WCAP-14489, Rev. 1

DONALD C. COOK NUCLEAR PLANT UNIT 2
3600 MWt UPRATING PROGRAM
LICENSING REPORT

May 1996

WESTINGHOUSE ELECTRIC CORPORATION
Energy Systems Business Unit
P.O. Box 355
Pittsburgh, Pennsylvania 15230

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

LIST OF TABLES

LIST OF TABLES (continued)

LIST OF TABLES (continued)

LIST OF TABLES (continued)

LIST OF TABLES (continued)

LIST OF FIGURES

LIST OF FIGURES (continued)

LIST OF FIGURES (continued)

LIST OF FIGURES (continued)

LIST OF FIGURES (continued)

LIST OF FIGURES (continued)

LIST OF ACRONYMS AND ABBREVIATIONS

AEPSC	American Electric Power Service Corporation
AFWS	Auxiliary Feedwater System
ANS	American Nuclear Society
ART	Adjusted Reference Temperature
ASME	American Society of Mechanical Engineers
APC	Alternate Plugging Criteria
BIT	Boron Injection Tank
BOP	Balance of Plant
CCWS	Component Cooling Water System
C_D	Discharge Coefficient
C&FS	Condensate and Feedwater System
CHG/SI	Charging/Safety Injection
COLR	Core Operating Limits Report
CRDM	Control Rod Drive Mechanism
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
DBE	Design Basis Earthquake
DECL	Double-Ended Cold Leg
DEHL	Double-Ended Hot Leg
DEPS	Double-Ended Pump Suction
DER	Double-Ended Rupture
DF	Decontamination Factor
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFPM	Effective Full Power Months
EFPY	Effective Full Power Years
EOP	Emergency Operating Procedure
ERG	Emergency Response Guidelines
ESF	Engineered Safety Features
ESFAS	Engineered Safety Feature Actuation System
ESW	Essential Service Water
$F\Delta H$	Hot Channel Enthalpy Rise Factor
F_Q	Total Peaking Factor
FHA	Fuel Handling Accident
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GPM	Gallons per Minute
HELB	High Energy Line Break
HFP	Hot Full Power

LIST OF ACRONYMS AND ABBREVIATIONS (continued)

HHSI	High Head Safety Injection
HTC	Heat Transfer Coefficient
HZP	Hot Zero Power
IFBA	Integral Fuel Burnable Absorbers
IFM	Intermediate Flow Mixing
ISI	In-service Inspection
ITDP	Improved Thermal Design Procedure
LB	Large Break
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
LOL/TT	Loss of Load/Turbine Trip
LOOP	Loss of All AC Power to the Station Auxiliaries
LPZ	Low Population Zone
M/E or M&E	Mass and Energy
MCO	Moisture Carryover
MMF	Minimum Measured Flow.
MSLB	Main Steam Line Break
MTC	Moderator Temperature Coefficient
MWt	Megawatt Thermal
NIS	Nuclear Instrumentation System
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OPΔT	Overpower Delta T
OTΔT	Overtemperature Delta T
PCT	Peak Clad Temperature
PLOF	Partial Loss of Reactor Coolant Flow
PORV	Power Operated Relief Valve
PRT	Pressurizer Relief Tank
PTS	Pressurized Thermal Shock
PSSM	Power Shape Sensitivity Model
PSV	Pressurizer Safety Valve
PWR	Pressurized Water Reactor
RC	Reactor Coolant
RCCA	Rod Cluster Control Assembly
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RPS	Reactor Protection System
RSE	Reload Safety Evaluation
RSG	Replacement Steam Generator

LIST OF ACRONYMS AND ABBREVIATIONS (continued)

RSR	Relative Stability Ratio
RTDP	Revised Thermal Design Procedure
RT_{NDT}	Reference Temperature for Nil-Ductility Transition
RT_{PTS}	Reference Temperature for Pressurized Thermal Shock
RTP	Rated Thermal Power
RTS	Reactor Trip System
RTSR	Reload Transition Safety Report
RWST	Refueling Water Storage Tank
RWFS	RCCA Bank Withdrawal from a Subcritical Condition
SAL	Safety Analysis Limit
SDM	Shutdown Margin
SECL	Safety Evaluation Checklist
SER	Safety Evaluation Report
SB	Small Break
SI	Safety Injection
SIS	Safety Injection System
SG	Steam Generator
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SLB	Steam Line Break
SLB-CR	Steam Line Break Core Response
SR	Surveillance Requirement
TA	Total Allowance
T_{avg}	RCS Average Temperature
T_{HOT}	Vessel Outlet Temperature
T_{COLD}	Vessel Inlet Temperature
TDF	Thermal Design Flow
UFSAR	Updated Final Safety Analysis Report
VCT	Volume Control Tank

DEFINITIONS

Rerating Program:

WCAP-11902 documented the analyses and evaluations performed to support reduced temperature and pressure operation of Donald C. Cook Nuclear Plant Unit 1. Subsequently, a supplement to WCAP-11902 was issued to summarize the additional efforts performed to support an uprated power level of Cook Nuclear Plant Unit 1 and to provide support for a Unit 2 rerating (increased power level, reduced temperature and pressure). These analyses and evaluations are described in Sections 2.0 of this report and are documented in References 1 and 2 of Section 2.0. Throughout this report, the analyses and evaluations documented in WCAP-11902 and Supplement are referred to as the Rerating Program.

Uprating Program (Unit 2):

Analyses and evaluations to support operation of Donald C. Cook Nuclear Plant Unit 2 with an NSSS power level of 3600 MWt. In addition to the increased power level, several increases in operating margin were addressed in the Uprating Program. These operating margin increases are described in Section 1.0.

SGTP Program (Unit 1):

Analyses and evaluations to support operation of Donald C. Cook Nuclear Plant Unit 1 with a steam generator tube plugging level of 30%. This program is described in WCAP-14285, and includes some analyses which bound both Unit 1 and Unit 2.

VANTAGE 5 Reload Transition Safety Report (RTSR):

The analyses and evaluations performed as part of the VANTAGE 5 fuel upgrade program for Donald C. Cook Nuclear Plant Unit 2 are documented in the VANTAGE 5 Reload Transition Safety Report, Revision 1, dated March 1990. All of the accident analyses, with the exception of the Large Break LOCA and the Containment Integrity analyses, were performed to support an uprated NSSS power level of 3600 MWt.

SUMMARY AND CONCLUSIONS

PROGRAM SUMMARY

The purpose of this document is to provide the safety analysis and evaluation results to support operation of Donald C. Cook Nuclear Plant Unit 2 with an NSSS power level of 3600 MWt. The analyses and evaluations were performed over a range of primary temperatures (547°F and 581.3°F), two values of primary pressure (2100 psia and 2250 psia), an average steam generator tube plugging (SGTP) level of 10%, and a peak SGTP level of 15%. The evaluations in this report are based on analyses performed for the Rerating Program and the VANTAGE 5 RTSR.

In addition to addressing the increased NSSS power level (3600 MWt), the following increased operating margins were also addressed:

(1) Reduction of SI and RHR discharge pressure on recirculation - The SI and RHR minimum safeguards pump head curves were reduced by 15%, an additional 5% reduction from the current analysis degradation of 10%. The charging pump head curve degradation remained at the current analysis value of 10%.

(2) To support increased ΔT drift, the margin between the safety analysis limits (SAL) and the nominal values of the K_1 and K_4 gains of the Donald C. Cook Nuclear Plant Unit 2 OTΔT and OPΔT setpoint equations were adjusted.

(3) An increase in the pressurizer code safety valve (PSV) setpoint tolerance from +/-1% to +/-3%

The analyses and evaluations in this document support all of these changes. Discussions of specific analyses address issues most relevant to those analyses.

The operating parameters for the increase in NSSS power level and the additional operating margins listed above will be referred to throughout this report as the "Uprating Program". This report provides the necessary documentation to support the Technical Specification changes associated with the Uprating Program. The topics addressed in this report are as follows:

- Description of License Amendment
- Summary of Technical Specification Changes
- Basis for Evaluations/Analyses Performed
- Loss of Coolant Accident Analyses
- Post-LOCA Hydrogen Production
- Post-LOCA Hot Leg Recirculation Time
- LOCA Hydraulic Forces

- Non-LOCA Analyses
- Containment Analyses
- Steam Generator Tube Rupture Analyses
- Reactor Cavity Pressure Evaluation
- Radiological Analysis
- Primary Components Evaluations
- Fluid and Auxiliary Systems Evaluations
- Fuel Structural Evaluation

Also provided in the Appendix to this report are the proposed Technical Specification changes. A brief summary of the results of each analysis and evaluation is provided below.

ACCIDENT ANALYSIS CONCLUSIONS

The results of the accident analyses and evaluations performed for the Uprating Program demonstrate that safe operation of Donald C. Cook Nuclear Plant Unit 2 is maintained. The bases for the evaluations and analyses performed are provided in Section 2.1.

Large Break LOCA (Section 3.1.1)

The results of the VANTAGE 5 RTSR large break LOCA analysis indicated that plant operation with the RHR crosstie valves open would be required in order to support operation at the uprated power of 3588 MWt. Because AEPSC could not operate Cook Nuclear Plant Unit 2 with the RHR crosstie valves open due to the possibility of dead heading the RHR pumps, the RHR crosstie valves were closed and an uprated power level was not pursued. A modification to the RHR System was subsequently designed to move the RHR crosstie valves downstream of the miniflow line and permit operation with the crosstie valves open. The large break LOCA calculation was reanalyzed assuming the new location of the RHR crosstie valves, 15% pump head degradation and the crosstie valves open.

Since the time of the Rerating Program analysis, the grid heat transfer model in the LOCBART computer program was revised. The model revision resulted in an improvement in the ability of the grids to wet, which translated into a Peak Clad Temperature (PCT) benefit. When compared to the Rerating Program, the revised grid model resulted in a significant PCT benefit for the large break LOCA analysis at a power level of 3588 MWt with the RHR crosstie valves open. Consequently, analyses were also perfomed for other break sizes and conditions to ensure that the revised grid model did not cause the limiting break size to change. The results of the additional analyses confirmed that the high pressure/high temperature conditions with minimum safeguards for the 0.6 C_D break remained limiting.

Considering the significant PCT benefit resulting from the model revision, the large break LOCA analysis was reanalyzed for the crosstie valves closed case to determine if acceptable

results could be achieved. Additional analysis assumptions included 3588 MWt core power, a total peaking factor of 2.335, a hot channel enthalpy rise peaking factor of 1.644, and an accumulator water temperature of 100°F. The limiting case resulted in a PCT of 2051°F, which is less than the 2200°F limit in 10 CFR 50.46. It is concluded that the Donald C. Cook Nuclear Plant Unit 2 large break LOCA analysis is acceptable for operation at 3588 MWt with the RHR crosstie valves closed, and therefore, no modification to the RHR system is required.

It should be noted that the large break LOCA analysis was not impacted by the PSV tolerance increase or the K1/K4 values.

Small Break LOCA (Section 3.1.2)

The small break LOCA analysis performed as part of the Rerating Program was performed for a core power level of 3588 MWt using ECCS flows with the high head safety injection (HHSI) crosstie valves open. Analyses with the HHSI crosstie valves closed could not support uprated power conditions.

Since the time of the Rerating Program, changes in the small break LOCA evaluation with NOTRUMP, including the modeling of safety injection in the broken loop with the COSI condensation model, have resulted in net PCT benefits. Therefore, the small break LOCA transient was reanalyzed for the uprated power level of 3588 MWt with the HHSI crosstie valves closed. The analysis also included revised ECCS flows due to 15% pump head degradation. The small break LOCA analysis was not impacted by the pressurizer code safety valve tolerance increase or the revised K1/K4 values.

The small break LOCA analysis was performed with the Westinghouse small break LOCA ECCS Evaluation Model using the NOTRUMP code (including the recent model changes submitted in WCAP-10054-P, Addendum 2 and WCAP-10081-NP, Addendum 2). The key analysis input assumptions included ECCS flows with the HHSI crosstie valves closed, a total peaking factor of 2.32, a hot channel enthalpy rise peaking factor of 1.62, a hot assembly average power (P_{HA}) of 1.443, and a power shape based on an axial offset of 13%. The small break analysis was performed for the reduced pressure and high temperature operating conditions which were previously demonstrated to be limiting at the uprated power. The analysis was performed for the 3-inch break first since this was the limiting break size for the previous analysis at reduced power with the HHSI crosstie valves closed. The analysis was then extended to the 2 and 4-inch breaks to ensure that the limiting break size did not change. The analysis demonstrated that the 3-inch break remains the limiting break with the high head crosstie valves closed. The peak cladding temperature calculated for the 3-inch break is 2065°F which is less than the 2200°F limit in 10 CFR 50.46.

LOCA Hydraulic Forcing Functions (Section 3.2)

The Donald C. Cook Nuclear Plant LOCA hydraulic forces were most recently analyzed for the VANTAGE 5 RTSR. The RCS parameters used in the existing analysis-of-record bound the conditions for the Uprating Program. Therefore, the existing LOCA forces analyses remain bounding relative to the Uprating Program.

Non-LOCA Analyses (Section 3.3)

The non-LOCA events were addressed by a combination of evaluations and analyses for the impact of the increased NSSS power level, revised ECCS flows, PSV tolerance increase, and revised K1/K4 values. The computer codes and methods used for the non-LOCA analyses have been previously approved by the NRC. The non-LOCA safety analyses were reviewed on the basis of both DNB and non-DNB acceptance criteria. All DNB event reanalyses were found to yield a minimum DNBR which remains above the limit value. The analyses demonstrate that all licensing basis criteria continue to be met and the conclusions presented in the UFSAR remain valid.

Post-LOCA Hydrogen Generation (Section 3.4)

The post-LOCA hydrogen generation rates that were reviewed as part of the Rerating Program were determined to remain applicable to the Uprating Program.

Containment Integrity (Section 3.5)

The containment integrity analyses were addressed for the impact of the increased NSSS power level and revised ECCS flows. The containment analyses were not impacted by the PSV tolerance increase or the revised K1/K4 values. The increase in the containment pressure and temperature following a LOCA was analyzed. The LOCA mass and energy release rates calculated as part of the Uprating Program formed the basis to evaluate the structural integrity of the containment following a postulated accident to satisfy the acceptance criterion, General Design Criterion (GDC) 38. Even though Cook Nuclear Plant is licensed to GDCs in Appendix H of the original FSAR, more conservative acceptance criteria were used. The containment integrity analysis for the most limiting case (i.e., RHR crosstie valve closed) resulted in a maximum calculated containment pressure of 11.66 psig, for the double-ended pump suction minimum safeguards break case. Since the calculated pressure is below the design pressure of 12.0 psig, the results of the LOCA containment integrity analysis are acceptable.

As part of the Donald C. Cook Nuclear Plant Steam Generator Tube Plugging Program, a containment integrity analysis was performed to demonstrate that the peak containment temperature resulting from a design basis MSLB will not exceed the equipment qualification

criterion for the plant. The analysis was performed to bound Donald C. Cook Nuclear Plant Units 1 and 2 operation at uprated conditions (3600 MWt). The containment pressure response determined for the LOCA containment integrity analysis is calculated to be more severe than for the MSLB, and therefore bounds the MSLB analysis. For the large break case, the limiting case is the 1.4 ft^2 double-ended rupture at 102% power with MSIV failure. This case yielded a calculated peak temperature of 322.7°F. For the small break case, the most limiting case in terms of peak calculated temperature is the 0.942 ft^2 split break at 30% power with an MSIV failure. This case resulted in a calculated peak temperature of 326°F. Both cases are within the Environmental Acceptance Criteria (Section 3.5.4.1, Reference 3). Therefore, the analysis demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a MSLB. General Design Criterion 50 and Appendix K are satisfied. This analysis is discussed in WCAP-14285.

<u>Short Term Containment Analysis</u> **(Section 3.5.1)**

The short term containment analysis that was performed for the Rerating Program was reviewed and it was determined that the conclusions provided for the Rerating Program remain valid for the Uprating Program. That is, the resulting peak pressures remain below the allowable design peak pressures for the pressurizer enclosure, the fan accumulator room, and the steam generator enclosure.

<u>Steam Generator Tube Rupture (Section 3.6)</u>

The SGTR event was analyzed for the impact of the Uprating Program parameters. The SGTR analysis was not impacted by any of the Uprating Program increased operating margins. A review of the SGTR analysis to support an uprating to 3600 MWt showed that the thermal-hydraulic results remain applicable for the uprating power conditions. Therefore, the conclusions of the UFSAR remain valid.

<u>Post-LOCA Hot Leg Recirculation Time</u> **(Section 3.7)**

The hot leg recirculation time analysis has been updated for Donald C. Cook Nuclear Plant Unit 2 to determine the time when switchover to hot leg recirculation should be initiated following a LOCA. This analysis addresses the concern related to the potential for boron precipitation in the reactor vessel during cold leg recirculation following a LOCA.

The results of the updated hot leg switchover analysis for Donald C. Cook Nuclear Plant Unit 2 indicates that the maximum allowable boric acid concentration of 23.53 weight percent will not be exceeded in the vessel if the hot leg recirculation is initiated at 8.5 hours after the LOCA occurs. An assessment also indicates that the cold leg recirculation flow rate is sufficient to meet the requirement for core heat removal when sump recirculation is initiated,

and that the hot leg recirculation flow rate is adequate to meet the recirculation requirement when switchover to hot leg recirculation is performed.

Reactor Cavity Pressure Analysis (Section 3.8)

The Reactor Cavity Pressure Analysis that was performed for the Rerating Program was reviewed and it was determined that the conclusions provided for the Rerating Program remain valid for the Uprating Program.

Radiological Doses (Section 3.9)

The Loss-of-Offsite Power, Loss of Load, Loss of Normal Feedwater, SGTR, Steamline Break and Fuel Handling Accident were reanalyzed for the Uprating Program. All radiological doses were found to be within acceptable limits. The Large Break LOCA dose calculations performed for the Cook Unit 1 SGTP Program bound the Cook Unit 2 Uprating Program.

FLUID AND AUXILIARY SYSTEMS EVALUATION CONCLUSIONS (Section 3.10)

The fluid systems proof of design calculations were reviewed for the Uprating Program conditions. This review demonstrated that the NSSS fluid systems will continue to function adequately as designed for all conditions of the Uprating Program. ECCS flowrates were revised as part of the Uprating Program and were used in the safety analyses and evaluations.

In the NSSS/BOP interface area, the proposed NSSS Performance Parameters for the Uprating Program were compared with those of the Rerating Program. The results of the evaluation show that the Uprating Program will have no adverse effects on the BOP systems performance (Main Steam System, Condensate and Feedwater System, Steam Dump System, Auxiliary Feedwater System, Steam Generator Blowdown and Sampling System). They will continue to perform acceptably at the conditions associated with Uprating Program. The current minimum stored volume in the Condensate Storage Tank is 175,000 gallons and has been verified to be adequate for the Uprating Program.

PRIMARY COMPONENTS EVALUATION CONCLUSIONS

Steam Generators (Section 3.11.1)

As part of the Rerating Program, thermal hydraulic performance parameters were evaluated for a range of thermal powers and steam pressures. The conclusion of the Rerating Program was that the performance characteristics of the steam generators, including moisture carryover, continue to be acceptable at all the Rerating Program conditions. Since the

envelope of uprating conditions is bounded by the Rerating Program, the conclusion continues to apply for the Cook Nuclear Plant Unit 2 Uprating Program.

Structural analyses and evaluations performed for the Cook Unit 2 steam generators indicate that the steam generator components remain in compliance with the applicable ASME Code requirements under the Uprating Program conditions.

Reactor Vessel (Section 3.11.2)

The results of the structural evaluations performed for the reactor vessel demonstrate that operation of Cook Nuclear Plant Unit 2 within the parameters of the Uprating Program does not result in stress intensities or fatigue usage factors which exceed the acceptance criteria of the applicable ASME Code versions. The changes in the neutron fluence resulting from the uprated power level were evaluated for the impact on reactor vessel integrity. The assessment included a review of the upper shelf energy values, current material surveillance capsule withdrawal schedule, heatup and cooldown pressure-temperature limit curves, RT_{PTS} values, and the Emergency Response Guideline (ERG) limits. The evaluation concluded that the upper shelf energy values of all beltline region plates and weld materials are expected to remain above 50 ft-lb through the life of the vessel (32 EFPY). All RT_{PTS} values of the Cook Nuclear Plant Unit 2 reactor vessel beltline materials remain below the PTS screening criteria using projected uprating fluence values through 32 EFPY. The removal schedule contained in the Cook Nuclear Plant Unit 2 Technical Specifications remains unchanged. Upon implementation of the Uprating Program, the heatup and cooldown pressure-temperature limit curves currently contained in the Technical Specifications will be applicable to 14.5 EFPY and Cook Nuclear Plant Unit 2 will be in Category II of the ERG pressure temperature limits.

Reactor Internals (Section 3.11.3)

Results of the thermal-hydraulic analyses performed for the reactor internals indicate that the Uprating Program for Cook Nuclear Plant Unit 2 results in acceptable values of core bypass flow, pressure drops, component lift forces, and momentum flux values. It was also confirmed that the RCCA scram performance was not impacted by the Uprating Program conditions.

From the component stress analysis and the flow induced vibration evaluations, it is concluded that the margins of safety are within acceptable limits per the original design basis.

Control Rod Drive Mechanisms (Section 3.11.4)

The conclusion of structural evaluations performed for the Uprating Program conditions for the CRDMs demonstrate that the operability, service life, and structural integrity of the CRDM latch assembly, drive rod, and coil stack will not be adversely affected.

Reactor Coolant Pumps (Section 3.11.5)

The review performed of the reactor coolant pumps for the Uprating Program conditions demonstrate that the conditions are acceptable for the 93A RCP, and no additional thermal or structural analyses are required to demonstrate compliance with the applicable codes and standards. The RCP motor evaluation revealed that the motors are acceptable for operation at the Uprating Program conditions.

Pressurizer (Section 3.11.6)

A fatigue analysis performed for the Cook Nuclear Plant Unit 2 pressurizer, incorporating the most conservative conditions of the Uprating Program, demonstrated that the pressurizer remains in compliance with the applicable ASME Code criteria.

Reactor Coolant Piping and Supports (Section 3.11.7)

As part of the Rerating Program, analyses were performed to determine the effects of the Rerating Program conditions on the primary loop piping, primary equipment supports, and the primary equipment nozzles. The analyses demonstrated the acceptability of these components at the Rerating Program conditions. The Rerating Program analyses were reviewed for their applicability to the Uprating Program, and it was confirmed that the primary loop piping, primary equipment supports, and primary equipment nozzles are acceptable for operation at the Uprating Program conditions.

It was also concluded that the Uprating Program conditions will have an insignificant impact on the design basis analysis for the NRC Bulletin 88-08 evaluation of the auxiliary spray piping and the NRC Bulletin 88-11 evaluation of the pressurizer surge line piping.

Auxiliary Components (Section 3.11.8)

Evaluations were performed for the auxiliary tanks, pumps, valves, and heat exchangers to determine the effects of the revised RCS parameters due to the Uprating Program. The results of these evaluations demonstrated that, due to conservatively specified parameters used in the procurement of the auxiliary equipment, the Uprating Program parameters will not adversely affect the function or structural integrity of this equipment.

Ice Condenser (Section 3.11.9)

The potential impact of increased blowdown forces on the ice condenser, due to changes associated with the Rerating Program and the Uprating Program, on the structural integrity of the ice condenser have been assessed. It was determined that the ice condenser is structurally adequate for the rerating/uprating conditions.

Fuel Structural Evaluation (Section 3.12)

Evaluations were performed for the Cook Nuclear Plant Unit 2 Uprating Program in the areas of fuel rod and fuel assembly structural integrity, core design, and thermal-hydraulic design.

The fuel assembly structural integrity is not affected by the Uprating Program, and the core coolable geometry is maintained for the 17x17 Vantage 5 fuel in the Unit 2 core. The evaluation of the fuel rod structural integrity indicates these conditions will be acceptable, although it is noted that cycle-specific verification during the normal reload will still be performed.

The results of the core design evaluation indicated that the Uprating Program conditions will not impact the core design and that the analysis performed for the VANTAGE 5 RTSR remain applicable.

Thermal-hydraulic analyses were performed for the fuel for the limiting Uprating Program parameters using RTDP methodology. The analysis showed that the DNBR design basis was met for the limiting DNB events. This analysis caused the available DNB margin to increase. This margin can be used for flexibility of design and to offset unanticipated DNBR penalties.

1.0 INTRODUCTION - DESCRIPTION OF LICENSE AMENDMENT REQUEST

1.1 PURPOSE FOR CHANGE

Currently, the licensing basis analyses for Donald C. Cook Nuclear Plant Unit 2 are documented in the Updated Final Safety Analysis Report (UFSAR). This amendment request reflects the changes to the safety analysis assumptions and results due to the revised operating conditions resulting from an increased NSSS power level. While the analyses and evaluations were being performed for the increased NSSS power level, several operating margins were increased and incorporated into the analyses and evaluations in order to maximize the benefit of the reanalysis. Therefore, in addition to addressing an increased NSSS power level of 3600 MWt, the following increased operating margins were also addressed:

(1) Reduction of SI and RHR discharge pressure on recirculation - The SI and RHR minimum safeguards pump head curves were reduced by 15%, an additional 5% reduction from the current analysis degradation of 10%. The charging pump head curve degradation is maintained at the current analysis value of 10%.

(2) To support increased ΔT drift, the margin between the safety analysis limits (SAL) and the nominal values of the K_1 and K_4 gains of the Donald C. Cook Nuclear Plant Unit 2 OTΔT and OPΔT setpoint equations were adjusted.

(3) An increase in the pressurizer code safety valve (PSV) setpoint tolerance from ± 1% to ±3%

The parameters associated with an increased NSSS power of 3600 MWt, both directly and indirectly, are referred to throughout this report as the "Uprating Program". This program resulted in changes to the Donald C. Cook Nuclear Plant Unit 2 Technical Specifications, as described in Section 1.2 of this report.

1.2 CURRENT LICENSE BASIS AND FUNCTION OF IDENTIFIED TECHNICAL SPECIFICATIONS AND DESCRIPTION OF PROPOSED CHANGE

The proposed changes to the Donald C. Cook Nuclear Plant Unit 2 Technical Specifications are summarized in Table 1.2-1. These changes reflect the impact on the design, analytical methodology, and safety analysis assumptions outlined in this amendment request. The proposed Technical Specification changes are included in Appendix A of this report. A brief overview of the significant Technical Specification changes follows.

The changes are based on analyses and evaluations associated with the Uprating Program. Since new analyses and evaluations were required to establish the acceptability of the NSSS Power level, several related Technical Specification relaxations were verified. In addition, selected editorial changes are proposed to improve consistency with the Unit 1 Technical Specifications. For completeness, Technical Specification changes associated with operation with a full core of VANTAGE 5 fuel are also included.

Power Uprating

Technical Specification Definition 1.3, RATED THERMAL POWER, is changed to be consistent with the uprated core thermal power that is assumed in the Uprating Program.

The safety limits in Figure 2.1-1 are revised to reflect limits at the uprated power and the bases discussion is revised to reflect the current fuel design.

OPΔT/OTΔT Setpoints

Technical Specification Table 2.2-1 lists the reactor protection system instrumentation trip setpoints for the various trip functions. The reactor trip setpoint limits specified in Table 2.2-1 are the nominal values at which the reactor trips are set for each functional unit.

The Thermal Overpower ΔT (OPΔT) trip function provides assurance of fuel integrity (e.g., no fuel melting and less than 1% cladding strain) under all possible conditions, limits the required range for Thermal Overtemperature ΔT (OTΔT) protection, and provides a backup to the High Neutron Flux trip.

The OTΔT trip function provides sufficient core protection to preclude departure from nucleate boiling (DNB) over a range of operating and transient conditions. The setpoint is automatically varied with temperature, pressure, and the axial power distribution. The F(ΔI) penalty function adjusts the trip setpoint for axial peaks greater than design.

Revisions to the limiting safety system settings for the OTΔT and OPΔT trip functions (Table 2.2-1, Notes 1, 2, 3, and 4) are proposed to maintain consistency with the non-LOCA

Accident Analysis. These trip functions provide primary protection against DNB and fuel centerline melting (excessive kw/ft) during postulated transients. The proposed settings have been based on the new core safety limits and account for instrument uncertainties. The reference temperatures in the trip function equations are now indicated values that apply over a temperature range that was analyzed and are specified in the notes (i.e., 547°F to 581.3°F).

DNB Parameters, RCS Pressure, T_{avg} and Flow

LCO 3.2.5, DNB Parameters, specifies RCS parameters assumed as initial conditions in the transient and accident analyses. The DNB parameters are limits on RCS temperature, pressure and flow derived from the initial conditions assumed in the accident analysis. The DNB pressure limits are given for both normal pressure operation and reduced pressure operation. The DNB limits represent indicated values which include an appropriate allowance for indication uncertainty. For the Unit 2 Uprating Program, AEPSC is responsible for defining the DNB limits. Information related to DNB parameter uncertainty on Unit 2 can be found in WCAP-12576, Rev. 1 of June 1990. Safety analysis limits used for the Uprating Program are summarized in Table 3.3-5.

ECCS Pump Flows

The restriction in LCO 3.5.2 that requires all safety injection crosstie valves to be open is deleted, based on the safety analysis at the uprated condition that supports crosstie valves closed.

Containment Internal Pressure

The maximum calculated post-accident containment pressure must remain below the containment design pressure of 12.0 psig. The results of the containment integrity analyses performed for the Uprating Program resulted in a maximum calculated containment pressure of 11.66 psig. Thus, the value in the Bases for LCO 3.6.1.4 and 3.6.1.5 (9.4 psig) are being revised to reflect the analysis results.

Condensate Storage Tank Minimum Volume

The current minimum usable volume in the Condensate Storage Tank is 175,000 gallons and has been verified to be adequate for the Uprating Program.

Pressure/Temperature Limits

Due to the increased irradiation of the reactor pressure vessel at the uprated power level, the applicable time limit of the current heatup and cooldown limits is slightly reduced.

Miscellaneous Changes

DNB Limits

The limit on minimum RCS average temperature is deleted from LCO 3.2.5. This change is included to make the DNB limits similar in format to the DNB limits for Unit 1 and other plants with Standard Technical Specifications.

ESF Trip Setpoints

The steam line pressure - low trip setpoint is reduced based on an earlier reload safety evaluation for Unit 2, cycle 10.

The low pressure SI trip setpoint is revised based on SECL-93-193. However, since SECL-93-193 did not include nominal allowable values, these values must be determined by AEPSC.

Low Flow Trip Setpoint

The reference flow used in Table 2.2-1 for the low flow trip setpoint is related to the total RCS flow in Table 3.2-1 of the DNB LCO. This change is included to make the low flow trip setpoint format similar to Unit 1.

Pressurizer Code Safety Valve Lift Setting Tolerance

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. Accident and safety analyses which require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The limit protected by this specification is the reactor coolant pressure boundary Safety Limit of 110% of design pressure.

The LCO 3.4.2 (MODES 4 and 5) and LCO 3.4.3 (MODES 1, 2, and 3) pressurizer code safety valve lift setting tolerance has been increased from ± 1% to ± 3%.

The acceptability of the increased safety valve tolerance has been established by evaluation or analysis of applicable events including loss of load, turbine trip, locked rotor, loss of normal feedwater, feedwater line break, and loss of all power to station auxiliaries.

TABLE 1.2-1
SUMMARY OF TECHNICAL SPECIFICATION CHANGES

PAGE	SECTION	DESCRIPTION OF CHANGE	BASIS
1-1	1.0	Revise definition of Rated Thermal Power	Uprated Power is 3600 MWt
2-2	2.1	Revised Safety Limit Figure	Uprated Power is 3600 MWt
B 2-1	B 2.1.1	Revise Bases to reflect current fuel design	CY-10 Fuel Design
2.5	2.2	Use DNB flow limit for the Reference flow for low flow trip setpoint	Provide format consistency with Unit 1
2-7 2-8 2-9 B 2-5	2.2 B 2.2.1	**OTΔT & OPΔT Setpoints** Revise OTΔT and OPΔT Trip Setpoint and Allowable Value notes.	Proposed settings based on new instrument uncertainties and analysis limit.
3/4 2-15 B 3/4 2-5	3.2.5 B3.2.5	**DNB Parameters** Revise DNB Parameters, RCS T_{avg} from 578.7°F to (581.3 + 5.1 - indication error) °F. Add RCS pressure limits for normal and reduced pressure operation. Limit = nominal pressure -63 psi + indication error. Indication error is responsibility of AEPSC. Delete min T_{avg} limit.	T_{avg} input assumption verified by reanalyses. Format based on Unit 1 DNB format and DNB limits in the LCO.
B 3/4 2-4a	3/4 2.3	Revise Bases to reflect current fuel design.	CY-10 fuel design
3/4 3-23		**Low Pressure SI Trip Setpoint**	Revised based on SECL-93-193
3/4 3-23 3/4 3-25	3.2	**Low Steamline Trip Setpoint**	Relaxed trip setpoint based on Rerating Program and RTSR.
3/4 4-16	4.4.6.2.1	Add P_{SI} term for reduced pressure operation.	Revision Based on Limits for Reduced Pressure operation.

3/4 4-4 3/4 4-5	3.4.2 3.4.3	**Safety Valve Lift Setting** Pressurizer code safety valve lift setting pressure tolerance increased to 3%	Relaxation based on evaluation or analysis of several limiting events.
3/4 4-25 3/4 4-26 B 3/4 4-6	3.4.9.1	**Heatup and Cooldown EFPY Limit**	Limits apply at reduced vessel EFPY due to uprated conditions
3/4 5-3 B 3/4 5-1a	3.5.2	**ECCS Crosstie Valves** Delete requirement that the safety injection crosstie valves must be open	The safety analysis does not assume that the crosstie valves must be open.
B 3/4 6-2	B 3.6.1.4	**Containment Internal Pressure & Temp.** The value of peak containment pressure in the Bases for LCO 3.6.1.4 and 3.6.1.5 are being revised to 11.66 psig to reflect the analysis results	The results of the containment integrity analyses performed for the SGTP program resulted in a maximum calculated containment pressure of 11.66 psig.
3/4 7-7		Define CST volume limit in terms of usable volume	Utility preference.
B 3/4 7-3		Define minimum usable CST volume in Bases	Basis revised to clarify CST minimum contained volume.

2.0 BASIS FOR EVALUATIONS/ANALYSES PERFORMED

The purpose of the Uprating Program was to perform the necessary NSSS-related efforts to support an increase in the NSSS power level to 3600 MWt and continue operational flexibility in terms of primary temperature and pressure. In addition to the change in parameters associated with the increased NSSS power level, additional changes were incorporated into the analyses, as described in Section 1.0 (e.g., ECCS pump degradation, pressurizer safety valve tolerance, etc.).

Previously, AEPSC submitted a report for NRC review in October 1988, which provided the necessary analysis, documentation, and licensing effort to support operation at reduced primary temperatures and pressures. These analyses were performed in an effort to reduce the propensity for the initiation and propagation of corrosion in the Cook Nuclear Plant Unit 1 Series 51 steam generator tubes. The Westinghouse input for this submittal was provided in WCAP-11902 (Reference 1). The efforts performed for WCAP-11902 supported 100% thermal power operation (3250 MWt core power) in the range of vessel average temperatures between 547°F and 576.3°F, at primary pressure values of 2100 psia and 2250 psia. The primary pressures were intended as two discrete values; the program was not structured to support a continuous range of primary pressures. The intent of the reduced primary pressure value is to minimize the primary to secondary pressure drop across the steam generator tubes at reduced temperature operation. In addition, the analyses and evaluations performed support a maximum average tube plugging level of 10%, with a peak steam generator tube plugging (SGTP) level of 15%.

A supplement to WCAP-11902 was issued as the Westinghouse input for a second submittal to the NRC to summarize the additional efforts performed to support a rerating of Cook Nuclear Plant Unit 1 and to provide part of the support for a Unit 2 rerating (Reference 2). The impact of this document on Cook Nuclear Plant Unit 2 is to support the licensing of a power uprating (in addition to supporting the range of operating conditions described above) to 3600 MWt NSSS. Only the reduced temperature and pressure portion of this program and associated operational improvements have been approved at this time. AEPSC currently selects the desired operating conditions from within the range addressed in the Rerating Program on a cycle-to-cycle basis. The efforts documented in WCAP-11902 and Supplement are referred to throughout this report as the "Rerating Program".

Subsequent to the submittal of WCAP-11902, Supplement 1, a submittal was made to the NRC for the core related accident analyses at 3600 MWt as part of the Cycle 8 reload analysis (Reference 3). Only the Large Break Loss-of-Coolant Accident (LOCA) and long term containment integrity analyses did not support operation at 3600 MWt.

The RCS temperatures of the Uprating Program were chosen to be within the bounds of the Rerating Program. The two primary pressure values of 2100 psia and 2250 psia were

evaluated. The maximum average and peak SGTP level remain unchanged at 10% average SGTP and 15% peak. Because the range of NSSS parameters was chosen to be within the bounds of the Rerating Program, many of the analyses performed for the Rerating Program (Reference 1 and 2) remain applicable to the Uprating Program. Upon approval of the analyses and evaluations in this report, AEPSC will select the desired operating conditions from within the range addressed in the Uprating Program on a cycle to cycle basis.

References

1. WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report", October 1988.

2. WCAP-11902, Supplement, "Rebated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 and 2 Licensing Report," September 1989.

3. VANTAGE 5 Reload Transition Safety Report for Donald C. Cook Nuclear Plant Unit 2, Revision 1, March 1990.

2.1 DESIGN POWER CAPABILITY PARAMETERS

This section describes the parameters which were used as the basis for the evaluations and analyses performed to support the Uprating Program for Cook Nuclear Plant Unit 2. The NSSS performance parameters feature the uprated NSSS power of 3600 MWt, a T_{avg} temperature range from 547°F to 581.3°F, two primary pressure values of 2250 psia or 2100 psia, an average SGTP level of 10%, a peak SGTP level of 15% and a TDF of 88,500 gpm/loop. The RCS temperature range is bounded by the Rerating Program.

A brief description of each set of parameters is provided below:

Case 1: These are the original NSSS performance parameters for Unit 2 and are shown for comparison with the revised parameters. These parameters incorporate an NSSS power of 3403 MWt and 0% SGTP.

Case 2: These parameters incorporate a core power level of 3588 MWt, an NSSS power level of 3600 MWt (which includes 12 MWt of reactor coolant pump heat), an average steam generator tube plugging level of 10%, primary pressures of 2100 psia, and a lower bound vessel average temperature of 547.0°F.

Case 3: These parameters incorporate the same features as case 2, except that the primary temperatures incorporate an upper bound vessel average temperature of 581.3°F. This case was used as the basis for selected analyses, where high primary temperatures were limiting.

Case 4: These parameters incorporate the same features as case 2, except that the primary pressure was revised to 2250 psia.

Case 5: These parameters incorporate the same features as case 3, except that the primary pressure was revised to 2250 psia.

The Uprating Program NSSS performance parameters incorporate the current fuel, 17x17 VANTAGE 5, and also replacement steam generators.

TABLE 2.1-1
COOK NUCLEAR PLANT UNIT 2 NSSS PERFORMANCE PARAMETERS FOR UPRATING PROGRAM

Parameter	(Unit 2, Original) Case 1
NSSS Power, MWt	3403
Core Power, MWt	3391
RCS Flow,(gpm/loop)*	88,500
Minimum Measured Flow, (total gpm)	366,400
RCS Temperatures, °F	
Core Outlet	609.1
Vessel Outlet	606.4
Core Average	576.8
Vessel Average	573.8
Vessel/Core Inlet	541.3
Steam Generator Outlet	541.0
Zero Load	547.0
RCS Pressure, psia	2250
Steam Pressure, psia	820
Steam Flow, (10^6 lb/hr.tot.)	14.74
Feedwater Temperature, °F	431
% SG Tube Plugging	0

Flow Definitions:

* RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

TABLE 2.1-1 (continued)
COOK NUCLEAR PLANT UNIT 2 NSSS PERFORMANCE PARAMETERS
FOR UPRATING PROGRAM[1]

Parameter	Case 2	Case 3	Case 4	Case 5
NSSS Power, MWt	3600	3600	3600	3600
Core Power, MWt	3588	3588	3588	3588
RCS Flow, (gpm/loop)[2]	88,500	88,500	88,500	88,500
Minimum Measured Flow, (total gpm)[3]	366,400	366,400	366,400	366,400
RCS Temperatures, °F				
Core Outlet	585.7	618.2	585.8	618.4
Vessel Outlet	582.2	615.0	582.3	615.2
Core Average	550.1	584.9	550.1	584.8
Vessel Average	547.0	581.3	547.0	581.3
Vessel/Core Inlet	511.8	547.6	511.7	547.3
Steam Generator Outlet	511.5	547.4	511.4	547.1
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2100	2100	2250	2250.
Steam Pressure, psia	587	820	587	820
Steam Flow, (10^6 lb/hr.tot.)	15.90	16.00	15.90	16.00
Feedwater Temperature, °F	449	449	449	449
% SG Tube Plugging	10 avg/ 15 peak	10 avg/ 15 peak	10 avg/ 15 peak	10 avg/ 15 peak

1 Cases 2, 3, 4 & 5 are identical to cases 7, 5, 6, & 4, respectively, of Table B-2.1 of the VANTAGE 5 RTSR.

2 RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based upon this flow.

3 Minimum Measured Flow - The flow specified in the Technical Specifications which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Revised Thermal Design Procedure. MMF based upon a 3.5% flow measurement uncertainty.

2.2 NSSS DESIGN TRANSIENTS

The NSSS design transients evaluation for the Donald C. Cook Nuclear Plant Unit 2 Uprating Program was completed and confirmed that the NSSS design transients developed as part of the Donald C. Cook Nuclear Plant Units 1 and 2 Rerating Program continue to apply to Donald C. Cook Nuclear Plant Unit 2 at the uprating conditions with the exception of the Loss of Load and Loss of Offsite Power design transients. The evaluation consisted of a comparison of the NSSS performance parameters for the Uprating Program with the parameters for the Rerating Program. The comparison concluded that the Uprating Program parameters do not impact the NSSS design transients (i.e., temperatures, pressures, and power levels) and are bounded by the parameters used in the Rerating Program. The Loss of Load and Loss of Offsite Power design transients were modified to support the 3% Pressurizer Safety Valve setpoint tolerance.

2.3 CONTROL/PROTECTION SYSTEM SETPOINTS

Control Systems were evaluated and found to be bounded by the analyses performed as part of the Rerating Program. These analyses reflected the objective of optimizing control parameters, primarily with respect to two aspects of plant behavior: stability of the control systems and operating margins to the various reactor protection system trips.

The flexibility identified during the Rerating Program to adjust the full load vessel average temperature and primary pressure as necessary on a cycle-to-cycle basis remains applicable to the Uprating Program. Control systems setpoints are selected for each fuel cycle from those analyzed for the Rerating Program. Therefore, the plant will be adequately stable for all Uprating Program operating conditions and will operate with adequate margin to reactor protection system setpoints.

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.

<u>Analysis for RHR Crosstie Valves Open</u>

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 Conclusions

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.

REFERENCES

1. Gergos, B. W., Editor, "VANTAGE 5 Reload Transition Safety Report for Donald C. Cook Nuclear Plant Unit 2," Revision 2, September 1990.

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. 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.

3.1.2 Small Break LOCA

3.1.2.1 Introduction

The current licensing basis small break LOCA analysis for Donald C. Cook Nuclear Plant Unit 2 was performed to support operation with VANTAGE 5 fuel, and the analysis was subsequently updated to support an increase in the Main Steam Safety Valve (MSSV) setpoint tolerance from ±1% to ±3%. The analyses are described in the VANTAGE 5 Reload Transition Safety Report (RTSR), Reference 1, and in the safety evaluation for the MSSV setpoint tolerance relaxation, Reference 2. The RTSR and MSSV setpoint tolerance small break LOCA analyses are summarized in Section 3.1.2.2 and Section 3.1.2.3 of this report, respectively, 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 3.1.2.4. The RTSR and MSSV setpoint tolerance analyses were used as the basis for the uprating analyses, and thus the new analyses represent sensitivity studies relative to the current licensing basis analysis. The uprating sensitivity studies have confirmed the limiting break size which was established in the RTSR analysis. However, the operating conditions for the proposed licensing basis for the Uprating Program will rest on the RTSR analysis, since the RTSR analyses to establish the limiting operating temperature and pressure conditions were not repeated in the Uprating Program sensitivity analyses.

The RTSR and MSSV setpoint tolerance analyses indicated that acceptable small break LOCA results could be obtained for operation at a core power level of 3588 MWt with the High Head Safety Injection (HHSI) crosstie valves open, but that it would be necessary to reduce the core power level below 3588 MWt in order to obtain acceptable results with the HHSI crosstie valves closed. However, the uprating analyses have demonstrated that an acceptable peak clad temperature (PCT) can be obtained at 3588 MWt with the HHSI crosstie valves closed. The improvement in the small break LOCA analysis results for Donald C. Cook Nuclear Plant Unit 2 is due to the combined effect of the changes in the small break LOCA evaluation model to model SI in the broken loop and to incorporate an improved condensation model. The results of the Uprating Program analyses are discussed in Section 3.1.1.4.

3.1.2.2 RTSR Small Break LOCA Analysis

The licensing basis small 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 small break LOCA analysis for the VANTAGE 5 reload is presented in the VANTAGE 5 RTSR (Reference 1), and the results of the analysis are summarized below.

The small break LOCA analysis for the RTSR was performed to support operation at a core power level of 3588 MWt with the HHSI crosstie valves open. The analysis was performed with the Westinghouse Small Break LOCA ECCS Evaluation Model using the NOTRUMP

Code (References 3 and 4). Peak clad temperature calculations were performed with the LOCTA-IV code (Reference 5) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. The analysis was performed for a spectrum of break sizes in the cold leg, since the cold leg has been shown to represent the limiting break location for the small break LOCA analysis. The pumped safety injection flow rates used in the analysis were based on minimum emergency core cooling system availability with the HHSI crosstie valves open. In addition, the HHSI and charging pump performance curves were degraded 10 percent from the design head. The effect of flow from the RHR pumps is not considered in the small break LOCA analyses since their shutoff head is lower than the RCS pressure during the time portion of the transient considered in the analysis.

The RTSR small break LOCA analysis was performed for operation at a reactor power level of 3588 MWt for a range of RCS temperatures, and for a nominal RCS pressure of 2100 or 2250 psia, similar to the large break LOCA analysis. Table 3.1-13 summarizes the key input parameters used for the analysis. The analysis was performed for 3-inch, 4-inch, and 6-inch diameter breaks in the cold leg in order to determine the limiting break size. The limiting break was found to be a 4-inch diameter cold leg break initiated at reduced pressure and high temperature conditions (initial RCS pressure of 2100 psia and initial hot leg temperature of 615.2°F). The PCT attained during the transient was 1357°F, which is well below the acceptance limit of 2200°F in 10 CFR 50.46 (Reference 6). The key transient event times for the three break spectrum analysis performed at the reduced RCS pressure and high temperature conditions are listed in Table 3.1-14, and the results are summarized in Table 3.1-16.

Analyses were also performed to determine the influence of the initial RCS coolant operating temperatures and operating pressures on the small break LOCA PCT. To support operation of Donald C. Cook Nuclear Plant Unit 2 at RCS pressures of 2100 psia or 2250 psia for a range of loop operating temperatures, two additional analyses were performed. Calculations were performed for the limiting 4-inch diameter break for an initial RCS pressurizer pressure of 2250 psia and initial hot leg temperature of 615.2°F, and for an initial RCS pressurizer pressure of 2250 psia and initial hot leg temperature of 582.2°F. The sequence of events for these calculations are shown in the Table 3.1-15 and the results are summarized in Table 3.1-17. The results of these analyses demonstrated that the reduced pressure and high temperature conditions analyzed resulted in the limiting PCT for the 4-inch diameter break.

Additional calculations were made to support closure of the HHSI crosstie valves. Since the amount of pumped injection flow is reduced with the HHSI crosstie valves closed, it was necessary to lower core power in order to maintain the PCT within the 10 CFR 50.46 limit. The calculation which supports plant operation with the high head crosstie valves closed assumed an initial RCS pressurizer pressure at 2100 psia and an initial hot leg temperature of 615.2°F at a core power level of 3413 Mwt. Past experience indicated that the reduced injection flow for the HHSI crosstie valves closed, in conjunction with the reduced reactor

power, would shift the limiting break from the 4-inch diameter cold leg break to the 3-inch diameter break. Thus, analyses were performed for both 3-inch and 4-inch break diameters at the reduced pressure, high temperature initial conditions. The sequence of events for the crosstie closed cases are presented in Table 3.1-18 and the results are summarized in Table 3.1-19. The results for the crosstie closed cases in Table 3.1-19 show that the 3-inch diameter break is limiting with the reduced safety injection flow. The PCT calculated for the 3-inch break at 3413 MWt with the HHSI crosstie valves closed is 2124°F, which is below the acceptance limit of 2200°F in 10 CFR 50.46.

Thus, the RTSR small break LOCA analyses demonstrated that the results would be acceptable for operation of Donald C. Cook Nuclear Plant Unit 2 at 3588 MWt with the HHSI crosstie valves open, and at 3413 MWt with the HHSI crosstie valves closed.

3.1.2.3 MSSV Setpoint Tolerance Relaxation

Subsequent to the RTSR analyses, additional small break LOCA analyses were performed to support an increase in the MSSV lift setpoint tolerance from ±1% to ±3%. The results of the small break LOCA analyses performed for a ±3% MSSV setpoint tolerance are reported in Reference 2. The RTSR analyses demonstrated that reduced pressure and high temperature represent the limiting operating conditions for the small break LOCA analysis, and that the 4-inch break is the limiting break size with the HHSI crosstie valves open. The RTSR analyses also demonstrated that the reduction in pumped safety injection flow associated with closing the HHSI crosstie valves caused the limiting break size to shift from the 4-inch to the 3-inch break, and that the HHSI crossties closed case with reduced power resulted in the most limiting clad temperature. Thus, the small break LOCA analysis for a ±3% MSSV setpoint tolerance was performed for the limiting 3-inch break with the crosstie valves closed. A reduced core power level of 3250 MWt was assumed in the analysis, since the power level of 3413 MWt used in the RTSR analysis with the crossties closed could not be supported. An additional analysis was performed for a 4-inch break at 3588 MWt with the crossties open, which confirmed that the 3-inch break with the crossties closed remained the limiting case. A list of the major input parameters used in the analyses for the HHSI crosstie valves closed at 3250 MWt and for the HHSI crosstie valves open at 3588 MWt are presented in Table 3.1-20.

The sequence of events and the results of the 3-inch break analysis at 3250 MWt with the HHSI crosstie valves closed are presented in Table 3.1-21 and Table 3.1-22, respectively. The comparable results for the 4-inch break at 3588 MWt with the HHSI crosstie valves open are provided in Tables 3.1-23 and 3.1-24. As shown in Table 3.1-22, the PCT for the 3-inch break at 3250 MWt with the crosstie valves closed is 2124.9°F, which remains less than the 2200°F limit. From Table 3.1-24, the PCT for the 4-inch break at 3588 MWt with the crosstie valves open is 1543°F. Thus, the 3-inch break at 3250 MWt with the crossties closed is more limiting than the 4-inch break at 3588 MWt with the crossties open.

3.1.2.4 Uprating Program Small Break LOCA Analysis

The RTSR small break LOCA analyses indicated that acceptable results could be obtained for operation at a core power level of 3588 MWt with the HHSI crosstie valves open, and that the core power would have to be reduced from 3588 to 3413 MWt for operation with HHSI crosstie valves closed. The analyses perfomed to support the increase in the MSSV setpoint tolerance to ±3% indicated that the core power would have to be reduced from 3588 to 3250 MWt for operation with the HHSI crosstie valves closed. However, subsequent changes in the small break LOCA evaluation model with NOTRUMP, including the modelling of safety injection in the broken loop in conjunction with the COSI condensation model, have resulted in net PCT benefits for many plants. Therefore, the small break LOCA transient was reanalyzed for the uprated power of 3588 MWt with the HHSI crosstie valves closed to determine if operation at the higher power level with the crosstie valves closed would be acceptable.

Previously, safety injection into the broken loop was not modeled in the Westinghouse small break LOCA analysis since it was assumed that the additional safety injection would result in a PCT benefit. Because more recent studies have shown that the response to broken loop safety injection can result in an increase in the calculated PCT, modelling of safety injection into the broken loop has now been incorporated into the NOTRUMP small break evaluation model. A more realistic model for condensation of steam in the cold leg by pumped safety injection based on data from the COSI test facility has also been incorporated, which provides a benefit larger than the penalty for safety injection in the broken loop. The methodology for modelling safety injection to the broken loop in small break LOCA analyses and application of the COSI condensation model, are presented in the NOTRUMP Small Break ECCS Evaluation Model, Addendum 2 (Reference 7). The small break LOCA analysis perfomed for Donald C. Cook Nuclear Plant Unit 2 at the uprated power of 3588 MWt with the HHSI crosstie valves closed modelled pumped safety injection and an accumulator in the broken loop, and used the more realistic COSI condensation model described in Reference 7.

The safety injection flows used for the analysis are based on minimum emergency core cooling system availability with the HHSI crosstie valves closed. The safety injection flow rates are also based on a pump head degradation of 15% for the HHSI pumps and 10% for the centrifugal charging pumps (CCP). As noted previously, the effect of flow from the RHR pumps is not incorporated in the small break analysis. The pumped safety injection characteristics as a function of RCS pressure used as boundary conditions in the analysis are shown in Figure 3.1-27 and in Table 3.1-25. The safety injection flow rates are provided for the intact loops and also for the broken loop.

A list of the major input parameters used in the analysis is presented in Table 3.1-26. Figure 3.1-28 depicts the hot rod axial power shape used to perform this analysis at the uprated power. This shape is based on the maximum axial offset of 13%, and was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small-break LOCAs because it minimizes coolant level swell,

while maximizing vapor superheating and fuel rod heat generation in the uncovered elevations.

The small break LOCA analysis for the Uprating Program was performed at 3588 MWt for the reduced pressure and high temperature conditions (initial RCS pressure of 2100 psia and initial hot leg temperature of 615.2°F) which were demonstrated to be limiting in the RTSR analysis. The analysis was performed for 2-inch, 3-inch, and 4 inch diameter breaks in the cold leg. The limiting break was found to be a 3-inch diameter cold leg break, and the peak clad temperature attained during the transient was 2065°F. As shown in the RTSR and increased MSSV setpoint tolerance analyses, the reduction in delivered safety injection flow with HHSI crosstie valves closed places more importance on the break size as the primary energy removal mechanism, resulting in the same limiting 3-inch diameter break. The key transient event times and the results for the 3-inch break analysis performed at reduced RCS pressure and high temperature conditions are presented in Table 3.1-27 and Table 3.1-28, respectively.

Plots for the following parameters are shown in Figures 3.1-29 through 3.1-36 for the limiting 3-inch break transient:

- RCS pressure
- Core mixture level
- Peak clad temperature
- Core outlet steam mass flow rate
- Hot spot rod surface heat transfer coefficient
- Hot spot fluid temperature
- Cold leg break mass flow rate
- Safety injection mass flow rate

The results of the calculations for the 2-inch and 4-inch breaks are also shown in the Sequence of Events Table 3.1-27 and Results Table 3.1-28. Plots of the following parameters are shown in Figures 3.1-37 through 3.1-43 for the 2-inch break and Figures 3.1-44 through 3.1-50 for the 4-inch break:

- RCS pressure
- Core mixture level
- Peak clad temperature
- Core outlet steam mass flow rate
- Hot spot rod surface heat transfer coefficient
- Hot spot fluid temperature
- Cold leg break mass flow rate

As seen in Table 3.1-28, the peak clad temperatures were calculated to be less than that for the 3-inch break.

As shown in Table 3.1-28, the calculated PCT for operation at 3588 MWt with the HHSI crosstie valves closed is 2065°F for the limiting 3-inch break, which is well below the 10 CFR 50.46 limit of 2200°F. The maximum local metal-water reaction is 13.70%, 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 small break LOCA for operation at 3588 MWt with the HHSI crosstie valves closed.

3.1.2.5 Conclusions

Based on the small 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 HHSI crosstie valves closed is acceptable.

REFERENCES

1. Gergos, B. W., Editor, "VANTAGE 5 Reload Transition Safety Report for Donald C. Cook Nuclear Plant Unit 2," Revision 2, September 1990.

2. "Main Steam Safety Valve Lift Setpoint Tolerance Relaxation," SECL-91-429, Revision 2, December 1993.

3. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.

4. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," WCAP-10054-P-A, August 1985.

5. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974, WCAP-8201, (Proprietary), June 1974.

6. "Acceptance Criteria for Emergency Core Cooling Systems for Water Cooled Nuclear Power Reactors" 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.

7. Thompson, C. M., et. al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model," WCAP-10054-P, Addendum 2, (Proprietary) and WCAP-10081-NP, Addendum 2, (Non-Proprietary), August 1994.

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

<u>CALCULATION ASSUMPTIONS</u>

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, 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%

DONALD C. COOK NUCLEAR PLANT UNIT 2

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	C_D=0.4 Min SI 3588 Mwt	C_D=0.8 Min SI 3588 Mwt	C_D=0.6 Min SI 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
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 Min SI 3588 Mwt	C_D=0.4 Min SI 3588 Mwt	C_D=0.8 Min SI 3588 Mwt	C_D=0.6 Min SI 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

CALCULATION ASSUMPTIONS

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

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-13
RTSR ANALYSIS
SMALL BREAK LOCA ANALYSIS
PLANT INPUT PARAMETERS

Core Power	102% of 3588 MWt
Total Core Peaking Factor	2.32
Steam Generator Tube Plugging Level	15% (peak uniform)
Accumulator Conditions:	
Cover Gas Pressure	600 psia
Water Volume	946.0 ft^3
Total Volume	1350 ft^3
RCS Initial Conditions:	
Nominal Vessel Inlet Temperature	511.7 to 547.6°F
Nominal Vessel Outlet Temperature	582.2 to 615.2°F
Nominal Vessel Average Temperature	547.0 to 581.3°F
Nominal RCS Pressure	2100 or 2250 psia
Vessel Flowrate	354000 gpm
Reactor Trip Signal	1860 psia
Safety Injection Signal	1715 psia
Safety Injection Delay Time	27 seconds
Rod Drop Time	2.7 seconds

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-14
RTSR ANALYSIS
SMALL BREAK LOCA ANALYSIS
TIME SEQUENCE OF EVENTS

HHSI Crosstie Valves Open at 3588 Mwt

Event	Time (s) High Temperature, Reduced Pressure		
Break Size:	3-Inch	4-Inch	6-inch
Break occurs	0	0	0
Reactor trip signal	12.17	7.26	4.97
Safety injection signal	21.53	14.99	10.59
Start of safety injection delivery	48.53	41.99	37.59
Loop seal venting	589.7	333.1	154.9
Loop seal core uncovery	N/A	335.2	141.2
Loop seal core recovery	N/A	345.3	173.4
Boil-off core uncovery	1072.	640.9	388.3
Accumulator injection begins	1960.	861.6	366.4
Peak clad temperature occurs	1662.	919.7	168.6
Top of core covered	N/A	N/A	425.2
SI flow rate exceeds break flow rate	1515.	1606.	N/A

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-15
RTSR ANALYSIS
SMALL BREAK LOCA ANALYSIS
TIME SEQUENCE OF EVENTS

HHSI Crosstie Valves Open at 3588 Mwt

Event	Time (s) High Temp. High Pressure 4-Inch	Reduced Temp. High Pressure 4-Inch
Break occurs	0	0
Reactor trip signal	11.75	9.85
Safety injection signal	18.95	13.08
Start of safety injection delivery	45.95	40.08
Loop seal venting	333.7	346.9
Loop seal core uncovery	327.4	407.8
Loop seal core recovery	344.0	428.1
Boil-off core uncovery	673.1	664.7
Accumulator injection begins	878.7	874.3
Peak clad temperature occurs	943.6	942.0
Top of core covered	1722.0	1658.0
SI flow rate exceeds break flow rate	1515.0	1516.0

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-16
RTSR ANALYSIS
SMALL BREAK LOCA ANALYSIS RESULTS

HHSI Crosstie Valves Open at 3588 Mwt

PARAMETER	VALUE High Temperature, Reduced Pressure		
Break Size:	3-Inch	4-Inch	6-Inch
Peak clad temperature (°F)	1133	1357	959
Elevation (ft)	11.50	11.50	10.50
Zr/H_2O cumulative reaction			
Maximum local (%)	0.07	0.15	0.03
Elevation (ft)	11.50	11.50	10.50
Total core (%)	< 0.3	< 0.3	< 0.3
Rod Burst	None	None	None

CALCULATION:

Core Power MWt 102% of	3588
Peak Linear Power kw/ft 102% of	12.825
Accumulator Water Volume, cu. ft.	946

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-17
RTSR ANALYSIS
SMALL BREAK LOCA ANALYSIS RESULTS

HHSI Crosstie Valves Open at 3588 Mwt

PARAMETER	VALUE High Temp. High Pressure 4-Inch	Reduced Temp. High Pressure 4-Inch
Peak clad temperature (°F)	1325	1315
Elevation (ft)	11.50	11.50
Zr/H_2O cumulative reaction		
Maximum local (%)	0.13	0.11
Elevation (ft)	11.50	11.50
Total core (%)	< 0.3	< 0.3
Rod Burst	None	None

CALCULATION:

Core Power MWt 102% of	3588
Peak Linear Power kw/ft 102% of	12.825
Accumulator Water Volume, cu. ft.	946

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-18
RTSR ANALYSIS
SMALL BREAK LOCA ANALYSIS
TIME SEQUENCE OF EVENTS

HHSI Crosstie Valves Closed at 3413 Mwt

Event	Time (s) High Temp. Reduced Pressure 3-Inch	Time (s) High Temp. Reduced Pressure 4-Inch
Break occurs	0	0
Reactor trip signal	10.88	6.74
Safety injection signal	20.36	13.46
Start of safety injection delivery	47.36	40.46
Loop seal venting	611.9	357.1
Loop seal core uncovery	N/A	359.2
Loop seal core recovery	N/A	368.3
Boil-off core uncovery	962.0	611.2
Accumulator injection begins	1566.	839.2
Peak clad temperature occurs	1640.	908.0
Top of core covered	N/A	2356.
SI flow rate exceeds break flow rate	1915.	N/A

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-19
RTSR ANALYSIS
SMALL BREAK LOCA ANALYSIS RESULTS

HHSI Crosstie Valves Closed at 3413 Mwt

PARAMETER	VALUE High Temp. Reduced Pressure 3-Inch	VALUE High Temp. Reduced Pressure 4-Inch
Peak clad temperature (°F)	2124	1530
Elevation (ft)	12.00	11.50
Zr/H_2O cumulative reaction		
Maximum local (%)	8.64	0.37
Elevation (ft)	12.00	11.50
Total core (%)	< 0.3	< 0.3
Rod Burst	None	None

CALCULATION:

Core Power MWt 102% of	3413
Peak Linear Power kw/ft 102% of	12.756
Accumulator Water Volume, cu. ft.	946

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-20
±3% MSSV SETPOINT TOLERANCE ANALYSIS
SMALL BREAK LOCA ANALYSIS
PLANT INPUT PARAMETERS

	HHSI Crossties Closed	HHSI Crossties Open
Core Power (MWt)	102% of 3250	102% of 3588
Peak Linear Power (kW/ft)	102% of 12.226	102% of 12.764
Total Core Peaking Factor, F_Q	2.357	2.32
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.666	1.62
Maximum Assembly Average Power, P_{HA}	1.46	1.46
Axial Offset (%)	+13	+13
Fuel Assembly Array	17x17 V5	17x17 V5
Steam Generator Tube Plugging Level (%)	15	15
Accumulator Water Volume (ft^3/tank)	946	946
Accumulator Tank Volume (ft^3/tank)	1350	1350
Minimum Accumulator Gas Pressure (psia)	600	600
Accumulator Water Temperature (°F)	120	120
Refueling Water Storage Tank Temperature (°F)	120	120
Thermal Design Flowrate (gpm/loop)	88,500	88,500
Nominal Vessel Inlet Temperature (°F)	546.4	547.6
Nominal Vessel Outlet Temperature (°F)	611.0	615.0
Nominal Vessel Average Temperature (°F)	578.7	581.3
Nominal Initial RCS Pressure (psia)	2100	2100
Nominal Steam Pressure (psia)	820	820
Safety Injection Delay Time (sec)	27	27
HHSI Pump Head Degradation (%)	10	10
Charging Pump Head Degradation (%)	10	10
Charging Pump Flow Imbalance (gpm)	25	25
HHSI Cross-Tie Valve Position	Closed	Open

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-21
±3% MSSV SETPOINT TOLERANCE ANALYSIS
SMALL BREAK LOCA ANALYSIS
TIME SEQUENCE OF EVENTS

HHSI Crosstie Valves Closed at 3250 Mwt

Event	Time (sec) High Temperature Reduced Pressure 3-inch
Break Occurs	0
Reactor trip signal	11.01
Safety injection signal	20.92
Start of safety injection	47.92
Loop seal venting	620.0
Loop seal core uncovery	NA
Loop seal core recovery	NA
Boil-off core uncovery	976.9
Accumulator injection begins	1604.3
Peak clad temperature occurs	1691.0
Top of core covered	NA
SI flow rate exceeds break flow rate	1683.0

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-22
±3% MSSV SETPOINT TOLERANCE ANALYSIS
SMALL BREAK LOCA ANALYSIS RESULTS

HHSI Crosstie Valves Closed at 3250 Mwt

	High Temperature Reduced Pressure 3-Inch
NOTRUMP Peak Clad Temperature (°F)	1955.9
Peak Clad Temperature Location (ft)	11.75
Peak Clad Temperature Time (sec)	1691.0
Local Zr/H_2O Reaction Maximum (%)	4.26
Local Zr/H_2O Reaction Location (ft)	11.75
Total Zr/H_2O Reaction (%)	<1.0
Rod Burst	None
Artificial Leak-By Penalty (°F)	12
Burst and Blockage Penalty (°F)	157
Total Peak Clad Temperature (°F)	2124.9

CALCULATION:	
NSSS Power MWt 102% of	3250*
Peak Linear Power kw/ft 102% of	12.226
Accumulator Water Volume, cu. ft.	946

* Does not include pump heat

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-23
±3% MSSV SETPOINT TOLERANCE ANALYSIS
SMALL BREAK LOCA ANALYSIS
TIME SEQUENCE OF EVENTS

HHSI Crosstie Valves Open at 3588 Mwt

Event	Time (sec) High Temperature Reduced Pressure 4-inch
Break Occurs	0
Reactor trip signal	7.10
Safety injection signal	16.02
Start of safety injection	43.02
Loop seal venting	340
Loop seal core uncovery	402
Loop seal core recovery	448
Boil-off core uncovery	487
Accumulator injection begins	785
Peak clad temperature occurs	846
Top of core covered	1754
SI flow rate exceeds break flow rate	961

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-24
±3% MSSV SETPOINT TOLERANCE ANALYSIS
SMALL BREAK LOCA ANALYSIS RESULTS

HHSI Crosstie Valves Open at 3588 Mwt

	High Temperature Reduced Pressure 4-inch
NOTRUMP Peak Clad Temperature (°F)	1531
Peak Clad Temperature Location (ft)	11.25
Peak Clad Temperature Time (sec)	846.1
Local Zr/H_20 Reaction Maximum (%)	0.459
Local Zr/H_20 Reaction Location (ft)	11.25
Total Zr/H_20 Reaction (%)	<1.0
Rod Burst	None
Artificial Leak-By Penalty (°F)	12
Burst and Blockage Penalty (°F)	None
Total Peak Clad Temperature (°F)	1543

CALCULATION:

NSSS Power MWt 102% of	3588*
Peak Linear Power kw/ft 102% of	12.764
Accumulator Water Volume, cu. ft.	946

* Does not include pump heat

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-25
UPRATING PROGRAM
SMALL BREAK LOCA ANALYSIS
SAFETY INJECTION FLOW RATE
WITH HHSI CROSSTIE VALVES CLOSED

RCS Pressure (psia)	Intact Loop Charging Flow Rate (lb/s)	Broken Loop Charging Flow Rate (lb/s)	Intact Loop HHSI Flow Rate (lb/s)	Broken Loop HHSI Flow Rate (lb/s)	Intact Loop Total Flow Rate (lb/s)	Broken Loop Total Flow Rate (lb/s)
214.7	37.79	15.75	22.05	23.45	59.84	39.20
314.7	36.91	15.38	21.06	22.41	57.97	37.79
414.7	35.99	15.00	19.99	21.27	55.98	36.27
514.7	34.99	14.57	18.81	20.01	53.80	34.58
614.7	33.95	14.14	17.56	18.69	51.51	32.83
714.7	32.91	13.71	16.22	17.27	49.13	30.98
814.7	31.87	13.28	14.81	15.75	46.67	29.03
914.7	30.78	12.83	13.28	14.13	44.06	26.96
1014.7	29.70	12.39	11.41	12.15	41.11	24.54
1114.7	28.61	11.93	9.15	9.83	37.76	21.76
1214.7	27.53	11.47	6.27	6.67	33.80	18.14
1314.7	26.32	10.97	2.04	2.17	28.36	13.14
1414.7	25.07	10.44	0.00	0.00	25.07	10.44
1514.7	23.77	9.90			23.77	9.90
1614.7	22.44	9.34			22.44	9.34
1714.7	21.10	8.79			21.10	8.79
1814.7	19.56	8.15			19.56	8.15
1914.7	17.93	7.47			17.93	7.47
2014.7	16.27	6.77			16.27	6.77
2114.7	14.39	5.99			14.39	5.99
2214.7	12.14	5.06			12.14	5.06
2314.7	0.00	0.00			0.00	0.00

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-26
UPRATING PROGRAM
SMALL BREAK LOCA ANALYSIS
PLANT INPUT PARAMETERS

Core Power (MWt)	102% of 3588
Peak Linear Power (kW/ft)	102% of 12.735
Total Core Peaking Factor, F_Q	2.32
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.62
Maximum Assembly Average Power, $\overline{P}_{HA}$	1.443
Axial Offset (%)	+13
Fuel Assembly Array	17x17 V5
Steam Generator Tube Plugging Level (%)	15
Accumulator Water Volume (ft^3/tank)	946
Accumulator Tank Volume (ft^3/tank)	1350
Minimum Accumulator Gas Pressure (psia)	600
Accumulator Water Temperature (°F)	130
Refueling Water Storage Tank Temperature (°F)	120
Thermal Design Flowrate (gpm/loop)	88,500
Nominal Vessel Inlet Temperature (°F)	547.6
Nominal Vessel Outlet Temperature (°F)	615.0
Nominal Vessel Average Temperature (°F)	581.3
Nominal Initial RCS Pressure (psia)	2100
Nominal Steam Pressure (psia)	820
Safety Injection Delay Time (sec)	27
HHSI Pump Head Degradation (%)	15
Charging Pump Head Degradation (%)	10
Charging Pump Flow Imbalance (gpm)	25
HHSI Cross-Tie Valve Position	Closed

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-27
UPRATING PROGRAM
SMALL BREAK LOCA ANALYSIS
TIME SEQUENCE OF EVENTS

HHSI Crosstie Valves Closed at 3588 Mwt

Event	Time (s) High Temperature, Reduced Pressure		
Break Size:	2-Inch	3-Inch	4-Inch
Break occurs	0	0	0
Reactor trip signal	27.04	11.72	7.08
Safety injection signal	42.34	21.67	15.86
Start of safety injection delivery	69.34	48.67	42.86
Loop seal core uncovery	1307.	634.	311.
Loop seal venting	1332.	657.	325.
Loop seal core recovery	1333.	656.	325.
Boil-off core uncovery	1746.	752.	540.
Accumulator injection begins	>6000.	1520.	819.
Peak clad temperature occurs	3551.	1562.	894.
Top of core covered	>6000.	3826.	2955.

