3.2.2 OTHER OPERATIONAL TRANSIENTS

Chapter 15 of the Final Safety Analysis Report (FSAR) for a nuclear power plant includes analyses of several events expected to occur during the life of an individual plant. Limiting faults, not expected to occur during a plant lifetime, are also considered. These events are analyzed to demonstrate the acceptable performance of the reactor and balance of plant systems during transients and accidents. Bounding assumptions are made with respect to such variables as initial plant conditions and the performance of mitigating systems. As a result, each event description in the FSAR is often applicable to only one of several sequences of events which can follow from a single initiating disturbance, and the calculated results may be atypically severe.

In the standard (Reg. Guide 1.70 Rev. 2) FSAR, "operator actions" are specified for each event. An FSAR Chapter 15 Event Review was conducted to assess the validity of the specified actions with respect to reactor safety, making realistic assessments of plant conditions and responses for each event. As agreed in the September 6, 1979, meeting between the BWR Owners' Group and the Bulletins and Orders Task Force staff, this study comprised the following:

1. Consideration of each Chapter 15 event which results in a reactor transient (excluding ATWS) for a <u>typical</u> BWR/4;

2. Determination of effects of best-estimate assumptions on predicted symptoms or anticipated operator actions;

3. Identification of events where effects are significant;

4. Revision of symptoms and operator actions (if appropriate).

The results of this study are documented in this section.

This study is not intended to cover all BWR product lines in complete detail (although most of its conclusions have generic applicability). Generic actions to respond to emergencies are specified in the Emergency Procedure Guidelines. Further, the results of this study are not appropriate for use in training (even for the sample plant from which it was developed) due to the inherent

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narrow definition of each Chapter 15 event. The usefulness of this study lies in its conclusion that even within the narrow definitions of Chapter 15 events and the conservative assumptions used in their analysis, there is nothing inherent in the analyses or specified actions of the FSAR which would mislead the operator or procedure writer.

It is therefore concluded that there are no specific safety deficiencies in the "operator actions" specified in the sample plant's FSAR. However, definitions of the events chosen for safety analysis purposes are narrow, and the FSAR analyses characteristically assume minimal operator actions. Further, most of the events are in fact not emergencies; they become so only if accompanied by further equipment failures or operator errors. These facts make the FSAR less than ideally useful as a training guide (a purpose for which it was not intended but for which it has sometimes been used). The Emergency Procedure Guidelines will serve this function.

3.2.2.1 DECREASE IN REACATOR COOLANT TEMPERATURE

3.2.2.1.1 LOSS OF FEEDWATER HEATING

3.2.2.1.1.1 Event Description

3.2.2.1.1.1.1 FSAR Overview

The reactor vessel receives cooler feedwater which causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations. This event has been conservatively estimated to incur a loss of up to 100°F of the feedwater heating capability of the plant. The event can occur with the reactor in either the automatic or manual flow control mode.

In the automatic flow control mode, the recirculation flow control system responds to the power increase by reducing core flow so that steam flow from the reactor vessel to the turbine remains essentially constant. In order to maintain the initial steam flow with the reduced inlet temperature, reactor thermal power increases above the initial value and settles at about 120% NBR (115% of initial power), below the flow-referenced APRM thermal power scram setting, and core flow is reduced to approximately 93% of rated flow. The sequence of events is listed in Table 3.2.2.1.1-1. The smaller power increase makes this event less severe than the manual flow control case given below. Nuclear system pressure does not change and consequently the reactor coolant pressure boundary is not threatened. If scram occurs, the results become very similar to the manual flow control case.

In manual flow control mode, no compensation is provided by core flow and thus the power increase is greater than in the automatic mode. A scram on high APRM thermal power occurs. The sequence of events is listed in Table 3.2.2.1.1-2. Vessel steam flow increases and the initial system pressure increase is slightly larger. Peak heat flux is 116% of its initial value and peak fuel center temperature increases 544°F. The increased core inlet subcooling aids core thermal margins and minimum MCPR reaches 1.06.

3.2.2.1.1.1.2 Critique

1. Mathematical Model

   The FSAR analyses were performed using the REDY code in which the neutron kinetics is represented by a point kinetic model which assumes that the axial power shape of the core remains unchanged throughout the transient. Recent studies have demonstrated that the consideration of axial power shape in the analysis of this event is important. The increase in the core inlet subcooling due to loss of feedwater heating will shift the axial power shape toward more bottom-peaked which acts to mitigate the extent of the decrease in MCPR. Therefore, the transient is expected to be less severe than predicted in the FSAR, under most expected conditions.

Table 3.2.2.1.1-1

SEQUENCE OF EVENTS*

LOSS OF FEEDWATER HEATING (AUTOMATIC CONTROL)

Time-sec	Event
0	Initiate a 100°F temperature reduction in the feedwater system.
4	Initial effect of unheated feedwater starts to raise core power level but flow control system automatically reduces core flow to maintain initial steam flow.
80	Reactor settles out at a higher thermal power level.

*Note: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

Table 3.2.2.1.1-2

SEQUENCE OF EVENTS*

LOSS OF FEEDWATER HEATING (MANUAL CONTROL)

Time-sec	Event
0	Initiate a 100°F temperature reduction into the feedwater system.
4	Initial effect of unheated feedwater starts to raise core power level and steam flow.
44	APRM initiates reactor scram on high thermal power.
50	L8 vessel level set point trips main turbine and feedwater pumps.
50.01	Recirculation pump trip (RPT) is actuated by turbine stop valve position switches.
100 (est)	L2 vessel level set point initiates main steam line isolation, and HPCI and RCIC operation.
>100 (est)	SRVs cycle to release decay heat.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

2. Exposure Condition

The FSAR analyses were performed at the end of equilibrium cycle exposure condition in which the scram and void characteristics would be the worst. In general, for a given cycle, the event will be most severe at the point in the cycle at which the void coefficient is most negative since doppler and void reactivity coefficients have a stronger impact on severity of the transient than do scram characteristics. Therefore, for reload licensing evaluations, the event is analyzed as a function of exposure.

3. Initial Power Level

The FSAR analyses were performed assuming the reactor was initially operating at the 105% steamflow condition. The event could produce a larger thermal power increase if the reactor were initially operating at other power/flow conditions. However, because of limitations on initial operating MCPRs the event initiated from a lower power/flow condition will not produce the minimum MCPR.

4. Loss of Feedwater Heating Capability

The maximum loss of feedwater heating capability of a plant was assumed to be 100°F in the FSAR analyses, which was consistent with the GE specification. Results of startup tests performed on various plants ear full power indicate that this is plant dependent and ranges from 30 to 70°F. For reload licensing evaluations, the utility confirms the validity of this assumption.

5. Feedwater Heater Time Constant

The feedwater heater time constant was assumed to be 30 seconds in the FSAR analyses. The actual time constant is expected to be in the order of a minute, which would produce a slower transient than analyzed.

3.2.2.1.1.2 Operator Actions

3.2.2.1.1.2.1 FSAR Identified Actions

In the automatic flow control mode, the reactor settles out at a lower recirculation flow with no change in steam output. An average power range monitor

(APRM) neutron flux or thermal power alarm will alert the operator that he must insert control rods to get back down to the rated flow control line, or that he must reduce flow if in the manual mode. The operator must determine from existing tables the maximum allowable turbine-generator output with feedwater heaters out of service. If reactor scram occurs, as it would in manual flow control mode, the operator must monitor the reactor water level and pressure controls and the T-G auxiliaries during coastdown.

3.2.2.1.1.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.1.2 FEEDWATER CONTROL FAILURE - MAXIMUM DEMAND

3.2.2.1.2.1 Event Description

3.2.2.1.2.1.1 FSAR Overview

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure with maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

With excess feedwater flow the water level rises to the high-level trip point at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. Table 3.2.2.1.2-1 lists the sequence of events. The MCPR reaches 1.10 and peak fuel center temperature increases 218°F. The peak vessel bottom pressure is 1183 psig and the nuclear system process barrier pressure limit is not endangered. The bypass valves subsequently close to reestablish pressure control in the vessel during shutdown.

Table 3.2.2.1.2-1

SEQUENCE OF EVENTS*

FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND

Time-Sec	Event
0	Initiate simulated failure of 129% upper limit on feedwater flow.
16.65	L8 vessel level set point trips main turbine and feedwater pumps.
16.66 (est)	Reactor scram initiated from main turbine stop valve position switches.
16.66	Recirculation pump trip (RPT) actuated by stop valve position switches.
16.70	Main turbine bypass valves open.
18.30	First group of safety/relief valves open due to high pressure.
24.60	First group of safety/relief valves close.
>50 (est)	Turbine bypass valves closed.
>50 (est)	L2 vessel level set point initiates main steam line isolation and HPCI and RCIC operation.
>50 (est)	SRVs cycle to release decay heat.

*Note: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

3.2.2.1.2.1.2 Critique

1. Mathematical Model

   The FSAR analysis was performed using the REDY code. REDY does not consider the effects of steamline pressure wave transmission and core axial power shape. However, these two effects tend to compensate each other to some degree. Also, the FSAR analysis utilized conservative multipliers on both void reactivity coefficients and scram characteristics. Therefore, the transient is expected to be less severe than predicted in the FSAR, under most expected conditions.

2. Initial Power Level

   The FSAR analysis was performed assuming the reactor was initially operating at the 105% steamflow condition. The event could be more severe if the reactor were initially operating at a lower power level. This was due to the fact that the initial feedwater flow at a lower power level would be smaller. Therefore, with a given feedwater flow runout capacity, the magnitude of feedwater/steam mismatch would be greater if the initial feedwater flow were smaller. However, because of limitations on initial operating MCPRs the event initiated from a lower power/flow condition will not produce the minimum MCPR.

3. Initial Water Level

   In the FSAR analysis the initial water level was assumed to be at the low end of the normal range of water level. This would make the transient results more severe since it allows a longer time, and therefore a larger thermal power increase, before feedwater pumps are tripped on L8.

4. Maximum Feedwater Runout Flow

   In the FSAR analysis the feedwater flow was assumed to run out to its maximum capacity instantly. This gave rise to a larger thermal power increase and, therefore, more severe transient results. The maximum runout capacity assumed, 129% of rated, was also conservative.

3.2.2.1.2.2 Operator Actions

3.2.2.1.2.2.1 FSAR Identified Actions

a. Observe that high-level feedwater pump trip has terminated the failure event.

b. Switch the feedwater controller from auto to manual control in order to try to regain a correct output signal.

c. Identify causes of the failure and report all key plant parameters during the event.

3.2.2.1.2.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.1.3 PRESSURE REGULATOR FAILURE - OPEN

3.2.2.1.3.1 Event Description

3.2.2.1.3.1.1 FSAR Overview

This event assumes that either the controlling pressure regulator or the backup regulator fails to the open position. When this happens, the turbine admission valves can be fully opened and the turbine bypass valves can be partially opened until the maximum steam flow is established. The total steam flow rate to the main turbine resulting from a pressure regulator malfunction is limited by a maximum flow limiter imposed at the turbine controls. This limiter is set to limit maximum steam flow to approximately 110% NBR.

The increase of steam flow to a maximum value causes rapid vessel depressurization and water level swell. The water level rises to the high level trip set point and initiates trip of the main turbine and feedwater pumps. Closure of the turbine stop valves initiates scram and recirculation pump trip (RPT). Since the pressure regulator failure will now signal the bypass to open to 115% steam flow, the turbine inlet pressure will drop below the low pressure isolation set point and cause an isolation of the main steam lines. The sequence of events is listed in Table 3.2.2.1.3-1. Because the rapid portion of the transient results in only momentary depressurization of the nuclear system, the nuclear system process barrier is not threatened by high internal pressure.

3.2.2.1.3.1.2 Critique

The major uncertainty in the FSAR analysis was in the maximum flow limiter setting assumed (115% of rated). If the maximum flow limiter were set higher or lower than normal, there would result a faster or slower vessel depressurization rate. The vessel depressurization rate and the initial water level would determine whether the L8 vessel level set point or the low turbine inlet pressure set point would be reached first. Therefore, main steamline isolation could precede L8 turbine trip.

3.2.2.1.3.2 Operator Actions

3.2.2.1.3.2.1 FSAR Identified Actions

When regulator trouble is preceded by spurious or erratic behavior of the controlling device, it may be possible for the operator to transfer operation to the backup controller in time to prevent the full transient. If the reactor scrams as a result of the isolation caused by the low pressure at the turbine inlet (825 psig) in the run mode, the following is the sequence of operator actions expected during the course of the event. Once isolation occurs the pressure will increase to a point where the relief valves open. The operator should:

a. Monitor that all rods are in.

b. Monitor reactor water level and pressure.

c. Observe turbine coastdown and break vacuum before the loss of steam seals. Check turbine auxiliaries.

d. Observe that the reactor pressure relief valves open at their set point.

Table 3.2.2.1.3-1

SEQUENCE OF EVENTS*

PRESSURE REGULATOR FAILURE - OPEN

Time-sec	Event
0	Maximum limit on steam flow to main turbine initiated.
0.2	Main turbine bypass opens.
19.97	L8 vessel level set point trips main turbine and feedwater pumps.
17.98 (est)	Reactor scram initiated from main turbine stop valve position switches.
17.98	Recirculation pump trip (RPT) actuated by stop valve position switches.
>50 (est)	Main steam line isolation valves closed on low turbine inlet pressure.
>50 (est)	SRVs cycle to release decay heat.
50 (est)	L2 vessel level set point initiates RCIC and HPCI operation.

*Note: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

e. Observe that RCIC and HPCI initiate on low-water level.

f. Secure both HPCI and RCIC when reactor pressure and level are under control.

g. Monitor reactor water level and continue cooldown per the normal procedure.

h. Complete the scram report and initiate a maintenance survey of pressure regulator before reactor restart.

3.2.2.1.3.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.2 INCREASE IN REACTOR PRESSURE

3.2.2.2.1 PRESSURE REGULATOR FAILURE - CLOSED

3.2.2.2.1.1 Event Description

3.2.2.2.1.1.1 FSAR Overview

It is assumed for purposes of this transient that a single failure occurs which erroneously causes the controlling regulator to close the main turbine control valves and thereby increases reactor pressure. If this occurs, the backup regulator is ready to take control. Postulating a failure of the primary or controlling pressure regulator in the closed mode will cause the turbine control valves to close momentarily. The pressure will increase, because the reactor is still generating the initial steam flow. The backup regulator will reopen the turbine control valves and reestablish steady-state operation above the initial pressure equal to the difference of 5 psi in the set point of the primary and backup regulators. The pressure disturbance in the vessel is not expected to exceed flux or pressure scram trip set points.

3.2.2.2.1.1.2 Critique

The FSAR event description is appropriate.

3.2.2.2.1.2 Operator Actions

3.2.2.2.1.2.1 FSAR Identified Actions

The operator will verify that the backup regulator assumes proper control.

3.2.2.2.1.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.2.2 GENERATOR LOAD REJECTION

3.2.2.2.2.1 Event Description

3.2.2.2.2.1.1 FSAR Overview

Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The turbine control valves and the combined intermediate valves are required to close as rapidly as possible to prevent overspeed of the turbine-generator.

1. Generator Load Rejection
   A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 3.2.2.2.2-1. Peak neutron flux rises to 163% NBR. The average surface heat flux peaks at 102% of its initial value and MCPR reaches 1.13.

2. Generator Load Rejection with Failure of Bypass
   A loss of generator electrical load at high power with these conditions produces the sequence of events listed in Table 3.2.2.2.2-2. Peak neutron flux reaches about 257% of rated, average surface heat flux reaches 107% of its initial value. Since this event is classified

Table 3.2.2.2-1

SEQUENCE OF EVENTS*

GENERATOR LOAD REJECTION

Time-sec	Event
(-)0.015	Turbine-generator detection of loss of electrical load.
0	Turbine-generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Turbine-generator PLU trip initiates main turbine bypass system operation.
0.01	Fast control valve closure (FCV) initiates scram trip and recirculation pump trip.
0.1	Turbine bypass valves start to open.
0.15	Turbine control valves closed.
1.8	Group 1 relief valves actuated.
1.9	Group 2 relief valves actuated.
2.1	Group 3 relief valves actuated.
4.3	L8 vessel level set point initiates feedwater pump trips.
7.9	All relief valves close.
47.0 (est)	L2 vessel level set point initiates main steamline isolation, and HPCI and RCIC operation.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence only as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

Table 3.2.2.2-1 (Continued)

Time-sec	Event
>50 (est)	SRVs cycle to release decay heat.

Table 3.2.2.2.2-2

SEQUENCE OF EVENTS*

GENERATOR LOAD REJECTION WITH BYPASS FAILURE

Time-sec	Event
(-)0.015 (approx)	Turbine-generator detection of loss of electrical load.
0	Turbine-generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0.01	Fast control valve closure (FCV) initiates scram trip and recirculation pump trip.
0.15	Turbine control valves closed.
1.3	Group 1 relief valves actuated.
1.4	Group 2 relief valves actuated.
1.5	Group 3 relief valves actuated.
6.7	L8 vessel level set point initiates feedwater pump trips.
14.5	All relief valves close.
20	SRVs actuate and then close.
32.8	L2 vessel level set point initiates main steamline isolation and HPCI and RCIC operations.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence only as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

Table 3.2.2.2.2-2 (Continued)

Time-sec	Event
>35	SRVs cycle to release decay heat.

an an infrequent incident, the MCPR limit is permitted to fall below the safety limit for incidents of moderate frequency. MCPR reaches 1.03 for this event. The peak nuclear system pressure reaches 1209 psig at the bottom of the vessel, well below the transient pressure limit of 1375 psig.

3.2.2.2.2.1.2 Critique

The FSAR analyses were performed using the REDY code. REDY does not consider the effects of steamline pressure wave transmission and core axial power shape. However, these two effects tend to compensate each other to some degree. Also, the FSAR analysis untilizes conservative multipliers on both void reactivity coefficients and scram characteristics. Therefore, the transient is expected to be less severe than predicted in the FSAR, under most expected conditions.

3.2.2.2.2.2 Operator Actions

3.2.2.2.2.2.1 FSAR Identified Actions

a. Verify proper bypass valve performance.

b. Observe that the feedwater/level controls have maintained the reactor water level at a satisfactory value.

c. Observe that the pressure regulator is controlling reactor pressure at the desired value.

d. Record peak power and pressure.

e. Verify relief valve operation.

3.2.2.2.2.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.2.3 TURBINE TRIP

3.2.2.2.3.1 Event Description

3.2.2.2.3.1.1. FSAR Overview

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are moisture separator and heater drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum, and reactor high water level.

1. Turbine Trip
   Turbine trip at high power produces the sequence of events listed in Table 3.2.2.2.3-1.

Table 3.2.2.2.3-1

SEQUENCE OF EVENTS*

TURBINE TRIP

Time-sec	Events
0	Turbine trip initiates c osure of main stop valves.
0	Turbine trip initiates bypass operation.
0.1	Main turbine stop valves reach 90% open position and initiate reactor scram trip and recirculation pump trip.
0.1	Turbine stop valves closed.
0.1	Turbine bypass valves start to open to regulate pressure.
1.8	Group 1 relief valves actuated.
2.0	Group 2 relief valves actuated.
2.2	Group 3 relief valves actuated.
4.3	L8 vessel level set point initiates feedwater pump trips.
7.7	All relief valves close.
47.5 (est)	L2 vessel level set point initiates main steamline isolation, and HPCI and RCIC operation.
>50 (est)	SRVs cycle to release decay heat.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 144% of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed 100% of its initial value. Peak pressure in the bottom of the vessel reaches 1182 psig, which is below the ASME code limit of 1375 psig for the reactor coolant pressure boundary.

2. <u>Turbine Trip with Failure of Bypass</u>

Turbine trip at high power with bypass failure produces the sequence of events listed in Table 3.2.2.2.3-2.

Peak neutron flux reaches 231% of its rated value, and peak fuel center temperature increases approximately 175°F. MCPR for the transient is 1.05. Peak nuclear system pressure reaches 1209 psig at the vessel bottom. This event is classified as an infrequent incident.

3. <u>Turbine Trip with Bypass Valve Failure, Low Power</u>

This transient is less severe than a similar one at high power. Below 30% of rated power, the turine stop valve closure and turbine control valve closure scrams are automatically bypassed. At these lower power levels, turbine first stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief set points are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve set points and will be significantly below the RCPB transient limit of 1375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR is expected to remain well above the safety limit.

Table 3.2.2.2.3-2

SEQUENCE OF EVENTS*

TURBINE TRIP WITH BYPASS FAILURE

Time-sec	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip and recirculation pump trip.
0.1	Turbine stop valves closed.
1.4	Group 1 relief valves actuated.
1.5	Group 2 relief valves actuated.
1.6	Group 3 relief valves actuated.
6.6	L8 vessel level set point initiates feedwater pump trips.
19.3	All relief valves actuated again.
20	SRVs actuate and then close.
33.0	L2 vessel level set point initiates main steamline isolation, and HPCI and RCIC operations.
>35	SRVs cycle to release decay heat.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

3.2.2.2.3.1.2 Critique

The FSAR analyses were performed using the REDY code. REDY does not consider the effects of steamline pressure wave transmission and core axial pwer shape. However, these two effects tend to compensate each other to some degree. Also, the FSAR analysis utilizes conservative multipliers on both void reactivity coefficients and scram characteristics. Therefore, the transient is expected to be less severe than predicted in the FSAR, under most expected conditions.

3.2.2.2.3.2 Operator Actions

3.2.2.2.3.2.1 FSAR Identified Action

The operator must:

a. Verify auto transfer of buses supplied by generator to incoming power - if automatic transfer does not occur, manual transfer must be made.

b. Monitor and maintain reactor water level at required level.

c. Check turbine for proper operation of all auxiliaries during coastdown.

d. Depending on conditions, initiate normal operating procedures for cool-down, or maintain pressure for restart purposes.

e. Put the mode switch in the startup position before the reactor pressure decays to <850 psig.

f. Secure the RCIC operation if auto initiation occurred due to low water level.

g. Monitor control rod drive positions and insert both the IRMs and SRMs.

h. Investigate the cause of the trip, make repairs as necessary, and complete the scram report.

i. Cool down the reactor per standard procedure if a restart is not intended.

3.2.2.2.3.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.2.4 MSIV CLOSURE

3.2.2.2.4.1 Event Description

3.2.2.2.4.1.1 FSAR Overview

Various steam line and nuclear system malfunctions, or operator actions, can initiate main steam isolation valve (MSIV) closure. Examples are low steam line pressure, high steam line flow, high steam line radiation, low water level or manual action.

1. Closure of All Main Steam Line Isolation Valves
   The sequence of events is listed in Table 3.2.2.2.4-1. Position switches on the valves are less than 90% open. Peak neutron flux reaches 175% of rated after approximately 2.4 seconds. The nuclear system relief valves begin to open at approximately 2.5 seconds after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 1210 psig, well below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steam line is 1173 psig.

2. Closure of One Main Steam Line Isolation Valve
   One MSIV may be manually closed at a time for testing purposes. Operator error or equipment malfunction may cause a single MSIV to be closed inadvertently. If reactor power is greater than about 80% when this occurs, a high APRM flux scram or high steam line flow isolation may result.

No significant effect is imposed on the RCPB, since if closure of the valve occurs at an unacceptably high operating power level, a flux or pressure scram will result. The main turbine bypass system will continue to regulate system pressure via the other three "live" steam lines.

Table 3.2.2.2.4-1

SEQUENCE OF EVENTS*

CLOSURE OF ALL MAIN STEAM LINE ISOLATION VALVES

Time-sec	Event
0	Initiate closure of all main steam line isolation valves (MSIV).
0.3	MSIVs reach 90% position.
0.3	MSIV position trip scram initiated.
2.62	High pressure set point reached.
2.7	Relief valves open (3 groups) due to pressure relief set point action.
2.92	High pressure recirculation pump trip initiated.
15.9	All relief valves close.
20	SRVs actuate and then close.
28.5	L2 vessel level set point initiate HPCI and RCIC systems operation.
>30	SRVs cycle to release decay heat.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

3.2.2.2.4.2 Critique

1. Mathematical Model

The FSAR analyses were performed using the REDY code. REDY does not consider the effects of steamline pressure wave transmission and core axial power shape. However, these two effects tend to compensate each other to some degree. Also, the FSAR analysis utilizes conservative multipliers on both void reactivity coefficients and scram characteristics. Therefore, the transient is expected to be less severe than predicted in the FSAR, under most expected conditions.

2. Feedwater Flow

In the FSAR analysis it was conservatively assumed that feedwater flow was terminated within 5 seconds due to MSIV closure. Results of startup tests at several plants with turbine-driven feed pumps have demonstrated that feedwater flow actually remained near rated for a least 20 seconds, and was sufficient to bring the vessel water level back to normal before HPCI/RCIC flow was injected into the vessel.

3.2.2.2.4.2 Operator Actions

3.2.2.2.4.2.1 FSAR Identified Actions

The following is the sequence of operator actions expected during the course of the event assuming no restart of the reactor. The operator should:

a. Observe that all rods have inserted.

b. Observe that the relief valves have opened for reactor pressure control.

c. Check that RCIC auto starts on the impending low reactor water level condition.

d. Switch the feedwater controller to the manual position.

e. Initiate operation of the RHR system in the steam condensing mode only.

f. When the reactor vessel level has recovered to a satisfactory level, switch RCIC controller to manual and secure when level is under control.

g. When the reactor has cooled sufficiently for RHR operation, put it into service per procedure.

h. Before resetting the MSIV isolation, determine the cause of valve closure.

i. Observe turbine coastdown and break vacuum before the loss of sealing steam. Check T-G auxiliaries for proper operation.

j. Not reset and open MSIVs unless conditions warrant and be sure the pressure regulator set point is above vessel pressure.

k. Survey maintenance requirements and complete the scram report.

3.2.2.2.4.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.2.5 LOSS OF CONDENSER VACUUM

3.2.2.2.5.1 Event Description

3.2.2.2.5.1.1 FSAR Overview

Various system malfunctions which can cause a loss of condenser vacuum are designated in Table 3.2.2.2.5-1. Trip signals associated with loss of condenser vacuum are designated in Table 3.2.2.2.5-2.

The event presented here is a hypothetical case with a conservative 2 inches Hg per second vacuum decay rate. Under this hypothetical 2 inches Hg per second vacuum decay condition, the turbine bypass valve and main steam line isolation valve closure would follow main turbine and feedwater turbine trips by about 5 seconds after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of main steam line isolation valve closure tends to be minimal since the closure of

Table 3.2.2.2.5-1
TYPICAL RATES OF DECAY FOR CONDENSER VACUUM

Cause	Estimated Vacuum Decay Rate
a. Failure or Isolation of Steam Jet Air Ejectors	<1 inch Hg/minute
b. Loss of Sealing Steam to Shaft Gland Seals	~1 to 2 inches Hg/mintue
c. Opening of Vacuum Breaker Valves	~2 to 12 inches Hg/minute

Table 3.2.2.2.5-2
TRIP SIGNALS ASSOCIATED WITH LOSS OF CONDENSER VACUUM

Vacuum (inches of Hg)	Protective Action Initiated
27 to 28	Normal Vacuum Range
20 to 23	Main Turbine Trip and Feedwater Turbine Trip (Stop Valve Closure)
7 to 10	Main Steam Line Isolation Valve (MSIV) Closure an Bypass Valve Closure

main turbine stop valves and subsequently the bypass valves have already shut off the main steam line flow. Table 3.2.2.2.5-3 lists the sequence of events. Peak neutron flux reaches 112% of NB rated power while average fuel surface heat flux remains at the initial value. Safety/relief valves open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated. Peak nuclear system pressure is 1174 psig at the vessel bottom. The over-pressure transient is below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Vessel dome pressure does not exceed 1155 psig. A comparison of these values to those for the turbine trip with bypass failure at higher power shows the similarities between these two transients. The prime differences are the loss of feedwater and main steam line isolation.

Table 3.2.2.2.5-3

SEQUENCE OF EVENTS*

LOSS OF CONDENSER VACUUM

Time-sec	Event
-0.0 (est)	Initiate simulated loss of condenser vacuum at 2 inches Hg per second.
0 (est)	Low condenser vacuum main turbine trip actuated.
0 (est)	Low condenser vacuum feedwater trip actuated.
0.01 (est)	Main turbine trip initiates reactor scram and recirculation pump trip.
1.9	Group 1 relief valves set points actuated.
2.1	Group 2 relief valves set points actuated.
2.2	Group 3 relief valves set points actuated.
5.0	Low condenser vacuum initiates main steam line isolation valve closure.
5.0	Low condenser vacuum initiates bypass valve closure.
13.6	All relief valves close.
17	SRVs actuate and then close.
26.0 (est)	HPCI/RCIC system initiation on low level (L2).
>26	SRVs cycle to release decay heat.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

3.2.2.2.5.1.2 Critique

Because the protective actions are actuated at various levels of condenser vacuum, the severity of the transient is directly dependent upon the rate at which vacuum is lost. Normal loss of vacuum due to loss of cooling water pumps or steam jet air ejector problems produces a very slow rate of loss of vacuum (minutes, not seconds). See Table 3.2.2.2.5-1. If corrective actions by the reactor operators are not successful, the simultaneous trips of the main and feedwater turbines, and ultimately complete isolation by closing the bypass valves (opened with the main turbine trip) and the MSIVs, will occur. A faster rate of loss of the condenser vacuum would reduce the anticipatory action of the scram and the overall effectiveness of the bypass valves since they would be closed more quickly.

3.2.2.2.5.2 Operator Actions

3.2.2.2.5.2.1 FSAR Identified Actions

The operator must:

a. Verify auto transfer of buses supplied by generator to incoming power - if automatic transfer does not occur, manual transfer must be made.

b. Monitor and maintain reactor water level at required level.

c. Check turbine for proper operation of all auxiliaries during coastdown.

d. Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes.

e. Put the mode switch in the startup position before the reactor pressure decays to <850 psig.

f. Secure the RCIC operation if auto initiation occurred due to low water level.

g. Monitor control rod drive positions and insert both the IRMs and SRMs.

h. Investigate the cause of the trip, make repairs as necessary, and complete the scram re . c.

i. Cool down the reactor per standard procedure if a restart is not intended.

3.2.2.2.5.2.2 <u>Critique</u>

The FASR-specified operator actions are not inappropriate for the event.

3.2.2.2.6 <u>LOSS OF AC POWER</u>

3.2.2.2.6.1 <u>Event Description</u>

3.2.2.2.6.1.1 <u>FSAR Overview</u>

1. <u>Loss of Auxiliary Power Transformer</u>
   Causes for interruption of the auxiliary power transformer can arise from normal operation or malfunctioning of transformer protection circuitry. These can include high transformer oil temperature, reverse or high current operation as well as operator error which trips the transformer breakers.

   The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. Estimates of the responses of the various reactor systems (assuming loss of the auxiliary transformer) provide the following simulation sequence:

   a. The recirculation pumps are tripped at a reference time, t=0, with normal coastdown times.

   b. At approximately 2 seconds, independent main steam line isolation valve closure and scram are initiated due to loss of power to the respective solenoids.

c. At approximately 4 seconds, feedwater pump trips are initiated. The sequence of events is listed in Table 3.2.2.2.6-1.

The initial portion of the transient is similar to the loss of recirculation pumps. At approximately 2.0 seconds, turbine trip and scram occur. There is no significant increase in fuel temperature or decrease in MCPR, fuel thermal margins are not threatened, and the design basis is satisfied.

Table 3.2.2.2.6-1

SEQUENCE OF EVENTS*

LOSS OF AUXILIARY POWER TRANSFORMER

Time-sec	Event
0	Loss of auxiliary power transformer occurs.
0	Recirculation system pump motors are tripped.
0	Booster pumps are tripped.
0	Condenser circulating water pumps tripped.
2.0	Reactor scram initiated.
2.0	Closure of main steam line isolation valves.
4.0	Feedwater pumps are tripped.
5.0	All SRVs actuated.
12.0	All SRVs close.
>17.0	SRVs cycle to release decay heat.
43 (est)	L2 vessel level set point initiates HPCI and RCIC operation.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapterl 5. Other event sequences can occur, depending on the assumptions made.

2. Loss of All Grid Connections

Loss of all grid connections can result from major shifts in electrical loads, loss of load, lightning, storms, wind, etc., which contribute to electrical grid instabilities. These instabilities will cause equipment damage if unchecked. Protective relay schemes automatically disconnect electrical sources and loads to mitigate damage and regain electrical grid stability.

The sequence of events is the same as that for loss of auxiliary power transformer except that this event would add a generator load rejection to the above sequence at time t=0. The load rejection immediately forces the turbine control valves closed, causing a scram. The sequence of events is listed in Table 3.2.2.2.6-2.

3.2.2.2.6.1.2 Critique

The FSAR event description is appropriate.

3.2.2.2.6.2 Operator Actions

3.2.2.2.6.2.1 FSAR Identified Actions

The operator should maintain the reactor water level by use of the RCIC system, control reactor pressure by use of the relief valves and steam condensing mode of the RHR. Verify that the turbine d-c oil pump is operating satisfactorily to prevent turbine bearing damage. Also, he should verify proper switching and loading of the emergency diesel generators.

The following is the sequence of operator actions expected during the course of the events when no immediate restart is assumed. The operator should:

a. Following the scram, verify all rods in.

b. Check that diesel generators start and carry the vital loads.

Table 3.2.2.2.6-2

SEQUENCE OF EVENTS*

LOSS OF ALL GRID CONNECTIONS

Time-sec	Event
(-)0.015	Loss of Grid causes turbine-generator to detect a loss of electrical load.
0	Generator load rejection is initiated (control valve fast closure).
0	Turbine-generator PLU trip initiates main turbine bypass system operation.
0	Recirculation system pump motors are tripped.
0	Fast control valve closure (FCV) initiates a reactor scram trip.
0.1	Turbine bypass valves open.
0.15	Turbine control valves closed.
1.8	Group 1 safety relief valves actuated.
2.0	Closure of main steam line isolation valves.
2.0	Group 2 safety relief valves actuated.
2.2	Group 3 safety relief valves actuated.
4.0	Feedwater pump trips initiated.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

Table 3.2.2.2.6-2 (Continued)

Time-sec	Event
14.1	All safety relief valves close.
>17.0	SRVs cycle to release decay heat.
41.9 (est)	L2 vessel level set point initiates.

c. Check that relays on the reactor protection system (RPS) drop out.

d. Check that both RCIC and HPCI start when reactor vessel level drops to the initiation point after the relief opens.

e. Break vacuum before the loss of sealing steam occurs.

f. Check T-G auxiliaries during coastdown.

g. When both the reactor pressure and level are under control, secure both HPCI and RCIC as necessary.

h. Continue cooldown per the normal procedure.

i. Complete the scram report and survey the maintenance requirements.

3.2.2.2.6.2.2 Critique

The FASR-specified operator actions are not inappropriate for the event.

3.2.2.2.7 LOSS OF ALL FEEDWATER FLOW

This transient is completely treated in the Loss of Feedwater Analyses and the Emergency Procedure Guidelines.

3.2.2.2.8 FEEDWATER LINE BREAK

Refer to Section 3.2.2.6.4.

3.2.2.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

3.2.2.3.1 Recirculation Pump Trip

3.2.2.3.1.1 Event Description

3.2.2.3.1.1.1 FSAR Overview

Recirculation pump motor operation can be tripped off by design for intended objectives as well as randomly by unpredictable operational failures. Intentional tripping will occur in response to:

a. Reactor vessel water level L2 set point trip.

b. High pressure set point trip.

c. Motor branch circuit over-current protection.

d. Motor overload protection.

e. Suction block valve not fully open.

f. Main turbine trip.

Random tripping will occur in response to:

a. Operator error.

b. Loss of electrical power source to the pumps.

c. Equipment or sensor failures and malfunctions which initiate the above intended trip response.

1. Trip of One Recirculation Pump

The vessel water level swell due to rapid flow coastdown may reach the high level trip thereby shutting down the main turbine and the feed pump turbines, and indirectly initiating scrams as a result of the main turbine trip. Recirculation pump trip action is initiated by tripping of the main turbine. The sequence o. events is listed in Table 3.2.2.3.1-1.

2. Trip of Two Recirculation Pumps

The vessel water level swell due to rapid flow coastdown may reach the high level trip thereby shutting down the main turbine and the feed pump turbines, and indirectly initiating scrams as a result of the main turbine trip. The sequence of events is listed in Table 3.2.2.3.1-2.

3.2.2.3.1.1.2 Critique

The major uncertainity in the FSAR analysis lies in the prediction of L8 vessel water level trip. Should the water level swell due to a rapid flow coastdown be less than predicted, the reactor could settle into a new steady state without L8 trip. If this happened with both recirculation pumps tripped, the reactor would be operating at the intersection of the natural circulation and rated rod lines, on the power flow map, which are conditions of minimum core flow and core power stability.

3.2.2.3.1.2 Operator Actions

3.2.2.3.1.2.1 FSAR Identified Actions

As soon as possible, the operator must verify that no operating limits are being exceeded. If they are, corrective actions must be initiated. Also, the operator must determine the cause of the trip prior to returning the system to normal.

3.2.2.3.1.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

Table 3.2.2.3.1-1

SEQUENCE OF EVENTS*

TRIP OF ONE RECIRCULATION PUMP

Time-sec	Event
0	Trip of one recirculation pump initiated.
5.0	Vessel water level (L8) trip initiates turbine trip.
5.0	Feedwater pumps are tripped off.
5.0	Turbine trip initiates bypass operation.
5.0	Turbine trip initiates reactor scram trip.
5.2	Turbine trip initiates trip of the second recirculation pump.
9.3	Group 1 pressure relief valves open.
12.6	Group 1 pressure relief valves close.
48.0 (est)	L2 vessel level set point initiates main steam line isolation and RCIC and HPCI operation.
>50 (est)	SRVs cycle to release decay heat.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

Table 3.2.2.3.1-2

SEQUENCE OF EVENTS*

TRIP OF TWO RECIRCULATION PUMPS

Time-sec	Event
0	Trip of both recirculation pumps initiated.
3.4	Vessel water level (L8) trip initiates turbine trip.
3.4	Feedwater pumps are tripped off.
3.4	Turbine trip initiates bypass operation.
3.4	Turbine trip initiates reactor scram trip.
6.7	Group 1 pressure relief valves open.
10.1	Group 1 pressure relief valves open.
46.6 (est)	L2 vessel level set point initiates main steam line isolation and RCIC and HPCI operation.
>50 (est)	SRVs cycle to release decay heat.

*NOTE: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

3.2.2.3.2 RECIRCULATION FLOW CONTROL FAILURE - DECREASING FLOW

3.2.2.3.2.1 Event Description

3.2.2.3.2.1.1 FSAR Overview

Recirculation flow controller malfunctions can be due to:

a. Failure of an individual recirculation M/G set speed controller (one per loop) or the positioning control of an individual scoop tube actuator which can result in a rapid flow decrease in only one recirculation loop.

b. A downscale failure of the master flow controller or an excessive manual speed demand set point change which can generate a zero flow demand signal to both recirculation flow control loops.

1. Individual Speed Controller Failure - Closed

The sequence of events for this transient is similar to, and can never be more than, that listed in Table 3.2.2.3.1-1 for the trip of one recirculation pump.

2. Master Controller Failure - Closed

The sequence of events for this transient is similar to and can never be more severe than that listed in Table 3.2.2.3.1-2 for the trip of both recirculation pumps.

3.2.2.3.1.2.2 Critique

Same as Section 3.2.2.3.1.2.2.

3.2.2.3.2.2 Operator Actions

Same as Section 3.2.2.3.1.2.

3.2.2.3.3 RECIRCULATION PUMP SEIZURE

3.2.2.3.3.1 Event Description

3.2.2.3.3.1.1 FSAR Overview

The case of recirculation pump seizure represents the extremely unlikely event of instantaneous stoppage of the pump motor shaft of one recirculation pump. This produces a very rapid decrease of core flow as a result of the large hydraulic resistance introduced by the stopped rotor.

The level swell produces a trip of the main and feedwater turbines, and stop valve closure scram. The sequence of events is listed in Table 3.2.2.3.3.-1. Since, after the time at which MCPR occurs, heat flux decreases much more rapidly than the rate at which heat is removed by the coolant, the scram conditions impose no threat to thermal limits. Additionally, the bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

3.2.2.3.3.1.2 Critique

Same as Section 3.2.2.3.1.1.2.

3.2.2.3.3.2 Operator Actions

Same as Section 3.2.2.3.1.2.

Table 3.2.2.3.3-1

SEQUENCE OF EVENTS*

RECIRCULATION PUMP SEIZURE

Time-sec	Event
0	Single pump seizure.
0.77	Jet pump diffuser flow reverses in seized loop.
2.5	Vessel level (L8) trip initiates turbine trip.
2.5	Feedwater pumps are tripped off.
2.5	Turbine trip initiates bypass operation.
2.5	Turbine trip initiates reactor scram trip.
2.7	Turbine trip initiates trip of the second recirculation pump.
5.5	Group 1 pressure relief valves open.
8.7	Group 1 pressure relief valves close.
45.2 (est)	Initiation of L2 vessel level set point isolation of main steam line and HPCI and RCIC operation.
>50 (est)	SRVs cycle to release decay heat.

*Note: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

3.2.2.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

3.2.2.4.1 ROD WITHDRAWAL ERROR - LOW POWER

3.2.2.4.1.1 Control Rod Removal Error During Refueling

3.2.2.4.1.1.1 Event Description

3.2.2.4.1.1.1.1 FSAR Analysis Overview

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod during refueling. The probability of the initial causes alone is considered low enough to warrant its being categorized as an infrequent incident, since there is no postulated set of circumstances which results in an inadvertent Rod Withdrawal Error (RWE) while in the REFUEL mode based on single equipment failure or single operator error under the assumption that refueling interlocks are not bypassed.

A. Initial Control Rod Removal or Withdrawal

During refueling operations safety system interlocks provide assurance that inadvertent criticality does not occur because a control rod has been removed or is withdrawn in coincidence with another control rod.

B. Fuel Insertion With Control Rod Withdrawn

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods be fully inserted when fuel is being loaded into the core. This requirement is backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "REFUEL" position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

C. Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the "REFUEL" position, only one control rod can be withdrawn. Any attempt to withdraw a second rod results in a rod block by the refueling interlocks. Since the core is designed to meet shutdown requirements with the highest worth rod withdrawn, the core remains subcritical even with one rod withdrawn.

D. Control Rod Removal Without Fuel Removal

Finally, the design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control rod without the simultaneous or prior removal of the four adjacent fuel bundles. This precludes any hazardous condition.

3.2.2.4.1.1.1.2 **FSAR Analysis Critique**

As stated, this event is precluded by plant design; therefore, consideration of uncertainties is inappropriate for this event. However, it is assumed that refueling interlocks are not bypassed for the FSAR analysis.

3.2.2.4.1.1.2 **Operator Actions**

3.2.2.4.1.1.2.1 **FSAR Identified Operator Actions**

No operator actions are required to preclude this event since plant design prevents its occurrence.

3.2.2.4.1.1.2.2 **Critique of FSAR Identified Operator Actions**

Not applicable.

3.2.2.4.1.2 Continuous Rod Withdrawal During Reactor Startup

3.2.2.4.1.2.1 Event Description

3.2.2.4.1.2.1.1 FSAR Analysis Overview

The probability of initiation of this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further development of this event makes the total event probability even lower because it is contingent upon the simultaneous failure of two redundant systems, the Rod Sequence Control System (RSCS) and the Rod Worth Minimizer (RWM) systems, concurrent with a high worth rod, out-of-sequence rod selection contrary to procedures, plus operator nonacknowledgement of continuous alarm annunciations prior to safety system actuation. A key assumption to the above argument is that the control rod drive system is operational.

For the typical plant analyzed in this study, which has Bank Position RSCS, control rod withdrawal errors are not considered credible in the startup and low power ranges. The RSCS and RWM prevent the operator from selecting and withdrawing an out-of-sequence control rod.

Continuous control rod withdrawal errors during reactor startup are precluded by the RSCS. The RSCS prevents withdrawal of an out-of-sequence control rod in the 100% to 75% control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75% rod density to the preset power level. Since only in-sequence control rods can be withdrawn in the 100% to 75% control rod density and control rods are withdrawn in the banked position mode from the 75% control rod density point to the preset power level, there is no basis for the continuous control rod withdrawal error in the startup and low power range. The low power range is defined as zero power to the RSCS low power set point, i.e., 20% of rated core power. For RWE above the low power set point see section 3.2.2.4.2.

3.2.2.4.1.2.1.2 FSAR Analysis Critique

The FSAR text is appropriate. However, even if the RWM and RSCS fail to block the continuous withdrawal of an out-of-sequence rod, the licensing basis criteria with respect to fuel failure are still satisfied.

3.2.2.4.1.2.2 Operator Actions

3.2.2.4.1.2.2.1 FSAR Identified Operator Actions

No operator actions are required to preclude this event since the plant design prevents its occurrence.

3.2.2.4.1.2.2.2 Critique of FSAR Identified Operator Actions

Not applicable.

3.2.2.4.2 ROD WITHDRAWAL ERROR - AT POWER

3.2.2.4.2.1 Event Description

3.2.2.4.2.1.1 FSAR Analysis Overview

While operating in the power range in a normal mode of operation the reactor operator makes a procedural error and continuously withdraws the maximum worth control rod until the Rod Block Monitor (RBM) System inhibits further withdrawal. A key assumption is that the control rod drive system is operational.

The sequence of events for this transient, as calculated with conservative assumptions, is presented in Table 3.2.2.4.2-1. No operator actions are required during this event. However, operator procedural actions expected to occur are shown.

The focal point of this event is localized to a small portion of the core. Therefore, although reactor control and instrumentation is assumed to function normally, credit is taken only for the RBM system. A discussion of the event follows below.

Table 2.2.4.2-1

SEQUENCE OF EVENTS*

RWE IN POWER RANGE

Elapsed Time	
0	Core is assumed to be operating at rated conditions.
0	Operator selects and withdraws the maximum worth control rod.
1 sec	The total core power and the local power in the vicinity of the control rod increase.
5 sec	The LPRM system indicates excessive localized peaking.
5 sec	The operator ignores warning and continues withdrawal.
15 sec	The RBM system indicates excessive localized peaking.
15 sec	The operator ignores warning and continues withdrawal.
20 sec	The RBM system initiates a rod block inhibiting further withdrawal.
40 sec	Reactor core stabilizes at higher core power level.
60 sec	Operator reinserts control rod to reduce core power level.
80 sec	Core stabilizes at rated conditions.

*NOTE: This sequence of event represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumption made.

While operating in the power range in a normal mode of operation the reactor operator makes a procedural error and withdraws the maximum worth control rod until the RBM system inhibits further withdrawal.

Under most normal operating conditions no operator action is required since the transient which would occur would be very mild. Should the peak linear power design limits be exceeded, the nearest Local Power Range Monitor (LPRM) would detect this and sound an alarm. The operator should acknowldege this alarm and take appropriate action to rectify the situation.

If the rod withdrawal error is severe enough, the Rod Block Monitor (RBM) system would sound alarms, at which time the operator would acknowledge the alarm and take corrective action. Even for extremely severe conditions (i.e., for highly abnormal control rod patterns, operating conditions, and assuming that the operator ignores all alarms and warnings and continues to withdraw the control rod), the RBM system will lock further withdrawal of the control rod before the fuel reaches the point of boiling transition or the 1% plastic strain limit imposed on the clad.

3.2.2.4.2.1.2 FSAR Analysis Overview Critique

1. Mathematical Model

For this transient the reactivity insertion rate is very slow; therefore, it is adequate to assume that the core has sufficient time to equilibriate (i.e., that both the neutron flux and the heat flux are in phase). The transient is calculated using a steady-state, three-dimensional, coupled nuclear-thermal-hydraulics computer program.

The primary output from this code, in addition to the basic nuclear parameters, is: The variation of the Linear Heat Generation Rate (LHGR); the variation of the Minimum Critical Power Ratio (MCPR); the total reactor power; and the variation of the incore instruments during the transient. A detector response code uses the instrument responses to predict the Rod Block Monitor action under the specified condition for the Rod Withdrawal Error.

2. Input Parameters and Initial Conditions

The number of possible RWE transients is extremely large due to the number of control rods and the wide range or exposures and power levels. In order to encompass all of the possible RWEs which could conceivably occur, a limiting analysis is defined such that conservative assessment of the consequences is provided.

The assumptions are:

a. It is assumed that the operator has fully inserted the maximum worth rod prior to its removal and selected the remaining control rod pattern in such a way as to approach thermal limits in the fuel bundles in the vicinity of the rod to be withdrawn. It should be emphasized that this control rod configuration would be highly abnormal and could only be achieved by deliberate operation action or by numerous operator errors.

b. The operator is assumed to ignore all warnings during the transient.

c. Of the four LPRM strings nearest to the control rod being withdrawn, the two highest reading LPRMs during the transient are assumed to have failed.

d. One of the two instrument channels is assumed to be bypassed and out-of-service. The A and C LPRM chambers input to one channel while the B and D chambers input to the other. The channel with the greatest response is assumed to be bypassed.

3. RBM System Operation

The RBM System minimizes the consequences of a RWE by blocking the motion of the control rod before the safety limits are exceeded.

The RBM has three trip levels (rod withdrawal permissive removed). The trip levels may be adjusted and are nominally 8% of reactor power apart. The highest trip level is set so that the safety limit is not exceeded. The lower two trip levels are intended to provide a warning to the operator. Settings (typical) are 107%, 99%, and 91% of initial, steady-state, operating power at 100% flow. The trip levels are automatically varied with reactor coolant flow to protect against fuel damage at lower flows. The operator may encounter any number of trip points (up to three) depending on the starting power of a given control rod withdrawal. The lower two points may be passed up (reset) by manual operation of a pushbutton. The reset permissive is actuated (and indicated by a light) when the RBM reaches 2% power less than the trip point. The operator should then assess his local power and either reset or select a new rod. The highest power trip point may not be reset.

3.2.2.4.2.2 Operator Actions

3.2.2.4.2.2.1 FSAR Identified Operator Actions

No operator actions are required to mitigate this event, as analyzed. However as outlined in Table 3.2.2.4.2-1, the operator is assumed to ignore two warnings. Though credit is not taken for action, the following operator actions will reduce the severity of this event.

1. When withdrawing control rods in the power range, monitor the local LPRM readings and terminate the withdrawal if excessive localized peaking occurs (quantification of "excessive" is left to operator discretion as no credit is taken for operator action).

2. When withdrawing control rods in the power range, monitor the RBM response and terminate withdrawal if excessive RBM response occurs (quantification of "excessive" is left to operator discretion as no credit is taken for operator action).

3.2.2.4.2.2.2 Critique of FSAR Identified Operator Actions

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.4.3 ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP

3.2.2.4.3.1 Event Description

3.2.2.4.3.1.1 FSAR Overview

This event results directly from the operator's manual action to initiate pump operation. (The idle recirculation pump suction valve is open, but the pump discharge valve is closed.) It is assumed that the remaining loop is already in operation.

Shortly after the pump begins to move, a surge in flow from the started jet pump diffusers causes the core inlet flow to rise sharply. A short-duration neutron flux peak reaches the flow referenced APRM flux set point and initiates reactor scram. The sequence of events is listed in Table 3.2.2.4.3-1. The neutron flux peaks at 206% of NB rated. Surface heat flux follows the slower response of the fuel and peaks at 71% of NB rated. Nuclear system pressures do not increase significantly above initial. The water level does not reach either the high or low level set points.

3.2.2.4.3.1.2 Critique

The major uncertainties in the FSAR analysis are in the initial power level (55% of rated) and the initial idle recirculation loop temperature (100^{o}F) assumed. Normal procedures require startup of an idle recirculation loop at a lower power and also require heating recirculation loops within 50^{o}F of core inlet temperature prior to loop startup. Therefore, the severity of the transient could be much less than predicted, and the reactor might settle out without scram.

3.2.2.4.3.2 Operator Actions

3.2.2.4.3.2.1 FSAR Identified Actions

It is assumed that the transient will be over before the operator can insert the control rods to limit the flux peak.

Table 3.2.2.4.3-1

SEQUENCE OF EVENTS*

ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP

Time-sec	Event
0	Start pump motor.
9.8	Startup loop flow reverses.
10.3	Reactor high flux scram initiated.
50	Vessel level returns to normal and stabilizes.

*NOTE: This sequence of events represents one possible or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

Key operator action is required before the startup to choose proper procedures with minimize the transient.

3.2.2.4.3.2.2 <u>Critique</u>

The FASR-specified operator actions are not inappropriate for the event.

3.2.2.4.4 RECIRCULATION FLOW CONTROL FAILURE WITH INCREASING FLOW

3.2.2.4.4.1 Event Description

3.2.2.4.4.1.1 FSAR Overview

Failure of the master flow controller can cause a speed increase for both recirculation pumps. However, both individual speed controllers have error limiters so that this case is less severe than the failure (maximum demand) of one of the M/G set speed controllers. A rapid swing of the coupler is simulated at its maximum rate of 25%/sec.

The rapid increase in core coolant flow causes an increase in neutron flux which initiates a reactor APRM high flux scram. The sequence of events is listed in Table 3.2.2.4.4-1. The peak neutron flux rise reaches 279% of NBR flux, and the accompanying transient fuel surface heat flux reaches 88% of rated. MCPR remains above the safety limit of 1.06 and fuel center temperature increases only 503°F. The changes in nuclear system pressure are not significant with regard to overpressure. The water level does not reach either the high or low level set point.

3.2.2.4.4.1.2 Critique

The major uncertainties in the FSAR analysis are in the initial power level (65% of rated) and the maximum rate of coupler swing (25%/second) assumed. However, these are expected to have no impact on the sequence of events and operator actions.

Table 3.2.2.4.4-1

SEQUENCE OF EVENTS*

RECIRCULATION FLOW CONTROL FAILURE WITH INCREASING FLOW

Time-Sec	Event
0	Simulate failure of single loop control.
2.7	Reactor high flux scram trip initiated.
40	Vessel level returns to normal and stabilizes.

*Note: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

3.2.2.4.4.2 Operator Actions

3.2.2.4.4.2.1 FSAR Identified Actions

Initial action by the operator will include:

a. Transfers flow control to manual and reduces flow to minimum.

b. Identify cause of failure.

Reactor pressure will be controlled as required, depending on whether a restart or cooldown is planned. In general, the corrective action would be to hold reactor pressure and condenser vacuum for restart after the malfunctioning flow controller has been repaired. The following is the sequence of operator actions expected during the course of the event, assuming restart. The operator should:

a. Observe that all rods are in.

b. Check the reactor water level and maintain between low-low and high level to prevent MSIVs from isolating.

c. Switch the reactor mode switch to the "startup" position.

d. Continue to maintain vacuum and turbine seals.

e. Transfer the recirculation flow controller to the manual position and reduce set point to zero.

f. Survey maintenance requirements and complete the scram report.

g. Monitor the turbine coastdown and auxiliary systems.

h. Establish a restart of the reactor per the normal procedure.

3.2.2.4.4.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.4.5 CONTROL ROD DROP ACCIDENT (CRDA)

3.2.2.4.5.1 Event Description

3.2.2.4.5.1.1 FSAR Analysis Overview

The control rod drop accident is the result of a postulated event in which a high worth control rod, within the constraints of the banked position RSCS, drops from the fully inserted or intermediate position in the core. The highest worth rod becomes decoupled from its drive mechanism. The mechanism is withdrawn but the decoupled control rod is assumed to be stuck in place. At a later moment, the control rod suddenly falls free and drops to the control rod drive position. This results in the removal of a large negative reactivity from the core and results in a localized power excursion.

Before the control rod drop accident (CRDA) is possible, the sequence of events presented in Table 3.2.2.4.5-1 must occur. No operator actions are required to terminate this transient.

The unlikely set of circumstances referenced above makes possible the rapid removal of a control rod. The dropping of the rod results in high reactivity in a small region of the core. For large, loosely coupled cores this would result in a highly peaked power distribution and subsequent operation of shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion.

The Rod Sequence Control System (RSCS) limits the worth of any control rod which could be dropped by regulating the withdrawal sequence. This system prevents the movement of an out-of-sequence rod in the 100% to 75% rod density range, and from the 75% rod density point to the preset power level the RSCS will only allow banked position mode rod withdrawals or insertions.

The RSCS is assumed to operate throughout the event. The RWM would provide the same protection as the RSCS if the RSCS was not functioning and the RWM was.

Table 3.2.2.4.5-1

SEQUENCE OF EVENTS*

CONTROL ROD DROP ACCIDENT

Approximate Elapsed Time	Event
	Reactor is operator at 50% rod density pattern.
	RWM is not functioning.
	Maximum worth control rod blade becomes decoupled from the CRD.
	Operator selects and withdraws the control rod drive of the decoupled rod.
	Decoupled control rod sticks in the fully inserted or an intermediate bank position.
0	Control rod becomes unstuck and drops to the drive position at the nominal measured velocity plus three standard deviations.
<1 second	Reactor goes on a positive period and the initial power increase is terminated by the Doppler coefficient.
<1 second	APRM 120% power signal scrams reactor.
<5 seconds	Scram terminates accident.

*Note: This sequence of events represents one possible sequence of a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

The termination of this excursion is accomplished by inherent shutdown mechanisms. Therefore, no operator action during the excursion is required. Although other normal plant instrumentation and controls are assumed to function, no credit for their operation is taken in the analysis of this event.

3.2.2.4.5.1.2 **FSAR Analysis Critique**

1. **Mathematical Model**

   As justified in the FSAR, the mathematical mode, initial parameters, and initial conditions used to analyze the postulated CRDA are chosen to be conservative and bounding with respect to the event results. If a CRDA were to occur, the event results would most likely be much less severe.

   The sequence of events would be essentially the same for a realistic CRDA except at low power an Intermediate Range Monitor (IRM) scram trip would occur prior to the APRM trip. Also, if the rod that drops is of very low worth or drops only a short distance, a scram might not occur and the reactor would undergo a mild transient, eventually leveling off at slightly higher power level.

2. **Input Parameters and Initial Conditions**

   The core at the time of the rod drop accident is assumed to be at the point in cycle which results in the highest incremental control rod worth, to contain no xenon, to be in a hot-startup condition, and to have the control rods in sequence A at 50% rod density (groups 1-4 withdrawn). Removing xenon, which competes well for neutron absorptions, increases the fractional absorptions, or worth, of the control rods. The 50% control rod density ("black and white" rod pattern), which nominally occurs at the hot-startup condition, ensures that withdrawal of the next rod results in the maximum increment of reactivity.

   Since the maximum incremental rod worth is maintained at very low values, the postulated CRDA cannot result in peak enthalpies in excess of 280 calories per gram for any plant condition.

3. **Results**

The radiological evaluations are based on the assumed failure of 770 fuel rods. The number of rods which exceed the damage threshold is less than 770 for all plant operating conditions or core exposure provided the peak enthalpy is less than the 280 cal/gm design limit.

The results of the compliance check calculation indicate that the maximum incremental rod worth is well below the worth required to cause a CRDA which would result in 280 cal/gm peak fuel enthalpy. The conclusion is that the 280 cal/gm design is not exceeded and the assumed failure of 770 pins for the radiological evaluation is conservative.

3.2.2.4.5.2 Operator Actions

3.2.2.4.5.2.1 FSAR Identified Operator Actions

The termination of this excursion is accomplished by inherent shutdown mechanism. Therefore, no operator action is required.

3.2.2.4.5.2.2 Critique of FSAR Identified Operator Actions

Not applicable.

3.2.2.5 INCREASE IN REACTOR COOLANT INVENTORY

3.2.2.5.1 INADVERTENT HPCI PUMP START

3.2.2.5.1.1 Event Description

3.2.2.5.1.1.1 FSAR Overview

Manual startup of the HPCI system is postulated for this analysis, i.e., operator error.

The event begins with the introduction of cold water into the feedwater sparger. Within 1 second the full HPCI flow is established at approximately 20% of the rated feedwater flow rate. Addition of cooler water to the core causes the neutron flux to increase to the APRM scram set point at approximately 16 seconds. The water level reaches the high level trip set point at approximately 20 seconds initiating turbine trip, trip of feedwater pumps, and HPCI. The sequence of events is listed in Table 3.2.2.5.1-1.

3.2.2.5.1.1.2 Critique

The FSAR analyses were performed assuming the reactor was initially operating at the 105% steam flow condition. The event could produce a larger thermal power increase if the reactor were initially operating at other power/flow conditions. However, because of limitations on initial operating MCPRs the event initiated from a lower power/flow condition will not produce the minimum MCPR.

3.2.2.5.1.2 Operator Actions

3.2.2.5.1.2.1 **FSAR Identified Actions**

With the recirculation system in either the automatic or manual mode, relatively small changes would be experienced in plant conditions. The operator should, after hearing the alarm that the HPCI has commenced operation, check reactor water level and drywell pressure. If conditions are normal, the operator should shut down the system.

3.2.2.5.1.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

Table 3.2.2.5.1-1

SEQUENCE OF EVENTS*

INADVERTENT HPCI PUMP START

Time-sec	Event
0	HPCI cold water injection initiated.
1	Full flow established for HPCI.
16.1	Reactor high flux scram initiated.
20.3	L8 vessel level set point trips main turbine, feedwater pumps, and HPCI.
20.3	Turbine trip initiates bypass operation.
21.31	Turbine trip initiates recirculation pump trip (RPT).
>50 (est)	L2 vessel level set point initiates isolation of main steam lines, and HPCI and RCIC operations.
>50 (est)	SRVs cycle to release decay heat.

*Note: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

3.2.2.6 DECREASE IN REACTOR COOLANT INVENTORY

3.2.2.6.1 Instrument Line Break

3.2.2.6.1.1 Event Description

3.2.2.6.1.1.1 FSAR Overview

A circumferential rupture of an instrument line which is connected to the primary coolant system is postulated to occur outside the primary containment but inside the secondary containment. This failure results in the release of primary system coolant to the secondary containment, until the reactor is depressurized. This event could be postulated to occur in the drywell; however, the effects would not be as significant as those from a failure in the secondary containment. A break in the drywell would result in a high drywell pressure signal and would result in the initiation of the reactor safety systems (scram and ECCS). Table 3.2.2.6.1-1 is the sequence of events assumed in the FSAR analysis.

Normal plant instrumentation and controls are assumed to be fully operational during the entire plant transient to ensure positive identification of the break and safe shutdown of the plant. Minimum reactor and plant protection system operations are assumed for the analysis, e.g., minimum ECCS flow and suppression pool cooling capability. As a consequence of the accident, the reactor is scrammed and the reactor vessel cooled and depressurized over a 5 hour period.

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Subsections 3.2.2.6.2, 3.2.2.6.3, and 3.2.2.6.4. Consequently instrument line breaks are considered to be bounded by the steamline break, Subsection 3.2.2.6.2.

Table 3.2.2.6.1-1

SEQUENCE OF EVENTS*

INSTRUMENT LINE BREAK

Time	Event
0	Instrument line fails.
0-10 min	Identification of break attempted.
10 min	Activation of RHR and initiate orderly shutdown.
5 hours	Reactor Vessel depressurized and break flow terminated.

*Note: This sequence of events represents one possible sequence or a bounding sequence as defined in Chapter 15. Other event sequences can occur, depending on the assumptions made.

3.2.2.6.1.1.2 Critique

The instruments affected by the broken line can give erroneous indications. If the instruments affected control reactor systems, such as feedwater flow controls, the instrument line break can adversely affect the automatic operation of those systems. Also the instrument line break can spuriously initiate safety systems as a result of the erroneous instrument response.

3.2.2.6.1.2 Operator Actions

3.2.2.6.1.2.1 FSAR Identified Actions

The operator shall, if possible, isolate the affected instrument line. Depending on which line is broken, the operator shall determine whether to continue plant operation until a scheduled shutdown can be made or to proceed with an immediate, orderly plant shutdown, and initiate SGTS or other ventilation effluent treatment systems.

Operator action can be initiated by any one or any combination of the following:

1. Operator comparing radiation, temperature, humidity, fluid and noise readings with several instruments monitoring the same process variable such as reactor level, jet pump flow, steam flow, and steam pressure.

2. By annunciation of the control function, either high or low in the main control room.

3. By a half-channel scram if rupture occurred on a reactor protection system instrument line.

4. By a general increase in the area radiation monitor readings.

5. By an increase in the ventilation process radiation monitor readings.

6. By increases in area temperature monitor readings in the containment.

7. Leak detection system actuations.

Upon recognizing one or more of the above symptoms and having made the decision to shut down the plant, the operator should proceed to shut down the reactor in an orderly manner.

3.2.2.6.1.2.2 Critique

The FSAR-specified operator actions are not inappropriate for the event.

3.2.2.6.2 STEAM SYSTEM PIPING BREAK OUTSIDE CONTAINMENT

The steamline break outside the containment has been analyzed in the Small Break LOCA Analysis in Section 3.1.1. The system response, automatic and operator actions, and conclusions presented in Section 3.1.1 are applicable to this event.

3.2.2.6.3 LOSS-OF-COOLANT ACCIDENTS INSIDE CONTAINMENT

The loss of coolant accident inside the containment has been analyzed in detail in Section 3.1.1 for small break sizes. The system response, automatic and operator actions, and conclusions presented in Section 3.1.1 are applicable to this event.

3.2.2.6.4 FEEDWATER LINE BREAK OUTSIDE THE CONTAINMENT

This event is covered by the analysis for a steamline break outside the containment presented in Secton 3.1.1. The conclusions from that analysis are applicable to a feedwater line break outside the containment as both breaks are isolated almost immediately. Once isolated, the system responses to both events are very similar. The automatic and operator actions and conclusions presented in Section 3.1.1 are applicable to this event.

3.2.1 Loss of Feedwater

The Loss of Feedwater (LOF) event is an operational transient which occurs with a frequency of approximately 1-2 times per plant-year. The LOF event is a mild transient with respect to maintaining acceptable fuel thermal and pressure margins. However, the LOF event is the most challenging abnormal operational transient with respect to coolant inventory control since it results in the most rapid reactor coolant inventory loss. Unlike operational transients in which feedwater remains available, the LOF relies on the successful operation of other makeup systems (e.g., HPCI) for core cooling. General Electric Boiling Water Reactors are designed so that the high pressure makeup and inventory maintenance systems (RCIC, HPCI, and isolation condensers) are independently capable of maintaining the water level above the top of the active fuel given a loss of feedwater. Redundancy of systems and components is provided in the BWR design to provide margin against core uncovery and a high probability that core uncovery will be avoided.

The adequacy of the mitigating systems in providing core cooling has been conservatively demonstrated in SAR analysis. However, as pointed out in the introduction to Chapter 3, the SAR analysis may differ from the expected reactor response by the operator due to the excessive conservatisms assumed in SAR analyses. Therefore analyses have been performed using more realistic assumptions, covering normal and degraded conditions to provide guidance to operators and support for the operator guidelines.

3.2.1.1 Analysis - Loss of Feedwater Flow Event

This section supports the Emergency Procedure Guidelines as they relate to the Loss of Feedwater event. It covers all BWR/1-5 plants and a range of reactor operation from normal cases to extreme degradations.

3.2.1.1.1 Description of the LOF Transient

Feedwater serves two fundamental purposes:

1. It replenishes the reactor coolant inventory loss due to steam flow to the turbine and other paths;

2. It mixes with the relatively hot steam separator return flow to provide proper core inlet subcooling so that the void reactivity control is achieved.

A LOF may occur as a result of loss of AC power, feedwater pump failures, condensate pump failures, feedwater controller failures, operator errors, or trip on reactor high water level. The following is a description of reactor behavior following LOF initiation. A standard BWR/4-251 plant is described.

3.2.1.1.1.1 Short Term Phenomena (Before Scram)

This paragraph provides a description of the LOF event for a typical BWR/4. Upon a Loss of Feedwater, vessel water level starts to decrease due to the mismatch between coolant inventory loss (steam) and supply (feedwater). The rate of level decrease depends on the initial power level: higher initial power will cause faster level decrease. Because of diminishing injection of relatively cold feedwater, core inlet flow becomes warmer. This causes more void generation in the core, hence neutron flux decreases. When the plant is in the automatic flow control mode, control systems will function to attempt to maintain the core power by increasing the recirculation pump speed (hence, the core flow). When the level decreases to the low level alarm setpoint (L4), a vessel low level annunciator comes on and runback of the recirculation pump is initiated to protect the recirculation pumps from cavitation. Runback will cause a rapid increase of void fraction in the core, which in turn will cause a rapid reduction in neutron flux (hence thermal power). The increase of core void fraction leads to redistribution of some liquid from the core to the vessel downcomer. This causes a temporary slowdown of sensed vessel level (level indication in the control room and on trip instruments) decrease. The level will continue to decrease and reach the low level scram setpoint L3) where reactor scram is initiated.

3.2.1.1.1.2 Long Term Phenomena (After Scram)

At the low level scram setpoint (L3) most of the primary containment isolation valves (except MSIV) are closed. Scram will cause a further rapid level reduction due to the redistribution of vessel downcomer water to fill the collapsed voids inside the core. The amount of water redistribution depends on the initial void fraction of the core: for a given recirculation flow, higher initial power leads to a larger level reduction after scram.

Once the voids in the core have collapsed, level continues decreasing due to steaming to the main condenser through the turbine bypass valves. Level eventually decreases to the low-low (L2) trip setpoint.

The low-low trip will close the MSIVs and valves not closed at L3, trip the recirculation pumps, and initiate HPCI and RCIC. Recirculation pump coastdown maintains higher than natural circulation core flow for a period of time. The vessel pressure soon rises to the safety relief valve (SRV) setpoint. The pressure then remains at approximately the setpoint pressures as one or more SRVs cycle open and closed to maintain pressure control. The vessel pressure is maintained by steam generated by the decay heat of the fuel. Vessel inventory continues to be lost as steam through the SRVs.

Under normal conditions, the high pressure makeup water systems will provide sufficient water to restore the level to the normal range. Plant shutdown or restart can then be accomplished. For degraded conditions where all of the high pressure systems are unavailable, the water level will continue to drop. Under these conditions, depressurization of the reactor to the range where the low pressure systems can inject water into the reactor is necessary.

Due to the large capacity of the low pressure systems, they will rapidly reflood the reactor. Once the vessel is reflooded, the operator can then proceed to place the reactor in cold shutdown.

Under some of the degraded conditions analyzed here, core uncovery is calculated to occur. Core uncovery in itself is not a critical safety factor as long as the cladding temperature is minimized during the event. This is accomplished through timely injection of low pressure ECCS for such degraded conditions.

3.2.1.1.2 **Models**

In licensing basis transient analysis the main concerns are:

1. Fuel thermal margin (MCPR); and

2. Integrity of Reactor Coolant Pressure Boundary (RCPB).

The LOF transient is categorized as a pressure increase event in FSAR Chapter 15 analyses. However, it does not become a pressurization event until after scram and MSIV closure. Consequently, reactor pressure rises moderately, (e.g., compared with turbine trip) and actuates only the Group 1 relief valves, well below the RCPB limiting pressure. Typically, the peak pressure is reached 30 seconds into the event. Having demonstrated meeting the RCPB pressure margin requirement, the licensing analysis usually terminates slightly after the time of peak pressure (typically, 50 seconds into the event). By this time, the S/RVs have cycled a few times, and the vessel level has stabilized below L2 but well above L1. The actuation and injection of high pressure ECCS is not included in the licensing analysis, because the subsequent reactor behavior is of little significance as far as thermal margins and pressure margins are concerned.

Thermal and pressure margins are very fine-scale parameters. Local effects as well as the dynamic behavior of reactor systems are very critical to these parameters. The REDY code is utilized in licensing analyses, due to its fine nodalization representation of the NSSS and BOP systems and its comprehensive thermal-hydraulic and neutronics models.

Analysis to support Emergency Procedure Guidelines, on the other hand, demands an analytical model designed for simulating long term inventory behavior. Local and short term effects are of little significance to an operator in an emergency. The SAFE code is suitable for long-term inventory modeling, and is used in the analyses of this section. Its use is more thoroughly described in Section 3.2.1.3.

3.2.1.1.3 **Determination of System Configurations to be Analyzed**

In this section, the specific system configurations to be analyzed are determined.

The reactor systems which are available for providing makeup water and removing decay heat are described in Section 3.1.1.1.2.1.

Due to the large injection capacity and redundant loop features of the low pressure injection systems, low pressure injection systems provide ultimate assurance of core cooling.

However, high pressure injection systems are preferable for the following reasons:

1. If after scram the reactor is in a high pressure condition, the high pressure systems can immediately provide makeup water;

2. If the reactor is maintained at high pressure, the plant can be restarted immediately;

3. Avoidance of depressurization eliminates a further reactor and containment system transient.

4. High-pressure systems take suction from high-quality water sources.

Principal emphasis, therefore, is given to high pressure systems in the analyses and in the guidelines.

After scram and MSIV closure, inventory maintenance and decay heat removal are the major objectives to be accomplished by the operator and the mitigating systems. Hence, the spectrum of realistic LOF cases is determined by the various combinations of inventory maintenance and decay heat removal systems assumed to be available. The case of a stuck-open relief valve (SORV) is also considered.

The various combinations of mitigating systems under normal and degraded conditions are illustrated in Figure 3.2.1.1.3-1 and 3.2.1.1.3-2 respectively for BWR/1, 2, 3 with Isolation Condensers (IC) and BWR/3, 4, 5 with RCIC. The combinations of assumed degradations lead to system configurations labeled by (A), (B), (C), etc. The cases to be analyzed are based on the severity of the systems configurations. For example, for the most severe cases where complete loss of high pressure makup water systems occur, analyses are performed for each product line. For the less severe cases where at least one mitigating system is capable of maintaining core cooling, analysis is performed for a representative product line instead of every product line. For example, when IC and FWCI are both available, BWR/2 response is very similar to BWR/3, and analysis is only performed for BWR/3. When only IC is available, BWR/3 response is very similar to BWR/2, and analysis is only performed for BWR/2. These cases also cover BWR/2 without FWCI. The essential point is that each system configuration is covered by at least one analysis.

The specific system configurations analyzed are summarized below.

BWR/1, 2, 3 with IC

Refer to Figure 3.2.1.1.3-1. The BWR/1 is covered in more detail in Section 3.2.1.3.3.

Configur tion (A), (E): Reactor responses for BWR/2 and 3 are very similar. Analysis is performed for BWR/3.

Configuration (B), (F): Reactor response is very similar to BWR/4. These configurations are covered by BWR/4 analysis.

Co figuration (C): Reactor response for BWR/2 and 3 are very similar. Analysis is performed for BWR/2.

Configuration (D), (H): Analysis is performed for BWR/2 and 3.

Configuration (G): Reactor response is very similar to (H). This configuration is covered by (H).

BWR/3, 4, 5 with RCIC

For BWR/3 and 4 equipped with RCIC and HPCI, the reactor responses are very similar. Hence analyses are performed for a BWR/4 reference plant, and BWR/5 analyses are performed where appropriate.

Refer to Figure 3.2.1.1.3-2.

Configuration (A), (C), (E), (F), (H). Analyses is performed for BWR/4 and 5.

Configuration (B), (G). When HPCI/HPCS is on, the incremental injection flow from RCIC is negligible. Hence (B), (G) are covered by (A), (H).

Configuration (D). When SORV is not present, RCIC alone can maintain level according to the design basis for RCIC. Hence (D) is covered by (A).

In performing the above analyses, automatic initation of mitigating systems is assumed. Reactor responses following manual initiation are very similar to auto-initiation. Hence they are covered by the above analyses.

3.2.1.1.4 **Analysis Assumptions**

The most appropriate analyses for developing operator guidelines and emergency procedures are best estimate predictions of the system performance. Nominal (or best estimate) input data and conditions are assumed in the analyses of this section. These assumptions include:

Assumption	Rationale
1. Reactor is initially at 105% steam flow and 100% core flow. Other reactor parameters are at nominal value.	It realistically bounds other reactor conditions.
2. 1978 ANS Standard decay heat model is used.	This is the most realistic decay heat model.
3. h=12 Btu/hr-ft$^{2\circ}$F is assumed for core spray heat transfer coefficient.	Recent experiments in the Two-Loop Test Apparatus (TLTA) have demonstrated higher heat transfer coefficients than the presently used ECCS licensing model value of ,tu/hr-ft$^{2\circ}$F.

Assumption	Rationale
4. Nucleate boiling is assumed when the core is covered.	Recent TLTA tests have demonstrated that the core remains in nucleate boiling as long as the core is covered.
5. Simultaneous trip of all the feedwater pumps is assumed at t = 0.	This leads to the most rapid level reduction, and the subsequent events encompass a complete spectrum of phenomena pertaining to the LOF event. Realistically, this can correspond to a spurious high level trip signal.
6. Feedwater flow coastdown time is 5 seconds.	This is a realistic value for motor-driven feed pumps and conservative (although not unduly so) for steam-turbine-driven feed pumps.
7. CRD flow is neglected.	Due to the small CRD flow rate, it has a small effect by comparison with feedwater flow. This assumption does not alter the conclusions of this section.
8. When SORV is postulated, it is assumed not to occur to an ADS valve.	In some plants, the ADS valves are higher pressure group relief valves. The effect of this assumption will be discussed.

The effects of inputs differing from the ones listed above are examined.

In performing the analysis, the following approach is adopted.

1. A representative plant is used for each product line.

2. BWR/1 results are inferred from BWR/2, 3 analysis. This is explained in Section 3.2.1.3.

3. The SAFE code is used for predicting the long term thermal hydraulic behavior. This is explained in Section 3.2.1.3.

4. Analyses for cases requiring low-pressure ECCS extend only to the time when injection is assured. Subsequent behavior is covered by the SBA analyses of 3.1.1.1.

3.2.1.1.5 **Analysis and Results**

3.2.1.1.5.1 **All Systems Operable**

In the event of a total Loss of Feedwater Flow, the most likely situation is that all the mitigating systems function as designed. The high pressure systems are initiated automatically to mitigate the consequence of the event. Figure Groups 3.2.1.1.5-1 to 3.2.1.1.5-4 illustrate the reactor systems responses for BWR/2, BWR/3 with IC, BWR/3, 4 with RCIC and BWR/5. In each figure group, the time histories of the key variables are plotted. They include:

- Vessel pressure
- Steam line flow
- Total SRV flow
- ADS flow
- Total recirculation flow
- Actual level inside shroud
- Actual level outside shroud
- Fuel temperature
- Reactor power
- Total heat to coolant
- Feedwater flow
- ECCS flow

To aid in interpretation of these figure groups, event sequences are summarized in Tables 3.2.1.1.5-1 to 3.2.1.1.5-4.

With respect to the level outside the shroud, note that the actual level outside shroud is plotted. The actual level differs from the sensed level in that the actual level corresponds to a two-phase mixture level while the latter corresponds to a collapsed level (the two-phase mixture is collapsed into an equivalent liquid-only level). Usually, there is little steam present in the liquid inside the annulus, hence the actual level is approximately equal to the sensed level.

Figure Goup 3.2.1.1.5-1 is for BWR/2 with two ICs in operation without makeup water. Due to the large heat removal capacity of two ICs, SRVs are not actuated and the system experiences no inventory loss after MSIV closure. Reactor water level remains well above the top of active fuel (TAF) throughout the event.

Figure Group 3.2.1.1.5-2 is for BWR/3 with one IC in operation and with FWCI providing makeup water. The initial decay heat exceeds the heat removal capacity of one IC, hence SRVs are actuated a few times. However, the inventory loss through SRV actuation is made up by FWCI. After the decay power is below the capacity of IC, SRV actuation ceases. Water level is recovered to the normal range by FWCI.

Figure Groups 3.2.1.1.5-3 and 3.2.1.1.5-4 are very similar. They are for BWR/3, 4 with RCIC and BWR/5. Inventory is lost after MSIV closure through SRV actuations and RCIC/HPCI steam extraction. However the injection capacity of the high pressure systems (RCIC & HPCI for BWR/3, 4 and RCIC & HPCS for BWR/5) is so large that reactor level is quickly recovered. The high pressure injection systems are tripped off on high water level (L8). HPCI and HPCS will restart automatically if a subsequent low water level is reached.

These figure groups demonstrate that under design conditions BWRs are capable of mitigating the consequences of LOF event automatically without the operator's assistance. This is achieved through the automatic functioning of various mitigating systems.

3.2.1.1.5.2 Degraded Conditions - Partial Failure of High Pressure Systems

3.2.1.1.5.2.1 BWR/2

3.2.1.1.5.2.1.1 Failure of One IC

The reactor behavior with only one IC in operation is illustrated in Figure Group 3.2.1.1.5-5. The sequence of events is summarized in Table 3.2.1.1.5-5. The initial decay heat exceeds the heat removal capacity of one IC, hence SRVs are actuated. However, shortly afterwards, decay heat falls below the capacity of one IC, SRV actuation ceases, and the reactor no longer experiences inventory loss. Water level remains high above the TAF. Adequate core cooling is provided throughout the event.

3.2.1.1.5.2.1.2 Failure of One IC, SORV

The condition is very similar to failure of both ICs with SORV, which is analyzed in Section 3.2.1.1.5.3.

3.2.1.1.5.2.2 BWR/3 with IC

3.2.1.1.5.2.2.1 FWCI and IC On, SORV

The reactor response is illustrated in Figure Group 3.2.1.1.5-6. The sequence of events is summarized in Table 3.2.1.1.5-6. With inventory makeup due to FWCI, reactor water level is recovered to the normal range. The SORV and IC depressurize the reactor without operator action to where shutdown cooling can be put into operation.

3.2.1.1.5.2.2.2 IC Only, SORV

This condition is very similar to the failure of IC and FWCI with SORV, which is analyzed in Section 3.2.1.1.5.3.

3.2.1.1.5.2.3 BWR/3, 4 with RCIC and BWR/5

3.2.1.1.5.2.3.1 RCIC, HPCI/HPCS On, SORV

The results are depicted in Figure Groups 3.2.1.1.5-7 and 3.2.1.1.5-8 for BWR/3, 4 and BWR/5 respectively. The sequence of events are summarized in Tables 3.2.1.1.5-7 and 3.2.1.1.5-8. With the large injection capacity of RCIC and HPCI/HPCS, reactor water level is quickly recovered beyond normal range. RCIC and HPCI/HPCS are tripped off on L8. The reactor depressurizes to where shutdown cooling can be put into operation via steam discharge through the SORV, through the HPCI/RCIC turbines, or through manually opened S/RVs.

3.2.1.1.5.2.3.2 RCIC Only, SORV

The results are depicted in Figure Groups 3.2.1.1.5-9 and 3.2.1.1.5-10 for BWR/3, 4 and BWR/5 respectively. The sequence of events are summarized in Tables 3.2.1.1.5-9 and 3.2.1.1.5-10. The injection capacity of RCIC is initially below the SORV steam discharge, and water level decreases at a slow rate. As the reactor depressurizes, steam discharge decreases. The reactor reaches a quasi-steady state where steam discharge approximately equals RCIC flow and the energy efflux approximately equals decay heat. Reactor pressure decreases and level increases, both at very slow rates. This qausi-steady condition will persist for a relatively long time. The pressure will eventually decrease to the low pressure ECCS or condensate pump shutoff head, and low pressure systems can quickly restore level to normal range. (Low pressure ECCS are either automatically initiated on L1 or manually actuated by the operator.) The operator can enhance response by manually depressurizing through one or more S/RVs.

Figure Group 3.2.1.1.5-9 illustrates the BWR/4 response under the RCIC only, SORV condition, with enhancement by manual depressurization (BWR/5 response would be similar). When the quasi-steady state is reached, the operator is assumed to manually open one SRV (at 1900 seconds). The low pressure ECCS are enabled, and the reactor level is quickly restored to the normal range. (The plot is shown to the time slightly before LP ECCS reaching full injection capacity. The water level will quickly recover. In-shroud level remains well above the TAF throughout the event.

Figure Group 3.2.1.1.5-10 illustrates the BWR/5 responses under the RCIC only, SORV condition without operator action (BWR/4 response would be similar). During the quasi-steady condition, the reactor pressure decreases to the low pressure ECCS injection point. However, the injection of low pressure ECCS is not simulated to demonstrate the capability of RCIC to turn the water level around. Figure 3.2.1.1.5-10.4 clearly indicates that the in-shroud level reaches a minimum of 31.4 ft (which is approximately 2 feet above the TAF), then starts to increase.

Figure Group 3.2.1.1.5-10 indicates that for the reference plant studied here RCIC alone can maintain the reactor in a safe and stable condition for a prolonged period (>1 hour) even with the presence of an SORV. This leaves the operator abundant time to either attempt to reclose the SORV, to restart the high-pressure ECCS or feedwater, or to establish a low-pressure cooling system.

3.2.1.1.5.3 Degraded Conditions, Failure of All High Pressure Systems

Under the postulated conditions where failure of all high pressure systems occurs, the reactor experiences a quasi-steady inventory loss through repetitive actuation of SRVs. Inventory loss will be increased if a relief valve sticks open. The level transients are depicted in Figure 3.2.1.1.5-11 to 3.2.1.1.5-14 for BWR/2 to 5 with and without SORV. These figures indicate:

1. Without operator action, core uncovery will occur. The need for operator action to assure adequate core cooling under these conditions is addressed in Section 3.5.2.1.

2. With a SORV, the initial flashing causes the swollen level to decrease more slowly than without a SORV, even though there is more mass lost from the RPV. As the flashing subsides, water level decreases more rapidly than without a SORV.

3. A SORV will eventually depressurize the reactor to where the low pressure ECCS can inject water. Without an SORV, reactor pressure will cycle between the opening and reclosing setpoints of the SRVs until the reactor is manually depressurized.

It is clear that in the event of failure of all high pressure systems, core cooling relies on timely operator action and low pressure ECCS injection. Due to the lack of containment high pressure, ADS will not be initiated automatically.

With manual depressurization of the reactor, low pressure ECCS systems can inject. As soon as they inject, the reactor is quickly reflooded.

Figure Groups 3.2.1.1.5-19 to 3.2.1.1.5-26 depict typical BWR response with no high pressure systems available and with manual ADS. Figure Groups 3.2.1.1.5-19 to 3.2.1.1.5-22 are for BWR/2-5 without SORV and Figure Groups 3.2.1.1.5-23 to 3.2.1.1.5-26 are for cases with SORV. The corresponding sequence of events tables are provided in Table 3.2.1.1.5-11 to 3.2.1.1.5-14 and Table 3.2.1.1.5-15 to 3.2.1.1.5-18. In these figure groups, except for the BWR/3 analysis manual ADS is arbitrarily assumed to be accomplished when the vessel level descends to 5 ft above the TAF. In the representative BWR/3 analysis, vessel level descends to 5 ft above the TAF before the pressure reaches the lowest S/RV setpoint. Therefore, for consistency with the other analyses, manual ADS in this case is taken with 120-second delay after level reaches 5 ft above the TAF.

These figure groups illustrate that BWRs behave similarly under the same degraded conditions. They also demonstrate the BWR's in-depth protection to mitigate the consequences of highly degraded multiple failure conditions with minimal operator actions.

3.2.1.1.5.4 **Summary and Conclusions**

The analyses performed in this section have encompassed a complete spectrum of LOF events analysed with realistic assumptions, ranging from all mitigating systems available through the very degraded case of no high-pressure systems (HPCI, HPCS, RCIC, CRD, FWCI, IC) available. The results are summarized below according to available mitigating systems:

1. All Mitigating Systems Available

This is the most likely situation to be encountered. For all product lines, the reactor reaches a safe and stable condition automatically without operator assistance. Operator actions are limited to verifying and confirming the initiation and completion of automatic actions.

2. Partial Failure of Mitigating System

a. BWR/2, 3 with IC

When a SORV is not present, the operation of one IC or the operation of FWCI (if so equipped) suffices to mitigate the consequences of an LOF event. No operator action is required.

When SORV is present for BWR/2 or 3, FWCI (if so equipped) can mitigate the consequences of an LOF event, but IC by itself cannot. Hence if FWCI is unavailable, the operator must manually depressurize to mitigate the consequence of the LOF event.

b. BWR/3, 4, 5 with RCIC and HPCI/HPCS

The operation of either RCIC or HPCI/HPCS can mitigate the consequences of an LOF event whether SORV is present or not.

3. Complete Failure of All High Pressure Injection and Inventory Maintenance Systems

This is a highly improbable condition. Should it occur, the consequences can be mitigated by timely manual depressurization if necessary, to enable the injection of the low pressure ECCS.

Based on the above, it can be concluded that:

1. The BWRs covered in this report are adequately equipped to mitigate the consequence of the LOF event as it relates to core cooling without operator assistance under all conditions within the design basis, with or without a stuck-open relief valve.

2. Operator actions are required only under the highly improbable conditions where complete loss of high pressure injection and inventory maintenance systems occurs. In such an event, timely manual depressurization followed by injection of low-pressure systems suffices to mitigate the consequences.

3. A stuck-open relief valve, even with a complete loss of feedwater, is a controllable event. The consequences of a SORV can be mitigated by the injection of either FWCI, RCIC, HPCI, or HPCS. If none of these injection systems are available, the consequences can be mitigated by the injection of low pressure systems following manual reactor depressurization.

3.2.1.2 Loss of Feedwater Operator Guidelines

The operator guidelines for the loss-of-feedwater event are included in the Emergency Procedure Guidelines.

3.2.1.3 Model Justification and Sensitivity Studies

This section justifies the analytical model used in the analyses of Section 3.2.1.1, and investigates the sensitivity of the results to certain important parameters.

3.2.1.3.1 SAFE Analytical Model

The analyses in Section 3.2.1.1 utilize the SAFE code to predict system thermal-hydraulic response. The use of the SAFE code instead of the transient code REDY (which is used in licensing analyses of LOF) is justified below.

The licensing basis LOF analysis employs the transient code REDY. REDY is designed for analyzing situations where short term thermal margins (MCPR) and pressure margins are of primary concern. In general, these concerns exist only when the core power and core flow are at relatively high levels (from the initiation of the event to shortly after scram).

REDY is not intended to handle very low power and low flow conditions (which are typical of long term LOF behavior). REDY also lacks the capability to simulate the various combinations of core cooling systems because their performance is not required for evaluating thermal margin and pressure margin. Thus, while REDY is most suitable for simulating short-term LOF behavior, it is not suitable for developing operator guidelines because the guideline development is dominated by the long term behavior of the reactor system.

The SAFE code is judged to be the most suitable analysis tool for the development of LOF Operator Guidelines. The SAFE model is a flexible tool for the evaluation of emergency core cooling systems. The lumped parameter model can accurately predict the long term reactor vessel pressure transient and water inventory over the entire range of credible break sizes and locations, down to and including a "zero break" (loss of feedwater only). Any combination of core cooling systems can be simulated.

Because SAFE is intended to predict long term transients, short-term and local effects are generally ignored. During the first few seconds after an initiating event, local effects dominate reactor response. However, after an initial period of rapid change, pressure traces from SAFE agree very well with those from codes which allow spatial pressure variation based on more detailed models of the internal structure.

As stated in Section 3.2.1.1, the initial transient of an LOF event elapses in a very short period of time (< 30 seconds). The LOF analyses in this report, on the other hand, are intended to provide guidance for operator action during the longer-term decay heat removal and level recovery stage. While SAFE lacks details in simulating the initial transient of LOF event, the lack of detail has minimal impact on the long term behavior. This has been discussed in Section 3.1.1.3 and 3.1.1.4, and discussed below in greater detail.

Table 3.2.1.3.1 is a comparison of the modeling assumptions incorporated in REDY and SAFE.

The major differences between REDY and SAFE predictions are as follows:

1. REDY employs a neutronics model to simulate the effect of decreasing core inlet subcooling and recirculation flow runback. Hence, REDY realistically predicts the reduction of neutron flux prior to scram. SAFE treats the power generation prior to scram more simply, assuming initial power until scram. After scram, the heat generation models are compatible: the standard ANS decay heat model can be input into both codes. These differences have practically no effect on the longer-term level and pressure transients of interest to operator action.

2. Because SAFE's main steam flow model is simplified compared to REDY, the timing of the L3 and L2 trips predicted by SAFE is slightly different from that by REDY. However, due to the rapidity of level decrease, the difference in timing is on the order of only a few seconds. Again, the differences have practically no effect on the longer-term pressure and level transients.

To demonstrate the above, a comparison of REDY and SAFE for BWR/4-251 LOF condition is presented in Figures 3.2.1.3.1-1 and 3.2.1.3-2. Figure 3.2.1.3.1-1 is for level comparison and Figure 3.2.1.3.1-2 is for vessel pressure comparison. Figure 3.2.1.3.1-1 shows that the two codes predict almost identical timing for low level (L3) scram. The isolation time (time corresponding to L2) predicted by SAFE is approximately 9 seconds later than that predicted by REDY. Both codes predict almost identical level.

The pressure comparison (Figure 3.2.1.3.1-2) shows that the two codes predict almost identical minimum pressures reached within a difference of approximately 9 seconds (corresponding to the difference in reaching L2 isolation). Both codes predict the reactor pressure cycling between the opening and reclosing setpoint of SRVs with a difference of approximately 30 seconds in reaching the first SRV actuation.

REDY will not run beyond the time domain shown in these figures due to the nearly vanishing core flow. However for the short-term comparison presented here, it can be concluded that while SAFE lacks certain details in simulating the initial LOF transient behavior, it reasonably well simulates the initial critical events which are important to the long term behavior.

3.2.1.3.2 Sensitivity to System Parameters

The analyses of Section 3.2.1.1 included various combinations of mitigating systems degradations. This section demonstrates that over the realistic range of the critical parameters used in the analyses, the predicted event sequence will not differ significantly. While there will be variations in details, the major trends remain the same, and the variations have no significance to an operator in an emergency, except for timing.

3.2.1.3.2.1 Effect of Initial Power/Flow Condition

In Section 3.2.1.1.5.1 to 3.2.1.1.5.3, the initial conditions were 105% steam flow and 100% core flow. In this section, the effect of different reactor conditions are studied for a standard BWR/4-251 plant. Refer to Figure 3.2.1.3.2-1, the power-flow map. Reactor conditions are selected from the 105% load line, Points A, B, and the minimum pump speed line, Points C, D, E, F. The corresponding power/flow are conditions summarized in Table 3.2.1.3-2.

Analyses are performed for the most severe failure combination, all high pressure systems unavailable (without SORV). Manual ADS was arbitrarily assumed to take place at 5 feet above TAF. The results for Points A, B, C are presented in Figure Groups 3.2.1.3.2-2 to 3.2.1.3.2-4. It can be seen that the reactor behavior depicted in these figure groups is very similar to that for the 105% steam flow analysis (Figure Group 3.2.1.1.5-21). No new phenomena are introduced as a consequence of changing the power/flow condition. The different initial conditions affect only the timing of manual ADS and the minimum in-shroud level achieved, as summarized in Table 3.2.1.3-2 for Point A to F.

Table 3.2.1.3-2 shows that as initial power decreases, the operator gains more time for manual depressurization.

Table 3.2.1.3-2 also indicates that Points A, B, C, D, and E result in approximately the same minimum in-shroud level. Point F results in a higher minimum value. Minimum in-shroud levels for Points A, B, C, D, and E are achieved just before the low pressure systems start to inject. For Point F, the minimum level is achieved slightly before manual ADS. The different behavior of the lower-power case is simply explained. When manual depressurization is acccomplished, the rapid depressurization causes a strong flashing inside the core. The entire core is covered with a two-phase mixture. As depressurization subsides, so does the flashing. For higher power, flashing subsides before the injection of low pressure ECCS. Hence minimum level is achieved during depressurization. For very low power, such as Point F, flashing is still underway when the low pressure systems start to inject. Hence minimum level is achieved at the moment of initiation of depressurization for the low-power case.

From the above discussions, it can be concluded that the analyses performed in Section 3.2.1.1.5.1 to 3.2.1.1.5.3 are sufficient to encompass the entire range of power/flow conditions.

3.2.1.3.2.2 Effect of Decay Power

The 1978 ANS Standard decay heat is assumed in all the analyses. To investigate the effect of decay heat on reactor response, an analysis is performed using ANS Standard + 20% decay heat for a standard BWR/4-251 plant. The initial condition is 105% steam flow and 100% core flow. Failure of all high pressure systems and no SORV is assumed for the system configuration. Figure Group 3.2.1.3.2-5 depicts the reactor response for manual ADS at 5 ft above TAF. It can be seen that the reactor response is very similar to that for ANS Standard decay heat (Figure Group 3.2.1.1.5-21). Due to higher decay heat, the time at which the level reaches 5 feet above TAF is earlier than in the base analysis. The minimum in-shroud level is about the same.

The increase in the duration of core uncovery due to 20% greater decay heat is at most on the order of 20 seconds, so the impact on fuel performance is minimal. Hence, an increase of 20% decay heat over ANS Standard value has no impact on the essential operator actions.

3.2.1.3.2.3 Effect of Number of ADS Valves

In the preceding sections, the stuck-open valve (when assumed) is taken to be a non-ADS valve. Hence when manual ADS is initiated there is one more than the number of ADS valves contributing to the depressurization. In some plants it is possible that the SORV will be one of the ADS valves. The number of valves, however, will not affect the basic event sequence. It only affects the depressurization rate, hence the minimum in-shroud level.

3.2.1.3.2.4 Power Source Availability

With the exception of plants with FWCI, the analyses are not sensitive to a loss of offsite power (LOOP) during the transient because CRD flow was not simulated in the analyses. Reactor response for FWCI-equipped plants with FWCI unavailable has been analyzed in Section 3.2.1.1.

If LOOP is combined with a DC power source failure in one division, the availability of both high pressure and some low pressure ECC systems may be affected for some plants. The system responses for these cases are covered by the complete loss of high pressure systems analysis performed in Section 3.2.1.1. Section 3.5.2.1 covers such degraded cases in more detail.

3.2.1.3.3 BWR/1 Evaluation

No specific BWR/1 analyses were made in Section 3.2.1.1. However it will be demonstrated below that the BWR/1 responses under all LOF conditions are qualitatively covered by the analyses performed in Section 3.2.1.1 for BWR/2 through 5, and therefore that they support the Emergency Procedure Guidelines as applied to BWR/1.

For BWR/1 there are two high pressure auxiliary systems available for use in the event of loss of feedwater: (a) the isolation condenser (IC) which has a capacity greater than 7% of nuclear-boiler-rated power, and (b) the control rod drive (CRD) system which provides make-up water of similar capacity.

For some plants, MSIV closure occurs at the same time of reactor low level scram. The IC acts to remove decay heat. The reactor no longer experiences inventory loss after the auto-initiation of IC, and the reactor automatically reaches a safe and stable condition. This is very similar to the "All Systems Operable" analysis performed in Section 3.2.1.1.5.1 for BWR/2.

For some plants, the MSIV closure setpoint is much lower than the low level scram setpoint. Steam continuously flows to the main condenser after scram. However, in these plants the steam flow (which is equal to the steam flow generated by the decay heat) can be made up by the CRD flow. The reactor also reaches a safe and stable condition automatically. It has been estimated in the absence of CRD flow, approximately one hour will elapse before the level descends to the MSIV closure setpoint for this class of reactors. Once this occurs, the operation of IC terminates the event. The entire event is automatic; no operator action is required.

For the degraded condition where the IC fails to operate, safety valves will be actuated due to the pressure rise. If the safety valves function properly, (i.e., no SORV is present), manual depressurization (which could be accomplished with power operated relief valves in plants so equipped) is required to depressurize the reactor so that the low-pressure ECCS can inject. A stuck-open valve, if present, would depressurize the reactor to the low-pressure ECCS injection point due to the large capacity of the safety valves in the BWR/1 units.

The design power for BWR/1s is considerably smaller than that for BWR/2-5. However the volumes of the BWR/1 NSSS systems are not scaled-down proportionally. Qualitatively, this means that the steam-feedwater mismatch as well as decay heat experienced per unit volume of BWR/1 primary system is much smaller than that for BWR/2-5. In that case, the rate of level decrease as well as the pressurization rate will be slower for BWR/1 than for BWR/2-5 given a LOF. The slower response of the BWR/1 will in general allow the operator more time to act.

Due to the similarities of the mitigating systems available for BWR/1 and BWR/2, the sequences of events presented in Section 3.2.1.1 are also applicable to BWR/1 except that the timing for BWR/1 events will be slower than the corresponding BWR/2 timing.

3.2.1.4 Response to NRC Questions

This section specifically addresses the NRC questions to the BWR Owners' Group dated September 28, 1979 (Ref. 1). These questions were answered in part by Ref. 2. The questions dealt with Section 3.2.1 of the August, 1979 version of NEDO-24708, which is entirely superseded by this version of Section 3.2.

1. Provide a detailed discussion of how the SAFE code simulates a transient such as loss of feedwater (LOFW). Describe all input parameters and output parameters used for determining the Sequence of Events tables in NEDO-24708. Provide a comparison of the results of a SAFE code simulation with the normal transient code (REDY/ODYN) for each reactor class. Describe all modifying assumptions made when using the SAFE code to simulate transients.

Response to Question 1

(i) The adequacy of utilizing SAFE in analyzing the LOF events for the development of operator guidelines has been discussed in Section 3.2.1.1.2 and Section 3.2.1.3.1.

(ii) Comparison of LOF transients predicted by REDY and SAFE is provided in Section 3.2.1.3.1.

(iii) The assumptions adopted for the simulation of LOF event by utilizing SAFE are provided in Section 3.2.1.1.4.

(iv) Level setpoints are the most important input parameters used for determining the sequence-of-events tables. These include the L3 scram setpoint, L2 MSIV closure setpoint (also for HP systems auto-initiation, trip of recirculation pumps etc.) and L1 LP systems auto-initiation setpoint. In some cases, pressure setpoints, such as SRV opening and reclosing setpoint, are also important parameters for determining the sequence-of-events.

The reactor pressure and the differential pressure of the water inventory above the pressure instrument tap are the important SAFE output parameters used for determining the sequence-of-events table.

2. Provide details on how BWR/1 transients were derived from the BWR/2 analyses.

Response to Question 2

In the August 1979 LOF analysis, the BWR/1 LOF sequence-of-events tables were derived from BWR/2 analyses since there was a great deal of similarity between the mitigating systems on these two designs. The sequence of events was inferred from BWR/2 analysis for illustrative purpose only, taking into consideration certain BWR/1 unique characteristics (e.g. see response to question 5). As discussed previously, the exact timings are not required for the development of operator guidelines. In section 3.2.1.3.3, a discussion has been presented which describes why a BWR/1 response is similar to a BWR/2, and a unique BWR/1 analysis is not necessary for the development of LOF operator guidelines.

3. Provide a complete set of curves for the BWR/4 LOFW analyses. These should include: vessel level, vessel pressure, steam and feedwater flow, safety relief valve flow, ECC flows steam line pressure, peak fuel temperature, bypass valve flow, wide range and narrow range sensed level, core inlet flow, drive flow neutron flux. For other reactor classes provide vessel pressure, vessel level, SRV flow, ECCS flows, steam flow, feedwater flow.

Response to Question 3

Complete sets of curves for BWR/2-5 LOF events are provided in Section 3.2.1.1.5.1.

4. Identify the representative plant in each reactor class and the rationale for selection. Describe how representative plants provide plant specific transient response when systems characteristics of plants differ within each reactor class.

Response to Question 4

The representative plant selected for LOF analysis in each reactor class (BWR/2-5) is as follows:

BWR/2	(EC/FWCI):	Nine Mile Point
BWR/3	(EC/FWCI or HPCI):	Millstone
BWR/3, 4	(RCIC/HPCI):	Browns Ferry
BWR/5	(RCIC/HPCS):	Zimmer

It has been demonstrated in Sections 3.2.1.1 and 3.2.1.3 that BWRs of different product lines behave very similarly under normal and degraded conditions. This is to be expected since common design philosophy and technology has been implemented in all the GE BWRs. This is demonstrated when the results between different product lines (e.g., BWR/4 versus BWR/5) are compared. Greater similarity is of course expected for plants in the same reactor class.

5. For the BWR/1 reactor with LOFW and no control rod drive (CRD) flow, show that the operator has one hour to manually isolate the reactor before core uncovery.

Response to Question 5

For the steam drum-equipped BWR/1 reactor which does not isolate automatically during a high-power LOF event with CRD available (i.e., the low-level isolation setpoint is considerably below the low-level scram setpoint), the liquid inventory between the two level setpoints is approximately 30,000 lbm. Based on this and the ANS-5 decay heat, it was estimated that it would take about one hour (after scram) for the water level to drop to the isolation setpoint due to loss of feedwater and CRD. Therefore, the operator has at least one hour to manually isolate the reactor before it isolates automatically.

6. It is not apparent that additional failure in shutdown methods would not aggravate or change the course of a simulated transient as stated in NEDO-24708. Clarify.

Response to Question 6

The event descriptions have been modified in Section 3.2.1.1. The shutdown process is addressed in the Emergency Procedure Guidelines.

7. For BWR/1 with no emergency condenser (EC) or CRD flow, provide the system response when the SRV recloses instead of remaining open. The pressure will rise again to the SRV setpoint and continue this cycling at high pressure while inventory is being depleted. If manual action is required, provide the instrumentation available to alert the operator and what actions are required to maintain acceptable core inventory.

Response to Question 7

This condition is covered in Section 3.2.1.3.3.

