EXHIBIT G

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Dated January 13, 1986

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

Westinghouse Electric Corporation

Nuclear Technology Division

Nuclear Safety Department

Safeguards Engineering and Development

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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 with Exxon fuel assemblies, 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 fuel assembly thimble plug removal, as well as increased levels of Fq to 2.5 and 10% steam generator tube plugging.

II. Method of Analysis

The analysis was performed for Exxon fuel assemblies in the Prairie Island Units and was an extension of an analysis for small breaks with Westinghouse fuel in the same Units. Exxon fuel parameters which affect hydraulic performance are similar to those of Westinghouse fuel, with only 3.4% greater core pressure loss and 6% less core flow area. Given the significant similarity in hydraulic characteristics and the limited impact of core hydraulics during the quasi-steady state small break LOCA transient, the Westinghouse fuel NOTRUMP calculations adequately defined the hydraulic transient for the Exxon fuel. Thus, the fuel rod response calculation using the Exxon fuel parameters in the LOCTA code was performed with the NOTRUMP inputs from the Westinghouse fuel analysis.

The initial analysis for Westinghouse fuel used 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." Analysis of the Westinghouse fuel showed that the 4 inch break size is limiting for the Prairie Island Units, as the 3 inch and 6 inch breaks showed no core uncovery. Consequently, the Exxon fuel analysis was performed for only the 4 inch break size.

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 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. In this evaluation, Exxon fuel rod parameters were input to the LOCTA code. The fuel design parameters were prepared by the Westinghouse Nuclear Fuel Division using NRC approved methodology and fuel performance models, modified to describe measured Exxon fuel operating performance data. The similiarity of Exxon and Westinghouse fuel as-built and irradiated mechanical properties supports the validity of the model development. The LOCA fuel design parameters calculated by Westinghouse were then compared to with LOCA fuel parameters used by Exxon in the previous cycle LOCA evaluation. This comparison showed good agreement on the fuel temperatures and stored energy and somewhat lower fuel rod internal pressures as a function of fuel rod power. To assure that the new LOCA fuel performance parameters conservatively bound the values used by Exxon in the prior cycle analysis, the values calculated with the Westinghouse models were adjusted upward to match the prior cycle limiting values. The fuel parameters, calculated with the modified models, were used as input to the LOCTA calculations.

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. 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.

III. Results and Conclusions

The 4-inch break shows two brief periods of core uncovery, Figure 5, prior to accumulator injection. During the second period of uncovery a PCT of 978°F occurs, Figure 6. 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, FQ	2.50
Power Shape	See Figure 2
Fuel Assembly Array	14x14 OFA
Nominal Cold Leg Accumulator Water Volume (ft3/accumulator)	1266
Nominal Cold Leg Accumulator Tank Volume (ft3/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

