XN-NF-84-74
SUPPLEMENT 1

PLANT TRANSIENT ANALYSIS FOR H.B. ROBINSON UNIT 2 AT 2300 MWt WITH INCREASED $F^{N}_{\Delta H}$ SUPPLEMENT 1: ANALYSIS OF CONTROL ROD MISOPERATION EVENTS (RCCA MISALIGNMENTS)

SEPTEMBER 1984

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, INC.

XN-NF-84-74
Supplement 1

Issue Date: 9/13/84

PLANT TRANSIENT ANALYSIS FOR H. B. ROBINSON UNIT 2 AT 2300 MWt WITH INCREASED $F^N_{\Delta H}$

SUPPLEMENT 1: ANALYSIS OF CONTROL ROD MISOPERATION EVENTS (RCCA MISALIGNMENTS)

Prepared by: 12 Sept 84
F. T. Adams, Lead Engineer
PWR Safety Analysis

Prepared by: I. Z. Stone 12 Sept 84
I. Z. Stone, Engineer
PWR Neutronics

Reviewed by: W V Kayser 9/13/84
W. V. Kayser, Manager
PWR Safety Analysis

Reviewed by: MRK F B Skogen 13 SEP 84
F. B. Skogen, Manager
PWR Neutronics

Concur: for 13 Sept 84
J. C. Chandler, Lead Engineer
Reload Fuel Licensing

Approve: W V Kayser for 9/13/84
R. B. Stout, Manager
Licensing & Safety Engineering

Approve: Williamson 9/13/84
H. E. Williamson, Manager
Neutronics & Fuel Management

Approve: for 13 SEP 84
G. A. Sofer, Manager
Fuel Engineering & Technical Services

gf

EXXON NUCLEAR COMPANY, INC.

NUCLEAR REGULATORY COMMISSION DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

This technical report was derived through research and development programs sponsored by Exxon Nuclear Company, Inc. It is being submitted by Exxon Nuclear to the USNRC as part of a technical contribution to facilitate safety analyses by licensees of the USNRC which utilize Exxon Nuclear-fabricated reload fuel or other technical services provided by Exxon Nuclear for light water power reactors and it is true and correct to the best of Exxon Nuclear's knowledge, information, and belief. The information contained herein may be used by the USNRC in its review of this report, and by licensees or applicants before the USNRC which are customers of Exxon Nuclear in their demonstration of compliance with the USNRC's regulations.

Without derogating from the foregoing, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document will not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this document.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

This document presents the analysis of the Control Rod Misoperation events in support of H.B. Robinson Unit 2 Cycle 10 and subsequent cycles bounded by the conditions of this analysis. The Standard Review Plan (SRP)[1] identifies this event as 15.4.3. Included are four sub-events: Single Full Length Rod Control Cluster Assembly (RCCA) Withdrawal, Static Misalignment of a Full Length RCCA, Dropped Full Length RCCA, and Dropped Full Length RCCA Bank. The disposition of SRP events for H.B. Robinson is presented in Reference 2. Analysis of other SRP events for H.B. Robinson Cycle 10 is presented in References 3, 4 and 5.

Section 2.0 of this document presents a summary of the analytical results for the Control Rod Misoperation events. Section 3.0 presents the conditions used in the safety analysis and the results of the analysis.

Analysis of the RCCA drop events presented here is structured to address a potential unreviewed safety issue for rod drop protection on turbine runback plants.[6] In Reference 6, a more limiting scenario for the rod drop event is postulated than had been previously analyzed. The results of this analysis bound that scenario with results satisfying acceptance criteria.

The General Design Criteria, in existence at the time that H.B. Robinson Unit 2 was licensed for operation in July 1970, are contained in the Proposed Appendix A to 10CFR50, General Design Criteria for Nuclear Plants, published in the Federal Register on July 11, 1967. The proposed 1967 criteria, together with the responses, are presented in Sections 3.1.1 and 3.1.2 of the Updated H.B. Robinson Unit 2 FSAR.

2.0 SUMMARY

The results of Control Rod Misoperation events for H.B. Robinson Unit 2 Cycle 10 are summarized in Table 2.1. These results confirm that H.B. Robinson Unit 2 may be operated in Cycle 10 as described in the Cycle 10 Safety Analysis Report, while meeting the acceptance criteria for each event as defined in the licensing basis for H.B. Robinson Unit 2. These results are applicable to future cycle operation where operating conditions are bounded by those employed in this analysis. Since no rod bow penalty is assessed to H.B. Robinson Unit 2 for assembly burnups less than 44 GWD/MTU,(3) the analysis accounts for the effects of rod bow on MDNBR.

The analysis of the dropped full length RCCA and RCCA bank events considers the effect of turbine runback with both Beginning of Cycle (BOC) and End of Cycle (EOC) neutron kinetics feedbacks. This analysis addressed the potential unreviewed safety issue stated in Reference 6. The analysis demonstrated acceptable results for the dropped full length RCCA and RCCA bank events.

Table 2.1 Summary of Results of Control Rod Misoperation Analyses

	Maximum Power Level, MWt	Maximum Core Average Heat Flux Btu/hr-ft^2	Maximum Pressurizer Pressure, psia	MDNBR (XNB)
Anticipated Operational Occurrences				
Statically Misaligned Full Length RCCA	2346	187940	2220	$\geq$1.17†
Dropped Full Length RCCA (-80 pcm)	2346	187940	2303	$\geq$1.17†
Dropped Full Length RCCA (-200 pcm)	Power, heat flux and pressure decrease monotonically from the initial condition			$\geq$1.17†
Postulated Accidents (Condition III Events)				
Single Full Length RCCA Withdrawal at Power	2728	212850	2275	††

† Includes effects of radial power redistribution.

†† Less than 10% of the core experiences boiling transition. Radiological release less than 10% of 10 CFR 100 limits.

3.0 ANALYSIS OF EVENTS

This section presents results of analyses of Control Rod Misoperation Events for H.B. Robinson Unit 2. Subsections in the report are numbered in accordance with the Standard Review Plan to facilitate review. The reader is referred to Subsections 15.0.1 through 15.0.10 of Reference 3 for classification of plant conditions and a discussion of nominal operating parameters, control and protection system functions, setpoints, and capacities. Values of key input parameters employed in these analyses are given in the following results section.

15.4.3 CONTROL ROD MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

15.4.3.1 Identification of Causes and Event Description

Rod Cluster Control Assembly (RCCA) misoperation events include:

a) Withdrawal of a single full length RCCA

b) Static misalignment of a single full length RCCA

c) Dropped full length RCCA

d) Dropped full length RCCA bank

Each RCCA has a position indicator which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod bottom light. Group demand position is also indicated. The full-length RCCAs are always moved in preselected banks and the banks are always moved in the same preselected sequence.

The statically misaligned RCCA, dropped RCCA, and dropped RCCA bank events are classified as Condition II events. The withdrawal of a single RCCA event is classified as a Condition III event.

15.4.3.1.1 Withdrawal of a Single Full Length RCCA

The withdrawal of a single full length RCCA is initiated by the inadvertant withdrawal of a single control rod at power. The ensuing reactivity insertion causes core power to increase. In the event that the secondary steam dump control system does not respond to the increased power production, secondary system temperature and pressure will increase, causing a corresponding increase in primary coolant temperature. This increase in primary coolant temperature occurs slowly enough that the pressurizer pressure control system, if available, is capable of suppressing the primary pressure increase. The degradation of coolant conditions coupled with the power increase is essentially the same as expected for RCCA bank withdrawals at power, and may approach DNB conditions in the hot channel.

The single RCCA withdrawal is distinguished from the withdrawal of an RCCA bank by a severe radial power redistribution. High radial power peaking is quite localized in the region of the single withdrawn RCCA and may, in severe cases, surpass the design limits. Thus assemblies in the immediate vicinity of the withdrawn RCCA may experience boiling transition. Such exposure would be limited to short time periods. Some fuel damage might occur.

