SIEMENS

EMF-91-172

Grand Gulf Unit 1 LOCA Analysis for Single Loop Operation

October 1991

Siemens Nuclear Power Corporation

SIEMENS

EMF-91-172
Issue Date: 10/11/91

GRAND GULF UNIT 1 LOCA ANALYSIS
FOR SINGLE LOOP OPERATION

Prepared by:

D. J. Braun
BWR Fuel Engineering
Fuel Engineering and Licensing

October 1991

/smg

Siemens Nuclear Power Corporation

Engineering and Manufacturing Facility

2101 Horn Rapids Road PO Box 130 Richland WA 99352-0130 Tel (509) 375-8100 Fax (509) 375-8402

CUSTOMER DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

The results of a loss-of-coolant-accident (LOCA) analysis with emergency core cooling systems (ECCS) for the Grand Gulf Unit 1 to support single loop operation (SLO) are reported in this document. These calculations are performed with the generically approved Siemens Nuclear Power Corporation (SNP) EXEM/BWR Evaluation Model[1] in accordance with Appendix K of 10 CFR 50,[9] and the results comply with the USNRC 10 CFR 50.46[9] criteria.

The initial SLO condition selected for this analysis is 70.6% power and 54.1% flow. This analysis establishes the multiplier that is to be applied to the two-loop MAPLHGRs of the fuel during single loop operation. The analysis is performed with the same multiplier for both SNP 8x8 and 9x9-5 fuel.

2.0 SUMMARY

The results of the LOCA-ECCS analysis presented herein support the use of a 0.86 SLO multiplier on the two-loop MAPLHGRs for SNP fuel when the Grand Gulf Unit 1 reactor is in single loop operation.

Single loop operation of Grand Gulf Unit 1 with a multiplier of 0.86 on the two loop SNP fuel MAPLHGRs for 8x8 and 9x9-5 fuels assures that the emergency core cooling systems for the Grand Gulf Unit 1 plant will meet the USNRC acceptance criteria of 10 CFR 50.46[9] 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 calculated total oxidation of the cladding nowhere exceeds 17% of the total cladding thickness before oxidation.

3. The calculated core wide metal-water reaction does not exceed 1% of the zircaloy associated with the active fuel cladding in the reactor.

4. The LOCA cladding temperature transient is calculated to be terminated at a time when the core is still amenable to cooling.

5. The system long-term cooling capabilities provided for the initial core and subsequent reloads remain applicable to SNP fuel.

3.0 JET PUMP BWR ECCS EVALUATION MODEL

3.1 LOCA During Single Loop Operation

The loss-of-coolant accident (LOCA) break spectrum analysis for a BWR/6 for two loop operation conditions is described in Reference 7, and the Grand Gulf Unit 1 plant specific MAPLHGR analysis for two loop operation is described in Reference 8. This document describes LOCA-ECCS analysis and MAPLHGR justification for Grand Gulf Unit 1 SLO operation. The same limiting break and single failure are assumed in both the two loop and the SLO LOCA analyses.

During SLO the recirculation pump in the inactive loop is not in operation and the recirculation flow control valve is put at minimum position. Both intake and discharge block valves in the idle recirculation loop remain open during SLO. A significant resistance to idle loop recirculation flow does exist, but a small amount of flow can pass through the idle loop during LOCA conditions. A break can be hypothesized to occur in either loop. However, a break in the inactive loop would behave essentially like a break during two loop operation except that substantial break flow would come from only one side of the break (because the recirculation flow control valve is at its minimum position). System performance would then be like that resulting from a somewhat smaller break during two loop operation. The scenario of breaks smaller than the limiting break has already been covered by the BWR/6 LOCA break spectrum analysis.(7) Further consideration in this report will be given only to the case where a break occurs in the active loop. This case differs from the two loop case in one important respect: there is no flow coastdown in the intact (idle) recirculation loop.

Previous SLO analysis(10) assumed that the consequence of a lack of recirculation loop coastdown flow (which continues to supply liquid from the downcomer to the lower plenum during two loop operation) would be an almost immediate flow stagnation in the core and a very early CHF (0.1 sec); this resulted in degraded heat transfer very early in the transient and required that a MAPLHGR reduction factor of 0.86(11) be imposed for SLO conditions on the MAPLHGR for NSSS vendor fuel for Grand Gulf Unit 1.

3.2 EXEM/BWR Application To Grand Gulf Unit 1

The SNP EXEM/BWR ECCS Evaluation model codes are used for this SLO LOCA-ECCS calculation. The codes which comprise EXEM consist of RELAX,[2] FLEX,[3] HUXY,[4,5] RODEX2.[6] The latest versions of these codes are used for this SLO analysis, and they are referenced in the two loop analyses.[7,8]

The approved SNP generic break spectrum analysis[7] for the BWR/6 class of reactors identified the limiting break as the double-ended guillotine break of the recirculation pump discharge pipe with a discharge coefficient of 1.0 (1.0 DEG/RD) and a worst single failure.

In the unlikely event that a LOCA would occur during SLO, a rapid drop in core flow would be expected to occur during the early phase of the event because the idle loop pump is not operating. The core flow transient during the early phase (0 to 5 seconds) of a single loop LOCA is the principal event which could distinguish such an accident from a LOCA occurring during normal two loop operation. Other than the core flow, the LOCA results for two and single loop operations are quite similar, as shown by event times (time of uncovery of jet pumps, lower plenum flashing, time of rated spray, etc.). Therefore the worst single failure, the size, location, and discharge coefficient of the break for the single loop LOCA will be the same as for the two loop LOCA.

The 70.6% power/54.1% core flow (70.6/54.1) operating point is selected as the initial operating condition for this analysis. The conditions are those used by the NSSS vendor in their SLO LOCA-ECCS analysis.[11]

The system behavior during a LOCA is determined primarily by the LOCA break parameters: break location, break size, and break configuration together with the ECCS systems and plant geometry. Variation in core geometric parameters produce only secondary effects on the system behavior. Thus, by using bounding core neutronic parameters, the LOCA-ECCS results established by this analysis will apply for future cycles unless substantial changes are made in the plant operating conditions, plant hardware, or core design such that the analysis no longer bounds the plant conditions.

The system blowdown calculations for the Grand Gulf Unit 1 SLO differ from those for two loop operation in that the back flow through the idle loop jet pump is modeled consistent with expected SLO steady state conditions. The initial conditions are shown in Table 3.1, and the system blowdown nodalization diagram is shown in Figure 3.1.

The system blowdown calculations are followed by HOT CHANNEL calculations. The hot channel geometry is modeled to that of SNP 8x8 and 9x9-5 fuel residing in the Grand Gulf reactor. The initial conditions are adjusted to correspond to those for the single loop operating point. The power, axial peaking, and flow of the hot assembly are determined by an XCOBRA thermal hydraulics calculation to support a MCPR value considerably below the appropriate MCPR operating limit. The nodalization and geometry used in the reflood calculation are identical to those of the two loop analysis. In the FLEX code the intact loop is not modeled in detail because intact loop flows are insignificant at the time of rated low pressure core spray. Thus, no changes are required in the FLEX nodalization or geometry compared to the two loop analysis. The initial conditions for the reflood calculation are entirely determined by the system blowdown calculation.

The HUXY heatup calculation of the hot plane (center one foot node) is done identically with previous two loop analyses: fuel stored energy, gap thermal conductivity and dimensions from RODEX2 as a function of power and exposure; time of rated spray, decay power, heat transfer coefficients and coolant conditions from RELAX; and time of hot-node-reflood from FLEX. 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 heatup calculation is conservative. The spray heat transfer coefficients identified in Appendix K of 10 CFR 50[9] are used for the SLO analysis in an identical manner as used in the approved two loop analyses[8,12]. This includes the use of 5.0 Btu/hr-ft^2-°F[12] for all unheated surfaces in the SNP 9x9-5 fuel. Peak cladding temperature (PCT) and the cladding oxidation percentage are specifically determined for the SNP fuel geometries.

TABLE 3.1 GRAND GULF UNIT 1 REACTOR SYSTEM DATA FOR SINGLE LOOP OPERATION

Primary Heat Output, MW	(1.02 x .706 x 3833) = 2760.2
Total Reactor Flow Rate, lb/hr	(.541 x 112.5 x 10^6) = 60.86 x 10^6
Active Core Flow Rate, lb/hr	54.28 x 10^6
Steam Dome Pressure, psia	1001.7
Reactor Inlet Enthalpy, Btu/lb	511.8
Recirculation Loop Flow Rate, lb/hr	15.93 x 10^6
Steam Flow Rate, lb/sec	11.47 x 10^6
Feedwater Flow Rate, lb/sec	11.44 x 10^6
Rated Recirculation Pump Head, ft	765.
Rated Recirculation Pump Speed, rpm	1785.
Moment of Inertia, lbm-ft^2/rad	18,875
Recirculation Suction Pipe I.D., in	21.825
Recirculation Discharge Pipe I.D., in	21.825

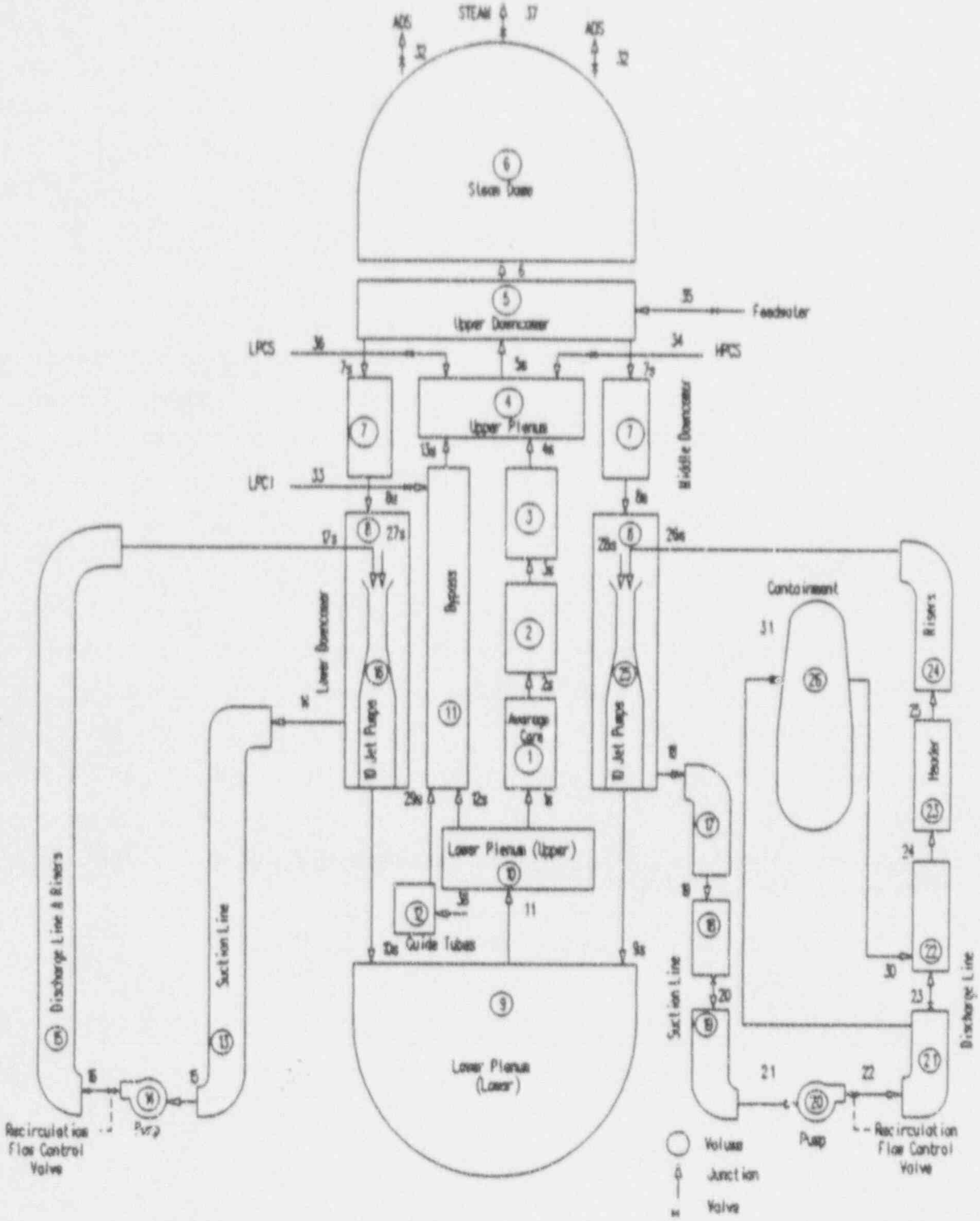

FIGURE 3.1 GRAND GULF UNIT 1 SYSTEM BLOWDOWN NODALIZATION

4.0 ANALYSIS RESULTS AND CONCLUSIONS

The results are obtained by a LOCA-ECCS heatup analysis of the 1.0 DEG/RD break over the exposure range of the SNP 8x8 and 9x9-5 fuels in the Grand Gulf reactor. The fuel stored energies at 5 GWd/MTU are used to initialize the RELAX/HOT CHANNEL calculations for both fuel types. These stored energies are the maximum calculated by the RODEX2 code to occur during the life of the fuels. This means that the fuel temperatures in the hot channels are at their maximum calculated values regardless of the fuel exposure at which the pipe rupture is assumed to occur.

Shown in Table 4.2 are the highest PCTs over the exposure range for both fuels in both single and two[8,13] loop operation. The two loop MAPLHGRs with a .86 multiplier used in this single loop analysis produce lower PCTs than the acceptance criteria of 2200 °F and are less than the PCTs for the two-loop analysis by about 100 °F. Thus, a .86 multiplier on the two loop SNP 8x8 and 9x9-5 MAPLHGR curves assures that the 10 CFR 50.46[9] criteria are met under SLO conditions.

