WESTINGHOUSE CLASS 3

WCAP-11324

RTD BYPASS ELIMINATION LICENSING REPORT
FOR
BYRON UNITS 1 & 2
AND
BRAIDWOOD UNITS 1 & 2

NOVEMBER, 1986

W. R. RICE

S. M. DITOMMASO

WESTINGHOUSE ELECTRIC CORPORATION
Pittsburgh, Pennsylvania

ACKNOWLEDGEMENT

The authors wish to recognize contributions by the following individuals:

E. Manz

R. A. Holmes

E. K. Hackmann

J. C. Bass

J. C. Mesmeringer

M. Weaver

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

1.0 INTRODUCTION

Westinghouse Electric Corporation has been contracted by Commonwealth Edison Co. (CECo) to remove the existing Resistance Temperature Detector (RTD) Bypass System and replace this hot leg and cold leg temperature measurement method with fast-response RTDs installed in the reactor coolant loop piping. This report is submitted for the purpose of supporting operation of the Byron Units 1 and 2 and Braidwood Units 1 and 2 (Byron/Braidwood) with the new RTD System installed.

1.1 HISTORICAL BACKGROUND

Prior to 1968, PWR designs had been based on the assumption that the hot leg temperature was uniform across the pipe. Therefore, placement of the temperature instruments was not considered to be a factor affecting the accuracy of the measurement. The hot leg temperature was measured with direct-immersion RTDs extending a short distance into the pipe at one location. By the late 1960s, as a result of accumulated operating experience at several plants, the following problems associated with direct immersion RTDs were identified.

- o Temperature streaming conditions; the incomplete mixing of the coolant leaving regions of the reactor core at different temperatures produces significant temperature gradients within the pipe.
- o The loops required cooling and draining before the RTDs could be replaced.

The RTD bypass system was designed to resolve these problems; however, operating plant experience has now shown that operation with the RTD bypass loops has created it's own obstacles such as:

- o Plant shutdowns caused by excessive primary leakage through valves, flanges, etc., or by interruptions of bypass flow due to valve stem failure.

- Increased radiation exposure due to maintenance on the bypass line and to crud traps which increase radiation exposure throughout the loop compartments.

The proposed temperature measurement modification has been developed in response to both sets of problems encountered in the past. Specifically:

- Removal of the bypass lines eliminates the components which have been a major source of plant outages as well as Occupational Radiation Exposure (ORE).

- Three thermowell-mounted hot leg RTDs provide an average measurement (equivalent to the temperature measured by the bypass system) to account for temperature streaming.

- Use of thermowells permits RTD replacement without draining the loops.

Following is a detailed description of the effort required to perform this modification.

1.2 MECHANICAL MODIFICATIONS

The individual loop temperature signals required for input to the Reactor Control and Protection System will be obtained using RTDs installed in each reactor coolant loop.

1.2.1 <u>Hot Leg</u>

a) Byron 1 and Braidwood 1

The hot leg temperature measurement on each loop will be accomplished with three fast response narrow range RTDs mounted in thermowells. To accomplish the sampling function of the RTD bypass manifold system and eliminate the need for additional hot leg piping penetrations, the thermowells will be located within the three existing RTD bypass manifold scoops. A hole will be drilled through the end of each scoop so that water will flow in through the existing holes in the leading edge of the scoop, past the RTD, and out through the new hole (Figure 1.2-1). These three RTDs will measure the hot leg temperature which is used to calculate the reactor coolant loop differential temperature (ΔT) and average temperature (T_{avg}).

b) Byron 2 and Braidwood 2

In order to take advantage of a non-radioactive environment inside containment prior to plant operation, independent bosses and RTD thermowells (without scoops) have been installed. The RTD thermowells (Figure 1-2.3) are located 120 degress apart in the same plane, thereby providing the same averaging function as if the RTDs had been mounted in the existing scoops. These three RTDs are used in the same manner as described in paragraph 1.2.1(a).

c) This modification will not affect the single wide range RTD currently installed near the entrance of each steam generator. This RTD will continue to provide the hot leg temperature used to monitor reactor coolant temperature during startup, shutdown, and post accident conditions.

