REVISED PRAIRIE ISLAND

LARGE BREAK

LOCA CALCULATION USING

WCAP-10924 VOL. 1 ADD. 4

MODEL CHANGES

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I. Introduction

This document reports the results of an analysis that was performed to demonstrate that Prairie Island. meets the requirements of Appendix K and 10CFR50.46 for Large Break Loss-of-Coolant-Accidents (LOCA) (Reference 1).

II. Method of Analysis

The analysis was performed using the Westinghouse Large Break LOCA Best-Estimate Methodology as given in (Reference 2) and as amended in Reference 6. The Westinghouse Best-Estimate Methodology was developed consistent with guidelines set forth in the SECY-83-472 document (Reference 3). These guidelines provide for the use of realistic models and assumptions, with the exception of specific models and assumptions required by Appendix K. The technical basis for the use of this model is discussed in detail in Reference 2.

The Best Estimate Methodology is comprised of the WCOBRA/TRAC and COCO computer codes (References 2 and 4, respectively). The WCOBRA/TRAC code was used to generate the complete transient (blowdown through reflood) system hydraulics as well as the cladding thermal analysis. The COCO code was used to generate the containment pressure response to the mass and energy release from the break. This containment pressure curve was used as an input to the WCOBRA/TRAC code.

The fuel parameters used as input for the LOCA analysis were generated using the Westinghouse fuel performance code (PAD 3.3) (Reference 5). The fuel parameters input to the code were at beginning-of-life (maximum densification) values.

The analysis was performed using the four channel core model developed in Reference 2 for the 0.4 double-ended cold leg guillotine (DECLG) break. The transient was considered to be terminated if the hot rod cladding temperature began to decline and the injected ECCS flows exceeded the break flow.

III. Results and Conclusions

Table 1 shows the time sequence of events for the Large Break LOCA transients. Table 2 provides a brief summary of the important results of the LOCA analyses for this calculation. Figures 1 through 8 show important transient results for the limiting 0.4 DECLG break (four channel core model). Note on these figures that the break occurs at time 0.0 (the results from -20.0 to 0.0 are from the steady state). Figure 1 shows the core pressure during the transient. Figure 2 shows the vapor and liquid mass flowrate at the top of the hot assembly. Figures 3 and 4 show the collapsed liquid level in the downcomer and core hot assembly channel, respectively, indicating the refilling of the vessel. Figures 5 and 6 show the flow of ECCS water into the cold leg (accumulator and high head safety injection flow) with Figure 7 showing the flow of low head safety injection into the upper plenum (UPI flow). Figure 8 shows the resulting peak cladding temperature for the 0.4 DECLG break as a function of time for each of the five fuel rods modeled. Rod 1 is the hot rod in the hot assembly channel, Rod 2 is the hot assembly average rod, Rods 3 and 4 represent average assemblies in the center of the core and Rod 5 represents the lower power assemblies at the edge of the core. The safety injection (SI) system was assumed to be delivering to the RCS five seconds after the generation of a safety injection signal. This five second delay includes the time required for developing full flow from the SI pumps. No additional delay was required for diesel startup and sequencing since the analysis assumed reactor coolant pumps remain in operation in conjunction with no loss of offsite power. Sensitivity studies (Reference 2) show that this assumption results in the worst peak cladding temperature. Minimum safeguards ECCS capability and operability has also been assumed.

No additional penalties were required for upper plenum injection since this model properly models the location of the RHR flow. This analysis result is below the 2200^{0}F Acceptance Criteria limit established by Appendix K of 10CFR50.46 (Reference 1).

REFERENCES

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors: 10CFR50.46 and Appendix K of 10CFR50.46," Federal Register, Vol. 39, No. 3, January 4, 1974.

2. Dederer, S.I., et. al., Westinghouse Large-Break LOCA Best-Estimate Methodology, Volumes 1 and 2, WCAP-10924-P/A (Proprietary Version), December, 1988.

3. NRC Staff Report, "Emergency Core Cooling System Analysis Methods," USNRC-SECY-83-472, November, 1983.

4. Bordelon, F. M., and C. T. Murphy, Containment Pressure Analysis Code (COCO), WCAP-8327 (Proprietary Version,) WCAP-8326 (Non-Proprietary Version), June, 1974.

5. Westinghouse Revised Pad Code Thermal Safety Model, WCAP-8720, Addendum 2 (Proprietary), and WCAP-8785 (Non-Proprietary).

6. Nissley, M. E., et. al., Westinghouse Large Break LOCA Best-Estimate Methodology, Vol. 1, Add. 4, WCAP-10924-P, (Proprietary Version), August 1990

TABLE 1

LARGE BREAK

TIME SEQUENCE OF EVENTS

EVENT	Four Channel Core 0.4 DECLG (seconds)
Start	0.0
Reactor Trip Signal	0.1
Safety Injection (S.I.) Signal	2.0
High Head S.I. Begins	7.0
Accumulator Injection	9.0
Blowdown PCT Occurs	10.0
Low Head S.I. Begins	21.0
End of Bypass	26.2
Hot Rod Burst	27.8
Hot Assembly Average Rod Burst	34.7
Bottom of Core Recovery	35.5
Accumulator Water Empty	45.7
Accumulators Nitrogen Injection Ends	71.0
Reflood PCT Occurs	83.21

