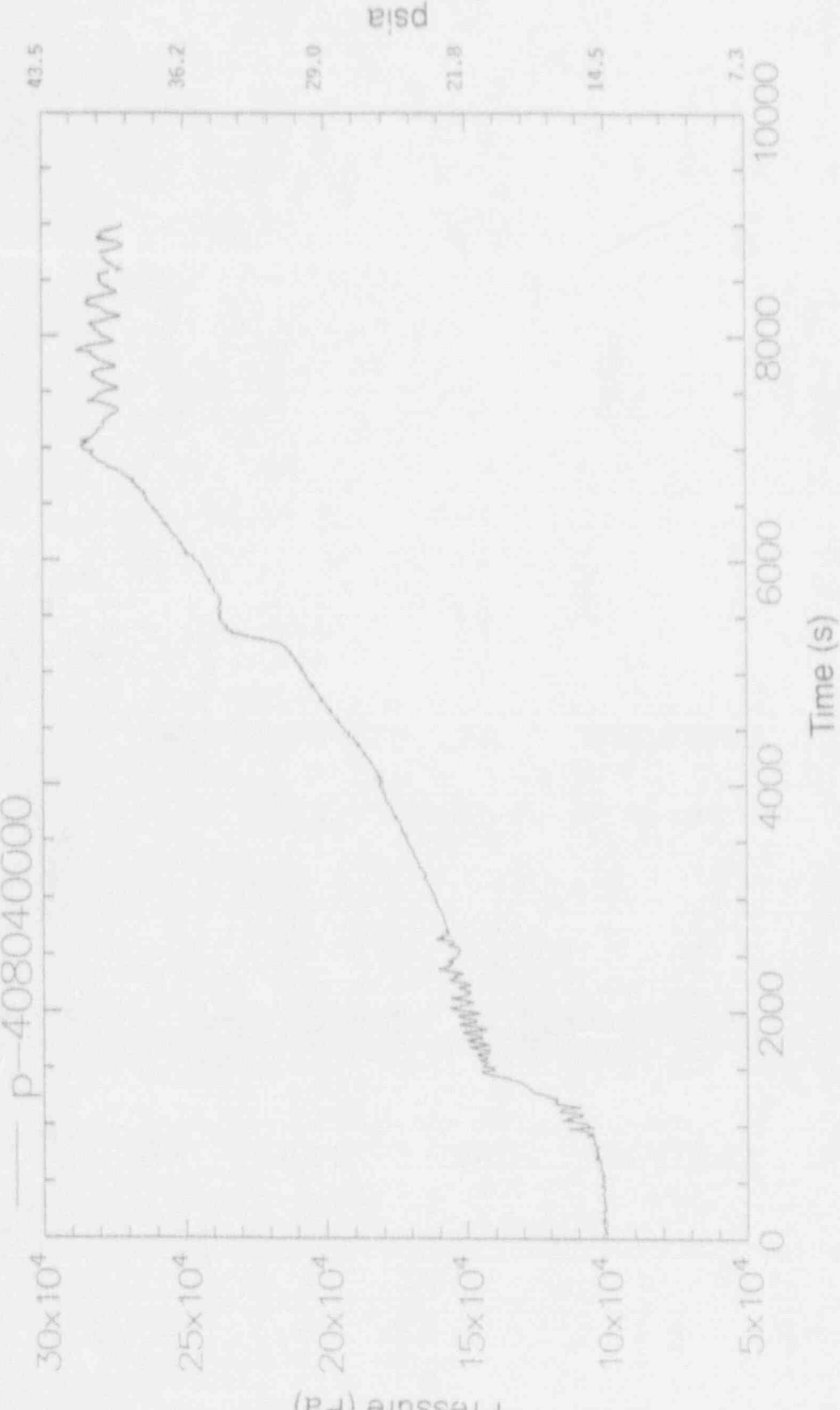

Figure 9. Primary system pressure for Case 1.

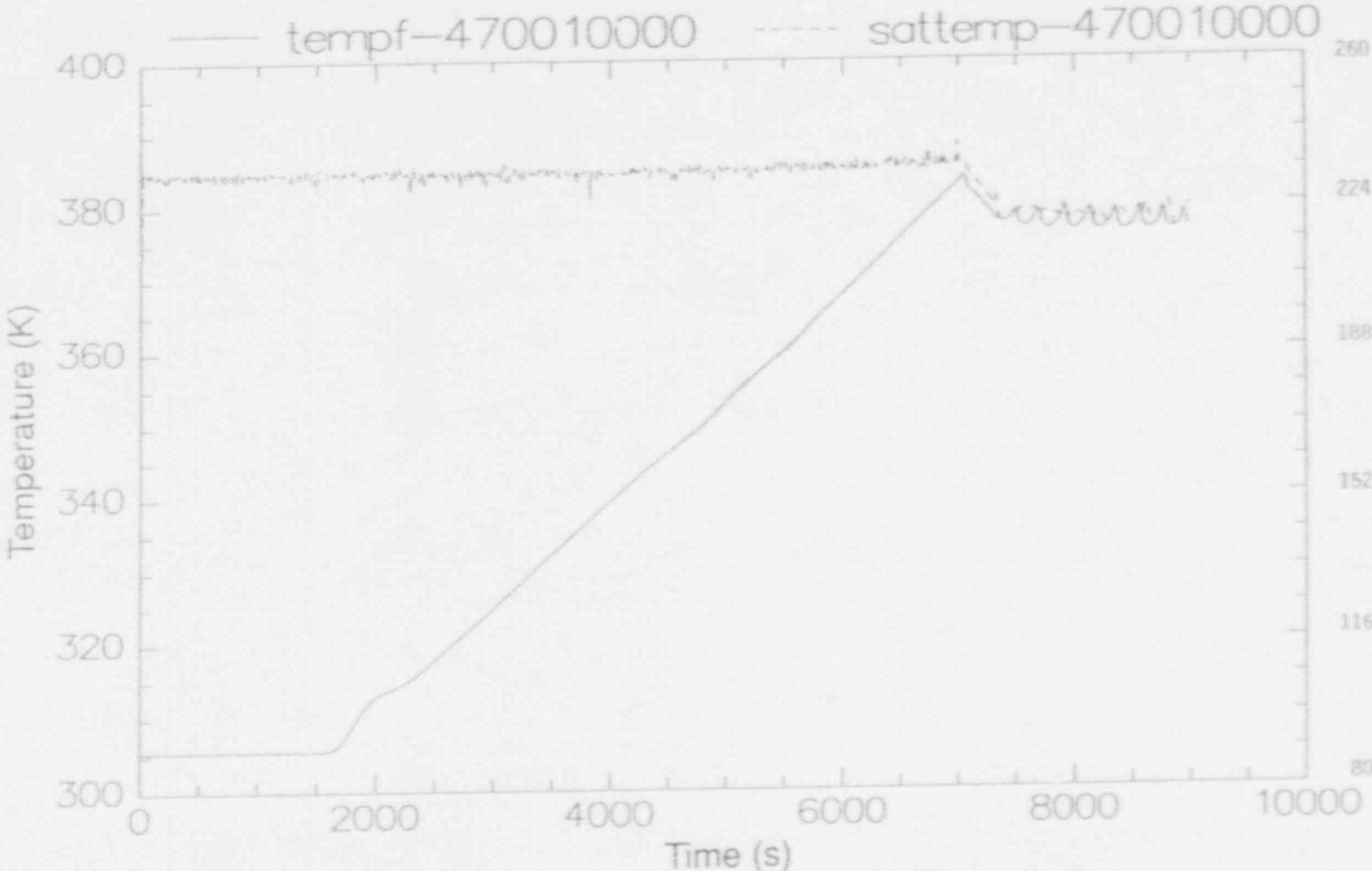

Figure 10. Comparison of water temperature to saturation temperature in steam generator secondary—Case 1.

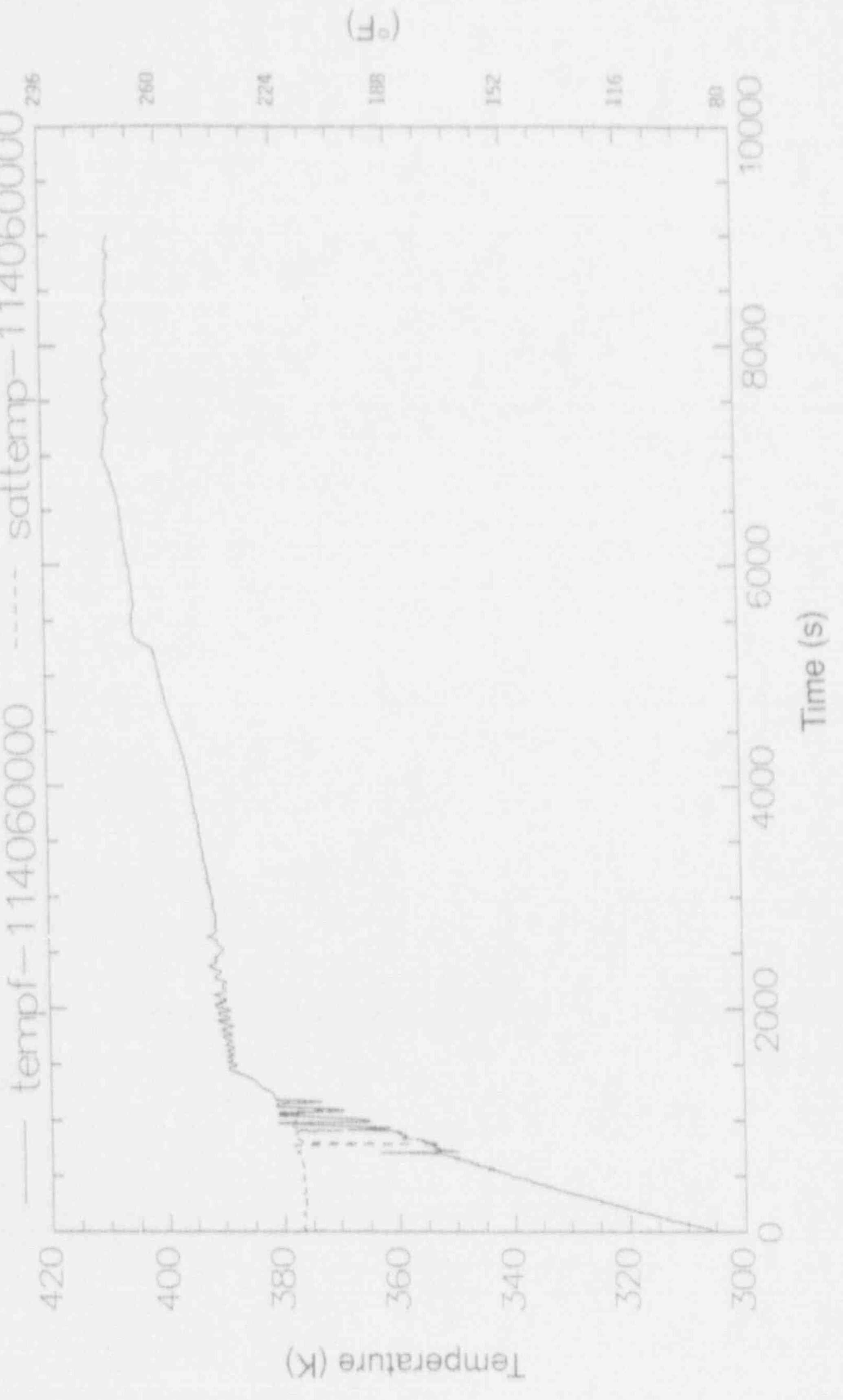

Figure 11. Water temperature in the upper core volume (114-06).

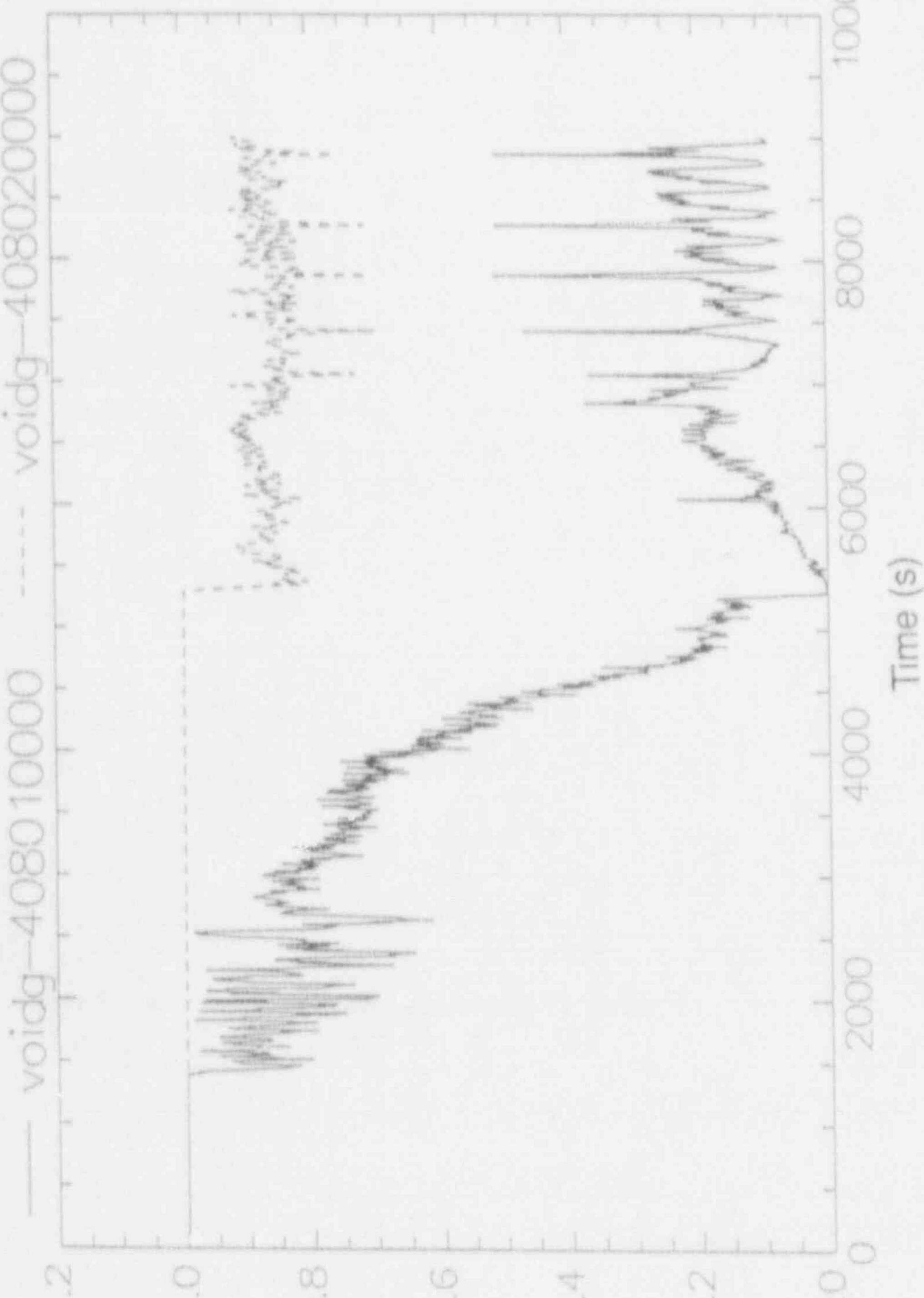

Figure 12. Void fraction in the first two volumes of the steam generator tubes for Case 1.

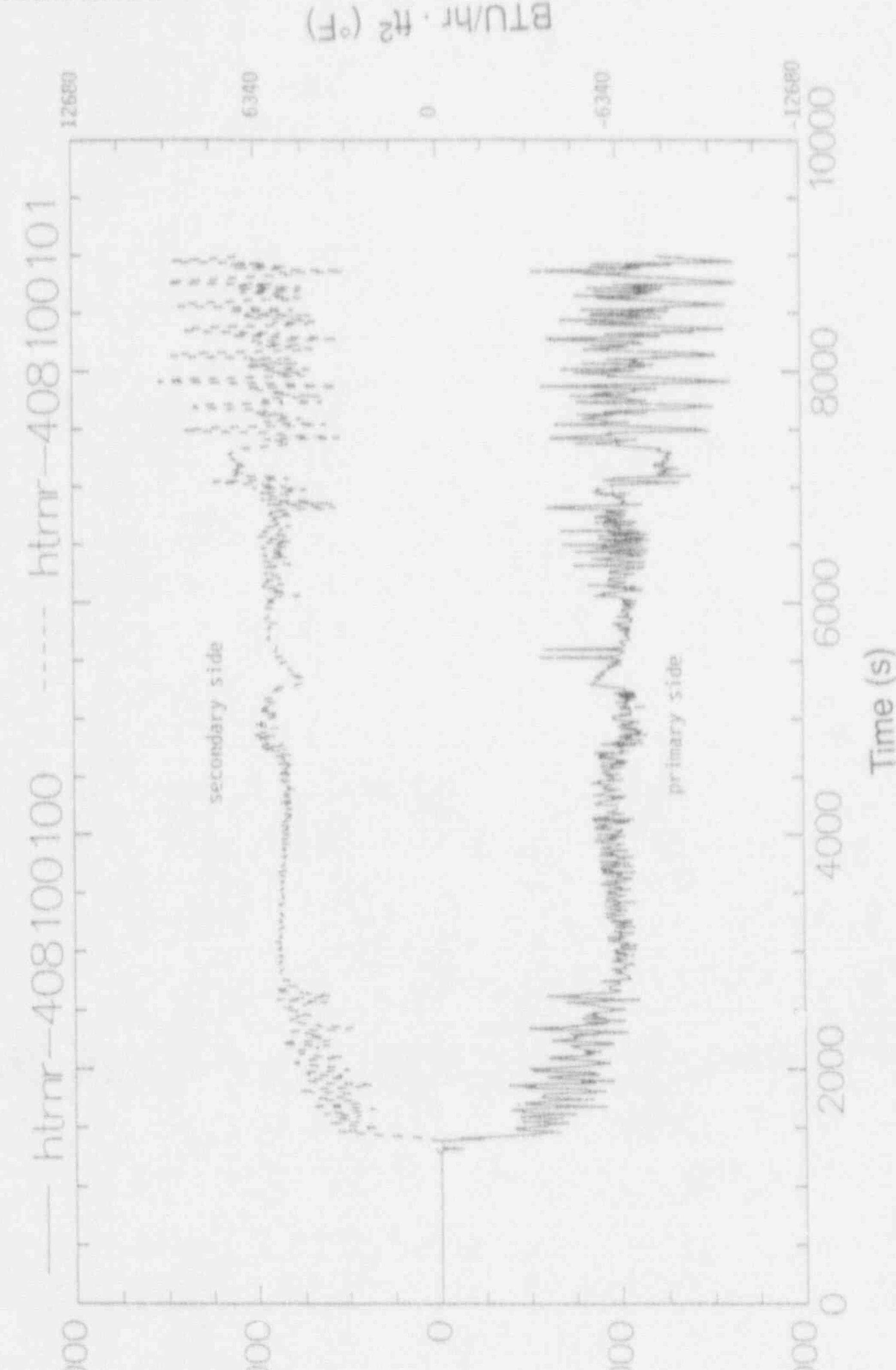

Figure 13. Heat flux through the tubes in the first steam generator volume for Case 1.

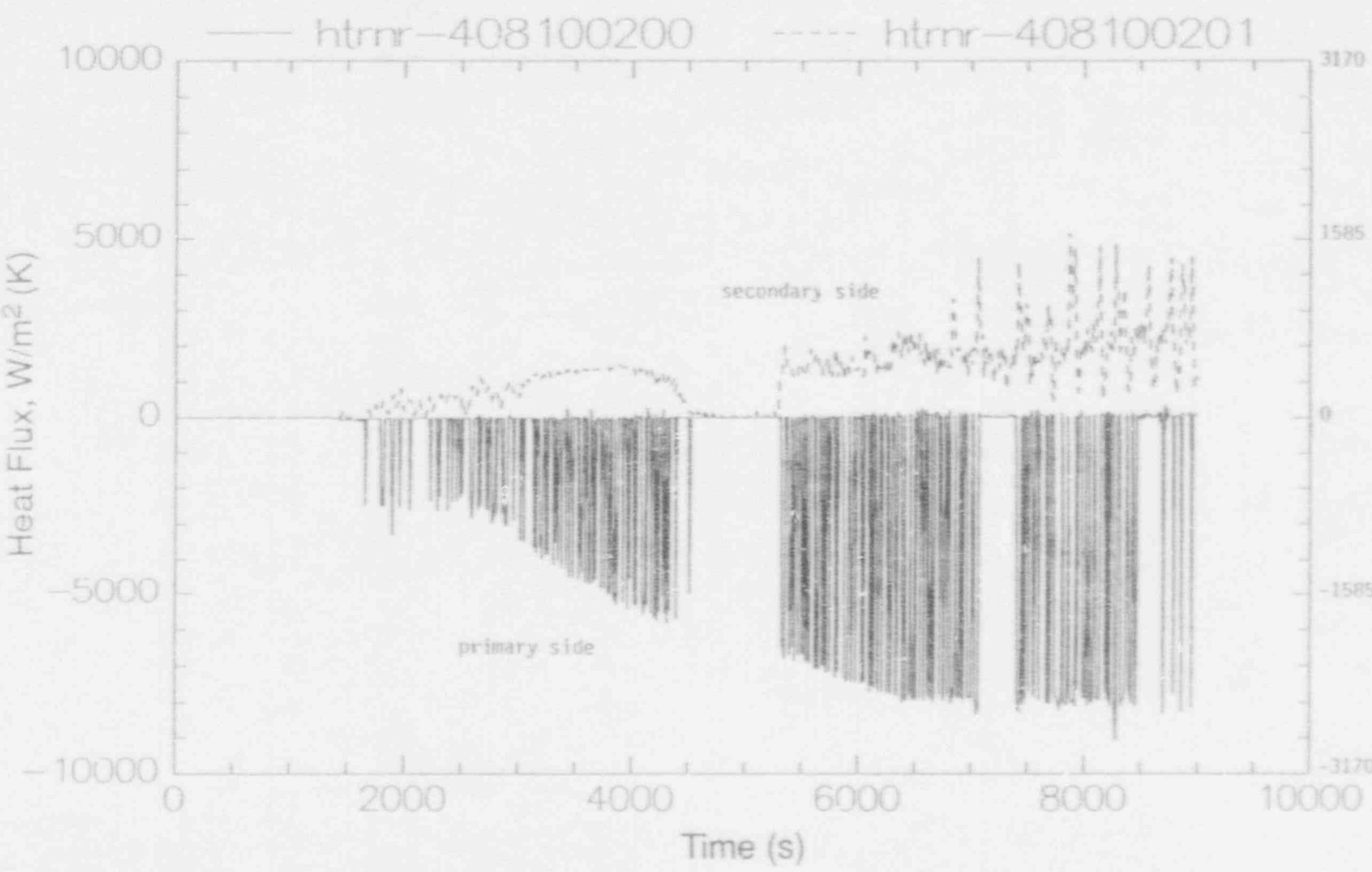

Figure 14. Heat flux through the tubes in the second steam generator volume for Case 1.

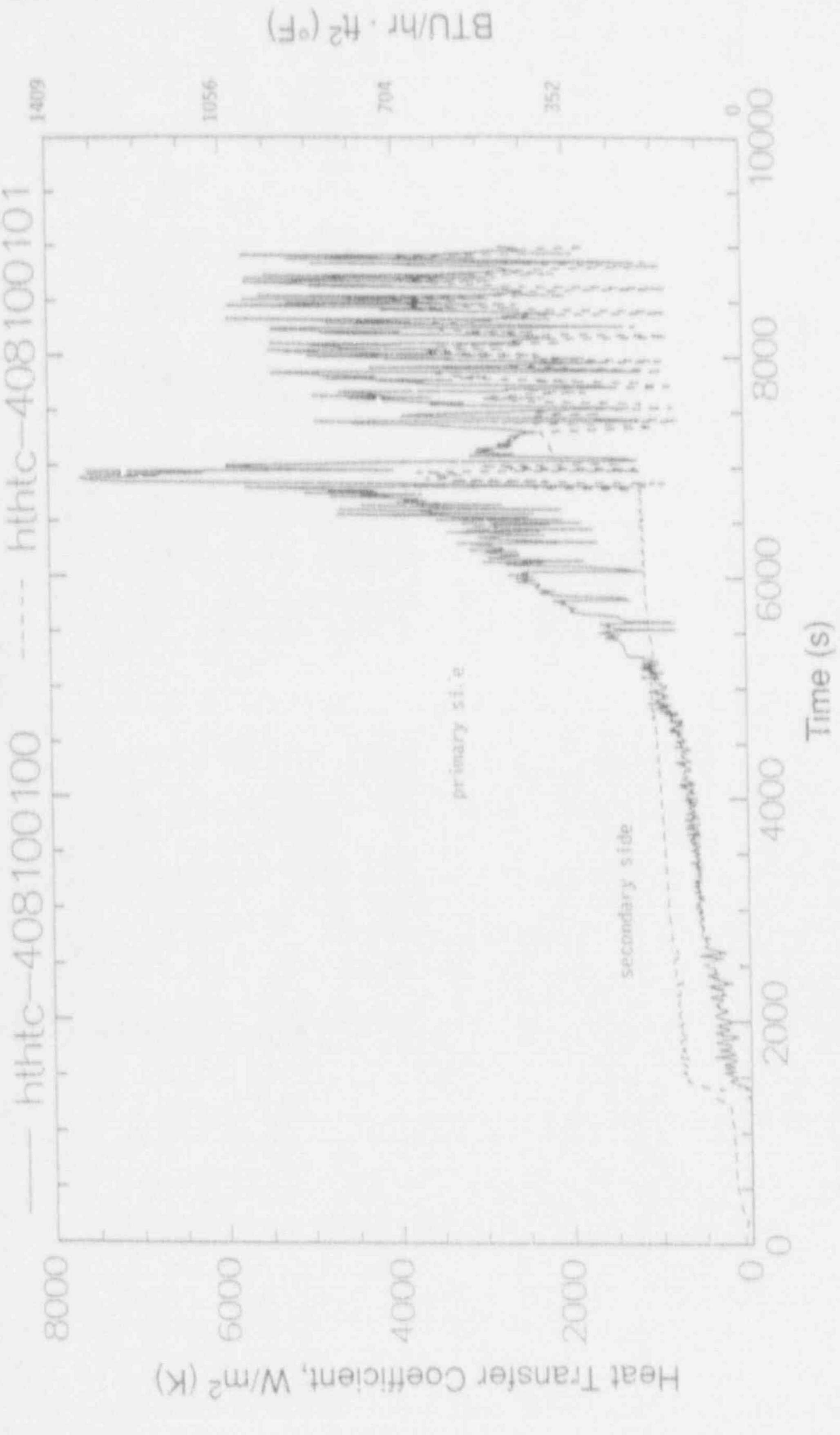

Figure 15. Heat transfer coefficients in the first volume of the steam generator primary tubes for Case 1.

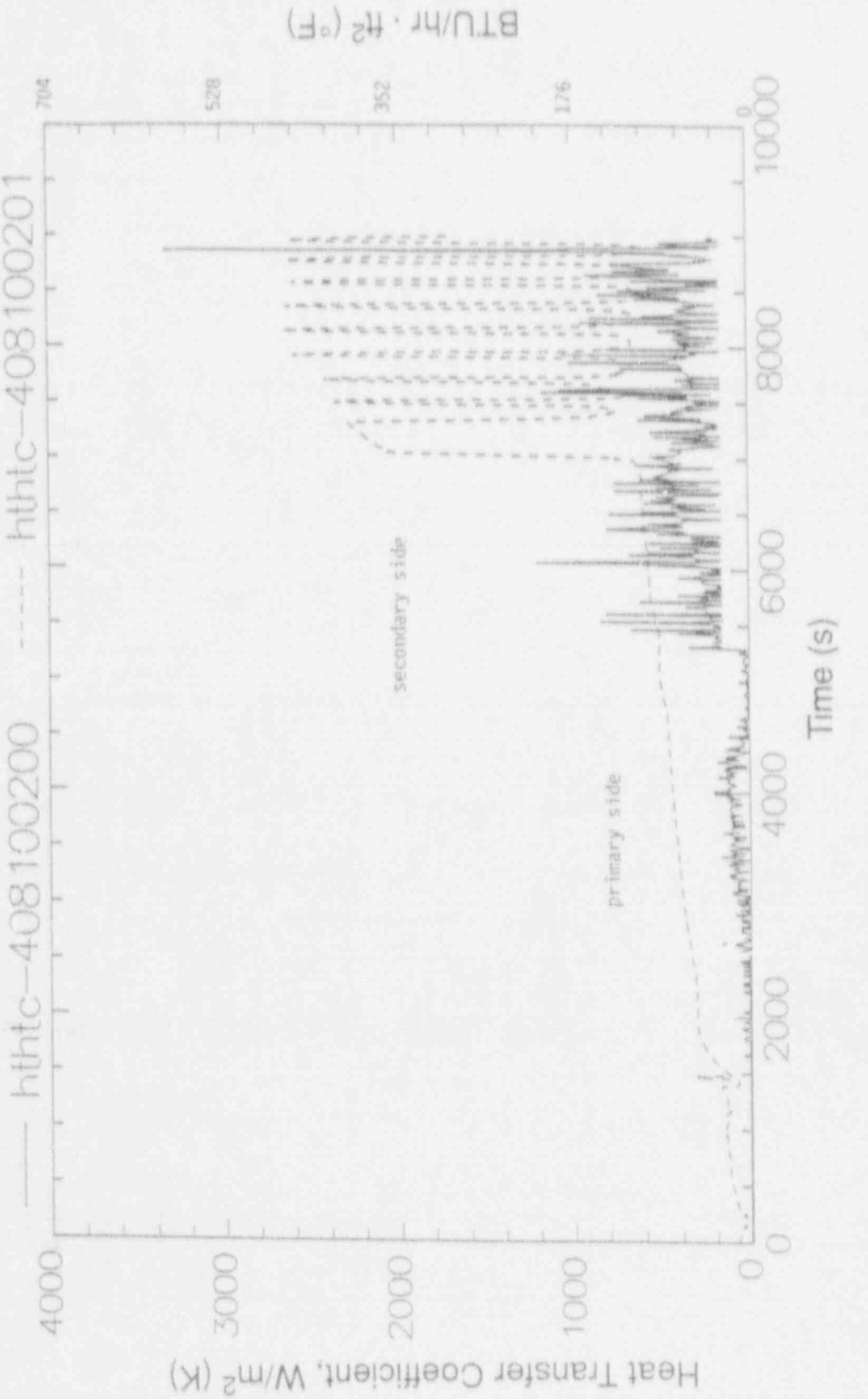

Figure 16. Heat transfer coefficients in the second volume of the steam generator primary tubes for Case 1.

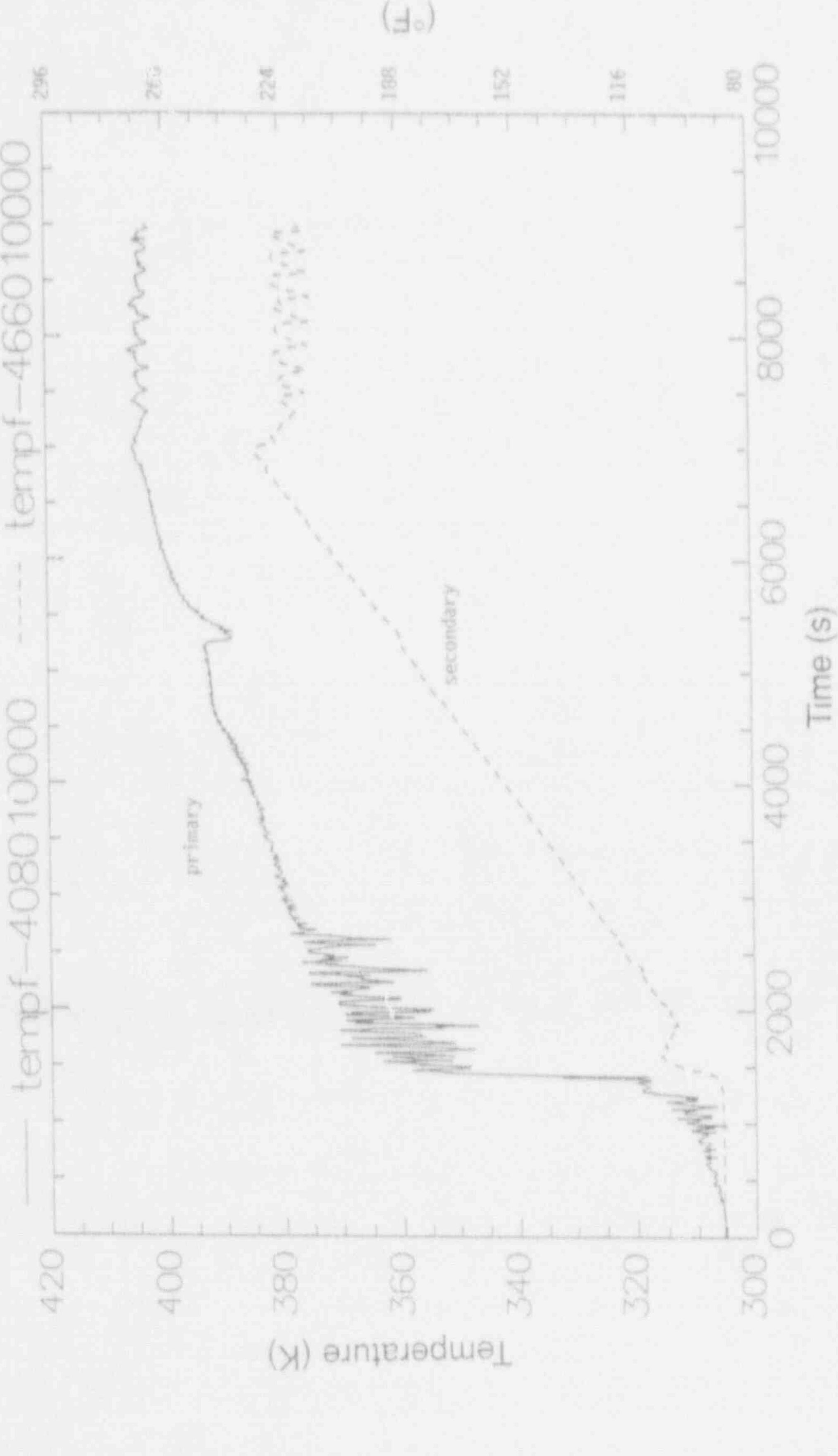

Figure 17. Primary and secondary temperatures in first steam generator volume—Case 1.

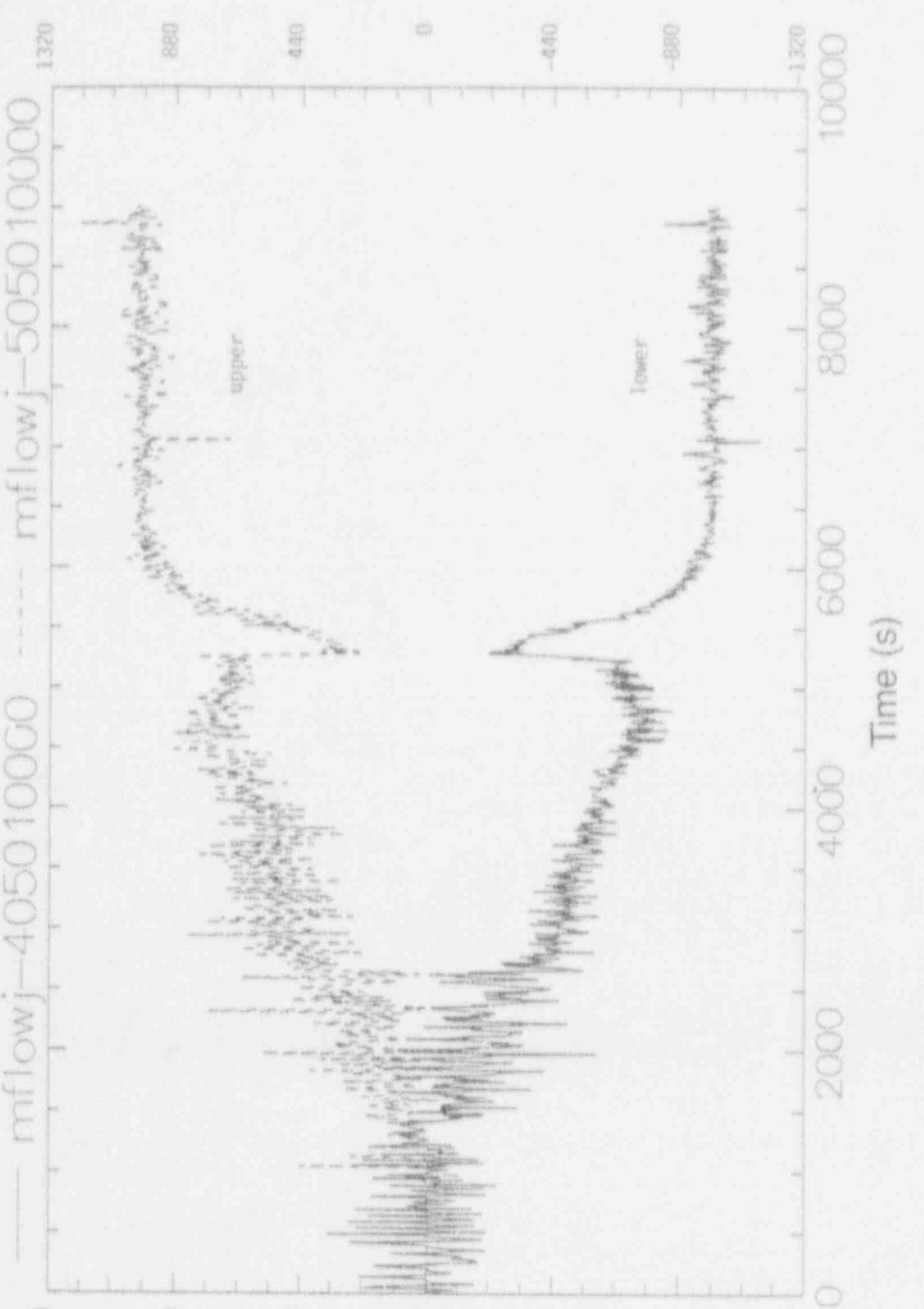

Figure 18. Mass flow in the upper and lower parts of the hot leg—Case 1.

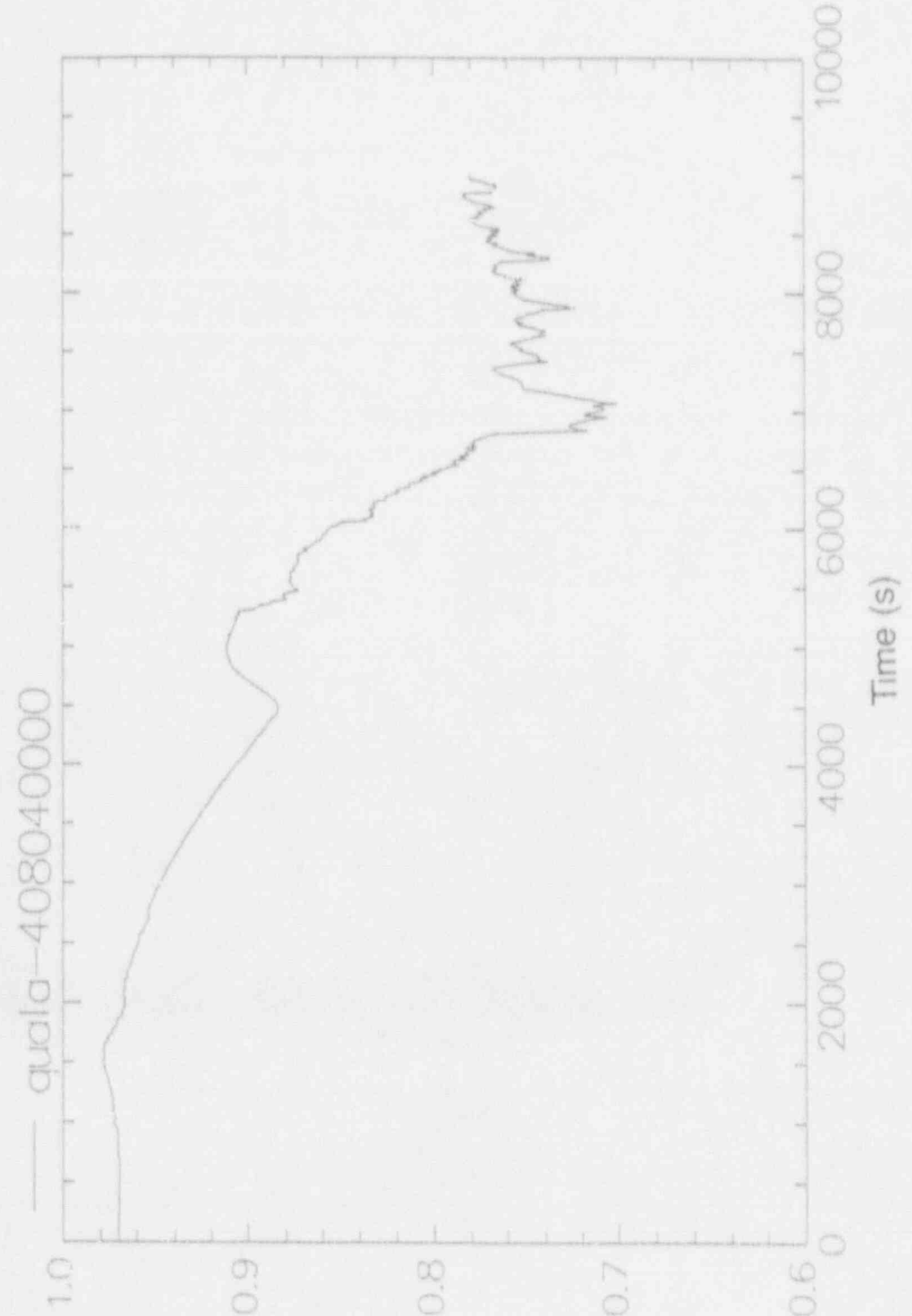

Figure 19. Noncondensible mass fraction in the U-bend of the steam generator tubes—Case 1

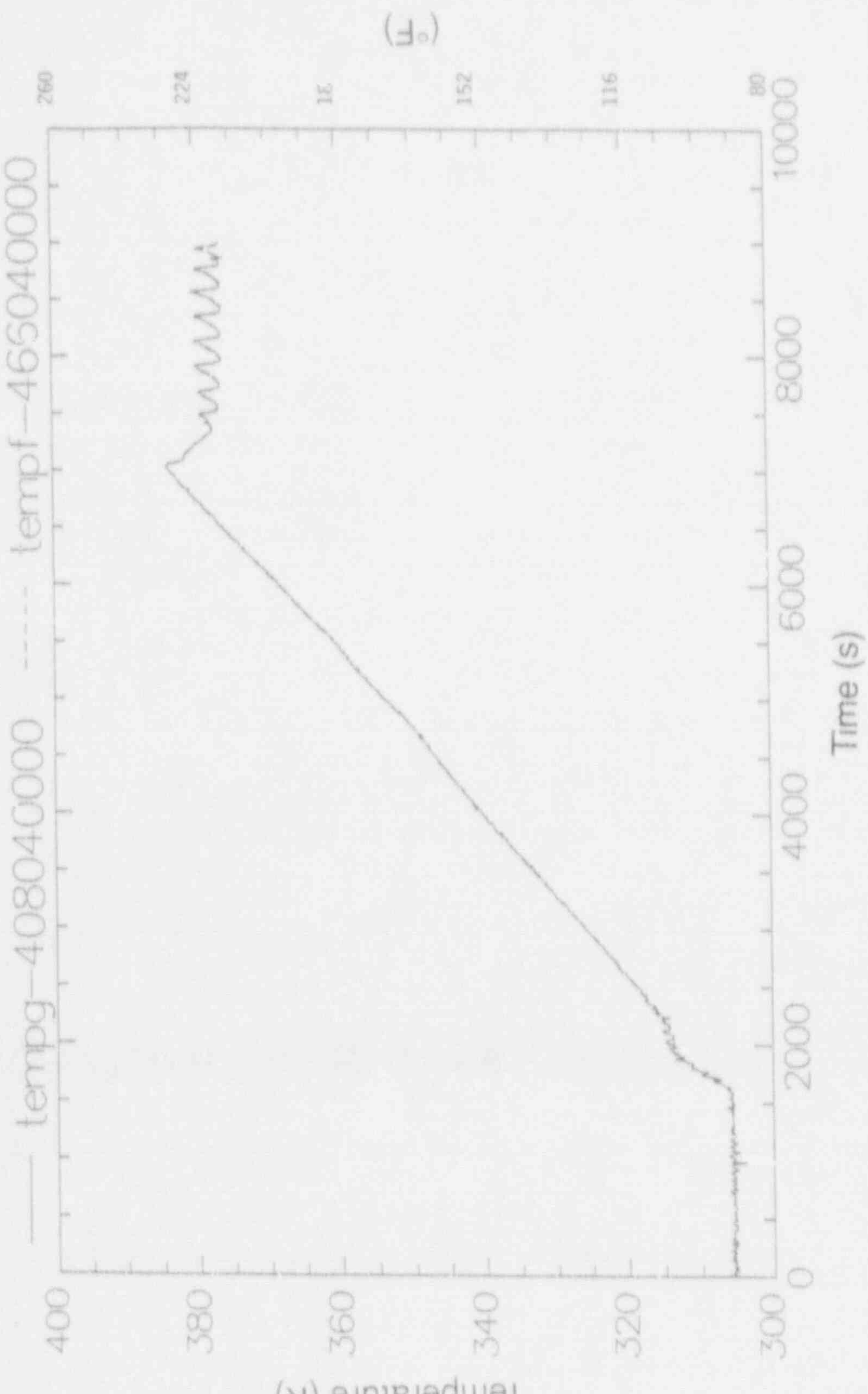

Figure 20. Primary and secondary temperatures at the U-bend of the steam generator tubes—Case 1.

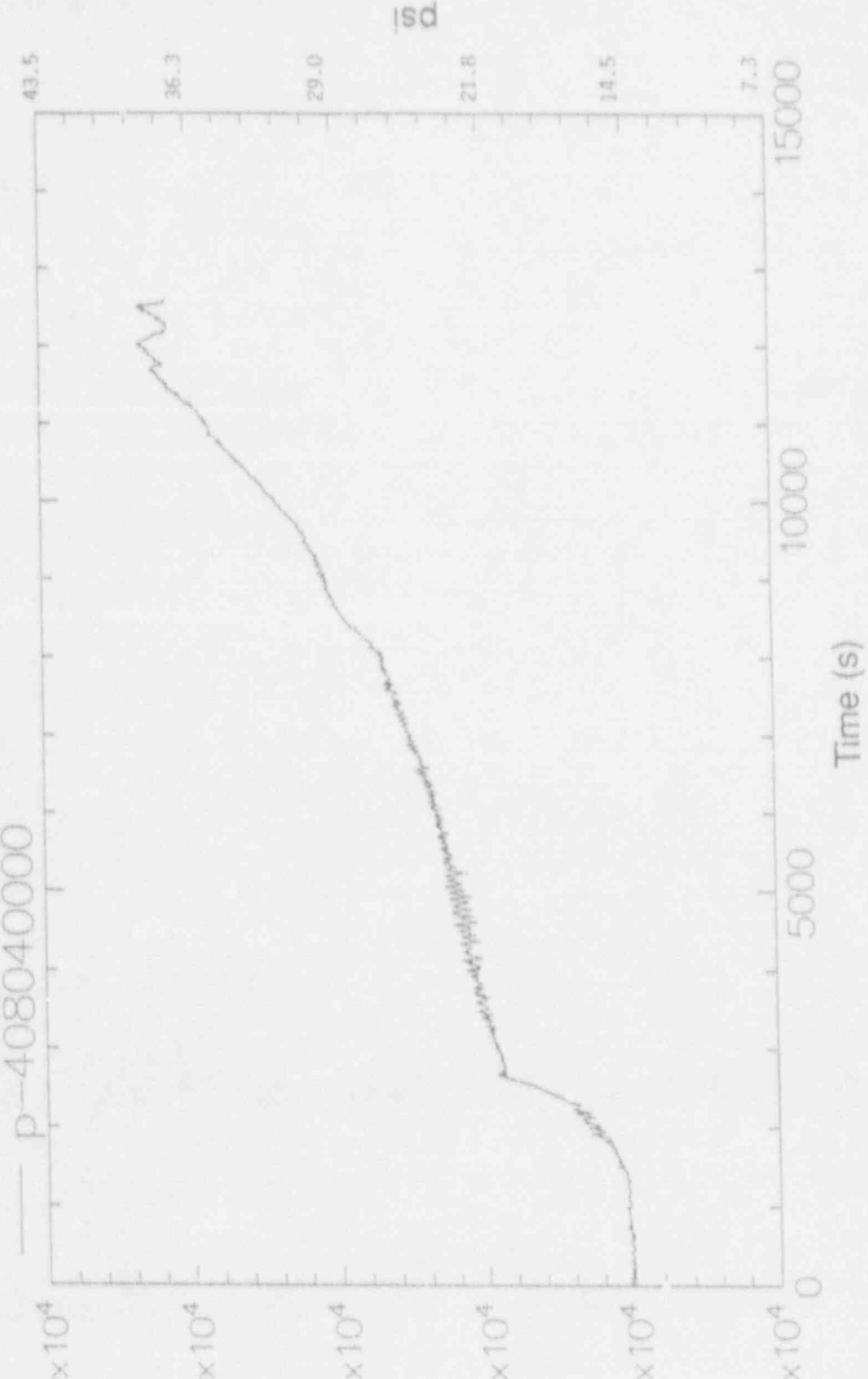

Figure 21. Primary system pressure for Case 2.

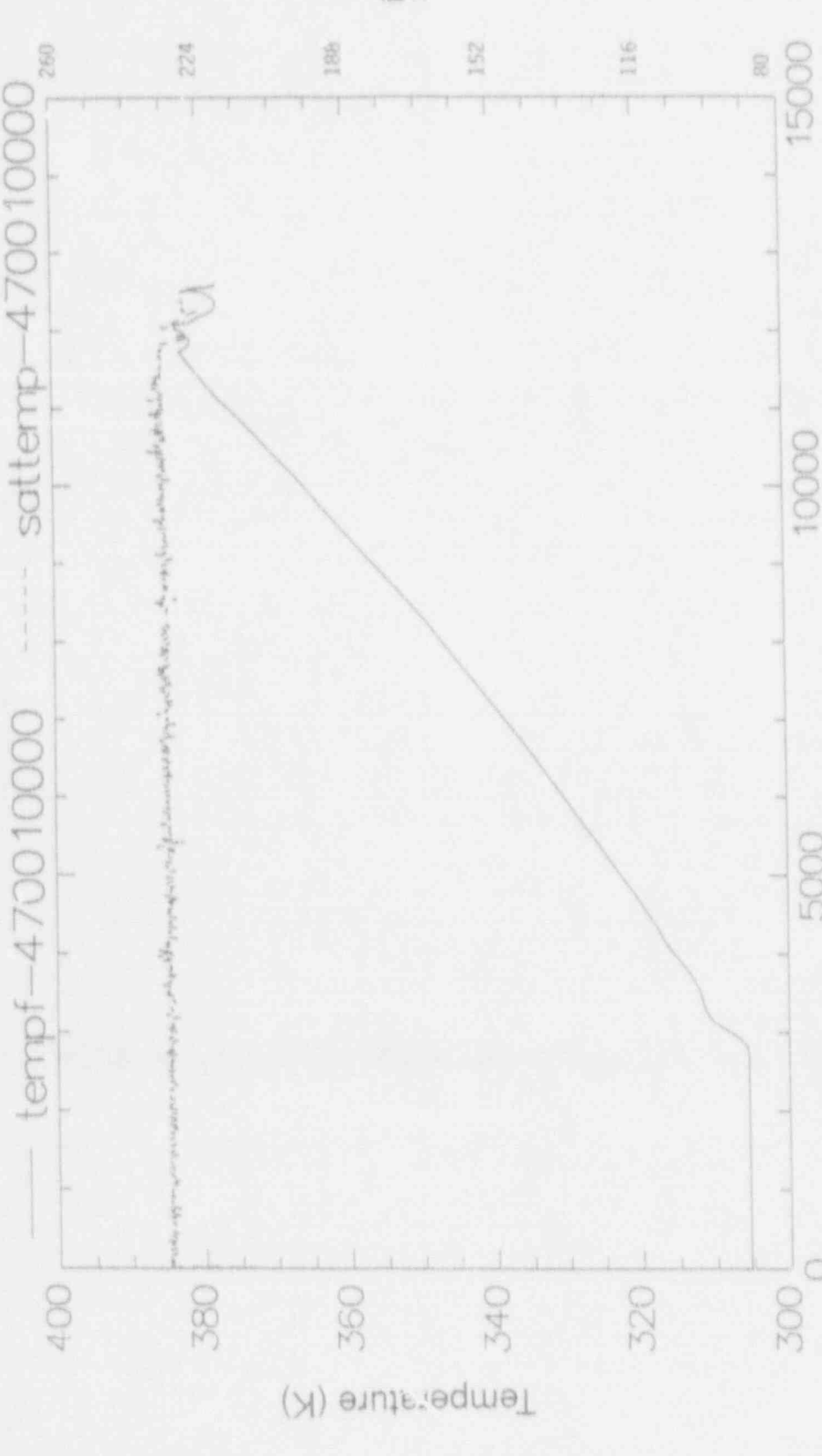

Figure 22. Comparison of water temperature to saturation temperature in steam generator secondary—Case 2.

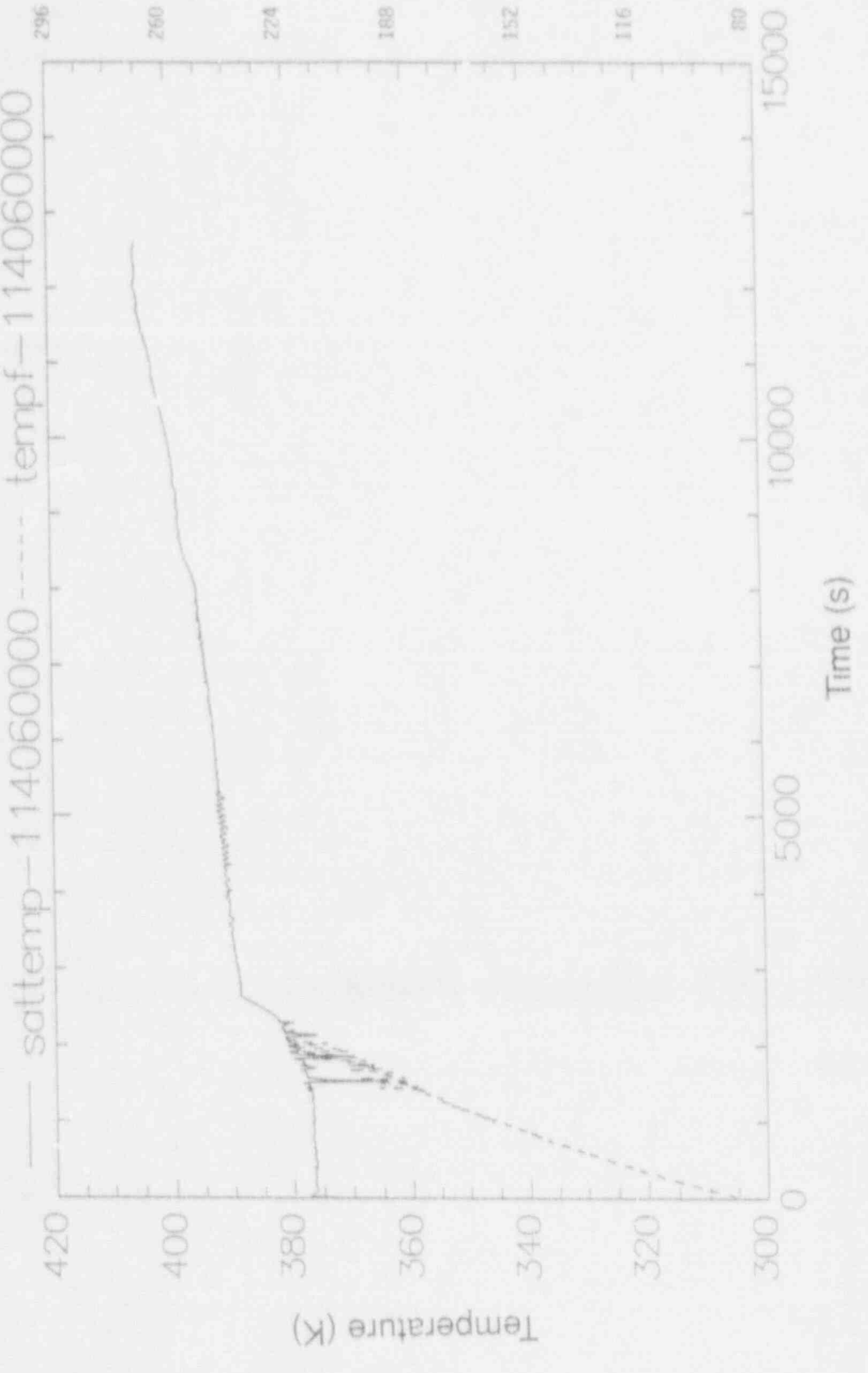

Figure 23. Water temperature in the upper core volume (114-06

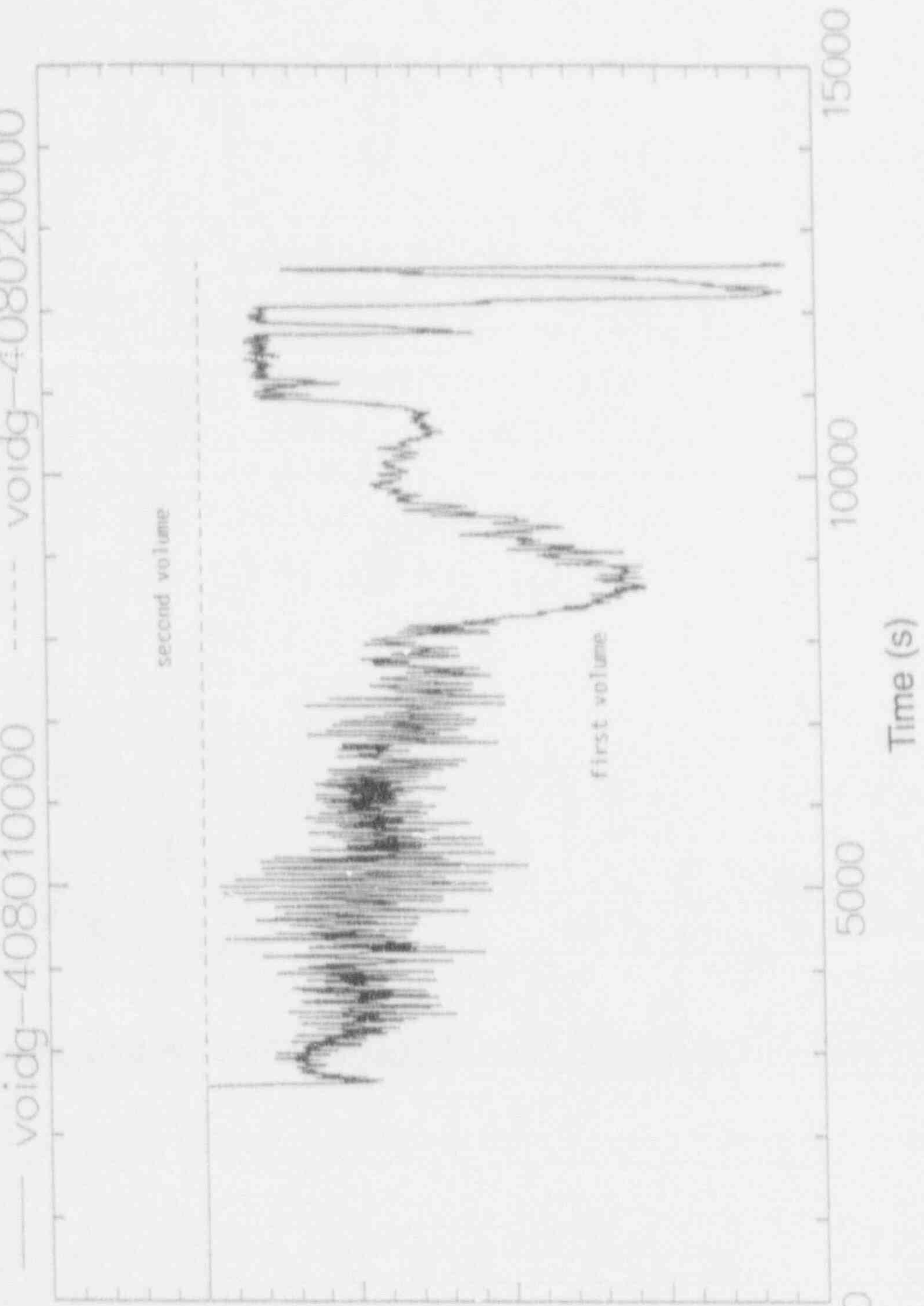

Figure 24. Void fraction in the first two volumes of the steam generator for Case 2.