Primary protection for this event is afforded by the high nuclear flux trip and the overtemperature ΔT trip.

No single electrical or mechanical failure in the Rod Control System could cause the accidental withdrawal of a single RCCA from the inserted RCCA bank during full power operation. Procedures are available to permit the operator to withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event can occur only as the result of multiple wiring failures or multiple operator actions. The probability of such a combination of conditions is low. This event is therefore classified as a Class III event during which some fuel damage is permitted.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, the rod position indicators would indicate the relative positions of the assemblies in the bank. Withrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would similarly result in the same visual indications. Withdrawal of a single RCCA results both in a positive reactivity insertion tending to increase core power, and in an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the high neutron flux reactor trip, although due to the increase in local power density it is not possible in all cases to provide assurance that the core safety limits will not be violated.

15.4.3.1.2 Static Misalignment of a Single Full Length RCCA

The static misalignment of an RCCA is defined as a malfunction of the Control Rod Drive (CRD) mechanism, or of the rod control power supply, which causes an RCCA to be out of alignment with its bank; i.e., either higher or lower than any of the other RCCAs in the same bank. The reactor is in the steady state at rated full power conditions, and no excursion of core temperature, pressure, flow, or power occurs. For extreme RCCA misalignments, the core radial power distribution may be characterized by peaking factors in excess of design limits. Highly localized increases in clad surface heat flux, coolant temperature, and flow diversion may occur. In severe cases, the SAFDL on DNB may be approached.

The most severe misalignment event occurs at full power operation, with D Bank fully inserted (beyond its PDIL) and one D Bank RCCA fully withdrawn.

The full-length RCCAs are always moved in preselected sequence. A quadrant tilt monitor alarm (upper and lower ex-core neutron detectors) is provided to indicate significant power tilts. If this alarm is temporarily out of service, periodic checks of individual rod positions and ex-core detector currents, and even core symmetry checks using in-core thermocouples and movable detectors can be made.

The operator is provided with rod position indication for each RCCA. An alarm is actuated when any RCCA bottom defeat switch is actuated so that an RCCA can be inserted into the core. This defeat switch must be actuated to prevent a load cutback.

The operator is thus presented with sufficient information to prevent inadvertent insertion of an RCCA. Should this occur, automatic rod withdrawal is blocked by one out of four high nuclear power indications, high ΔT in any two loops, or T_{avg} deviation in any single loop. Alarms are sounded in all cases and in the case of the ΔT deviation a load cutback also occurs.

15.4.3.1.3 <u>Dropped Full Length RCCA and Dropped Full Length RCCA Bank</u>

The event is initiated by a de-energized control rod drive mechanism or by a malfunction associated with an RCCA bank during power operation. The result is that a single RCCA or RCCA bank falls into the core. Core power decreases rapidly at first in response to the negative reactivity insertion. A decrease in moderator temperature results from the initial power reduction. At EOC conditions, a strongly negative moderator temperature coefficient can return the reactor to the full power condition with an elevated radial power peaking factor consequent to the dropped RCCA. Elevated clad heat flux in the hot assembly may result in an approach to the DNBR SAFDL.

An automatic and redundantly actuated reduction in turbine load demand (turbine runback) is provided as protection for this event. The turbine load reduction reduces secondary steam flow, causing a tendency for the secondary side temperature and pressure to increase. Thus the primary coolant temperature decrease characteristic of the event is mitigated, reducing the reactivity insertion contingent on cooldown and reducing the ultimate power level at which the reactor stabilizes. Coolant conditions in the hot channel are thus improved as a result of turbine runback.

With small dropped rod worths and consequent small initial power reductions, the occurrence of the turbine runback can result in a net primary system heatup. At BOC conditions, moderator feedback may be insufficient to damp out the primary temperature increase. The reactor may trip on overtemperature ΔT or reach the upper limit on attainable primary temperature imposed by the secondary safety valves. The radial peaking factor may exceed design limits in severe cases, so the SAFDL on DNB may be approached in the hot channel.

If a RCCA drops into the core during power operation, it would be detected by either a rod bottom signal device or by the use of the excore chambers. The rod bottom signal device provides an individual position indication signal for each RCCA. The other independent indication of an RCCA drop is obtained through the excore power range channel signals. This rod drop detection circuit is actuated upon sensing a rapid decrease in local flux such as could occur from depression of flux in one region by a dropped RCCA.

A rod drop signal from any rod position indication channel, or from one or more of the four power range channels, initiates protective action by reducing turbine load by a preset adjustable amount and blocking of further automatic rod withdrawal. Either action individually prevents core damage. The turbine runback is redundantly obtained by acting upon the turbine load limit and on the turbine governor control system. The rod stop is also redundantly actuated.

15.4.3.2 Analysis Method

The analyses are performed by coupling a conservative power peak to transient response and DNB calculations. The power peak associated with each event is characterized through an augmentation factor which relates the maximum power peak to the steady state power peak. The steady state power distributions and augmentation factors are calculated with the XTGPWR reactor simulator[7].

15.4.3.2.1 Withdrawal of a Single Full Length RCCA

The transient response of the reactor system exclusive of radial power redistribution effects is as calculated with the PTSPWR2 code[8] for the most limiting case of SRP event 15.4.2, uncontrolled RCCA withdrawal at power[3]. The coolant flow rate, primary pressure, and core inlet coolant temperature boundary conditions at the time of MDNBR (determined by PTSPWR2 hot channel calculations) are transferred to the XCOBRA-IIIC methodology[9] for calculation of MDNBR. The core average heat flux at the time of MDNBR is adjusted to include design power peaking and a radial peaking augmentation factor calculated to describe the radial power peaking redistribution due to the single withdrawn RCCA.

The percent of the fuel to experience boiling transition for the event is determined using the XNB critical heat flux correlation statistics, the calculated MDNBR for the event, and a core radial power distribution representative of the power redistribution associated with the event.

15.4.3.2.2 Static Misalignment of a Single Full Length RCCA

Primary system pressure, core inlet temperature, and coolant flow rate at the rated full power operating point are input to the XCOBRA-IIIC code to calculate MDNBR. The rated full power core average clad surface heat flux is input to the MDNBR calculation after having been adjusted to include the design radial and axial power peaking distribution factors and a radial peaking augmentation factor calculated to bound the radial power redistribution characteristics of a misaligned RCCA.

15.4.3.2.3 Dropped Full Length RCCA and RCCA Bank

The analysis is performed using the PTSPWR2 code and XCOBRA-IIIC. The PTSPWR2 code models the salient system components and calculates neutron power, fuel thermal response, and fluid conditions. The fluid conditions and rod surface heat transport at the time of MDNBR are transferred to the XCOBRA-IIIC methodology for calculation of the MDNBR. A radial power peaking augmentation factor on $F^N_{\Delta H}$ is included in the MDNBR calculation to account for radial power redistribution effects typical of the event. The dropped RCCA bank is distinquished from the dropped RCCA in the PTSPWR2-XCOBRA analysis only by the greater magnitude of rod bank worth and radial peaking augmentation factors associated with the dropped RCCA bank.

15.4.3.3 Definition of Events Analyzed and Bounding Input

For control rod misoperation events, the maximization of power peaking results in a reduction in the DNBR. To assure that bounding values are determined for the radial power peaking, the following approach is used for each event. The increase in power peaking above that associated with equilibrium steady state conditions is determined for a spectrum of cycle exposures and applicable control rod configurations. Based on these results, a conservative augmentation factor is derived. This augmentation factor is then applied to the allowable $F_{\Delta H}$ to ensure a bounding value for the peak pin power input to the DNB analysis.

15.4.3.3.1 Withdrawal of a Single Full Length RCCA

The initial input is selected such that the analysis bounds power operation. The initial input for the case analyzed is the same as the previously identified to provide the limiting transient response for the uncontrolled RCCA bank withdrawal at power(3). That case employed positive neutron kinetics feedbacks in an RCCA bank withdrawal from rated power. For the withdrawal of a single full length RCCA, a radial power peaking augmentation factor of 1.27 is employed which is calculated to bound Cycle 10 operation.

