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# SINGLE FAILURE EFFECTS ON SURGE LINE BREAK TRANSIENTS FOR THE VVER-440 REACTOR*

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## ABSTRACT

This paper describes the analysis of surge line break transients for the Soviet designed, water cooled, light water moderated, power reactors referred to as VVERs. These events represent an intermediate size loss of coolant accident (LOCA) for these plants and provide a severe challenge to the safety system design. The pressurizer surge line represents the largest diameter connection to the primary system and the break results in a relatively rapid blowdown of the primary system when compared to more conventional small break LOCAs (e,g., stuck open pressurizer relief valves).

The VVER unit selected for this analysis is designated as VVER-440 Model V213. This plant generates 440 Mwe and is of current interest since fifteen are now operating and additional units are in various stages of construction. In addition to a base case surge line break analysis, this paper also presents the results of several sensitivity studies related to single failures in various plant safety systems that have been included in the design to mitigate the effects of such a LOCA on the plant and fuel system performance. Examples of the safety systems selected for these sensitivity studies include the scram system, the accumulators, and the high pressure injection system.

## INTRODUCTION

Surge line break transients have been simulated for the Soviet designed VVER-440's using the RELAP5/MOD2 computer code (1). The VVER-440 unit selected for this analysis, designated as Model V213, has six primary coolant loops and generates 1375 Mwth. Existing V213 plants have two different pressurizer surge line configurations: certain plants have two surge lines connecting the pressurizer to the hot leg while other plants have the conventional, single surge line. The design selected for this analysis was the double surge line configuration and the transient analyzed was the double-ended rupture of one 20 cm (approximately 8 inch) surge line. The following sections of this paper provide a description of the RELAP5 computer code, the VVER-440 plant model, and the analysis results. Included in the results are a base case analysis and several sensitivity studies that examine the importance of certain safety systems included in the VVER-440 design.

## COMPUTER CODE DESCRIPTION

The RELAP5/MOD2 computer code has been used extensively for the analysis of pressurized water reactor LOCAs, abnormal occurrences, and anticipated transients without scram. The hydrodynamic model included in the code is a one-dimensional, twofluid model for flow of a two-phase steam water mixture. The basic field equations include the continuity, momentum, and energy equations solved for each of the two phases. Heat transfer is modelled with the one dimensional heat conduction equation that is coupled with the hydrodynamic calculation through a number of surface heat transfer regimes. Process models are available for choked flow, branching, and representation of pipe ruptures. Component models can be used to represent pumps, valves, and accumulators. Considerable flexibility is available for the representation of very complex systems.

## VVER SIMULATION MODEL

A detailed model of the VVER-440 plant has been developed for use with the RELAP5 code. This model, shown in Figure 1, consists of two main coolant loops with one loop representing the response of five VVER-440 loops and the second loop representing a single plant loop. Each loop contains models for steam generators, main coolant pumps, and the associated piping. Overall, the entire plant is represented by 84 control volumes and 84 flow junctions. Specific areas of the modelling important for the current analysis are discussed in the following paragraphs. Data for the development of these models has been obtained from Reference 2.

Pressure Vessel Model: As shown in Figure 1, the reactor pressure vessel is represented by seven control volumes with additional nodalization in the heated core region as discussed below. The downcomer was split, with individual connections to the lumped intact loop nozzles and the single loop nozzle connecting to the pressurizer. Individual control volumes are provided for the lumped nozzles, the intact nozzle, the lower plenum, and the outlet plenum.

Core Model: The core model developed for this analysis consisted of a single channel with five axial nodes. The heated core region was represented with three of these nodes; a chopped-cosine function was used to model the axial power distribution. Each axial core region is divided into 3 radial zones to model the fuel, gap, and clad regions of the fuel rods. The radial nodalization assigned four nodes to the fuel region, one node to the gap , and three nodes to the clad. Important results calculated from this model include the fuel center-line temperature and the inner and outer clad temperatures.

Neutronics Model: The neutronic response of the VVER-440 is not explicitly calculated in the current model. Core power is specified as a function of time with the power being held constant until the occurrence of a low core outlet pressure signal (1335 psia). Following this low pressure signal, designated as a first order safety signal or AZ-1 in VVER-440 design, the power is reduced to decay heat levels in thirty seconds. The assumption of constant power until the AZ-1 signal is considered conservative for this analysis since the surge line break is expected to cause voiding in the core region and some subsequent power reduction prior to the low pressure signal.

Primary System Piping: The VVER-440 plant has two loop seals in each primary coolant system loop (the vertical pipe sections located upstream and downstream of the steam generator). These loop seals are particularly important for LOCA analysis since natural circulation flow through the loops can be impeded by the accumulation of steam in the seals. Without natural circulation flow, the water level in the core will quickly drop to the level of the inlet and outlet nozzles; further reductions in water level could be expected on a longer term due to boiling in the core region. In addition, the accumulation of water in the seals can have an important effect on the steam generator heat transfer. Thus, considerable detail has been included in the modelling of the hot and cold leg piping in each loop. The control volumes designated as 102 and 202 in Figure 1 have each been subdivided into 17 nodes.

Steam Generator Model: The VVER-440 steam generators have horizontal once through steam generator tubes. The RELAP5 model represented the primary side with ten control volumes and ten heat slabs that transferred heat to the secondary system. Each of the heat slabs was sub-divided into 3 radial regions to allow the calculation of a temperature distribution through the walls of the steam generator tubes. This nodalization is considered adequate for the tube wall thickness of 0.056 inches.

Main Coolant Pump Model: The reactor coolant pumps were modelled with the homologous curves built into the RELAP5 code. Rated pump data and other pump parameters were obtained from Reference 2.

Secondary System Model: Each secondary steam system was represented by a single control volume that was sub-divided into three regions. All heat from the primary side was transferred to the middle, two phase region; feedwater flow entered the bottom region that was filled with liquid, subcooled by 58°F. Turbine protection is provided by fast acting isolation valves located at the end of the steam line. These valves have been modelled to close on a high steam pressure signal of 748 psia and a low pressure signal of 570 psia. The closing time of the valves is 2.5 seconds. Since there is a possibility of steam line isolation during the transients, the steam generator safety and relief valves were also modelled. The safety valves were set to open at 824 psia with a flow rate of 153 lbm/sec; the relief valves open at 769 psia and the flow rate is 122 lbm/sec.

Passive Safety Injection (Accumulators): Model V213 of the VVER-440 design includes four accumulators that provide a large source of coolant to the primary system following various LOCAs. These accumulators are arranged such that two deliver water to the downcomer region and two deliver water to the outlet plenum. The initial operating pressure of the accumulators is 798 psia.

Active Emergency Core Cooling System Models: This VVER-440 has three High Pressure Injection (HPI) Systems and three Low Pressure Injection (LPI) Systems. Each HPI system is connected to the cold leg piping; in the RELAP5 model, two systems are connected to the lumped, intact loop and one system is connected to the ruptured loop. The HPI systems are activated by a low core outlet pressure signal (1580. psia) with flow initiated following a thirty second time delay. The head/flow characteristic of the pumps is represented with data from Reference 2. The maximum flow rate per pump is 70. lbm/sec. at atmospheric conditions.

Each LPI system has a high volume, low shutoff head pump. The maximum flow of 292 lbm/sec. is reached at atmospheric conditions and the shutoff head is approximately 100 psia. Thus, this system is only effective in the long term analysis of primary system blowdown transients. Two LPI pumps deliver water to two accumulators and thus, to the downcomer and outlet plenum. The third LPI pump injects to both the hot and cold leg piping of a loop not connected to the pressurizer.

## ANALYSIS RESULTS

The surge line break transient was initiated at time zero by the instantaneous rupture of one 20 cm. diameter surge line. The rupture was assumed to be double-ended such that the primary system blows down through one side and the pressurizer blows down through the other side. Coolant flow between the primary system and the pressurizer is maintained through the intact surge line. However, this flow is not sufficient to control system pressure and both the primary system and the pressurizer rapidly depressurize. Outlet plenum pressure and pressurizer pressure (Figure 2) rapidly decrease during the initial 15 seconds of the transient. The rate of pressure decrease is reduced as saturated conditions are reached in the primary and significant voiding occurs in the outlet plenum (Figure 3). However, the initial depressurization is sufficient to trip the HPI (at 1580 psia) and injection flow is initiated, following a 30 second time delay, at approximately 32 seconds (Figure 4). As discussed previously, the VVER-440 RELAP5 model includes three HPI pumps with two pumps injecting to the intact loops and one pump injecting to the ruptured loop.