1.2.2 <u>Cold Leg</u>

a) One fast response, narrow range, RTD will be located in each cold leg at the discharge of the reactor coolant pump (as replacements for the cold leg RTDs located in the bypass manifold). Temperature streaming in the cold leg is not a concern due to the mixing action of the RCP. For this reason, only one RTD is required. This RTD will measure the cold leg temperature which is used to calculate reactor coolant loop ΔT and T_{avg}. For Byron 1 and Braidwood 1, the existing cold leg RTD bypass penetration nozzle will be modified (Figure 1.2-2) to accept the RTD thermowell. For Byron 2 and Braidwood 2, a new penetration will be made and an RTD thermowell installed as described in paragraph 1.2.2(c).

b) This modification will not affect the single wide range RTD in each cold leg currently installed at the discharge of the reactor coolant pump. This RTD will continue to provide the cold leg temperature used to monitor reactor coolant temperature during startup, shutdown, and post-accident conditions.

c) A new penetration will also be made to each cold leg to accept an additional thermowell mounted narrow range RTD, for use as an installed spare. This will give the new modification a tolerance for RTD failures equivalent to the bypass loops. A new cold leg boss will be added (Figure 1.2-3) to accept the RTD thermowell.

1.2.3 Crossover Leg

When RTD bypass elimination is implemented for each unit, the RTD bypass manifold return line will be capped at the nozzle on the crossover leg.

1.3 ELECTRICAL MODIFICATIONS

1.3.1 Function

Figure 1.3-1 shows a block diagram of the modified electronics. The hot leg RTD measurements (three per loop) will be electronically averaged in the process protection system. The averaged T_{hot} signal will then be input to the appropriate protection function. This will be accomplished by additions to the existing process control equipment.

1.3.2 Qualification

Equipment seismic and environmental qualification will be to IEEE standards 344-1975 and 323-1974, respectively, as described in WCAP-8587, Rev. 5, "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment".

1.3.3 RTD Operability Indication

Existing control board ΔT and T_{avg} indicators and alarms will provide the means of identifying RTD failures. The spare cold leg RTD provides sufficient spare capacity to accommodate a single cold leg RTD failure per loop. Failure of a hot leg RTD will require manual action to defeat the failed signal, and a manual rescaling of the electronics to average the remaining signals (see Figure 1.3-1).

a,c

Figure 1.2-1 Hot Leg RTD Scoop Modification for Fast Response RTD Installation

a,c

Figure 1.2-2 Cold Leg Pipe Nozzle Modification for Fast Response RTD Installation

a,c

Figure 1.2-3 Additional Bosses for Hot and Cold Leg Fast-Response RTD Installation

a,c

Figure 1.3-1 RTD Averaging Block Diagram, Typical for Each of 4 Channels

2.0 TESTING

There are two specific tests which have been performed to support the installation of the fast-response RTDs in the reactor coolant piping: a response time test and a hot leg temperature streaming test.

2.1 RESPONSE TIME TEST

Westinghouse has performed an RTD response time test at its Forest Hills Test Facility. This test placed a fast response RTD, manufactured by RdF Corporation, inside a scoop, within a thermowell, which modelled the actual in-plant installation. The flow conditions were adjusted to equal the high velocity reactor coolant system flows of approximately [ ]^{a,b,c}. The RTD's response time is determined based on a comparison of the RTD with thermocouples which had been previously calibrated and response time characterized. Sixty-five test runs were made at various flow rates while gathering data on 2 RTDs. The test results demonstrated a mean response time for the RTD, thermowell and scoop of less than [ ]^{a,b,c} seconds. Table 2.1-1 provides a comparison of the original RTD Bypass System response time and how it differs from the new fast response thermowell system.

2.2 STREAMING TEST

Past testing at Westinghouse PWRs has established that temperature stratification exists in the hot leg pipe with a temperature gradient from top to bottom of [ ]^{b,c,e}. A test program was implemented at an operating plant to confirm the temperature streaming magnitude and stability with measurements of the RTD bypass branch line temperatures on two adjacent reactor coolant loops. Specifically, it was intended to determine the magnitude of the differences between branch line temperatures, confirm the short-term and long-term stability of the temperature streaming patterns and evaluate the impact on the indicated temperature if only 2 of the 3 branch line temperatures are used to determine an average temperature. This plant specific data is used in conjunction with data taken from other Westinghouse designed plants to determine an appropriate temperature error for use in the safety analysis and calorimetric flow calculations. Section 3 will discuss the specifics of these uncertainty considerations.

