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ECCS ANALYSIS FOR SUSQUEHANNA UNIT 1 AT FULL POWER AND 87% FLOW USING THE ENC EXEM BWR EVALUATION MODEL

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1.0 INTRODUCTION AND SUMMARY

ENC provided ECCS analysis for Susquehanna Unit 1 in the report XN-NF-84-119. The analysis in that report was performed to support the 100% power/100% flow operating point. Since that report was submitted, Pennsylvania Power and Light Company has requested that ENC provide an ECCS analysis to support the 100% power/87% flow operating point which is the lowest flow at which full power operation is permitted. This supplement presents the results of the requested analysis. The analysis was performed with the generically approved Exxon Nuclear Company EXEM Evaluation Model[1,2] according to Appendix K of 10 CFR 50[3], and the results comply with the USNRC 10 CFR 50.46 criteria.

The purpose of this analysis was to determine if that the MAPLHGR limit established at 100% flow in XN-NF-84-119 is applicable to the range of flow conditions currently allowed by the Susquehanna Unit 1 power/flow operating domain. To assure the range of allowable flow conditions are covered, this supplement reports the results of an analysis at the 100% power (102% as required by Appendix K)/87% flow condition. Other conditions for this analysis were the same as for the 100% power/100% flow case described in the base document and are consistent with the limiting break that is applicable to the Susquehanna plant established in Reference 4. An assumed minimum critical power ratio (MCPR) operating limit of 1.24 was used to establish the maximum power in the limiting assembly for both analyses.

The results of this analysis confirm that the 100% power/100% flow analysis is bounding and that the MAPLHGR limit established for the

Susquehanna plant in the base document[5] is applicable over the Susquehanna Unit 1 power flow map. The applicable exposure dependent MAPLHGR of the base document is repeated in Figure 1.1. The peak clad temperatures and metal-water reactions versus burnup have been shown to be below the 10 CFR 50.46 criteria of 2200°F and 17% respectively for all exposures.

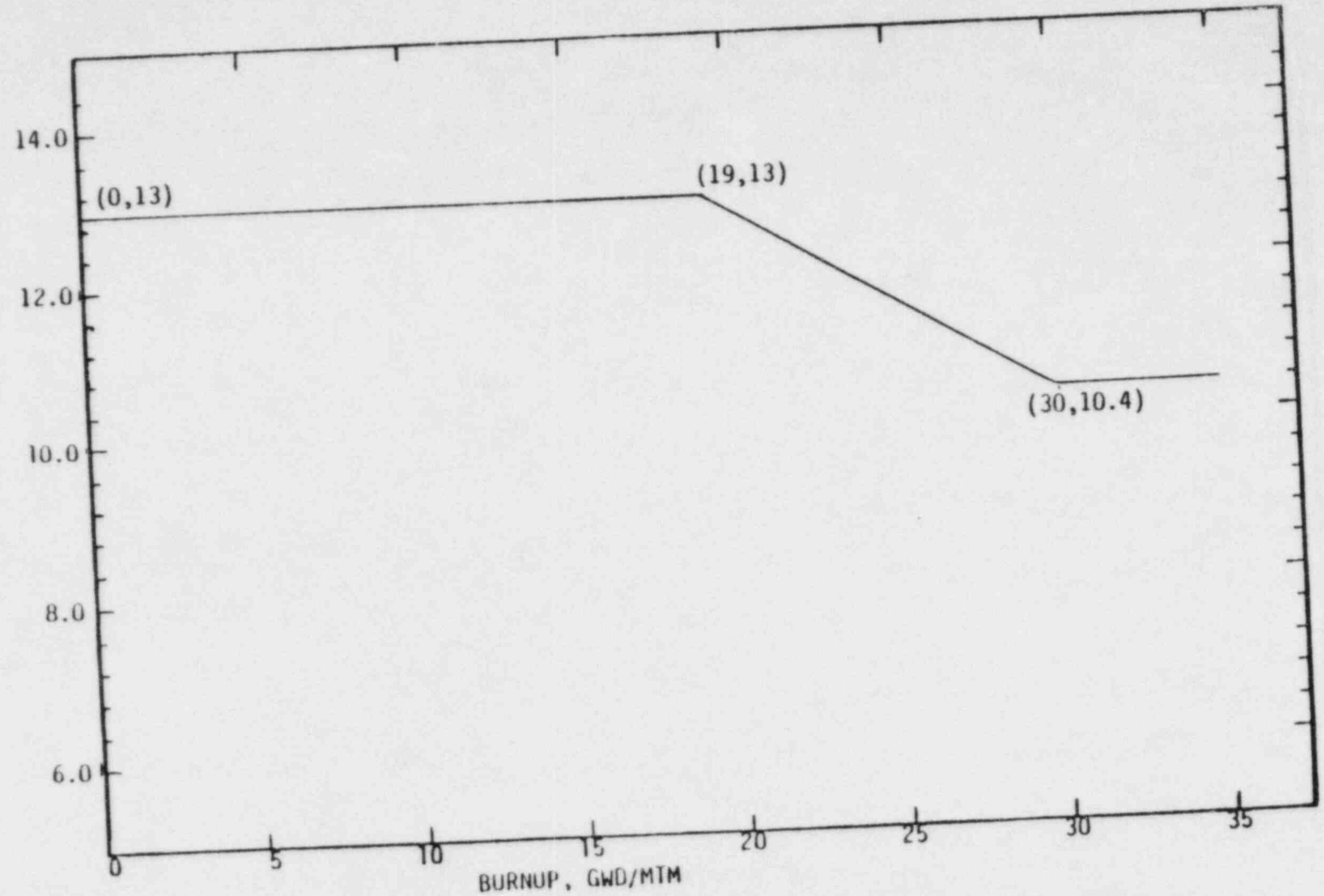

Figure 1.1 Susquehanna Unit 1 MAPLHGR vs. Hot Assembly Burnup Results for ENC 8x8 Reload Fuel

2.0 JET PUMP BWR ECCS EVALUATION MODEL

2.1 LOCA DESCRIPTION

A loss-of-coolant accident (LOCA) is defined as a hypothetical rupture of the reactor coolant system piping, up to and including the double-ended rupture of the largest pipe in the reactor coolant system or of any line connected to that system up to the first closed valve. In the unlikely event a LOCA occurs in the Susquehanna plant, the reactor system coolant inventory loss would result in a high containment drywell pressure and a decreasing coolant mixture level in the reactor vessel. The earlier of these two events provides a safety injection signal which brings coolant injection systems into operation to limit the accident consequences.

During the early phase of the LOCA depressurization transient, core cooling is provided by the existing coolant inventory. In the latter stage of system depressurization and after depressurization has been achieved, the core spray provides core cooling and supplies liquid to refill the lower portion of the reactor vessel and reflood the core. The reflood process provides sufficient heat removal to terminate the core temperature transient.

2.2 EXEM APPLICATION TO SUSQUEHANNA UNIT 1

The ENC EXEM codes were used for the LOCA-ECCS analysis for Susquehanna Unit 1. EXEM is comprised of the RODEX2, RELAX, FLEX, and HUXY/BULGEX computer codes.

The RELAX code is used to calculate the reactor system behavior during the initial portion (up to time of rated core spray) of the reactor system depressurization transient. RELAX predicts mass distribution, core and system thermal hydraulics, and break flow rates. The blowdown calculation also provides reactor coolant system conditions at the time of rated low pressure core spray for the initialization of the refill/reflood transient calculation using the FLEX code. The reactor core is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback and decay heating required by Appendix K of 10 CFR (U.S. Code of Federal Regulations) Part 50. For the blowdown calculation, the reactor coolant system is nodalized into control volumes representing reasonably homogeneous regions interconnected by junctions shown in Figure 2.1. Reactor system data for the Susquehanna LOCA system analysis is summarized in Table 2.1. The data of Table 2.1 for the previous 100% flow analysis differs only in the active core, total reactor and recirculation flows.

For the maximum power fuel assembly, a separate RELAX/HOT CHANNEL calculation is used to calculate the cladding-to-coolant heat transfer coefficients, and the coolant thermodynamic properties. The HOT CHANNEL analysis uses time dependent plenum boundary conditions from the RELAX blowdown calculation. The calculated results from the HOT CHANNEL calculation are used as input data for subsequent hot assembly heatup calculation that encompass the LOCA transient from its initiation through peak clad temperature (PCT). The RELAX/HOT CHANNEL nodalization is shown in Figure 2.2.

The initial stored energies for the RELAX blowdown and RELAX/HOT CHANNEL calculations are determined with the RODEX2 code. RODEX2 also provides inital gap conductance and initial gap dimensions as a function of power for the hot assembly heatup calculation using HUXY/BULGEX. The same conservative power history was used for this analysis as in the original analysis[5] which maximized the stored energy at all exposures.

The FLEX system refill/reflood analysis predicts the latter segment of the reactor coolant system depressurization, lower plenum refill and core reflood. The time of hot-node-reflood is determined by FLEX and is an input quantity for the hot assembly heatup calculation. The FLEX refill/reflood calculation is performed with leakage paths minimized to conservatively bound the first cycle with ENC reload fuel as well as future cycles. The FLEX system refill/reflood nodalization is shown in Figure 2.3. Compared to the 100% core flow analysis, the input for FLEX differs only in the initial conditions provided by the blowdown run.

The HUXY/BULGEX heatup calculation uses: (1) fuel stored energy, gap conductivity and dimensions calculated from RODEX2, (2) time of rated spray, decay power, heat transfer coefficients and coolant thermodynamic properties calculated from RELAX, and (3) time of hot-node-reflood calculated from FLEX. These data are combined to determine the peak clad temperature (PCT) and the cladding oxidation percentage. Bounding fission and actinide product decay heat obtained with end-of-cycle neutronics in the system blowdown calculation assure that the power input to the HUXY/BULGEX heatup calculation is conservative. Conservative heat transfer coefficients and fluid thermodynamic properties for the heatup calculation are assured by using the maximum stored energy over the life of the ENC 8x8 fuel in the HOT CHANNEL calculation for the generation of this information.

2.3 EXEM CODE

Since the completion of the ECCS analysis for 100% core flow, there has been no code c!anges. The EXEM code versions used for the Susquehanna analysis at 87% of rated core flow are specified below:

Code	Version
RELAX	RELXE0104
FLEX	FLEXE0105
HUXY/BULGEX	JAN83
RODEX2	UAUG84

Table 2.1 Susquehanna Unit 1 Reactor System Data
100% Power/87% Flow Conditions

Primary Heat Output, MW	3358.9
Total Reactor Flow Rate, lb/hr	87.00×10^6
Active Core Flow Rate, lb/hr	78.27×10^6
Nominal Reactor System Pressure, (upper plenum) psia	1039
Reactor Core Inlet Enthalpy, Btu/lb	520.7
Recirculation Loop Flow Rate, lb/hr	13.44×10^6
Steam Flow Rate, lb/hr	13.78×10^6
Feedwater Flow Rate, lb/hr	13.74×10^6
Rated Recirculation Pump Head, ft	710
Rated Recirculation Pump Speed, rpm	1670.
Moment of Inertia, lbm-ft^2/rad	15710
Recirculation Suction Pipe I.D., in.	25.34
Recirculation Discharge pipe I.D., in.	25.34