Table 4.1 lists the major event times for this analysis. System blowdown results are given in Figures 4.1 through 4.22. System refill and reflood results are shown in Figures 4.23 through 4.25. Results from the SNP 9x9-5 RELAX/HOT CHANNEL calculation are given in Figures 4.26 through 4.28. The RELAX/HOT CHANNEL results are applied as boundary conditions for the HUXY heatup calculations. A clad temperature history determined by HUXY for the SNP 9x9-5 fuel at the beginning-of-life (BOL) exposure is shown in Figure 4.29.

An examination of these plots reveals the following information:

1. The sudden loss of drive flow in the operating (broken loop) jet pumps results in a sudden drop in lower plenum pressure of sufficient magnitude to allow flow through the inactive jet pumps to "reverse" from their initial negative flow to a positive flow at one second into the blowdown.

2. The exit junction flows of the operating jet pumps reverse direction at one second into the blowdown.

3. Because of 2. above, the initial drop in core flow is sufficient to cause an early CHF at the mid-plane of the SNP 8x8 and 9x9-5 hot channels. CHF occurs at less than 2 seconds for the SNP 8x8 and 9x9-5 fuel types.

With the results presented here, it is concluded that the application of a .86 multiplier during SLO conditions to the SNP 8x8 and 9x9-5 fuel MAPLHGRs in the Technical Specifications will protect SNP fuel in the Grand Gulf Unit 1 reactor from exceeding 10 CFR 50.46[9] limits.

TABLE 4.1 LOCA EVENT TIMES - SINGLE LOOP OPERATION

Event	Time (sec)
Start	0.00
Initiate Break	0.05
Feedwater Flow Stops	3.05
Steam Flow Stops	6.05
Low Mixture Level for HPCS	10.2
Low Mixture Level for LPCS/LPCI	17.2
Jet Pumps Uncover	19.5
HPCS Flow Starts	20.2
Lower Plenum Flashes (Quality > 0)	21.3
Recirculation Suction Nozzle Uncovers	31.1
LPCS/LPCI Pressure Permissive Cleared	67.5
LPCS Flow Starts	82.7
LPCI Flow Injection Starts	91.5
End of Blowdown (Upper Plenum Pressure for Rated LPCS flow)	113.6
Depressurization Ends (vessel pressure reaches 1 atm)	180.4
Start of Reflood (high density fluid enters core)	183.9
Peak Clad Temperature Reached	194.5

TABLE 4.2 PCT RESULTS FOR SNP FUEL - SINGLE AND TWO LOOP OPERATION

	Maximum PCT (°F) Two Loop Operation	Maximum PCT (°F) .86 x MAPLHGR SLO
SNP 8x8 Fuel	1738[8]	1631
SNP 9x9-5 Fuel	1713[13]	1609

