Indiana Michigan
Power Company
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
616-465-5901

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INDIANA
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U. S. Nuclear Regulatory Commission
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Donald C. Cook Nuclear Plant Unit 1
ANNUAL REPORT OF LOSS-OF-COOLANT ACCIDENT
EVALUATION MODEL CHANGES
AND SUBMITTAL OF NEW LARGE BREAK LOSS-OF-COOLANT
ACCIDENT ANALYSIS OF RECORD FOR UNIT 1

Pursuant to 10 CFR 50.46, Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP), is submitting an annual report of loss-of-coolant accident (LOCA) model changes affecting the peak cladding temperature (PCT) for Unit 1. This submittal includes a new large break LOCA (LBLOCA) analysis of record provided for NRC information.

Attachment 1 to this letter describes the background for the new LBLOCA analysis of record and describes assessments against the small break LOCA analysis of record. Attachment 2 provides the analysis report for the new LBLOCA analysis of record. Attachment 3 provides PCT margin utilization tables for large break and small break LOCAs showing that the calculated PCTs remain within the limit specified in 10 CFR 50.46. Attachment 4 contains a summary of new I&M commitments made in this submittal.

Should you have any questions, please contact Mr. Robert C. Godley, Director of Regulatory Affairs, at (616) 466-2698.

Sincerely,

M. W. Rencheck
Vice President, Engineering

/dms

Attachments

A001

Printed on recycled paper

c: J. E. Dyer
MDEQ – DW & RPD, w/o attachments
NRC Resident Inspector
R. Whale, w/o attachments

ATTACHMENT 1 TO C0300-07

BACKGROUND FOR NEW LARGE BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS OF RECORD AND ASSESSMENTS AGAINST THE SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS OF RECORD

New Large Break Loss-of-Coolant Accident Analysis of Record

Indiana Michigan Power Company's (I&M's) most recent annual 10 CFR 50.46 submittal for Donald C. Cook Nuclear Plant (CNP) Unit 1 (Reference 1) provided two peak cladding temperature (PCT) margin utilization tables for the large break loss-of-coolant accident (LBLOCA) analyses of record. NRC review of these analyses is documented in the Safety Evaluation Report (SER) associated with Reference 2. The first PCT margin utilization table presented the PCT resulting from the limiting LBLOCA analysis that modeled a symmetric chopped cosine power distribution. In that analysis, skewed power shapes were addressed by the power shape sensitivity model (PSSM) methodology. It was determined that the symmetric power distribution case remained limiting. The PSSM methodology was withdrawn from NRC review shortly after the first analysis was completed, but before the NRC issued the Reference 2 SER.

The second PCT margin utilization table reflected the application of the explicit shape analysis for PCT effects (ESHAPE) methodology. To compensate for the PCT penalty that resulted from the ESHAPE methodology, a benefit was applied for the incorporation of the hot leg nozzle gap (HLNG) into the BASH methodology. However, the NRC staff has not accepted loss-of-coolant accident (LOCA) emergency core cooling system evaluation models that credit this HLNG effect. Accordingly, I&M is replacing both LBLOCA analyses of record identified in Reference 1 with an analysis that uses the ESHAPE methodology, without credit for the HLNG. Since the analysis continues to use the previously approved BASH computer model and since the NRC has approved replacement of PSSM with ESHAPE (Reference 2), the new analysis of record is provided in Attachment 2 for NRC information. Attachment 2 is a non-proprietary report, prepared by Westinghouse Electric Company.

The new LBLOCA analysis of record also incorporates a change in the fuel assembly design to be used in the next Unit 1 operating cycle (Cycle 17). The fuel used for previous Unit 1 operating cycles, and assumed in previous LBLOCA analyses, was Westinghouse 15x15 Optimized Fuel Assembly (OFA) fuel. I&M is transitioning to a more advanced Westinghouse fuel assembly design which modifies the OFA fuel assembly design currently used at CNP by replacing the Zircaloy-4 cladding with ZIRLO cladding. With the design features of the fuel assemblies to be used in the Unit 1 core, including ZIRLO cladding, this revised fuel design is referred to as VANTAGE+, or by the Westinghouse marketing label of Performance+. Reference 3 approved a change to the Technical Specifications (T/S) to allow the use of this new fuel assembly design. The Unit 1 Cycle 17 core will include both OFA fuel assemblies and VANTAGE+ fuel with ZIRLO cladding and intermediate flow mixer grids. The new LBLOCA analysis of record provided in Attachment 2 supports this change.

By Reference 4, I&M reported errors in the LOCBART computer code used in the LBLOCA analysis for fuel rod heat-up calculations. Pursuant to 10 CFR 50.46(a)(3)(ii), the estimated effect of these errors was included in the notification to the NRC. The reported effects of these errors included PCT changes resulting from the spacer grid heat transfer error (13°F PCT benefit) and the metal-water oxidation error (39°F PCT penalty), for a net PCT penalty of 26°F. These values were derived from plant-specific re-analyses, assuming the use of OFA fuel. As noted above, the new LBLOCA analysis of record, provided in Attachment 2, assumes VANTAGE+ fuel with ZIRLO cladding (i.e., Performance+). This new analysis of record results in a 14°F penalty from the spacer grid heat transfer error and a 44°F penalty from the metal-water oxidation error. To help offset the overall 58°F PCT penalty, margin was obtained by assuming a steam generator tube plugging level of 25%, vice the currently assumed 30% level. This resulted in a 29°F PCT benefit. I&M is replacing the Unit 1 steam generators and will revise the LBLOCA analysis to reflect the effect of the replacement steam generators prior to Unit 1 start-up. The revised LBLOCA analysis will assume a lower tube plugging limit (i.e., fewer than 25%) for the replacement steam generators. The average Unit 1 steam generator tube plugging level prior to replacing the steam generators during current outage was fewer than 17% (Reference 5). The effects of these assessments on the calculated PCT are shown in the PCT margin utilization table (Attachment 3, Table 1) for the new LBLOCA analysis of record.

In an amendment request involving the containment sump water inventory (Reference 6), several issues were identified as potentially affecting LOCA analyses. As documented in Section 3.5.4 of Attachment 10 of Reference 6, one of these issues consisted of shortening the delay in CEQ fan start time. The new analysis of record assumes a value for the delay in starting the containment air recirculation system (CEQ) fans following a LOCA. The effect of this shortened delay was evaluated further and it was determined that it would have a negligible effect on the PCT determined by the new analysis of record. In Reference 7, the NRC approved the amendment and noted in the SER that analyses had demonstrated that licensing criteria such as that of 10 CFR 50.46 remained satisfied.

This submittal satisfies I&M's commitment, made in Reference 4, to submit PCT margin utilization tables addressing the LBLOCA model errors identified in that reference.

Assessments Against the Small Break Loss-of-Coolant Accident Analyses of Record

By Reference 4, I&M also notified the NRC of an error in the small break LOCA (SBLOCA) analysis of record. The error involved the inability of the version of the NOTRUMP model used for the analysis of record to evaluate an asymmetric safety injection (SI) system delivery that could result if the two SI pump discharge lines were not cross-tied. Operation with the SI pump discharge lines not cross-tied occurs during quarterly T/S required surveillance testing.

As reported in Reference 4, a CNP-specific, four-loop version of the NOTRUMP model was used to estimate the effect on PCT that would result from asymmetric SI delivery during a SBLOCA. The estimated effect was determined to be a penalty of 50°F. Attachment 3 to this letter includes a

PCT margin utilization table (Table 2) showing assessments against the SBLOCA analysis of record, using the CNP-specific four-loop version of the NOTRUMP model at a power level of 3250 MWt. The margin utilization table includes the estimated 50°F penalty resulting from asymmetric SI delivery.

The current Unit 1 SBLOCA analysis of record assumes the use of Westinghouse 15x15 OFA fuel. I&M will assess the effects of steam generator replacement on the SBLOCA analysis. This assessment, to be completed prior to Unit 1 start-up, will also address the change in fuel assembly design to VANTAGE+ fuel, as approved by Reference 3.

This submittal completes I&M's commitment, made in Reference 4, to submit PCT margin utilization tables addressing the SBLOCA model errors identified in that reference, including a quantified estimate of the burst and blockage/time-in-life penalty. The SBLOCA burst and blockage/time-in-life assessment is a function of base PCT plus permanent margin allocation (i.e., net PCT). This penalty need only be applied for cases for which the net PCT exceeds 1700°F. Because the net PCT for the Unit 1 SBLOCA is fewer than 1700°F, the burst and blockage/time-in-life PCT penalty is not applicable for this case.

Summary

The PCT margin utilization tables provided in Attachment 3 show that compliance with the PCT limit of 2200°F specified in 10 CFR 50.46(b)(1) is maintained for the current LBLOCA and SBLOCA analyses of record for Unit 1.

REFERENCES

1. Letter from J. R. Sampson, I&M, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2, Annual Report of LOCA Evaluation Model Changes," submittal AEP:NRC:1118M, dated June 3, 1998.

2. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant Units Nos. 1 and 2 – Issuance of Amendments Re: Increased Steam Generator Plugging Limit (TAC Nos. M92587 and M92588)," dated March 13, 1997.

3. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2, Re: Fuel Rod ZIRLO Cladding and Integral Fuel Burnable Absorber Requirements (TAC Nos. MA7041 and MA7042)," dated January 6, 2000.

4. Letter from M. W. Rencheck, I&M, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2, Errors In Loss-Of-Coolant-Accident Evaluation Models," submittal C1299-04, dated December 9, 1999.

5. Letter from J. R. Sampson, I&M, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2, 1997, Annual Operating Report," submittal AEP:NRC:1147H, dated May 14, 1998.

6. Letter from R. P. Powers, I&M, to U. S. Nuclear Regulatory Commission, "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.

7. 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.

ATTACHMENT 2 TO C0300-07

NEW ANALYSIS OF RECORD FOR LARGE BREAK LOSS-OF-COOLANT ACCIDENT

DONALD C. COOK NUCLEAR PLANT, UNIT 1

(Non-Proprietary Report prepared by Westinghouse Electric Corporation)

D. C. COOK NUCLEAR PLANT UNIT 1

LARGE BREAK LOCA ANALYSIS

FOR CHANGE TO ZIRLO™ CLAD AND IFM GRIDS

D. C. Cook Nuclear Plant Unit 1
Large Break LOCA Analysis
for Change to ZIRLO™ Clad and IFM Grids

Introduction

The fuel currently in Donald C. Cook Nuclear Plant Unit 1 is 15x15 OFA fuel, and it is planned to change to 15x15 Performance+ fuel with ZIRLO™ clad and Intermediate Flow Mixer (IFM) grids for Cycle 17. A large break LOCA analysis has been performed to support the change to fuel with ZIRLO™ clad and IFM grids, and the results of the analysis are presented in this report.