Conservative conditions established for the event are given below:

Control	Manual
Core Power	Nom. +2% Rated
Core Coolant Inlet Temperature	Nom. +4°F
Primary Pressure	Nom. -30 psi
Pressurizer Spray	Available
Pressurizer PORVs	Available
Pressurizer Level	Nom. -10%
Steam Bypass	Disable
Steam Line PORVs	Disable
Reactor Trips	Power Range High Flux (High)
Pellet-to-Clad Heat Transfer Coefficient	Maximum
Reactivity Insertion Rate	$2.01 \times 10^{-5}\ \Delta\rho$/sec

Moderator Temperature Coefficient	1.2 x BOC value ($6.0 \times 10^{-5} \Delta\rho/^{o}F$)
Doppler Coefficient	.8 x BOC value ($-0.8 \times 10^{-5} \Delta\rho/^{o}F$)

15.4.3.3.2 Static Misalignment of a Single Full Length RCCA

The event is analyzed at the rated full power operating point to bound power operation. Analysis inputs are listed in Table 15.4.3.3. These values reflect the following allowance from nominal full power operating conditions:

Power	Nom. +2%
Core Inlet Temperature	Nom. +4^{o}F
Pressurizer Pressure	Nom. -30 psi
Coolant Flow	Nom. -3%

The radial peaking factor augmentation is 1.176, calculated to bound Cycle 10 operation.

15.4.3.3.3 Dropped Full Length RCCA and RCCA Bank

The characteristic system response for this event is strongly dependent on the neutron kinetics feedbacks, the worth of the dropped RCCA or RCCA bank, and on the availability of the automatic rod control (ARC) system. The effects of operable ARC are not considered here and must be addressed in subsequent analysis. Cases involving dropped RCCA banks (-700 pcm to -1800) do not result in appreciable degradation of MDNBR due to the action of t turbine runback, combined with the large initial power reduction. In the absence of ARC, negative feedbacks bounding of EOC conditions result in system responses to dropped RCCAs (-80 pcm to -250 pcm) which are bounded with respect to MDNBR (exclusive of power redistribution) by the rated full power operating point. With a maximum radial power peaking augmentation factor of 1.084, MDNBR calculations for the dropped RCCA event at EOC will result in MDNBRs greater than that calculated for the RCCA misalignment event.

The remaining case is a dropped RCCA with positive reactivity feedback bounding of BOC conditions. The case is analyzed for a spectrum of dropped RCCA worths ranging from -80 pcm to -250 pcm. This range of worth bounds Cycle 10 RCCA worths and includes allowances for cycle to cycle variations. The case analyzed is described below:

Control	Manual
Core Power	Nom. +2% Rated
Core Coolant Inlet Temperature	Nom. +4°F
Primary Pressure	Nom. -30 psi
Pressurizer Spray	Available
Pressurizer PORVs	Available
Pressurizer Level	Nom. -10%
Steam Bypass	Disable
Steamline PORVs	Disable
Reactor Trip	OT-ΔT
Turbine Runback	Available

Pellet to Clad Heat Transfer Coefficient	Maximum
Dropped RCCA Worth	-80 pcm to -250 pcm
Moderator Temperature Coefficient	1.2 x BOC ($6.0 \times 10^{-5}\ \Delta\rho/^{o}F$)
Doppler Coefficient (large RCCA worth)	1.2 x BOC ($-1.2 \times 10^{-5}\ \Delta\rho/^{o}F$)
Doppler Coefficient (small RCCA worth)	0.8 x BOC ($-0.8 \times 10^{-5}\ \Delta\rho/^{o}F$)

The system state is set to minimize MDNBR and conservatively bounds operating conditions. For larger RCCA worths the reactor does not return to full power; the nominal BOC Doppler coefficient is increased by a factor of 1.2 to maximize the power level reached. For smaller RCCA worths, reactor power can increase above the initial condition and a 0.8 multiplier or the nominal BOC Doppler feedback is conservative.

15.4.3.4 Analysis of Results

15.4.3.4.1 Withdrawal of a Single Full Length RCCA

Initial plant operating conditions assumed in the analysis are summarized in Table 15.4.3.1. The event sequence is summarized in Table 15.4.3.2. A high neutron flux trip occurs at 25.8 seconds, followed shortly by occurrence of MDNBR. A peak core average heat flux of 113% of rated occurs coincident with MDNBR. System pressure is controlled by the pressurizer pressure control system, reaching a peak of 2275 psia at 20.7 seconds. The system response is typical of RCCA withdrawal events.

MDNBR for the event is 0.64, less than the XNB correlation safety limit of 1.17. The extreme radial power peaking calculated for the single RCCA withdrawal is localized in the neighborhood of the withdrawn RCCA. Less than 10% of the rods in the core are calculated to experience boiling transition.

The radiological consequences evaluation(5) for the Design Basis LOCA event assumed all fuel assemblies in the core failed. Acceptable results were obtained for the LOCA, i.e., less than 10% of the 10 CFR 100 limits. This is also an acceptable result for a Condition III event. Using the conservative assumption that boiling transition results in fuel failure, less than the full core exhibited DNB and the integrity of the primary system is maintained; therefore, the radiological results of the withdrawal of a single full length RCCA event are bounded by those of the LOCA event.

The single RCCA withdrawal event is classified as a Condition III event(3). Less than 10% of the core experiences boiling transition, with radiological release less than 10% of 10 CFR 100 limits. Reactor vessel pressurization is well below 110% of the design limit. It is not anticipated that core cooling would be significantly hindered by less than 10% fuel failures. No more limiting fault is engendered by the occurrence of the event. The result of the analysis is thus in conformance with the acceptance criteria for a Condition III event and is therefore acceptable.

15.4.3.4.2 <u>Static Misalignment of a Single Full Length RCCA</u>

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which Bank D is fully inserted with one RCCA fully withdrawn; this 12 ft. misalignment error is analyzed. The condition for which the analysis is performed are presented in Table 15.4.3.3.

The calculated MDNBR for the Static Misalignment of a Full Length RCCA is above the XNB critical heat flux correlation safety limit of 1.17. The peak pellet linear heat generation is well below 21 kw/ft, so that fuel centerline melt does not occur. Since no fuel failure is calculated to occur, there is no radiological release consequent to this event. The result of the analysis is thus in conformance with the acceptance criteria for Condition II events and is therefore acceptable.

15.4.3.4.3 Dropped Full Length RCCA or RCCA Bank

The event is initiated by a step negative reactivity insertion representative of dropped RCCA worth. Concurrently, the turbine steam demand runs back at a rate of 200% per minute, reaching a minimum value of 70% of rated load after about 9 seconds. Core power drops rapidly in response to the dropped rod. This power decrease is accompanied by an initial decrease in core average moderator temperature which, coupled with BOC kinetics feedbacks, further depresses power in the initial few seconds of the event.

For the low rod worth cases, the initial power reduction is smaller than the reduction in steam demand caused by the turbine runback. This mismatch between primary power production and secondary heat removal results eventually in a heatup of the primary system which, when coupled with the BOC kinetics feedback, results in a gradual power ascension. Average coolant temperature reaches about 600°F. The power and coolant temperature increases result in a reactor trip on the OT-ΔT setpoint. MDNBR calculations demonstrate that sufficient design margin exists in the

OT-ΔT setpoint to preclude penetration of the SAFDL on DNB with the maximum calculated radial peaking augmentation factor for a dropped RCCA (1.084).

For the higher worth cases, the initial power reduction is large, and more heat is extracted at the 70% load demand point than is produced by the core. The resulting cooldown of the moderator coupled with BOC kinetics feedbacks acts to further reduce core power. Pressure decays toward the low pressurizer pressure reactor trip setpoint. MDNBR (exclusive of power redistribution effects) is monotonically increasing throughout the event. With consideration of radial power redistribution, MDNBR does not fall below the initial value.

