BWR OWNERS' GROUP

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U.S. NUCLEAR REGULATORY COMMISSION

U.S. Nuclear Regulatory Commission
Division of Licensing
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Darrell G. Eisenhut, Director

SUBJECT: BWR Owners' Group Evaluations of NUREG-0737
Requirements II.K.3.16 and II.K.3.18

Gentlemen:

This letter transmits feasibility studies performed by the BWR Owners' Group for the following NUREG-0737 items:

II.K.3.16: Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification

II.K.3.18: Modification of Automatic Depressurization System Logic - Feasibility Study for Increased Diversity for Some Event Sequences

The submittal of an Owners' Group position developed in response to an NRC requirement does not indicate that the Owners' Group unanimously endorses that position; rather, it indicates that a substantial number of members believe the position is responsive to the NRC requirement and adequately satisfies the requirement. Each member must formally endorse a position so developed and submitted in order for the position to become the member's position.

General Electric will provide sixty (60) additional copies of the attachment in a separate mailing.

Please contact me at (919) 836-6584 if you have any questions concerning the enclosed information.

Sincerely,

David B. Waters

D. B. Waters, Chairman
BWR Owners' Group

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BWR OWNERS GROUP EVALUATION

OF NUREG-0737 ITEM II.K.3.16

REDUCTION OF CHALLENGES AND FAILURES OF RELIEF VALVES

TABLE OF CONTENTS

ABSTRACT

This report documents a study performed in response to NUREG-0737 item II.K.3.16 which requires an evaluation of the feasibility and contraindications of reducing challenges to the relief valves by various methods in BWRs by April 1, 1981. The report reviews potential methods of reducing the likelihood of stuck open relief valve (SORV) events in BWRs and estimates the reduction in such events that can be achieved by implementing these methods. The reduction was estimated by computing the reduction in Safety Relief Valve (SRV) actuations achievable by various design and operating modifications and by estimating the relative probability of various types of SRVs to stick open. Using the BWR/4 plant as a measure of operating experience, it was concluded that BWR/2, BWR/3 with isolation condenser, BWR/5 and BWR/6 plants already include design features which yield a significant reduction in the occurrence of SORV events. The remaining plants can reduce the SORV event frequency by methods evaluated herein. The report applies to the plants listed in Appendix A.

1. INTRODUCTION

This report documents a study performed in response to NUREG-0737 item II.K.3.16 which requires an evaluation of the feasibility and contraindictations of reducing challenges to the relief valves by various methods in BWRs by April 1, 1981. The report reviews potential methods of reducing the likelihood of stuck open relief valve (SORV) events in BWRs and estimates the reduction in such events that can be achieved by implementing these methods.

Reducing the likelihood of S/RV challenges will directly reduce the likelihood of a SORV. In addition, attention is also given to modifications which could reduce spurious SRV blowdowns and to modifications which could reduce the probability of SRVs to stick open when challenged. The report applies to the plants listed in Appendix A.

1.1 NRC Requirement

NUREG-0737 item II.K.3.16 requires that a feasibility study be performed to identify modifications to reduce S/RV challenges. The NUREG-0737 position states, "An investigation of the feasibility and contraindications of reducing the challenges to the relief valves...should be conducted. Those changes which are shown to reduce relief-valve challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude)."

1.2 Objective, Standard and Goal

Although the NUREG-0737 position deals primarily with reduction of challenges to S/RVs, its clear intent is to reduce the incidence of SORV events. Reducing challenges is only one of three approaches to reducing SORV events. The others are reducing the causes of spurious blowdowns and reducing the probability of S/RVs to stick open when challenged. All three of these approaches present feasible and effective opportunities for reducing the incidence of uncontrolled blowdowns via S/RVs. Further, as discussed in the study, the feasible approaches to reducing S/RV challenges do not by themselves accomplish the desired "order of magnitude" (factor of ten) reduction in SORV events. Consideration of the other two approaches, however, shows a factor of ten reduction in the incidence of SORVs to be feasible. Such a reduction is the objective of this study.

The NUREG-0737 position does not state a standard by which the desired factor of ten reduction should be judged. Using the number of S/RV-years in plant operation as a criterion, it can be concluded from Table 1.1 that operating BWR/4 units provide the most representative basis for such judgements. The methods employed in this study make it desirable to take a single BWR product line as a benchmark, due to the differences in transient response, valve types, and other reactor systems (e.g., isolation condensers) among the product lines. Thus, the SORV experience of currently operating BWR/4s without the design improvements described in this study was selected as the criterion by which all plants are to be judged.

Considering all of the above, the goal of this study is to identify feasible modifications to BWR design and operation which reduce the frequency of uncontrolled S/RV blowdowns for each product line to a factor of ten below the frequency experienced in BWR/4 units.

TABLE 1.1

S/RV OPERATING EXPERIENCE**

Product Line	SRV-Years of Operation* End of 1977	SRV-Years of Operation* End of 1980
BWR/1	N/A	N/A
BWR/2	93	126
BWR/3	260	361
BWR/4	342	715
BWR/5	None	None
BWR/6	None	None

*Number of power-operated relief or safety/relief valves per plant, times number of years of operation of plant, totalled for all plants in product line through end of year.

**Includes US experience only.

N/A - Not applicable since these plants were not considered in the study.

2. BWR RESPONSE TO A TRANSIENT WITH A STUCK OPEN RELIEF VALVE

The Boiling Water Reactor design assures that core integrity will be maintained following a stuck open relief valve (SORV) transient. The response of a typical BWR to a SORV transient is discussed in this section in order to provide a perspective on typical sequences of events which lead to and follow SORV events and the consequence of various manual or automatic actions.

The safety/relief valves (S/RVs) of a BWR are designed to protect the reactor coolant pressure boundary from overpressurization. Transients resulting in pressurization frequently raise the reactor vessel pressure to the S/RV setpoint causing the S/RVs to open so that safety limits are not reached. If any relief valve fails to close after the pressure peaks and decreases, further steam release will deplete the water inventory in the reactor vessel and challenge the numerous water delivery systems which assure adequate core cooling. In a few instances, S/RVs have spuriously opened. Either event is termed a "Stuck Open Relief Valve (SORV)" event.

For any anticipated transient, if a SORV is the only additional failure, the vessel inventory lost through the SORV can be easily made up by various high pressure and/or low pressure water delivery systems. The consequences of a SORV are not a safety concern and reactor shutdown is uncomplicated, as proven by numerous field occurrences.

Following transients which result in loss of feedwater flow, a SORV could challenge the emergency core cooling systems. There are several such transients which result in the loss of the feedwater system, e.g., "loss of feedwater flow," "loss of AC power," "MSIV Closure" or "Feedwater controller failure - maximum demand." The loss of feedwater flow event is a typical and bounding transient from the core cooling viewpoint. The BWR response to this transient with a SORV and with more severe degradations is discussed in detail in NEDO-24708, Sections 3.1, 3.2 and 3.5 (Refs. 1-4). These evaluations demonstrate that adequate core cooling is assured in the BWR following an SORV event.

It is concluded that adequate core cooling is maintained in a BWR following an SORV event even under degraded conditions. It follows, then, that reduction of the frequency of SORV events is not of great concern from the standpoint of assuring adequate core cooling.

3. CANDIDATE MODIFICATIONS FOR REDUCING SORV EVENT FREQUENCY

Three different approaches can be taken to reduce the frequency of SORV events:

1. Reduction of challenges to the S/RVs;

2. Reduction of the probability of the S/RVs to stick open when challenged;

3. Reduction of spurious blowdown of S/RVs.

Each of these approaches leads to the identification of feasible and effective opportunities for reducing the incidence of uncontrolled blowdowns via S/RVs. Based on the recommendations in NUREG-0737 and on the judgment and experience of GE and utility personnel, a number of candidate modifications have been selected for consideration in this study. A description of these candidate modifications is provided in this section. The benefit associated with the implementation of each candidate modification, as estimated in this study, is also presented. Other candidate modifications may exist which are not addressed in this study.

The effectiveness of many of the candidate modifications will vary amongst the BWR product lines, due to design variations. Thus, a range of potential benefits are presented. For instance, lower water level isolation is expected to result in a 24-36% reduction in S/RV challenges. The 24% reduction is applicable for BWR/5 plants and 36% for BWR/3 plants without Isolation Condenser. Further, the values cited are maximum achievable benefits evaluated based on the assumption that the candidate modification will completely eliminate all of the challenges associated with the modification.

3.1 Candidate Modifications Which Reduce S/RV Challenges

Most of the candidate modifications to reduce S/RV challenges reduce the frequency of transient events which cause S/RVs to open. The remaining candidates reduce the number of relief valves which open during a given transient event.

3.1.1 Main Steam Line Isolation

Two candidate modifications will reduce the frequency of main steam line isolation during transients. One involves lowering the water level isolation setpoint; and the other, lowering the pressure isolation setpoint.

3.1.1.1 Lower Water Level Isolation Setpoint

Definition - Lower the RPV water level isolation setpoint for MSIV closure from Level 2 to Level 1.

Discussion - This candidate modification would reduce the number of times the reactor is isolated from the main condenser. This results in reduced S/RV challenges by eliminating isolation cycling of the S/RV's resulting from transients such as feedwater controller failure, trip of both recirculation pumps, and loss of feedwater flow. This modification is expected to result in a reduction of 24 to 36% in S/RV challenges for plants without isolation condensers.

This modification is feasible for BWR/4-5 plants which do not already include the feature. It is not feasible for BWR/2-3 because of the need for additional reactor water level instrumentation.

3.1.1.2 Lower Reactor Pressure Isolation Setpoint

Definition - If the reactor is in the "run" mode and the main steam line pressure drops below 825 psig, the reactor is isolated in order to prevent a rapid cooldown resulting from a pressure control malfunction. This candidate modification would reduce the pressure at which the reactor is isolated under these conditions. The extent of reduction in pressure setpoint has not been established but is expected to be in the neighborhood of 50 psig.

Discussion - Prior to 1975, the main steam line isolation was initiated at 850 to 880 psig depending upon the product line. Operating experience at that time showed that this setpoint was too close to the normal operating pressure. As a result, the noise level of the pressure switch hydraulic sensing line or small pressure transients in the main steam lines could initiate reactor isolation. After a careful review, General Electric determined that the isolation setpoint could be safely lowered to 825 psig. This setpoint provides adequate protection against spurious isolation events. A review of recent operating plant experience shows that the additional reduction in S/RV challenges resulting from further reduction in isolation setpoint would be less than 1%.

3.1.2 Feedwater Control System Modifications

A number of candidate modifications improve the feedwater control system as a means to reduce S/RV challenges by reducing the number of main steam line isolation events. Feedwater control system failures have contributed to about 0.7 isolation events per plant year. The thrust of the modifications is to control water level between the high water level trip and the ECCS initiation/MSIV isolation trip setting for various transients.

Water level is controlled in a BWR by a Feedwater Control System that utilizes a single primary channel for control. The control system utilizes a water level sensor input (the primary element) and the difference between two secondary elements, namely feedwater flow and steam flow. Water level must be under positive control by the Feedwater Control System during plant operation, because feedwater flow must respond to changes in steam flow while maintaining the water level in the vessel. Therefore, when a component in the control system or in the power supply to the control system fails, the water level in the reactor vessel can drift out of limits, ultimately causing the reactor to scram. Such a water level excursion is usually so rapid that the reactor operator is unable to respond in time to prevent a scram from an abnormal water level condition. Frequently these scrams are followed by reactor isolation with consequent vessel pressurization causing the S/RV's to open.

3.1.2.1 Triple Redundant or Single Failure Proof Control System

Definition - A triple redundant control system is a candidate modification which could reduce isolations. Such a system would have three channels of control, with the highest and lowest values being ignored. Thus failure of the controlling element (either upscale or downscale) would result in another channel taking control. The single failure proof control system would have two channels of control, in which the second channel acts as a backup for transfer of control when a failure of the controlling channel is detected.

Discussion - The improved control system would reduce transients resulting from failure of components in the existing single channel Feedwater Control System. By eliminating these transients, the associated reactor scrams and isolation events are also eliminated, which reduces the number of S/RV challenges. This results in a reduction of about 2 to 4% in S/RV challenges. The small reduction in S/RV challenges alone does not justify the high cost of implementing the modification.

3.1.2.2 Uninterrupted & Redundant Control System Power

Definition - Failures in Feedwater Control System power supplies have caused reactor isolations in operating plants. This candidate provides for an uninterruptible and redundant source of power such that the controller is not affected by failures in the power supply system.

Discussion - This change could eliminate S/RV challenges associated with isolation events arising out of failures in the power supply. A maximum of 0.07 isolation events per plant year can be eliminated by implementing this modification, resulting in about a 1% reduction in S/RV challenges.

3.1.2.3 Condensate System Modifications and Condensate/Feedwater Integration

Definition - The controls of the condensate system (including the demineralizers), on which the feedwater system depends for proper operation, could be integrated with the Feedwater Control System so that failures in the condensate system would be detected in such a way that reactor scram and isolation could be avoided.

This candidate modification calls for an integration of the condensate and feedwater control systems in providing input signals for reactor operation. Examples of the possible integration and design modifications of the condensate system are as follows:

a) Typically, three 50% or four 33% capacity condensate pumps are provided per plant. If one of the condensate pumps fails, the redundant pump could be automatically started.

b) If there is high differential pressure across a condensate demineralizer, a signal could be provided to cut back reactor power by running back recirculation flow.

c) The Feedwater Control System could be designed to assure that a loss of a single condensate or condensate booster pump or feed pump will not result in reactor scram or isolation.

Discussion - Implementation of these candidate modifications could result in a reduction in reactor isolation events for BWR/3 plants without Isolation Condenser, BWR/4 and BWR/5 plants, and consequently, relief valve challenges resulting from failures in the condensate and feedwater systems. This could result in a reduction of 3-4% in S/RV challenges.

The implementation of this modification would increase the complexity of the feedwater and recirculation control systems, and thereby introduce additional failure modes. Therefore this modification could have an adverse impact on the reliability of these systems.

3.1.2.4 Feedwater Runback

Definition - Feedwater runback is a method of controlling reactor water level to avoid high vessel water level (L8) trip, following certain transients. This would prevent tripping the feedwater pumps and subsequent reactor isolation on low water level.

Discussion - The implementation of this candidate modification would result in the elimination of S/RV challenges associated with the trip of both recirculation pumps and recirculation controller failure. A reduction of 6 to 12% in S/RV challenges can be expected by implementing this modification.

The implementation of this modification would increase the complexity of the feedwater control system, and thereby introduce additional failure modes. Therefore, this modification could have an adverse impact on the reliability of that system.

3.1.2.5 Additional Anticipatory Scram on Loss of Feedwater

Definition - A description of this candidate modification is quoted from NUREG-0626: "...The challenge rate could be reduced by providing anticipatory signals on the feedwater pump trip similar to the scram signal derived from turbine stop valve closure on a turbine trip. This modification will reduce the reactor power quickly and thereby reduce the severity or magnitude of the pressure spike."

Discussion - A review of the Loss of Feedwater Flow (LFWF) event in NEDO-24708 shows that LFWF causes a low water level scram approximately 7-15 seconds following initiation of the transient at full power, depending upon the product line. LFWF results in reactor isolation for BWR/1 thru

BWR/5, and anticipatory scram on feedwater pump trip does not prevent reactor isolation at reactor level 2 or the associated cycling of relief valves. LFWF does not result in reactor isolation for BWR/6 due to isolation at reactor level 1. This candidate modification is of no benefit in reducing S/RV challenges for BWR 1-5 if the low water level isolation modification is also carried out, since the latter modification prevents reactor isolation following the LFWF event. A possible disadvantage of implementing this candidate modification is that it denies the operator the opportunity to prevent a scram by restarting feedwater pumps. In summary, anticipatory scram or loss of feedwater is considered to be an insignificant contributor (less than 1%) to S/RV challenge reduction for all BWR product lines.

3.1.3 SRV Control Logic/SRV Setpoint Revision

The following candidate modifications are expected to reduce S/RV challenges through changes to SRV control logic or through revision of S/RV setpoints.

3.1.3.1 Low-Low Set Relief or Equivalent Manual Actions

Definition - Some BWR plants are equipped with a 'Low-Low Set' design feature which changes the setpoints of selected SRVs following the initial opening of a number of S/RVs. This assures that following the initial pressurization the pressure will be relieved by the 'Low-Low Set' valve alone, and the remaining S/RVs will not experience any subsequent actuation This feature could be applied to plants which do not currently include it. However, the BWR Emergency Procedure Guidelines (Ref. 5) call for the equivalent manual action.

Discussion - The 'Low-Low Set' design or equivalent manual action will reduce the total number of S/RV challenges by limiting the second and subsequent opening of the S/RVs to the Low-Low Set valve. It is estimated that a 23-62% reduction in S/RV challenges can be achieved by implementing this modification. This modification is practical for all BWR product lines.

3.1.3.2 Revised Relief Valve Setpoints

Definition - A description of this candidate modification is quoted from NUREG-0626: "The relief valve setpoints could be revised upward to allow more margin to the relief valve opening setpoint. Another method to provide margin is to lower the operating pressure. A combination of the relief valve setpoint and operating pressure will increase the plant's ability to withstand a pressure increase transient without causing the relief valve to open."

Discussion - There are two setpoints associated with safety/relief valves, namely relief setpoint and safety setpoint. The relief setpoint, which is the lower of the two setpoints, is used to provide pressure relief following an overpressure transient. The safety setpoint limits the reactor pressure to the ASME code allowable limits.

Both the relief and spring setpoint values are constrained by a number of factors. In the case of the Target Rock valve, the factors are:

1. The ASME Code

2. High pressure injection system (High Pressure Coolant Injection (HPCI), High Pressure Core Spray (HPCS), Reactor Core Isolation Cooling (RCIC)) design discharge pressure. If the spring setpoints are higher than the pump discharge pressure, high pressure coolant cannot be injected into the reactor under all design conditions.

3. A need to offset relief valve setpoints where applicable. This is done to prevent all valves from opening simultaneously.

4. Tolerance on the relief valve setpoints. Setpoint drifts in one direction should not result in the valve opening in the safety mode, nor should a drift in the opposite direction result in relief valve operation for minor overpressure transients.

In the case of Crosby and Dikkers valves there exists another factor which is a practical consideration of requiring the valve to reclose in a relief mode rather than the spring mode, i.e., the spring mode reclosure setpoint should be always higher than the relief mode reclosure setpoint, even after allowing for setpoint drifts.

In the case of Three Stage Target Rock valves, it has been determined that the spring setpoint could be raised by about 15 to 50 psig depending upon the plant. The impact of such a modification is a reduced incidence of spurious S/RV actuations, through increased simmer margin. Based on a review of failure data and engineering judgment, a 5% reduction in SORV events in plants with Three Stage Target Rock valves is expected through increasing the spring setpoint to the maximum value possible. No reduction in S/RV challenges is likely because a 15-50 psi increase in setpoint is insignificant compared to the pressure rise experienced in most overpressure transients.

In the case of Crosby and Dikkers valves (BWR/5-6 plants) the setpoints are already near their maximum possible value and can be increased by no more than about 15 psig. Such an increase in the relief setpoint will not cause any significant reduction in SRV challenges.

In the case of Two Stage Target Rock valves, pilot valve leakage does not lead to spurious opening. Therefore, the conclusion for Crosby and Dikkers valves also applies for the Two Stage Target Rock valve.

In conclusion, the plants with Three Stage Target Rock valves may be able to achieve a 5% reduction in spurious SRV openings through an increased spring setpoint. None of the plants can achieve any significant reduction in SRV challenges through an increased relief setpoint.

The NUREG-0737 suggestion also refers to lowering the operating pressure. But as stated above for setpoint increases, modest changes are insignificant compared to the pressure rise experienced in most overpressure transients. Thus, lowering the operating pressure by a modest amount would not result in any significant reduction of S/RV challenges. In addition such a change would result in undetermined penalties in terms of plant thermal efficiency and fuel utilization.

3.1.3.3 Offset Relief Valve Setpoints

Definition - A description of this candidate modification is quoted from NUREG-0626 as follows: "The valve pressure setpoint could also be modified or offset such that fewer valves are challenged."

Discussion - Offsetting relief valve setpoints does not contribute to S/RV challenge reduction during isolation cycling since only one or two valves participate in such cycling. During the initial blowdown there could be some reduction in S/RV openings for some transients. It is noted that small but unavoidable setpoint drifts result in a de facto offsetting even when several valves are nominally set at the same value.

3.1.3.4 Increase Main Steam Line Flow Setpoint

Definition - A description of this candidate modification is quoted from NUREG-0626 as follows: "Increasing the high steam line flow setpoint for main steam line isolation valve (MSIV) closure (can reduce SRV challenge and failure rates)."

Discussion - The MSIVs are designed to close when a break occurs in the main steam lines. An abnormal increase in the main steam line flow is taken as an indication of a main steam line break. High steam line flow setpoints are selected in a manner as to assure a high probability of isolation on a steam line break while keeping the probability of inadvertent closure resulting from operational transients small. A review of the BWR experience data has revealed no instance of spurious MSIV closure resulting from plant transient events. However, a number of inadvertent isolation events have occurred during MSIV closure surveillance testing. These occurred when a second MSIV was closed without resetting the first MSIV that was tested. The sudden increase in steam flow in the remaining lines results in reactor isolation. The maximum reduction in SRV challenges that can theoretically be obtained through an increased high steam flow setpoint is about 0.5%. However, such a reduction may not be practical since increasing the setpoints will reduce the reliability of isolation following a main steam pipe break. A more practical approach to achieve the same goal is through reduction in MSIV test frequency, discussed in 3.1.4.4.

3.1.4 Other Candidate Modification

Candidate modifications pertaining to other systems are discussed in this section.

3.1.4.1 Analog Transmitter/Trip Unit System

Definition - Most operating BWRs use direct acting pressure, differential pressure and water level switches as input into the reactor protection, main steamline isolation, and emergency core cooling systems. Technical specifications for this type of process sensor typically require surveillance testing once a month while the plant is at power. In the past, during the monthly surveillance tests, errors have caused scrams and challenges to relief valves. If an improved system were installed which uses an analog transmitter and bi-stable trip unit instead of the pressure switch, the number of unnecessary scrams (and associated SRV challenges) could be reduced. The transmitter-trip unit combination can also be designed to be highly stable and easily testable. Calibration requirements for this new system are thus greatly reduced.

Discussion - The use of the analog transmitter/trip unit system would reduce the number of reactor scrams resulting from procedural and physical errors during surveillance tests. A 2 to 5% reduction in S/RV challenges could be expected due to the implementation of this candidate modification.

3.1.4.2 Improved Recirculation Flow Control System

Definition - Failures in the recirculation flow electronic control systems can result in reactor isolation. If an augmented recirculation flow control system with signal deviation alarms and signal rate alarms to detect failures in the control electronics were provided, the significance of flow changes could be reduced. The failure detection scheme in the augmented system would cause the logic signal to change from automatic flow control to a steady recirculation flow to prevent a core flow excursion and eventual scram.

Discussion - It is estimated that approximately 2% to 6% of the S/RV challenges could be eliminated with this equipment. However, the cost and increased complexity of the control system must be evaluated further before this candidate modification can be considered feasible.

3.1.4.3 Reduce Isolations Caused by Surveillance Testing

Definition - This candidate calls for developing an improved method of carrying out surveillance tests without causing inadvertent isolations. This may involve hardware and design changes. In addition, reduction of surveillance testing frequency could reduce the inadvertent closures.

Discussion - A maximum of 4 to 5% reduction in S/RV challenges could be achieved through the implementation of this candidate modification.

3.1.4.4 Reduce MSIV Testing Frequency

Definition - This candidate modification is suggested in NUREG-0737. A number of isolation events occur while the MSIV closure tests are being conducted. A reduction in the MSIV test frequency would result in a reduction in number of isolation events.

Discussion - The frequency of MSIV tests is contained in a plant's technical specifications and generally conforms to ASME Code Section XI recommendations. The extent of frequency reduction that is possible without impacting reliability of isolation capability should be considered in the detailed design of this modification. However, the maximum benefit that could be expected is about 2 to 3% reduction in S/RV challenges.

3.1.4.5 Installation of New Relief Valve With Block Valve in Series

Definition - A description of this candidate modification is quoted from NUREG-0626: "Plants could also be modified by installing new relief valves with normally open isolation or block valves that would eliminate the opening of present S/RVs that may fail to close and cannot be isolated."

Discussion - The following factors would have to be considered in the implementation of this candidate modification:

1. The suggested modification could be in violation of the ASME Code, unless some valves are dedicated for the safety function and others for the relief function. Any modification would need to be reviewed to assure compliance with the ASME Code.

2. The new relief valves, pipes and block valves would have to be designed for the reactor design pressure (1250 psig). Currently, the relief valve discharge flange and piping is designed for about 500 psig.

3. Inadvertent closure of the block valves would cause the new relief valves to become unavailable for the relief function.

This candidate modification would not reduce the SORV event frequency, but would reduce the consequence of such an event. Theoretically it is possible to design a system that will mitigate any future SORV event; however, from a practical standpoint this may be an impossible modification to implement since there would not be sufficient room in the drywell of most plants to accommodate the additional piping and equipment. The large expense in terms of cost and personnel exposure required to implement this concept is not justified when the low risk of core damage resulting from a SORV event is considered.

3.1.4.6 Earlier Initiation and Increased Flow of Emergency Core Coolant

Definition - A description of this candidate modification is quoted from NUREG-0626: "Another method that could be employed is to provide additional emergency core coolant (ECC) flow to act as a heat sink (steam condenser) to accommodate the pressure increase due to swelling of the coolant. This could also be accomplished by modifying the plant instrumentation to provide earlier ECC system initiation. The combination of increased ECC flow at an earlier time in the transient could provide the necessary heat sink to absorb the power or pressure spike before the relief valve setpoint is reached."

Discussion - ECC flow could not be initiated early enough by any practical means to result in any significant reduction of the number of valves that open during the initial blowdown, because of the steep rate of pressure rise following a transient. Further, earlier initiation of ECC flow could result in ECC initiation and L8 feedwater pump trip on transients such as turbine trip (following which ECCS is not expected to initiate), causing simultaneous loss of the preferred coolant source (feedwater) and the preferred heat sink (the main condenser). Such a modification cannot be justified.

3.2 Reduction of the Relative Probability of the Valve to Stick Open

Many operating BWR plants are equipped with Three Stage Target Rock valves. The Three Stage Target Rock valves have exhibited a higher probability to stick open in the past than other types of valves. A detailed review of the SORV events associated with the Three Stage Target Rock valve was carried out by General Electric and Target Rock Company, and the results have been used to identify valve modifications which improve the valve performance. Design and operational modifications have been identified for the Three Stage Target Rock valve which reduce the probability of the Three Stage Target Rock valves in service to stick open.* In addition, the valve topworks have been redesigned to minimize the probability of the valve to stick open. The new design is referred to as the "two stage" design. Such valves have been installed in some operating plants. The valves thus modified are referred to as Two Stage Target Rock valves in this study.

Some operating BWR plants are equipped with Dresser Electromatic relief valves. BWR/5-6 plants are equipped with Crosby and Dikkers dual function safety/relief valves.

Assigning a normalized SORV probability factor = 1.0 for the Three Stage Target Rock valve (which is taken as a benchmark valve), the relative SORV probability factors for other valves were determined as follows:

Two Stage Target Rock Valve	=	0.50
Dresser Electromatic Valve	=	0.25
Crosby Valve	=	0.125
Dikkers Valve	=	0.125

3.3 Reducing Causes of Spurious Blowdowns

The following candidate design modifications are expected to directly affect the number of SORVs by eliminating the causes of spurious blowdowns.

3.3.1 Eliminate Spurious Safety/Relief Valve Openings Resulting from DC Power Supply Ground Faults

Definition - Inadvertant S/RV openings can be reduced by providing double pole single throw switches, or other means of protection that disconnect both the positive and the negative sides of the DC power supply, for energizing and deenergizing the solenoids of safety relief valves.

*Plants with improved Three Stage Target Rock valves and plants employing operational modifications will address their valve reliability on a plant-unique basis.

Discussion - The potential benefit of this candidate modification is the avoidance of spurious depressurization of the reactor as the result of grounding faults in the DC power supply. It is estimated that approximately one spurious relief valve opening or failure to reclose after proper opening per 50 reactor years would be eliminated by this modification.

Detailed design should assure that the new switching device will not be less reliable than the existing device in performing the functions of energizing and deenergizing the solenoid coil on demand.

3.3.2 Control of Pneumatic Supply Pressure to S/RV's.

Definition - High pneumatic supply pressure to the actuating solenoids of Target Rock S/RVs caused one spurious blowdown in an operating plant. Improved pneumatic supply pressure control would eliminate the cause.

Discussion - The implementation of this candidate modification would assure that this mode of spurious S/RV actuation will be eliminated. This modification results in a maximum reduction of 2% in spurious blowdowns.

3.3.3 Revised S/RV Spring Setpoint. See discussion in Section 3.1.3.3.

3.3.4 More Stringent Leakage Criteria and Early Removal of Leaking Valves

Definition - These candidate modifications were suggested in NUREG-0626. "More stringent leakage criteria" is assumed here to refer to leaking safety/relief valves while in operation. "Early removal of leaking valves" refers to a planned action of removing the valves which begin to leak.

Discussion - Leaking Three Stage Target Rock valves can result in spurious blowdown. Analysis has shown that a maximum of 40 to 60% reduction in spurious operation of S/RV's could be obtained by identifying and replacing valves with high leakage. Since all valves leak to some extent, it is difficult to develop an absolute leakage criterion. Additional study is required to develop a leakage criterion which is practical and a system to detect the leakage. With the use of the Two Stage Target Rock, Crosby or Dikkers valves, the leakage is not a concern because leakage does not significantly affect the spurious blowdown probability. The implementation of this candidate modification will not reduce spurious blowdowns in the Two Stage Target Rock, Crosby and Dikkers valves.

3.3.5 Use of Two Stage Target Rock Valves

Definition - The Three Stage Target Rock valves could be changed to Two Stage Target Rock Valves.

Discussion - The Two Stage design eliminates most spurious blowdown modes associated with the Three Stage Valves. A 40-60% reduction in spurious blowdowns can be achieved by changing the Three Stage Target Rock valves to Two Stage valves.

4. METHODOLOGY

This section discusses the methodology used in this study.

4.1 Introduction

Although the NUREG-0737 position deals primarily with the reduction of challenges to S/RVs, its clear intent is to reduce the incidence of SORV events. Reducing challenges is only one of three approaches to reduction of SORV events. The others are reducing the causes of spurious blowdowns and reducing the probability of S/RVs to stick open when challenged. All three approaches are required to achieve an "order of magnitude" (factor of ten) reduction in SORV events. BWR/4 units equipped with Three Stage Target Rock valves were used as the basis for such judgments. The BWR/4 with Three Stage Target Rock valves is referred to as the "benchmark" plant in this discussion.

4.2 Approach

A comparison of the SORV event frequency that can be expected over the lifetime of each BWR product line is made by multiplying the expected number of S/RV openings during a plant's lifetime by the relative probability factor for the S/RV to stick open. The SORV event frequencies thus computed for each product line were normalized to that of the benchmark plant taken as 100.

The reduction of spurious operation of relief valves was estimated based on operating experience and engineering judgment.

4.3 Probability of S/RV's to Stick Open

For comparing the various valves, the Three Stage Target Rock valve was taken as the benchmark valve with an assumed normalized factor of 1.0 for probability to stick open when challenged. Similar factors for other types of valves were obtained as described below.

4.3.1 Two Stage Target Rock

A detailed review of all the SORV events in operating plants was made, and the failure modes associated with the Three Stage Target Rock valve were tabulated. Then based on a study of the Two Stage valve design, an assessment was made of all the failure modes that are eliminated by the Two Stage design. Consideration was also given to any new failure modes which might develop in going from the three stage to the two stage design. With this information, a relative probability factor of 0.50 was assigned for the Two Stage valve to stick open, when challenged.

4.3.2 Dresser Electromatic

Based on a review of operating experience and engineering judgment, the Dresser valve was assigned a factor of 0.25 for its relative probability to stick open, when challenged.

4.3.3 Crosby and Dikkers

The actual experience with the Crosby & Dikkers valves is too limited to be used for estimating the relative SORV probability factor. However, since those valves are direct acting (unlike the Three Stage Target Rock Valve which is pilot operated) seat leakage is not likely to be a significant concern as in the case of Target Rock valves. Based on valve qualification test data and limited operating experience, a factor of 0.125 was assigned for their relative probability to stick open, when challenged.

4.4 Estimation of S/RV Challenges

The total number of S/RV challenges expected over the design life of a plant was estimated as described below for each BWR product line. The total S/RV challenges during a plant lifetime was taken to be the summation of the product of the frequency of various design transients and the estimated number of valve openings per occurrence of a transient. These values were then normalized to the benchmark plant whose value was taken as 100.

4.4.1 Frequency of Transients

It was assumed that each plant will experience the same number of various transients as were considered in its design basis. To estimate the impact of various design improvements on the frequency of transients, data from operating BWR plants spanning 120 reactor-years of operation and approximately 1400 reactor scram events (which include 720 S/RV challenge events) were investigated. By analyzing the data an estimate was made of the percentage by which various transient event frequencies would be modified if each of the candidate modifications discussed in Section 3 were implemented. This estimate was used to modify the frequency of design basis transients.

4.4.2 Total Number of Valve Openings

The total number of valve openings for various transients was computed by using the General Electric long-term thermal hydraulics model (SAFE and REDY codes).

There are many operational transients which can result in a pressure rise in the reactor vessel. The safety/relief valves will open if necessary to prevent the pressure from exceeding allowable limits. For most of these events, the safety/relief valves will open only once. However, there are several types of transient events which can result in a closure of the main steam isolation valves. Although a scram occurs immediately when the isolation valves close, the reactor continues to generate steam due to decay heat. The safety/relief valves are then the primary means of reactor pressure control. One or more valves may open with the initial pressurization following MSIV closure. These safety/relief valves will open when their pressure setpoints are reached and will discharge steam to the suppression pool until the vessel pressure decreases to the closure setpoint of the valve. Reactor pressure will then increase again until the lowest safety/relief valve's opening setpoint is reached. In most instances, only one S/RV will open on subsequent actuations. If no operator action occurs, one valve will continue to cycle open and closed. The total number of safety/relief valve lifts is thus based on three factors--the number of transient events which result in opening of the safety/ relief valves, the number of valves which open in the initial pressurization, and the number of cycles which subsequently occur.

4.4.3 Discussion of Assumptions

Following are the key bases and assumptions used in the analysis.

4.4.3.1 The frequencies of transient events are based upon the BWR/6 design document for plant duty requirements. Overall the estimated number of relief valve openings based on the design transient frequency differed only by 13 to 21% from the estimate based on transient frequency actually experienced by the plant. Since these numbers were used only to determine the relative contribution of various modifications, this difference of 13 to 21% is not significant.

4.4.3.2 The maximum specified relief valve reclosure setpoint is used since a smaller difference between the opening and closing setpoints results in a larger number of cycles. A 25 psi blowdown per relief valve opening is used for pilot operated valves such as Target Rock and Dresser, and a 50 psi blowdown is used for direct-acting valves such as Crosby and Dikkers.

4.4.3.3 Initial plant operating conditions are at 105% nuclear boiler rated steam flow.

4.4.3.4 BWR/2 and 3 plants equipped with isolation condensers are assumed to be capable of avoiding relief valve cycling after the initial relief valve discharge due to a transient.

4.4.3.5 For isolation transients, subsequent single S/RV discharges continue for 30 minutes.

5. RESULTS

5.1 Expected SORV Frequency

The expected SORV event frequency (normalized to the benchmark plant) for some of the most effective modifications are shown in Table 5.1. The following conclusions can be reached from this table.

5.1.1 BWR/2, BWR/3 with isolation condenser, BWR/5 and BWR/6 plants are estimated to have a SORV frequency which is a factor of 10 less than the benchmark plant (BWR/4).

5.1.2 BWR/4 plants can reduce the SORV frequency by a factor of ten or more by implementing selected modifications from Section 3.

5.1.3 BWR/3 plants without isolation condensers can reduce the SORV frequency by a factor of ten or more by implementing selected modifications from Section 3.