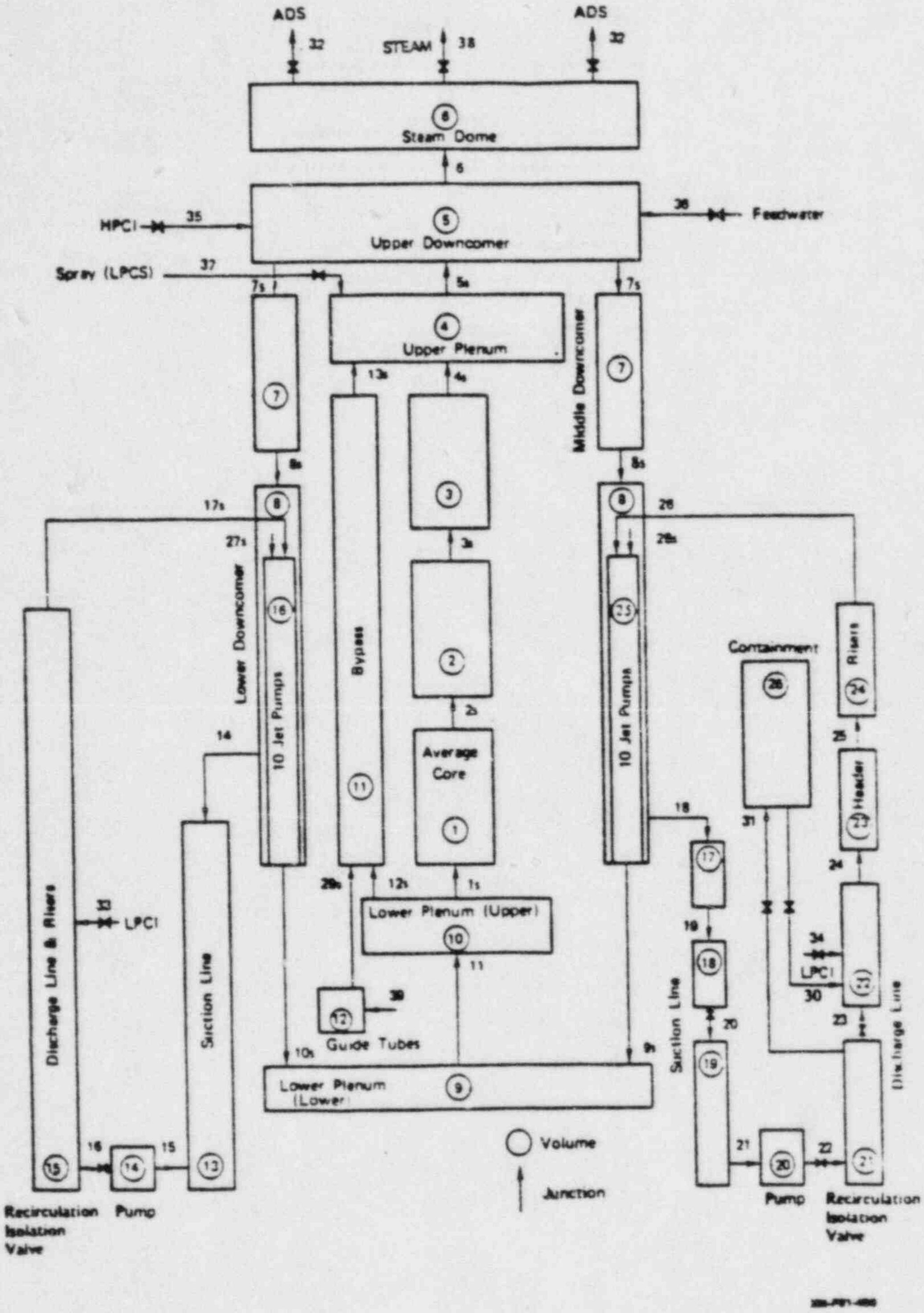

Figure 2.1 System Blowdown Nodalization for Susquehanna Unit 1

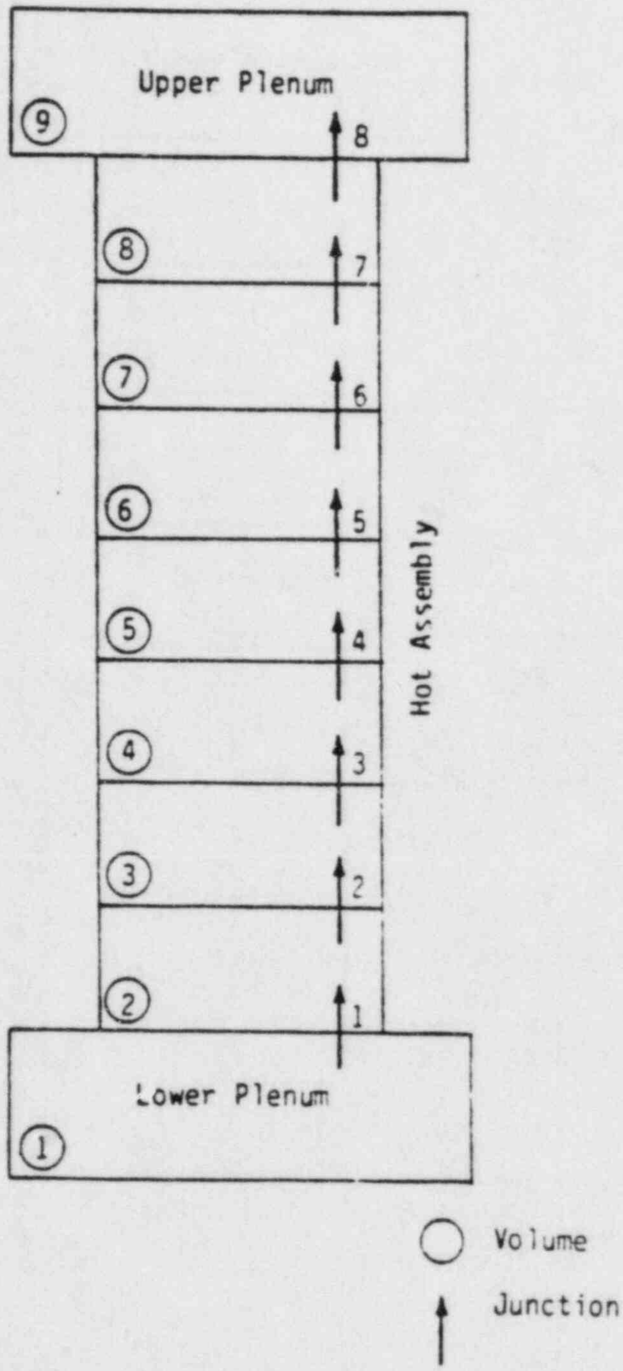

Figure 2.2 Hot Channel Nodalization for Susquehanna Unit 1

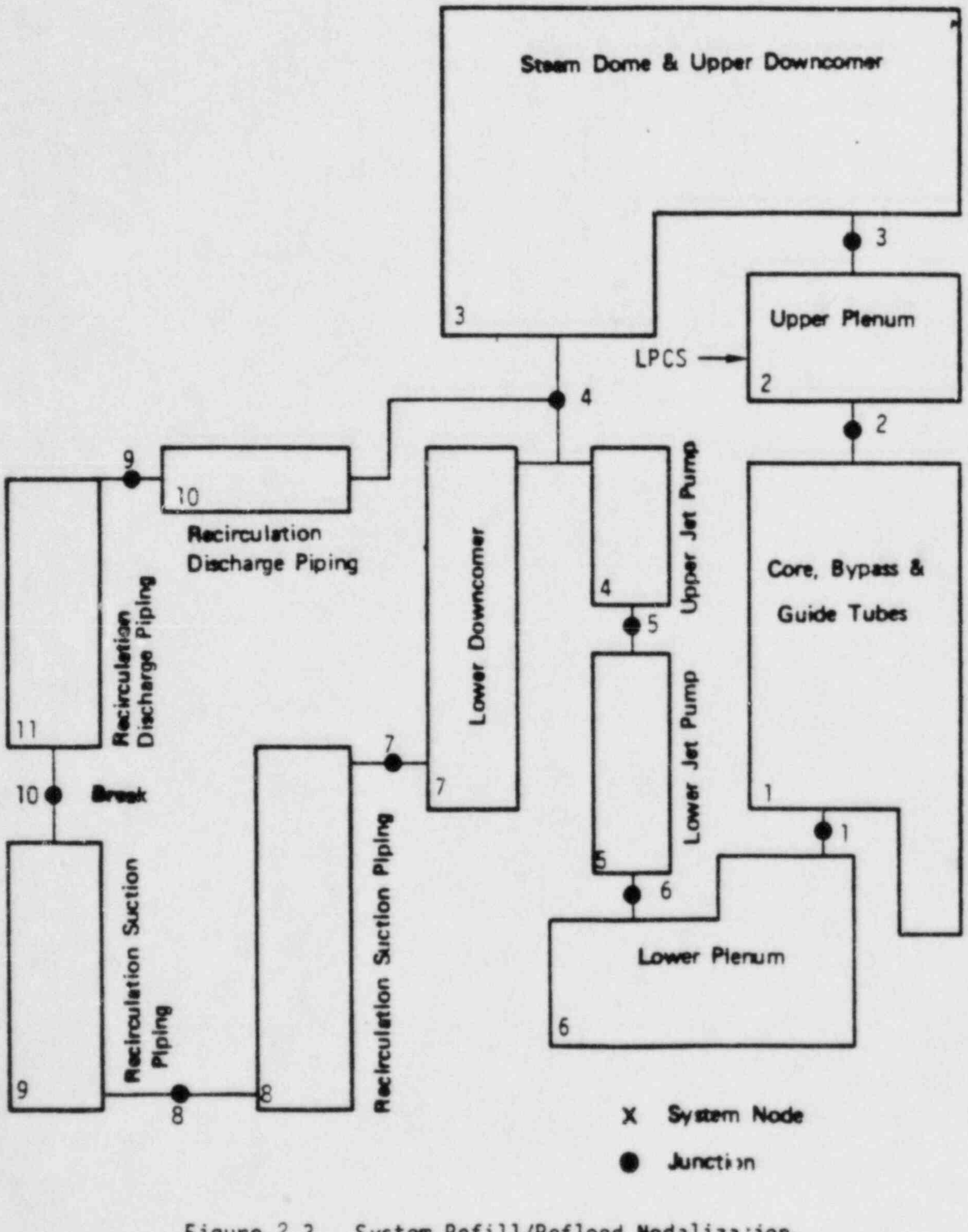

Figure 2.3 System Refill/Reflood Nodaliza:ion for Susquehanna Unit 1

3.0 ANALYSIS RESULTS

3.1 LOCA ANALYSIS

An ECCS analysis of the limiting break has been performed for Susquehanna Unit 1 operating at full power and 87% of rated core flow. The calculated results are summarized herein. The break spectrum calculations for BWR 3 and BWR 4 plants without LPCI loop selection logic have been previously performed and reported(4) on a generic bases. In the generic analysis, the limiting break location was found to be in the recirculation discharge piping. The configuration and size were found to be a double ended guillotine (DEG) with a discharge coefficient of 0.4.

The LOCA system behavior is a function of the system geometry and break size. The LOCA analysis for Susquehanna Unit 1 has been previously analyzed at the full power and 100% of rated core flow point. In the current analysis, operation at full power but with only 87% of rated core flow is considered. This change in core flow results in small changes in system behavior during the LOCA, the most notable of which are later times of rated spray and reflood. At 87% flow, the maximum power assembly has a reduced radial peaking to yield an assumed operating minimum critical power ratio (MCPR) limit of 1.24.

Major event times and results for the limiting break for Susquehanna Unit 1 are shown in Tables 3.1 and 3.2; Figures 3.1 through 3.19 are system blowdown parameters; Figures 3.20 through 3.22 are system refill parameters; Figures 3.23 through 3.25 are hot channel blowdown parameters; and Figure 3.26 shows the transient temperature of the peak clad temperature

(PCT) rod, peak power rod and cannister temperature at the maximum axial power plane for the HUXY/BULGEX calculation.

3.2 MAPLHGR RESULTS

The MAPLHGR results for the Susquehanna Unit 1 reactor is based on a limiting break calculation described in Section 3.1 which is a double-ended guillotine (DEG) with a discharge coefficient of 0.4 in the recirculation discharge piping. MAPLHGR results are obtained using LOCA system analysis boundary conditions but require an additional RELAX/HOT CHANNEL calculation and a single HUXY/BULGEX calculation at a fuel exposure of 19 GWD/MTM.

A bounding HOT CHANNEL calculation has been performed for this MAPLGHR analysis with the same maximized fuel stored energy as in the previous 100% flow case[5]. The hot channel calculation was also performed with a lower hot channel power level in order to be consistent with the hot channel in the 100% flow case[5] in terms of its operating minimum critical power ratio (MCPR) of 1.24. In order to maintain the same MCPR of 1.24 for the 87% flow case, the hot channel's core wide radial peaking factor was decreased from 1.585 to 1.508. With this lower power, the fluid conditions during blowdown in the hot channel did not degrade as would have been anticipated in a reduced flow analysis. The fluid conditions actually improved with the decreased flow and decreased hot channel power.

The fluid conditions provided by the bounding HOT CHANNEL calculation are the heat transfer coefficient, the fluid temperature, and the fluid quality at the plane of interest for the HUXY/BULGEX calculation. These HOT CHANNEL calculated parameters are shown in Figures 3.23 through 3.25. Figure 3.26 is the heatup vs. time plot calculated by the HUXY/BULGEX code.

The HUXY/BULGEX calculated result is a single calculation done at 19 GWD/MTM which was the most limiting burnup in terms of PCT in the original analysis[5]. The result of the above calculation is shown in Table 3.2. As is shown in the table, the PCT for the 87% core flow case is 28^{o}F less than the PCT for the 100% core flow case at 19 GWD/MTM. Since the fuel parameters used in the HUXY/BULGEX calculations for both flows are equal, the difference in PCT is caused by improved fluid conditions during the blowdown for the 87% flow case. This results in a lower stored energy at the beginning of the heatup period for the 87% flow case and hence a lower PCT at the time of reflood.

Since the fuel parameters for the HUXY/BULGEX calculations do not vary from the 100% to 87% flow cases over the life of the fuel, it can be concluded that the PCTs for the 87% flow case will be about 28^{o}F less than the PCTs for the 100% flow case over the life of the fuel. Thus, no further HUXY/BULGEX calculations were needed to conclude that the 100% flow case is the bounding analysis and the 87% flow case meets the 10 CFR 50.46 PCT limit of 2200^{o}F by a margin of 80^{o}F at its most limiting exposure. The margin to the metal-water reaction oxidation limit of 17% of the cladding is again substantial.

Table 3.1 Susquehanna Unit 1 Limiting Break Event Times
100% Power/87% Flow Conditions

Event	Time (sec)
Start	0.00
Initiate Break	0.05
Feedwater Flow Stops	0.55
Steam Flow Stops	5.05
Low Mixture Level (LI)	12.4
Jet-Pumps Uncover	15.4
Recirculation Suction Uncovers	24.9
Lower Plenum Flashes (Quality > 0.)	21.8
HPCI Flow Starts	33.3
LPCS Flow Injection Starts	80.1
Rated Spray Calculated	122.5
Depressurization Ends (Vessel pressure reaches 1 atmosphere)	262.9
Start of Reflood (High density fluid enters core)	286.5
Peak Clad Temperature Reached	292.5

Table 3.2 Susquehanna LOCA Analysis for 100% and 87% Core Flow Results for ENC 8x8 Reload Fuel

Assembly Average Burnup	MAPLHGR (kw/ft)	Local MWR* (%) 100% Flow	Local MWR* (%) 87% Flow	Peak Clad Temperature (^{o}F) 100% Flow	Peak Clad Temperature (^{o}F) 87% Flow
0	13.0	1.9		2074	
5	13.0	2.0		2093	
10	13.0	2.1		2116	
19	13.0	2.3	2.1	2147	2119
25	12.2	1.6		1977	
30	11.3	1.0		1846	
35	10.4	1.2		1852	