The limiting case with respect to MDNBR is the least worth RCCA drop at BOC conditions, which terminates on the OT-ΔT trip with MDNBR greater than the XNB correlation safety limit of 1.17. An intermediate worth case (-150 pcm) reached the OT-ΔT trip simultaneous with the opening of the steam generator safety values, thus locating the intersection of the OT-ΔT trip with the steam generator safety valve (SGSV) limit line. The SGSV limit line describes an upper limit on attainable primary coolant temperature as a function of power level due to the opening of the SGSVs at their setpoint pressure. At this intersection, MDNBR margin is minimized for points on the SGSV limit line not protected by the OT-ΔT trip. Calculated MDNBR for the -150 pcm case thus bounds points on the unprotected (low power) part of the SGSV limit line.

A summary of initial conditions for RCCA drop event is given in Table 15.4.3.4. Event summaries for the -80 pcm dropped RCCA and the -200 pcm dropped RCCA are provided in Table 15.4.3.5. Figures 15.4.3.1 through 15.4.3.4 depict the system response to a dropped RCCA of -80 pcm. Figures 15.4.3.5 through 15.4.3.8 depict system response to the -200 pcm dropped rod event.

Calculated MDNBR for the event remains above the XNB correlation safety limit and peak pellet LHGR remains below 21 kw/ft. Fuel failure is thus precluded for this event, and no radiological consequences are expected. System pressure response does not challenge the vessel design limits. The event is not anticipated to result in a more serious plant condition. The results of the analysis are thus in conformance with the acceptance criteria for a Condition II event and are acceptable on that basis.

15.4.3.5 Conclusion

15.4.3.5.1 Withdrawal of a Single Full Length RCCA

For the case of the accidental withdrawal of a single RCCA, with the reactor initially operating at full power with Bank D at the insertion limit, an upper bound of the number of fuel rod experiencing DNBR 1.17 is 10 percent of the total fuel rods in the core. This event therefore satisfies 10 CFR 100 criteria.

15.4.3.5.2 Static Misalignment of a Single Full Length RCCA

An RCCA out of position can result only from a malfunction in the mechanism or its associated power supply and, in such a case, it is clearly indicated to the operator by independent monitoring systems. The cases discussed above have indicated that the DNB ratio remains greater than 1.17 in the event of a rod misalignment. The DNB SAFDL is therefore satisfied for this event.

15.4.3.5.3 Dropped Full Length RCCA

Protection for a dropped full length RCCA is provided by automatic turbine power cutback. As the analyses show, a DNBR greater than 1.17 is maintained. The DNB SAFDL is therefore satisfied for this event.

15.4.3.5.4 Dropped Full Length RCCA Bank

The impact of a dropped full length RCCA bank is mitigated by automatic turbine cutback. The results of this event are bounded by the results of the dropped full length RCCA. The DNB SAFDL is therefore satisfied for this event.

Table 15.4.3.1 Single Full Length RCCA Withdrawal
Summary of Initial Conditions

Condition	Value
Power (MWt)	2346
Core Inlet Temperature (^{o}F)	550.2
Pressurizer Pressure (PSIA)	2220
Pressurizer Level	Programmed Full Power Level -10%
Reactor Coolant System Flow Rate (lbm/hr)	97.29×10^6
Steam Dome Pressure (PSIA)	828.3

Table 15.4.3.2 Single Full Length RCCA Withdrawal
Event Summary

Event	Time (sec)
Uncontrolled RCCA Withdrawal begins	0.0
High Neutron Flux Setpoint Reached	25.8
Scram Results in Rod Motion	26.4
Minimum DNBR Occurs	26.6

Table 15.4.3.3 Static Misalignment of a Full Length RCCA
Summary of Conditions

Condition	Value
Power (MWt)	2346
Core Inlet Temperature (^oF)	550.2
Pressurizer Pressure (PSIA)	2220.
Reactor Coolant System Flow Rate (lbm/hr)	97.29×10^6
$F^N_{\Delta H}$*	1.94
F_Z	1.65

* $F^N_{\Delta H}$ = Technical Specification Limit of 1.65 x 1.176 Augmentation Factor.

Table 15.4.3.4 Dropped Full Length RCCA
Summary of Initial Conditions

Condition	Value
Power (MWt)	2346.
Core Inlet Temperature (^{O}F)	550.2
Pressurizer Pressure (PSIA)	2220.
Pressurizer Level	Nom. -10%
Reactor Coolant System Flow Rate (lbm/hr)	97.29×10^6
Steam Power Pressure (PSIA)	828.3

Table 15.4.3.5 Dropped Full Length RCCA
Event Summary

Case 1: Low Worth RCCA (-80 pcm)

Event	Time (sec)
Dropped RCCA Fully In	0.0
Turbine Runback Begins	0.0
Turbine Runback Reaches Low Load Limit	9.0
Overtemperature T Setpoint Reached	47.3
Scram Results in Rod Motion	53.3
Minimum MDNBR Occurs	53.6

Case 2: High Worth RCCA (-200 pcm)

Event	Time (sec)
Dropped RCCA Fully In	0.0
Turbine Runback Begins	0.0
Turbine Runback Reaches Low Load Limit	9.0
MDNBR Occurs	0.0