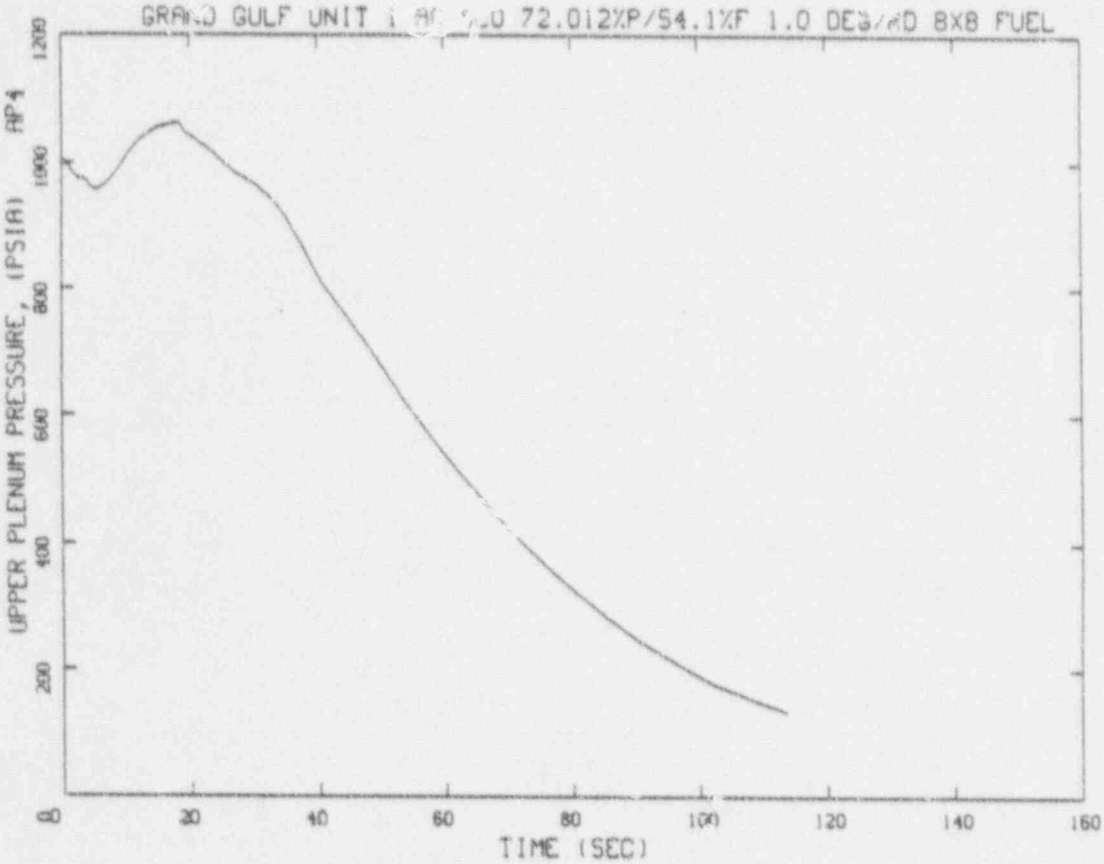

FIGURE 4.1 UPPER PLENUM PRESSURE

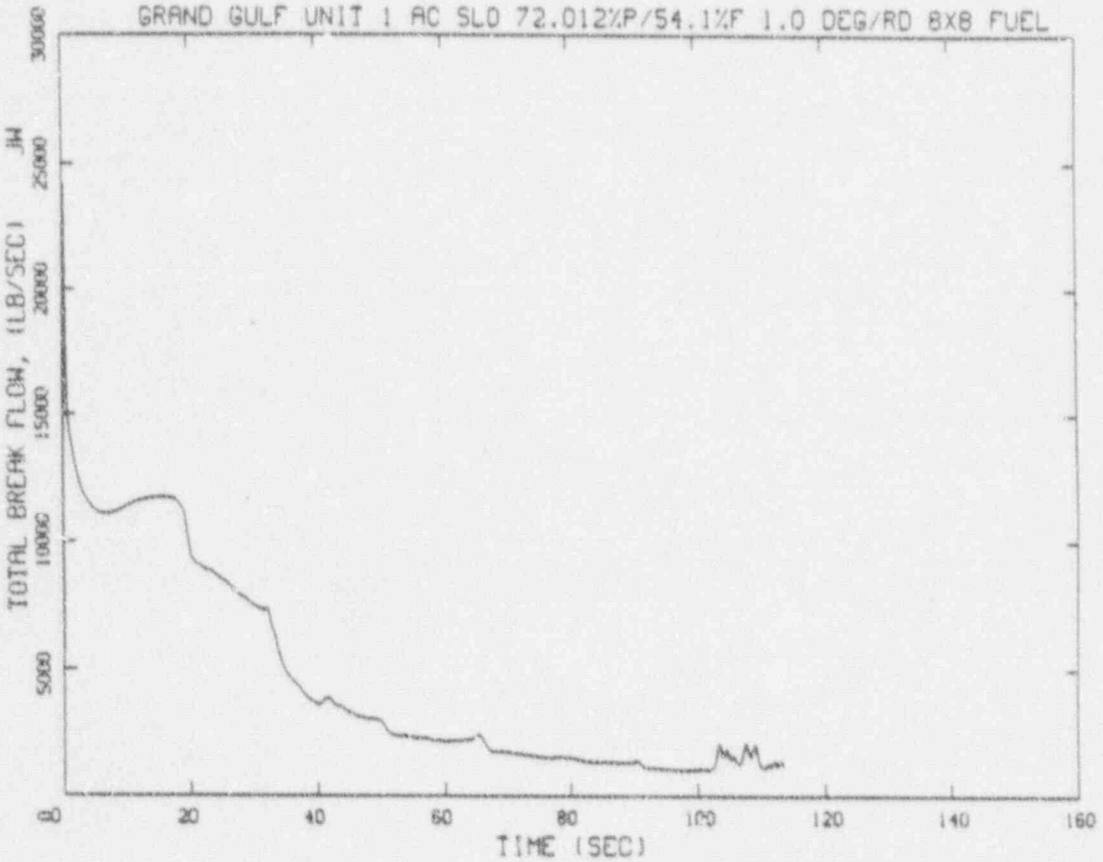

FIGURE 4.2 TOTAL BREAK FLOW

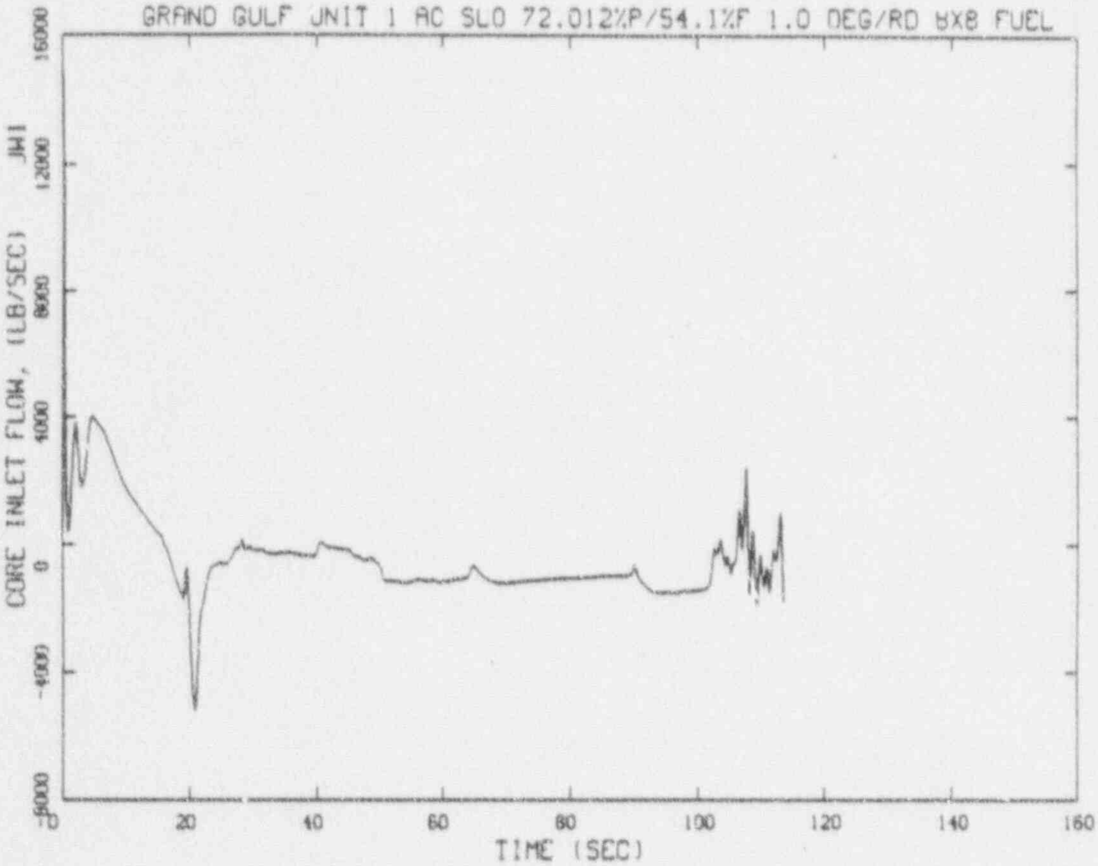

FIGURE 4.3 AVERAGE CORE INLET FLOW

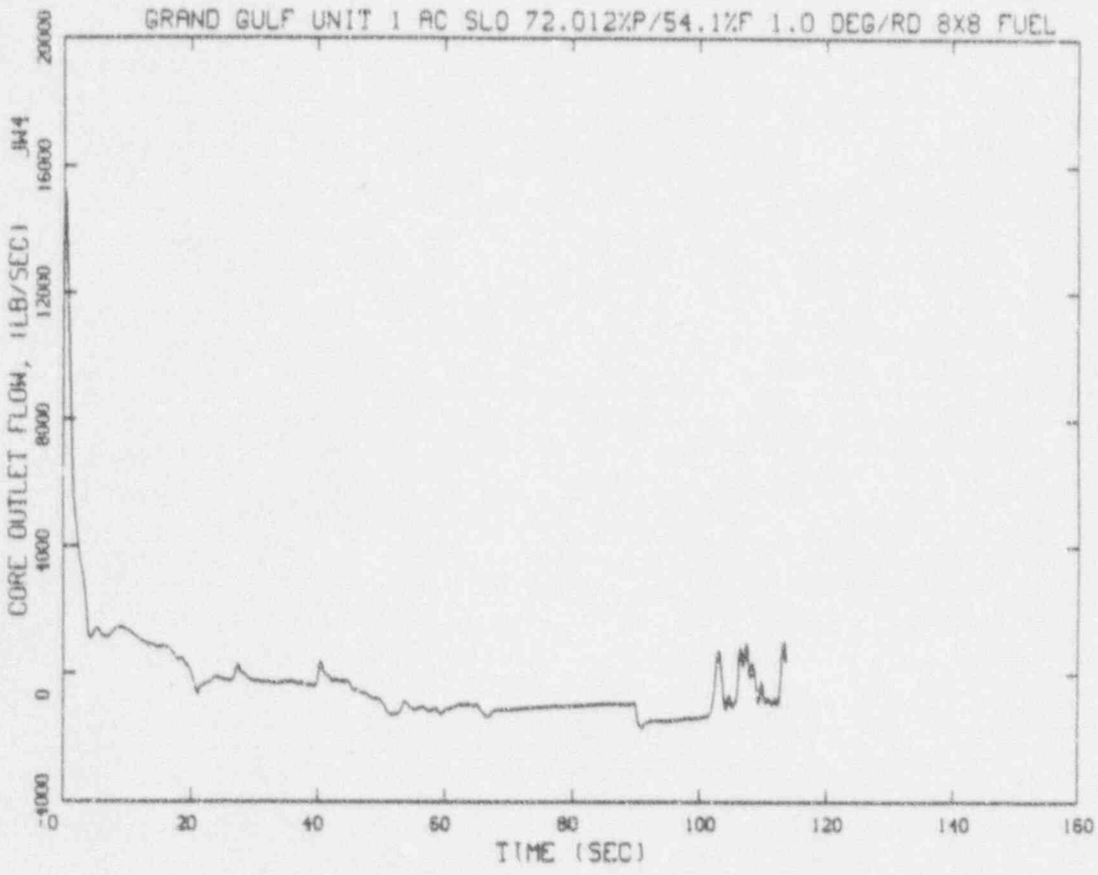

FIGURE 4.4 AVERAGE CORE OUTLET FLOW

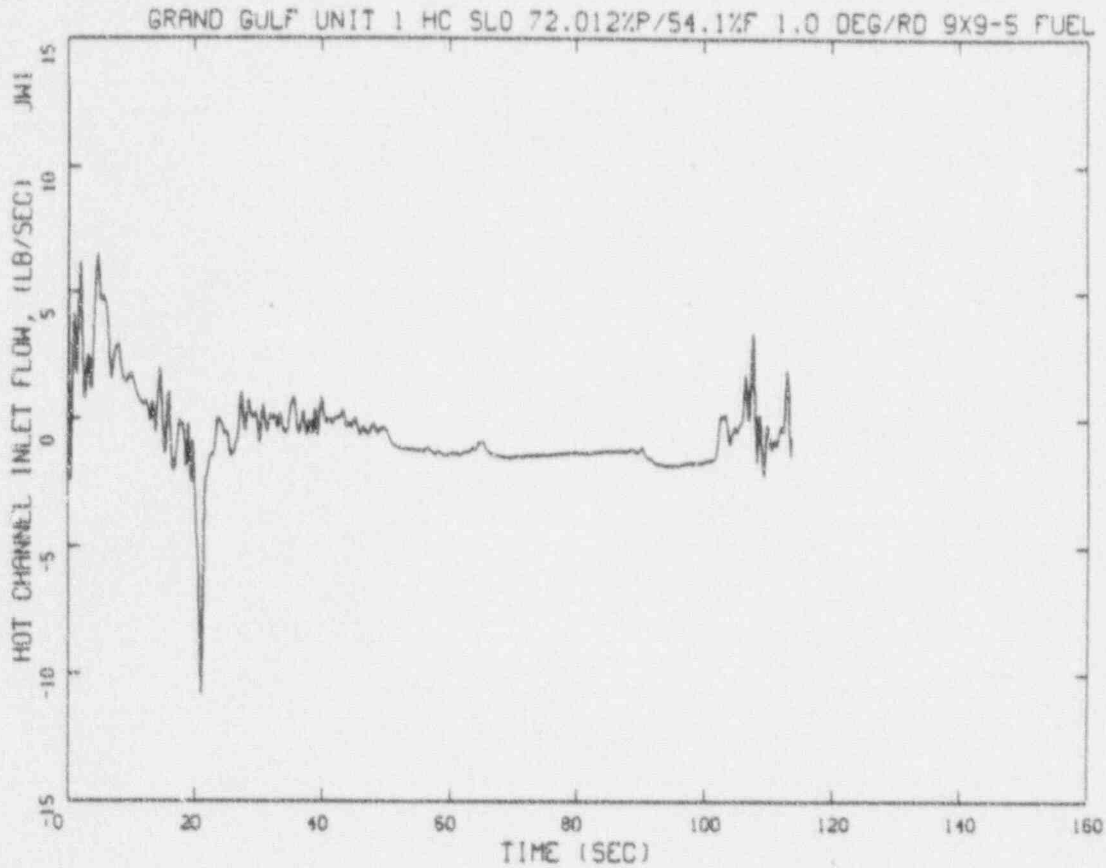

FIGURE 4.5 HOT CHANNEL INLET FLOW

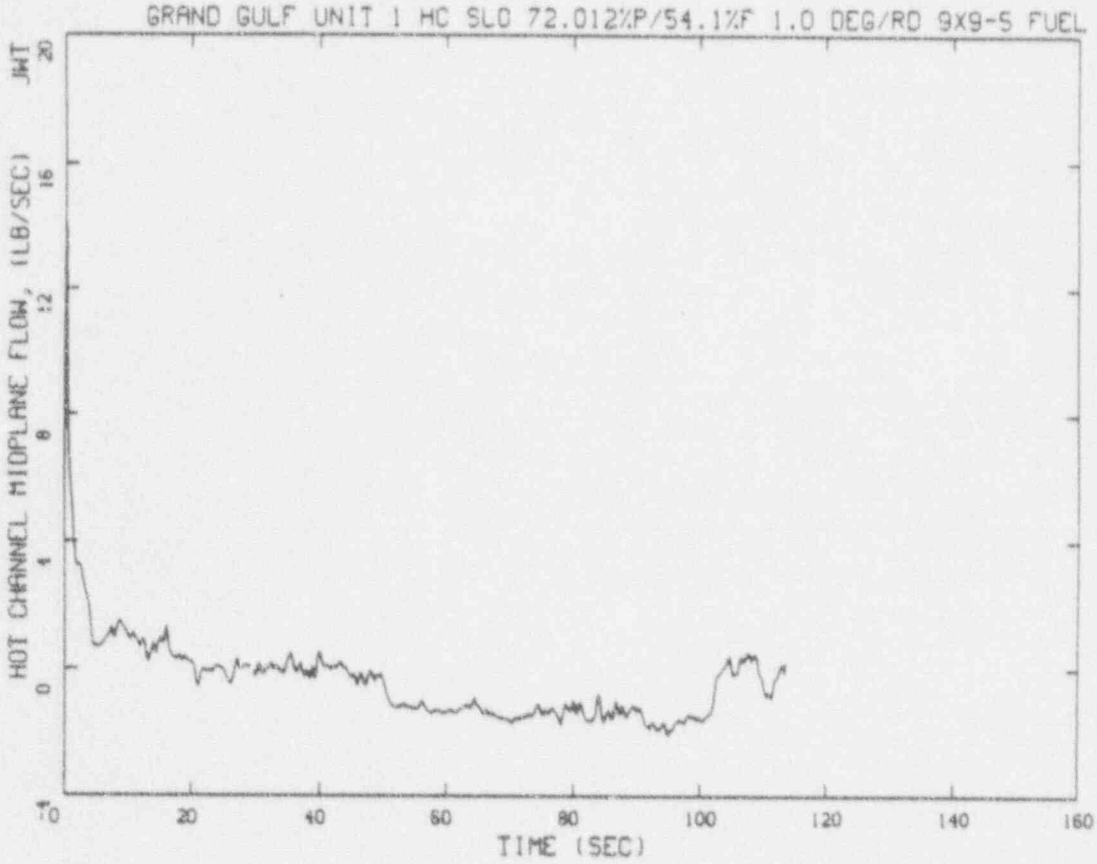

FIGURE 4.6 HOT CHANNEL MIDPLANE FLOW

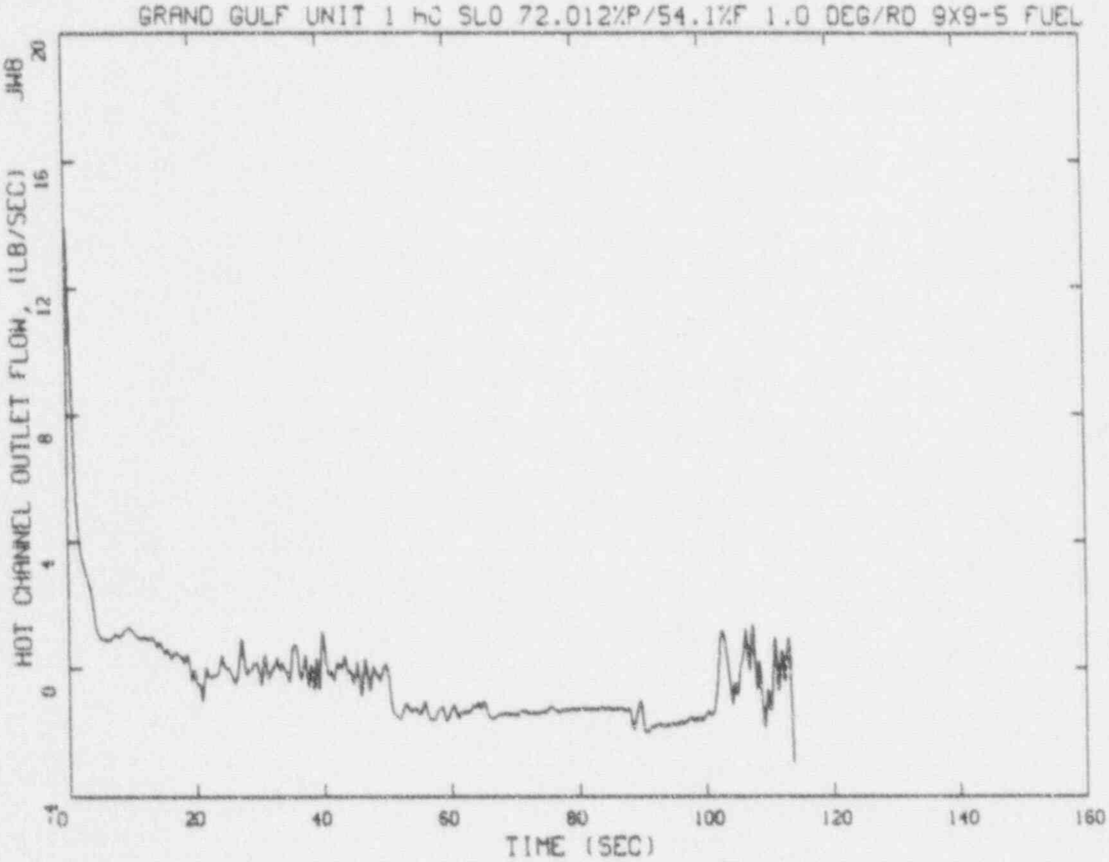

FIGURE 4.7 HOT CHANNEL OUTLET FLOW

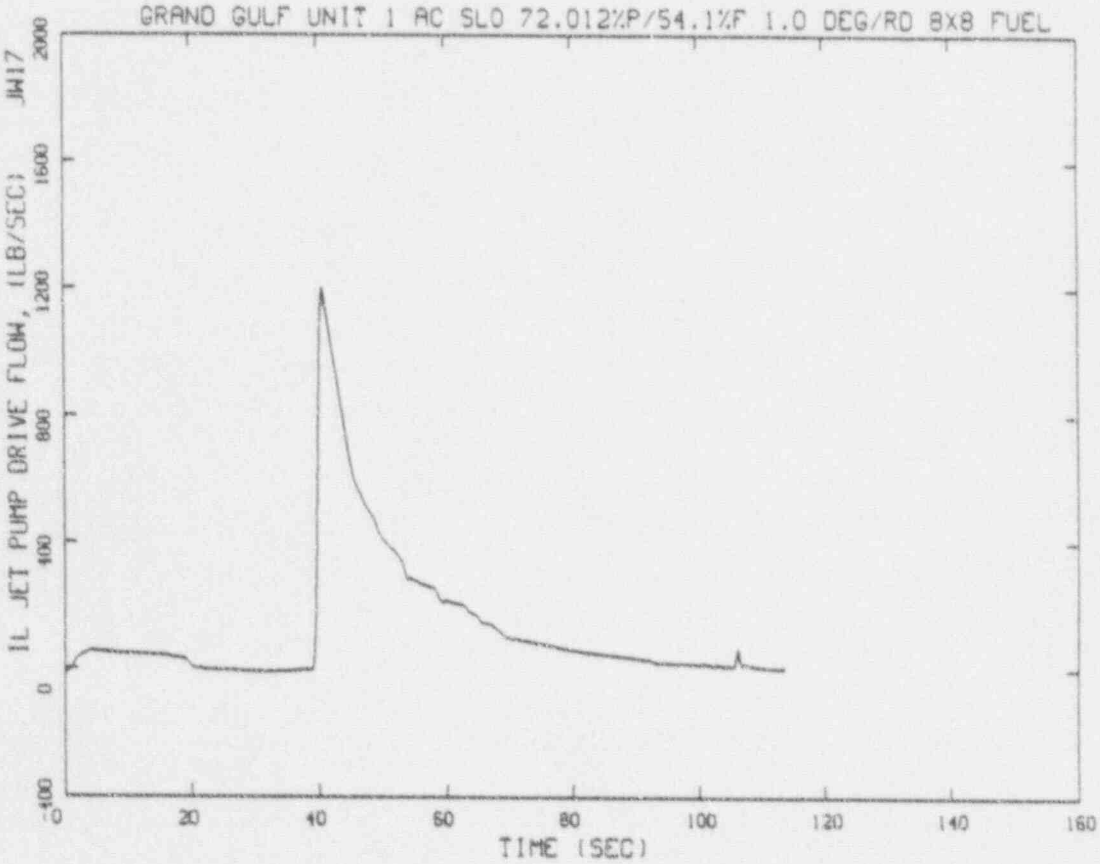

FIGURE 4.8 INTACT LOOP JET PUMP DRIVE FLOW

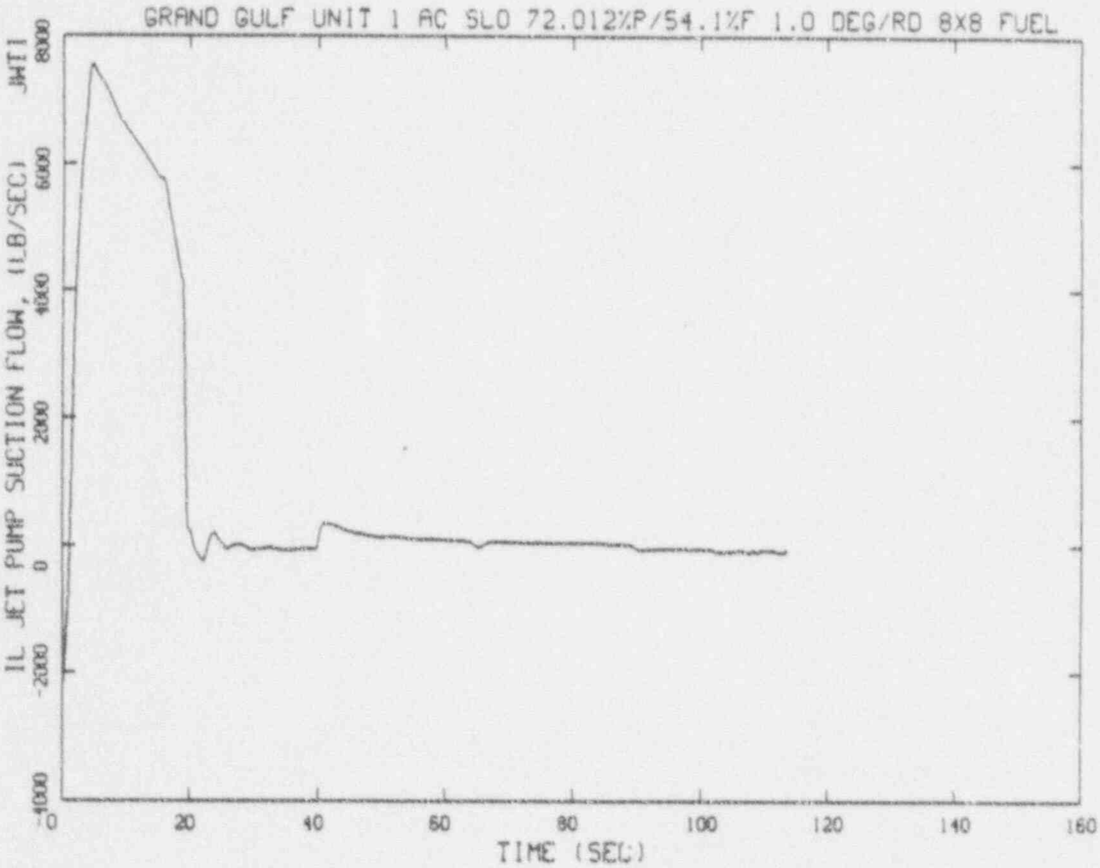

FIGURE 4.9 INTACT LOOP JET PUMP SUCTION FLOW