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 3.1-28
UPRATING PROGRAM
SMALL BREAK LOCA ANALYSIS RESULTS

HHSI Crosstie Valves Closed at 3588 MWt

PARAMETER	VALUE High Temperature, Reduced Pressure		
Break Size:	2-Inch	3-Inch	4-Inch
Peak clad temperature (°F)	1252.	2065.	1576.
Elevation (ft)	11.50	11.75	11.25
Zr/H_2O cumulative reaction			
Maximum local (%)	0.29	13.70	0.57
Elevation (ft)	11.50	11.75	11.25
Total core (%)	< 1.0	< 1.0	< 1.0
Rod Burst			
Time (s)	N/A	1560.	N/A
Elevation (ft)	N/A	11.75	N/A

CALCULATION:

Core Power MWt 102% of	3588
Hot Rod Peak Linear Power kw/ft 102% of	12.735
Hot Rod Linear Power Distribution (kw/ft)	See Figure 3.1-28
Accumulator Water Volume, cu. ft.	946

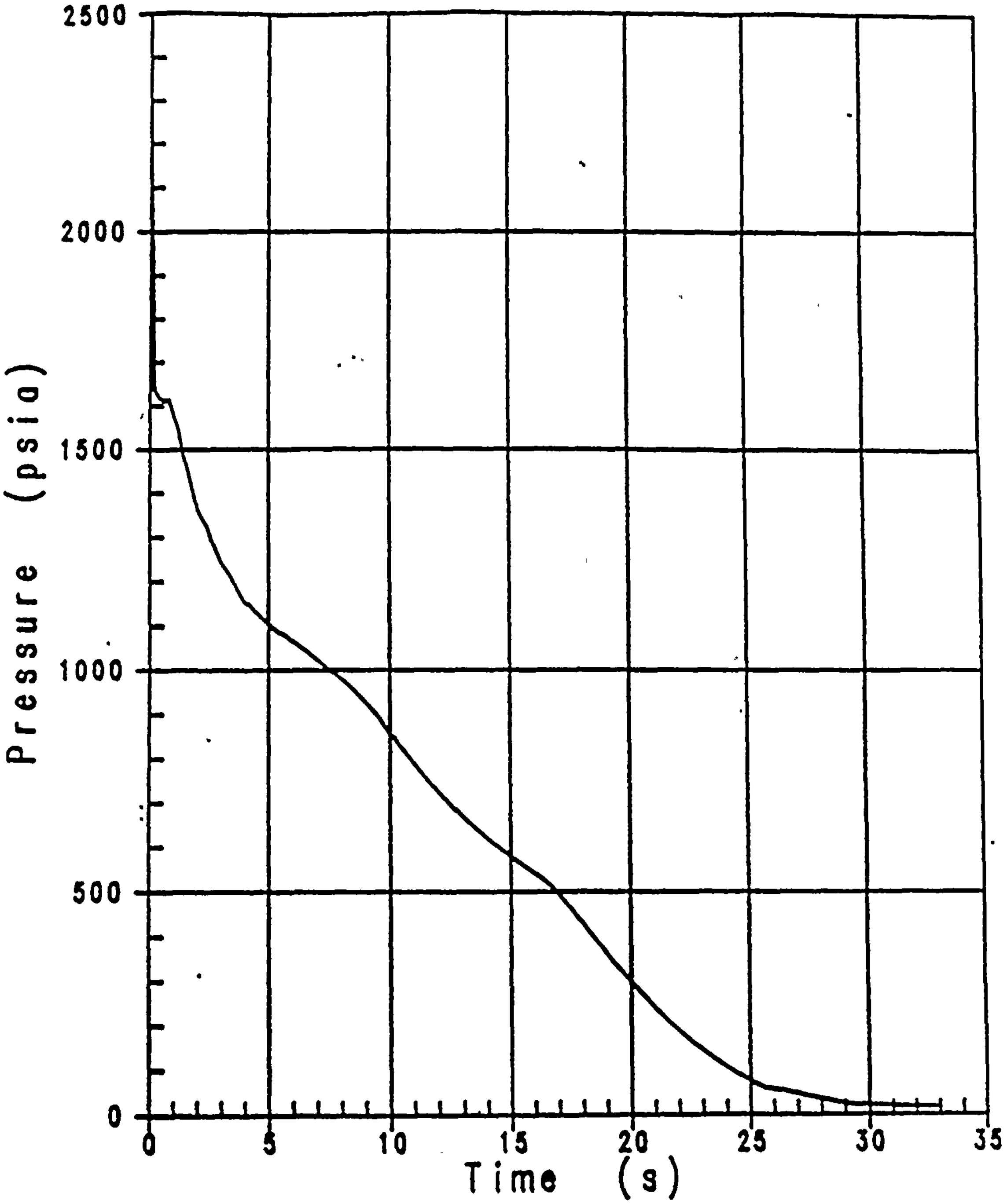


Figure 3.1-1 Reactor Coolant System Pressure Uprating Analysis–RHR Crosstie Closed

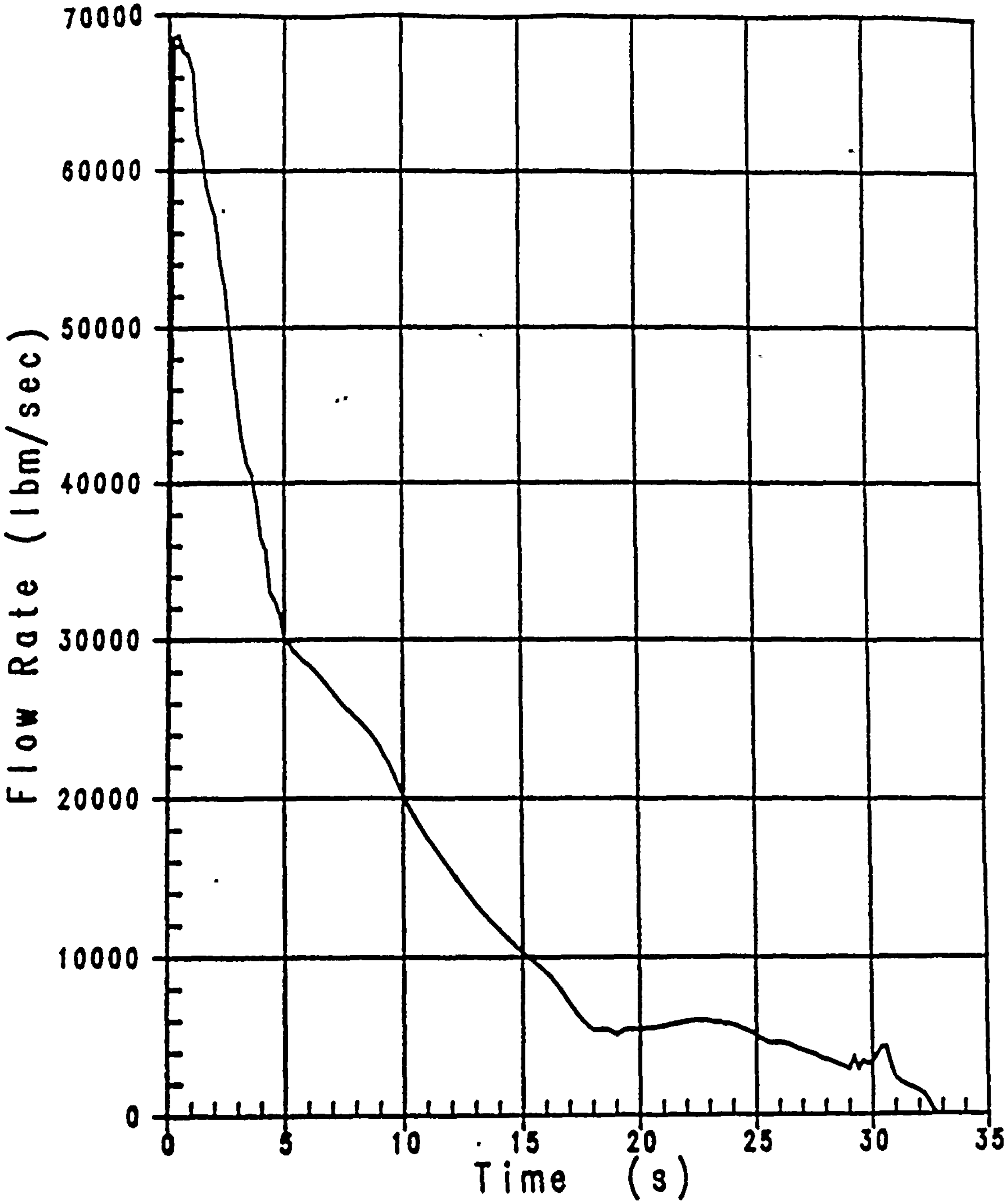


Figure 3.1-2 Break Flow During Blowdown
Uprating Analysis–RHR Crosstie Closed

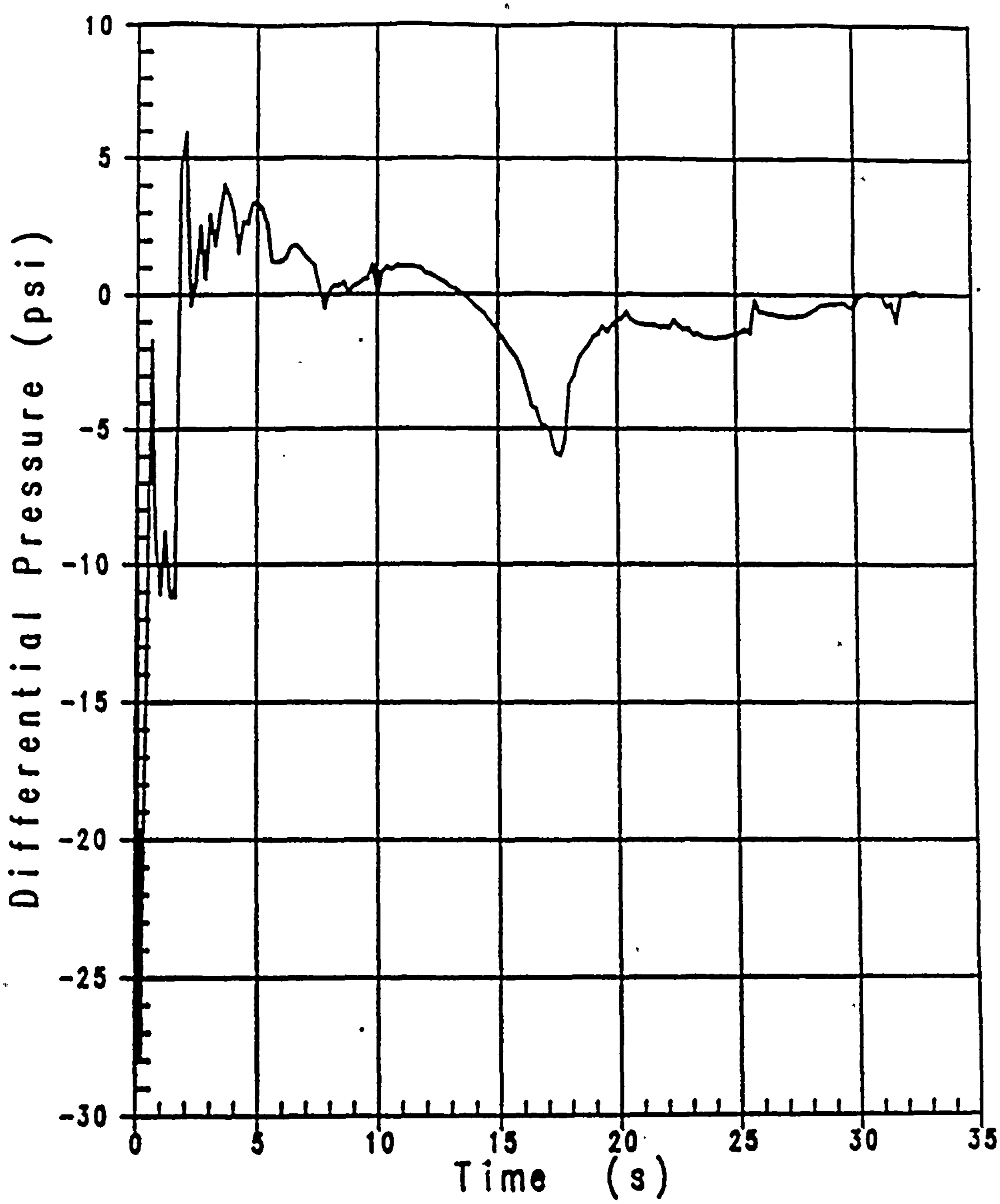


Figure 3.1-3 Core Pressure Drop
Uprating Analysis–RHR Crosstie Closed

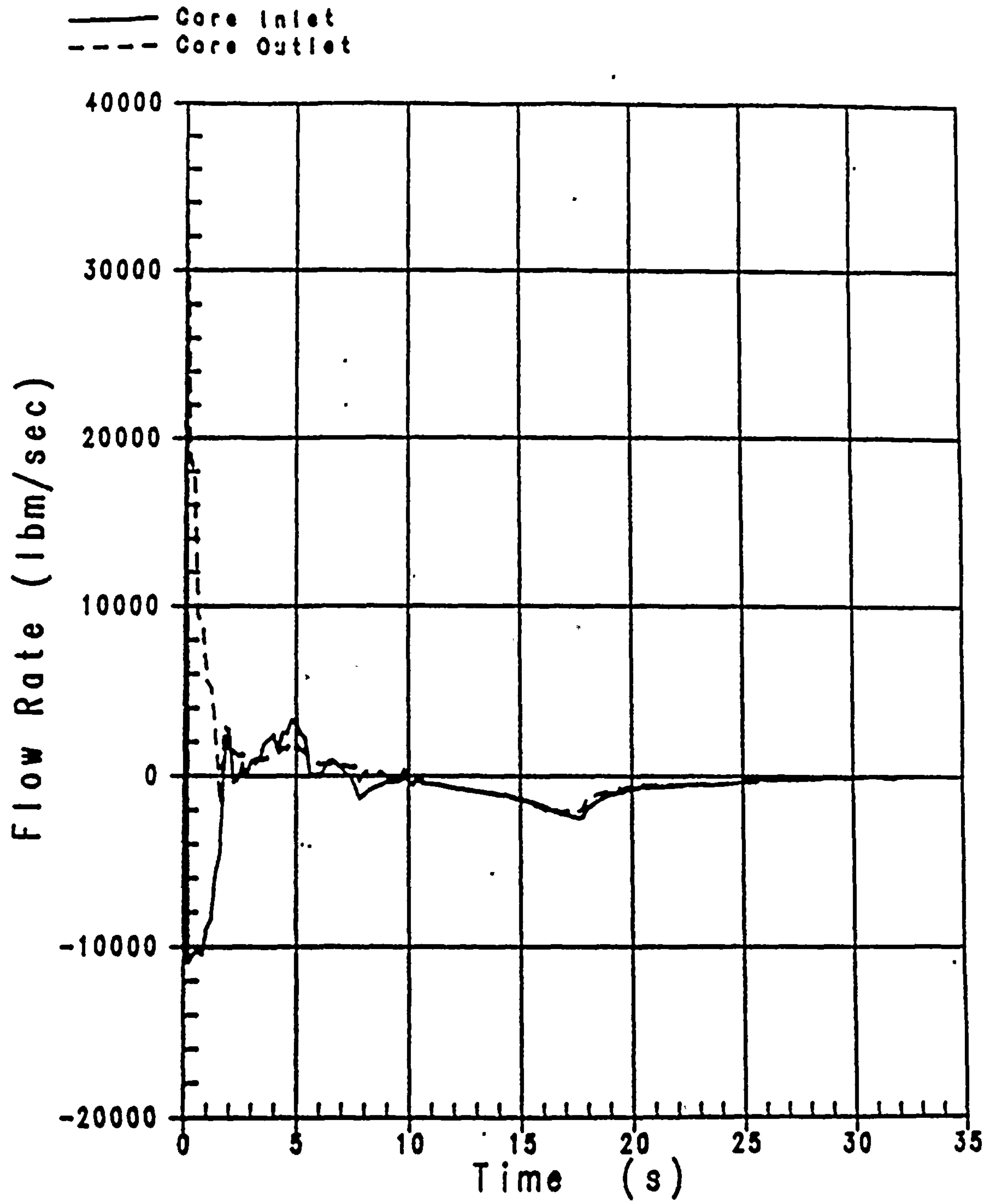


Figure 3.1-4 Core Flowrate
Uprating Analysis–RHR Crosstie Closed

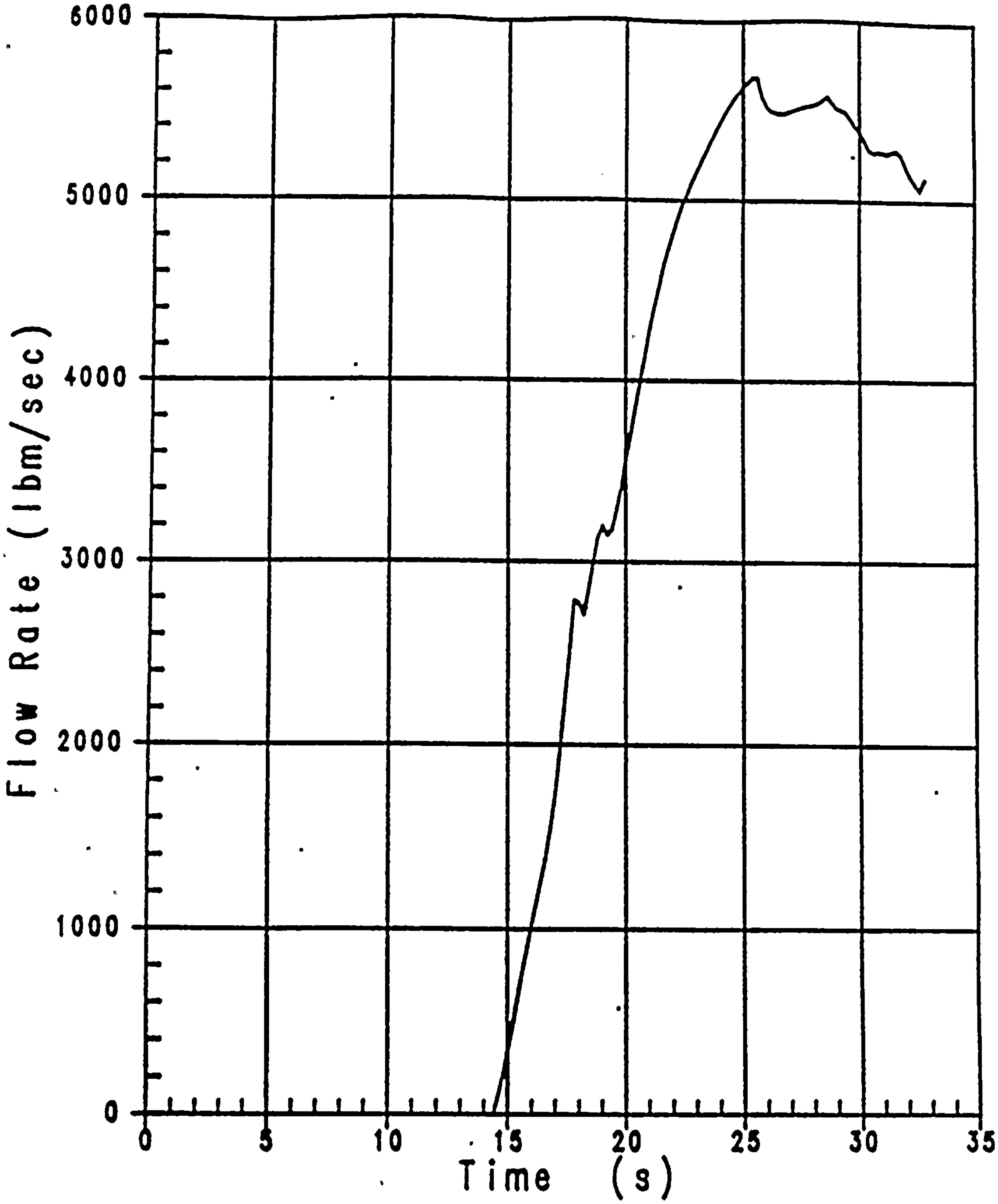


Figure 3.1-5 Accumulator Flow During Blowdown
Uprating Analysis–RHR Crosstie Closed

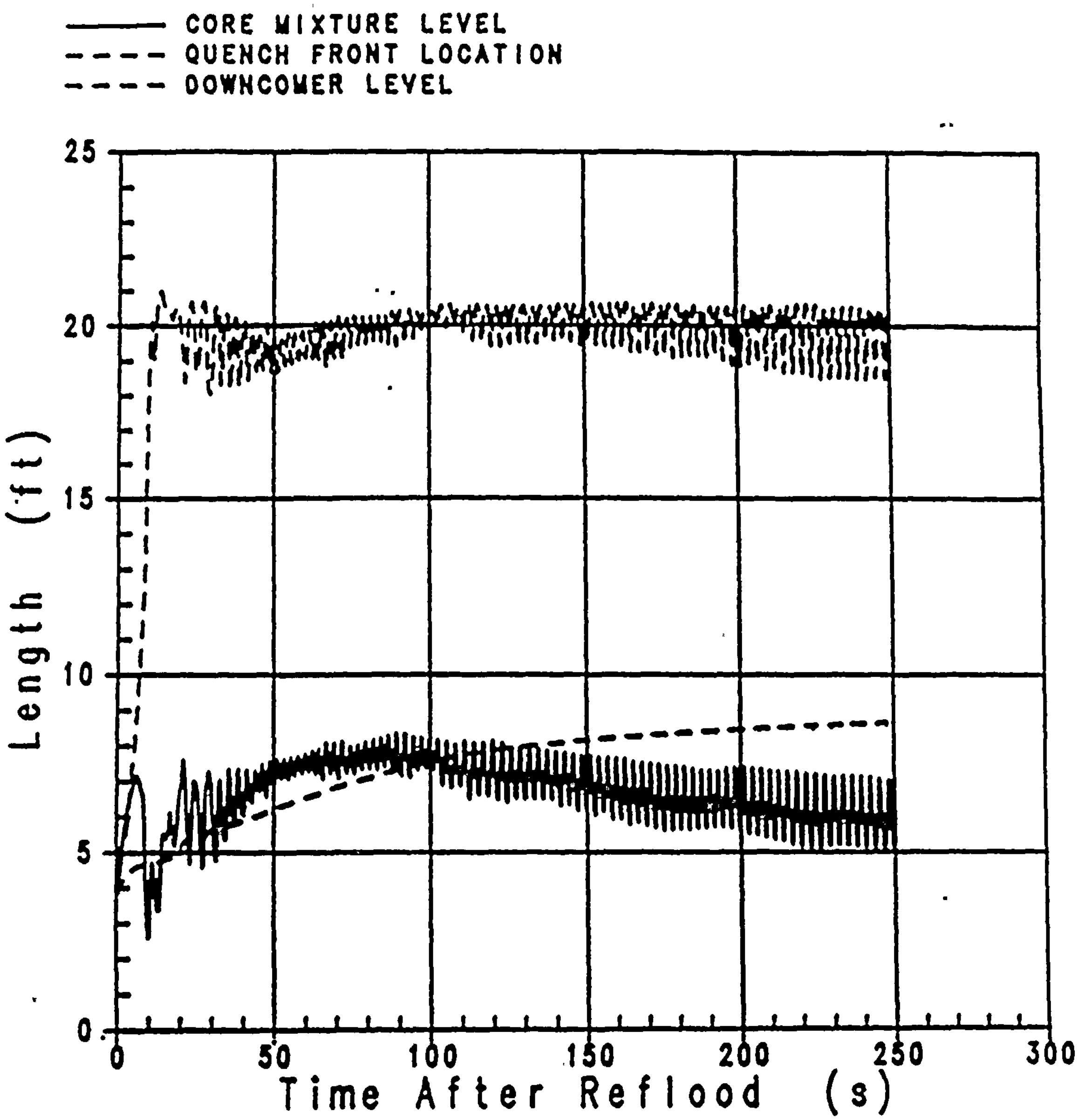


Figure 3.1-6 Vessel Liquid Levels During Reflood Uprating Analysis–RHR Crosstie Closed

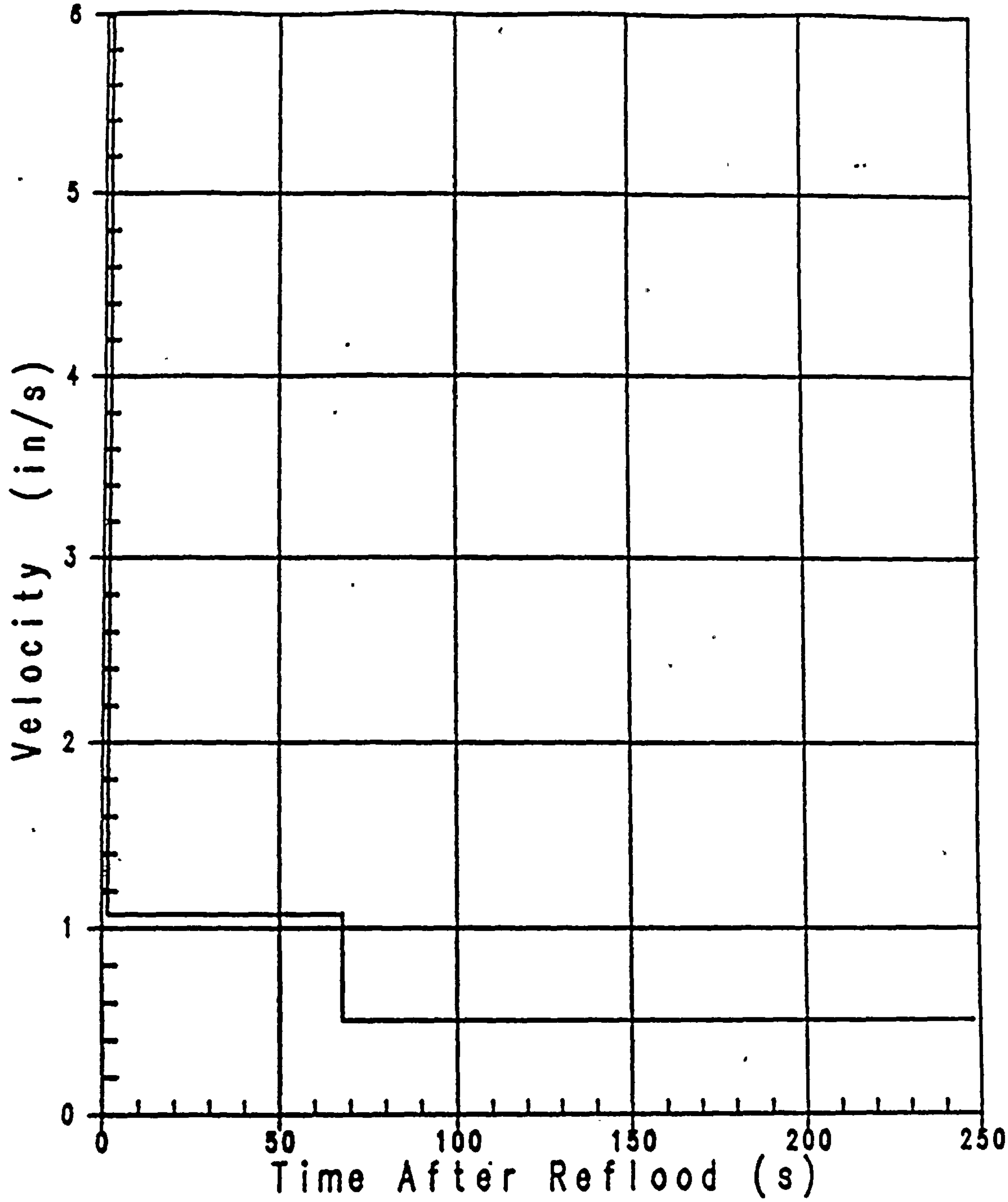


Figure 3.1-7 Core Inlet Flow During Reflood
Uprating Analysis–RHR Crosstie Closed

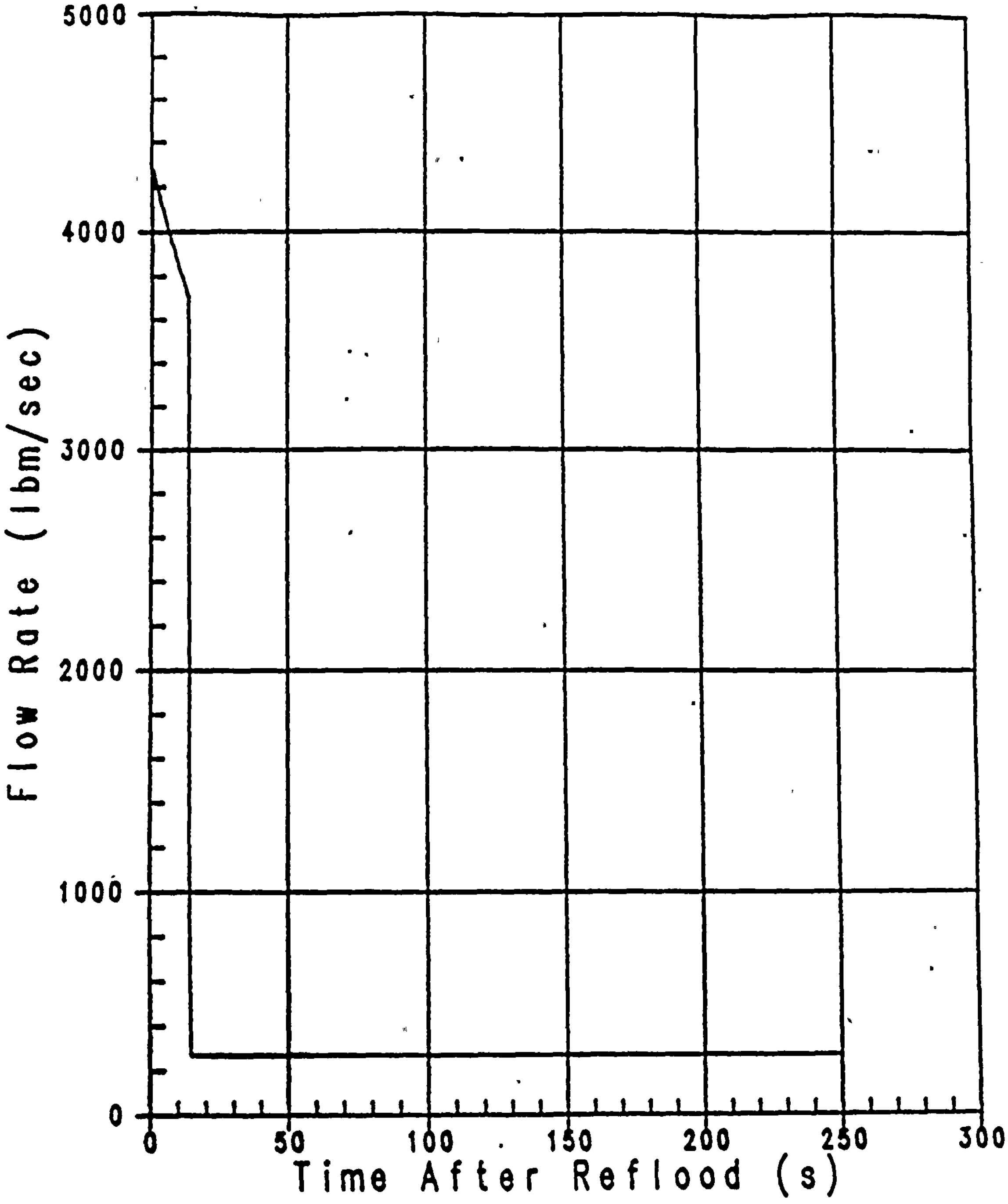


Figure 3.1-8 Accumulator and SI Flow During Reflood Uprating Analysis–RHR Crosstie Closed

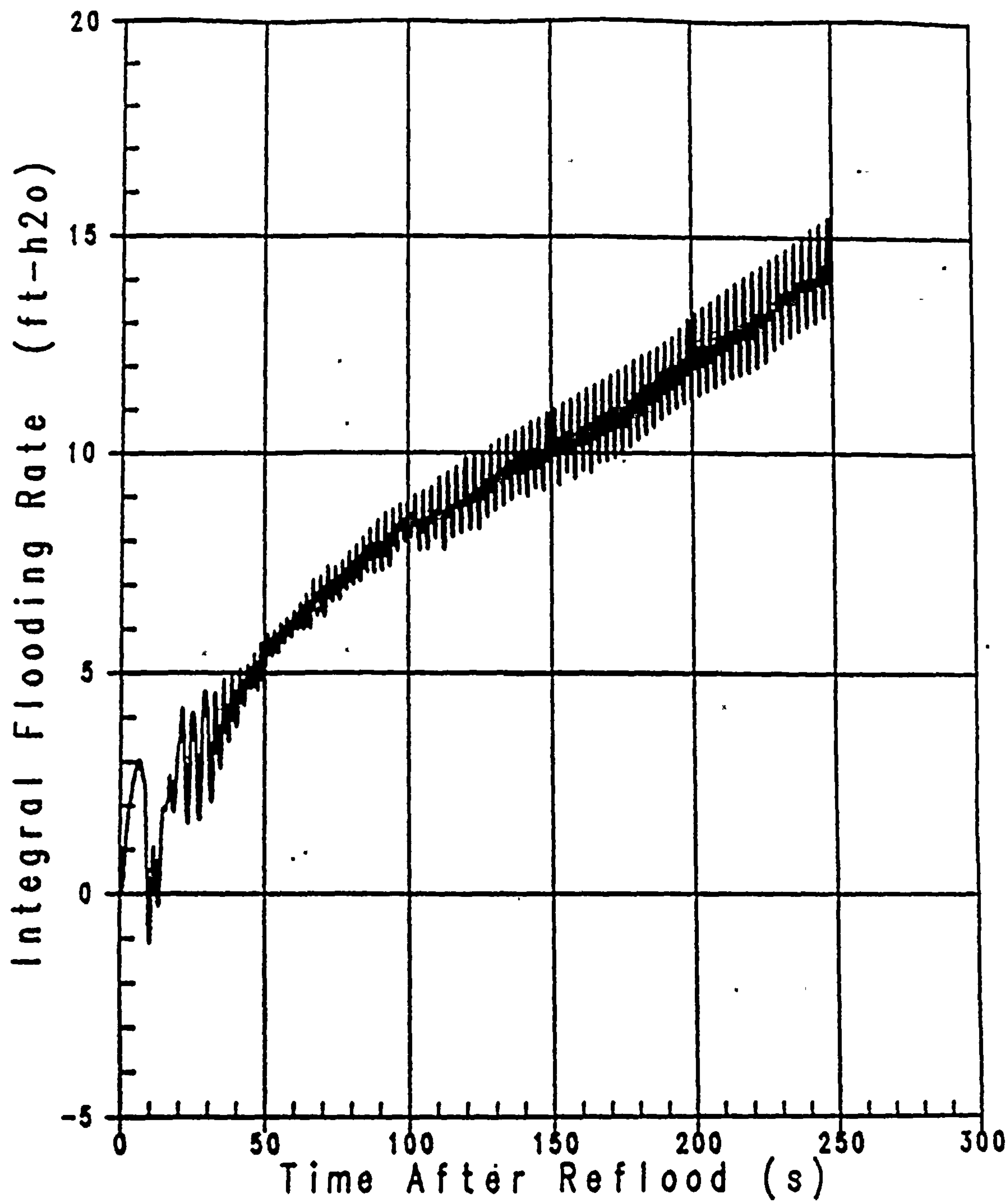


Figure 3.1-9 Integral of Core Inlet Flow
Uprating Analysis–RHR Crosstie Closed

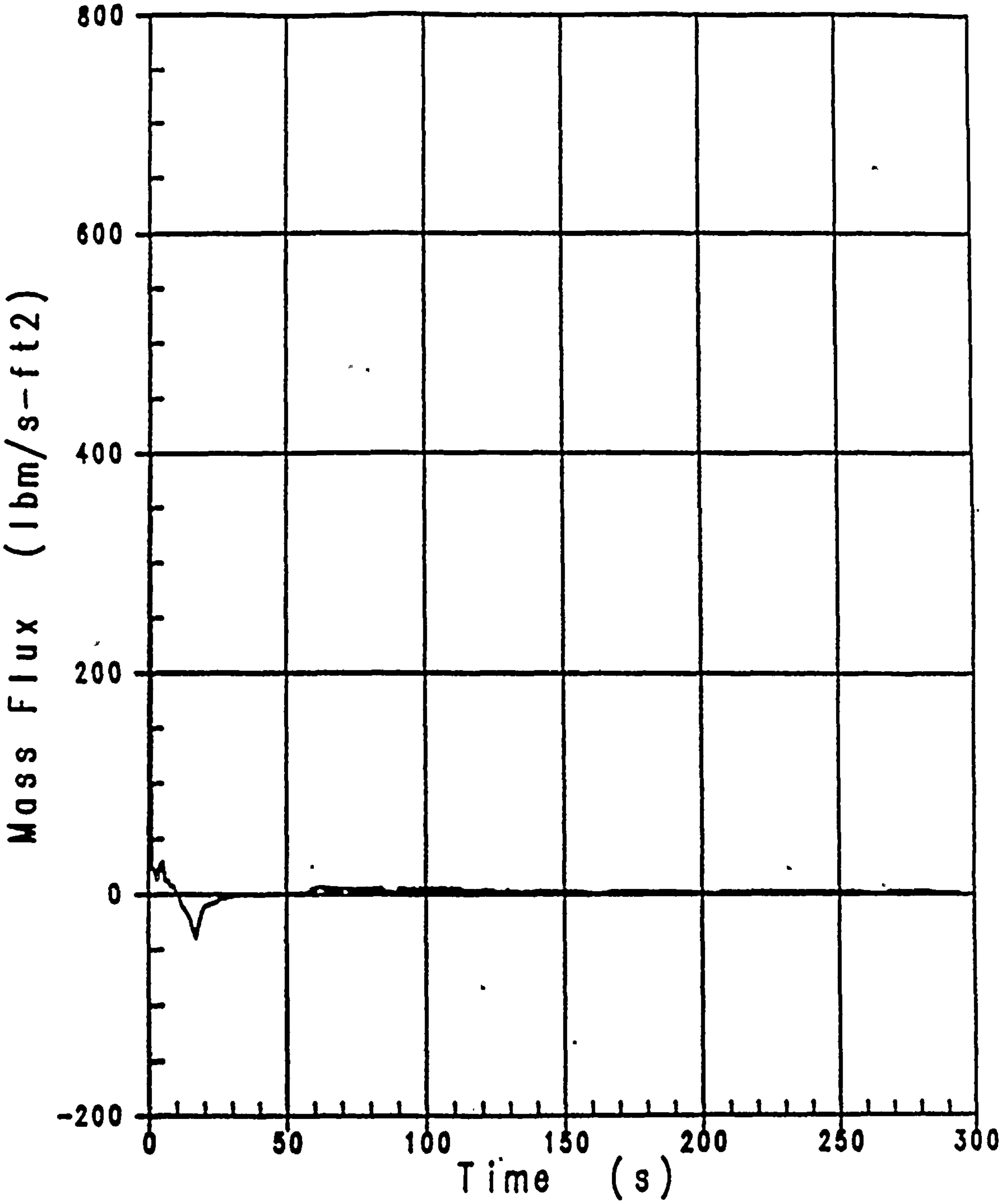


Figure 3.1-10 Mass Flux at Peak Temperature Elevation Uprating Analysis–RHR Crosstie Closed

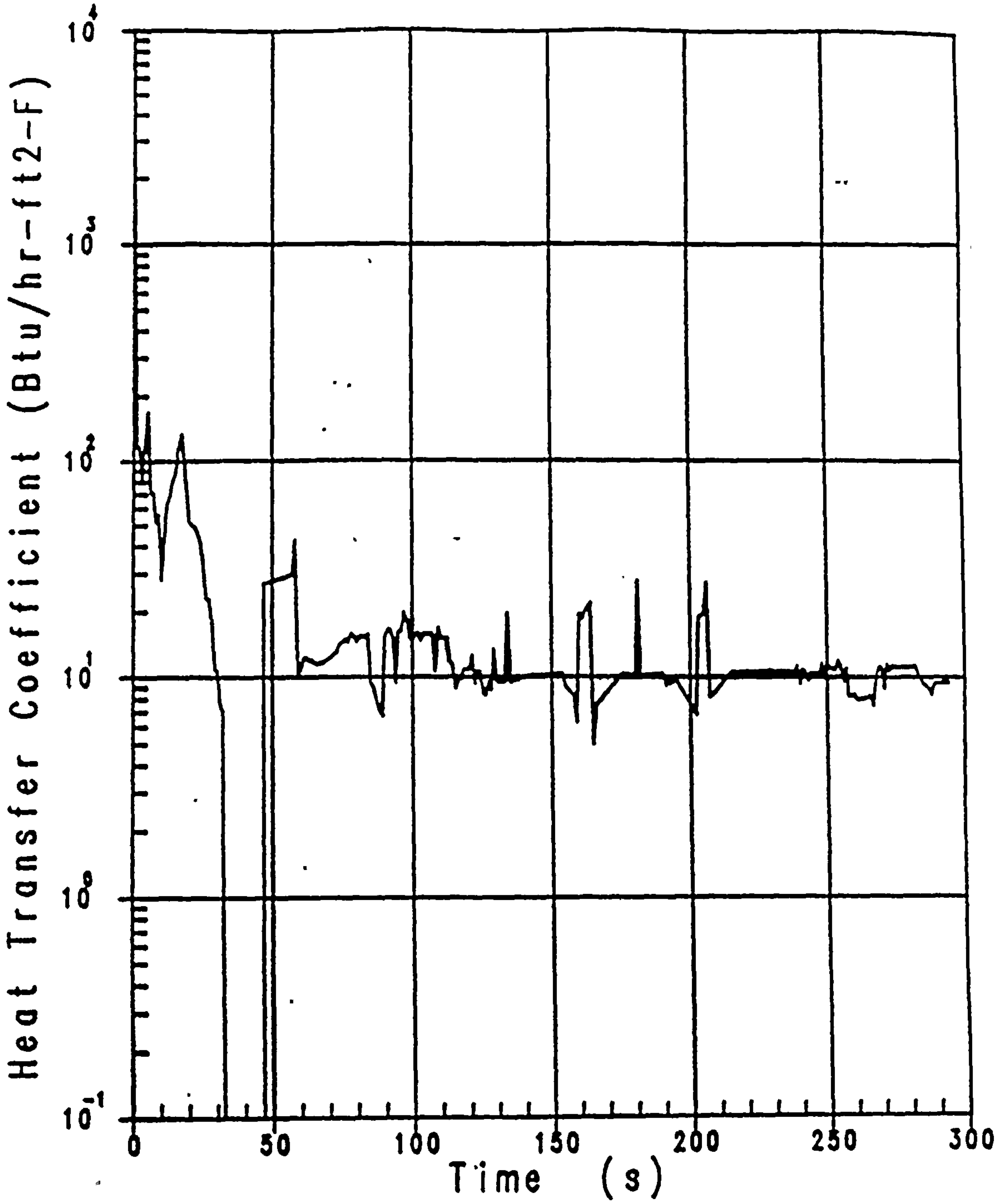


Figure 3.1-11 Rod H.T.C. at Peak Temperature Elevation
Uprating Analysis–RHR Crosstie Closed

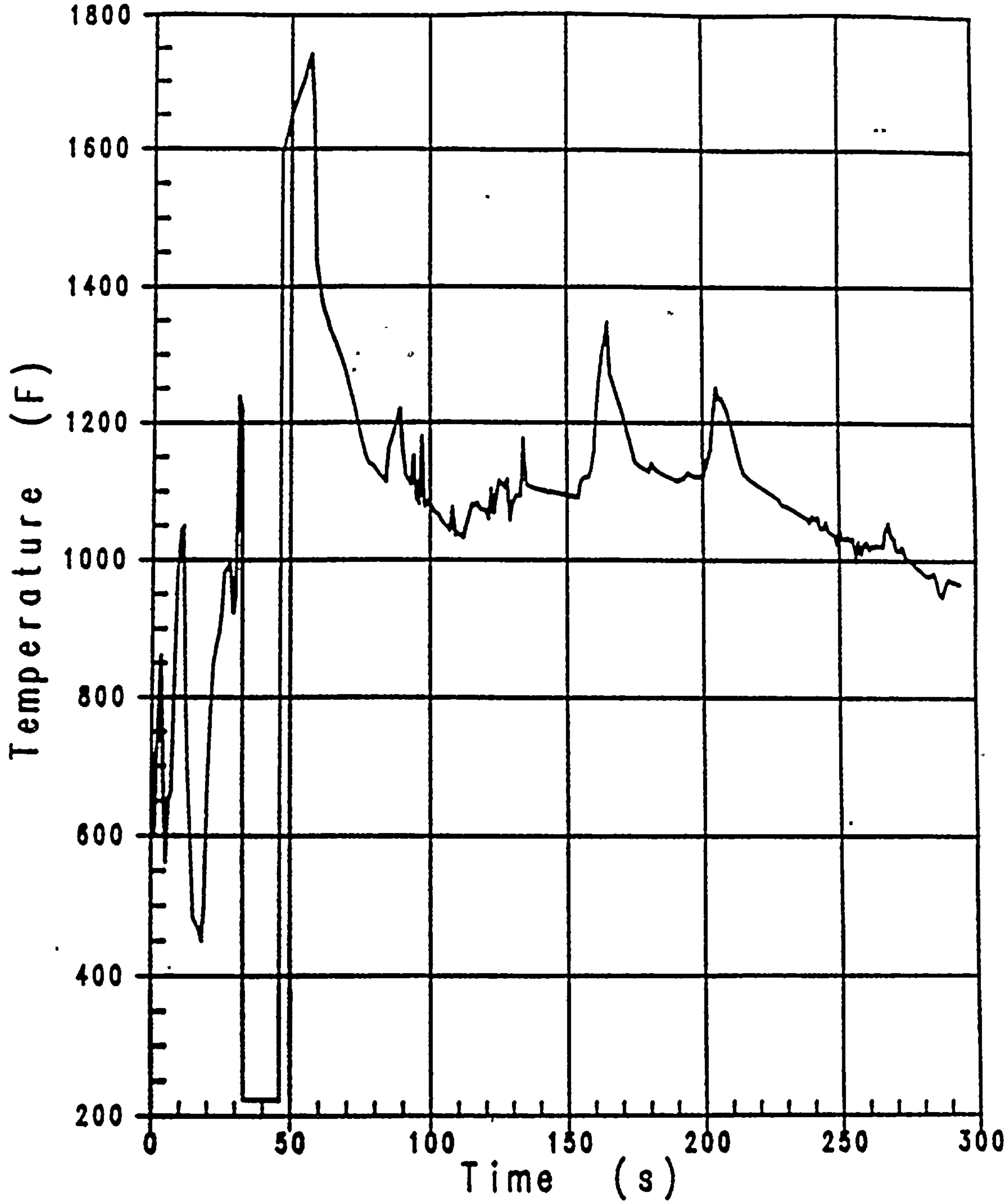


Figure 3.11-12 Vapor Temperature
Uprating Analysis–RHR Crosstie Closed

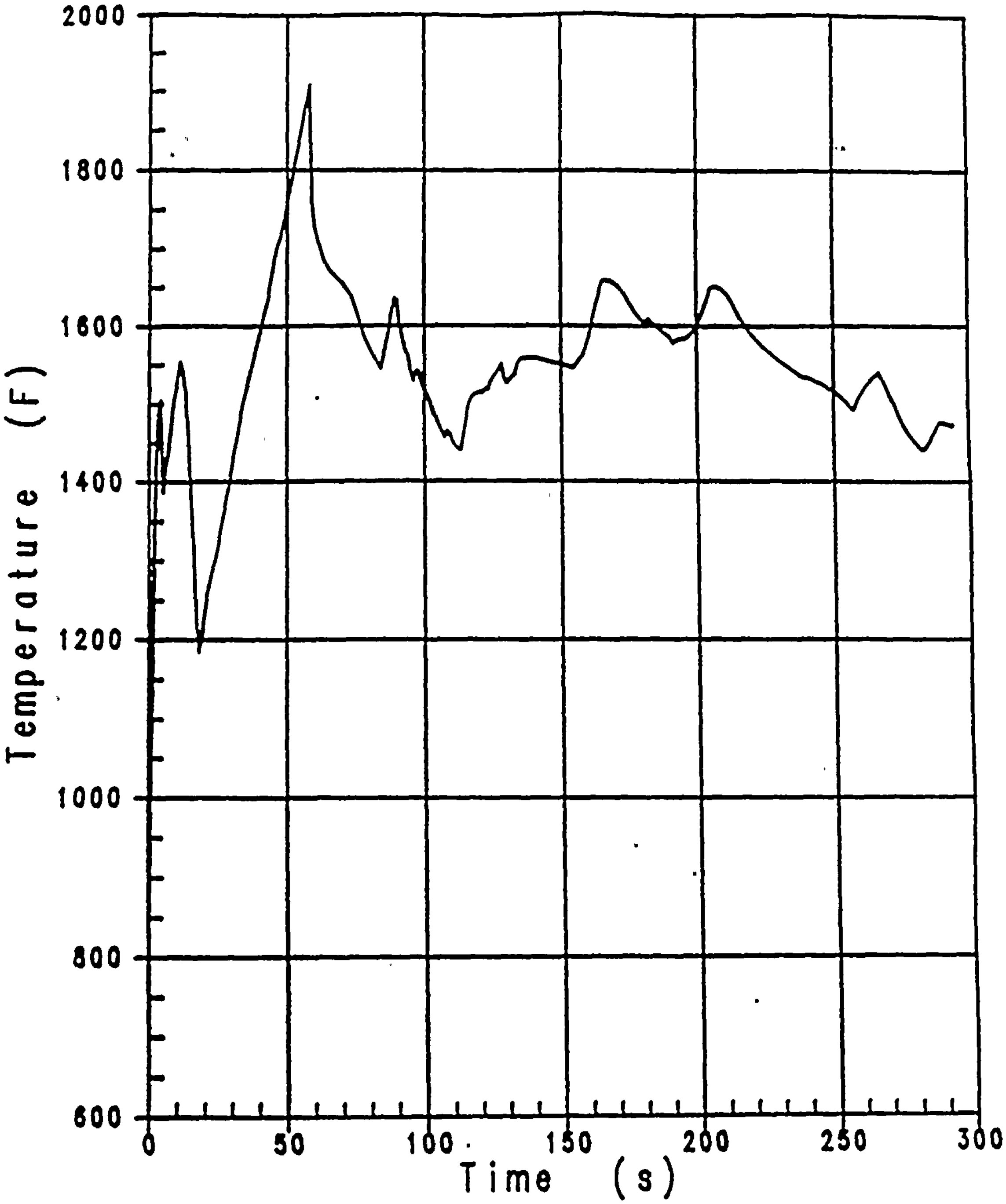


Figure 3.1-13 Fuel Rod Peak Clad Temperature Uprating Analysis–RHR Crosstie Closed

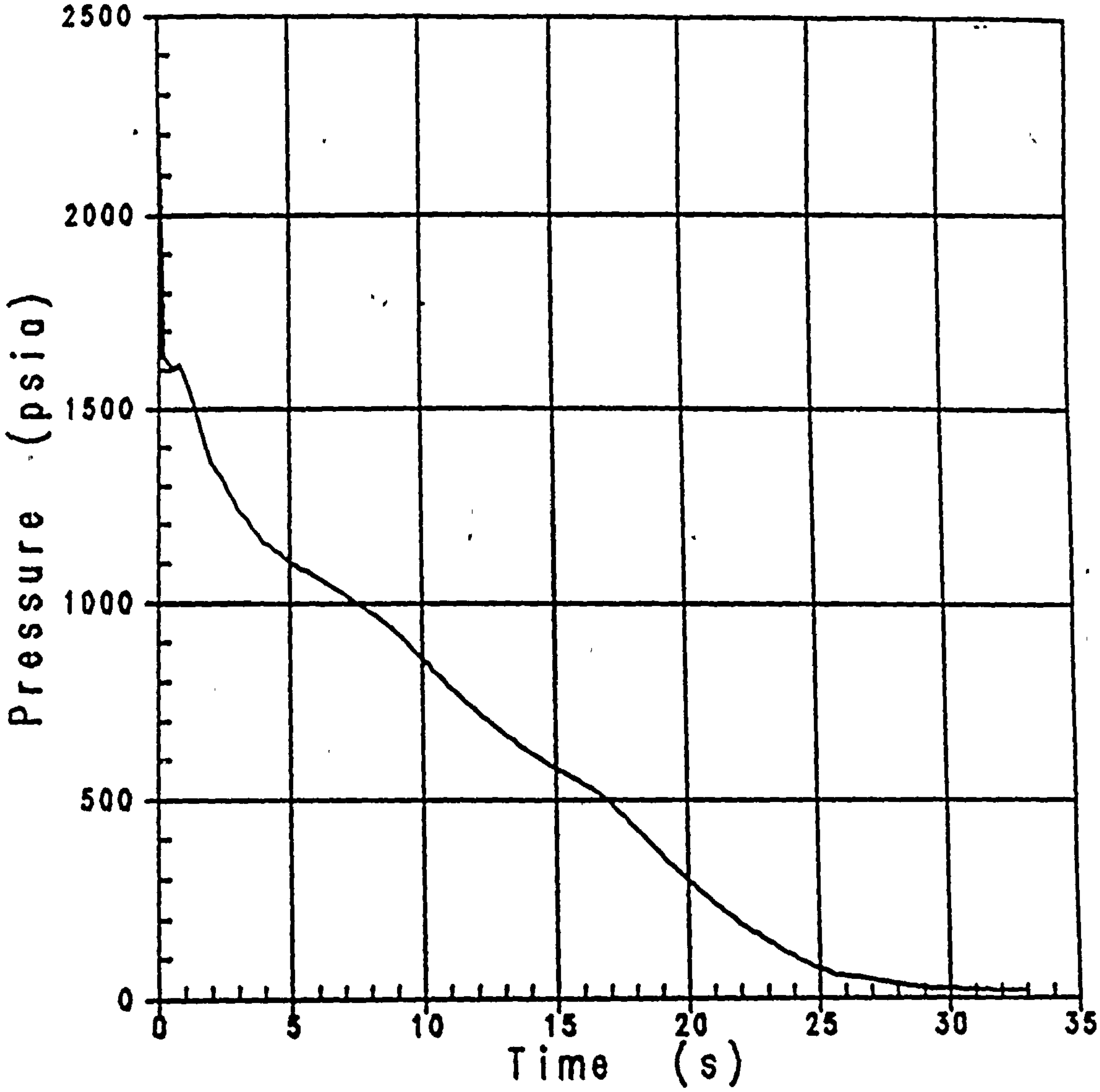


Figure 3.1-14 Reactor Coolant System Pressure
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

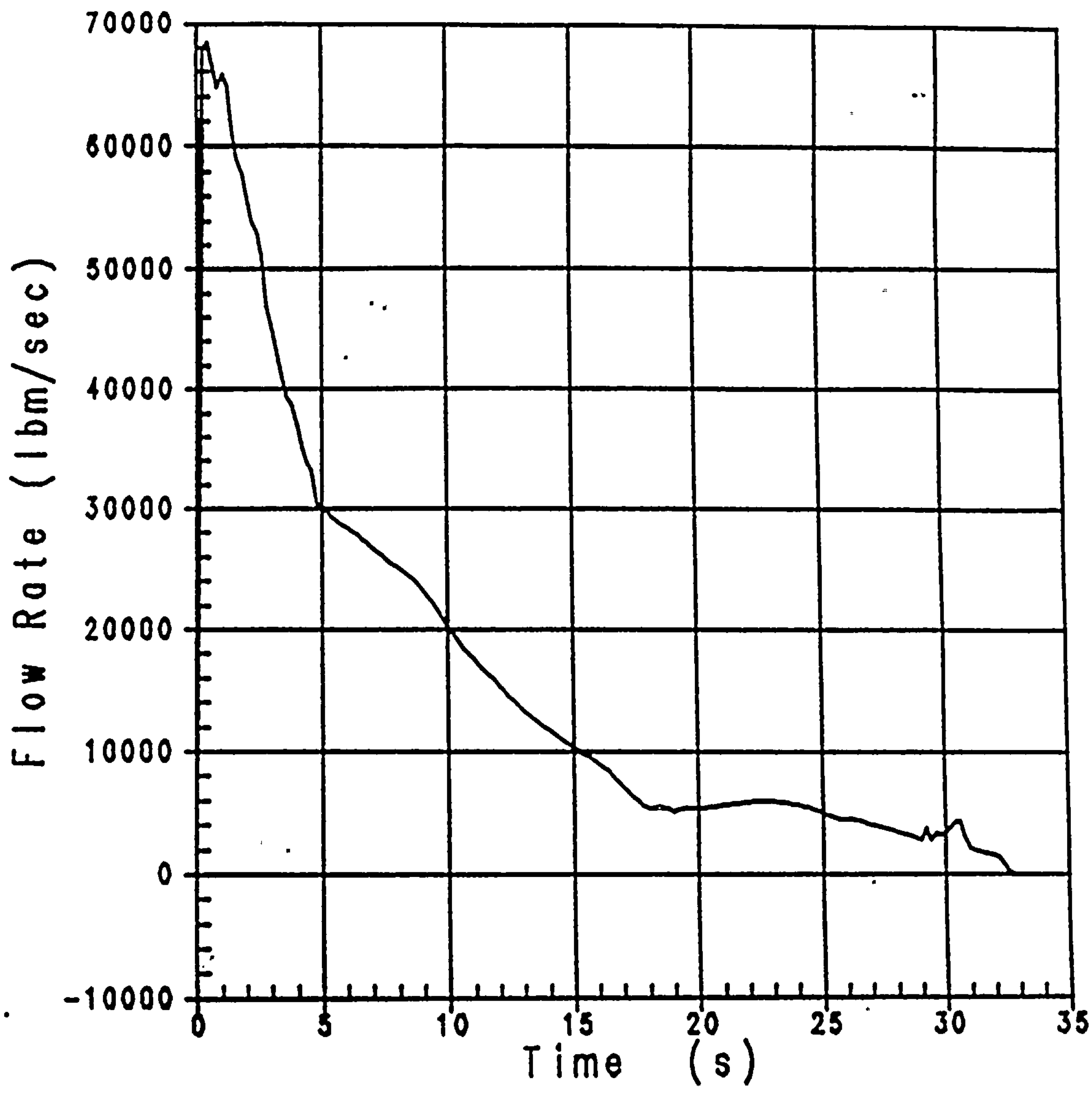


Figure 3.1-15 Break Flow During Blowdown
Uprating Analysis - RHR Crosstie Closed, $F_Q = 2.335$

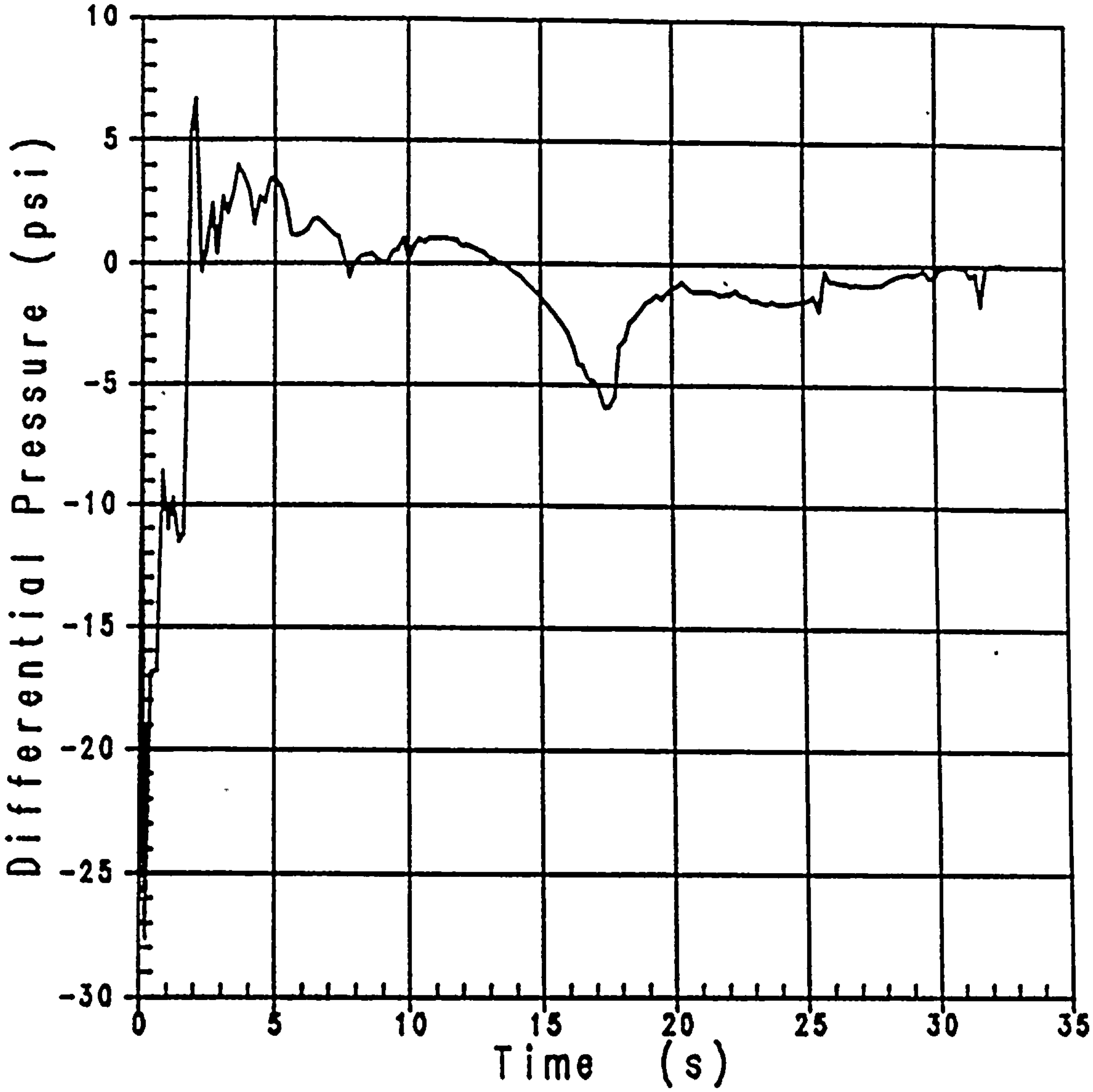


Figure 3.1-16 Core Pressure Drop
Uprating Analysis - RHR Crosstie Closed, $F_Q = 2.335$

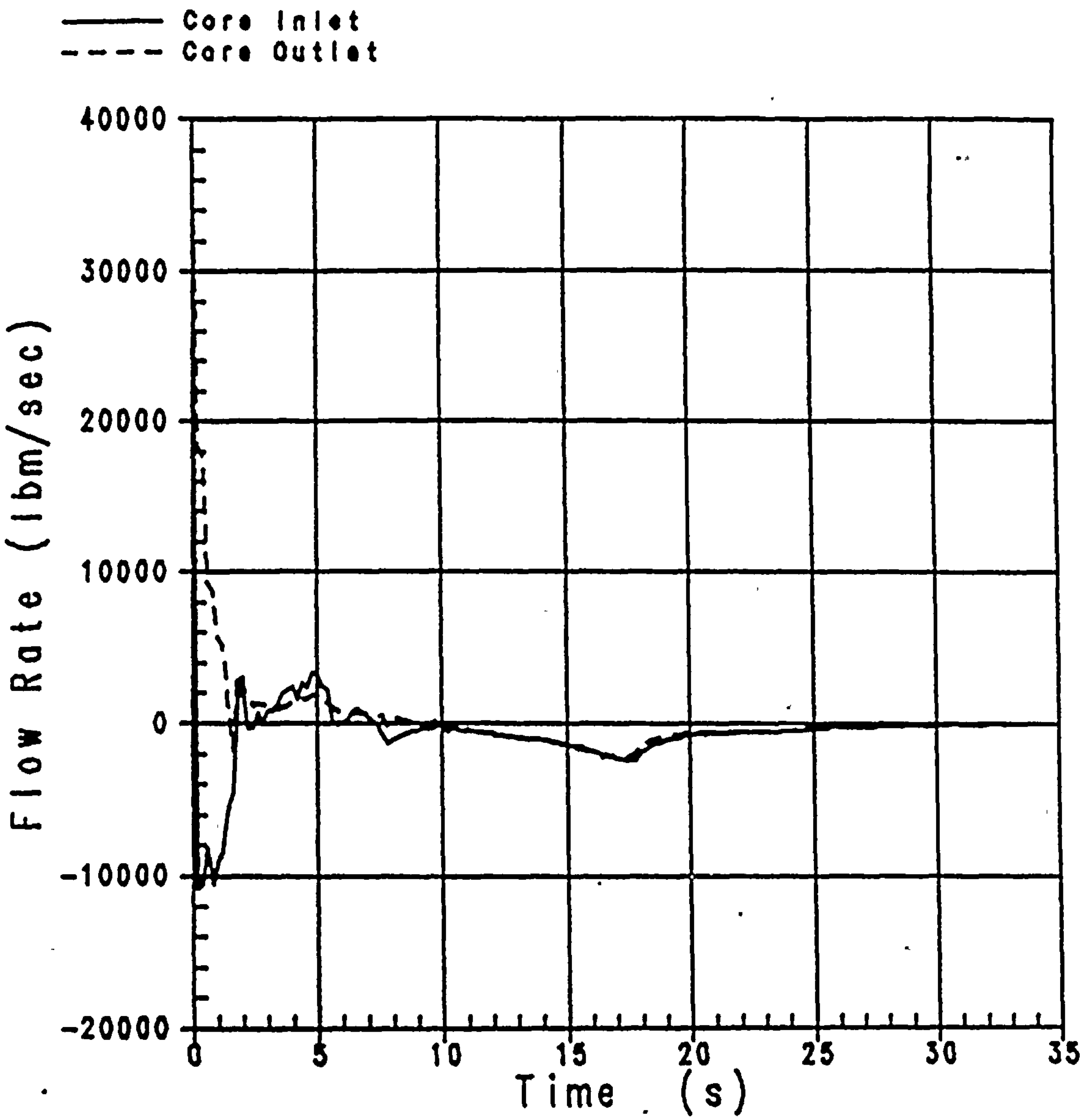


Figure 3.1-17 Core Flowrate
Uprating Analysis - RHR Crosstie Closed, $F_Q = 2.335$

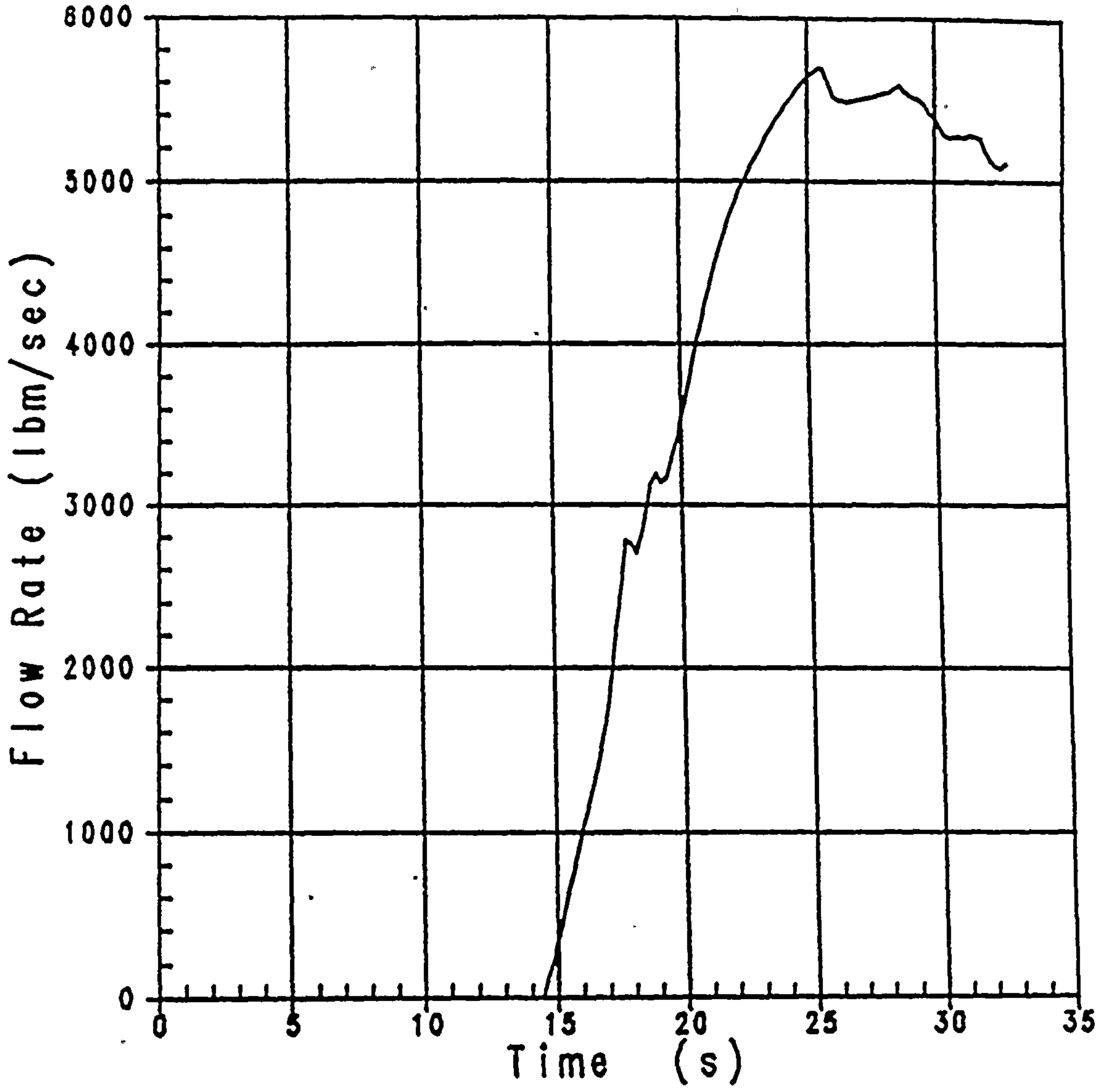


Figure 3.1-18 Accumulator Flow During Blowdown
Uprating Analysis - RHR Crosstie Closed, $F_Q = 2.335$

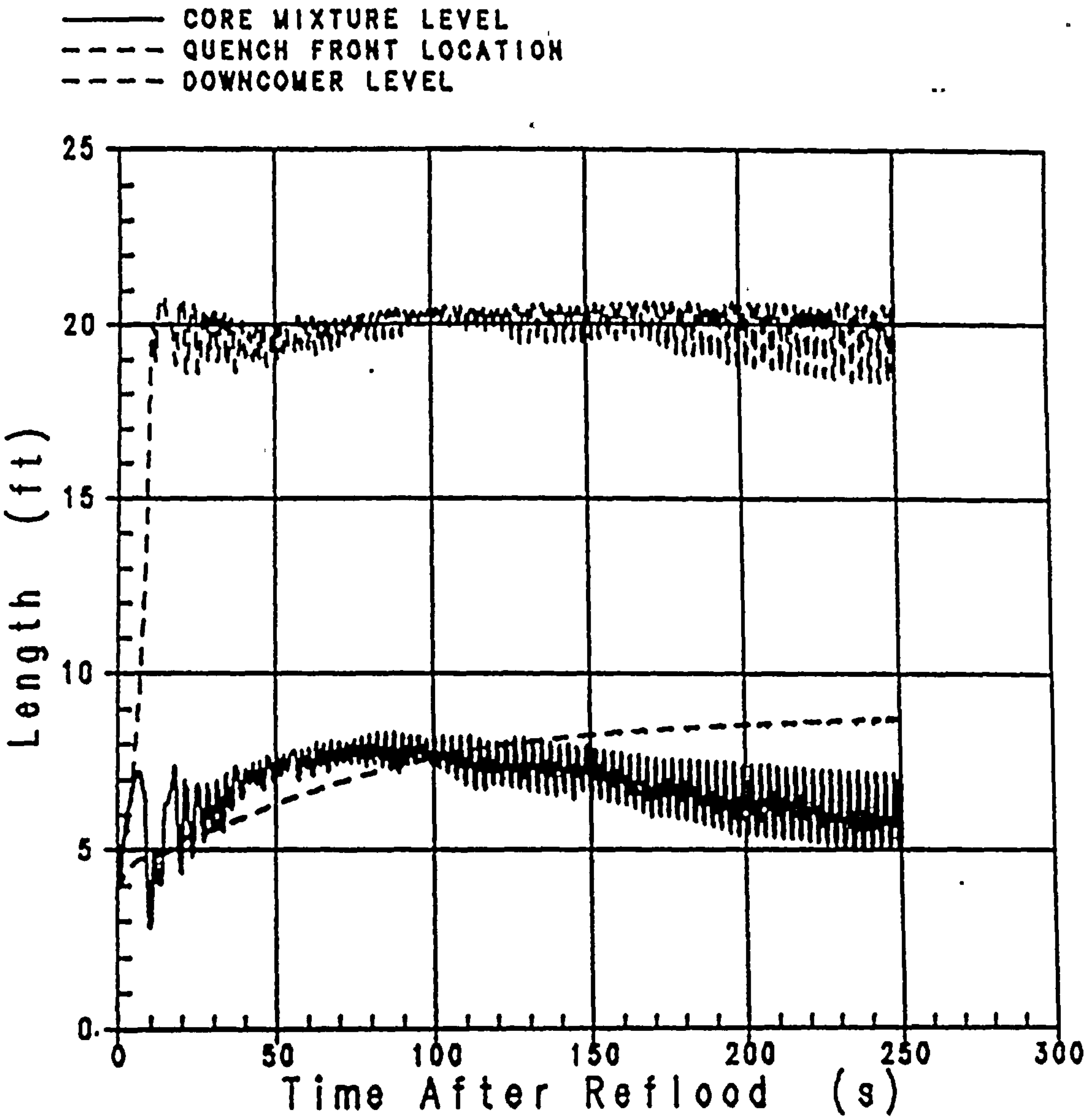


Figure 3.1-19 Vessel Liquid Levels During Reflood
Uprating Analysis - RHR Crosstie Closed, $F_Q = 2.335$

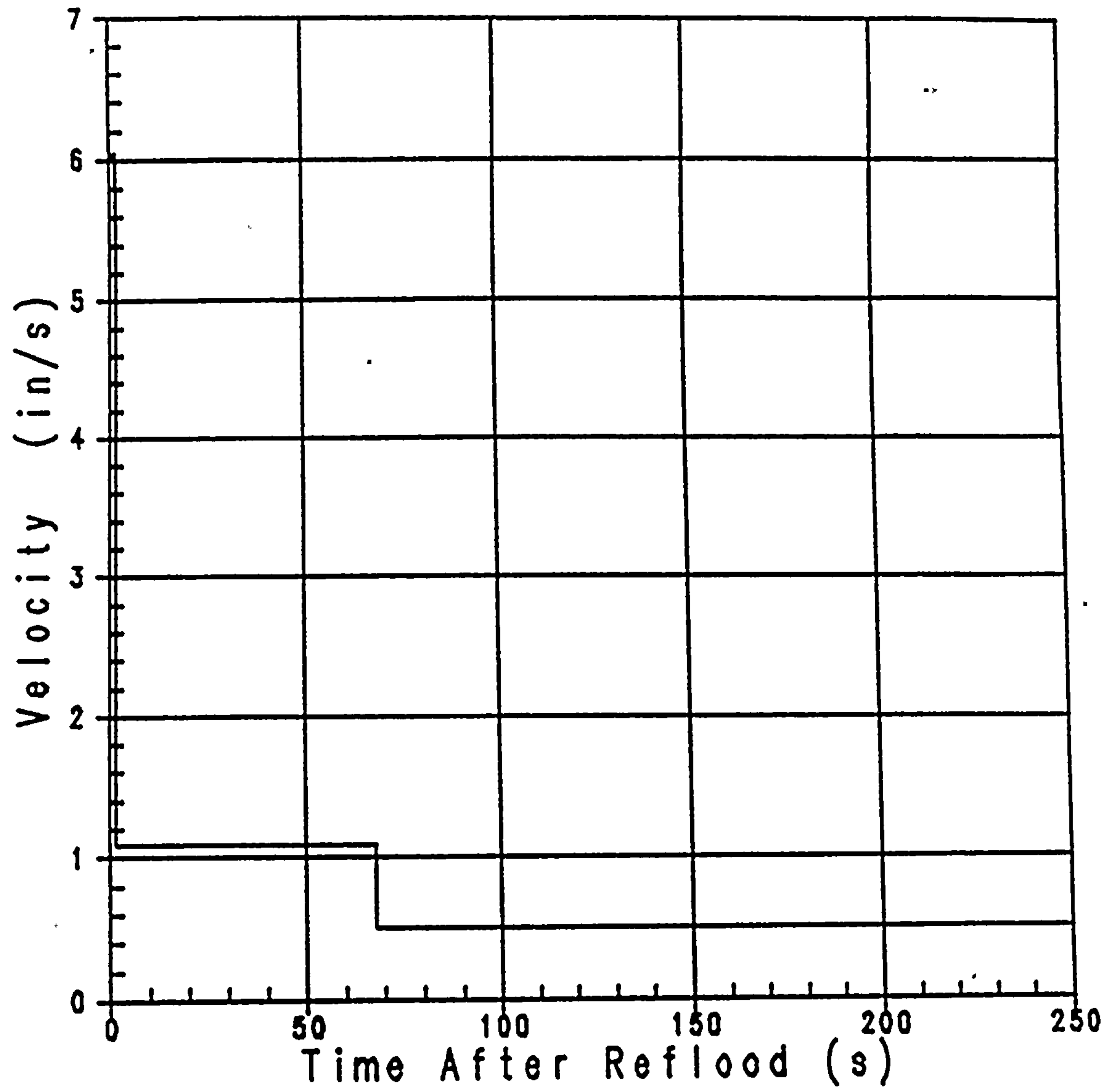


Figure 3.1-20 Core Inlet Flow During Reflood
Uprating Analysis - RHR Crosstie Closed, $F_Q = 2.335$

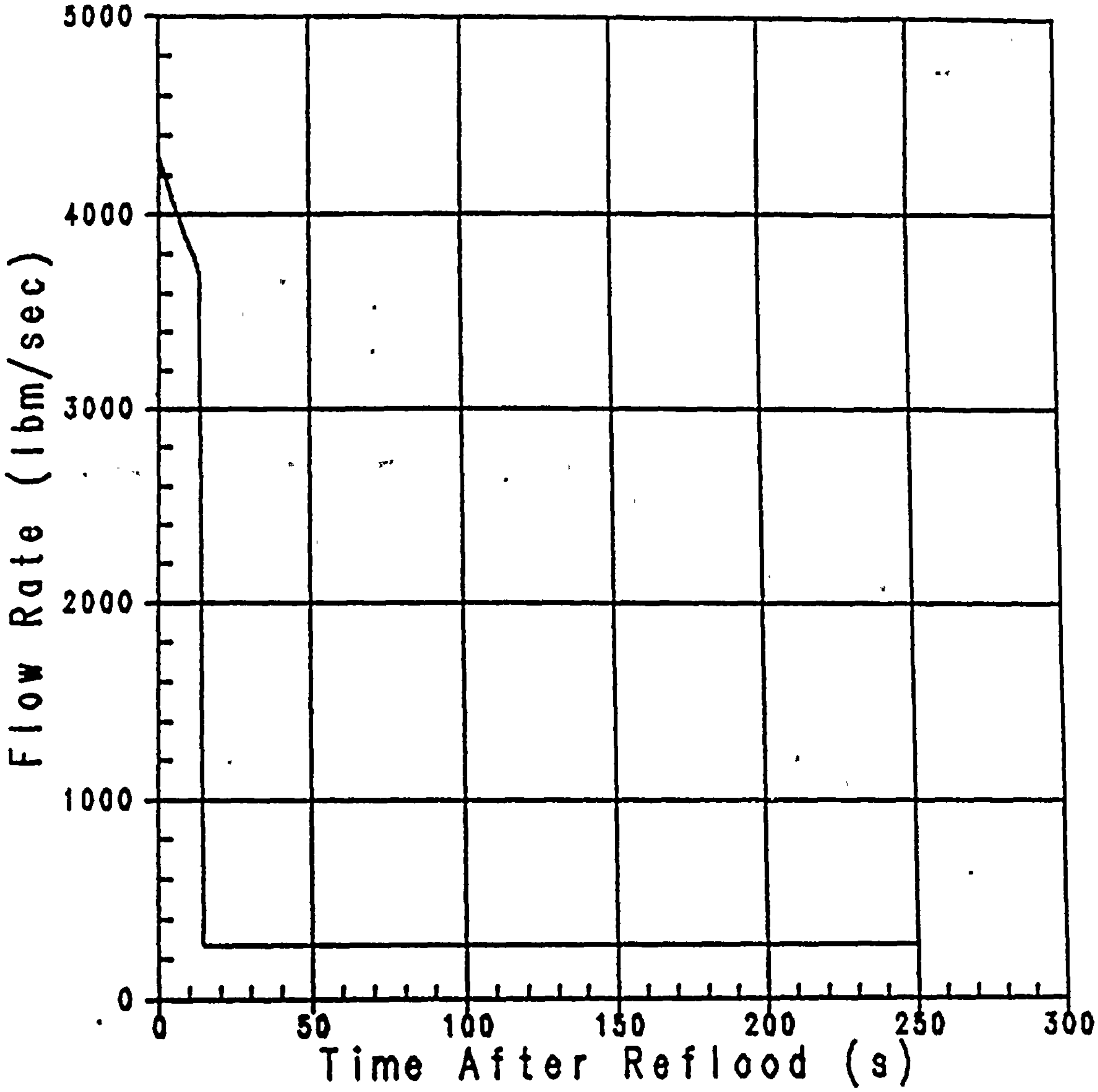


Figure 3.1-21 Accumulator and SI Flow During Reflood
Uprating Analysis - RHR Crosstie Closed, $F_Q = 2.335$

