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February 29, 2008

AEP:NRC:8046
10 CFR 50.46

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
Response to Request for Additional Information Regarding the Reanalysis of Unit 1 Small Break Loss-of-Coolant Accident

References:

1. Letter from M. A. Peifer, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Small Break Loss-of-Coolant Accident Evaluation Model Reanalysis," AEP:NRC:7046, dated March 29, 2007 (ML071000431).
2. Electronic mail message from P. S. Tam, NRC, to M. K. Scarpello, I&M, et. al., "D. C. Cook Unit 1 – Draft RAI re: SBLOCA Reanalysis (TAC MD5297)," dated May 31, 2007 (ML071510338).
3. Letter from P. S. Tam, NRC, to M. K. Nazar, I&M, "D. C. Cook Nuclear Plant Unit 1 (DCCNP-1) – Request for Additional Information Regarding Reanalysis of Small-Break Loss-of-Coolant Accident (TAC MD5297)," dated August 10, 2007 (ML072050570).
4. Letter from M. A. Peifer, I&M, to U. S. NRC Document Control Desk, "Proposal for Response to Small Break Loss-of-Coolant Accident Reanalysis Request for Additional Information," AEP:NRC:7046-04, dated October 5, 2007 (ML072890355).

Dear Sir or Madam:

By Reference 1, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, provided the Nuclear Regulatory Commission (NRC) with the re-analyzed small break loss-of-coolant accident (SBLOCA) analysis to meet commitments made in accordance with 10 CFR 50.46(a)(3)(ii). The analysis used the NRC-approved Westinghouse NOTRUMP SBLOCA emergency core cooling system (ECCS) Evaluation Model methodology.

By Reference 2, the NRC transmitted a draft request for additional information (RAI) on the CNP Unit 1 SBLOCA reanalysis. A telephone conference was held between I&M and NRC staff on

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July 20, 2007, to clarify the RAI. Subsequent to the telephone conference call, the NRC sent a formal RAI, by Reference 3, with one of the questions from Reference 2 removed. The NRC requested a 60-day response to the RAI from the date of that letter. By Reference 4, I&M proposed to respond to the RAI by February 29, 2008. Enclosure 1 to this letter provides I&M's responses to the RAI. The figures and table associated with the response to RAI Question 8 are proprietary to Westinghouse and are provided in Enclosure 2. Enclosure 3 provides the non-proprietary version of Enclosure 2. Attachment 1 to this letter provides the regulatory commitment made in this submittal.

The proprietary information in Enclosure 2 is supported by an affidavit signed by Westinghouse, the owner of the proprietary information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. Attachment 2 contains the Westinghouse authorization letter, CAW-08-2392, accompanying affidavit, Proprietary Information Notice, and Copyright Notice for Enclosure 2.

During recent reviews of CNP Unit 2 ECCS modeling calculations that are in-progress, it was determined that the ECCS flow rates assumed for the cold leg recirculation alignment improperly reflect the residual heat removal (RHR) system discharge cross-tie valves being open. This is not correct since, upon aligning the RHR pump suction to the containment sump, the two trains of RHR would be isolated from each other. This incorrect modeling also affects the Unit 1 analysis submitted by Reference 1. The erroneously high ECCS recirculation alignment flow rates yield a non-conservative reactor coolant system inventory recovery rate for the 8.75-inch break case following the five-minute RHR flow interruption. The other analyzed break cases are not affected, as discussed in the enclosure for the response to RAI Question 10. The 8.75-inch break case is in the process of being re-analyzed and will be submitted to the NRC by June 30, 2008.

Should you have any questions, please contact Mr. James M. Petro, Jr., Regulatory Affairs Manager, at (269) 466-2491.

Sincerely,

Mark A. Peifer
Site Vice President

KS/rdw

Enclosures:

1. Indiana Michigan Power Company's Responses to the RAI
2. Table 8-1 and Figures 8-1 through 8-4 (Proprietary)
3. Table 8-1 and Figures 8-1 through 8-4 (Non-Proprietary)

Attachments:

1. Regulatory Commitment
2. Application for Withholding Proprietary Information From Public Disclosure

c: J. L. Caldwell, NRC Region III
K. D. Curry, Ft. Wayne AEP, w/o enclosures/attachments
J. T. King, MPSC
MDEQ – WHMD/RPMWS
NRC Resident Inspector
P. S. Tam, NRC Washington, DC

Enclosure 1 to AEP:NRC:8046

References for this attachment are identified on Page 27.

By Reference 1, Indiana Michigan Power Company (I&M) provided the Nuclear Regulatory Commission (NRC) with the Donald C. Cook Nuclear Plant (CNP) Unit 1 reanalyzed small break loss-of-coolant accident (SBLOCA) analysis to meet commitments made in accordance with 10 CFR 50.46(a)(3)(ii). The analysis used the NRC-approved Westinghouse NOTRUMP SBLOCA emergency core cooling system (ECCS) Evaluation Model methodology. By Reference 2, the NRC transmitted a draft Request for Additional Information (RAI) to I&M regarding the reanalysis. A telephone conference was held between I&M and NRC staff on July 20, 2007, to clarify the RAI. Subsequent to the telephone conference call, the NRC sent a formal RAI, by Reference 3, with one of the questions from Reference 2 removed. This enclosure provides I&M's response to the formal RAI.

NRC RAI Question 1

Table 1 on page 9 identifies the calorimetric uncertainty as 1.0034 percent. However, the text on page 3 quotes the full power analysis value of 100.34 percent. Please explain the discrepancy. Also, please also provide the reference for the calorimetric uncertainty determination.

Response to RAI Question 1

The calorimetric uncertainty value in Table 1 of the enclosure to Reference 1 is incorrect. The analysis was performed with an initial power level of 100.34%. A value of "0.34" should have been presented for the input parameter described as "Calorimetric Uncertainty, %" in Table 1. The uncertainty value of 0.34% is consistent with Unit 1 operation at the current licensed thermal power level of 3304 megawatt thermal. The calorimetric uncertainty was understood by NRC staff as documented in NRC Safety Evaluation dated December 20, 2002 (Reference 4). This error is tracked within the CNP Corrective Action Program.

NRC RAI Question 2

Please provide the results of the severed emergency core cooling system (ECCS) injection line case that utilizes the degraded ECCS injection into the intact lines.

Response to RAI Question 2

During the telephone conference between I&M and NRC staff on July 20, 2007, the NRC staff stated that the concern was that ECCS injection flows used in the analysis may not have properly accounted for the increased spill flow with this particular scenario, which would consequently reduce the flow rates in the injection lines further. As described below, I&M has confirmed that the hydraulic effects were properly modeled or provide the severed injection line case if it is found to be limiting.

The severed 10-inch nominal pipe diameter cold leg injection line that delivers residual heat removal (RHR), safety injection (SI), and accumulator flow to the reactor coolant system (RCS) is analyzed as an 8.75-inch (inside pipe diameter) break case. The charging injection paths to the

cold legs are via separate 1.5-inch nominal pipe diameter connections. The inside diameter of the charging injection line is less than that assumed in the minimum break case of the analyzed break spectrum of the CNP Unit 1 SBLOCA analysis (from 1.5-inch to 8.75-inch diameters).

A review of the supporting analysis calculation indicates that the non-limiting 8.75-inch case properly accounted for RHR and SI spilling to zero pounds per square inch gauge (psig) containment pressure. The ECCS flow rates presented in Tables 2 and 3 of the enclosure to Reference 1 present the ECCS injection phase flow rates. The ECCS injection phase flow rates used for the severed injection line case properly accounted for the increased spill flow to zero psig backpressure, as discussed on Page 3 of the analysis report provided in the enclosure to Reference 1.

During recent reviews of CNP Unit 2 ECCS modeling calculations that are in-progress, it was determined that the ECCS flow rates assumed for the cold leg recirculation alignment improperly reflect the RHR system discharge cross-tie valves being open. This is not correct since, upon aligning the RHR pump suction to the containment sump, the two trains of RHR would be isolated from each other. This incorrect modeling also affects the Unit 1 analysis submitted by Reference 1. The erroneously high ECCS recirculation alignment flow rates yield a non-conservative RCS inventory recovery rate for the 8.75-inch break case following the five-minute RHR flow interruption. The improper modeling of the RHR alignment after switchover to recirculation is tracked within the CNP Corrective Action Program. The other analyzed break cases are not affected, as discussed in the response to RAI Question 10. The 8.75-inch break case is in the process of being re-analyzed and will be submitted to the NRC by June 30, 2008.

NRC RAI Question 3

Does the reduced high-pressure safety injection (HPSI) impact the timing for precipitation and the switch time to simultaneous injection? Please explain. Please also provide the boric acid vs. time for the limiting large- and small- break LOCA and identify the time to switch to simultaneous injection.

Response to RAI Question 3

The timing to perform switchover of RHR and SI from the cold leg recirculation alignment to the hot leg recirculation alignment (charging continues to inject into the cold leg injection lines), is based upon calculations that are separate from the SBLOCA analysis for peak cladding temperature (PCT) determination. The calculations pertaining to post-LOCA boron precipitation were most recently performed during extensive work on the CNP emergency operating procedures (EOPs) during the extended dual unit shutdown beginning in 1997, and were described in Attachment 10 of Reference 5. Reference 5 provided part of the supporting information for a Technical Specification change request approved by NRC Safety Evaluation, dated December 13, 1999 (Reference 6). Since then there has been no need to change these calculations. The switchover timing prescribed in the EOPs is confirmed to remain valid during the core reload analysis process.

During the telephone conference call between I&M and NRC staff on July 20, 2007, the NRC staff considered the above information sufficient in responding to the requested information.

NRC RAI Question 4

Please explain the cause of the abrupt change in the core two-phase level which suddenly remains constant from 800 to 1000 seconds in Fig. 6, for the 3.25-inch break. Since this is a boil-off process, and the broken loop seal has cleared earlier, the abrupt termination of the level decrease during the boildown and uncovery period is not understood. Please provide the core inlet and outlet mass flow rates and the core liquid mass vs. time for this break.

Response to RAI Question 4

Figure 4-1 of this enclosure illustrates the sudden increase in pressure differential between the upper plenum (UP) and the downcomer (DC) at approximately 860 seconds, resulting in a depression of the mixture level. This excessive differential pressure is attributed to model numerics and is considered unrealistic. Moreover, this behavior is considered to be a conservative response. However, this situation corrects itself and as the pressure differential decreases, the depression is relieved and the mixture level demonstrates this as a termination of the level decrease during boil-down. After approximately 1000 seconds the normal boil-down process drives the mixture level response.

Figure 4-2 provides the core inlet mass flowrate, Figure 4-3 provides the core outlet mass flowrate, and Figure 4-4 provides the total core liquid mass.

Figure 4-1

Differential Pressure Between UP and DC vs Core Mixture Level
(3.25-inch break)

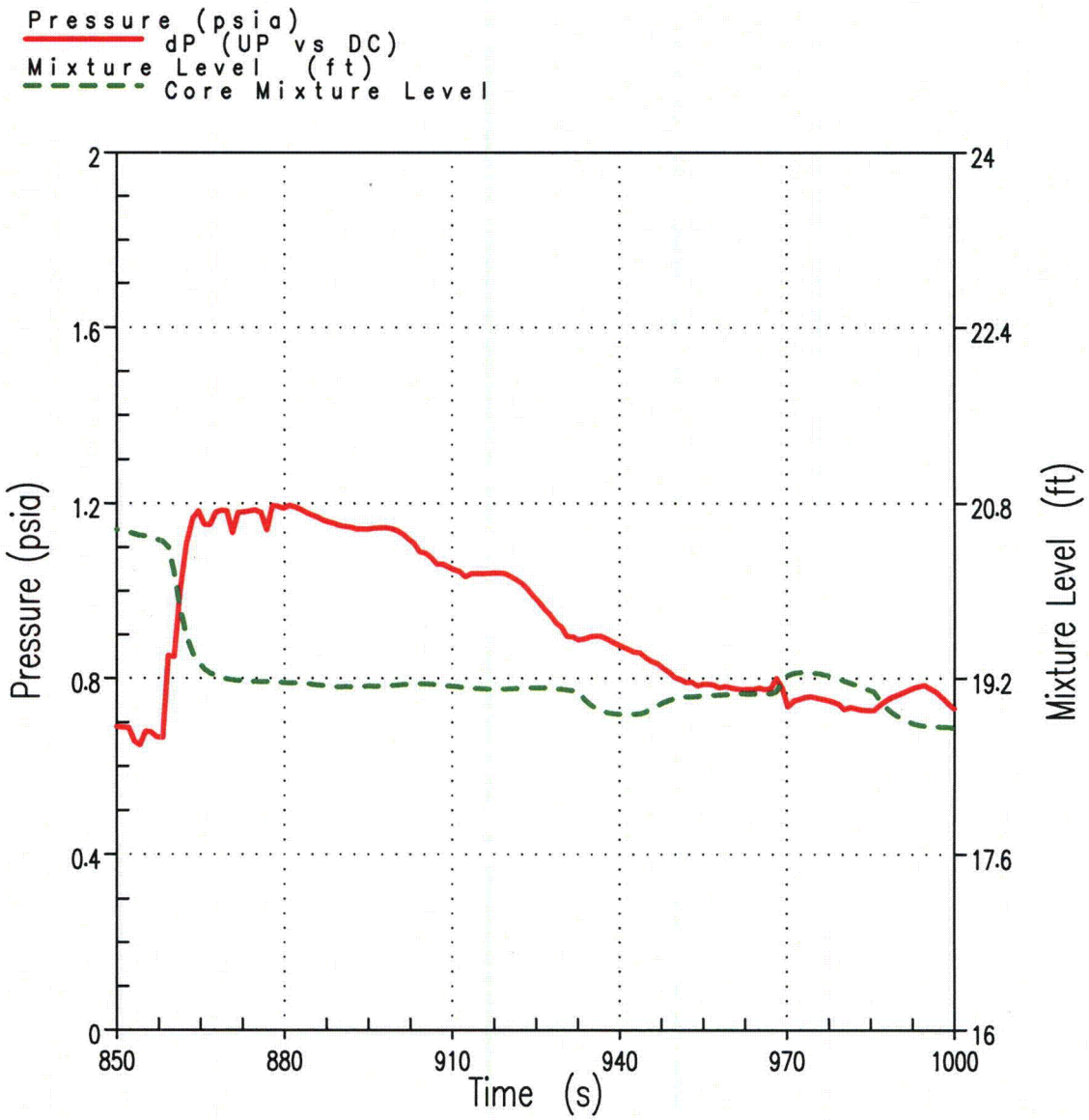


Figure 4-2

Core Inlet Mass Flowrate
(3.25-inch break)