8. For the BWR classes where the SRV cycle before decay heat is removed by ECCS, what happens to the vessel inventory. Provide plots of level, pressure, ECCS, and SRV flows.

Response to Question 8

These cases have been analyzed in Section 3.2.1.1.5. The corresponding curves are provided.

9. It appears that a stuck open relief valve (SORV) combined with a LOFW and failure of high pressure systems is not as severe as a properly operating SRV or one that is partially stuck open. In determining the course of a LOFW transient a sensitivity study should be performed for determining operator action times for event recognition and proper mitigation.

Response to Question 9

These cases have been analyzed in Section 3.2.1.1.5.

10. Justify the assumptions used in the analyses to show operator action times as provided in the sequence of events. For example, justify the selection used for decay heat which varied for reactor class. How sensitive is the analysis to your assumptions.

Response to Question 10

The assumptions adopted for the analyses of Section 3.2.1.1 and the associated rationales have been provided in Section 3.2.1.1.4. Sensitivity studies have been done in Section 3.2.1.3.2. These results show that the guidelines are fundamentally insensitive to the variation of system parameters and are solely dependent on the actual experienced sequence of events.

11. It appears that operationally it is desired to manually restart a failed high pressure system prior to using the automatic depressurization system (ADS) for the low pressure (LPCI/LPCS) ECCS. However, the core inventory recovery is faster with ADS (no high pressure ECCS) and LPCI/LPCS. What will the guidelines suggest to the operator?

Response to Question 11

This area is specifically addressed in the Emergency Procedure Guidelines.

12. Supply curves to show the differences in SRV opening times and level recovery times for BWR/4 and BWR/5 reactors.

Response to Question 12

See response to Question 3.

13. Provide the analyses and sequence of events for the LOFW coupled with a stuck open SRV and the following: loss of offsite power; loss of all A-C power; and loss of one train of D-C power with loss of offsite power. Provide the following time-dependent variables: SRV flow; vessel pressure; ECCS flows; vessel water level; and fuel cladding, and coolant temperatures. The initial conditions assumed in the analyses should be provided and the time at which stable conditions are reached. If core uncovery results, provide the basis for assessing core damage (duration, extent).

Response to Question 13

The effect of power source availability on the LOF event has been discussed in Section 3.2.1.3.2.4.

References:

(1) Letter, D. F. Ross to T. D. Keenan, **Additional Information Required to Evaluate NEDO-24708**, September 28, 1979.

(2) Letter, R. H. Buchholz to D. F. Ross, **Additional Information on Loss of Feedwater Transient Analysis**, December 28, 1979.

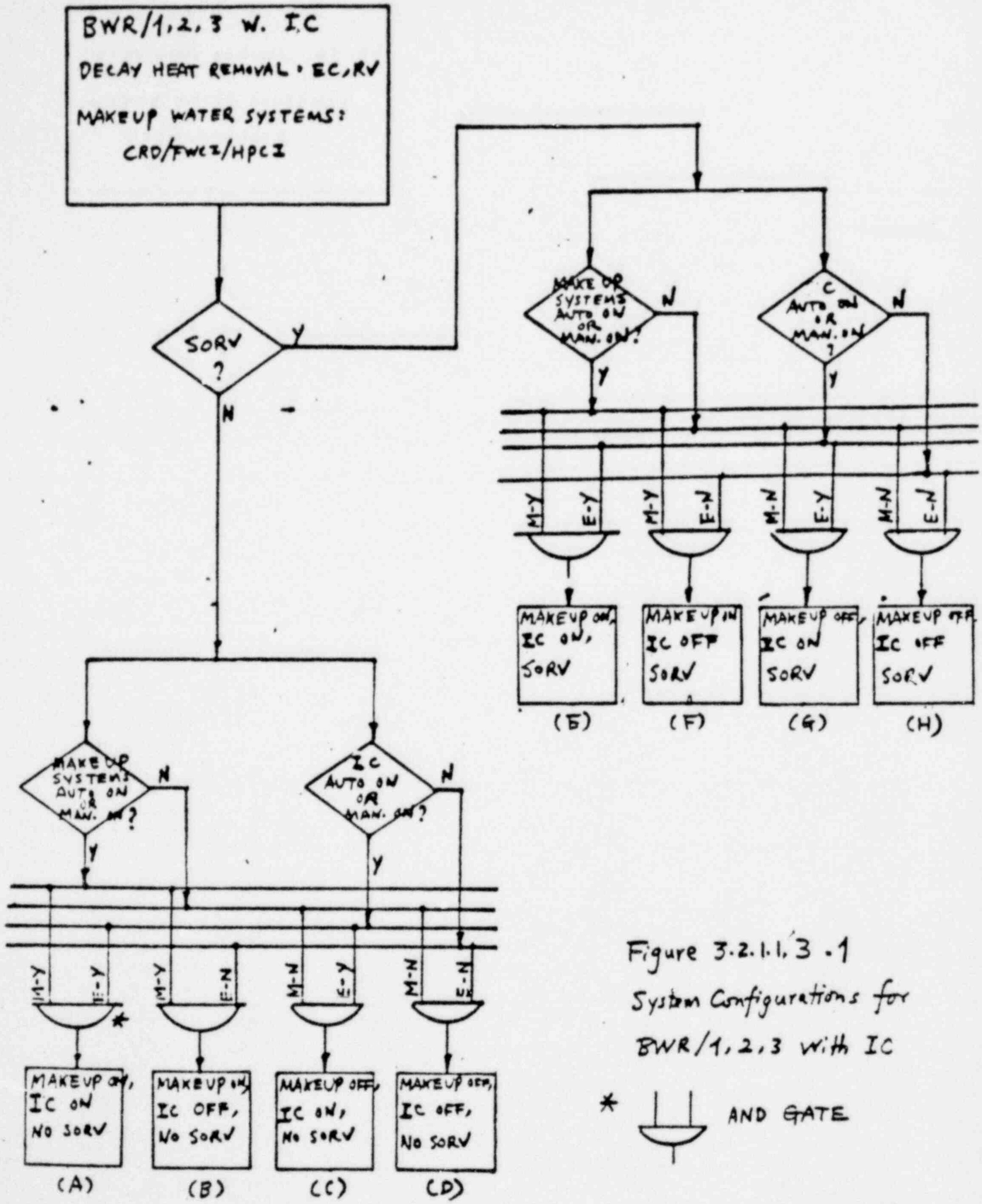


Figure 3.2.1.1.3-1
System Configurations for BWR/1,2,3 with IC

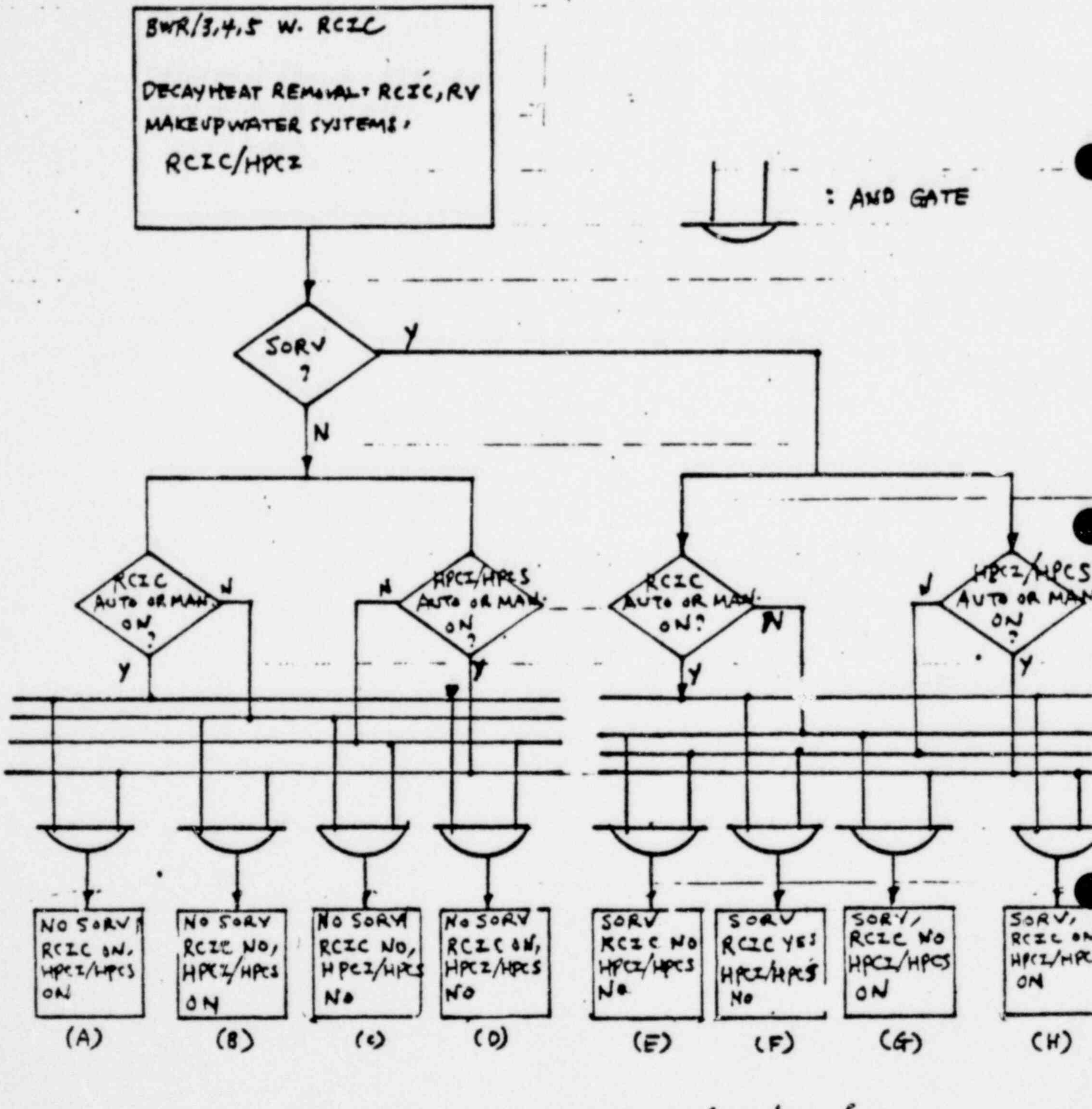
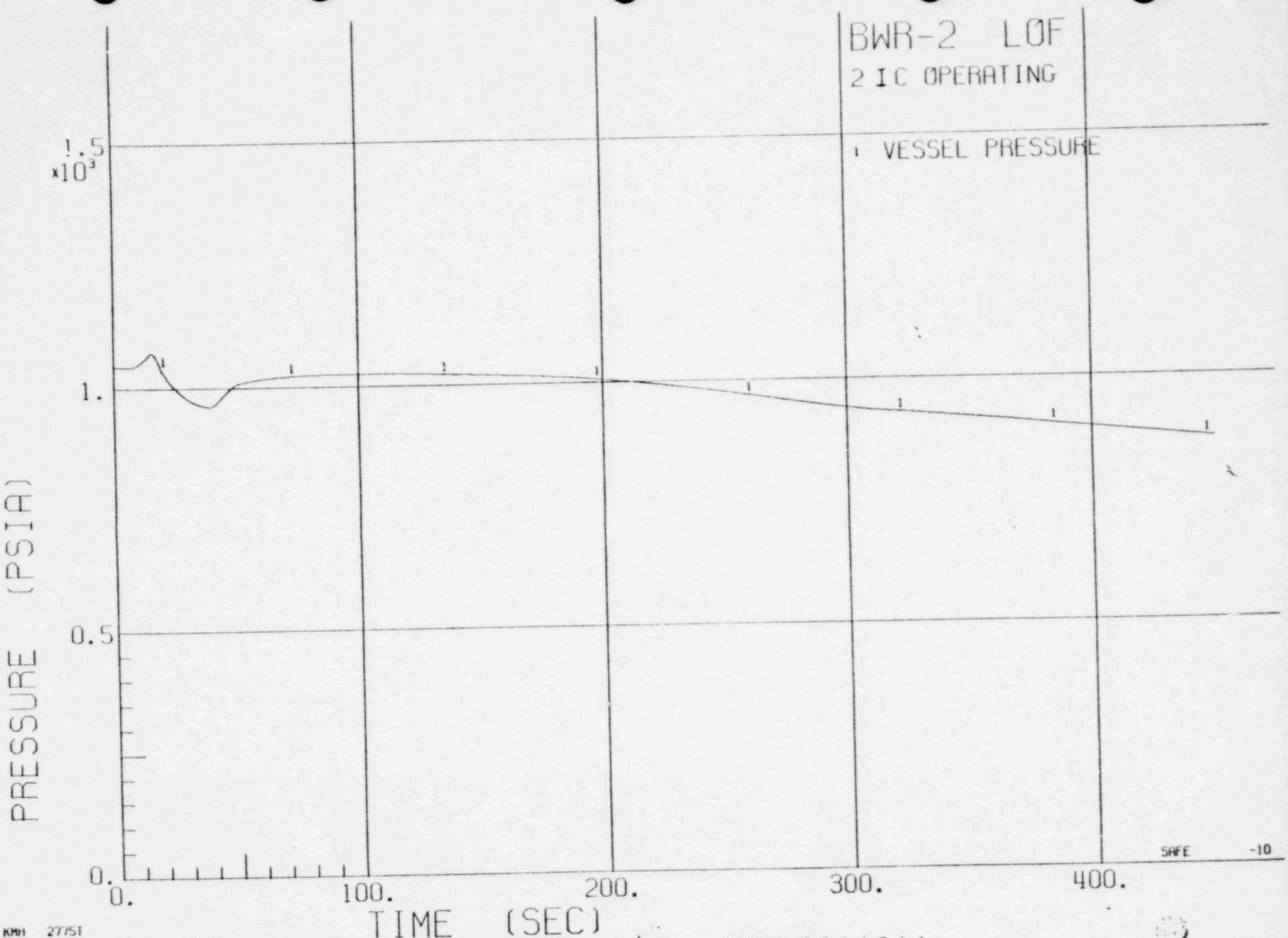
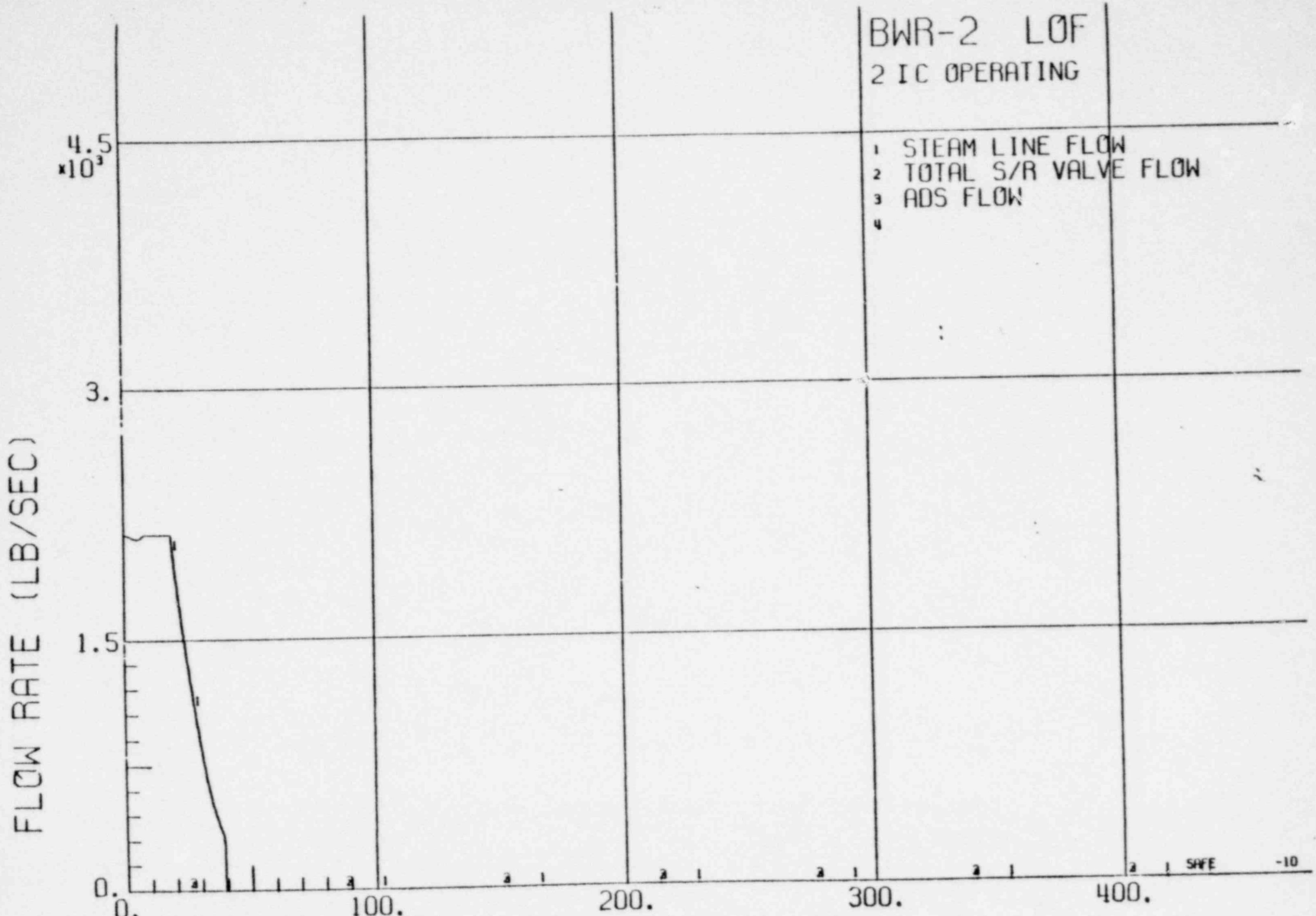


Figure 3.2.1.1.3 · 2 System Configurations for BWR/3,4,5 with RCIC



KMH 2775T
120579 0604.4

FIGURE 3.2.1.1.5-1.1



KMH 2775T

FIGURE 3.2.1.1.5-1.2

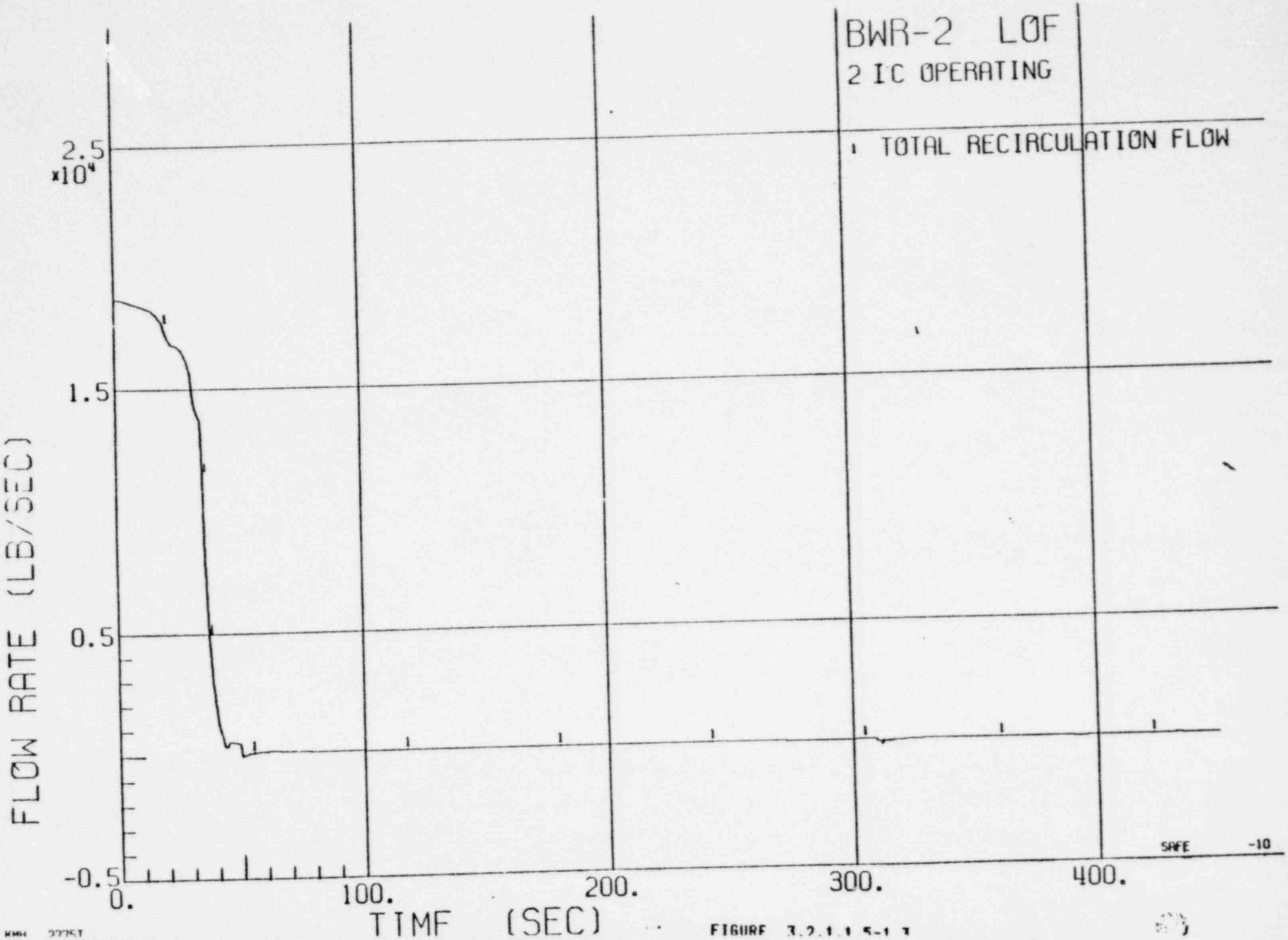


FIGURE 3.2.1.1.5-1.3

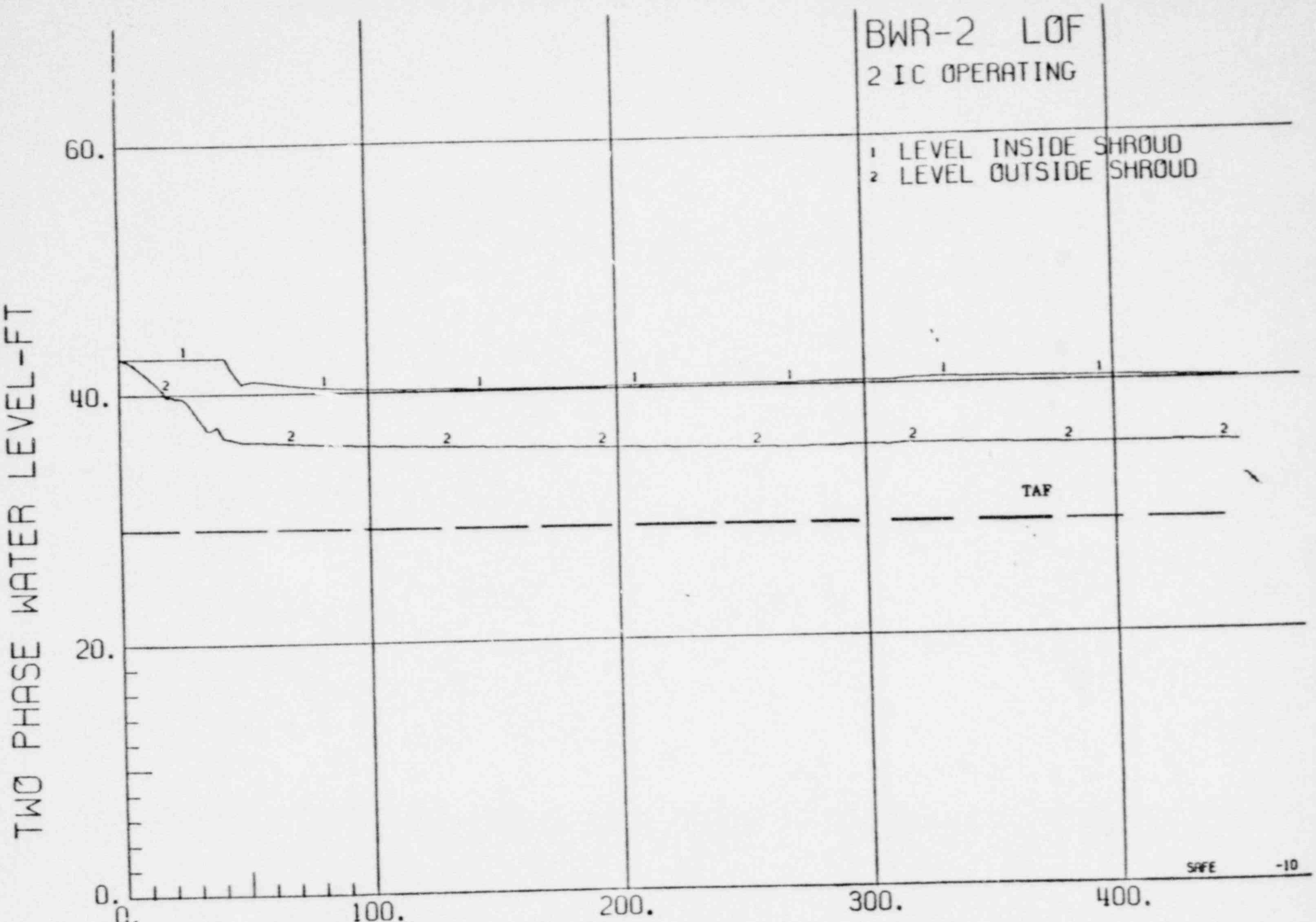
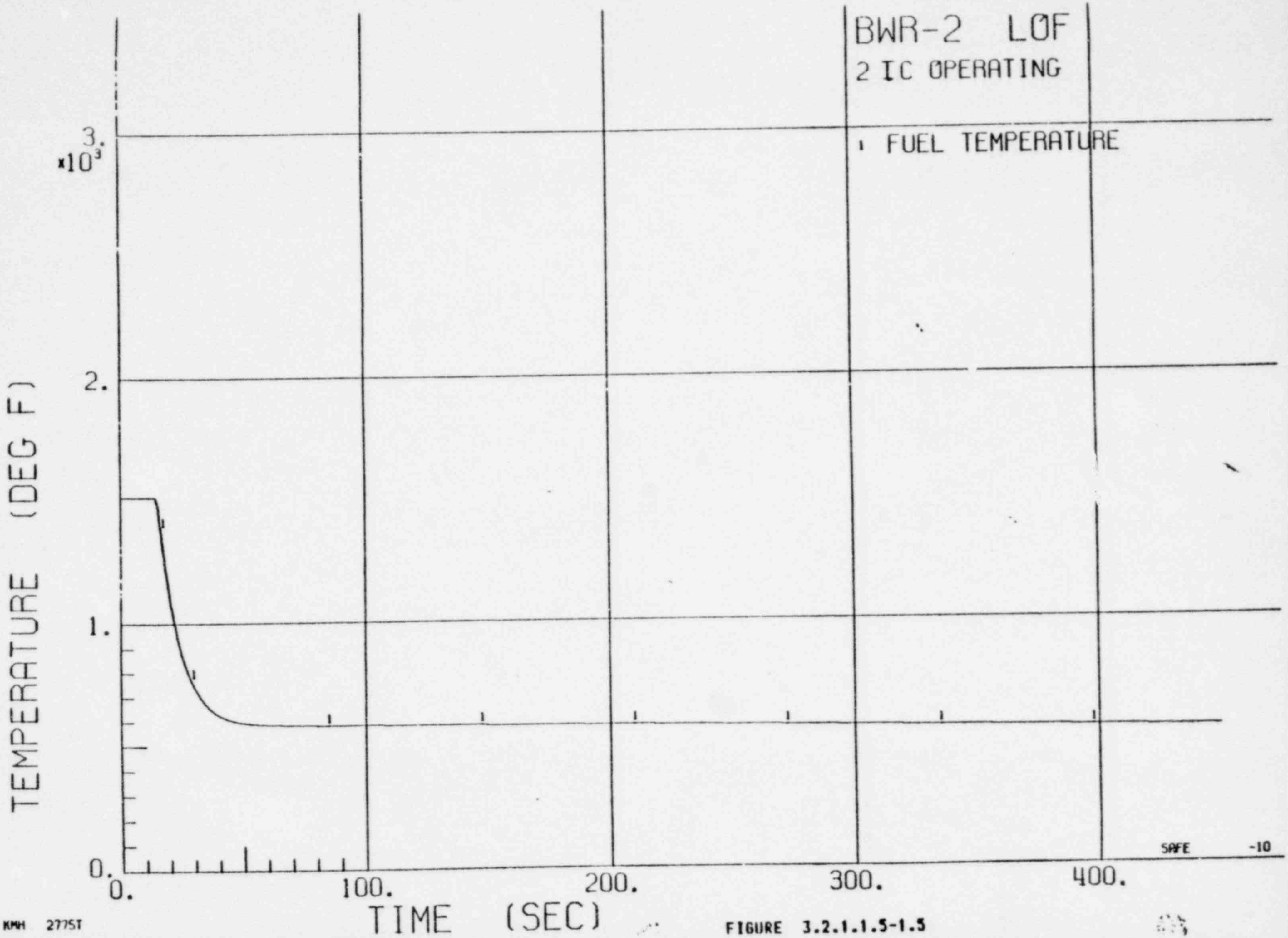


FIGURE 3.2.1.1.5-1.4

KMH 2775T



KMH 2775T

FIGURE 3.2.1.1.5-1.5

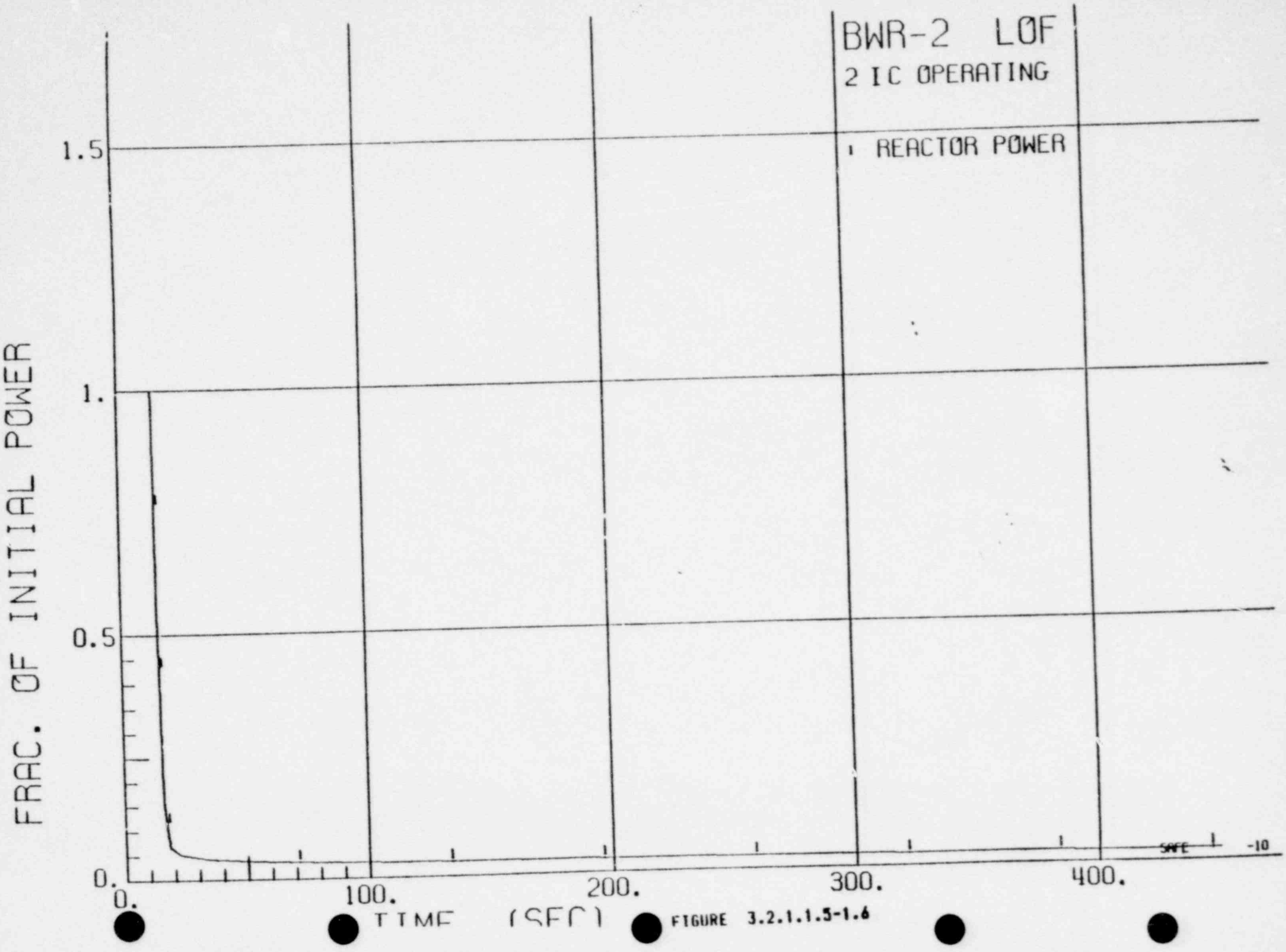


FIGURE 3.2.1.1.5-1.6

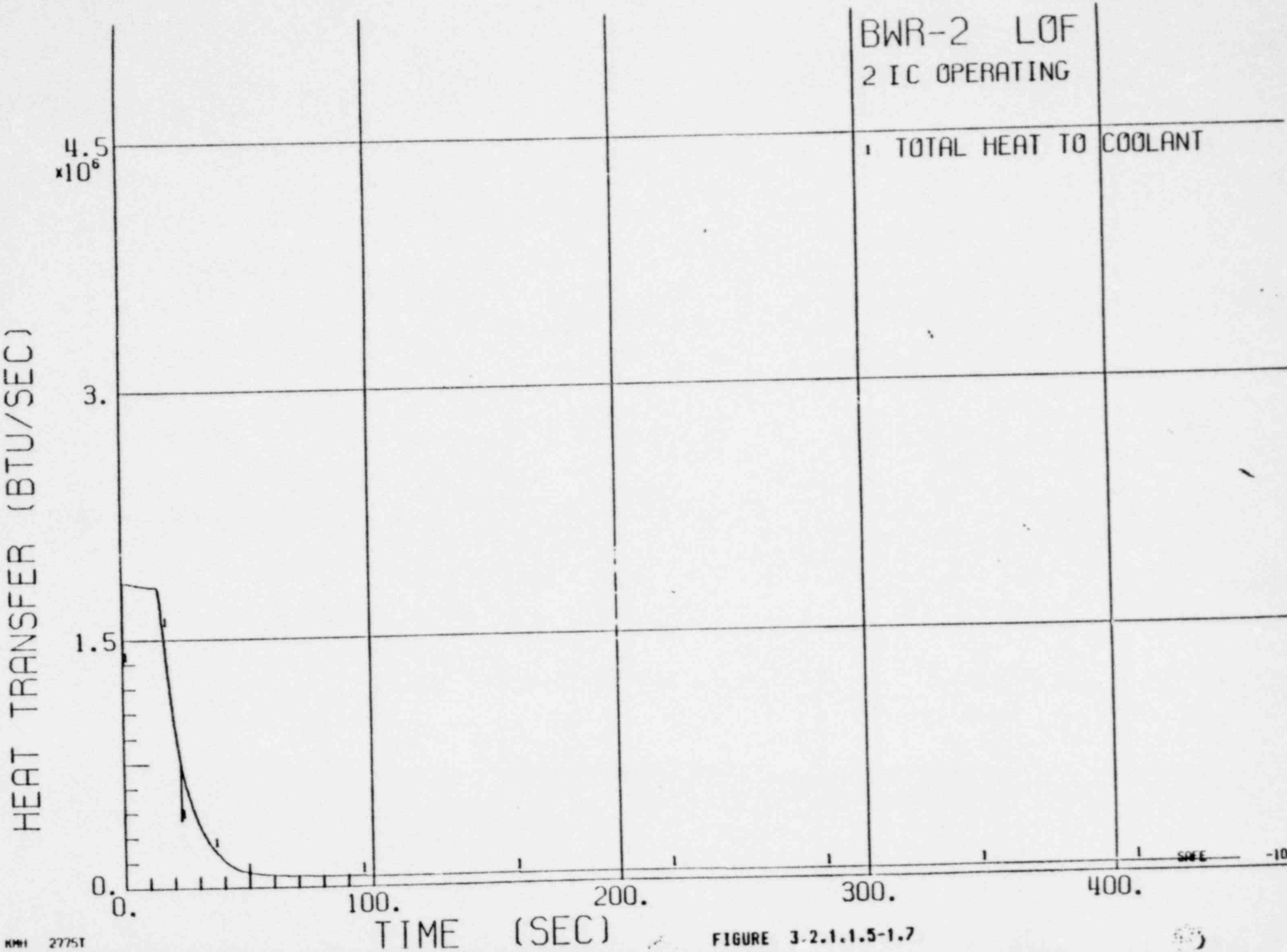


FIGURE 3.2.1.1.5-1.7

KMH 2775T

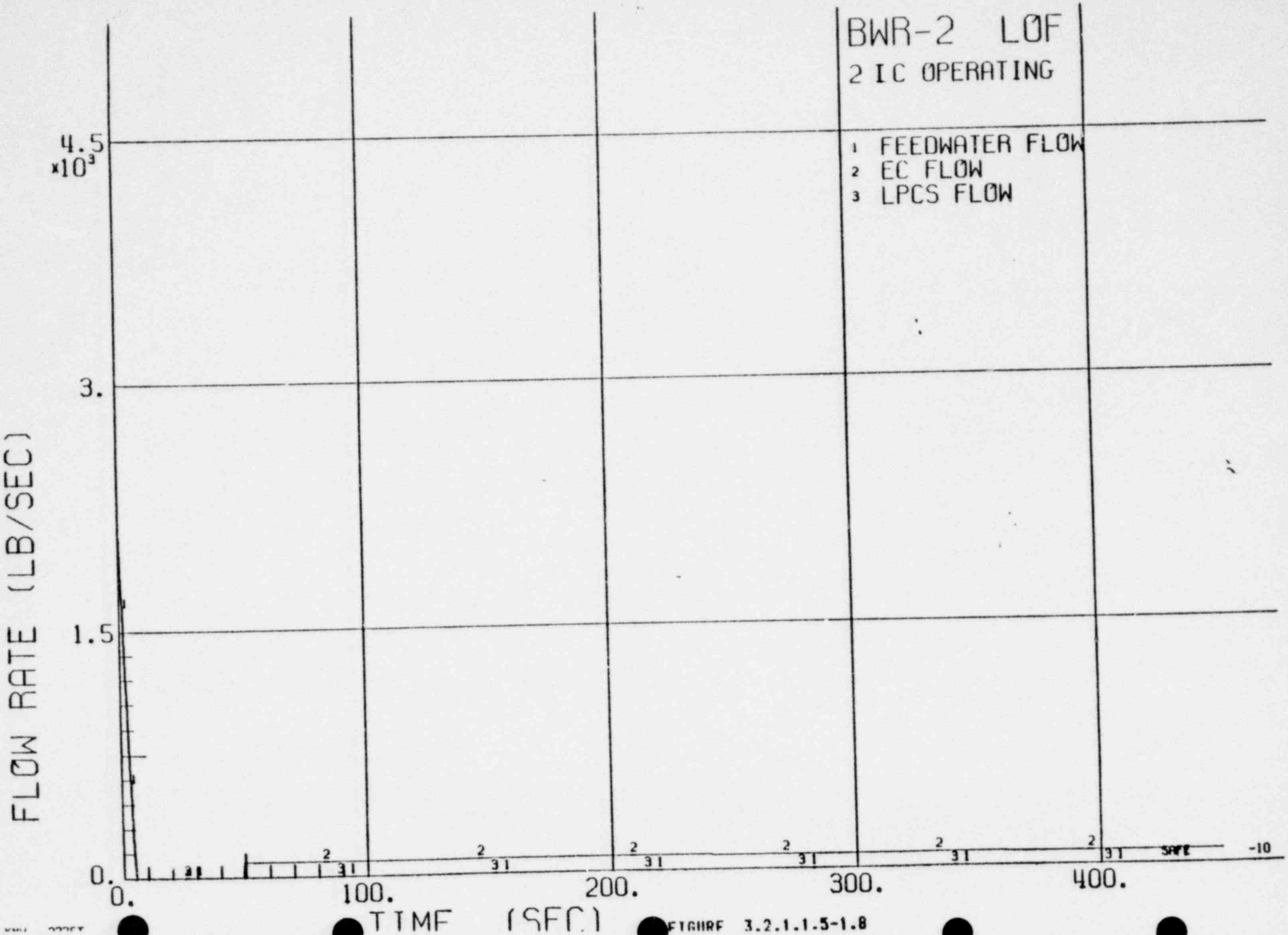
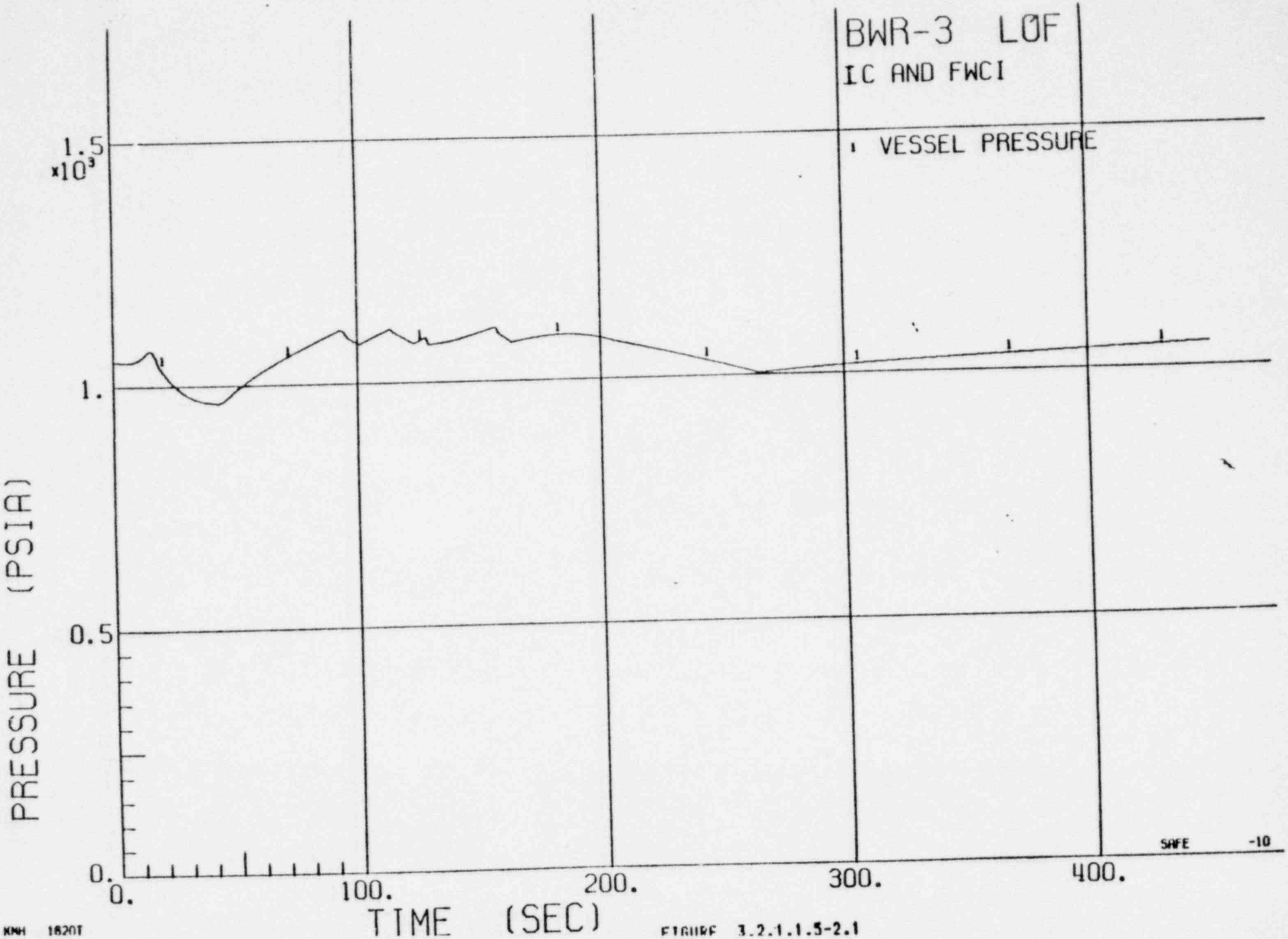
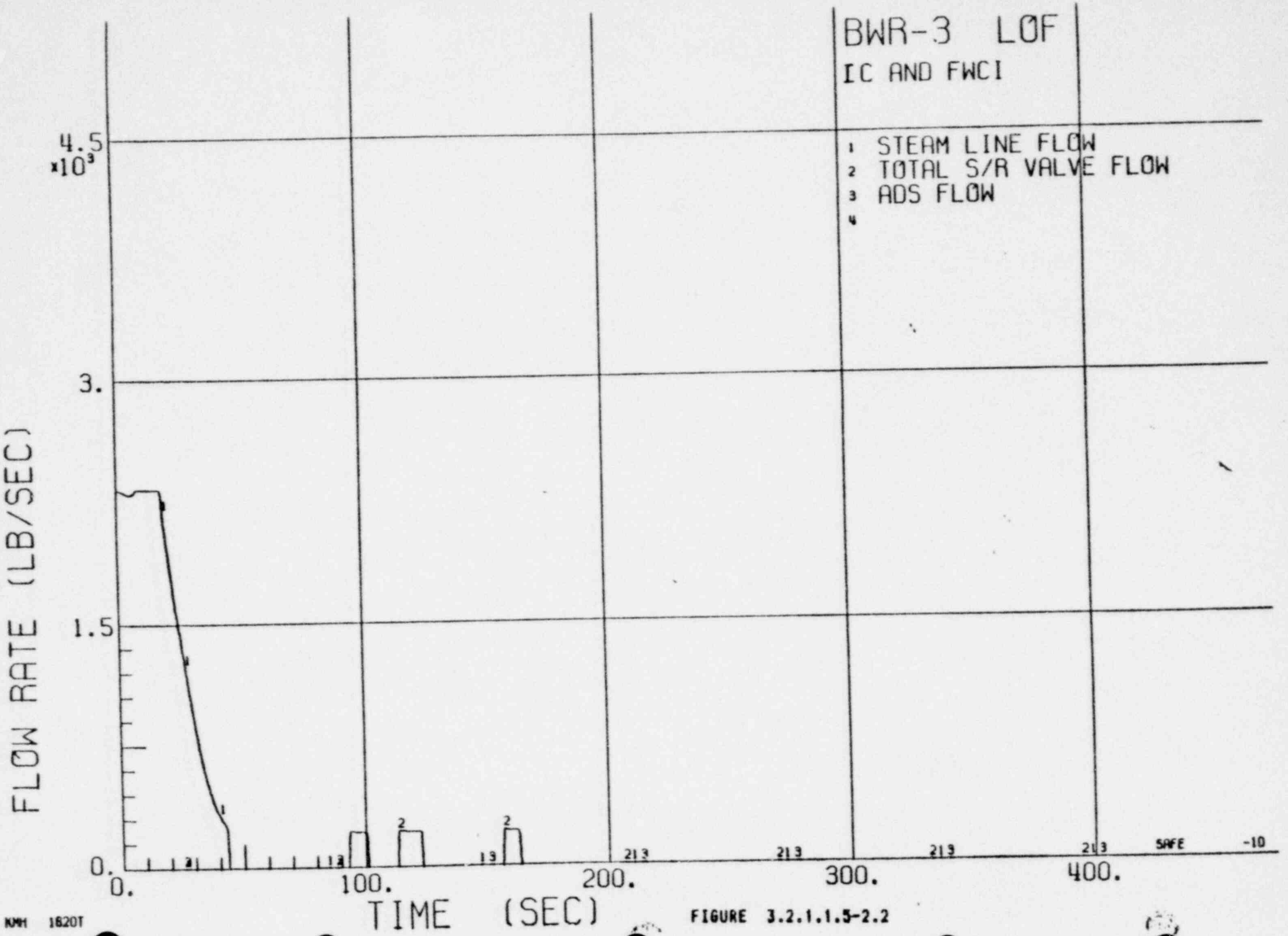


FIGURE 3.2.1.1.5-1.8



KNH 1820T

FIGURE 3.2.1.1.5-2.1



KMH 1620T

FIGURE 3.2.1.1.5-2.2

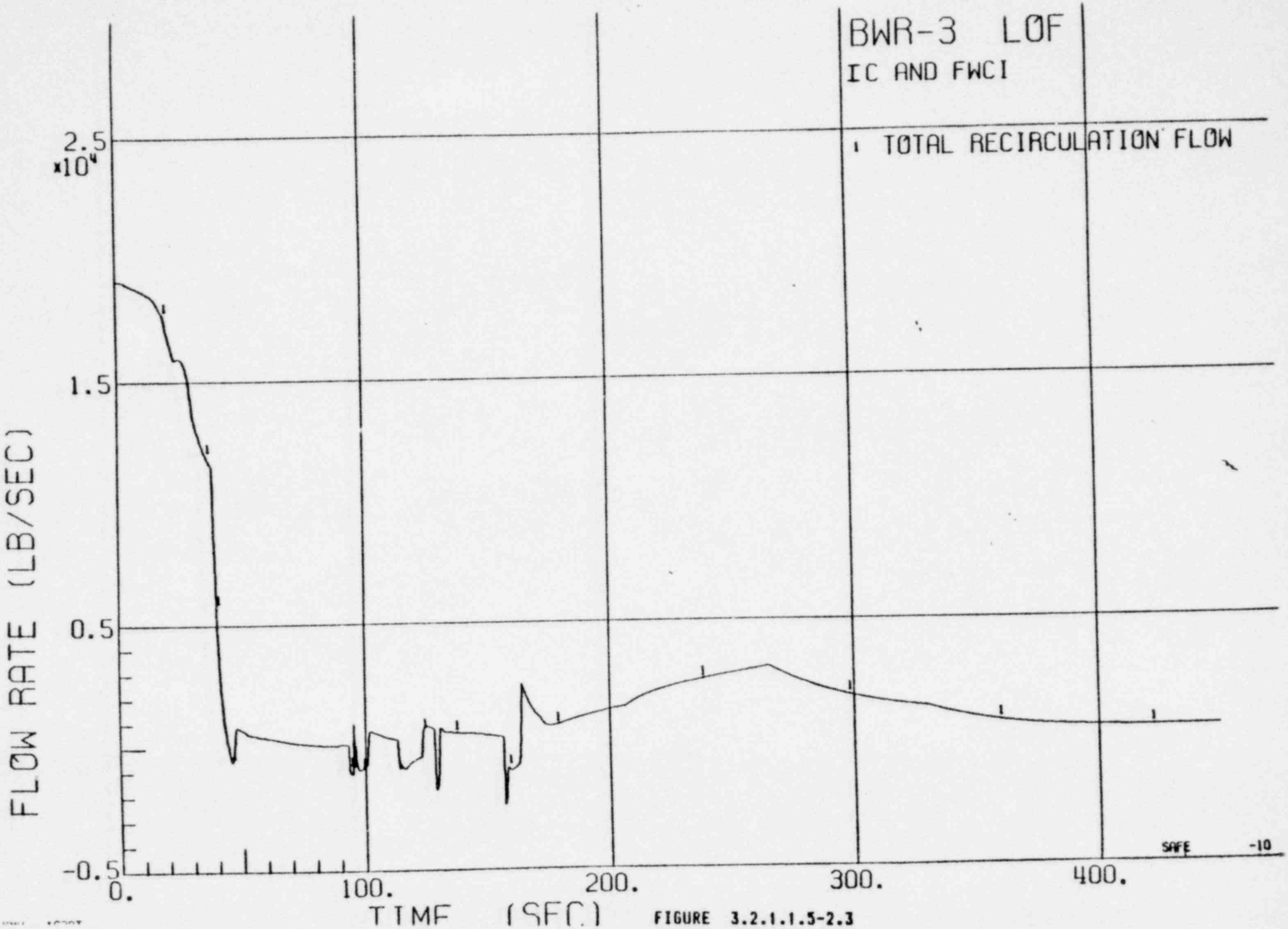
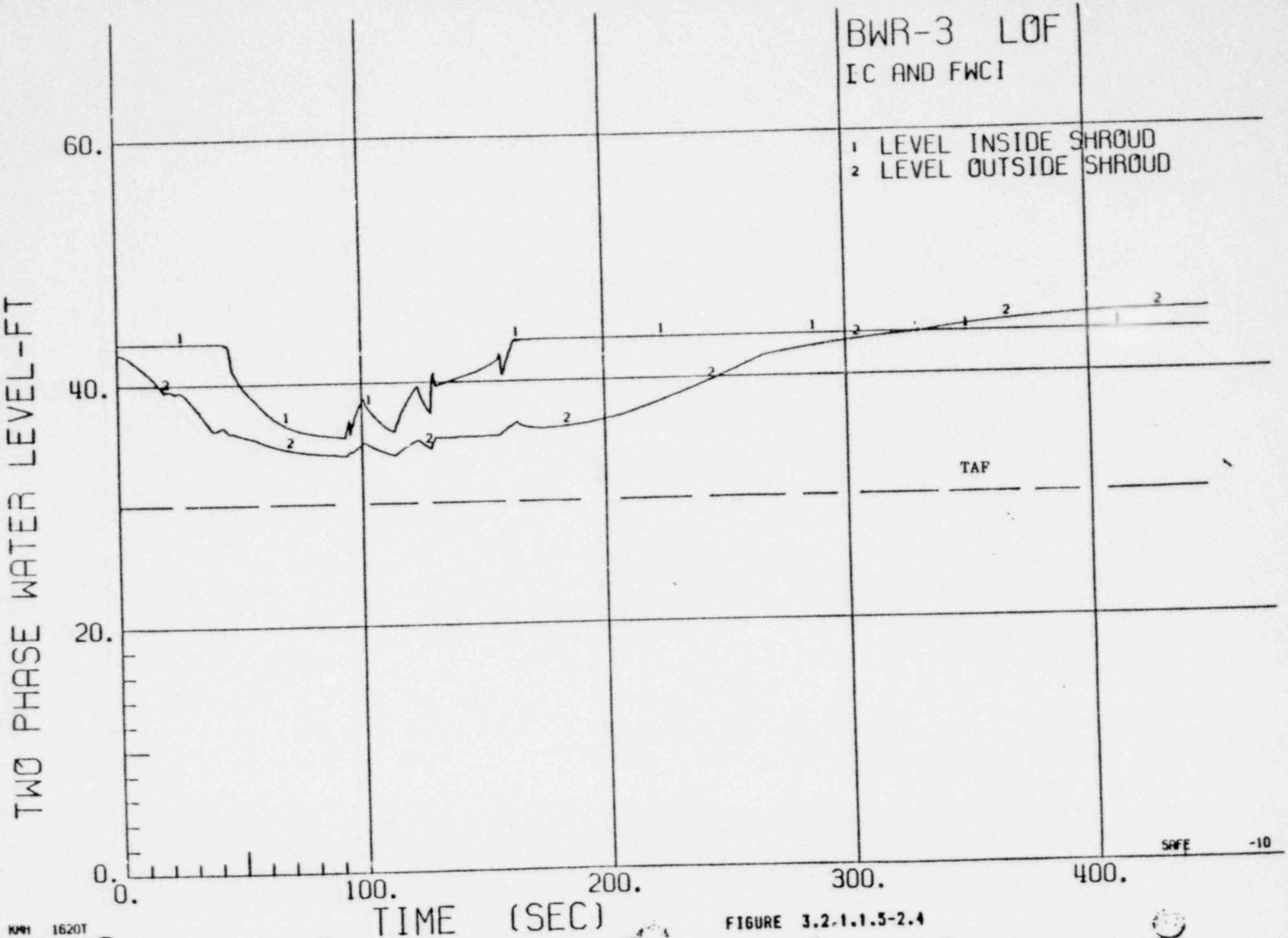
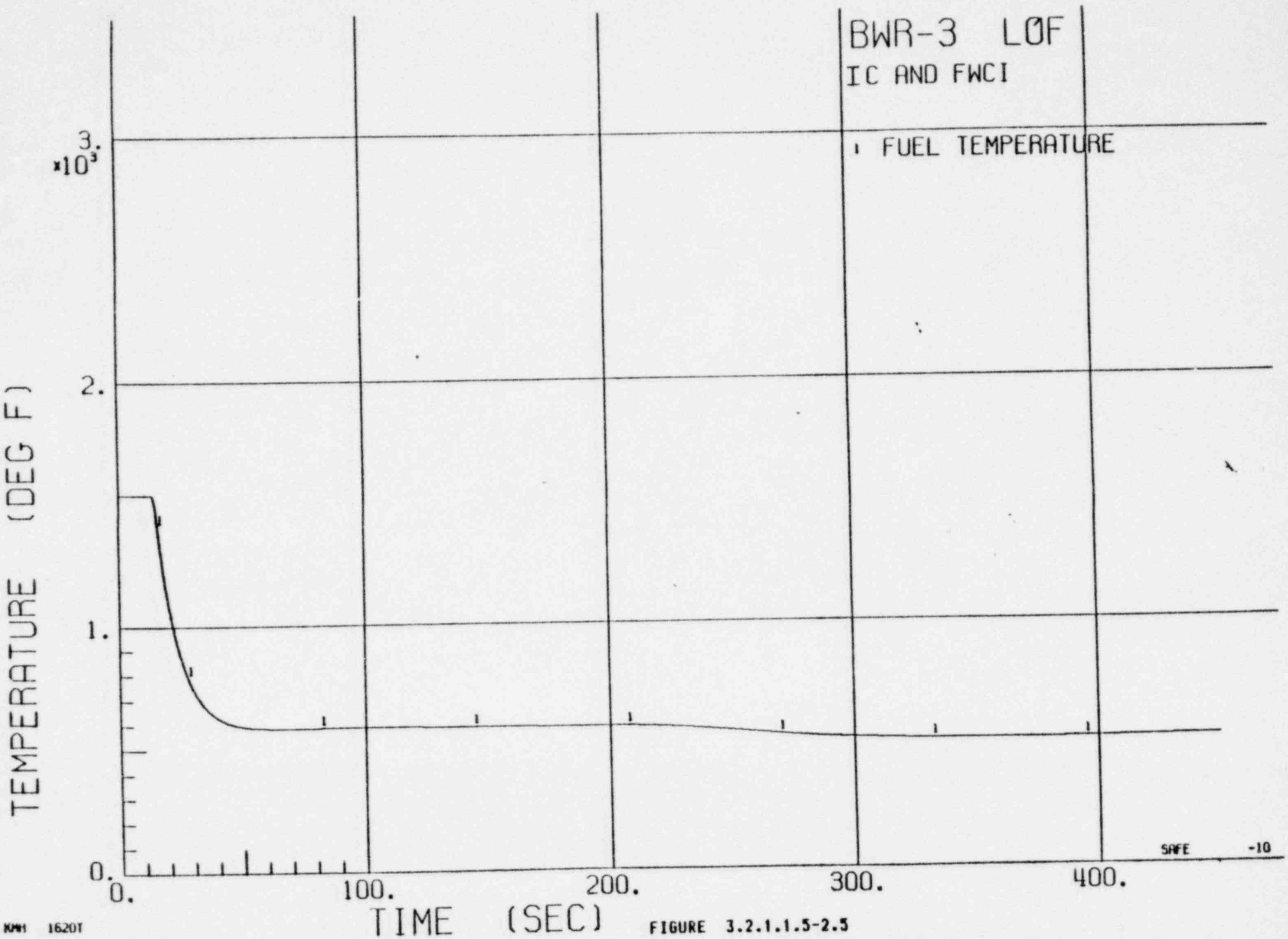


FIGURE 3.2.1.1.5-2.3



KMH 1620T

FIGURE 3.2.1.1.5-2.4



KMH 1620T

FIGURE 3.2.1.1.5-2.5

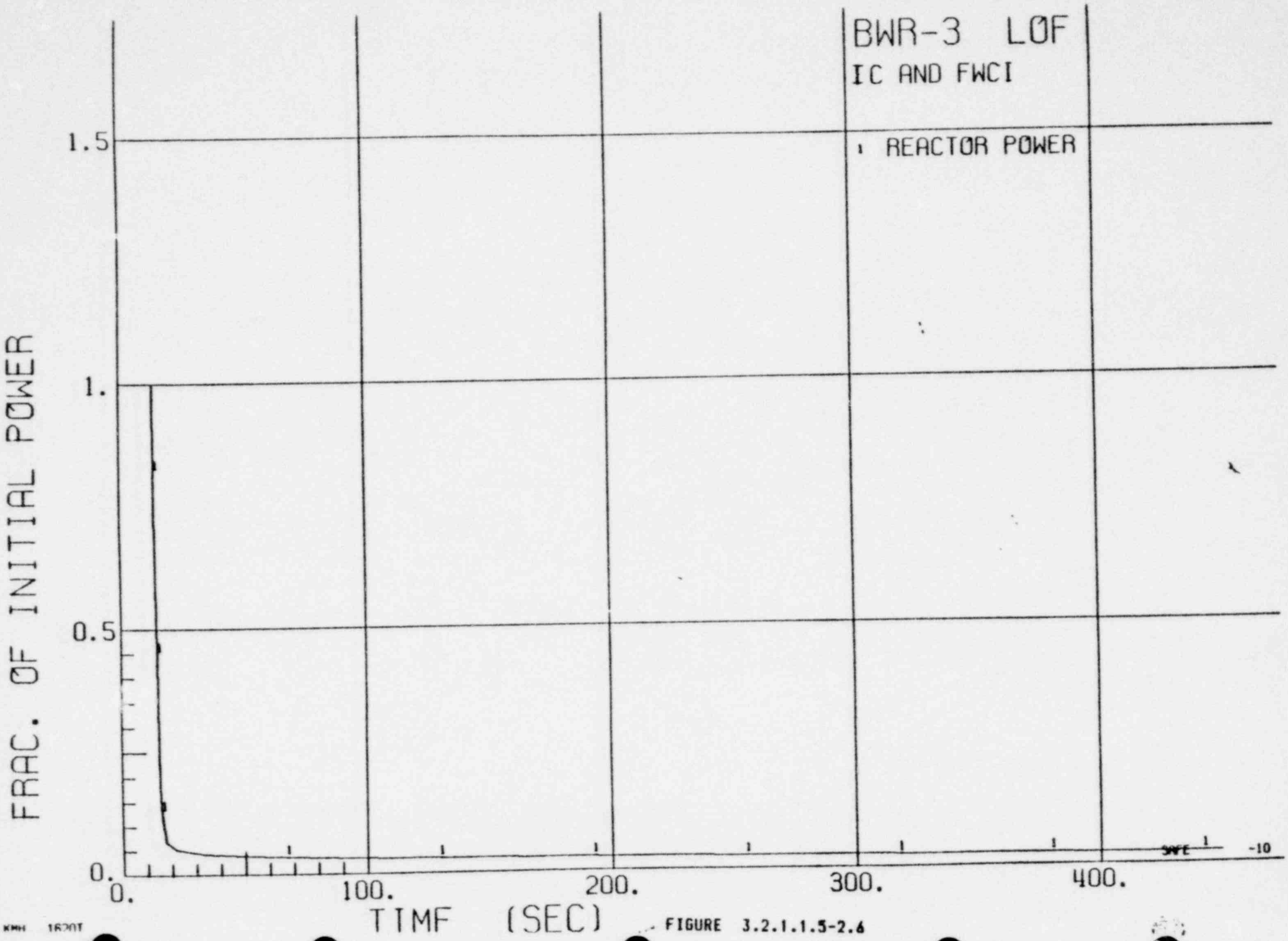
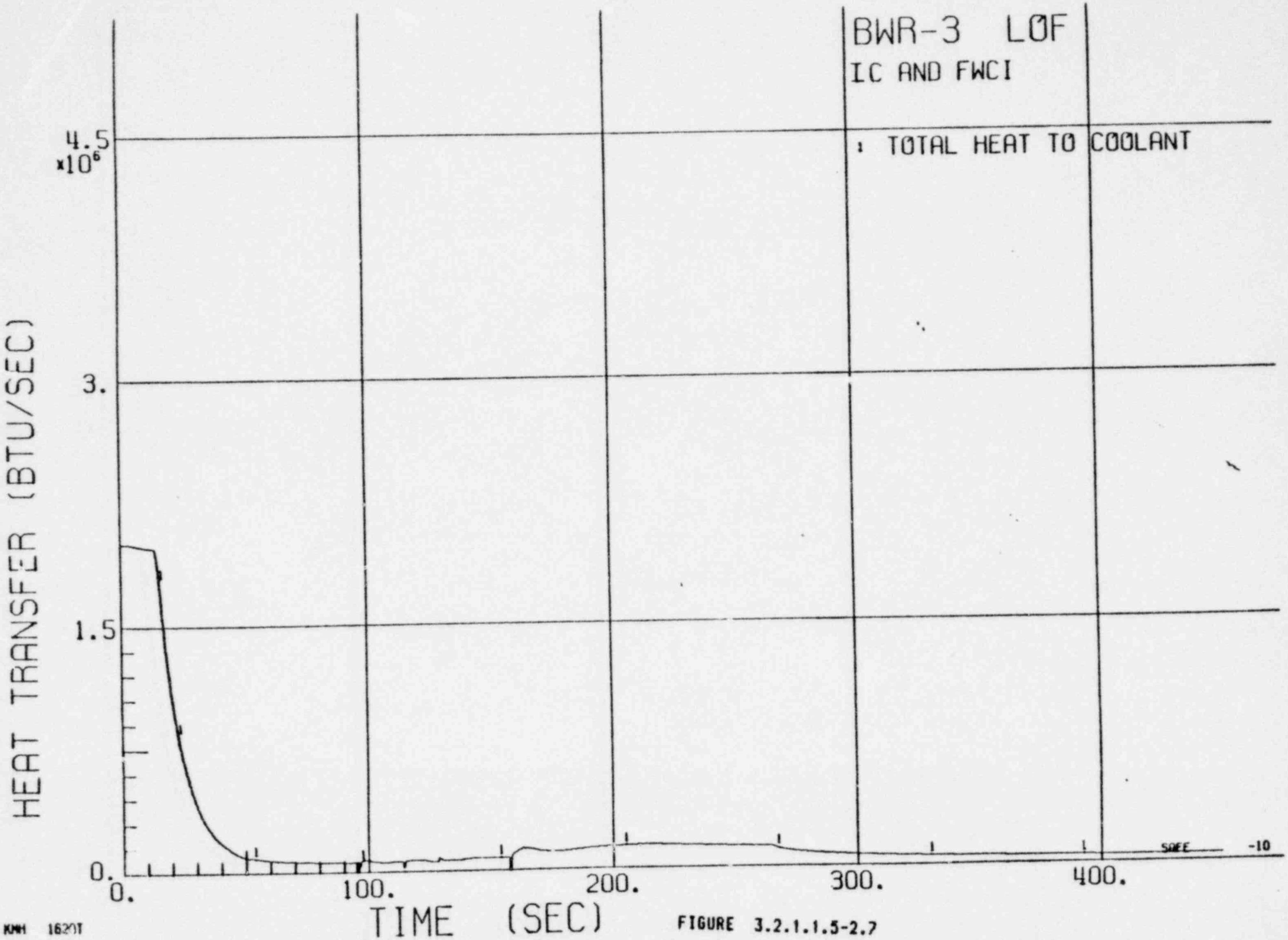


FIGURE 3.2.1.1.5-2.6

KMH 1620T



KNH 1620T

FIGURE 3.2.1.1.5-2.7

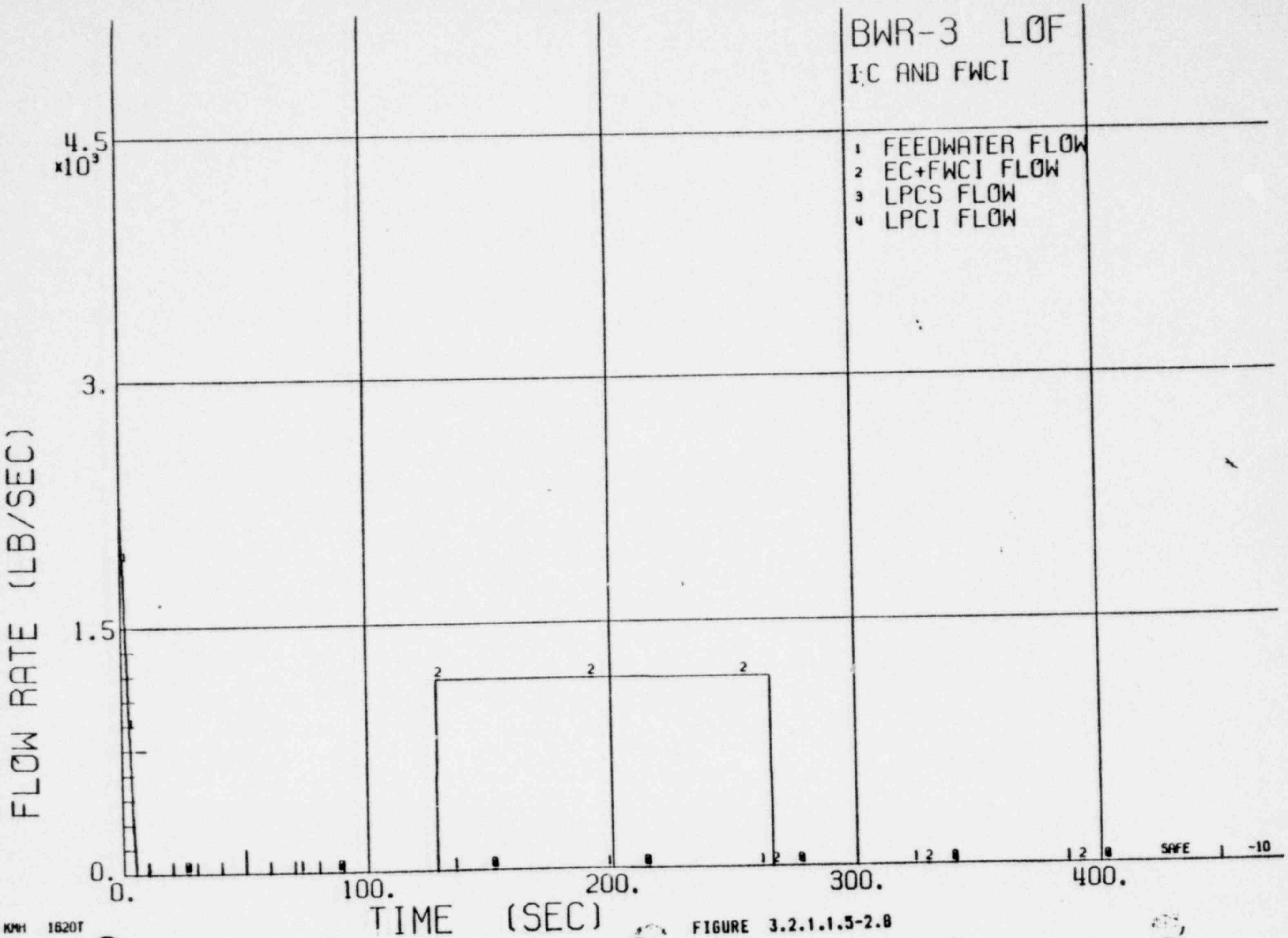


FIGURE 3.2.1.1.5-2.8

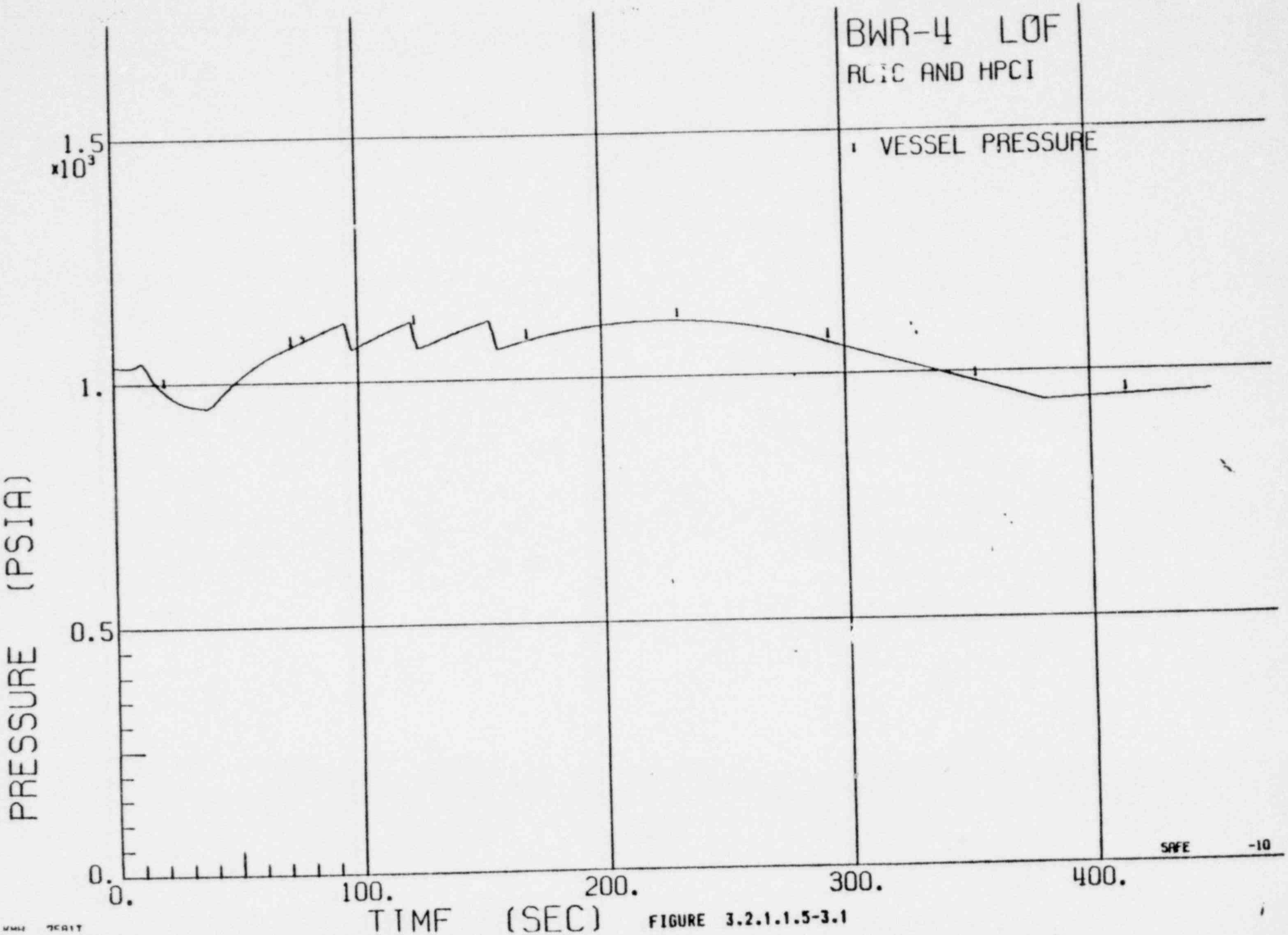
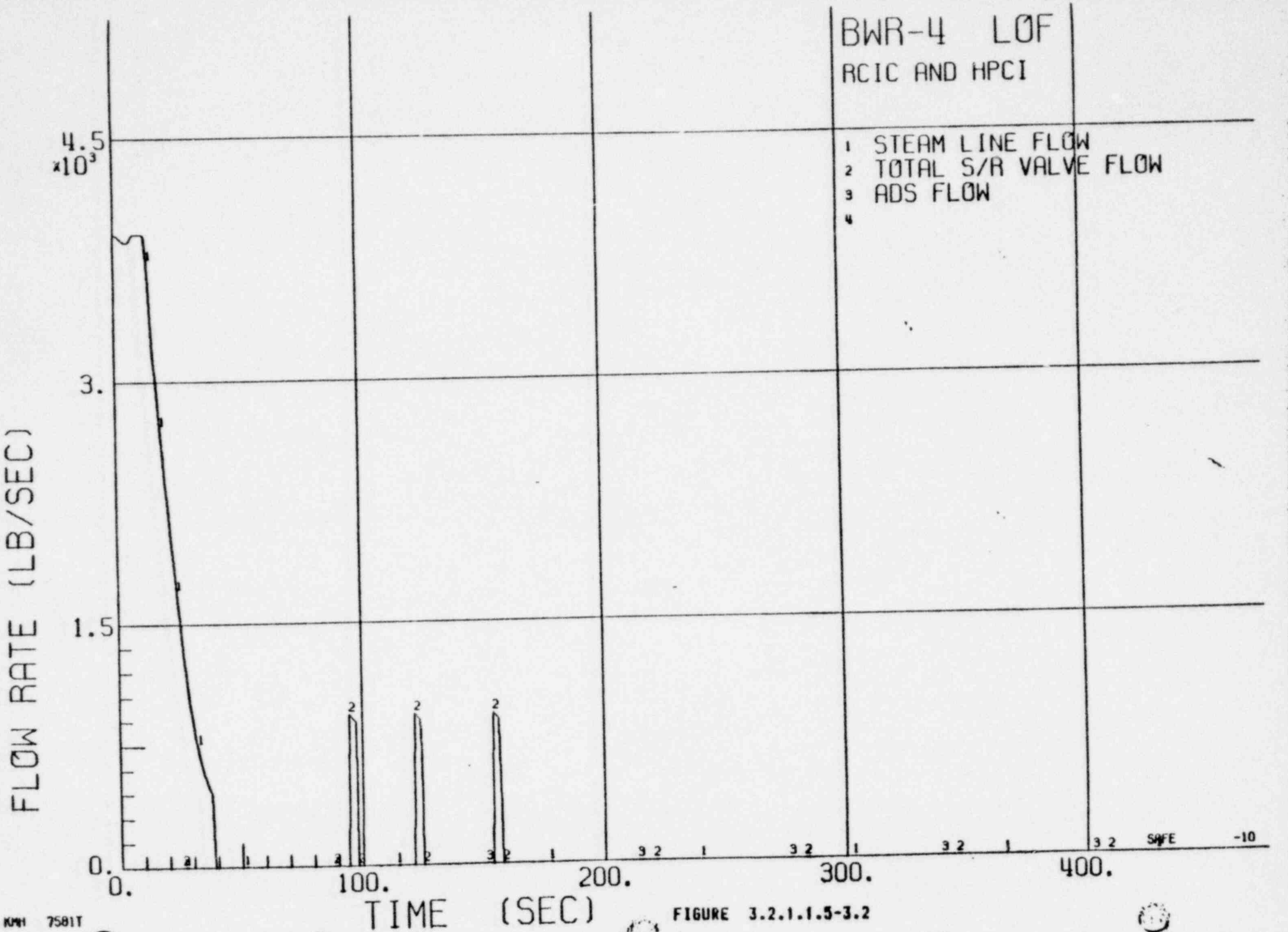


FIGURE 3.2.1.1.5-3.1



KMH 7581T

FIGURE 3.2.1.1.5-3.2

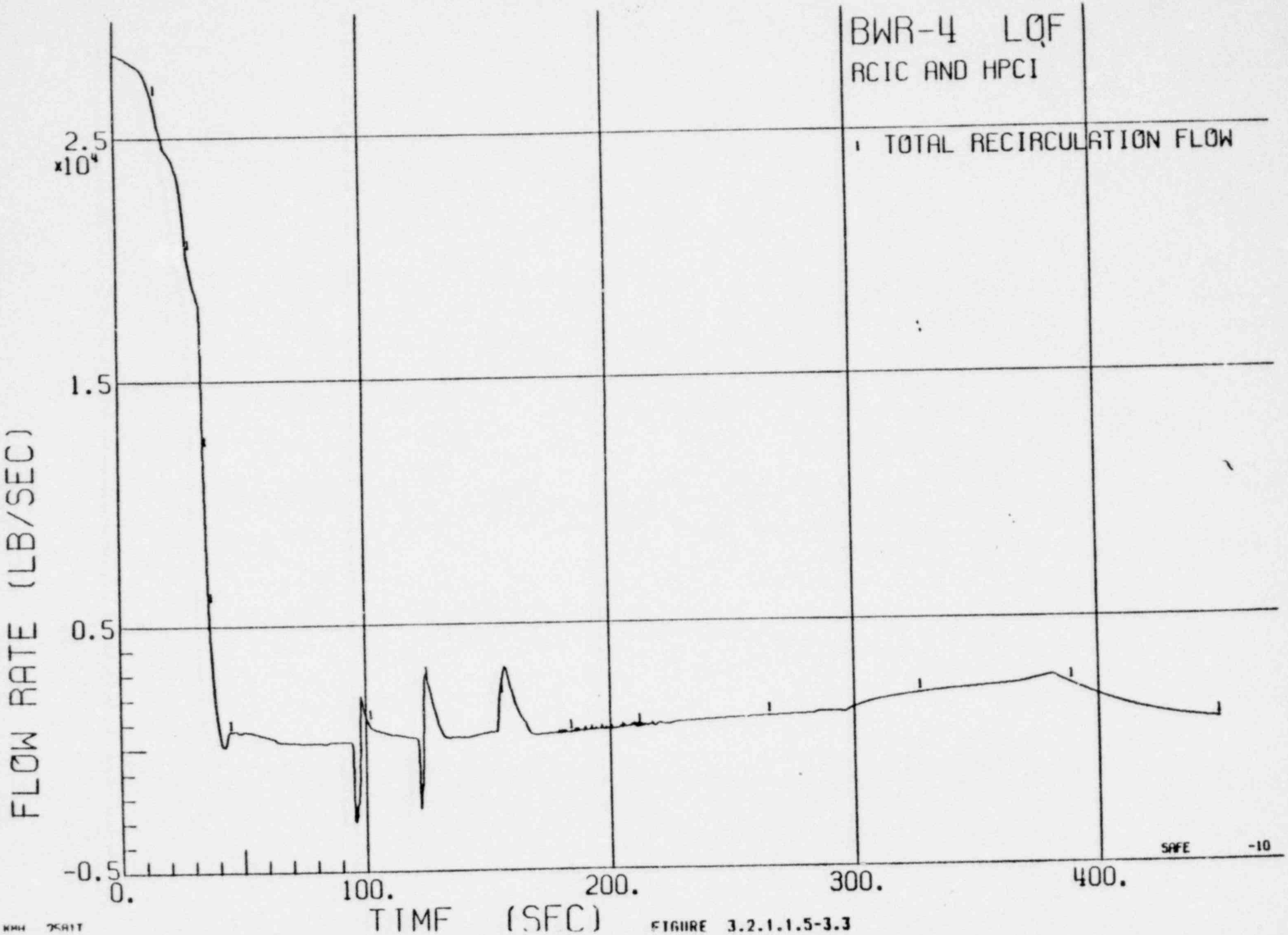
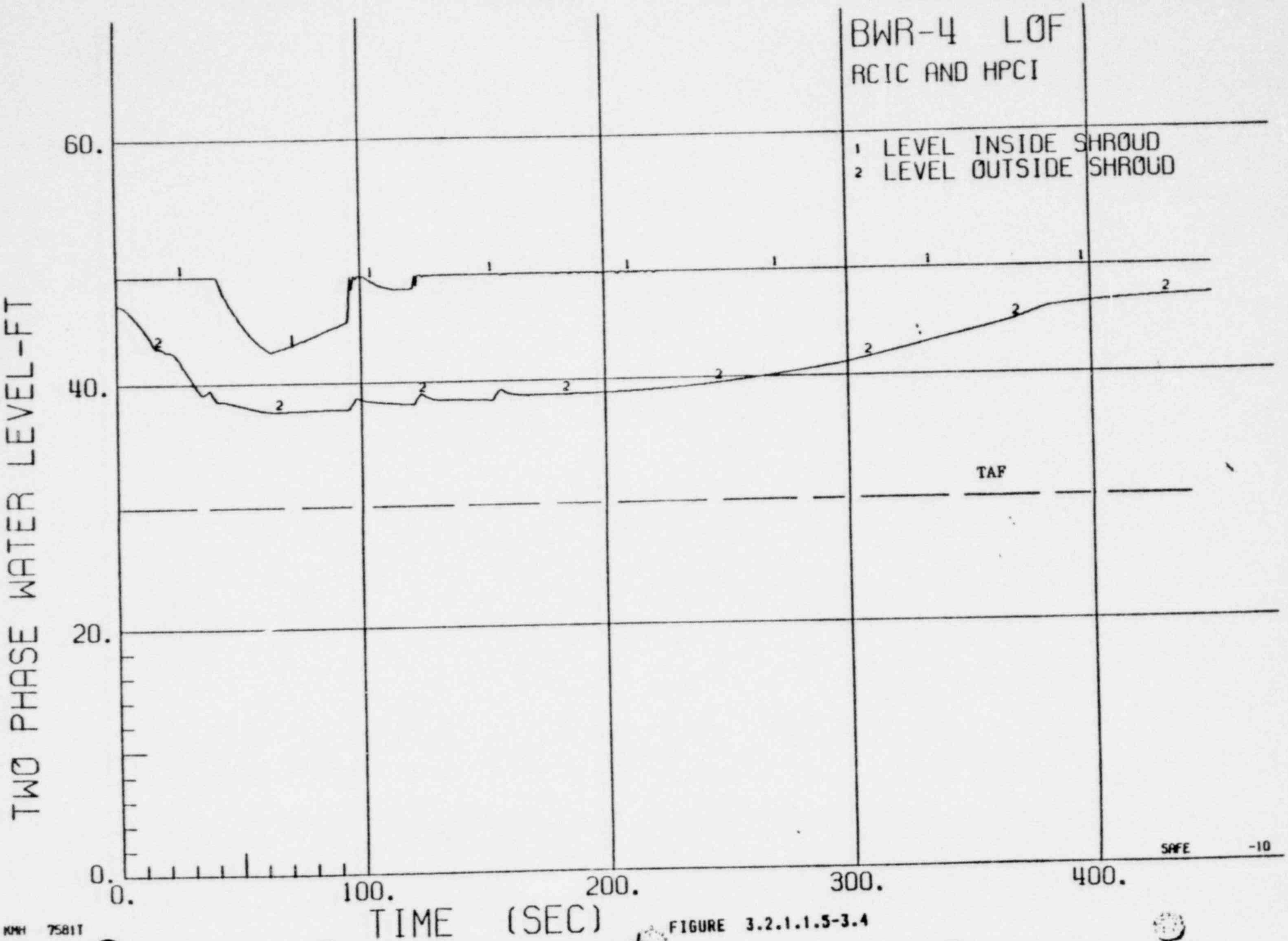


FIGURE 3.2.1.1.5-3.3



KMH 7581T

FIGURE 3.2.1.1.5-3.4

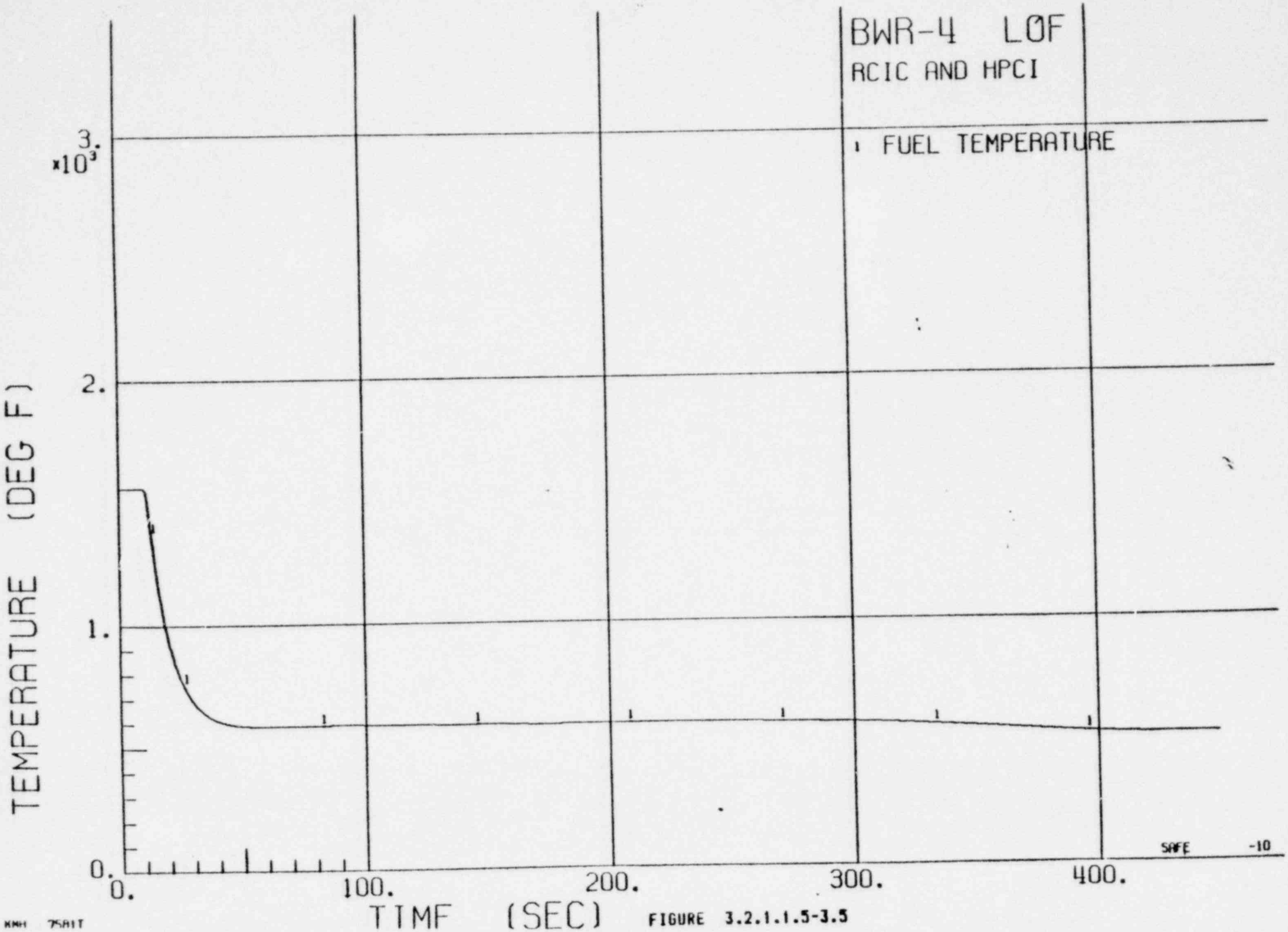


FIGURE 3.2.1.1.5-3.5

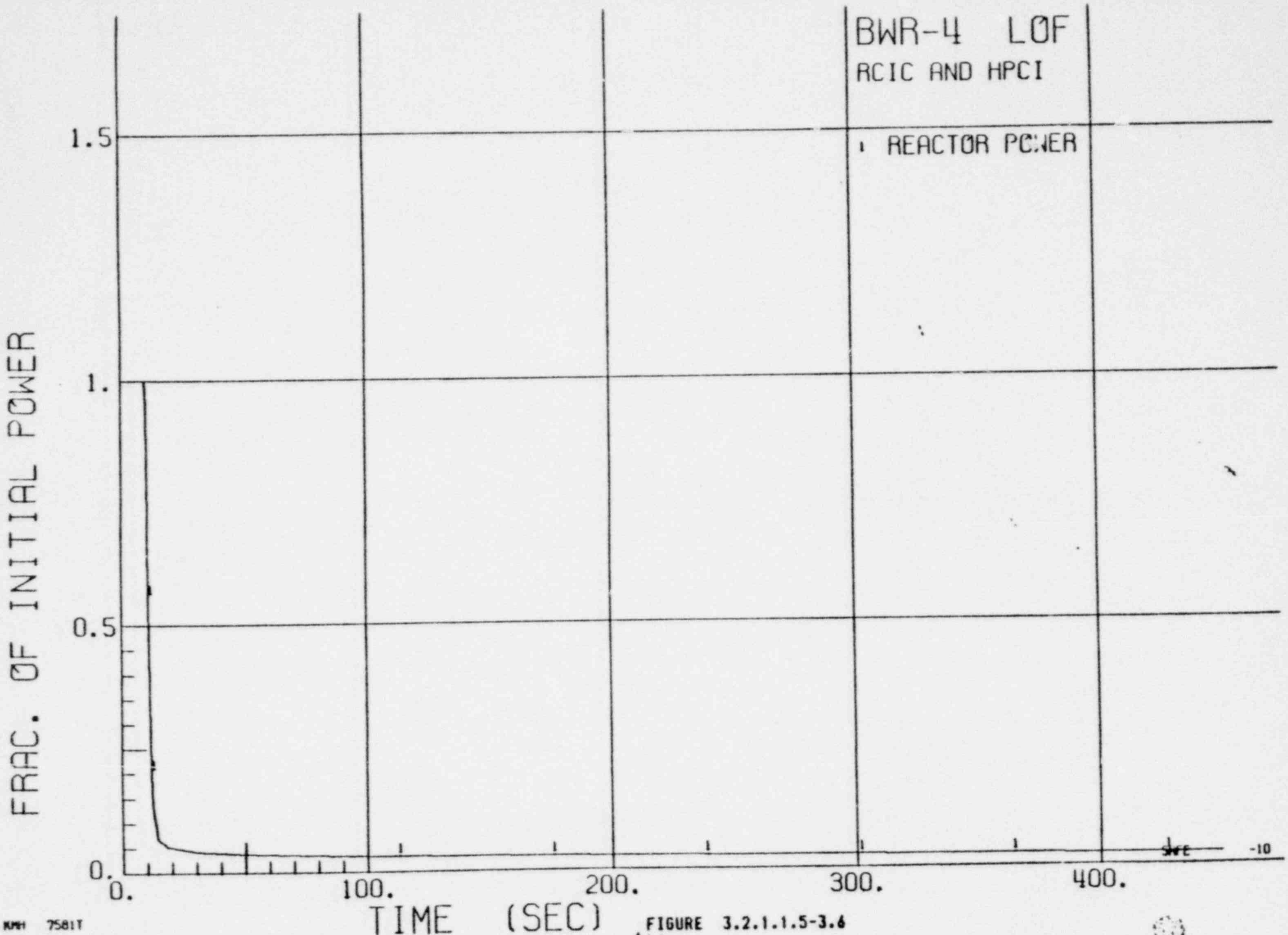


FIGURE 3.2.1.1.5-3.6

KMH 7581T

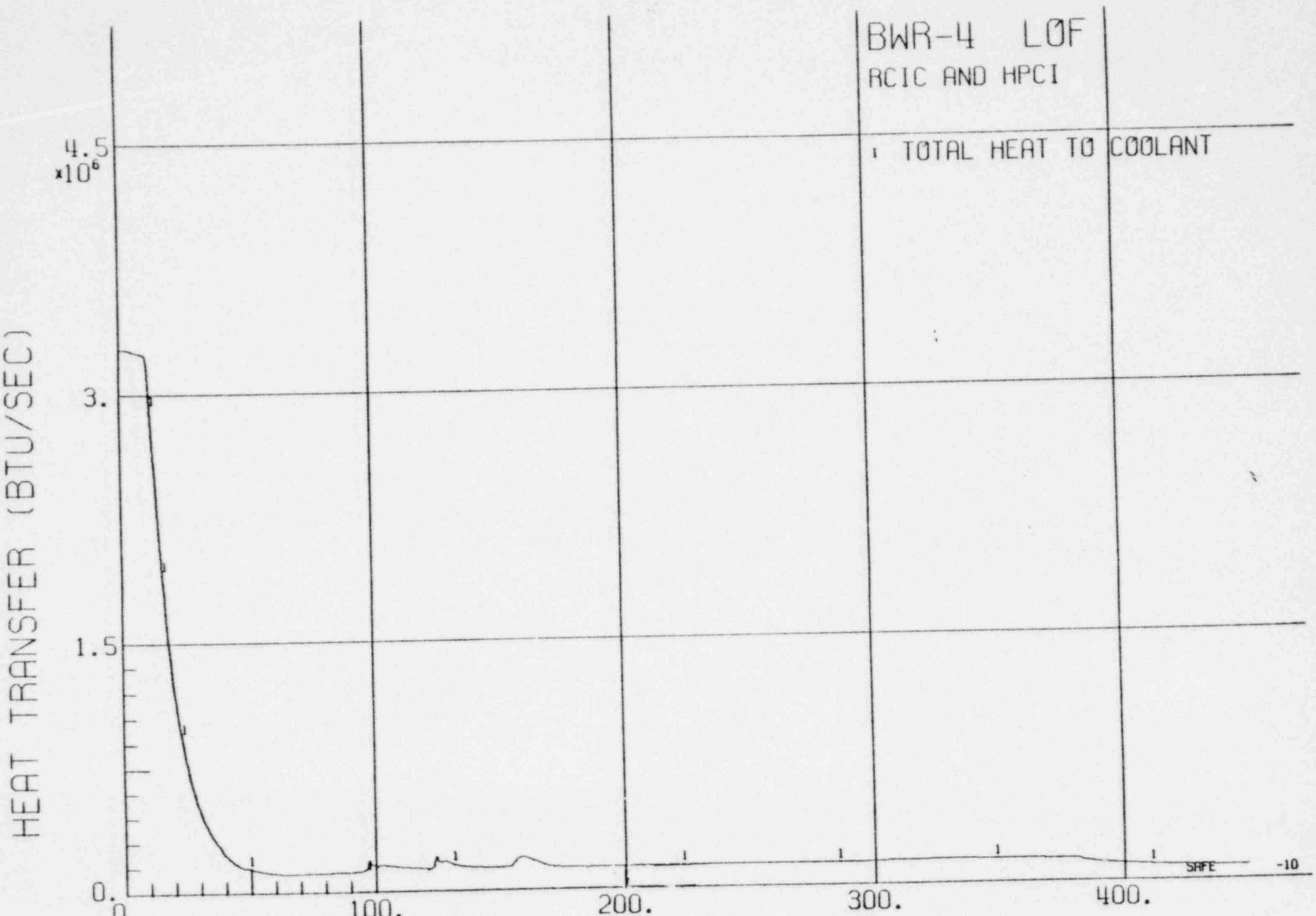


FIGURE 3.2.1.1.5-3.7

KMH 7581T
120779 0355.6

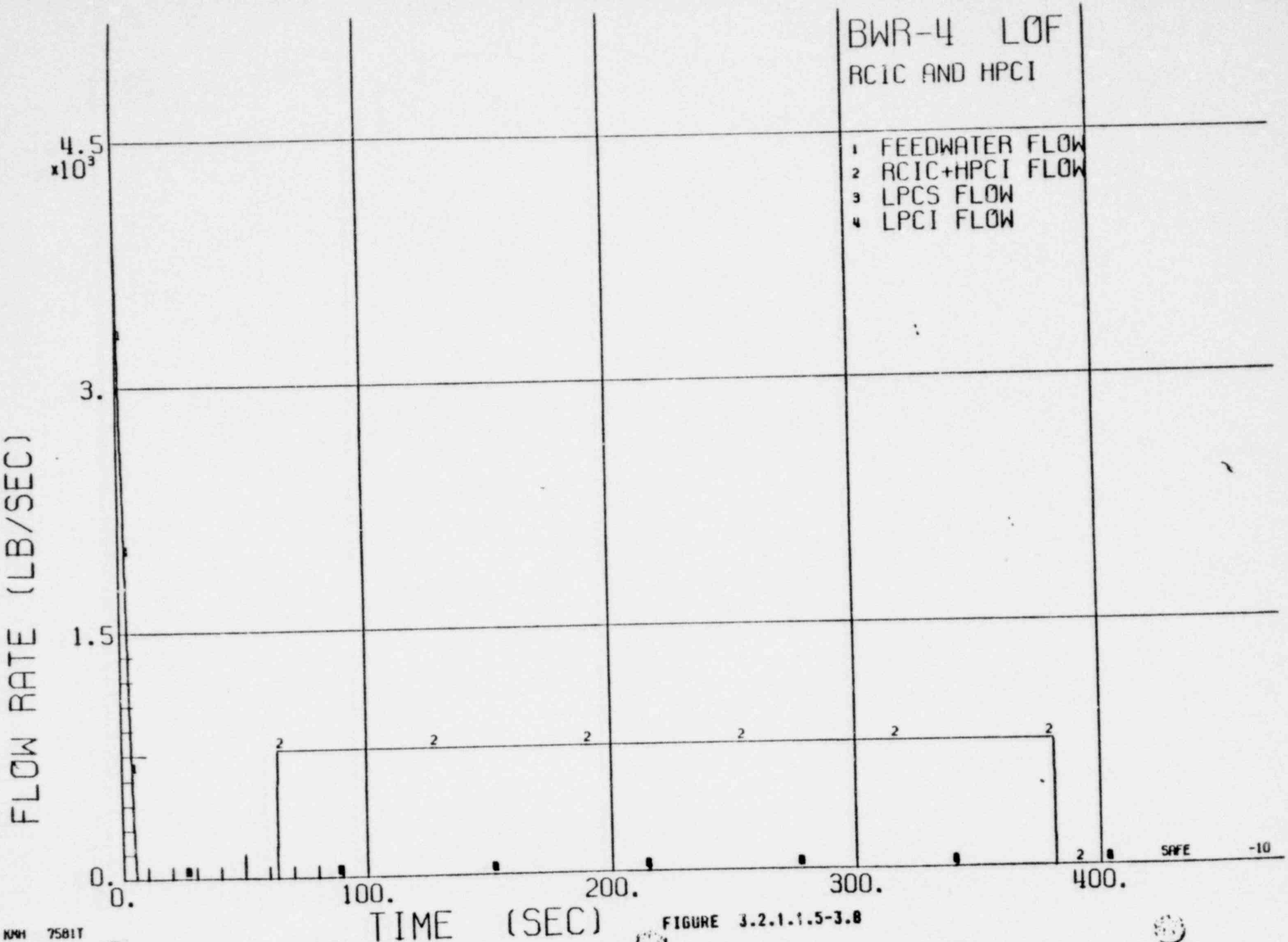
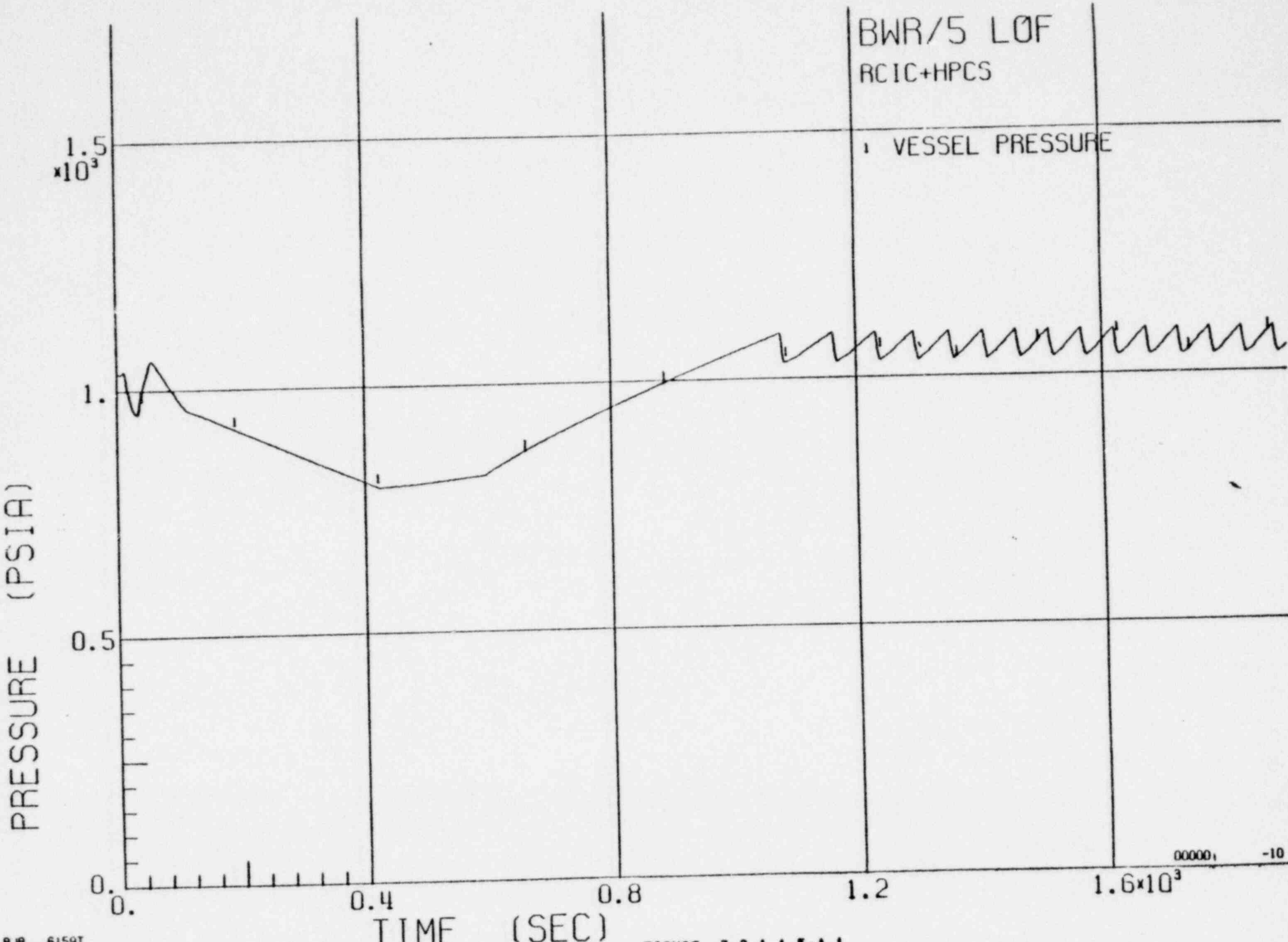