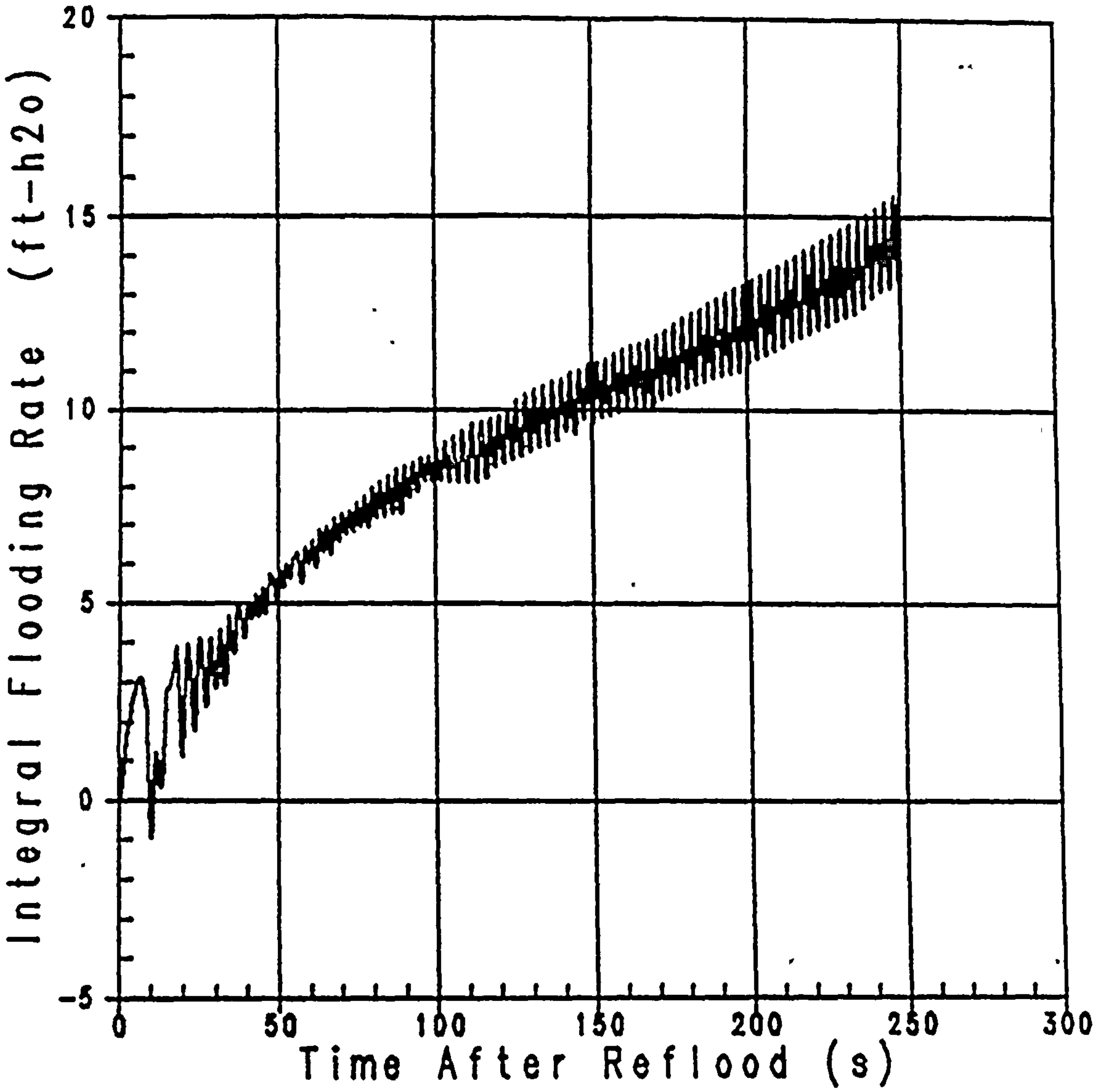


Figure 3.1-22 Integral of Core Inlet Flow
Uprating Analysis - RHR Crosstie Closed, $F_Q = 2.335$

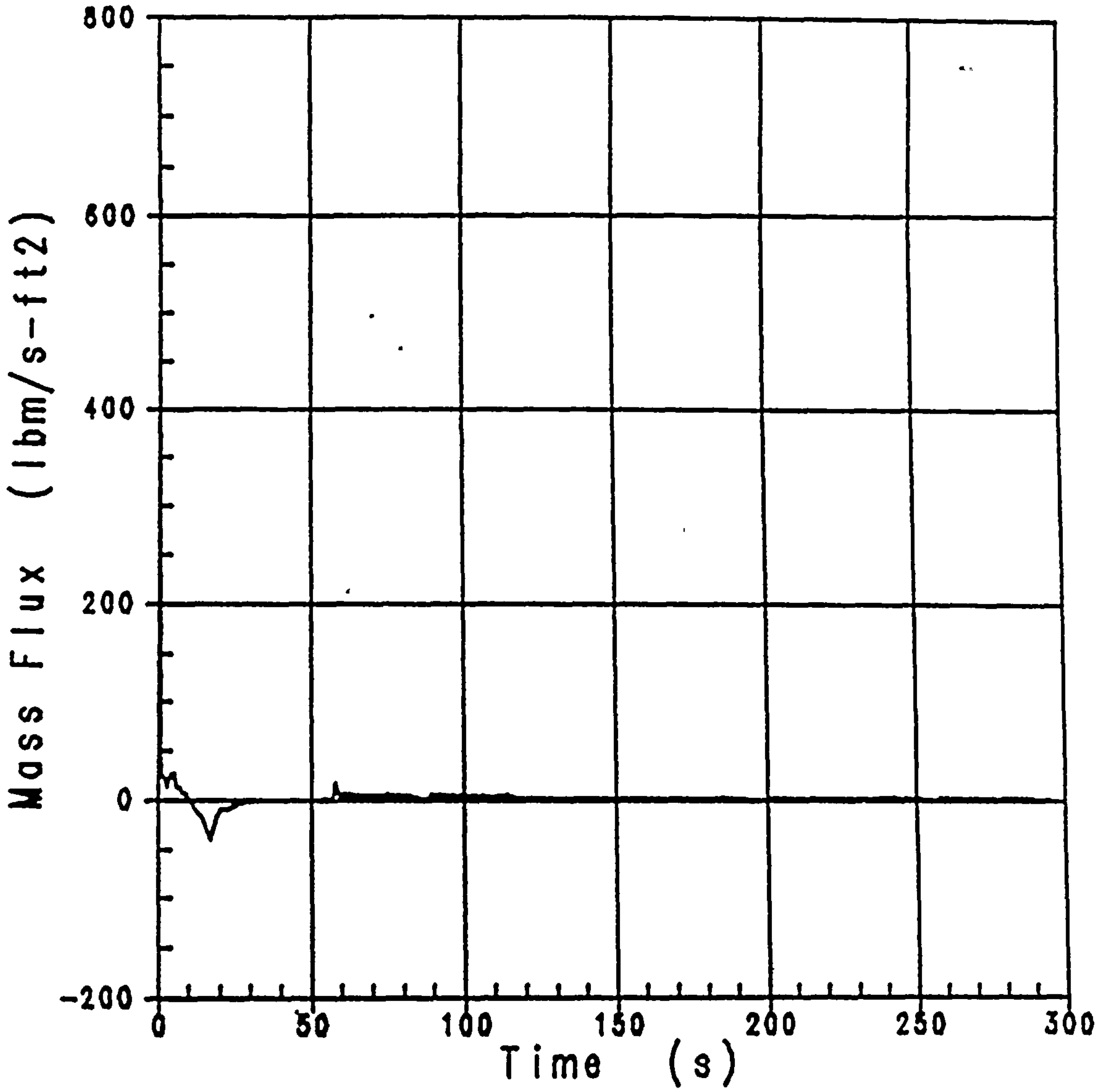


Figure 3.1-23 Mass Flux at Peak Temperature Elevation
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

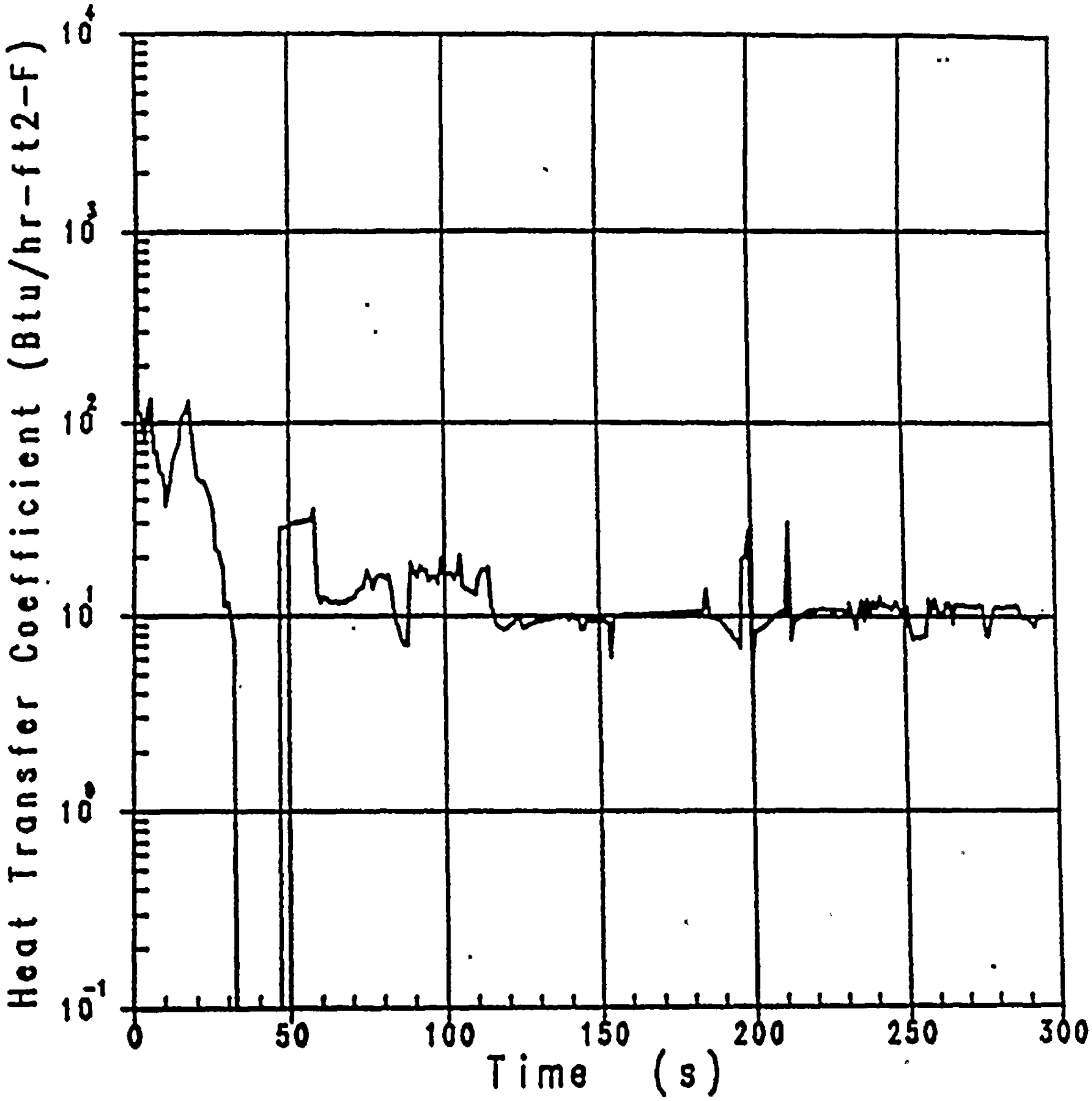


Figure 3.1-24 Rod H.T.C. at Peak Temperature Elevation
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

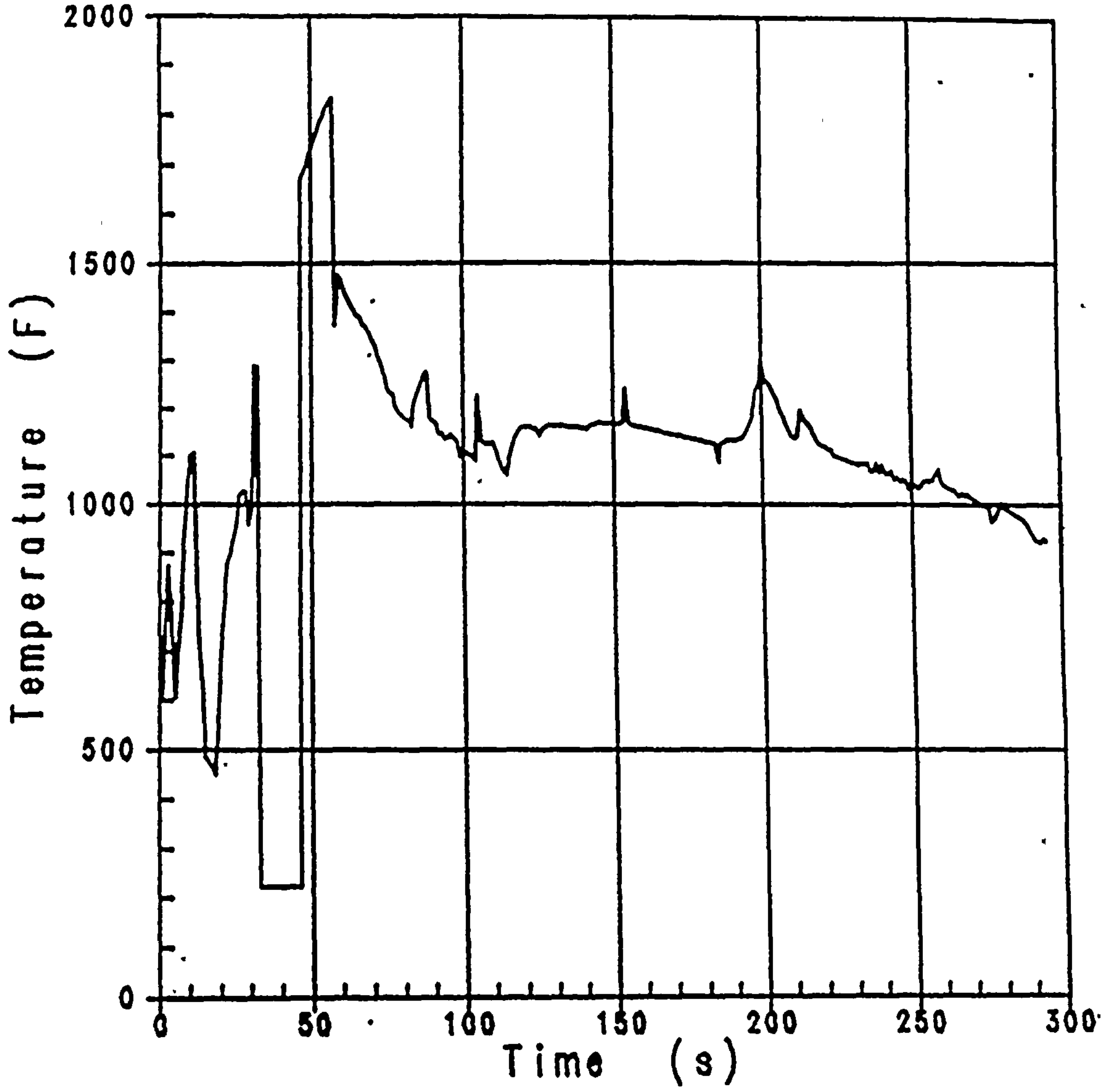


Figure 3.1-25 Vapor Temperature
Uprating Analysis - RHR Crosstie Closed, F_Q = 2.335

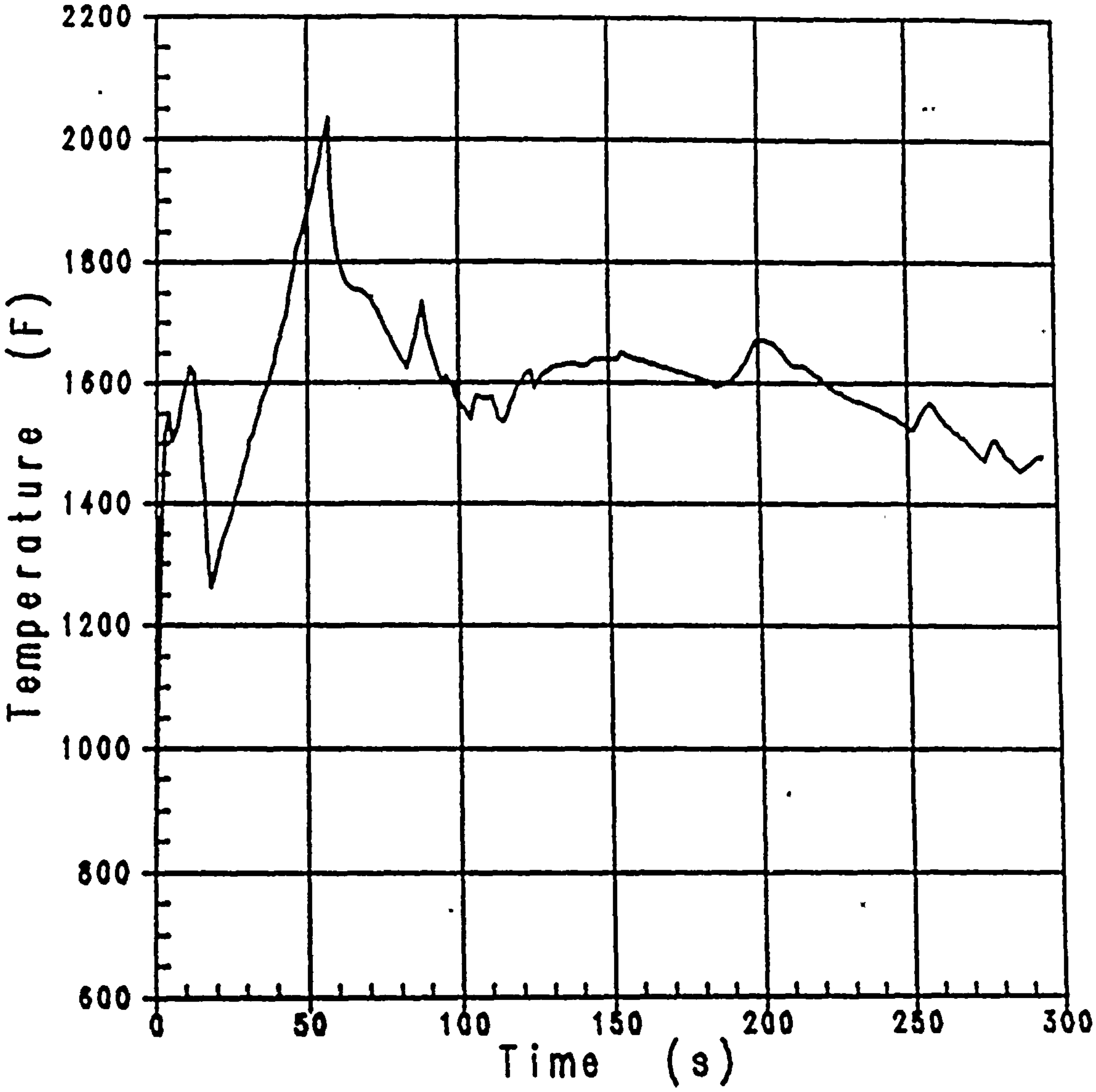


Figure 3.1-26 Fuel Rod Peak Clad Temperature
Uprating Analysis - RHR Crosstie Closed, $F_Q = 2.335$

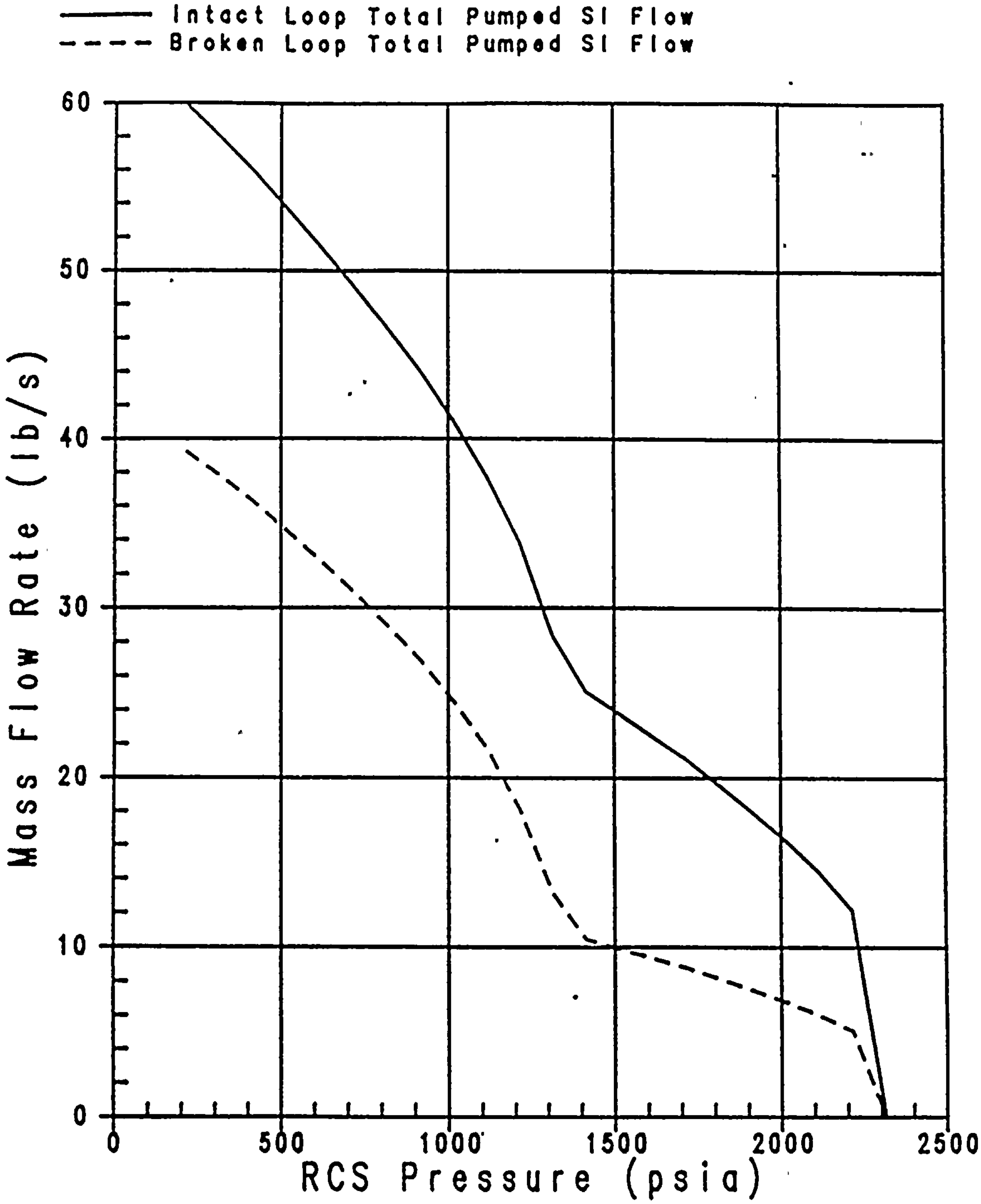


Figure 3.1-27. Safety Injection Flow Rates
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

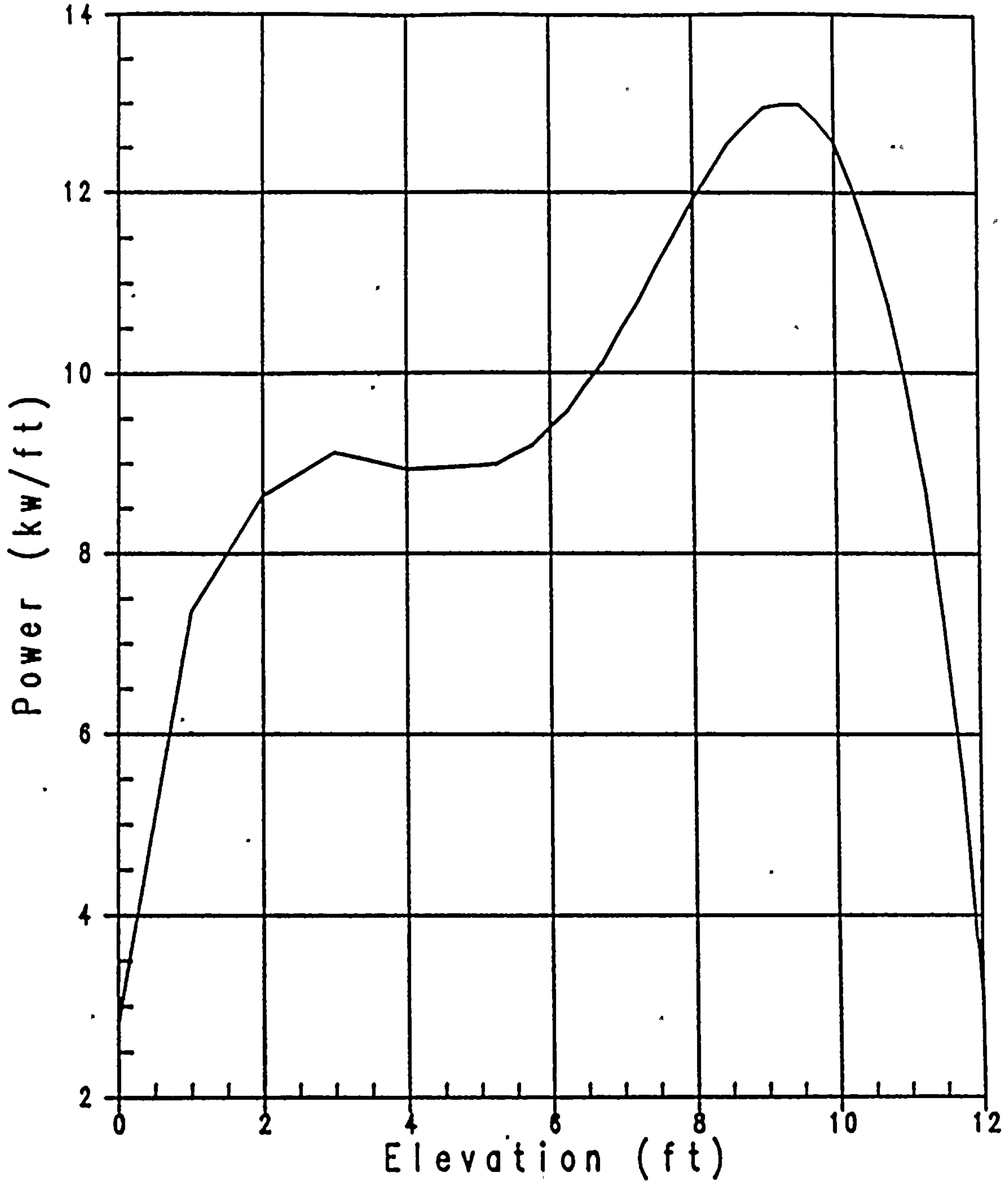


Figure 3,1-28 Hot Rod Axial Power Distribution
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

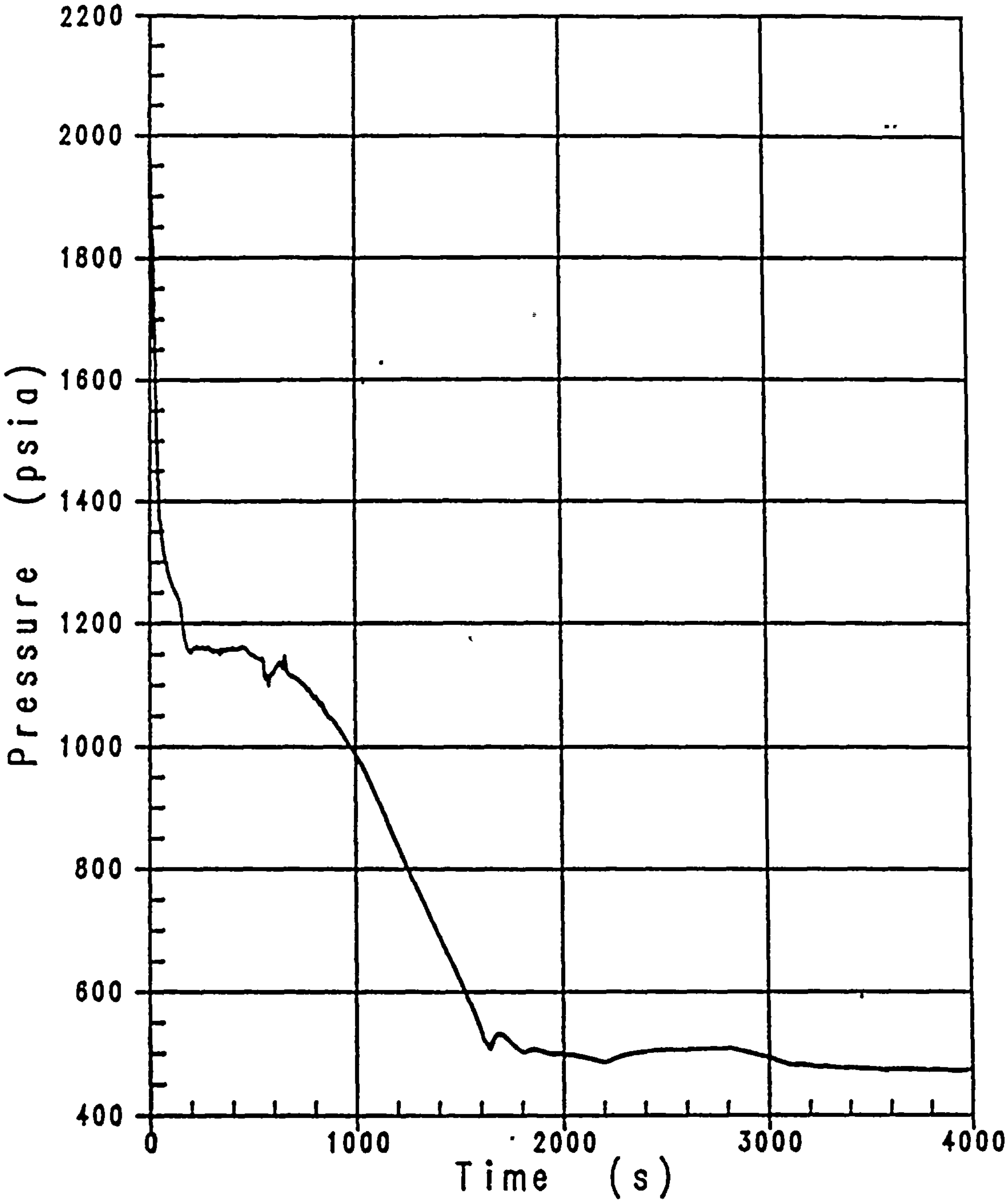


Figure 3.1-29 RCS Pressure (3 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

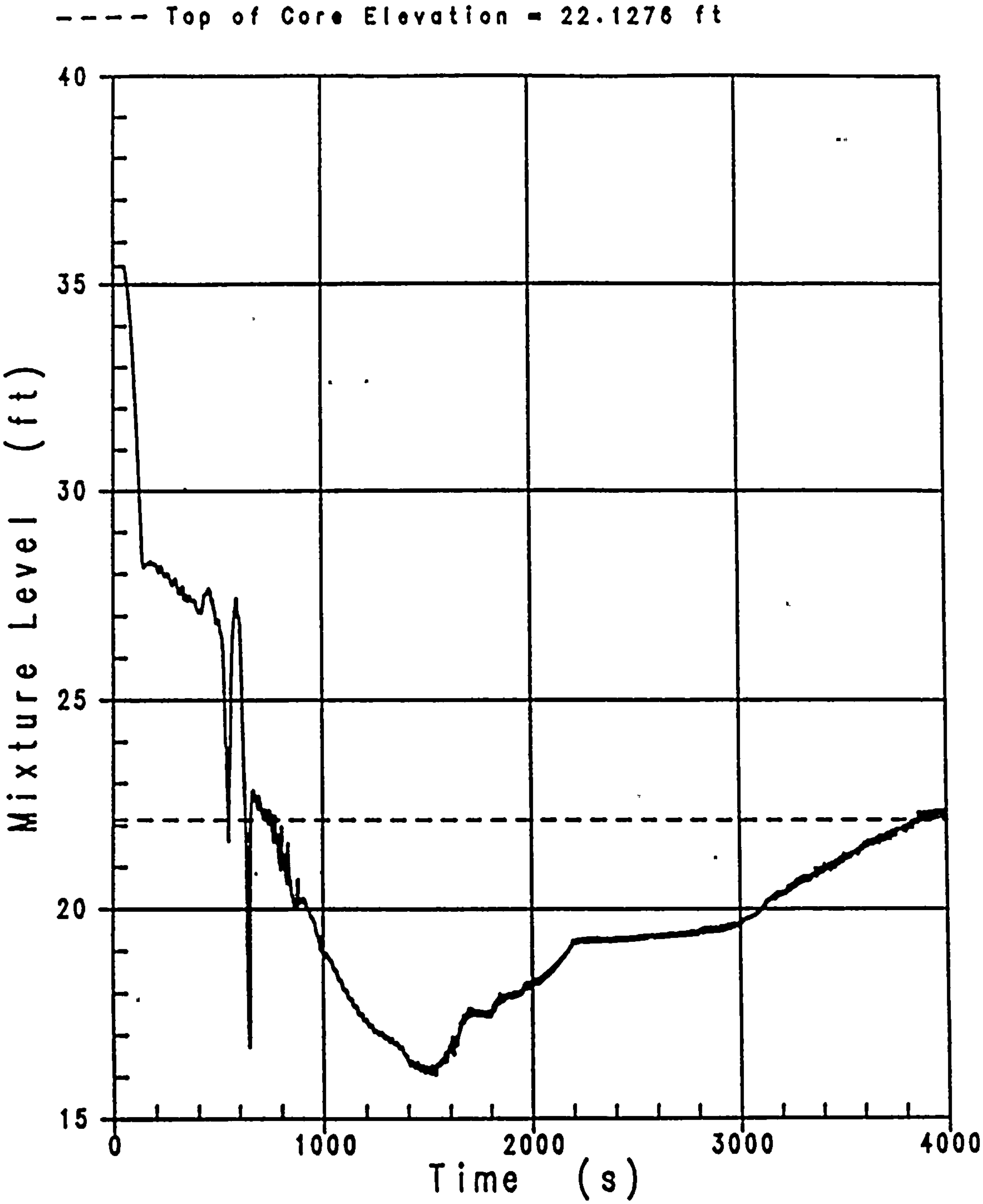


Figure 3.1-30 Core Mixture Level (3 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

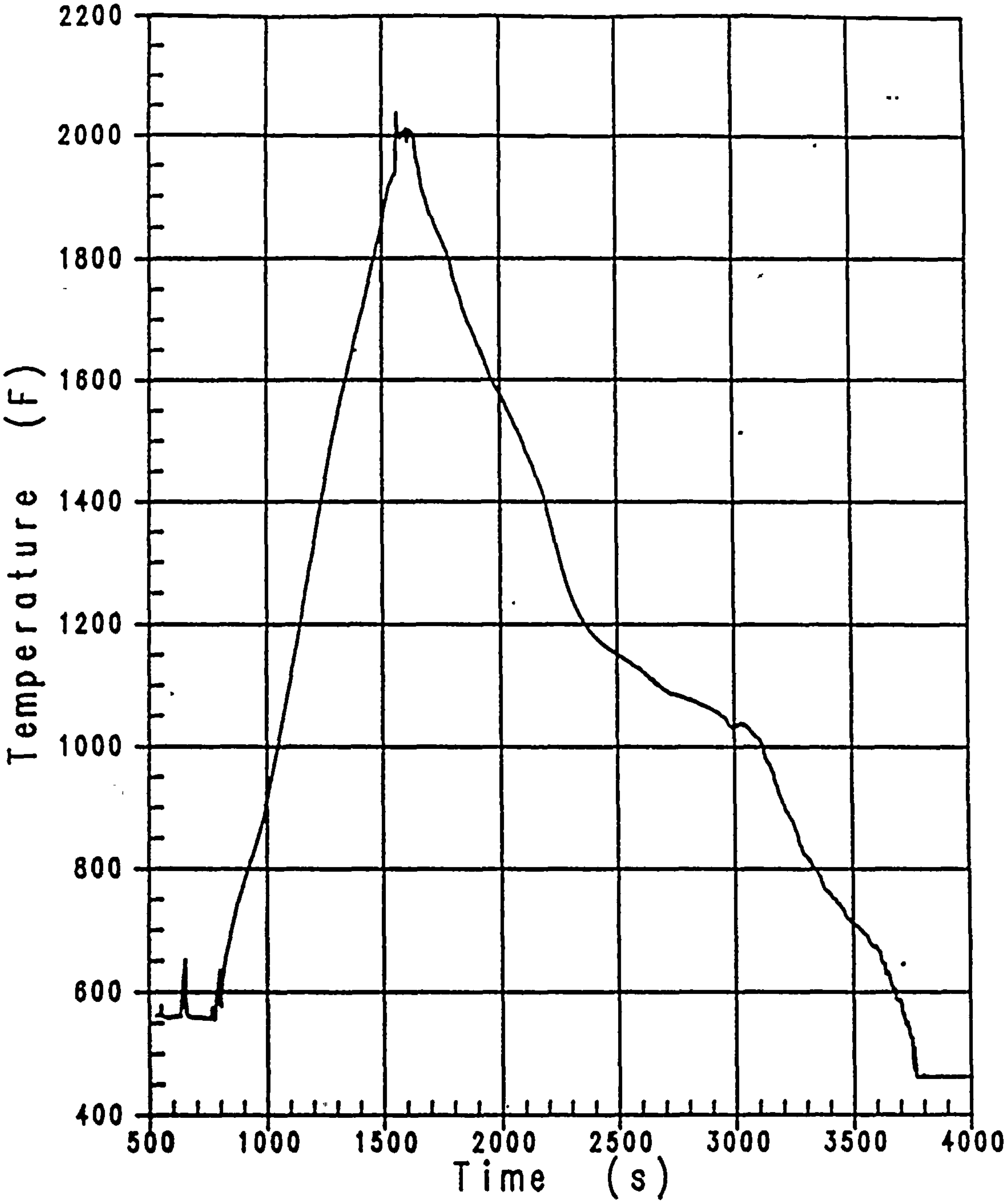


Figure 3.1-31 Peak Clad Temperature (3 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

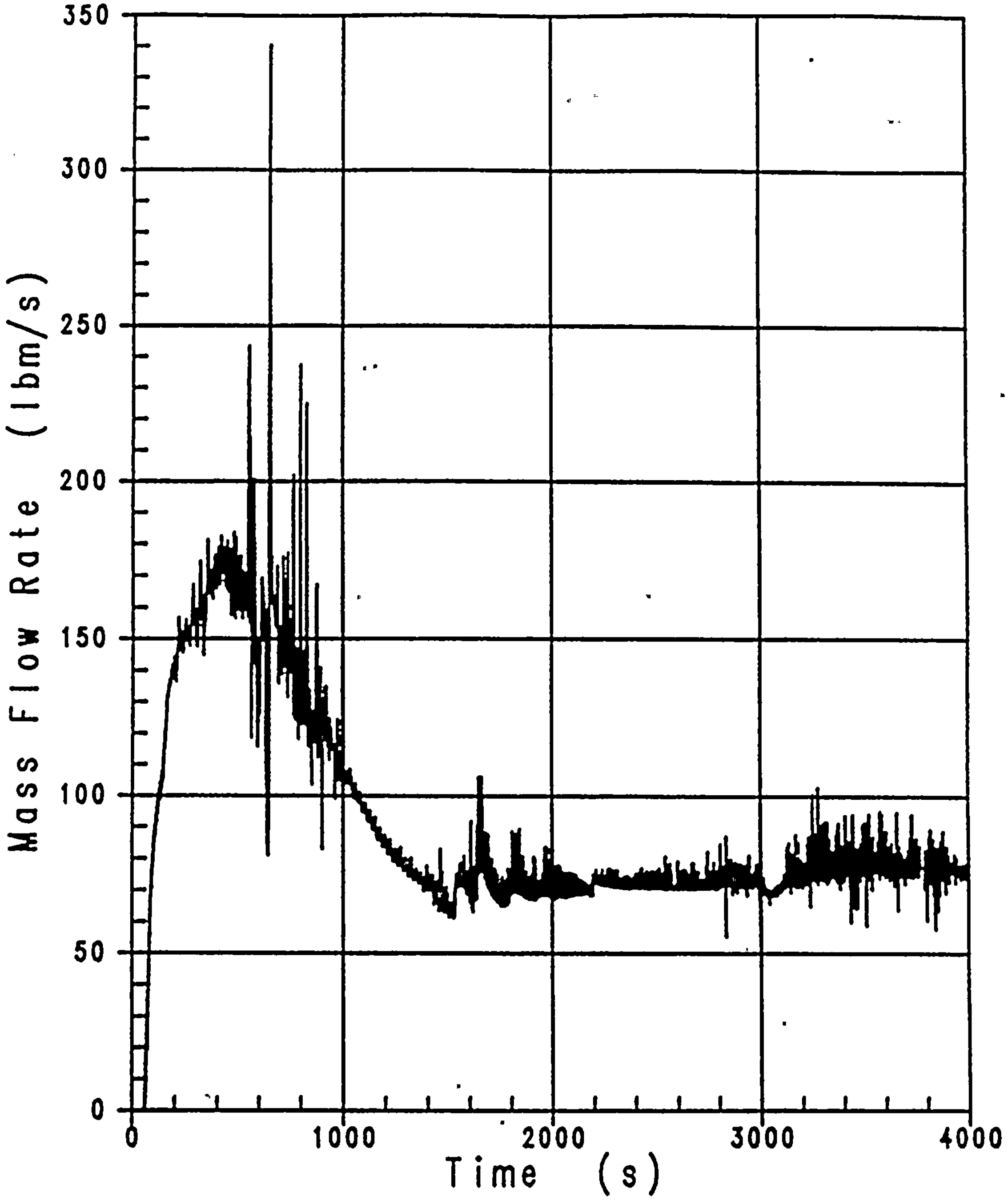


Figure 3.1-32 Core Outlet Steam Mass Flow Rate (3 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

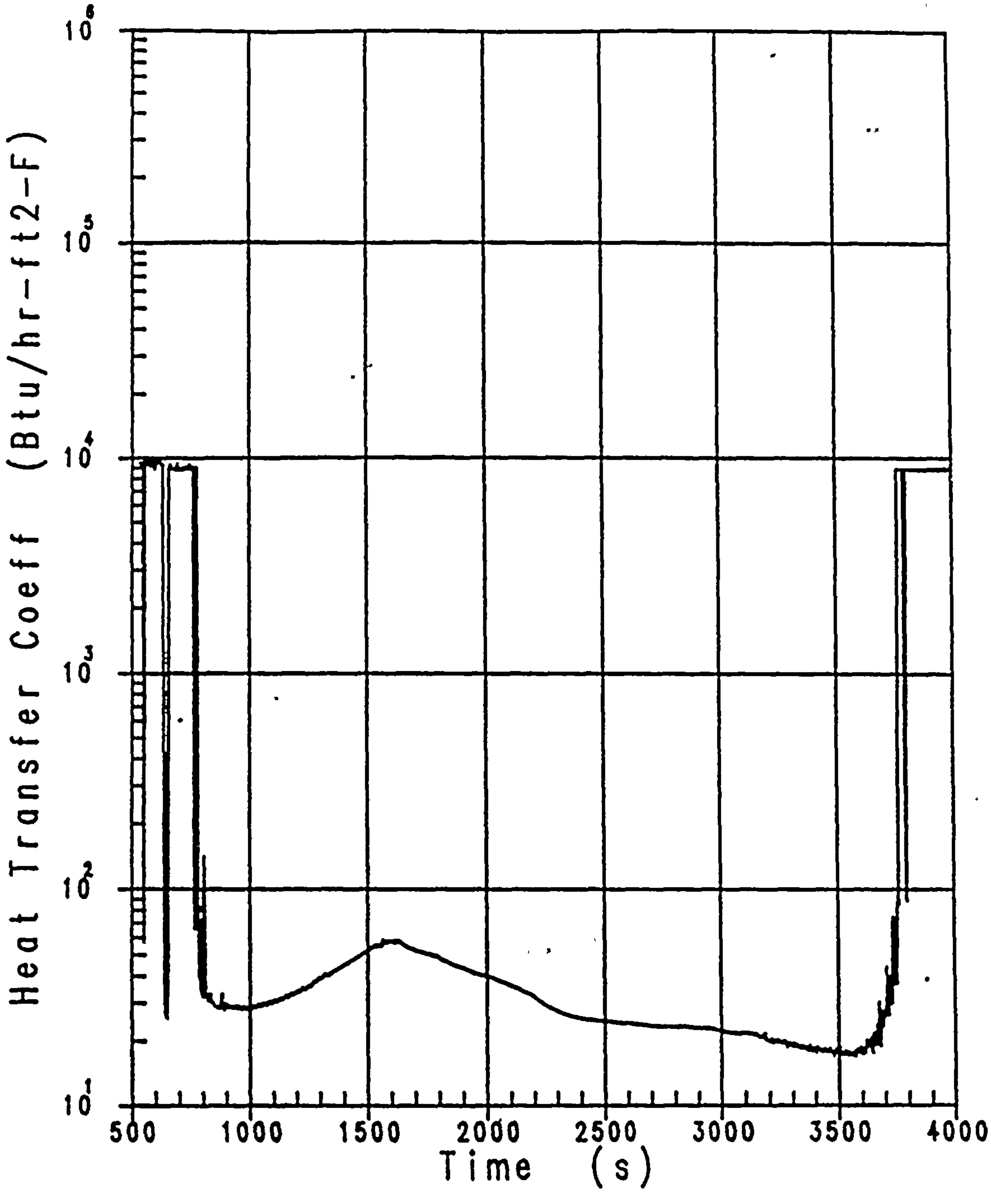


Figure 3.1-33 Hot Spot Heat Transfer Coefficient (3 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

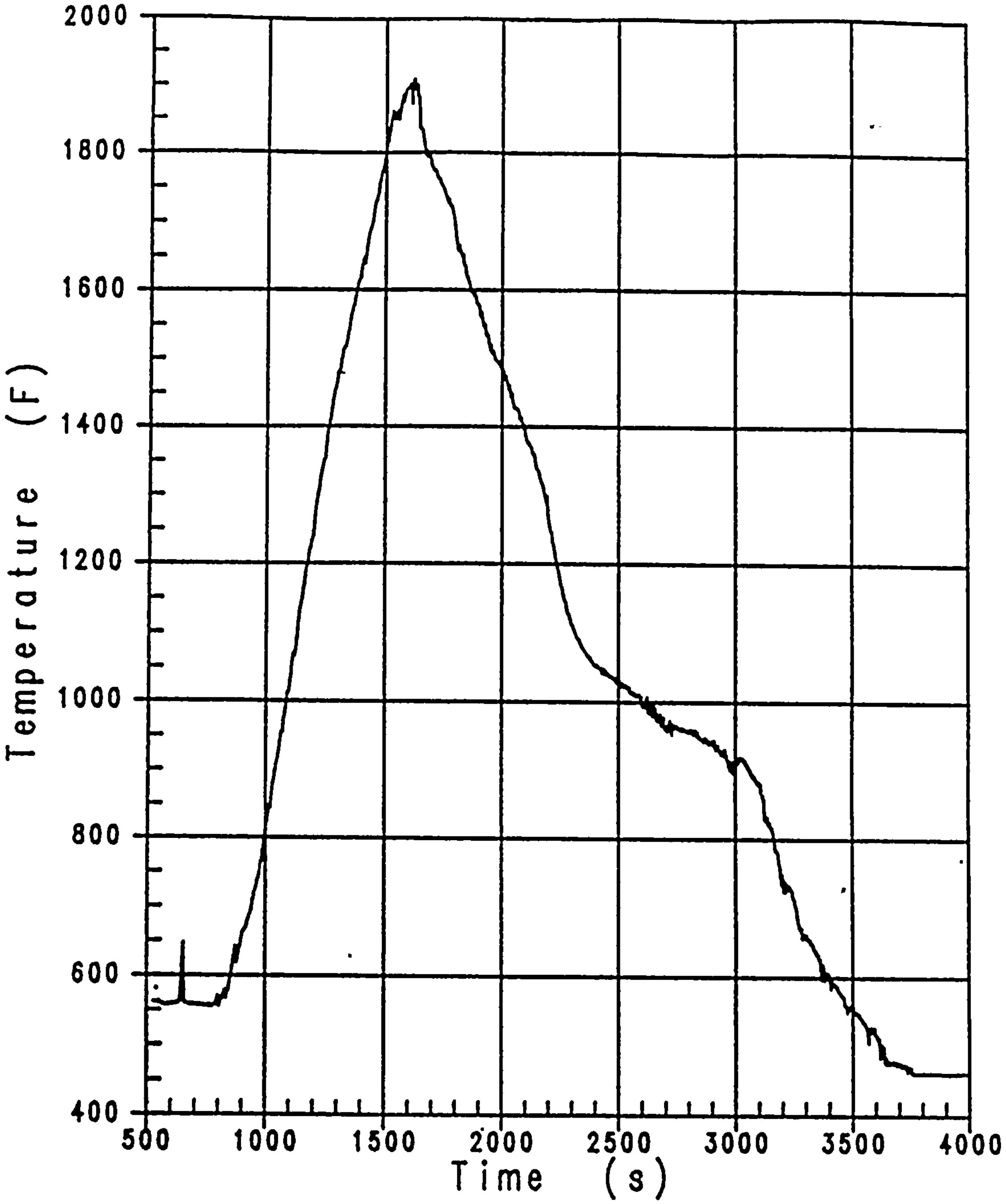


Figure 3.1-34 Hot Spot Fluid Temperature (3 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

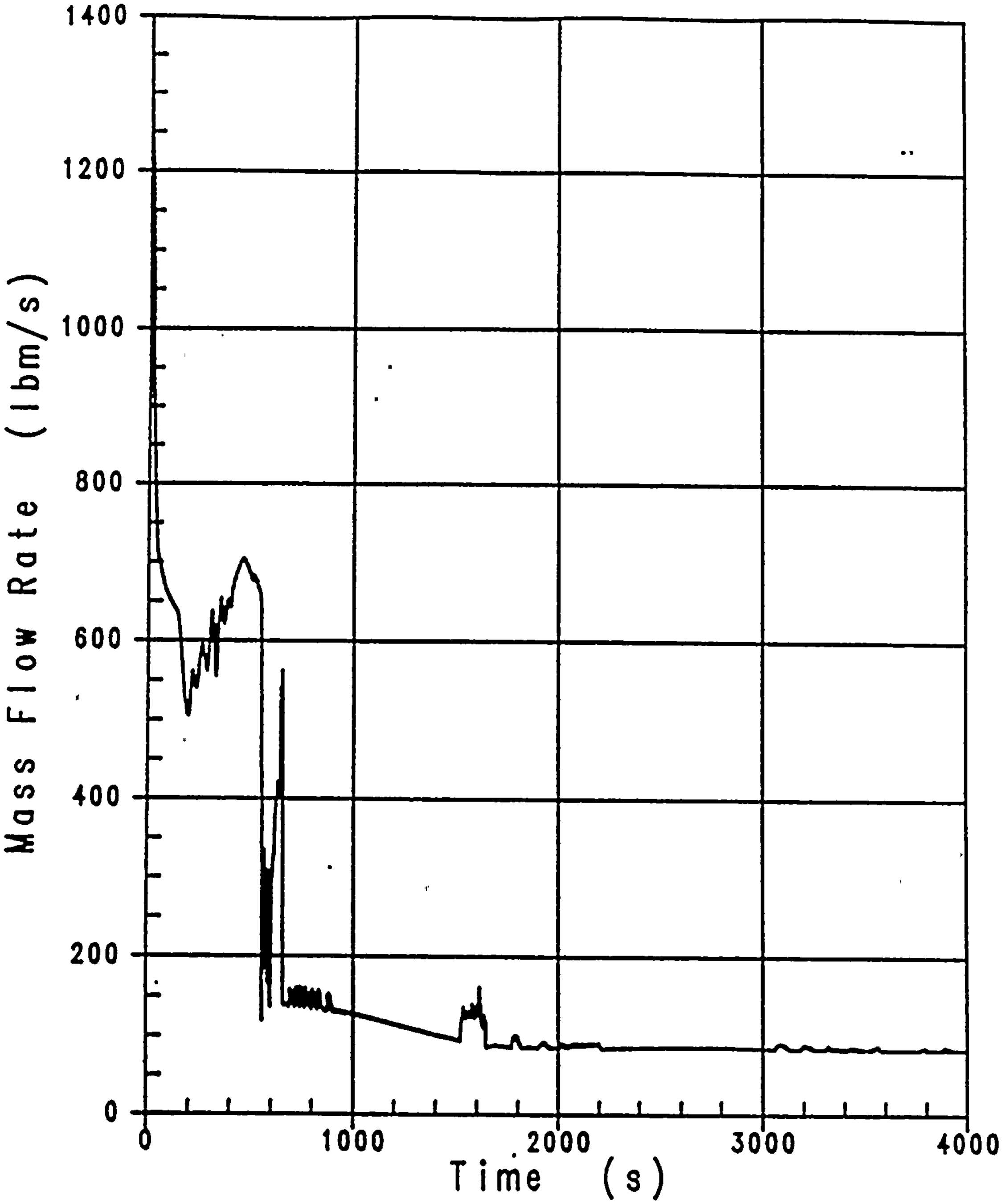


Figure 3.1-35 Cold Leg Break Mass Flow Rate (3 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

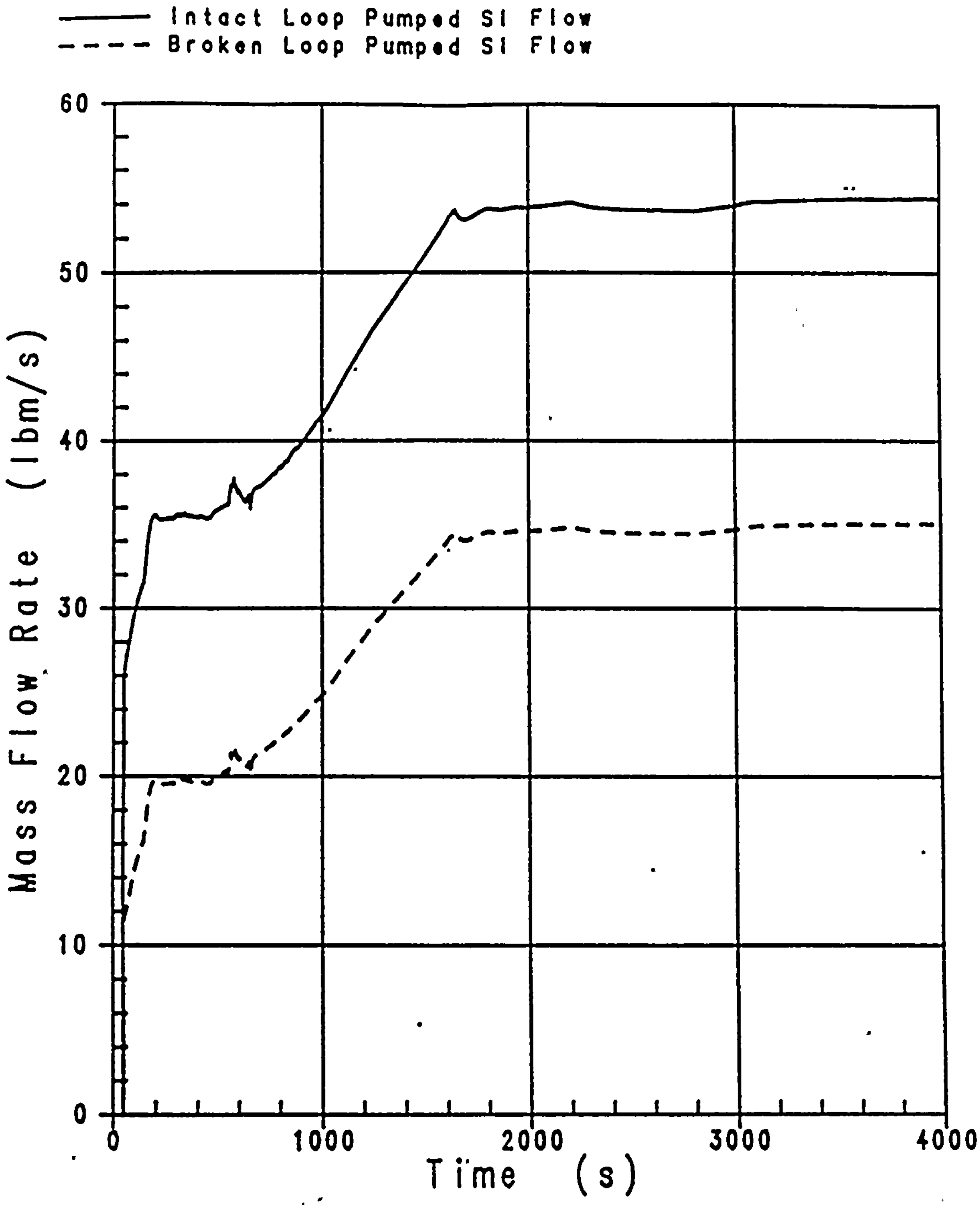


Figure 3.1-36 Safety Injection Mass Flow Rate (3 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

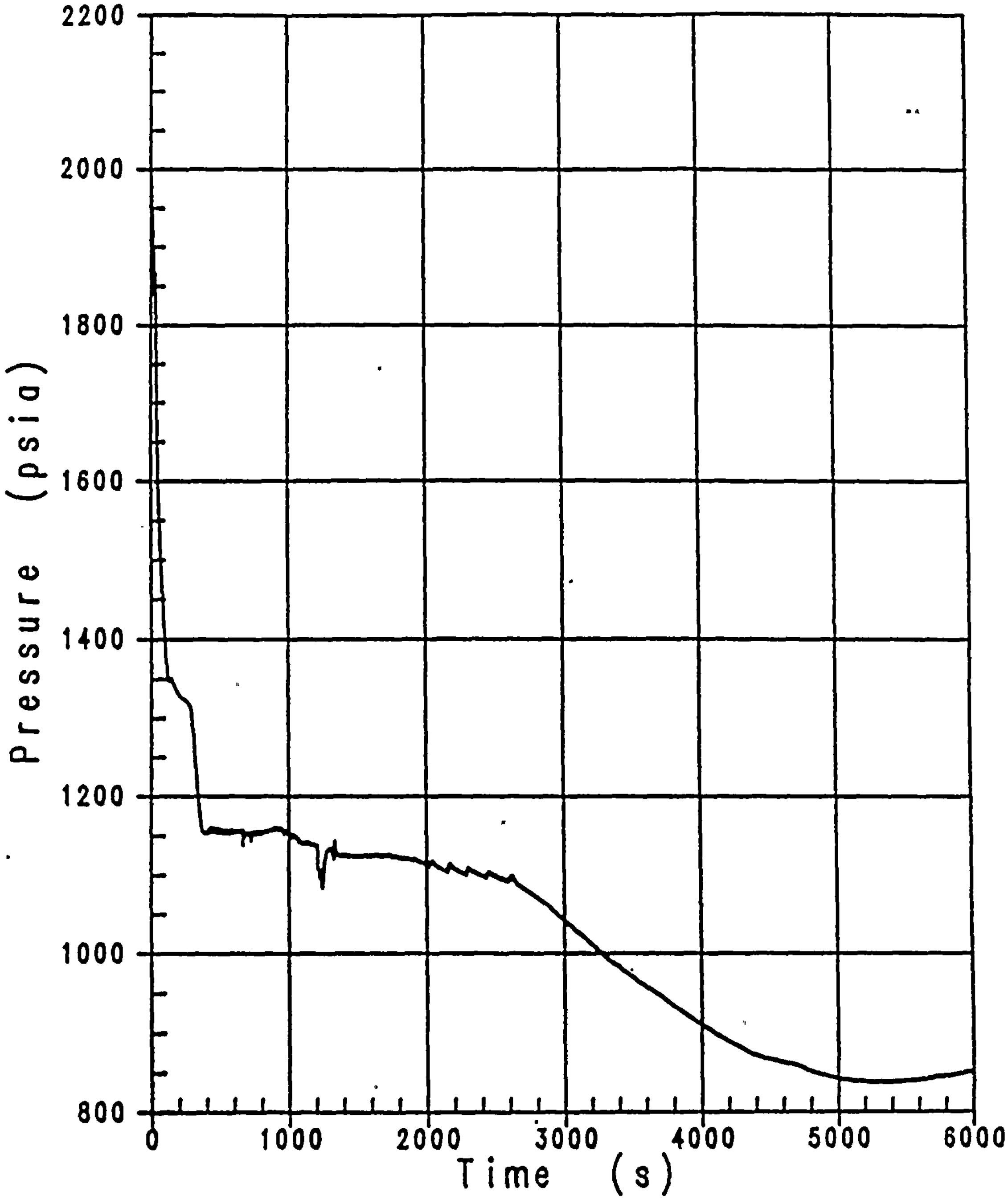


Figure 3.1-37 RCS Pressure (2 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

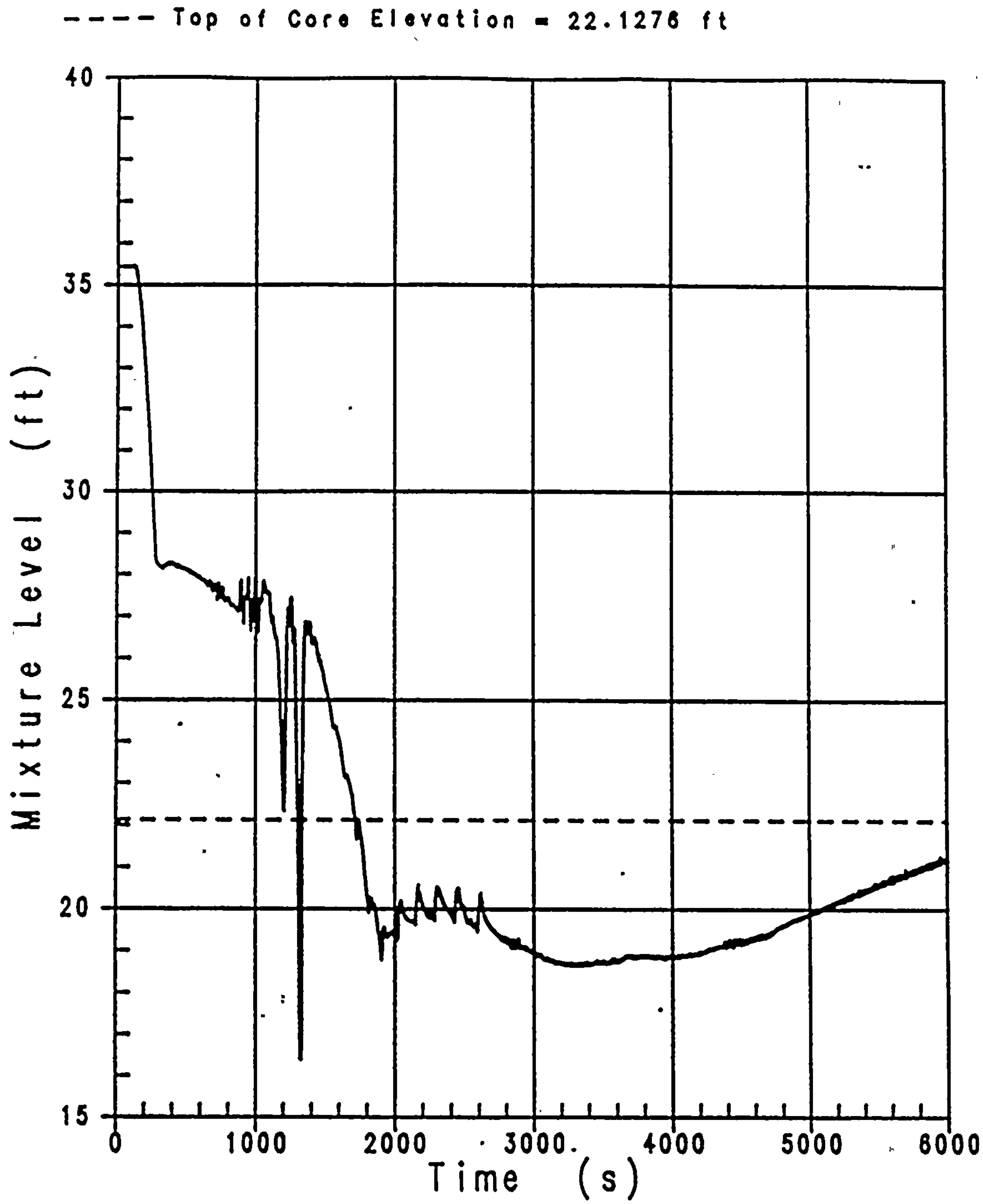


Figure 3.1-38 Core Mixture Level (2 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

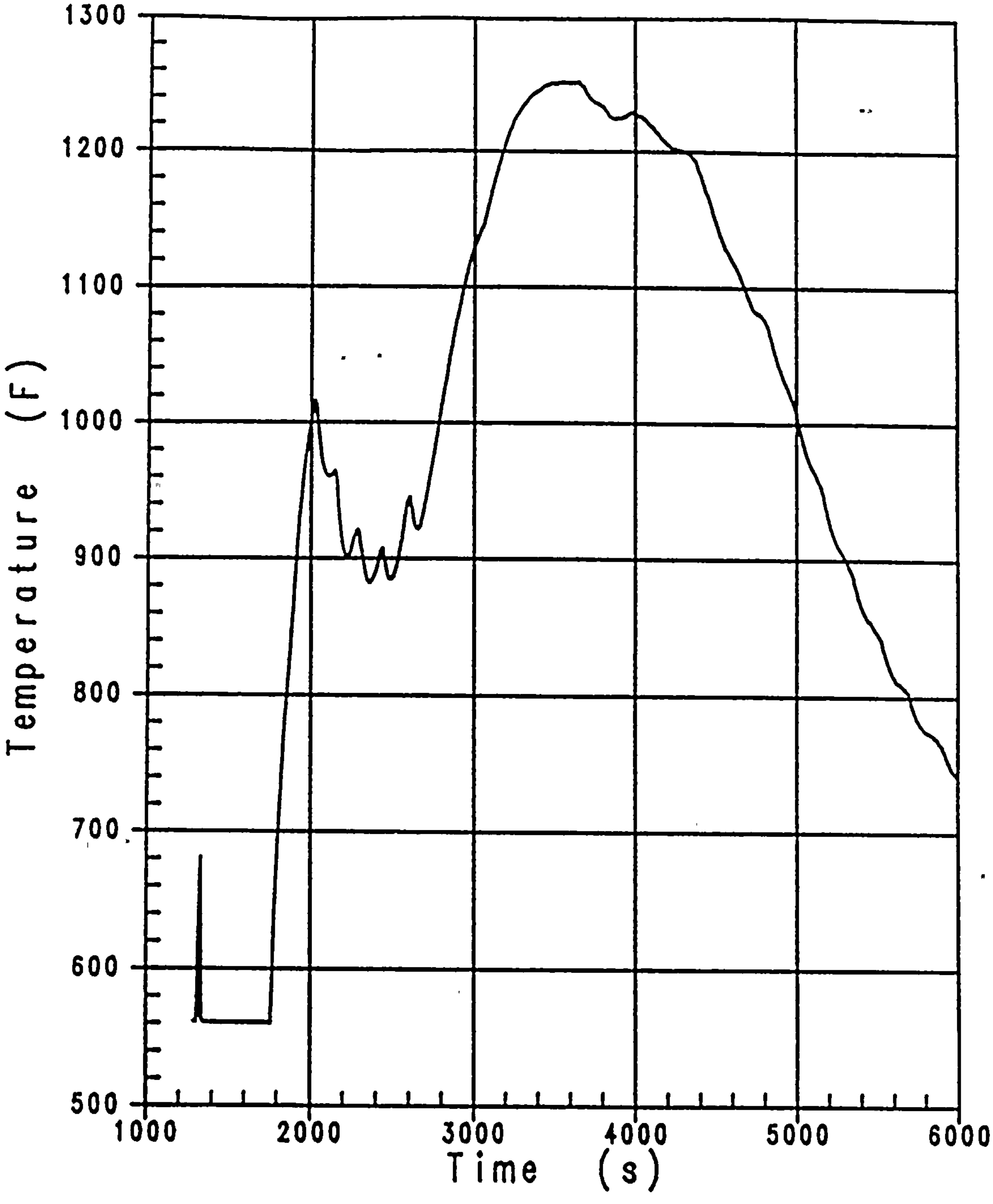


Figure 3.1-39 Peak Clad Temperature (2 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

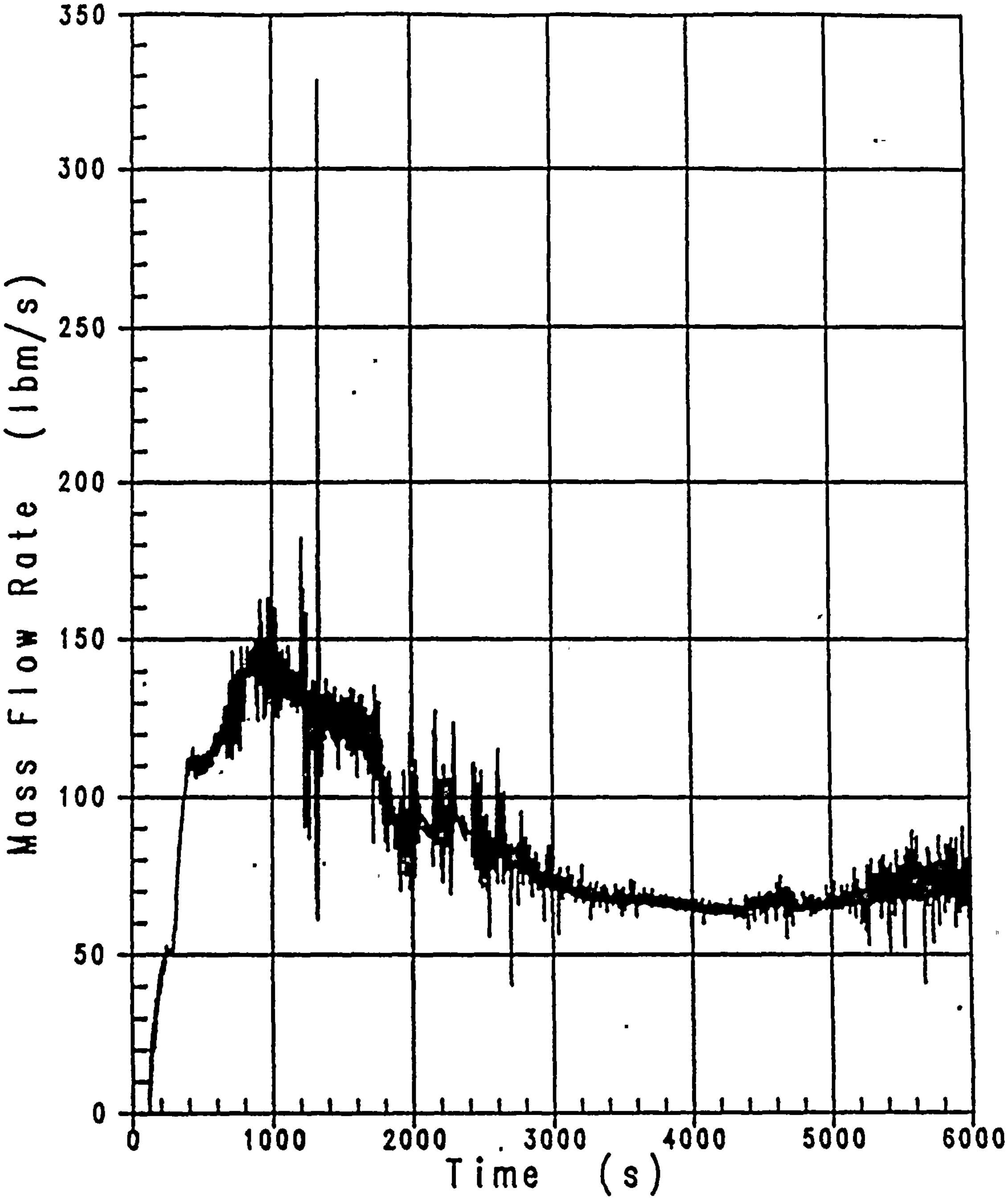


Figure 3.1-40 Core Outlet Steam Mass Flow Rate (2 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

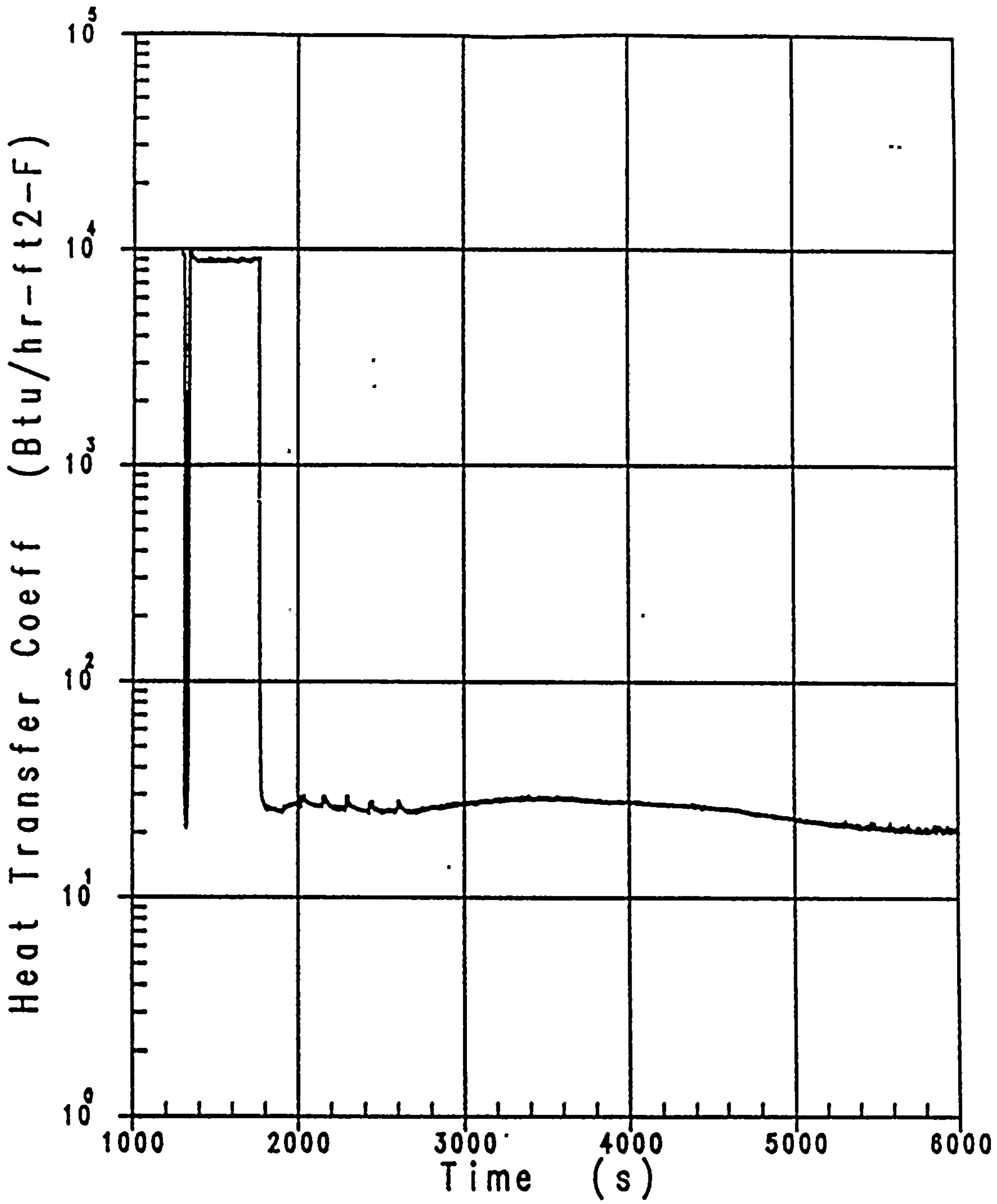


Figure 3.1-41 Hot Spot Heat Transfer Coefficient (2 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

