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Document Control Desk
U. S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Attention: Mr. Robert B. Samworth, Project Engineer
PWR Project Directorate III-3

Gentlemen:

DOCKETS 50-266 AND 50-301
ADDITIONAL INFORMATION ON ECCS LOCA ANALYSIS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On March 19, 1991 you telephoned Wisconsin Electric requesting additional information on the Emergency Core Cooling System (ECCS) large-break Loss of Coolant Accident (LOCA) analysis performed for Point Beach Nuclear Plant (PBNP). The additional information is needed to close out your review of recent changes to the analysis described in our letter dated March 5, 1991. This letter provides the information you requested.

Attachment 1 is a list of input changes from the analysis described in WCAP-10924, Volume 2, Revision 1, Addendum 2 dated December 1988. You requested that we identify any differences in the inputs for this analysis when compared to the previous analysis of record.

Attachment 2 is a justification for using sensitivity studies from previous analyses to determine limiting conditions for this analysis. You requested justification for assuming that the limiting break size and worst single failure are the same as in previous analyses.

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Attachment 3 is a copy of changes to section 14.3.2 of the PBNP Final Safety Analysis Report. Section 14.3.2 is the section titled "Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)." Changes in the text are hand-written on the copy and the figures are revised based on the new analysis.

Please contact us if you have any additional questions.

Very truly yours,

C. W. Fay
Vice President
Nuclear Power

Enclosures

Copy to: NRC Resident Inspector
NRC Regional Administrator

Attachment 1

WCOBRA/TRAC INPUT CHANGES*

Change	Reason for Change
Revised PAD 3.4 fuel data (higher avg. temps) with lower (275 psig) backfill pressure.	New RSAC (reload) data had more limiting temperatures.
Increased peak rod power to 14.54 kw/ft (from 14.47 kw/ft).	Correction to match current power level and peaking factors.
Corrected Neutron redistribution factors.	To match corrected code model (Addendum 4).
Added Gamma redistribution factors.	For new code version (Addendum 4).
Modeled ACCUM/SI interaction.	To match plant configuration.
Corrected pressure drop calcs to account for gravitational and velocity heads. Expanded area in cold leg cell next to vessel for transient deck to match the steady state deck.	To match current input methods.
Corrected broken loop piping to specify no heat transfer or friction in broken loop while using Moody flow model.	Deck correction.
Added core barrel wetted perimeter to channel 13.	To match current input methods.
Corrected loop elevations.	Deck correction.

* The changes described are relative to the original Point Beach analysis documented in WCAP-10924-P, Volume 2, Revision 1, Addendum 2.

Attachment 2

Justification of Sensitivity Study Results for Point Beach Reanalysis

The Point Beach reanalysis using the methodology in Addendum 4 of WCAP-10924, Volume 1, was performed for the worst break (0.4 DECL guillotine) with the worst single failure (loss of one low head SI pump), using the limiting axial power shape determined from the lead two-loop plant studies. The worst single failure, loss of a low head SI pump, has not changed from even the initial two-loop sensitivity performed with the pre-WCOBRA/TRAC methodology. The 0.4 DECL guillotine break is the most limiting break, as confirmed by the original sensitivity studies, and represents the lower bound uncertainty on the Moody critical flow model. The break spectrum sensitivity results shown in WCAP-10924 Volume 2, showed that the 0.4 DECL guillotine was clearly the limiting break.

The axial power shape sensitivity studies given in WCAP-10924, Volume 2, also showed that a center skewed axial power shape with the peak displaced upward was most limiting for two-loop plants. This limiting shape was used with the worst break and worst single failure for the Point Beach analysis.

The above assumptions were based upon sensitivity studies documented in WCAP-10924, Volume 2, Revision 2, and performed with three- and four-channel WCOBRA/TRAC UPI models. These sensitivity studies were used to address the impact and direction, increase or decrease, of the peak cladding temperature for each change studied. The model corrections and improvements described in Addendum 4 either would not affect these sensitivity studies (e.g., the decay heat correction has no impact since the sensitivity studies employed the 1979 decay heat model, while the correction affected only the 1971 decay heat model) or would not significantly change the relative PCT differences between the calculations for the studies performed. In other words, while the PCTs for these studies could change, the relative differences between the calculations should be preserved such that the original decisions on worst case assumptions should still apply. Chapter 5 of Addendum 4 provides a more detailed discussion of the effect of the model changes on the sensitivity studies.

Attachment 3

Changes to Final Safety Analysis Report

Section 14.3.2

14.3.2 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS-OF-COOLANT ACCIDENT)

The analysis specified by 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors"[1], is presented in this section. The results of the loss-of-coolant accident analysis are shown in Table 14.3.2-4 and show compliance with the acceptance criteria. The analytical techniques used are in compliance with Appendix K of 10 CFR 50 with an exemption to requirements I.D.3 and I.D.5 regarding liquid carryover fraction and refill/reflood heat transfer.

Should a major break occur, depressurization of the reactor coolant system results in void formation and pressure decrease in the pressurizer. Rapid voiding in the core shuts down reactor power. A safety injection system signal is actuated when the low pressurizer pressure setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid which transfers heat from the core by forced convection with some nucleate boiling. After the break develops the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50. After departure from nucleate boiling, the fuel rods are cooled by transition and film boiling processes. As the two-phase cooling changes to single phase steam flow, both turbulent and laminar forced convection are considered as core heat transfer mechanisms.

When the reactor coolant system pressure falls below the pressure in the accumulators, the accumulators begin to inject borated water. The conservative assumption is made that ECCS water injected into the cold leg bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10 CFR 50.

Thermal Analysis

Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss of coolant accident including the double ended severance of the largest reactor cooling system cold leg pipe. The reactor core and internals together with the emergency core cooling system are designed so that the reactor can be safely shut-down and the essential heat transfer geometry of the core preserved following the accident. Long-term coolability is maintained.

The emergency core cooling system, even when operating during the injection mode with the most severe single failure, is designed to meet the acceptance criteria. The most severe single failure is the loss of one RHR pump. Loss of one diesel generator is a less severe single failure since such a failure results in the loss of one containment spray pump which increases containment pressure. Higher containment pressure increases the rate of core reflood thereby reducing PCT. Loss of one RHR pump results in a higher PCT.[3]

Method of Thermal Analysis

The analysis was performed using the Westinghouse Large Break LOCA Best-Estimate Methodology.[2,3] The Westinghouse Best-Estimate Methodology was developed consistent with guidelines set forth in the SECY-83-472 document.[4] 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 References 2 and 3.

The SECY-83-472 document states that there are three areas of conservatism in the current licensing models: the required Appendix K conservatism, the conservatism added by both the NRC staff and industry to cover uncertainties, and the conservatism imposed by the industry in some cases to reduce the complexity of the analysis. Based on a review of the available experimental data and the best estimate computer code calculations, the NRC staff concluded that there is more than sufficient safety margin to assure adequate performance of the ECCS, and that this excess margin can be reduced without an adverse effect on plant safety. Therefore, in the SECY-83-472 approach, the NRC staff suggests that the licensee utilize a realistic model of the PWR to perform three calculations.

The first two calculations predict the plant response to a LOCA at the most realistic or most probable level (50 percent probability) and at a more conservative 95 percent probability level. The calculation at the 95 percent probability level accounts for uncertainties in such things as power level, fuel initial temperature, nuclear parameters, and computer code uncertainties. The parameters to be examined, and the methods for combining uncertainties (either statistically or as a one-sided bias) need to be justified. The realistic PWR model and the uncertainty analysis can be performed on a generic PWR model which is representative of a class of similar plants, that is, two-, three-, or four-loop PWRs so that generic uncertainties are applicable to the individual plants.

The third calculation suggested in SECY-83-472 is to use the realistic model augmented only with the required features of Appendix K. Required Appendix K features are 1971 ANS decay heat plus 20 percent, Moody break flow model, no return to nucleate boiling during blowdown, and so forth. The Appendix K calculation is acceptable if the peak cladding temperature is greater than the peak cladding temperature calculated at the 95 percent probability level but below the licensing limit of 2200°F. In order to comply with the Appendix K requirements, the code was prevented from returning to nucleate boiling after CHF during blowdown even if the fluid and rod surface conditions would permit this to occur.

The SECY-83-472 interpretation of these results is that the required features of Appendix K have sufficient margin to cover all uncertainties inherent in a LOCA analysis combined at a 95 perent probability level. Such a series of calculations provides an acceptable licensing basis for the NRC and is expected to result in peak cladding temperature margin which the licensees can use for improved operational flexibility, low leakage loading patterns to address PTS concerns, and to accommodate more economical fuel designs.

The Best Estimate Methodology is comprised of the WCOBRA/TRAC and COCO computer codes[3,5]. 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 parameters used in the containment analysis to determine this pressure curve are presented in Table 14.3.2-1.

In determining the conservative direction for bounding values and assumptions of plant parameters, many sensitivity studies were performed, as documented in Reference 3. These sensitivities were performed using the Prairie Island Nuclear Power Plant model. Since Prairie Island has a higher peak linear heat rate and a higher core power to ECCS flow ratio than Point Beach, it will yield a greater change in peak cladding temperature for changes in plant parameters. These sensitivity studies were used to determine the direction of conservatism for choosing the bounding conditions for the 95 percentile calculation for Point Beach.