The AZ-1 scram signal is also generated due to low core outlet pressure and a scram is initiated at approximately two seconds after the rupture. As discussed previously, neutron kinetics have not been explicitly modelled; however, the effect of the reactor scram has been represented by reducing the reactor power to decay heat levels over a thirty second time period. Thus, as shown in Figure 5, the core inlet and outlet temperatures approach each other as the power level decreases. At approximately thirty seconds, the inlet and outlet temperatures are maintaining the same decreasing trend due to the cooling provided by the addition of HPI water. However, at 57 seconds, the primary system pressure has decreased below the pressure in the accumulators (798 psia) and accumulator flow is initiated to both the downcomer and the outlet plenum (Figure 6). The accumulator injection to the outlet plenum, a design feature of the V213 plant that differs from U.S. designs, allows the vessel to refill from the top in a counter-current flow regime and causes the core outlet temperature to decrease more than the core inlet temperature (Figure 5). In addition, the injection of relatively cold accumulator flow causes the outlet plenum pressure to decrease at a slightly greater rate (Figure 2) and collapses some of the void in the outlet plenum (Figure 3). The outlet plenum void fraction remains below 0.25 until approximately 145 seconds when the accumulators are empty. At this time, void in the outlet plenum increases as the water level decreases to the level of the hot and cold leg nozzles.

The initiation of accumulator flow also effects the void fraction in the core. As shown in Figure 7, the core void fraction increases during the initial 60 seconds due to a reduction in the core inlet flow (Figure 9) and the decreasing system pressure. However, soon after initiation of the accumulator flow, all voids in the core have been collapsed and core cooling is maintained for the remainder of the transient. The effects of the loop seals located in the hot and cold leg piping is demonstrated by the core inlet flow rate shown in Figur- 8. As voids are generated in the loop seals during the depressurization, the capability of the main coolant pumps to force flow to the core inlet is reduced; in addition, natural circulation flow is inhibited by the void blockage generated in the loop seals. Thus, the core inlet flow is rapidly reduced soon after the surge line ruptured and remains at a low value for the remainder of the transient.

The fuel center-line temperature and clad surface temperatures are shown in Figures 9 and 10. Both figures show temperature decreases in the early phase of the transient due to the reactor scram. These reduced temperatures are maintained by the addition of water from the HPI, accumulator, and low pressure injection systems. At approximately 170 seconds, the transient has been terminated following the initiation of the high flow, low pressure injection system. At this time, the primary system has blown down to near to atmospheric conditions and the break flow rates (Figure 11) have been reduced to less than the injection flow rates; no additional fuel temperature transients are anticipated.

Break flow rates from both the primary and pressurizer side are shown in Figures 11. These flow rates are subcooled and saturated liquid initially. They decrease as two phase conditions are generated during the early phase of the transient. Shortly after 60 seconds, the effect of the cooler injection flow from the HPI and accumulator systems is evident as the break flows increase due the density reduction. Finally, at approximately 165 seconds, the break flows are terminated as the primary system has depressurized to approximately atmospheric conditions. No fuel damage is anticipated during this transient.

## SENSITIVITY STUDIES

The effects of various single failures in the safety systems included in this VVER-440 design have been evaluated by performing several sensitivity studies on the base case RELAP5 model. The single failures considered included a delay in the scram, the failure of one HPI system, the delay of initiation of all HPI systems, and a failure in the accumulator system. Each of these analyses are discussed in the following paragraphs.

<u>Delayed Scram</u>: The base case results described previously have shown that the scram is very effective in reducing system pressure following the surge line break such that injection flow from the HPI and accumulator systems can enter the primary system. Thus, an interesting single failure that should be evaluated is the delay in initiation of the scram. Delay of scram initiation may occur due to failure of automatic scram with a subsequent manual operator scram. The evaluation of scram delay has been performed by repeating the base case analysis with the scram delayed by 2 minutes. The results, shown in Figures 12 through 14, indicate that the scram delay will cause primary system pressure to remain high and prevent the injection of HPI and accumulator water. Outlet plenum pressure (Figure 13), after an initial decrease due to the rupture, begins to increase due to the core heat and void generation (Figure 14) such that HPI and accumulator flow cannot activate. Following the scram at 120 seconds, system pressure rapidly decreases and injection flow is initiated. However, fuel and clad surface temperatures (Figures 12) have already reached unacceptable levels (i.e., exceeding the zirconium-water reaction threshold temperature of 1500°F). It should be noted that these results include the conservative assumption that power remains constant over the time period prior to scram. Some reduction in power level would be expected from the reactivity effects of core

void formation. However, the results indicate the importance of a timely scram initiation following the surge line break. Additional calculations assuming a one minute scram delay have shown that excessive fuel temperatures will not result during this transient.

Failure of One HPI System: The base case RELAP5 model was modified to remove the HPI system that provided injection to the ruptured loop. The results indicate that the injection flow from the accumulators provides a much greater contribution to core cooling than the HPI system. Thus, the failure of one HPI system does not have a significant effect on the fuel and clad surface temperature response.

HPI Initiation Delayed: The sensitivity described above has been extended to evaluate the delay of initiation of all three HPI systems by five minutes after the expected initiation time. This analysis showed that in the absence of the HPI pumps, the accumulators provided sufficient inventory to the primary system that core cooling was maintained and fuel temperatures remained low. Although the accumulators emptied at approximately 160 seconds, the depressurization of the primary system was sufficient to activate the low pressure injection system that provided additional flow to prevent a fuel temperature excursion. Since the transient is similar to the previous case and the fuel temperature transients are relatively benign, no graphical results are provided.

Accumulator Failure: The final sensitivity analysis performed was the failure of accumulator flow to either the downcomer or the outlet plenum. Both cases have been analyzed; however, the loss of accumulator flow to the outlet plenum is more severe due to the loss of the counter current flow from the outlet plenum to the core region. The results indicate that the fuel temperature and clad surface temperature continually decrease during the transient. Core voids formed during the depressurization are collapsed by the combination of HPI injection and injection from the operable accumulators to the downcomer.

## SUMMARY

The analysis of the VVER-440 Model V213 presented in this paper has shown that the plant safety system design provides adequate protection against intermediate size LOCAs such as the surge line break. The base case results indicate that no fuel or clad surface temperature excursions are experienced during the transient. The addition of cooling water from the HPI systems and the accumulators coupled with a timely scram are sufficient to collapse any core voids generated and provide adequate core cooling. Sensitivity studies have shown that the timely occurrence of a scram is particularly important in maintaining core cooling. However, failures in the injection systems (HPI and accumulators) can be compensated for by the performance of other systems.

## ACKNOWLEDGEMENTS

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1. Ransom, V.H. and Wagner R.J., RELAP5/MOD2 Code Manual Volumes 1 and 2, EGG-SAAM-6377, April 1984.

2. Central Research Institute for Physics, Specification For a VVER-440 Type NPP With Pressure Suppression System, IAEA Workshop Report, February 1988.