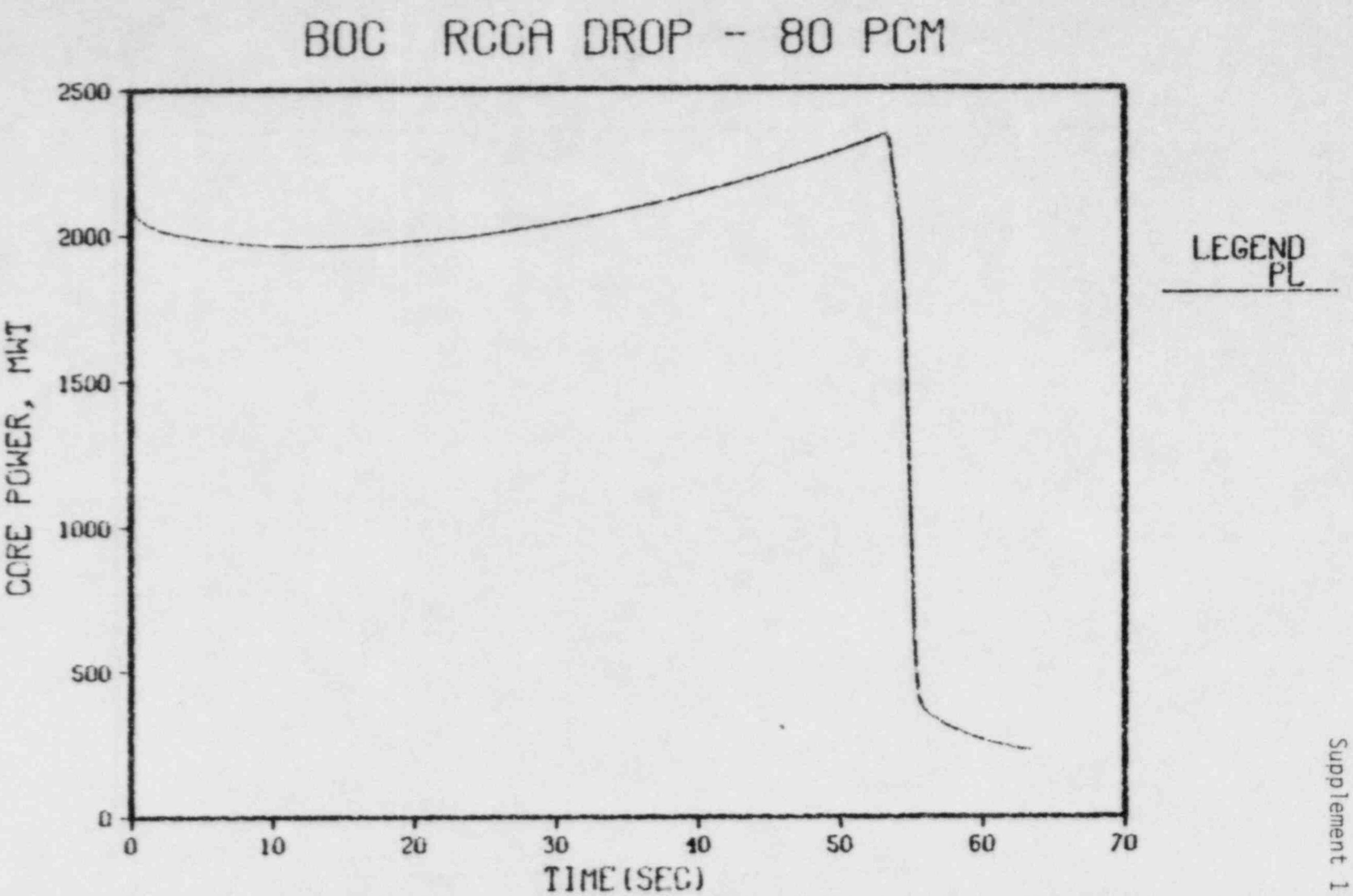

Figure 15.4.3-1 Dropped Full Length RCCA, Low Worth BOC, Reactor Power

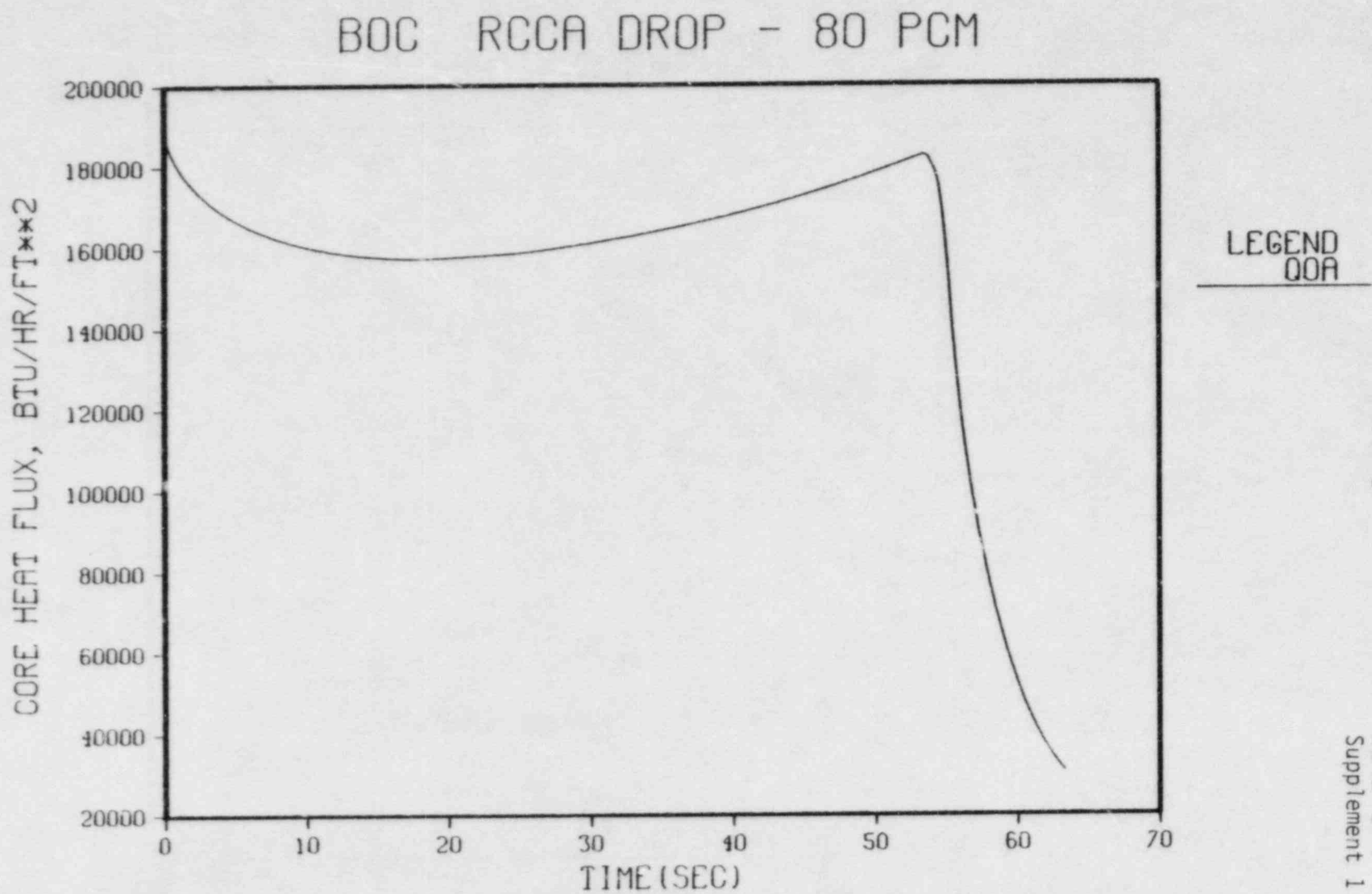

Figure 15.4.3-2 Dropped Full Length RCCA, Low Worth BOC,
Core Average Heat Flux

11.25.18 MON 10 SEP. 1984 JOB=CERNDSC , U C C DISSPLA VER 8.2

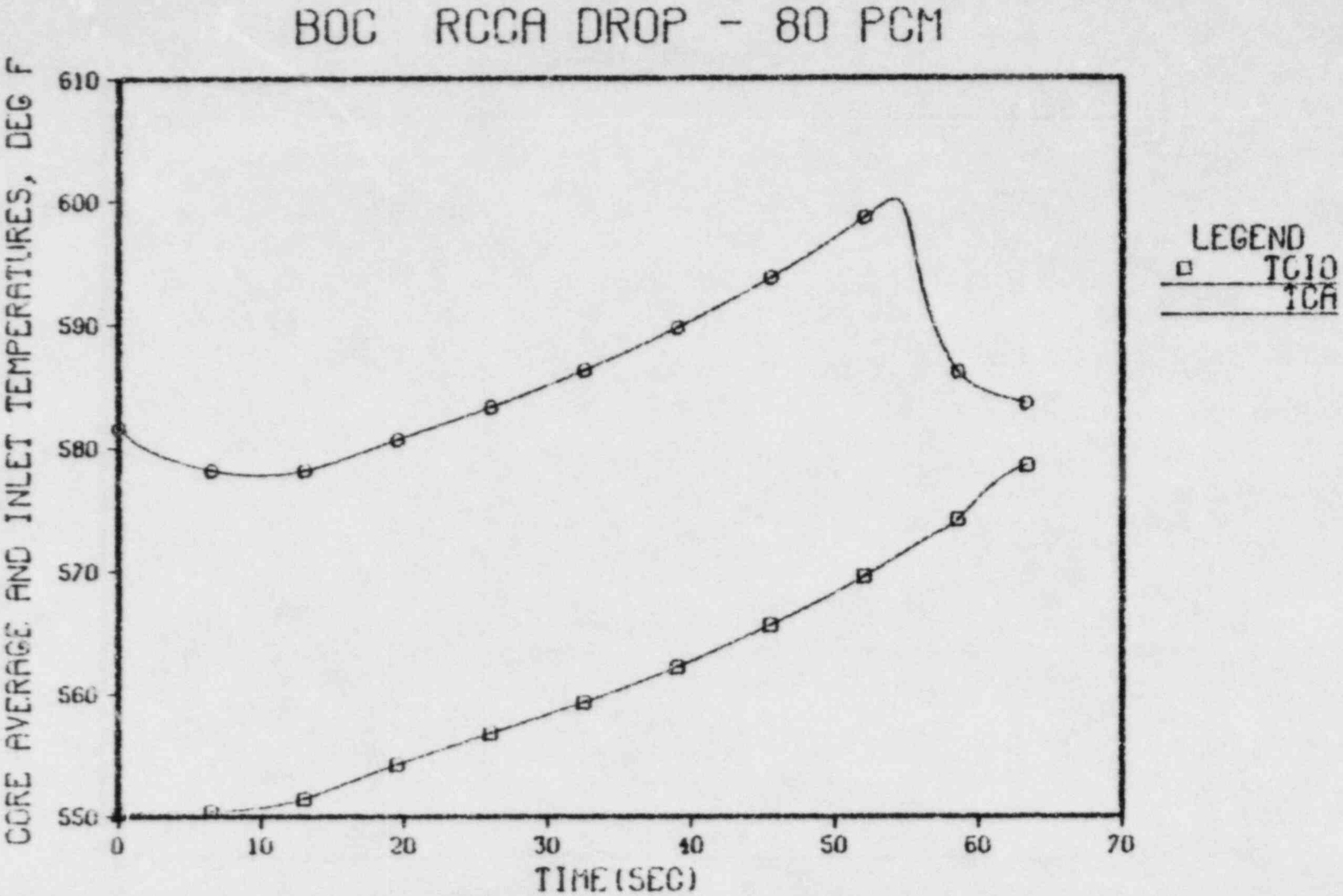

Figure 15.4.3-3 Dropped Full Length RCCA, Low Worth BOC,
Core Coolant Inlet (TCIØ) and Average (TCA) Temperatures