FIGURE 3.2.1.1.5-3.8

KMH 7581T



RJB 6159T

FIGURE 3.2.1.1.5-4.1

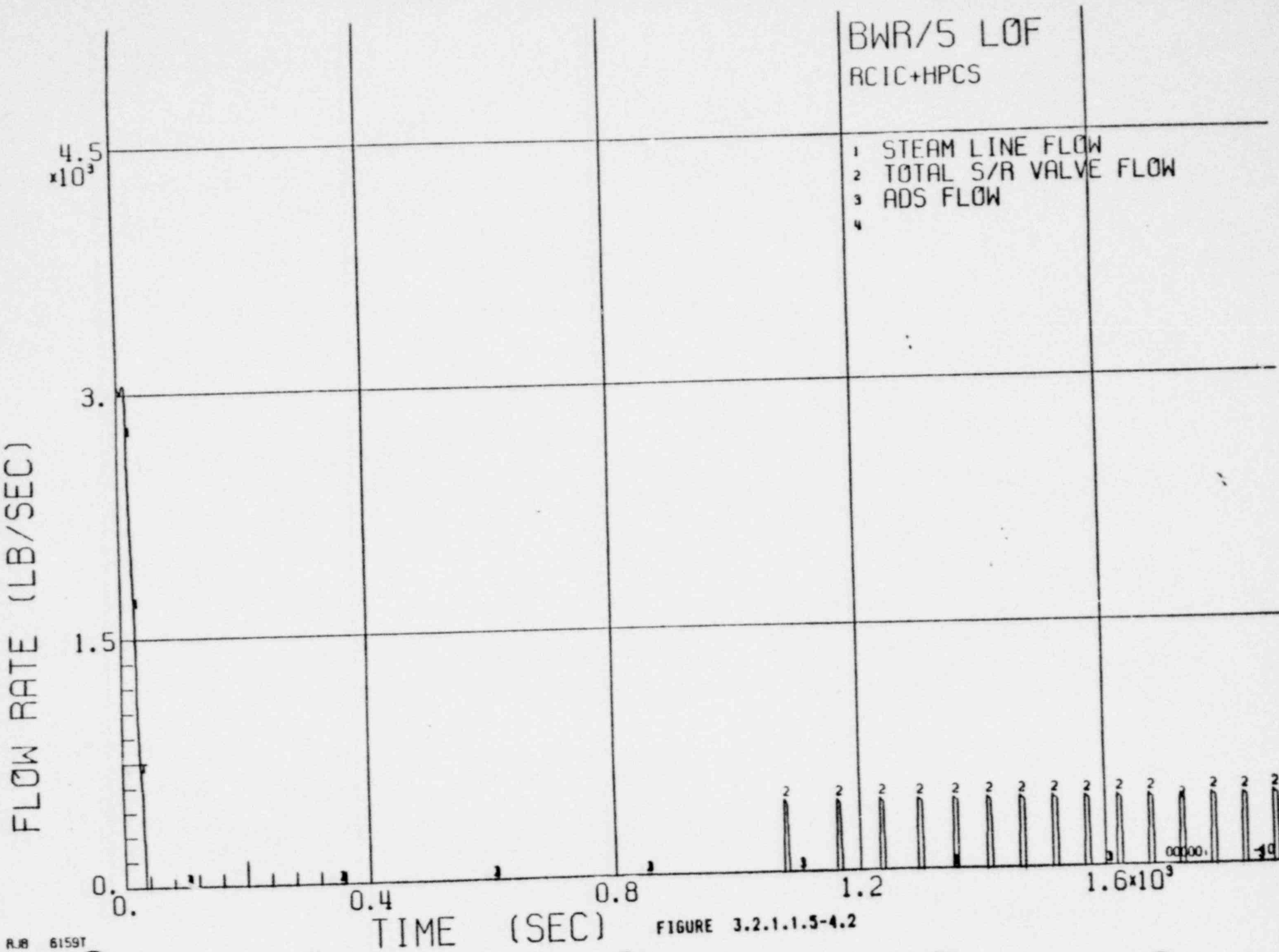


FIGURE 3.2.1.1.5-4.2

RJB 6159T

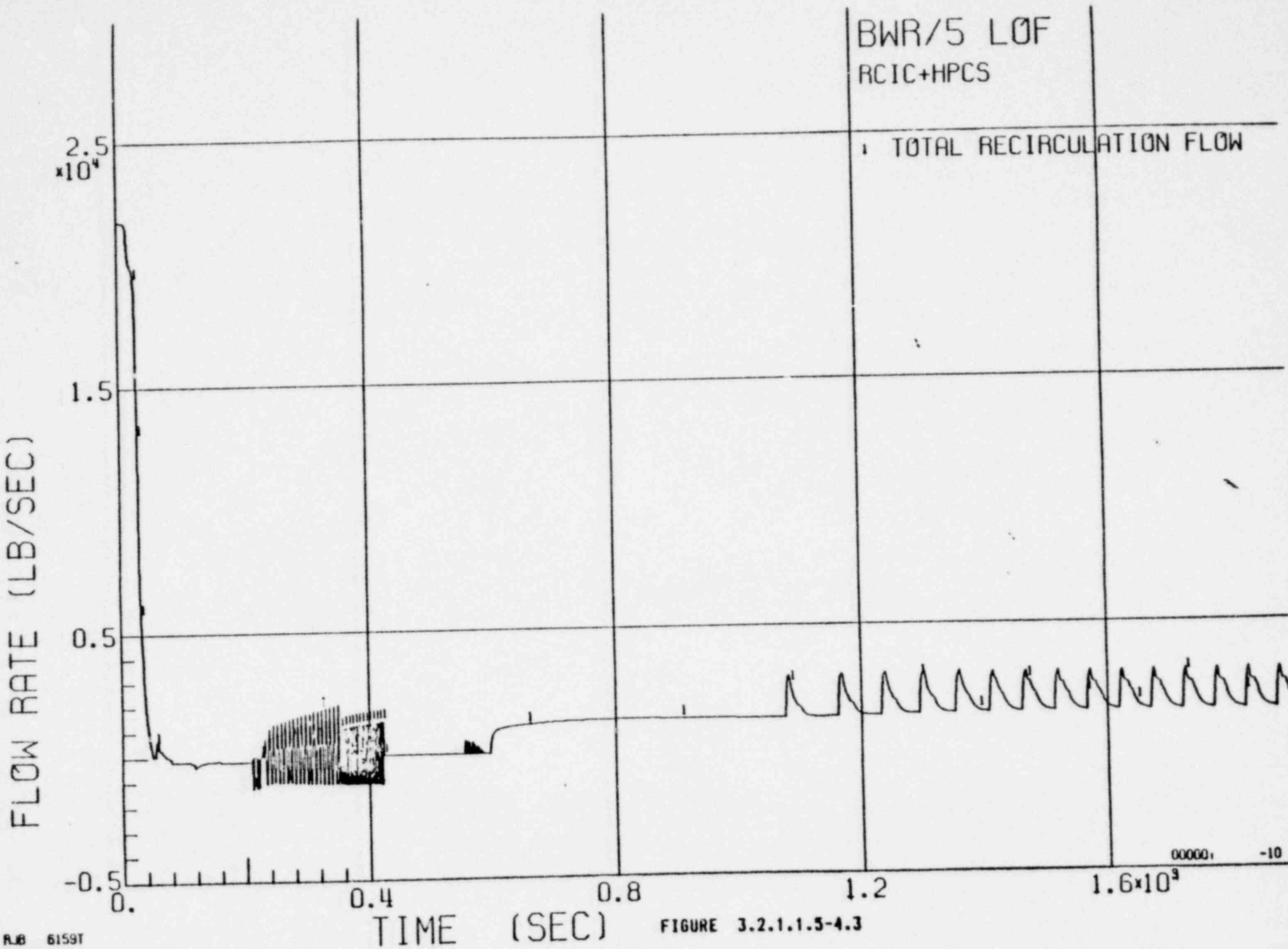


FIGURE 3.2.1.1.5-4.3

RJB 6159T
113079 1512.0

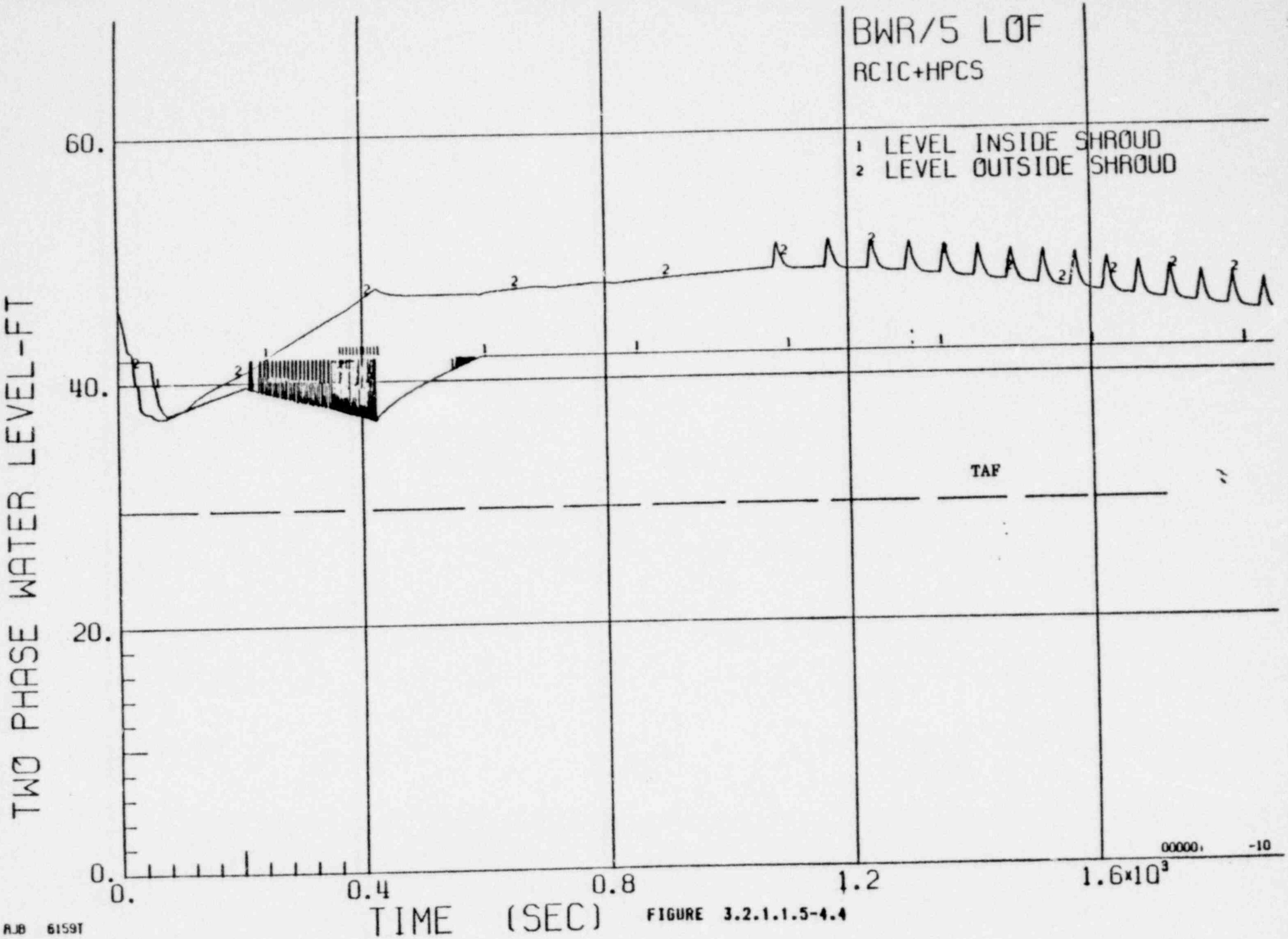


FIGURE 3.2.1.1.5-4.4

RJB 6159T

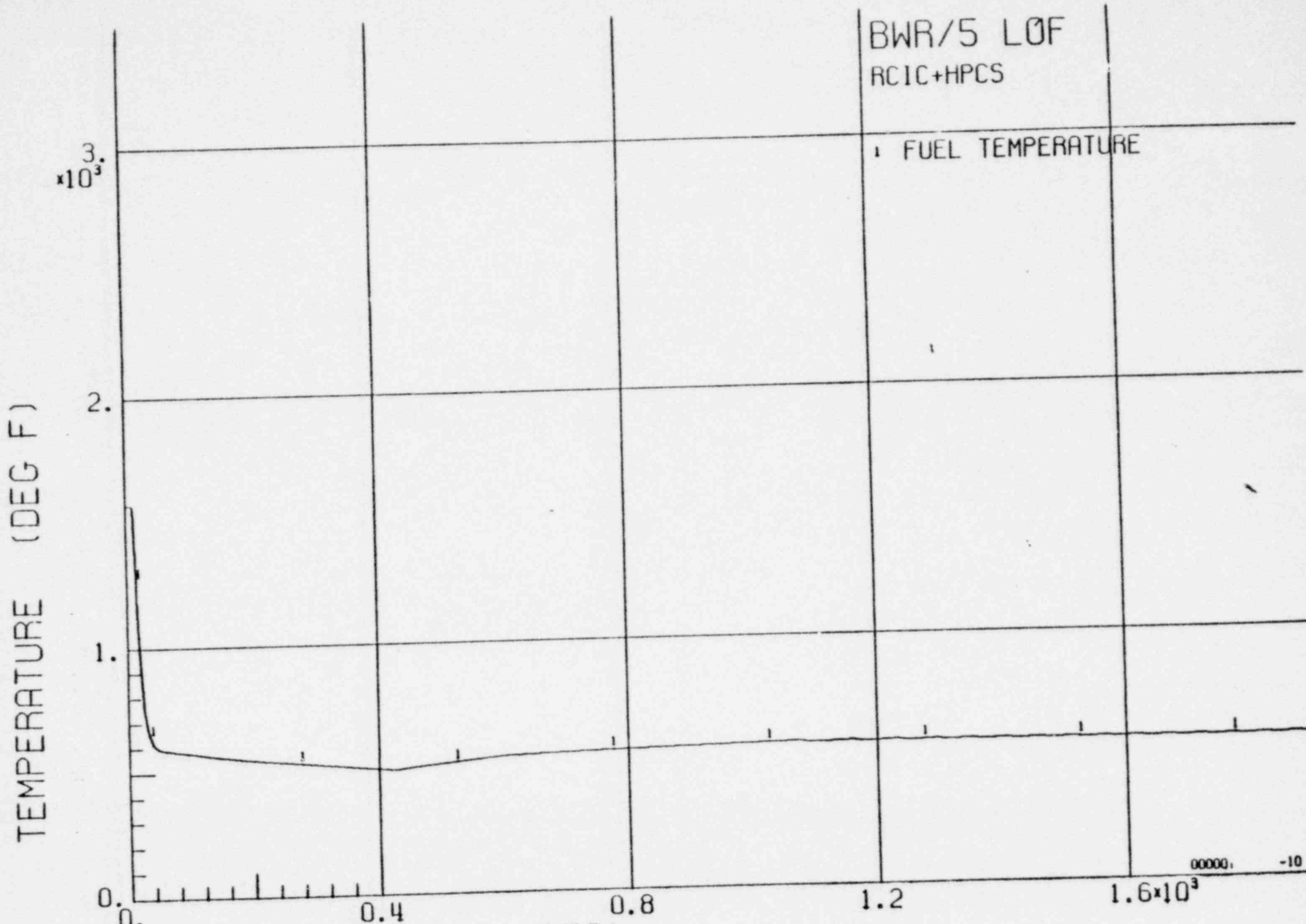


FIGURE 3.2.1.1.5-4.5

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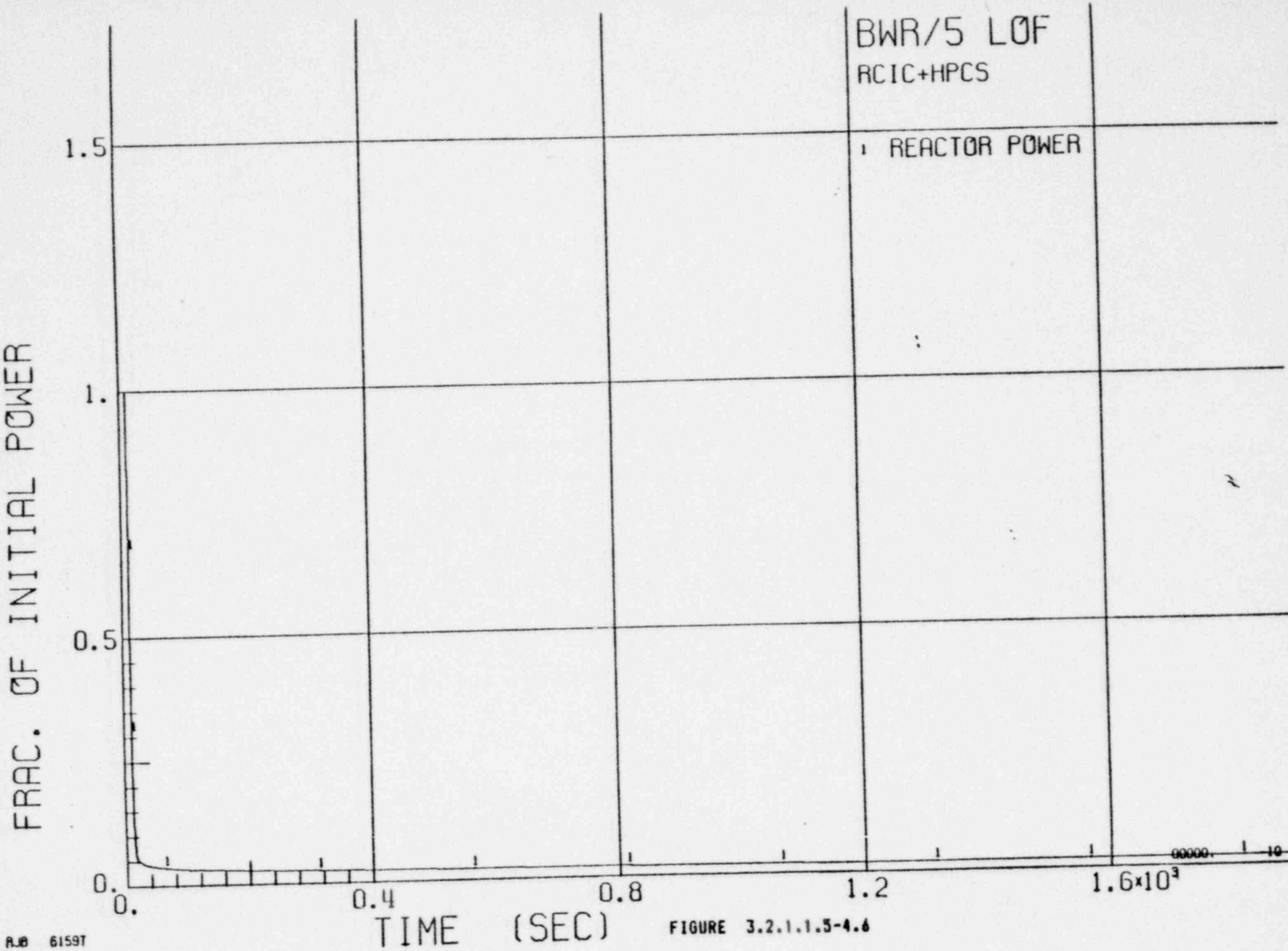


FIGURE 3.2.1.1.5-4.6

RJB 6159T

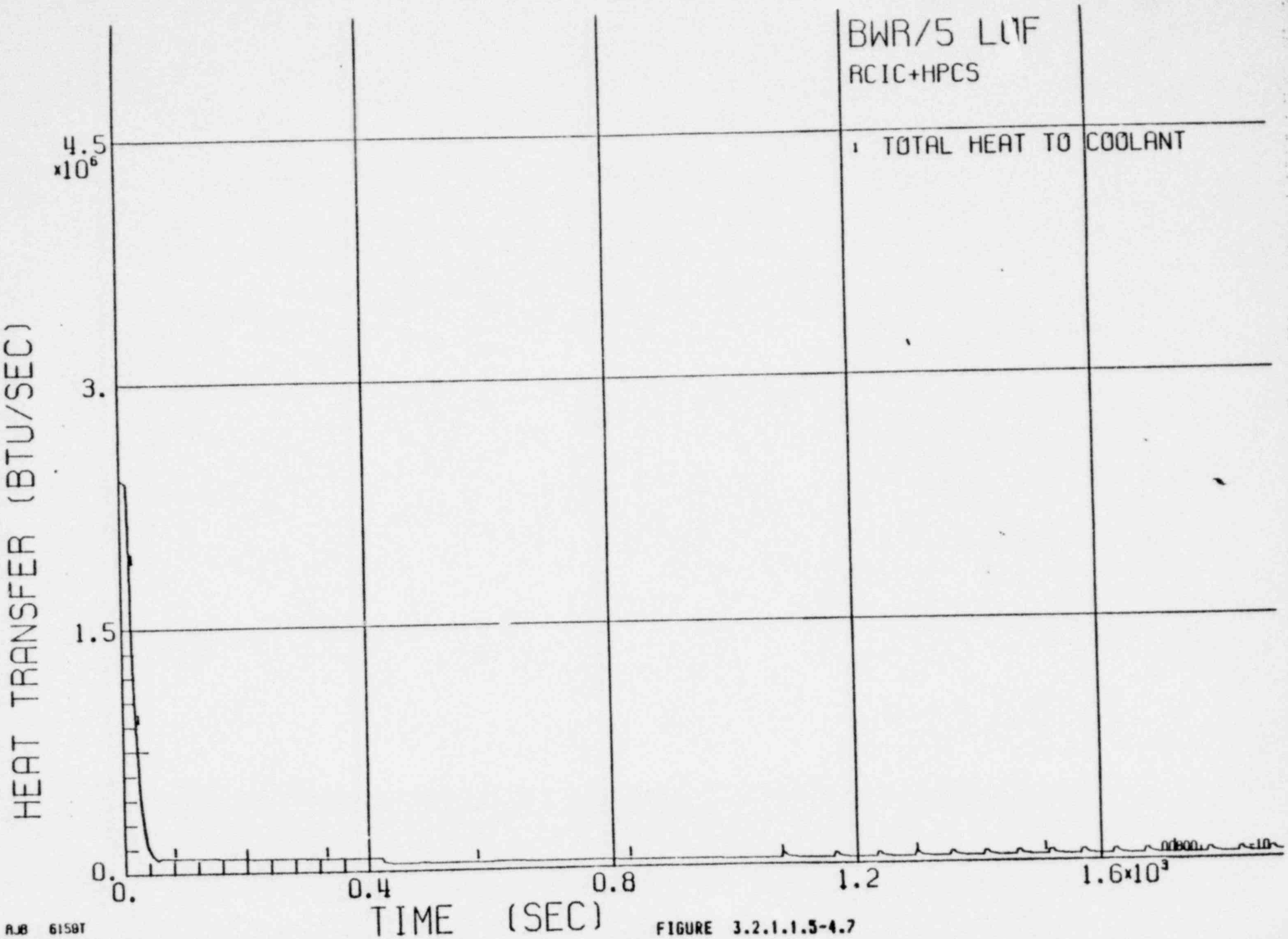


FIGURE 3.2.1.1.5-4.7

RJB 6158T
119079 1518.2

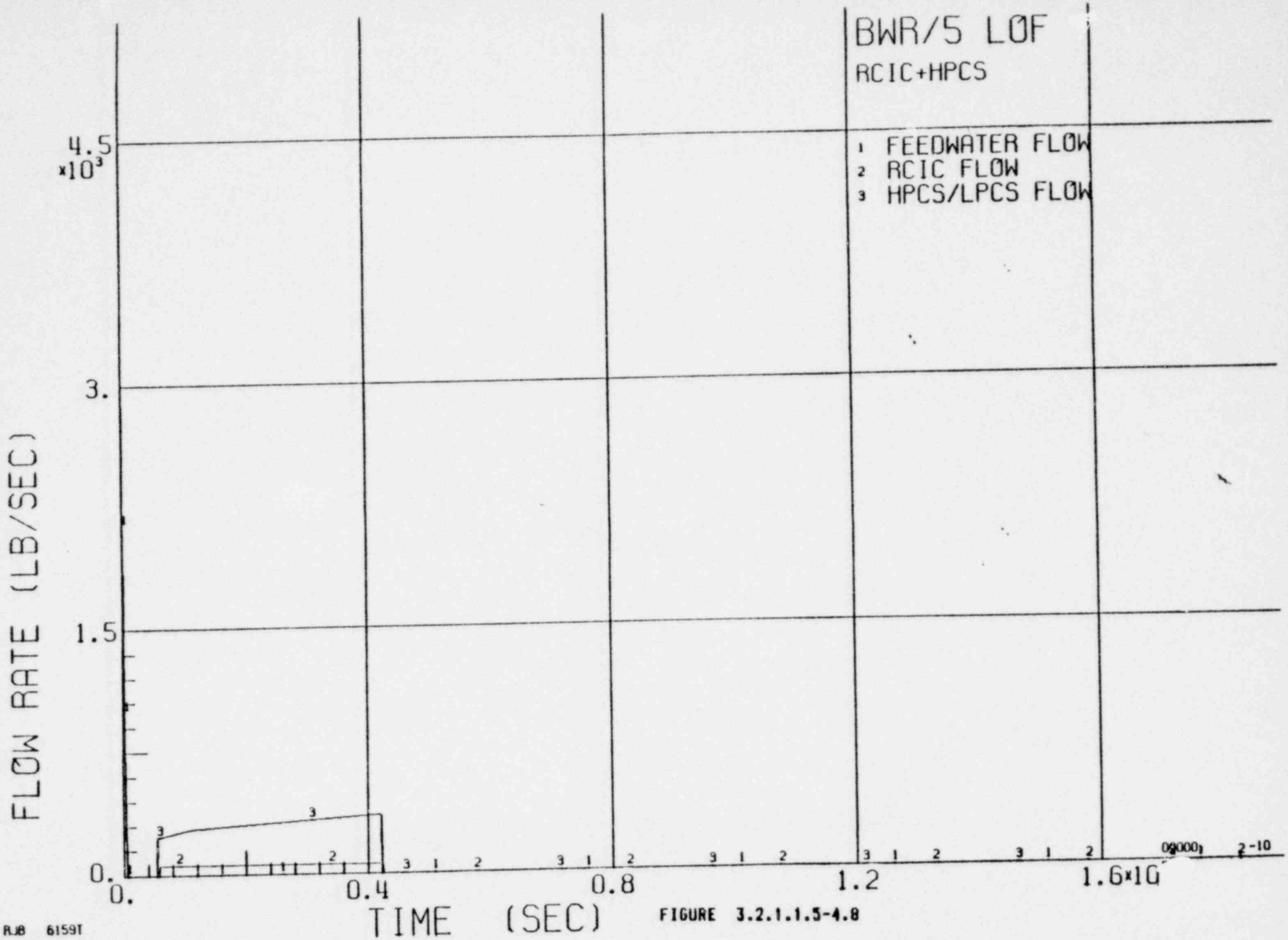
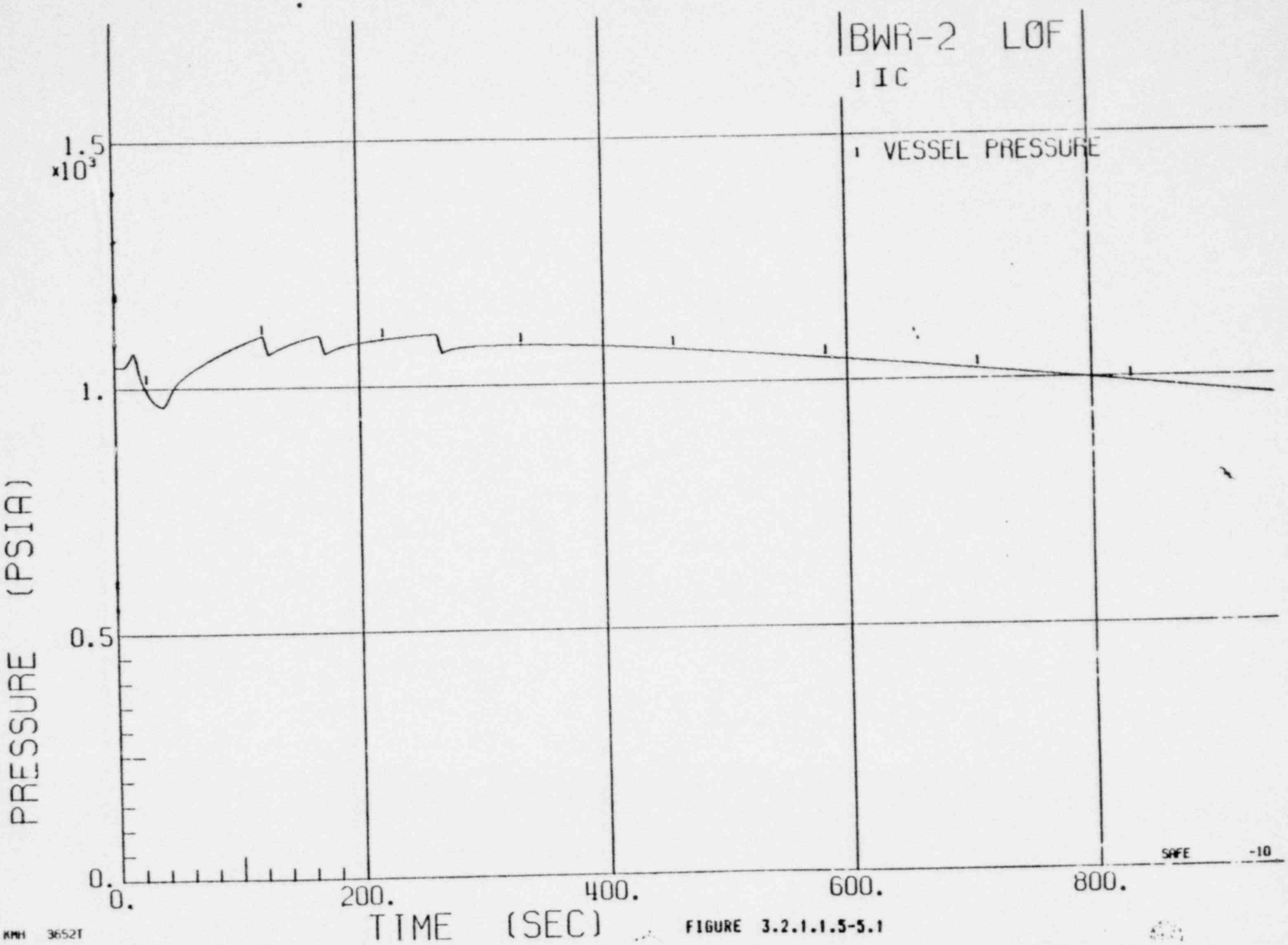


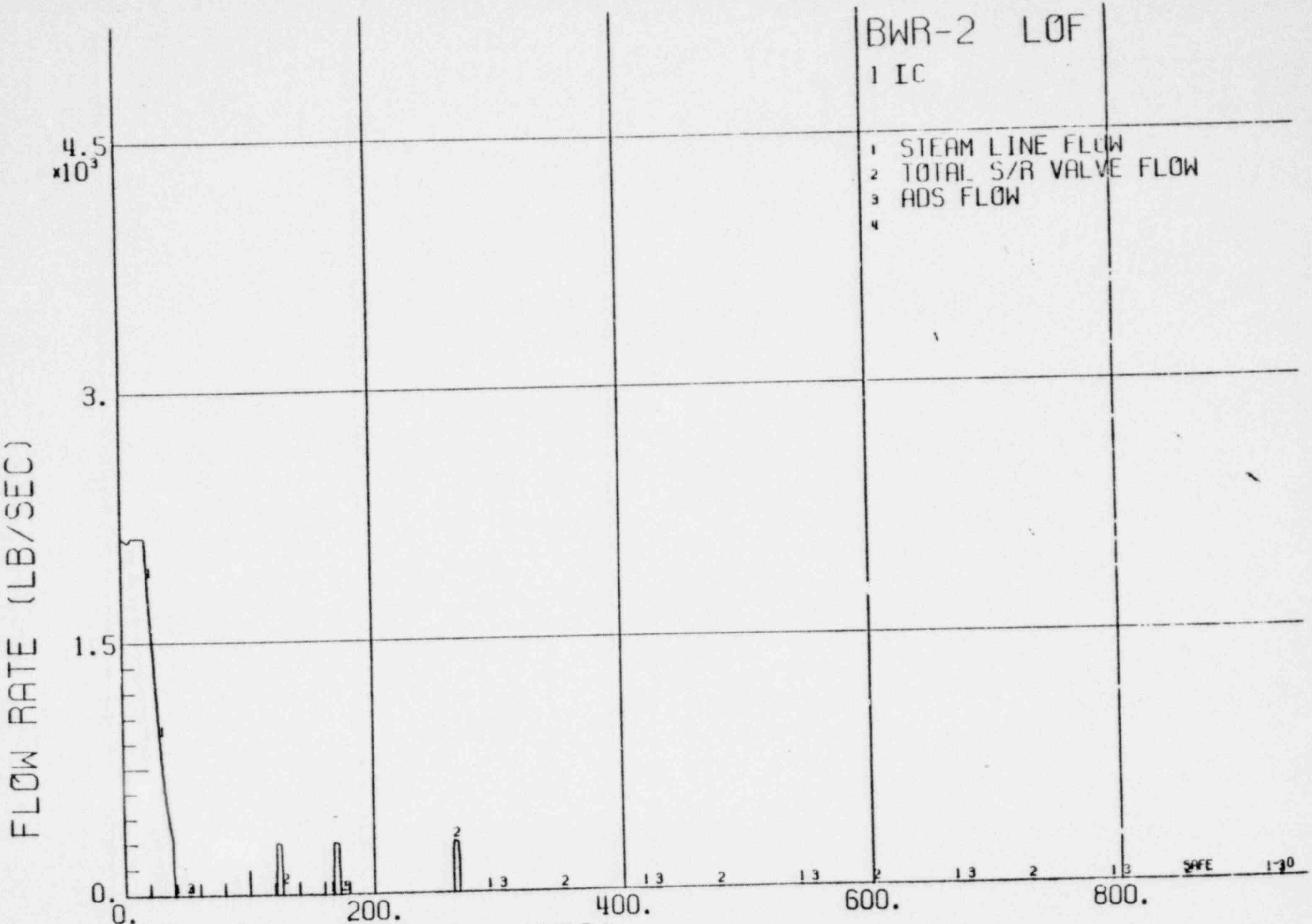
FIGURE 3.2.1.1.5-4.8

RJB 6159T



KMH 3652T

FIGURE 3.2.1.1.5-5.1



KMH 3652T

FIGURE 3.2.1.1.5-5.2

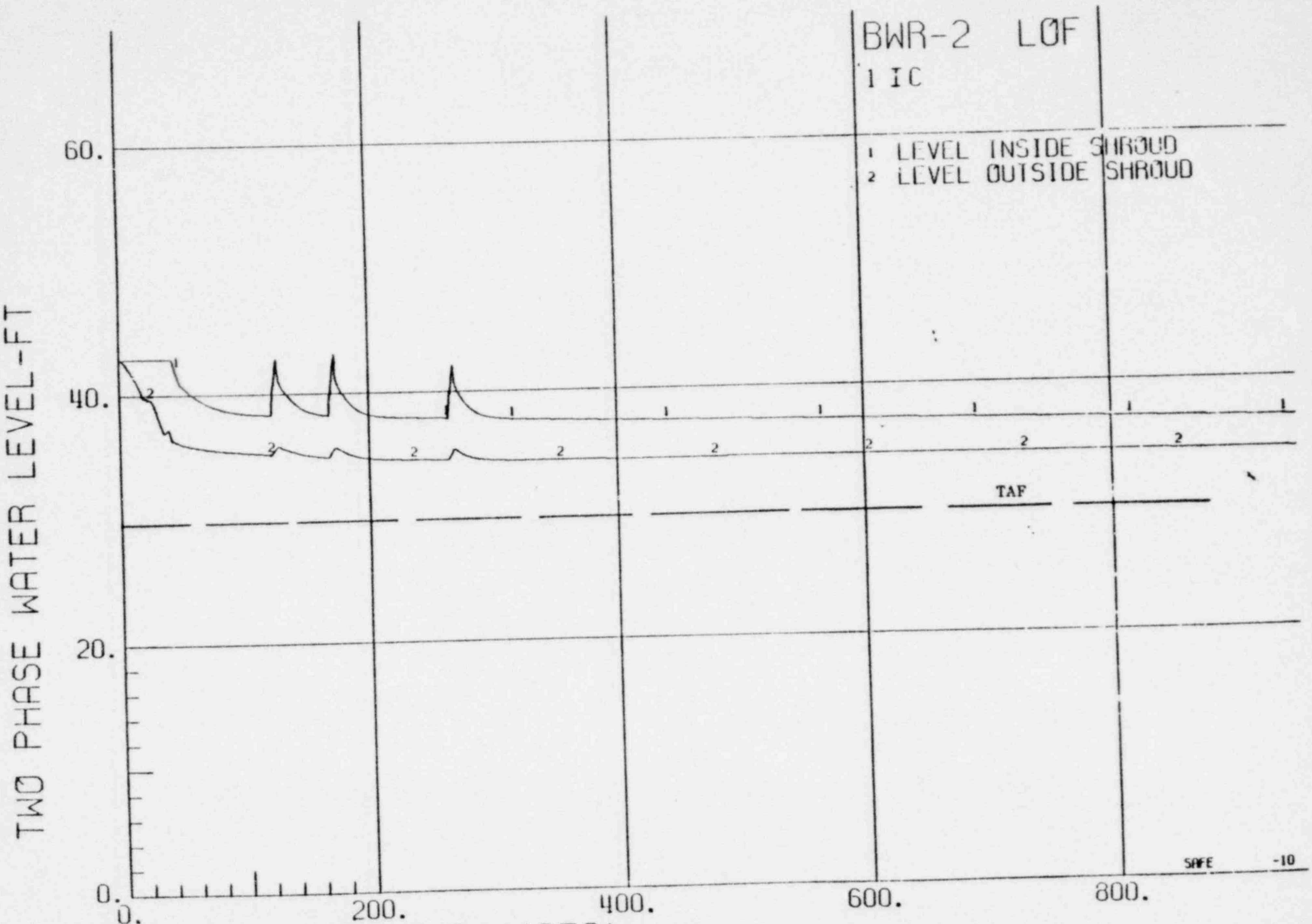
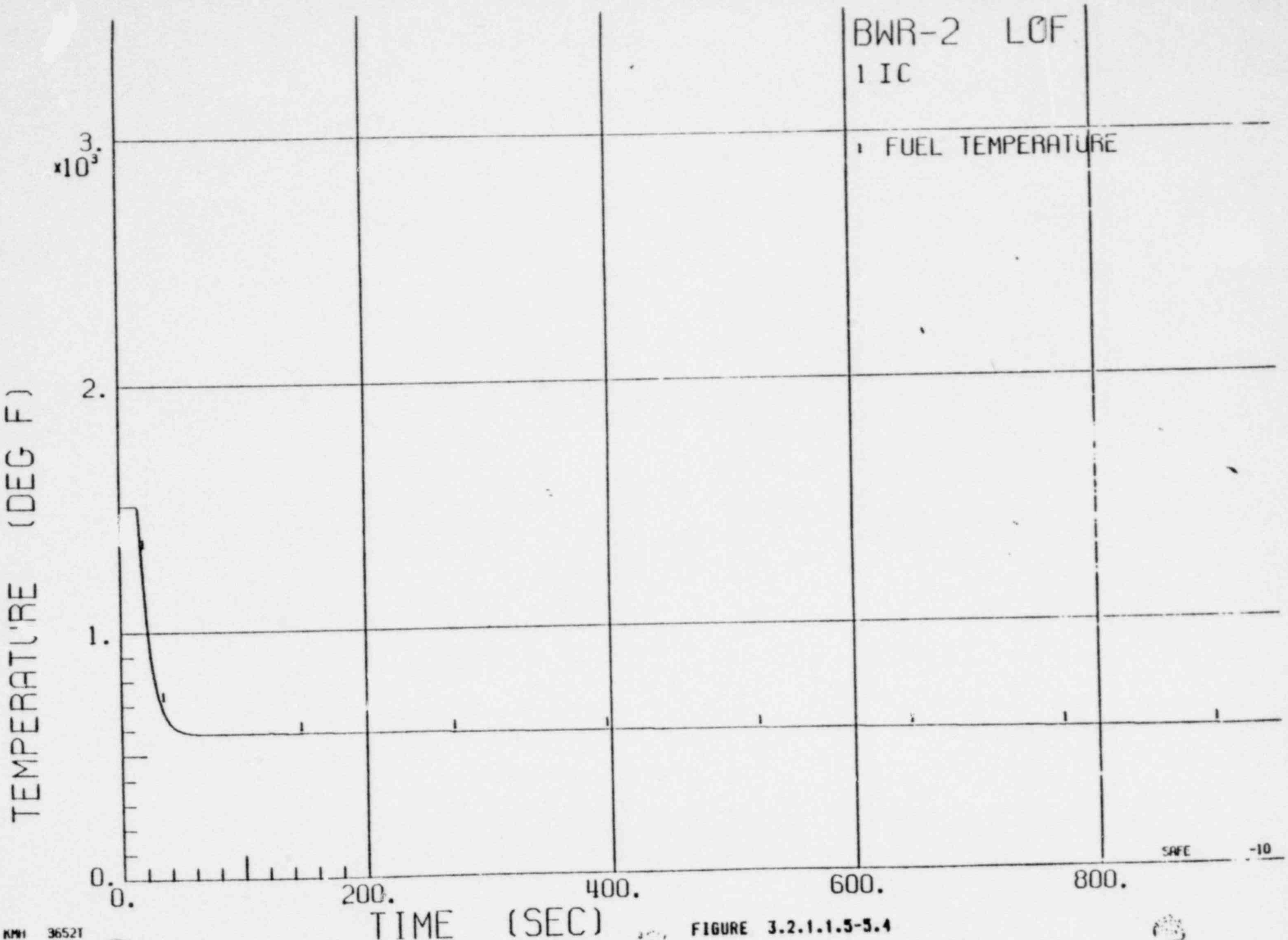


FIGURE 3.2.1.1.5-5.3

KMH 3652T



KMH 3652T

FIGURE 3.2.1.1.5-5.4

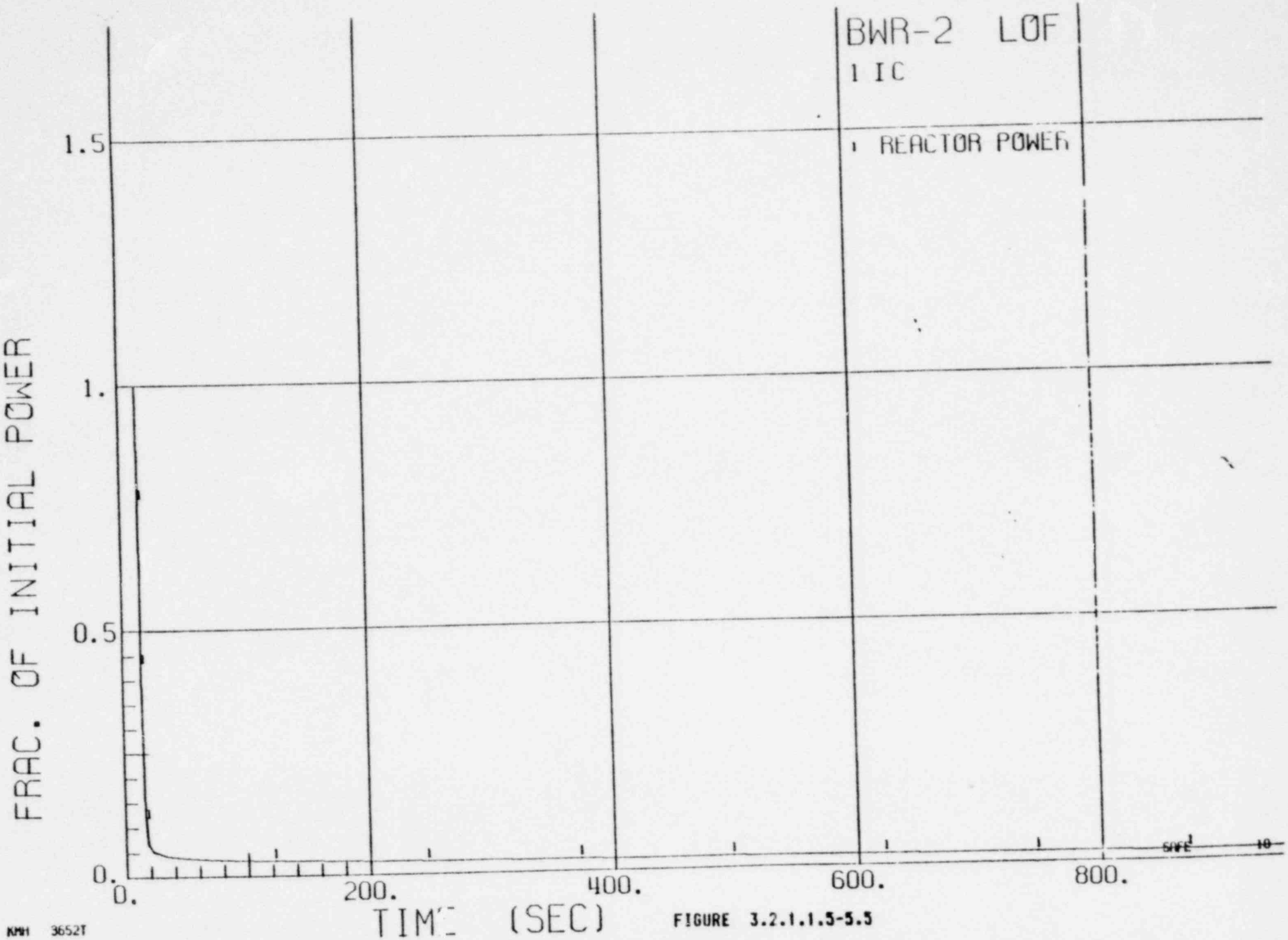


FIGURE 3.2.1.1.5-5.5

KMH 3652T

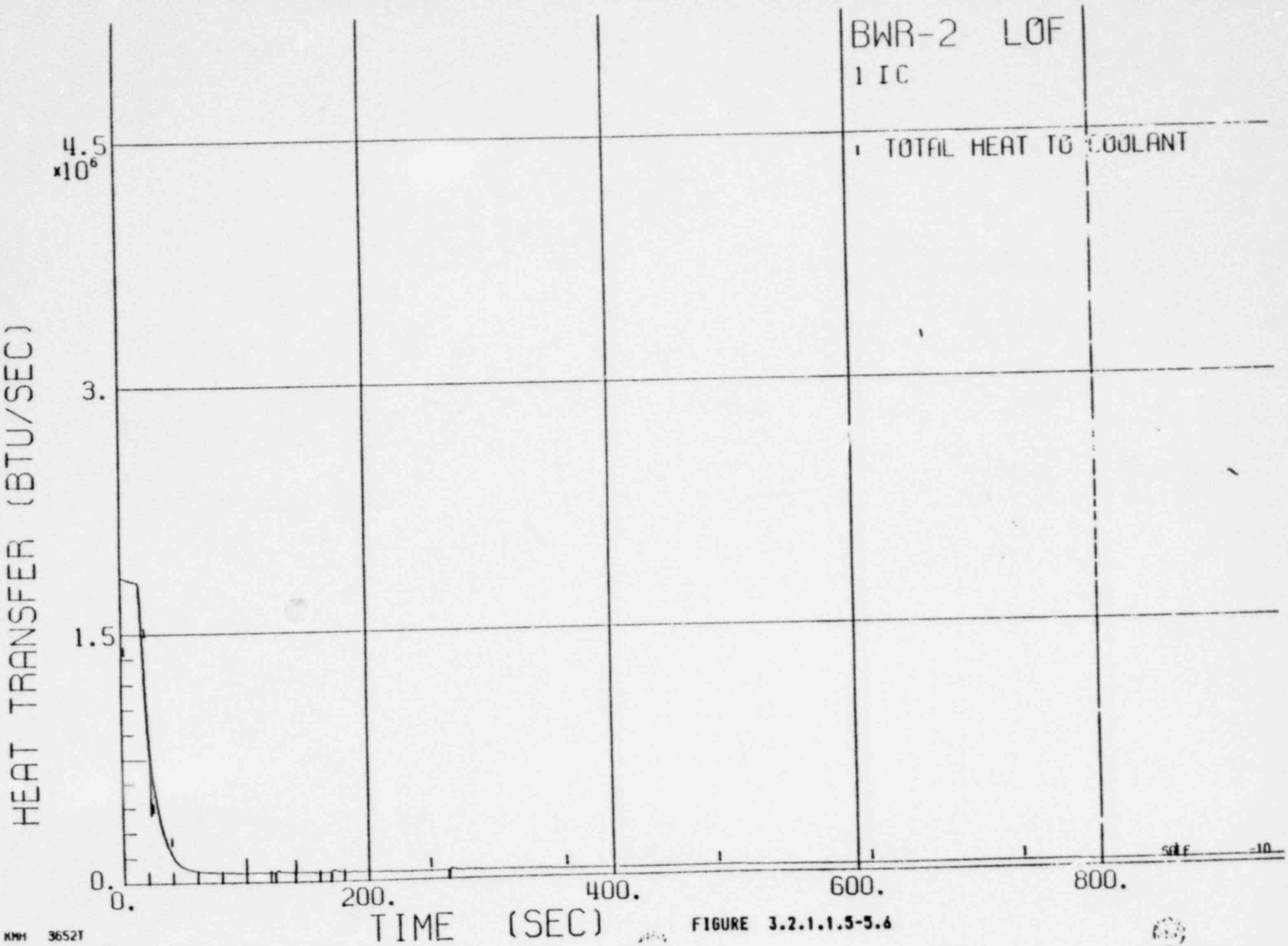


FIGURE 3.2.1.1.5-5.6

KMH 3652T

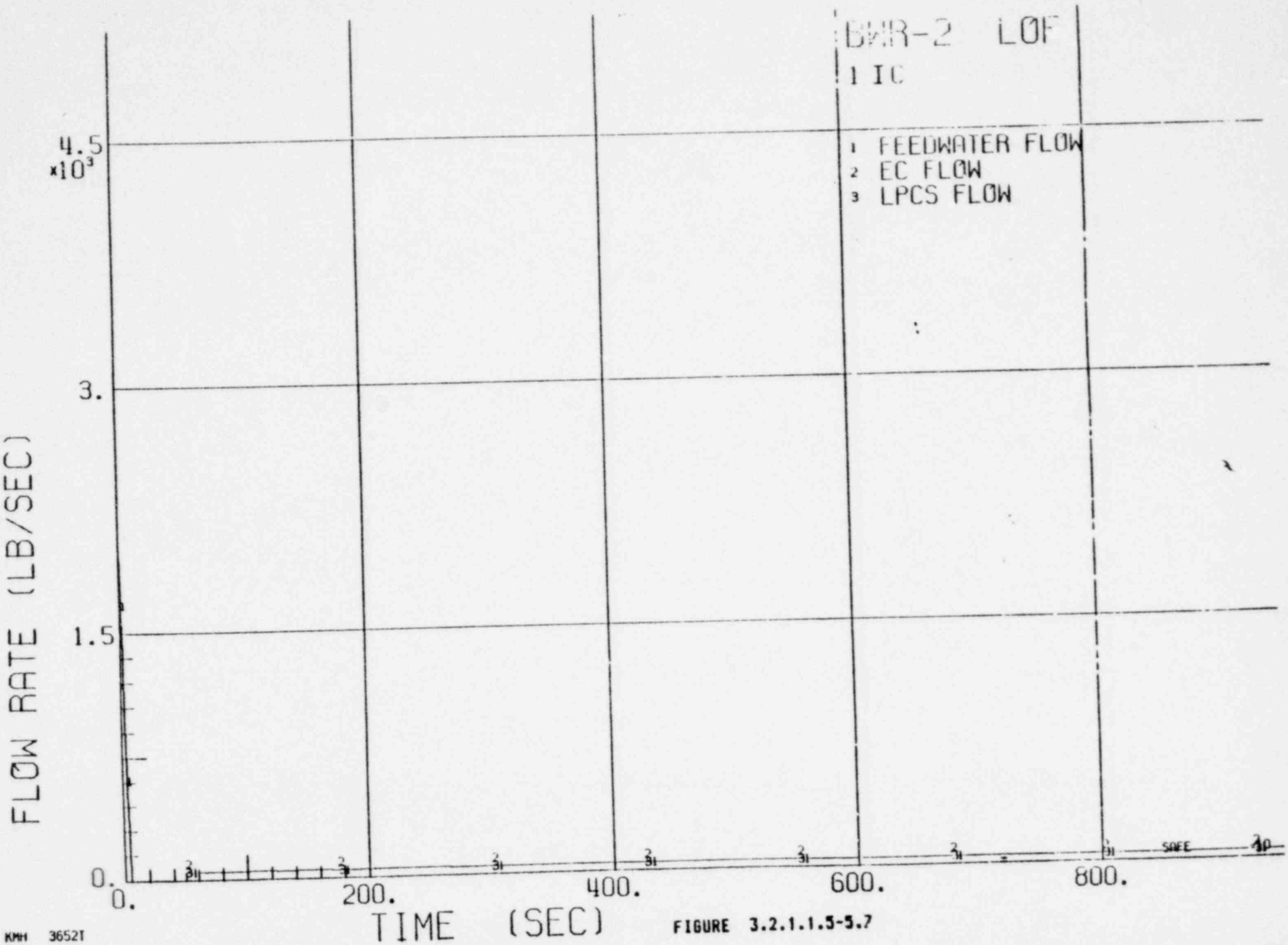


FIGURE 3.2.1.1.5-5.7

KMH 3652T

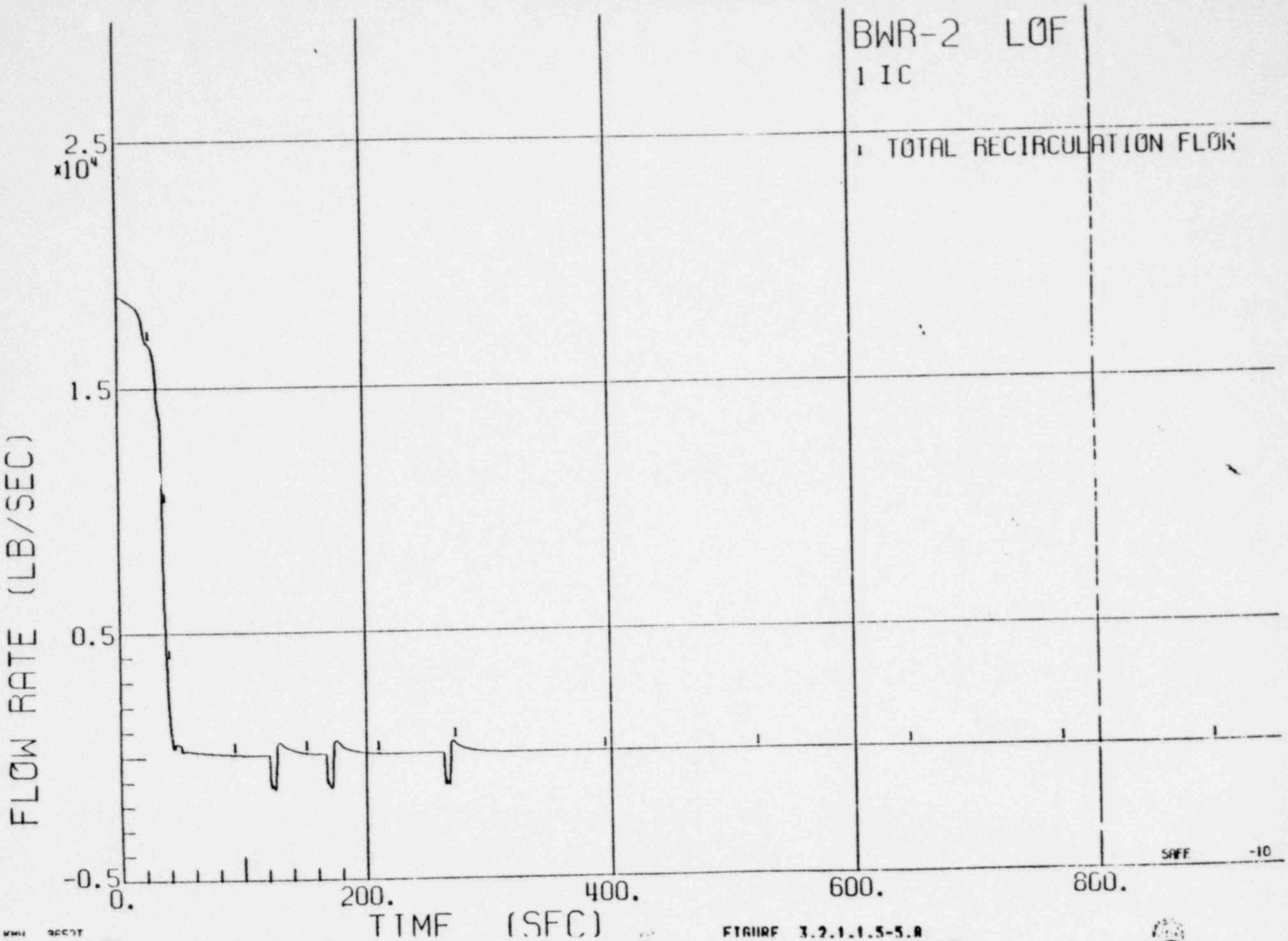


FIGURE 3.2.1.1.5-5.A

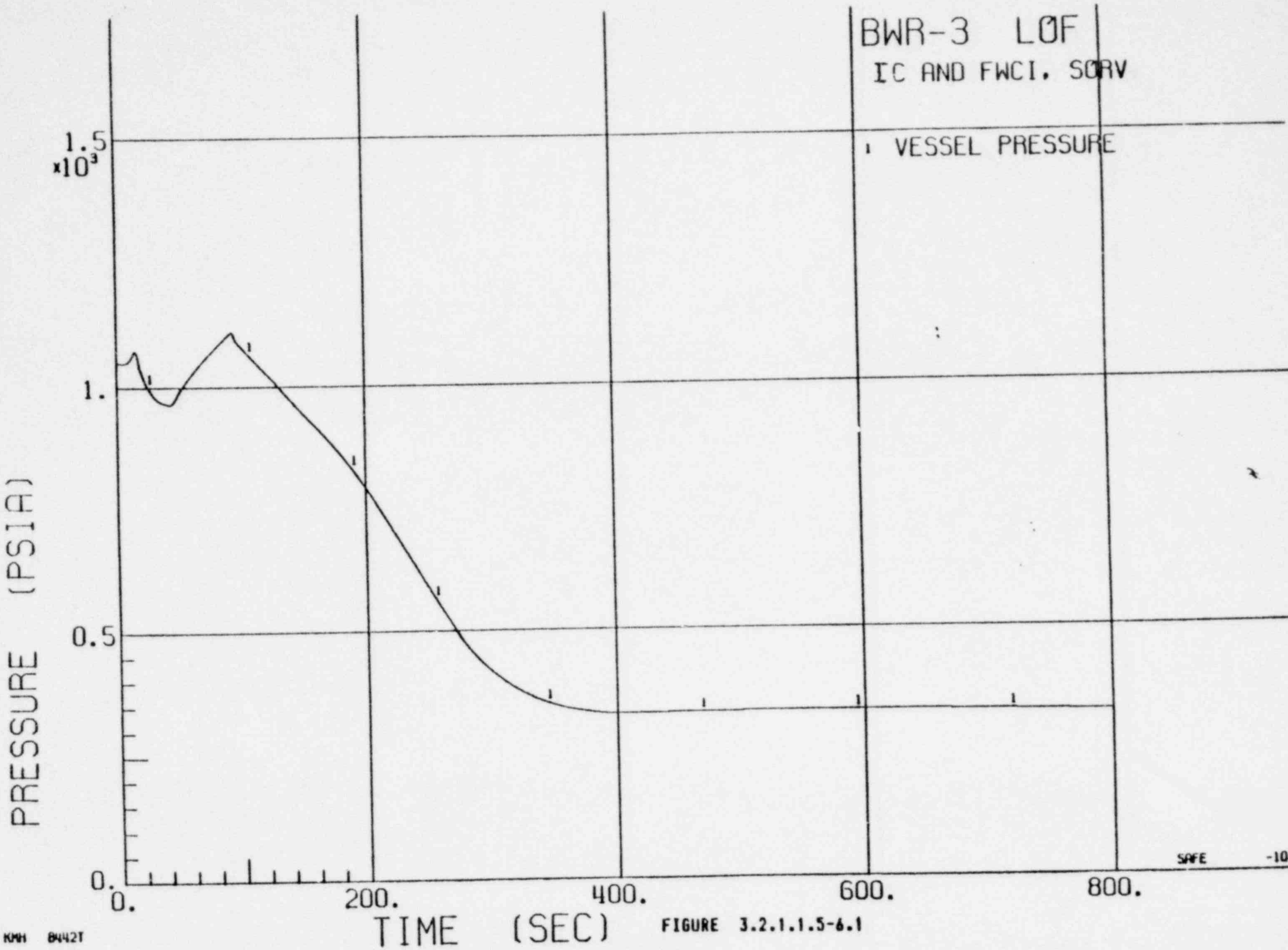


FIGURE 3.2.1.1.5-6.1

KMH 8442T

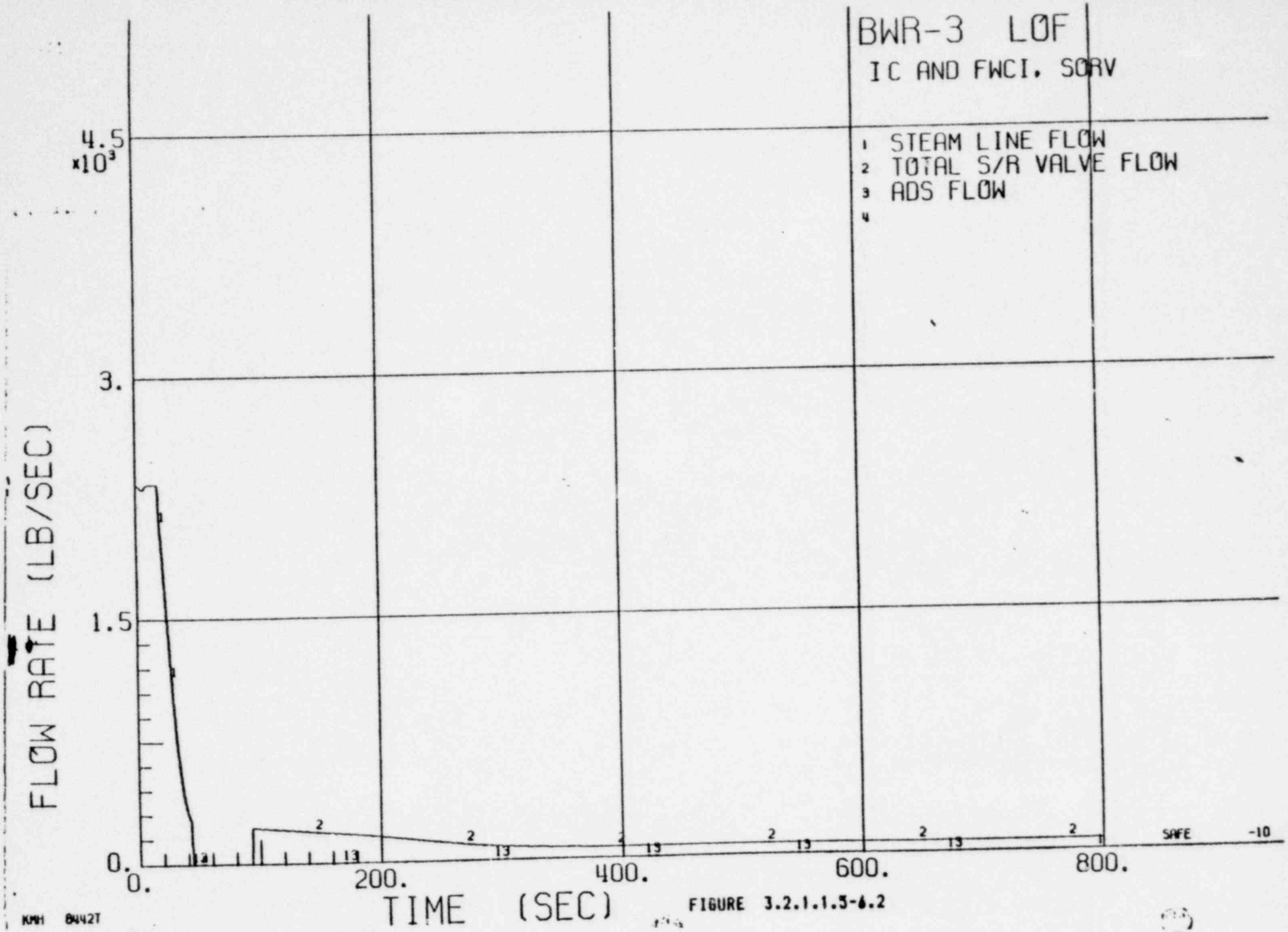


FIGURE 3.2.1.1.5-6.2

KMH 8442T

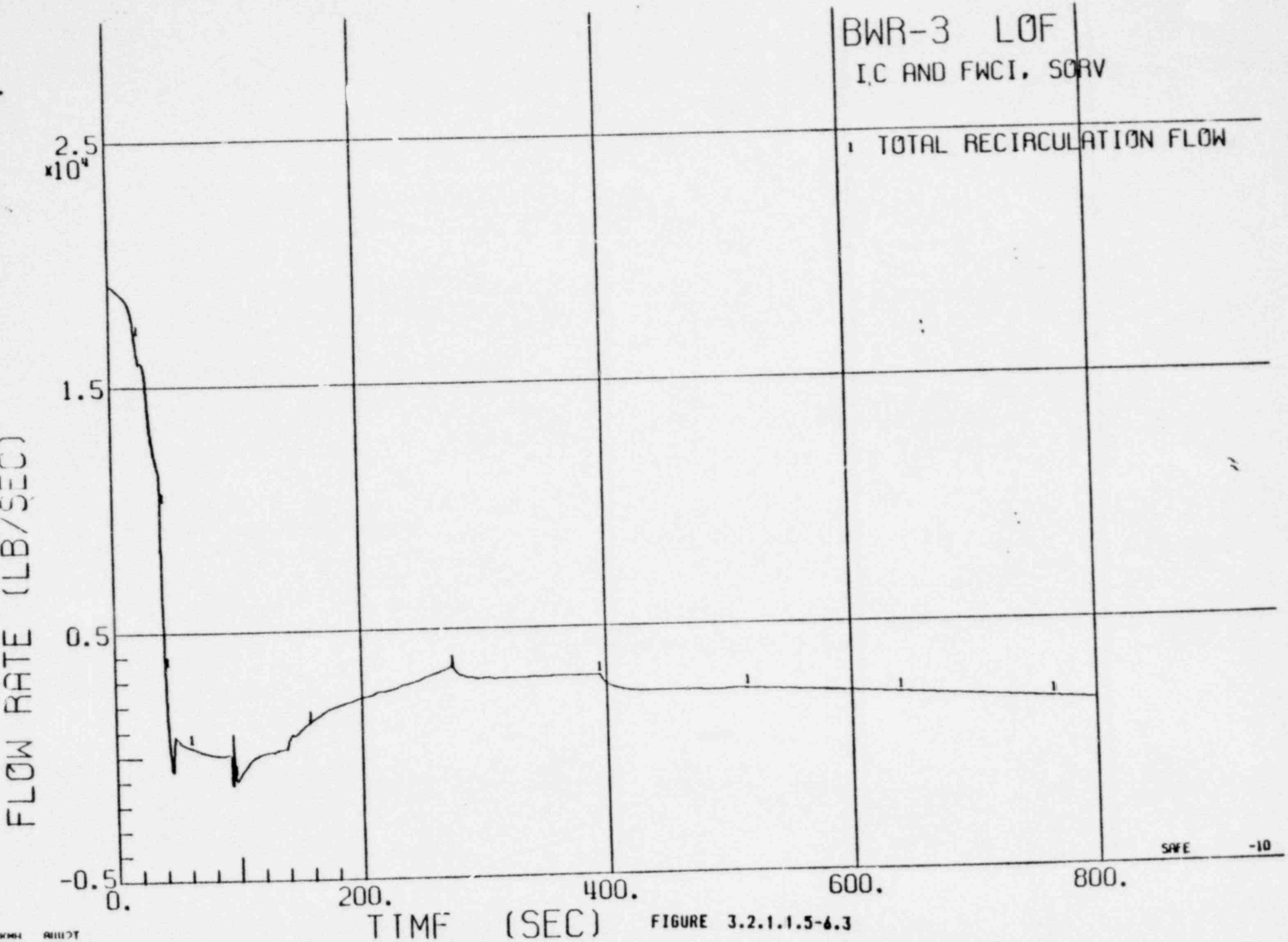


FIGURE 3.2.1.1.5-6.3

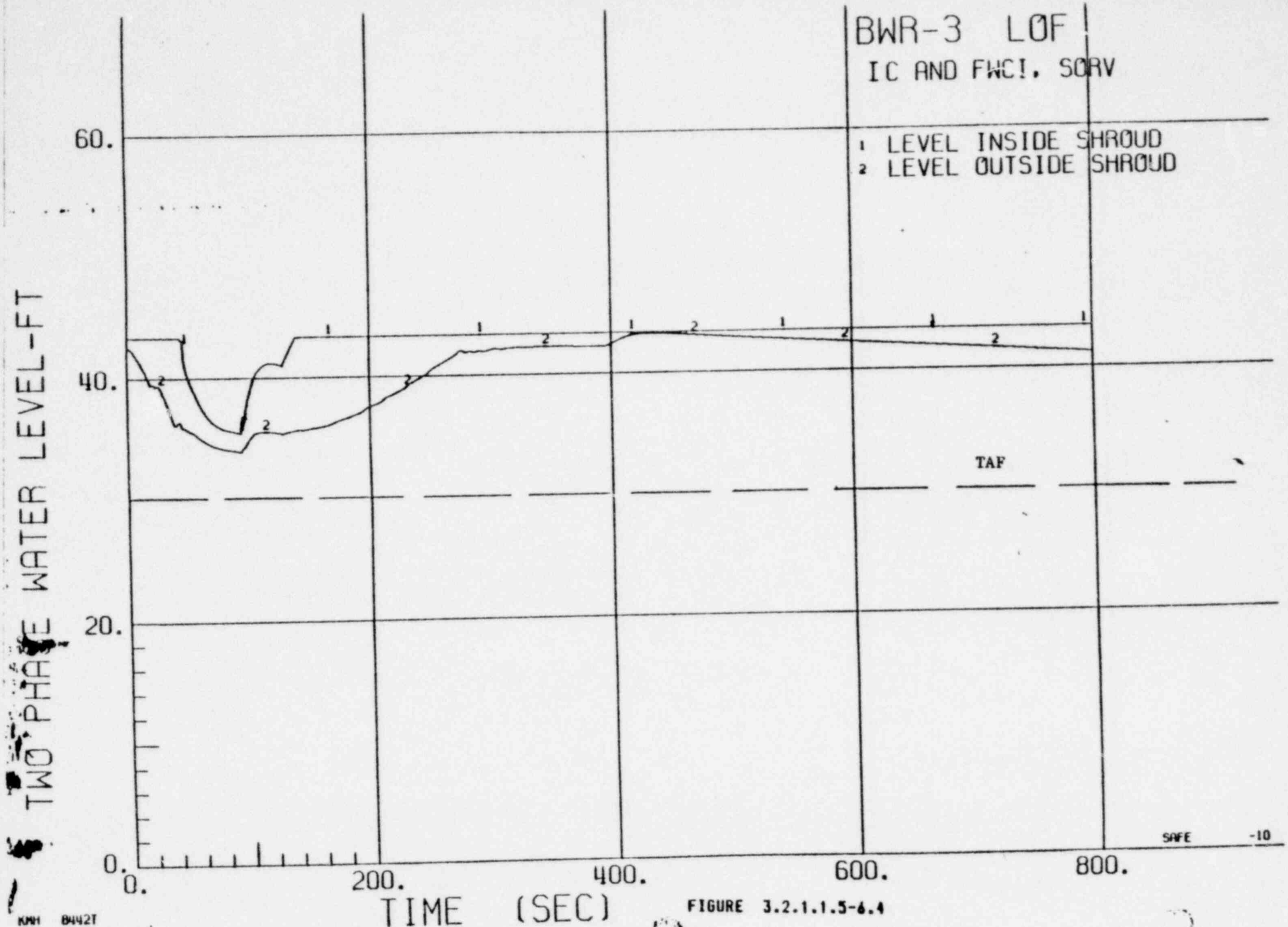


FIGURE 3.2.1.1.5-6.4

KNH 8442T

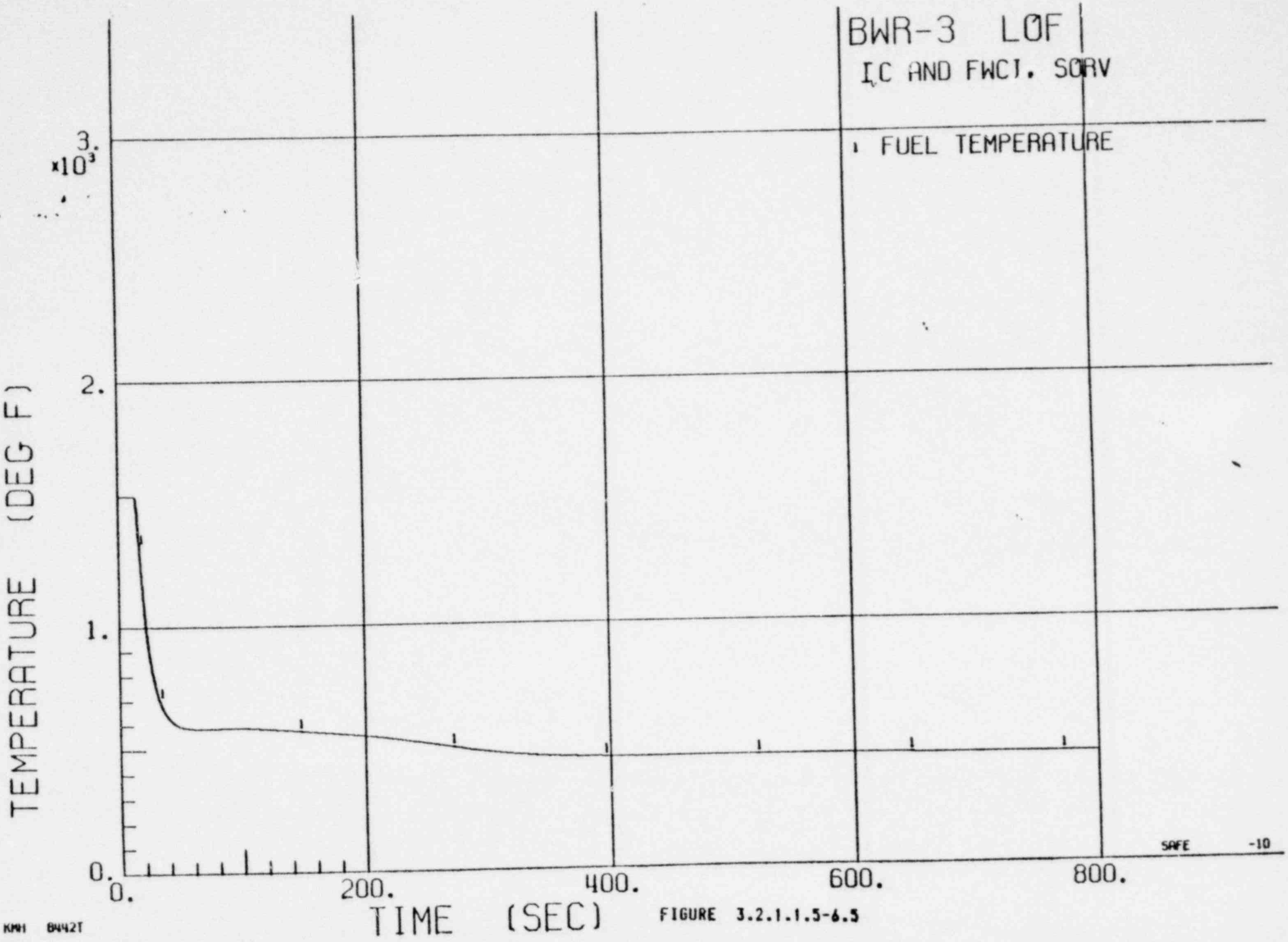


FIGURE 3.2.1.1.5-6.5

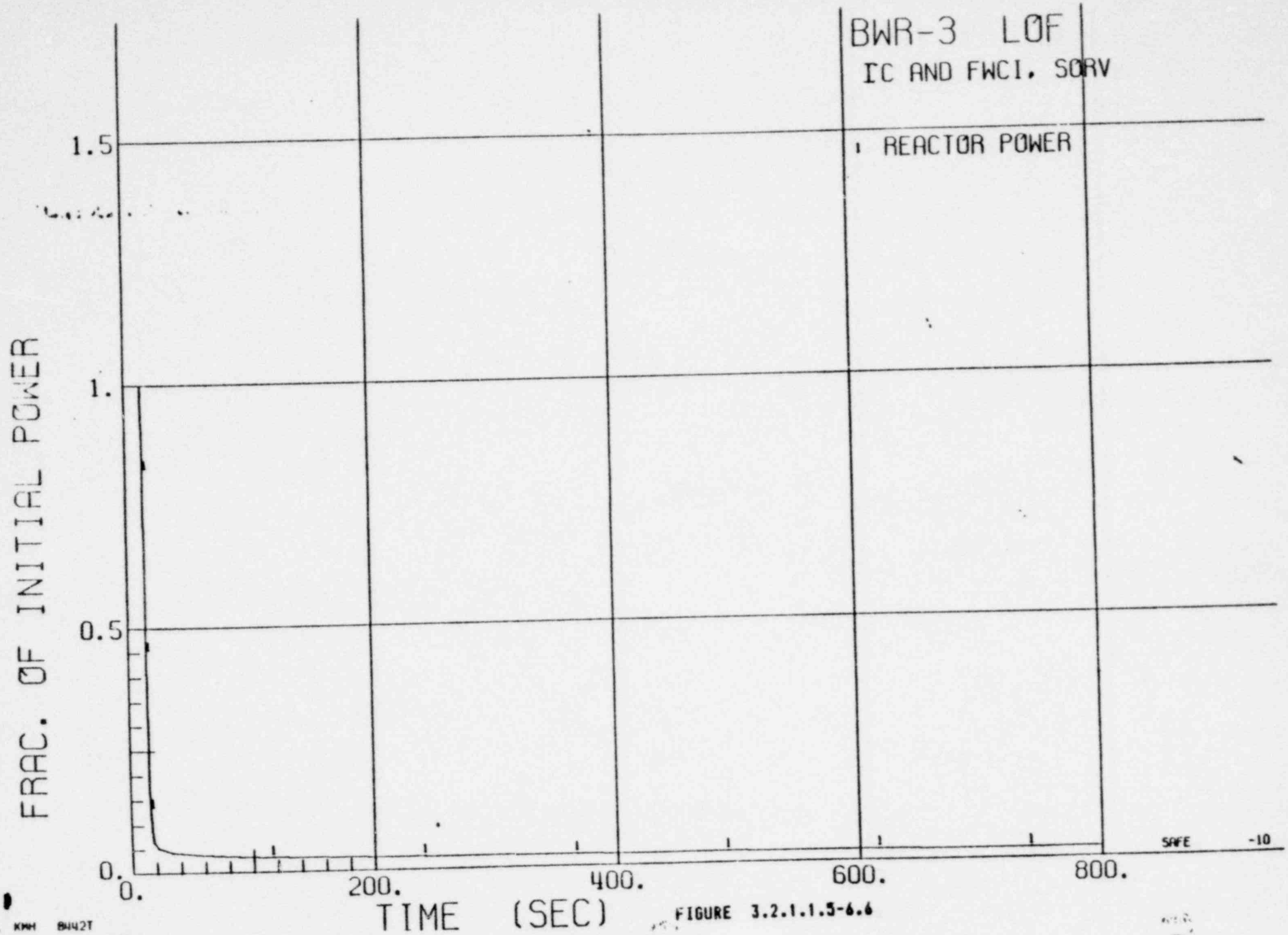


FIGURE 3.2.1.1.5-6.6

KMH 8442T

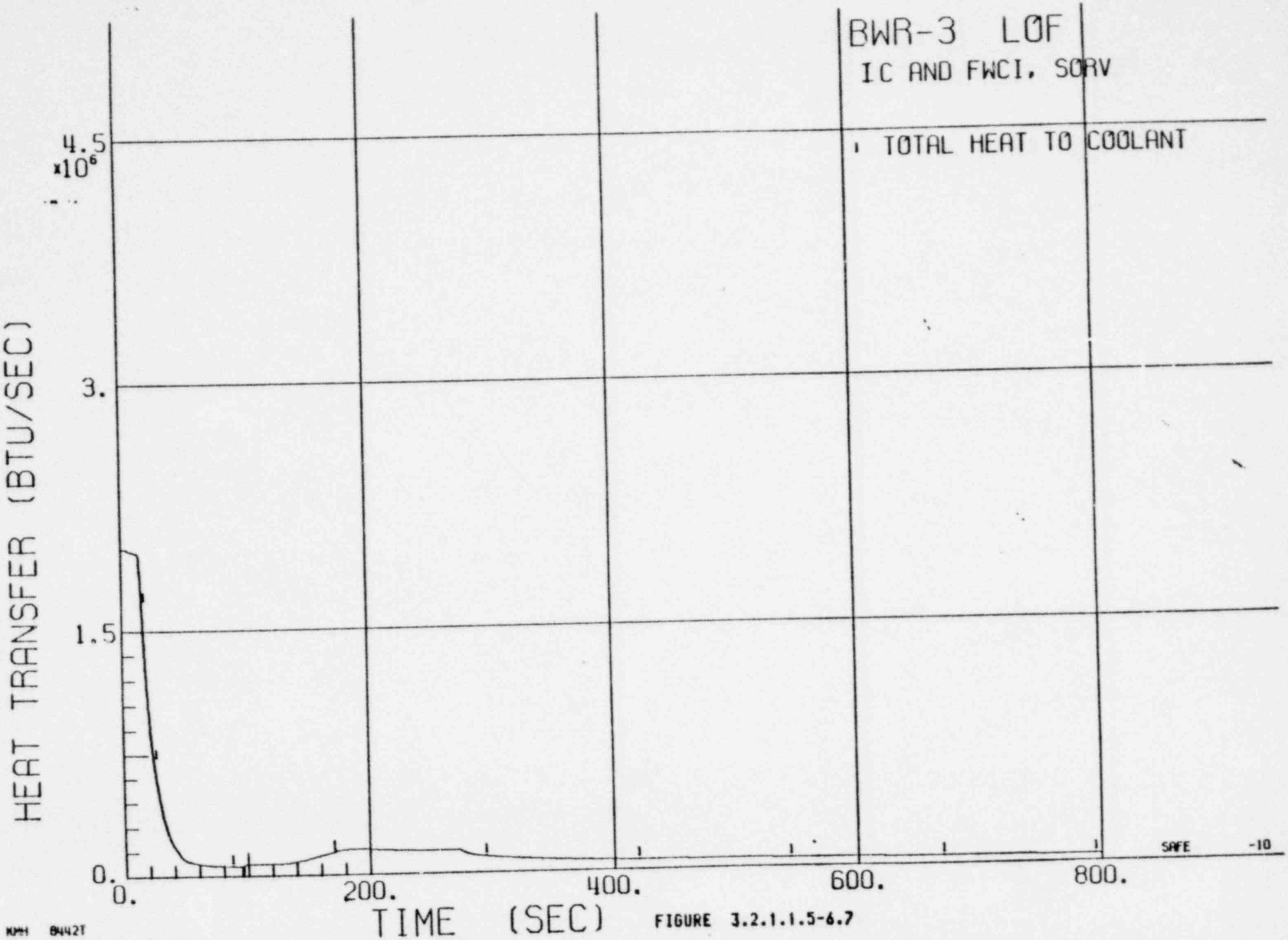


FIGURE 3.2.1.1.5-6.7

KMH 8442T

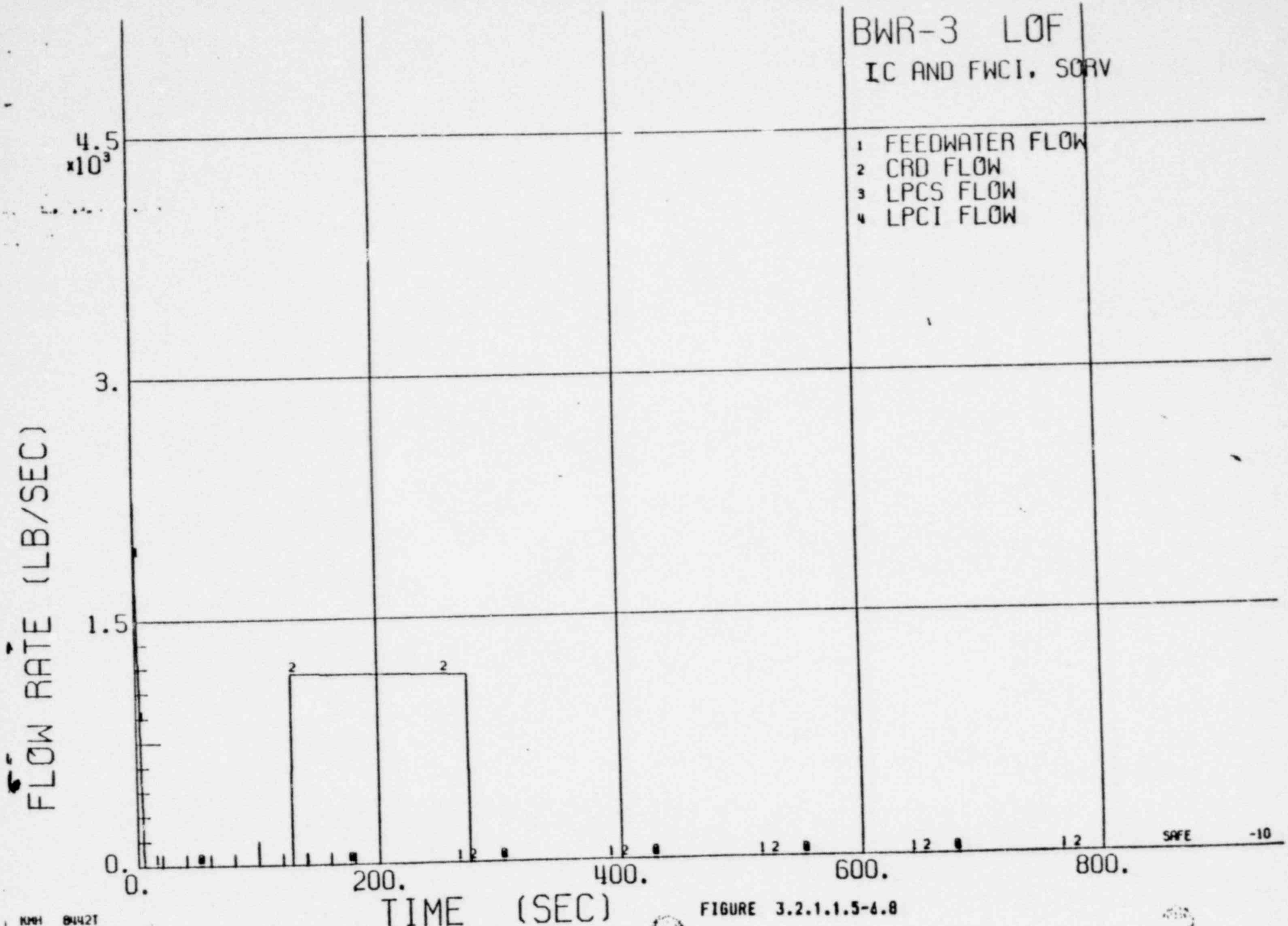


FIGURE 3.2.1.1.5-6.8

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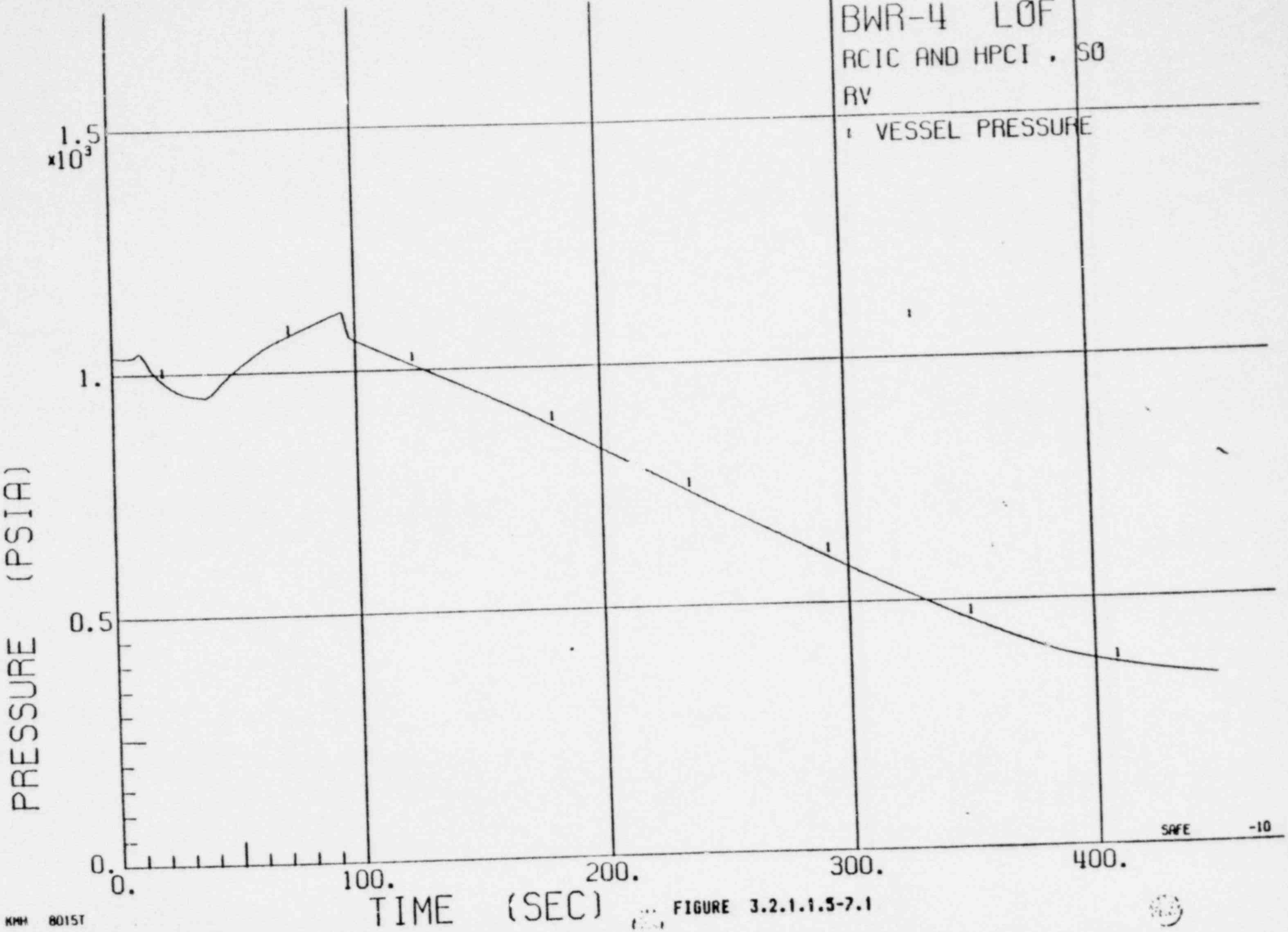


FIGURE 3.2.1.1.5-7.1

KMH 8015T

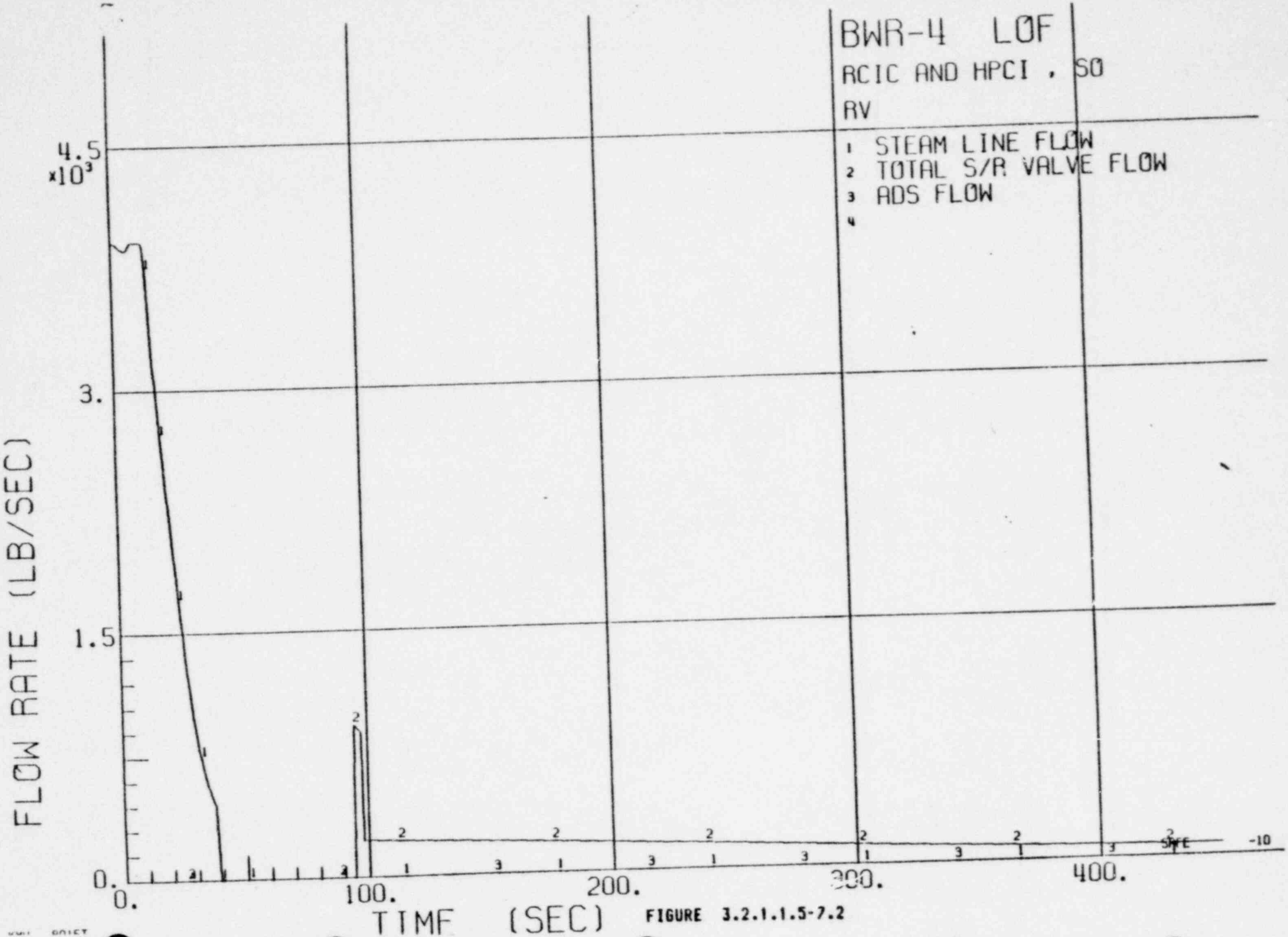


FIGURE 3.2.1.1.5-7.2

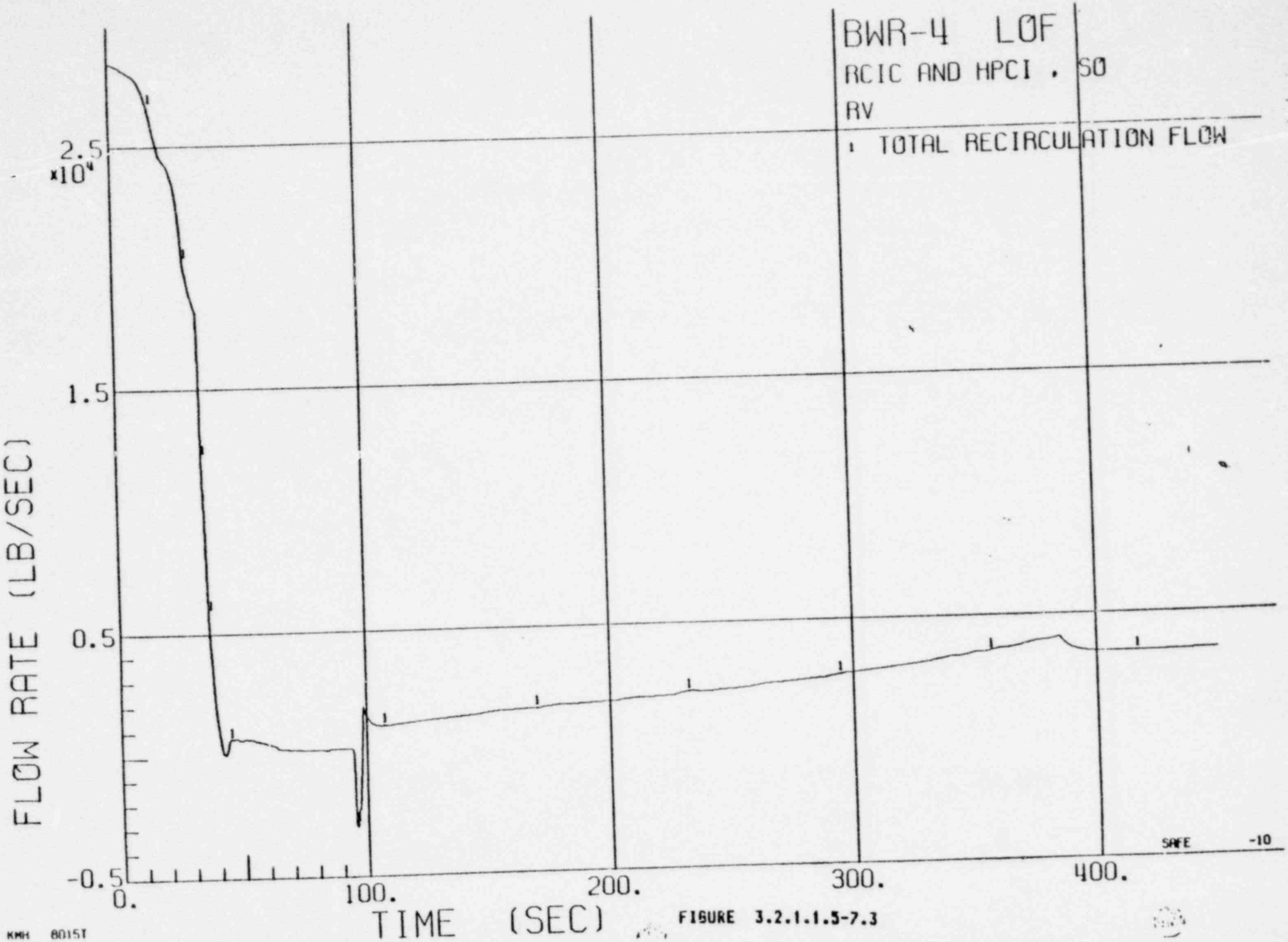
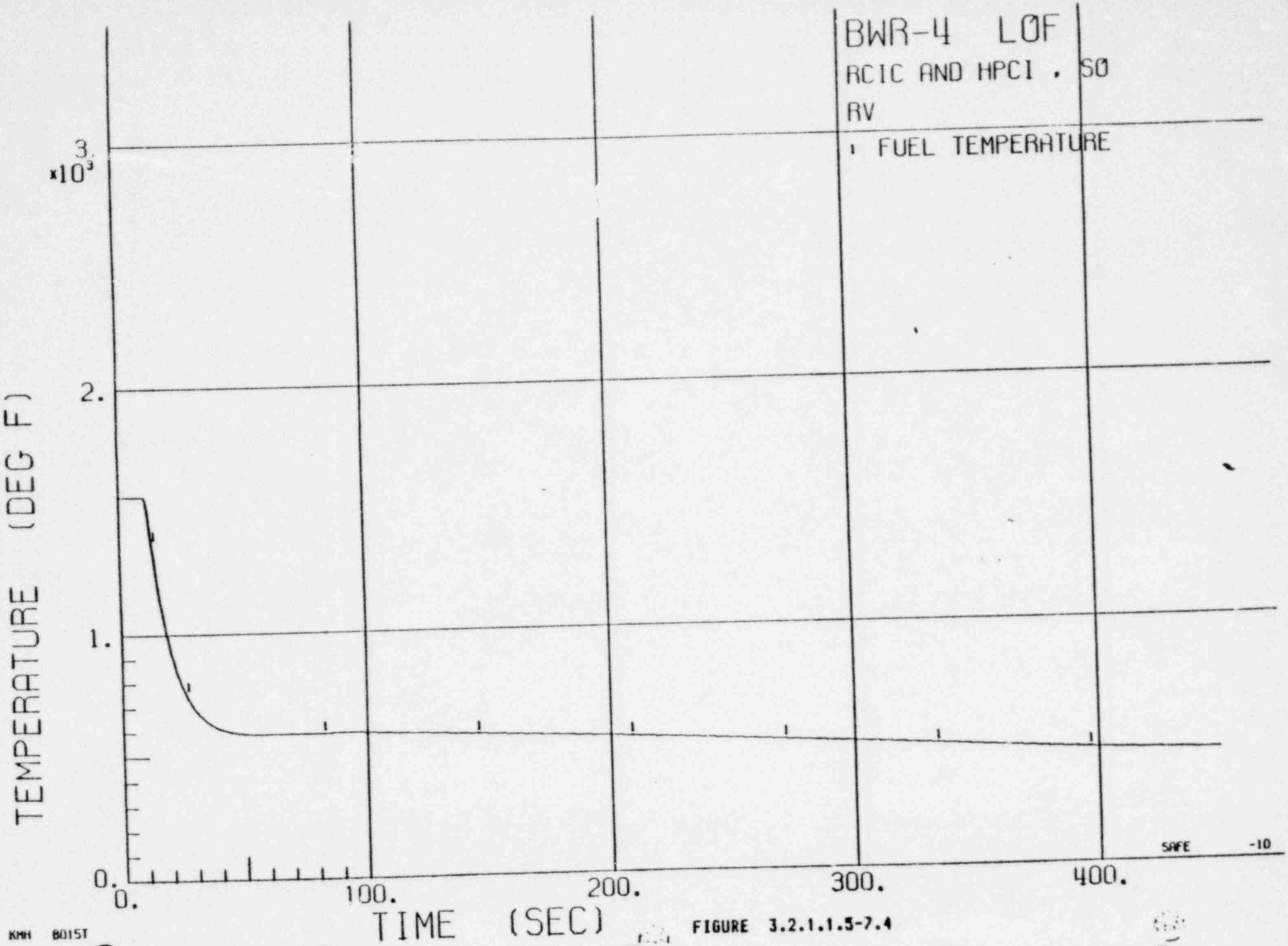