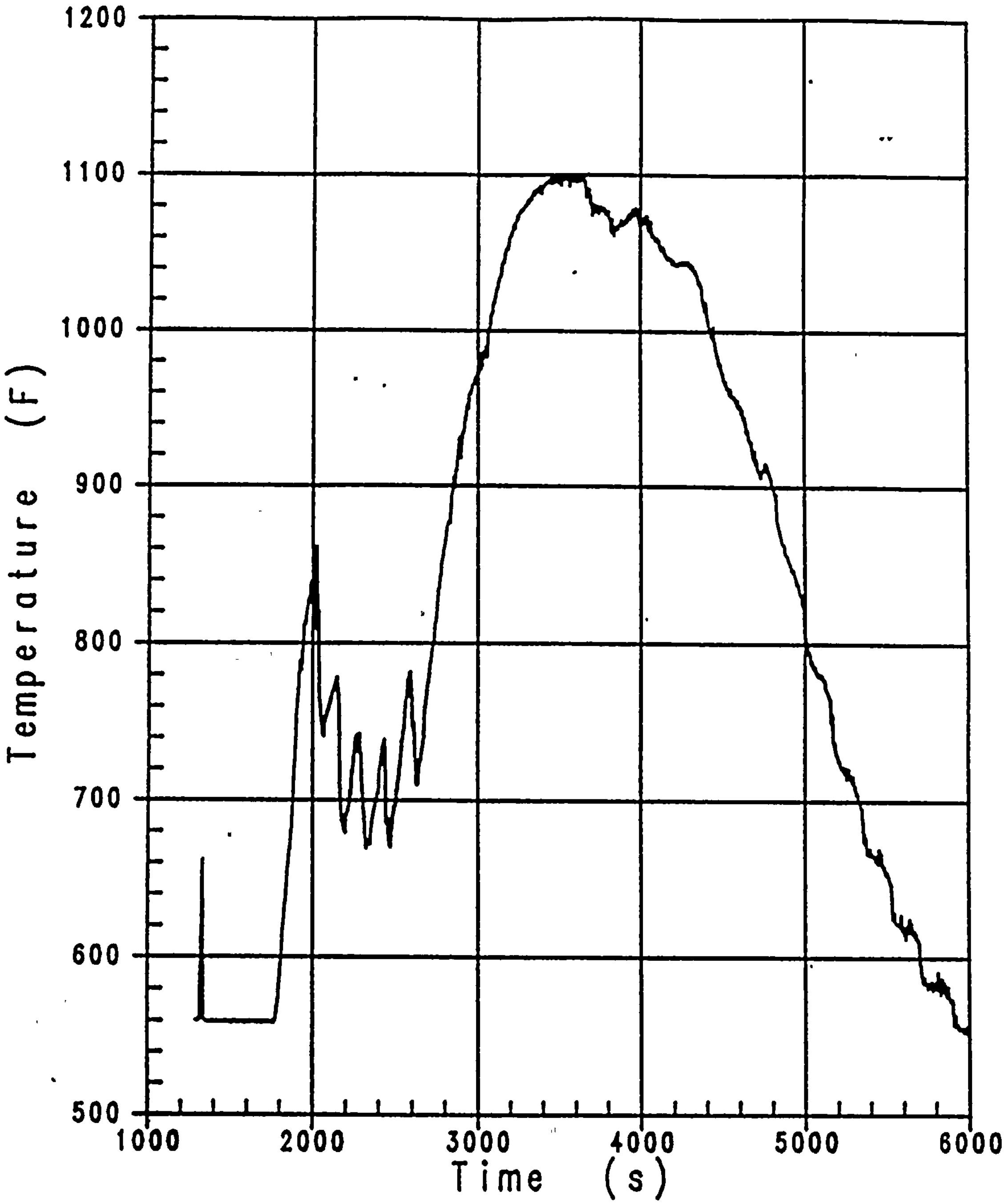


Figure 3.1-42 Hot Spot Fluid Temperature (2 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

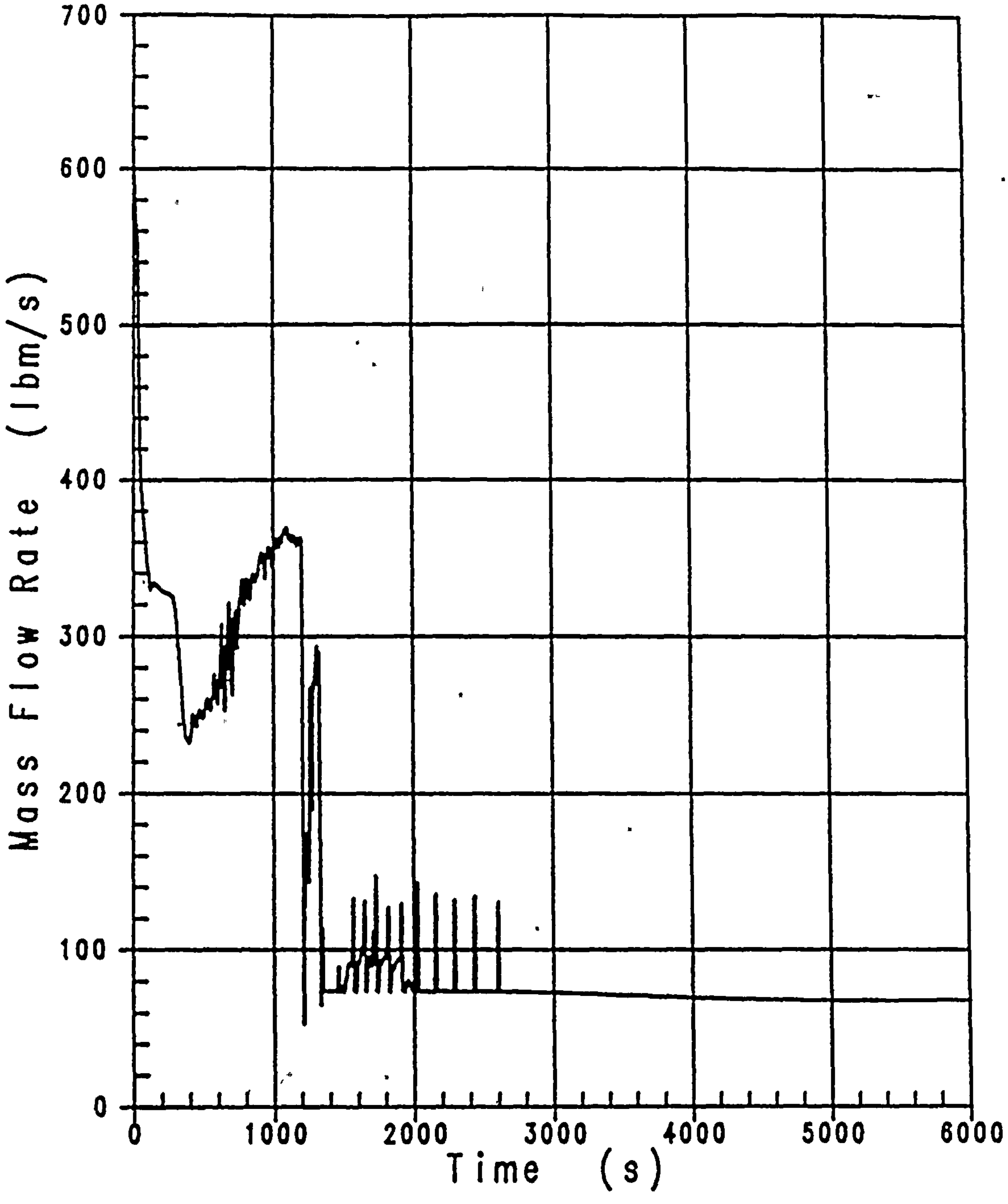


Figure 3.1-43 Cold Leg Break Mass Flow Rate (2 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

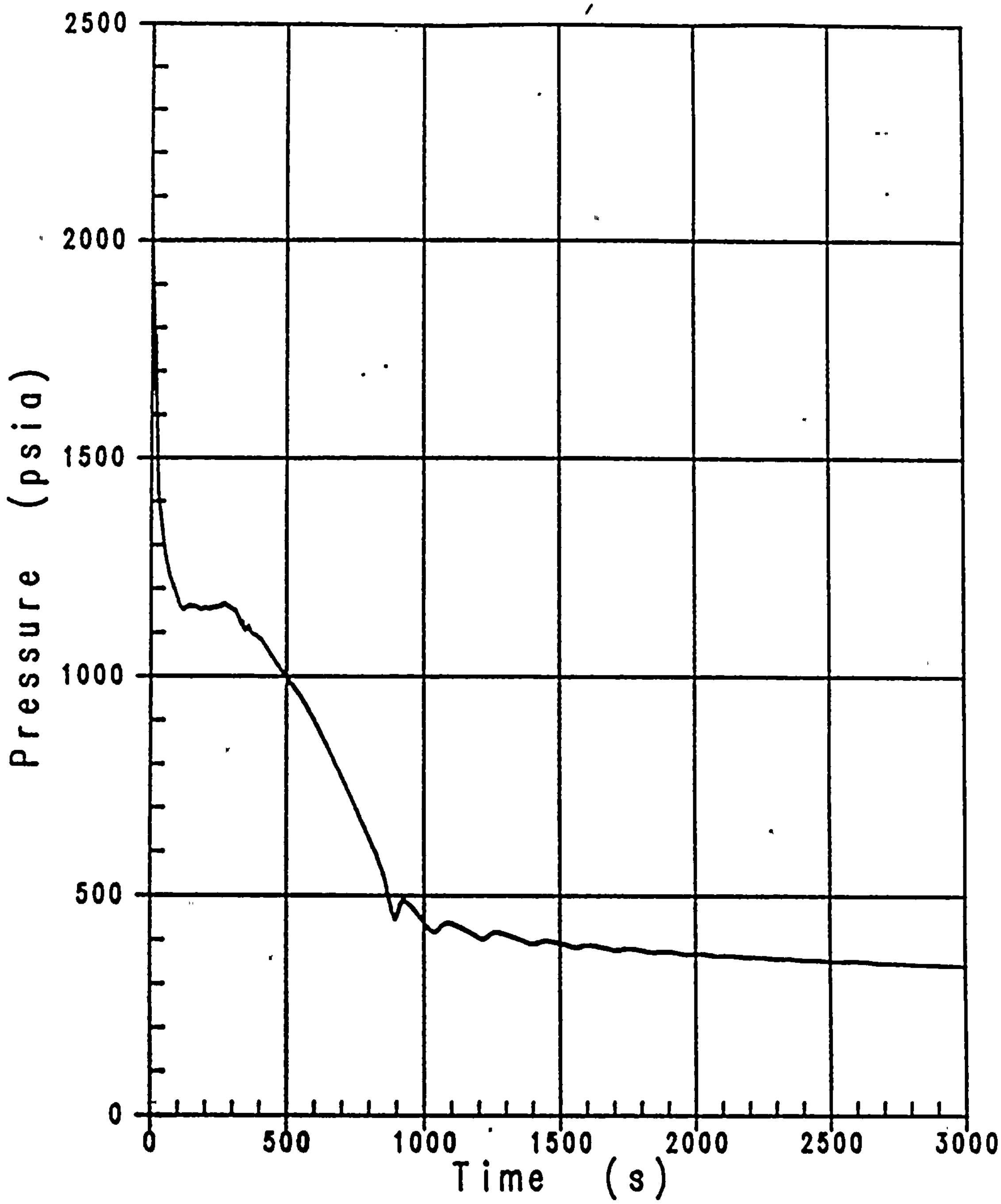


Figure 3.1-44 RCS Pressure (4 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

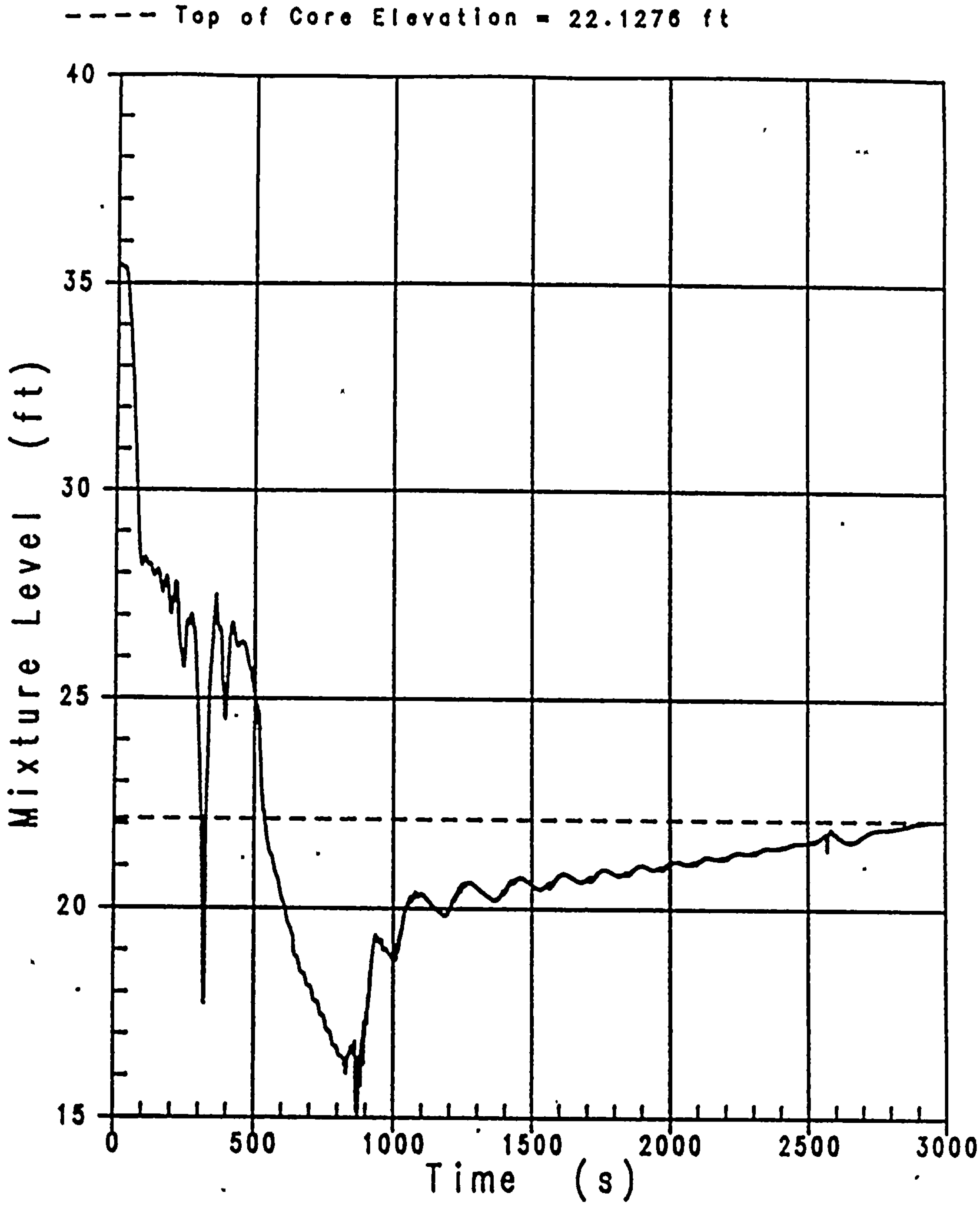


Figure 3.1-45 Core Mixture Level (4 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

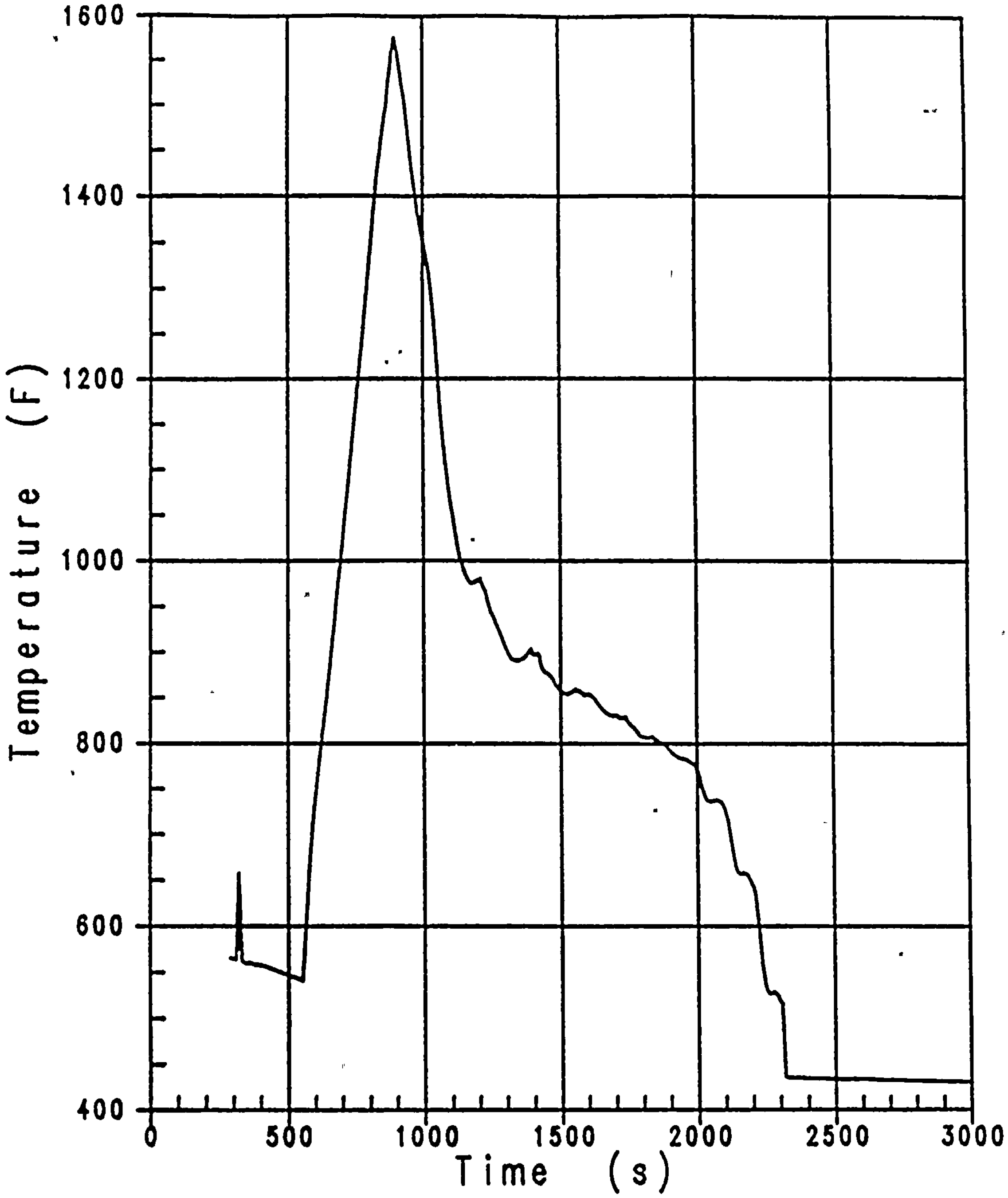


Figure 3.1-46 Peak Clad Temperature (4 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

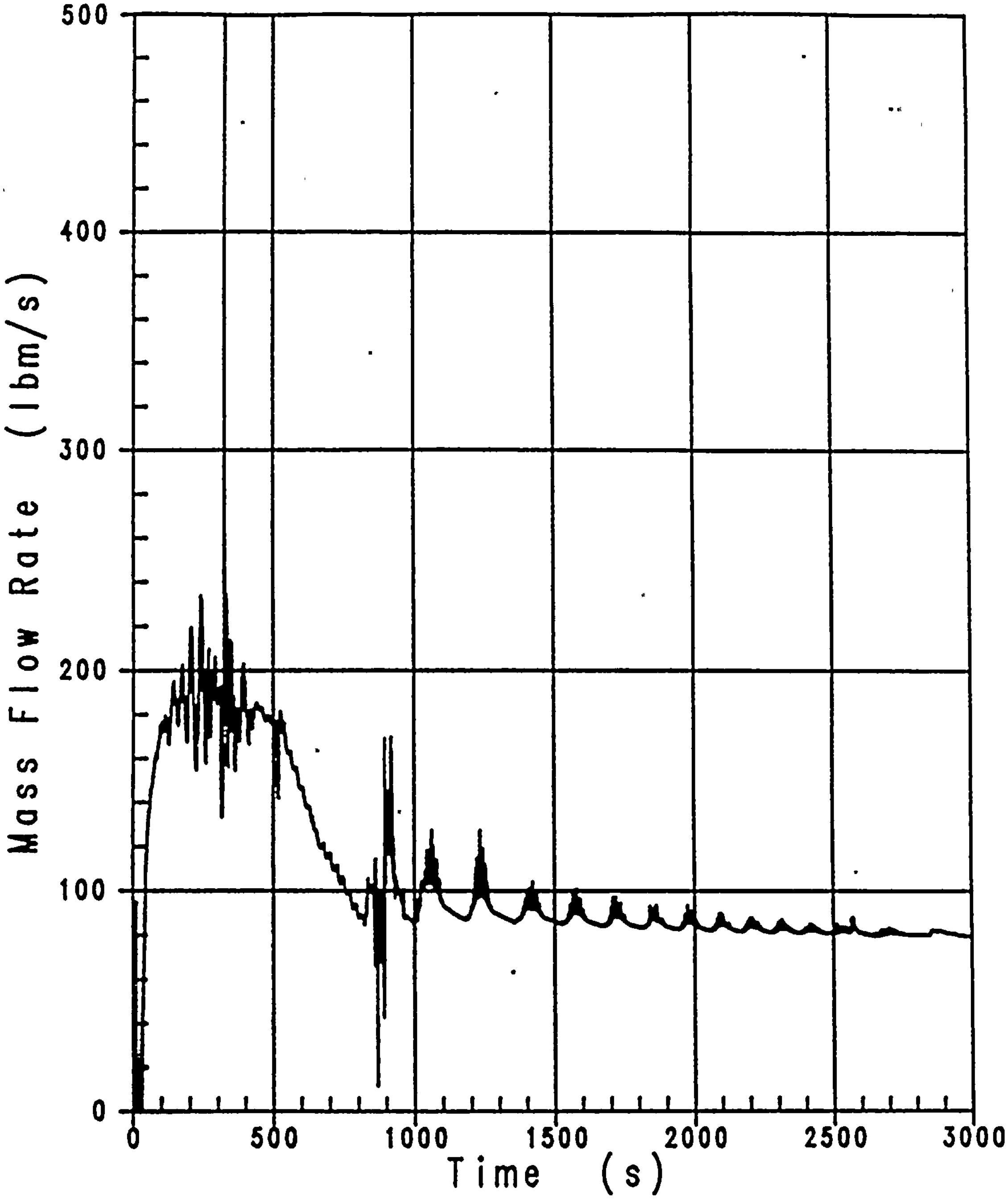


Figure 3.1-47 Core Outlet Steam Mass Flow Rate (4 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

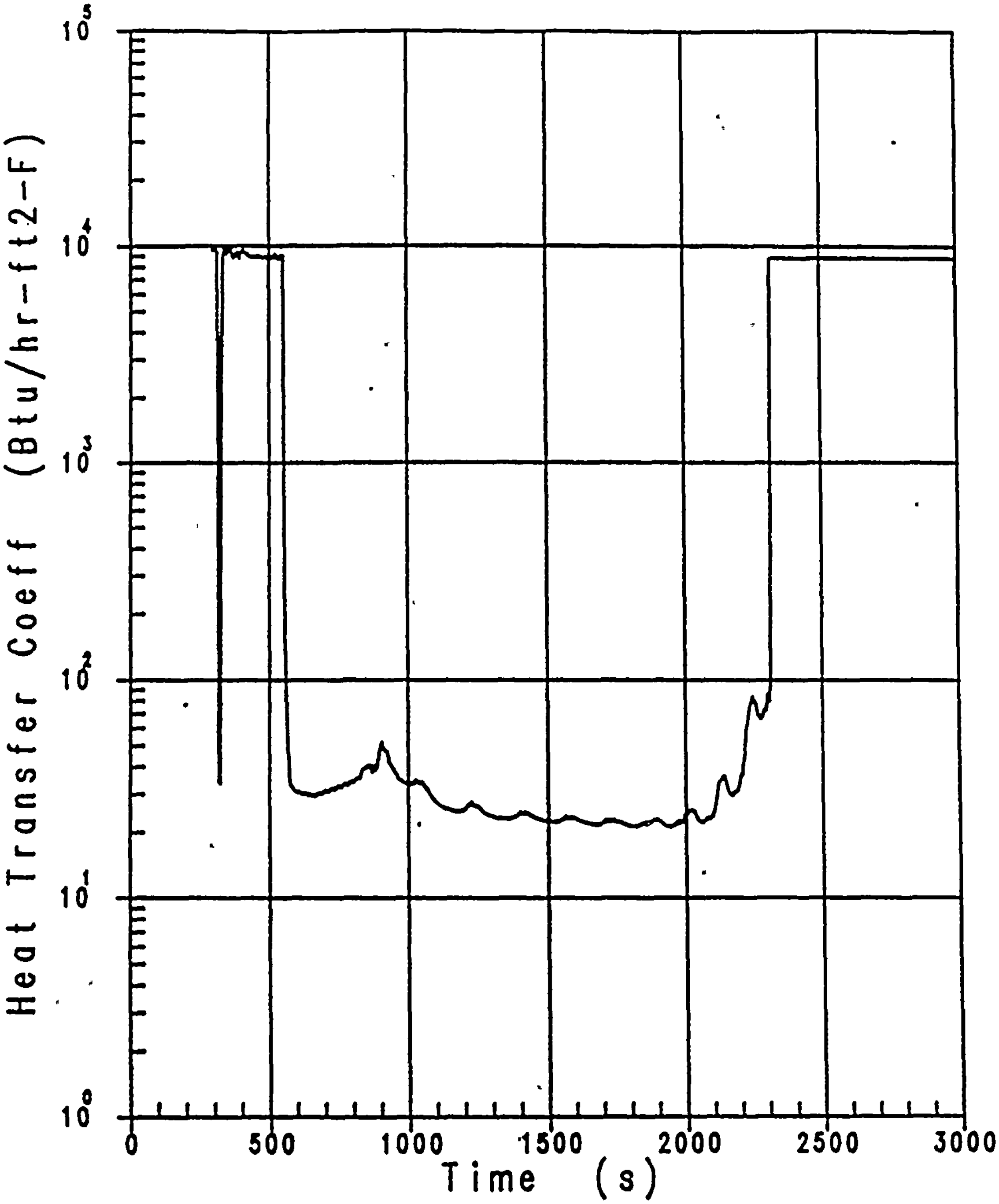


Figure 3.1-48 Hot Spot Heat Transfer Coefficient (4 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

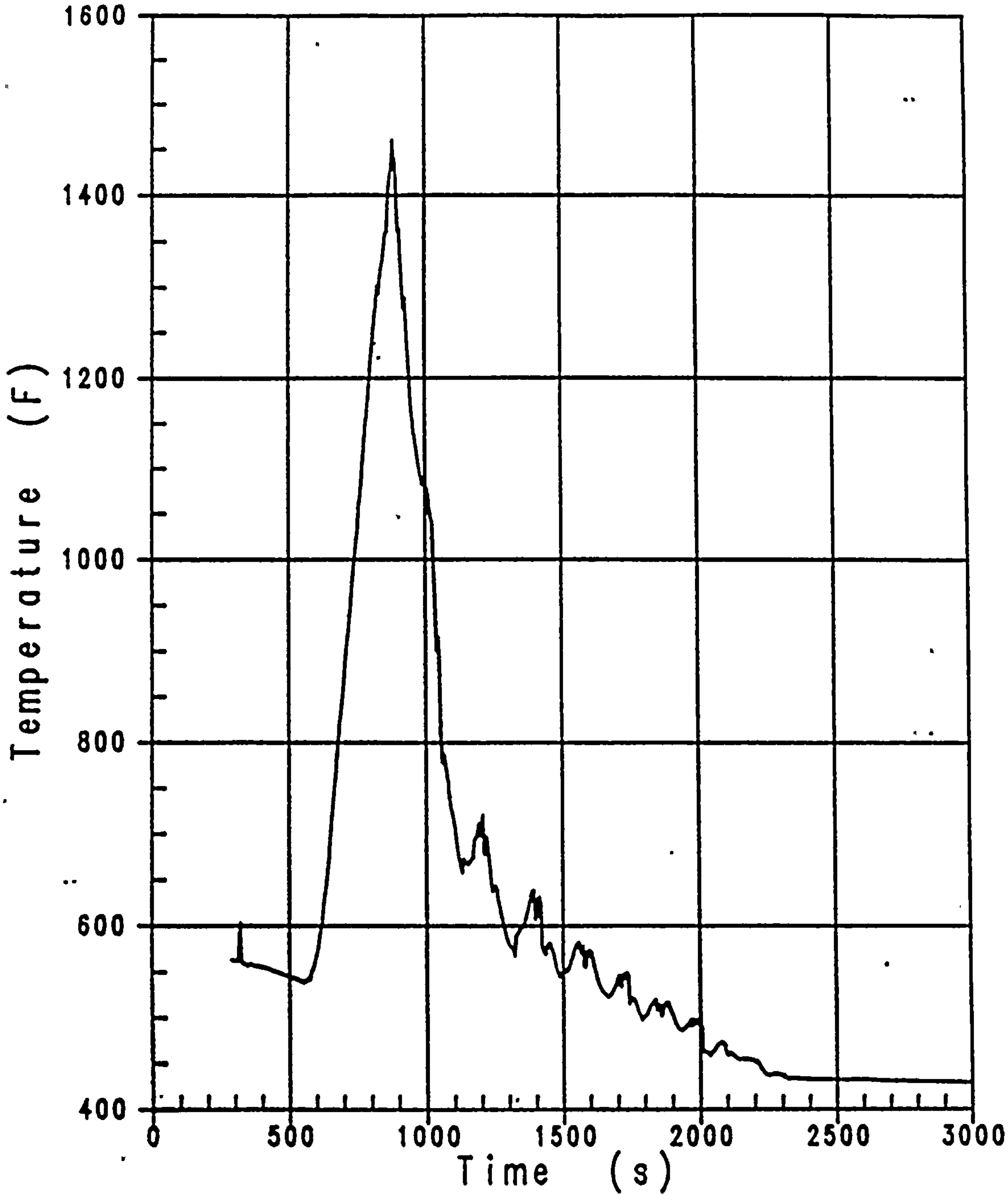


Figure 3.1-49 Hot Spot Fluid Temperature (4 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

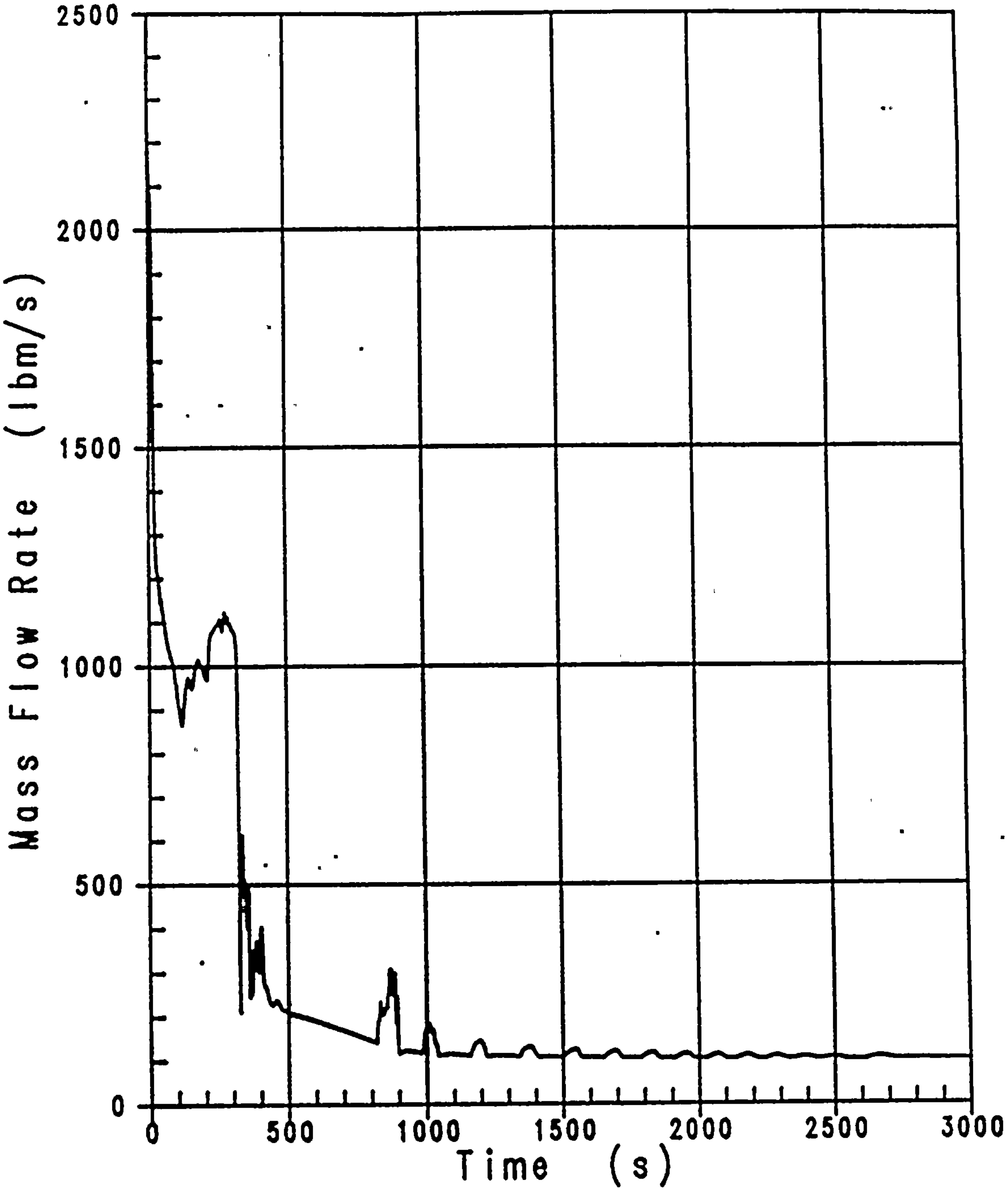


Figure 3.1-50 Cold Leg Break Mass Flow Rate (4 inch)
High Temperature, Reduced Pressure
HHSI Crosstie Valves Closed

3.2 LOCA HYDRAULIC FORCES

The LOCA hydraulic forces were analyzed for the VANTAGE 5 RTSR and used conservative values of 582.3°F and 511.7°F for T_{hot} and T_{cold}, respectively. These LOCA hydraulic forcing functions conservatively bound the RCS parameters in Table 2.1-1 of this report. Therefore, the LOCA hydraulic forces analyzed for the VANTAGE 5 RTSR remain valid and no changes and/or additions are required.

3.3 NON-LOCA ANALYSES

3.3.1 Introduction

This section evaluates the effects of the Donald C. Cook Nuclear Plant Unit 2 Uprating Program with respect to the non-LOCA safety analyses. The effort performed is to support Unit 2 operation with a core power of 3588 MWt in the range of full-power reactor vessel average temperatures between 547°F and 581.3°F at primary pressure values of 2100 psia or 2250 psia, with a maximum average steam generator tube plugging level of 10%, with a peak steam generator tube plugging level of 15% (Cases 1 through 4 of Table 3.3-1, which are identical to Cases 2 through 5 of Table 2.1-1).

The current non-LOCA analyses of record for Unit 2 support an uprated core thermal power of 3588 MWt (3600 MWt NSSS) with a full power vessel average temperature between 547°F and 581.3°F at a primary system pressure of 2100 psia or 2250 psia. Operation of Unit 2 at the uprated NSSS power of 3600 MWt, was supported as part of the transition to W 17x17 VANTAGE 5 fuel, as documented in the RTSR (Reference 1). Furthermore, it should be noted that the parameters presented as Cases 1 through 4 of Table 3.3-1 correspond to Cases 7, 5, 6, and 4 of Table B.2-1 of the VANTAGE 5 RTSR, respectively.

The Donald C. Cook Nuclear Plant Unit 2 licensing basis, as reported in the UFSAR (Reference 9) includes analyses and evaluations of sixteen non-LOCA events, which are delineated on the next two pages. This licensing-basis has been reviewed to assess the impact associated with the Uprating Program.

The following events were re-analyzed since the VANTAGE 5 RTSR work was completed. The first two events identified below were re-analyzed as part of this Uprating Program, whereas the remaining events were previously reanalyzed for reasons outside of this Uprating Program.

Unit 2 UFSAR Section	Analysis (Reason)
14.1.6	Loss of Reactor Coolant Flow (Including Locked Rotor) {Locked Rotor for Peak Pressure case only} (Revised Pressurizer Safety Valve Setpoint, Section 3.3.5.1)
14.1.8	Loss of External Electrical Load and/or Turbine Trip (Revised Pressurizer Safety Valve Setpoint, Section 3.3.5.2)
14.1.10	Excessive Heat Removal Due to Feedwater System Malfunctions {Hot Full Power cases only} (Revised Failed Feedwater Flow Rate, UFSAR)

14.3.4.4 (Unit 1) (bounds both units)	Mass and Energy Releases Analysis for Postulated Secondary System Pipe Ruptures Inside Containment (Revised Feedwater Isolation Delay time, and Containment Spray Flow and Response Time, Reference 14)

The following events/analyses have been evaluated since the VANTAGE 5 RTSR effort was completed. Some of these evaluations were performed prior to this Uprating Program, but are discussed herein for completeness, since the analyses performed in support of the VANTAGE 5 RTSR effort were used as a starting point for all of the non-LOCA events.

Unit 2 UFSAR Section	Analysis (Reason)
14.1	Overtemperature and Overpower ΔT Setpoints (Revised to support increased drift, Section 3.3.2.1)
14.1.2	Uncontrolled RCCA Bank Withdrawal At Power (Revised OTΔT Setpoints, Section 3.3.4.1)
14.1.3 / 14.1.4	Dropped RCCA (New Methodology Applied to Support Deletion of the Negative Flux Rate Trip, UFSAR and Section 3.3.4.2)
14.1.9	Loss of Normal Feedwater (Revised Main Steam and Pressurizer Safety Valve Setpoint Tolerance, Section 3.3.4.3)
14.1.12	Loss of All AC Power to the Plant Auxiliaries (Revised Main Steam and Pressurizer Safety Valve Setpoint Tolerance, Section 3.3.4.4)
14.2.5	Rupture of a Steam Pipe (core response analysis) (Revised Low Pressurizer Pressure Safety Injection Setpoint and Auxiliary Feedwater Flow Rates, and Consideration of the Centrifugal Charging Pump Minimum Flow Isolation Valves, Section 3.3.4.5)
14.2.8	Major Rupture of a Main Feedwater Pipe (Revised Low Pressurizer Pressure Safety Injection Setpoint and Auxiliary Feedwater Flow Rates, and Consideration of the Centrifugal Charging Pump Minimum Flow Isolation Valves, Section 3.3.4.6)
14.4.11.3 (Unit 1)	Steamline Break Mass/Energy Releases Outside Containment (Revised OTΔT Setpoints, Section 3.3.4.7)

Notice that Section 3.3.2.1 of this report discusses the revised OTΔT and OPΔT reactor trip function setpoint equation coefficients.

The following events/analyses have not been analyzed nor evaluated since the completion of the VANTAGE 5 RTSR effort. As such, the analyses performed to support the transition to 17x17 VANTAGE 5 and coincident uprating of Unit 2 continue to remain valid.

Unit 2 UFSAR Section	Analysis
14.1.1	Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition
14.1.3 / 14.1.4	Rod Cluster Control Assembly Misalignment
14.1.5	Chemical and Volume Control System Malfunction
14.1.6	Loss of Reactor Coolant Flow (Including Locked Rotor) (Locked Rotor Rods-in-DNB and PCT cases only)
14.1.7	Start-up of an Inactive Loop
14.1.10	Excessive Heat Removal Due to Feedwater System Malfunctions (Hot Zero Power cases only)
14.1.11	Excessive Load Increase Incident
14.2.6	Rupture of a CRDM Housing (Rod Ejection)

3.3.2 Non-LOCA Safety Analysis Assumptions Requiring Technical Specification Changes

To enhance operating flexibility over the range of parameters defined in Table 3.3-1, certain Reactor Protection System (RPS) setpoints were revised. The pressurizer code safety valve setpoint tolerance applicable to the safety analyses has been relaxed to account for drift during the fuel cycle (i.e., the lift setting for the PSVs is calibrated to ±1%, however, the Non-LOCA safety analyses support ±3% for as found operability).

The revised RPS setpoints include the overtemperature ΔT (OTΔT) and the overpower ΔT (OPΔT) reactor trips. The general equations for the OTΔT and OPΔT reactor trip setpoints and the safety analysis limit coefficient values are presented in Table 3.3-4. A detailed discussion of the revised setpoint equations for these reactor trip functions is provided in Section 3.3.2.1.

The discussions of the "as found" pressurizer code safety valve setpoint tolerance adjustment is presented in the Section 3.3.2.2. The applicable Technical Specification updates for these revisions/relaxations are provided in Appendix A.

3.3.2.1 Reactor Protection System Trip Setpoints

Revised OTΔT and OPΔT setpoints are based upon 17x17 VANTAGE 5 core thermal safety limits (the same limits used for the VANTAGE 5 RTSR Program), which employ the methodology described in Reference 7. The OPΔT and OTΔT setpoints were revised to increase the available margin between the safety analysis setpoint values and the nominal, or Technical Specification values, such that more ΔT-drift could be accommodated between instrumentation calibrations during the fuel cycle. Presently, the power margin associated with VANTAGE 5 RTSR Program is being utilized to offset the ΔT-drift that is being experienced during core burnup (i.e., the core power of 3588 MWt is supported by the analyses, but the plant has been operated with a core full-power value of 3411 MWt). However, once the plant is licensed to operate at a core power of 3588 MWt, this power margin will no longer be available. Thus, there is a need to revise the OTΔT and OPΔT setpoints as part of this Uprating Program.

Figures 3.3-1 through 3.3-4 present the allowable reactor coolant loop average temperature and ΔT conditions as a function of primary coolant pressure, based upon a minimum measured flow (MMF) of 366,400 gpm and a 1.55 chopped cosine axial power distribution. Figure 3.3-1 represents the most limiting Uprated operating configuration (nominal full-power T_{avg} = 581.3°F, nominal pressure = 2100 psia) of the range of conditions described in Table 3.3-1 (Case 2) for the calculation of the OTΔT and OPΔT setpoints. The boundaries of operation defined by the OTΔT and OPΔT trips are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, a trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the limit value (for full VANTAGE 5 cores, 1.69 and 1.61 for typical and thimble cells, respectively; see Table 3.12-3 of this report and Table 4-2 of the RTSR). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the Safety Analysis Limit DNBR value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable Safety Analysis Limit DNBR at any point.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high and low RCS pressure (fixed setpoints); overpower and overtemperature ΔT (variable setpoints), and the opening of the steam generator safety valves which limit the maximum RCS average temperature. The Safety Analysis Limit DNBR value (1.69 typical and 1.61 thimble), which was used as the DNBR limit for all accidents analyzed with the Revised Thermal Design Procedure (RTDP; Reference 2), is conservative compared to the actual Design Limit DNBR value (for full VANTAGE 5 cores, 1.23 and 1.22 for typical and thimble cells, respectively; see Table 3.12-3 of this report and Table 4-2 of the RTSR), required to meet the DNB design basis.

Table 3.3-2 presents the limiting trip setpoints assumed in the accident analyses and the time delay values assumed for each trip function. The difference between the limiting trip point assumed for the analysis and the normal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications. During plant start-up tests, it is demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times are determined periodically in accordance with the Technical Specifications.

The safety analysis methodology is based upon the assumption that the reference average temperatures (T' and T") used in the OTΔT and OPΔT setpoint equations are scaled to the appropriate full-power values each time the cycle average temperature is changed. It is also assumed that the reference pressure (P') in the OTΔT equation is set equal to the appropriate nominal primary system pressure for a particular cycle (either 2100 psia or 2250 psia). These assumptions are key to ensure that the actual plant conditions required to result in an OTΔT and/or OPΔT trip signal to be generated are conservative with respect to assumptions made in the safety analyses. Figures 3.3-1 through 3.3-4 illustrate the OTΔT and OPΔT protection setpoints for the endpoints of the range of full-power vessel average temperatures for the Uprating Program at either 2100 psia or 2250 psia. The calibration of the NIS excore detectors, to compensate for the changes in coolant density each time the cycle operating conditions are changed, is also assumed in the analyses.

The OTΔT and OPΔT reactor trip functions provide primary protection against fuel centerline melting, among other concerns (i.e., DNB and hot-leg boiling). The criterion for no fuel melt is, the uranium dioxide melting temperature shall not be exceeded for at least 95 percent of the limiting fuel rods at a 95 percent confidence level (Reference 7). This criterion is met by limiting the calculated fuel centerline temperature to 4700°F (valid for approximately 60000 MWD/MTU burnup per Reference 10). In many cases, fuel centerline melting can be prevented by limiting gross core thermal power to a prescribed limit (historically 118% of nominal power) independent of axial power distribution. As part of the reload process (via the Reload Safety Analysis Checklist, or RSAC), the peak linear heat generation rate of the core (i.e., peak kw/ft) is determined specifically for fuel centerline melting concerns associated with Condition II events. Even though the revised OTΔT and OPΔT reactor trip setpoint equations allow the typical gross core average thermal power to slightly exceed the historical value of 118% (Cook Unit 2 safety analyses indicate that a peak overpower of 119.31% can be supported with the revised setpoints), fuel centerline melt concerns are specifically evaluated on a cycle-by-cycle basis as part of the formal reload process to ensure that fuel centerline melting does not occur as a result of a Condition II event.

Since the revised OTΔT and OPΔT reactor trip setpoint equations allow the typical gross core average thermal power to slightly exceed 118%, as noted above, this fact was used to address concerns associated with a full power steamline break - core response event. The full-power steamline break analysis for core response considerations is not in the Donald C.

Cook Nuclear Plant licensing basis. Nevertheless, it has been determined that the revised OTΔT and OPΔT setpoint equations, with the OPΔT reference average temperature (T") restricted to values from 547°F to no greater than 576.0°F, provides sufficient assurance that the minimum DNBR will be protected during an at-power steamline break accident.

The time constant, τ_1, of the OTΔT T_{avg} lead-lag compensator is 22 seconds. This value was previously supported by the VANTAGE 5 RTSR program analyses for operation with a core fully loaded with W VANTAGE 5 fuel. The analyses and evaluations performed as part of this Uprating Program assumed a τ_1 value consistent with unit operation with only W VANTAGE 5 fuel loaded in the core.

3.3.2.2 Pressurizer Code Safety Valve Setpoint Tolerance Increase

The following events, which are potentially impacted by an increased pressurizer code safety valve setpoint tolerance, have been shown to support an "as found" value of ±3% setpoint tolerance: Loss of External Electrical Load and/or Turbine Trip; Loss of Normal Feedwater; Loss of All AC Power to the Station Auxiliaries; Major Rupture of a Main Feedwater Pipe; and Locked Rotor/Shaft Break events. Thus, an "as found" setpoint tolerance of ±3% for the pressurizer code safety valves is acceptable.

3.3.3 Methodology

The Unit 2 non-LOCA safety evaluation for the Uprating Program was performed using current Westinghouse methodology and computer codes. The following three sub-sections discuss: the Initial Conditions assumed, which reflect the Revised Thermal Design Procedure (RTDP) for most of the events that are DNB limited; the Computer Codes Utilized; and a Clarification of Steam Generator Tube Plugging Levels.

3.3.3.1 Initial Conditions

For most accidents which are DNB limited, nominal values of initial conditions and minimum measured flow (366,400 gpm) are assumed. The allowances on reactor power, RCS temperature and pressure are determined on a statistical basis and are included in the limit DNBR as described in WCAP-11397-A (Reference 2). This procedure is known as the "Revised Thermal Design Procedure" or RTDP.

For accidents that are not DNB limited or in which RTDP is not employed, the initial conditions are obtained by adding the maximum steady-state errors to rated values. The following steady-state errors are considered:

a.	Core Power	+2% calorimetric error allowance
b.	Average RCS Temperature	+4.1°F / -5.6°F controller deadband and measurement error allowance (see discussion of cold-leg streaming below)
c.	Pressurizer Pressure	±62.6 psi steady-state fluctuations and measurement error allowance (see paragraph below)
d.	Reactor Flow	Thermal Design Flow (354,000 gpm)

An allowance for cold-leg streaming is not included in the values presented above. However, an allowance has been made by American Electric Power Service Corporation as documented in the Reference 15 evaluation. This evaluation, which continues to remain applicable, biases the hot full power Tavg by 1°F.

The pressurizer pressure controller uncertainty is ±38.2 psi. AEPSC has calculated a "readability" uncertainty of ±18.9 psi, which is only applicable for DNB considerations. However, the ±62.6 psi uncertainty, which is based upon uncertainty calculations performed as part of the RTSR Program, was conservatively applied to all non-LOCA analyses for simplicity.

Table 3.3-5 summarizes initial conditions and computer codes used for all of the Unit 2 accident analyses, and shows which accidents employed a DNB analysis using the RTDP.

3.3.3.2 Computer Codes Utilized

Summaries of the principal computer codes used in the transient analyses are given below. The codes used in the analysis of each transient have been listed in Table 3.3-5.

FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad using as input the nuclear

power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel rod model which simultaneously exhibits the following features:

a. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

b. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.

c. The necessary calculations to handle post-departure from nucleate boiling transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the materials.

FACTRAN is further discussed in Reference 3.

LOFTRAN

The LOFTRAN program is used for transient response studies of a pressurized water reactor (PWR) system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), and the pressurizer. The pressurizer heaters, spray, relief, and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogenous, saturated mixture for the thermal transients and a water level correlation for indication and control. The Reactor Protection System is simulated to include reactor trips on high neutron flux, overpower ΔT, overtemperature ΔT, high and low pressurizer pressure, low flow, and high pressurizer water level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The ECCS, including the accumulators, can also be modeled.

LOFTRAN also has the capability of estimating the transient value of DNBR based on the input from the core limits. The core limits represent the minimum value of DNBR as calculated for a typical or thimble fuel cell.

LOFTRAN is further discussed in Reference 4.

TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady-state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and

moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperatures, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided, e.g., channel-wise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 5.

THINC-IV

The THINC-IV computer program, as approved by the NRC, is used to determine coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distributions along parallel flow channels within a reactor core under all expected operating conditions. The THINC-IV code is described in detail in Reference 6.

3.3.3.3 Clarification of Steam Generator Tube Plugging Levels

The analyses of the VANTAGE 5 RTSR Program support a maximum average steam generator tube plugging level of 10%. This report will continue to support this level of plugging with the added clarification that the peak plugging level allowed may be as high as 15%, provided that the average plugging level is still less than or equal to 10%. The power capability parameters presented in Table 3.3-1 reflect this level of average steam generator tube plugging, with asymmetry, as indicated.

3.3.4 Non-LOCA Safety Evaluation: Transients Evaluated

The sections that follow contain the detailed descriptions of the individual non-LOCA analyses, and how they support the uprating of Unit 2. This first grouping of transients are those which have/had issues, concerns, or changes associated with the event that can/could be evaluated, and the second grouping (Section 3.3.5) are transients which required re-analysis. In all cases, the appropriate UFSAR acceptance criteria are satisfied. Recall that eight events have not undergone any evaluations to date since the VANTAGE 5 RTSR effort, which included operation at a NSSS power of 3600 MWt, as discussed previously in this report (see the end of Section 3.3.1). Therefore, these eight analyses that were performed for the VANTAGE 5 transition support the uprating of Unit 2 and continue to be the analysis of record for these eight transients.

3.3.4.1 Uncontrolled RCCA Bank Withdrawal At Power

An uncontrolled Rod Cluster Control Assembly (RCCA) withdrawal at power results in an increase in core heat flux. Since the heat extraction from the steam generator lags behind the power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise may eventually result in DNB. Therefore, to minimize the possibility of breaching the cladding, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the limit value.

The automatic features of the Reactor Protection System, which minimize adverse effects to the core in a RCCA Bank Withdrawal incident at power, include the following:

1. Nuclear power range instrumentation actuates a reactor trip on high neutron flux if two-out-of-four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an overtemperature ΔT (OTΔT) setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two-out-of-four ΔT channels exceed an overpower ΔT (OPΔT) setpoint. This setpoint is automatically varied with coolant average temperature so that the allowable fuel power rating is not exceeded.

4. A high pressurizer pressure reactor trip, actuated from any two-out-of-four pressure channels, is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip, actuated from any two-out-of-three level channels, is set at a fixed point.

In addition to the above listed reactor trips, there are the following RCCA Withdrawal blocks.

a. High neutron flux (one-out-of-four)
b. Overpower ΔT (two-out-of-four)
c. Overtemperature ΔT (two-out-of-four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of Reactor Coolant System conditions is illustrated in Figures 3.3-1 through 3.3-4. These figures, which reflect the OTΔT and OPΔT setpoints that have been revised as part of this Uprating Program, represent the allowable conditions of

reactor coolant loop average temperature and power (i.e., ΔT is proportional to core power) with the design power distribution in a two-dimensional plot.

The purpose of this analysis is to demonstrate the manner in which the above protective systems function for various reactivity insertion rates from different initial conditions. Reactivity insertion rates and initial conditions govern which protective function occurs first.

The current analysis of record for the Unit 2 Uncontrolled RCCA Withdrawal At Power event was performed in support of the transition to W 17x17 VANTAGE 5 fuel, as documented in Reference 1. The revised OTΔT and OPΔT setpoint equations that have been created as part of this Uprating Program do not impact the current licensing basis safety analysis, as the OPΔT trip is not credited in the analysis, and the revised OTΔT setpoint is bounded by that modeled in the existing analysis. Specifically, the safety analysis limit K_1 gain value has been revised from 1.39 to 1.37. Thus, the lower K_1 will result in a OTΔT reactor trip signal being generated earlier in the transient than that currently analyzed. A quicker reactor trip will yield results less severe (i.e., higher minimum DNBR values) than those predicted by the current analysis of record for all cases that trip on the OTΔT reactor protection function. Therefore, the conclusions presented in the VANTAGE 5 RTSR and the Donald C. Cook Nuclear Plant Unit 2 UFSAR (Reference 9) remain applicable.

3.3.4.2 Dropped RCCA

The rod cluster control assembly (RCCA) dropped events are primarily examined to demonstrate core protection. The dropped RCCA events include:

a. A dropped RCCA
b. A dropped RCCA bank.

Typically the Statically Misaligned RCCA event is grouped here as well. However, the Statically Misaligned RCCA safety analysis remains unchanged from that performed in support of the VANTAGE 5 RTSR, as noted in Section 3.3.1 of this report. Thus, the current licensing basis analysis for the Statically Misaligned RCCA event, as presented in the VANTAGE 5 RTSR and the Donald C. Cook Nuclear Plant Unit 2 UFSAR (Reference 9) remain applicable.

Since the VANTAGE 5 RTSR effort, the licensing-basis analysis for the Dropped RCCA(s) event(s) has not credited protection provided by the negative flux rate trip. This methodology (Reference 8) has been approved by the NRC and has been utilized for Unit 2 for the past several fuel cycles. The discussion of the Dropped RCCA(s) event(s) is not presented in this report, as it is already bounded by the Unit 2 licensing basis (see Section 14.1.3 of the Unit 2 UFSAR).

3.3.4.3 Loss of Normal Feedwater

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The reactor trip on low-low water level in any steam generator provides the necessary protection against a loss of normal feedwater.

The auxiliary feedwater system is started automatically. The turbine driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor driven auxiliary feedwater pumps are supplied by power from the diesel generators if a loss of offsite power occurs. The pumps take suction directly from the condensate storage tank for delivery to the steam generators. An evaluation (Reference 12) to support an increase in the "as found" steam generator safety valve setpoint tolerance to ±3% has been performed (and licensed by the NRC via the SER for Technical Specification Amendment 167) since the VANTAGE 5 RTSR program.

An evaluation of the system transient has been performed as part of this Uprating Program to support an increase in the "as found" pressurizer safety valve (PSV) setpoint tolerances to ±3%. The evaluation concluded that the increase in the "as found" pressurizer safety valve setpoint tolerance to ±3% will not adversely impact the plant response for this event. Specifically, the pressurizer pressure during this analysis does not increase to a value that could be influenced by the PSV setpoint tolerance relaxation, since the pressurizer power operated relief valves (PORVs) are assumed operable during this transient to maximuze the pressurizer water volume (i.e., modeling the pressurizer PORVs maximizes insurge). The overpressure transient for this event due to the PSVs lifting 3% higher than their setpoint is addressed via the Loss of Load analysis (see Section 3.3.5.2 of this report), since the turbine trip for the Loss of Normal Feedwater transient occurs after reactor scram, which provides additional heat removal via steam flow to the turbine, thereby providing a benefit with respect to peak paressure behavior. Thus, the auxiliary feedwater system is capable of returning the plant to a safe condition by removing the stored and residual heat, thereby preventing either overpressurization of the RCS or uncovery of the core. Therefore, the conclusions presented in the Unit 2 UFSAR remain valid.

3.3.4.4 Loss of All AC Power to the Station Auxiliaries

The loss of all AC power to the station auxiliaries event, as with the loss of normal feedwater incident, is a limiting transient with respect to pressurizer overfill. The deviation in the pressurizer safety valve behavior from that modeled in the current licensing basis analysis for

this event due to the increase in the setpoint tolerance to ±3% could potentially aggravate the transient and increase the potential for pressurizer filling. As such, the loss of all AC power to the station auxiliaries was evaluated for the Uprating Program.

A complete loss of all (non-emergency) AC power (e.g., offsite power) may result in the loss of all power to the plant auxiliaries, i.e., the RCPs, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system.

This transient is analyzed to show the adequacy of the heat removal capability of the auxiliary feedwater system. The transient more severely reduces the core heat removing capability than the loss of load or loss of normal feedwater events analyzed because, in this case, the decrease in heat removal by the secondary system is accompanied by a RCS flow coast down, which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip due to: (1) turbine trip; (2) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or (3) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of power with turbine and reactor trips, the sequence described below will occur:

a. Plant vital instruments are supplied from emergency DC power sources.

b. As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for steam dump. If the steam flow rate through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat contained in the fuel and coolant plus the residual decay heat produced in the reactor. An existing evaluation (Reference 12) for this event, which was performed after the VANTAGE 5 RTSR and was subsequently approved by the NRC via the SER for Technical Specification Amendment 167, supports an increased "as found" steam generator safety valve setpoint tolerance of ±3%. This evaluation continues to support the uprated conditions defined previously and is currently reflected in the Unit 2 UFSAR.

c. As the no load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition.

d. The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply plant vital loads.

The motor driven auxiliary feedwater pumps are supplied power by the diesels and the turbine-driven pump utilizes steam from the main steam system. Both type pumps are designed to supply rated flow within 60 seconds of the initiating signal even if a loss of all non-emergency AC power occurs simultaneously with loss of normal feedwater. The turbine exhausts the used steam to the atmosphere. The auxiliary feedwater pumps take suction from the condensate storage tank for delivery to the steam generators.

Following the RCP coastdown caused by the loss of AC power, the natural circulation capability of the RCS will remove decay heat from the core, aided by auxiliary feedwater in the secondary system. An evaluation has concluded that the increase in the "as found" pressurizer safety valve setpoint tolerance to $\pm 3\%$ will not adversely impact the plant response. Specifically, the pressurizer pressure during this analysis does not increase to a value that could be influenced by the PSV setpoint tolerance relaxation, since the pressurizer power operated relief valves (PORVs) are assumed operable during this transient to maximize the pressurizer water volume (i.e., modeling the pressurizer PORVs maximizes insurge). The overpressure transient for this event due to the PSVs lifting 3% higher than their setpoint is addressed via the Loss of Load analysis (see Section 3.3.5.2 of this report), since the turbine trip for the Loss of All AC Power to the Station Auxiliaries transient occurs after reactor scram, which provides additional heat removal via steam flow to the turbine, thereby providing a benefit with respect to peak pressure behavior.

3.3.4.5 Rupture of a Steam Pipe

A rupture of a steam pipe results in an uncontrolled steam release from a steam generator. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (MTC), the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential concern mainly because of the high hot channel factors which exist when the most reactive assembly is assumed stuck in its fully withdrawn position. The core is ultimately shut down by boric acid delivered by the ECCS.

The current Unit 2 licensing basis steam pipe rupture safety analysis, which was performed as part of the VANTAGE 5 RTSR program, demonstrates that:

- Assuming a stuck assembly, with or without offsite power, and assuming a single failure in the engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

It should be noticed that the existing rupture of a steam pipe analysis-of-record supports:

a. End-of-life shutdown margin of 1.3 %Δk/k at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. The relaxation of the shutdown margin for Unit 2 was included as part of the Cook Unit 1 30% SGTP Program Licensing Submittal (Reference 17).

b. Minimum capability for the injection of boric acid solution corresponding to the most restrictive single failure in the safety injection system. The ECCS consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the high head safety injection system (referred to as the "intermediate head safety injection system" in the VANTAGE 5 RTSR), and 4) the charging system (referred to as the "high head safety injection system" in the VANTAGE 5 RTSR). Only the charging system and the passive accumulators are modeled for the Rupture of a Steam Pipes analysis for core response. Centrifugal Charging pump head degradation of 10% was also assumed.

Since the VANTAGE 5 RTSR effort, three evaluations have been performed which affect this event. These evaluations are described below:

- A reduction in the safety analysis value for the Low Pressurizer Pressure setpoint that generates a SI signal from 1815 psia to 1715 psia was conducted (References 13 and 18.

- An internal Westinghouse study that reviewed the potential impact of the operation of the Centrifugal Charging Pump Minimum Flow Isolation Valves on the SI flow delivered during the transient. These valves are assumed to close following the receipt of a SI signal and reopen when RCS pressure rises above 2000 psig. The SI flow rates assumed in the steamline break analysis correspond to the Centrifugal Charging Pump Minimum Flow Isolation Valves being in the closed position.

- Revised values for the maximum total auxiliary feedwater flow (AFW) being delivered to the steam generators have been calculated and are slightly higher than those considered in the analysis-of-record. The maximum AFW flow rates are based upon a worst-case, flow retention failure. The evaluation considered a maximum AFW flow rate of 1905 gpm, conservatively directed to the faulted steam generator. Although there are other AFW configurations that yield a slightly higher calculated total AFW flow rate, it is key to notice that the skewed distribution assumed in this latter evaluation yields the most conservative results. As such, the 1905 gpm maximum flow

rate configuration is applicable, as it corresponds to the most skewed flow distribution for those scenarios considering an AFW flow retention failure.

All three of the aforementioned evaluations, which continue to support the uprated conditions, concluded that the results presented in Section 14.2.5 of the Unit 2 UFSAR remain applicable. Therefore, the conclusions of UFSAR Section 14.2.5 continue to be valid and support the Donald C. Cook Unit 2 Uprating.

3.3.4.6 Major Rupture of a Main Feedwater Pipe

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Furthermore, a break in this location could preclude the subsequent addition of emergency feedwater to the affected steam generator. (A break upstream of the feedwater line check valve would affect the nuclear steam supply system only as a loss of normal feedwater.)

The severity of the feedwater line rupture transient depends upon a number of system parameters including break size, initial reactor power level, and credit taken for the functioning of various control and safety systems. Sensitivity studies have shown that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. Analyses have been performed at full power, with and without offsite power.

It should be noticed that the major rupture of a feedwater line analysis-of-record, which was performed as part of the VANTAGE 5 RTSR effort, supports:

a. End-of-life shutdown margin of 1.3 %Δk/k at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position.

b. Minimum capability for the injection of boric acid solution corresponding to the most restrictive single failure in the safety injection system. The ECCS consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the high head safety injection system, and 4) the charging system. Only the charging system, with an assumed pump head degradation of 10%, is modeled for this accident analysis.

A detailed evaluation was performed which included sensitivity cases using the LOFTRAN code. Sensitivity cases investigated the effects of increasing the pressurizer safety valve setpoint tolerance from ±1% to ±3%. These sensitivity cases also included the effects associated with the steam generator safety valve setpoint tolerance increasing from ±1% to ±3%, and modeling a reduction in the Low Pressurizer Pressure - SI setpoint from 1815 psia

to 1715 psia. These latter two sensitivities were previously evaluated in References 12 and 13, respectively.

It should also be noted that the operation of the Centrifugal Charging Pump Minimum Flow Isolation Valves was also considered in the evaluation performed as part of the Unit 1 Steam Generator Tube Plugging Program, as discussed in the previous paragraph. These valves are assumed to close following the receipt of a SI signal and reopen when RCS pressure rises above 2000 psig. The SI flow rates assumed in the feedline break analysis correspond to the Centrifugal Charging Pump Minimum Flow Isolation Valves being in the closed position. The evaluation concluded that the reduction in the SI flow rates between the pressures of 2000 psig and 2300 psig due to the automatic operation to open the Centrifugal Charging Pump Minimum Flow Isolation Valves would have an insignificant effect on the Unit 2 Major Rupture of a Feedwater Pipe safety analysis.

The evaluations discussed above have concluded that the results presented in the Donald C. Cook Nuclear Plant Unit 2 UFSAR for the Major Rupture of Main Feedwater Pipe event (UFSAR Section 14.2.8) continue to be applicable for the Uprating Program.

3.3.4.7 Steamline Break Mass/Energy Releases Outside Containment

The existing mass and energy (M/E) release calculations following a steamline break (SLB) outside containment were performed to support the range of conditions possible for the Rerating Program of Unit 1, as well as to position Unit 2 for a future uprating (i.e., 3600 MWt NSSS). The superheated mass and energy release data are documented in Reference 19. As such, the M/E releases are based upon a rated thermal power of 3600 MWt. The core reactivity parameters were chosen to conservatively maximize the reactivity feedback effects of the cooldown resulting from a blowdown of either Donald C. Cook Nuclear Plant Unit 1 or Unit 2.

The adjustment in the K_4 safety analysis value of the OPΔT setpoint equation (discussed in Section 3.3.2.1) does not impact the SLB M/E Release Outside Containment analysis, which is the only non-LOCA safety analysis that relies on this trip function for primary protection. This is because a conservatively larger K_4 value of 1.18, which delays reactor trip, was assumed in the SLB M/E Release Outside Containment analysis that was performed as part of the Rerating Program, as discussed in the previous paragraph. (The revised safety analysis K_4 value for Unit 2 is 1.170.) Therefore, it can be concluded that the current licensing basis outside containment SLB M/E releases , which are presented in Section 14.4.11.3 of the Donald C. Cook Nuclear Plant.Unit 1 UFSAR, continue to bound Unit 2 operation as defined by the Uprating Program.

It is key to notice that the existing outside containment SLB M/E release analysis-of-record became part of the Cook Nuclear Plant Unit 1 and Unit 2 licensing basis following the approval of the Boron Injection Tank (BIT) Removal submittal (Reference 11). The existing

SLB M/E Release Outside Containment analysis was performed as part of the program to support rerating both Cook units (Reference 16). The analysis assumed:

a. End-of-life shutdown margin of 1.3 %Δk/k at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position.

b. Minimum capability for the injection of boric acid solution corresponding to the most restrictive single failure in the safety injection system. The ECCS consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the high head safety injection system (referred to as the "intermediate head safety injection system" in Reference 16), and 4) the charging system (referred to as the "high head safety injection system" in Reference 16). Only the charging system and the passive accumulators are modeled for the steamline break accident analysis for M/E releases outside containment. Centrifugal Charging pump head degradation of 10% was also assumed.

c. Coincidence logic required for safety injection and steamline isolation consistent with the originally installed Unit 1 steamline break protection system (which is referred to as "OLD" Steamline Break Protection in Section 3.5.4.1 of Reference 14). However, the current analysis, which assumes the "OLD" steamline break protection system, bounds the Unit 2 steamline break protection system (which is referred to as "HYBRID" Steamline Break Protection in Section 3.5.4.1 of Reference 14). A detailed description of the two steamline break protection systems is presented on pages 3.5-18 through 3.5-20 of Reference 14.

Thus, it can be seen that the existing mass and energy release calculations following a steamline break located outside containment support operation of Unit 2 under the conditions defined for this Uprating Program (Table 3.3-1 of this report).

3.3.5 Non-LOCA Safety Evaluation: Transients Analyzed

The subsections that follow contain the details of the accidents re-analyzed to support operation of Unit 2 under uprated conditions. In all cases, the applicable UFSAR acceptance criteria are satisfied.

3.3.5.1 Locked Rotor Accident (Peak Pressure Case Only)

A transient analysis has been performed for the instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools

down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer causes a pressure increase which in turn actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in a sequence dependent on the rate of insurge and pressure increase. The power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray are not included in this analysis.

The locked rotor event is examined with respect to peak RCS pressure, Rods-in-DNB, and peak cladding temperature. The increase in the "as found" pressurizer safety valve setpoint tolerance to ±3%, which is being performed as part of this Uprating Program, is non-conservative with respect to the peak RCS pressure evaluation. As such, the locked rotor analysis for peak RCS pressure was re-analyzed. Since no changes are being implemented which would adversely impact the Locked Rotor Rods-in-DNB or peak cladding temperature analyses, these two aspects of the Locked Rotor event need not be reanalyzed. Therefore, the Locked Rotor Rods-in-DNB and peak cladding temperature evaluations performed in support of the VANTAGE 5 RTSR, which support the uprated power level and associated operating conditions, remain applicable to Unit 2 and are not discussed further in this report.

<u>Method of Analysis</u>

The LOFTRAN digital-computer code is used to analyze the peak RCS pressure transient for this event. The LOFTRAN code is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak RCS pressure.

The analysis is performed based on the limiting conditions associated with the Uprating Program, and support the relaxation of the "as found" PSV setpoint tolerance to ±3%. As in the previous UFSAR analysis, the analysis assumes that offsite power is available following the reactor trip and turbine trip.

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion begins 1 second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or main feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak RCS pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are assumed to initially open at 2575 psia and achieve rated flow at 2580 psia. This analysis assumed an initial pressurizer pressure of 2312.6 psia. Table 3.3-5 presents the initial conditions assumed for the peak pressure transient.

Results

The transient results for the locked rotor accident are shown in Figures 3.3-5 and 3.3-6. The peak RCS pressure (2645 psia) reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits (this peak pressure is also below 110% of the design pressure). The pressure response shown in Figure 3.3-6 is the response at the point in the Reactor Coolant System having the maximum pressure. The sequence of events is included in Table 3.3-6.

Conclusions

Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted conditions stress limits, the integrity of the primary coolant system is not endangered.

3.3.5.2 Loss of External Electrical Load and/or Turbine Trip

The complete loss of steam load from full power is examined primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection. The increase in the "as found" pressurizer safety valve setpoint tolerance to ±3%, which is being performed as part of this Uprating Program, is non-conservative with respect to the NSSS pressure response to this transients. Expected protection signals for this event are from the high pressurizer pressure, OTΔT, high pressurizer water level, and low-low steam generator water level reactor trips.

The loss of external electrical load may result from an abnormal variation in network frequency or other adverse network operating conditions. It may also result from a trip of the turbine generator, or in an unlikely opening of the main breaker from the generator which fails to cause a turbine trip but causes a rapid/large NSSS load reduction by the action of the turbine control.

Method of Analysis

The loss of load transients are analyzed by employing the detailed digital computer program LOFTRAN. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

An analysis is performed to bound the conditions of the Uprating Program. Nominal values are assumed for the initial reactor power, temperature, and pressure, since this accident is analyzed with the RTDP. Plant characteristics and initial conditions are listed in Table 3.3-5.

Major assumptions are summarized below:

a. Initial Operating Conditions - nominal conditions for reactor power, pressure, and RCS temperatures are assumed for statistical DNB analyses.

b. Moderator and Doppler Coefficients of Reactivity - the loss of load is analyzed with both maximum and minimum reactivity feedback. The maximum feedback cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback cases assume a positive moderator temperature coefficient and the least negative Doppler coefficients.

c. Reactor Control - from the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control. If the reactor was in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

d. Pressurizer Spray and Power-Operated Relief Valves - two cases for both the minimum and maximum moderator feedback cases are analyzed:

   1. Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available, with a -3% tolerance on the opening set pressure.

   2. No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable, with a +3% tolerance on the opening set pressure.

e. Steam Release - no credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through the safety valves limits the secondary steam pressure at the setpoint value.

f. Feedwater Flow - main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

g. Reactor trip is actuated by the first Reactor Protection System trip setpoint reached. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high pressurizer water level, and low-low steam generator water level.

Results

The transient responses for a loss of load from full power operation are shown for four cases: minimum and maximum reactivity feedback, with and without pressure control (Figures 3.3-7 through 3.3-26).

Figures 3.3-7 through 3.3-11 show the transient responses for the loss of load with minimum reactivity feedback assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip signal.

The minimum DNBR remains well above the limit value. The pressurizer safety valves are actuated for this case to limit primary system pressure. The steam generator safety valves prevent overpressurization of the secondary system, maintaining pressure below 110 percent of design value.

Figures 3.3-12 through 3.3-16 show the responses for the total loss of steam load with maximum reactivity feedback assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. The DNBR increases throughout the transient and never drops below its initial value. Pressurizer relief valves and steam generator safety valves prevent overpressurization in primary and secondary systems, respectively. The reactor is tripped by the low-low steam generator water level signal. The pressurizer safety valves are not actuated for this case.

In the event that feedwater flow is not terminated at the time of turbine trip for this case, flow would continue under automatic control with the reactor at a reduced power. The operator would take action to terminate the transient and bring the plant to a stabilized condition. If no action was taken by the operator, the reduced power operation would continue until the condenser hotwell was emptied. A low-low steam generator water level reactor trip would be generated along with auxiliary feedwater initiation signals. Auxiliary feedwater would then be used to remove decay heat with the results less severe than those presented in Section 3.3.4.3, Loss of Normal Feedwater.

The loss of load accident was also studied assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 3.3-17 through 3.3-21 show the transient responses with minimum reactivity feedback. The neutron flux remains essentially constant at full power until the reactor is tripped. The DNBR never goes below its initial value throughout the transient. In this case, the pressurizer

safety valves are actuated and maintain system pressure below 110 percent of the design value.

Figures 3.3-22 through 3.3-26 show the transient responses with maximum reactivity feedback with the other assumptions being the same as in the preceding case. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated to limit primary pressure.

The sequence of events following each of these transients is included in Table 3.3-7.

Conclusions

Results of the analyses show that the plant design is such that a loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits. The integrity of the core is maintained by operation of the Reactor Protection System, i.e., the DNBR will be maintained above the safety limit value.

3.3.5.3 Excessive Heat Removal due to Feedwater System Malfunctions (HFP Cases Only)

Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The high neutron flux, overpower ΔT, and overtemperature ΔT trips prevent any power increase which could lead to DNBR less than the minimum allowable value in the event that the steam generator high-high water level protection function has not been actuated.

Excessive feedwater flow may be caused by full opening of a feedwater control valve due to a Feedwater Control System malfunction or an operator error. At power conditions, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

The excessive heat removal due to Feedwater System Malfunctions are examined primarily to demonstrate core protection. For the Uprating Program, a review of the licensing basis analyses has been performed. The analyses performed in support of the VANTAGE 5 RTSR Program continue to be the analyses of record, with the exception of the "Feedwater System Malfunctions Causing an Increase in Feedwater Flow" scenario which is performed at hot full power (HFP) conditions (see Section B.3.8B.2 of Reference 1). (Hereafter, this accident will be referred to as the HFP Feedwater Malfunction (FWM) event.) Since the VANTAGE 5 RTSR submittal, the HFP FWM event has been re-analyzed for both Cook units to address an

increase in the feedwater flow rate from 150% to 200% of nominal full power feed flow. This re-analysis also considered an initiating failure that could result in the full opening of one or more feedwater control valves, under uprated conditions (i.e., the same parameters defined in Cases 1 through 4 of Table 3.3-1), with the reactor being operated under both automatic and manual rod control conditions. A safety review of this HFP FWM re-analysis was performed by AEPSC (Reference 20) prior to the work being incorporated into the UFSARs for the Cook units.