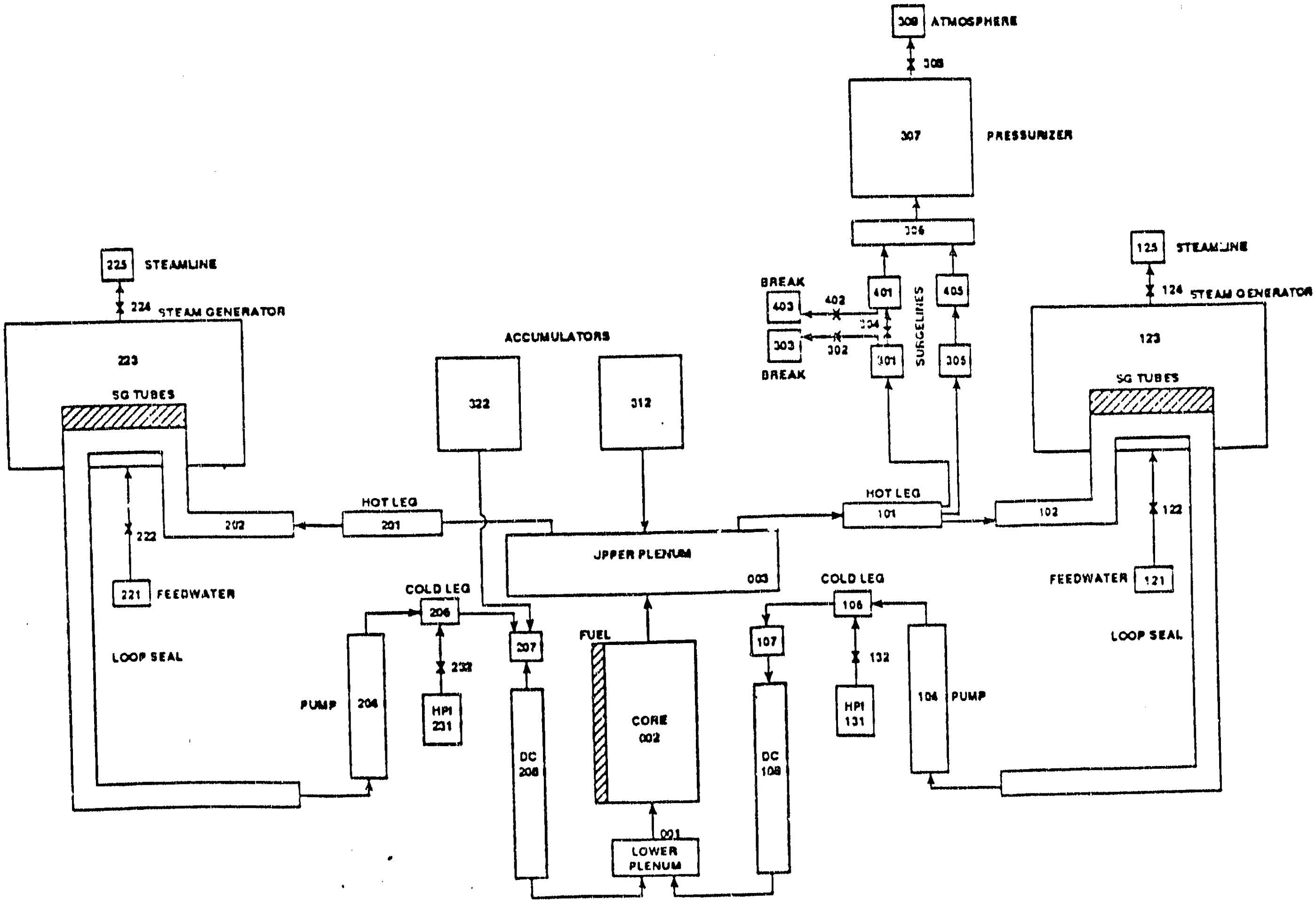
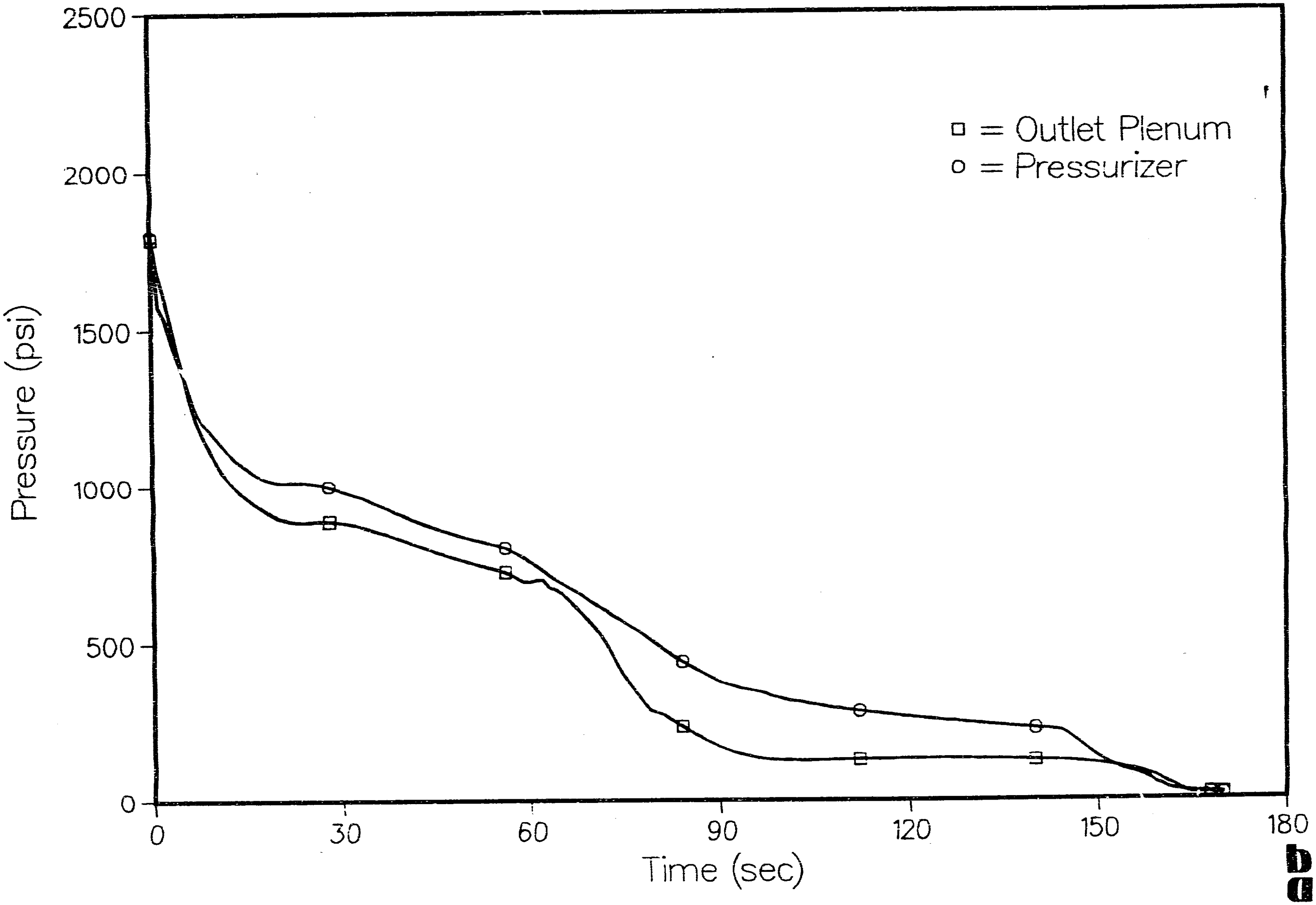


Figure 1 VVER-440 Model V213 Nodal Diagram

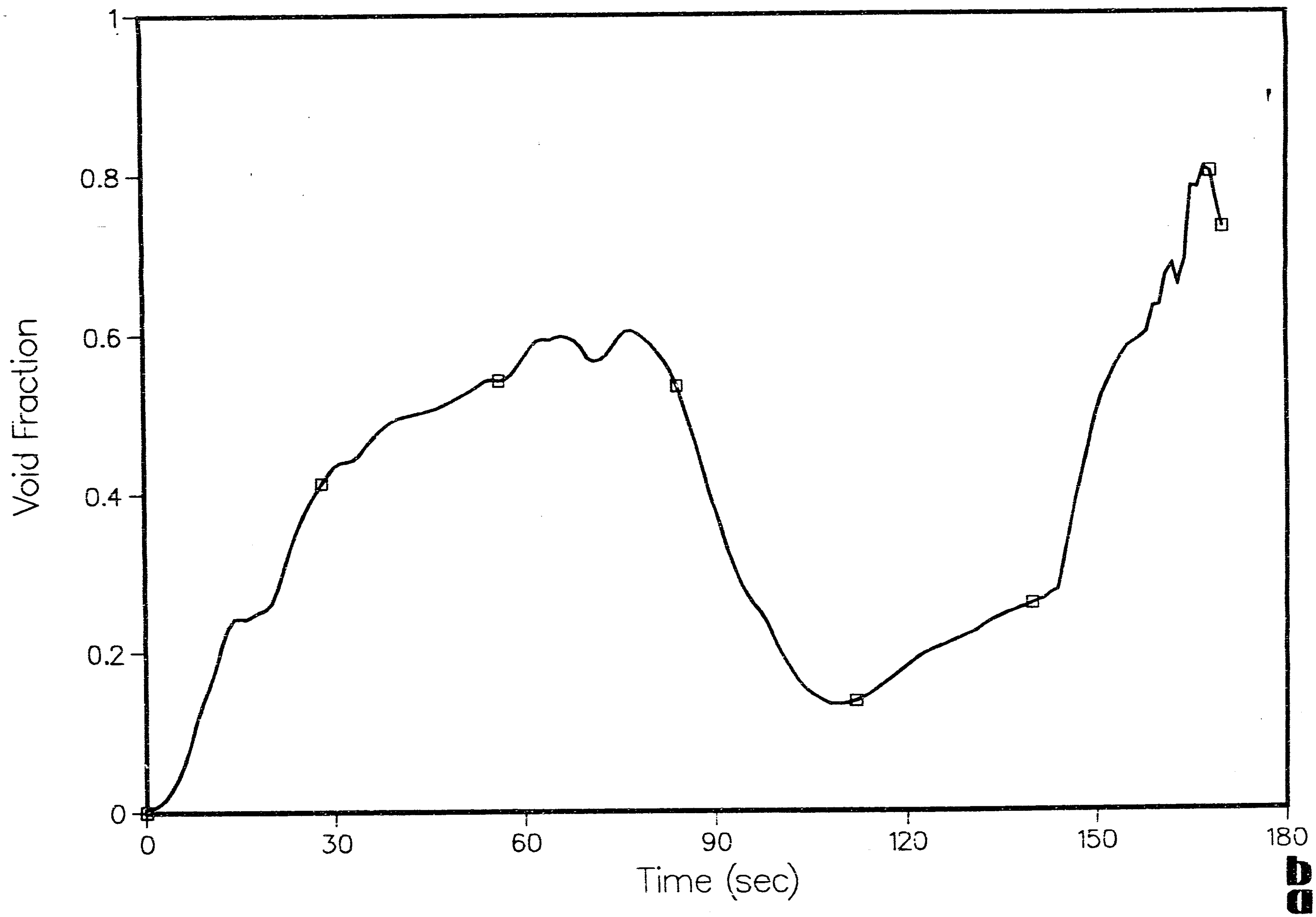
VVER–440 V213 Surge Line Break
Base Case