The test data has been reduced and characterized to answer the three objectives of the test program. First, it is conservative to state that the streaming pattern [ ]^{b,c,e}. Steady state data taken at 100% power for a period of four weeks indicates that the streaming pattern [ ]^{b,c,e}. In other words, the temperature gradient [ ]^{b,c,e}. This is inferred by [ ]^{b,c,e} observed between branch lines. Since the [ ]^{b,c,e} into the RTD averaging circuit if a hot leg RTD fails and only 2 RTDs are used to obtain an average hot leg temperature. The operator can review temperatures recorded prior to the RTD failure and determine an [ ]^{b,c,e} into the "two RTD" average to obtain the "three RTD" expected reading. This significantly reduces the error introduced by a failed RTD.

The test data also supports previous calculations of streaming errors determined from tests at other Westinghouse plants. The recent data is consistent with the upper bound temperature gradients that characterize the previous data. There were no new discoveries, but the data did add a dimension previous tests did not have. The test sampled temperatures from the pipe interior while all previous tests investigated temperature gradients at the pipe surface. The pipe internal temperature data has greatly strengthened the assumptions and inferences made with previous test data.

The streaming test and response time test have both provided valuable information needed to support the design of the fast-response RTDs installed in the reactor coolant piping.

TABLE 2.1-1

RESPONSE TIME PARAMETERS FOR RCS TEMPERATURE MEASUREMENT

	RTD Bypass System	Fast Response Thermowell RTD System
	[ ]^{a,c}	[ ]^{a,c}
RTD Bypass Piping and Thermal Lag (sec)		
RTD Response Time (sec)		
RTD Filter Time Constant (sec)		
Electronics Delay (sec)		
Total Response Time (sec)	6.0 sec	7.0 sec

3.0 UNCERTAINTY CONSIDERATIONS

This new method of hot leg temperature measurement has been analyzed to determine if it will have an impact on two uncertainties included in the Safety Analysis: Calorimetric Flow Measurement Uncertainty and Hot Leg Temperature Streaming Uncertainty.

3.1 CALORIMETRIC FLOW MEASUREMENT UNCERTAINTY

Reactor coolant flow is verified with a calorimetric measurement performed after the return to power operation following a refueling shutdown. The two most important instrument parameters for the calorimetric measurement are the narrow range hot leg and cold leg coolant temperatures. The accuracy of the RTDs has, therefore, a major impact on the accuracy of the flow measurement.

The current licensed flow measurement uncertainty for Byron/Braidwood for the sum of the four loop flows including elbow taps, is about ± 2.1% flow (not including 0.1% flow for feedwater venturi fouling allowance). However, with the use of three T_{hot} RTDs (resulting from the elimination of the RTD Bypass lines) and the latest Westinghouse RTD cross-calibration procedure (resulting in lower RTD calibration uncertainties at the beginning of a fuel cycle), it is possible to reduce the RCS flow measurement uncertainty to approximately ± 1.8% flow (including the cold leg elbow taps and excluding feedwater venturi fouling). Utilizing the uncertainty calculational methodology explicitly described in WCAP-11168-R1 (Reference 1), Tables 3.1-1 through 3.1-4 were generated to provide the Byron specific instrument uncertainties, calorimetric sensitivities, and flow uncertainties. Prior to bypass elimination implementation at the Braidwood units, these values must be reviewed for applicability to Braidwood.

3.2 HOT LEG TEMPERATURE STREAMING UNCERTAINTY

The safety analyses incorporate an uncertainty to account for the difference between the actual hot leg temperature and the measured hot leg temperature caused by the incomplete mixing of coolant leaving regions of the reactor core at different temperatures. This temperature streaming uncertainty is based on an analysis of test data from other Westinghouse plants, and on calculations to evaluate the impact on temperature measurement accuracy of numerous possible temperature distributions within the hot leg pipe. The test data has shown that the circumferential temperature variation is no more than [ ]^{b,c,e}, and that the inferred temperature gradient within the pipe is limited to about [ ]^{b,c,e}. The calculations for numerous temperature distributions have shown that, even with margins applied to the observed temperature gradients, the three-point temperature measurement (scoops or thermowell RTDs) is very effective in determining the average hot leg temperature. The most recent calculations for the thermowell RTD system have established an overall streaming uncertainty of [ ]^{b,c,e} for a hot leg measurement. Of this total, [ ]^{b,c,e}. The overall temperature streaming uncertainty applied to the calorimetric flow measurement is only slightly larger than the uncertainty used in previous analyses.

