EXHIBIT F

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Dated January 13, 1986

DEMONSTRATION OF THE CONFORMANCE OF PRAIRIE ISLAND UNITS TO APPENDIX K AND 10CFR50.46 FOR SMALL BREAK LOCAs

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I. Introduction

This document reports the results of an analysis that was performed to demonstrate that Prairie Island, Units I and II, meet the requirements of Appendix K and 10CFR50.46 for small break LOCA. The analysis incorporates anticipated plant hardware modifications, i.e., the new upper reactor internals package and the thimble plug removal, as well as increased levels of F_Q and steam generator tube plugging.

II. Method of Analysis

The analysis was performed using the W NOTRUMP and LOCTA computer codes for a spectrum of small break sizes. These codes are incorporated in the approved Westinghouse ECCS Small Break Evaluation Model developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

The NOTRUMP computer code is a one-dimensional general network code consisting of a number of advanced features, including the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. NOTRUMP includes the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Cladding thermal analyses are performed with the LOCTA-IV (Reference 3) code which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations, as input.

Table 1 lists important input parameters and initial conditions used in the NOTRUMP analysis. The core power decay and axial power distribution are shown in Figures 1 and 2. For these analyses, the SI delivery considers pump injection flow which is depicted in Figure 3 as a function of RCS pressure. This figure represents injection flow from the SI pumps based on performance curves degraded 5 percent from the design head. The Safety Injection (SI) system was assumed to be delivering to the RCS 25 seconds after the generation of a safety injection signal. The 25-second delay includes time required for diesel startup and loading of the safety injection pumps onto the emergency buses. Minimum safeguards Emergency Core Cooling System capability and operability has, also, been assumed in this analysis.

The hydraulic analyses are performed with the NOTRUMP code using 102 percent of the licensed NSSS core power. The core thermal transient analyses are performed with the LOCTA-IV code using 102 percent of licensed NSSS core power.

Three break size transients were evaluated, 3 inch, 4 inch, and 6 inch. These transients were considered to be terminated when the following criteria were met:

1. RCS system pressure had decreased below the accumulator set point and accumulator flow had been initiated.

2. The core had been recovered and the core/upper plenum mixture level was at or above the bottom of the vessel outlet nozzles.

3. The net flow to the RCS was positive with accumulator and SI flow exceeding the break flow.

III. Results and Conclusions

Of the three break sizes evaluated, core uncovery occurred only for the 4-inch break case, Figures 5, 7, and 10. Accumulator injection terminates the 3-inch break transient; producing a positive slope on the core mixture level curve. A similar result occurs for the 6-inch break. The 4-inch break shows two brief periods of core uncovery prior to accumulator injection. During the second period of uncovery a PCT of 1000°F occurs, Figure 8. This value is well below all Acceptance Criteria limits of 10CFR50.46 and is non-limiting in comparison to large break analysis results.

REFERENCES

1. Lee, H., Rupprecht, S. D., Tauche, W. D., Schwarz, W. R., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, August 1985.

2. Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.

3. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (Proprietary), and WCAP-8305 (Non-Proprietary), June 1974.

TABLE 1

INPUT PARAMETERS USED IN THE SMALL BREAK ANALYSES

Parameter	Small Break
Peak Linear Power (kw/ft) (includes 102% factor)	15.03
Total Peaking Factor, F_Q	2.50
Power Shape	See Figure 2
Fuel Assembly Array	14x14 OFA
Nominal Cold Leg Accumulator Water Volume (ft^3/accumulator)	1266
Nominal Cold Leg Accumulator Tank Volume (ft^3/accumulator)	2000
Minimum Cold Leg Accumulator Gas Pressure (psia)	715
Pumped Safety Injection Flow	See Figure 3
Steam Generator Initial Pressure (psia)	733.0
Steam Generator Tube Plugging Level %	10
Fuel Assembly Thimble Plugs	Removed
Reactor Upper Internals Package	New Design

TABLE 2

SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

	3 in. Break (sec)	4 in. Break (sec)	6 in. Break (sec)
Start	0.0	0.0	0.0
Reactor Trip	4.5	2.8	1.7
Top of Core Uncovered	N/A	183.7	N/A
Cold Leg Accumulator Injection	703.0	371.5	162.6
Peak Clad Temperature Occurs	N/A	297.8	N/A
Top of Core Covered	N/A	325.8	N/A

TABLE 3

SMALL BREAK LOCA RESULTS
FUEL CLADDING DATA

	3 in. Break	4 in. Break	6 in. Break
Results			
Peak Clad Temperature (^{o}F)	N/A	1000	N/A
Peak Clad Location (ft)	N/A	12.00	N/A
Local Zr/H_2O Reaction (max), (%)	N/A	0.066	N/A
Local Zr/H_2O Reaction Location (ft)	N/A	12.00	N/A
Total Zr/H_2O Reaction (%)	<.3	<.3	<.3
Hot Rod Burst Time (sec)	N/A	N/A	N/A
Hot Rod Burst Location (ft)	N/A	N/A	N/A