The Cook Unit 1 and Unit 2 UFSARs were updated in 1993 to reflect this HFP FWM re-analysis effort. Thus, Section 14.1.10B of the current Cook Unit 2 UFSAR presents two FWM cases for a full VANTAGE 5 core at HFP conditions; Case 1a, which is the accidental full opening of one feedwater control valve assuming automatic and manual rod control, and Case 1b, which is the accidental full opening of all four feedwater control valves assuming automatic and manual rod control. Both of these cases assumed a feedwater flow rate of 200% of nominal full power feed flow being delivered to the affected steam generator(s).

Since a complete discussion of the reanalysis is already part of the Cook Unit 2 licensing basis, further discussions will not be presented herein. However, it is noted that the Cook Unit 2 UFSAR demonstrates that all acceptance criteria are met for the reanalyzed HFP FWM event under uprated conditions.

3.3.5.4 Mass and Energy Releases Analysis for Postulated Secondary System Pipe Ruptures Inside Containment

The mass and energy releases following a steamline break inside containment were recalculated as part of the Cook Unit 1 30% steam generator tube plugging program and the results are applicable to both Cook units. A bounding analysis was performed to address the range of conditions applicable for both Unit 1 and the future uprated conditions for Unit 2.

The non-LOCA discussion regarding the reanalysis of the steamline break mass and energy releases inside containment can be found in Section 3.5.4 (and Section 3.5.5) of the Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report (Reference 14), which has been submitted for NRC approval via Reference 17.

3.3.6 Conclusions of the Non-LOCA Safety Evaluation

The non-LOCA safety analyses and evaluations discussed in this section support the operation of Donald C. Cook Nuclear Plant Unit 2 under uprated conditions, as described in Table 3.3-1 (Cases 1 through 4).

References

1. VANTAGE 5 Reload Transition Safety Report for Donald C. Cook Nuclear Plant Unit 2, Revision 1, March 1990.

2. Friedland, A. J., Ray, S., "Revised Thermal Design Procedure," WCAP-11397-A, April, 1989.

3. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June, 1972.

4. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1, 1984.

5. Risher, D. H., Jr., and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-8028-A, January, 1975.

6. Friedland, A. J. and Ray, S., "Improved THINC-IV Modeling for PWR Core Design," WCAP-12330-P, August 1989.

7. Ellenberger, S. L., et al., "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," WCAP-8746, March, 1977.

8. Haessler, R. L., et al., "Methodology for the Analysis of the Dropped Rod Event," WCAP-11394-P-A and WCAP-11395-NP-A, January 1990.

9. Donald C. Cook Nuclear Plant Unit 2 Updated Final Safety Analysis Report, USNRC Docket Number 50-316, updated through July 1995.

10. Christensen, J. A., et al., "Melting Point of Irradiated Uranium Dioxide," Transactions of the American Nuclear Society, 7, 1964.

11. Letter from William O. Long, Sr. (USNRC) to Eugene E. Fitzpatrick (AEPSC), Subject: Amendment Nos. 158 and 142 to Facility Operating License Nos. DPR-58 and DPR-74 (TAC Nos. 80262 and 80263), dated November 20, 1991.

12. "Donald C. Cook Units 1 and 2 Main Steam Safety Valve Lift Tolerance Relaxation," SECL-91-429, Revision 2, dated December 1993.

13. "Donald C. Cook Nuclear Plant Unit 2 Low Pressurizer Pressure Safety Injection SAL Decrease," SECL-93-193, dated August 1993.

14. "Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report," WCAP-14285, Revision 1, dated May 1995.

15. AEP1-W/0315 and AEP2-W/0157 dated December 31, 1992, with attachment titled "Evaluation of Impact of Potential Cold Leg Temperature Gradient on Cook Nuclear Plant Operation".

16. Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 & 2 Licensing Report, WCAP-11902 Supplement 1, dated September 1989.

17. AEP:NRC:1207, "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," dated May 26, 1995.

18. Sharma, R. S., "Memo to AEPSC SBICI Readiness Review," dated August 16, 1993.

19. Donald C. Cook Nuclear Plant Units 1 and 2 Rerating Engineering Report, Appendix V: "Steam Line Break Mass and Energy Release Outside Containment," WCAP-12135, Appendices, Volume 2, dated September 1989.

20. Roulett, G. P., "AEPSC Safety Review, 50.59 Evaluation for Feedwater Malfunction FSAR Update Impact of PR 91-290 on the Cook Units 1 and 2 Analysis of Record," dated April 15, 1993.

TABLE 3.3-1

DONALD C. COOK NUCLEAR PLANT UNIT 2 NSSS PERFORMANCE PARAMETERS USED IN NON-LOCA SAFETY ANALYSES

Parameter	Case 1	Case 2	Case 3	Case 4
NSSS Power, MWt	3600	3600	3600	3600
Core Power, MWt	3588	3588	3588	3588
RCS Flow, gpm/loop [1]	88500	88500	88500	88500
Minimum Measured Flow, total gpm [2]	366,400	366,400	366,400	366,400
RCS Temperature, °F				
Core Outlet	585.7	618.2	585.8	618.4
Vessel Outlet	582.2	615.0	582.3	615.2
Core Average	550.1	584.9	550.1	584.8
Vessel Average	547.0	581.3	547.0	581.3
Vessel/Core Inlet	511.8	547.6	511.7	547.3
Steam Generator Outlet	511.5	547.4	511.4	547.1
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2100	2100	2250	2250
Steam Pressure, psia	587	820	587	820
Steam Flow (10^6 lb/hr total)	15.90	16.00	15.90	16.00
Feedwater Temp., °F	449	449	449	449
SG Tube Plugging, %	10% avg/ 15% peak	10% avg/ 15% peak	10% avg/ 15% peak	10% avg/ 15% peak

(1) RCS Flow (Thermal Design Flow, or TDF) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based upon this flow.

(2) Minimum Measured Flow - The flow specified in the Technical Specifications which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Revised Thermal Design Procedure. MMF is based upon the TDF plus a 3.5% flow measurement uncertainty.

TABLE 3.3-2

RPS TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN NON-LOCA ACCIDENT ANALYSIS[c]

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delay (Seconds)
Power range high neutron flux, high setting	118 percent	0.5
Power range high neutron flux, low setting	35 percent	0.5
Overtemperature ΔT	Variable, see Figure 3.3-1 through 3.3-4 and Table 3.3-4	8.0[a]
Overpower ΔT	Variable, see Figure 3.3-1 through 3.3-4 and Table 3.3-4	8.0[d]
High pressurizer pressure	2428 psig	2.0
Low pressurizer pressure	1907 psig	2.0
High pressurizer water level	100% NRS	2.0
Low reactor coolant flow (From loop flow detectors)	87 percent loop flow	1.0
Undervoltage trip	b	1.5
Low-low steam generator water level	0.0 percent of narrow range level span	2.0

---

[a] Total time delay (including RTD bypass loop fluid transport delay effect, bypass loop piping thermal capacity, RTD time response, and trip circuit, channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall. The time delay assumed in the analysis supports the 6 second response time of the RTD time response, trip circuit delays, and channel electronics delay presented in the Technical Specifications.

[b] No explicit value assumed in the analysis. Undervoltage trip setpoint assumed reached at initiation of analysis.

[c] The control rod scram time to dashpot is 2.7 seconds.

[d] Overpower ΔT reactor trip was assumed in the steamline break mass/energy release outside containment calculations.

TABLE 3.3-3

ESF TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN NON-LOCA ACCIDENT ANALYSIS

ESF Actuation System	Limiting Trip Point Assumed In Analysis	Time Delay (Seconds)
Safety Injection (SI)		
- Low pressurizer pressure	1700 psig	27 w/ offsite power[1] 37 w/out offsite power[2]
- Low steamline pressure	344 psig	27 w/ offsite power[1] 37 w/out offsite power[2]
Auxiliary Feedwater (AFW)		
- Low-low steam generator water level	0.0 percent of narrow range level span	60 [3]
High steam generator water level Turbine Trip	82 percent of narrow range level span	2.5
Steamline Isolation on low steamline pressure	Not applicable	11.0[4]
Feedwater Isolation on high steam generator water level	Not applicable	11.0[5]
Feedwater Isolation on low steamline pressure	Not applicable	8.0[5]

[1] Diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

[2] Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

[3] For the Loss of Normal Feedwater and Loss of Offsite Power to Station Auxiliaries events, the delay time assumed is 60 seconds from the generation of the signals.

For the Feedline Break event, the delay time assumed is 600 seconds (10 minute operator action delay) from the initiation of the break.

[4] Steamline isolation total delay time includes valve closure time, electronics and sensor delay. Technical Specifications require 8.0 second valve closure time.

[5] Feedline isolation total delay time includes valve closure time and electronics and sensor delay time.

TABLE 3.3-4

OTΔT AND OPΔT SETPOINT EQUATION AND SAFETY ANALYSIS LIMIT COEFFICIENT VALUES

Overtemperature ΔT equation:

$$OT\Delta T \leq \Delta T_o \left[K_1 - K_2 \left[\frac{1+\tau_1 s}{1+\tau_2 s}\right](T - T') + K_3 (P - P') - f_1(\Delta I)\right]$$

where,

K_1 = 1.37
K_2 = 0.0268
τ_1 = 22 seconds
τ_2 = 4 seconds
s = Laplace transform operator
T' = 547.0 to 581.3°F
K_3 = 0.00111
$f_1(\Delta I)$: Dead-band: from -16 to +6%ΔI

Positive Wing: 2.7%/%ΔI reduction in the ΔT trip setpoint for each percent ΔI > +6%ΔI

Negative Wing: 2.05%/%ΔI reduction in the ΔT trip setpoint for each percent ΔI < -16%ΔI

Overpower ΔT equation:

$$OP\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 s}{1+\tau_3 s}\right] T - K_6 (T - T'') - f_2(\Delta I)\right]$$

where,

K_4 = 1.170
K_5 = 0.02; this gain is not modeled in the non-LOCA safety analyses
τ_3 = 10 seconds
s = Laplace transform operator
T" = 547.0 to 576.0°F
K_6 = 0.00197
$f_2(\Delta I)$: = 0

TABLE 3.3-5
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Moderator Density (ΔK/gm/cc)	Reactivity Coefficients Assumed: Doppler	DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output (8) (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Uncontrolled Rod Cluster Assembly Bank Withdrawal from a Subcritical Condition	TWINKLE FACTRAN THINC-IV	Refer to Sec. B.3.1.2 of V-5 RTSR	NA	Min (9)	W-3/WRB-2	No	0	162,840	547	2037.0 (5)
Uncontrolled Rod Cluster Assembly Bank Withdrawal At Power (2)	LOFTRAN	+5	.54	Min (1) & Max (3)	WRB-2	Yes	3608 2165 361	366,400	581.3 567.6 550.4	2100.0
Rod Cluster Control Assembly Misalignment	LOFTRAN THINC-IV	NA	NA	NA	WRB-2	Yes	3600	366,400	581.3	2100.0
Uncontrolled Boron Dilution	NA	NA	NA	NA	NA	NA	3600 0	NA	NA	NA
Loss of Forced Reactor Coolant Flow	LOFTRAN FACTRAN THINC-IV	+5	NA	Max (3)	WRB-2	Yes	3608	366,400	581.3	2100.0
Locked Rotor (Peak Pressure)	LOFTRAN	+5	NA	Max (3)	NA	NA	3680	354,000	585.4	2312.6
Locked Rotor (Peak Cladding Temp)	LOFTRAN FACTRAN	+5	NA	Max (3)	NA	NA	3680	354,000	585.4	2037.0
Locked Rotor (Rods-In-DNB)	LOFTRAN FACTRAN THINC-IV	+5	NA	Max (3)	WRB-2	Yes	3608	366,400	581.3	2100.0

NA - Not Applicable
(1) Minimum Doppler only power coefficient (pcm/%power) = -9.55 + 0.03732Q, where Q is in % power.
(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.
(3) Maximum Doppler only power coefficient (pcm/% power) = -19.4 + 0.07176Q, where Q is in % power
(4) Zero Power Doppler defect at BOL and EOL assumed to be -965 pcm and -849 pcm, respectively.
(5) Core pressure.
(6) Full Power Doppler defect at BOL and EOL assumed to be -966 pcm and -893 pcm, respectively.
(7) Core thermal power.
(8) Includes reactor coolant pump heat, if applicable.
(9) Zero Power Doppler defect at BOL assumed to be -1081 pcm.

TABLE 3.3-5 (continued)
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Moderator Density (ΔK/gm/cc)	Doppler	Revised DNB Correlation	Initial NSSS Thermal Design Procedure	Reactor Thermal Power Output (8) (MWt)	Vessel Vessel Coolant Flow (GPM)	Average Temperature (°F)	Pressurizer Pressure (PSIA)
Loss of Electrical Load and/or Turbine Trip	LOFTRAN	+5	.54	Min (1) and Max (3)	WRB-2	Yes	3600	366,400	581.3	2100.0
Loss of Normal Feedwater	LOFTRAN	+5	NA	Max (3)	NA	NA	3680	354,000	585.4	2312.6
Excessive Heat Removal Due to Feedwater System Malfunction	LOFTRAN	NA	.54	Min (1)	WRB-2	Yes	3600 0	366,400	581.3 547	2100.0
Excess Load Increase Incident	LOFTRAN	NA	0 and .54	Min (1) and Max (3)	WRB-2	Yes	3600	366,400	581.3	2100.0
Loss of Offsite Power to the Station Auxiliaries	LOFTRAN	+5	NA	Max (3)	NA	NA	3680	354,000	541.4	2312.6
Rupture of a Steam Pipe	LOFTRAN THINC IV	See Figure B.3-55 of the V-5 RTSR	NA	See Figure B.3-56 of the V-5 RTSR	W-3	NA	0	354,000	547.0	2100.0
Rupture of a Control Rod Drive Mechanism Housing	TWINKLE FACTRAN	See Section B.3.12 of the V-5 RTSR	NA	(6) (4)	NA	NA	3660 (7) 0	354,000 162,840	585.4 547.0	2037.4 (5)
Rupture of Feedwater Pipe	LOFTRAN	NA	.54	Max (3)	NA	NA	3680	354,000	585.4	2162.6

NA - Not Applicable
(1) Minimum Doppler only power coefficient (pcm/%power) = -9.55 + 0.03732Q, where Q is in % power.
(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.
(3) Maximum Doppler only power coefficient (pcm/% power) = -19.4 + 0.07176Q, where Q is in % power
(4) Zero Power Doppler defect at BOL and EOL assumed to be -965 pcm and -849 pcm, respectively.
(5) Core pressure.
(6) Full Power Doppler defect at BOL and EOL assumed to be -966 pcm and -893 pcm, respectively.
(7) Core thermal power.
(8) Includes reactor coolant pump heat, if applicable.
(9) Zero Power Doppler defect at BOL assumed to be -1081 pcm.

TABLE 3.3-6

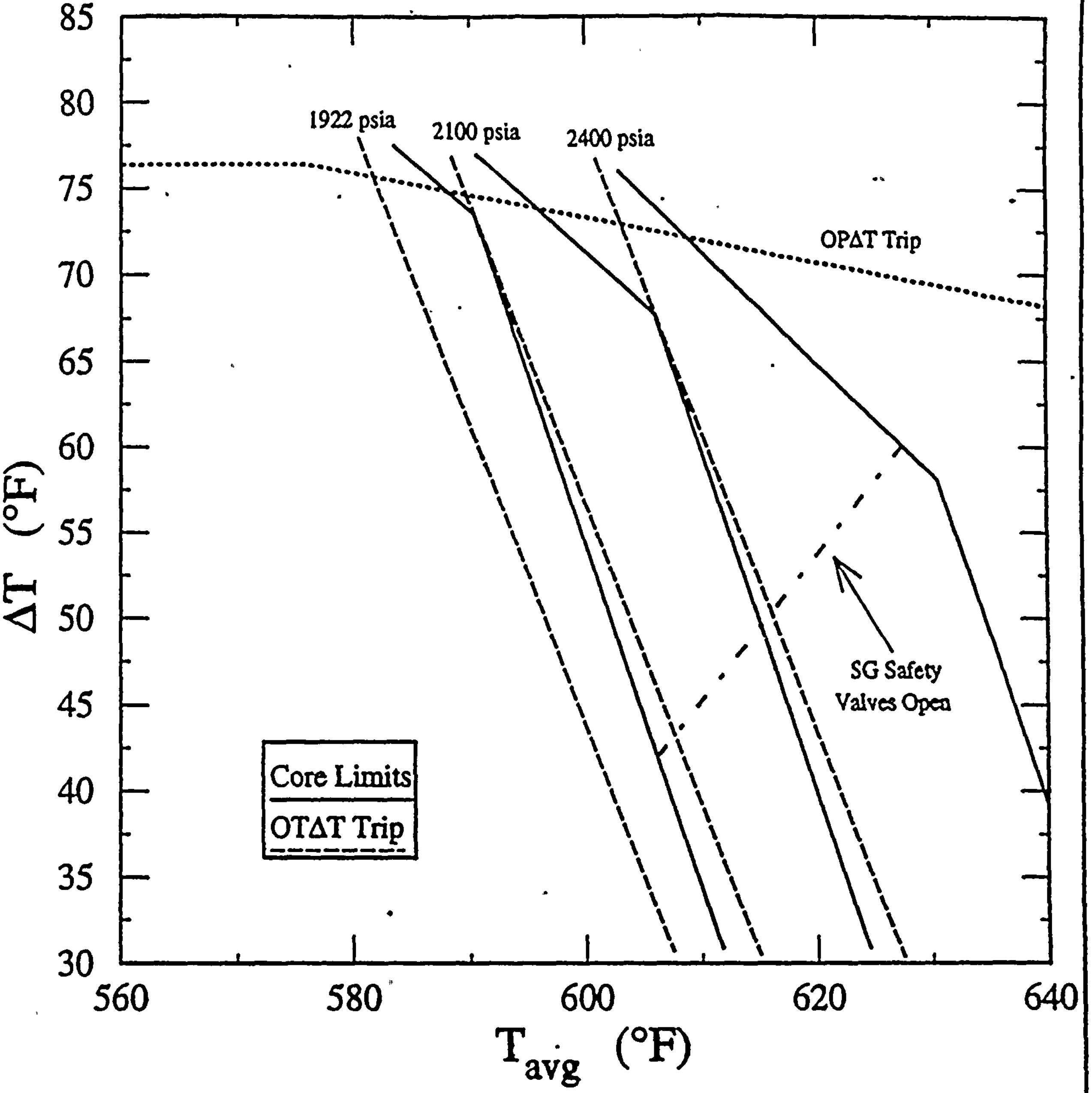
SEQUENCE OF EVENTS FOR THE
LOCKED ROTOR FOR PEAK PRESSURE ANALYSIS

Accident	Event	Time (sec.)
Locked Rotor	One pump rotor seizes	0.0
	Low reactor coolant flow trip setpoint reached in faulted loop	0.02
	Rods begin to drop	1.02
	Maximum RCS pressure occurs	3.10

TABLE 3.3-7
SEQUENCE OF EVENTS FOR LOSS OF EXTERNAL ELECTRICAL LOAD

Case	Event	Time (sec.)
Minimum Feedback with Pressure Control	Loss of external electrical load	0.0
	OTΔT reactor trip setpoint reached	12.2
	Peak RCS pressure occurs	12.5
	Rods begin to drop	14.2
	Minimum DNBR occurs	16.0
Maximum Feedback with Pressure Control	Loss of external electrical load	0.0
	Peak RCS pressure occurs	8.5
	Low-low steam generator water level reactor trip setpoint reached	53.7
	Rods begin to drop	55.7
	Minimum DNBR occurs	*
Minimum Feedback without Pressure Control	Loss of external electrical load	0.0
	High pressurizer pressure reactor trip setpoint reached	7.3
	Peak RCS pressure occurs	9.0
	Rods begin to drop	9.3
	Minimum DNBR occurs	*
Maximum Feedback without Pressure Control	Loss of external electrical load	0.0
	High pressurizer pressure reactor trip setpoint reached	7.4
	Rods begin to drop	9.4
	Peak RCS pressure occurs	9.5
	Peak RCS pressure occurs	*

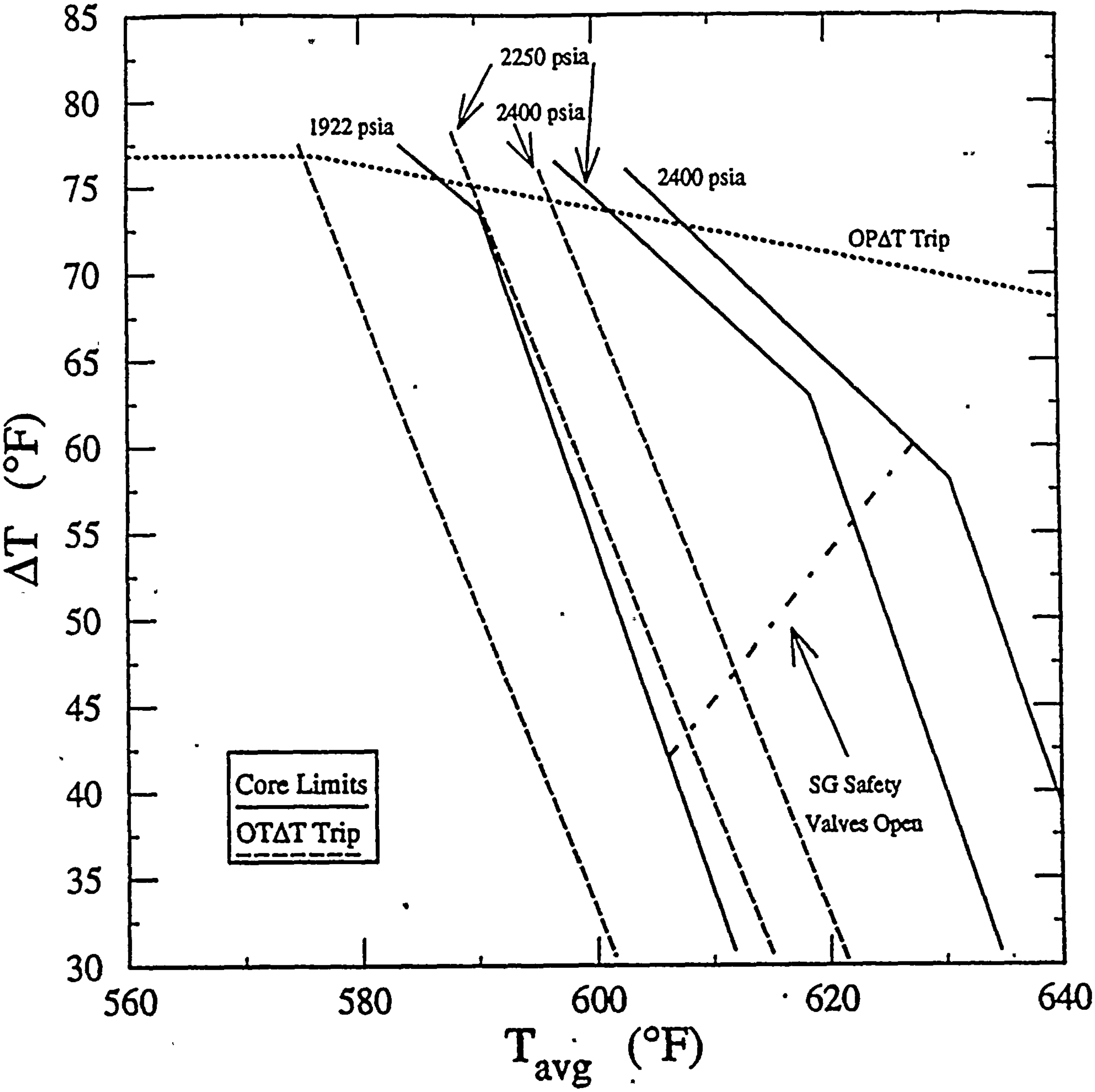
* - DNBR never decreases below its initial value



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-1

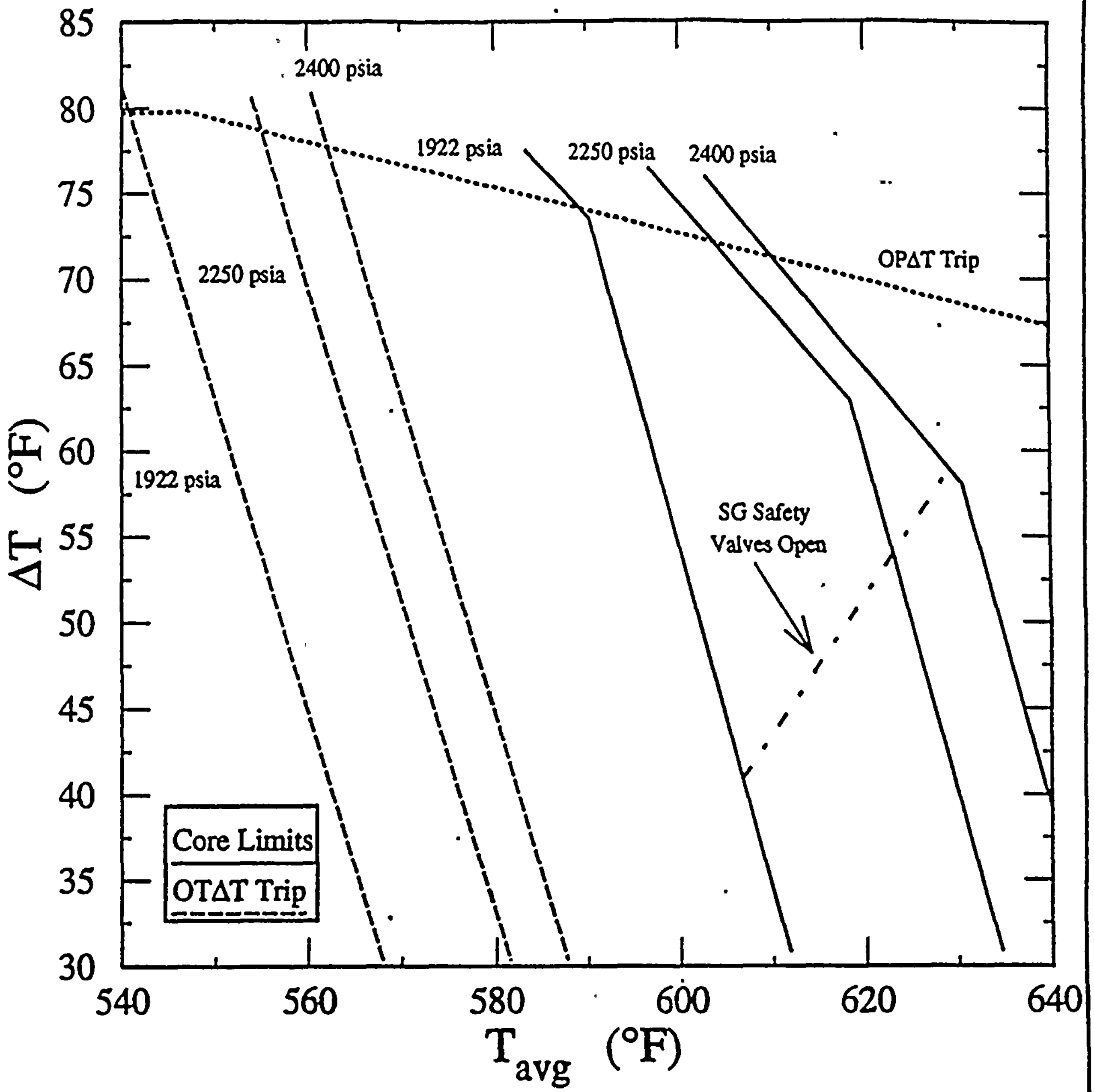
Illustration of Overtemperature and Overpower ΔT Protection
Nominal Tavg = 581.3°F
Nominal Pressure = 2100 psia



**DONALD C. COOK
NUCLEAR PLANT
UNIT 2**

FIGURE 3.3-2

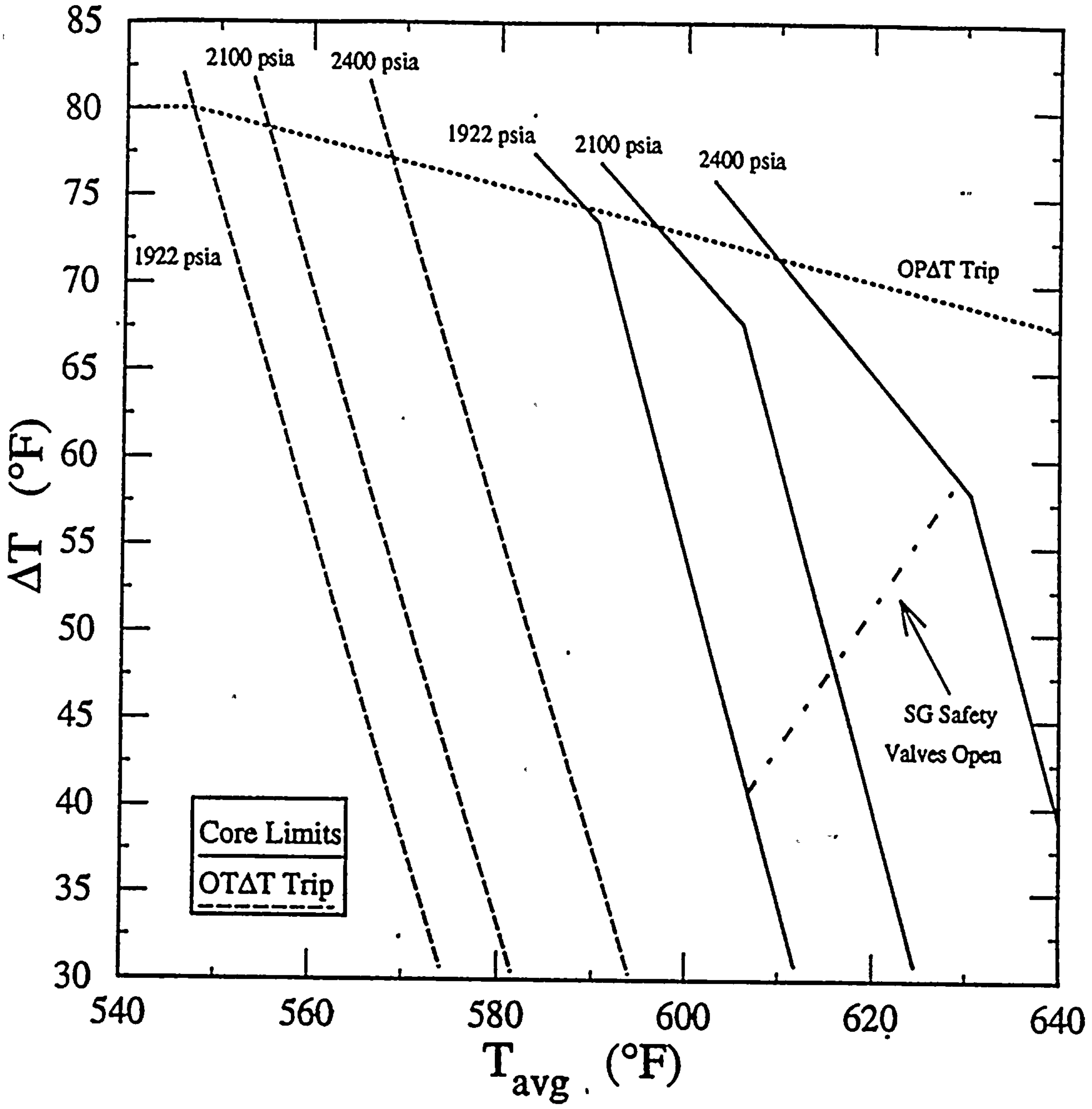
Illustration of Overtemperature and Overpower ΔT Protection
Nominal Tavg = 581.3°F
Nominal Pressure = 2250 psia



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-3

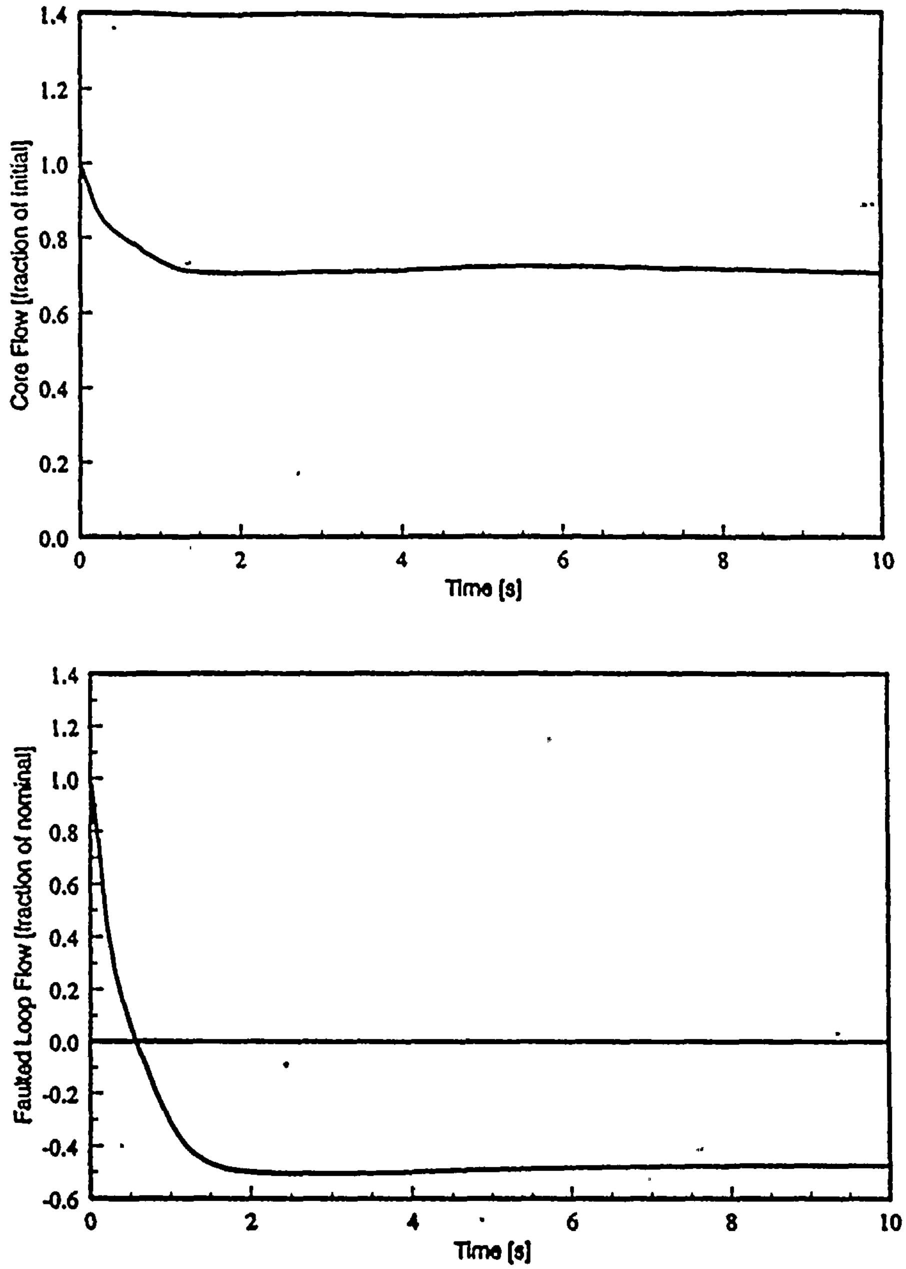
Illustration of Overtemperature and Overpower ΔT Protection
Nominal Tavg = 547.0°F
Nominal Pressure = 2250 psia



DONALD C. COOK NUCLEAR PLANT UNIT 2

FIGURE 3.3-4

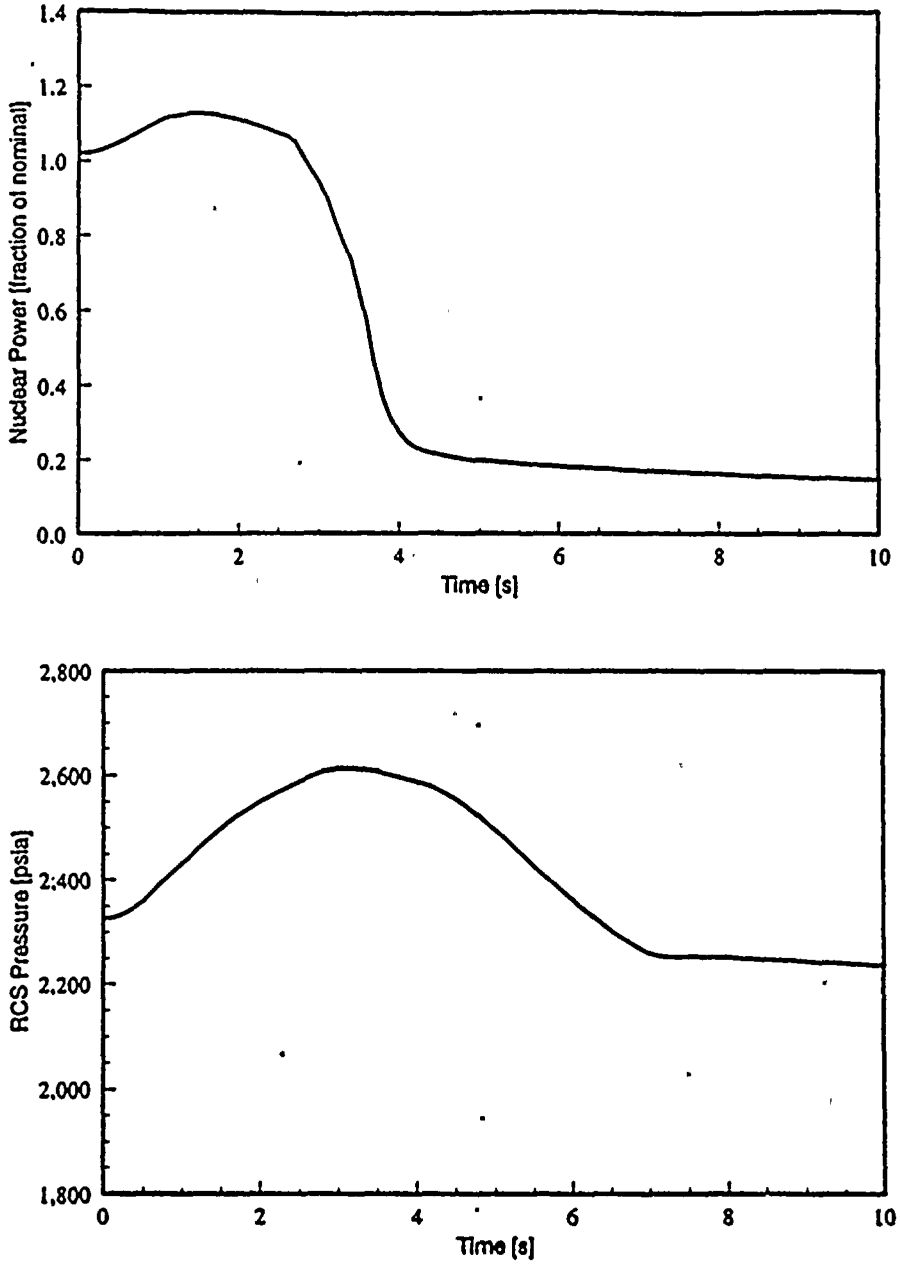
Illustration of Overtemperature and Overpower ΔT Protection
Nominal Tavg = 547.0°F
Nominal Pressure = 2100 psia



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-5

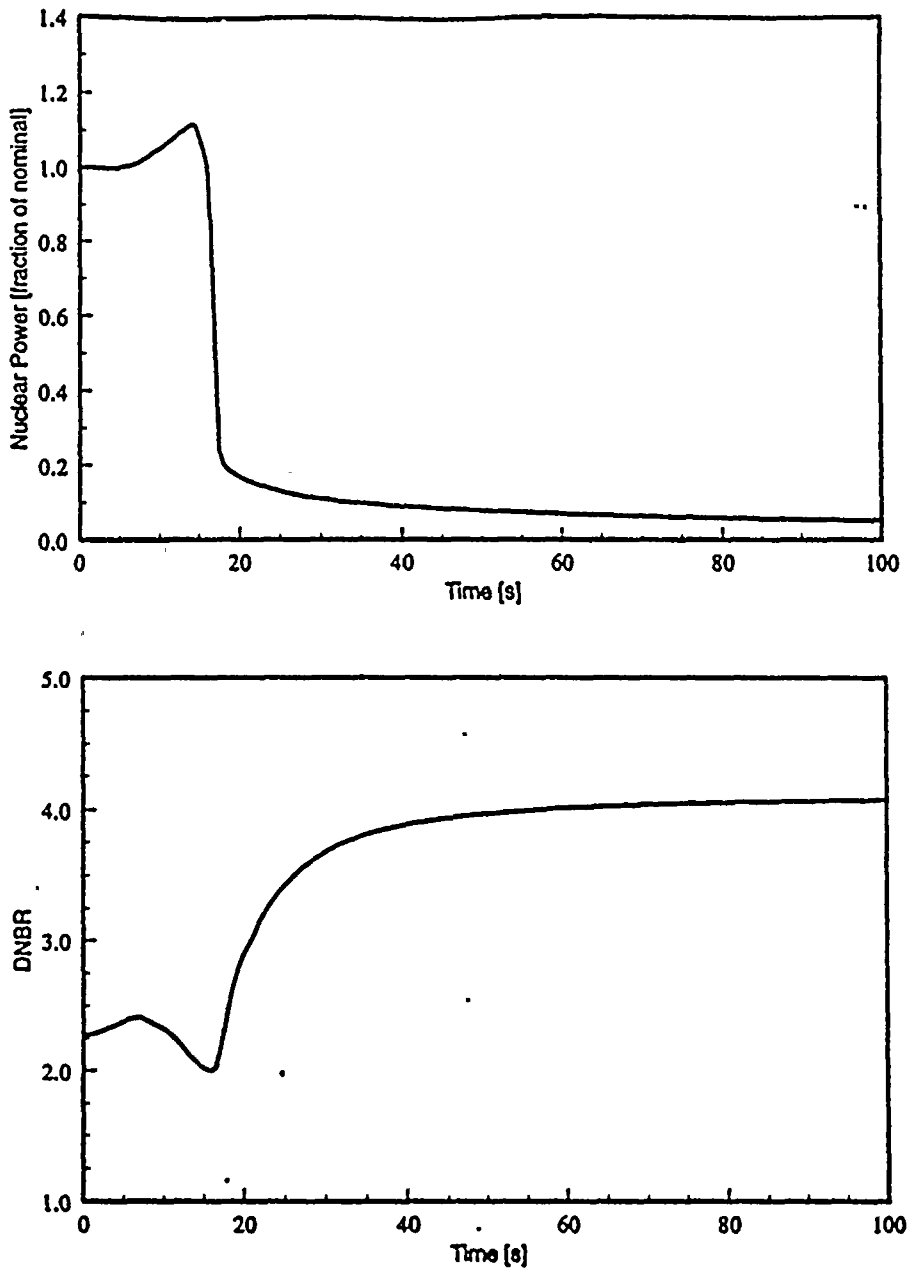
Total Core Flow and Faulted Loop Flow vs. Time
For The Locked Rotor Event



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-6

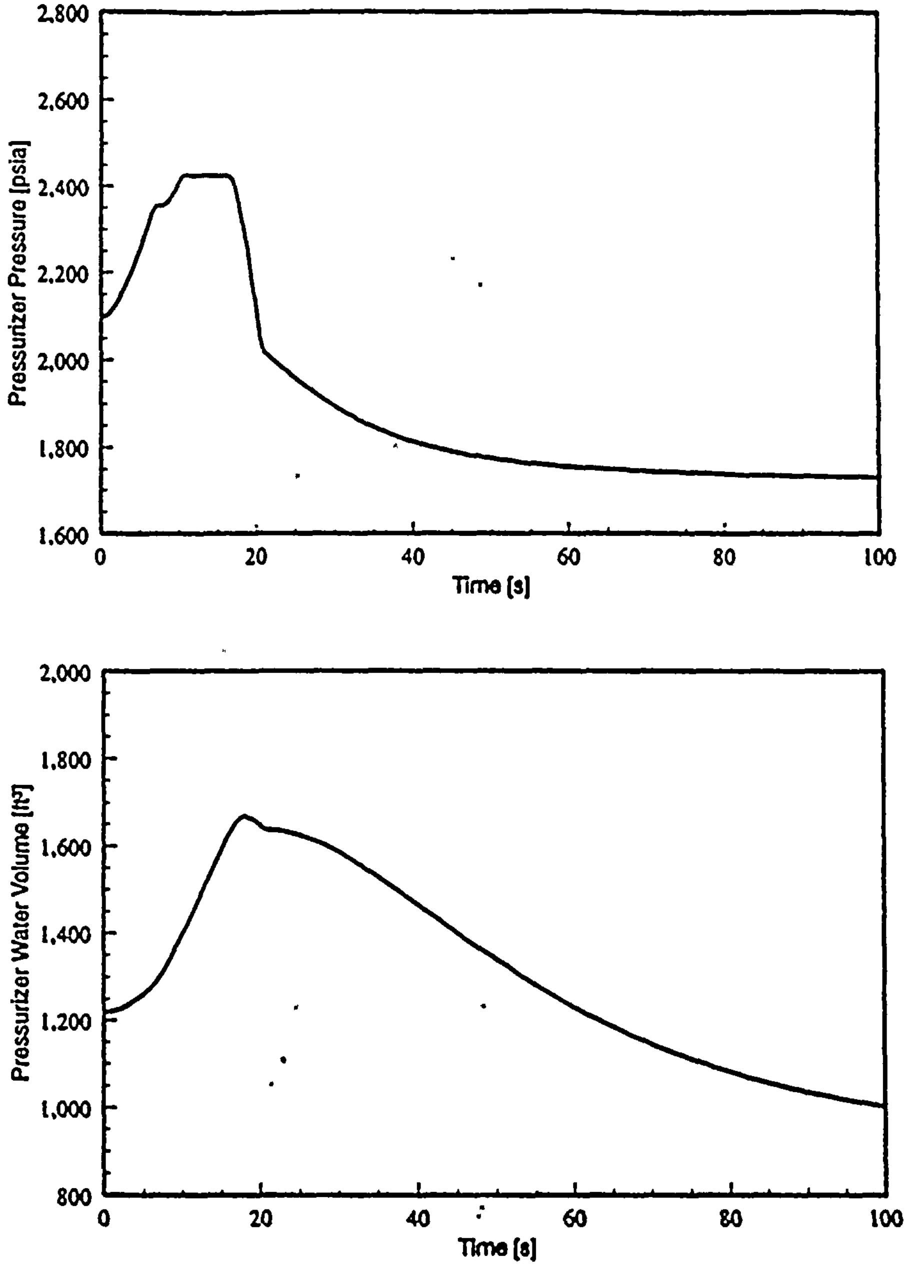
Nuclear Power and RCS Pressure vs. Time
For The Locked Rotor Event



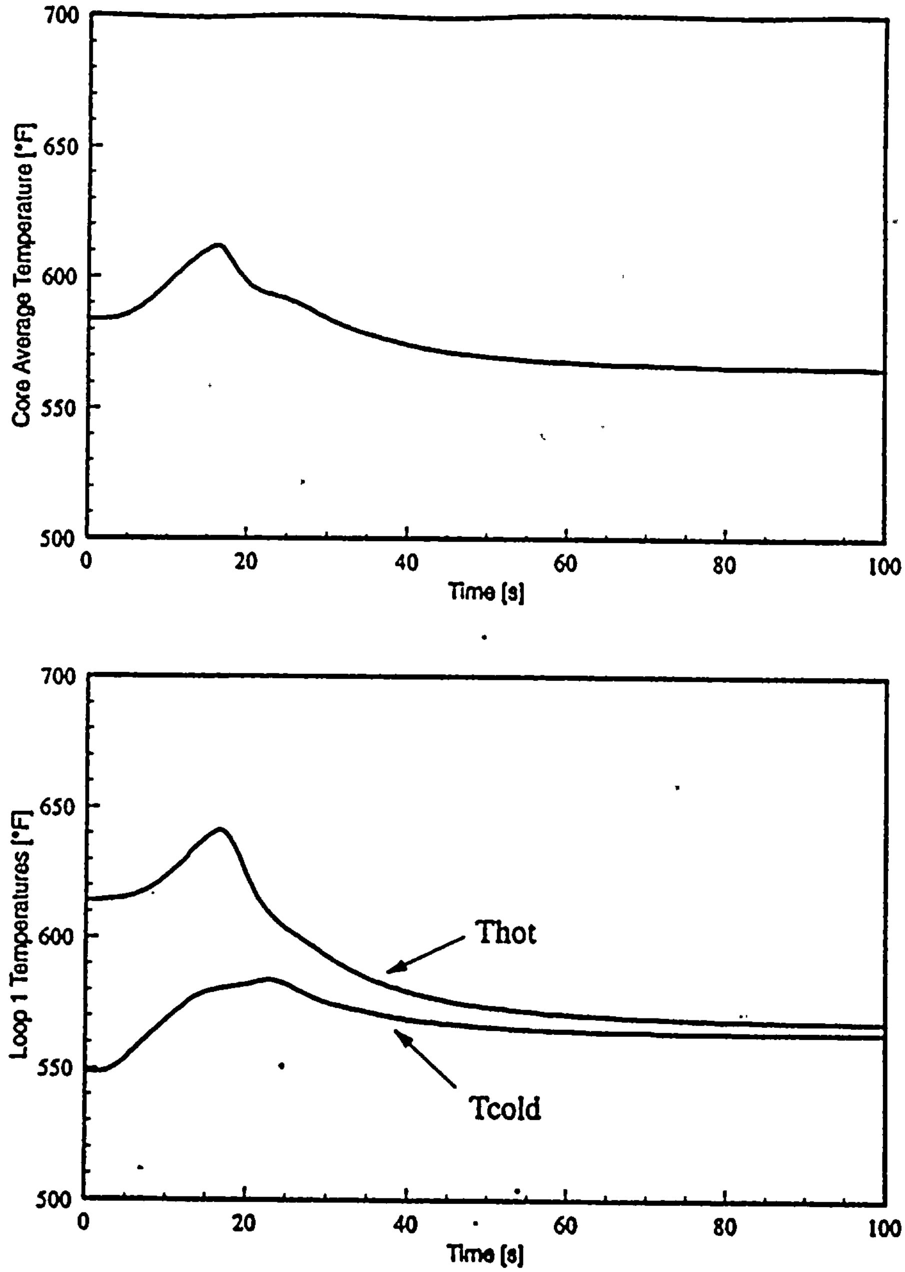
DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-7

Nuclear Power and DNBR vs.Time
For Loss of Load, Minimum Reactivity Feedback
With Pressurizer Spray and PORVs



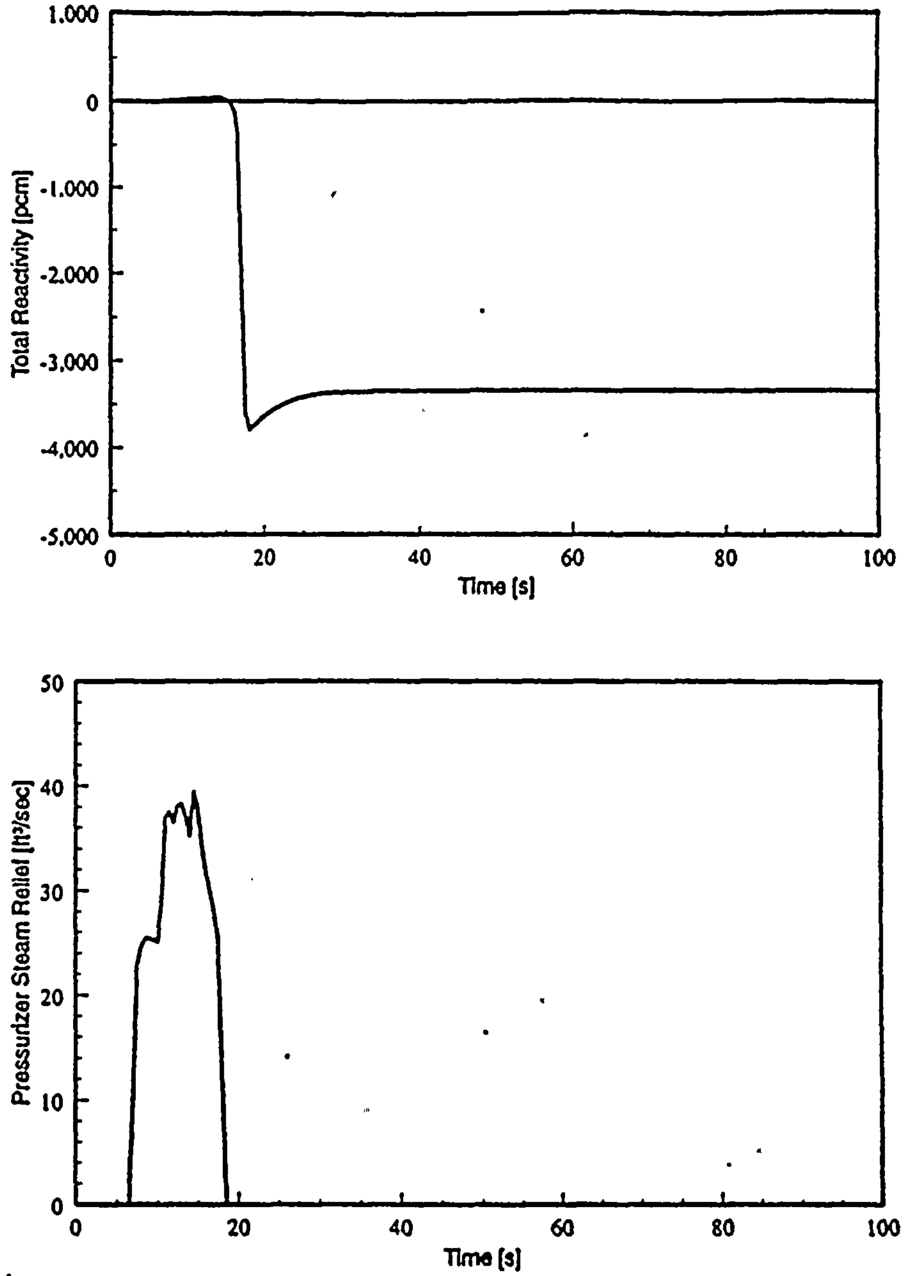
DONALD C. COOK NUCLEAR PLANT UNIT 2
FIGURE 3.3-8 Pressurizer Pressure and Pressurizer Water Volume vs.Time For Loss of Load, Minimum Reactivity Feedback With Pressurizer Spray and PORVs



DONALD C. COOK NUCLEAR PLANT UNIT 2

FIGURE 3.3-9

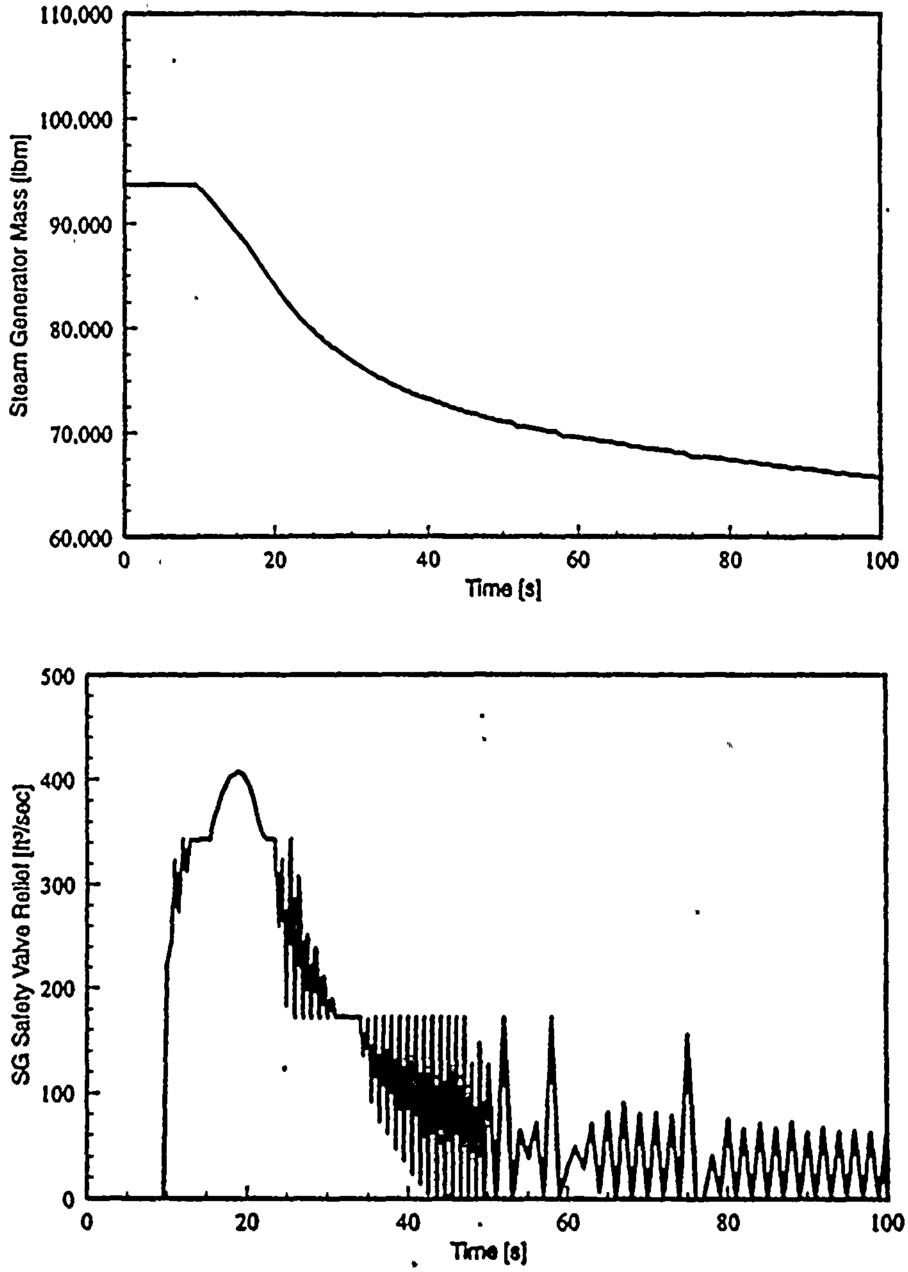
Core Average and Loop 1 Temperatures vs. Time For Loss of Load, Minimum Reactivity Feedback With Pressurizer Spray and PORVs



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-10

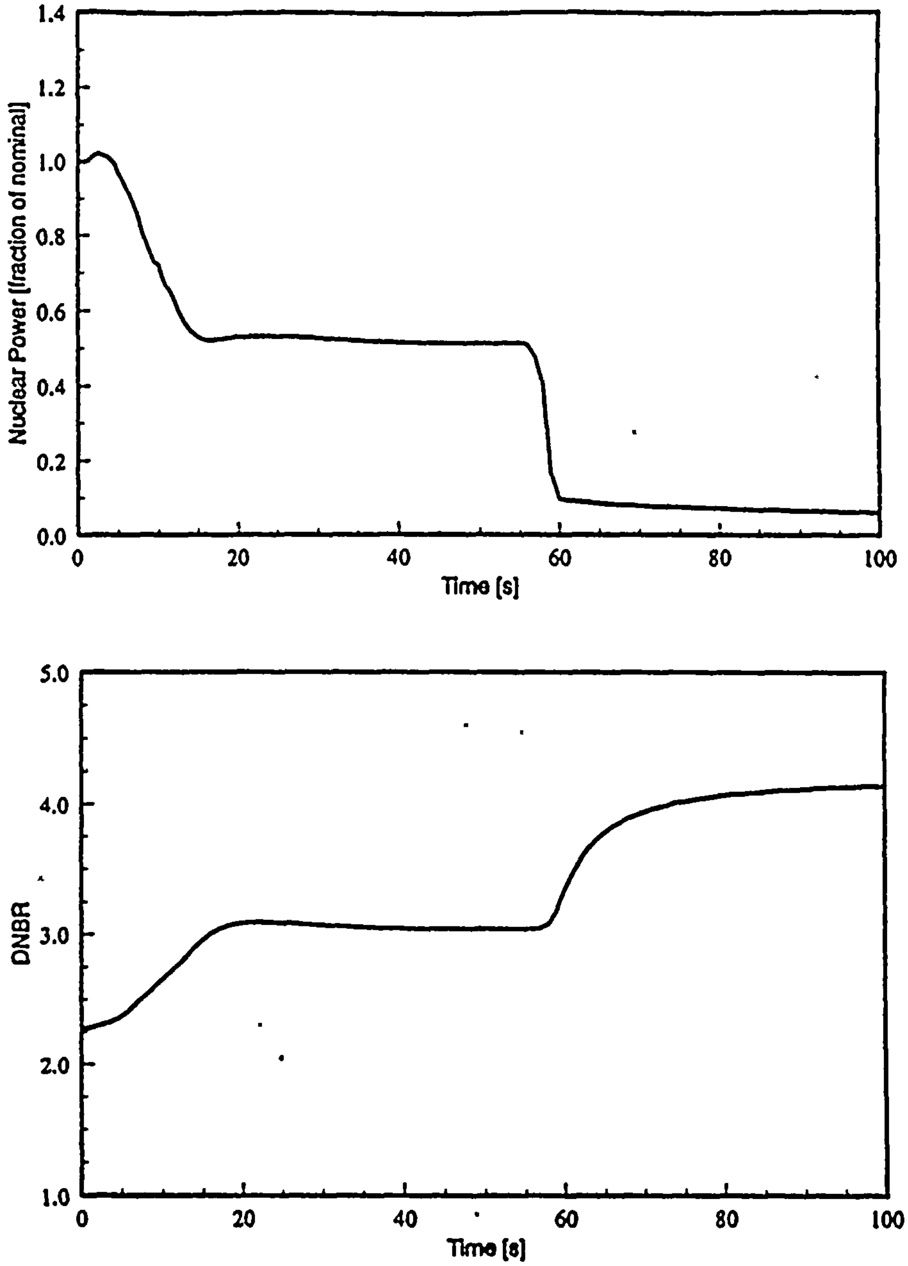
Total Reactivity and Pressurizer Steam Relief vs.Time
For Loss of Load, Minimum Reactivity Feedback
With Pressurizer Spray and PORVs



DONALD C. COOK NUCLEAR PLANT UNIT 2

FIGURE 3.3-11

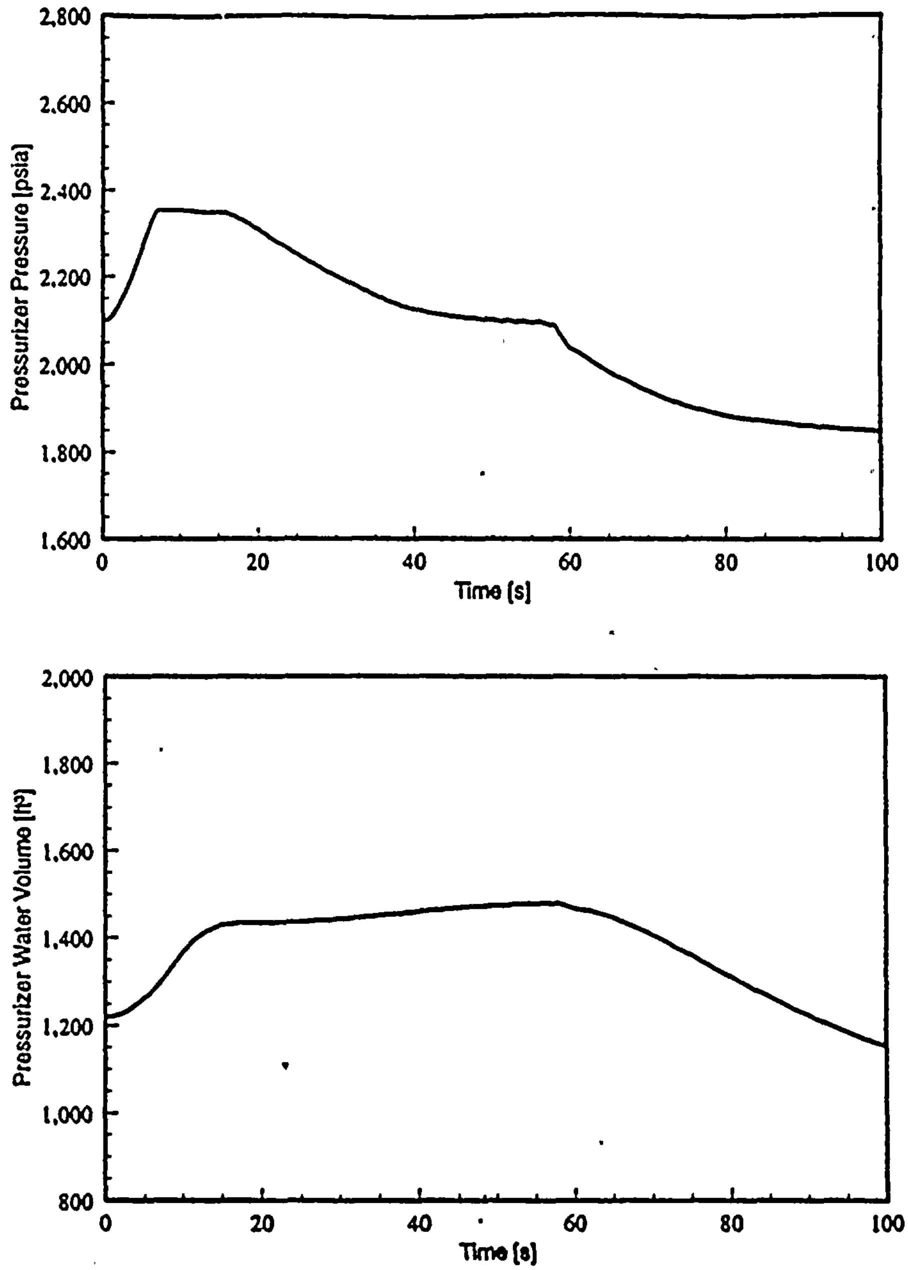
Steam Generator Mass and Safety Valve Relief vs.Time For Loss of Load, Minimum Reactivity Feedback With Pressurizer Spray and PORVs



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-12

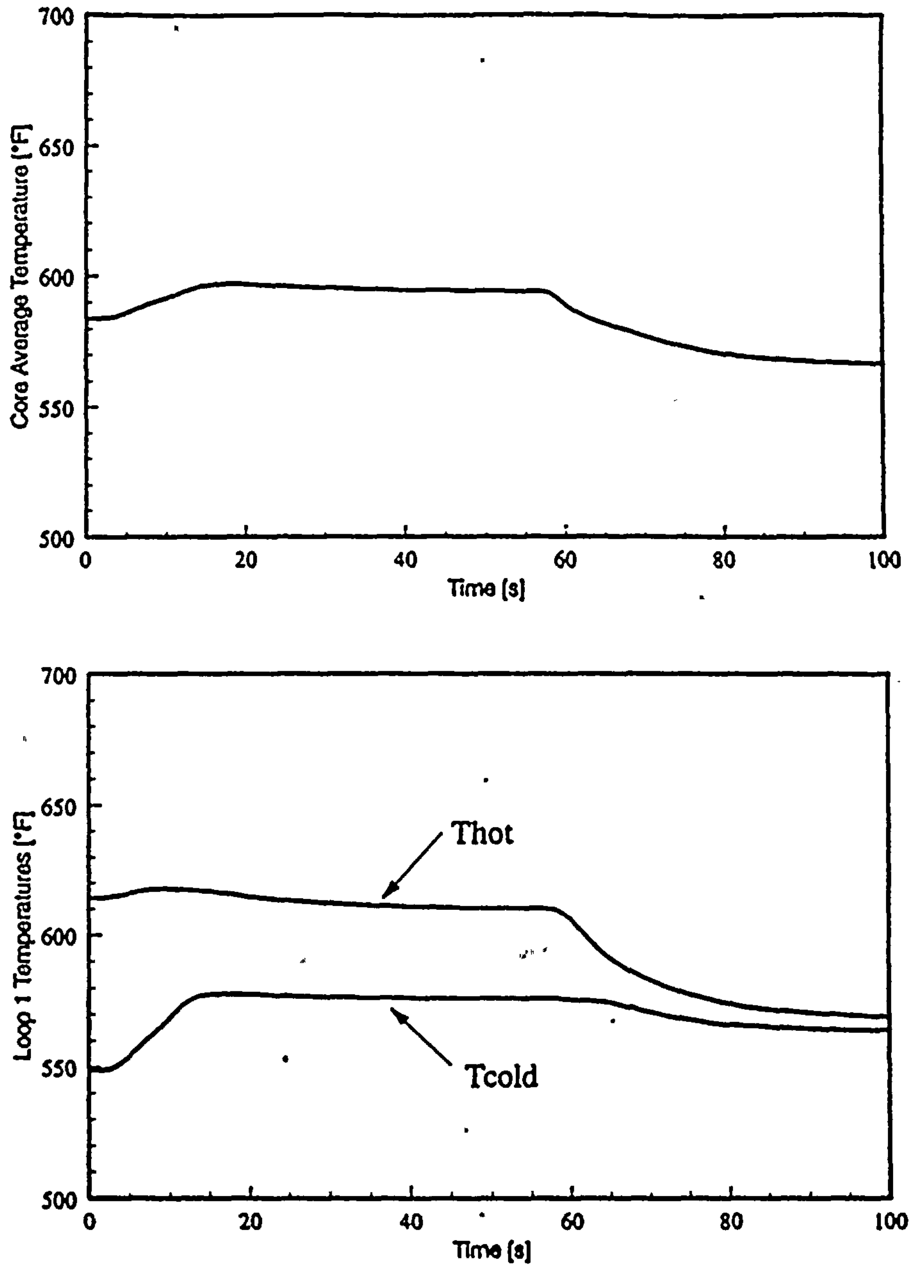
Nuclear Power and DNBR vs.Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-13

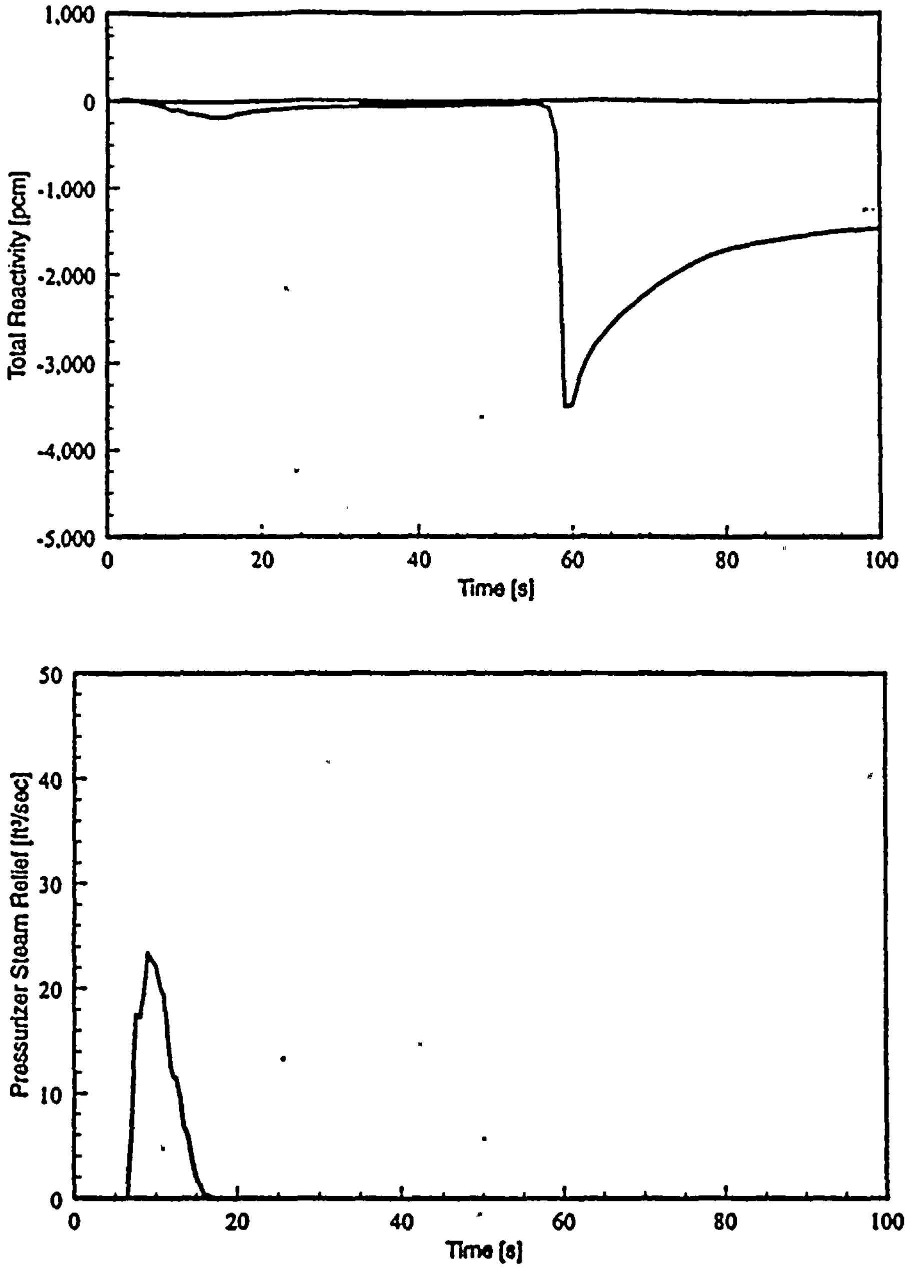
Pressurizer Pressure and Pressurizer Water Volume vs.Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-14

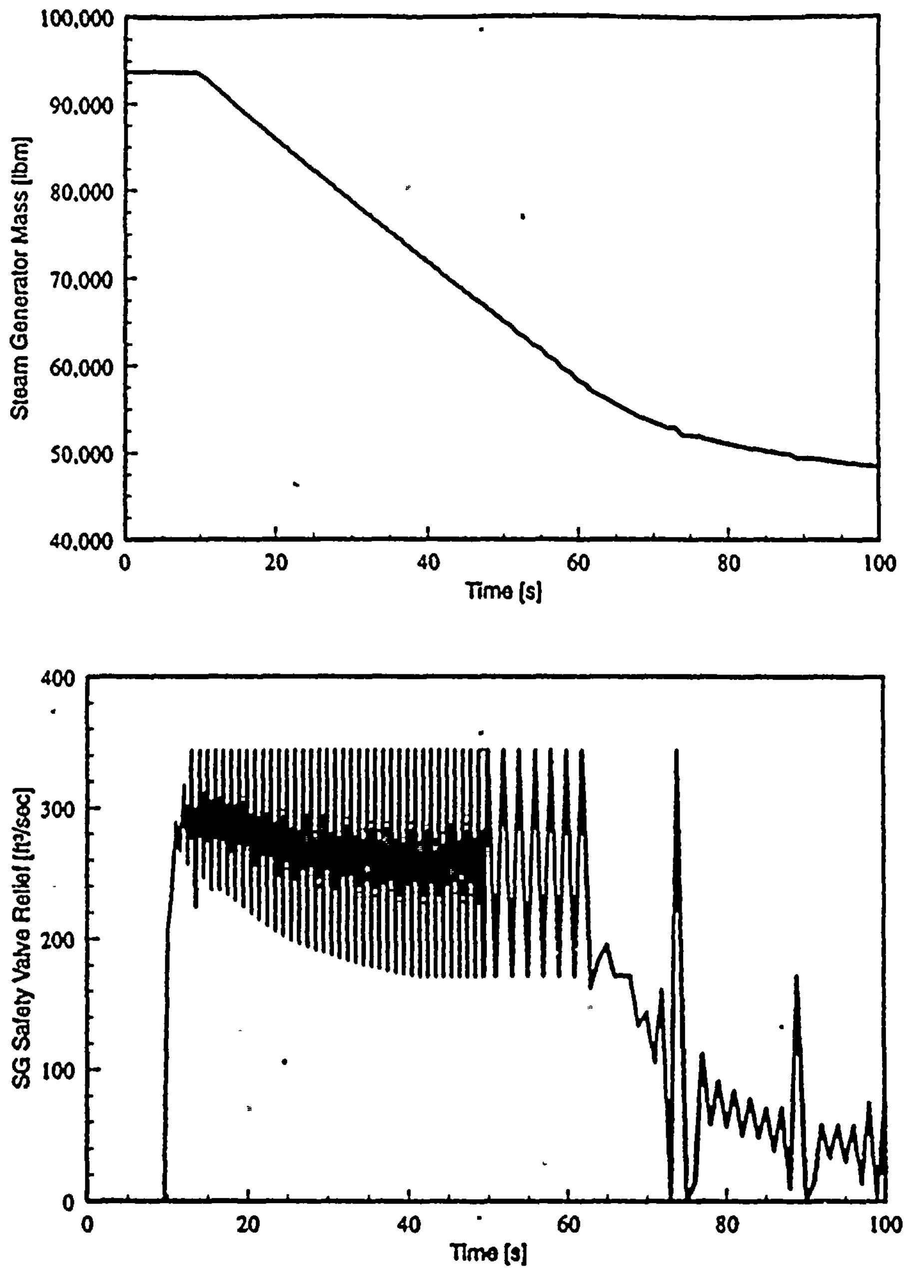
Core Average and Loop 1 Temperatures vs.Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-15