5.1.4 The effect of isolation condensers on plants not so equipped is shown for comparison even though they are not practical for backfit application. The effect of high steam bypass capability (here, 110%) is shown because some plants are so equipped.

5.1.5 The relative impact of each candidate modification on the S/RV challenge reduction alone is shown in Table 5.2. It should be noted that more than one candidate modification could reduce S/RV challenges by addressing a common characteristic; therefore the percentage reductions in S/RV challenge rates attributable to the candidate modifications are not necessarily additive.

5.2 Expected Spurious Blowdown Frequency Reduction

The expected reduction in frequency of spurious blowdowns alone through implementation of various candidate modifications is summarized in Table 5.3.

TABLE 5.1

SORV EVENT FREQUENCY

CANDIDATE MODIFICATION	TOTAL SORV EVENT FREQUENCY (NORMALIZED) BWR/4 w/3 Stage Target Rock Valve (Benchmark Plant)	BWR/3 without Isolation Condenser	BWR/2/3 with Isolation Condenser	BWR/5	BWR/6
(A): None	100	78	8	8	6
(B): Low Water Level Isolation Setpoint	69	50	8	6	*
(C): Low-Low Set Relief or Equivalent Manual Action	44	29	8	6	*
(D): (B + C)	35	21	8	5	*
(E): Low-Low Set Relief or Equivalent Manual Action + 2 Stage Target Rock Valve	11	7	2	N/A	N/A
(F): (C) + Early Removal of Leaky 3-Stage Target Rock Valve	22	15	4	N/A	N/A
(G): Low Water Level Isolation + Low-Low Set Relief or Equivalent Manual Action + 2 Stage Target Rock Valve	9	5	2	N/A	N/A
(H): (B) + (C) + Early Removal of 3 Stage Target Rock Valve	18	10	4	N/A	N/A
(I): 110% Steam Bypass†	88	73	4	6	4
(J): 110% Steam Bypass† + 2 Stage Target Rock Valve	22	18	2	N/A	N/A
(K): Isolation Condenser†	23	8	*	3	4
(L): Isolation Condenser† + 2 Stage Target Rock Valve	6	2	2	N/A	N/A

*Already implemented

† for comparison only

N/A Not Applicable since these plants are equipped with Crosby/Dikkers valves.

NOTES:

1. This table shows the SORV event frequency reduction due to some of the most effective contributors.

2. SORV event frequencies (normalized) shown above are obtained by multiplying total S/RV challenges (normalized) in Table 5.2 by relative SORV probability factor. In the case of Two Stage Target Rock Valves the benefit in reduction of spurious blowdowns (from Table 5.3) has been included above.

TABLE 5.2
S/RV CHALLENGES

TOTAL S/RV CHALLENGES (NORMALIZED)

CANDIDATE MODIFICATION	BWR/4 2/3 Stage Target Rock Valve (Bench-mark Plant)	BWR/3 without Isolation Condenser	BWR/2/3 with Isolation Condenser	BWR/5	BWR/6
None	100	78	8	63	47
Low Water Level Isolation Setpoint	69	50	8	48	*
Low-Low Set Relief or Equivalent Manual Action	44	29	8	49	*
Feedwater Runback	91	69	7	58	44
Reduce Surveillance Test Error**	95	74	7	60	45
Reduce MSIV Test Frequency**	98	76	8	62	46
Feedwater Control System Modification	89	68	7	56	45
Feedwater System Improvement**	97	75	8	62	47
Turbine System Improvement**	85	67	6	53	38
Analog Transmitter/Trip Unit**	97	75	8	61	46
Improved Recirculation Flow Control	96	73	8	60	46
Isolation Condenser†	23	8	*	27	34
110% Steam Bypass†	88	73	4	47	28

*Already implemented

**See Note 3

† for comparison only

NOTES:

1. To obtain SORV event frequency (normalized) multiply the values in the table by relative SORV probability factor.
2. Relative SORV probability factor for various valves are as follows:

   3 Stage TR Valve: 1.0 — Dresser Electromatic: 0.25
   2 Stage TR Valve: 0.50 — Dikkers & Crosby: 0.125
3. The benefits due to various candidate modifications are not additive except where noted by **.
4. Values shown above may vary from plant to plant depending upon utility operating practice.

TABLE 5.3

REDUCTION IN SPURIOUS BLOWDOWN EVENT FREQUENCY

Candidate Modification	Percentage Reduction in IORV Events	Applicability
Eliminate S/RV Ground Faults	1-2%	All Plants
Improved Pneumatic Supply Control System	2-3%	Plants with Target Rock valves only
Revise Spring Setpoint to Increase Simmer Margin	5%	Plants with Target Rock valves only
More Stringent Leakage Criteria & Early Replacement of Leaking Valves	40-60%	Plants with 3-Stage Target Rock Valves Only
Replace Valve Topworks with Two-Stage Design	40-60%*	Plants with 3-Stage Target Rock Valves Only

*This modification also reduces the probability of the valve to stick open. See Table 5.1 for the total impact on SORV event frequency.

6. CONCLUSIONS

Adequate core cooling is maintained in a BWR following an SORV event even under degraded conditions. It follows, then, that reduction of the frequency of SRV events is not of great concern from the standpoint of assuring adequate core cooling.

It is concluded that BWR/2, BWR/3 with isolation condenser, BWR/5 and BWR/6 plants are expected to have a SORV frequency which is a factor of at least ten below that for the benchmark plant. The use of selected modifications from a list of candidates can produce a factor of ten reduction in stuck open relief valve event frequency for BWR/4 plants and BWR/3 plants without isolation condenser. It should be noted that additional candidate modifications may exist which could reduce SORV event frequencies but have not been addressed in this report.

7. REFERENCES

1. NEDO-24708A (December, 1980), "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," Sections 3.1.1, 3.2.1.

2. NEDO-24708A, Section 3.2.1.

3. NEDO-24708A, Section 3.2.2.

4. NEDO-24708A, Section 3.5.2.1.

5. BWR Emergency Procedure Guidelines, Rev. 1 (prepublication form), submitted January 31, 1981.

APPENDIX A

Participating Utilities

NUREG-0737 II.K.3.16

This report applies to the following plants, whose Owners participated in the report's development.

Utility	Plant
Boston Edison	Pilgrim 1
Carolina Power & Light	Brunswick 1&2
Commonwealth Edison	LaSalle 1&2, Dresden 2-3, Quad Cities 1&2
Georgia Power	Hatch 1&2
Iowa Electric Light & Power	Duane Arnold
Jersey Central Power & Light	Oyster Creek 1
Niagara Mohawk Power	Nine Mile Point 1&2
Nebraska Public Power District	Cooper
Northeast Utilities	Millstone 1
Northern States Power	Monticello
Philadelphia Electric	Peach Bottom 2&3, Limerick 1&2
Power Authority of the State of New York	FitzPatrick
Detroit Edison	Enrico Fermi 2
Long Island Lighting	Shoreham
Mississippi Power & Light	Grand Gulf 1&2
Pennsylvania Power & Light	Susquehanna 1&2
Washington Public Power Supply System	Hanford 2
Cleveland Electric Illuminating	Perry 1&2
Houston Lighting & Power	Allens Creek
Illinois Power	Clinton Station 1&2
Public Service of Oklahoma	Black Fox 1&2
Vermont Yankee Nuclear Power	Vermont Yankee
Tennessee Valley Authority	Browns Ferry 1-3; Hartsville 1-4; Phipps Bend 1-2
Gulf States Utilities	River Bend

BWR OWNERS' GROUP

EVALUATION OF

NUREG-0737 ITEM II.K.3.18

MODIFICATION OF AUTOMATIC DEPRESSURIZATION SYSTEM LOGIC

CONTENTS

APPENDIX

ABSTRACT

A study was performed to determine the feasibility and benefits of extending the operation of the BWR Automatic Depressurization System (ADS) to include transient events which do not result in a release of steam to the drywell. Five options, including retaining the current design, were considered. The current design, with implementation of the Emergency Procedure Guidelines, is adequate. However, an ADS modification is believed to reduce plant risk, and is therefore proposed. The results showed that the addition of a bypass of the high drywell pressure trip if reactor water level remains below the low pressure ECCS initiation setpoint for a sustained period, or elimination of the high drywell pressure trip, are the preferred concepts. Detailed implementation will require consideration of broader scope issues, such as the final resolution of ATWS (which may affect the ADS logic, and which is specifically not considered in this study).

1. INTRODUCTION

The feasibility and reliability assessment study reported herein addresses NUREG-0737 Item II.K.3.18 which states, "The Automatic Depressurization System (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme which should be considered is ADS actuation on low reactor vessel water level provided no HPCI or HPCS system flow exists and a low pressure ECC system is running. This logic would complement, not replace, the existing ADS actuation logic."

The automatic depressurization system, through selected safety/relief valves,* functions as a backup to the operation of the high pressure coolant systems [feedwater, High Pressure Core Spray (HPCS)/High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC)/Isolation Condenser (IC)] for protection against excessive fuel cladding heatup upon loss of coolant, over a range of steam or liquid line breaks inside the drywell. The ADS depressurizes the vessel, permitting the operation of the low pressure coolant systems [condensate, Low Pressure Coolant Injection (LPCI), Low Pressure Core Spray (LPCS)]. The ADS is typically activated automatically upon coincident signals of low water level in the reactor vessel, high drywell pressure, ** and low pressure ECCS pumps running. A time delay of approximately 2 minutes after receipt of the signals allows time for the automatic blowdown to be bypassed if the water level is restored (or to be bypassed manually if the signals are erroneous). The ADS can be manually initiated as well.

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* A few plants have dedicated ADS valves.

**A few plants do not require coincident signals.

For transient and accident events which do not directly produce a high drywell pressure signal and are degraded by a loss of all high pressure coolant systems, adequate core cooling is assured by manual depressurization of the reactor vessel. For these events, the operator has sufficient time to manually depressurize the reactor in order to permit operation of the low pressure ECCS in these highly degraded events. The intent of this NUREG-0737 item is to provide additional assurance of adequate core cooling for these additional events. This study evaluates the feasibility of automating the vessel depressurization for isolation events with and without a stuck open relief valve (SORV), and assesses the changes in overall plant risk resulting from such automation.

The intent of the NUREG-0737 item may be addressed in two ways: the ADS logic may be modified to assure adequate core cooling for these additional events, or the operator can be given specific guidance and training for performing manual actions under degraded conditions. Following the accident at Three Mile Island, this second course of action has been undertaken, resulting in the development of symptom-oriented Emergency Procedure Guidelines (EPG's). Implementation of the EPGs will improve the operator response to degraded transients by giving him explicit guidance under these conditions and a better awareness and understanding of the plant response as a result of improved training. The events in question are slow, well behaved, well understood transients which allow the operator sufficient time to actuate ADS if the situation warrants. In addition the operator has had extensive training in this class of events and will receive additional instruction with the implementation of the EPGs.

It was shown in NEDO-24708, "Additional Information Required For NRC Staff Generic Report on Boiling Water Reactors," Section 3.5.2.1, that the operator has at least 30 to 40 minutes to initiate the ADS and prevent excessive fuel cladding heatup for both of the above events. This minimum time represents a "worst case" situation starting from full power with equilibrium core exposure and complete failure of all the high pressure makeup systems. Lower initial core power, low fuel exposure, control rod drive leakage flow, or partial operation of the high pressure systems would significantly increase the time available for operator action.

In addition, the operator has explicit guidance for transients under severely degraded conditions with the implementation of emergency procedures based on the symptom-oriented Emergency Procedure Guidelines (EPG's). The symptom oriented procedures lead the operator through transients with increasing levels of degradation and give him specific guidance on when to initiate ADS if it is needed. Thus with the implementation of the new emergency procedures, the operator has an increased understanding of the system and can reliably perform the actions necessary to assure core cooling.

Transients which may require manual depressurization can be characterized by two general events: 1) reactor isolation with loss of normal inventory maintenance, and 2) reactor isolation with loss of normal inventory maintenance degraded by a stuck open relief valve. Both of these events were considered in the design of the modified ADS even though the second event type is beyond the current design basis which assumes a single failure. Transients resulting in a loss of feedwater for the most part also cause a reactor isolation due to low reactor water level. Steamline breaks outside the containment result in an isolation due to several signals (e.g., high steamline flow, high steam tunnel radiation or temperature). For transients that do not cause an isolation, the main condenser is available for depressurization and the ADS is not required. Therefore such events are not included in this study.

The isolation with SORV is considered separately because the additional inventory loss through the open valve increases the required high pressure makeup flow. The additional loss also reduces the time available for the operator to manually depressurize the vessel if necessary.

Four ADS logic modifications are considered, and the current logic is reviewed using the same basis as the modifications. These five options are evaluated as to system performance, feasibility of implementation, cost of additional design and hardware, and impact on plant operation.

Section 2 gives a detailed description of each of the logic alternatives considered. Section 3 demonstrates the acceptability of the system performance of each of the modifications. Section 4 discusses the reliability of ADS actuation with increased automation of the ADS. The advantages and disadvantages of each option and the feasibility of implementation are discussed in Section 5.

This study is devoted to the feasibility of concepts as opposed to a detailed design assessment. The goal is to determine, using simple concepts and arguments, whether or not ADS should be further automated, and if so, which conceptual design is most favorable for further development. Detailed design implementation of any changes will require consideration of broader scope issues, such as the final resolution of ATWS (which may affect the ADS logic, and which is specifically not considered in this study) and the desirability of other changes in the ADS based on recent studies in support of the Emergency Procedure Guidelines.

2. ADS LOGIC OPTIONS CONSIDERED

Five ADS logic options are considered: the current design, and four logic modifications. These four modifications are 1) elimination of the high drywell pressure trip, 2) addition of a timer that bypasses the high drywell pressure trip if reactor water level is low for a sustained period, 3) addition of a suppression pool temperature trip in parallel with the high drywell pressure trip, and 4) the addition of high pressure system flow measurement and logic in parallel with the high drywell pressure trip.

2.1 CURRENT DESIGN

The first option to be considered is the present ADS logic design. With the implementation of the symptom-oriented EPG's, the current logic satisfies the intent of the NUREG-0737 item and its incorporation in the NSSS design meets all of the applicable design and licensing requirements. It is not obvious that the advantages of further automation outweigh the disadvantages. The current design is thus a viable option in its own right. Figure 2-1 shows the current ADS logic design for a typical plant. The design requires a LOCA signal consisting of concurrent high drywell pressure and low reactor water level in order to actuate the ADS. The actuation sequence depends on receipt of the high drywell pressure (2 psig) signal. Once this signal is received, it is sealed into the initiation sequence and does not reset even if the high drywell pressure subsequently clears. The next signal is low water level (low pressure ECCS actuation level).* When this is satisfied, the logic confirms the water level is indeed below the scram water level (to prevent spurious actuations) and starts the 120 second delay timer. The

*In many plants this level is commonly referred to as "Level 1" and will be referred to as such in this report.

timer is reset if the low water level trip clears before the timer times out. The timer also allows the operator time to bypass the automatic blowdown if the conditions have corrected themselves or if the signals are erroneous. To complete the sequence, the low pressure ECCS pumps are automatically checked to provide some assurance that makeup water will be delivered to the vessel once it is depressurized.

Drywell cooling is lost for a number of plants when the reactor water level reaches low level (Level 1). The loss of cooling results in a heatup of the drywell air space (if the operator is unable to restore cooling) and subsequent pressurization of the drywell above the 2 psig required for ADS initiation and actuating the ADS. Thus in plants where drywell cooling is lost on low level, the current logic will act as a satisfactory backup to manual action for the events considered.

The symptom-oriented EPG's are written incorporating the current ADS logic. For the events in question, the operator is given explicit instructions on when to manually depressurize the vessel if the high pressure systems cannot maintain inventory. These instructions are based on the conditions of vessel pressure and water level, and the availability of high and low pressure injection systems. As a result of the symptom-oriented nature of these guidelines, the appropriate operator actions are specified for all levels of degradation and plant conditions.

2.2 ELIMINATE HIGH DRYWELL PRESSURE TRIP

The second option is simply to eliminate the high drywell pressure trip from the current logic sequence. The ADS sequence would then be activated on low reactor water level only. The remainder of the sequence remains unchanged. The effect of high drywell pressure on other safety systems, such as reactor scram and the ECCS that initiate on high drywell pressure, are unaffected. The logic design for this alternative is shown in Figure 2-2.

2.3 BYPASS HIGH DRYWELL PRESSURE TRIP

The third option is to bypass the high drywell pressure requirement after a set timer delay. This is accomplished by installing a second ("bypass") timer actuated on low water level. When this timer runs out, the high drywell pressure trip is bypassed and the ADS is initiated on water level alone. The additional logic does not affect the high drywell pressure-low water level initiation sequence for pipe breaks inside the drywell. Once the timer runs out, this option becomes the same as that discussed in Section 2.2. The only difference is that for events which do not produce a high drywell pressure signal, the bypass timer gives the operator additional time to bypass the automatic blowdown if the situation is corrected or ADS is not needed for some other reason.

Figures 2-3A and 2-3B show the logic for this alternative, with the bypass timer started at either scram or low (Level 1) water level. A time delay of approximately eight minutes was chosen for preliminary evaluations. Starting the 8-minute delay at scram water level results in the high drywell pressure trip being bypassed at about the same time as the water level outside the shroud reaches the top of the active fuel under the worst case conditions of an isolation event with an SORV from full power and no high pressure makeup. If the water level recovers to the initiation level of either timer, that timer is automatically reset.

The exact delay of the bypass timer is subject to a more precise evaluation, which is based on avoidance of excessive fuel cladding heatup using the realistic evaluation models of NEDO-24708A. It is specifically noted that the current system, which requires operator action for certain transients, is in compliance with all applicable design and licensing bases. The modification is regarded as a backup to operator action.

Starting the bypass timer at low water level (Level 1) allows the operator the greatest time to control the system manually and still assure automatic depressurization in time to prevent excessive fuel heatup even under the worst case conditions described above. For BWR/1-3, the bypass timer would be reset if the low water level trip is recovered. For BWR/4-6, once the bypass timer times out, the bypass permissive would be sealed in and the bypass timer would not reset. This accounts for the lower Level 1 trip elevation and prevents repeated partial core uncovery.

2.4 ADD SUPPRESSION POOL TEMPERATURE TRIP

The fourth option is to add a suppression pool temperature trip in parallel with the high drywell pressure trip. The ADS initiation sequence would then be initiated by either high drywell pressure or a rise in suppression pool temperature, with the remainder of the logic unchanged. Just as high drywell pressure is symptomatic of a loss of inventory inside the drywell, a rapid rise in suppression pool temperature indicates inventory loss through the relief valves or a break in the drywell. There are two conditions that could be used to provide the pool temperature permissive; when the pool temperature reaches a specified value, or if the pool heats up faster than a specified rate. The heatup rate trip would require a simple data processing system that would record the present pool temperature, compare it to the reading ten minutes previous to determine the heatup rate, and store the present information to be used later. As shown in Figure 2-4, the remainder of the logic is unchanged from the current design.

2.5 NO HIGH PRESSURE SYSTEM FLOW

The fifth option is to measure high pressure system flow (Feedwater, HPCI/HPCS, RCIC/isolation condenser), and bypass the high drywell pressure trip if no flow is present in all of these systems. This option was identified in NUREG-0737. The remainder of the logic would remain unchanged. Since this signal would complement the current ADS logic, the ADS is not inhibited for LOCA events if a high pressure system is operating. The additional logic is shown in Figure 2-5. There are two methods of accomplishing the high pressure flow measurement. The more direct method is to actually measure the flow of each high pressure system, with a minimum flow for the system (approximately full rated flow for the high pressure ECCS and about 10 percent of full feedwater) required to block the "no high pressure flow" permissive. This method gives the greatest assurance that makeup water is reaching the vessel.

However, trips based on flow measurement may not be as reliable as desired. A theoretically more reliable but less direct measurement scheme would use pump operation and valve position to infer the lack of high pressure system operation. Because the scheme is less direct, the overall ADS actuation reliability may or may not be improved. During loss of feedwater transients, the high drywell pressure signal is bypassed and the ADS initiation sequence is started for the short time between the time feedwater is lost and the time the water level falls to the high pressure ECCS initiation trip (Level 2) and the high pressure ECCS start up. During this time, blowdown would be prevented by the Level 1 trip (or the 120 second timer for BWR/1-3 because the high and low pressure ECCS start at the same water level). Once one of the high pressure systems start up, the permissive is removed and the initiation sequence is halted. If no high pressure systems start up, level alone starts the timer and initiates ADS.

3. SYSTEM PERFORMANCE

This section analyzes each of the options as to whether it ensures adequate core cooling for isolations and SORV's. For these analyses it is assumed that all high pressure systems have failed and the ADS must depressurize the vessel and allow the low pressure systems to inject. The modeling used in these analyses is the same as that used in NEDO-24708.

3.1 CURRENT DESIGN

The current design does not directly satisfy the above criteria because the logic does not actuate the ADS specifically for the events considered. However, as stated earlier, the operator has 30 to 40 minutes to depressurize the vessel under these worst case conditions and prevent core damage. This is sufficient time to assess the situation and take the necessary actions.

In addition to the time available to the operator to blow down the vessel, ADS actuation is assured for these events in plants which lose the drywell cooling on a low water level signal. The loss of drywell cooling will cause the drywell temperature to increase, and consequently the drywell pressure to rise. The drywell pressure reaches the 2 psig setpoint required for ADS initiation in 5 to 10 minutes resulting in ADS actuation if the water level has not been restored above Level 1. The time required for the drywell to heat up and pressurize is insensitive to the power level of the reactor or the ambient conditions inside and outside the containment. Thus ADS actuation would most likely occur without operator action within about 10 minutes after Level 1 even for events which do not directly pressurize the drywell. Analyses presented in NEDO-24708 (Figure Group 3.5.2.1-33) demonstrate that adequate core cooling is assured for isolation events with the ADS blowdown delayed 10 minutes after Level 1. Figure Group 3-1 shows the same analysis assuming an SORV. The results shown are typical of BWR/4-6. These results bound

BWR/1-3 because of the latter's higher ADS level trip. Because the trip is at a higher level, the resulting core uncovery is shorter and the core heatup is less.

3.2 ELIMINATE HIGH DRYWELL PRESSURE TRIP

By eliminating the high drywell pressure trip, the system response to the transients considered in this study is similar to that for small LOCA events inside the containment. With a pipe break inside the drywell, the high drywell pressure trip occurs before the low water level trip. Eliminating the high drywell pressure trip can be thought of as assuring this signal exists for all transients. The water level response for an isolation event is bounded by small break LOCA analyses where the majority of the inventory lost is not through the break but through the cycling of relief valves. The water level response to a stuck open relief valve is essentially the same as that shown for a small recirculation line break. Thus the break spectrum analyses provided in Safety Analysis Reports provide verification that adequate core cooling would be assured for this option.

3.3 BYPASS HIGH DRYWELL PRESSURE TRIP

There are two cases studied for this option: 1) the 8-minute bypass timer started at scram water level, and 2) the 8-minute bypass timer started at low water level (the current ADS level trip). For the first case the eight minute bypass timer delay plus the two minute bypass timer delay is consistent with the operator action assumed in Safety Analysis Reports. The analyses presented in these reports for the steamline break outside the containment demonstrate adequate core cooling is assured for this case. Figure Group 3-2 presents typical results for an isolation with an SORV. The second case results in the ADS actuation occurring approximately ten minutes after low level (Level 1). The analyses and justification presented in Section 3.2 are applicable here.

3.4 SUPPRESSION POOL TEMPERATURE TRIP

Since the suppression pool temperature trip performs the same function as the high drywell pressure trip, the discussion and justification presented in Section 3.2 is applicable for this option provided that the system would be designed to reliably produce a high pool temperature signal before low reactor water level is reached for all events in question (see Section 5.4).

3.5 NO HIGH PRESSURE MAKEUP FLOW

For this modification, if no high pressure system flow is indicated, the high drywell signal is bypassed. Section 3.2 is applicable to this situation. For isolation and SORV events where adequate high pressure injection is indicated, the high drywell pressure trip is not bypassed and ADS is not actuated. For isolation events, any one of the high pressure systems alone (feedwater, HPCI/HPCS, RCIC/IC) is adequate to maintain adequate core cooling.

For an SORV, the additional lost inventory must be made up by the high pressure system. The HPCI and HPCS systems have adequate flow capacities to make up the lost inventory. The RCIC has a much smaller capacity but can provide inventory makeup and in some plants can actually prevent core uncovery. The isolation condensers (IC) do not provide additional makeup to replace that lost through the SORV and cannot maintain the level. Figure Group 3-3 shows the results for an isolation with a stuck open relief valve with only RCIC available for BWR/6. As shown for the plant analyzed, the RCIC has sufficient capacity to maintain the water level above the core. Figure Groups 3-4, 3-5 and 3-6 show the same event for a BWR/4 and a BWR/3 with only RCIC and a BWR/2 with only the isolation condenser available. Because for these plants the RCIC or IC does not make up the inventory lost through the valve, the water level slowly falls. The depressurization of the vessel due to the SORV and the operation of the IC soon allows the low pressure systems to inject and assure adequate core cooling without ADS actuation. Thus it appears that adequate core cooling is assured by this alternative, but that conclusion would have to be confirmed by a detailed analysis if the modifications were implemented.

3.6 SUMMARY

The current logic meets all of the applicable design and licensing requirements, and in addition the current ADS logic will be sufficient for the events in question when accounting for the high drywell pressure trip resulting from the loss of drywell cooling on low water level. Each of the four ADS logic modifications provides adequate core cooling by automating vessel depressurization for isolations and SORV events, so all are equally viable from a system performance viewpoint.

4.0 RELIABILTY ASSESSMENT

This section assesses each of the first four options described in Section 2 as to whether it reliably actuates the ADS for the events considered, and whether it increases the probability of spurious or inadvertent actuation. The fifth option (ADS permissive if no high pressure system flow is available) was not considered in this analysis since it is concluded in Section 5 that the approach is not feasible. Also included is a discussion of the expected improvement in operator reliability as a result of implementation of the EPG's and improved operator training.

4.1 RELIABILITY OF ADS INITIATION

4.1.1 CURRENT LOGIC

With the current logic, the operator is relied upon to manually initiate the vessel depressurization if required by failure of the high pressure injection systems to maintain level for isolations and SORV's. These events are slow, uncomplicated, and well understood, for which the operator is extensively trained and is familiar with both the equipment and the overall system response. Following the TMI accident, reviews of emergency procedures and operator training indicated that the operator reliability could be improved for degraded situations by providing better guidance and training. The symptom-oriented EPG's were developed as a result of these reviews. The EPG's give the operator the additional guidance required for degraded situations. Informal demonstrations of the EPG's on control room simulators using both new and experienced operators have shown a significant improvement in the reliability of operator performance under degraded conditions. Thus, implementation of the symptom-oriented EPG's and improved operator training in the use of the new procedures results in an improved probability that the operator will depressurize the vessel if required for the events considered.

In addition, for plants which lose drywell cooling on low reactor water level and satisfy the high drywell pressure trip as a result of the drywell heatup, a backup to operator action is provided.

4.1.2 LOGIC MODIFICATIONS

The logic modifications considered either eliminate the high drywell pressure permissive or provide instrumentation that serves as an alternate to the high drywell pressure permissive for transients that do not pressurize the drywell. This instrumentation can be designed to be more reliable than the operation of the ADS valves themselves. Eliminating the drywell pressure trip or installing the timed bypass of the drywell pressure trip are simple changes that utilize hardware and instrumentation similar to that used in the current logic. Thus, these schemes have about the same reliability for the transients considered as the current logic has for LOCAs. The suppression pool temperature monitoring system required for the other alternative might be designed to provide roughly the same level of reliability. Thus, the overall ADS reliability is approximately the same regardless of which modification is considered.

4.2 SPURIOUS AND INADVERTENT ACTUATION

Fault tree analyses of the various alternatives were performed in order to estimate the probability of unneeded depressurization. These analyzes included inadvertent manual depressurization, false signals, and testing and maintenance errors. The results of the studies show that the probability of an unneeded depressurization is not significantly affected by the additional modifications to the ADS logic.

The probability of inadvertent operator actuation of the ADS when depressurization is not needed is believed to be slightly improved if the system is automated for the events considered. The manual inadvertent initiation probability is higher for the current ADS logic because the operator, knowing he is responsible for manual depressurization, may be more apt to err in the conservative direction and depressurize the vessel.

The probability of spurious actuation due to equipment failure or testing/maintenance error is believed to be slightly greater for the two logic modifications where the high drywell pressure permissive is bypassed or eliminated. For the suppression pool temperature permissive option, the probability of spurious actuation is believed to be about the same as that for the current logic.

Because these results are based on conceptual logic designs, it is difficult to precisely quantify these effects, but it is believed that the decrease in probability of inadvertent manual depressurization and the increase in the probability of spurious actuation are approximately offsetting. The overall probability of unneeded actuation is believed to be slightly improved. Thus from the standpoint of inadvertent actuation, all the alternatives are about the same.

4.3 SUMMARY

Each of the alternatives considered reliably actuates the ADS when required, and there is no significant difference between the alternatives from the standpoint of inadvertent actuation.

5. FEASIBILITY OF IMPLEMENTATION

This section compares the advantages and disadvantages of each of the alternatives. Included is a discussion of the practicality of each concept, the resources required for implementation, the impact on plant operations, and the impact on the overall plant design. A comparative summary of the feasibility of each option is presented in Table 5-1. In this table the current design is used as the basis for comparison. Three major areas are addressed in this study: ADS performance, installation and maintenance, and impact on plant operation.

5.1 CURRENT DESIGN

5.1.1 ADS Performance

With implementation of the EPG's, the current ADS logic design meets the intent of the NUREG-0737 item of assuring adequate core cooling for the additional transients considered, and has several advantages. It is a system with which the operator has had significant training and experience. In addition, the operator reliability in degraded situations will be improved with implementation of the EPG's, which give explicit instructions on when to initiate the vessel blowdown.

Isolations and SORV's are slow, uncomplicated, and familiar transients, for which the operator is extensively trained and for which he is familiar with both the equipment response and the overall system response and behavior. Because of this familiarity, the reliability of the operator to perform the required actions is high, to the point of being reflexive. Sufficient time (30 to 40 minutes) is available to assess the overall plant situation and initiate blowdown if required. In addition, the current design allows the operator the flexibility to control the systems as required by the plant conditions and symptoms.

5.1.2 Installation and Maintenance

The current design forms the basis for comparison of the other four options with respect to installation and maintenance.

5.1.3 Impact on Plant Operation

The current design forms the basis for comparison of the other four options with respect to impact on plant operation.

5.2 ELIMINATE HIGH DRYWELL PRESSURE TRIP

5.2.1 ADS Performance

Elimination of the high drywell pressure trip for the ADS is a simple modification that is effective in initiating the ADS if it is needed to assure core cooling. If the high pressure systems are unable to maintain the water level, the ADS is actuated and the low pressure systems provide core cooling.

5.2.2 Installation and Maintenance

Implementation of this modification requires only a few simple wiring changes with no additional hardware additions. Maintenance and testing is somewhat easier as fewer trip circuits need to be tested and repaired.

5.2.3 Impact on Plant Operation

The primary drawback to this alternative is that the removal of one of the trip signals results in a slight increase in the probability of spurious actuation as a result of improper testing or due to spurious signals. This does not present a core cooling concern, since the low pressure systems would provide adequate inventory makeup; however, it does tend to decrease the plant availability, and increase the duty cycles on the vessel and containment due to unneeded depressurizations.

However, the probability of ADS operation when not really needed is believed to be slightly improved. The two effects are believed to be approximately offsetting.

An additional drawback is present in the earlier (BWR/1-3) plant designs having both the high and low pressure ECCS actuated at one common reactor water level. This common actuation level allows the high pressure systems about two minutes to start and restore the water level above the trip setpoint before the ADS is actuated. If the level is not restored before the ADS timer times out, the vessel is blown down. The RCIC and isolation condensers for these plants are sized to prevent core uncovery for isolations, but they are not large enough to restore the level above the initiation setpoint and reset the ADS within the allotted two minutes. Thus, with an isolation and loss of high-capacity, high-pressure makeup, the RCIC could bring the water level under control and ADS would not be needed; however with this modification ADS would occur unless manually defeated. Requiring it to be manually defeated is undesirable from the human engineering standpoint.

5.3 BYPASS HIGH DRYWELL PRESSURE TRIP

5.3.1 ADS Performance

This option is essentially the same as the preceding one, however, the addition of the delay in bypassing the high drywell pressure trip gives the high pressure systems additional time to recover the water level and reduces the chance of undesired ADS actuation described earlier for BWR/1-3s. Two variations are considered, one with the bypass timer started at scram water level and one with the bypass timer started at the low pressure ECCS initiation level (Level 1).

The scram water level trip has the advantage of actuating ADS before the core is uncovered and thus minimizing core uncovery. However, since transients resulting in the water level dipping below the scram water level are fairly common, the ADS system would be challenged more frequently than with the lower level trip. An increase in the probability of spurious actuation would thus result. Starting the bypass timer at the lower water level (Level 1) does not present this problem and will provide the operator additional time to assess plant conditions while still providing adequate core cooling; however, the likelihood of core uncovery before ADS actuation is increased. It is judged that starting the bypass timer at low water level (Level 1) is preferred.

5.3.2 Installation and Maintenance

The cost of this modification is low with the installation of the necessary hardware easy to perform.

The additional maintenance and surveillance is minimal as the system does not have complicated interfaces with other systems. In general this alternative is very similar to the preceding option.

5.3.3 Impact on Plant Operation

The impact on plant operation is about the same as that presented for the previous modification. However, the addition of the bypass timer gives the operator additional time to initiate the high pressure coolant injection systems, and thus precludes unnecessary vessel blowdowns.

5.4 SUPPRESSION POOL TEMPERATURE TRIP

5.4.1 ADS Performance

The major advantage of adding a suppression pool temperature trip in parallel with the high drywell pressure trip is that a rise in suppression pool temperature is indicative of an inventory loss from the reactor coolant system, analagous to the high drywell pressure signal. Since a rise in suppression pool temperature is virtually assured for isolation and SORV events, this option automates vessel depressurization for these events while including a permissive signal in the ADS logic which would reduce the likelihood of spurious ADS actuation relative to the second option (elimination of the high drywell pressure trip). The advantages and disadvantages of that option are thus applicable here, particularly the problem of unnecessary ADS actuation for the earlier plants. In addition, the additional hardware required for this option is complex, reducing the reliability of this system to perform on demand. The suppression pool temperature monitoring and averaging equipment must be precise anough to measure the relatively slow pool heatup in order to give the ADS a permissive signal and actuate the ADS in a timely manner. Variations in suppression pool mixing as a function of SRV discharge location and RHR operation raise the possibility of the temperature monitoring system "missing" the local temperature rise resulting from an SORV or detecting a non-representatively high local temperature. Operation of the pool cooling during the initial stages of the transient does not, however, affect the initial heatup rate of the pool due to the low temperature differences across the RHR heat exchangers.

In Section 2.4, two approaches were suggested for providing the pool temperature permissive; measuring the pool temperature heatup rate or giving the permissive once the pool temperature reaches a specified value. Measuring the pool heatup rate requires additional hardware compared to the absolute trip. The rate measurement trip is less reliable than the absolute trip, however the rate measurement automatically compensates for normal changes in the pool temperature. The rate measurement scheme can be adequately and reliably approximated by the simpler absolute trip by periodically resetting the trip to reflect a predetermined temperature difference above the actual pool temperature. A detailed analysis would have to be performed to determine an acceptable temperature rise or rate trip.

5.4.2 Installation and Maintenance

The temperature measurement concerns expressed in the previous section could be reduced by installing a large number of temperature sensors and sophisticated averaging and monitoring equipment. Such hardware would be very expensive to purchase and install. Maintenance and surveillance testing would be complex and would increase exposure to maintenance personnel.

5.4.3 Impact on Plant Operation

Due to the complexity of the system, its overall reliability is judged to be somewhat lower than that for the other options. Though this concept is an indication of inventory loss, this benefit is far outweighed by the disadvantages of implementation and operation.