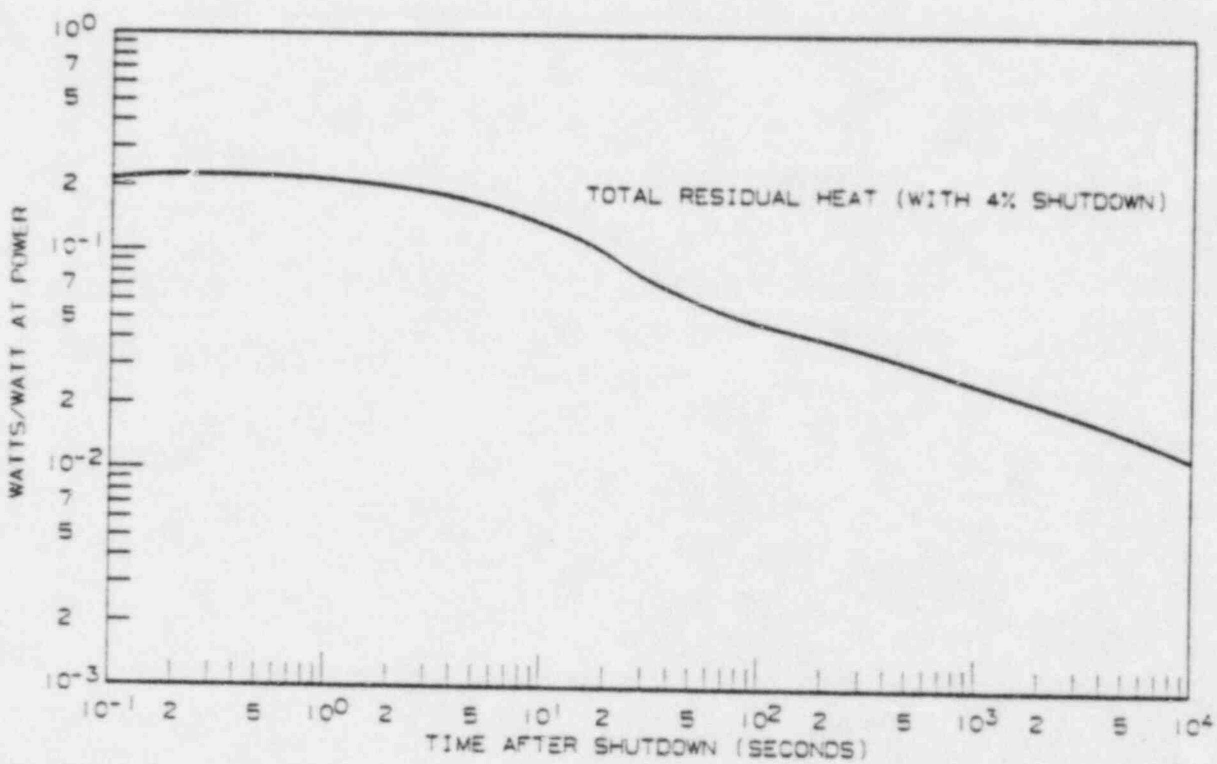

Figure 1. Core Power After Reactor Trip

