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DRESDEN UNIT 3 LOCA-ECCS ANALYSIS
MAPLHGR Results for 9x9 Fuel

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MAPLHGR RESULTS FOR ENC 9x9 FUEL

Prepare: D. R. Swope 9-12-85
D. R. Swope
BWR Safety Analysis

Prepare: RE Collingham for 9/12/85
D. J. Braun
BWR Safety Analysis

Approve: R E Collingham 9/12/85
R. E. Collingham, Manager
BWR Safety Analysis

Approve: R B Stout 12 Sept 85
R. B. Stout, Manager
Licensing and Safety Engineering

Concur: 9/12/85
J. N. Morgan, Manager
Customer Services Engineering

Approve: K N Woods for G L Ritter 9/12/85
G. L. Ritter, Manager
Fuel Engineering and Technical Services

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1.0 INTRODUCTION AND SUMMARY

This report provides the results of loss-of-coolant accident emergency core cooling system (LOCA-ECCS) analyses performed by Exxon Nuclear Company (ENC) for ENC XN-3 9x9 fuel in the Dresden Unit 3 reactor. The results of this analysis are presented in terms of the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit as a function of fuel exposure for normal operation and for operation with one relief valve out-of-service. These calculations were performed with the generically approved Exxon Nuclear Company EXEM/BWR ECCS Evaluation Model(1,2) according to Appendix K of 10 CFR 50 (3), and the results comply with the U. S. NRC 10 CFR 50.46 criteria. The results are summarized in the Reload Analysis Report for Dresden Unit 3 Cycle 10. The remainder of Section 1.0 and Sections 2, 3, and 4 will cover the case of a LOCA during normal operation. The analysis for a LOCA during operation with one relief valve out-of-service will be covered separately in Appendix A.

A generic LOCA-ECCS break spectrum analysis applicable to jet pump BWR 3 reactors of the Dresden 2 and 3 design has been reviewed and approved by the USNRC (4). The worst or limiting LOCA break from the generic break spectrum during normal operation is the double-ended guillotine break of the recirculation suction pipe with a discharge coefficient of 1.0 (1.0 DEG). The analyses contained in this document were performed for the ENC 9x9 fuel design for the identified limiting LOCA break at the expected worst points bounding the operating power-flow map for Dresden 3 Cycle 10. ENC has previously performed LOCA analyses for the Dresden Unit 3 reactor utilizing ENC 8x8 fuel assemblies (5). These analyses provide MAPLHGR limits which remain applicable for the ENC 8x8 fuel.

For normal operation, limiting LOCA break calculations were performed for the Dresden 3 reactor with a full core of ENC 9x9 fuel. Two calculations were performed, one for full-power full-flow (100/100) and the other at full-power and 87% flow (100/87) which is the minimum flow for allowed operation at full power from the current Dresden 3 operating power-flow map. Both operating conditions were found to result in essentially identical LOCA transients. The power/flow of 100/87 resulted in the highest peak cladding temperature (PCT) and was used to verify the LOCA-ECCS MAPLHGR limits.

MAPLHGR limits for 9x9 fuel during normal operation as a function of exposure were determined based on the limiting operating conditions for the identified limiting LOCA break. ENC MAPLHGR limits also protect against exceeding fuel design limits for 9x9 fuel. When the fuel design limit is more restrictive than LOCA-ECCS criteria, the LOCA-ECCS results will be significantly below the 2200 F criteria at higher exposures. This is the case for Dresden 3 with ENC 9x9 fuel. The exposure dependent MAPLHGR limits are shown in Figure 1.1, and the computed points used to determine this curve are given in Table 1.1. The MAPLHGR limits apply to ENC 9x9 fuel in the Dresden Unit 3 reactor during normal operation for both the initial and subsequent reloads of the current 9x9 fuel design.

All calculations were performed with the NRC approved EXEM/BWR ECCS Evaluation Model according to Appendix K of 10 CFR 50. Operation of the Dresden Unit 3 reactor with ENC 9x9 fuel at or below the MAPLHGR limits of Figure 1.1 satisfies the criteria specified by 10 CFR 50.46 of the U. S. Code of Federal Regulations, and assures that the emergency core cooling system for the Dresden Unit 3 reactor will meet the U. S. NRC acceptance criteria for Loss-of-Coolant Accident breaks up to and including the double-ended severance of a reactor coolant pipe. That is:

1. The calculated peak fuel element clad temperature does not exceed the 2200 F limit.
2. The amount of fuel element cladding which reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor.
3. The cladding temperature transient is terminated at a time when the core is still amenable to cooling. The hot fuel rod cladding oxidation limit of 17% is not exceeded during or after quenching.
4. The system long term cooling capabilities provided for the intial core and subsequent reloads remains applicable to ENC fuel.

Table 1.1 Dresden Unit 3 MAPLHGR Summary for ENC 9x9 Reload Fuel (Types XN-3 and XN-3A)

Assembly Average Burnup (GWD/MTM)	Cycle 10 MAPLHGR Limits (kW/ft)
	Normal Operation
0.	11.40
5.	11.75
10.	11.40
15.	10.55
20.	9.70
25.	8.85
30.	8.00
35.	7.15
40.	6.30

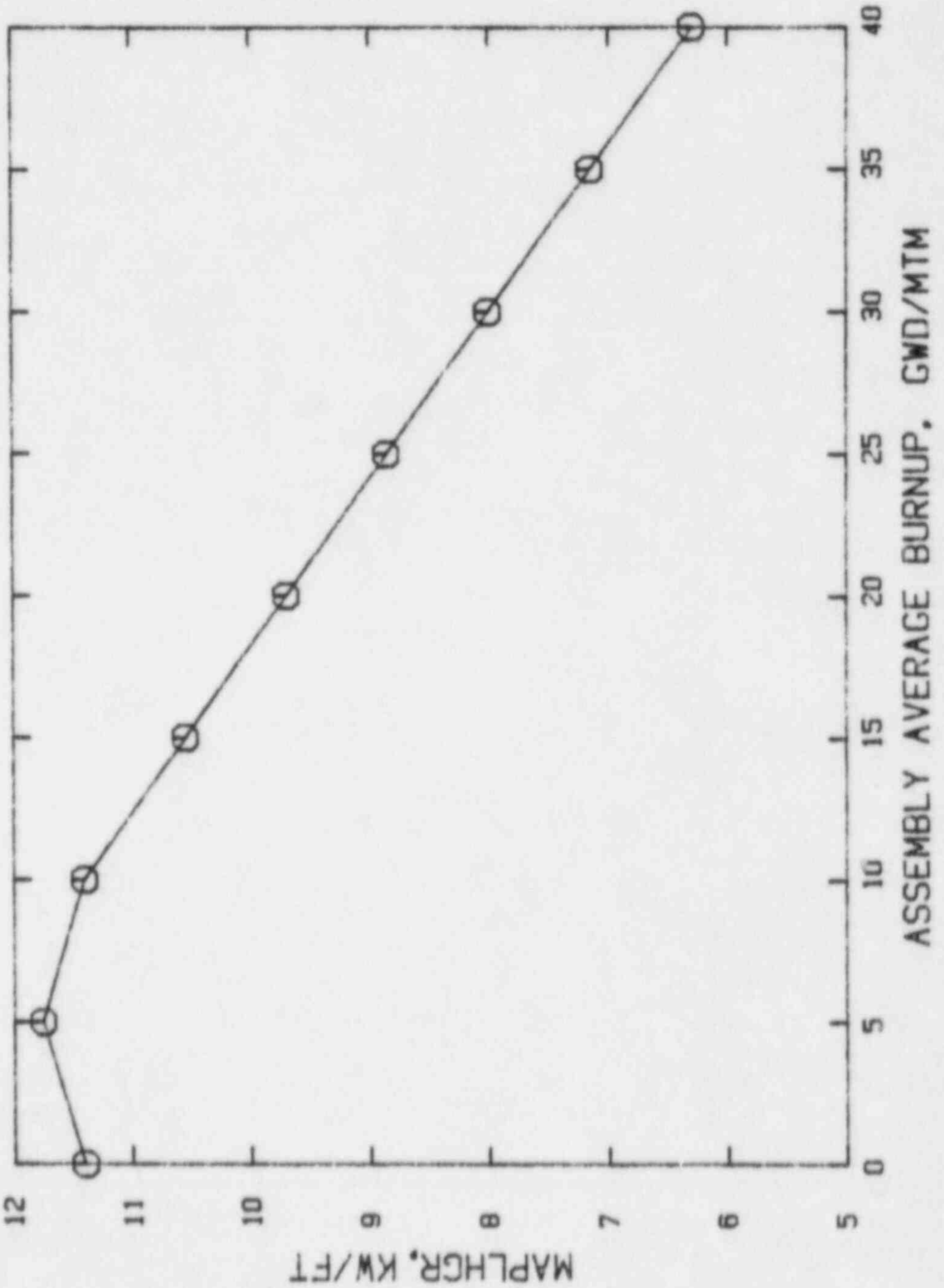

Figure 1.1 Dresden 3 MAPLHGR vs Assembly Average Burnup
(Types XN-3 and XN-3A, Normal Operation)

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 Dresden Unit 3 reactor, the reactor coolant system inventory loss would result in a high contaiment drywell pressure and reduced reactor vessel pressure. The concurrent high drywell pressure and low reactor vessel pressure provide 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 exiting 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/BWR APPLICATION TO DRESDEN UNIT 3

The EXEM/BWR ECCS Evaluation Model codes were used for the LOCA-ECCS calculations for Dresden Unit 3. The EXEM/BWR Codes consist of RODEX2 (7), RELAX (8), FLEX (9), and HUXY/BULGEX (10,11). The latest versions of the approved codes were used for the Dresden Unit 3 9x9 analysis. It was found that some modifications to the FLEX computer code were required to perform the reflood calculation for a 9x9 array fuel.

Specifically array dimensions were increased in order to perform the radiation calculation for the 9x9 fuel assemblies. Other than the required changes necessary to analyze 9x9 fuel assemblies, the calculations used code versions identical or equivalent to the approved versions applied in previous ENC LOCA analyses.

The initial stored energy and fission gas release calculations for fuel at various exposures are performed with the RODEX2 code. The system LOCA depressurization from the time of break until the core spray system has reached rated flow is calculated using the RELAX code. The FLEX code is used to compute the system depressurization from the time of rated core spray flow and to calclate the system refill and time of reflood when significant entrainment occurs at the core midplane during the core reflood process. The HUXY/BULGEX code is used to compute the thermal transient at the midplane of the hot or maximum power assembly using initial conditions from RODEX2 and system boundary conditions from RELAX and FLEX. HUXY/BULGEX also computes clad swelling and rupture and the extent of metal-water reaction.

The RELAX system blowdown calculation determines the reactor system behavior during the initial portion of the system depressurization transient. The RELAX system blowdown nodalization for the Dresden Unit 3 9x9 analysis is shown in Figure 2.1. A separate RELAX/HOT CHANNEL calculation is used to calculate the cladding-to-coolant heat transfer coefficients and coolant thermodynamic properties for the maximum power fuel assembly. This calculation considers one fuel assembly with time-dependent boundary conditions from the RELAX system blowdown results being applied for the reactor vessel upper and lower plenum volumes. The

RELAX/HOT CHANNEL nodalization is given in Figure 2.2. The RELAX system blowdown results also provide initial condition input at the time of rated core spray flow for the FLEX refill/reflood calculation.

The FLEX system refill/reflood analysis calculates the later portion of the system depressurization, reactor vessel lower plenum refill, core reflood, and the time at which the reflooding liquid is entrained to the maximum power plane in the core (time of hot node reflood). The time of hot node reflood is an input parameter to the heatup calculation. Figure 2.3 gives the nodalization used for the FLEX code calculations.

The HUXY/BULGEX heatup calculation uses calculated parameters from RODEX2 (fuel stored energy and fission gas release), RELAX (time of rated spray, decay power, heat transfer coefficients, and coolant conditions) and FLEX (time of hot node reflood) to determine the peak clad temperature (PCT) and the percent oxidation of the cladding. A symmetric center-peaked axial power profile is used. A series of heatup calculations are performed at different burnups, and appropriate exposure-dependent MAPLHGR limits are determined.

The Dresden Unit 3 9x9 fuel LOCA analysis was performed assuming an entire core of ENC 9x9 fuel assemblies. The ENC 9x9 assemblies have been demonstrated to be neutronically and hydraulically compatible with both ENC and NSSS vendor 8x8 fuel assemblies. Dresden Unit 3 reactor system data used in this analysis are given in Table 2.1.

Table 2.1 Dresden Unit 3 Reactor System Data

Primary Heat Output, MW	2577.5*
Total Reactor System Volume, ft^3	20160
Total Reactor Flow Rate, lb/hr	98.0×10^6**
Active Core Flow Rate, lb/hr	88.08×10^6**
Nominal Reactor System Pressure (upper plenum) psia	1,017.**
Core Inlet Enthalpy, Btu/lb	525.3**
Recirculation Loop Flow Rate, lb/hr	17.11×10^6**
Steam Flow Rate, lb/hr	9.95×10^6*
Feedwater Flow Rate, lb/hr	9.95×10^6*
Rated Recirculation Pump Head, ft	570.
Rated Recirculation Pump Speed, rpm	1,670.
Moment of Inertia, lbm-ft^2/rad	10,950.
Recirculation Suction Pipe I.D., in.	25.78
Recirculation Discharge Pipe I.D., in.	25.46
Fuel Assembly Rod Diameter, in***	0.424
Fuel Assembly Rod Pitch, in***	0.572
Active Core Height, in***	145.24