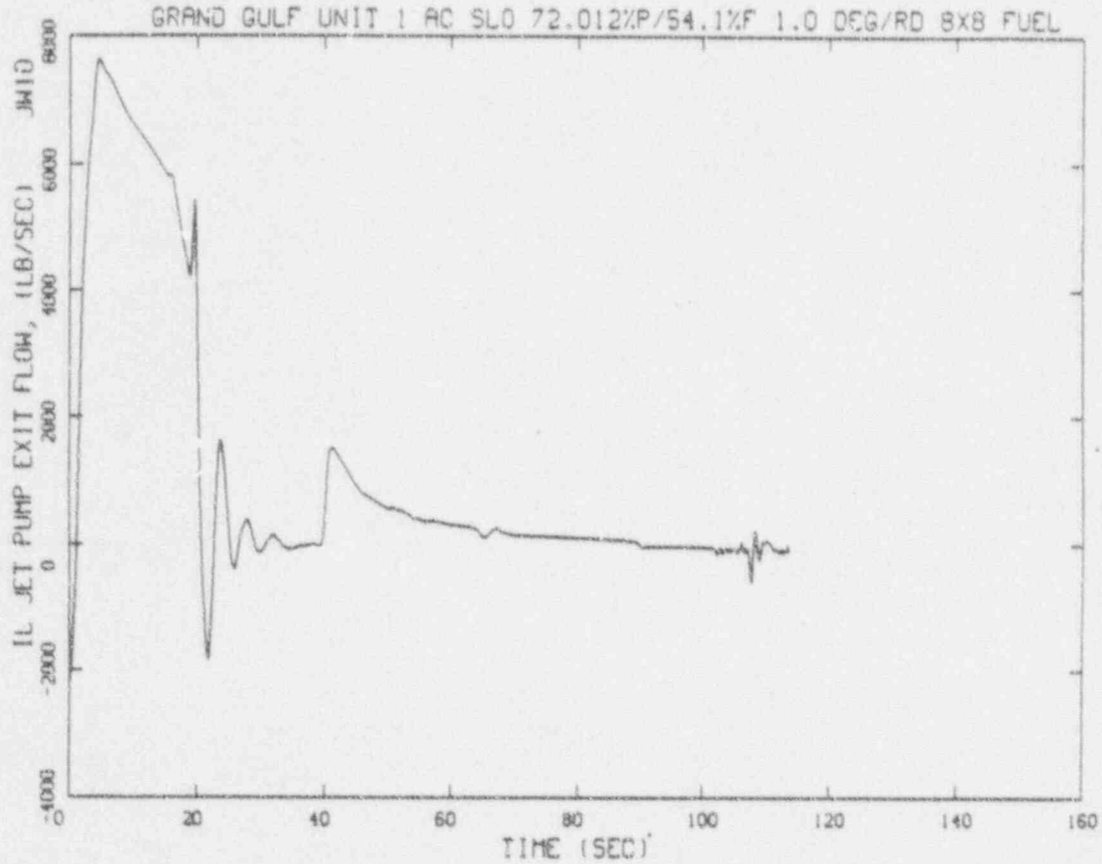

FIGURE 4.10 INTACT LOOP JET PUMP EXIT FLOW

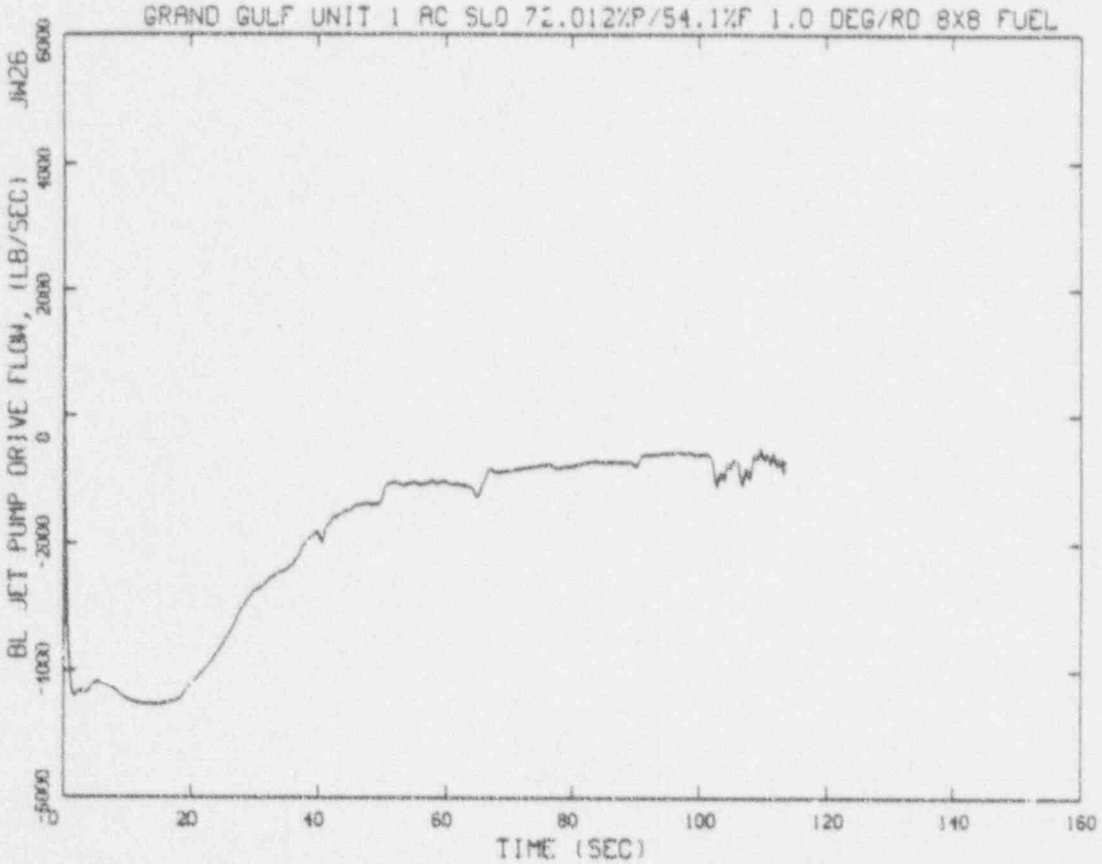

FIGURE 4.11 BROKEN LOOP JET PUMP DRIVE FLOW

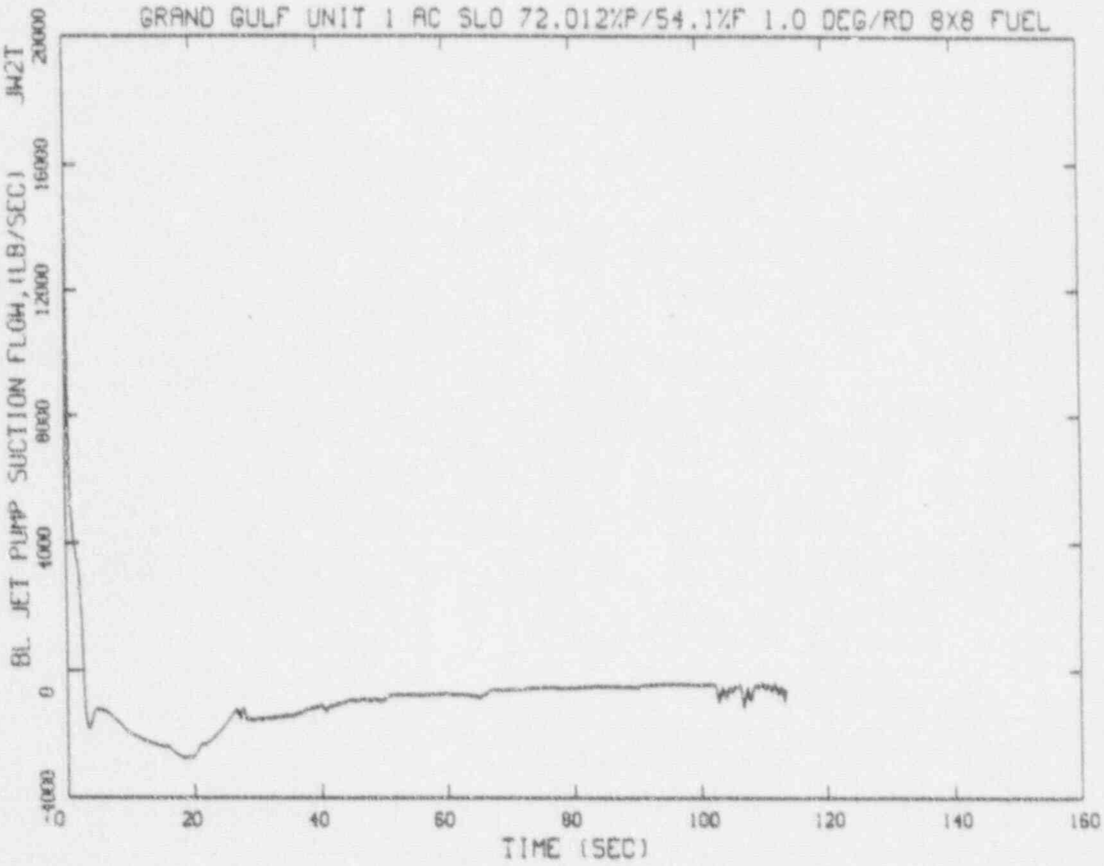

FIGURE 4.12 BROKEN LOOP JET PUMP SUCTION FLOW

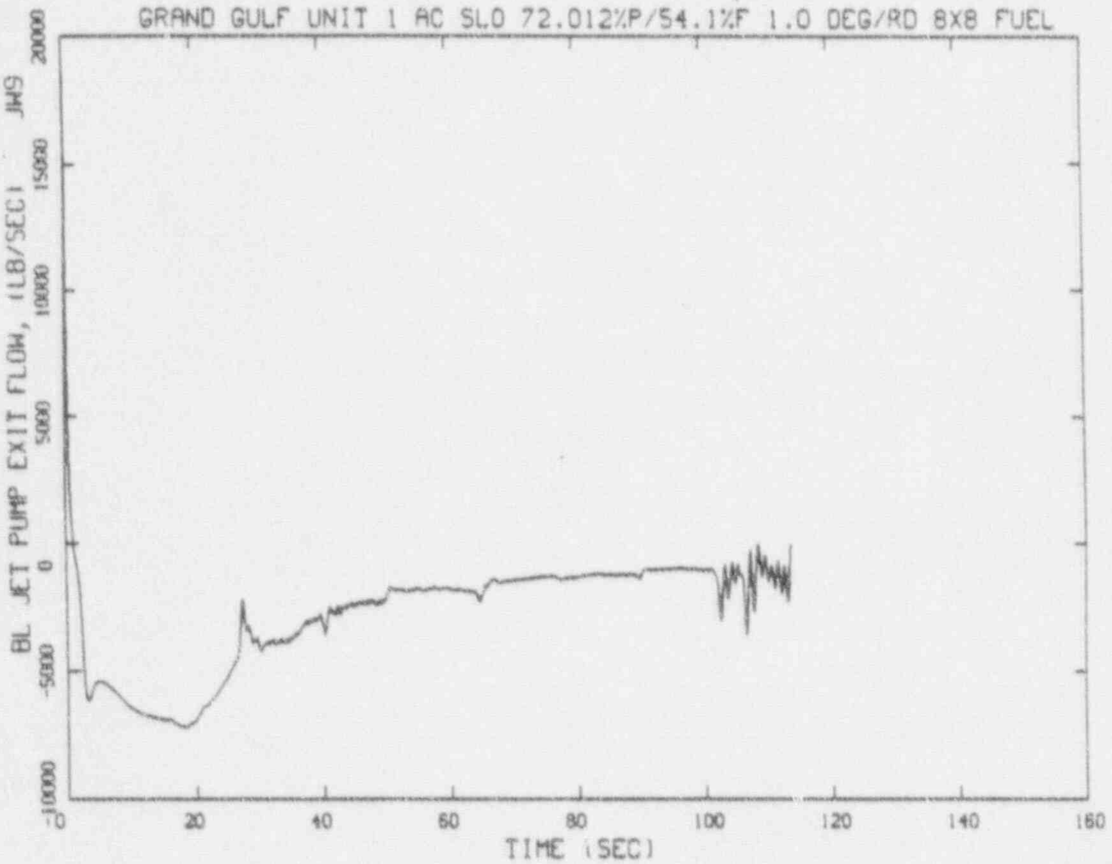

FIGURE 4.13 BROKEN LOOP JET PUMP EXIT FLOW

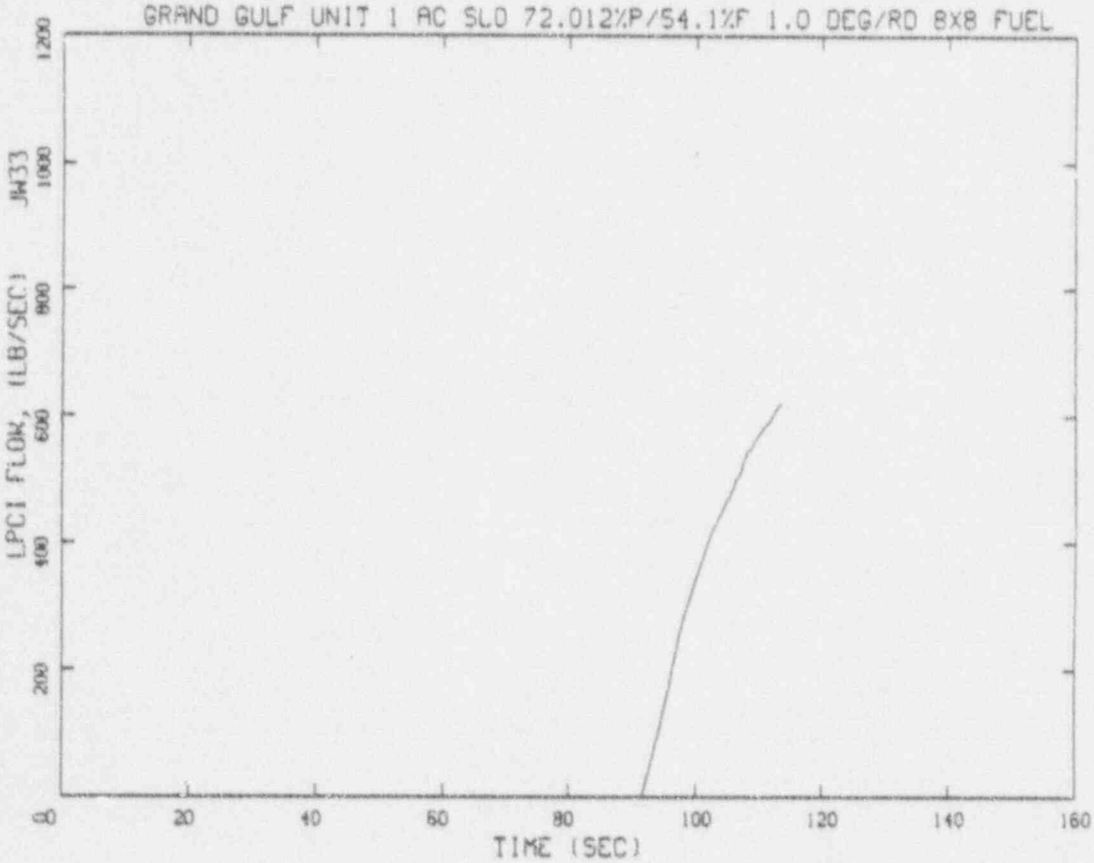

FIGURE 4.14 LOW PRESSURE COOLANT INJECTION FLOW

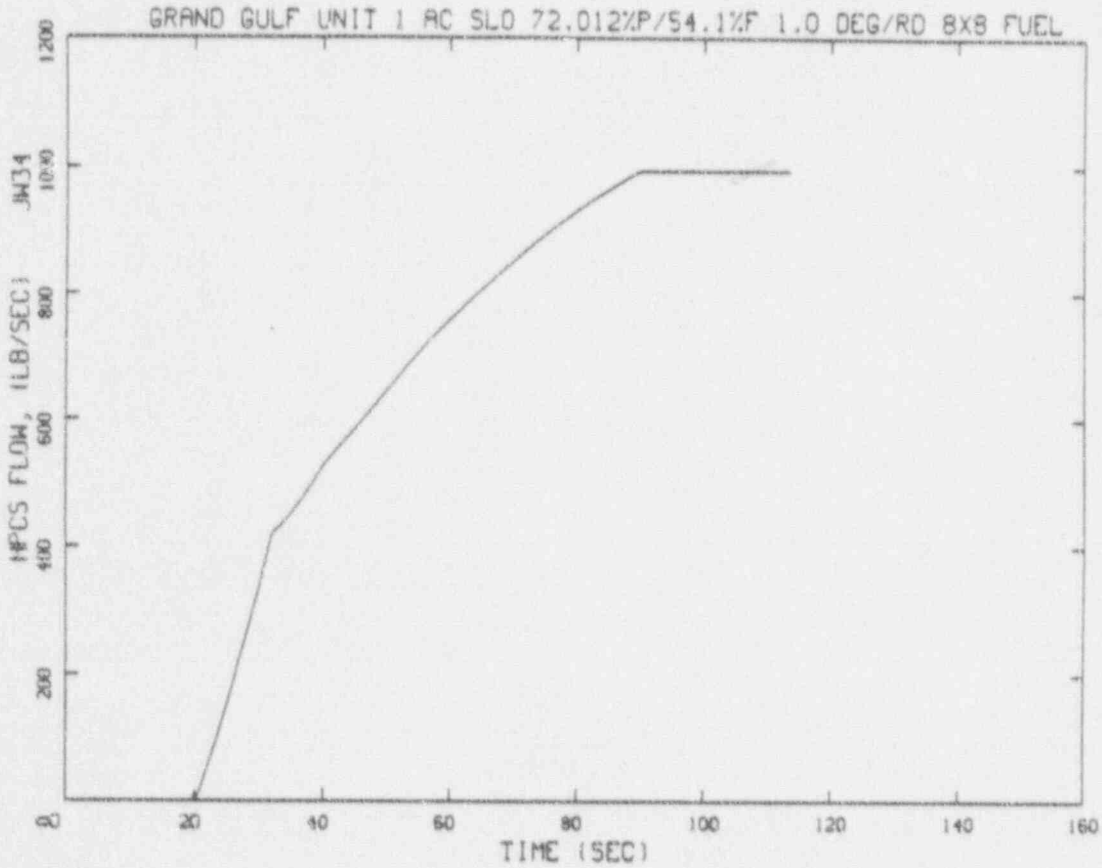

FIGURE 4.15 HIGH PRESSURE CORE SPRAY FLOW

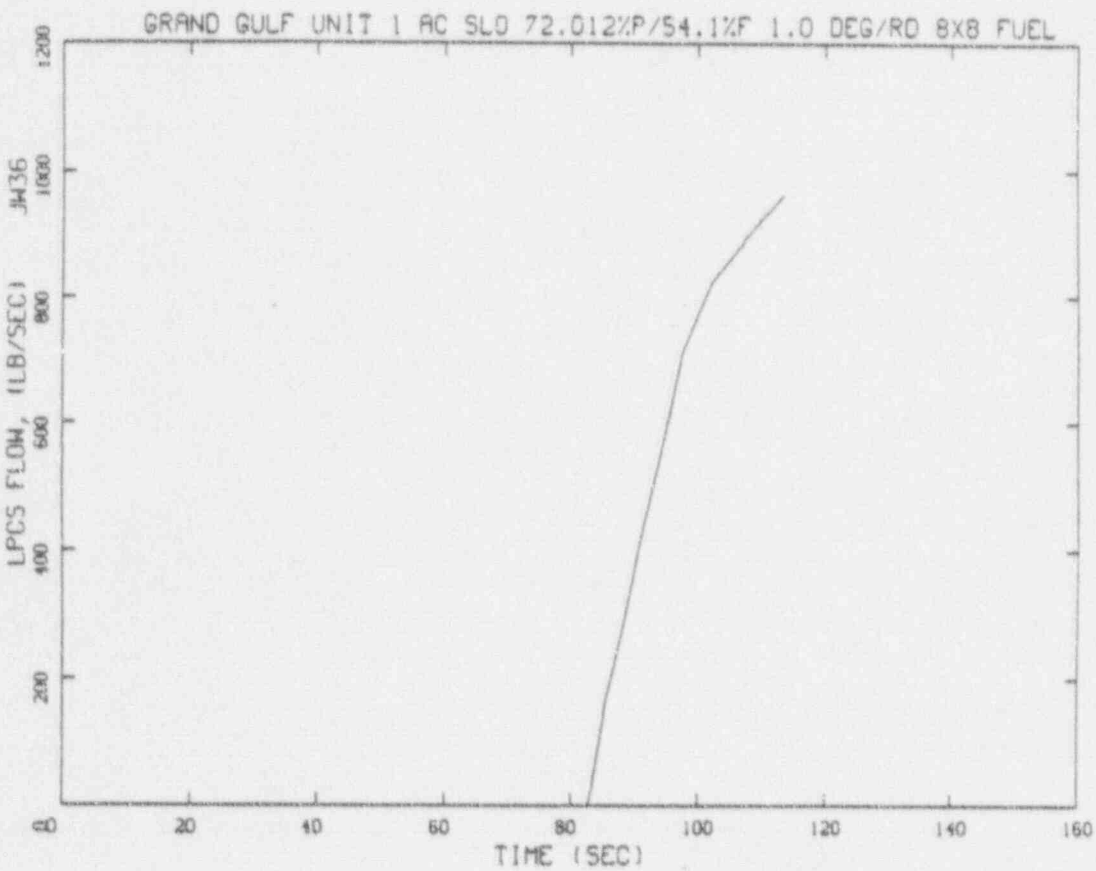

FIGURE 4.16 LOW PRESSURE CORE SPRAY FLOW

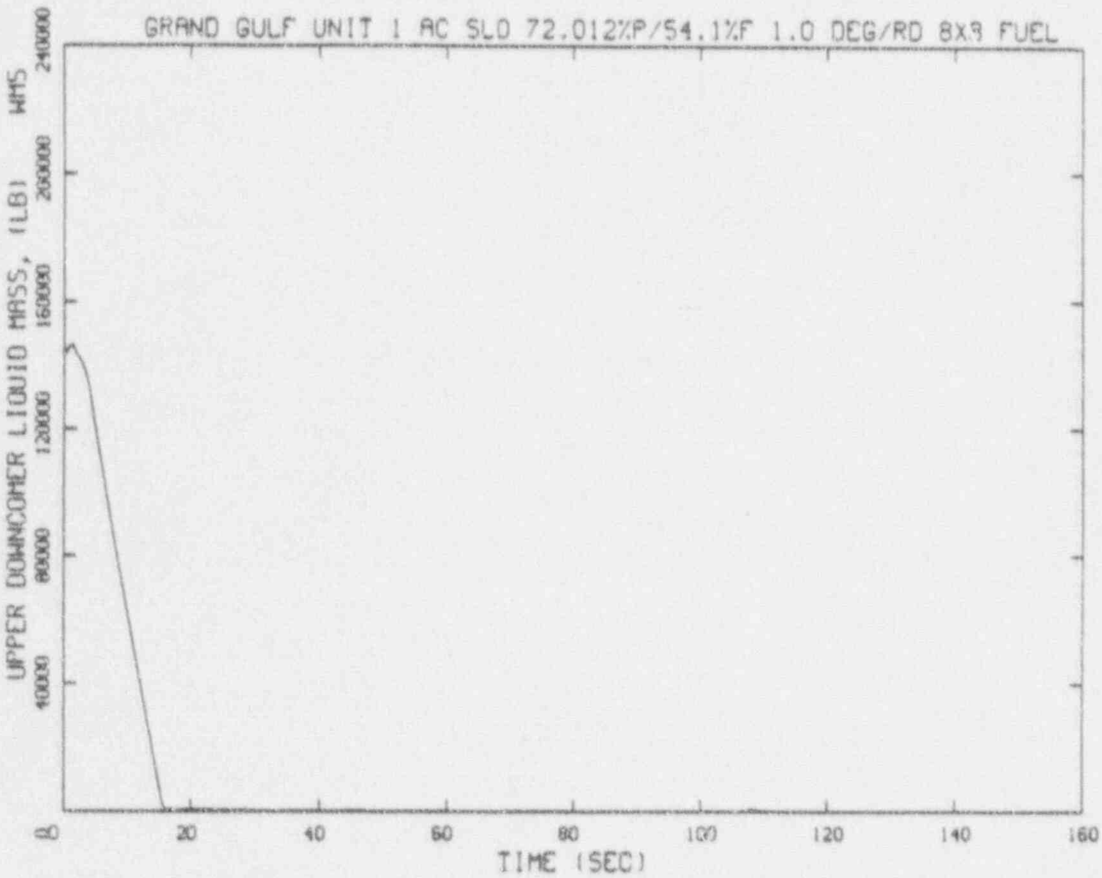

FIGURE 4.17 UPPER DOWNCOMER LIQUID MASS

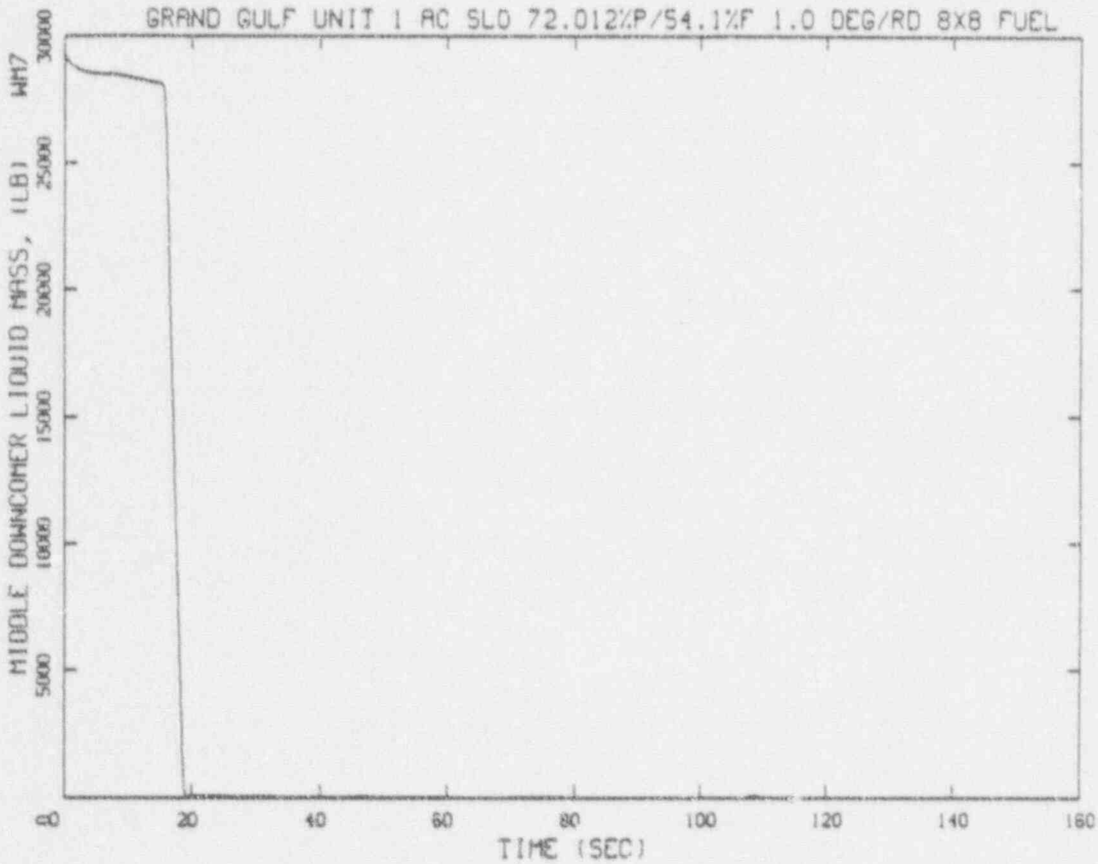