Figure 2: Primary Pressure



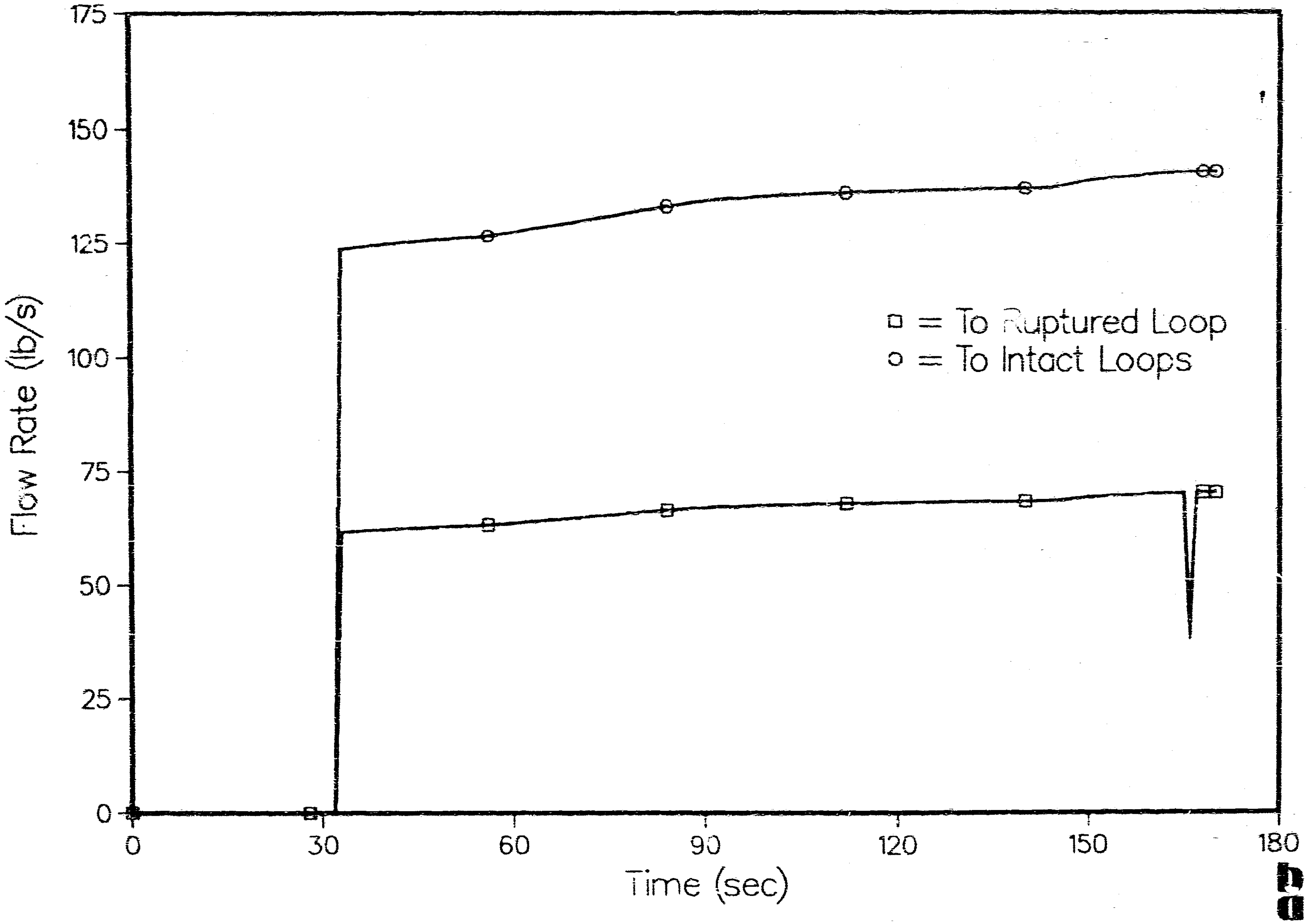
VVER–440 V213 Surge Line Break
Base Case

Figure 3: Outlet Plenum Void Fraction



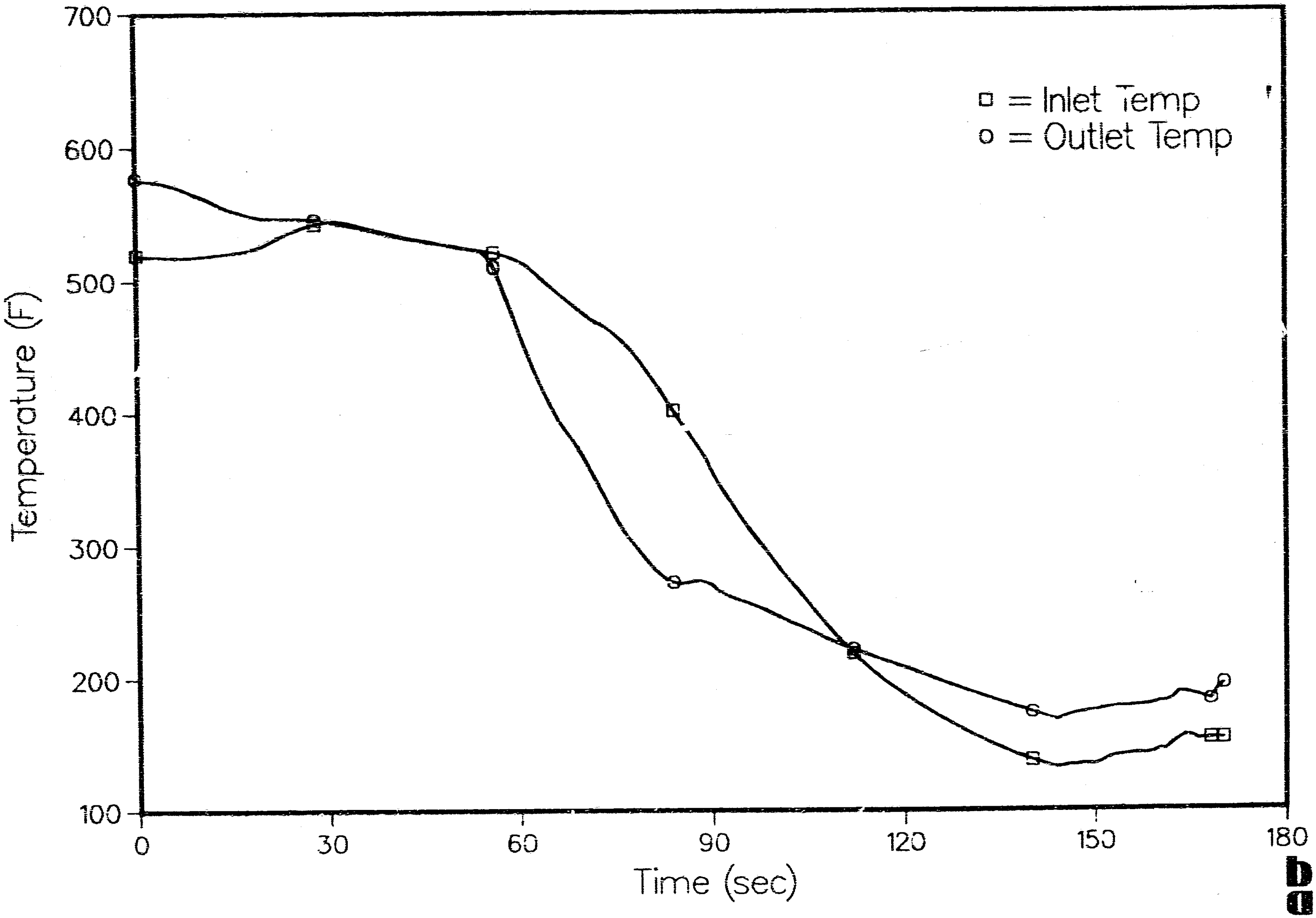
VVER–440 V213 Surge Line Break
Base Case