The new method of measuring hot leg temperatures, with the thermowell RTDs located within the three scoops, is at least as effective as the existing RTD bypass system, [ ]^{a,c}. Although the new method measures temperature at one point within the thermowell, compared to the five sample points in a 5-inch span of the scoop measurement, the thermowell measurement point is opposite the center hole of the scoop and therefore measures the equivalent of the average scoop sample if a linear radial temperature gradient exists in the pipe. The thermowell measurement may have a small error relative to the scoop measurement if the temperature gradient over the 5-inch scoop span is nonlinear. Assuming that the maximum inferred temperature gradient of [ ]^{b,c,e} exists from the center to the end of the scoop, the difference between the thermowell and scoop measurement is limited to [ ]^{b,c,e}. Since three RTD measurements are averaged, and the nonlinearities at each scoop are random, the effect of this error on the hot leg temperature measurement is limited to [ ]^{b,c,e}. On the other hand, imbalanced scoop flows can introduce temperature measurement uncertainties of up to [ ]^{a,c}.

In all cases, the flow imbalance uncertainty will equal or exceed the [ ]^{b,c,e} sampling uncertainty for the thermowell RTDs, so the new measurement system tends to be a more accurate measurement with respect to streaming uncertainties.

Temperature streaming measurements from testing at an operating plant have been obtained. The measurements confirm the [ ]^{b,c,e}.

Over the 4-week testing period, there were only minor variations of less than [ ]^{b,c,e} in the temperature differentials between scoops, and smaller variations in the average value of the temperature differentials. [ ]^{b,c,e}.

Provisions were made in the RTD electronics for operation with only two hot leg RTDs in service. The two-RTD measurement will be biased to correct for the difference compared with the three-RTD average. Based on recent test data, the bias would be limited to between [ ]^{b,c,e}. Data comparisons show that the magnitude of this bias varied less than [ ]^{b,c,e} over the test period.

TABLE 3.1-1
FLOW CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

	FW TEMP	FW PRES	FW d/p	STM PRESS	T_H	T_c	PRZ PRESS	
# OF INST USED					3	1	1 **	
	°F	psia	% d/p	psia	°F	°F	psia	
INST SPAN =	618.	2000.	120.	1500.	100.	100.	3000.	
INST UNC. (RANDOM) =								a,c
INST UNC. (BIAS) =								
NOMINAL =	440.	990.		990.	618.4	558.4	2250.	

** Number of Hot Leg and Cold Leg RTDs used for measurement in each loop and the number of Pressurizer Pressure transmitters used overall, i.e., one per loop.

TABLE 3.1-2

FLOW CALORIMETRIC SENSITIVITIES

FEEDWATER FLOW			
F_a			a,c
TEMPERATURE	=		
MATERIAL	=		
DENSITY			
TEMPERATURE	=		
PRESSURE	=		
DELTA P	=		
FEEDWATER ENTHALPY			
TEMPERATURE	=		
PRESSURE	=		
h_s	=	1193.3 BTU/LBM	
h_f	=	419.4 BTU/LBM	
Dh(SG)	=	773.8 BTU/LBM	
STEAM ENTHALPY			
PRESSURE	=		a,c
MOISTURE	=		
HOT LEG ENTHALPY			
TEMPERATURE	=		
PRESSURE	=		
h_H	=	640.5 BTU/LBM	
h_c	=	557.7 BTU/LBM	
Dh(VESS)	=	82.8 BTU/LBM	
$Cp(T_H)$	=	1.550 BTU/LBM-°F	
COLD LEG ENTHALPY			
TEMPERATURE	=		a,c
PRESSURE	=		
$C_p(T_c)$	=	1.264 BTU/LBM-°F	
COLD LEG SPECIFIC VOLUME			
TEMPERATURE	=		a,c
PRESSURE	=		