The current licensing basis large break LOCA analysis for Donald C. Cook Nuclear Plant Unit 1 was performed to support an increase in the steam generator tube plugging (SGTP) level to 30%, and the analysis results were presented in WCAP-14285 (Reference 1). The large break LOCA analysis for 30% SGTP was implemented for Cycle 16 operation, and has been incorporated into Section 14.3.1 of the Unit 1 UFSAR. The analysis for 30% SGTP was used as the basis for performing the additional large break LOCA analyses to support Cycle 17 operation with the change to ZIRLO™ clad and IFM grids. Since the large break LOCA analysis for ZIRLO™ clad and IFM grids is an extension of the 30% SGTP analysis, a summary of the 30% SGTP analysis is presented in the following section. A detailed description of the additional large break LOCA analyses performed for ZIRLO™ clad and IFM grids is then presented in a subsequent section.

Large Break LOCA Analysis for 30% SGTP

The large break LOCA analysis for the 30% SGTP program was performed with the 1981 version of the Westinghouse ECCS Evaluation Model using the BASH computer code (Reference 2). The analysis was performed for plant operation at 3250 MWt with 15x15 OFA fuel using a total core peaking factor (F_Q) of 2.15, a hot channel enthalpy rise factor ($F_{\Delta H}$) of 1.55, and a hot assembly average power (P_{HA}) of 1.38. The analysis conditions were based on a total RCS thermal design flow of 332,800 gpm which is consistent with 30% SGTP. The analysis was performed for a range of operating temperatures in order to justify plant operation between 609.1°F and 586.8°F in the hot legs and 543.5°F and 519.2°F in the cold legs. These temperatures correspond to a T_{AVG} operating range between 576.3°F and 553.0°F. Initial RCS pressure was also varied to justify plant operation at 2100 and 2250 psia. An RCS pressure uncertainty of 67 psia was used for the analysis, which resulted in initial RCS pressures of 2033

and 2317 psia, respectively.

The safety injection flows used for the large break LOCA analysis were based on operation with the RHR cross-tie valves closed. The safety injection flow rates with the RHR cross-tie valves closed also reflected a degradation of 15% in the pump performance curves for the high head safety injection and RHR pumps. For the centrifugal charging pumps, a 10% pump head degradation and a 25 gpm flow imbalance were assumed. The diesel generator start time was assumed to be 30 seconds, which results in a total delay of 47 seconds from the time of safety injection actuation until pumped safety injection flow is initiated.

A full break spectrum analysis was performed for nominal RCS conditions (hot leg temperature of 609.1°F and initial RCS pressure of 2250 psia) with minimum safety injection flow rates, which indicated that the limiting break is the 0.4 C_D break. The limiting 0.4 C_D break was then reanalyzed for the reduced hot leg temperature of 586.8°F and nominal RCS pressure of 2250 psia. The limiting 0.4 C_D break was also reanalyzed for the nominal hot leg temperature of 609.1°F and RCS pressure of 2100 psia. The results of these analyses indicated that the limiting operating condition is the nominal hot leg temperature of 609.1°F and a pressure of 2100 psia, and a peak clad temperature (PCT) of 2164°F was calculated for this case. An analysis for the limiting 0.4 C_D break and limiting operating conditions with maximum safety injection demonstrated that maximum safety injection is not limiting for the large break LOCA analysis.

The large break LOCA analysis for the 30% SGTP program was performed using a cosine power shape, and it was intended to use the Westinghouse Power Shape Sensitivity Model (PSSM) to demonstrate that the cosine shape remained the limiting power distribution for each operating cycle. However, after the analysis was completed and submitted to the NRC, the PSSM methodology was replaced by the Westinghouse ESHAPE methodology which requires an explicit analysis with a set of skewed power shapes. The application of the ESHAPE methodology resulted in a PCT penalty, and thus a compensatory benefit for the incorporation of the hot leg nozzle gap into the BASH Methodology was used to offset the PCT penalty. The net result was that the PCT for the limiting skewed power shape with the application of the hot leg nozzle gap, was less than the PCT calculated for the cosine power shape without the hot leg nozzle gap. The effect of the hot leg nozzle gap would not reduce the PCT significantly for the cosine power shape, and thus it was concluded that the PCT for the cosine power shape would remain limiting. It is noted that the hot leg nozzle gap was not required for the cosine shape analysis, and was only used to demonstrate that the cosine shape remained limiting. The large

break LOCA analysis for 30% was approved by the NRC on this basis, with a commitment to update the analysis using the ESHAPE methodology prior to Cycle 18.

Large Break LOCA Analysis for ZIRLO™ Clad and IFM Grids

The large break LOCA analysis for the 30% SGTP program has been extended to support a change from the 15x15 OFA fuel to 15x15 Performance+ fuel with ZIRLO™ clad and IFM grids. The large break LOCA analysis for ZIRLO™ and IFMs was also performed with the 1981 version of the Westinghouse ECCS Evaluation Model using BASH. The analysis assumptions and input parameters that were used for the large break LOCA analysis for the 30% SGTP program have also been used for the analysis to support the fuel upgrade to 15x15 Performance+ fuel. A list of the key plant input parameters used for the fuel upgrade large break LOCA analysis extension is provided in Table 1, which are the same as those used for the 30% SGTP analysis with the exception of the fuel type. The Westinghouse ESHAPE methodology has also been explicitly used to evaluate skewed power shapes for the fuel upgrade analysis to eliminate any need to credit the hot leg nozzle gap for skewed power shapes.

As noted previously, the limiting case for the 30% SGTP analysis was the 0.4 C_D break with a nominal hot leg temperature of 609.1°F and a pressure of 2100 psia with minimum SI flow rates. Since previous analysis experience has shown that the limiting break size and the sensitivity to maximum SI flow rates are not affected for a fuel upgrade for ZIRLO™ clad and IFM grids, the limiting 0.4 C_D break with minimum SI flow rates was used as the basis for the large break analysis for the change to ZIRLO™ and IFMs. Analyses were performed for the range of operating temperatures at pressures of 2100 and 2250 psia used for the 30% SGTP analysis to ensure that the current plant operating flexibility is maintained. Analyses were performed for both the 275 psig non-IFBA fuel and the 150 psig IFBA fuel with ZIRLO™ and IFMs to ensure that the analysis results are applicable for both fuels.

The initial large break LOCA analysis was performed for the new 275 psig non-IFBA fuel with ZIRLO™ and IFMs for the limiting 0.4 C_D break with a hot leg temperature of 609.1°F and a pressure of 2100 psia using a cosine power shape. Analyses were then performed for the other operating conditions (hot leg temperature of 609.1°F with a pressure of 2250 psia, and hot leg temperature of 586.8°F with a pressure of 2250 psia) using a cosine power shape. The results of the analyses and the sequence of events for the different operating conditions (Cases 1, 2, and 3) are presented in Table 2. These results demonstrate that the limiting condition with a hot leg

temperature of 609.1°F and a pressure of 2100 psia established in the 30% SGTP analysis will remain limiting for Cycle 17 operation with the new fuel with ZIRLO™ and IFMs.

Analyses were then performed for skewed power shapes for the new 275 psig non-IFBA fuel with ZIRLO™ and IFMs for the limiting 0.4 C_D break and operating conditions (hot leg temperature of 609.1°F and pressure of 2100 psia) using the ESHAPE methodology. Analyses were performed for skewed power shapes with peaks at elevations of 8.5 ft, 9.5 ft, and 10.5 ft, and the results and sequence of events for these analyses (Cases 4, 5, and 6) are also presented in Table 2. The results in Table 2 show that the highest PCT was obtained for the 8.5 ft peak skewed power shape. The initial skewed power shape analysis results indicated that the 10.5 ft peak shape would be the most limiting power shape for BOL conditions, but it was determined that shapes with peaks above 10.0 ft typically do not occur until after 4000 MWD/MTU burnup. Therefore, an analysis was performed for the 10.5 ft peak shape at 4000 MWD/MTU burnup, and the results for this case in Table 2 indicate that the 10.5 ft peak shape will not be limiting after 4000 MWD/MTU. A restriction will be placed on the core design to ensure that 10 ft peak power shapes will be precluded prior to a burnup of 4000 MWD/MTU for Unit 1, Cycle 17. On this basis, it is concluded that the 8.5 ft peak shape will be the limiting power shape for Unit 1, Cycle 17 operation with the new fuel with ZIRLO™ and IFMs.

An analysis was also performed for the new 150 psig IFBA fuel with ZIRLO™, IFMs, and annular blankets for the limiting 0.4 C_D break, operating conditions, and 8.5 ft peak shape which were established from the analysis for the 275 psig non-IFBA fuel. The results and sequence of events for the 150 psig IFBA fuel (Case 7) are presented in Table 3. A comparison of the limiting results for the 275 psig non-IFBA fuel in Table 2 (Case 4) with the results in Table 3 shows that the PCT of 2133°F for the 150 psig IFBA fuel is limiting. Therefore, it is concluded that the 150 psig IFBA fuel is limiting for the large break LOCA analysis for Unit 1 Cycle 17 operation with ZIRLO™ and IFMs. The transient results for selected parameters are presented in the attached Figures 1 through 13 for the limiting Case 7 analysis.

An analysis was also performed for the once-burned OFA fuel with Zirc-4 clad and no IFM grids from Cycle 16 to demonstrate that the partially burned Cycle 16 fuel will be less limiting than the new fuel with ZIRLO™ and IFMs. The analysis was performed for the limiting 275 psig non-IFBA fuel, at a minimum Cycle 16 burnup of 8000 MWD/MTU. Analyses were performed for the limiting 0.4 C_D break and operating conditions (hot leg temperature of 609.1°F and pressure of 2100 psia) using the ESHAPE methodology without taking credit for the hot leg nozzle gap.