FIGURE 3.2.1.1.5-7.3

KMH 8015T



KMH 8015T

FIGURE 3.2.1.1.5-7.4

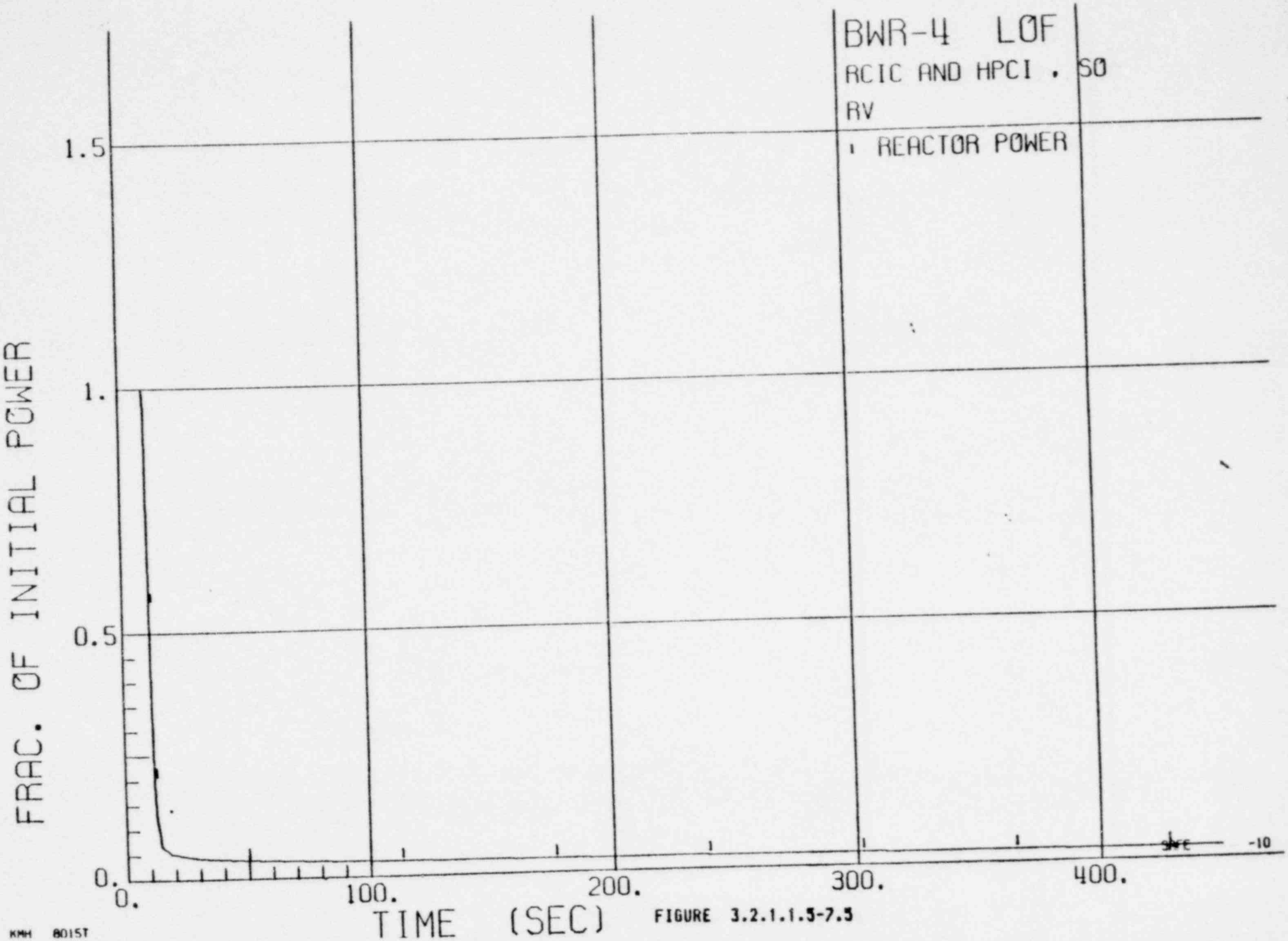


FIGURE 3.2.1.1.5-7.5

KMH 8015T

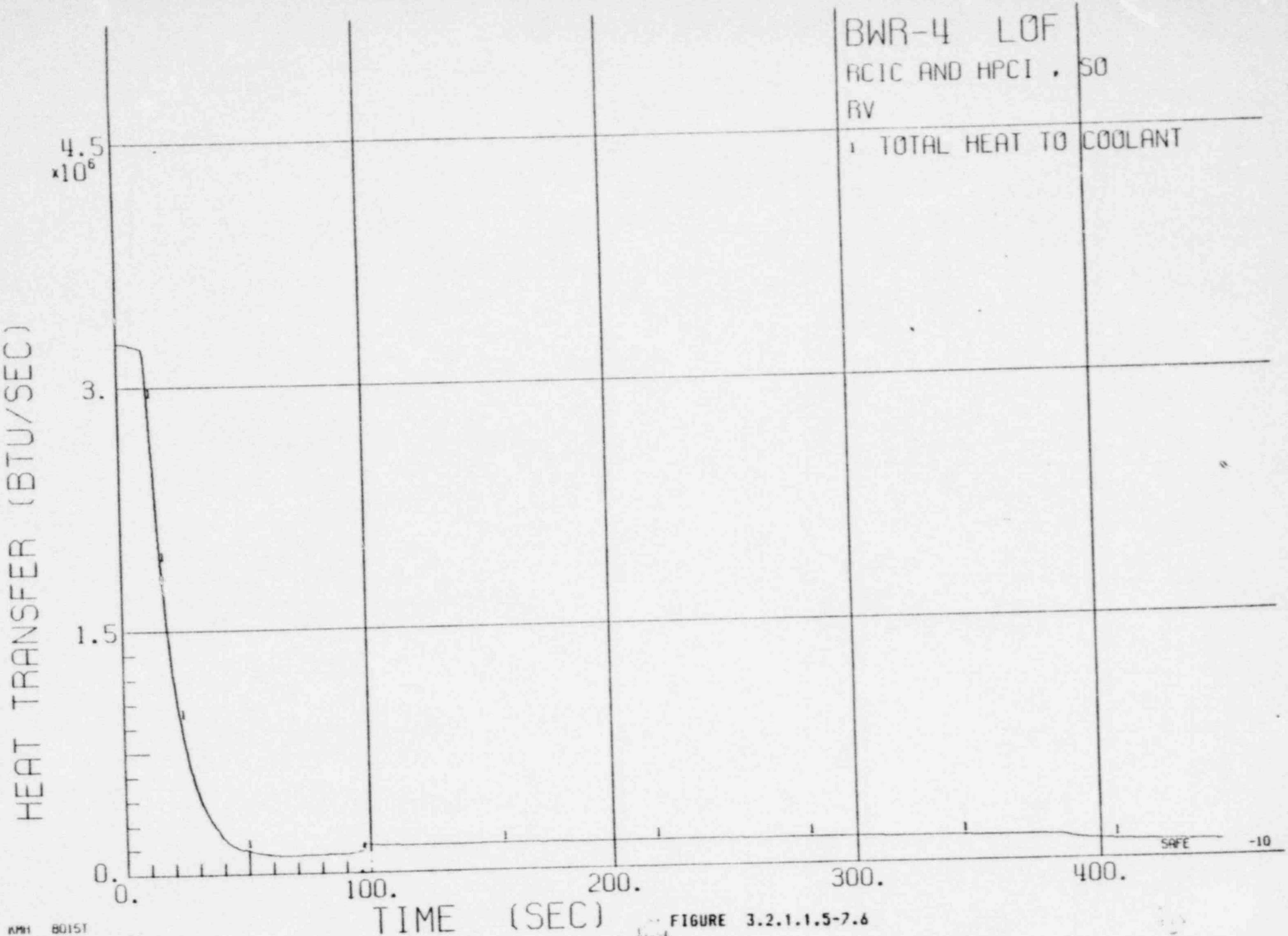


FIGURE 3.2.1.1.5-7.6

KMH 80151
120579 1919.7

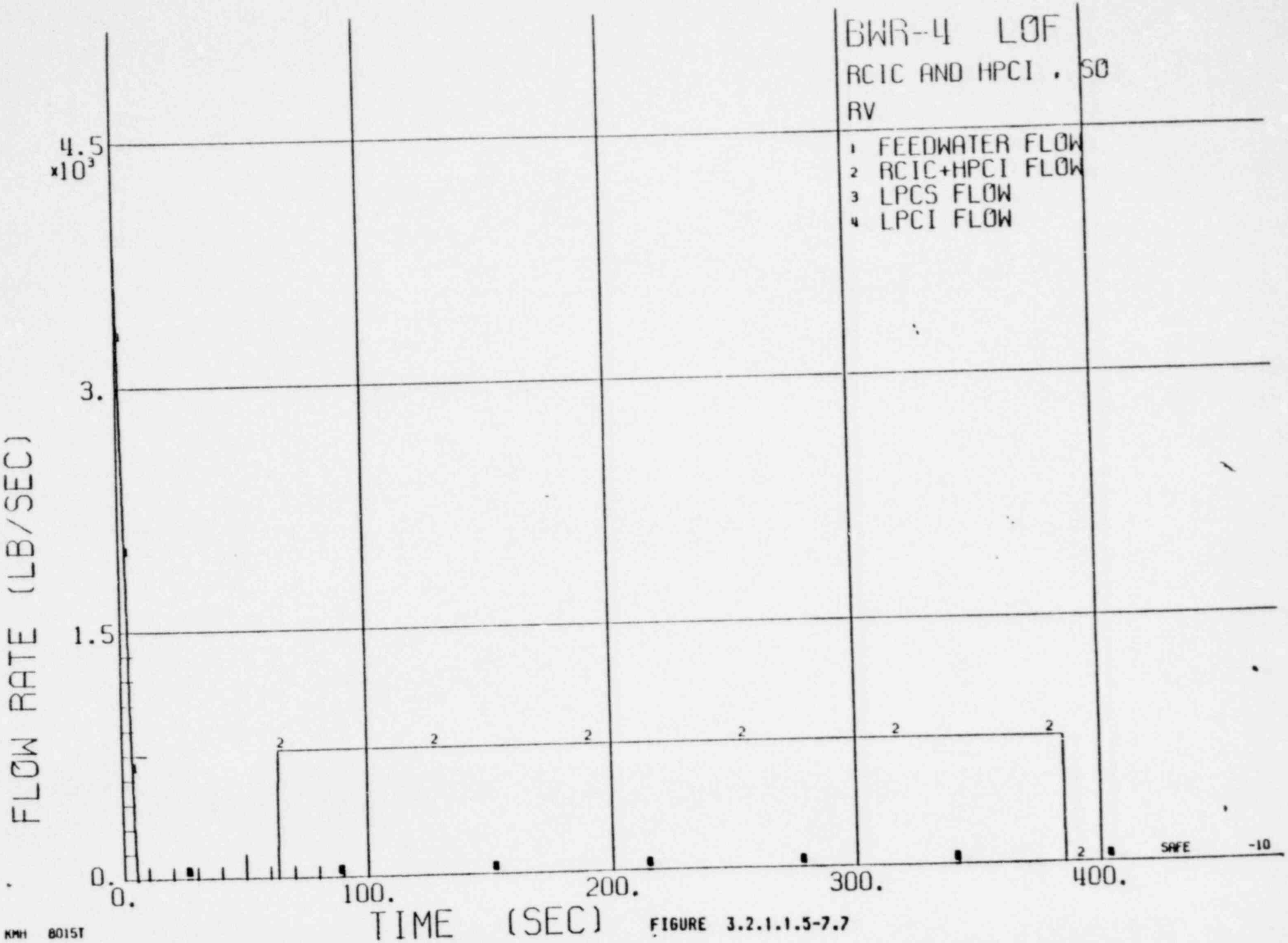
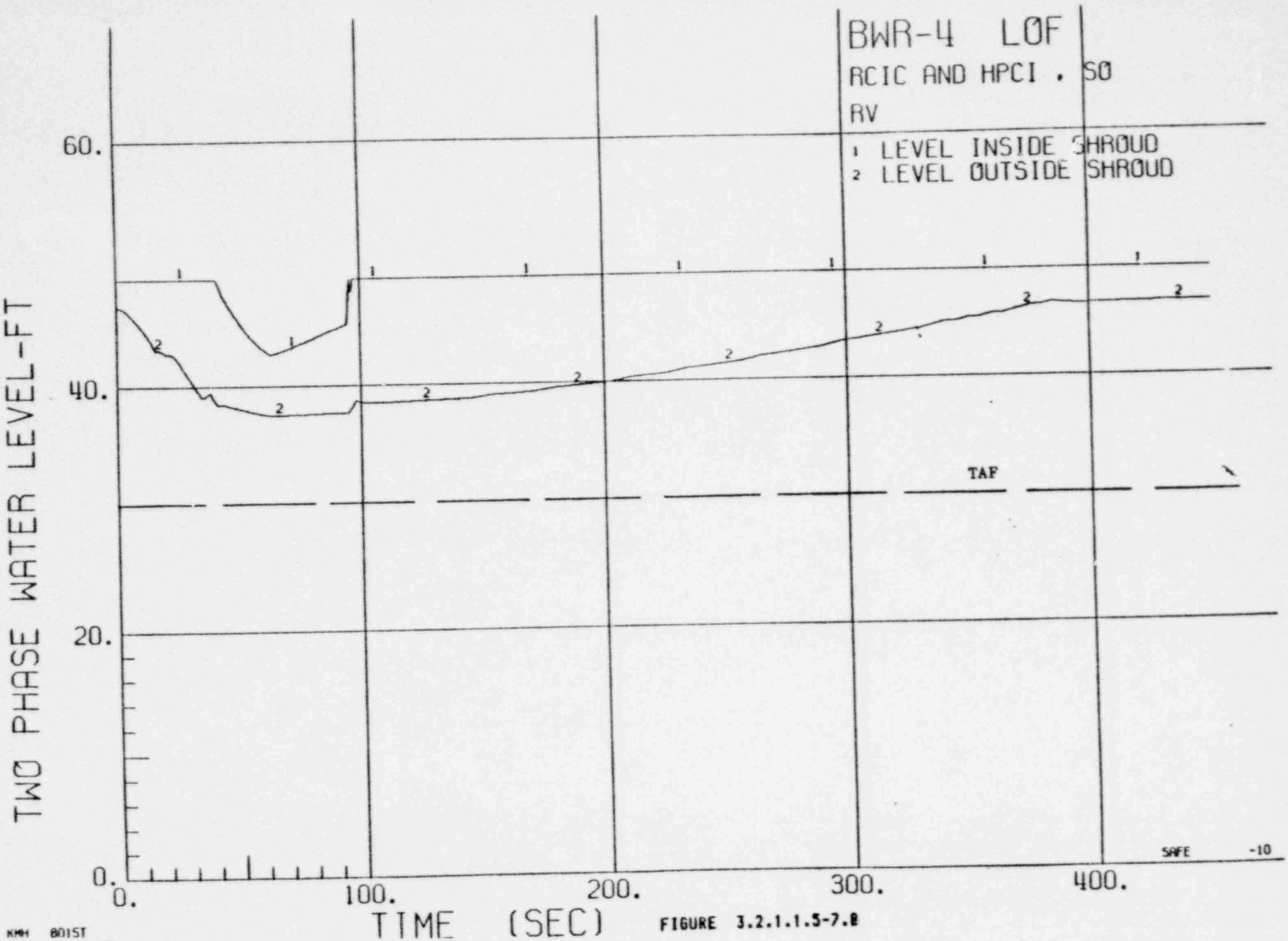


FIGURE 3.2.1.1.5-7.7

KMH 8015T



KMH 8015T

FIGURE 3.2.1.1.5-7.B

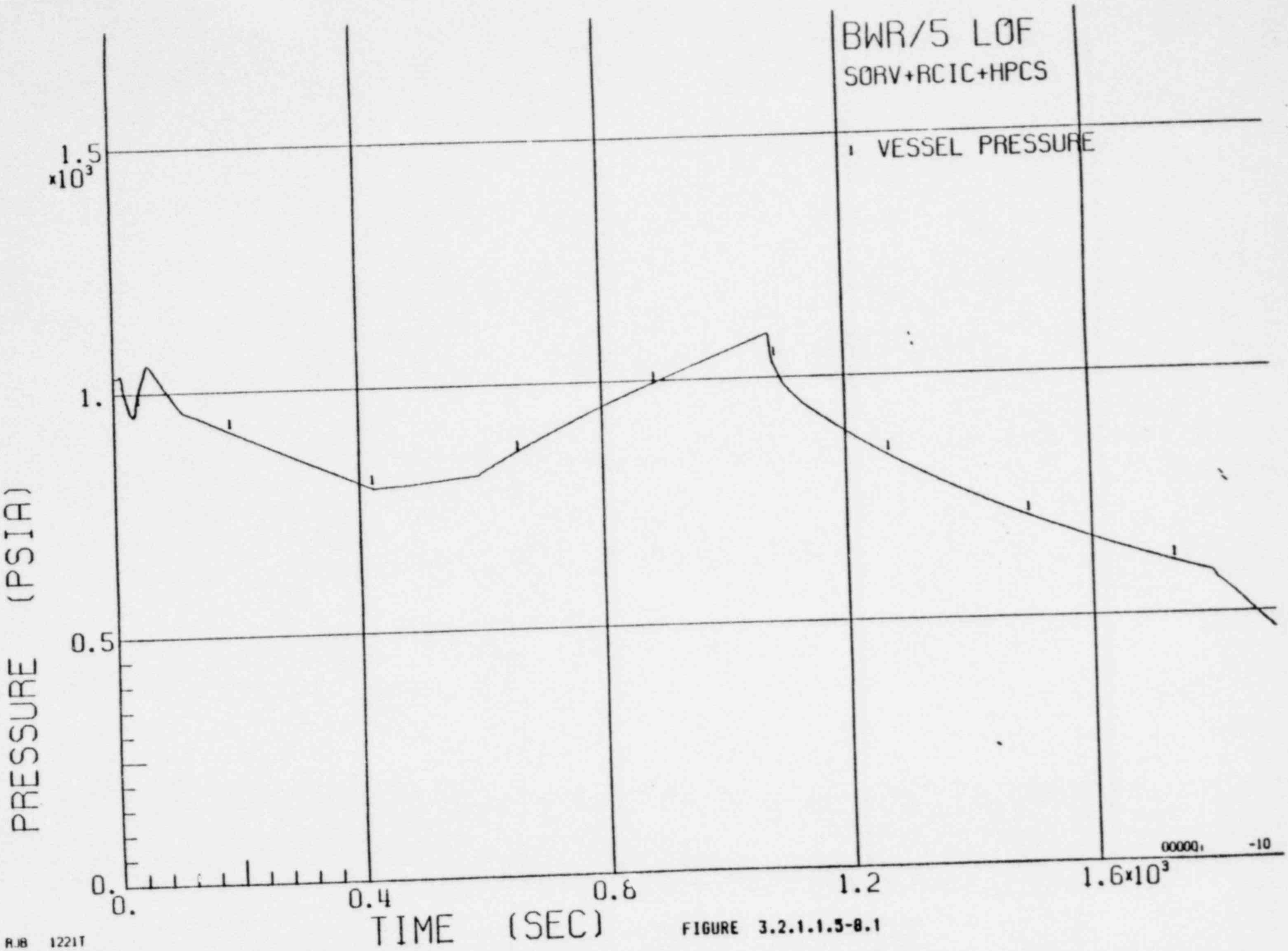


FIGURE 3.2.1.1.5-8.1

RJB 1221T

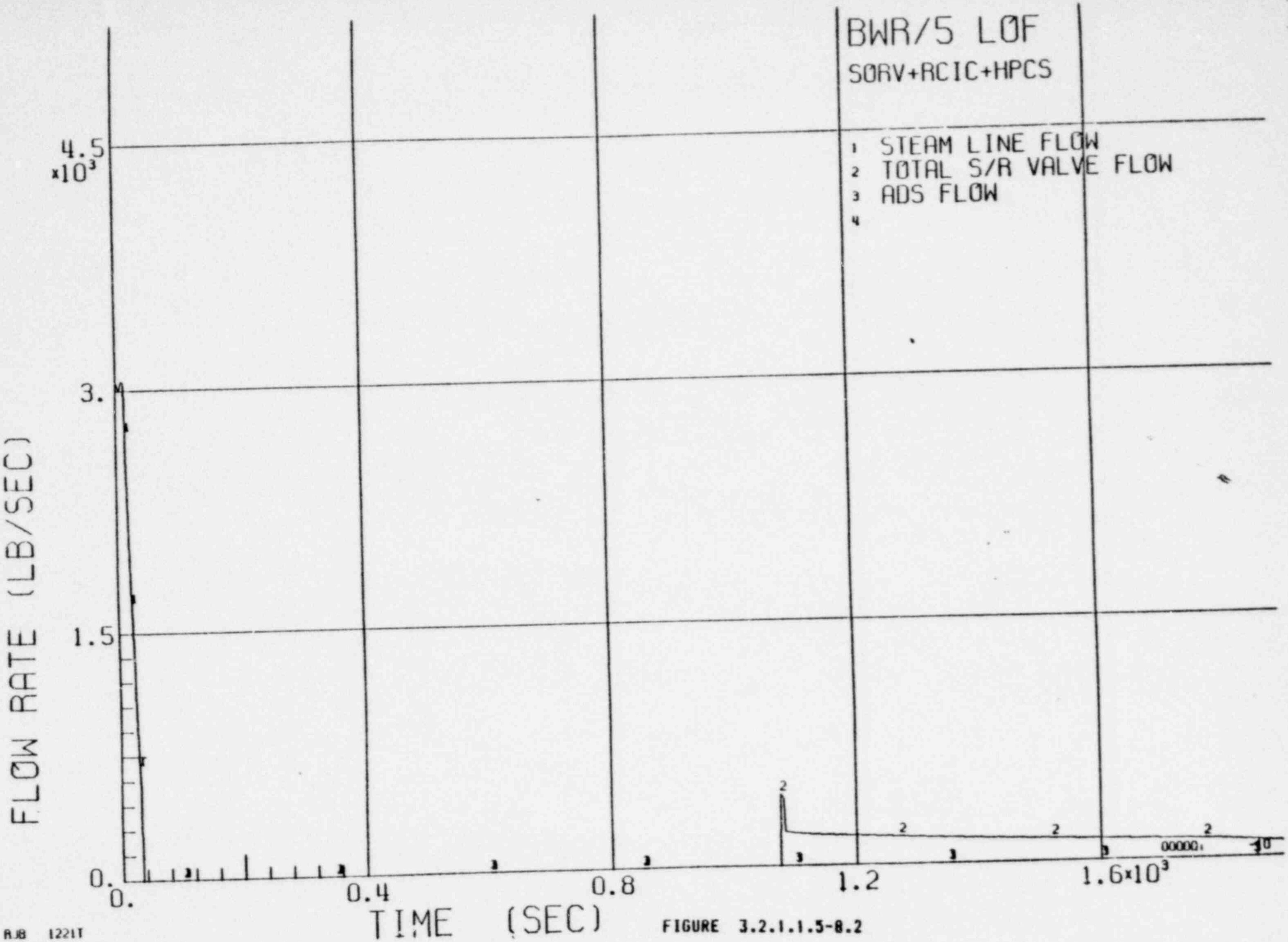


FIGURE 3.2.1.1.5-8.2

RJB 1221T
121379 0015.8

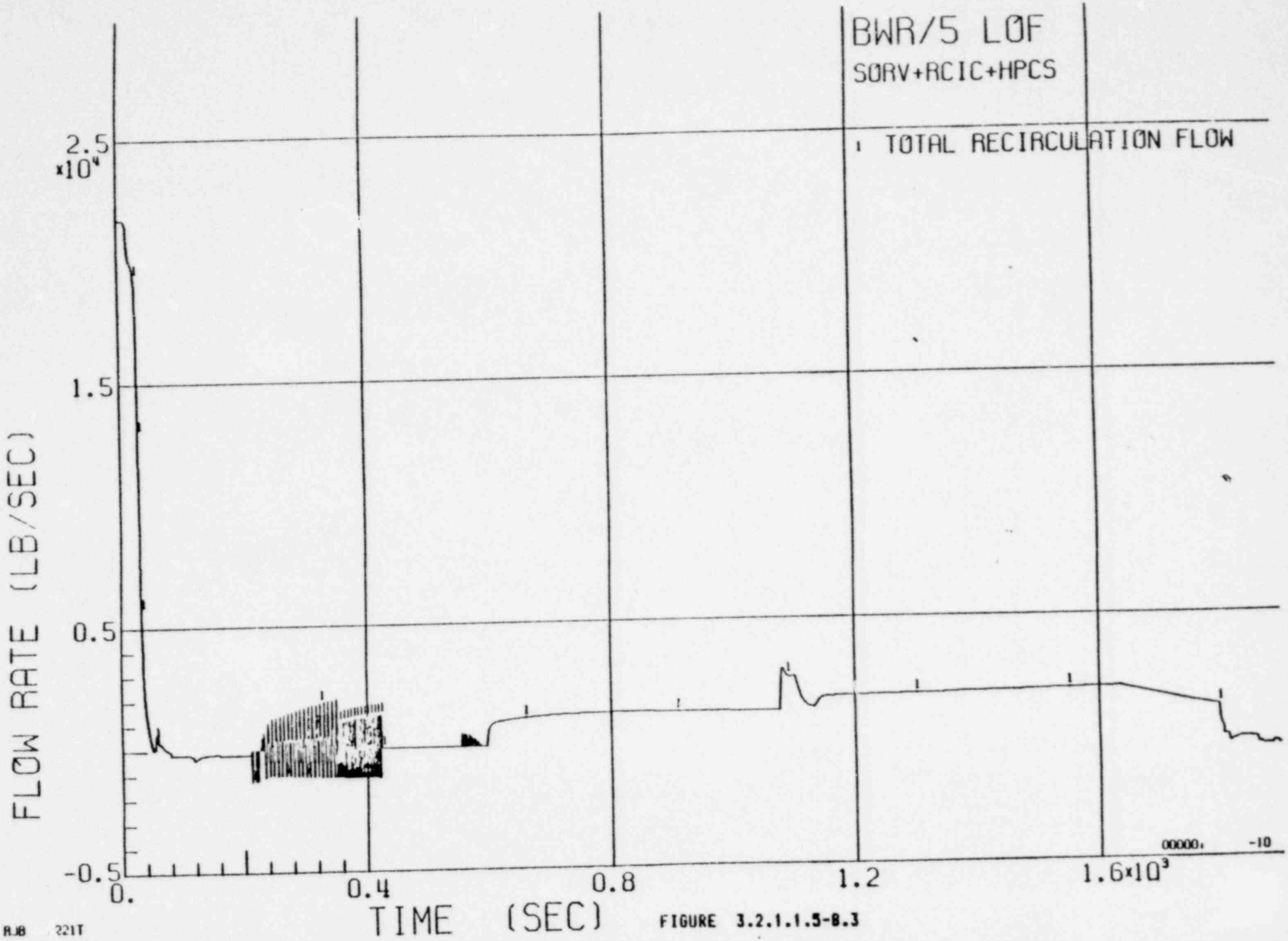


FIGURE 3.2.1.1.5-B.3

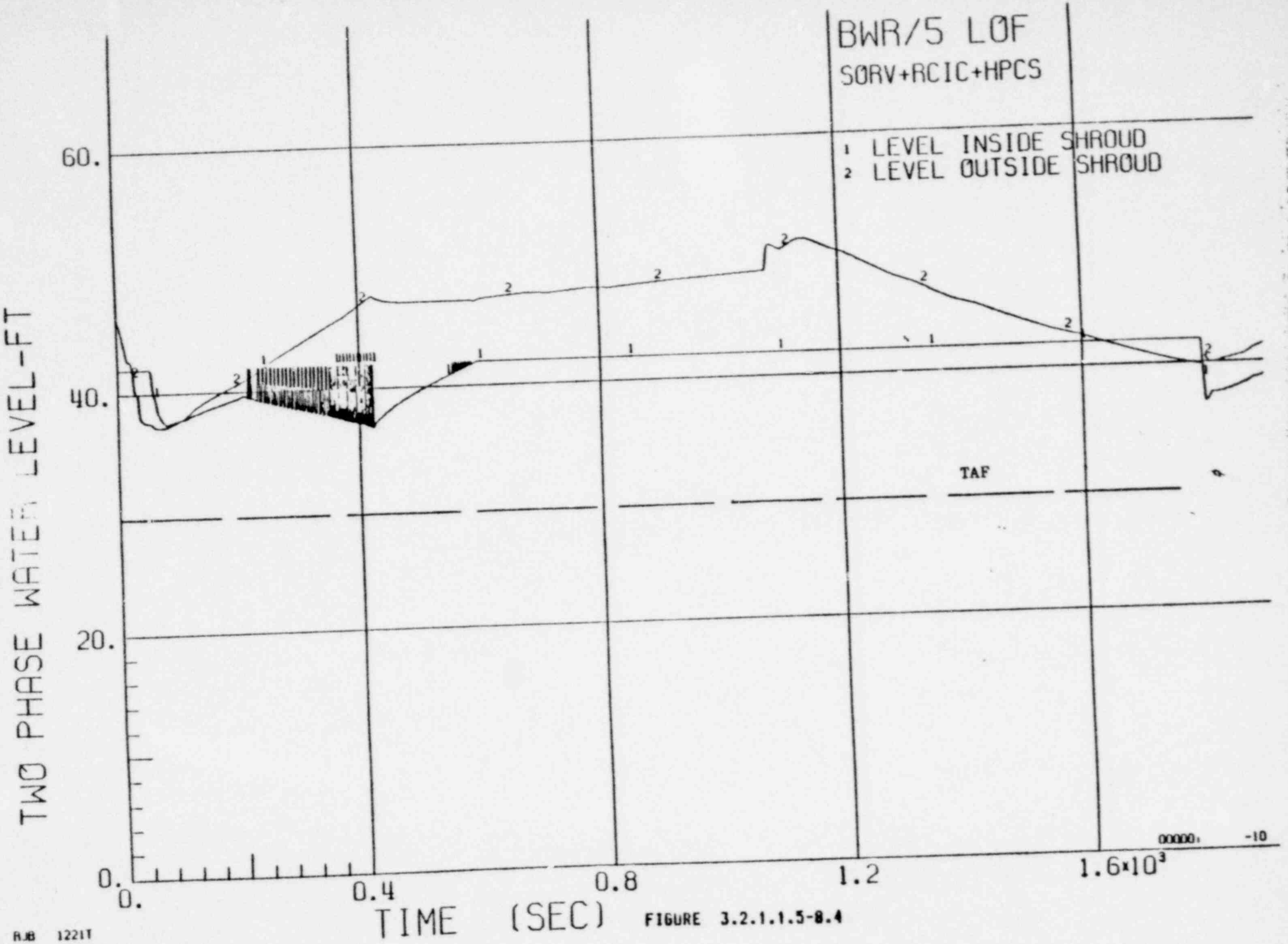


FIGURE 3.2.1.1.5-8.4

RJB 1221T

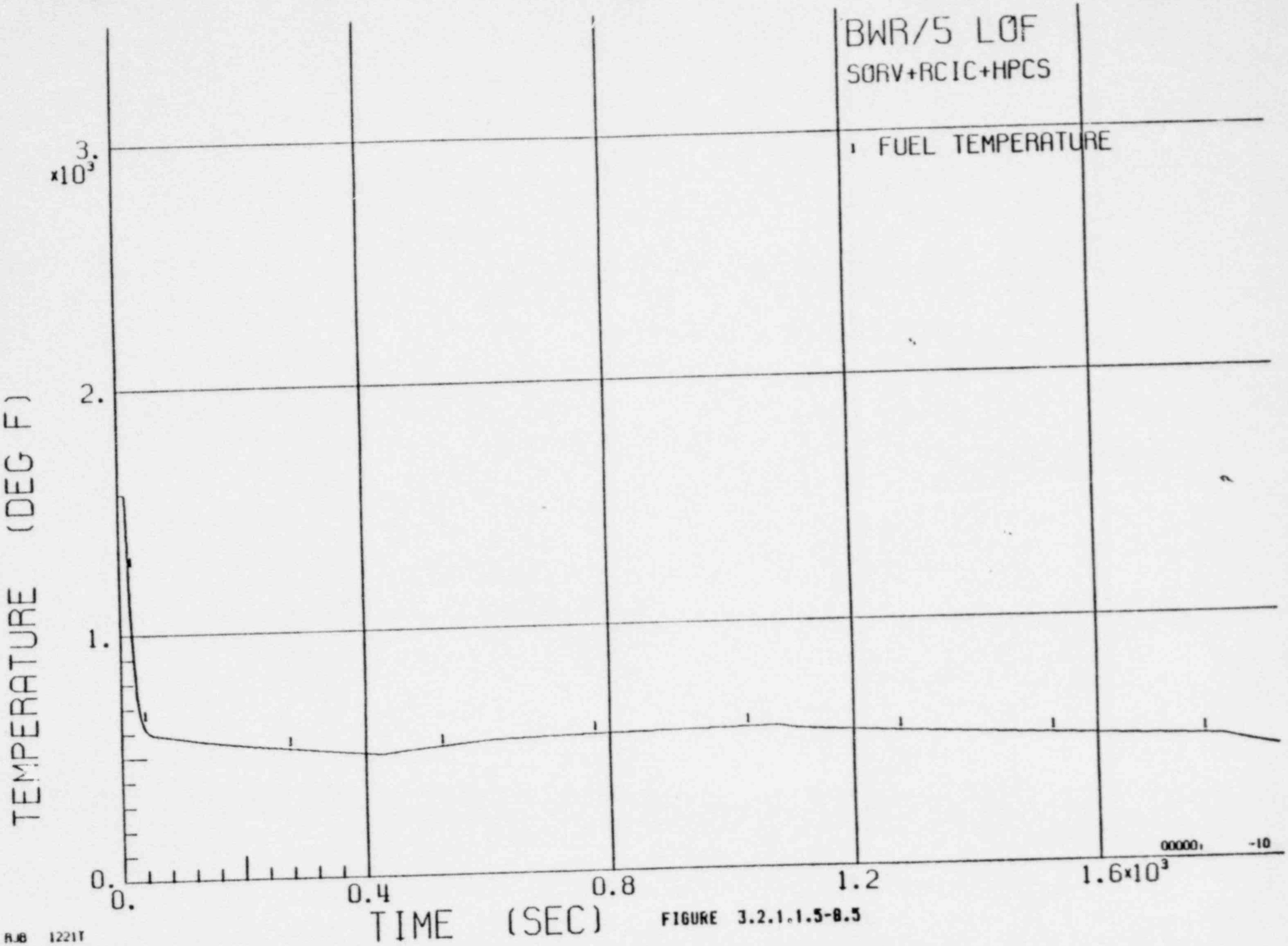


FIGURE 3.2.1.1.5-8.5

RJB 1221T
121379 0015 8

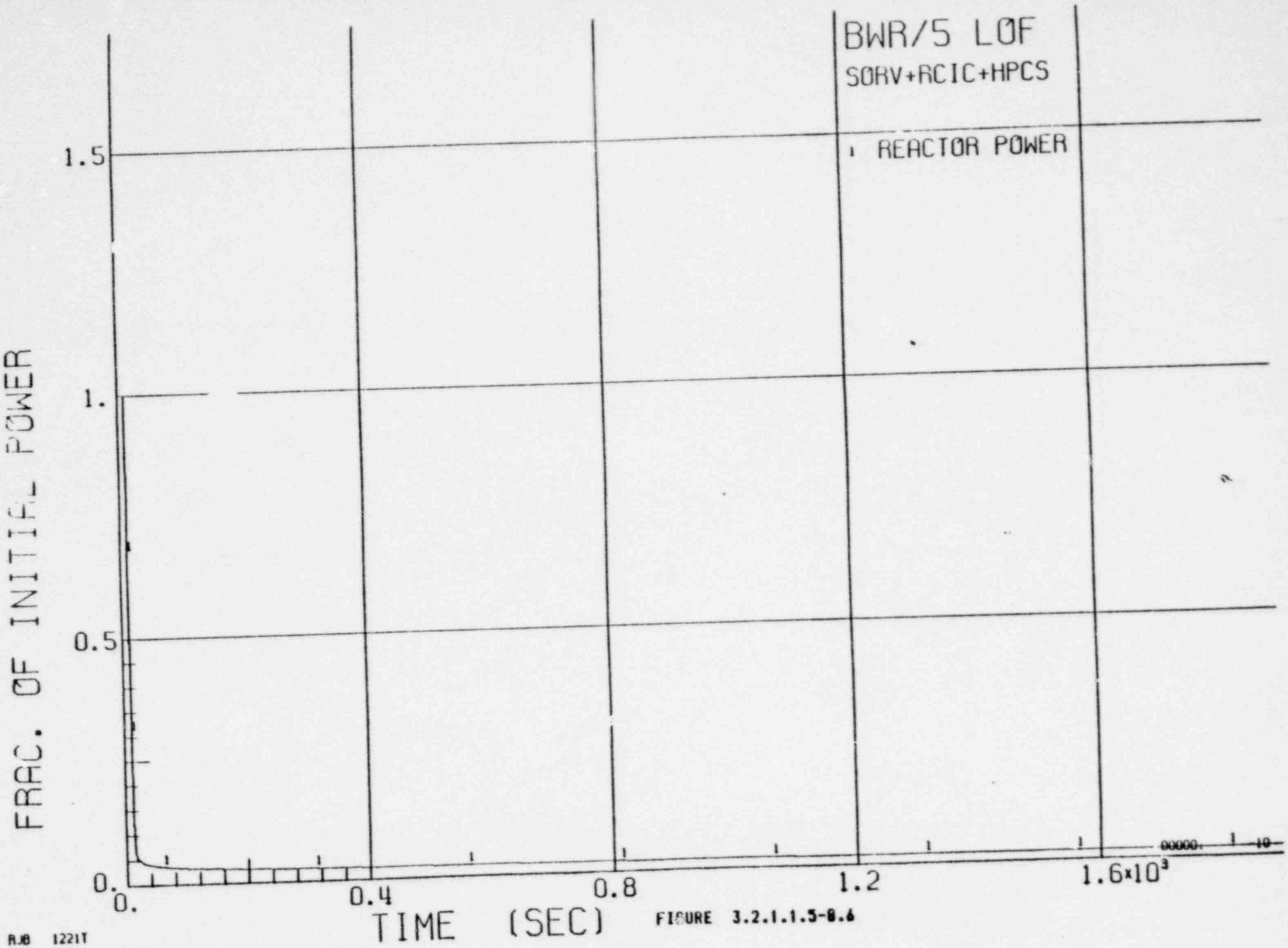


FIGURE 3.2.1.1.5-8.6

RJB 1221T

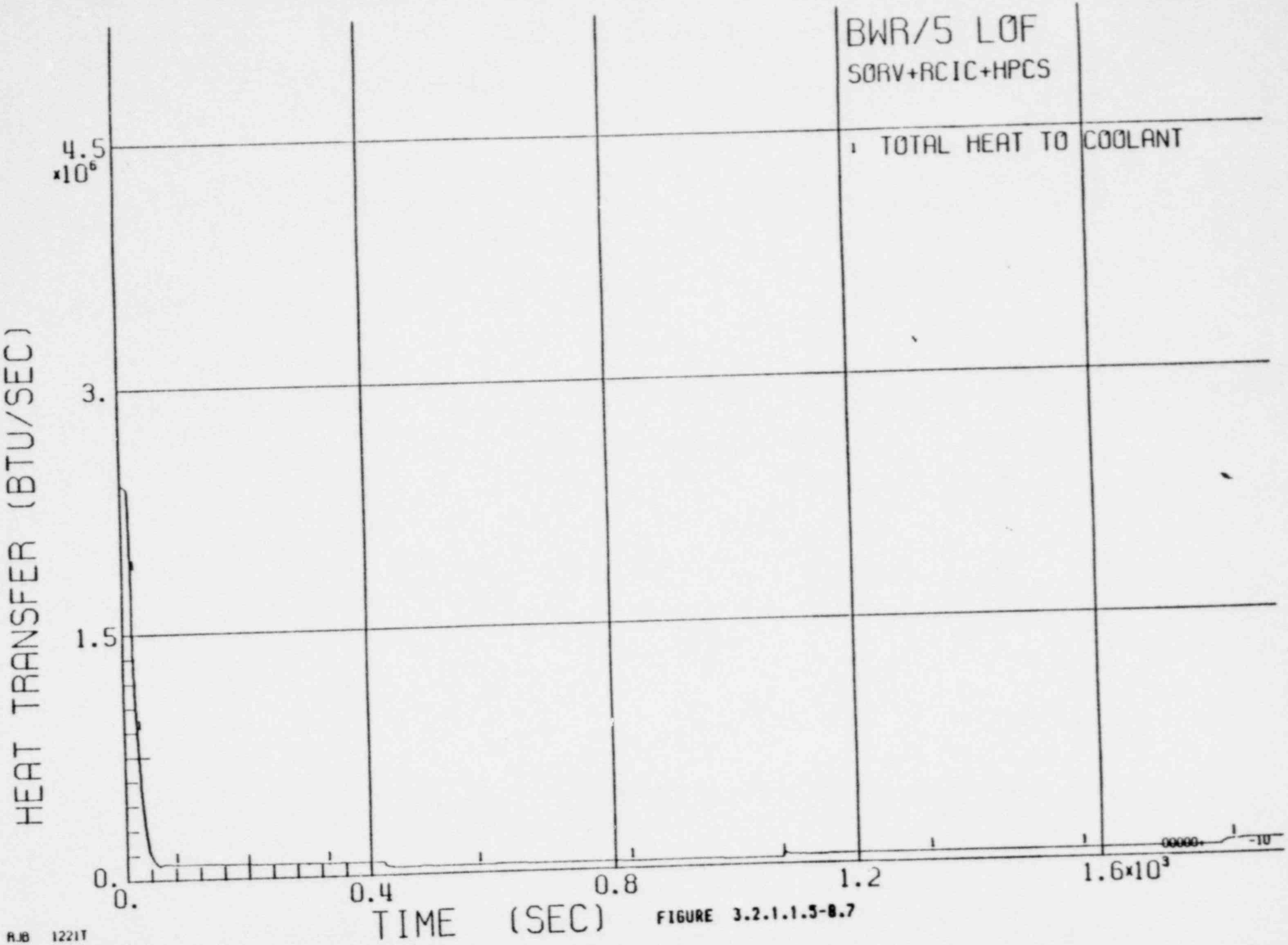


FIGURE 3.2.1.1.5-8.7

RJB 1221T

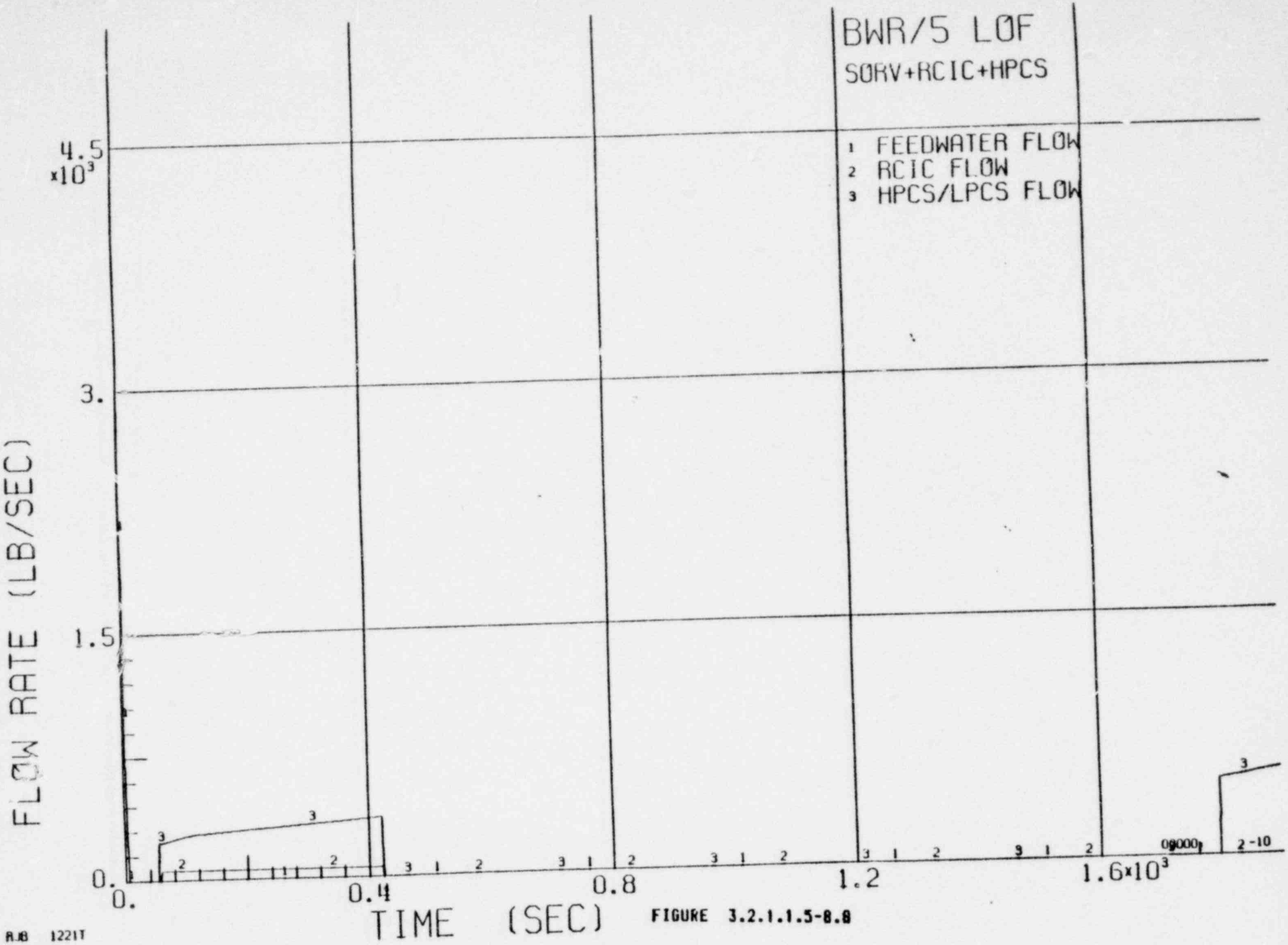


FIGURE 3.2.1.1.5-8.8

RJB 1221T

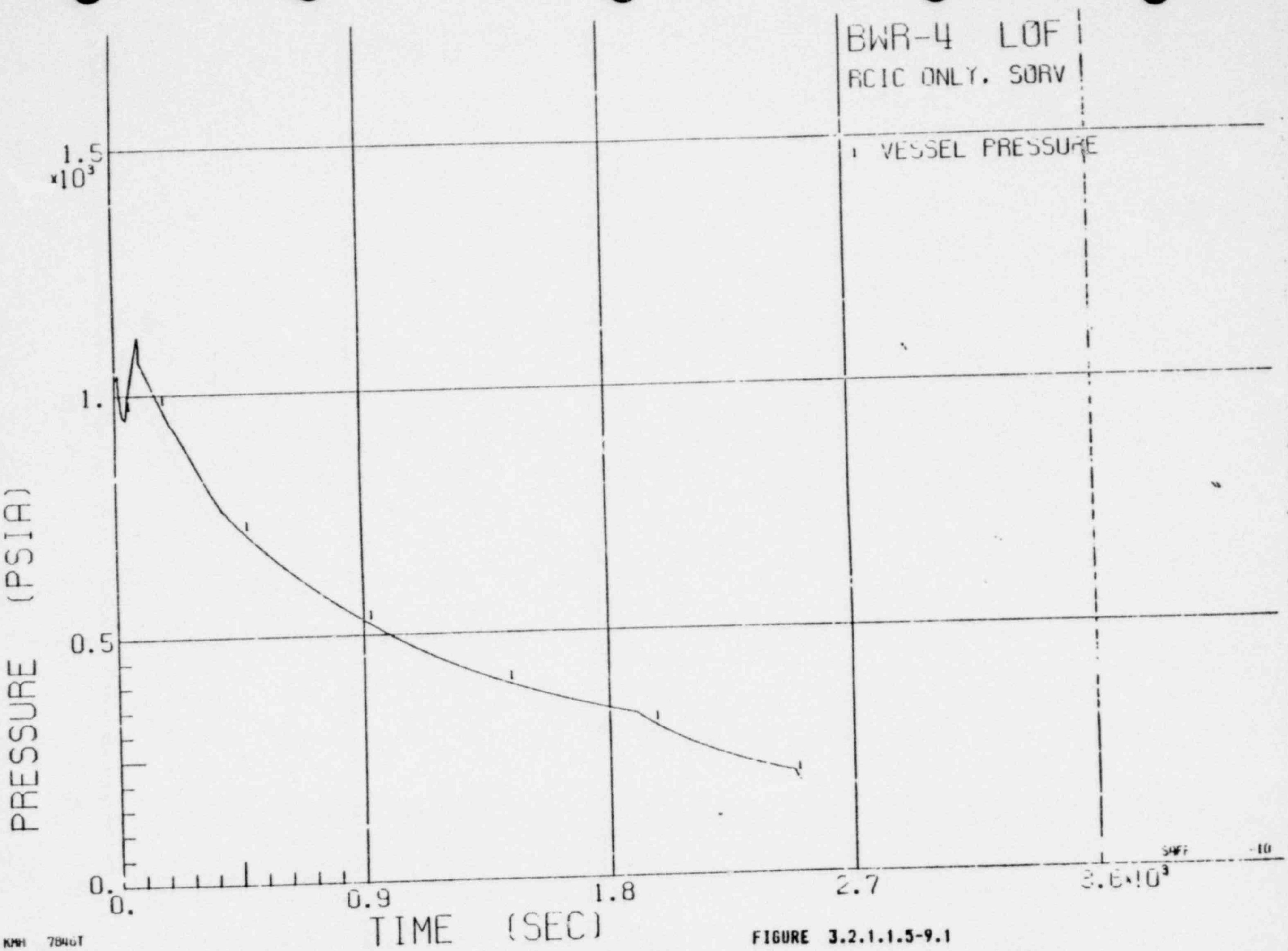
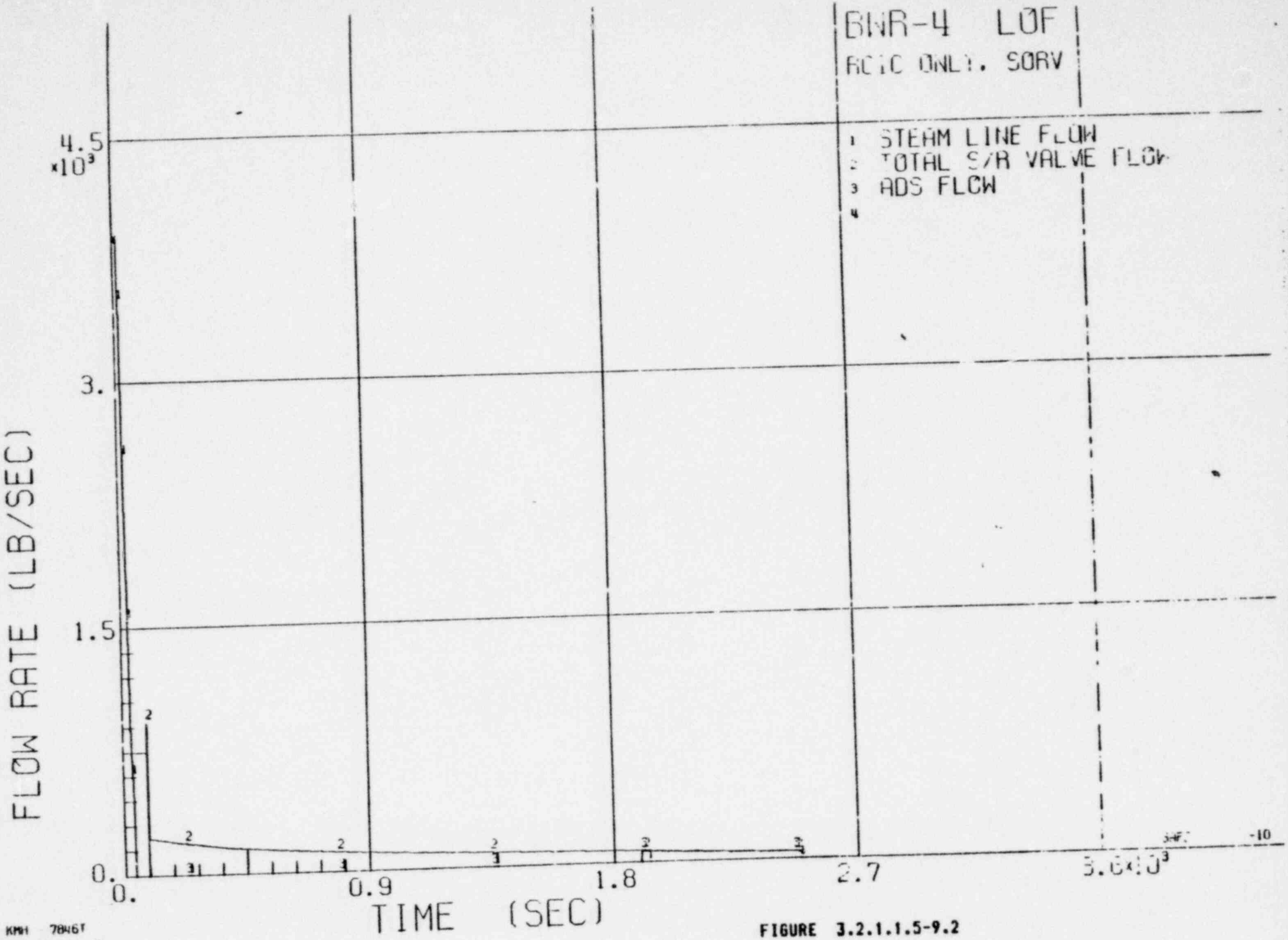


FIGURE 3.2.1.1.5-9.1

KMH 784GT
010480 2314.5



KMH 7846T
010480 2314.5

FIGURE 3.2.1.1.5-9.2

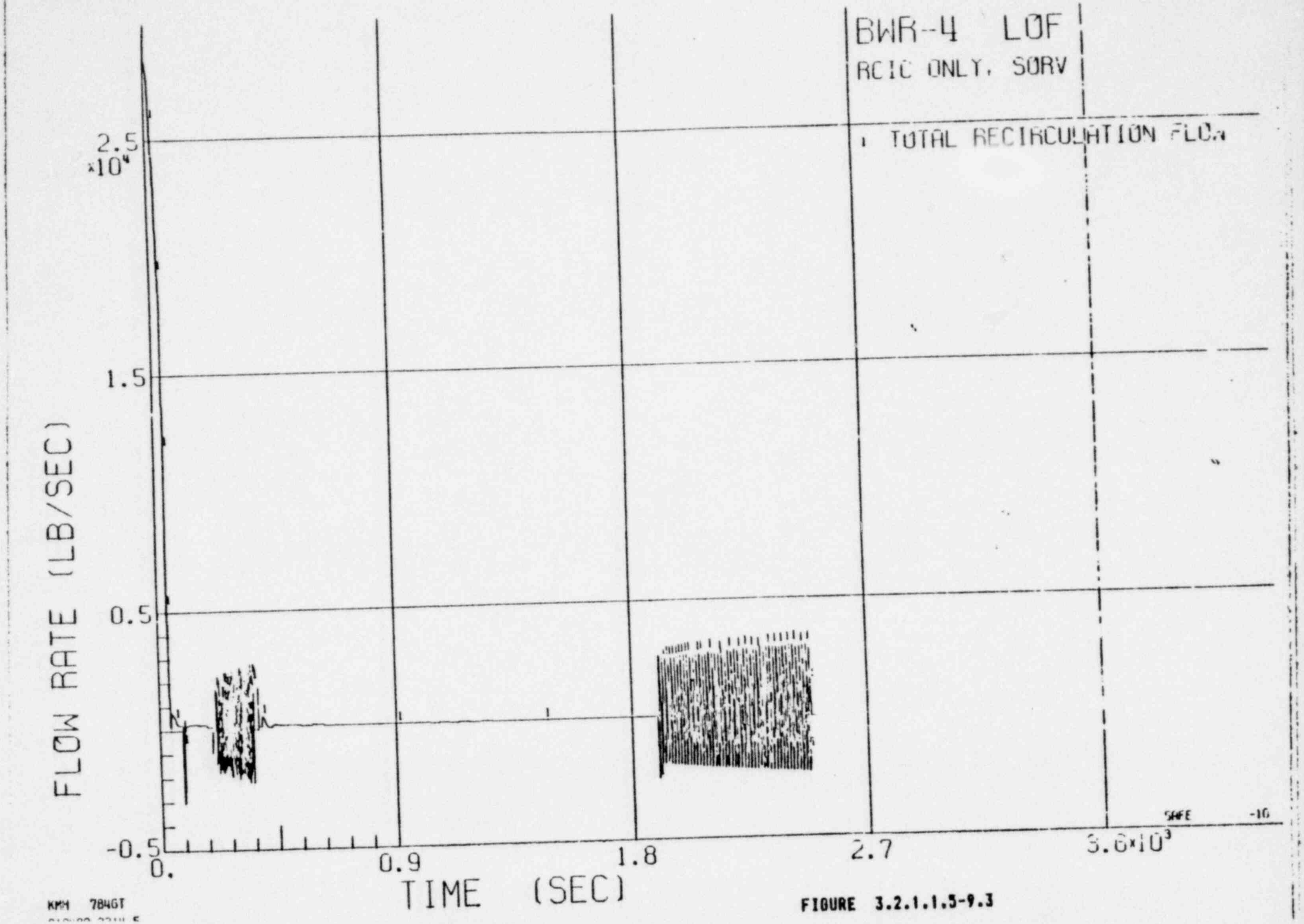
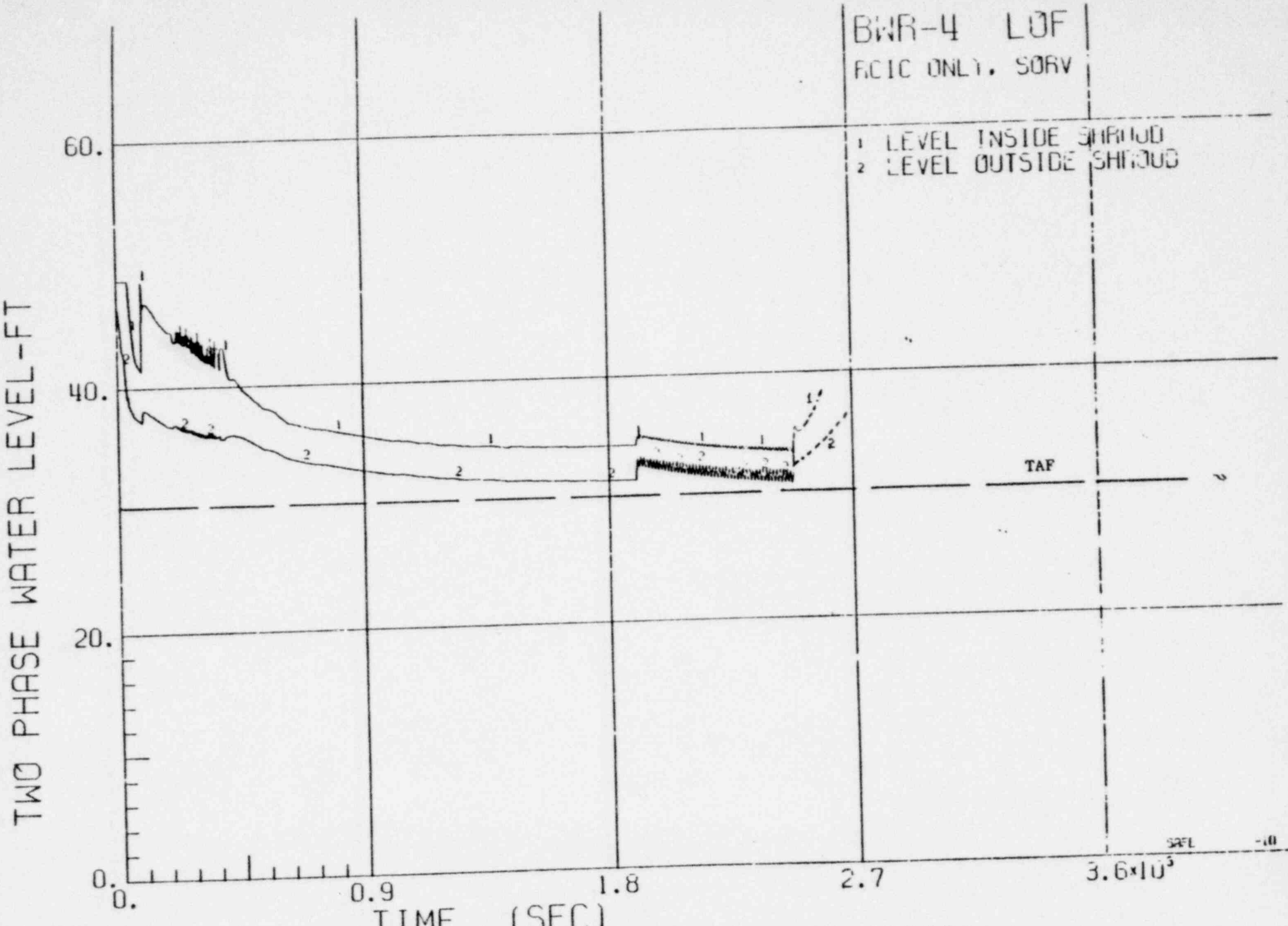
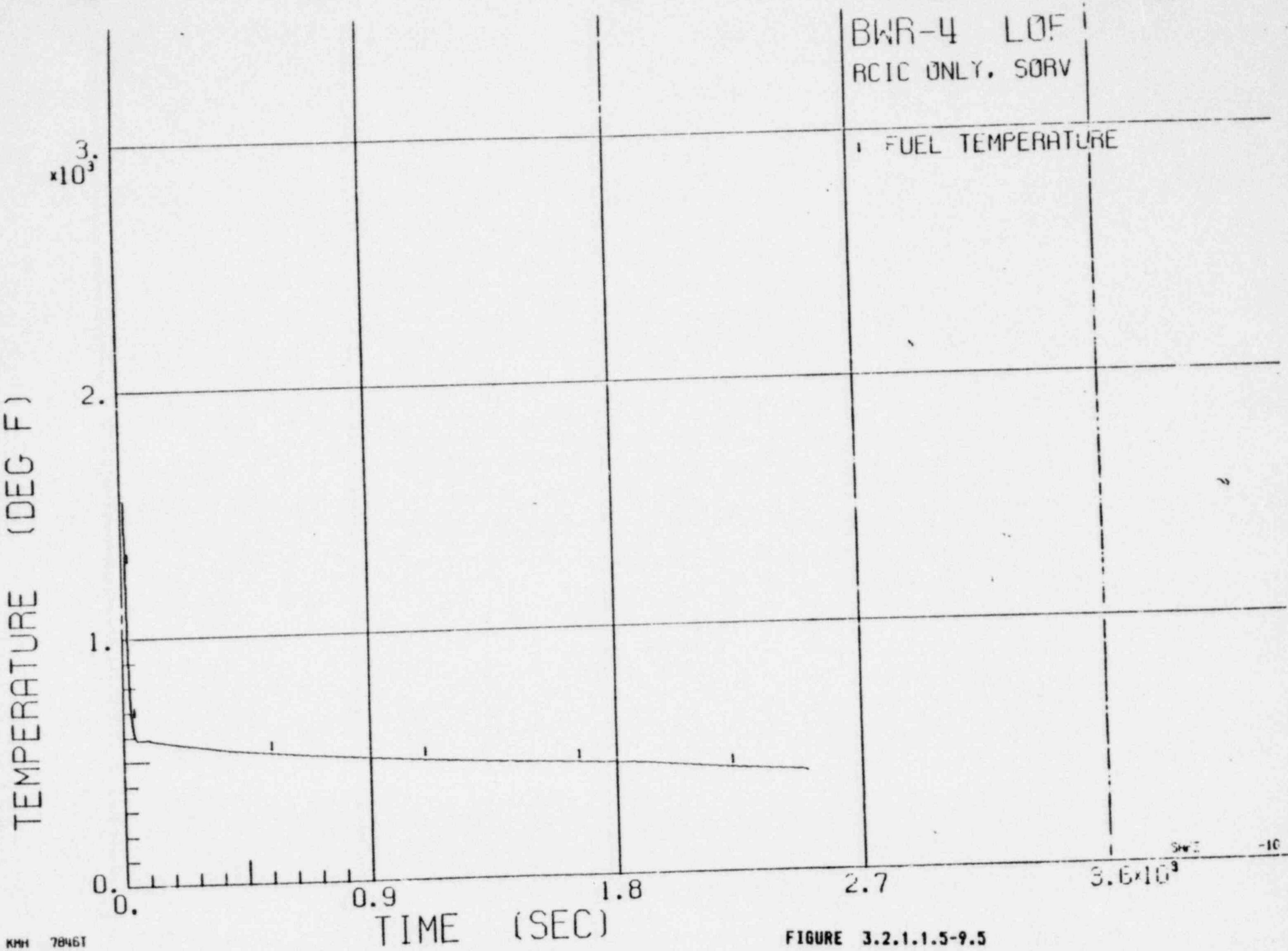


FIGURE 3.2.1.1.5-9.3



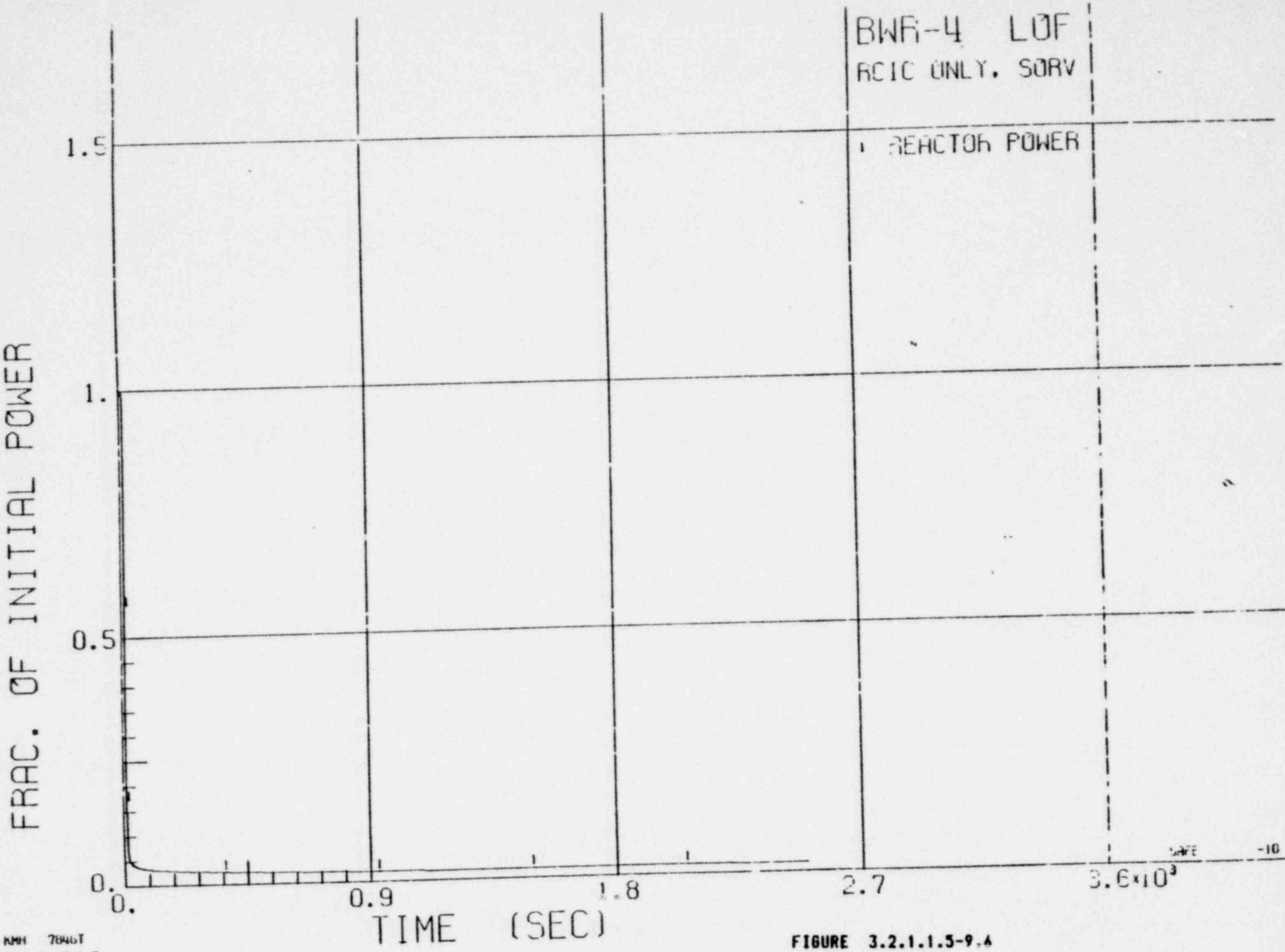
KMH 7846T
010480 2514.5

FIGURE 3.2.1.1.5-9.4



KMH 7846T
010480 2314.5

FIGURE 3.2.1.1.5-9.5



KMH 7846T
010480 2314.5

FIGURE 3.2.1.1.5-9.4

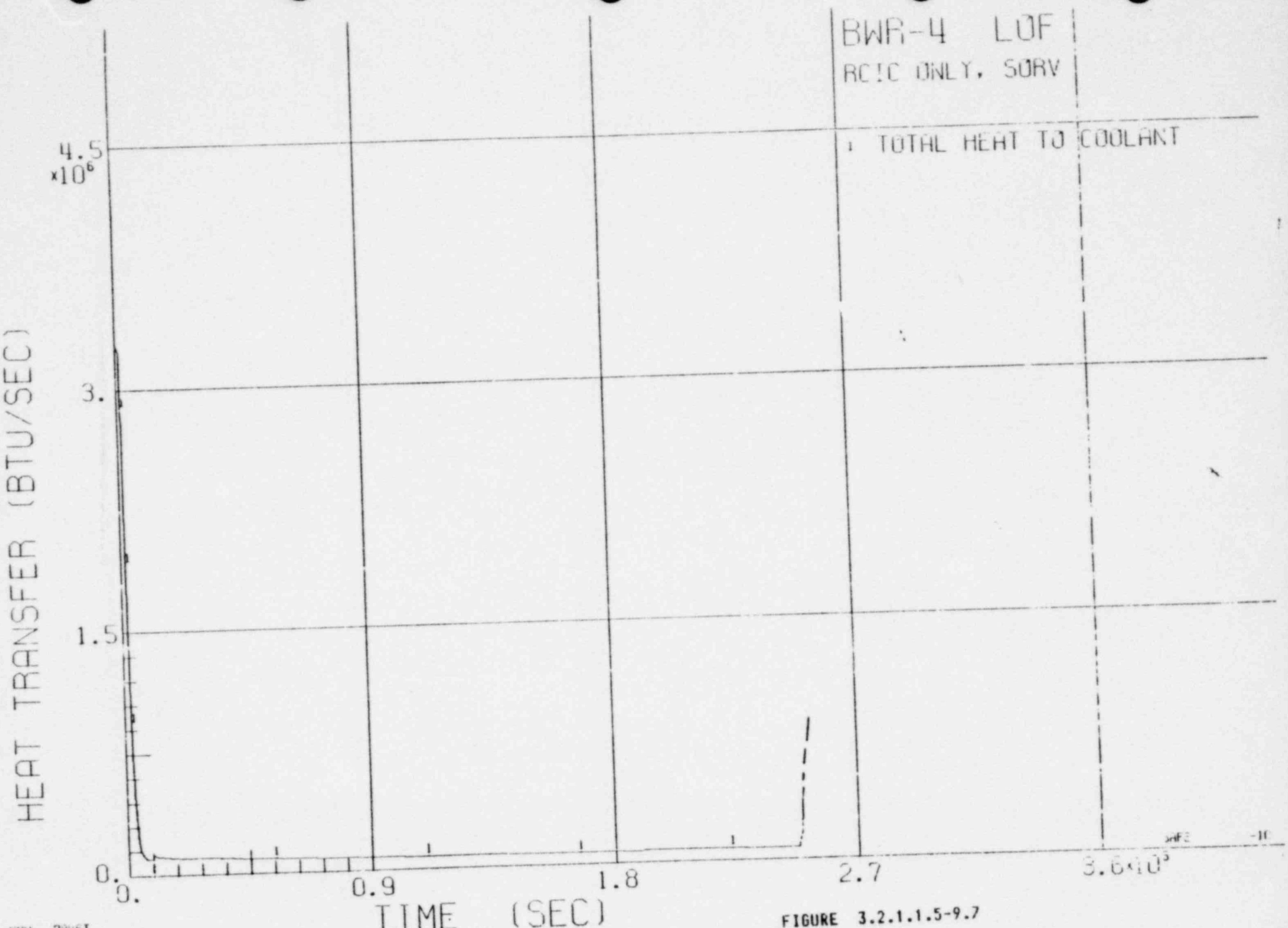
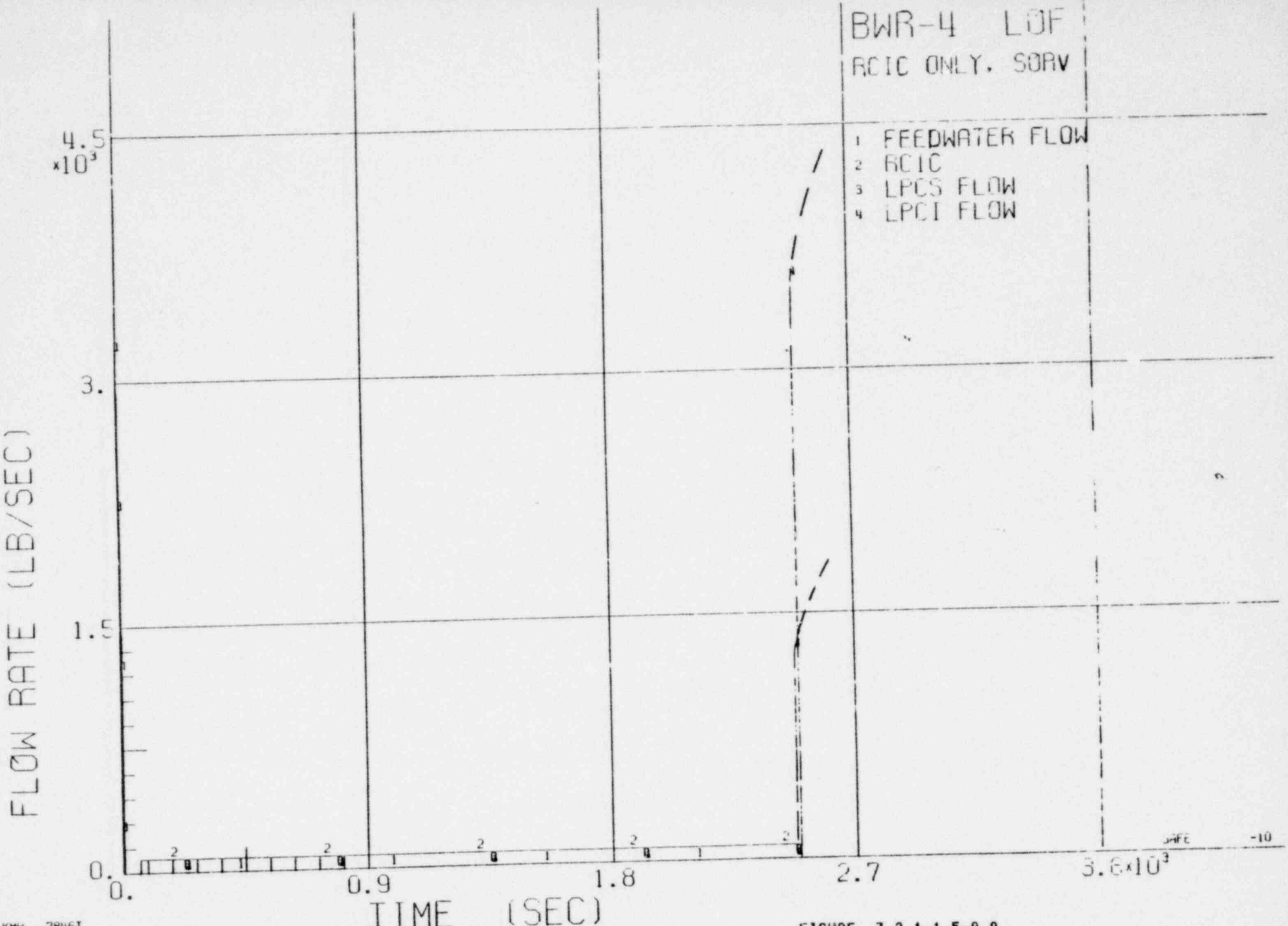


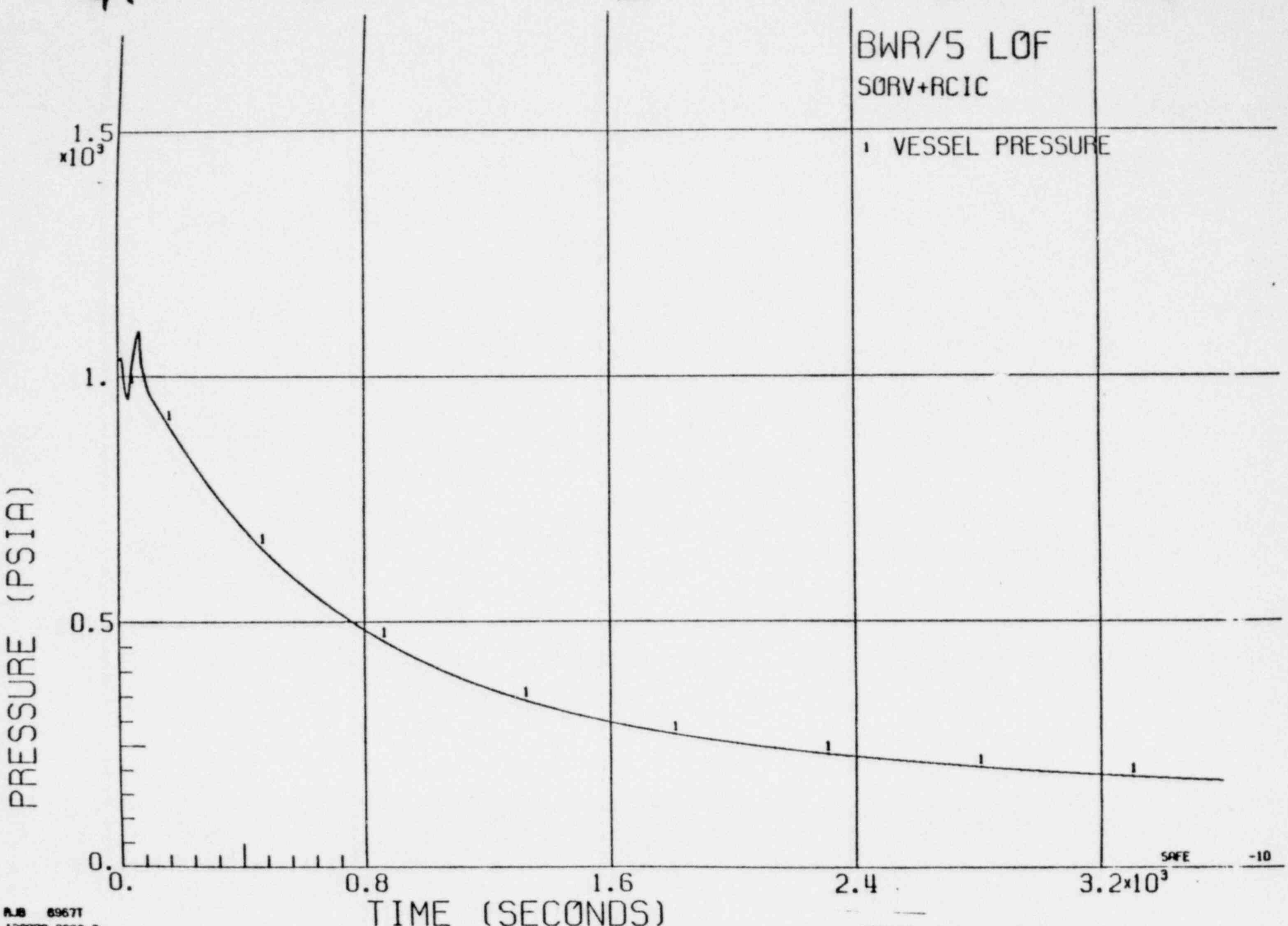
FIGURE 3.2.1.1.5-9.7

KMH 7846T
010480 2354.6



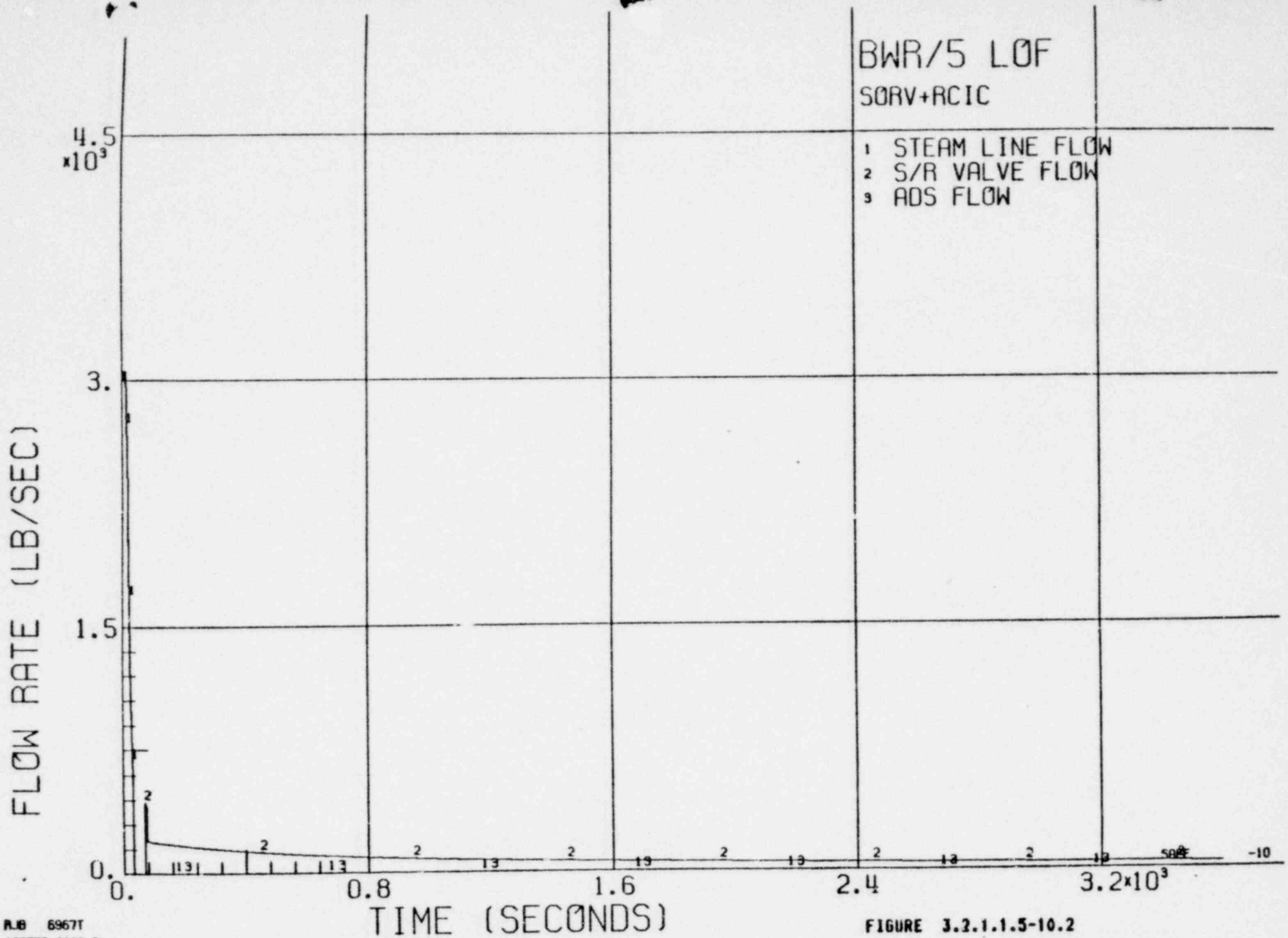
KMH 7846T
010480 2354.6

FIGURE 3.2.1.1.5-9.8



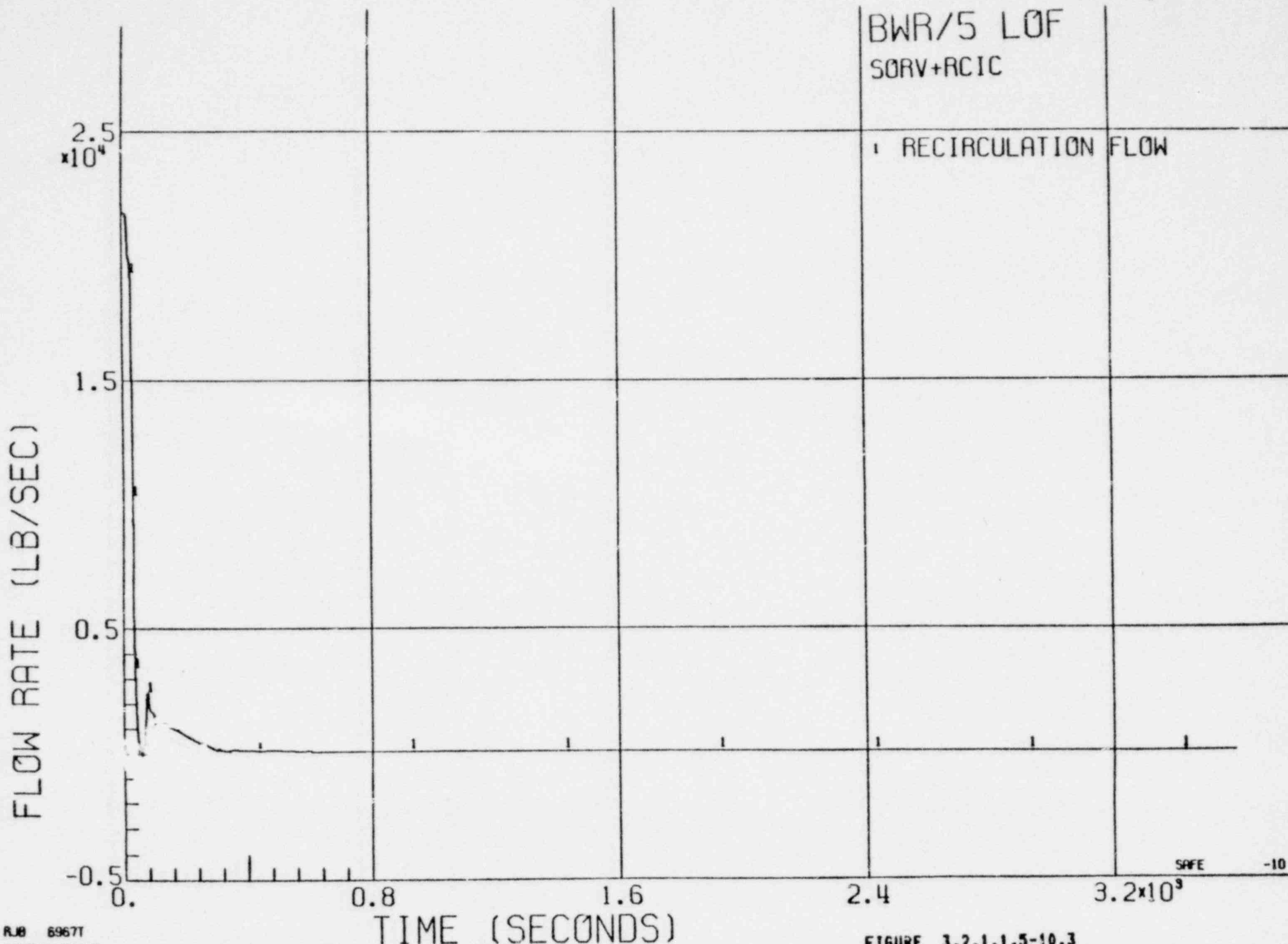
RJB 6967T
120779 2036.2

FIGURE 3.2.1.1.5-10.1



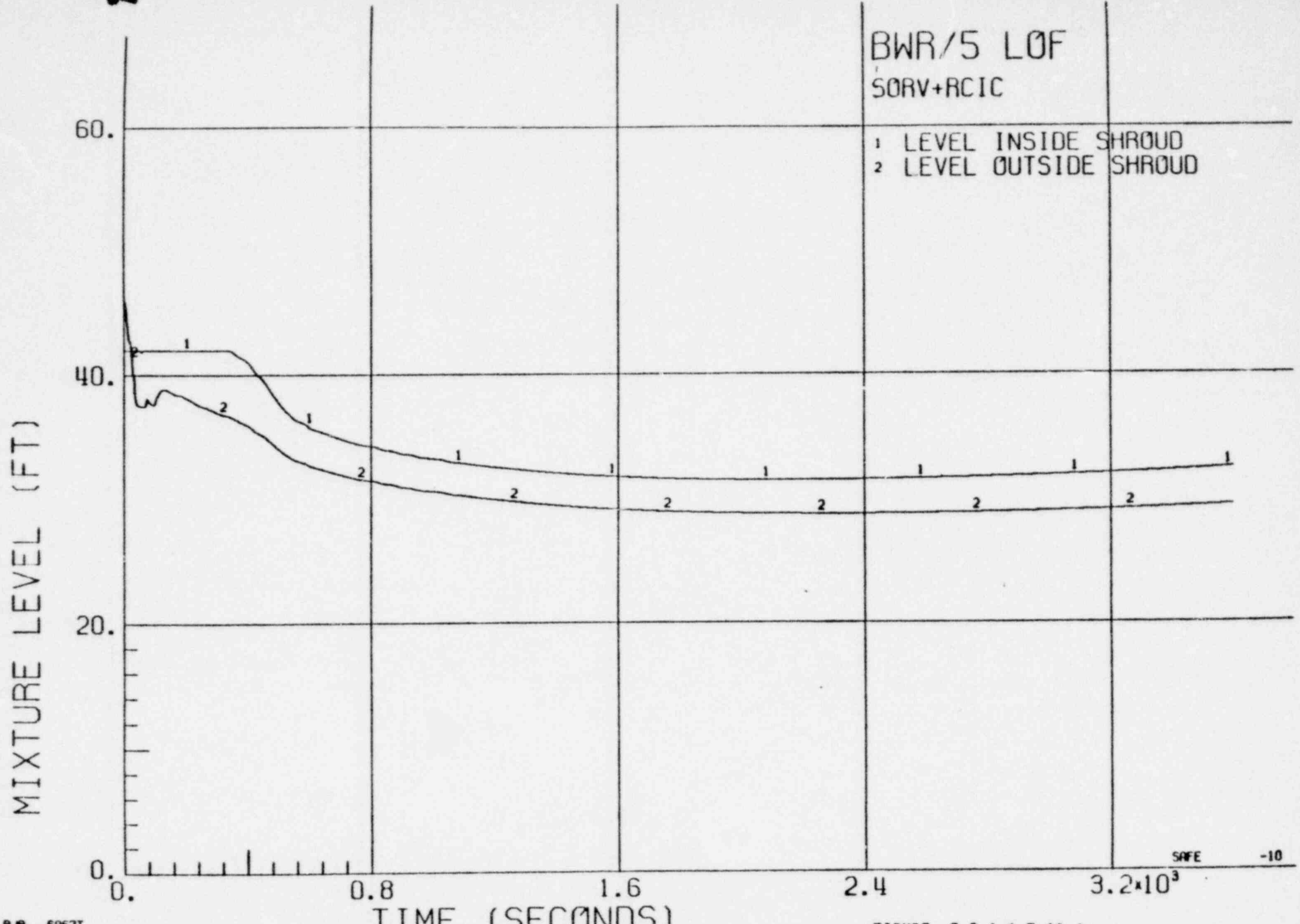
RJB 6967T
120779 2038.2

FIGURE 3.2.1.1.5-10.2



RJB 6967T
120779 2203.5

FIGURE 3.2.1.1.5-10.3



RJB 6967T
120779 2038.2

FIGURE 3.2.1.1.5-10.4

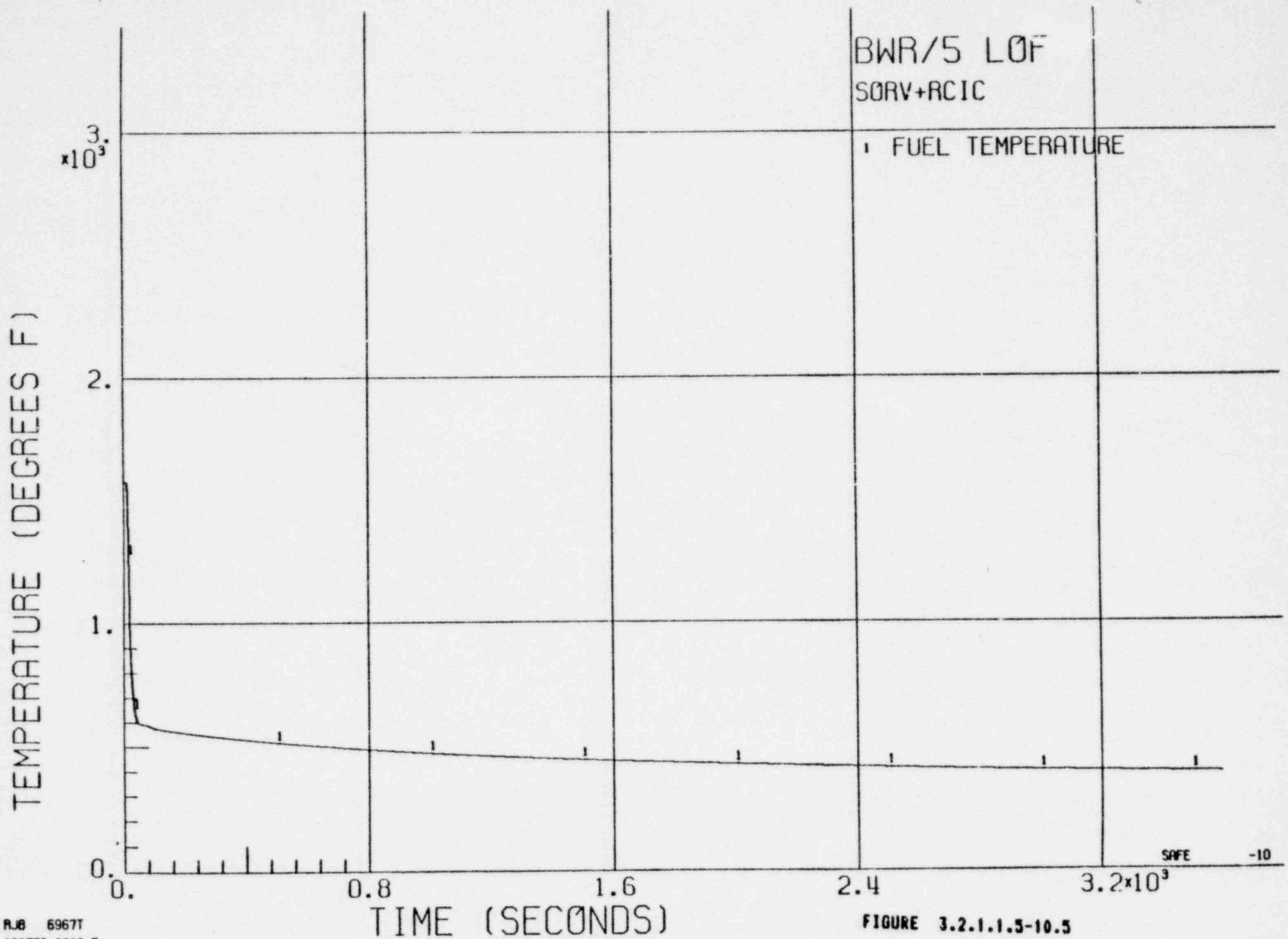
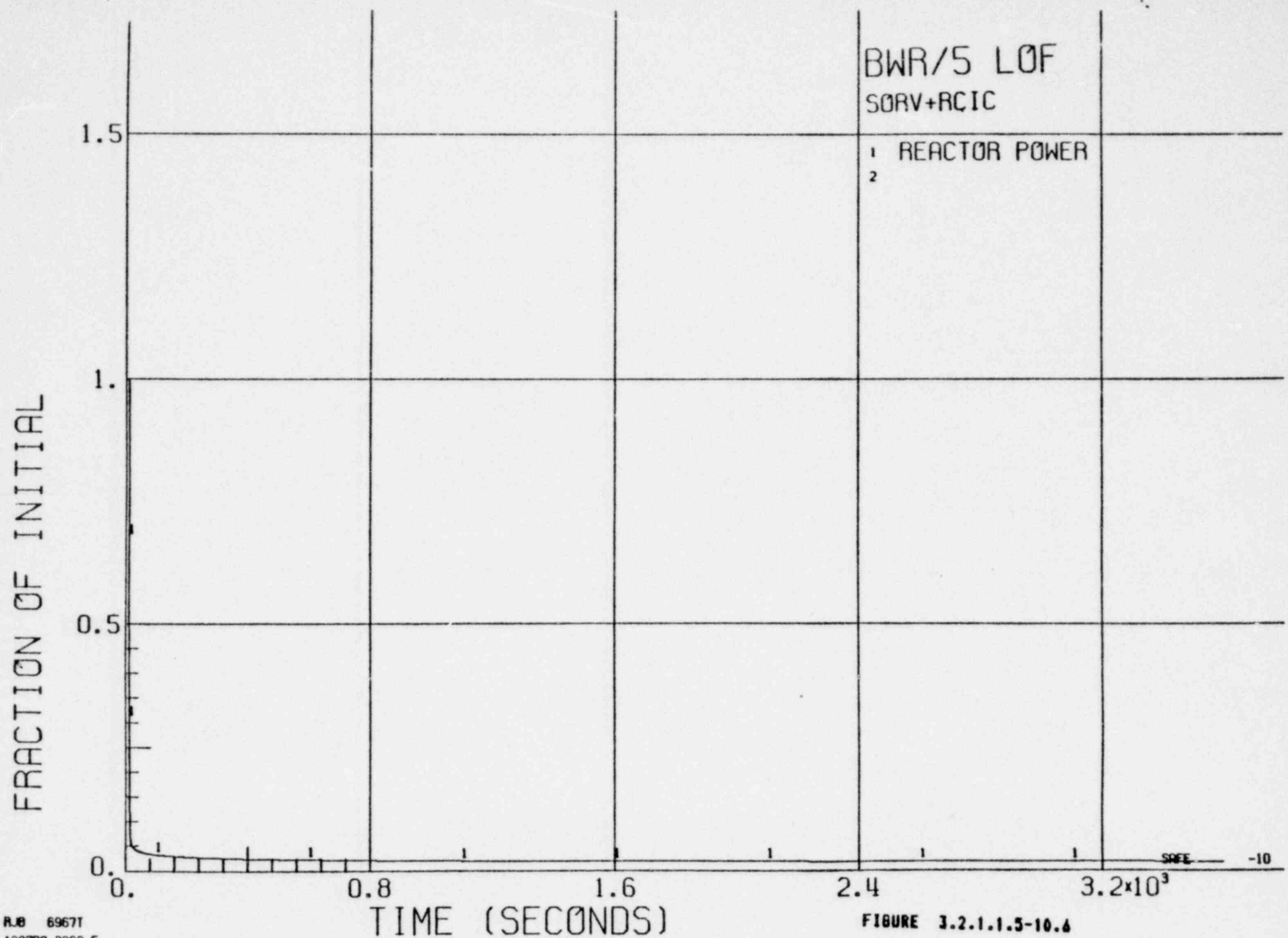