Figure 4: HPI Flow Rate



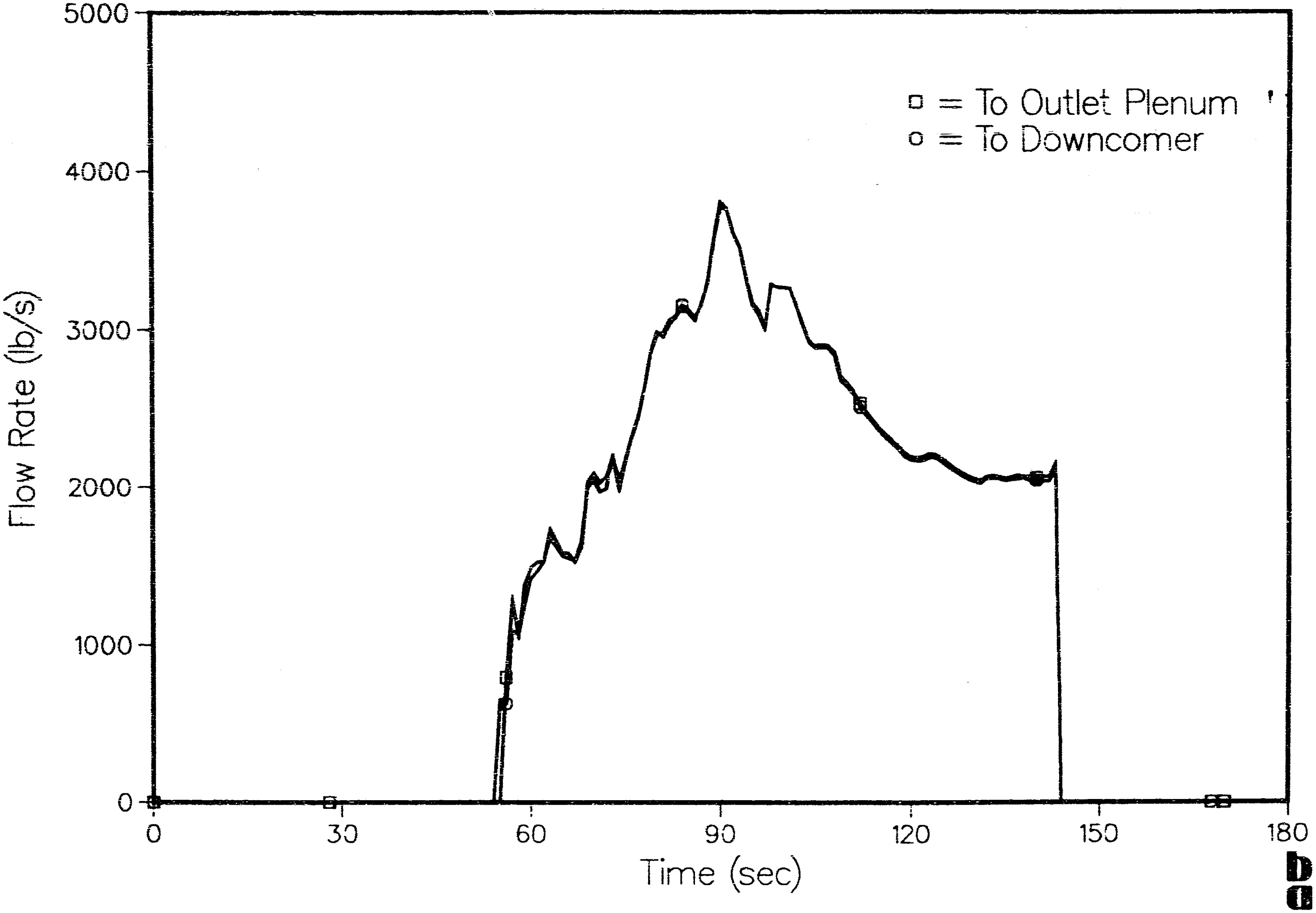
VVER–440 V213 Surge Line Break
Base Case

Figure 5: Core Inlet and Outlet Temperatures



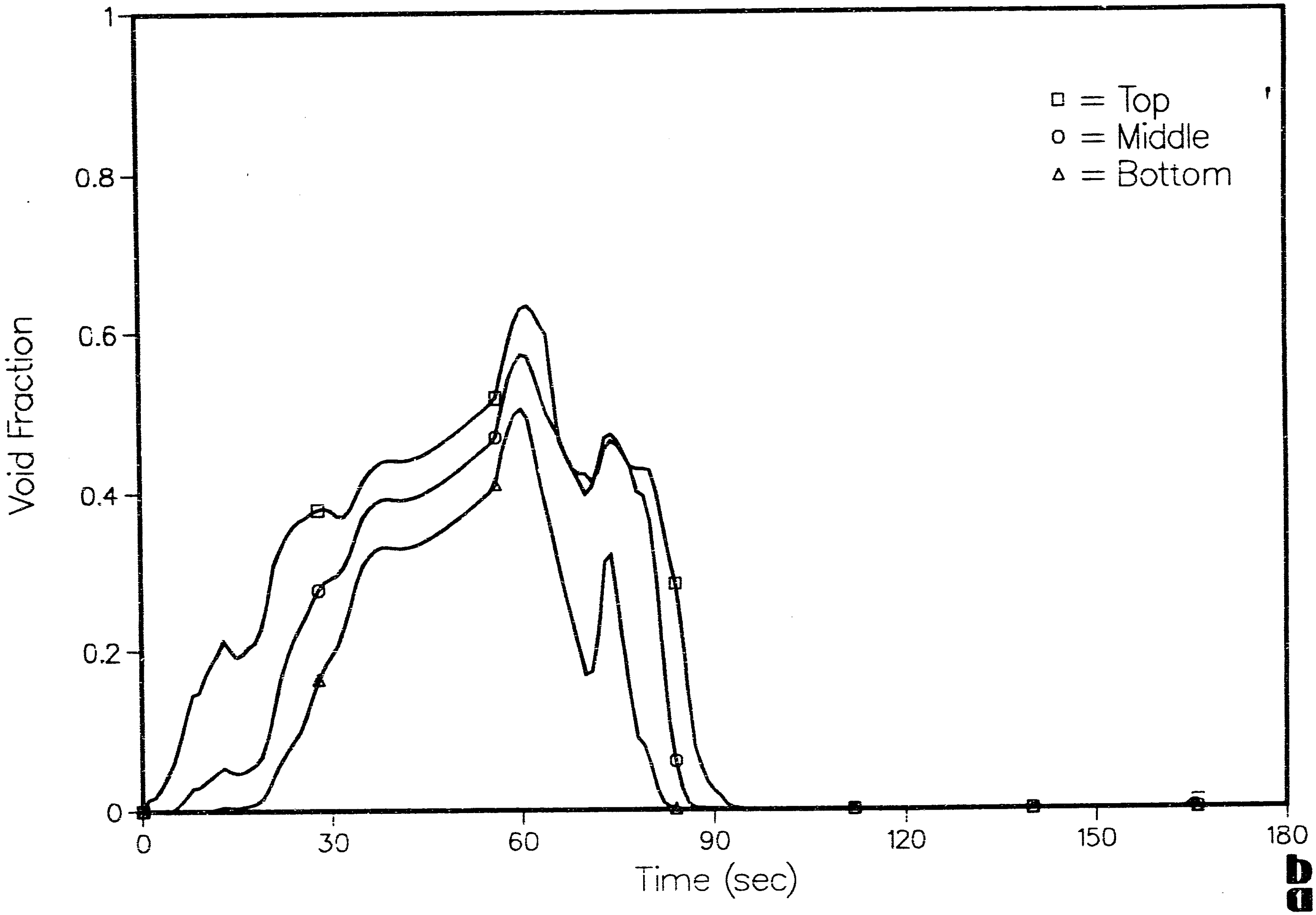
## VVER-440 V213 Surge Line Break
Base Case