The results indicated that the 8.5 ft peak skewed shape is also the limiting power shape for the once burned Cycle 16 fuel. The results for this analysis indicated that the Cycle 16 fuel with a minimum burnup of 8000 MWD/MTU will be less limiting than the new 150 psig IFBA fuel with ZIRLO™ and IFMs.

The new fuel assemblies with IFM grids use low pressure drop (LPD) structural grids to offset the increased pressure drop for the IFM grids, such that the overall assembly pressure drop for the new fuel with IFMs is not significantly higher than for the existing assemblies. However, the use of the IFM grids does result in local flow diversions from the new assemblies located adjacent to the existing assemblies without IFMs, which can affect the calculated PCT. Therefore, an evaluation was performed to determine the large break LOCA PCT penalty for the transition core due to the diversion of flow from the new fuel assemblies with IFMs. It was determined that the PCT for the new 150 psig IFBA fuel with ZIRLO™ and IFMs would be increased by 31°F, for a net PCT of 2164°F for the transition core period. The limiting PCT of 2133°F for the new 150 psig IFBA fuel with ZIRLO™ and IFMs represents the analysis of record PCT for Unit 1, Cycle 17. A PCT penalty of 31°F for the transition core will be added to the analysis of record, and the resulting PCT of 2164°F will be applicable until a full core of the new fuel with ZIRLO™ and IFMs is obtained.

Conclusions

The large break LOCA analysis results in Table 3 for the new 150 psig IFBA fuel with ZIRLO™ and IFMs show that the PCT is 2133°F, which is less than the 10 CFR 50.46 limit (Reference 3) of 2200°F. With a PCT penalty of 31°F for the transition core, the resulting PCT for the transition core is 2164°F which is still below the 2200°F limit. In addition, the maximum zirconium-water reaction is 9.35% which is less than the 10 CFR 50.46 limit of 17% and the total core zirconium-water reaction is less than the limit of 1%. Furthermore, this analysis was evaluated for a 1999 BASH Evaluation Model error in Reference 4. As demonstrated in the Reference 4 PCT summary, the PCT is 2193°F. Since the PCT and the zirconium-water reaction are less than the 10 CFR 50.46 limits, it is concluded that the fuel with ZIRLO™ and IFMs is acceptable for Donald C. Cook Nuclear Plant Unit 1, Cycle 17 operation.

REFERENCES

1. WCAP-14285, Revision 1, "American Electric Power Service Corporation, Donald C. Cook Nuclear Plant Unit 1, Steam Generator Tube Plugging Program Licensing Report", May 1995.

2. Young, M. Y. et al, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A Rev. 2 (Proprietary), March 1987.

3. "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register 1974, Volume 39, Number 3.

4. AEP-99-396, "American Electric Power Service Corporation, D. C. Cook Units 1 and 2, 10 CFR 50.46 BART/BASH Evaluation Model Mid-Year Notification and Reporting for 1999," November 8, 1999.

TABLE 1

PLANT INPUT PARAMETERS USED IN LARGE BREAK LOCA ANALYSIS FOR FUEL WITH ZIRLO™ AND IFMS

Core Power (MWt)	102% of 3250
Peak Linear Power (kW/ft)	102% of 14.434
Total Core Peaking Factor, F_Q	2.15
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.55
Maximum Assembly Average Power, P	1.38
Fuel Assembly Array	15 X 15 Performance+
Steam Generator Tube Plugging Level (%)	30
Accumulator Water Volume (ft^3/tank)	946
Accumulator Tank Volume (ft^3/tank)	1350
Minimum Accumulator Gas Pressure (psia)	600
Accumulator Water Temperature (°F)	100
Refueling Water Storage Tank Temperature (°F)	70 - 105
Thermal Design Flow Rate (gpm/loop)	83,200
RCS Loop Average Temperature (°F)	553.0 and 576.3
Nominal Initial RCS Pressure (psia)	2100 and 2250
Nominal Steam Pressure (psia)	595 and 749
Safety Injection Delay Time (sec)	47
RHR Pump Head Degradation (%)	15
HHSI Pump Head Degradation (%)	15
Charging Pump Head Degradation (%)	10
Charging Pump Flow Imbalance (gpm)	25
RHR Cross-Tie Valve Position	Closed

TABLE 2
LARGE BREAK LOCA ANALYSIS RESULTS
275 PSIG NON-IFBA FUEL WITH ZIRLO™ AND IFMS

	Case 1 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI Cosine Shape	Case 2 C_D=0.4 T_{HOT}=609.1°F P=2250 psia Min. SI Cosine Shape	Case 3 C_D=0.4 T_{HOT}=586.8°F P=2250 psia Min. SI Cosine Shape	Case 4 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI 8.5 ft Peak Shape	Case 5 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI 9.5 ft Peak Shape	Case 6 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI 10.5 ft Peak Shape(1)
Peak Clad Temperature (°F)	2048	1969	1957	2073	2046	2039
Peak Clad Location (ft)	6.50	8.75	8.75	8.75	8.75	11.00
Local Zr/H_2O Reaction (Max %)	<17.0	<17.0	<17.0	<17.0	<17.0	<17.0
Local Zr/H_2O Location (ft)	6.25	5.50	6.50	8.75	8.75	11.00
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0	<1.0	<1.0	<1.0
Hot Rod Burst Time (s)	41.5	43.0	47.7	42.5	46.1	57.0
Hot Rod Burst Loc. (ft)	6.25	5.50	6.50	7.50	7.75	8.25

(1) At 4000 MWD/MTU burnup

TABLE 2 (Continued)
LARGE BREAK LOCA ANALYSIS TIME SEQUENCE OF EVENTS
275 PSIG NON-IFBA FUEL WITH ZIRLO™ AND IFMS

	Case 1 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI Cosine Shape	Case 2 C_D=0.4 T_{HOT}=609.1°F P=2250 psia Min. SI Cosine Shape	Case 3 C_D=0.4 T_{HOT}=586.8°F P=2250 psia Min. SI Cosine Shape	Case 4 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI 8.5 ft Peak Shape	Case 5 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI 9.5 ft Peak Shape	Case 6 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI 10.5 ft Peak Shape(1)
Start	0.0	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	0.49	0.65	0.56	0.49	0.49	0.49
Safety Injection Signal	4.10	4.90	4.50	4.10	4.10	4.10
Accumulator Injection	18.60	18.40	17.60	18.60	18.60	18.60
End of Blowdown	39.53	39.90	39.99	39.99	39.96	40.06
Pump Injection	51.10	51.90	51.50	51.10	51.10	51.10
Bottom of Core Recovery	53.80	53.90	54.70	53.60	53.50	53.60
Accumulator Empty	67.40	67.50	68.40	67.40	67.40	67.50

(1) At 4000 MWD/MTU burnup

TABLE 3

LARGE BREAK LOCA ANALYSIS RESULTS

150 PSIG IFBA FUEL WITH ZIRLO™ AND IFMS

	Case 7 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI 8.5 ft Peak Shape
Peak Clad Temperature (°F)	2133
Peak Clad Location (ft)	8.75
Local Zr/H_2O Reaction (Max %)	9.35
Local Zr/H_2O Location (ft)	7.50
Total Zr/H_2O Reaction (%)	<1.0
Hot Rod Burst Time (s)	47.3
Hot Rod Burst Loc. (ft)	7.50

TABLE 3 (Continued)
LARGE BREAK LOCA ANALYSIS TIME SEQUENCE OF EVENTS
150 PSIG IFBA FUEL WITH ZIRLO™ AND IFMS

	Case 7 C_D=0.4 T_{HOT}=609.1°F P=2100 psia Min. SI 8.5 ft Peak Shape
Start	0.0
Reactor Trip Signal	0.49
Safety Injection Signal	4.10
Accumulator Injection	18.60
End of Blowdown	39.99
Pump Injection	51.10
Bottom of Core Recovery	53.60
Accumulator Empty	67.40

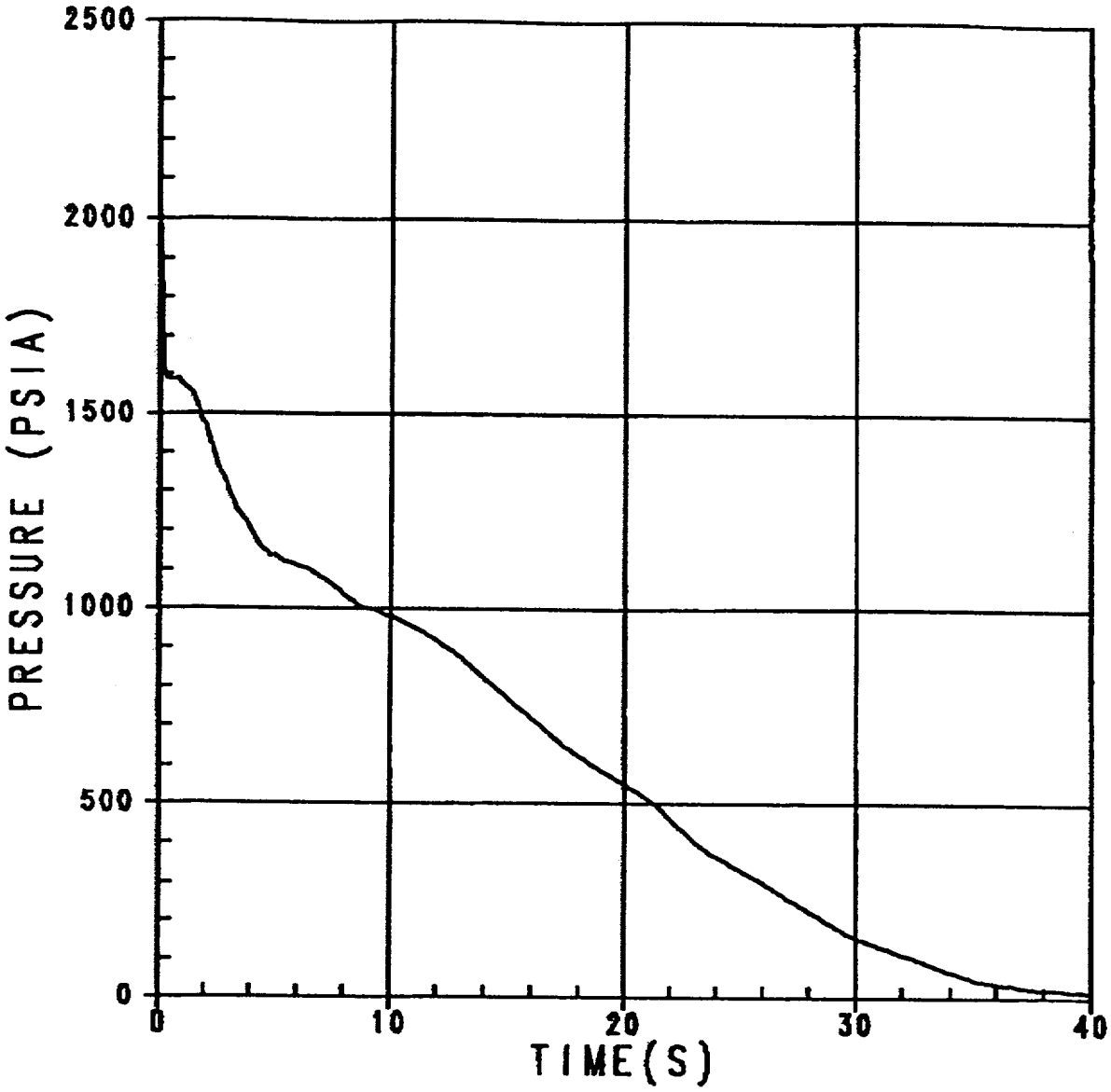

Figure 1 Reactor Coolant System Pressure
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

