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Donald C. Cook Nuclear Plant Unit 2
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 2

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 2. This submittal includes a new large break LOCA (LBLOCA) analysis of record provided for NRC information along with responses to previous NRC requests for information pertaining to that analysis.

Attachment 1 to this letter describes the background for the new LBLOCA analysis of record and describes assessments against the small break LOCA analyses 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 I&M's responses to previous NRC requests for information pertaining to the new LBLOCA analysis of record. Attachment 5 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,

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A001

/dms

Attachments

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ATTACHMENT 1 TO C0200-08

BACKGROUND FOR NEW LARGE BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS OF RECORD AND ASSESSMENTS AGAINST THE SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSES 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 (Reference 1) provided Peak Cladding Temperature (PCT) margin utilization tables for two large break loss-of-coolant accident (LBLOCA) analyses of record for Unit 2; one in which the residual heat removal (RHR) pump discharge lines are assumed to be cross-tied and one in which pump discharges are assumed not to be cross-tied. These analyses credited the hot leg nozzle gap (HLNG) for providing a steam vent path for the lower portion of the reactor vessel, thus facilitating core re-flood. However, the NRC staff has not accepted LOCA analysis models that credit this HLNG effect. Accordingly, I&M is replacing the LBLOCA analyses of record identified in Reference 1 with another analysis (described in Attachment 2 to this letter) that assumes no credit for the HLNG.

This new analysis of record was previously submitted to the NRC in support of a proposed license amendment (Reference 2) to increase (uprate) the Unit 2 licensed power level to 3588 MWt. Although the uprate amendment request was later withdrawn, the associated LBLOCA analysis model remains valid. As described in Attachment 2, the analysis was performed for both the RHR cross-tied and not cross-tied conditions. The PCT calculated for the RHR not cross-tied condition was greater than that calculated for the RHR cross-tied condition and therefore bounds both conditions. Accordingly, the LBLOCA analysis of the RHR not cross-tied condition, without credit for HLNG is now the new LBLOCA analysis of record, replacing both previous LBLOCA analyses of record. This analysis uses the Explicit Shape Analysis for PCT Effects (ESHAPE) methodology in place of the Power Shape Sensitivity Model (PSSM) for axial power shape evaluation used in the previous analyses. Since the analysis continues to use the previously approved BASH computer model and since the NRC has approved replacement of PSSM with ESHAPE (Reference 8), the new analysis of record is provided in Attachment 2 for NRC information.

Subsequent to conducting the new analysis of record described in Attachment 2, errors were identified in the LOCBART computer code used in the analysis for fuel rod heatup calculations. The estimated effect of these errors (a 58°F penalty) was reported to the NRC pursuant to 10 CFR 50.46(a)(3)(ii) by Reference 3. As a result of these errors, it was necessary to perform plant specific analyses based on the currently-licensed power level, 3413 MWt, rather than the previously requested uprated power level, 3588 MWt, to demonstrate that compliance with the PCT limit of 2200°F specified in 10 CFR 50.46 could be maintained. Two previous 10 CFR 50.46 model changes that had resulted in a net PCT penalty of 2°F were incorporated in the reanalyses. 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.

Prior to I&M withdrawing the proposed uprate amendment, the NRC requested additional information (References 4 and 5) regarding various aspects of the proposed amendment. I&M has reviewed References 4 and 5 and determined that some of the requested information pertained to the LBLOCA analysis submitted in support of the proposed amendment. Since that analysis is being resubmitted as the new LBLOCA analysis of record, I&M is providing responses to those NRC requests for information, or portions thereof, that it has identified as being relevant to the analysis. These responses are provided in Attachment 4 to this letter.

The new analysis of record assumes a value for the delay in starting the containment air recirculation system (CEQ) fans following a LOCA. 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 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 requested amendment and noted in the Safety Evaluation Report that analyses had demonstrated that licensing criteria such as that of 10 CFR 50.46 remained satisfied. Consequently, I&M considers that the new analysis of record remains valid.

This submittal satisfies I&M's commitments, made in Reference 1, to resubmit LOCA analysis portions of the Unit 2 uprate proposal and to include responses to pertinent NRC requests for information. This submittal also completes I&M's commitment, made in Reference 3, to submit PCT margin utilization tables addressing the LBLOCA model errors identified in that reference. However, the date by which I&M had committed in Reference 3 to submit these tables, January 31, 2000, was not achieved.

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

In Reference 3, 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 Technical Specification (T/S) required surveillance testing. This is reflected in the Unit 2 licensing basis, in that T/S 3.5.2 limits core power to 3250 MWt in this configuration. This value is supported by an SBLOCA analysis conducted specifically for the condition in which the SI pump discharge lines are not cross-tied.

As reported in Reference 3, a different, 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. Additionally, it was recognized that the SBLOCA burst and blockage/time-in-life PCT penalty that had been previously assessed for Unit 2 would change since the magnitude of the penalty depends on the net PCT resulting from other assessments for a given case, such as the asymmetric SI delivery assessment.

When the SBLOCA burst and blockage/time-in-life PCT penalty was evaluated, it was found that the PCT limits of 10 CFR 50.46 were approached. The SBLOCA analysis of record was re-baselined which consisted of re-performing the NOTRUMP code and LOCTA (fuel rod heatup) code calculations for the limiting SBLOCA scenario using the latest input assumptions on plant parameters. This resulted in a benefit of 214°F in the calculated PCT. In conjunction with this re-baselining, previously identified model errors, for which Donald C. Cook Nuclear Plant Unit 2 had been assessed generic penalties, were corrected, resulting in an additional net benefit of 32°F. Reference 3 documents I&M's commitment to submit a new SBLOCA analysis for the SI pump discharge lines not cross-tied condition.

Attachment 3 to this letter includes a PCT margin utilization table (Table 2) showing assessments against the SBLOCA analysis of record for a power level of 3250 MWt, which is the T/S limit for operation when the SI pump discharge lines are not cross-tied. The margin utilization table includes the estimated 50°F penalty resulting from asymmetric SI delivery, the 214°F benefit resulting from re-baselining, the 32°F benefit from correcting previously identified model errors, and SBLOCA burst and blockage/time-in-life PCT penalty, which was determined to be 15°F after all other penalties had been assessed.

Attachment 3 also includes a PCT margin utilization table (Table 3) for the previously established analysis of record for a SBLOCA assuming the SI pump discharge lines <u>are</u> cross-tied, thereby eliminating the need to consider asymmetric SI delivery. Although there have been no new PCT assessments for this analysis since the previous annual 10 CFR 50.46 report, the margin utilization table has been included in this submittal for completeness. The associated analysis, which is based on a power level of 3588 MWt, remains the bounding analysis of record for operation at the full licensed power level of 3413 MWt when SI pump discharge lines are cross-tied.

The previous annual 10 CFR 50.46 report also contained a margin utilization table for an analysis of record of a SBLOCA with the SI pump discharge lines not cross-tied and at an assumed power level of 3413 MWt. A margin utilization table for this case has not been included in Attachment 4 of this letter since the re-baselined analysis at 3250 MWt described above is the analysis of record for the SI pump discharge lines not cross-tied condition.

This submittal completes I&M's commitment, made in Reference 3, 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. However, the date by which I&M had committed in Reference 3 to submit these tables, January 31, 2000, was not achieved.

<u>Summary</u>

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 all current LBLOCA and SBLOCA analyses of record for Unit 2.

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 E. E. Fitzpatrick, I&M, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plants 1 and 2, License Nos. DPR-58 and DPR-74, Proposed License and Technical Specification Changes Supported by Analyses to Increase Unit 2 Rated Thermal Power and Certain Proposed Changes for Unit 1 Supported by Related Analyses," submittal AEP:NRC:1223, dated July 11, 1996.

3. 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.

4. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Re: Power Uprate Program (TAC Nos. M96363 and M96364)," dated July 9, 1997.

5. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Re: Power Uprate Program (TAC Nos. M96363 and M96364)," dated August 6, 1997.

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.

8. 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.

ATTACHMENT 2 TO C0200-08

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

DONALD C. COOK NUCLEAR PLANT, UNIT 2

3.0 SAFETY EVALUATION/ANALYSES

3.1 LOCA (LARGE BREAK AND SMALL BREAK)

3.1.1 Large Break LOCA

3.1.1.1 Introduction

The current licensing basis large break LOCA analysis for Donald C. Cook Nuclear Plant Unit 2 was performed to support operation with VANTAGE 5 fuel. The analysis is described in the VANTAGE 5 Reload Transition Safety Report (RTSR), Reference 1. The RTSR large break LOCA analysis is summarized in Section 3.1.1.2 of this report in order to present a complete picture of the proposed licensing basis analysis for the Uprating Program. Additional analyses have also been performed to support the Uprating Program, and these analyses are described in Section 3.1.1.3. The RTSR analyses were used as the basis for the uprating analyses, and thus the new analyses constitute sensitivity studies relative to the current licensing basis analyses. The uprating sensitivity studies have confirmed the limiting break size and operating conditions which were established in the RTSR analysis. However, the operating conditions for the proposed licensing basis for the Uprating Program will rest on the RTSR analyses, since the RTSR analyses include a low temperature, high pressure case and a maximum safeguards analysis which were not repeated in the Uprating Program sensitivity analyses.

Although the RTSR analyses indicated that it would be necessary to operate the unit with the RHR crosstie valves open to obtain an acceptable large break LOCA result for a core power level of 3588 MWt, the uprating analyses have demonstrated that an acceptable PCT can be obtained with the RHR crosstie valves closed. It was determined that revisions to the large break LOCA model, which have been made to resolve issues identified in the 10 CFR 50.46 reports since the RTSR analysis was performed, have resulted in a significant PCT benefit for Donald C. Cook Nuclear Plant Unit 2. It was determined that the revision to the grid heat transfer model in the LOCBART program was the major contributor to the reduction in the PCT. The results of the uprating analyses, including the effect of the grid heat transfer model change, are discussed in Section 3.1.1.3.

3.1.1.2 RTSR Large Break LOCA Analysis

The licensing basis large break LOCA analysis for Donald C. Cook Nuclear Plant Unit 2 was performed to support plant operation with VANTAGE 5 fuel installed. A detailed description of the large break LOCA analysis performed for the VANTAGE 5 reload is presented in the VANTAGE 5 RTSR (Reference 1), and the results of the analysis are summarized below.

The large break LOCA analysis for the RTSR was performed to support operation at a core power level of 3588 MWt with the RHR crosstie valves open, and also at 3413 MWt with the

RHR crosstie valves closed. The analysis was performed with the December 1981 version of the Westinghouse ECCS Evaluation Model modified to incorporate the BASH (Reference 2) computer code. The analysis was performed for a double-ended cold leg guillotine (DECLG) break, which has been shown to represent the limiting break for the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. Although the large break single failure for the Westinghouse design is the loss of one RHR pump, it was conservatively assumed that only one train of ECCS is available for delivery of water to the RCS. However, both Emergency Diesel Generators were assumed to start and full containment heat removal systems operation was modelled. The safety injection flow for the analysis was based on the operation of one charging pump with the minimum resistance branch injection line spilling to containment backpressure, and the operation of one safety injection pump and one RHR pump with the minimum resistance accumulator injection line spilling to containment backpressure. In addition, all safety injection pump performance curves were degraded by 10%.

A range of reactor operating temperatures was analyzed in the RTSR in order to justify plant operation at a reactor power level of 3588 MWt between 582.2°F to 615.2°F in the hot legs and 511.7°F to 547.6°F in the cold legs. In addition to the temperature range analyzed, initial RCS pressurizer pressure was also varied to justify plant operation at 2100 or 2250 psia (2037 or 2313 psia, respectively, with the pressure uncertainty included). The analyses were performed using minimum safeguards assumptions, with safety injection flows based on the RHR crosstie valves open. A full spectrum break analysis was done at the high pressure/high temperature RCS conditions (initial RCS pressurizer pressure, with uncertainty, of 2313 psia and initial hot leg temperature of 615.2°F) from which the limiting break size was determined. The limiting break was then reanalyzed for low temperature and high RCS pressure, and also for high temperature and low initial RCS pressure. The limiting case was also reanalyzed assuming maximum safeguards ECCS flow rates.

The analysis also considered plant operation at 3413 MWt with the RHR crosstie valves closed. The lower power level was considered necessary to offset the reduction in safety injection flow due to the closure of the RHR crosstie valve. This case assumed a power level of 3413 MWt and minimum safeguards with the RHR crosstie valves closed at the limiting RCS conditions. All cases conservatively assumed 15% steam generator tube plugging in all four steam generators. Table 3.1-1 describes the cases analyzed. Tables 3.1-2 and 3.1-3 summarize the key input parameters and setpoints modelled in the RTSR large break LOCA analysis. The analysis was performed with a reactor vessel upper head temperature equal to the RCS hot leg temperature.

The results of these calculations are summarized in Tables 3.1-4 and 3.1-5. The peak clad temperature (PCT) was calculated for the 0.6 C_D cold leg break initiated at 3588 MWt with high RCS pressure and high temperature conditions, and with minimum safeguards ECCS flows (Case A). The PCT calculated for this case was 2140°F, which is less than the acceptance criterion limit of 2200°F in 10 CFR 50.46 (Reference 3). The PCT calculated for the limiting break and operating conditions at 3413 MWt (Case G) was 2090°F, which is also less than the acceptance limit of 2200°F.

3.1.1.3 Uprating Program Large Break LOCA Analysis

Introduction

The results of the RTSR large break LOCA analysis indicated that core power would be limited to 3413 MWt with the RHR crosstie valves closed, and that operation with the RHR crosstie valves open would be required in order to support the uprated core power of 3588 MWt. Therefore, a modification to the RHR System was proposed to enable continuous plant operation with the RHR crosstie valves open. The RTSR large break LOCA analysis at the uprated power level of 3588 MWt was then updated using safety injection flow rates based on the proposed modification to the RHR System with the RHR crosstie valves open. An analysis was also performed using safety injection flow rates for the current RHR System with the RHR crosstie valves closed. The safety injection flows are based on an increase in the pump head degradation from 10% to 15% for the high head safety injection pumps and the RHR pumps, and a pump head degradation of 10% for the centrifugal charging pumps. The safety injection flow rates used in the large break LOCA analyses for the proposed modified RHR System with the RHR crosstie valves open, and for the current system with the RHR crosstie valves closed are presented in Table 3.1-6. It is noted that the safety injection flow rates for the proposed modified RHR System with the crosstie valves open are approximately 20% less than the flow rates used in the RTSR analysis with the crosstie valves open, for the lower RCS pressures of primary interest for the large break LOCA analysis. The safety injection flow rates for the current RHR System with the crosstie valves closed are also slightly lower than the comparable values used in the RTSR analysis due to the increased degradation in the pump performance curves assumed for the analysis.

Evaluation Model Changes

The uprating large break reanalysis was also performed with the ECCS Evaluation Model with BASH. However, it is noted that the WREFLOOD code, which was previously used to calculate the RCS behavior during vessel lower plenum refill, has been replaced by the REFILL code as reported in Reference 4. The REFILL code is identical to the section of the WREFLOOD code that modelled the refill phase of the transient. There has also been a recent change in the methodology for execution of the BASH Evaluation Model as reported in Reference 5. The changes involve revisions to the procedures used to couple the various codes in the entire execution stream, but no changes were made to any of the approved physical models or basic techniques which form the basis of the methodology. The pertinent change which was made is the incorporation of the REFILL and LOCTA codes directly into the BASH code as subroutine modules. In addition, the LOTIC code which is used for containment backpressure calculations for ice-condenser plants has been coupled with the BASH code, so that the codes run interactively. The BASH Evaluation Model now utilizes the

SATAN code for the blowdown calculations, the BASH code for the refill and reflood phases with interactive LOTIC calculations for containment backpressure, and the LOCBART code for the core fuel rod heatup calculations.

The large break reanalysis also incorporates other model and analysis changes that have resulted from the resolution of issues which have been identified in the 10 CFR 50.46 reports since the RTSR analysis was completed. The large break LOCA model issues which have been identified and resolved since the RTSR analysis was completed include the inconsistency between the LOCA fuel rod model and the fuel rod design model, fuel rod burst and blockage assumptions, effect of steam generator tube crush for a combined LOCA and seismic event, the structural heat modelling in WREFLOOD, spacer grid heat transfer error in BART, vessel and steam generator calculation errors in LUCIFER, a revised burst strain limit model, corrections in the BASH loop/core interface, error in the pellet power radial flux depression factor, and the use of the ESHAPE methodology to explicitly evaluate the effect of skewed power shapes. The uprating large break LOCA reanalysis reflects the changes resulting from the resolution of these issues. Although the sum of the estimated PCT changes for the individual issues is relatively small, the combined effect of the changes on the PCT may be significantly different than the sum of the estimated individual effects.

Analysis for RHR Crosstie Valves Open

A large break LOCA analysis was first performed at the uprated power level of 3588 MWt using safety injection flow rates based on the proposed modification to the RHR System with the RHR crosstie valves open. The LOCA reanalysis was performed for the 0.6 C_D break at high pressure and high temperature conditions with minimum safeguards, which was previously demonstrated to be limiting for the RTSR analysis. The analysis was performed using essentially the same conditions and assumptions used in the RTSR analysis, with the exception of the safety injection flow rates. The analysis was performed using a total peaking factor of 2.220 and a hot channel enthalpy rise peaking factor of 1.620, which were used for the RTSR analysis. The key input parameters and assumptions used in the analysis are summarized in Table 3.1-7. The containment data used in the LOTIC program to generate the containment backpressure transient is presented in Table 3.1-8.

The results of this analysis are presented in the first column of Tables 3.1-9 and 3.1-10. The calculated PCT for this case is 1884°F at a core elevation of 6.25 feet, and occurs at a transient time of 56.8 seconds. A comparison of these results with the previous RTSR analysis indicates that the PCT is significantly less than the previous value of 2140°F. In addition, the PCT for this analysis occurs at a lower core elevation and much earlier in the transient than for the original analysis. Since the only significant difference in input parameters between the two analyses was the reduction in the safety injection flow rates which was expected to result in a PCT penalty, an evaluation was performed to determine the reason for the significant PCT benefit.

The reduction in the PCT was attributed to the combination of the model changes which have been incorporated to resolve the issues identified in the 10 CFR 50.46 reports for Donald C. Cook Nuclear Plant Unit 2 and in Westinghouse reports to the NRC since the RTSR analysis was completed. It was determined that the revision in the grid heat transfer model in the LOCBART program used for the fuel rod temperature transient calculation was a major contributor to the reduction in the PCT. A description and evaluation of the revised grid heat transfer model, along with the NRC SER for the model change, is provided in Addendum 1 to WCAP-10484-P-A (Reference 6). As indicated in Addendum 1 to WCAP-10484-P-A, the grid model revision generally resulted in a significant improvement in the ability of the grids to wet. It was noted that the effect of the revision on the PCT was very transient specific, with observed PCT changes ranging from a small penalty to benefits as much as 150°F. However, a 0°F PCT change was conservatively assigned for this issue for the purpose of 10 CFR 50.46 reporting. For the uprating analysis, the change resulted in a significant increase in the heat transfer due to grid wetting, which effectively reversed the clad temperature excursion at the higher core elevations earlier in the transient. This resulted in the PCT occurring at a lower core elevation and much earlier in the transient, with a corresponding reduction in the PCT.

Because the grid model revision resulted in a significant PCT benefit, analyses were also performed for the 0.4 and 0.8 C_D breaks to determine if the limiting break discharge coefficient would change. The results of these analyses, which are also summarized in Tables 3.1-9 and 3.1-10, confirmed that the 0.6 C_D break remained limiting. An analysis was also performed for the 0.6 C_D break at reduced pressure and high temperature conditions to ensure that the limiting conditions did not change for the uprated power conditions. The results of this analysis in Tables 3.1-9 and 3.1-10 show that the high pressure and high temperature conditions remained more limiting than the low pressure and high temperature conditions. Evaluations were performed for high pressure and reduced temperature operating conditions and maximum safeguards conditions which demonstrated that the high pressure and high temperature conditions with minimum safeguards would also remain limiting.