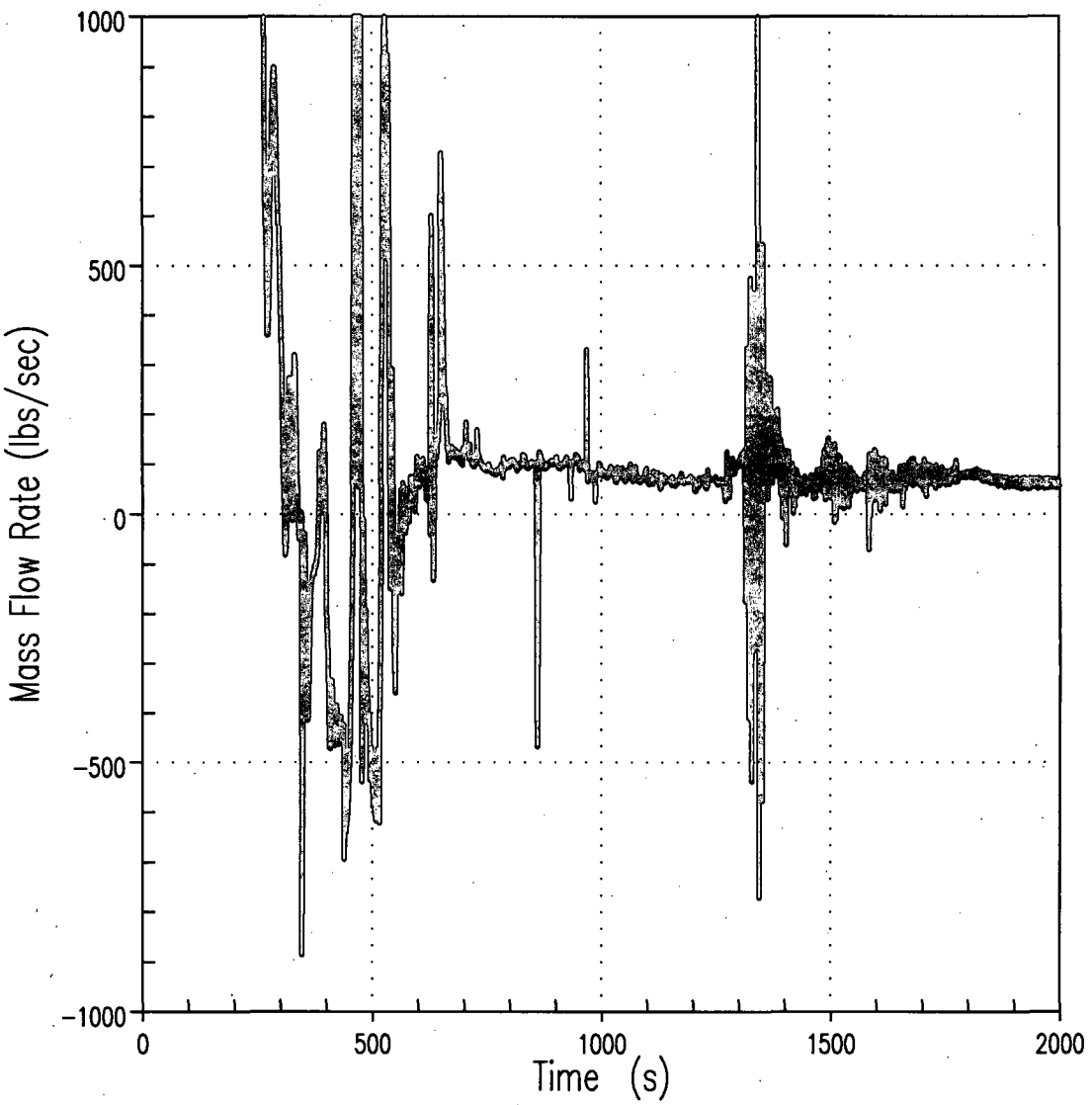


Figure 4-3

Core Outlet Mass Flowrate
(3.25-inch break)

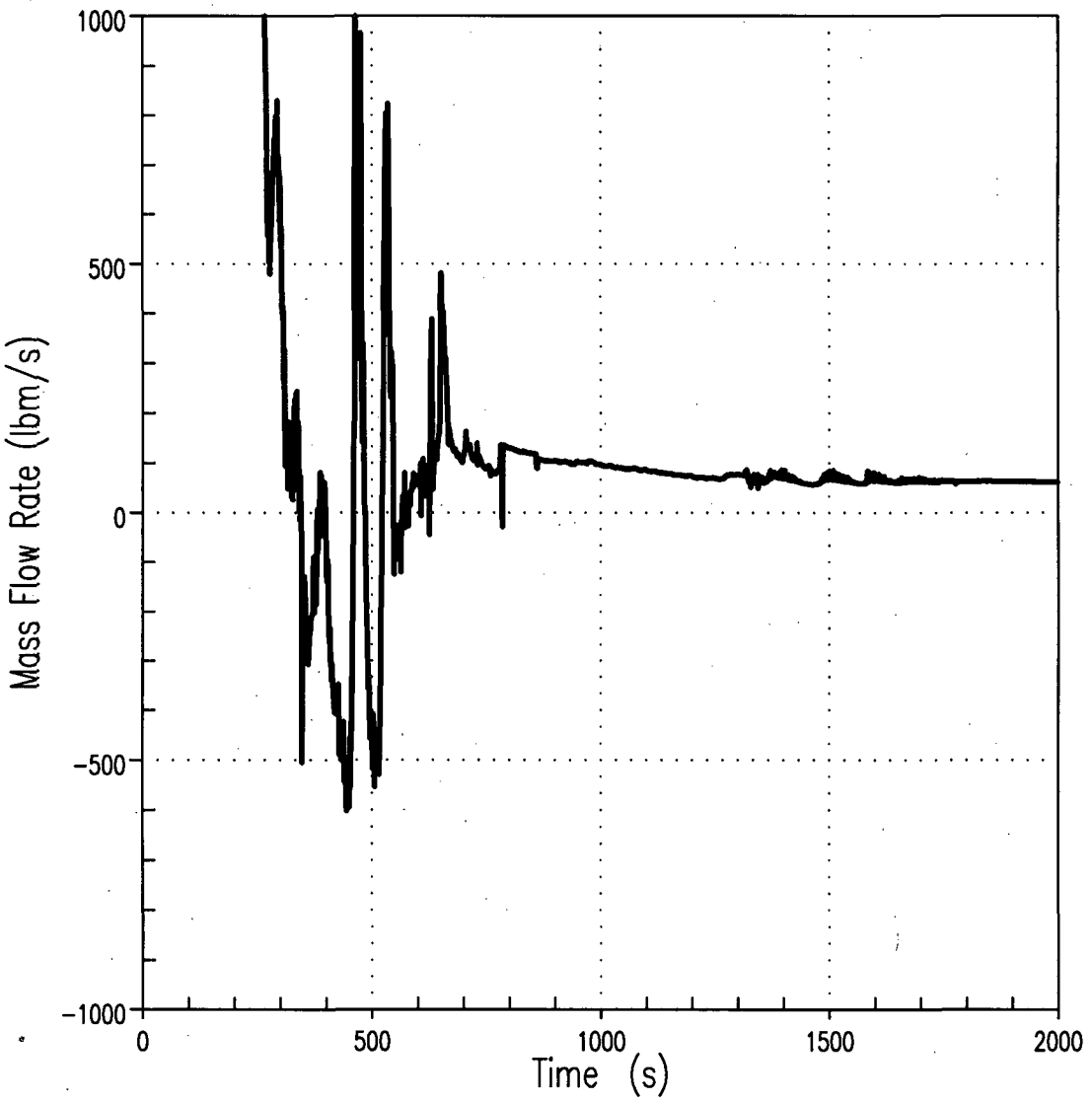
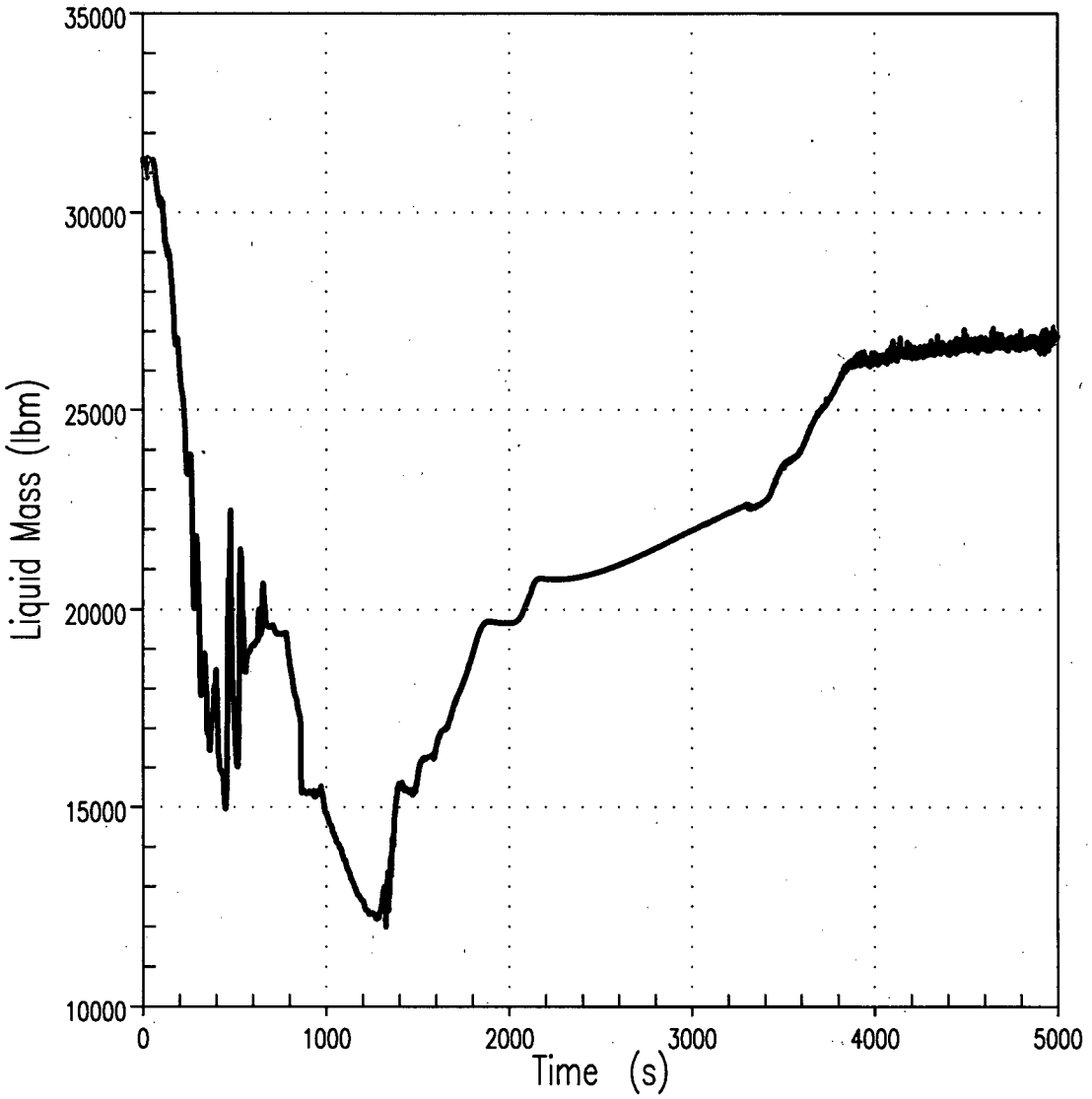


Figure 4-4

Total Core Liquid Mass
(3.25-inch break)



NRC RAI Question 5

The peak cladding temperature (PCT) turns over at about 1500 seconds for the 3.25-inch break in Fig. 14, at which time the heat transfer coefficient peaks during steam cooling shown in Fig. 15. At 1800 - 2000 seconds, the heat transfer coefficient drops below the value that produced the earlier heatup from 800 to 1500 seconds, yet the clad temperature continues to decrease. Please explain.

Response to RAI Question 5

The steady decrease in PCT from approximately 1500 to 1900 seconds for the 3.25-inch break can be attributed to higher heat flux (clad heat removal rate) during this period. As seen in Figure 5-1, after the PCT turns around at about 1500 seconds, the heat flux (heat transfer between the clad and the surrounding fluid) at the PCT elevation of 11.75 feet continues to remain above the value that produced the earlier heat up until approximately 1900 seconds. Figure 5-1 also illustrates that the rate of PCT decrease slows down after 1900 seconds due to the lower heat flux at this time.

Figure 5-2 is provided to clarify this further. The clad heat removal depends on both the heat transfer coefficient as well as the temperature difference (delta-T) between the clad and the surrounding fluid. The magnitude of the fluid temperature is impacted by accumulator injection during approximately 1500 to 1900 seconds which results in a fluid temperature decrease and subsequent delta-T increase as illustrated by Figure 5-2. Figure 5-2 also illustrates that during the time period of interest here (1500-1900 seconds), the variation in heat transfer coefficient is smaller in magnitude and is offset by the increase in delta-T, which in turn aids in the PCT decrease.

Figure 5-3 is presented to show that clad heat up and cool down corresponds to the uncovery and recovery period.