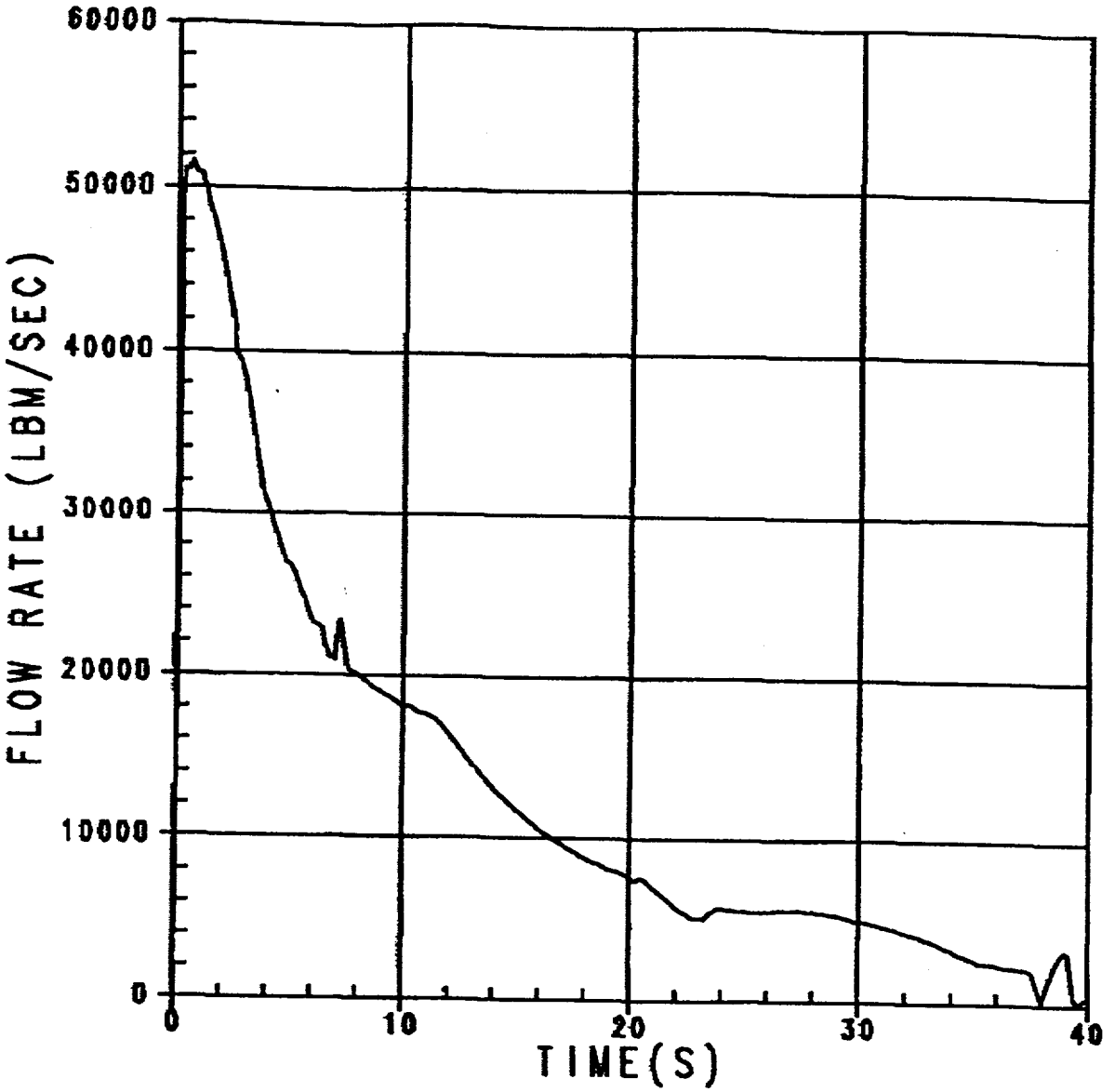

Figure 2 Break Flow During Blowdown
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

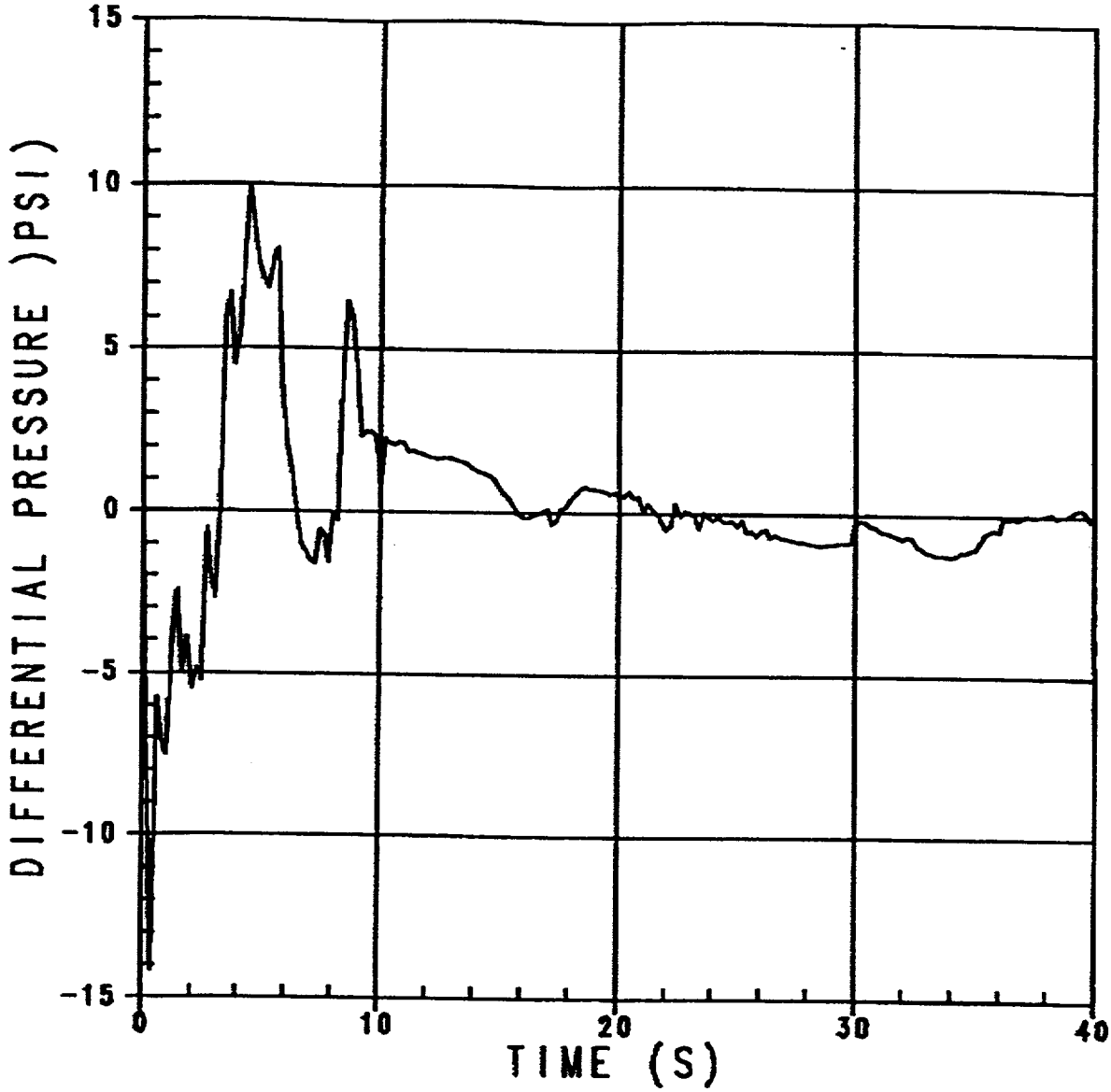

Figure 3 Core Pressure Drop
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