The Power Shape Sensitivity Model (PSSM) which was previously used to evaluate the effects of skewed axial core power distributions in the large break LOCA analysis was recently replaced by an alternate methodology, designated as ESHAPE (Explicit SHape Analysis for PCT Effects). The ESHAPE methodology is based on explicit analysis of the large break LOCA transient with a set of skewed axial power shapes to supplement the standard analysis done with the chopped cosine power shape. The ESHAPE methodology was used to evaluate the effect of skewed power shapes for the limiting 0.6 C_D break for high pressure and high temperature conditions. The analysis for skewed power shapes demonstrated that the cosine power shape would be limiting for the Donald C. Cook Nuclear Plant Unit 2 uprating analysis.

Analysis for RHR Crosstie Valves Closed

Since the large break LOCA analysis at the uprated power level of 3588 MWt with the RHR crosstie valves open resulted in substantial PCT margin to the 2200°F limit, it appeared that acceptable results could be obtained with the RHR crosstie valves closed. This would eliminate the need for the proposed RHR System modification, such that the RHR crosstie valves could remain closed for operation at the uprated power. Therefore, an analysis was performed at the uprated power level of 3588 MWt using safety injection flow rates for the RHR crosstie valves closed. The safety injection flow rates with the RHR crosstie valves closed which were used in the analysis are presented in Table 3.1-6.

The analysis was performed for the 0.6 C_D break at high pressure and high temperature conditions with minimum safeguards, which was demonstrated to be limiting for the RTSR analysis and confirmed to remain limiting in the uprating analysis. The parameters and assumptions used for the Uprating Program with the RHR crosstie valves open were also utilized for this analysis, with the exception of the safety injection flow rates. The results of the analysis with the RHR crosstie valves closed are summarized in Tables 3.1-11 and 3.1-12. The results for this analysis are also presented in Figures 3.1-1 to 3.1-13, which show the transient behavior of selected parameters.

As shown in Table 3.1-12, the calculated PCT for operation at 3588 MWt with the RHR crosstie valves closed is 1908°F, which is well below the 10 CFR 50.46 limit of 2200°F. The maximum local metal-water reaction is 4.64%, which is well below the embrittlement limit of 17% as required by 10 CFR 50.46. The total metal-water reaction which corresponds to the amount of hydrogen generation is also less than the 1% criterion in 10 CFR 50.46. Therefore, adequate protection is provided by the Emergency Core Cooling System in the event of a large break LOCA for operation at 3588 MWt with the RHR crosstie valves closed. Based on this analysis, it is concluded that the planned RHR modification is not required and that plant operation can continue with the RHR crosstie valves closed.

Analysis for Increased Peaking Factors

As noted previously, the large break LOCA analyses for the RTSR and the current Uprating Program at 3588 MWt have been performed using a total peaking factor (F_Q) of 2.220 and a hot channel enthalpy rise peaking factor ($F_{\Delta H}$) of 1.620. However, the RTSR large break LOCA analysis for operation at 3413 MWt with the RHR crosstie valves closed is based on F_Q = 2.335 and $F_{\Delta H}$ = 1.644. Since the calculated PCT for the large break LOCA analysis at the uprated power level of 3588 MWt with the RHR cross tie valves closed is significantly less than 2200°F limit, it appeared that the current peaking factors for 3413 MWt could also be accommodated at 3588 MWt. If the peaking factors for operation at 3413 MWt could be maintained for the uprated power of 3588 MWt, this would provide additional core design flexibility for future operating cycles. Therefore, an analysis was also performed at the uprated power level of 3588 MWt for the RHR crosstie valves closed, with F_Q = 2.335 and $F_{\Delta H}$ = 1.644. The hot assembly average power was also changed from 1.443 to 1.464 to be consistent with the increase in $F_{\Delta H}$ to 1.644.

The analysis was performed for the 0.6 C_D break at high pressure and high temperature conditions with minimum safeguards which was demonstrated to be limiting for the uprating analysis. The parameters and assumptions used for the uprating analysis with the RHR crosstie valves closed were also utilized for this analysis, with the exception of the peaking factor changes. The results of the analysis with F_Q = 2.335 and $F_{\Delta H}$ = 1.644 are summarized in Tables 3.1-11 and 3.1-12, along with results of the analysis for F_Q = 2.220 and $F_{\Delta H}$ = 1.620. The transient behavior of selected parameters for this analysis are also presented in Figures 3.1-14 to 3.1-26.

As shown in Table 3.1-12, the calculated PCT is 2051°F for operation at 3588 MWt with the RHR crosstie valves closed, and with F_Q = 2.335 and $F_{\Delta H}$ = 1.644, which is still below the 10 CFR 50.46 limit of 2200°F. The maximum local metal-water reaction is 6.42%, which is well below the embrittlement limit of 17% as required by 10 CFR 50.46. The total metal-water reaction which corresponds to the amount of hydrogen generation is also less than the 1% criterion in 10 CFR 50.46. Therefore, adequate protection is provided by the Emergency Core Cooling System in the event of a large break LOCA for operation at 3588 MWt with the RHR crosstie valves closed, and with a total peaking factor of 2.335 and a hot channel enthalpy rise peaking factor of 1.644.

3.1.1.4 <u>Conclusions</u>

Based on the large break LOCA analyses performed for the Uprating Program, it is concluded that Donald C. Cook Nuclear Plant Unit 2 operation at 3588 MWt with the RHR crosstie valves closed is acceptable, and that the proposed modification to the RHR System to permit continuous plant operation with the RHR crosstie valves open is not required. It is also concluded that operation will be acceptable with a total core peaking factor of 2.335 and a hot channel enthalpy rise peaking factor of 1.644.

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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. Liparulo, N. J. (Westinghouse) letter to W. T. Russel (USNRC), May 23, 1994, NTD-NRC-94-4143, "Change in Methodology for Execution of BASH Evaluation Model".

5. Liparulo, N. J. (Westinghouse) letter to W. T. Russel (USNRC), August 29, 1995, NTD-NRC-95-4520, "Change in Methodology for Execution of BASH Evaluation Model".

6. Shimeck, D. J., "Spacer Grid Heat Transfer Effects During Reflood", Addendum 1 to WCAP-10484-P-A (Proprietary), September 1993.

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-1
RTSR ANALYSIS
LARGE BREAK LOCA ANALYSIS
CASES ANALYZED

CASE A - C_D=0.6, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.2°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.220, $F^N_{\Delta H}$ =1.620, Minimum SI with crosstie valves open. Limiting break case, i.e., this case had highest PCT for all cases analyzed.

CASE B - C_D=0.4, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.2°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.240, $F^N_{\Delta H}$=1.620, Minimum SI with crosstie valves open.

CASE C - C_D=0.8, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.2°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.240, $F^N_{\Delta H}$=1.620, Minimum SI with crosstie valves open.

CASE D - C_D=0.6, 3588 Mwt Core Power, Low Temperature (T_{HOT}=582.3°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.220, $F^N_{\Delta H}$=1.620, Minimum SI with crosstie valves open.

CASE E - C_D=0.6, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.0°F), Low Pressure (P_{RCS}=2037 psia), F_Q=2.220, $F^N_{\Delta H}$=1.620, Minimum SI with crosstie valves open.

CASE F - C_D=0.6, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.2°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.220, $F^N_{\Delta H}$=1.620, Maximum SI with crosstie valves open.

CASE G - C_D=0.6, 3413 Mwt Core Power, High Temperature (T_{HOT}=611.2°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.335, $F^N_{\Delta H}$=1.644, Minimum SI with RHR crosstie valves closed.

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-2
RTSR ANALYSIS
LARGE BREAK LOCA ANALYSIS
INPUT PARAMETERS

	Cross Ties Open	RHR Cross Ties Closed
License Core Power[a], (MWt)	3588	3413
Peak Linear Power[a], (kw/ft)	12.714	12.721
Total Peaking Factor, F_Q^T	2.220	2.335
Axial Peaking Factor, F_Z	1.370	1.420
Hot Channel Enthalpy Rise Factor, $F^N_{\Delta H}$	1.620	1.644
Power Shape:	Chopped Cosine	
Fuel Assembly Array	17 X 17 VANTAGE 5	
Accumulator Water Volume, Nominal (ft^3/accumulator)	946	946
Accumulator Tank Volume, Nominal (ft^3/accumulator)	1350	1350
Accumulator Gas Pressure, Minimum (psia)	600	600
Safety Injection Pumped Flow Rate	All pumps degraded 10%, Charging pump flow rate imbalance = 25 gpm)	
Initial Loop Flow (GPM)	88,500	88,500
Vessel Inlet Temperature (°F)	511.7 to 547.6	513.3 to 546.4
Vessel Outlet Temperature (°F)	582.2 to 615.2	580.6 to 611.2
Average Reactor Coolant Pressure (psia)	2037.4 or 2312.6	2037.4 or 2312.6
Steam Pressure (psia)	587 to 820	603 to 820
Steam Generator Tube Plugging Level (%)	15	15
Refueling Water Storage Tank Temperature (°F)	85 (Range 70-100)	85 (Range 70-100)

(a) Two percent is added to this power to account for calorimetric error.

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-3
RTSR ANALYSIS
LARGE BREAK LOCA ANALYSIS
SYSTEMS MODELLING

Pressurizer Low Pressure Reactor Trip (psia)	1860.0
Pressurizer Low Pressure Safety Injection (psia)[(a)]	1715.0
Containment HI Pressure for Safety Injection (psia)	15.8
Safety Injection Delay (includes signal processing, EDGs start-up, sequencer and pumps to full speed, sec)	27.0
Feedwater Isolation Delay after Reactor Trip (sec)[(b)]	0.0
Steamline Isolation Delay after Reactor Trip (sec)[(b)]	0.0

(a) This setpoint causes actuation of the safety injection at the times shown in Table 3.1-4, for all seven cases.

(b) Conservative modelling for Large Break LOCA

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-4
RTSR ANALYSIS
LARGE BREAK LOCA ANALYSIS TIME SEQUENCE OF EVENTS

	Case A C_D=0.6 Min SI 3588 Mwt	Case B C_D=0.4 Min SI 3588 Mwt	Case C C_D=0.8 Min SI 3588 Mwt	Case D C_D=0.6 Min SI 3588 Mwt	Case E C_D=0.6 Min SI 3588 Mwt	Case F C_D=0.6 Max SI 3588 Mwt	Case G C_D=0.6 RHR X-Tie 3413 Mwt
T_{HOT} =	615.2°F	615.2°F	615.2°F	582.3°F	615.0°F	615.2°F	611.2°F
P_{RCS} =	2313 psia	2313 psia	2313 psia	2313 psia	2037 psia	2313 psia	2313 psia
Start (sec)	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Reactor Trip Signal (sec)	0.669	0.681	0.661	0.527	0.515	0.669	0.642
Safety Injection Signal (sec)	4.70	4.99	4.54	4.14	3.93	4.70	4.62
Accumulator Injection Begins (sec)	14.6	20.4	12.0	13.0	14.8	14.6	14.6
End-of-Bypass (sec)	31.69	40.51	26.94	33.48	31.70	31.69	32.02
End-of-Blowdown (sec)	31.69	41.13	26.94	33.48	31.70	31.69	32.02
Pump Injection Begins (sec)	31.70	31.99	31.54	31.14	30.93	31.70	31.62
Bottom of Core Recovery (sec)	45.99	56.00	40.87	48.88	45.95	45.39	46.79
Accumulator Empty (sec)	59.40	66.64	55.66	60.00	59.40	59.57	59.40

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-5
RTSR ANALYSIS
LARGE BREAK LOCA ANALYSIS RESULTS

		Case A	Case B	Case C	Case D	Case E	Case F	Case G
		C_D=0.6	C_D=0.4	C_D=0.8	C_D=0.6	C_D=0.6	C_D=0.6	C_D=0.6
		Min SI	Min SI	Min SI	Min SI	Min SI	Max SI	RHR X-Tie
		3588 Mwt	3588 Mwt	3588 Mwt	3588 Mwt	3588 Mwt	3588 Mwt	3413 Mwt
T_{HOT} =		615.2°F	615.2°F	615.2°F	582.3°F	615.0°F	615.2°F	611.2°F
P_{RCS} =		2313 psia	2313 psia	2313 psia	2313 psia	2037 psia	2313 psia	2313 psia
Peak Clad Temperature	(°F)	2140.0	1848.2	1766.0	1878.4	2074.7	2102.7	2090.0
Peak Clad Temperature Location	(ft)	9.75	8.75	6.25	9.75	9.75	9.75	9.75
Peak Clad Temperature Time	(sec)	258.9	250.1	57.9	239.9	255.4	253.1	244.4
Local Zr/H_2O Reaction Maximum	(%)	6.80	3.56	2.97	3.30	5.71	6.18	6.08
Local Zr/H_2O Reaction Location	(ft)	9.75	6.25	5.25	9.75	9.75	9.75	9.75
Total Zr/H_2O Reaction	(%)	<0.3	<0.3	<0.3	<0.3	<0.3	<0.3	<0.3
Hot Rod Burst Time	(sec)	45.79	60.93	50.66	50.11	46.05	46.04	46.10
Hot Rod Burst Location	(ft)	6.00	6.25	5.25	6.00	6.00	6.00	6.00

CALCULATION ASSUMPTIONS

Peak Linear Power (Kw/ft), 102% of	12.714 (12.721 for Case G)
Peaking Factor (at License Rating)	2.220 (2.335 for Case G)
Accumulator Water Volume (ft^3) per accumulator	946
Cycle Analyzed	All

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-6
UPRATING PROGRAM
LARGE BREAK LOCA ANALYSIS
SAFETY INJECTION FLOW RATES

	SI Flow Rates (lbm/sec)	
RCS Pressure (psig)	With Proposed RHR Modification RHR Crossties Open	W/O Proposed RHR Modification RHR Crossties Closed
0	426.2	277.0
20	345.4	228.7
40	257.0	173.9
60	201.1	106.8
80	154.5	75.4
100	96.8	74.6

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-7
UPRATING PROGRAM
LARGE BREAK LOCA ANALYSIS
INPUT PARAMETERS

Licensed Core Power[a] (MWt)	3588
Peak Linear Power[a] (kW/ft)	12.714
Total Core Peaking Factor, F_Q	2.220
Axial Peaking Factor, F_Z	1.370
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.620
Maximum Assembly Average Power, $\bar{P}_{HA}$	1.443
Power Shape[b]	Cosine
Fuel Assembly Array	17x17 V5
Accumulator Water Volume[c] (ft^3/tank)	946
Accumulator Tank Volume (ft^3/tank)	1350
Minimum Accumulator Gas Pressure (psia)	600
Accumulator Water Temperature (°F)	100
Thermal Design Flow Rate (gpm/loop)	88,500
Nominal Vessel Inlet Temperature (°F)	511.7 to 547.6
Nominal Vessel Outlet Temperature (°F)	582.2 to 615.2
Nominal Vessel Average Temperature (°F)	547.0 to 581.3
Initial RCS Pressure Including Uncertainty[d] (psia)	2037.4 or 2312.6
Nominal Steam Pressure	587 to 820
Steam Generator Tube Plugging Level (%)	15
Refueling Water Storage Tank Temperature (°F)	87.5 (Range 70 - 105)

(a) Two percent is added to this power to account for calorimetric error.
(b) Cosine power shape was found to be more limiting than skewed power shapes.
(c) Additional accumulator line volume of 32 ft^3 per accumulator used in analysis.
(d) The pressure uncertainty is 62.6 psia.

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-8
UPRATING PROGRAM
LARGE BREAK LOCA ANALYSIS
ICE CONDENSER CONTAINMENT DATA

NET FREE VOLUME		
(Includes Distribution Between Upper, Lower, and Dead-Ended Compartments)	UC	746,829 ft^3
	LC	249,446 ft^3
	DE	116,168 ft^3
	IC	163,713 ft^3
Initial Conditions		
Pressure		14.7 psia
Maximum Temperature for the Upper, Lower, and Dead-Ended Compartments	UC	100°F
	LC	120°F
	DE	120°F
Minimum Temperature for the Upper, Lower, and Dead-Ended Compartments	UC	60°F
	LC	60°F
	DE	60°F
RWST Temperature		70°F
Temperature Outside Containment		-22°F
Initial Spray Temperature		70°F
Spray System		
Runout Flow for a Spray Pump		3600 gpm
Number of Spray Pumps Operating		2
Post-Accident Initiation of Spray System		36 sec
Distribution of Spray Flow to the Upper and Lower Compartments	LC	2700 gpm
	UC	4500 gpm
Deck Fan		
Post-Accident Initiation of Deck Fans		480 sec
Flow Rate per Fan		43,890 cfm per fan
Assumed Spray Efficiency of Water from Ice Condenser Drains		100%

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TABLE 3.1-8 (continued)
UPRATING PROGRAM
LARGE BREAK LOCA ANALYSIS
ICE CONDENSER CONTAINMENT DATA

STRUCTURAL HEAT SINKS

wall	compartment	area (ft^2)	thickness (ft)	material
1	LC	12,105	0.0469/2.0	steel/concrete
2	LC	11,701	2.0	concrete
3	LC	65,979	4.0	concrete
4	LC	5,462	0.0833	steel
5	LC	5,273	0.0103	steel
6	LC	290	0.25	lead
7	LC	14,896	0.0078	steel
8	LC	4,515	0.1042	steel
9	LC	5,775	0.009	steel
10	LC	57,317	0.00833	steel
11	LC	9,404	0.0313	steel
12	LC	2,623	0.0313	steel
13	UC	378	0.0365/0.1667	steel/concrete
14	UC	34,895	0.0078	steel
15	UC	8,060	0.0208	steel
16	UC	420	0.0052	steel
17	UC	29,332	2.0	concrete
18	UC	34,125	0.0469/2.0	steel/concrete
19	UC	420	0.0052	steel

UC: Upper Compartment
LC: Lower Compartment
DE: Dead-Ended Compartment
IC: Ice Compartment

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-9
UPRATING PROGRAM
LARGE BREAK LOCA ANALYSIS
TIME SEQUENCE OF EVENTS

RHR CROSSTIE VALVES OPEN

	C_D=0.6 Min SI 3588 Mwt 615.2°F 2313 psia	C_D=0.4 Min SI 3588 Mwt 615.2°F 2313 psia	C_D=0.8 Min SI 3588 Mwt 615.2°F 2313 psia	C_D=0.6 Min SI 3588 Mwt 615.2°F 2037 psia
T_{HOT} = / P_{RCS} =				
Start (sec)	0.0	0.0	0.0	0.0
Reactor Trip Signal (sec)	0.67	0.68	0.66	0.51
Safety Injection Signal (sec)	4.7	5.0	4.5	3.9
Accumulator Injection Begins (sec)	14.0	20.0	12.0	15.0
End-of-Bypass (sec)	31.8	39.9	28.4	30.5
End-of-Blowdown (sec)	32.8	39.9	28.4	32.3
Pump Injection Begins (sec)	31.7	32.0	31.5	30.9
Bottom of Core Recovery (sec)	46.5	55.8	42.8	45.8
Accumulator Empty (sec)	60.3	67.1	56.7	60.4

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-10
UPRATING PROGRAM
LARGE BREAK LOCA ANALYSIS RESULTS

RHR CROSSTIE VALVES OPEN

		C_D=0.6	C_D=0.4	C_D=0.8	C_D=0.6
		Min SI	Min SI	Min SI	Min SI
		3588 Mwt	3588 Mwt	3588 Mwt	3588 Mwt
	T_{HOT} =	615.2°F	615.2°F	615.2°F	615.2°F
	P_{RCS} =	2313 psia	2313 psia	2313 psia	2037 psia
Peak Clad Temperature	(°F)	1884.4	1842.2	1866.7	1852.9
Peak Clad Temperature Location	(ft)	6.25	6.25	5.50	5.50
Peak Clad Temperature Time	(sec)	56.8	68.0	194.2	56.2
Local Zr/H_2O Reaction Maximum	(%)	4.16	3.19	3.31	3.70
Local Zr/H_2O Reaction Location	(ft)	6.25	6.25	5.50	5.50
Total Zr/H_2O Reaction	(%)	<1.0	<1.0	<1.0	<1.0
Hot Rod Burst Time	(sec)	44.1	54.7	48.2	43.9
Hot Rod Burst Location	(ft)	6.25	6.25	6.25	5.75