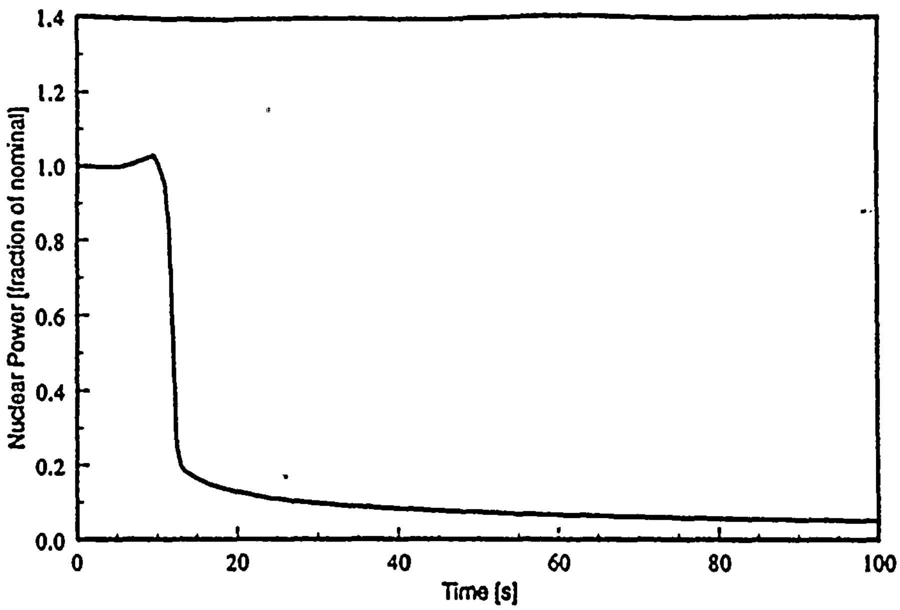
Total Reactivity and Pressurizer Steam Relief vs.Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs

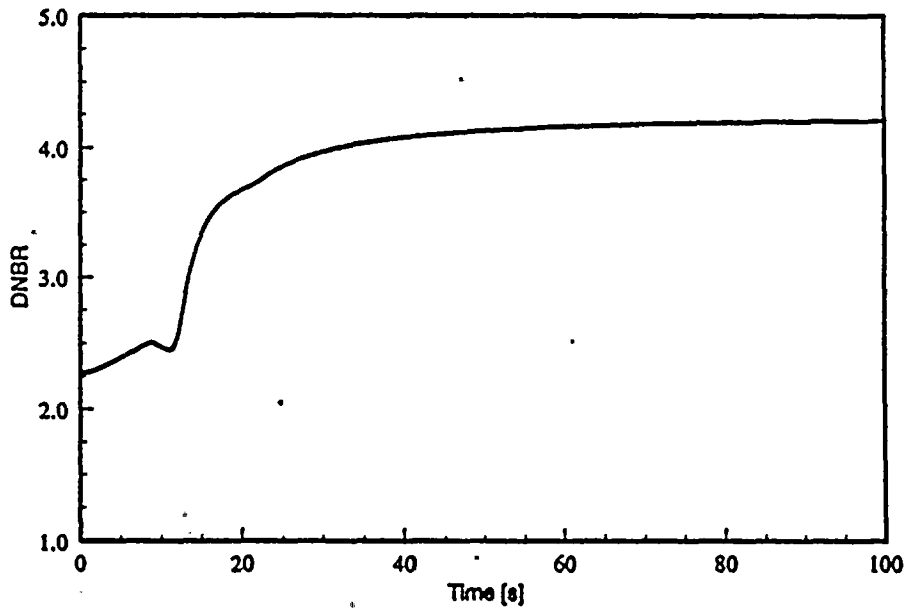


DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-16

Steam Generator Mass and Safety Valve Relief vs.Time
For Loss of Load, Maximum Reactivity Feedback
With Pressurizer Spray and PORVs

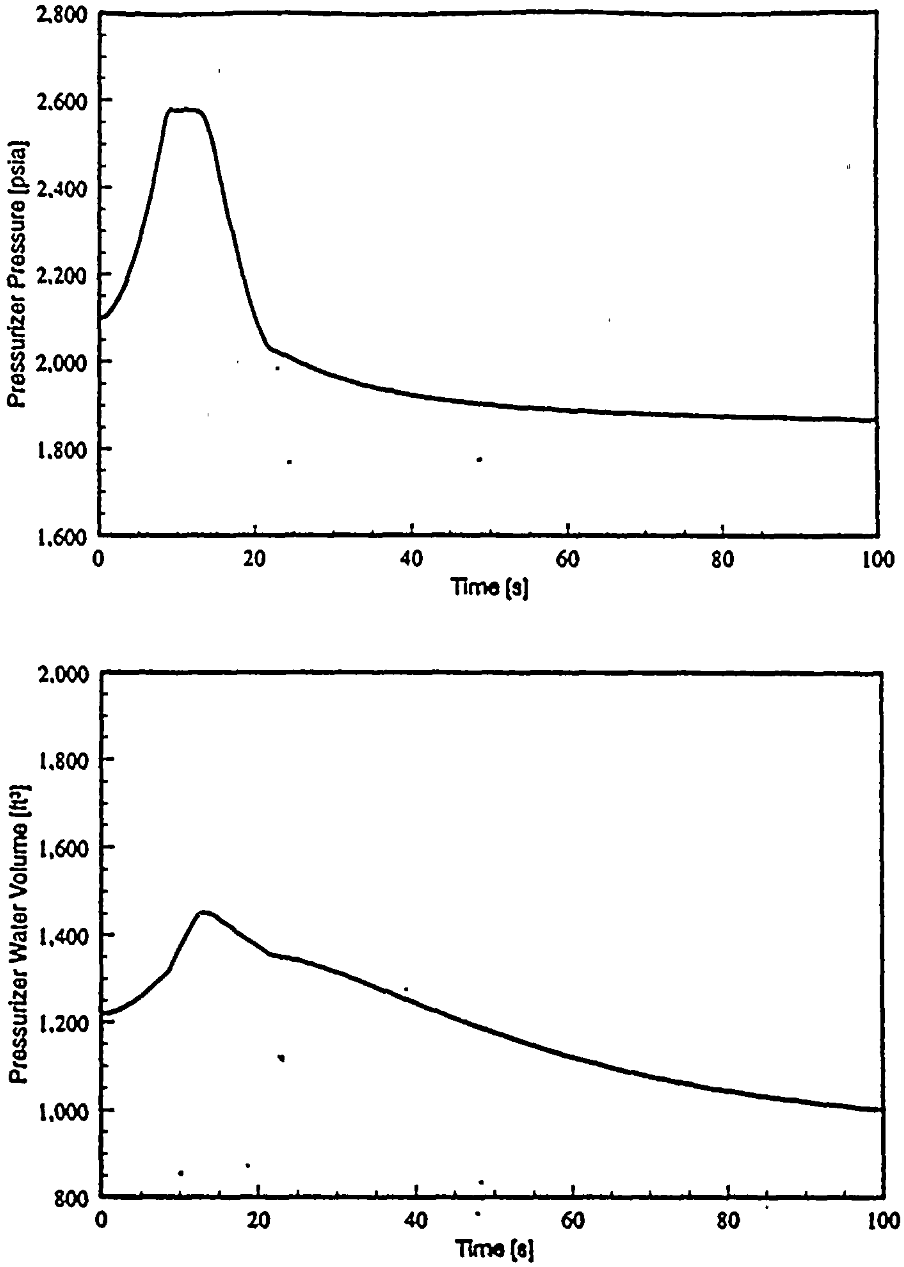




DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-17

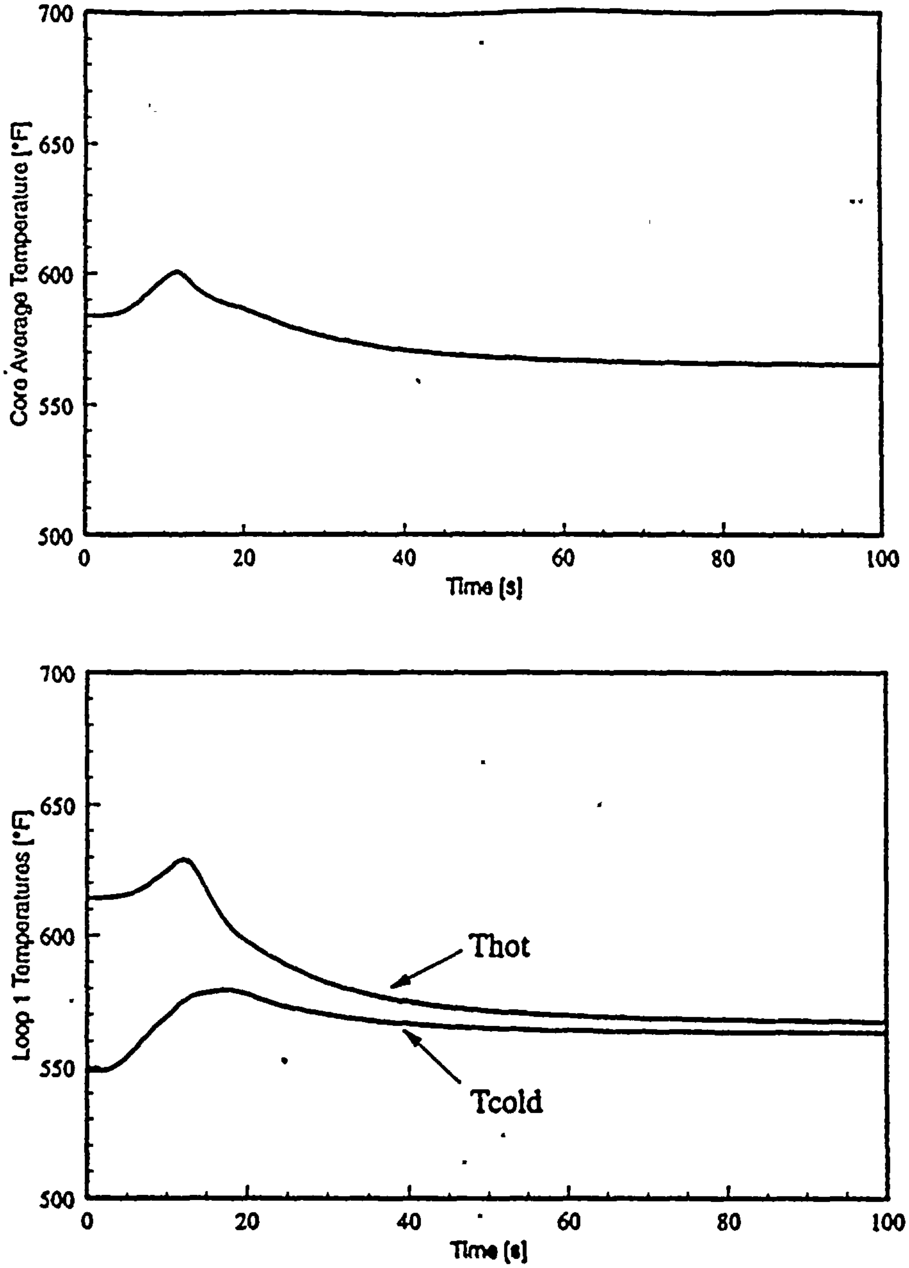
Nuclear Power and DNBR vs.Time
For Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-18

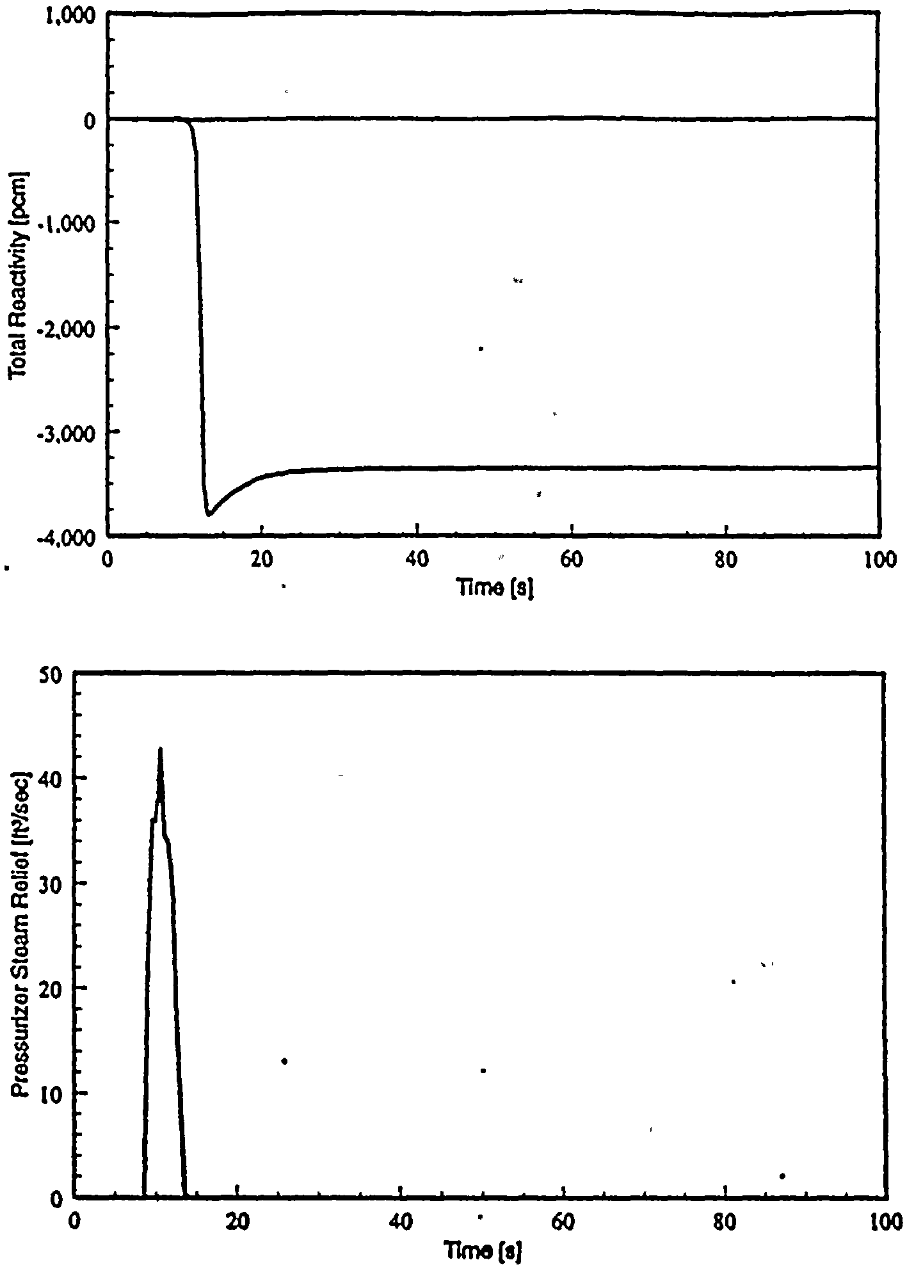
Pressurizer Pressure and Pressurizer Water Volume vs.Time
For Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs



DONALD C. COOK NUCLEAR PLANT UNIT 2

FIGURE 3.3-19

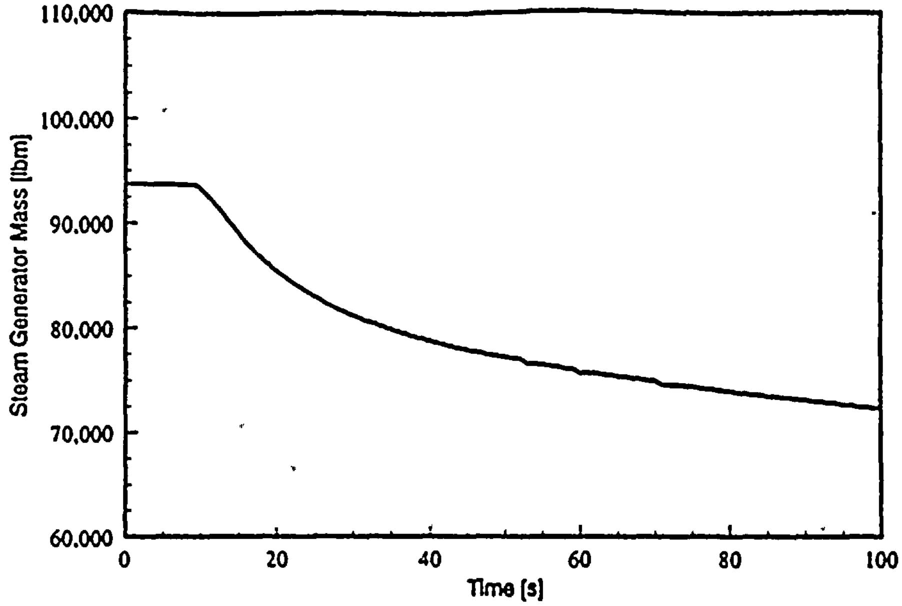
Core Average and Loop 1 Temperatures vs. Time For Loss of Load, Minimum Reactivity Feedback Without Pressurizer Spray and PORVs

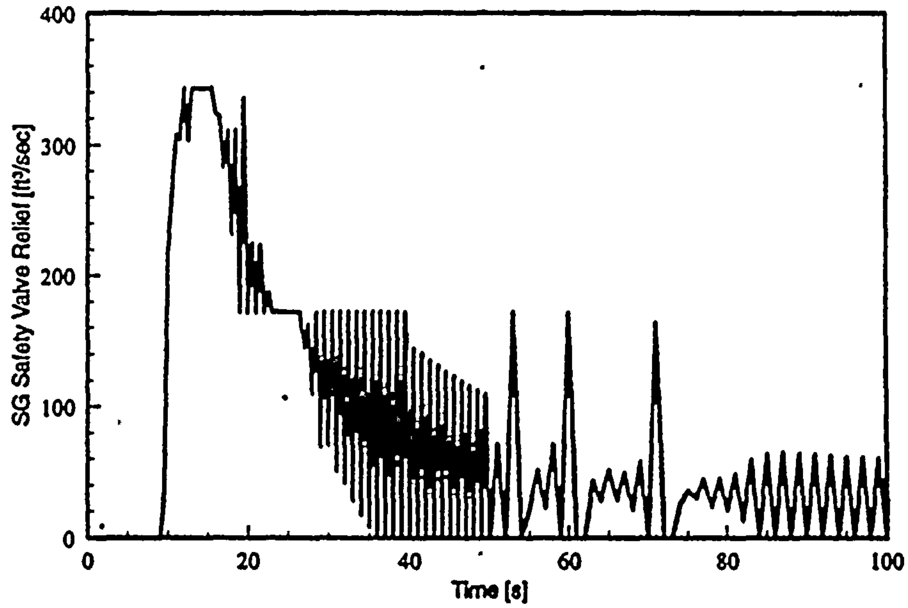


DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-20

Total Reactivity and Pressurizer Steam Relief vs.Time
For Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs

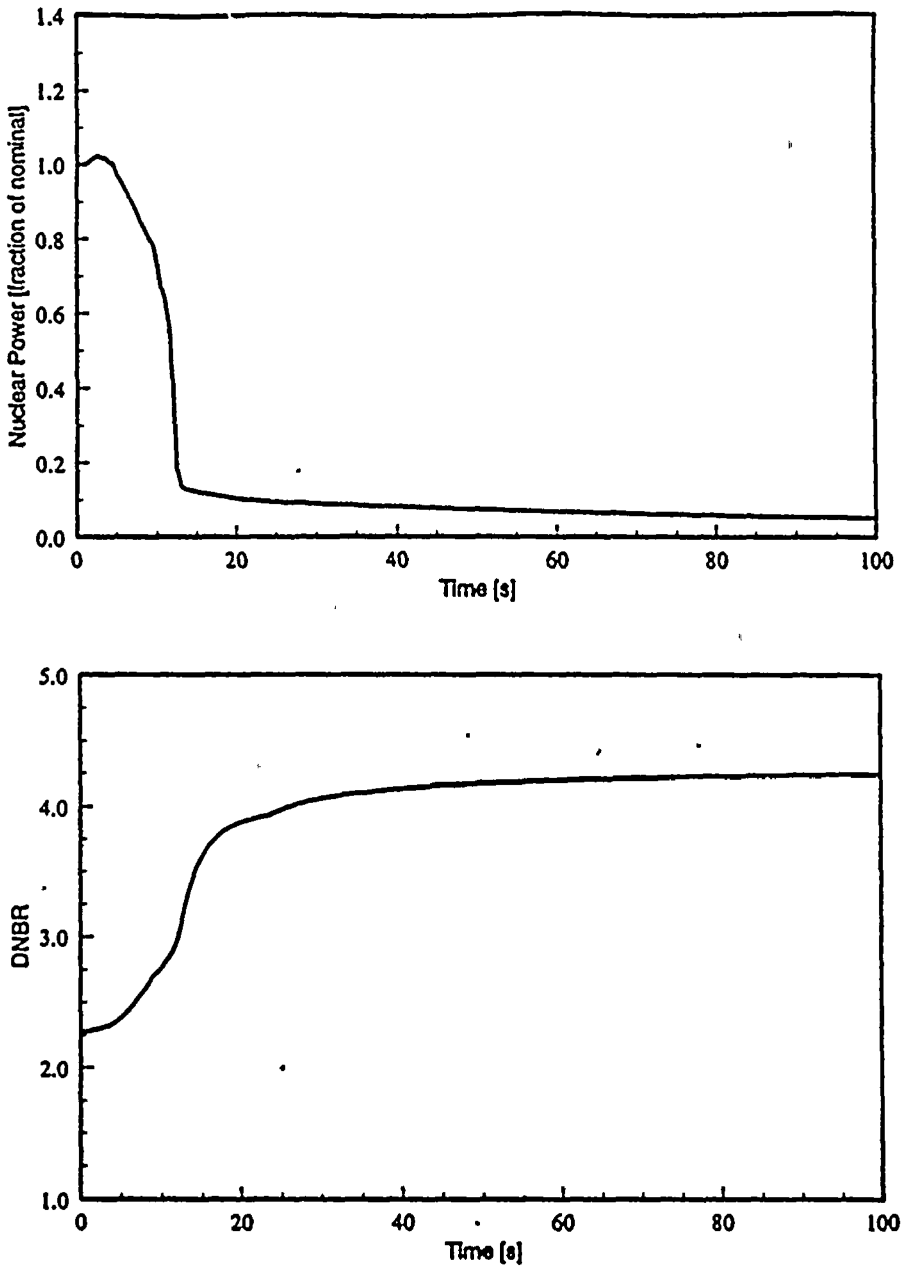




DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-21

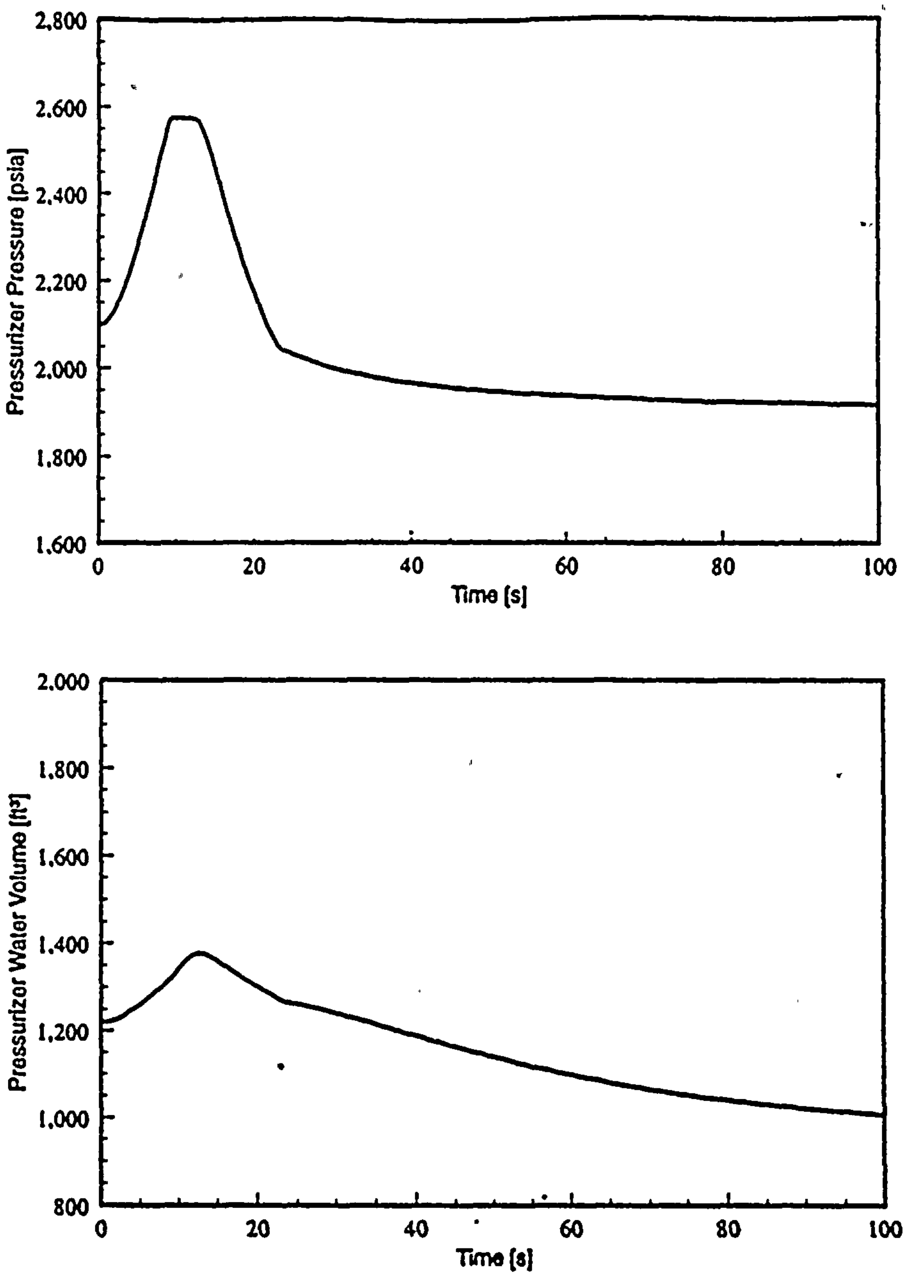
Steam Generator Mass and Safety Valve Relief vs.Time
For Loss of Load, Minimum Reactivity Feedback
Without Pressurizer Spray and PORVs



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-22

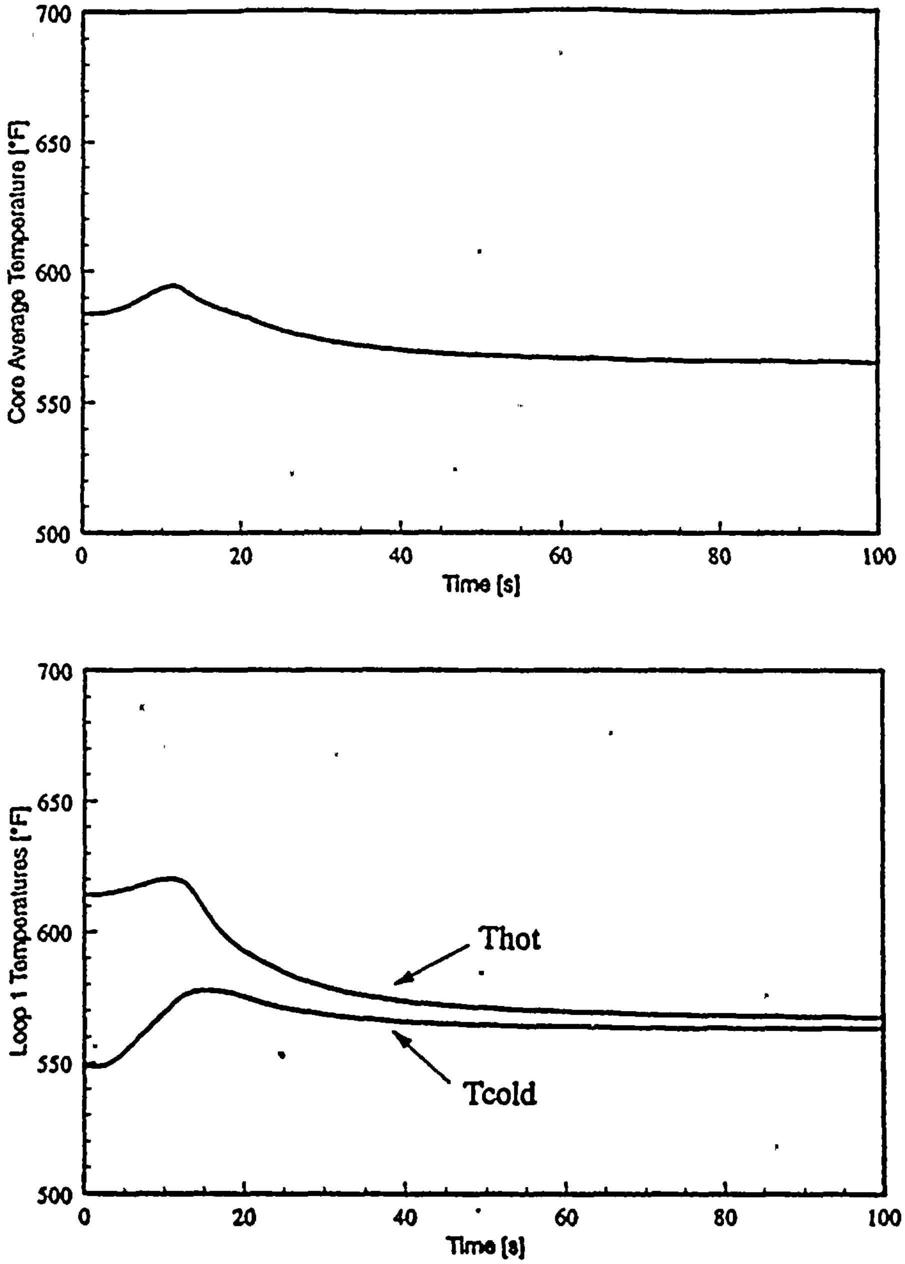
Nuclear Power and DNBR vs.Time
For Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs



DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-23

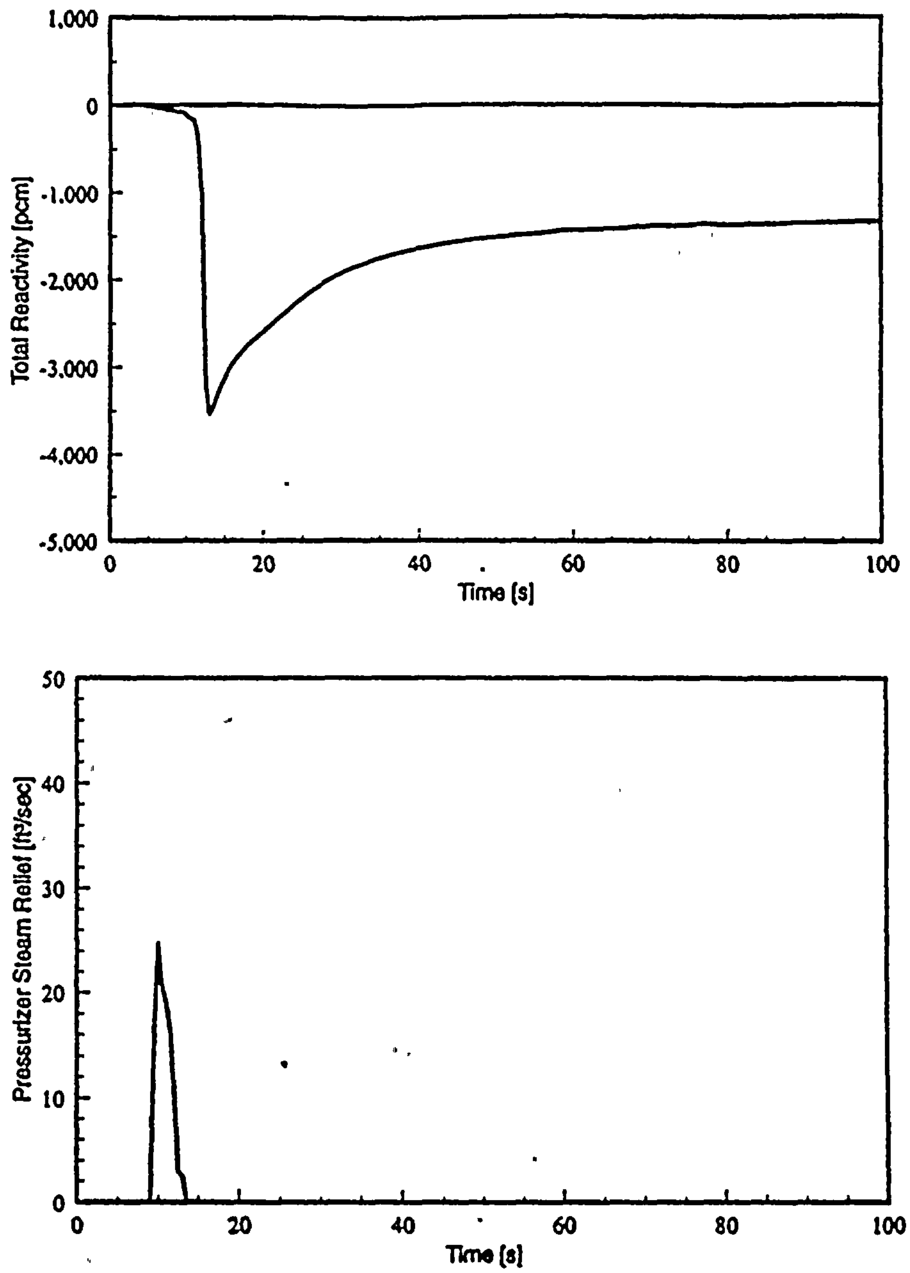
Pressurizer Pressure and Pressurizer Water Volume vs.Time
For Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs



DONALD C. COOK NUCLEAR PLANT UNIT 2

FIGURE 3.3-24

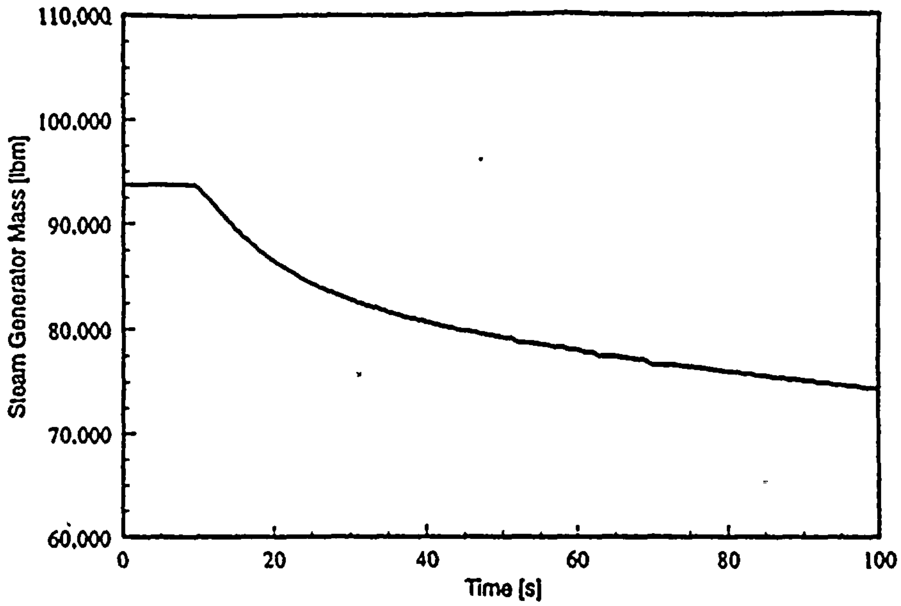
Core Average and Loop 1 Temperatures vs.Time For Loss of Load, Maximum Reactivity Feedback Without Pressurizer Spray and PORVs

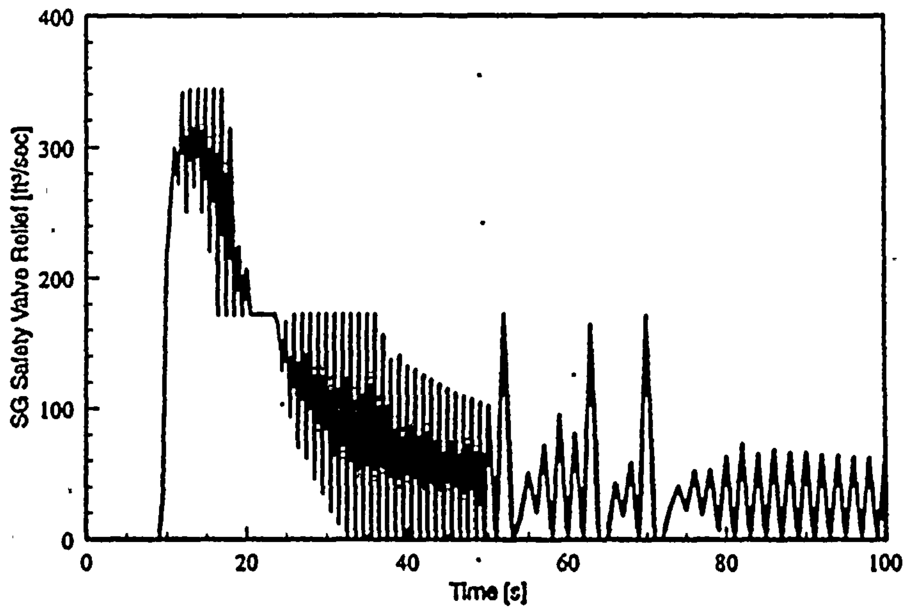


DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-25

Total Reactivity and Pressurizer Steam Relief vs.Time
For Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs





DONALD C. COOK
NUCLEAR PLANT
UNIT 2

FIGURE 3.3-26

Steam Generator Mass and Safety Valve Relief vs.Time
For Loss of Load, Maximum Reactivity Feedback
Without Pressurizer Spray and PORVs

3.4 POST LOCA HYDROGEN PRODUCTION

As part of the Rerating Program, Westinghouse provided hydrogen generation rates and inventories inside containment from various sources; including core radiolysis, sump radiolysis, corrosion-generated hydrogen and the zirconium/water reaction. The hydrogen generation inside containment was recalculated based on the results of analyses performed for the Unit 2 Uprating Program; i.e. a zirconium/water reaction of less than 1%, and revised post-accident temperatures.

Considering the 10 CFR 50.44 (d) (1) factor of 5 increase to the Zirconium-water reaction percentage, hydrogen generation analyses were performed based on a Zirconium-water reaction of 5%. Also, corrosion of the materials listed in Table 14.3.6-3 of the July 1990 revision of the UFSAR was considered based on the revised time-temperature profile. Other assumptions employed in the analysis are consistent with the values currently in the Donald C. Cook Nuclear Plant Unit 2 UFSAR.

The analysis indicates that the time to reach four (4) volume percent hydrogen inside the containment is 20 hours after the accident. However, the evaluation indicates that the maximum hydrogen concentration in containment remains below 4% based on start of the recombiner at the end of the sixth hour. Thus, the basic conclusion stated in the UFSAR remains valid and applicable; i.e. as stated on page 14.3.6-14 of the July 1990 revision to the UFSAR:

> *"... It is intended that the recombiner will be turned on approximately 6 hours after the accident or when dictated by gas analysis (the recombiner actuation times was revised from 24 hours to 6 hours as a result of procedure reviews subsequent to the TMI-2 accident) to limit the containment hydrogen concentration to a safe value...."*

3.5 CONTAINMENT ANALYSES (SHORT-TERM, LONG-TERM MSLB, LONG-TERM LOCA)

3.5.1 Short-Term Containment Analysis

The short term containment integrity analysis is used to verify the adequacy of interior structures and walls by demonstrating that calculated differential pressures are less than design limits. The functionality of the ice condenser is demonstrated and containment integrity is also verified. The efforts performed for the short term containment analysis, applicable to the Pressurizer Enclosure, as described in Section 3.4.1 of WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," support operation of Cook Nuclear Plant Units 1 and 2 over the full range of rerated parameters. The major impacts, on LOCA short term mass and energy release rate calculations and containment subcompartment response analysis, are the effects due to reactor coolant system temperature changes. The short term mass and energy releases are linked directly to the critical mass flux, which increase with decreasing temperatures. The temperatures applicable for the Uprating Program remain unchanged, bounded by the temperatures assessed to support WCAP-11902, Supplement 1. For the steam generator enclosure, mass and energy releases and the subsequent containment response are performed at zero power, which maximizes effects because steam pressure is maximized. All relevant analyses and evaluations performed for the Rerating Program assumed values which would bound both Units 1 and 2, at the rerated power levels and revised temperatures and pressures described in Table 2.1-1 of WCAP-11902, Supplement 1. The results of the short term containment analyses and evaluations for the Uprating Program demonstrate that, for the pressurizer enclosure, the fan accumulator room and the steam generator enclosure, the resulting peak pressures remain below the allowable design peak pressures. Since the calculated pressures in WCAP-11902, Supplement 1 for the loop compartments exceeded the design pressure, demonstration of structural adequacy was required. This issue was addressed by AEPSC and is documented in UFSAR Section 14.3.4.2.3.4.

3.5.2 Loss-of-Coolant Mass and Energy Release

3.5.2.1 Purpose

The purpose of this analysis was to calculate the long term Loss of Coolant Accident (LOCA) mass and energy releases with the proposed revised plant conditions and increased operating margins. The increased operating margins include assumed conditions which would bound both Units 1 and 2.

This section provides the analytical basis with respect to the LOCA containment mass and energy release for the operation of the Donald C. Cook Nuclear Plant Units 1 and 2 at the Uprating Program conditions. This containment integrity analysis bounds both units.

Rupture of any of the piping carrying pressurized high temperature reactor coolant, termed a LOCA, will result in release of steam and water into the containment. This, in turn, will result in

an increase in the containment pressure and temperature. The mass and energy release rates described in this document form the basis of further computations to evaluate the structural integrity of the containment following a postulated accident to satisfy the Nuclear Regulatory acceptance criteria, General Design Criterion 38, which is more restrictive than the old GDC criteria, Appendix H of the original FSAR, to which the Donald C. Cook Plants are licensed. Section 3.5.3 presents the long term containment integrity analysis for containment pressurization evaluations.

3.5.2.2 <u>System Characteristics and Modeling Assumptions</u>

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Some of the most critical items are the: RCS initial conditions, core decay heat, safety injection flow, and metal and steam generator heat release modeling. (Note: Initial conditions reflecting the Unit 2 replacement steam generator, which bound the Unit 1 steam generator conditions with respect to steam pressure and steam temperature were modeled to maximize the steam generator heat release.) Specific assumptions concerning each of these items are discussed next. Tables 3.5-1 and 3.5-2 present key data assumed in the analysis.

For the long term mass and energy release calculations, operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The modeled core rated power of 3588 MWt adjusted for calorimetric error (+2 percent of power) was the basis in the analysis. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. Additionally, an allowance of +5.1°F is reflected in the temperatures in order to account for instrument error and deadband. The initial reactor coolant system (RCS) pressure in this analysis is based on a nominal value of 2250 psia. Also included is an allowance of +67 psi, which accounts for the measurement uncertainty on pressurizer pressure. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2317 (2250 + 67) psia initial pressure was selected as the limiting case for the long term mass and energy release calculations. These assumptions conservatively maximize the mass and energy in the RCS.

The selection of the fuel design features for the long term mass and energy calculation is based on the need to conservatively maximize the core stored energy. The margin in core stored energy was chosen to be +15 percent. Thus, the analysis very conservatively accounts for the stored energy in the core. The fuel conditions were adjusted to provide a bounding analysis for

current Cook Nuclear Plant Units 1 and 2 fuel design features. The following items serve as the basis to ensure conservatism in the core stored energy calculation: time of maximum fuel densification and the highest BOL temperatures.

Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty) is modeled.

With respect to safety injection flow, the mass and energy calculation considered the limiting scenario of minimum safety injection flow, with the RHR crosstie valve closed, 15% pump head degradation for the RHR and SI pumps, and 10% pump head degradation for the charging pumps. This configuration conservatively bounds other respective alignments. Closure of the RHR crosstie was considered over the HHSI crosstie because this would have a more severe impact on the analysis (i.e., closure of the RHR crosstie would bound closure of the HHSI crosstie). This results in the conservative minimum safety injection flowrate used.

The following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperature of the reactor coolant system (100% full power conditions)

2. An allowance in temperature for instrument error and dead band (+5.1°F)

3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)

4. Core rated power of 3588 MWt

5. Allowance for calorimetric error (+2 percent of power)

6. Conservative coefficient of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)

7. Allowance in core store energy for effect of fuel densification

8. A margin in core stored energy (+15 percent included to account for manufacturing tolerances)

9. An allowance for RCS initial pressure uncertainty (+67 psi)

10. Steam generator tube plugging leveling (0% uniform)

- Maximizes reactor coolant volume and fluid release

- Maximizes heat transfer area across the SG tubes

- Reduces coolant loop resistance, which reduces delta-p upstream of break and increases break flow

Thus, based on the above conditions and assumptions, a bounding analysis of Cook Nuclear Plant Units 1 and 2 is made for the release of mass and energy from the RCS in the event of a LOCA to support the Uprating Program.

3.5.2.3 Long Term Mass and Energy Release Analysis

3.5.2.3.1 Introduction

The evaluation model used for the long term LOCA mass and energy release calculations was the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved by the NRC, and has been used in the analysis of other ice condenser plants.

This report section presents the long term LOCA mass and energy releases that were generated in support of the D. C. Cook Unit 2 Uprating Program. These mass and energy releases are then subsequently used in the LOTIC-1 containment integrity analysis peak pressure calculation.

3.5.2.3.2 LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy analysis, is typically divided into four phases:

1. Blowdown - the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium state at containment design pressure.

2. Refill - the period of time when the lower plenum is being filled by accumulator and ECCS water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

3.5.2.3.3 Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long term LOCA mass and energy releases for the Cook Nuclear Plant Units 1 and 2.

SATAN calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass and energy flowrates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the Emergency Core Cooling System refills the reactor vessel and provides cooling to the core. The most important feature is the steam/water mixing model (See Section 3.5.2.6.2).

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

EPITOME continues the FROTH post-reflood portion of the transient calculation from the time at which the secondary equilibrates to containment design pressure to the end of the transient. It also compiles a summary of data on the entire transient, including instantaneous mass and energy release tables and mass and energy balance tables, with data provided at critical times.

3.5.2.4 <u>Break Size and Location</u>

Generic studies have been performed with respect to the effect of postulated break size on the LOCA mass and energy releases. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break location analyzed for the Uprating Program is the pump suction double ended rupture, DEPS (10.48 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for each case analyzed. The following information provides a discussion on each break location.

The hot leg double ended rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid which exits the core bypasses the steam generators venting directly to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, generic studies (addressing both dry and ice condenser design containments) have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). In addition, for ice condenser containment design, the large inventory of ice is available to condense the steam release. The reflood period for the Uprating Program analysis extends from approximately 29.6 seconds to 258.6 seconds. The mass and energy releases for the hot leg break have not been included in the scope of this containment integrity analysis, because, for the hot leg break, only the blowdown phase of the transient is of any significance. Since there are no reflood and post-reflood phases to consider for the hot leg break, the limiting peak pressure calculated would be the compression peak pressure.

The cold leg break location has also been found in previous studies to be much less limiting than the pump suction break in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is not included in the scope of this Uprating program.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the Reactor Coolant System in calculating the releases to

containment. This break location has been determined to be the limiting break for all ice condenser plants relative to containment pressure.

In summary, the analysis of the limiting break location for an ice condenser containment has been performed and is shown in this report. The double-ended pump suction guillotine break has historically been considered to be the limiting break location, by virtue of its consideration of all energy sources present in the Reactor Coolant System (RCS). This break location provides a mechanism for the release of the available energy in the RCS, including both the broken and intact loop steam generators.

3.5.2.5 Application of Single Failure Criteria

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for the pump suction (DEPS) break. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, which are required to power the safety injection system. This is not an issue for the blowdown period which is limited by the compression peak pressure.

The limiting minimum safety injection case has been analyzed for the effects of a single failure. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train, thereby minimizing the safety injection flow. An additional conservatism has been included in this analysis in that the closure of the RHR crosstie valve has been considered because it results in a further reduction in safety injection flow. The analysis further considers the RHR and SI pump head curves to be degraded by 15% and the charging pump head curve to be degraded by 10%. This results in the greatest SI flow reduction for the minimum safeguards case.

3.5.2.6 Mass and Energy Release Data

3.5.2.6.1 Blowdown Mass and Energy Release Data

A version of the SATAN-VI code is used for computing the blowdown transient, which is the code used for the Emergency Core Cooling System (ECCS) calculation in Reference 2.

The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is determined appropriately by conditions in the transient. The methodology for the use of this model is described in Reference 1.

Table 3.5-3 presents the calculated mass and energy releases for the blowdown phase of the DEPS break. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break; break path 2 refers to the mass and energy exiting from the pump side of the break.

3.5.2.6.2 Reflood Mass and Energy Release Data

The WREFLOOD code used for computing the reflood transient, is a modified version of that used in the 1981 ECCS evaluation model, Reference 2.

The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator release are included as auxiliary equations which interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break; i.e. the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and emergency core cooling injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the Reference 1 mass and energy release evaluation model, used in the analysis of other ice condenser plants. Even though the Reference 1 model credits steam/mixing only in the intact loop and not in the broken loop, justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 3). This assumption is justified and supported by test data, and is summarized as follows:

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two phase interaction with condensation of steam by cold ECCS water. The second is a single phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3 scale tests (Reference 4), which are the largest scale data available and thus most clearly simulates

the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flowrates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double ended rupture break. For this break, there are two flowpaths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results is contained in References 1 and 4.

Table 3.5-4 presents the calculated mass and energy release for the reflood phase of the pump suction double ended rupture with minimum safety injection.

The transients of the principal parameters during reflood are provided in Table 3.5-5.

3.5.2.6.3 Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 5) is used for computing the post-reflood transient.

The FROTH code calculates the heat release rates resulting from a two-phase mixture level present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two phase fluid exits the core, flows through the hot legs and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two phase. The methodology for the use of this model is described in Reference 1.

After containment depressurization, the mass and energy release available to containment is generated directly from core boiloff/decay heat.

Table 3.5-6 presents, the two phase post-reflood (froth) mass and energy release data for the pump suction double ended case.

3.5.2.7 Decay Heat

On November 2, 1978 the Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society approved ANS standard 5.1 for the determination of decay heat. This standard was used in the mass and energy release model with the following input:

Significant assumptions in the generation of the decay heat curve:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and N_p-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Table 10, of Reference 6.

5. Operation time before shutdown is 3 years.

6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

7. Two sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

3.5.2.8 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (Tsat) at the containment design pressure. After the FROTH calculations, steam generator secondary energy is removed based on first and second stage rates. The first stage rate is applied until the steam generator reaches Tsat at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment

pressure. Then the second stage rate is used until the final depressurization. The heat removal of the broken loop and intact loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature and a secondary side heat transfer coefficient determined using a modified McAdam's correlation [Reference 7]. Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization.

3.5.2.9 Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are given in Table 3.5-7. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Table 3.5-8. The energy sources include:

1. Reactor Coolant System Water

2. Accumulator Water

3. Pumped Injection Water

4. Decay Heat

5. Core Stored Energy

6. Reactor Coolant System Metal - Primary Metal (includes SG tubes)

7. Steam Generator Metal (includes transition cone, shell, wrapper, and other internals)

8. Steam Generator Secondary Energy (includes fluid mass and steam mass)

9. Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

It should be noted that the inconsistency in the energy balance tables from the end of Reflood to 3600 seconds, i.e., "Total Available" data versus "Total Accountable" resulted from the omission of the reactor upper head in the analysis following blowdown. It has been concluded that the results are more conservative when the upper head is neglected. This does not affect the instantaneous mass and energy releases, or the integrated values, but causes an increase in the total accountable energy within the energy balance table.

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time of broken loop steam generator equilibration to pressure setpoint
6. Time of intact loop steam generator equilibration to pressure setpoint

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature did not rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

The LOCA mass and energy release analysis has been performed in accordance with the criteria shown in the Standard Review Plan Section 6.2.1.3. In this analysis, the relevant requirements of General Design Criteria (GDC) 50 and 10 CFR Part 50 Appendix K, as it relates to sources of energy during the LOCA to assure all energy sources have been considered. These sources include: reactor power, decay heat, core stored energy, energy stored in the reactor vessel and internals, and stored energy in the secondary system. Although Cook Units 1 and 2 are not Standard Review Plan Plants, the review guidelines presented in the Standard Review Plan Section 6.2.1.3 have been satisfied.

3.5.2.10 References

1. "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version", WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Non-Proprietary).

2. "Westinghouse ECCS Evaluation Model - 1981 Version", WCAP-9220-P-A, Rev. 1, February 1982 (Proprietary), WCAP-9221-A, Rev.1 (Non-Proprietary)

3. Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 7106), for D.C. Cook Nuclear Plant Unit 1", June 9, 1989.

4. EPRI 294-2, Mixing of Emergency Corė Cooling Water with Steam; 1/3 Scale Test and Summary, (WCAP-8423), Final Report June 1975.

5. "Westinghouse Mass and Energy Release Data For Containment Design", WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A (Non-Proprietary).

6. ANSI/ANS-5.1 1979, American National Standard for Decay Heat Power in Light Water Reactors", August 1979.

7. W. H. McAdam, Heat Transmission , McGraw-Hill 3rd edition, 1954, p.172.

3.5.3 LOCA Containment Integrity Analysis

3.5.3.1 Description of LOTIC-1 Model

Early in the ice condenser development program it was recognized that there was a need for modeling of long term ice condenser performance. It was realized that the model would have to have capabilities comparable to those of the dry containment (COCO) model. These capabilities would permit the model to be used to solve problems of containment design and optimize the containment and safeguards systems. This has been accomplished in the development of the LOTIC code, described in Reference 1.

The model of the containment consists of five distinct control volumes, the upper compartment, the lower compartment, the portion of the ice bed from which the ice has melted, the portion of the ice bed containing unmelted ice, and the dead ended compartment. The ice condenser control volume with unmelted and melted ice is further subdivided into six subcompartments to allow for maldistribution of break flow to the ice bed.

The conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three phases in time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the ice condenser test facility described in Reference 3. These phases are the blowdown period, the depressurization period, and the long term.

The most significant simplification of the problem is the assumption that the total pressure in the containment is uniform. This assumption is justified by the fact that after the initial blowdown of the Reactor Coolant System, the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between the control volumes will also be relatively small. These flow rates then are unable to maintain significant pressure differences between the compartments.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. The air is considered a perfect gas, and the thermodynamic properties of steam are taken from the ASME steam table.

The condensation of steam is assumed to take place in a condensing node located, for the purpose of calculation, between the two control volumes in the ice storage compartment. The exit temperature of the air leaving this node is set equal to a specific value which is equal to the temperature of the ice filled control volume of the ice storage compartment. Lower compartment exit temperature is used if the ice bed section is melted.

3.5.3.2 Containment Pressure Calculation

The following are the major input assumptions used in the LOTIC analysis for the pump suction pipe rupture case with the steam generators considered as an active heat source for the Cook Nuclear Plant Containment:

1. Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two RHR pumps and one of two RHR heat exchangers providing flow to the core; one of two safety injection pumps and one of two centrifugal charging pumps; and one of two air return fans.

2. 2.11×10^6 lbs. of ice initially in the ice condenser.

3. The blowdown, reflood, and post reflood mass and energy releases described in Section 3.5.2 are used.

4. Blowdown and post-blowdown ice condenser drain temperatures of 190°F and 130°F are used, respectively.

5. Nitrogen from the accumulators in the amount of 4510 lbs. is included in the calculations.

6. Essential service water temperature of 87.5°F is used on the spray heat exchanger and the component cooling heat exchanger.

7. The air return fan is effective, 10 minutes after the transient is initiated.

8. No maldistribution of steam flow to the ice bed is assumed. (This assumption is conservative, contributes to early ice bed melt out time.)

9. No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)

10. The initial conditions in the containment are a temperature of 57°F in the upper compartment volume, and 60°F in the lower and dead-ended compartment volumes. All volumes are at a pressure of 0.3 psig.

11. Containment structural heat sinks are assumed with conservatively low heat transfer rates. (See Tables 3.5-11,and 3.5-12)

12. The operation of one containment spray heat exchanger (UA = 3.107 x 10^6 Btu/hr-°F), for containment cooling and the operation of one RHR heat exchanger (UA = 2.22 x 10^6 Btu/hr-°F) for core cooling. The component cooling heat exchanger was modeled at 3.58 x 10^6 Btu/hr-°F.

13. The air return fan returns air at a rate of 39,000 cfm from the upper to the lower compartment.

14. An active sump volume of 40,600 ft^3 is used.

15. 102% of 3588 MWt power is used in the calculations.

16. Subcooling of ECCS water from the RHR heat exchanger is assumed.

17. Essential service water flow to the containment spray heat exchanger was modeled as 2000 gpm. Also the essential service water flow to the component cooling heat exchanger was modeled as 5000 gpm.

18. RHR Spray initiation is assumed after switchover from injection to recirculation has been completed and containment pressure is greater than or equal to 8 psig.

3.5.3.3 Structural Heat Removal

Provision is made in the containment pressure analysis for heat storage in interior and exterior walls. Each wall is divided into a number of nodes. For each node, a conservation of energy equation expressed infinite difference forms accounts for transient conduction into and out of the containment structural heat sinks used in the analysis. The material property data used is found in Tables 3.5-11 and 3.5-12.

The heat transfer coefficient to the containment structure is based primarily on the work of Tagami [Reference 2]. When applying the Tagami correlations, a conservative limit was placed on the lower compartment stagnant heat transfer coefficients. They were limited to a steam-air ratio of 1.4 according to the Tagami correlation. The imposition of this limitation is to restrict the use of the Tagami correlation within the test range of steam-air ratios where the correlation was derived.

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure below the design pressure.

3.5.3.4 Analysis Results

The results of the analysis shows that the maximum calculated containment pressure is 11.66 psig, for the double-ended pump suction minimum safeguards break case. This pressure peak occurs at approximately 9005 seconds, with ice bed meltout at approximately 5453 seconds.

The following plots show the containment integrity transient, as calculated by the LOTIC-1 code.

Figure 3.5-1, Containment Pressure Transient
Figure 3.5-2, Upper Compartment Temperature Transient
Figure 3.5-3, Lower Compartment Temperature Transient
Figure 3.5-4, Active and Inactive Sump Temperature Transient
Figure 3.5-5, Ice Melt Transient

Tables 3.5-9 and 3.5-10 give energy accountings at various points in the transient.

3.5.3.5 Relevant Acceptance Criteria

The LOCA mass and energy analysis has been performed in accordance with the criteria shown in the Standard Review Plan (SRP) Section 6.2.1.3. In this analysis, the relevant requirements of General Design Criteria (GDC) 50 and 10 CFR Part 50 Appendix K have been included by confirmation that the calculated pressure of 11.66 psig is less than the design pressure of 12 psig, and because all available sources of energy have been included, which is more restrictive than the old GDC criteria, Appendix H of the original FSAR, to which the Donald C. Cook Nuclear Plant Units 1 and 2 are licensed. These sources include: reactor power, decay heat, core stored energy, energy stored in the reactor vessel and internals, and stored energy in the secondary system.

The containment integrity peak pressure analysis has been performed in accordance with the criteria shown in the SRP section 6.2.1.1.b, for ice condenser containments. Conformance to GDC's 16, 38, and 50 is demonstrated by showing that the containment design pressure is not exceeded at any time in the transient. This analysis also demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a LOCA.

3.5.3.6 Conclusions

Based upon the information presented, it may be concluded that operation with the revised plant conditions and increased operating margins noted in Section 3.5.2.2 for Donald C. Cook Nuclear Plant Units 1 and 2 is acceptable. Operation with the RHR crosstie valve closed was also shown

to be more limiting than operation with the valve open since there is less safety injection water available for steam condensation. Operation with the revised plant conditions, increased operating margins and the RHR crosstie valve closed results in a calculated peak containment pressure of 11.66 psig, as compared to the design pressure of 12.0 psig. Thus, the most limiting case has been considered, and has been demonstrated to yield acceptable results.

3.5.3.7 References

1. "Long Term Ice Condenser Containment Code - LOTIC Code", WCAP-8354-P-A, April 1976 (Proprietary), WCAP-8355-A (Non-Proprietary).

2. Tagami, Takashi, Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June, 1965 (No. 1).

3. "Final Report Ice Condenser Full-Scale Section Tests at the Waltz Mill Facility," WCAP-8282-P, February, 1974 (Proprietary), WCAP-8110, Supplement 6 (Non-Proprietary).

3.5.4 Main Steam Line Break Containment Integrity

A series of main steam line split and double-ended breaks were analyzed as part of the Rerating Program for Cook Nuclear Plant Units 1 and 2 to determine the most severe break condition for containment temperature and pressure response for this design basis event. The analysis and evaluation conducted are discussed in Reference 1.

Subsequently an analysis consistent with Reference 1 was performed as a part of the 30% Steam Generator Tube Plugging Program (SGTP), to demonstrate that the peak containment temperature resulting from a design basis main steam line break will not exceed the equipment qualification criterion for the plant (Reference 3). For this analysis in particular, the peak containment temperature was calculated using the LOTIC-3 computer code (Reference 4). The analysis was performed to bound Cook Nuclear Plant Units 1 and 2 operation under uprated conditions (3600 Mwt NSSS). Main steam line break is not limiting for containment pressure. The containment pressure response generated for the Loss of Coolant Accident, double-ended pump suction break (Section 3.5.3) is calculated to be more severe. The calculated peak containment pressure from a postulated LOCA was 11.66 psig.

The results from the analyses show that the worst case of the double-ended breaks was a 1.4 square foot break, occurring at 102% power with a main steam isolation valve failure. The worst case for the split breaks was the 0.942 square foot break, occurring at 30% power, with a main steam isolation failure. The calculated peak containment temperature was 322.7°F and 326°F respectively.

The main steam line break containment integrity analysis has been performed consistent with the current licensing basis analysis, considering the bounding SGTP and Uprating operating plant conditions. The results of this analysis show that the Environmental Acceptance Criteria (Reference 3) applicable for Cook Nuclear Plant Units 1 and 2 is met. This analysis therefore demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a main steam line break accident. General Design Criteria (GDC) 50 and 10CFR Part 50 Appendix K are satisfied, which is more restrictive than the old GDC criteria, Appendix H of the original FSAR, to which Donald C. Cook Nuclear Plant Units 1 and 2 are licensed.

3.5.4.1 References

1. WCAP-11902, Supplement 1, September 1989, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 & 2 Licensing Report:"

2. WCAP-14285, "Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report," April 1995.

3. AEP/W SGTP-19, "Donald C. Cook Nuclear Plant Steam Generator Tube Plugging Analysis Technical Documentation Transmittal," August 10, 1994

4. WCAP-8354-P-A (Proprietary) Supplement 2, "Long Term Ice Condenser Containment Code - LOTIC-3 Code," February 1979.

TABLE 3.5-1

DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
SYSTEM PARAMETERS
INITIAL CONDITIONS

PARAMETERS	VALUE
Core Thermal Power (MWt)	3588
Reactor Coolant System Flowrate, per Loop (gpm)	79000
Vessel Outlet Temperature* (°F)	615.2
Core Inlet Temperature* (°F)	547.4
Vessel Average Temperature (°F)	581.3
Initial Steam Generator Steam Pressure (psia)	858.2
Steam Generator Design	Model 54F
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	115216
Accumulator	
Water Volume (ft^3)	946
N_2 Cover Gas Pressure (psia)	600
Temperature (°F)	120
Safety Injection Delay (sec) (includes time to reach pressure setpoint)	48.0

* (analysis value includes an additional +5.1°F allowance for instrument error and deadband)

TABLE 3.5-2

DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
SAFETY INJECTION FLOW
Minimum SI

INJECTION MODE

RCS Pressure (psig)	Total Flow (gpm)
0	3635.5
20	3447.2
40	3235.3
60	3003.7
80	2738.0
100	2425.6
120	2041.3
140	1493.3
160	889.5
180	883.0
200	876.4

RECIRCULATION MODE
(w/o RHR Spray)

Total Flow (gpm)
2983.4

RECIRCULATION MODE
(w/ RHR Spray)

Total Flow (gpm)
764

TABLE 3.5-3
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MINIMUM SI
BLOWDOWN MASS AND ENERGY RELEASE

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
.000	.0	.0	.0	.0
.100	41063.7	22455.0	21836.2	11889.1
.200	46773.1	25791.2	23712.8	12919.1
.401	47539.6	26808.1	23094.0	12611.5
.500	45637.6	26062.5	22482.7	12287.7
.801	44763.4	26388.1	19980.1	10928.1
1.20	40422.9	24546.3	18825.7	10303.8
1.90	35318.0	22494.3	18472.4	10112.3
2.30	31635.0	20795.7	18119.2	9924.5
2.80	26078.4	17714.8	17109.7	9383.5
3.00	21755.7	14941.8	16724.7	9178.5
3.30	19827.4	13782.0	16157.2	8876.3
3.60	19070.7	13307.0	15595.3	8576.2
4.40	16286.5	11363.0	14276.8	7868.8
5.00	14530.9	10127.1	13540.8	7471.6
6.00	12979.6	8984.2	12719.8	7024.4
6.20	12791.3	8833.9	13250.0	7322.4
6.80	12608.4	8631.1	13208.3	7297.1
8.00	12820.0	8620.2	12515.3	6921.0
8.40	12522.1	8507.3	12479.1	6902.8
8.80	10215.2	7667.3	12192.9	6742.9
9.00	9873.8	7519.2	12094.2	6688.4
11.0	9446.5	6883.0	10831.1	5988.5
12.0	8677.2	6369.0	10239.6	5660.1
13.6	7269.2	5640.1	9325.4	5154.1
16.2	5577.4	4692.3	7900.5	4372.7
18.0	4560.9	3971.7	6712.3	3509.6
18.2	4462.8	3909.8	7442.1	3855.6
18.4	4360.8	3858.0	6637.8	3358.0
18.6	4237.6	3796.0	10284.5	5297.0
18.8	4080.2	3719.3	7210.8	3809.4
19.0	4095.4	3808.5	7232.9	3667.2
19.2	3925.1	3759.2	9357.4	4860.9
19.4	3771.2	3752.3	4772.7	2460.8
19.6	3600.7	3690.9	8045.6	4006.4
19.8	3361.3	3597.2	5990.8	3052.9
20.2	2889.2	3382.0	6352.5	3041.3
20.4	2734.9	3288.8	4367.5	2004.2
20.6	2477.7	3029.1	5536.5	2445.5
20.8	2290.9	2820.4	3937.2	1744.6
21.2	1975.2	2447.1	3666.5	1520.1
21.8	1598.7	1993.2	4648.0	1809.2
22.4	1273.7	1595.8	4928.6	1871.1
22.8	1111.6	1396.1	3312.1	1222.0
23.2	990.3	1246.6	2508.0	887.2
24.4	434.8	549.9	2433.4	684.1
25.2	221.7	282.0	2659.9	729.4
26.0	101.5	129.7	2336.8	649.2
27.4	27.9	36.0	280.3	85.8
29.6	.0	.0	85.5	37.7

TABLE 3.5-4
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MINIMUM SI
REFLOOD MASS AND ENERGY RELEASE

TIME	BREAK PATH NO. 1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
29.6	.0	.0	.0	.0
29.9	119.6	140.7	1759.5	157.5
30.1	121.5	142.7	1775.4	158.9
30.3	109.8	129.0	1763.4	157.8
31.1	106.4	124.9	1711.9	153.2
31.3	105.7	124.1	1703.8	152.5
31.7	118.6	139.3	1677.4	150.1
32.4	124.2	145.9	1643.5	147.1
35.7	146.3	171.9	1499.2	134.2
37.7	158.2	185.9	1427.8	127.8
39.7	169.1	198.8	1364.7	122.1
40.7	191.9	225.7	1673.5	170.7
41.7	341.1	402.4	3712.4	513.3
42.7	367.2	433.5	3993.9	576.0
43.7	365.3	431.3	3974.1	578.6
47.7	342.9	404.6	3738.4	552.3
48.7	363.3	428.9	3982.5	572.5
50.7	353.4	417.1	3879.2	560.3
52.7	344.2	406.1	3781.9	548.8
54.7	335.6	395.9	3690.1	537.9
56.7	327.6	386.4	3603.6	527.6
58.7	320.1	377.5	3521.7	517.9
60.7	313.1	369.1	3444.2	508.7
62.7	306.5	361.3	3370.7	499.9
63.7	407.9	481.7	281.1	215.6
64.7	449.8	532.1	299.4	241.3
68.7	417.3	493.3	284.6	221.6
70.7	402.3	475.3	277.8	212.6
74.7	375.5	443.3	265.6	196.7
78.7	351.2	414.4	254.8	182.4
81.7	334.5	394.5	247.3	172.7
88.7	299.9	353.3	232.1	152.9
92.7	282.5	332.7	224.5	143.1
96.7	266.8	314.1	217.8	134.4
100.7	252.6	297.3	211.7	126.6
104.7	239.8	282.1	206.3	119.6
106.7	233.9	275.1	203.8	116.4
110.7	223.0	262.2	199.3	110.6
111.2	221.7	260.7	198.7	109.9
118.7	204.6	240.4	191.6	100.8
126.7	190.0	223.1	185.7	93.2
134.7	178.7	209.8	181.2	87.4
144.7	168.3	197.5	177.1	82.1
148.7	165.1	193.7	175.9	80.5
158.7	158.1	186.3	173.5	77.3
170.7	154.0	180.6	171.7	74.7
182.7	151.4	177.5	170.7	73.3
196.7	150.0	175.9	170.3	72.4
220.7	150.1	175.9	170.3	71.9
258.6	152.7	178.8	171.5	72.6

TABLE 3.5-5
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MINIMUM SI
PRINCIPAL PARAMETERS DURING REFLOOD

TIME	FLOODING TEMP	FLOODING RATE	CARRYOVER FRACTION	CORE HEIGHT	DOWNCOMER HEIGHT	FLOW FRACTION	INJECTION TOTAL	INJECTION ACCUMULATOR	INJECTION SPILL	ENTHALPY
SECONDS	DEGREE F	IN/SEC		FT	FT		(POUNDS MASS PER SECOND)			BTU/LBM
29.6	227.8	.000	.000	.00	.00	.250	.0	.0	.0	.00
30.1	224.8	19.505	.000	.55	.34	1.000	7098.2	7098.2	.0	89.50
30.5	222.8	8.224	.000	1.04	.55	1.000	7003.1	7003.1	.0	89.50
30.8	222.4	3.723	.013	1.16	.96	1.000	6934.0	6934.0	.0	89.50
31.0	222.4	4.661	.035	1.23	1.25	1.000	6889.0	6889.0	.0	89.50
32.2	222.6	2.523	.318	1.50	3.04	.595	6605.9	6605.9	.0	89.50
34.7	224.0	2.115	.533	1.76	6.89	.485	6152.4	6152.4	.0	89.50
38.0	225.9	2.197	.642	2.00	11.73	.443	5665.0	5665.0	.0	89.50
42.7	228.3	3.482	.713	2.33	15.99	.577	4695.1	4695.1	.0	89.50
44.9	229.4	3.332	.730	2.50	16.00	.574	4496.2	4496.2	.0	89.50
45.7	229.8	3.279	.735	2.58	16.00	.572	4434.7	4434.7	.0	89.50
47.7	230.9	3.162	.744	2.70	16.00	.568	4290.1	4290.1	.0	89.50
48.7	231.4	3.271	.747	2.77	16.00	.580	4560.6	4137.0	.0	87.97
52.3	233.5	3.109	.756	3.01	16.00	.574	4343.0	3916.0	.0	87.88
60.8	238.8	2.832	.768	3.50	16.00	.561	3922.0	3488.4	.0	87.67
62.7	240.0	2.782	.770	3.61	16.00	.558	3841.4	3406.6	.0	87.63
63.7	240.6	3.452	.772	3.66	15.95	.623	419.3	.0	.0	72.99
64.7	241.2	3.639	.769	3.73	15.76	.627	404.4	.0	.0	72.99
68.8	243.6	3.368	.772	4.00	15.01	.624	412.4	.0	.0	72.99
77.7	244.3	2.905	.775	4.53	13.72	.619	424.6	.0	.0	72.99
87.0	242.1	2.535	.774	5.00	12.75	.612	433.9	.0	.0	72.99
98.7	243.1	2.172	.773	5.52	11.96	.603	442.3	.0	.0	72.99
111.2	244.3	1.891	.772	6.00	11.50	.592	448.4	.0	.0	72.99
126.7	243.0	1.661	.769	6.52	11.32	.580	453.2	.0	.0	72.99
142.4	244.3	1.515	.770	7.00	11.41	.570	456.4	.0	.0	72.99
160.7	243.3	1.422	.769	7.52	11.73	.564	459.1	.0	.0	72.99
178.9	244.3	1.372	.771	8.00	12.17	.561	461.1	.0	.0	72.99
198.7	243.6	1.351	.773	8.51	12.72	.561	462.8	.0	.0	72.99
218.0	244.3	1.341	.776	9.00	13.29	.563	464.3	.0	.0	72.99
224.7	244.1	1.340	.777	9.17	13.49	.563	464.8	.0	.0	72.99
226.7	244.0	1.340	.777	9.22	13.55	.563	464.9	.0	.0	72.99
238.7	243.9	1.340	.779	9.52	13.90	.565	465.8	.0	.0	72.99
258.6	244.3	1.343	.783	10.00	14.50	.568	467.1	.0	.0	72.99