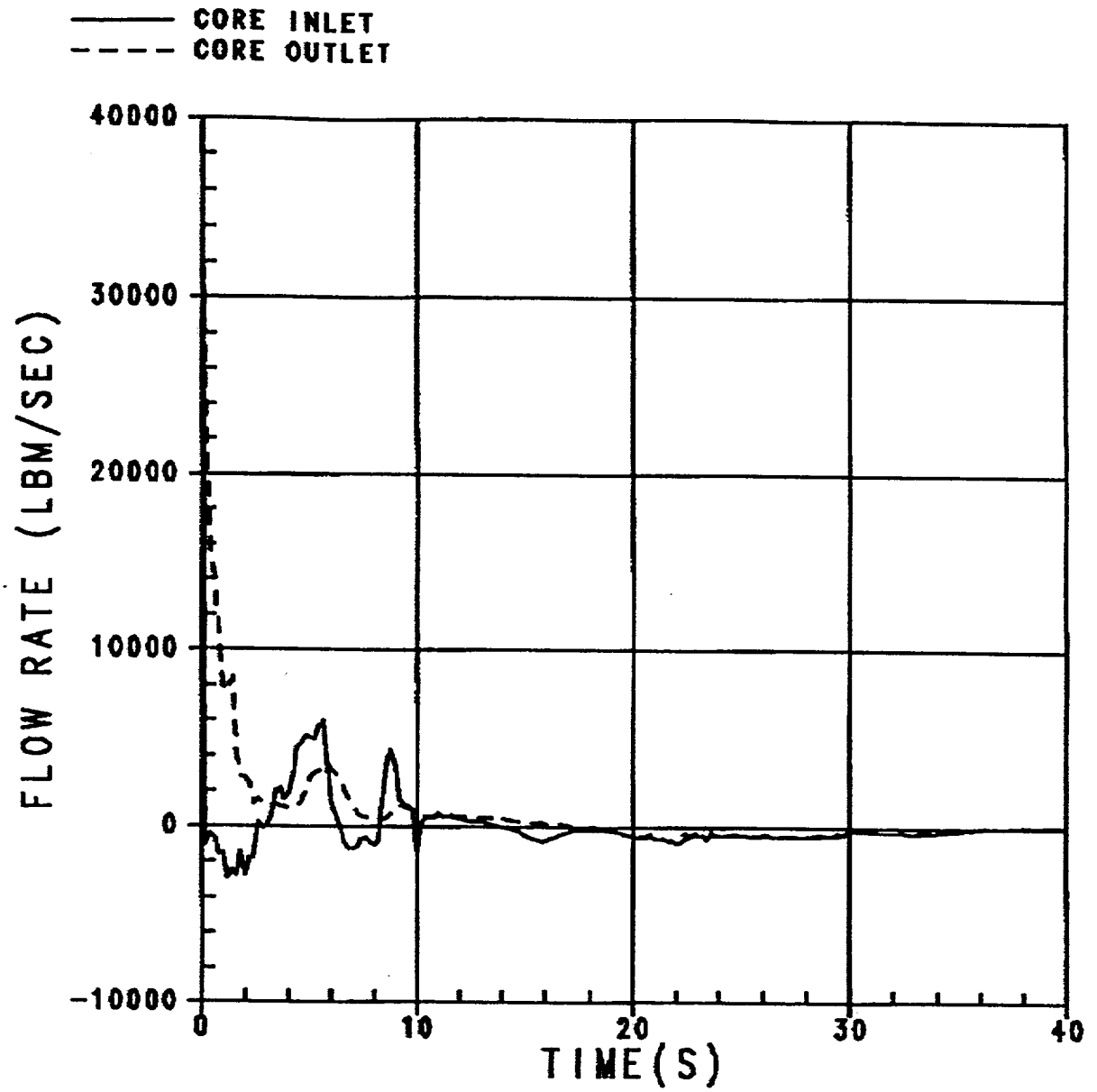

Figure 4 Core Flowrate
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

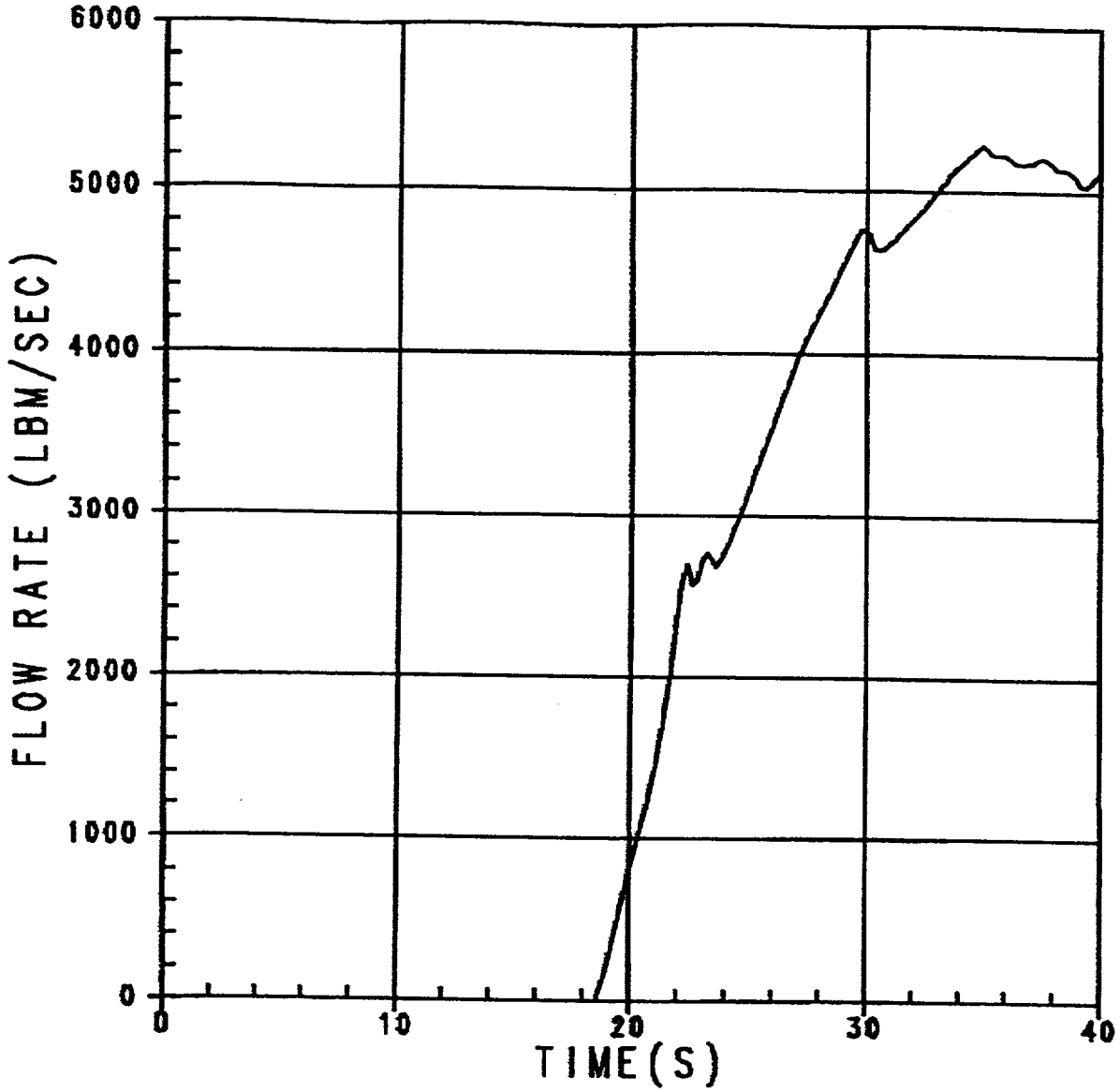

Figure 5 Accumulator Flow During Blowdown
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

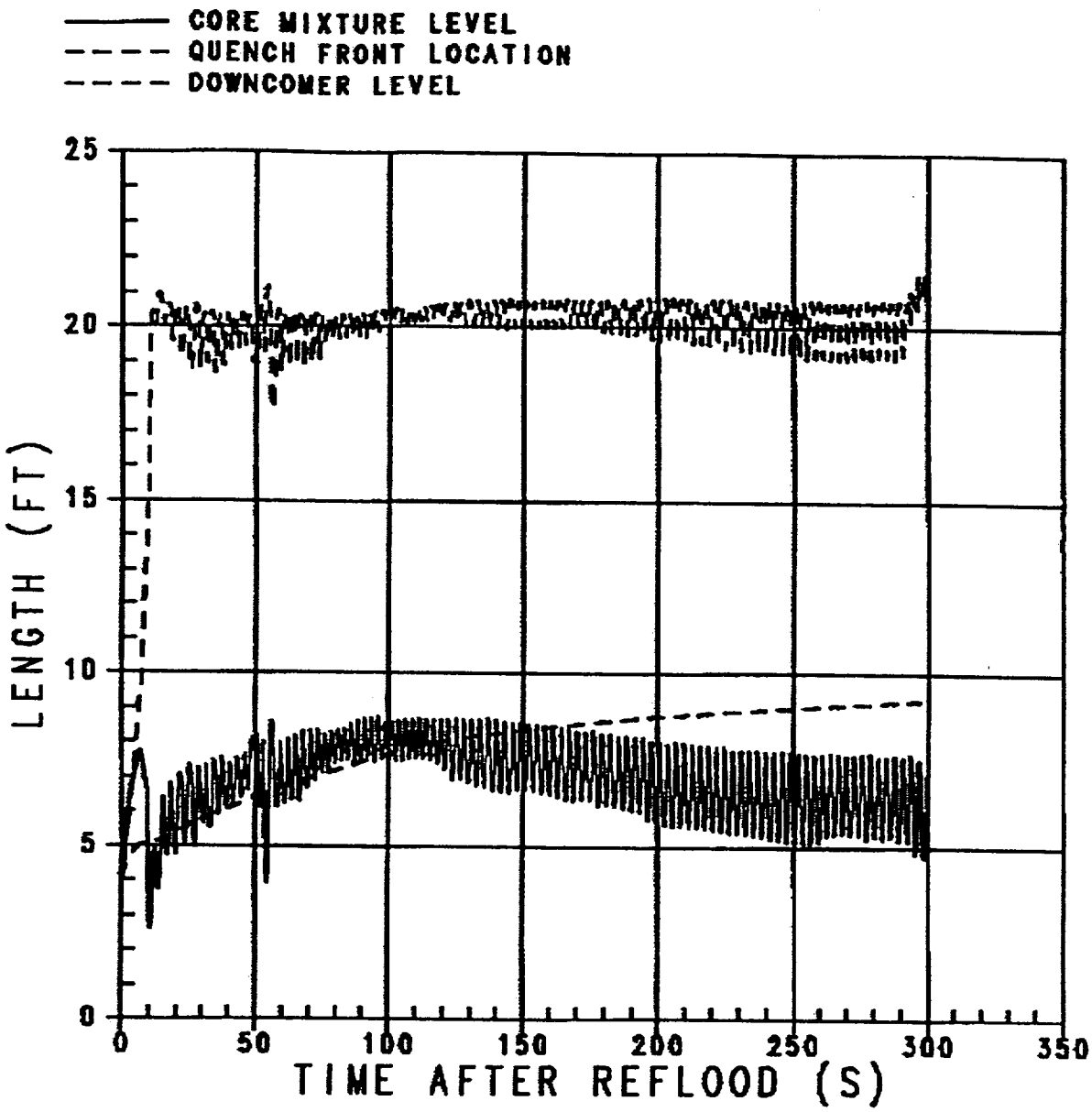

Figure 6 Vessel Liquid Levels During Reflood
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