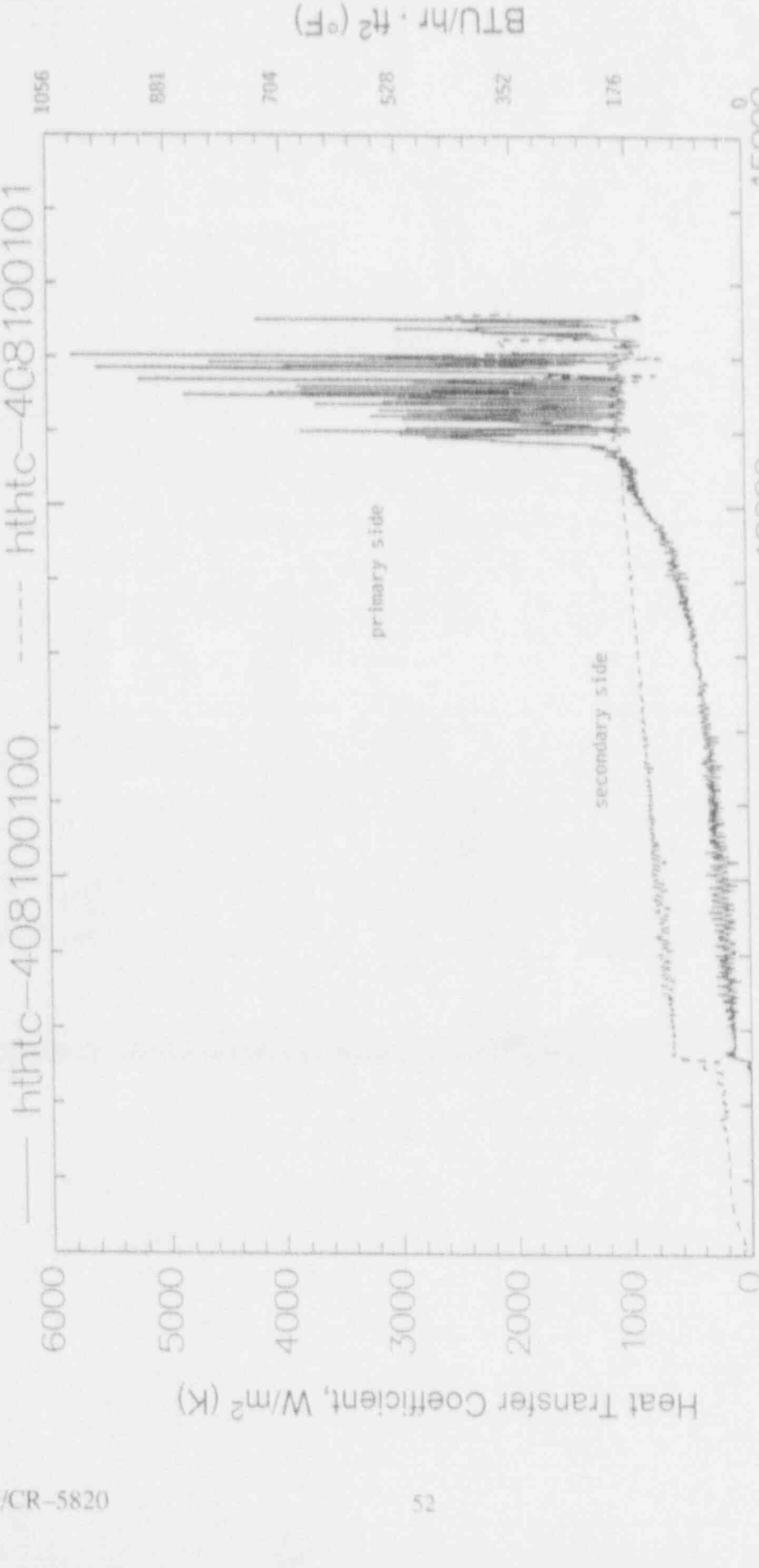

Figure 25. Heat transfer coefficients in the first volume of the steam generator primary tubes for Case 2.

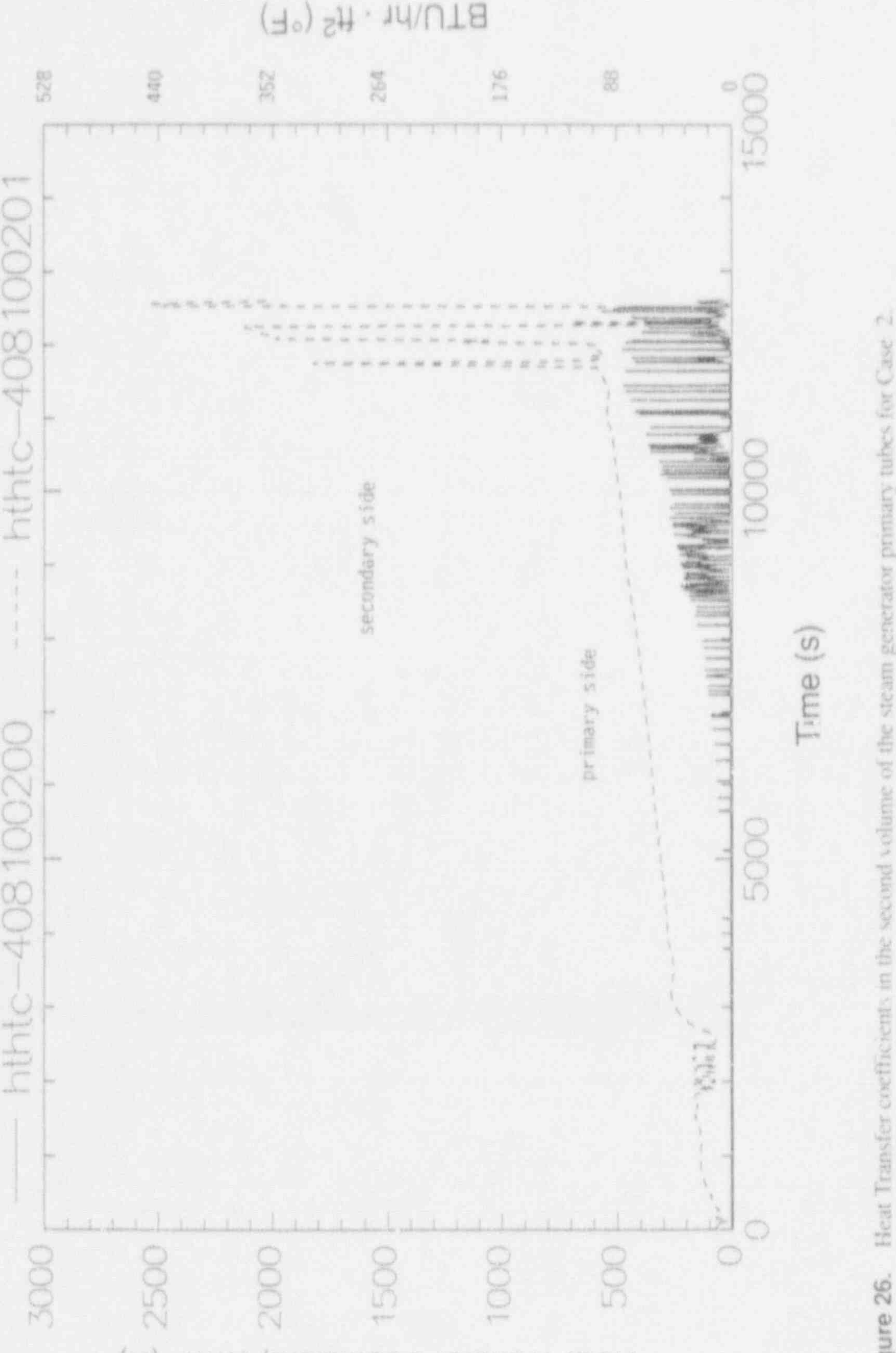

Figure 26. Heat Transfer coefficients in the second volume of the steam generator primary tubes for Case 2.

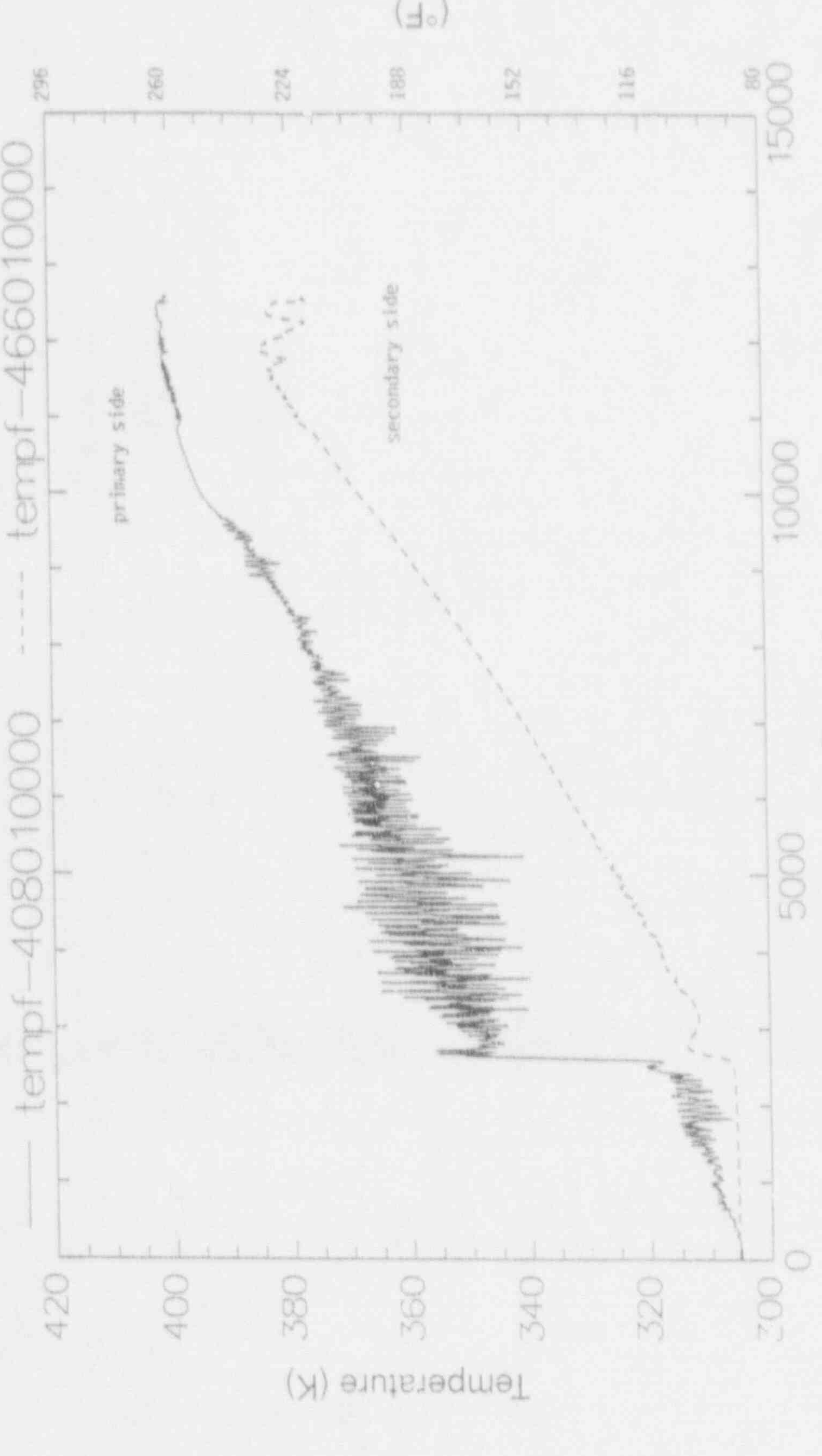

Figure 27. Primary and secondary temperature in the first steam generator volume—Case 2.

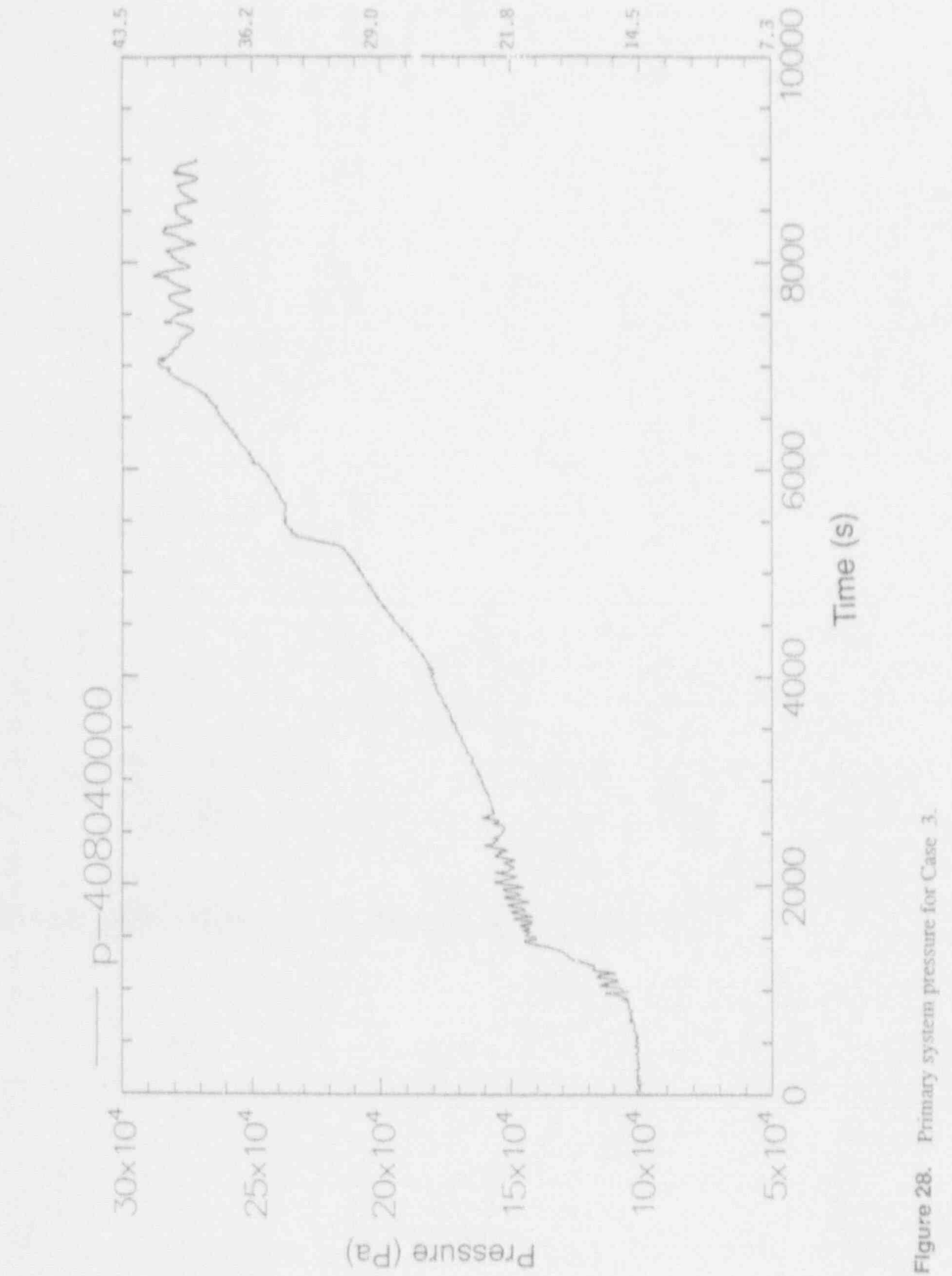

Figure 28. Primary system pressure for Case 3.

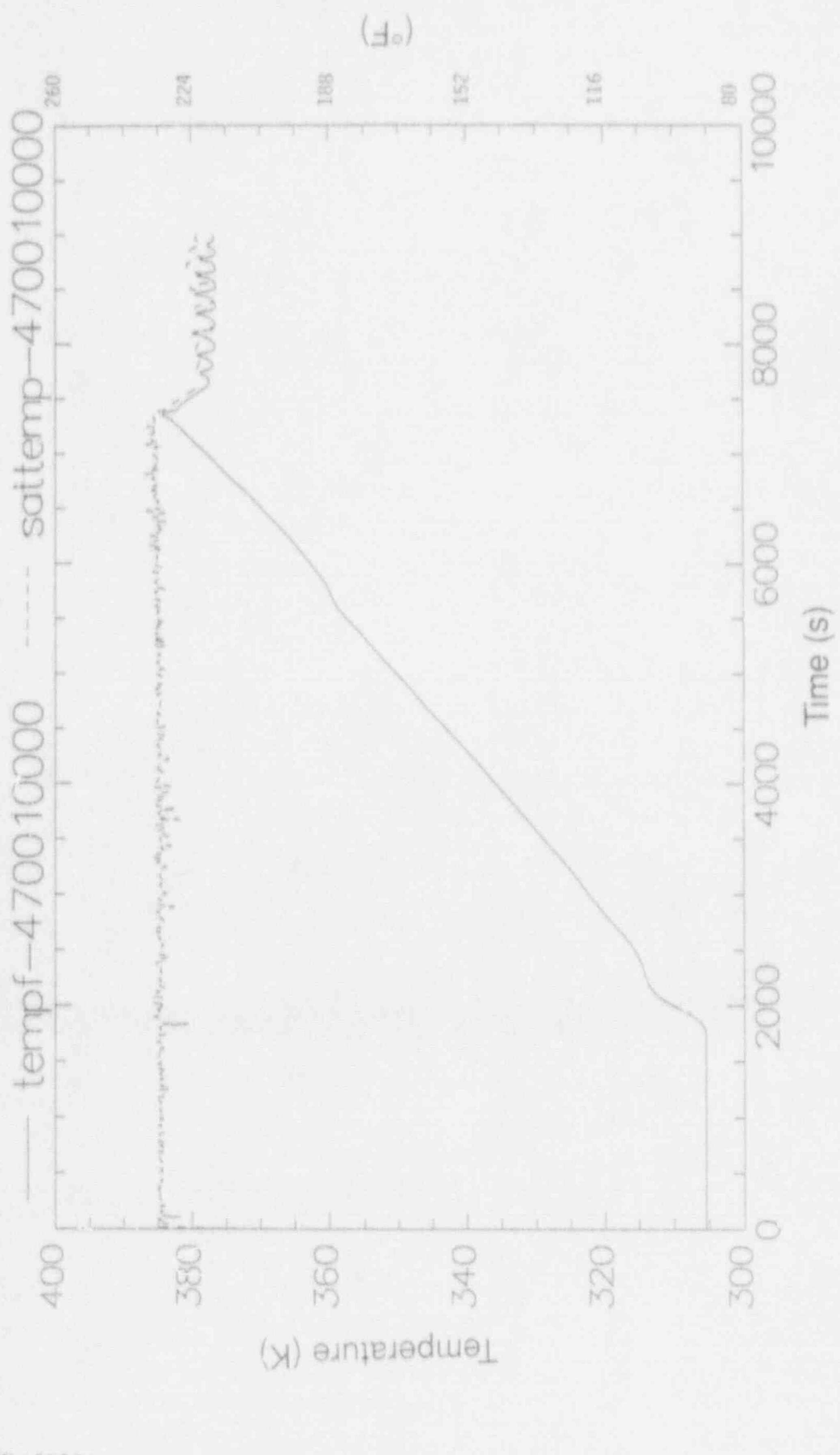

Figure 29. Comparison of water temperature to saturation temperature in steam generator secondary—Case 3.

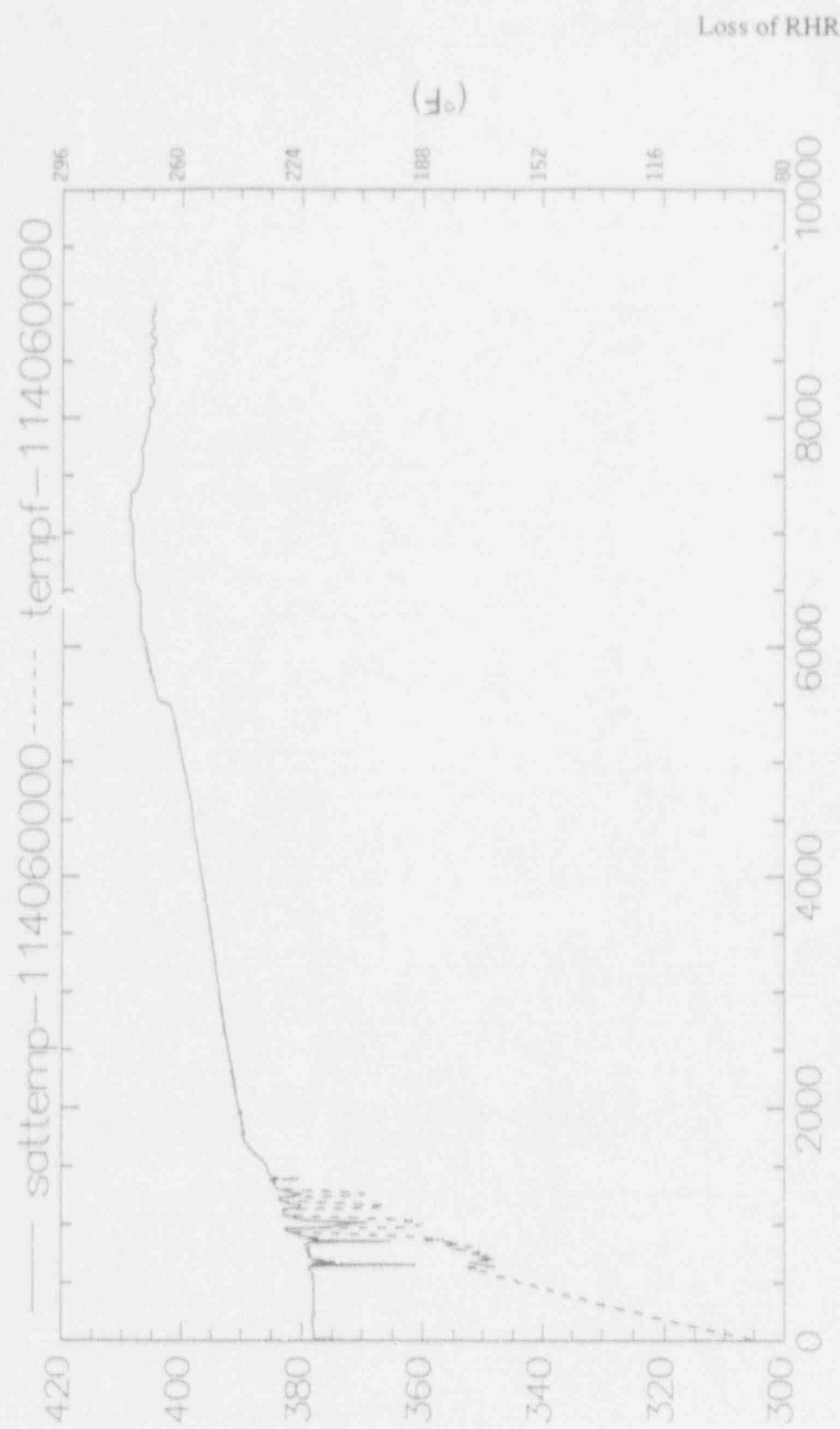

Figure 30. Water temperature in the upper core volume (114-06).

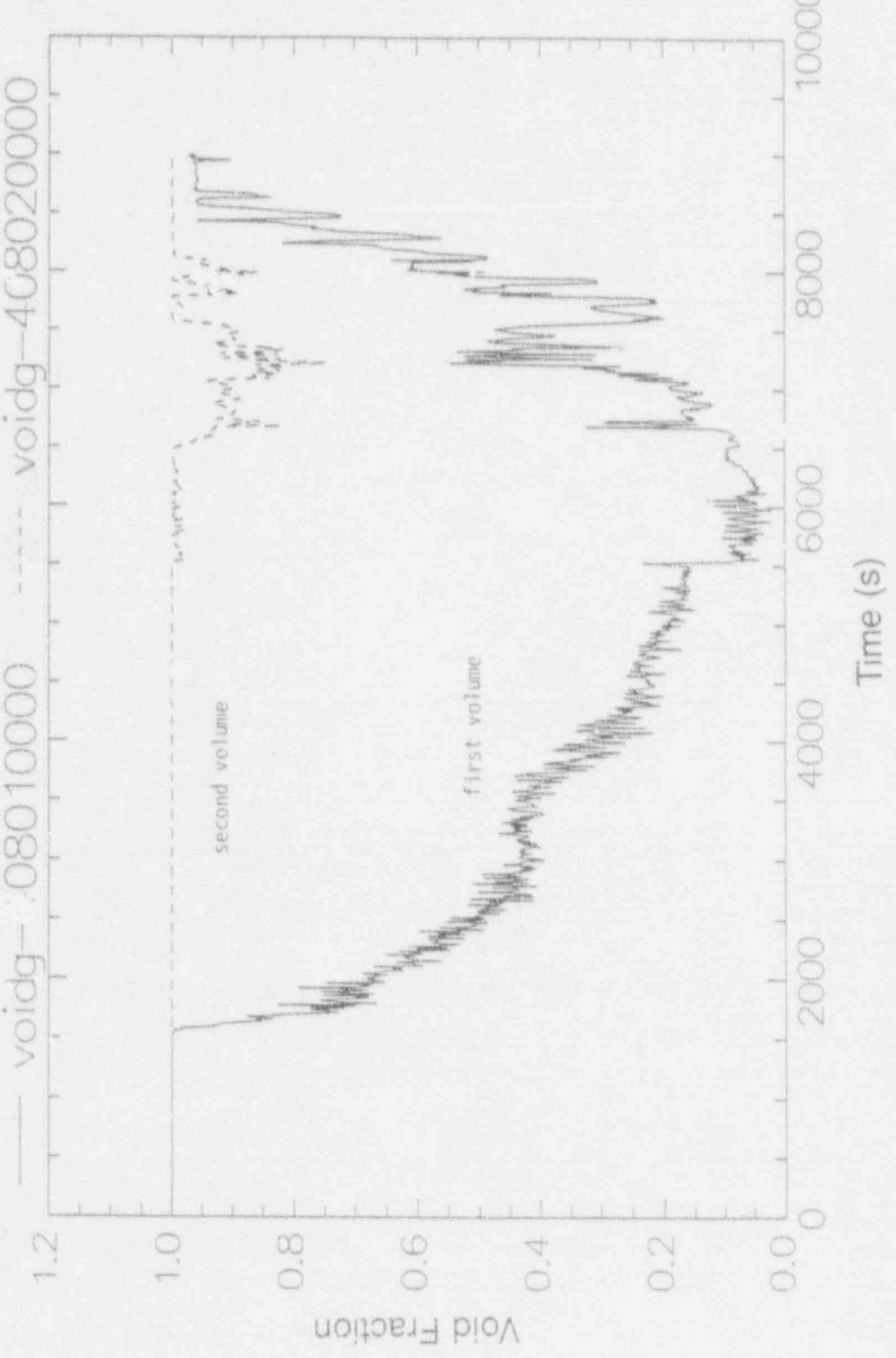

Figure 31. Vapor void fraction in the first two volumes of the steam generator tubes for Case 3.

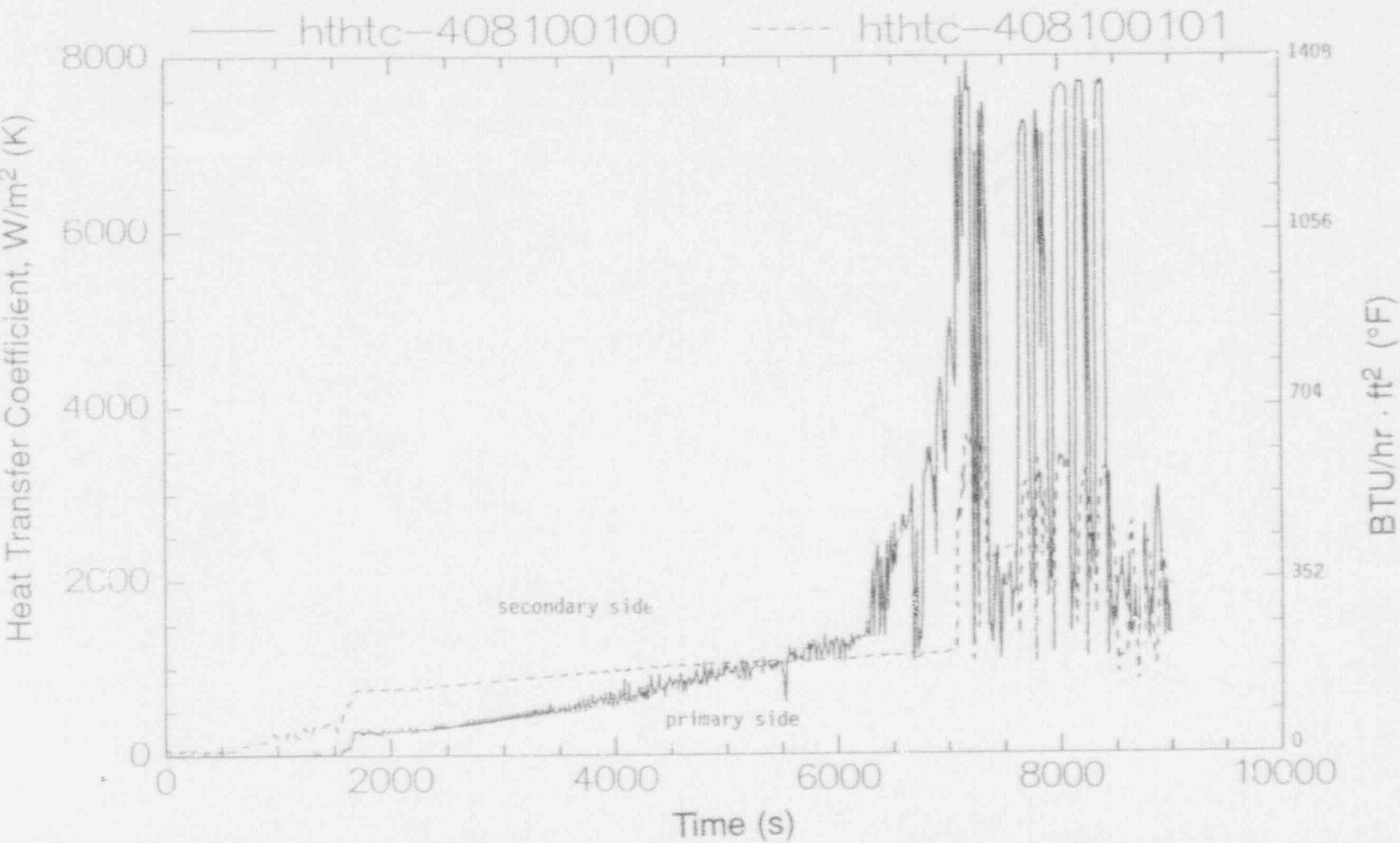

Figure 32. Heat transfer coefficients in the first volume of the steam generator primary tubes for Case 3.

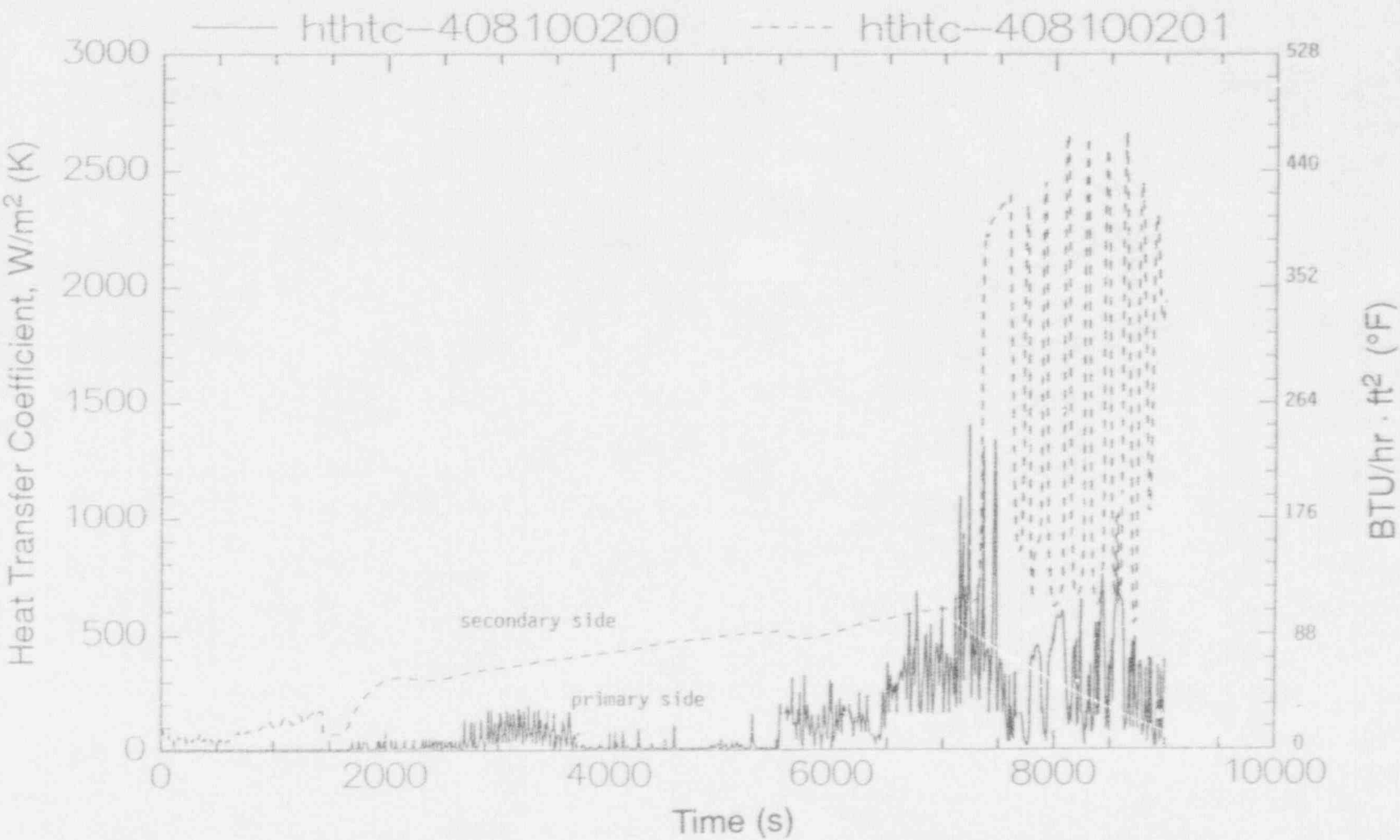

Figure 33. Heat transfer coefficients in the second volume of the steam generator primary tubes for Case 3.

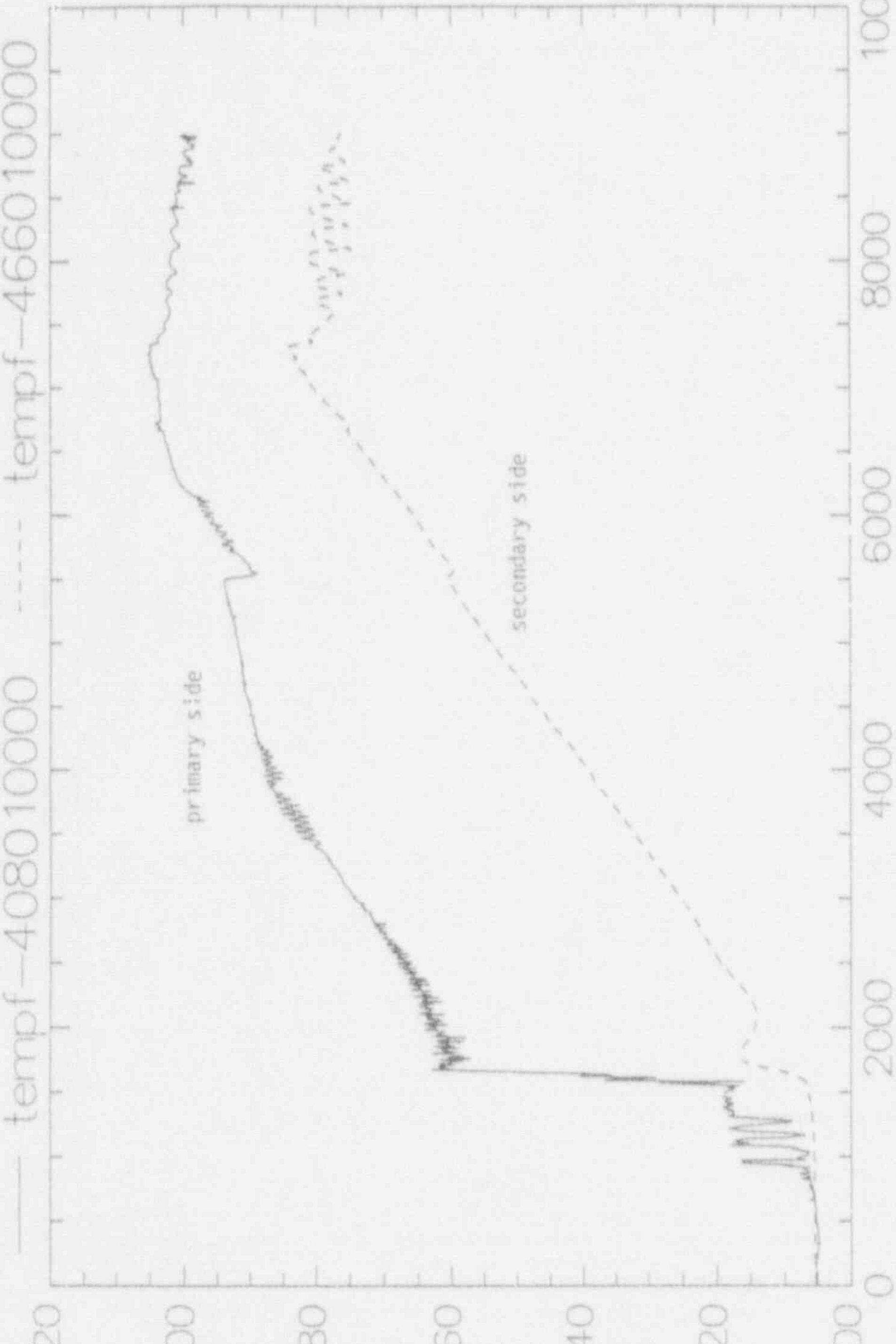

Figure 34. Primary and secondary temperature in the first steam generator volume—Case 3.

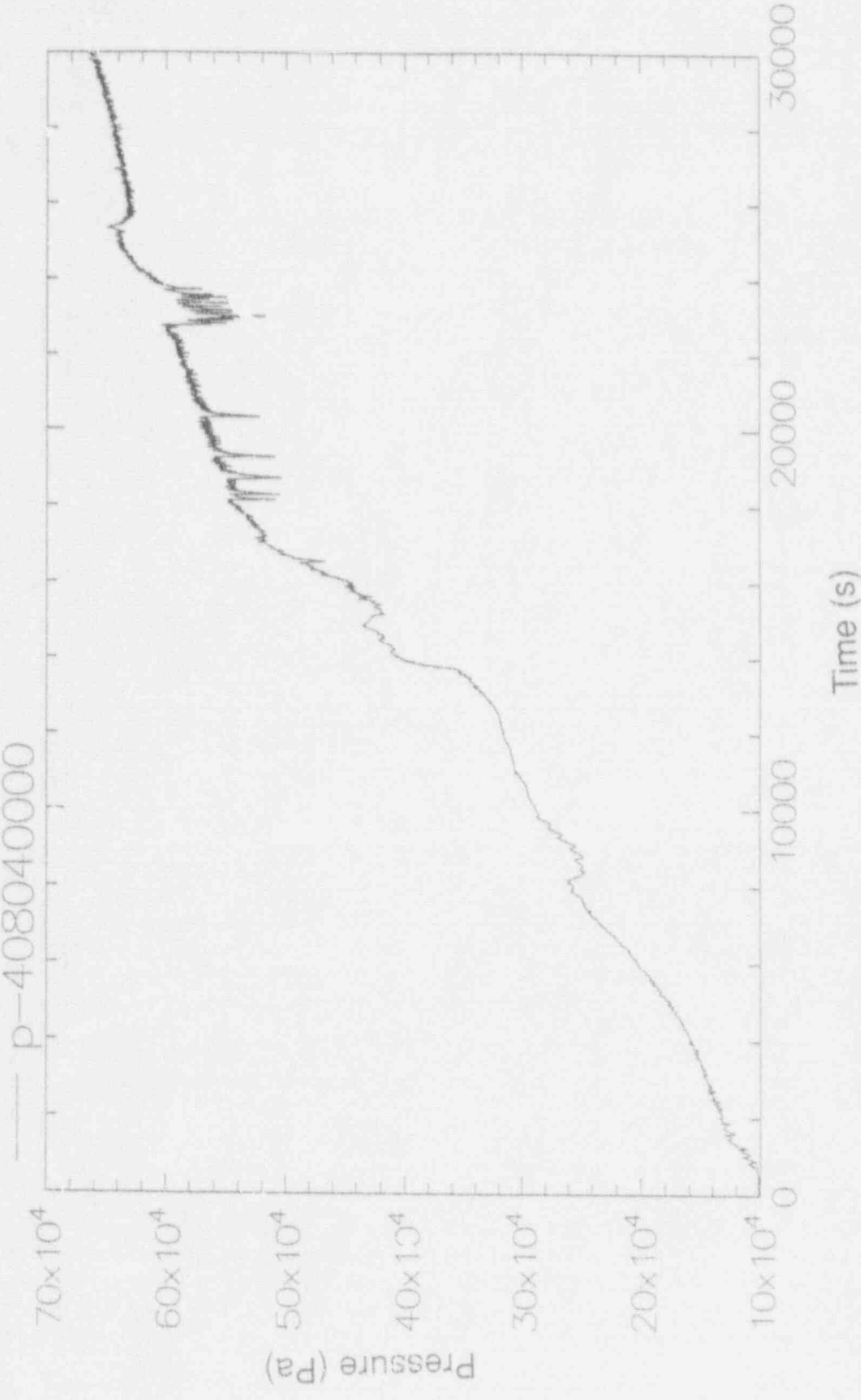

Figure 35. Primary system pressure for Case 4.

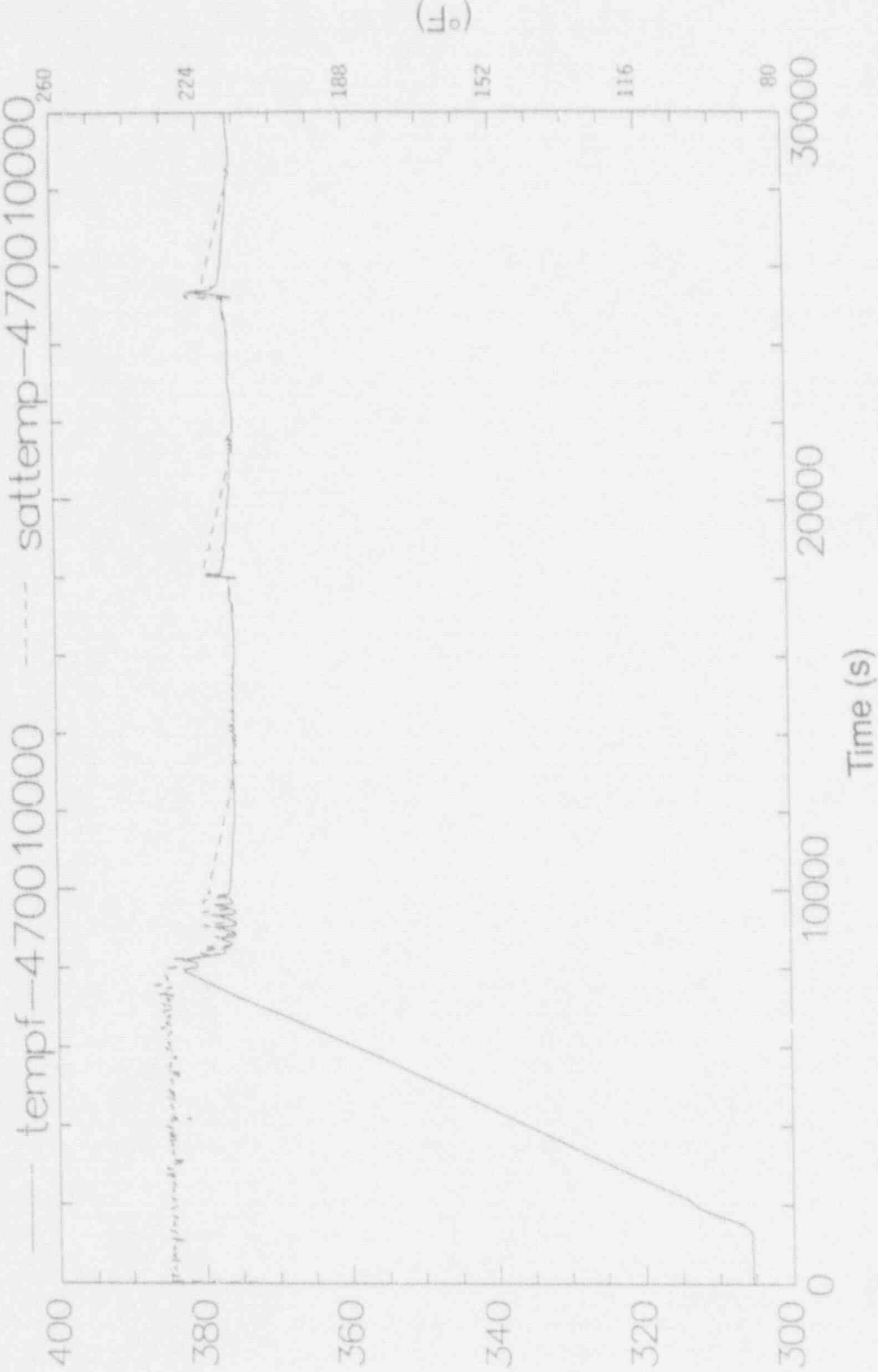

Figure 36. Comparison of water temperature to saturation temperature in steam generator secondary—Case 4.

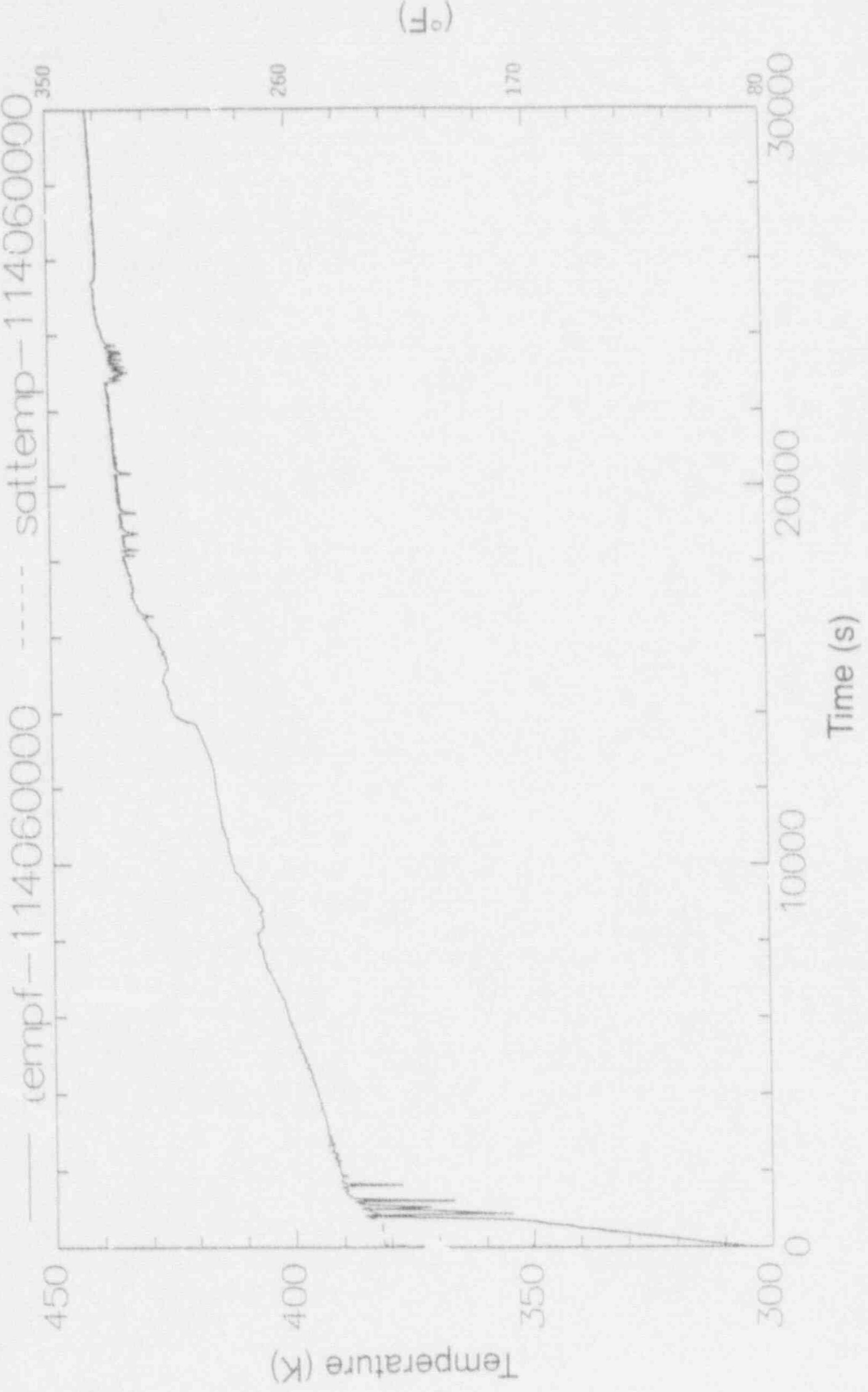

Figure 37. Water temperature in the upper core volume (114-06)—Case 4

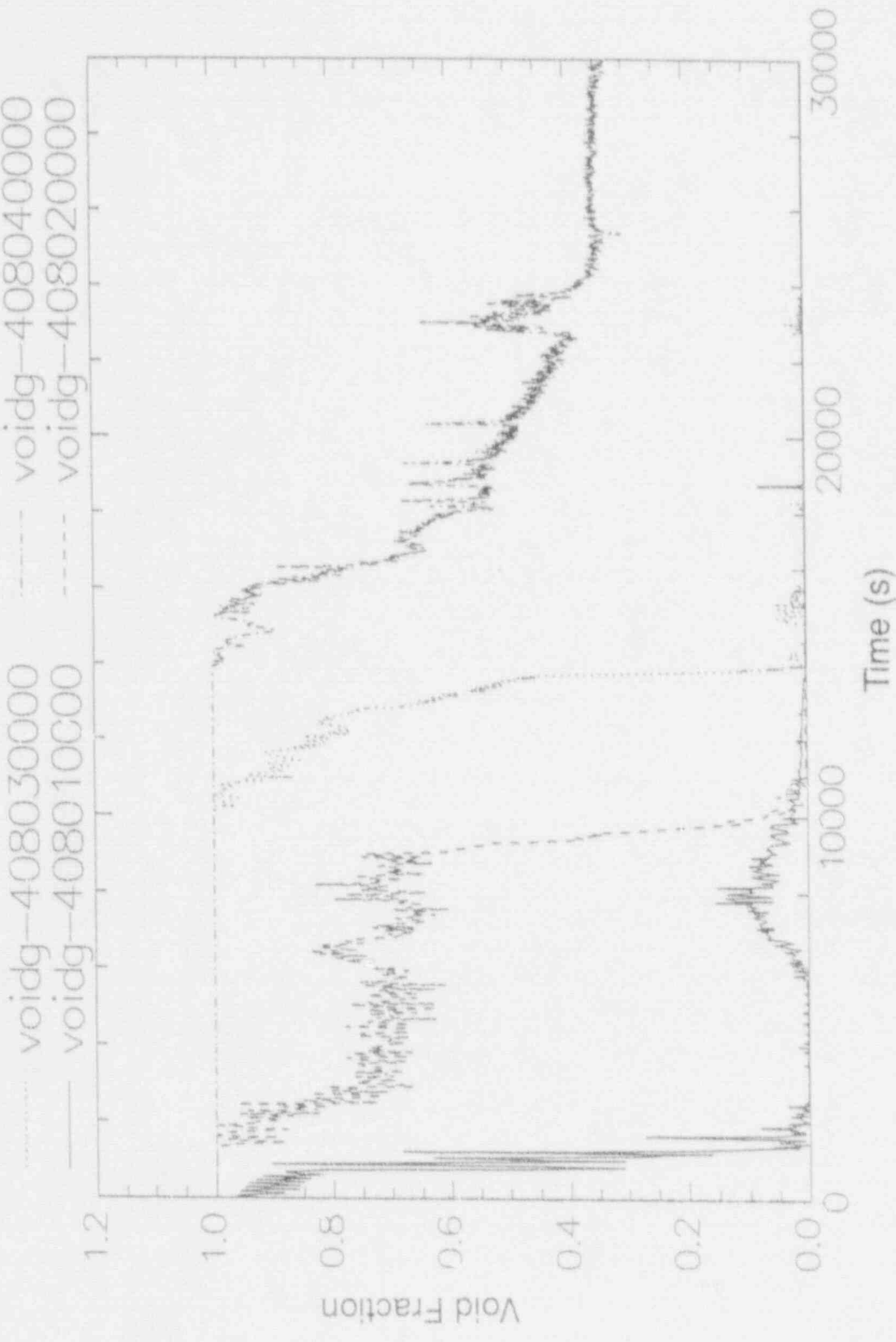

Figure 38. Void fraction in the steam generator tube volumes for the inlet plenum to the U-bend—Case 4.