TABLE 2

LARGE BREAK RESULTS

EVENT	Four Channel Core 0.4 DECLG
Peak Cladding Temp., ^{o}F	2109.
Peak Clad Temp. Location, Ft.	7.375
Local Zr/Water Reaction (max), %	8.265
Local Zr/Water Reaction Location ft.	7.625
Total Zr/Water Reaction, %	< 0.3
Hot Rod Burst Time, Sec.	27.8
Hot Rod Burst Location, Ft.	7.75
Hot Assembly Burst Time, Sec.	34.7
Hot Assembly Burst Location, Ft.	7.75
Hot Assembly % Blockage	27.85
Calculation Input Values:	
NSSS Power, Mwt, 102% of	1650.
Peak Linear Power, kw/ft, 102% of	14.862
Peaking Factor	2.4
Accumulator Water Volume (Cubic Ft. Per Tank, Nominal)	1270.
Accumulator Pressure, psia	754.7
Number of Safety Injection Pumps (Operating [1 RHR + 2 HHSI])	3
Steam Generator Tubes Plugged	10%

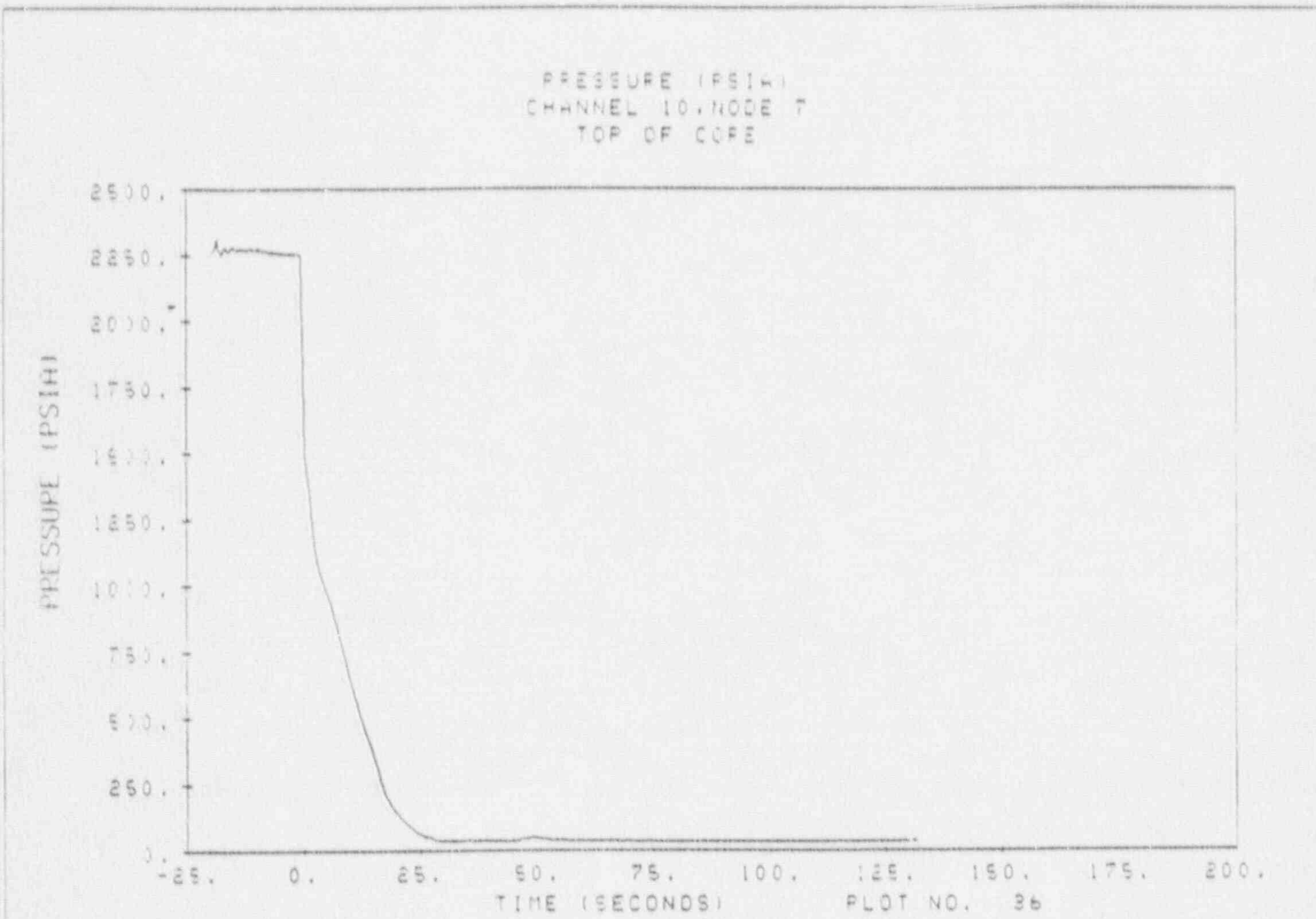

FIGURE 1. CORE PRESSURE DURING THE TRANSIENT

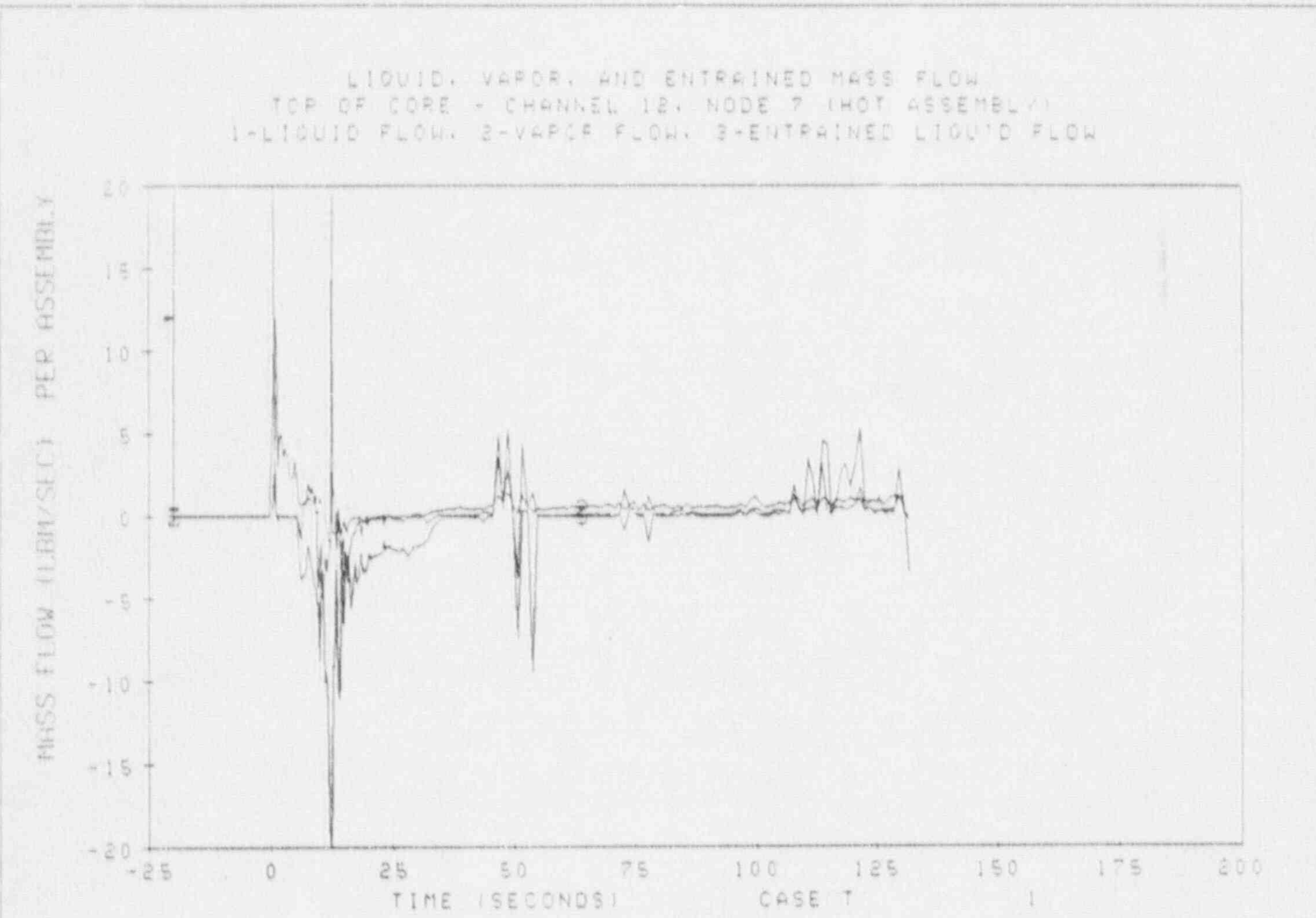