* 102% of rated power
** At 100% of rated flow
*** ENC 9x9 fuel parameters

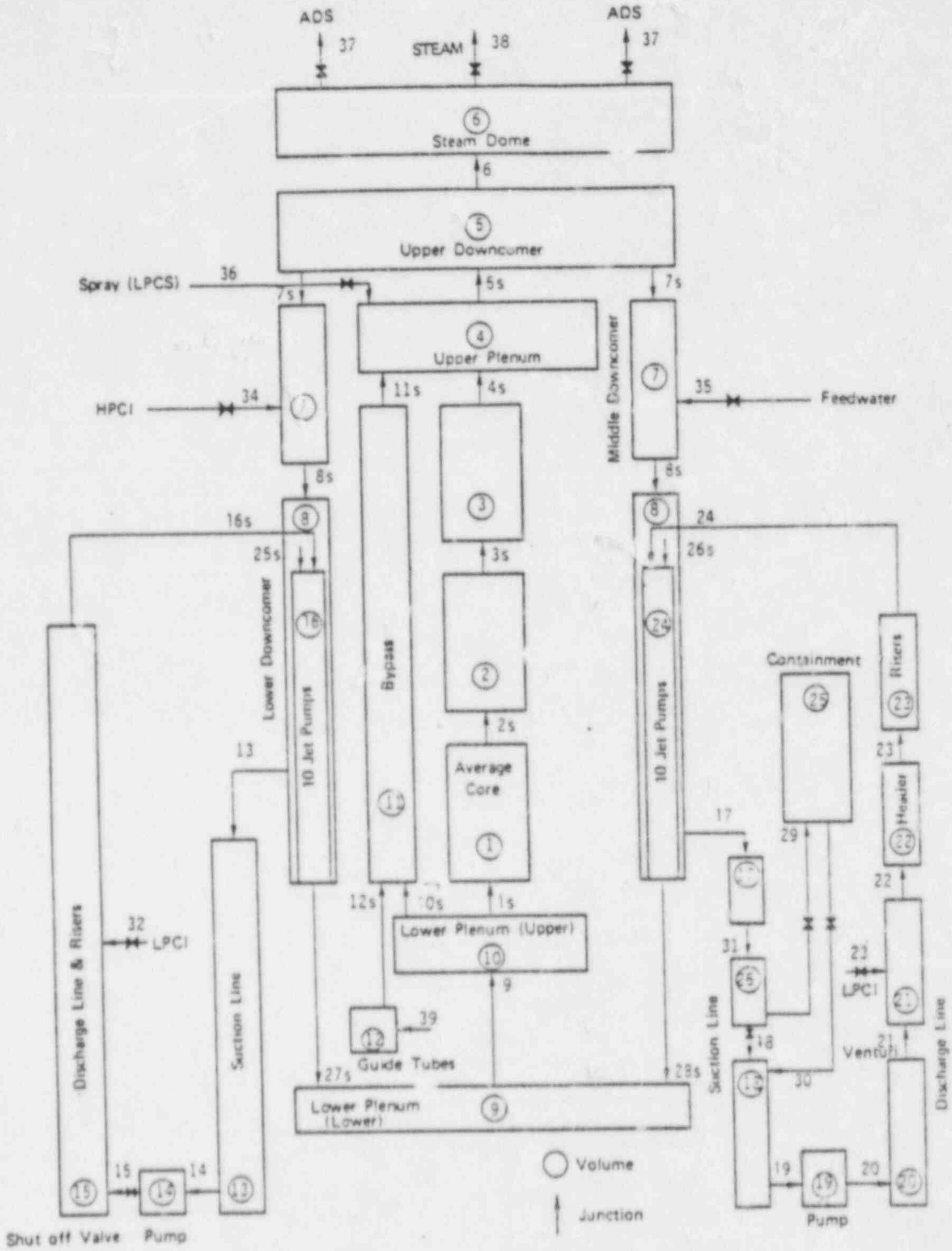

Figure 2.1 System Blowdown Nodalization for Jet-Pump BWR 3

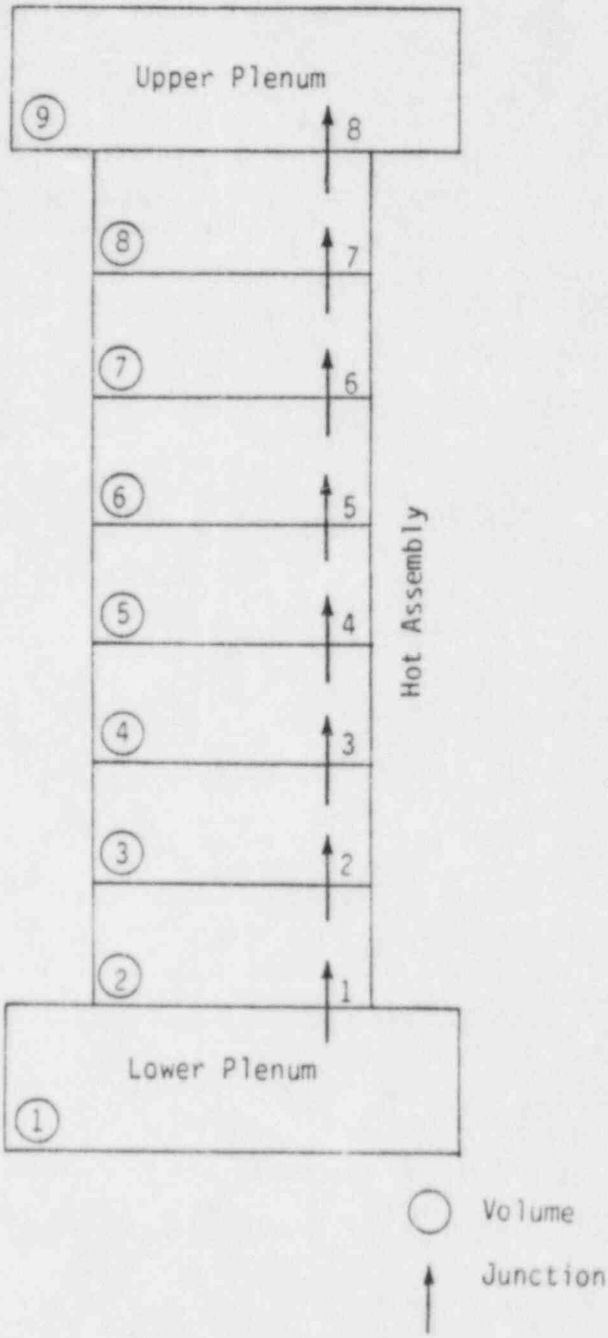

Figure 2.2 Hot Channel Nodalization

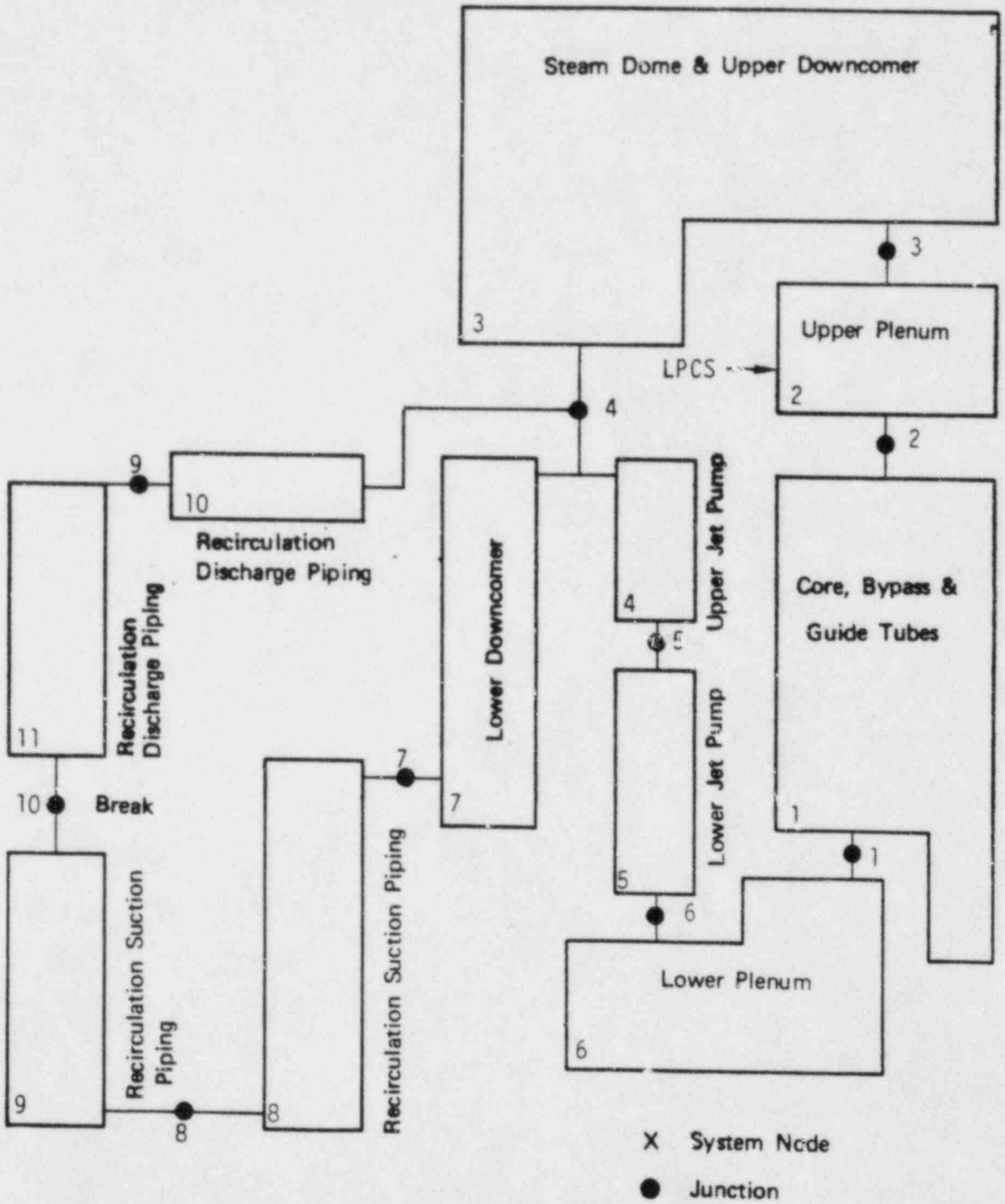

Figure 2.3 System Refill/Reflood Nodalization

3.0 ANALYSIS RESULTS

A complete LOCA-ECCS limiting break calculation was performed for Dresden Unit 3 with a full core of ENC 9x9 fuel. The approved generic break spectrum analysis for the BWR 3 reactor, of which Dresden Unit 3 is typical, identified the limiting LOCA break as the double-ended guillotine break of the recirculation pump suction pipe with a discharge coefficient of 1.0 (1.0 DEG/PS). The LOCA-ECCS calculations for Dresden Unit 3 with 9x9 fuel were performed for this limiting LOCA break.

Average core blowdown calculations were made assuming a full core of ENC 9x9 fuel. In comparing the results of these calculations with those for a full 8x8 core, no significant differences are found in the overall system performance. Event times, for example, changed by less than .5 s (Table 3.3). The results of a mixed core (8x8 and 9x9) blowdown calculation would be expected to be in between and would therefore be nearly identical. It can be concluded that the use of a full 9x9 core (or a full 8x8 core or a mixed core) blowdown calculation for boundary conditions for the hot channel and heatup calculations will not impact results.

The initial reflood calculations made for this analysis utilized a full 9x9 core. There are some significant differences between the results of these calculations and those of the previous reflood calculations made for Cycles 8 and 9 which utilized mixed GE and ENC 8x8 cores. They will not impact the hot channel blowdown calculation since the reflood calculation begins when the blowdown calculation ends. They will impact heatup results. The key parameter that is different is the time of hot node reflood, which advanced from 169 s in the Cycle 9 (mixed GE and ENC

8x8 core) analysis to 160 s in the Cycle 10 calculation (full 9x9 core). The time for a mixed GE 8x8, ENC 8x8 and ENC 9x9 core would be expected to fall in between those times. A later time of hot node reflood allows more time for heatup and therefore a higher PCT. Thus, it can be concluded that the Cycle 9 analysis, which utilized the later time, is bounding for ENC 8x8 fuel in Cycle 10 and future cycles containing a mixed core of ENC 8x8 and ENC 9x9 fuel and that use of the time of hot node reflood from the Cycle 9 analysis in the 9x9 heatup calculations gives results that are bounding for ENC 9x9 fuel in Cycle 10 and future cycles, whether containing a mixed 8x8 and 9x9 core or a full 9x9 core.

Two conditions for full-power operation were evaluated, full power full flow (100/100) and full power and minimum allowed flow of 87 percent (100/87). Both conditions assumed operation at an operating MCPR of 1.33 which results in reduced power for the maximum power assembly when operating at reduced flow conditions. Both calculations were performed with consistent exposure conditions. Table 3.1 gives the calculated PCT results for the two operating conditions. Calculated LOCA transient results for the two operating conditions are nearly the same, with the low-flow (100/87) case giving a slightly higher PCT (3 degrees F) than the full-flow (100/100) case. The more limiting boundary conditions for the full-power low-flow (100/87) operating conditions were then used to verify the exposure dependent MAPLHGR limits. These MAPLHGR limits for worst case operation bound operation within the allowed power-flow operating map.

The NSSS thermal-hydraulic behavior during a LOCA is determined primarily by the LOCA break parameters; break location, break size, and break configuration, together with the system components and geometry. Variations in core parameters produce only secondary effects on the system behavior. Thus, by using bounding core parameters, the LOCA-ECCS limits established by this analysis for ENC 9x9 fuel in Dresden Unit 3 will apply for future cycles unless significant changes are made in the plant operating conditions, plant hardware, or core design such that the analysis no longer bounds the plant conditions.

Calculated event time results and LOCA-ECCS results for the limiting break and worst case conditions are given in Tables 3.2 and 3.3. The results of Table 3.2 (metal-water reaction and peak clad temperature) are from the low flow case since these are more bounding. The full flow case done at the same MAPLHGR limits resulted in lower PCTs and therefore these limits apply to both cases. System blowdown results are presented in Figures 3.1 through 3.19. System refill and reflood results are given in Figures 3.20 through 3.22. These system conditions are used as boundary conditions for a series of exposure dependent maximum power assembly heatup calculations. Results from a RELAX/HOT CHANNEL calculation are given in Figures 3.23 through 3.25. Typical clad temperature as calculated by HUXY/BULGEX are shown in Figure 3.26. The time of hot node reflood used in the heatup calculations is the bounding (later) time from the Cycle 9 analysis.

The final MAPLHGR calculation results from HUXY/BULGEX were given in Table 1.1, Figure 1.1, and Table 3.2. Table 3.2 gives the analyzed MAPLHGR, local metal-water reaction, and peak cladding temperature as a function of the hot assembly average burnup. These LOCA-ECCS results are in conformance to the U. S. NRC 10 CFR 50.46 criteria.

It should be noted that the analyzed MAPLHGR values chosen correspond to the ENC 9x9 fuel design limit for REMACCX, Ref. 6, Figure 1, and therefore significant margin exists to the LOCA-ECCS 2200 F limit at these MAPLHGR values.

Table 3.1 Dresden Unit 3 Operating Conditions Comparison

	100% Flow	87% Flow
Peak Cladding Temperature, F	2042	2045
Local Zr/H2O Reaction (Max), %	2.41	2.44

Table 3.2 Dresden Unit 3 LOCA Analysis Results For ENC 9x9 Reload Fuel (Types XN-3 and XN-3A) Normal Operating Conditions

Assembly Average Burnup (GWD/MTM)	MAPLHGR (kW/ft)	Local MWR (%)	PCT (F)
0.	11.40	2.20	2006.
5.	11.75	2.44	2045.
10.	11.40	0.91	1893.
15.	10.55	0.63	1805.
20.	9.70	0.44	1710.
25.	8.85	0.29	1623.
30.	8.00	0.18	1529.
35.	7.15	0.12	1421
40.	6.30	0.08	1309

Table 3.3 Dresden 3 9x9 Limiting Break Event Times

Event	Time (sec)
Start	0.00
Initiate Break	0.05
Feedwater Flow Stops	0.55
Steam Flow Stops	5.05
Low Low Mixture Level	4.5
Jet-Pumps Uncover	7.6
Recirculation Pipe Uncovers	10.8
Lower Plenum Flashes	12.0
HPCI Flow Starts	14.5
LPCS Starts	37.3
Rated Spray Calculated	59.7
Depressurization Ends	116.2
Start of Reflood	142.
Time of Hot Node Reflood	
a. Cycle 9 Analysis (mixed GE and Enc 8x8 core)	169.0
b. Cycle 10 Analysis (Full 9x9 core)	160.0
Peak Clad Temperature Reached	169.0