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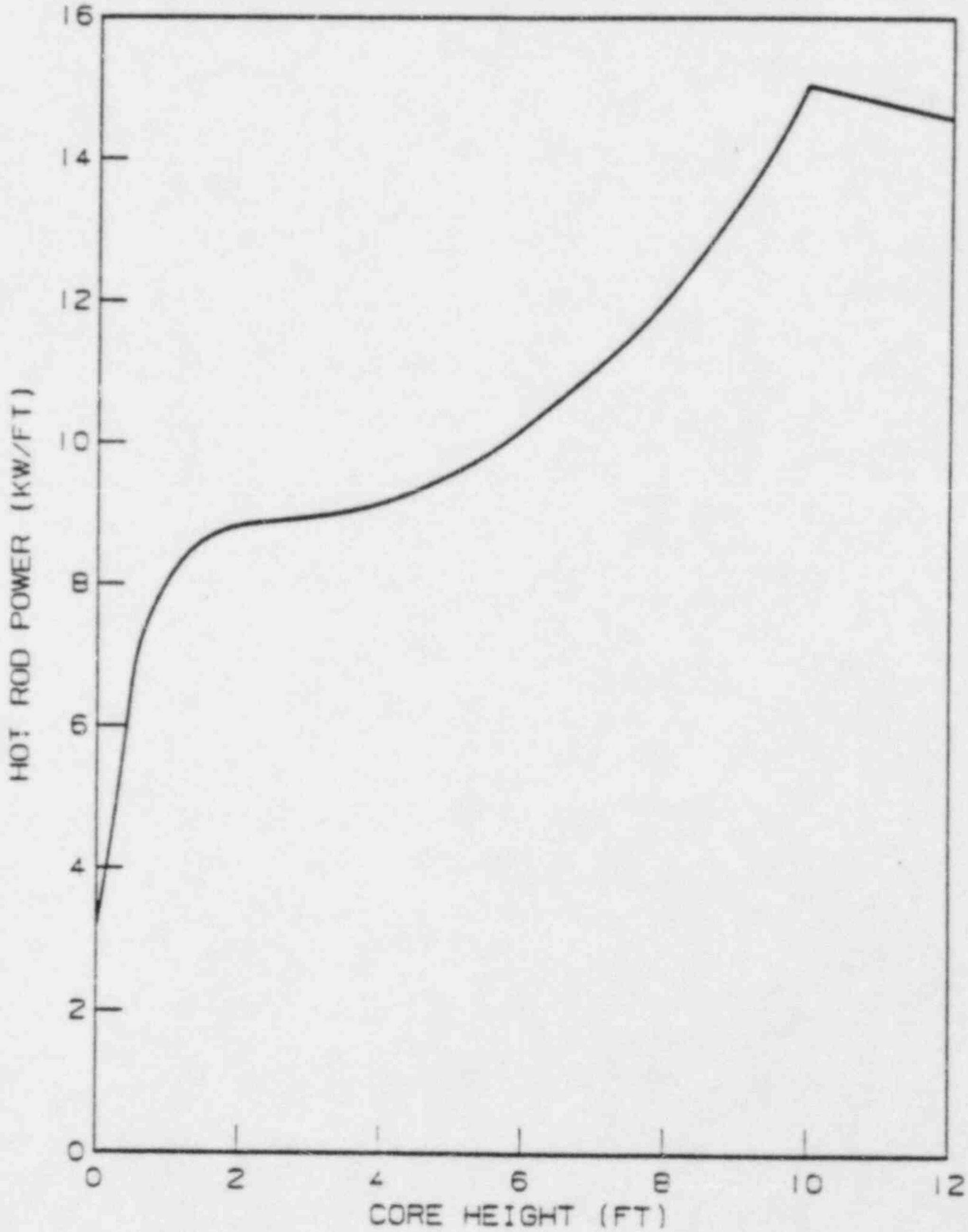