FIGURE 4.18 MIDDLE DOWNCOMER LIQUID MASS

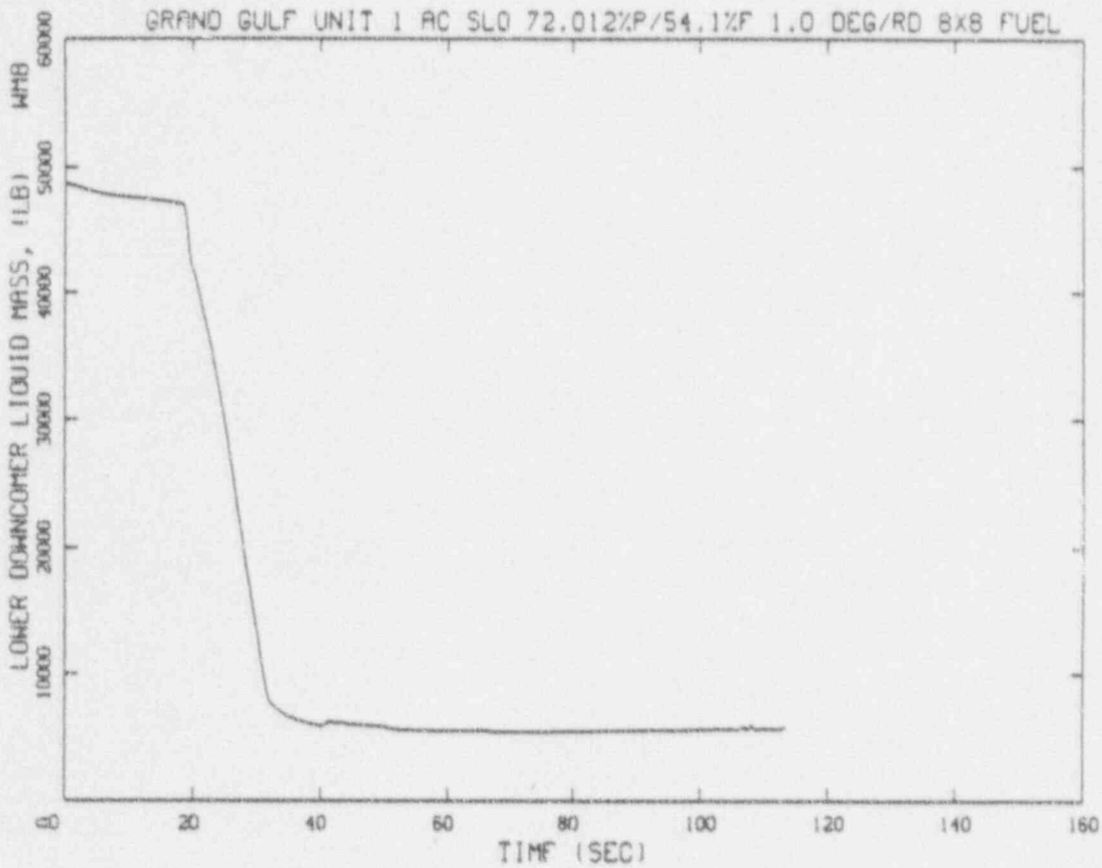

FIGURE 4.19 LOWER DOWNCOMER LIQUID MASS

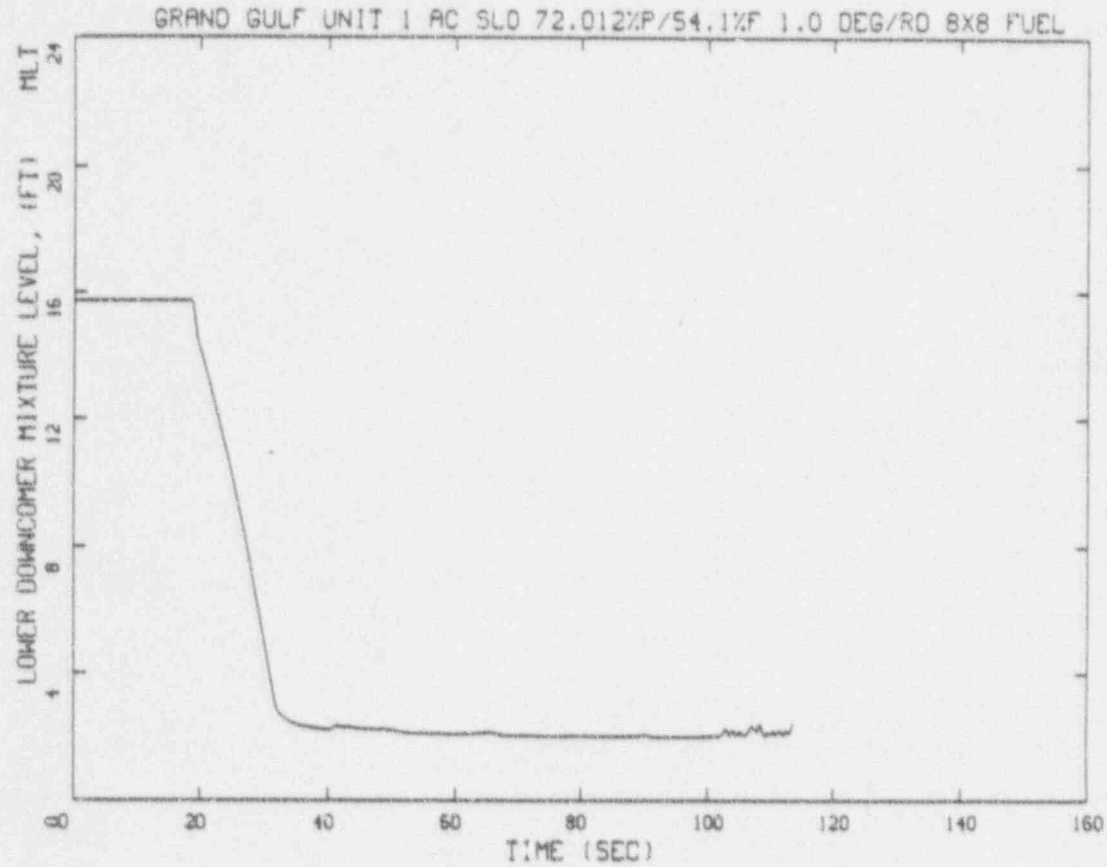

FIGURE 4.20 LOWER DOWNCOMER MIXTURE LEVEL

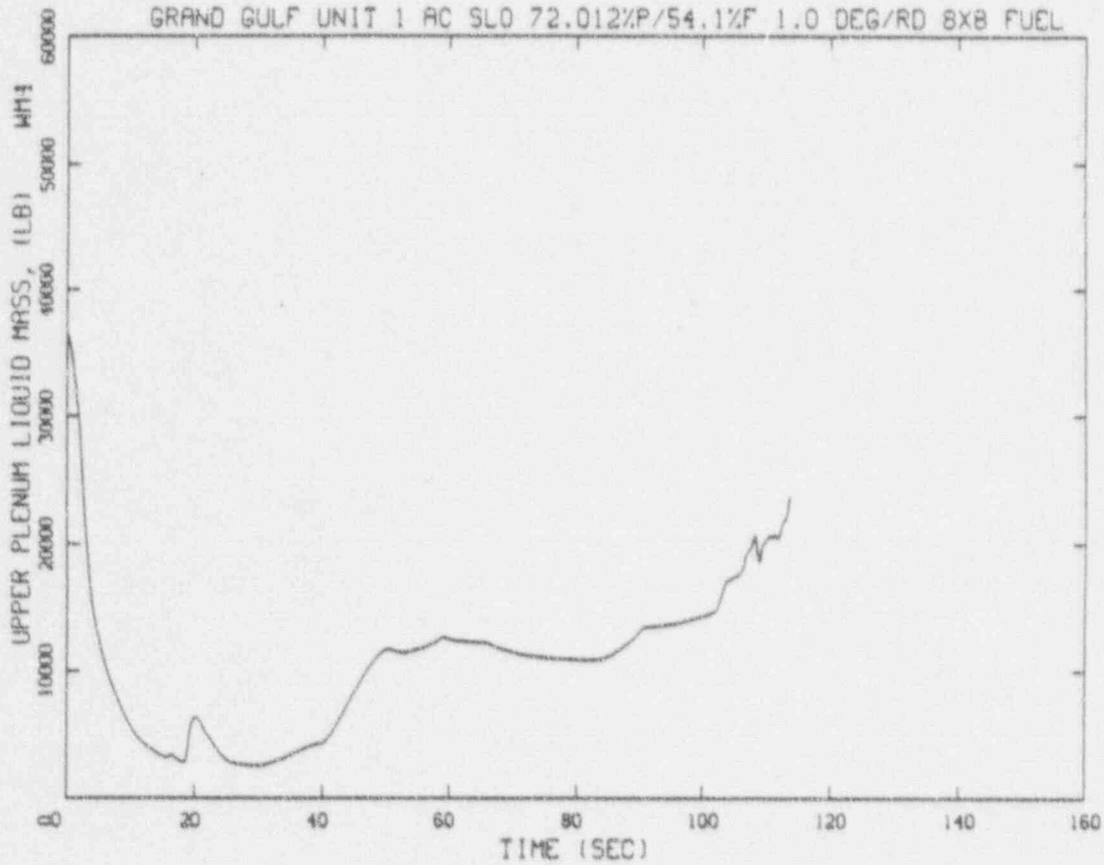

FIGURE 4.21 UPPER PLENUM LIQUID MASS

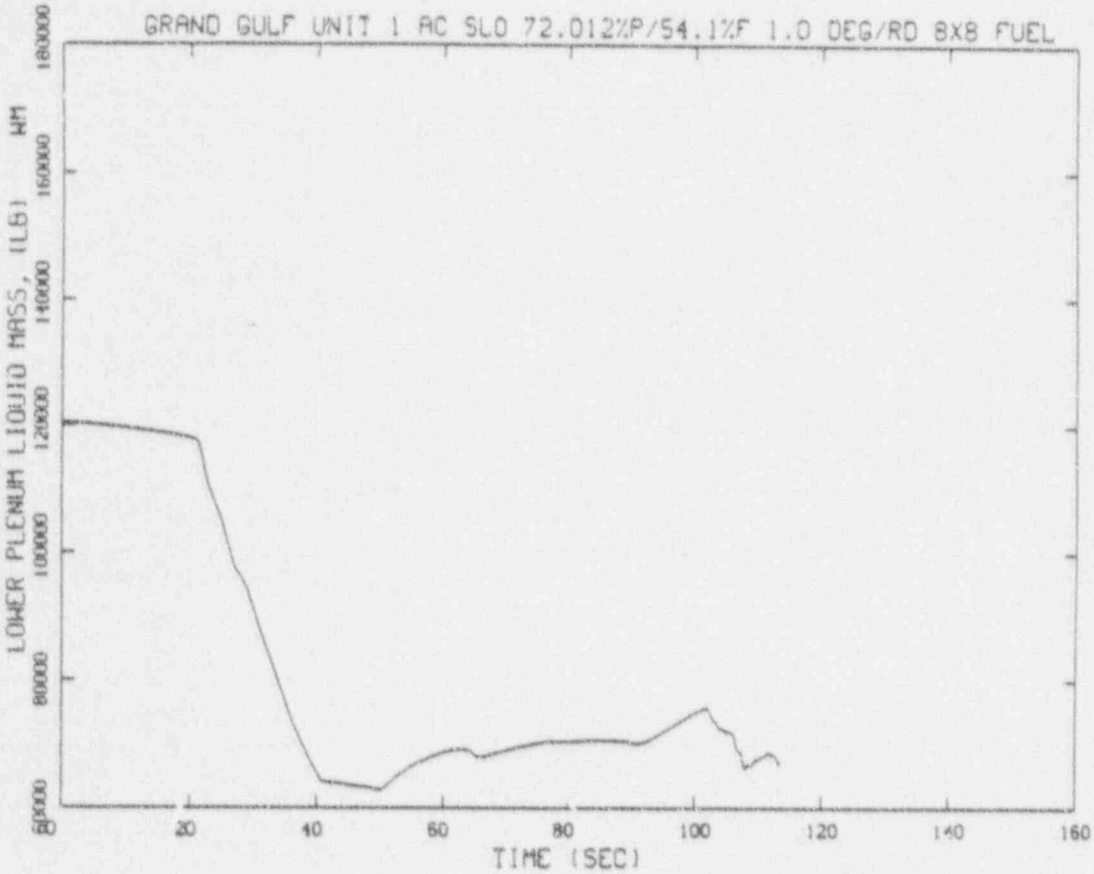

FIGURE 4.22 LOWER PLENUM LIQUID MASS

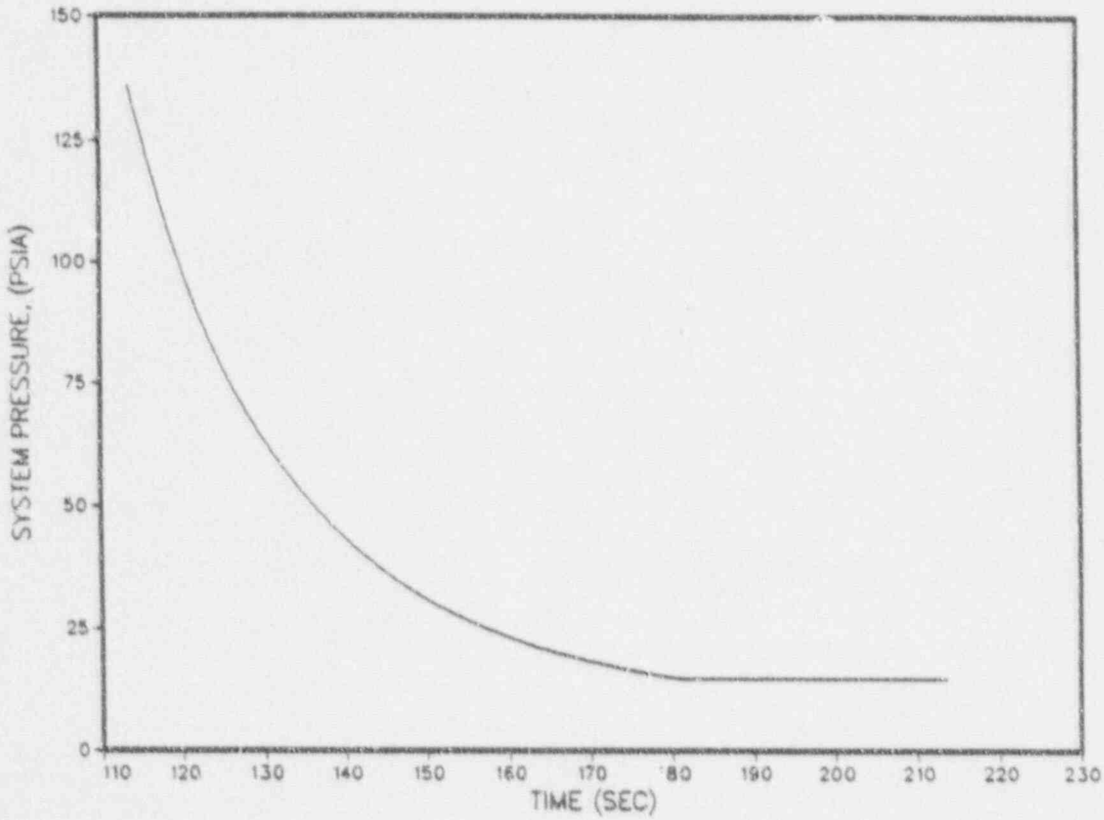

FIGURE 4.23 REFILL/REFLOOD SYSTEM PRESSURE

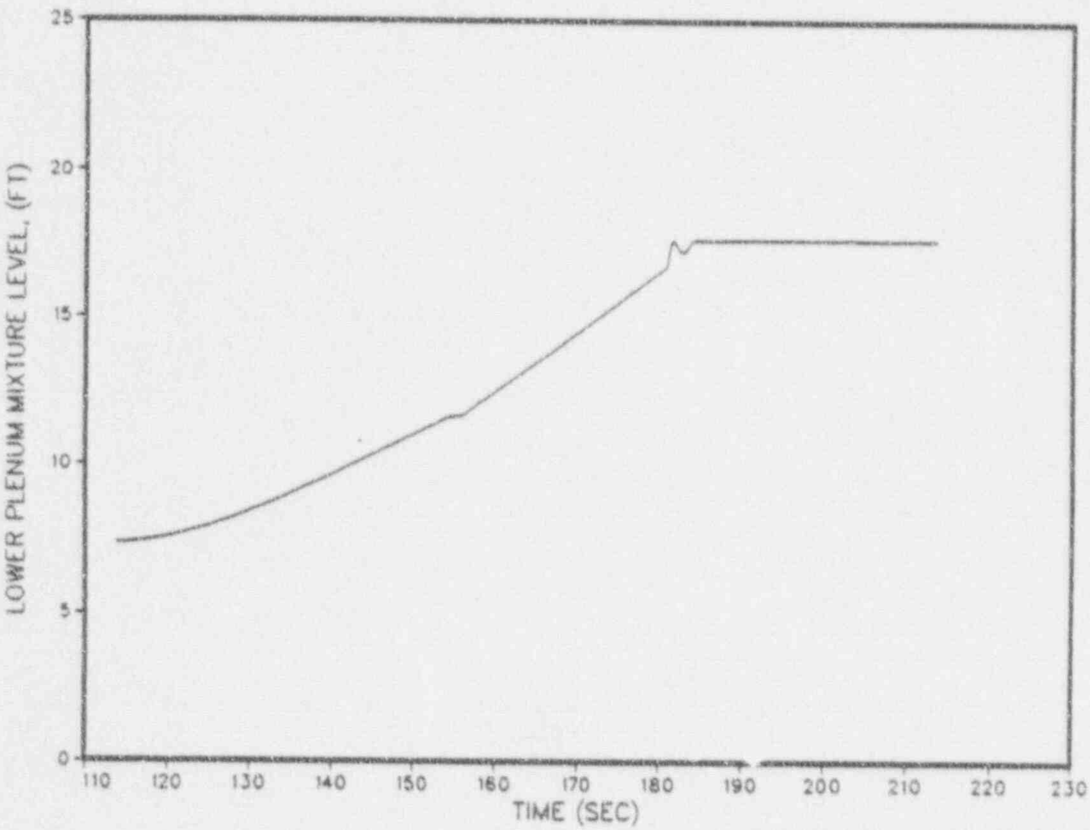

FIGURE 4.24 REFILL/REFLOOD LOWER PLENUM MIXTURE LEVEL

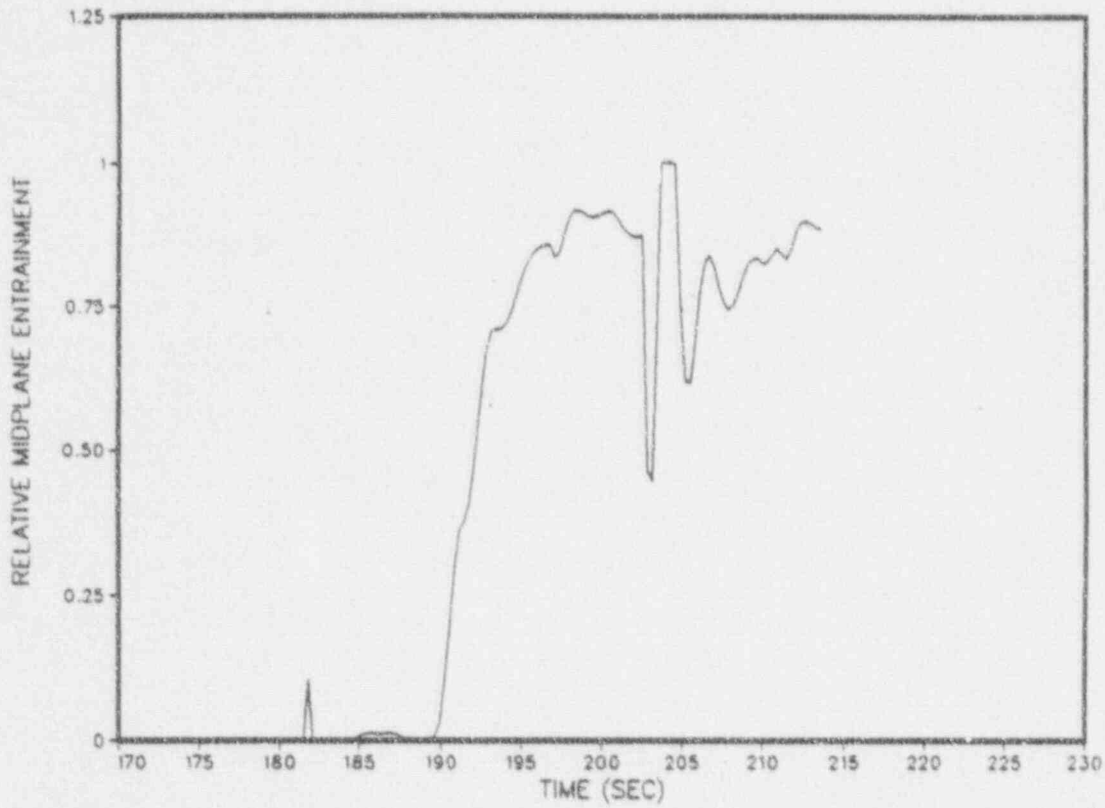

FIGURE 4.25 REFILL/REFLOOD RELATIVE CORE MIDPLANE ENTRAINMENT

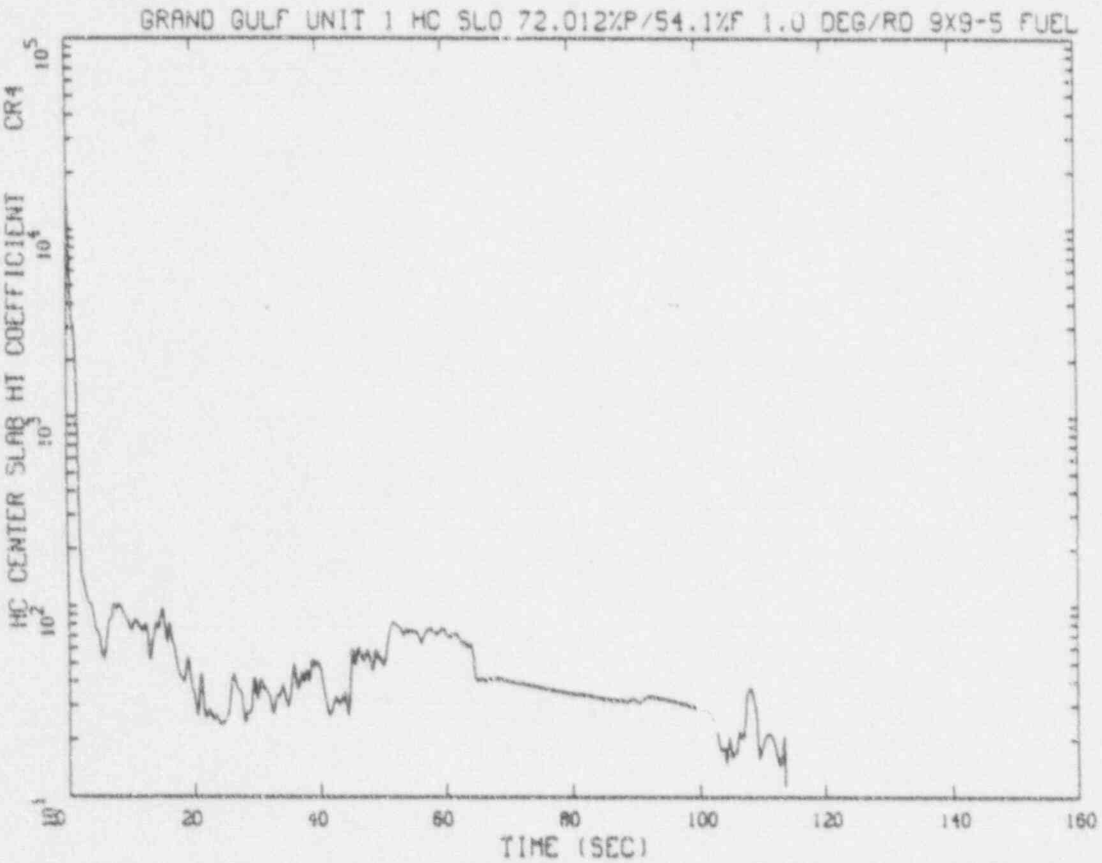