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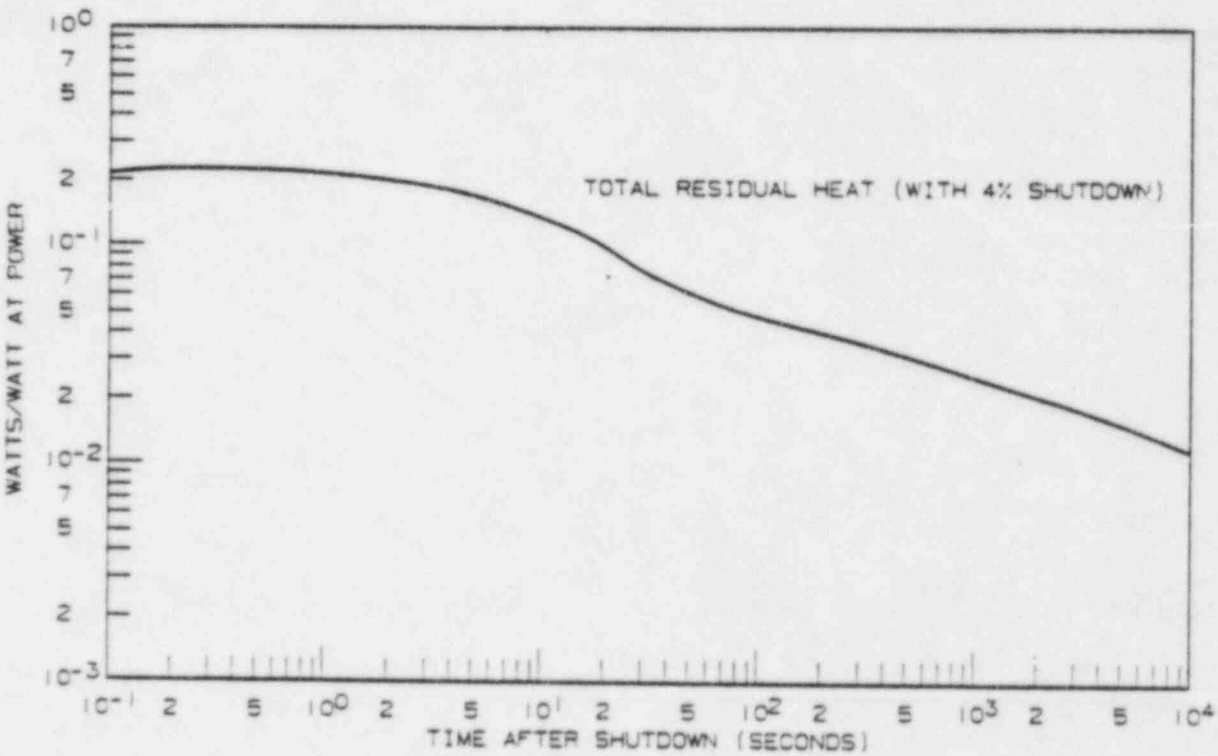