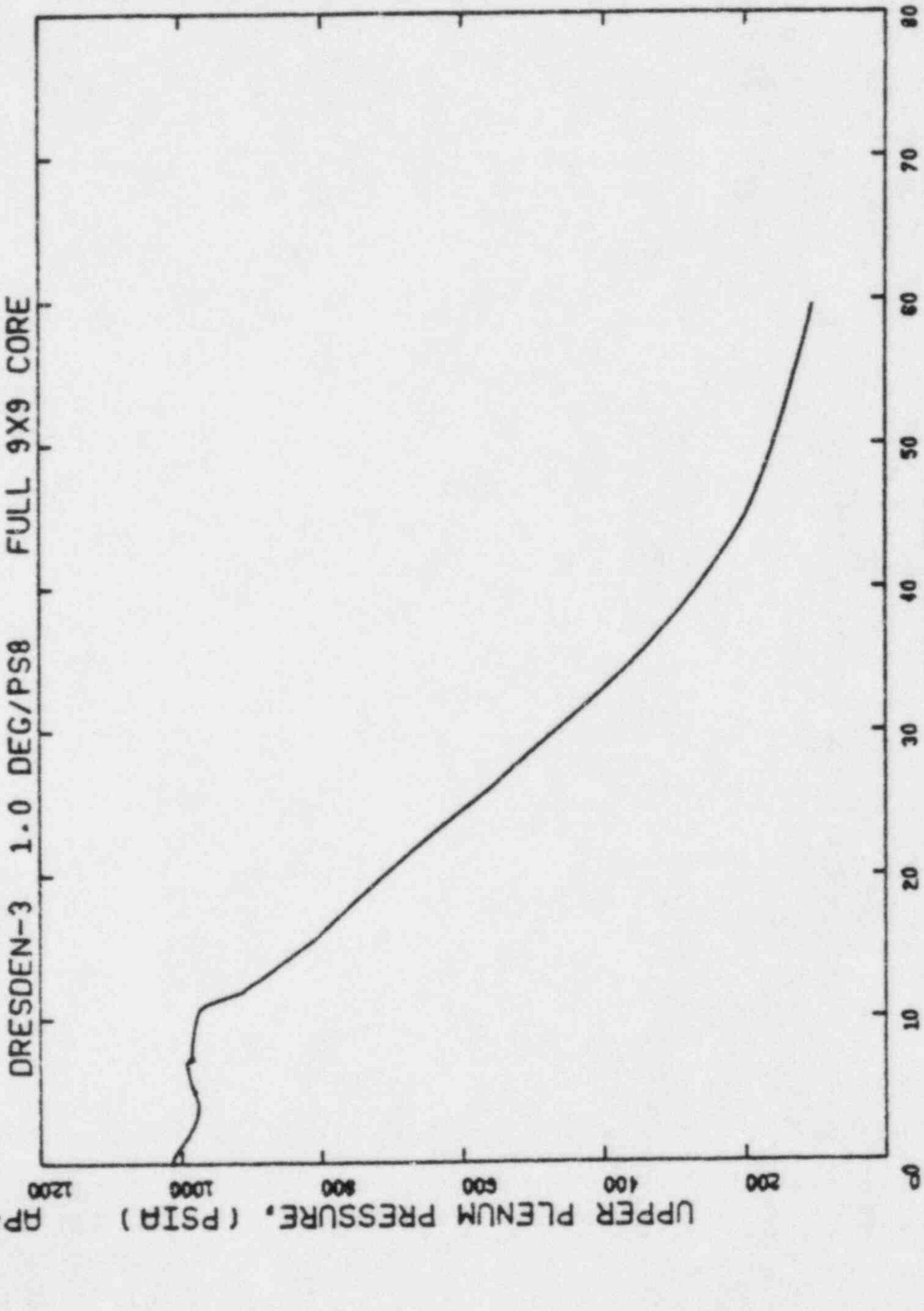

Figure 3.1 Blowdown System Pressure

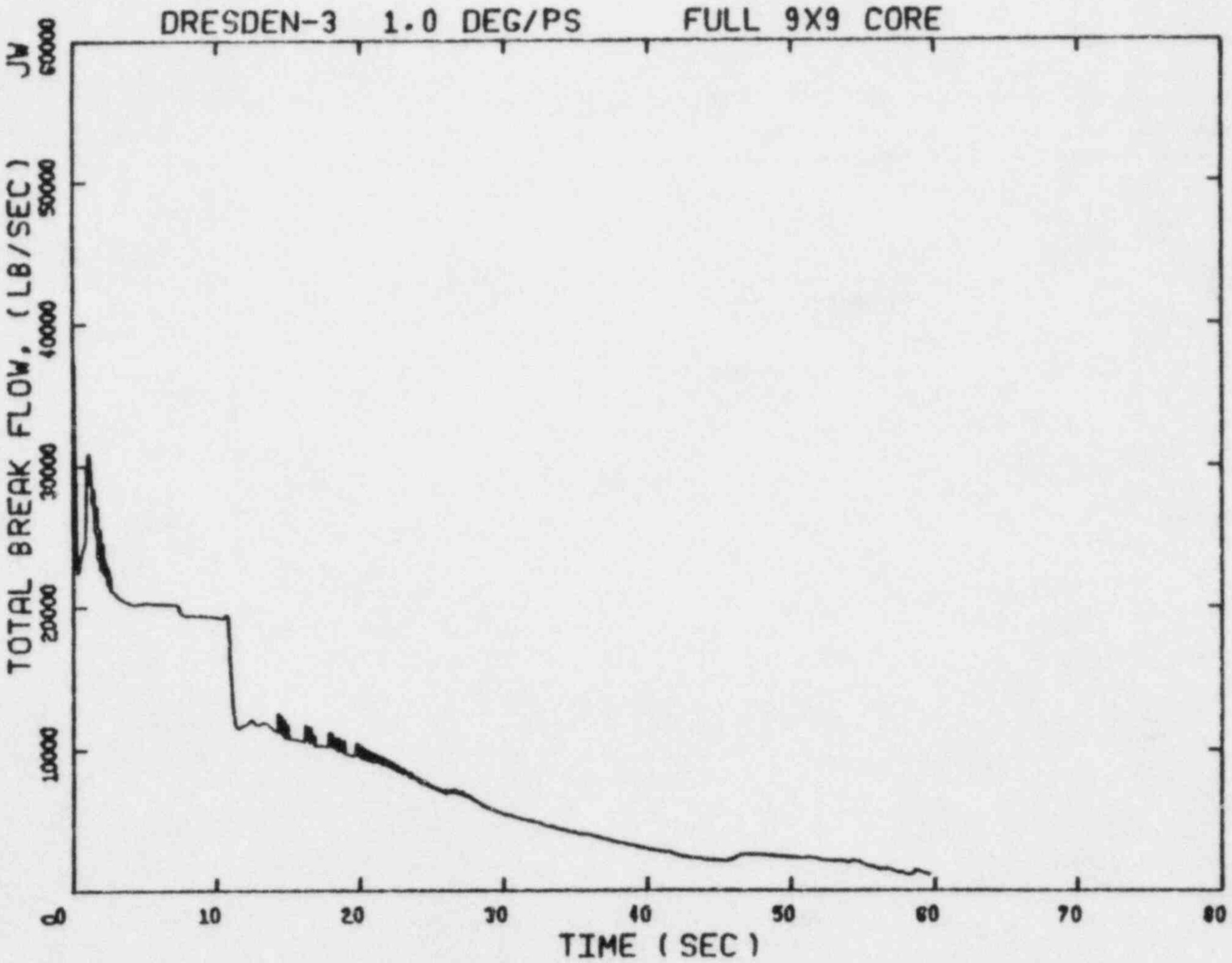

Figure 3.2 Blowdown Total Break Flow

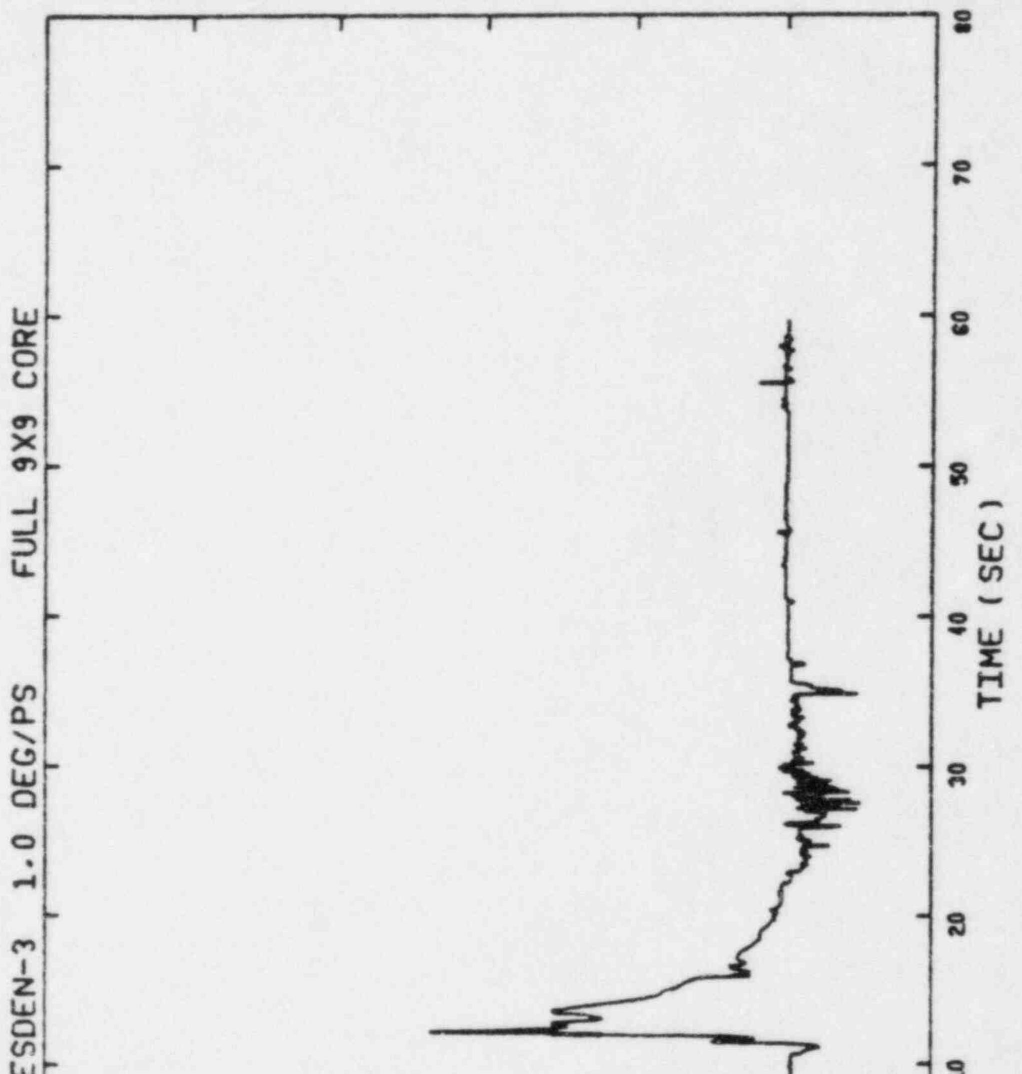

Figure 3.3 Average Core Inlet Flow

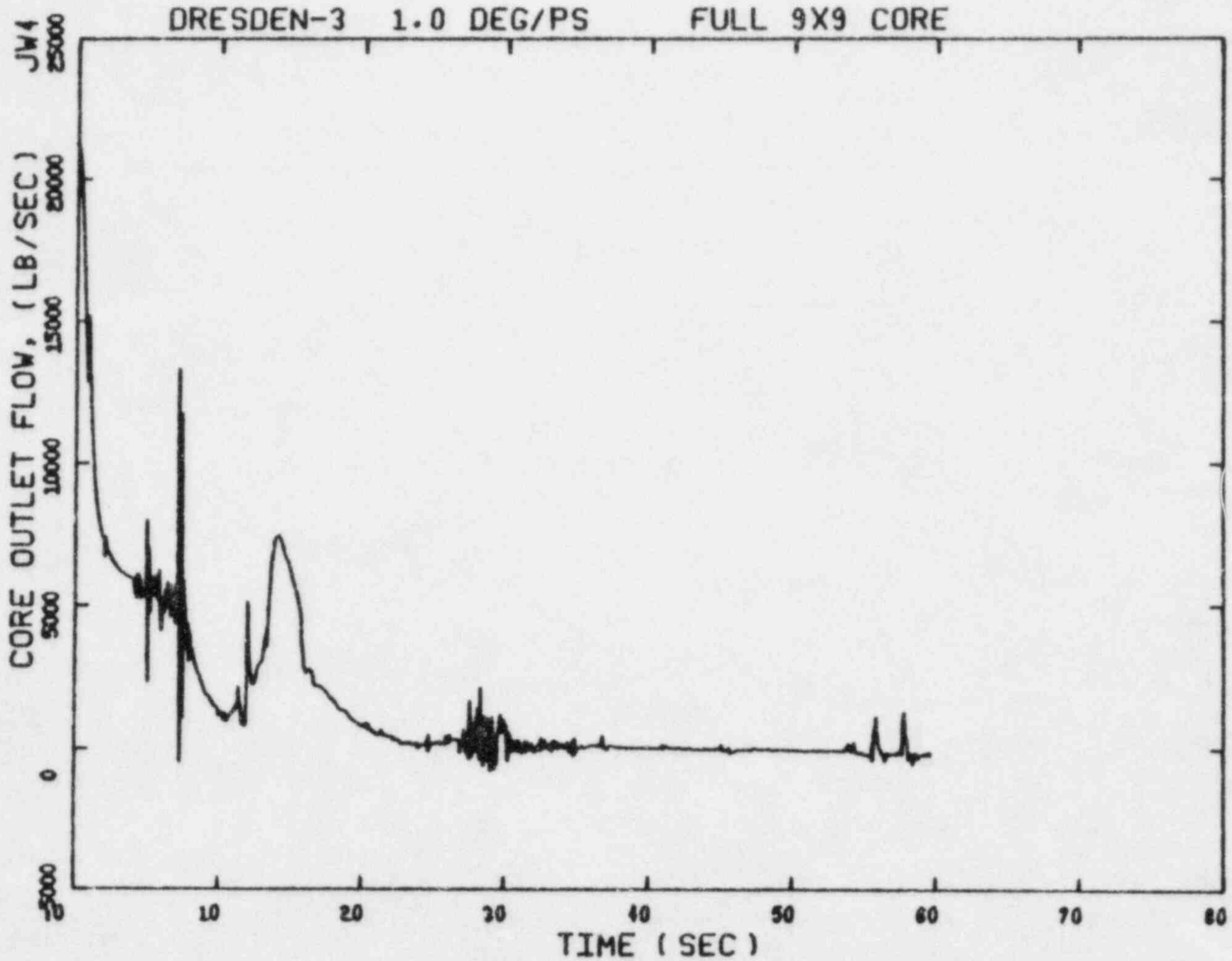

Figure 3.4 Blowdown Average Core Outlet Flow

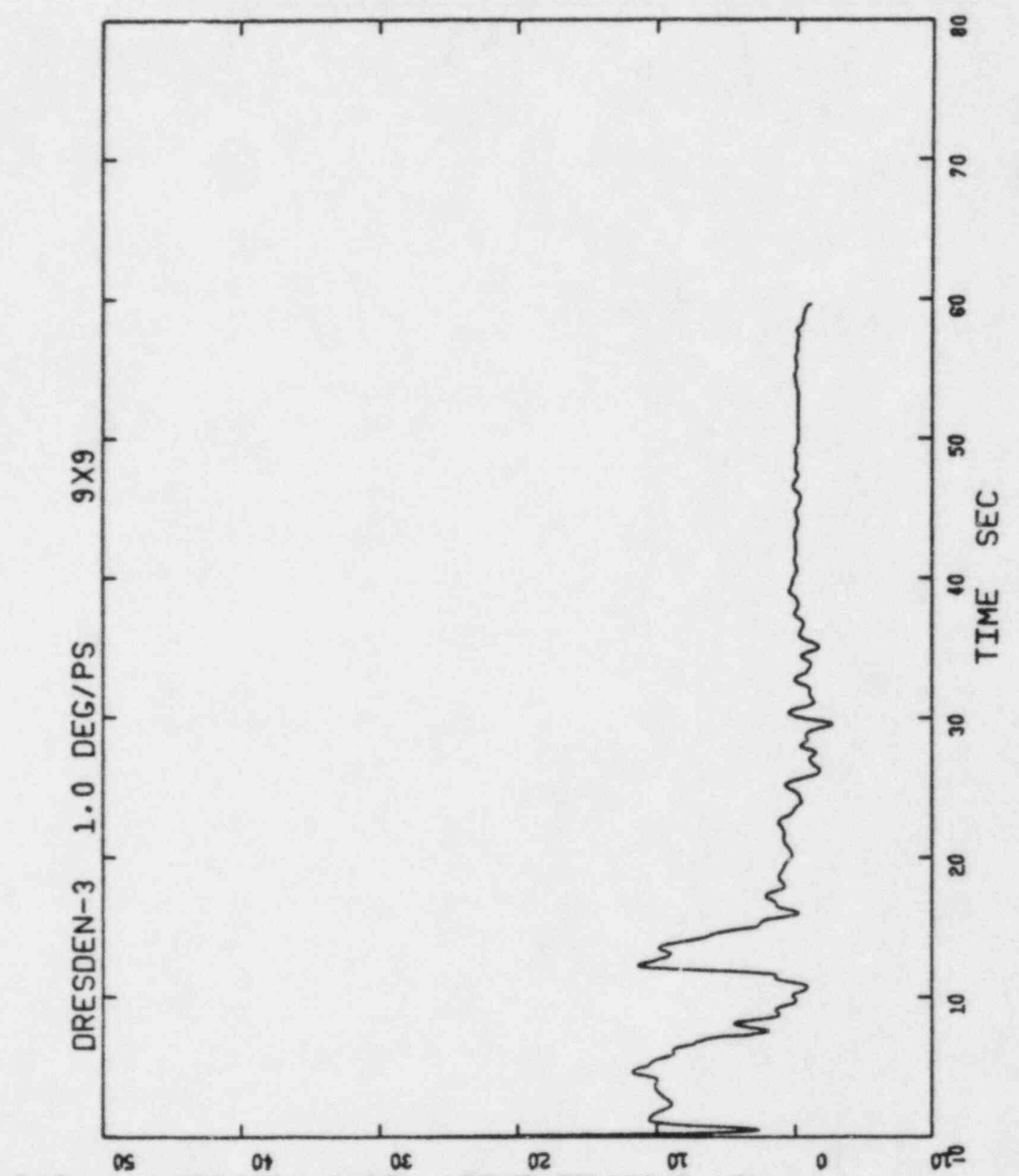

Figure 3.5 Blowdown HOT CHANNEL Inlet Flow

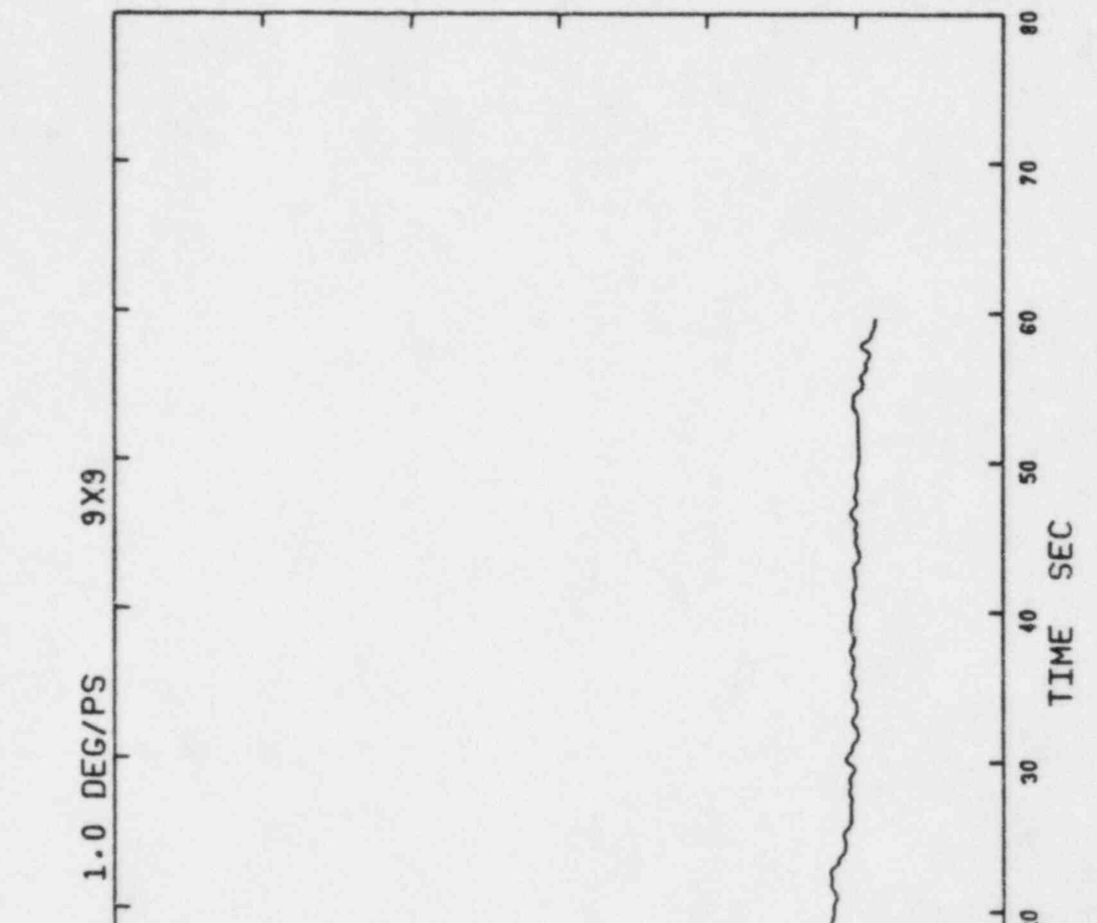

Figure 3.6 Blowdown HOT CHANNEL Outlet Flow

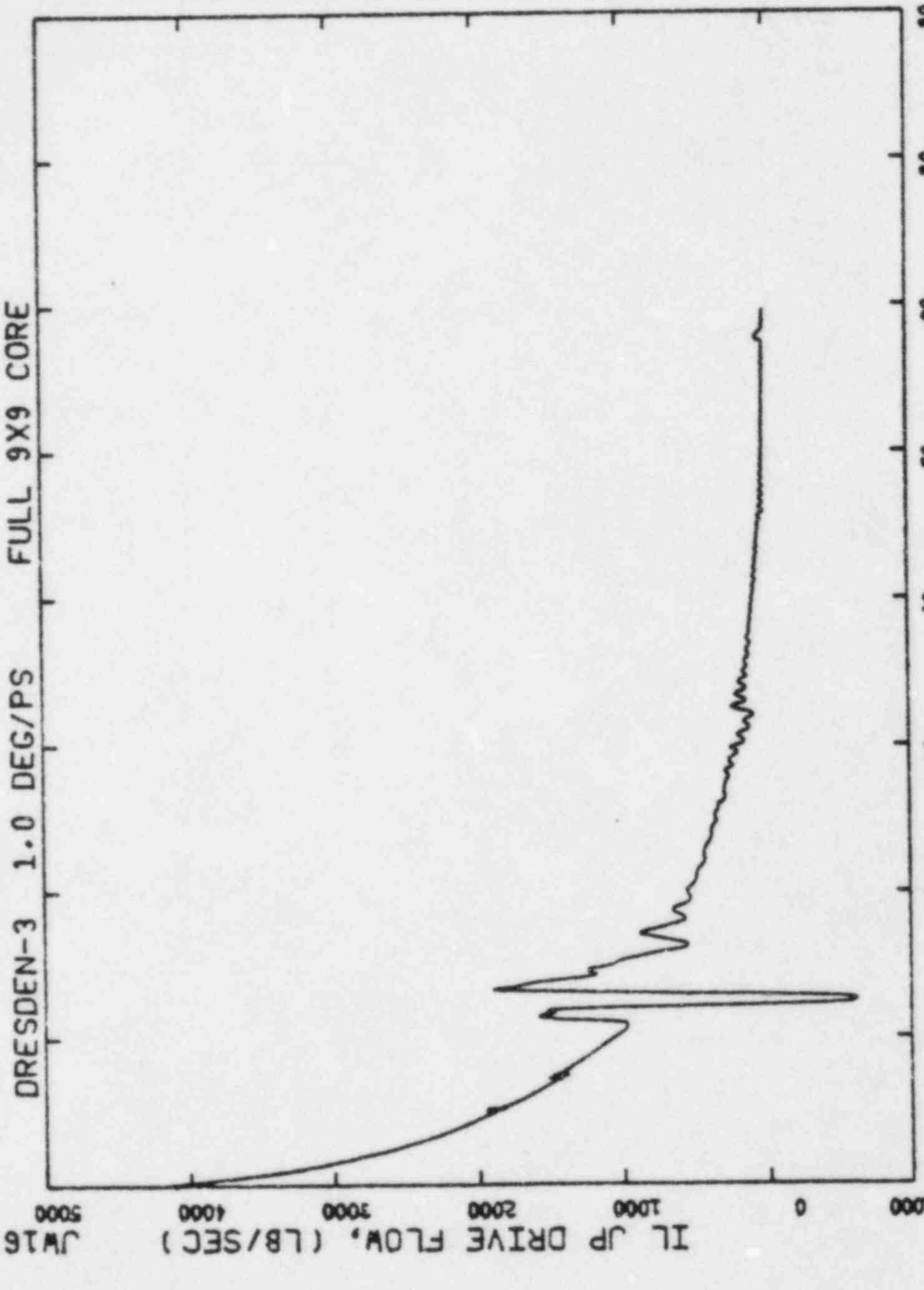

Figure 3.7 Blowdown Intact Loop Jet Pump Drive Flow

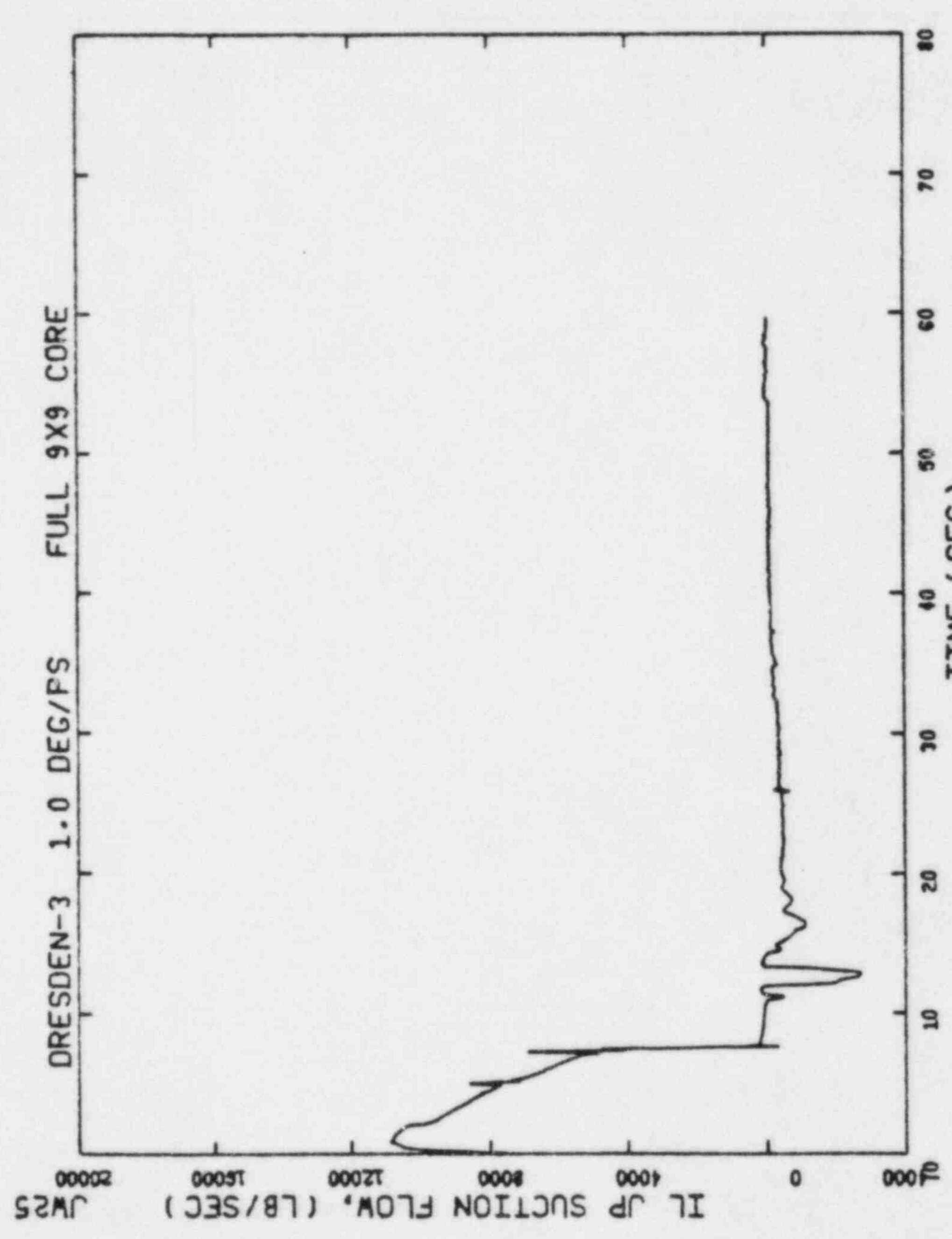