<u>CALCULATION ASSUMPTIONS</u>

Peak Linear Power (Kw/ft), 102% of	12.714
Peaking Factor (at License Rating)	2.220
Accumulator Water Volume (ft^3) per accumulator	946
Cycle Analyzed	All

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-11
UPRATING PROGRAM
LARGE BREAK LOCA ANALYSIS
TIME SEQUENCE OF EVENTS

RHR CROSSTIE VALVES CLOSED

C_D = 0.6
Min SI
3588 Mwt
T_{HOT} = 615.2°F
P_{RCS} = 2313 psia

	F_Q = 2.220, $F_{\Delta H}$ = 1.620	F_Q = 2.335, $F_{\Delta H}$ = 1.644
Start (sec)	0.0	0.0
Reactor Trip Signal (sec)	0.67	0.67
Safety Injection Signal (sec)	4.7	4.7
Accumulator Injection Begins (sec)	14.0	14.0
End-of-Bypass (sec)	31.8	31.8
End-of-Blowdown (sec)	32.8	32.8
Pump Injection Begins (sec)	31.7	31.7
Bottom of Core Recovery (sec)	46.6	46.6
Accumulator Empty (sec)	60.2	60.2

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-12
UPRATING PROGRAM
LARGE BREAK LOCA ANALYSIS RESULTS

RHR CROSSTIE VALVES CLOSED

		C_D = 0.6 Min SI 3588 Mwt T_{HOT} = 615.2°F P_{RCS} = 2313 psia	
		F_Q = 2.220 $F_{\Delta H}$ = 1.620	F_Q = 2.335 $F_{\Delta H}$ = 1.644
Peak Clad Temperature	(°F)	1908.1	2051.2
Peak Clad Temperature Location	(ft)	6.25	6.25
Peak Clad Temperature Time	(sec)	58.0	57.8
Local Zr/H_2O Reaction Maximum	(%)	4.64	6.42
Local Zr/H_2O Reaction Location	(ft)	6.25	6.25
Total Zr/H_2O Reaction	(%)	<1.0	<1.0
Hot Rod Burst Time	(sec)	44.1	39.7
Hot Rod Burst Location	(ft)	6.25	6.25

CALCULATION ASSUMPTIONS		
Peak Linear Power (Kw/ft), 102% of	12.714	13.373
Peaking Factor (at License Rating)	2.220	2.335
Accumulator Water Volume (ft^3) per accumulator	946	946
Cycle Analyzed	All	All

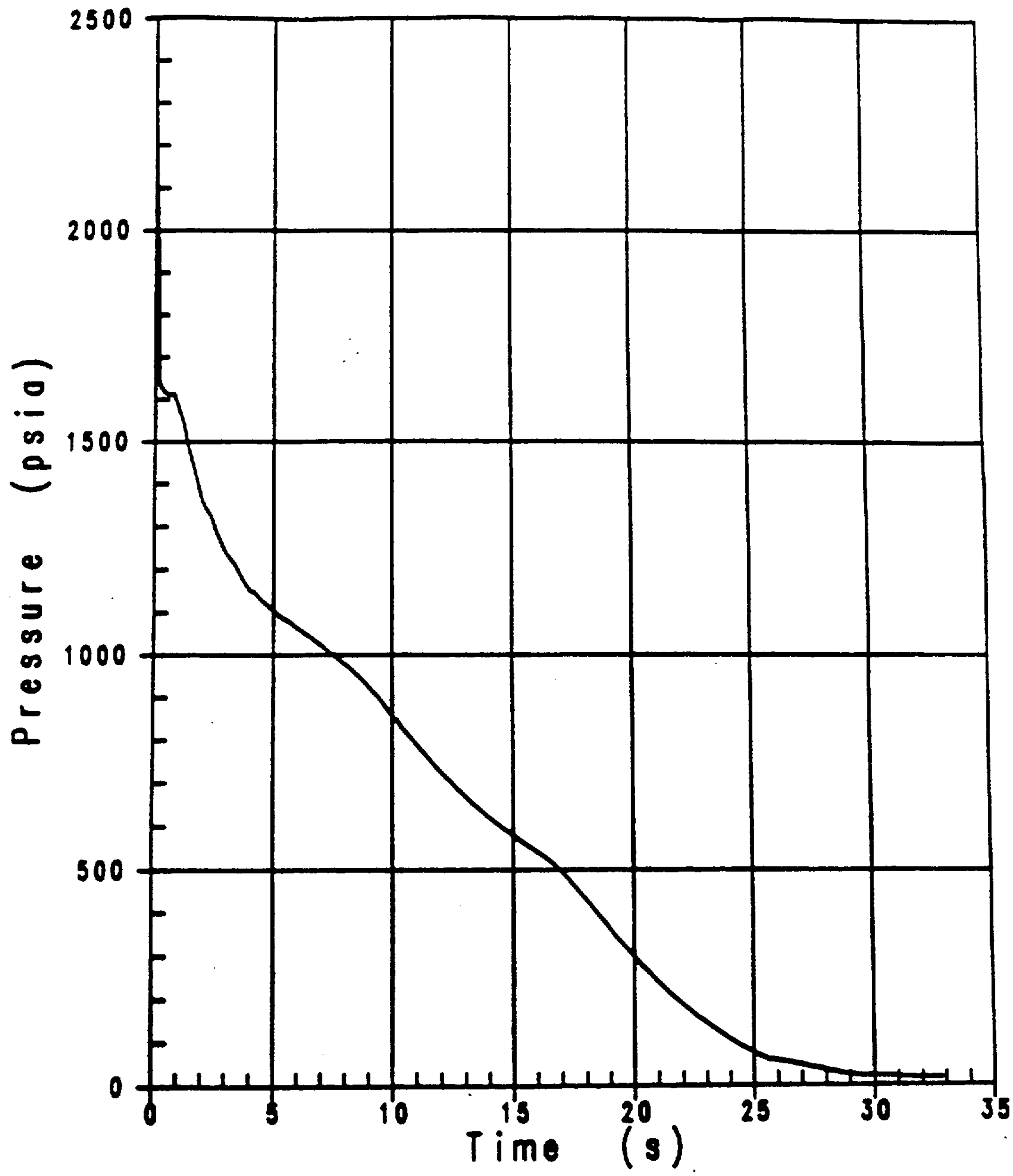

Figure 3.1-1 **Reactor Coolant System Pressure Uprating Analysis–RHR Crosstie Closed**

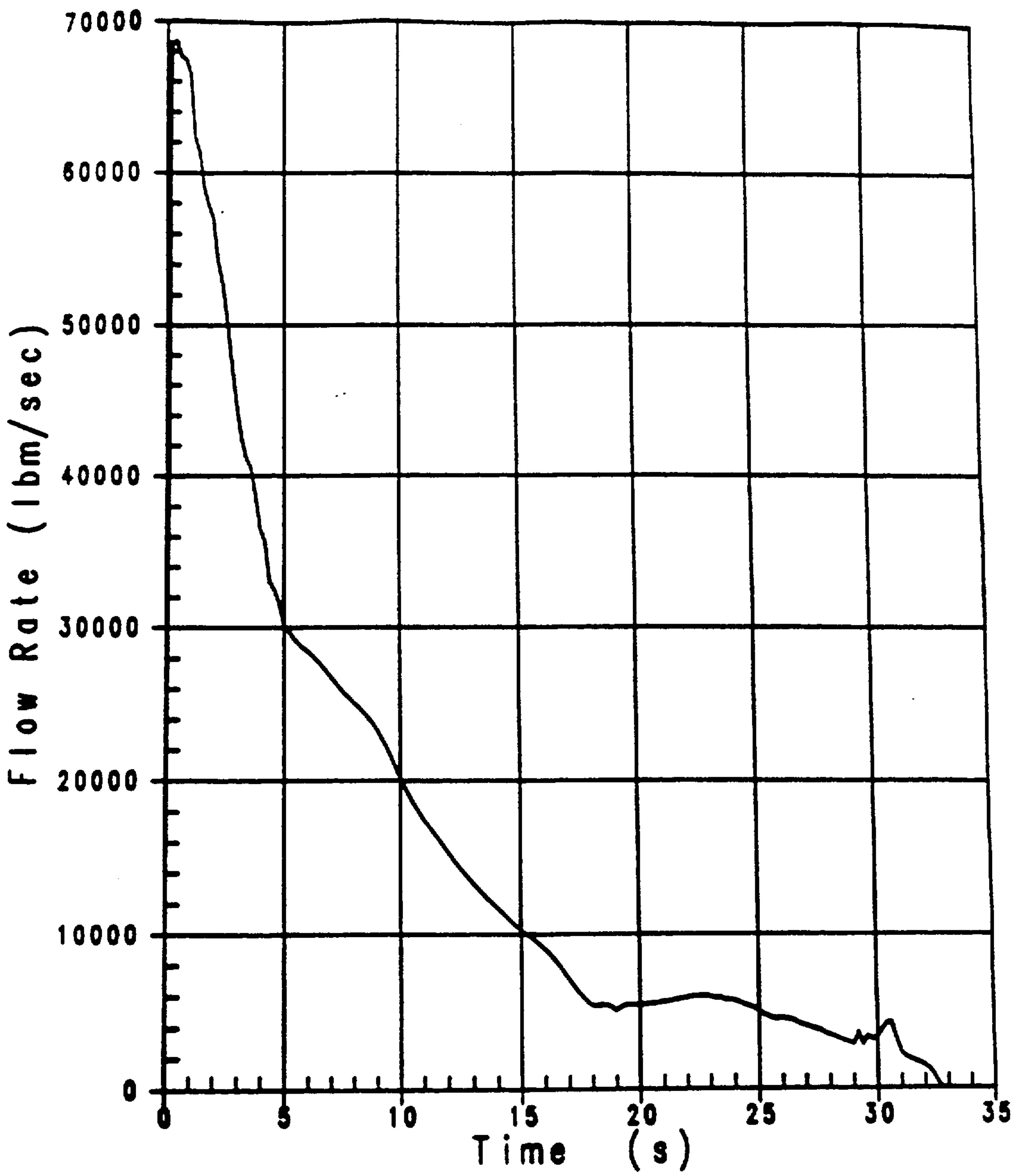

Figure 3.1-2 **Break Flow During Blowdown**
Uprating Analysis–RHR Crosstie Closed

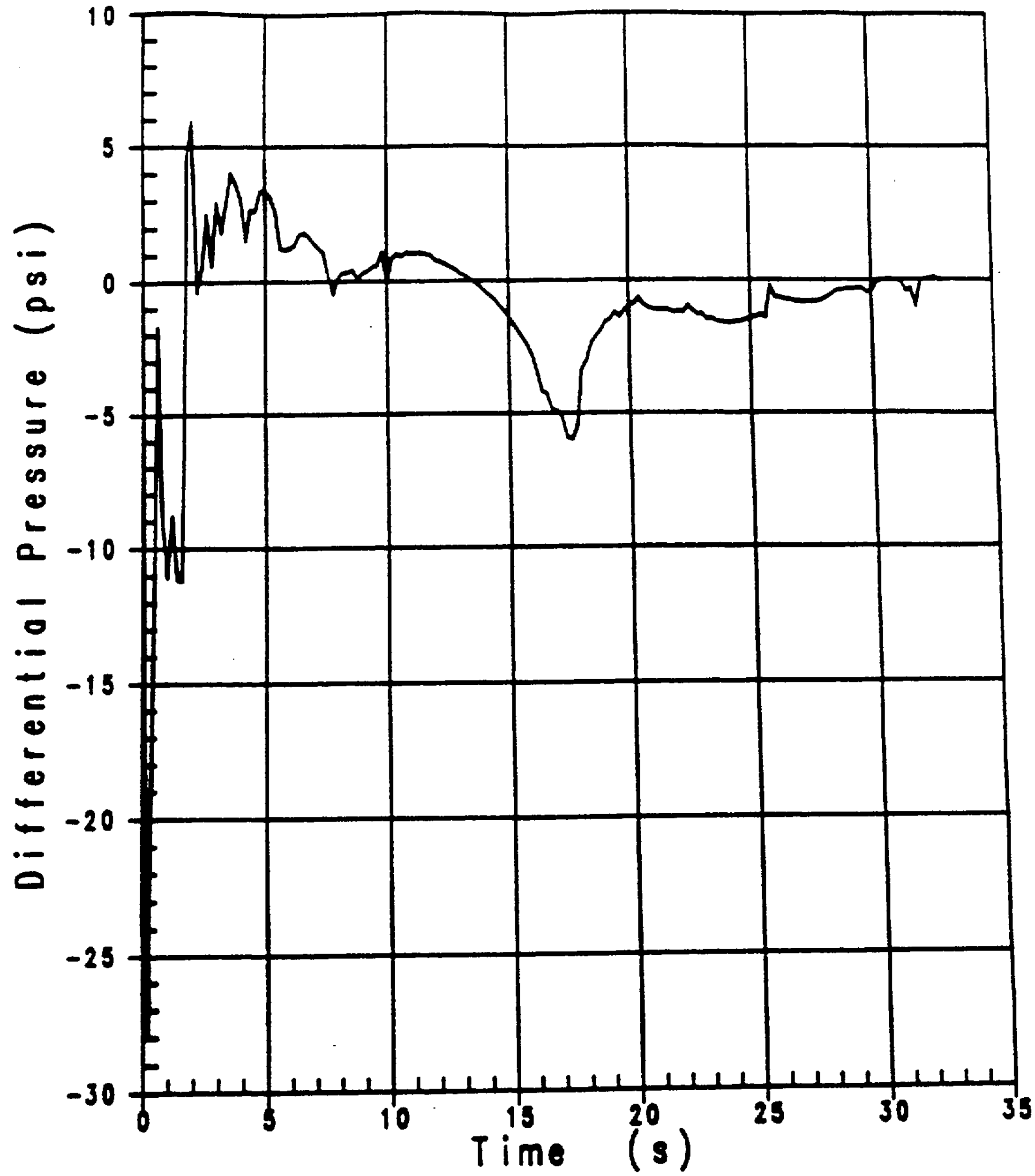

Figure 3.1-3 Core Pressure Drop
Uprating Analysis–RHR Crosstie Closed

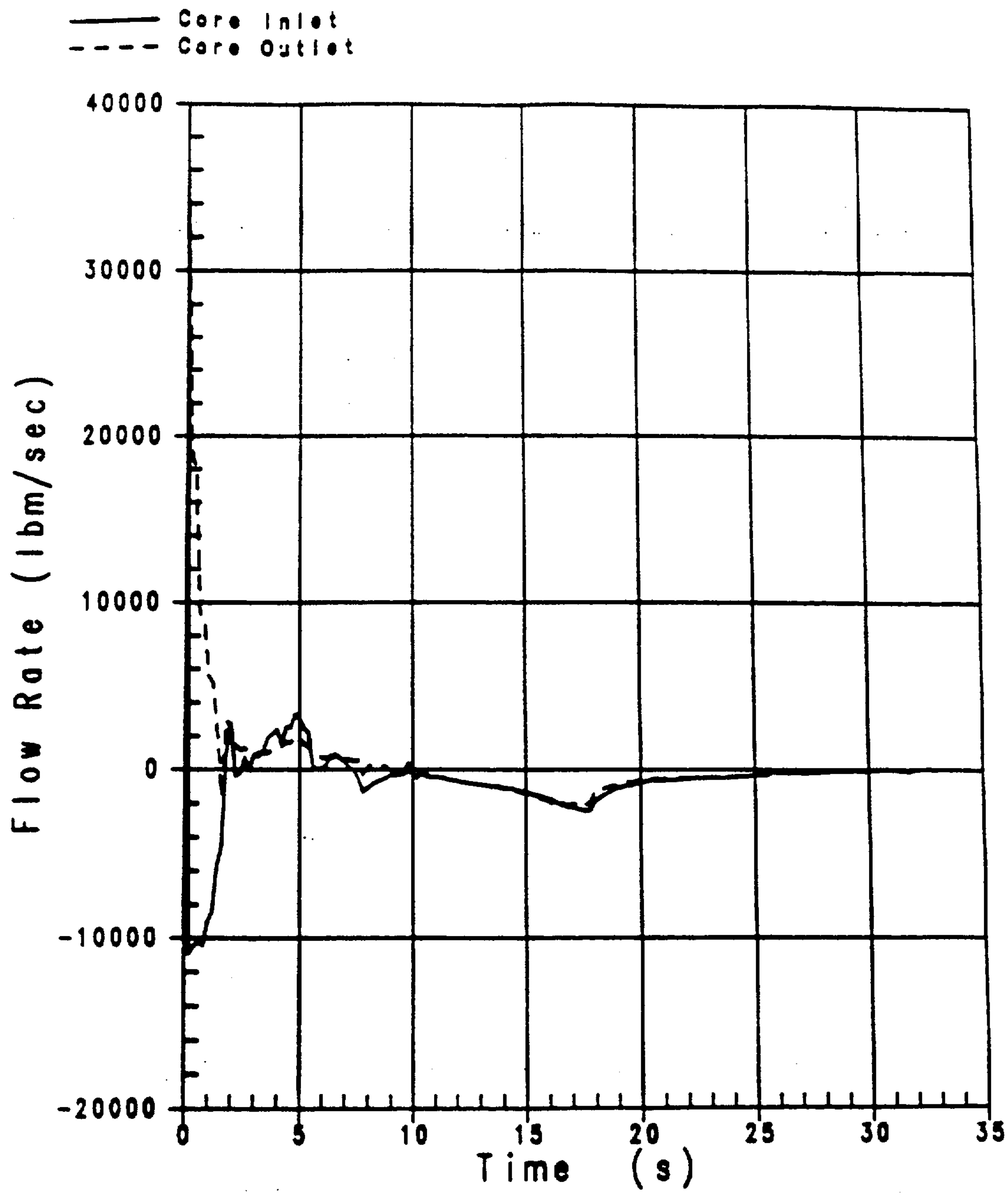

Figure 3.1-4 Core Flowrate
Uprating Analysis–RHR Crosstie Closed

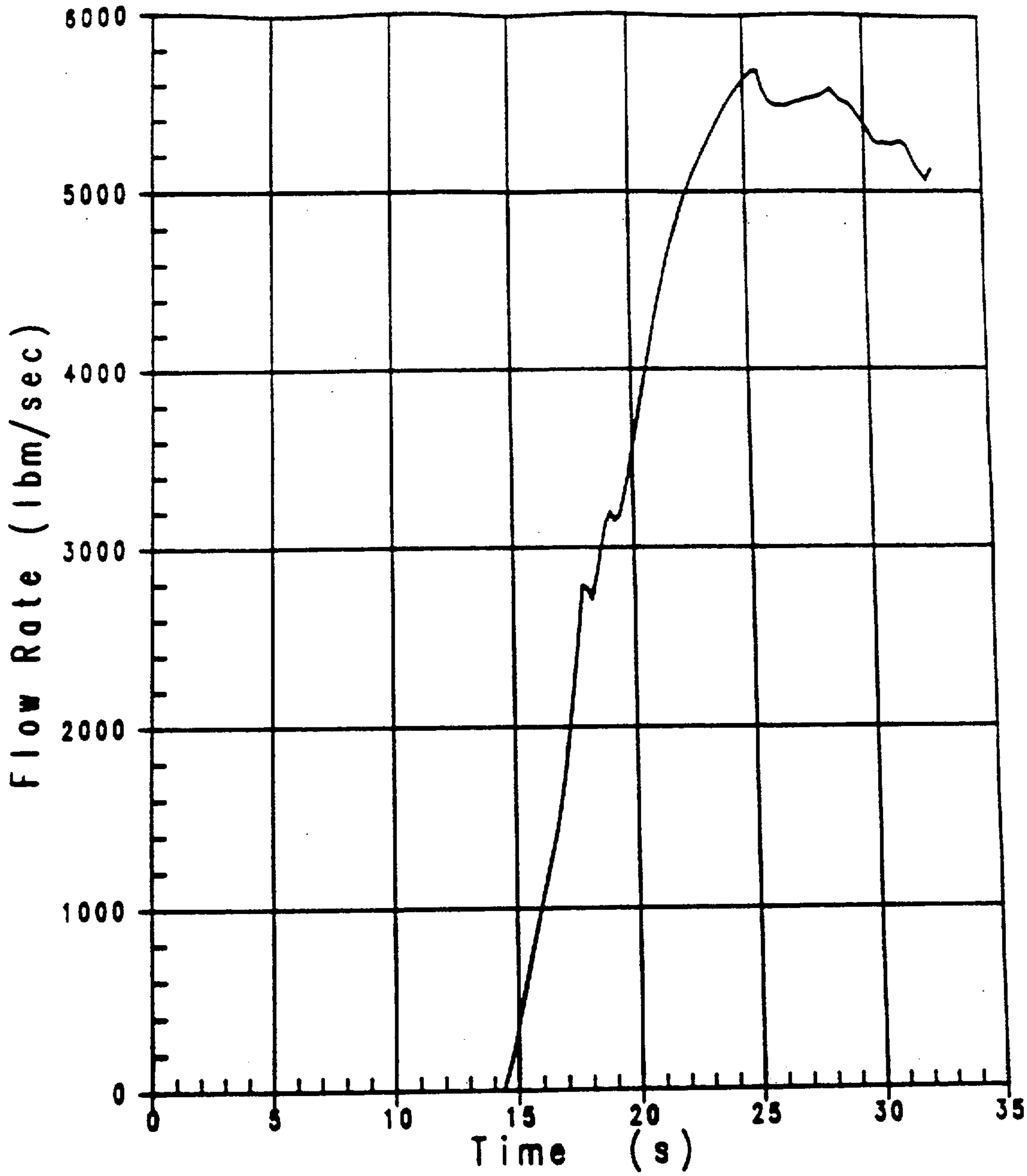

Figure 3.1-5 Accumulator Flow During Blowdown
Uprating Analysis–RHR Crosstie Closed