Figure 5-1

Clad Heat Flux versus PCT
(3.25-inch break)

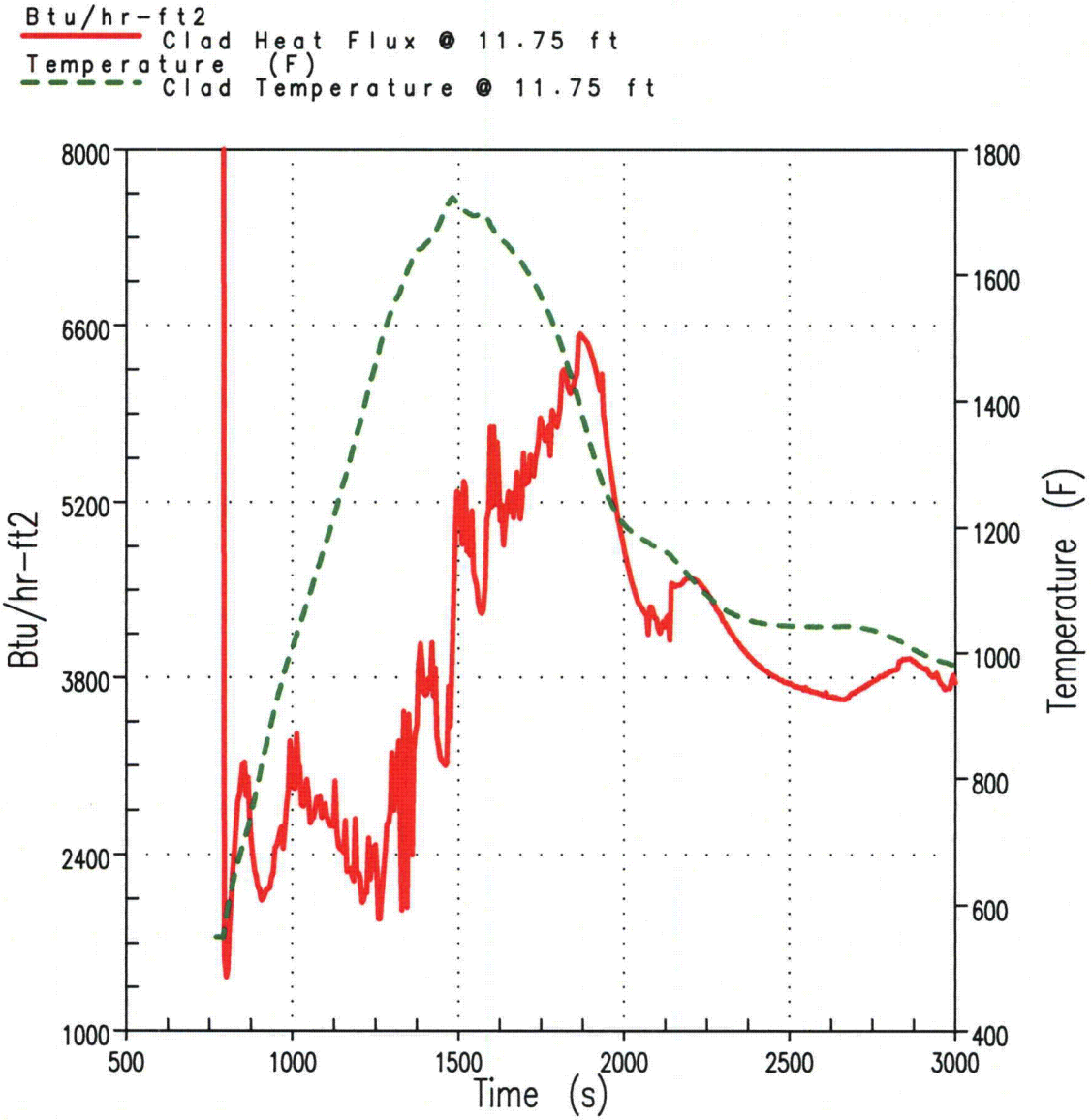


Figure 5-2

Delta-T (T_{Clad} minus T_{Fluid}) versus Heat Transfer Coefficient
(3.25-inch break)

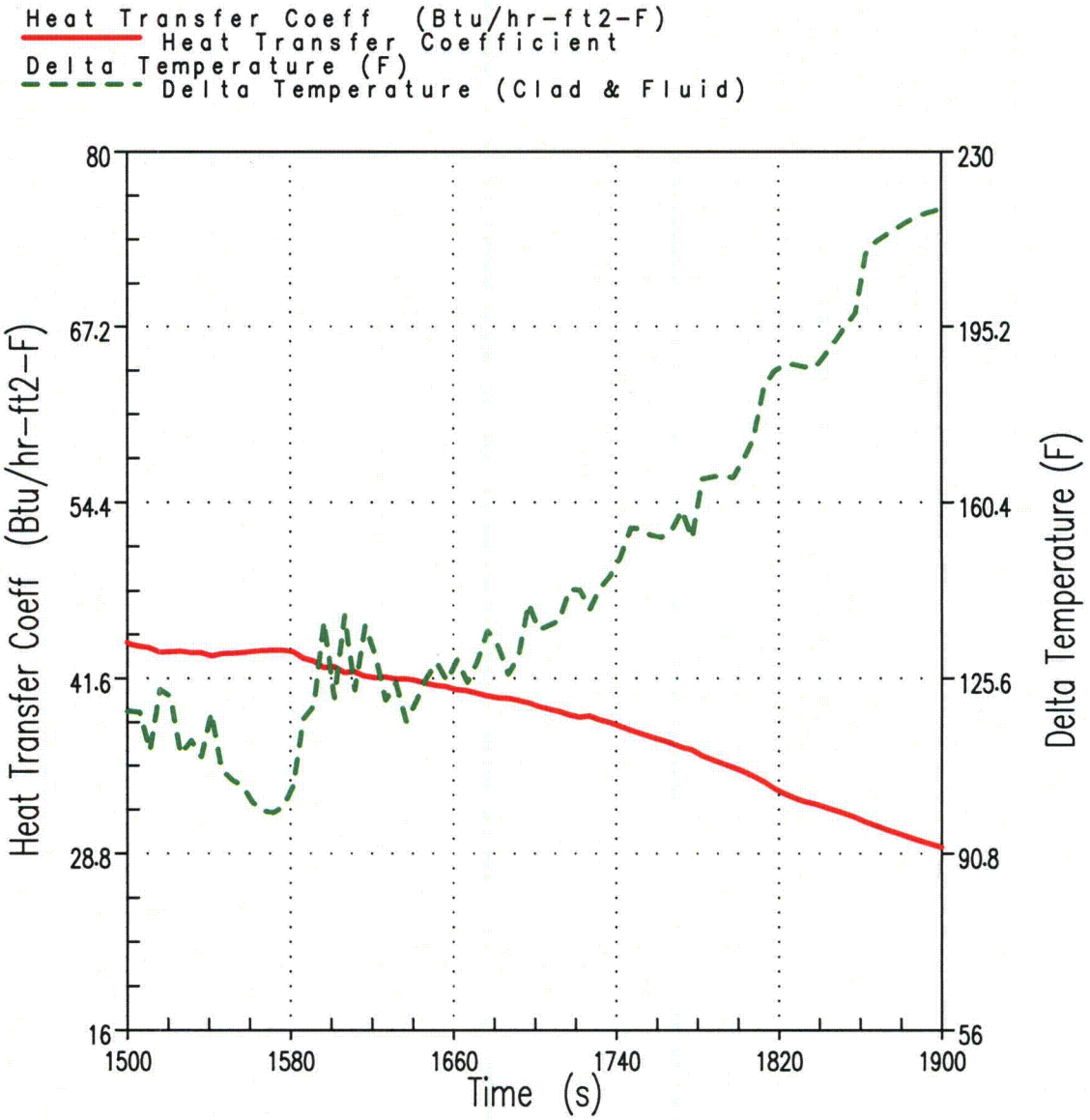
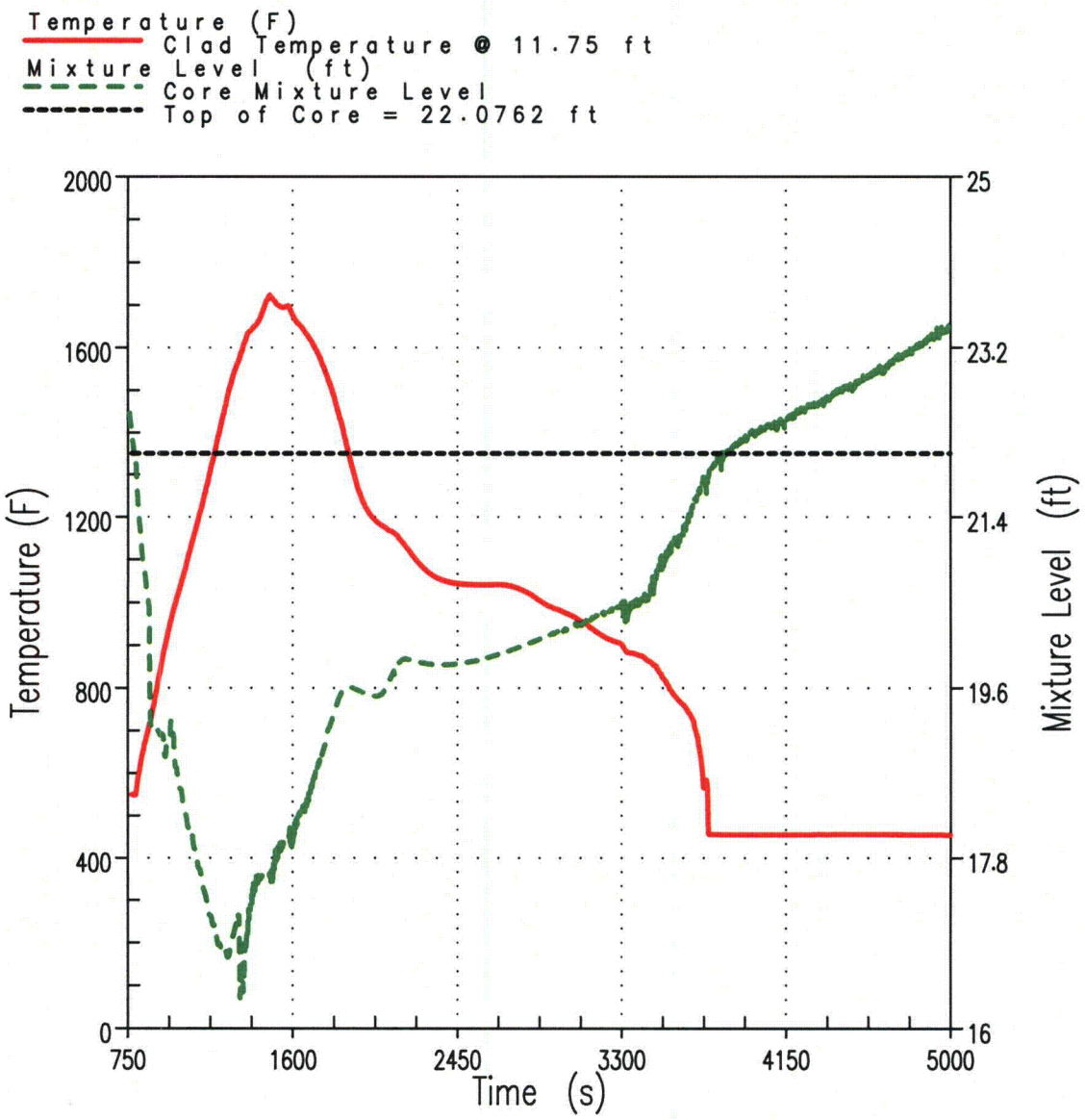


Figure 5-3

Clad Temperature versus Core Mixture Level
(3.25-inch break)



NRC RAI Question 6

Please explain the termination in the core two-phase level for the 3-inch break from 1000 to 1300 seconds in Fig. 28 for the 3-inch break. The 2.5-inch break has a similar behavior from 1200 to 1500 seconds in Fig. 25 as does the 2.5-inch break from 1100 to 1600 seconds in Fig. 22. Please explain the abrupt deviation from the boil-down and initial rapid loss of level behavior for these breaks. The froth-up behavior for the 2.5-inch break terminates and cools the hot spot during this period and does not appear physical. Please also provide the liquid mass and core inlet flow rates vs. time for these breaks.

Response to RAI Question 6

The core mixture level termination seen in the 2.5-, 2.75-, and 3-inch breaks can be attributed to the steady decrease in pressure differential between the UP and the DC at the same time, as seen for the 3-inch break case in Figure 6-1. A similar behavior also occurs for both the 2.5- and 2.75-inch breaks. Note that, as the break size decreases, this differential pressure effect becomes more pronounced as can been seen in Figures 6-2 for the 2.5-inch case and Figures 6-3 for the 2.75-inch case.

The froth up behavior seen at 1400 seconds in the 2.5-inch break case can be attributed to the mixture level variations during that time which causes a corresponding peak in the PCT at this time as seen in Figure 6-4.

The core inlet flow rates versus time for the 2.5-, 2.75-, and 3-inch breaks are provided in Figures 6-5, 6-6, and 6-7, respectively. The total core liquid mass for the 2.5-, 2.75-, and 3-inch breaks are provided in Figures 6-8, 6-9, and 6-10, respectively.