Figure 3.8 Blowdown Intact Loop Jet Pump Suction Flow

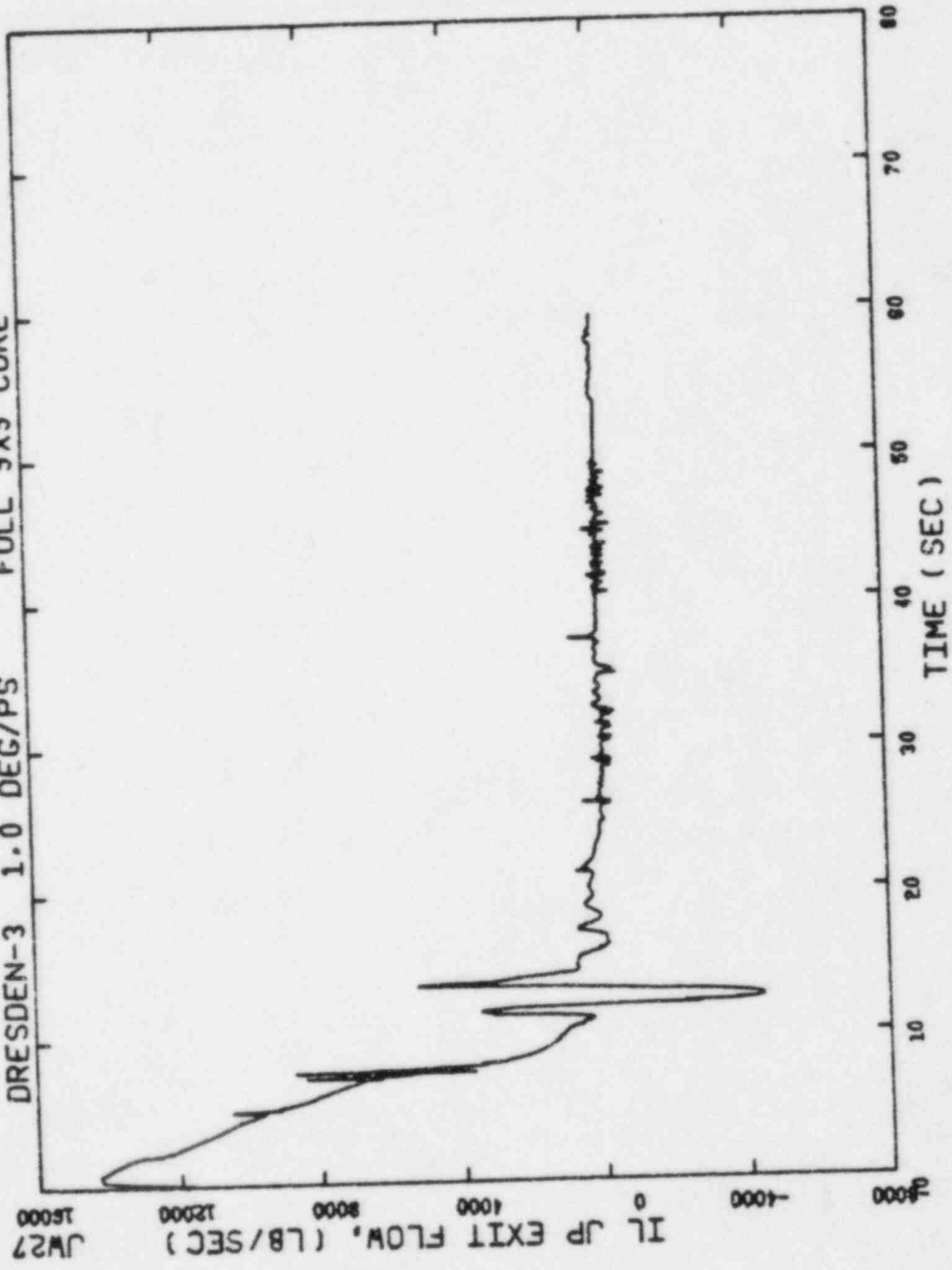

Figure 3.9 Blowdown Intact Loop Jet Pump Exit Flow

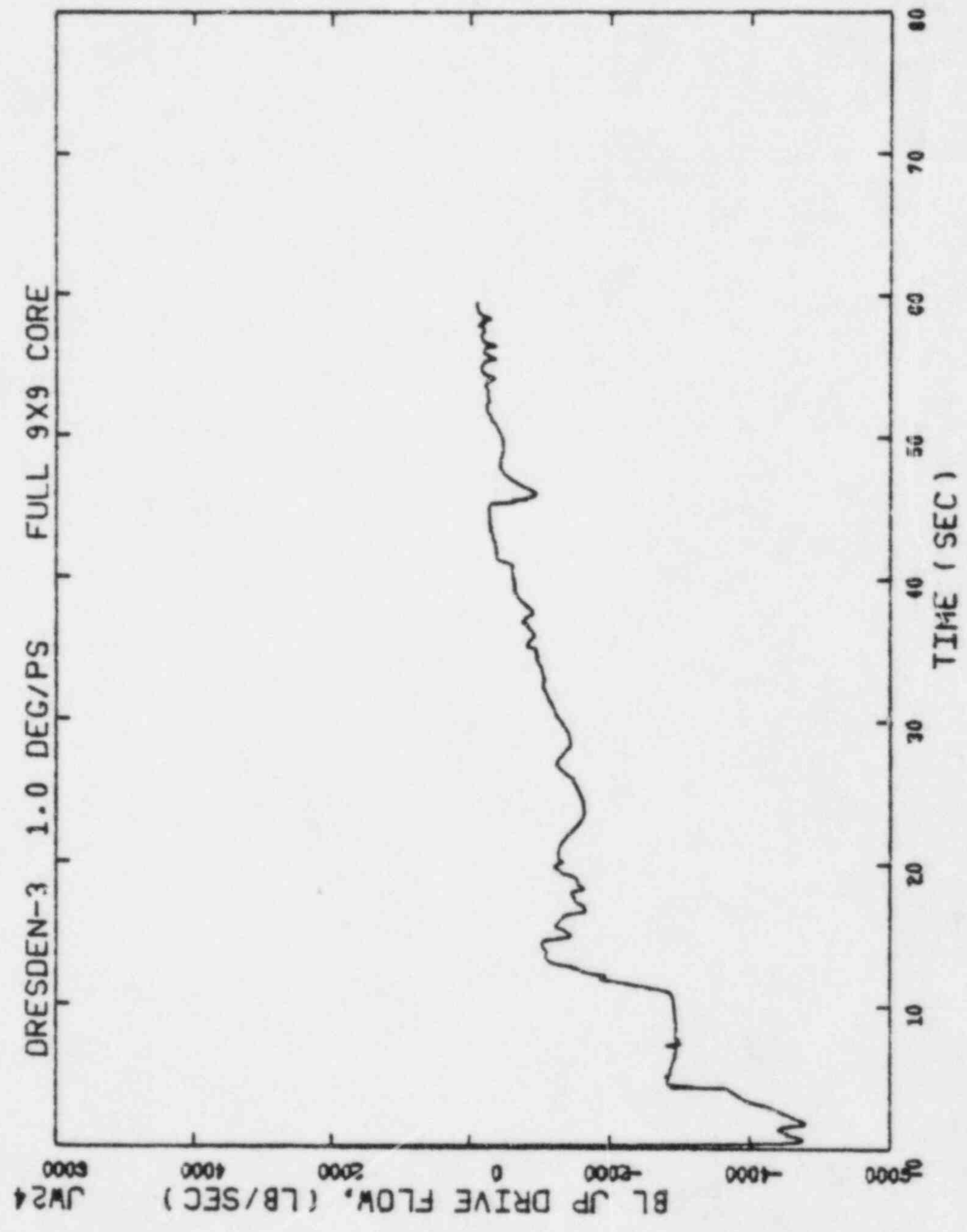

Figure 3.10 Blowdown Broken Loop Jet Pump Drive Flow

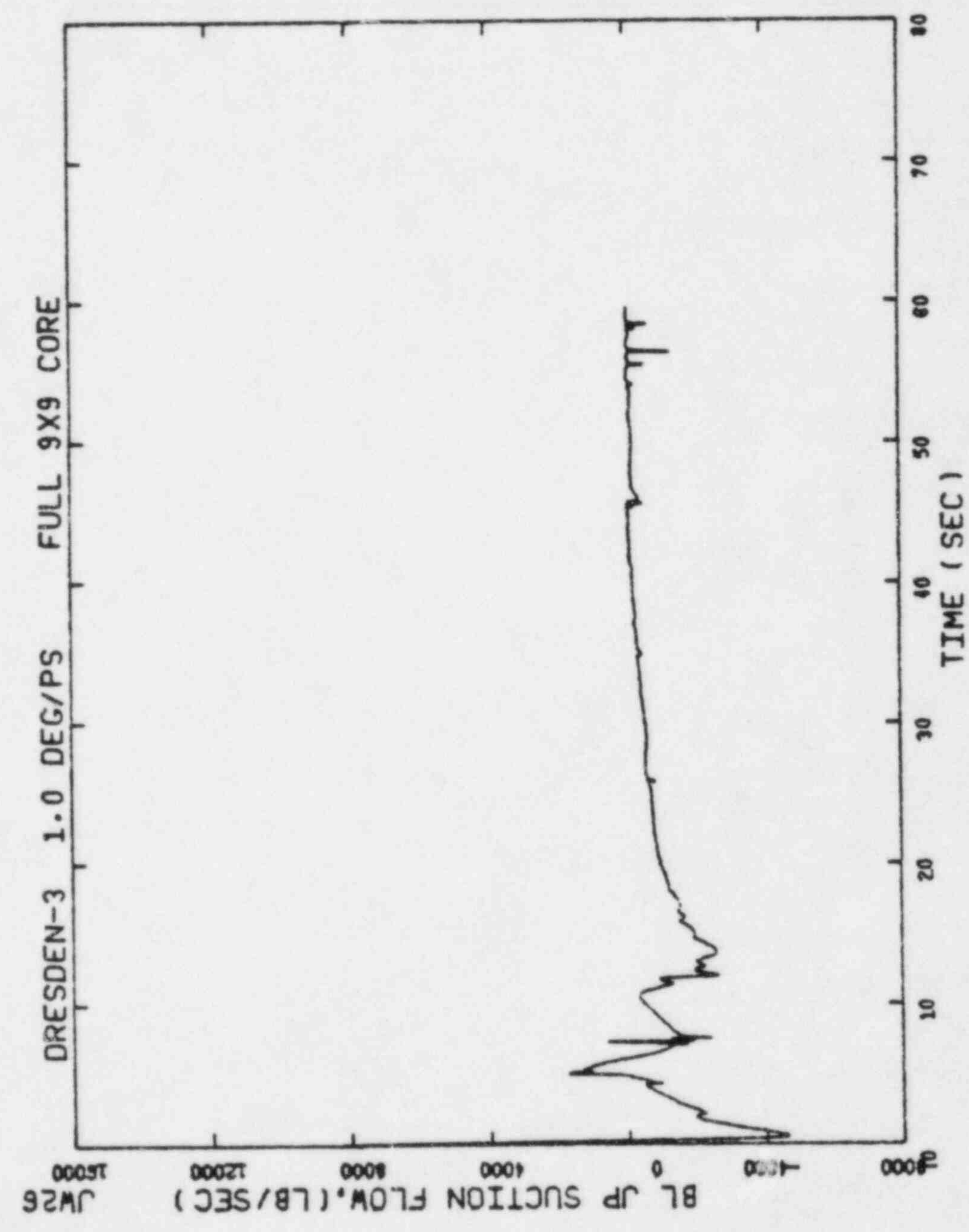

Figure 3.11 Blowdown Broken Loop Jet Pump Suction Flow

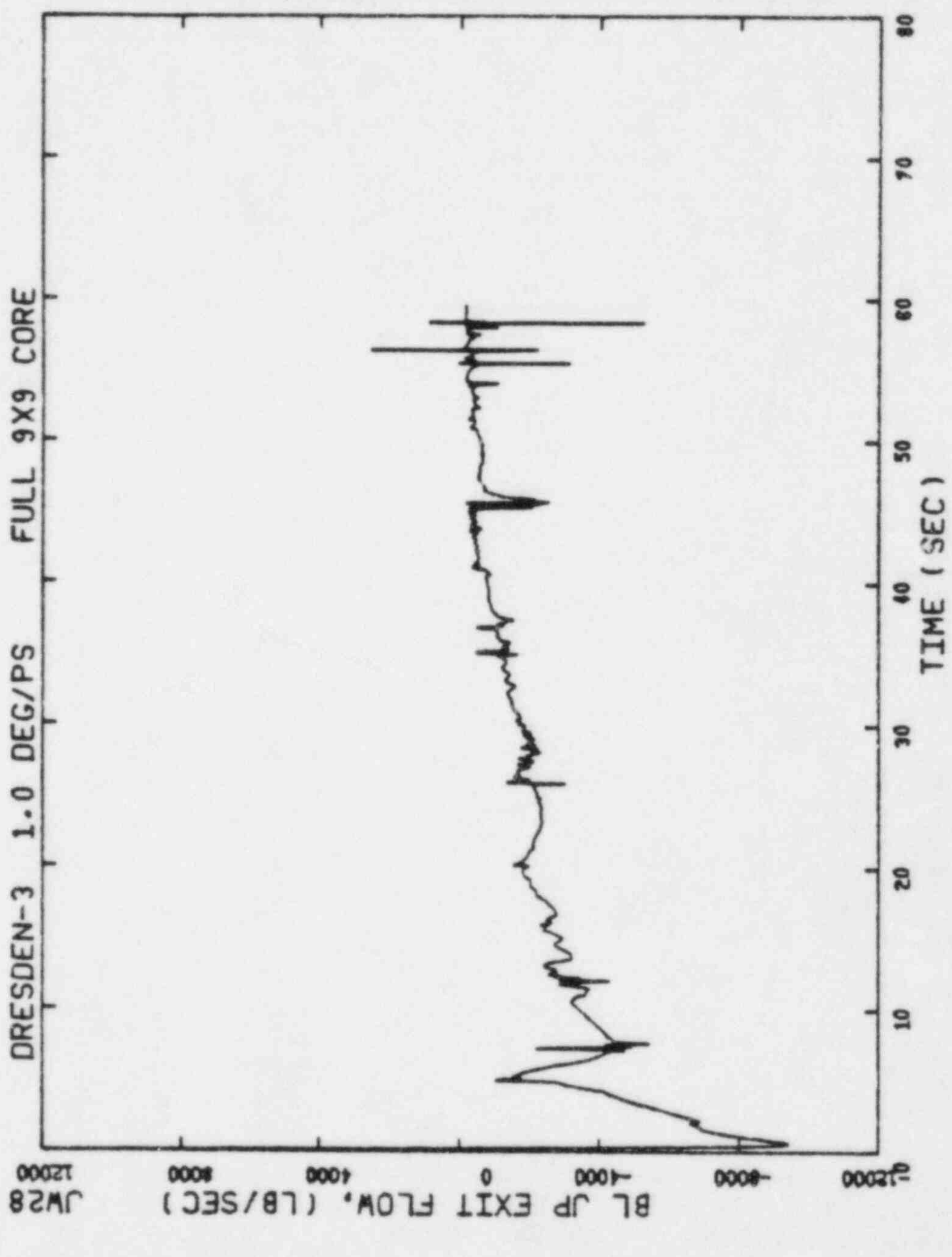

Figure 3.12 Blowdown Broken Loop Jet Pump Exit Flow

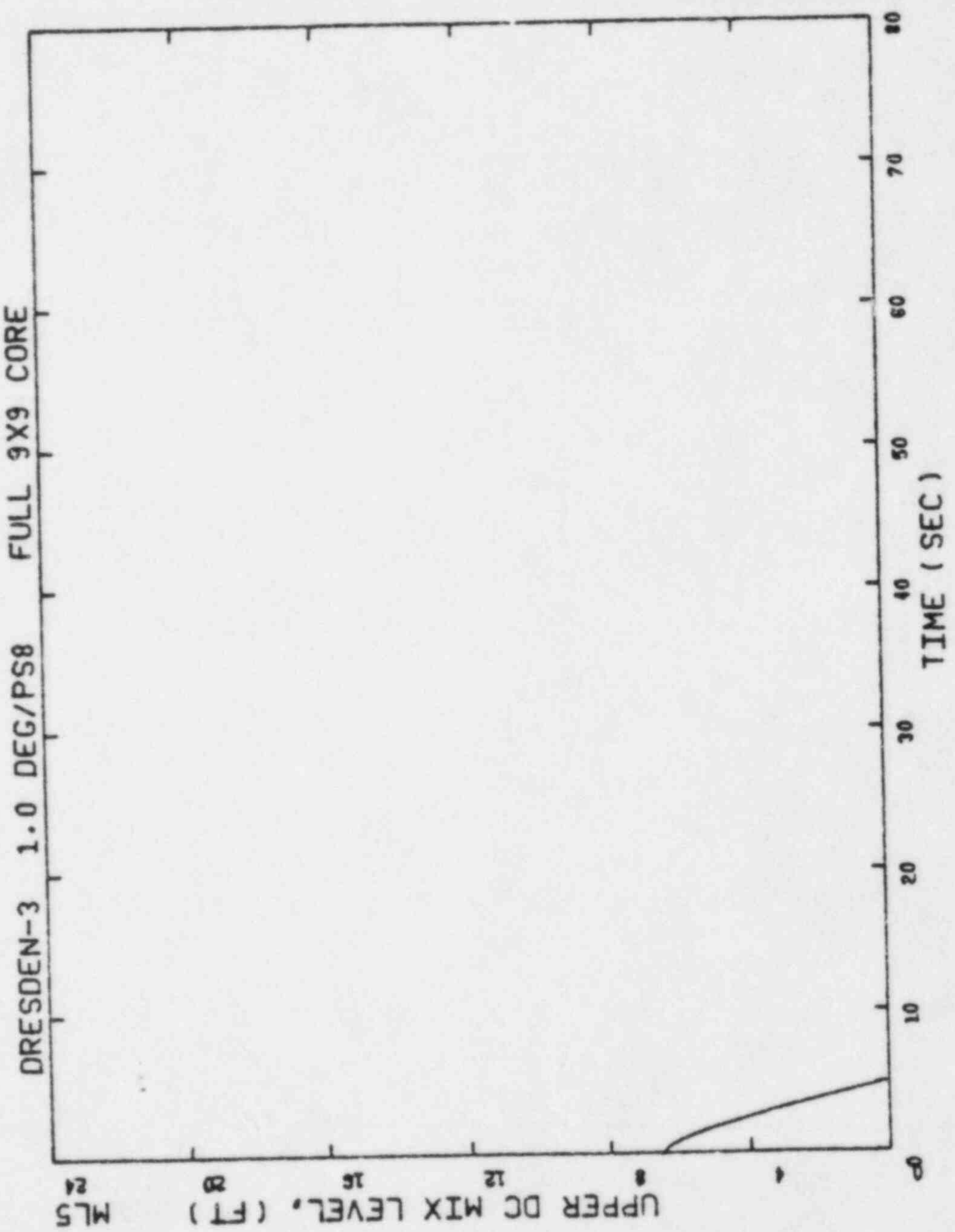

Figure 3.13 Blowdown Upper Downcomer Mixture Level

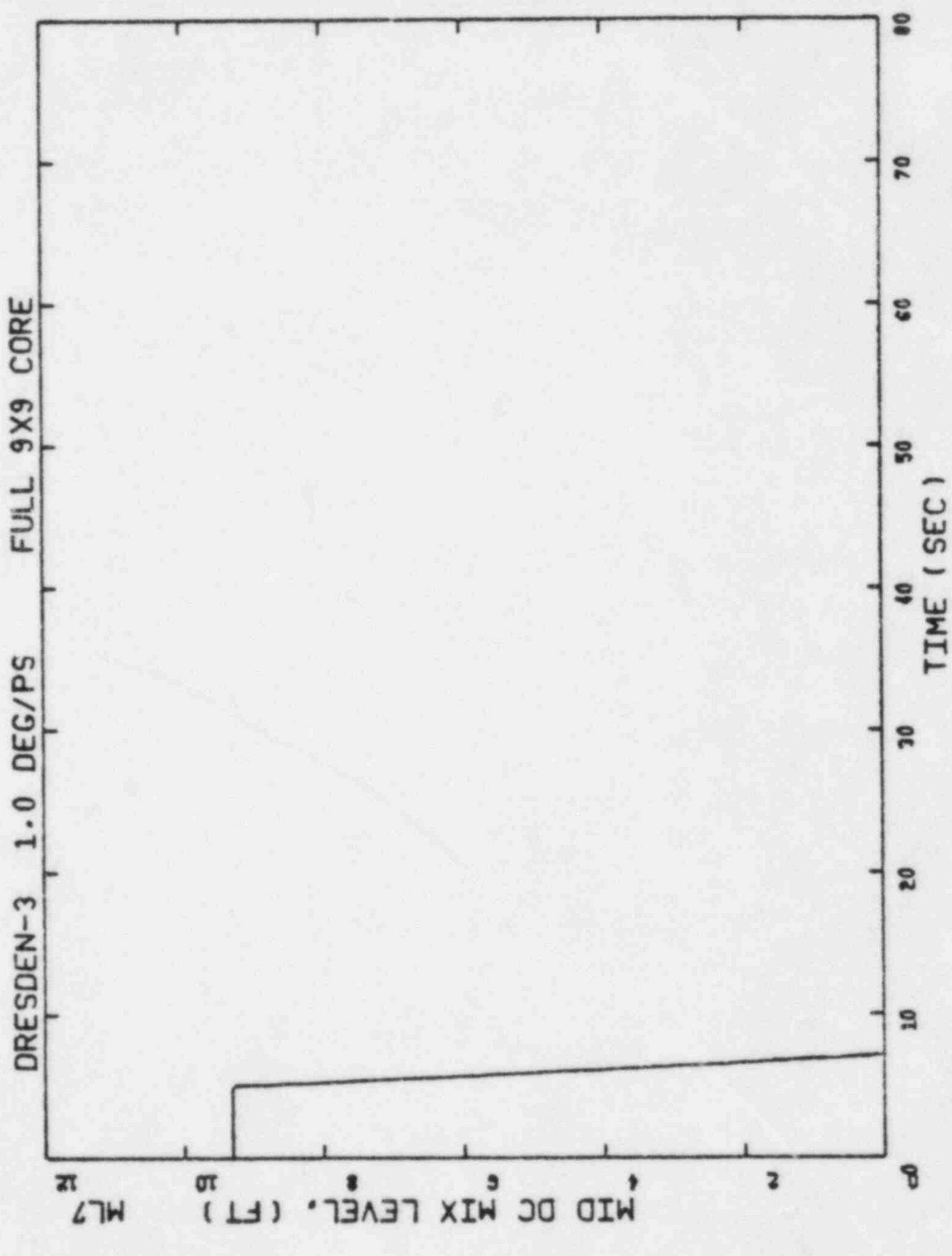

Figure 3.14 Blowdown Middle Downcomer Mixture Level