TABLE 3.5-6
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE MINIMUM SI
POST REFLOOD MASS AND ENERGY RELEASE

TIME	BREAK PATH NO. 1 FLOW		BREAK PATH NO 2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
258.7	202.6	253.5	283.6	104.1
263.7	203.0	254.0	283.2	103:9
268.7	202.2	253.1	283.9	103.9
273.7	202.6	253.6	283.5	103.7
278.7	201.8	252.6	284.3	103.8
283.7	202.2	253.0	283.9	103.6
288.7	201.4	252.1	284.7	103.7
298.7	202.1	252.9	284.1	103.3
303.7	201.3	251.9	284.9	103.4
308.7	201.6	252.2	284.6	103.2
313.7	200.7	251.2	285.4	103.3
318.7	201.0	251.5	285.1	103.1
323.7	200.2	250.5	286.0	103.2
333.7	200.6	251.0	285.5	102.9
338.7	199.7	249.9	286.4	102.9
348.7	200.1	250.4	286.1	102.6
353.7	199.2	249.3	287.0	102.7
363.7	199.4	249.6	286.7	102.4
368.7	198.5	248.4	287.7	102.5
383.7	198.6	248.6	287.5	102.2
388.7	197.6	247.3	288.5	102.3
408.7	197.7	247.4	288.4	101.8
413.7	196.8	246.3	289.4	101.9
428.7	196.8	246.3	289.4	101.6
453.7	196.2	245.5	290.0	101.2
478.7	194.7	243.6	291.5	101.0
483.7	195.1	244.2	291.0	100.8
493.7	194.1	243.0	292.0	100.8
508.7	193.9	242.7	292.2	100.5
533.7	192.6	241.1	293.5	100.2
538.7	193.0	241.5	293.1	100.0
553.7	191.8	240.0	294.3	100.0
563.7	191.9	240.1	294.2	99.7
588.7	190.5	238.4	295.6	99.5
593.7	190.6	238.5	295.5	99.4
618.7	189.4	237.0	296.8	99.1
628.7	189.0	236.6	297.1	98.9
633.7	85.7	107.3	400.4	120.8
901.0	85.7	107.3	400.4	120.8
901.1	83.2	100.0	403.0	98.2
903.7	83.1	103.6	403.0	120.6
1853.7	69.3	80.5	416.8	83.3
1857.0	69.3	86.3	106.2	99.9
2096.9	69.3	86.3	106.2	99.9
2097.0	67.1	83.6	324.9	158.9
2300.6	67.1	83.6	324.9	158.9

m:\2422w-6.wpf:1b-122895

TABLE 3.5-7
DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SI

MASS BALANCE

	TIME (SECONDS)	.00	29.60	29.60	258.63	901.08	2300.57
		MASS (THOUSAND LBM)					
INITIAL	IN RCS AND ACC	770.60	770.60	770.60	770.60	770.60	770.60
ADDED MASS	PUMPED INJECTION	.00	.00	.00	95.14	407.42	1068.61
	TOTAL ADDED	.00	.00	.00	95.14	407.42	1068.61
*** TOTAL AVAILABLE ***		770.60	770.60	770.60	865.74	1178.02	1839.21
DISTRIBUTION	REACTOR COOLANT	536.60	68.20	68.59	136.89	136.89	136.89
	ACCUMULATOR	234.00	159.36	158.96	.00	.00	.00
	TOTAL CONTENTS	770.60	227.56	227.55	136.89	136.89	136.89
EFFLUENT	BREAK FLOW	.00	543.03	543.03	728.84	1041.12	1702.15
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	543.03	543.03	728.84	1041.12	1702.15
TOTAL ACCOUNTABLE		770.60	770.59	770.58	865.73	1178.01	1839.04

m:\2422w-6.wpf:1b-122895

TABLE 3.5-8
DONALD C COOK NUCLEAR PLANT UNITS 1 AND 2
DOUBLE-ENDED PUMP SUCTION GUILLOTINE
MINIMUM SI

ENERGY BALANCE

TIME (SECONDS)		.00	29.60	29.60	258.63	901.08	2300.57
		ENERGY (MILLION BTU)					
INITIAL ENERGY	IN RCS,ACC,S GEN	903.70	903.70	903.70	903.70	903.70	903.70
ADDED ENERGY	PUMPED INJECTION	.00	.00	.00	6.94	29.76	89.17
	DECAY HEAT	.00	9.59	9.59	36.80	93.76	190.09
	HEAT FROM SECONDARY	.00	-5.57	-5.57	-5.57	-2.56	3.45
	TOTAL ADDED	.00	4.02	4.02	38.17	120.96	282.71
*** TOTAL AVAILABLE ***		903.70	907.72	907.72	941.87	1024.66	1186.41
DISTRIBUTION	REACTOR COOLANT	318.14	14.10	14.14	29.63	29.63	29.63
	ACCUMULATOR	20.94	14.26	14.23	.00	.00	.00
	CORE STORED	28.93	14.50	14.50	3.19	3.16	2.92
	PRIMARY METAL	179.10	168.07	168.07	144.87	92.62	63.42
	SECONDARY METAL	84.70	84.39	84.39	78.14	59.18	35.55
	STEAM GENERATOR	271.89	275.60	275.60	251.90	187.85	115.49
	TOTAL CONTENTS	903.70	570.92	570.93	507.73	372.44	247.01
EFFLUENT	BREAK FLOW	.00	336.32	336.32	425.89	643.98	896.91
	ECCS SPILL	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	336.32	336.32	425.89	643.98	896.91
TOTAL ACCOUNTABLE		903.70	907.24	907.25	933.62	1016.42	1143.92

TABLE 3.5-9
ENERGY ACCOUNTING IN MILLIONS OF BTU

	Approx. End of Blowdown (t = 10.0 sec.)	Approx. End of Reflood (t = 258.6 sec.)
Ice Heat Removal*	201.7	241.3
Structural Heat Sinks*	21.04	59.88
RHR Heat Exchanger Heat Removal*	0	0
Spray Heat Exchanger Heat Removal*	0	0
Energy Content of Sump	186.0	247.65
Ice Melted (Pounds) (10^6)	0.65	0.81

* Integrated Energies

TABLE 3.5-10
ENERGY ACCOUNTING IN MILLIONS OF BTU

	Approx. Time of Ice Melt Out (t = 5453 sec.)	Approx. Time of Peak Pressure (t = 9005 sec.)
Ice Heat Removal*	566.18	566.18
Structural Heat Sinks*	104.82	161.16
RHR Heat Exchanger Heat Removal*	53.07	97.46
Spray Heat Exchanger Heat Removal*	62.0	115.31
Energy Content of Sump	566.51	589.74
Ice Melted (Pounds)(10^6)	2.11	2.11

* Integrated Energies

TABLE 3.5-11
STRUCTURAL HEAT SINK TABLE

SURFACES		AREA (Ft^2)	THICKNESS (Ft)
Upper Compartment Material			
1.	Paint (topcoat)	29958.25	0.0008333
	Paint (undercoat)		0.00025
	Carbon Steel		0.029741
	Concrete		3.0364
2.	Paint (topcoat)	12571.4	0.0004167
	Paint (undercoat)		0.000833
	Concrete		2.710421
3.	Paint (topcoat)	15526.8	0.0004167
	Paint (undercoat)		0.000833
	Concrete		2.2728
4.	Paint (topcoat)	1306.25	0.0008333
	Paint (undercoat)		0.00025
	Carbon Steel		0.209107
5.	Paint (topcoat)	4207.55	0.0008333
	Paint (Undercoat)		0.00025
	Carbon Steel		0.064932
6.	Paint (topcoat)	22443.75	0.0008333
	Paint (undercoat)		0.00025
	Carbon Steel		0.017571
7.	Paint (topcoat)	30014.3	0.0008333
	Paint (undercoat)		0.00025
	Carbon Steel		0.0102872
Lower Compartment Material			
8.	Paint (topcoat)	6734.55	0.0008333
	Paint (undercoat)		0.00025
	Carbon Steel		0.0167
	Concrete		1.0103

TABLE 3.5-11(continued)
STRUCTURAL HEAT SINK TABLE

9.	Paint (topcoat)	14642.35	0.0004167
	Paint (undercoat)		0.000833
	Concrete		5.8355
10.	Paint (topcoat)	25872.3	0.0004167
	Paint (undercoat)		0.000833
	Concrete		2.699
11.	Paint (topcoat)	3214.8	0.0008333
	Paint (undercoat)		0.00025
	concrete		0.092852
12.	Paint (topcoat)	3499.8	0.008333
	Paint (undercoat)		0.00025
	Carbon Steel		0.069179
13.	Paint (topcoat)	12312.0	0.0008333
	Paint (undercoat)		0.00025
	Carbon Steel		0.013136
14.	Paint (topcoat)	61696.8	0.0008333
	Paint (undercoat)		0.00025
	Carbon Steel		0.0089655
Ice Condenser			
15.	Steel	180600.	0.00663
16.	Steel	76650.	0.0217
17.	Steel	28670.	0.0267
18.	Paint	3336.	0.000833
	Concrete		0.333
19.	Steel and Insulation	19100.	1.0
	Steel		0.0625
20.	Steel and Insulation	13055.	1.0
	Concrete		1.0

TABLE 3.5-12
MATERIAL PROPERTIES TABLE

Material	Conductivity Btu/hr - ft -°F	Volumetric Heat Capacity Btu/ft^3 - °F
Upper and Lower Compartments		
Concrete	0.81	30.4
Steel	26.0	58.8
Paint (concrete)		
Undercoat	0.19	29.3
Topcoat	0.19	75.0
Paint (steel)		
Undercoat	0.4	29.3
Topcoat	0.4	75.0
Ice Condenser Compartment		
Paint (concrete)	0.0833	28.4
Insulation (steel)	0.15	2.75
Insulation (concrete)	0.2	3.663

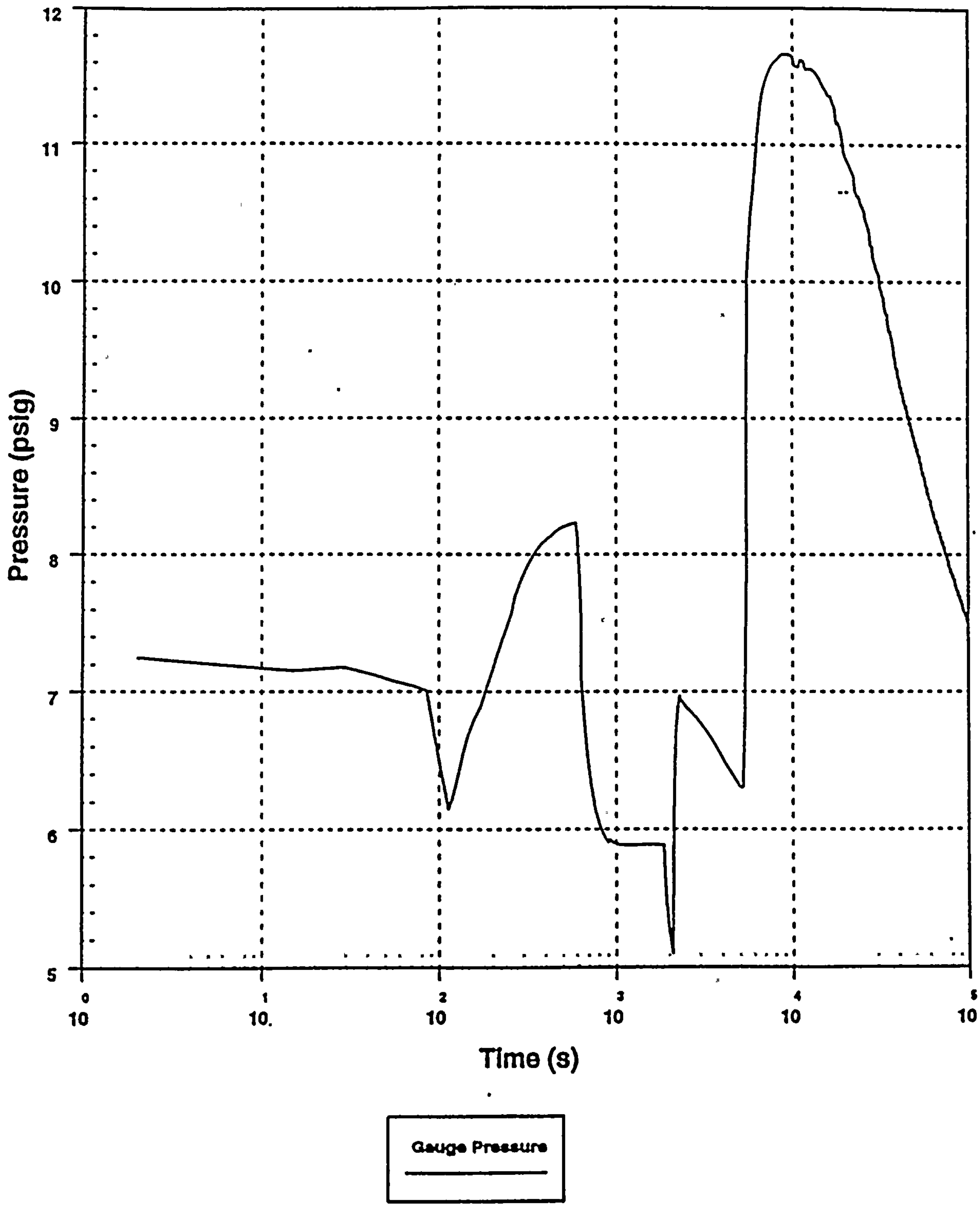


Figure 3.5-1 LOCA Mass and Energy Release Containment Integrity Containment Pressure

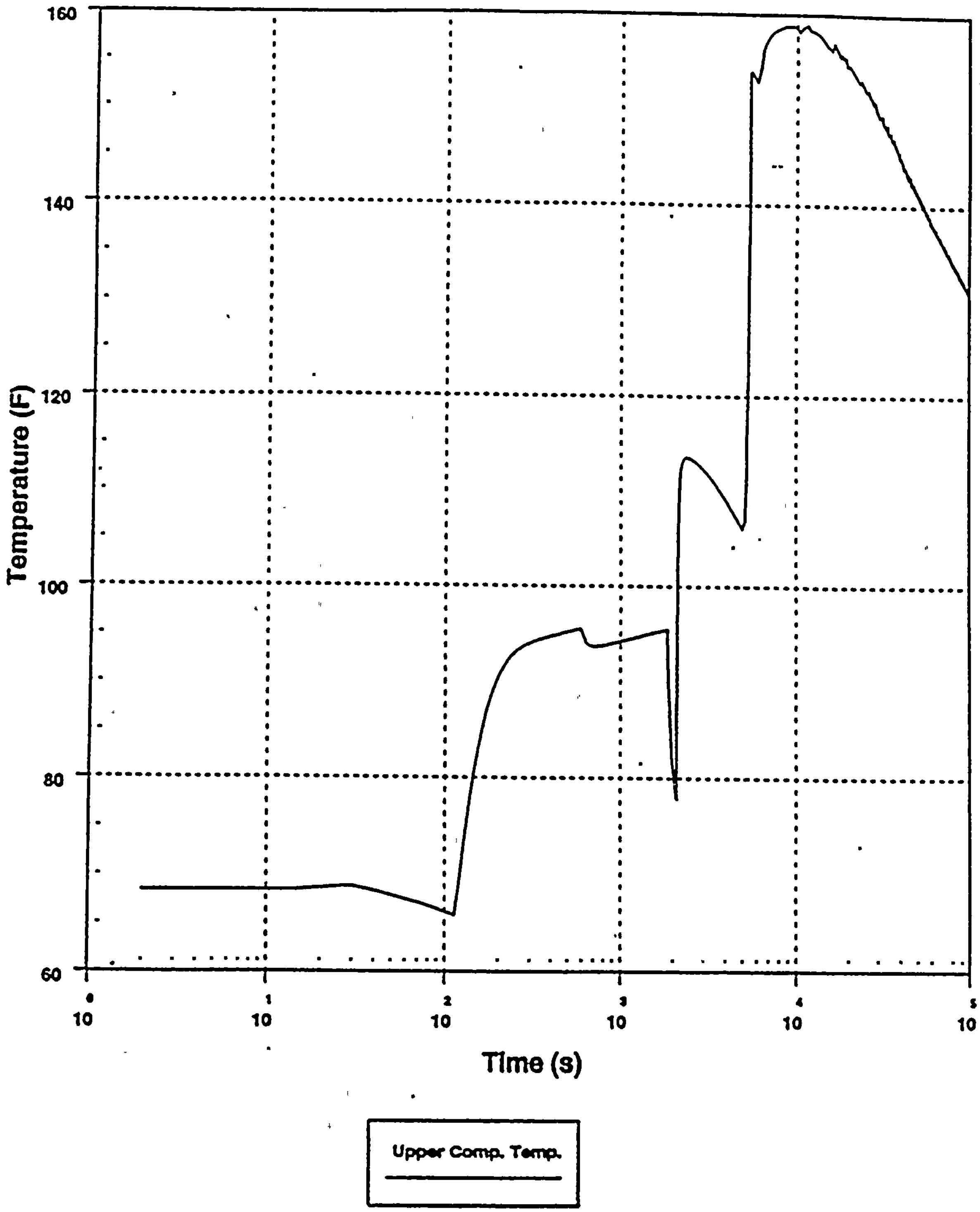


Figure 3.5-2 **LOCA Mass and Energy Release Containment Integrity Upper Compartment Temperature**

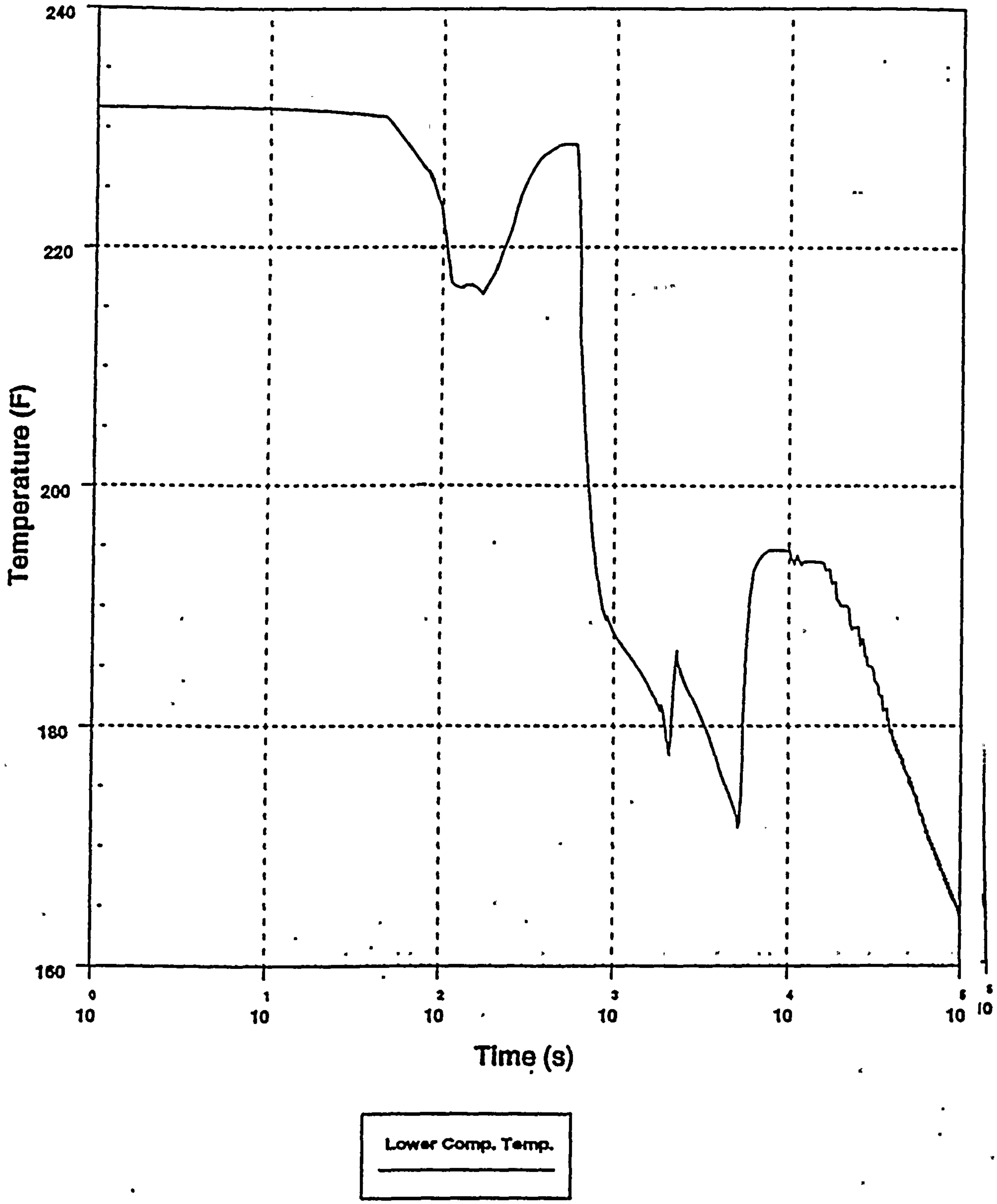


Figure 3.5-3 LOCA Mass and Energy Release Containment Integrity Lower Compartment Temperature

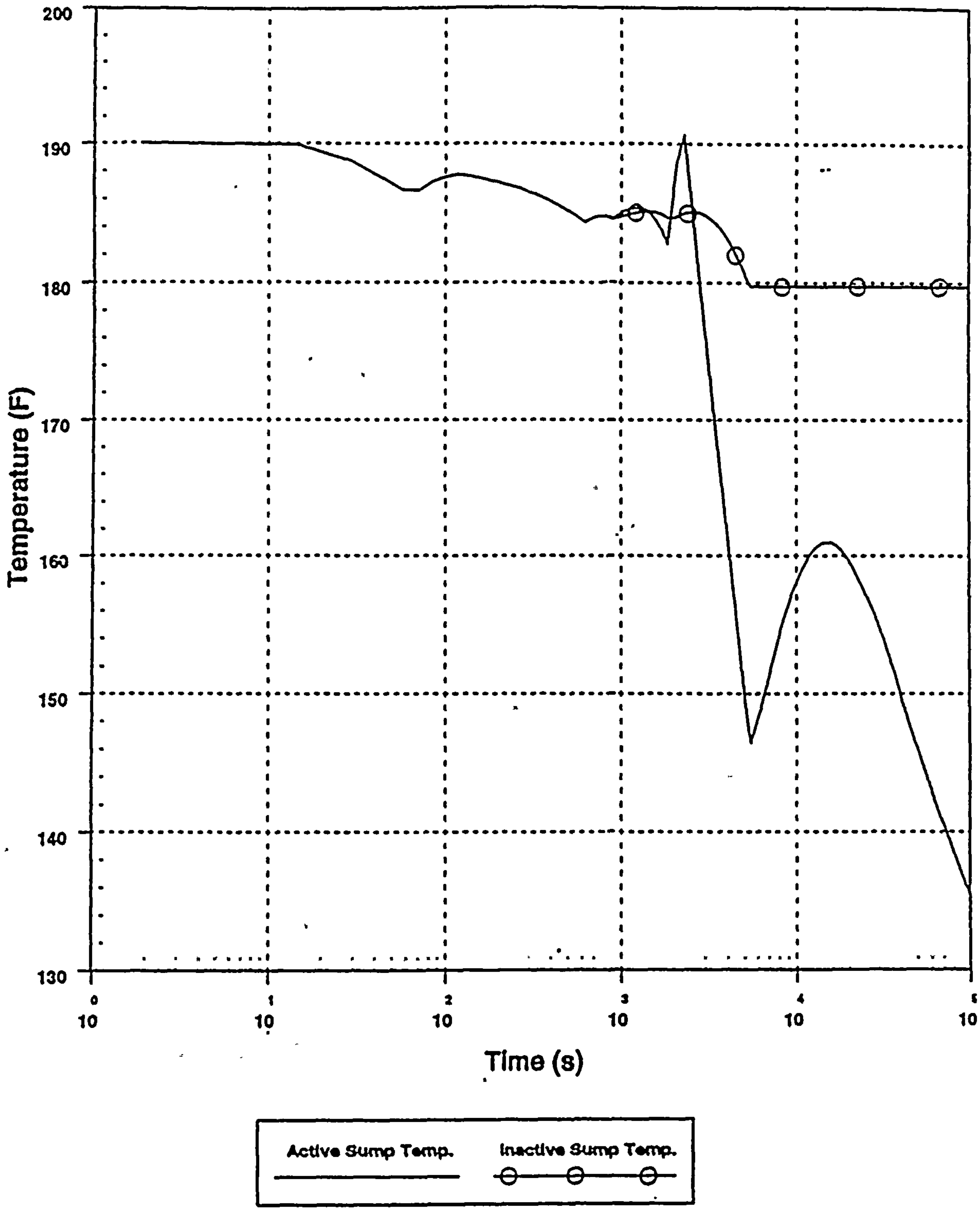


Figure 3.5-4 LOCA Mass and Energy Release Containment Integrity Active Sump and Inactive Sump Temperature Transient

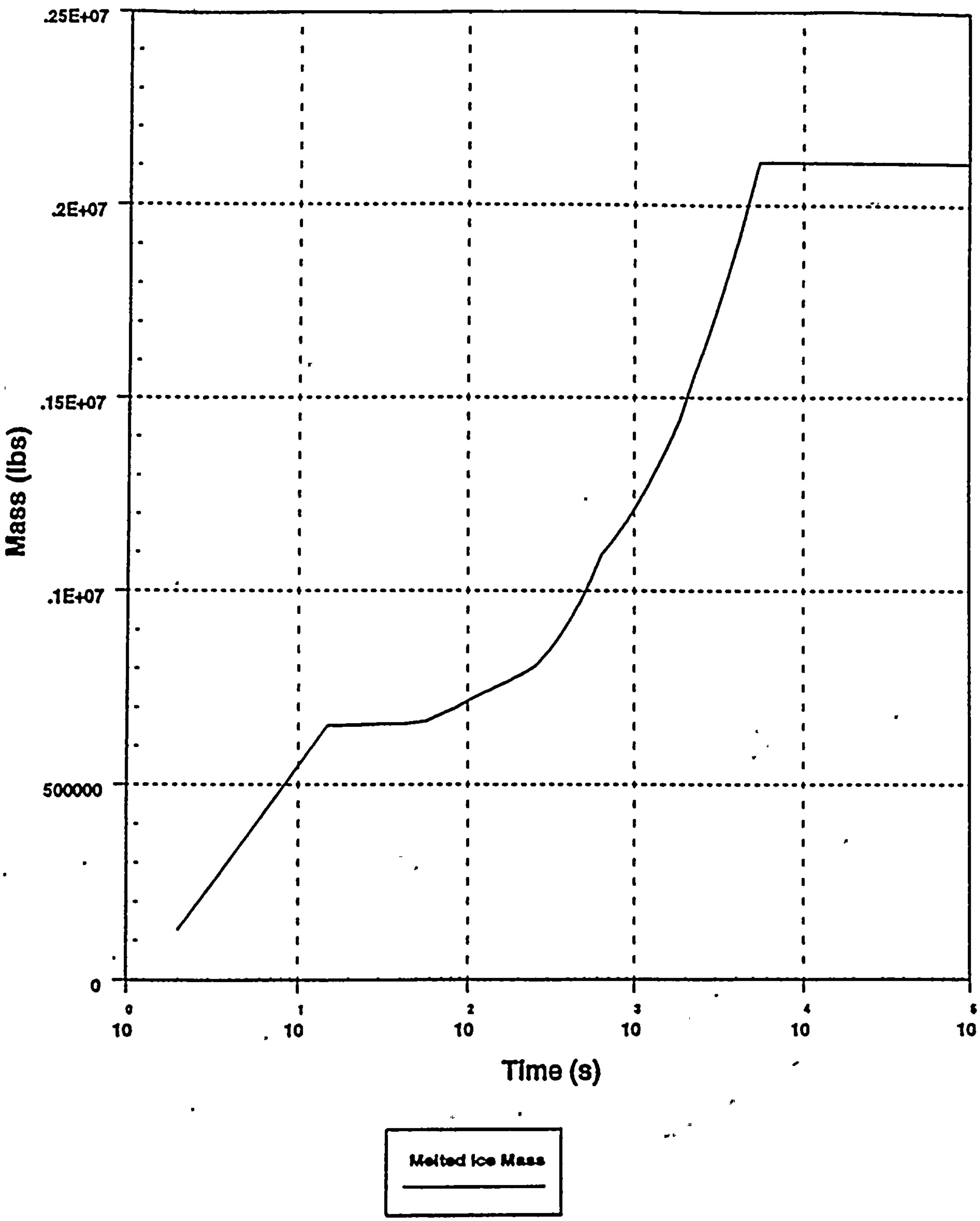


Figure 3.5-5 LOCA Mass and Energy Release Containment Integrity Ice Melt Transient

3.6 STEAM GENERATOR TUBE RUPTURE ACCIDENT ANALYSIS

The UFSAR analysis of a steam generator tube rupture (SGTR) transient is performed to conservatively predict the radiological consequences of such an event. An evaluation of this transient, supporting an increase in NSSS power to 3600 MWt for Donald C. Cook Nuclear Plant Unit 2, has been completed to determine the impact on the dose releases. In addition, plant changes implemented at Donald C. Cook Nuclear Plant Unit 2 since the Rerating Program was completed were reviewed to assess their impact on the SGTR design basis assumptions. These plant changes included the reduced RHR and SI pump head.

The SGTR analysis to support an uprate to 3600 MWt at Cook Nuclear Plant Unit 2 was performed in 1988 and presented in the UFSAR to ensure that the offsite radiation doses remain below the limits defined in 10 CFR 100. The principal thermal-hydraulic parameters which affect the offsite radiation doses for an SGTR accident are the amount of reactor coolant transferred to the secondary side of the ruptured steam generator and the amount of steam released from the ruptured steam generator to the atmosphere. A review of the SGTR analysis performed for the Rerating Program indicates that the analysis was based on NSSS performance parameters supporting a plant uprate to 3600 MWt and incorporated replacement steam generators, T_{avg} range from 547°F to 581.3°F, a steam generator tube plugging level of 15%, a thermal design flow of 88500 gpm/loop, and two distinct primary side pressures, namely, 2100 psia and 2250 psia. Since these NSSS performance parameters are identical to those proposed for the Uprating Program, the thermal-hydraulic results of the SGTR analysis remain bounding.

Subsequent to the Rerating Program analysis, an evaluation was performed to determine the impact of an increase in the main steam safety valve (MSSV) setpoint to 3% on the SGTR accident consequences (Reference 1). It was concluded from the results of this evaluation that there was no significant thermal-hydraulic impact and that the SGTR radiological doses continue to be bounded by the original analysis. On this basis, therefore, the thermal-hydraulic analysis completed to support an uprating to 3600 MWt bound both the proposed uprated NSSS performance parameters and the increased MSSV setpoint tolerance.

Plant changes made since completion of the UFSAR SGTR analysis were reviewed and found to have no effect on the SGTR accident consequences.

Reduced RHR Head After reactor trip and safety injection actuation, it is assumed that the RCS pressure stabilizes at the equilibrium point where safety injection flow rate equals the out going break flow rate and that the equilibrium break flow persists until 30 minutes after the initiation of the SGTR. The cut-in pressure for the RHR pumps is below the equilibrium RCS pressure so that there is no flow from these pumps during an SGTR event. Therefore, the reduced RHR pump head have no impact on the SGTR accident consequences.

Reduced HHSI Head With regards to HHSI pumps, it is beneficial to reduce the safety injection flow during an SGTR. A reduction in the HHSI flow results in a lower break flow rate and, therefore, a reduction in the mass accumulation in the ruptured steam generator. For this reason, maximum HHSI flow rates are assumed. Consequently, reducing HHSI head (flow) has no impact on the analysis. Note that maximum safety injection flow rates have not changed.

Therefore, the UFSAR SGTR thermal and hydraulic analysis remains bounding with respect to the Donald C. Cook Nuclear Plant Unit 2 Uprating Program.

Reference

1) SECL-91-429, Rev. 2, "Donald C. Cook Units 1 and 2 Main Steam Safety Valve Lift Setpoint Tolerance Relaxation," December 1993

3.7 POST-LOCA HOT LEG RECIRCULATION TIME

3.7.1 Introduction

The hot leg recirculation time analysis has been updated for Donald C. Cook Nuclear Plant Unit 2 to determine the time when switchover to hot leg recirculation should be initiated following a LOCA. This analysis addresses the concern related to the potential for boron precipitation in the reactor vessel during cold leg recirculation following a LOCA. The analysis was updated to support operation of Unit 2 at the uprated core power level of 3588 MWt with an increase in the RHR and high head safety injection pump head degradation from 10% to 15%.

3.7.2 Event Description

During a large break LOCA the plant switches to cold leg recirculation after the RWST switchover setpoint has been reached. If the break is in the cold leg there is a concern that the cold leg injection water will fail to establish flow through the core. Safety injection entering the broken loop will spill out the break, while safety injection entering the intact cold legs may circulate around the downcomer and out the break. With no flow path established, the core may remain stagnant while steam continues to be produced from the core decay heat. Since not all of the boron is carried off by the steam, the boron concentration increases in the vessel as the water boils off. The boron concentration in the vessel could potentially increase until the solubility limit of the boric acid solution is reached, at which time boron will begin to precipitate. Precipitation may result in boron plating out on the fuel rods which would adversely affect their heat transfer characteristics. Boron precipitation may also block flow channels which would adversely affect the ability of the ECCS to ensure long term core cooling. Switching to hot leg recirculation provides flow to the core which terminates the boron buildup transient by diluting the vessel boron concentration through natural circulation of the incoming safety injection, reverse flushing of the core, and/or terminating boiling in the core.

3.7.3 Methodology

The analysis requires the identification of all sources of borated water in the sump following a LOCA and their pre-accident boron concentrations. Since the analysis considers the build-up of boric acid in the vessel, conservative assumptions are used which maximize the sump boron concentration. The boron concentration in the RWST and accumulators was assumed to be at the Technical Specification maximum of 2600 ppm for Cook Unit 2. The initial RCS boron concentration was conservatively assumed to be at the RWST Technical Specification minimum of 2400 ppm, and the boron concentration of the ice mass in the ice condenser was assumed to be 2300 ppm. The calculation of the hot leg switchover time is conservatively based on the core decay heat for a LOCA which is assumed to occur during full power operation. This basis results in a switchover time which is bounding for lower operating modes such as startup after a refueling shutdown when the RCS boron concentration may be higher than the RWST Technical Specification minimum value.

The analysis assumes that the coolant in the core is stagnant during cold leg injection and that the steam boil-off from the core does not carry any boron. Thus, as steam is boiled off and boron is left behind, the boric acid concentration of the vessel increases. A conservatively small effective vessel volume is used which includes only the free volumes of the reactor core and upper plenum below the bottom of the hot leg nozzles. This assumption conservatively neglects the mixing of boric acid solution with directly connected volumes, such as the reactor vessel lower plenum. The core boiloff and boric acid concentration in the vessel volume is calculated as a function of time after the LOCA occurs. The hot leg switchover should be initiated when the boric acid solution reaches the solubility limit less 4 weight percent. The solubility limit less 4 weight percent is 23.53% at a solution temperature of 212°F and 14.7 psia. Thus, when the boric acid solution concentration in the vessel reaches 23.53%, hot leg circulation should be initiated.

3.7.4 Results

The results of the updated hot leg switchover analysis for Donald C. Cook Nuclear Plant Unit 2 indicates that the maximum allowable boric acid concentration of 23.53 weight percent will not be exceeded in the vessel if the hot leg recirculation is initiated at 8.5 hours after the LOCA occurs. An assessment also indicates that the cold leg recirculation flow rate is sufficient to meet the requirement for core heat removal when sump recirculation is initiated, and that the hot leg recirculation flow rate is adequate to meet the recirculation requirement when switchover to hot leg recirculation is performed.

3.8 REACTOR CAVITY PRESSURE ANALYSIS

The Reactor Cavity Pressure Analysis is performed to calculate the initial pressure response in the reactor cavity to a loss of coolant accident. The Reactor Cavity Pressure Analysis that was performed for the Rerating Program was reviewed and it was determined that the conclusions provided for the Rerating Program (WCAP-11902 Supplement 1) remain valid for the Uprating Program.

3.9 RADIOLOGICAL ANALYSIS

A review was performed of the radiological analysis in the UFSAR for Cook Nuclear Plant Unit 2 to determine the effects of the Uprating Program. The Loss-of Offsite Power, SGTR, Steamline Break and Fuel Handling Accident radiological consequences have been reanalyzed and these analyses are summarized below. The Large Break LOCA doses were recalculated for the SGTP Program for Donald C. Cook Nuclear Plant Unit 1 (Refer to WCAP-14285) and remain bounding for the Uprating Program. The Large Break LOCA doses do not exceed the acceptance criteria given in 10 CFR 100.

3.9.1 Loss of Off-site Power Radiological Consequences

3.9.1.1 Introduction

A loss of all A.C. power to the plant auxiliaries (i.e., a loss of off-site power) is assumed to occur. Due to primary to secondary SG tube leakage, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is assumed to be released to the outside atmosphere through the atmospheric dump/SG safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this activity is released to atmosphere as a result of steaming of the SGs following the event. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite doses resulting from this release.

Conservative SG steam releases are used such that the off-site doses determined will not only be conservative for a loss of off-site power (LOOP) but also conservative for both a loss of load event and a loss of normal feedwater event. Steam is released to the atmosphere from the steam generators via the atmospheric dump valves. A LOOP is assumed, since this provides direct release to the atmosphere rather than to the main condenser. The steam released is dependent on the decay heat and sensible energy released from the primary and secondary systems while bringing the plant to cold shutdown (RHR conditions). At 2 hours it is assumed that the plant has stabilized at no-load conditions. From 2 hours to 8 hours, the RCS is assumed to be cooled and depressurized to the RHR operating conditions.

3.9.1.2 Input Parameters and Assumptions

The off-site doses following a LOOP were determined considering both existing and concurrent iodine spikes. For the existing iodine spike it is assumed that a reactor transient has occurred prior to the LOOP and has raised the RCS iodine concentration to 60 µCi/gm of dose equivalent (DE) I-131. For the event initiated iodine spike, the reactor trip associated with the LOOP creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the maximum equilibrium RCS Technical Specification concentration of 1.0 µCi/gm of DE I-131. The duration of the accident initiated iodine spike is 1.8 hours.

The noble gas activity concentration in the RCS at the time the event occurs is based on a fuel defect level of 1.0%. The iodine activity concentration of the secondary coolant at the time the LOOP occurs is assumed to be equivalent to the Technical Specification limit of 0.10 μCi/gm of DE I-131.

The amount of primary to secondary SG tube leakage is conservatively assumed to be equal to the Unit 2 Technical Specification limit on total leakage of 1.0 gpm. This leak rate is bounding for Unit 1.

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of off-site power.

An iodine partition factor in the SGs of 0.01 (curies I/gm steam) / (curies I/gm water) is used. All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Eight hours after the event the RHR System is assumed to be placed into service for heat removal, and there are no further steam releases to atmosphere from the secondary system.

The major assumptions and parameters used in the analysis are itemized in Table 3.9-1. The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 3.9-2.

3.9.1.3 Description of Analyses

The TITAN5 computer code is used to calculate the activity release to atmosphere from the SGs and the resulting off-site doses. Both the existing iodine spike and concurrent iodine spike models are analyzed for this release path.

3.9.1.4 Acceptance Criteria

The off-site dose acceptance criteria for a LOOP are a "small fraction of" the guideline values of 10CFR100, or 30 rem thyroid and 2.5 rem whole body.

3.9.1.5 Results

The off-site thyroid and whole body doses due to the LOOP are given in Table 3.9-3.

3.9.1.6 Conclusions

The off-site doses due to the LOOP are less than the acceptance criteria.

3.9.2 Steam Generator Tube Rupture Radiological Consequences

3.9.2.1 Introduction

The complete severance of a single steam generator tube is assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is assumed to be released to the outside atmosphere through the atmospheric dump/SG safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident and some of this activity is released to atmosphere as a result of steaming of the SGs following the accident. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the off-site doses resulting from this release.

3.9.2.2 Input Parameters and Assumptions

The off-site doses following a SGTR were determined considering both pre-accident and accident initiated iodine spikes. For the pre-accident iodine spike it is assumed that a reactor transient has occurred prior to the SGTR and has raised the RCS iodine concentration to 60 µCi/gm of dose equivalent (DE) I-131. For the accident initiated iodine spike, the reactor trip associated with the SGTR creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the maximum equilibrium RCS Technical Specification concentration of 1.0 µCi/gm of DE I-131. The duration of the accident initiated iodine spike is 1.8 hours.

The noble gas activity concentration in the RCS at the time the accident occurs is based on a fuel defect level of 1.0%. The iodine activity concentration of the secondary coolant at the time the SGTR occurs is assumed to be equivalent to the Technical Specification limit of 0.10 µCi/gm of DE I-131.

The amount of primary to secondary SG tube leakage is conservatively assumed to be equal to the Unit 2 Technical Specification limit on total leakage of 1.0 gpm, with 0.25 gpm in the ruptured SG and 0.75 gpm in the 3 intact SGs. This leak rate is also bounding for Unit 1.

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of off-site power.

An iodine partition factor in the SGs of 0.01 (curies I/gm steam) / (curies I/gm water) is used. All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Flow through the ruptured SG tube is assumed to be terminated at 30 minutes following accident initiation.

Eight hours after the accident the RHR System is assumed to be placed into service for heat removal, and there are no further steam releases to atmosphere from the secondary system.

The major assumptions and parameters used in the analysis are itemized in Table 3.9-4. The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 3.9-5.

3.9.2.3 Description of Analyses

The TITAN5 computer code is used to calculate the activity release to atmosphere from the ruptured and intact SGs and the resulting off-site doses. Both the pre-accident iodine spike and accident initiated iodine spike models are analyzed for these release paths.

3.9.2.4 Acceptance Criteria

The off-site dose acceptance criteria for a SGTR with a pre-accident iodine spike are the guideline values of 10CFR100. These guideline values are 300 rem thyroid and 25 rem whole body. For a SGTR with an accident initiated iodine spike the acceptance criteria are a "small fraction of" the 10CFR100 guideline values, or 30 rem thyroid and 2.5 rem whole body.

3.9.2.5 Results

The off-site thyroid and whole body doses due to the SGTR are given in Table 3.9-6.

3.9.2.6 Conclusions

The off-site doses due to the SGTR are less than the acceptance criteria.

3.9.3 Steamline Break Radiological Consequences

3.9.3.1 Introduction

The complete severance of a main steamline outside containment is assumed to occur. The affected SG will rapidly depressurize and release radioiodines initially contained in the secondary coolant and also the primary coolant activity, transferred via SG tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact SGs and the activity due to tube leakage is released to atmosphere through the atmospheric dump/SG safety valves. This section describes the assumptions and analyses performed to determine the amount of radioactivity released.and the offsite doses resulting from this release.

3.9.3.2 Input Parameters and Assumptions

The offsite doses following a steamline break (SLB) were determined considering both pre-accident iodine spike and accident initiated iodine spike. For the pre-accident iodine spike it is assumed that a reactor transient has occurred prior to the SLB and has raised the RCS iodine concentration to 60 µCi/gm of dose equivalent (DE) I-131. For the accident initiated iodine spike the reactor trip associated with the SLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the maximum equilibrium RCS Technical Specification concentration of 1.0 µCi/gm of DE I-131. Although the duration of the accident initiated iodine spike is 1.8 hours, it is conservatively assumed to last for 2.0 hours for this analysis.

The noble gas activity concentration in the RCS at the time the accident occurs is based on a fuel defect level of 1.0%. The iodine activity concentration of the secondary coolant at the time the SLB occurs is assumed to be equivalent to the Technical Specification limit of 0.10 µCi/gm of DE I-131.

For Unit 2, the amount of primary to secondary SG tube leakage is assumed to be equal to the Technical Specification limit on total leakage of 1.0 gpm.

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power.

The SG connected to the broken steamline is assumed to boil dry within the initial two hours following the SLB. The entire liquid inventory of this SG is assumed to be steamed off and all of the iodine initially in this SG is released to the environment. Also, iodine carried over to both the faulted SG and the 3 intact SGs by SG tube leaks is assumed to be released directly to the environment.

An iodine partition factor in the intact SGs of 0.10 (curies I /gm steam)/(curies I/gm water) is used for the iodine initially in the intact SGs which is released to the environment via steaming of the intact SGs.

All noble gas activity carried over to the secondary through SG tube leakage is assumed to be immediately released to the outside atmosphere.

At 8 hours after the accident the RHR System is assumed to be placed into service for heat removal, and there are no further steam releases to atmosphere from the secondary system.

The major assumptions and parameters used in the analysis are itemized in Table 3.9-7. The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 3.9-8.

3.9.3.3 Description of Analyses Performed

The TITAN5 computer code is used to calculate the activity release to atmosphere and the resulting offsite doses. Both the pre-accident iodine spike and accident initiated iodine spike models are analyzed for these release paths.

3.9.3.4 Acceptance Criteria

The offsite dose limits for a SLB with a pre-accident iodine spike are the guideline values of 10CFR100. These guideline values are 300 rem thyroid and 25 rem whole body. For a SLB with an accident initiated iodine spike the acceptance criteria are a "small fraction of" the 10CFR100 guideline values, or 30 rem thyroid and 2.5 rem whole body.

3.9.3.5 Results

The offsite thyroid and whole body doses due to the SLB are given in Table 3.9-9.

3.9.3.6 Conclusions

The offsite doses due to the SLB do not exceed the acceptance criteria.

3.9.4 Fuel Handling Accident Radiological Consequences

3.9.4.1 Introduction

A fuel assembly is assumed to be dropped and damaged during refueling. Analysis of the accident is performed for the accident occurring both inside containment and in the auxiliary building. Activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the auxiliary building ventilation systems. This section describes the assumptions and analyses performed to determine the amount of activity released and the offsite doses resulting from this release.

3.9.4.2 Input Parameters and Assumptions

The offsite doses following a fuel handling accident (FHA) reflect the uprated core power level of 3588 MWt. Also addressed is a 20% increase in the I-131 gap fraction for high burnup fuel (References 1 and 2). The gap fractions applied to the remaining iodine and noble gas isotopes remain as given in Regulatory Guide 1.25 (Reference 3), or 0.10 for these iodine and noble gas isotopes with the exception of 0.30 for Kr-85.

The analysis assumes that all of the fuel rods in the equivalent of one assembly are damaged to the extent that all their gap activity is released.

The reactor has been assumed to be subcritical for 100 hours before fuel is moved; therefore, 100 hours of radioactive decay is assumed in the analysis. Per Technical Specifications, it is assumed that there is a minimum of 23 feet of water above the reactor pressure vessel flange. With this water depth, decontamination factors (DF) of 133 for elemental iodine and 1 for methyl iodine are used for pool scrubbing (Reference 3). The iodine activity in the fuel rod gap is assumed to be 99.75% elemental and 0.25% methyl (Reference 3). The resulting overall pool scrubbing DF for iodine is 100.

All of the noble gas released from the damaged assembly is assumed to be released from the pool water (i.e., the pool scrubbing DF is 1) (Reference 3).

A conservatively high radial peaking factor of 1.65 is assumed for the damaged assembly (Reference 3).

No credit is taken for filtration of iodine for the FHA inside containment. Although the containment purge will be automatically isolated on a purge line high radiation alarm, isolation is not modeled in the analysis. The activity released from the damaged assembly is assumed to be immediately released to the outside atmosphere.

Credit is taken for filtration of iodine for the FHA inside the auxiliary building. The auxiliary building iodine removal efficiency is assumed to be 90% elemental and 70% methyl (Reference 3).

The major assumptions and parameters used in the analysis are itemized in Table 3.9-10. The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 3.9-11.

3.9.4.3 Description of Analyses Performed

The activity releases and offsite doses are determined for both a FHA inside containment and a FHA in the spent fuel pit.

3.9.4.4 Acceptance Criteria

The dose limits for a FHA are appropriately within the guideline values of 10CFR100.

3.9.4.5 Results

The offsite thyroid and whole body doses due to the FHA are given in Table 3.9-12.

3.9.4.6 Conclusions

The offsite doses due to the FHA do not exceed the acceptance criteria.

3.9.5 Summary of Conclusions

The Loss-of-Offsite power, SGTR, Steamline Break, and FHA radiological consequences have been reanalyzed for the Uprating Program. The off-site dose acceptance criteria for a LOOP are a "small fraction of" the guideline values of 10CFR100, or 30 rem thyroid and 2.5 rem whole body. The off-site doses due to the LOOP are less than the acceptance criteria.

Conservative SG steam releases were used such that the off-site doses determined are not only conservative for a LOOP but also conservative for both a loss of load event and a loss of normal feedwater event.

The off-site dose acceptance criteria for an SGTR with a pre-accident iodine spike are the guideline values of 10CFR100. These guideline values are 300 rem thyroid and 25 rem whole body. For a SGTR with an accident initiated iodine spike the acceptance criteria are a "small fraction of" the 10CFR100 guideline values, or 30 rem thyroid and 2.5 rem whole body. The offsite doses due to the SGTR are less than the acceptance criteria.

The offsite dose limits for an SLB with a pre-accident iodine spike are the guideline values of 10CFR100. For an SLB with an accident initiated iodine spike the acceptance criteria are a "small fraction of" the 10CFR100 guideline values. The offsite doses due to the SLB do not exceed the acceptance criteria.

The dose limits for an FHA are appropriately within the guideline values of 10CFR100. The offsite doses due to the FHA do not exceed the acceptance criteria.

The Large Break LOCA doses were recalculated for the SGTP Program (WCAP-14285) and remain bounding for the Uprating Program.

3.9.6 References

1. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors", D. A. Baker, et. al., February 1988.

2. Federal Register/Vol. 53, No. 39/Monday, February 29, 1988/pages 6040 through 6043.

3. US AEC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Safety Guide 25), 3/23/72.

TABLE 3.9-1
ASSUMPTIONS FOR LOOP DOSE ANALYSIS

Core Power Level	3588 MWt
Reactor Coolant Noble Gas Activity Prior to accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	60 µCi/gm of DE I-131
Accident Initiated Spike	1.0 µCi/gm of DE I-131
Iodine Release from Fuel to Reactor Coolant Due to Accident Initiated Spike	500 times equilibrium release rate from fuel for initial 1.8 hours after LOOP
Secondary Coolant Activity Prior to Accident	0.10 µCi/gm of DE I-131
SG Tube Leak Rate per SG During Accident	1.0 gpm
SG Iodine Partition Factor	0.01
Duration of Activity Release from Secondary System	8 hr
Steam Release from SGs to Environment	600,000 lb (0-2 hr) 1,200,000 lb (2-8 hr)

TABLE 3.9-2
DOSE CONVERSION FACTORS, BREATHING RATES AND ATMOSPHERIC DISPERSION FACTORS

Isotope	Thyroid Dose Conversion Factors[1] (rem/curie)
I-131	1.48×10^{6}
I-132	5.35×10^{4}
I-133	4.0×10^{5}
I-134	2.5×10^{4}
I-135	1.24×10^{5}

Time Period (hr)	Breathing Rate[2] (m^3/sec)
0-8	3.47×10^{-4}

Atmospheric Dispersion Factors, sec/m^3

Site Boundary (0-2 hr)	3.15×10^{-4}
Low Population Zone (0-8 hr)	7.5×10^{-4}

[1] TID-14844

[2] Regulatory Guide 1.4

TABLE 3.9-3
LOOP OFF-SITE DOSES

	Dose (Rem)	
	SB (0-2 Hr)	LPZ (0-8 Hr)
Thyroid: Accident Initiated Spike	$5.4x10^{-2}$	$8.4x10^{-2}$
Thyroid: Pre-Accident Spike	$6.6x10^{-2}$	$8.5x10^{-2}$
Whole Body	$4.2x10^{-4}$	$3.0x10^{-4}$

TABLE 3.9-4
ASSUMPTIONS FOR SGTR DOSE ANALYSIS

Core Power Level	3588 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	60 μCi/gm of DE I-131
Accident Initiated Spike	1.0 μCi/gm of DE I-131
Iodine Release from Fuel to Reactor Coolant Due to Accident Initiated Spike	500 times equilibrium release rate from fuel for initial 1.8 hours after SGTR
Secondary Coolant Activity Prior to Accident	0.10 μCi/gm of DE I-131
SG Tube Leak Rate During Accident	
Ruptured SG	0.25 gpm
Intact SGs	0.75 gpm
Break Flow to Ruptured SG	140,264 lb (0-30 min)
SG Iodine Partition Factor	0.01
Duration of Activity Release from Secondary System	8 hr
Off-site Power	Lost
Steam Release from SGs to Environment	
Ruptured SG	56,525 lb (0-0.5 hr)
Intact SGs	413,000 lb (0-2 hr) 978,000 lb (2-8 hr)

TABLE 3.9-5
DOSE CONVERSION FACTORS, BREATHING RATES AND ATMOSPHERIC DISPERSION FACTORS

Isotope	Thyroid Dose Conversion Factors[1] (rem/curie)
I-131	1.48×10^{6}
I-132	5.35×10^{4}
I-133	4.0×10^{5}
I-134	2.5×10^{4}
1-35	1.24×10^{5}

Time Period (hr)	Breathing Rate[2] (m^3/sec)
0-8	3.47×10^{-4}

Atmospheric Dispersion Factors, sec/m^3

Site Boundary (0-2 hr)	3.15×10^{-4}
Low Population Zone (0-8 hr)	7.5×10^{-5}

[1] TID-14844

[2] Regulatory Guide 1.4

TABLE 3.9-6
SGTR OFF-SITE DOSES

	Dose (Rem)	
	SB (0-2 Hr)	LPZ (0-8 Hr)
Thyroid: Accident Initiated Spike	0.27	0.12
Thyroid: Pre-Accident Spike	1.64	0.44
Whole Body	0.081	0.019

TABLE 3.9-7
ASSUMPTIONS USED FOR SLB DOSE ANALYSIS

Core Power Level	3588 MWt
Reactor Coolant Noble Gas Activity Prior to Accident	1.0% Fuel Defect Level
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	60 µCi/gm of DE I-131
Accident Initiated Spike	1.0 µCi/gm of DE I-131
Iodine Release from Fuel to Reactor Coolant Due to Accident Initiated Spike	500 times equilibrium release rate from fuel for initial 2.0 hours after SLB
Secondary Coolant Activity Prior to Accident	0.10 µCi/gm of DE I-131
SG Tube Leak Rate During Accident	1.0 gpm
Iodine Partition Factor for SG Tube Leakage	
Faulted SG	1.0 (SG assumed to steam dry)
Intact SGs	1.0 (leak assumed to be above mixture level)
Iodine Partition Factor for Activity Release Due to Steaming	
Faulted SG	1.0 (SG assumed to steam dry)
Intact SGs	0.1
Duration of Activity Release from Secondary System	8 hr
Offsite Power	Lost
Steam Release from SGs to Environment	
Faulted SG	99,300 lb (0-2 hr)
Intact SGs	463,000 lb (0-2 hr) 978,000 lb (2-8 hr)

TABLE 3.9-8
DOSE CONVERSION FACTORS, BREATHING RATES AND ATMOSPHERIC DISPERSION FACTORS

Isotope	Thyroid Dose Conversion Factors[1] (rem/curie)
I-131	1.48×10^{6}
I-132	5.35×10^{4}
I-133	4.0×10^{5}
I-134	2.5×10^{4}
I-135	1.24×10^{5}

Time Period (hr)	Breathing Rate[2] (m^3/sec)
0-8	3.47×10^{-4}

Atmospheric Dispersion Factors, sec/m^3

Site Boundary (0-2 hr)	3.15×10^{-4}
Low Population Zone (0-8 hr)	7.5×10^{-5}

[1] TID-14844

[2] Regulatory Guide 1.4

TABLE 3.9-9
SLB OFFSITE DOSES

	Dose (Rem)	
	SB (0-2 Hr)	LPZ (0-8 Hr)
Thyroid: Accident Initiated Spike	3.3	3.8
Thyroid: Pre-Accident Spike	4.1	3.1
Whole Body	4.1×10^{-4}	3.0×10^{-4}

TABLE 3.9-10
ASSUMPTIONS USED FOR FHA DOSE ANALYSIS

Power	3588 MWt
Radial Peaking Factor	1.65
Damaged Fuel	1 Fuel Assembly
Fuel Rod Gap Fractions[1] [2]	0.10 for iodines and noble gases, except 0.12 for I-131 and 0.30 for Kr-85
Percent of Gap Activity Released	100%
Pool Decontamination Factors	
Elemental Iodine	133[1]
Methyl Iodine	1[1]
Noble Gas	1[1]
Iodine Species in Fuel Rod Gap	
Elemental Iodine	99.75%[1]
Methyl Iodine	0.25%[1]
Minimum Water Depth Above Reactor Pressure Vessel Flange	23 feet
Radioactive Decay Time	100 hours
Containment Purge Filter Efficiency	No Filtration Assumed
Containment Isolation	No Containment Isolation
Auxiliary Building Filter Efficiency	90% elemental[1] 70% methyl[1]

(1) Regulatory Guide 1.25
(2) NUREG/CR-5009

TABLE 3.9-11
DOSE CONVERSION FACTORS, BREATHING RATES
AND ATMOSPHERIC DISPERSION
FACTORS

Isotope	Dose Conversion Factor[1] (rem/curie)
I-131	1.48×10^{6}
I-132	5.35×10^{4}
I-133	4.00×10^{5}
I-134	2.50×10^{4}
I-135	1.24×10^{5}

Time Period (hr)	Breathing Rate[2] (m^3/sec)
0-8	3.47×10^{-4}

Atmospheric Dispersion Factors, sec/m^3

Exclusion Boundary (0-2 hr)	3.15×10^{-4}
Low Population Zone	7.5×10^{-5}

[1] TID-14844

[2] Regulatory Guide 1.4

TABLE 3.9-12
FUEL HANDLING ACCIDENT OFFSITE DOSES

FHA INSIDE CONTAINMENT

		Dose (Rem)	
		SB (0-2 Hr)	LPZ (0-2 Hr)
I.	Thyroid	119	28
II.	Whole Body	0.25	0.06

FHA INSIDE AUXILIARY BUILDING

		Dose (Rem)	
		SB (0-2 Hr)	LPZ (0-2 Hr)
I.	Thyroid	18	4.2
II.	Whole Body	0.25	0.06

3.10 FLUID AND AUXILIARY SYSTEMS EVALUATIONS

3.10.1 Fluid Systems Evaluation

3.10.1.1 Introduction

This section addresses the impact of the Uprating Program on the ability of the Reactor Coolant System and auxiliary fluid systems to perform their required functions. The parameters considered are listed in Table 2.1-1.

In order to support the operation of Cook Nuclear Plant Unit 2 at the Uprating Program conditions, the following systems were evaluated at the new conditions: 1) Reactor Coolant System (RCS), 2) Chemical and Volume Control System (CVCS), 3) Emergency Core Cooling System (ECCS) and 4) Residual Heat Removal System (RHR). A brief description of each system is provided below.

The Emergency Core Cooling System flowrates were revised as part of the Uprating Program. These ECCS flowrates reflected a charging pump head degradation of 10% (differential pressure of 2290 psid on recirculation), a SI and RHR pump head degradation of 15% (differential pressures of 1326 psid and 150 psid, respectively, on recirculation). These ECCS flowrates were used in the safety analyses and evaluations for the Uprating Program.

3.10.1.2 Description of Fluid Systems

Reactor Coolant System

The RCS consists of four identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump and a steam generator. In addition, the system includes a pressurizer, a pressurizer relief tank, interconnecting piping, and instrumentation necessary for operational control.

During operation, the RCPs circulate pressurized water through the reactor vessel and the four coolant loops. The water, which serves both as a coolant, moderator, and solvent for boric acid (chemical shim control), is heated as it passes through the core. It then flows to the steam generators, where the heat is transferred to the steam system, and returns to the RCPs to repeat the cycle.

RCS pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water sprays. Steam can be formed (by the heaters) to increase RCS pressure or condensed (by the pressurizer spray) to reduce the pressure. Three spring loaded safety valves and three power operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

Chemical and Volume Control System

The CVCS provides for boric acid addition, chemical additions for corrosion control, reactor coolant clean-up and degasification, reactor coolant make-up, reprocessing of water letdown from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the shell side of the regenerative heat exchanger and then through a letdown orifice. The regenerative heat exchanger reduces the temperature of the reactor coolant and the letdown orifice reduces the pressure. The cooled, low pressure water leaves the reactor containment and enters the auxiliary building. A second temperature reduction occurs in the tube side of the letdown heat exchanger followed by a second pressure reduction due to the low pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the Volume Control Tank (VCT).

Emergency Core Cooling System

The ECCS injects borated water into the reactor following a break in either the reactor or steam systems in order to cool the core and prevent an uncontrolled return to criticality. Two SI pumps and two residual heat removal pumps take suction from the RWST and deliver borated water to four cold leg connections via the accumulator discharge lines. In addition, two centrifugal charging pumps take suction from the RWST on SI actuation and provide flow to the RCS via separate SI connections on each cold leg. This arrangement of SI pumps can provide safety injection flow at any RCS pressure up to the set pressure of the pressurizer safety valves.

Residual Heat Removal System

The RHRS is designed to remove sensible and decay heat from the core and reduces the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the RHRS is used to transfer refueling water between the RWST and the refueling cavity at the beginning and end of refueling operations.

The RHRS consists of two residual heat exchangers, two RHR pumps and associated piping, valves and instrumentation. During system operation, coolant flows from one hot leg of the RCS to the RHR pumps, through the tube side of the residual heat exchangers and back to two RCS cold legs. The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell.

3.10.1.3 Fluid Systems Evaluation

The impact of the Uprating Program on the ability of the following fluid systems to perform their required functions has been evaluated for the RCS, CVCS, ECCS, and RHRS.

Reactor Coolant System

The capability of the RCS to operate was evaluated at the Uprating Program conditions listed in Table 2.1-1.

The capacities of the pressurizer spray and power operated relief valves, pressurizer surge line, relief line, RTD bypass delay times and pressurizer relief tank setpoints were evaluated. It was concluded that the Uprating Program conditions were acceptable.

The pressurizer surge line pressure drop was evaluated during a design basis surge. The design basis surge results from three safety valves relieving at the design capacity. It was determined that the RCS maximum pressure at the discharge of the RCP is 2745 psia which is below the ASME maximum allowable pressure for the RCS.

The pressurizer relief line pressure drop calculation was unaffected by the revised NSSS parameters for the SGTP Program.

The results of the evaluations showed that the installed PORV capacity of 630,000 lbm/hr is adequate for the design basis load swings for operation at all Uprating Program operating conditions.

The RTD Bypass delay time calculations indicate that the fluid transport delay times for the existing piping network remain below 1.0 second in all loops and are, therefore, acceptable.

The pressurizer relief tank (PRT) setpoints were found to be acceptable. The PRT pressure will be maintained below the rupture disc set pressure following a design basis discharge with the current level setpoints.

Chemical and Volume Control System

The regenerative and letdown heat exchangers are designed to cool letdown flow from T_{cold} to 115°F. This reduction in temperature is required to ensure that the normal RCP seal injection temperature requirement of 130°F will be maintained, including an allowance for a 15°F temperature rise across the centrifugal charging pump. The variations in T_{cold} considered for the Uprating Program are bounded by the design-inlet temperature of 547°F for the regenerative heat exchanger. Therefore, the cooling requirements of the letdown function are met with the revised operating parameters.

The letdown function is designed to reduce the static pressure of the reactor letdown stream from the RCP suction pressure to VCT operating pressure, such that the design pressure of intervening piping and components is not exceeded and fluid is maintained in a subcooled condition throughout the system. The majority of the pressure reduction is taken across the letdown orifices. The pressure control valve, QRV-301, ensures that adequate back pressure is maintained on the letdown orifices to ensure subcooled fluid conditions. The pressurizer

pressures considered (2100 or 2250 psia) are bounded by the design pressurizer operating pressure. In addition, it has been verified that QRV-301 is capable of maintaining sufficient backpressure on the letdown orifices to ensure subcooled fluid conditions when the pressurizer pressure is reduced to 2100 psia. Therefore, the pressure reduction requirements of the letdown function are met with the revised operating parameters.

In addition, the Boric Acid Storage Tank (BAST) Volumes in the Technical Specifications have been verified to be adequate for the Uprating Program.

<u>Emergency Core Cooling System</u>

The primary system pressures considered for this program are less than or equal to the primary system pressure against which the original system was designed to deliver. The required core cooling flow rate is proportional to reactor power level which has not changed since the VANTAGE 5 RTSR. The revised primary system parameters do not require an increase in either the motive pressure or core cooling capacity of the ECCS. ECCS flows were recalculated for the 15% pump head degradation and used in the accident analyses.

<u>Residual Heat Removal System</u>

The RHRS is normally placed in operation approximately four hours after reactor shutdown when the pressure and temperature of the RCS are approximately 400 psig and 350°F, respectively. Under normal operating conditions, the RHRS is designed to reduce the temperature of the reactor coolant to 140°F within 20 hours following reactor shutdown, with both trains operating. In the event of a train failure, the RHRS is designed to reduce the reactor coolant temperature to 200°F within 36 hours after reactor shutdown. Since the initiation temperature and decay heat generation rates (power level) have not changed from those previously evaluated for the Rerating Program, the demands on the RHRS are not affected. Therefore, the RHRS is still capable of reducing the reactor coolant temperature to 140°F within the 20 hour limit for normal operating conditions, when both trains are operating. In the event of a train failure, the RHRS is still capable of reducing the reactor coolant temperature to 200°F within the 36 hour limit.

3.10.2 NSSS/Balance of Plant Interface Systems Evaluation

3.10.2.1 <u>Introduction</u>

As part of the Donald C. Cook Nuclear Plant Unit 2 Uprating Program, the following Balance-of-Plant (BOP) fluid systems were reviewed to assess compliance with Westinghouse Nuclear Steam Supply Systems (NSSS)/BOP interface requirements (Reference 1):

- Main Steam System
- Steam Dump System
- Condensate and Feedwater System

- Auxiliary Feedwater System
- Steam Generator Blowdown and Sampling System

The review was performed based on the range of NSSS operating parameters developed to support an NSSS power level of 3600 MWt as provided in Section 2.0.

The evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface requirements are delineated below.

3.10.2.2 Main Steam System

The following summarizes the Westinghouse evaluation of the major steam system components relative to the NSSS performance parameters for the Uprating Program:

Steam Generator Safety Valves

The setpoints of the steam generator safety valves are determined based on the design pressure of the steam generators (1085 psig) and the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the SGs has not changed, there is no need to revise the setpoints of the safety valves.

The steam generator safety valves must have sufficient capacity to ensure that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV code) for the worst-case loss-of-heat-sink event (Reference 2, Chapter 14). Based on this requirement, Westinghouse applies the conservative criterion that the valves should be sized to relieve 105 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the Main Steam System design pressure (Reference 1). Additionally, the capacity of any single safety valve is presently limited to 265 lb/sec at 1100 psia based on the present steam break analysis of record for a stuck-open steam generator safety valve (Reference 2, Chapter 14.2.5).

Donald C. Cook Nuclear Plant Unit 2 has 20 safety valves with a total capacity of 17,153,800 lb/hr (Reference 2, page 10.2-2), which provides about 107.2 percent of the maximum calculated steam flow of 16.00 x 10^6 lb/hr approved for the Uprating Program. Therefore, based on the range of NSSS performance parameters approved for the Uprating Program, the capacity of the installed MSSVs meets the Westinghouse sizing criterion.

The Unit 2 safety analysis of record also confirms that the setpoints (with a tolerance of ±3 percent) and capacity of the installed safety valves are adequate to preclude overpressure for the range of NSSS performance parameters approved for the Uprating Program (Reference 2, Sections 14.1.8 and 14.1.9).

Steam Generator Power Operated Relief Valves

The steam generator power operated relief valves (PORVs), which are located upstream of the main steam isolation valves (MSIVs) and adjacent to the MSSVs, are automatically controlled by steam line pressure during plant operations. The steam generator PORVs automatically modulate open and exhaust to atmosphere whenever the steam line pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. As the steam line pressure decreases to the PORV opening setpoint, the valves modulate closed. The steam generator PORV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set MSSVs. Since neither of these pressures changes for the range of NSSS performance parameters, there is no need to change the PORV setpoint.

The steam generator PORVs also provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating pumps, or steam dump to the condenser is not available. Under such circumstances, the PORVs in conjunction with the Auxiliary Feedwater System (AFWS) permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the Residual Heat Removal System (RHRS) can be placed in service. During cooldown, the PORVs are automatically controlled by steam line pressure with remote manual adjustment of the pressure setpoint from the Control Room.

In the event of a tube rupture event in conjunction with loss of offsite power, the PORVs are used to cool down the Reactor Coolant System (RCS) to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere (Reference 2, Chapter 14.2.4).

The four steam generator PORVs are sized to have a capacity equal to about 10 percent of the steam flow used for plant design, at no-load steam pressure (Reference 1). This capacity permits a plant cooldown to RHRS operating conditions in 4 hours (at an assumed cooldown rate of 50°F/hr) assuming 2 hours at hot standby. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFWS.

Based on the range of NSSS operating parameters approved for the uprated power level the installed PORV capacity (1.468×10^6 lb/hr at 1020 psia) is only about 9 percent of the required maximum steam flow (16.0×10^6 lbs/hr). An evaluation of this capacity in terms of cooldown capability at the uprated conditions indicates that the PORVs can still achieve a 4 hour plant cooldown assuming the cooldown rate is allowed to exceed the nominal administrative limit of 50°F/hr by a few degrees. Note that the design cooldown rate is assumed to be 100°F/hr for component fatigue analysis. Therefore, the PORVs are adequate based on the range of NSSS performance parameters for Uprating Program.

Main Steam Isolation Valves

The MSIVs are located outside the containment and downstream of the steam generator safety and relief valves. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the MSIVs must be capable of closure within 8 seconds of receipt of a closure signal against steam break flow conditions in either the forward or reverse direction (Reference 2, page 14.2.5-7). This valve closure requirement was assumed in the accident analysis performed to support the Uprating Program.

Rapid closure of the MSIVs following postulated steam line breaks causes a significant differential pressure across the valve seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs and check valves. The worst cases for pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flow rate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables are not impacted by the Uprating Program, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change.

3.10.2.3 Steam Dump System

The NSSS Reactor Control Systems and the associated equipment (pumps, valves, heaters, control rods, etc.) are designed to provide satisfactory operation (automatic in the range of 15 to 100 percent power) without reactor trip when subjected to the following load transients:

- Loading at 5 percent of full power per minute with automatic reactor control.
- Unloading at 5 percent of full power per minute with automatic reactor control.
- Instantaneous load transients of plus or minus 10 percent of full power (not exceeding full power) with automatic reactor control.
- Load reductions of 50 percent of full power with automatic reactor control and steam dump.

The steam dump system creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. The Westinghouse sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of plant rated electrical load without a reactor trip (Reference 1). A steam dump capacity of 40 percent of rated steam flow at full load steam pressure also prevents steam generator safety valve lifting following a reactor trip from full power.

Condenser Steam Dump Valves

The Donald C. Cook Nuclear Plant Unit 2 is provided with nine condenser steam dump valves and each valve is currently specified to have a maximum flow capacity of 890,000 lbs/hr at a valve inlet pressure of 1100 psig. This total capacity provides a steam dump capability of about 39 percent of current rated steam flow (15.07 x 10^6 lb/hr), or 5.85 x10^6 lbs/hr at a full load steam generator pressure of 820 psia versus the Westinghouse sizing criterion of 40 percent of rated steam flow.

NSSS operation within the range of NSSS performance parameters at lower steam generator pressures and higher steam flows will result in a reduced steam dump capability relative to the original Westinghouse sizing criteria. An evaluation indicates that the total steam dump capacity could be as low as 26 percent of rated steam flow (15.9 x 10^6 lb/hr), or 4.2 x 10^6 lbs/hr at a full-load steam pressure equal to 587 psia. These operating conditions are based on an NSSS power level of 3600 MWt, an assumed steam generator tube plugging level of 10 percent average and a T_{avg} in the lower end of the operating range. Although a total steam dump capacity of only 26 percent of full load steam flow is significantly less than the Westinghouse recommended capacity, it may be adequate since plant operating experience has demonstrated that the recommended capacity of 40 percent of rated flow includes about 15 percent margin. Note at the upper end of the T_{avg} operating range and a full-load steam generator pressure of 820 psia, steam dump capacity is about 37 percent of rated flow (16.0 x 10^6 lbs/hr), or 5.85 x 10^6 lbs/hr.

In order to confirm adequate steam dump capacity to support the design basis load rejection transient, a control systems operability assessment (i.e. margin to trip analysis) must be performed using the dump system capacities determined for the proposed range of uprating operating conditions.

To provide effective control of flow on large step load reductions or plant trip, the steam dump valves are required to go from full-closed to full-open in 3 seconds at any pressure between 50 psi less than full load pressure and steam generator design pressure.

The dump valves are also required to modulate to control flow. The positioning response required for this purpose is permitted to be slower than the 3 second requirement discussed above. The maximum full stroke time for modulation of 20 seconds is still applicable for the NSSS performance parameters for the Uprating Program (Reference 1).

3.10.2.4 Condensate and Feedwater System

The Condensate and Feedwater System (C&FS) must automatically maintain steam generator water levels during steady-state and transient operations. The range of NSSS performance parameters will result in a required feedwater volumetric flow increase of up to 6.8 percent during full-power operation. The higher feedwater flow and lower feedwater temperatures will have an impact on system pressure drop, which may increase by as much as 13.4 percent.

Also, a comparison of the range of NSSS performance parameters with the reference operating parameters indicates that the S/G full-power operating steam pressure may be decreased by as much as 233 psi (820 psia - 587 psia).

Feedwater Isolation Valves

The feedwater isolation valves (FIVs) are located outside containment and downstream of the feedwater control valves (FCVs). The FIVs function in conjunction with the FCVs and backup trip signals to the feedwater pumps to provide redundant isolation of feedwater flow to the steam generators following a steam line break or a malfunction in the steam generator level control system. Isolation of feedwater flow is required to prevent containment overpressurization and excessive reactor coolant system cooldowns. To accomplish this function, the FIVs and the backup FCVs must be capable of closure in less than or equal to 41 seconds (Reference 4, page 3.5-22) and 8 seconds (Reference 2, page 14.2.5-14), respectively, after receipt of a closure signal under all operating and accident conditions, including a maximum flow condition with all main feedwater pumps delivering to one steam generator. These valve closure requirements were assumed in the accident analysis performed to support the uprating.

The quick-closure requirements imposed on the FIVs and the backup FCVs, causes dynamic pressure changes that may be of large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam break from no load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not impacted by the Uprating Program, the design loads and associated stresses resulting from rapid closure of these valves will not change.

Feedwater Control Valves, Condensate and Feedwater System (C&FS) Pumps

The C&FS available head in conjunction with the feedwater control valve characteristic must provide sufficient margin for feed control to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous steady feed flow should be maintained at all loads. To assure stable feedwater control, with variable speed feedwater pumps, the pressure drop across the FCVs valves at rated flow (100 percent power) should be approximately equal to the dynamic losses from the feed pump discharge through the steam generator (i.e. equal to the frictional resistance of feed piping, FIV, high pressure feed water heaters, feed flow meter, and steam generator). In addition, adequate margin should be available in the FCVs at full load conditions to permit a C&FS delivery of 96 percent of rated flow with a 100 psi pressure increase above the full load pressure with the FCVs fully open (Reference 1).

The hydraulics of the C&FS in conjunction with the allowable range of feedwater pump speed control should permit operation over the entire range of NSSS performance parameters. However, to optimize feedwater control and minimize the duty on the feedwater control valves

the feedwater pump speed control program must be set to a specific set operating conditions and subsequently revised if NSSS performance parameters are varied within the approved range.

To provide effective control of flow during normal operation, the feedwater control valves (FCVs) are required to stroke open or closed in 20 seconds over the anticipated inlet pressure control range (approximately 0-1600 psig). Additionally, rapid closure of the FCVs is required in 8 seconds after receipt of a trip close signal in order to mitigate certain transients and accidents. These requirements are still applicable at the uprated conditions (References 1 and 2).

3.10.2.5 Auxiliary Feedwater System

The AFWS serves as a backup system for supplying feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the heat sink of the steam generators. The system provides an alternate to the Main Feedwater System during startup, hot standby, and cooldown and also functions as an Engineered Safeguards System. In the latter function, the AFWS is directly relied upon to prevent core damage and system overpressurization in the event of transients and accidents such as a loss of normal feedwater or a secondary system pipe break. The minimum flow requirements of the AFWS are dictated by accident analysis and since the uprating impacts these analyses, evaluations of the limiting transients and accidents were performed and confirmed that the current AFWS performance is acceptable at the uprated conditions.

Auxiliary Feedwater Storage Requirements

The AFWS pumps are normally aligned to take suction from the condensate storage tank (CST). Sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The Donald C. Cook Nuclear Plant Unit 2 Technical Specifications require that the CST contain a minimum usable volume of 175,000 gallons to ensure that sufficient water is available to maintain the RCS at hot standby conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. Based on Reference 3, the inventory required to bring the plant from full load to hot shutdown conditions and then hold the plant at hot shutdown conditions for 9 hours is about 161,000 gallons.

Based on the range of NSSS performance parameters for the Cook Unit 2 Uprating Program, the minimum usable inventory required in the Cook Unit 2 CST is 174,500 gallons to ensure hot standby capability following plant trip from full load for a period of 9 hours in accordance with the plant Technical Specifications. The usable water volume limit reflects the volume of water above the centerline of the discharge pipe. An allowance for water not usable because of tank discharge line location or other physical characteristics is not required. The plant

plant Technical Specifications bases have been revised for clarification purposes. Technical Specification markups are contained in Appendix A.

The recommended minimum usable inventory to bound the power uprating is based on the following assumptions:

- Reactor trip occurs from 102 percent of rated power (3588 MWt), from a low low water level in the steam generators. A two second delay is assumed before reactor trip following loss of offsite power.

- Steam is released from the steam generators at first safety valve set point plus 3 percent accumulation pressure plus 3 percent setting tolerance.

- The steam generators are filled back up to no load programmed level.

- The CST operating fluid temperature is 120°F.

3.10.2.6 Steam Generator Blowdown and Sampling System

The Steam Generator Blowdown and Sampling System is used in conjunction with the Chemical Addition System to control the chemical composition of the steam generator shell water within the specified limits. The blowdown system also controls the buildup of solids in the steam generator water.

The Unit 2 Replacement Steam Generators (RSG) are designed to handle a design maximum blowdown flow rate equal to 1 percent of total feedwater flow rate continuously and 3 percent of total feedwater flow rate for short periods during normal operation.

The proposed uprate feedwater flow is increased from rated flow for the RSGs of 14.78×10^6 lb/hr to a maximum of 16.0×10^6, an increase of approximately 8 percent. If the normal blowdown flow rate (i.e., the continuous flow rate) is allowed to increase to a maximum of one percent of the uprated main feedwater flow rate, the existing flow bases for the RSGs will be exceeded by approximately 8 percent. Theoretically, the potential for erosion/corrosion (E/C) will increase with any increase in the normal blowdown flow rated above the existing design bases. However, the overall effect of the small increases in the steam generator blowdown velocities associated with an 8 percent increase in blowdown flow rate is not expected to alter the E/C rates appreciably. However, it should be noted that the actual required normal blowdown flows are based on chemistry control requirements and these requirements are not expected to change at the uprated power level.