Figure 1. Core Power After Reactor Trip

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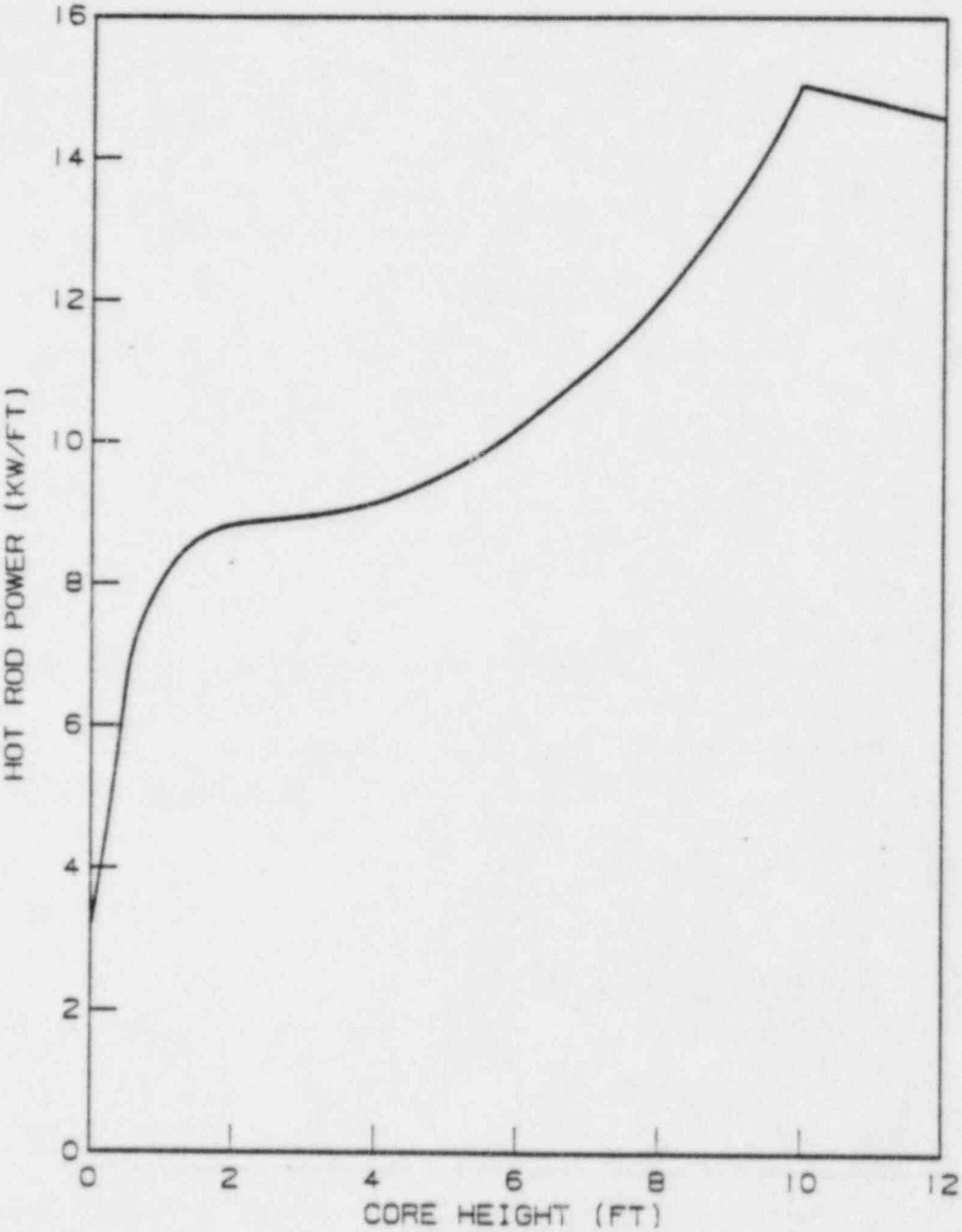