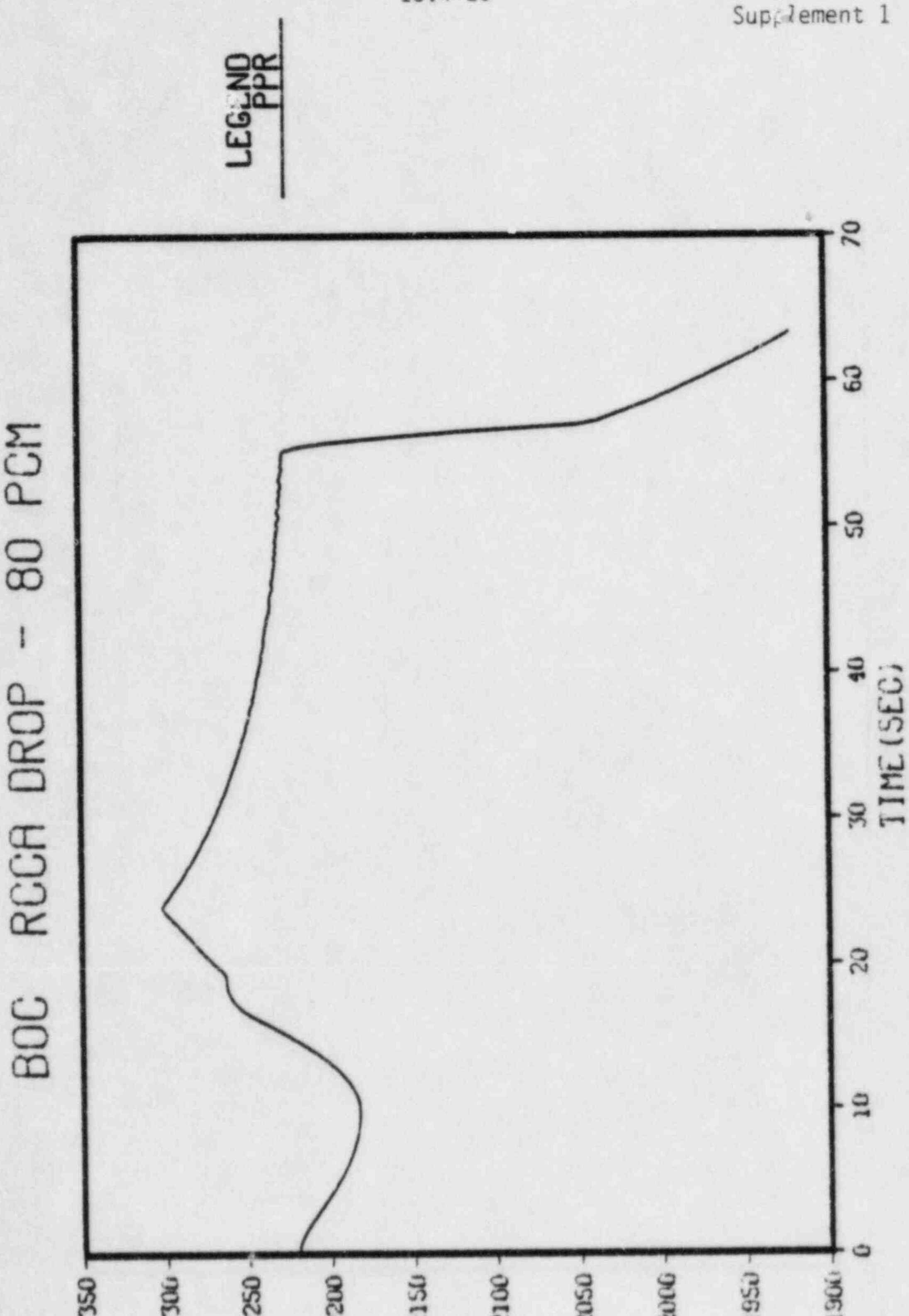

Figure 15.4.3-4 Dropped Full Length RCCA, Low Worth BOC, Pressurizer Pressure

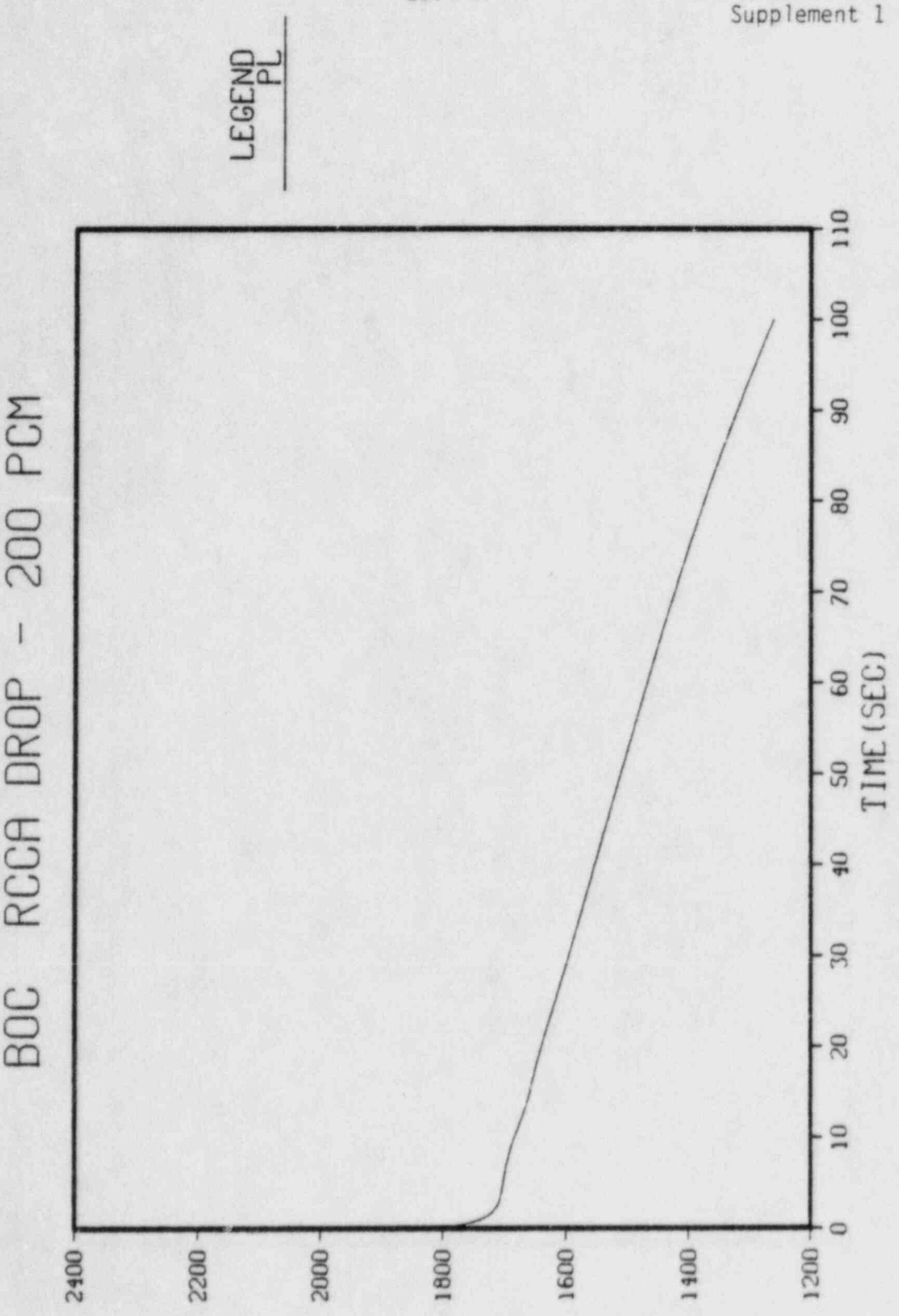

Figure 15.4.3-5 Dropped Full Length RCCA, High Worth BOC, Reactor Power

PLOT 1 11.38.10 MON 10 SEP, 1984 JOB=GEAN087, U C C DISSPLA VER 8.2

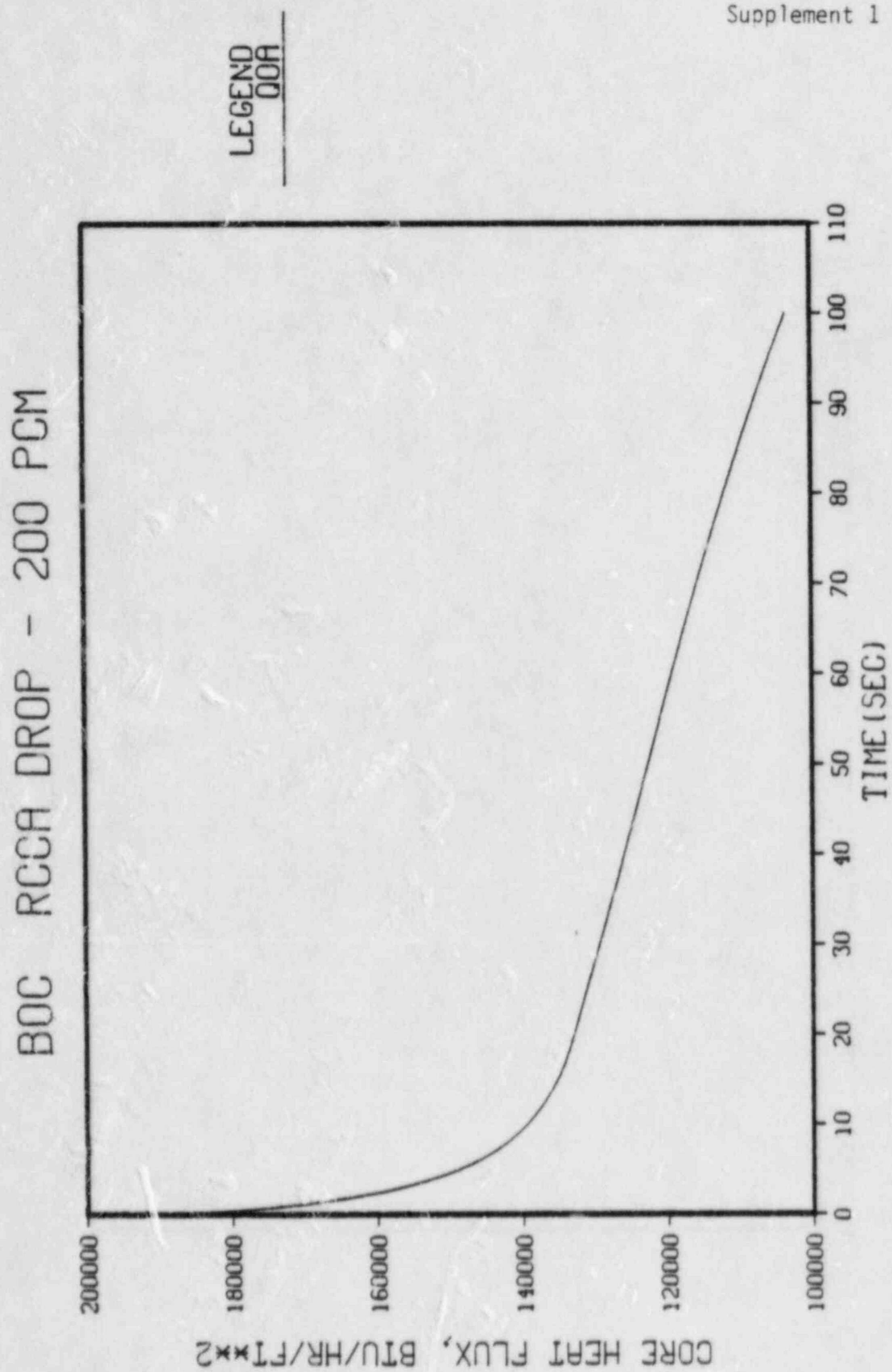

Figure 15.4.3-6 Dropped Full Length RCCA, High Worth BOC, Core Average Heat Flux

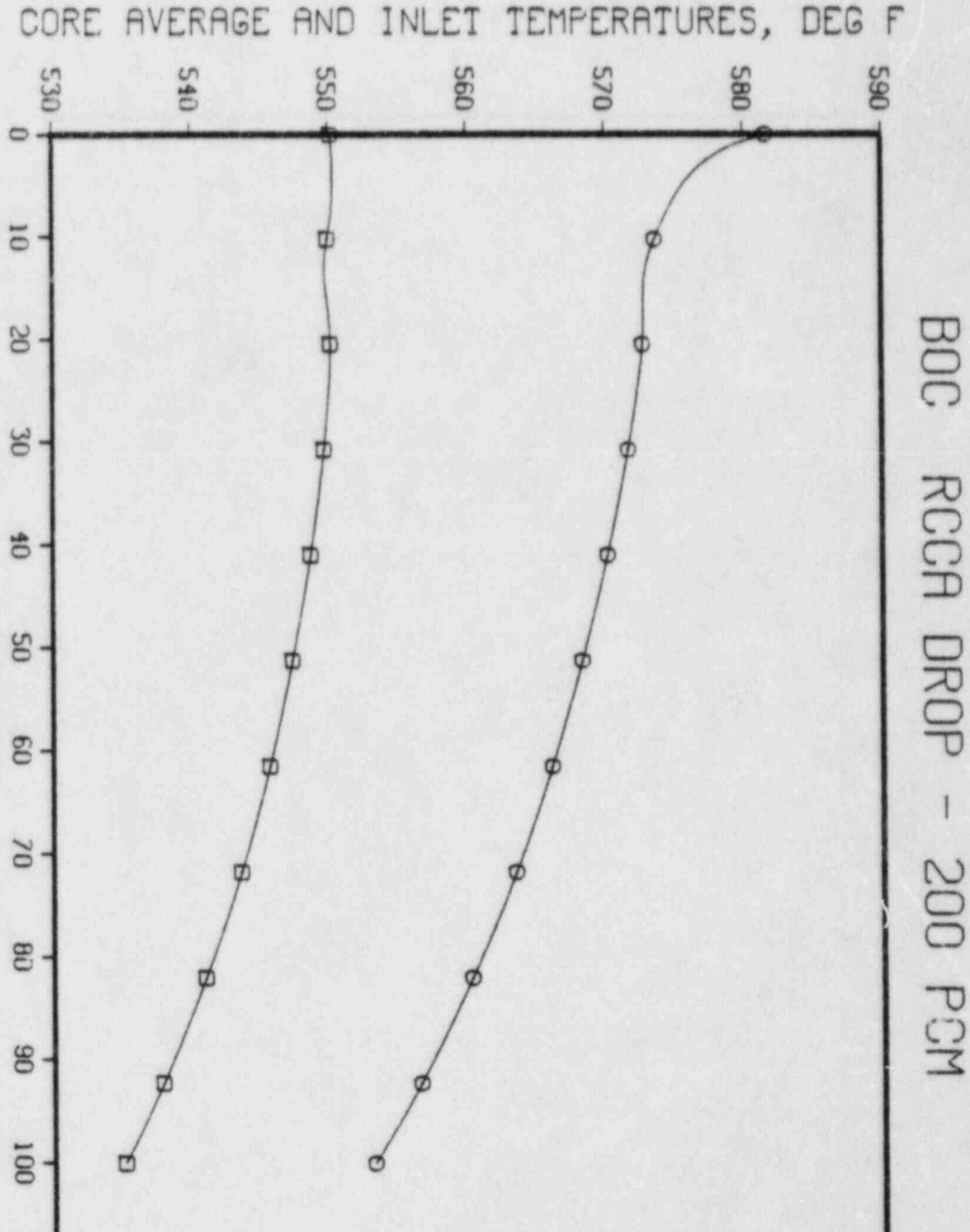

Figure 15.4.3-7 Dropped Full Length RCCA, High Worth BOC, Core Coolant Inlet (TCIØ) and Average (TCA) Temperatures

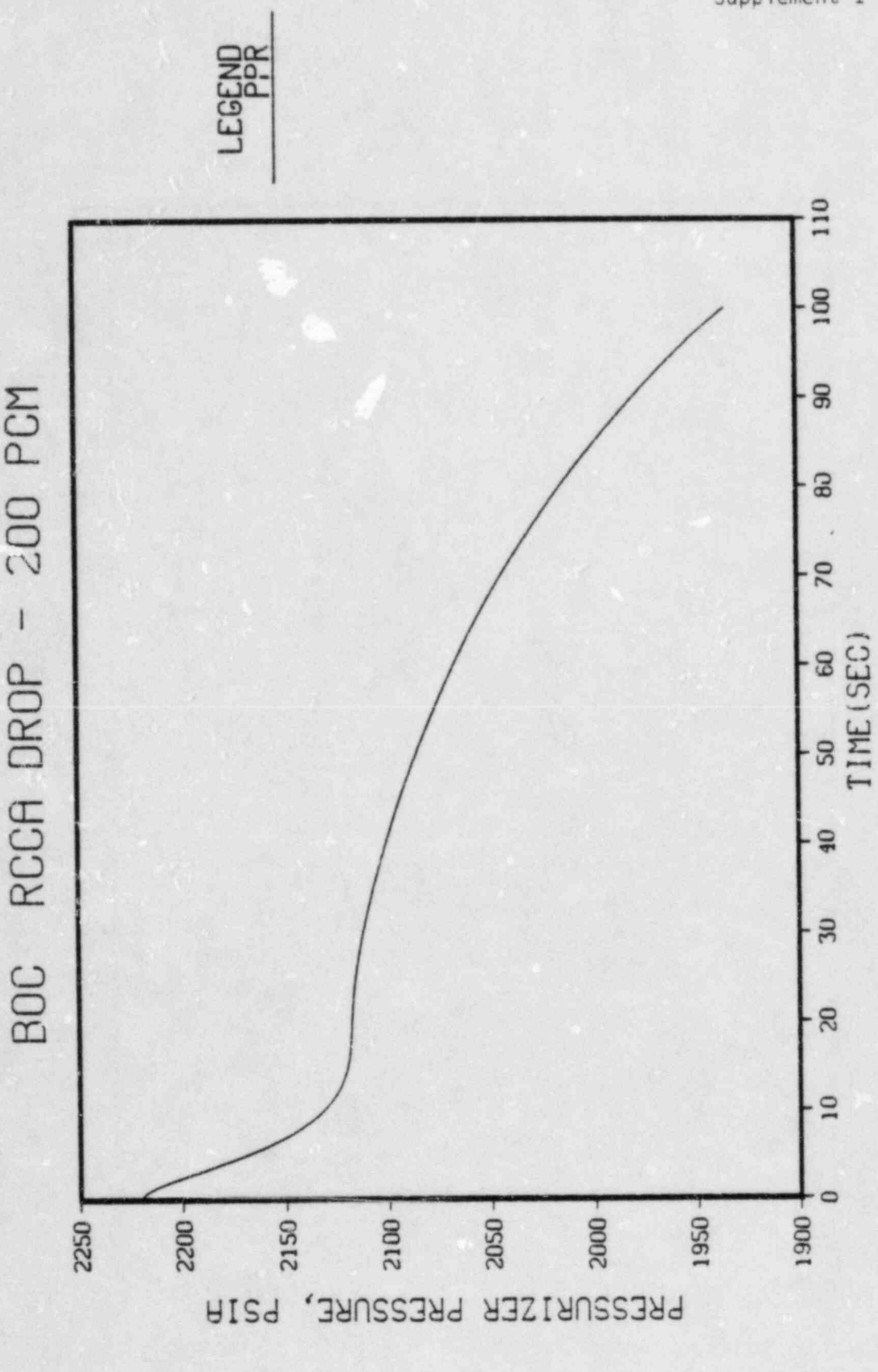

Figure 15.4.3-8 Dropped Full Length RCCA, High Worth BOC, Pressurizer Pressure

4.0 REFERENCES

1. NUREG-0800, Rev. 2, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," United States Nuclear Regulatory Commission, Washington, D.C., July 1981.