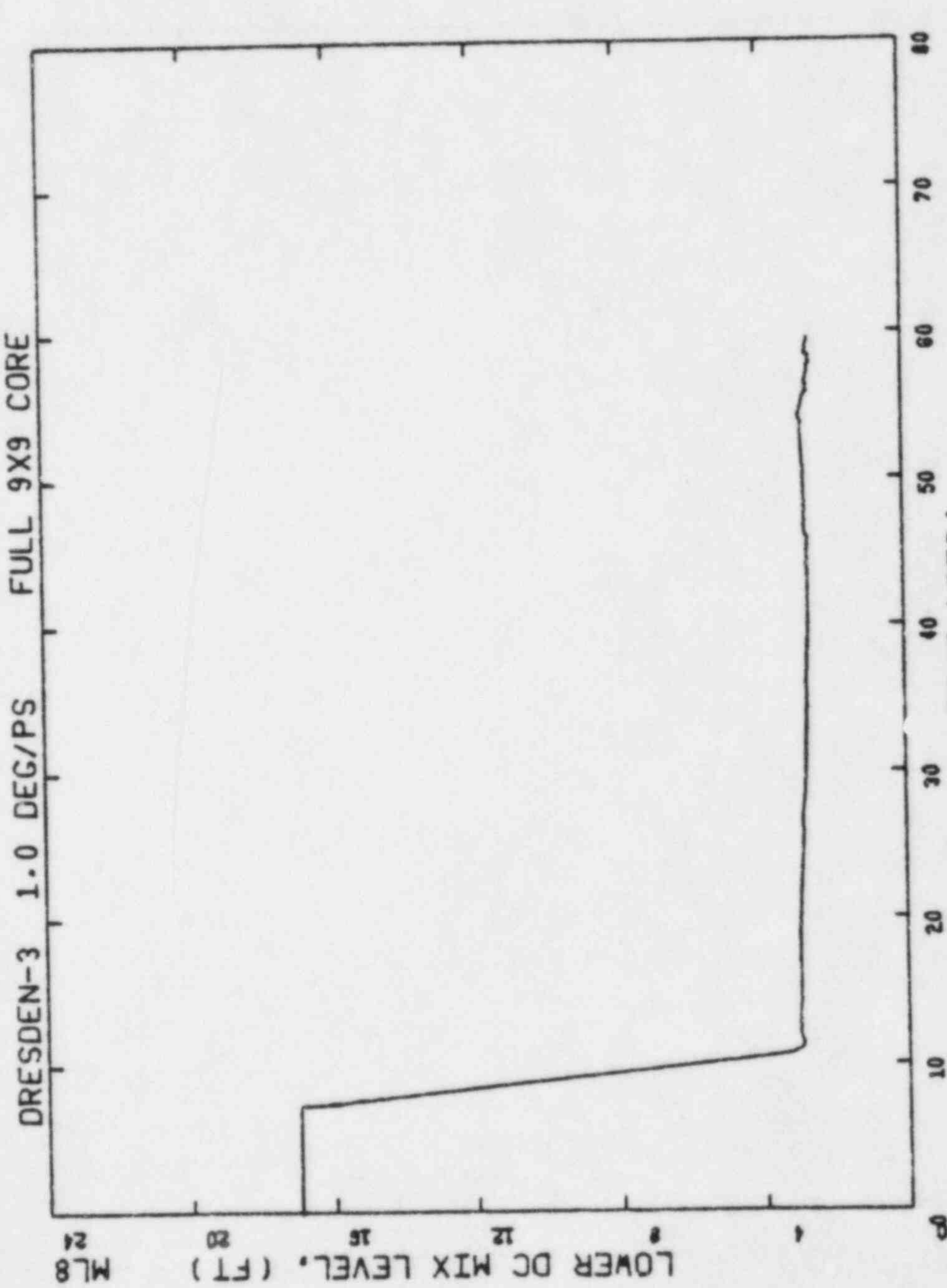

Figure 3.15 Blowdown Lower Downcomer Mixture Level

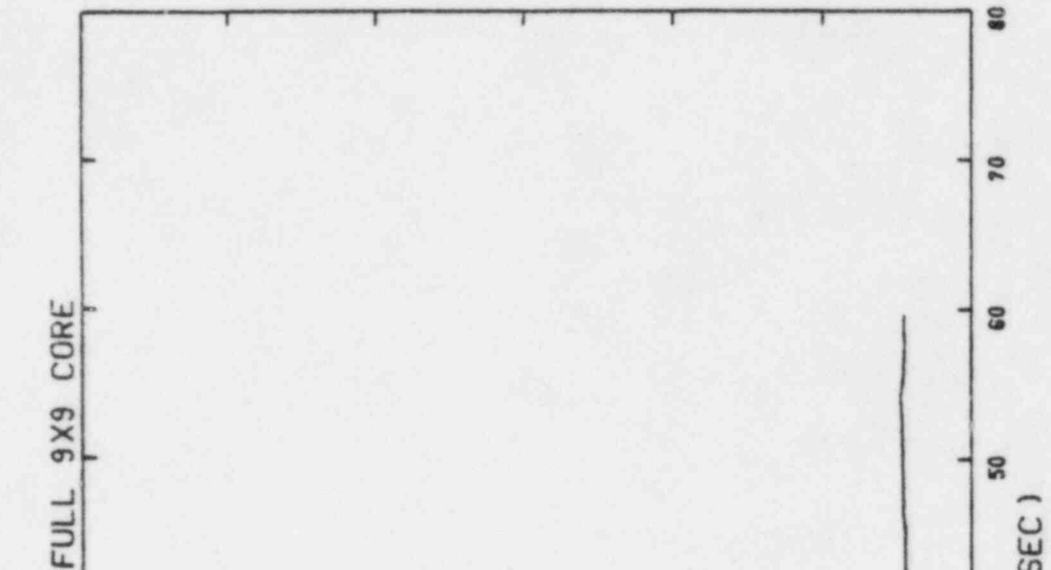

Figure 3.16 Blowdown Lower Downcomer Liquid Mass

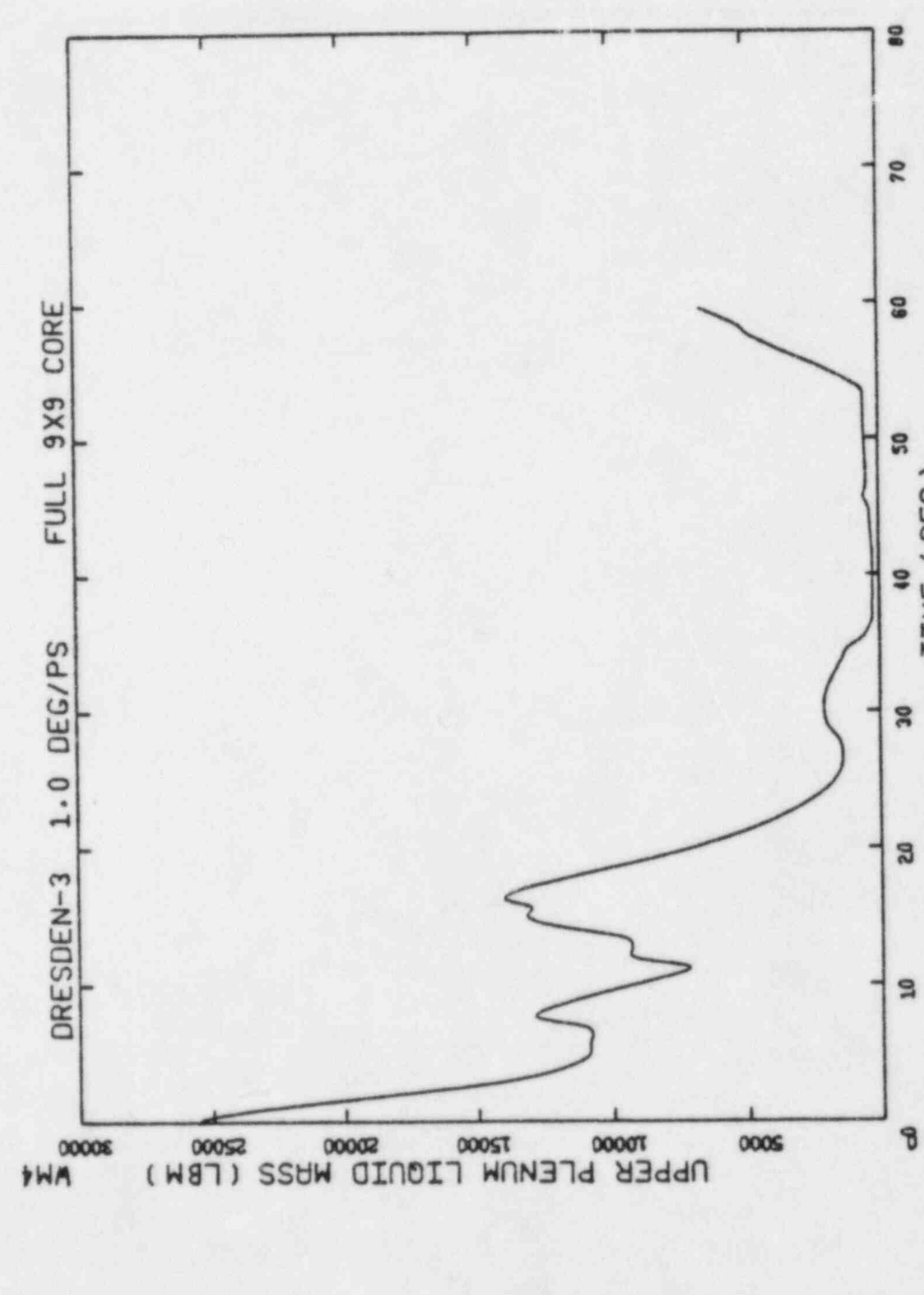

Figure 3.17 Blowdown Upper Plenum Liquid Mass

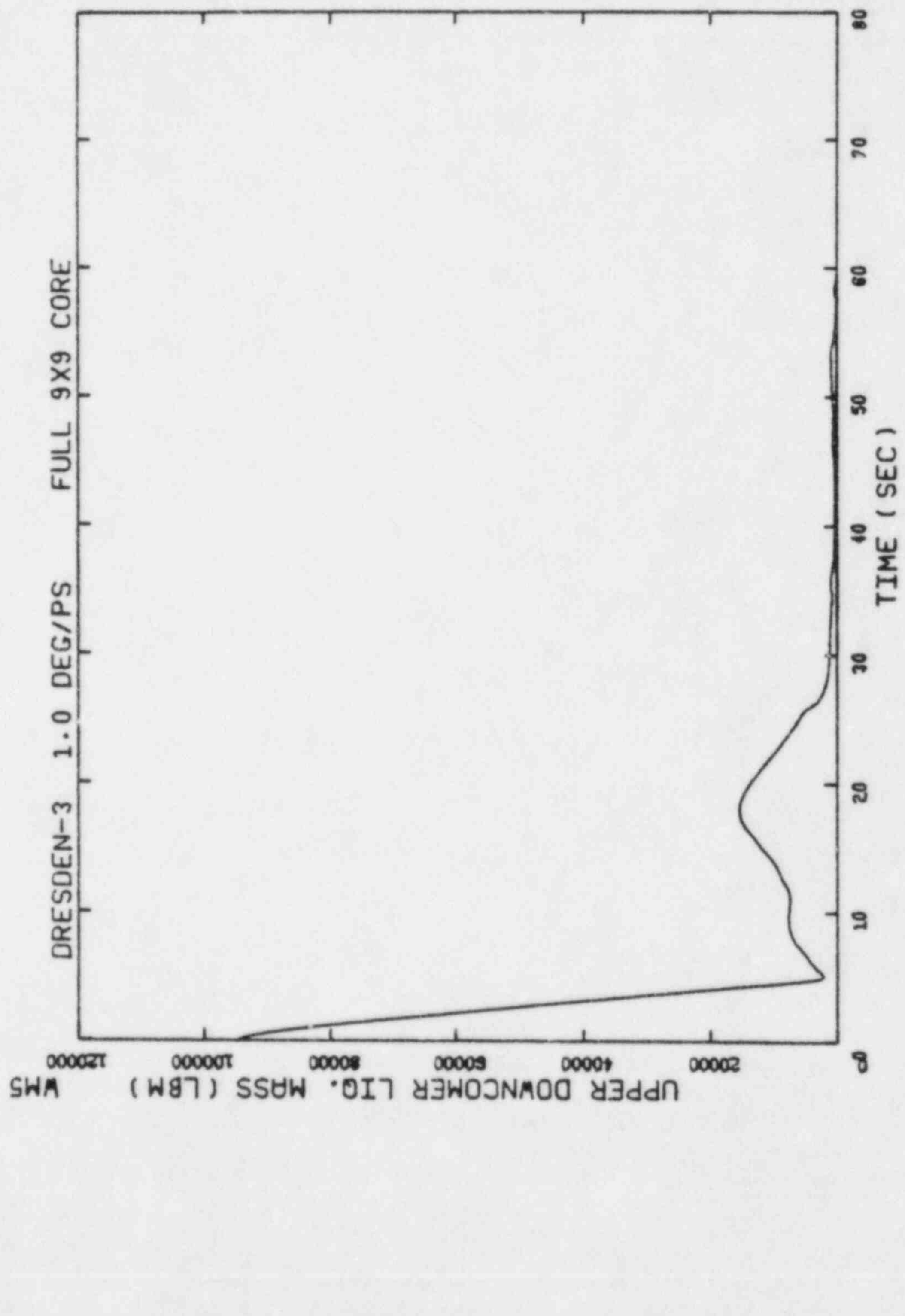

Figure 3.18 Blowdown Upper Downcomer Liquid Mass

Figure 3.19 Blowdown Lower Plenum Liquid Mass

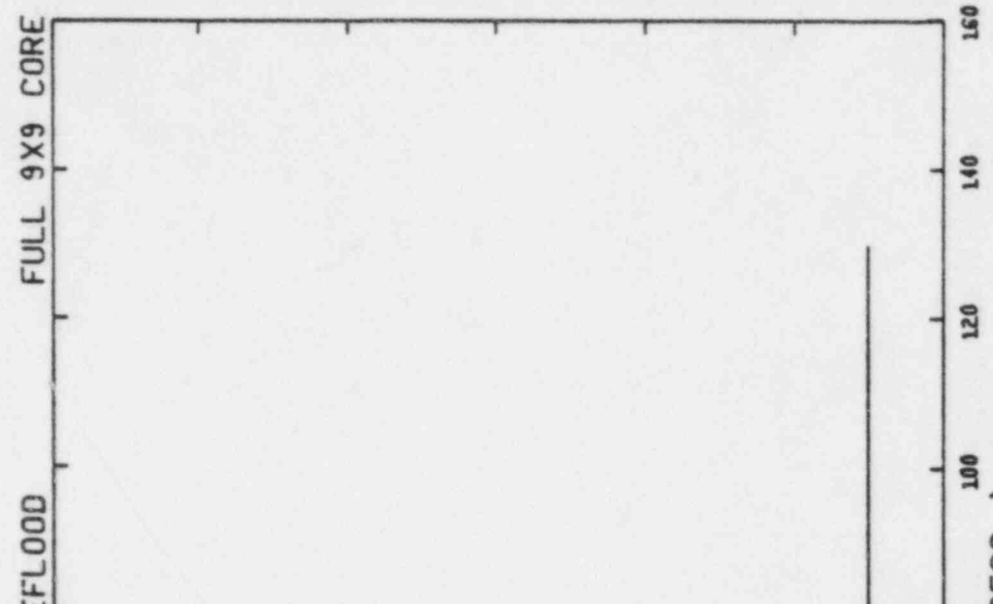

Figure 3.20 Refill/Reflood System Pressure

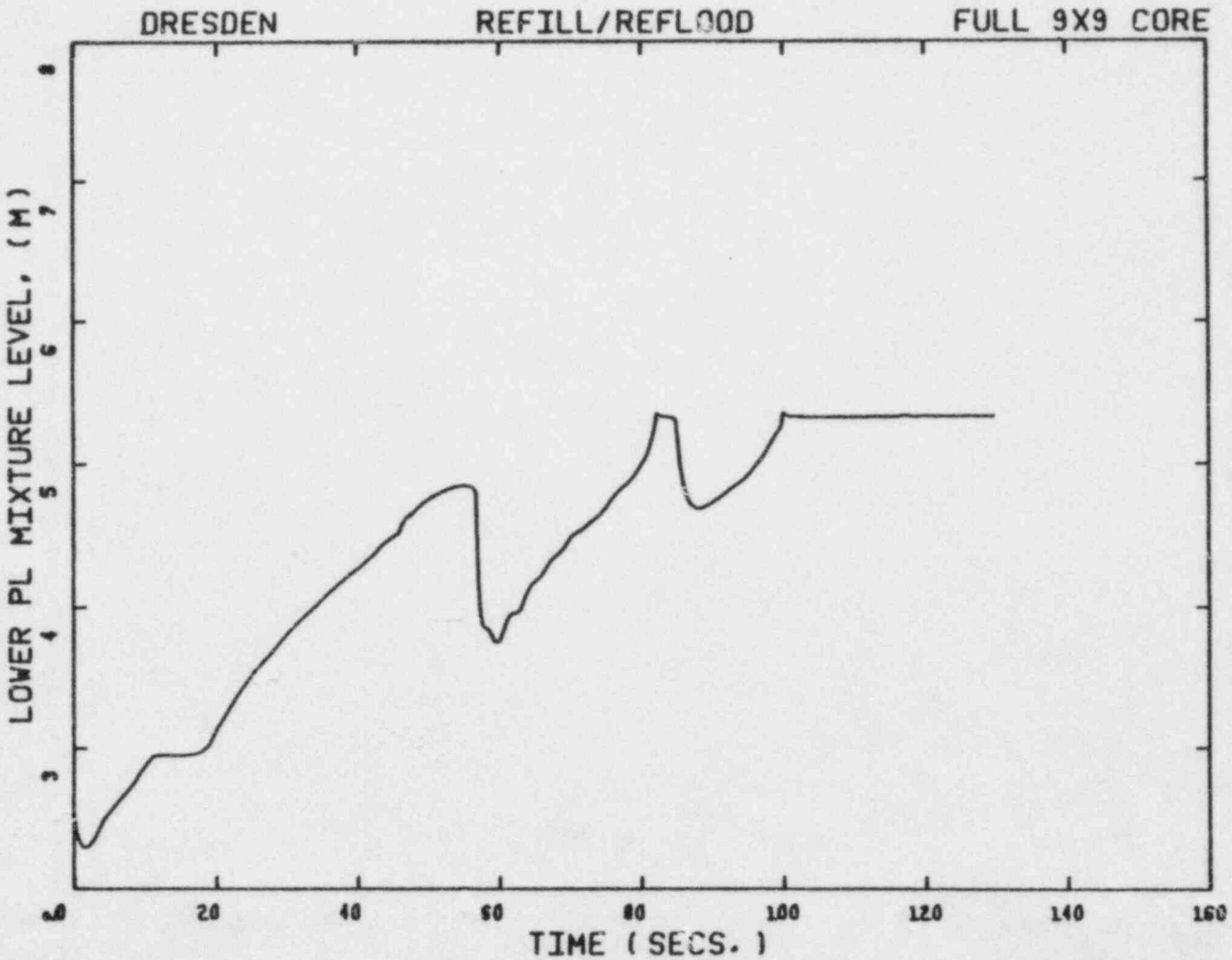

Figure 3.21 Refill/Reflood Lower Plenum Mixture Level