Figure 2. Small Break Power Distribution Assumed for Loca Analysis

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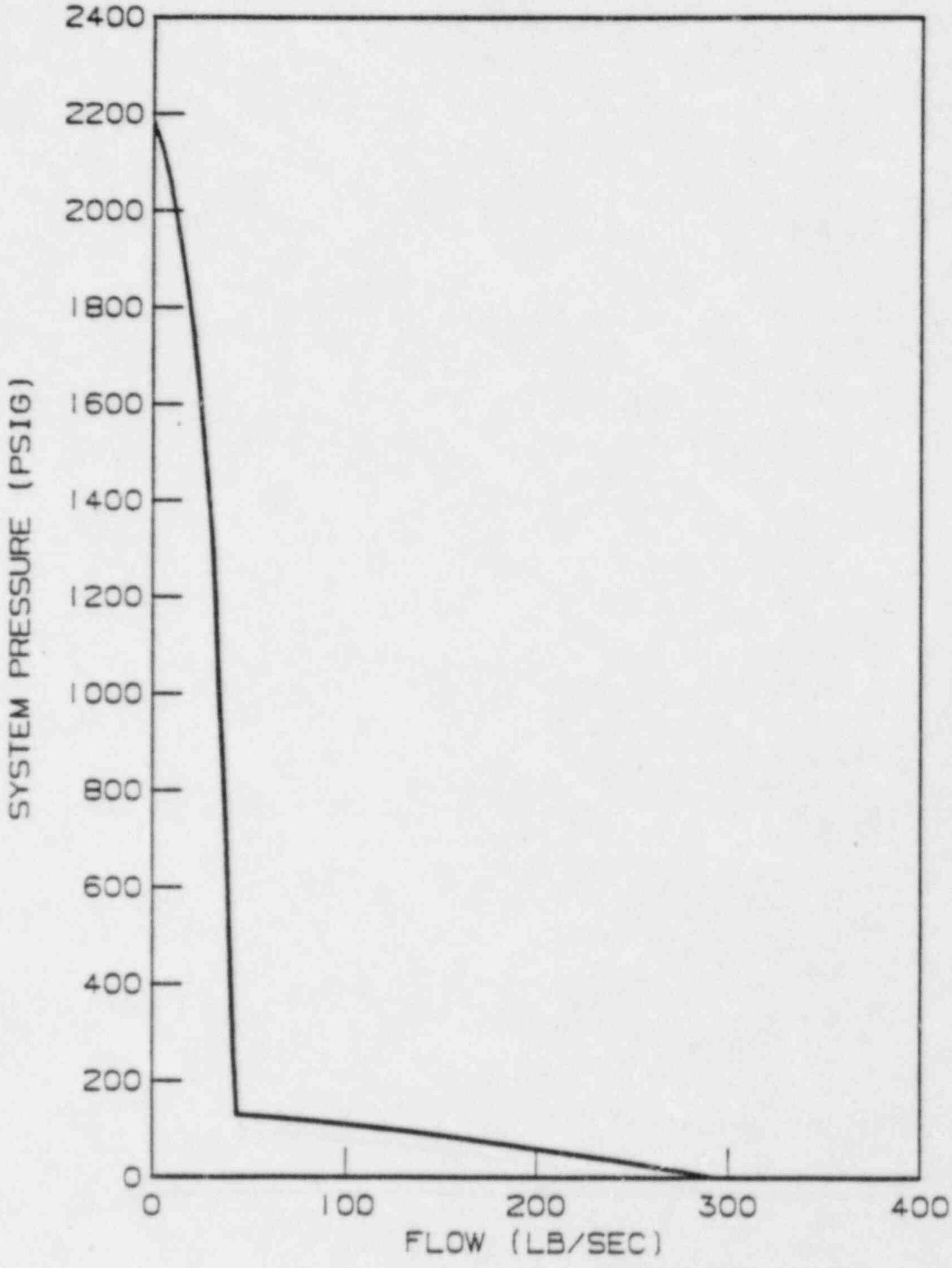