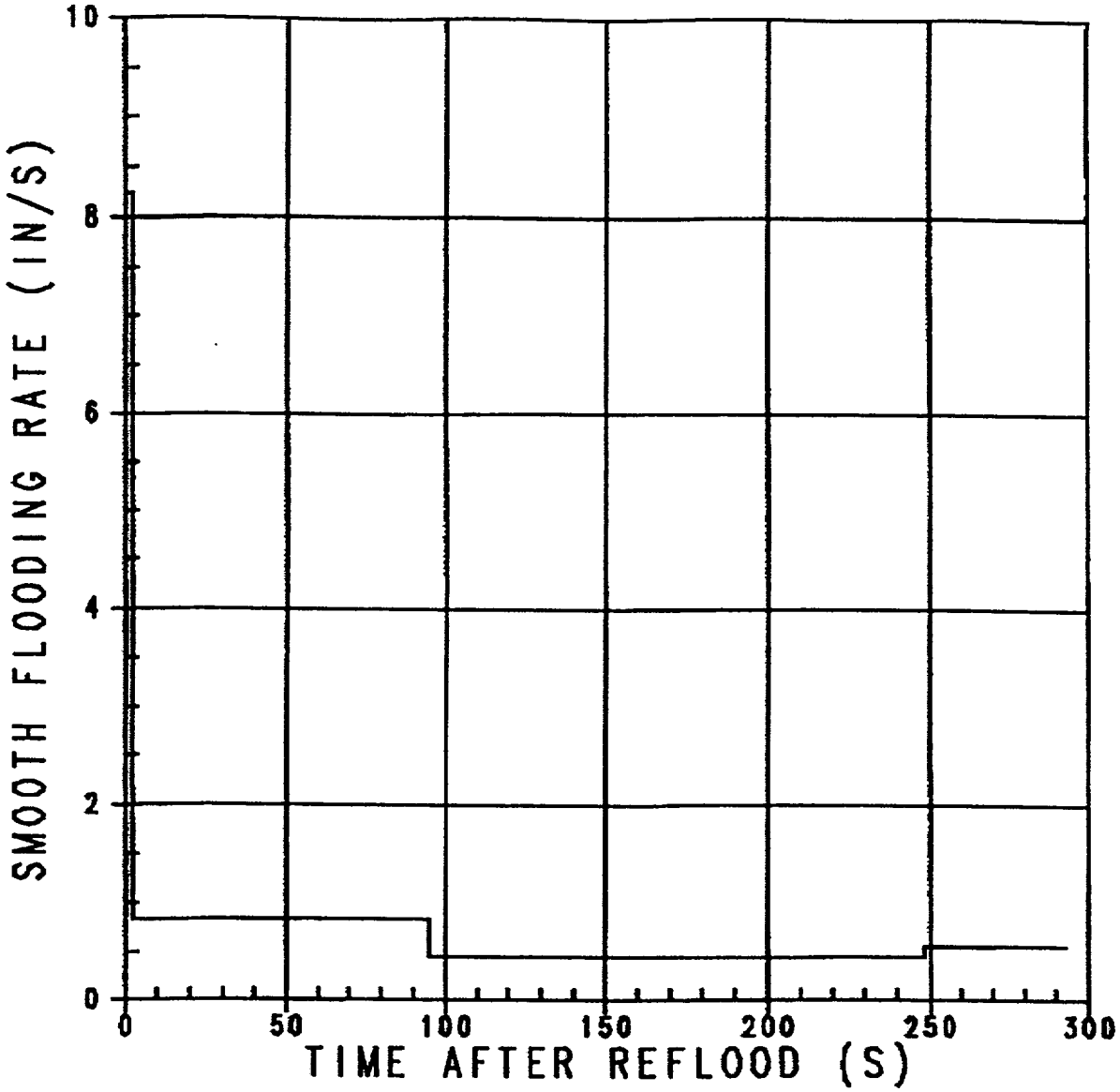

Figure 7 Core Inlet Flow During Reflood
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

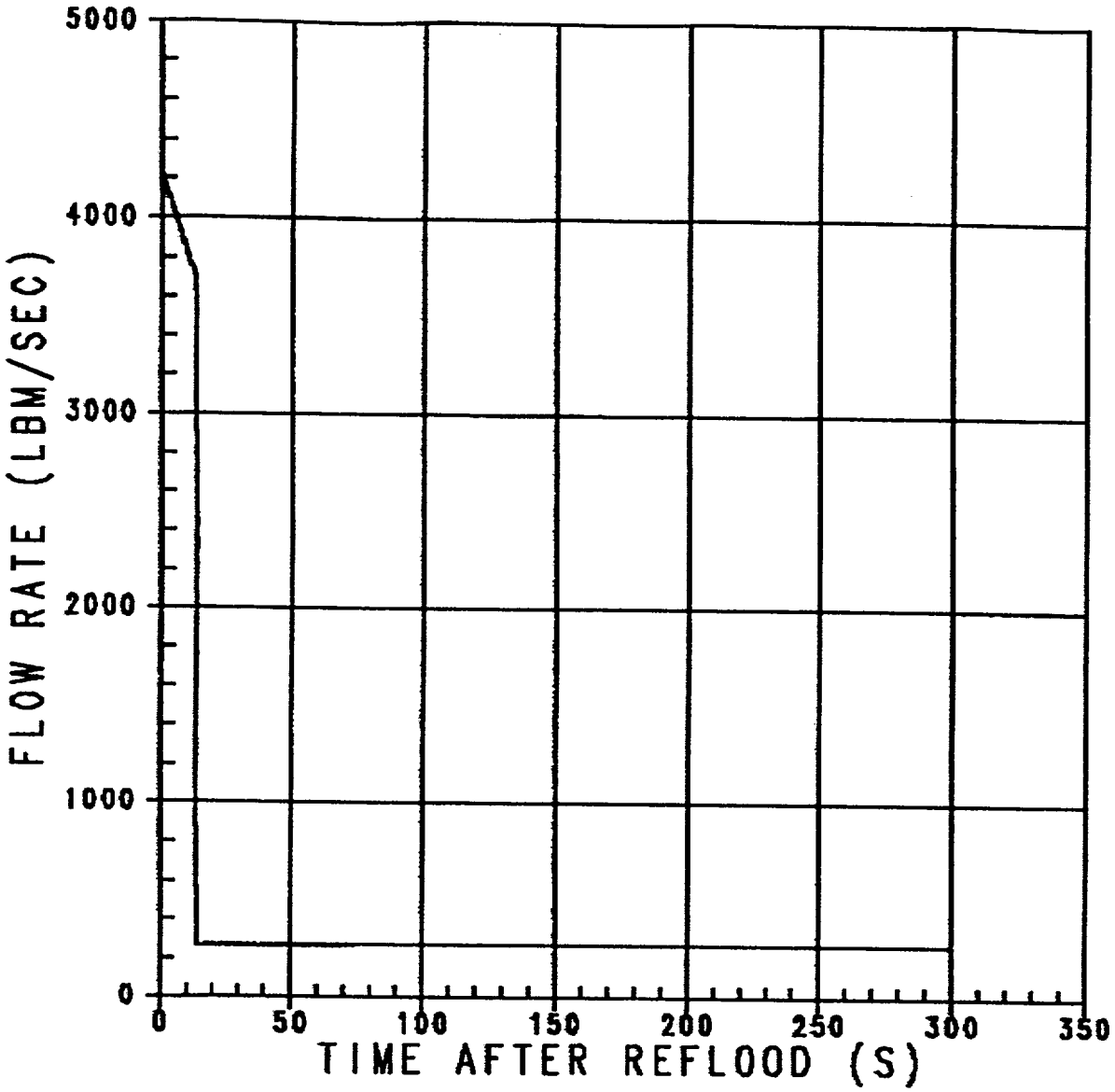

Figure 8 Accumulator and SI Flow During Reflood
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