Figure 6: Accumulator Flow Rate



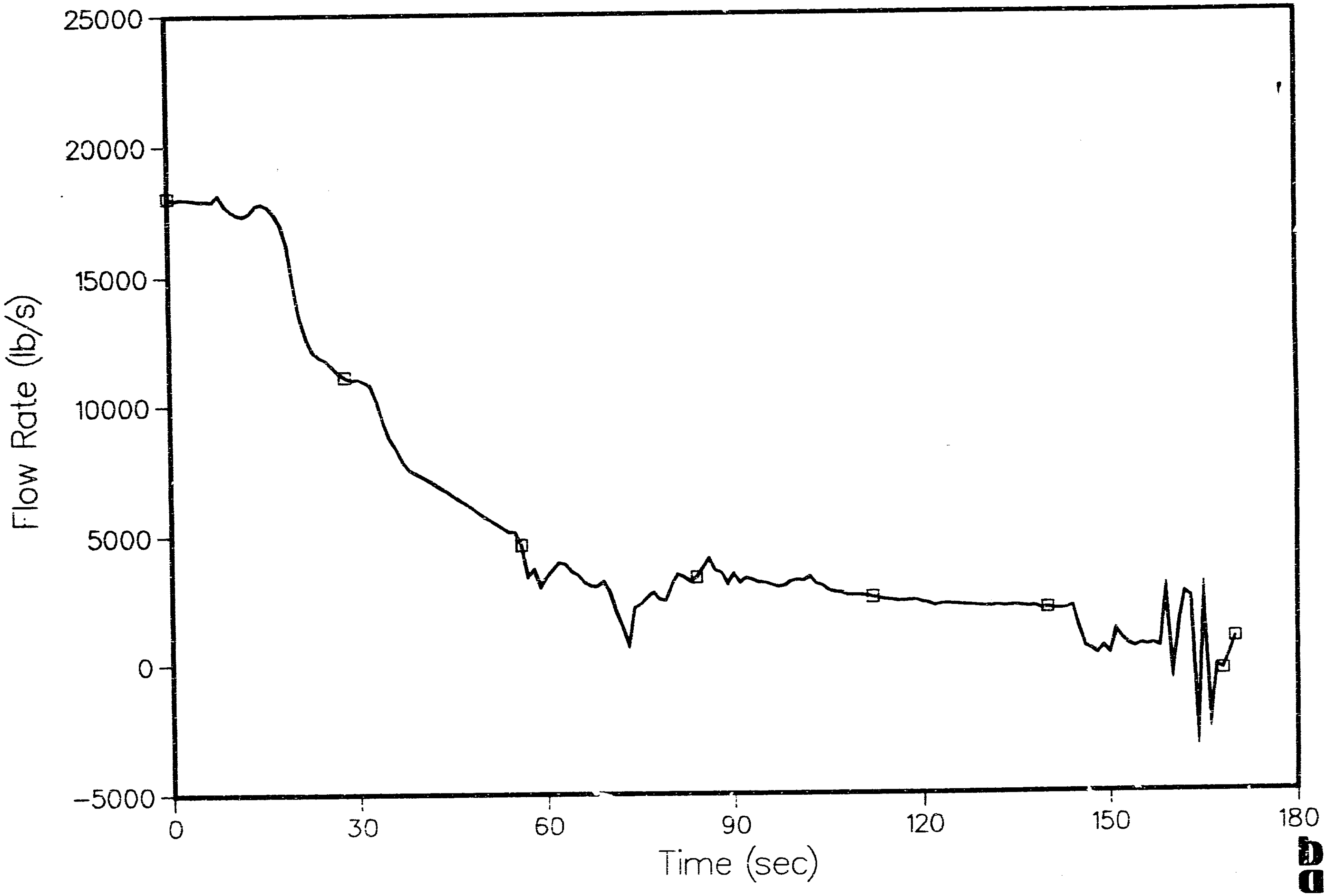
VVER–440 V213 Surge Line Break
Base Case

Figure 7: Core Void Fractions



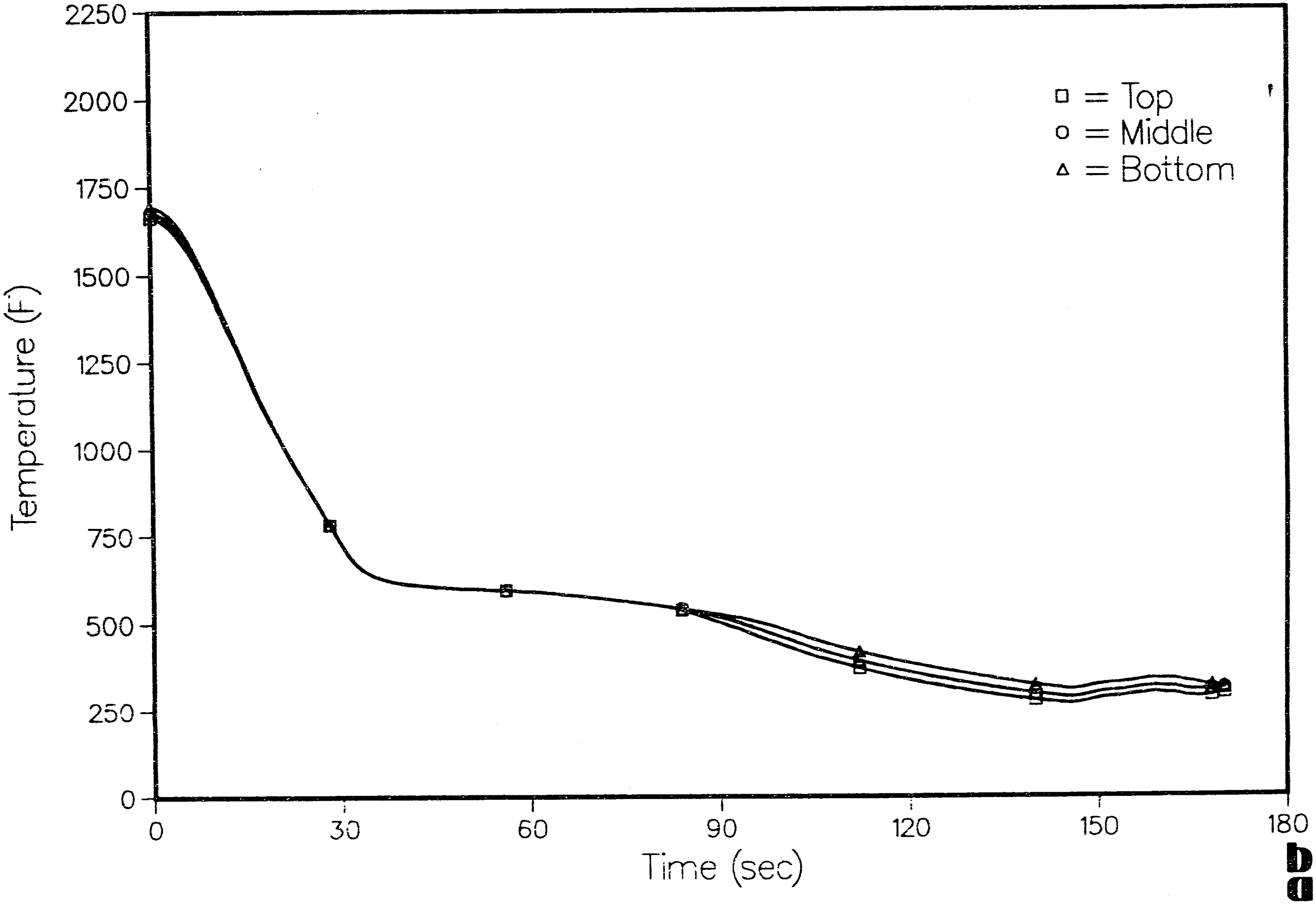
## VVER-440 V213 Surge Line Break
## Base Case

Figure 8: Core Inlet Flow



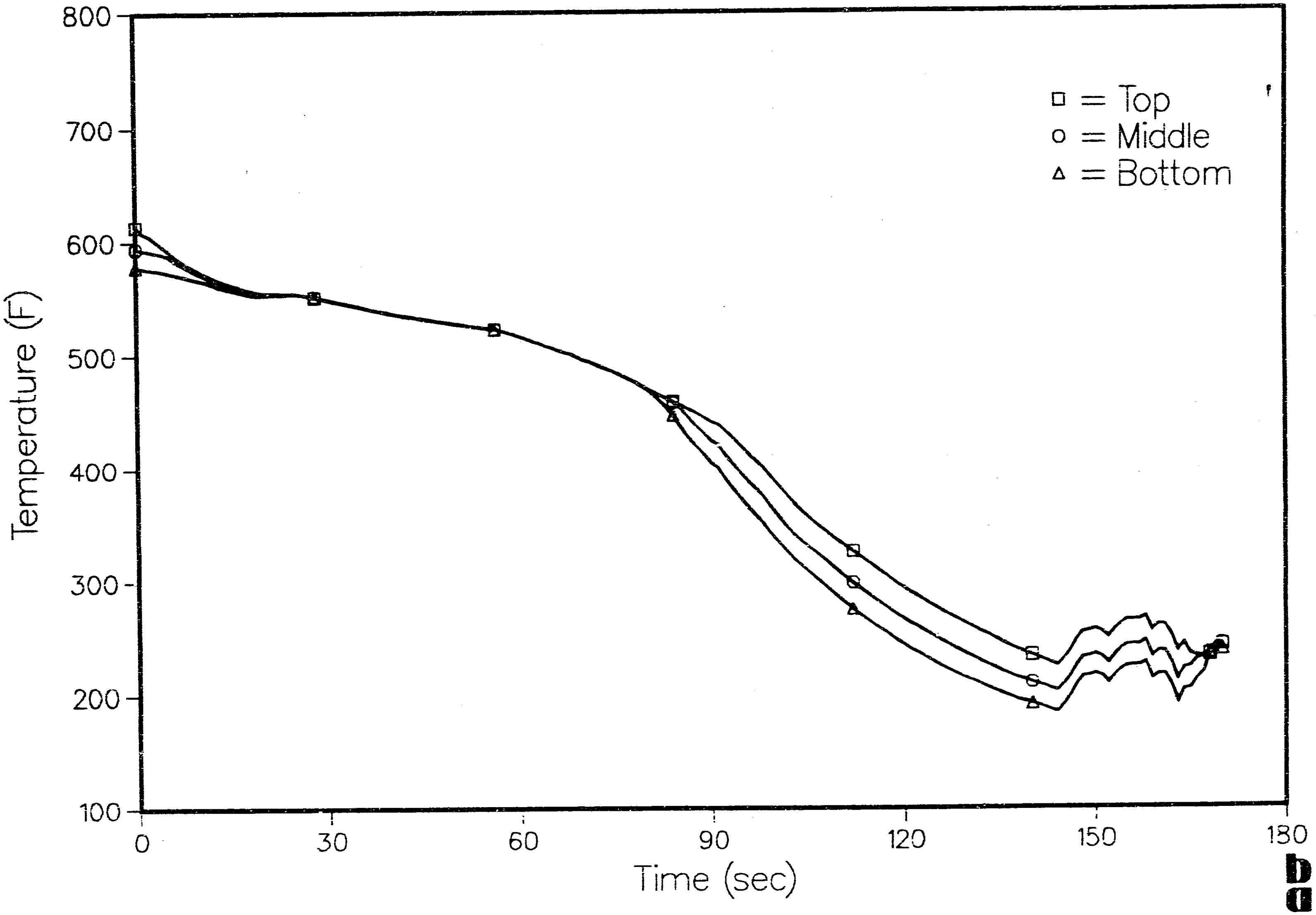
VVER–440 V213 Surge Line Break
Base Case