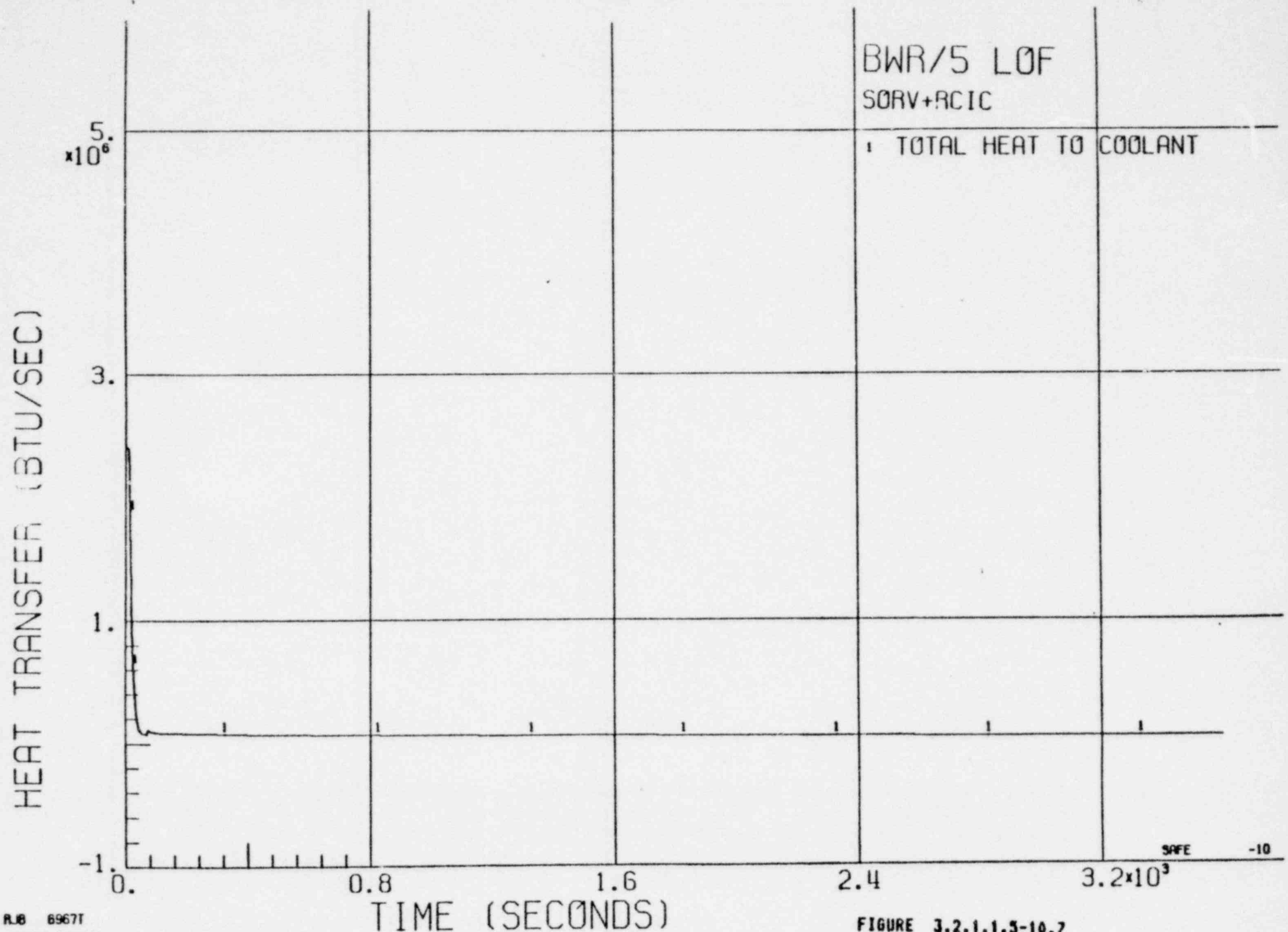
FIGURE 3.2.1.1.5-10.5

RJB 6967T
120778 2203.5



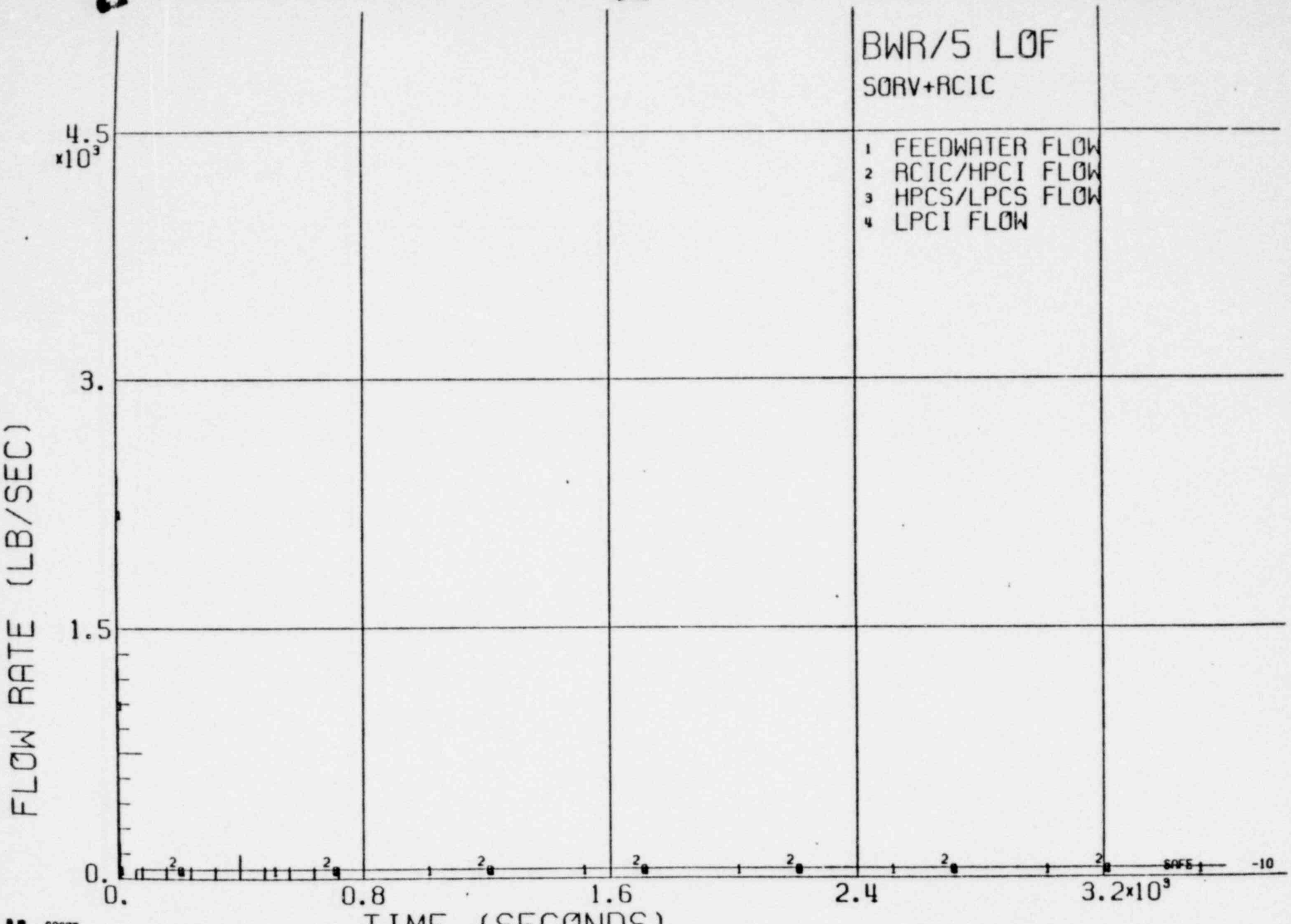
RJB 6967T
120779 2203.5

FIGURE 3.2.1.1.5-10.6



RJB 6967T
120779 2203.5

FIGURE 3.2.1.1.5-10.7



RJB 6967T
120779 2038.2

FIGURE 3.2.1.1.5-10.8

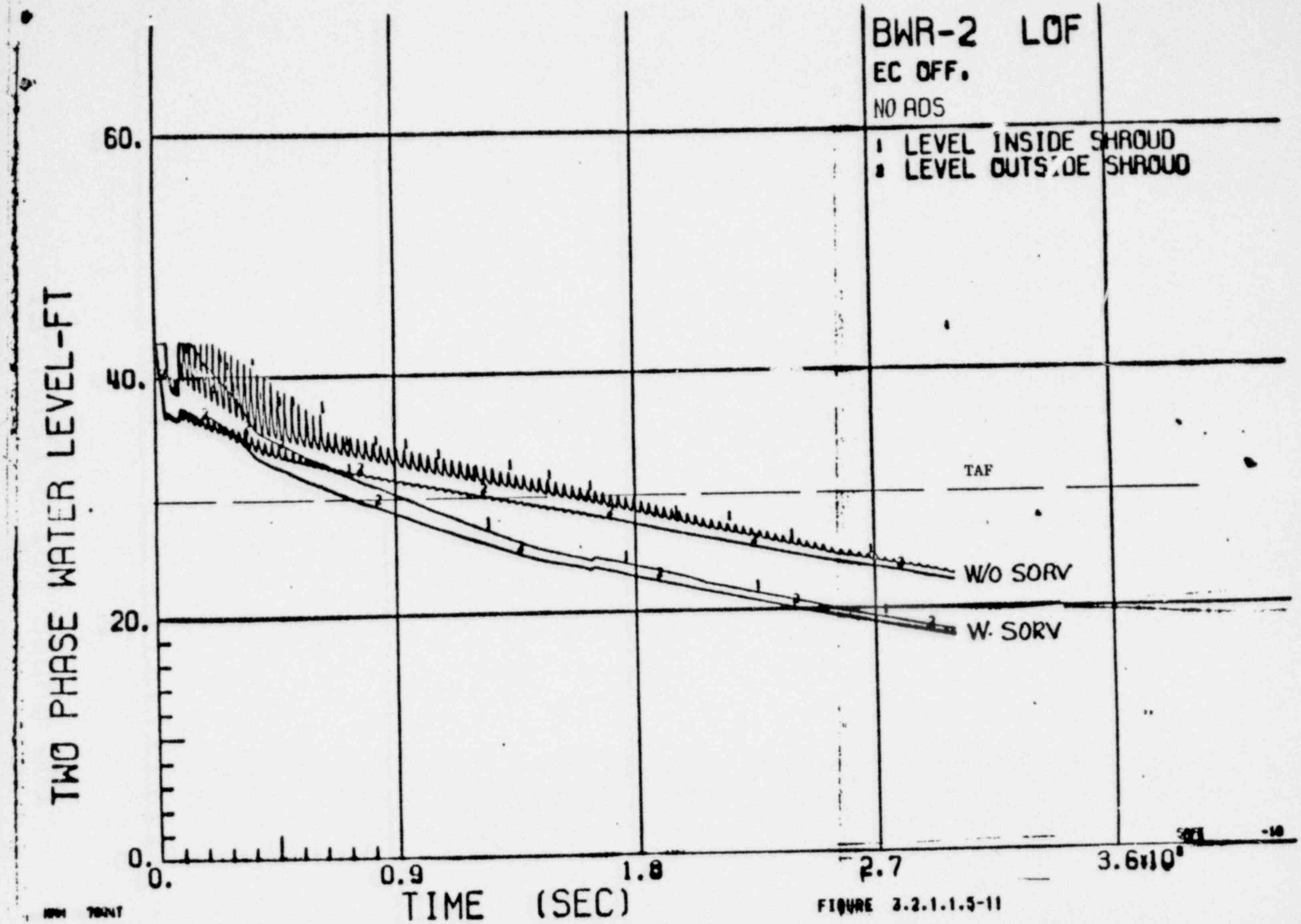


FIGURE 3.2.1.1.5-11

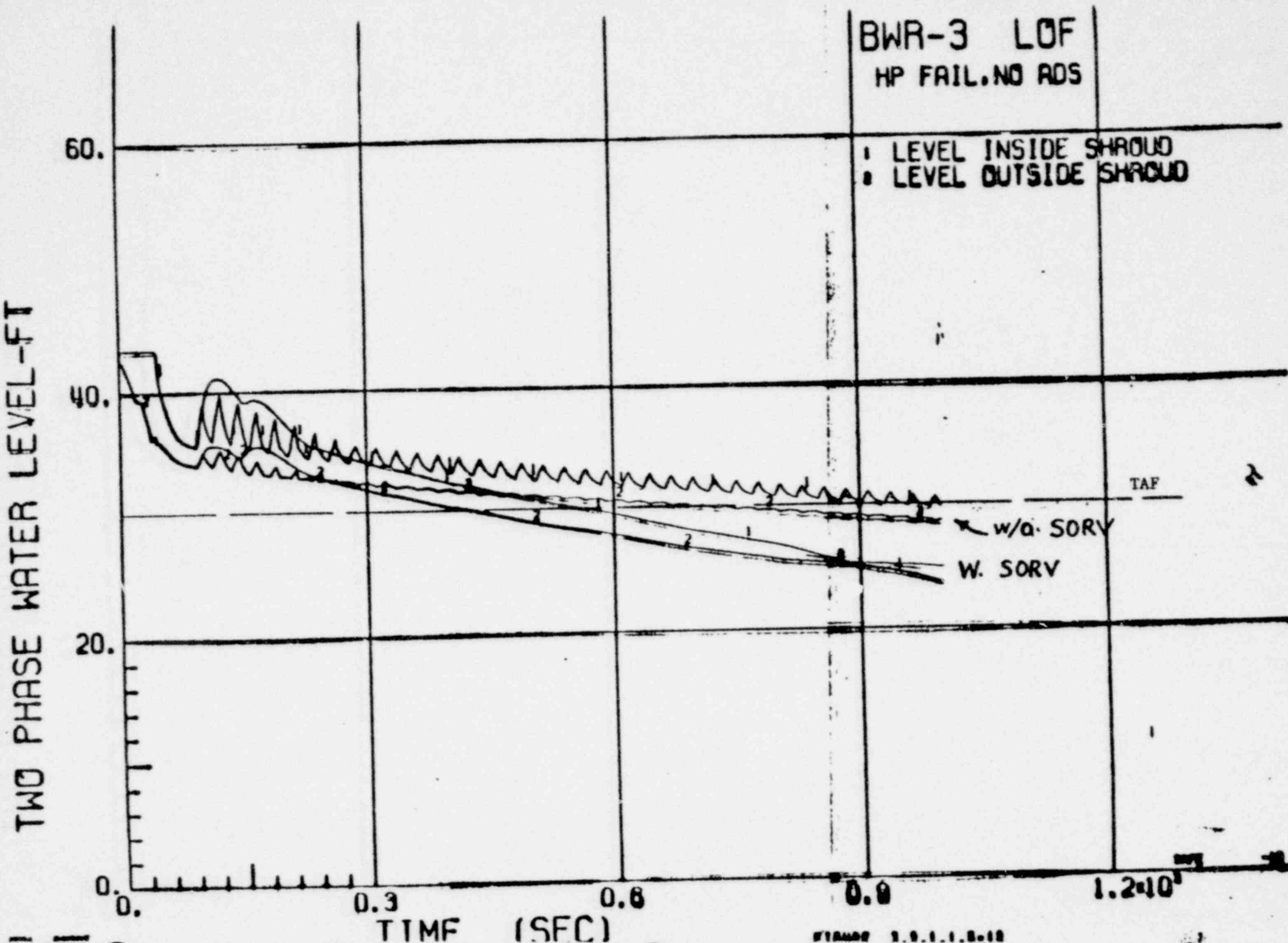

BWR-3 LOF
HP FAIL, NO ADS
1 LEVEL INSIDE SHROUD
2 LEVEL OUTSIDE SHROUD
TAF
w/o SORV
W. SORV
60.
40.
20.
0.
0.
0.3
0.6
0.9
1.2×10³
TWO PHASE WATER LEVEL-FT
TIME (SEC)

FIGURE [illegible]

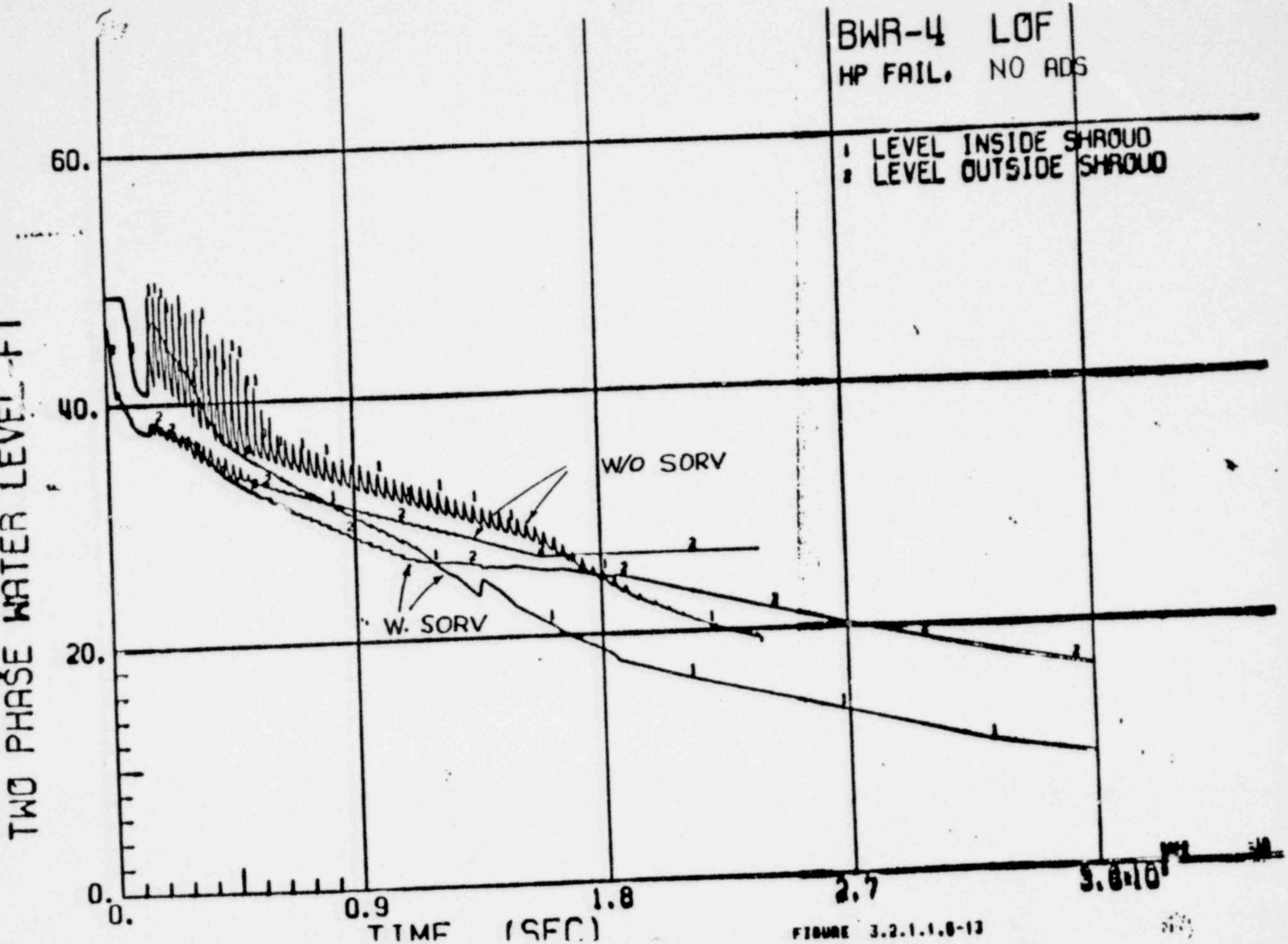


FIGURE 3.2.1.1.8-13

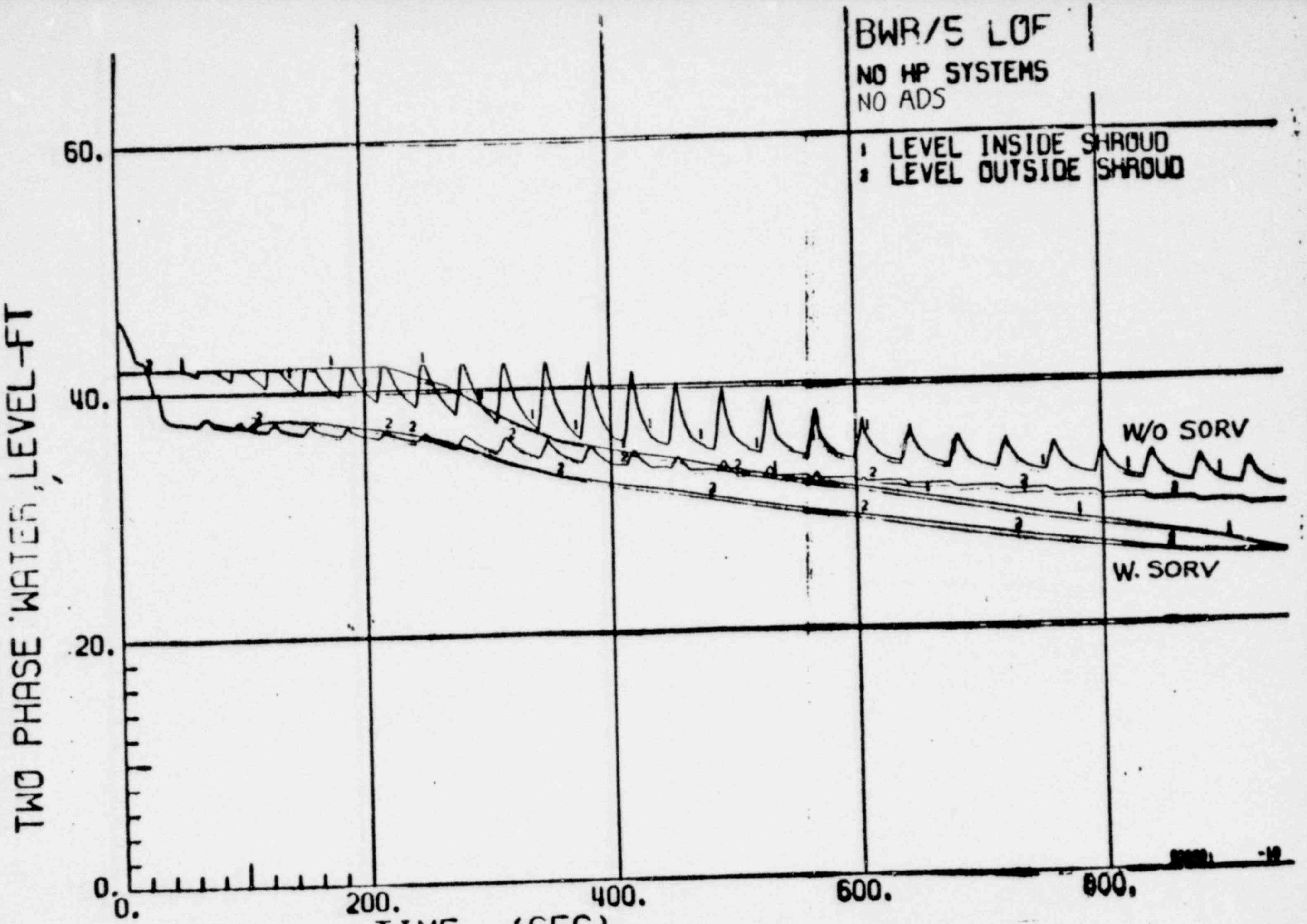
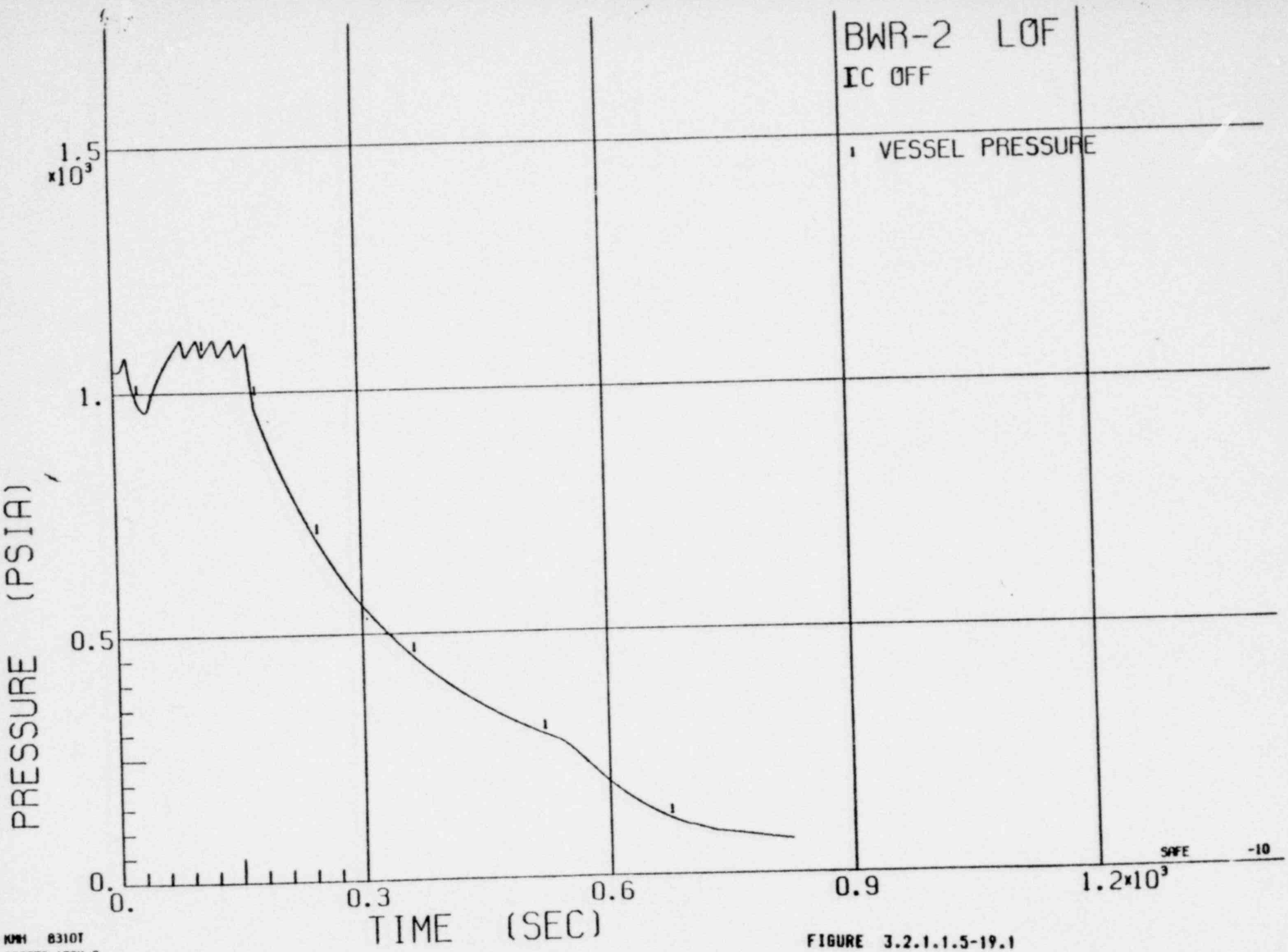


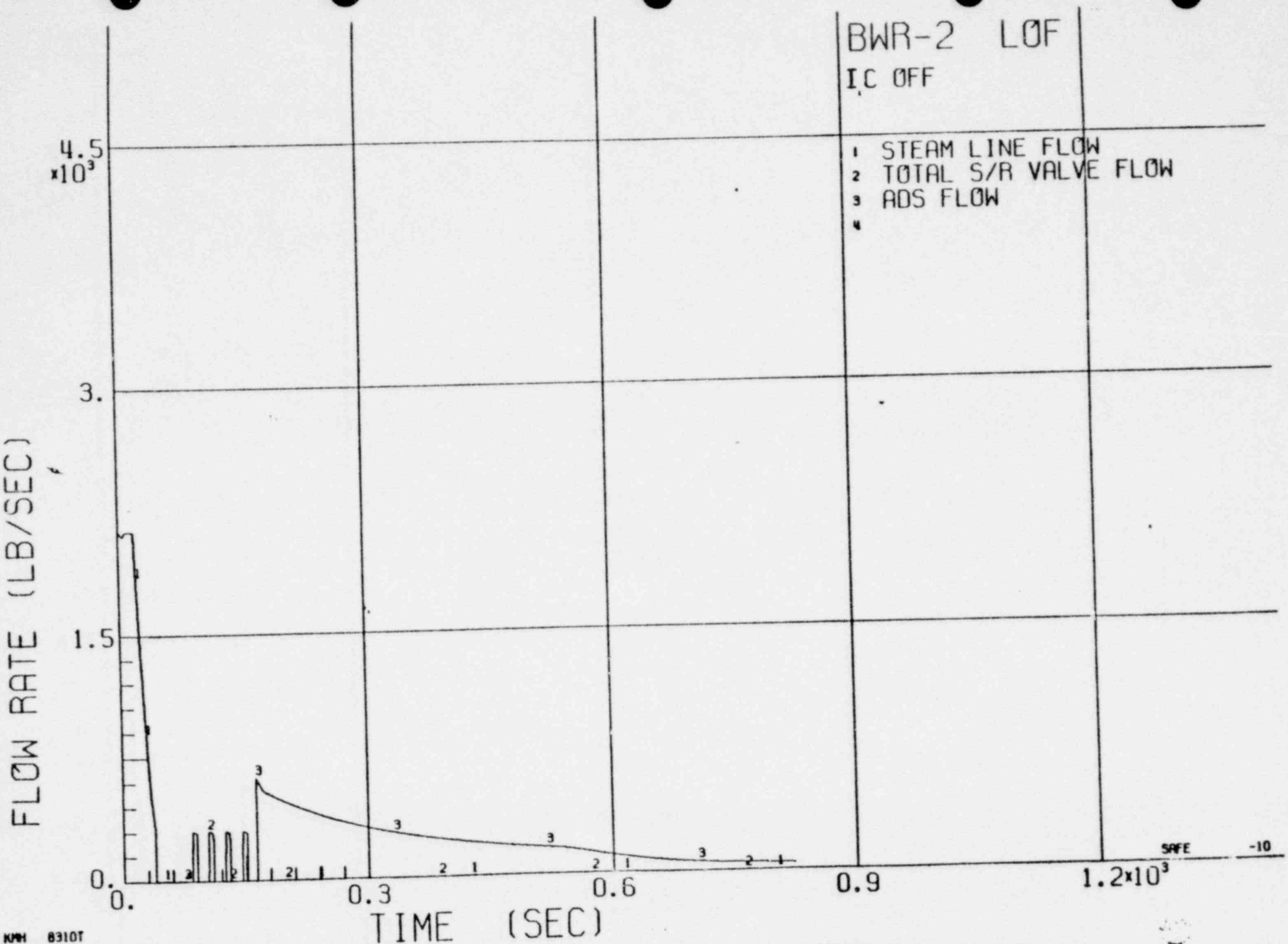
FIGURE 3.2.1.1.8-14

(There are no Figures 3.2.1.1.5-15 through 18.)



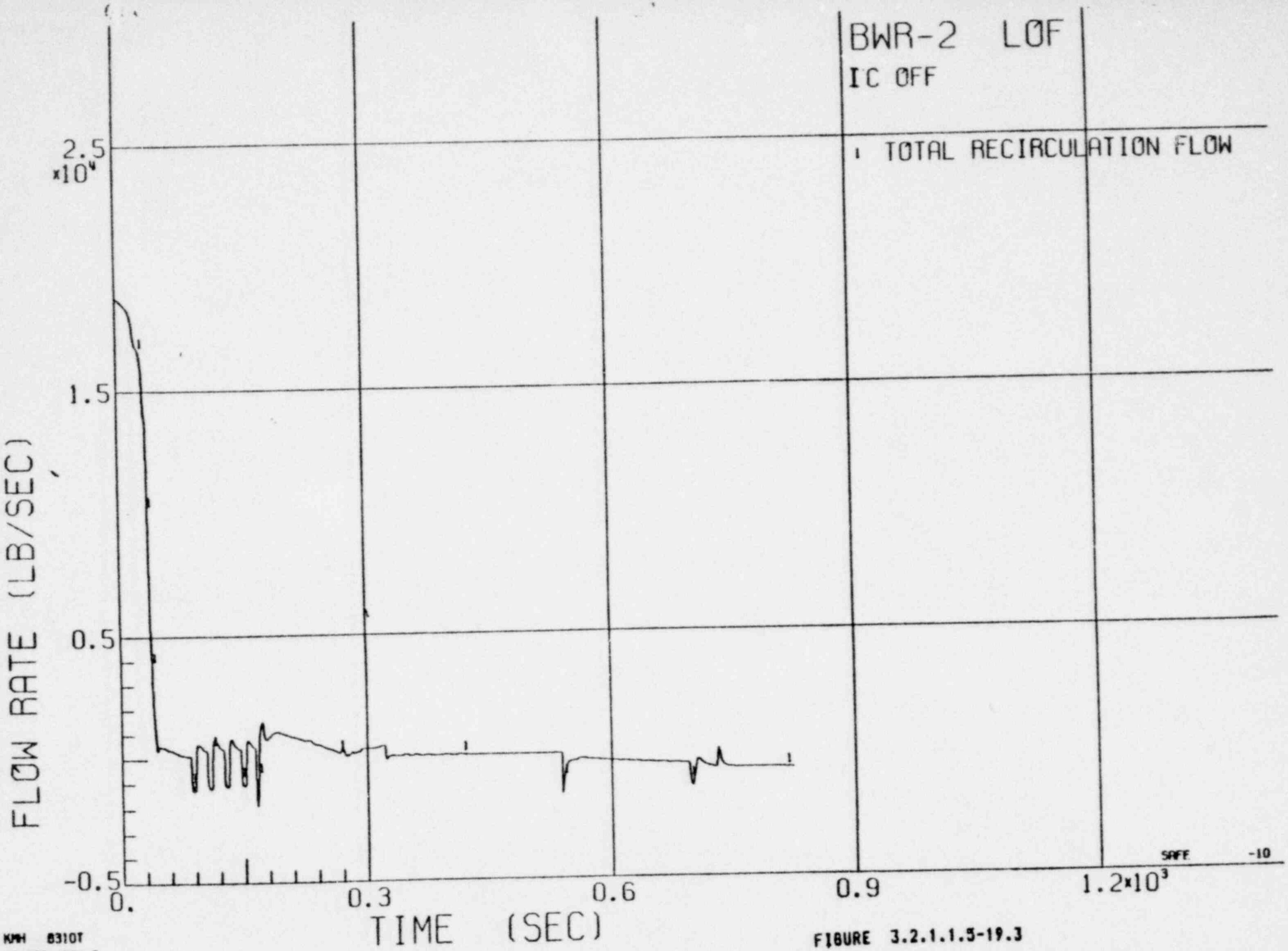
KMH 8310T
122879 1734.9

FIGURE 3.2.1.1.5-19.1



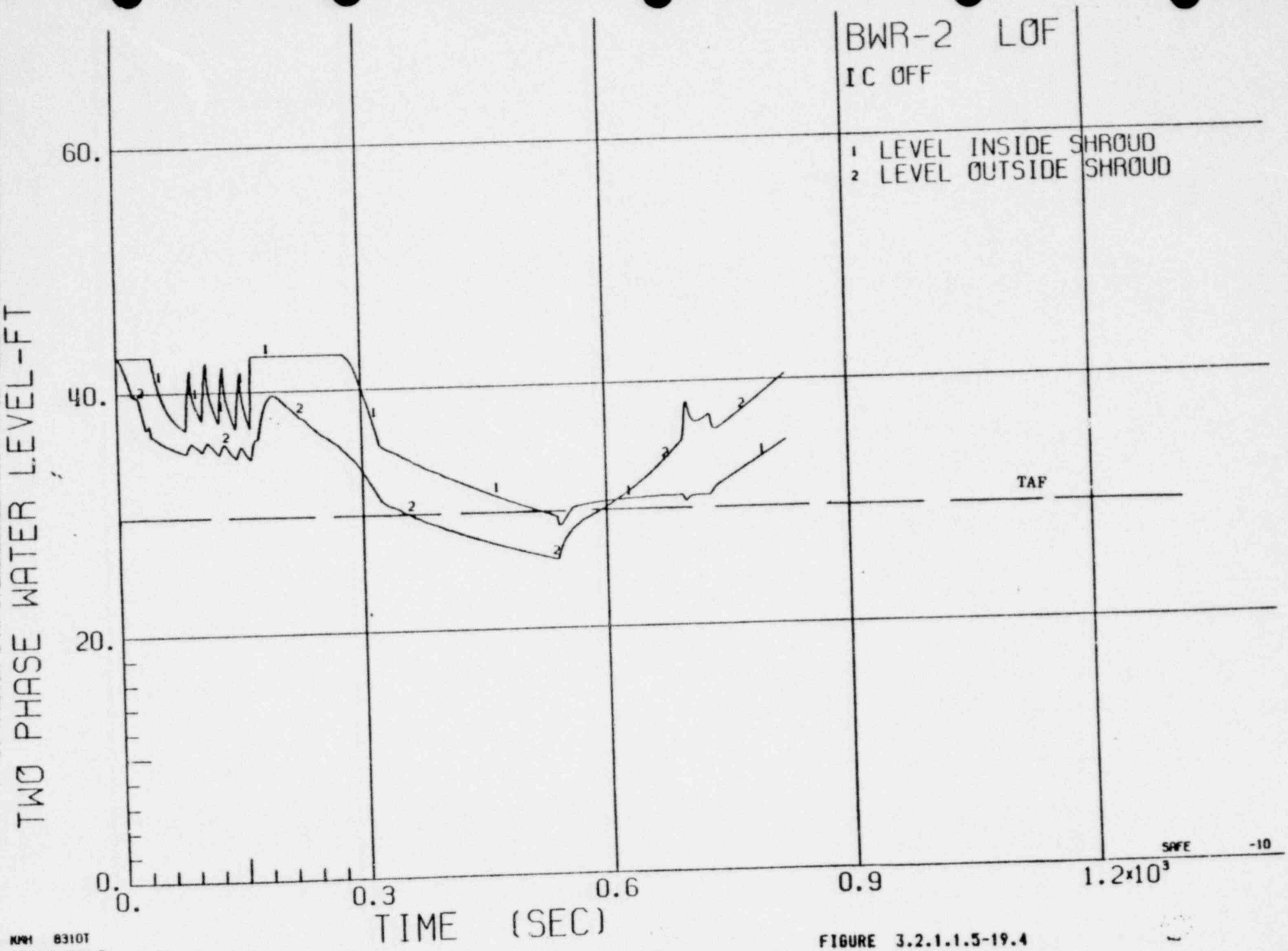
KMH 8310T
122879 1"34.9

FIGURE 3.2.1.1.5-19.2



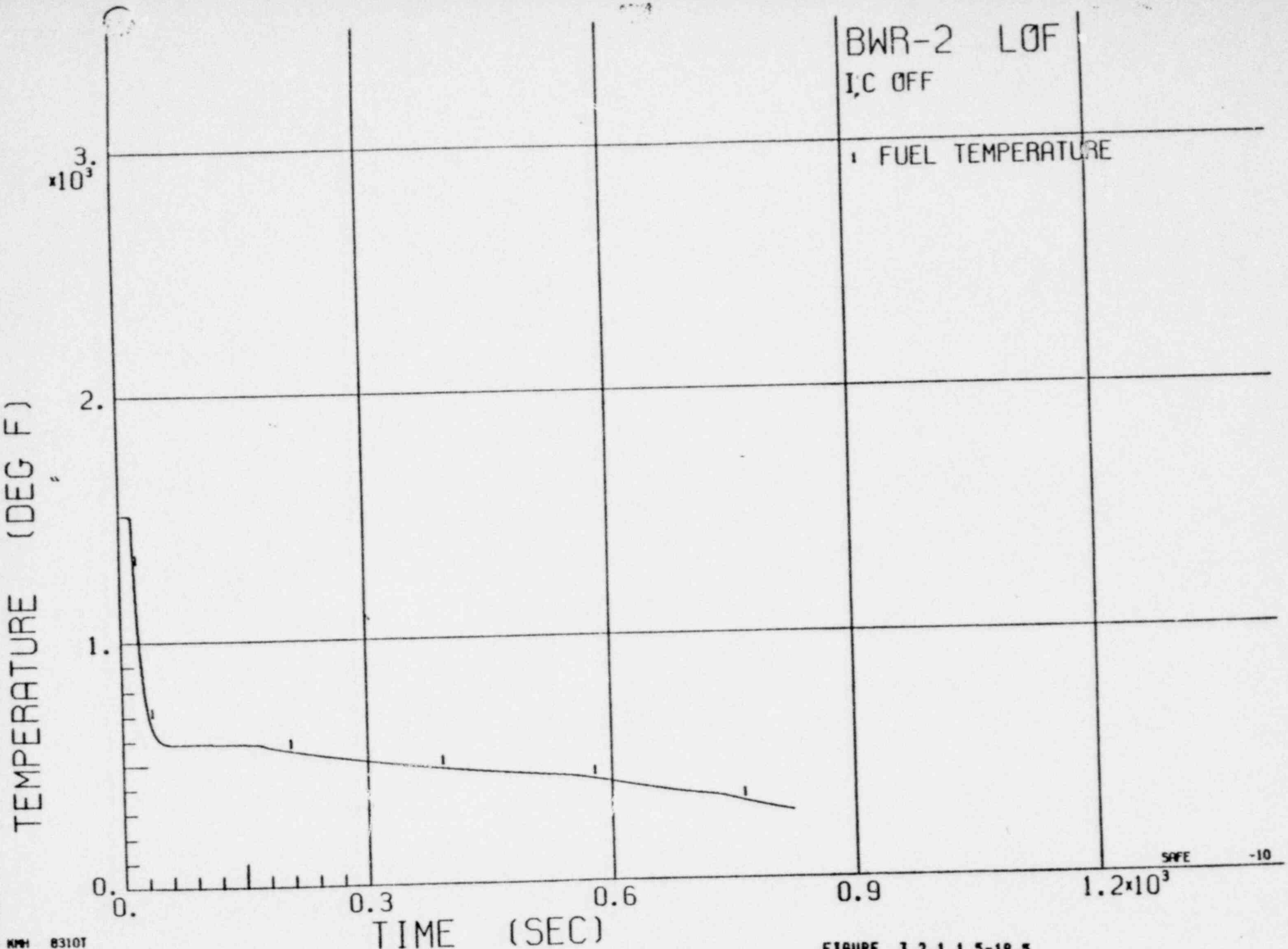
KMH 8310T
122879 1734.9

FIGURE 3.2.1.1.5-19.3



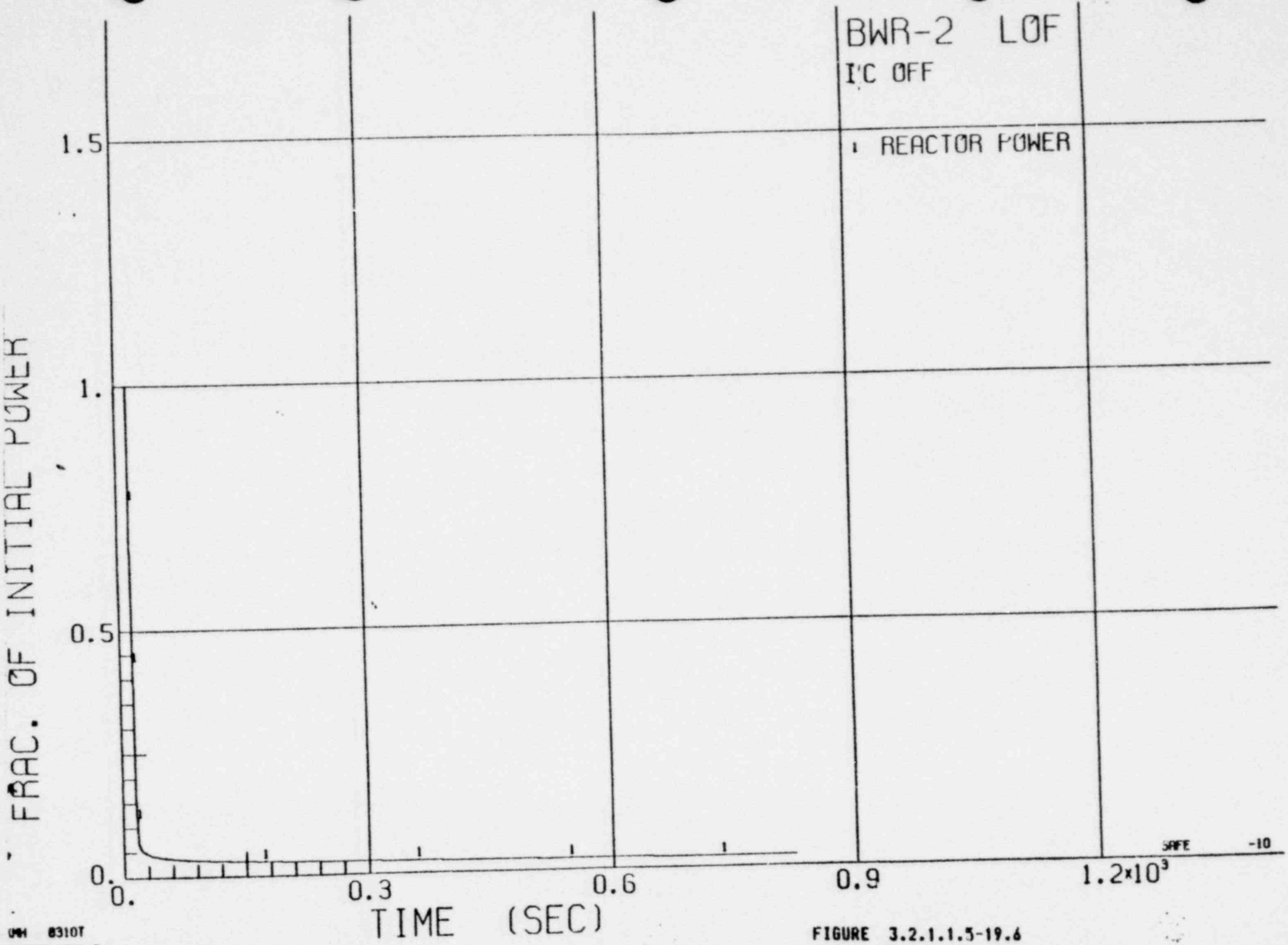
KMH 8310T
122879 1734.9

FIGURE 3.2.1.1.5-19.4



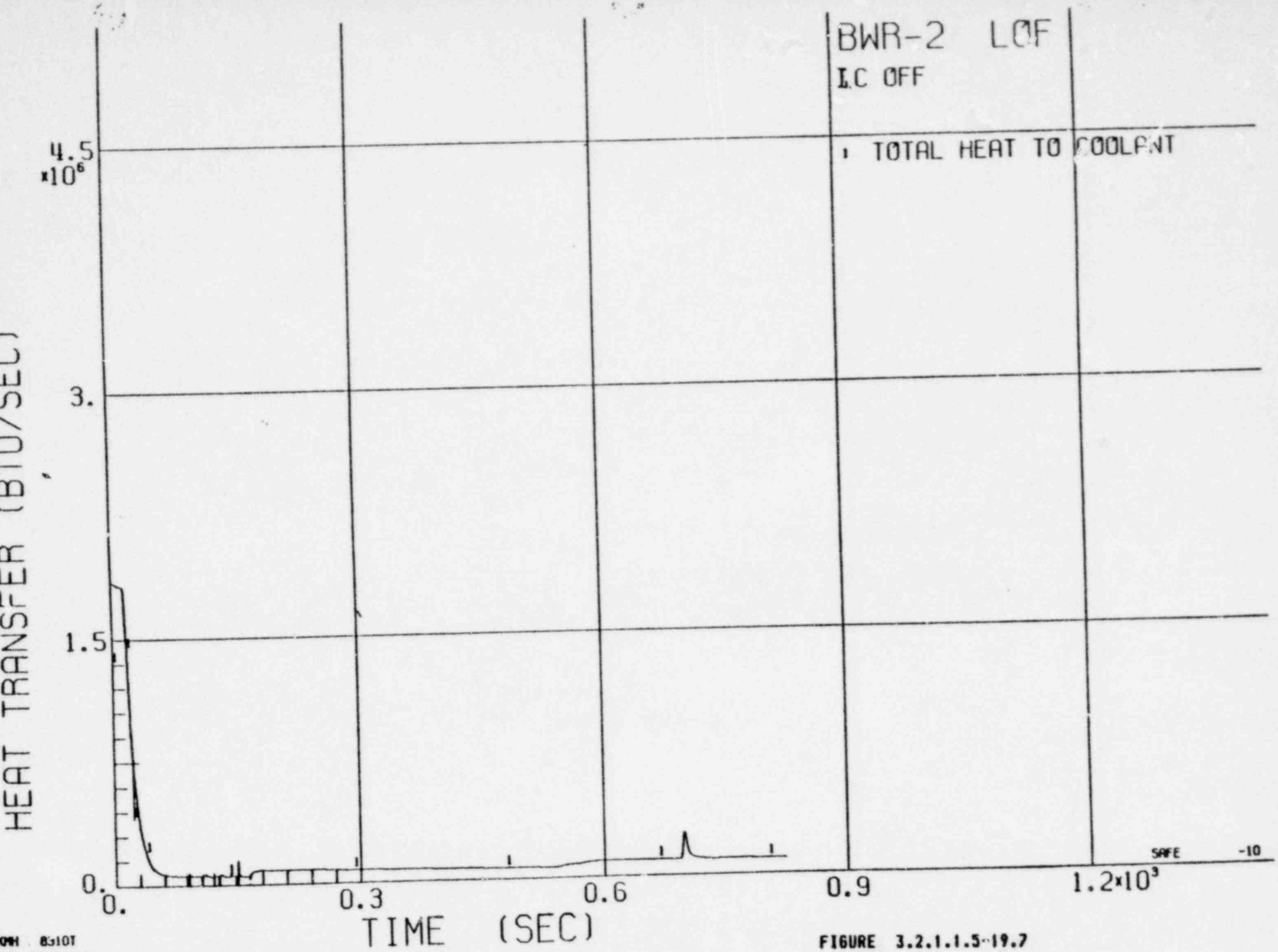
KMH 8310T
122879 1734.9

FIGURE 3.2.1.1.5-19.5



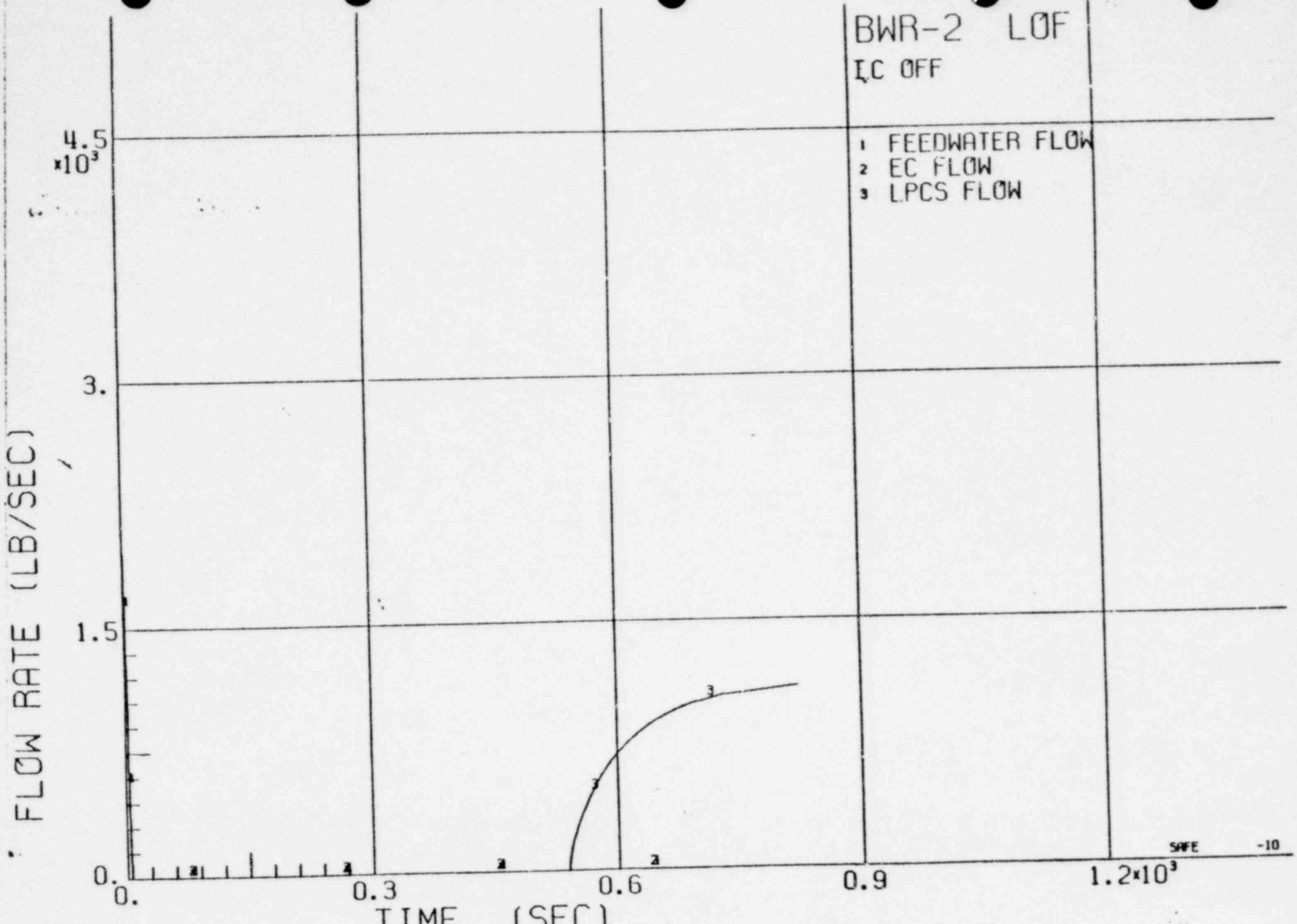
UMH 8310T
122879 1734.9

FIGURE 3.2.1.1.5-19.6



KMH 83101
122878 1748.1

FIGURE 3.2.1.1.5-19.7



KMH 8310T
122879 1748.1

FIGURE 3.2.1.1.5-19.8

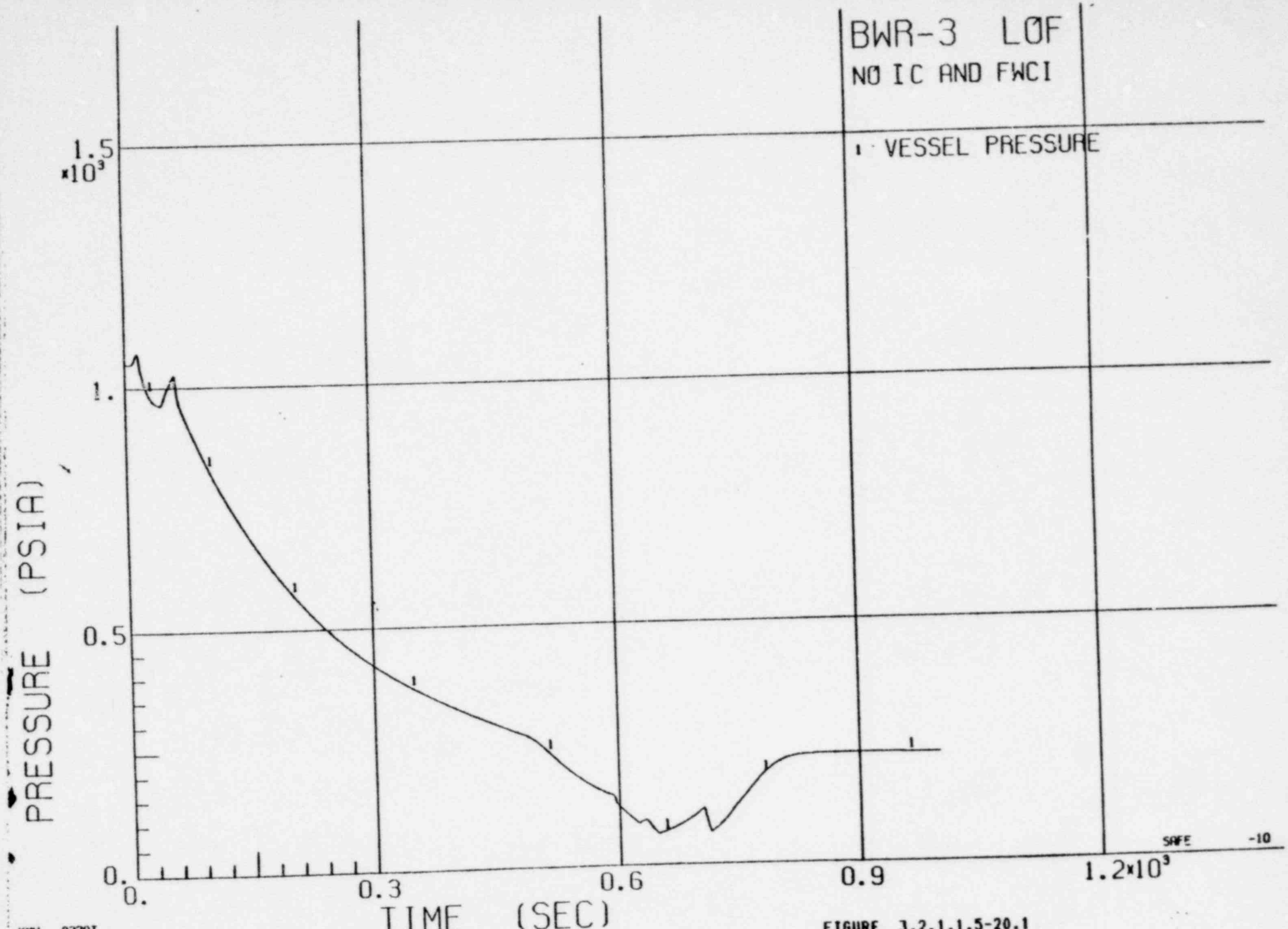
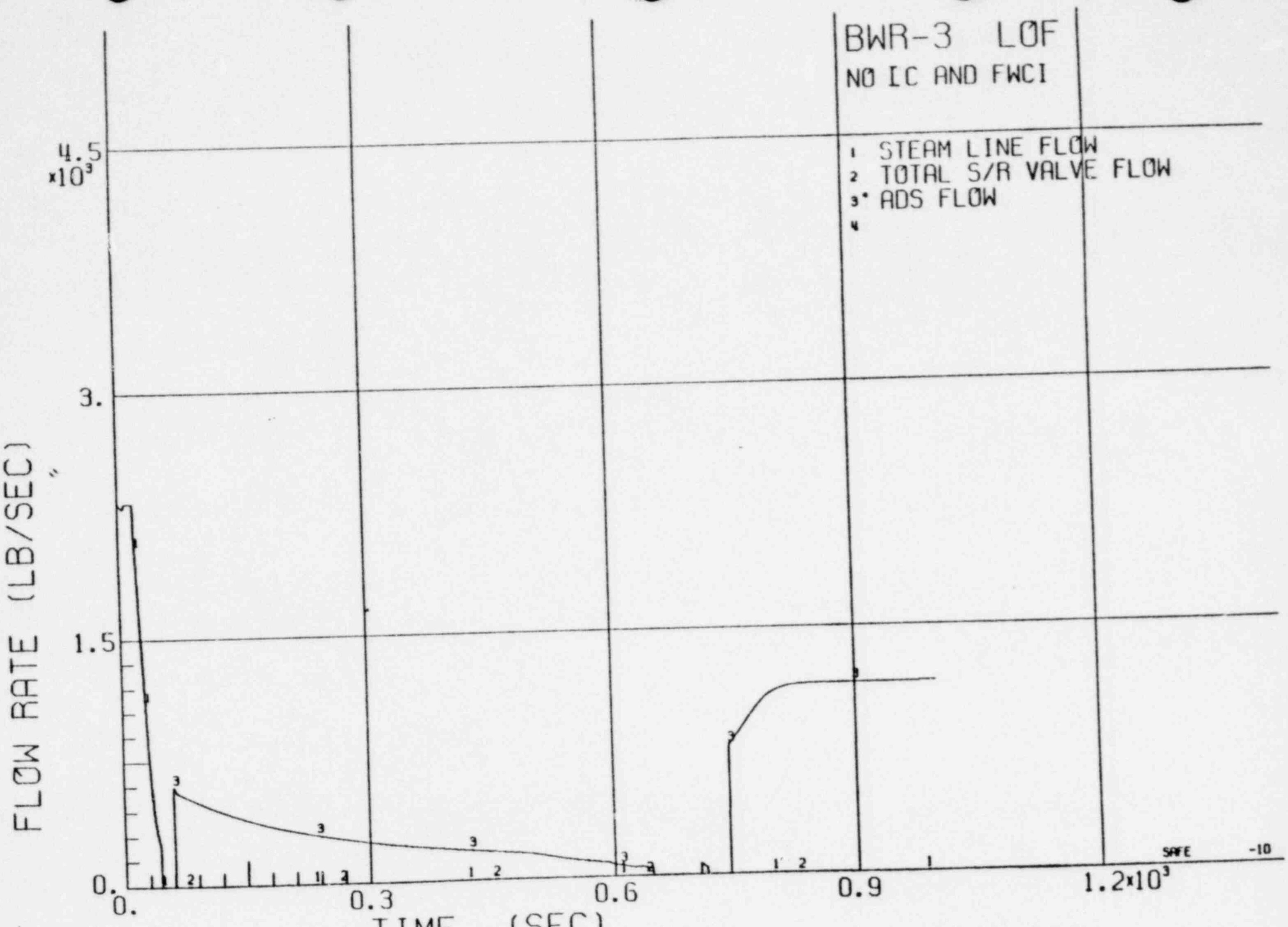


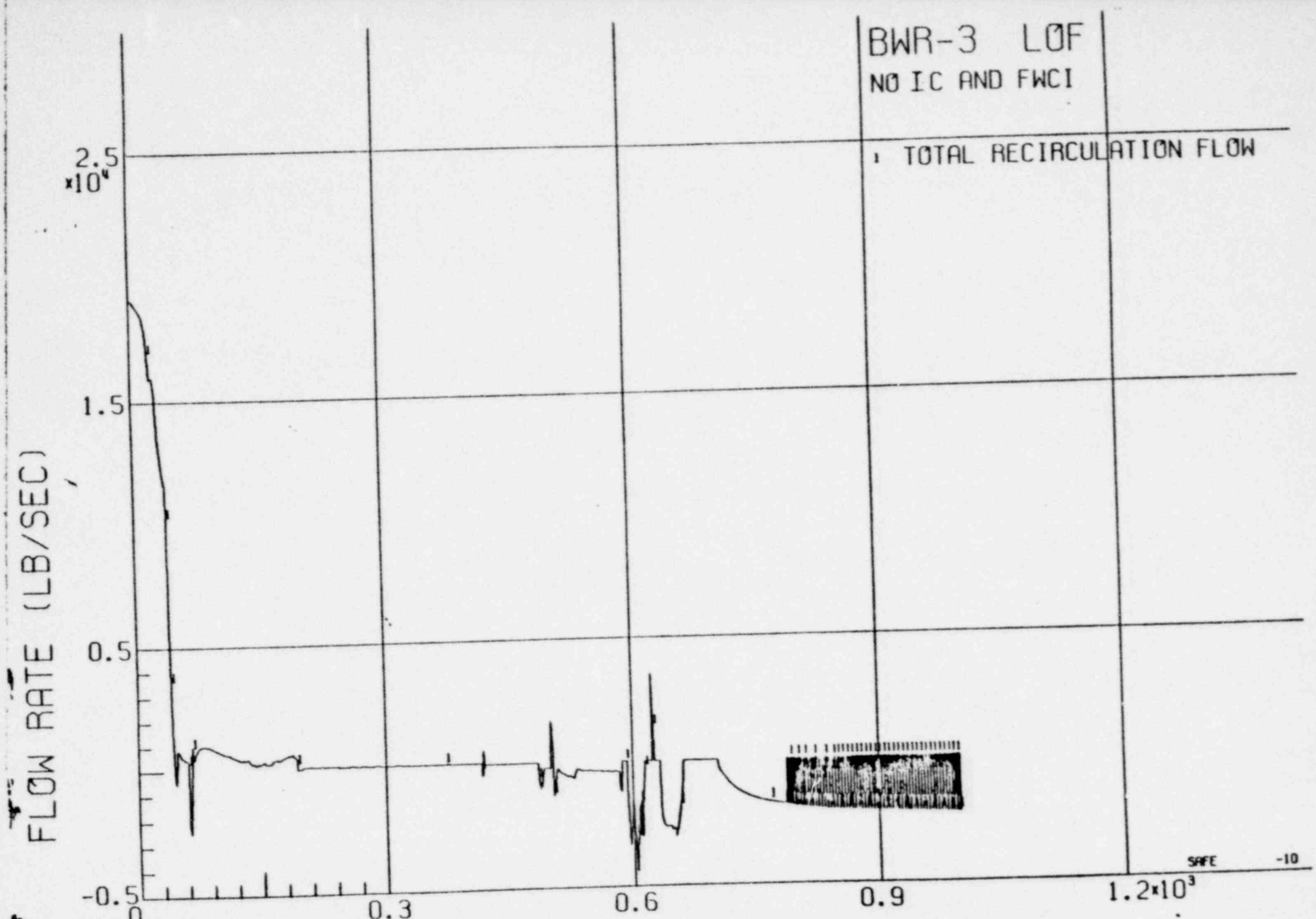
FIGURE 3.2.1.1.5-20.1

KMH 8379T
122879 1754.8



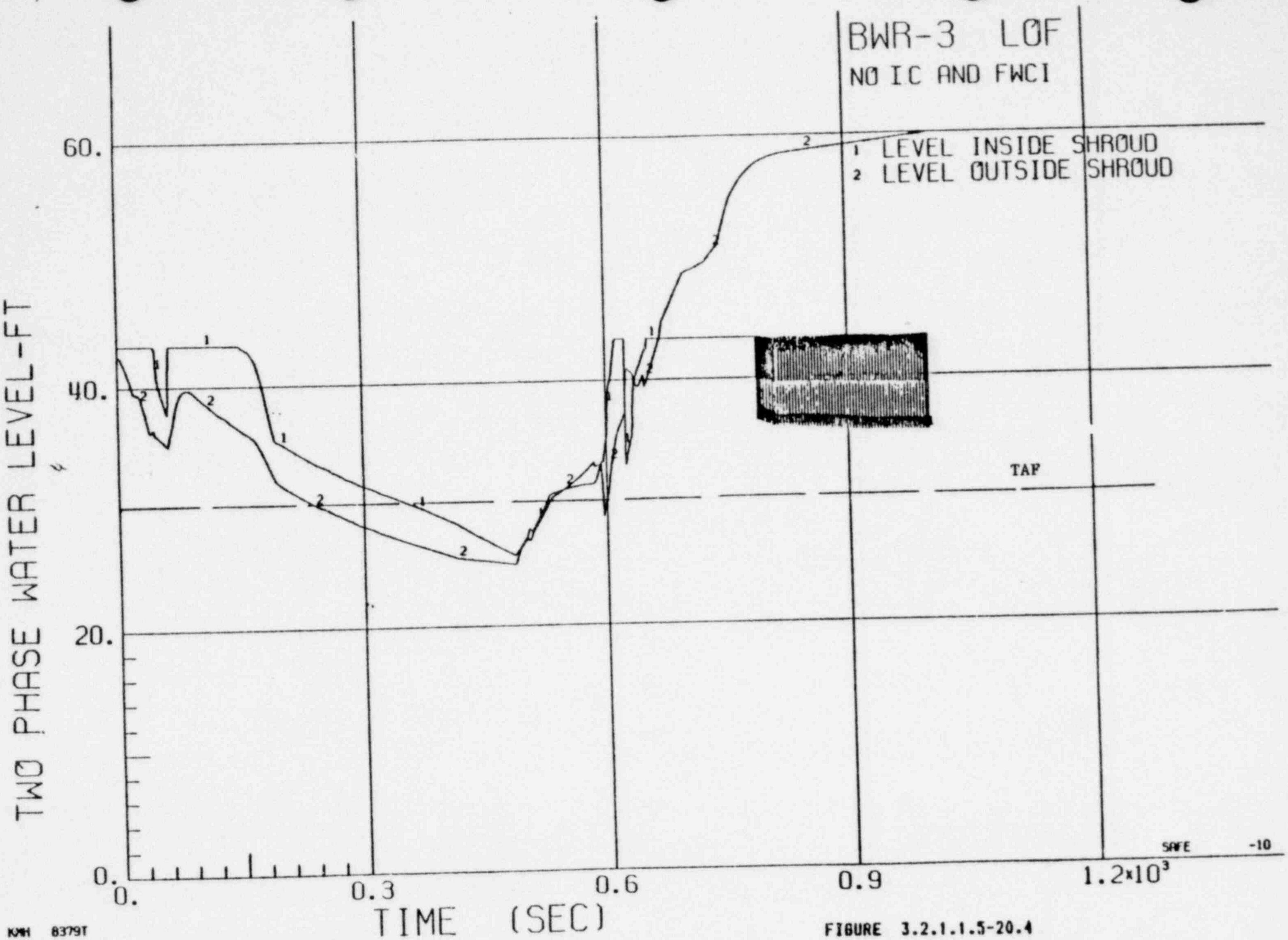
KMH 8379T
122879 1754.8

FIGURE 3.2.1.1.5-20.2



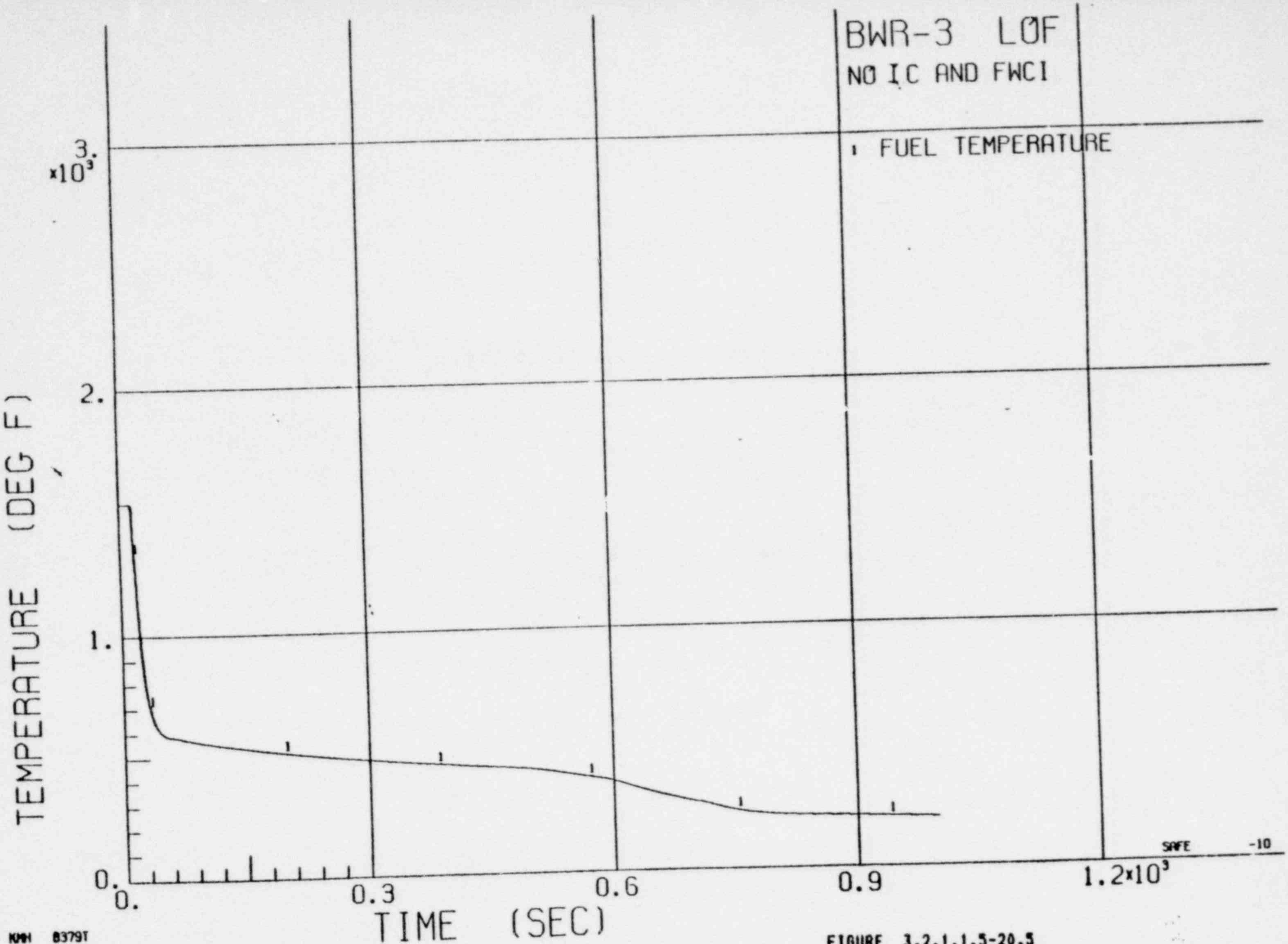
KMH 8379T
122879 1754.8

FIGURE 3.2.1.1.5-20.3



KMH 8379T
122879 1754.8

FIGURE 3.2.1.1.5-20.4



KMH 8379T
122879 1754.8

FIGURE 3.2.1.1.5-20.5

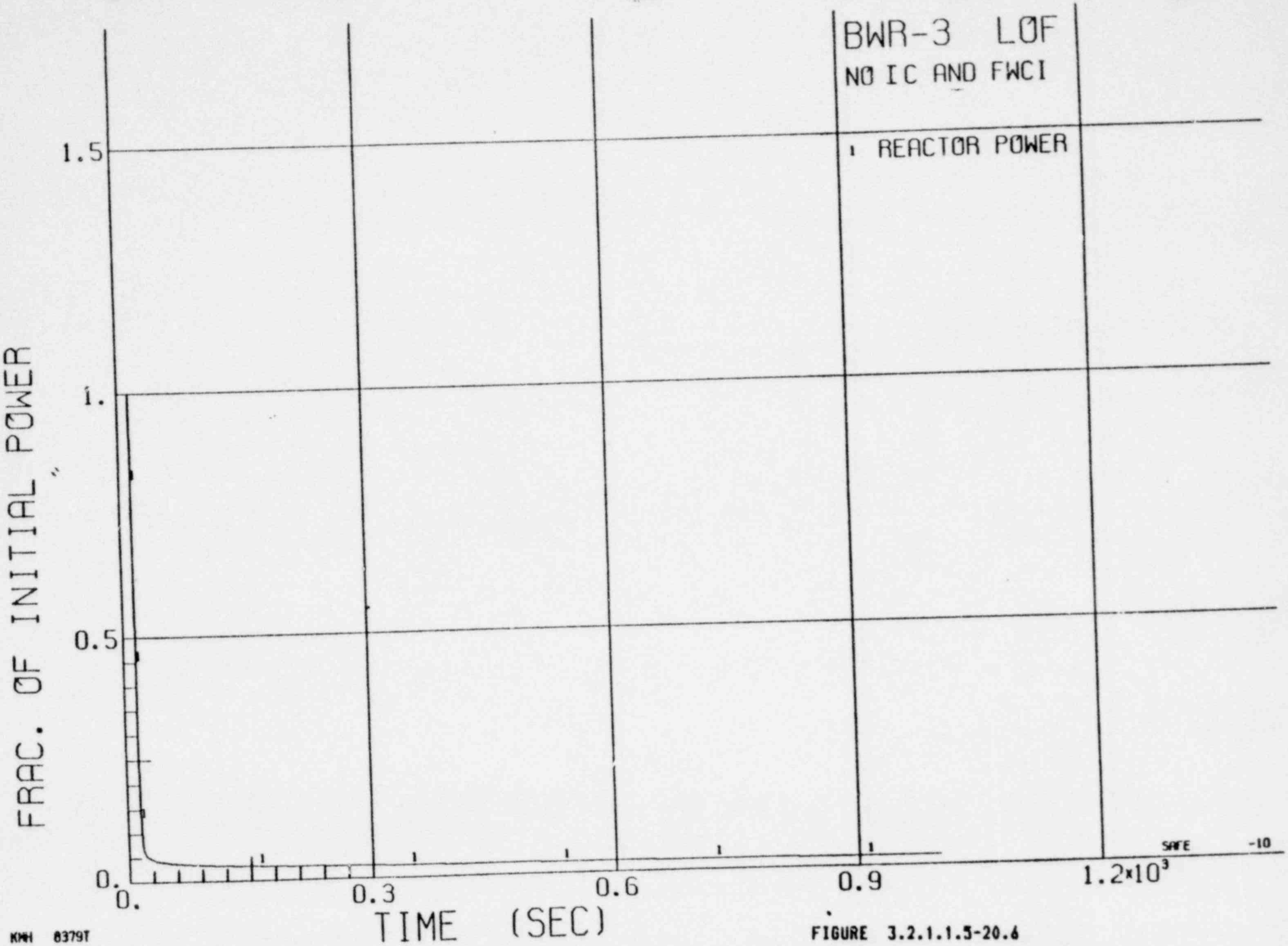
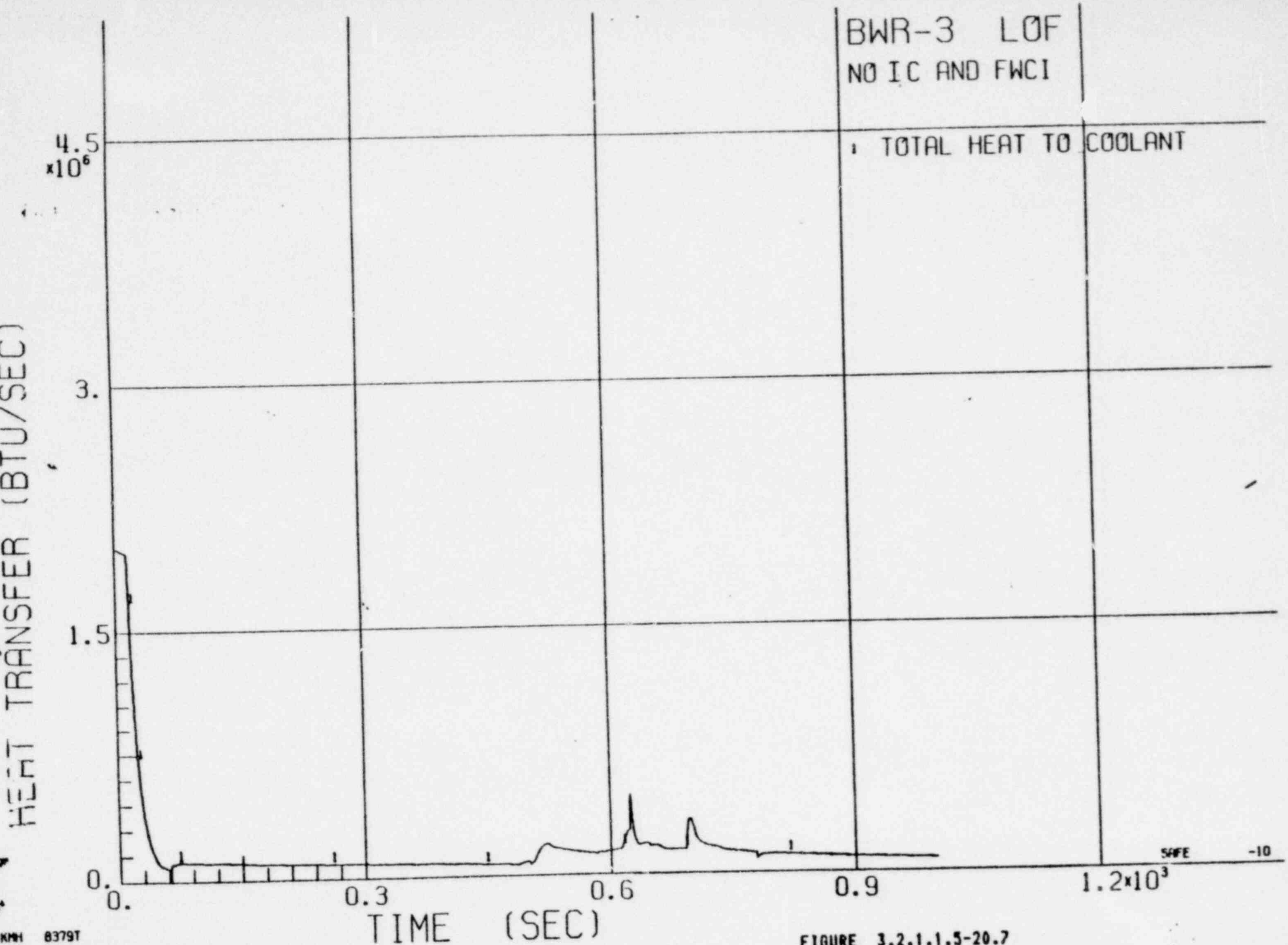


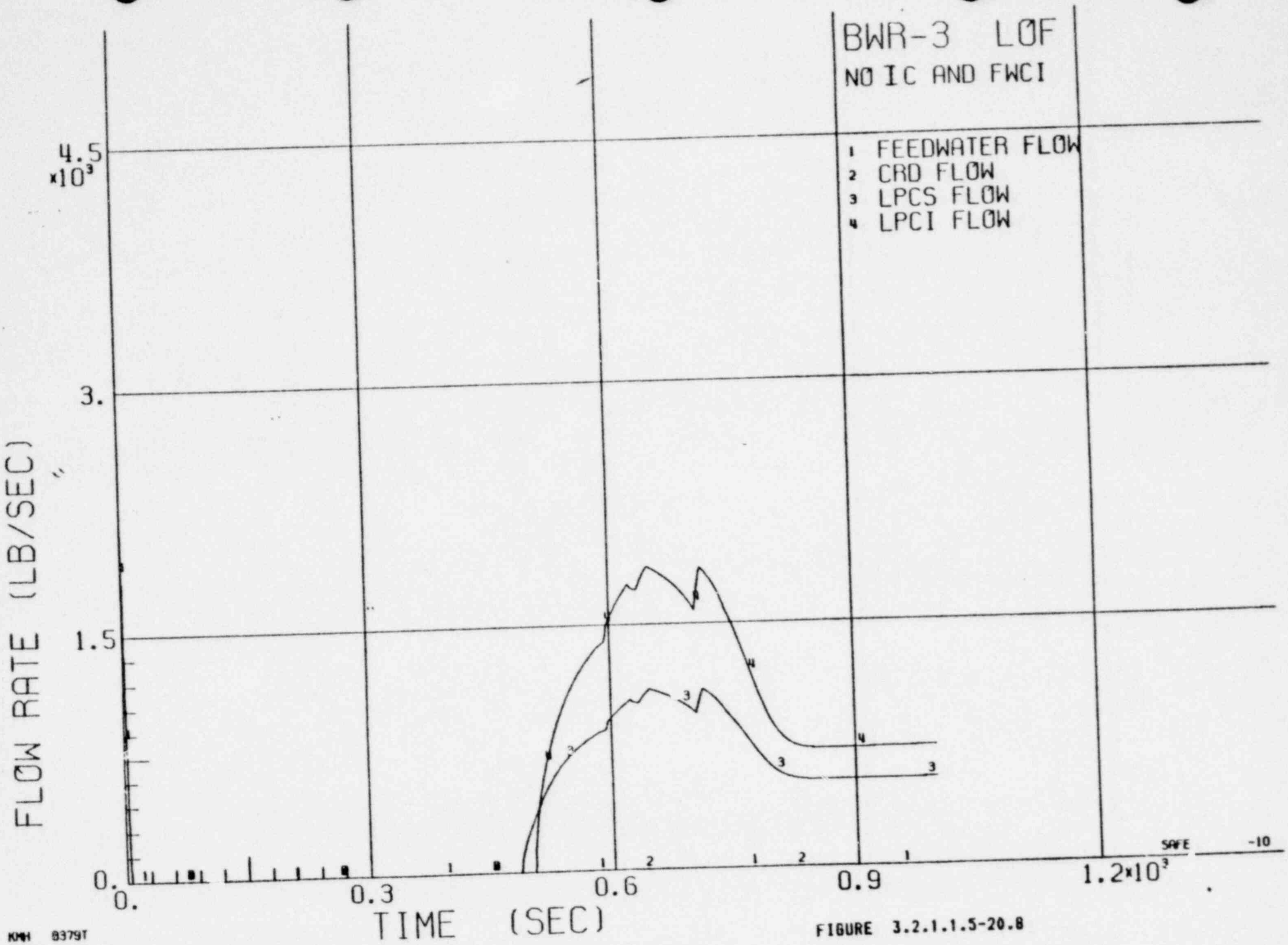
FIGURE 3.2.1.1.5-20.6

KMH 8379T
122879 1754.6



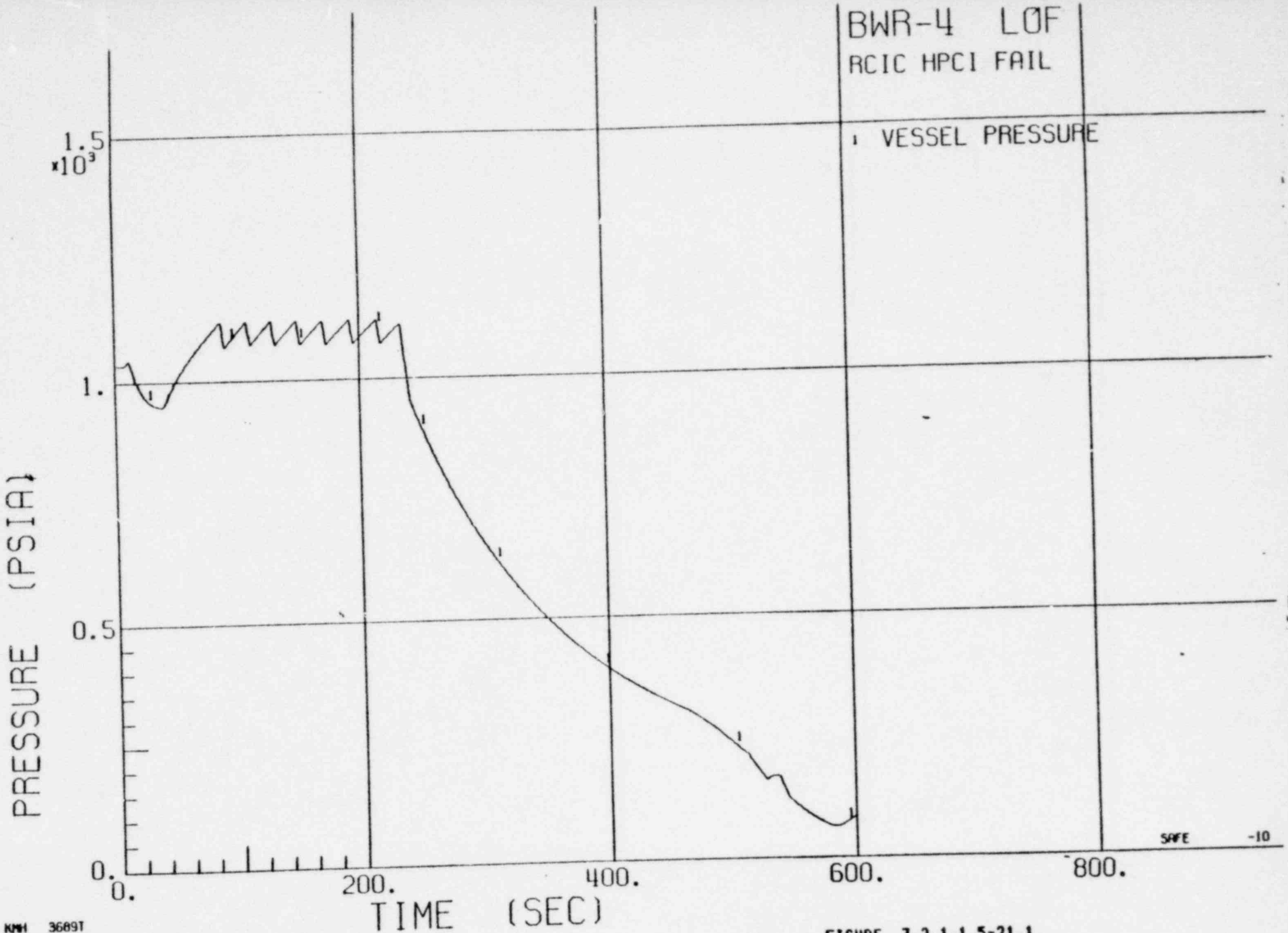
KMH 8379T
122879 1818.8

FIGURE 3.2.1.1.5-20.7



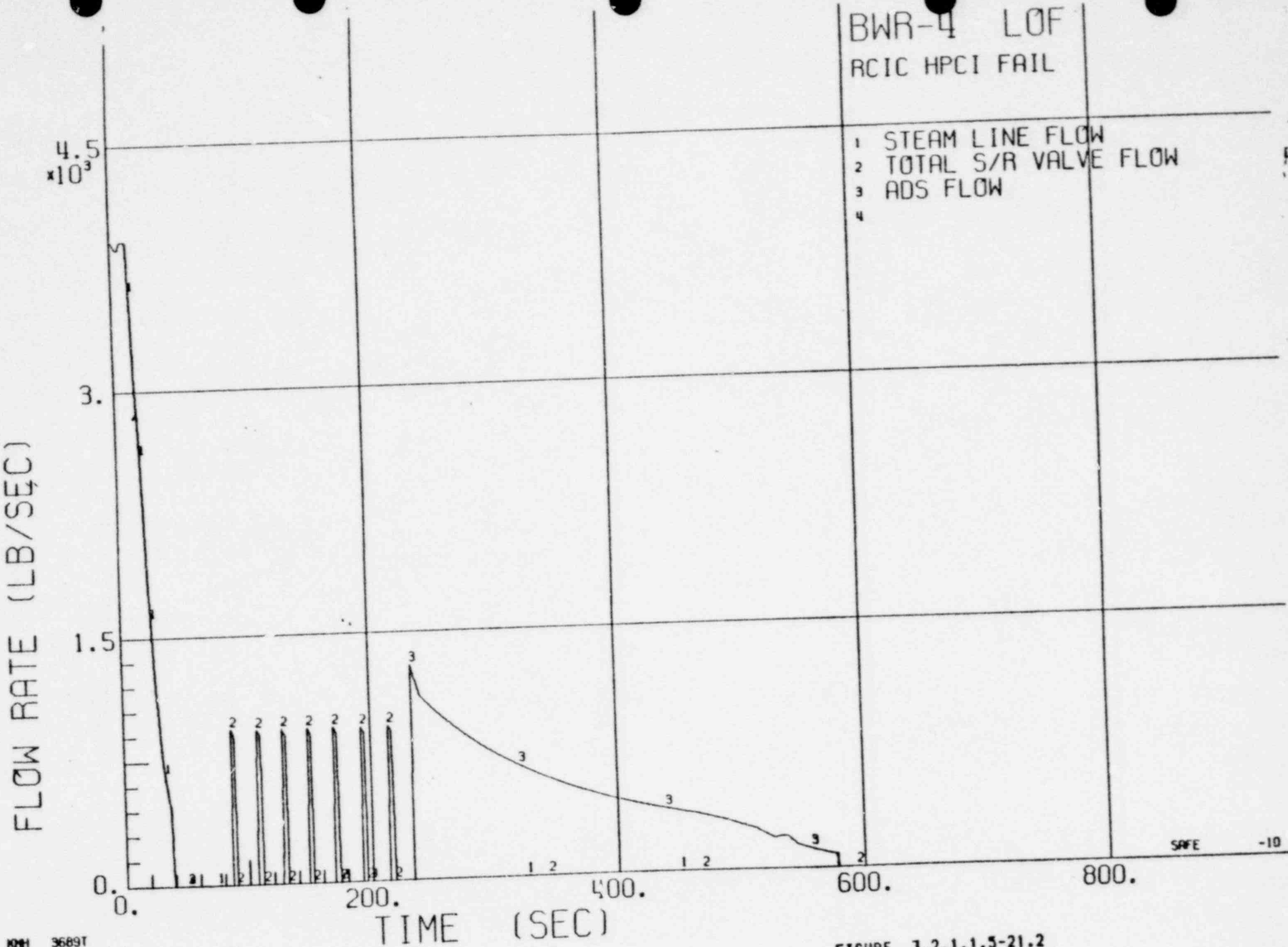
KMH 8379T
122879 1818.8

FIGURE 3.2.1.1.5-20.8



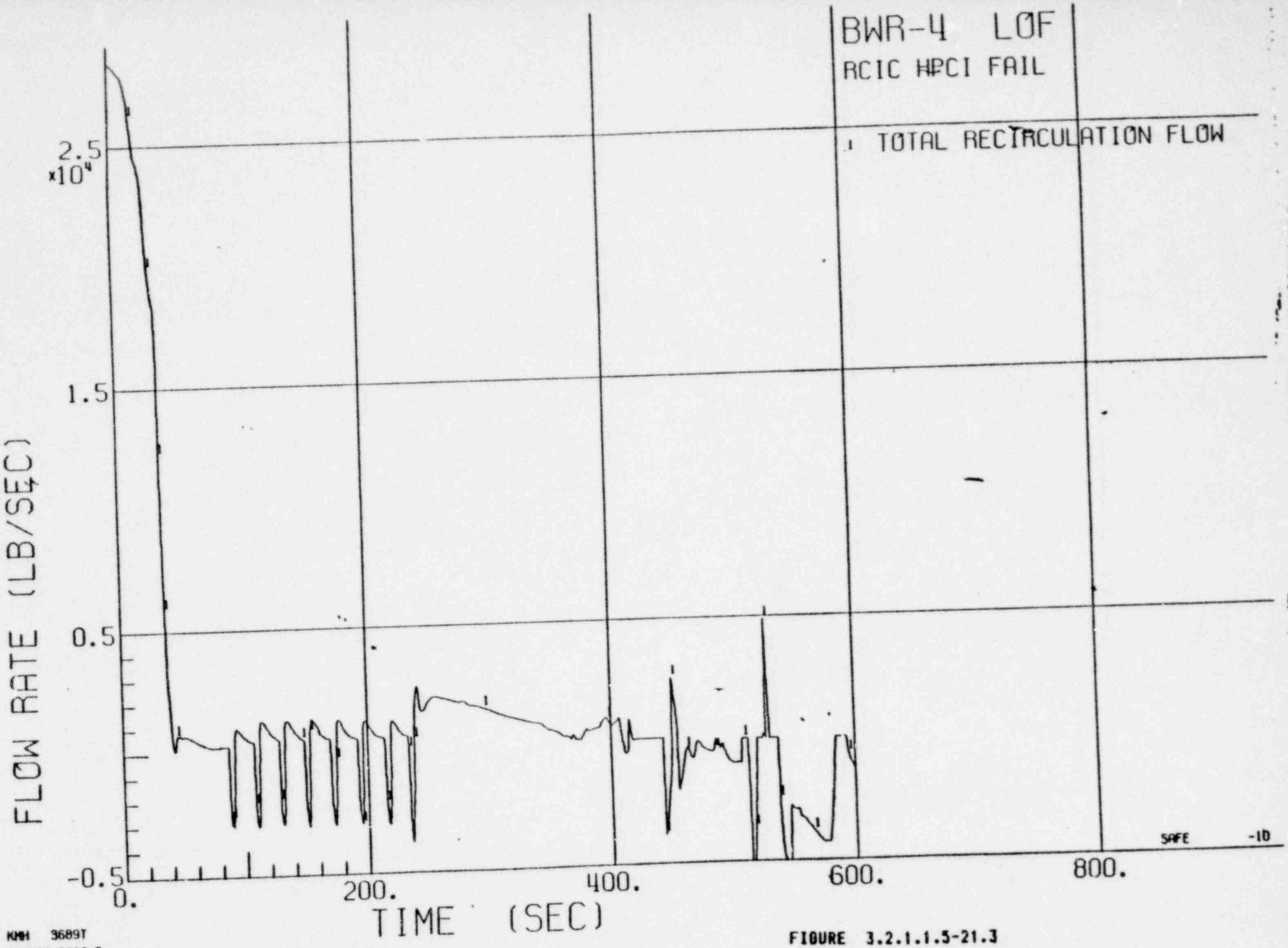
KMH 3689T
011080 0002.9

FIGURE 3.2.1.1.5-21.1



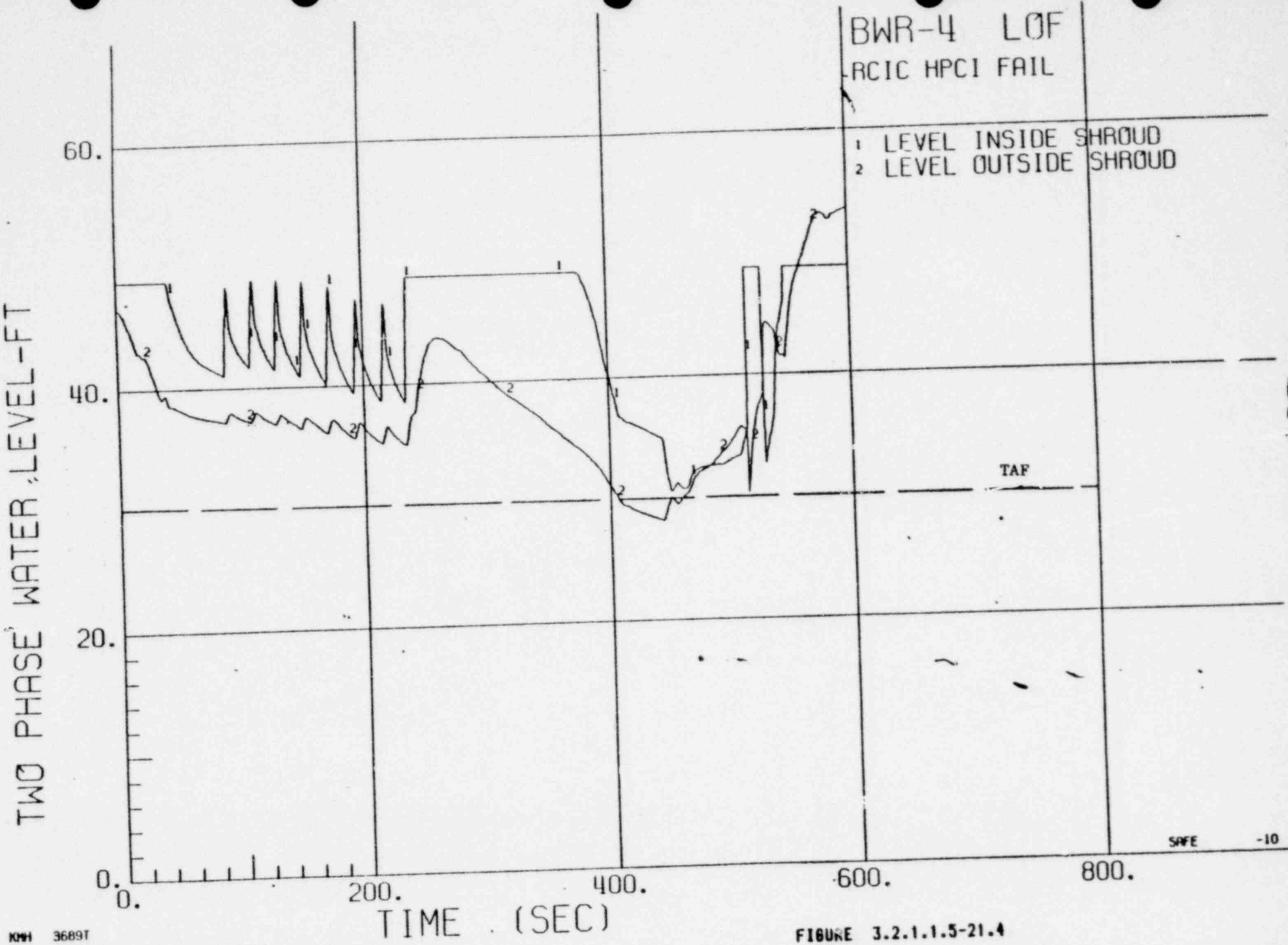
KMH 3689T
011080 0002.9

FIGURE 3.2.1.1.5-21.2



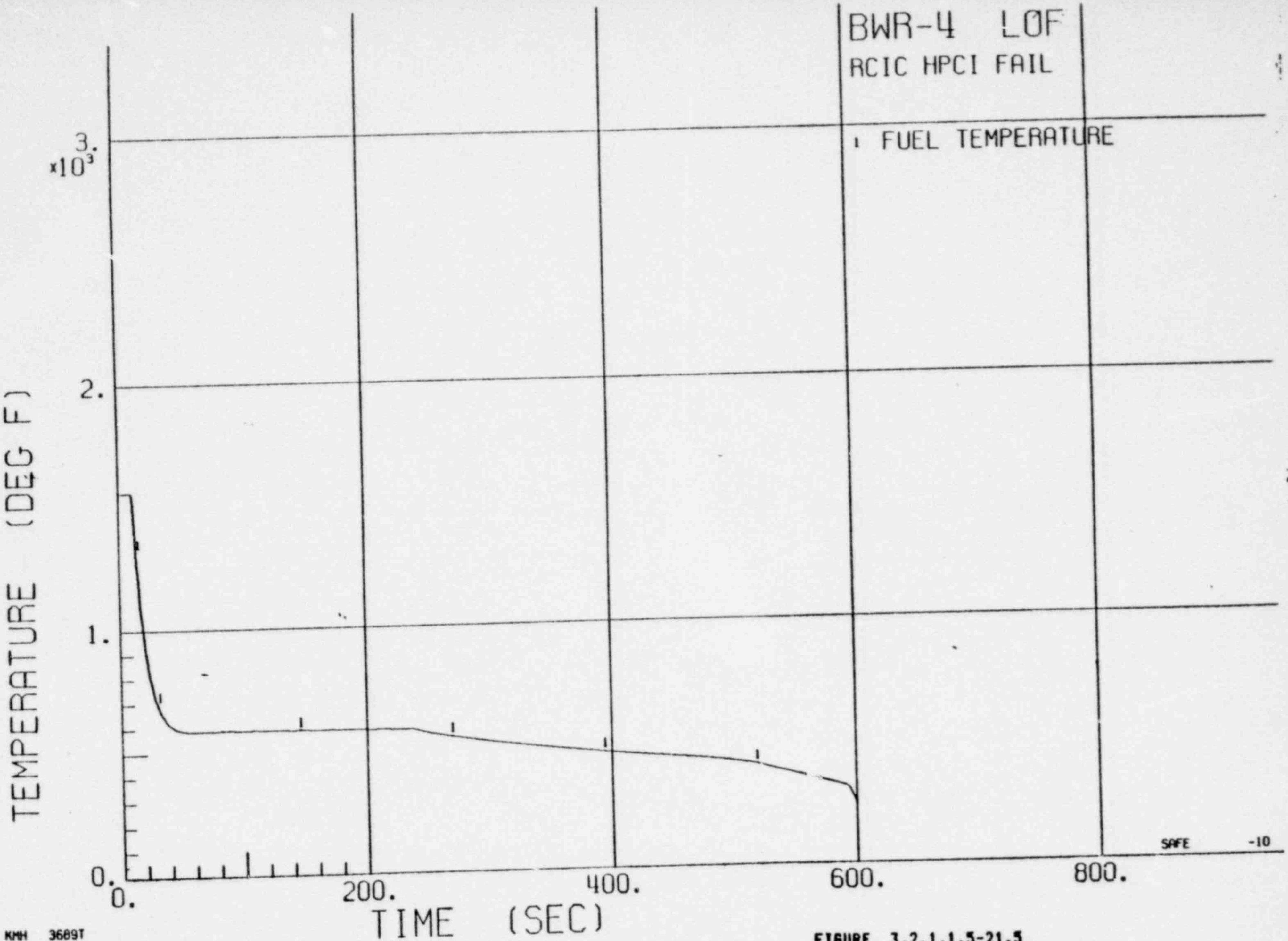
KMH 3689T
011080 0002.9

FIGURE 3.2.1.1.5-21.3



KMH 3689T
011080 0002.9

FIGURE 3.2.1.1.5-21.4



KMH 3689T
011080 0002.9

FIGURE 3.2.1.1.5-21.5

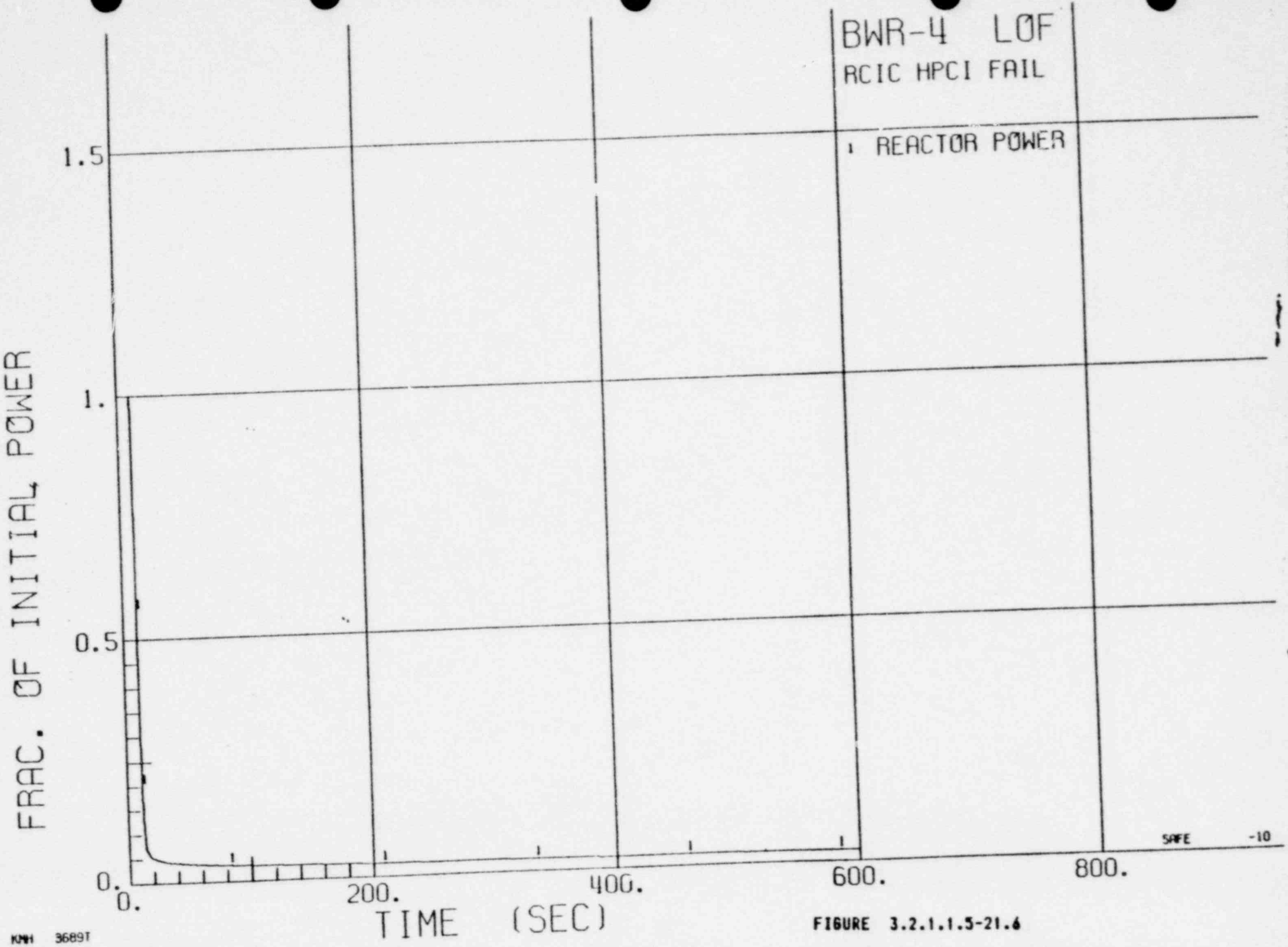
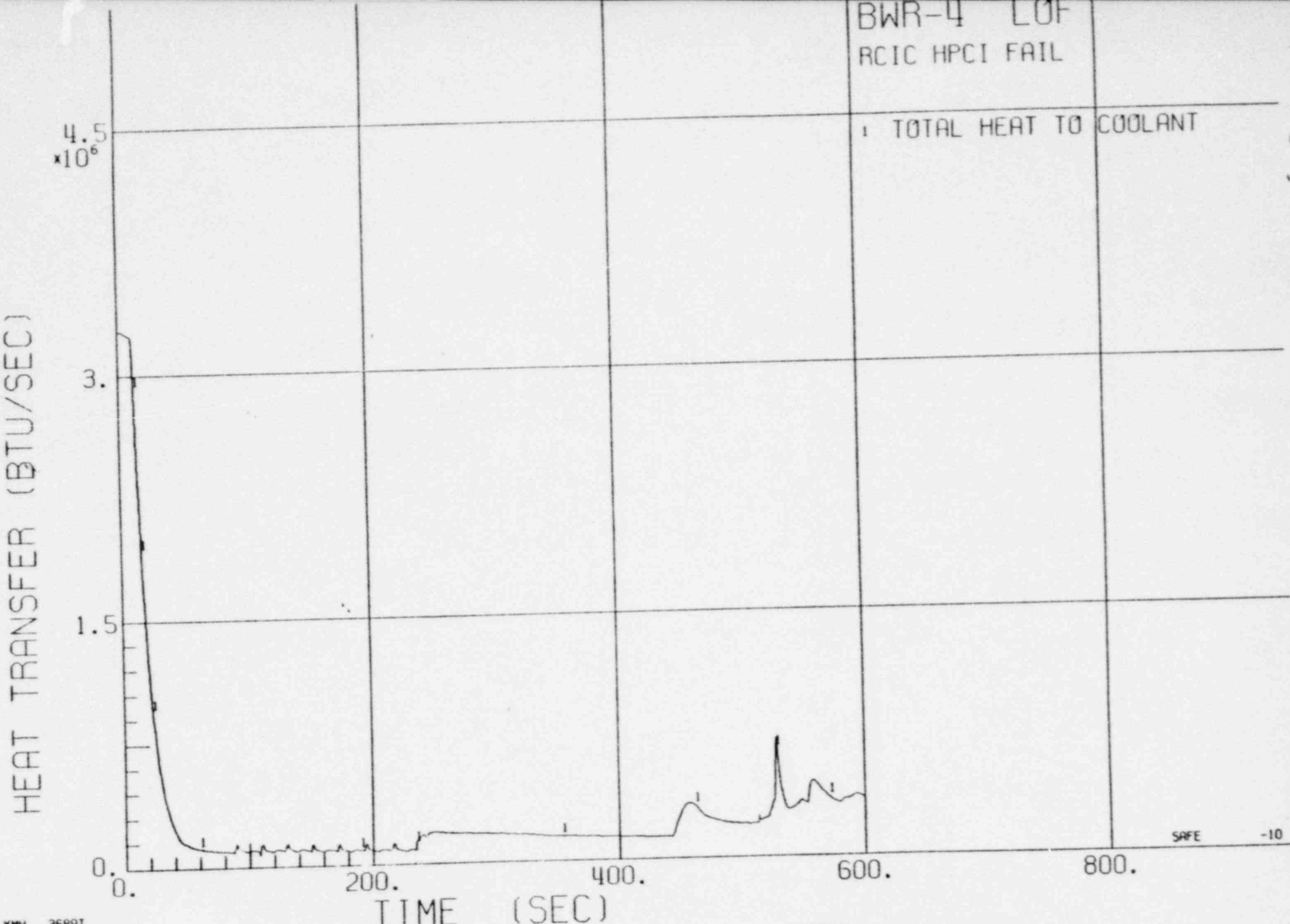