5.5 NO HIGH PRESSURE SYSTEM FLOW

5.5.1 ADS Performance

The measurement of high pressure system flow gives a fairly direct indication of the unavailability of inventory makeup flow. A lack of high pressure makeup flow would in effect verify the falling water level indication in the vessel. In addition, this alternative does not cause the undesired ADS for the earlier (BWR/1-3) plants when the RCIC or isolation condenser are working but do not clear the ADS water level trip. If high pressure injection systems are operating, the ADS logic would not be initiated without a high drywell pressure signal. The high pressure systems for these plants, however, must start up within two minutes in order to prevent ADS actuation, because the high pressure ECCS and ADS use the same water level initiation setpoint. This time limit effectively eliminates any chance the operator has of restarting the high pressure systems and thus increases the chance of needlessly actuating the ADS.

In addition to the limited time this concept gives the operator for restarting systems, the flow measuring system is vulnerable to a high pressure system pipe break or incorrect valving downstream of the measuring point. The flow would register and no permissive signal would be generated, even if the water was not reaching the vessel. The major disadvantages of this option are the difficulty of determining *a priori* what the proper flow criterion is, and the difficulty of measuring such low flows in high flow systems. For example, only about 3% of rated feedwater flow is required to maintain the reactor water level for isolation and SORV events. It is difficult to accurately and reliably measure such a small flow using devices that would not interfere with normal operation. In addition, HPCI or HPCS normally cycle on and off as required to maintain the reactor water level. Because of the high capacity of these systems, the water level is quickly restored and the "off" cycle is long compared to the "on" cycle, erroneously initiating the ADS sequence. Thus the flow measurement scheme has a low probability of producing a true signal that reflects the availability of the high pressure systems.

5.5.2 Installation and Maintenance

The other drawbacks of this alternative are the high cost and difficulty of installation of the hardware required for the flow measurement. Additional flow taps would be required to bring the present flow indications up to safety grade requirements. Maintenance and surveillance would be comparatively difficult.

5.5.3 Impact on Plant Operation

The reliability of the system would be governed by the reliability of the flow measuring instrumentation. The reliability of the system might or might not be improved by inferring the lack of flow from pump operation and injection valve position instead of measuring flow directly. Spurious operation of the ADS would be more likely since for every loss of feedwater transient, the ADS sequence would be initiated for a short time during the beginning of the transient before the high pressure ECCS have been signaled to start. The ADS sequence would also be reinitiated during the subsequent cycling of the high pressure systems as they maintain reactor water level. Because of these impacts on plant operation, and the cost of installation, this option is less desirable than the first four. For these reasons this option was not addressed in Section 4.

5.6 SUMMARY

Based on the above study, the first three concepts presented (the current logic, eliminating the high drywell pressure trip, and bypassing the high drywell pressure trip) are viable options for the ADS design. The current logic, though it does not explicitly address the NUREG-0737 item, meets all of the applicable design and licensing requirements. The operator has sufficient time to assess the plant conditions and manually depressurize the vessel if warranted. In plants which lose drywell cooling on low reactor water level, ADS operation will occur without manual action for isolations and SORVs. The fourth option (the suppression pool temperature trip) provides no additional benefit as to the reliability of ADS initiation for the events considered when compared to the other options and add needless complexity to the overall plant design and operation. The fifth option (the high pressure system flow measurement) is not recommended because of its impact on plant operation.

The second option, eliminating the high drywell pressure trip works well for BWR/1-6. The third option, the bypass timer, is suited for BWR/1-3, with the timer started at the ADS water level initiation setpoint. The timer should reset when the water level trip clears. This alternative is also suitable for BWR/4-6; however, the bypass timer should not reset once it has run out. Starting the timer at low water level rather than at scram water level is recommended. Either of these options is a low cost, easily implemented means of automating vessel depressurization for outside breaks, isolations, and SORVs if required.

6. CONCLUSIONS

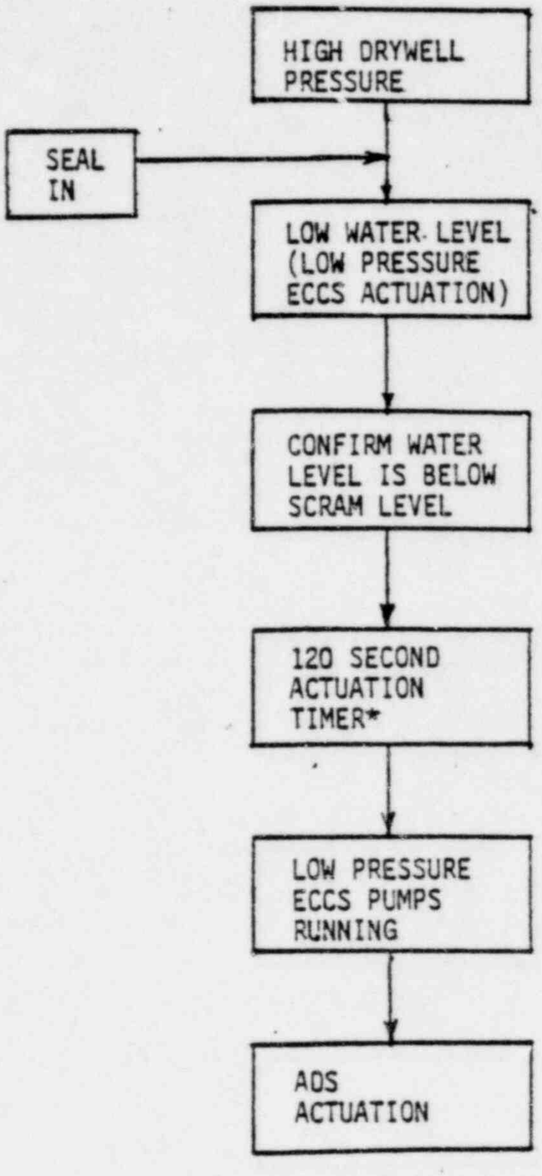
The intent of the NUREG-0737 item is to provide more assurance of adequate core cooling in the event of transients and accidents not producing a high drywell pressure signal (e.g. isolation, SORV) under conditions such that high-pressure makeup systems are unable to maintain reactor inventory. The intent may be satisfied in two ways: the ADS logic may be modified to automate the depressurization for these additional events, or the operator may be given specific guidance and training for performing manual actions under degraded conditions.

This second course of action has already been undertaken with the implementation of the symptom-oriented Emergency Procedure Guidelines. The transients considered are slow and well understood, and the operator has sufficient time to assess the plant conditions and initiate the depressurization. In addition, it was shown that for plants which lose drywell cooling on low level, the current logic will act as a satisfactory backup to manual action for the events in question.

A feasibility study of possible ADS options showed that of the five alternatives presented, the first three (the current design with implementation of the EPG's; eliminating the high drywell pressure trip; and bypassing the high drywell pressure trip after runout of a timer started at the low pressure ECCS initiation level) were the most viable. The fourth option (a suppression pool temperature trip in parallel with the high drywell pressure trip) is also feasible; however, the added complexity of the system provides no additional benefit as to the reliability of ADS actuation when compared to the first three options. The fifth option (bypassing the high drywell pressure trip if low flow of high pressure injection systems is indicated) was shown to be impractical. Of all the alternatives, addition of a bypass to the high drywell pressure trip if reactor water level remains below the low pressure ECCS initiation setpoint for a sustained period, or elimination of the high drywell pressure trip, are the preferred solutions.

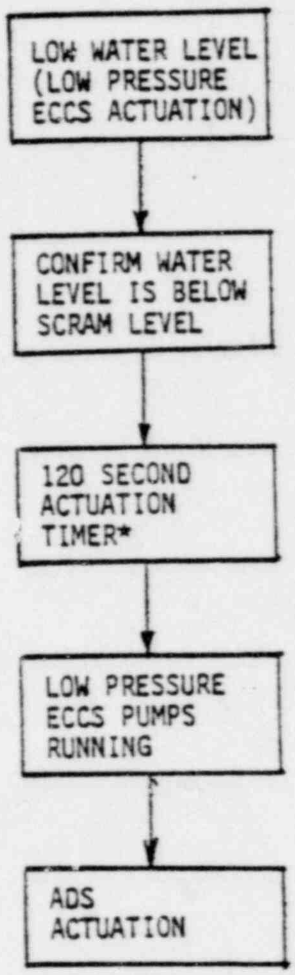
This report does not attempt to demonstrate the absolute reduction in plant risk due to the ADS modification, although it is believed that a reduction will be achieved. The ADS modification is proposed because it provides more assurance of core cooling in the event of isolations, it brings the automatic system operation more closely in line with the Emergency Procedure Guidelines, and it does not increase the probability of rapid depressurization if such is not needed.

It is stressed that detailed implementation will require consideration of broader scope issues, such as the final resolution of ATWS (which may affect the ADS logic, and which is specifically not considered in this study).



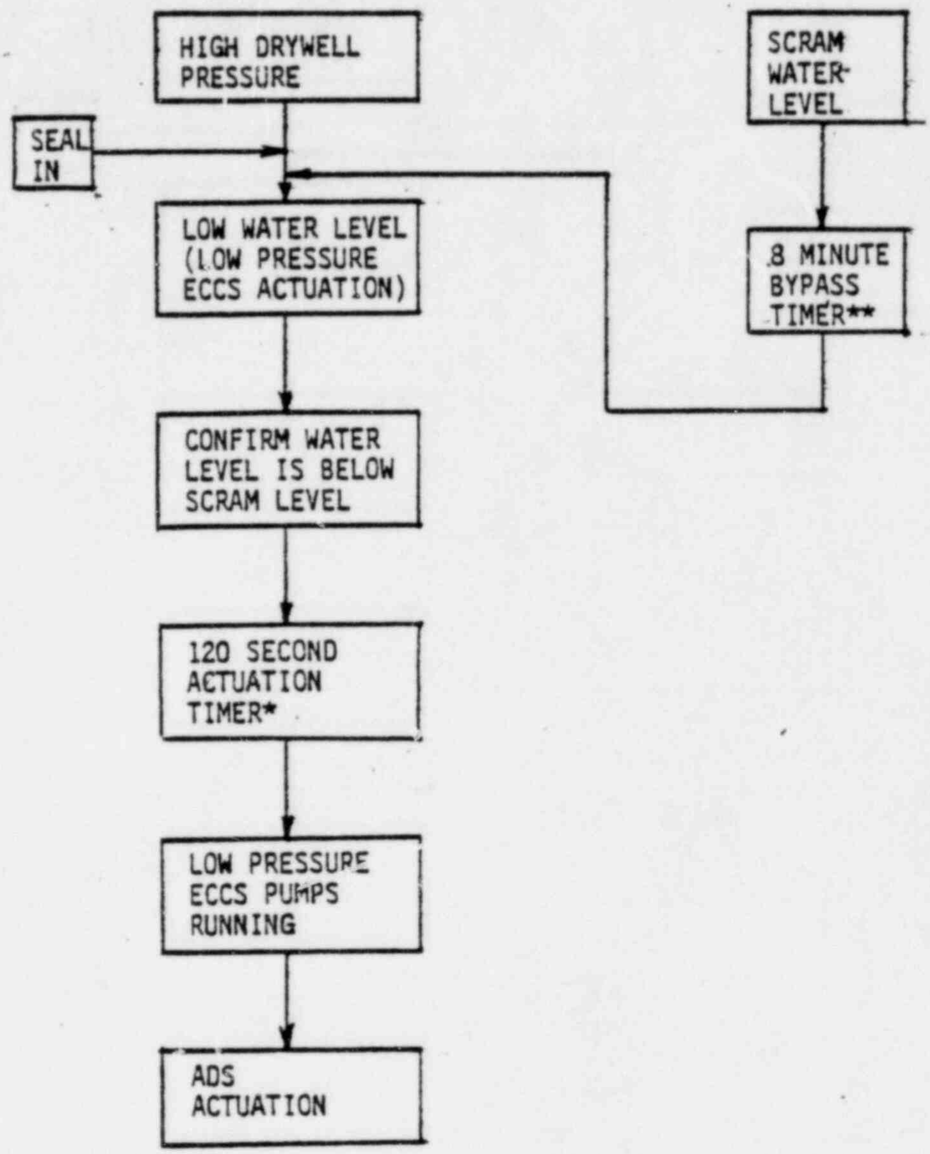
*120 SECOND ACTUATION TIMER WILL RESET IF LOW WATER LEVEL TRIP RECOVERS BEFORE IT TIMES OUT. THE TIMER WILL RESTART IF THE LOW LEVEL SIGNAL OCCURS AGAIN.

FIGURE 2-1: CURRENT ADS LOGIC FOR A TYPICAL BWR



*120 SECOND ACTUATION TIMER WILL RESET IF LOW WATER LEVEL TRIP RECOVERS BEFORE IT TIMES OUT. THE TIMER WILL RESTART IF THE LOW LEVEL SIGNAL OCCURS AGAIN.

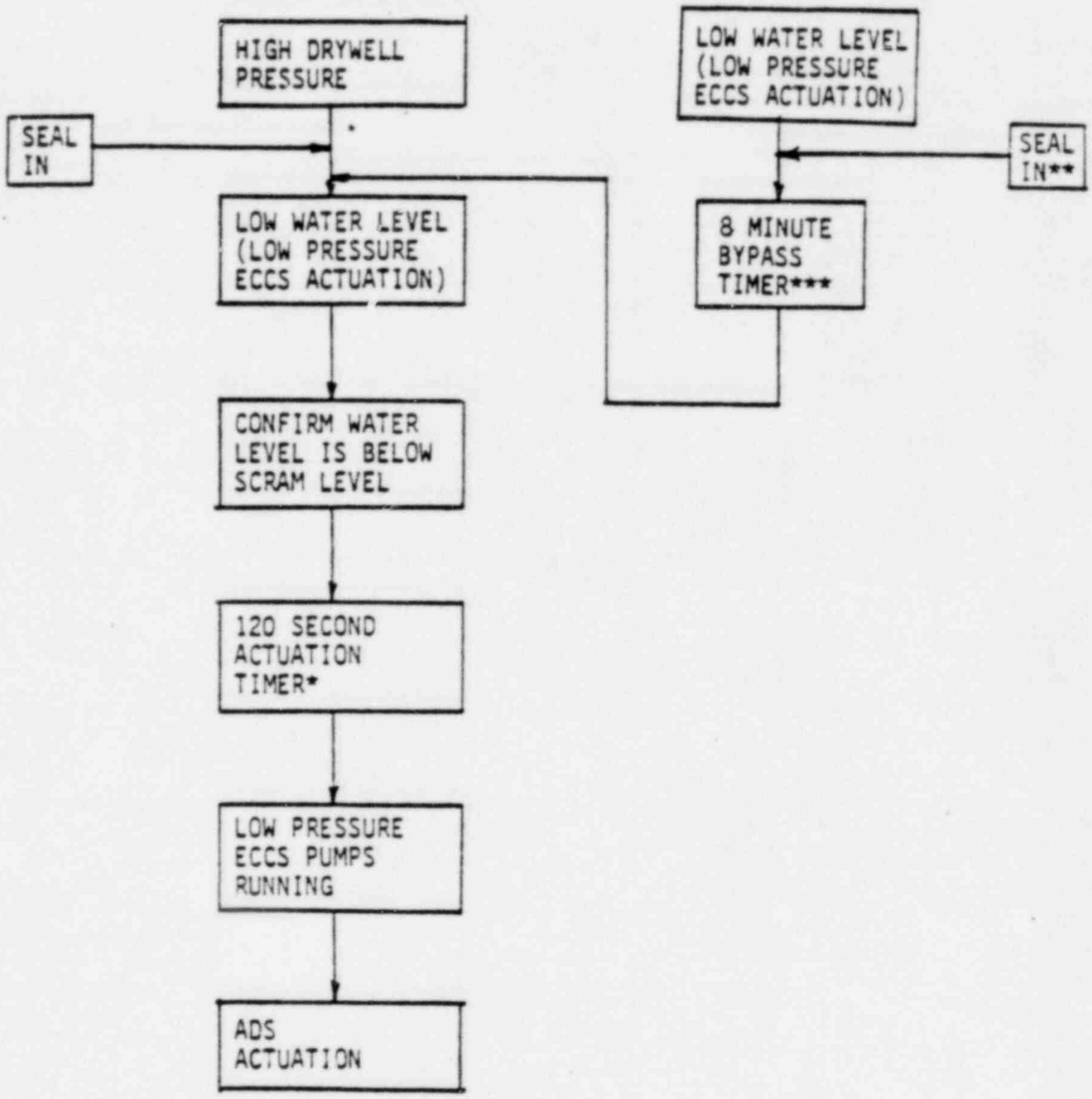
FIGURE 2-2: HIGH DRYWELL PRESSURE TRIP ELIMINATED



*120 SECOND ACTUATION TIMER WILL RESET IF LOW WATER LEVEL TRIP RECOVERS BEFORE IT TIMES OUT. THE TIMER WILL RESTART IF THE LOW LEVEL SIGNAL OCCURS AGAIN.

**RESET SAME AS 120 SECOND ACTUATION TIMER.

FIGURE 2-3A: ADD DELAYED BYPASS OF HIGH DRYWELL PRESSURE TRIP WITH BYPASS TIMER STARTED AT SCRAM WATER LEVEL

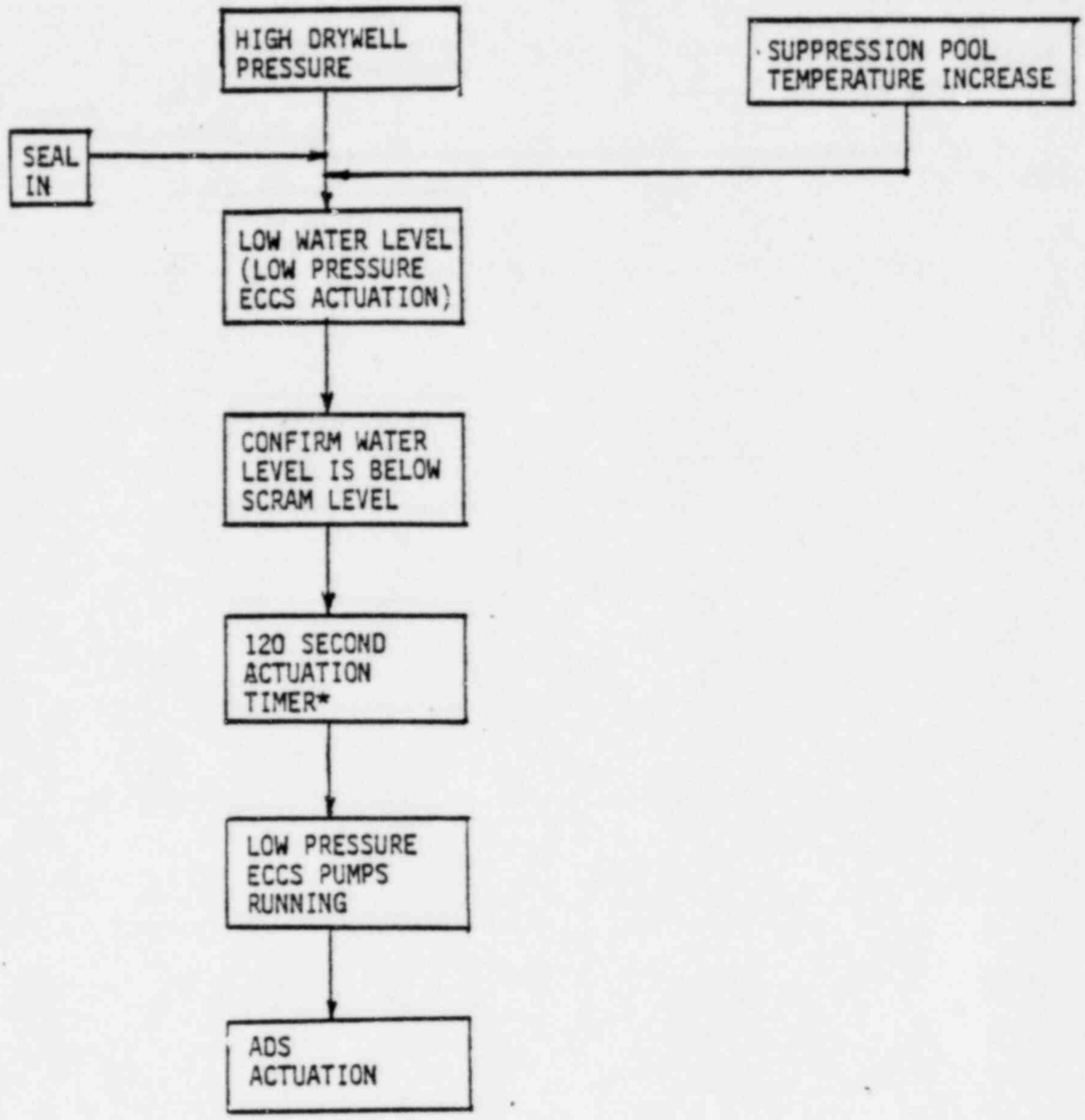


*120 SECOND ACTUATION TIMER WILL RESET IF LOW WATER LEVEL TRIP RECOVERS BEFORE IT TIMES OUT. THE TIMER WILL RESTART IF THE LOW LEVEL SIGNAL OCCURS AGAIN.

** SEAL IN FOR BWR/4-6 ONLY

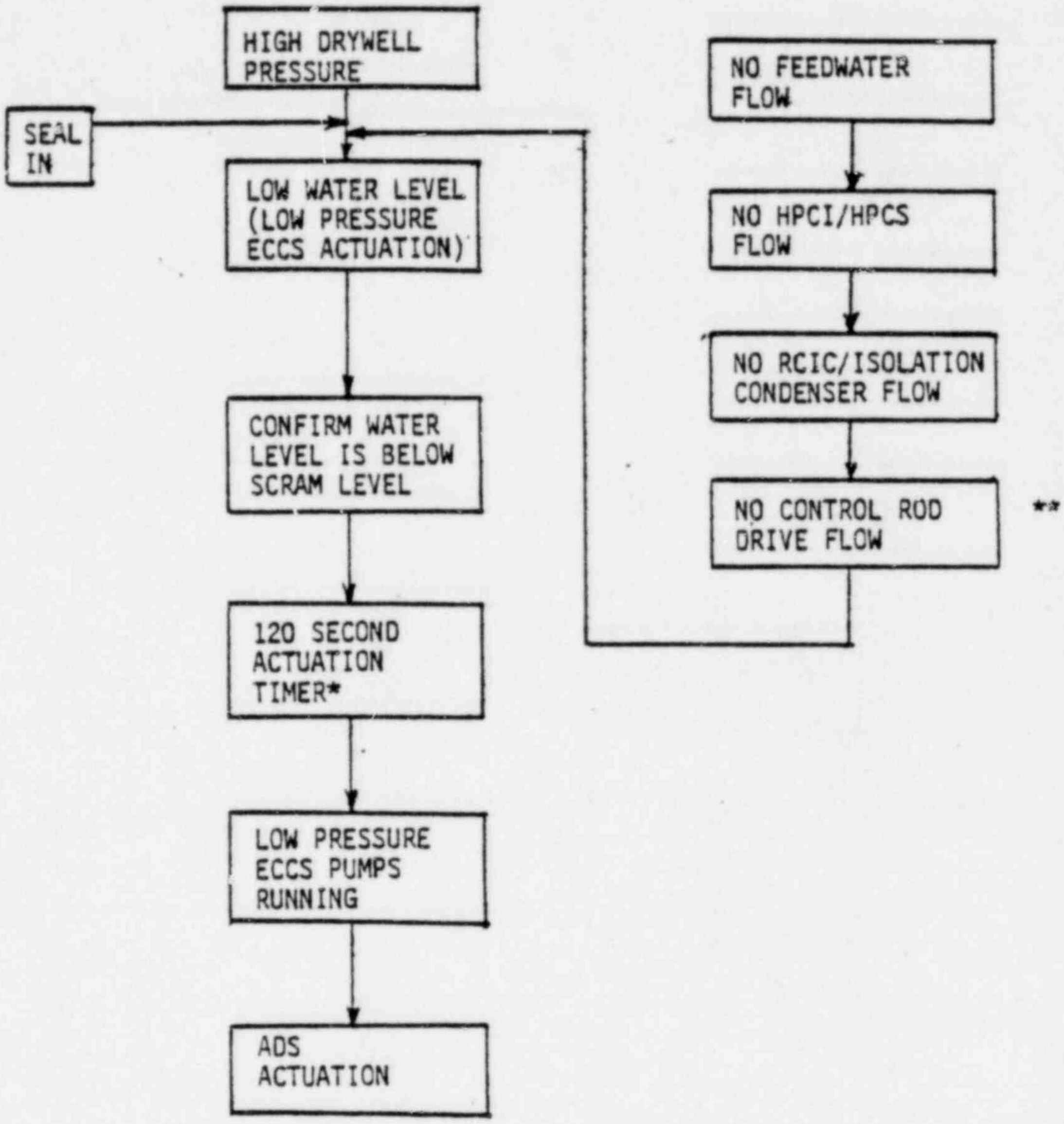
*** RESET SAME AS 120 SECOND ACTUATION TIMER FOR BWR/1-3 ONLY

FIGURE 2-3B: ADD DELAYED BYPASS OF HIGH DRYWELL PRESSURE TRIP WITH BYPASS TIMER STARTED AT LOW WATER LEVEL



*120 SECOND ACTUATION TIMER WILL RESET IF LOW WATER LEVEL TRIP RECOVERS BEFORE IT TIMES OUT. THE TIMER WILL RESTART IF THE LOW LEVEL SIGNAL OCCURS AGAIN.

FIGURE 2-4: ADD SUPPRESSION POOL TEMPERATURE INCREASE TRIP IN PARALLEL WITH HIGH DRYWELL PRESSURE TRIP



*120 SECOND ACTUATION TIMER WILL RESET IF LOW WATER LEVEL TRIP RECOVERS BEFORE IT TIMES OUT. THE TIMER WILL RESTART IF THE LOW LEVEL SIGNAL OCCURS AGAIN.