MWR*: Metal Water Reaction

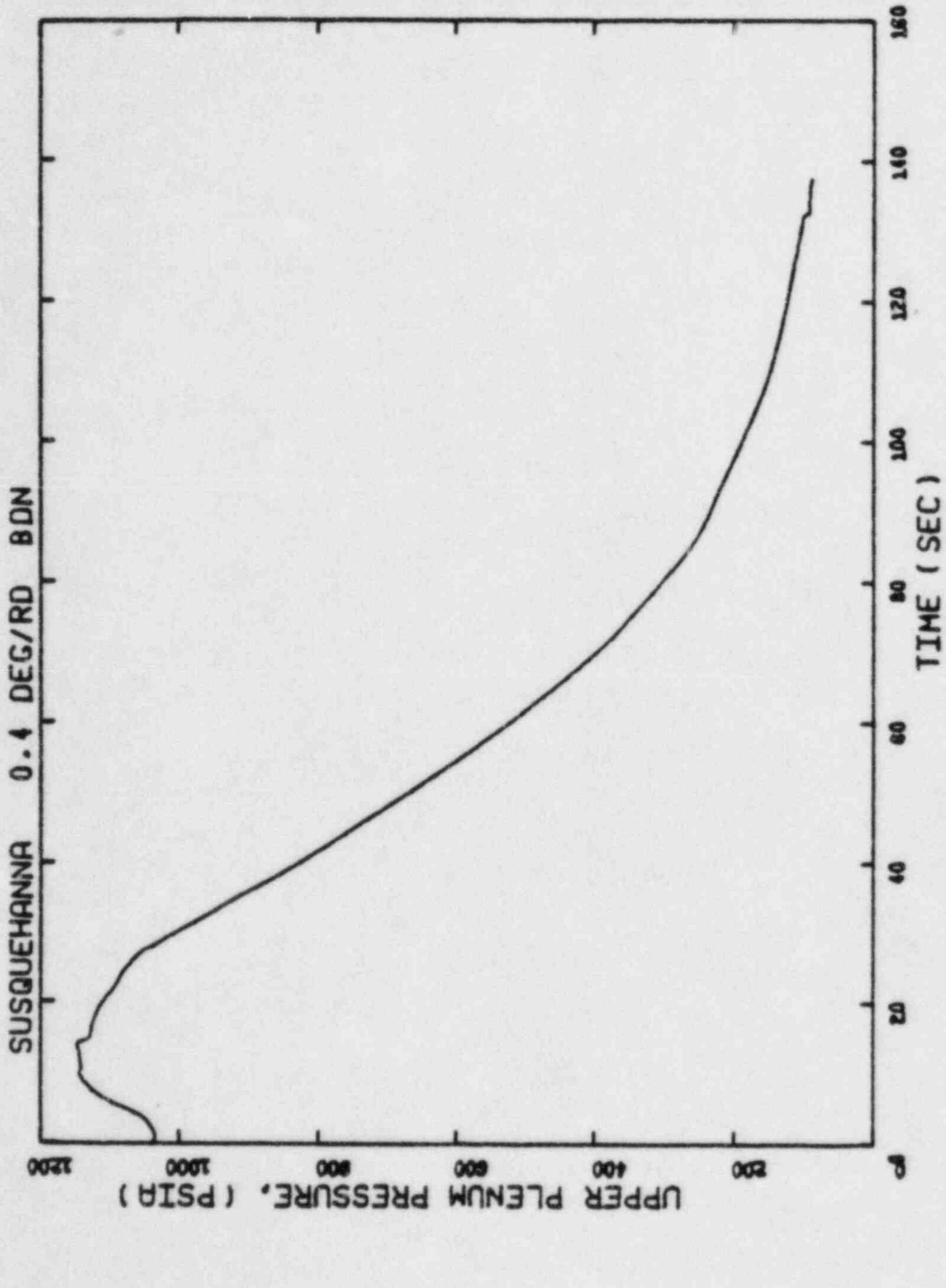

Figure 3.1 Blowdown System Pressure, 0.4 DEG/RD Break

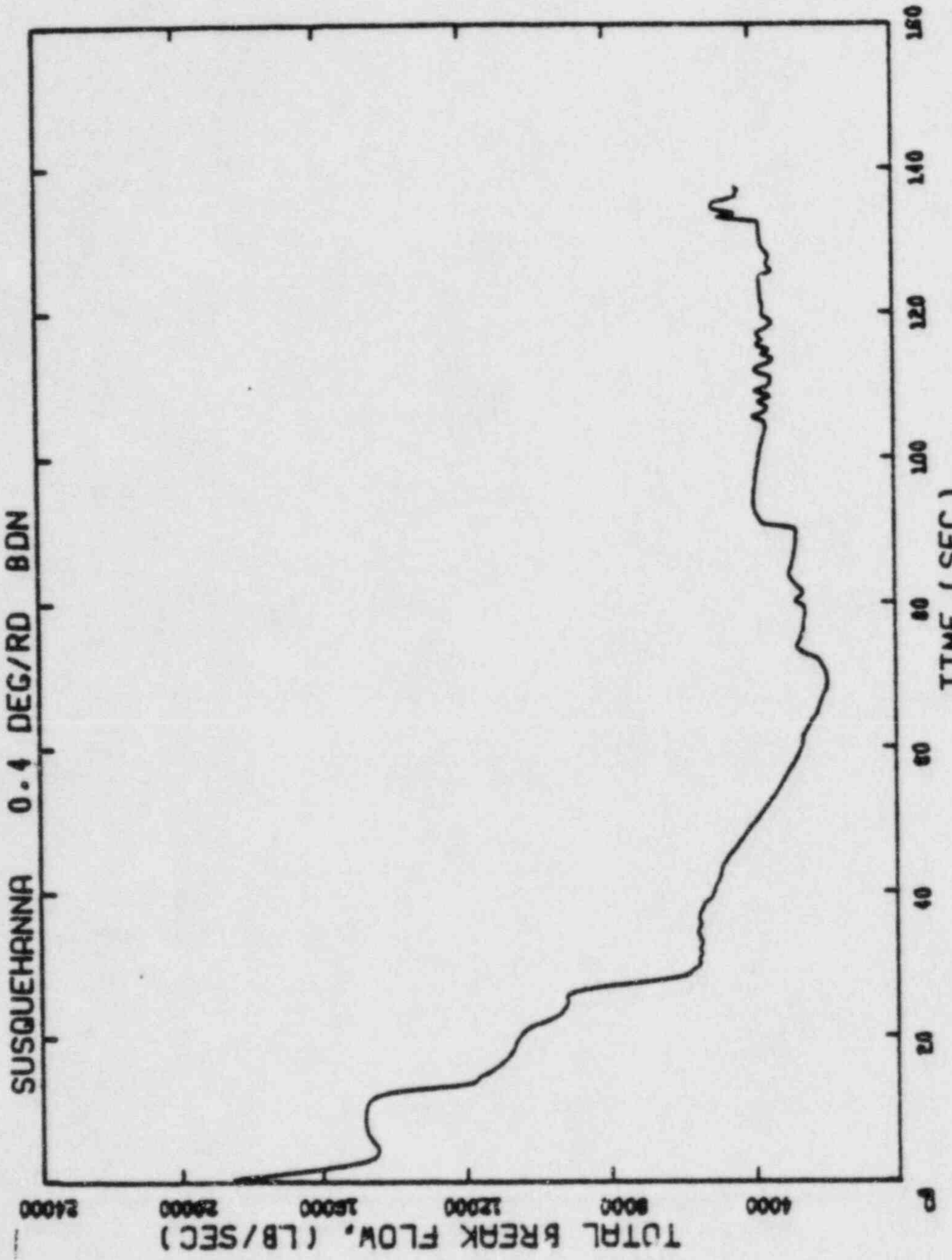

Figure 3.2 Blowdown Total Break Flow,
0.4 DEG/RD Break

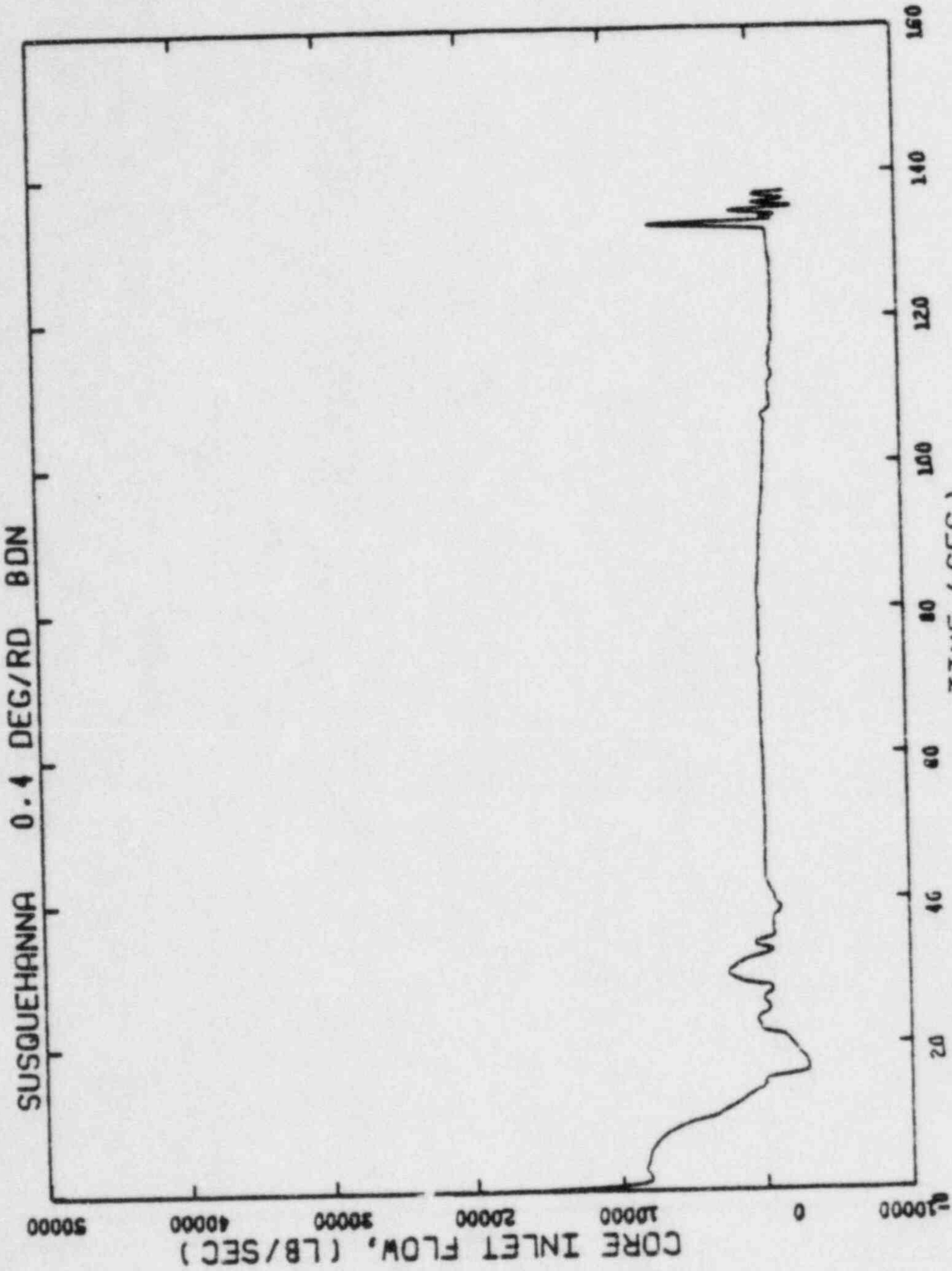

Figure 3.3 Blowdown Average Core Inlet Flow, 0.4 DEG/RD Break

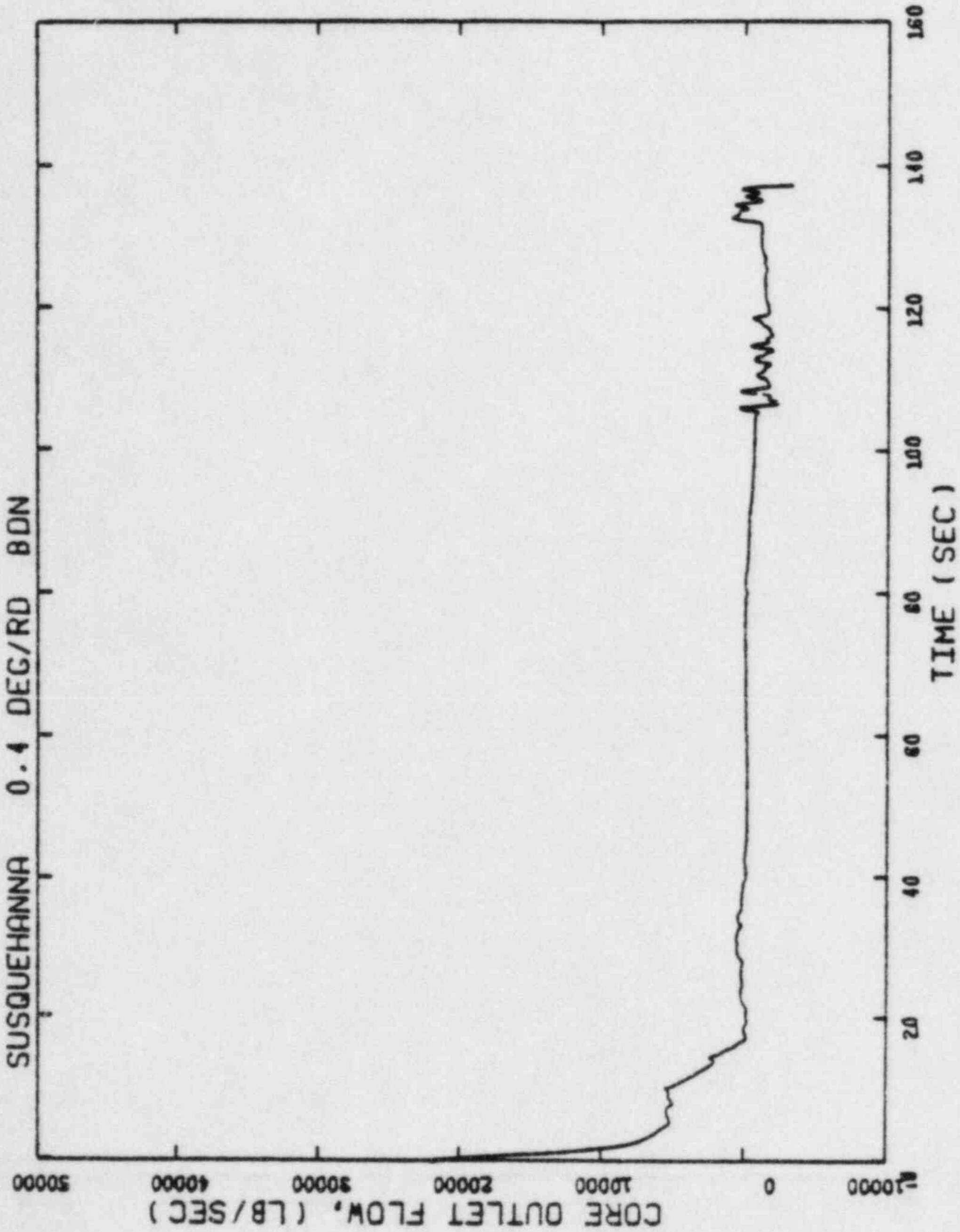

Figure 3.4 Blowdown Average Core Outlet Flow, 0.4 DEG/RD Break

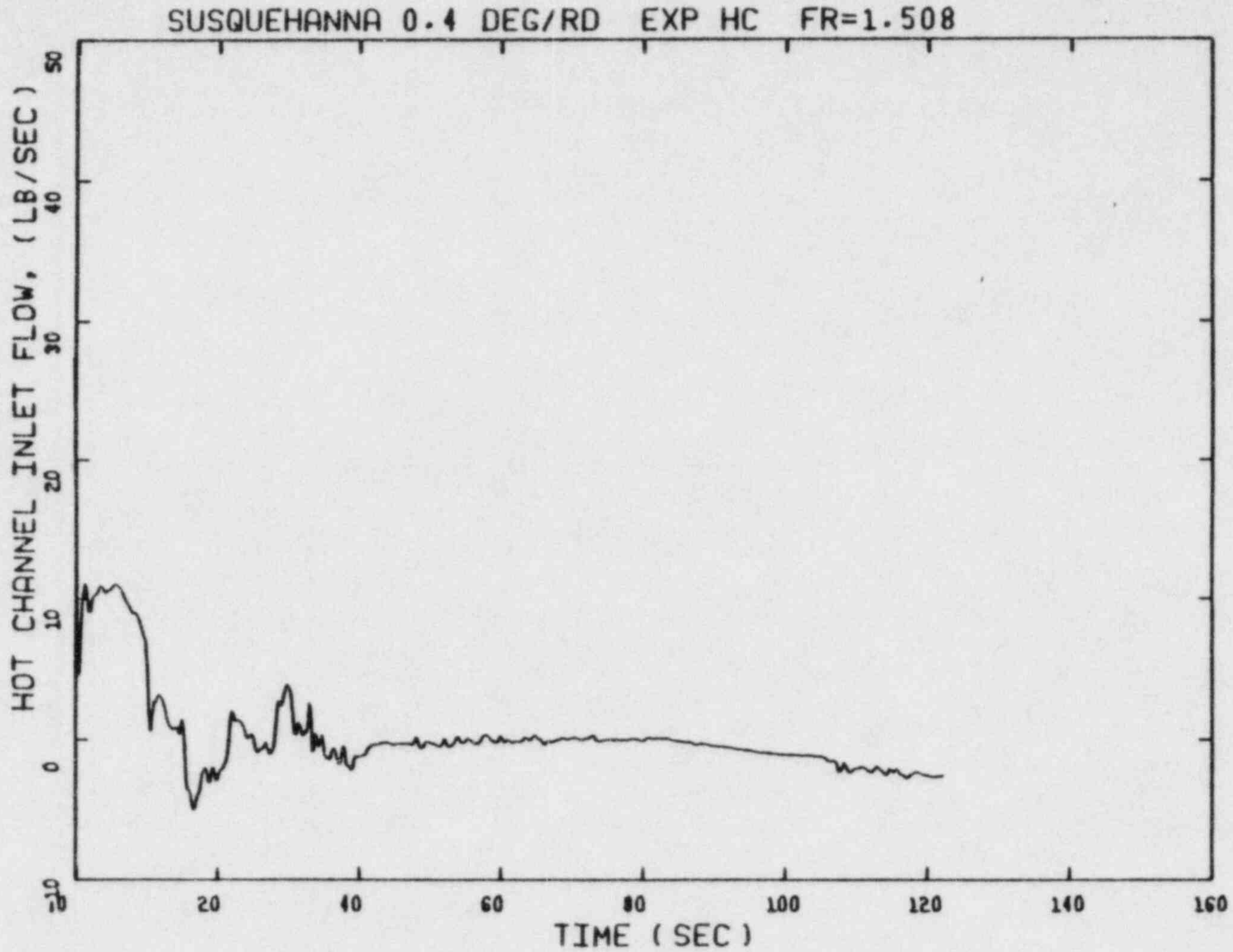

Figure 3.5 Blowdown HOT CHANNEL Inlet Flow,
0.4 DEG/RD Break

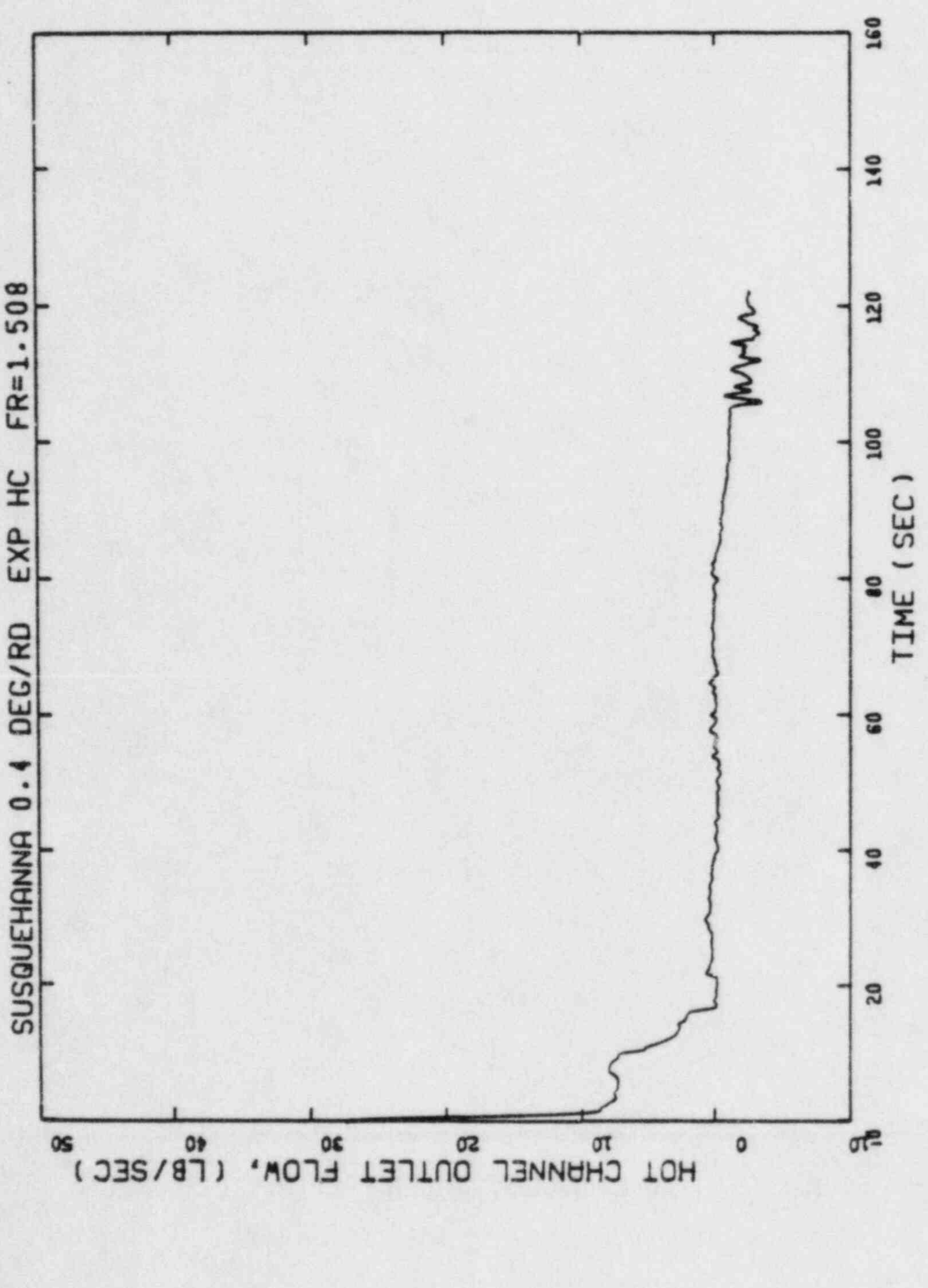