Studies were also performed to determine the limiting break type and the limiting break size. Figure 14.3.2-1 shows the results of these calculations, performed using the Prairie Island Nuclear Plant model. Cold Leg Split (CLS) breaks are defined as breaks formed in the wall of the cold leg such that flow must make a right angle turn to exit the break, such as would be formed if a pipe had a longitudinal crack. The Double-Ended Cold Leg Guillotine (DECLG) break is defined as the full severence of a

cold leg such that water spills out from both sides of break, with various discharge coefficients modeled to account for possible variations in flow. These results show a clear trend that a DECLG break with a discharge coefficient of 0.4 will be limiting for the Appendix K calculation. Double-Ended Hot Leg Guideline (DEHLG) break results are not shown in Figure 14.3.2-1. Calculations described in Reference 3 show that DEHLG breaks are not limiting. These results justify performing only a 0.4 DECLG break for the Point Beach Appendix K calculation, results of which will be discussed later.

For the Appendix K calculation, exemptions from items I.D.3 and I.D.5 of Appendix K to 10 CFR Part 50[1] were requested in a letter dated November 30, 1988.[9] These exemptions were necessary because Item I.D.3, which requires the use of a carryover fraction to calculate the reflood core exit fluid flow, and Item I.D.5, which sets specific requirements for refill and reflood heat transfer calculation, were intended for conventional cold-leg injection plants and are not applicable to the UPI plants. This exemption has been granted.[7]

The PBNP analysis also models the removal of the thimble plugs with the corresponding increase in core bypass flow. Other input specifications for the LOCA analysis are delineated in Table 14.3.2-2. These parameters were chosen at their limiting values in order to provide a conservative estimate of the capability of this plant to recover from a large break LOCA analysis. If the direction of conservatism was unknown, the limiting value was determined by sensitivity studies documented in Reference 3. Results of sensitivity studies are similar to previous Evaluation Model results with the exception of having the reactor coolant pump running. A sensitivity study shows that having the reactor coolant pumps running increases peak cladding temperature.

The fuel parameters used as input for the PBNP LOCA analysis were generated using the Westinghouse fuel performance code (revised PAD 3.~~2~~4, Reference 6). The fuel parameters input to the code were at beginning-of-life (maximum densification) values.

The PBNP Appendix K analysis was performed at a system operating pressure of 2250 psia using the four channel core model developed in Reference 3 for the 0.4 DECLG break. The 0.4 DECLG break was shown to be limiting by calculations performed with the three channel core model in Reference 3. A sensitivity study on system pressures of 2250 and 2100 psia was performed for a 0.4 DECLG and showed that the normal operating pressure of 2250 psia yielded the highest peak cladding temperature. The hot assembly was located under an open hole in the upper core plate, which was shown in sensitivity studies to be the limiting location for peak cladding temperature.[3] These transients were considered to be terminated if the hot rod cladding temperature began to decline and the injected ECCS flows exceeded the break flow.

Results

Results of the Appendix K 0.4 DECLG break for Point Beach are described in this section. Results of the 50 percent probability calculation, the 95 percent probability calculation and sensitivity studies performed for the Point Beach Nuclear Plant can be found in Addendum 2 to Reference 3 entitled PBNP Plant Specific Analysis. Sensitivity studies done to determine the conservative direction for bounding values and assumptions for plant parameters are described in Reference 3. Table 14.3.2-3 shows the time sequence of events for the Appendix K Large Break LOCA transient. Table 14.3.2-4 provides a brief summary of the important results of the LOCA analysis and shows compliance with the 10 CFR 50, Appendix K requirements. Table 14.3.2-5 shows the mass and energy release to containment from the broken loop accumulator. Figures 14.3.2-2 through 14.3.2-14 show important transient results for the limiting 0.4 DECLG break (2250 psia case). Note on these figures that the break occurs at time 0.0. Figure 14.3.2-2 shows the core pressure during the transient. Figure 14.3.2-3 shows the vapor and liquid mass flowrate at the top of the hot assembly. Figures 14.3.2-4 and 14.3.2-5 show the collapsed liquid level in the downcomer and core hot assembly channel, respectively, indicating the refilling of the vessel. Figures 14.3.2-6 and 14.3.2-7 show the flow of the ECCS water into the cold leg

(accumulator and high head safety injection flow) with Figure 14.3.2-8 showing the flow of low head safety injection into the upper plenum (UPI flow). Figure 14.3.2-9 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 at a value of 0.6 times the core average power. Figures 14.3.2-10 and 14.3.2-11 show the core power history for the blowdown period and for the complete transient. Figure 14.3.2-12 shows the containment pressure response as calculated by the COCO computer program. Figure 14.3.2-13 shows the mixture flow rate out of the vessel side of the broken cold leg. Positive flow is out of the vessel. Figure 14.3.2-14 shows the mixture flow rate, in LBM/SEC, out of the loop side of the broken cold leg. Positive flow is out of the cold leg. No flow is shown prior to time 0.0 seconds because the component being plotted is added at time 0.0 seconds to initiate the break.

The safety injection (SI) system delivers to the RCS 5.~~4~~0 seconds after generation of a SI signal or 7.~~5~~4 seconds after the break. Injection begins 7.~~5~~4 seconds after the break occurs because pressure in the RCS falls below the effective shutoff head of the SI system at that time. The effective shutoff head of the SI system is much lower than the shutoff head of an SI pump due to the system configuration assumed in the analysis. Two SI pumps are operating, but one of the two injection lines is dumping to containment. Prior to 7.~~5~~4 seconds, the pressure in the RCS prevents injection. SI pumps are capable of providing full flow within 5 seconds after the generation of a SI signal. The delay is the time required for the pumps to develop full flow. No delay is required for diesel startup because the analysis assumes that reactor coolant pumps remain in operation in conjunction with no loss of offsite power. Sensitivity studies[3] show that continued operation of the reactor coolant pumps results in the worst peak cladding temperature.

The upper plenum injection system (RHR) delivers to the RCS ~~12.9~~ 17.7 seconds after generation of a SI signal or ~~15~~ 20.1 seconds after the break. Injection begins ~~15~~ 20.1 seconds after the break occurs because pressure in the RCS falls below the effective shutoff head of the RHR system at that time. Prior to ~~15~~ 20.1 seconds, the pressure in the RCS prevents injection. RHR pumps are capable of providing full flow within 10 seconds after a SI signal. Five seconds are needed for sequencing of loads on the 4160V bus and five seconds are needed for the pump to develop full flow after starting. Minimum safeguards capability and operability has also been assumed.

The CD=0.4 break proved to be the limiting (highest predicted peak cladding temperature, PCT) case with a PCT of 202~~3~~8°F for the four channel core model with 2250 psia system pressure. Previous sensitivity studies indicated that a 0.6 or 0.8 DECLG case would yield even lower temperatures.[3]

The cladding temperature analysis is based on a total peaking factor of 2.50. The hot spot metal-water reaction reached is ~~13.24%~~ 4.85%, which is below the embrittlement limit of 17%, as required by 10 CFR 50.46. In addition, the total core metal-water reaction is less than ~~.0001%~~ .007% as compared with the 1% criterion of 10 CFR 50.46.

Conclusions

For breaks up to and including the double ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46. These criteria are as follows:

1. The calculated peak fuel element cladding temperature is below the requirements of 2,200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

The analysis techniques used for this evaluation allow for the Point Beach Units to operate at an RCS pressure of either 2250 or 2~~0~~100 psia. A sensitivity study on RCS pressure demonstrates the 2250 psia case to be limiting, and therefore, the results shown here assume an RCS pressure of 2250 psia.

No additional penalties are required for upper plenum injection since the Westinghouse Large Break LOCA Best-Estimate Methodology models the RHR flow to be injected into the upper plenum. Interim ECCS Evaluation Models did not consider the effect of upper plenum injection and required addition of a penalty on peak cladding temperature. This analysis result is below the 2200°F Acceptance Criteria limit established by Appendix K of 10 CFR 50.56.[9]

In keeping with the SECY-83-472 approach, additional large break LOCA analyses were performed for most probable (50 percent probability - also called nominal) level and the 95 percent probability level (known as a "superbounded" calculation). The nominal calculation had a peak cladding temperature of 1382°F. The superbounded calculation resulted in a peak cladding temperature of 193~~3~~8°F while the Appendix K calculation had a PCT of 202~~3~~8°F. These results clearly meet the SECY-83-472 requirement that the Appendix K calculation have sufficient margin to cover all uncertainties inherent in a LOCA analysis at a 95 percent probability level.

REFERENCES - Section 14.3.2

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

2. Hochreiter, L. E., Schwarz, W. R., Takeuchi, K., Tsai, C. K., and Young, M. Y., Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation, WCAP-10924-P-A, Volume 1, Revision 1, (Proprietary Version), December 1988.

3. Dederer, S. I., Hochreiter, L. E., Schwarz, W. R., Stucker, D. L., Tsai, C. K., and Young, M. Y., Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection, WCAP-10924-P-A, Volume 2, Revision 2, December 1988.

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

5. Bordelon, F.M., and Murphy E.T., Containment Presure Analysis Code (COCO), WCAP-8327 (Proprietary Version), WCAP-8326 (Non-Proprietary Version), June 1974.

6. Westinghouse revised PAD Code Thermal Safety Model, WCAP-8720, Addendum 2 (Proprietary), and WCAP-8785 (Non-Proprietary).

7. Enclosure to Letter from W. Swenson (NRC) to C. W. Fay (WEPCO), dated March 8, 1989, (54 FR 11095).

8. "Safety Evaluation Report on Interim ECCS Evaluation Model For Westinghouse Two-Loop Plants", March 1988.

9. Letter from C. W. Fay (WEPCo) to USNRC, "Dockets 50-266 and 50-301, Large-Break Loss of Coolant Accident Analysis For Technical Specification Change Request 127, Increased Allowable Core Power Peaking Factors, Point Beach Nuclear Plant, Units 1 and 2", VPNPD-88-581, NRC-88-119, November 30, 1988.