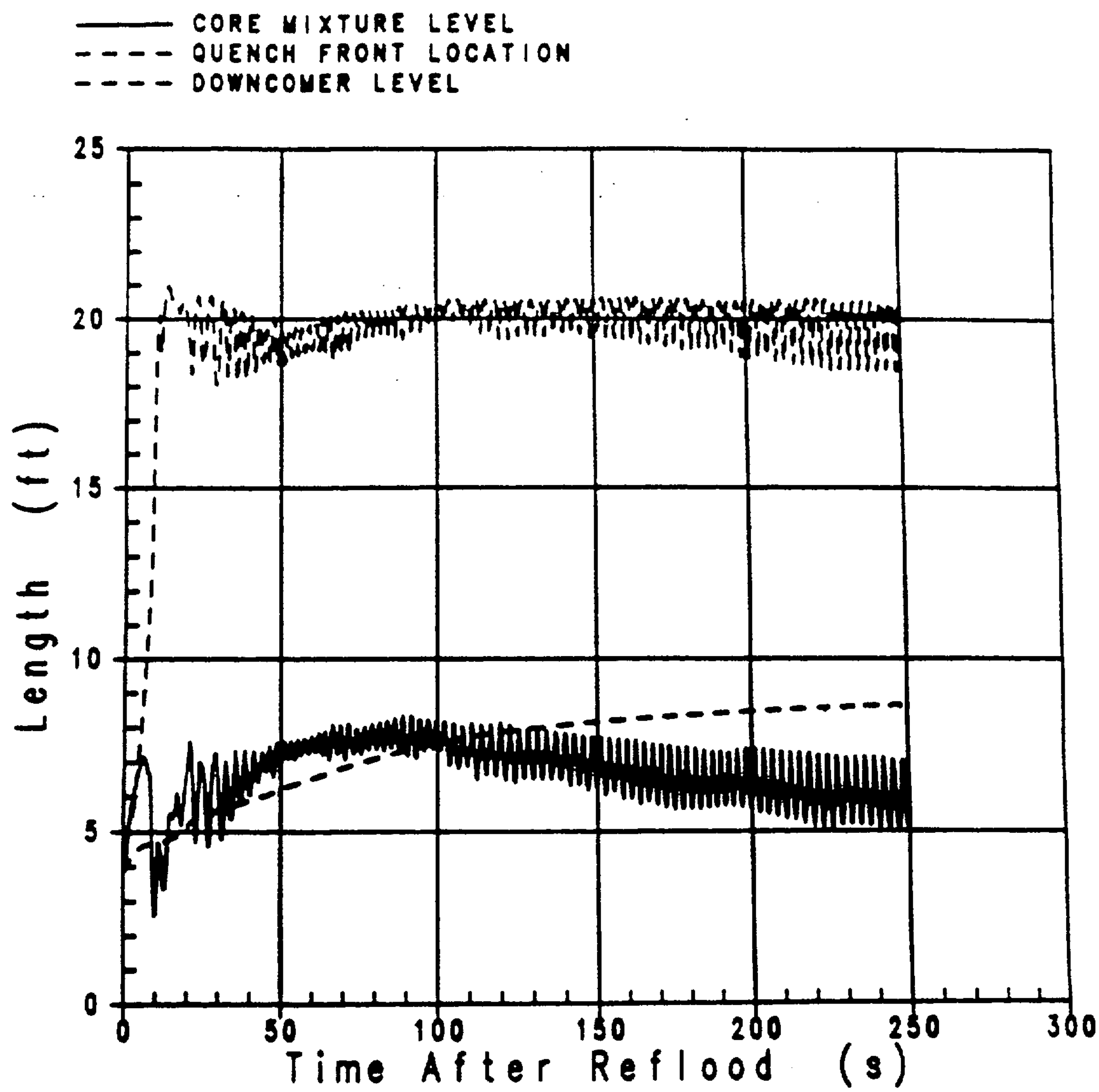

Figure 3.1-6 Vessel Liquid Levels During Reflood
Uprating Analysis–RHR Crosstie Closed

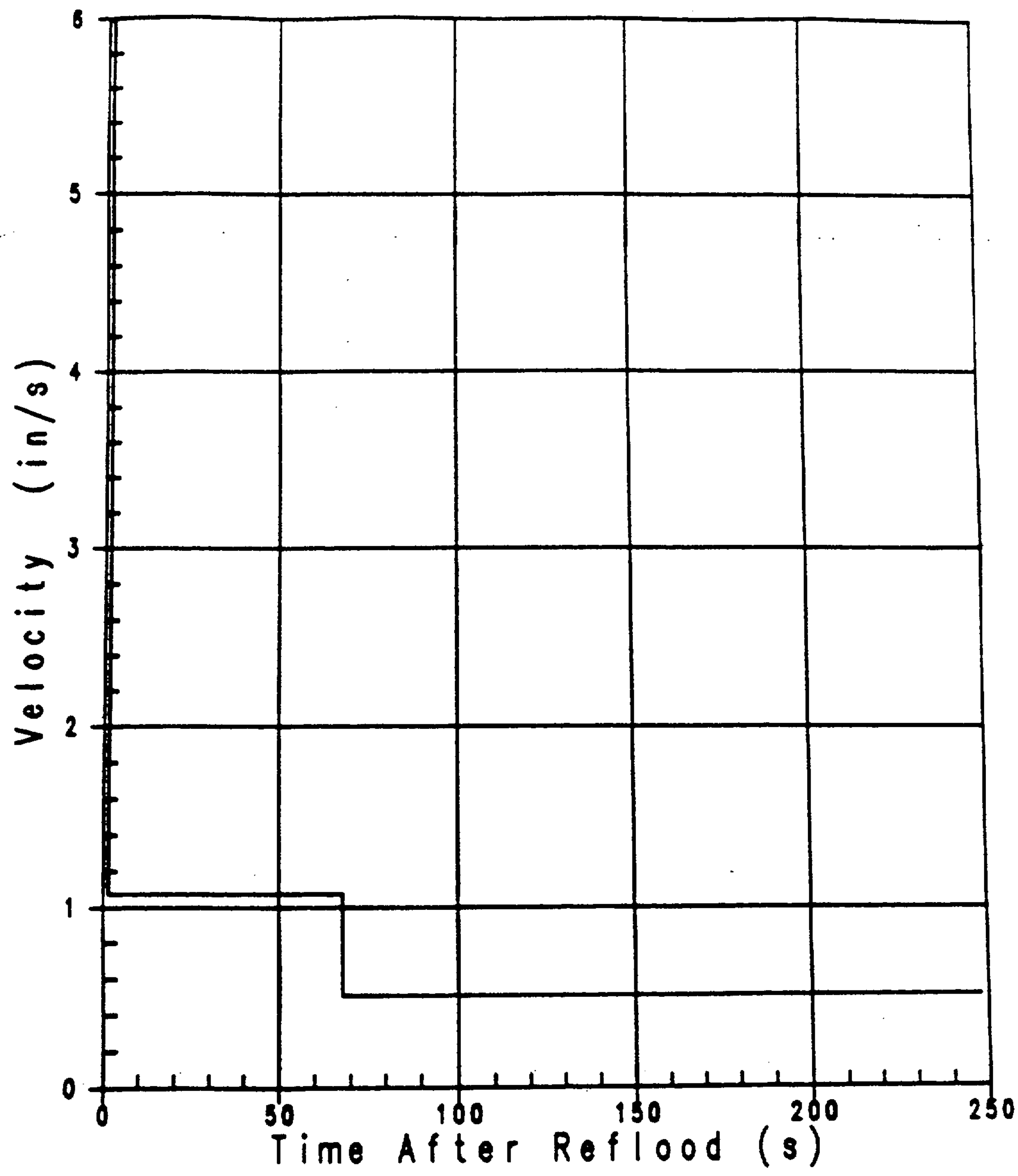

Figure 3.1-7 Core Inlet Flow During Reflood
Uprating Analysis–RHR Crosstie Closed

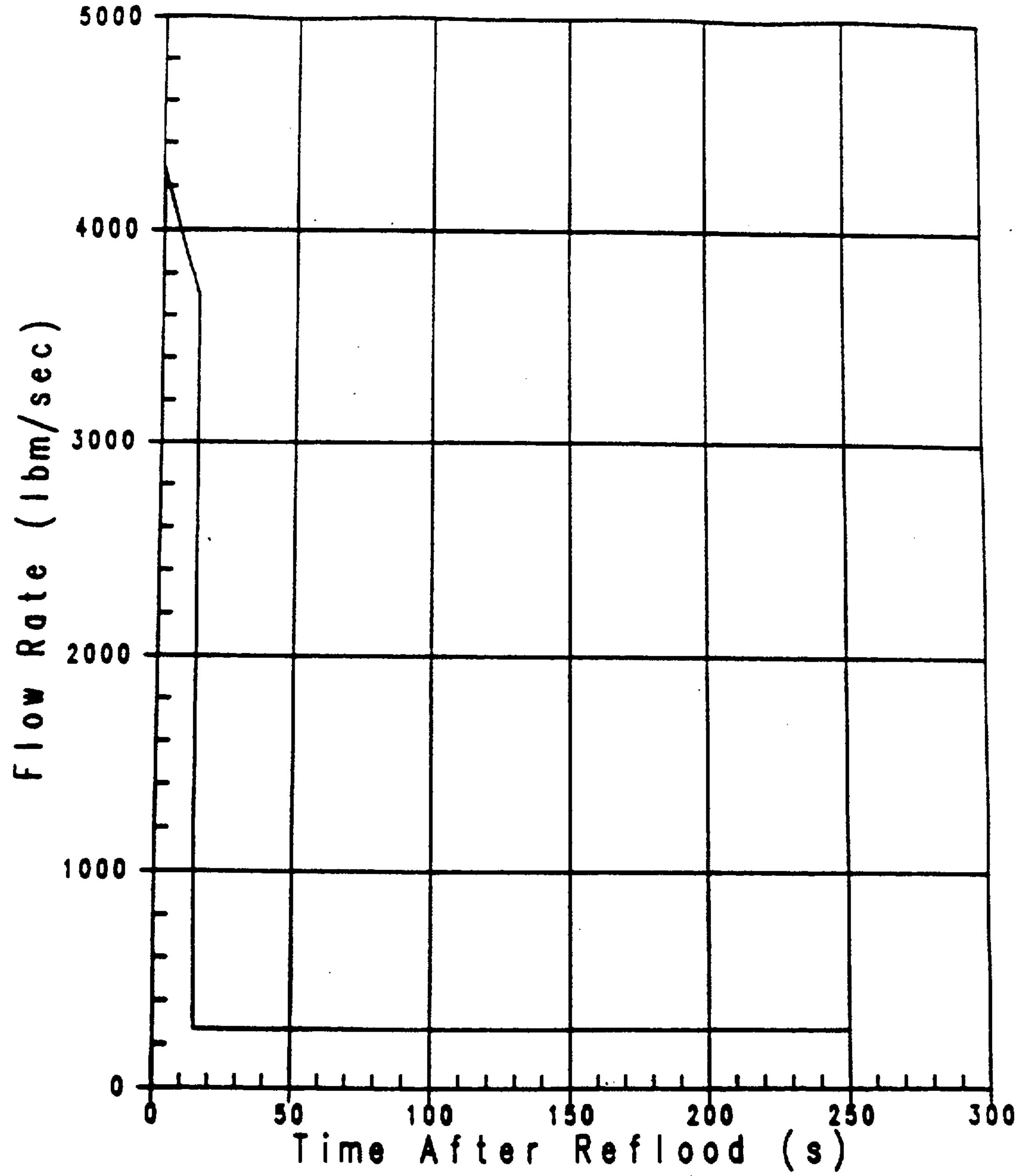

Figure 3.1-8 Accumulator and SI Flow During Reflood Uprating Analysis–RHR Crosstie Closed

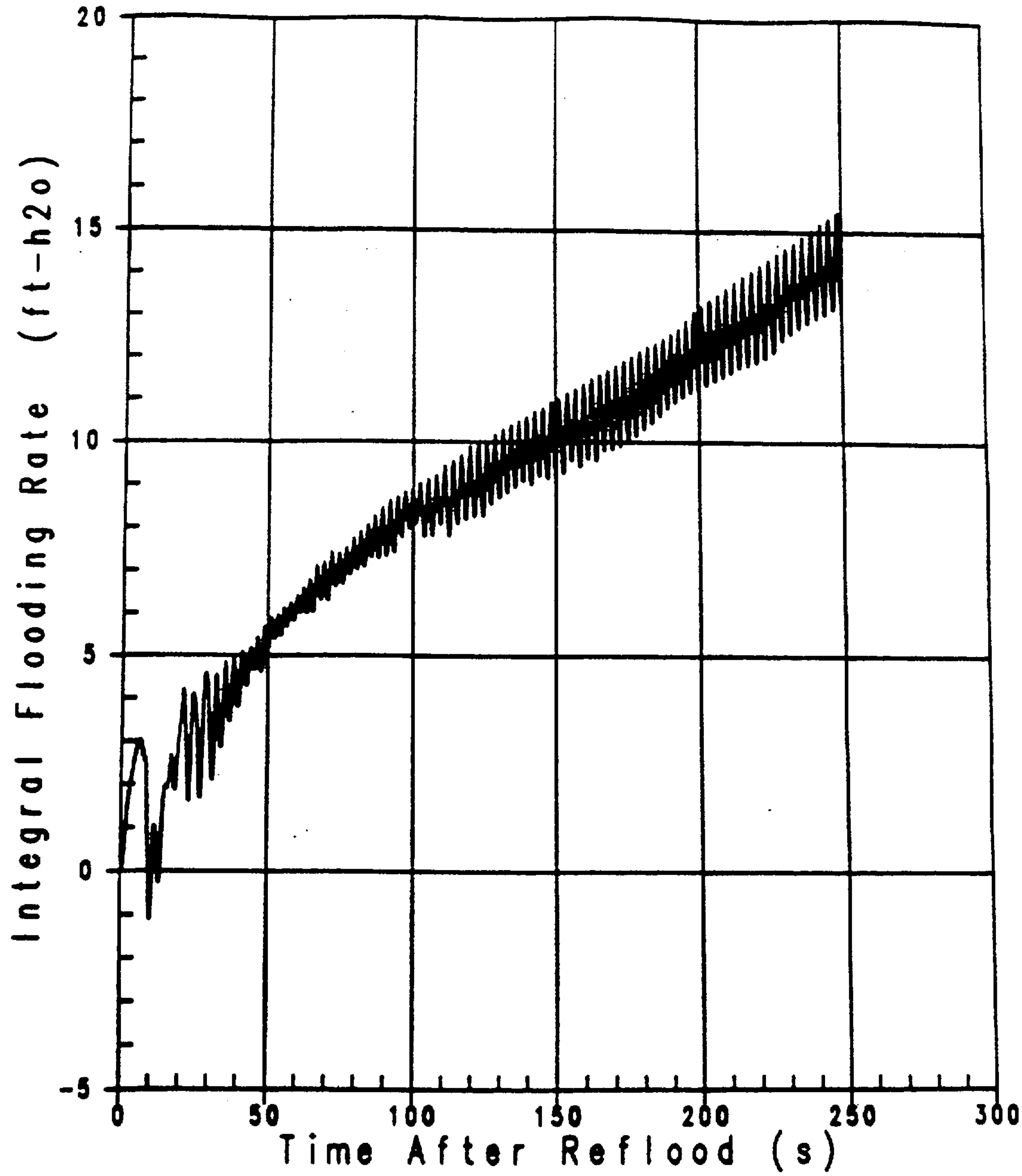

Figure 3.1-9 Integral of Core Inlet Flow
Uprating Analysis–RHR Crosstie Closed

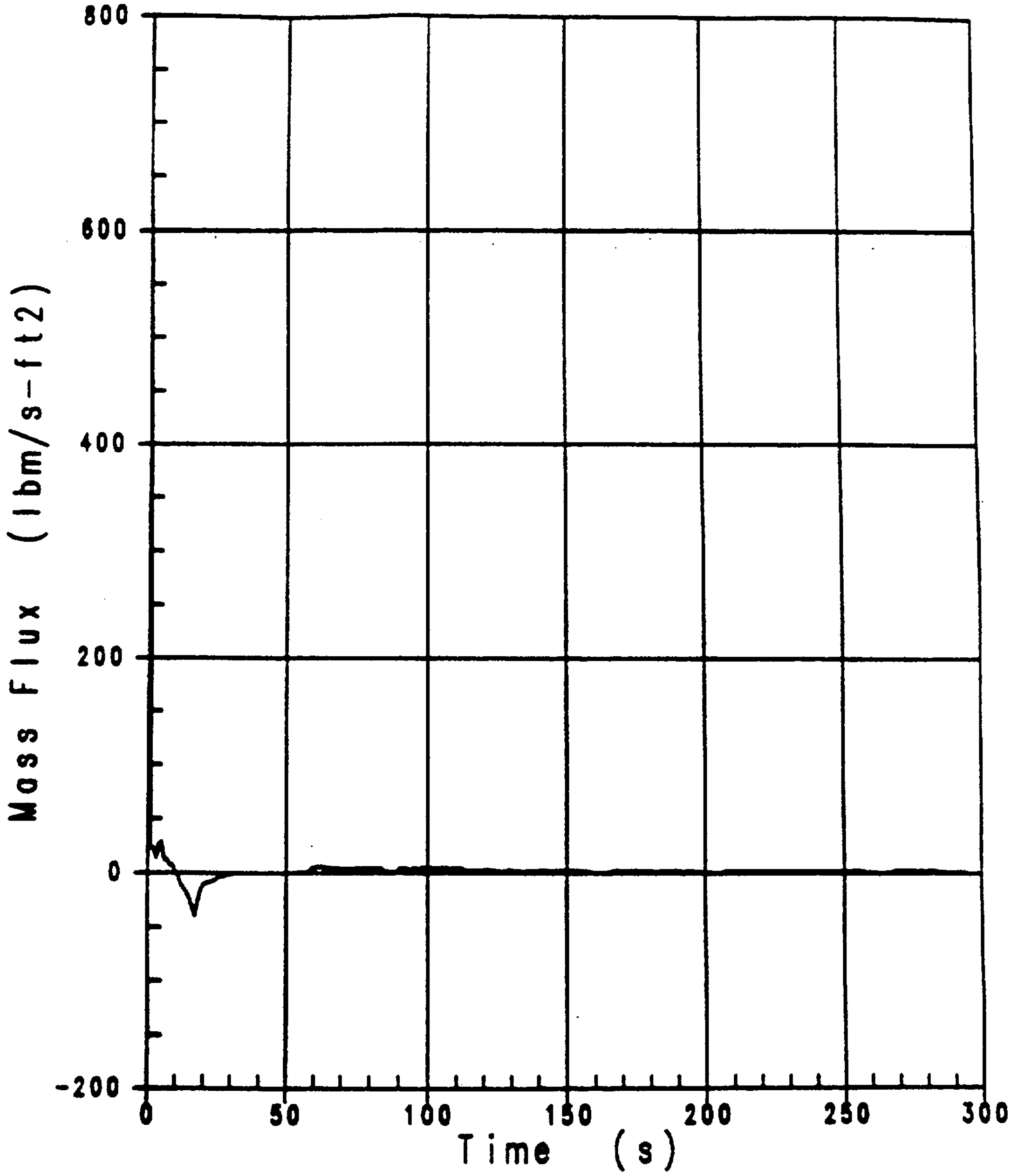

Figure 3.1-10 Mass Flux at Peak Temperature Elevation
Uprating Analysis–RHR Crosstie Closed