Figure 3.6 Blowdown HOT CHANNEL Outlet Flow, 0.4 DEG/RD Break

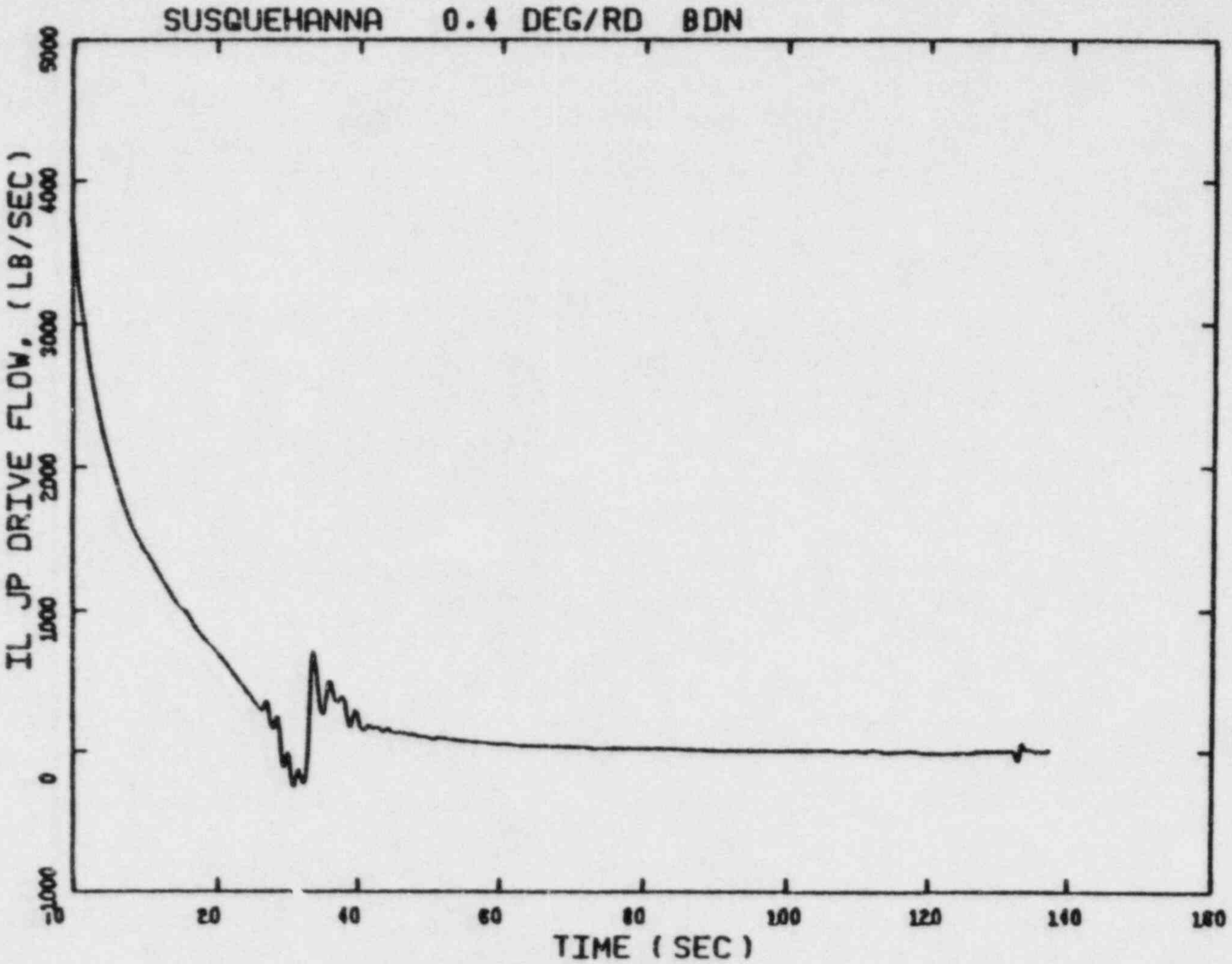

Figure 3.7 Blowdown Intact Loop Jet Pump Drive Flow, 0.4 DEG/RD Break

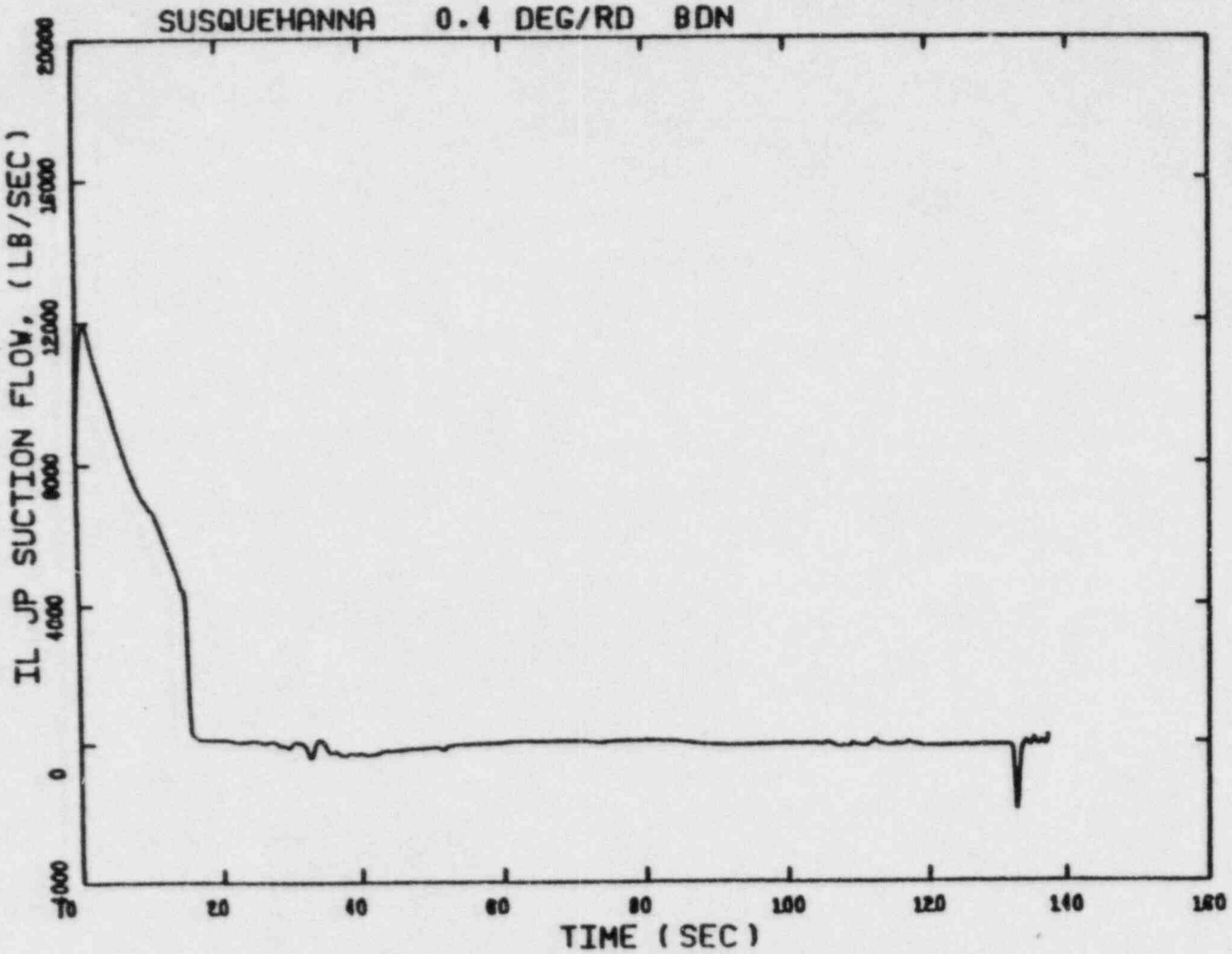

Figure 3.8 Blowdown Intact Loop Jet Pump
Suction FLow, 0.4 DEG/RD Break

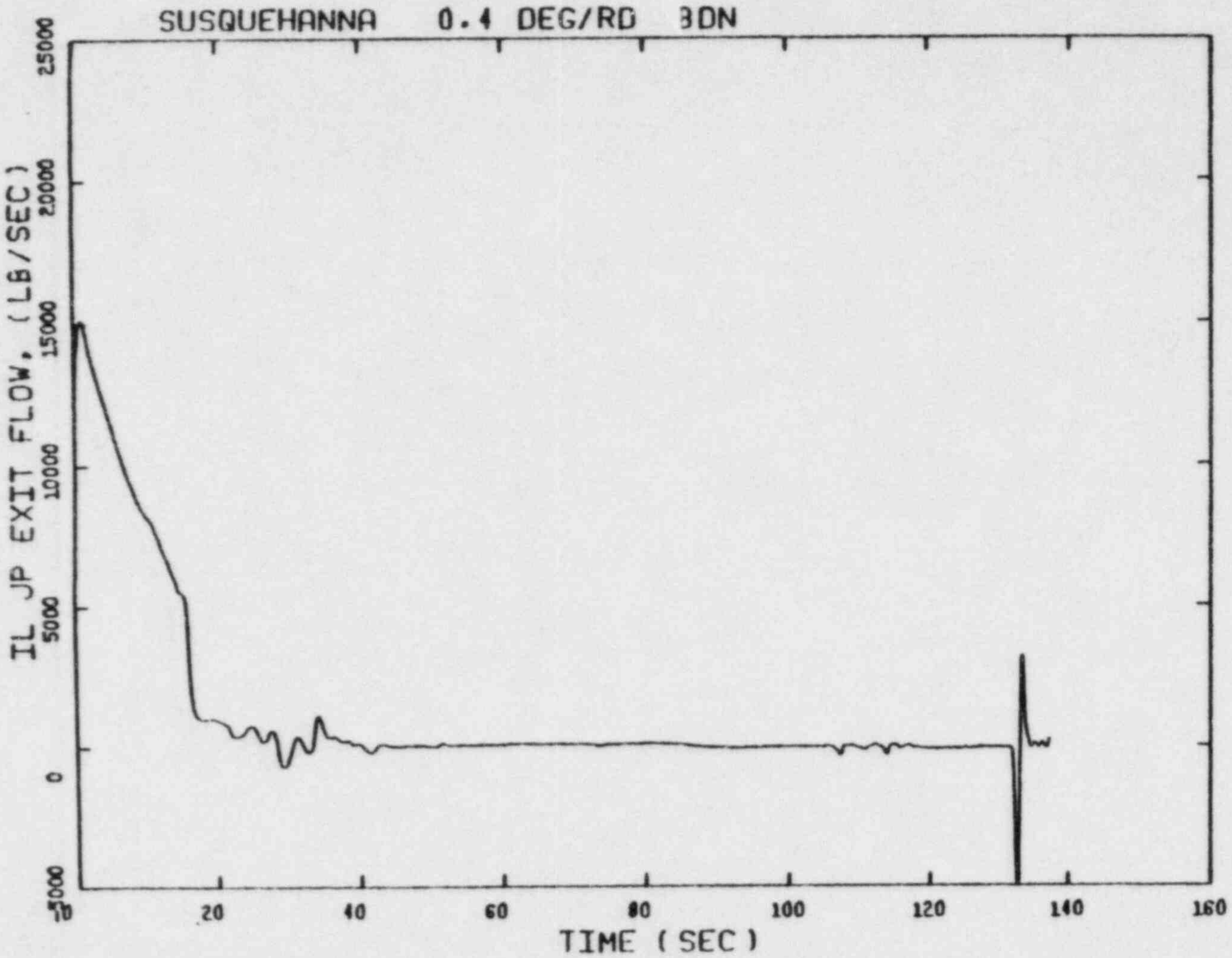

Figure 3.9 Blowdown Intact Loop Jet Pump Exit Flow, 0.4 DEG/RD Break

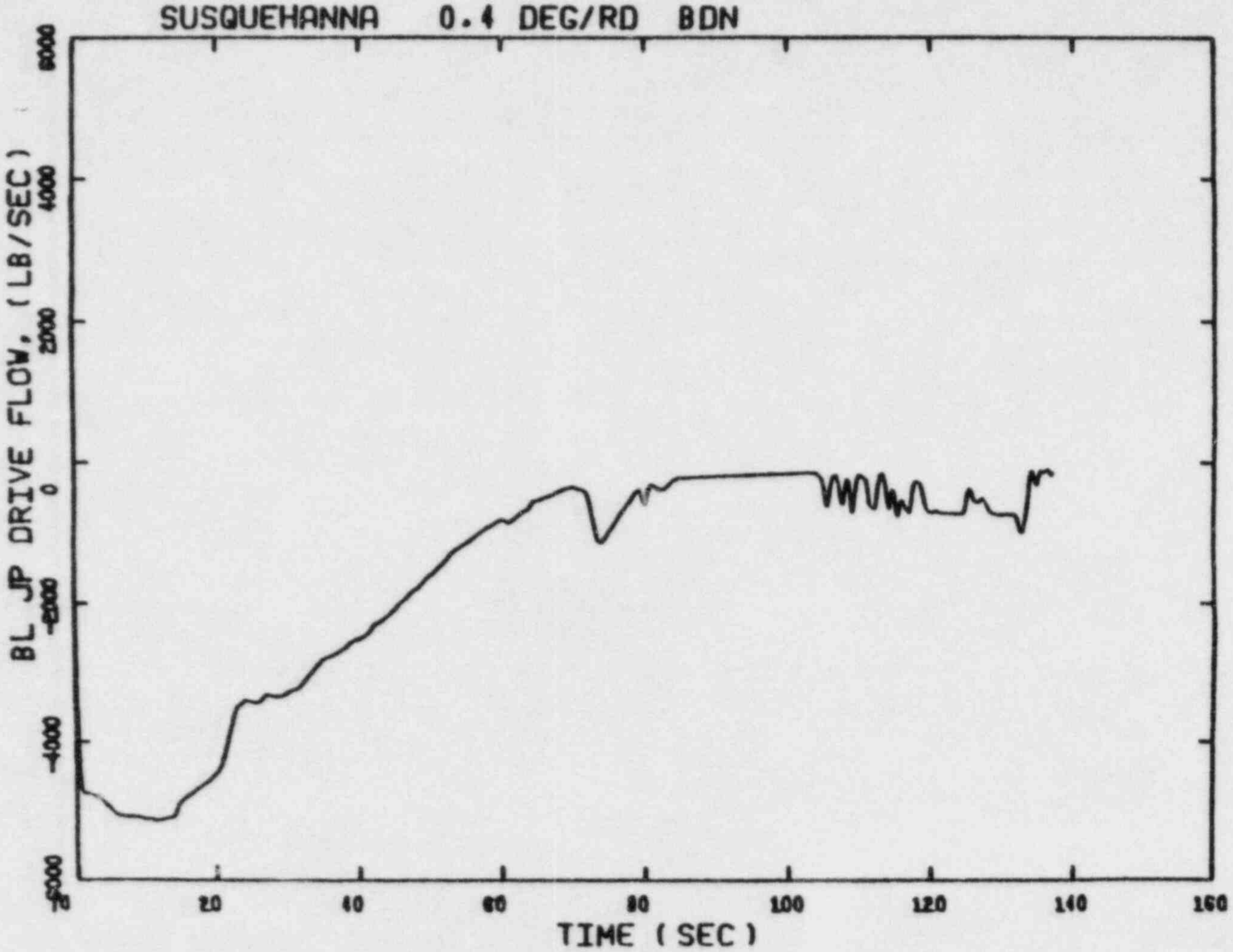

Figure 3.10 Blowdown Broken Loop Jet Pump Drive Flow, 0.4 DEG/RD Break

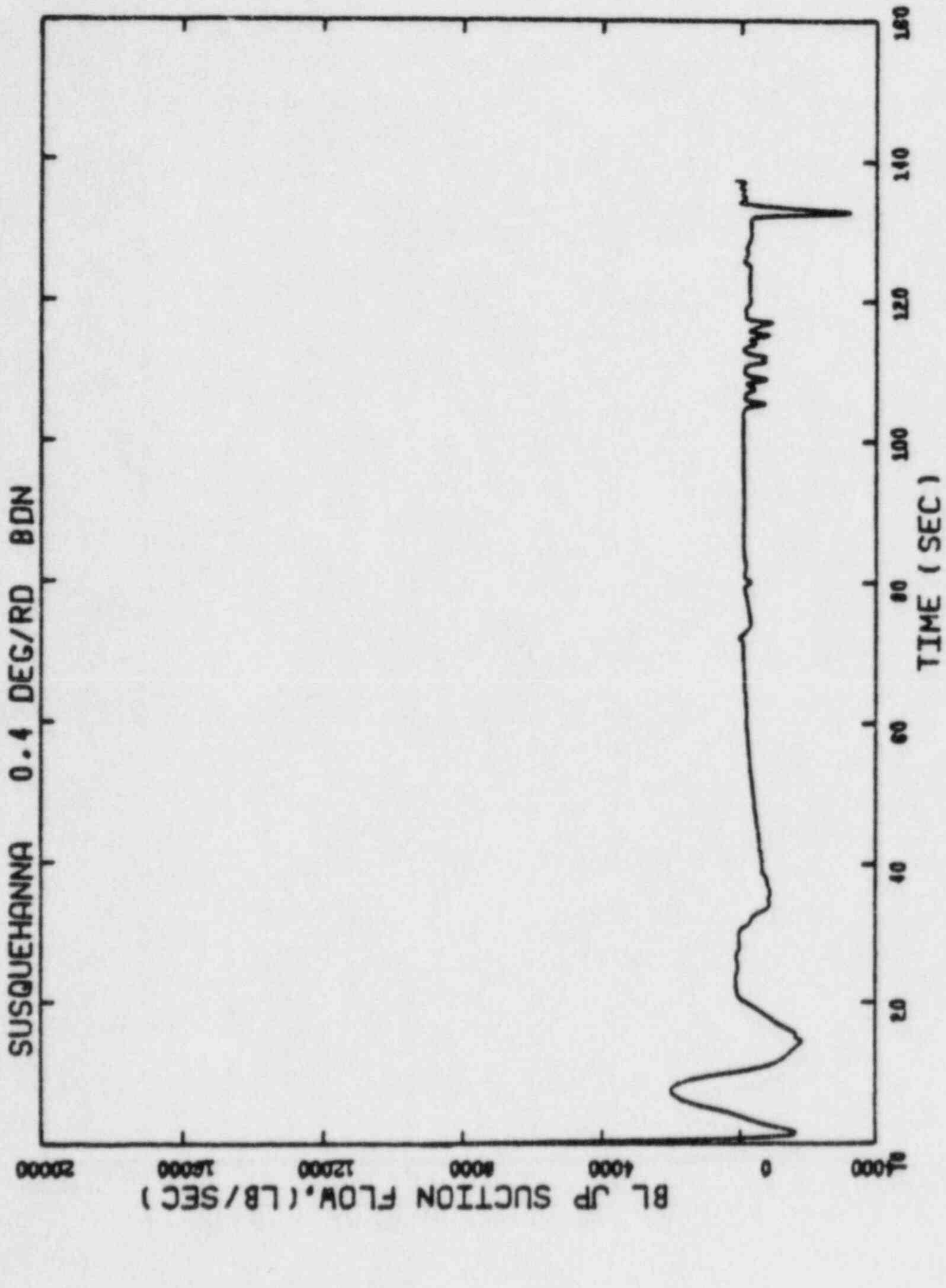