~~Ref.~~ Insert

Weiner, R., et al, "~~Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluation~~," WCAP-10851-P-A, August, 1988.

Weiner

TABLE 14.3.2-1

LARGE BREAK
CONTAINMENT DATA
(DRY CONTAINMENT)

NET FREE VOLUME	1.065×10^6 ft^3
INITIAL CONDITIONS	
Pressure	14.7 psia
Temperature	90°F
RWST Temperature	34°F
Service Water Temperature	33°F
Outside Temperature	-25°F
SPRAY SYSTEM	
Number of Pumps Operating	2
Runout Flow Rate	1950 gpm each
Actuation Time	10 secs
SAFEGUARDS FAN COOLERS	
Number of Fan Coolers Operating	4
Fastest Post Accident Initiation of Fan Coolers	35 secs

TABLE 14.3.2-1 (Continued)

PAINTED STRUCTURAL HEAT SINK DATA

Structural Heat Sink Surface Area (ft^2)	Structural Heat Sink Thickness (in)	Paint Thickness (mils)
56020	0.322	7.5
2480	0.25	7.5
103724	0.094	7.5
11710	0.304	7.5
4730	0.443	7.5
5441	0.584	7.5
4490	0.712	7.5
957	1.0	7.5
3667	2.634	6.0
10221	0.125	6.0
16551	0.209	6.0
2707	0.5	6.0

TABLE 14.3.2-2

(page 1 of 2)

INPUT SPECIFICATIONS FOR THE POINT BEACH
APPENDIX K LARGE BREAK LOCA ANALYSIS

PARAMETER	ANALYSIS VALUE
Plant Internals	Flat Upper Support Plate
Barrel Baffle Design	Upflow
Core Bypass Flow	6.5%
NSSS Power, 102% of (MWT)	1518.5
System Pressure (psia)	2250.
Primary System Fluid Temperatures	
T hot (°F)	~~549.0~~ 609.0
T cold (°F)	~~609.0~~ 549.0
T upper head (°F)	602.0
Fuel Type	14 x 14 OFA with axial blankets
Fuel Stored Energy	Beginning of Life
Fuel Data Source	~~Revised~~ Pad 3.~~3~~ 4
Fuel Rod Backfill Pressure (psig)	275.
FQ_T	2.50
F_{deltaH}	1.70
Peak Linear Power, kw/ft	~~14.47~~ 14.54
Maximum Average Power in the Outer Core Channel (24 assemblies)	0.6
Loop Flowrate (GPM)	89000
Reactor Coolant Pumps	Running
Steam Generator Tube Plugging (Symmetric)	25%
Steam Generator Isolation	No Steam or Feedwater Flow

TABLE 14.3.2-2
(page 2 of 2)

INPUT SPECIFICATIONS FOR THE POINT BEACH
APPENDIX K LARGE BREAK LOCA ANALYSIS

PARAMETER	ANALYSIS VALUE
Steam Generator Secondary Pressure (psia)	775.53
Accumulator Conditions	
Water Volume (cu. ft.)	1100.
Nitrogen Pressure (psig)	700.
Water Temperature (°F)	90.
Safety Injection Conditions	
Pumps in Operation	1 RHR + 2 HHSI
Pump Flow	Degraded
Water Temperature (°F)	60.
Delay Time (seconds) (no loss of offsite power)	5.0
Containment Pressure	Standard LOCA COCO curve

TABLE 14.3.2-3

LARGE BREAK

TIME SEQUENCE OF EVENTS FOR A 0.4 DECLG BREAK

	<u>Time</u> (seconds)
Start	0.0
Reactor Trip Signal	~0.1
S.I. Signal	~~2.1~~ 2.4
High Head Safety Injection Begins	~~7.5~~ 7.4
Accumulator Injection Begins	~~8.2~~ 8.3
Blowdown Peak Cladding Temperature Occurs	~~8.5~~ 9.0
Low Head Safety Injection Begins	~~15.0~~ 20.1
End of Bypass	~~18.1~~ 18.8
Hot Rod Burst	~~28.4~~ 26.3
Bottom of Core Recovery	~~32.3~~ 33.1
Hot Assembly Average Rod Burst	~~35.3~~ 33.4
Accumulator Water Empty	~~63.5~~ 67.5
System Mass Inventory Equibrates	~~83.0~~ 80.0
Accumulator N2 Injection Ends	~~86.0~~ 87.5
Reflood Peak cladding Temperature Occurs	~~104.5~~ 136.1

TABLE 14.3.2-4

LARGE BREAK

Results	DECLG (C_D=0.4)
Peak Cladding Temp., °F	2023. 8
Peak Cladding Temp. Location, Ft.	~~6.25~~ 8.375
Local Zr/H_2O Rxn (max), %	~~13.24~~ 4.85
Local Zr/H_2O Location, Ft.	~~6.25~~ 7.875
Total Zr/H_2O Rxn, %	<0.3
Hot Rod Burst Time, sec	~~28.4~~ 26.3
Hot Rod Burst Location, Ft.	~~6.25~~ 8.0
Hot Assembly Burst Time, sec	~~35.3~~ 33.4
Hot Assembly Burst Location, Ft.	~~6.25~~ 8.0
Hot Assembly % Blockage	~~54.2~~ 41.8