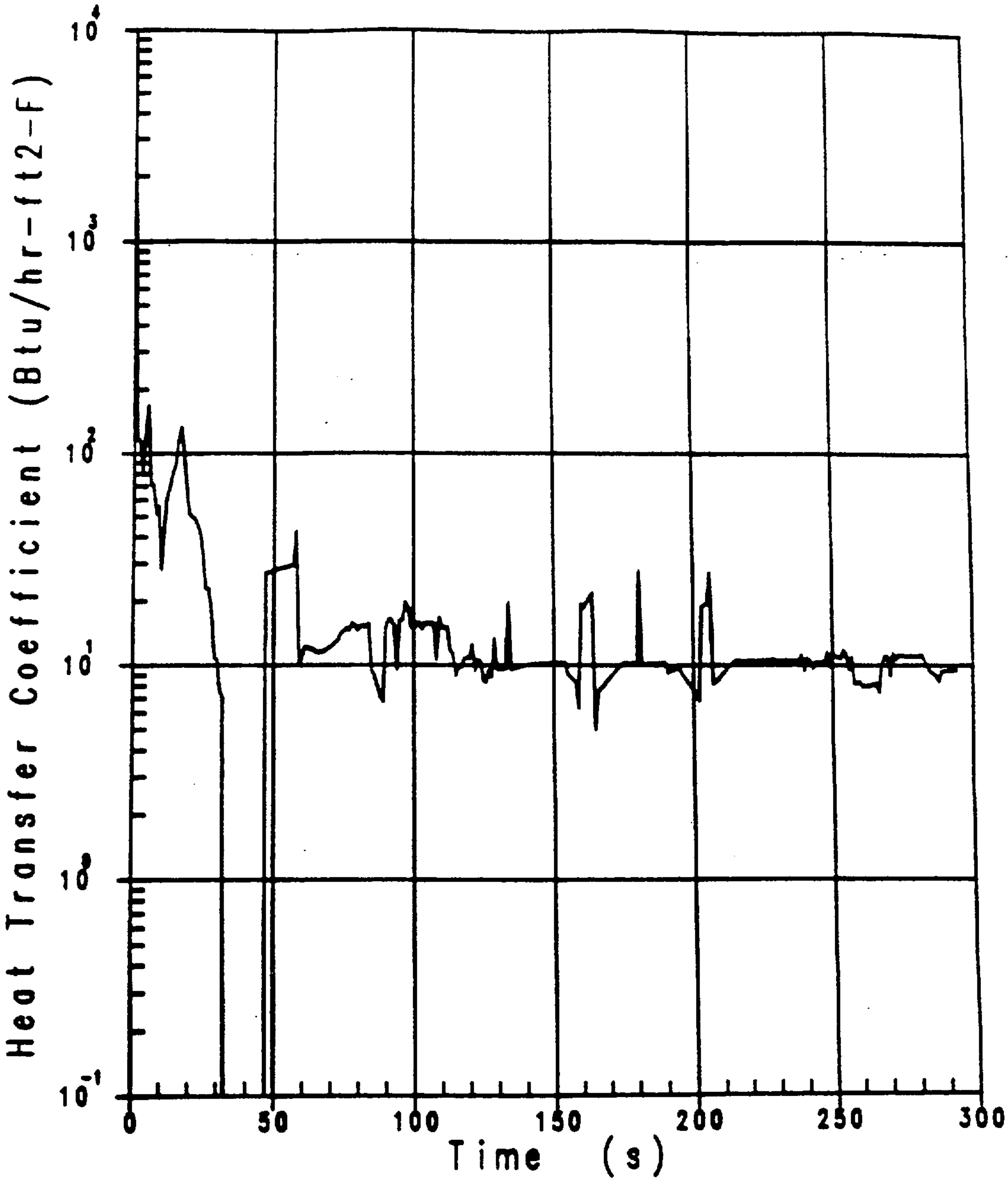

Figure 3.1-11 Rod H.T.C. at Peak Temperature Elevation
Uprating Analysis–RHR Crosstie Closed

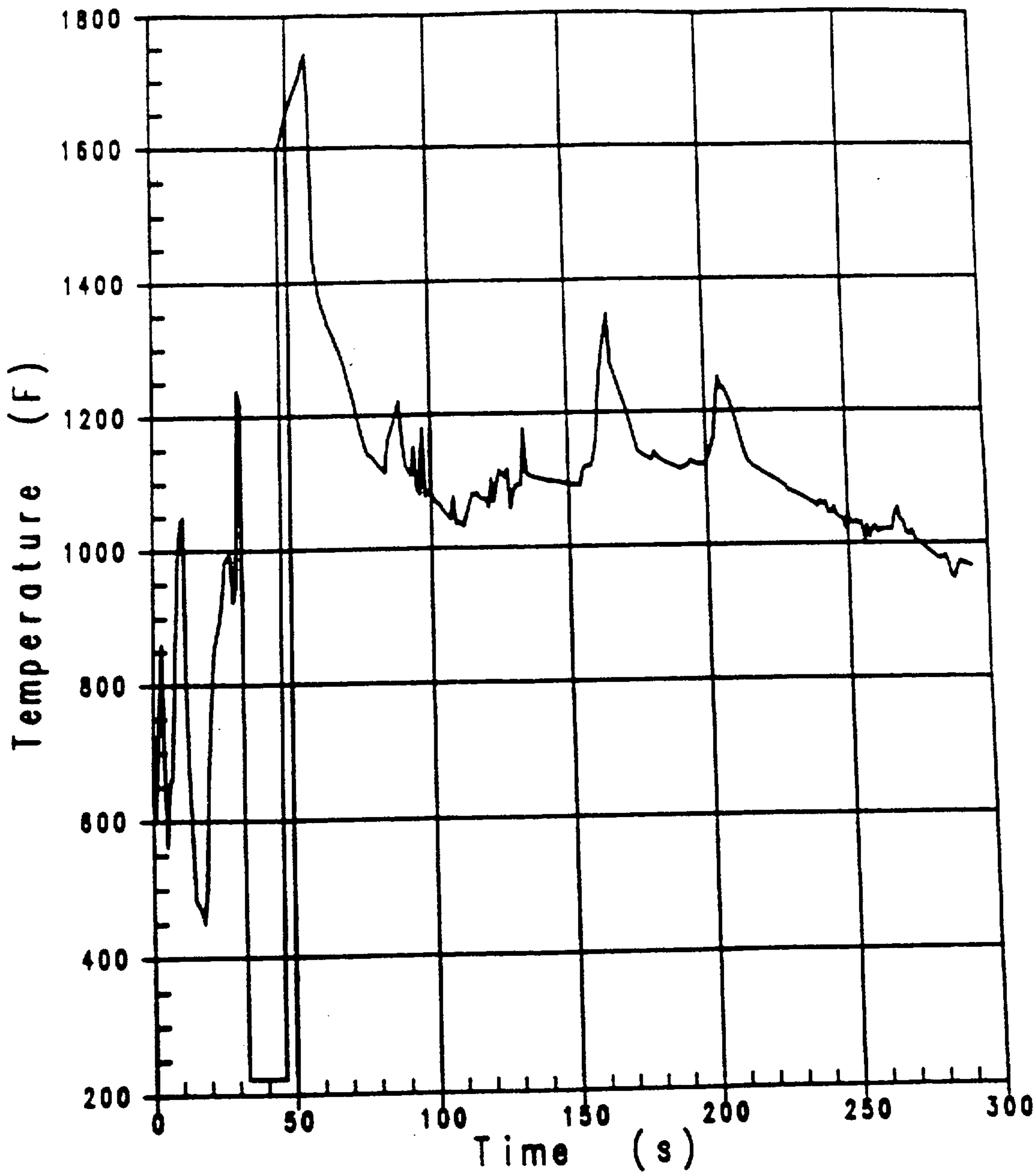

Figure 3.11-12 Vapor Temperature
Uprating Analysis–RHR Crosstie Closed

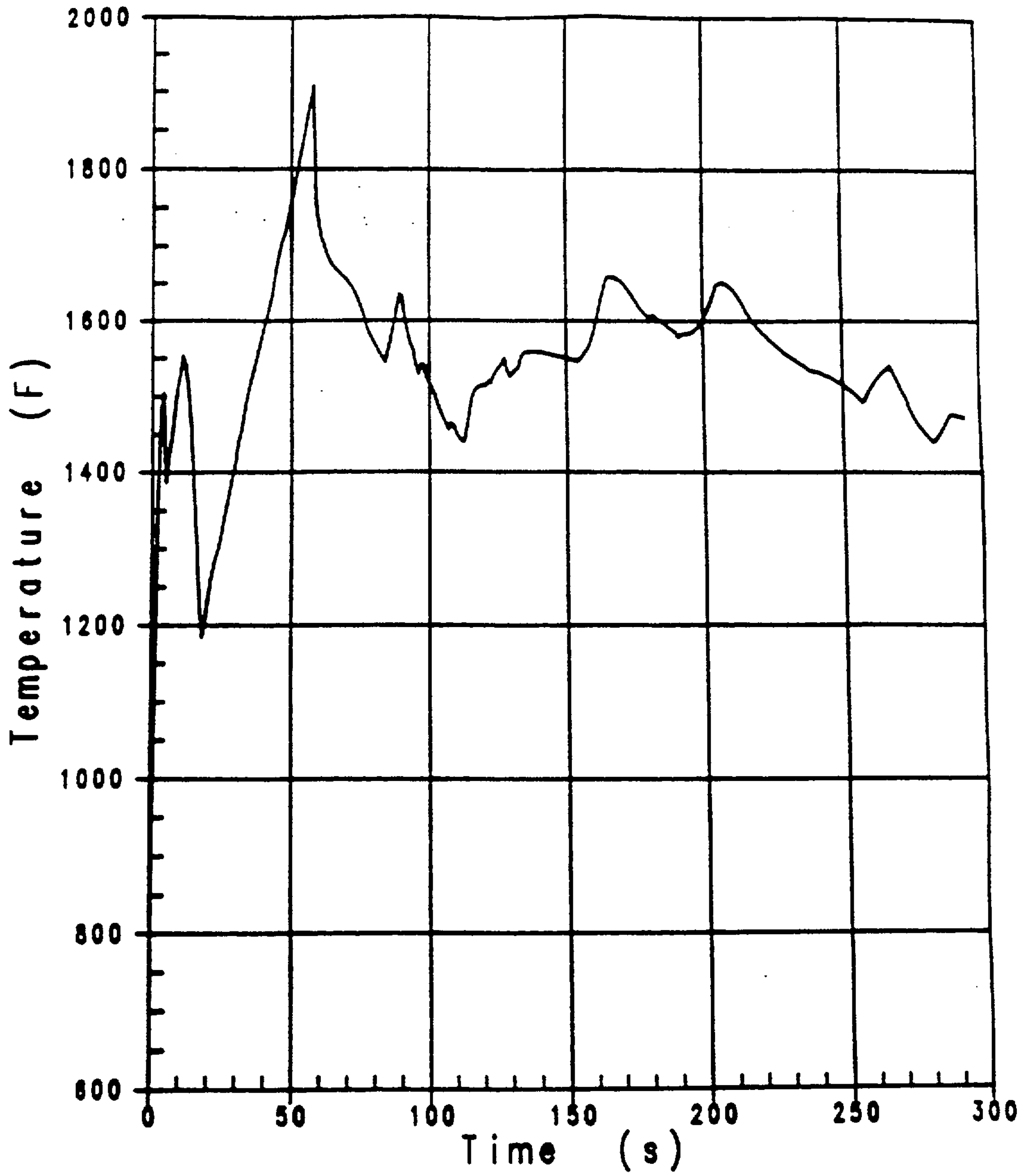

Figure 3.1-13 Fuel Rod Peak Clad Temperature
Uprating Analysis–RHR Crosstie Closed

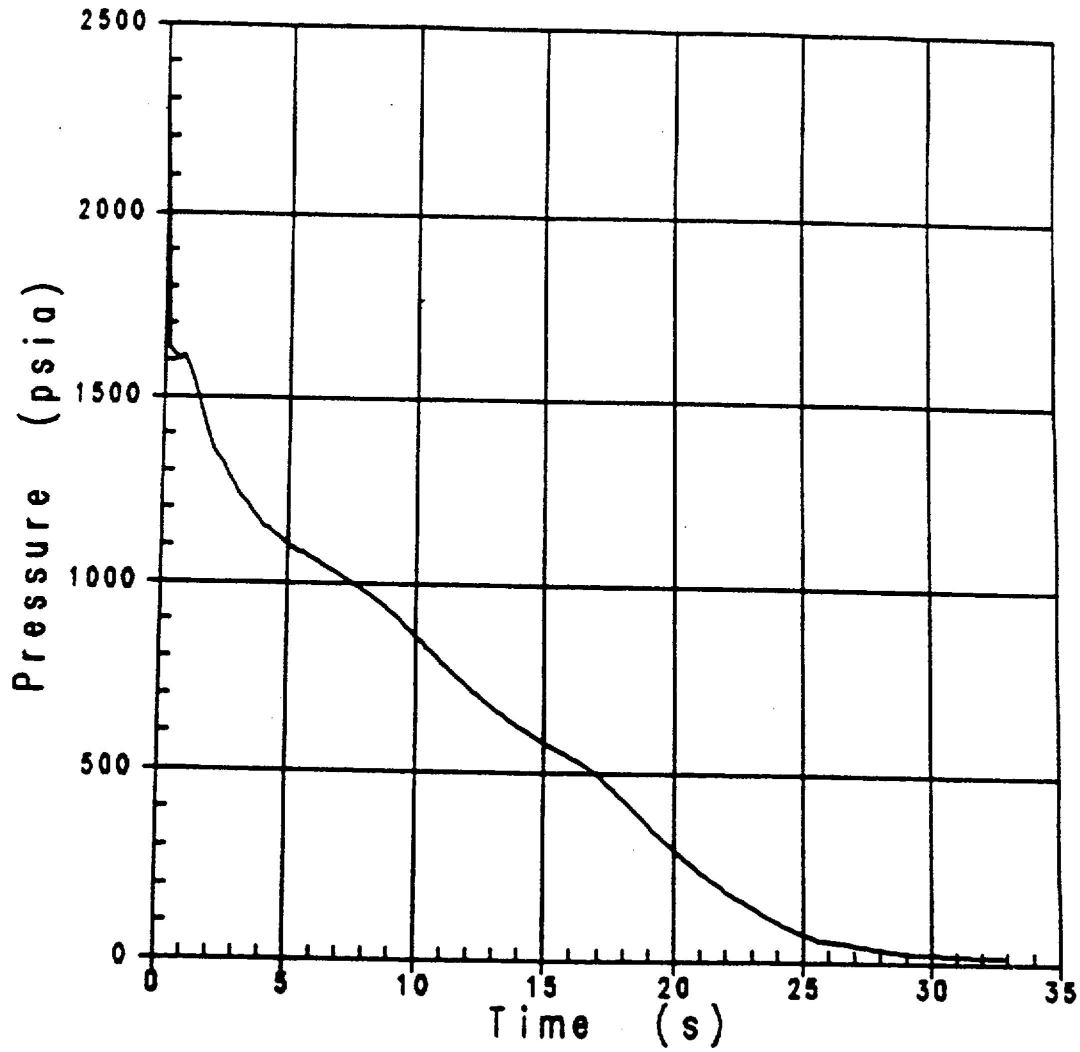

Figure 3.1-14 Reactor Coolant System Pressure
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

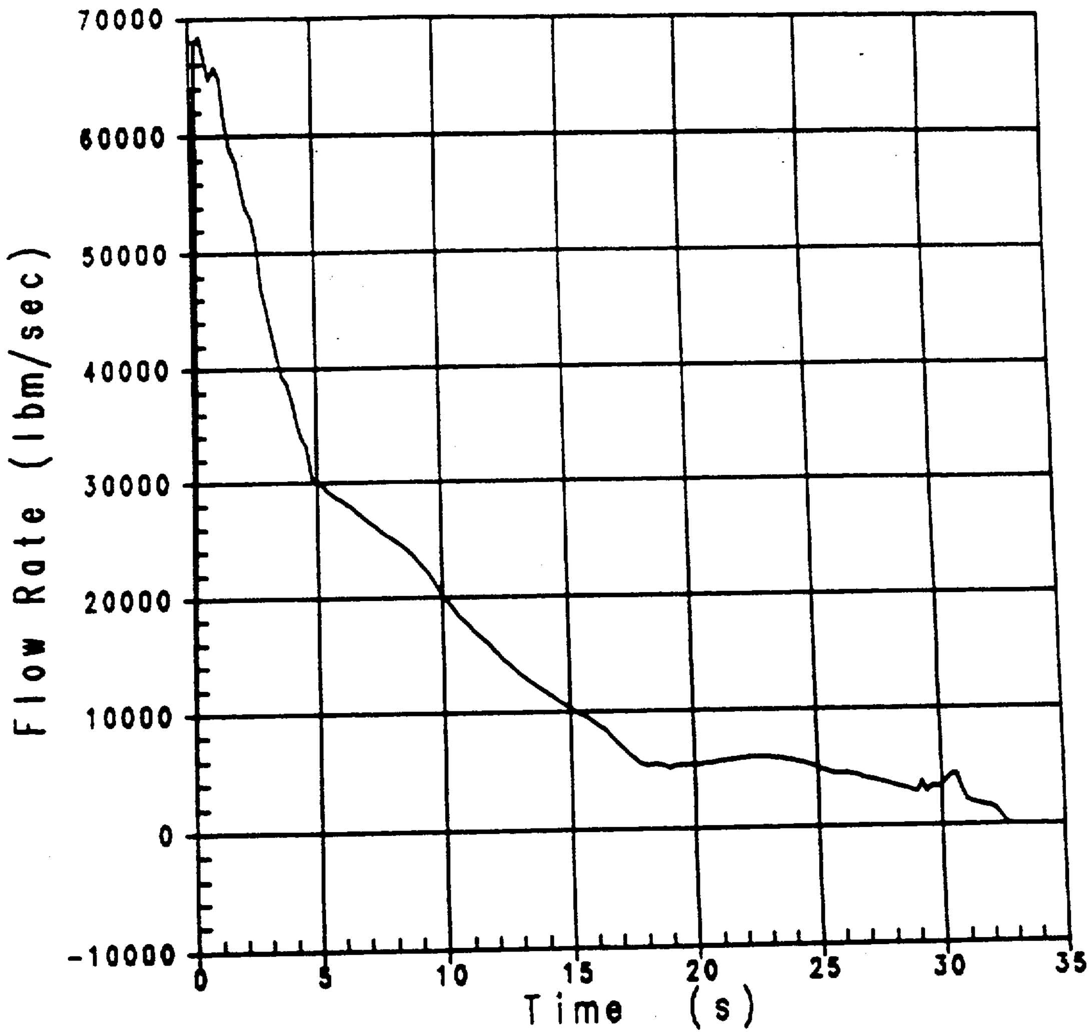

Figure 3.1-15 Break Flow During Blowdown
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