**MAY BE DESIRABLE TO ADD FOR BWR/1-3

FIGURE 2-5: BYPASS HIGH DRYWELL PRESSURE TRIP IF NO HIGH PRESSURE MAKEUP SYSTEM FLOW IS AVAILABLE

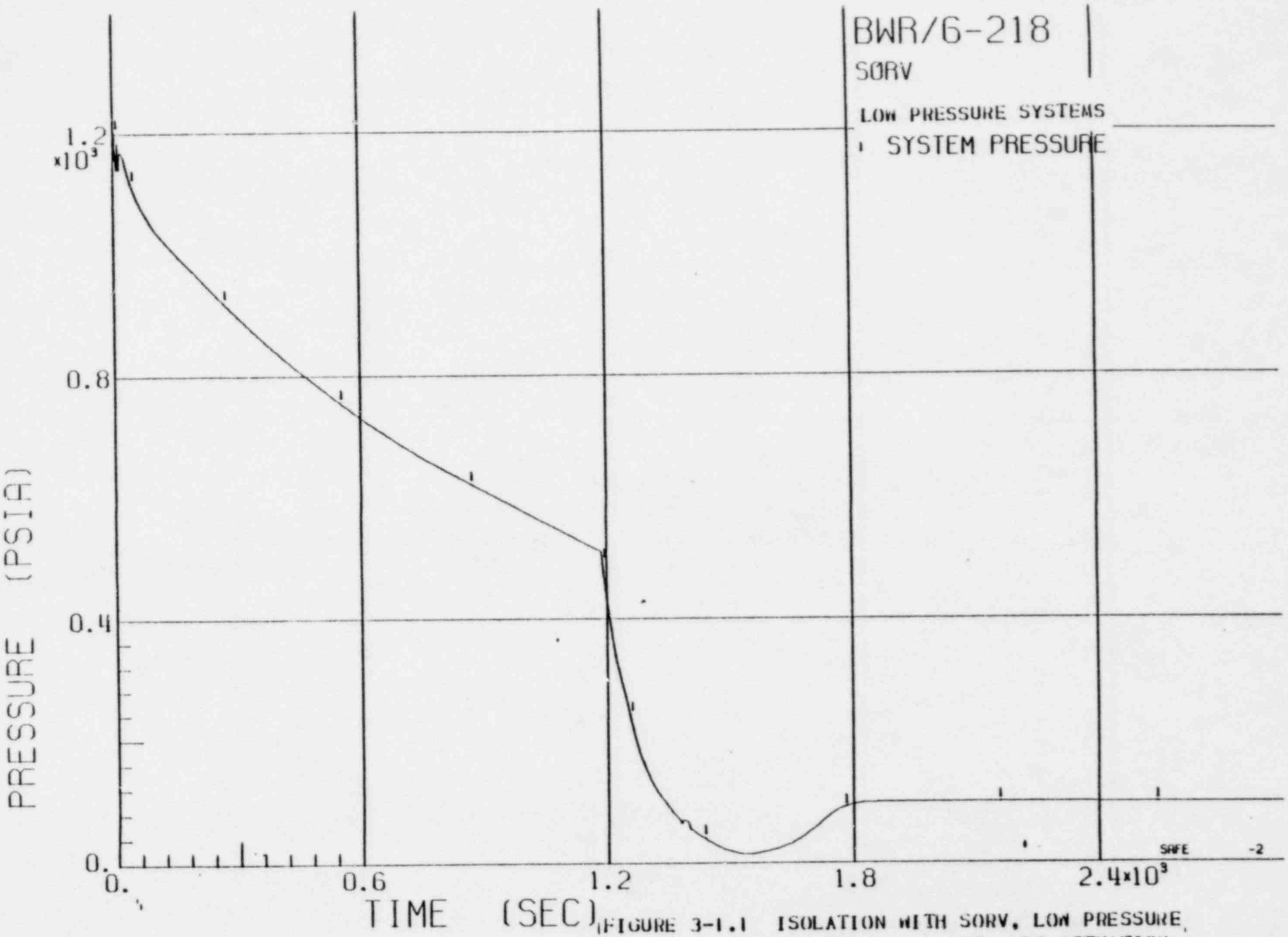


FIGURE 3-1.1 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS ACTUATION 10 MINUTES AFTER LEVEL ONE

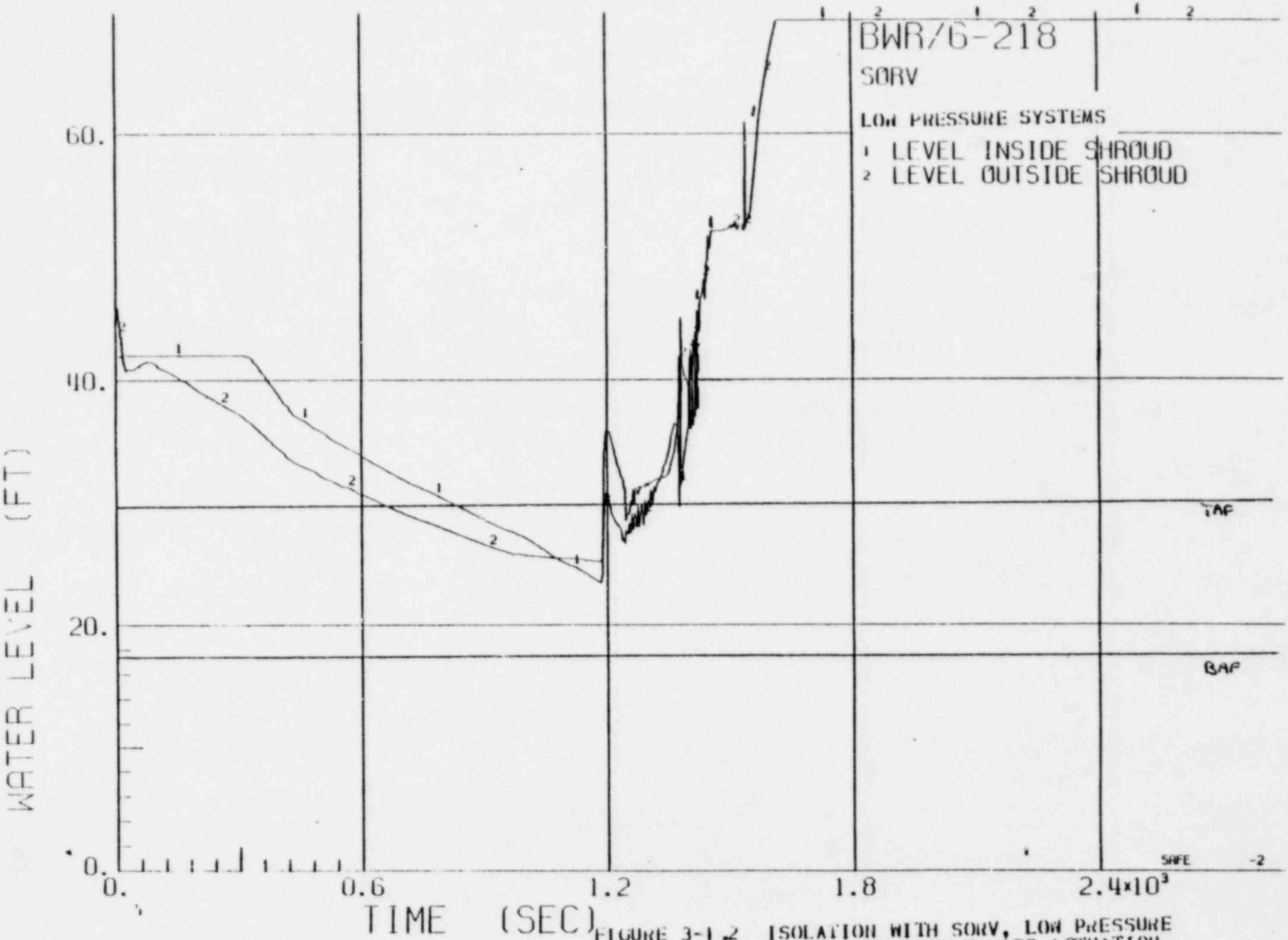


FIGURE 3-1.2 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS ACTUATION 10 MINUTES AFTER LEVEL ONE

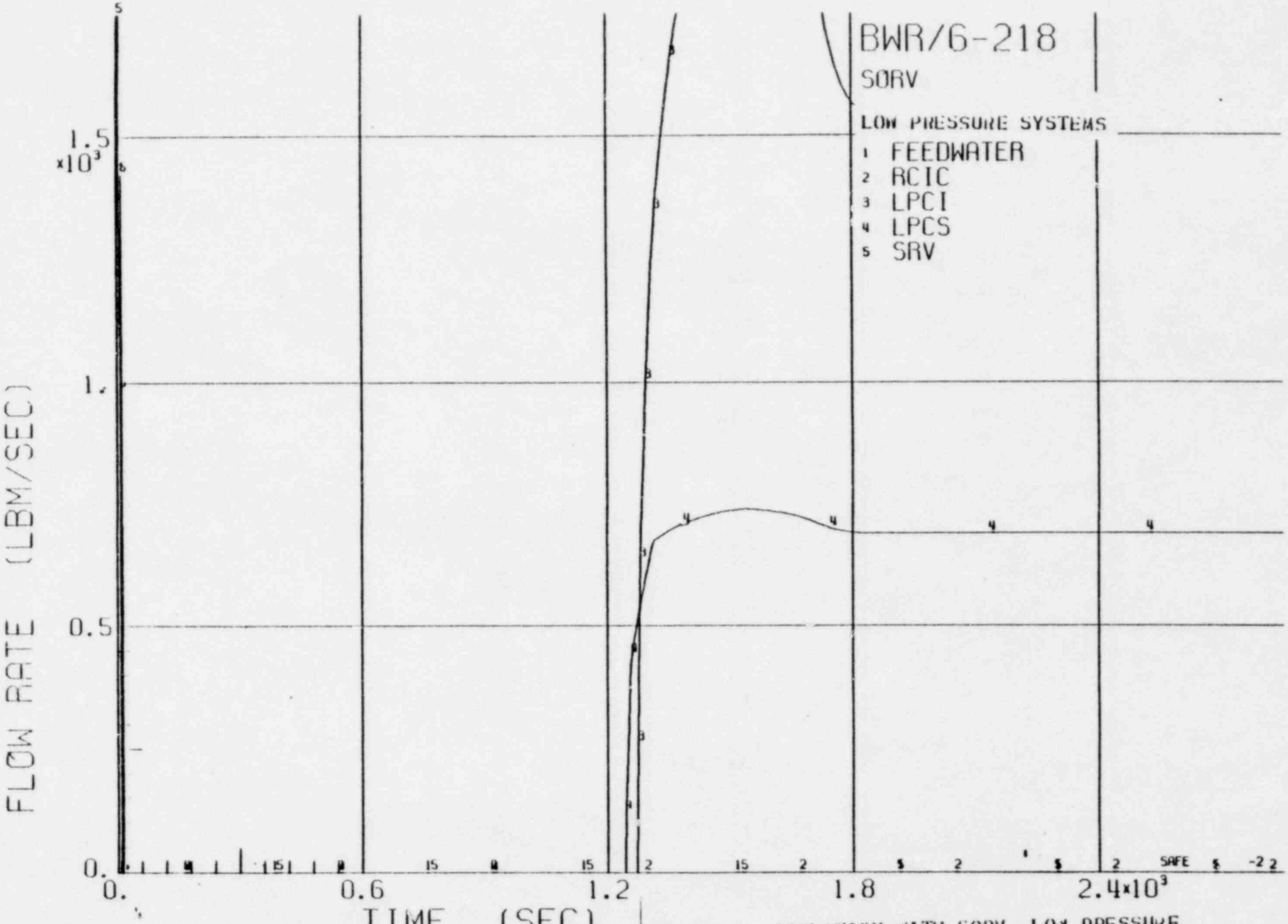


FIGURE 3-1.3 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS ACTUATION 10 MINUTES AFTER LEVEL ONE

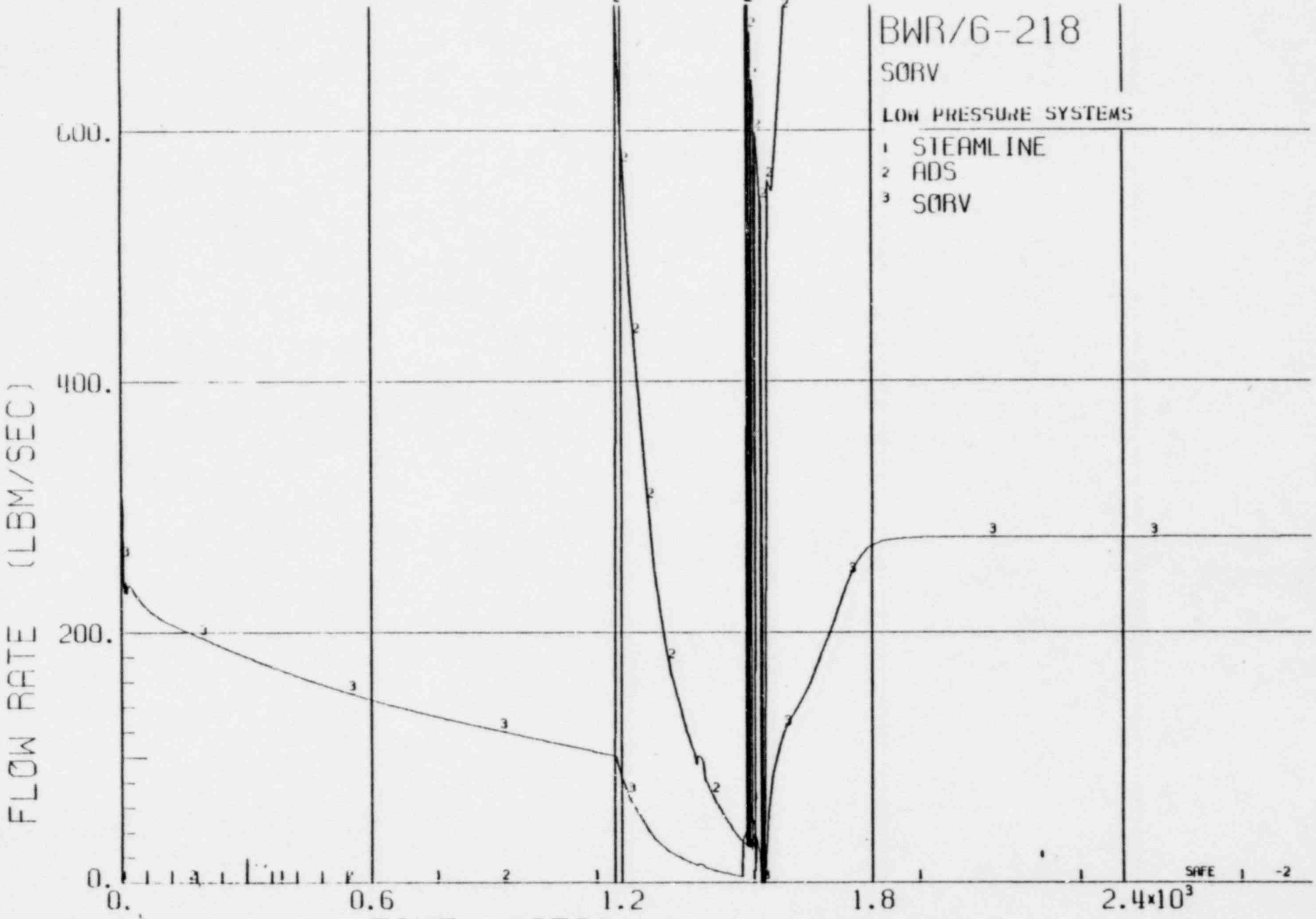


FIGURE 3-1.4 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS ACTUATION 10 MINUTES AFTER LEVEL ONE

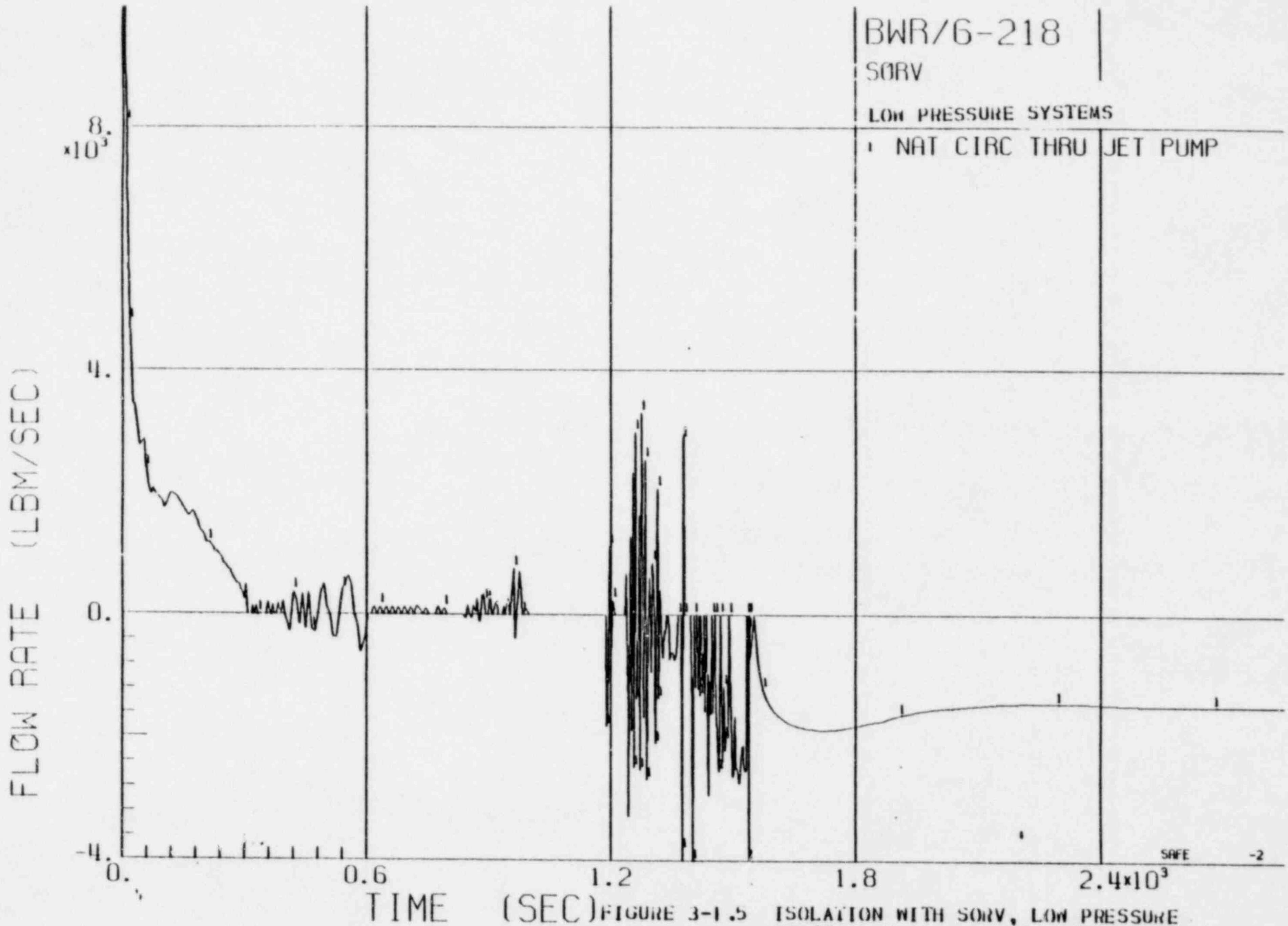


FIGURE 3-1.5 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS ACTUATION 10 MINUTES AFTER LEVEL ONE

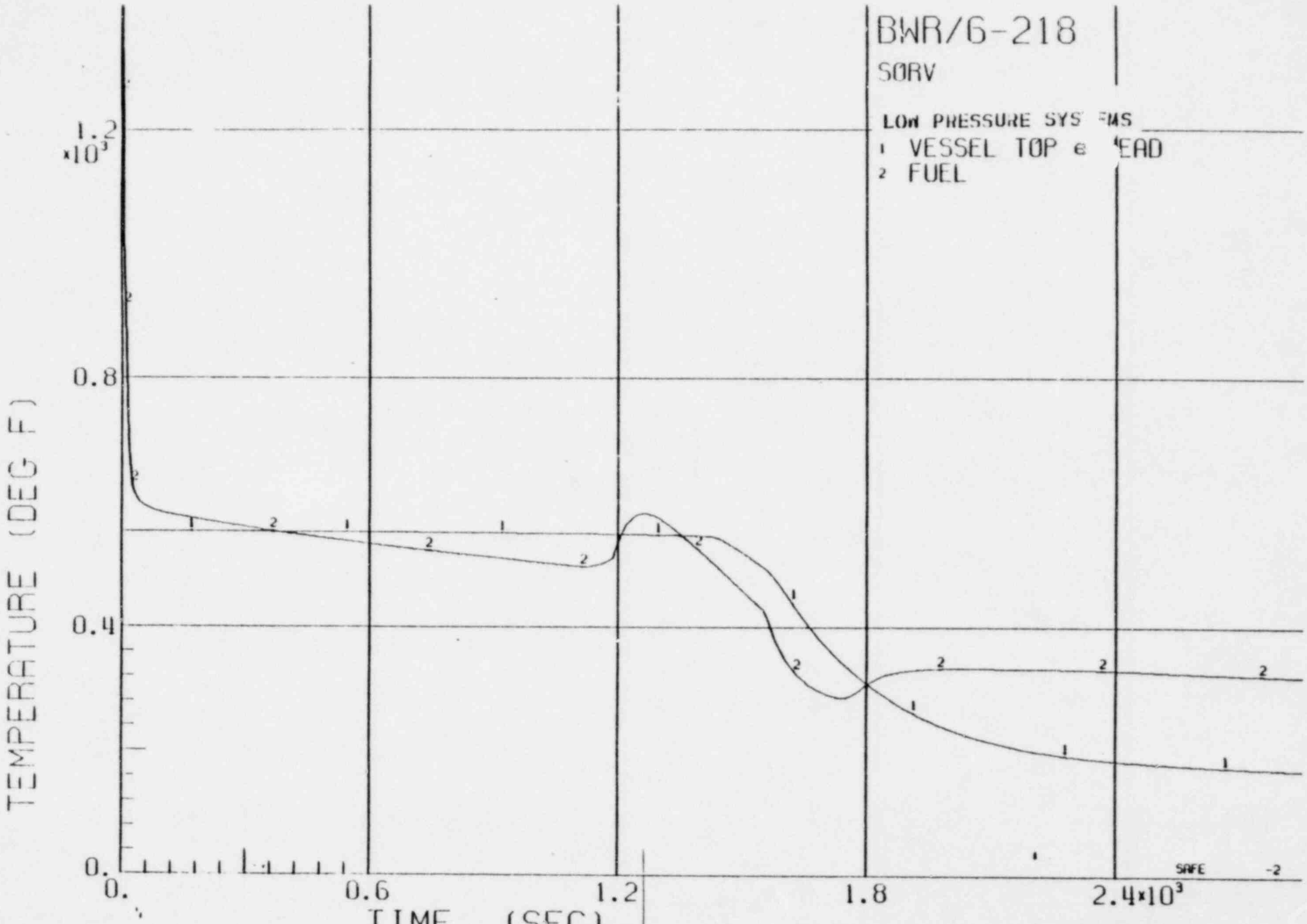


FIGURE 3-1.6 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS ACTUATION 10 MINUTES AFTER LEVEL ONE

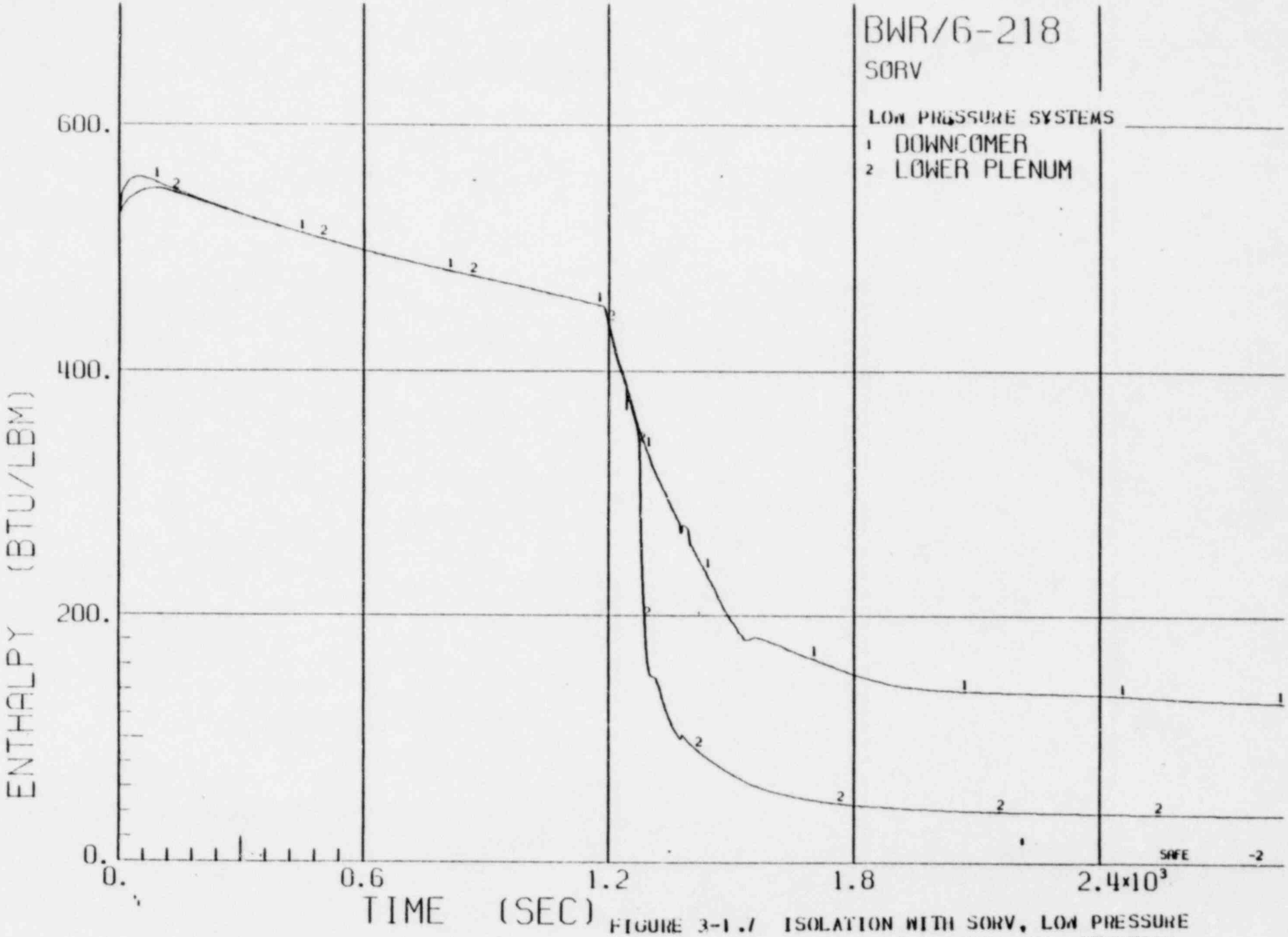