Figure 3.11 Blowdown Broken Loop Jet Pump Suction Flow, 0.4 DEG/RD Break

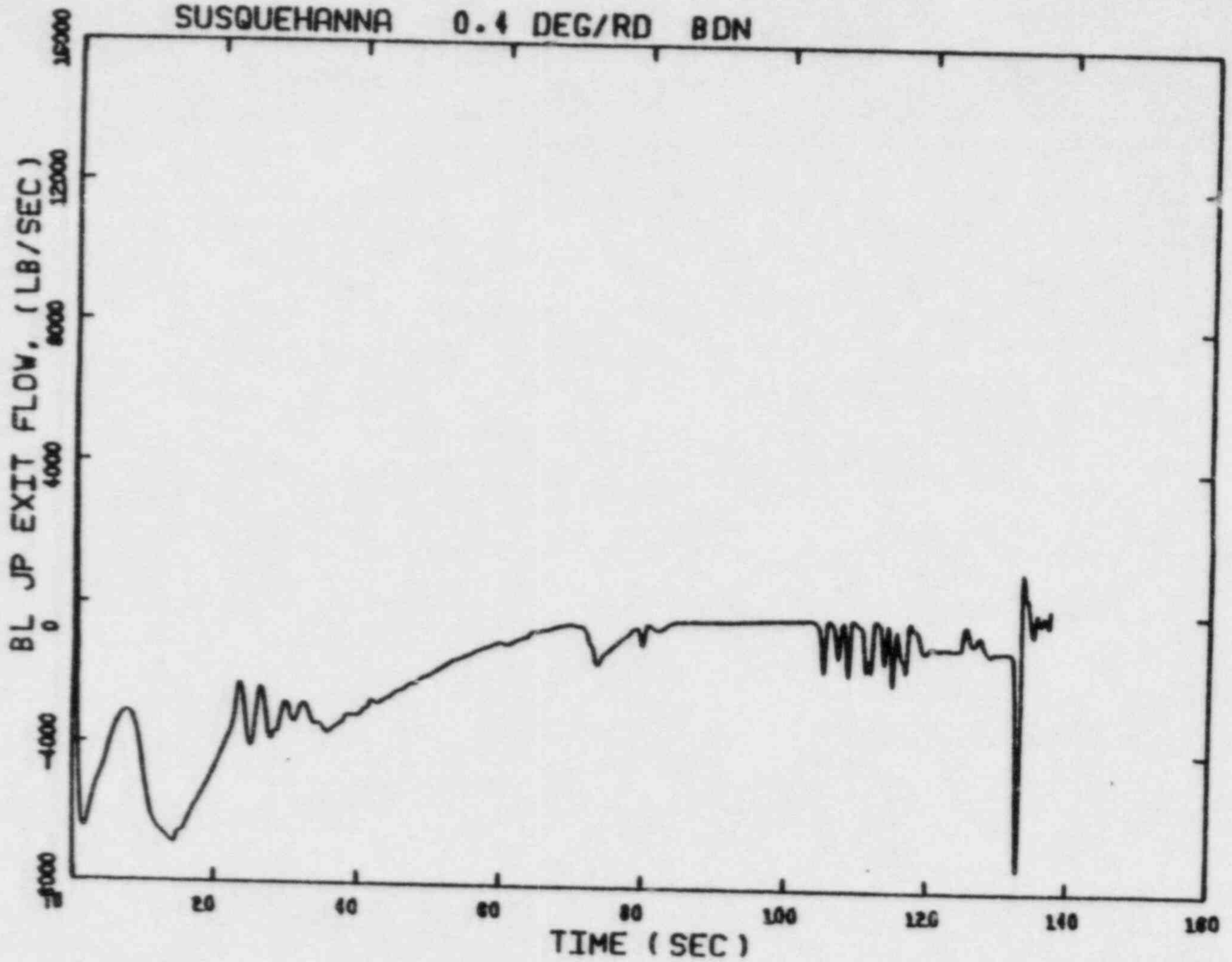

Figure 3.12 Blowdown Broken Loop Jet Pump Exit Flow, 0.4 DEG/RD Break

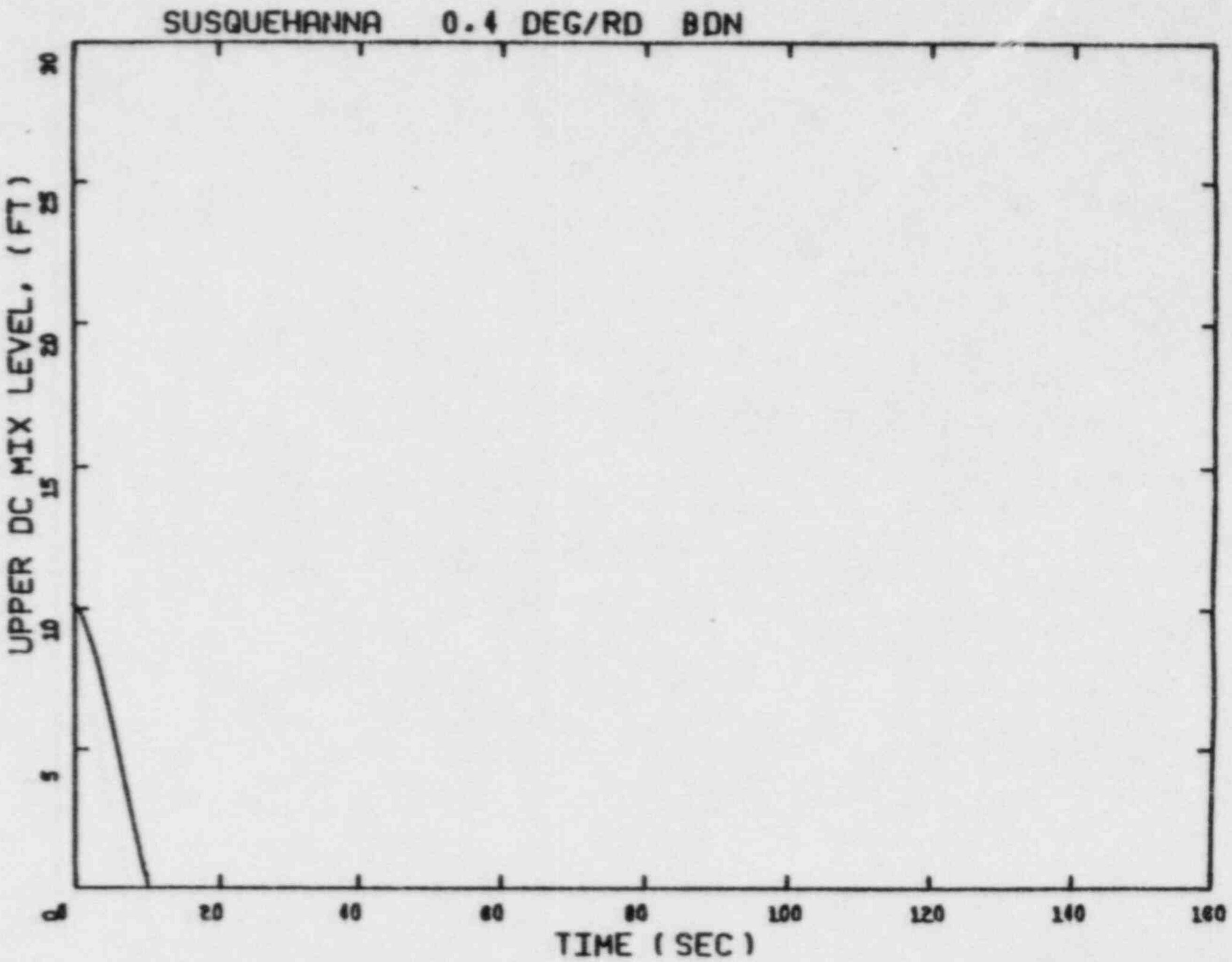

Figure 3.13 Blowdown Upper Downcomer Mixture Level, 0.4 DEG/RD Break

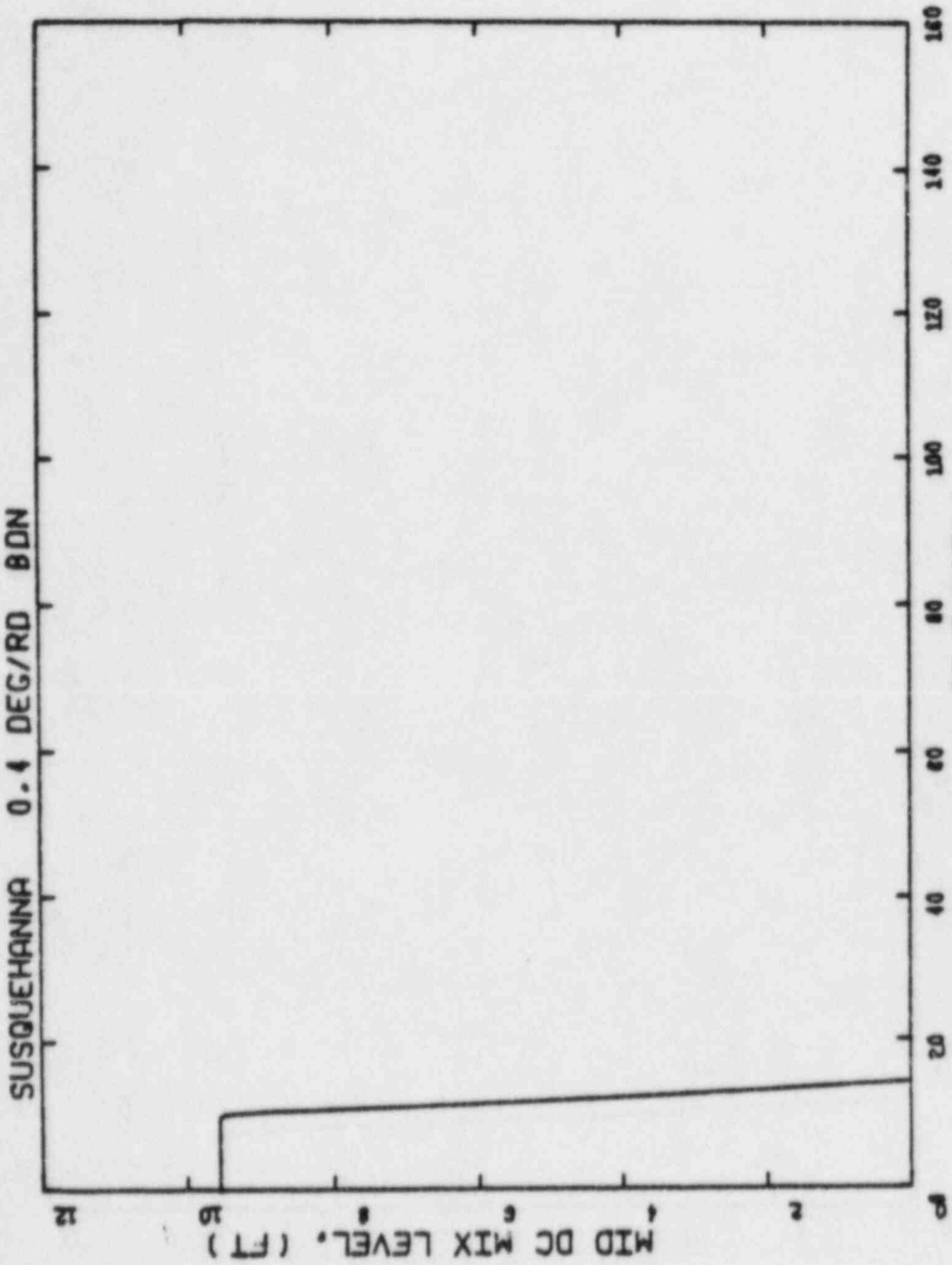

Figure 3.14 Blowdown Middle Downcomer Mixture
0.4 DEG/RD Break

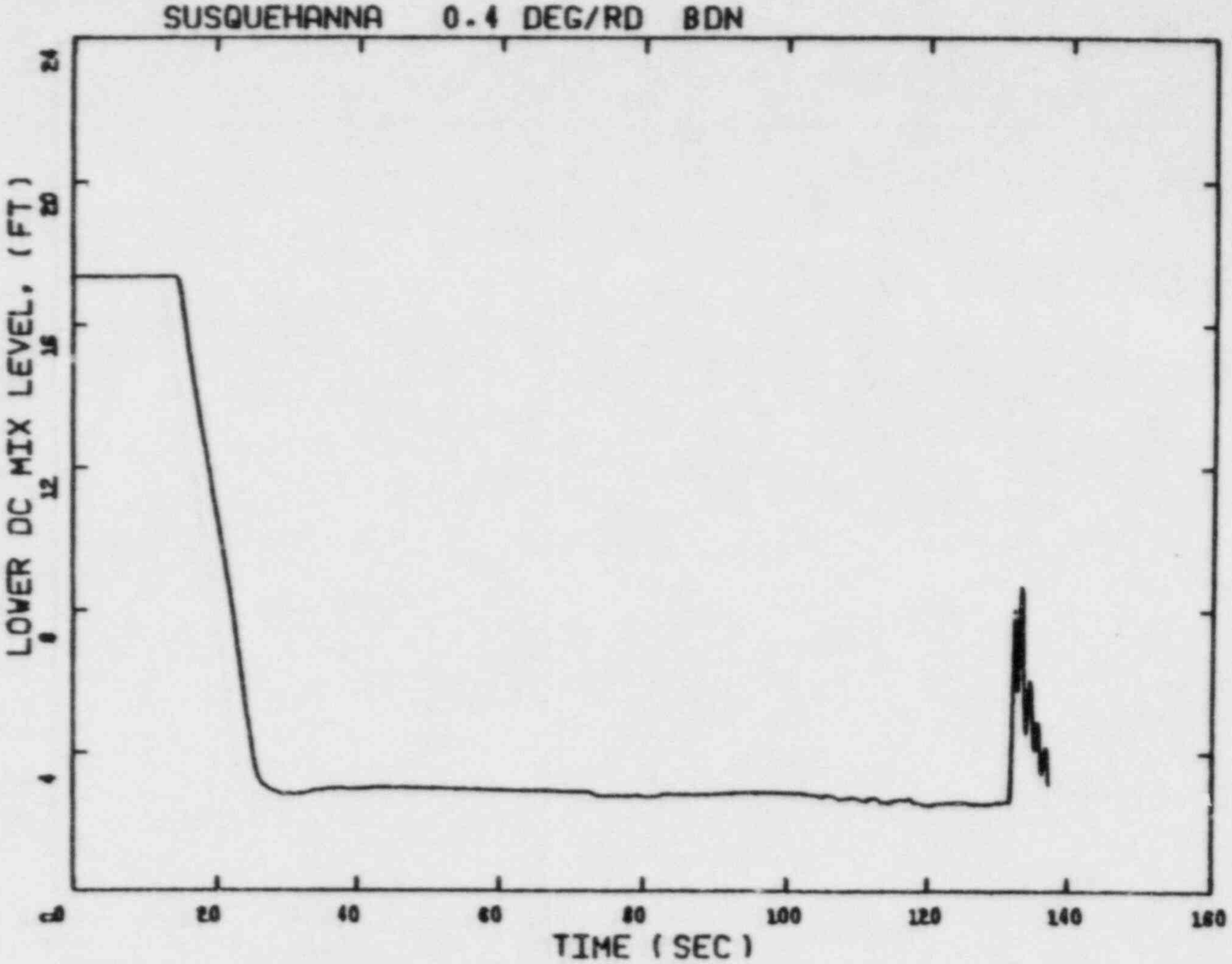

Figure 3.15 Blowdown Lower Downcomer Mixture Level, 0.4 DEG/RD Break

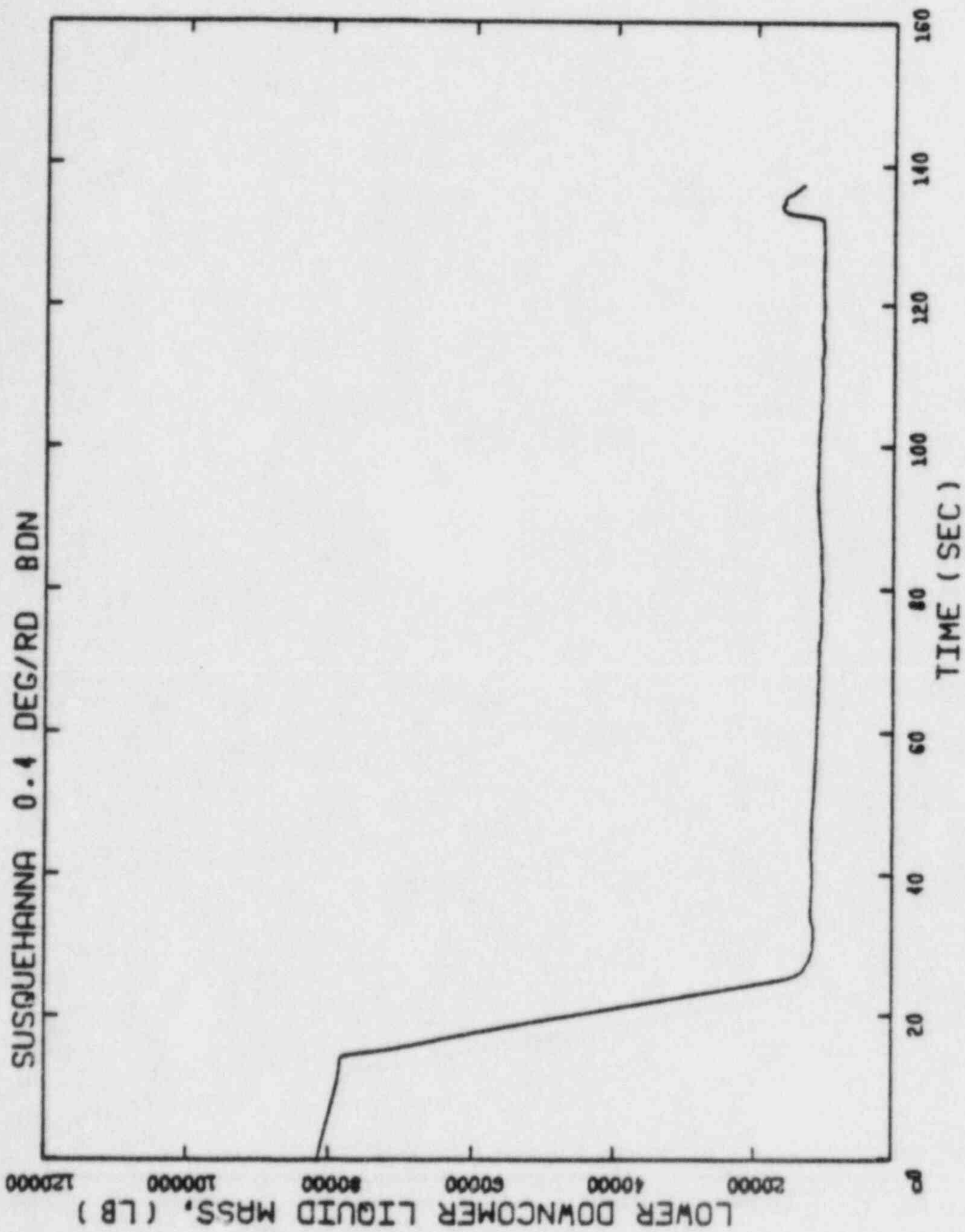