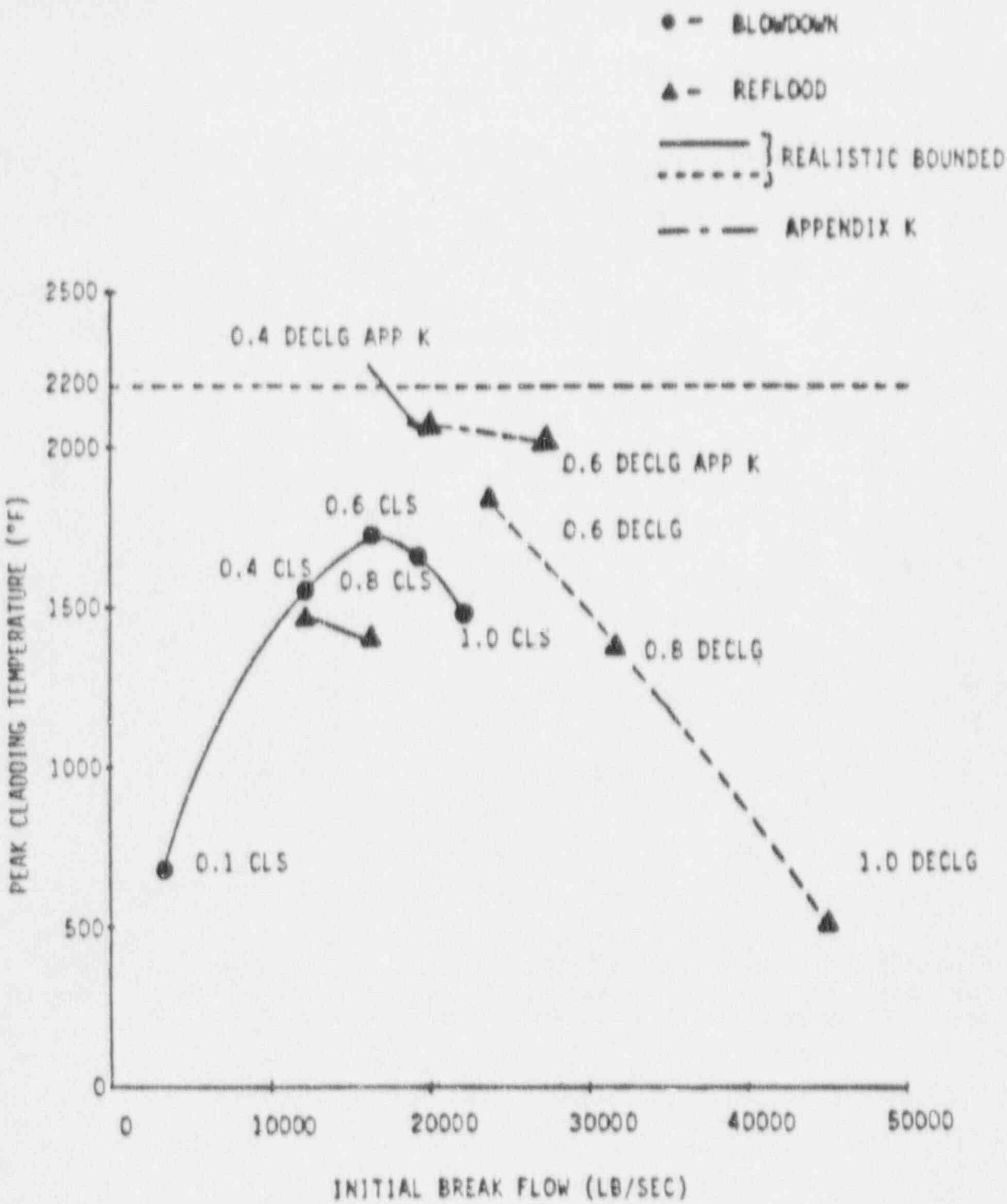

Figure 14.3.2-1. Comparison of Appendix K and Bounded Break Spectrums

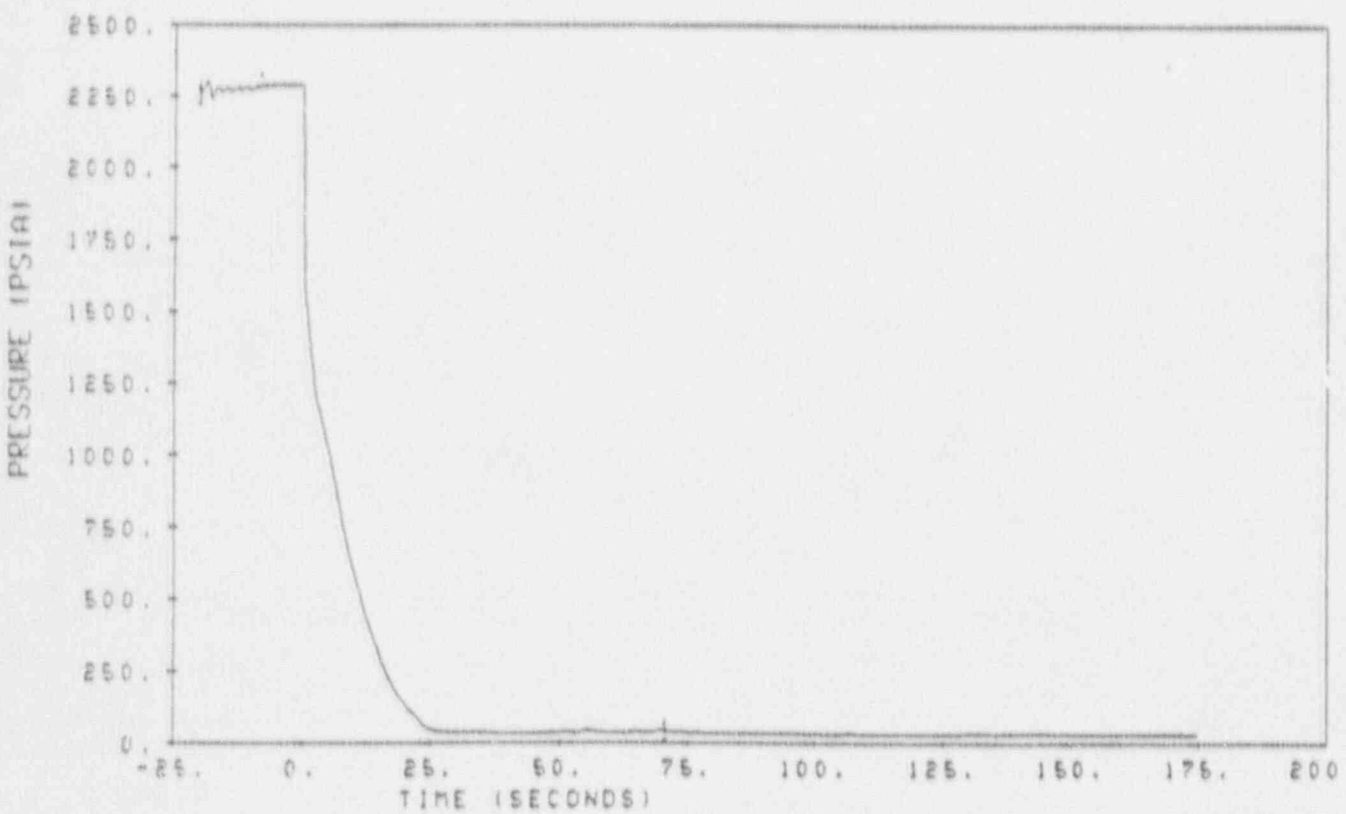

Figure 14.3.2-2. Core Pressure History (2250 psia case) (0.4 DECLG)

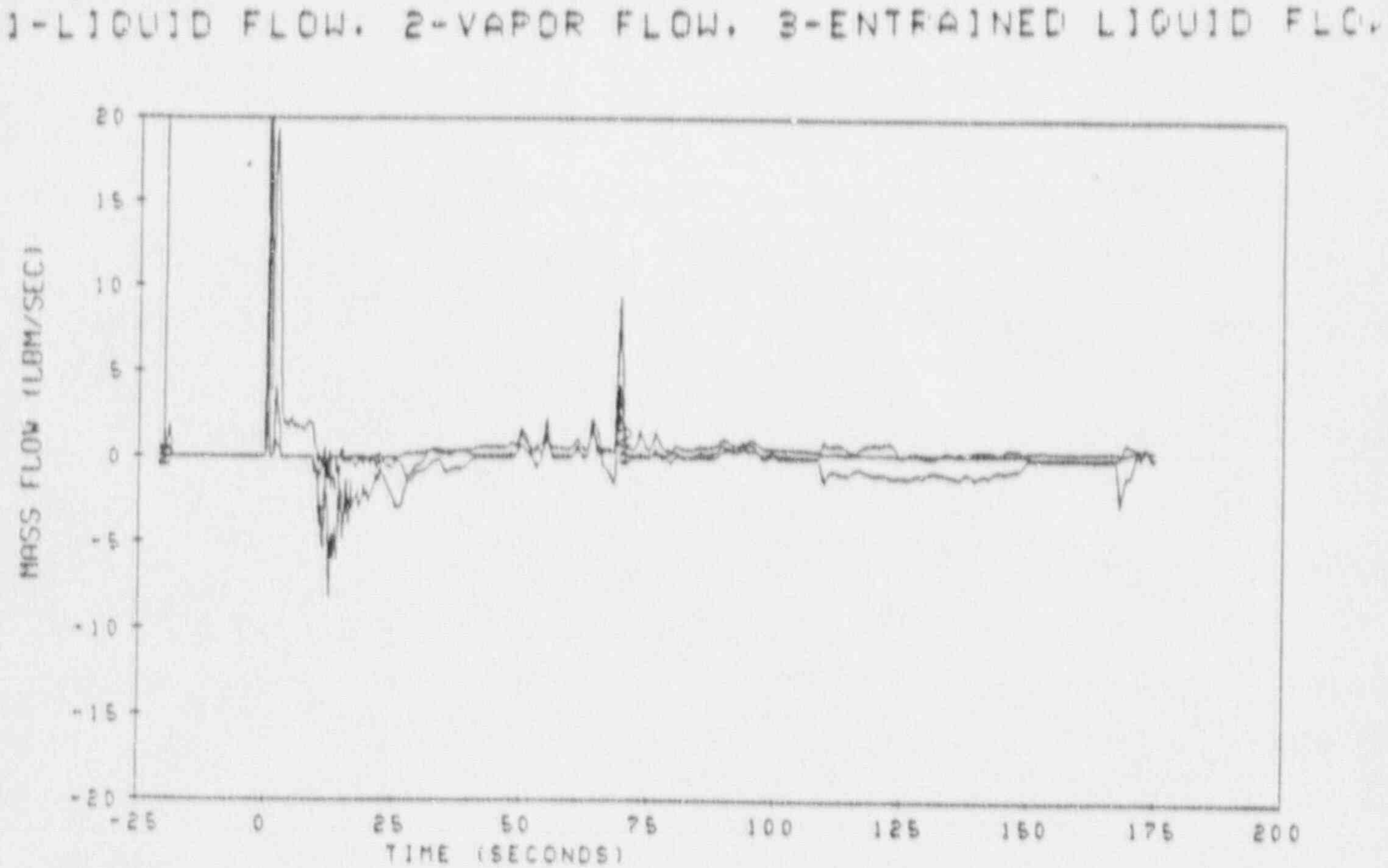

Figure 14.3.2-3. Core Flow Rate at Top of the Hot Assembly
(2250 psia case) (0.4 DECLG)

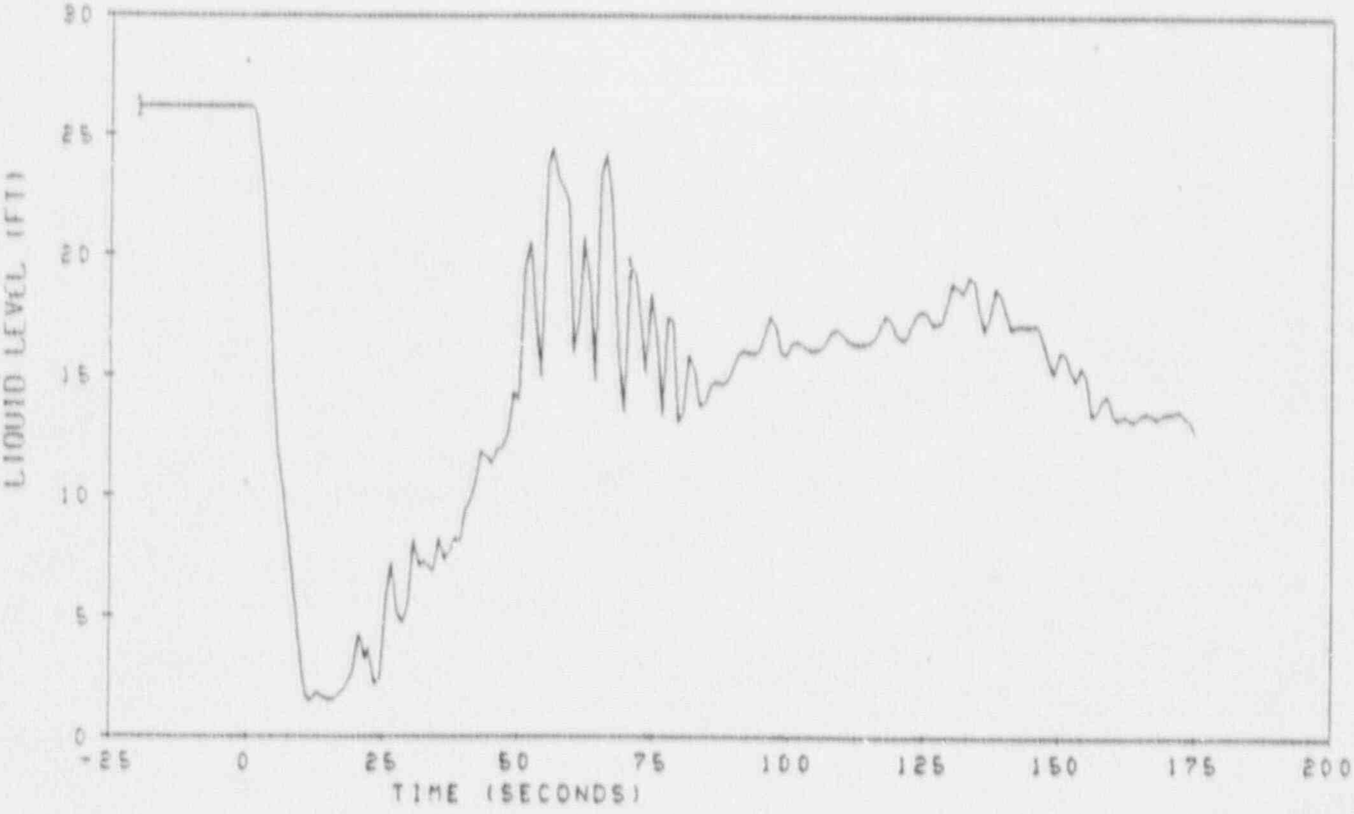

Figure 14.3.2-4. Collapsed Liquid Level in the Downcomer
(2250 psia case) (0.4 DECLG)