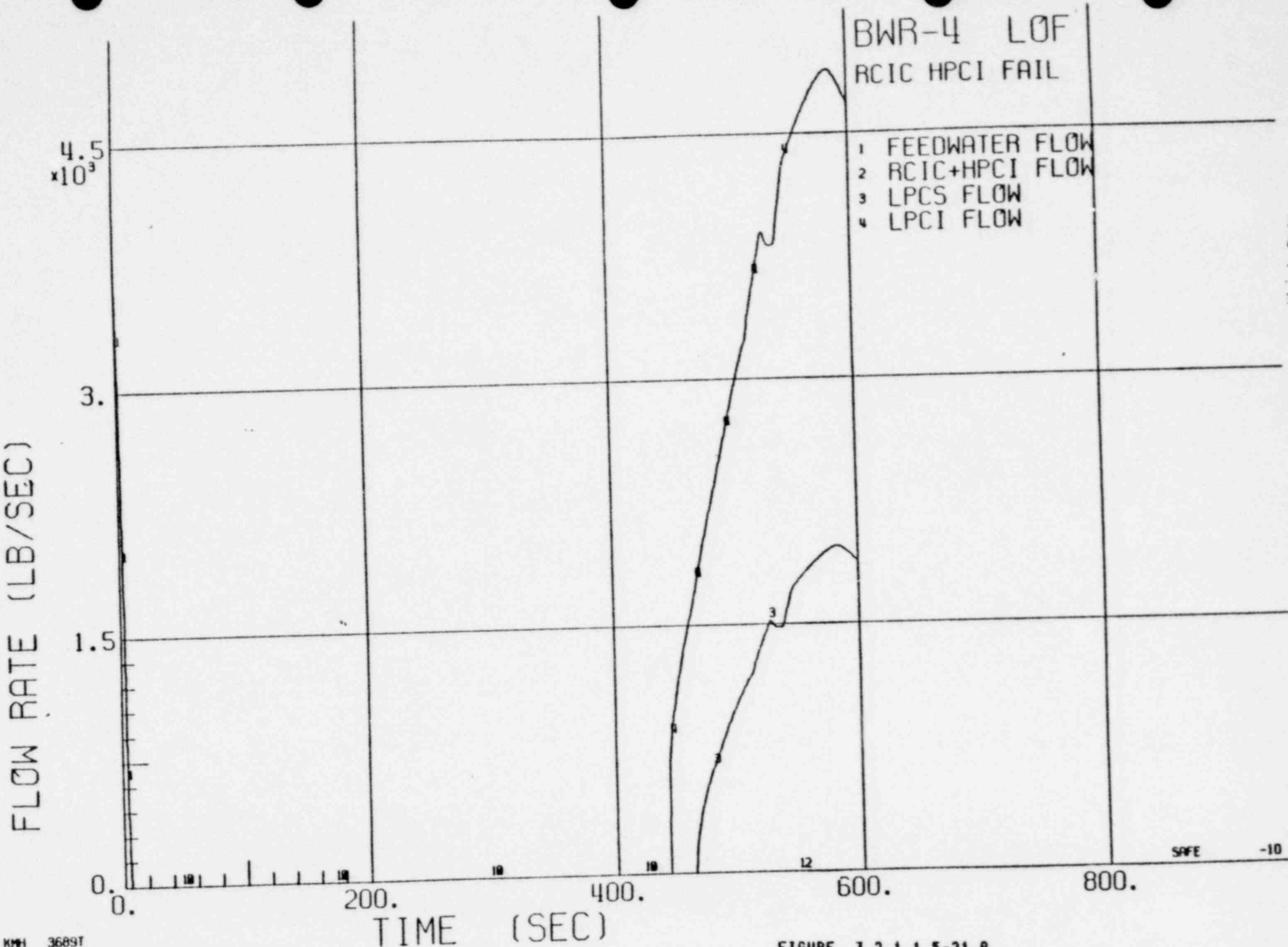
FIGURE 3.2.1.1.5-21.6

KMH 3689T
011080 0002.9



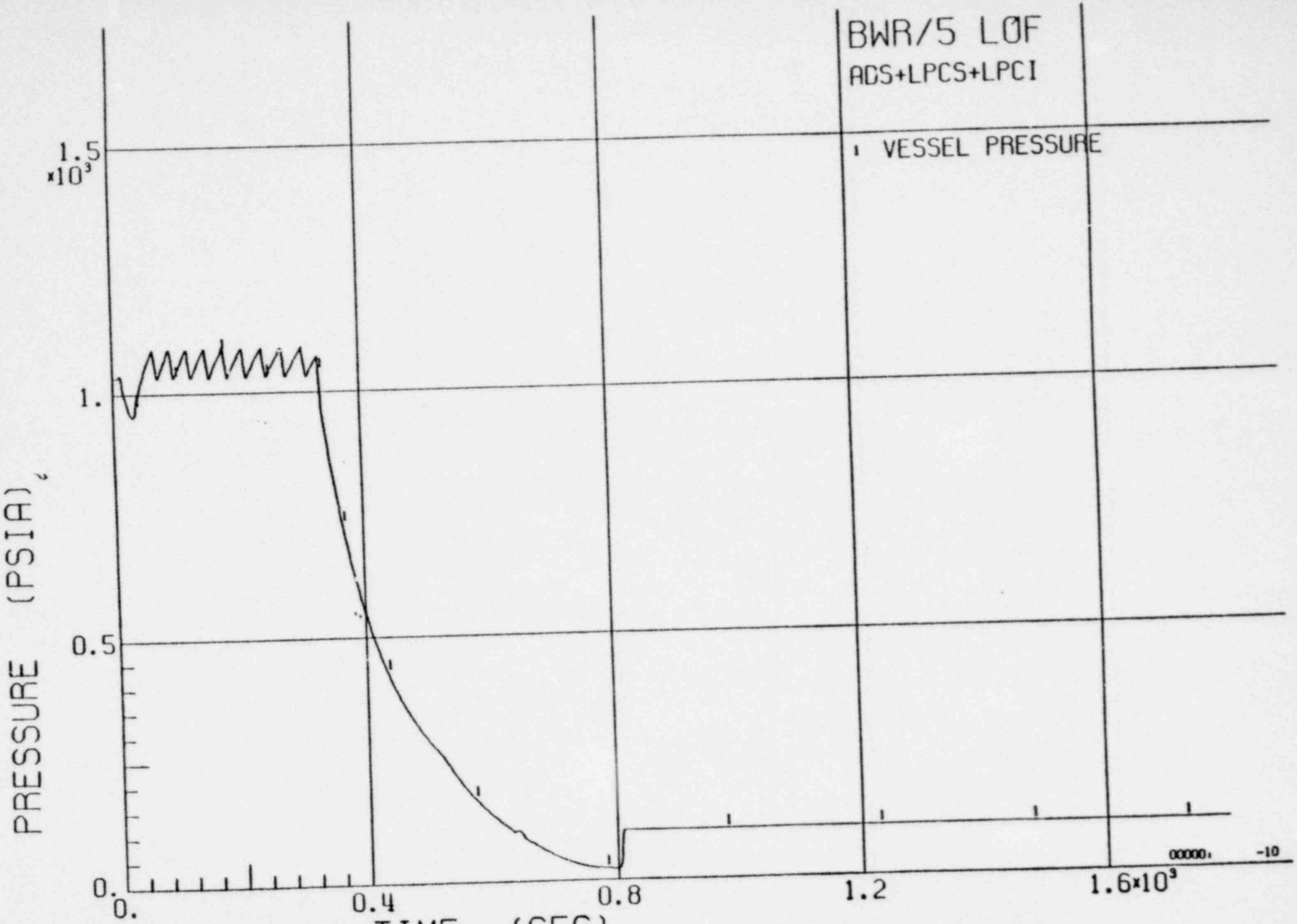
KMH 3689T
011080 0010.3

FIGURE 3.2.1.1.5-21.7



KMH 3689T
011080 0010.3

FIGURE 3.2.1.1.5-21.8



RJB 7089T
120879 0555.8

FIGURE 3.2.1.1.5-22.1

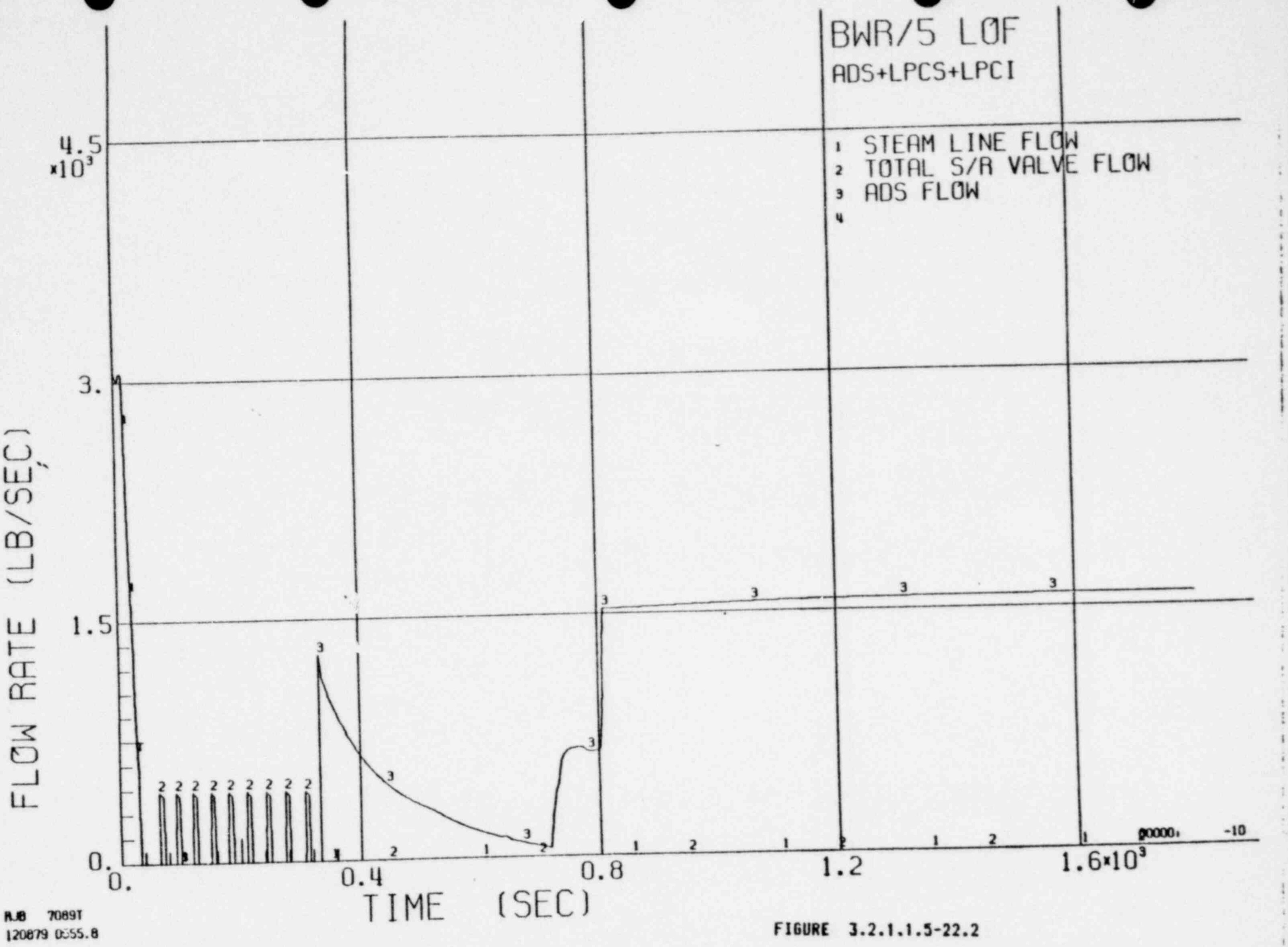
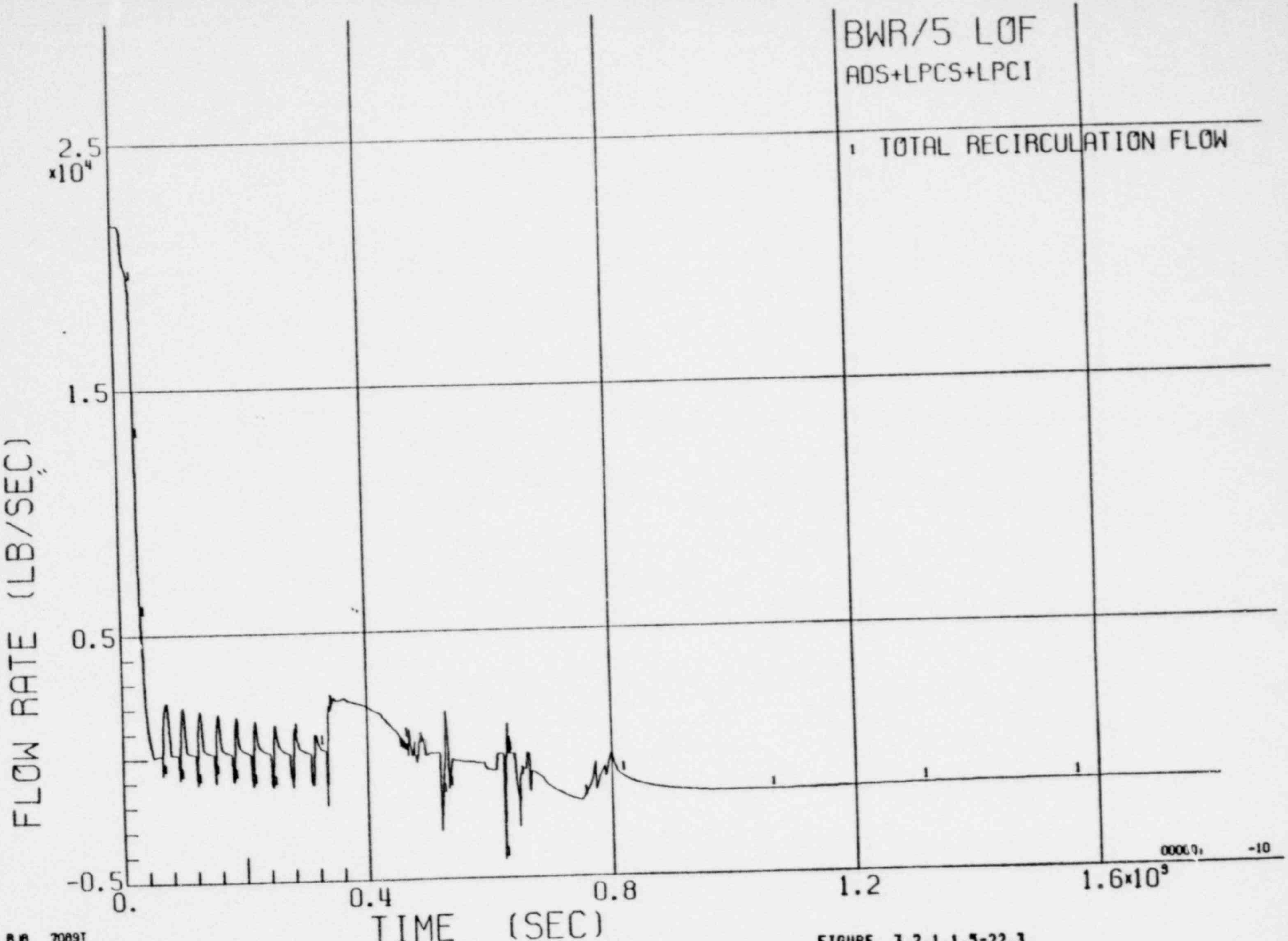


FIGURE 3.2.1.1.5-22.2



RJB 7089T
120879 0555.8

FIGURE 3.2.1.1.5-22.3

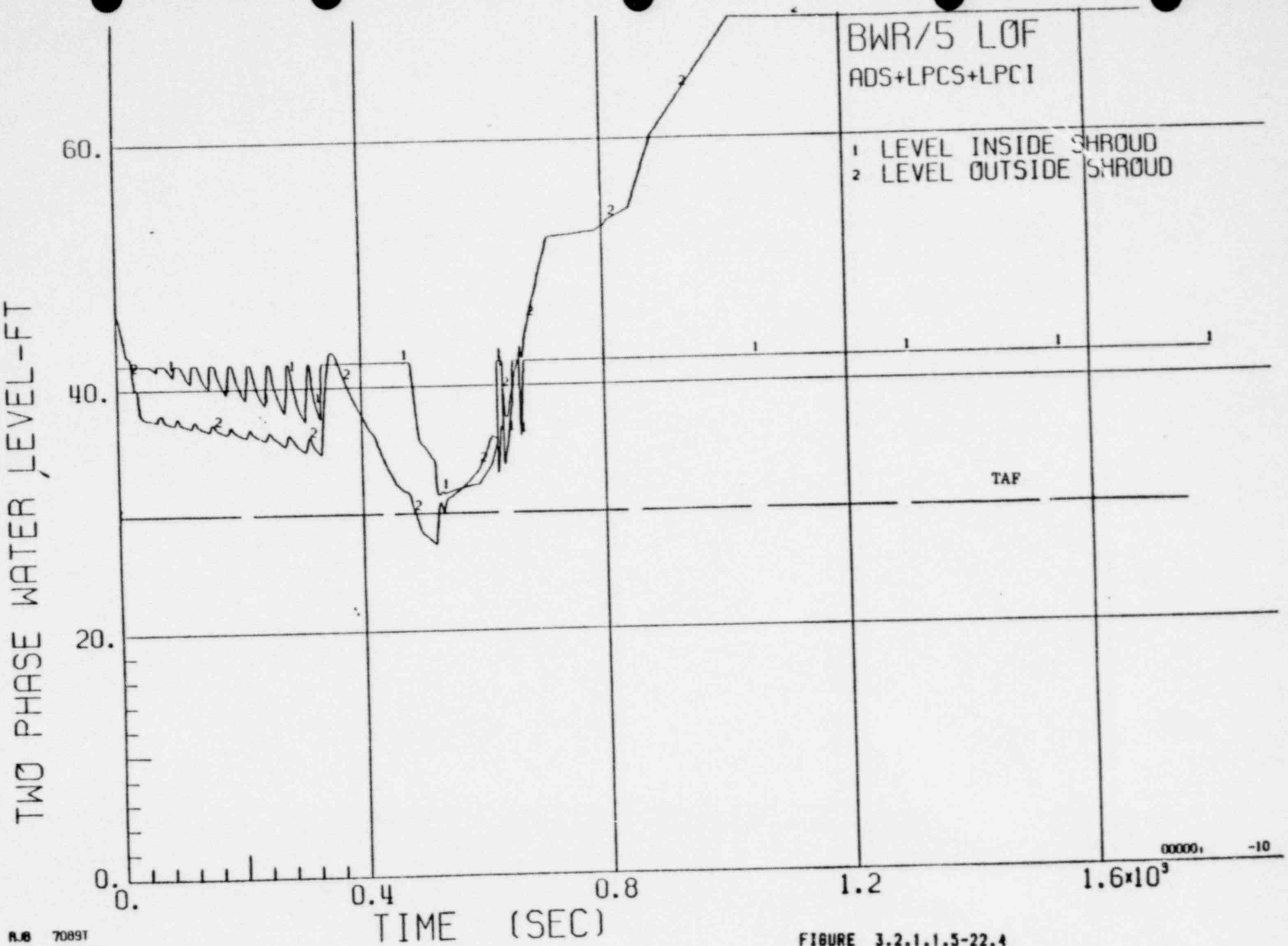


FIGURE 3.2.1.1.5-22.4

RJB 7089T
120879 0555.8

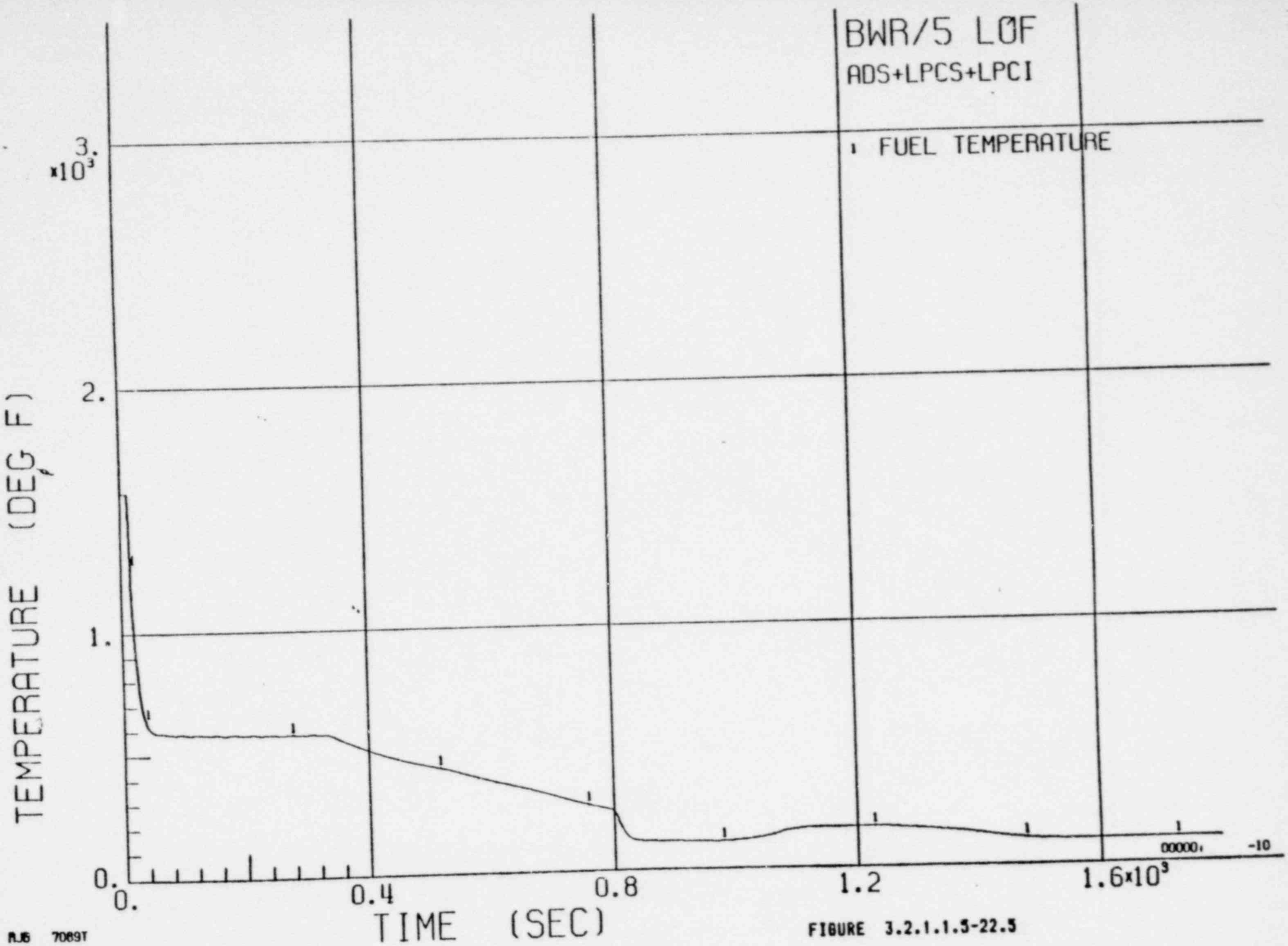
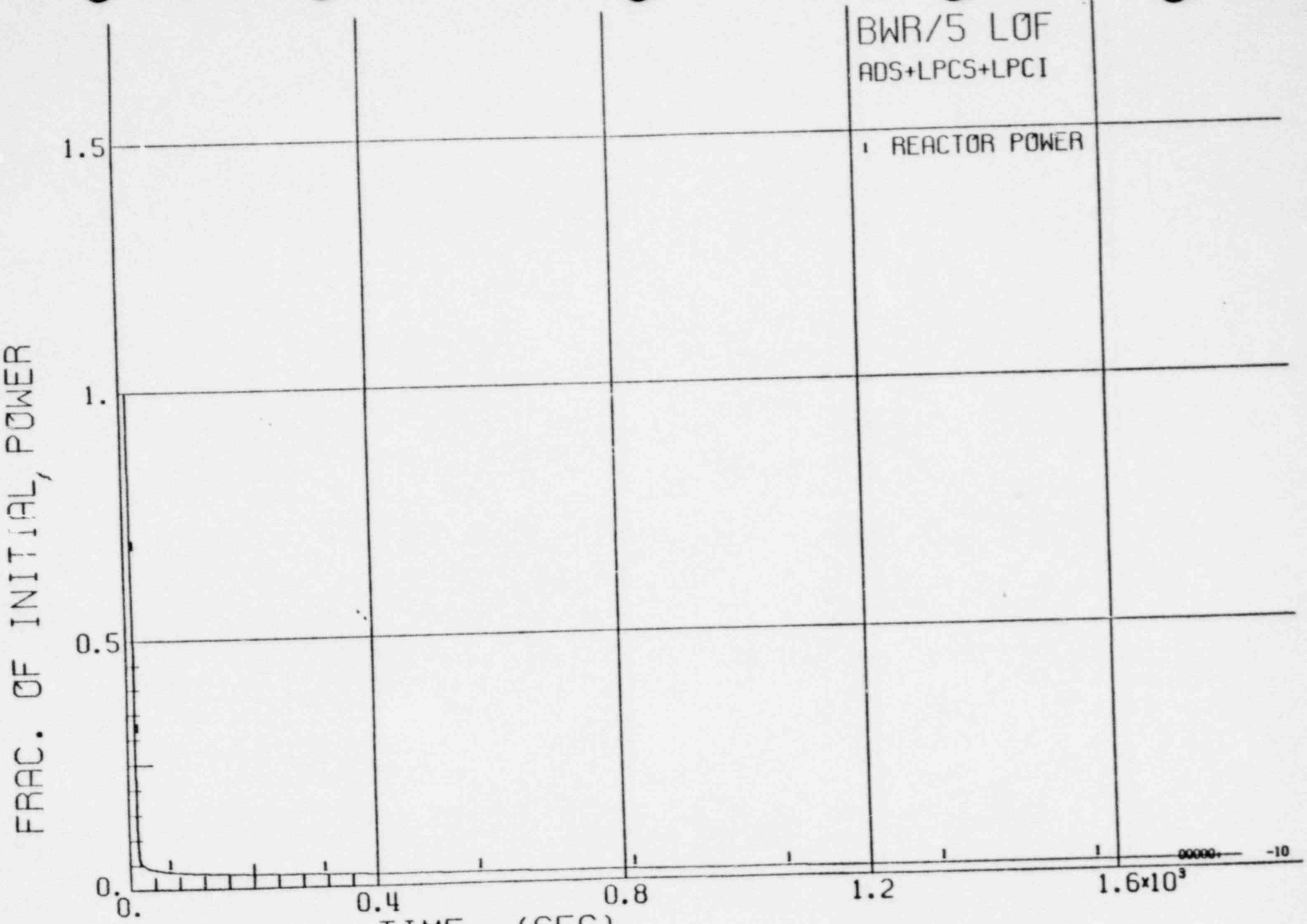


FIGURE 3.2.1.1.5-22.5

RJB 7089T
120878 0555.8



RJB 70A9T
120879 0555.8

FIGURE 3.2.1.1.5-22.6

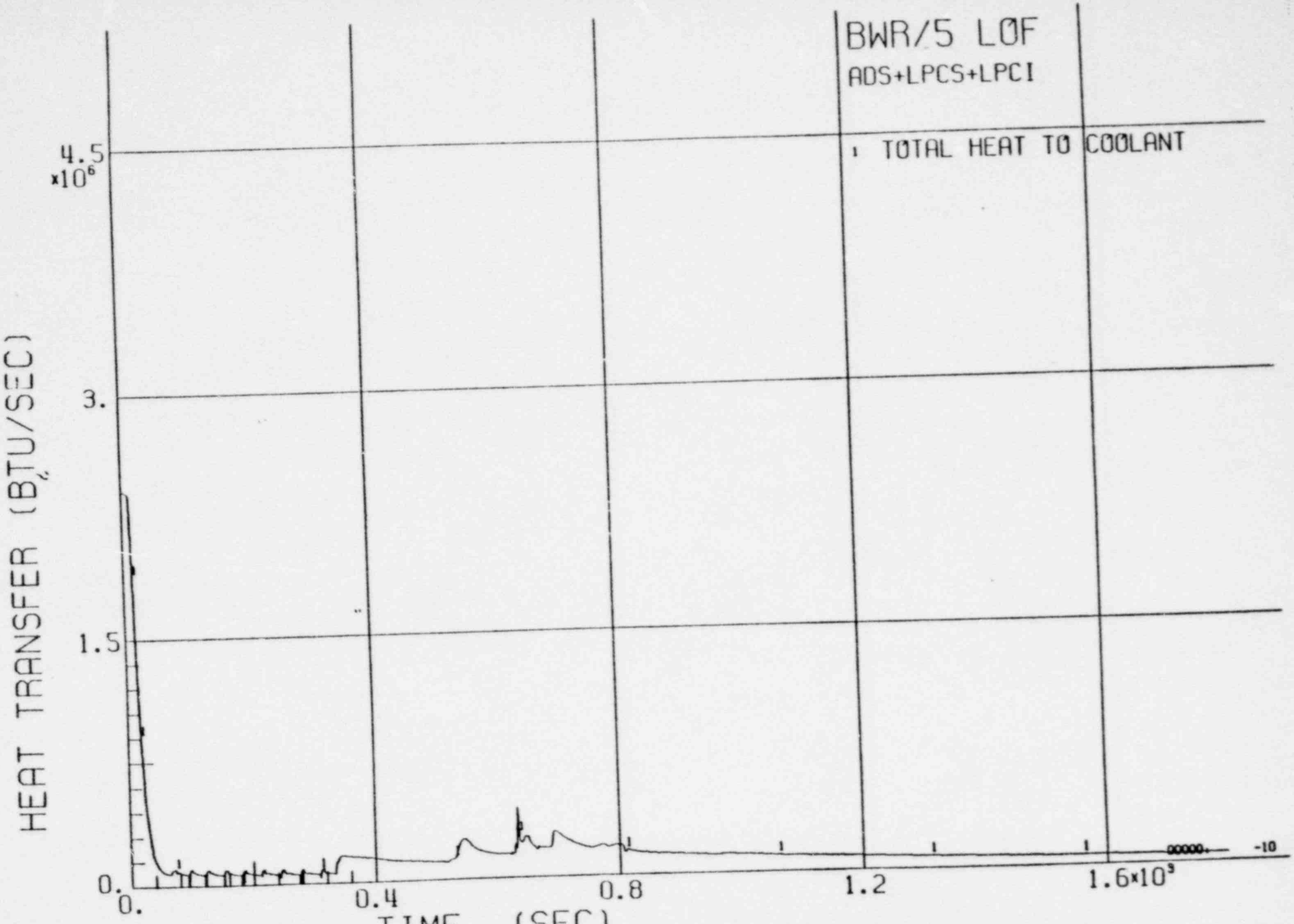


FIGURE 3.2.1.1.5-22.7

RJB 7089T
120879 0558.0

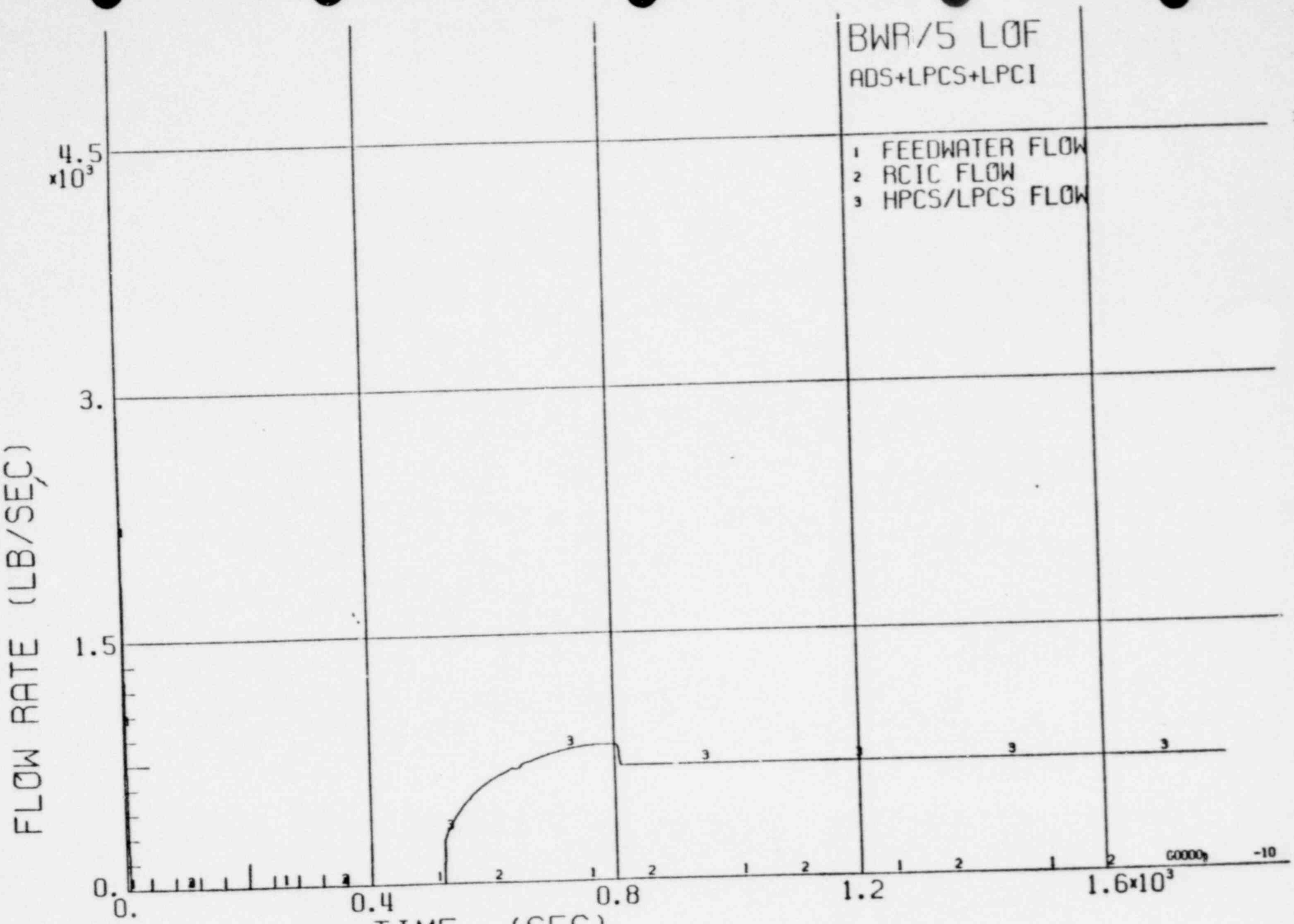
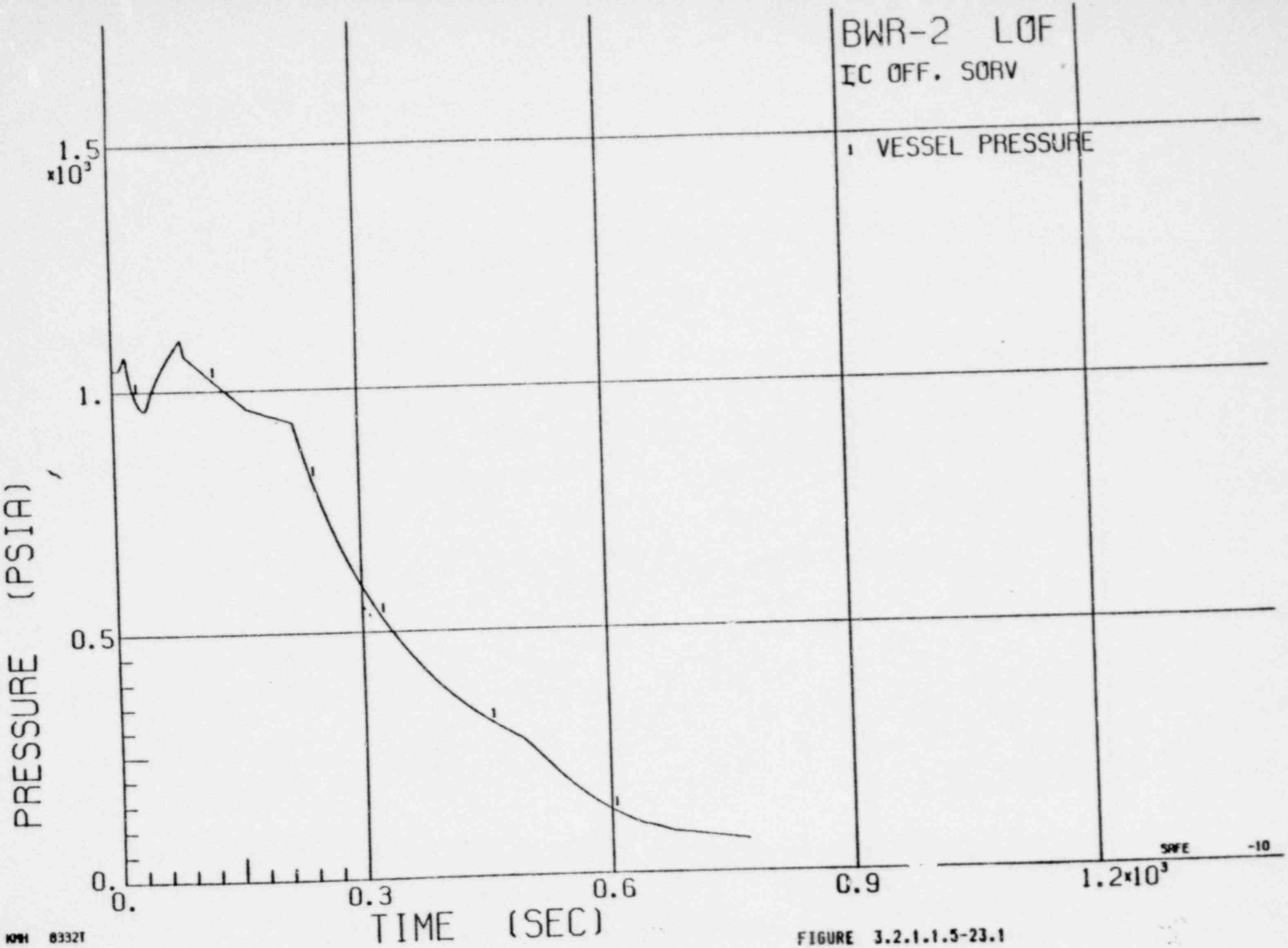


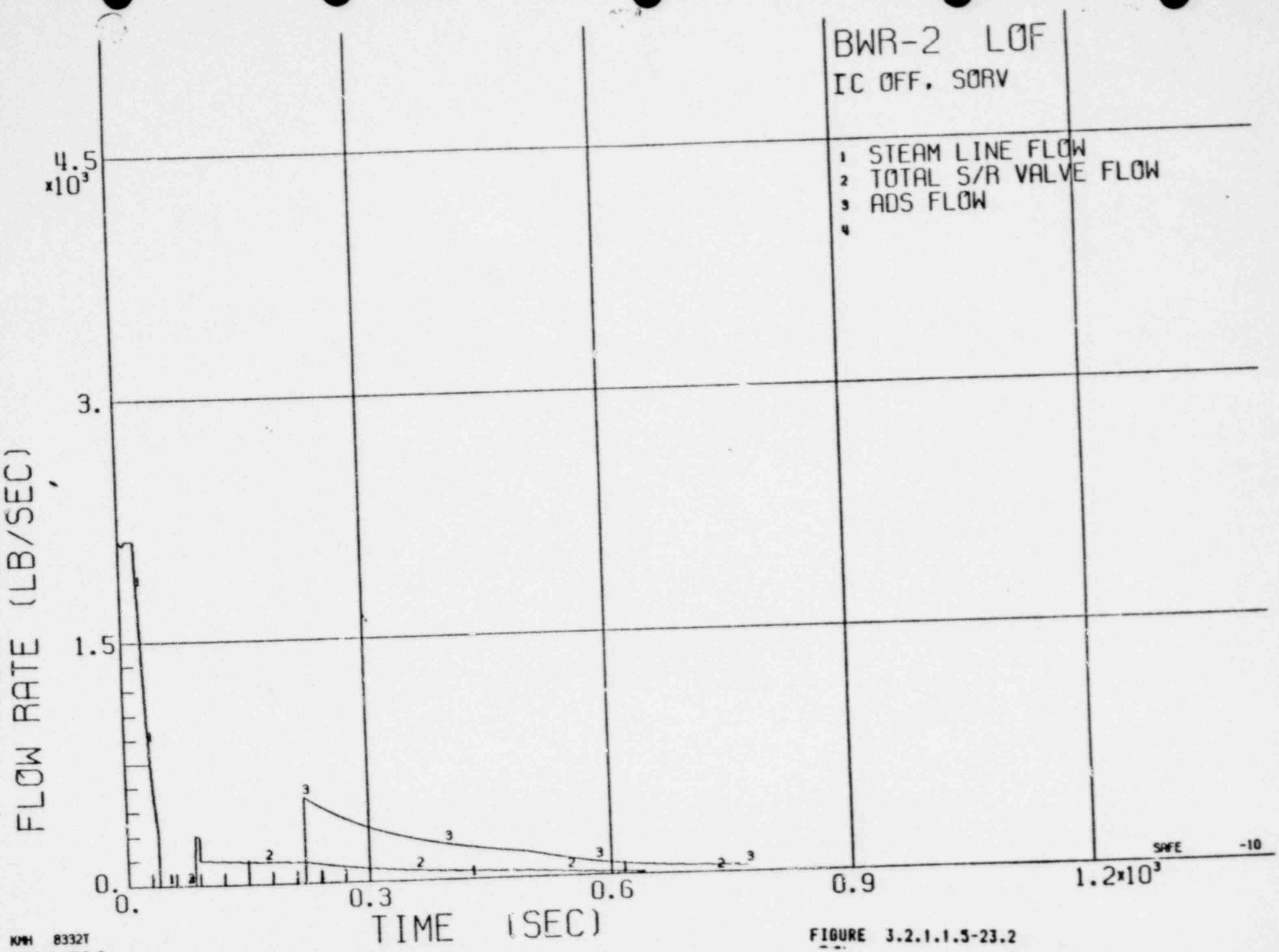
FIGURE 3.2.1.1.5-22.8

AJB 7089T
120879 0558.0



KMH 8332T
122876 1745.2

FIGURE 3.2.1.1.5-23.1



KMH 8332T
122878 1745.2

FIGURE 3.2.1.1.5-23.2

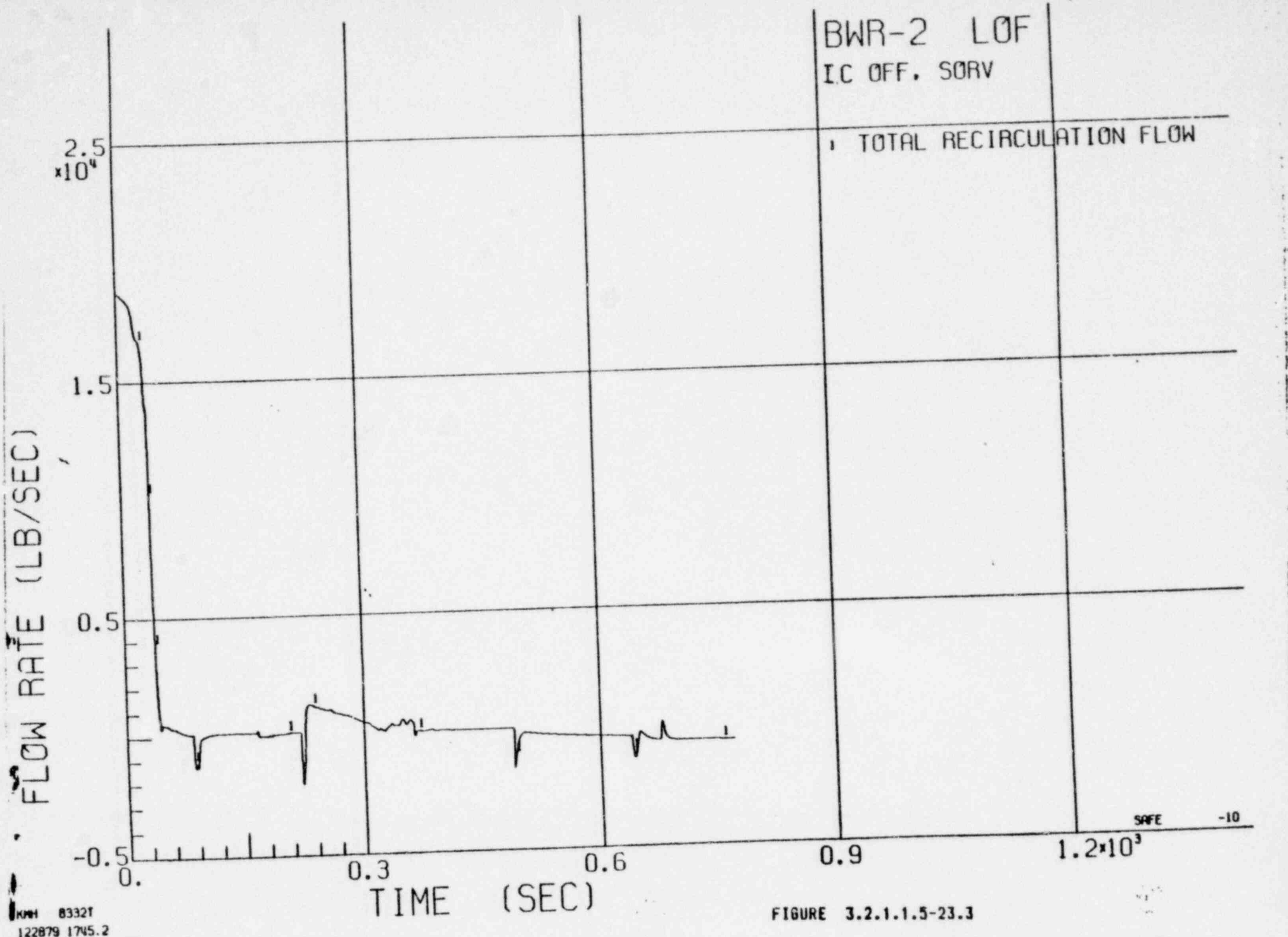


FIGURE 3.2.1.1.5-23.3

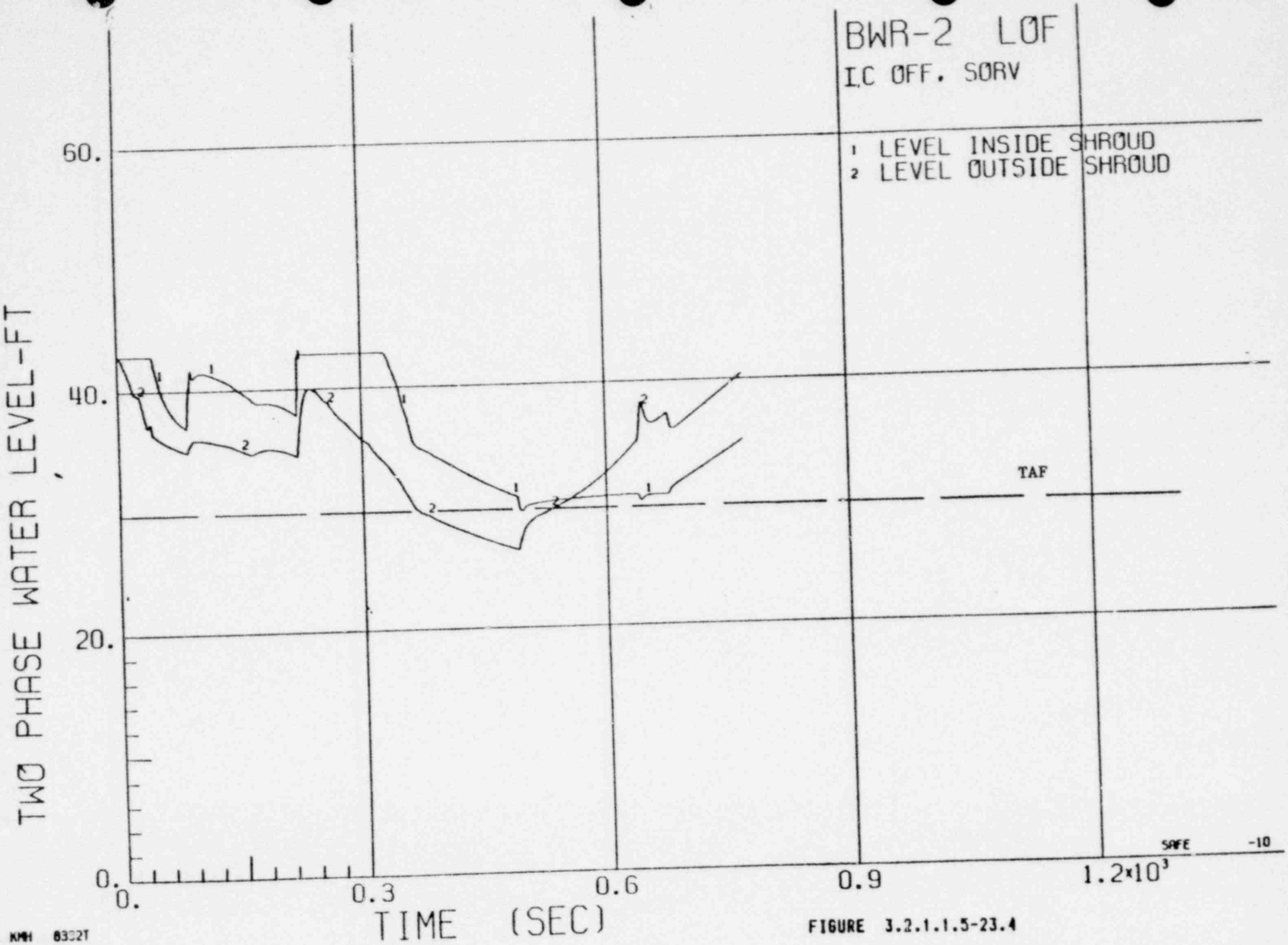


FIGURE 3.2.1.1.5-23.4

KMH 8332T
4 122879 1745.2

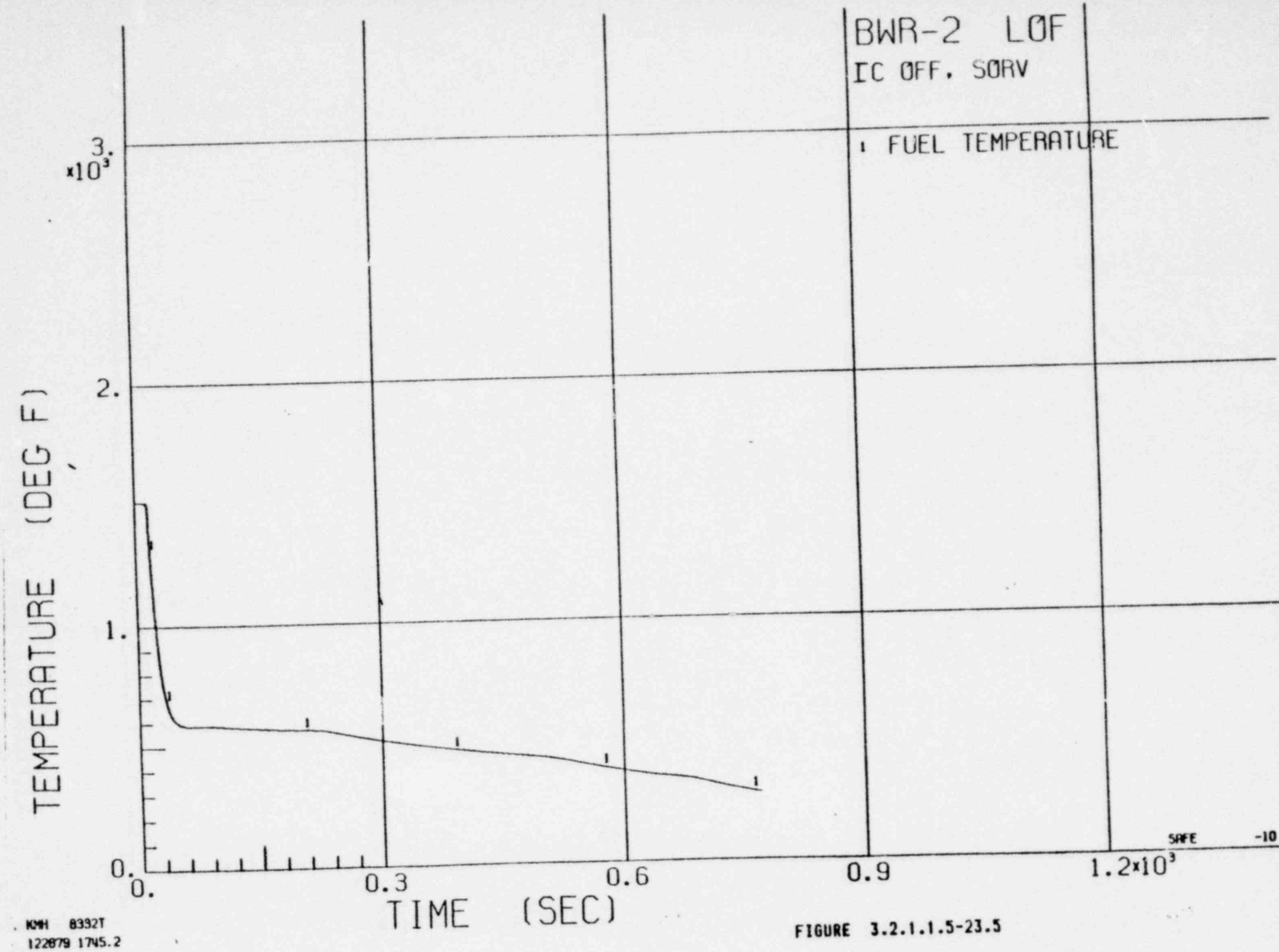


FIGURE 3.2.1.1.5-23.5

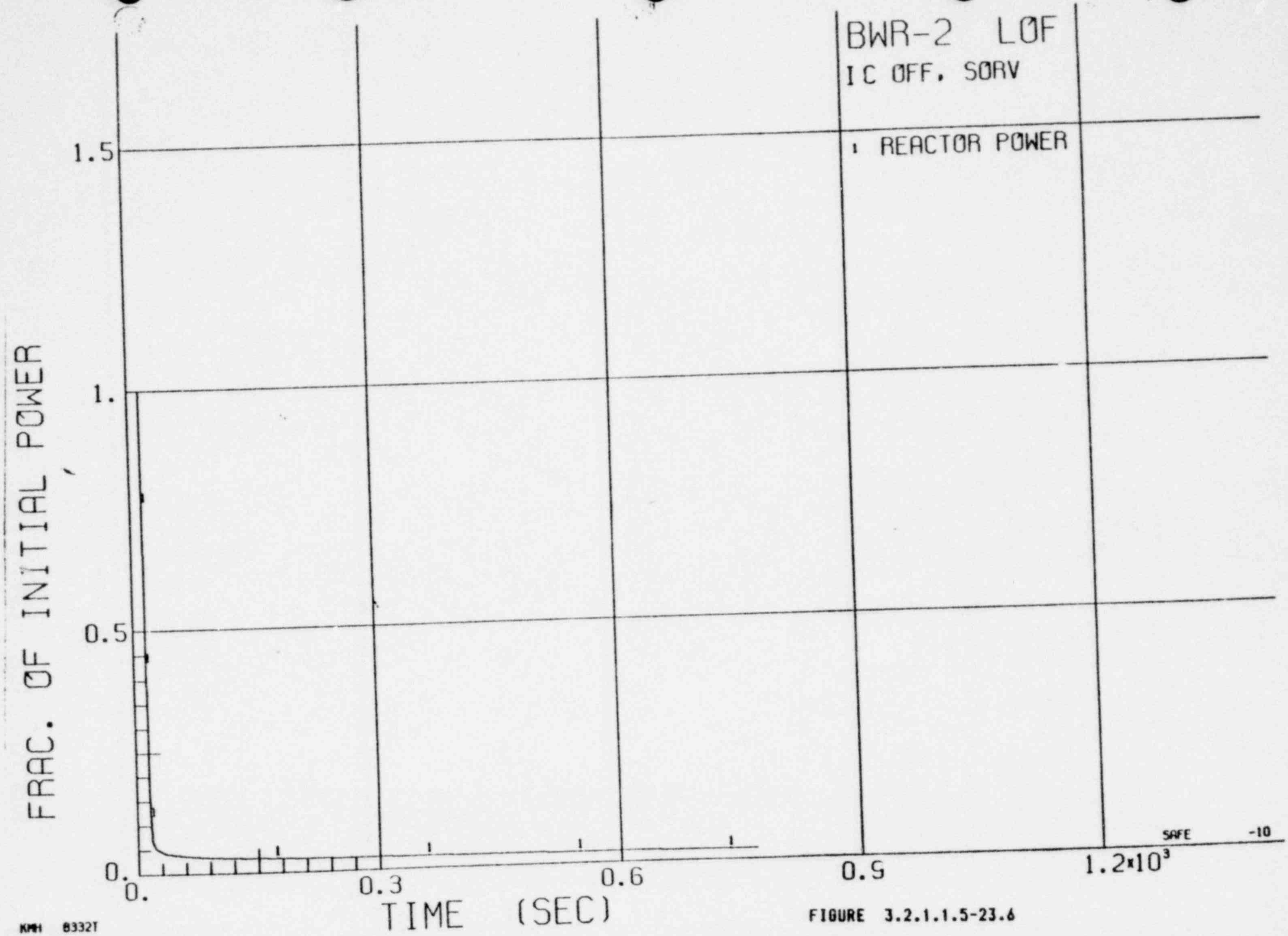
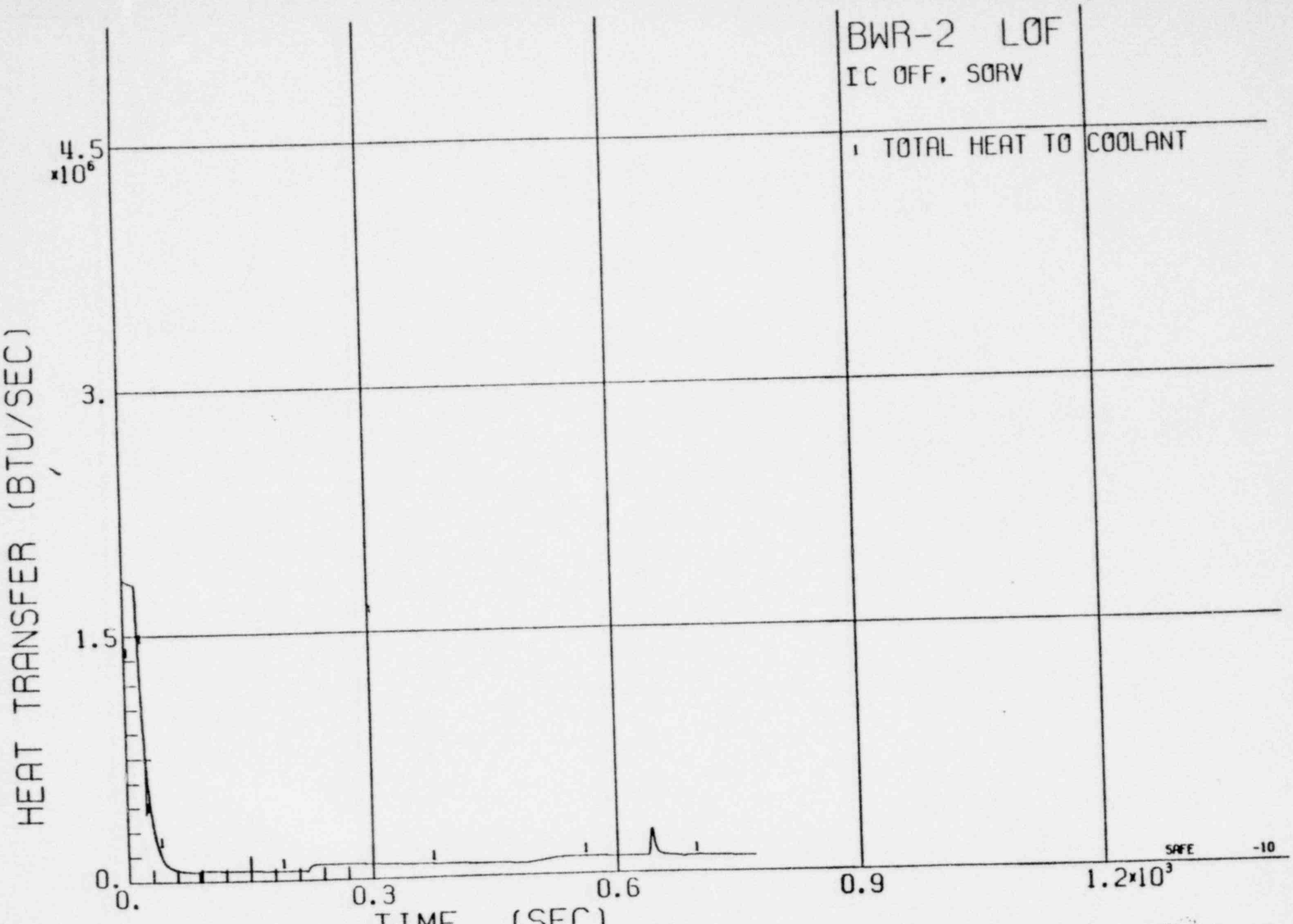


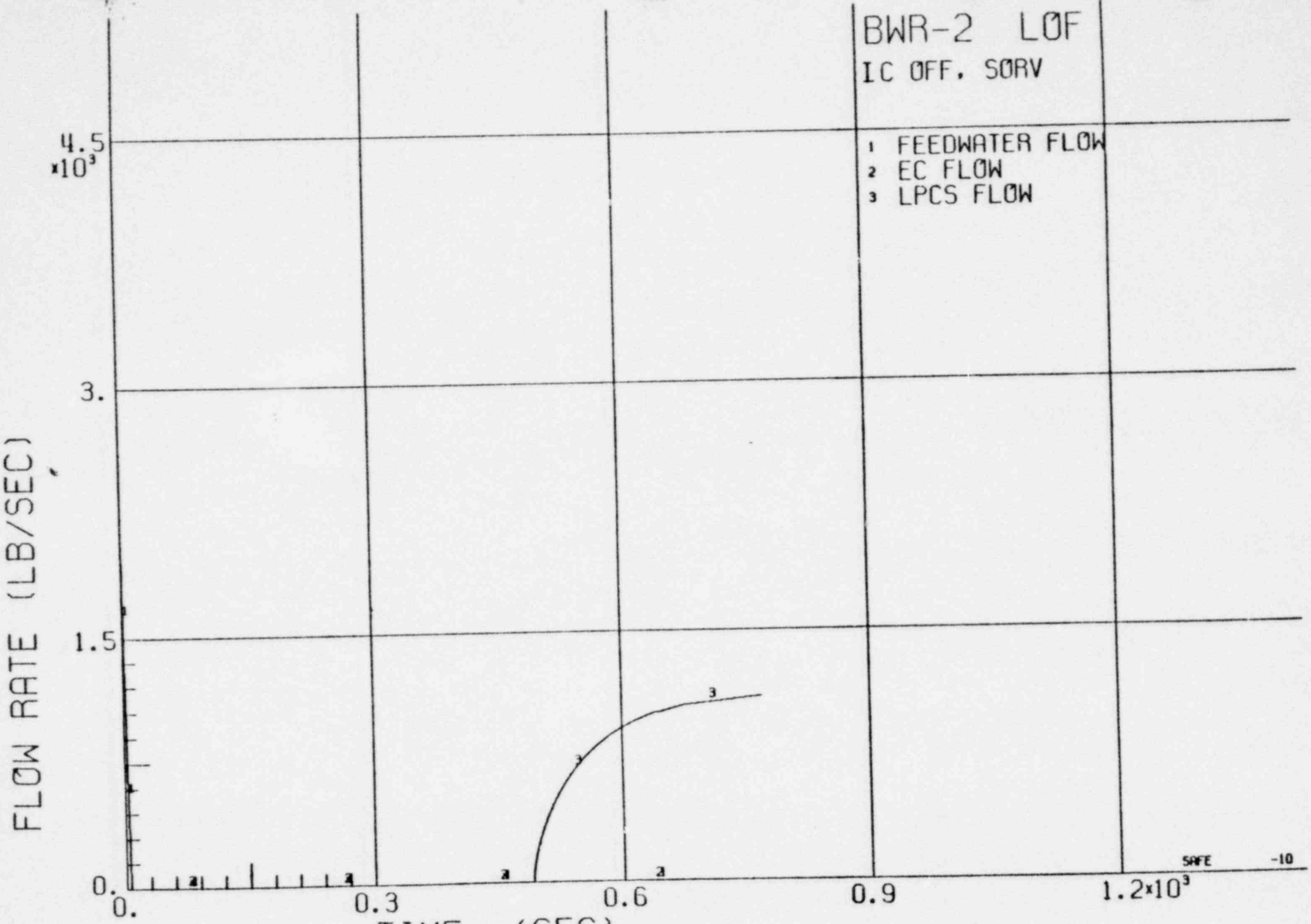
FIGURE 3.2.1.1.5-23.6

KMH 8332T
122879 1745.2



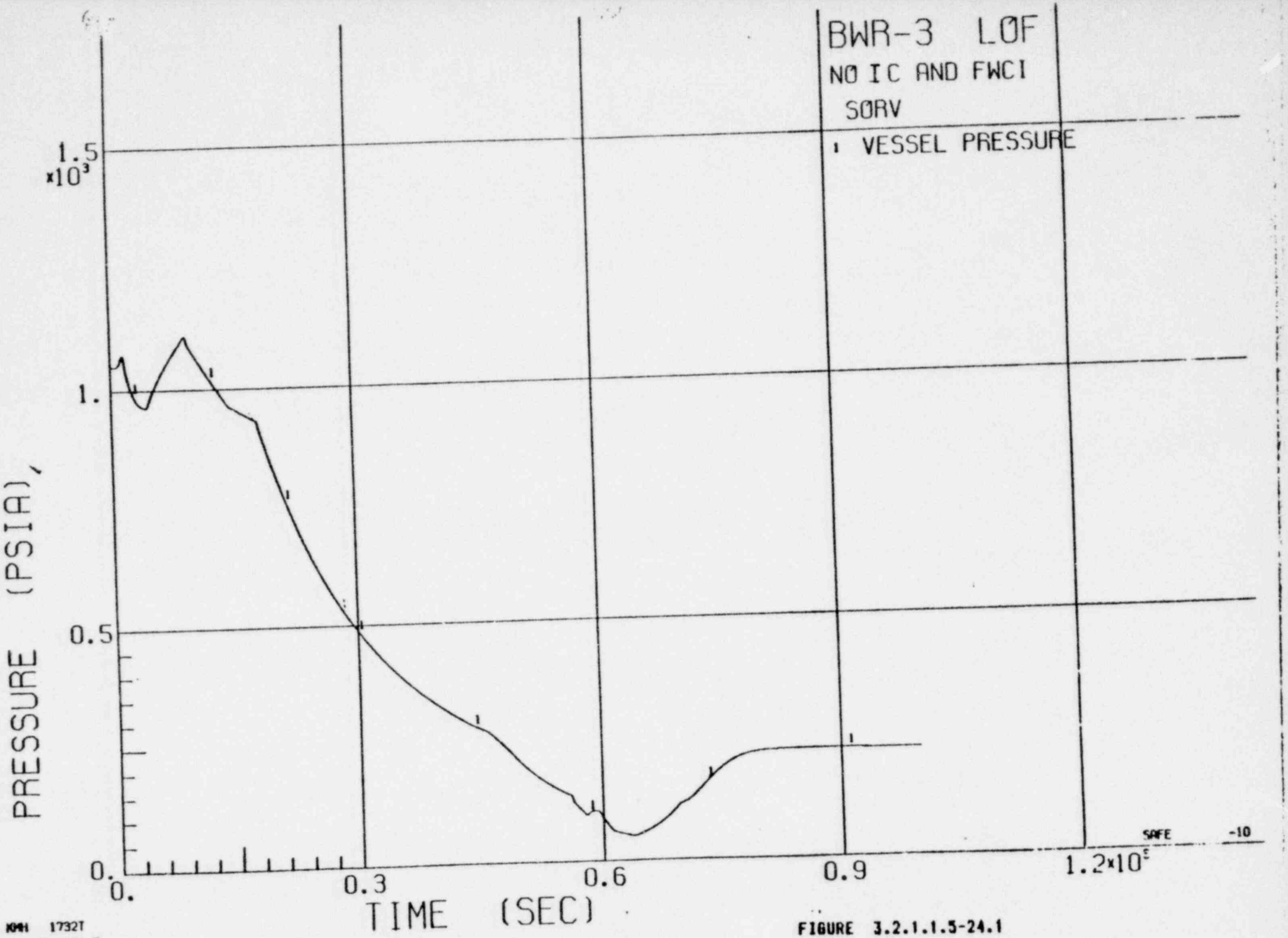
KMH 8332T
122879 1800.8

FIGURE 3.2.1.1.5-23.7



KMH 8332T
122879 1800.8

FIGURE 3.2.1.1.5-23.8



KMH 1732T
121179 2031.5

FIGURE 3.2.1.1.5-24.1

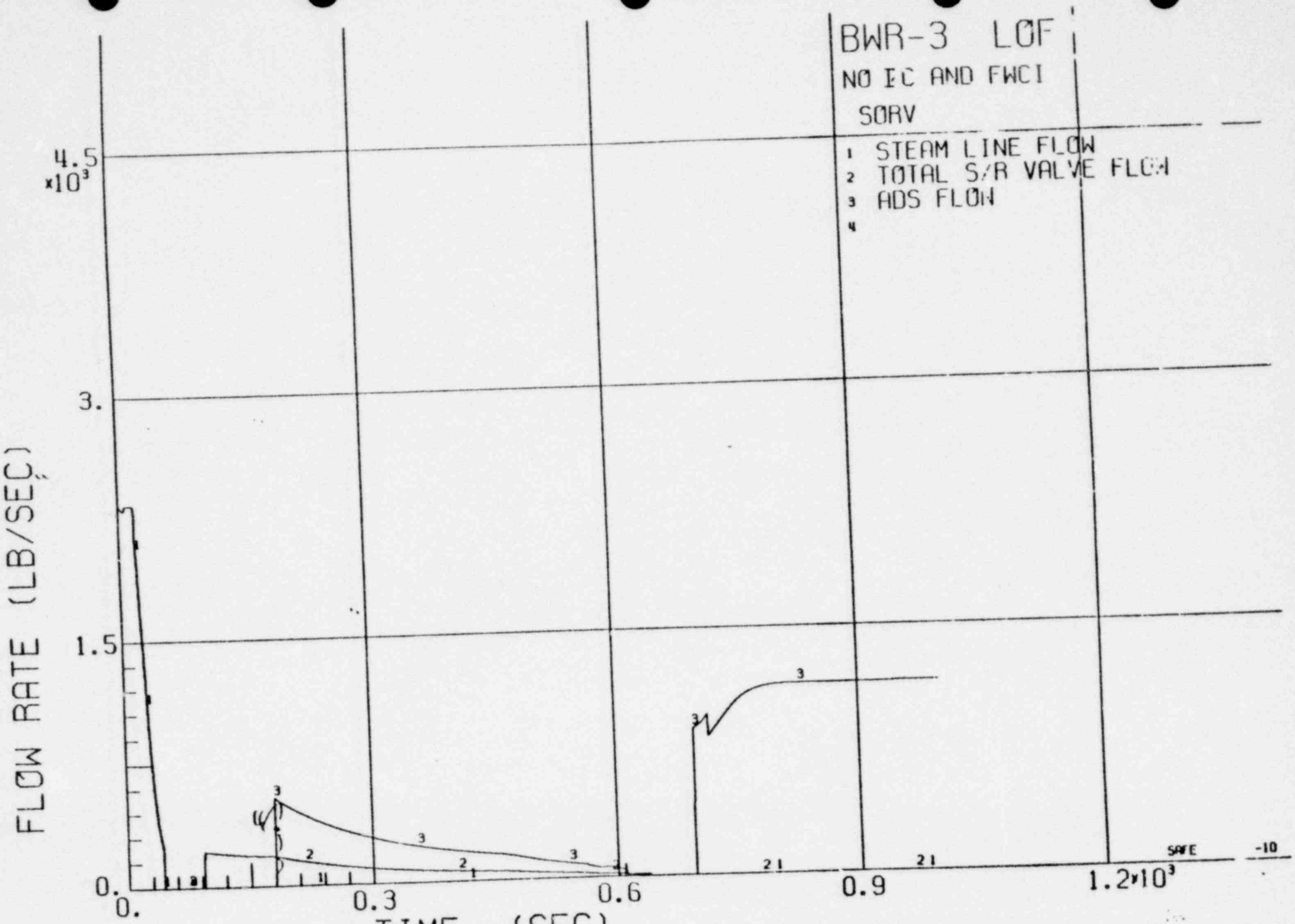
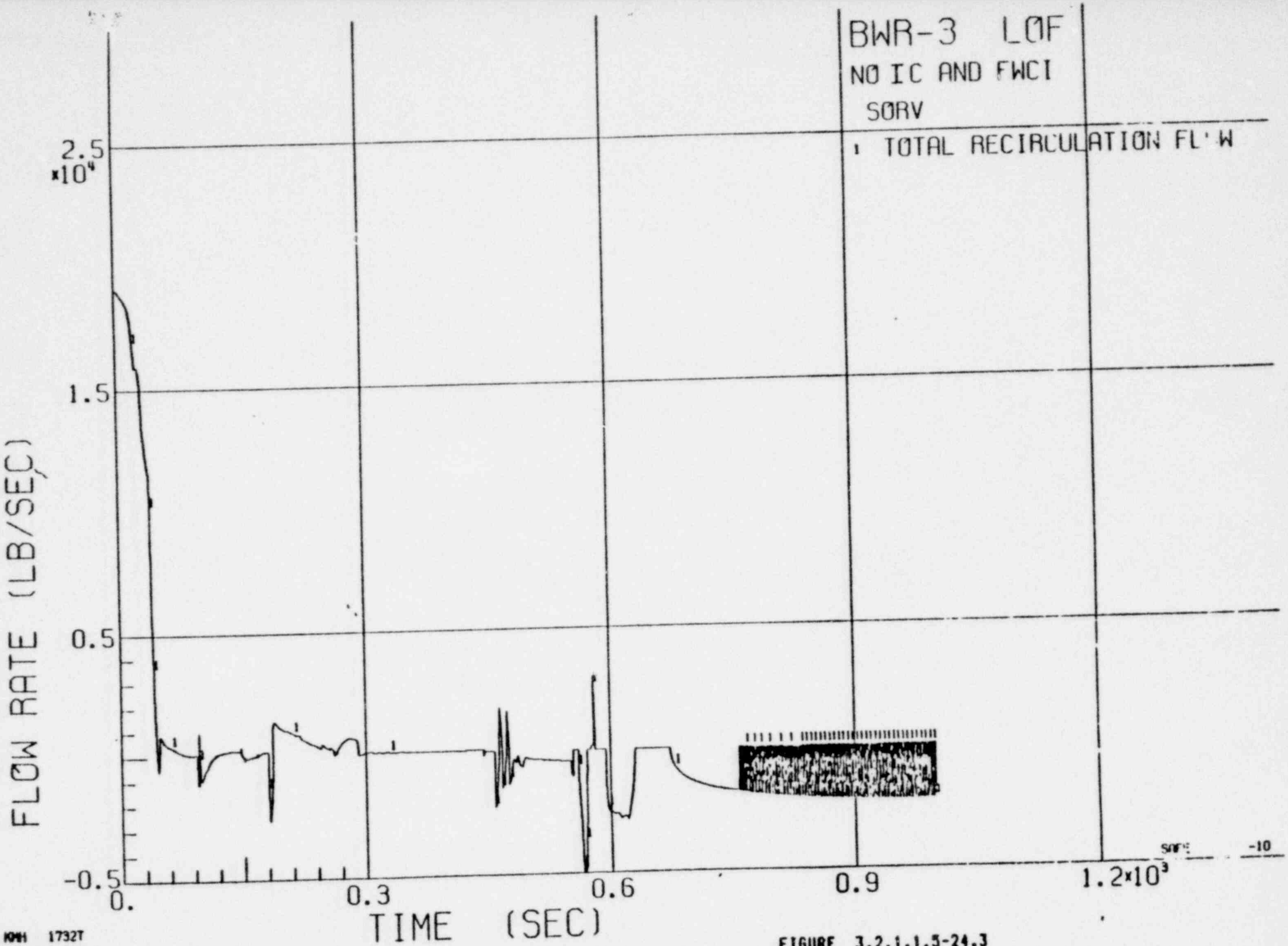


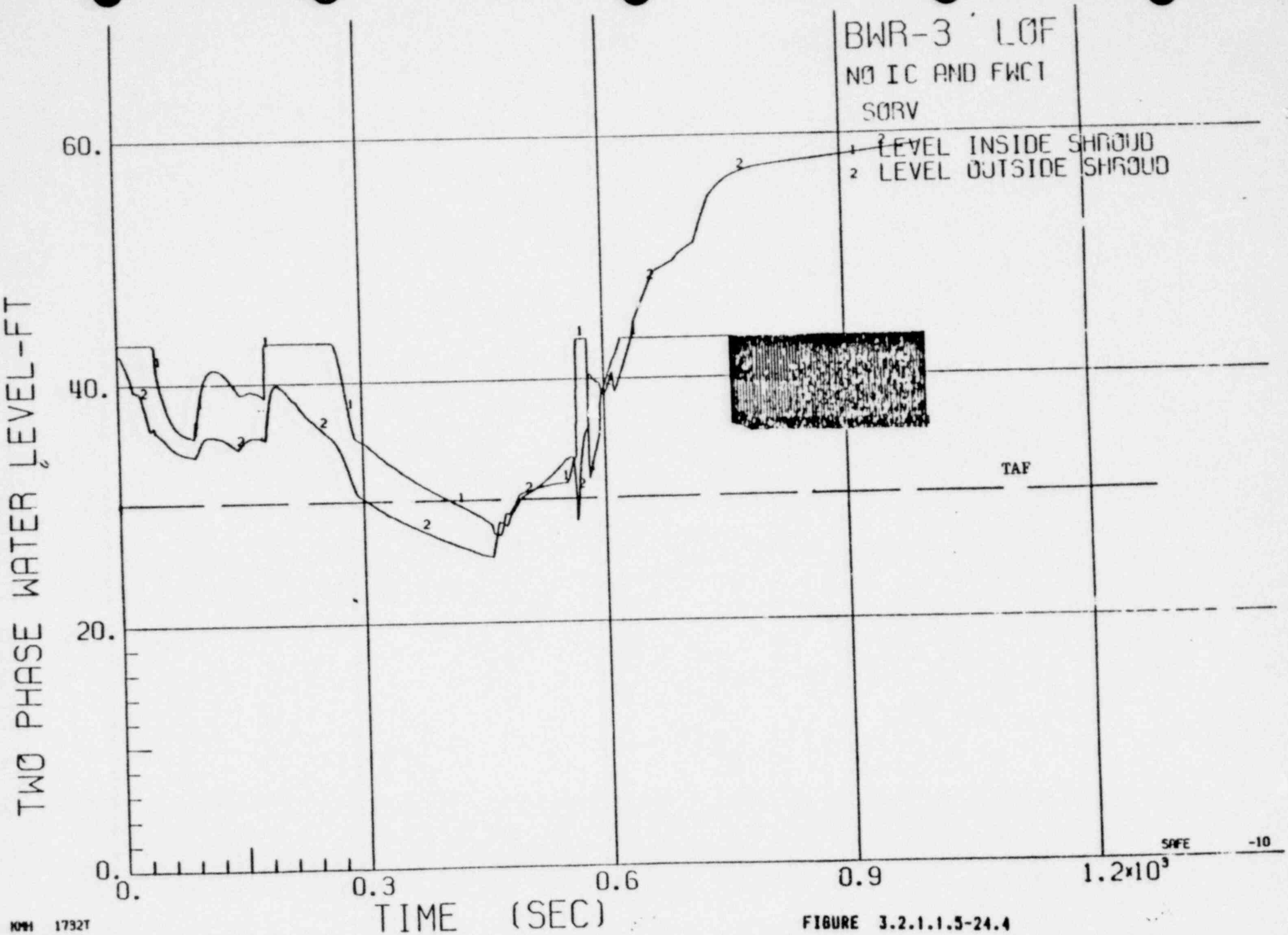
FIGURE 3.2.1.1.5-24.2

KMH 1732T
121179 2031.5



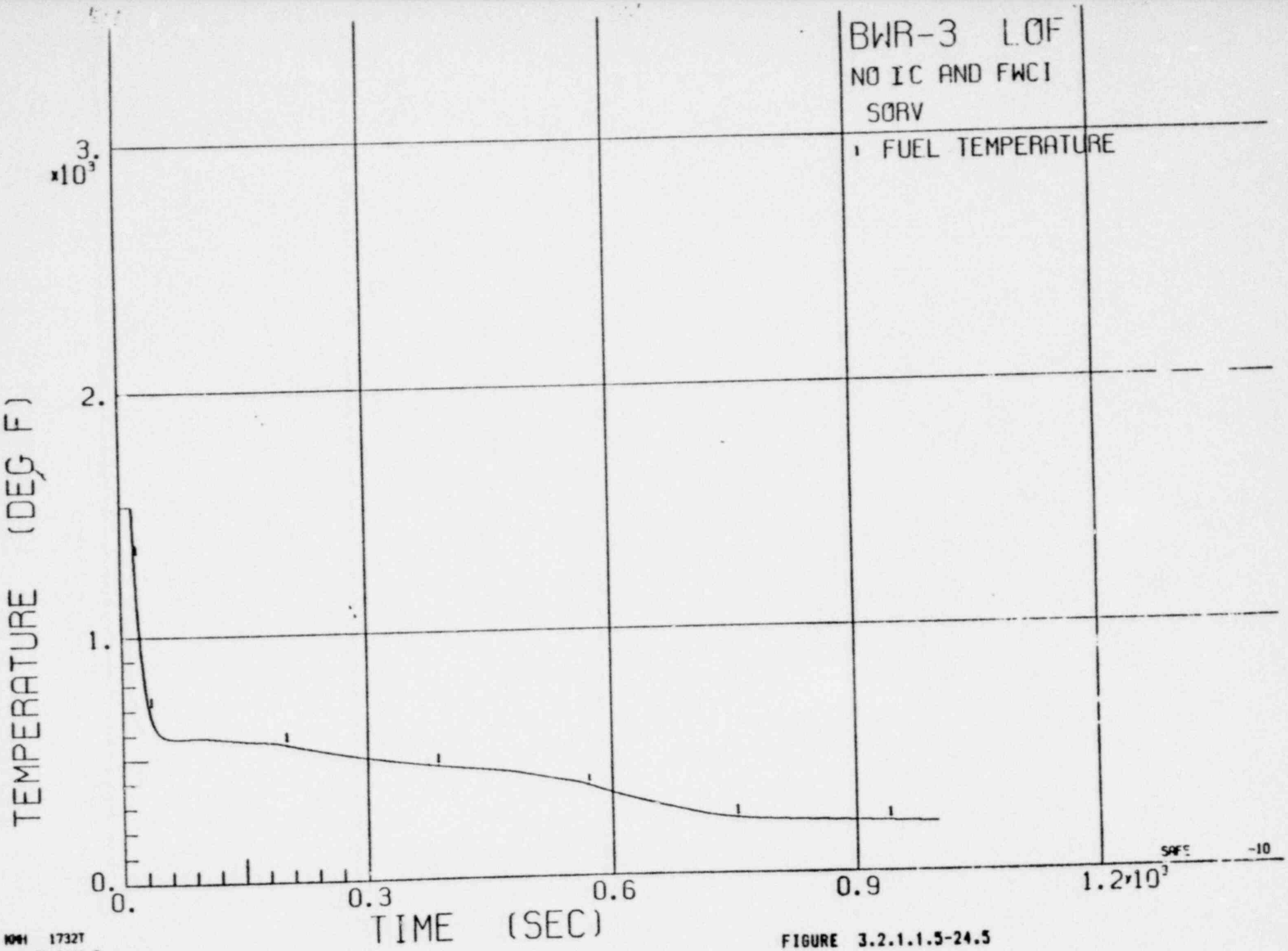
KMH 1732T
121179 2031.5

FIGURE 3.2.1.1.5-24.3



KMH 1732T
121179 2031.5

FIGURE 3.2.1.1.5-24.4



KMH 1732T
121179 2031.5

FIGURE 3.2.1.1.5-24.5

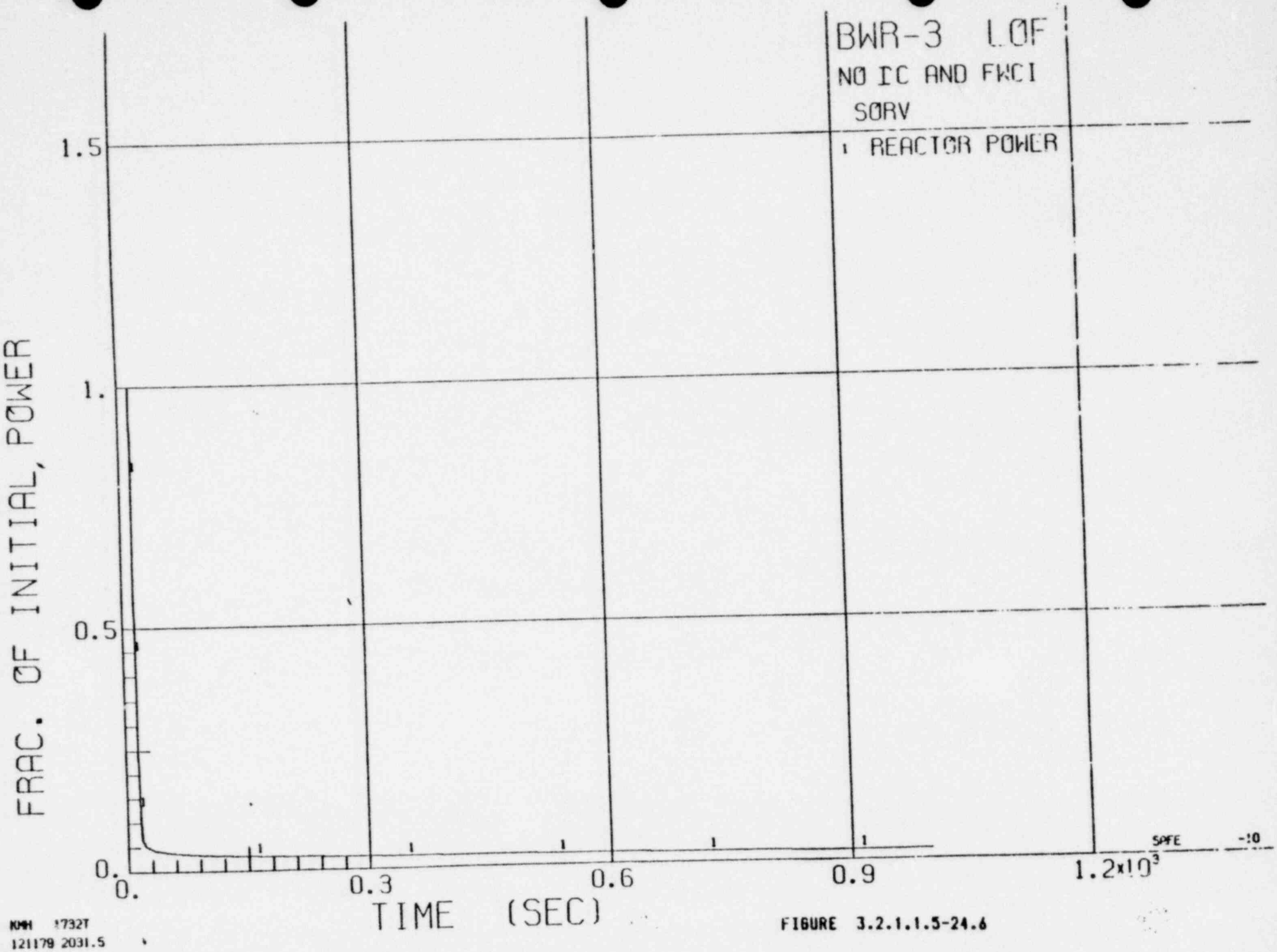
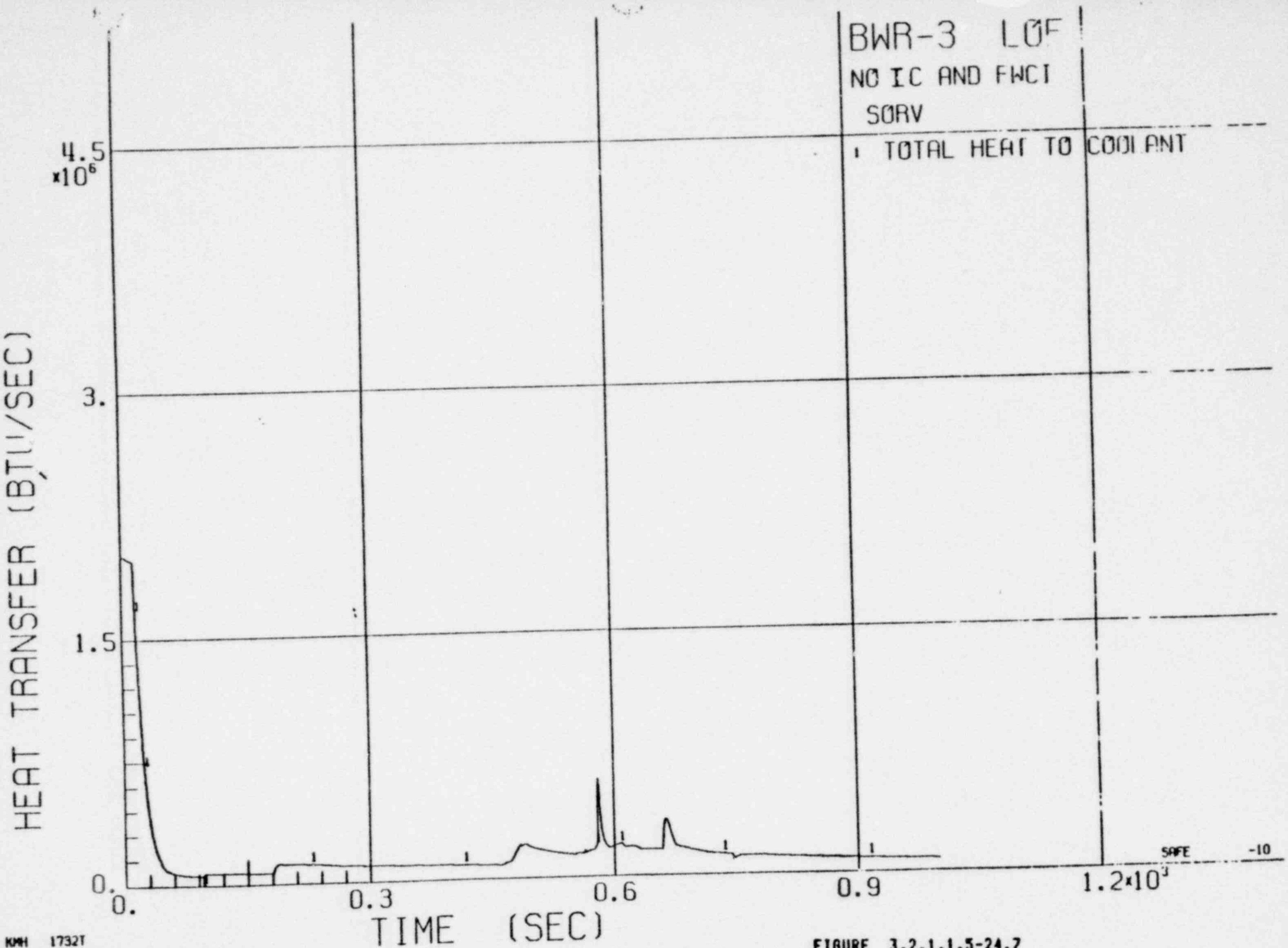
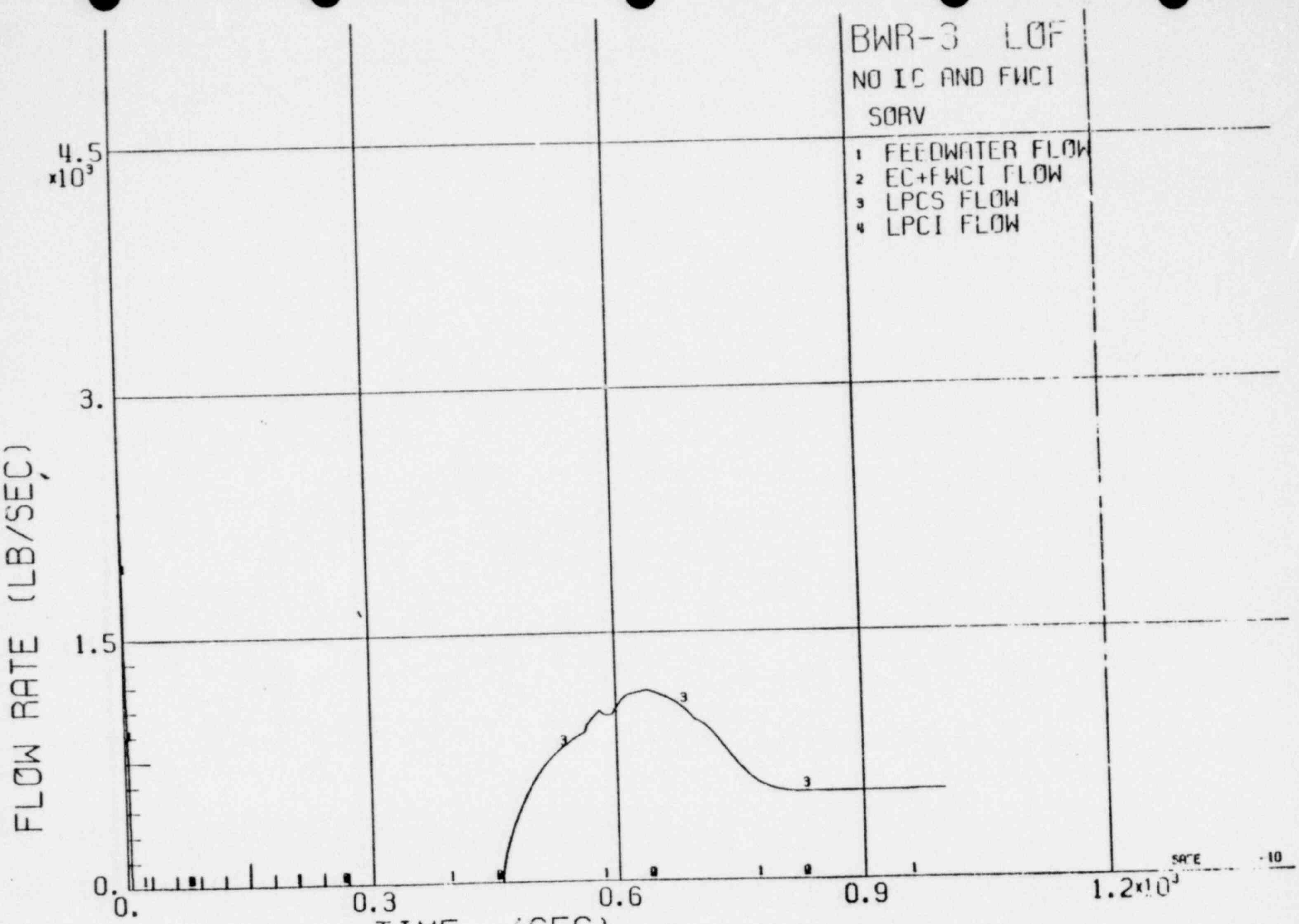


FIGURE 3.2.1.1.5-24.6



KMH 1732T
121179 2107.3

FIGURE 3.2.1.1.5-24.7



KMH 1732T
121179 2107.3

FIGURE 3.2.1.1.5-24.8

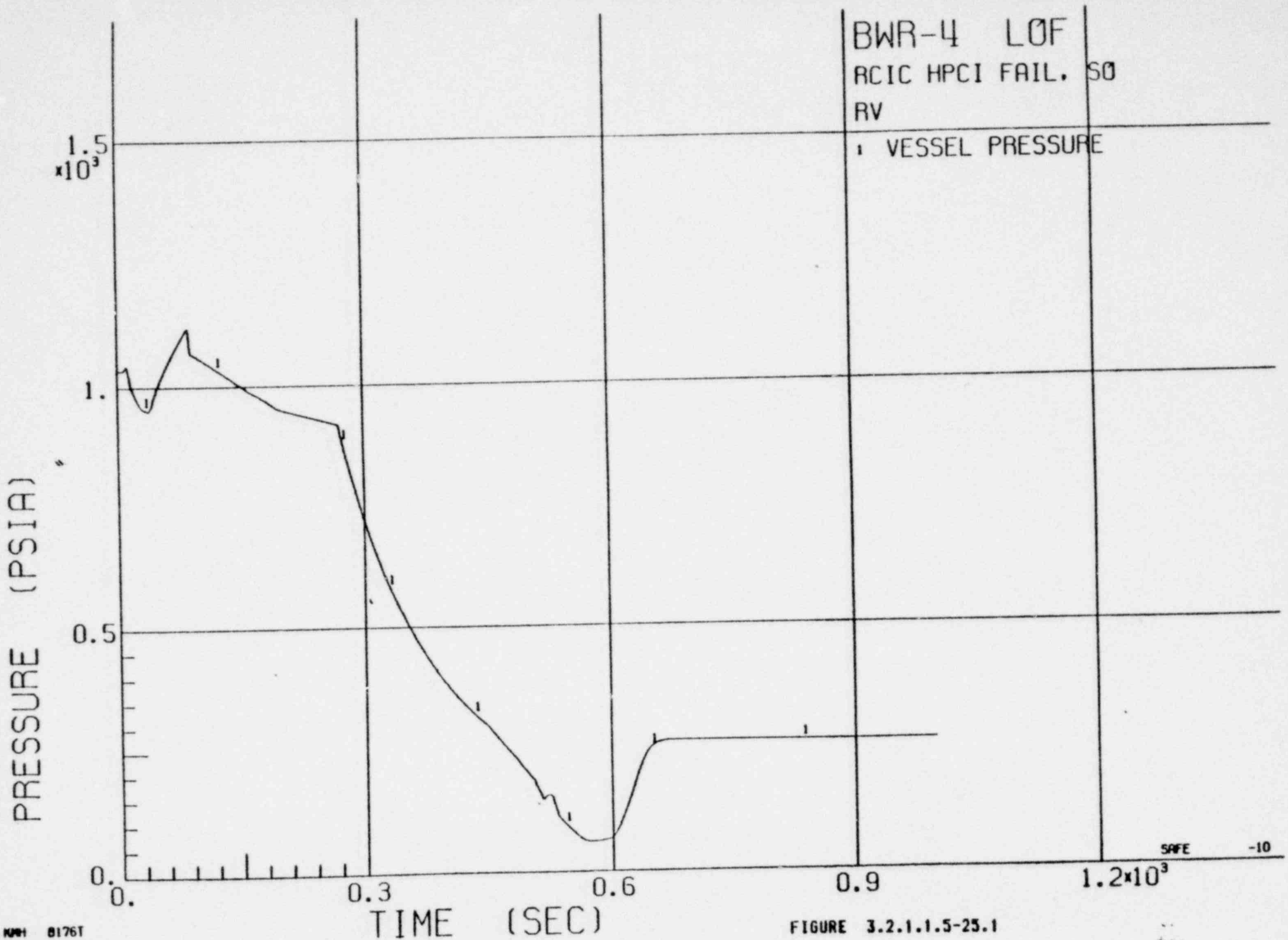


FIGURE 3.2.1.1.5-25.1

KMH 8176T
010280 2003.7

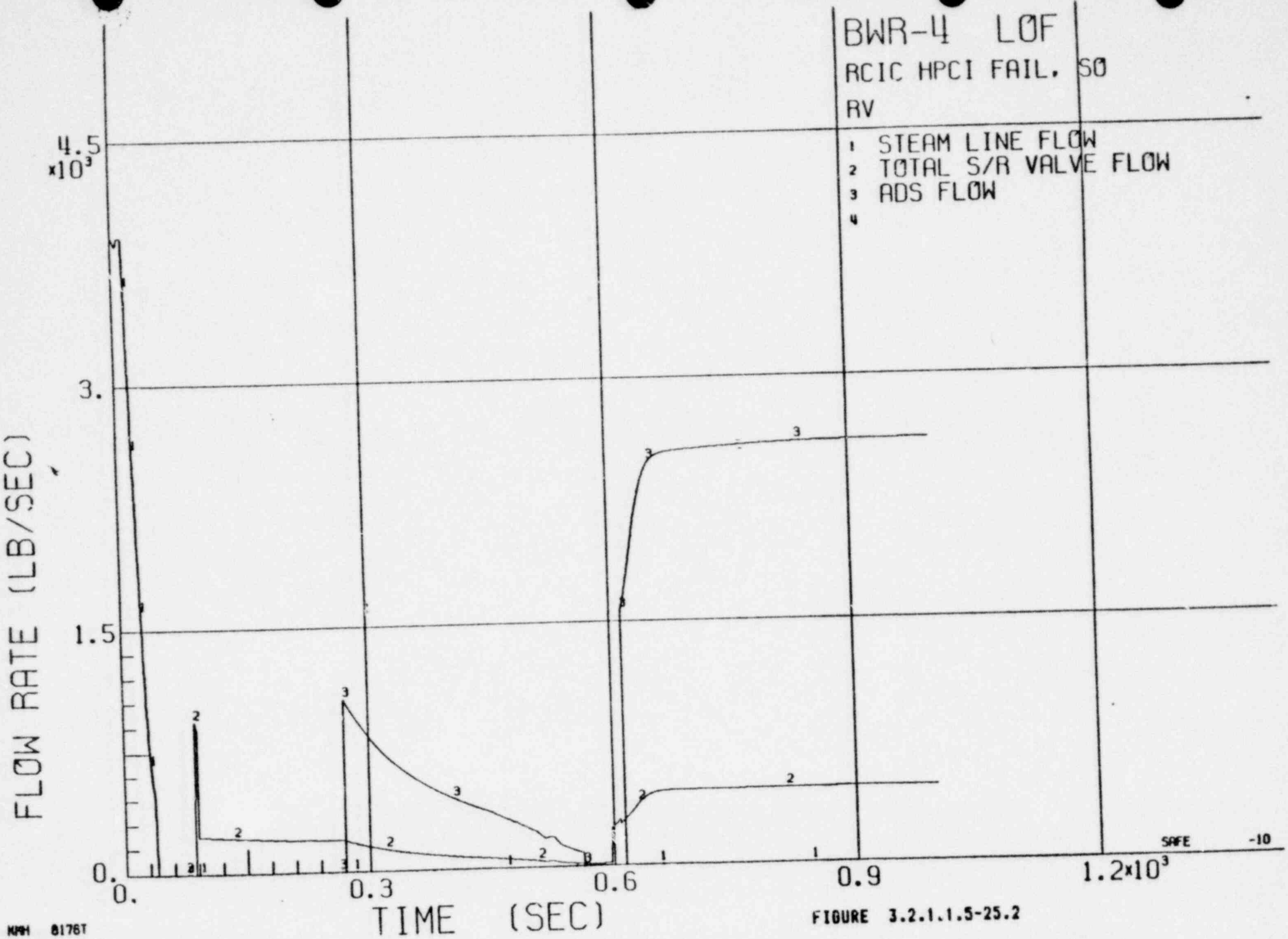
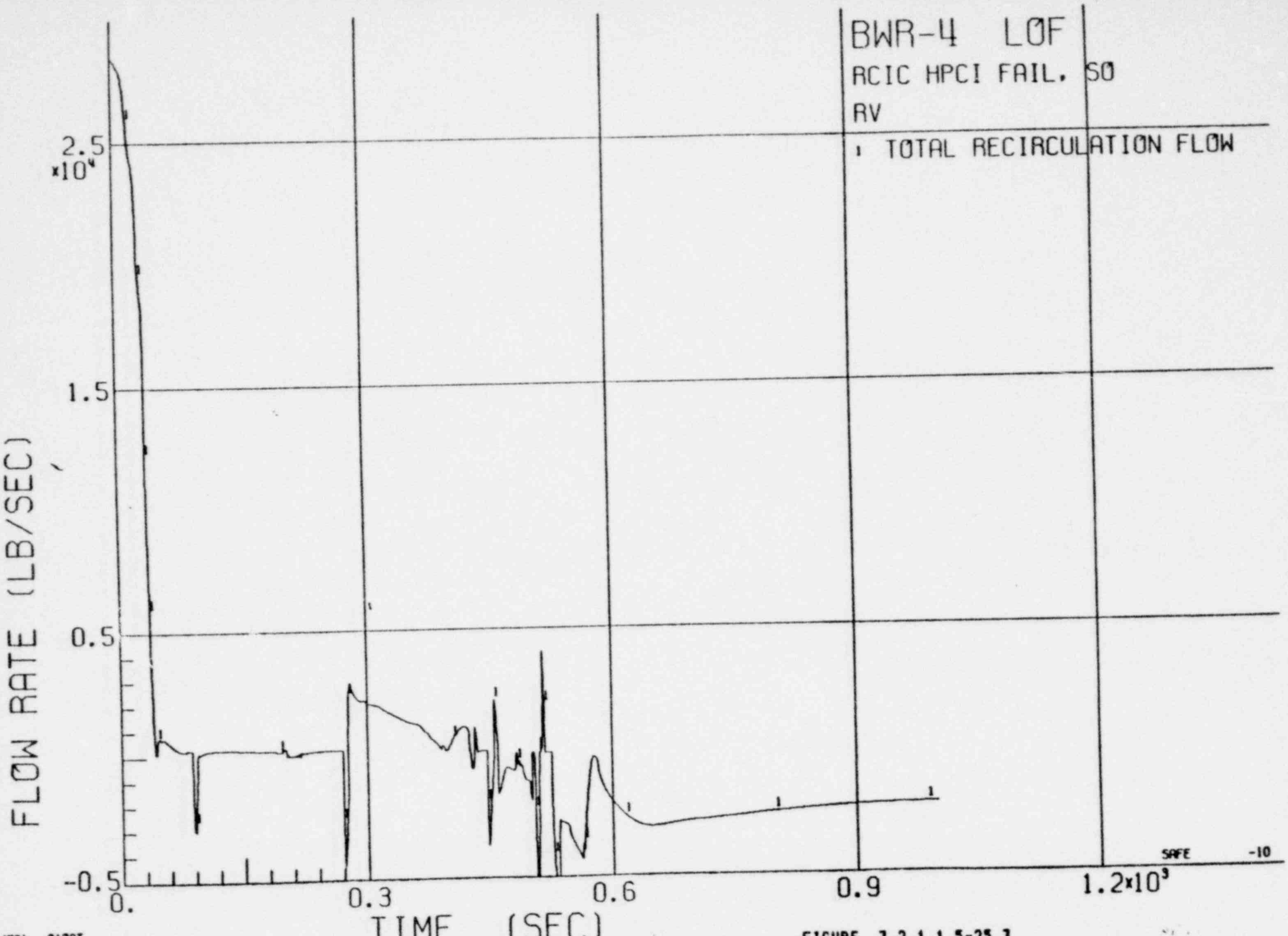