Figure 3. Safety Injection Flowrate Versus Pressure

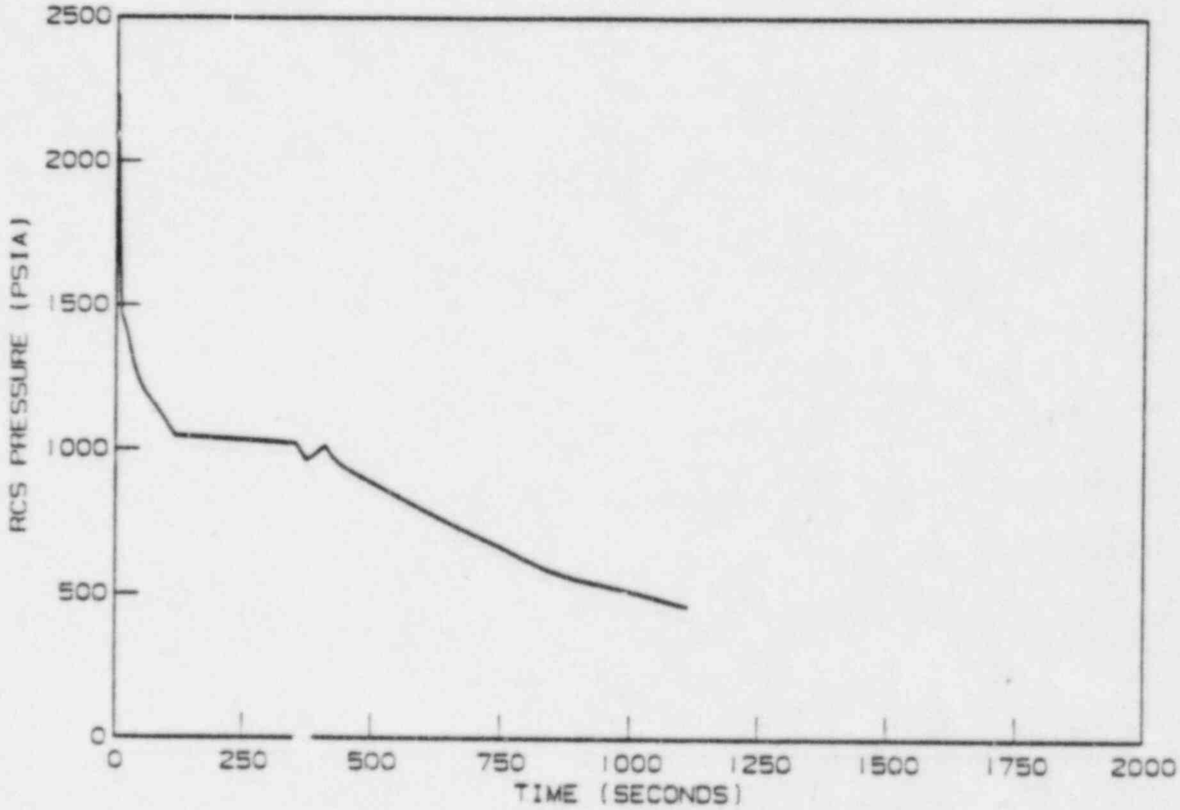

Figure 4. 3-Inch Cold Leg Break RCS Pressure Versus Time

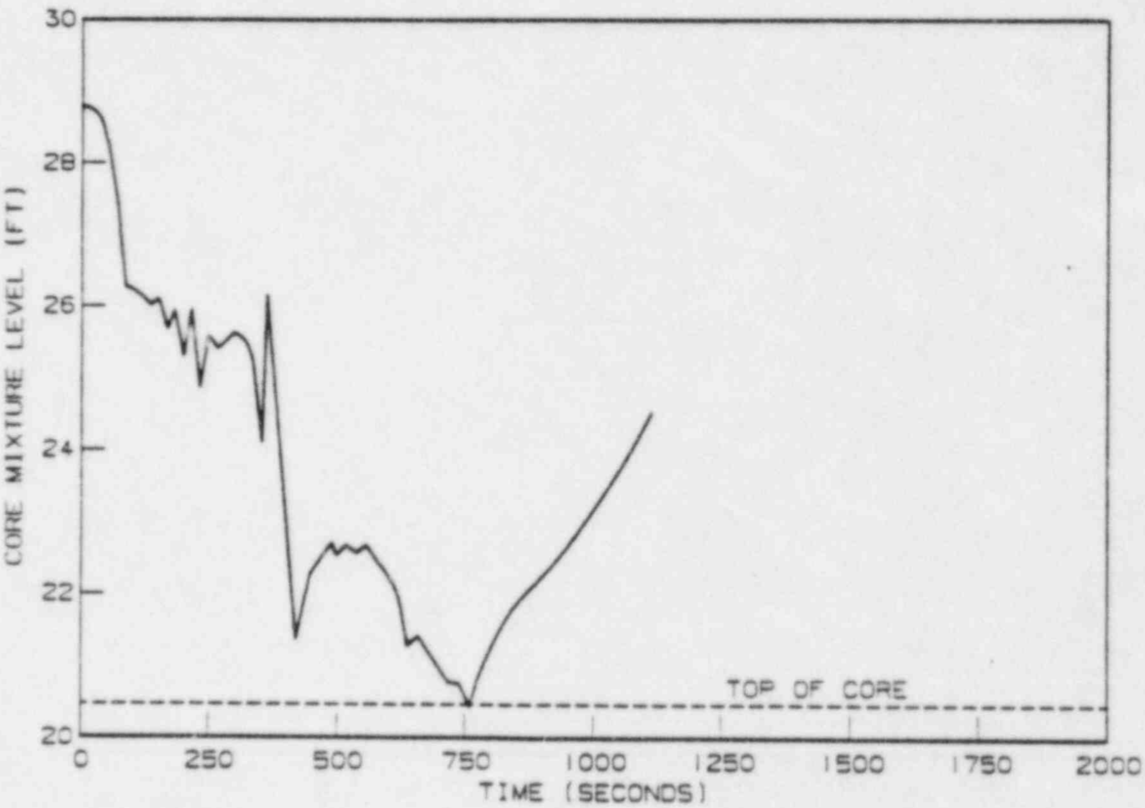