Figure 6-1

Differential Pressure Between UP and DC vs Core Mixture Level
(3-inch break)

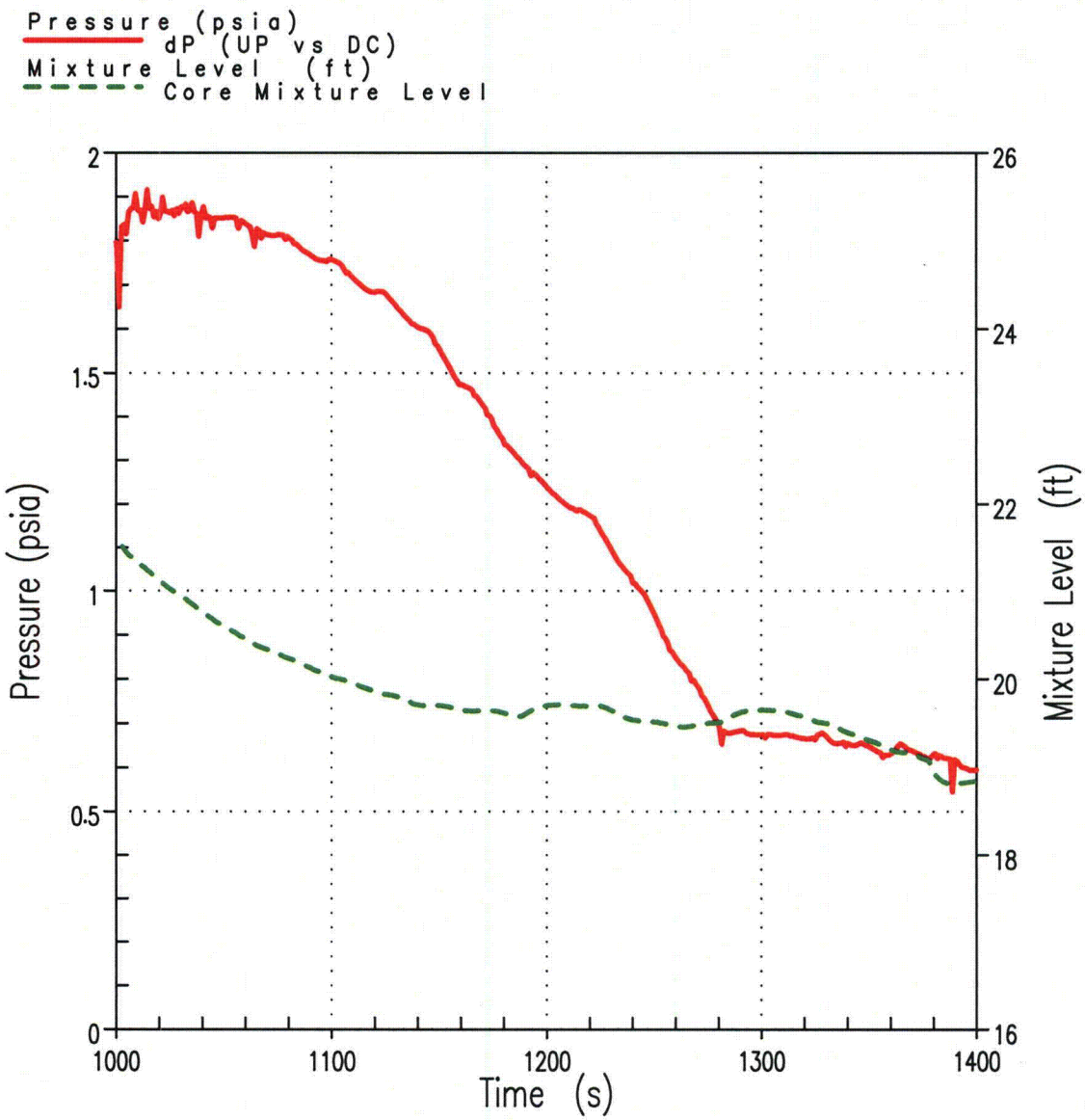


Figure 6-2

Differential Pressure Between UP and DC vs Core Mixture Level
(2.5-inch break)

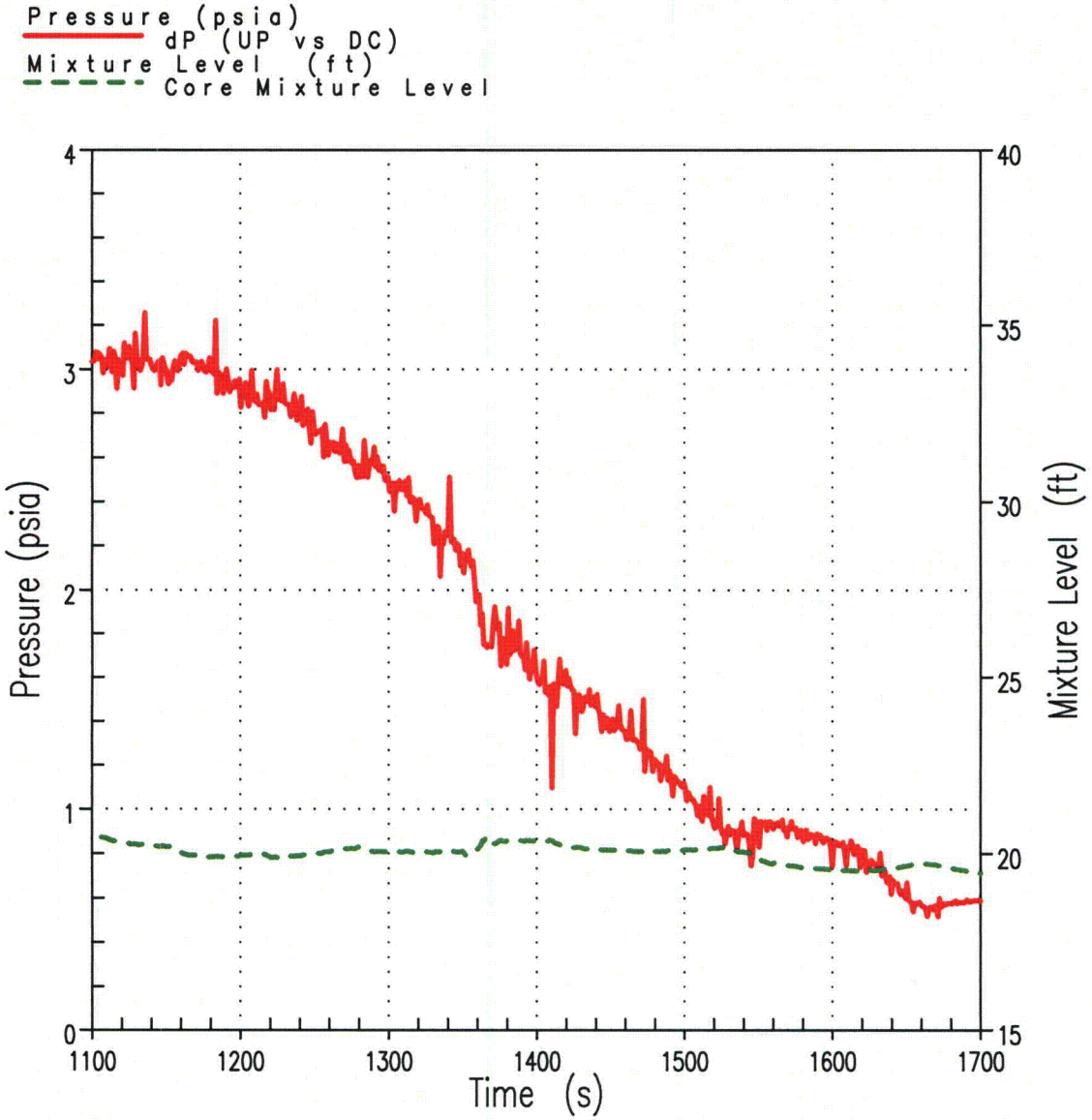


Figure 6-3

Differential Pressure Between UP and DC vs Core Mixture Level
(2.75-inch break)

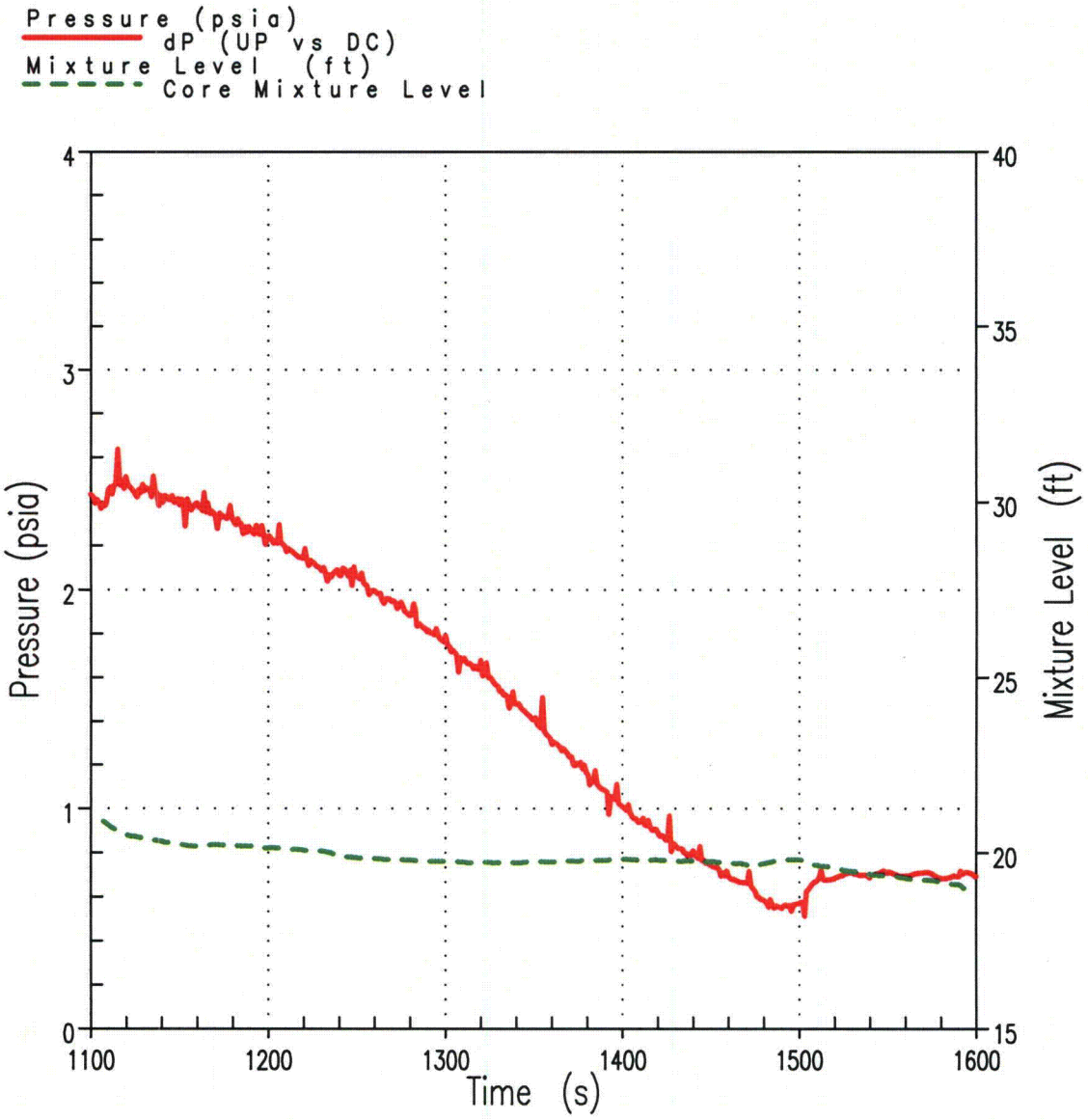


Figure 6-4

Core Mixture Level vs PCT
(2.5-inch break)

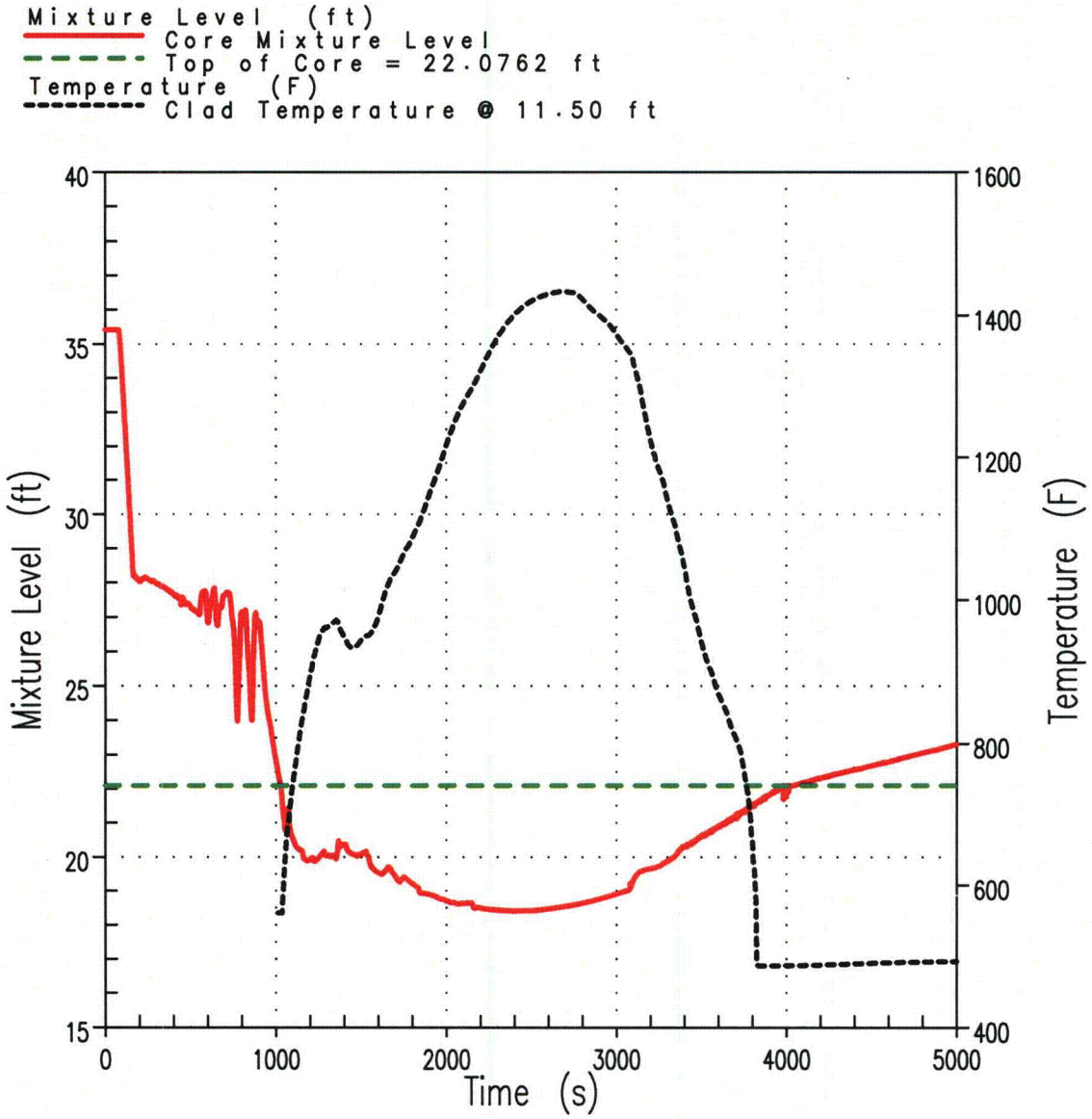


Figure 6-5

Core Inlet Flowrate
(2.5-inch break)