Figure 2. Small Break Power Distribution Assumed for Loca Analysis

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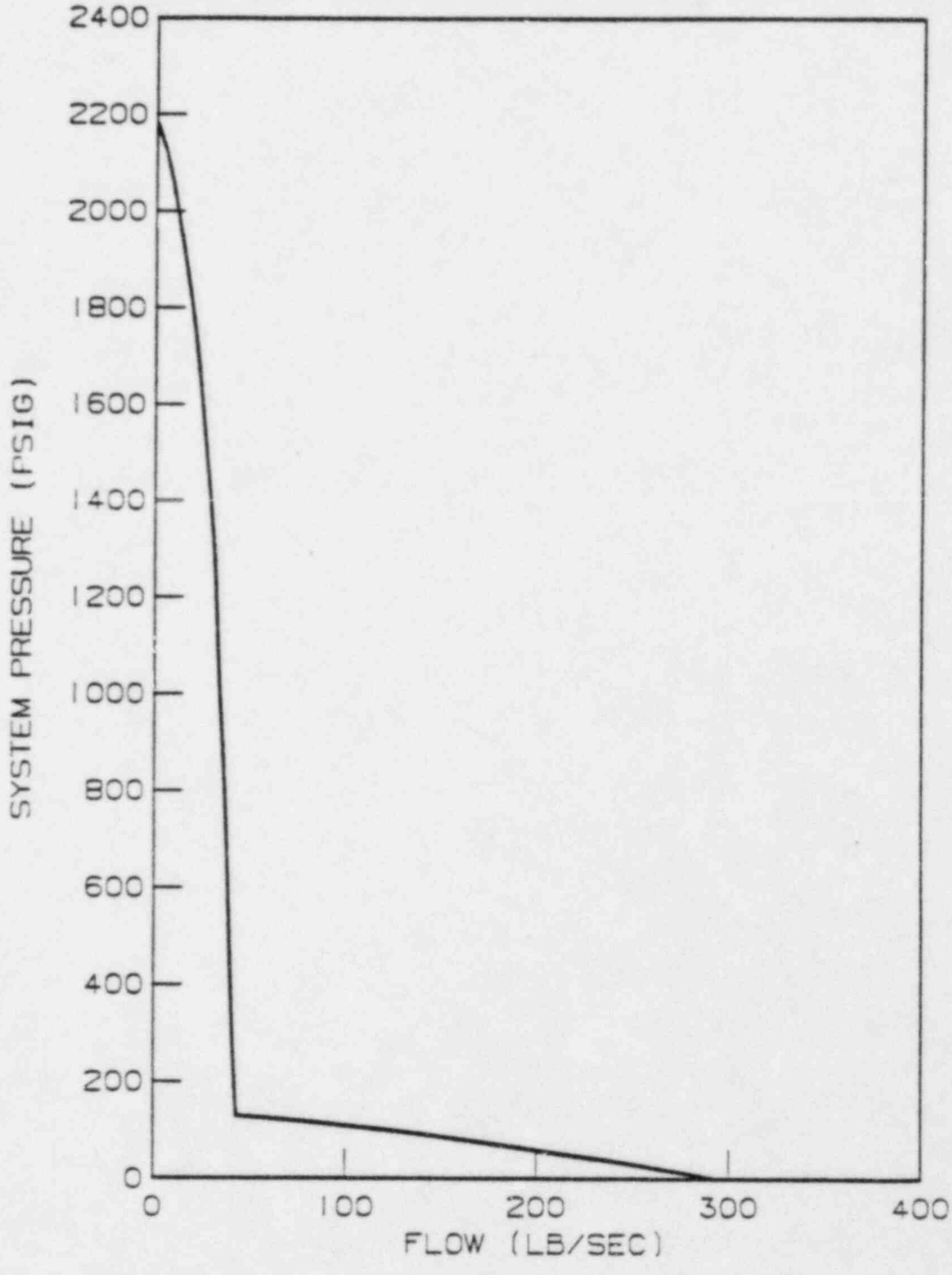