Figure 9: Fuel Centerline Temperatures



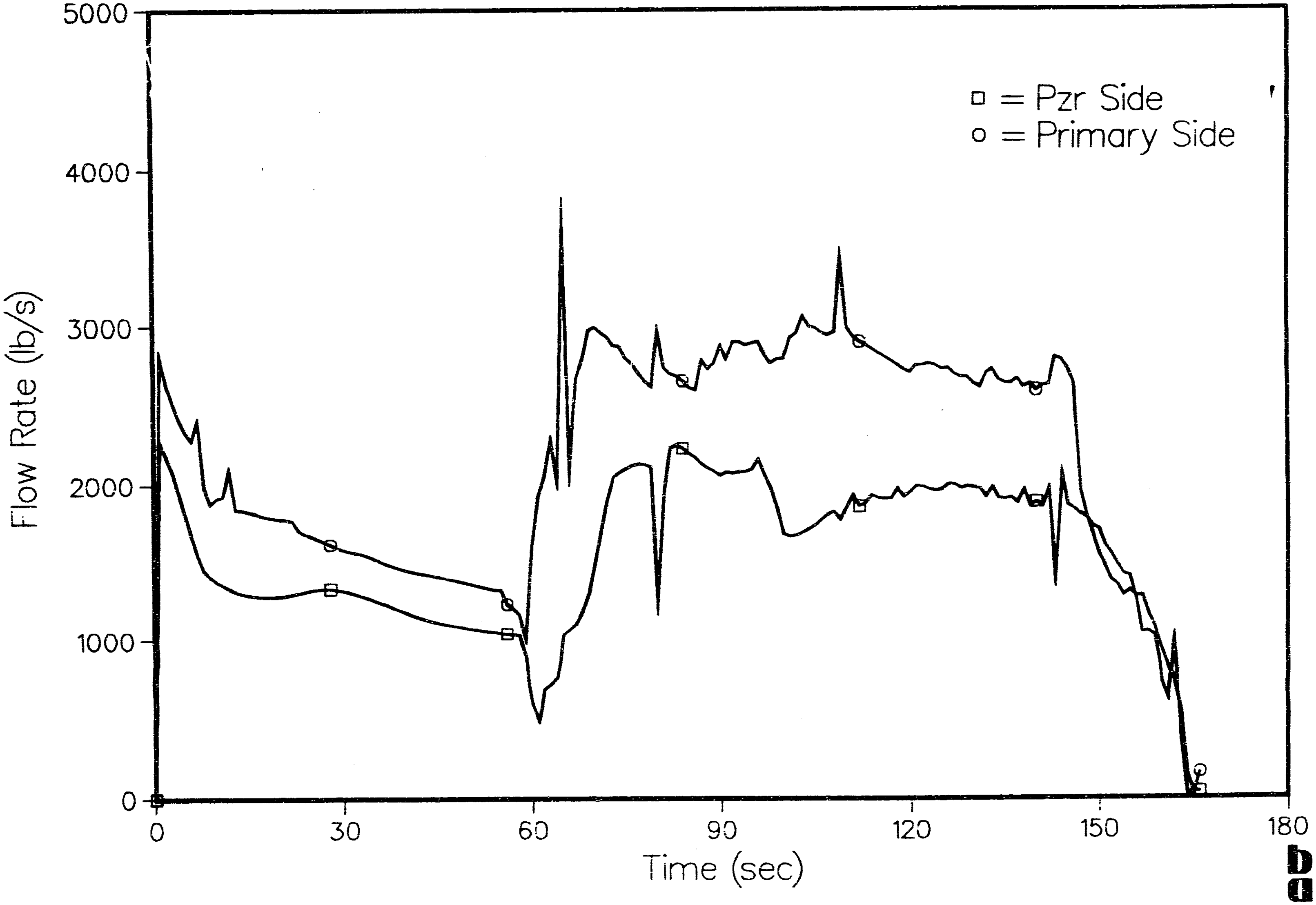
VVER–440 V213 Surge Line Break
Base Case

Figure 10: Clad Surface Temperatures



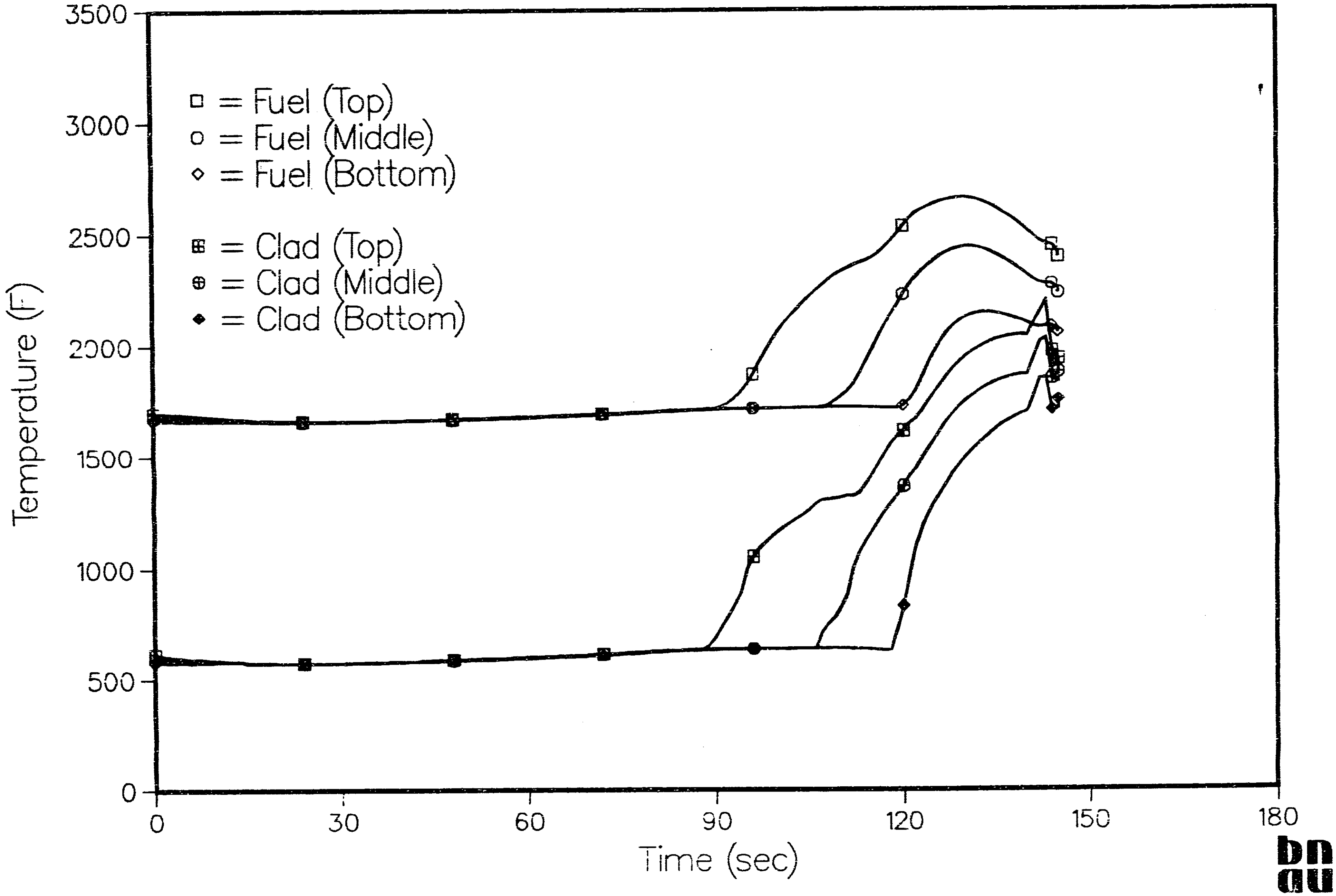
VVER–440 V213 Surge Line Break
Base Case