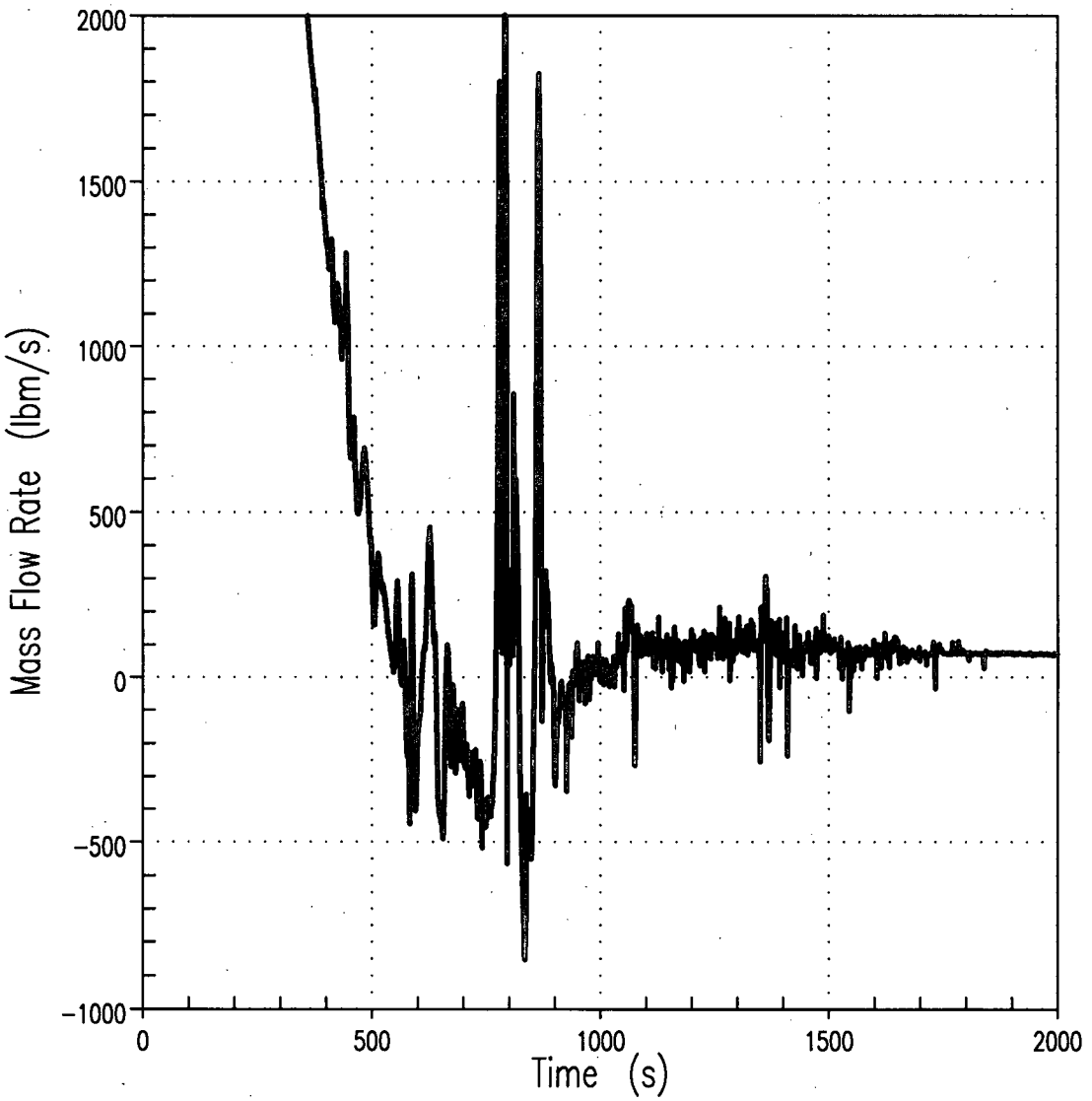


Figure 6-6

Core Inlet Flowrate
(2.75-inch break)

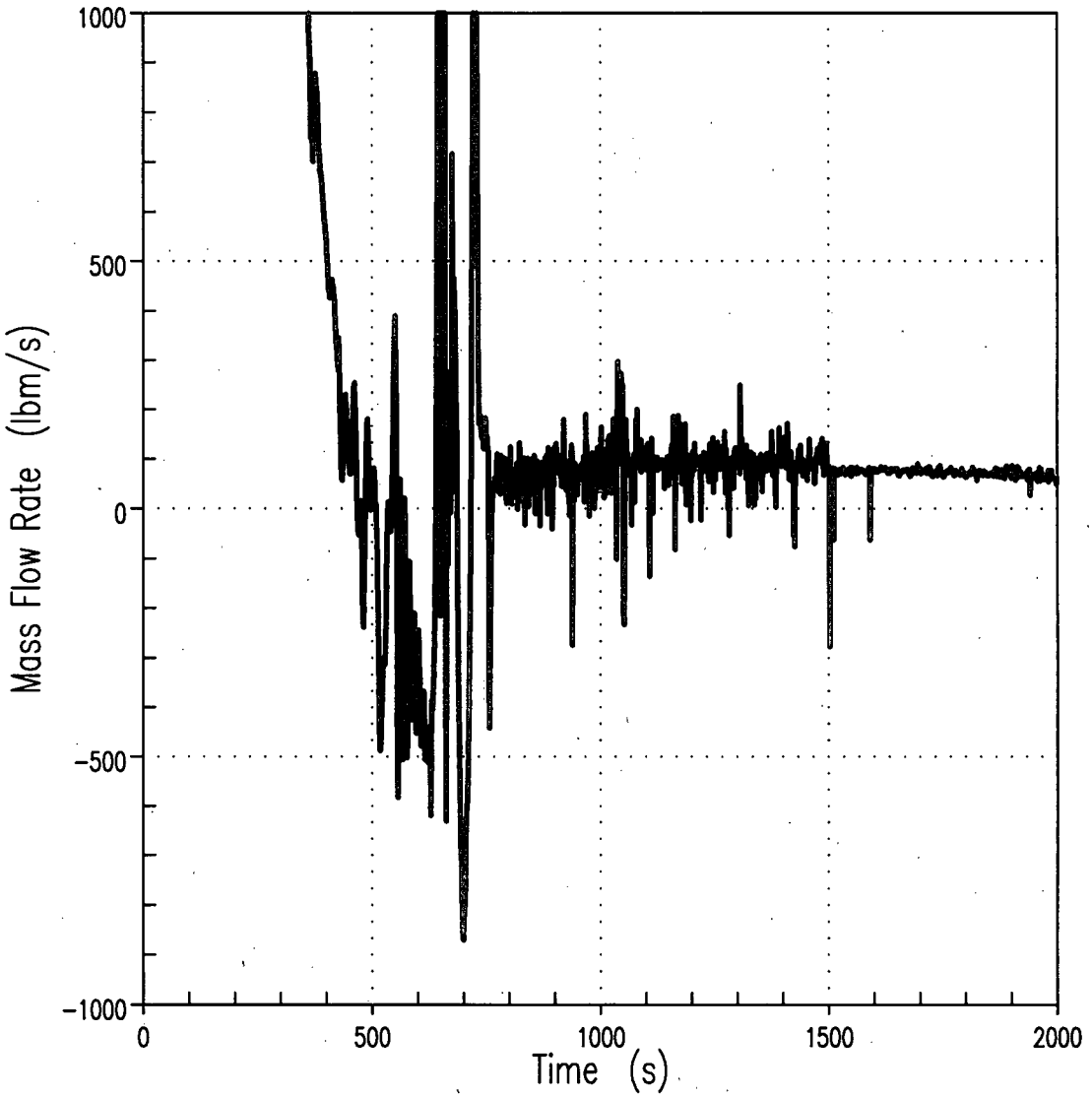


Figure 6-7

Core Inlet Flowrate
(3-inch break)

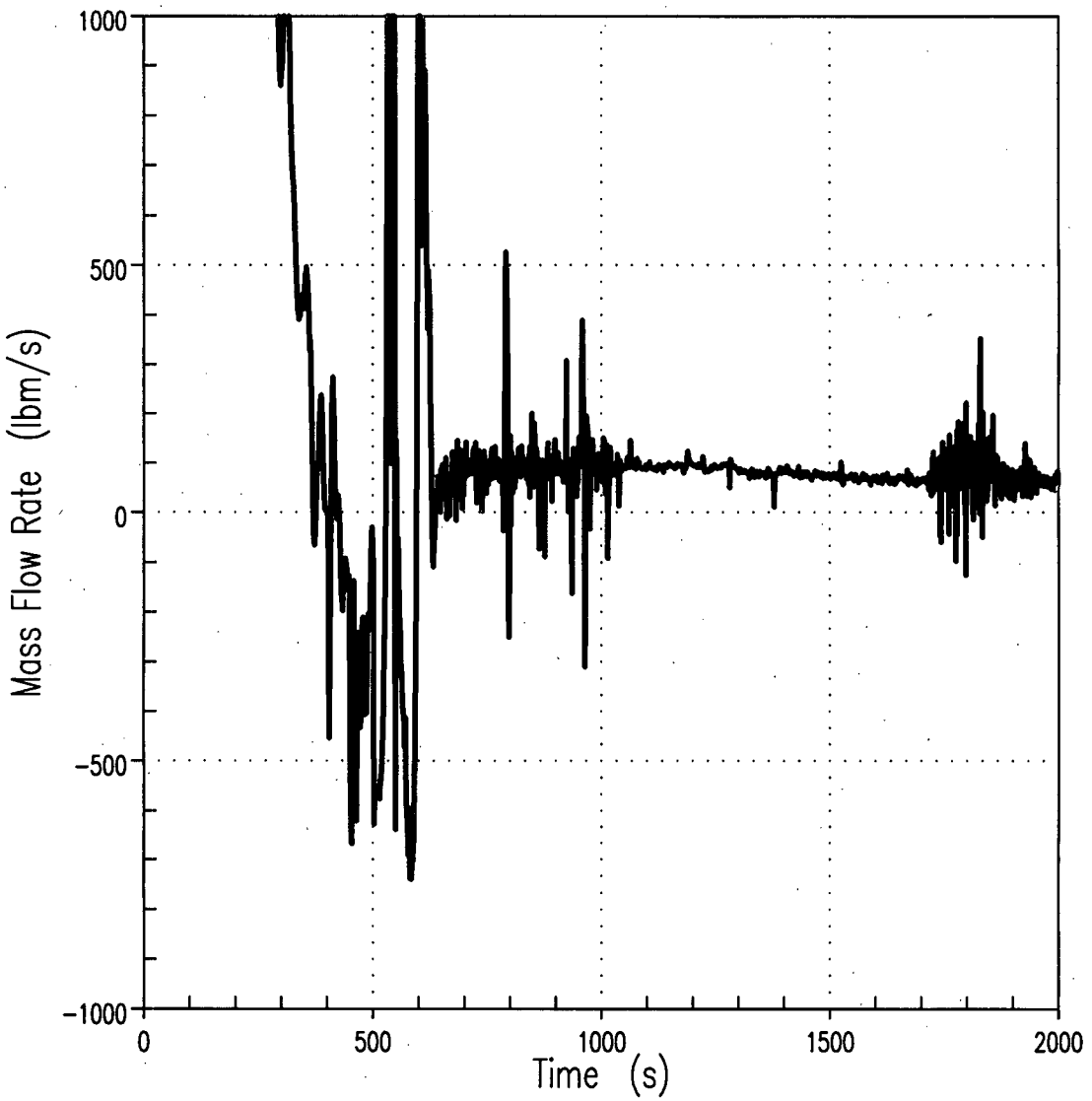


Figure 6-8

Total Core Liquid Mass
(2.5-inch break)