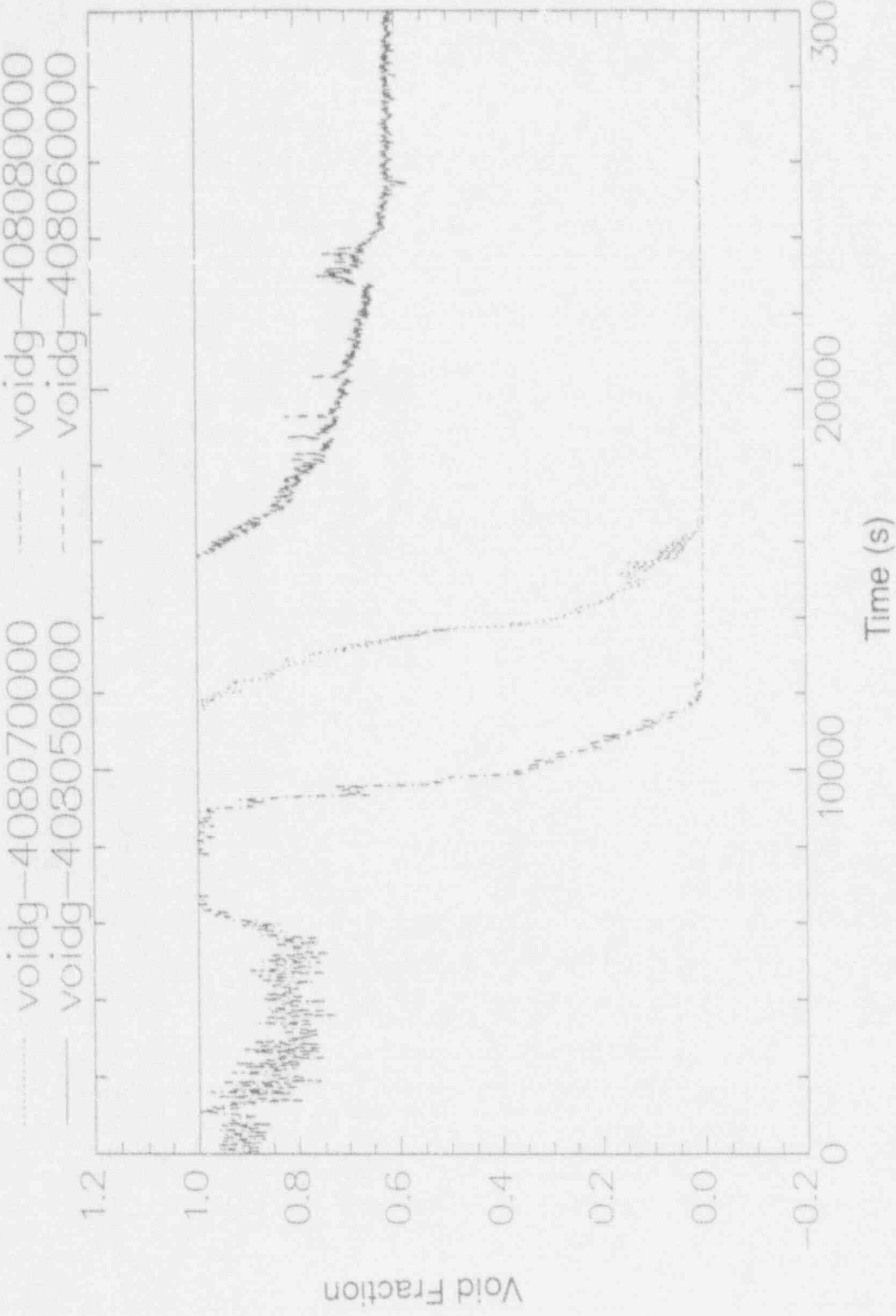

Figure 39. Void fraction in the steam generator tube volumes from the U-bend to the outlet plenum—Case 4.

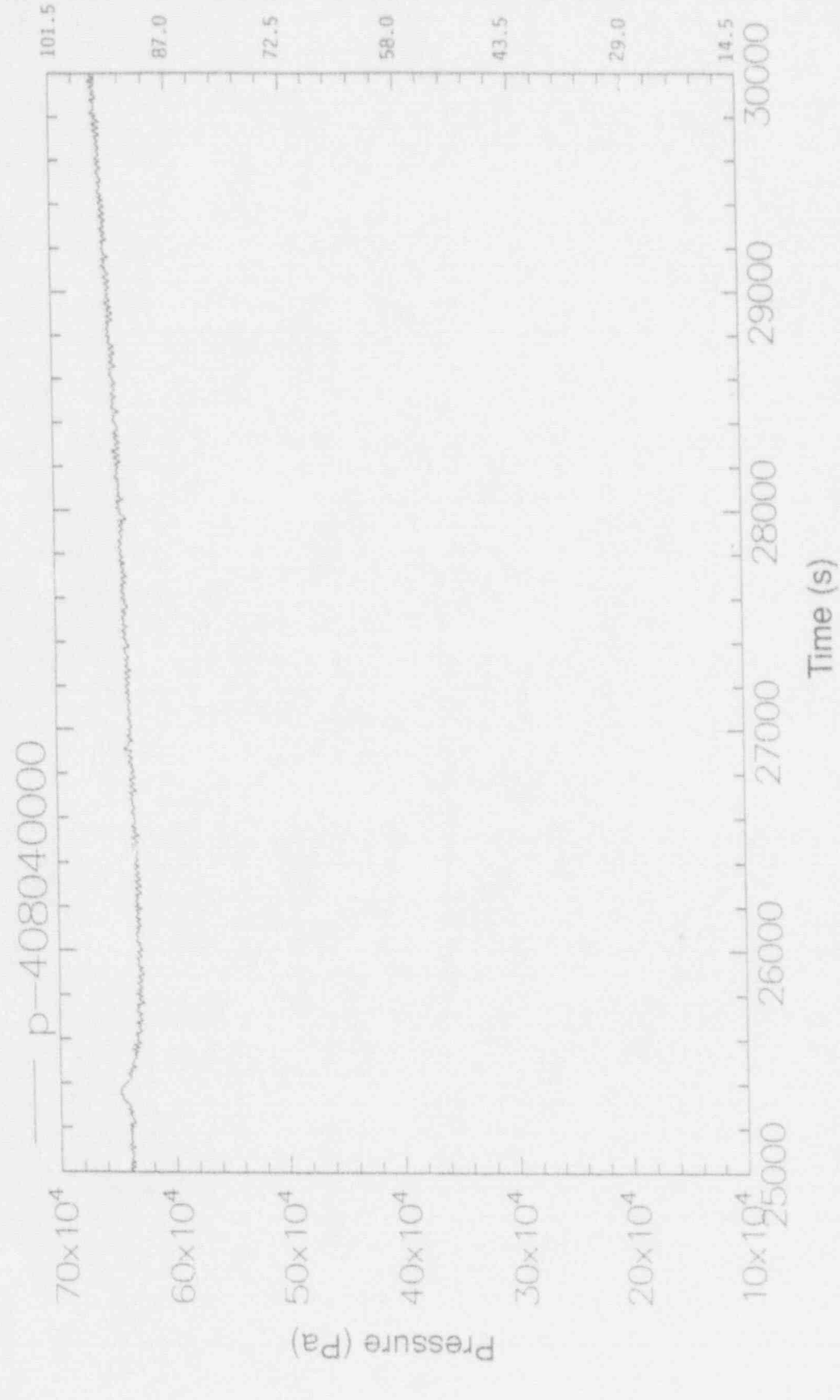

Figure 40. Steam generator primary pressure from 25,000 to 30,000 sec—Case 4.

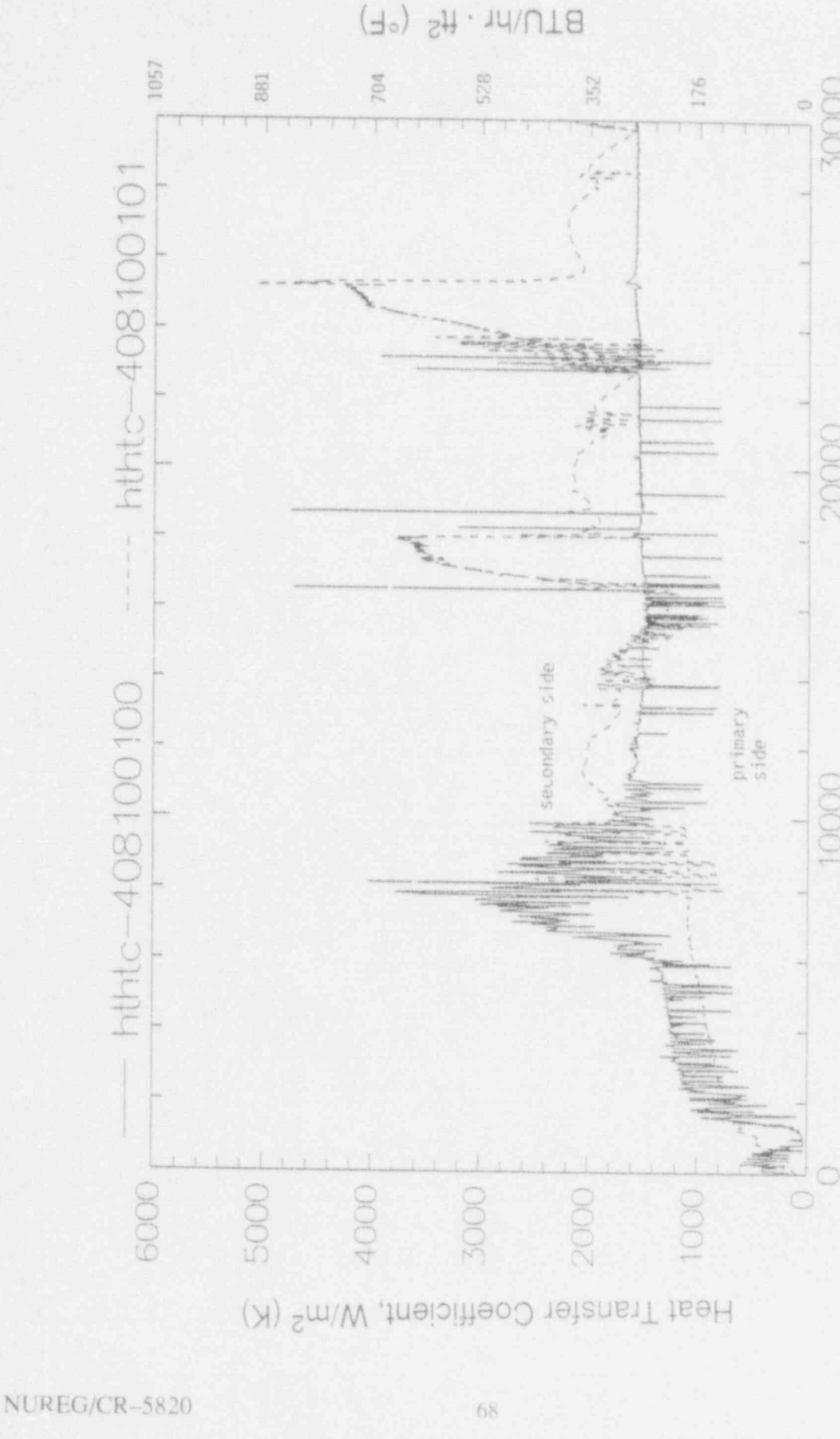

Figure 41. Heat transfer coefficients in the first volume of the steam generator primary tubes for Case 4.

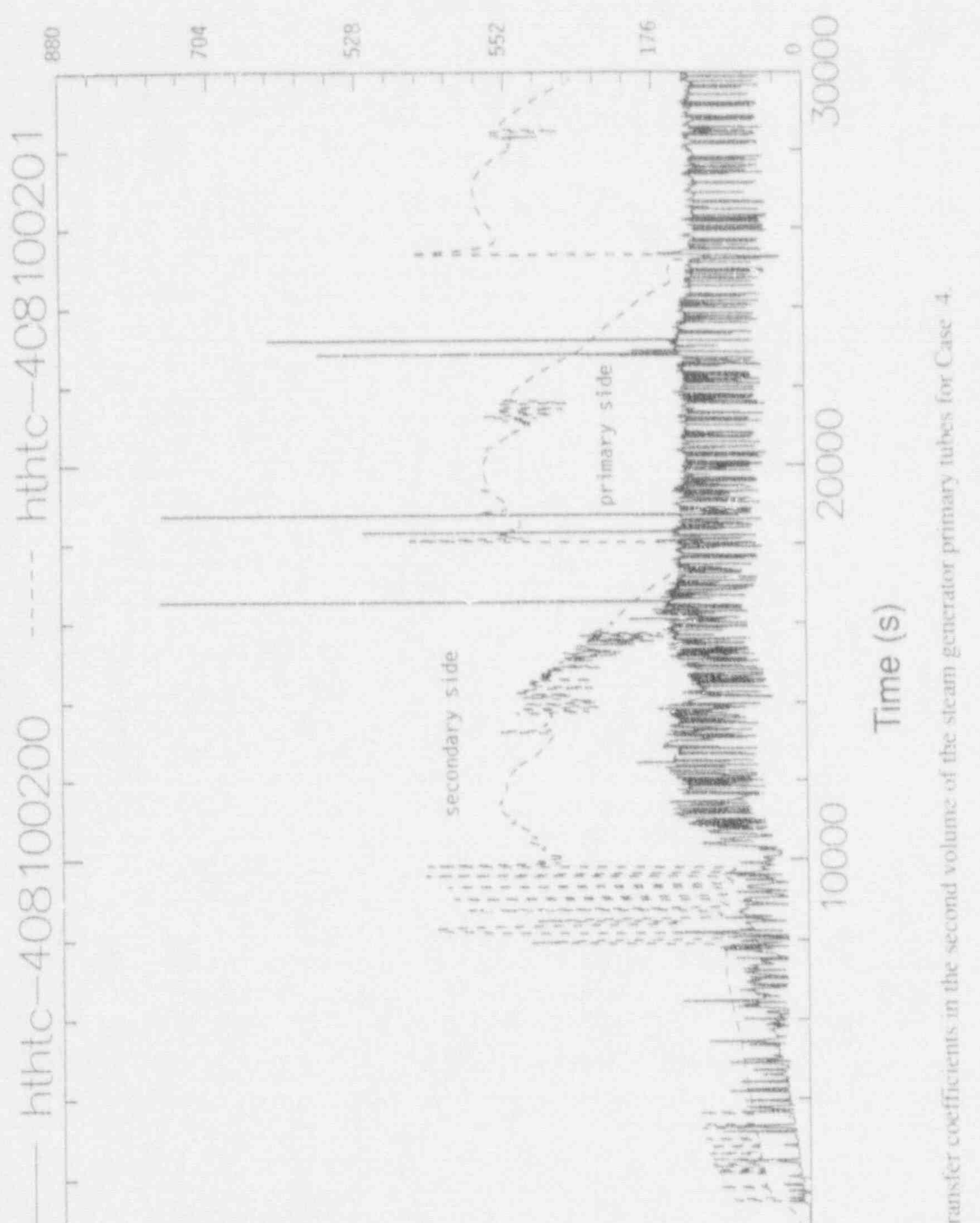

Figure 42. Heat transfer coefficients in the second volume of the steam generator primary tubes for Case 4.

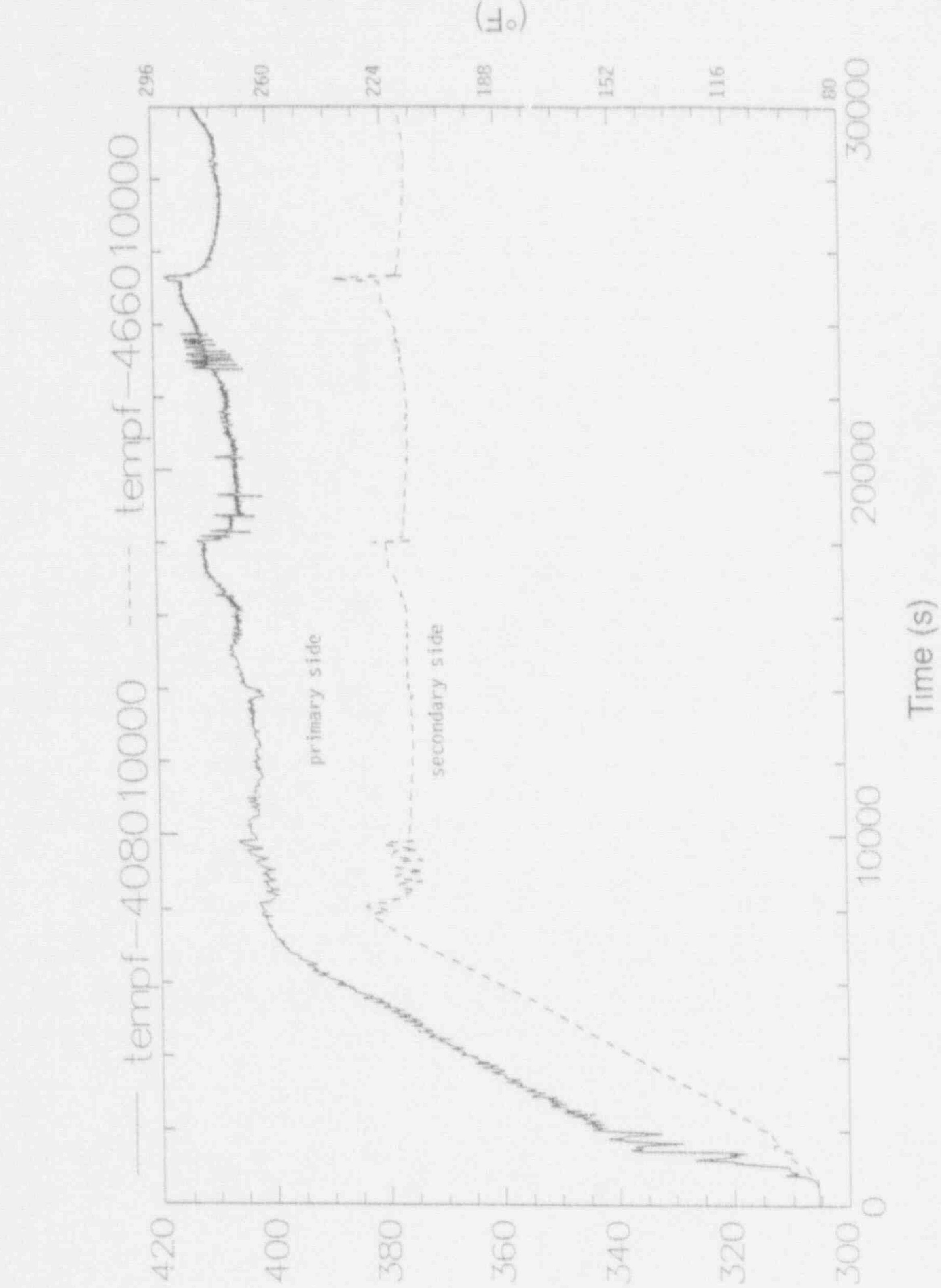

Figure 43. Primary and secondary temperature in the first steam generator volume—Case 4.

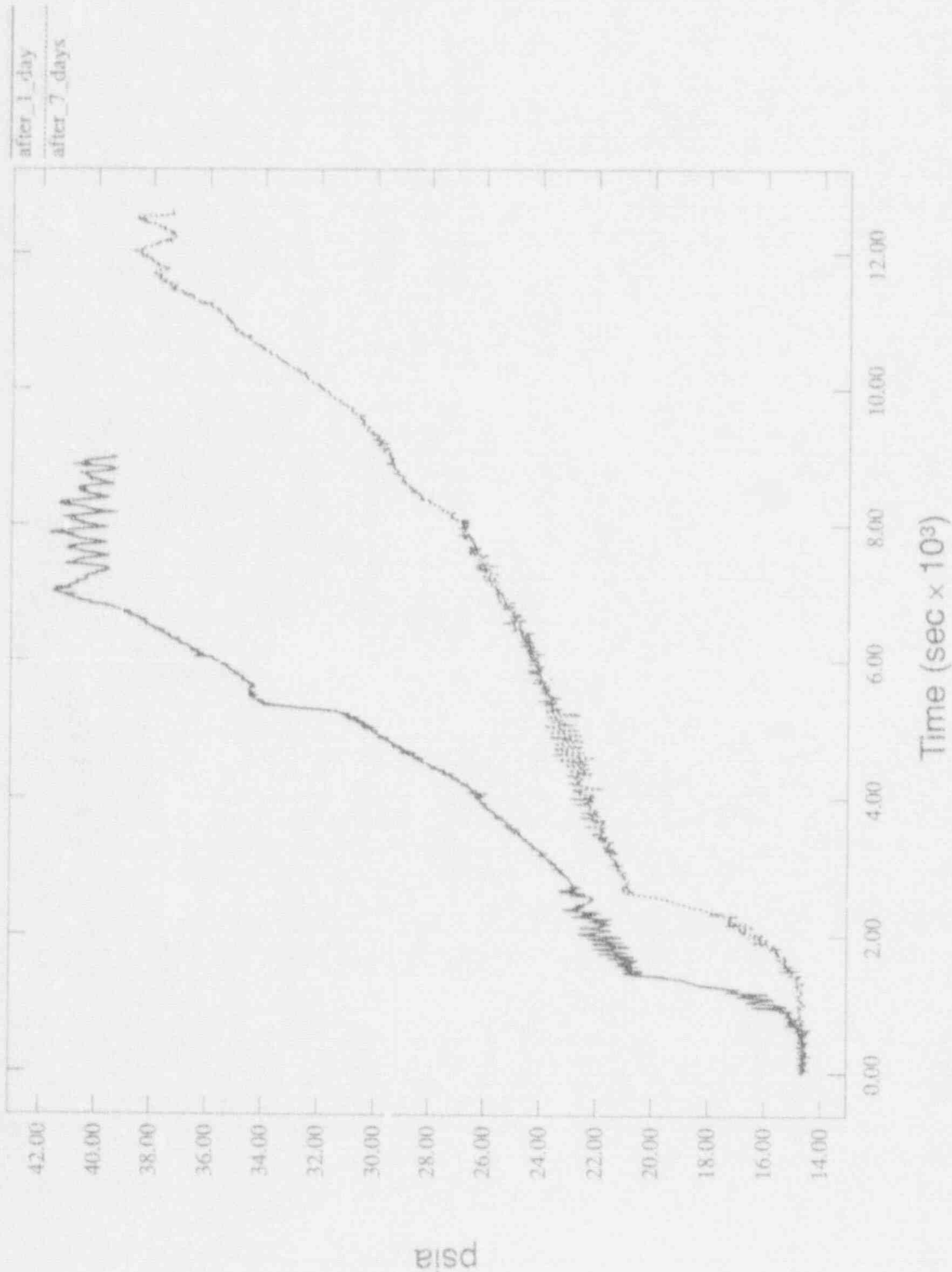

Figure 44. Comparison of primary system pressure for Cases 1 and 2.

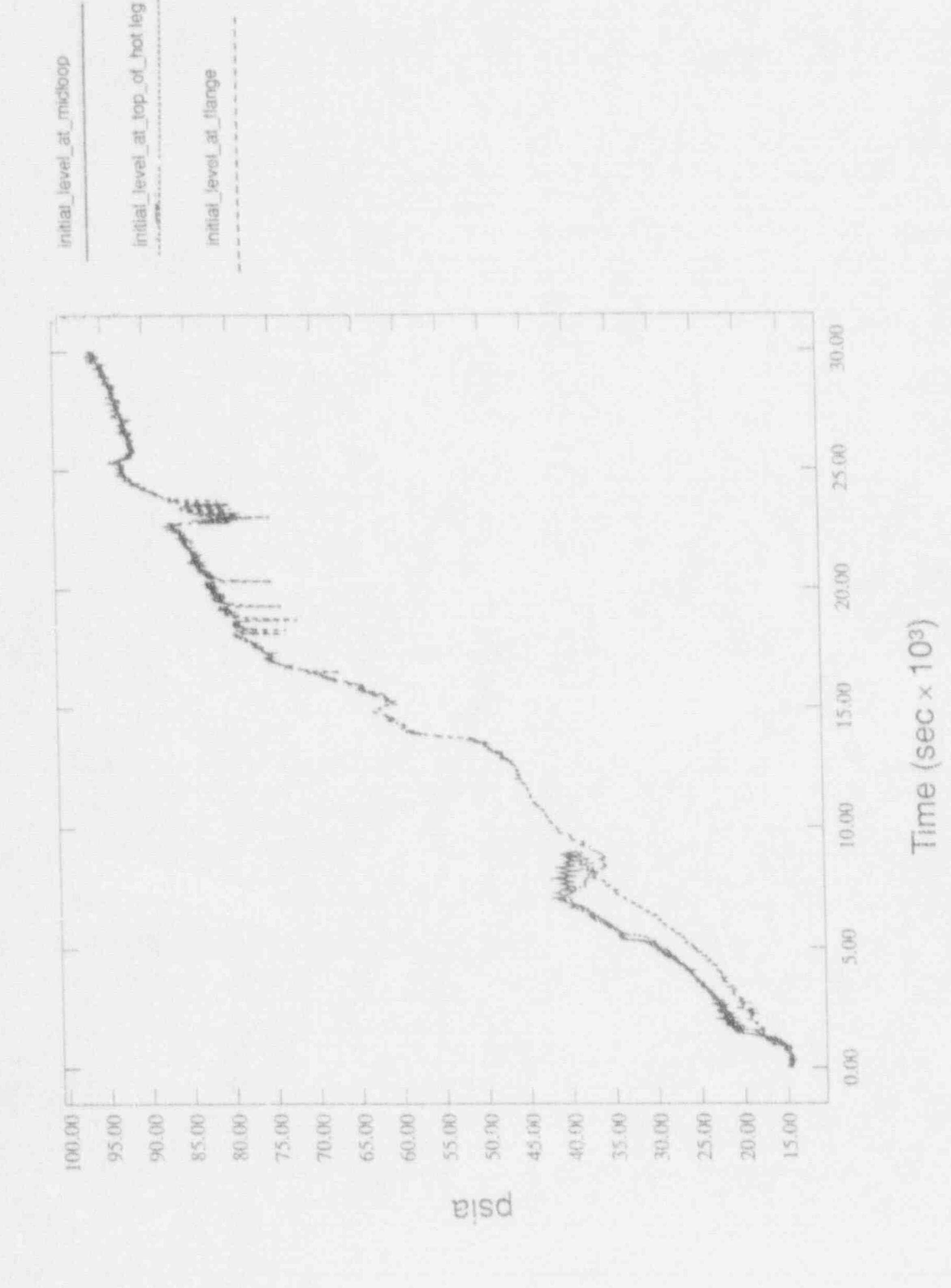

Figure 45. Sensitivity of RCS pressure to variation in vessel water level.

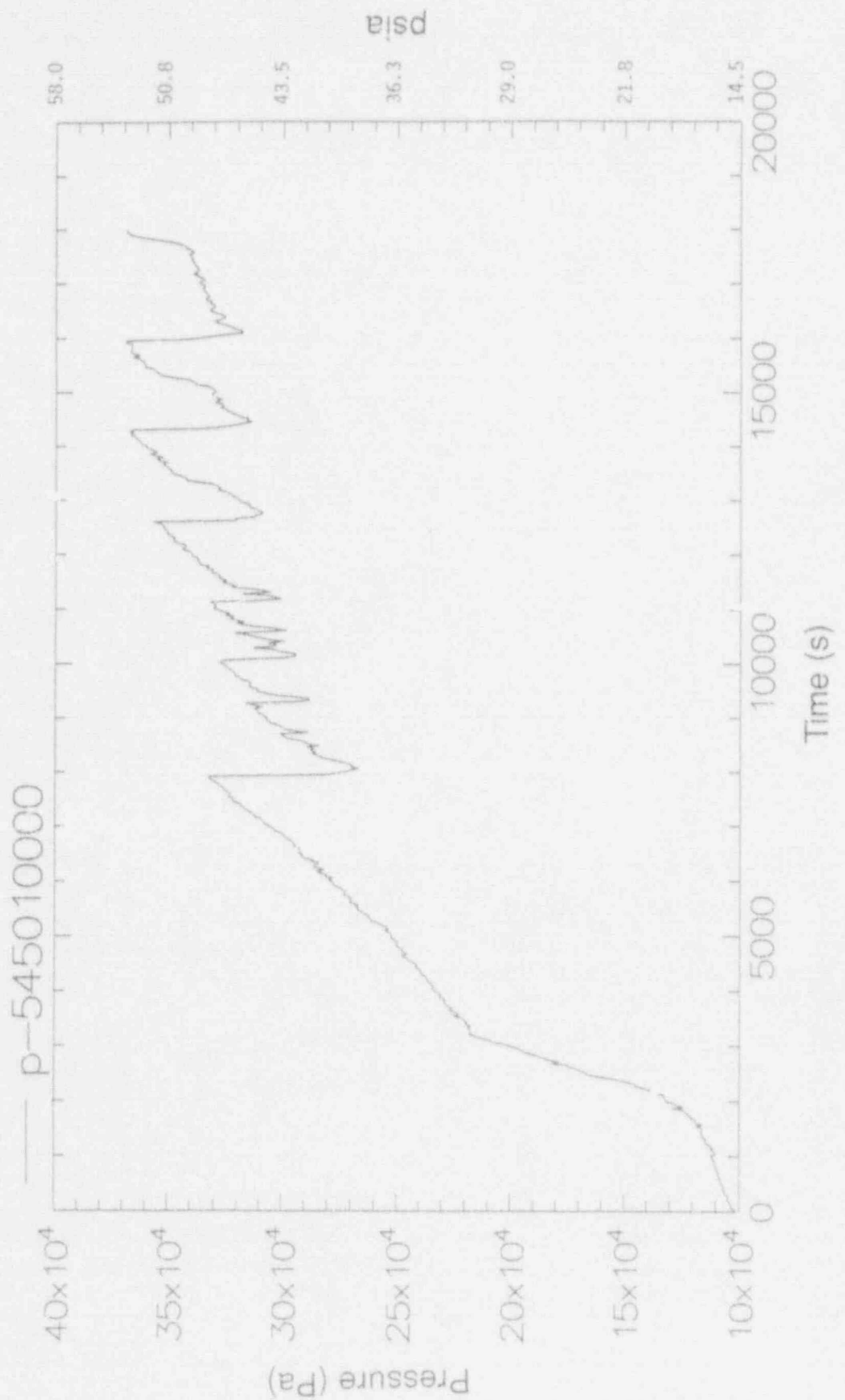

Figure 46. Primary system pressure for once through steam generator analysis—Oconee Case 1.

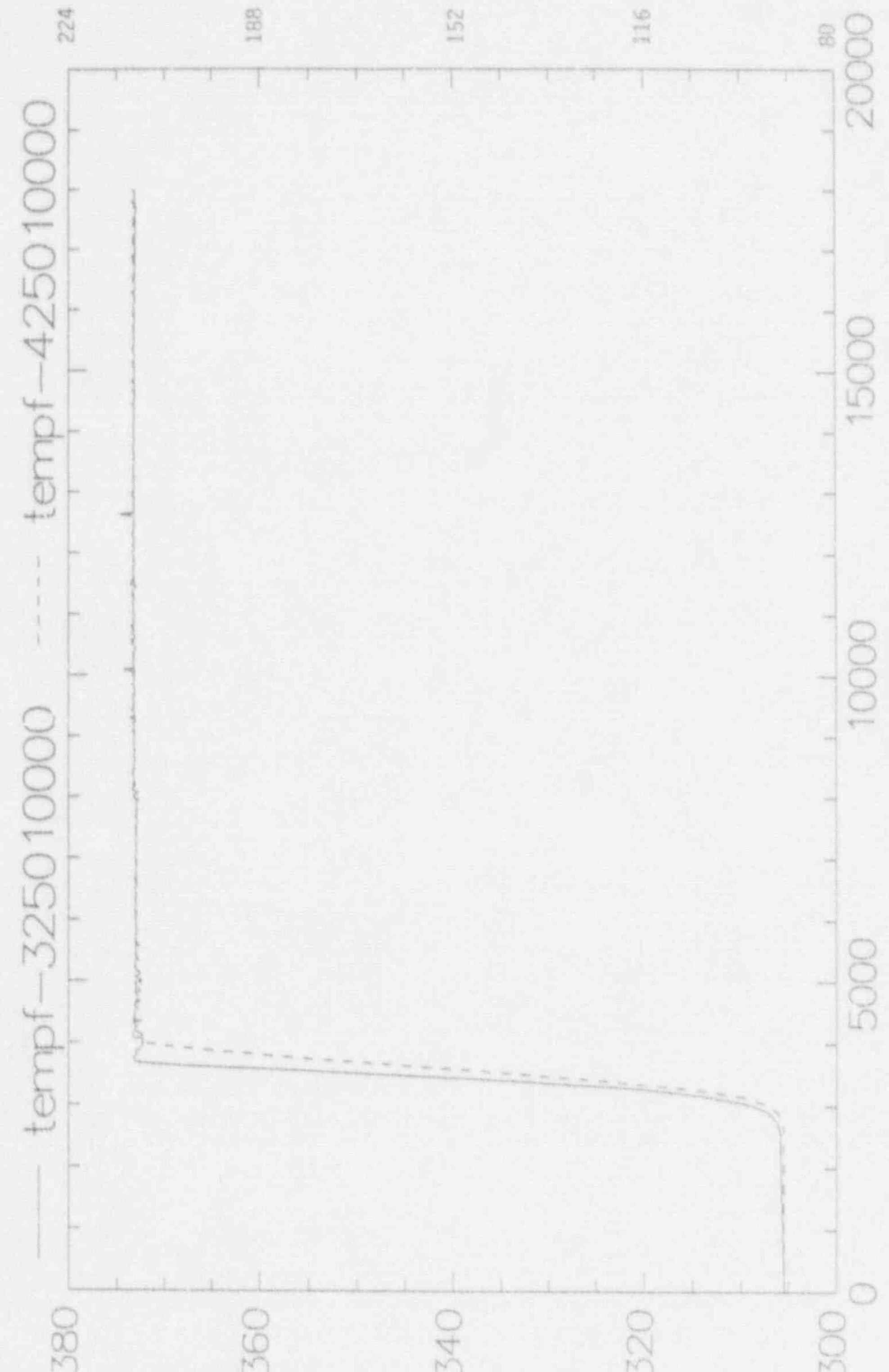

Figure 47. Water temperature in the topmost steam generator secondary volume—Oconee Case 1.

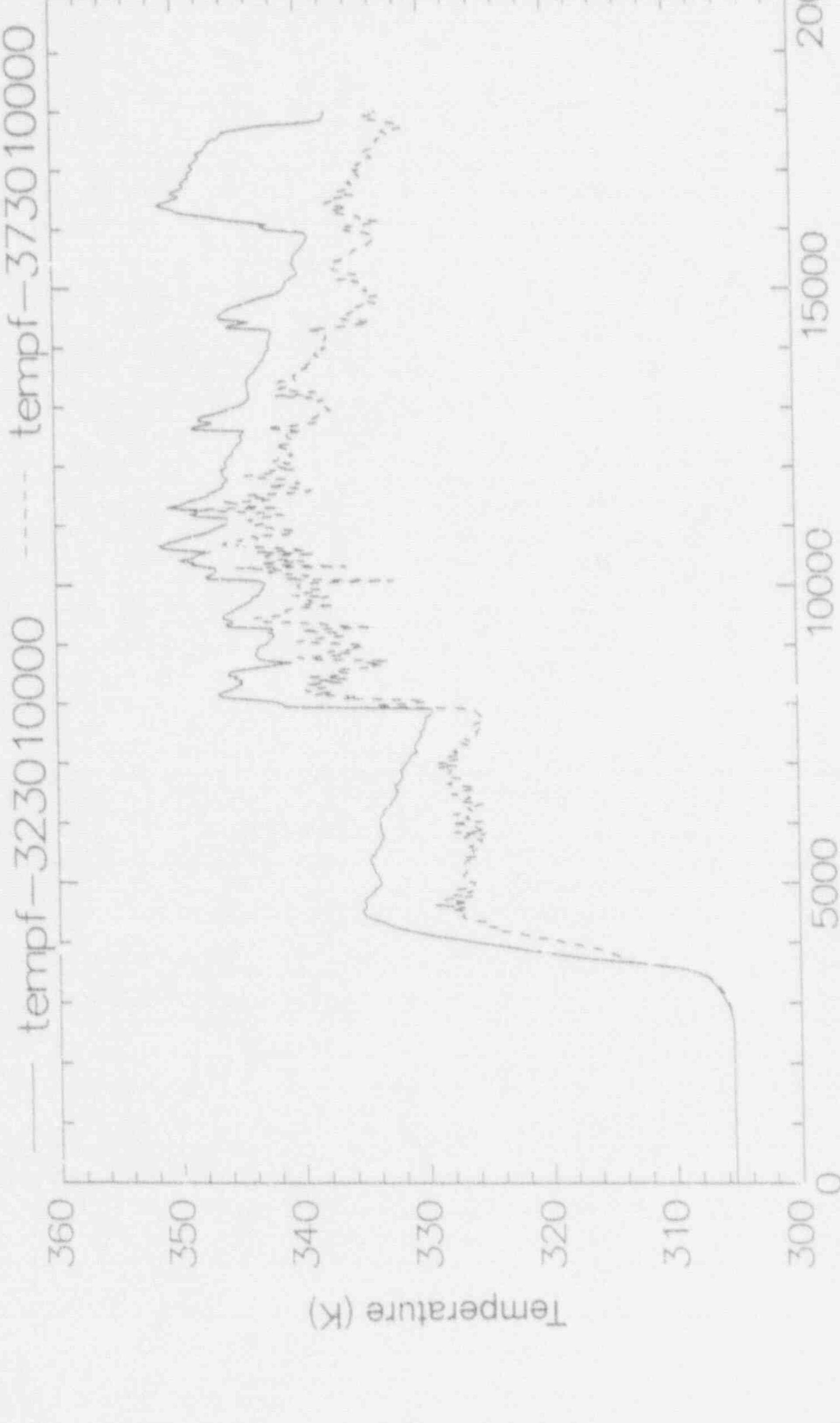

Figure 48. Water temperature in components 323 and 373 of Loop A SG secondary—Oconee Case 1.

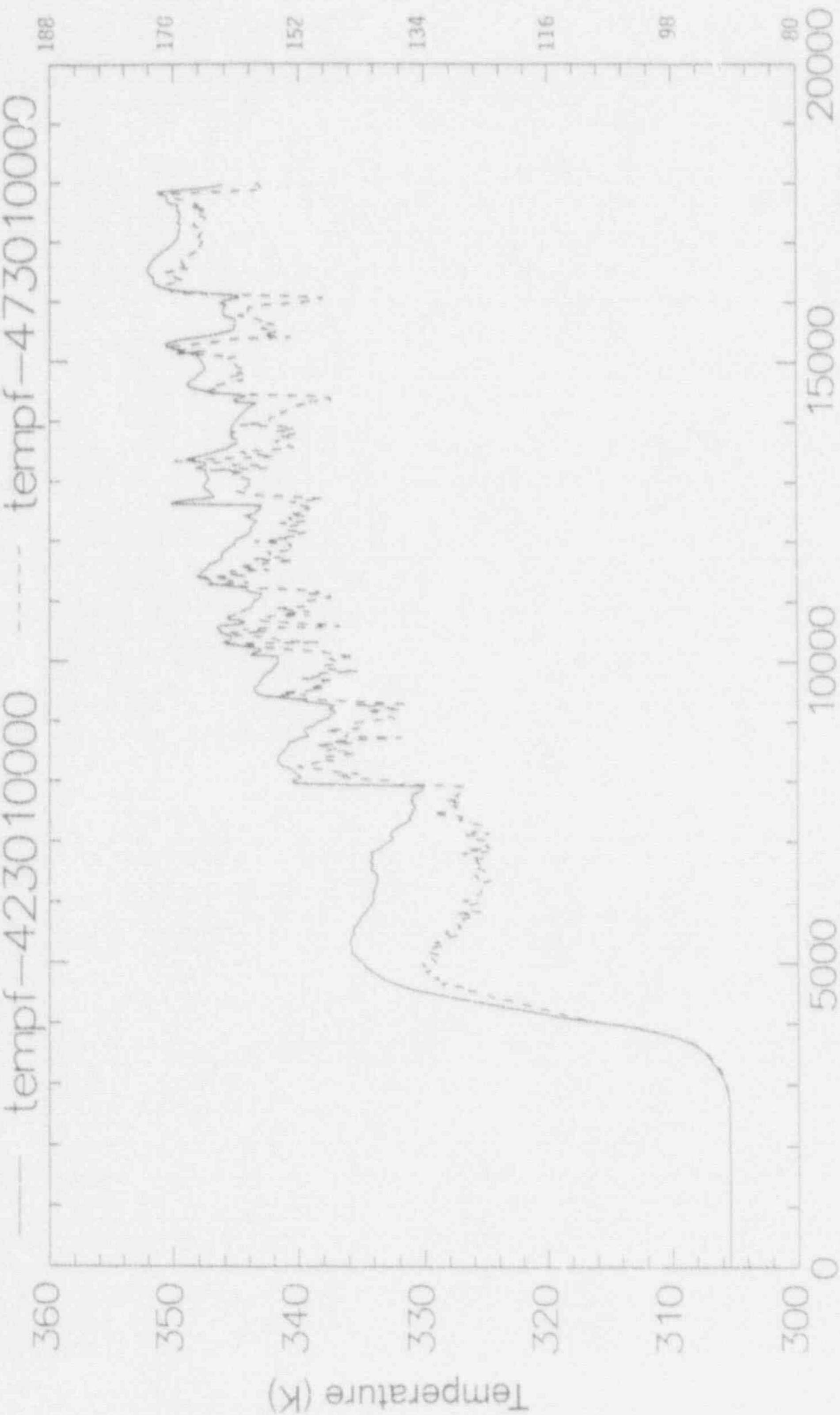

Figure 49. Water temperature in components 423 and 473 of Loop B SG secondary—Oconee Case 1.

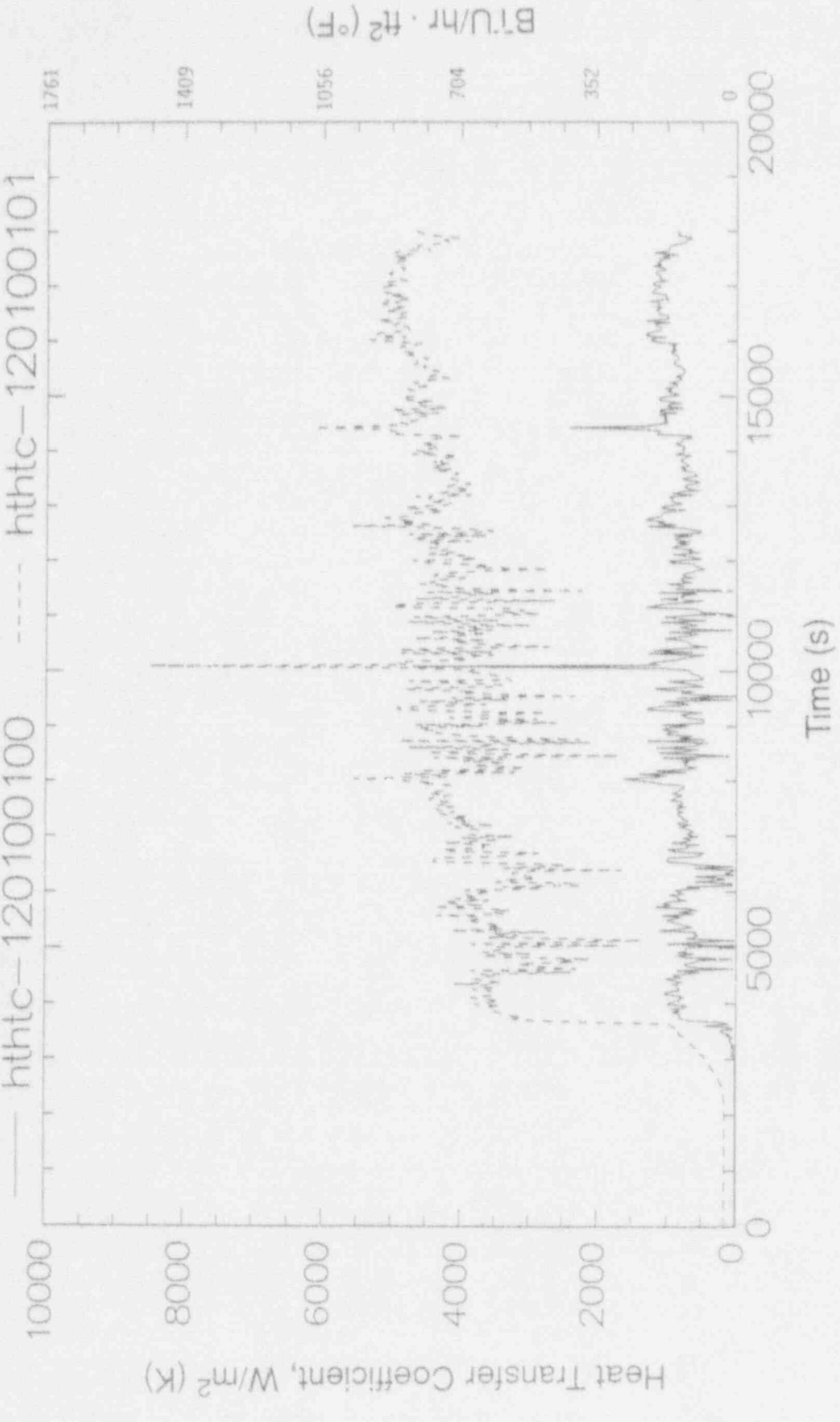

Figure 50. Heat transfer coefficient in volume 1 of Loop A SG primary tubes—Oconee Case 1.

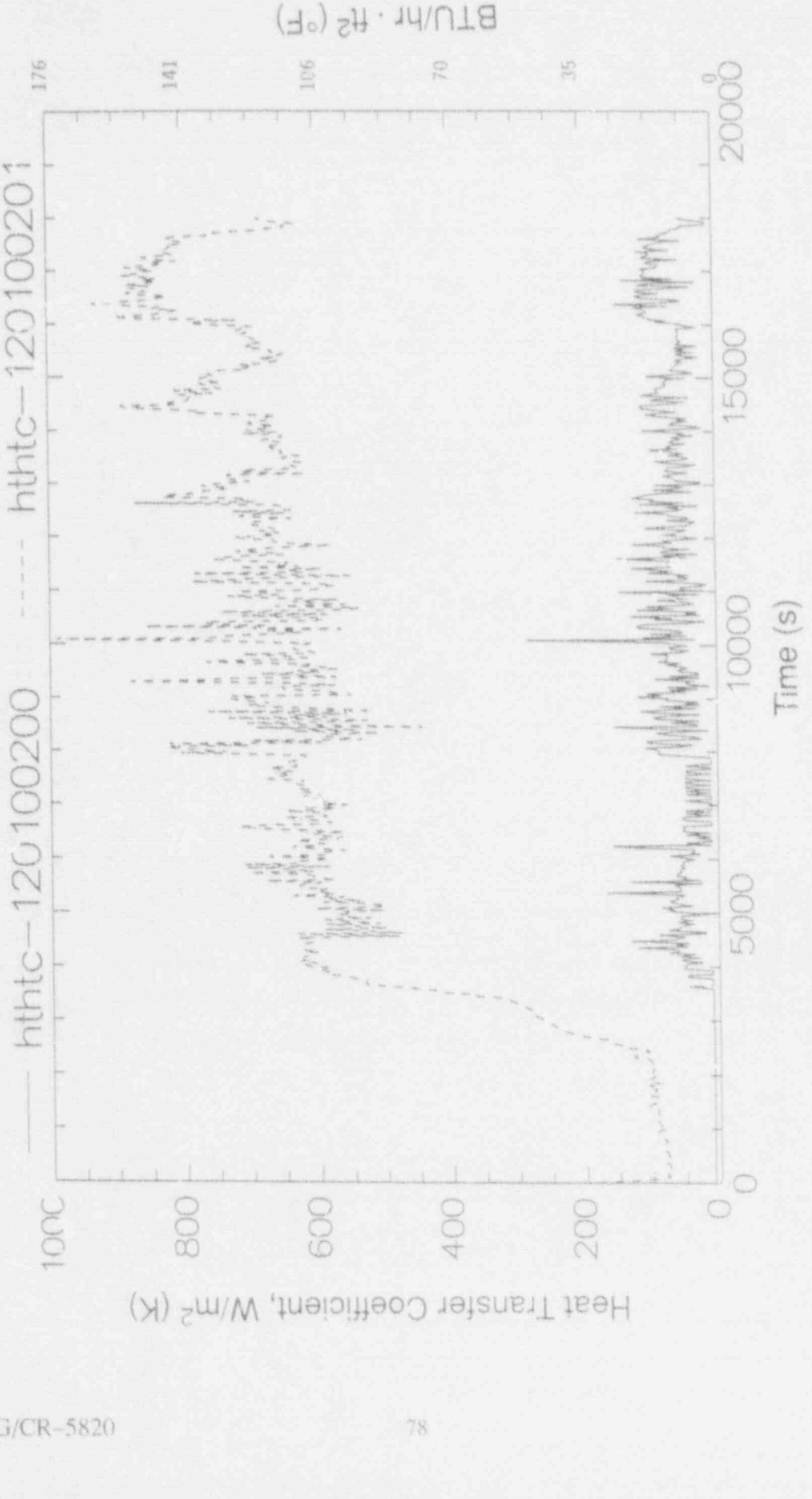

Figure 51. Heat transfer coefficient in volume 2 of Loop A SG primary tubes—Oconee Case 1.

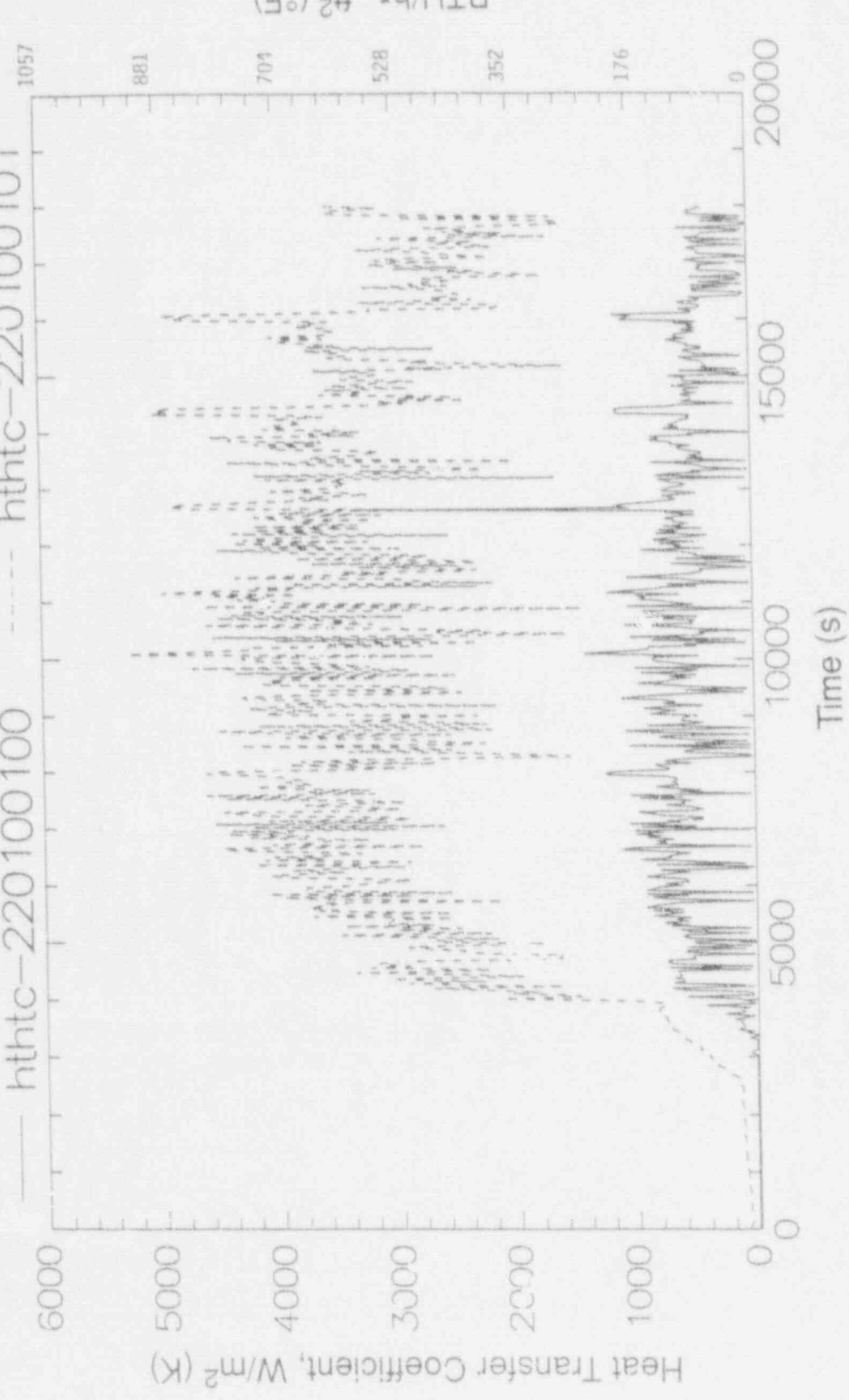

Figure 52. Heat transfer coefficient in volume 1 of Loop B SG primary tubes—Oconee Case 1

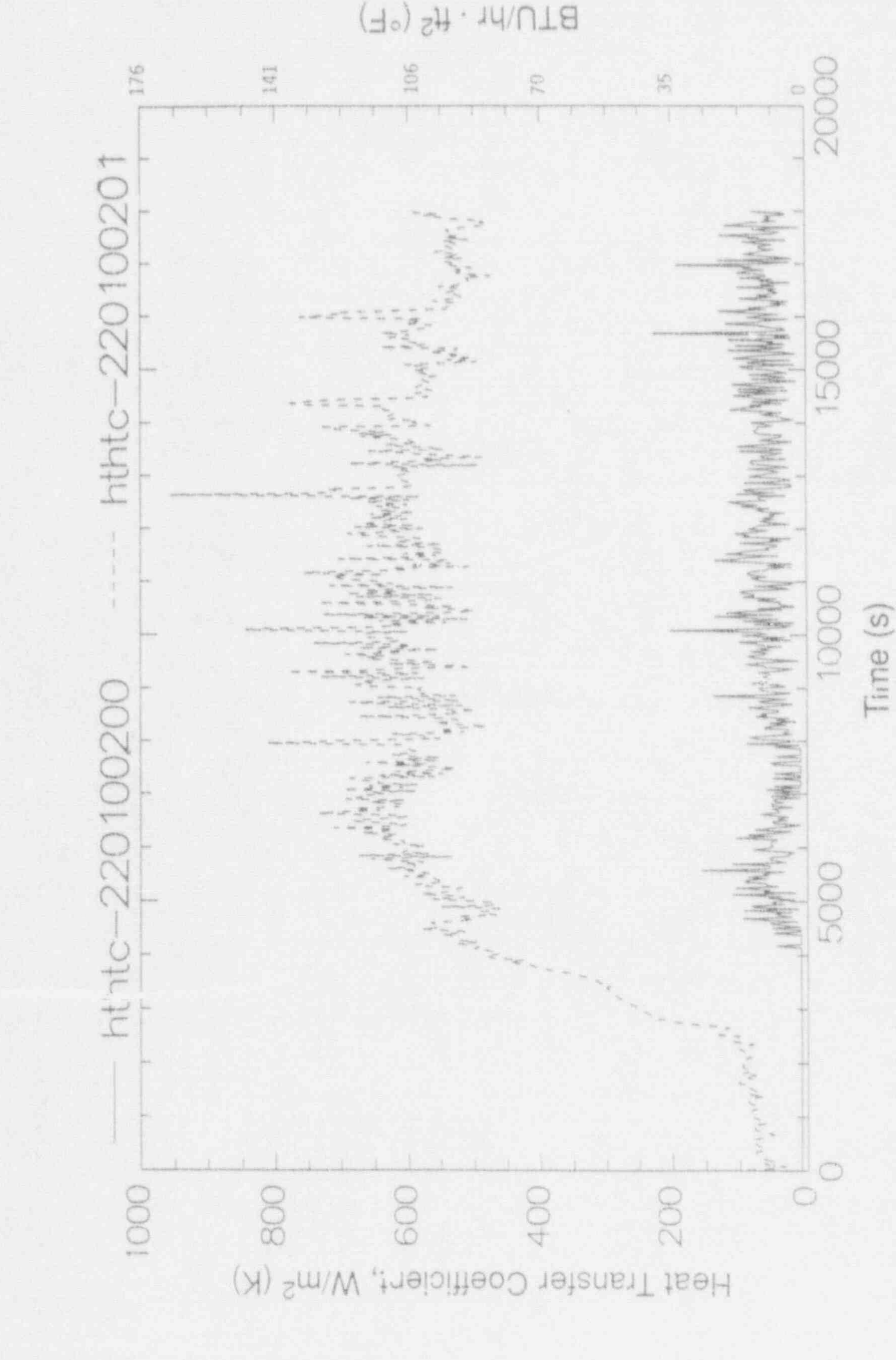

Figure 53. Heat transfer coefficient in volume 2 of Loop B SG primary tubes—Oconee Case 1.