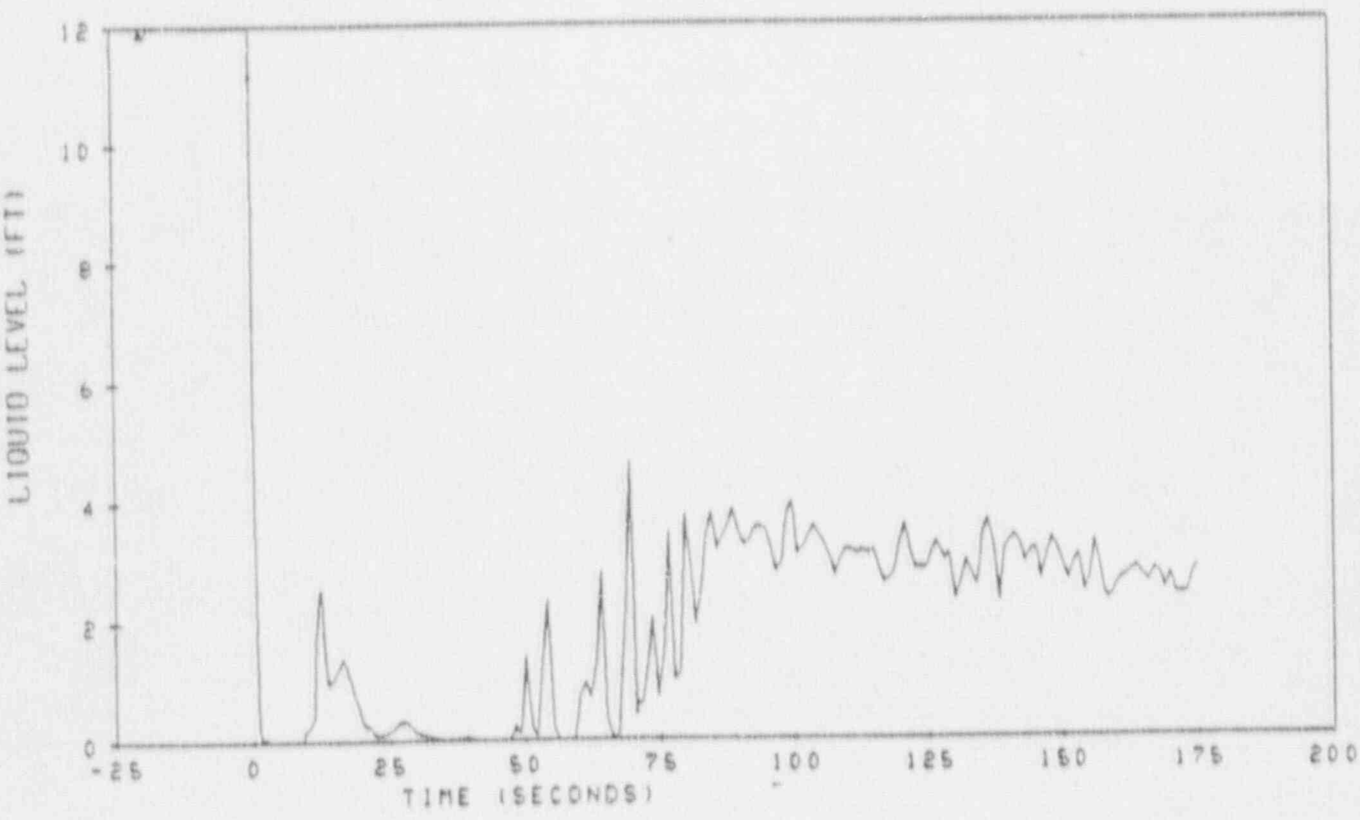

Figure 14.3.2-5. Collapsed Liquid Level in the Hot Assembly (2250 psia case) (0.4 DECLG)

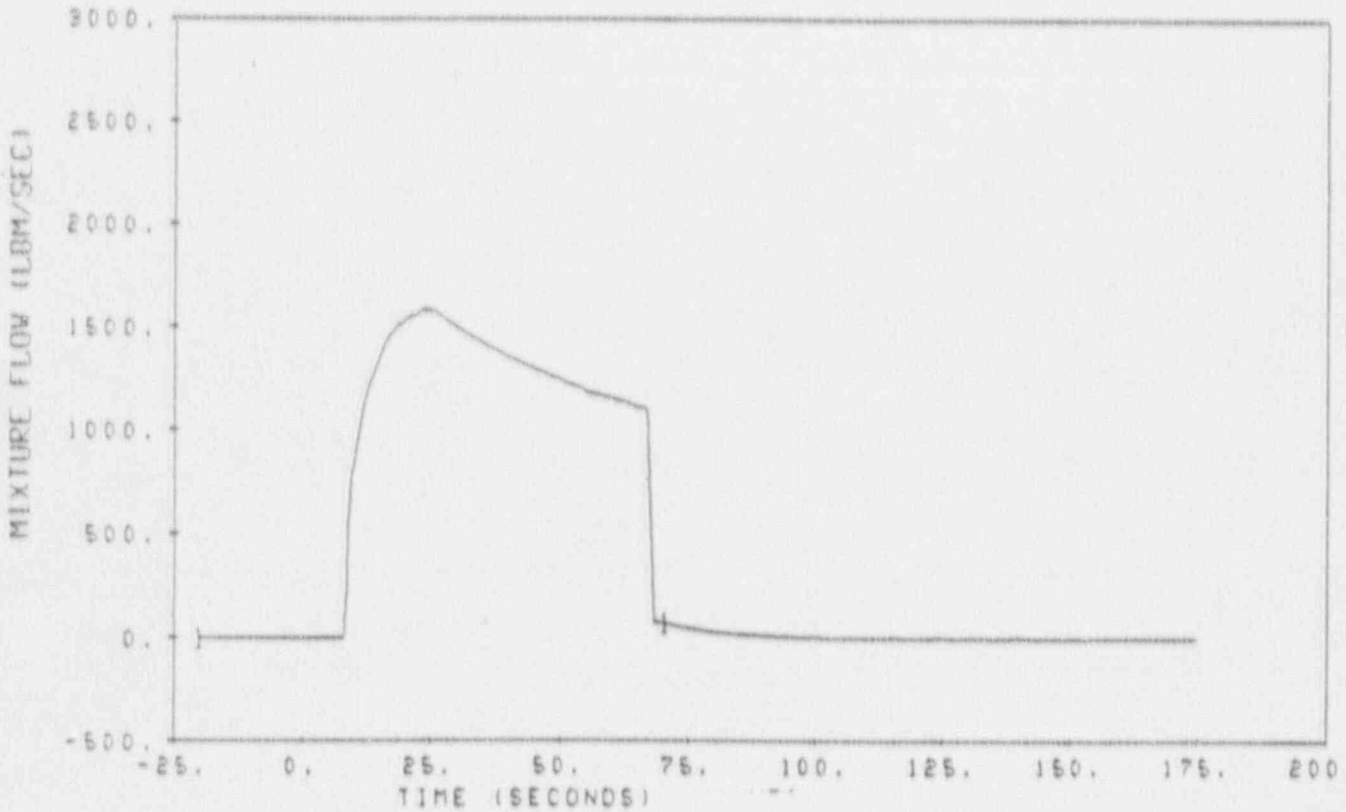

Figure 14.3.2-6. Accumulator Flow to the Cold Leg
(2250 psia case) (0.4 DECLG)