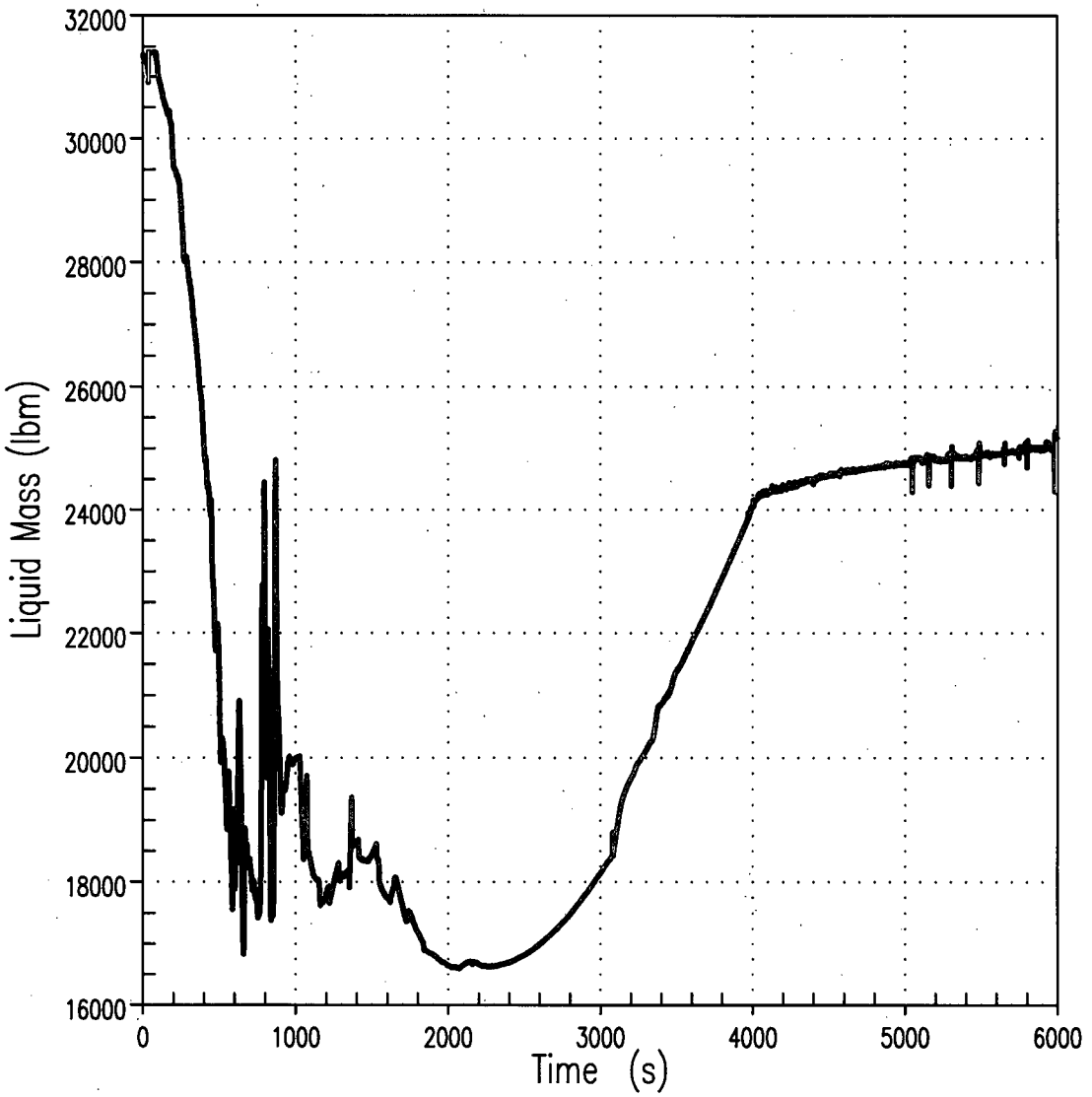


Figure 6-9

Total Core Liquid Mass
(2.75-inch break)

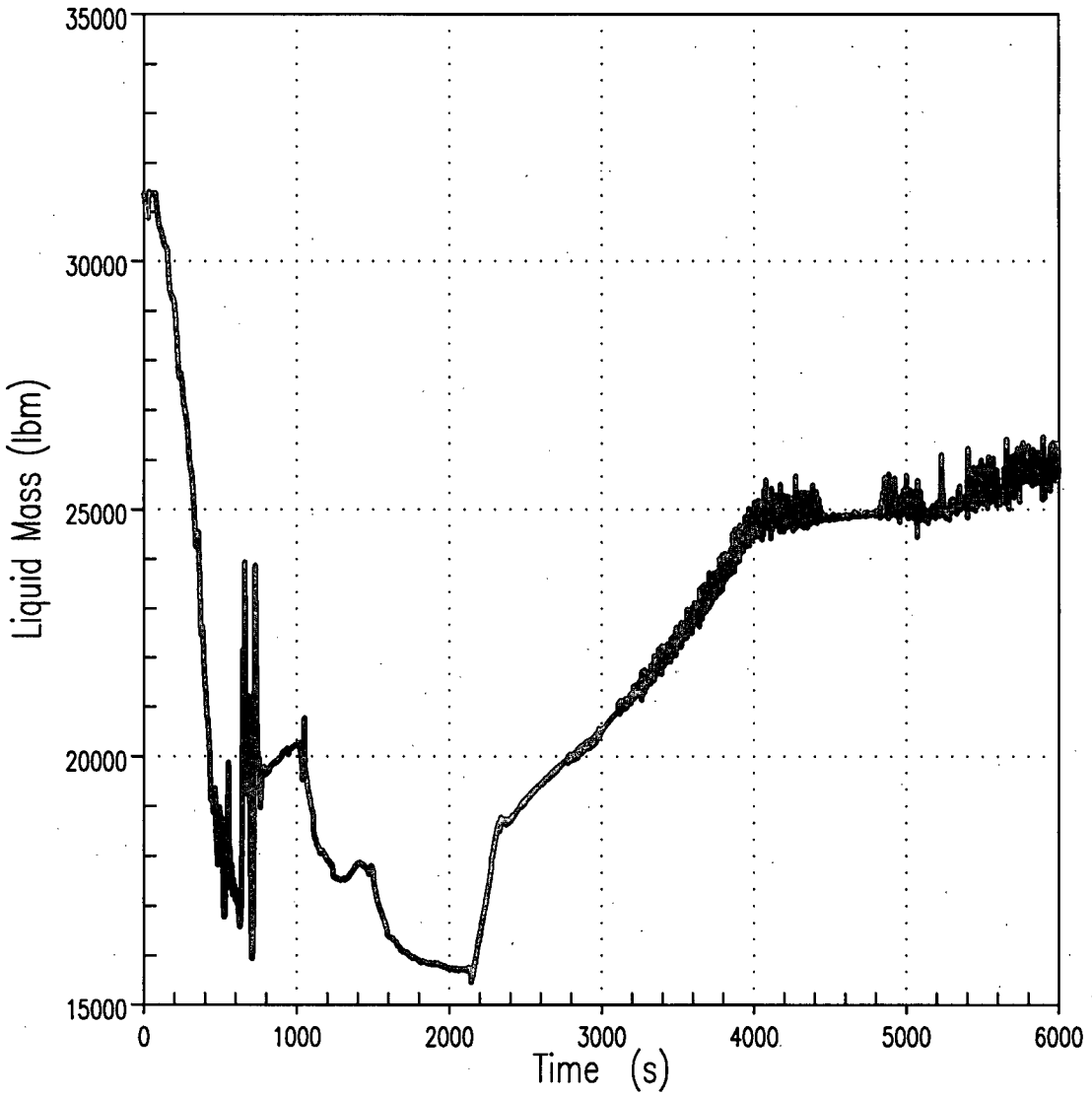
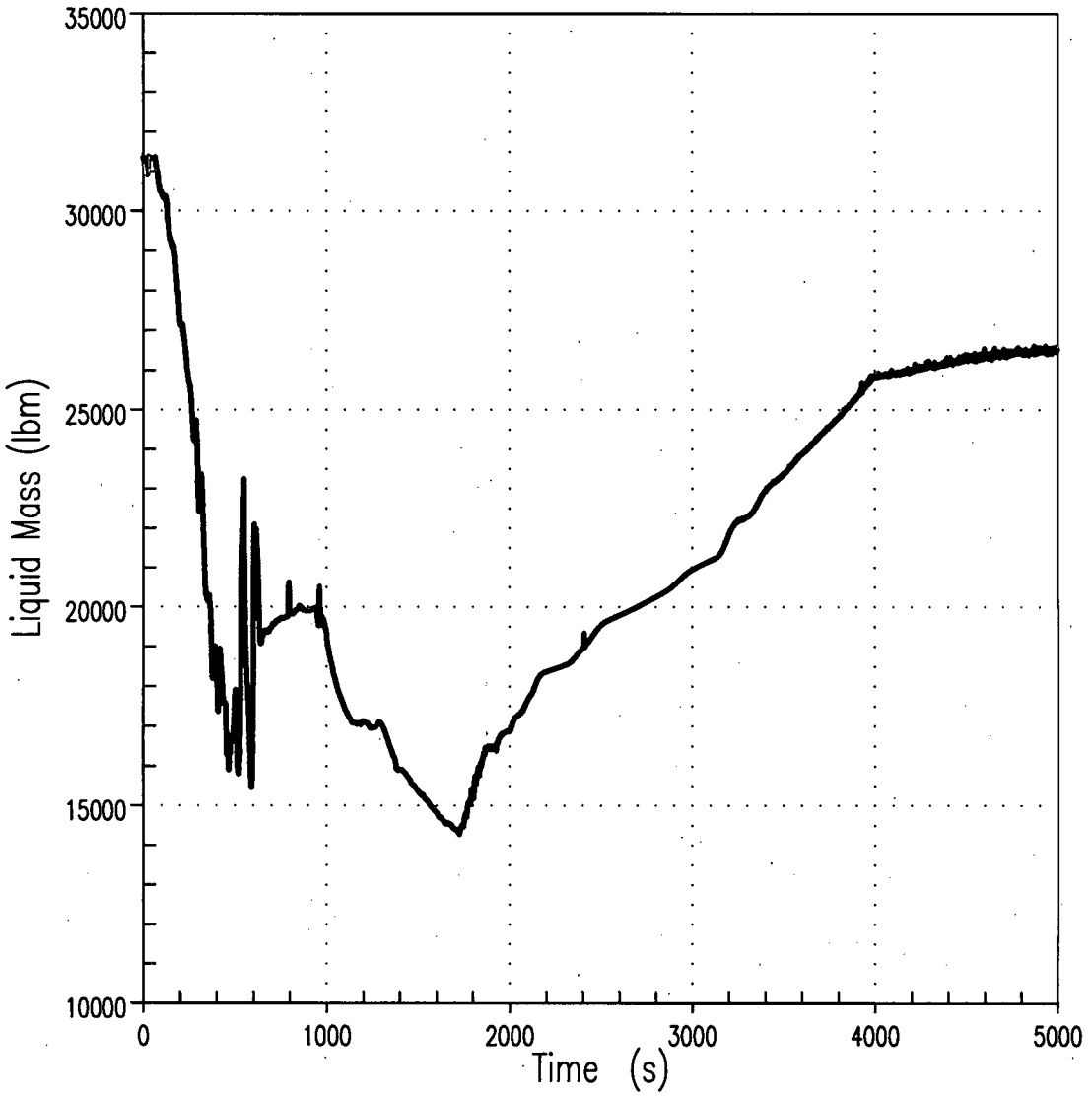


Figure 6-10

Total Core Liquid Mass
(3-inch break)



NRC RAI Question 7

The hot spot begins heatup at about 100 seconds in Fig. 29, when the level drops to about 21 feet. The clad heat-up is terminated at 200 seconds when the level increases from the minimum of 15 to 16 feet. Why does the PCT turn around at level of 16 feet when heat up of the hot spot begins at a level of 21 feet or less during the earlier portion of the event? How is the heat transfer at the hot spot during steam cooling calculated? Please provide the reference and briefly describe the correlations used to determine the heat transfer at the hot spot.

Response to RAI Question 7

By inspection of Figure 29, the 100 seconds and 200 seconds discussed in NRC RAI Question 7 is intended to mean 1000 seconds and 2000 seconds, respectively.

This question refers to the 3-inch break based on the reference to Figure 29. For the 3-inch break, the PCT does not turn around at 15 or 16 feet as discussed below.

For all the SBLOCA cases, analyzed clad heat up begins when the mixture level drops below the top of active fuel (approximately 22 feet). For the 3-inch break this occurs around 1000 seconds and the core mixture level continues to decrease to a minimum level of 17.7 feet and then begins to increase around 1700 seconds when accumulator injection begins. The hot rod PCT occurs around 2000 seconds when the mixture level increases to about 18 feet, by which time there is sufficient core cooling (by the combined effect of accumulators and SI flows) to terminate further clad heat up.

Section 3-13-1 of WCAP-10054-P-A (Reference 7), identifies that the LOCTA-IV code (Reference 8) is used in NOTRUMP-EM to calculate steam cooling heat transfer above the quench front for SB LOCA transients. The core heat transfer correlations used in NOTRUMP-EM during steam cooling are described in Reference 8, and modifications to the LOCTA-IV code for application with NOTRUMP-EM are described in Section 3-13-3 of Reference 7.

NRC RAI Question 8

For each break size, please identify the timing for and the location of the loop seals that clear of liquid. Please provide plots of the liquid levels in the vertical sections of the loop seals for the most limiting break.

Response to RAI Question 8

Loop-seal clearing information was provided with the time sequence of events information, which is presented in Table 6 of the enclosure to Reference 1. The information in Table 6 of Reference 1 includes a footnote describing that "loop-seal clearing is assumed to occur when the steam flow through the broken loop, loop-seal is sustained above 1 pound-mass per second (lbm/s)."

The loop-seal model is addressed in detail (including numerous associated experimental tests and staff audit analyses) in the May 2, 1985, safety evaluation report contained within Reference 7. The analysis methodology includes a restriction that prevents significant amounts of steam flow through the loop-seal in the intact loops. The result is that significant amounts of steam are only allowed to vent from one loop-seal (the broken loop). This assures conservative loop-seal behavior which yields a more severe core mixture level and cladding heat-up transient response during a small break LOCA. This artificial restriction was applied to the CNP Unit 1 analysis consistent with the modeling methodology clarified in Reference 9.