Higher blowdown flow rates are specified to address the short durations when additional flow is required for chemistry excursions, outage recoveries, or when it is desirable to achieve high steam generator tube sheet velocities in order to "sweep" the tube sheet area of crud

deposits. The minimum velocities to achieve the "sweeping" action are independent of the feedwater flow rate and therefore should not be affected by the uprate.

Since the range of NSSS performance parameters permits a large variation in full load steam pressure (820 to 587 psia), the inlet pressure to the steam generator blowdown and sampling system can also vary accordingly. To maintain a fixed blowdown capability, the control range of the steam generator blowdown and sampling system control valves must have adequate margin to accommodate the variation in system inlet pressure.

3.10.2.7 Conclusions

The following is a brief summary of the NSSS/BOP interface evaluation conclusions for the Donald C. Cook Nuclear Plant Unit 2 Uprating Program. Refer to the identified sections for a more detailed discussion.

Main Steam System

- Based on the range of NSSS operating parameters approved for the uprating, the capacity of the installed steam generator safety valves meets the Westinghouse sizing criterion.
- The steam generator PORVs are adequately sized for the Uprating.
- The MSIVs are not impacted by the Uprating.

Steam Dump System

- For the range of Uprating Program NSSS performance parameters, steam dump capacity is less than the Westinghouse recommended capacity. Since this is not a safety issue, the alternatives are to either not operate at a steam generator pressure below 820 psia or accept a load rejection capability without reactor trip of somewhat less than 50% load.

Condensate and Feedwater System

- The hydraulics of the C&FS in conjunction with the allowable range of feedwater pump speed control should permit operation over the entire range of full power NSSS. performance parameters for the Uprating Program.
- To optimize feedwater control and minimize the duty on the feedwater control valves the feedwater pump speed control program must be set to a specific set of operating parameters and subsequently revised if NSSS performance parameters are varied within the approved range.

Auxiliary Feedwater System

- The minimum flow requirements of the AFWS are dictated by accident analysis and since the Uprating Program impacts these analyses, evaluations of the limiting transients and accidents were perfomed to confirm that the current AFWS performance is acceptable for the Uprating Program.

- At the uprated operating conditions the minimum usable inventory required in the Unit 2 CST is 174,500 gals. This minimum inventory permits 9 hrs at hot standby following plant trip from full load in accordance with the plant technical specifications. The bases for the inventory specified in the plant technical specifications has been revised for clarification purposes. (Refer to Appendix A)

Steam Generator Blowdown and Sampling System

- The specified RSG blowdown capability is acceptable for the Uprating Program.

- To maintain a fixed blowdown capability, the control range of the Steam Generator Blowdown and Sampling System control valves must have adequate margin to accommodate the variation in system inlet pressure.

References

1. Westinghouse Steam Systems Design Manual, WCAP-7451, Revision 2, August 1973.

2. Donald C. Cook Nuclear Plant UFSAR.

3. Westinghouse reference operating instruction, A-6, "Station Blackout Operation" for D.C. Cook Units 1 and 2, dated 11-12-71.

4. WCAP-14285, Rev. 1, "Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report," May 1995.

3.11 PRIMARY COMPONENT EVALUATIONS

Evaluations were performed for all NSSS primary and auxiliary components to support the Uprating Program for Cook Nuclear Plant Unit 2. In some cases, structural reanalysis was performed. In general, the evaluations and analyses were performed assuming the associated NSSS performance parameters case(s) (from Table 2.1-1) most limiting for the particular component.

The NSSS components reviewed for the Uprating Program are as follows:

Section	Component
3.11.1	Steam Generators
3.11.2	Reactor Vessel
3.11.3	Reactor Internals
3.11.4	Control Rod Drive Mechanisms
3.11.5	Reactor Coolant Pumps
3.11.6	Pressurizer
3.11.7	Reactor Coolant Loop Piping and Supports
3.11.8	Auxiliary Components
3.11.9	Ice Condenser

A summary of the evaluations and analyses is provided below.

3.11.1 Steam Generators

The following sections describe the analyses and evaluations performed under the Cook Nuclear Plant Unit 2 Uprating Program for the Unit 2 Steam Generators. The Steam Generators evaluated are the replacement Model 51F series. Two separate areas of evaluation are addressed for the Uprating Program:

- Thermal-hydraulic performance characteristics (including moisture separator performance)
- Structural integrity

Note that a complete U-bend fatigue evaluation was not necessary for Cook Nuclear Plant Unit 2 because of the advanced design features incorporated into the replacement steam generators. It should be noted that the Design Specification for the Replacement Steam Generators (RSG) considered an uprated power level of 3600 MWt.

flow and tube plugging level are important insofar as they affect the steam pressure. Primary pressure, in the range under consideration, does not affect thermal hydraulic performance.

As part of the Rerating Program, thermal hydraulic performance parameters were evaluated for a range of thermal powers and steam pressures. The conclusion of the Rerating Program was that the performance characteristics of the steam generators, including moisture carryover, continue to be acceptable at all the Rerating Program conditions. Since the envelope of uprating conditions is bounded by the Rerating Program, the conclusion continues to apply for the Cook Nuclear Plant Unit 2 Uprating Program.

Moisture Separator Limits

The Unit 2 steam generator moisture separators were modified at the time of the steam generator replacement since the upper steam dome assemblies were retained. These modifications include the following elements:

- primary separator "top hats" which diffuse the jet issuing from the primary separators,
- "steam chimneys" which vent steam from below the mid deck plate without entraining liquid drops; and
- additional upper tier dryer drains.

Moisture carryover values, measured for other Model 51 units in the field which have the same modifications, have been near or below 0.1% over a wide range of power levels and steam pressures. These data are displayed in Figure 3.11-2. The data are plotted against steam pressure alone, since they show no consistent trend with power (steam flow). In earlier separator systems, including the unmodified Model 51, moisture carryover strongly increased with power (steam flow). A trend of increasing moisture with decreasing steam pressure is present, but its effect is small and the moisture level remains low to the lowest limit of the data, 700 psia.

Other Thermal Hydraulic Characteristics

In addition to moisture carryover, the Rerating Program evaluated circulation ratio (defined as bundle flow divided by feed flow), hydrodynamic stability, and steam generator total mass (steam + liquid) as additional indicators of acceptable performance. The change in these parameters from the design value to the values at each of the Rerating Program conditions were calculated. At the uprated power of 3600 MWt and at the steam pressures corresponding to the various primary temperature and plugging levels, these parameters remained within acceptable bounds.

Structural Integrity Evaluation

The stress report for the Model 51F Replacement Steam Generators considered an uprated power level of 3600 MWt. Section 2.2 indicates that the NSSS design transients for the Rerating Program remain applicable to the operation of Cook Nuclear Plant Unit 2 at the uprated conditions with the exception of the RCS pressure variations for Loss of Load and Loss of Offsite Power transients. Therefore, as part of the Uprating Program, the steam generator structural integrity was evaluated to account for the revised Loss of Load and Loss of Offsite Power design transients. Since there was no change in the thermal response, only the primary side pressure response was affected. The design pressure and the maximum primary-plus-secondary stress intensity range remain unchanged.

The evaluation considered the impact of the revised transients on the critical steam generator components. The component most affected by the Uprating Program is the tubesheet-to-channel head junction. The stress intensities continue to satisfy stress limits. However, the revised design transients affect the fatigue usage calculations. The inside point at the tubesheet-to-channel head junction is the most critical point for fatigue. The calculated value of the fatigue usage remains within the maximum allowable limit of the ASME Code.

The results of the analysis demonstrate that there will be a minimal effect on the steam generator components due to the changes in the design transients. The fatigue usage at the most critical point remains within the allowable limit. All stress and fatigue limits of the ASME Code are satisfied.

It was therefore concluded that the Cook Nuclear Plant Unit 2 steam generator structural integrity would be maintained for operations with an uprated power level.

3.11.2 Reactor Vessel

Section 2.1 identifies the Uprating Program NSSS performance parameters and confirms that the Uprating Program NSSS performance parameters are bounded by the Rerating Program NSSS performance parameters. The Uprating Program performance parameters (Table 2.1-1) identify a maximum vessel outlet temperature (T_{hot}) of 615.2°F and a minimum vessel inlet temperature (T_{cold}) of 511.7°F. These temperatures are also the bounds of the temperatures considered in the Rerating Program evaluations. Section 2.2 indicates that the NSSS design transients for the Rerating Program remain applicable to the operation of Cook Nuclear Plant Unit 2 at the uprated conditions with the exception of the RCS pressure variations for Loss of Load and Loss of Offsite Power transients. However, the changes to the pressure variations have a minimal effect on the reactor vessel maximum ranges of primary plus secondary stress intensity and peak stress intensities which were evaluated for the Rerating Program. The maximum change amounts to only 1.59 ksi increase in a maximum peak stress intensity for the bottom head instrumentation tubes. The small change in the alternating stress that results

has no effect on the reported cumulative fatigue usage factor. The maximum ranges of stress intensity and the maximum usage factors reported for the Rerating Program in Reference 1 remain applicable for the Cook Nuclear Plant Unit 2 Uprating Program.

3.11.2.1 Reactor Vessel Integrity Evaluation

Reactor vessel integrity is impacted by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the Donald C. Cook Unit 2 Uprating Program have been evaluated to determine the impact on reactor vessel integrity.

The most critical area in terms of reactor vessel integrity is the beltline region of the reactor vessel. The beltline region is defined in ASTM E185-82 (Reference 2) as "the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material". Figure 3.11-1 identifies and indicates the location of all beltline region material of the Cook Unit 2 reactor vessel.

Material Data

Fast neutron irradiation-induced changes in the tensile, fracture, and impact properties of reactor vessel materials are largely dependent on chemical composition, particularly in the copper concentration. Before performing the assessment of reactor vessel integrity for the changes associated with the uprating of Donald C. Cook Nuclear Plant Unit 2, a review of the latest plant-specific material properties was performed. Material property values were obtained from vessel fabrication test certificate results and subsequent chemical analyses that have been performed on the surveillance materials from the Cook Unit 2 surveillance program (Reference 3). The Cu and Ni weight percent values for the weld metal were obtained from the Cook Nuclear Plant Units 1 and 2 response to NRC Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity" (Reference 4). Table 3.11-1 summarizes the average copper and nickel weight percent values. A summary of the pertinent chemical and mechanical properties of the beltline region plates and weld material used in the uprating evaluation is given in Table 3.11-2.

Uprated Fluence Projections

Fast neutron exposure calculations for the Donald C. Cook Nuclear Plant Unit 2 reactor geometry were carried out using two-dimensional adjoined discrete ordinates transport techniques consistent with the requirements of draft Regulatory Guide DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence". The adjoined transport calculations for the Donald C. Cook Nuclear Plant Unit 2 reactor model were carried out in r,θ geometry using the DORT two-dimensional discrete ordinates code (Reference 5) and the BUGLE-93 cross-section library (Reference 6). The BUGLE-93 library is a 47 neutron

group, ENDF B-VI based data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P_3 expansion of the scattering cross-sections and the angular discretization was modeled with an S_8 order of angular quadrature. Adjoined source locations were chosen at several key azimuths on the pressure vessel inner radius. These calculations were run in r,θ geometry to provide neutron source distribution importance functions for the calculation of neutron exposure in terms of, $\phi(E > 1.0$ MeV).

The importance functions generated from these individual adjoined analyses provided the basis for all absolute exposure projections for the Donald C. Cook Nuclear Plant Unit 2 pressure vessel. The adjoined importance functions, when combined with cycle-specific neutron source distributions, yielded absolute predictions of neutron exposure at the locations of interest for each of the fuel cycles to date and established the means to perform similar predictions and dosimetry evaluations for all subsequent fuel cycles.

The cycle-specific neutron source distributions utilized with the adjoined importance functions permitted the use of not only fuel cycle-specific spatial variations of fission rates within the reactor core, but also allowed for the inclusion of the effects of the differing neutron yield per fission and the variation in fission spectrum introduced by the build-in of plutonium isotopes as the burnup of individual fuel assemblies increased.

Fluence projections on the vessel (Table 3.11-3) were calculated for the uprated power level for input to the reactor vessel integrity calculations. The reactor vessel integrity evaluation for the Donald C. Cook Unit 2 Uprating Program included the following objectives:

1. Calculate the EOL USE values for all of the beltline region materials using Figure 2 of Regulatory Guide 1.99, Revision 2 (Reference 7).

2. Review and revise, as necessary, the reactor vessel surveillance capsule removal schedule for Cook Unit 2 to reflect the changes in vessel fluence due to the Uprating Program. These calculations are consistent with the recommended practices of ASTM E185-82 and meet the requirements of Appendix H of 10 CFR Part 50 (Reference 8).

3. Calculate adjusted reference temperature (ART) values, following the methods of Regulatory Guide 1.99, Revision 2, to determine the applicability of the heatup and cooldown curves presently contained in the Cook Unit 2 technical specifications (Reference 9).

4. Calculate RT_{PTS} values for all beltline materials in the Cook Unit 2 reactor vessel based upon fluence values projected for the Uprating Program at the time of uprating and EOL (32 EFPY). The current PTS Rule, 10 CFR Part 50.61 (Reference 10), will be used to ensure that the screening criteria is met. Also, determine the Emergency Response Guidelines (ERG) pressure-temperature limit category (Reference 11).

Upper Shelf Energy (USE)

10 CFR Part 50, Appendix G (Reference 12) specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary in nuclear power plants. Reactor vessel beltline materials must maintain Charpy upper shelf energy of no less than 50 ft-lb throughout the life of the vessel.

The upper shelf energy values of the beltline region plates and weld materials were determined for EOL (32 EFPY) per Figure 2 of Regulatory Guide 1.99 Revision 2. The beltline region materials will maintain an upper shelf energy of no less than 50 ft-lb through EOL.

Surveillance Capsule Withdrawal Schedule

A surveillance capsule withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel in order to effectively monitor the condition of the reactor vessel materials under actual operating conditions. ASTM E185-82 defines the recommended number of surveillance capsules and the recommended withdrawal schedule, based on the vessel material predicted transition temperature shifts (ΔRT_{NDT}).

The current surveillance capsule withdrawal schedule has been reviewed for the Cook Unit 2 reactor vessel. The removal schedule currently contained in the Technical Specifications meets the requirements and thus remains unchanged. Four capsules have been removed, while four capsules remain as STANDBY capsules.

Applicability of Heatup and Cooldown Pressure-Temperature Limits Curves

A review of the applicability date of the heatup and cooldown curves currently contained in the Cook Unit 2 Technical Specifications and the Capsule U Report (WCAP-13515, Reference 13) was performed. First, adjusted reference temperature (ART) values were calculated using the current beltline material properties and uprated fluence projections per Regulatory Guide 1.99, Revision 2. Using the most limiting material properties, fluence values which give the same ART values used in the generation of the current curves are calculated. These fluence values are then projected to values in terms of EFPY. The most limiting EFPY value is the new applicability date of the curves. After the uprating is implemented, the current Technical Specification curves will remain applicable to 14.5 EFPY. The 32 EFPY curves contained in the Capsule U report (WCAP-13515) will remain applicable to 32 EFPY after the uprating is implemented.

Pressurized Thermal Shock (PTS)

A limiting condition on reactor vessel integrity known as Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a loss-of-coolant-accident (LOCA) or a steam line break. Such transients may challenge the integrity of a reactor vessel under the following conditions:

- severe overcooling of the inside surface of the vessel wall followed by high repressurization,

- significant degradation of vessel material toughness caused by radiation embrittlement, and

- the presence of a critical-size defect in the vessel wall.

The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The revised PTS Rule was published in the Federal Register, May 15, 1991 with an effective date of June 14, 1991 (Reference 9). This amendment makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2. The uprated RT_{PTS} values for all beltline materials shall not exceed the screening criteria of the PTS rule. Specifically, the RT_{PTS} values of the base metal (plates or forgings) and longitudinal welds shall not exceed 270°F, while the circumferential weld metal RT_{PTS} values shall not exceed 300°F through the end-of-license (32 EFPY).

Calculations were performed for the Uprating Program using the latest procedures specified by the NRC in the PTS Rule. The calculated neutron fluence values for the uprated conditions for Cook Unit 2 were input to the calculations. RT_{PTS} values were generated for all beltline region materials of the Cook Unit 2 reactor vessel for the time of uprating (11.1 EFPY) and EOL (32 EFPY).

All RT_{PTS} values remain below the NRC screening criteria values using the projected fluence values through 32 EFPY for Cook Unit 2. The most limiting RT_{PTS} value at 32 EFPY is 216°F for intermediate shell plate C5556-2. Therefore, the uprating will have no significant impact on the RT_{PTS} values through EOL.

Emergency Response Guideline (ERG) Limits

Emergency Response Guideline (ERG) pressure-temperature limits were developed in order to establish guidance for operator action in the event of an emergency situation. The main concern in this area is the reactor vessel wall and its ability to maintain integrity when

subjected to a rapid cooling or depressurization event, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT} at EOL. (The EOP RT_{NDT} value given in the Technical Specifications for the pressure temperature limits for the requirements of Regulatory Guide 1.99, Revision 2, is calculated based on the assumption that a 1/4T (thickness) flaw is present. The EOP RT_{NDT} is based on a near surface flaw (i.e., RT_{PTS}) and is used to determine the appropriate EOP limit category.) These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest EOL RT_{NDT} for which the generic category ERG pressure-temperature limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Thus, if the limiting vessel material has an EOL RT_{NDT} which exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG pressure-temperature limits must be developed.

The RT_{PTS} values at the end-of-life (32 EFPY) were calculated using the uprated fluence projections for Cook Unit 2 to determine the applicable ERG category. For Cook Unit 2, the most limiting inside surface RT_{NDT} (RT_{PTS}) value at EOL was calculated to be greater than 200°F but less than 250°F (Table 3.11-5). This result is within the Category II criteria. Therefore, Cook Unit 2 will be in ERG Pressure-Temperature Limit Category II after the uprating is implemented.

Plant Operation Below 525°F

The Donald C. Cook Nuclear Plant Unit 2 reactor vessel inlet temperature will be 511°F after the uprating is implemented. Per Regulatory Guide 1.99, Revision 2, "Irradiation below 525°F should be considered to produce greater embrittlement. The correction factor used should be justified by reference to actual data." A review was performed of all available surveillance capsule data from commercial power reactors for which the design operating temperature was below 550°F. The surveillance capsule test data from Cook Nuclear Plant Unit 2 for a radiation temperature of 550°F fell within or above the scatter band of all surveillance capsule test data for plants with an operating temperature of 510°F. Therefore, it is concluded that operation of Cook Nuclear Plant Unit 2 below 525°F down to 510°F is acceptable.

Conclusions

Based on the evaluations discussed above, it is concluded that the Uprating Program for Donald C. Cook Nuclear Plant Unit 2 will not have significant impact on the reactor vessel integrity.

References

1. WCAP-11968, "T_{hot} Reduction/Rerating Reactor Vessel Evaluation Addendum to Analytical Report for Indiana and Michigan Electric Company, Donald Cook Nuclear Power Plant Unit No. 2 Station 173. Pressurized Water Reactor Vessel" Chicago Bridge and Iron Contract No. 68-3262, August, 1988, by S. L. Abbott.

2. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706 (IF)[1]"

3. WCAP-8512, "American Electric Power Company Donald C. Cook Unit No. 2 Reactor Vessel Radiation Surveillance Program", J. A. Davidson, et al., November 1975.

4. AEP:NRC:1173F, "Donald C. Cook Nuclear Plant Units 1 and 2 Response to NRC Generic Letter 92-01, Rev. 1, Supplement 1, Reactor Vessel Structural Integrity," November 20, 1995.

5. RSIC Computer Code Collection CCC-543, "TORT-DORT Two- and Three-Dimensional Discrete Ordinates Transport, Version 2.8.14", January 1994.

6. RSIC Data Library Collection DLC-175, "BUGLE-93, Production and Testing of the VITAMIN-B6 Fine Group and the BUGLE-93 Broad Group Neutron/Photon Cross-Section Libraries Derived from ENDF/B-VI Nuclear Data", April 1994.

7. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", May 1988.

8. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements", 1/1/90 Edition.

9. Donald C. Cook Nuclear Plant Unit 2 Technical Specifications, pages 3/4 4-25, 26, 27, Amendment No. 171 (Heatup/Cooldown Curves) and Amendment No. 20 (Reactor Vessel Surveillance Schedule).

10. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", May 15, 1991.

11. Emergency Response Guidelines - Revision 1B, Westinghouse Owners Group, 2/28/92.

12. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements".

13. WCAP-13515, "Analysis of Capsule U from the Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program," J. M. Chicots, et al., February 1993.

3.11.3 Reactor Internals

3.11.3.1 Introduction

This section documents the results and conclusions of the evaluations performed to investigate the impact of the Uprating Program on the Cook Nuclear Plant Unit 2 reactor vessel internals. In order to assess the impact of the Uprating Program, the following design inputs were reviewed.

- NSSS Design Transients
- Power Level
- Heat Generation Rates
- Hydraulic Forcing Functions
- Core Cavity Pressurization Loads

NSSS Design Transients

The design transients of the Rerating Program remain applicable with the exception of the reactor coolant pressure transients for the loss of load and loss of off site power events. The revised reactor coolant pressure transients for the loss of load and loss of offsite power events do not affect the reactor internals analyses of the Rerating Program. Therefore, no reactor internals evaluations are required for changes in design transients.

Heat Generation Rates

The reactor internals evaluations of the Rerating Program were performed for 3600 MWt NSSS Power. Since the current applicable NSSS Performance parameters given in Table 2.1-1 are for this same power level, no reactor internals evaluations are required for heat generation rates.

Hydraulic Forcing Functions

The LOCA hydraulic forcing functions considered in the reactor pressure vessel system LOCA evaluation of Reference 1 remain bounding for the uprating.

Core Cavity Pressurization Loads

The LOCA analysis of the reactor pressure vessel and internals system documented in Reference 1 was performed for the core cavity pressurization loads based on NSSS Performance parameters identical to the current applicable values given in Table 2.1-1. Therefore, the core cavity pressurization loads are not affected by the Uprating Program.

Reactor Vessel Loop Stiffness

The reactor vessel loop stiffness used in the seismic and LOCA analyses of the reactor pressure vessel and internals system of Reference 1 is based on reactor coolant temperatures identical to those defined in the current applicable NSSS Performance parameters of Table 2.1-1. Therefore, the loop stiffness is not affected by the Uprating Program.

Fuel Characteristics

The VANTAGE 5 with IFM fuel design used in the analysis of the reactor pressure vessel and internals system evaluations of Reference 1 remains applicable for the Uprating Program. Therefore, no reactor internals evaluations are required for changes in fuel characteristics.

PCWG Parameters

The NSSS performance parameters issued for the Uprating Program remain unchanged from those considered in Reference 1. Therefore, no reactor internals evaluations are required for changes in NSSS performance parameters due to the Uprating Program.

RCCA Scram Performance

The RCCA scram performance evaluation documented in Reference 1 was performed for VANTAGE 5 fuel with IFMs and NSSS performance parameters identical to the current applicable values given in Table 2.1-1. Therefore, no RCCA scram performance evaluation is required for the Uprating Program.

Conclusion

The reactor vessel internals stress and fatigue evaluation is considered bounded by the Rerating Program evaluation.

Reference

1. WCAP-12828 "Reactor Pressure Vessel and Internals System Evaluations for the Donald C. Cook Unit 2 Vantage 5 Fuel Upgrade with IFMs, December 1990.

3.11.4 Control Rod Drive Mechanisms

This section addresses the acceptability of the Donald C. Cook Nuclear Plant Unit 2 Control Rod Drive Mechanisms (CRDM) for the Uprating Program. Cook Unit 2 has model L-106A full-length CRDMs manufactured by the Westinghouse Electro-Mechanical Division. There are part-length mechanisms manufactured by Royal Industries, which have the control rods removed but the pressure boundary components are still in place. This section addresses the ASME Code pressure boundary aspects of the NSSS performance parameters and design transients for the Uprating Program.

The Cook Unit 2 full-length CRDM's are Model L-106A mechanisms and the hot leg loop temperature defined as "Vessel Outlet" in Table 2.1-1 applies. Of the various cases, the maximum hot leg temperature is 615.2°F. The CRDM generic reports (References 1 and 2) conservatively used 650°F as the operating temperature, so the NSSS performance parameters for the Uprating Program are bounded by the generic analyses.

Section 2.2 gives the transient changes beyond the Rerating Program. The full-length original transients were given in the plant Equipment Specification (E-spec) (Reference 3). Based on Table 3.11-7, it was determined that the new "Loss of Load" and "Loss of Power" pressures do not affect previous code analysis of the full-length or part-length CRDMs.

Therefore, because the NSSS performance parameters and the new "Loss of Load" and "Loss of Power" pressures do not affect previous Code analysis of the full-length or part-length CRDMs for Cook Unit 2, it was concluded that the Uprating Program is acceptable for the full-length and part-length CRDM pressure boundary components. Westinghouse has also reviewed the Cook Unit 2 plant changes since Fuel Cycle 8 and found no affect on the CRDM code integrity.

References

1. E.M. 4531, Revision 2, "Stress and Thermal Report of Type L106A and L106B Control Rod Drive Mechanism Pressure Containing Components", WEMD, Cheswick, PA, by S. Ganguly, J. Raymond, and A. Reed, April 12, 1976.

2. Report No. 121X135, "Stress Report - Pressure Vessel Portion of 121J001 Series Control Rod Drive Mechanism", Royal Industries, by C. Hiers, February 26, 1969; and Seismic Data Report 121X141, June 19, 1969.

3. CRDM Equipment Specification 677470, Revision 1, "Standard L-106A Control Rod Drive Mechanism for Full Length Control Rods", Westinghouse Nuclear Energy Systems, Pittsburgh, PA, by G. Murray, June 7, 1973.

3.11.5 Reactor Coolant Pumps and Motors

This section addresses the acceptability of the Model 93A Reactor Coolant Pumps (RCPs) and motors for the Uprating Program for Donald C. Cook Nuclear Plant Unit 2. This section addresses the ASME Code aspects of the RCPs for the Uprating Program.

3.11.5.1 Reactor Coolant Pumps

Table 2.1-1 defines the Uprating Program NSSS performance parameters. The RCP parameters are defined by the "Steam Generator Outlet" values. The NSSS design transient changes are identified in Section 2.2 of this report. Reference 1 is the original Equipment Specification for the Cook Unit 2 RCPs. The effect of the Uprating Program on the RCP generic analysis reports given in References 2, 3, 4, 5 and 6 was reviewed.

The NSSS performance parameters are the same as previously addressed by the Rerating Program and are acceptable. The RCP temperature data for the various cases are still less than that used in the generic report (Reference 3). This report (Reference 3) used 2500 psi and 600°F to bound the transient cases at that time. Note the normal analysis RCP operating temperature is over 550°F and that exceeds all the NSSS performance parameter cases which have 547.4°F as the maximum. The Uprating Program parameters also bound a pressure of 2100 psia. This results in a lower stress than that already accepted in the generic reports of References 2, 3, 4, 5 and 6 which show Code Criteria (Reference 7) was met.

The transient changes only affect the "Loss of Load" and "Loss of Power" pressure values. A comparison of the previous values in the Rerating Program was made to the Uprating Program in Table 3.11-7. These transients were found to be acceptable for the Model 93A RCPs at Donald C. Cook Nuclear Plant Unit 2.

The changes in NSSS design transients do not affect the ASME Code structural integrity as covered by the generic reports. Fatigue usage is not affected by the changes to the pressure transients provided in Section 2.2. Westinghouse has reviewed the Cook Unit 2 plant changes since Fuel Cycle 8 and found no effect on RCP integrity.

3.11.5.2 Reactor Coolant Pump Motors

Westinghouse has calculated the worst case loads for the RCP motors based on the NSSS performance parameters for the Uprating Program Using the revised loads, the Cook Unit 2 RCP motors (Reference 8) have been evaluated in the four areas where parameter changes effect performance.

Continuous Operation at Revised Hot Loop Rating

Per the Equipment Specification (Reference 9), the motor is required to drive the pump continuously under hot loop conditions without exceeding a stator winding temperature rise of 75°C (corresponding to the NEMA Class B temperature rise limit in a 50°C ambient). Testing on duplicate motors has shown that the actual temperature rise at rated hot loop load (6000 HP) is between 44°C and 56°C. Therefore, adequate margin exists for continuous operation with loads in excess of 6000 HP nameplate rating.

The worst case hot loop load under the revised operating conditions is 6170 HP. This represents a 2.8% increase over the nameplate rating of the motor. The stator winding temperature rise at this new load is estimated to be no greater than 57°C. This is less than the NEMA Class B temperature limit and is therefore considered acceptable.

Continuous Operation at Revised Cold Loop Rating

Per the Equipment Specification (Reference 9), the motor is required to drive the pump for up to 50 hours (continuous) under cold loop conditions without exceeding a stator winding temperature rise of 100°C (corresponding to the NEMA Class F temperature rise limit in a 50°C ambient). Testing on duplicate motors has shown that the actual temperature rise at rated cold loop load (7500 HP) is between 61°C and 78°C. Therefore, adequate margin exists for continuous operation with loads in excess of 7500 HP nameplate rating.

The worst case cold loop load under the revised operating condition is 7940 HP. This represents a 5.9% increase over the nameplate rating of the motor. The stator winding temperature rise at this new load is estimated to be no greater than 86°C. This is less than the NEMA Class F temperature limit and is therefore considered acceptable.

Starting

Per the Equipment Specification (Reference 9), the motor is required to start across the line under cold loop conditions, with 80% starting voltage, against the reverse flow from the other pumps running at full speed. The limiting component for this type of starting duty is the rotor cage winding. A conservative all heat stored analysis is used to determine if the cage winding temperature exceeds the design limits (300°C on the bars and 50°C on the resistance rings). The starting temperature rise for the rotor bars and resistance rings was calculated. The results show bar temperature of 243.7°C and ring temperature of 40.3°C. Therefore, the motors can safely accelerate the load under worst case conditions.

Loads on Thrust Bearings

Performance of the thrust bearings in an RCP motor can be adversely effected by excessive or inadequate loading. The axial down thrust for the revised parameters decreased 400 lbs. for hot and 3400 lbs. for cold conditions. This represents only a 3.4% change in bearing

loading (thrust bearings were designed for continuous operation with 101,200 lbs. thrust due to normal system pressure, per Reference 9). This change is not considered significant.

3.11.5.3 <u>Conclusions</u>

The Uprating Program was reviewed for its affect on the Cook Unit 2 Model 93A RCPs and motors. The Structural analysis and thermal analysis were both found to be acceptable. In conclusion, the ASME Code Integrity of the Model 93A RCPs is not affected by the Cook Unit 2 Uprating Program. The RCP motors at Cook Unit 2 are considered acceptable for operation under the revised conditions without modification.

3.11.5.4 <u>References</u>

1. "Controlled Leakage Pump Assembly Equipment Specification 676588, Revision 2, Interim Revision 1", Westinghouse Nuclear Energy Systems, Pittsburgh, PA, 1977.

2. Parsons, L.E., "Structural Analysis of the Upper and Lower Seal Housings and Bolts for 93A Controlled Leakage Pump", EM 4546, Rev. 1, Westinghouse Electro-Mechanical Division, Cheswick, PA, July 1974.

3. Nee, J. D., "Stress Analysis of the Casing, Main Flange, Main Flange Bolts, and Thermal Barrier of the 93A Shaft Seal Pump", EM 4487, Westinghouse Electro-Mechanical Division, Cheswick, PA, September 18, 1973.

4. Oleyar, R.J., "Analysis of the 93A Casing Feet using Umbrella Loads", EM 4503, Revision 1, Westinghouse Electro-Mechanical Division, Cheswick, PA, February 8, 1974.

5. Brondyke, D.S., "Analysis of the 93A Casing Feet Using Umbrella Loads Addendum 3 - Analysis Using the Excessive Feedwater Transient," EM4528, Addendum 3, Westinghouse Electro-Mechanical Division, Cheswick, PA, July 17, 1980.

6. Pantano, S., and Cronin, P.J., "Model 93A Reactor Pump Auxiliary Nozzles Generic Structural Analysis", EM 5113, Westinghouse Electro-Mechanical Division, Cheswick, PA, August 1, 1978.

7. "ASME Boiler and Pressure Vessel Code, Section III NB, Nuclear Power Plant Components," American Society of Mechanical Engineers, Summer 1969.

8. Motor S/N 79P280 and 79P499 design and test records.

9. Westinghouse AED Equipment Specification E-565623, Revision G.

3.11.6 Pressurizer

3.11.6.1 Introduction

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and pressure and to keep the RCS at the desired pressure. The first function is accomplished by keeping the pressurizer approximately half full of water and half full of steam at normal conditions, connecting the pressurizer to the RCS at the hot leg of one of the reactor coolant loops and allowing inflow or outflow to or from the pressurizer as required. The second function is accomplished by keeping the temperature in the pressurizer at the water saturation temperature (T_{SAT}) corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer and can be lowered by introducing relatively cool water spray into the steam space at the top of the pressurizer.

The limiting locations from a structural standpoint on the pressurizer are the surge nozzle, the spray nozzle, and the upper shell at the point of spray impingement. The limiting operating condition (relative to the SGTP conditions) of the pressurizer occurs when the RCS pressure is high and the RCS hot leg temperature (T_{HOT}) and cold leg temperature (T_{COLD}) are low. This is explained as follows: Due to inflow and outflow to and from the pressurizer during various transients the surge nozzle alternately sees water at the pressurizer temperature (T_{SAT}) and water from the RCS hot leg at T_{HOT}. If the RCS pressure is high (which means that T_{SAT} is high) and T_{HOT} is low, then the surge nozzle will see maximum thermal gradients and thus experience the maximum thermal stress. Likewise the spray nozzle and upper shell temperatures alternate between steam at T_{SAT} and spray which for many transients is at T_{COLD}. Thus, if RCS pressure is high (T_{SAT} is high) and T_{COLD} is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients and thermal stresses.

3.11.6.2 Description of Analysis and Results

The analysis performed for the Cook Nuclear Plant Unit 2 Uprating Program for the pressurizer is based on the NSSS performance parameters provided in Table 2.1-1. The design transients for the Rerating Program remain applicable to the Uprating Program with the exception of the pressure transients for the Loss of Load and Loss of Offsite Power events.

The pressurizer component analyses performed (References 1 and 2), were previously modified to account for the off-normal cooldown event documented in Reference 3. In addition, the surge nozzle analysis was previously updated for the thermal stratification pipe loads (Reference 4). The analysis update performed for the Uprating Program considered all of the previous updates.

Analytical models of the pressurizer components were subjected to pressure loads, external loads (such as piping loads) and the thermal transients. Results of the analysis include primary, secondary, and peak stresses for the various conditions. The external loads do not

change and the changes in the pressure loads do not affect the primary stress calculations performed in the original stress report. Thus, the primary stresses calculated for the original analysis remain valid.

The changes in the design transients did not have any significant effect on the primary-plus-secondary stresses. However, for some components, the fatigue analysis is affected. The new calculated fatigue usage factors for each of the pressurizer components are listed in Table 3.11-6.

The results of the analysis show that the Donald C. Cook Nuclear Plant Unit 2 pressurizer components meet the stress/fatigue analysis requirements of the ASME Code, Section III (1965 Edition, including addenda up to Winter 1966) for operation at the Uprating Program conditions.

References

1. Donald C. Cook Unit 2 Pressurizer Analysis for T_{HOT} Reduction, Technical Record Book TR-0315, Westinghouse Electric Corporation, NCD, Pensacola, Florida, July 1988.

2. Pressurizer Stress Report Sections 1 and 2, American Electric Power Service Corporation, Donald C. Cook Unit 2, Westinghouse Electric Corporation, Tampa Division, January 1995.

   84 Series Pressurizer Stress Report, Sections 3.1 through 3.14, Westinghouse Electric Corporation, Tampa Division (Generic Stress Report)

3. D. C. Cook 2 (AMP) Pressurizer Cooldown Event Analysis, Technical Record Book TR-0389, Westinghouse Electric Corporation, NCD, Pensacola, February 1990.

4. Donald C. Cook 1 and 2 Pressurizer Surge Nozzle Evaluation for Thermal Stratification Pipe Loads, Technical Record Book TR-0543, Westinghouse Electric Corporation, NCD, Pensacola, January 1991

3.11.7 Reactor Coolant Loop Piping and Supports

As part of the Rerating Program, an analysis of the reactor coolant loop piping, primary equipment nozzles, and the primary equipment supports was performed for a set of thermal parameter cases bound various temperatures in the loop piping. The analysis concluded that the loop piping, the primary equipment nozzles, and the primary equipment supports were acceptable for operation at the Rerating Program conditions. The Uprating Program NSSS performance parameters are bounded by the Rerating Program performance parameters. Further, the NSSS design transients for the Rerating Program remain applicable to the Uprating program with the exception of the RCS pressure variations for Loss of Load and Loss of Offsite Power Transients. The Rerating Program analysis was reviewed for its applicability to the Uprating Program, and it was confirmed that the reactor coolant loop

piping, the primary equipment nozzles, and the primary equipment supports are acceptable for operation at the Uprating Program conditions.

WCAP-14070 (Reference 1) was prepared to address NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to the Reactor Coolant System". WCAP-12850 (Reference 2) was prepared to demonstrate compliance with NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification". As part of the 30% SGTP Program for Cook Nuclear Plant Unit 1, an evaluation was performed to assess the impact of the SGTP Program parameters on the design basis analysis for the NRC Bulletin 88-08 evaluation of the auxiliary spray piping and the NRC Bulletin 88-11 evaluation of the pressurizer surge line piping. The evaluation demonstrated that the conclusions reported in WCAP-14070 and WCAP-12850 remained valid for the SGTP Program conditions.

The evaluation performed as part of the SGTP Program (Reference 3) to address NRC Bulletins 88-08 and 88-11 was reviewed for its applicability to the Cook Nuclear Plant Unit 2 Uprating Program. The only significant difference that could impact the analyses performed is the hot leg lower bound temperature for Unit 2. Although the hot leg lower bound temperature for the Unit 2 Uprating Program (582.3°F) is lower than that used in the SGTP Program (586.8°F), it was determined that the evaluation performed for the Unit 1 SGTP Program applies to the Uprating Program. That is, the impact on the design basis analysis for the NRC Bulletin 88-08 evaluation of the auxiliary spray piping and NRC Bulletin 88-11 evaluation of the pressurizer surge line piping is insignificant.

<u>References</u>

1. WCAP-14070, "Evaluation of Donald C. Cook Units 1 and 2 Auxiliary Spray Piping per NRC Bulletin 88-08", July 1994.

2. WCAP-12850, "Structural Evaluation of Donald C. Cook Nuclear Plant Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification", January 1991.

3. WCAP-14285, Revision 1, "Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report," May 1995.

3.11.8 Auxiliary Components

The auxiliary components (pumps, valves, tanks and heat exchangers) were reviewed to determine the impact of the NSSS parameters for the SGTP Program, provided in Table 2.1-1 of this report. Because the NSSS parameters of the Uprating Program are bounded by those of the Rerating Program and the Auxiliary Equipment Transients are either unchanged or still bounded, there is no effect on the auxiliary components of Cook Nuclear Plant Unit 2.

3.11.9 Ice Condenser

The potential impact of increased blowdown forces on the ice condenser, due to changes associated with the Rerating Program and Uprating Program, on the structural integrity of the ice condenser have been assessed.

Based upon a conservative evaluation, it was determined that the LOCA mass & energy releases could increase due to penalties associated with the reduced RCS temperatures. It was further determined that the corresponding peak forces acting on the ice condenser could increase by approximately 20%. However, it has been concluded that this increase is more than offset by the margins in the methodology used to determine the ice condenser force loadings, and the structural margin in the ice condenser design, including the 40% design margin. Therefore, even though there is the potential that the loadings on the ice condenser could increase due to changes associated with this program, there is sufficient margin in the analysis to verify that the ice condenser is structurally adequate for the rerating/uprating conditions.

TABLE 3.11-1

CALCULATION OF AVERAGE CU AND NI WEIGHT PERCENT VALUES

Intermediate Shell Plate C5521-2		Intermediate Shell Plate C5556-2		Lower Shell Plate C5540-2		Lower Shell Plate C5592-1		Weld Metal	
Cu	Ni	Cu	Ni	Cu	Ni	Cu	Ni	Cu	Ni
0.15	0.59	0.16	0.56	0.10	0.60	0.14	0.59	0.055	0.96
0.14	0.57	0.14	0.56	0.11	0.64	0.14	0.60	0.051	0.98
0.14	0.58	0.15	0.58	0.10	0.63	0.14	0.57	0.06	0.81
0.11	0.58							0.06	0.9
								0.05	0.96
								0.06	0.97
								0.05	0.92
								0.055	0.97
								0.051	0.93
								0.06	0.97
0.14	0.58	0.15	0.57	0.10	0.62	0.14	0.59	0.055	0.937

TABLE 3.11-2

CHEMICAL AND MECHANICAL PROPERTIES OF THE BELTLINE REGION MATERIALS

Beltline Region Material	Cu weight %	Ni weight %	IRT_{NDT} (°F)
Intermediate Shell Plate C5521-2	0.14	0.58	58
Intermediate Shell Plate C5556-2	0.15	0.57	38
Lower Shell Plate C5540-2	0.1	0.62	-20
Lower Shell Plate C5592-1	0.14	0.59	-20
Weld Metal	0.055	0.937	-35

TABLE 3.11-3

FLUENCE (E>1.0 MEV) VALUES ON THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

Azimuthal Angle	Extrapolation Flux (n/cm^2-sec)	11.1 EFPY (n/cm^2)	12 EFPY (n/cm^2)	32 EFPY (n/cm^2)	48 EFPY (n/cm^2)
0°	5.252×10^{9}	2.243×10^{18}	2.392×10^{18}	5.707×10^{18}	8.359×10^{18}
10°	6.698×10^{9}	2.859×10^{18}	3.049×10^{18}	7.277×10^{18}	1.066×10^{19}
15°	8.382×10^{9}	3.493×10^{18}	3.731×10^{18}	9.021×10^{18}	1.325×10^{19}
30°	1.191×10^{10}	4.228×10^{18}	4.566×10^{18}	1.208×10^{19}	1.810×10^{19}
45°	1.645×10^{10}	6.152×10^{18}	6.619×10^{18}	1.700×10^{19}	2.530×10^{19}

TABLE 3.11-4

CALCULATION OF CHEMISTRY FACTORS USING CREDIBLE SURVEILLANCE CAPSULE (S/C) DATA

Material	Capsule	Fluence (10^{19} n/cm^2, E>1.0 MeV)	FF	ΔRT_{NDT} (°F)	$FF^*\Delta RT_{NDT}$ (°F)	FF^2
Inter. Shell Plate C5521-2 (Longitudinal)	T	0.218	0.590	55	32.5	0.348
	Y	0.662	0.884	90	79.6	0.782
	X	1.03	1.01	95	95.8	1.02
	U	1.50	1.11	95	105.7	1.24
Inter. Shell Plate C5521-2 (Transverse)	T	0.218	0.590	80	47.2	0.348
	Y	0.662	0.884	100	88.4	0.782
	X	1.03	1.01	103	103.9	1.02
	U	1.50	1.11	130	144.6	1.24
				Sum:	697.6	6.77
						103.1
Weld Metal	T	0.218	0.590	40	23.6	0.348
	Y	0.662	0.884	50	44.2	0.782
	X	1.03	1.01	70	70.6	1.02
	U	1.50	1.11	75	83.4	1.24
				Sum:	221.8	3.38
						65.5

NOTES:

f = fluence ÷ 10^{19} n/cm^2
FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$
CF = $\Sigma(FF^*\Delta RT_{NDT}) / \Sigma(FF^2)$

TABLE 3.11-5

DONALD C. COOK 1 PRESSURIZER COMPONENTS
CALCULATED FATIGUE USAGES FOR THE UPRATING PROGRAM

COMPONENT	FATIGUE USAGE
Surge Nozzle	<0.34
Spray Nozzle	0.991
Safety and Relief Nozzle	<0.15
Lower Head - Heater Well	<0.07
Lower Head - Perforations	<0.02
Upper Head and Shell	0.973
Support Skirt/Flange	<0.02
Manway Pad	0.0
Manway Cover	0.0
Manway Bolts	0.0
Support Lug	<0.05
Instrument Nozzle	<0.11
Immersion Heater	<0.01
Valve Support Bracket	0.01

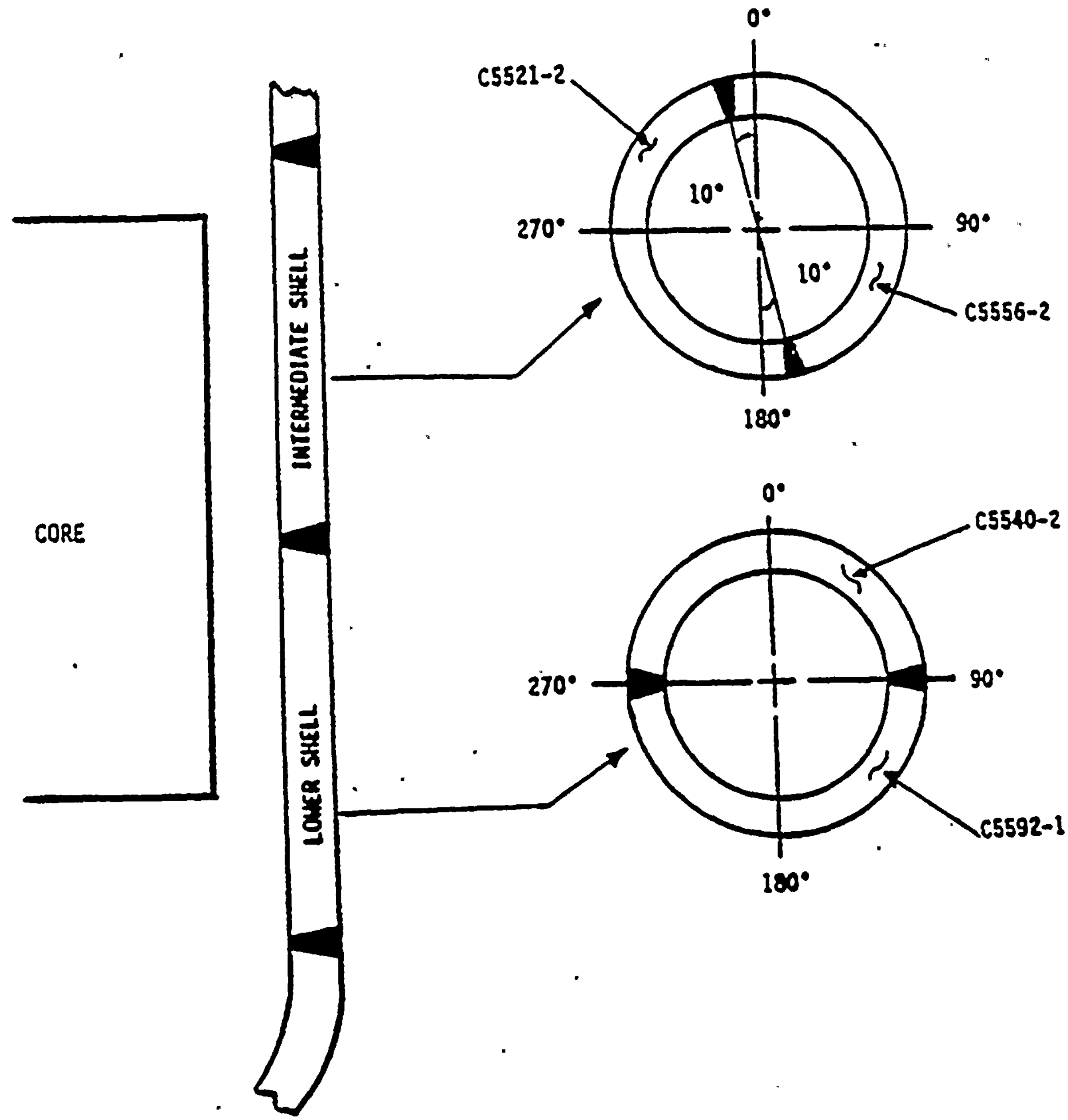


Figure 3.11-1 Identification and Location of Beltline Region Material for the Cook Unit 2 Reactor Vessel

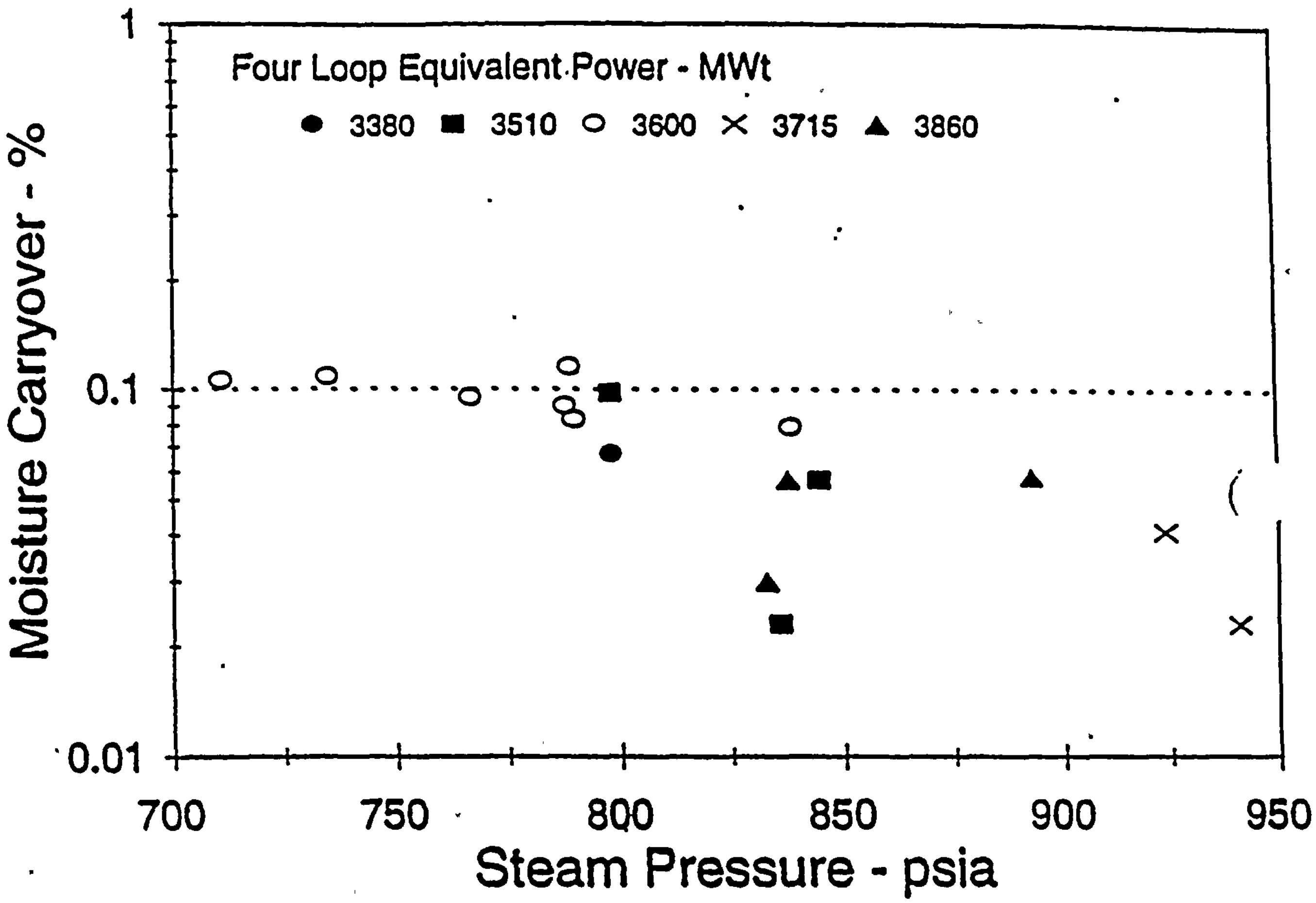


Figure 3.11-2 Moisture Separator Performance Modified Model 51 Separators

3.12 FUEL STRUCTURAL EVALUATION

Evaluations were performed of the fuel for Cook Nuclear Plant Unit 2 under the Uprating Program in the areas fuel rod and fuel assembly structural integrity, core design and thermal-hydraulic design. These evaluations assumed a maximum core power level of 3588 MWt and the associated range of operating conditions from Table 2.1-1.

3.12.1 Fuel Assembly Structural Evaluation

Fuel assemblies are designed to perform as described in the Technical Specifications. The combined effects of design basis loads are considered in the verification of the fuel assembly and its components to maintain the fuel assembly structural integrity. This is necessary so that the fuel assembly functional requirements are met, the core coolable geometry is maintained, and the reactor core can be shut down safely.

A structural evaluation of the fuel assembly was performed for the Uprating Program for Cook Nuclear Plant Unit 2, considering the range of operating parameters described in Table 2.1-1. This evaluation assumed 17 x 17 Vantage 5 fuel for Unit 2.

The NSSS performance parameters for the Uprating Program are bounded by those of the Rerating Program, and therefore, there is no impact on the fuel assembly seismic/LOCA structural evaluation due to the Uprating Program.

The Uprating Program for Cook Nuclear Plant Unit 2 does not increase the operating and postulated transient loads such that they will adversely affect the fuel assembly functional requirements. The fuel assembly structural integrity is not affected and the core coolable geometry is maintained for the 17 x 17 Vantage 5 fuel type for Cook Nuclear Plant Unit 2.

3.12.2 Fuel Rod Structural Evaluation

An evaluation was performed under the Uprating Program of the impact of NSSS performance parameters in Table 2.1-1 on the ability of fuel to satisfy fuel rod design criteria for Cook Nuclear Plant Unit 2.

The Uprating Program will have an impact on several key fuel rod design criteria. The Uprating Program parameters which impact fuel rod design are summarized in Table 3.12-1. The impacts of each of these parameters on margins to the fuel rod design criteria were evaluated.

<u>Rod Internal Pressure</u>

The rod internal pressure design basis is that the fuel system will not be damaged due to excessive fuel rod internal pressure. The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to

outward clad creep during steady state operation for extensive DNB propagation to occur. NRC-approved Westinghouse PAD3.4 fuel performance models, Reference 1, are used to evaluate rod internal pressure as a function of irradiation time and fuel duty. Margin to the rod internal pressure limit is impacted by changes in the core power rating, since the higher ratings will result in higher fuel operating temperature and higher fission gas release.

Rod internal pressure analyses, performed for the Donald C. Cook Nuclear Plant Unit 2 Uprating Program, indicate that the rod internal pressure criterion will be satisfied for the uprated conditions in Table 3.12-1.

Clad Corrosion

The clad corrosion design basis is that the fuel system will not be damaged due to excessive fuel clad oxidation. The fuel system will be operated to prevent significant degradation of mechanical properties of the clad at low temperatures, as a result of hydrogen embrittlement caused by the formation of zirconium hydride platelets. The calculated clad temperature (metal oxide interface temperature) will be less than accepted limits specified for steady state operation and for Condition II events. The hydrogen pickup level in the clad will also be restricted to specified limits predicted for the end of fuel operation. The uprating conditions in Table 3.12-1 will result in increased operating temperatures for the clad due to the increased rod average power rating. Since the corrosion process is a function of clad temperature, the uprating will impact these criteria. The uprated power may cause a decrease in the achievable burnup limit and potential modifications to core design or other options may be necessary to meet the corrosion related criteria. This will be evaluated on a cycle-specific basis.

Clad Stress

The Clad Stress design basis is that the fuel system will not be damaged due to excessive fuel clad stress. The volume average effective stress calculated with the Von Mises equation considering interference due to uniform cylindrical pellet-clad contact, caused by pellet thermal expansion, pellet swelling and uniform clad creep, and pressure differences, is less than the 0.2% offset yield stress with due consideration to temperature and irradiation effects under Condition I and II events. While the clad has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design limit. Westinghouse PAD3.4 models, Reference 1, are used to evaluate clad stress limits. The local power duty during Condition II events is a key factor in evaluating margin to clad stress limits. The fuel duty at the uprated conditions is expected to be more limiting, which will reduce margins to the clad stress limit. However, evaluations performed for the uprated core conditions indicate that sufficient margin is available to support the uprated conditions in Table 3.12-1.

Summary

The fuel rod design criteria most impacted by a change in core power rating have been reviewed with respect to available margin to support the proposed uprating. Although some design criteria are impacted, as stated above, the uprated conditions listed in Table 3.12-1 can be supported.

Finally, as in the past, cycle-specific fuel performance analyses will continue to be performed for each fuel region to confirm that this assessment, and all fuel rod design criteria, are satisfied for the operating conditions specific to each cycle of operation. These evaluations support the Reload Safety Evaluation (RSE), which is transmitted to AEPSC prior to each cycle of operation.

Reference:

1. Weiner, R.A., et. at., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations", WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), August 1988.

3.12.3 Core Design

The results of the core design evaluation indicated that the increased core power was evaluated as part of the Vantage 5 Reload Transition and Safety Report and the results and conclusions remain valid for the Uprating Program.

3.12.4 Thermal Hydraulic Design

3.12.4.1 Purpose of Analysis

The purpose of this section is to describe the thermal-hydraulic analysis necessary to support the Uprating Program conditions.

3.12.4.2 Assumptions

Table 3.12-2 summarizes the thermal-hydraulic design parameters used in this analysis. The core inlet temperature is consistent with the high temperature case from Table 2.1-1. Use of high inlet temperature bounds the range of RCS T_{avg} with regard to the Departure from Nucleate Boiling (DNB) analysis.

3.12.4.3 Discussion of Evaluation

The thermal hydraulic design criteria and methods remain the same as those presently in the Donald C. Cook Nuclear Plant Unit 2 RTSR (Reference 1).

DNB Methodology

The thermal-hydraulic analyses use the Revised Thermal Design Procedure (RTDP) (Reference 2). In the RTDP method, the following uncertainties are statistically combined with the DNBR correlation uncertainties to obtain the overall DNBR uncertainty factor used to define the design limit DNBR:

- Plant operating parameters (vessel coolant flow, core power, coolant temperature, system pressure, effective core flow fraction)

- Nuclear and thermal parameters ($F^N_{\Delta H}$)

- Fuel fabrication parameters ($F^E_{\Delta H,1}$)

- THINC IV and transient codes

The uncertainty factor obtained is used to define the design limit DNBR which satisfies the DNB design criterion. The DNB design criterion is that the probability that DNB will not occur on the most limiting fuel rod is at least 95 percent at a 95 percent confidence level during normal operation and operational transients (Condition I events) and during transient conditions arising from faults of moderate frequency (Condition II events).

To produce margin to offset penalties such as those due to rod bow and transition core, and for core design flexibility, the design limit DNBR values are increased to values designated as the Safety Analysis Limit DNBRs. The Safety Analysis Limit DNBRs are used when performing the thermal hydraulic analysis with RTDP. The DNBR limits, current penalties, and margin associated with RTDP analysis are listed in Table 3.12-3.

The Axial Offset limits were recalculated as a function of temperature at 80% and 119.6% power. These were used as input to the calculations associated with the f (ΔI) function for the OPΔT and OTΔT setpoint calculations.

Fuel Temperatures

The rod internal pressures were recalculated with the PAD3.4 code at uprated conditions. These were used as input to the LOCA analyses.

3.12.4.4 Conclusions

Thermal-hydraulic analyses were made for the fuel for the Uprating Program parameters (Table 3.12-2) using RTDP methodology. The analysis showed that the DNBR design basis was met for the limiting DNB events. This analysis caused DNBR margin to be created. This margin is given in Table 3.12-3. This margin can be used for flexibility of design and to offset unanticipated DNBR penalties.

References

1. VANTAGE 5 Reload Transition Safety Report for Donald C. Cook Nuclear Plant Unit 2, Revision 2, September 1990.

2. WCAP-8567-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1989.

TABLE 3.12-1

FUEL ROD DESIGN ANALYSIS PARAMETERS

PARAMETER	UNITS	CYCLE CONDITION	UPRATED CONDITION
Core Power	MWt	3411	up to 3588
Core Inlet Temperature	°F	542.8	up to 547.6
Mass Flow Rate	$\times 10^6$, $lbm/hr\text{-}ft^2$	2.36	>2.34
System Pressure	psia	2250	2100 to 2250
Cycle Lengths	EFPD	474	475
Fuel Designs Considered	–	Regions 10 thru 13	Regions 10 thru 13

TABLE 3.12-2

DONALD C. COOK NUCLEAR UNIT 2 UPRATING PROGRAM
THERMAL AND HYDRAULIC DESIGN PARAMETERS

Reactor Core Heat Output, MWt		3588
Reactor Core Heat Output, 10^6 Btu/hr		12,243
Heat Generator in Fuel, %		97.4
Pressurizer Pressure, Nominal, psia		2100 or 2250
Radial Power Distribution		1.65[1+0.3(1-P)]
Limit DNBR for Design Transients		
Flow Channel	Typical	Thimble
	1.69	1.61
DNB Correlation		WRB-2
HFP Nominal Conditions		
Vessel Thermal Design Flow, 10^6 lbm/hr		133.2
Core Flow Rate, 10^6 lbm/hr		126.4
Bypass Flow, %		5.1
Normal Vessel/Core Inlet Temp, °F[a]		547.6
Vessel Average Temp, °F		581.3
Core Average Temp, °F		584.9
Vessel Outlet Temp, °F		615.0
Average Temp Rise in Vessel, °F		67.4
Average Temp Rise in Core, °F		70.6
Heat Transfer[b]		
Average Heat Transfer Area, ft^2		57,700
Average Heat Flux, Btu/hr-ft^2 [b]		207,000
Average Linear Power, kw/ft[b]		5.72
Peak Linear Power for Normal Operation, kw/ft[c]		13.3
Temperature at Peak Linear Power for Prevention of Centerline Melt, °F		4700

(a) High inlet temperature bounds the proposed temperature range with respect to DNB
(b) Based on nominal 144 inch active fuel length
(c) Based on 2.32 F_Q Peaking Factor

TABLE 3.12-3

DONALD C. COOK NUCLEAR PLANT UNIT 2 UPRATING PROGRAM
RTDP DNBR LIMITS AND MARGIN SUMMARY

DNB Correlation	WRB-2	
Cell Type	Typical	Thimble
Design Limit	1.23	1.22
Safety Analysis Limit	1.69	1.61
Total DNBR Margin	27.2	24.2
DNBR Penalty - Rod Bow[b]	0	0
30 psi Issue[c]	0.9	0.9
THINC IV Penalty	6.0	4.0
Net Remaining DNBR Margin[a]	20.3	19.3

(a) Excludes temporary allocation of margin due to rotated grids
(b) A rod bow penalty of 1.3% is applied to the span without IFMs
(c) Penalty due to difference between assumed pressure drop between core and pressurizer (30 psi) and actual pressure drop

4.0 CONCLUSIONS

Provided in this document are the results and conclusions of the safety analyses and evaluations to support the implementation of the Uprating Program and the revised Technical Specification changes for Cook Nuclear Plant Unit 2. The safety analyses, evaluations, and supporting documentation provided in this submittal demonstrate acceptable results in each case, incorporating the revised operating conditions associated with the Uprating Program. A brief summary of the results of each analysis and evaluation is provided in the "Summary and Conclusions" section of this report.

ATTACHMENT 7 TO AEP:NRC:1223

BALANCE OF PLANT EVALUATIONS
AND
MISCELLANEOUS SAFETY EVALUATIONS

BALANCE OF PLANT EVALUATIONS

Evaluations were performed to determine the impact of the unit uprate on various balance of plant systems. Following is a discussion of the results of these evaluations:

Auxiliary Feedwater System

The AFW system provides water to the steam generators when the main feedwater system is not available due to a loss of main feedwater, unit trip, feedwater or steam line break, loss of offsite power, or loss of coolant accident. The source of normal water is the condensate storage tank or the lake if the CST is unavailable. The AFW system also provides water during start-up and shutdown when insufficient steam is available to drive the main feed pumps.

The AFW system consists of one turbine-driven auxiliary feed pump, which feeds all four steam generators, and two motor-driven auxiliary feed pumps, each of which feeds two steam generators.

Westinghouse's uprating evaluation has shown that the AFW system flows provided by AEP for the various accidents are acceptable with the unit operating at the uprated conditions. Therefore, no changes to the AFW system are required due to the uprating.

Chemical Feed Systems

The condensate and feedwater chemical feed systems supply the appropriate amount of chemical additive to the condensate and feedwater. The uprated chemical feedrate was reviewed and determined to be well within system capacity.

Circulating Water System

The circulating water system is an open loop system that provides a heat sink for waste heat from the plant thermal cycle. The system supplies Lake Michigan water to various coolers and condensers during all phases of plant operation.

The circulating water system essentially operates independent of the unit uprate parameters. A slight increase is expected in the thermal discharge to the lake, however, the discharge will be well within plant thermal discharge limits.

Component Cooling Water

The CCW system serves as an intermediate loop between the reactor coolant or other potentially intermediate radioactive heat sources and lake water to ensure that leakage of radioactive fluid from components being cooled is contained within the plant. The CCW system is a closed cooling water loop consisting of a surge tank, a chemical addition tank and associated piping, valves, and instrumentation, and the various equipment being cooled.

The CCW system is operated in conjunction with the essential service water system which is not affected by the rerating. Westinghouse's uprating evaluation has shown that the CCW system flows provided by AEP are acceptable with the unit operating at the uprated conditions. Therefore, no changes to the CCW system are required due to uprating.

Condensate System

The condensate system, in conjunction with the feedwater system, returns the condensed steam from the condensers and the feedwater heater drains to the steam generators while maintaining the overall water inventory throughout the cycle. The system is also required to compensate for the loss of fluid from the steam cycle when an atmospheric steam dump occurs.

Principal system components include: a main condenser, feedpump turbine condensers, hotwell pumps, condensate booster pumps, steam jet air ejectors, the turbine auxiliary cooling cycle, four stages of low pressure feedwater heating, heater drain pumps, a condensate storage tank and associated instrumentation, piping, and valves.

These components were reviewed and with the exception of the feedpump turbine condensers and the heater drain pumps, were found to be capable of operating satisfactorily at uprated conditions. The feedpump turbine condensers were determined to be undersized and will be replaced prior to the uprate. The heater drain pump performance has degraded due to operational wear. Maintenance activities will be performed to restore the pump's performance capability.

Additionally, the low pressure feedwater heaters (no.s 2 & 3) were determined to be slightly undersized. However, as the performance loss is estimated to result in only a 0.3 loss of megawatts, no corrective actions are planned.

Containment Spray System

The effect of uprating to 3588 MWt core power on the containment pressure transient and its impact on containment integrity has been addressed by Westinghouse. The iodine removal capability of the containment spray system is not affected by the small pressure changes associated with uprating. No changes to the containment spray system are required due to uprating.

Control and Plant Air

The control and plant air systems makeup the compressed air system which provides the air needs for general plant services, instrumentation and control containment integrated leak test, containment penetration and weld channel pressurization and respiratory use. The unit uprate is not expected to significantly impact compressed air needs. Ample system capacity is available to meet the expected demand, therefore, no modifications are anticipated.

Control Room Habitability

Calculations were performed confirming that the radiological dose received by plant operators in the control room was within the General Design Criterion 19 of 10CFR50 Appendix A. The calculations were performed using the source term resulting from the uprated power level of Unit 2.

The control room HVAC system was reviewed to determine the impact of uprate program. Principal system components include the air conditioning unit condenser and the liquid chiller

condenser/evaporator package. As there is no anticipated impact on the control room cooling requirements, existing equipment will not require any modification.

Electrical Systems

Principal BOP electrical components reviewed for uprate impact include; auxiliary and reserve transformers, generator step-up transformer, iso-phase bus, main generator/exciter and associated support systems. The various BOP motor/pump units were reviewed as part of the individual system review.

As no motor/pump upgrades are required, the existing auxiliary and reserve transformers are sufficient. The generator step-up transformer will require limiting reactive power operation during the winter months to meet uprated conditions. This operation has been reviewed and found to be acceptable. As a result of the uprate, the iso-phase bus temperature will increase slightly. The temperature increase will reduce the margin between the normal operating point and the alarm point. Additional temperature monitoring will be required because of the reduced margin. If warranted, based upon operating experience at the uprated condition, future bus duct cooling system upgrades will be pursued.

The turbine-generator supplier reviewed the uprate impact on the main generator/exciter and associated support systems. With the exception of the stator coolers, no other equipment was noted to be impacted. The stator water coolers were identified as requiring replacement to support the uprated operation. A follow-up evaluation of the heat loads on these coolers was performed. The evaluation determined one cooler could meet the expected demand, with the option of using the standby unit if necessary. Therefore, it was concluded that no modifications were required on this equipment to support uprated operation.

Essential Service Water

The ESW system provides the cooling water requirements for the component cooling water heat exchangers, the emergency diesel generator coolers, the containment spray heat exchangers, and the control room air conditioning condensers. The ESW system in each Unit consists of two ESW pumps, each with an automatic backwashing duplex strainer and associated piping, valves, and instrumentation. The ESW system in each unit is normally connected by open crossties and is comprised by two identical headers. Each header is served by two pumps and each header, in turn serves half of the system load in each unit.

The ESW system is operated in conjunction with the component cooling water and containment spray systems that are not affected by the rerating. An evaluation has shown that the ESW system flows are acceptable with the unit operating at the uprated conditions. Therefore, no changes to the ESW system are required due to uprating.

Feedwater System

The feedwater system consists of two feedpump suction strainers, two main feedpumps, two parallel strings of high pressure heaters, four feedwater control valves and associated piping, valves and instrumentation.

The feedwater system, in conjunction with the condensate system returns the condensed steam from the condensers and the feedwater heater drains to the steam generators. In transport to the steam generators, additional heat is input to the feedwater as it is passed through the high pressure heaters.

Review of the uprate heat balance and system calculations has determined that the feedwater system components are capable of operating at the uprated conditions with no modification required.

Fire Protection Systems

The fire protection system is independent of plant operating characteristics and are therefore not affected by unit uprate.

Flooding

Flooding in the auxiliary building due to a failure of non-seismic Class I piping has been reviewed. Only systems having access to large water volumes and/or potentially large flowrates were considered as discussed in the FSAR. The only such system is the main feedwater system. Since the changes in flow in the feedwater system are still within the design limits, the results discussed in the FSAR are still applicable.

High Energy Line Break Outside Containment

The mainsteam line break outside containment is described in Attachment 5 of our submittal AEP:NRC:1140, "Technical Specification Change Request, BIT Boron Concentration Reduction", reference 24 of Attachment 5. Submittal AEP:NRC:1140 was approved in reference 25 of Attachment 5. The AEP:NRC:1140, mainsteam line break analysis was performed at 3600MWt. Its continued applicability was verified in the course of preparing of this submittal.

Hydrogen in the Containment following a Loss of Coolant Accident

The post LBLOCA hydrogen analysis is discussed in Section 3.4 of Attachment 6. The subcompartment, hydrogen analysis is described in Attachment 3 of our submittal AEP:NRC:1067, "Reduced Temperature and Pressure Program Analyses and Technical Specification Changes", reference 4 of Attachment 5. Submittal AEP:NRC:1067 was approved in reference 9 of Attachment 5. The AEP:NRC:1067, subcompartment hydrogen analysis was reviewed and its continued applicability verified in the course of preparing this submittal.

Main Steam System

The main steam system delivers the steam produced in the steam generators to the main turbine and to other systems or auxiliary equipment requiring main steam.

This system consists of the piping from the steam generator to the turbine, turbine bypass piping, steam generator stop valves, safety valves, and power relief valves.

This equipment was reviewed by the turbine-generator supplier for possible uprate impact. Evaluation results determined the existing system components can satisfactorily support the unit uprate without modification.

Main/Feedpump Turbine & Auxiliary Equipment

The Unit 2 main turbine consists of a tandem (single shaft) arrangement of a double flow high pressure turbine and three identical double flow low pressure turbines. The feedpump turbine, used to drive the main feedpumps, is an eleven stage, double flow, variable speed turbine. Auxiliary equipment to support main and feedpump turbine operation includes; a lubrication oil system, a steam seal system, a seal oil system, supervisory instruments and control mechanisms.

During the turbine-generator uprate review, the main turbine was found to be adequately oversized such that it will support the uprate parameters. The review also determined that the auxiliary equipment would meet the uprate condition without modification. The feedpump turbine was noted to require maintenance reconditioning in order to restore it's original performance capability. The reconditioning will be performed prior to start-up in the uprated condition.

The planned uprating of unit 2 will not change any of the turbine or rotor configuration that would affect either the maximum theoretical overspeed potential of the turbine generator unit or the speed at which last stage blades would fracture or the turbine rotor discs would burst. These parameters are dependent on the blade and rotor configurations, material properties, and operating speed. None of these parameters will change with the uprating of unit 2.

Makeup Water System

The makeup water system is designed for continuous service and is a shared system supplying demineralized water to both units. The makeup system demand is unaffected by the unit uprate.

Miscellaneous sealing and Cooling Water (MSCW)

The MSCW system supplies cooling, sealing and lubrication water for various non-safety related equipment.

The MSCW system has ample capacity to deal with any increased demand as a result of the unit uprate. Three pumps are available, however, only one pump is needed to meet current requirements.

Moisture Separator Reheater (MSR)

The moisture separator reheater system removes moisture from the high pressure turbine exhaust steam and then superheats the steam before it enters the low pressure turbines. This minimizes water erosion and improves the thermal efficiency of the unit.

Based upon the results of an uprate study performed by the turbine-generator supplier, the right northwest bundle (RNW) of the MSR will not support uprated operation. Since it was determined that repairing or replacing the bundle is not cost effective, a design change has been originated to convert the RNW

bundle to a single pass configuration. Under these conditions, the MSR will operate satisfactorily at the uprated condition.

Nonessential Service Water System

The nonessential service water system is a shared system that provides Lake Michigan water to be used as cooling and makeup water to numerous plant systems and components.

The system consists of four supply pumps each with an automatic backwashing duplex strainer, cooling water suction supply lines from the circulating water intake and discharge tunnels, cooling water lines to the various components being serviced, and associated valves and instrumentation. The nonessential service water flows from the pumps to the equipment served and is then returned to the lake via the circulating water discharge tunnel.

System capacity has been reviewed and found to be sufficient to support the unit power uprate. Major system heat loads were reviewed and determined to be only marginally effected at the uprated condition.

Spent Fuel Pool Cooling System

The primary function of the spent fuel pool cooling system is to remove decay heat generated by the spent fuel assemblies stored in the pool. Decay heat generation is proportional to plant power level. The uprate will therefore result in an increase in the total decay heat that the spent fuel pool cooling system must remove. This increase in heat load has been evaluated and results have been submitted to NRR under a separate cover letter (AEP:NRC:1202A). The purification function is controlled by the spent fuel pool cooling system demineralization and filtration rates that are not affected by uprated power operation.

Steam Generator Blowdown and Blowdown Treatment Systems

The steam generator blowdown system is used to control secondary side water chemistry. It is also used to drain the steam generators during plant outages. The steam generator blowdown treatment system is used in the event of a primary to secondary steam generator tube leak to remove radioactive ions and particulates.

The blowdown rate is controlled by the operator, dependent upon system conditions. The uprated condition in-and-of-itself will not result in a change to the blowdown rate or the treatment system.

Turbine Auxiliary Cooling Water System (TACWS)

The TACWS is a subsystem of the condensate system. It uses main cycle condensate to remove heat from various heat exchangers associated with the turbine-generator unit.

The TACWS of each unit consists of two turbine auxiliary cooling pumps, one turbine auxiliary cooler and various other heat exchangers in the turbine-generator unit which are provided cooling water by the system.

An uprate study, performed by the turbine-generator supplier, identified the existing TACW pumps/coolers as adequate to support the unit uprate. As a result, no modifications are planned to this system.

Turbine Bypass (Steam Dump) System
The turbine bypass system allows steam to bypass the turbine and go directly to the condenser. The steam dump system is sized to provide sufficient capacity for a load rejection. No hardware modifications are planned for this system as a result of the unit uprate.

Vacuum Priming
The vacuum priming system removes air and non-condensable gasses from various heat exchangers/condensers and centrifugal pumps on lake water systems. Operation of this system is unaffected by the unit uprate.

Radiation Monitoring System

The design of the radiation monitoring system, as described in the UFSAR, is not affected by the uprating of Unit 2.

Waste Disposal System

The design of the waste disposal system, as described in the UFSAR, is not affected by the uprating of Unit 2.