TABLE 3.1-3

CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTIES

COMPONENT		INSTRUMENT ERROR	FLOW UNCERTAINTY
FEEDWATER FLOW			
VENTURI			
THERMAL EXPANSION COEFFICIENT			
TEMPERATURE			
MATERIAL			
DENSITY			
TEMPERATURE			
PRESSURE			
DELTA P			
FEEDWATER ENTHALPY			
TEMPERATURE			
PRESSURE			
STEAM ENTHALPY			
PRESSURE			
MOISTURE			
NET PUMP HEAT ADDITION			
HOT LEG ENTHALPY			
TEMPERATURE			
STREAMING, RANDOM			
STREAMING, SYSTEMATIC			
PRESSURE			
COLD LEG ENTHALPY			
TEMPERATURE			
PRESSURE			
COLD LEG SPECIFIC VOLUME			
TEMPERATURE			
PRESSURE			
RTD CROSS-CAL SYSTEMATIC ALLOWANCE			
BIAS VALUES			
FEEDWATER PRESSURE	DENSITY		
	ENTHALPY		
STEAM PRESSURE	ENTHALPY		
PRESSURIZER PRESSURE	ENTHALPY - HOT LEG		
	ENTHALPY - COLD LEG		
	SPECIFIC VOLUME - COLD LEG		
FLOW BIAS TOTAL VALUE			

a,c

*, **, +, ++ INDICATE SETS OF DEPENDENT PARAMETERS

SINGLE LOOP UNCERTAINTY	(WITHOUT BIAS VALUES)	
N LOOP UNCERTAINTY	(WITHOUT BIAS VALUES)	
N LOOP UNCERTAINTY	(WITH BIAS VALUES)	

a,c

TABLE 3.1-4
COLD LEG ELBOW TAP FLOW UNCERTAINTY

INSTRUMENT UNCERTAINTIES

	% d/p SPAN	% FLOW	
PMA =			a,c
PEA =			
SCA =			
SPE =			
STE =			
SD =			
RCA =			
M&TE =			
RTE =			
RD =			
ID =			
A/D =			
RDOT =			
BIAS =			
FLOW CALORIM. BIAS =			
FLOW CALORIMETRIC =			
INSTRUMENT SPAN =			

SINGLE LOOP ELBOW TAP FLOW UNC =		% FLOW	a,c
N LOOP ELBOW TAP FLOW UNC =			
N LOOP RCS FLOW UNCERTAINTY (WITHOUT BIAS VALUES) =			
N LOOP RCS FLOW UNCERTAINTY (WITH BIAS VALUES) =	1.8		

4.0 NON-LOCA SAFETY EVALUATION

4.1 RESPONSE TIME

The primary impact of the RTD bypass elimination on the FSAR Chapter 15 non-LOCA safety analyses (Reference 2) is the increased response time associated with the fast response thermowell RTD system. The secondary impact is the possible increase in instrument uncertainties. Currently, the overall response time of the Byron/Braidwood RTD bypass system assumed in the safety analyses is approximately 6.0 seconds (see Table 2.1-1). For the fast response thermowell RTD system, the overall response time will be approximately 7.0 seconds as described in Section 2.1 and as given in Table 2.1-1.

This increased RTD response time results in longer delays from the time when the fluid conditions in the RCS require an Overtemperature Delta-T or Overpower Delta-T reactor trip until a trip signal is actually generated. Therefore, those transients that rely on the above mentioned trips must be evaluated for the longer response time. The affected transients include the Uncontrolled RCCA Withdrawal at Power, the Loss of Load/Turbine Trip, and the Steamline Rupture at Power events and are discussed in Section 4.3.

4.2 RTD UNCERTAINTY

The proposed fast response thermowell RTD system will make use of RTDs, manufactured by the RdF Corporation, with a total uncertainty of [ ]a,c assumed for the analyses.

The FSAR analyses make explicit allowances for instrumentation errors for some of the reactor protection system setpoints. In addition, allowances are made for the initial average reactor coolant system (RCS) temperature, pressure and power as described in FSAR Section 15.0. These allowances are made explicitly to the initial conditions for the non-DNB events; for the DNB events, these allowances are statistically combined into the design limit DNBR value, consistent with the Improved Thermal Design Procedure (Reference 3).

The following protection and control system parameters were affected by the change from one hot leg RTD to three hot leg RTDs; the Overtemperature Delta-T (OTDT), Overpower Delta-T (OPDT), and Low RCS Flow reactor trip functions, RCS average temperature measurements used for control board indication and input to the rod control system, and the calculated value of the RCS flow uncertainty. System uncertainty calculations were performed for these parameters to determine the impact of the change in the number of hot leg RTDs. The results of these calculations indicate sufficient margin exists to account for all known instrument uncertainties.