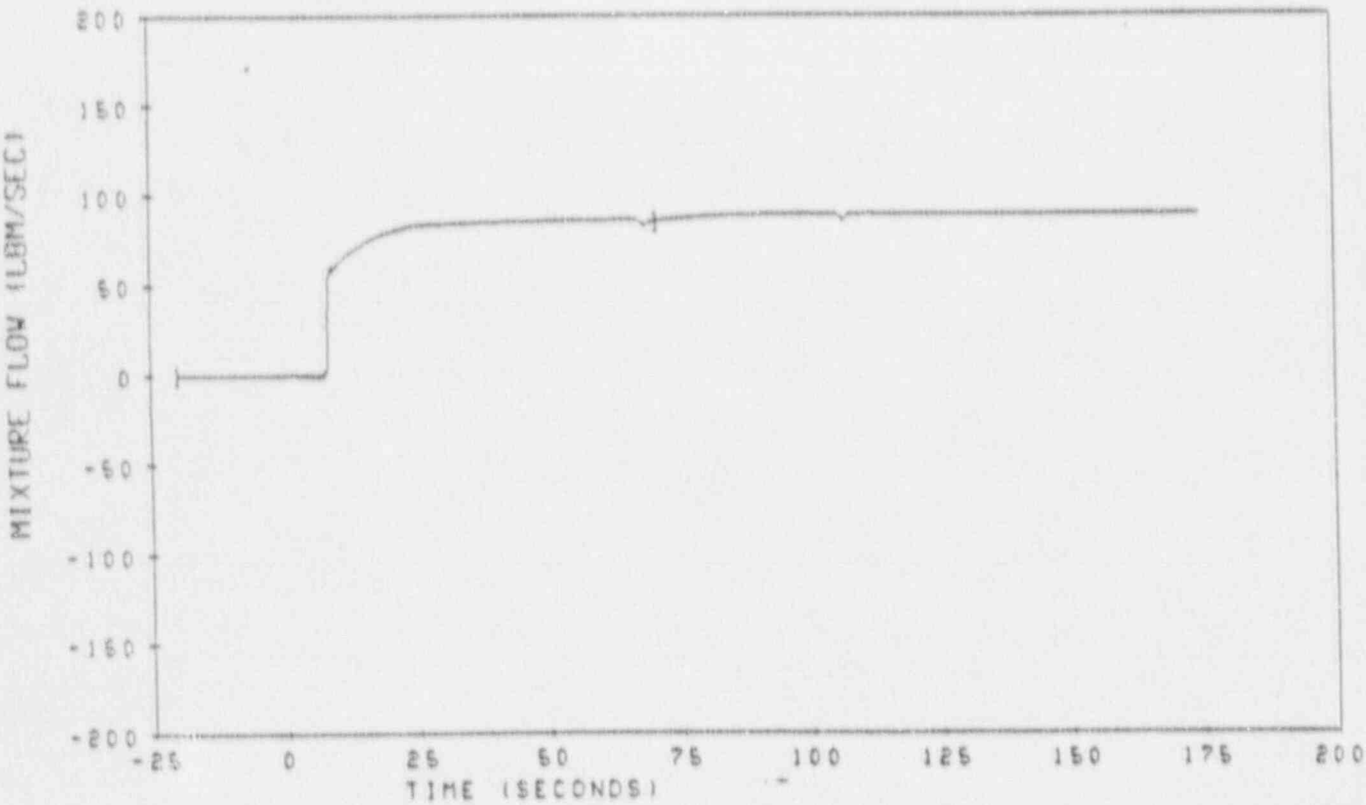

Figure 14.3.2-7. HHSI Flow to the Intact Cold Leg
(2250 psia case) (0.4 DECLG)

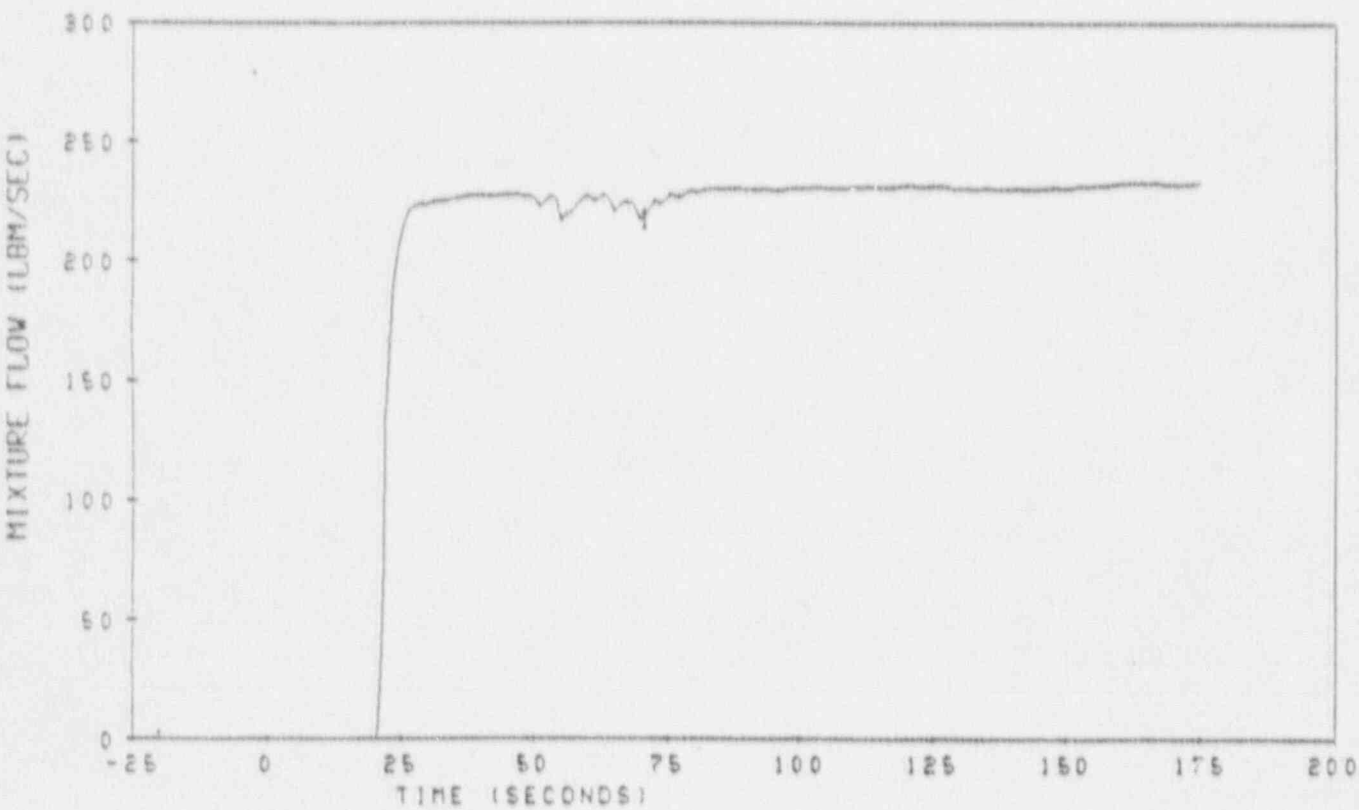

Figure 14.3.2-8. UPI Flow to the Upper Plenum
(2250 psia case) (0.4 DECLG)

ROD 1 - HOT ROD
ROD 2 - HOT ASSEMBLY
RODS 3 AND 4 - AVERAGE RODS
ROD 5 - LOW POWERED PERIPHERAL RODS

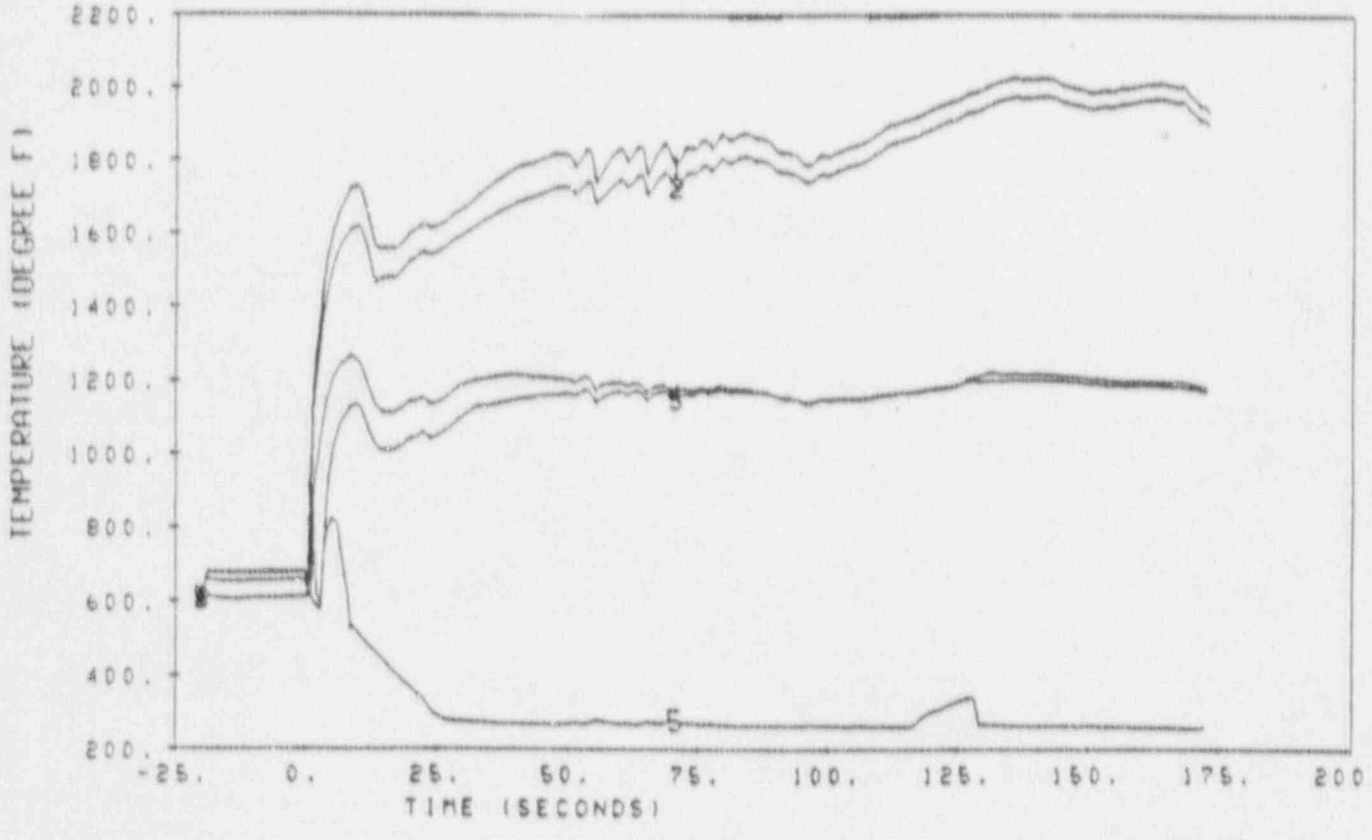

Figure 14.3.2-9. Cladding Temperature History at PCT Location (2250 psia case) (0.4 DECLG)