FIGURE 2. VAPOR, LIQUID, AND ENTRAINED MASS FLOW RATE AT THE TOP OF THE HOT ASSEMBLY

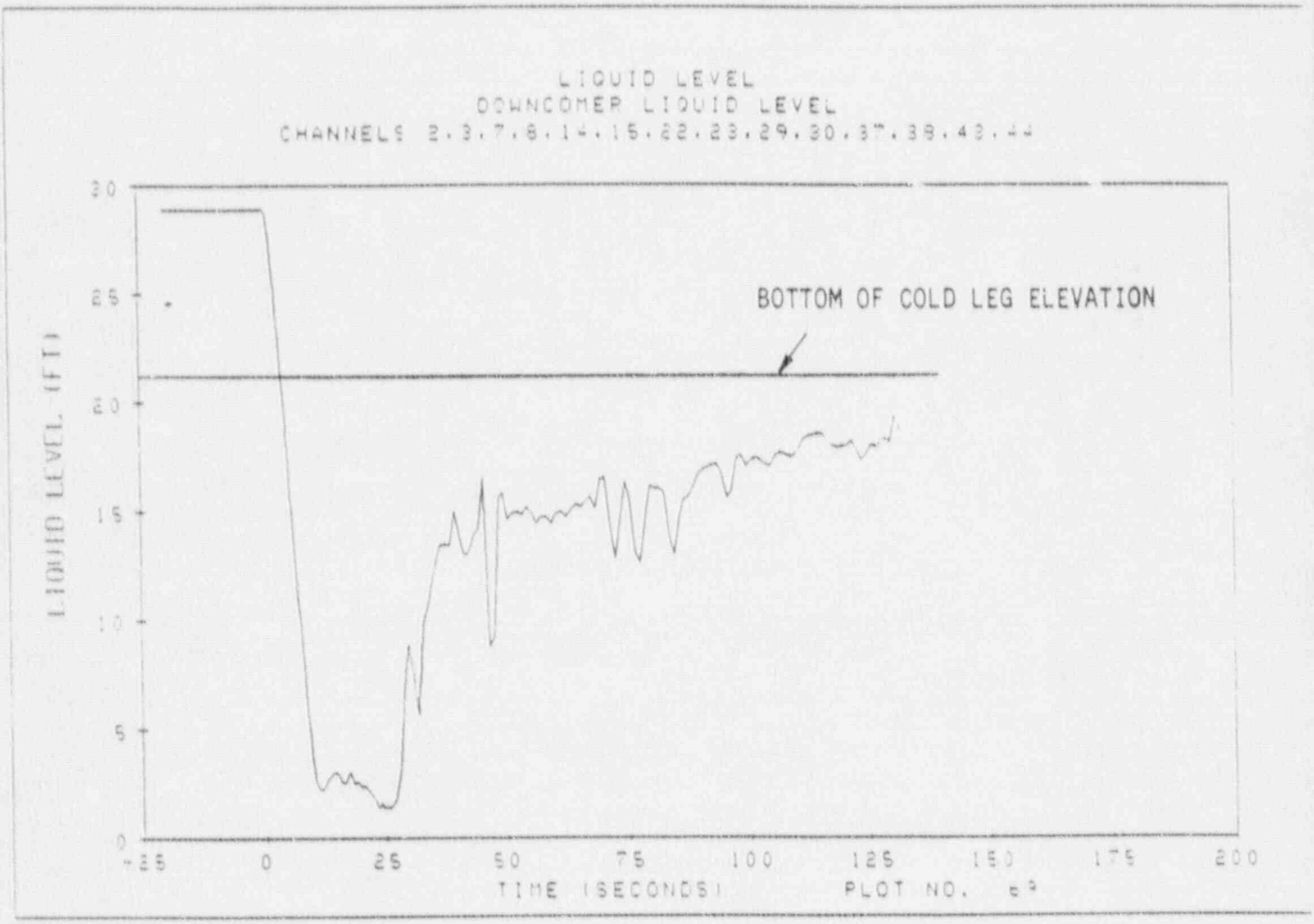

FIGURE 3. COLLAPSED LIQUID LEVEL IN DOWNCOMER

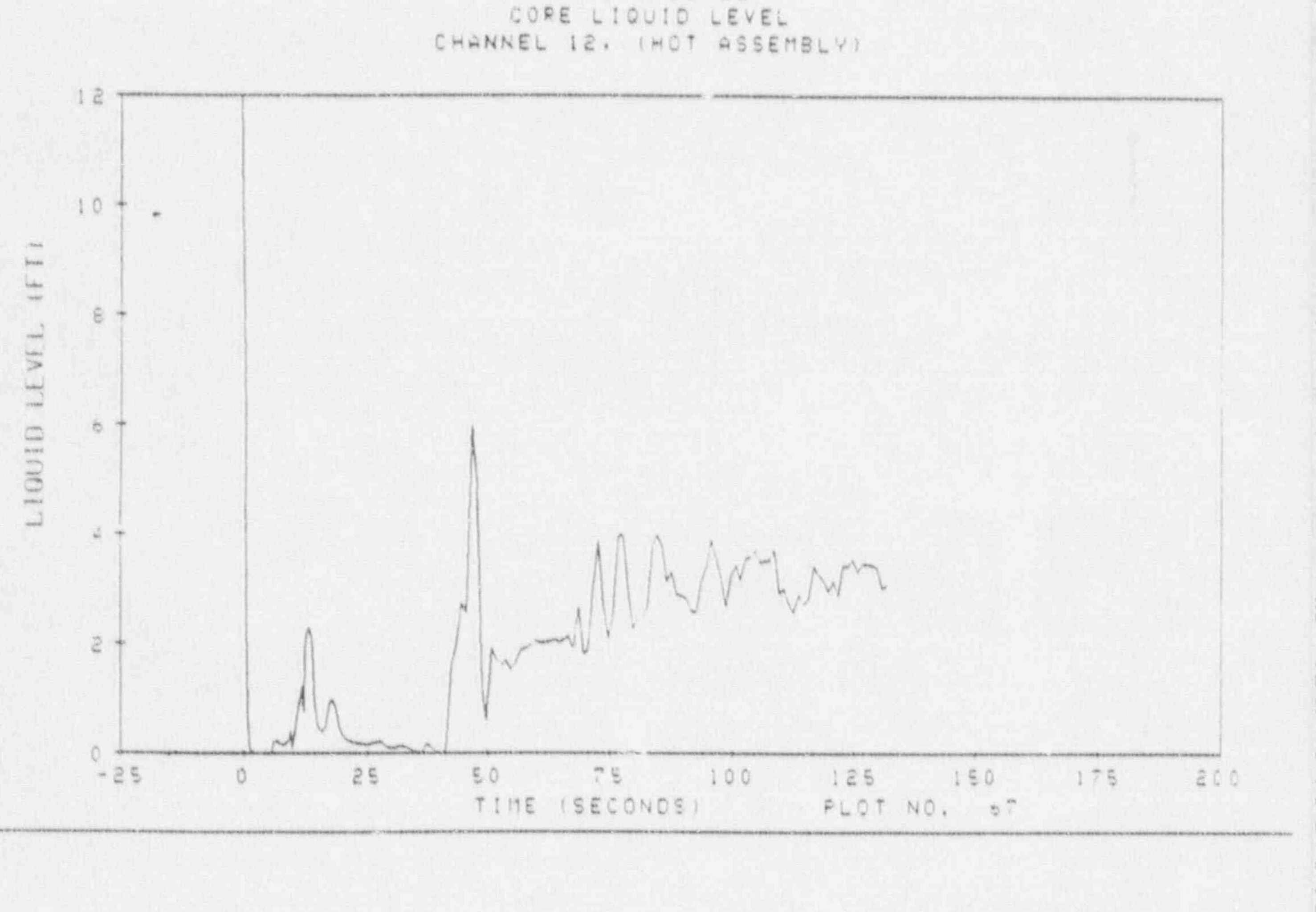

FIGURE 4. COLLAPSED LIQUID LEVEL IN HOT ASSEMBLY

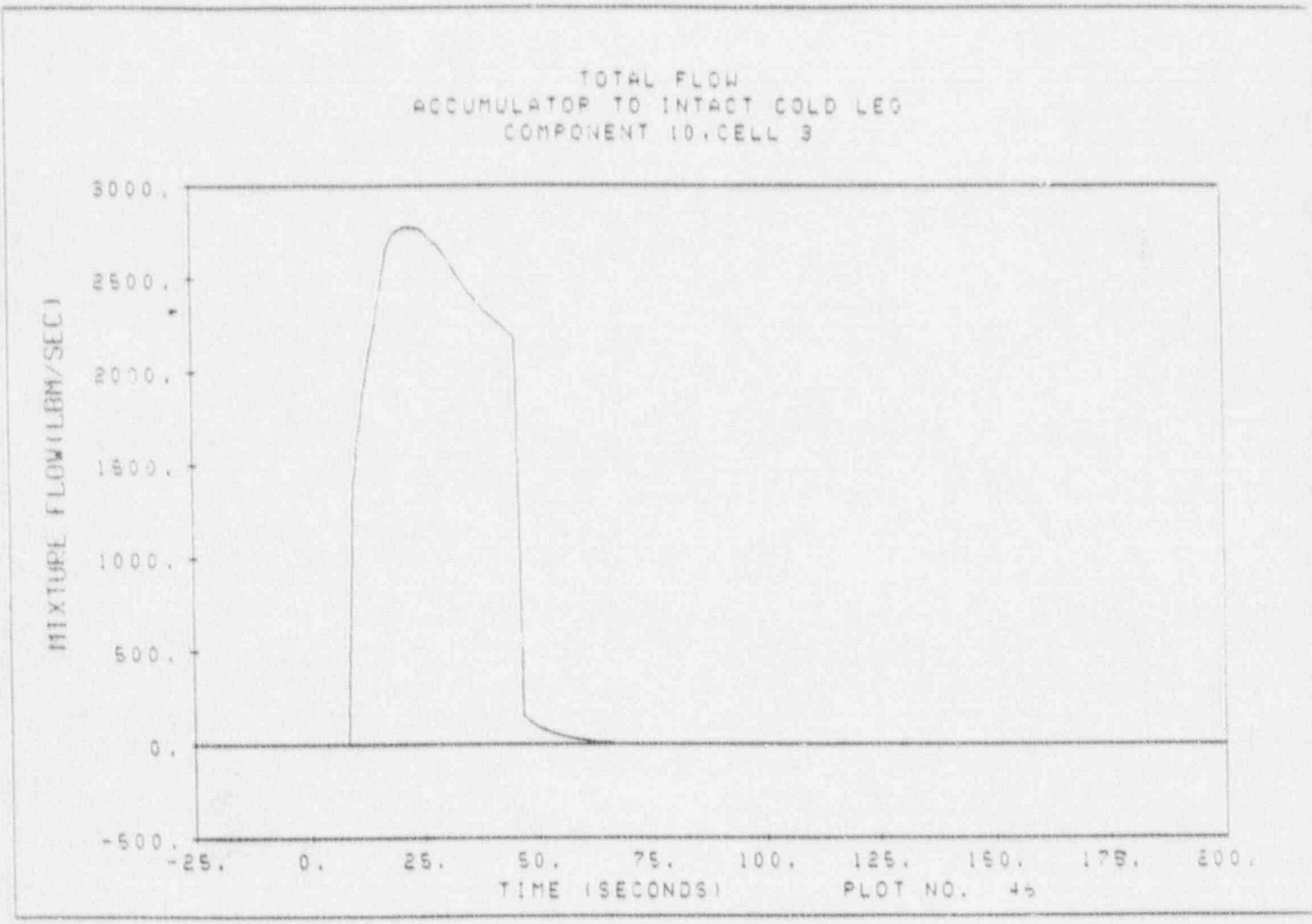

FIGURE 5. ACCUMULATOR FLOW

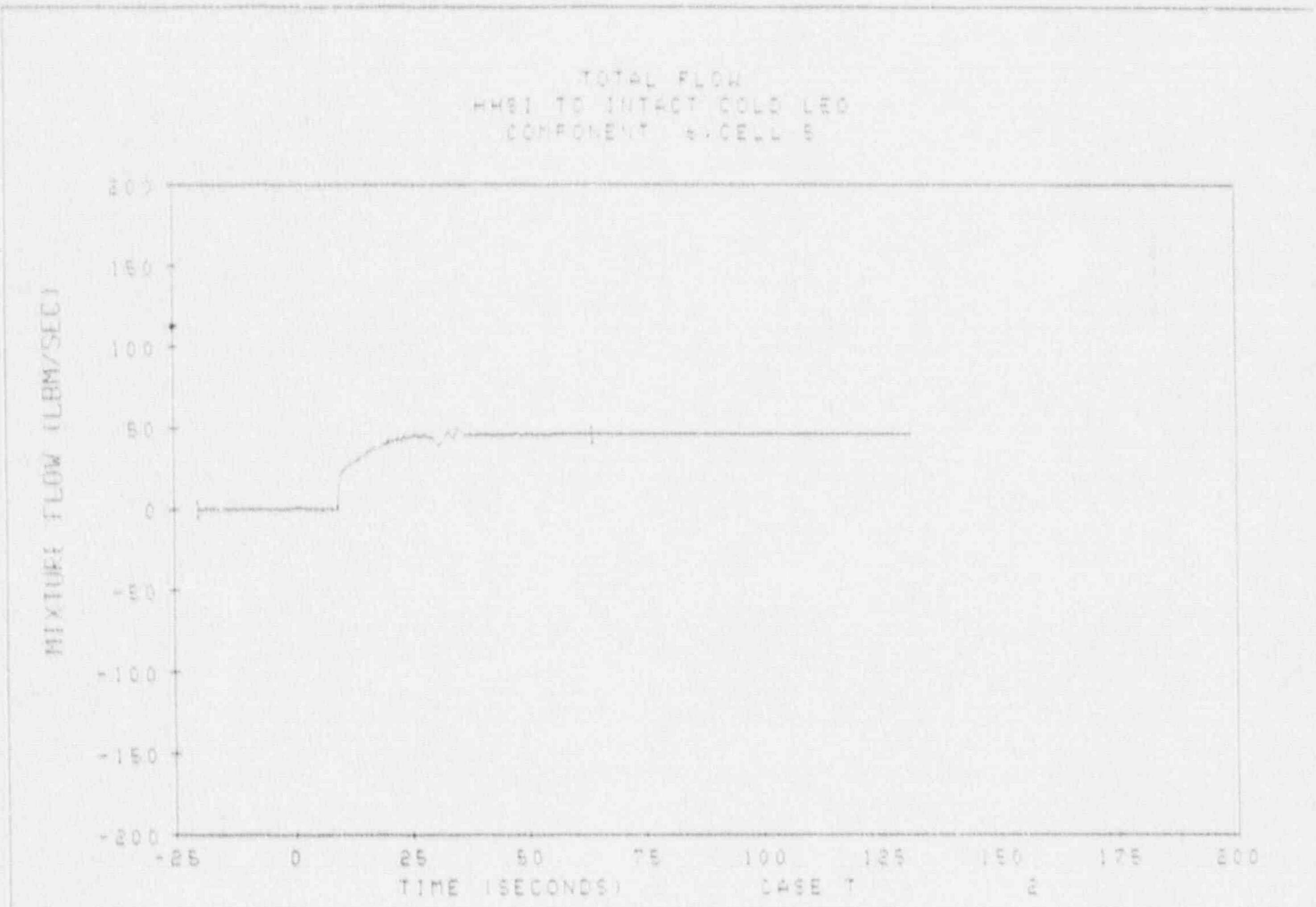

FIGURE 6. HIGH HEAD SI FLOW

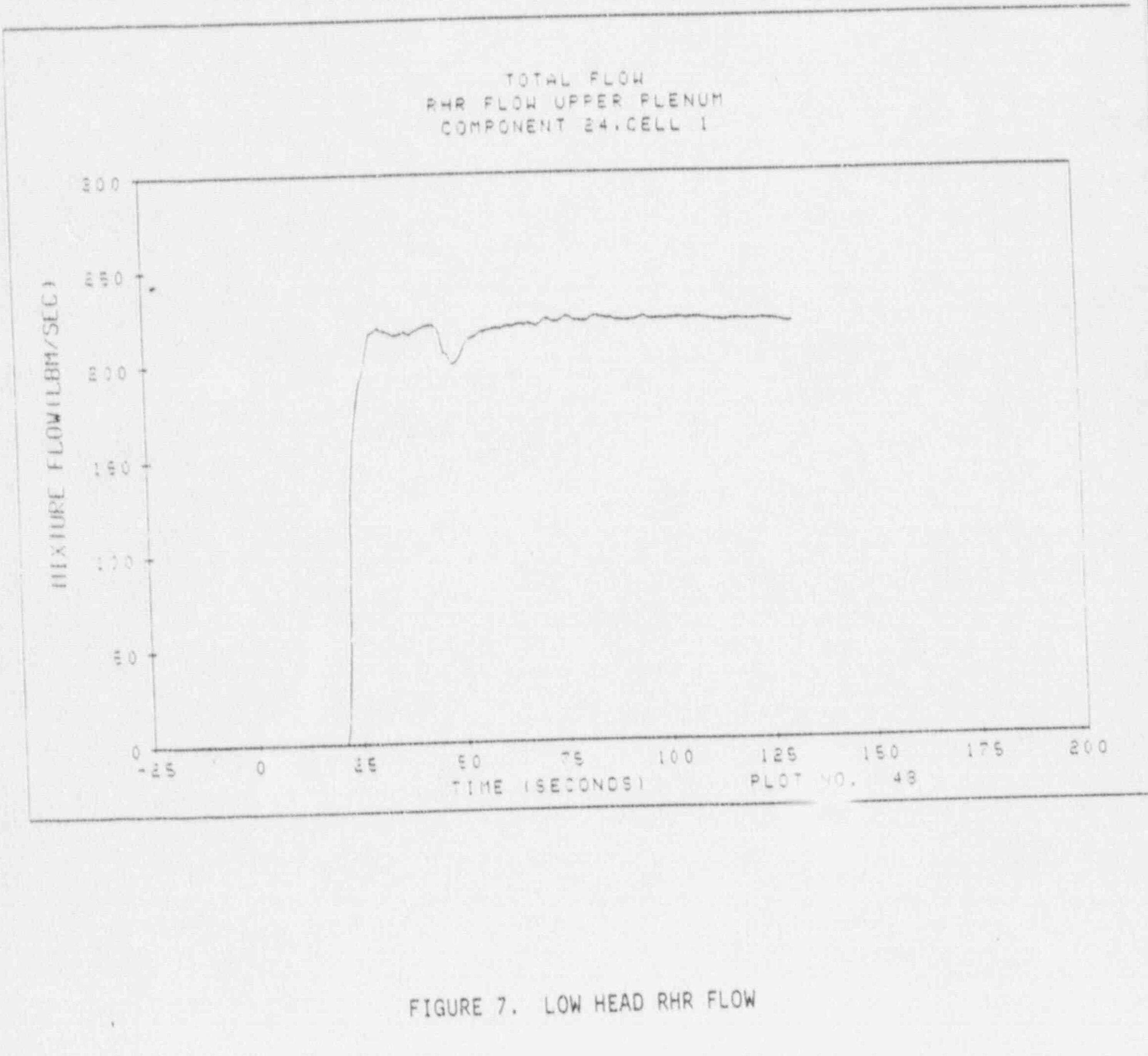

FIGURE 7. LOW HEAD RHR FLOW

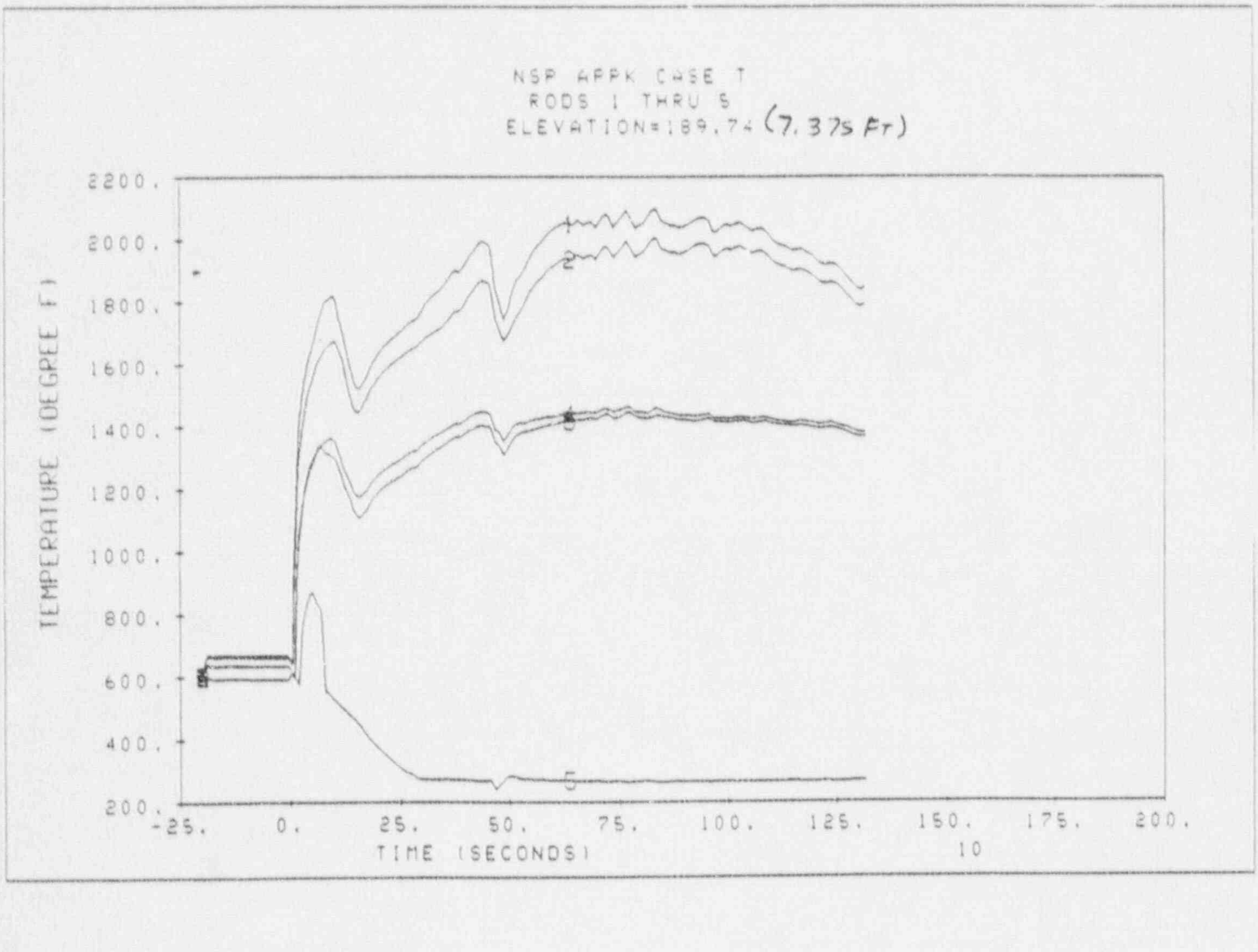

FIGURE 8. CLADDING TEMPERATURE AT PCT ELEVATION

Core Flow Difference Observed in
W̲COBRA/TRAC Calculations

W̲COBRA/TRAC calculations are initiated from a steady-state system condition which the code calculates to establish the vessel and loop flows as well as the fluid temperature distribution in the primary coolant system. There are a large number of parameters which must be specified to obtain a