FIGURE 3-1.7 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS ACTUATION 10 MINUTES AFTER LEVEL ONE

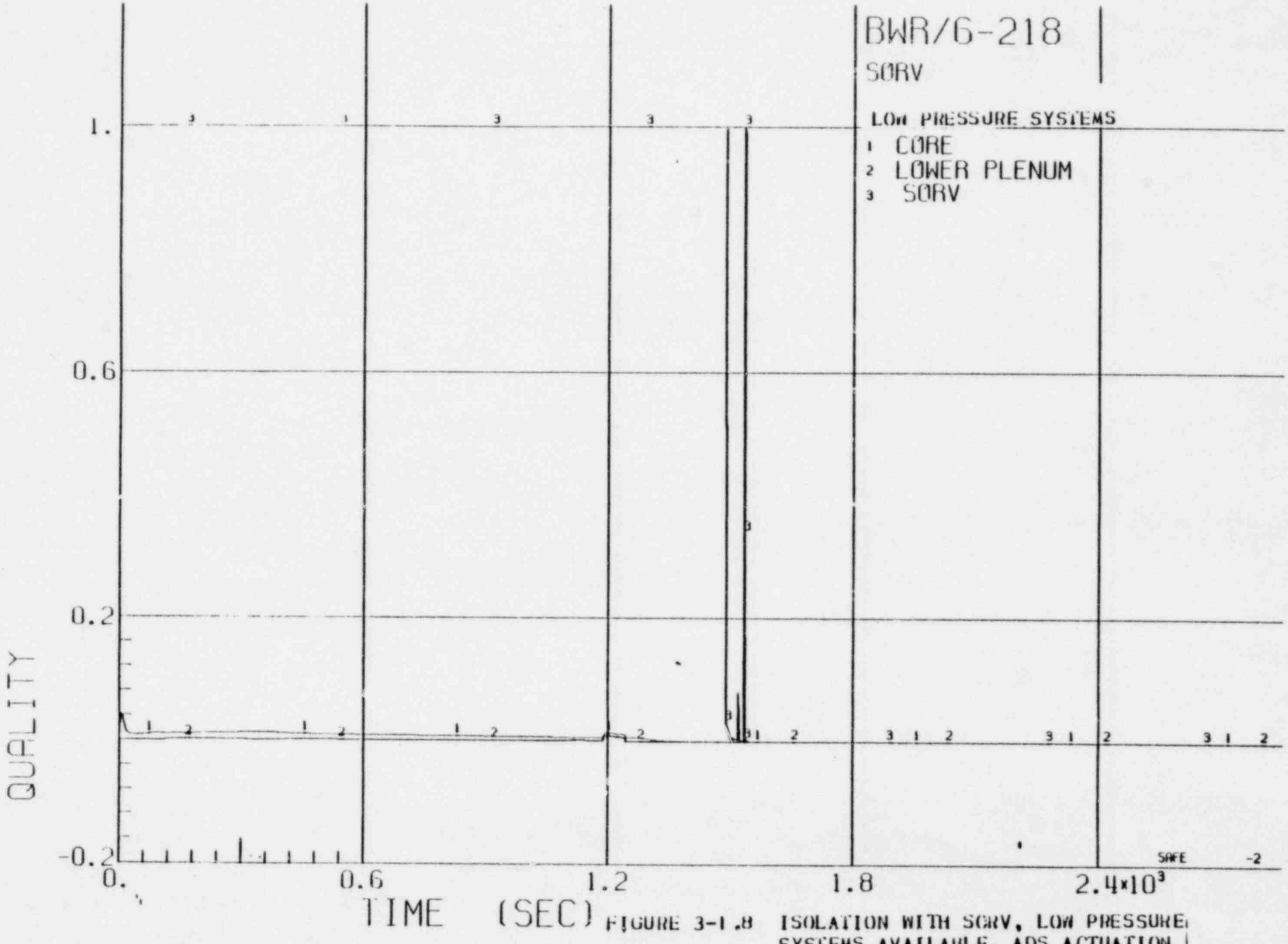


FIGURE 3-1.8 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS ACTUATION 10 MINUTES AFTER LEVEL ONE

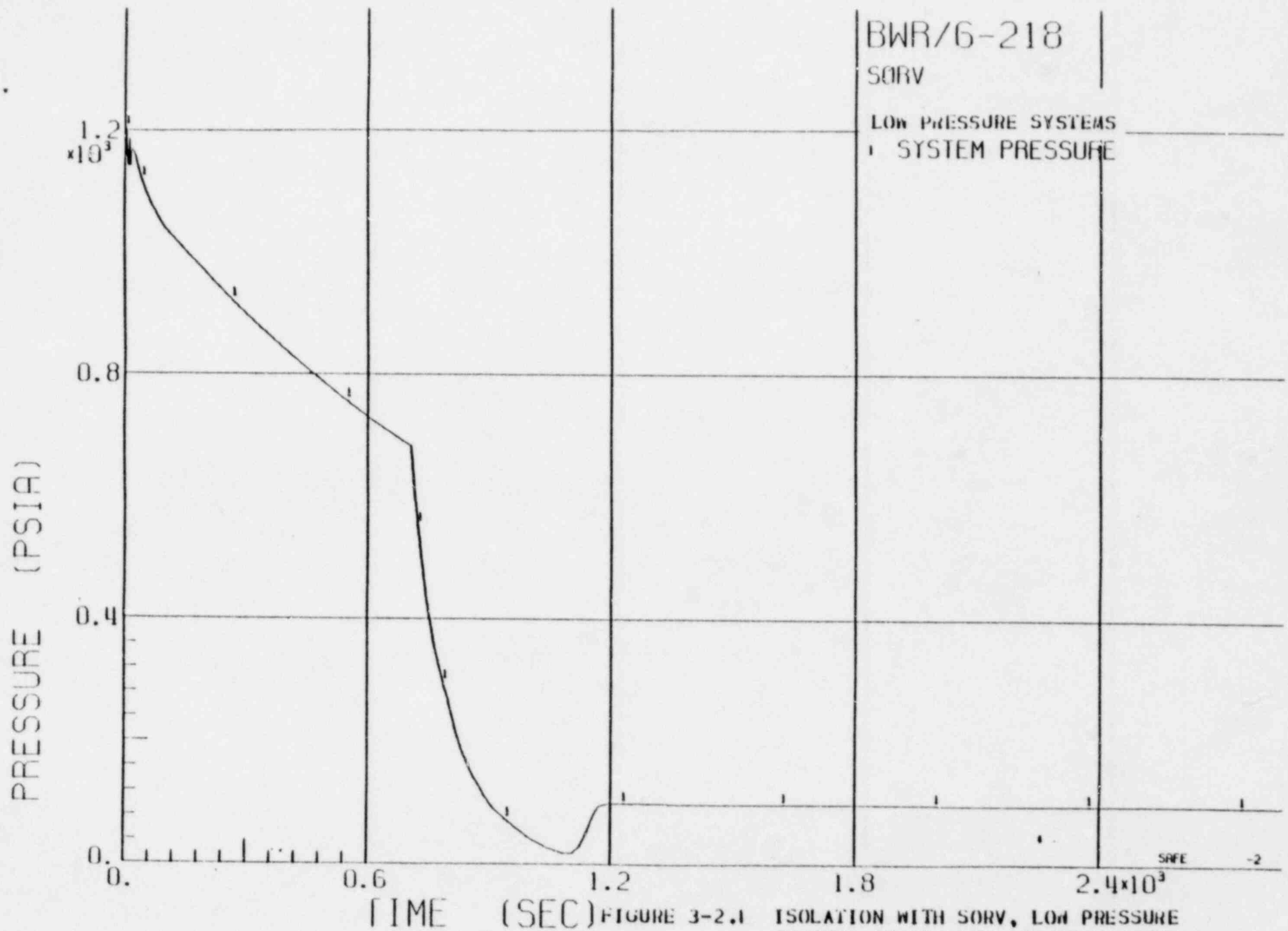


FIGURE 3-2.1 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS PERMISSIVE 8 MINUTES AFTER SCRAM LEVEL

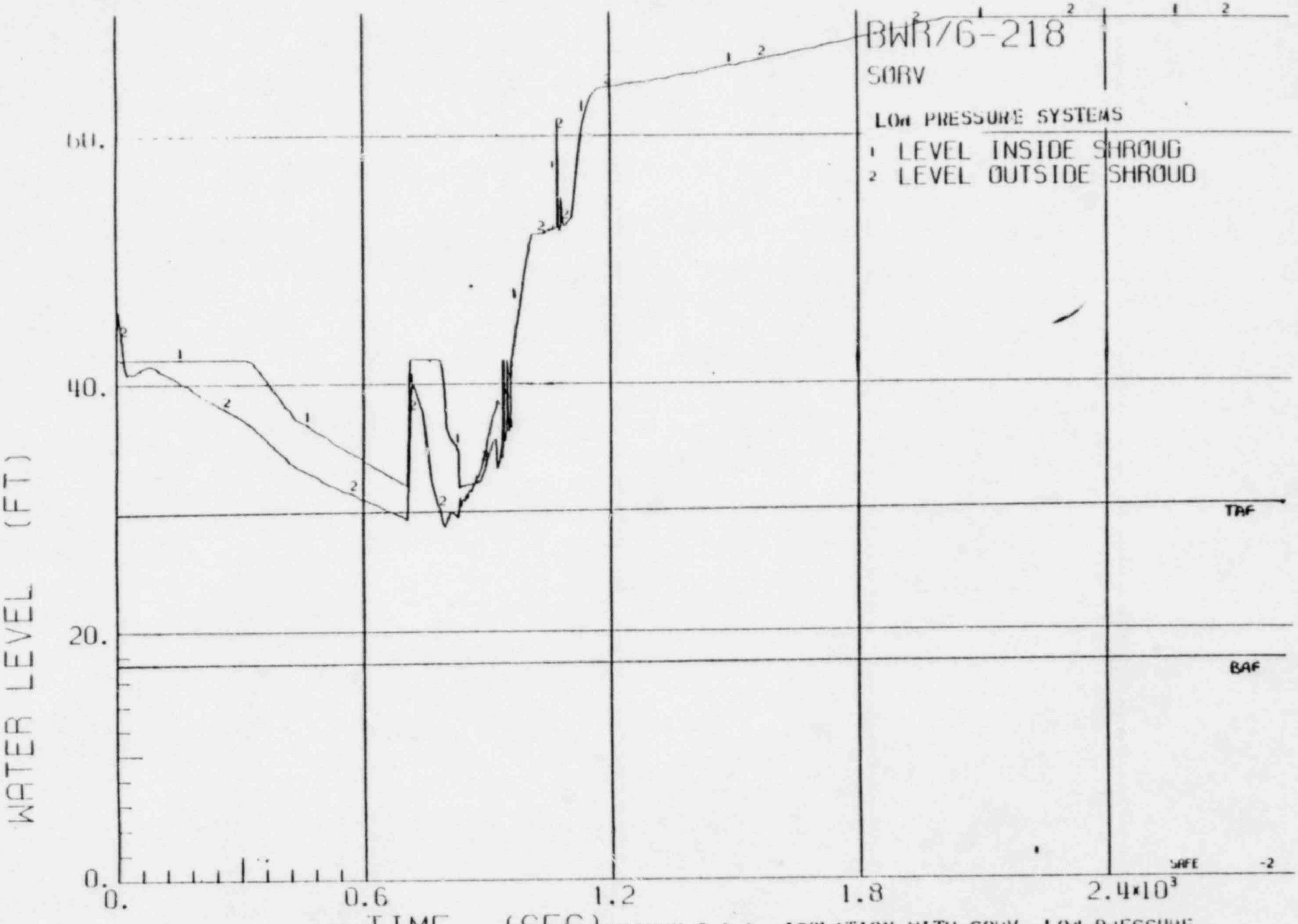


FIGURE 3-2.2 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS PERMISSIVE 8 MINUTES AFTER SCRAM LEVEL

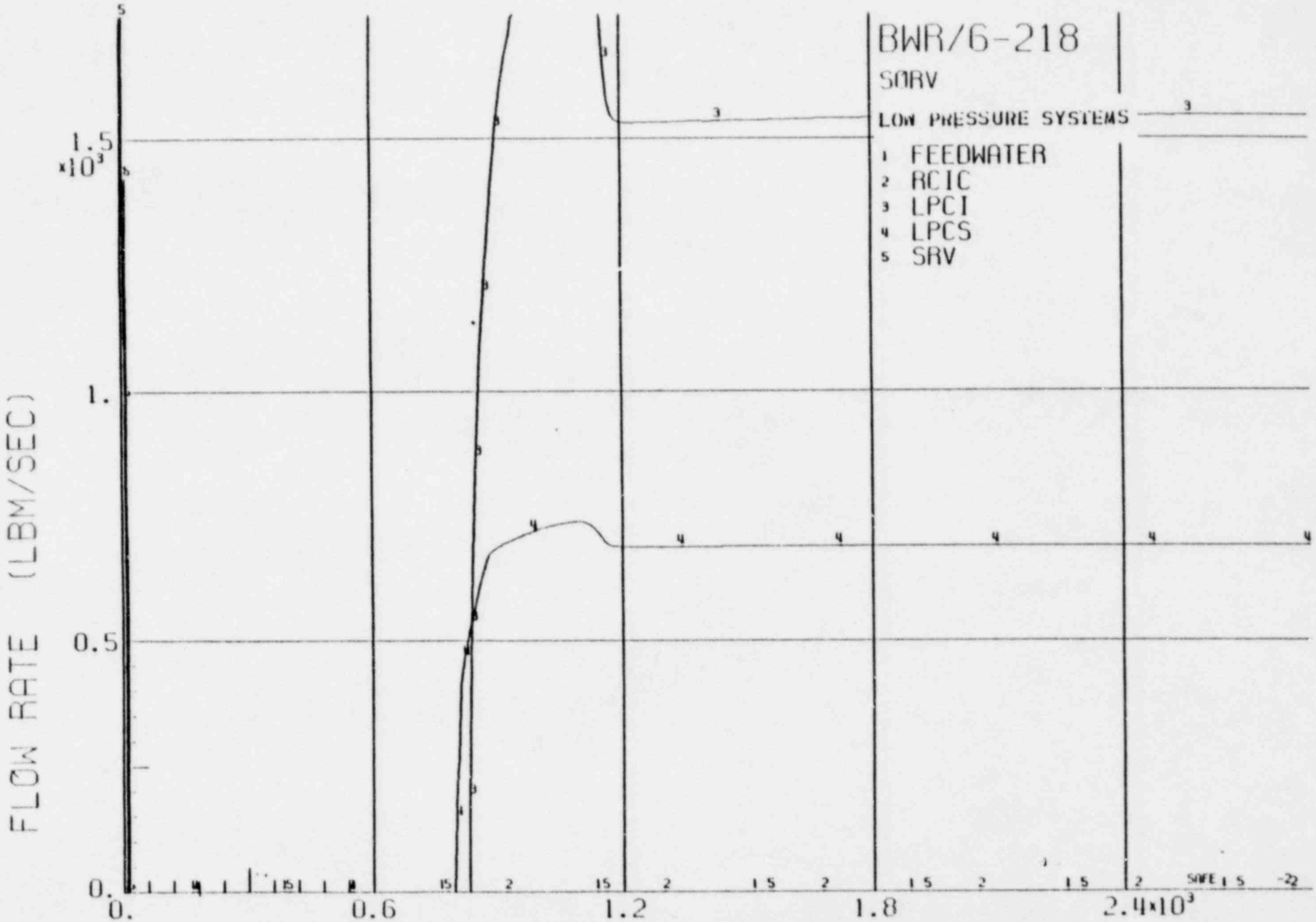


FIGURE 3-2.3 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS PERMISSIVE 8 MINUTES AFTER SCRAM LEVEL

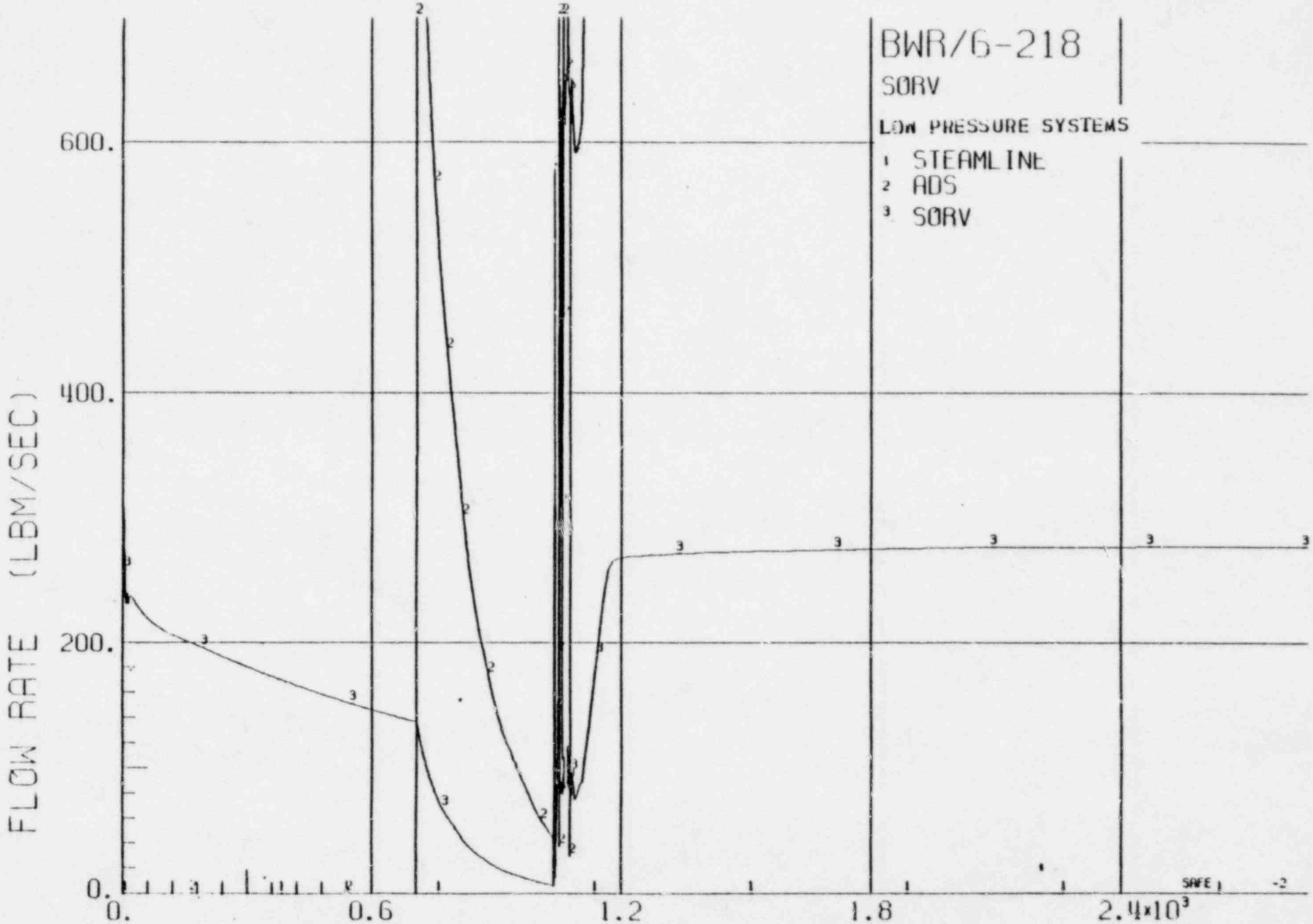


FIGURE 3-2.4 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS PERMISSIVE 8 MINUTES AFTER SCRAM LEVEL

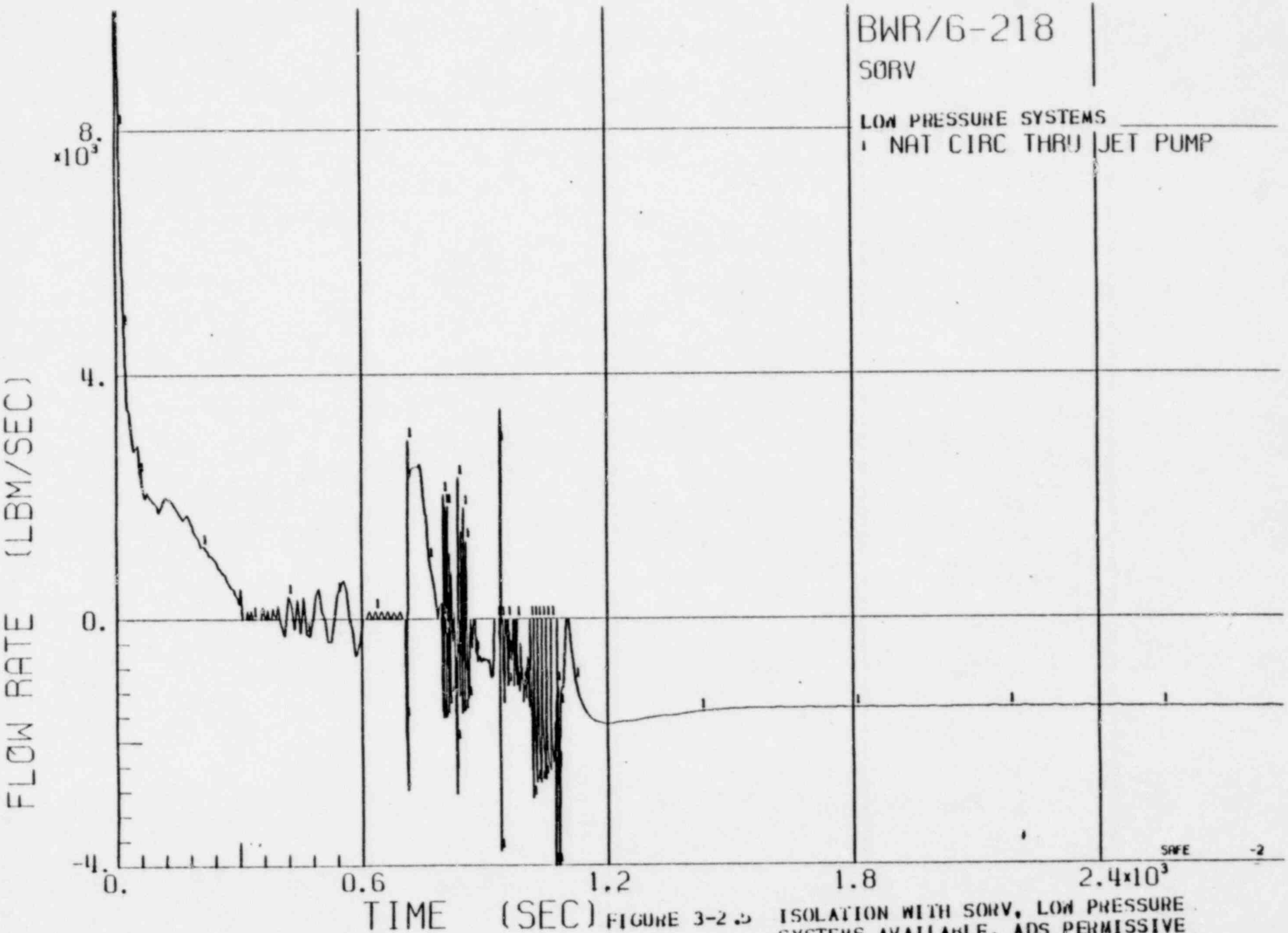


FIGURE 3-2.5 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS PERMISSIVE 8 MINUTES AFTER SCRAM LEVEL

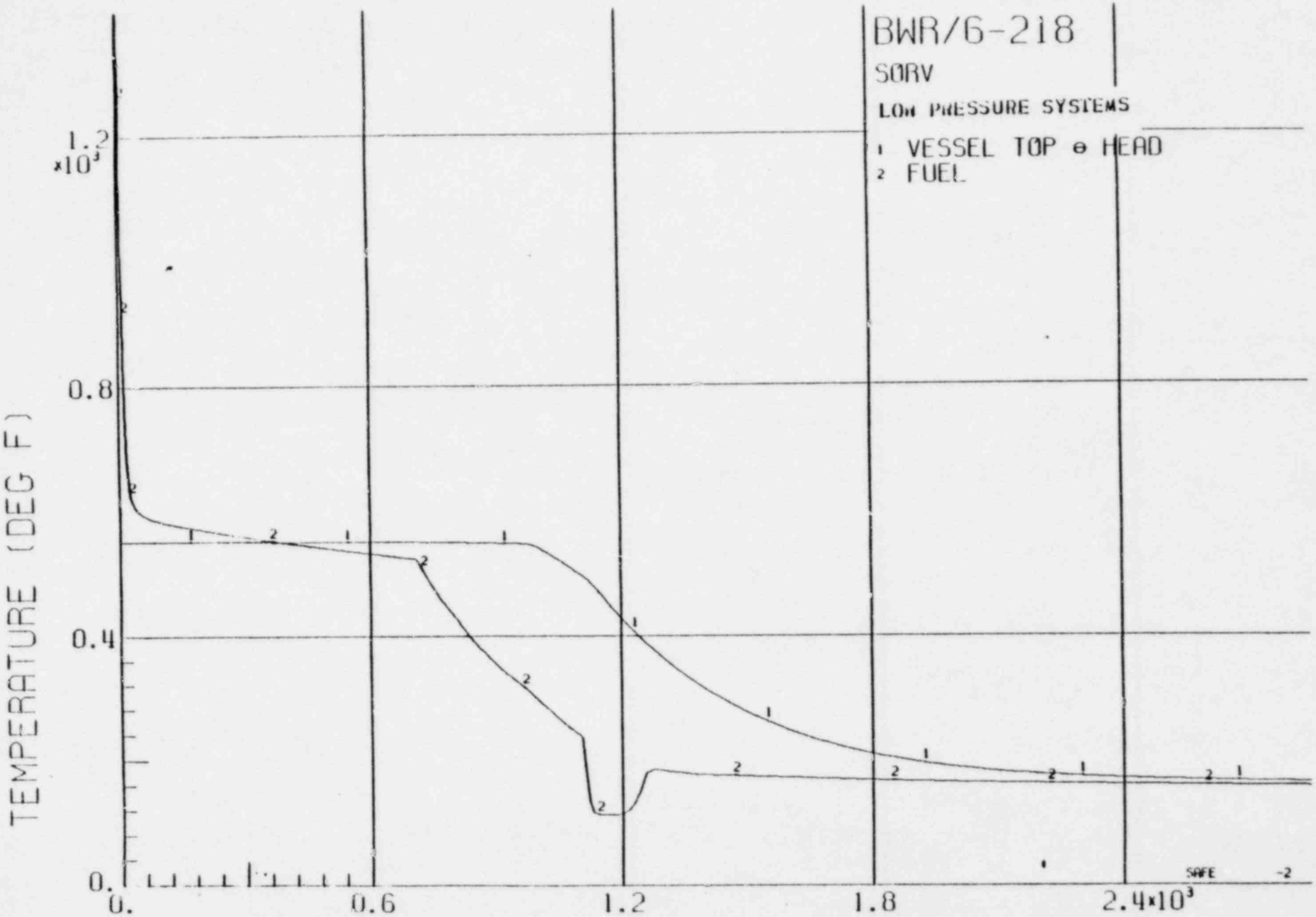


FIGURE 3-2.6 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS PERMISSIVE 8 MINUTES AFTER SCRAM LEVEL.