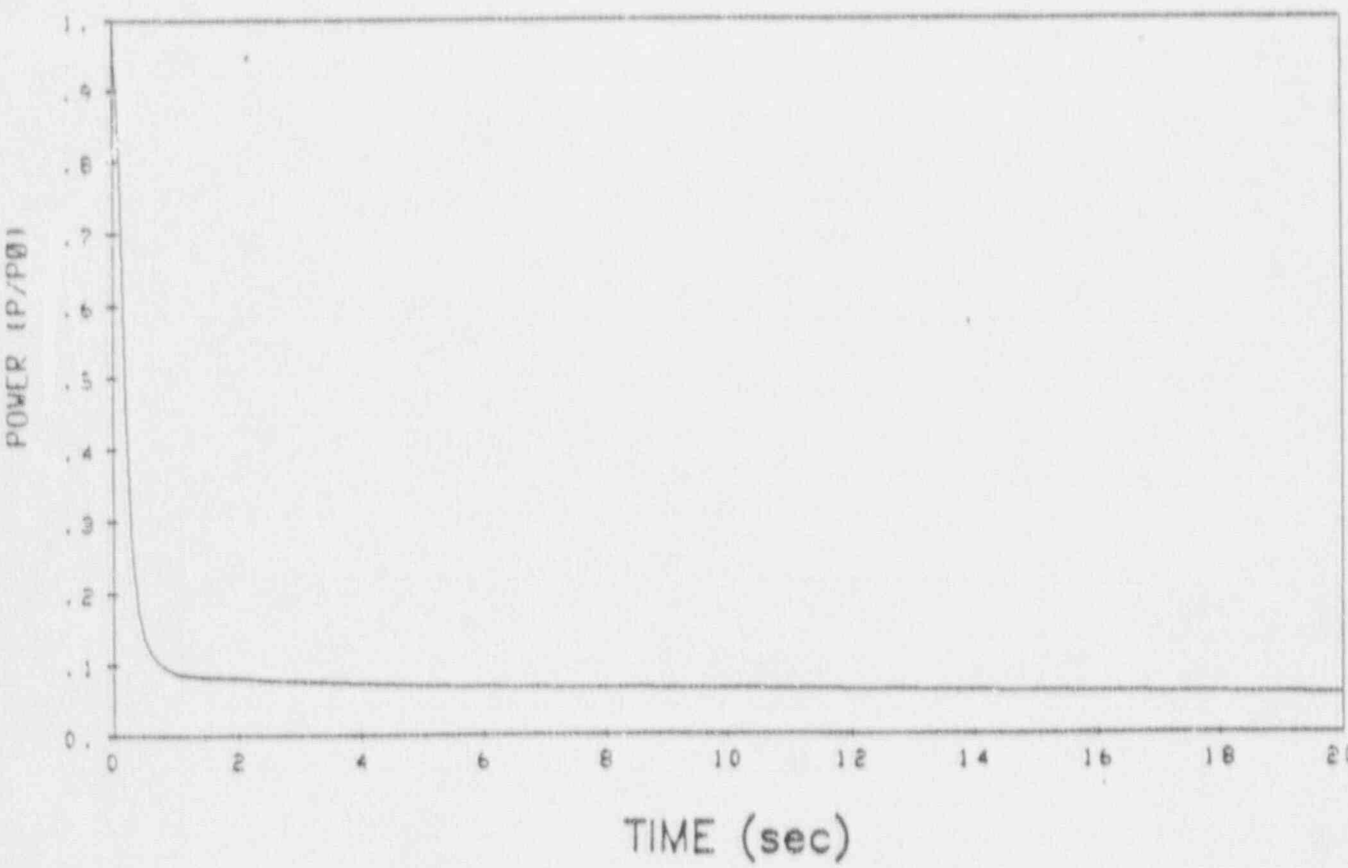

Figure 14.3.2-10. Core Power History for Blowdown

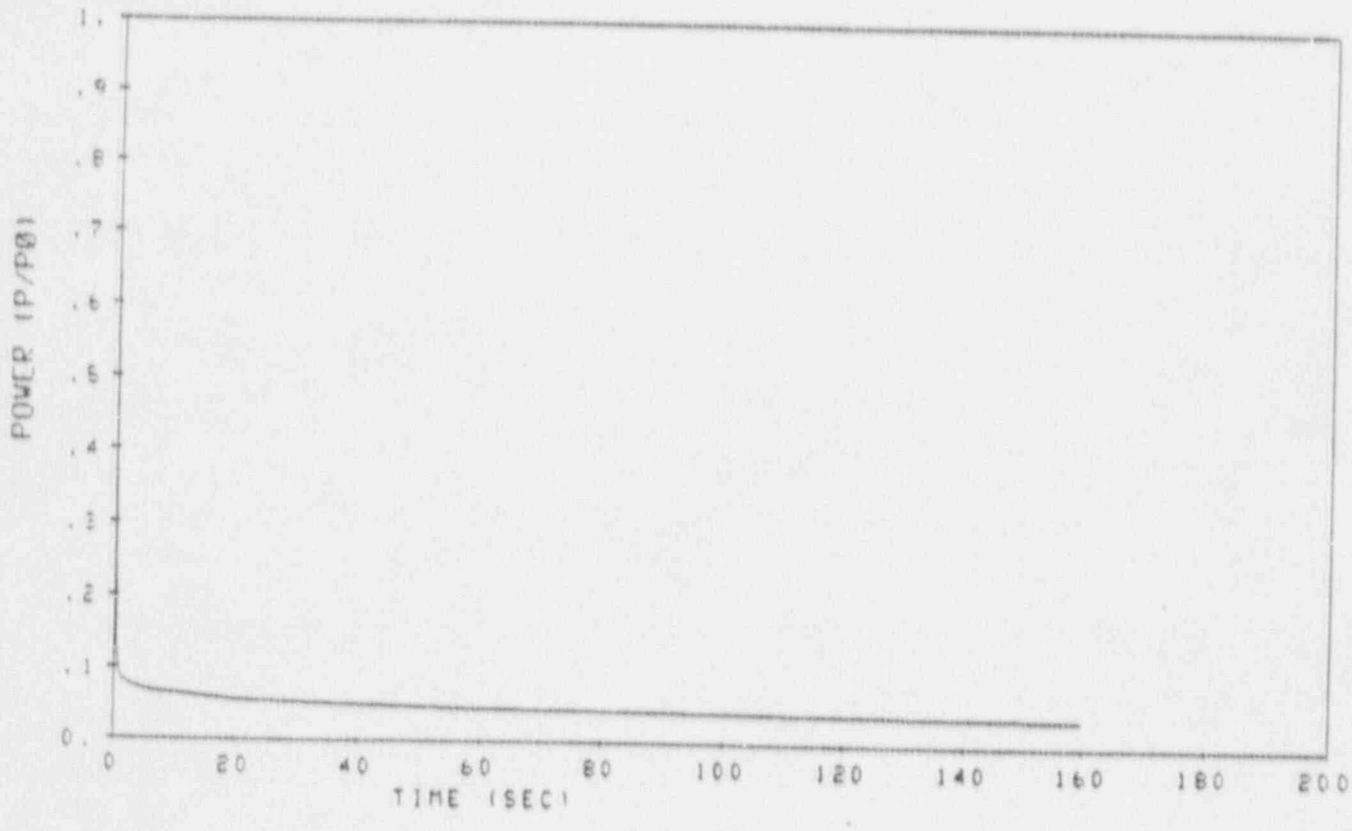

Figure 14.3.2-11. Core Power History for Complete Transient

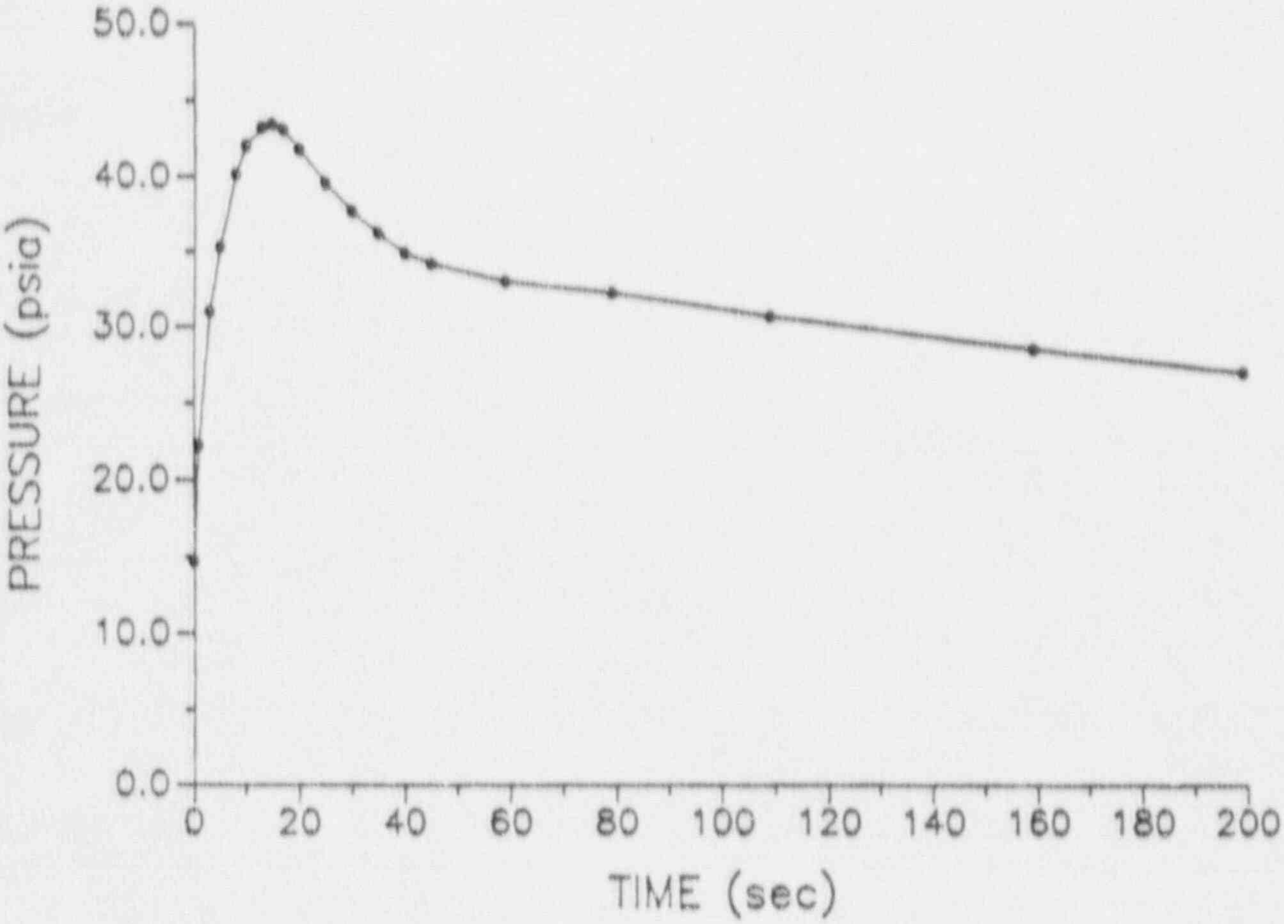