During a telephone conference call between I&M and NRC staff on July 20, 2007, the staff explained that in addition to the Table 6 loop-seal clearing information, the NRC staff would like: a) which loop-seals clear first and b) how many loop-seals are assumed to have cleared. NRC staff explained that the requested information is for the purpose of performing confirmatory checks of the method using the NRC staff's computer codes. As stated by I&M personnel during the July 20th call, the level of detail requested by NRC staff from the approved Westinghouse NOTRUMP model is not normally provided in an analysis report. The loop-seal clearing model is part of the NRC-approved NOTRUMP analysis methodology. However, I&M is providing the following additional information from Westinghouse concerning timing for, and the location of, the loop-seals that clear of liquid for each break size, as presented in Table 8-1. Figures 8-1 through 8-4 present the mixture level in the vertical section of the loop-seal of each of the four loops for the most limiting break (3.25-inch break case). Enclosures 2 and 3 provide the table and figures associated with this response.

NRC RAI Question 9

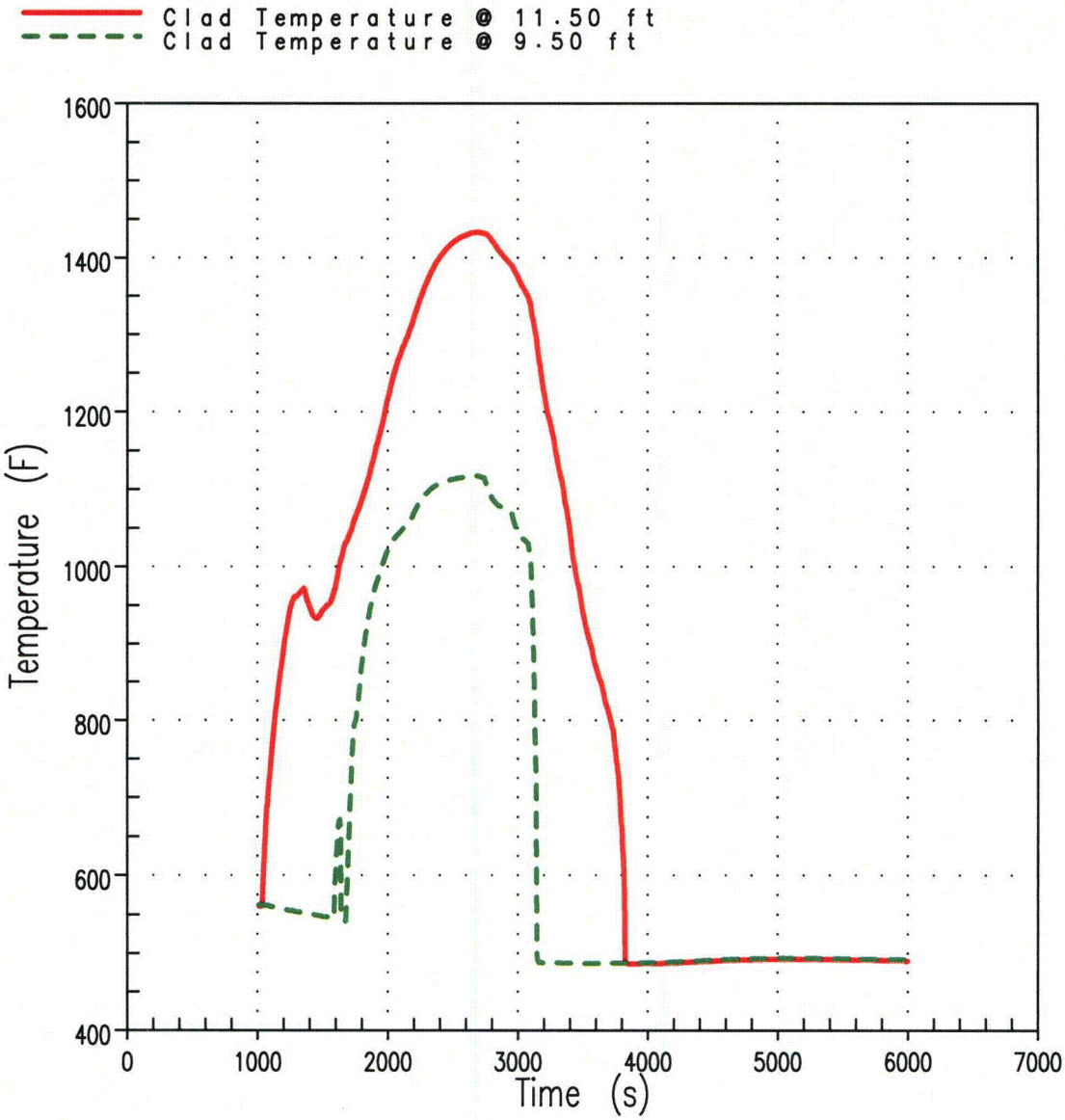
Please explain why the hot spot location at 9.5 feet does not produce the highest PCT and rupture location, since Table 5 identifies the PCT elevation as 11.75 ft, which has a local linear power level of about 6 kw/ft versus 12.5 kw/ft at 9.5 ft.

Response to RAI Question 9

For the 3.25-inch break case the hot rod PCT occurs at 1480 seconds when the mixture level is about 17.6 feet, (approximately 7.6 feet above the bottom of core). The core region above the quench front is exposed to vapor, with higher vapor temperatures at higher elevations. Therefore, the higher the elevation from the core mixture level the larger the integrated core heat up rate, due to less heat being removed at these elevations. This, in turn, results in higher clad temperatures at the higher elevations. This is illustrated in Figure 9-1, where the hot rod clad temperatures at 11.5 feet are higher than at 9.5 feet, indicating that the PCT elevation is more a function of vapor temperature and is less affected by the local linear power level at that elevation.

Figure 9-1

Clad Temperatures at 11.5 ft vs 9.5 ft
(2.5-inch break)



NRC RAI Question 10

Upon drainage of the refueling water storage tank [RWST], are HPSI pump flows also interrupted for 5 minutes? What is the impact of the 5-minute residual heat removal delay on the 8.75-inch break? What is the impact on the limiting large-break LOCA? Please explain.

Response to RAI Question 10

The SI and charging pumps continue to draw off of the RWST during the manual switchover of the RHR system from the cold leg injection alignment to the cold leg recirculation alignment. The manual evolution involves an RHR flow interruption of less than or equal to five minutes. Subsequently, the discharge flow of the RHR pumps is aligned to the suction-side of the SI and charging pumps, which causes the check valves in the RWST line to seat. Thus, the suction source for the SI and charging pumps is realigned to the containment sump without an interruption in SI or charging flow.

The SBLOCA analysis considered the five-minute RHR flow interruption, as described in the enclosure to Reference 1 under the subheading "Switchover from ECCS Injection Phase to ECCS Recirculation Phase." The limiting PCT case, which was the 3.25-inch break, as well as all other break sizes except for the 8.75-inch break case, have calculated RCS pressures that remained above the RHR cut-in pressure. As such, the five-minute RHR flow interruption has no effect on these analysis cases.

The 8.75-inch break case calculated RCS pressures that dropped below the RHR cut-in pressure and, thus, were affected by the RHR flow interruption. The analysis for the 8.75-inch break scenario was comprised of several runs to consider the ECCS flow rates delivered to the RCS at various times during the event. It was concluded that upon re-establishing RHR flow the RCS inventory recovery was such that no core uncovery occurred during the event. However, as described within the response to RAI Question 2, the ECCS flow rates assumed after the five-minute interruption improperly reflect RHR flow rates based upon its cross-tie valves being open. As such, the RCS inventory recovery rate for the 8.75-inch break case is erroneously high. The case is in the process of being re-analyzed and will be submitted to the NRC by June 30, 2008.

The impact of the 5-minute RHR flow interruption on large break LOCA PCT results were addressed in Attachment 10 to Reference 5 as part of supporting information for a Technical Specification change request approved by Reference 6.

NRC RAI Question 11

Were time step studies performed on the limiting small break? Please explain and provide the results of the time step study.

Response to RAI Question 11

During a telephone conference call between I&M and NRC staff on July 20, 2007, NRC staff stated that the core mixture level presented within Figure 22 appears to be unstable and that

smaller time steps may be required. NRC staff further stated that it is not necessary to provide a time step study. Figure 22, which presents core mixture level as a function of time for the 2.5-inch break case, is discussed in the response to RAI Question 6. The response to RAI Question 6 discusses the froth up behavior and apparent abrupt deviations in level which could be suspected as instabilities. Figure 6-4 of this enclosure provides the data included in Figure 22 along with cladding temperature versus time.

NRC RAI Question 12

Were any modifications made to the ECCS licensing models subsequent to the latest Nuclear Regulatory Commission approval and applied to the D.C. Cook SBLOCA analyses? Please identify any changes.

Response to RAI Question 12

No changes have been made to the NOTRUMP EM codes other than those reported through the 10 CFR 50.46 process. The 10 CFR 50.46 annual reporting describes error corrections and forward fit enhancements performed to these evaluation models. In the instance of error corrections, the impact for each plant is provided to the licensee by Westinghouse and Westinghouse provides both the NRC and the licensee a description of the correction/change and an estimated effect on the evaluation model. There have been no corrections or changes impacting the reanalysis provided by Reference 1.

References

1. Letter from M. A. Peifer, I&M, to U. S. NRC Document Control Desk, "Small Break Loss-of-Coolant Accident Evaluation Model Reanalysis," AEP:NRC:7046, dated March 29, 2007 (ML071000431).
2. Electronic mail message from P. S. Tam, NRC, to M. K. Scarpello, I&M, et. al., "D. C. Cook Unit 1 – Draft RAI re: SBLOCA Reanalysis (TAC MD5297)," dated May 31, 2007 (ML071510338).
3. Letter from P. S. Tam, NRC, to M. K. Nazar, I&M, "D. C. Cook Nuclear Plant Unit 1 (DCCNP-1) – Request for Additional Information Regarding Re-analysis of Small-Break Loss-of-Coolant Accident (TAC MD5297)," dated August 10, 2007 (ML072050570).
4. Letter from J. F. Stang, NRC, to A. C. Bakken III, I&M, "Donald C. Cook Nuclear Plant, Unit 1 – Issuance of Amendment 273 Regarding Measurement Uncertainty Recapture Power Uprate (TAC No. MB5498)," dated December 20, 2002 (ML023470126).
5. Letter from R. P. Powers, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Technical Specification Change Request, Containment Recirculation Sump Water Inventory," submittal C1099-08, dated October 1, 1999.
6. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Issuance of Amendments - Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6766 and MA6767)," dated December 13, 1999 (ML993570158).

7. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985.
8. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," June 1974.
9. Letter from H. A. Sepp, Westinghouse, to NRC Document Control Desk, "NRC Report for NOTRUMP Version 38.0 Changes (Proprietary), NRC Report for NOTRUMP Version 38.0 Changes (Non-Proprietary)," letter number NSBU-NRC-00-5972, dated June 30, 2000 (ML003731835 and ML003731862).

Enclosure 3 to AEP:NRC:8046

Table 8-1 and Figures 8-1 through 8-4 **(Non-Proprietary)**

a,c

[ ]

Figure 8-1 – Loop 1 (Broken Loop) Pump Suction Cross-Over Leg Mixture Level

3.25 inch Break

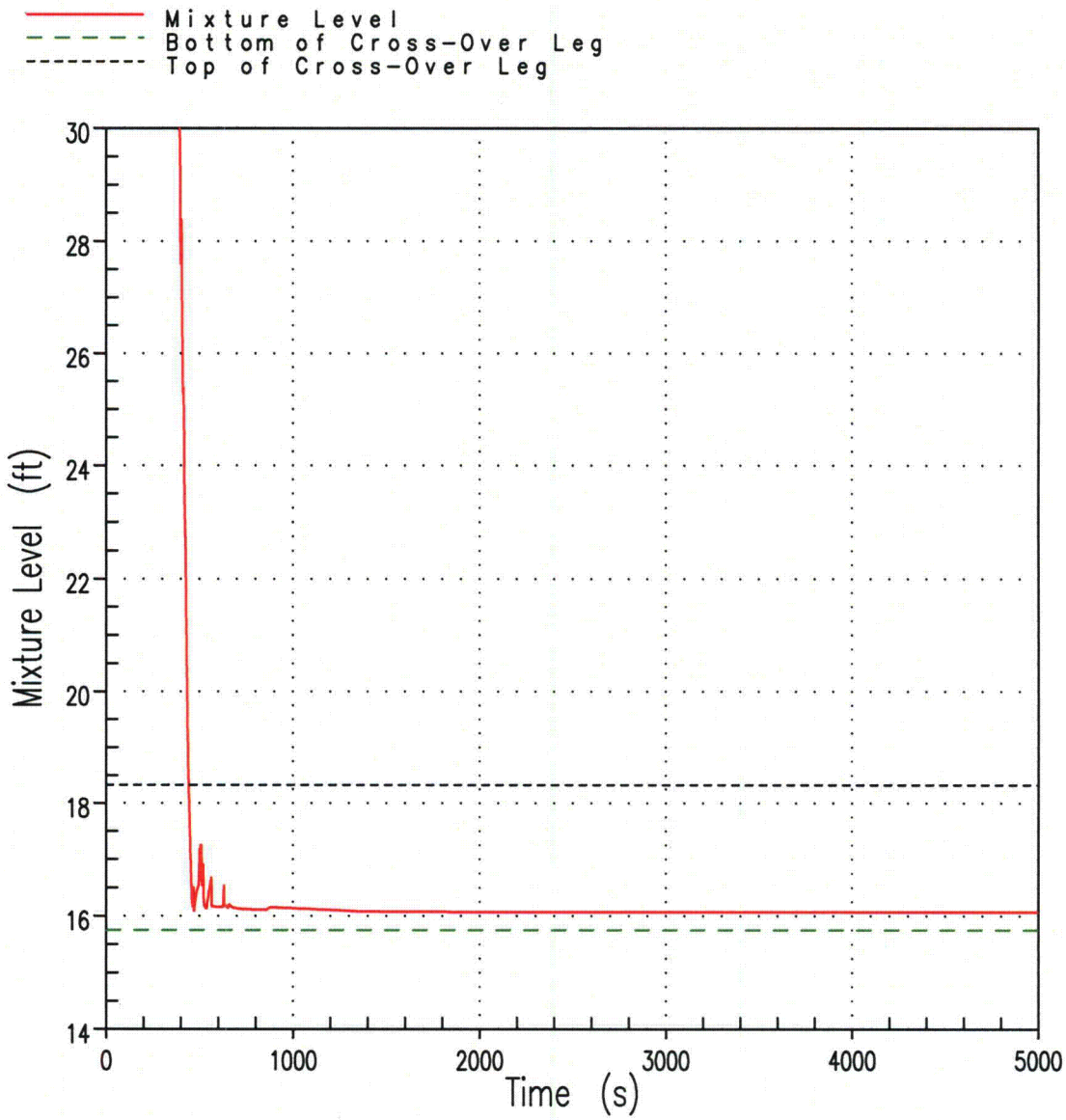


Figure 8-2 – Loop 2 (Intact Loop) Pump Suction Cross-Over Leg Mixture Level
3.25 inch Break