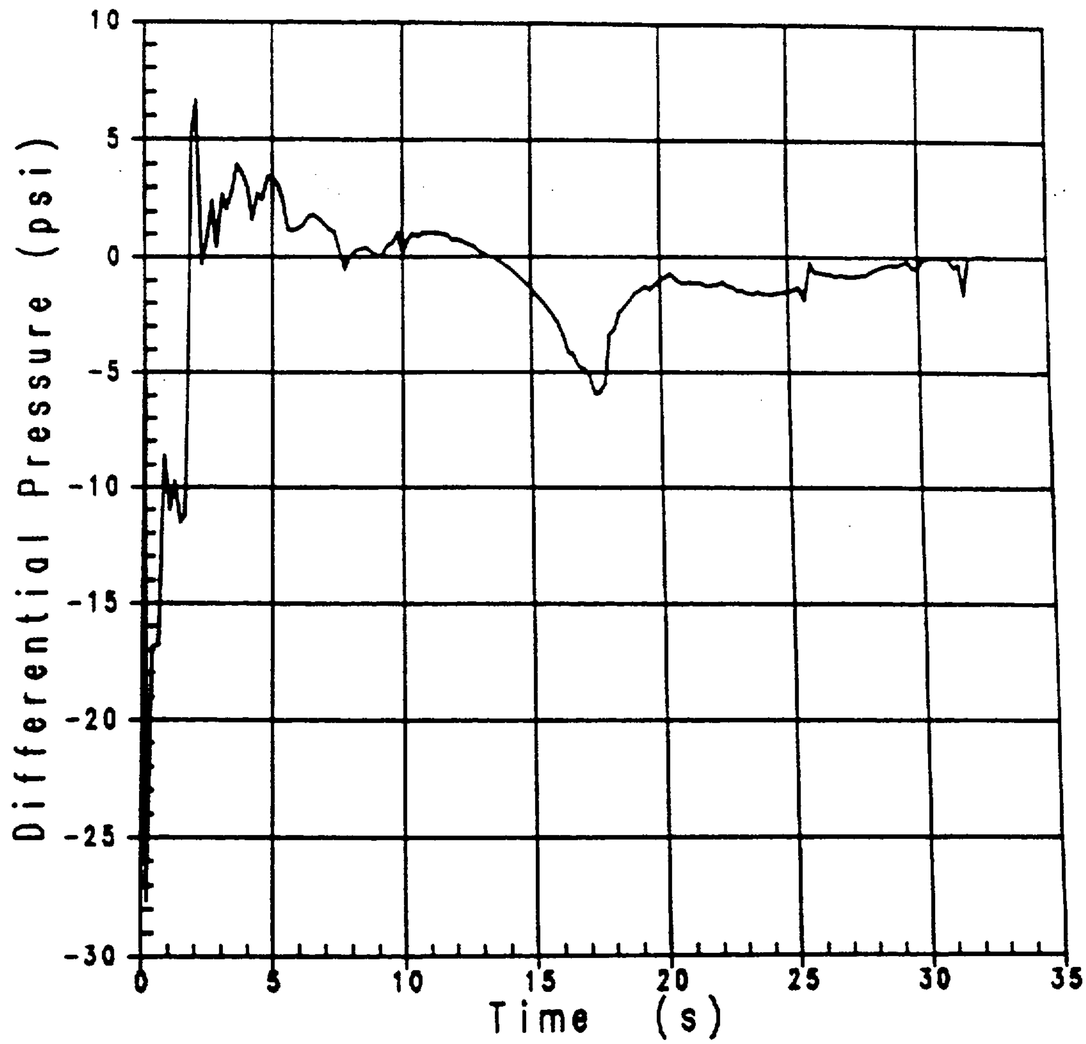

Figure 3.1-16 Core Pressure Drop
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

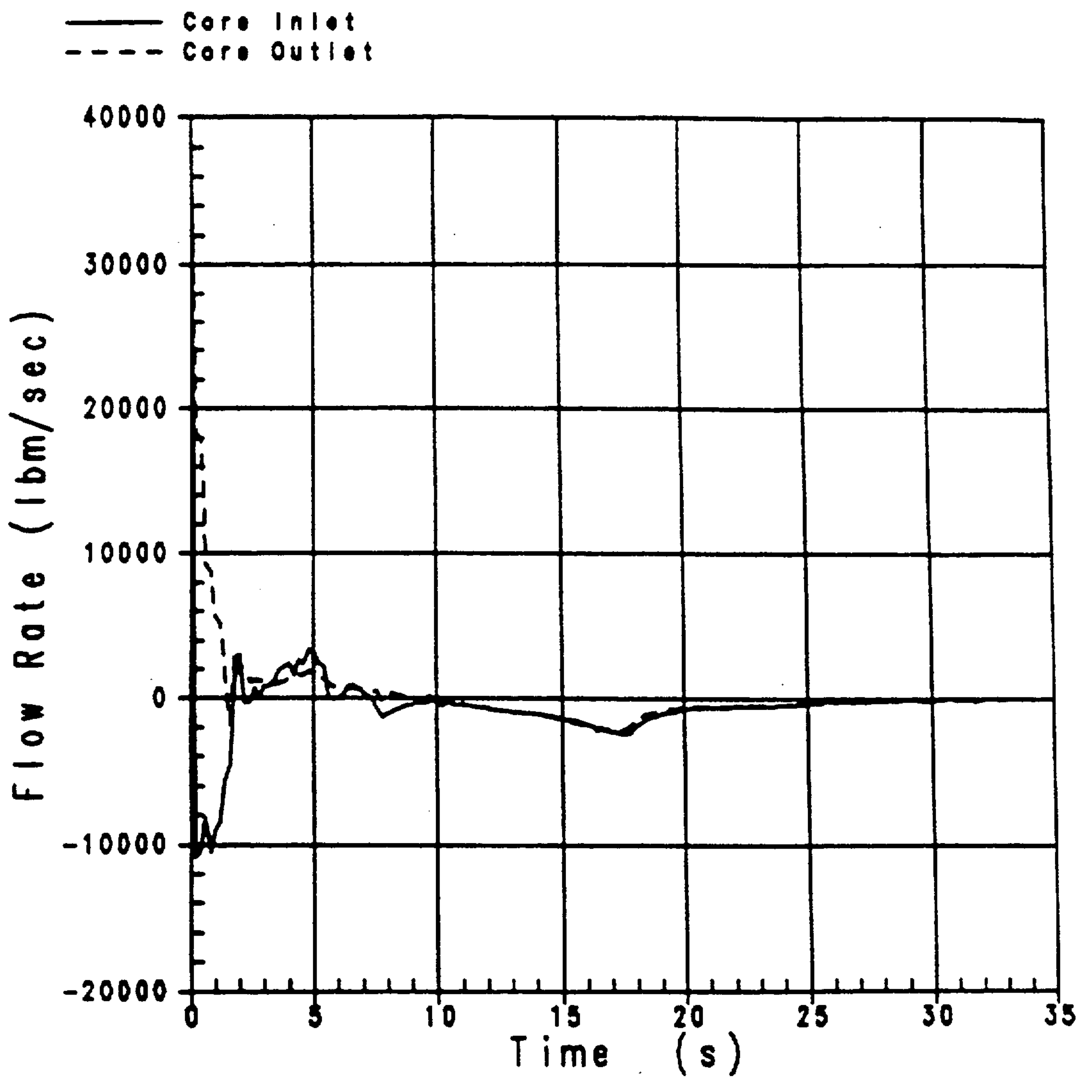

Figure 3.1-17 Core Flowrate
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

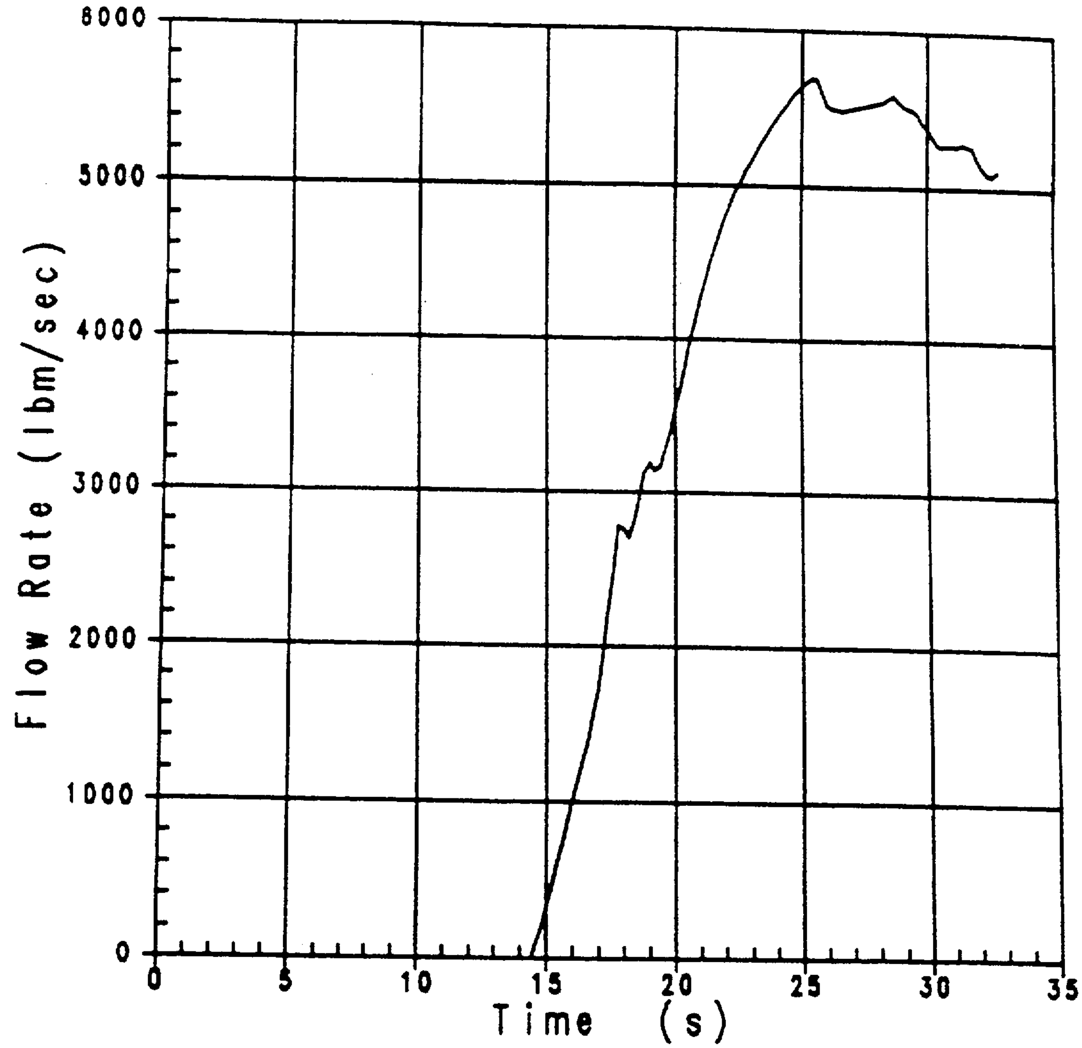

Figure 3.1-18 Accumulator Flow During Blowdown
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

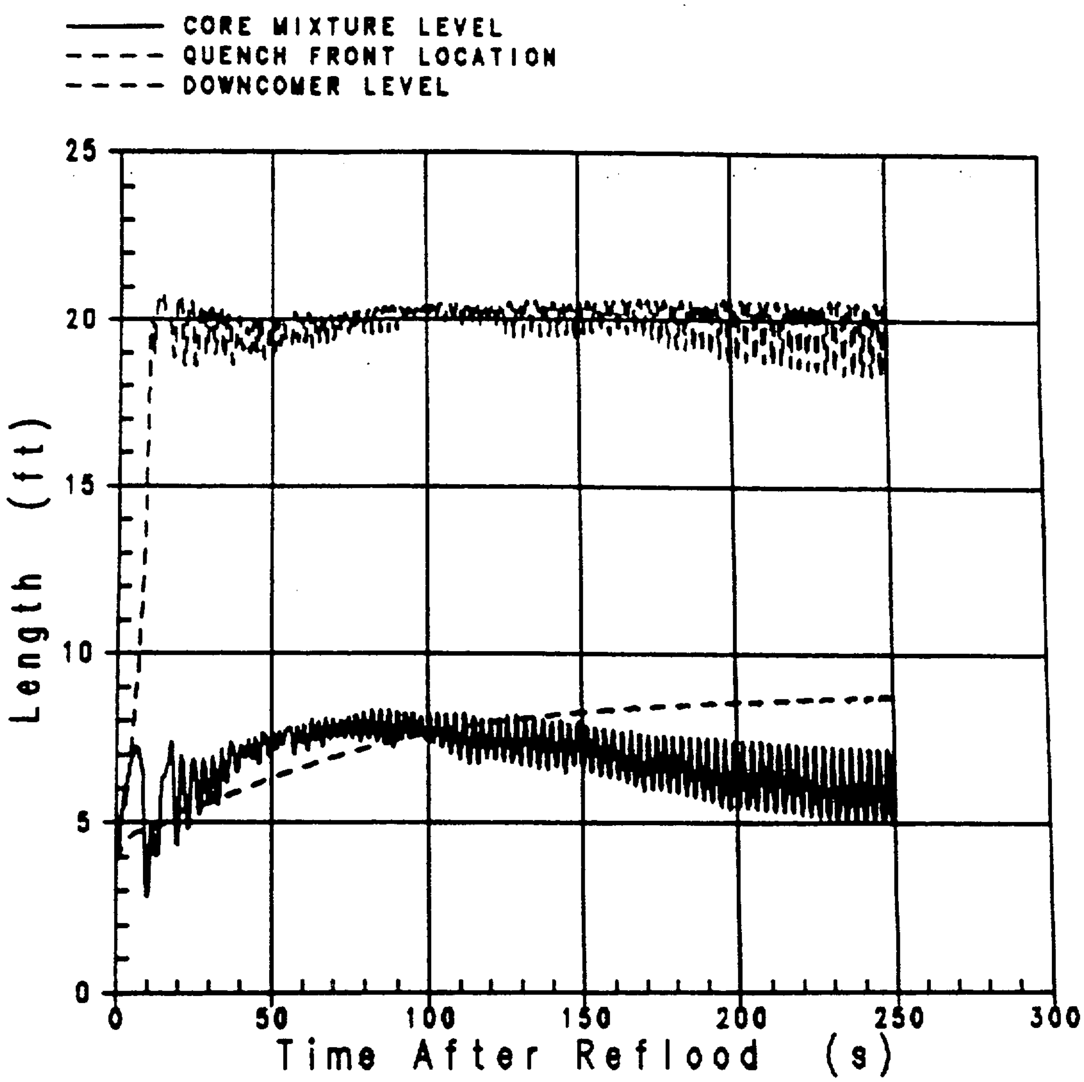

Figure 3.1-19 Vessel Liquid Levels During Reflood
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

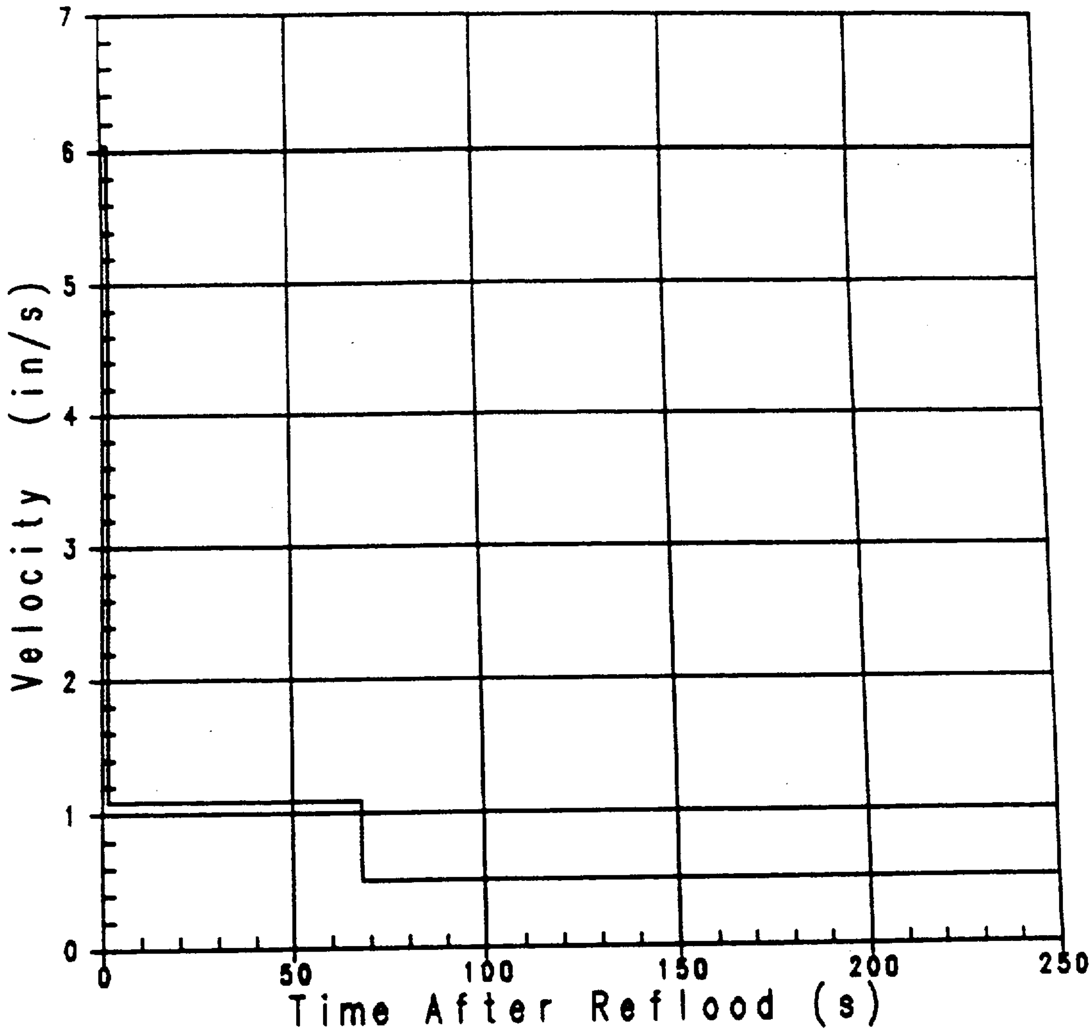

Figure 3.1-20 Core Inlet Flow During Reflood
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