Figure 14.3.2-12. Containment Pressure from COCO

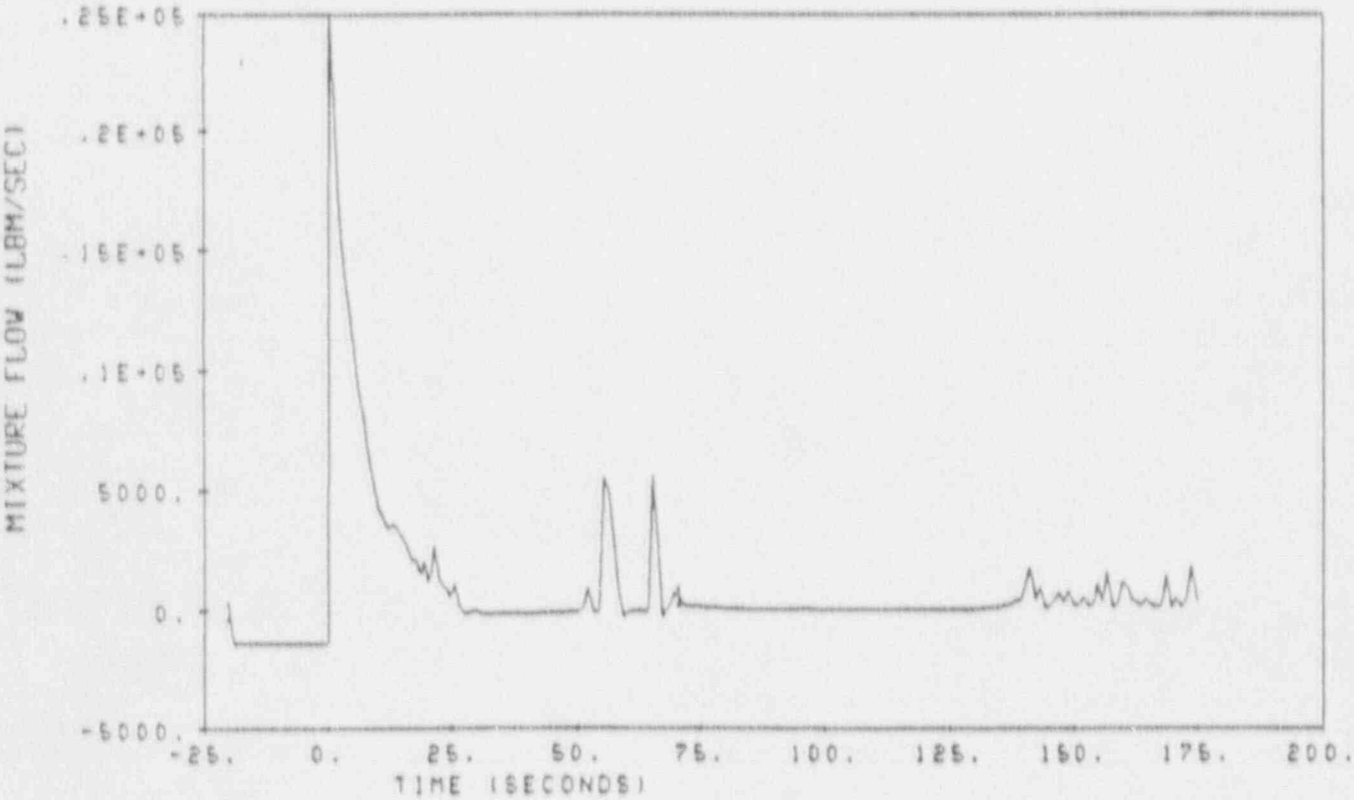

Figure 14.3.2-13. Vessel Side of Broken Cold Leg - Mixture Flow

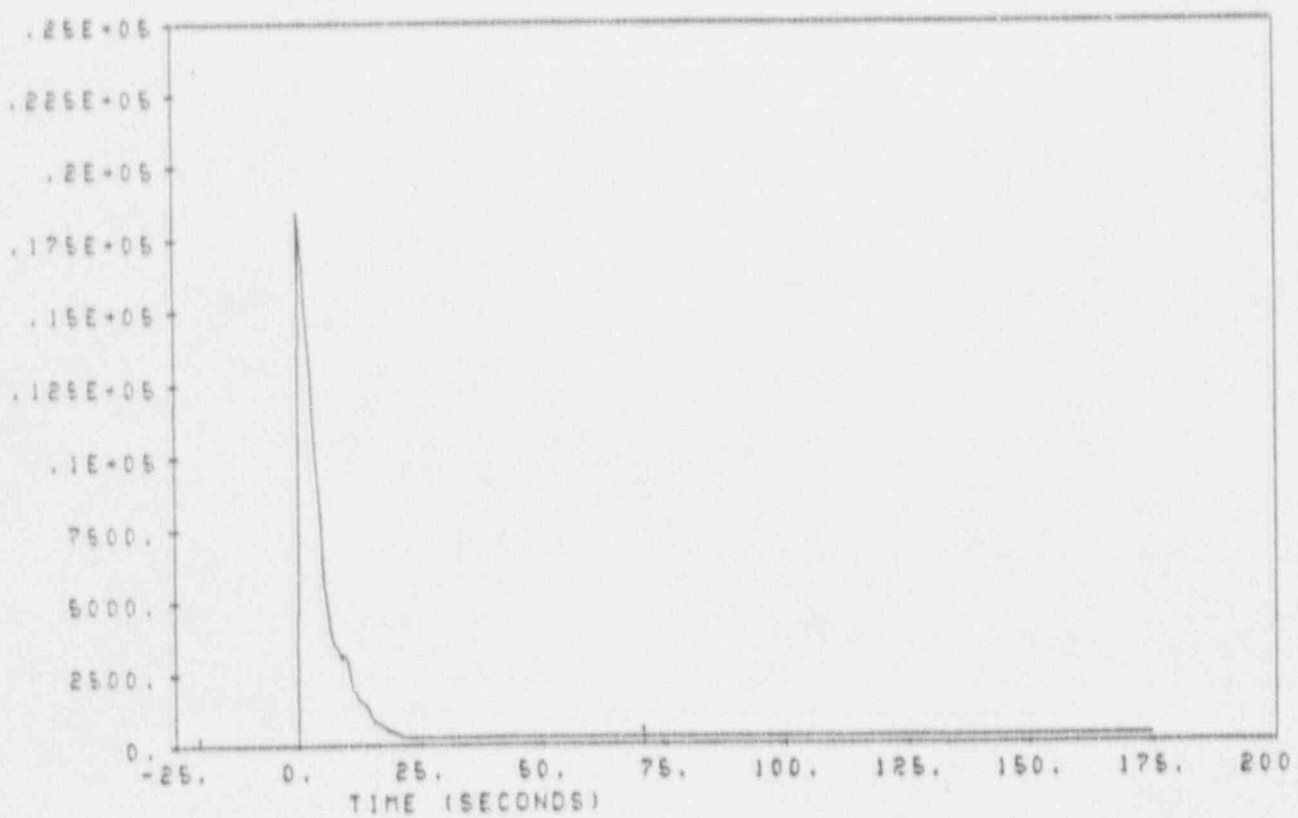

Figure 14.3.2-14. Loop Side of Broken Cold Leg - Mixture Flow