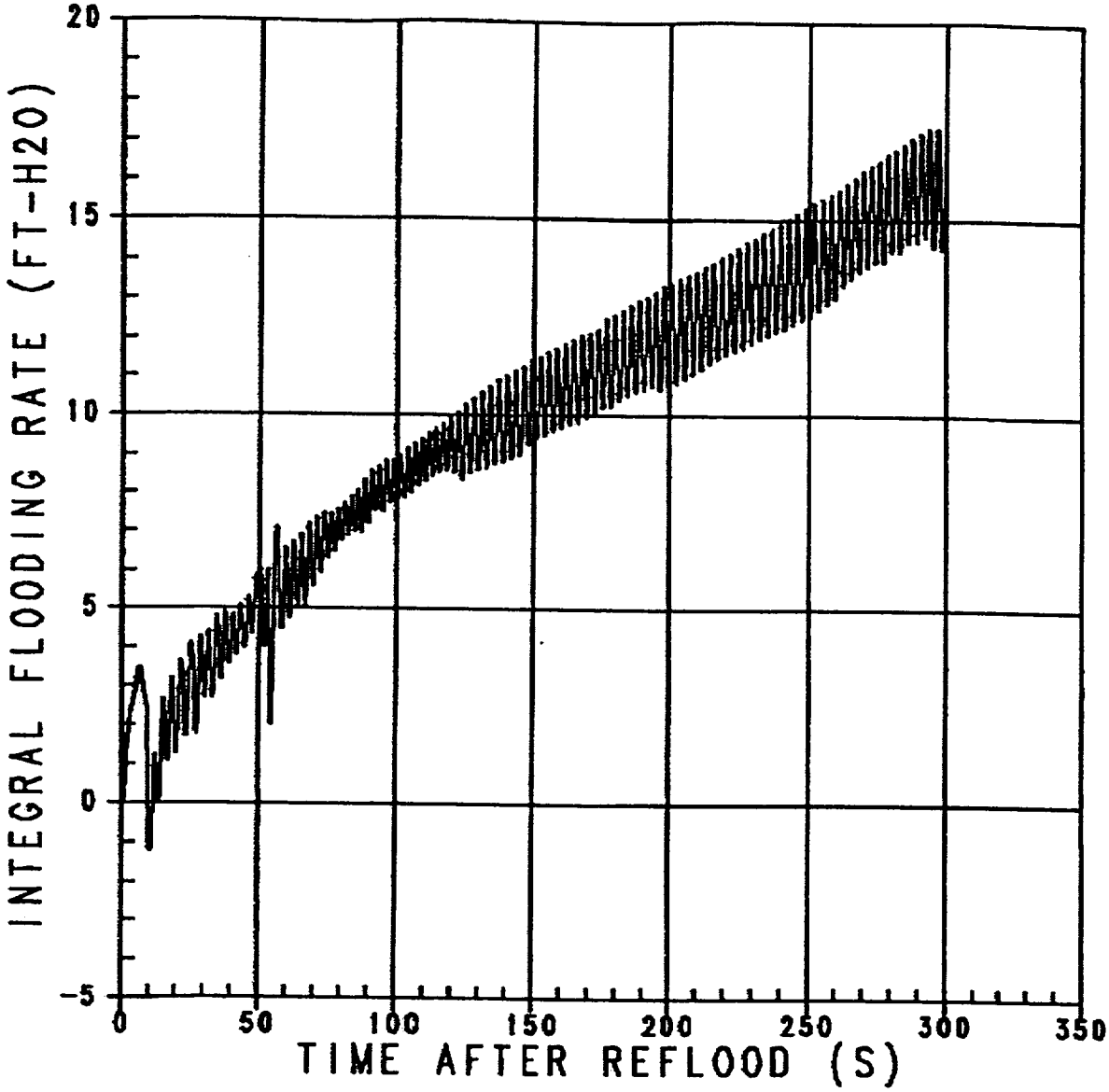

Figure 9 Integral of Core Inlet Flow
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit
With ZIRLO™ and IFM Grids

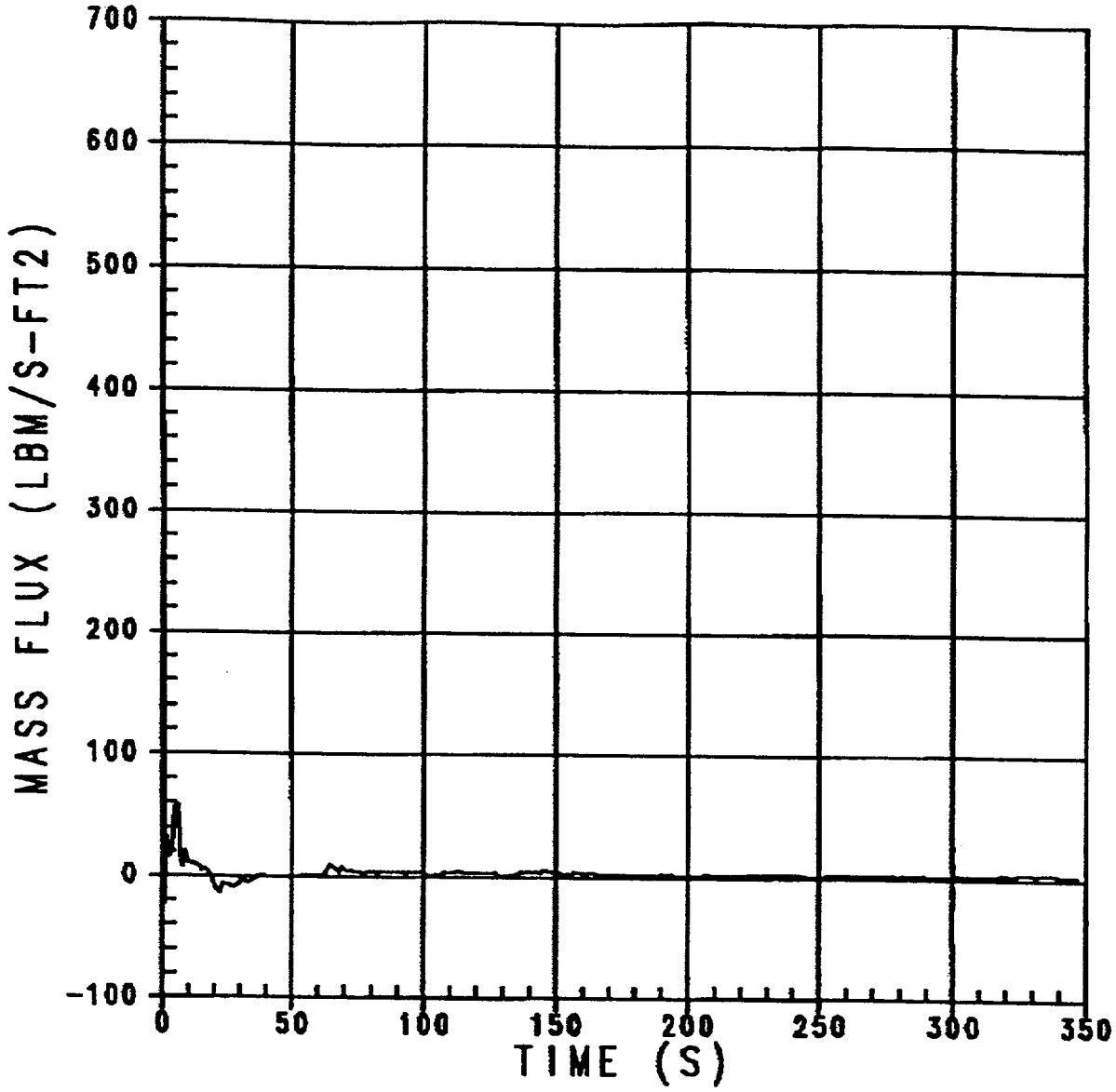

Figure 10 Mass Flux at Peak Temperature Elevation
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

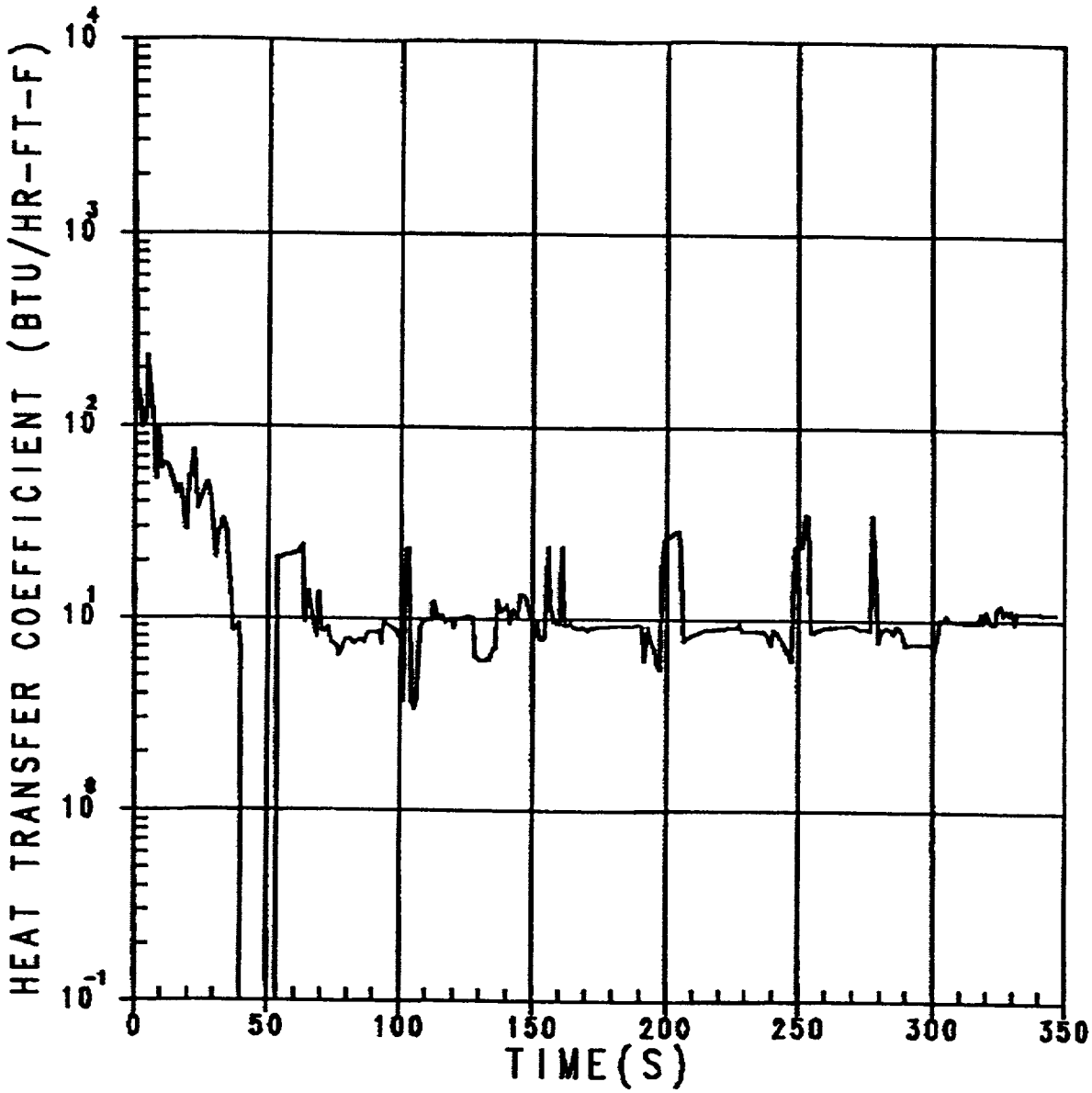

Figure 11 Rod H.T.C. at Peak Temperature Elevation
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