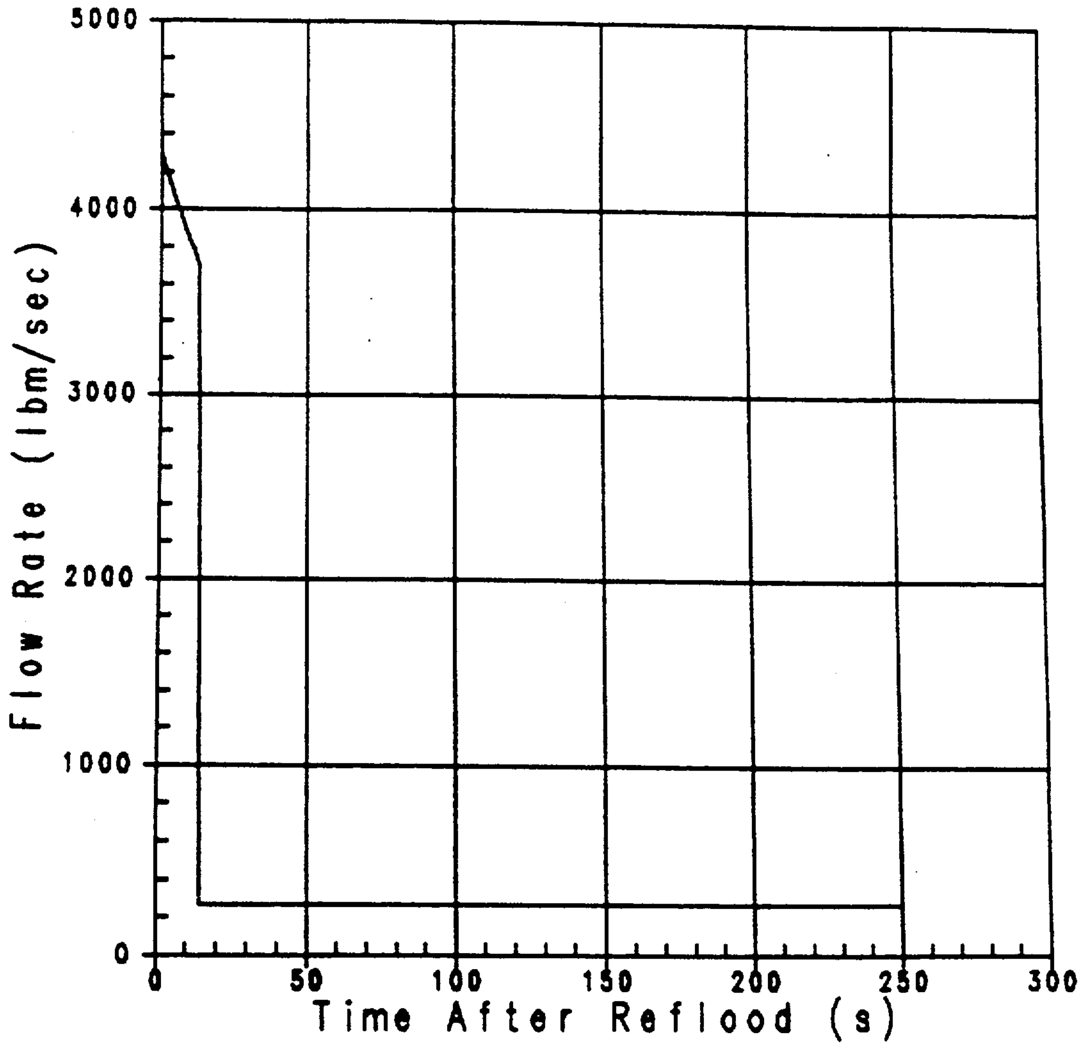

Figure 3.1-21 Accumulator and SI Flow During Reflood
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

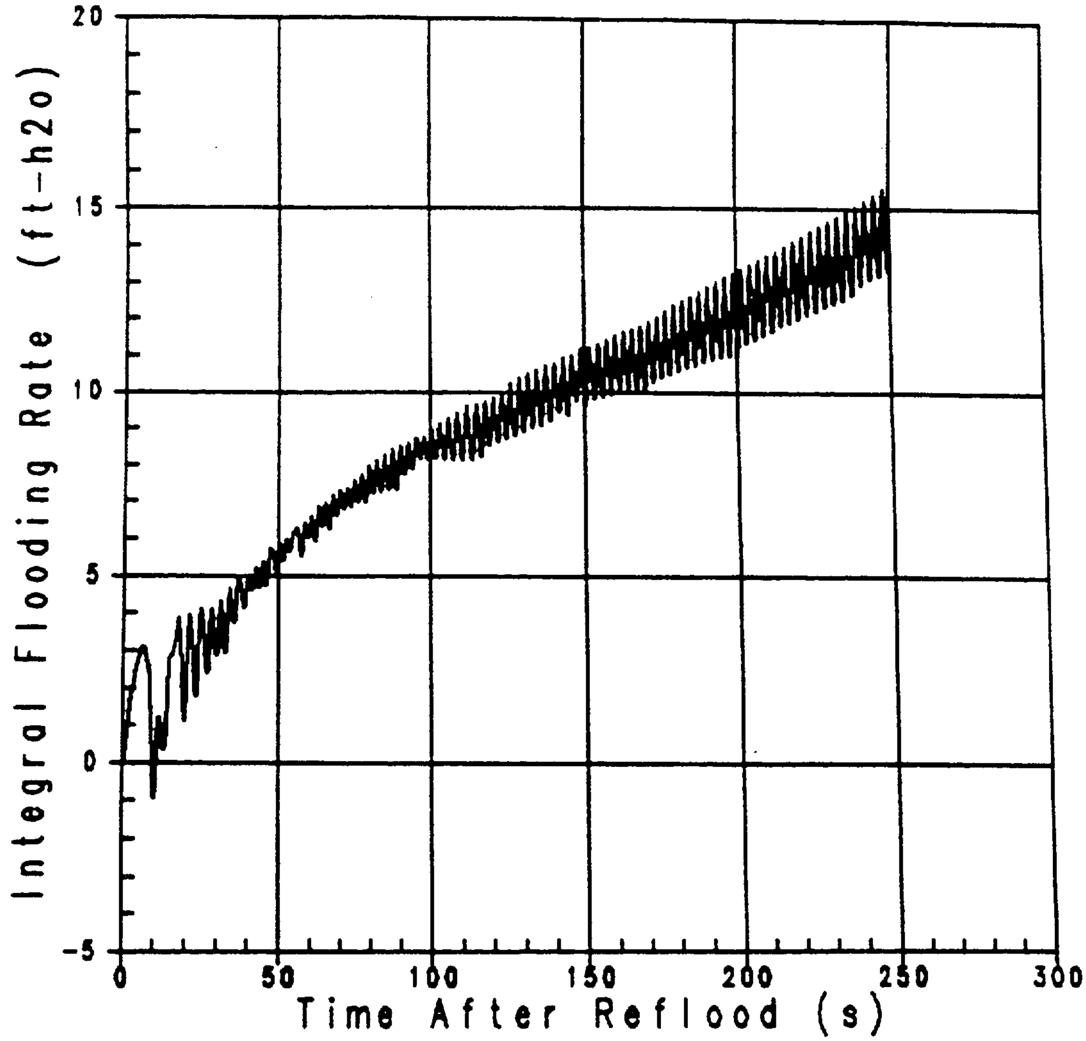

Figure 3.1-22 Integral of Core Inlet Flow
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

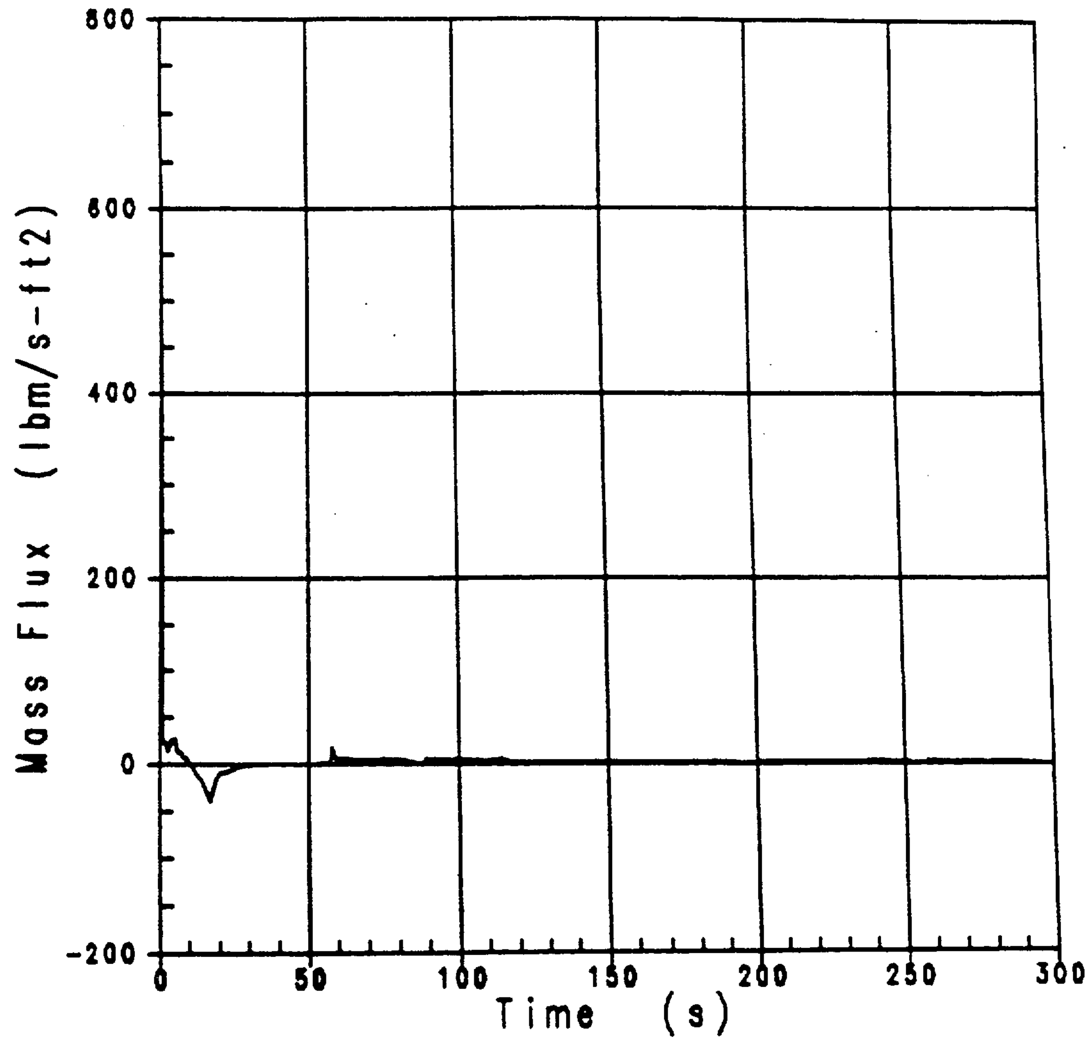

Figure 3.1-23 Mass Flux at Peak Temperature Elevation
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

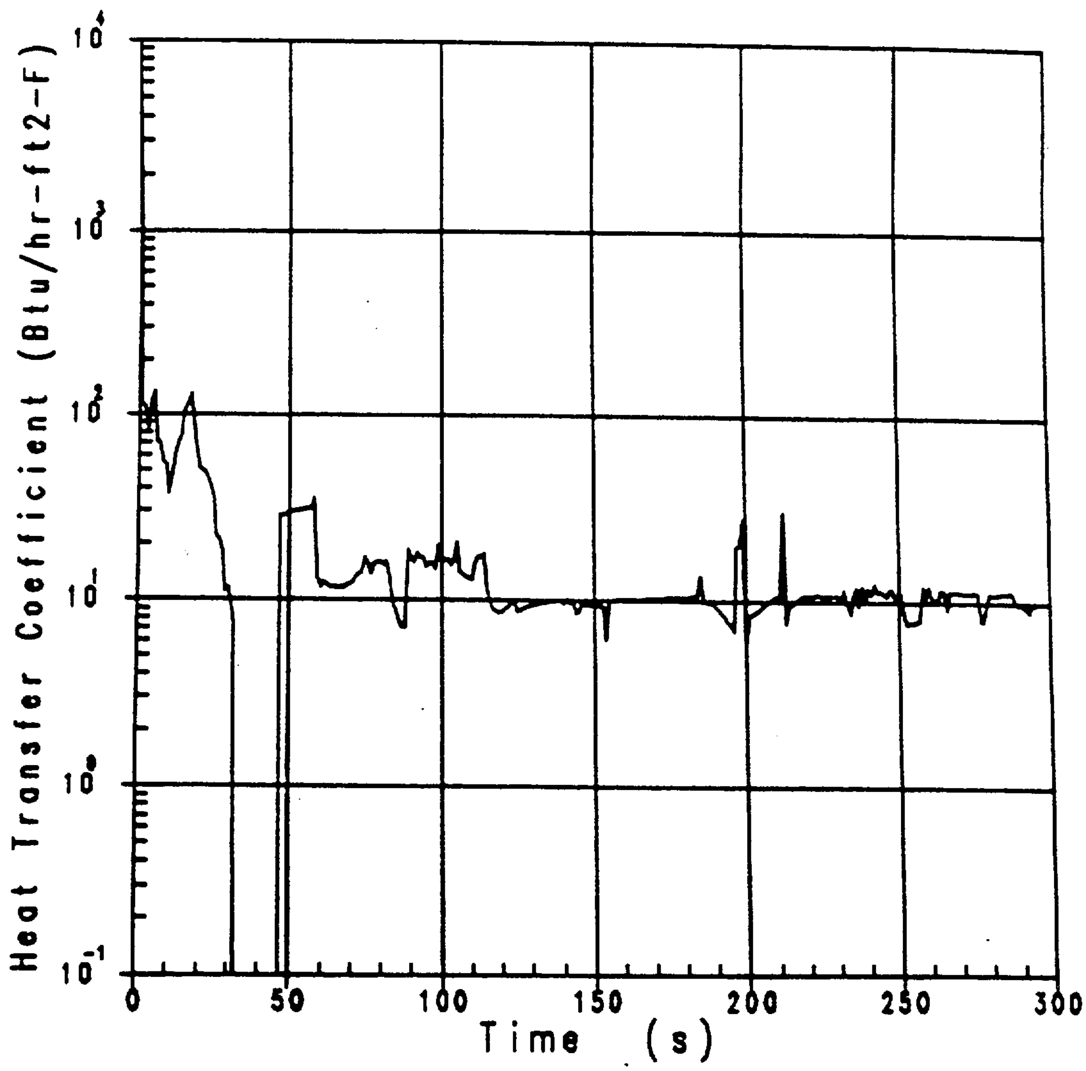

Figure 3.1-24 Rod H.T.C. at Peak Temperature Elevation
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

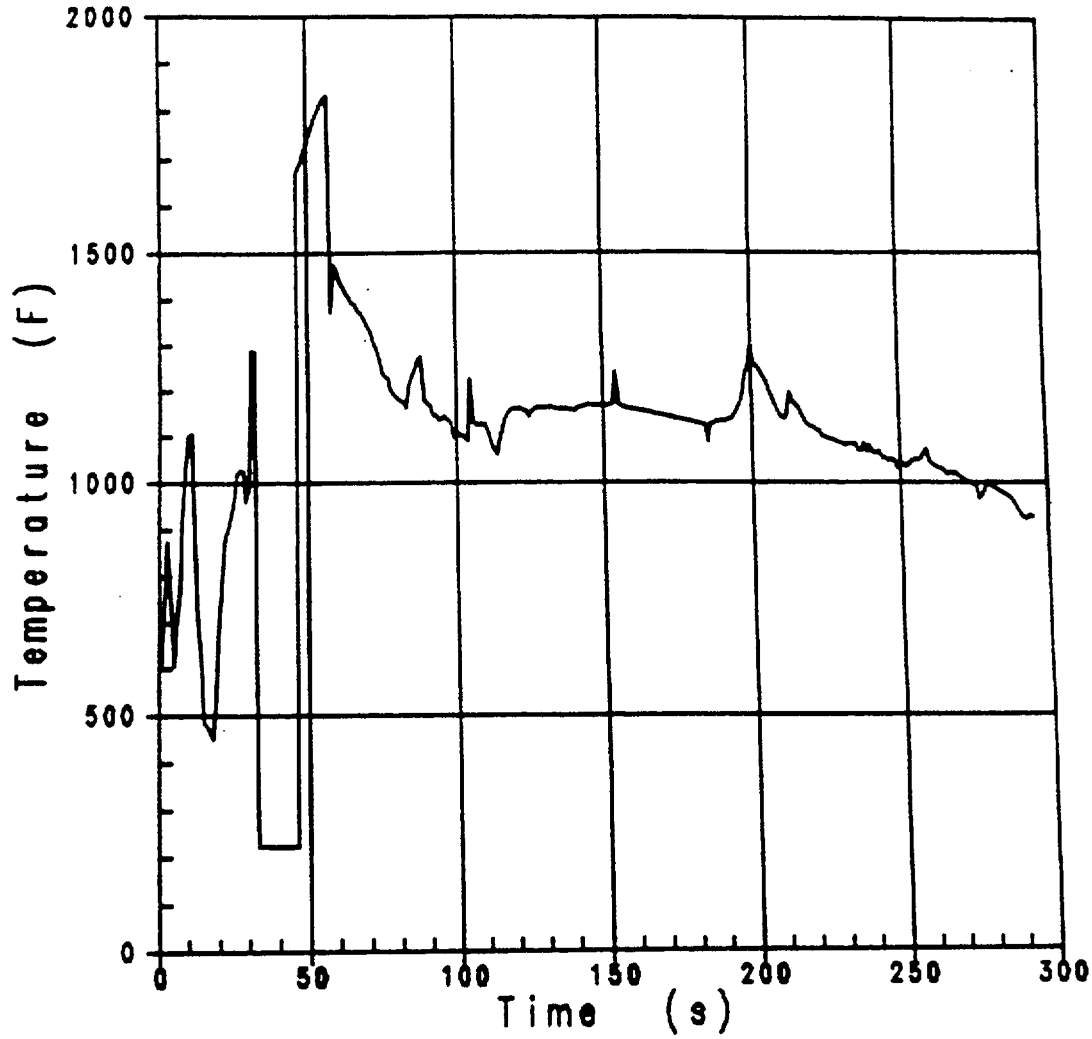

Figure 3.1-25 Vapor Temperature
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