Figure 5. 3-Inch Cold Leg Break Core Mixture Level Versus Time

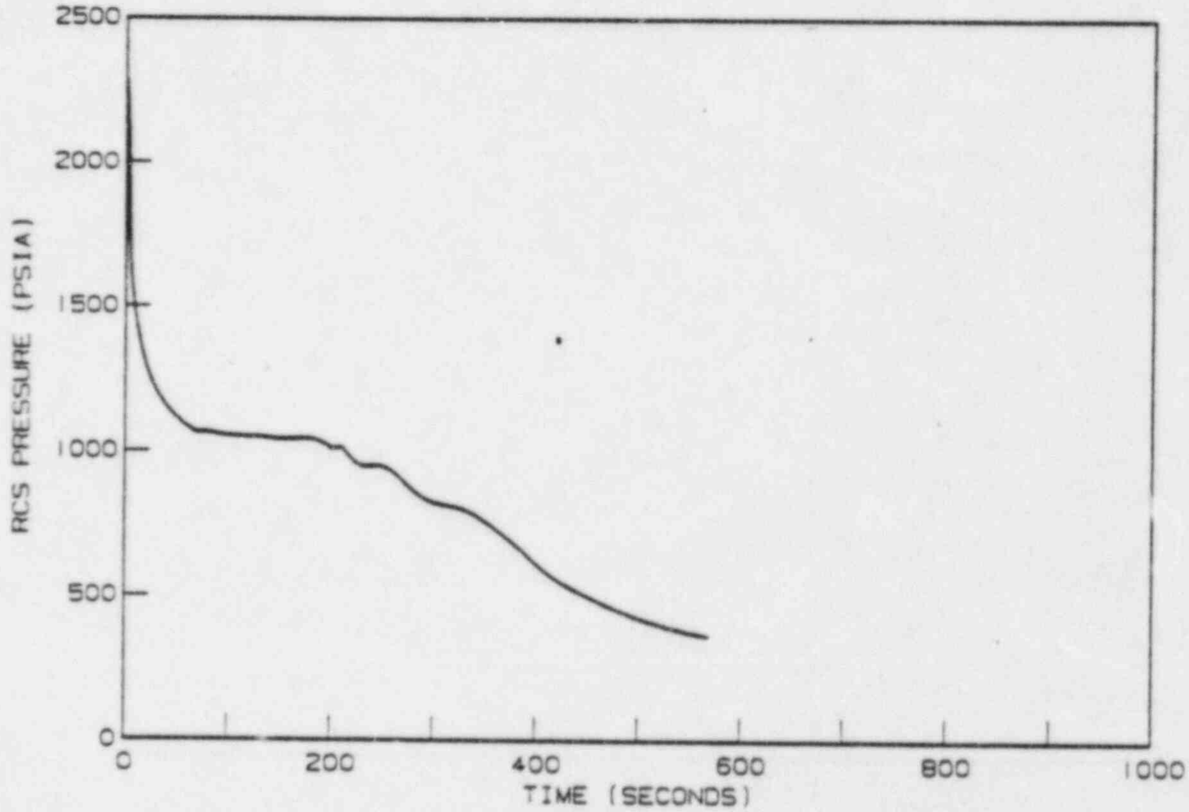

Figure 6. 4-Inch Cold Leg Break RCS Pressure Versus Time

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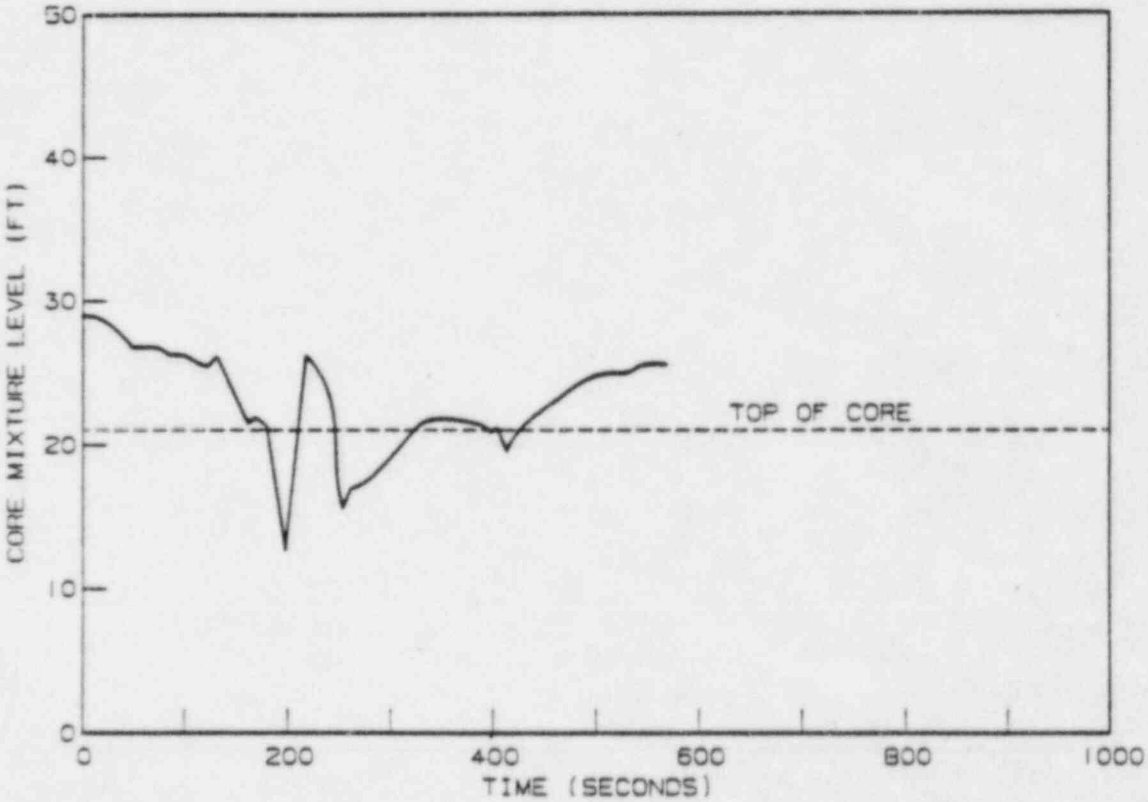