valid steady state such as reactor power, pump flows, steam generator tube plugging levels, fuel temperatures and internal fuel rod gas pressure, just to name a few. These parameters can vary slightly plant-to-plant. The fuel rod information is obtained from the PAD fuel rod code, while the reactor system hydraulic conditions are obtained from primary side and secondary system calculations which have been benchmarked to plant data as well as scaled hydraulic tests.

W̲COBRA/TRAC can yield slightly different values for the different parameters due to the different computational techniques and methods of modeling the reactor system between the reference plant calculations and W̲COBRA/TRAC modeling. It has been recognized that such differences can exist, hence acceptance criteria were established in WCAP-10924 to minimize these efforts such that consistent results would be obtained. The plant parameters which had to be matched were divided into two groups. The relative importance for large break LOCA of the parameters in each group were discussed in Chapter 3 of Volume 2 of WCAP-10924. The first level variables contain the reactor heat source parameters such as power, fuel temperature, coolant temperatures, peak kw/ft, and reactor pump delta-P and were held to a very tight tolerance. All power parameters were specifically made to exactly match the desired value or were made conservative. This criterion is given in Table 2-1 on page 2-32 of WCAP-10924 Volume 2, Revision 2. Most of the parameters had to be matched within +1%, -0%.

The secondary parameters included the reactor pressure drop values, core flow, and the ratios of pressure drops to ensure the proper hydraulic resistance distribution between the loops, and the reactor vessel. The pressure drop