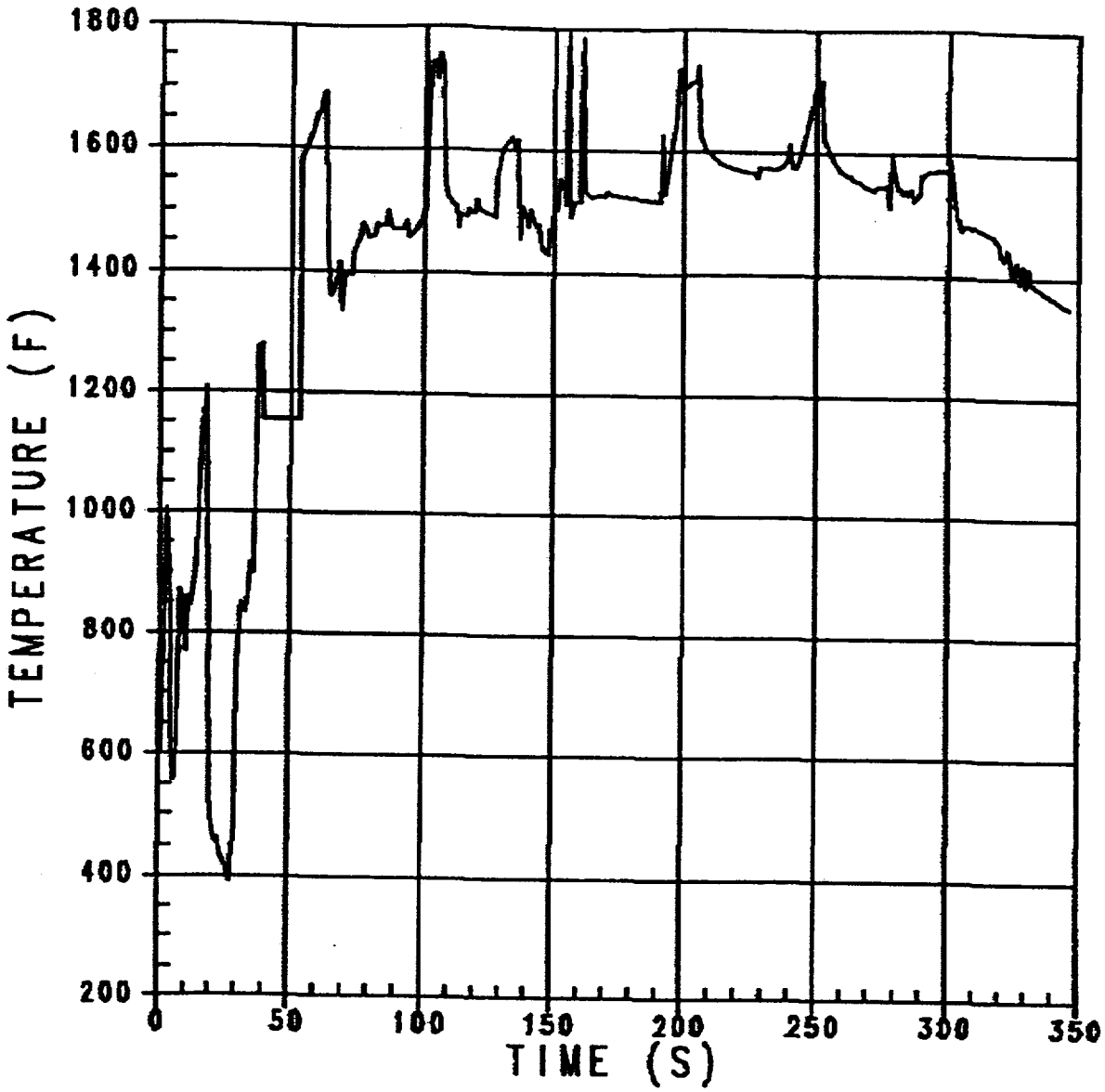

Figure 12 Vapor Temperature
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

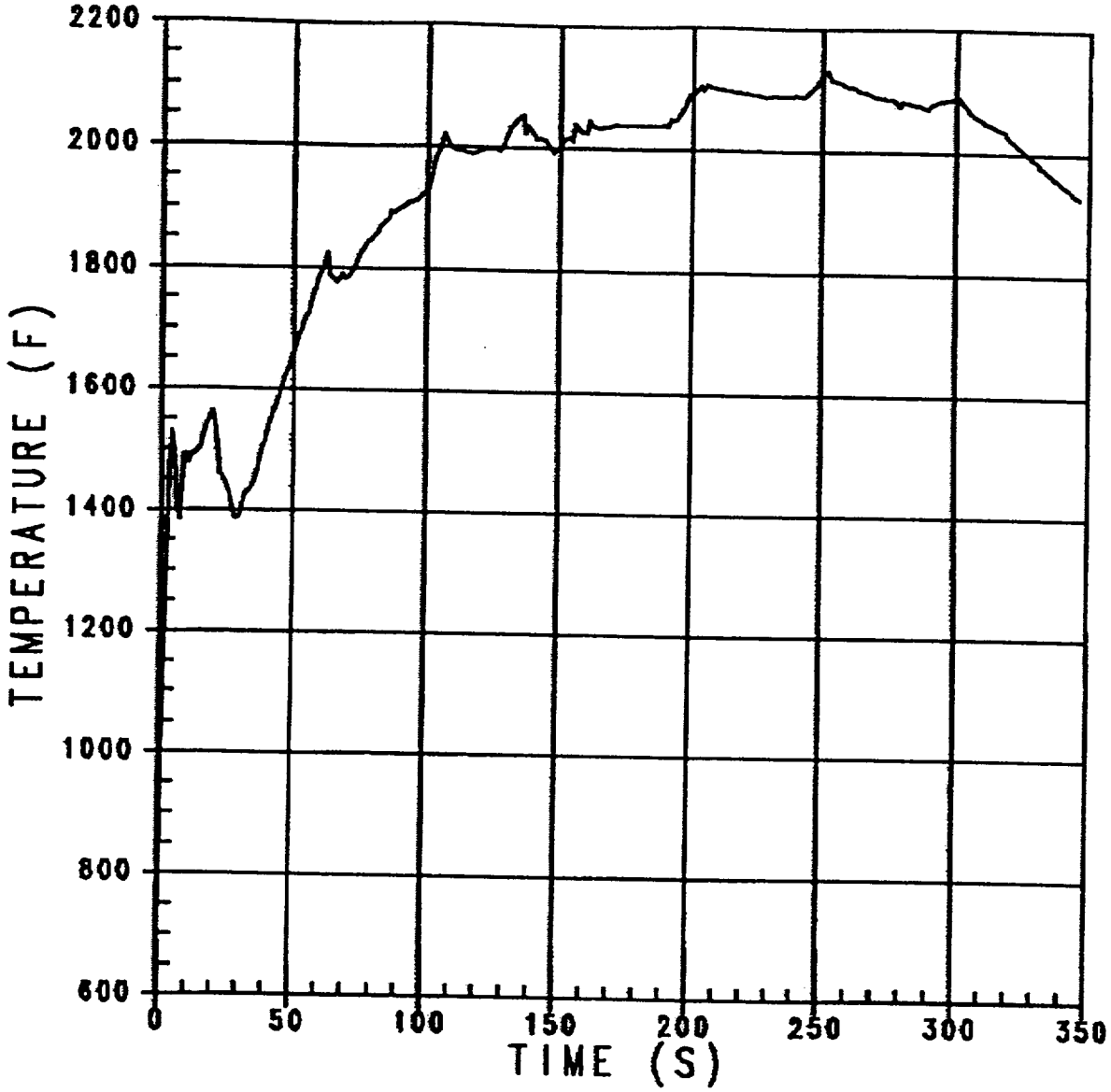

Figure 13 Fuel Rod Peak Clad Temperature
Case 7, CD=0.4, Thot=609.1°F, P=2100 psia
Donald C. Cook Unit 1
With ZIRLO™ and IFM Grids

ATTACHMENT 3 TO C0300-07

PEAK CLADDING TEMPERATURE MARGIN UTILIZATION TABLES FOR LARGE AND SMALL BREAK LOSS-OF-COOLANT ACCIDENTS ANALYSIS OF RECORD

DONALD C. COOK NUCLEAR PLANT, UNIT 1

TABLE 1

D. C. COOK UNIT 1

LARGE BREAK LOCA

Evaluation Model: BASH

$F_Q = 2.15$ $F_{\Delta H} = 1.55$ SGTP = 30% Break Size: $C_d = 0.4$

Operational Parameters: RHR System Cross-Tie Valve Closed, 3250 MWt Reactor Power

Fuel: Vantage+ (ZIRLO clad, IFM grids)

LICENSING BASIS		
	Analysis-of-Record - July 1998	PCT = 2133°F
MARGIN ALLOCATIONS (Delta PCT)		
A.	1999 10 CFR 50.46 MODEL ASSESSMENTS [1]	
	1. LOCBART Reanalysis for Spacer Grid Single Phase Heat Transfer Error [1]	ΔPCT= + 14°F
	2. LOCBART Reanalysis for Metal-Water Oxidation Error [1]	ΔPCT= + 44°F
B.	OTHER	
	1. Transition Core Penalty	ΔPCT= + 31°F
	2. Evaluation for Decreased Steam Generator Tube Plugging (from 30% to 25%)	ΔPCT= - 29°F
F.	LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT= 2193°F

Footnotes:

1. See Letter from M. W. Rencheck, I&M, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2, Errors in Loss-Of-Coolant-Accident Evaluation Models," submittal C1299-04, dated December 9, 1999.

TABLE 2

D. C. COOK UNIT 1

SMALL BREAK LOCA

Evaluation Model: NOTRUMP

$F_Q = 2.32$ $F_{\Delta H} = 1.55$ SGTP = 30% 3" cold leg break

Operational Parameters: SI System Cross-Tie Valve Closed, 3250 MWt Reactor Power

Fuel: OFA

LICENSING BASIS		
	Analysis of Record - December 1994	PCT = 1443°F
MARGIN ALLOCATIONS (Delta PCT)		
A.	PRIOR LOCA MODEL ASSESSMENTS – 1995 [1]	ΔPCT = + 5°F
B.	PRIOR LOCA MODEL ASSESSMENTS – 1996	ΔPCT = + 10°F
C.	PRIOR LOCA MODEL ASSESSMENTS – 1997	ΔPCT = 0°F
D.	PRIOR LOCA MODEL ASSESSMENTS – 1998	ΔPCT = 0°F
E.	1999 10 CFR 50.46 MODEL ASSESSMENTS	
	1. Asymmetric HHSI delivery [2]	ΔPCT = + 50°F
F.	LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT = 1508°F

Footnotes:

1. All permanent loss-of-coolant accident (LOCA) model assessments prior to 1995 were resolved by the new analysis.

2. See Letter from M. W. Rencheck, I&M, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2, Errors in Loss-Of-Coolant-Accident Evaluation Models," submittal C1299-04, dated December 9, 1999.