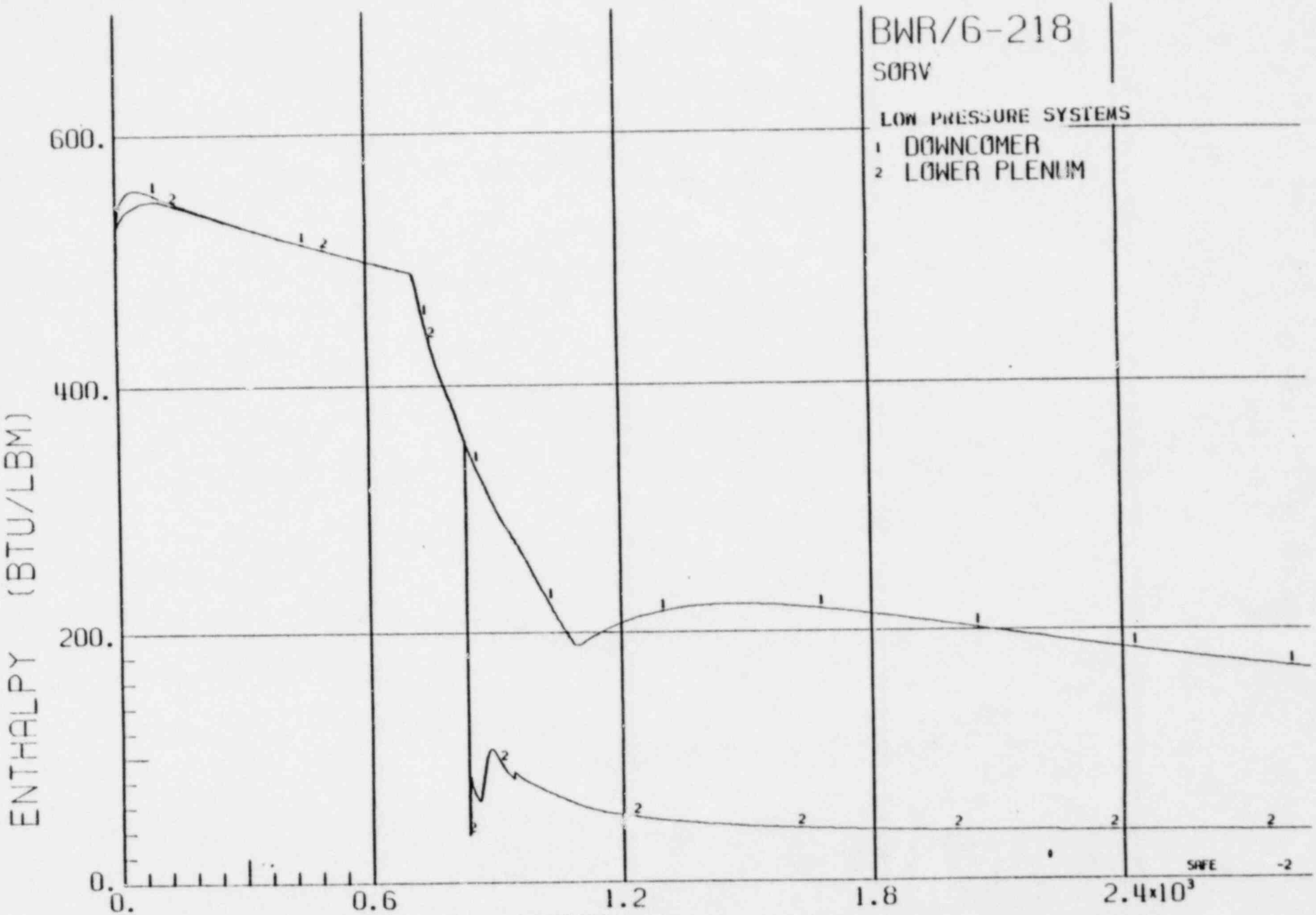


FIGURE 3-2.7 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS PERMISSIVE 8 MINUTES AFTER SCRAM LEVEL

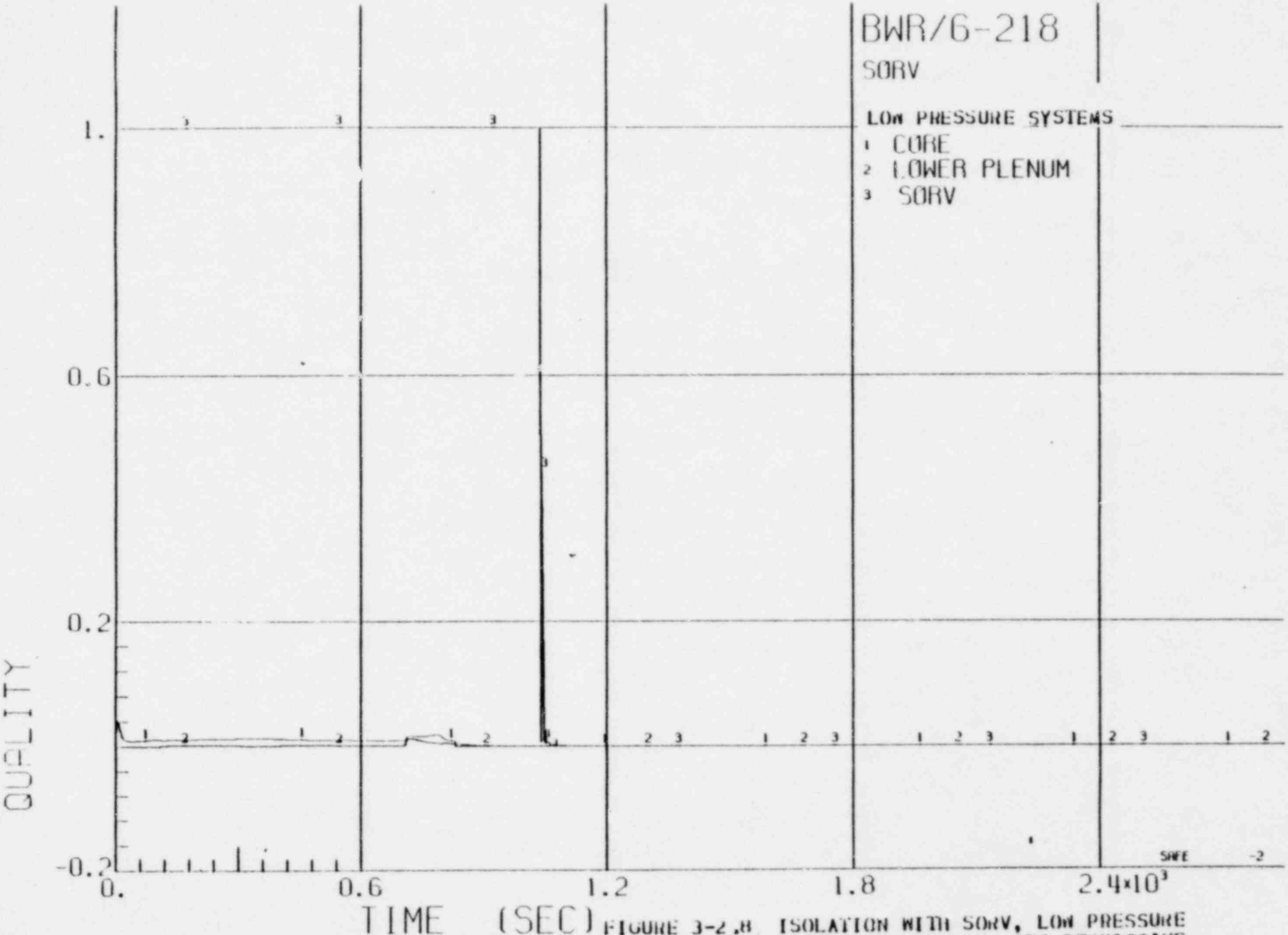


FIGURE 3-2.8 ISOLATION WITH SORV, LOW PRESSURE SYSTEMS AVAILABLE, ADS PERMISSIVE 8 MINUTES AFTER SCRAM LEVEL

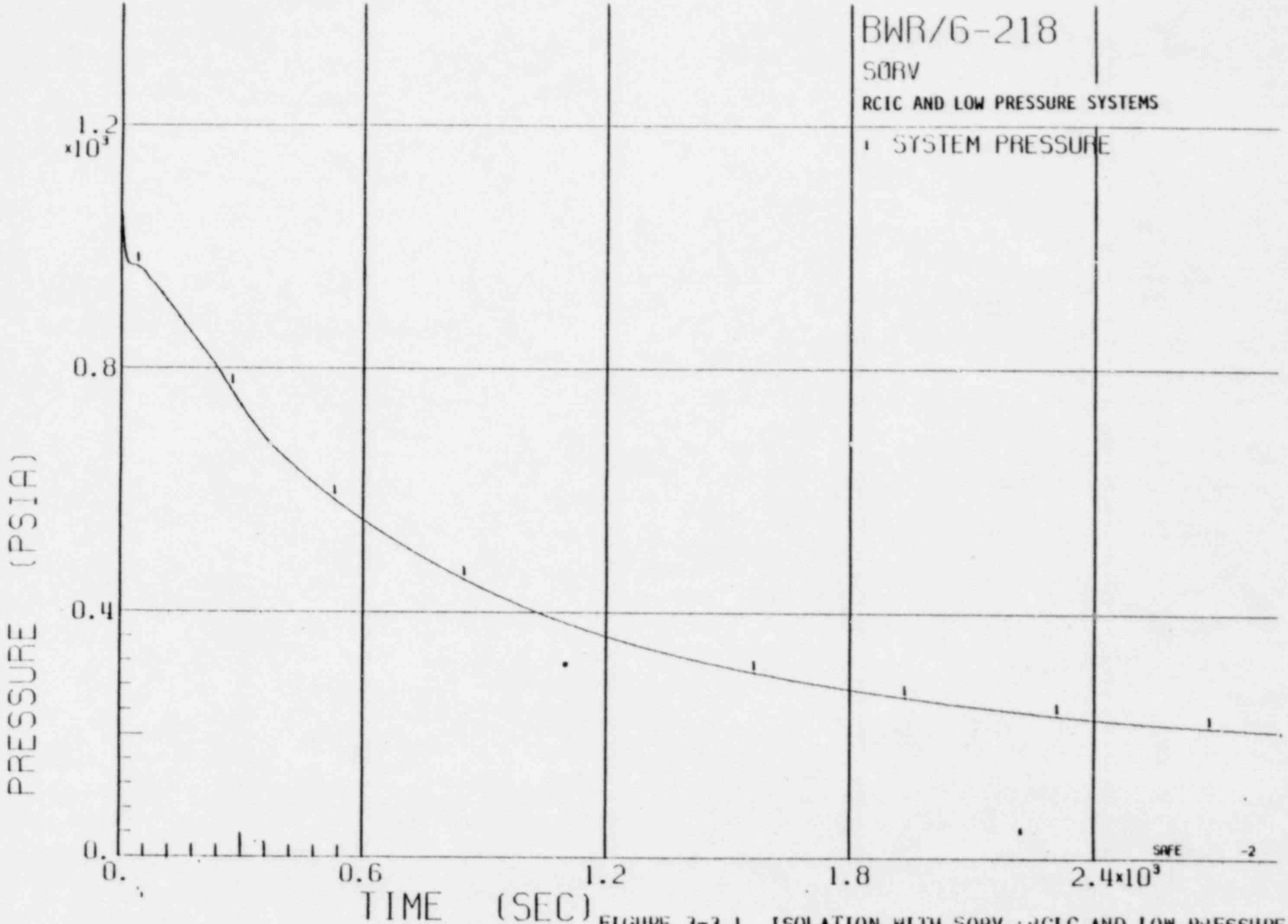


FIGURE 3-3.1 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

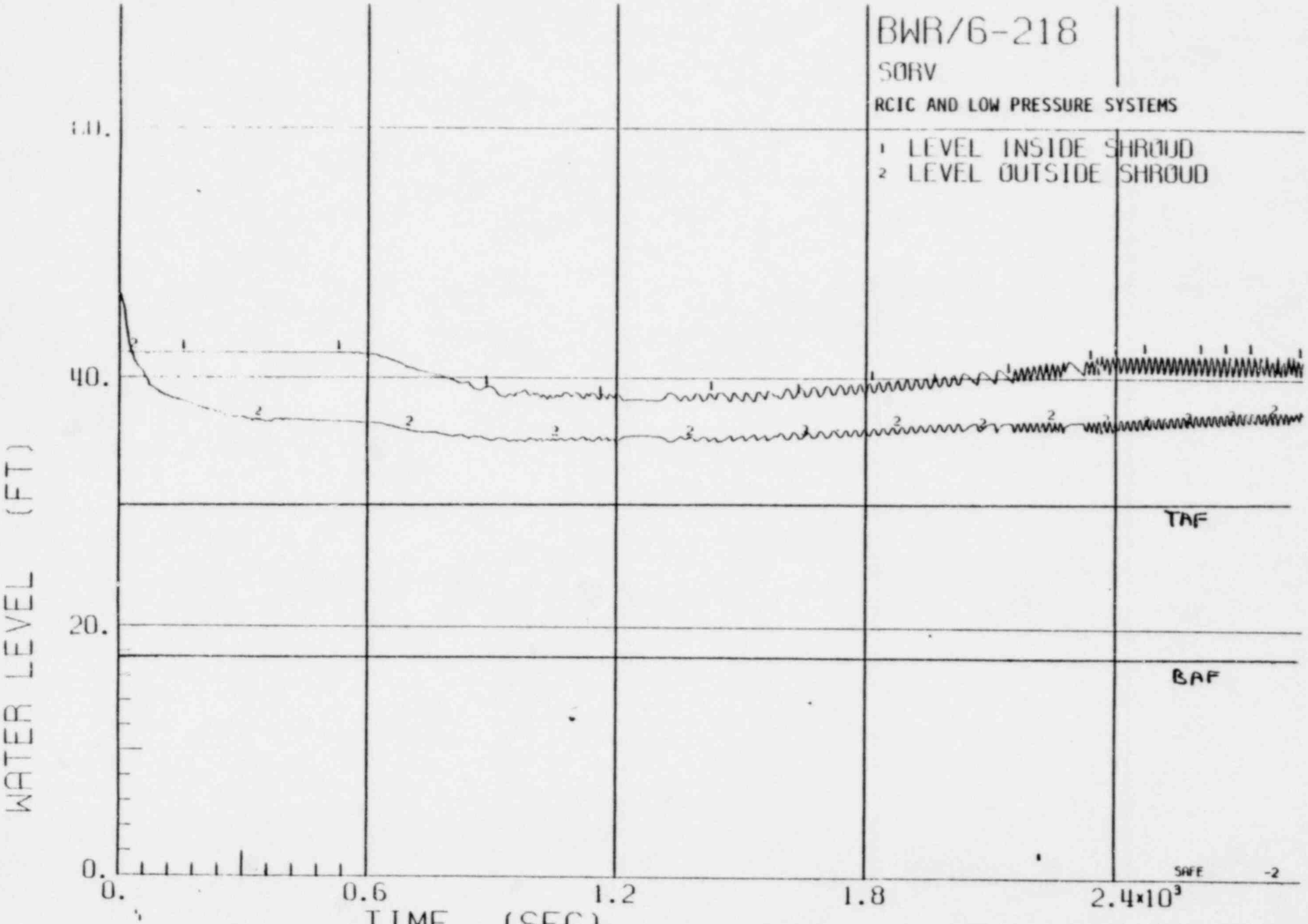


FIGURE 3-3.2 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

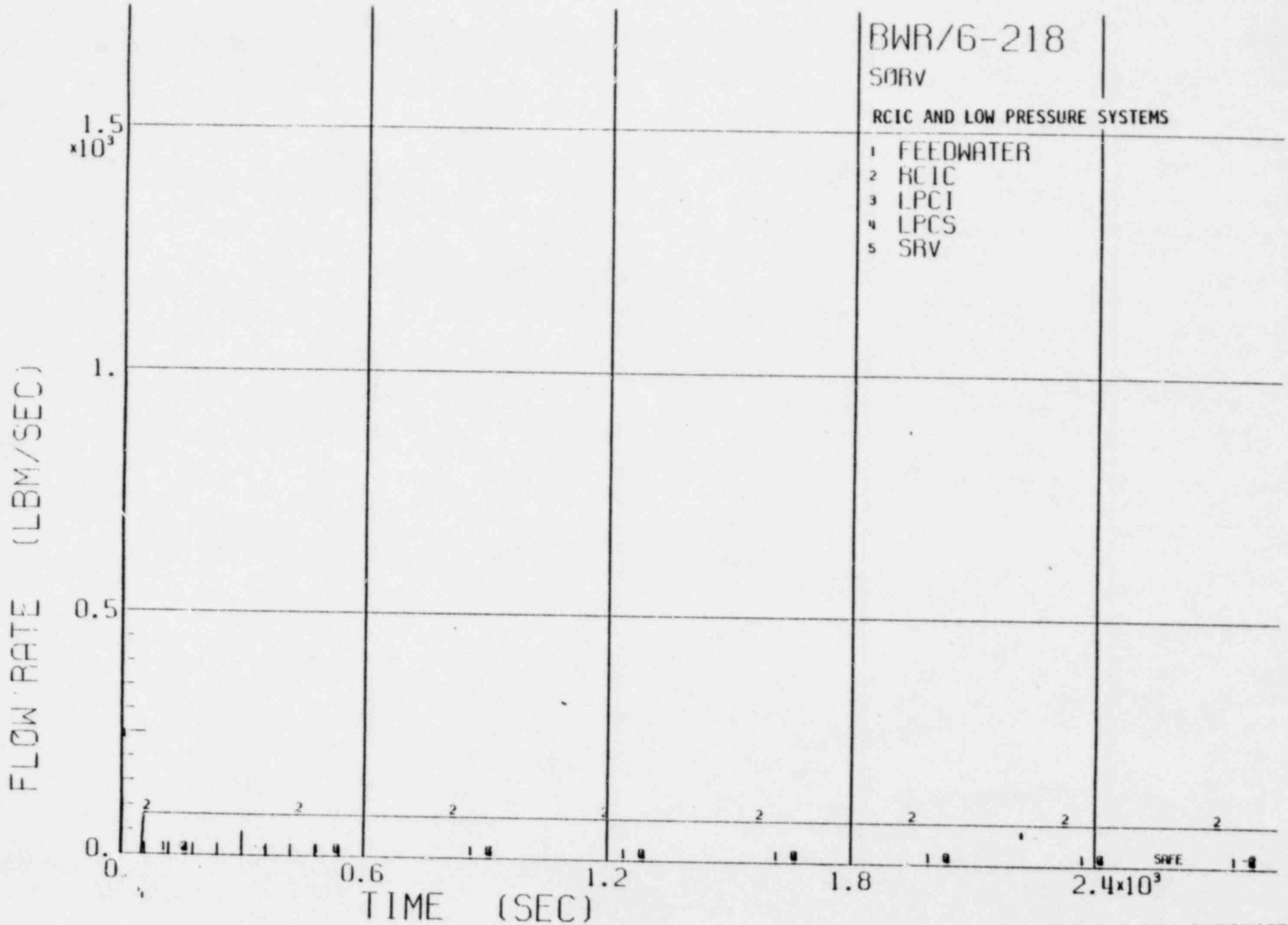


FIGURE 3-3.3 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

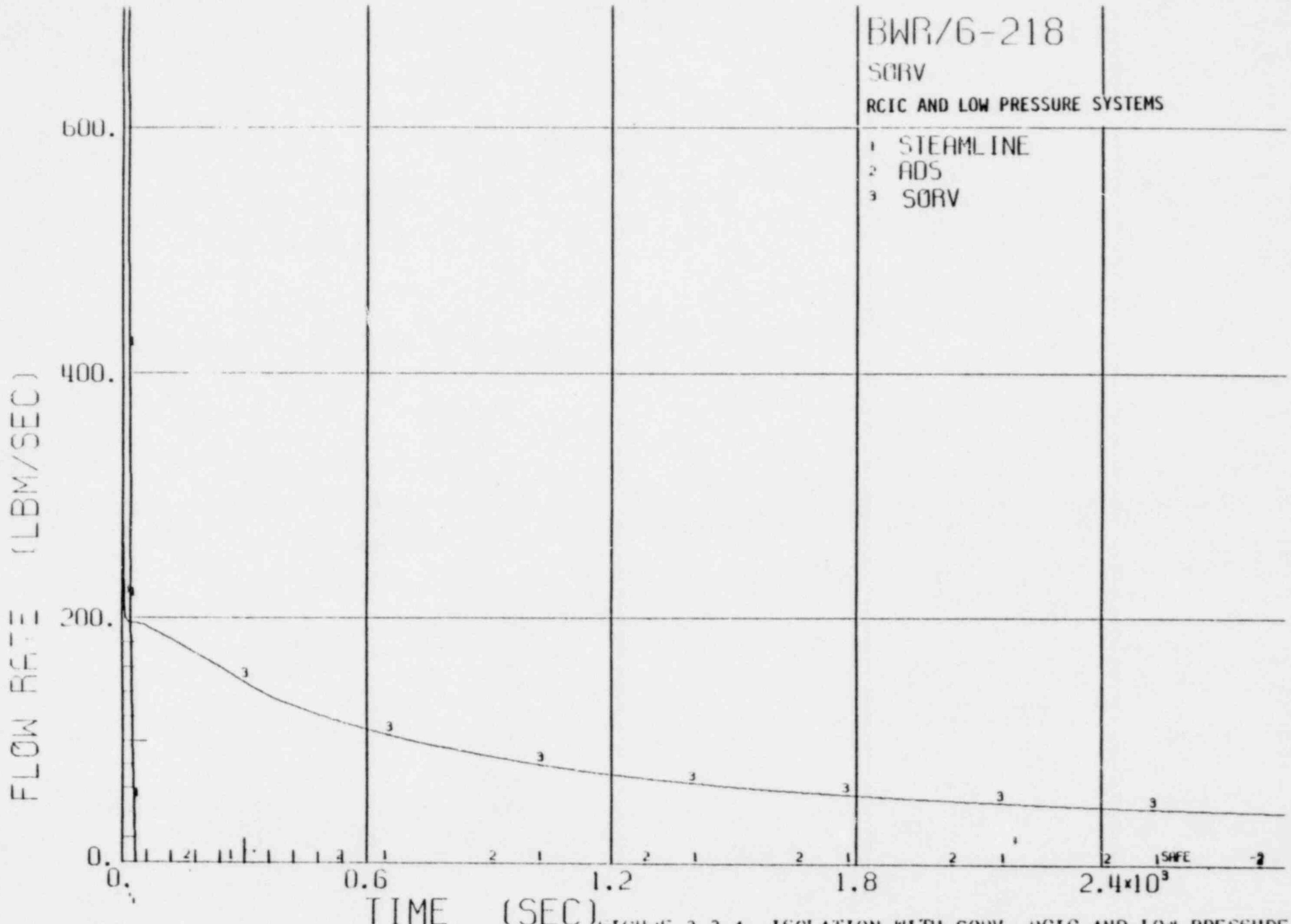


FIGURE 3-3.4 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

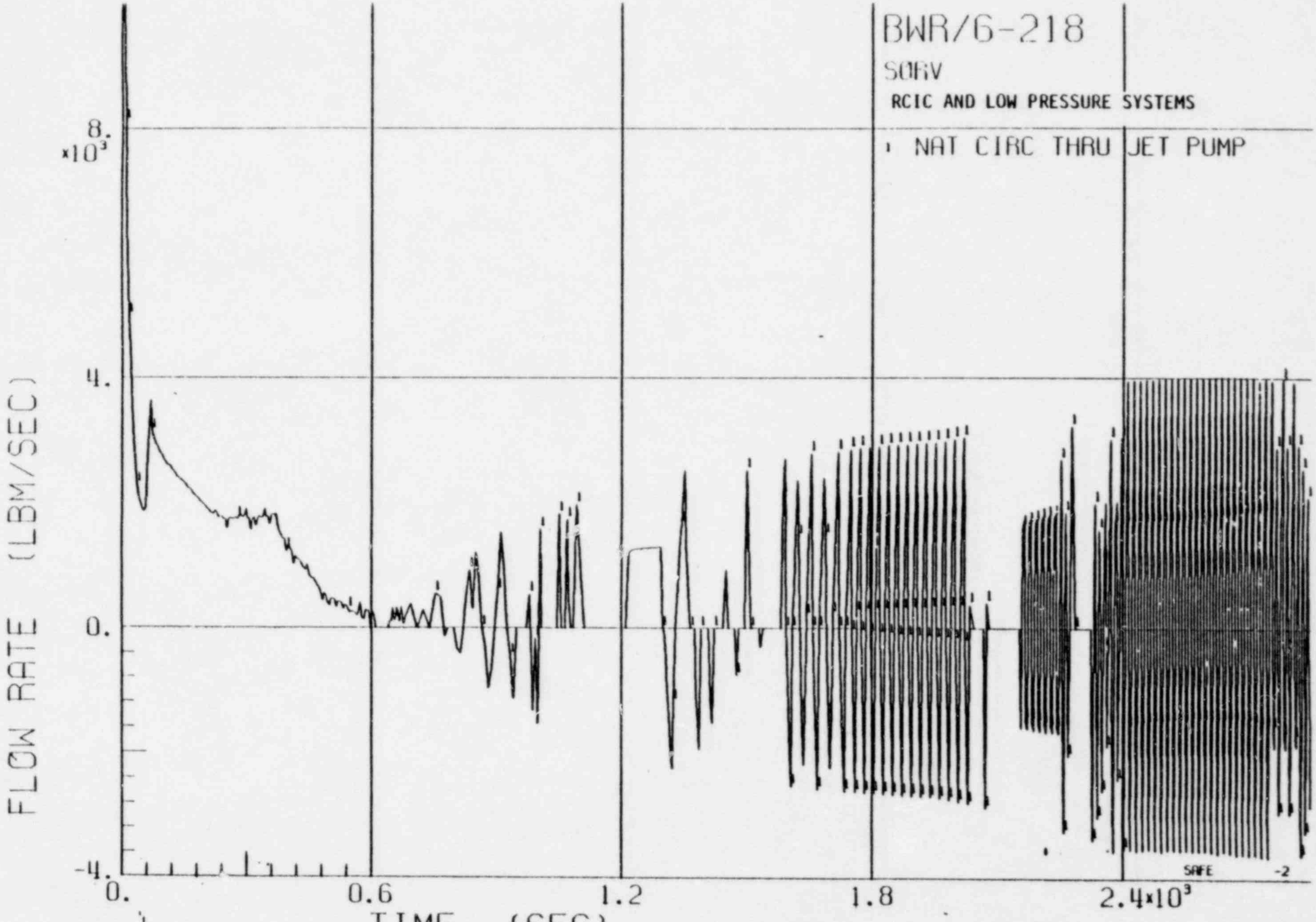


FIGURE 3-3.5 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

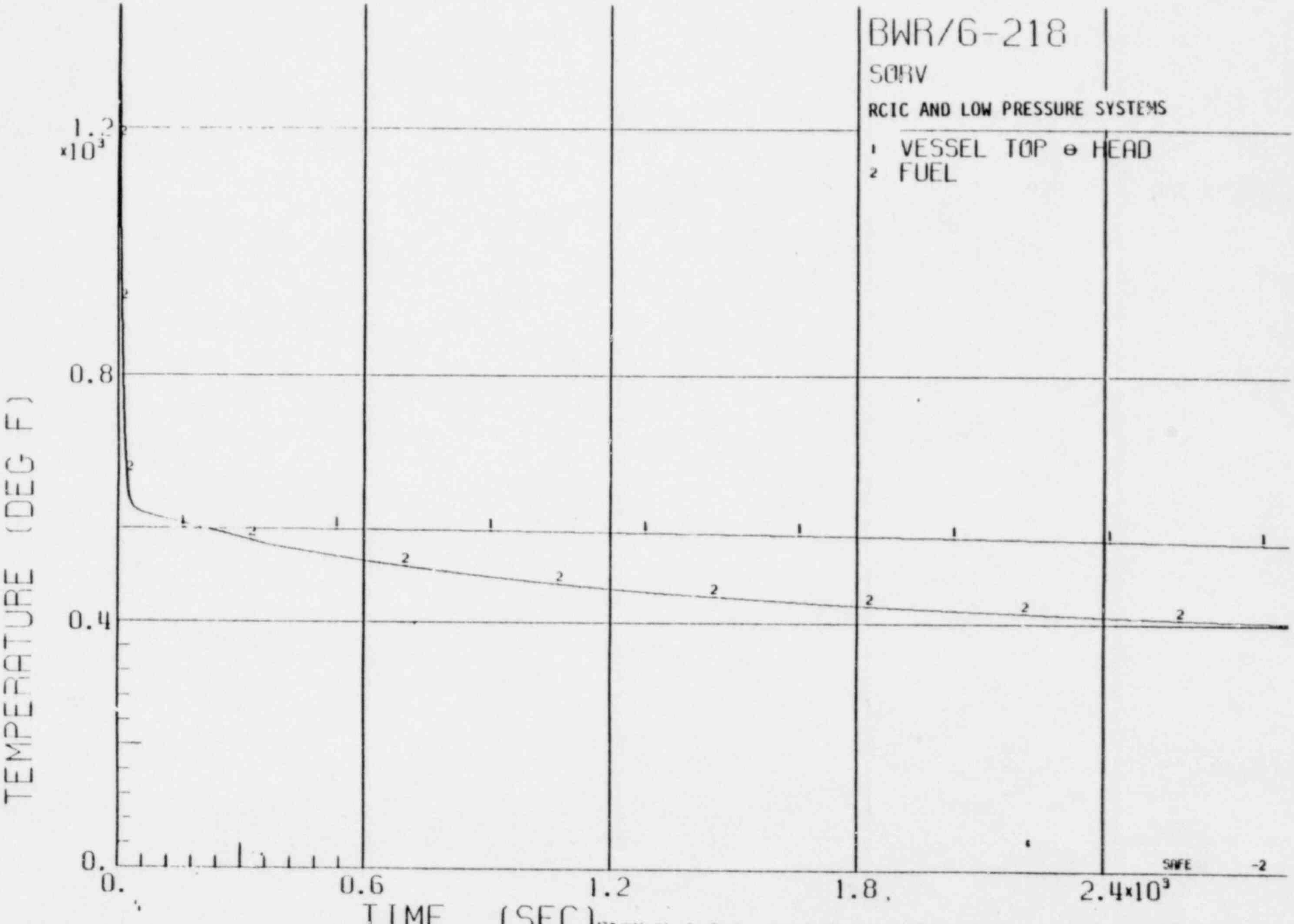


FIGURE 3-3.6 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

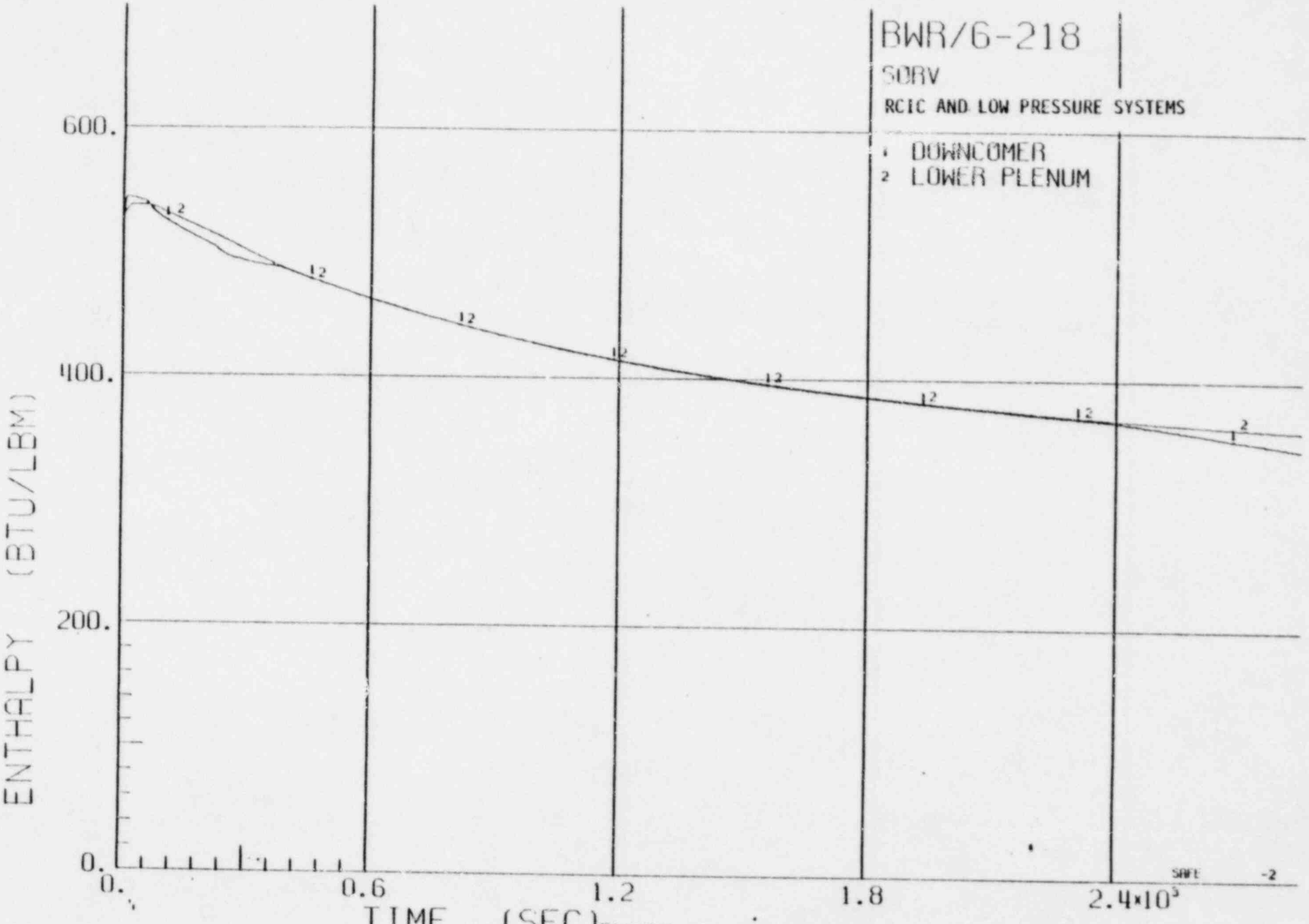


FIGURE 3-3.7 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

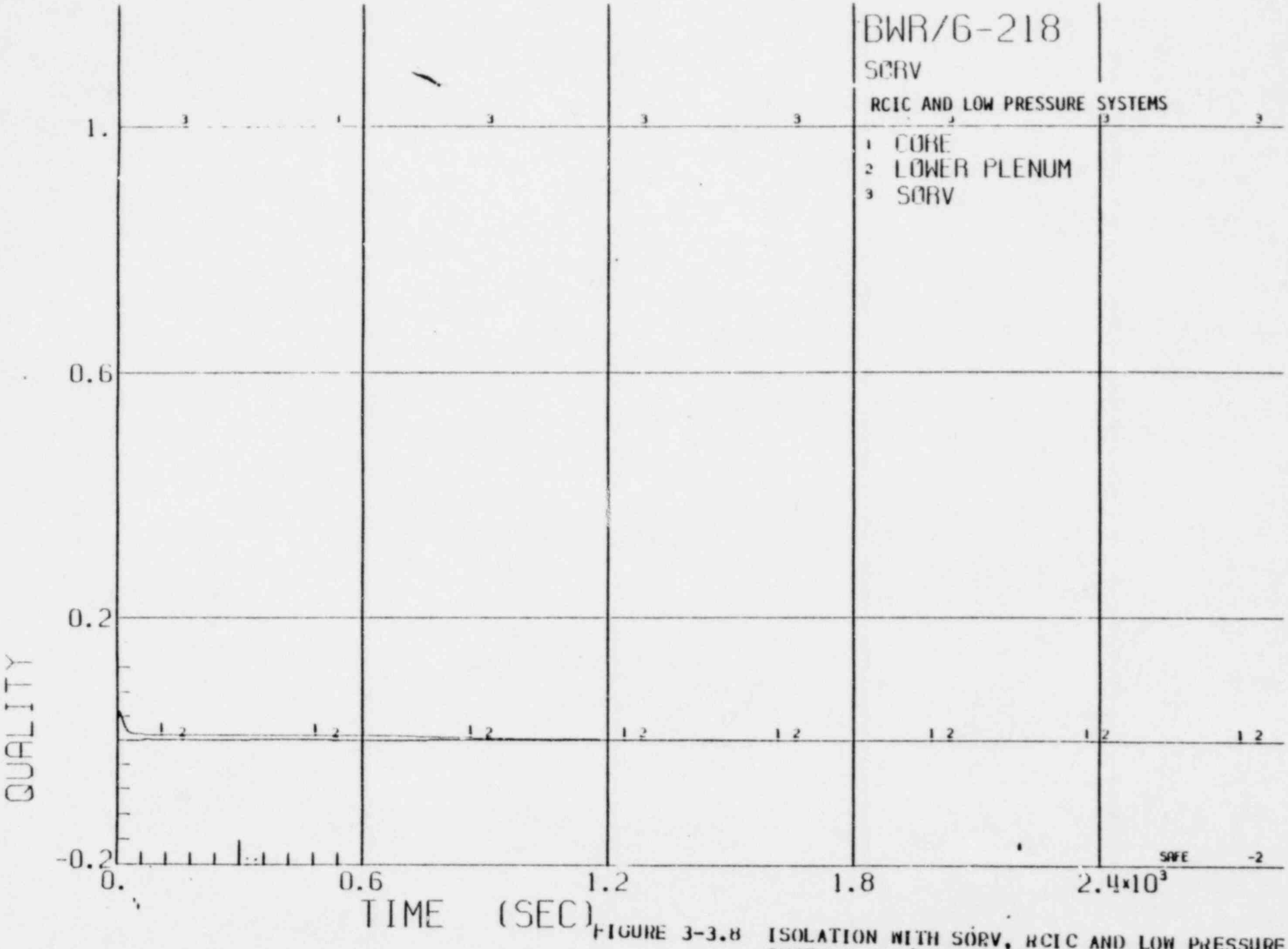


FIGURE 3-3.8 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

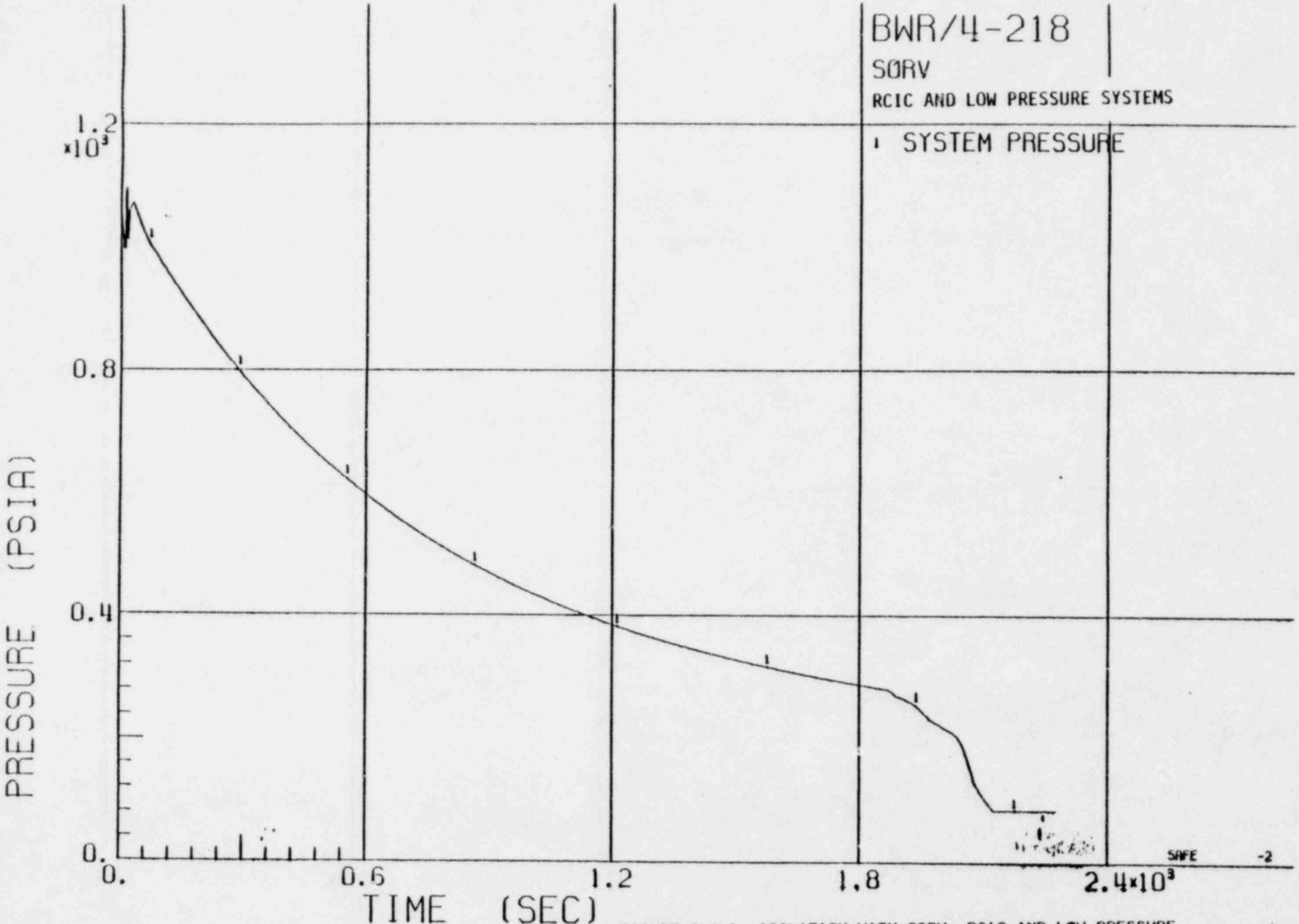


FIGURE 3-4.1 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

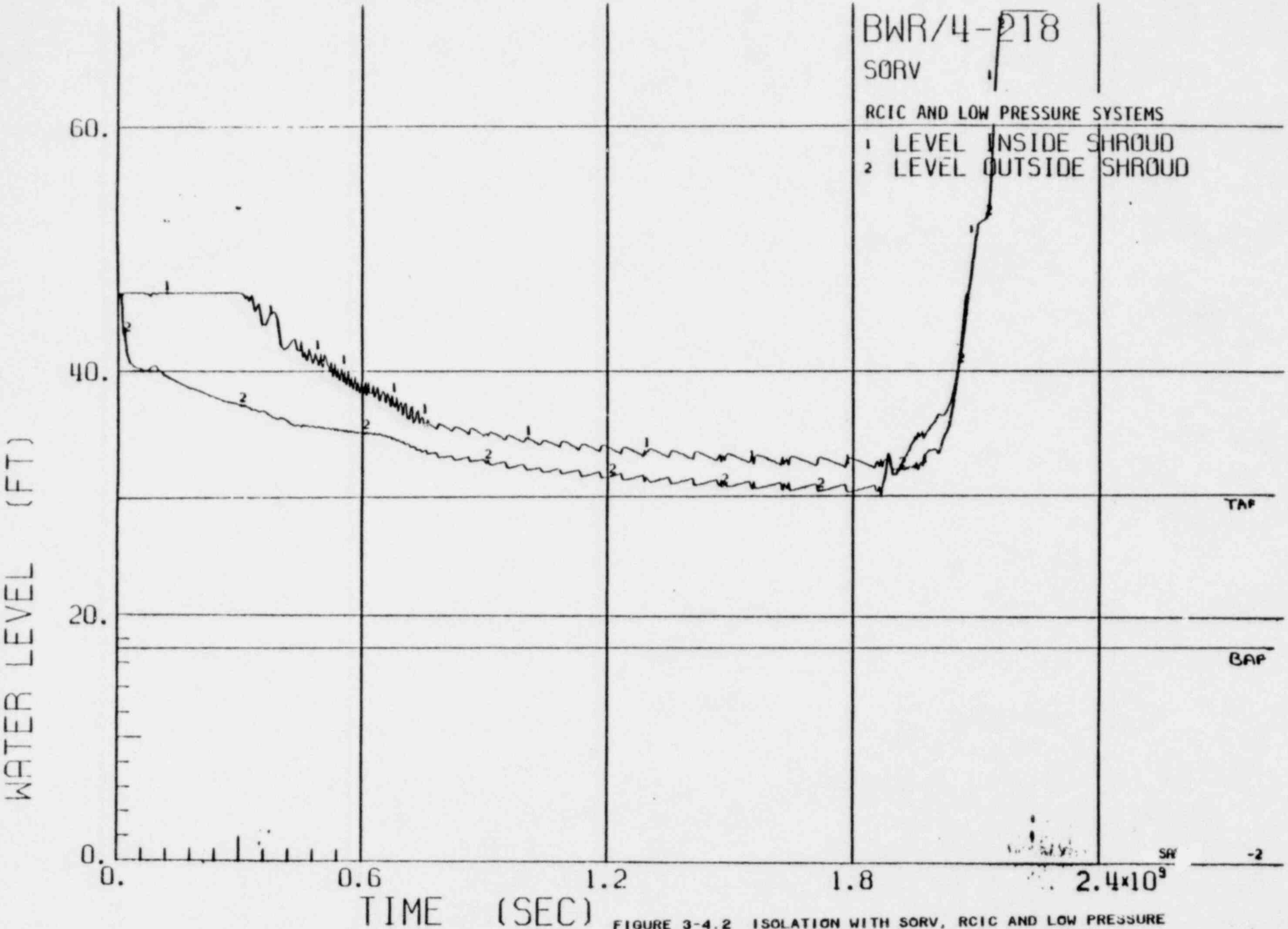


FIGURE 3-4.2 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

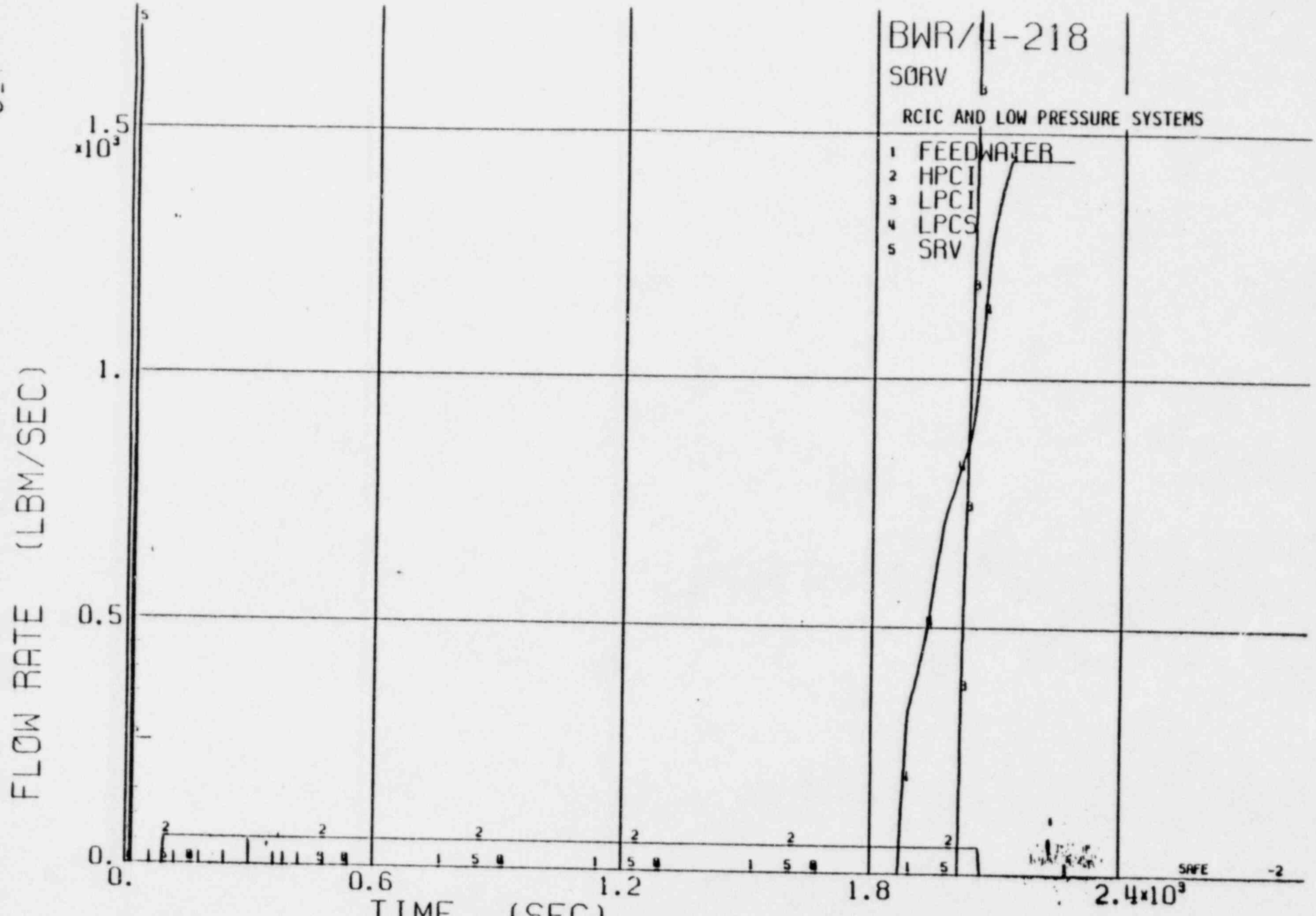


FIGURE 3-4.3 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE NO ADS ACTUATION