Figure 3.16 Blowdown Lower Downcomer Liquid Mass, 0.4 DEG/RD Break

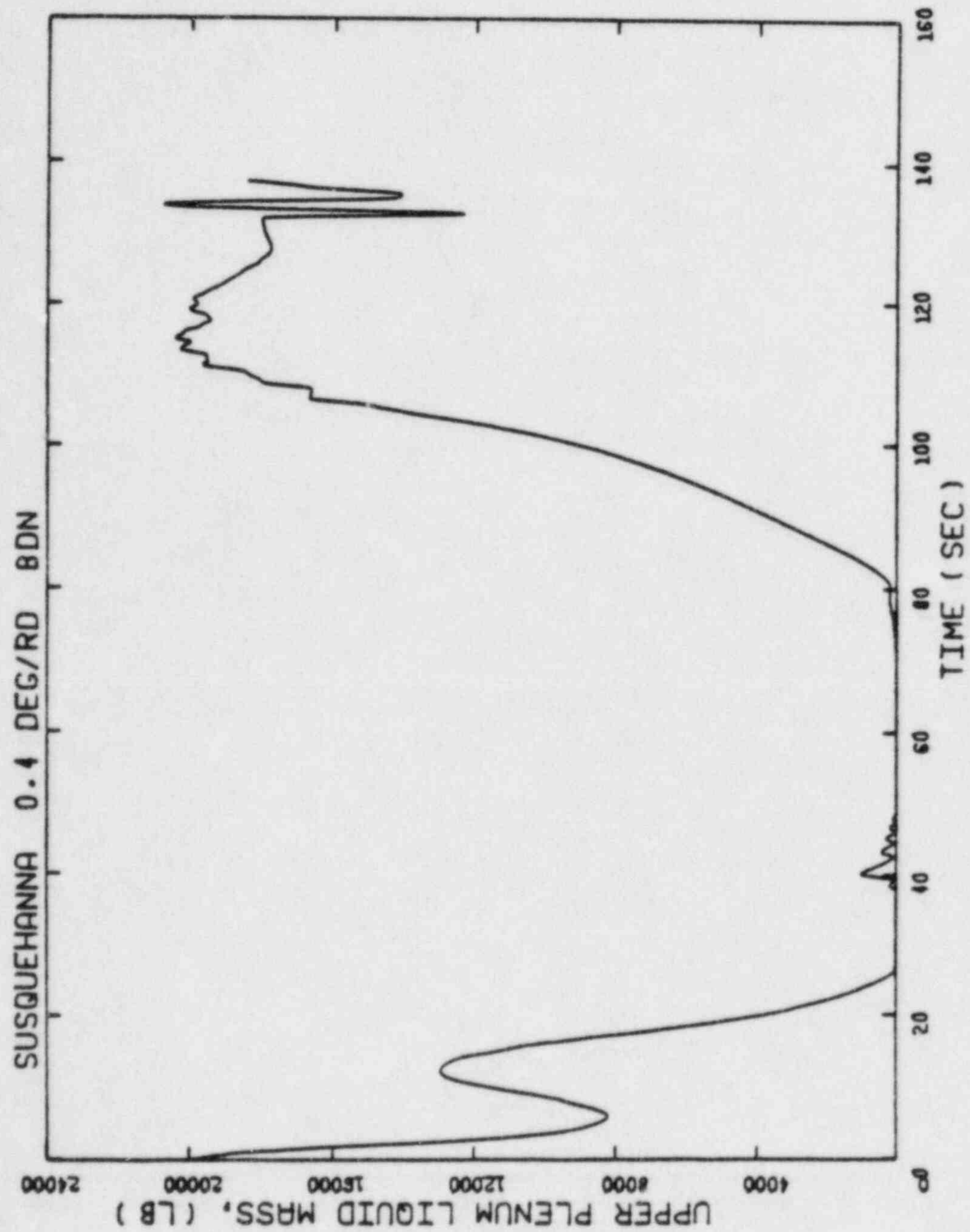

Figure 3.17 Blowdown Upper Plenum Liquid Mass,
0.4 DEG/RD Break

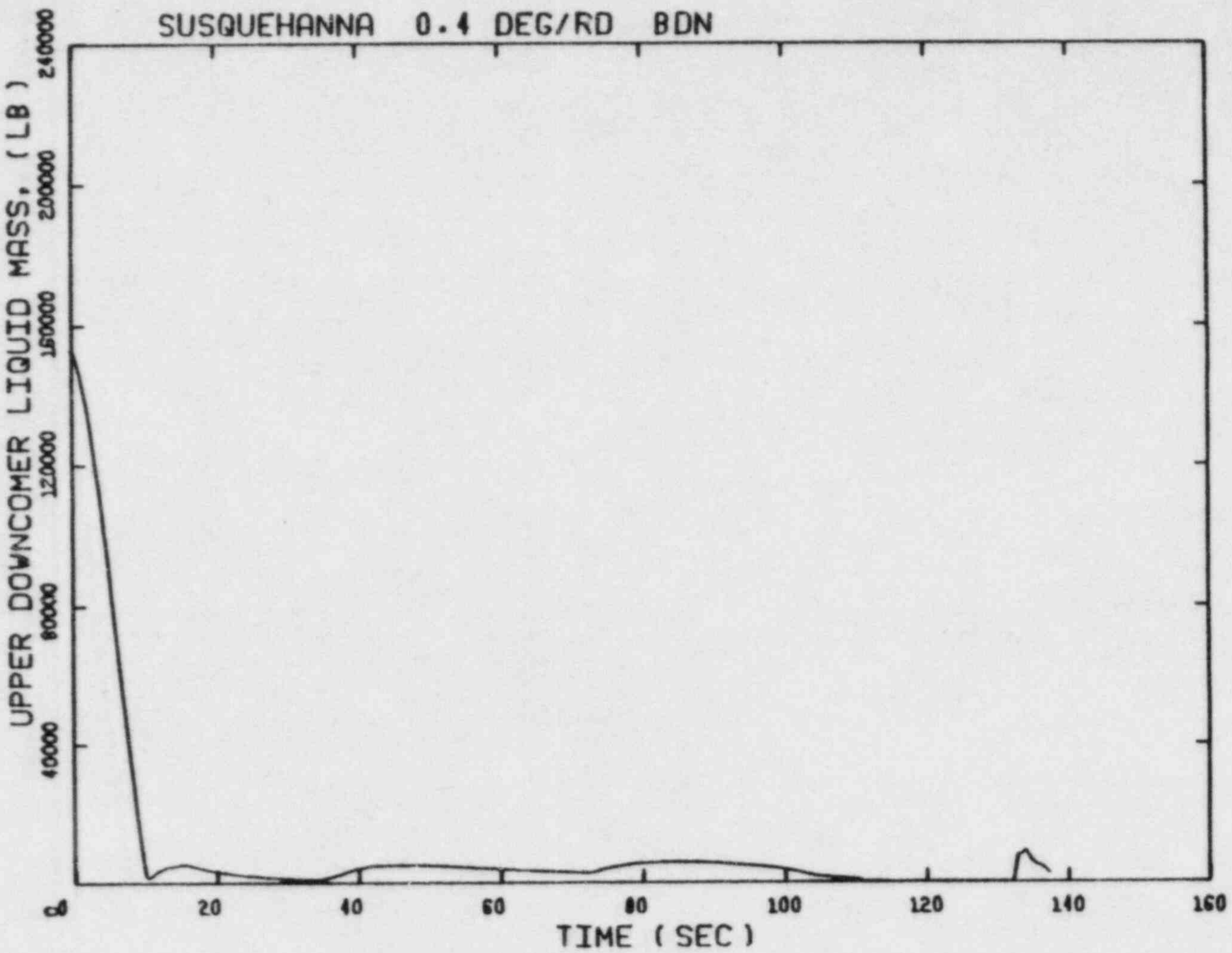

Figure 3.18 Blowdown Upper Downcomer Liquid Mass,
0.4 DEG/RD Break

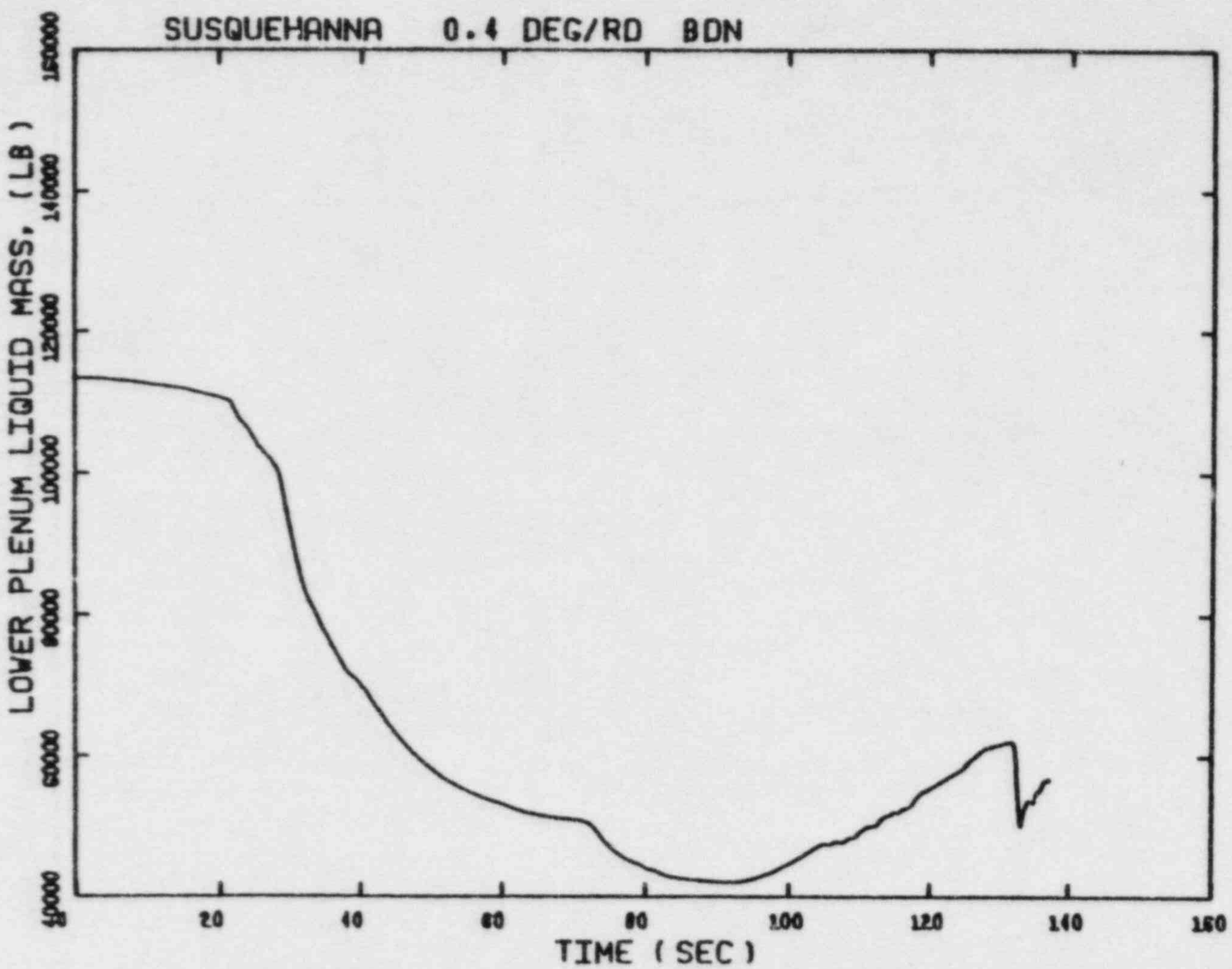

Figure 3.19 Blowdown Lower Plenum Liquid Mass, 0.4 DEG/RD Break

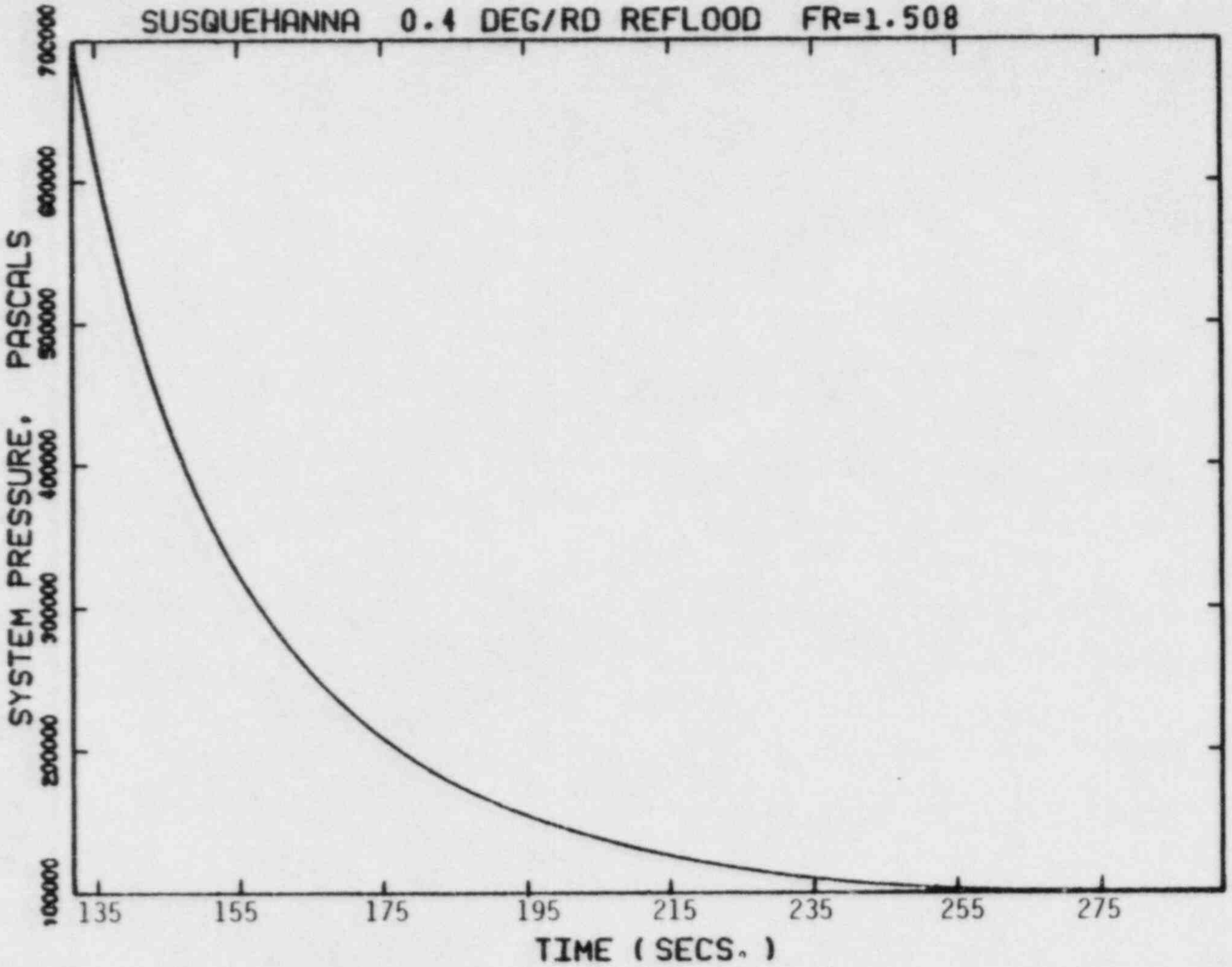

Figure 3.20 Refill/Reflood System Pressure, 0.4 DEG/RD Break

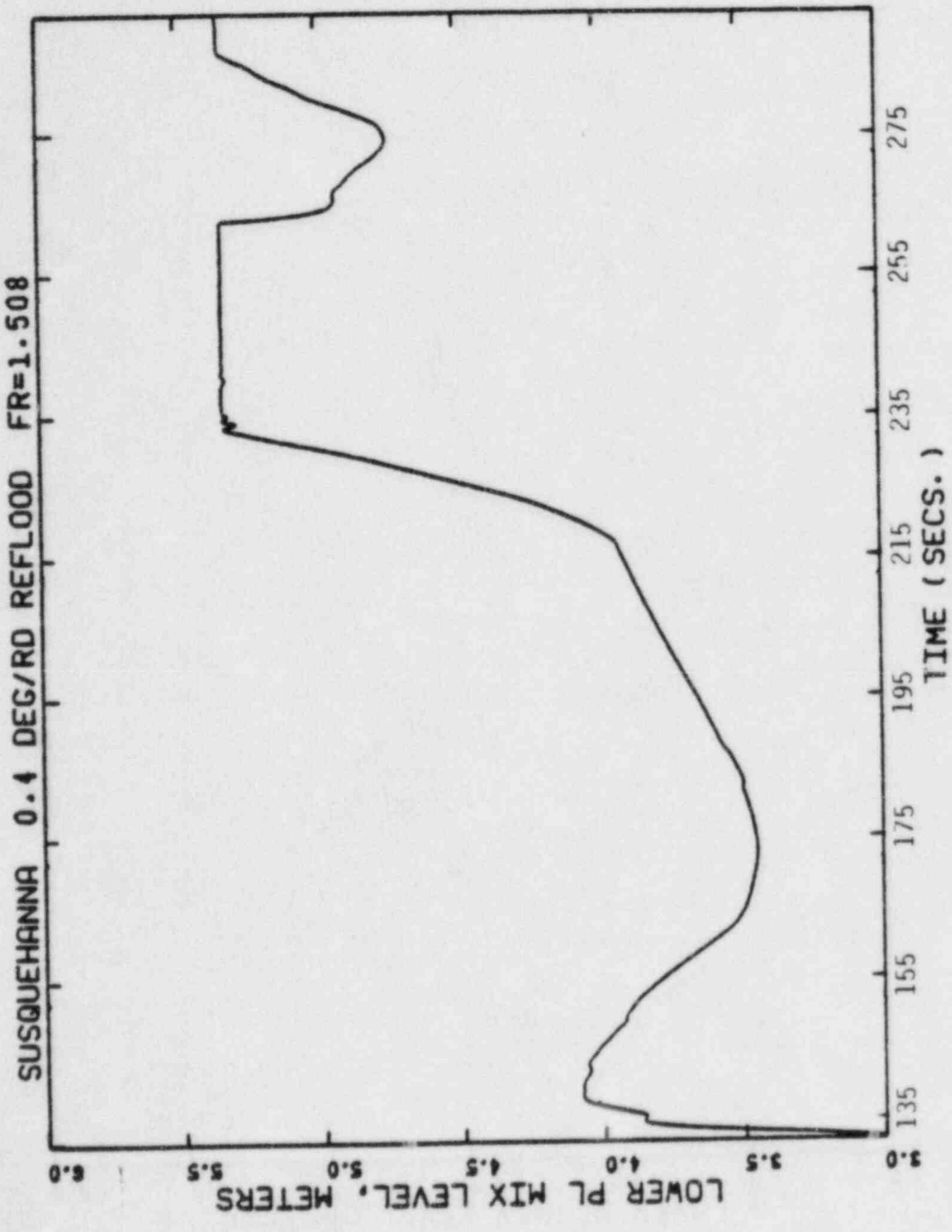