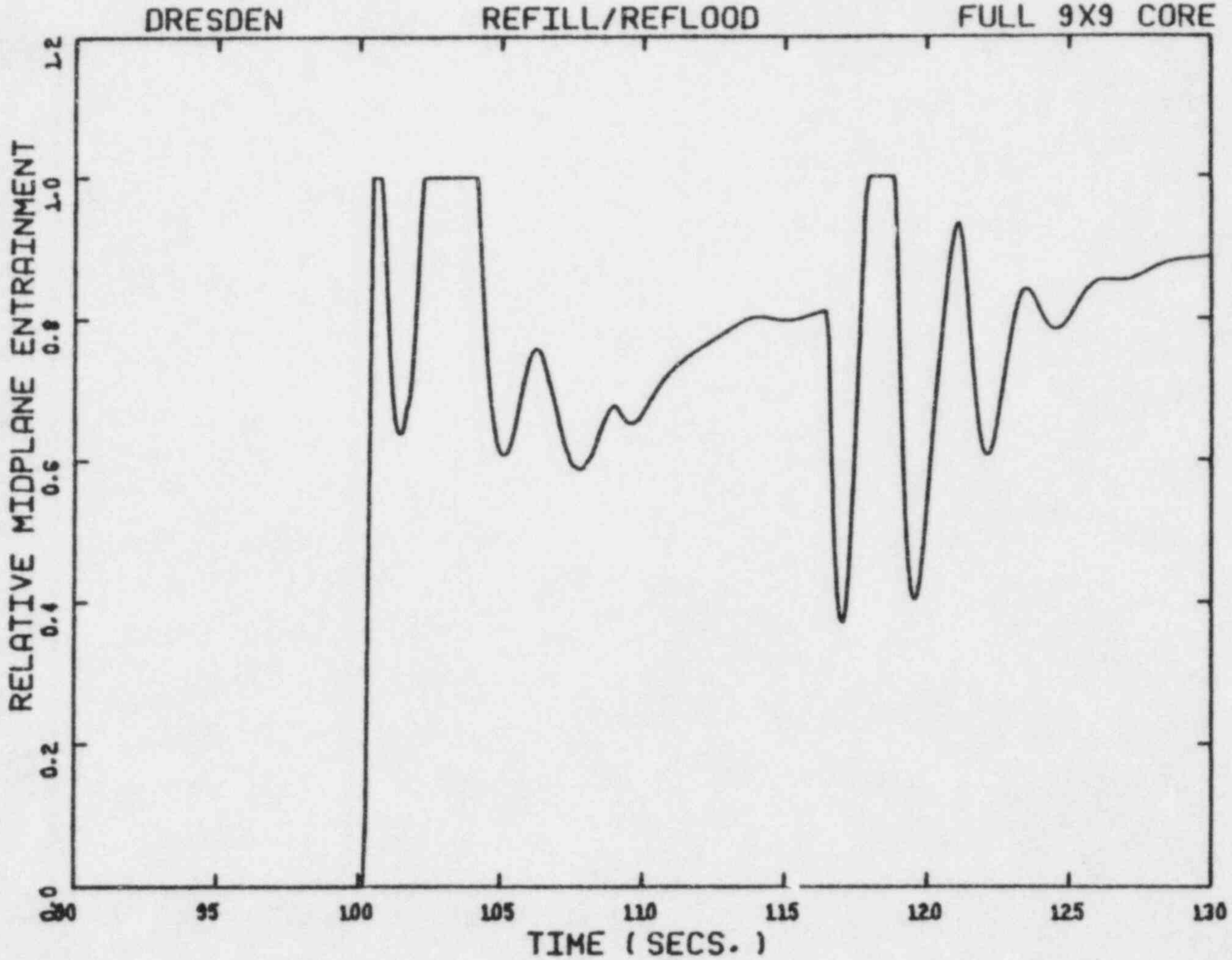

Figure 3.22 Refill/Reflood Relative Core Midplane Entrainment

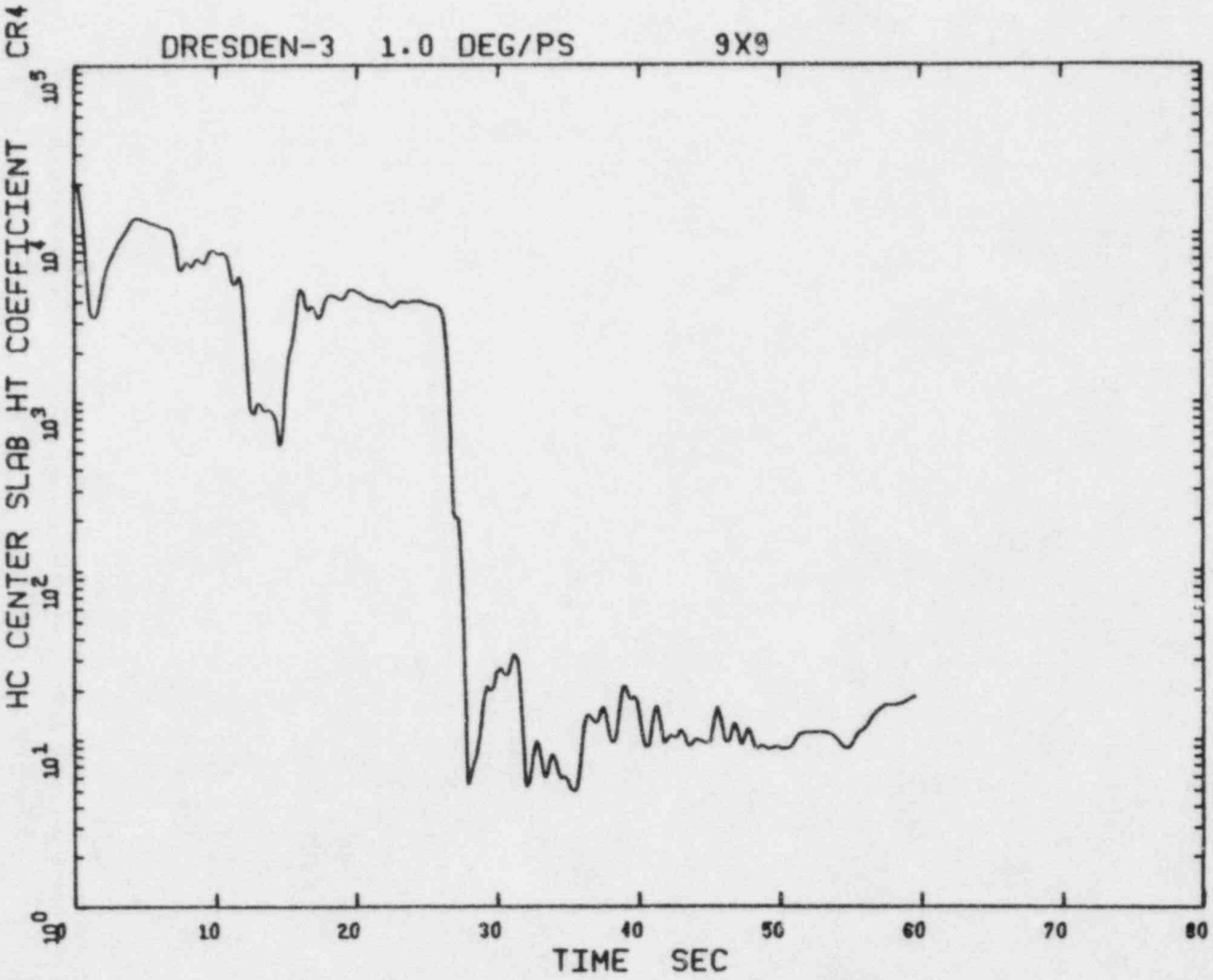

Figure 3.23 Blowdown HOT CHANNEL Heat Transfer Coefficient

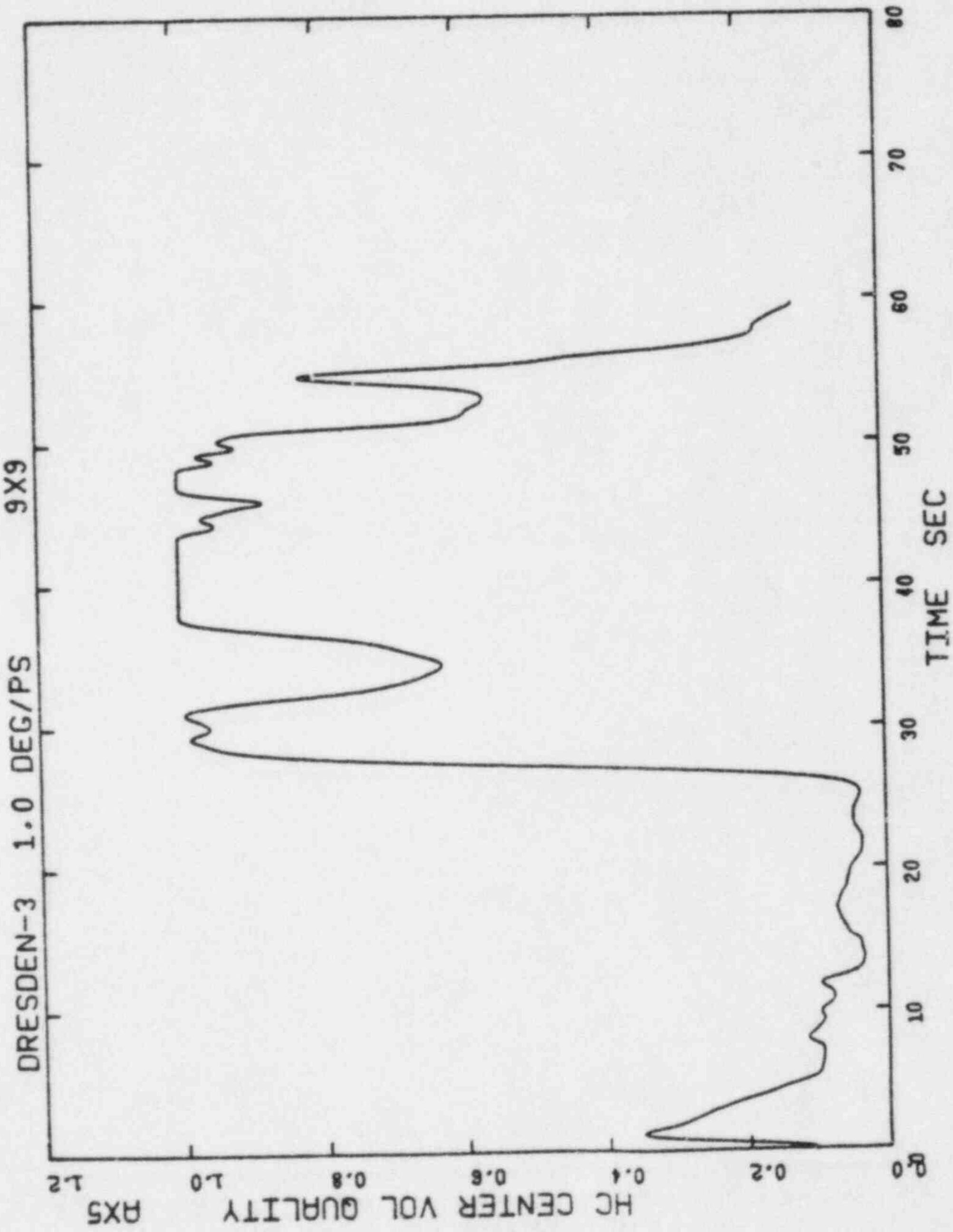

Figure 3.24 Blowdown HOT CHANNEL Center Volume Quality

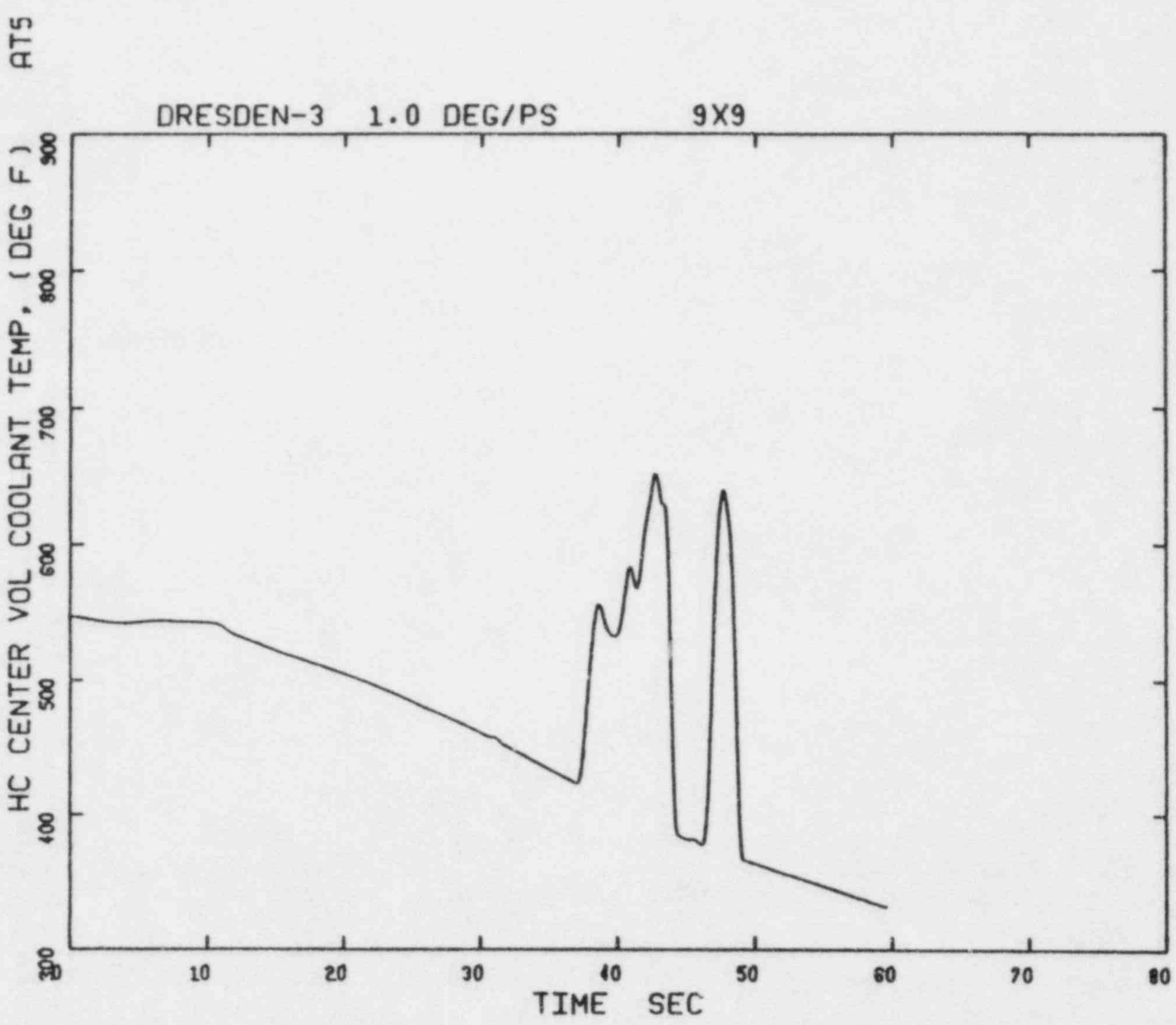

Figure 3.25 Blowdown HOT CHANNEL Center Volume Coolant Temperature

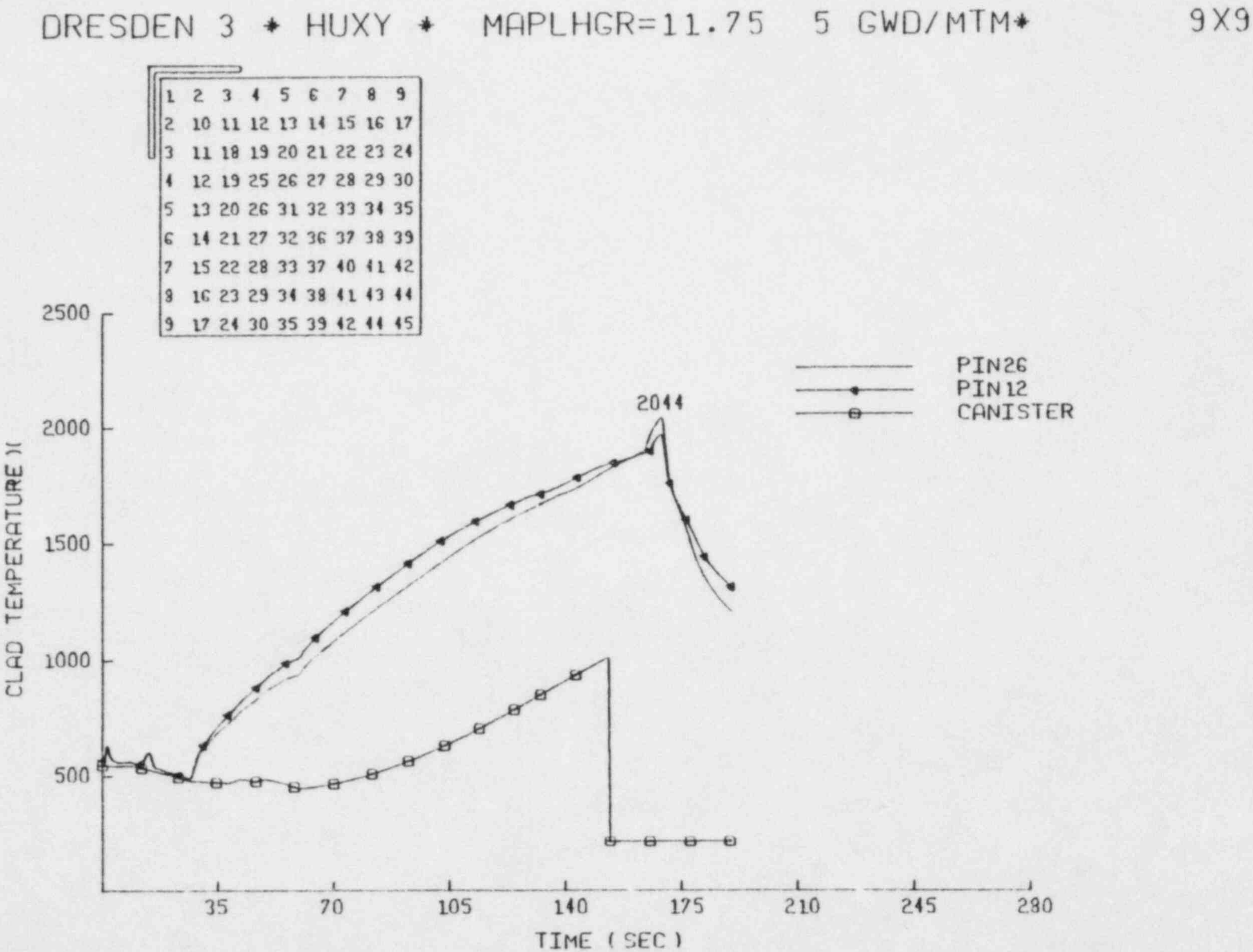

Figure 3.26 Typical Hot Assembly Heatup Results, 5000 MWD/MTM

4.0 CONCLUSIONS

A LOCA-ECCS analysis has been performed for the Dresden Unit 3 with ENC 9x9 fuel using the EXEM/BWR ECCS Evaluation model in conformance with Appendix K of 10 CFR 50. The limiting LOCA break was previously identified as the large double-ended guillotine break of the recirculation pump suction pipe with a discharge coefficient of 1.0. Limiting operating conditions were calculated to be for full-power low-flow operation. Based on the limiting break LOCA for the worst conditions, Maximum Average Planar Linear Heat Generation Rate MAPLHGR limits were determined as a function of exposure for ENC 9x9 fuel in Dresden Unit 3. These MAPLHGR limits are given in Tables 1.1 and 3.2, and in Figure 1.1.