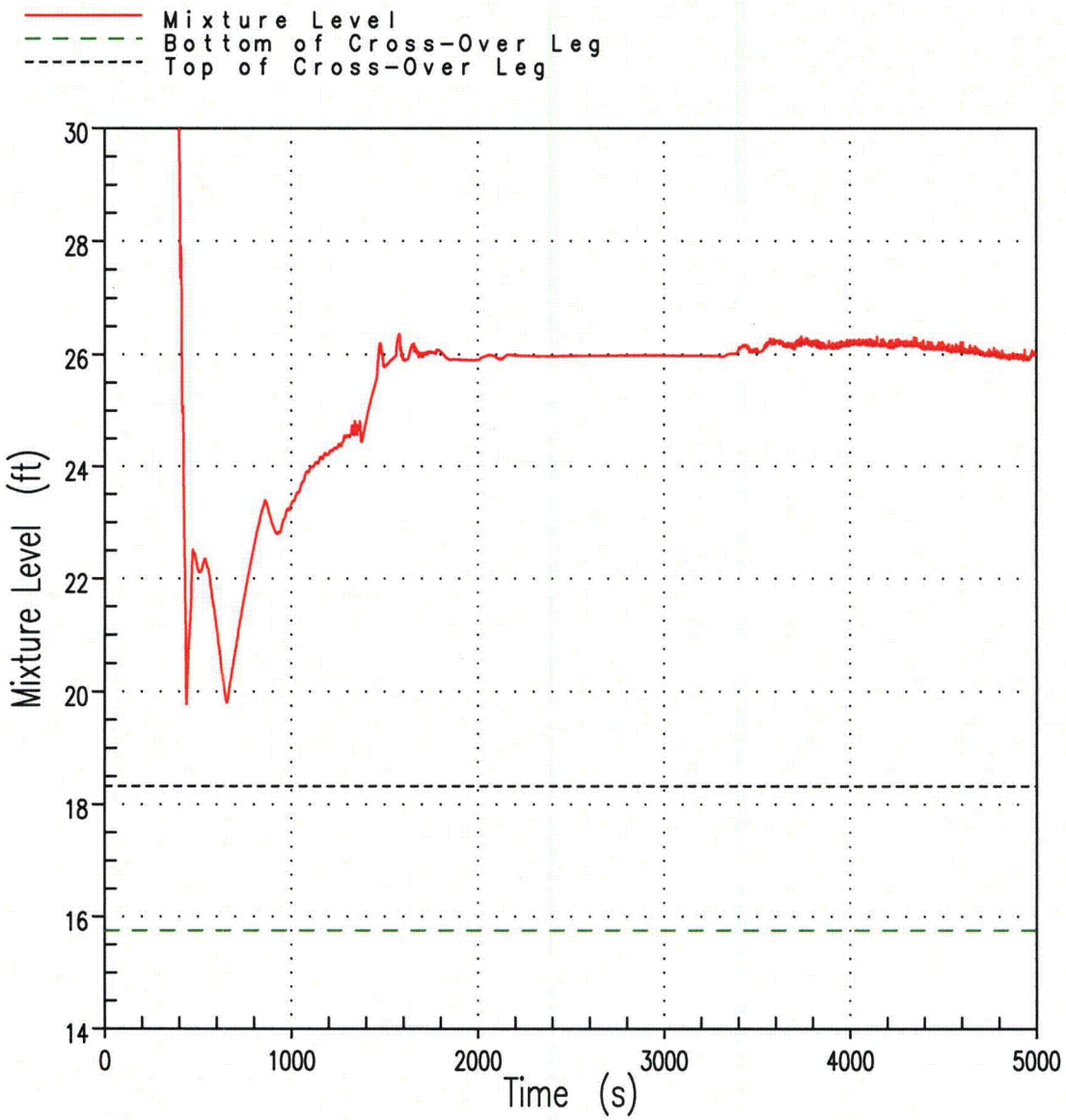


Figure 8-3 – Loop 3 (Intact Loop) Pump Suction Cross-Over Leg Mixture Level

3.25 inch Break

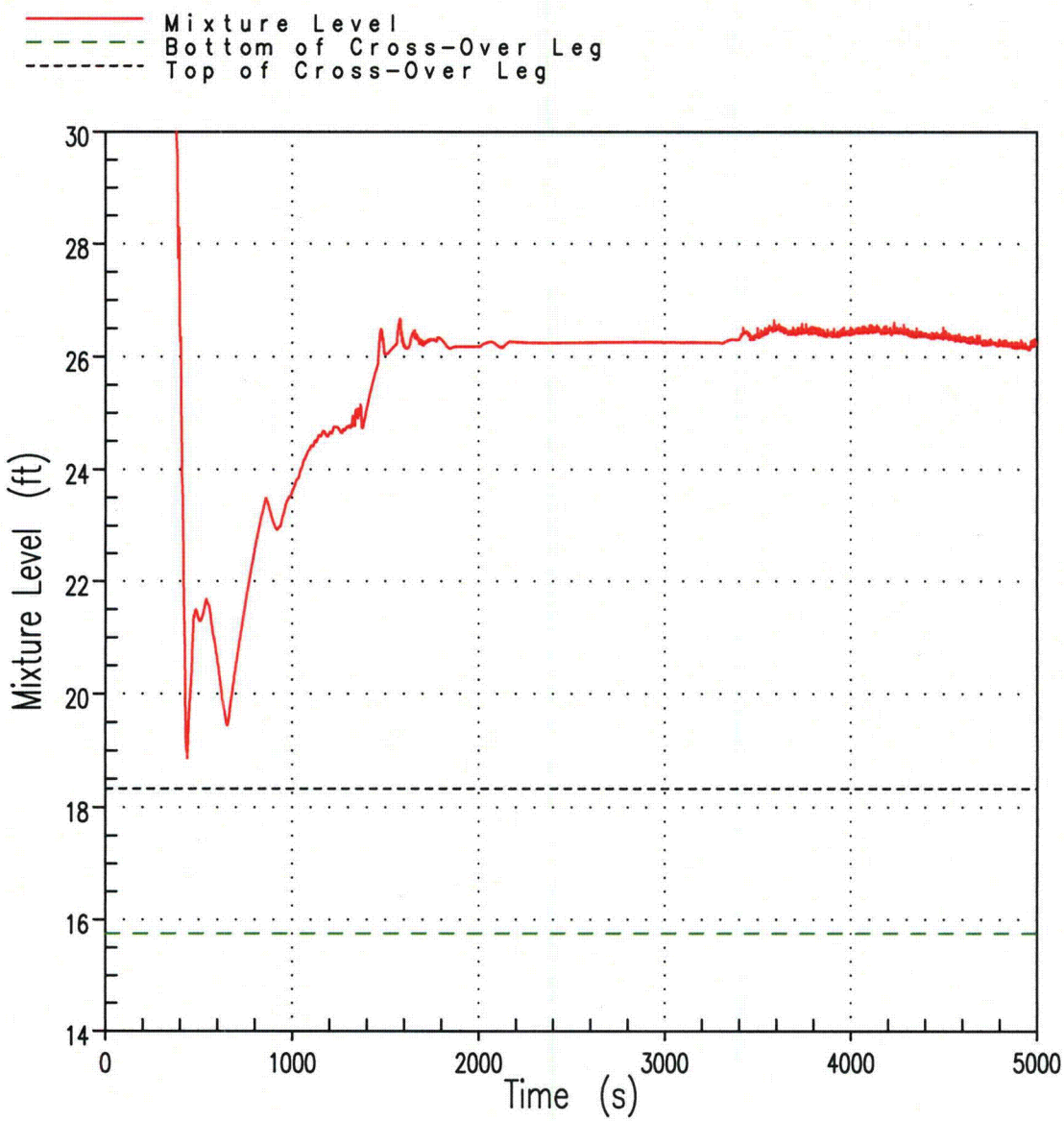
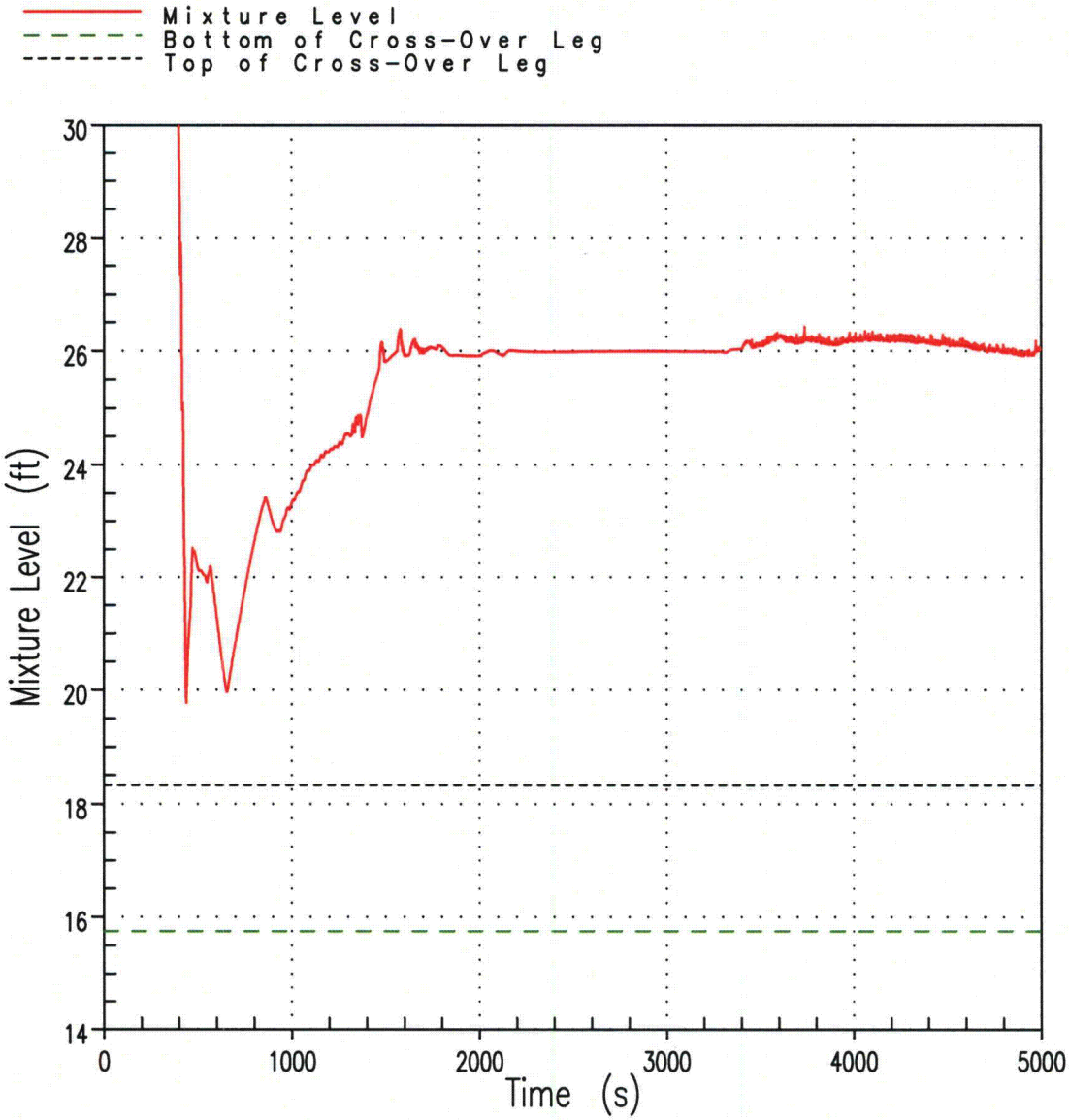


Figure 8-4 – Loop 4 (Intact Loop) Pump Suction Cross-Over Leg Mixture Level

3.25 inch Break



Attachment 1 to AEP:NRC:8046

REGULATORY COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this document. Any other actions discussed in this submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
I&M will provide information on the Unit 1 Small Break Loss-of-Coolant Accident analysis 8.75-inch case using the corrected flow rates.	June 30, 2008

Attachment 2 to AEP:NRC:8046

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Westinghouse

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Proj letter ref NF-AE-08-34

Our ref: CAW-08-2392

February 29, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "SBLOCA Phase 1 Reanalysis – Table and Figures for Response to the NRC Requests for Additional Information Question 8," (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-08-2392 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by American Electric Power.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2392 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Jon Thompson (NRC O-7E1A)

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 29th day of February, 2008

Sharon L. Markle
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's

competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "SBLOCA Phase 1 Reanalysis – Table and Figures for Response to the NRC Requests for Additional Information Question 8" (Proprietary) for submittal to the Commission, being transmitted by American Electric Power letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the NRC review of the D. C. Cook Unit 1 Small Break LOCA Phase 1 Reanalysis.

This information is part of that which will enable Westinghouse to:

(a) Assist the customer in obtaining NRC review of the D. C. Cook Unit 1 Small Break LOCA Phase 1 Reanalysis.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of this information to its customers for purposes of plant specific LOCA analysis for licensing basis applications

(b) It's use by a competitor would improve their competitive position in the design and licensing of a similar product for SBLOCA analyses.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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