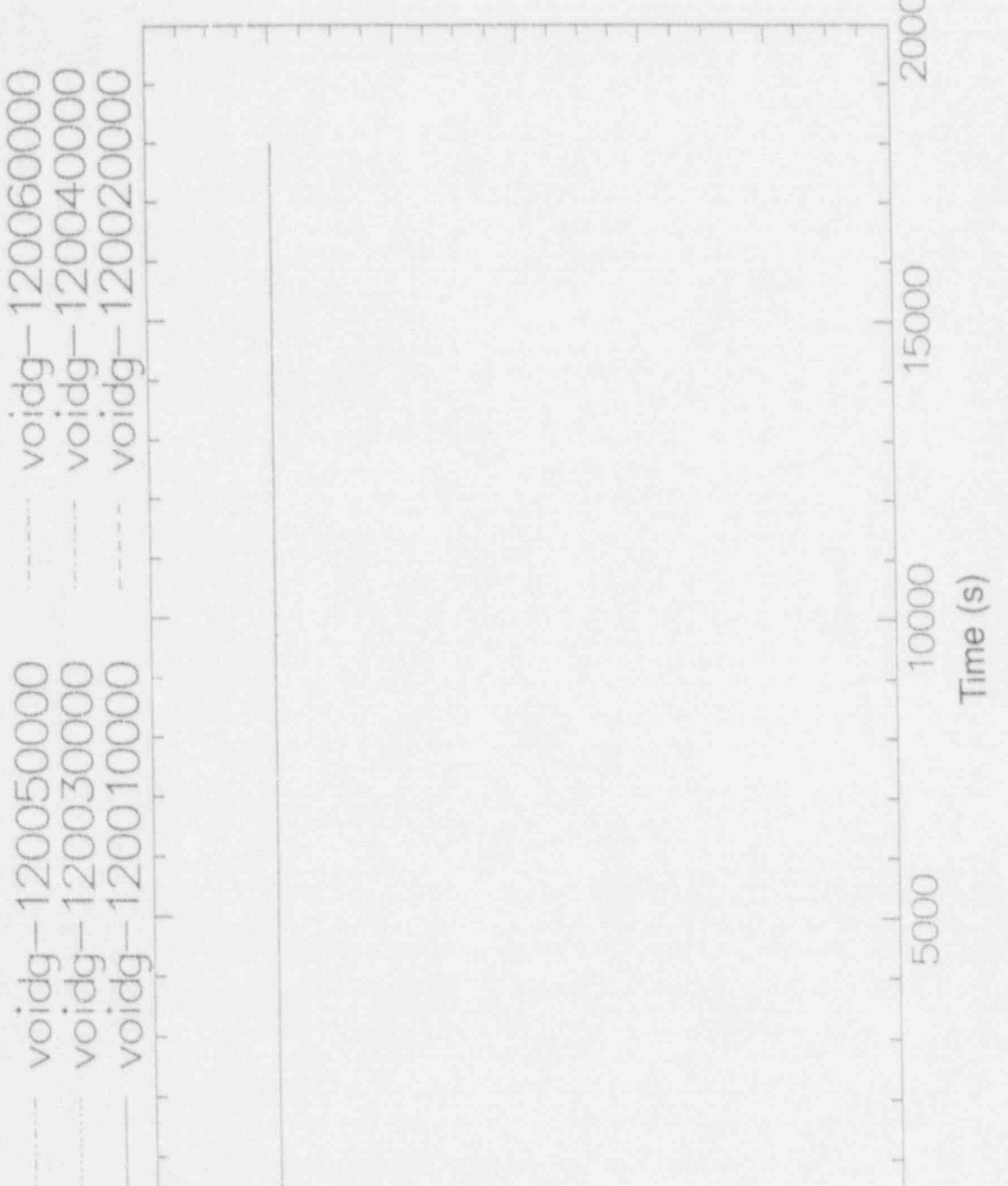

Figure 54. Vapor void fraction in volumes 1-6 of Loop A SG primary tubes—Oconee Case 1.

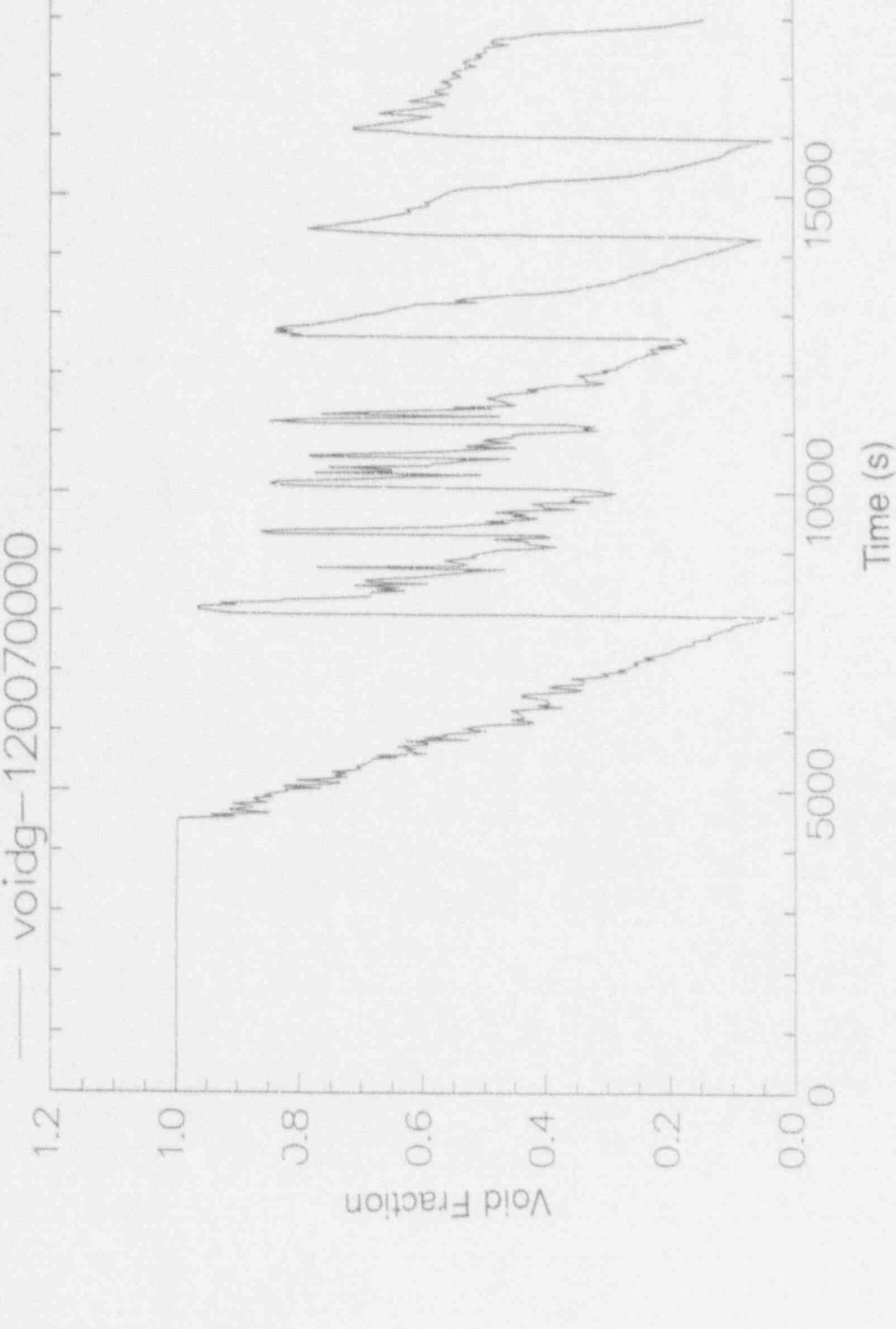

Figure 55. Vapor void fraction in volume 7 of Loop A SG primary tubes—Oconee Case 1.

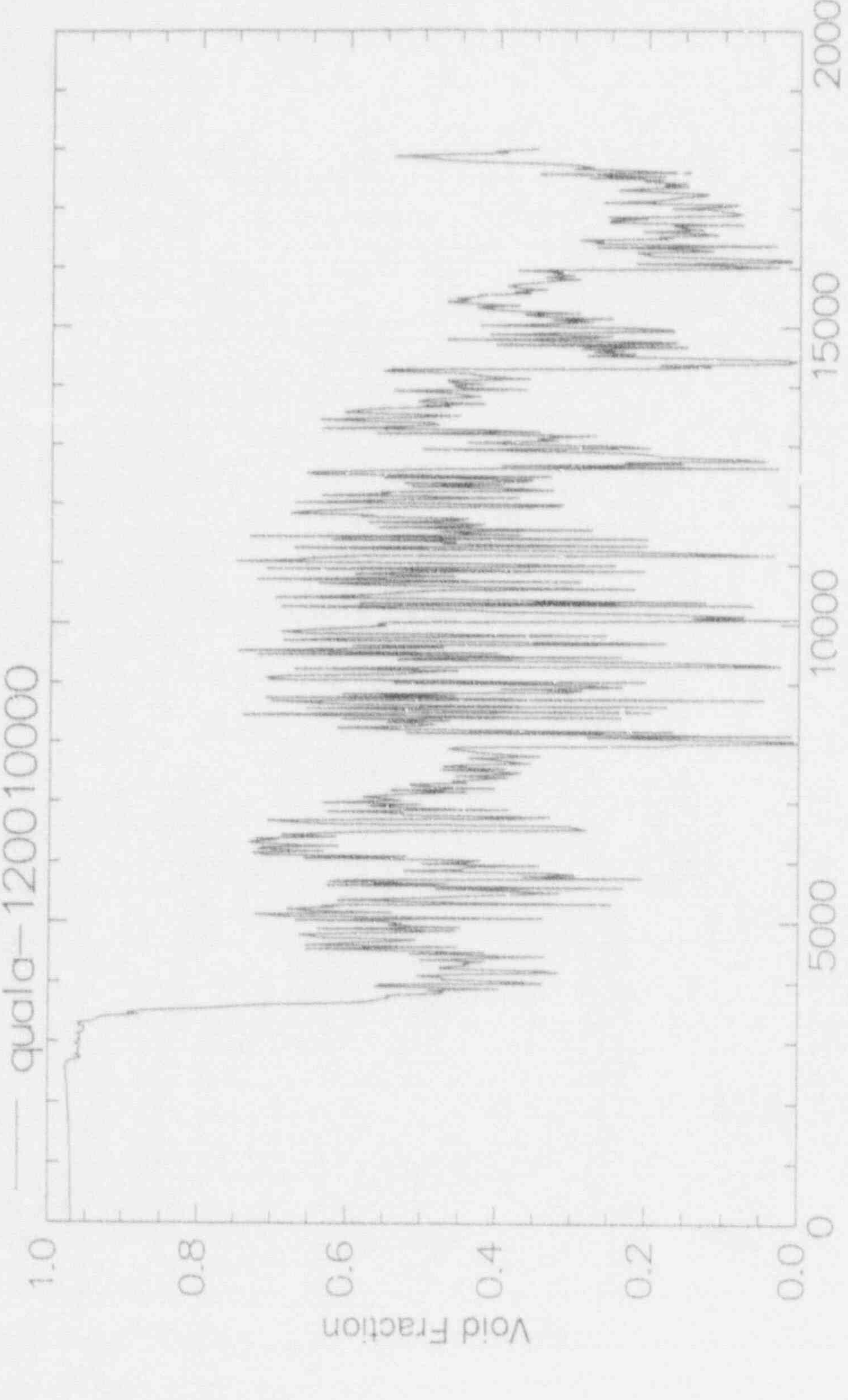

Figure 56. Noncondensible mass fraction in volume 1 of Loop A SG primary tubes—Oconee Case 1.

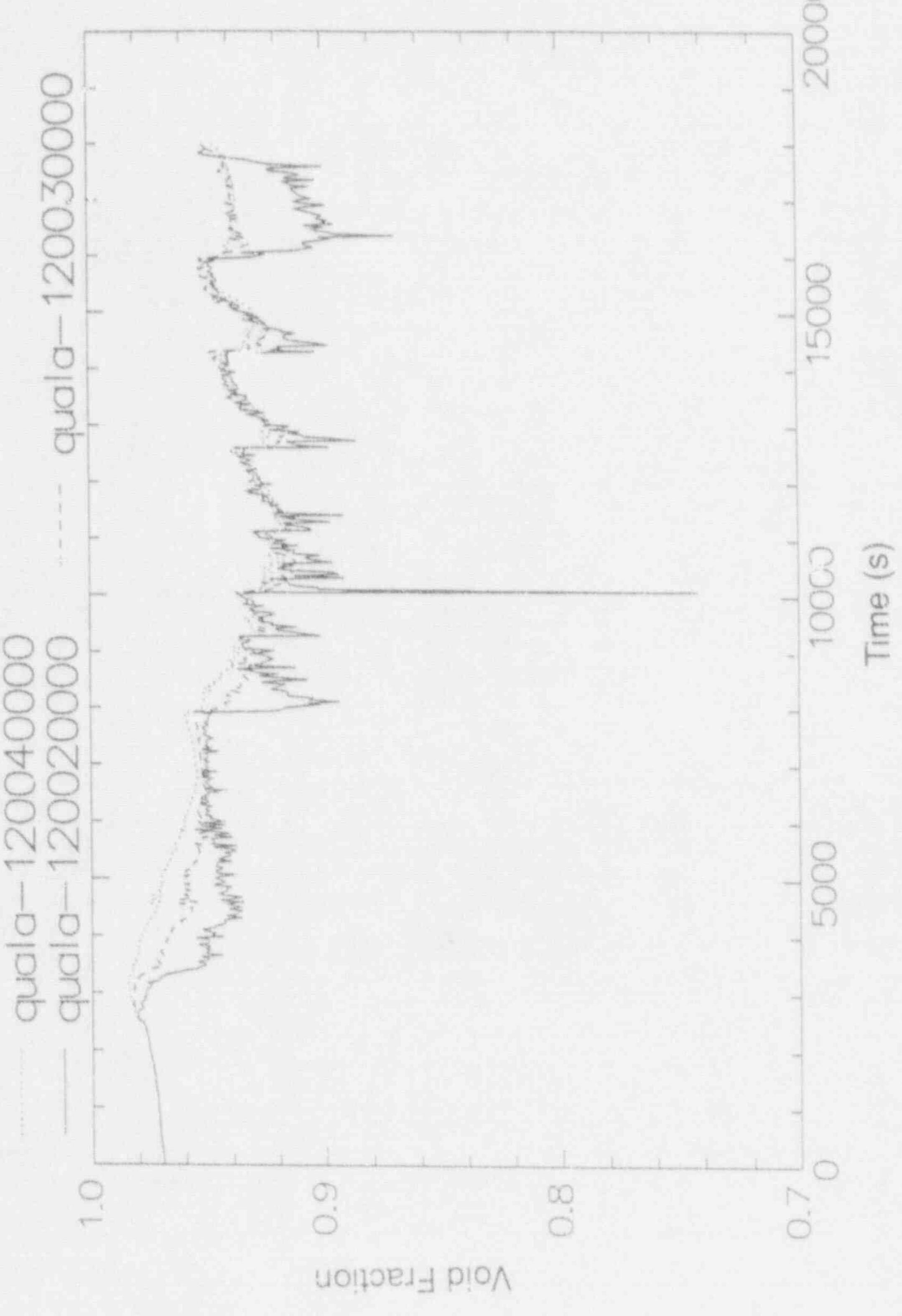

Figure 57. Noncondensible mass fraction in volumes 2-4 of Loop A SG primary tubes—Oconee Case 1.

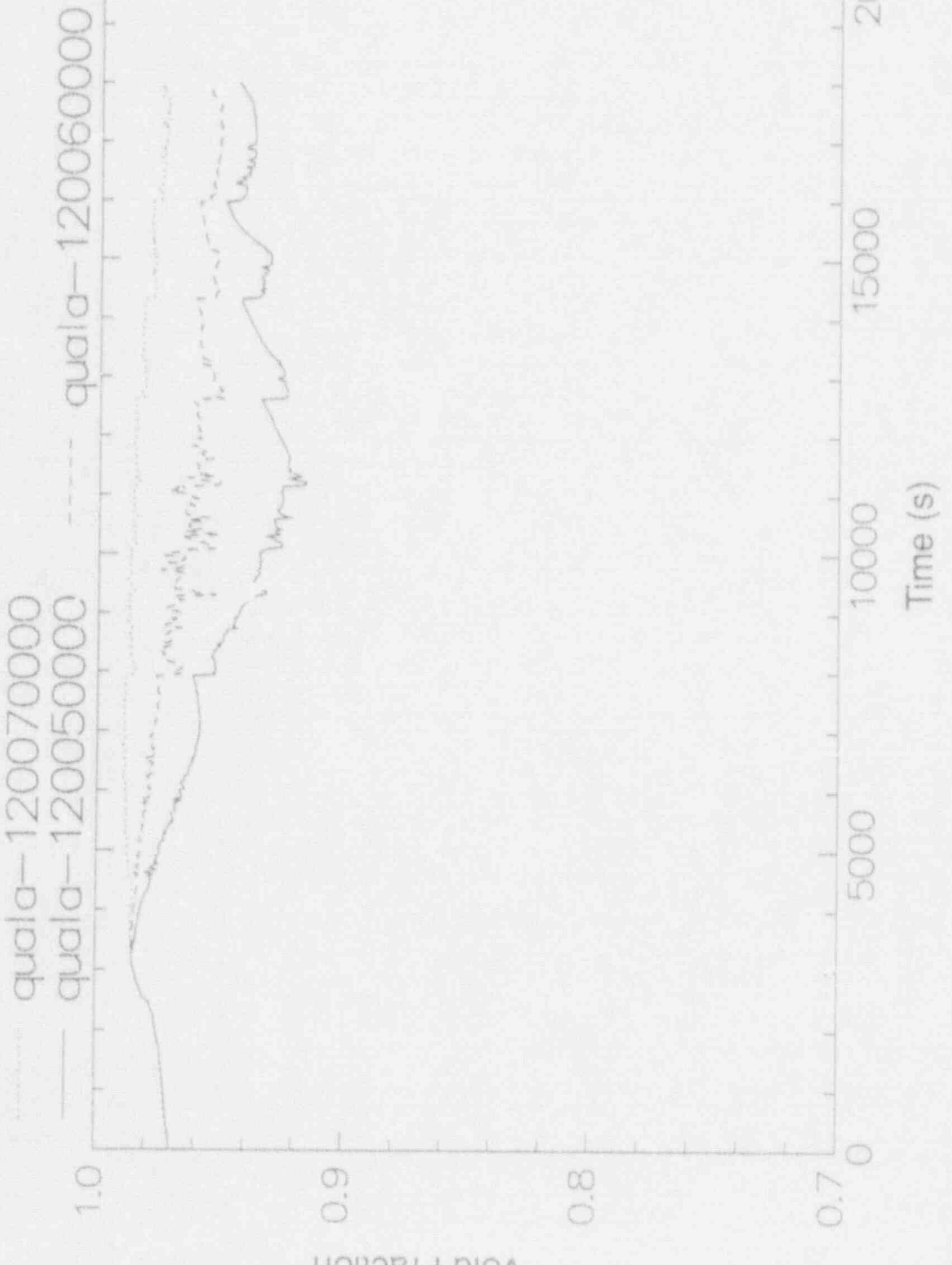

Figure 58. Noncondensible mass fraction in volumes 5-7 of Loop A SG primary tubes—Oconee Case 1.

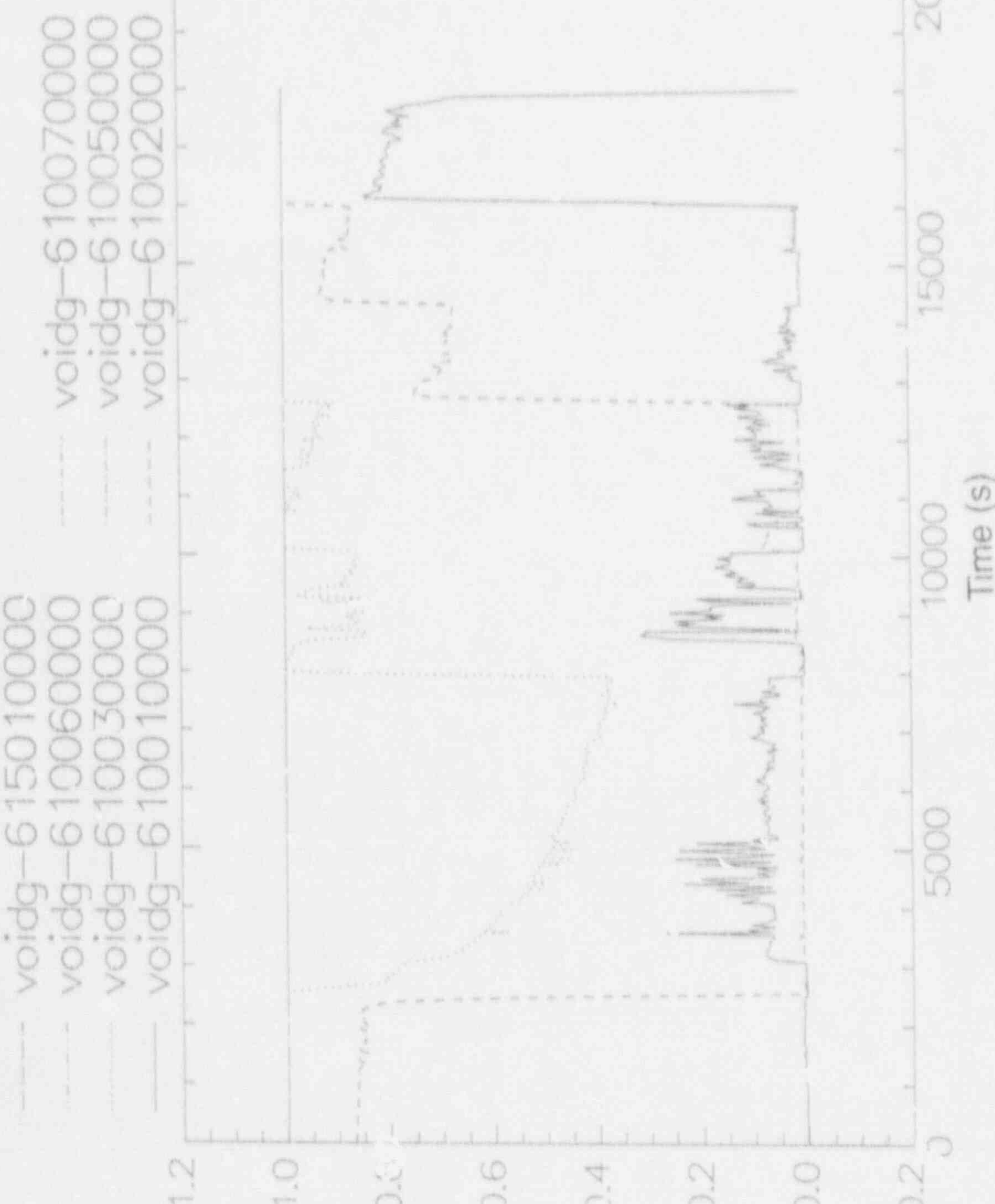

Figure 59. Vapor void fraction in the pressurizer volumes—Oconee Case 1.

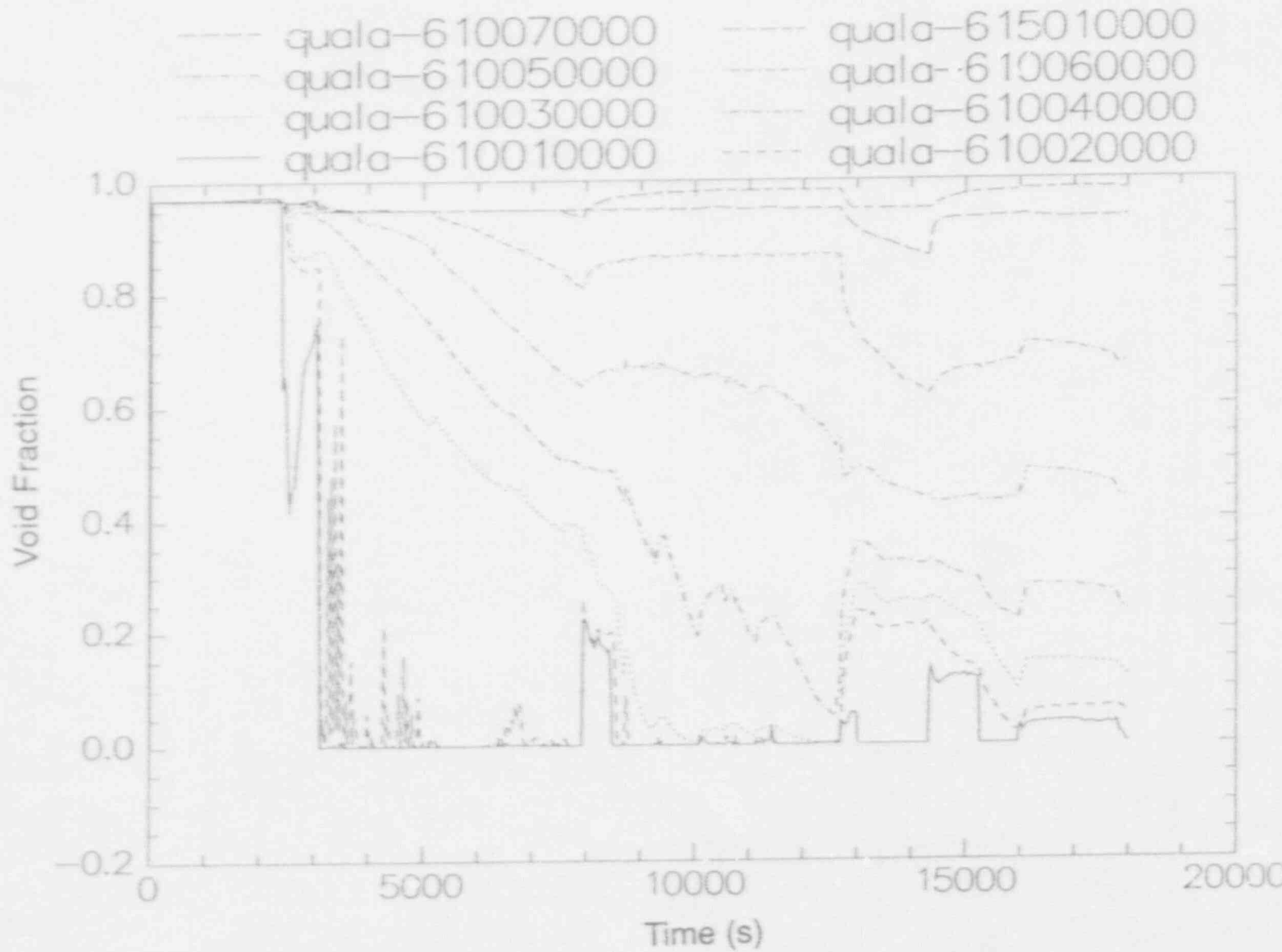

Figure 60. Noncondensible mass fraction in the pressurizer volumes—Oconee Case 1.

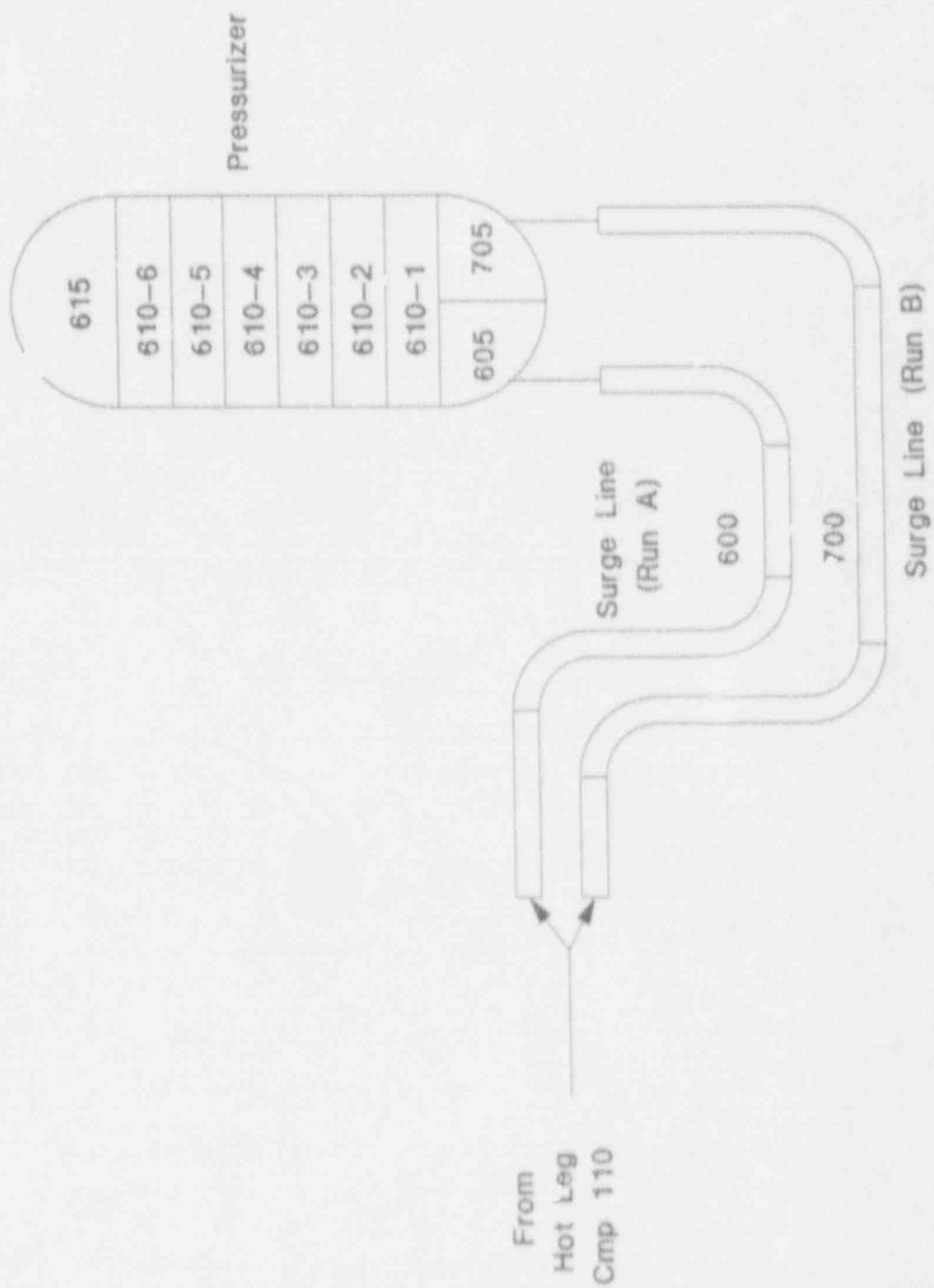

Figure 61. Dual surge line model for once through steam generator analysis.

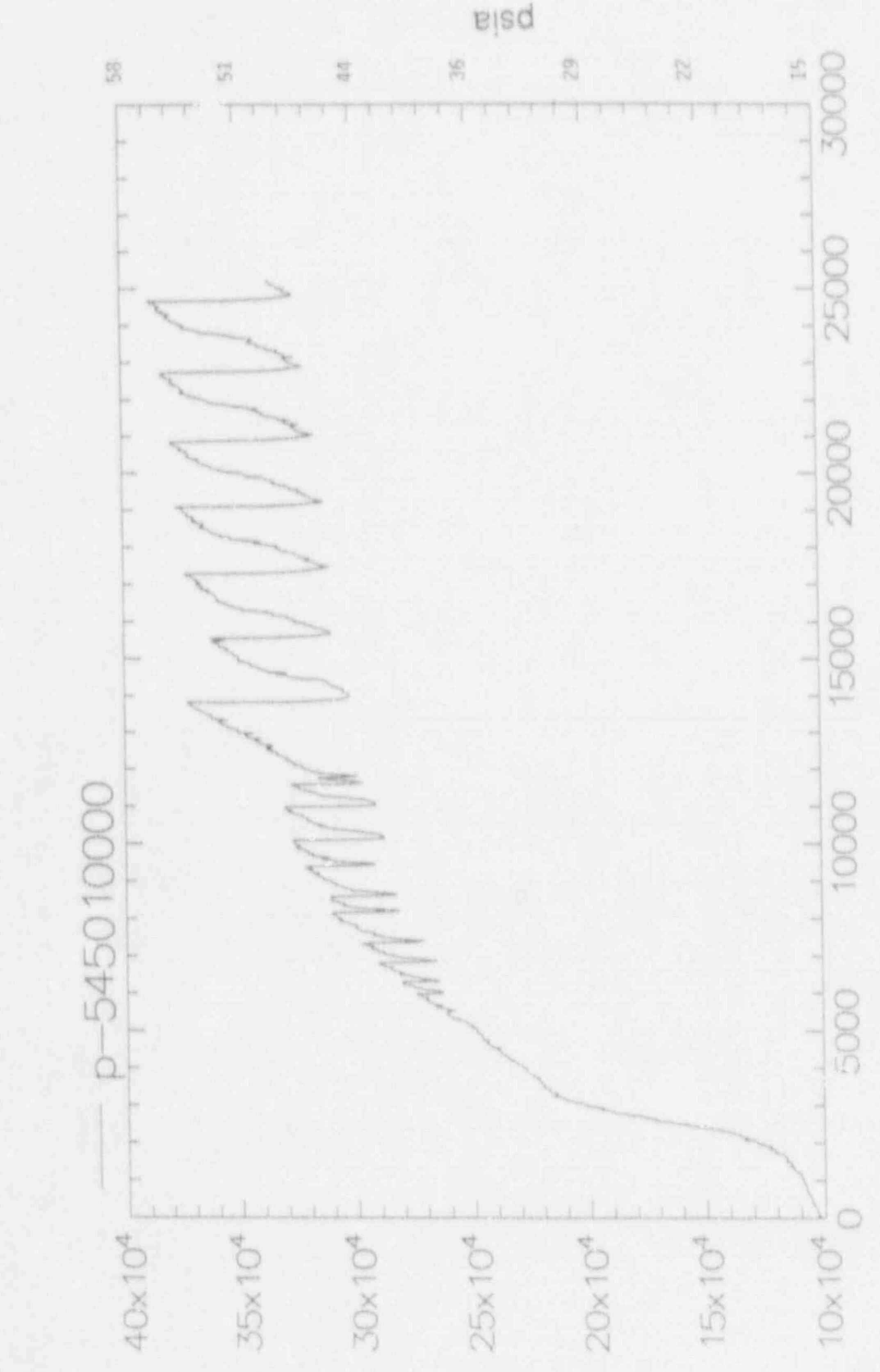

Figure 62. Primary system pressure for once through steam generator analysis—Oconee Case 2.

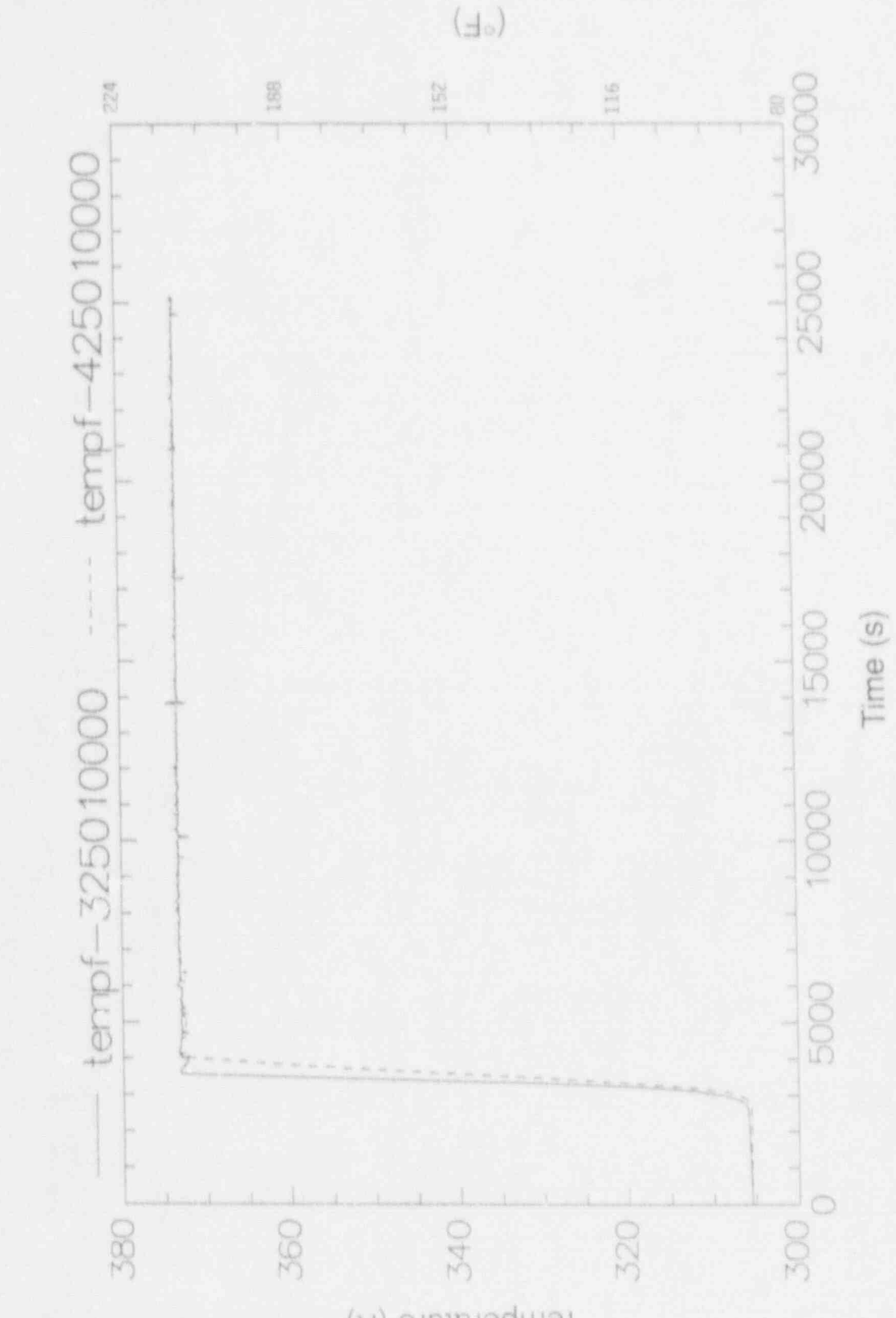

Figure 63. Water temperature in the topmost SG secondary volume—Oconee Case 2.

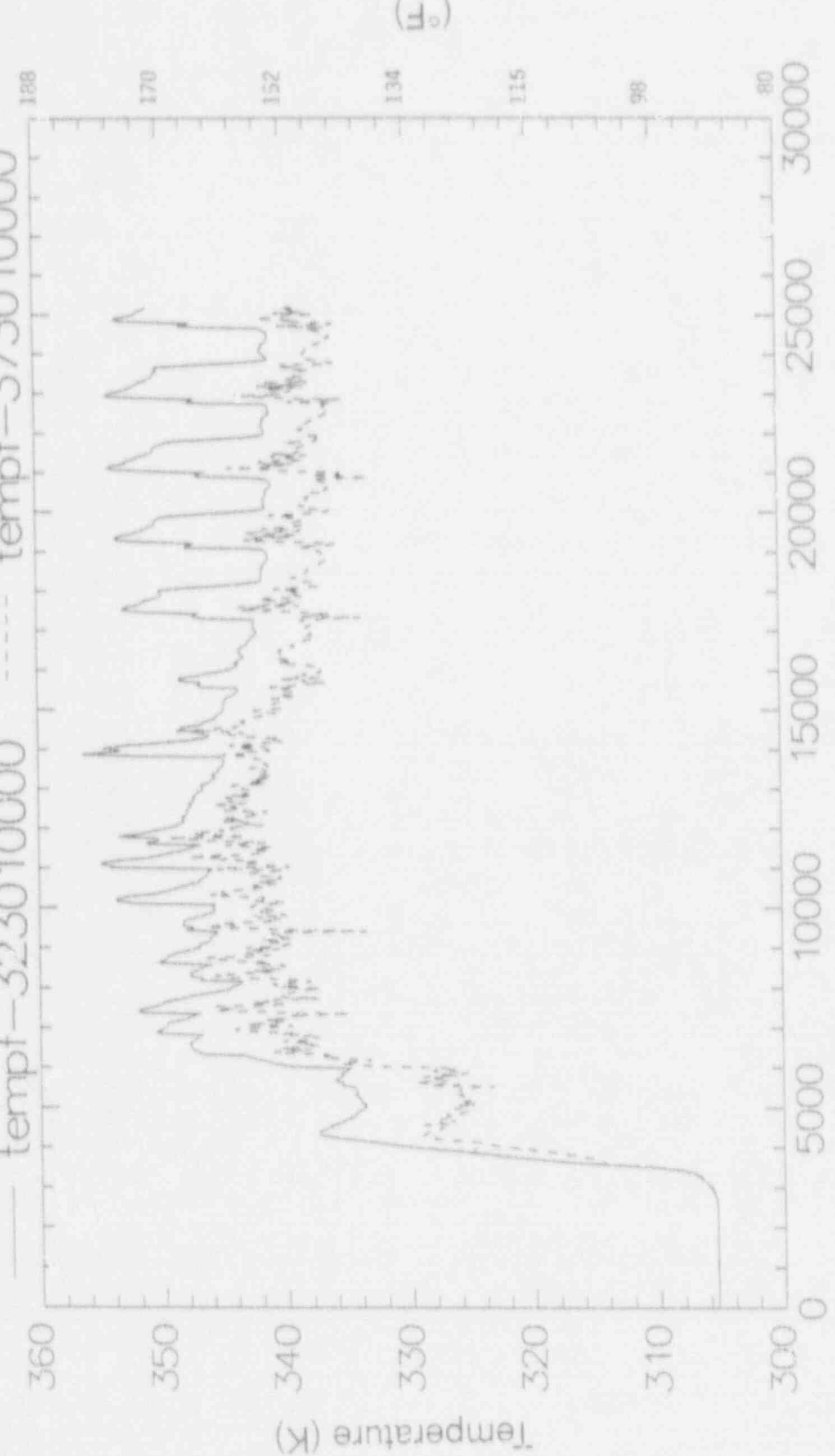

Figure 64. Water temperature in components 323 and 373 of Loop A SG secondary—Oconee Case 2.

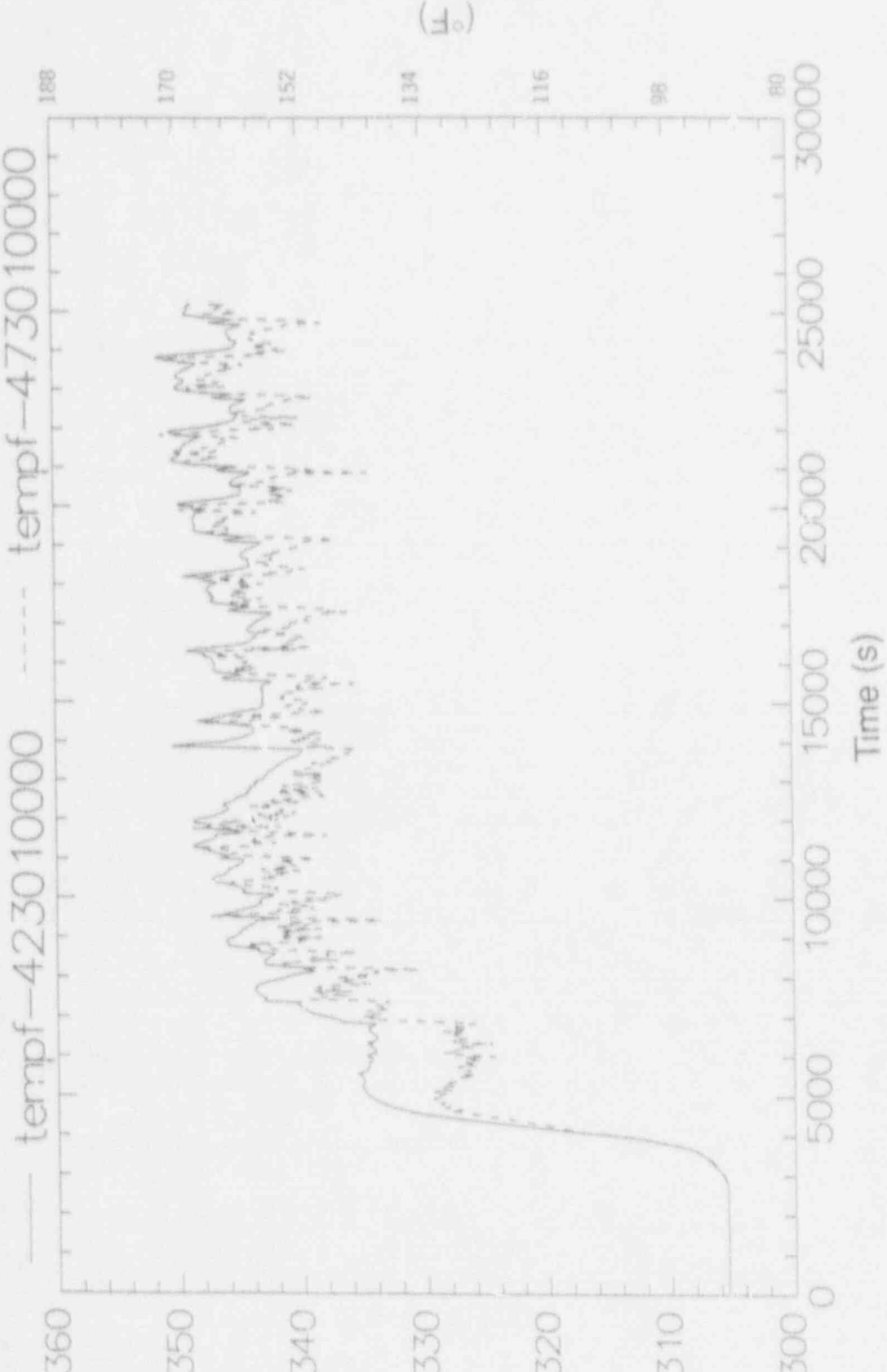

Figure 65. Water temperature in components 423 and 473 of Loop B SG secondary—Oconee Case 2.

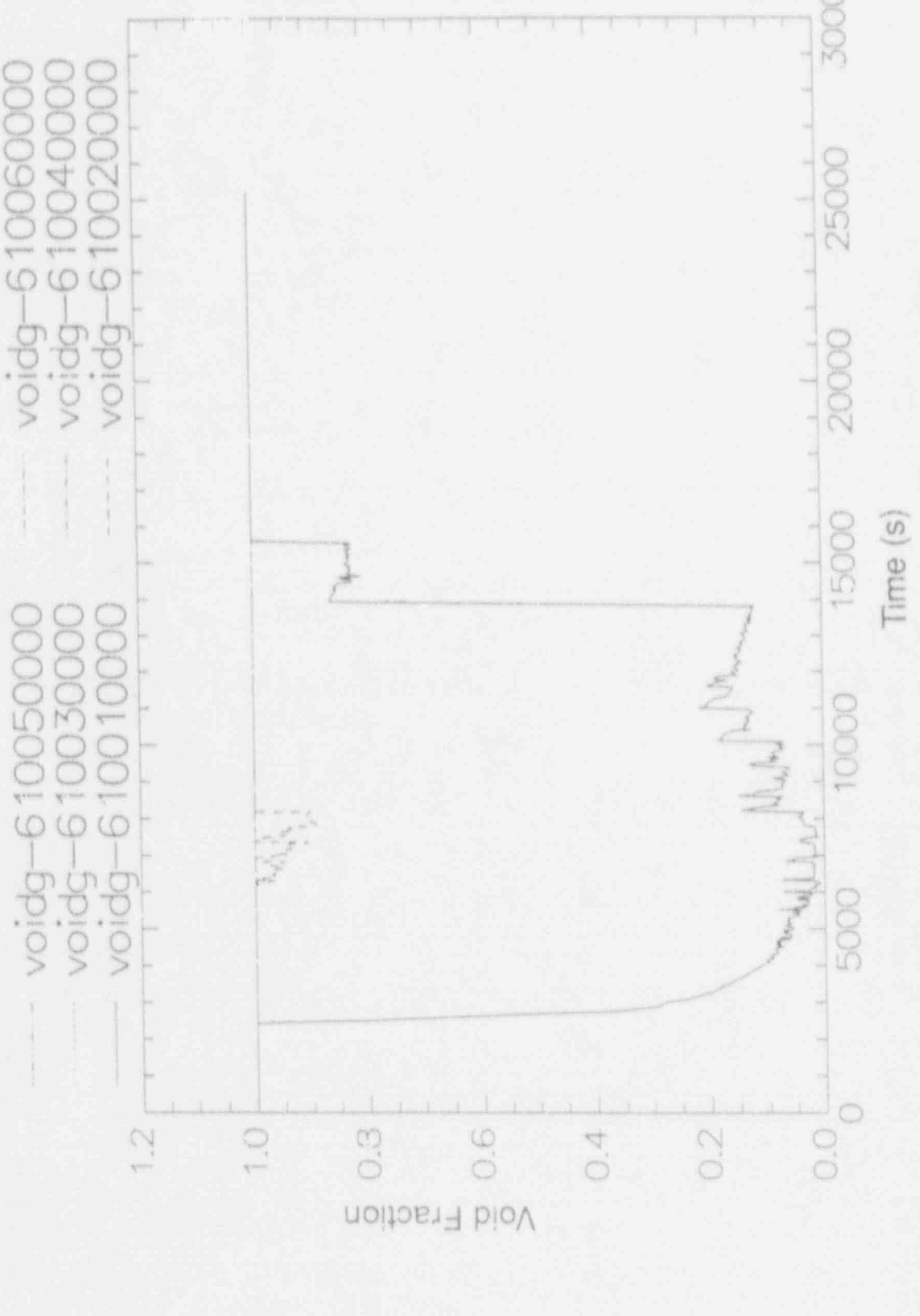

Figure 66. Vapor void fraction in the pressurizer volumes—Oconee Case 2.

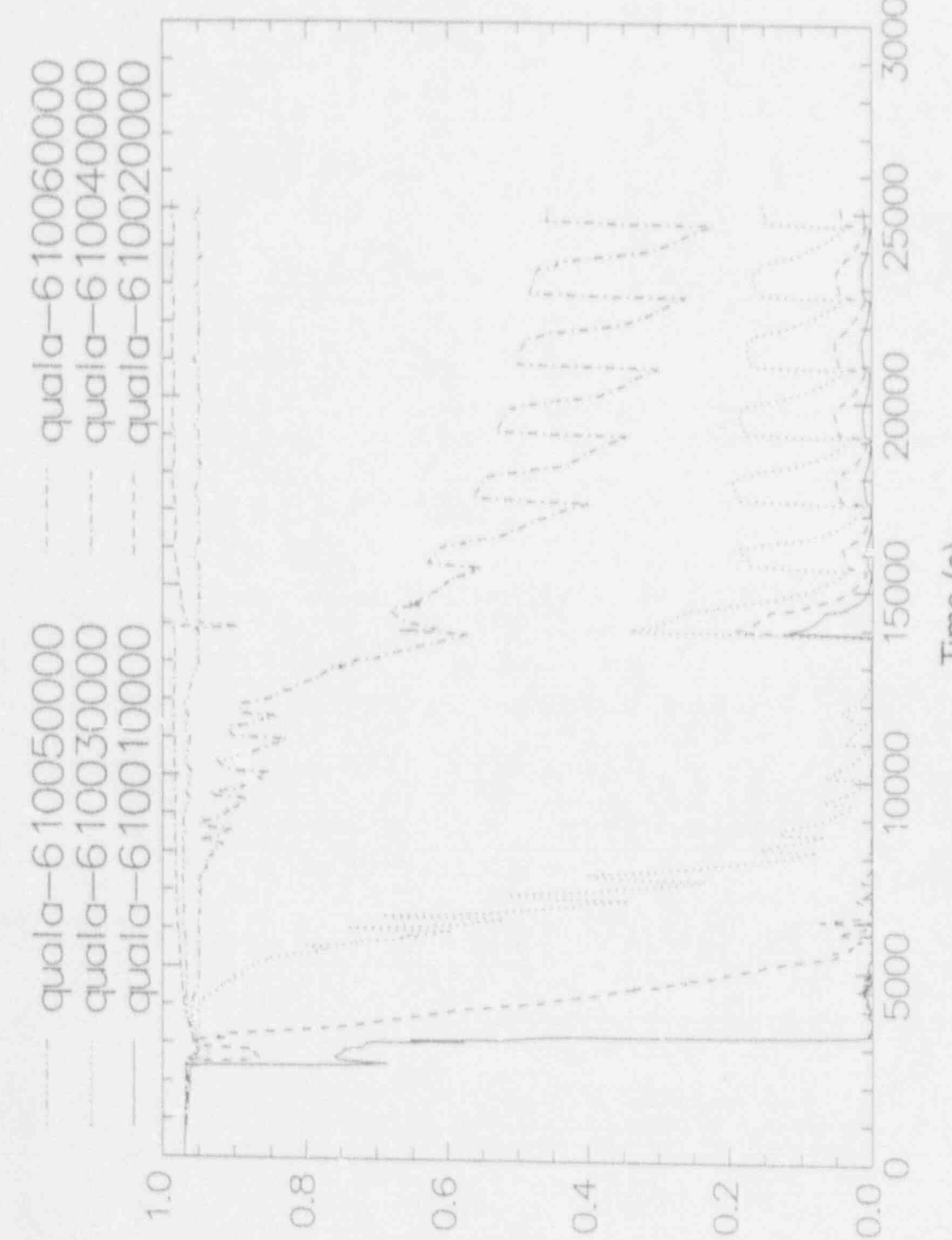

Figure 67. Noncondensible mass fraction in the pressurizer volumes—Oconee Case 2

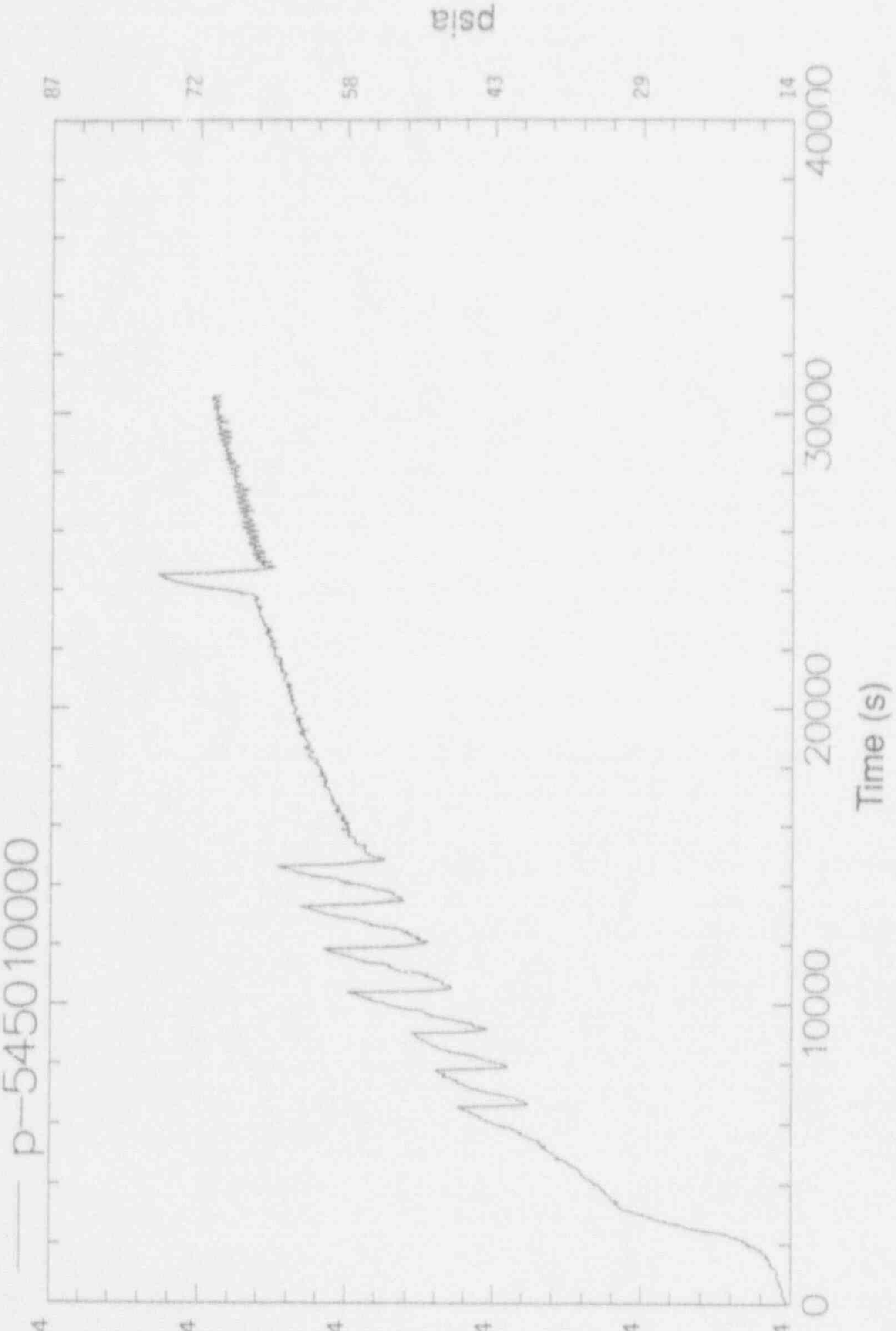

Figure 68. Primary system pressure for once-through steam generator analysis Oconee Case 3.