Figure 11: Break Flows



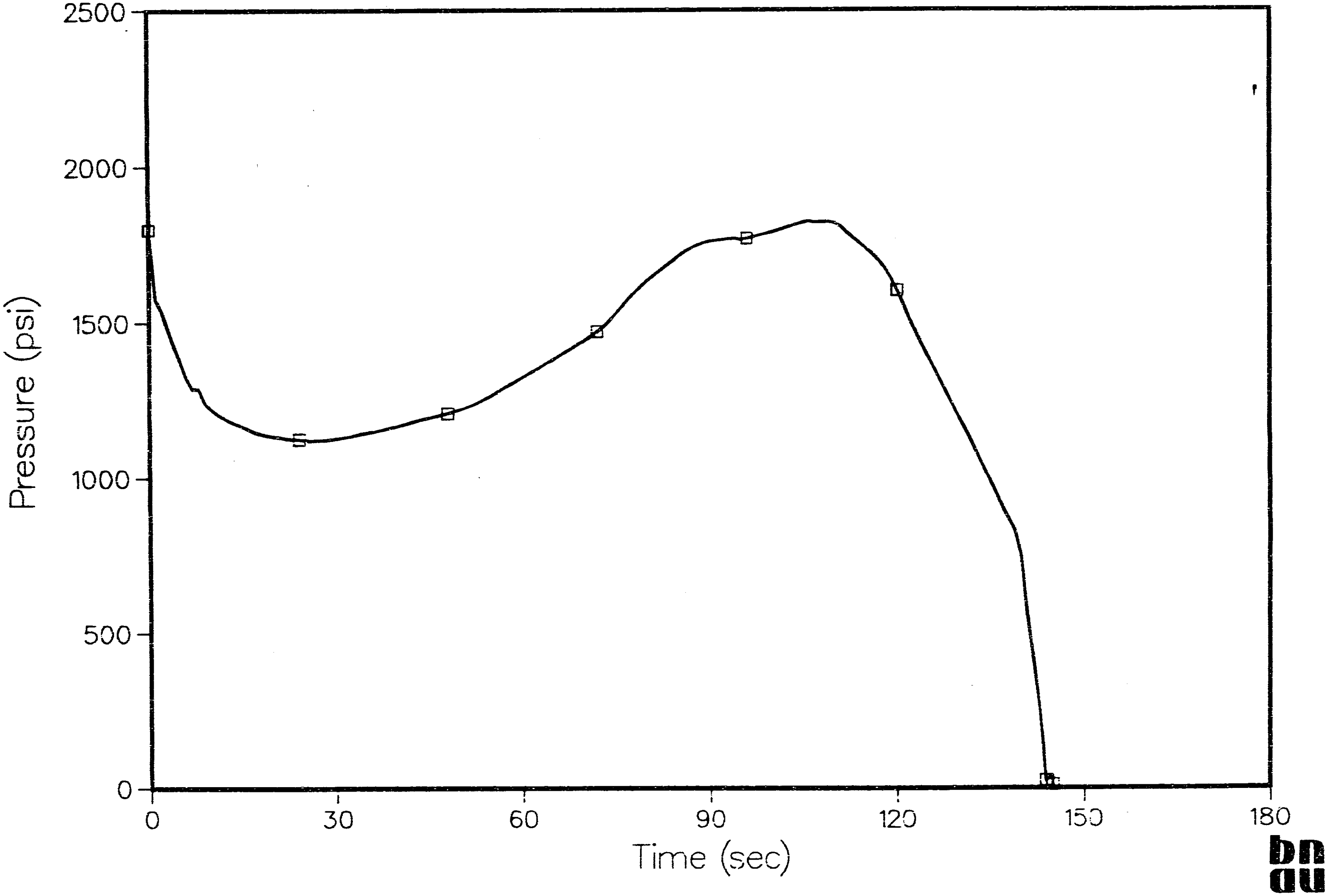
VVER–440 V213 Surge Line Break
Scram Delayed by 2 Minutes

Figure 12: Fuel and Clad Temperatures



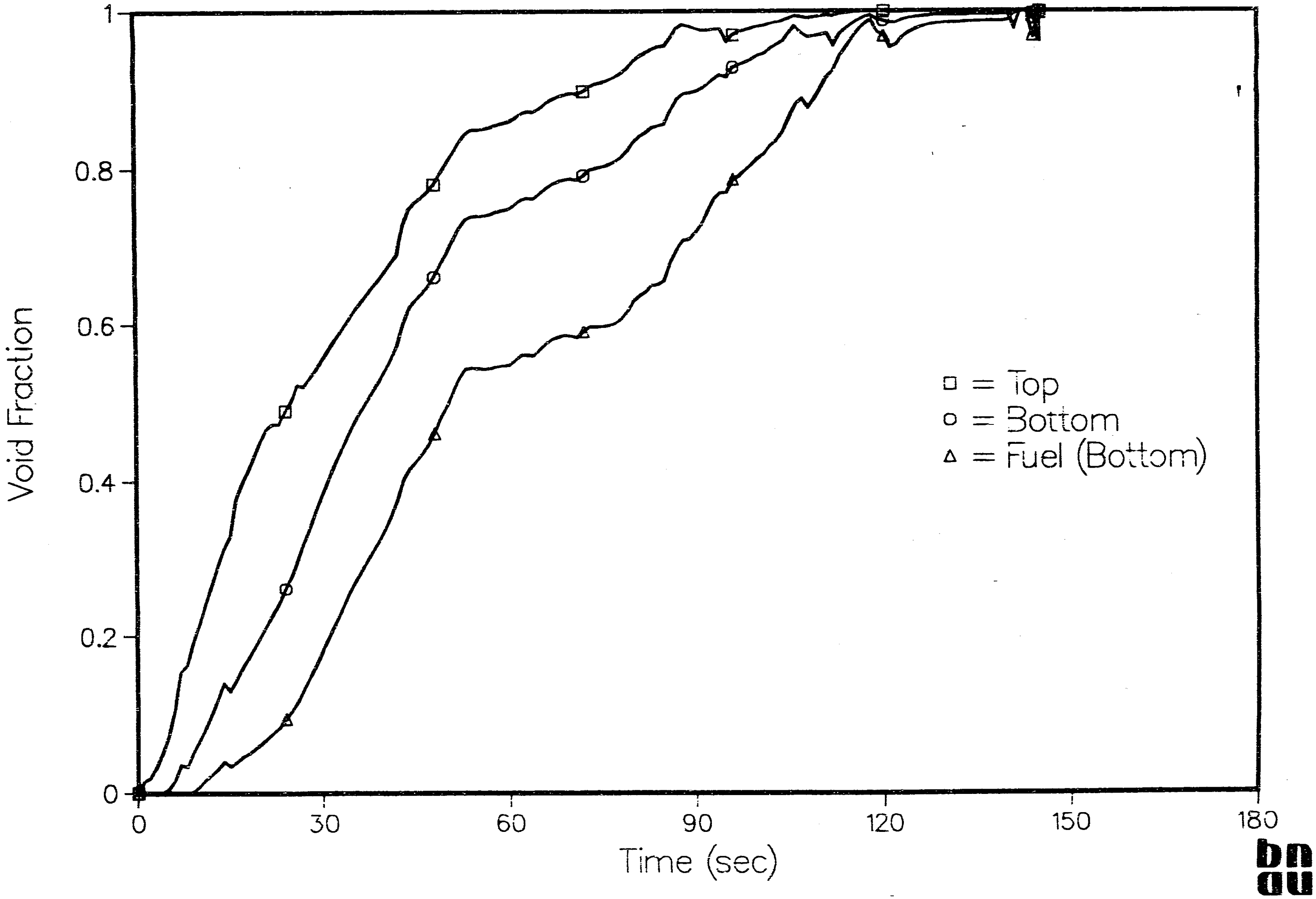
VVER–440 V213 Surge Line Break
Scram Delayed by 2 Minutes

Figure 13: Outlet Plenum Pressure



VVER–440 V213 Surge Line Break
Scram Delayed by 2 Minutes

Figure 14: Core Void Fractions



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