Figure 7. 4-Inch Cold Leg Break Core Mixture Level Versus Time

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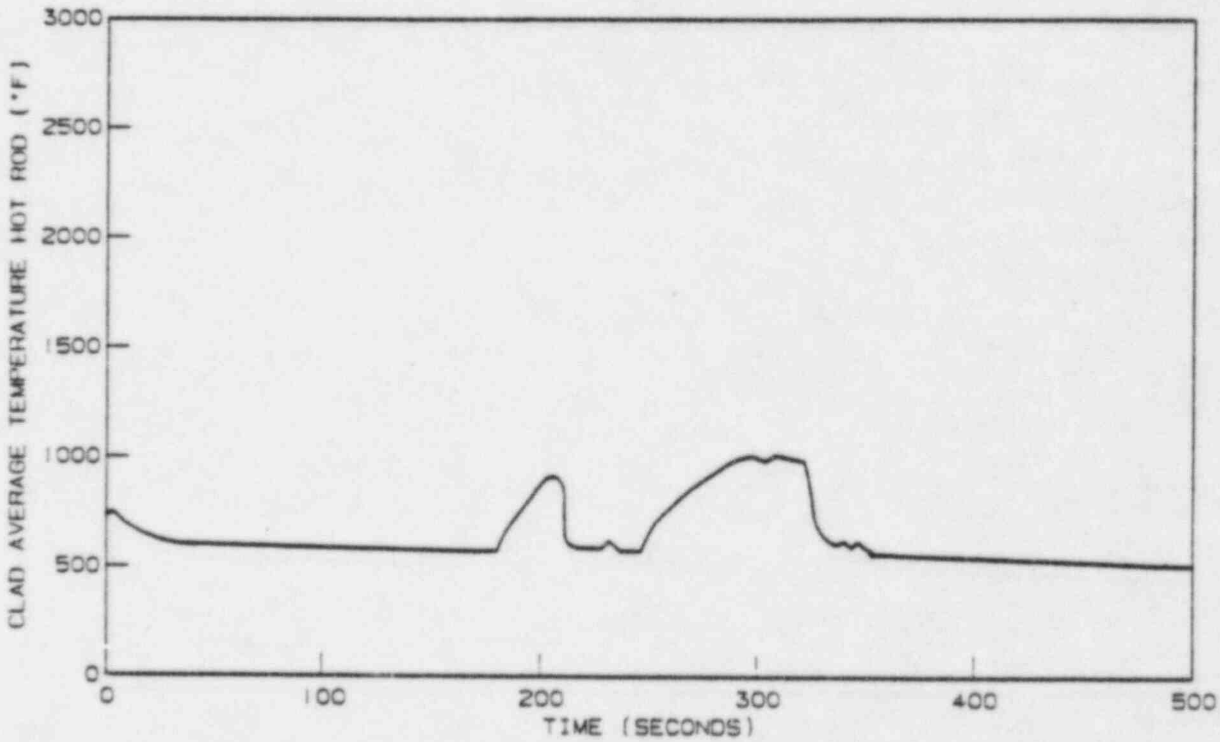

Figure 8. 4-Inch Cold Leg Break Clad Average Temperature, Hot Rod, Versus Time

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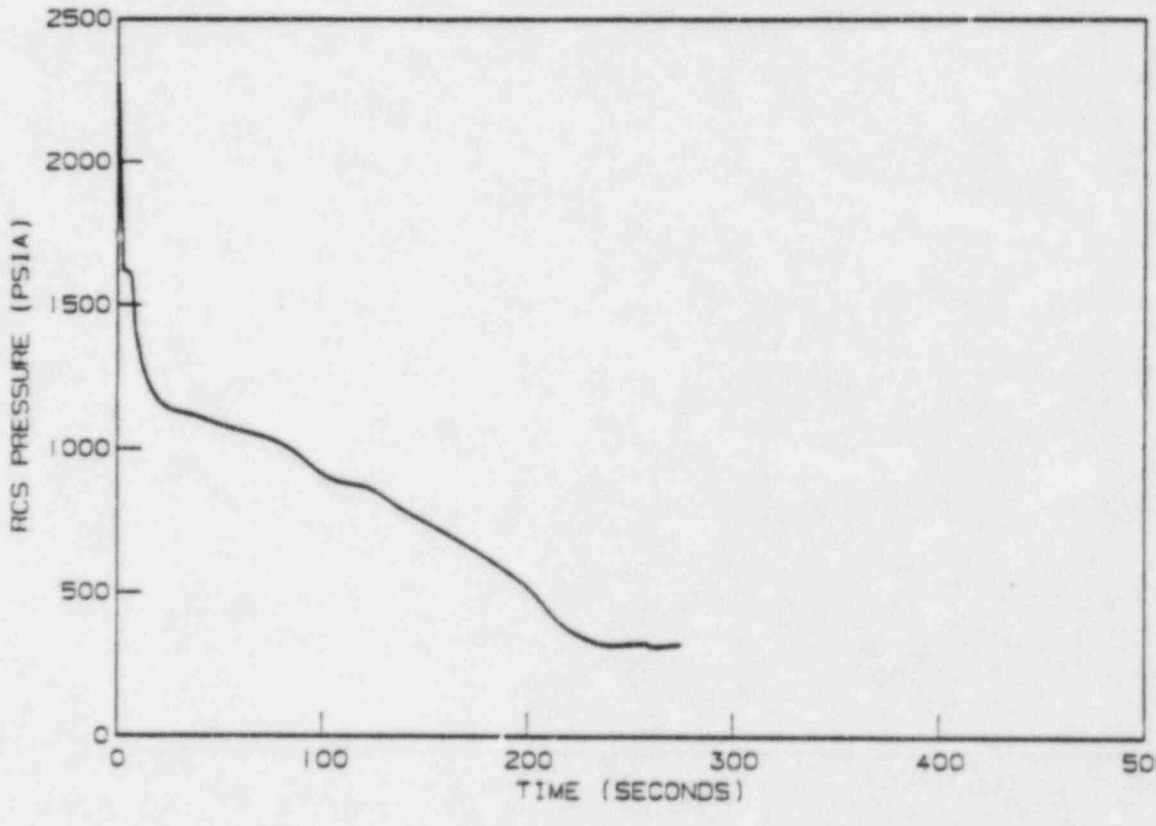