Figure 3.21 Refill/Reflood Lower Plenum Mixture Level, 0.4 DEG/RD Break

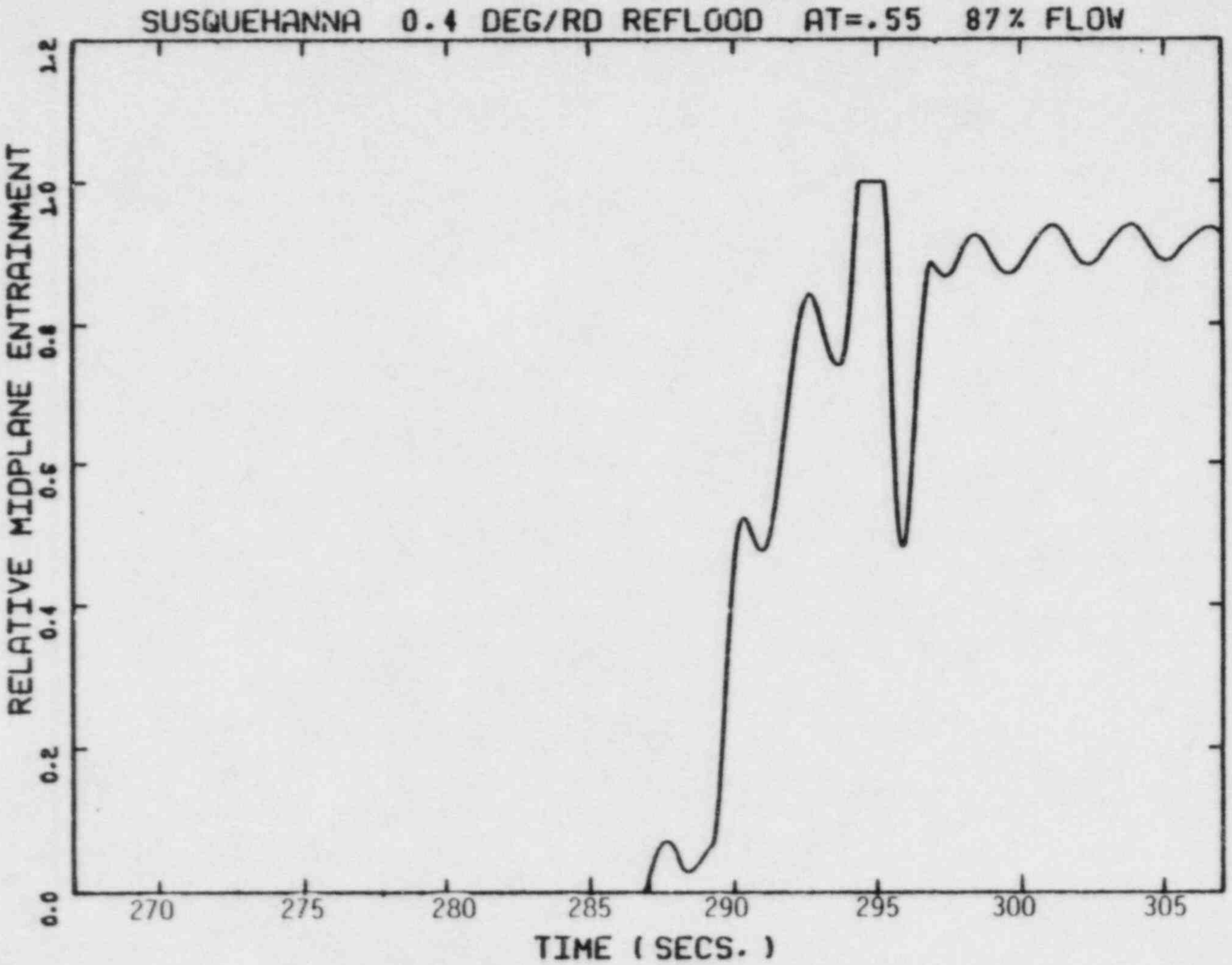

Figure 3.22 Refill/Reflood Relative Core Midplane Entrainment, 0.4 DEG/RD Break

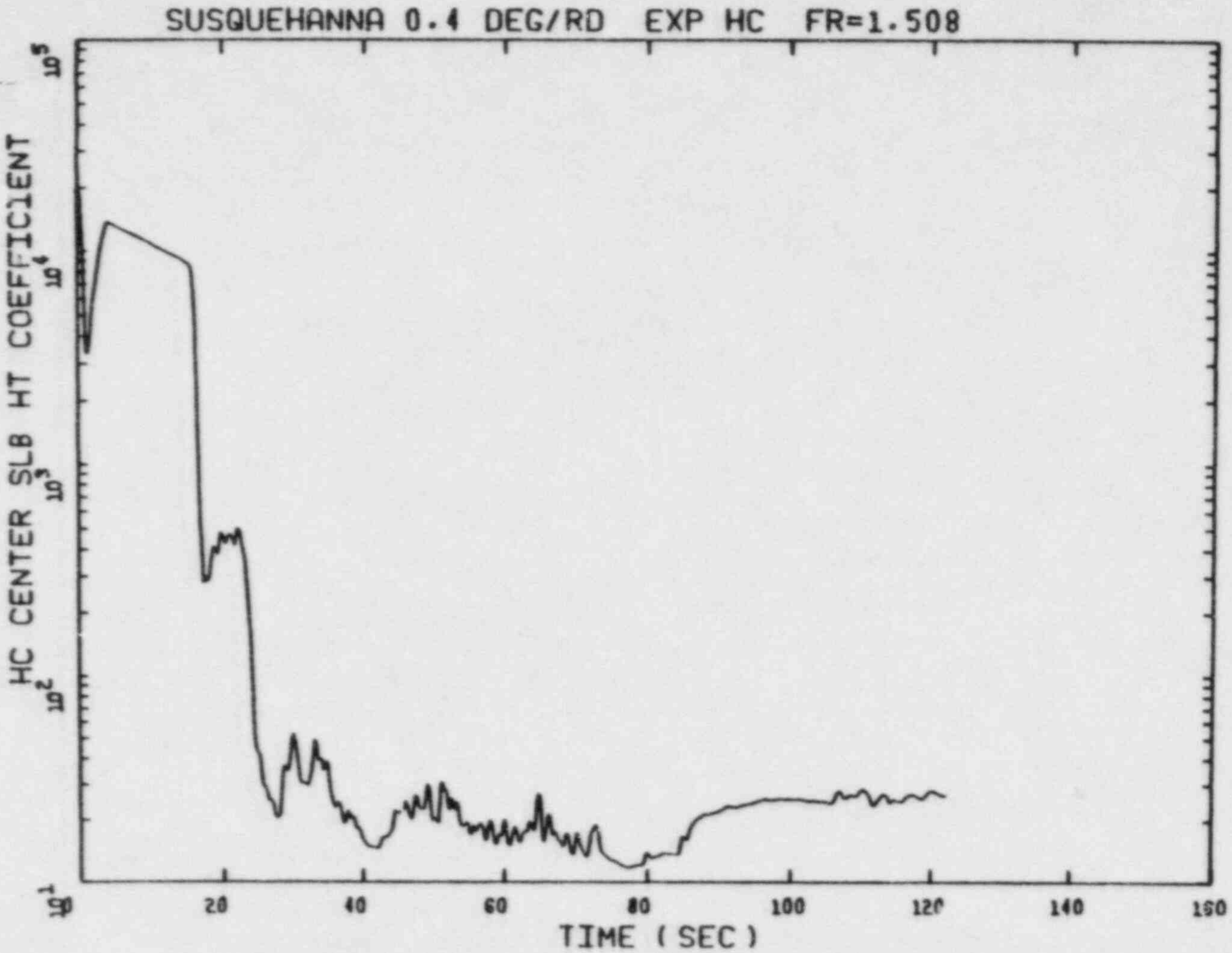

Figure 3.23 Blowdown HOT CHANNEL Heat Transfer Coefficient, 0.4 DEG/RD Break

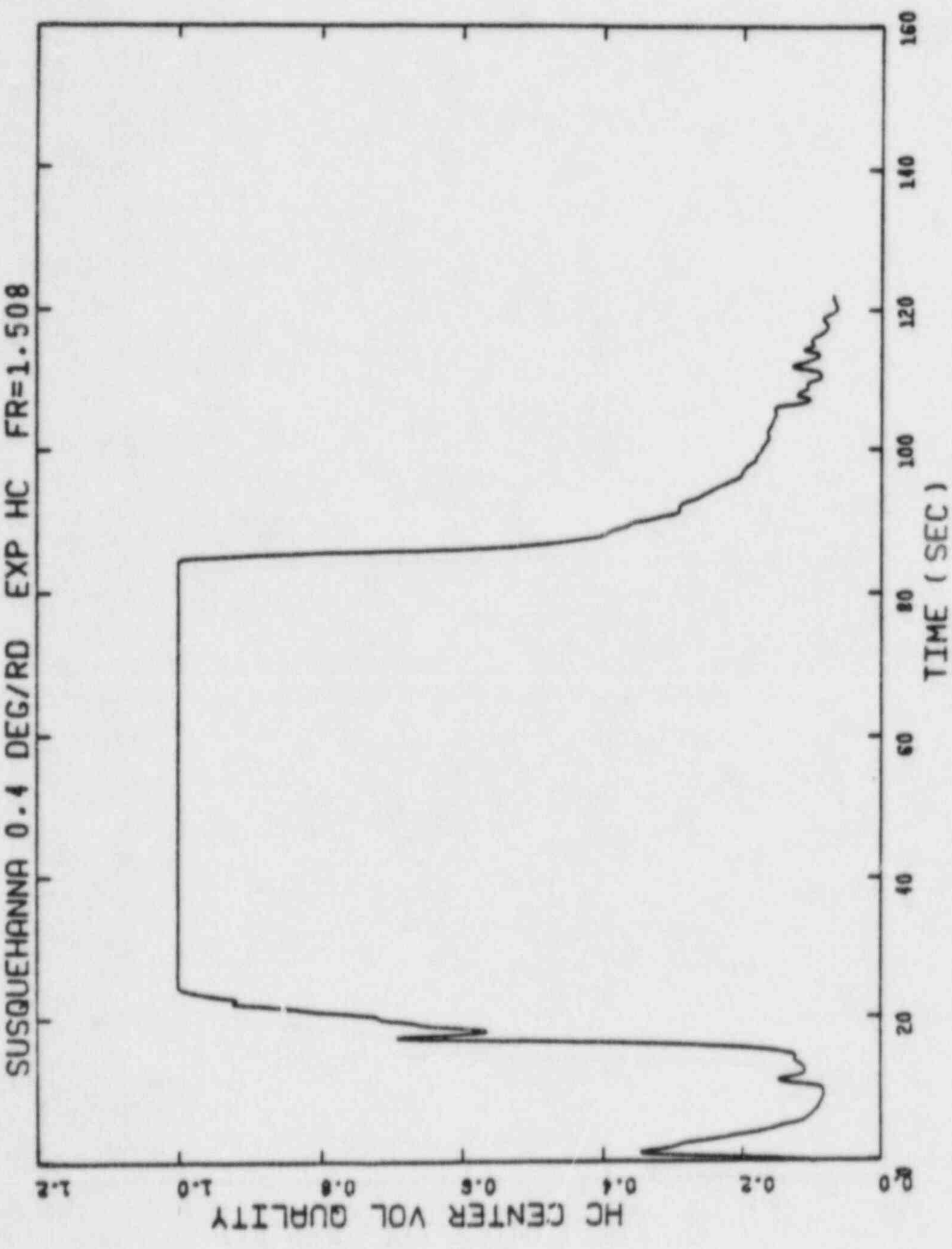

Figure 3.24 Blowdown HOT CHANNEL Center Volume Quality,
0.4 DEG/RD Break

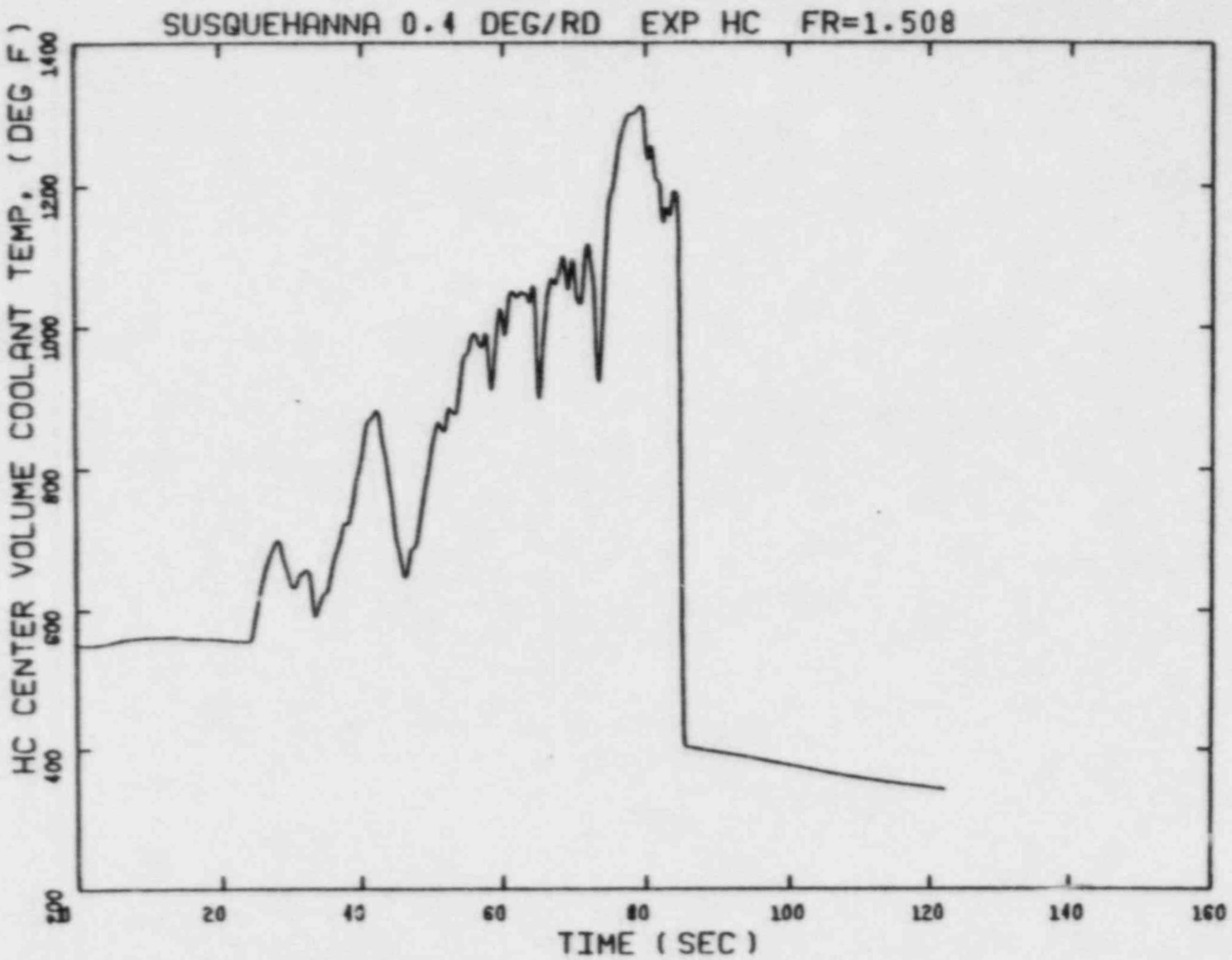

Figure 3.25 Blowdown HOT CHANNEL Center Volume Coolant Temperature, 0.4 DEG/RD Break