Operation of the Dresden 3 reactor with ENC 9x9 fuel within the limits defined by Table 1.1 assures that the Dresden 3 emergency core cooling system will meet the acceptance criteria as required by 10 CFR 50.46. That is:

1. The calculated peak fuel element clad temperature does not exceed the 2200 F limit.
2. The amount of fuel element cladding which reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the core.
3. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The hot fuel rod cladding oxidation limit of 17% is not exceeded during or after quenching.
4. The system long term cooling capabilities provided for previous cores remains applicable to cores containing ENC reload fuel.

5.0 REFERENCES

(1) "Exxon Nuclear Methodology for Boiling Water Reactors, Volume 2, EXEM: ECCS Evaluation Model Summary Description," XN-NF-80-19(P), Volume 2, Revision 1, Exxon Nuclear Company, June 1981.

(2) "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," XN-NF-82-07(A), Revision 1, Exxon Nuclear Company, November 1982.

(3) "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and 10 CFR 50 Appendix K, Federal Register, Volume 39, Number 3, January 4, 1974.

(4) "Generic Jet-Pump BWR 3 LOCA Analysis Using the ENC EXEM Evaluation Model," XN-NF-81-71(P)(A) and Supplement 1(P)(A), Exxon Nuclear Company, September 1982.

(5) "Dresden Unit 3 LOCA Analysis Using the ENC EXEM Evaluation Model MAPLHGR Results," XN-NF-81-75(P), Exxon Nuclear Company, November 1981.

(6) "Restricted Maneuvering and Conditioning Criteria for Exxon Nuclear Fuel," XN-NF-P31,022, Rev. 1, Exxon Nuclear Company, October 3, 1983.

(7) "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," XN-NF-81-58(A), Revision 2, Exxon Nuclear Company, February 1983.

(8) "RELAX: A RELAP4-Based Computer Code for Calculating Blowdown Phenomena," XN-NF-80-19(P), Volume 2B, Revision 1, Exxon Nuclear Company, June 1981.

(9) "FLEX: A Computer Code for the Refill and Reflood Period of a LOCA," XN-NF-80-19(P), Volume 2B, Revision 1, Exxon Nuclear Company, June 1981.

(10) "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option - User's Manual," XN-CC-33(A), Revision 1, Exxon Nuclear Company, November 1, 1975.

(11) "BULGEX: A Computer Code to Determine the Deformation and the Onset of Bulging of Zircaly Fuel Rod Cladding," XN-74-21, Revision 2, and XN-NF-27, Revision 2, Exxon Nuclear Company, December 31, 1974.

(12) "Analysis of Dresden Units 2 and 3 Operation with One Relief Valve Out of Service," XN-NF-84-49, Exxon Nuclear Company, September 1984.

(13) "Analysis for Operation with One Relief Valve Out of Service for Quad Cities Units 1 and 2," NEDO-30052, General Electric Co.

Appendix A

Operation with One Relief Valve Out-of-Service

A.1 INTRODUCTION AND SUMMARY

This appendix considers the impact of operation of Dresden Unit 3 with one relief valve out-of-service on MAPLHGR limits for ENC 9x9 fuel. This analysis uses the same approach as that previously reported in a similar analysis (Reference 12) for ENC 8x8 fuel. A heatup calculation was performed at the exposure (5 GWD/MTM) determined to be most limiting for normal operation. Coolant boundary conditions from the GE analysis for Quad Cities, Reference 13, were used as in the previous ENC analysis of Dresden Units 2 and 3 for operation with one relief valve out-of-service. A MAPLHGR limit was determined for this exposure and a MAPLHGR multiplier was calculated as the ratio of the MAPLHGR limit for operation with one relief valve out-of-service to the MAPLHGR limit for normal operation at the same exposure. Applying this MAPLHGR multiplier to MAPLHGR limits for normal operation yields MAPLHGR limits for operation with one relief valve out-of-service at other exposures. These limits are presented in Table A.1.

A.2 RESULTS

The system conditions of the limiting small break are first summarized. After break initiation and scram on high drywell pressure, the water level drops below the top of the active fuel at approximately 260 s. The core level which would experience the highest PCT uncovers at about 313 s, LPCI flow begins at 540 s, and rewetting of the plane of interest occurs at about 590 s. These event times determine the heat transfer coefficient (HTC) to be applied in the heatup analysis and correspond to the times when the HTC changed as reported by GE in Figure 2 of Reference 13. As in the 8x8 analysis, Reference 12, an HTC of 10,000 Btu/hr-ft^2-^{o}F is used until uncovery at 313 s, an HTC of 25 Btu/hr-ft^2-^{o}F after reflood at 589 s.

Figure A.1 shows the clad temperature of the limiting rod for ENC 9x9 fuel at a MAPLHGR = 8.95 kw/ft. The seven fueled rods surrounding the central water rod are all close to the average power of all fueled rods. The highest powered of these, rod 25, is the limiting rod with a PCT of 2192^{o}F. The PCT of the highest powered rod in the entire assembly, rod 12, is 2165^{o}F. As in the ENC 8x8 fuel, the highest powered rod and other high powered rods do not have the highest PCT because they are near the cannister wall and have better radiative heat transfer.

A.3 MAPLHGR MULTIPLIER

A MAPLHGR multiplier for ENC 9x9 fuel is calculated in the same manner as was done for ENC 8x8 fuel in Reference 12.

$$\text{Multiplier} = \frac{8.95}{(\text{Maximum MAPLHGR})} = \frac{8.95}{11.75} = 0.762$$

Applying this multiplier over the full range of exposure yields the MAPLHGRs shown in Table A.1 for ENC 9x9 fuel in Dresden Unit 3 when operating with one relief valve out-of-service.

Table A.1

9x9 MAPLHGRs with Relief Valve out of Service

Bundle Average Exposure (MWD/MTM)	Normal MAPLHGR (kw/ft)	Reduced MAPLHGR (kw/ft)
0	11.40	8.68
5000	11.75	8.95
10000	11.40	8.68
15000	10.55	8.04
20000	9.70	7.39
25000	8.85	6.74
30000	8.00	6.09
35000	7.15	5.45
40000	6.30	4.80

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DRESDEN UNIT 3 LOCA-ECCS ANALYSIS

MAPLHGR RESULTS FOR ENC 9X9 FUEL

Distribution

D. J. Braun
J. C. Chandler
R. E. Collingham
T. P. Currie
S. E. Jensen
W. V. Kayser
T. H. Keheley
J. E. Krajicek
T. L. Krysinski
J. L. Maryott
J. N. Morgan
G. L. Ritter
R. B. Stout
D. R. Swope

J. M. Ross/CECo (60)

Document Control (10)

ATTACHMENT 6

STABILITY ASSESSMENT OF ENC 9X9 FUEL AT DRESDEN-3

USING COTRANSA2 STABILITY METHODOLOGY

1.0 INTRODUCTION

A stability analysis was performed to quantify the relative core stability margin for the initial 9X9 reload and equilibrium 9x9 reload cores at Dresden Unit 3. This Cycle 10 specific analysis was extended to include an evaluation of the relative stability margins of 8X8 and 9X9 fuel types. These calculations not only provide a comparison of 8X8 and 9X9 stability margins, but also provide a direct comparison between the approved COTRAN methodology and the advanced system stability model, COTRANSA2.

The COTRANSA2 calculated decay ratio at the rod-block/minimum pump speed intercept was 0.45 for Cycle 10 compared to 0.46 calculated by COTRAN. Thus, the parallel COTRANSA2 analysis supports the relative stability margins calculated by the COTRAN methodology for cycle 10. Additional COTRANSA2 calculations were made to determine the inherent change in stability margins due to the 9X9 geometry and hydraulic characteristics. Again, at the rod-block/minimum pump speed intercept, a full core loading of 9X9 fuel, when compared to a full core of 8X8 fuel with the same neutronic characteristics, results in a decay ratio increase of only 0.03. Thus, transient results indicate that both the cycle 10 mixed core loading and a full core loading of 9X9 fuel at Dresden Unit-3 will exhibit a high degree of stability with respect to reactor core density wave oscillations.

2.0 DISCUSSION OF RESULTS

The stability margins of the Dresden Unit-3 reactor have been evaluated with COTRANSA2 (References 1 through 3) for both cycle 10 and an equilibrium 9X9 core loading. The calculations were performed at the rod-block power corresponding to the minimum pump speed flow.

The calculational results for COTRANSA2 are tabulated in Table 2.1 for the two fuel loadings. In addition, the calculational results from a parallel COTRAN analysis are also presented. To determine the degree of consistency between the two calculational models, the calculational biases can be removed based on the benchmark analysis of the two methodologies (References 3 and 4). This is accomplished by using the least-squares data fits between calculated and measured data to determine the corresponding "expected" decay ratios. Table 2.2 presents the "expected" decay ratios for the COTRANSA2 and COTRAN methodologies. As Table 2.2 shows, the two methodologies provide a consistent calculational basis for the Dresden Unit-3 stability margins.

The increase in core decay ratio between cycle 10 and the equilibrium 9X9 fuel loading was also investigated. The primary objective was to determine whether the increased core decay ratio was due to the reduced rod diameter of the 9X9 fuel or due to inherent differences in core conditions between the cycle specific analysis and the equilibrium cycle based on Haling solutions.

To assess these differences, comparisons between 8X8 and 9X9 fuel loadings were made with the COTRAN and COTRANSA2 models. The COTRAN calculations were performed for cycle 10 and equilibrium core loadings and address the relative magnitude that 9X9 fuel has on core stability margins. The COTRANSA2 calculations were performed to determine the relative stability margins between the 8X8 and 9X9 fuel if they were neutronically identical. This analysis was performed for the equlibrium cycle by recalculating the core input to COTRANSA2 u ing the 9X9 cross sections in XTGBWR, but specifying 8X8 rod geometry and hydraulic parameters. Thus, the results will give a direct estimation of the geometry effects between 8X8 and 9X9 fuel. The results for these analyses are presented in Table 2.3. As shown in Table 2.3, the cycle 10 results indicate that there is no quantifiable impact of a single 9X9 reload batch on core stability margins. The full core analysis for the equilibrium cycle compares the stability margins for equivalent 8X8 and 9X9 fuel designs. Thus, the same multi-cycle analysis was used to reach the equilibrium cycle. As Table 2.3 shows, the COTRAN and COTRANSA2 analysis indicate that the 9X9 fuel design results in a 4 to 5 percent increase in core decay ratio when compared to an equivalent 8X8 design capable of comparable cycle energies.

These calculational comparisons between 8X8 and 9X9 fuel loadings indicate that the relative stability margins between 8X8 and 9X9 fuel must be made on a consistent basis. This basis includes reactor power histories and loading patterns since both of these influence the end-of-cycle power distribution and core reactivity coefficients. The equilibrium cycle analysis, therefore, may be used for a relative measure of stability margins when comparing to equivalent equilibrium analyses but does not directly represent the expected decay ratio when compared to cycle specific analysis.

Table 2.1 Calculated Core Decay Ratios For Dresden Unit-3

Cycle	COTRAN Decay Ratio	COTRANSA2 Decay Ratio	Difference
10	0.46	0.45	+0.01
Equilibrium (Full Core 9X9)	0.76	0.68	-0.08

Table 2.2 "Expected" Core Decay Ratios For Dresden Unit-3

Cycle	COTRAN Decay Ratio	COTRANSA2 Decay Ratio	Difference
10	0.43	0.48	+0.06
Equilibrium (Full Core 9X9)	0.61	0.65	+0.04

Table 2.3 Comparison of 8X8 and 9X9 Stability Margins

	Cycle 10 COTRAN Analysis	
	All 8X8 Fuel	1/3 9X9 Fuel
100%LL/NC	0.33	0.33
Rod Block/NC	0.54	0.53
	Equilibrium COTRAN Analysis	
	8X8 Fuel	9X9 Fuel
Rod Block/MPS	0.73	0.76
	Equilibrium COTRANSA2 Analysis	
	8X8 Geometry	9X9 Geometry
Rod Block/MPS	0.65	0.68

3.0 REFERENCES

1. L.R. Zimmerman et al., "Stability Evaluation Methodology for BWR Cores: The COTRANSA2 Advanced BWR Stability Model and Application to Analysis of Anticipated Operation," XN-NF-84-67(P), Exxon Nuclear Company, Inc., Richland, WA 99352, June 1984.

2. L.R. Zimmerman et al., "Stability Evaluation Methodology for BWR Cores: Example Calculation with COTRANSA2," XN-NF-84-67(P), Supplement 1, Exxon Nuclear Company, Inc., Richland WA 99352, December 1984.

3. D.W. Pruitt, "Stability Evaluation Methodology for BWR Cores : Stability Benchmark Analysis with COTRANSA2," XN-NF-84-67(P), Supplement 2, Exxon Nuclear Company, Inc., Richland WA 99352, July 1985.

4. Stability Evaluation of Boiling Water Reactor Cores : Sensitivity Analyses & Benchmark Analysis, XN-Nf-691(P)(A) & Supplement 1, Exxon Nuclear Company Inc., Richland WA 99352, August 1984.

ATTACHMENT 7

D3C10 Evaluation of Significant Hazards Consideration

Description of Amendment Request

Commonwealth Edison proposes to amend Facility Operating License DPR-25 for Dresden Unit 3 to allow the use of Exxon 9x9 fuel, operation of the reactor in an expanded POWER/FLOW region and allow Single Loop Operation above 50% thermal power for Cycle 10.

Basis for Proposed No Significant Hazards Consideration Determination

Commonwealth Edison has evaluated the proposed Technical Specification amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10CFR50.92(c), operation of Dresden Unit 3 Cycle 10 in accordance with the proposed amendments will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated because:

   a. The Exxon 9x9 fuel was first introduced to Dresden 2 Cycle 9 as Lead Test Assemblies (LTA) and these assemblies are currently going through their second cycle of irradiation. The Dresden 3 XN-3 and XN-3A reload fuel is very similar in design to the Dresden 2 LTAs with the exception in the number of water rods and Gadolinia-bearing fuel rods. The XN-3 and XN-3A 9x9 fuel thermal-hydraulic performance falls between that of the ENC 8x8 fuel and the GE 8x8 fuel indicating adequate compatibility for coresidence in the Dresden 3 core. ENC evaluated the XN-3 and XN-3A reload fuel mechanical design using the methodology which has either received prior NRC approval or is currently under NRC review. The transient analyses were performed using plant transient analysis methodology which is similar to that which was used to establish thermal margin requirements for Cycles 8 and 9. Finally, the LOCA-ECCS analysis for the 9x9 fuel was performed with generically NRC-approved methods and the results comply with 10CFR.50.46 criteria. Thus, the XN-3 and XN-3A reload 9x9 fuel design is not significantly different from those previously found acceptable to the NRC for previous reloads at Dresden 3 and 2 and therefore does not increase the probability or consequences of an accident.

b. The removal of the provisions regarding SLO from the license and the incorporation of them into the Technical Specifications, with some minor revisions and, additionally, the allowing operation in SLO above 50% power will not increase the probability or consequences of an accident because GE has previously performed analyses supporting SLO above 50% power. Furthermore, recent SLO tests performed at another plant site have demonstrated that operation in Single Loop does not represent a less stable mode of operation. ENC has evaluated the results of the GE analyses and concludes the results are also applicable for ENC reload fuel; therefore, Dresden 3 may safely operate in SLO under the less restrictive conditions of the proposed license amendment.

c. ENC has performed an Extended Load Line Limit Analysis (ELLLA) that supports operation in an expanded POWER/FLOW region. Analysis shows that transients initiated from the most limiting point of this expanded region (100/87) would be bounded by the POWER/FLOW condition at 100%/100% and thus ensure that no safety limits would be violated. For LOCA-ECCS concern, limiting LOCA break calculations were performed for the 100/87 and the 100/100 conditions. Both operating conditions were found to result in essentially identical LOCA results with the POWER/FLOW condition of 100/87 giving the slightly higher peak cladding temperature which was used to verify the adequacy of LOCA-ECCS MAPLHGR limits. By observing the MAPLHGR limits, the consequences of accidents (LOCA) remain within the existing accident criteria established for Dresden.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because:

   a. 9x9 fuel has been previously used in Dresden 2 without a finding of new or different accidents;
   b. SLO has been previously allowed up to 50% power;
   c. Operation in the ELLLA region does not allow any new modes of operation nor any new equipment which could initiate or change the nature of accident sequences.

3. Involve a significant reduction in the margin of safety for the same reason as 1. above.

In consideration of the above, Commonwealth Edison expects that NRC approval of these amendments should not be predicated on satisfactory resolution of public comments or intervention as provided by for 10 CFR 50.91(a)(4).