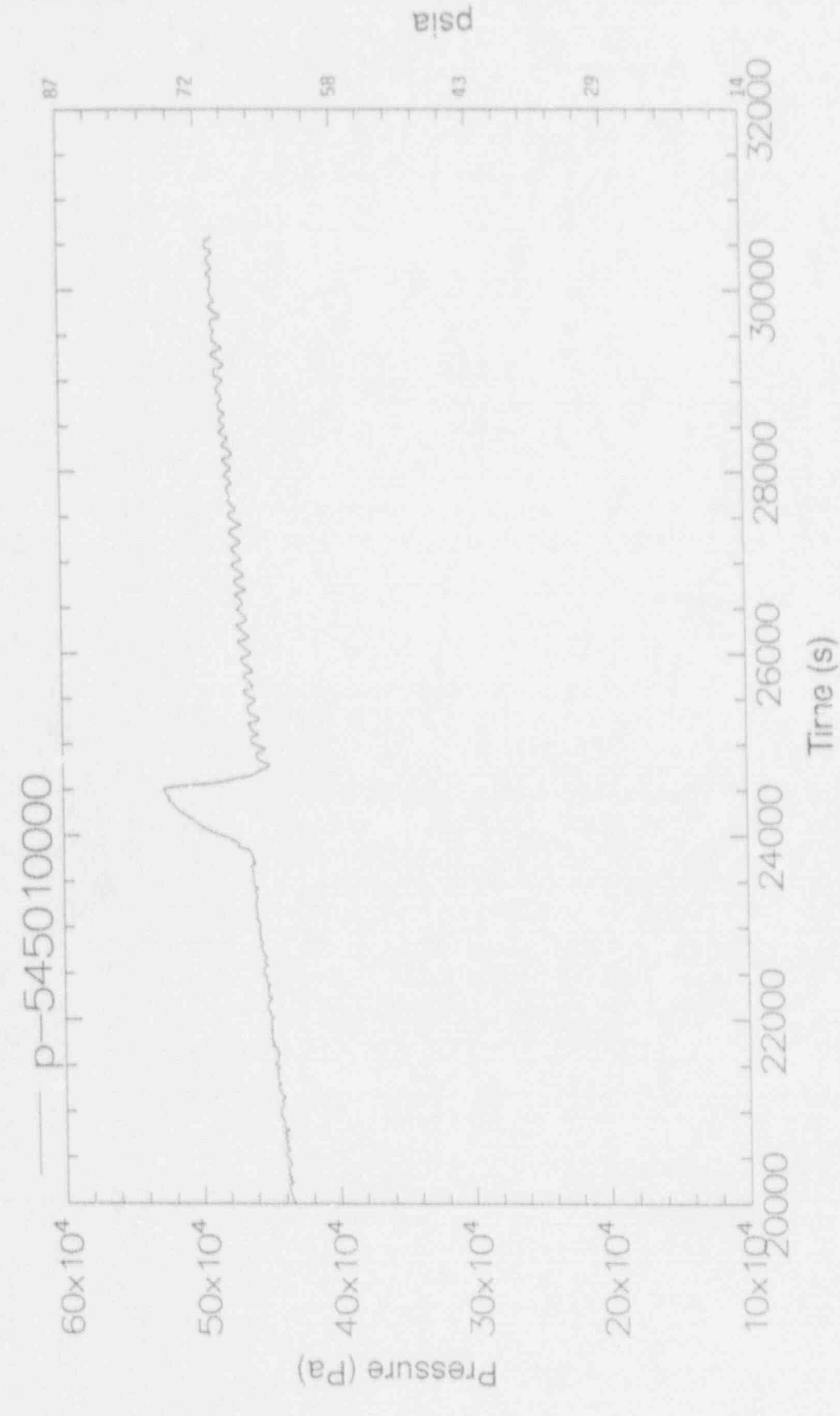

Figure 69. Primary system pressure for once-through steam generator analysis (expanded scale)—Oconee Case 3.

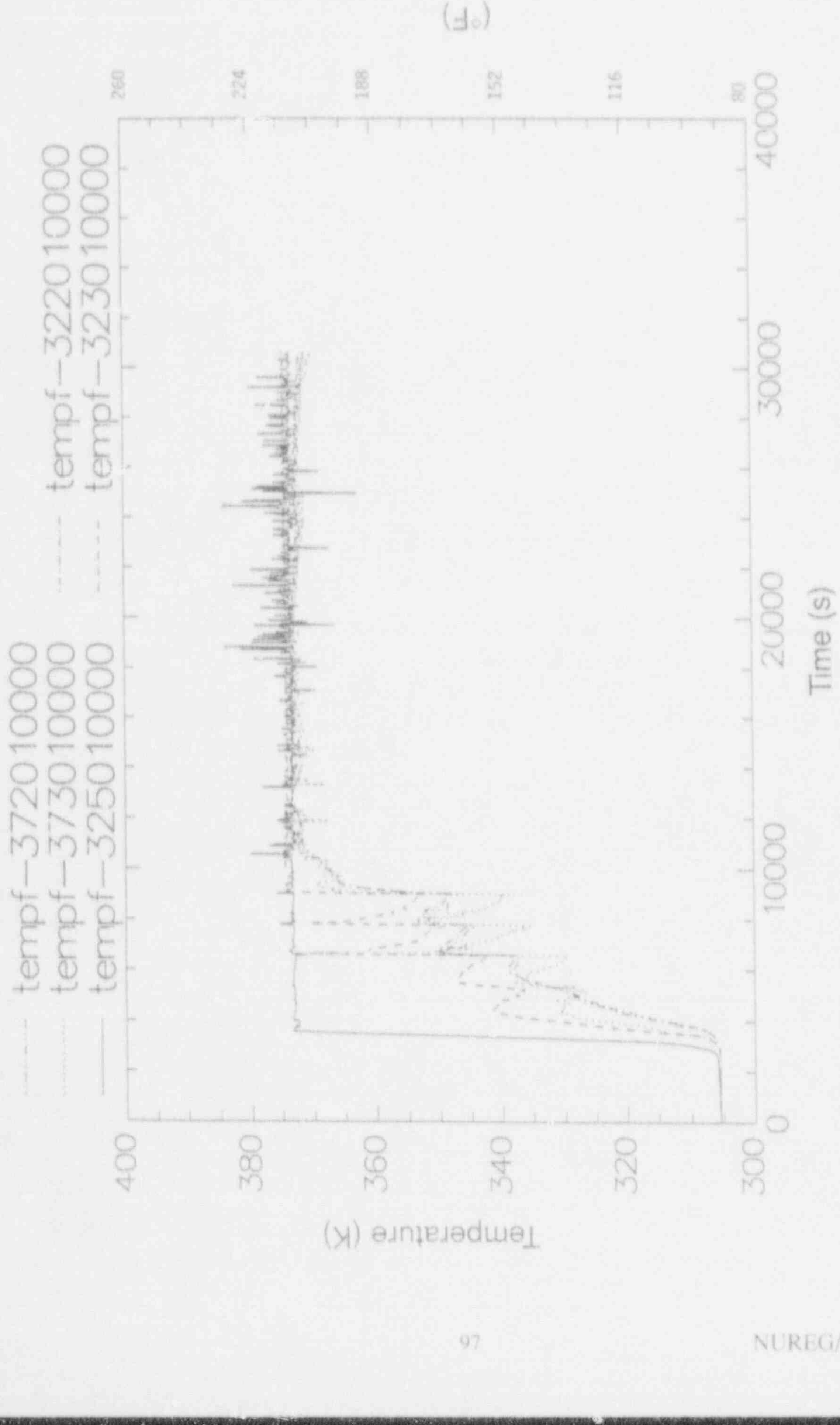

Figure 70. Water temperature in the upper portion of Loop A SG secondary—Oconee Case 1.

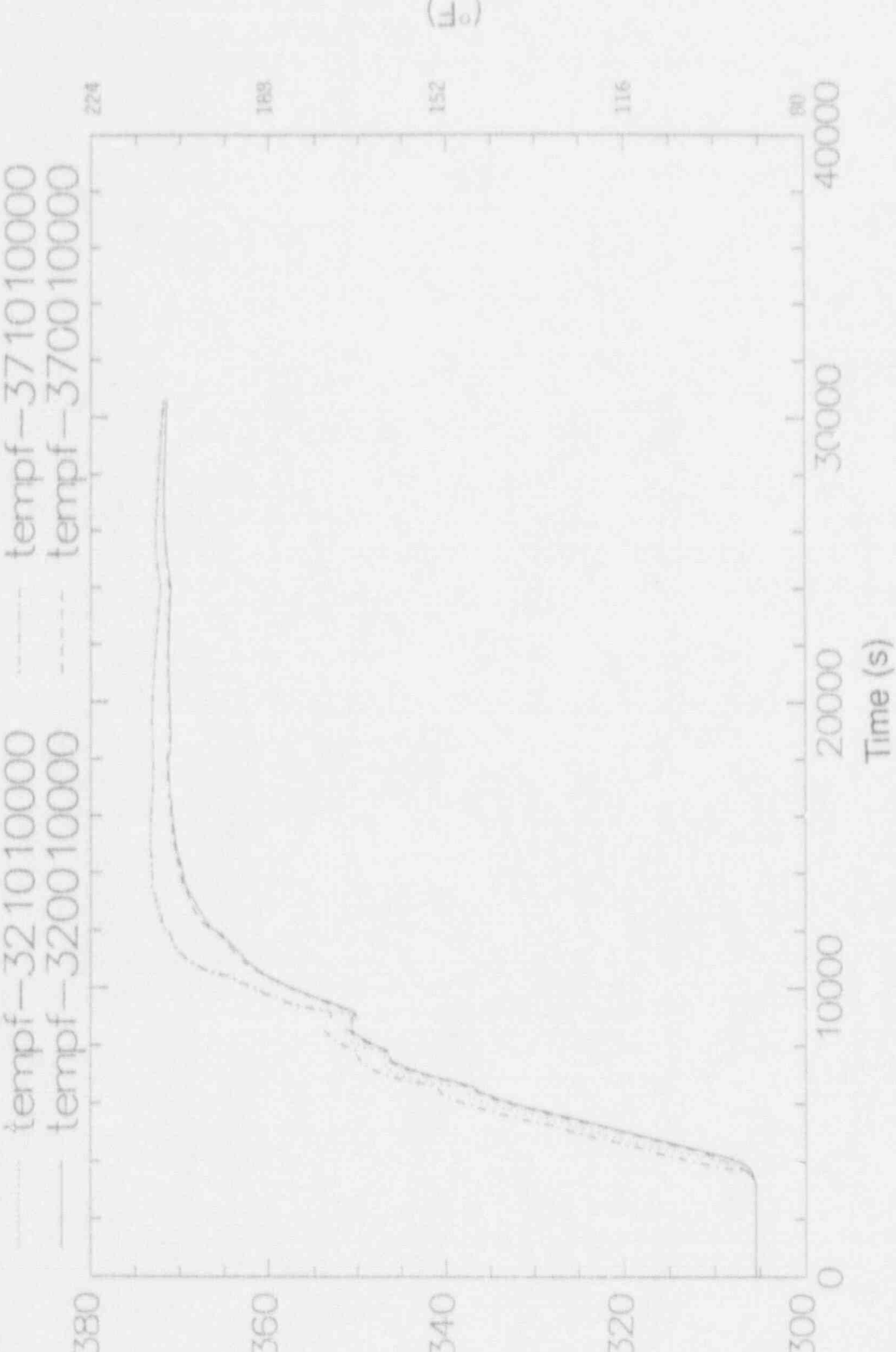

Figure 71. Water temperature in components 320, 321, 370, and 371 of Loop A SG secondary—Oconee Case 3.

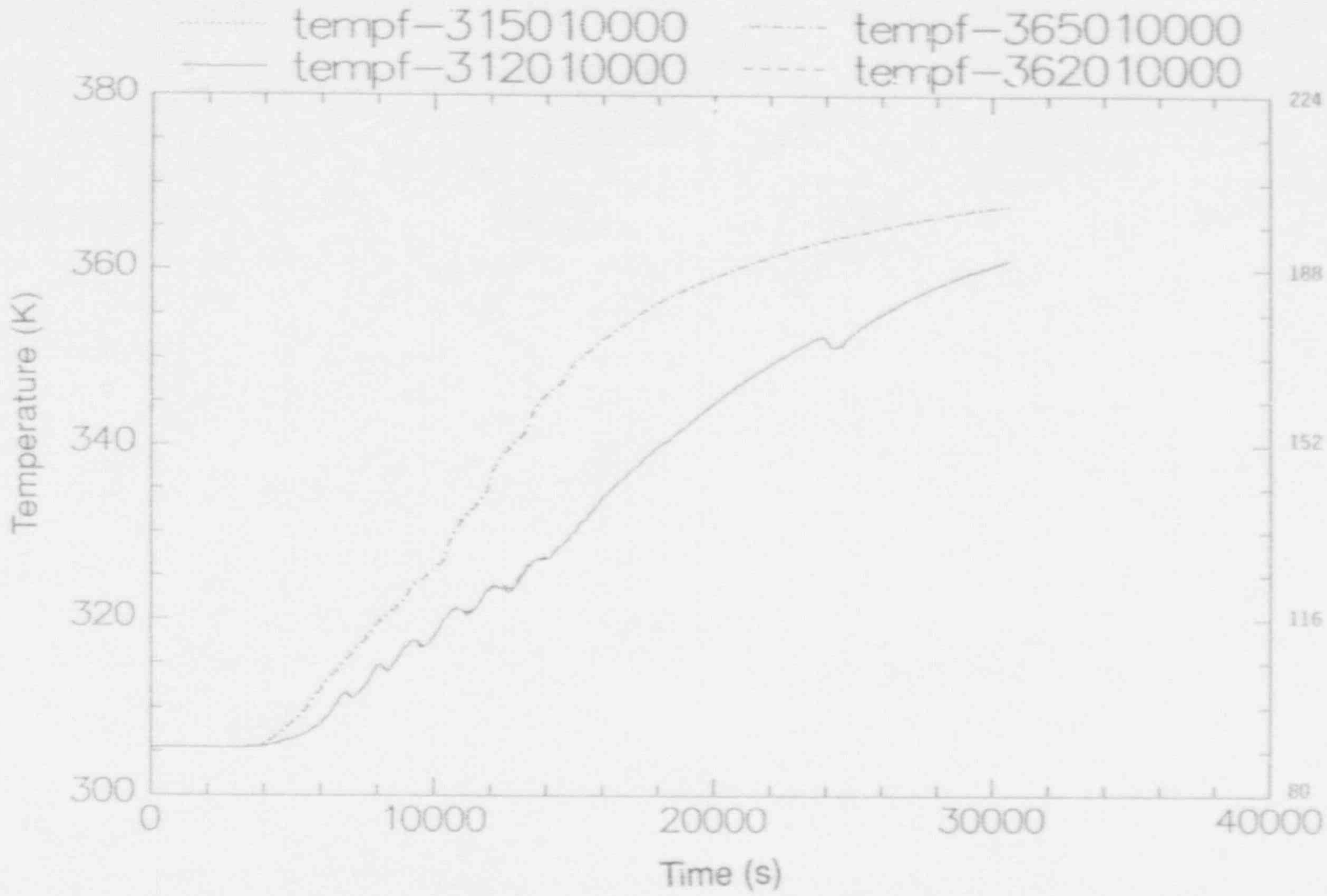

Figure 72. Water temperature in components 312, 315, 362, and 365 of Loop A SG secondary—Oconee Case 3.

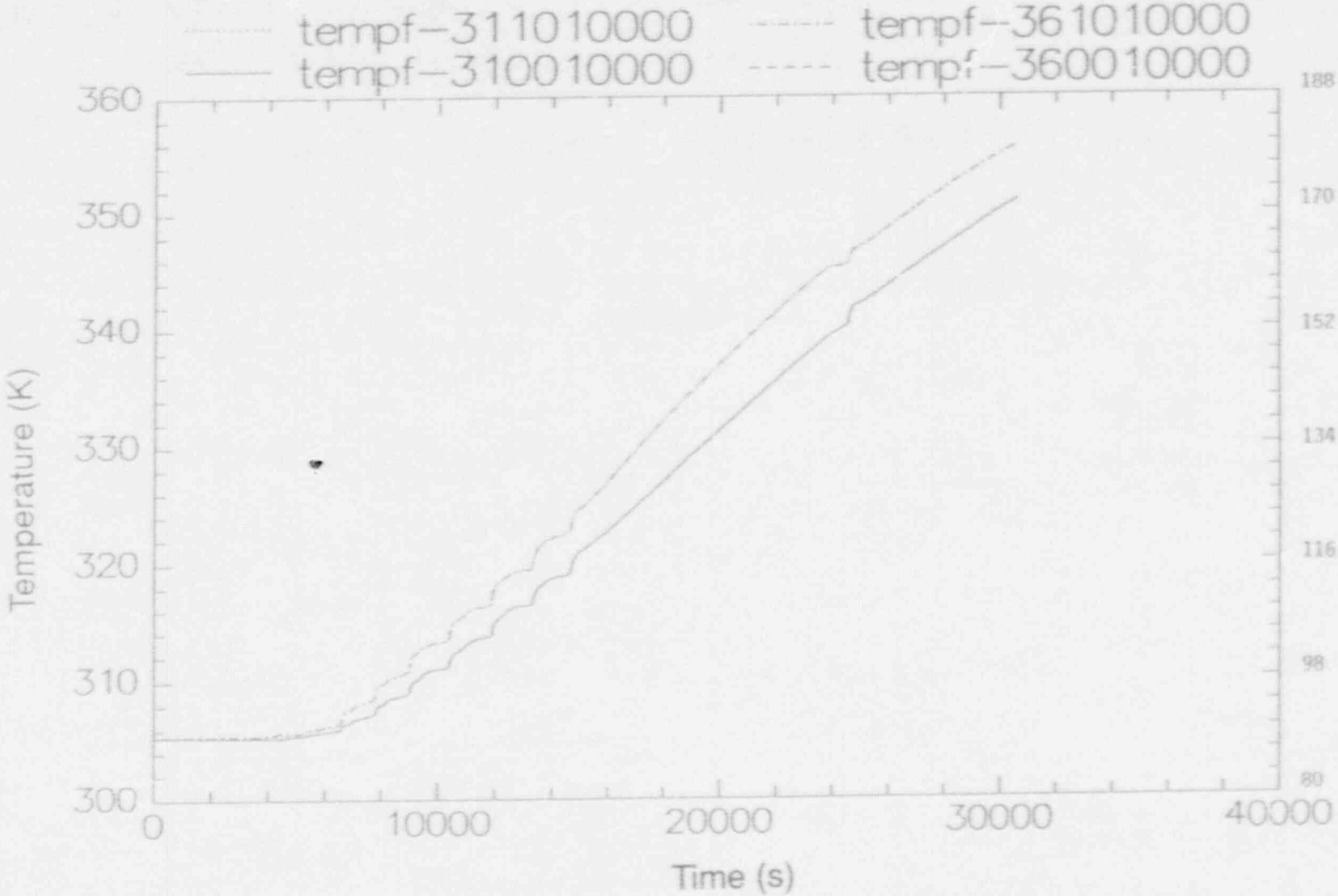

Figure 73. Water temperature in components 310, 311, 360, and 361 of Loop A SG secondary—Oconee Case 3.

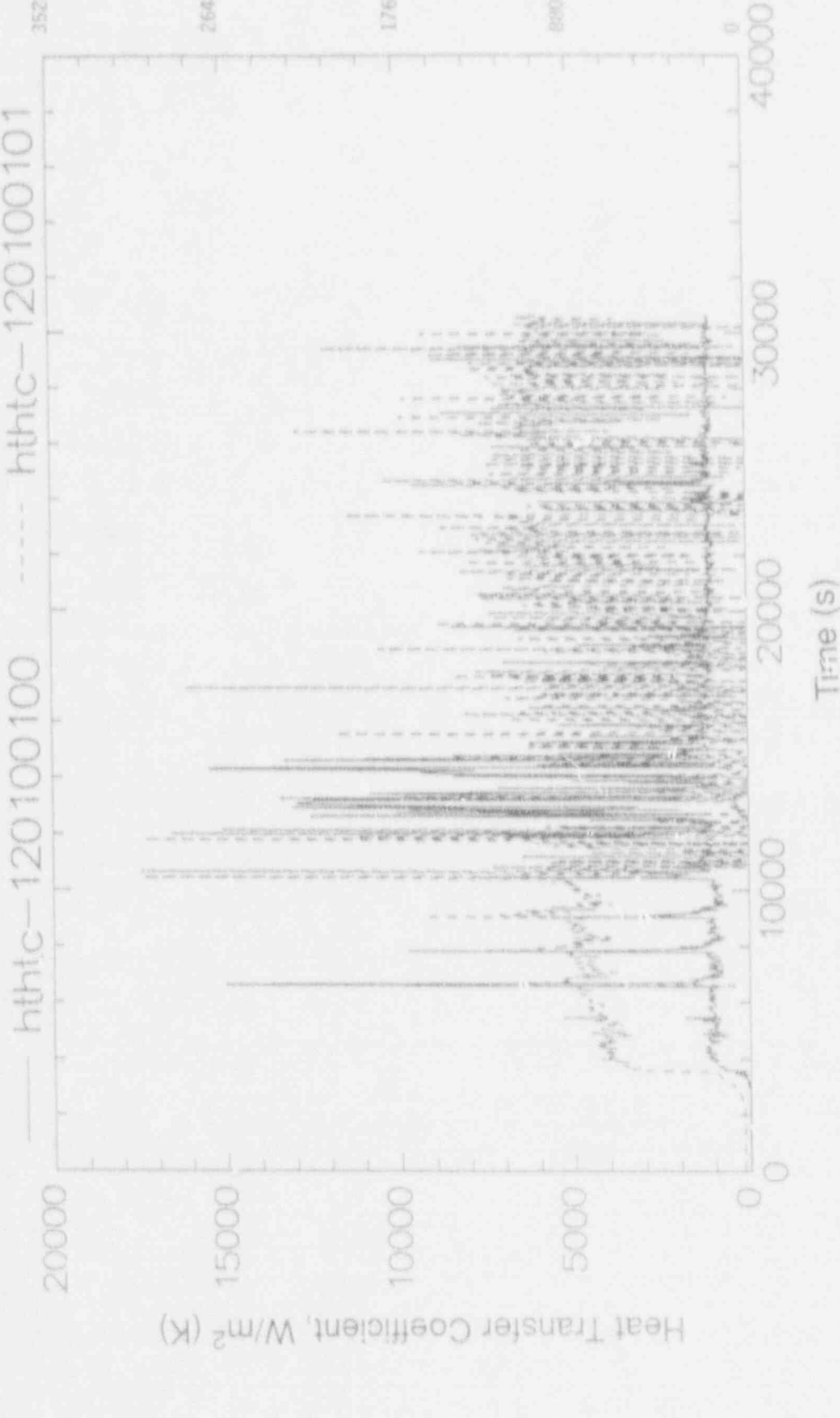

Figure 74. Heat transfer coefficient in volume 1 of Loop A SG primary tubes—Oconee Case 3.

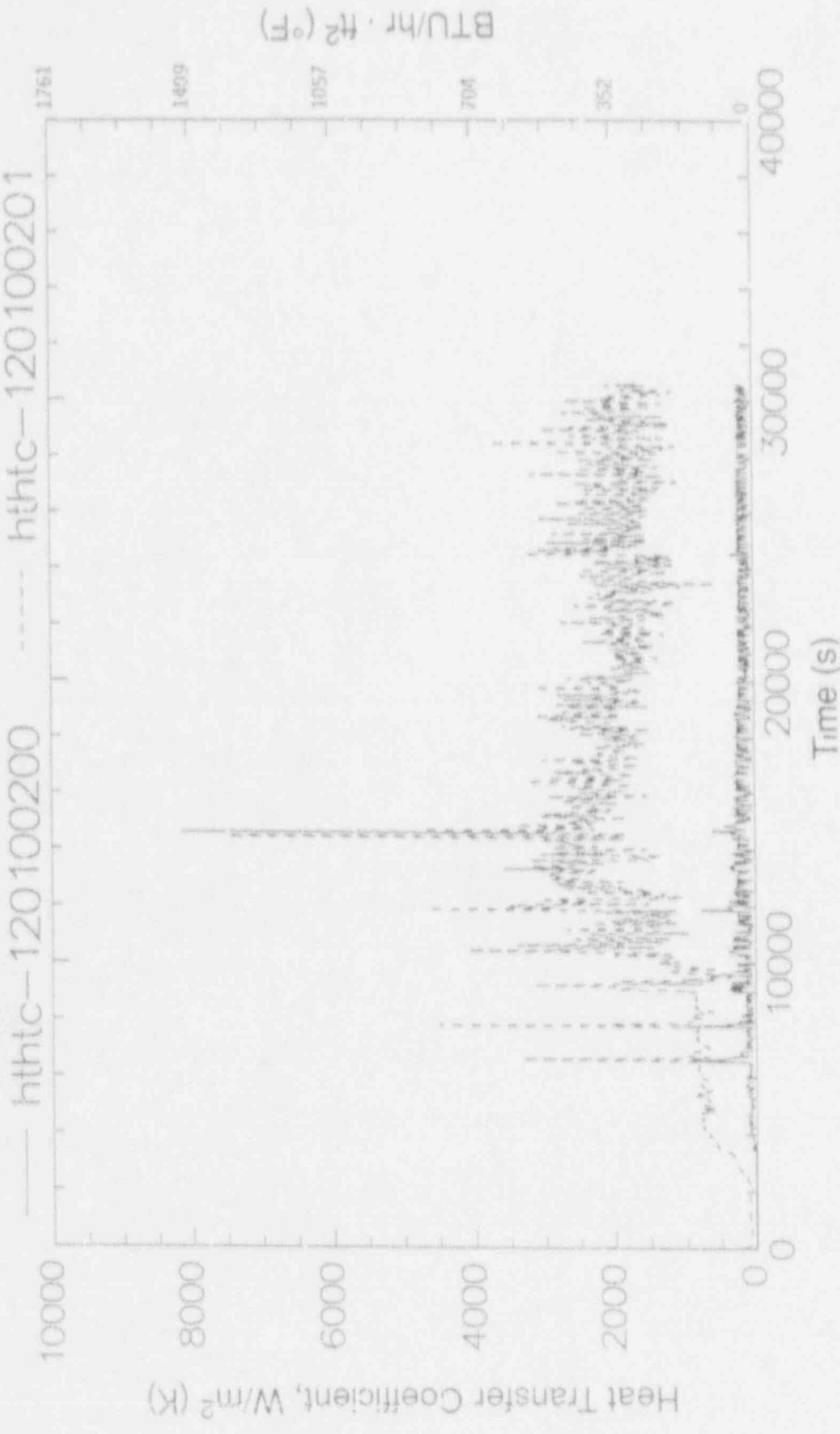

Figure 75. Heat transfer coefficient in volume 2 of Loop A SG primary tubes—Oconee Case 3.

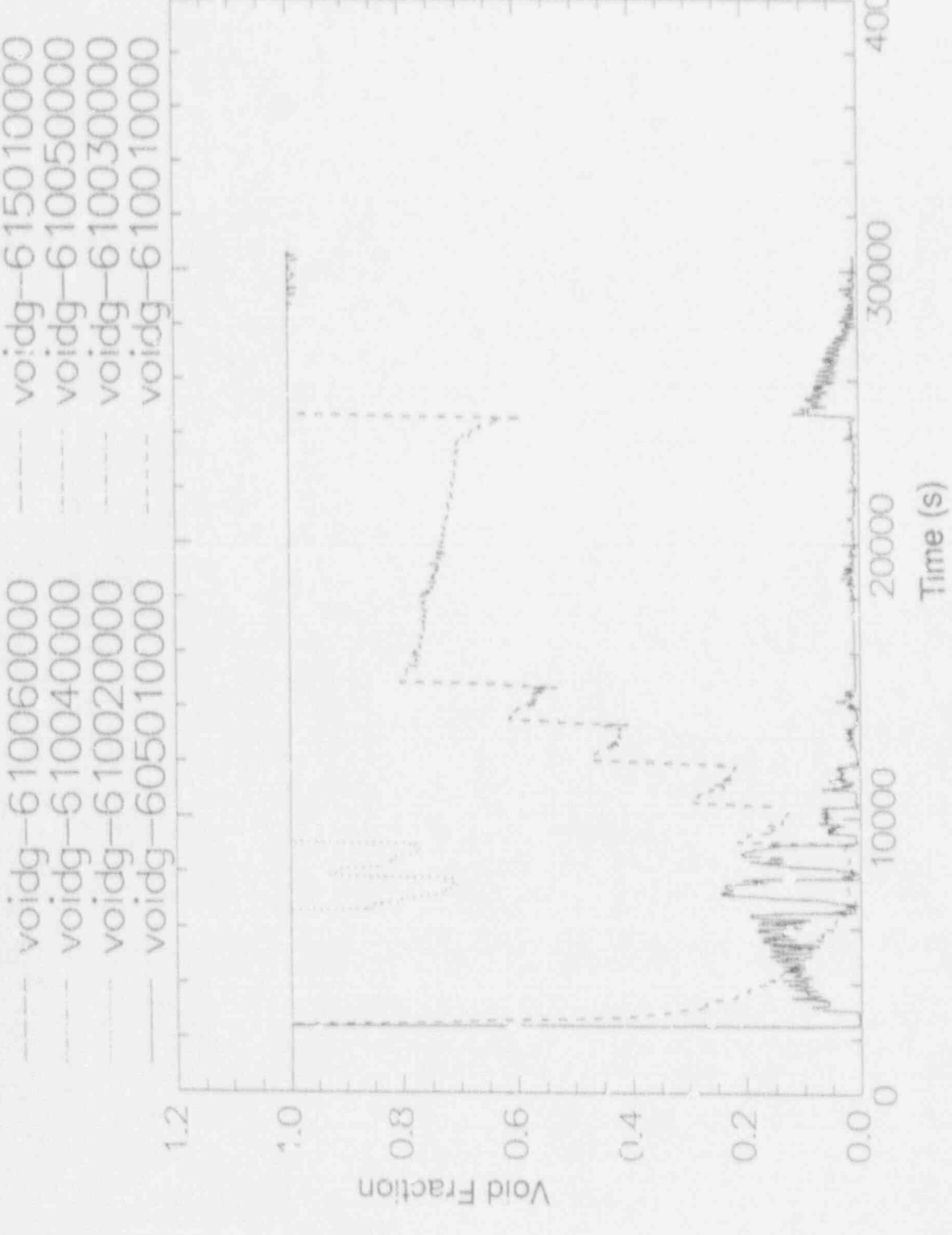

Figure 76. Vapor void fraction in the pressurizer volumes—Oconee Case 3.

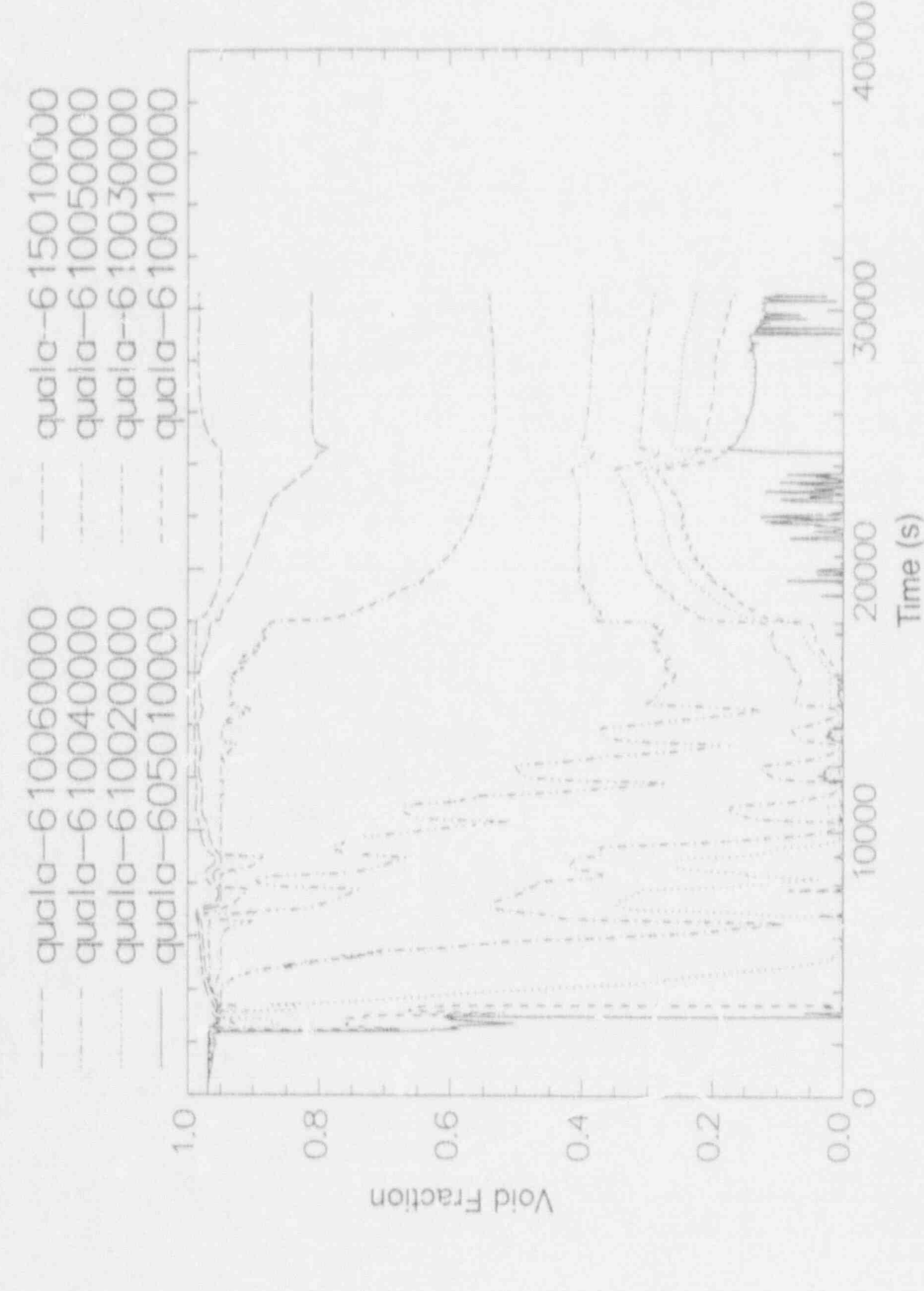

Figure 77. Noncondensible mass fraction in the pressurizer volumes—Oconee Case 3.

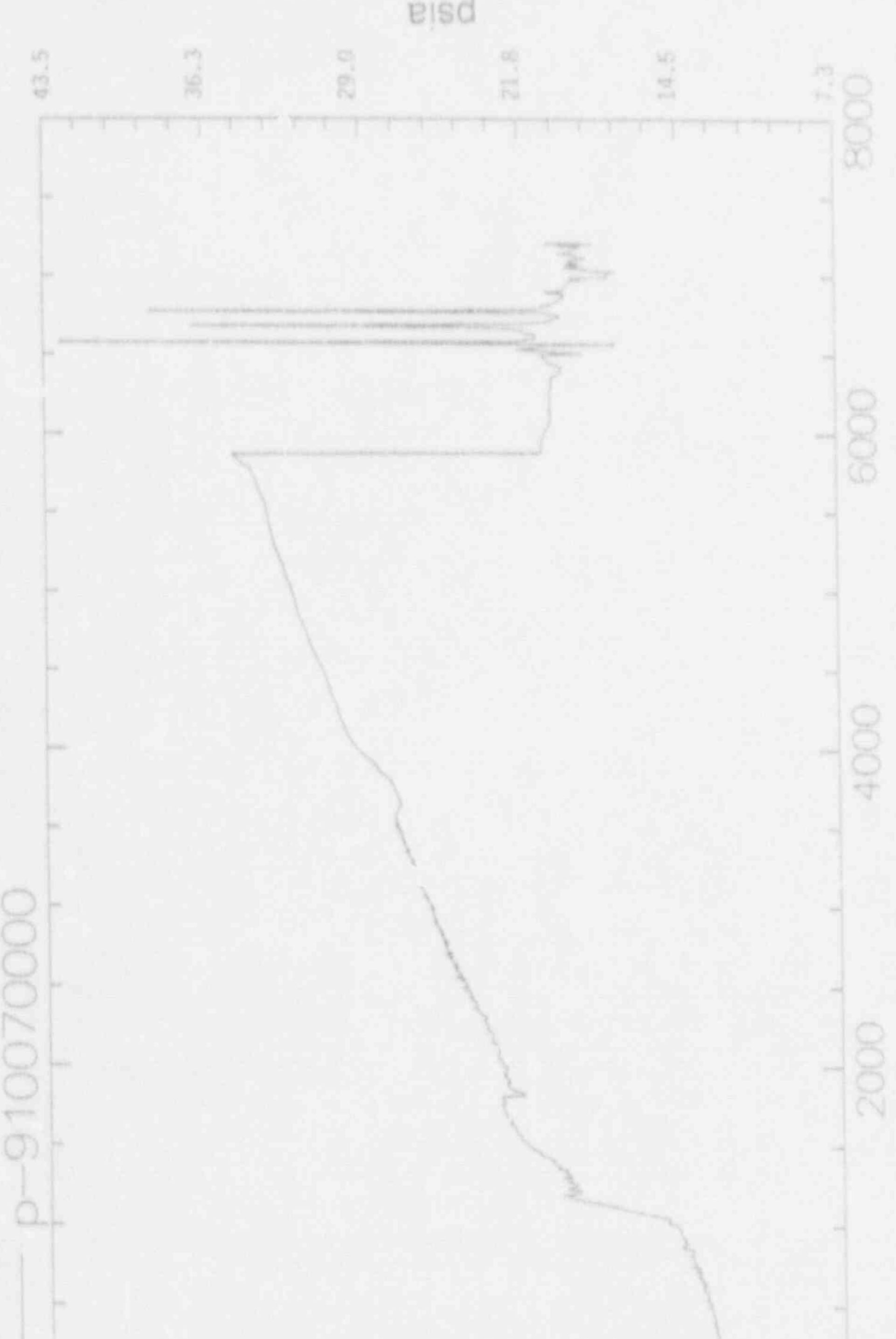

Figure 78. Primary system pressure for thimble seal failure analysis.

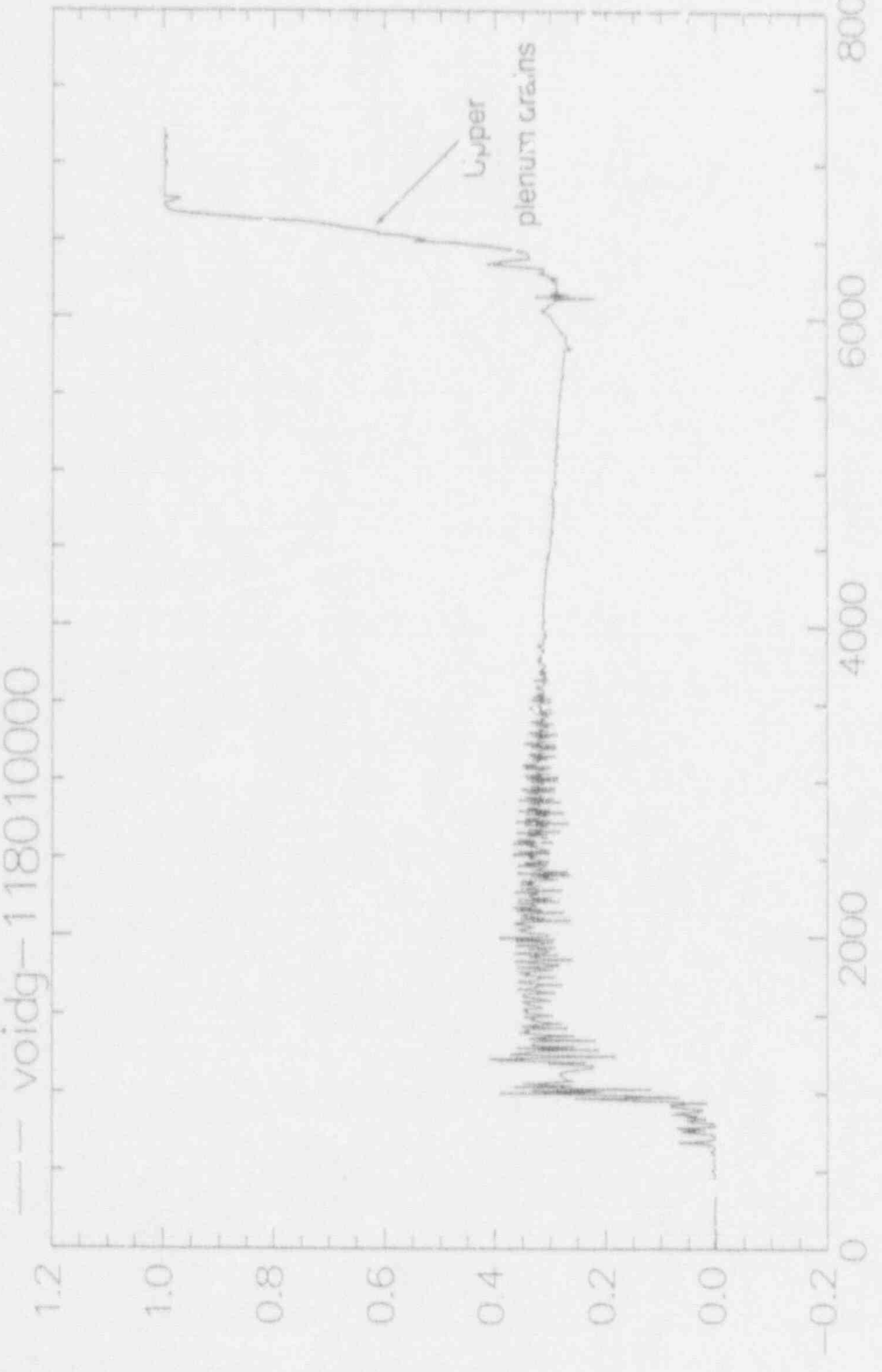

Figure 79. Void fraction for upper plenum volume just above core for thimble seal failure analysis.

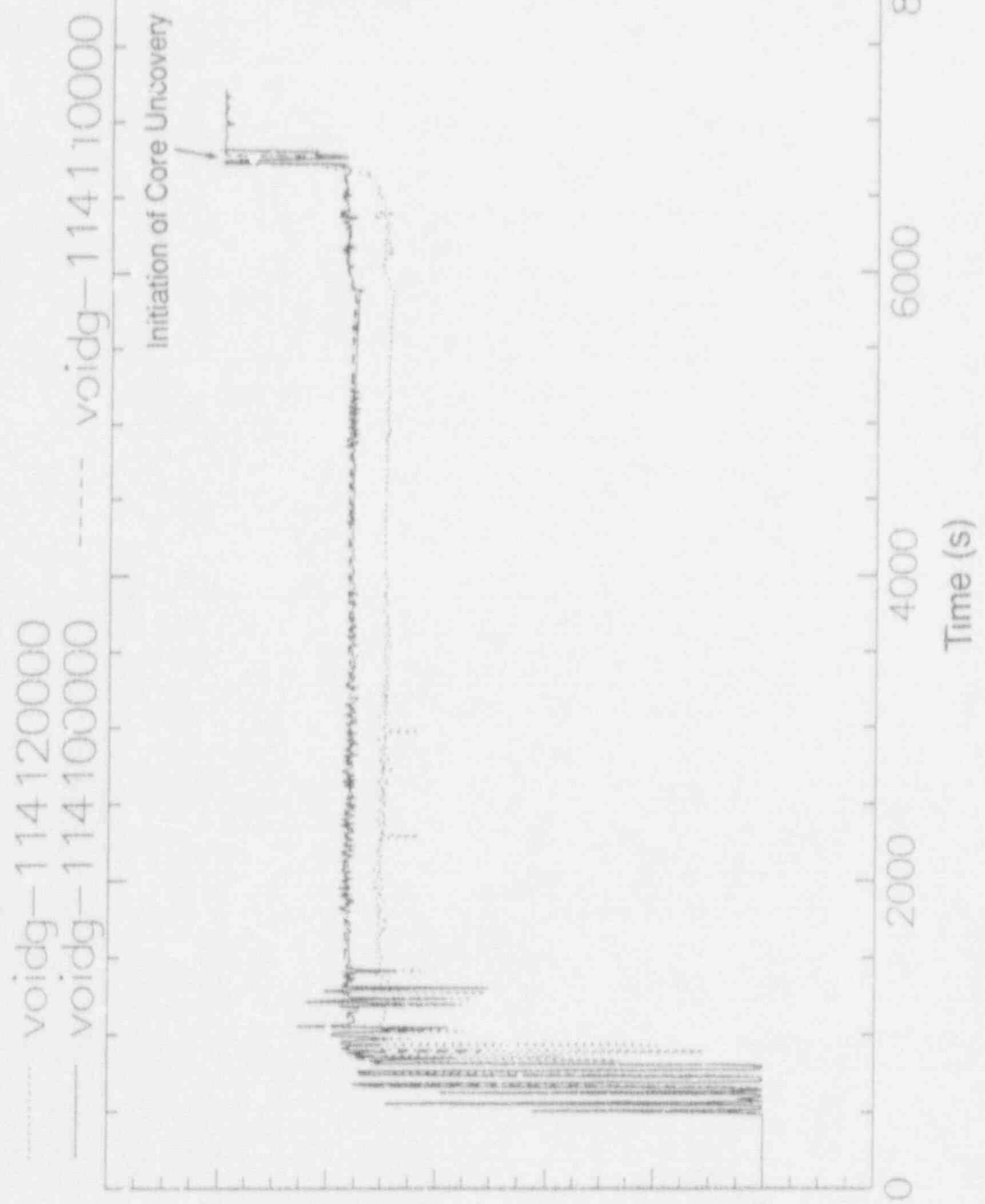

Figure 80. Void fraction for top three core volumes for thimble seal failure analysis.

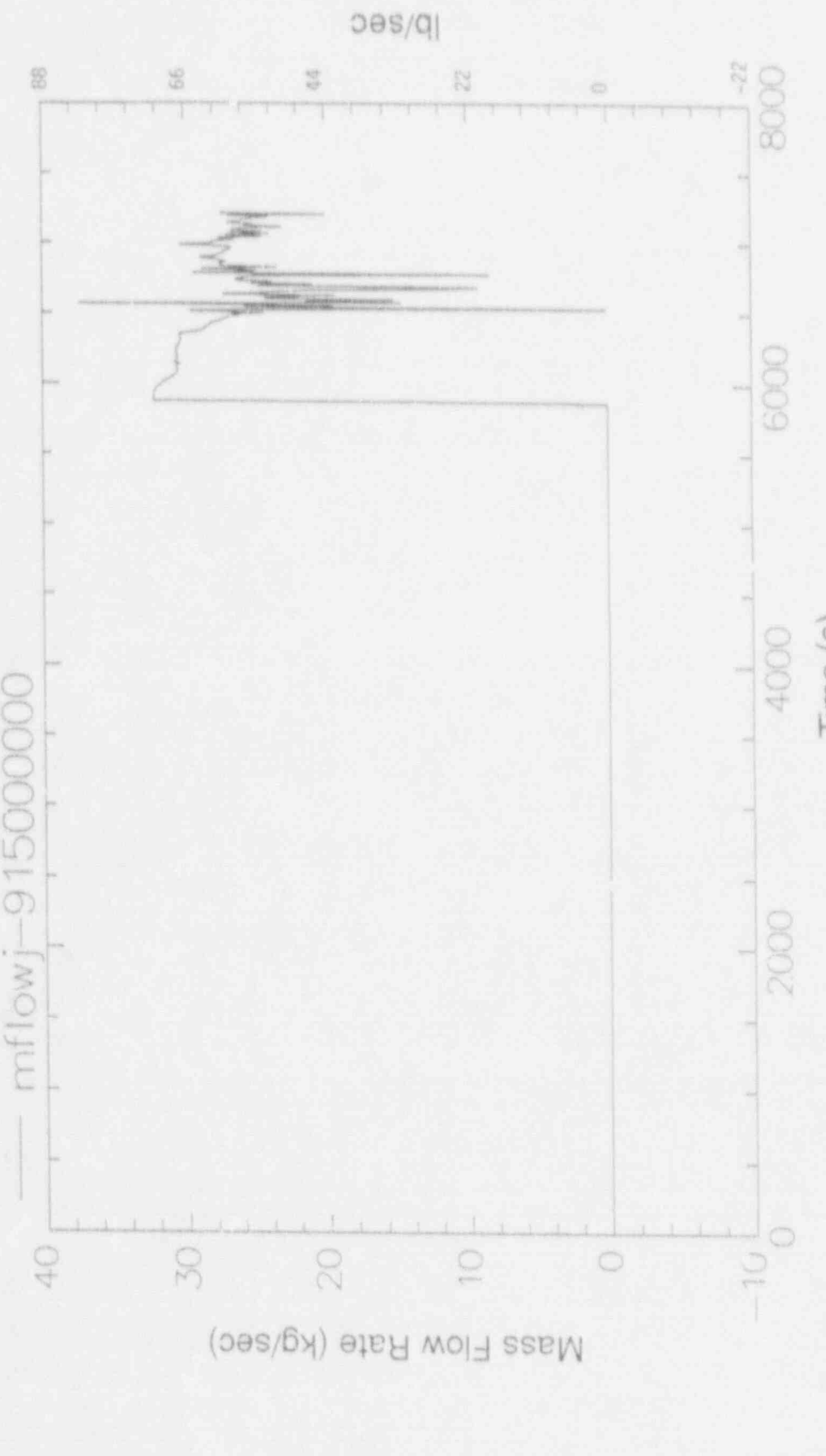

Figure 81. Break mass flow rate—thimble seal failure analysis.

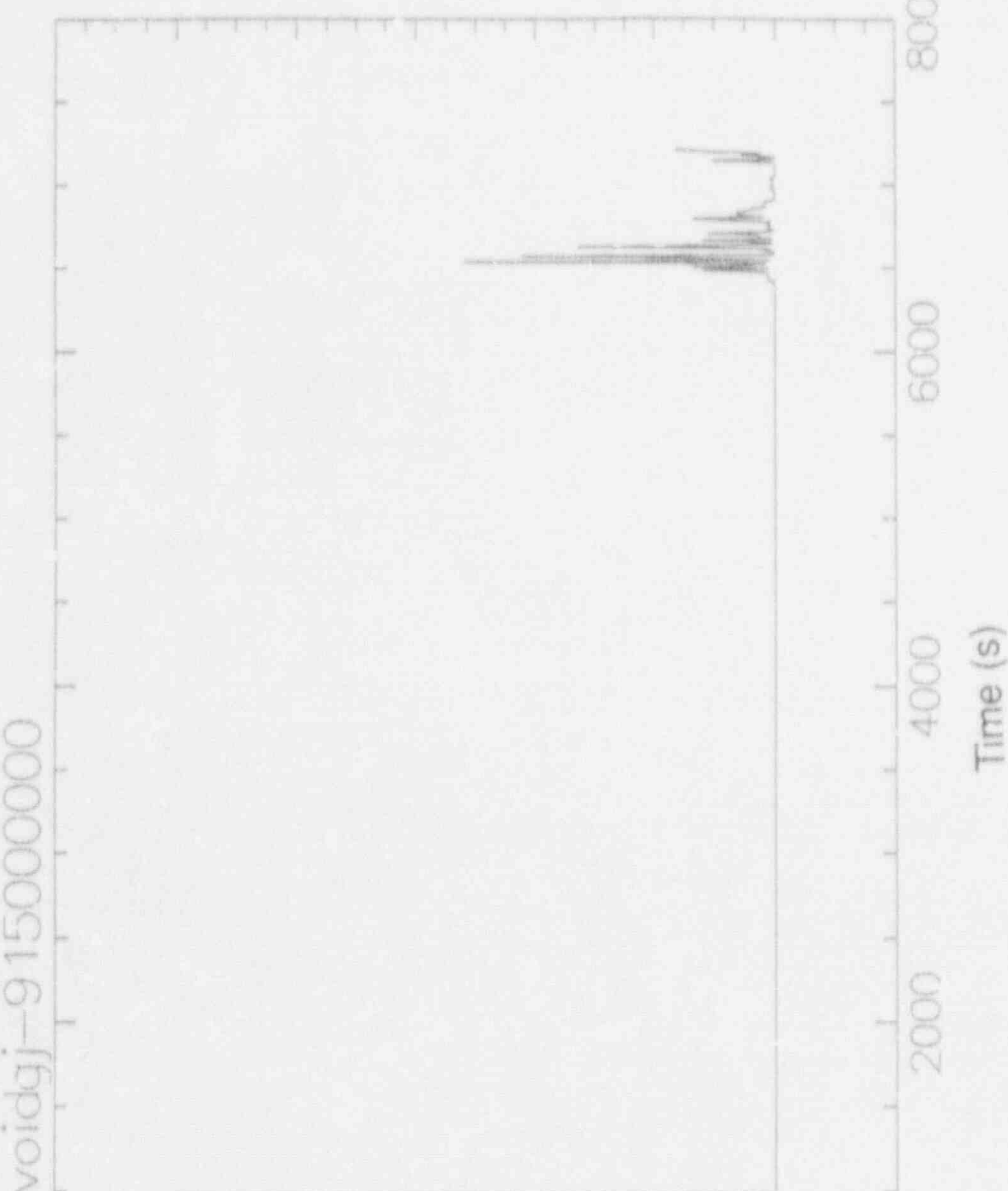

Figure 82. Break flow void fraction—thimble seal failure analysis.

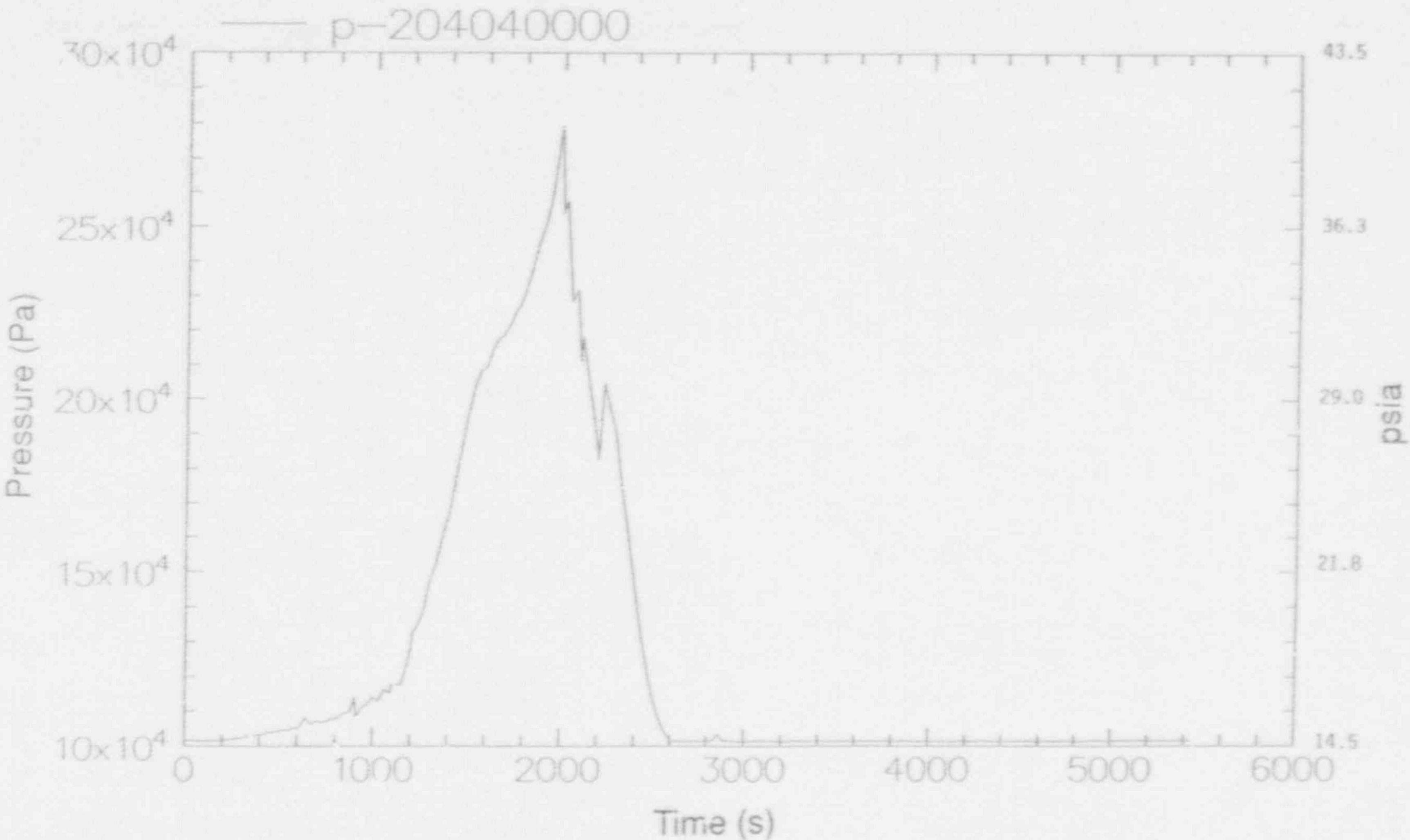

Figure 83. Primary system pressure results for nozzle dam failure analysis—Case 1.