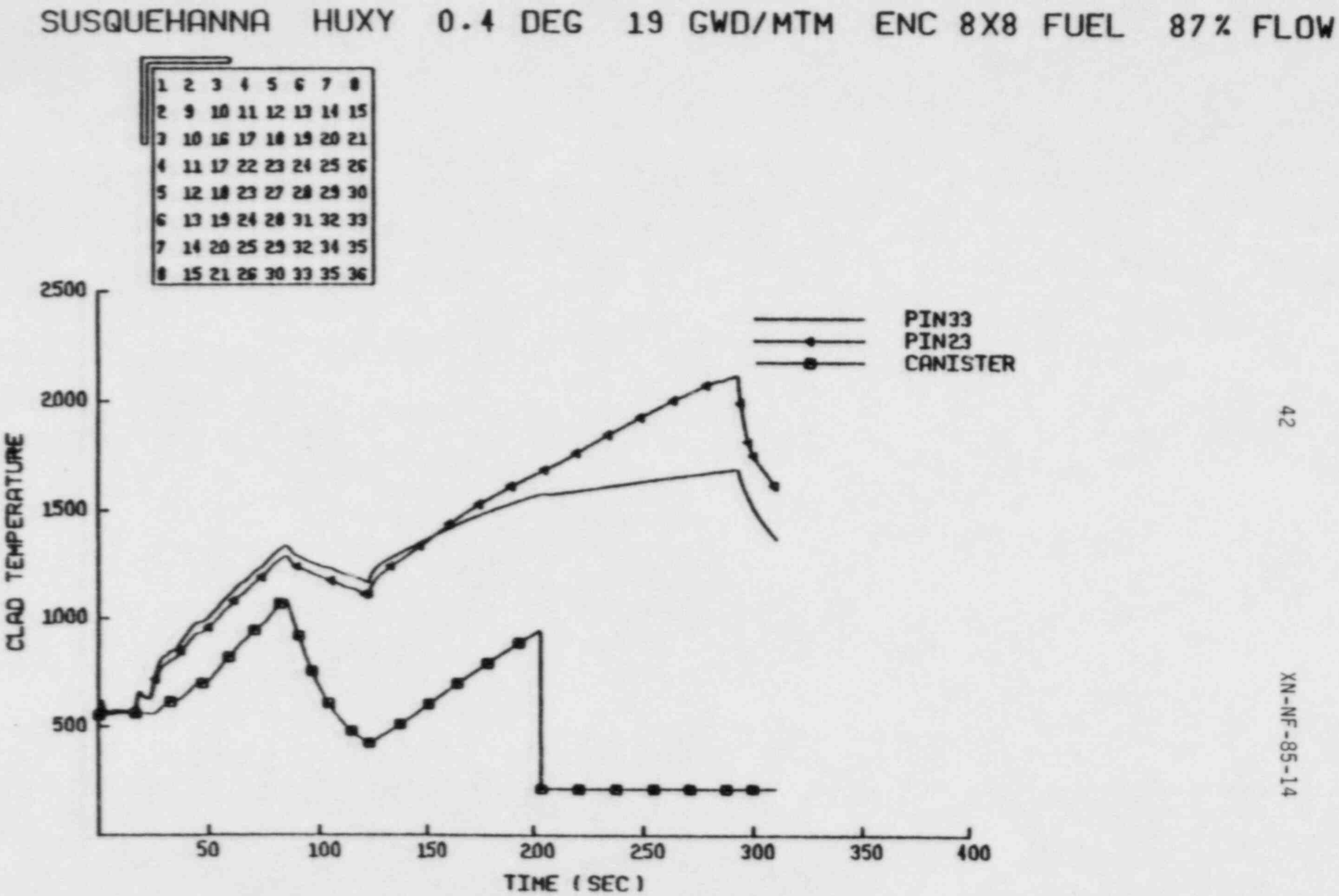

Figure 3.26 Hot Assembly Heatup Results, 19 GWD/MTM MAPLHGR = 13.0

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5. "Susquehanna Unit 1 LOCA-ECCS Analysis MAPLHGR Results," XN-NF-84-119, Exxon Nuclear Company, December 1984.

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ECCS ANALYSIS FOR SUSQUEHANNA UNIT 1 AT FULL POWER

AND 87% FLOW USING THE ENC EXEM BWR EVALUATION MODEL

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STARTUP TEST CRITERIA FOR SYMMETRIC TIP MEASUREMENT DIFFERENCES

The degree of TIP asymmetry in a particular cycle shall be considered consistent with the TIP uncertainty reported in XN-NF-80-19(P), Volume 1, Supplement 2, "Exxon Nuclear Methodology for Boiling Water Reactors, Volume 1, Neutronics Methods for Design and Analysis", unless the measured data provides statistical evidence that this assigned value is not valid. For the purpose of this test the variance of the integrated TIP responses shall be the quantity of interest. The χ^2 test of significance will be performed with the significance level fixed at 1%. The startup test criteria for symmetric TIP differences is that the χ^2 value calculated shall be less than the critical value at the 1% level of significance. The details of this test are as follows:

Define:

$$\chi^2 = M * S^2TIP_{ij} / \sigma^2TIP_{ij}$$

where:

M = the number of symmetric TIP pairs

S^2TIP_{ij} = The variance of the integrated TIP response determined from the differences between symmetric TIPs calculated for the current TIP data set.

σ^2TIP_{ij} = The variance of the integrated TIP response as determined in XN-NF-80-19(P), Volume 1, Supplement 2 (36.0).

The definitions of the above quantities in terms of the data measured at the plant are as follows:

Let:

T_{mk1} = The measured values of the first of two symmetric TIPs; pair m, axial level k.

T_{mk2} = The measured value of the second of two symmetric TIPs; pair m, axial level k.

T_{m1} = Sum of T_{mk1} over k.

T_{m2} = Sum of T_{mk2} over k.

d_m = Relative difference between T_{m1} and T_{m2}.

M = Total number of symmetric TIP pairs and the number of degrees of freedom.

Define:

$$T_{m1} = \sum_{k=3}^{22} T_{mk1}$$

$$T_{m2} = \sum_{k=3}^{22} T_{mk2}$$

$$d_m = 100 * (T_{m1} - T_{m2})/[(T_{m1} + T_{m2})/2]$$

Then:

$$S^2 TIP_{ij} = \frac{\sum_{m=1}^{M} d_m^2}{2M}$$

The critical values for χ^2 are a function of M. If the value of χ^2 calculated is less than the critical value then the criteria has been satisfied. The critical values of χ^2 for various values of M are shown in the table below.

M	χ^2
19	36.19
18	34.81
17	33.41
16	32.00

The startup test should be performed at the beginning of each cycle. The core power level should be above 75% of rated thermal power when the test is performed. The control rod pattern during the test should be 1/8 core symmetric. All of the TIP machines should be operable for the performance of the test.

If the χ^2 value calculated is greater than the critical value then one or more of the following actions should be taken:

1) The instrumentation and data processing system should be reviewed for any problems which may contribute to abnormal TIP asymmetries. A second determination of χ^2 should be made and compared to the critical value. If the new measured value of χ^2 is less than the critical value then the criteria has been satisfied.

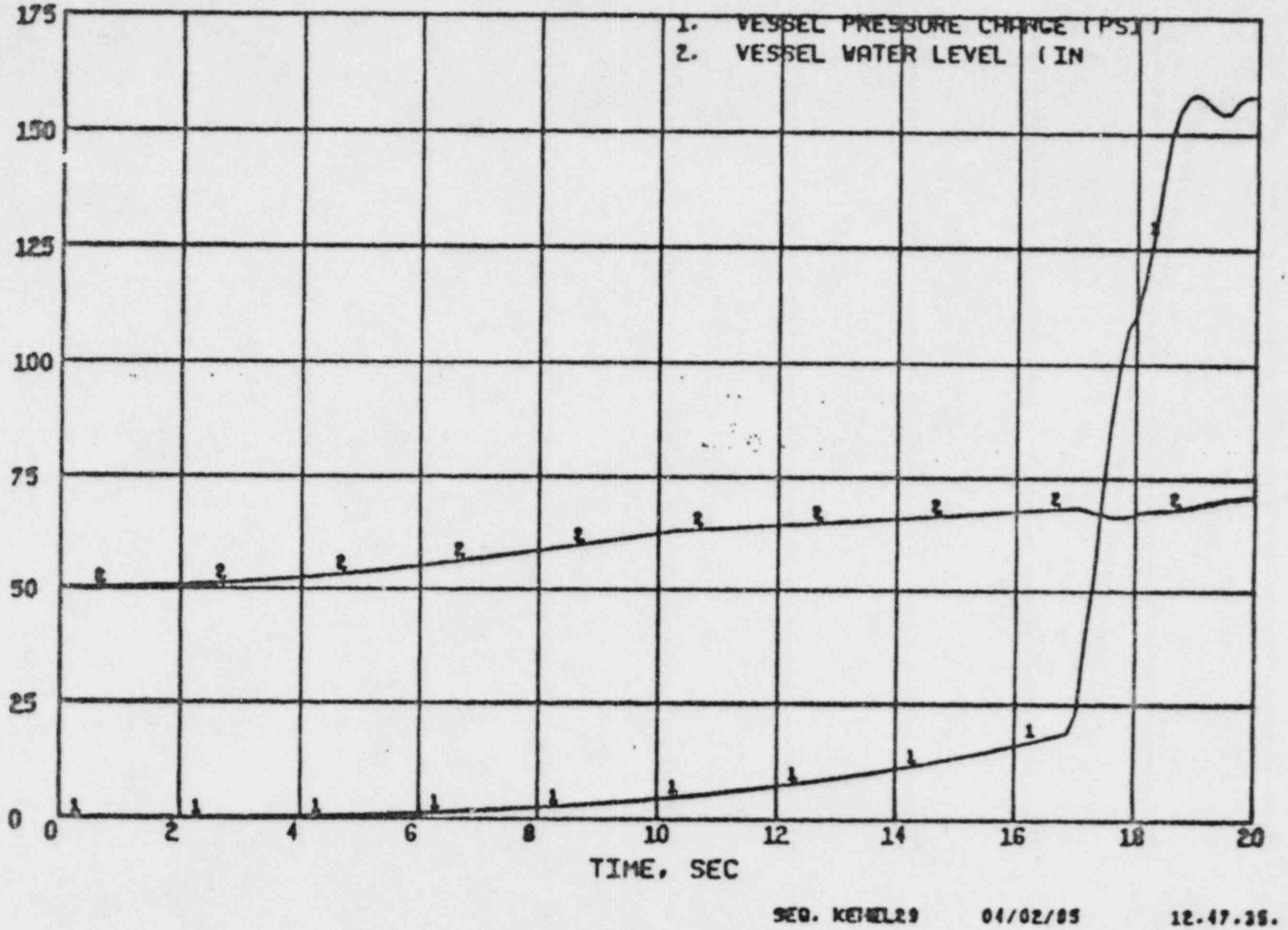
2) If the startup test criteria has not been satisfied by the above actions then the fuel vendor should be consulted and appropriate action taken to assure that a larger than anticipated TIP asymmetry does not adversely affect the safe operation of the reactor.

ADDITIONAL FEEDWATER CONTROLLER FAILURE TRANSIENT ANALYSIS FOR SUSQUEHANNA UNIT 1 CYCLE 2 WITH TURBINE BYPASS INOPERABLE

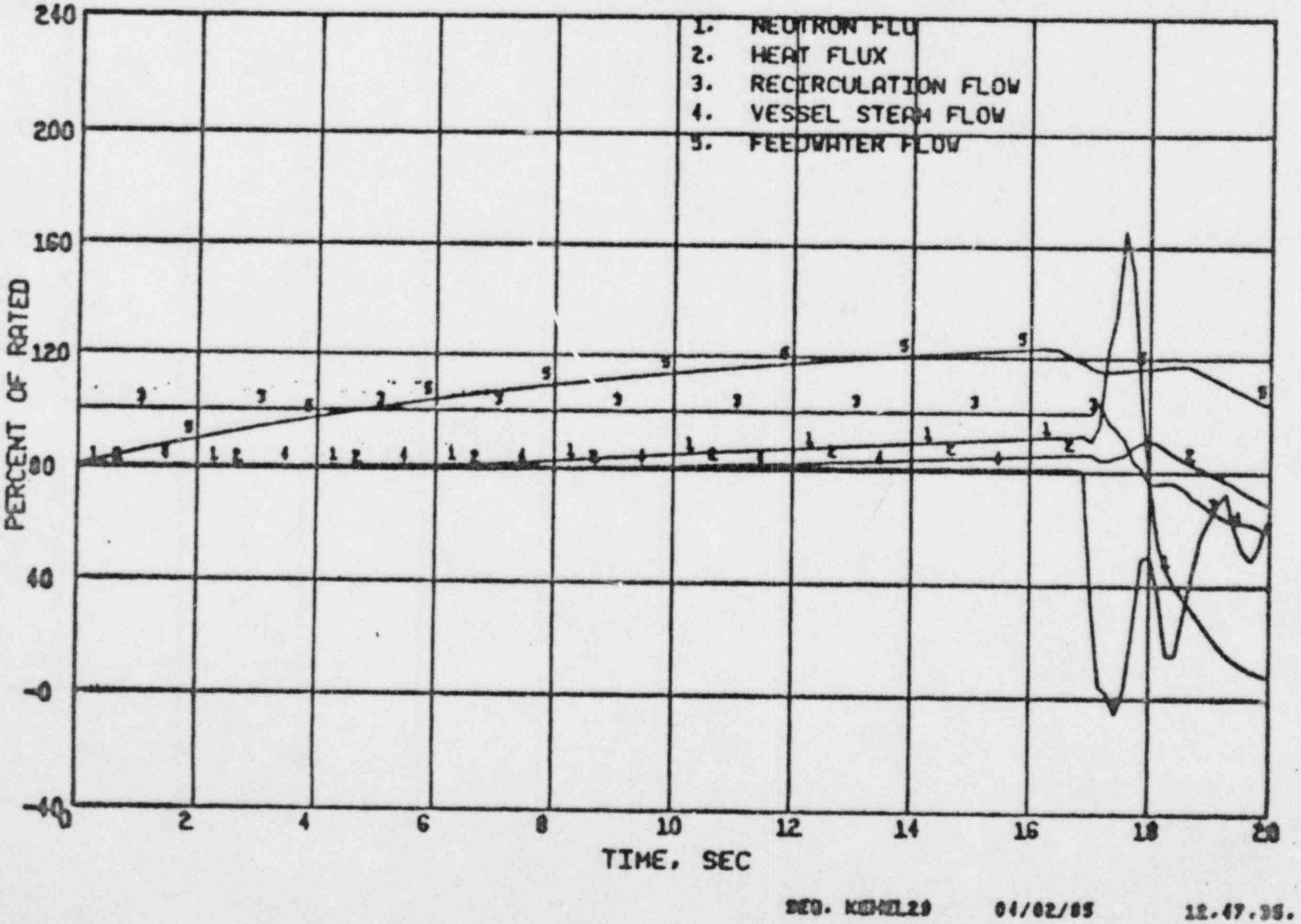
In the Susquehanna Unit 1 Cycle 2 plant transient analysis (XN-NF-84-118), ENC calculated the feedwater controller failure transient with the bypass assumed inoperable. This calculation was performed for 104% power/100% flow conditions, consistent with the FSAR analysis, and a nominal end of cycle 1 exposure of 10,700 MWD/MT. The calculated ΔCPR was 0.24 as reported in XN-NF-84-118. A MCPR limit based on this result is appropriate for the full power full flow conditions analyzed, however, the feedwater controller failure transient gives worse results at low power conditions than at full power. Thus, the low power high flow operation permitted by the power/flow map would not be bounded by a MCPR limit based on the XN-NF-84-118 results.

To provide a bounding analysis for the unlikely condition of operation with the main turbine bypass inoperable, ENC performed an additional analysis of the feedwater controller failure from 80% power/100% flow conditions. The results of this calculation were: (1) Reactor scram was calculated at 16.8 seconds due to the high vessel water level trip. (2) The highest calculated vessel pressure was 1083.5 psia. (3) The maximum calculated power was 174.3% of rated. (4) The calculated ΔCPR was 0.27. Plots of the transient parameters for this calculation are given in the two attached figures.

The bounding analysis was performed for the 10,700 MWD/MT cycle 1 exposure because neutronics input had previously been generated at the 80% power/100% flow conditions. To support a cycle 1 burnup of 11,800 MWD/MT, the MCPR limit resulting from the above transient may have to be increased at the end of cycle 2 operation, as noted in the transmittal of the 11,800 MWD/MT analysis results.



Feedwater Controller Failure without Bypass
at 80% Power/100% Flow



Feedwater Controller Failure without Bypass
at 80% Power/100% Flow

TABLE 3.2.3-1

MCPR OPERATING LIMITS FOR RATED CORE FLOW

EQUIPMENT STATUS	MCPR OPERATING LIMIT GE FUEL	MCPR OPERATING LIMIT EXXON FUEL
1. EOC-RPT and Main Turbine Bypass OPERABLE, RBM Setpoint ≤ 108 %	1.36	1.32
2. EOC-RPT Inoperable, Main Turbine Bypass OPERABLE, RBM Setpoint ≤ 108 %	1.36	1.32
3. Main Turbine Bypass Inoperable, EOC-RPT OPERABLE, RBM Setpoint ≤ 108 %	1.36	~~1.32~~ 1.34 ✱
4. EOC-RPT and Main Turbine Bypass OPERABLE, RBM Setpoint ≤ 106 %	1.32	1.29
5. EOC-RPT Inoperable, Main Turbine Bypass OPERABLE, RBM Setpoint ≤ 106 %	1.32	1.29
6. Main Turbine Bypass Inoperable, EOC-RPT OPERABLE, RBM Setpoint ≤ 106 %	~~1.32~~ 1.34 ✱	~~1.30~~ 1.34 ✱

✱ Revised