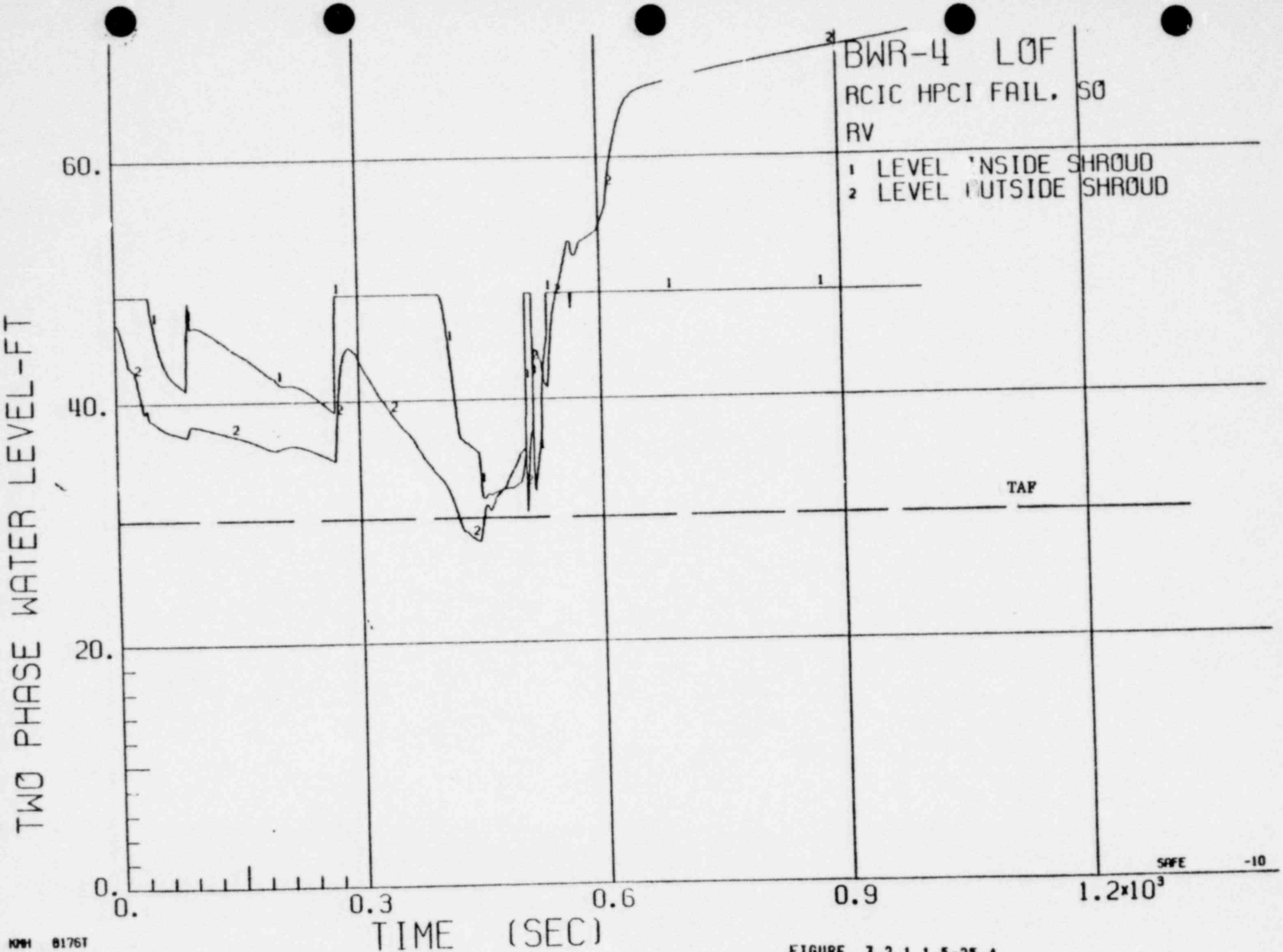
FIGURE 3.2.1.1.5-25.2

KMH 8176T
010280 2003.7



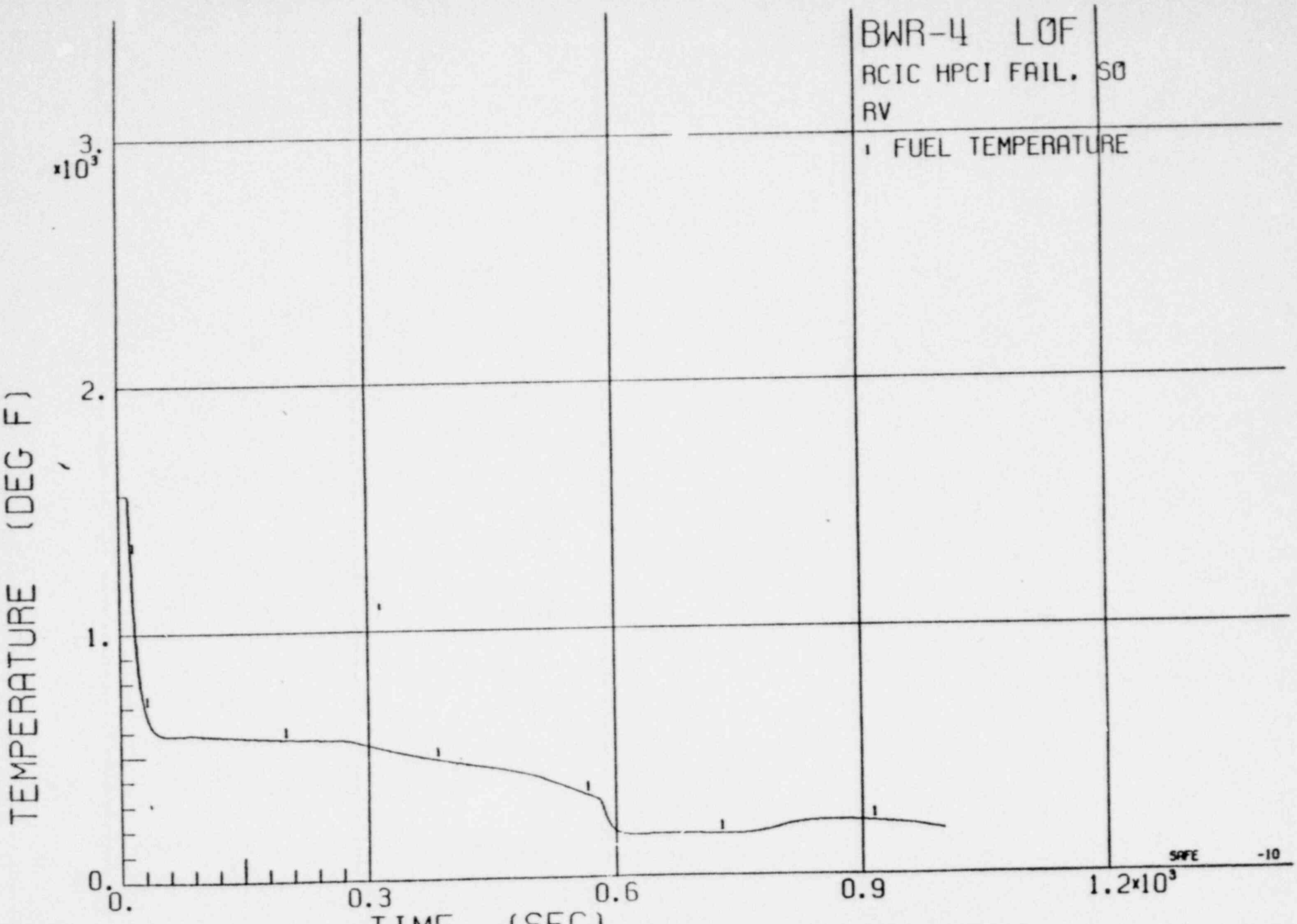
KMH 8178T
010280 2003.7

FIGURE 3.2.1.1.5-25.3



KMH 8176T
010280 2003.7

FIGURE 3.2.1.1.5-25.4



KMH 8176T
010280 2003.7

FIGURE 3.2.1.1.5-25.5

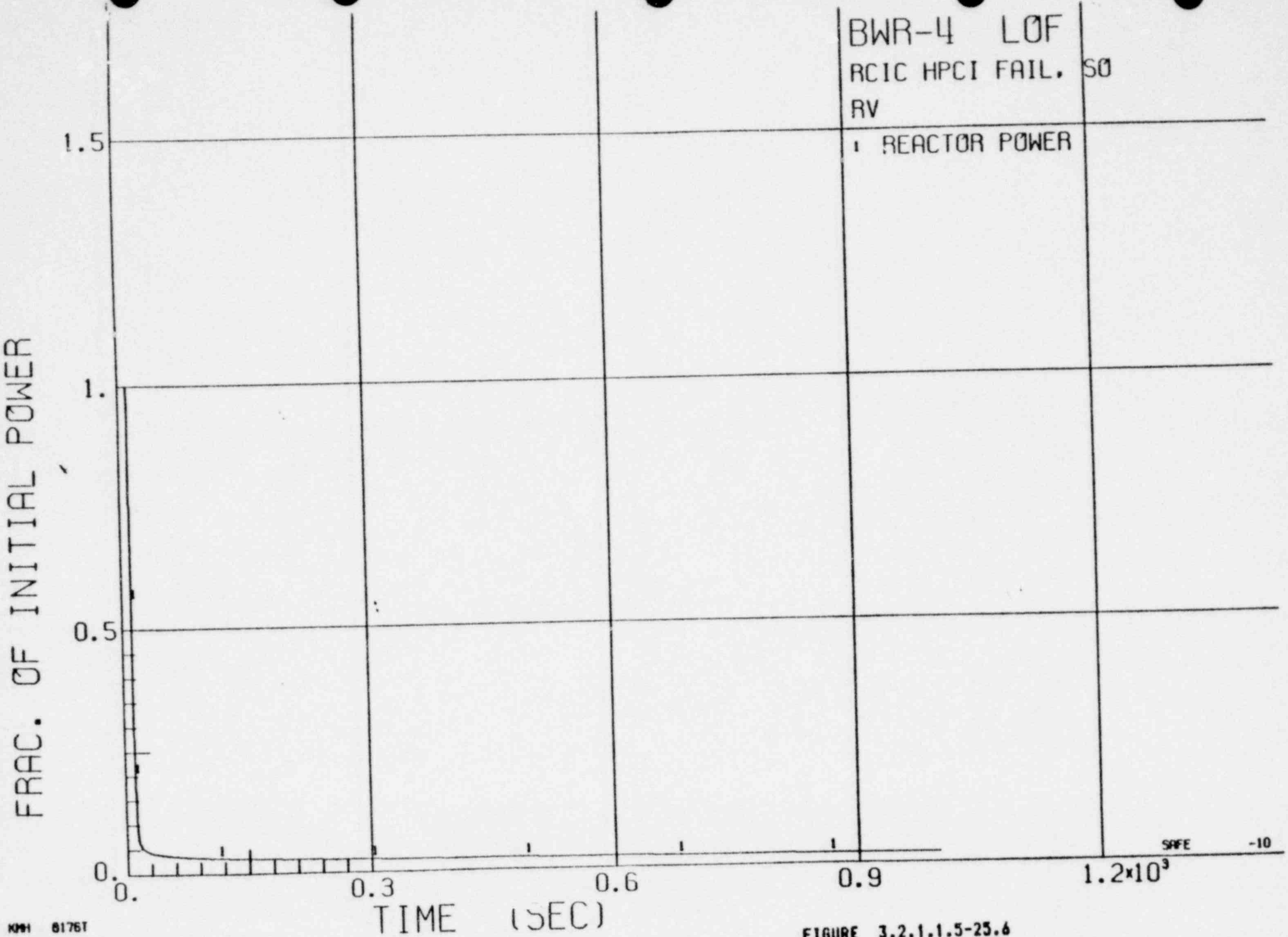


FIGURE 3.2.1.1.5-25.6

KMH 8176T
010280 2003.7

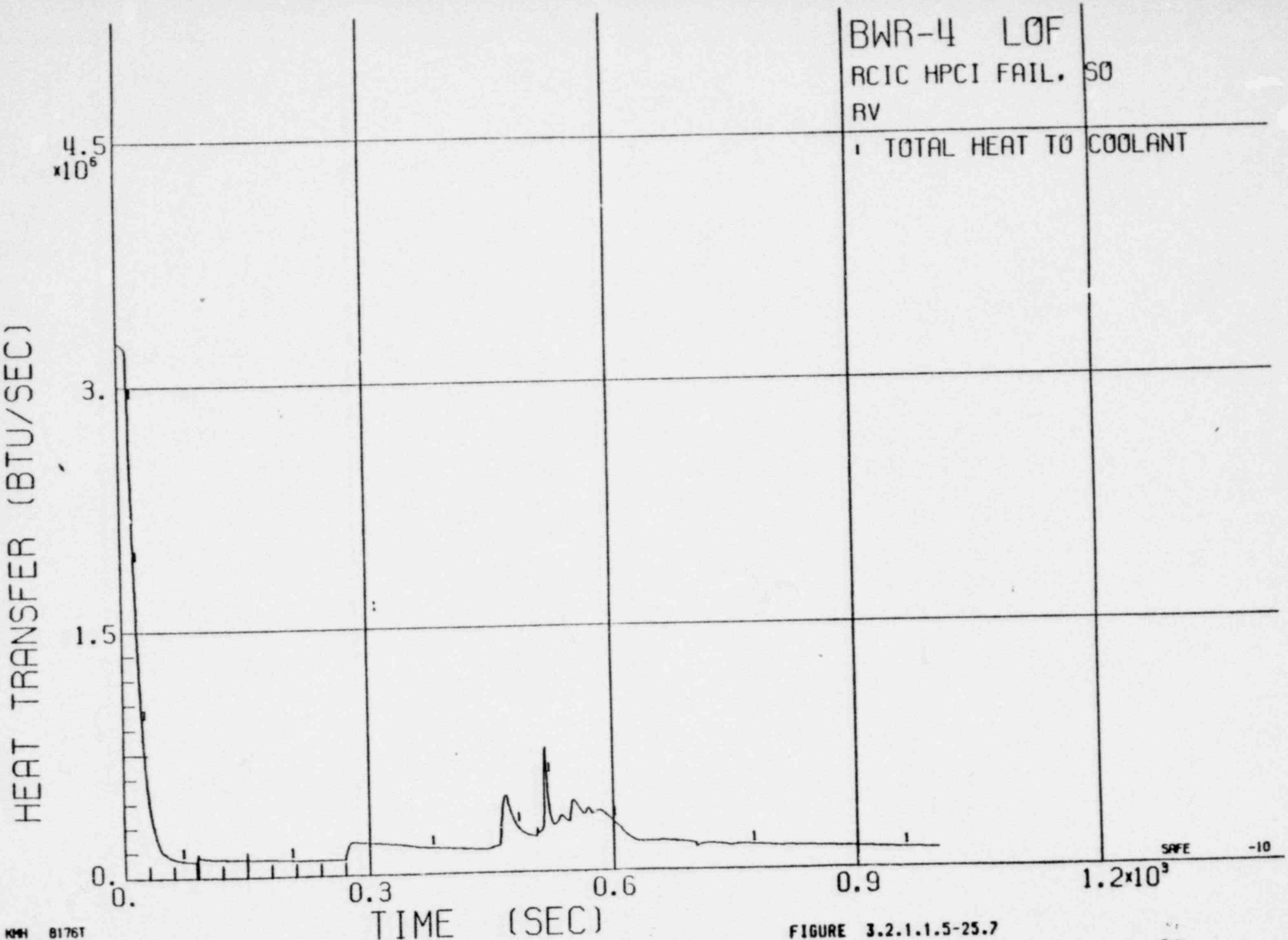
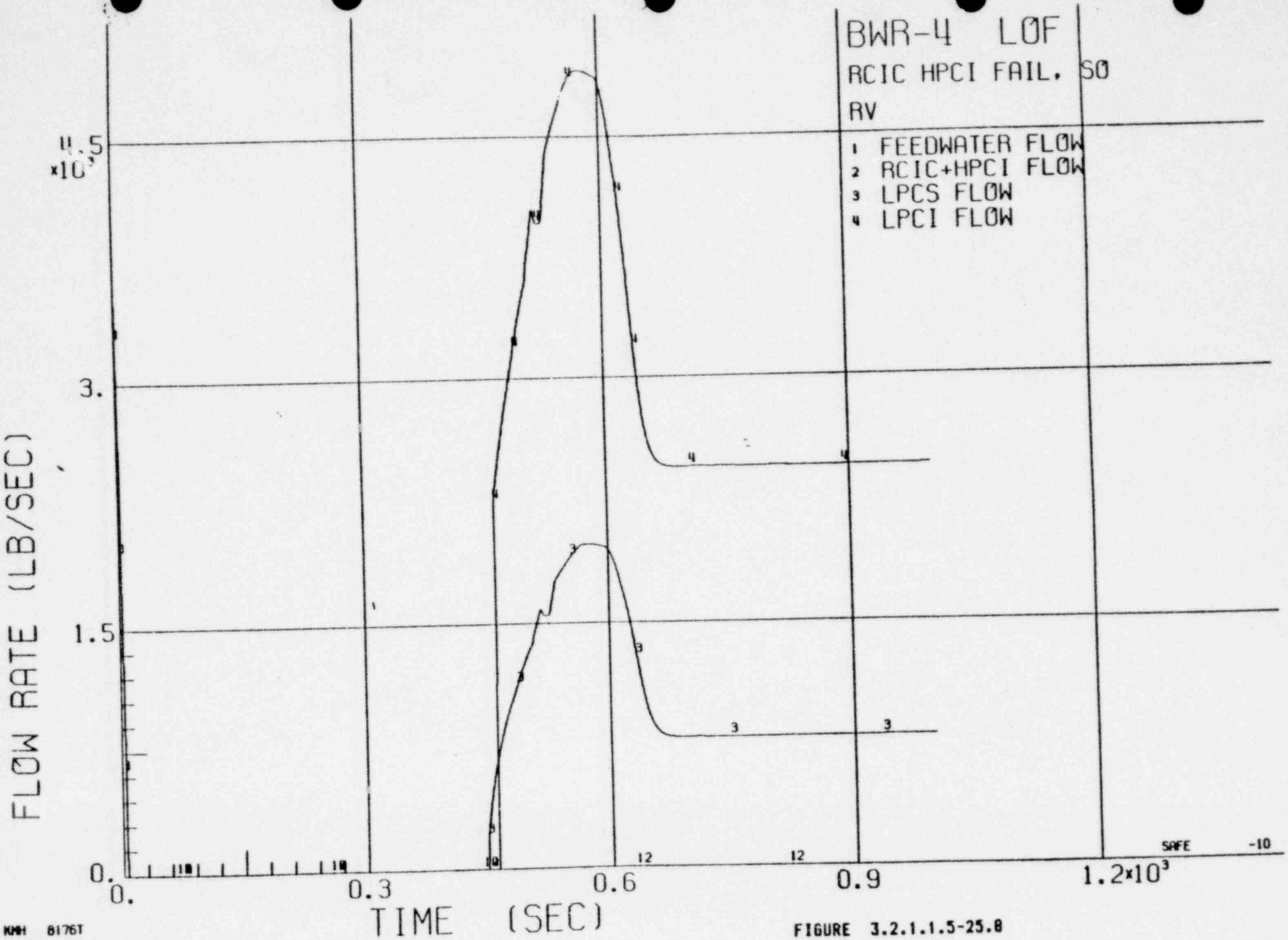


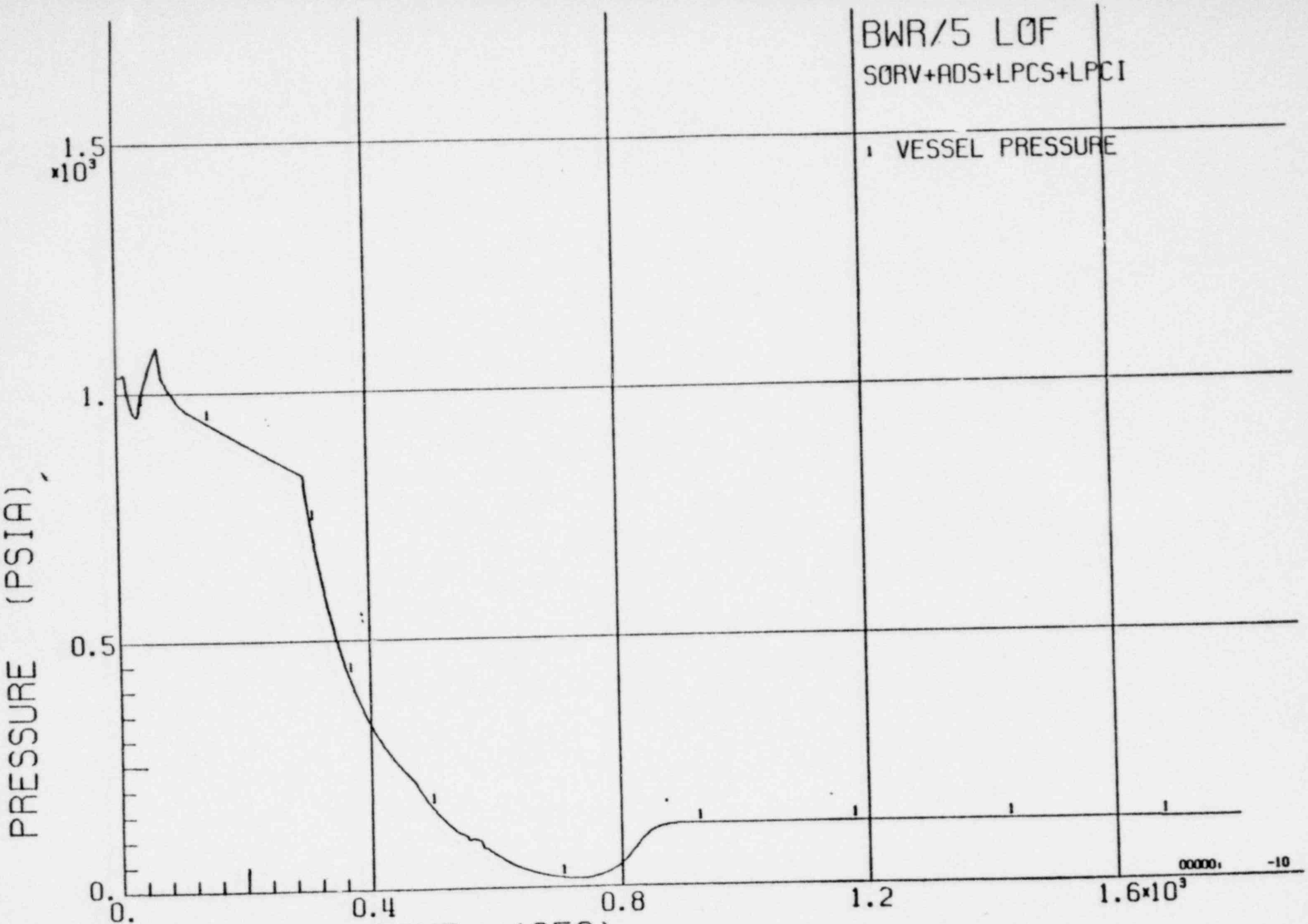
FIGURE 3.2.1.1.5-25.7

KMH 8176T
010280 2033.3



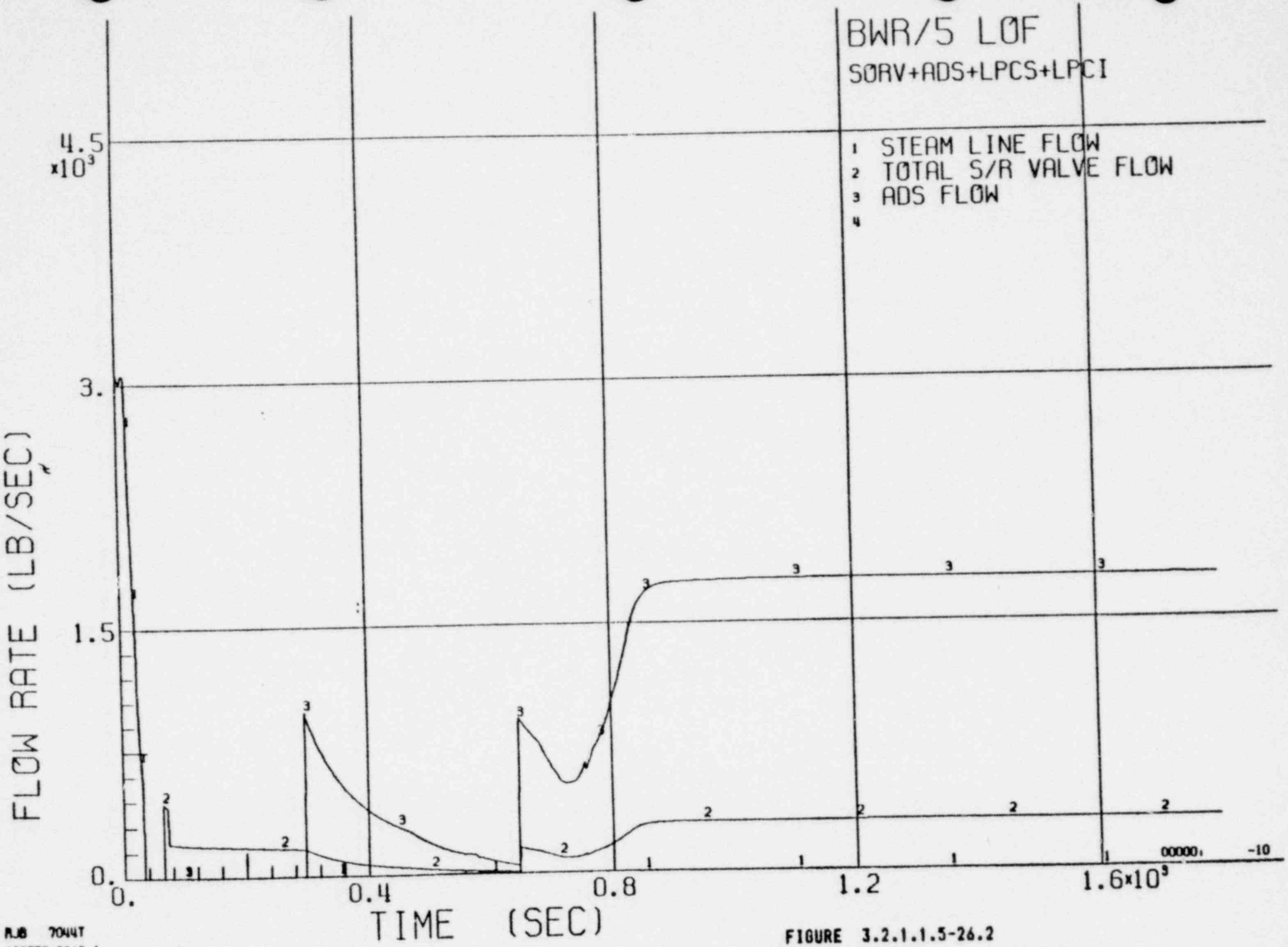
KMH 8176T
010280 2033.3

FIGURE 3.2.1.1.5-25.8



RJB 7044T
120779 0319.1

FIGURE 3.2.1.1.5-26.1



RJB 7044T
120779 0319.1

FIGURE 3.2.1.1.5-26.2

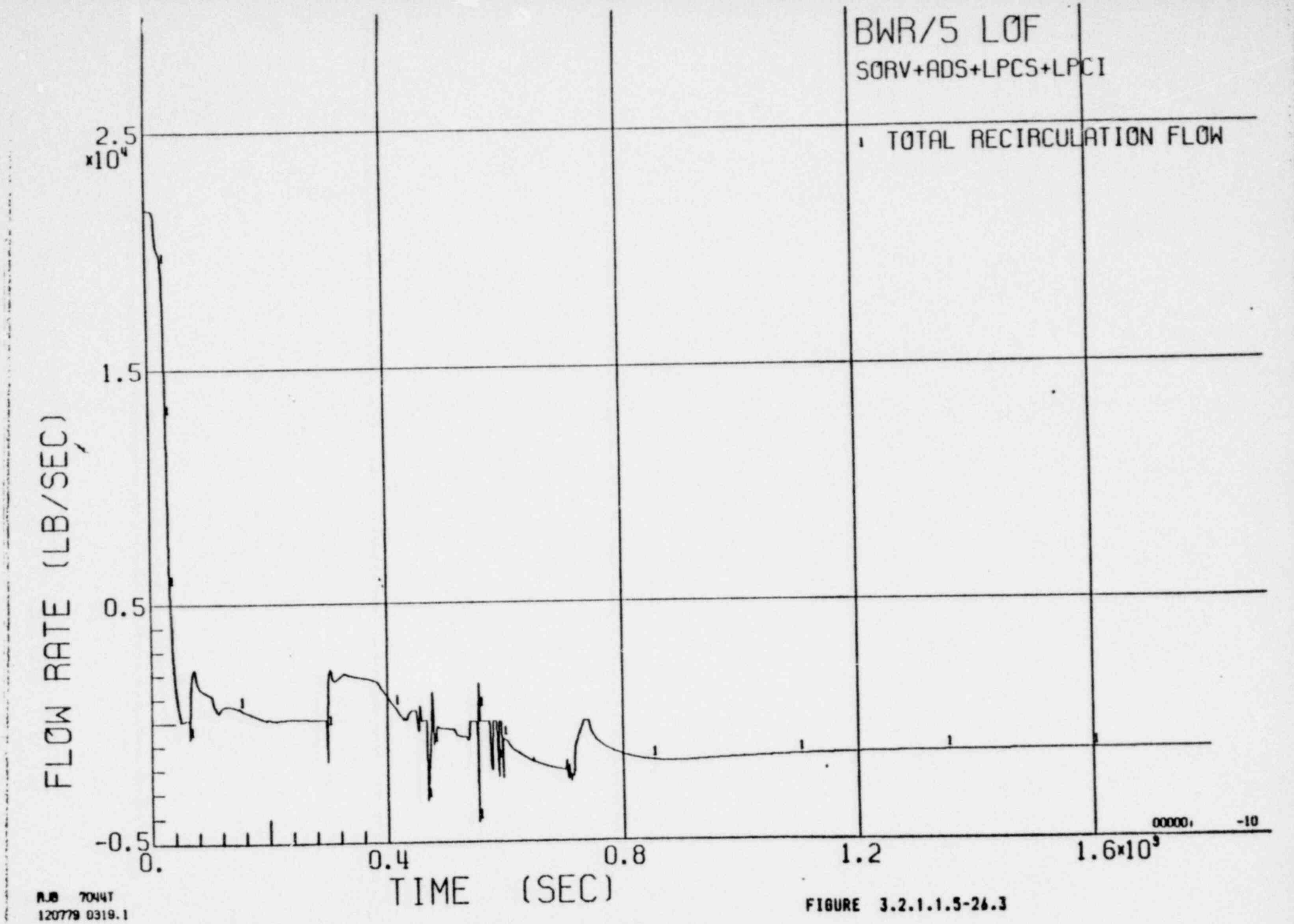
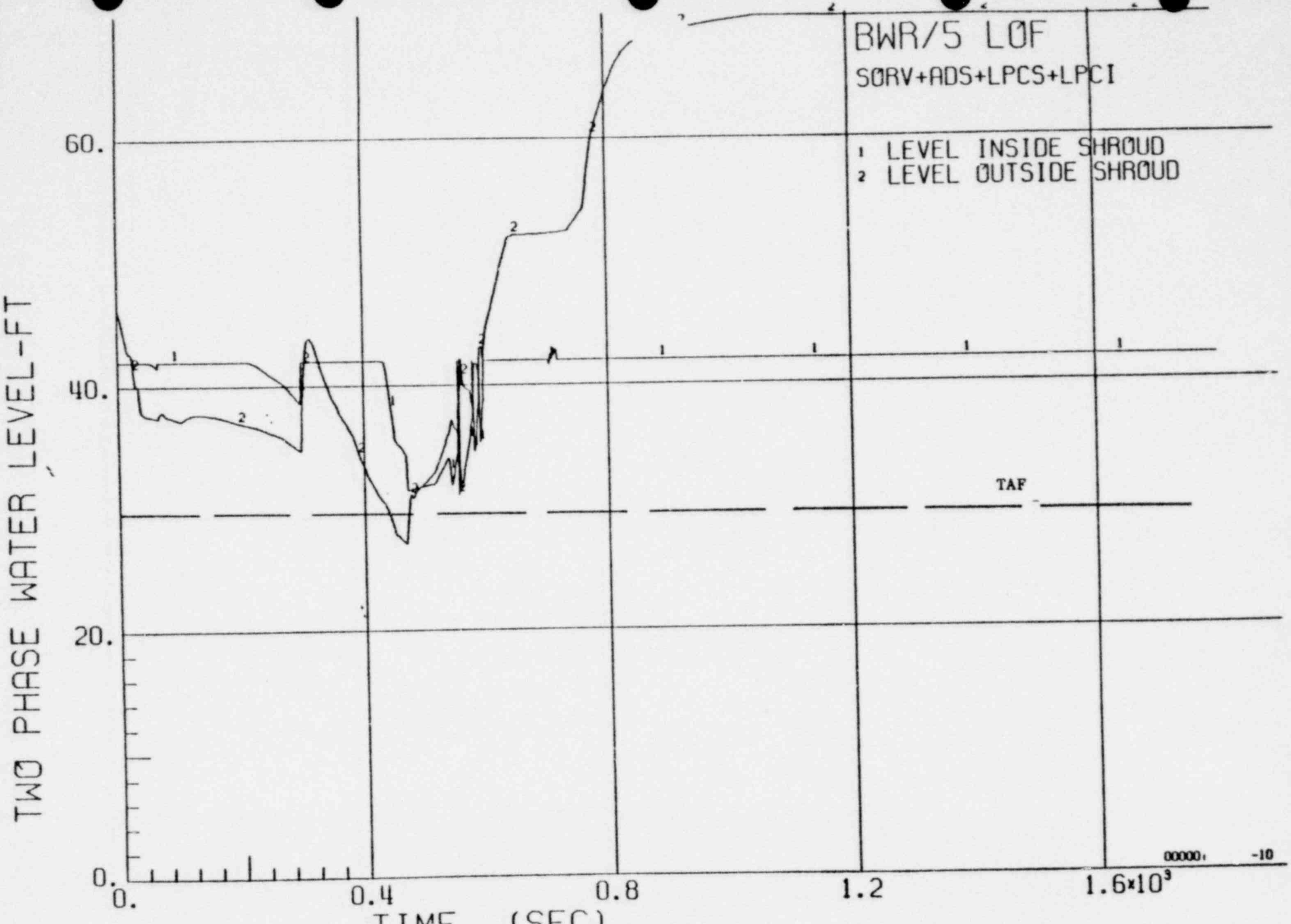
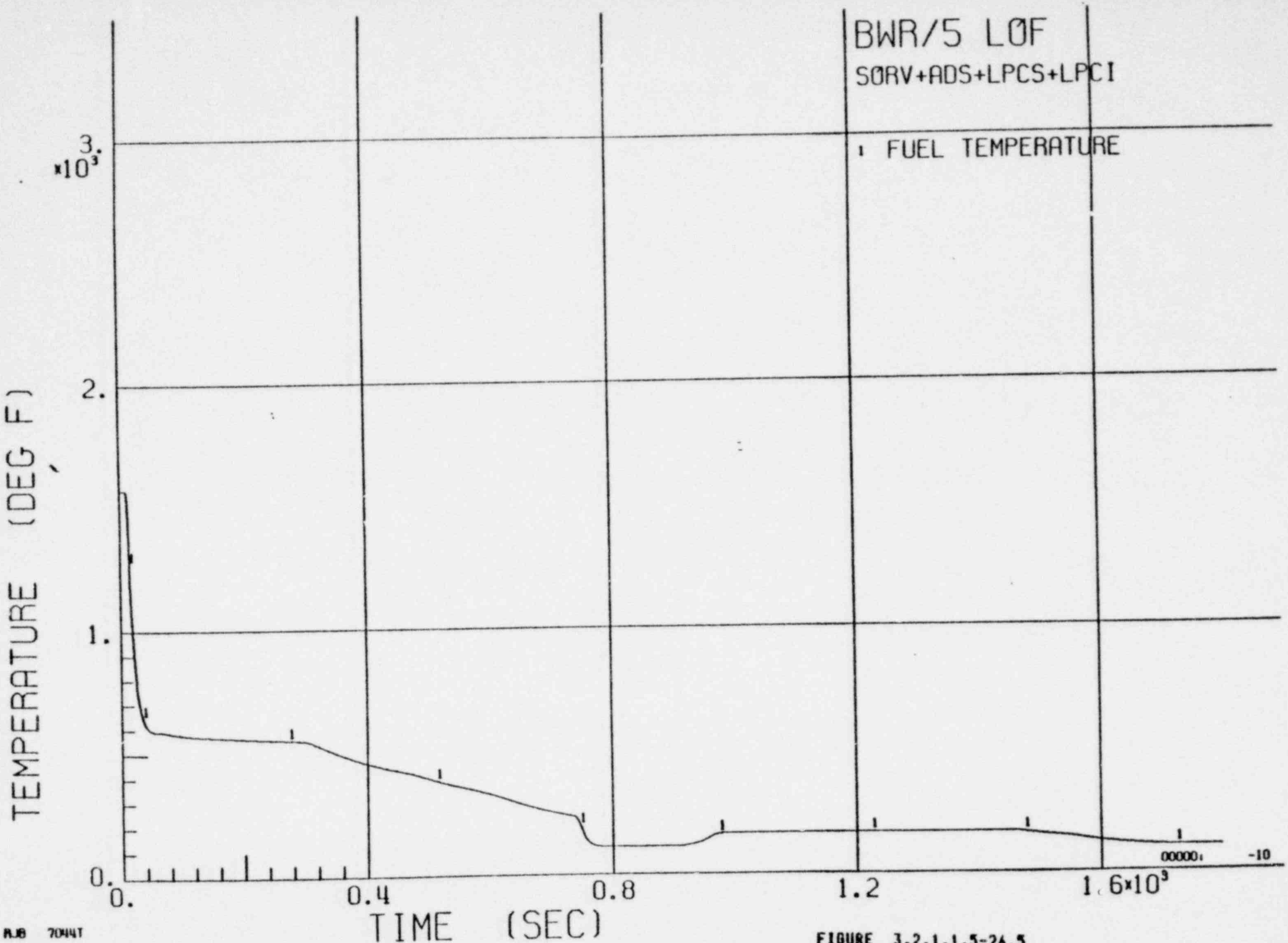


FIGURE 3.2.1.1.5-26.3



RJB 7044T
120779 0319.1

FIGURE 3.2.1.1.5-26.4



RJB 7044T
120779 0319.1

FIGURE 3.2.1.1.5-26.5

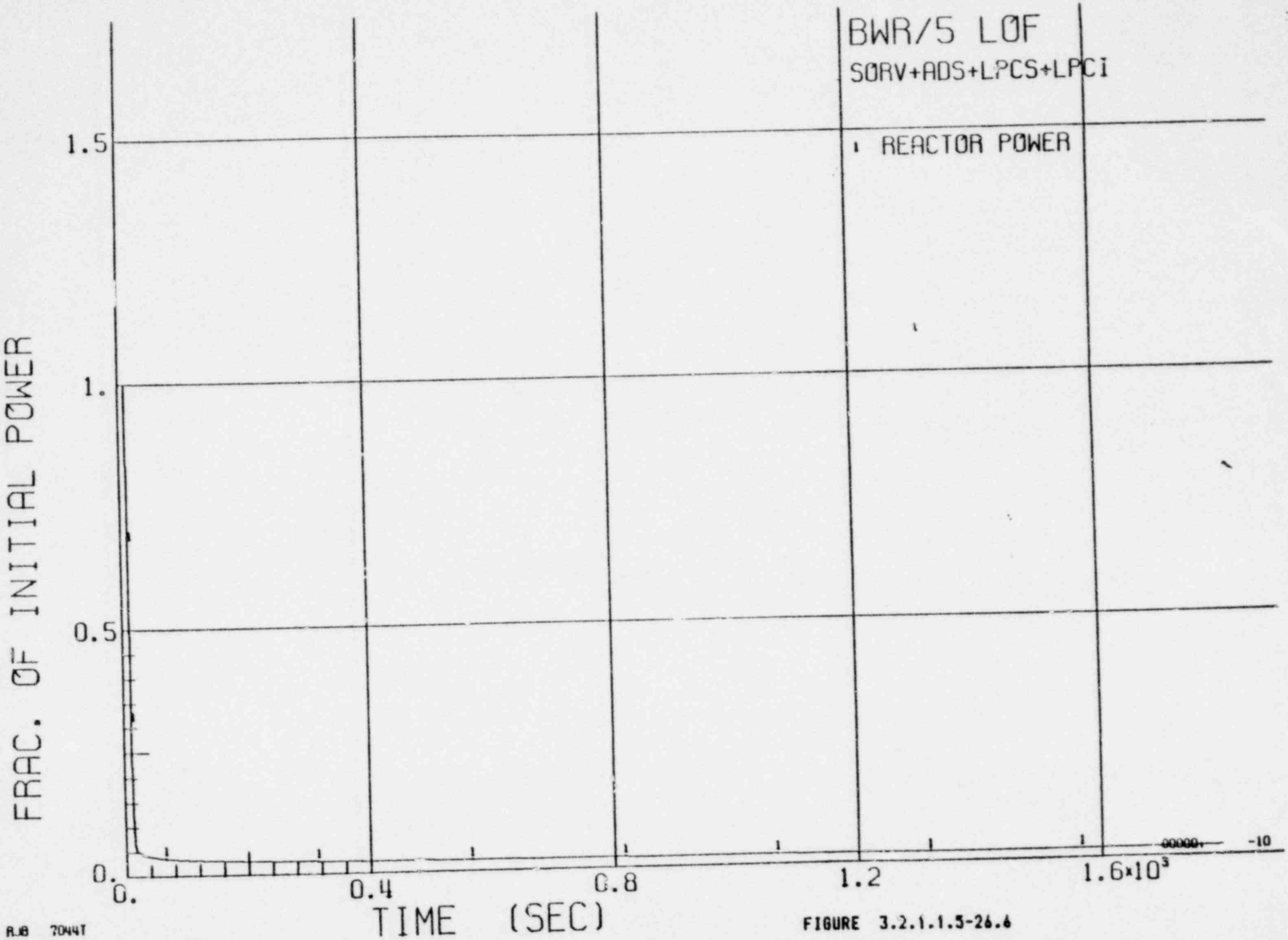
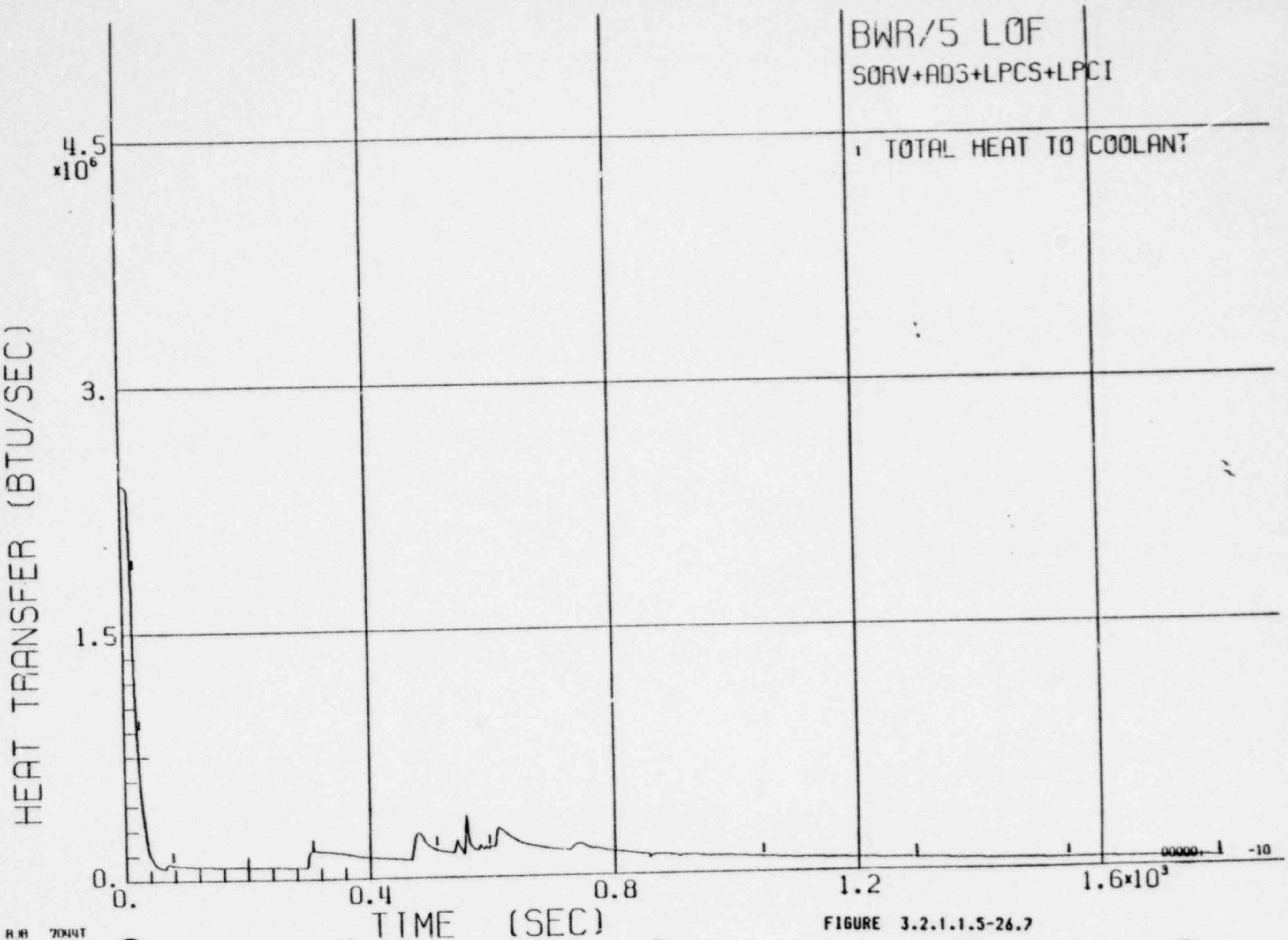


FIGURE 3.2.1.1.5-26.6

RJB 7044T



RJB 7044T

FIGURE 3.2.1.1.5-26.7

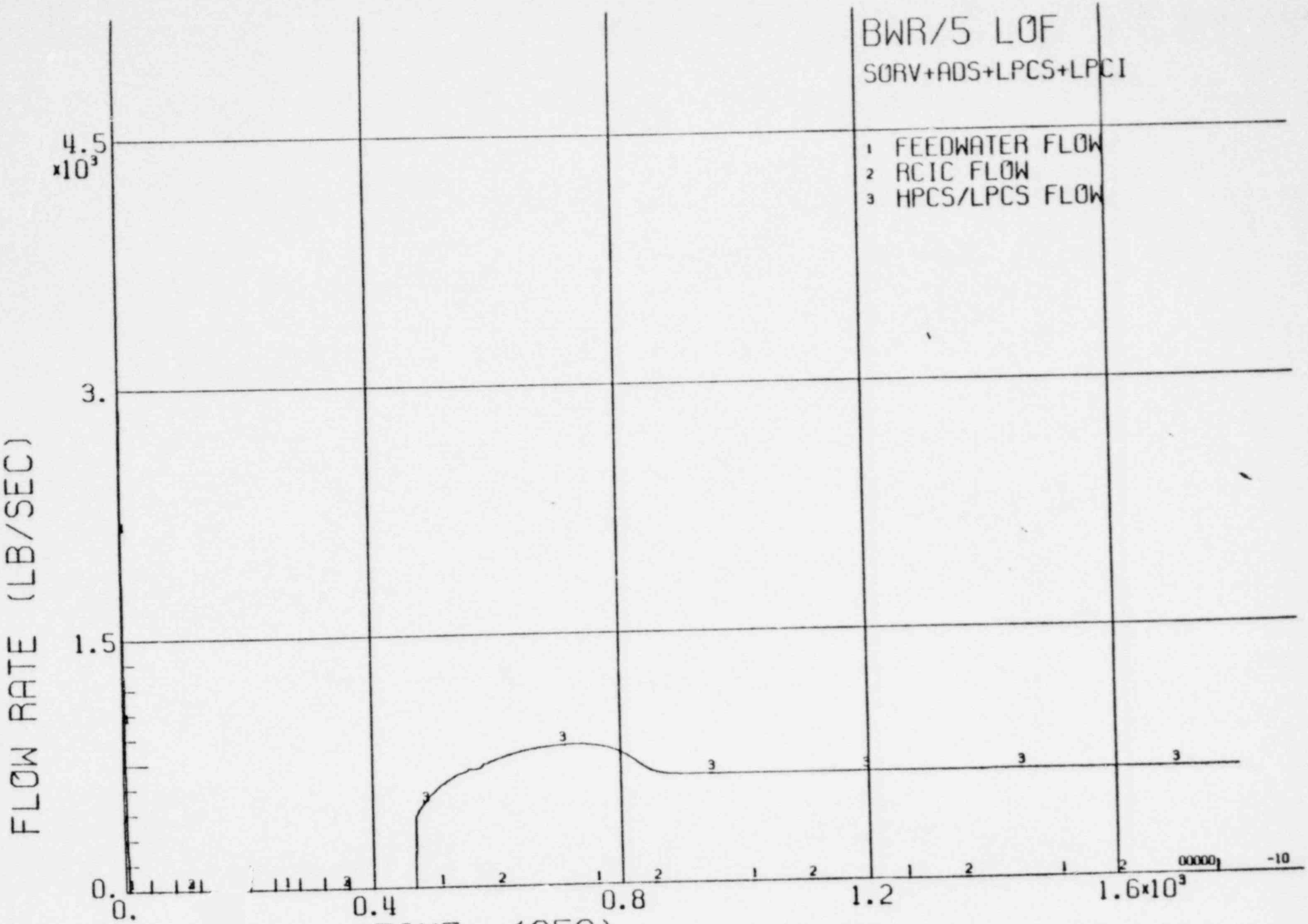


FIGURE 3.2.1.1.5-26.8

RJB 7044T

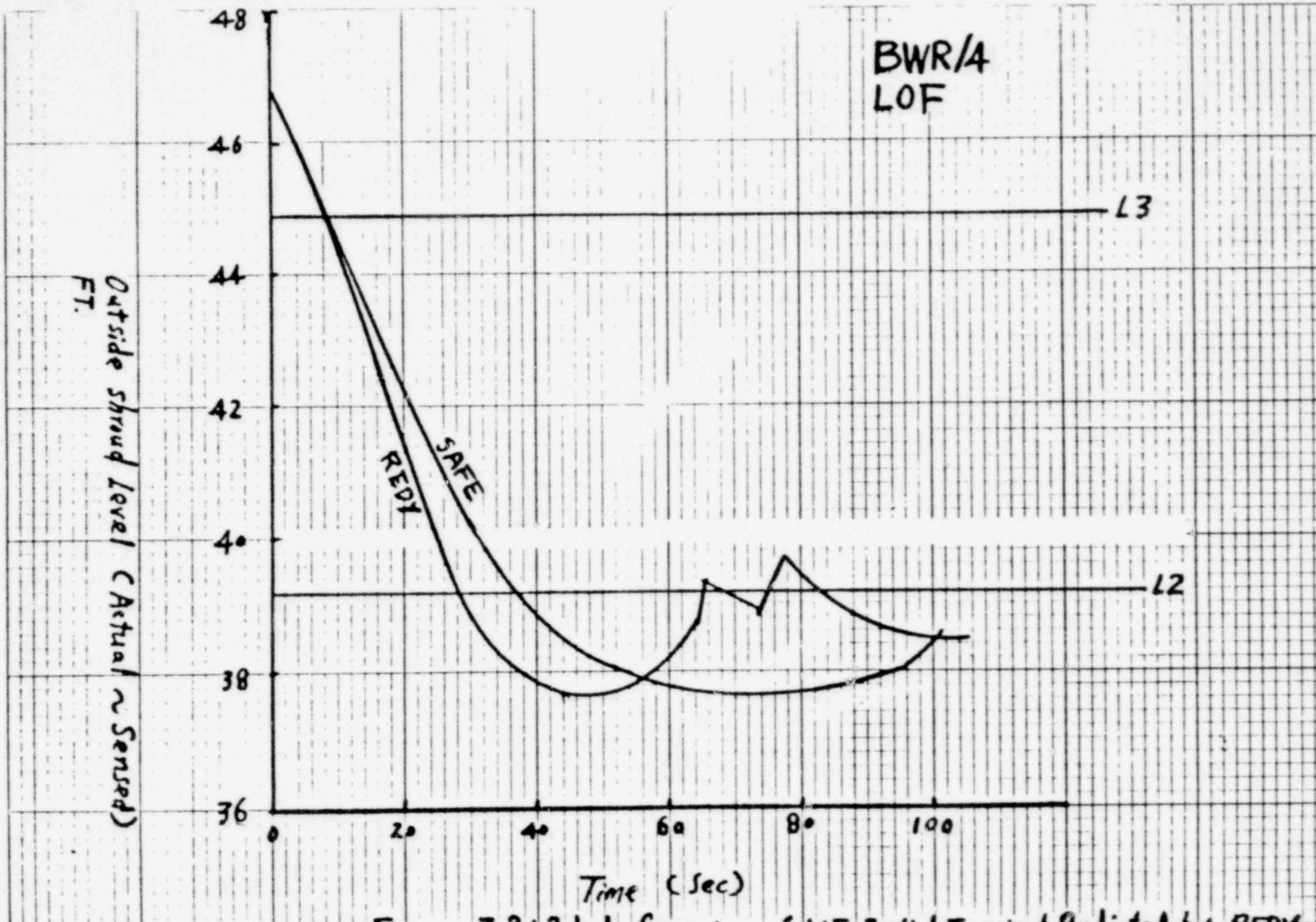


Figure 3.2.13.1-1 Comparison of LOF Initial Transient Predicted by REDY

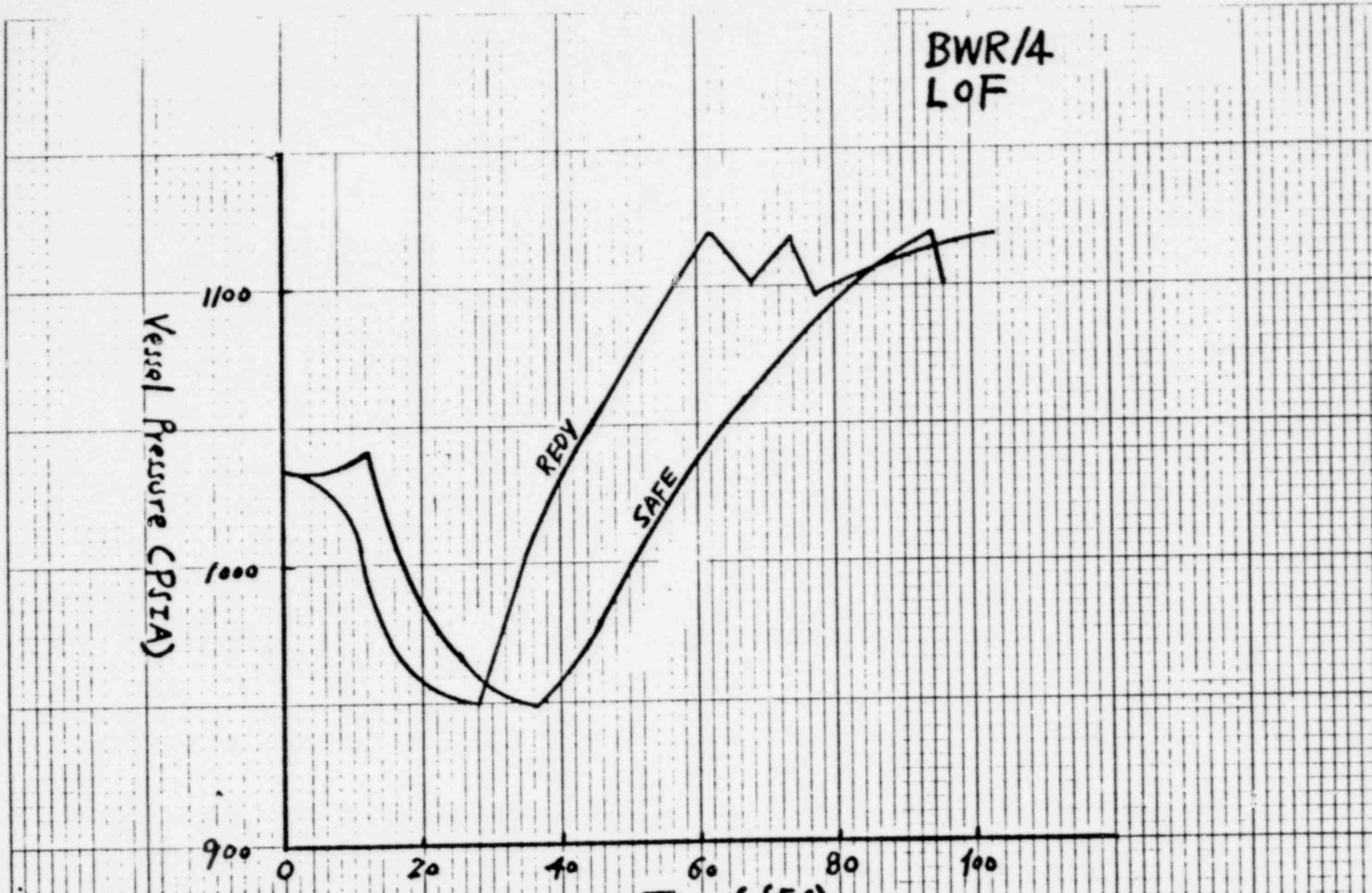


Figure 3.2.1.3.1-2 Comparison of LoF Initial Transient Predicted by REDY And SAFE, Level Comparison

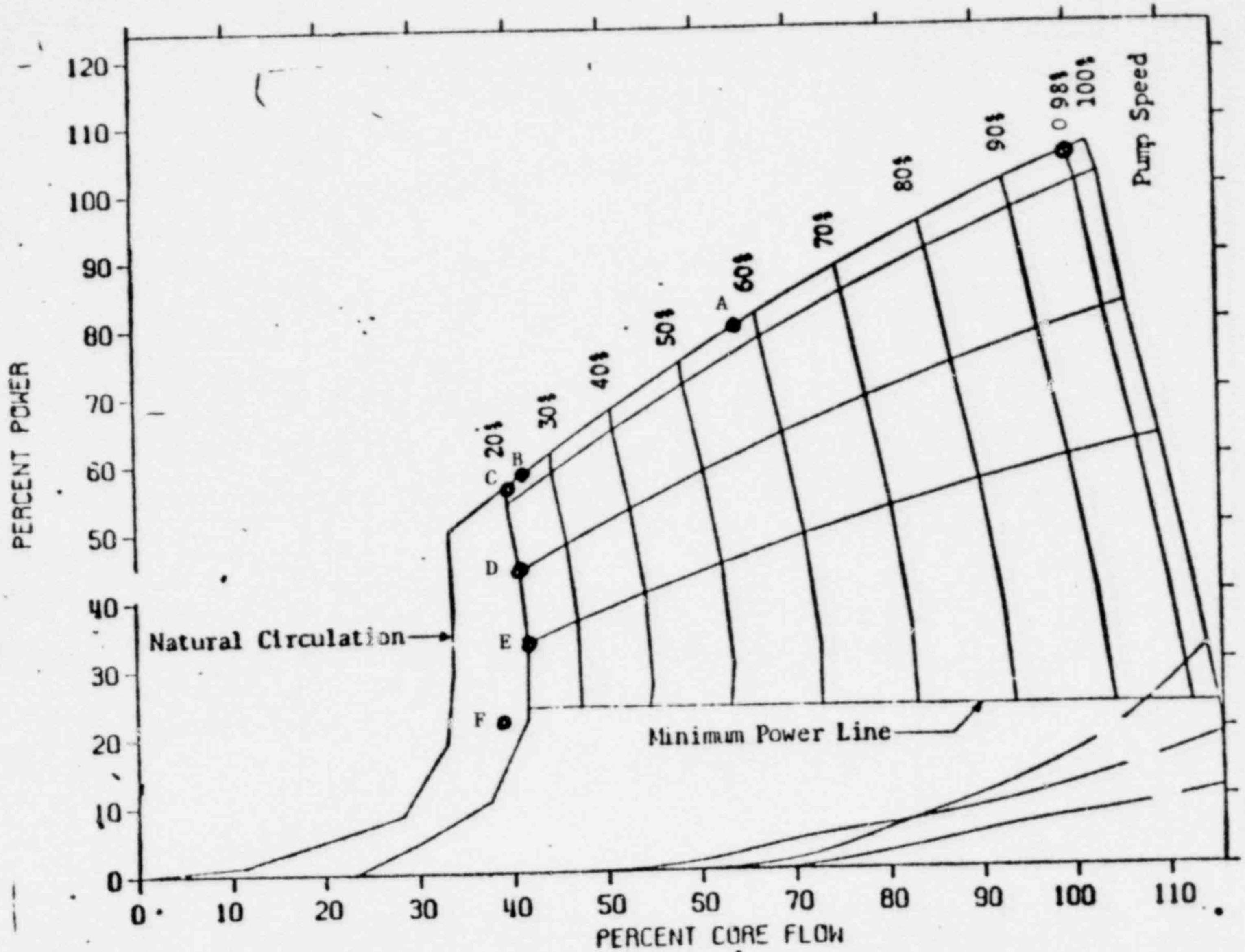


FIGURE 3.2.1.3.2-1 OPERATING MAP

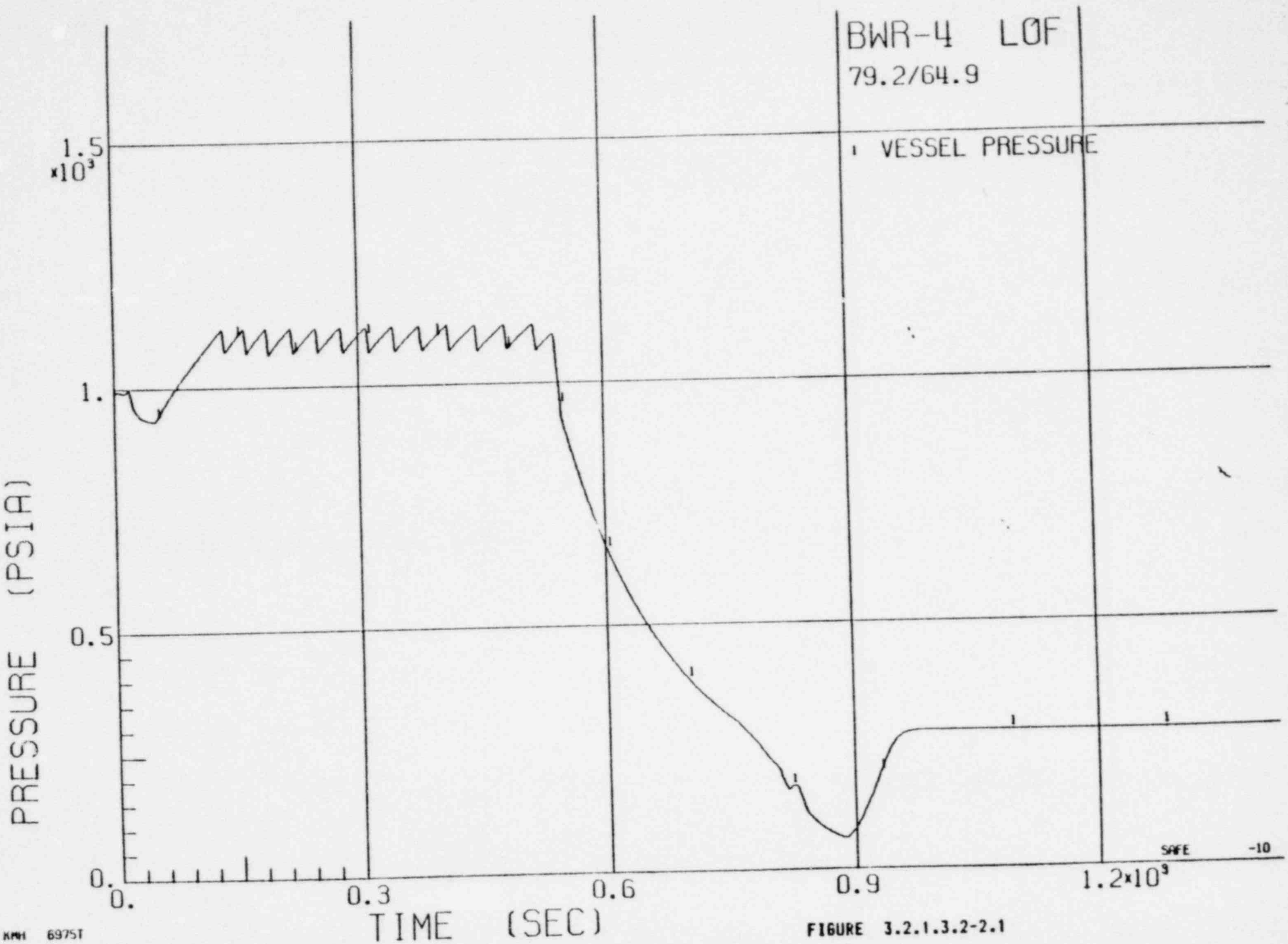
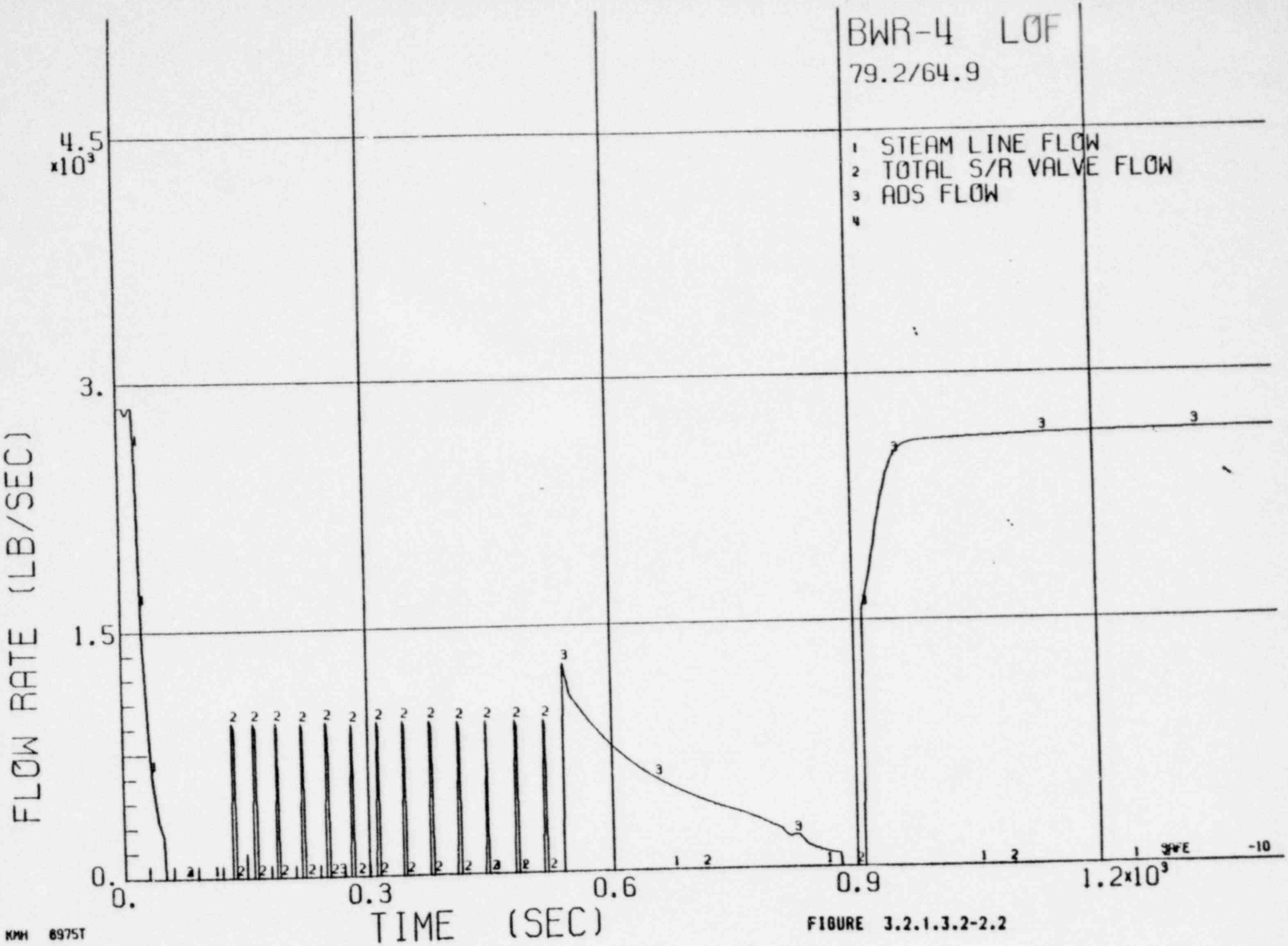


FIGURE 3.2.1.3.2-2.1

KMH 6975T



KMH 6975T

FIGURE 3.2.1.3.2-2.2

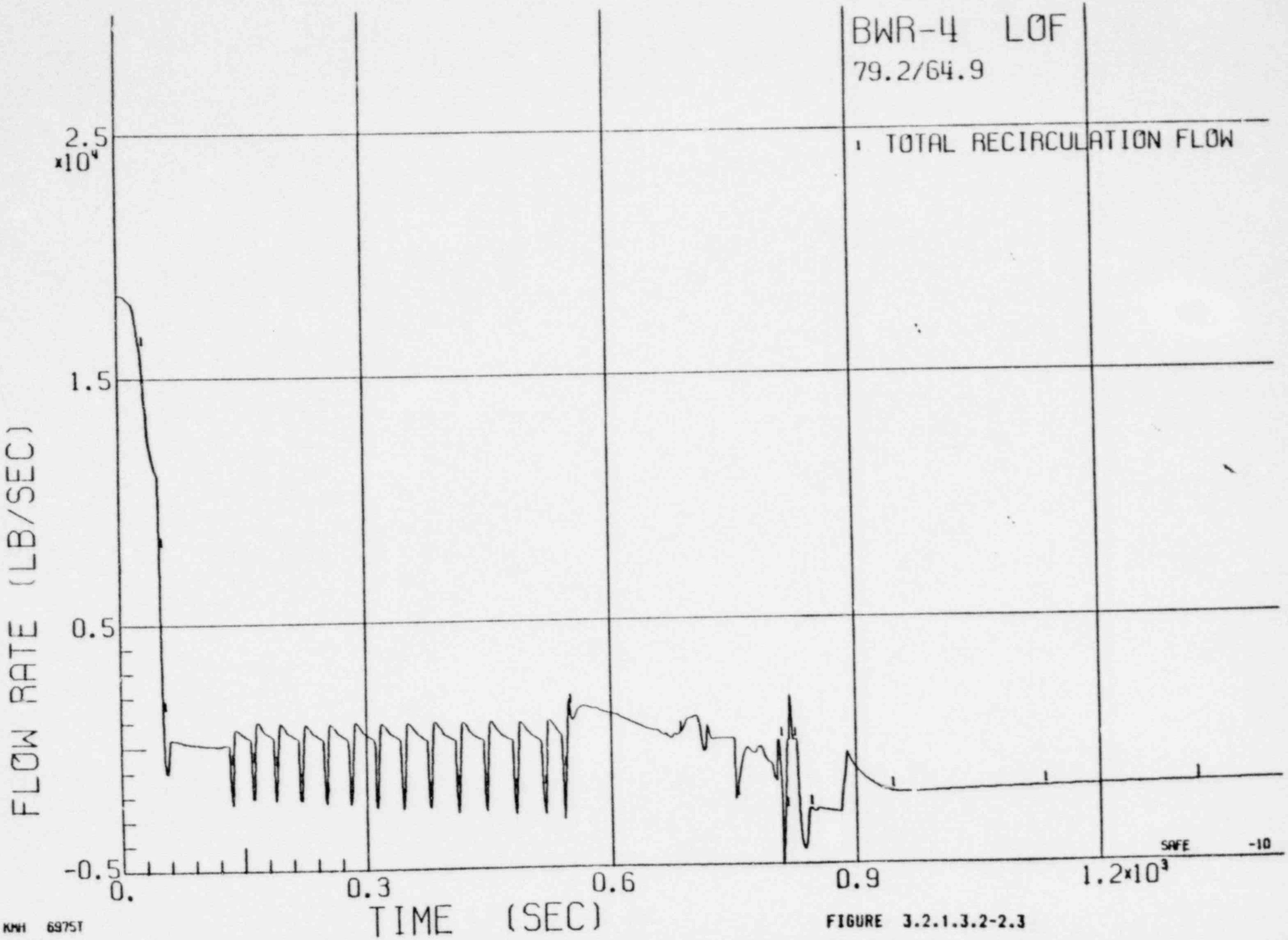
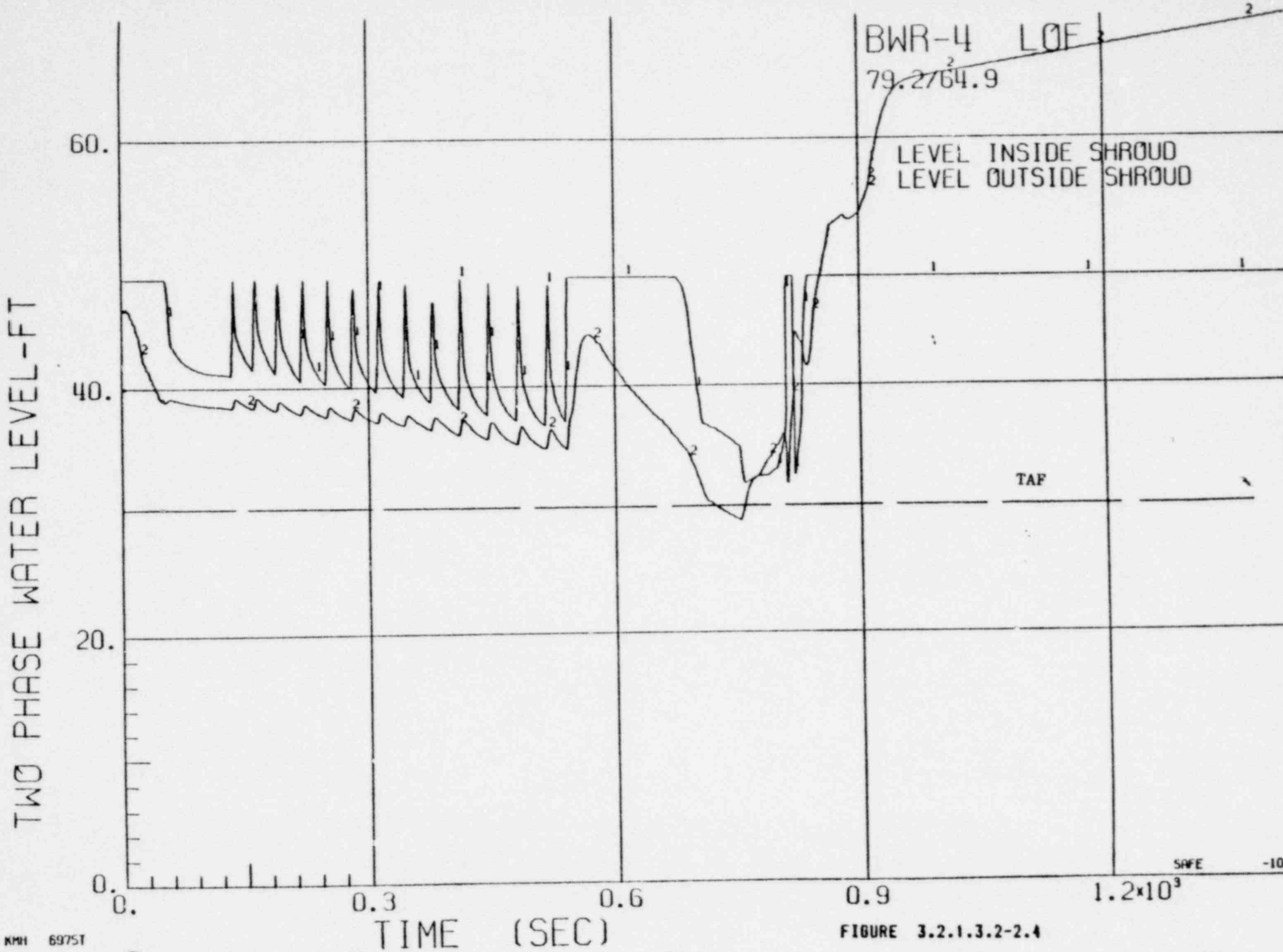


FIGURE 3.2.1.3.2-2.3

KMH 6975T



KMH 6975T

FIGURE 3.2.1.3.2-2.4

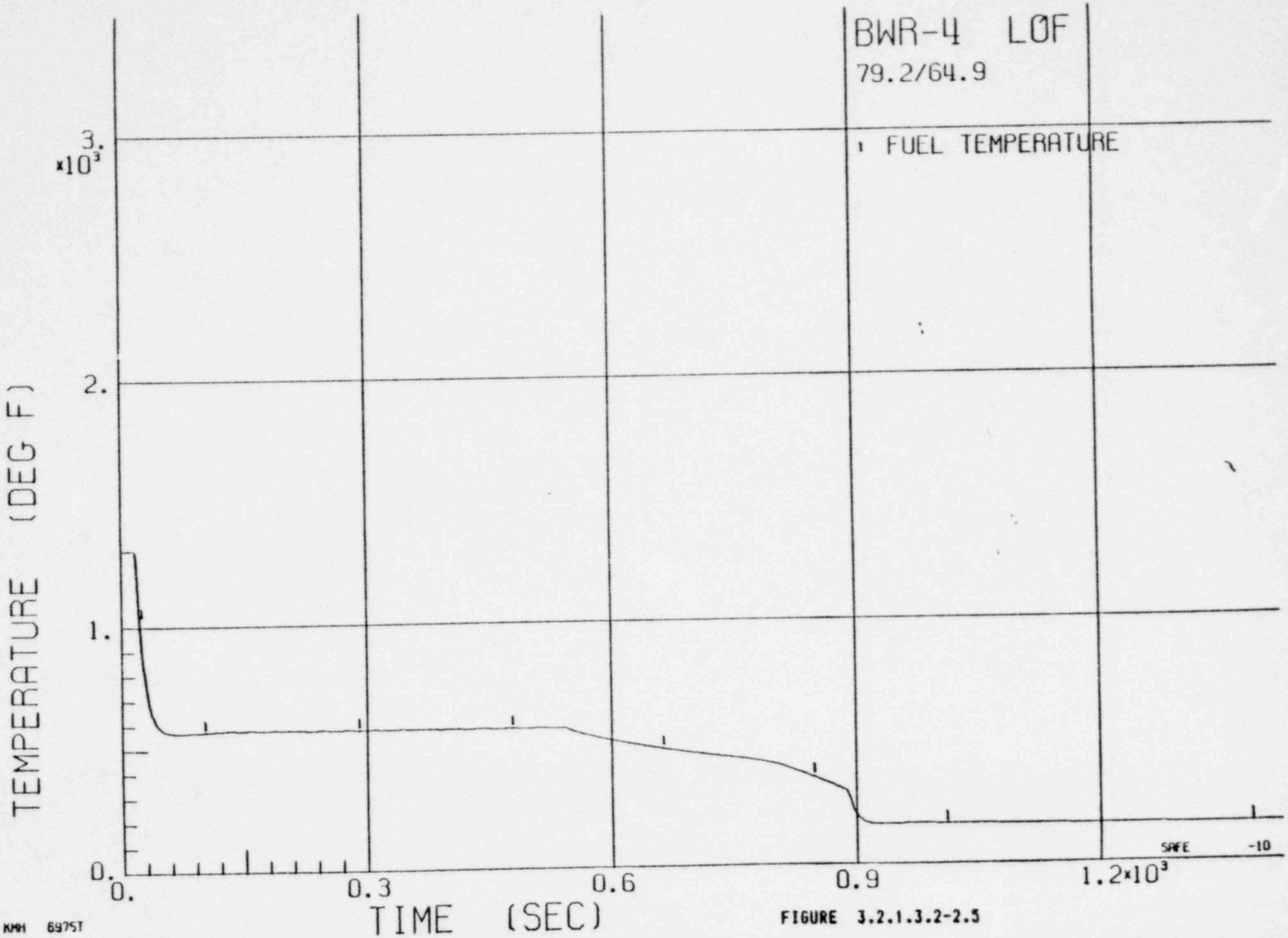
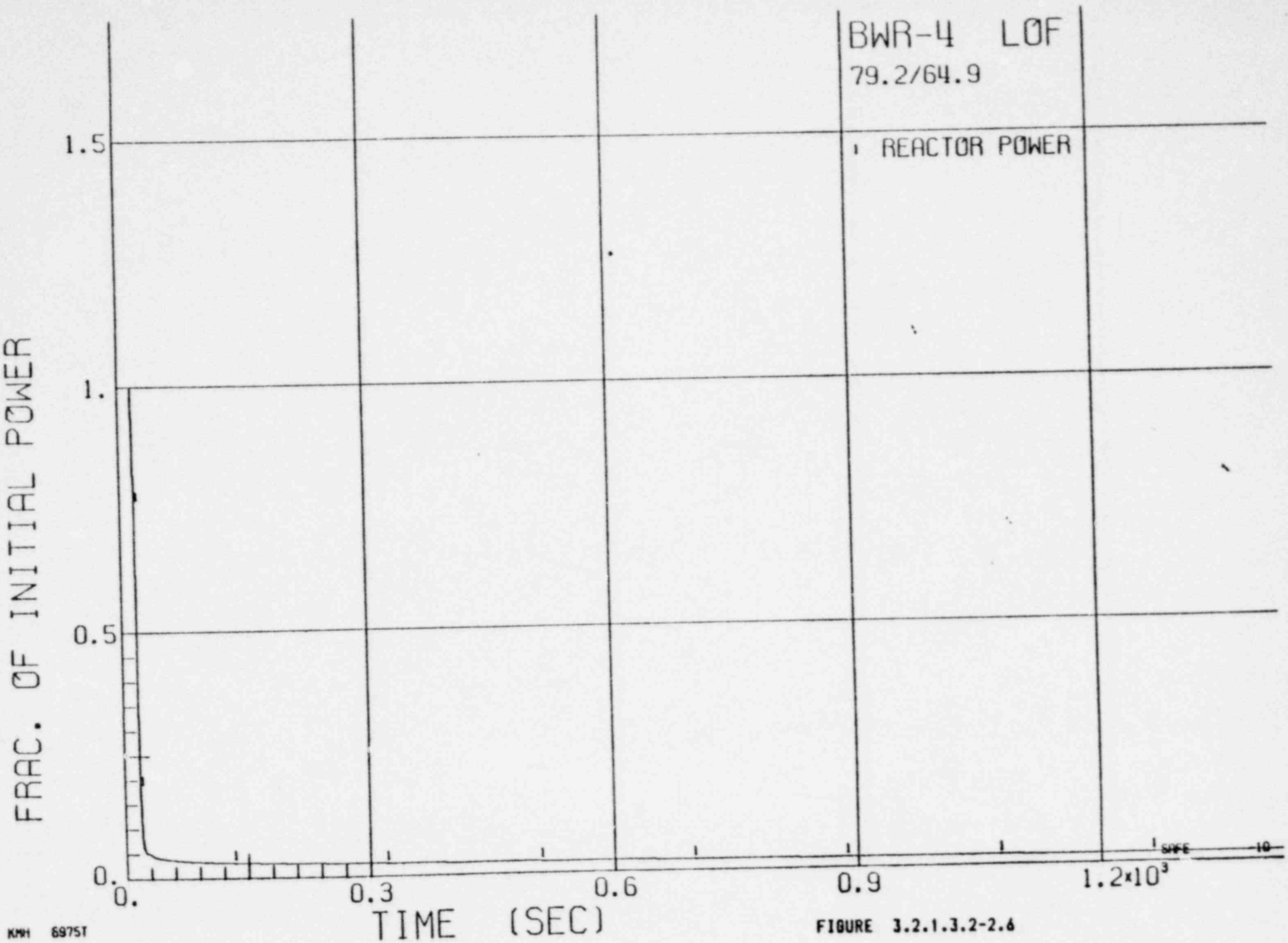


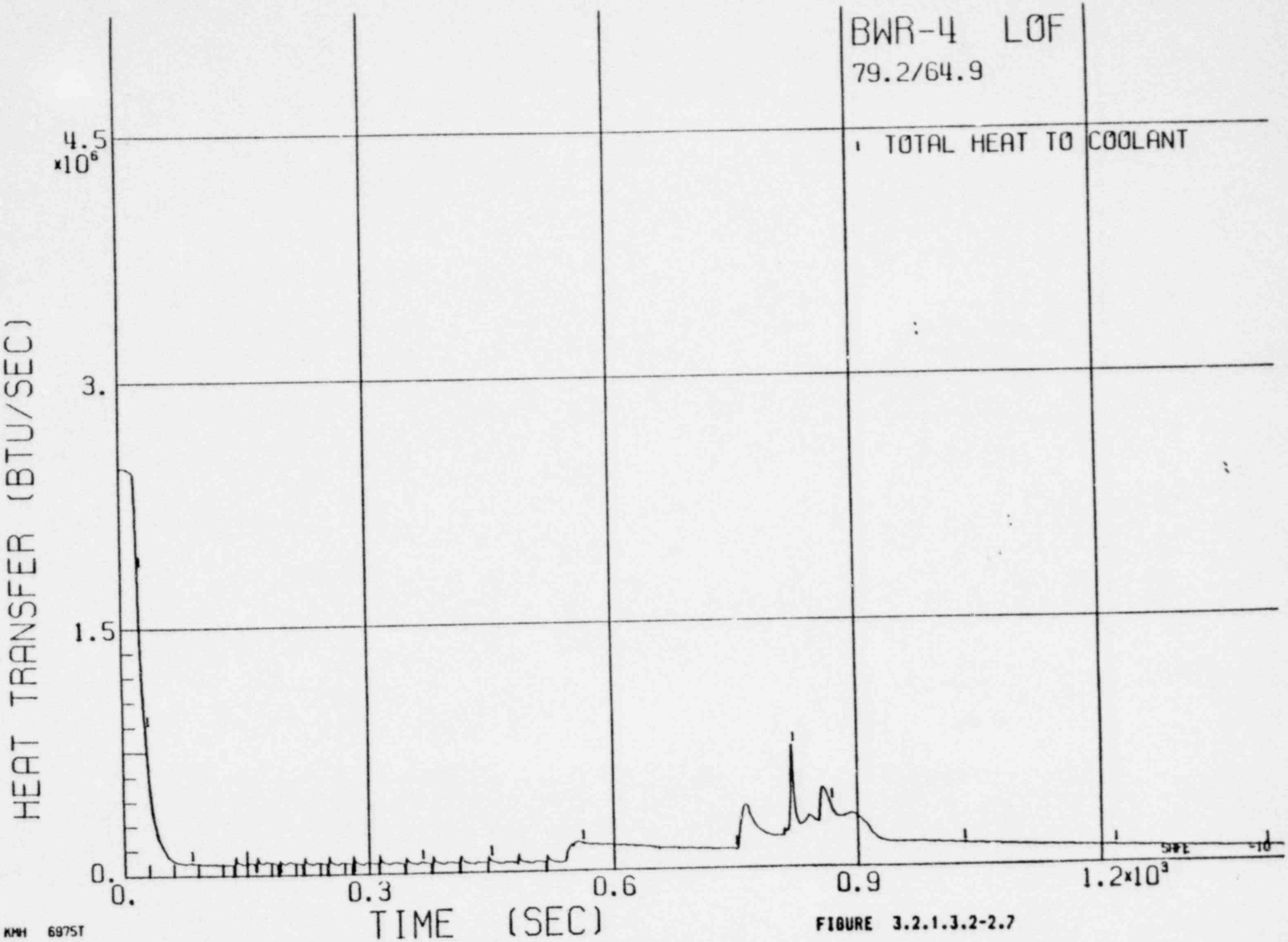
FIGURE 3.2.1.3.2-2.5

KMH 6975T



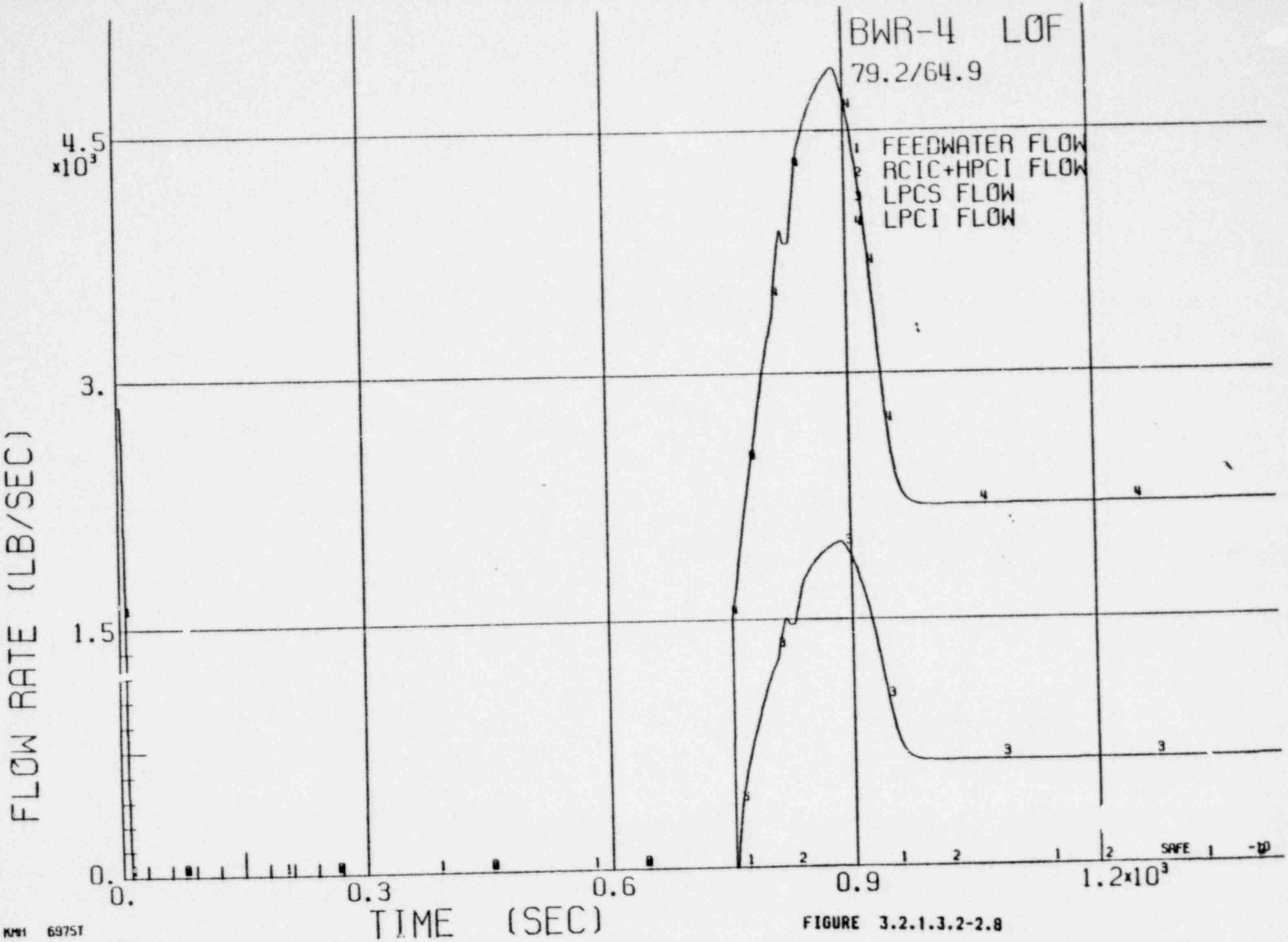
KMH 6975T

FIGURE 3.2.1.3.2-2.6



KMH 6975T

FIGURE 3.2.1.3.2-2.7



KMH 6975T

FIGURE 3.2.1.3.2-2.8

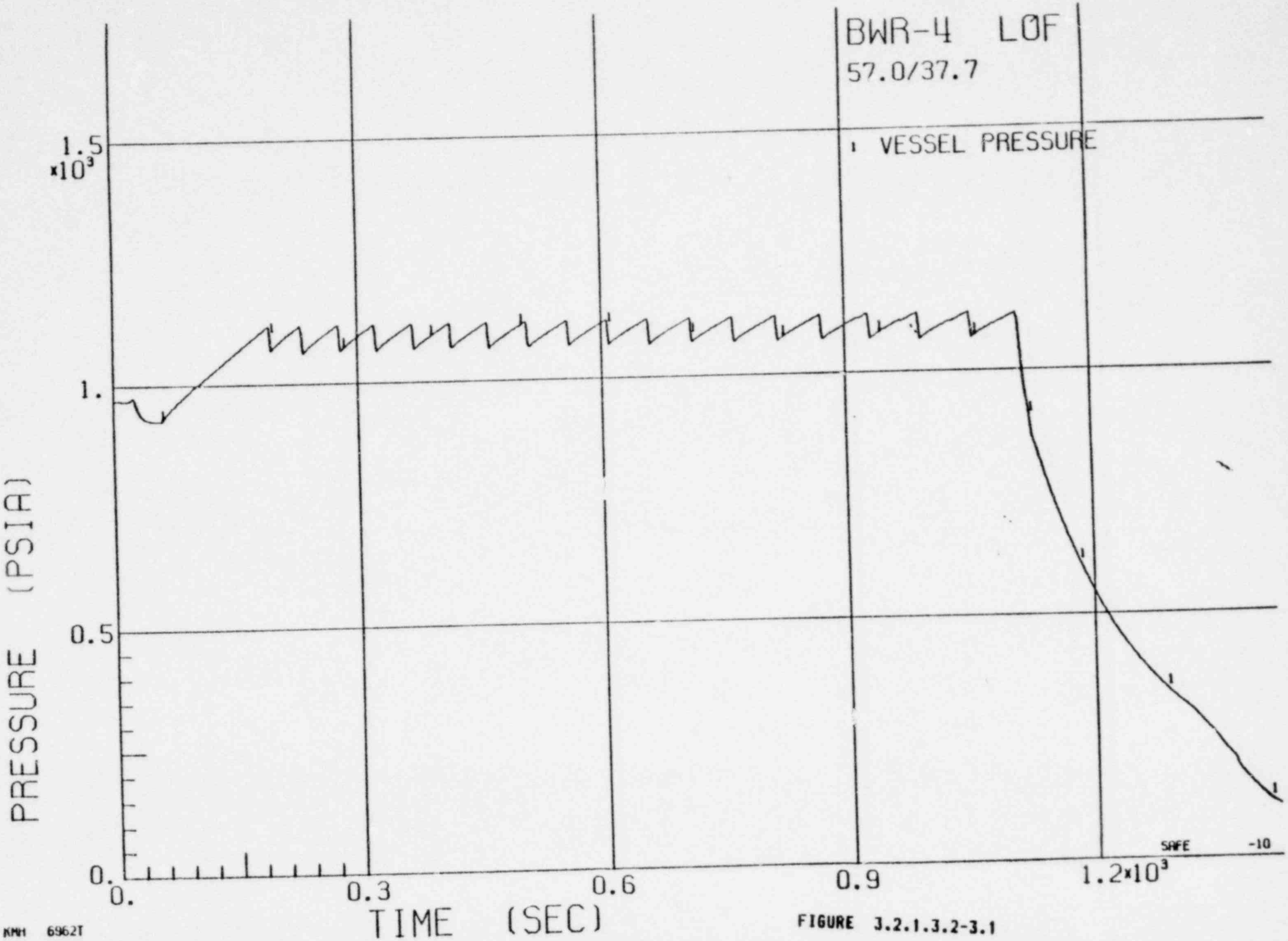
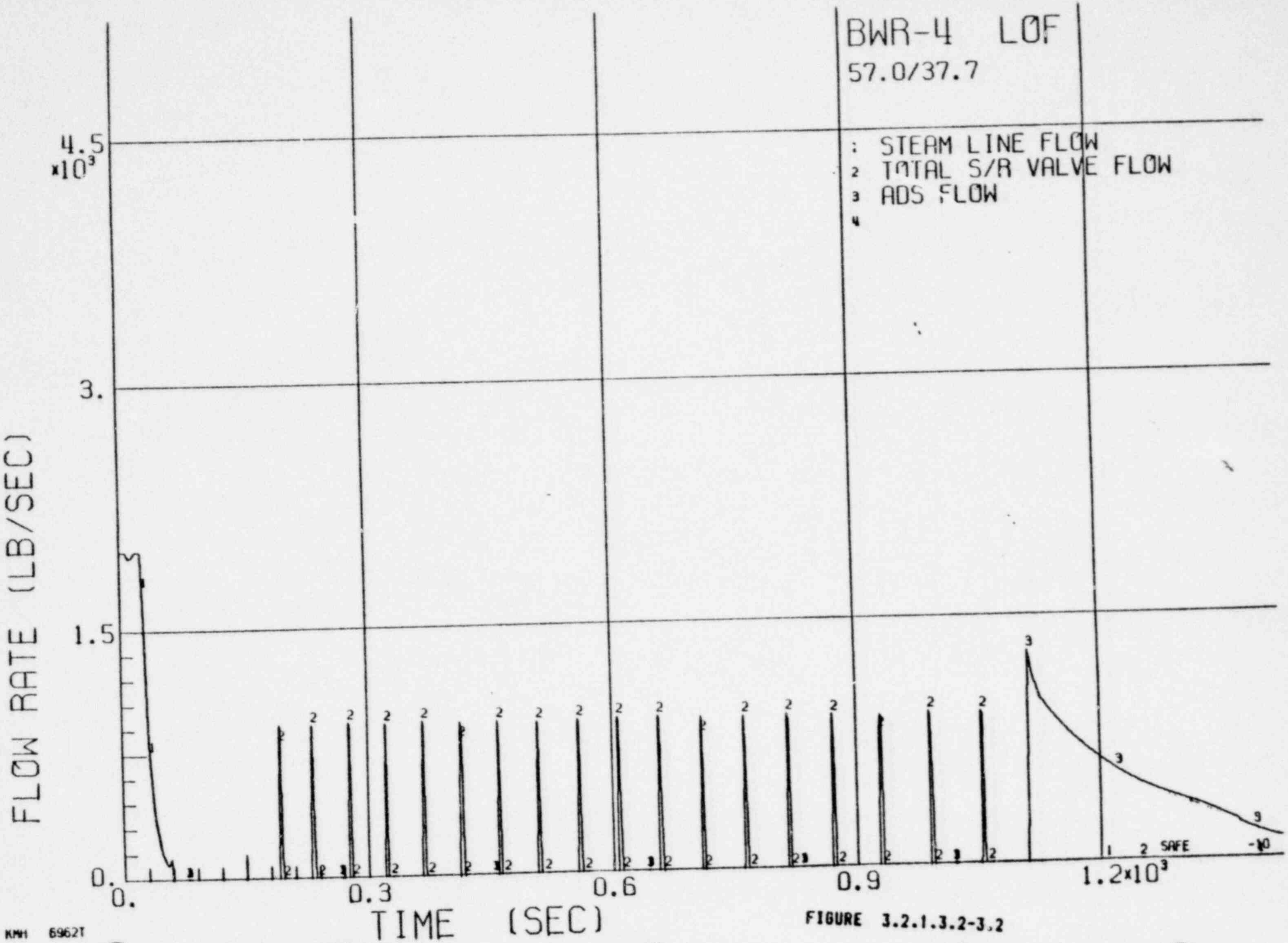