Figure 3. Safety Injection Flowrate Versus Pressure

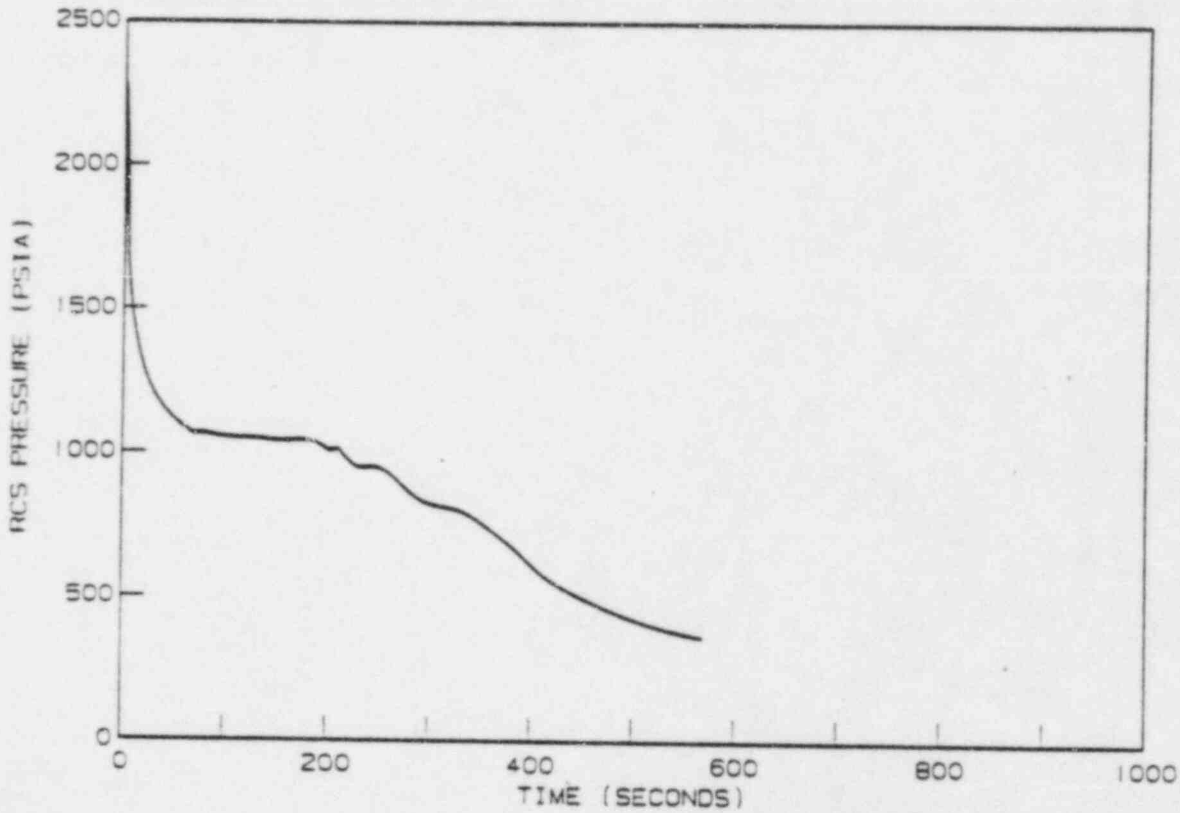

Figure 4. 4-Inch Cold Leg Break RCS Pressure Versus Time

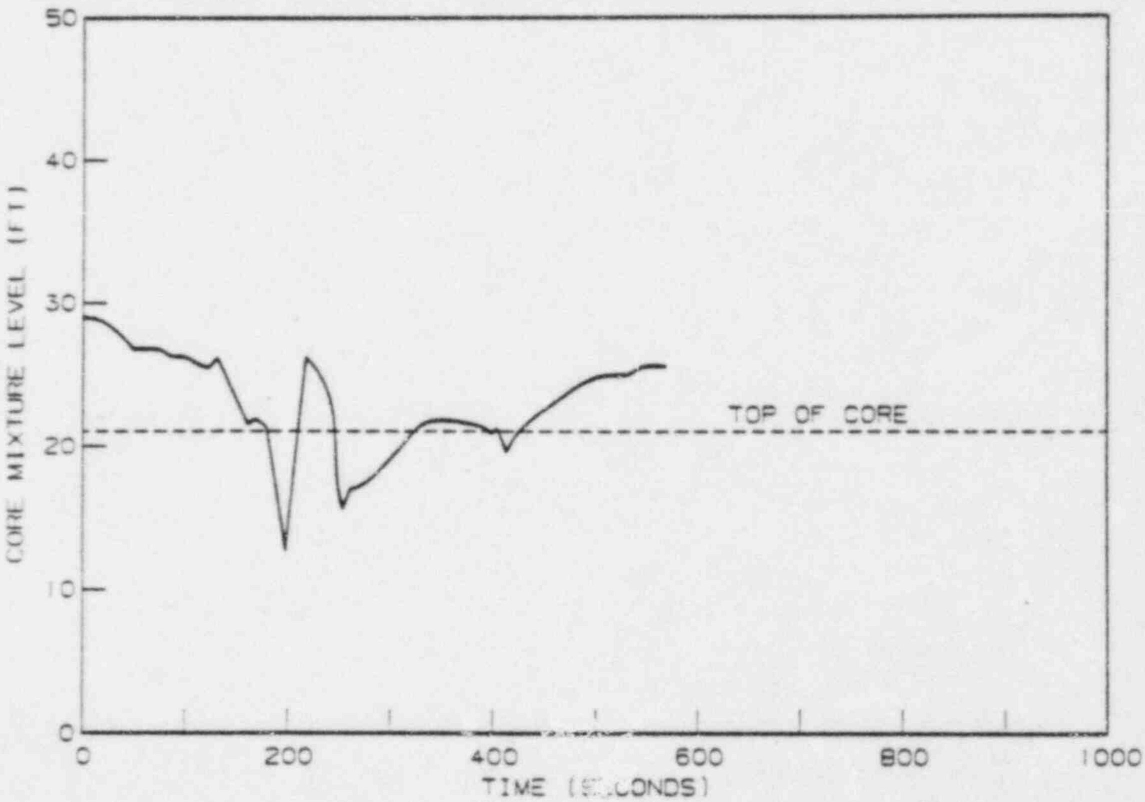