The results of the system uncertainty calculations verify that sufficient allowance has been made in the reactor protection system setpoints to account for the increased RTD error. Therefore, the current values of the nominal setpoints noted above as defined by the Byron Technical Specifications remain valid.

4.3 NON-LOCA TRANSIENTS REANALYZED

All the events reanalyzed in this section used the LOFTRAN computer code. LOFTRAN (Reference 4) is a digital computer code, developed to simulate transient behavior in a multi-loop pressurized water reactor system. The program simulates the neutron kinetics, thermal-hydraulic conditions, pressurizer, steam generators, reactor coolant pumps, and control and protection system operation. The secondary side of each steam generator utilizes a homogeneous saturated mixture for the thermal transients.

4.3.1 Uncontrolled RCCA Bank Withdrawal at Power

The Uncontrolled RCCA bank withdrawal at power event is described in Section 15.4.2 of the FSAR. An uncontrolled RCCA bank withdrawal at power causes a positive reactivity insertion which results in an increase in the core heat flux. Since the steam generator lags behind the core power generation, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the increase in coolant temperature and power could result in DNB. For this event, the Power Range High Neutron Flux and

Overtemperature Delta-T reactor trips are assumed to provide protection against DNB. Therefore, this event was reanalyzed with increased time constants to show that the DNBR limit is met.

Methods

The assumptions used are consistent with the FSAR for the ITDP methodology in that initial power, pressure, and RCS average temperature are assumed to be at the nominal values corresponding to 10%, 60%, and 100% power. Both minimum and maximum reactivity feedback cases were reanalyzed with the increased time response value. The analysis was done using the LOFTRAN Computer Code.

Results

For both minimum and maximum reactivity insertions, at the various power levels analyzed, the DNBR limit is met. A calculated sequence of events for a fast and slow insertion rate for each reactivity feedback assumption is presented on Table 4.3-1 for full power. Figures 4.3-1 through 4.3-8 show results for a fast insertion case and a slow insertion case corresponding to the 100% power case and both reactivity assumptions. The plots of minimum DNBR versus reactivity insertion rate at all three power levels are shown as Figures 4.3-9 through 4.3-11.

Conclusions

The limit DNBR is met, and therefore, the conclusions presented in the FSAR remain valid.

4.3.2 Loss of Load/Turbine Trip

The Byron/Braidwood FSAR only explicitly analyzes the Turbine Trip Event which is presented in Section 15.2.3. This event relies on any of three reactor trips for primary protection: High Pressurizer Pressure, Low Low Steam Generator Water Level, and Overtemperature Delta-T. Thus, the increase in RTD response time may have an effect on the results of this transient.

Methods

The assumptions used are consistent with the FSAR for the ITDP methodology in that initial power, pressure, and RCS average temperature are assumed to be at the nominal values corresponding to 100% power. All four cases presented in the FSAR were reanalyzed incorporating the assumptions of the RTD Bypass Elimination. These are Beginning of Life and End of Life, with and without pressure control (pressurizer spray and PORVs). The analysis was done using the LOFTRAN Computer Code.

Results

For all combinations of reactivity feedback and pressure control, the DNBR limit is met. The results of these four cases are presented as Figures 4.3-12 through 4.3-19. A calculated sequence of events is shown in Table 4.3-2. Figures 4.3-12 and 4.3-13 show the responses for a turbine trip event with minimum reactivity feedback (Beginning of Life) assuming operability of pressurizer sprays and PORV's. The reactor is tripped by the High Pressurizer Pressure trip function. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer safety valves are actuated and primary system pressure remains below the 110% design value.

Figures 4.3-14 and 4.3-15 show the responses for a turbine trip with maximum reactivity feedback (End of Life) and pressure control. The reactor is tripped by the Overtemperature Delta-T trip function, and the DNBR never drops below the initial value. The pressurizer safety valve lift set pressure is not reached.

Figures 4.3-16 and 4.3-17 show the responses for a turbine trip with minimum reactivity feedback (BOL) and without pressure control. The reactor is tripped by the High Pressurizer Pressure trip function, and the DNBR never drops below the initial value. The pressurizer safety valves are actuated and maintain system pressure below 110% of the design value.