Figure 9. 6-Inch Cold Leg Break RCS Pressure Versus Time

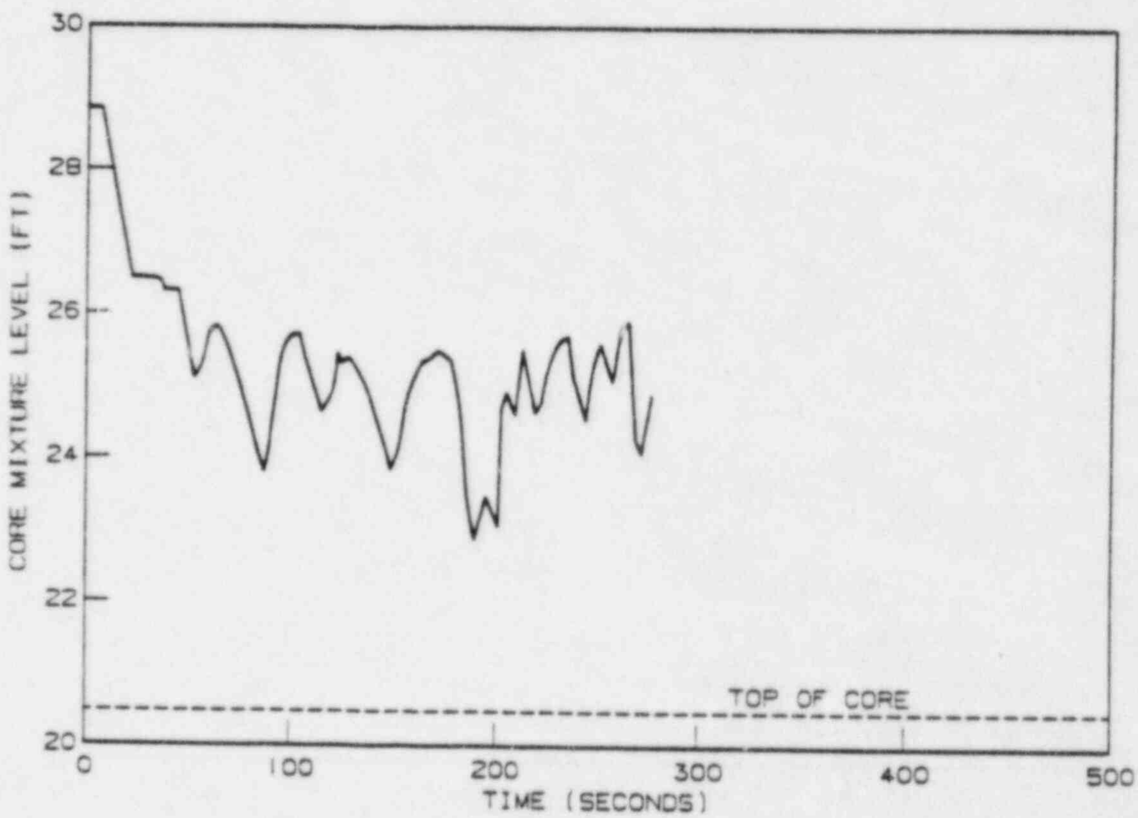

Figure 10. 6-inch Cold Leg Break Core Mixture Level Versus Time