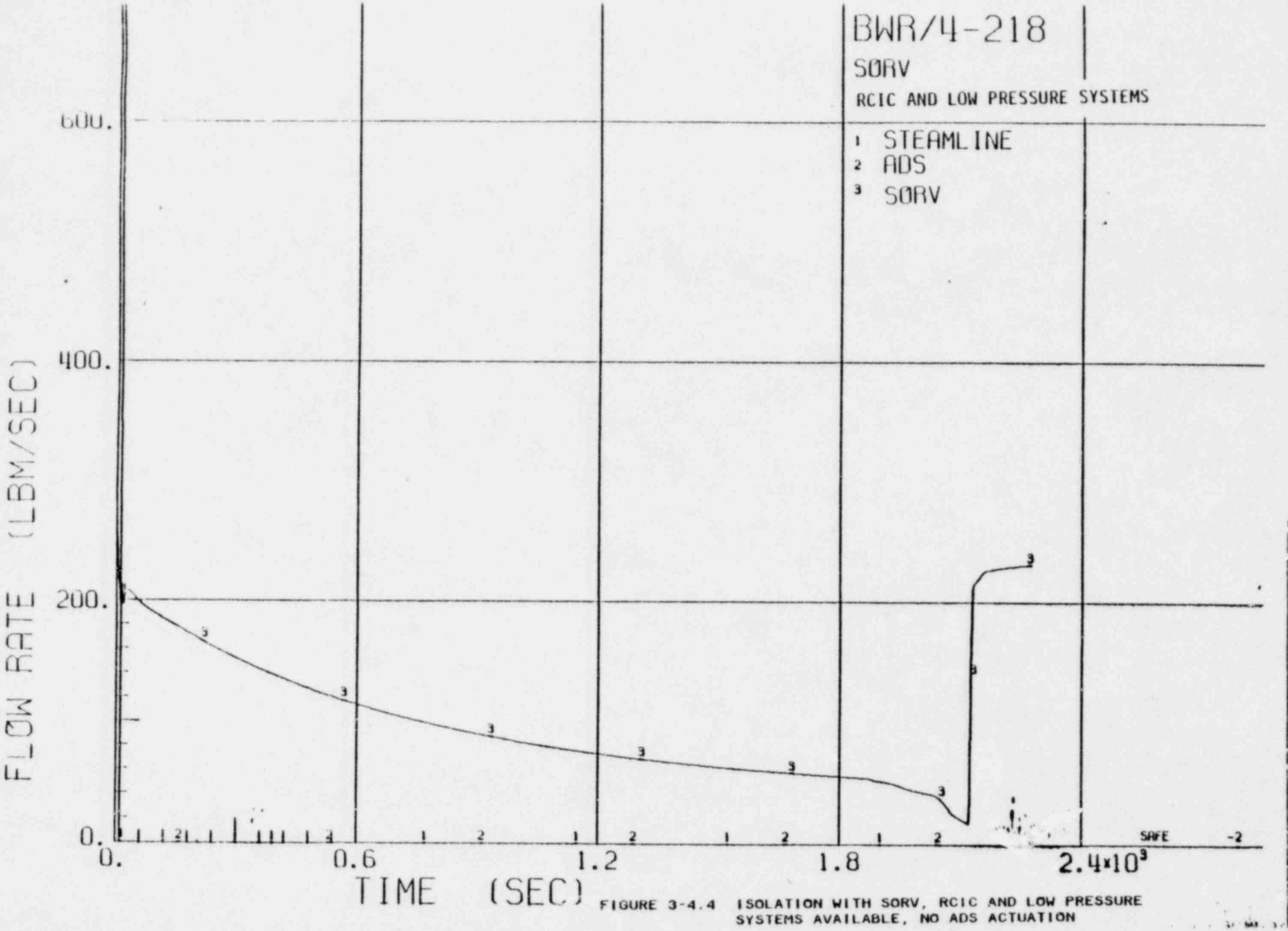


FIGURE 3-4.4 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

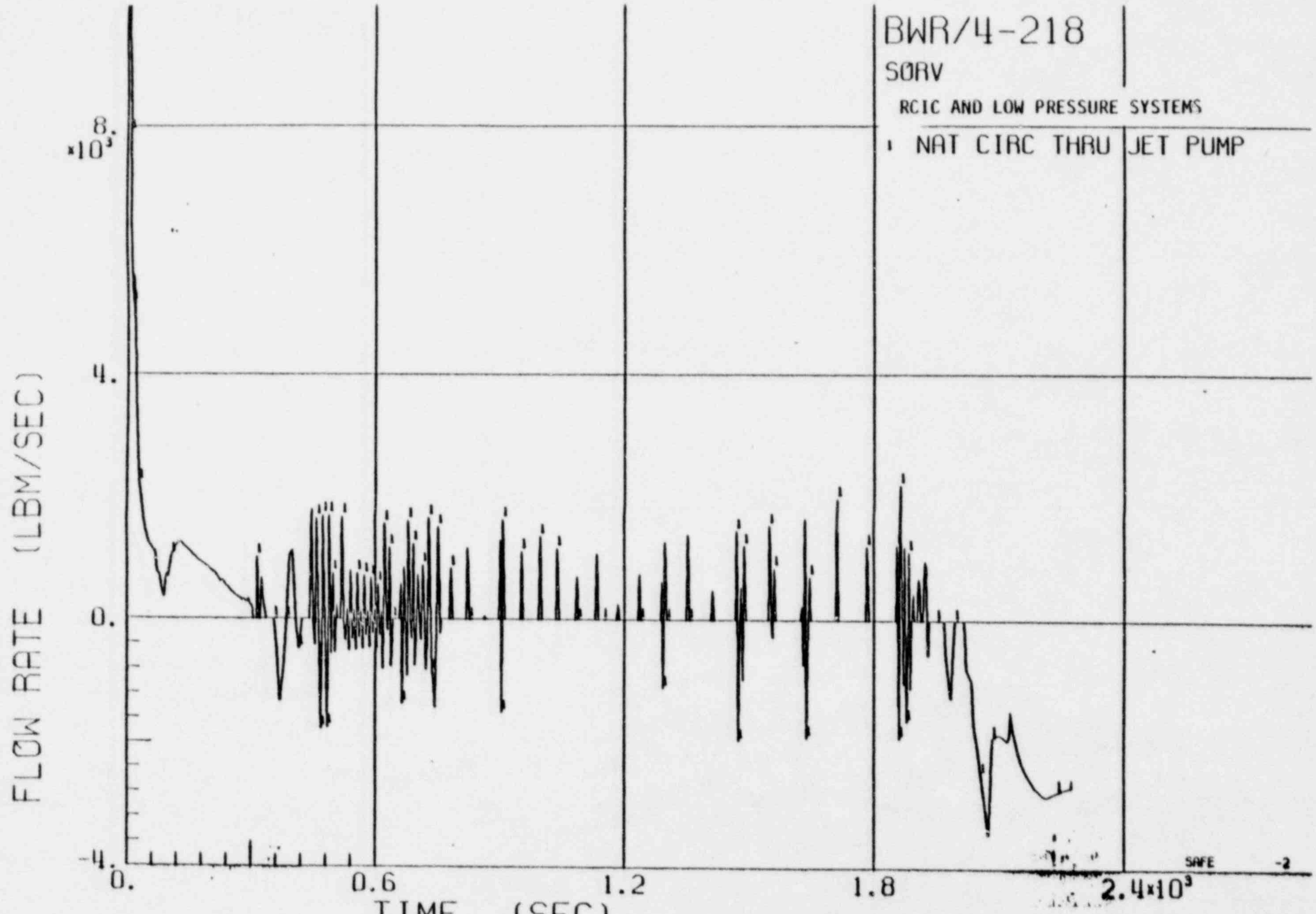


FIGURE 3-4.5 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

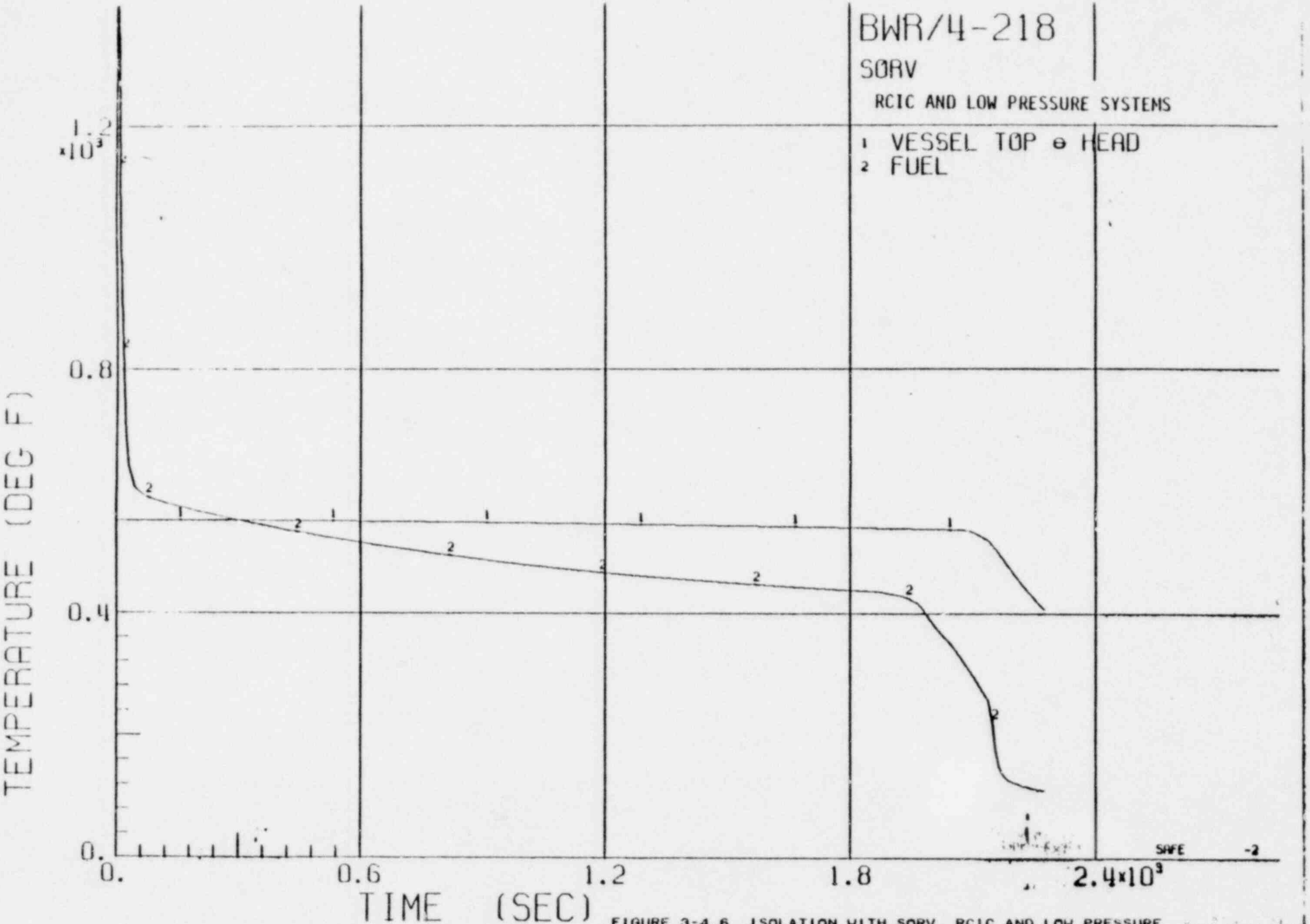


FIGURE 3-4.6 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

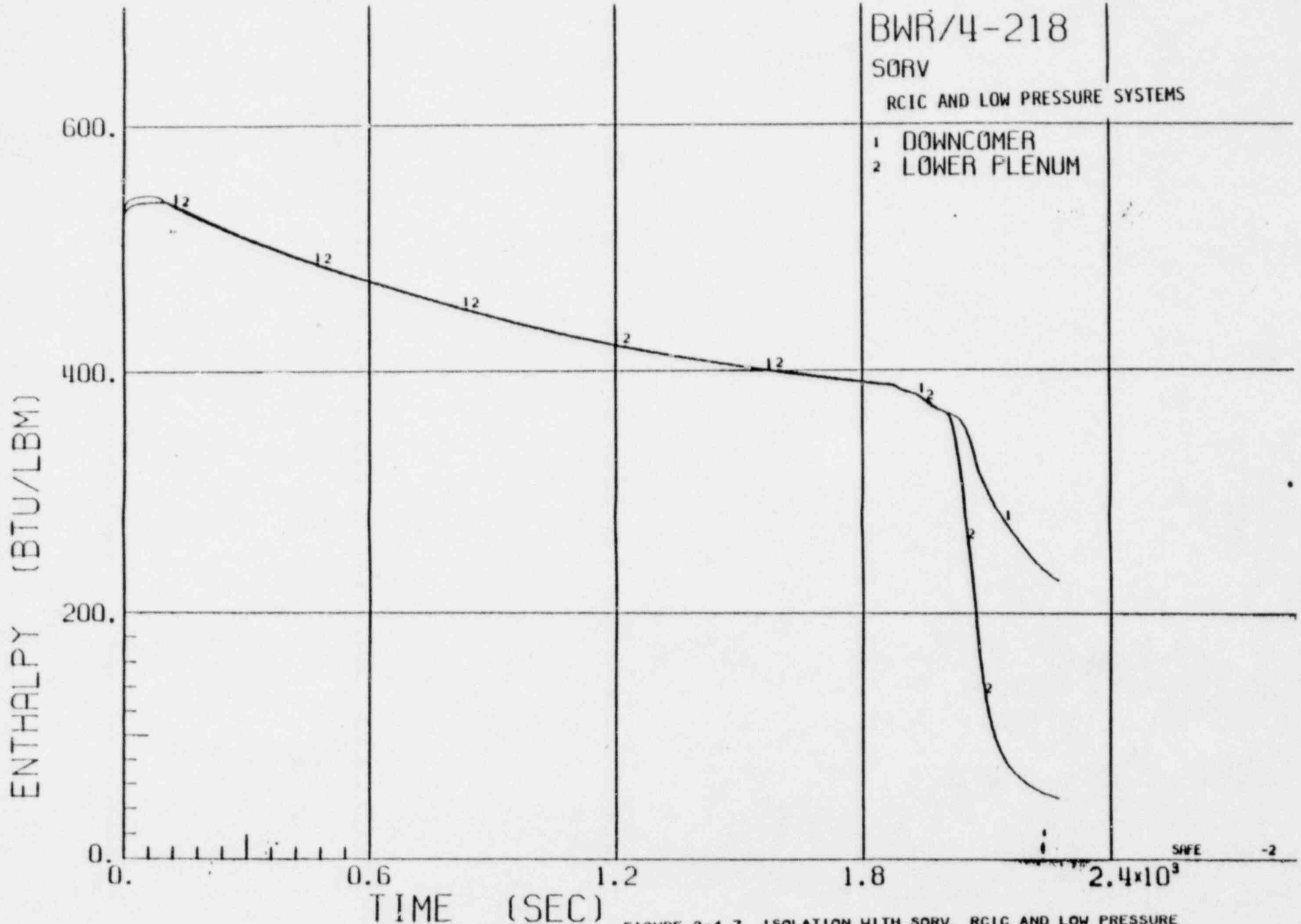


FIGURE 3-4.7 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

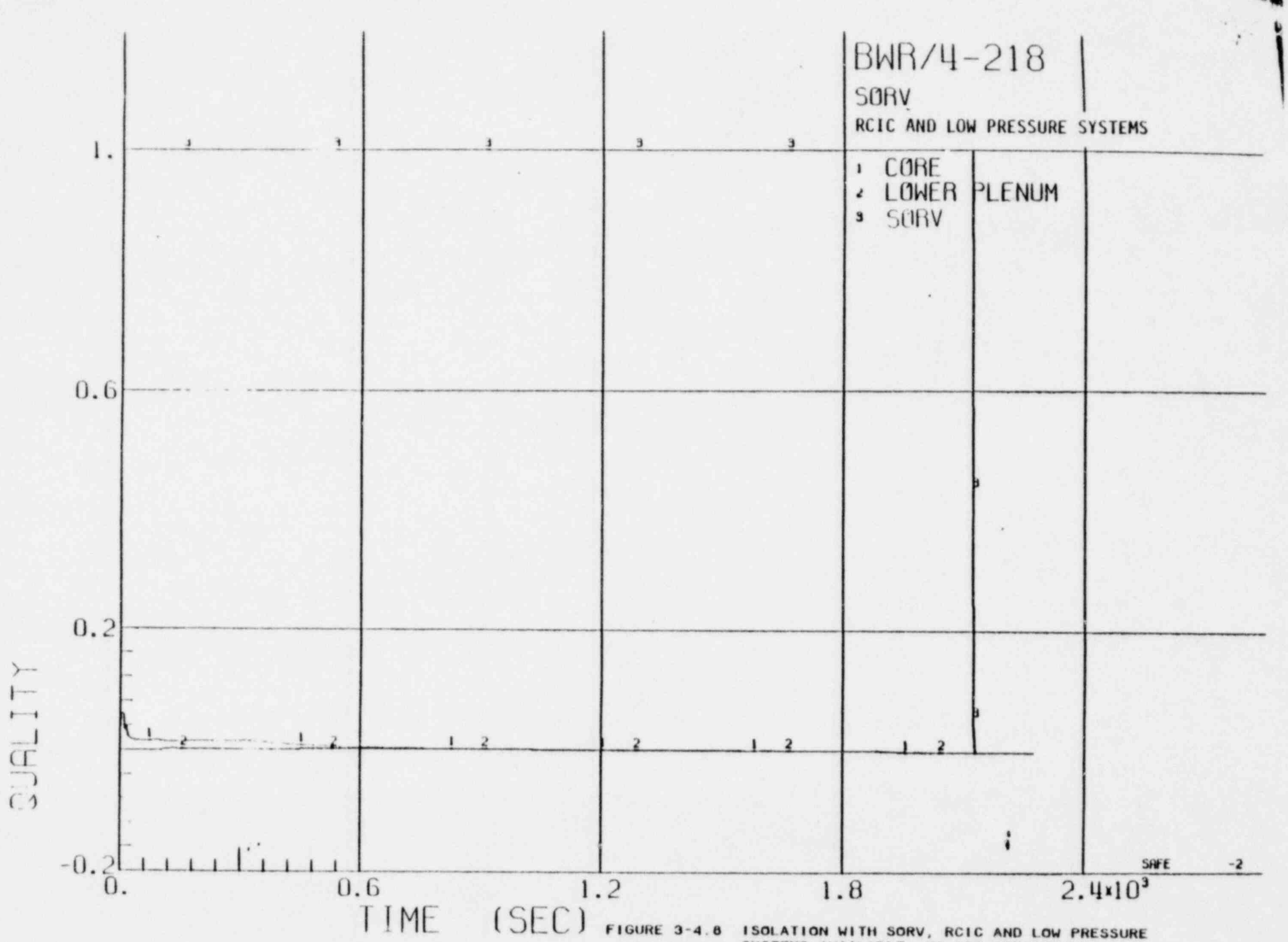


FIGURE 3-4.8 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

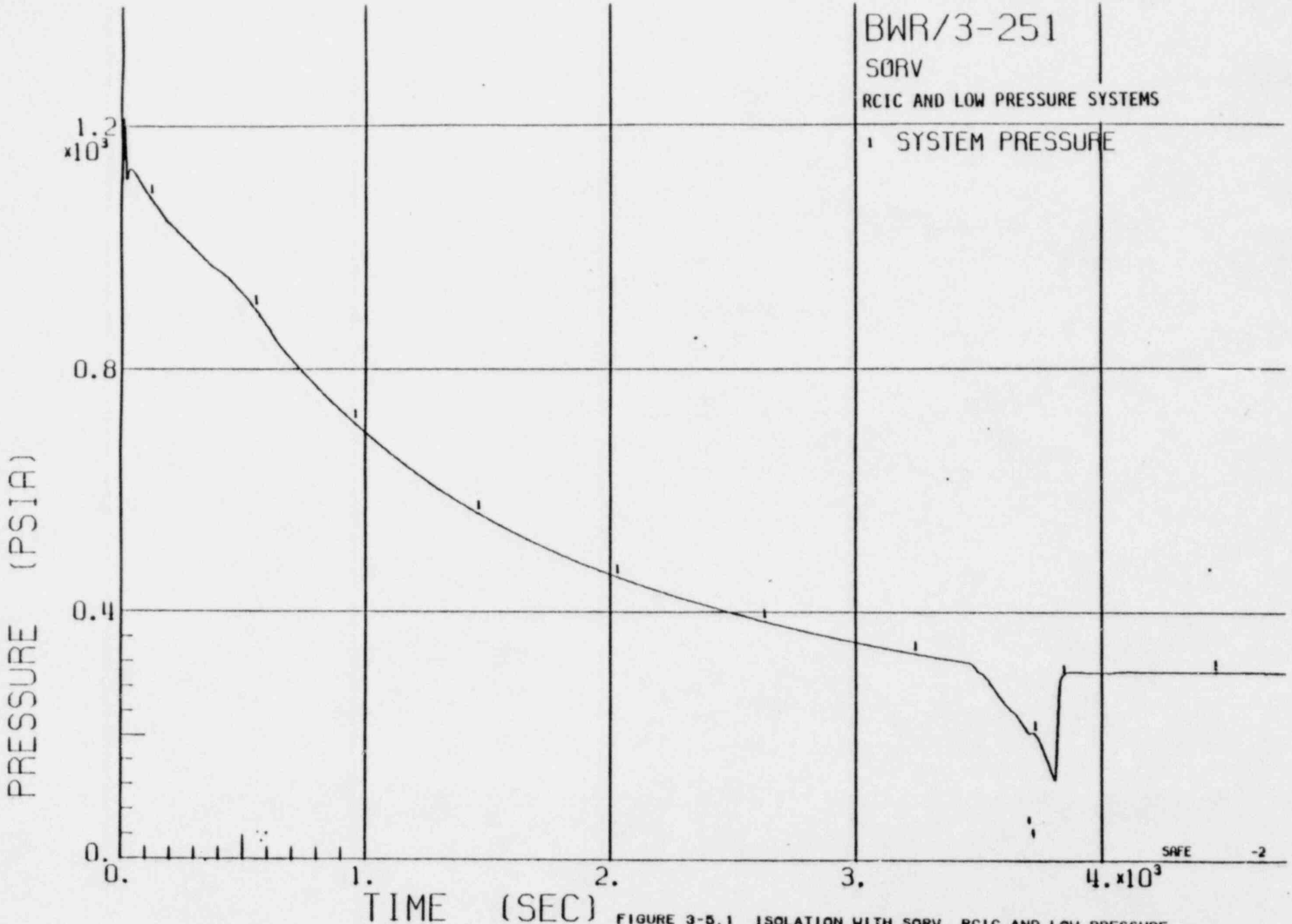


FIGURE 3-5.1 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

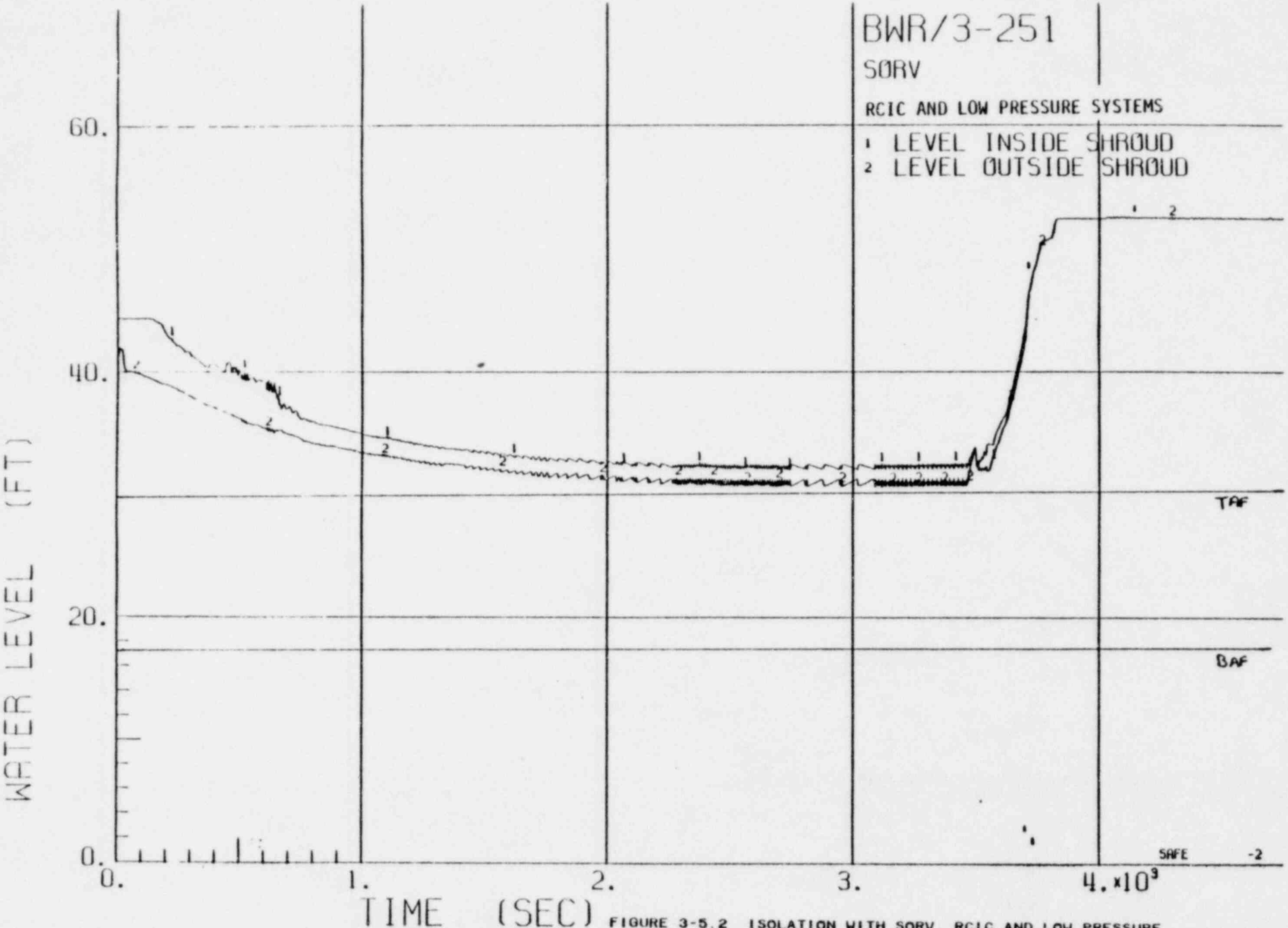


FIGURE 3-5.2 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

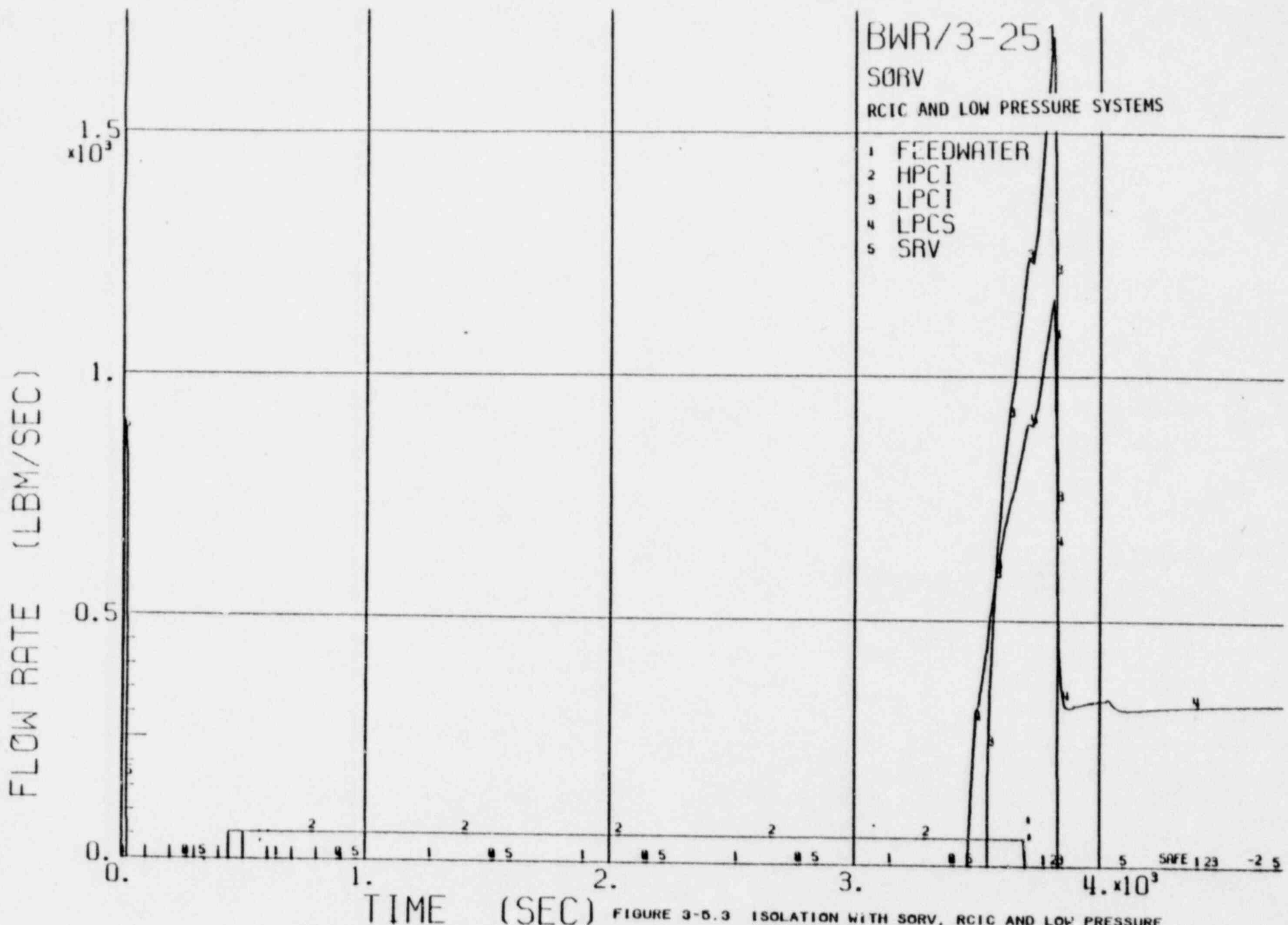


FIGURE 3-5.3 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

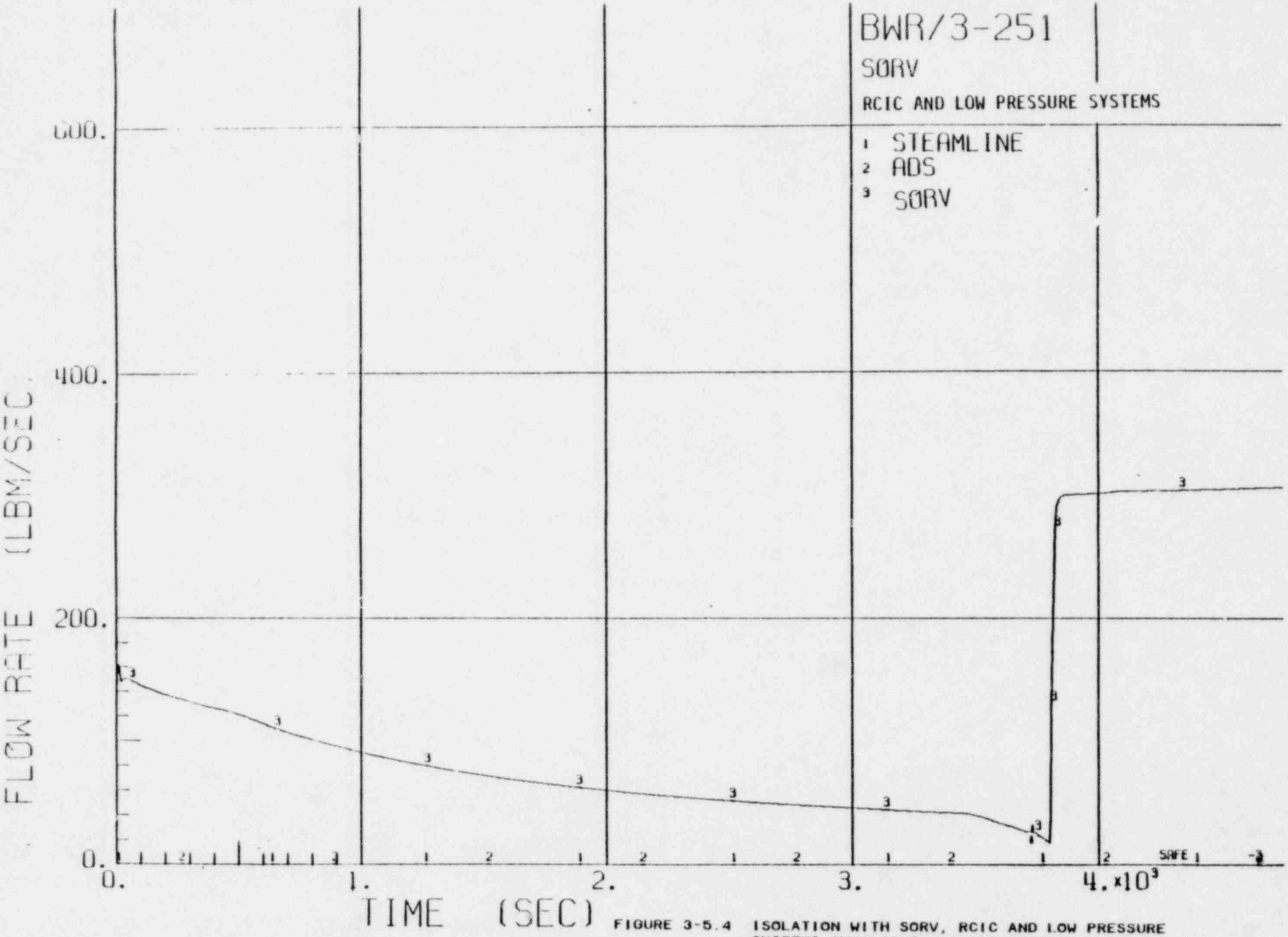


FIGURE 3-5.4 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

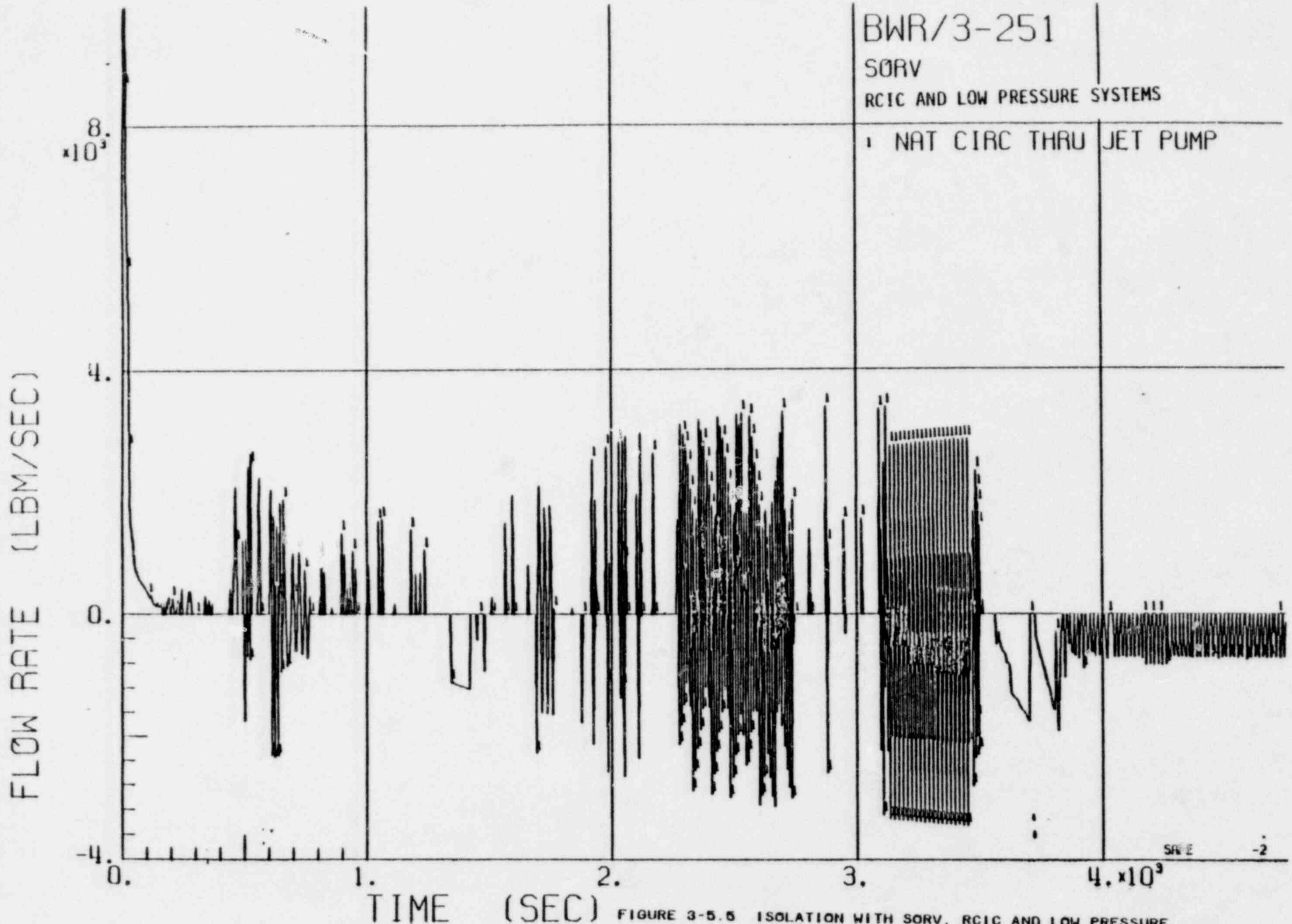


FIGURE 3-5.5 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

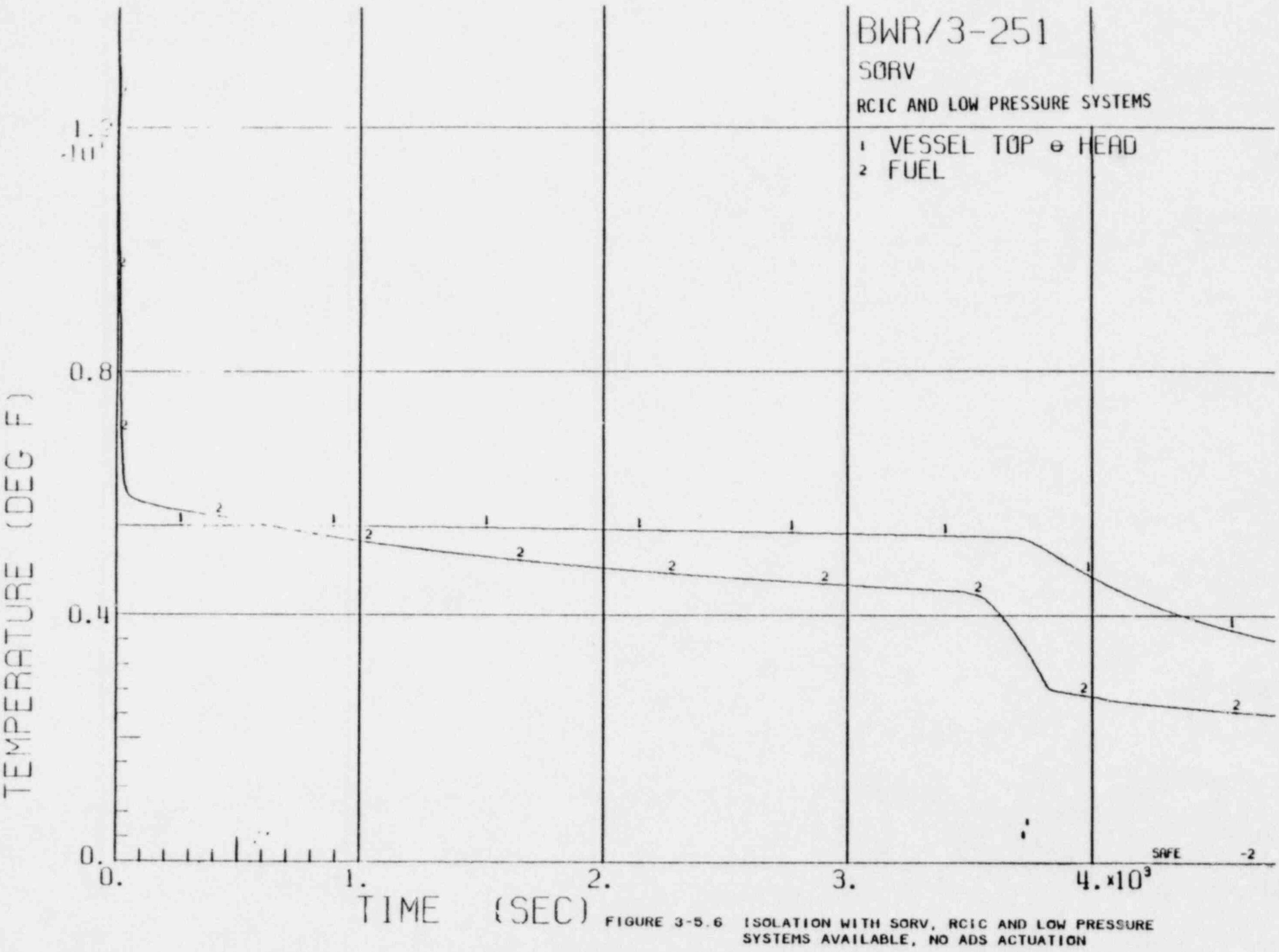


FIGURE 3-5.6 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

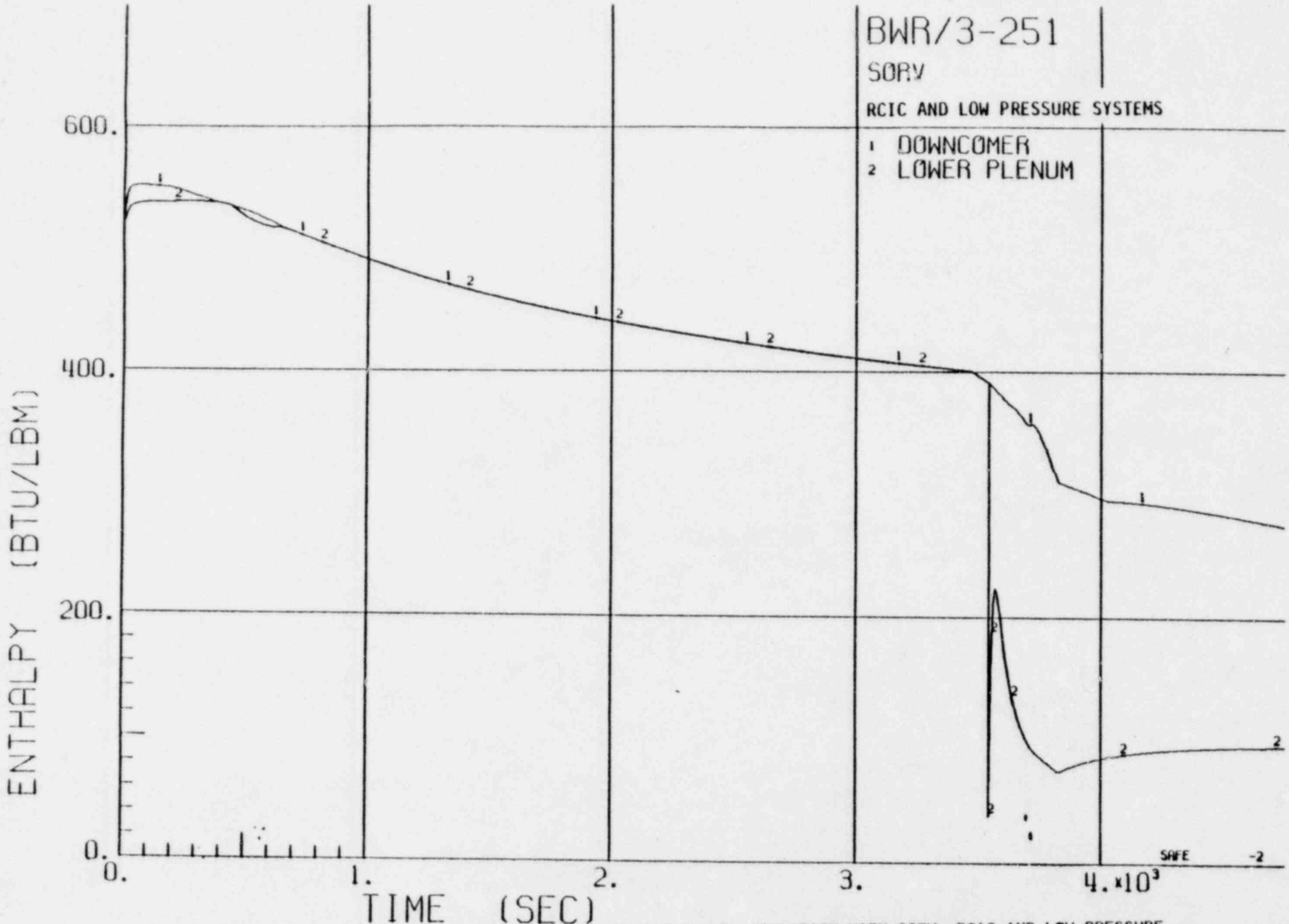


FIGURE 3-5.7 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

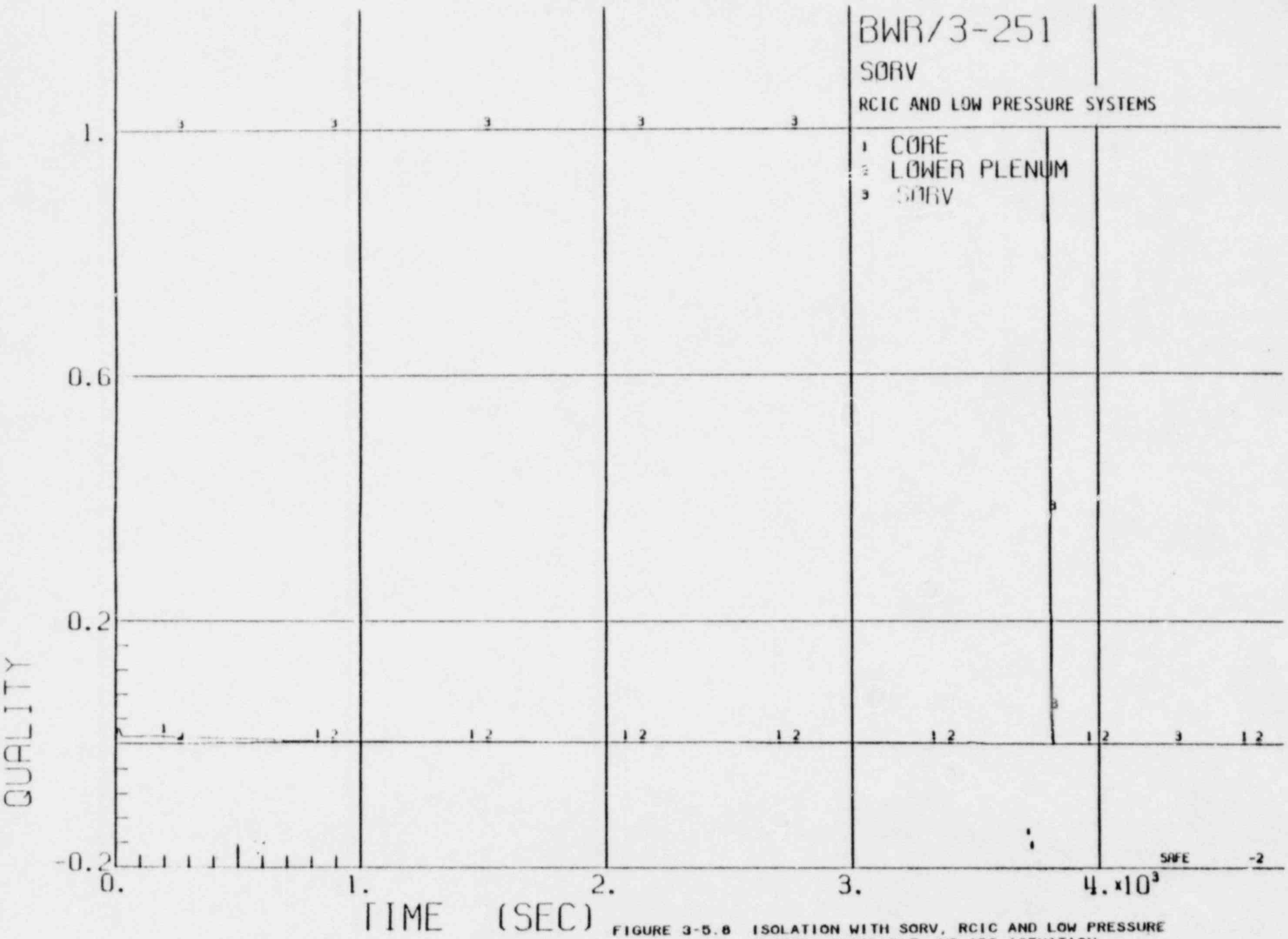


FIGURE 3-5.8 ISOLATION WITH SORV, RCIC AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