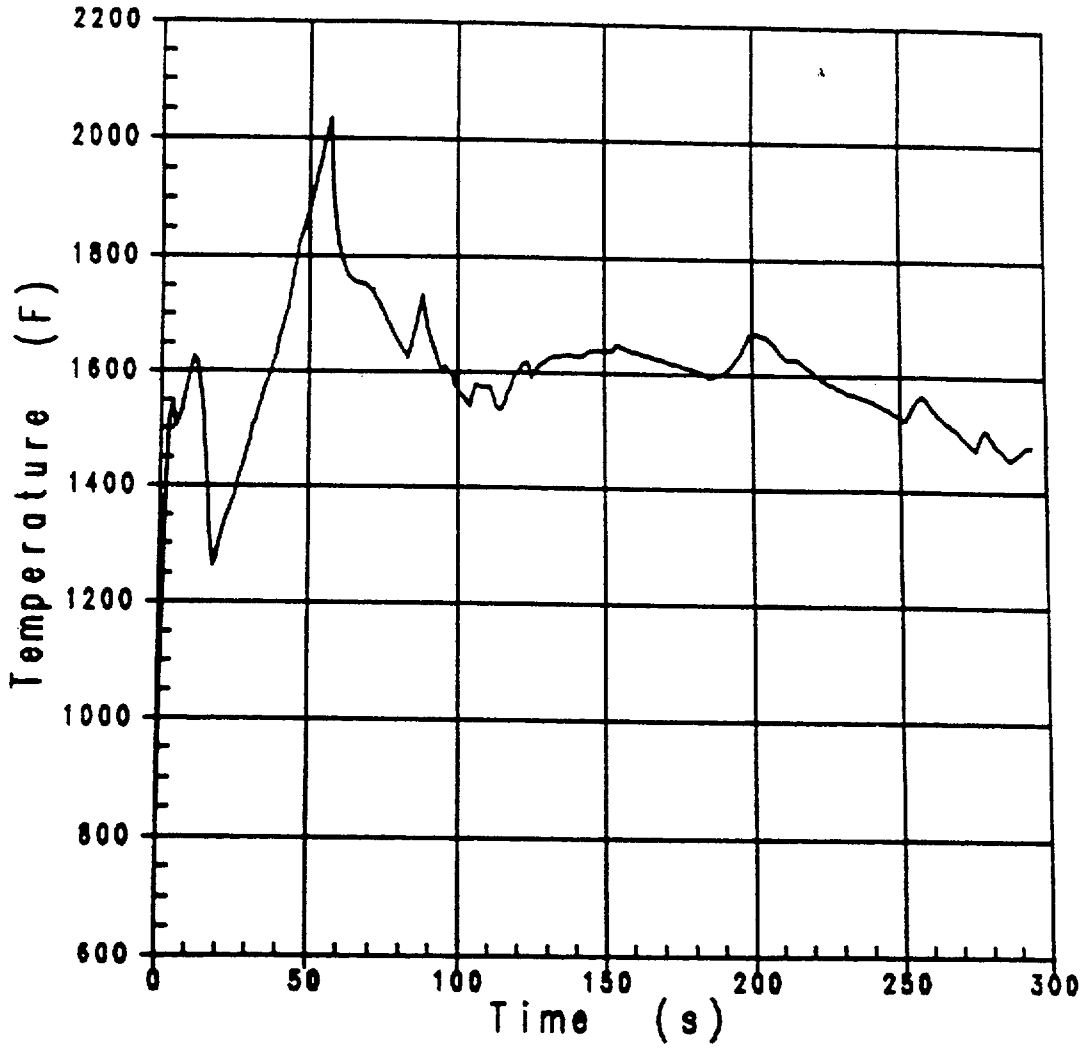

Figure 3.1-26 Fuel Rod Peak Clad Temperature
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

ATTACHMENT 3 TO C0200-08

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

DONALD C. COOK NUCLEAR PLANT, UNIT 2

TABLE 1

D. C. COOK UNIT 2

LARGE BREAK LOCA

Evaluation Model: BASH

$F_Q = 2.335$ $F_{\Delta H} = 1.644$ SGTP = 15% Break Size: $C_d = 0.6$

Operational Parameters: RHR System Cross-Tie Valve Closed, 3413 MWt Reactor Power [2]

A.	ANALYSIS OF RECORD - December 1995 (Performed for a Reactor Power of 3588 MWt)	PCT = 2051°F
B.	PRIOR LOCA MODEL ASSESSMENTS – 1996	ΔPCT= + 2°F
C.	PRIOR LOCA MODEL ASSESSMENTS – 1997	ΔPCT= + 8°F
D.	PRIOR LOCA MODEL ASSESSMENTS – 1998	ΔPCT= 0°F
E.	2000 10 CFR 50.46 MODEL ASSESSMENTS [2]	
	1. Reanalysis to incorporate LOCBART Spacer Grid Single Phase Heat Transfer Error and LOCBART Metal-Water Oxidation Error [1] (Performed for a Reactor Power of 3413 MWt)	ΔPCT= +58°F
	2. Removal of Item B above due to reanalysis	ΔPCT= –2°F
F.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT= 2117°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.

2. Power level used as basis for analysis reduced to 3413 MWt from 3588 MWt used in 1995 analysis of record.

TABLE 2

D. C. COOK UNIT 2

SMALL BREAK LOCA

Evaluation Model: NOTRUMP

$F_Q = 2.45$ $F_{\Delta H} = 1.666$ SGTP = 15% 3" cold leg break

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

A.	ANALYSIS OF RECORD - March 1992	PCT = 1956°F
B.	PRIOR LOCA MODEL ASSESSMENTS – October 1993	ΔPCT = - 13°F
C.	PRIOR LOCA MODEL ASSESSMENTS – March 1994	ΔPCT = - 16°F
D.	PRIOR LOCA MODEL ASSESSMENTS – December 1994	ΔPCT = + 69°F
E.	PRIOR LOCA MODEL ASSESSMENTS – 1995	ΔPCT = + 20°F
F.	PRIOR LOCA MODEL ASSESSMENTS – 1996	ΔPCT = - 28°F
G.	PRIOR LOCA MODEL ASSESSMENTS – 1997	ΔPCT = 0°F
H.	2000 10 CFR 50.46 MODEL ASSESSMENTS	
	1. Re-baseline for asymmetric SI	ΔPCT = - 214°F
	2. Removal of prior assessments	ΔPCT = - 32°F
	3. Asymmetric HHSI delivery [1]	ΔPCT = + 50°F
I.	BURST AND BLOCKAGE/TIME IN LIFE	ΔPCT = + 15°F
J.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT = 1807°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 3

D. C. COOK UNIT 2

SMALL BREAK LOCA

Evaluation Model: NOTRUMP
$F_Q = 2.32$, $F_{\Delta H} = 1.62$, SGTP= 15% 4" cold leg break
Operational Parameters: SI System Cross-Tie Valve Open, 3588 MWt Reactor Power

A.	ANALYSIS OF RECORD - August 1992	PCT = 1531°F
B.	PRIOR LOCA MODEL ASSESSMENTS – October 1993	ΔPCT = - 13°F
C.	PRIOR LOCA MODEL ASSESSMENTS – March 1994	ΔPCT = - 16°F
D.	PRIOR LOCA MODEL ASSESSMENTS – December 1994	ΔPCT = + 35°F
E.	PRIOR LOCA MODEL ASSESSMENTS – 1995	ΔPCT = + 20°F
F.	Prior LOCA MODEL ASSESSMENTS – 1996	ΔPCT = - 28°F
H.	Prior LOCA MODEL ASSESSMENTS – 1997	ΔPCT = 0°F
I.	Prior LOCA MODEL ASSESSMENTS 1998	ΔPCT = 0°F
J.	2000 10 CFR 50.46 MODEL ASSESSMENTS	ΔPCT = 0°F
K.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT = 1529°F

ATTACHMENT 4 TO C0200-08

RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO LARGE BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR UPRATING UNIT 2

This attachment provides responses to previous NRC requests for information pertaining to a large break loss-of-coolant accident (LBLOCA) analysis that Indiana Michigan Power Company (I&M) previously submitted to the NRC as part of supporting analyses for a proposed license amendment to increase (uprate) the Unit 2 licensed power level. That proposed amendment was subsequently withdrawn. However, prior to withdrawal the NRC had requested additional information (References 1 and 2) regarding various aspects of the proposed amendment, including the supporting LBLOCA analysis. Since that supporting LBLOCA analysis is now being resubmitted as the new LBLOCA analysis of record for Unit 2, I&M is providing responses to the pertinent NRC requests for information, or portions thereof as committed in Reference 3.

Request for Information Transmitted by NRC Letter Dated July 9, 1997 (Reference 1)

Question 2

"Clarify whether the rerating analyses of the pressure transients and the postulated loss-of-coolant accident (LOCA) include the proposed pressurizer safety and relief valve tolerance ±3%, and the previously NRC-approved main steam safety and relief valves tolerance of +3%. If not, state how the rerating analyses applies to the proposed Unit 2 power uprate."

Response to Question 2

The pressurizer safety valves, the pressurizer relief valves, the main steam safety valves and the main steam relief valves are not challenged on a LBLOCA. Consequently, the set point tolerance for these valves is not relevant to the new LBLOCA analysis of record.

Request for Information Transmitted by NRC Letter Dated August 6, 1997 (Reference 2)

Question 1

"Provide references to the staff's approval of the new SBLOCA analysis model submittal of December 14, 1994, identified as NTD-NRC-94-4278, and Unit 1's SGTP (steam generator tube plugging) submittal dated May 26, 1995, AEP:NRC:1207. Also, in paragraph 1 on page 4 of Attachment 1, you stated "the LBLOCA reanalysis with RHR cross-ties closed was also satisfactory using the current model. As described in Section 3.1.1.3 of WCAP-14489, the new reanalysis incorporates model changes that resulted from the resolution of issues identified in 10 CFR 50.46 reports and in Westinghouse reports to the NRC. These model changes were a significant benefit to Cook Nuclear Plant Unit 2 specific analysis." Provide a reference to the staff's approval of these changes to the LBLOCA model. Were any other new computer codes/models/methodologies used in analyses supporting this power uprate submittal? If yes, list the new methods and provide a reference to staff approval."

Response to Question 1

Reference to NRC staff approval of the 1994 small break loss-of-coolant accident (SBLOCA) analysis model is not provided in this response because the topic is not relevant to the new Unit 2 LBLOCA analysis of record.

NRC approval of the Unit 1 SGTP submittal, Reference 4, was provided by Reference 5. The NRC Safety Evaluation Report (SER) transmitted by that letter is relevant to this attachment in that both the LBLOCA analysis for the increased Unit 1 SGTP limit and the new Unit 2 LBLOCA analysis of record in that LBLOCA analysis used a BASH computer model. The SER also documented that the Power Shape Sensitivity Model (PSSM) for axial power shape evaluation had been replaced by the Explicit SHape Analysis for PCT Effects (ESHAPE) methodology which is used in the new Unit 2 LBLOCA analysis of record. The replacement of PSSM with ESHAPE has been approved by the NRC as noted in the Reference 5 SER.

Additionally, the Reference 5 SER documented that the NRC had not accepted use of the hot leg nozzle gap (HLNG) model into the BASH methodology. The previous LBLOCA analyses of record had relied on the benefits associated with HLNG to help offset the penalties that were applied when PSSM was replaced with ESHAPE. The absence of an approved HLNG model is the principal reason that I&M is changing to a new Unit 2 LBLOCA analysis of record that does not credit HLNG.

Regarding other new computer codes/models/methodologies used in the new LBLOCA analysis of record, a revision to the grid heat transfer model in the LOCBART computer program used for the fuel rod temperature transient calculation was a major contributor to the reduction in the peak cladding temperature (PCT). This is discussed on page 3.1-5 of Attachment 2 to this submittal, and the description and evaluation of the revised grid heat transfer model was approved by the NRC in Reference 6.

Question 4

"Identify the Chapter 15 [Chapter 14 for Cook Nuclear Plant] analyses which credit the OT delta-T reactor trip, OP delta-T reactor trip, low pressurizer pressure safety injection signal, low steam line pressure safety injection signal, and low steam line pressure steam line isolation. Provide the assumptions and results of these analyses for staff review (include FSAR markups). Justify or provide a reference of staff approval for the thermal safety limits used in deriving the proposed OT delta-T and OP delta-T set points. Also justify the new lower maximum allowable power range neutron high flux set points with inoperable steam line safety valves in a similar manner (by providing analyses/evaluations as appropriate)."

Response to Question 4

This question is relevant to the new LBLOCA analysis of record only in that the analysis credits low pressurizer pressure safety injection signal. Attachment 2 to this submittal documents the principal assumptions and results for the new LBLOCA analysis of record, including the low pressurizer pressure safety injection (SI) actuation pressure setpoint and times.

Markups of Chapter 14 of the Updated Final Safety Analysis Report (UFSAR) have not been included in this submittal because the type of LBLOCA information contained in Chapter 14 is essentially redundant to that provided in Attachment 2. The next UFSAR update submitted pursuant to 10 CFR 50.71(e) will include any changes to Chapter 14 needed to provide consistency with Attachment 2.

Justification of the thermal safety limits used in deriving OT delta-T and OP delta-T set points and lower maximum allowable power range neutron high flux set points with inoperable steam line safety valves is not provided since these set points are not relevant to the new LBLOCA analysis of record.

Question 10

"Were the changes to the grid heat transfer model in LOCBART computer code used to analyze large break LOCA approved by the staff? If yes, provide a reference of the approval, if no, then describe and justify the changes."

Response to Question 10

As indicated in I&M's response to Question 1, the changes to the grid heat transfer model in the LOCBART computer code used to analyze the LBLOCA were approved by the NRC in Reference 6.

Question 11

"In your submittal for Unit 1's SGTP, you credited the reactor vessel internal hot leg nozzle gap recirculation in your large break LOCA analysis. Was the hot leg nozzle gap credited in the analyses (large break or small break) for this submittal? If yes, justify."

Response to Question 11

Reactor vessel internal HLNG recirculation was not credited in the new LBLOCA analysis of record. Responses pertaining to uprate SBLOCA analyses are not within the scope of this submittal.

Question 12

"In your analysis of large break and small break LOCA, was loss of offsite power assumed at time zero (time of break)? If not, justify your assumption."

Response to Question 12

In the new LBLOCA analysis of record, a loss of offsite power was assumed to occur at time equal zero (time of the break) and the reactor coolant pumps were assumed to trip at that time. Responses pertaining to uprate SBLOCA analyses are not within the scope of this submittal.

Question 19

"How were pressure and temperature instrument uncertainties accounted for in your large break and small break LOCA analyses? Justify your answer."

Response to Question 19

For the new LBLOCA analysis of record, the pressure uncertainty of 62.6 pounds per square inch (psi) was added to the nominal pressure for the high pressure cases and subtracted from the nominal pressure for the reduced pressure cases. This ensures that the entire pressure range is covered by the analyses. The 62.6 psi value bounds the combination of the controller uncertainty calculated by Westinghouse and the readability uncertainty calculated by I&M.

The temperatures assumed in the new LBLOCA analysis of record are documented in Attachment 2, Table 3.1-7. These are the nominal values for the high and low average temperature cases. The temperature uncertainty is not explicitly modeled in the LBLOCA analyses, but is accounted for by using appropriately conservative initial coolant temperatures that were determined based on the best-estimate vessel average temperature, the thermal design flow, and 102% of the licensed core power for the high and low temperature cases. This method of accounting for temperature uncertainties is inherent in the LOCA analysis methodologies that have previously been approved by the NRC.

Responses pertaining to uprate SBLOCA analyses are not within the scope of this submittal.

Question 20

"Justify the use of an average RWST temperature in lieu of the limiting temperature for the range. Justify the additional assumed accumulator line volume of 32 cu. ft. per accumulator and the temperature assumption of 100°F."

Response to Question 20

The average refueling water storage tank (RWST) temperature of 87.5°F, the additional accumulator line volume of 32 cubic feet per accumulator, and the accumulator water temperature of 100°F were previously assumed for the Unit 1 LBLOCA analysis submitted by Reference 4 to support the increased Unit 1 SGTP limit, which was approved by the NRC in Reference 5. The basis for each of these assumptions is provided below.

The RWST water temperature has two principal effects in the BASH LBLOCA evaluation model: (1) an increase in the RWST temperature tends to increase the enthalpy of core reflood liquid early in the reflood transient which results in a PCT penalty, and (2) an increase in RWST temperature tends to reduce the effectiveness of containment spray causing increased containment pressure late in the transient, which provides a PCT benefit. Based on sensitivity studies, it was concluded that the effect of containment pressure on the reflood is more important than the effect of the small increase in lower plenum enthalpy. Since 10 CFR 50, Appendix K, directs that containment pressure be minimized, it is appropriate to conservatively assume the spray temperature is at the technical specification (T/S) minimum. Thus, the minimum T/S RWST temperature of 70°F was used in the calculation of the containment backpressure for the LBLOCA analysis, as indicated in Attachment 2, Table 3.1-8. However, since assuming the T/S minimum RWST temperature for the SI water results in a PCT benefit, the average RWST temperature was used in the BASH evaluation model for conservatism.

The accumulator water volume used in the LBLOCA analysis is typically based on the accumulator T/S requirements. However, it was determined that the accumulator water volume requirements in the T/S only apply for the accumulator tanks, and do not include the water volume in the accumulator lines. Since the water in the accumulator lines will also be injected following a LOCA, it is appropriate to include the water in the accumulator lines as part of the accumulator water volume. The volume in the accumulator lines from the check valve adjacent to the reactor coolant system to the accumulator tanks was determined to be 32 cubic feet per accumulator, and this volume was added to the accumulator tank water volume of 946 cubic feet per accumulator.

The 100°F accumulator water temperature used in the new LBLOCA analysis of record is an increase from the 90°F value used in the previous reload transition safety report (RTSR). Since the RTSR analysis was performed, Westinghouse performed sensitivity studies that indicated that higher accumulator water temperatures could result in an increase in the calculated PCT using the BASH evaluation model. After discussions with the Westinghouse Owners Group and the NRC staff, it was decided that the accumulator water temperature for the LBLOCA analysis should be established as the maximum expected value for the plant minus 12°F. The maximum expected water temperature is based on the average containment air temperature over approximately a two-week period during the hottest portion of the year. An average accumulator room air temperature of 112°F was established for the hottest two-week period of 1993 based on plant data for Unit 1, and the analysis value of 100°F was established on this basis. Since the containments for Unit 1 and Unit 2 are essentially identical, the Unit 1 analysis value of 100°F is applicable to Unit 2.

Question 21

"Confirm the total core peaking factor of 2.32 and hot channel enthalpy rise factor of 1.62 listed in Table 3.1-26 of Attachment 6. Explain the differences between the values assumed in the small break LOCA and those assumed in the large break LOCA."

Response to Question 21

The Attachment 6 referenced in the question is Attachment 6 of the I&M letter (Reference 7) proposing a license amendment to uprate Unit 2. Table 3.1-26 of that Attachment 6 lists the input parameters for the SBLOCA analysis that was submitted to support the proposed uprate. Responses pertaining to uprate SBLOCA analyses are not within the scope of this submittal.

As documented in Attachment 3 of this submittal, there are differences among the values for core peaking factor and hot channel enthalpy rise factor (F_Q and $F_{\Delta H}$) assumed in the new LBLOCA and current SBLOCA analyses of record. It is not unusual for the F_Q and $F_{\Delta H}$ assumptions to differ between such analyses since each analysis undergoes a unique evolutionary process to arrive at its current state. Moreover, it is not critical that the same F_Q and $F_{\Delta H}$ values be assumed for each analysis. It is only critical that cycle specific core operating limits established pursuant to T/S 6.9.1.9.1 be bounded by the most limiting values for F_Q and $F_{\Delta H}$ assumed in the current analyses of record.

REFERENCES

1. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Re: Power Uprate Program (TAC Nos. M96363 and M96364)," dated July 9, 1997.

2. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Re: Power Uprate Program (TAC Nos. M96363 and M96364)," dated August 6, 1997.

3. 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.

4. Letter from E. E. Fitzpatrick, I&M, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plant Units 1 and 2, License Nos. DPR-58 and DPR-74, Proposed Technical Specification Changes Supported by Analyses to Increase Unit 1 Steam Generator Tube Plugging Limit and Certain Proposed Changes for Unit 2 Supported by Related Analyses," submittal AEP:NRC:1207, dated May 26, 1995.

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

6. Letter from A. C. Thadani, NRC, to N. J. Liparulo, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-10484, Addendum 1 - Spacer Grid Heat Transfer Effects During Reflood," dated July 7, 1993.

7. Letter from E. E. Fitzpatrick, I&M, to U. S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Plants 1 and 2, License Nos. DPR-58 and DPR-74, Proposed License and Technical Specification Changes Supported by Analyses to Increase Unit 2 Rated Thermal Power and Certain Proposed Changes for Unit 1 Supported by Related Analyses," submittal AEP:NRC:1223, dated July 11, 1996.

ATTACHMENT 5 TO C0200-08

COMMITMENTS MADE IN THIS SUBMITTAL

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

COMMITMENT	DATE
The next Updated Final Safety Analysis Report update submitted pursuant to 10 CFR 50.71(e) will include any changes to Chapter 14 needed to provide consistency with Attachment 2 of this submittal.	As required by 10 CFR 50.71(e)