FIGURE 4.26 HOT CHANNEL CENTER SLAB HEAT TRANSFER COEFFICIENT

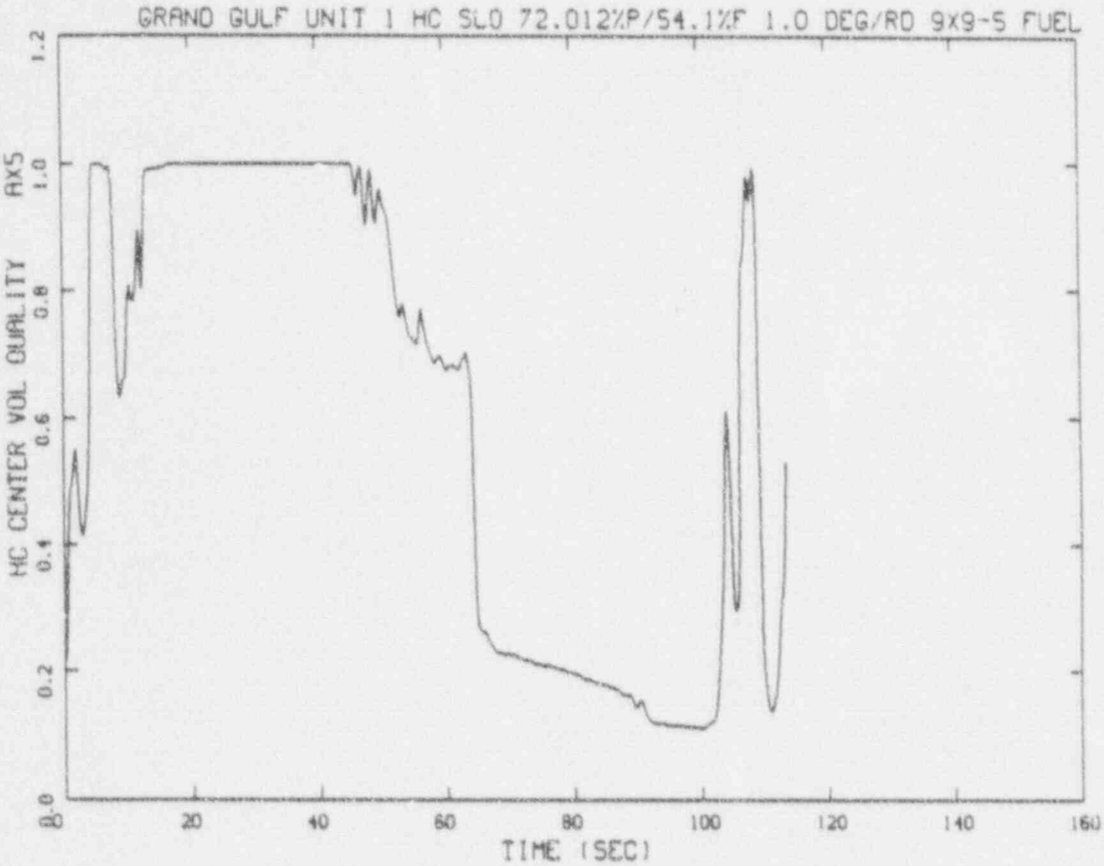

FIGURE 4.27 HOT CHANNEL CENTER VOLUME QUALITY

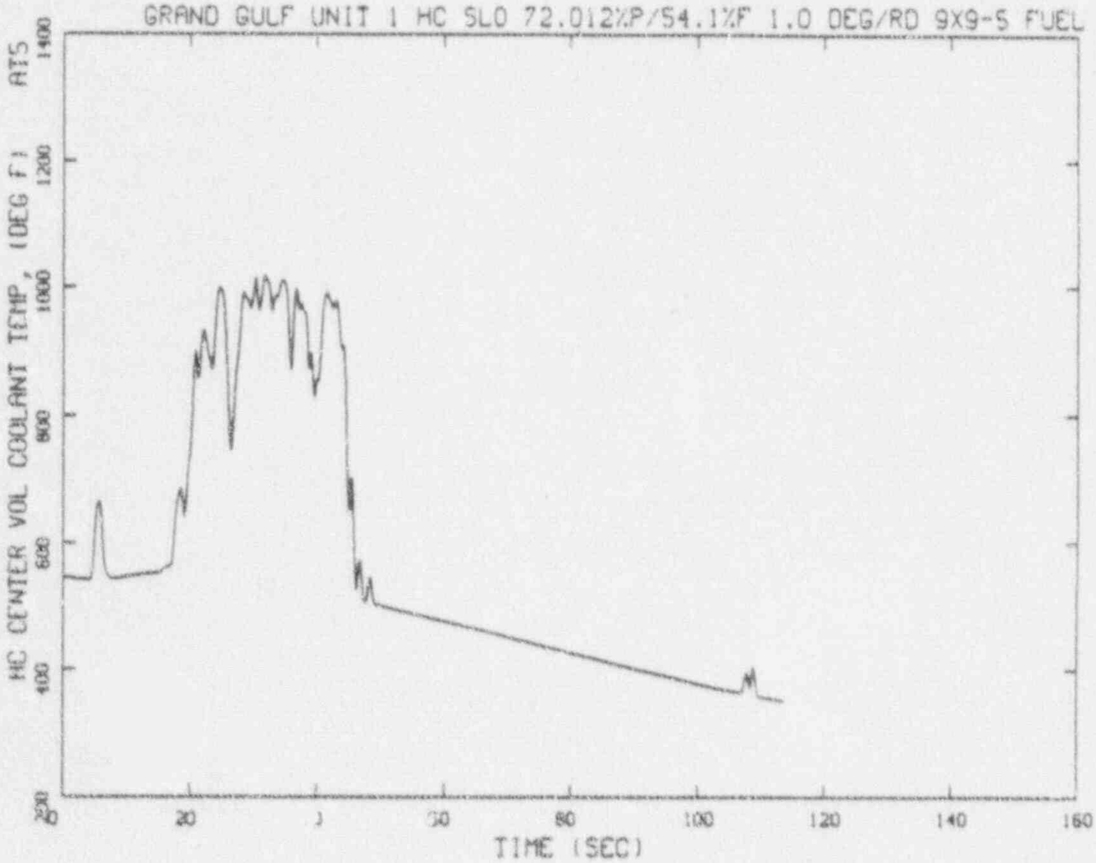

FIGURE 4.28 HOT CHANNEL CENTER VOLUME COOLANT TEMPERATURE

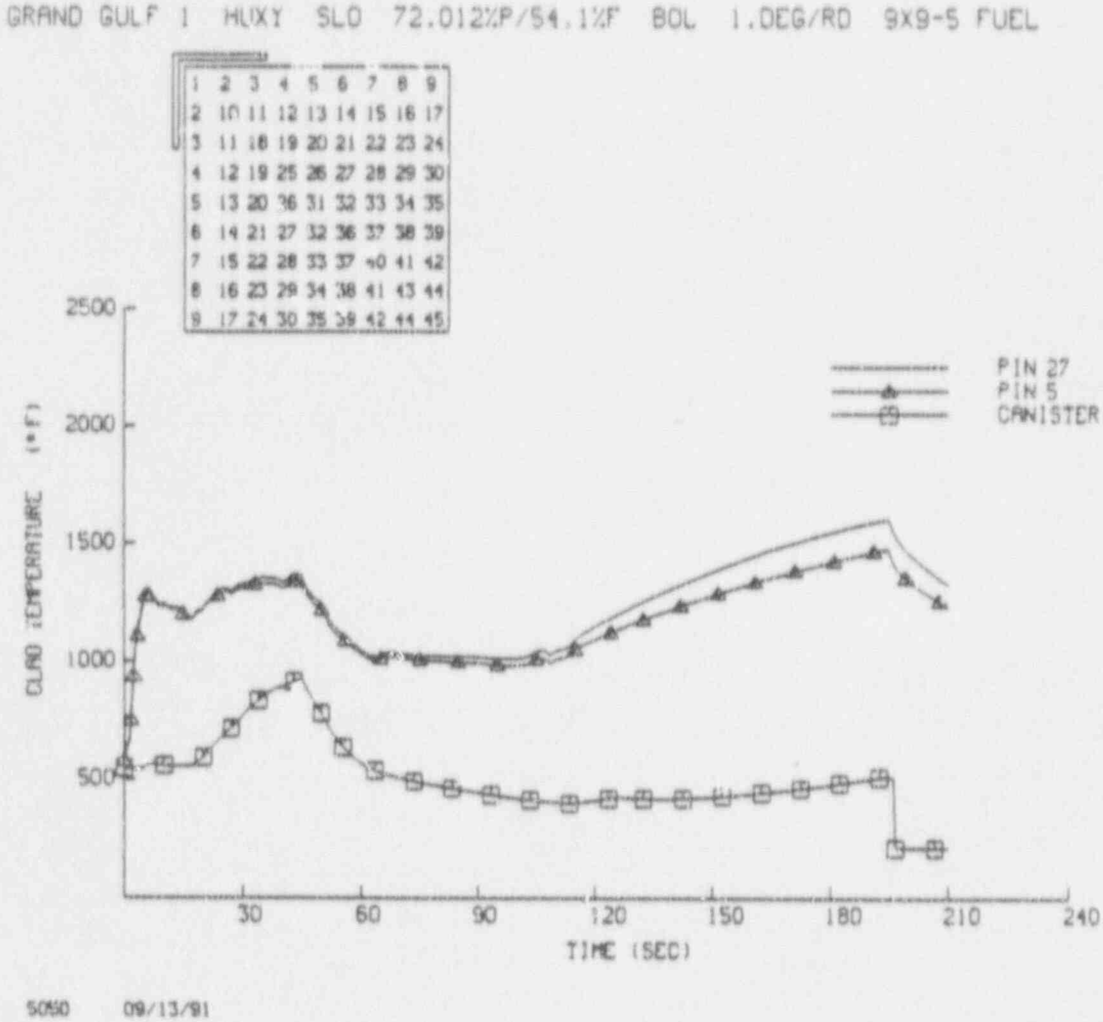

FIGURE 4.29 HOT ASSEMBLY HEATUP RESULTS FOR SNP 9x9-5 FUEL (BOL)

5.0 REFERENCES

1. "Exxon Nucle[illegible] [illegible]gy for Boiling Water Reactors: EXEM ECCS Evaluation Model, Summary Des[illegible]on," [illegible]NF-80-19(A), Volume 2, Revision 1, June 1981.

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

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

4. "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 1975.