Figure 5. 4-Inch Cold Leg Break Core Mixture Level Versus Time

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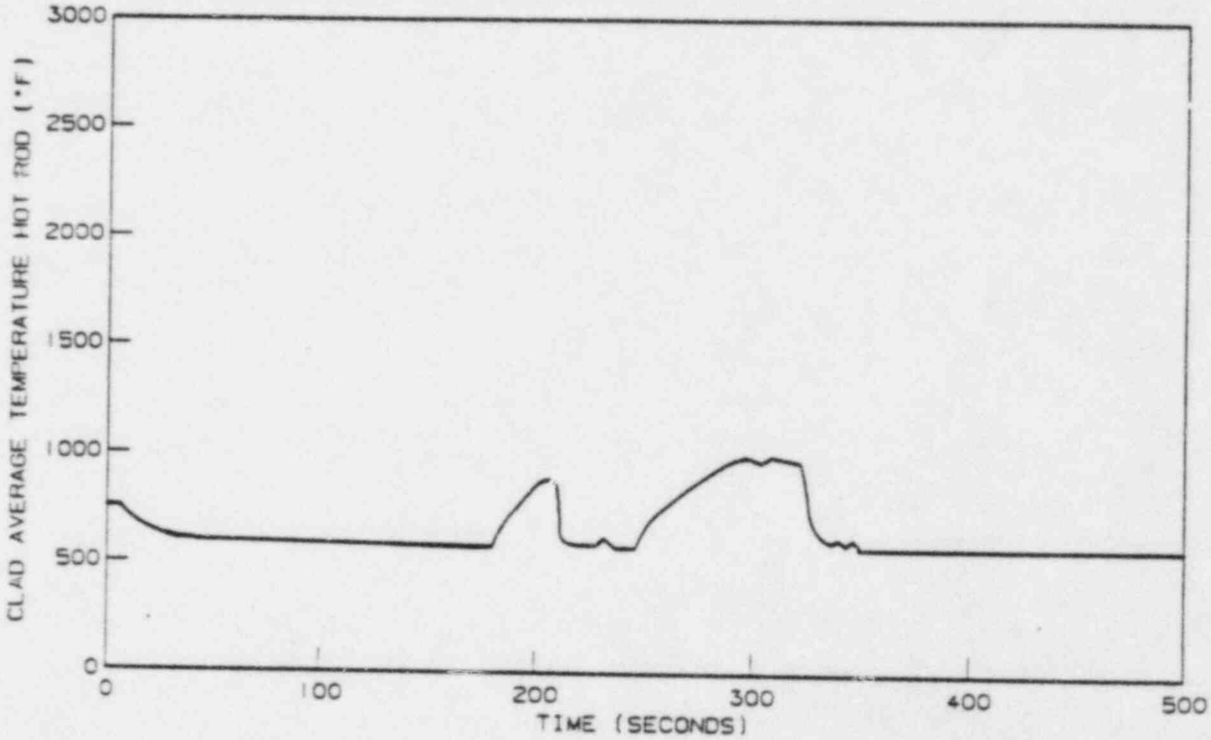

Figure 6. 4-Inch Cold Leg Break Clad Average Temperature, Hot Rod, Versus Time