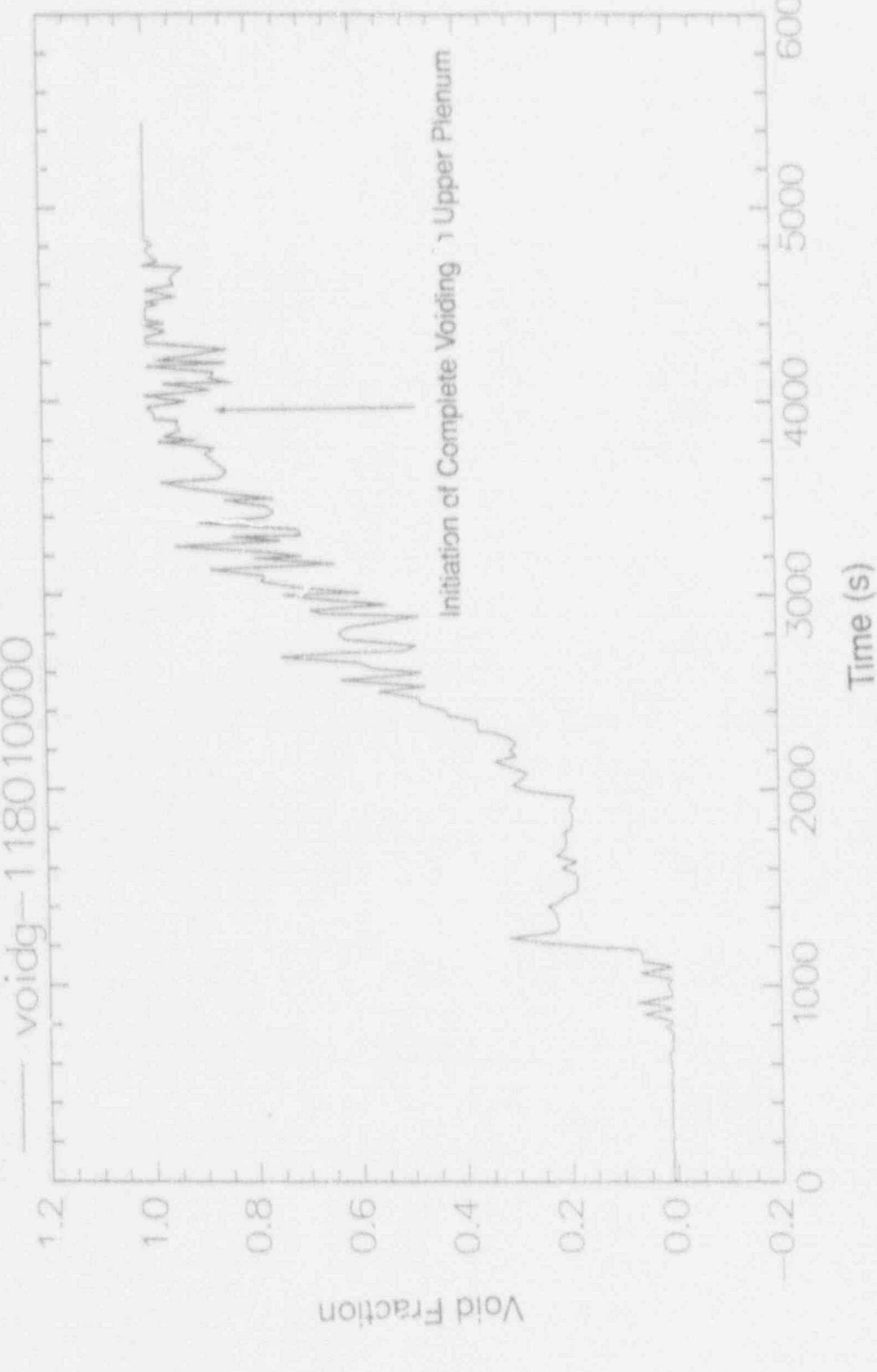

Figure 84. Vapor void fraction in the upper plenum volume just above core for nozzle dam failure analysis—Case 1.

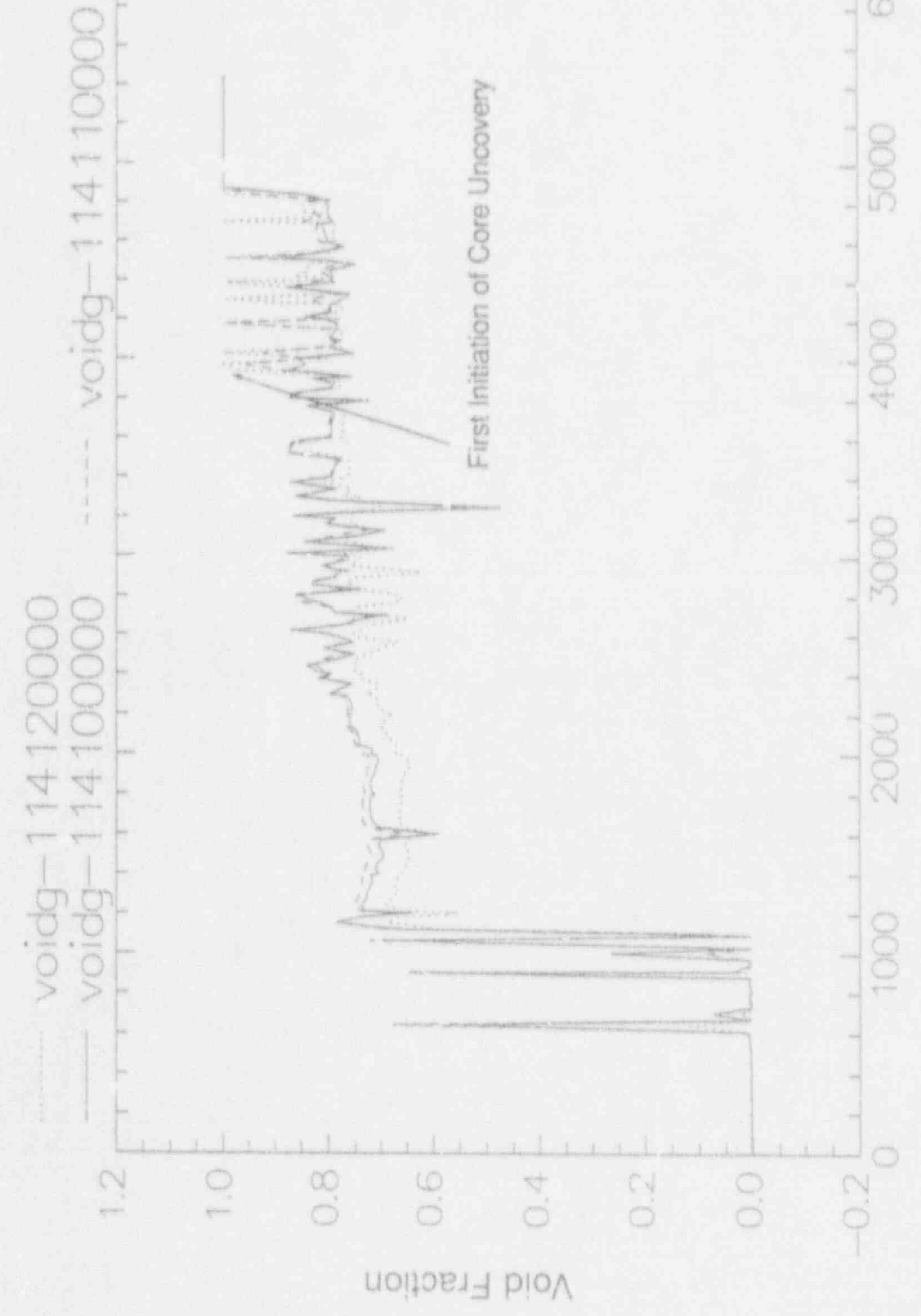

Figure 85. Vapor void fraction for the top three core volumes for the nozzle dam failure analysis—Case 1.

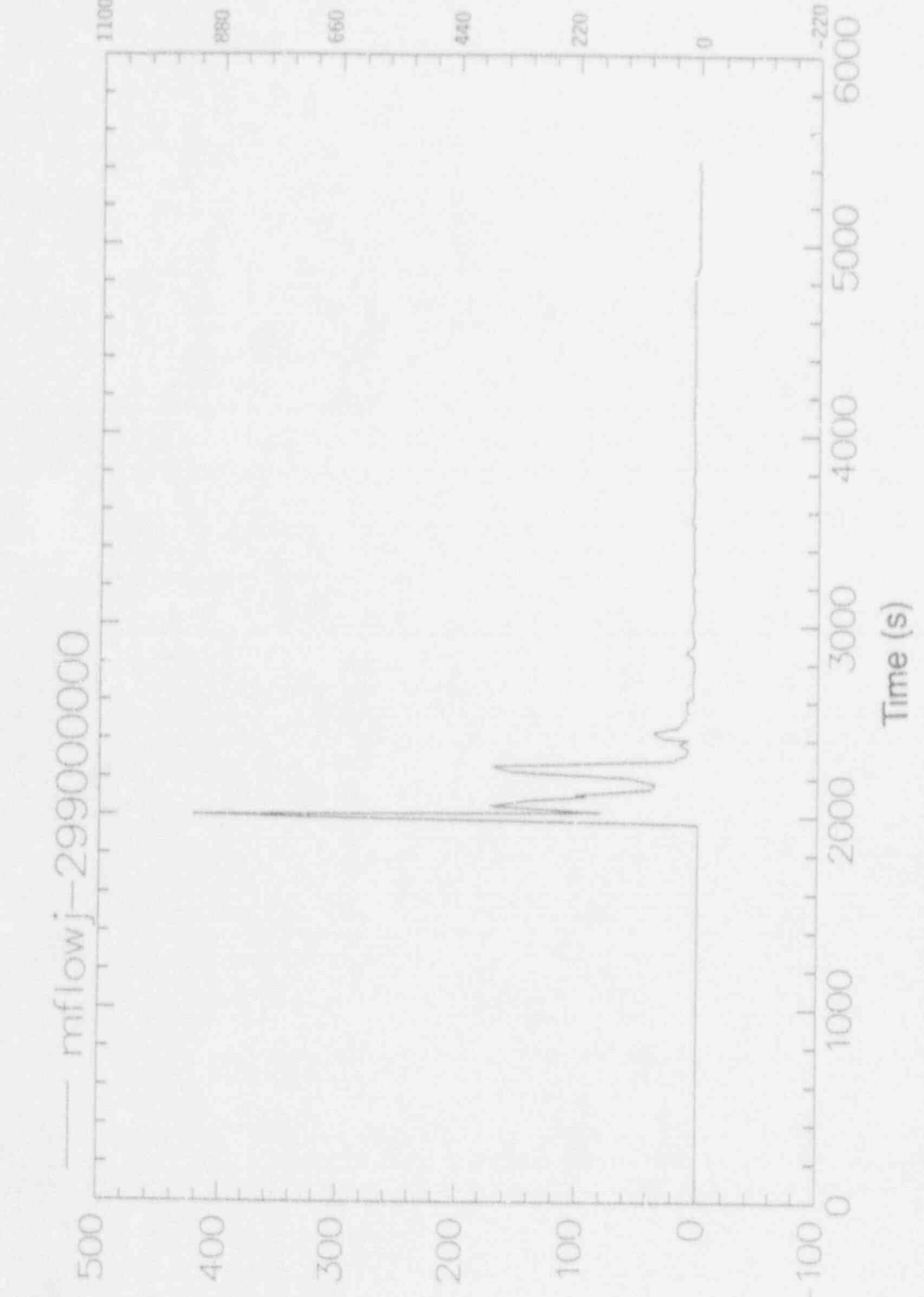

Figure 86. Break mass flow rate—nozzle dam failure analysis—Case 1.

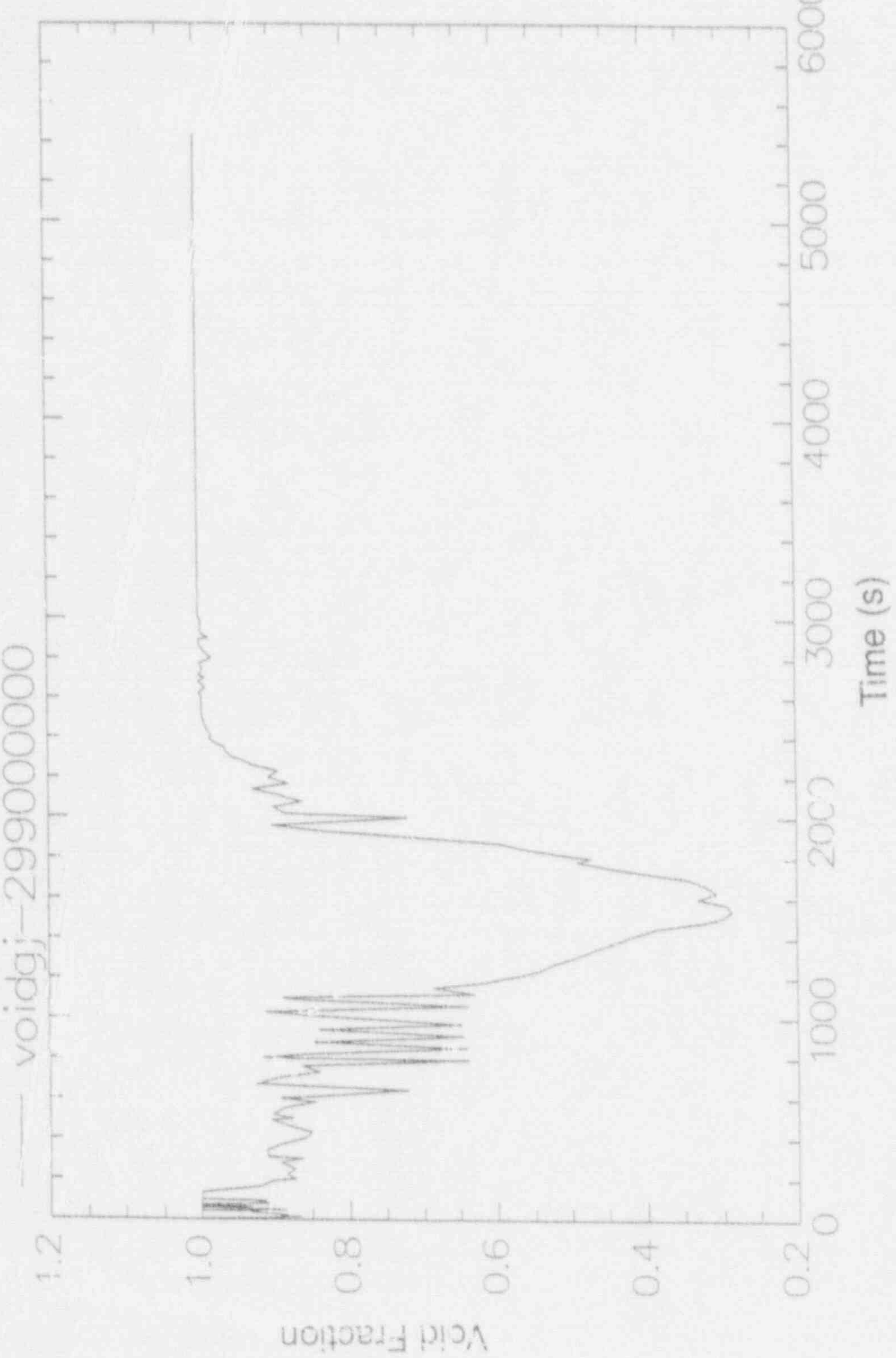

Figure 87. Break flow void fraction—nozzle failure analysis—Case 1.

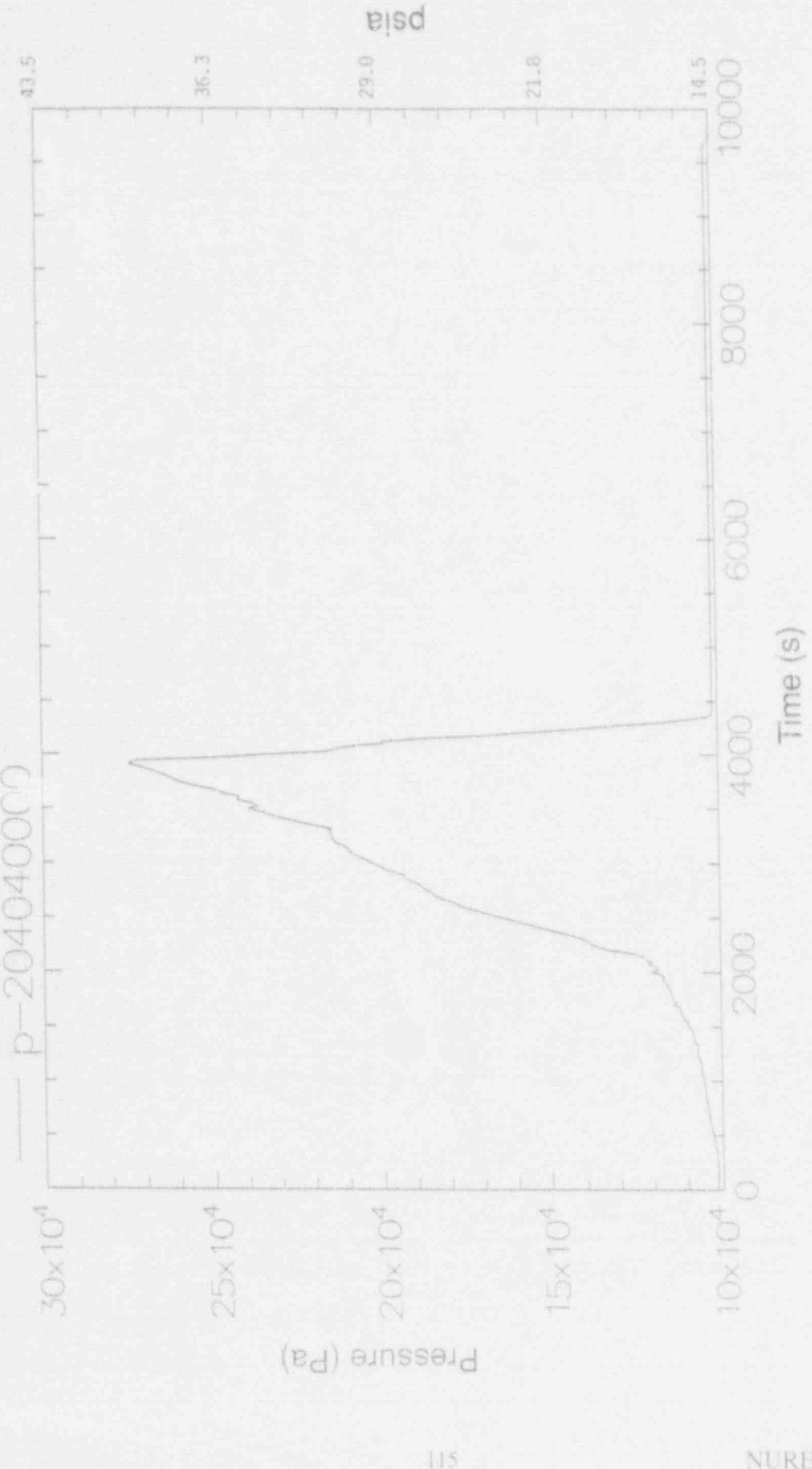

Figure 88. Primary system pressure for nozzle dam failure analysis—Case 2.

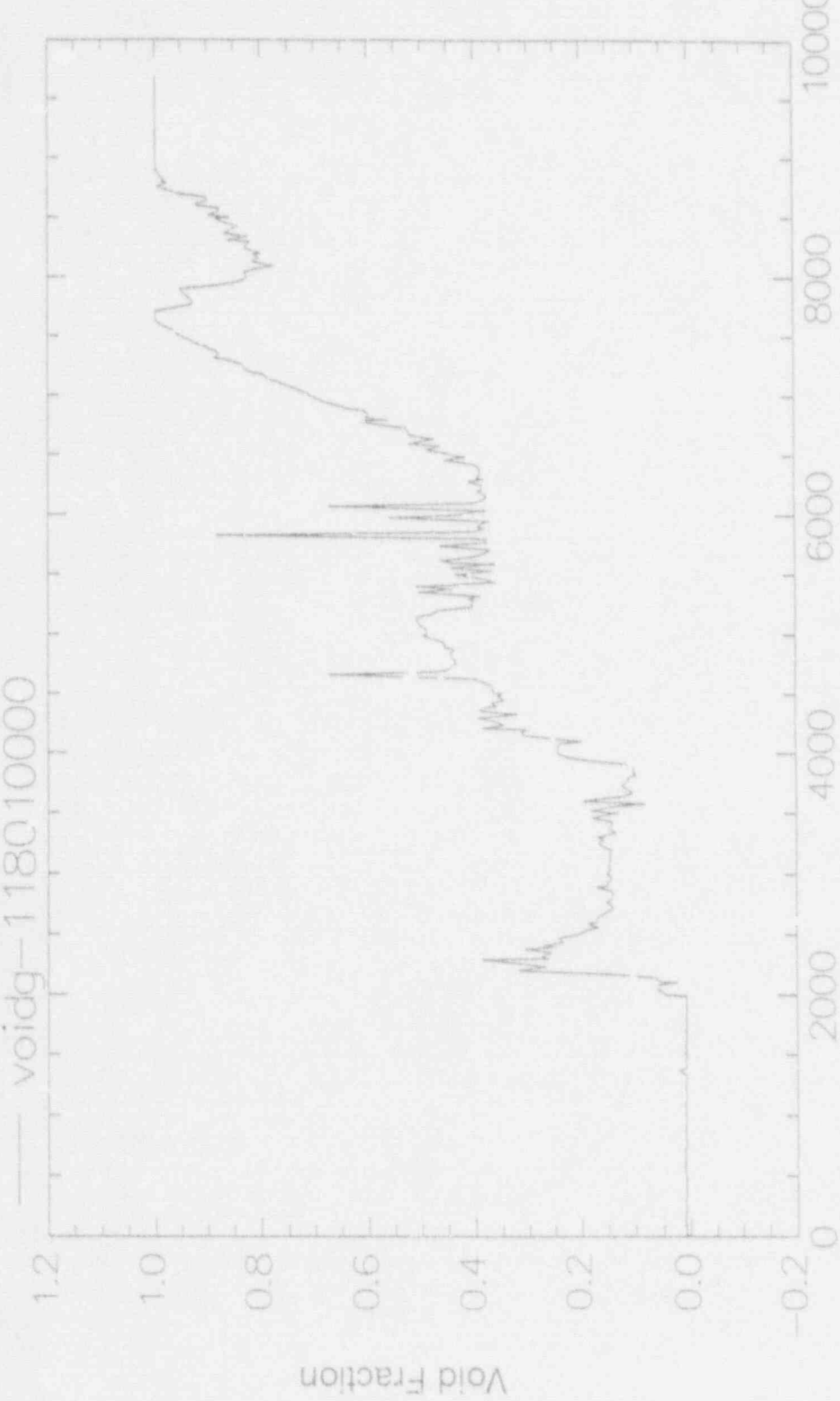

Figure 89. Void fraction in upper plenum volume just above core for nozzle dam failure analysis—Case 2.

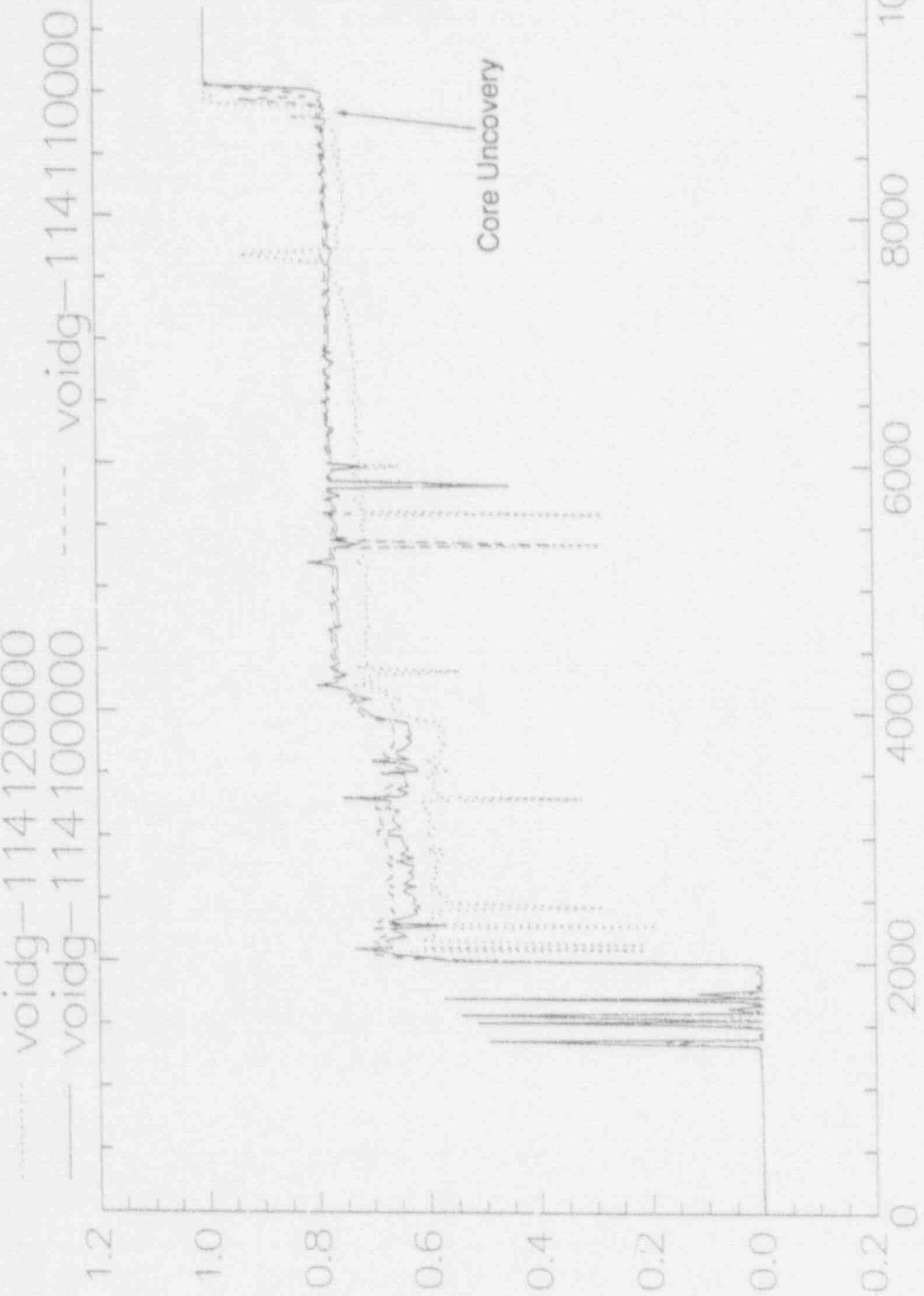

Figure 90. Void fraction for top three core volumes for nozzle dam failure analysis—Case 2.

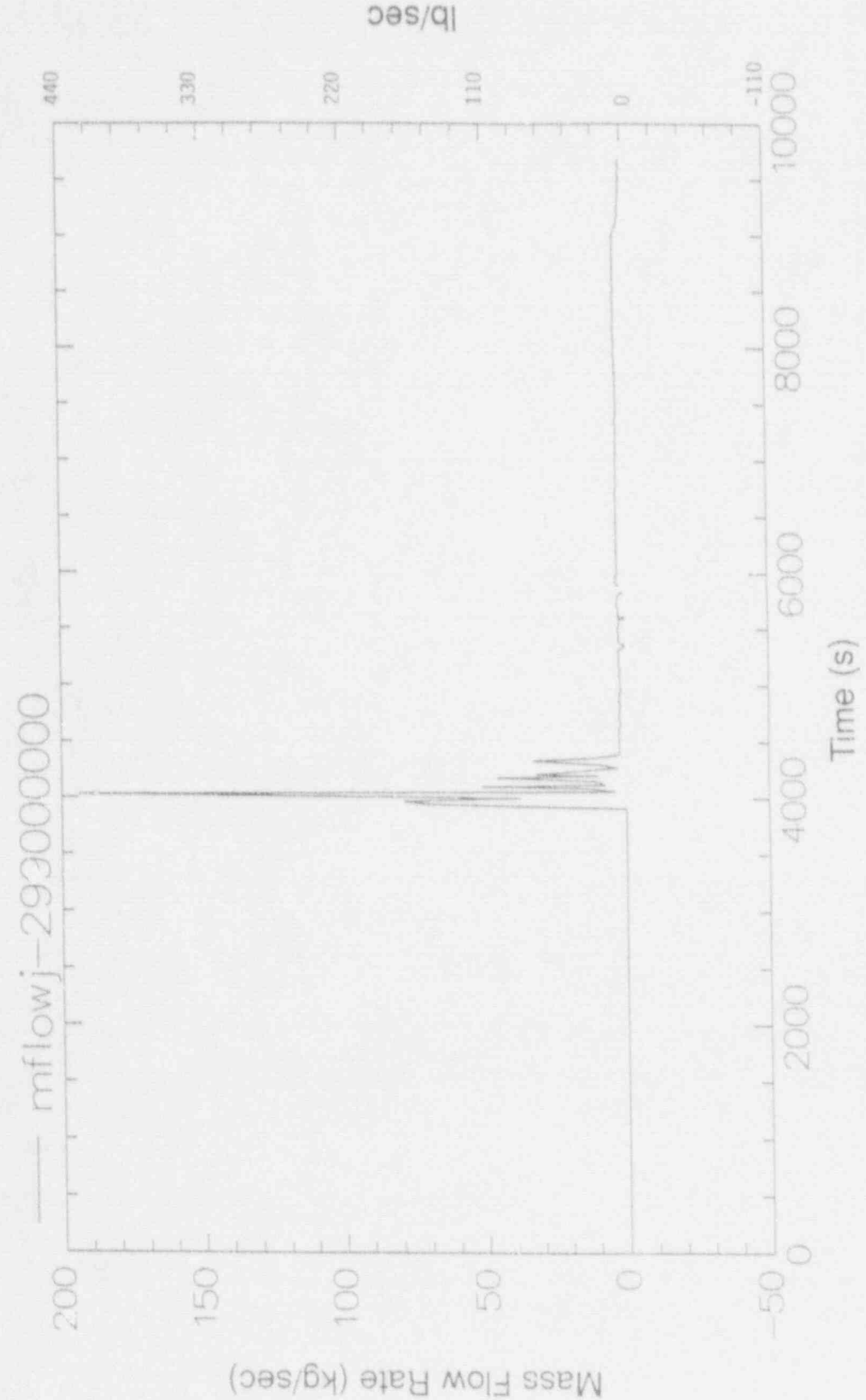

Figure 91. Break mass flow rate—nozzle dam failure analysis—Case 2.

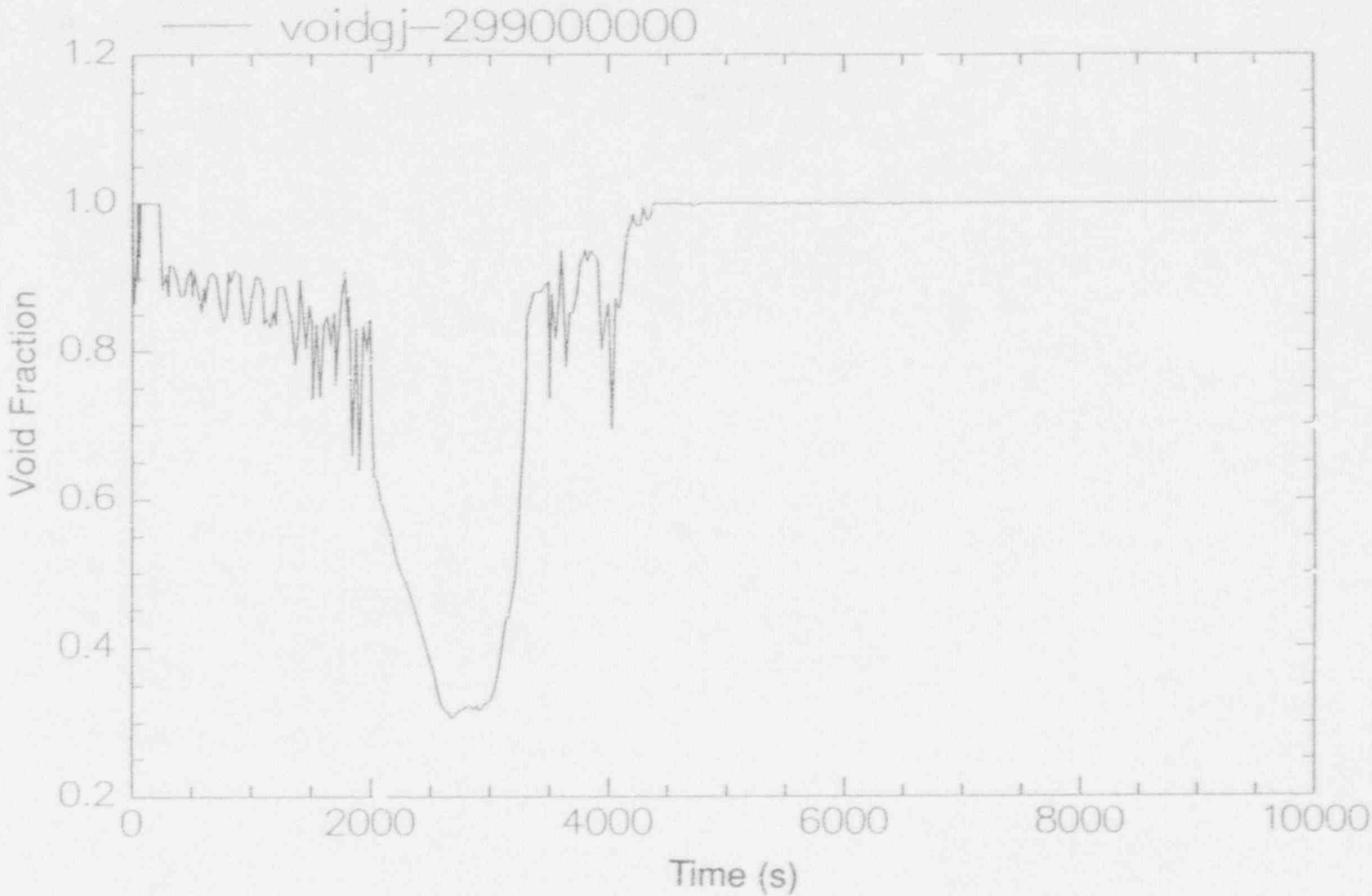

Figure 92. Break flow void fraction—nozzle dam failure analysis—Case 2.

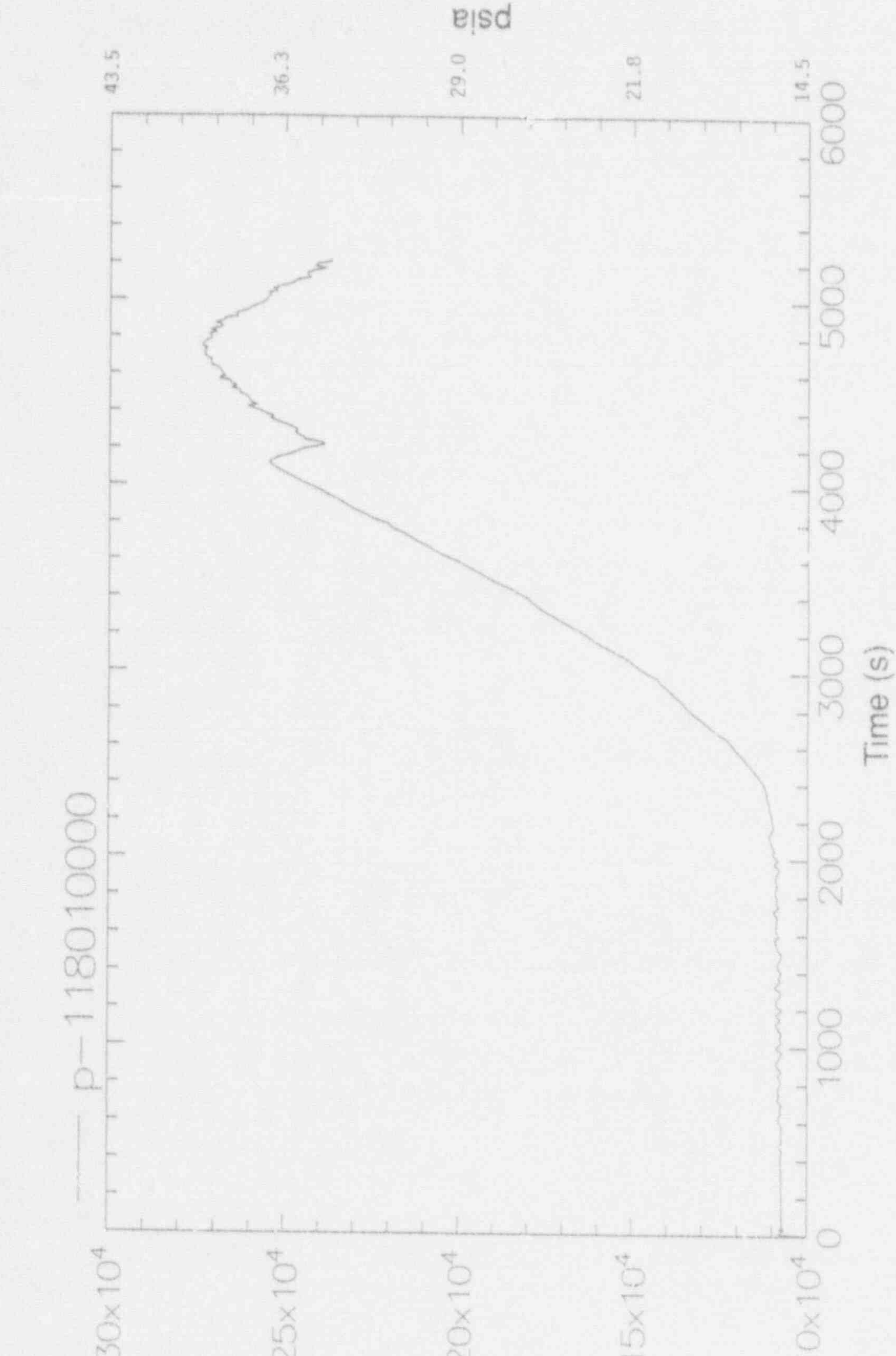

Figure 93. Pressure in the upper plenum volume just above the core for the SI line opened analysis.

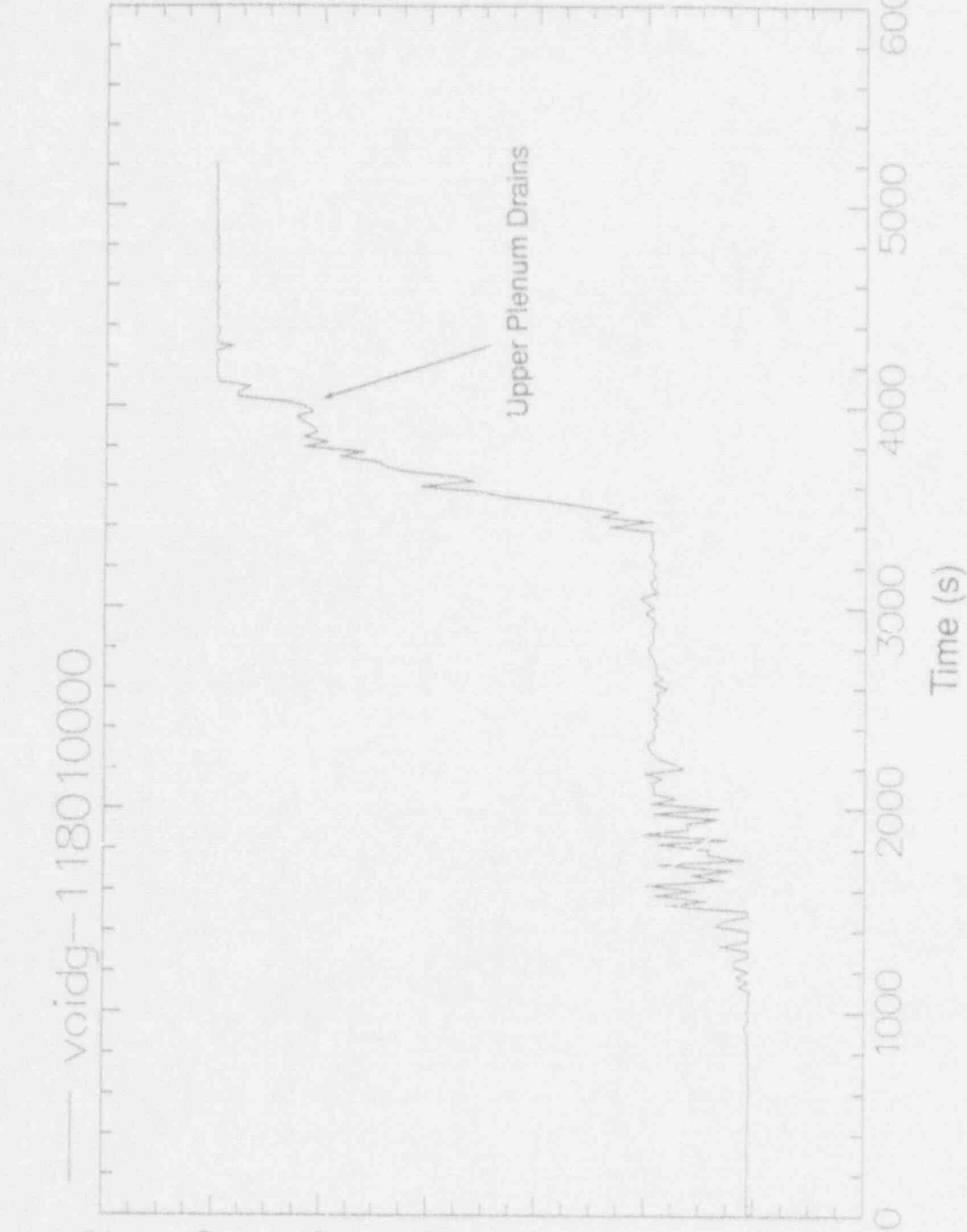

Figure 94. Void fraction in the upper plenum volume just above the core for the SI line opened analysis.

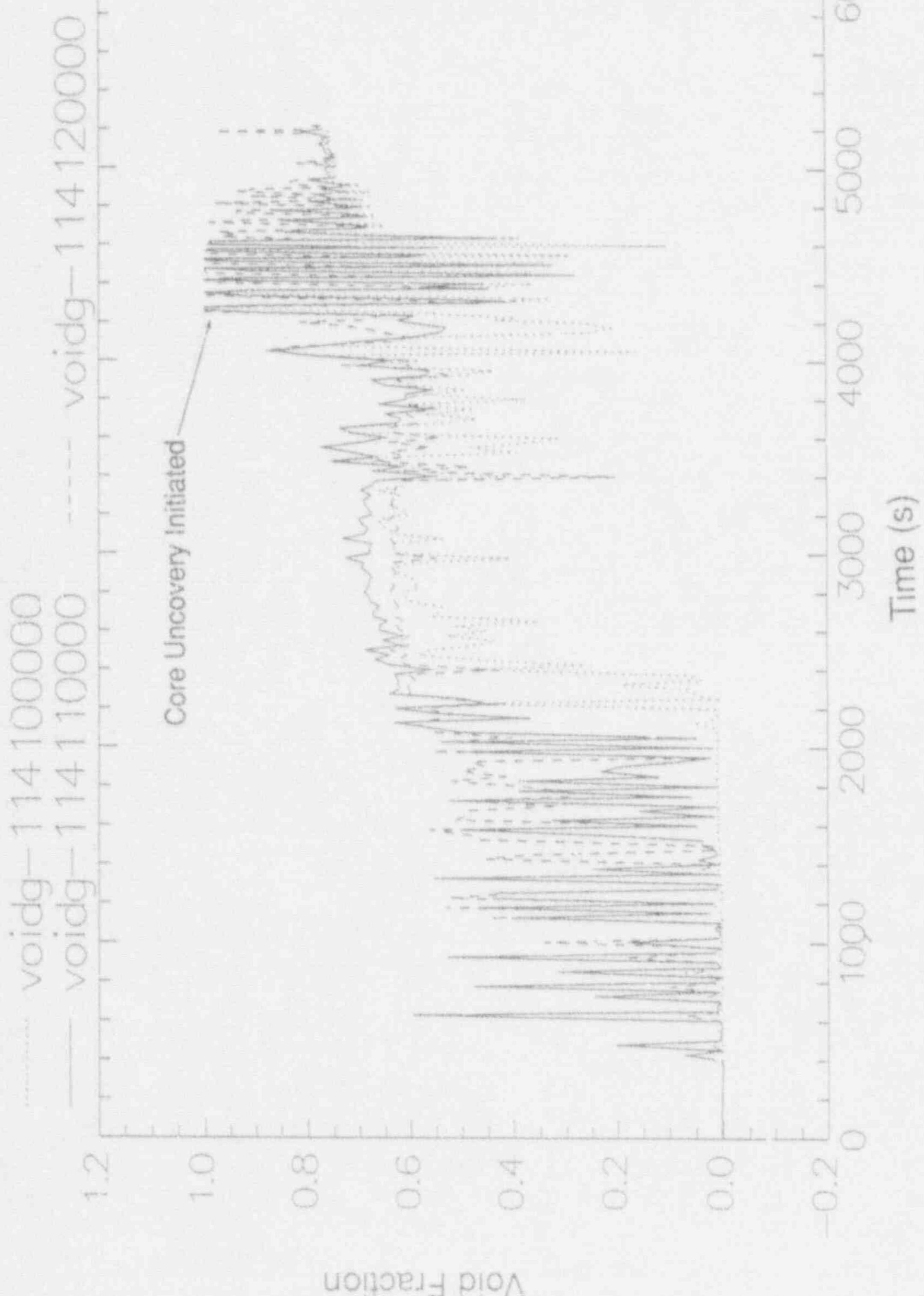

Figure 95. Void fraction in the top three core volumes for the SI line opened analysis.

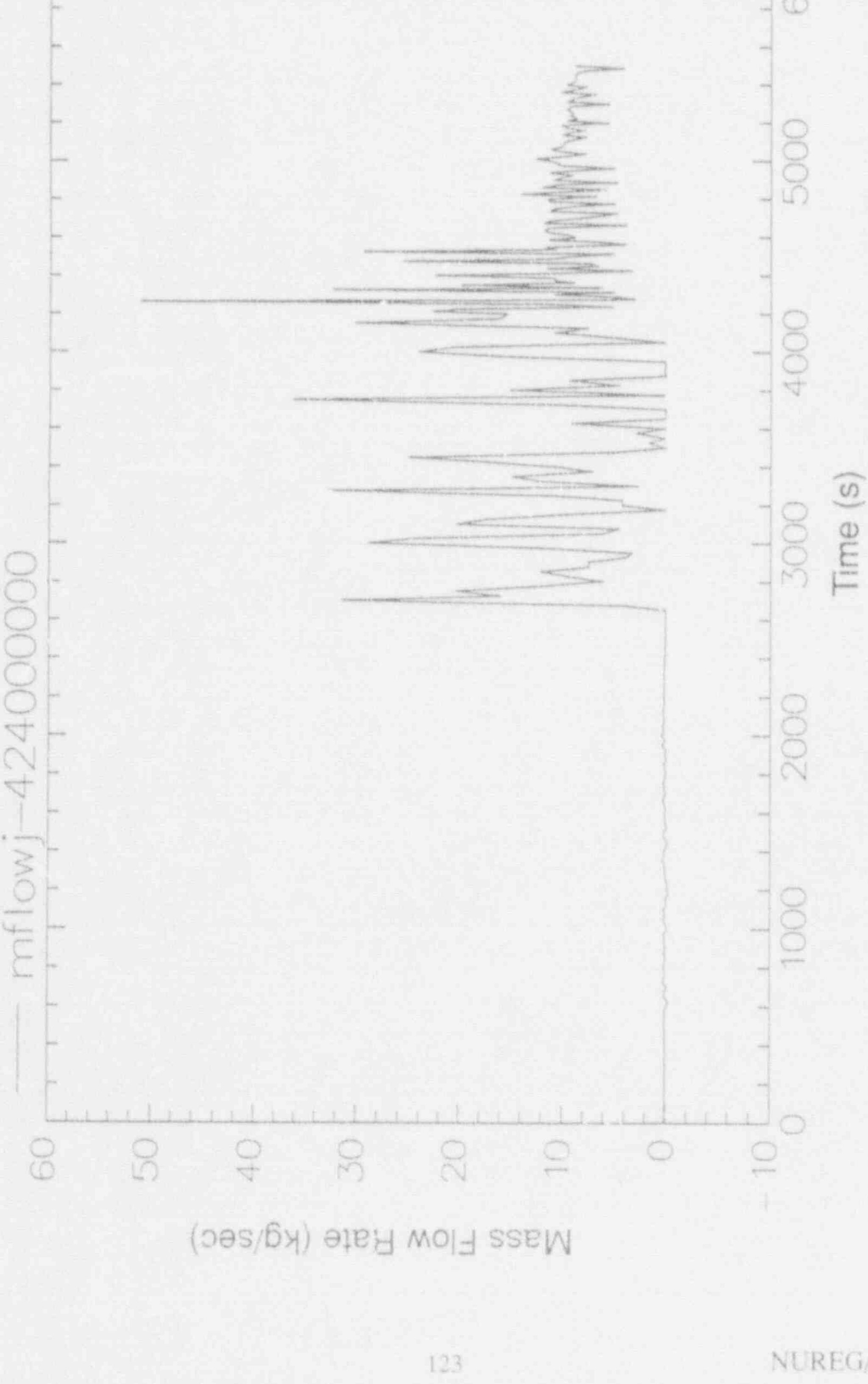

Figure 96. Break mass flow rate—SI line opened analysis.

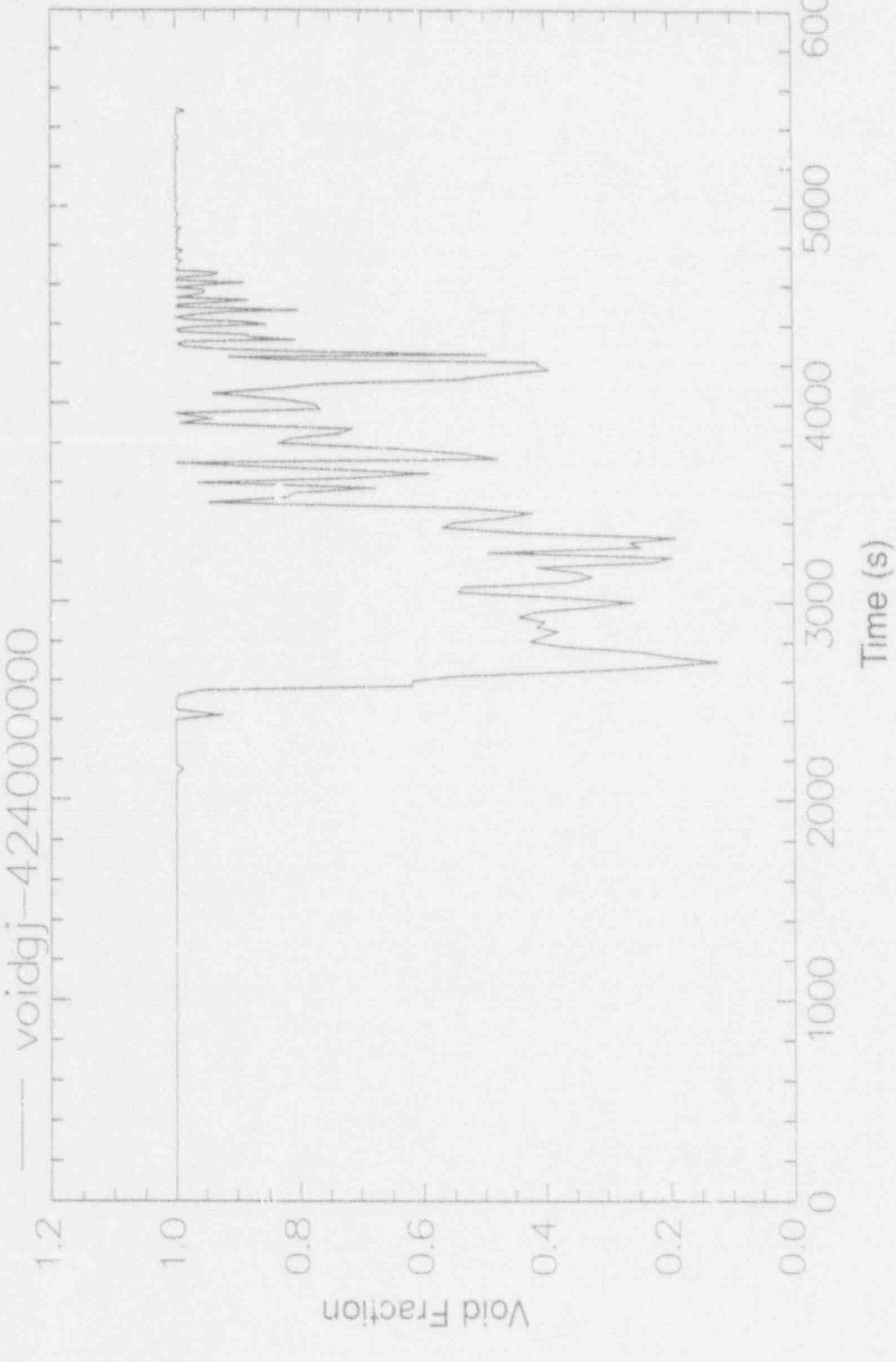

Figure 97. Break flow void fraction—SI line opened analysis.

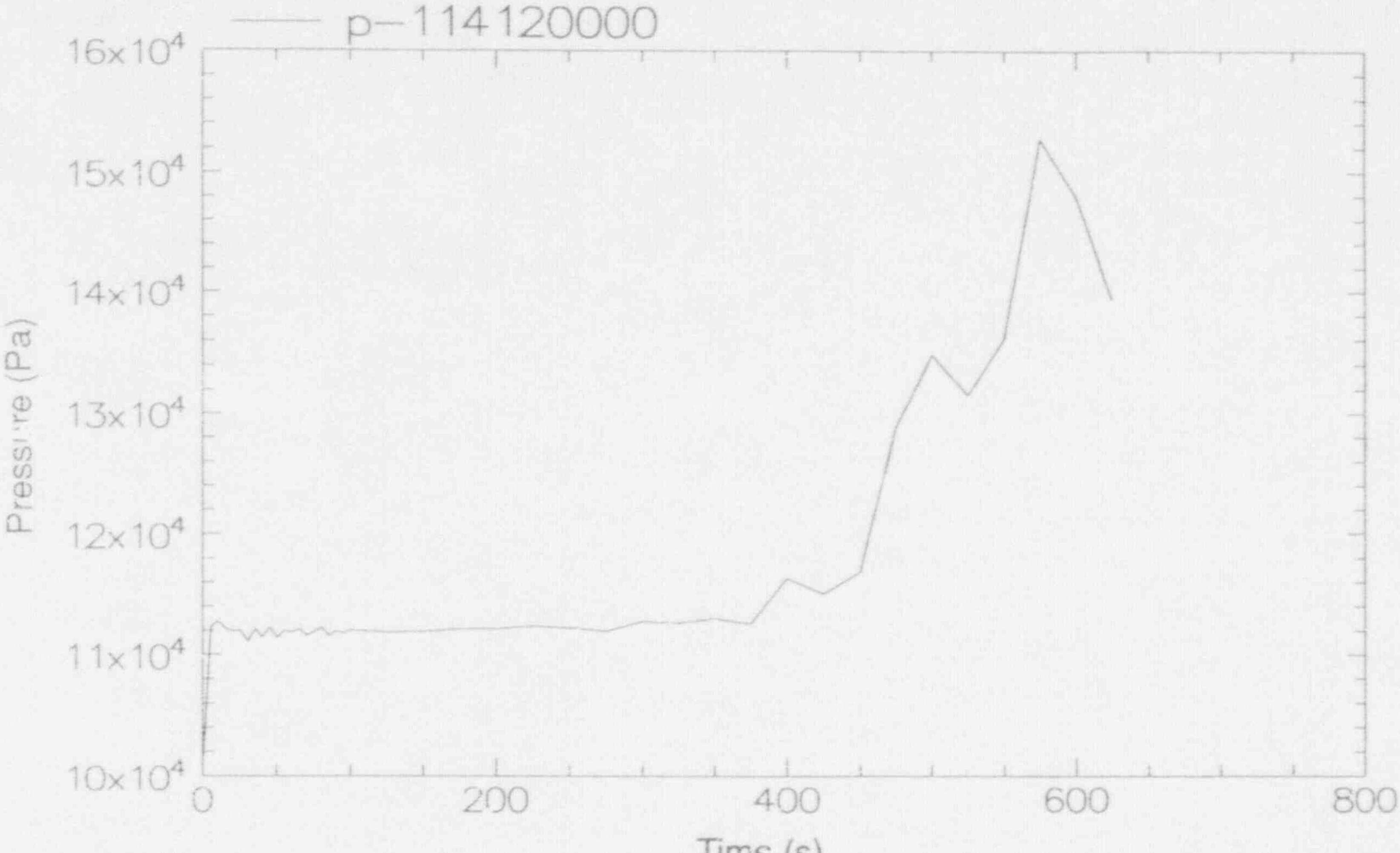

Figure 98. RCS pressure for open SI line bounding analysis.