2. XN-NF-83-72, Rev. 2, Supp. 1, "H.B. Robinson Unit 2, Cycle 10 Safety Analysis Report, Revision 2, Disposition of Chapter 15 Events," Exxon Nuclear Company, Richland, WA, July 1984.

3. XN-NF-84-74, "Plant Transient Analysis for H.B. Robinson Unit 2 at 2300 MWt with Increased $F^N_{\Delta H}$," Exxon Nuclear Company, Richland, WA, August 1984.

4. XN-NF-84-72, "H.B. Robinson Unit 2 Limiting Break LOCA-ECCS Analysis with Increased Enthalpy Rise Factor," Exxon Nuclear Company, Richland, WA, July 1984.

5. XN-NF-84-68(P), "H.B. Robinson Unit 2 Radiological Assessment of Postulated Accidents," Exxon Nuclear Company, Richland, WA, June 1984.

6. Letter, T.G. Satryan (Westinghouse) to W.L. Stewart (Virginia Electric & Power Company), Subject: Unreviewed Safety Issue for Dropped Rod on Turbine Runback Plants (VPA-E3-613), dated August 24, 1983.

7. XN-CC-28, Rev. 5, "XTG: A Two Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing (PWR Version)," Exxon Nuclear Company, Richland, WA, July 1979.

8. XN-NF-74-5(P), Rev. 2, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)," Exxon Nuclear Company, Richland, WA, October 1983, with updates [Letter to J. Guttmann (USNRC) from R.A. Copeland (ENC), dated May 25, 1984].

9. XN-NF-82-21(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Richland, WA, September 1983.

XN-NF-84-74
Supplement 1

Issue Date: 9/13/84

PLANT TRANSIENT ANALYSIS FOR H. B. ROBINSON UNIT 2

AT 2300 MWt WITH INCREASED $F^N_{\Delta H}$

SUPPLEMENT 1:

ANALYSIS OF CONTROL ROD MISOPERATION EVENTS

(RCCA MISALIGNMENTS)

Distribution

F. T. Adams
G. J. Busselman
J. C. Chandler
R. A. Copeland
L. J. Federico
N. F. Fausz
T. J. Helbling
J. S. Holm
W. V. Kayser
M. R. Killgore
C. E. Leach
J. N. Morgan
P. M. O'Leary
F. B. Skogen
G. A. Sofer
I. Z. Stone
R. B. Stout
T. Tahvili
H. E. Williamson

CP&L/TJ Helbling (85)

Document Control (5)