information generally has a higher uncertainty and is more difficult to match because of the uncertainty in the reference calculations as well as the WCOBRA/TRAC modelling. It should also be noted that the accuracy of the secondary parameters is sacrificed to obtain a more accurate fit to the first level parameter such as fluid temperatures. However, even the secondary level variables are required to be within ±5% as shown in Table 2-1 on page 2-32 of WCAP-10924, Volume 2, for a valid steady state calculation.

The difference between the original WCAP-10924, Volume 2, Revision " Prairie Island calculation and the revised calculation using the Addendum 4 to WCAP-10924 is two-fold. First there is some possible difference due to the decay power effects of correcting the decay heat error which is the basis for Addendum 4. It is expected that the decay heat effect is small since the time in question is very early in the transient and the integrated effect of the decay heat curve difference would be very small.

The second change between the two calculations is the method used to match the reference reactor system pressure drop and flow information during steady state. The revised calculation directly calculated the form loss pressure drop from the WCOBRA/TRAC code output to compare with the estimated unrecoverable pressure losses given in the reference reactor coolant system calculations. This is a more accurate method since the unrecoverable pressure losses are directly compared. The original calculation compared component pressure drops, not specific form losses, and the form losses were then adjusted to match the calculated pressure loss.

In the revised method, the velocity head effects are accounted for in a more accurate and systematic fashion. Both techniques yielded secondary level variables that were within the ±5% guidelines given in WCAP-10924, Volume 2, Revision 2, which had been reviewed and discussed with the NRC. The primary difference between the two calculations is the ratio of the lower plenum pressure drop to the vessel pressure drop. This ratio is slightly less (~1%) for the revised calculations which will result in slightly more downflow through the reactor vessel during blowdown. While this pressure drop adjustment, which is well within the allowable variation, may result in momentarily more core downflow, other such adjustments could have easily resulted in slightly reduced core flow. Since the reactor system pressure drops are difficult to exactly predict, the allowable tolerance of ±5% is reasonable and should result in only small changes run-to-run with a minimal effect on the final peak cladding temperature.