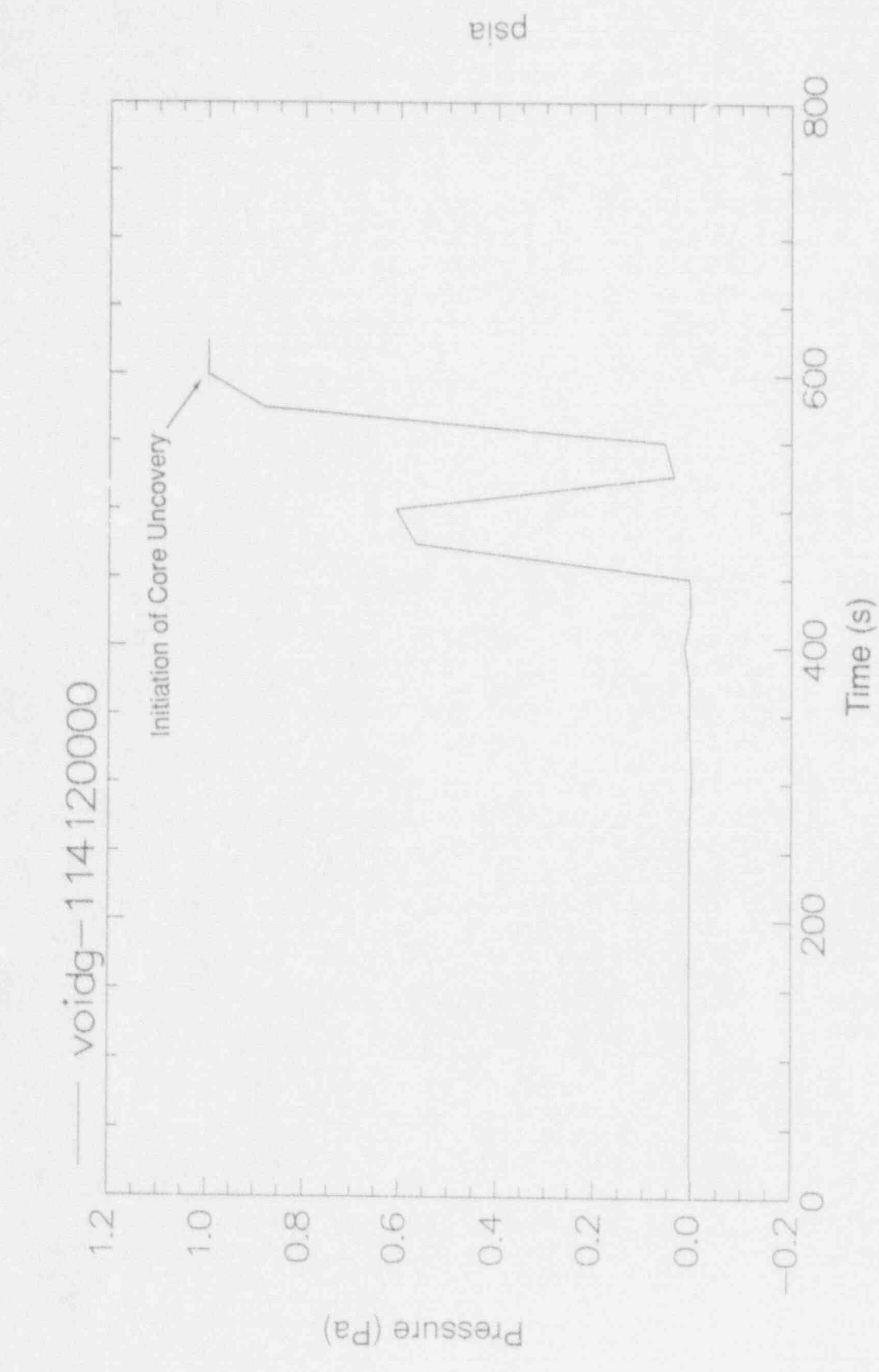

Figure 99. Void fraction in top core volume for open SI line bounding analysis.

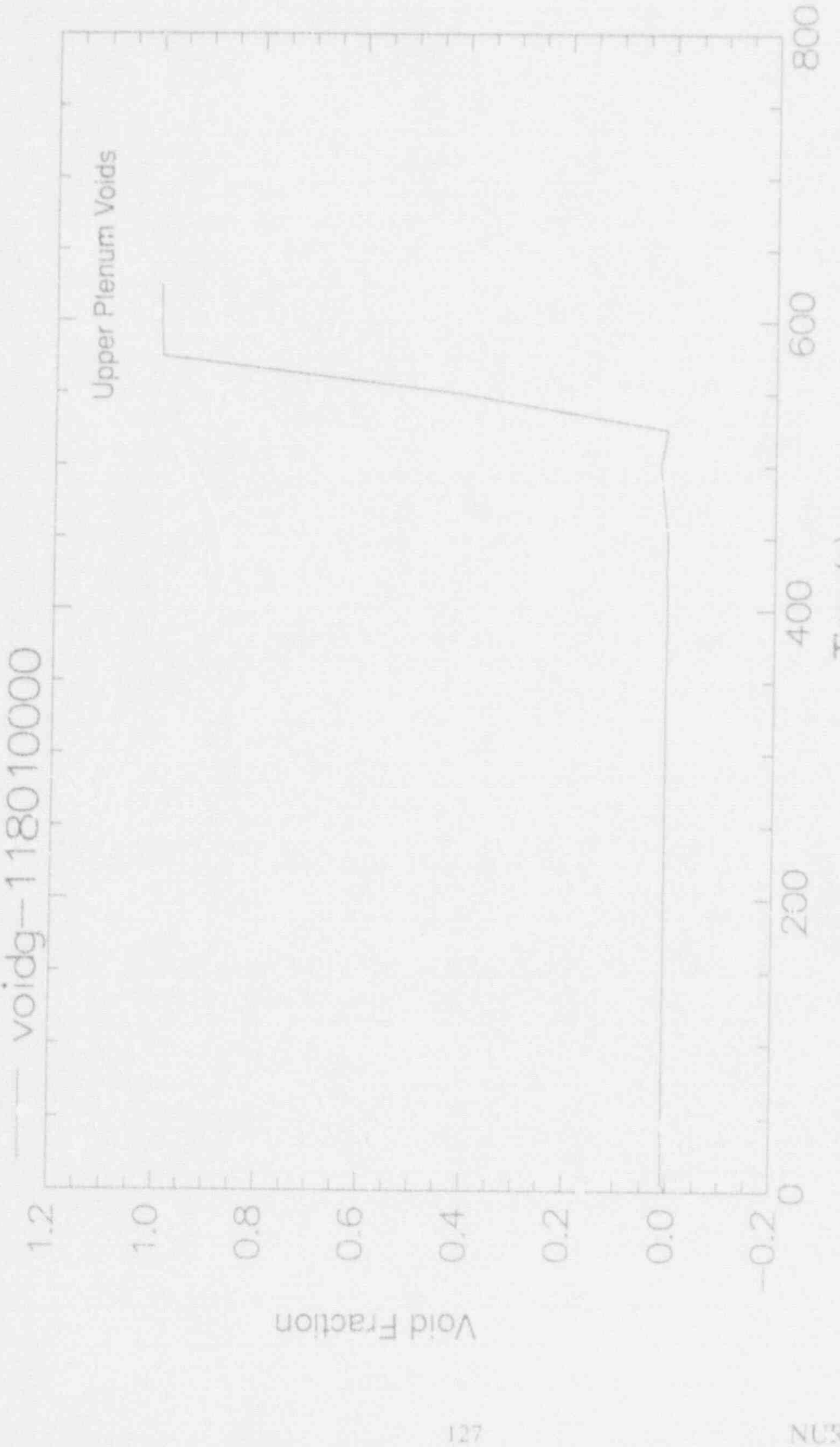

Figure 100. Upper plenum void fraction for open SI line bounding analysis.

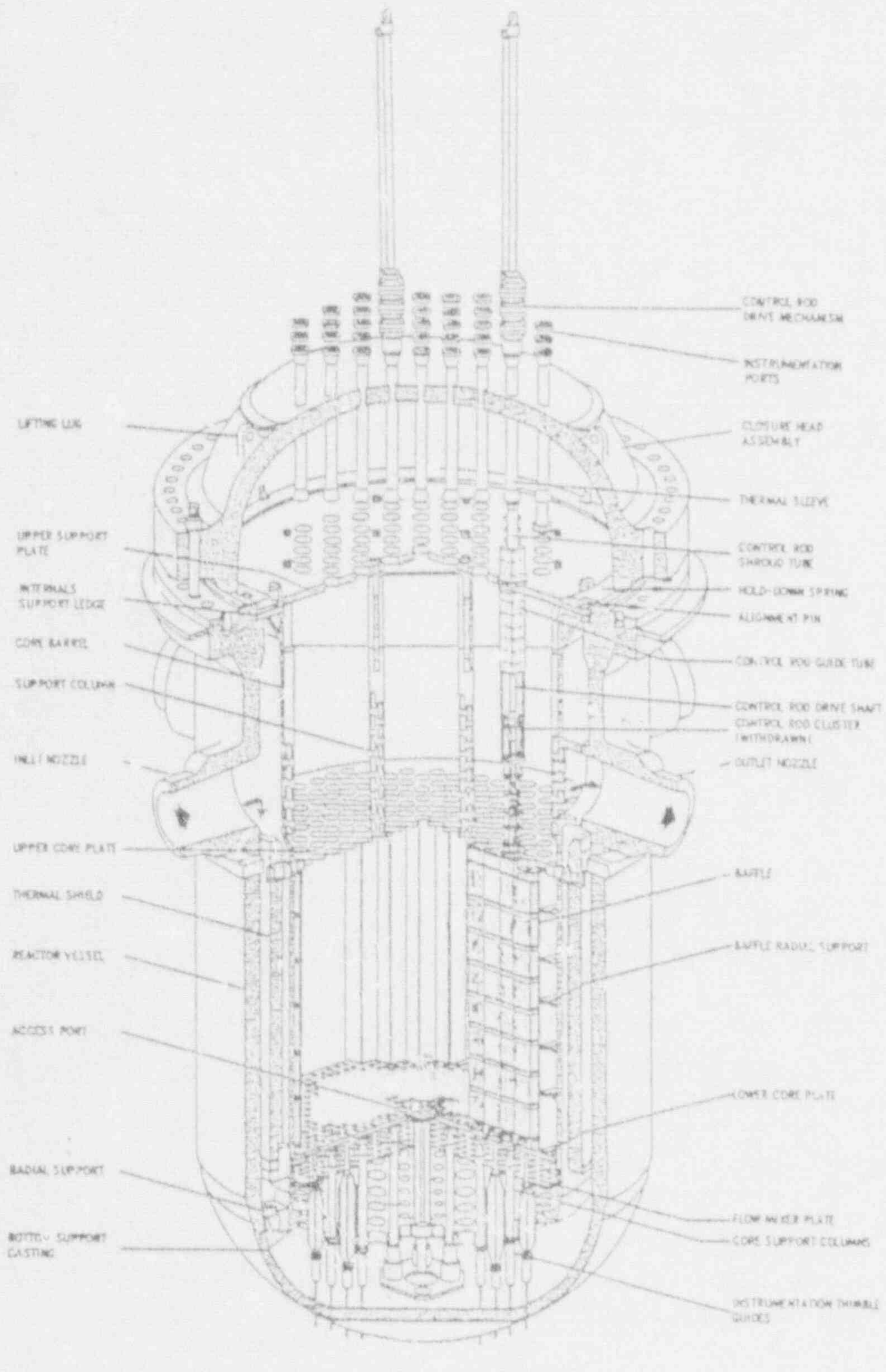

Figure 101. HBR vessel with upper head intact showing upper core support plate.

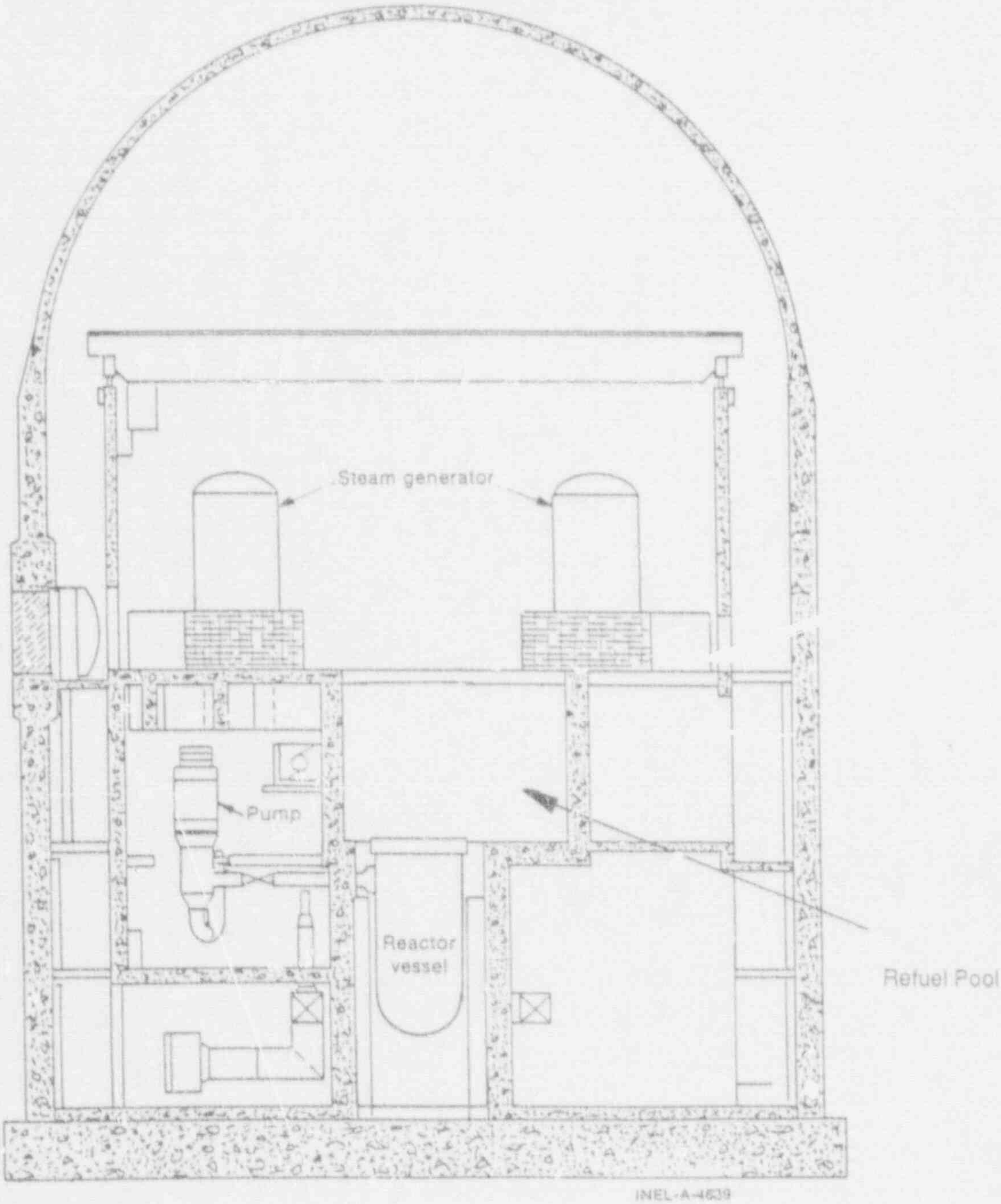

Figure 102. Typical dry containment, vessel head removed.

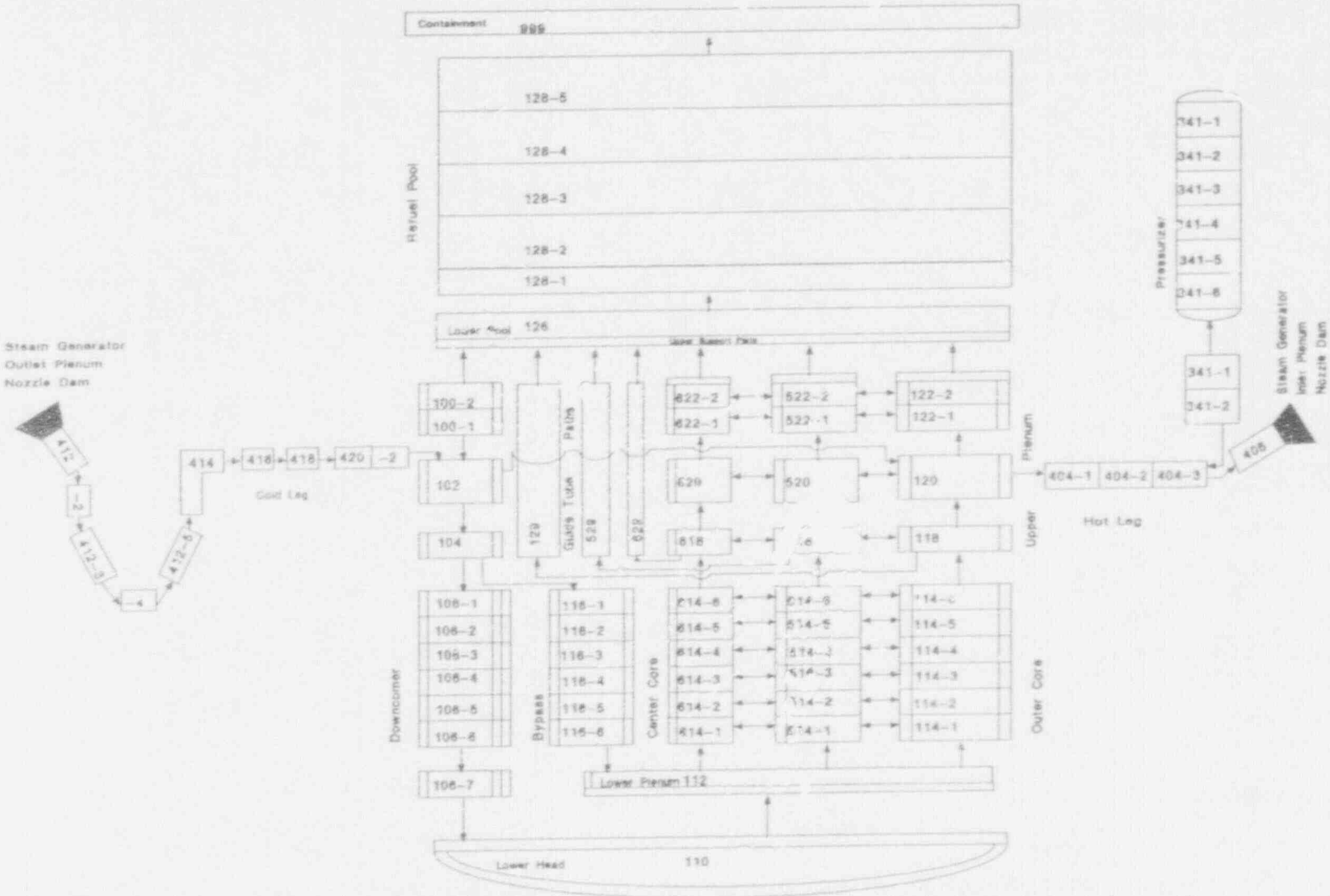

Figure 103. RELAP5 nodalization diagram for HBR-2.

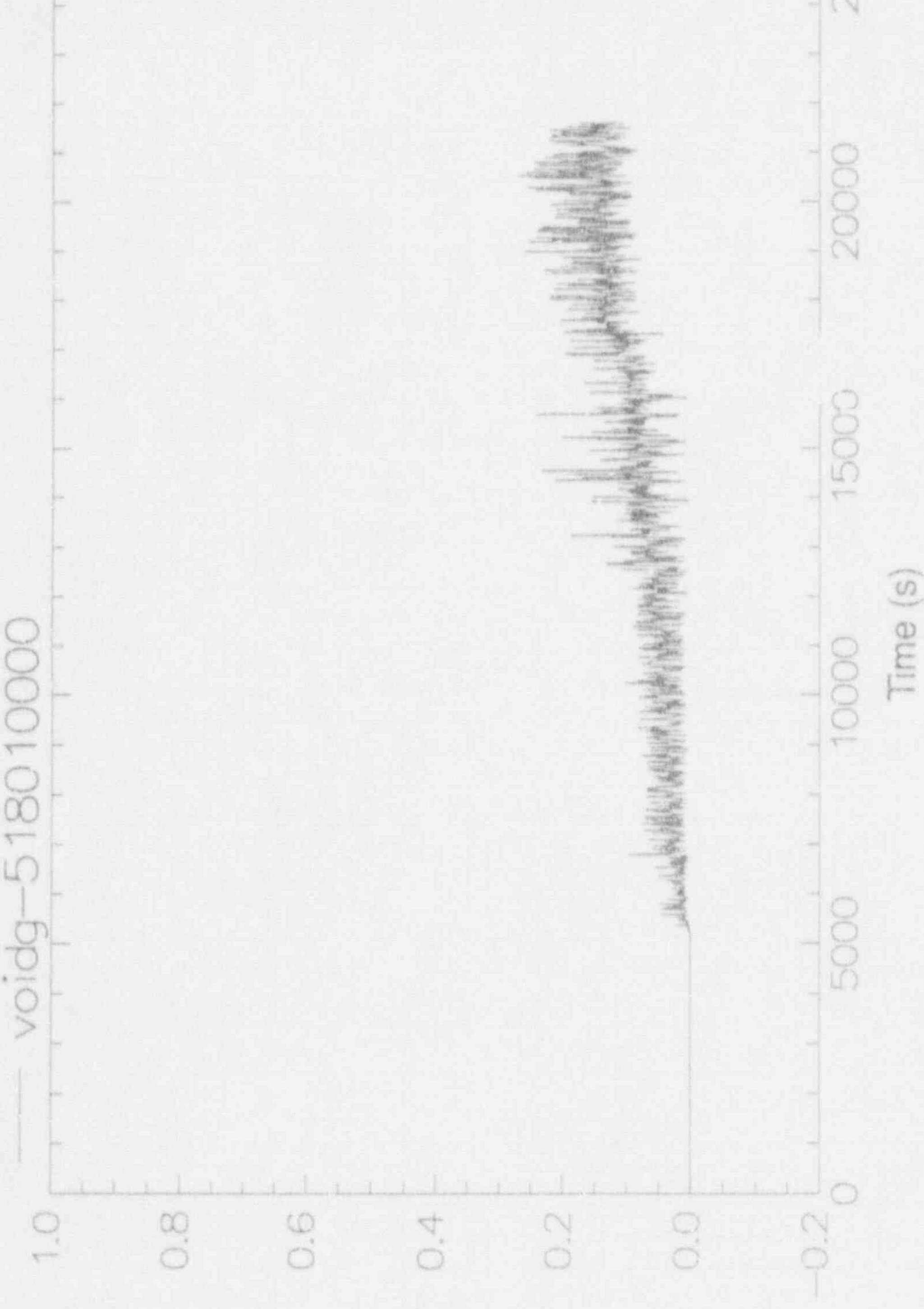

Figure 104. Void fraction just above core for Case 1.

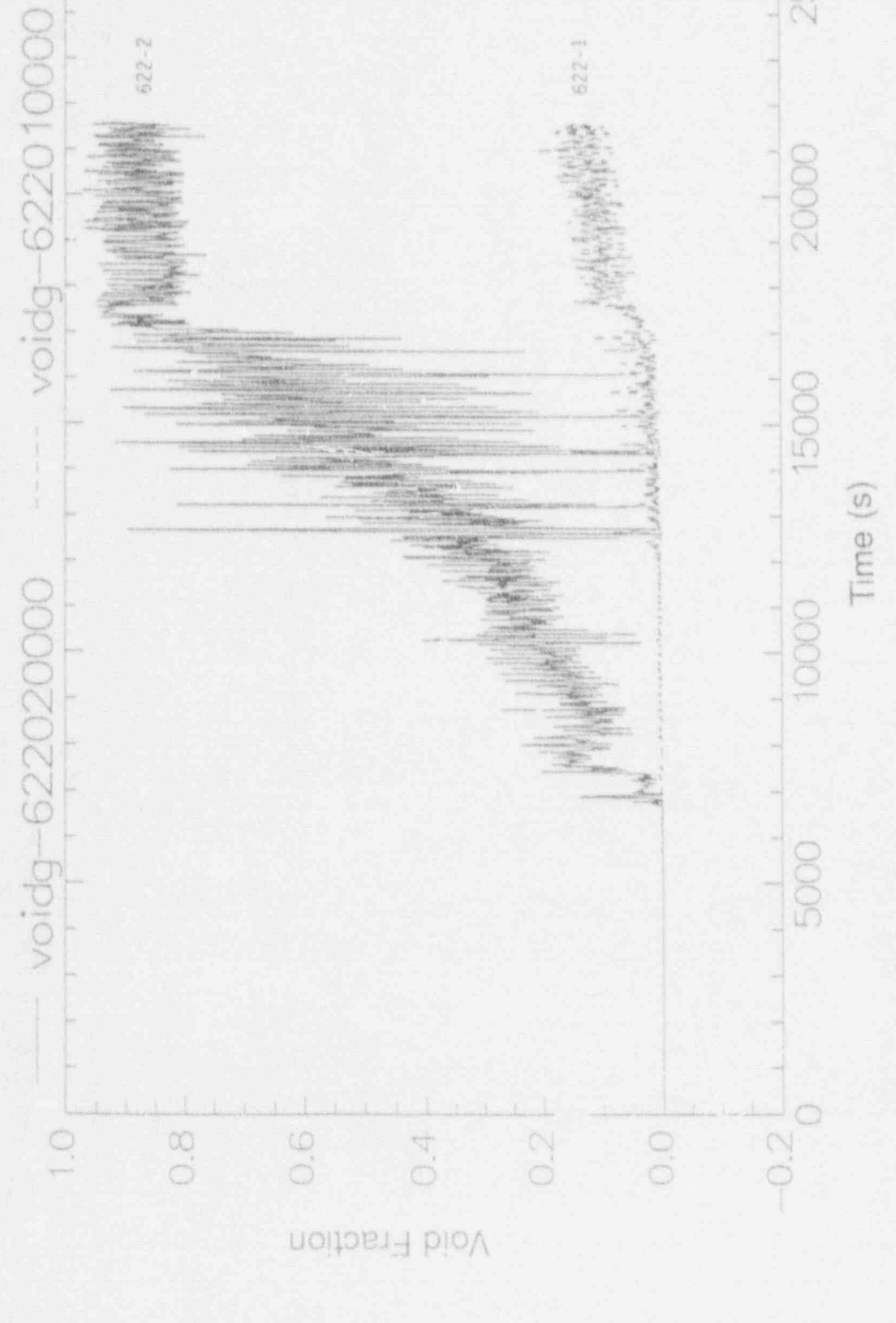

Figure 105. Void fraction in top of upper plenum for Case 1.

Figure 106. Upper plenum and pool temperatures Case 1.

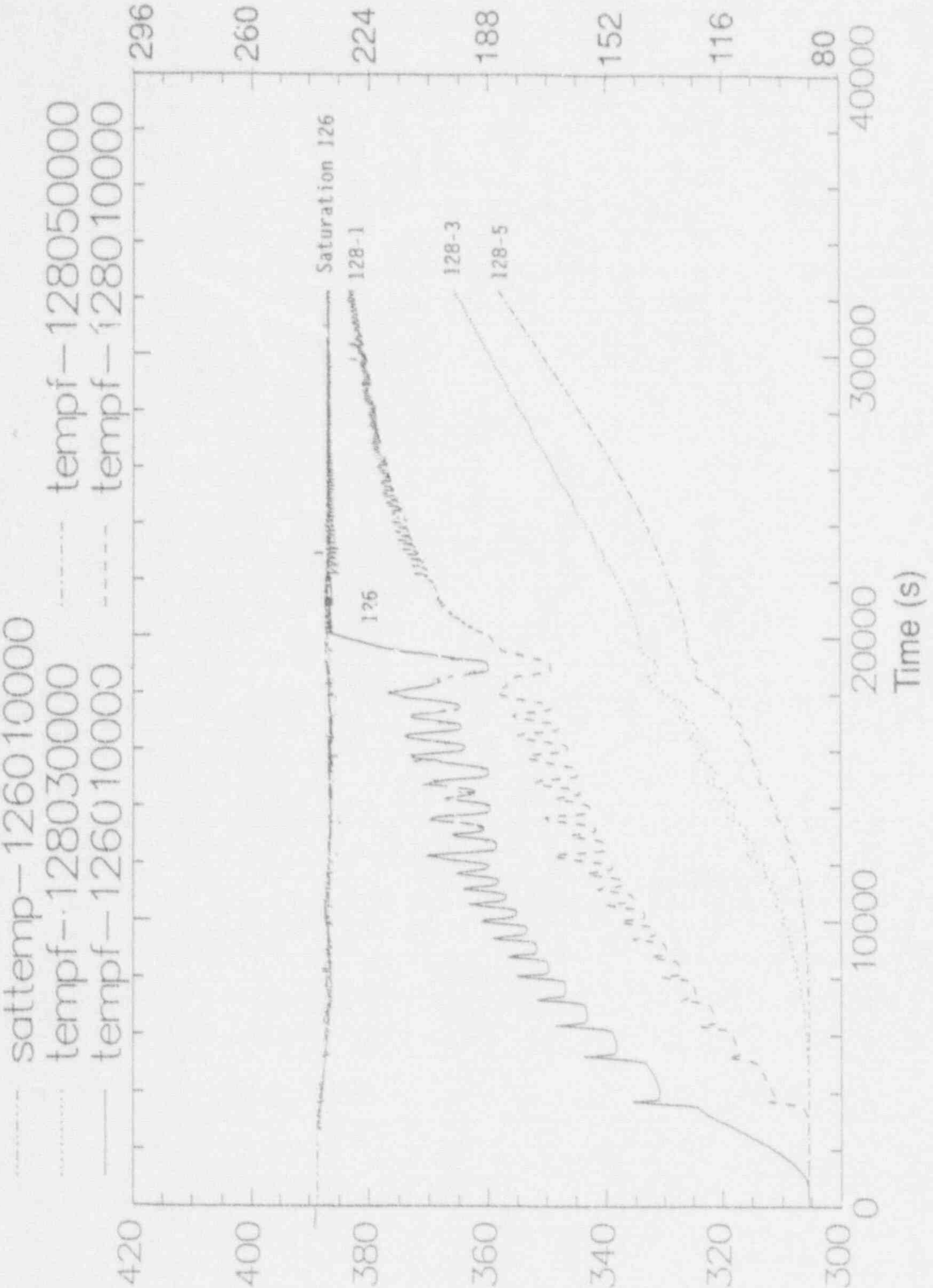

Figure 107. Pool and saturation temperatures Case 2.

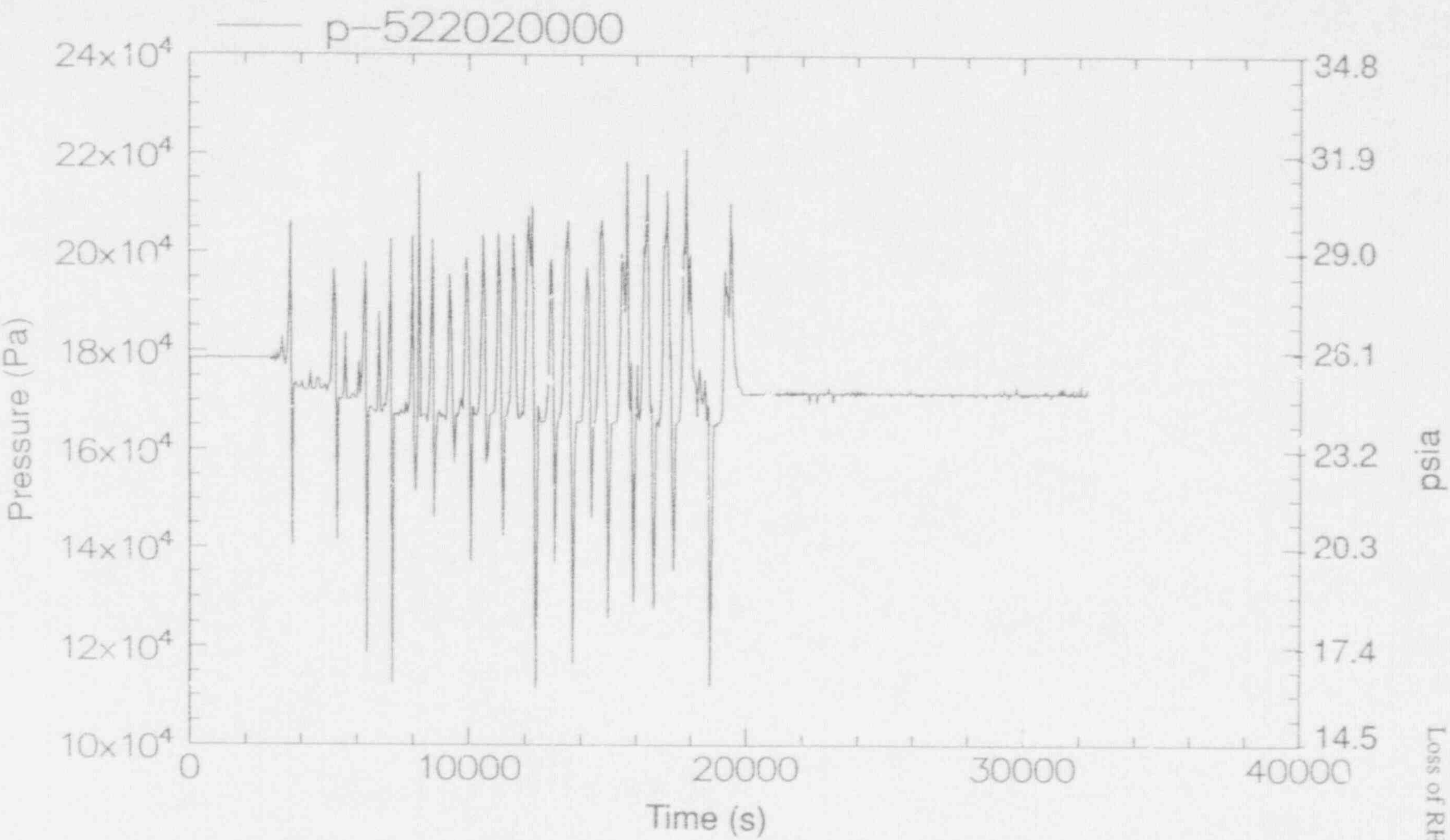

Figure 108. Vessel upper plenum pressure Case 2.

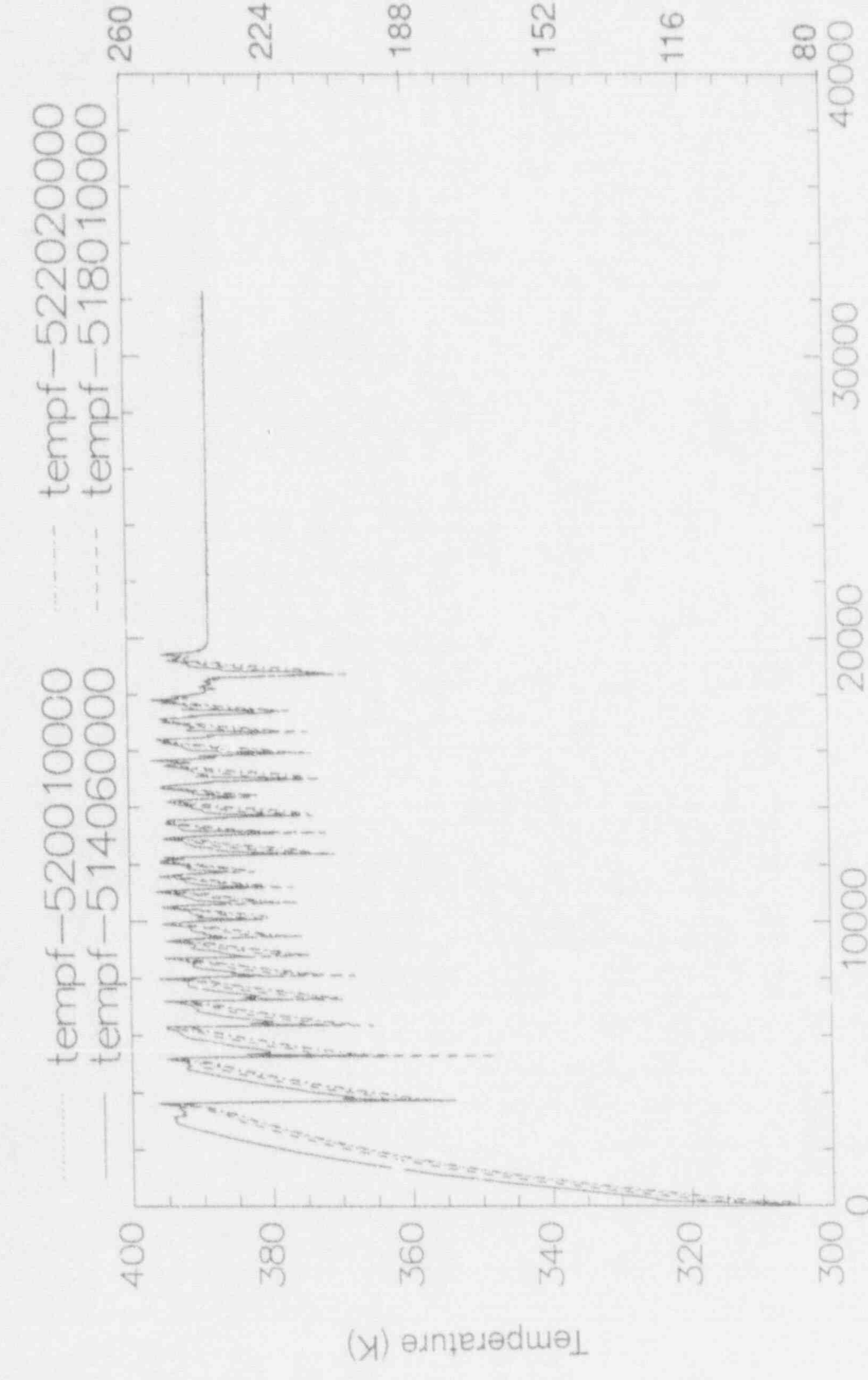

Figure 109. Top core and three upper plenum temperatures Case 2.

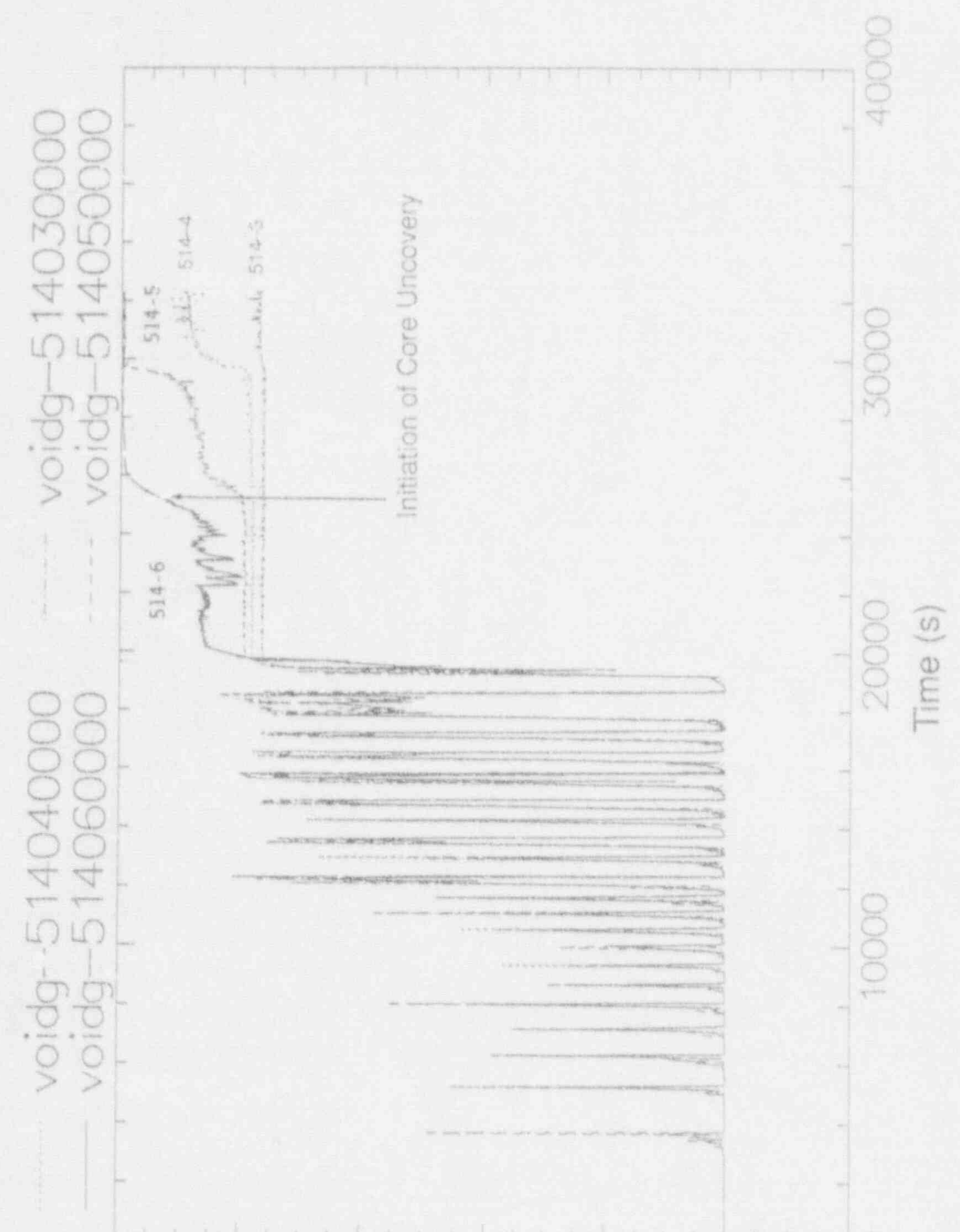

Figure 110. Void distribution for top four core volumes Case 2

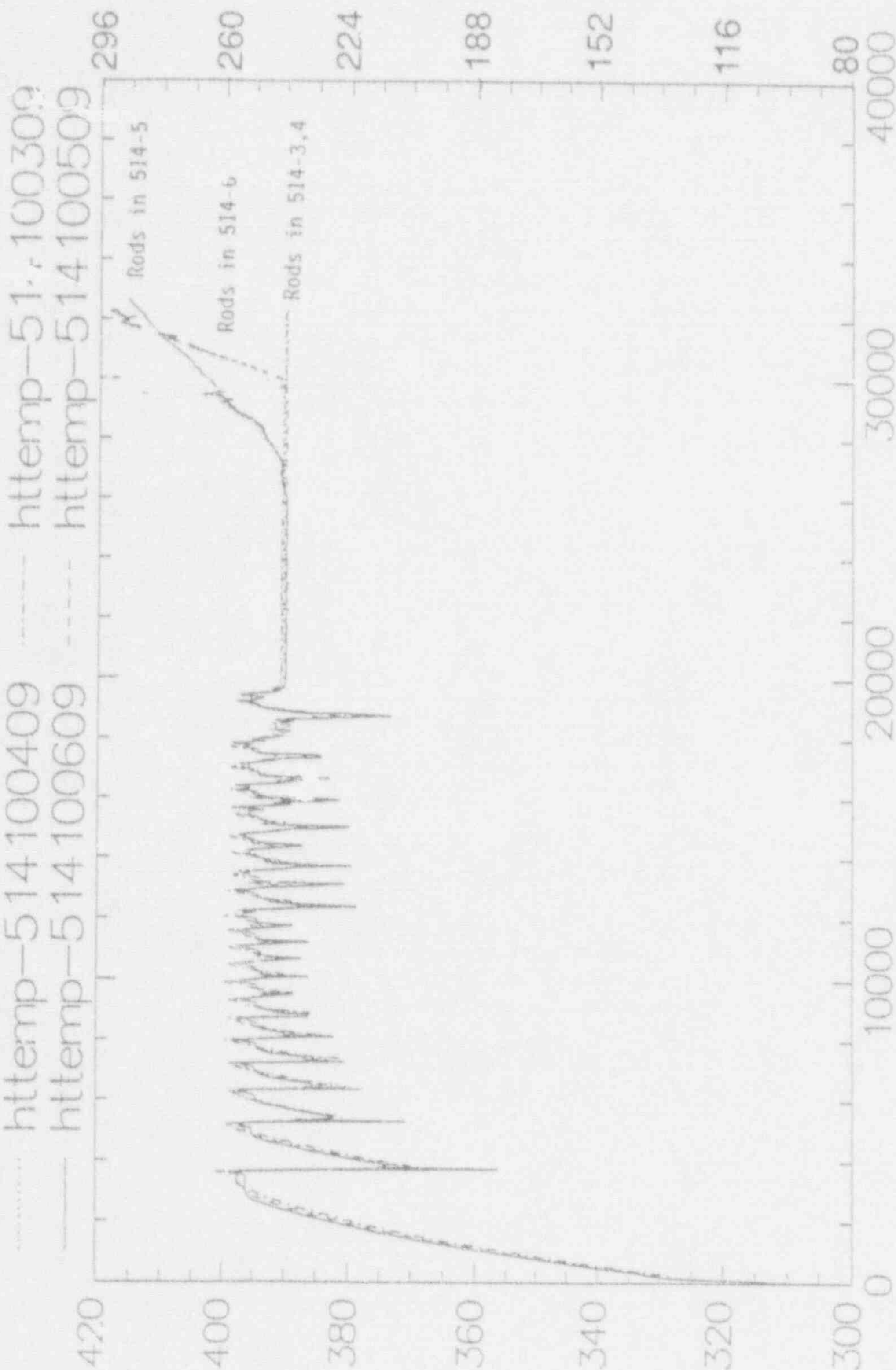

Figure 111. Clad temperatures in top four volumes of the core Case 2.

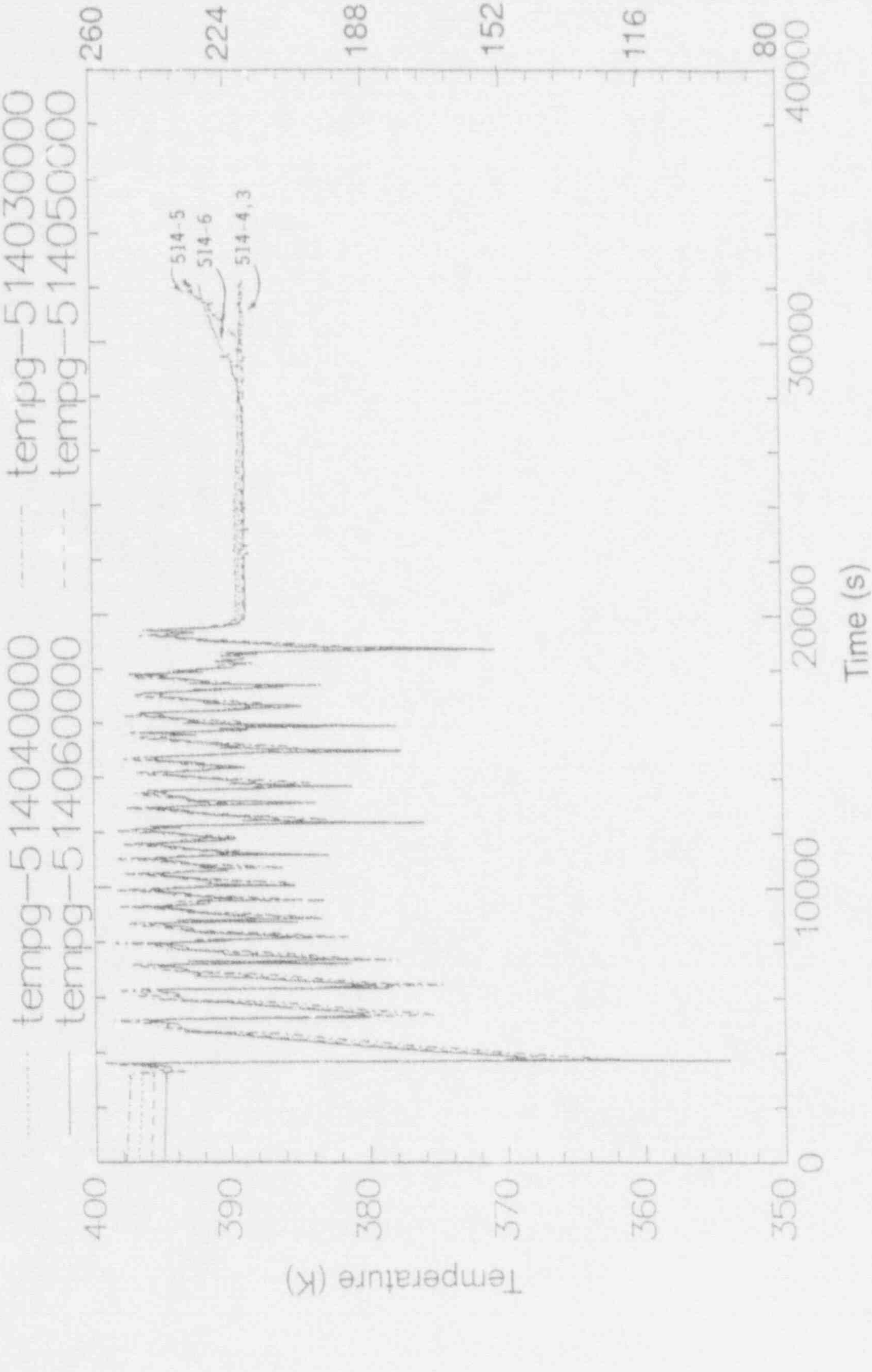

Figure 112. Steam temperature in top four volumes of the core Case 2.

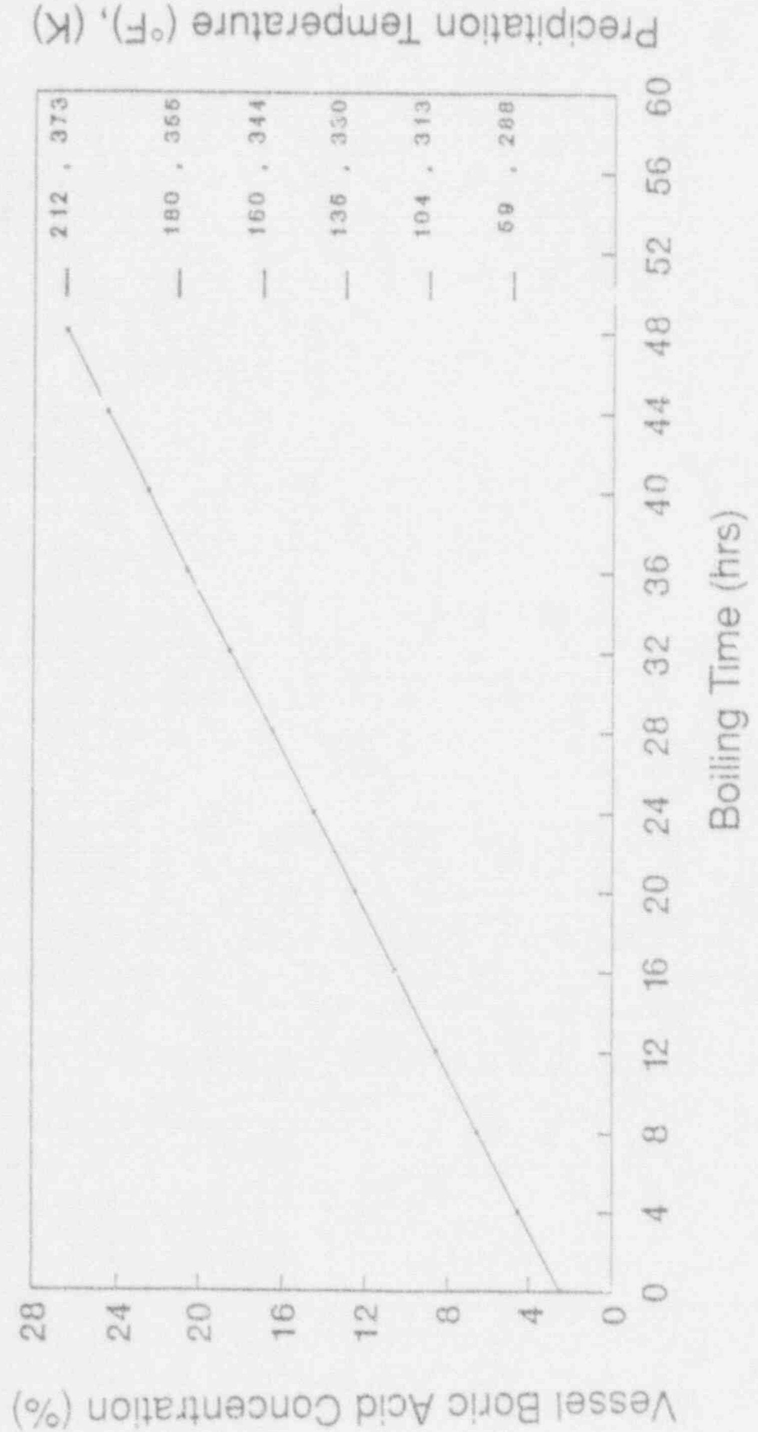

Figure 113. Vessel boron concentration vs. time injection source RWST (2200 ppm).

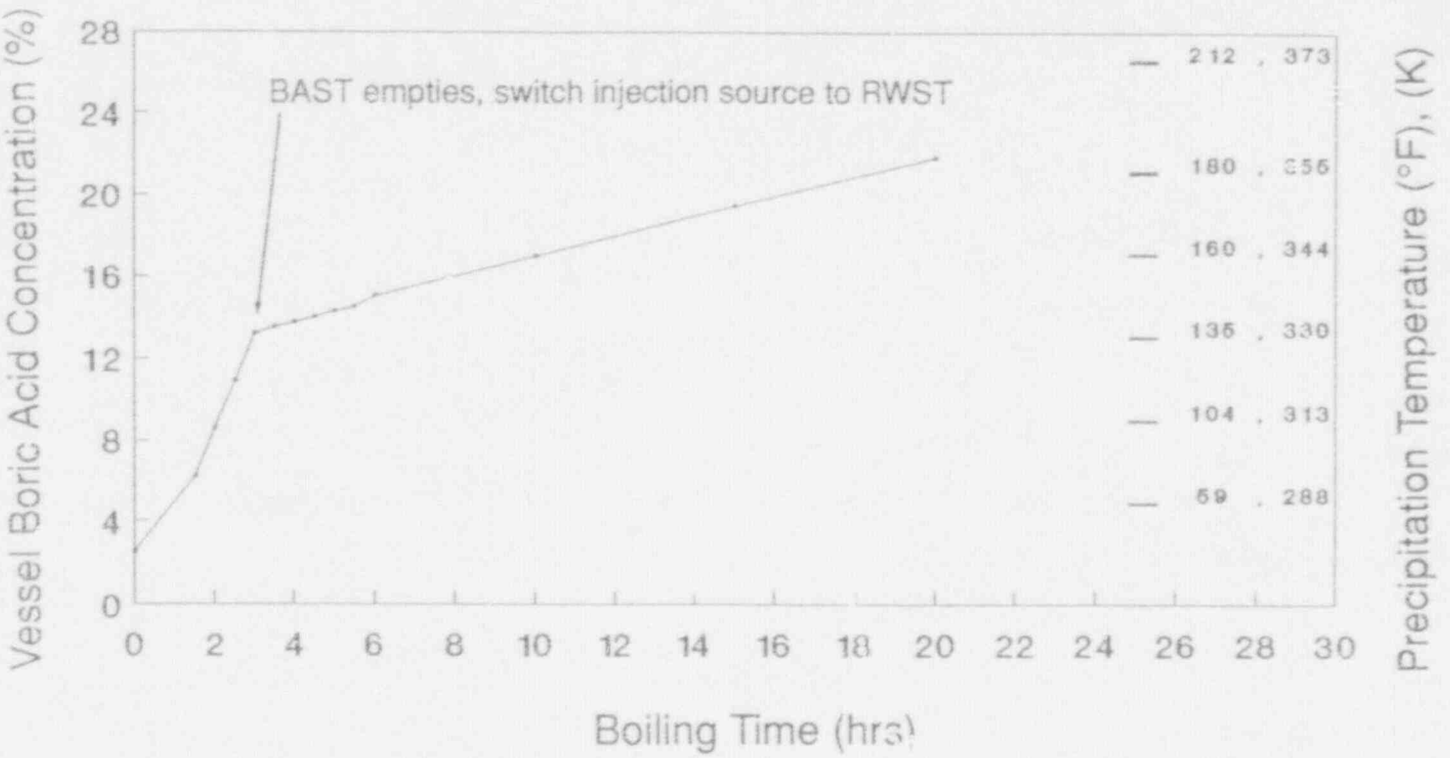

Figure 114. Vessel boron concentration vs. time injection source BAST (22,000 ppm) BAST empties at ~3 hours with injection switched to RWST.

3. ALTERNATE METHODOLOGY

This section presents a brief description of the alternate methods developed to assess the consequences of a loss of an RHRS event in PWRs. Section 3.1 describes the model while Section 3.2 presents the results of the model predictions to a PWR.

3.1 Model Description

Appendix A presents a description of the alternate methodology based on the work of Reference 3, which was used to perform the preliminary assessments of the consequences of loss of the RHRS. In this model, given in Figure A-1 of Appendix A, a mass and energy balance is performed on the primary system that is represented with two regions; a lower liquid region that can be either subcooled or saturated (region 1), and an upper region containing a noncondensible gas (region 2). The noncondensible gas consists of air that is treated as an ideal gas. In the development of this model, steam produced by boiling is assumed to act as a piston to compress the upper air region. The justification for this piston model—as opposed to a model that assumes that steam mixes with the upper air region—is provided in Section A-2 of Appendix A entitled, "Comparisons of the Methodology to Experimental Data."

The methodology consists of a computation of the mass, energy, and pressure for the steam-air-liquid fluid composition in the RCS with the overall depressurization rate given by

$$\frac{d}{dt}P = \frac{\left[h_1 \sum_{k=1}^{K} w_k - \left(\sum_{k=1}^{K} w_k h_k + Q_1\right)\right]\frac{\partial v_1}{\partial h_1} - v_1 \sum_{k=1}^{K} w_k}{M_1 \frac{\partial v_1}{\partial P} + \frac{1}{J}\frac{\partial v_1}{\partial h_1}V_1 + \frac{1}{J}\frac{V_2^2}{M_2 T_2}\left(\frac{1}{C_p} - \frac{J}{R}\right)} + \frac{\left[C_p T_2 \frac{dM_2}{dt} - \left(\sum_{n=1}^{N} w_n c_p T_n + Q_2\right)\right]\frac{V_2}{M_2 C_p T_2} - \frac{V_2}{M_2}\frac{dM_2}{dt}}{M_1 \frac{\partial v_1}{\partial P} + \frac{1}{J}\frac{\partial v_1}{\partial h_1}V_1 + \frac{1}{J}\frac{V_2^2}{M_2 T_2}\left(\frac{1}{C_p} - \frac{J}{R}\right)} \qquad (4)$$

where M_1 and M_2, V_1 and V_2, are the masses and volumes in the lower and upper regions respectively. The nomenclature section should be consulted for the remaining definition of the terms in Equation (4).