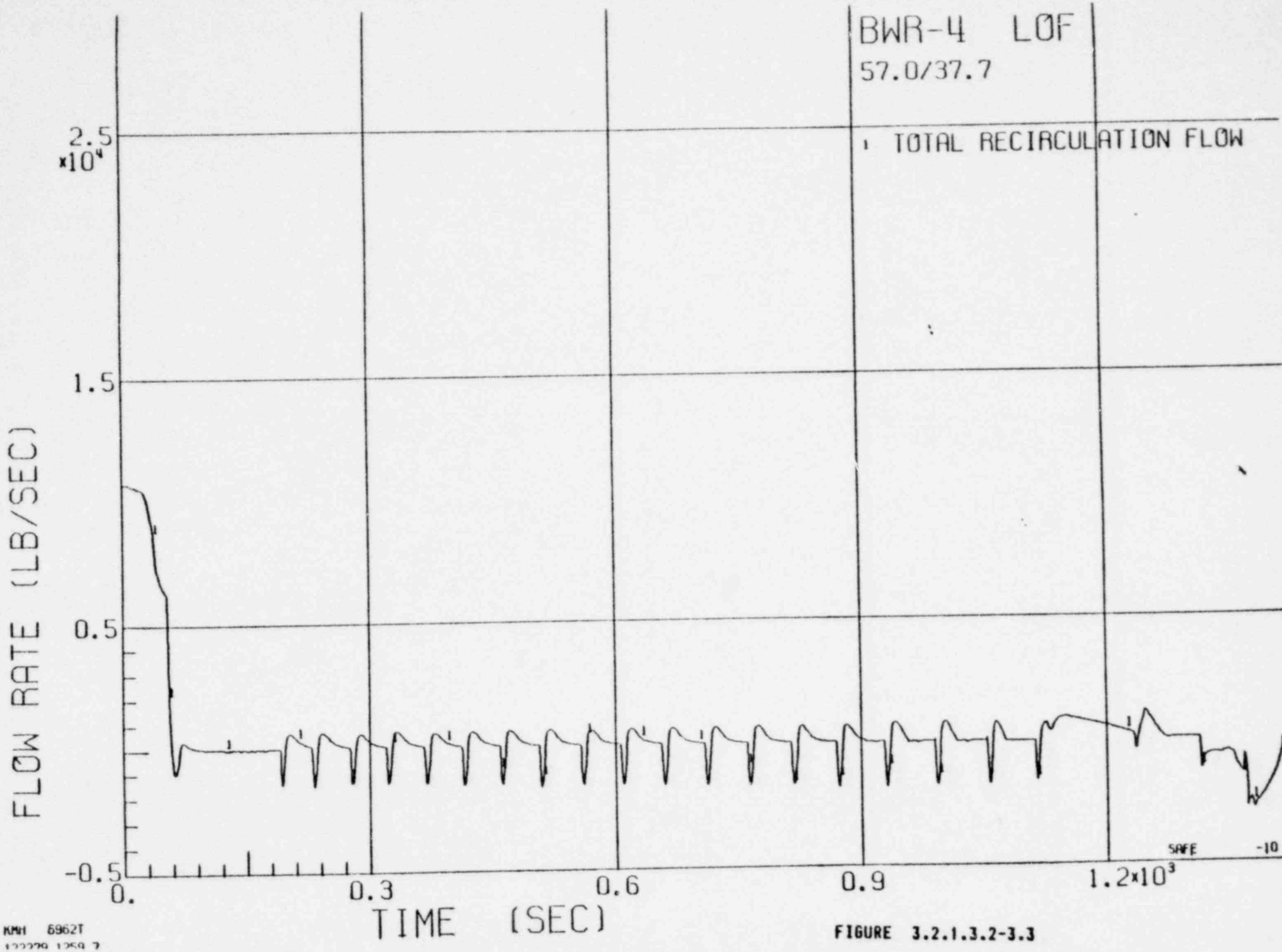
FIGURE 3.2.1.3.2-3.1

KMH 6962T



KMH 6962T

FIGURE 3.2.1.3.2-3.2



KMH 6962T

FIGURE 3.2.1.3.2-3.3

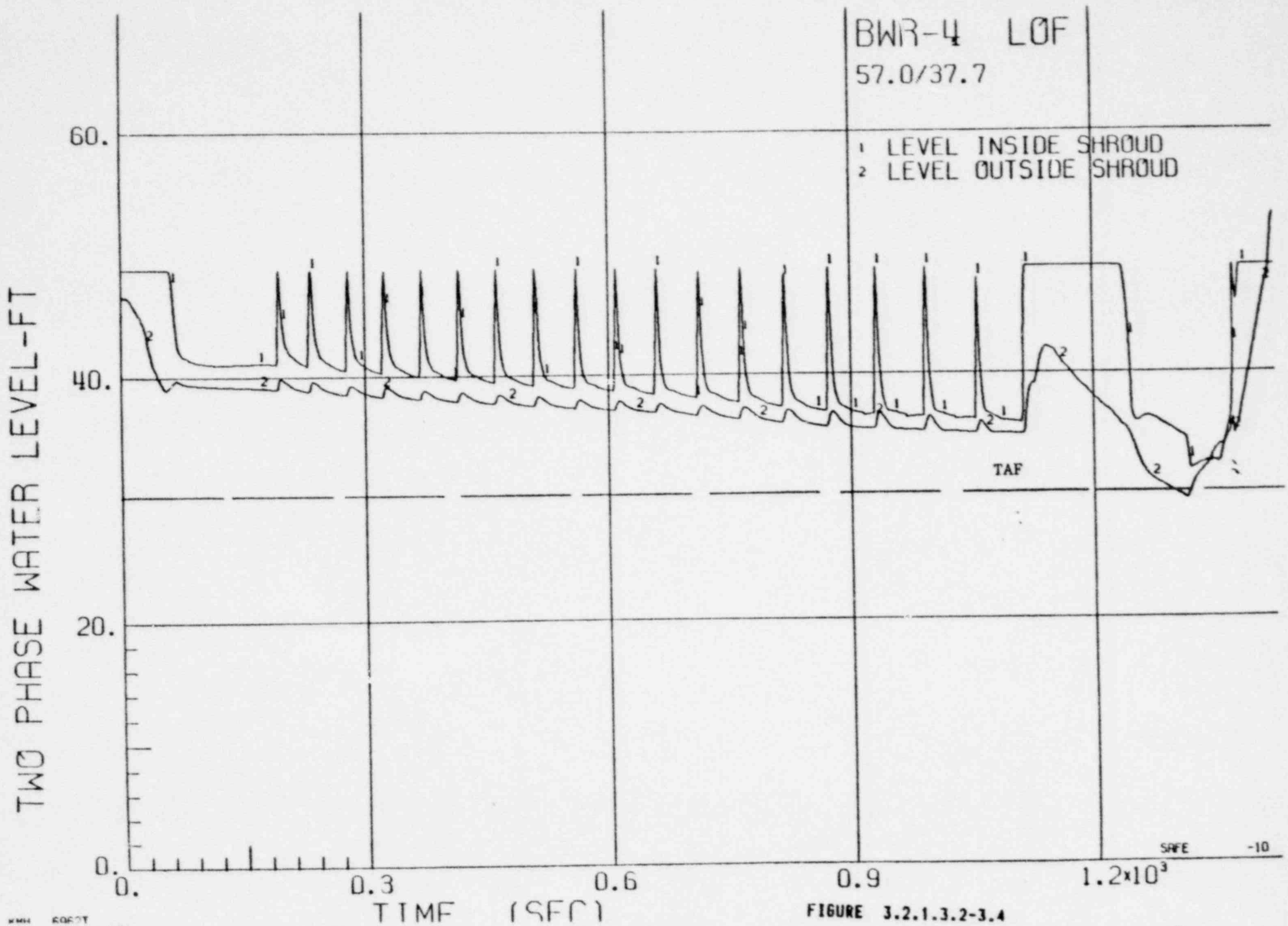


FIGURE 3.2.1.3.2-3.4

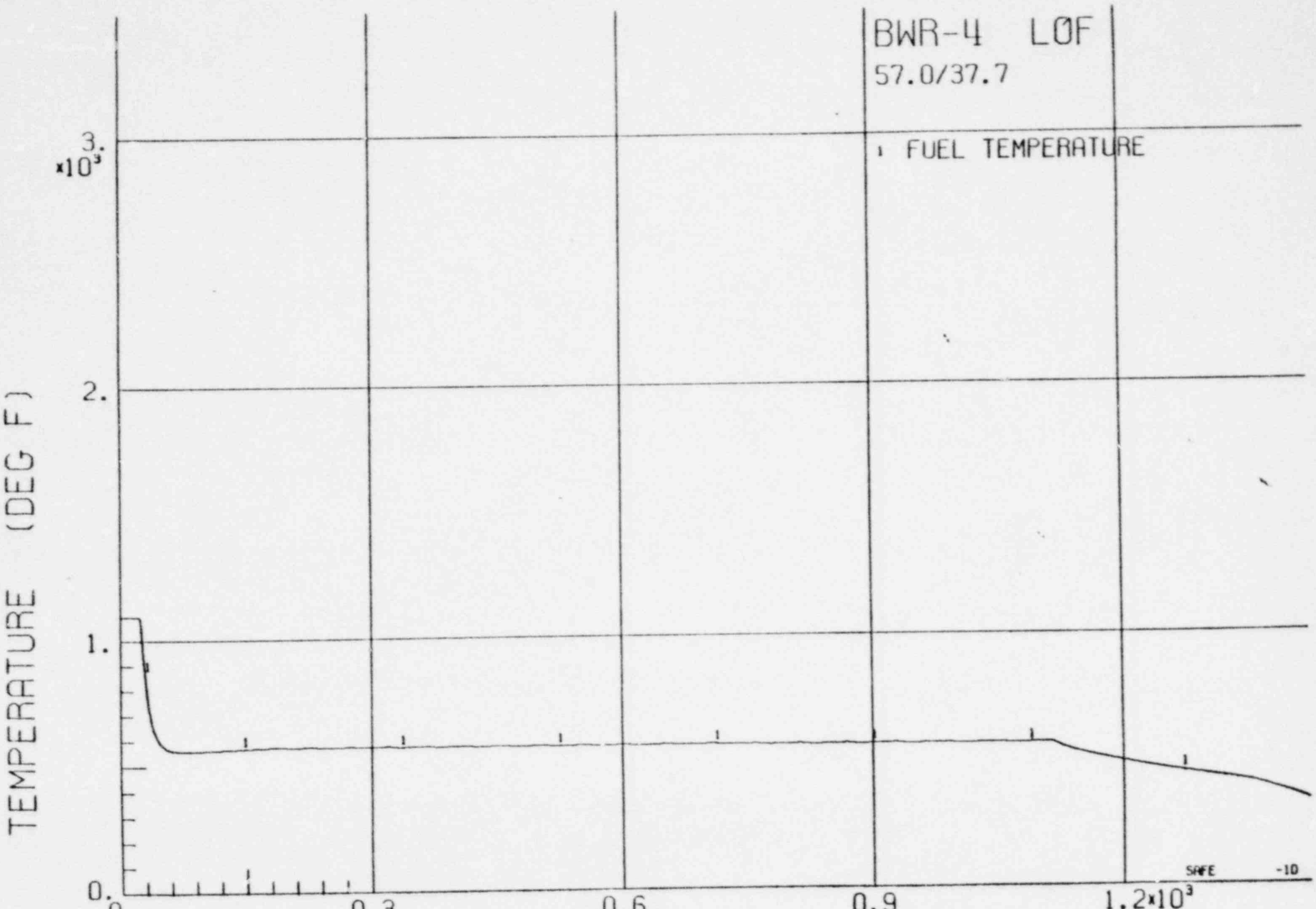


FIGURE 3.2.1.3.2-3.5

KMH 6962T

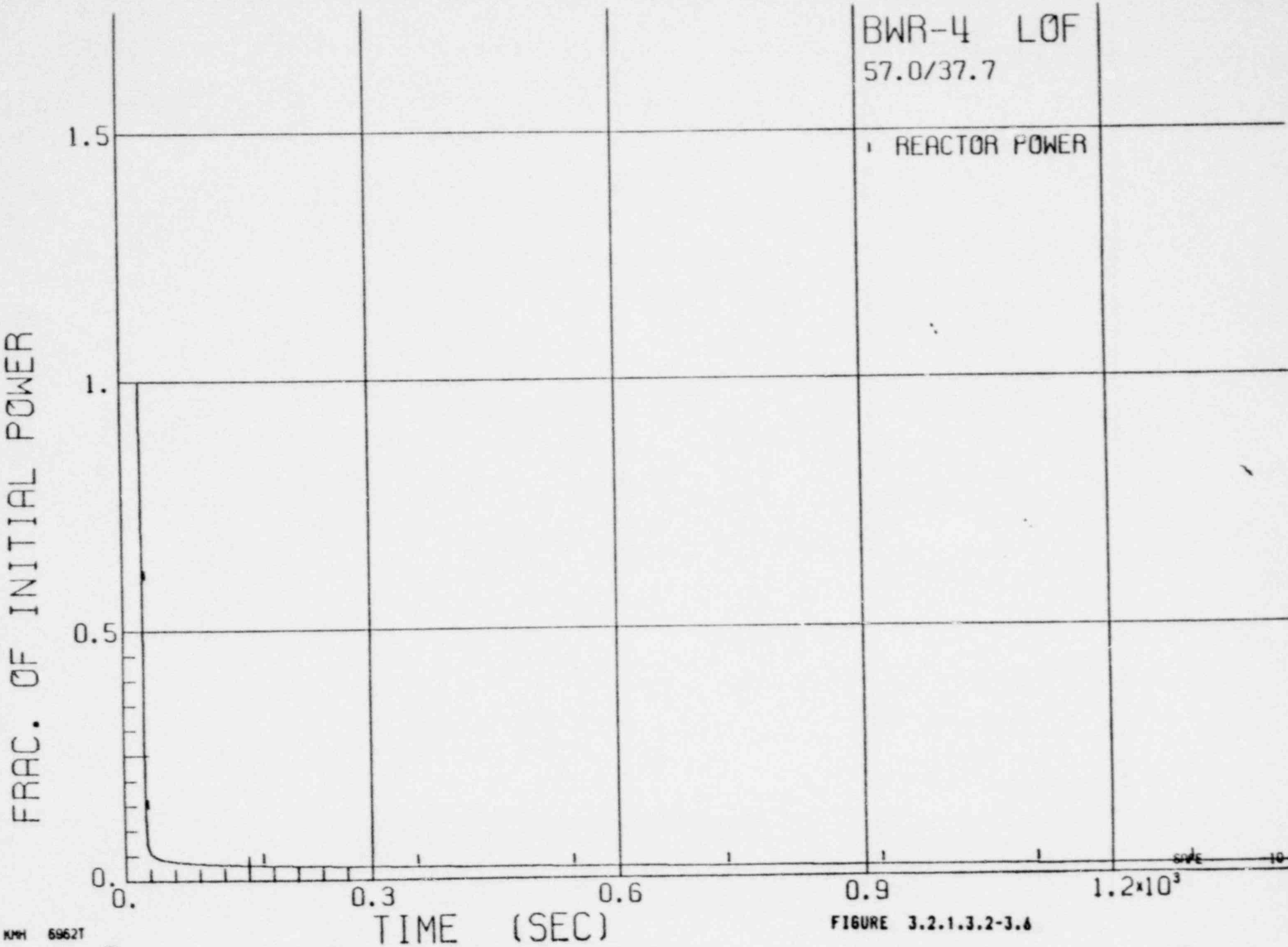
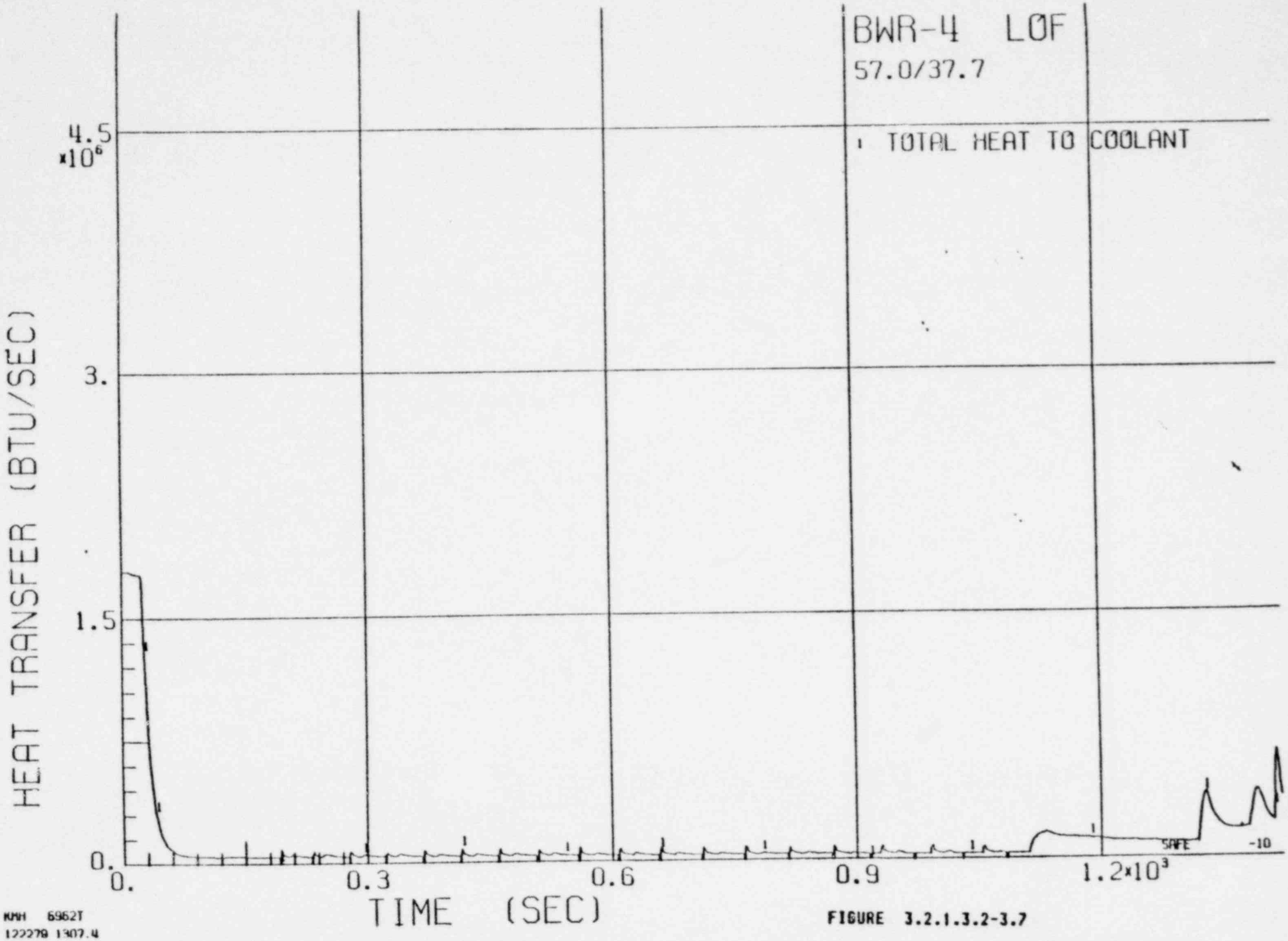


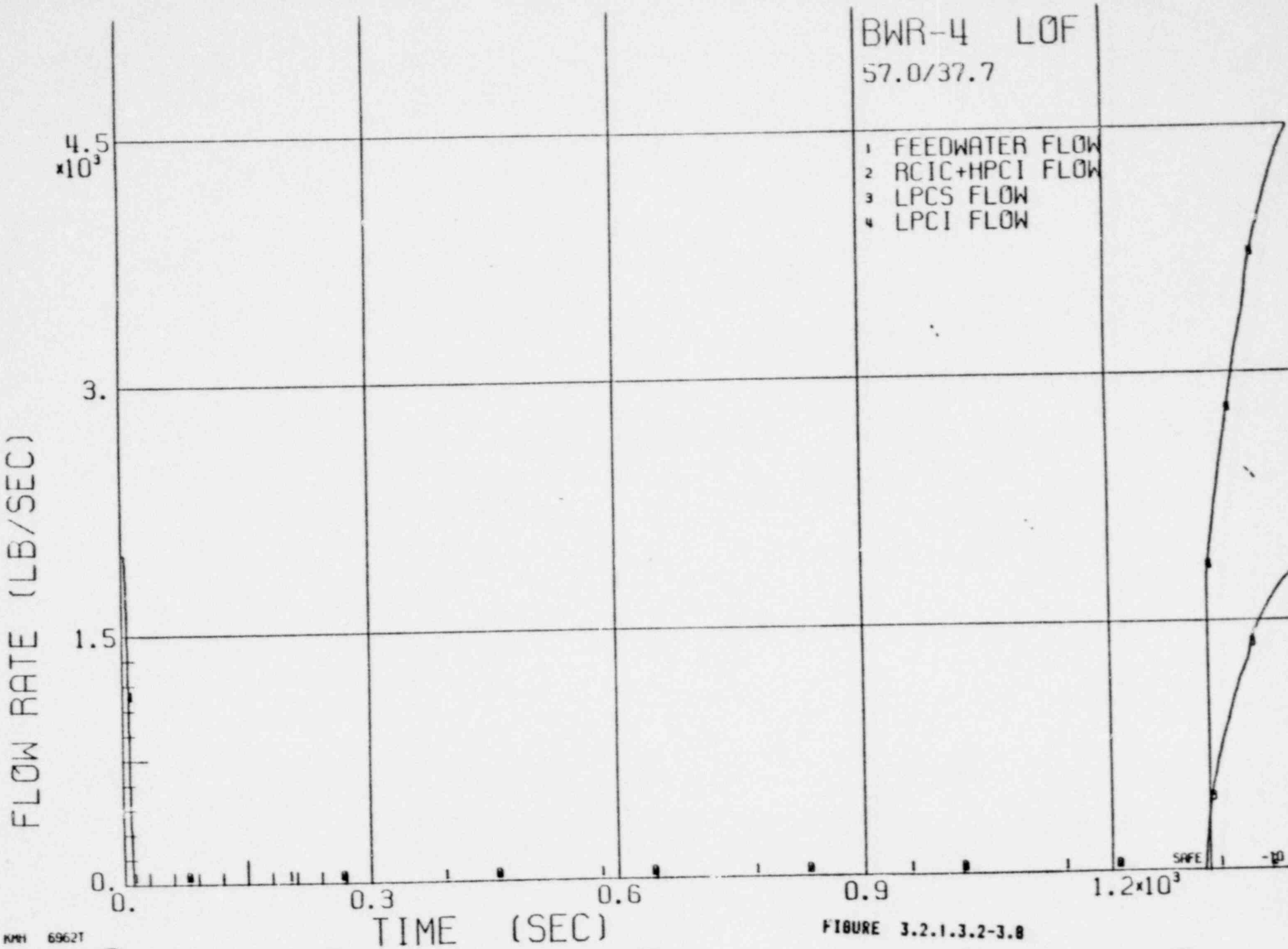
FIGURE 3.2.1.3.2-3.6

KMH 6962T



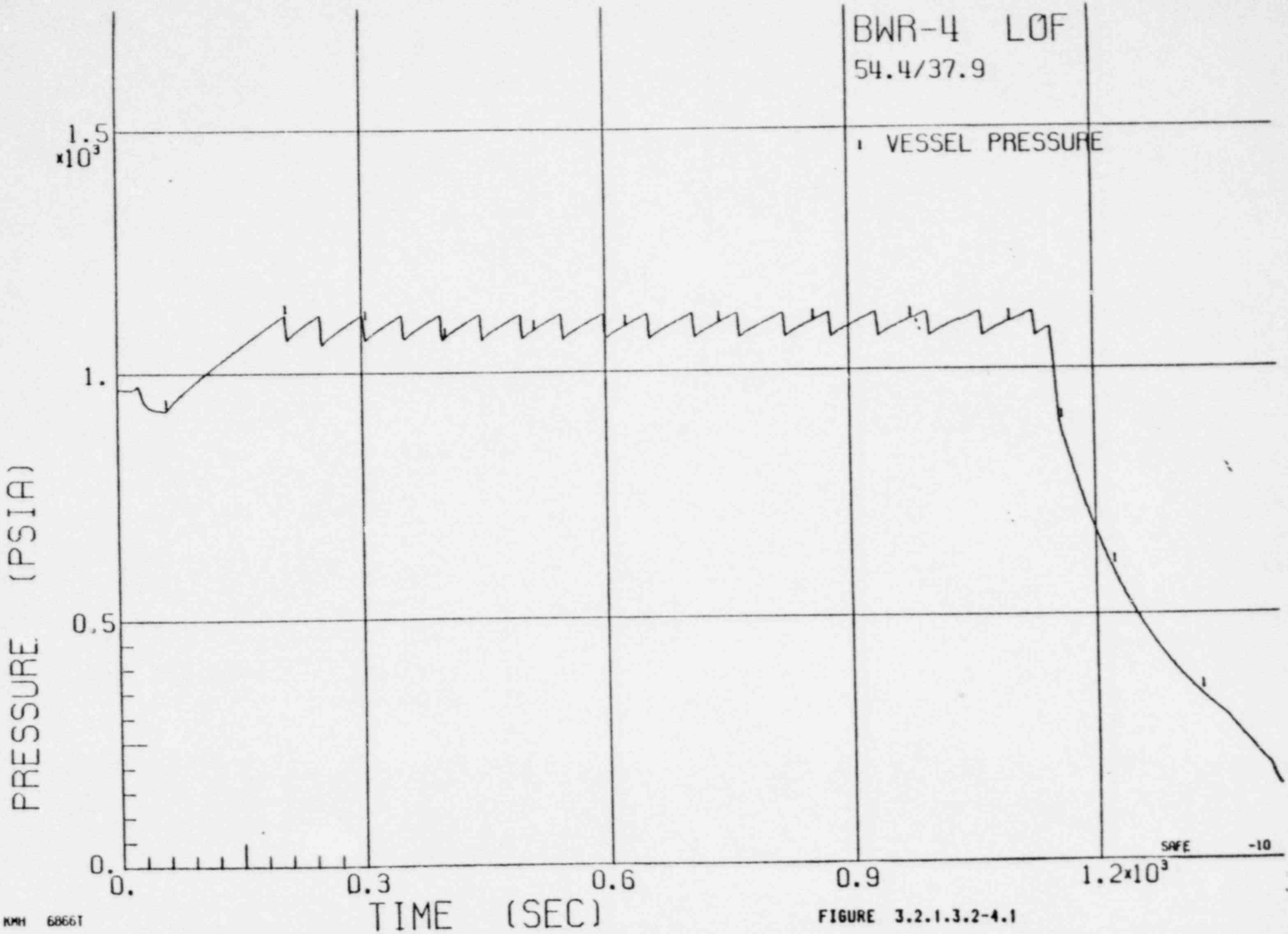
KMH 6962T
122279 1307.4

FIGURE 3.2.1.3.2-3.7



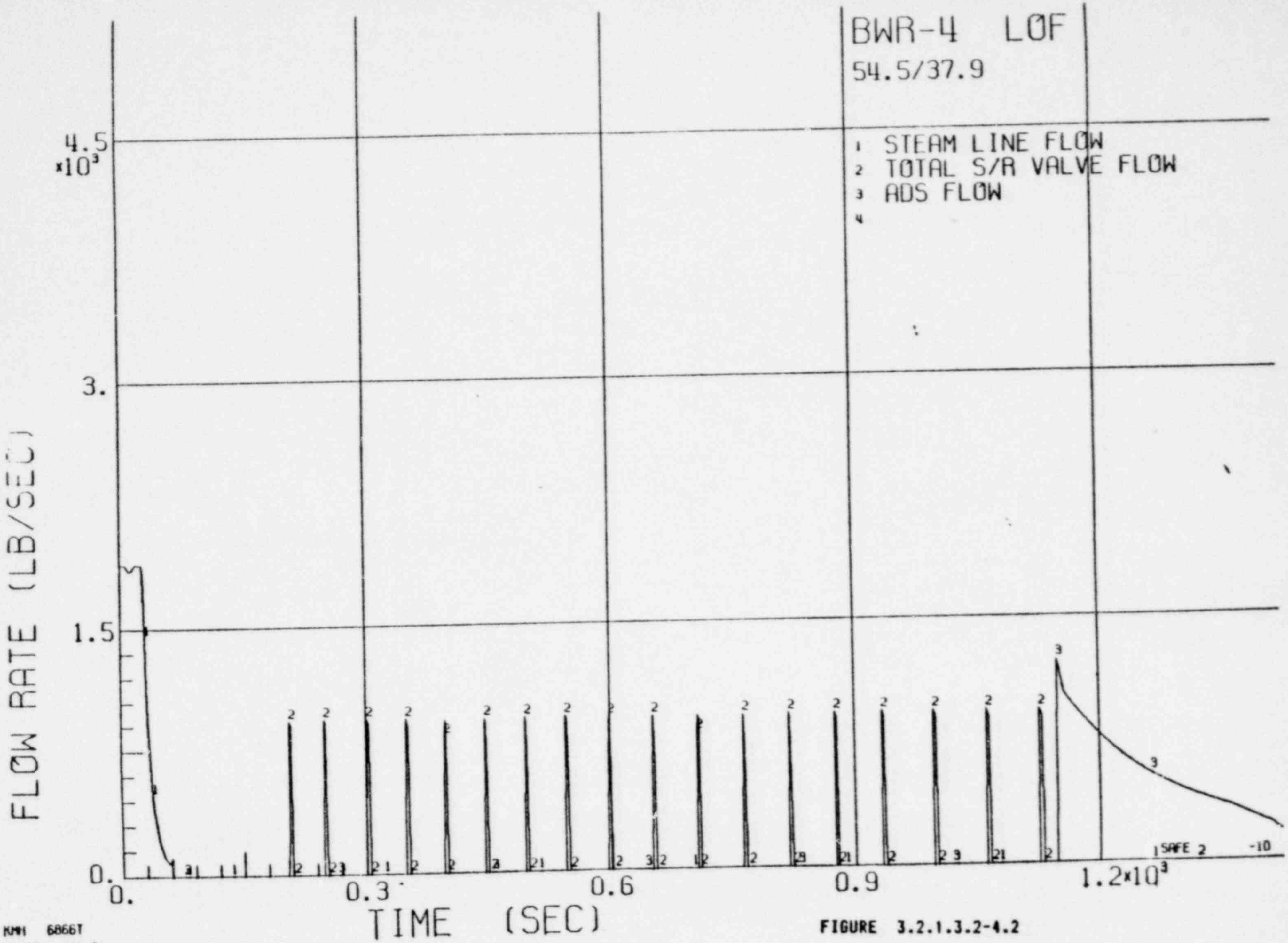
KMH 6962T

FIGURE 3.2.1.3.2-3.8



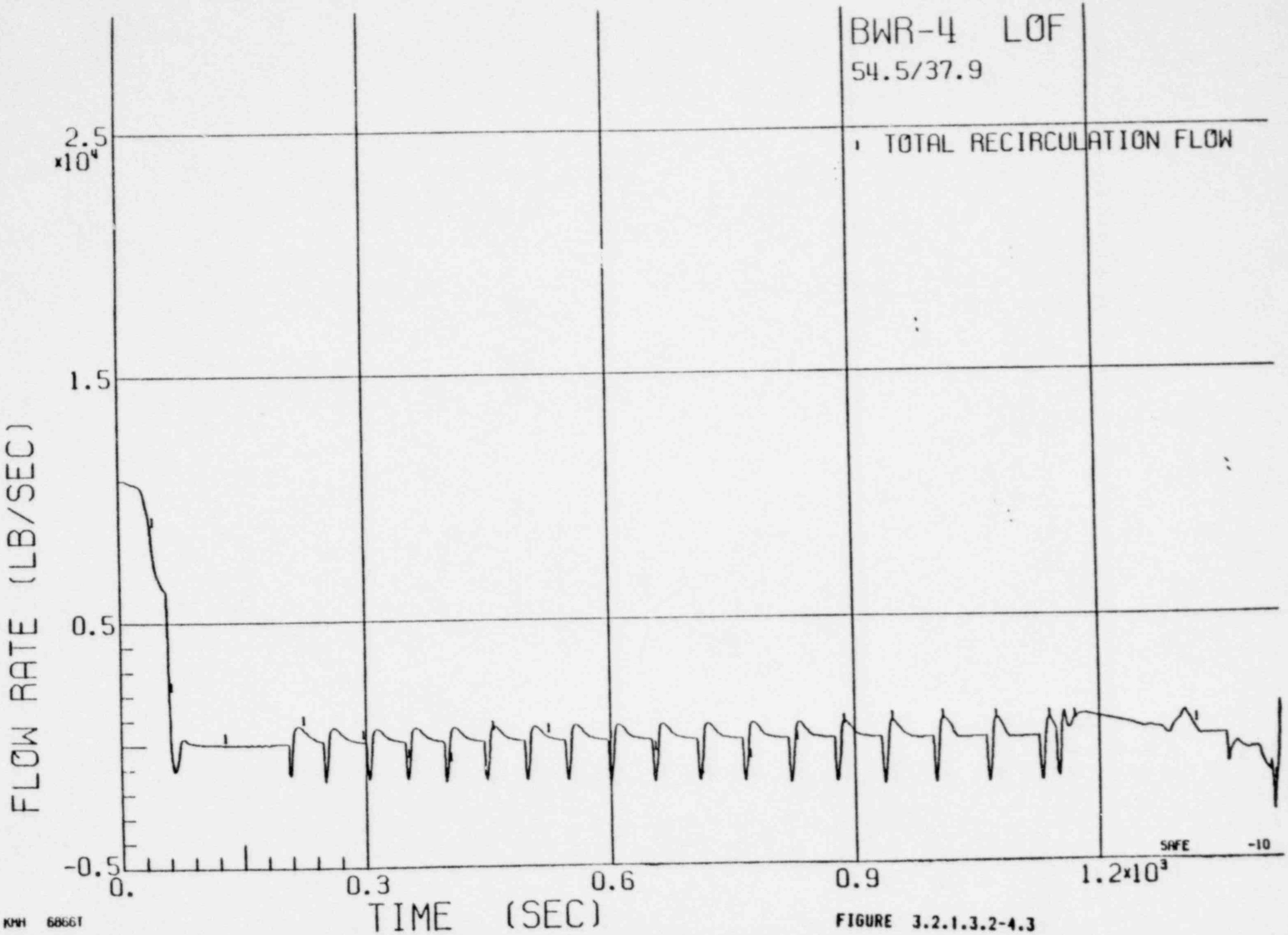
KMH 6866T

FIGURE 3.2.1.3.2-4.1



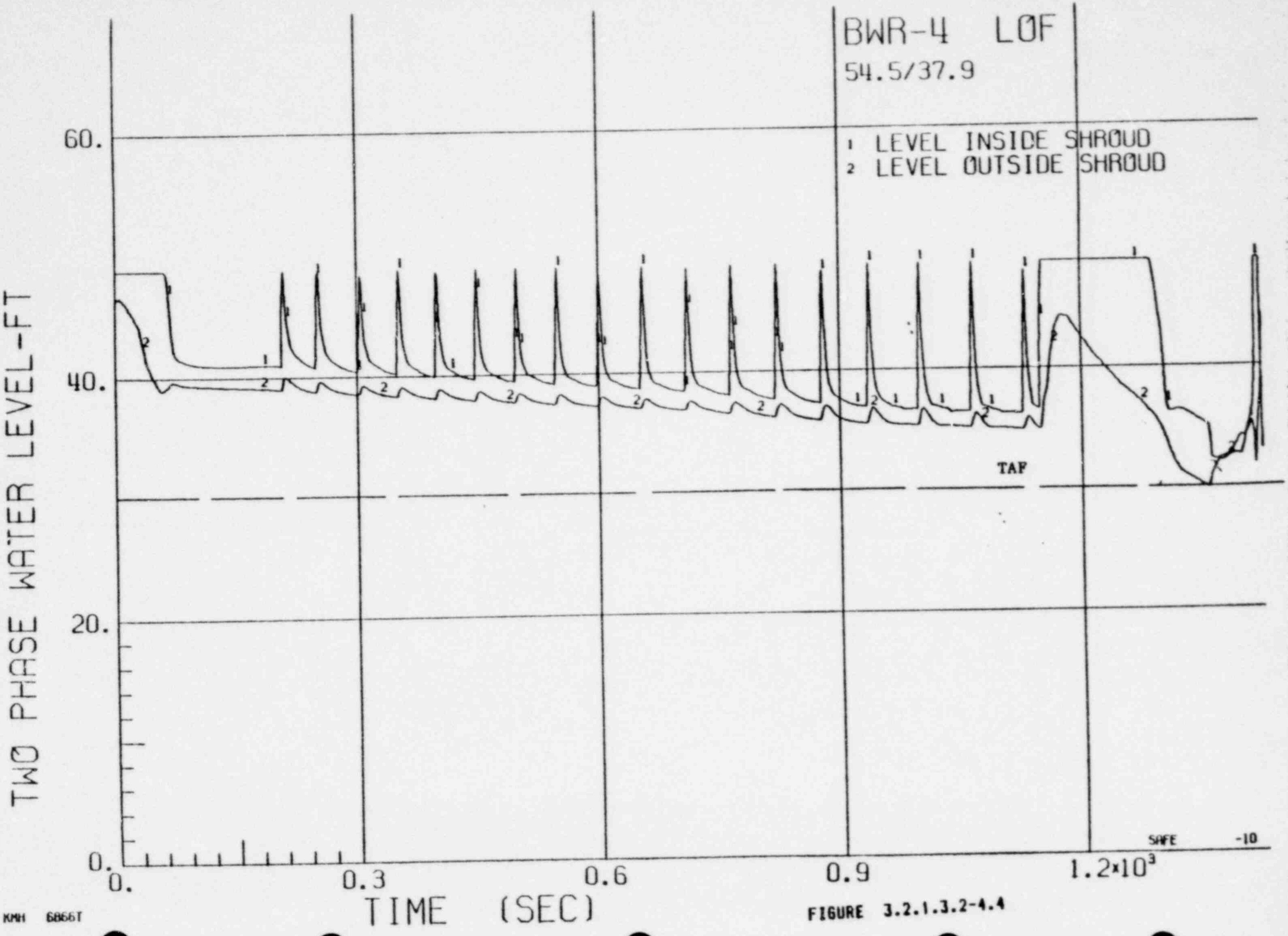
KMH 6866T

FIGURE 3.2.1.3.2-4.2



KMH 6866T

FIGURE 3.2.1.3.2-4.3



KMH 6866T

FIGURE 3.2.1.3.2-4.4

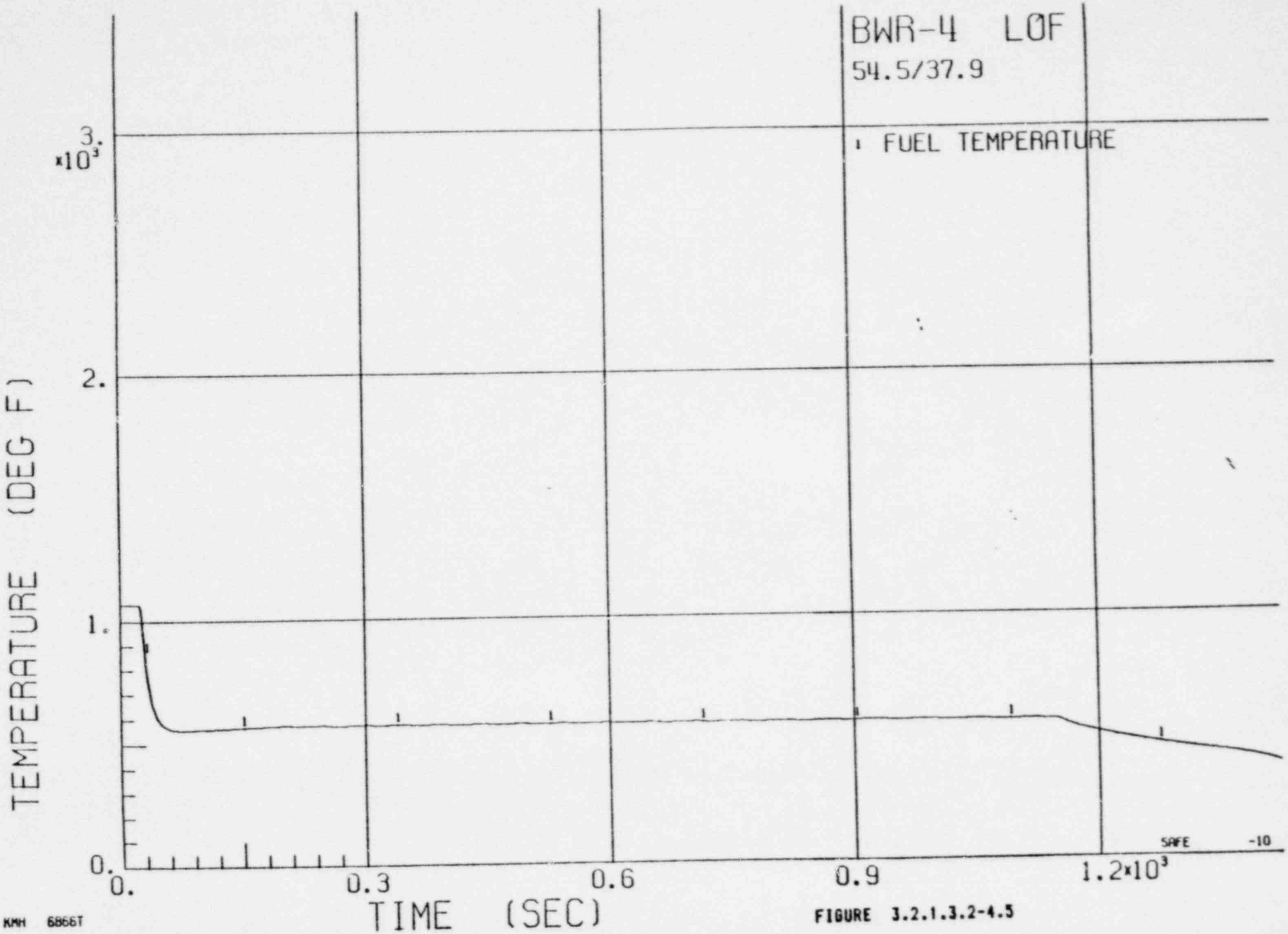
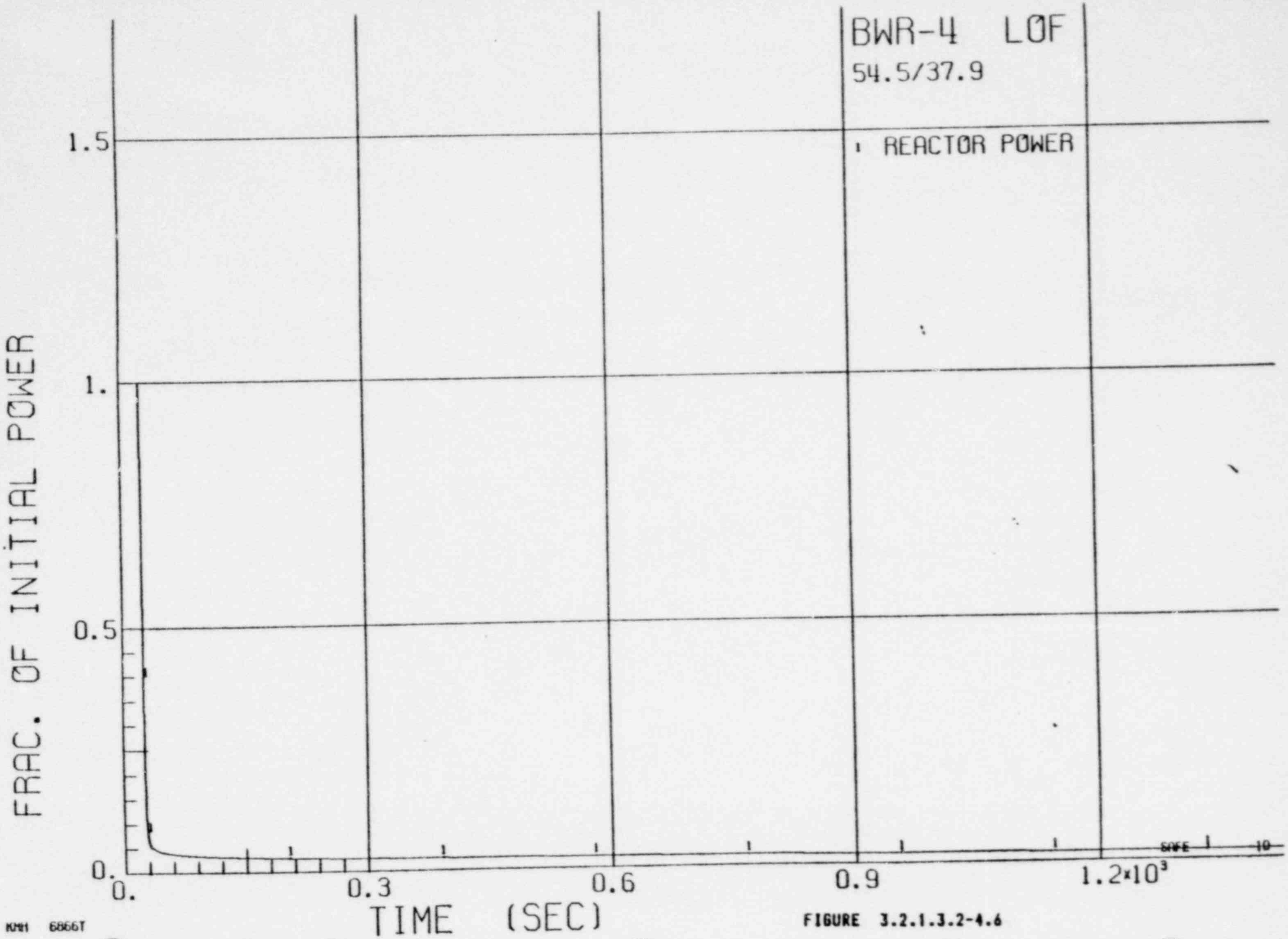


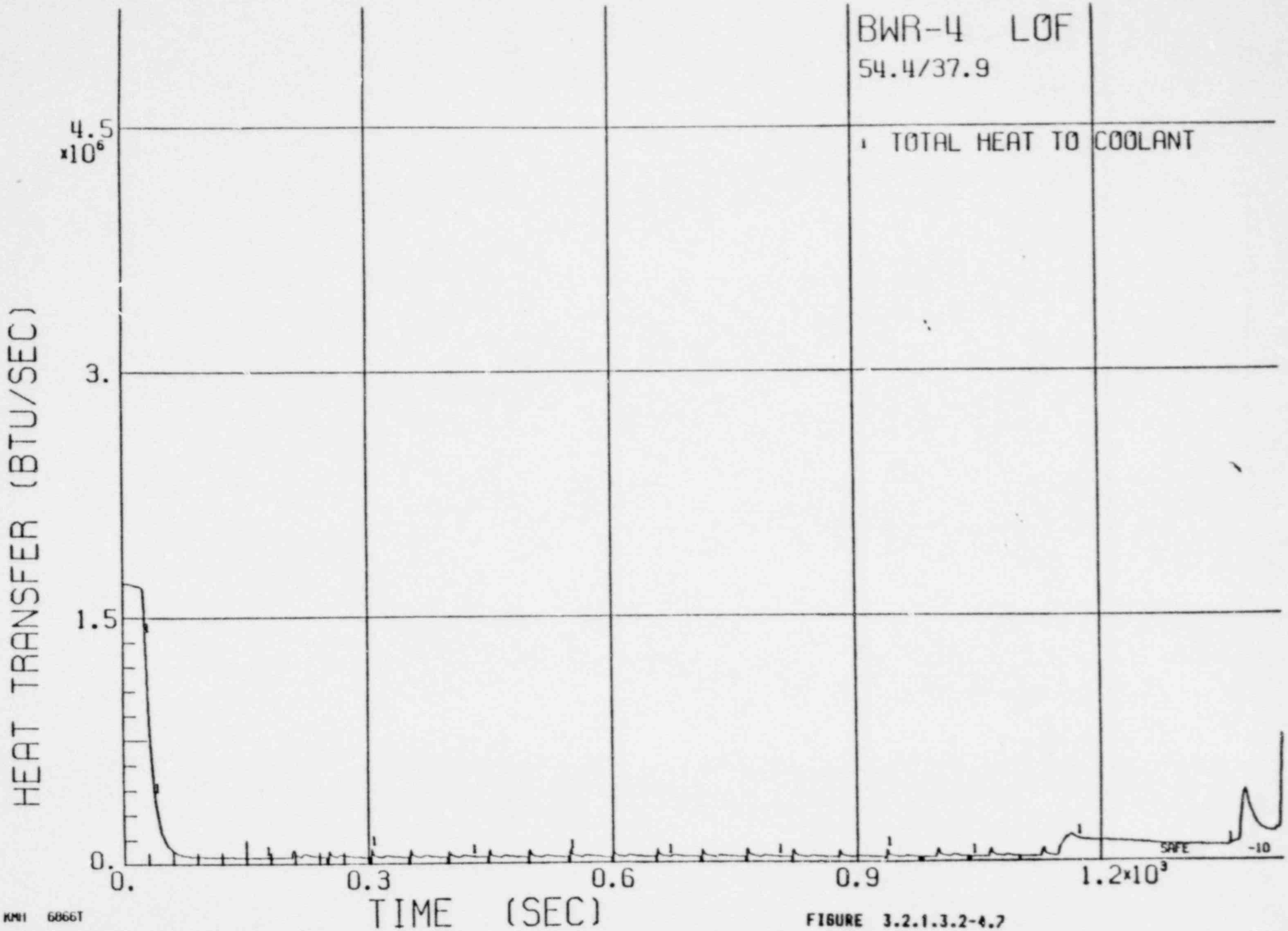
FIGURE 3.2.1.3.2-4.5

KMH 6866T



KMH 6866T

FIGURE 3.2.1.3.2-4.6



KMI 6866T

FIGURE 3.2.1.3.2-4.7

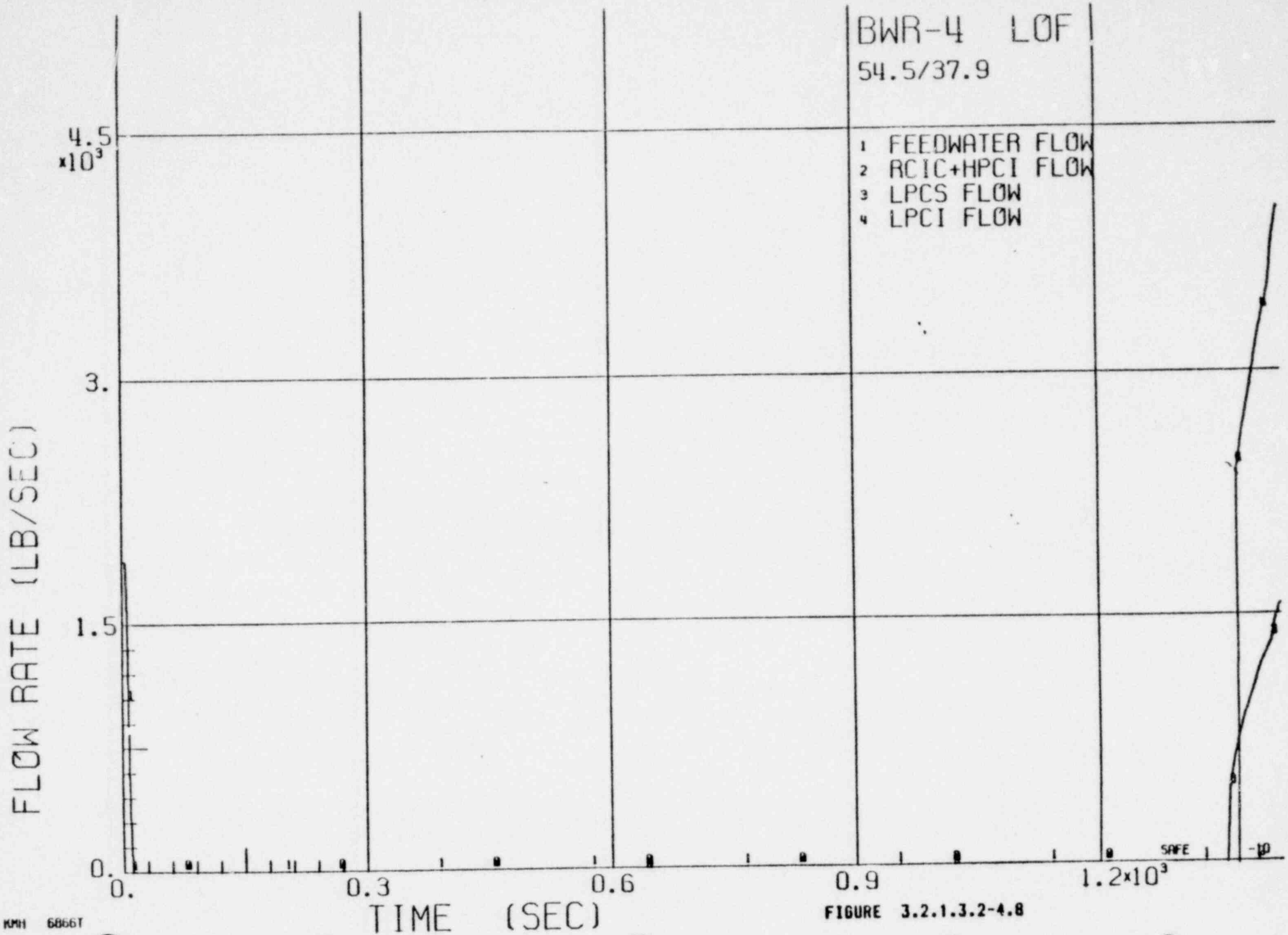


FIGURE 3.2.1.3.2-4.8

KMH 6866T

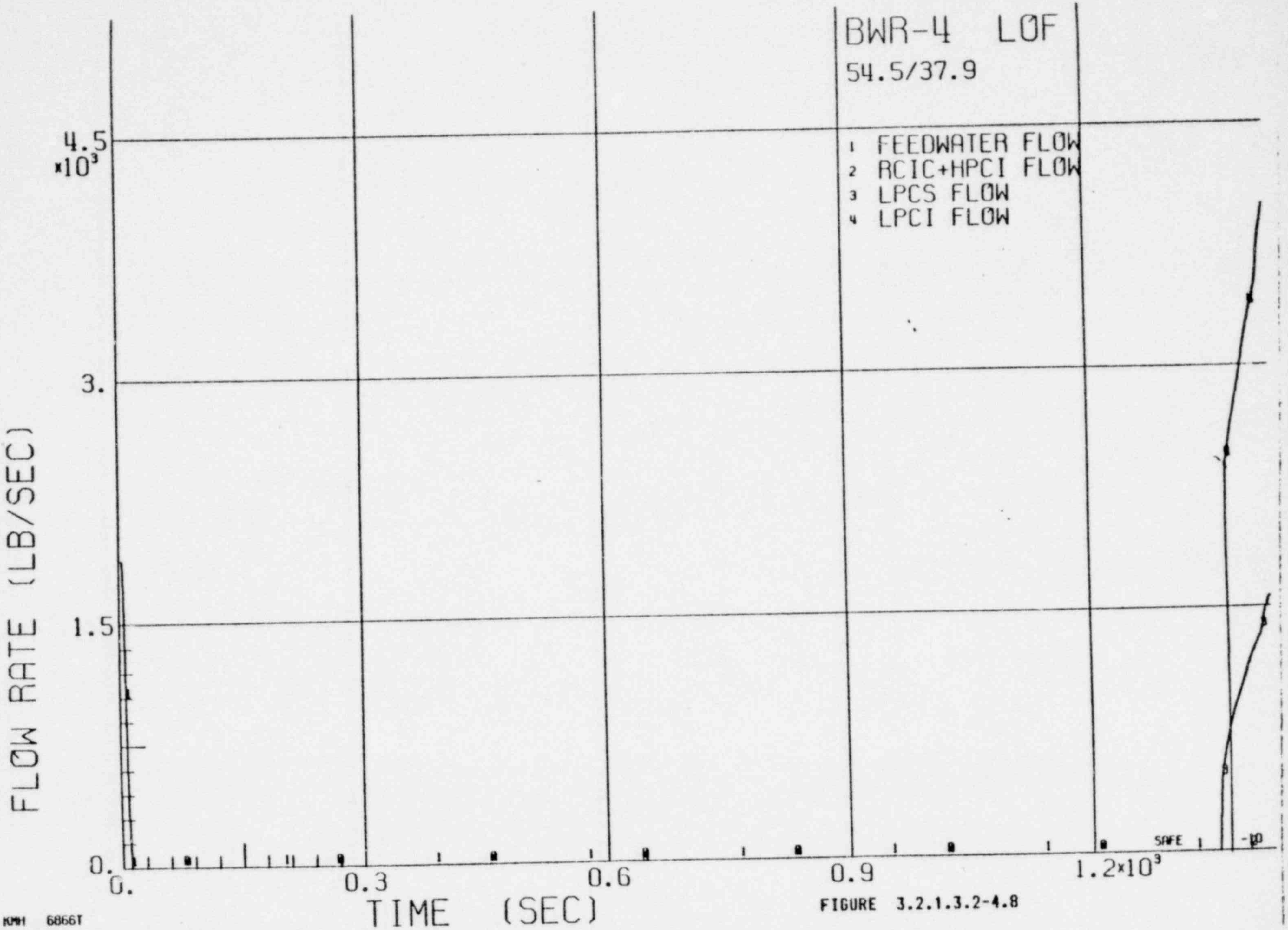


FIGURE 3.2.1.3.2-4.8

KMH 6866T

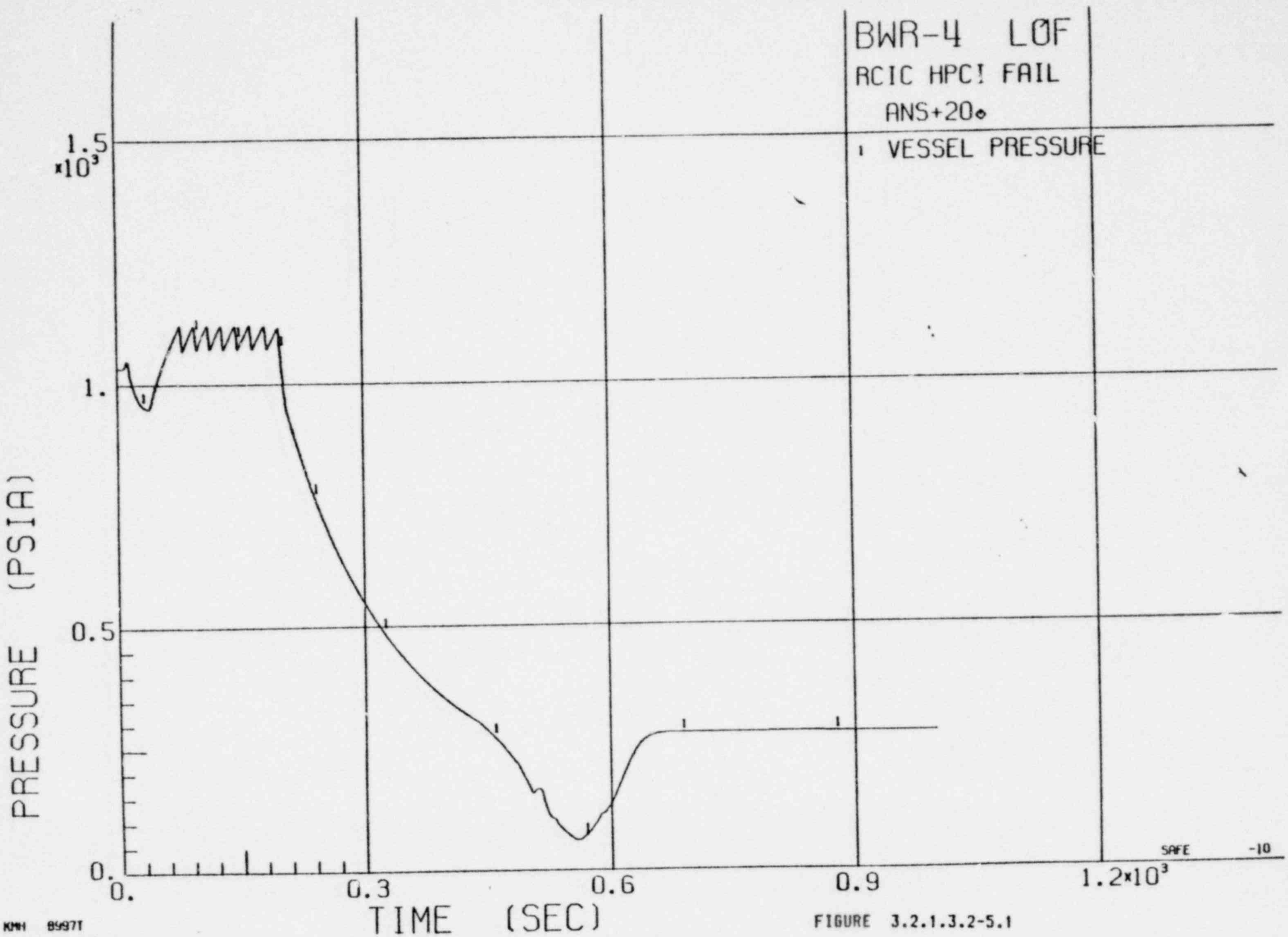


FIGURE 3.2.1.3.2-5.1

KMH 8997T

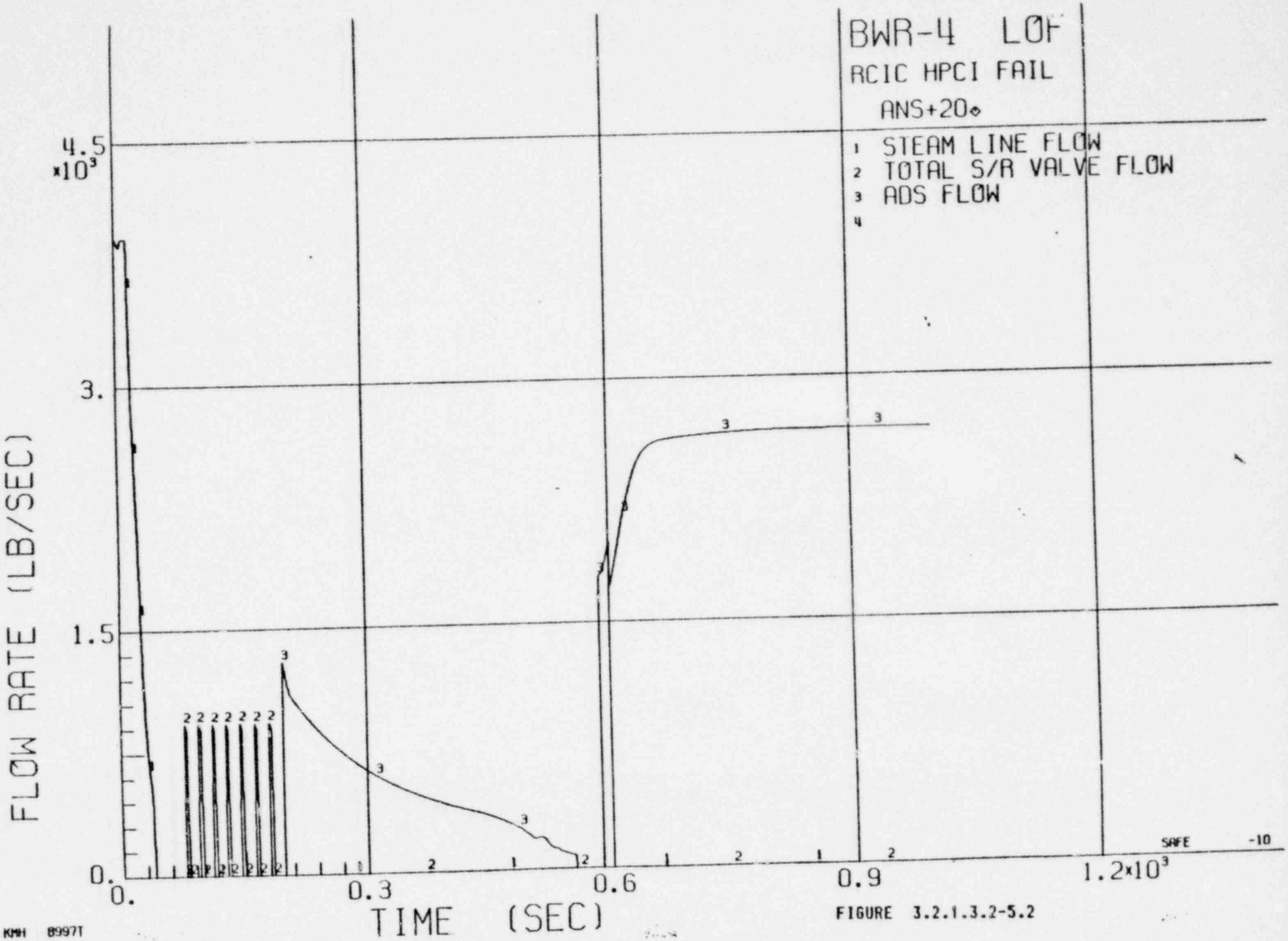


FIGURE 3.2.1.3.2-5.2

KMH 8997T

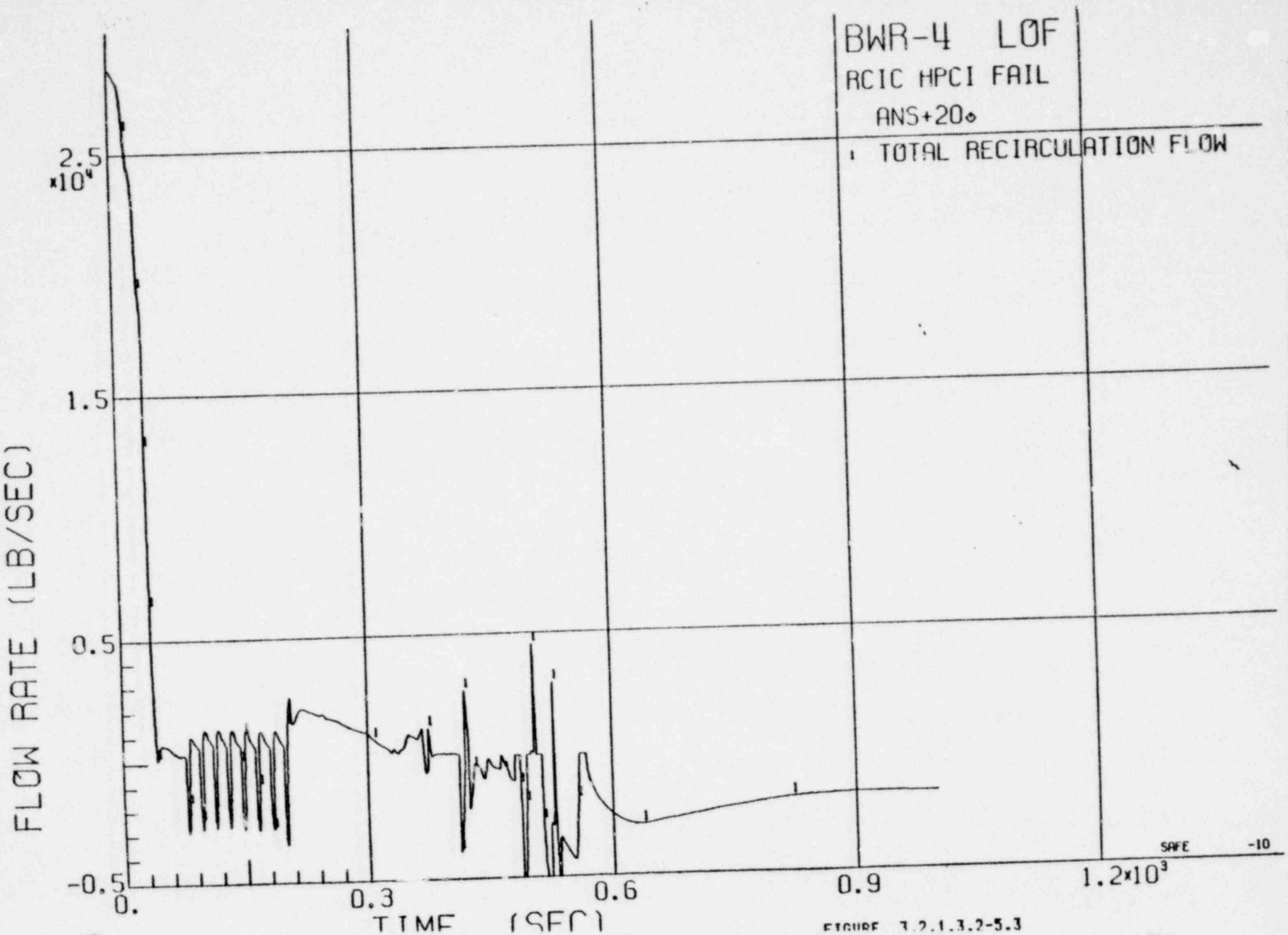
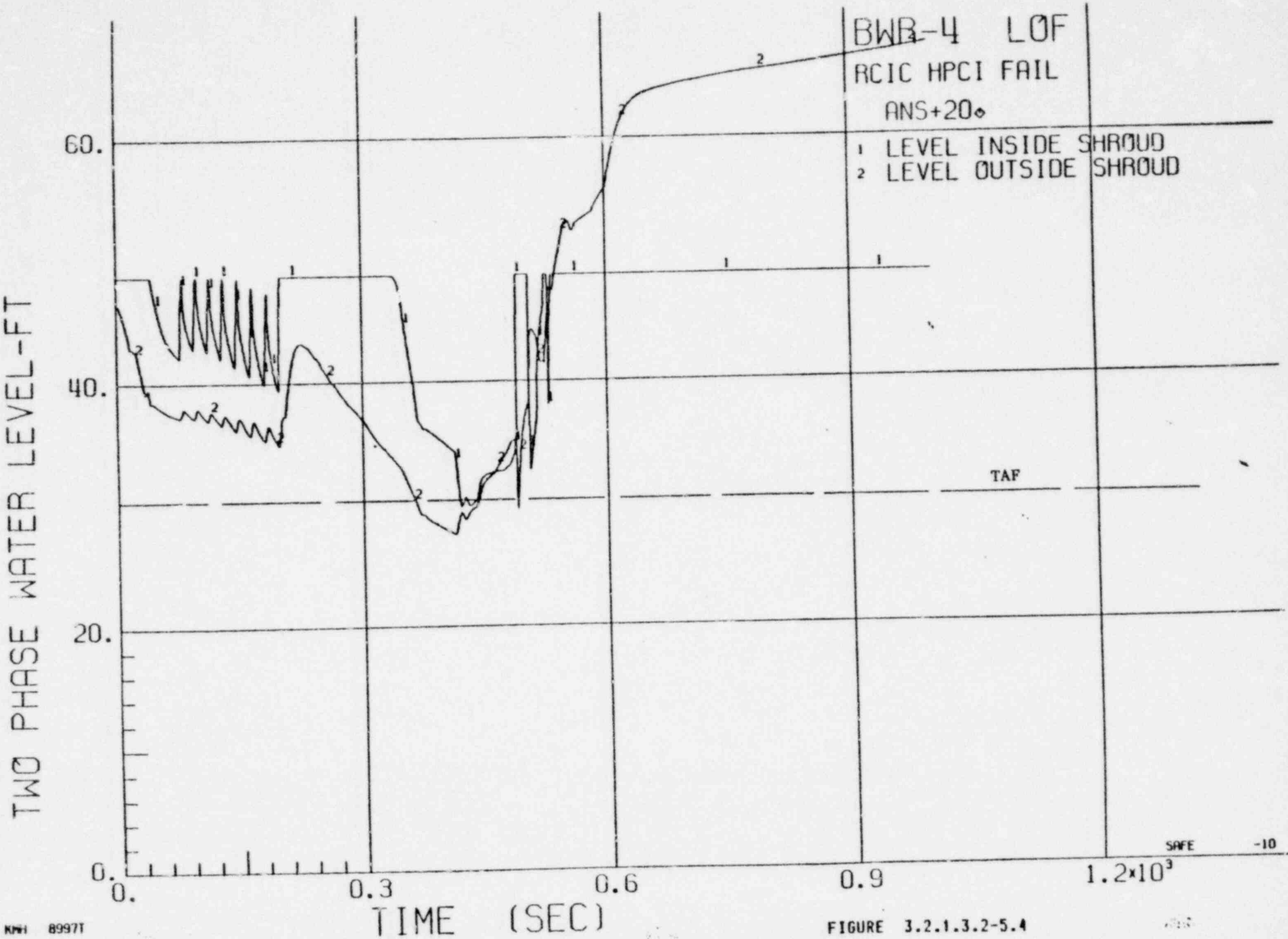


FIGURE 3.2.1.3.2-5.3



KMH 8997T

FIGURE 3.2.1.3.2-5.4

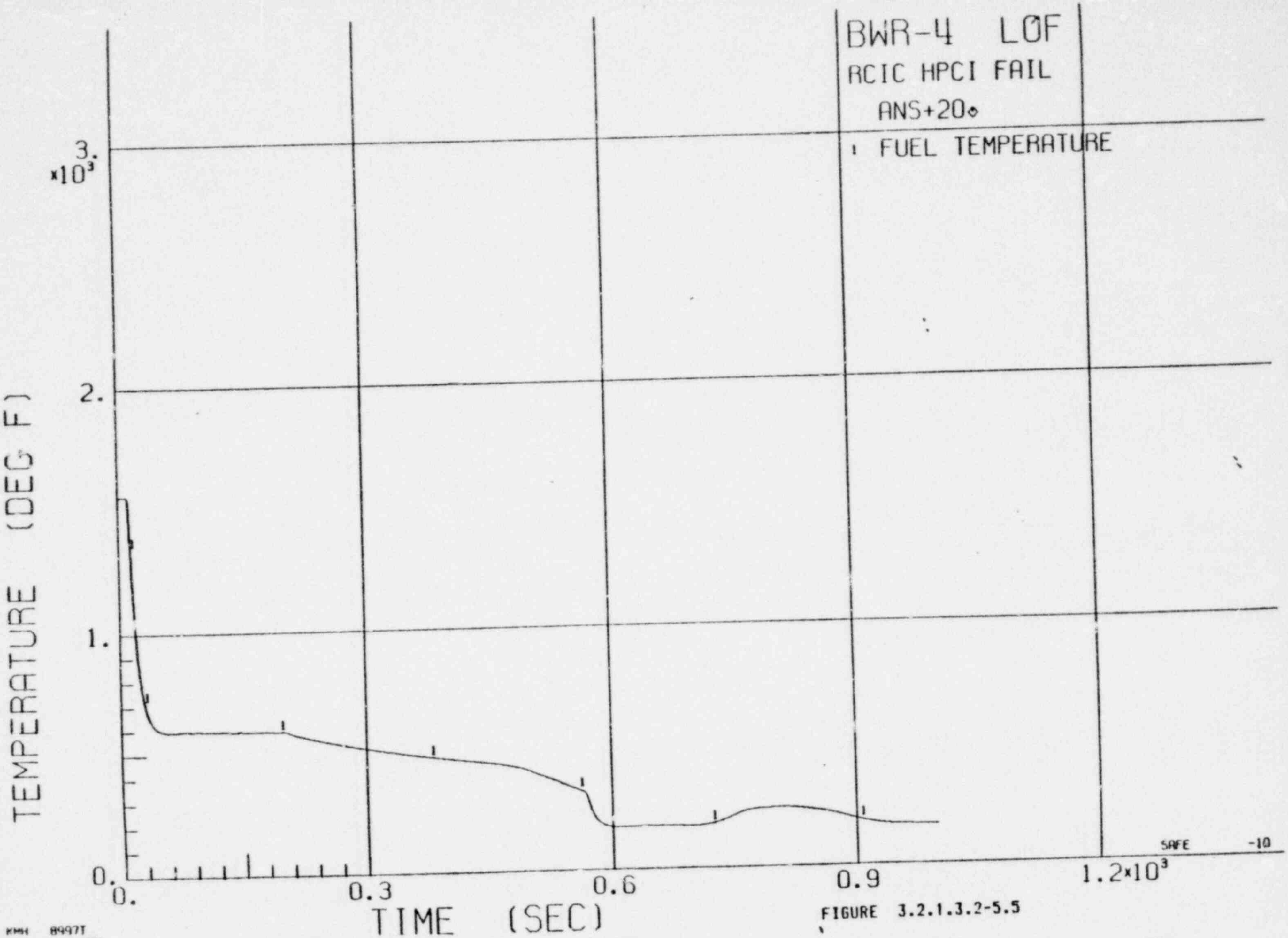


FIGURE 3.2.1.3.2-5.5

KMH 8997T

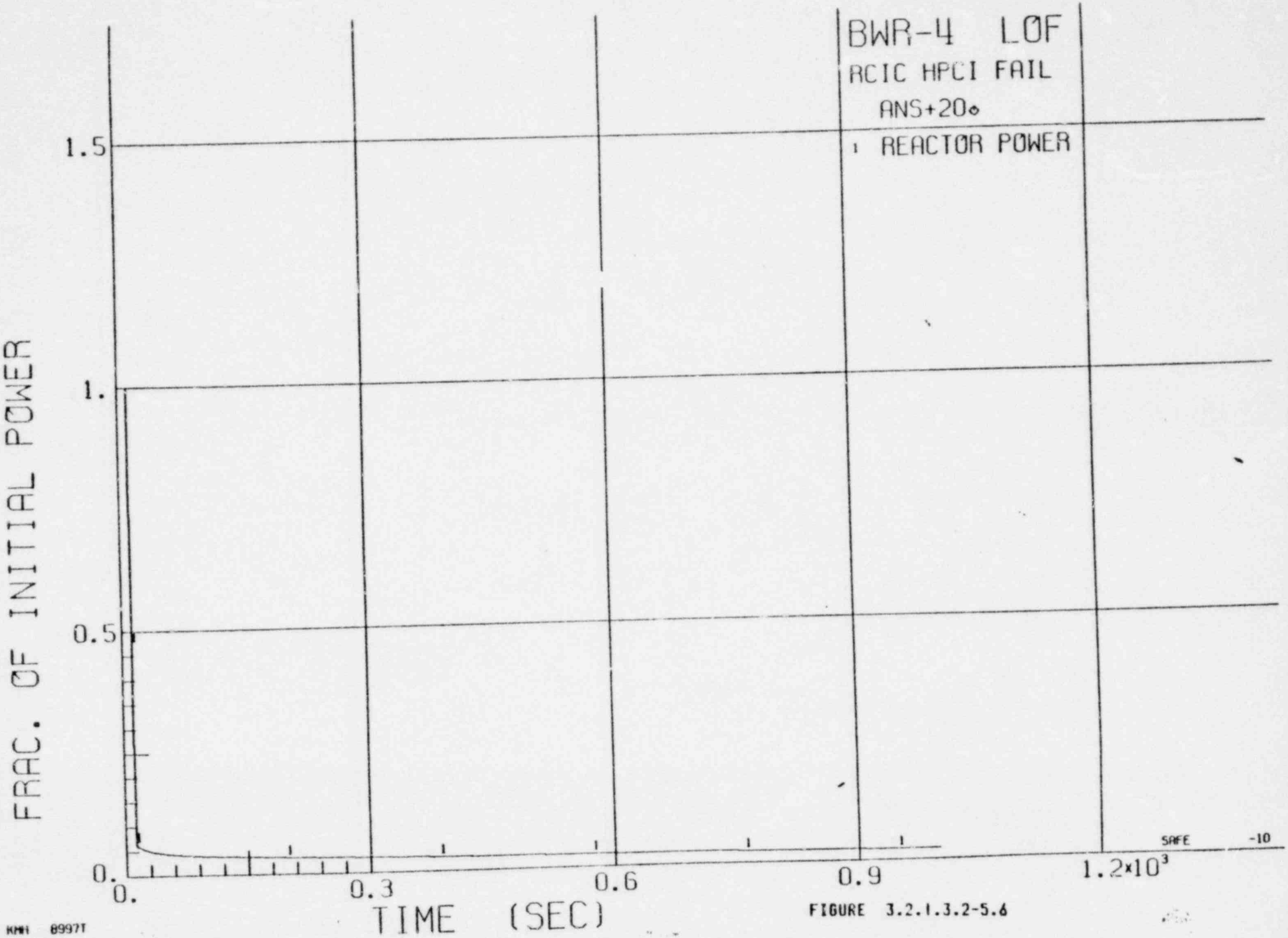


FIGURE 3.2.1.3.2-5.6

KMH 8997T

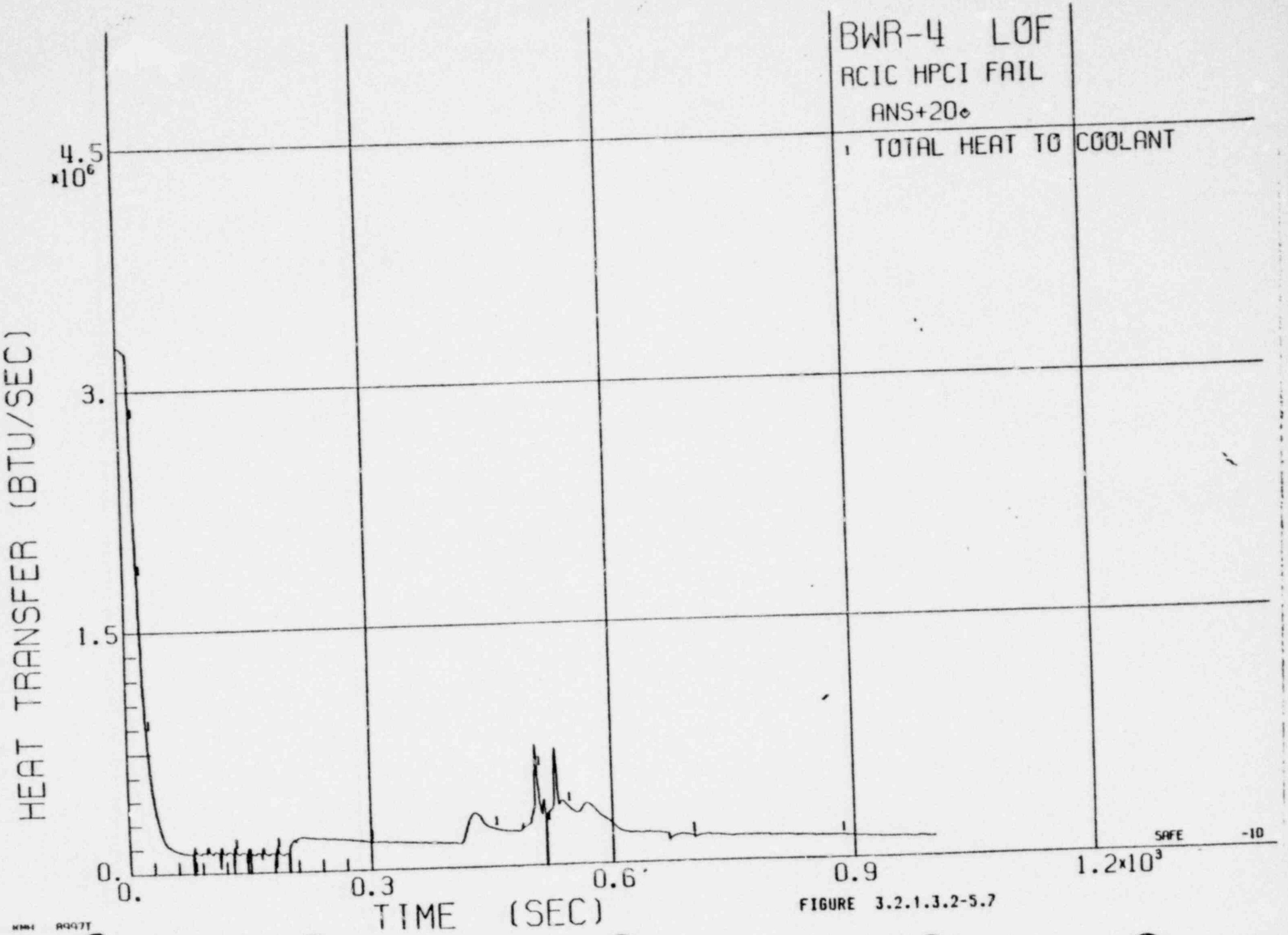
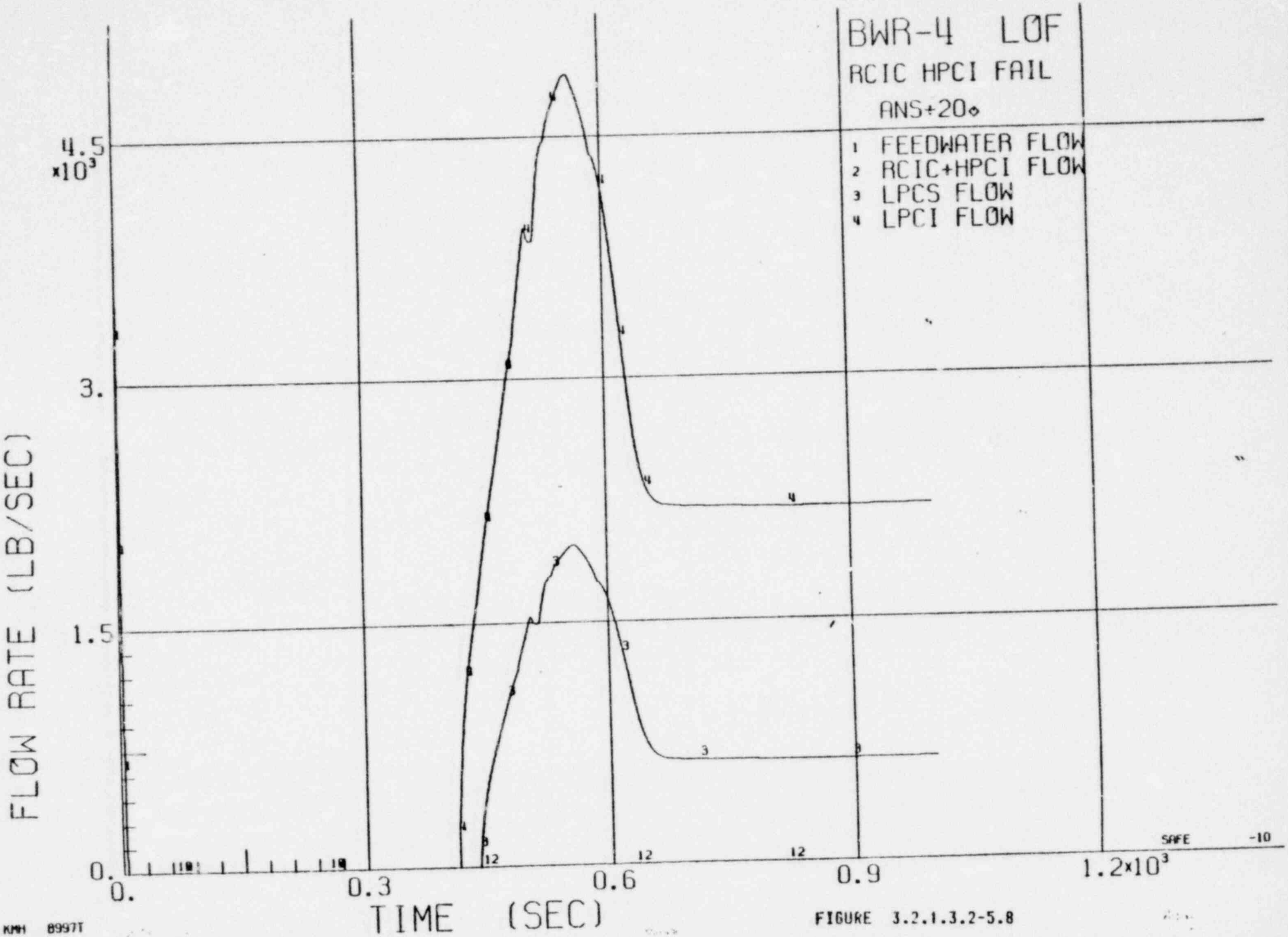


FIGURE 3.2.1.3.2-5.7



KMH 8997T

FIGURE 3.2.1.3.2-5.8

TABLE 3.2.1.1.5-1

BWR/2 LOF

SEQUENCE OF EVENTS

Time (Sec)	Event
	INITIAL VESSEL LEVEL = 43 FT., TAF = 29.44 ft.
0	Trip of all feedwater pumps initiated.
5	Feedwater flow decays to zero.
14	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated.
35	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. All primary system isolation valves including the main steam line isolation valves are initiated to close. The isolation condensers are initiated. The LPCS system is also initiated (partial logic). FWC[illegible] (if applicable) is also initiated.
40	MSIV's are fully closed.
40	Reactor pressure starts to rise due to MSIV closure.
50	Flow from both isolation condensers enter the vessel. Reactor pressure starts to decrease.
100	Reactor water level reaches a plateau and the reactor pressure continues to decrease at a slower rate. The reactor reaches a safe and stable condition.

TABLE 3.2.1.15-2

BWR/3 LOF

SEQUENCE OF EVENTS

Time (Sec)	Event
	Initial vessel level = 42.7 ft., TAF = 30.03 ft.
0	Trip of all feedwater pumps initiated.
5	Feedwater flow decays to zero.
14	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated.
38	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. All primary system isolation valves including the main steam line isolation valves are initiated to close. The FWCI or HPCI system is initiated. The LPCI/LPCS (partial logic) and ADS (partial logic) systems are also initiated.
43	MSIV's are fully closed, reactor pressure starts to rise due to MSIV closure.
90	Reactor pressure reaches the setpoint of the isolation condenser. The isolation condenser is initiated. Shortly afterwards, reactor pressure reaches the setpoint of the lowest safety/relief valve group. The lowest-set safety/relief valves start to open. Reactor pressure cycles between the opening and reclosing setpoint.
128	The isolation condenser flow and FWCI flow enter the vessel. Levels increase.
180	Decay heat falls below the capacity of IC. Relief valves stop cycling. Pressure slowly decreases due to the operation of IC. Reactor reaches a safe and stable condition.

Table 3.2.1.1.5-3

BWR/4: LOF

SEQUENCE OF EVENTS

Time (Sec)	Event
	Initial vessel level = 46.75 ft., TAF = 30.03 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
15	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated. All primary system isolation valves except the main steam line isolation valves (e.g., RHR shutdown isolation valves, RWCU isolation valves, and containment) are initiated to close. Automatic depressurization permissive.
34.0	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI systems are initiated.
39.0	MSIV's fully closed.
64.0	HPCI and RCIC flow starts to enter the vessel, In shroud level bottoms out 12.6 ft. above TAF and starts to rise.
93	Group 1 safety/relief valves start to open.
97	Group 1 safety/relief valves completely closed.
120	Group 1 safety/relief valves start to open again. Valves continue to cycle on and off on setpoints.
230	Group 1 safety/relief valves cease cycling due to HPCI and RCIC steam extraction.
380	Water level approaches normal water level. The operator takes manual control of HPCI and RCIC to maintain normal water level. Reactor reaches a safe and stable condition.

Table 3.2.1.1.5-4

SEQUENCE OF EVENTS
BWR/5: LOF

time (sec)	Event
	Initial vessel level = 46 ft., TAF = 29.55 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
10.0	NR sensed water level reaches low level scram(level 3). Reactor scram is initiated.
28.0	Wide range sensed water level reaches low low water level (level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI system are initiated.
33.0	MSIV's are fully closed.
55.0	HPCS flow starts to enter the vessel.
58.0	RCIC flow starts to enter the vessel. Due to the steam quenching effect of HPCS and RCTC reactor depressurizes.
70.0	Water level bottoms out 7.5 ft above TAF and starts to rise.
380.0	Water level approaches normal water level. The operator takes manual control of HPCS and RCIC to maintain normal water level. Reactor pressure starts to increase.
1080.	Group 1 safety/relief valves begin to cycle between the opening and reclosing setpoint. Reactor is in a safe and stable condition.

TABLE 3.2.1.1.5-5

BWR/2 LOF + 1 IC

SEQUENCE OF EVENTS

Time (Sec)	Event
	INITIAL VESSEL LEVEL = 43 ft., TAF = 29.44
0	Trip of all feedwater pumps initiated.
5	Feedwater flow decays to zero.
14	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated.
35	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. All primary system isolation valves including the main steam line isolation valves are initiated to close. The isolation condensers are initiated. The LPCS system is also initiated (partial logic). FWCI (if applicable is also initiated.
40	MSIV's are fully closed.
40	Reactor pressure starts to rise due to MSIV closure.
50	Flow from one isolation condenser enters the vessel. Reactor pressure continues to rise and reactor water level continues to decrease.
120	Reactor pressure reaches the setpoint of the lowest relief valve group. Reactor pressure cycles between the opening and reclosing setpoint. Reactor level decreases slowly.
300	Decay heat falls below the capacity of one IC, reactor pressure decreases slowly. Relief valves no longer open. Reactor reaches a safe and stable condition.

TABLE 3.2.1.1.5-6

BWR/3 LOF + SCRV + FWCI + IC

SEQUENCE OF EVENTS

Time (Sec)	Event
	Initial vessel level = 42.7 ft., TAF = 30.03 ft.
0	Trip of all feedwater pumps initiated.
5	Feedwater flow decays to zero.
14	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated.
38	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. All primary system isolation valves including the main steam line isolation valves are initiated to close. The FWCI or HPCI system is initiated. The LPCI/LPCS (partial logic) and ADS (partial logic) systems are also initiated.
43	MSIV's are fully closed, reactor pressure starts to rise due to MSIV closure.
90	Reactor pressure reaches the setpoint of the isolation condenser. The emergency condenser is initiated. Shortly afterwards, reactor pressure reaches the set point of the lowest safety/relief valve group. The lowest-set safety/relief valves starts to open. Reactor pressure peaks and then starts to decrease.
95	Reactor pressure drops below the reclosure set point of the lowest-set safety/relief valves. All valves, are closed with the exception of one, which is stuck open. As a result, reactor pressure continues to decrease. Reactor water level also slowly decreases.
128	The isolation condenser and FWCI start to enter the vessel. Level increases.Reactor depressurizes.
280	Vessel level is recovered to normal range. Operator manually trips FWCI and IC. Reactor reaches a safe and stable condition.

Table 3.2.1.1.5-7

BWR/4: LOF + SORV, HPCI/RCIC ON

SEQUENCE OF EVENTS

Time (Sec)	Event
	Initial vessel level - 46,75 ft,, TAF - 30.03 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
15.	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated. Most are initiated to close. Automatic depressurization permissive.
34.0	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI systems are initiated.
39.0	MSIV's and all other isolation valves fully closed.
64.0	HPCI and RCIC flow start to enter the vessel. In-shroud level bottoms out 12.6 ft. above TAF and begins to rise.
93.0	Group 1 safety/relief valves start to open.
97.0	All Group 1 safety relief valves except one close on setpoint. Operator should try to manually close the valve. The reactor will continue to depressurize through the stuck open S/R valve.
380.0	Water level approaches normal water level. The operator takes manual contro of HPCI and RCIC to maintain normal water level.
1800.	Operator initiates steam condensing mode if available. Operator can also choose to de-isolate and use the main condenser as a sink. When the reactor pressure reaches 150 psia the operator can switch RHR to shutdown cooling mode. (Not simulated).

Table 3.2.1.1.5-8

SEQUENCE OF EVENTS
BWR/5; LOF + SORV, WITH HPCS/RCIC

time (sec)	Event
	Initial vessel level = 46 ft., TAF = 29.55 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
10.0	NR sensed water level reaches low level scram(level 3). Reactor scram is initiated.
28.0	Wide range sensed water level reaches low low water level (level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI system are initiated.
33.0	MSIV's are fully closed.
55.0	HPCS flow starts to enter the vessel.
58.0	RCIC flow starts to enter the vessel, due to the steam quenching effect of HPCS and RCIC reactor depressurizes.
70.0	Water level bottoms out 7.5 ft above TAF and begins to rise.
380.0	Water level approaches normal water level. The operator takes manual control of HPCS and RCIC to maintain normal water level. Reactor pressure starts to increase.
1080	Group 1 safety/relief valves start to open.
1085	Group 1 safety/relief valves except one close on closing setpoint. The reactor will continue to depressurize through the stuck open valve. The operator should try to manually close the valve.
1800.	Operator selects steam condensing mode. The operator can also choose to deisolate and use the main condensor as a heat sink. When the reactor reaches 150 psia the operator can place the RHR in the shutdown cooling mode. (Not simulated)

Table 3.2.1.1.5-9

BWR/4: LOF + SORV, RCIC ONLY

SEQUENCE OF EVENTS

Time (Sec)	Event
	Initial vessel level = 46.75 ft., TAF = 30.03 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
15	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated. Most primary system isolation valves are initiated to close. Automatic depressurization permissive.
34.0	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI systems are initiated.
39.0	MSIV's and all other isolation valves fully closed.
64.0	RCIC flow enters the vessel.
88.0	Group 1 safety relief valves begin to open.
92.0	All group 1 safety relief valves except one close on closing setpoint. The reactor continues to depressurize through the stuck open relief valve. Reactor level decreases slowly.
1800	RCIC flow equals SORV flow. In-shroud level reaches a plateau (4.3 ft. above TAF). Reactor pressure is approximately 340 psia and decreasing slowly.
1900	Operator opens a relief valve to speed up the depressurization.
2480	LPCS flow enters the reactor. In-shroud level bottoms out (2.4 ft. above TAF), reactor is quickly reflooded.

Table 3.2.1.1.5-10

SEQUENCE OF EVENTS
BWR/5: LOF → SORV, RCIC ONLY

time (sec)	Event
	Initial vessel level = 46 ft., TAF = 29.55 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
10.0	NR sensed water level reaches low level scram(level 3). Reactor scram is initiated.
28.0	Wide range sensed water level reaches low low water level (level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI system are initiated.
33.0	MSIV's are fully closed.
55.0	No HPCS flow enters the vessel.
58.0	RCIC flow enters the vessel.
70.0	Group 1 safety/relief valves begin to open.
75.0	All Group 1 safety/relief valves except one close on closing setpoint. The reactor continues to depressurize through the open valve. Reactor level decreases slowly.
~1800	RCIC flow equals SORV flow. In-shroud levels reaches a plateau (~2.0 ft above TAF). Reactor pressure is approximately 280 psia. The operator can open a SRV to depressurize the reactor to enable the injection of low pressure ECCS systems (not simulated).
1900-3200	Reactor pressure slowly decreases, water level slowly increases. (The injection of low pressure ECCS is not simulated.)

TABLE 3.2.1.1.5-11

BWR/2 LOF, NO SORV, NO FWCI/IC

SEQUENCE OF EVENTS

Time (Sec)	Event
	INITIAL VESSEL LEVEL = 43 FT., TAF = 29.44 ft.
0	Trip of all feedwater pumps initiated.
5	Feedwater flow decays to zero.
14	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated.
35	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. All primary system isolation valves including the main steam line isolation valves are initiated to close. The isolation-- condensers are initiated. The LPCS system is also initiated (partial logic). FWCI (if applicable) is also initiated.
40	MSIV's are fully closed.
40	Reactor pressure starts to rise due to MSIV closure. No flow from the isolation condenser enters the vessel. Reactor pressure continues to rise and reactor water level continues to decrease.
84	Reactor pressure reaches the setpoint of the lowest relief valve group. Reactor pressure cycles between the opening and reclosing setpoint. Reactor level decreases.
165	Reactor water level decreases to about 5 ft. above TAF, the operator manually actuates ADS in order to depressurize the reactor in time to allow LPCS operation.
540	Reactor pressure decreases to 265 psig and LPCS flow enters the reactor. Reactor is quickly reflooded.

TABLE 3.2.1.1.5-12

BWR/3 LOF, NO SORV, NO FWCI/IC
SEQUENCE OF EVENTS

Time (Sec)	Event
	Initial vessel level = 42.7 ft., TAF = 30.03 ft.
0	Trip of all feedwater pumps initiated.
5	Feedwater flow decays to zero.
14	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated.
38	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. All primary system isolation valves including the main steam line isolation valves are initiated to close. The FWCI or HPCI system fails to be initiated. The LPCI/LPCS (partial logic) and ADS (partial logic) systems are also initiated.
43	MSIV's are fully closed, reactor pressure starts to rise due to MSIV closure.
60	Reactor level decreases to 5 ft. above TAF. The operator manually actuates ADS valves.
680	Reactor depressurizes to 275 psig, LPCS injects. Reactor is quickly reflooded.

Table 3.2.1.1. 5-13

BWR/4: LOF, NO SORV, NO HPCI/RCIC

SEQUENCE OF EVENTS

Time (Sec)	Event
	Initial vessel level = 46.75 ft., TAF = 30.03 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
15	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated. Most primary system isolation valves are initiated to close. Automatic depressurization permissive.
34.0	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI systems are initiated.
39.0	MSIV's and all other isolation valves fully closed.
64.0	No HPCI and RCIC flow enter the vessel. Reactor pressure increases.
88.0	Group 1 safety relief valves.
100.0	Group 1 safety relief valves cycle between opening and reclosing setpoint. Level decreases.
233.0	Reactor water level decreases to about 5 ft above TAF The operator manually initiates ADS.
444.0	Low pressure ECCS begin to inject water into the vessel as vessel pressure falls below shutoff head. The reactor is quickly reflooded.

Table 3.2.1.1.5-14

SEQUENCE OF EVENTS
BWR/5: LOF + NO SORV, NO HPCS/RCIC

time (sec)	Event
	Initial vessel level = 46 ft., TAF = 29.55 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
10.0	NR sensed water level reaches low level scram (level 3). Reactor scram is initiated.
28.0	Wide range sensed water level reaches low low water level (level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI fail to initiate.
33.0	MSIV's are fully closed.
65.0	Group 1 safety/relief valves being to open. No HPCS or RCIC flow enters the vessel.
70.0	Group 1 safety/relief valves close, reactor pressure cycles between the opening and closing setpoint of Group 1 SRVs.
335.	Reactor water level decreases to about five feet above TAF. The operator manually initiates ADS and low pressure ECCS.
500.	Reactor vessel pressure falls below the shutoff head of the low pressure ECCS and they begin to inject flow into the vessel. Reactor is quickly reflooded.

TABLE 3.2.1.1. 5-15

BWR/2 LOF + SORV, NO FWCI/IC

SEQUENCE OF EVENTS

Time (Sec)	Event
	INITIAL VESSEL LEVEL = 43 FT., TAF = 29.44 FT
0	Trip of all feedwater pumps initiated.
5	Feedwater flow decays to zero.
14	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated.
35	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. All primary system isolation valves including the main steam line isolation valves are initiated to close. The isolation condensers are initiated. The LPCS system is also initiated (partial logic). FWCI (if applicable) is also initiated.
40	MSIV's are fully closed.
40	Reactor pressure starts to rise due to MSIV closure. No flow from the isolation condenser enters the vessel. Reactor pressure continues to rise and reactor water level continues to decrease.
84	Reactor pressure reaches the setpoint of the lowest relief valve group. The lowest-set relief valves start to open. Reactor pressure peaks and starts to decrease.
90	Reactor pressure drops below the reclosure set point of the lowest relief valve group. All relief valves are closed with the exception of one, which is stuck open. As a consequence, reactor pressure continues to decrease and reactor water level also slowly decreases.
220	Reactor water level decreases to about 5 ft above TAF, the operator manually actuates ADS in order to depressurize the reactor in time to allow LPCS operation.
490	Reactor pressure decreases to 265 psig and LPCS flow enters the reactor. Reactor is quickly reflooded.

TABLE 3.2.1.1.5-16

BWR/3 LOF + SORV, NO FWCI/IC

SEQUENCE OF EVENTS

Time (Sec)	Event
	Initial vessel level = 42.7 ft., TAF = 30.03 ft.
0	Trip of all feedwater pumps initiated.
5	Feedwater flow decays to zero.
14	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated.
38	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. All primary system isolation valves including the main steam line isolation valves are initiated to close. The FWCI or HPCI system is initiated. The LPCI/LPCS (partial logic) and ADS (partial logic) systems are also initiated.
43	MSIV's are fully closed, reactor pressure starts to rise due to MSIV closure.
60	Reactor water level decreases to 5 ft above TAF. Operator waits 120 seconds to actuate ADS valves.
90	Reactor pressure reaches the set point of the emergency condenser. The emergency condenser is initiated. Shortly afterwards, reactor pressure reaches the set point of the lowest safety/relief valve group. The lowest-set safety/relief valves starts to open. Reactor pressure peaks and then starts to decrease.
95	Reactor pressure drops below the reclosure setpoint of the lowest-set safety/relief valves. All valves are closed with the exception of one which is stuck open. As a result, reactor pressure continues to decrease. Reactor water level also slowly decreases.
126	No flow from FWCI (if applicable) and IC enter the vessel.
180	The operator manually initiates ADS.
470	Reactor pressure decreases to 265 psig and LPCI/LPCS flow enters the reactor. Reactor is quickly reflooded.

Table 3.2.1.1.5-17

BWR/4: LOF + SORV + NO HPCI/RICI

SEQUENCE OF EVENTS

Time (Sec)	Event
	Initial vessel level - 46.75 ft., TAF - 30.03 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero
15	NR sensed water level reaches low level scram (Level 3). Reactor scram is initiated. Most are initiated to close. Automatic depressurization permissive.
34.0	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI systems are initiated.
39.0	MSIV's and all other isolation valves fully closed.
64.0	No HPCI and RCIC flow enter the vessel. Reactor pressure increases.
88.0	Group 1 safety relief valves start to open.
92.0	All group 1 safety relief valves except one close on closing setpoints. The reactor continues to depressurize through the stuck open valve.
270.0	Reactor water level decreases to about 5 ft. above TAF. The operator manually initiates ADS.
460.0	Low pressure ECCS begin to inject water into the vessel as vessel pressure falls below shutoff head. The reactor is quickly reflooded.

Table 3.2.1.1.5-18

SEQUENCE OF EVENTS
BWR/5: LOF + SORV, NO HPCS/RCIC

time (sec)	Event
	Initial vessel level = 46 ft., TAF = 29.55 ft.
0.0	Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
10.0	NR sensed water level reaches low level scram (level 3). Reactor scram is initiated.
28.0	Wide range sensed water level reaches low low water level (level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI fail to initiate.
33.0	MSIV's are fully closed.
65.0	Group 1 safety/relief valves being to open. No HPCS or RCIC flow enter the vessel.
70.0	Group 1 safety/relief valves close except one valve which remains stuck open. The reactor depressurizes through the stuck open valve.
300.	Reactor water level decreases to about five feet above TAF. The operator manually initiates ADS and low pressure ECCS.
425.	Reactor vessel pressure falls below the chutoff head of the low pressure ECCS and they begin to inject flow into the vessel. Reactor is quickly reflooded.

Table 3.2.1.3-1 - Comparison of Initial LOF Transient Predicted by REDY and SAFE

PARAMETER	REDY	SAFE	COMMENTS
Feedwater 5 seconds coastdown	Yes	Yes	
Core inlet enthalpy increase	Yes	Yes	
Recirc. runback	Yes	No	
Reactivity feedback	Yes, thermal power decrease	No, constant thermal power until scram	SAFE has no neutronics model
Mainsteam Flow	Dynamically calculated	Nearly constant until MSIV's close	SAFE uses simple main steam flow model
Dome Pressure	Decreasing, due to thermal power decrease	Increasing, due to constant thermal power and increasing inlet enthalpy	
L3 Scram	Yes, scram at lower power	Yes, scram at initial power	
Decay Heat	78 ANS Standard	78 ANS Standard	
L2 Trip	Yes	Yes	
MSIV 5 Seconds closure	Yes	Yes	
RPT	Yes	Yes	
Natural Recirculation Model	Yes	Yes	

Table 3.2.1.3-2

Summary of LOF Analysis For Off-Rated Reactor Conditions

Point	% Thermal Power (NBR)/% Core Flow	Timing for Manual ADS at 5ft Above TAF (sec)	Minimum In-Shroud Level (ft)
O	104.2/100	234	30.6
A	79.2/64.9	673.0	32.0
B	57.0/37.7	1095.0	32.1
C	54.5/37.9	1121.0	32.6
D	44.0/38.5	1488	32.2
E	33.1/38.7	2083	32.4
F	21.9/38.1	3230	36.2