5. "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," XN-NF-82-07(A), January 1982.

6. "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," XN-NF-81-58(A), Revision 2, Exxon Nuclear Company, March 1984.

7. "Generic LOCA Break Spectrum Analysis for BWR/6 Plants," XN-NF-86-37(P), Exxon Nuclear Company, Inc., April 1986.

8. "Grand Gulf Unit 1 LOCA Analysis," XN-NF-86-38, Exxon Nuclear Company, Inc., June 1986.

9. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46, and "ECCS Evaluation Models," Appendix K of 10 CFR 50.

10. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance With 10 CFR 50 Appendix K, Amendment 2-One Recirculation Loop Out-of-Service," NEDO-20566-2, Revision 1, Class 1, July 1978.

11. "Grand Gulf Nuclear Station Updated Final Safety Analysis Report," Appendix 15C.

12. "Grand Gulf Unit 1 Cycle 5 Reload Analysis," ANF-90-022, Revision 2, Advanced Nuclear Fuels Corporation," August 1990.

13. "Grand Gulf Unit 1 Cycle 6 Reload Analysis," EMF-91-169, Siemens Nuclear Power Corporation," October 1991.

EMF-91-172
Issue Date: 10/11/91

GRAND GULF UNIT 1 LOCA ANALYSIS FOR SINGLE LOOP OPERATION

Distribution

D. J. Braun
R. A. Copeland
L. J. Federico
D. E. Garber
N. L. Garner
D. E. Hershberger
M. J. Hibbard
J. N. Morgan
R. S. Reynolds
C. C. Roberts, Jr.
A. W. Will

Entergy Operations/S. L. Leonard (40)
Document Control (5)

Attachment 9 to GNRO-91/00186