The rate of heat transfer from the primary steam to the secondary system, represented as Q_1 in Equation (4), can be calculated from

$$Q_1 = UA_{sg}(T_s - T_{s1})\frac{Z_{cond}}{L} \qquad (5)$$

where

$$U = \frac{1}{\frac{1}{h_p} + \frac{1}{h_s} + \frac{A_{sg}\ln\left(\frac{d_o}{d_i}\right)}{2\pi k_s L N}} \qquad (6)$$

Thus, once the air volume in the RCS has been compressed to a value less than that of the steam generator active tube volume, primary steam condensation is computed using Equation (5). The heat rate between the primary air and secondary system, Q_2, was calculated in a similar manner except h_p is replaced by the constant coefficient of 5.0 Btu/hr-ft^2-°F (0.88 W/m^2-°C) and Z_{cond} is replaced by the quantity (L-Z_{cond}).

3.2 PWR System Response Using the Steam Generators as an Alternate Means of Decay Heat Removal

To apply the preceding methodology to a PWR, the generic Combustion Engineering 2700 MW(t) Class plant was used with the initial conditions described in Table 1.

It is noted that use of the decay heat at 24 hours is considered an upper limit. Mid-loop operation would not be expected to be initiated any earlier than 48 hours following shutdown. Furthermore, as previously discussed, the peak RCS pressure achieved following a loss of the RHRS is relatively insensitive to the decay heat power level. Decay power affects the timing of the peak RCS pressure while the lower decay heat power levels delay the initiation of boiling and hence simply delay the time the peak pressure is achieved.

A prerequisite for the success of heat removal through the steam generators is that a secondary vent path be opened to maintain the secondary pressure near 14.7 psia (0.101 MPa) once secondary boiling occurs. The implication of this assumption is that there is a need for a vent path to maintain a low secondary pressure and temperature should boiling initiate. Also, after the initiation of boiling auxiliary feedwater was also assumed to be added to the secondary side at the secondary boiling rate to prevent exposure of the steam generator tubes and loss of the heat sink.

Figures 115 and 116 present the results of the analyses using the RELAP5/MOD3 model discussed in Section 2. Figure 115 presents the RCS liquid temperature and total pressure, and secondary liquid temperature and pressure. Figure 115 shows that once the air volume, shown in Figure 116, has been sufficiently compressed to allow steam to enter the primary steam generator tubes, the secondary liquid begins heating up at about 2000 seconds following a loss of the RHRS. The secondary liquid temperature reaches saturation at about 3000 seconds (see Figure 116), and remains at saturation while the secondary pressure is maintained at atmospheric conditions due to the open secondary vent paths. Without this secondary vent path, once the secondary liquid reached saturation, the secondary temperature would continue to increase as boiling would then begin to pressurize the secondary side. Figure 116 shows that when boiling of the RCS liquid begins at about 1700 seconds, the increasing steam volume compresses the air volume until sufficient condensing surface is available to allow condensation to begin in the steam generator active tubes. Once approximately 4 ft (1.2 m) of steam generator primary tube length is exposed for condensation, the primary pressure stabilizes at about 45 psia (0.31 MPa).

A comparison of the RCS pressure response for the HBR-2 plant using this alternate method and the RELAP5/MOD3 code is given in Figure 117. The close agreement between both methods provides an alternate check of the RELAP5/MOD3 results.

The Reference 3 study also investigated the impact of changes in primary condensation heat transfer coefficient in the steam generator active tube region. The results of this study are shown in Figure 118 and show that the primary peak pressure is also insensitive to condensation coefficients in the range of 150 to 1000 Btu/hr-ft^2-°F (26.4 to 176.1 W/m^2-K). This insensitivity is due to the fact that the bulk of the RCS pressurization is attributable to the need to compress the air to a volume less than that of the steam generator active tube region. The additional changes in air volume in the entrance to the active steam generator tube region to accommodate changes in condensation coefficient have a small impact on RCS pressure.

It is important to note that the results of these analyses are limited to mid-loop operation with RCS liquid levels near and below the centerline elevation of the hot leg. As noted in Reference 4, under these conditions the initiation of boiling in the RCS is not expected to cause the swell of liquid and two-phase fluid into the steam generators active tube regions that could then block the establishment of a condensing surface. This implies that if the loss of the RHRS occurs with liquid levels above the top elevation of the hot leg, the swell of liquid into the steam generators with air trapped above could impede the two-phase flow in this region causing RCS pressures to increase well above that calculated herein for the lower initial water level. With the swell of liquid into the steam generators, the condensation of pure steam with the condensate exiting countercurrent to the steam is no longer applicable. In this condition, the liquid-air content in the steam generators could impede the establishment of two-phase natural circulation causing pressures to increase well above the 40 to 45 psia (0.28 to 0.31 MPa) pressure range computed for mid-loop operation described in this section. As noted in Section 2, the RELAP5/MOD3 analysis with higher initial liquid levels resulted in peak pressures of approximately 95 psia (0.66 MPa).

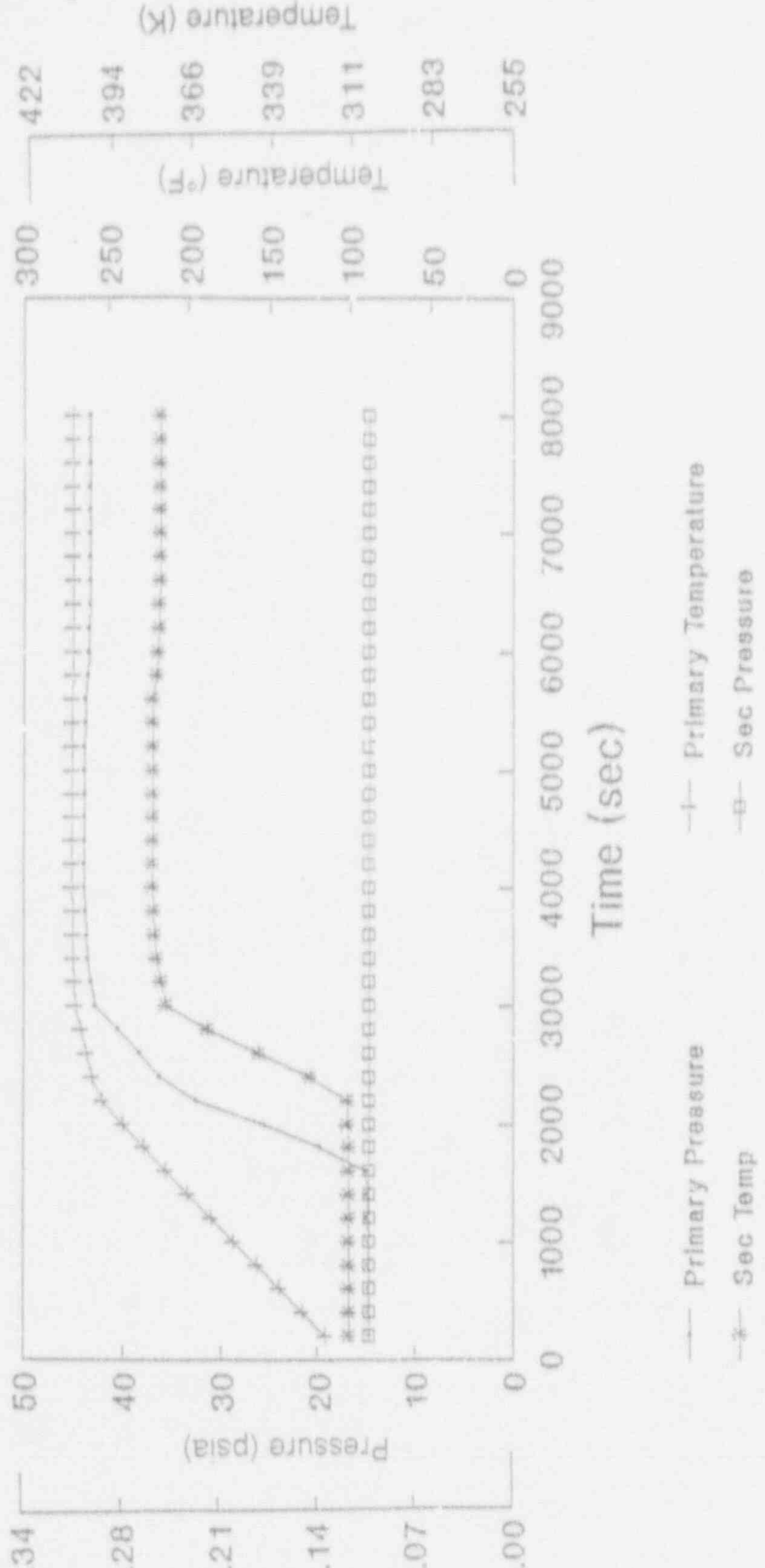

Figure 115. Primary and secondary pressure/temperature vs. time.

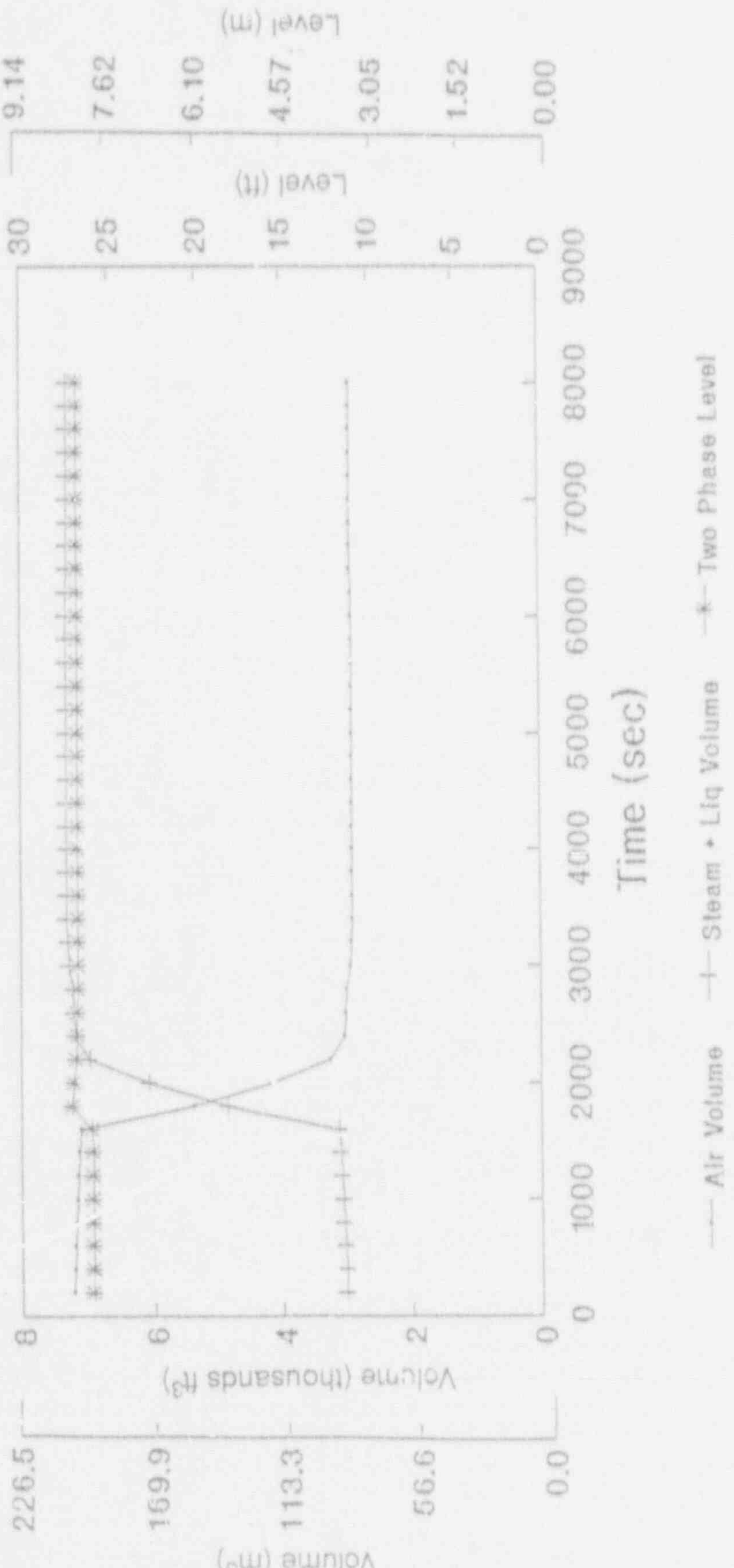

Figure 116. Air, steam/liquid volume and level vs. time.

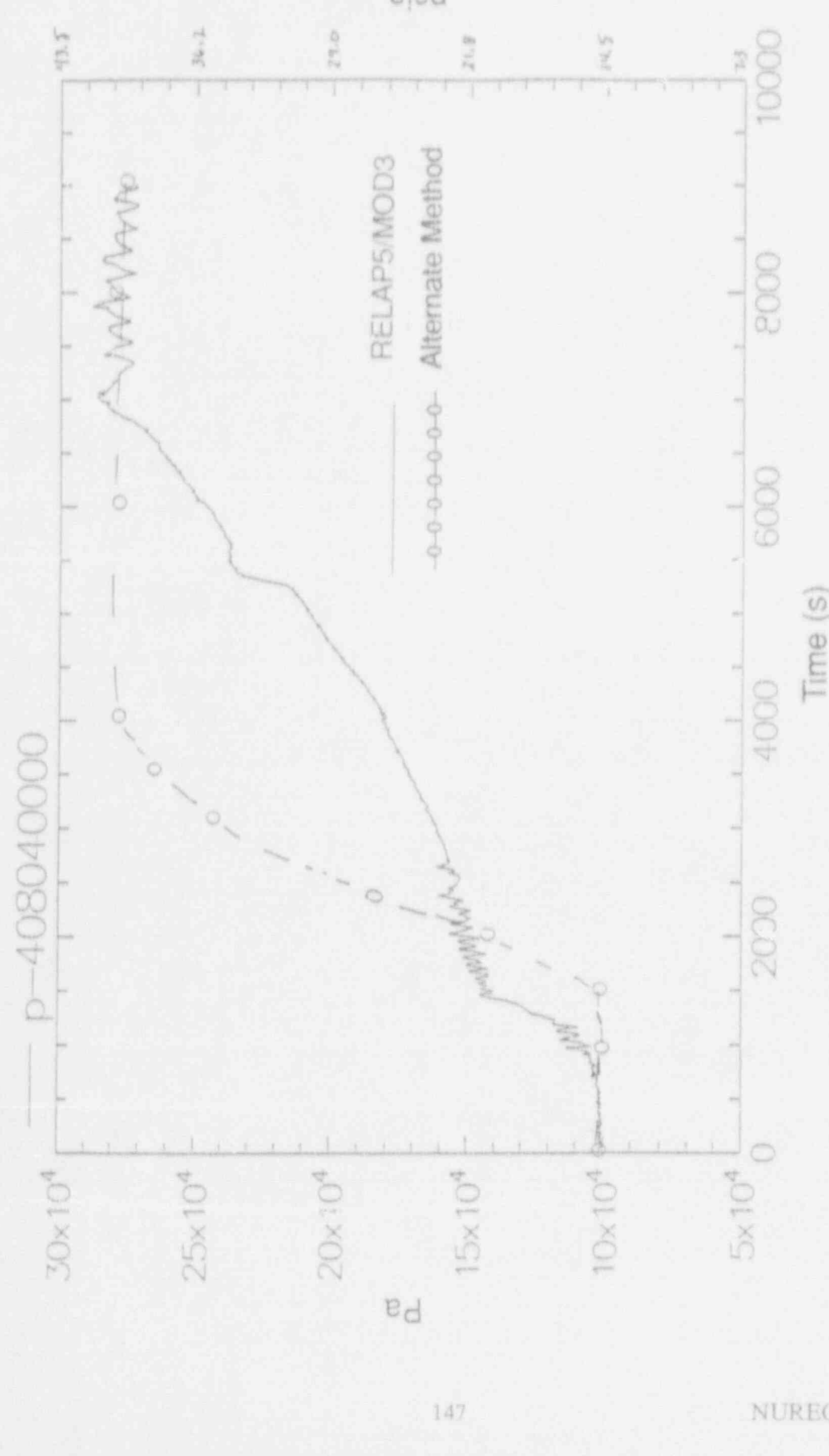

Figure 117. Steam generator primary pressure for Case 1.

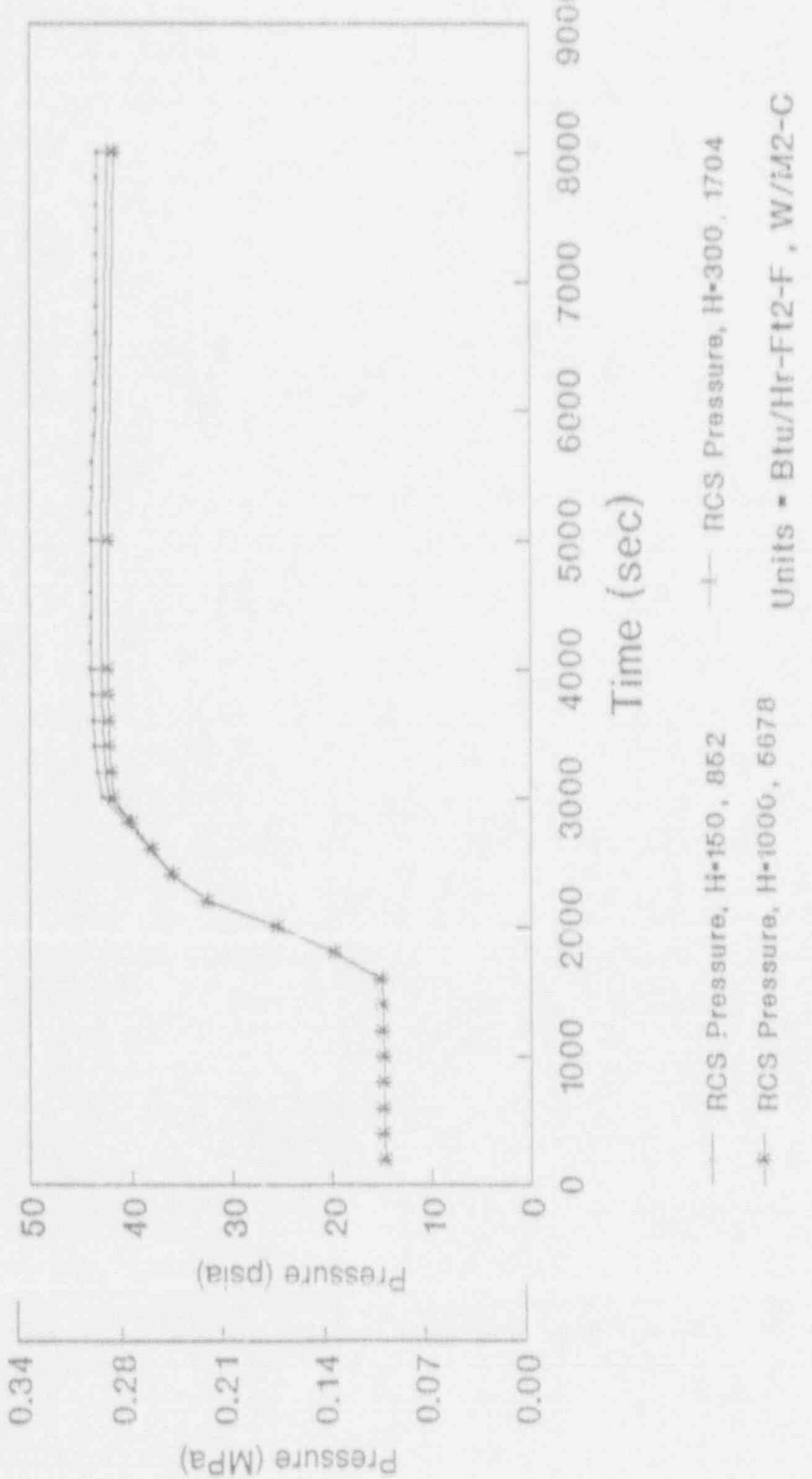

Figure 118. Sensitivity of RCS pressure to condensation coefficient.

4. REVIEW OF TEST DATA

As part of the evaluation of plant response to a loss of RHRS event, experimental test data regarding the heat transfer characteristics of a steam-air mixture in an inverted U-tube were reviewed. These data include the information contained in References 5 through 8 and include the following

- EPRI single tube test data investigating condensation of steam in the presence of air in an inverted U-tube.[5]

- PKL data simulating the loss of RHR following mid-loop operation with a noncondensible gas above the RCS liquid level.[6]

- Semiscale natural circulation tests when noncondensible gases are injected into the primary system.[7]

- The Kraftwerk Union (KWU) experimental investigation of condensation in an inverted U-tube during the injection of noncondensible gases at the entrance to a U-tube steam generator.[8]

The EPRI and PKL tests were based on low-pressure conditions. Pressures near atmospheric conditions were used for the EPRI tests while pressures of 90 psia (0.62 MPa) were achieved during the PKL tests. The KWU tests were performed at pressures near 1000 psia (6.89 MPa) under reflux boiling conditions. The importance of these data is that they all support the piston model characterization of the RCS fluid following the initiation of boiling in a system containing air or other noncondensible gases. That is, once boiling initiates the steam acts to compress the air into the high points of the system. These data also demonstrated that once the noncondensible gas is compressed sufficiently to allow a condensing surface to develop in the steam generator tubes, the steam generator would contain two regions: (a) an active region or a small region at the entrance of the steam generator tubes that contained pure steam and where the condensation occurred, and (b) a passive region in the remainder of the steam generator that contained the noncondensible gas and where no condensation took place. In all cases, an active region developed which contained the bulk of the condensation and also controlled the peak pressure achieved during the tests. Based on these test data, the piston model developed as part of the alternate methods of Reference 5 was chosen to characterize RCS fluid behavior following a loss of the RHRS.

As noted in Section 1, the RELAP5/MOD3 results of the reflux boiling cases also displayed this piston steam-noncondensible gas behavior.

5. CONCLUSIONS

Analyses of the consequences of the loss of the RHRS for PWRs with U-tube and once-through steam generator designs has been completed using the RELAP5/MOD3 code. The analyses of the loss of the RHRS event included the following:

- Use of a steam generator to mitigate the consequences of a loss of the RHRS following mid-loop operation for plants utilizing U-tube or once-through steam generators designs
- Consequences of failure of a hot leg nozzle dam during reflux boiling following mid-loop operation
- Consequences of a failure of all of the temporary thimble seals during reflux boiling from mid-loop operation
- Consequences of loss of the RHRS from mid-loop operation with an SI line open to the containment
- Consequences of a loss of the RHRS during refilling of the refueling pool cavity and the reactor vessel internals in place
- Consequences of long-term borated water addition to the vessel following a loss of RHRS during boiling in the RCS and when the steam generators are unavailable for decay heat removal.

Based on RELAP5/MOD3 analyses of the above events, the following conclusions are made:

- In the event there are no RCS temporary boundaries in use and the RCS is closed, the steam generators could be used as an alternate means of heat removal in the event the RHRS fails during reduced inventory conditions.
- Peak RCS pressure is insensitive to the time of loss of RHRS based on the results from the U-tube steam generator analyses for one and seven days after shutdown. Resulting peak RCS pressures near the design limit of the nozzle dams suggest that a better understanding of nozzle dam and all other temporary RCS boundary failures is needed before the steam generators can be used successfully as an alternate means of decay heat removal. Peak pressures for plants with once-through designs were calculated to be in the range 50 to 60 psia (0.34 to 0.41 MPa).
- For plants with U-tube steam generator designs, peak RCS pressures are sensitive to the initial RCS water level where levels above the top elevation of the hot leg result in pressures approaching 95 psia (0.66 MPa), well above the design conditions for temporary boundaries in the RCS during refueling operations.
- While peak pressures can approach the design pressure of the nozzle dams during reflux boiling, the resulting oscillatory RCS pressure behavior could initiate eventual loss of nozzle dam integrity. The pressure forces on the nozzle dams may be insufficient to cause immediate failure, however, the possible accumulated movement of the dams subjected to an oscillatory pressure may, with time, result in failure. Peak pressure alone therefore may be insufficient to judge success of steam generator reflux boiling to mitigate the consequences of the loss of the RHRS.
- The consequences of a hot leg nozzle dam failure with a steam generator manway removed shows there is little time to initiate the injection of water into the RCS to keep the core cooled. With the RHRS lost one day following shutdown, core uncovery ensues within 15 minutes following failure of the nozzle dam while uncovery occurs within one hour of nozzle dam failure if the RHRS is lost seven days after shutdown.
- Following a loss of the RHRS at one day following shutdown, a bounding analysis of the failure of all of the temporary thimble

seals resulted in core uncovery in approximately 15 minutes following opening of the break. This result was based on an arbitrarily assumed failure pressure of 35 psia (0.24 MPa), which was achieved about 1.5 hours following loss of the RHRS.

- The loss of the RHRS at one day with an open SI line to the containment resulted in core uncovery in about one hour. Fluid expulsion into the containment occurred at about 30 to 45 minutes following initiation of the event.

- The analysis of the loss of the RHRS with the reactor vessel internals in place suggest that, without intervention or recovery, a partial uncovery of the core may occur for those Westinghouse-designed plants where the upper support plate flow holes have been sealed. With the vessel internals in place, the restricted flow through the upper support plate inhibits communication of the refueling pool cavity water with the vessel upper plenum and core regions, which could result in the long-term uncovery of the top of the core in the event boiling occurs. While only the top portion of the core is uncovered very late in the event, the potential for core uncovery strongly supports the need for procedures and alternate strategies to prevent boiling under these circumstances.

- Analyses of the loss of the RHRS with the reactor vessel internals in place for those Westinghouse plants with the restricted flow through the upper support plate also demonstrated that the refueling pool cavity water temperature is not a good indication of the temperature in the core. Because the vessel coolant cannot readily circulate with the pool water above, pool cavity water temperature should not be used to infer time to boiling in the vessel.

- Should boiling in the RCS occur, the addition of borated water into the reactor vessel to prevent core uncovery could result in the precipitation of the boric acid in the RHRS lines preventing its further use for decay heat removal. Precipitation following the restoration of the RHRS after four or more hours of boiling could occur when the refueling water storage tank is used as a source of cooling, while precipitation as early as two hours into the event could occur if a high concentrated source of water is used such as the boric acid storage tank.

From the analyses presented in this report, it is important to note that because the system response to a loss of RHRS event and the availability of equipment to deal with the many consequences are highly plant dependent, analyses should be conducted on a plant-specific basis to properly evaluate the effectiveness of the proposed strategies and construct procedures to effectively deal with a loss of RHRS event. The analyses contained in this report should therefore be used only as guidance to identify those areas where further, more detailed, plant-specific analyses are needed to quantify the magnitude and timing of the key behaviors affecting a loss of the RHRS event.

Based on the analyses of the loss of RHRS events, the consequences of such an accident suggest that the use of the steam generators as an alternate means of decay heat removal, from mid-loop operation, may not be assured when temporary RCS boundaries are installed. In the unlikely event boiling occurs, the addition of borated water to the RCS can easily preclude the short-term potential for core uncovery, however the injection of borated water for extended periods of boiling presents a mechanism that could compromise decay heat removal and core coolability during the long term. More importantly, these analyses strongly suggest that specific procedures and strategies, in addition to careful attention to the instrumentation needed to monitor the accident progression, are needed to successfully deal with the range of potential consequences associated with a loss of the RHRS event.

6. REFERENCES

1. U.S. Nuclear Regulatory Commission, *Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990*, NUREG-1410, June 1990.

2. K. E. Carlson et al., *RELAP5/MOD3 Code Manual (4 Volumes)*, NUREG/CR-5535, EGG-2596, Draft, (available from EG&G Idaho, Inc., P.O. Box 1625, Idaho Falls, ID, 83415-2404) June 1990.

3. L. W. Ward et al., *Evaluation of the Loss of Residual Heat Removal Systems In Pressurized Water Reactors*, EGG-EAST-9853, February 1991.

4. C. D. Fletcher et al., *Thermal-Hydraulic Processes Involved in Loss of Residual Heat Removal During Mid-Loop Operation*, EGG-EAST-9337, October 1990.

5. Q. Nguyen et al., *Experimental Data Report on Condensation in a Single Inverted U-Tube*, EPRI NP-3471, May 1984.

6. R. Mandl, "Failure of PWR-RHRS Under Cold Shutdown Conditions, Experimental Results From the PKL Test Facility," *18th Water Reactor Safety Research Information Meeting, Rockville, MD, October 21-24, 1990.*

7. D. Hein, R. Rippel and P. Weiss, "The Distribution of Gas in a U-Tube Heat Exchanger and its Influence on the Condensation Process," *Proceedings of the International Heat Transfer Conference, 1982, Munich, Vol. 5*, pg. 467.

8. U.S. Nuclear Regulatory Commission, *Results of the Semiscale MOD-2A Natural Circulation Experiments*, NUREG/CR-2335, EGG-2200, September 1982.

Appendix A

An Alternate Methodology for Performing Loss of the RHRS Analyses

Appendix A

An Alternate Methodology for Performing Loss of the RHRS Analyses

A-1. DESCRIPTION

This Appendix presents a description of the alternate methodology used to perform the preliminary assessments of the consequences of a loss of the RHRS and to also check some of the RELAP5/MOD3 results. To assess RCS and secondary behavior following a loss of the RHRS, the methods of Reference 3 were employed. In this model, the primary system is represented with two regions; a lower liquid region that can be either subcooled or saturated (region 1), and an upper region containing a noncondensible gas (region 2). The noncondensible gas consists of air, which is treated as an ideal gas. In the development of this model, steam produced by boiling is assumed to act as a piston to compress the upper air region. The justification for this piston model, as opposed to a model that assumes the steam mixes with the upper air region, is provided in Section A-2.

Figure A-1 depicts the primary system with subcooled liquid below an upper air region to simulate mid-loop conditions. The methodology consists of a computation of the mass, energy, and pressure for the steam-air-liquid fluid composition in the RCS. To derive the time rate of change of pressure for the RCS with a steam-air-liquid composition, the two-region model depicted in Figure A-2 is used.

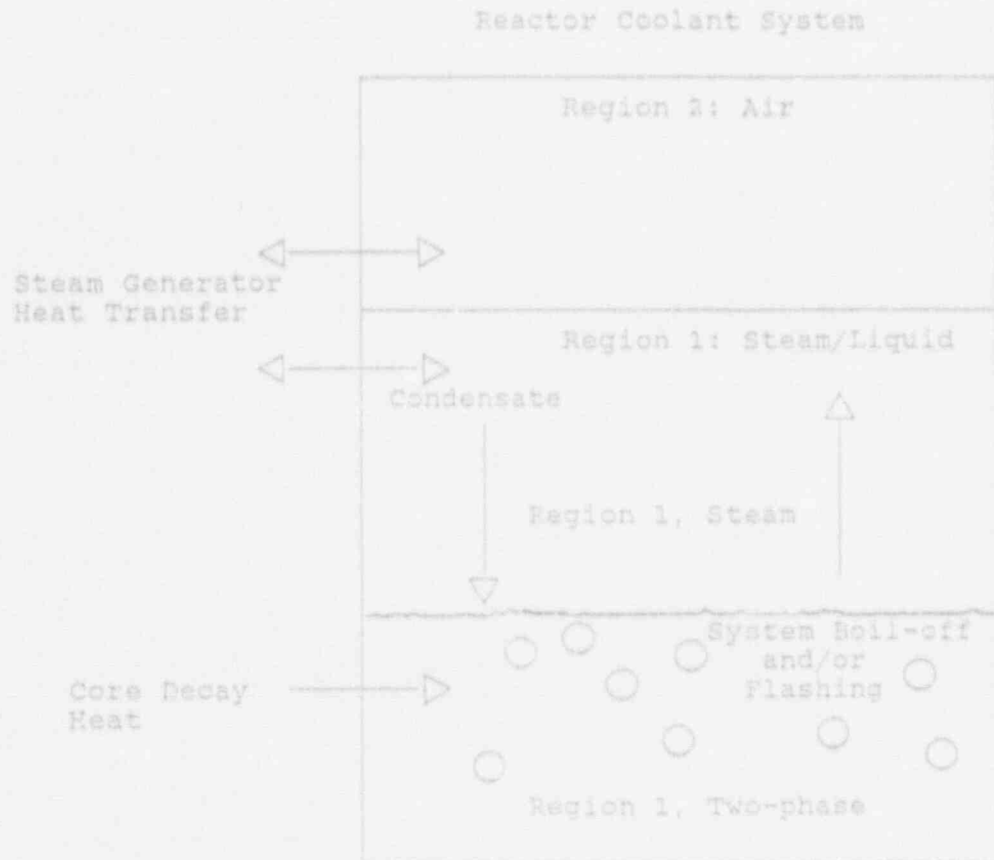

Figure A-1. Reactor coolant system two-region model.

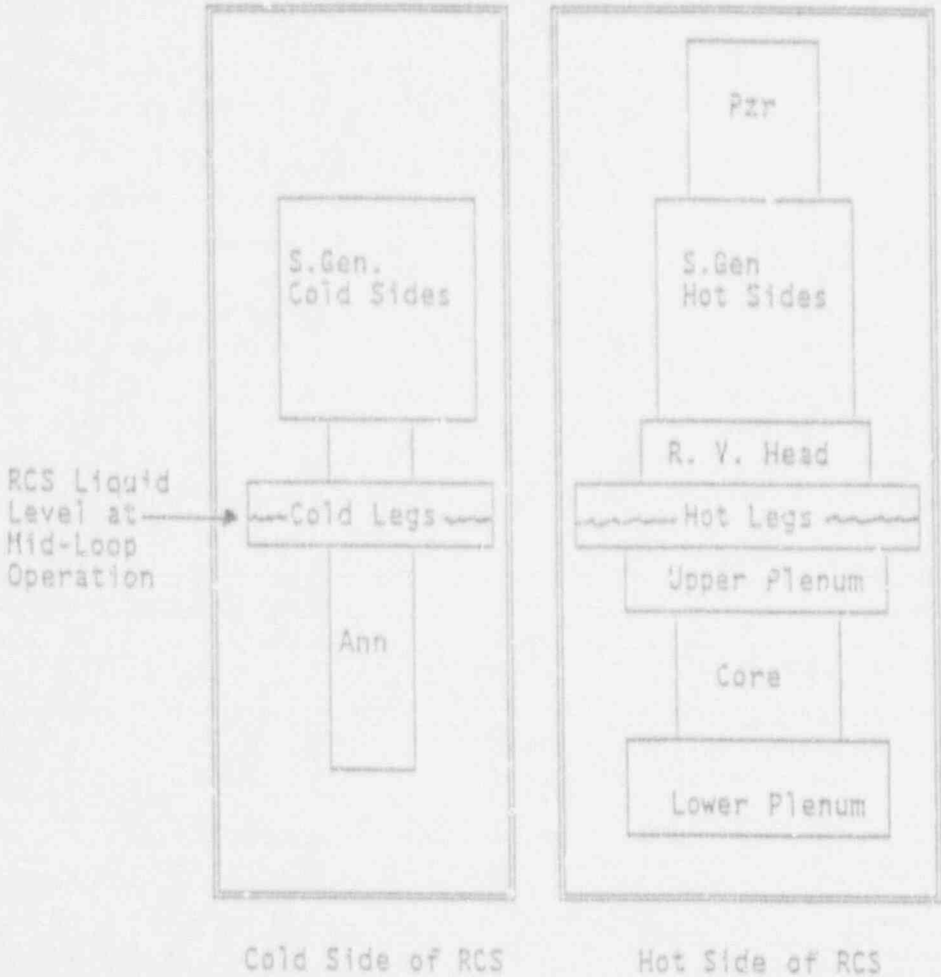

Figure A-2. Primary system representation for determining the liquid distribution in the RCS.

For the lower region containing liquid or a steam-liquid mixture, from continuity (see the nomenclature for definition of terms)

$$\frac{d}{dt}M_1 = \sum_{k=1}^{K} w_k \qquad \text{(A-1)}$$

The energy equation for this lower region 1 is defined by

$$\frac{d}{dt}U_1 = \sum_{k=1}^{K} w_k h_k + Q_1 - \frac{1}{J}P\frac{dV_1}{dt} \qquad \text{(A-2)}$$

Using the definition of specific volume where

$$M_1 v_1 = V_1 \qquad \text{(A-3)}$$

with

$$v_1 = v_1(h_1, P) \qquad \text{(A-4)}$$

and taking the derivative of Equation (A-3), the time rate of change in volume of region 1 is

$$\frac{d}{dt}V_1 = v_1\frac{dM_1}{dt} + M_1\left(\frac{\partial v_1}{\partial h_1}\frac{dh_1}{dt} + \frac{\partial v_1}{\partial P}\frac{dP}{dt}\right) \qquad \text{(A-5)}$$

The specific enthalpy of region 1 can be expressed as

$$U_1 = M_1 h_1 - \frac{1}{J}PV_1 \qquad \text{(A-6)}$$

From the derivative of Equation (A-6) and from Equation (A-2), the time rate of change of the region 1 enthalpy can be expressed as

$$\frac{d}{dt}h_1 = \frac{1}{M_1}\left(\sum_{k=1}^{K} w_k h_k + Q_1 - h_1 \frac{dM_1}{dt} + \frac{1}{J}V_1\frac{dP}{dt}\right) \quad \text{(A-7)}$$

Equations (A-5) and (A-7) define the time rate of change of volume and enthalpy for the lower region 1.

For the upper region 2 containing the air, the continuity equation is

$$\frac{d}{dt}M_2 = \sum_{n=1}^{N} w_n \quad \text{(A-8)}$$

The energy equation for the air is given by

$$\frac{d}{dt}U_2 = \sum_{n=1}^{N} w_n c_p T_n + Q_2 - \frac{1}{J}P\frac{dV_2}{dt} \quad \text{(A-9)}$$

The internal energy for an ideal gas can be written as

$$U_2 = M_2 C_p T_2 - \frac{1}{J}M_2 R T_2 \quad \text{(A-10)}$$

where for an ideal gas

$$PV_2 = M_2 R T_2 \quad \text{(A-11)}$$

Taking the derivative of the Equation (A-10) and using Equation (A-9), the time rate of change in air temperature in region 2 becomes

$$\frac{d}{dt}T_2 = \left(\frac{1}{M_2 C_p}\right)\left[\left(\sum_{n=1}^{N} W_n C_p T_n + Q_2\right) - C_p T_2 \frac{dM_2}{dt} + \frac{1}{J}V_2\frac{dP}{dt}\right] \quad \text{(A-12)}$$

From the derivative the Equation A-11 for an ideal gas, the time rate of change in volume becomes

$$\frac{d}{dt}V_2 = \frac{V_2}{M_2}\frac{dM_2}{dt} + \frac{V_2}{T_2}\frac{dT_2}{dt} - \frac{V_2^2}{M_2 R T_2}\frac{dP}{dt} \quad \text{(A-13)}$$

From the definition of total volume

$$V_t = V_1 + V_2 \quad \text{(A-14)}$$

The derivative of the total volume, which is constant, is therefore

$$\frac{d}{dt}V_1 = -\frac{d}{dt}V_2 \quad \text{(A-15)}$$

Using Equations (A-5), (A-7), (A-12), (A-13), and (A-15), the time rate of change in pressure for the RCS can be expressed as

$$\frac{d}{dt}P = \frac{\left[h_1\sum_{k=1}^{K} w_k - \left(\sum_{k=1}^{K} w_k h_k + Q_1\right)\right]\frac{\partial v_1}{\partial h_1} - v_1\sum_{k=1}^{K} w_k}{M_1\frac{\partial v_1}{\partial P} + \frac{1}{J}\frac{\partial v_1}{\partial h_1}V_1 + \frac{1}{J}\frac{V_2^2}{M_2 T_2}\left(\frac{1}{C_p} - \frac{J}{R}\right)} + \frac{\left[C_p T_2\frac{dM_2}{dt} - \left(\sum_{n=1}^{N} w_n C_p T_n + Q_2\right)\right]\frac{V_2}{M_2 C_p T_2} - \frac{V_2}{M_2}\frac{dM_2}{dt}}{M_1\frac{\partial v_1}{\partial P} + \frac{1}{J}\frac{\partial v_1}{\partial h_1}V_1 + \frac{1}{J}\frac{V_2^2}{M_2 T_2}\left(\frac{1}{C_p} - \frac{J}{R}\right)} \quad (4)$$

In this derivation, the changes in mass defined by Equations (A-1) and (A-8) are zero because there is no flow of fluid into or out of regions 1 and 2. Likewise, the energy convection terms defined as $w_ℓh_ℓ$ for the lower region, and $w_a c_p T_a$ for the upper air region, and which appear throughout the above formulation, also have a zero value. These terms are included in the d rivation for completeness.

Once boiling begins, the RCS steam distribution in the two-phase region is computed using a drift-flux formulation where the steam release rate is computed ..ased on the void fraction at the two-phase surface. The core void distribution is also power shape dependent and is based on the methods described in Reference 3. Although not pertinent to the calculations presented in this Appendix, this level swell modeling approach was required for those scenarios where core uncovery was calculated to occur. Once the masses, energies, and pressure are determined from the methods described above, the liquid mass is distributed between the hot and cold sides of the RCS, as depicted in Figure A-2, by hydrostatically balancing the liquid level on the cold side with the steam-liquid mixture on the hot side of the system. A variable area versus height geometric model of the hot and cold sides of the RCS is used to determine the final liquid and two-phase mixture heights in the RCS.

As the lower liquid region heats up following a loss of the RHRS, the expanding liquid region slowly compresses the upper air region. Upon boiling, the steam further compresses the air until the volume of the air is decreased to that necessary for steam to enter the primary tubes of the steam generators. A condensation height is then computed with the condensation coefficient on the primary side of the stea.n generator based on Nusselt falling film theory.

The secondary sides of the steam generators were modeled with a lower liquid region, containing either subcooled or saturated liquid, and an upper steam region that could be saturated or superheated. The fluid levels on the shell and tube sides of the secondary are also balanced hydrostatically for the purpose of determining the proper distribution of liquid and two-phase in a manner similar to that used in computing the primary fluid distribution discussed above. A variable area control vo'ume description is also used for the secondary sides of the steam generators. Heat transfer from both the primary steam and primary air to the secondary was also included. The secondary safety relief and atmospheric dump valves are also modeled with provisions for also adding auxiliary feedwater to the secondary. An approach similar to that for the primary system described above, without the air component, was used to compute the change in mass, energy, and pressure on the steam generator secondary sides.

A-2. COMPARISONS OF THE METHODOLOGY TO EXPERIMENTAL DATA

Experimental data on the condensation of steam in a single inverted U-tube, surrounded with a cooling water jacket, are presented in Reference 5. The test apparatus is shown in Figure A-3. The test section consists of a boiler, riser, U-bend, and the downside portion of the tube. Coolant is injected into the cooling jacket to simulate the secondary system. The boiler is partially filled with liquid and contains heating elements to produce steam for the riser portion of the U-tube. To establish the initial conditions for each test, the system was evacuated and air injected into the test section from a reservoir connected to the top of the boiler. Once the desired air content was achieved, the injection of air was terminated and the test was then initiated. Power was supplied to the boiler and the production of steam pressurized the system until the air was compressed sufficiently to allow steam to enter the bottom of the U-tube. Figures A-4 and A-5 present the comparison of the results using the above two-region methodology with Test 314 and demonstrates the ability of the model to adequately predict the test results. Note that the rise in secondary outlet temperature, marking the onset of primary steam condensation, occurred when the air volume was compressed to a value less than that of the volume of the U-tube region. Although the calculated results show the primary steam condensation and secondary heatup to initiate earlier than the data, the data and calculated response suggest that the steam acts as a piston, compressing the air into the upper portion of the test section and provides a basis for the methodology described above. Also note that the oscillations in secondary outlet temperature depicted in Figure A-5, result from the cyclical buildup and clearing of liquid in the riser portion of the condensing tube, but are of no significance to the prediction of the primary pressure for this test.

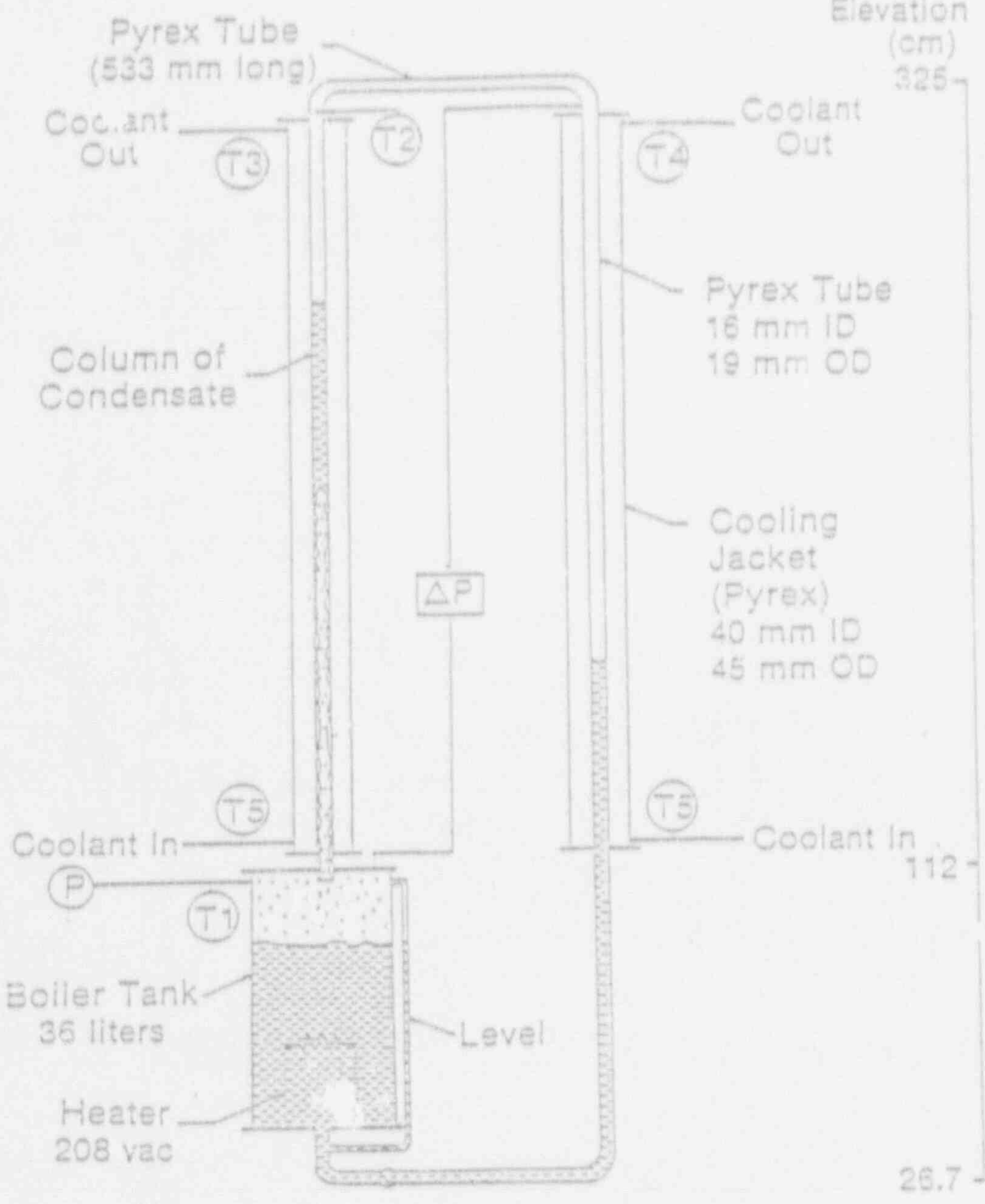

Figure A-3. Schematic of test apparatus.

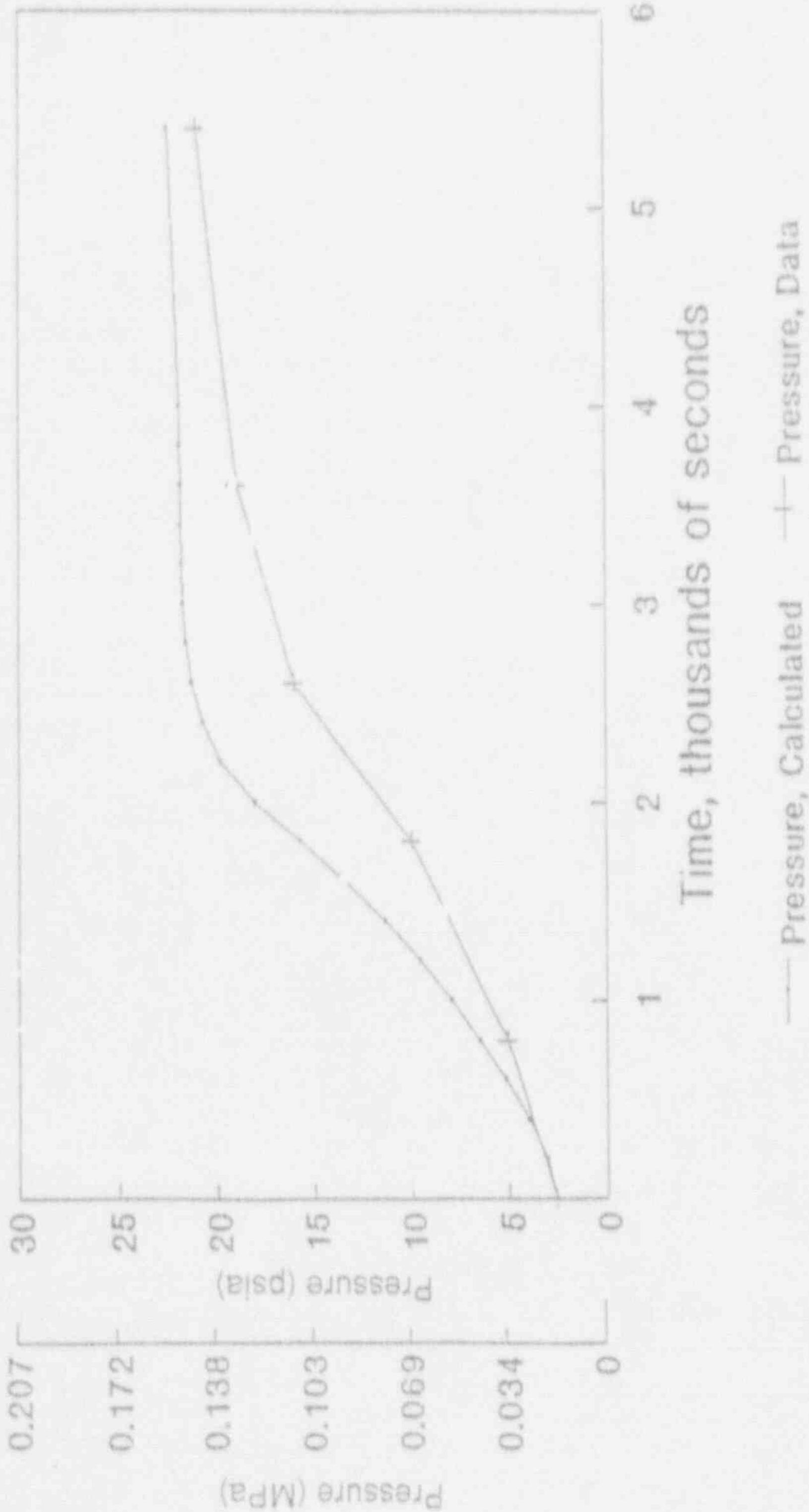

Figure A-4. UCLA Test 314 pressure vs. time.

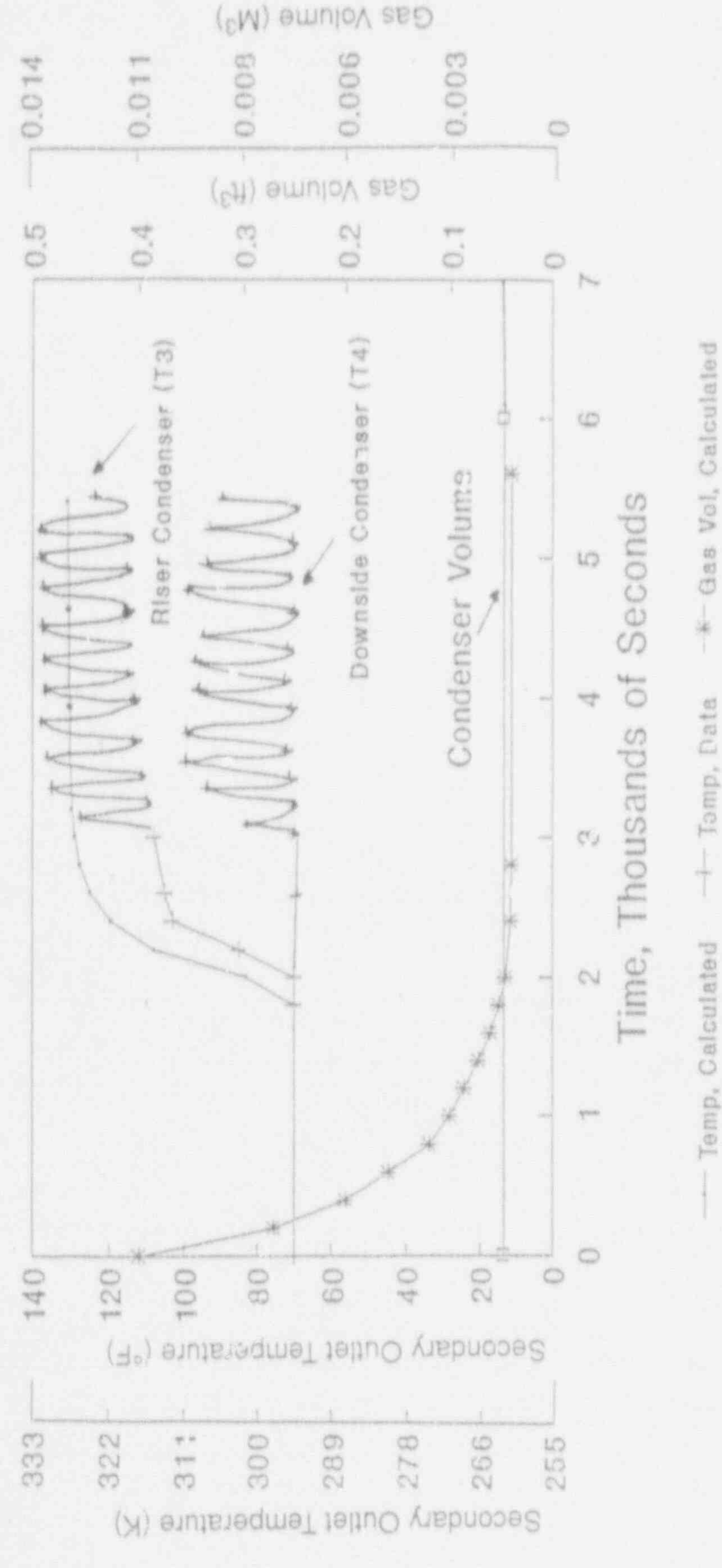

Figure A-5. UCLA Test 314 primary volume and secondary temperature vs. time.

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/CR-5820
EGG-2666

2. TITLE AND SUBTITLE

Consequences of the Loss of the Residual Heat Removal Systems in Pressurized Water Reactors

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1992

4. FIN OR GRANT NUMBER

L1696

5. AUTHOR(S)

L. W. Ward
W. Arcieri
C. Heath

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.)

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415

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 Systems Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

During shutdown at Vogtle Unit 1 on March 20, 1990, the loss of vital ac power and the Residual Heat Removal System (RHRS) focused much attention on the need to evaluate system performance following such an event in light water reactor (LWR) facilities. The RELAP5/MOD3 transient, non-equilibrium system performance code and an alternate methodology were used to evaluate this scenario and to investigate the accident consequences and identify key phenomenological and system behaviors characterizing these events. To investigate thermal hydraulic behavior following a loss of the RHRS, studies evaluated the use of the steam generators as an alternative for removing decay heat from mid-loop operation. Additional studies investigated the effects of decay power and changes in reactor coolant system (RCS) water level on system behavior when attempting to use the steam generators for heat removal. Under these alternate heat removal conditions, analyses identified the time to core uncovery in the event a nozzle dam or the temporary thimble tube seal fails. Other evaluations included assessing the impact of a loss of the RHRS with the reactor vessel internals in place and the upper head removed. For some plant designs, the flow restrictions through the upper internals may inhibit the downflow of water from the refuel pool cavity to the core during boiling, which could lead to the long-term uncovering of the fuel. Lastly, in the event boiling occurs in an open RCS, the impact of the addition of borated water for extended periods of time was investigated to identify the potential for boric ac. I precipitation.

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

Residual Heat Removal System
RELAP5/MOD3
PWR
Loss
Steam Generator

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

120555139531 1 1AN1RM
US NRC-OADM
DIV FOIA & PUBLICATIONS SVCS
TPS-PDR-NUREG
P-211
WASHINGTON DC 20555