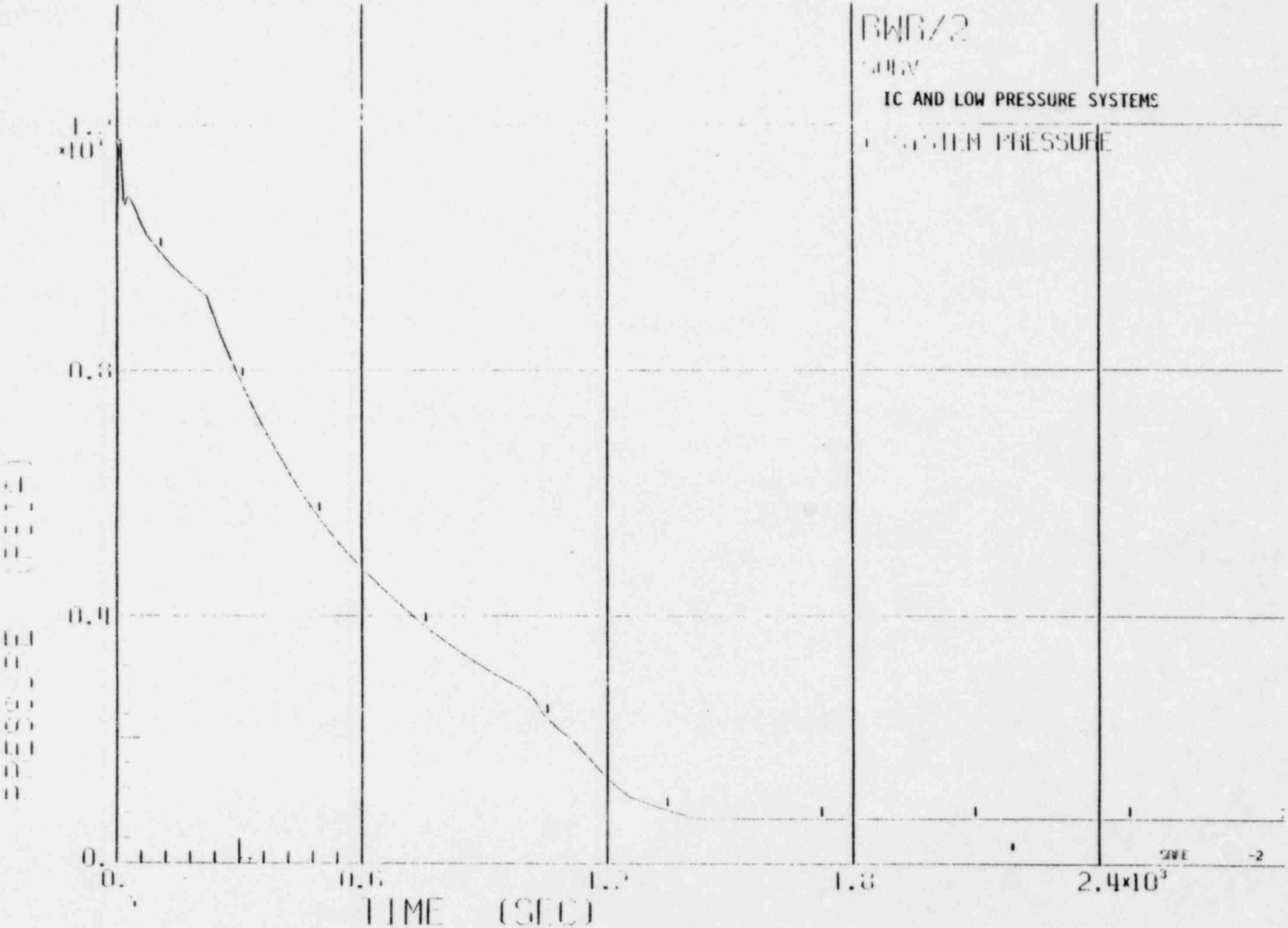


FIGURE 3-6.1 ISOLATION WITH SORV, ISOLATION CONDENSERS AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

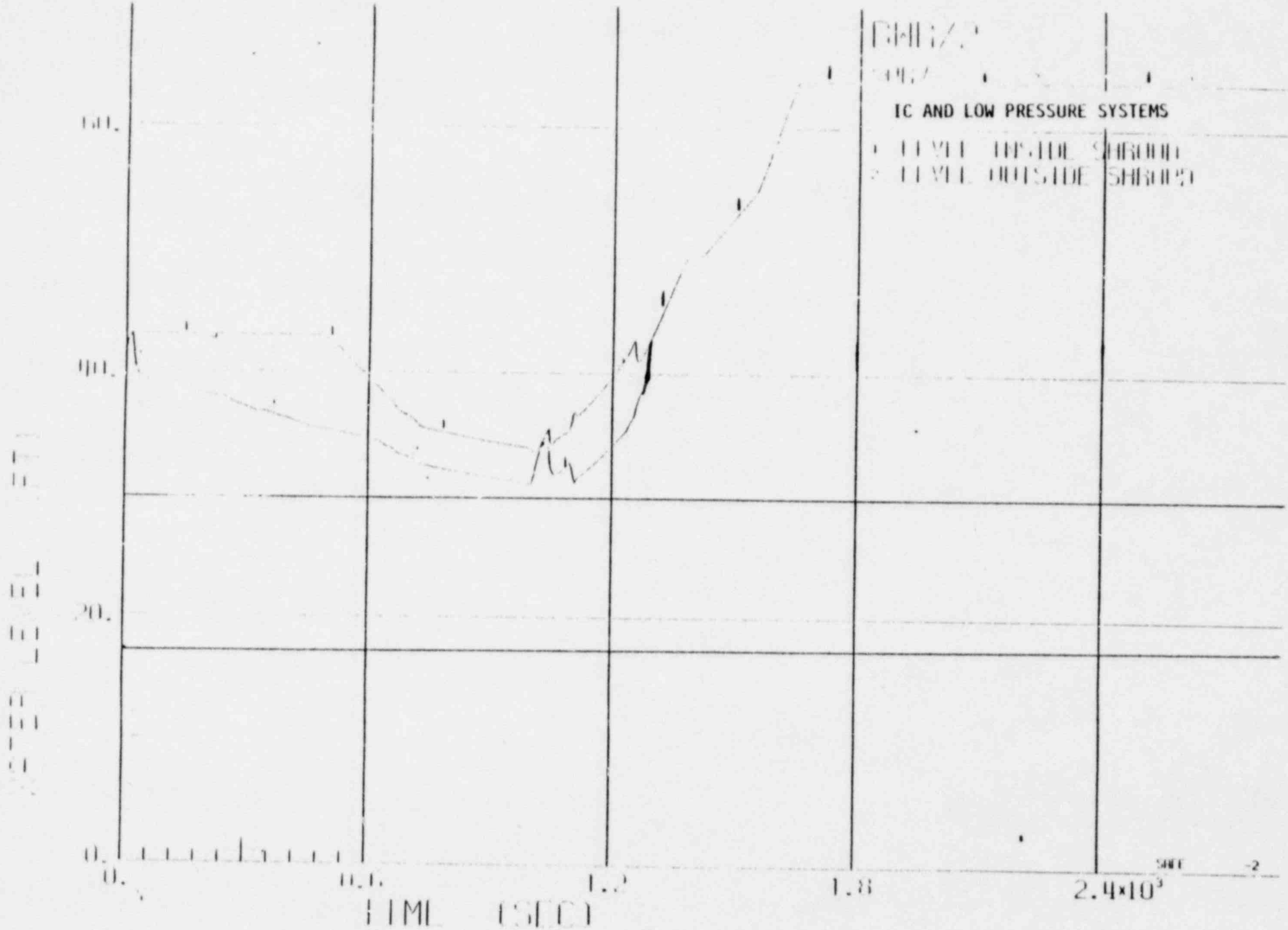


FIGURE 3-6.2 ISOLATION WITH SORV, ISOLATION CONDENSERS AND LOW

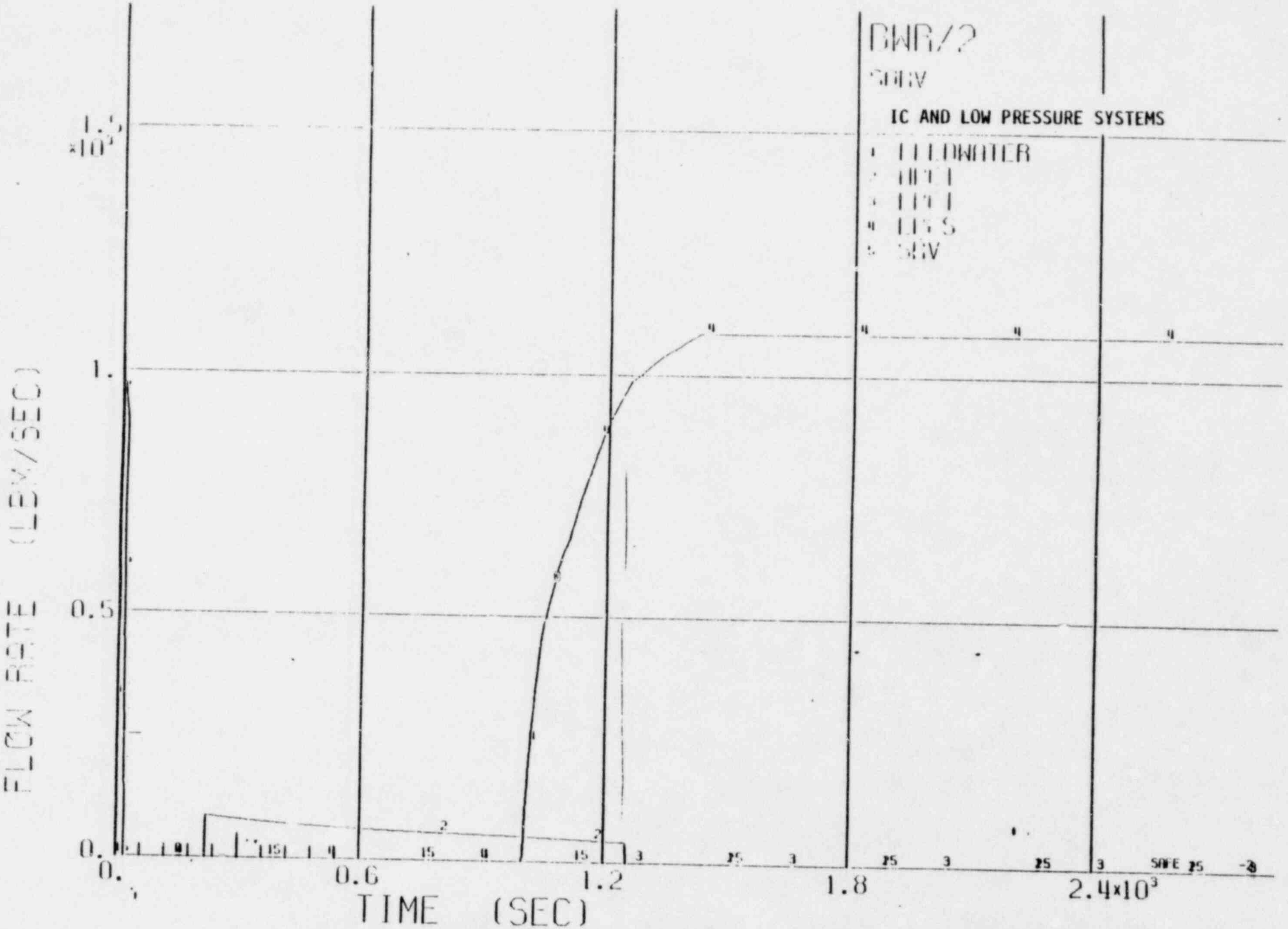


FIGURE 3-6.3 ISOLATION WITH SORV, ISOLATION CONDENSERS AND LOW

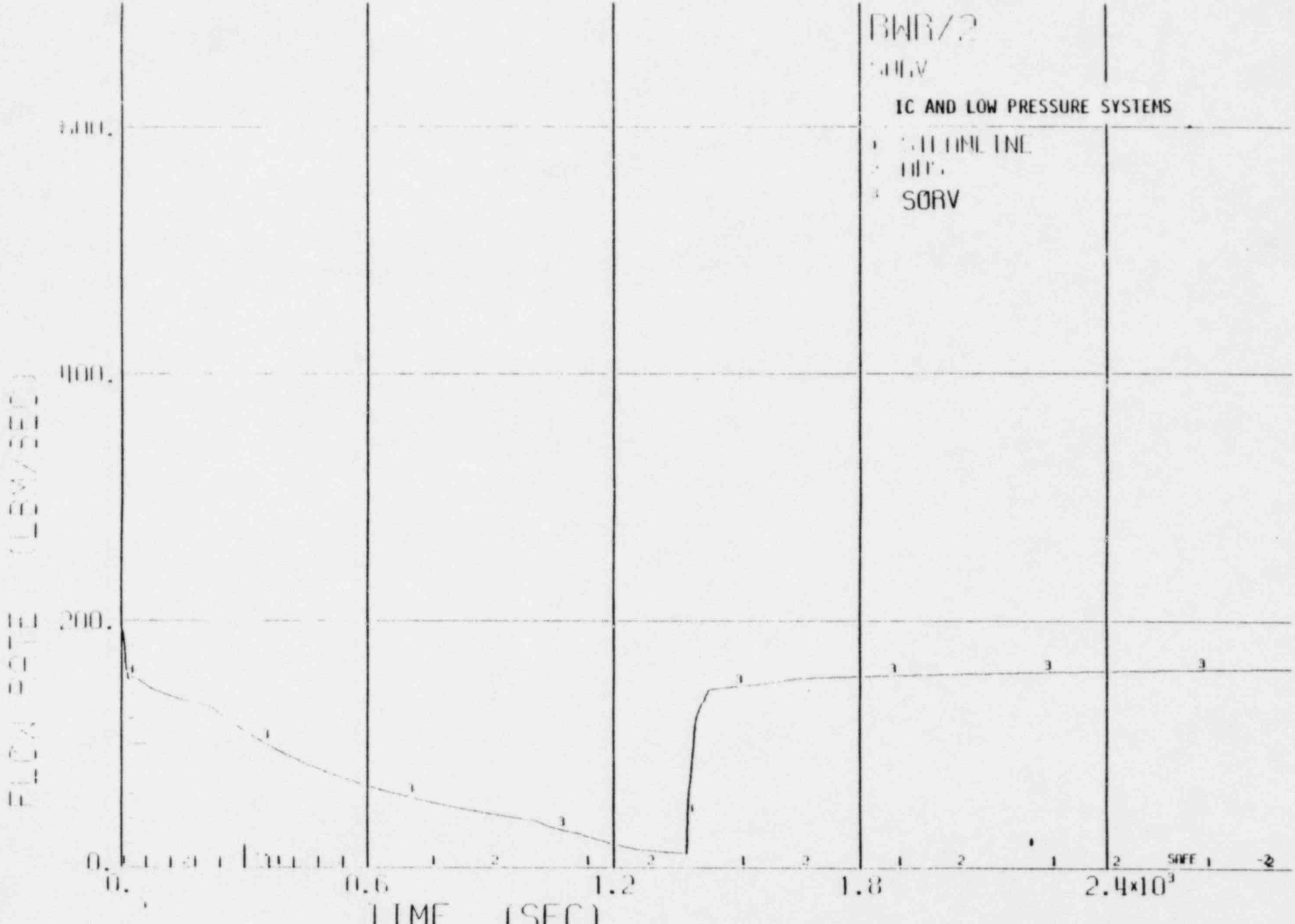


FIGURE 3-6.4 ISOLATION WITH SORV, ISOLATION CONDENSERS AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

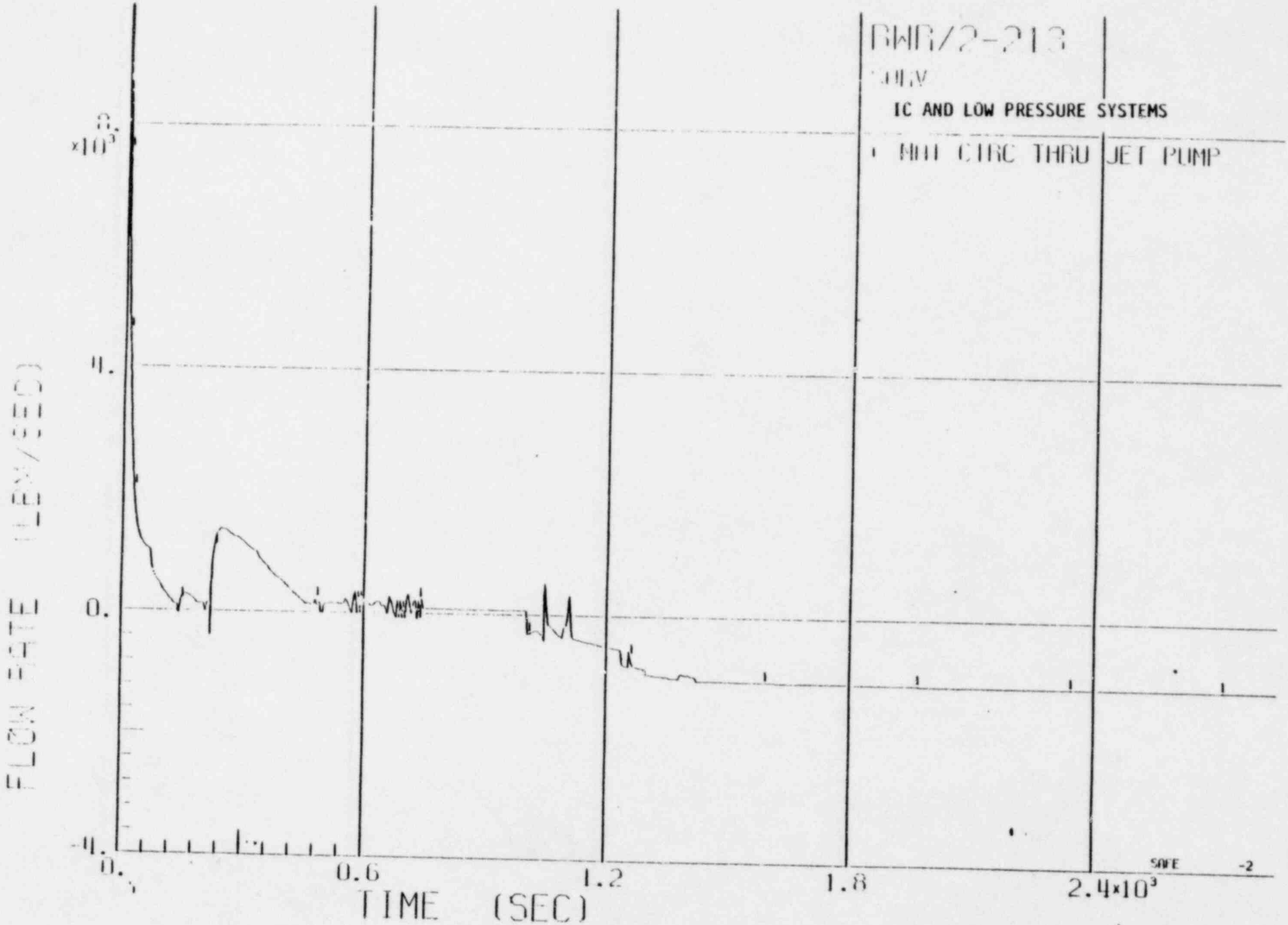


FIGURE 3-6.5 ISOLATION WITH SORV, ISOLATION CONDENSERS AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

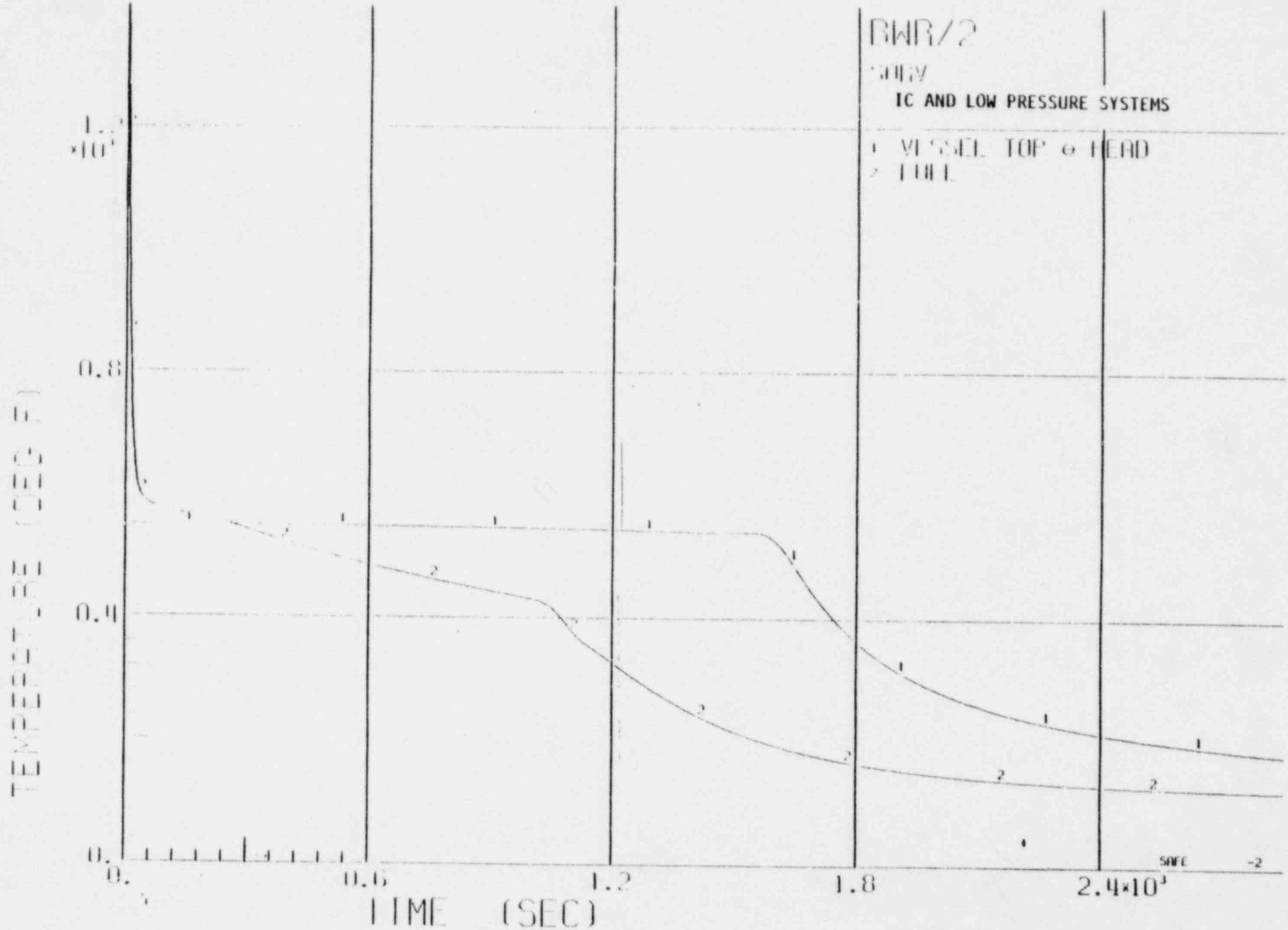


FIGURE 3-6.6 ISOLATION WITH SORV, ISOLATION CONDENSERS AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

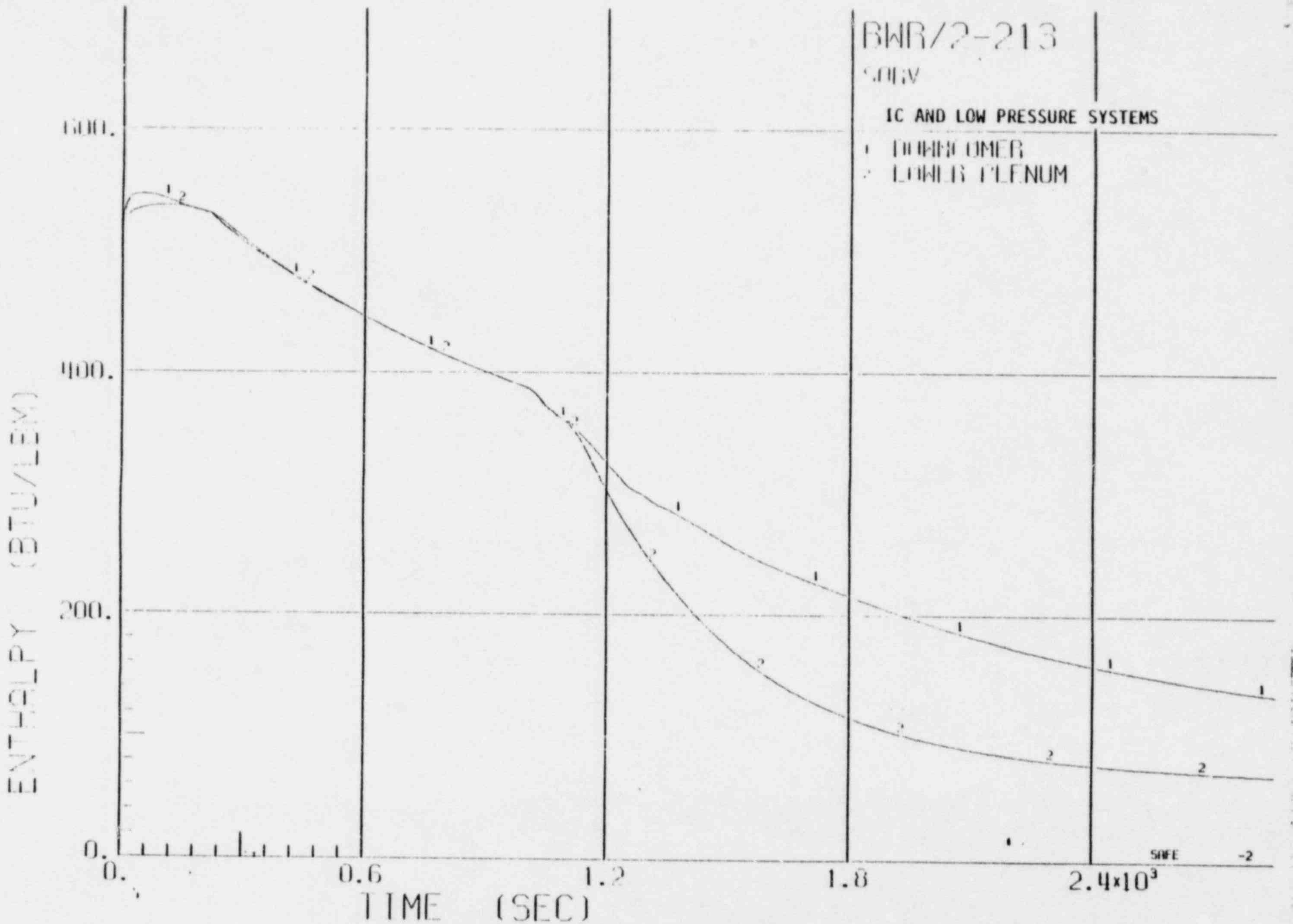


FIGURE 3-6.7 ISOLATION WITH SORV, ISOLATION CONDENSERS AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

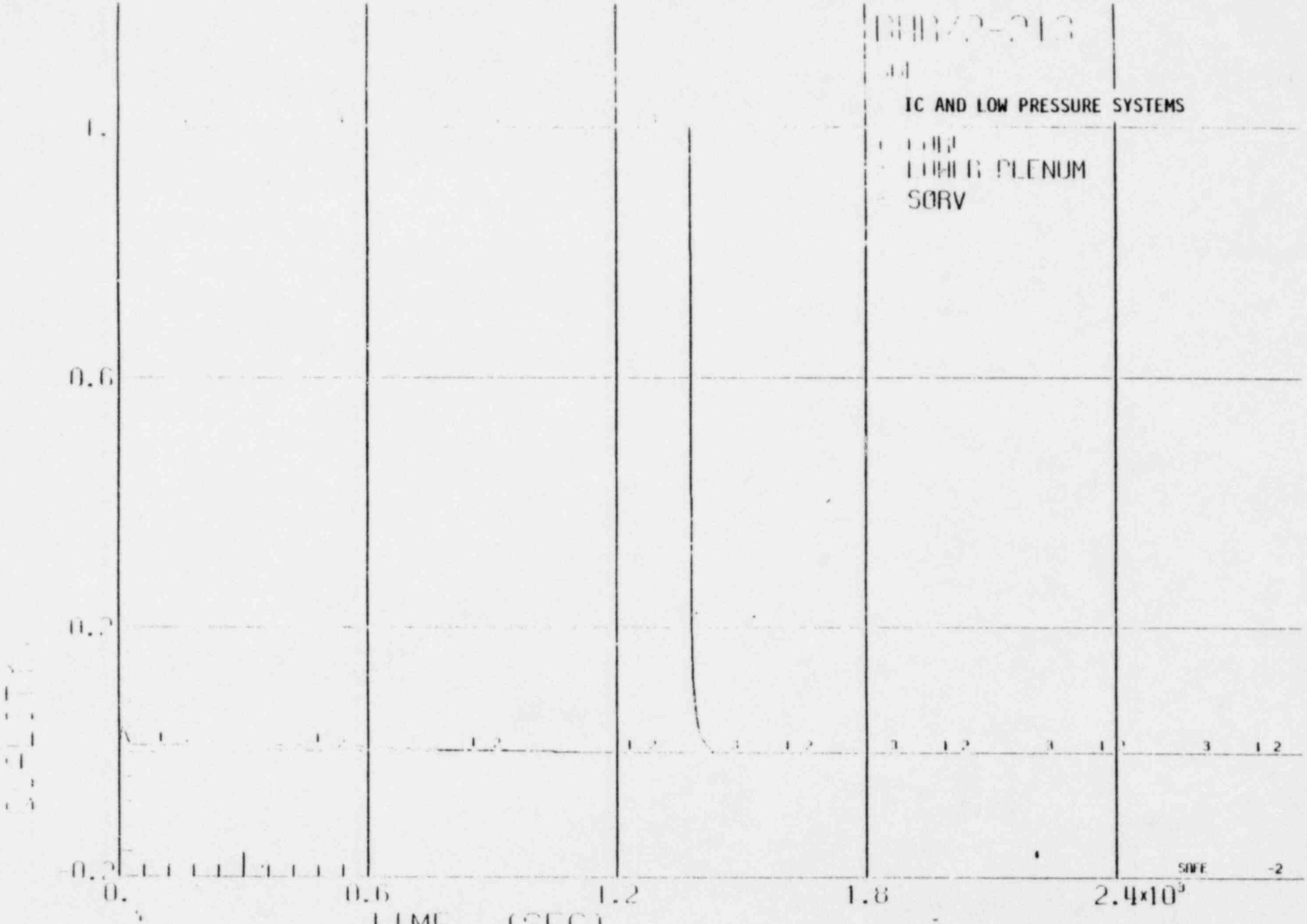


FIGURE 3-6.8 ISOLATION WITH SORV, ISOLATION CONDENSERS AND LOW PRESSURE SYSTEMS AVAILABLE, NO ADS ACTUATION

TABLE 5-1: COMPARISON OF ADS LOGIC OPTIONS
ADS LOGIC OPTIONS

CRITERIA	ELIMINATE HIGH DRYWELL PRESSURE TRIP	ADD BYPASS TIMER STARTED ON SCRAM WATER LEVEL	ADD BYPASS TIMER STARTED ON ECCS INITIATION LEVEL	AND SUPPRESSION POOL TEMPERATURE TRIP	ADD NO HIGH PRESSURE SYSTEM TRIP
I. ADS PERFORMANCE**					
1. HARDWARE COMPLEXITY	SIMPLE REWIRING NEEDED	SIMPLE HARDWARE ADDITION	SIMPLE HARDWARE ADDITION	VERY COMPLEX POOL TEMPERATURE SYSTEM REQUIRED	COMPLEX FLOW MEASURING HARDWARE REQUIRED
2. RELIABILITY - ADS ACTUATION WHEN NEEDED FOR ISOLATION, SORV	HIGHLY RELIABLE	HIGHLY RELIABLE	HIGHLY RELIABLE	RELIABILITY DEPENDENT ON POOL TEMPERATURE SYSTEM RELIABILITY AND NUMBER OF SENSORS IN POOL CAN BE MADE RELIABLE	RELIABILITY DEPENDENT ON FLOW MEASUREMENT LOW FEEDWATER FLOW MEASUREMENT RELIABILITY QUESTIONABLE MAY PREVENT ADS ACTUATION IF FLOW PRESENT BUT NOT REACHING VESSEL (I.E., BREAK IN LINE)
3. MEETS NUREG-0737 INTENT	YES	YES	YES	YES	YES

**Compared to current design.

TABLE 5-1: COMPARISON OF ADS LOGIC OPTIONS
ADS LOGIC OPTIONS**

CRITERIA	ELIMINATE HIGH DRYWELL PRESSURE TRIP	ADD BYPASS TIMER STARTED ON SCRAM WATER LEVEL	ADD BYPASS TIMER STARTED ON ECCS INITIATION LEVEL	AND SUPPRESSION POOL TEMPERATURE TRIP	ADD NO HIGH PRESSURE SYSTEM TRIP
II. INSTALLATION AND MAINTENANCE**					
1. COST	LOW	LOW	LOW	VERY HIGH WITH ADDITION OF POOL TEMPERATURE SYSTEM, MODERATELY HIGH IF POOL SYSTEM IN PLACE	HIGH
2. IMPACT ON MAINTENANCE OR CONSTRUCTION SCHEDULE	NONE	SMALL	SMALL	MAY CAUSE SIGNIFICANT DELAY	MAY CAUSE SIGNIFICANT DELAY
3. MAINTENANCE	MAINTENANCE REDUCED	SLIGHT INCREASE IN MAINTENANCE	SLIGHT INCREASE IN MAINTENANCE	SIGNIFICANT INCREASE IN MAINTENANCE	INCREASED MAINTENANCE
4. SURVEILLANCE TESTING	TESTING REDUCED	SMALL INCREASE IN TESTING	SMALL INCREASE IN TESTING	SIGNIFICANT INCREASE IN TESTING	INCREASED TESTING
III. IMPACT ON PLANT OPERATIONS**					
1. SPURIOUS ACTUATIONS*	SLIGHT INCREASE IN PROBABILITY OF SPURIOUS ACTUATION	SLIGHT INCREASE IN PROBABILITY OF SPURIOUS ACTUATION	SLIGHT INCREASE IN PROBABILITY OF SPURIOUS ACTUATION (LESS LIKELY THAN SCRAM LEVEL START)	ABOUT SAME PROBABILITY AS CURRENT DESIGN	ABOUT SAME PROBABILITY AS CURRENT DESIGN
2. INADVERTENT ACTUATIONS-LOGIC PERFORMS AS DESIGNED BUT ADS NOT REQUIRED	ADS MAY ACTUATE FOR BWR/1-3 WHEN NOT NEEDED	ADS MAY ACTUATE FOR BWR/1-3 WHEN NOT NEEDED	ADS MAY ACTUATE FOR BWR/1-3 WHEN NOT NEEDED (LESS LIKELY	ADS MAY ACTUATE FOR FOR BWR/1-3 WHEN NOT NEEDED	NORMAL HPCI/HPCS CYCLIC OPERATION INITIATES ADS LOGIC

* INADVERTENT MANUAL ACTION IS EXPECTED TO DECREASE WITH AUTOMATION AND IS EXPECTED TO OFFSET ANY INCREASE IN SPURIOUS ACTUATIONS.
** COMPARED TO CURRENT DESIGN.

TABLE 5-1: COMPARISON OF ADS LOGIC OPTIONS
ADS LOGIC OPTIONS

CRITERIA	ELIMINATE HIGH DRYWELL PRESSURE TRIP	ADD BYPASS TIMER STARTED ON SCRAM WATER LEVEL	ADD BYPASS TIMER STARTED ON ECCS INITIATION LEVEL	AND SUPPRESSION POOL TEMPERATURE TRIP	ADD NO HIGH PRESSURE SYSTEM TRIP
IV. CONCLUSIONS	VIABLE	VIABLE	VIABLE	VIABLE	NOT RECOMMENDED

APPENDIX A

Participating Utilities

NUREG-0737 II.K.3.18

This report applies to the following plants, whose Owners participated in the report's development.

Utility	Plant
Boston Edison	Pilgrim 1
Carolina Power & Light	Brunswick 1&2
Commonwealth Edison	LaSalle 1&2, Dresden 2-3, Quad Cities 1&2
Georgia Power	Hatch 1&2
Iowa Electric Light & Power	Duane Arnold
Niagara Mohawk Power	Nine Mile Point 1&2
Nebraska Public Power District	Cooper
Northeast Utilities	Millstone 1
Northern States Power	Monticello
Philadelphia Electric	Peach Bottom 2&3, Limerick 1&2
Power Authority of the State of New York	FitzPatrick
Detroit Edison	Enrico Fermi 2
Long Island Lighting	Shoreham
Mississippi Power & Light	Grand Gulf 1&2
Pennsylvania Power & Light	Susquehanna 1&2
Washington Public Power Supply System	Hanford 2
Cleveland Electric Illuminating	Perry 1&2
Houston Lighting & Power	Allens Creek
Illinois Power	Clinton Station 1&2
Public Service of Oklahoma	Black Fox 1&2
Vermont Yankee Nuclear Power	Vermont Yankee
Jersey Central Power and Light	Oyster Creek 1
Tennessee Valley Authority	Browns Ferry 1-3; Hartsville 1-4; Phipps Bend 1-2
Gulf States	River Bend