Figures 4.3-18 and 4.3-19 show the responses for a turbine trip with maximum reactivity feedback (End of Life) and without pressure control. The reactor is tripped by the High Pressurizer Pressure trip function, and the DNBR never drops below the initial value. The pressurizer safety valves are actuated and maintain system pressure below 110% of the design value.

Conclusions

The DNBR limit value is met in all four cases, and therefore, the conclusions presented in the FSAR remain valid. The Overpressure Protection Report is also not impacted by the RTD bypass elimination effort, and thus, the conclusions presented in that document remain unchanged.

4.3.3 Steamline Rupture at Power

The Steamline Rupture at Power transient was analyzed consistent with WCAP-9226-R1. The analysis included the increased time constants and the increased temperature uncertainties mentioned in Section 4.2. For this event the design basis as described in WCAP-9226-R1 was met.

4.3.4 Conclusion

The impact of the RTD bypass elimination for Byron and Braidwood Units 1 and 2 on the FSAR Chapter 15 non-LOCA accident analyses has been evaluated. For the events impacted, it was demonstrated that the conclusions presented in the FSAR remain valid.

4.4 LOCA Evaluation

The elimination of the RTD bypass system impacts the uncertainties associated with RCS temperature and flow measurement. The magnitude of the uncertainties are such that RCS inlet and outlet temperatures, thermal design flow rate and the steam generator performance data used in the LOCA analyses will not be affected. Past sensitivity studies have shown that the variation of the core inlet temperature (T_{in}) used in the LOCA anlayses affects the predicted core flow during the blowdown period of the transient. The amount of flow into the core is influenced by the two-phase vessel-side break flow, and the core cooling is affected by the quality of the fluid. These sensitivity studies concluded that the inlet temerature effect on peak clad temeprature is dependent on break size. As a result of these studies, the LOCA analyses are performed at a nominal value of T_{in} without consideration of small uncertainties. The RCS flow rate and steam generator secondary side temperature and pressure are also determined using the loop average temperature (T_{avg}) output. These nominal values used as inputs to the analyses are not affected due to the RTD bypass elimination. It is concluded that the elimination of the RTD bypass piping will not affect the LOCA analyses input and hence, the results of the analyses remain unaffected. Therefore, the plant design changes due to the RTD bypass elimination are acceptable from a LOCA analysis standpoint without requiring any reanalysis.

4.5 INSTRUMENTATION AND CONTROL (I&C) SAFETY EVALUATION

The RTD Bypass Elimination modification for Byron/Braidwood does not functionally change the $\Delta T/T_{avg}$ protection channels. The implementation of the fast response RTDs in the reactor coolant piping will change the inputs into the $\Delta T/T_{avg}$ Protection Sets I, II, III, and IV as follows:

1. The Narrow Range (NR) cold leg RTD in the cold leg manifold will be replaced with a fast response NR RTD well mounted in the RCP pump discharge pipe. The signal from this fast response NR RTD will perform the same function as the existing RTD T_{cold} signal. A second narrow range RTD will be installed as a spare.

2. The NR hot leg RTD in the bypass manifold will be replaced with 3 fast response NR RTDs well mounted in hot leg scoops that are electronically averaged in the process protection system. The signal from this average T_{hot} circuit obtained from these 3 NR T_{hot} RTDs will perform the same function as the existing RTD T_{hot} signal.

3. Identification of failed signals will be by the same means as before the modifications, i.e., existing control board alarms and indications.

4. Signal process and the added circuitry to the Protection Set racks will be accomplished by additions to the process control (Westinghouse Model 7300) racks using 7300 technology. When one T_{hot} signal is removed from the averaging process, the electronics will allow a bias to be manually added to a 2-RTD average T_{hot} (as opposed to a 3-RTD average T_{hot}) in order to obtain a value comparable with the 3-RTD average T_{hot} prior to the failed RTD. In the event of a cold leg RTD failure, the spare cold leg RTD will be manually connected to the 7300 circuitry in place of the failed RTD.

Other than the above changes, the instrumentation and control will remain the same and unchanged from what has previously been utilized. For example, two out of four voting logic continues to be utilized for protection functions, with the model 7300 process control bistables continuing to operate on a "de-energize to actuate" principle. Non-safety related control signals continue to be derived from protection channels.

The above principles of the modification have been reviewed to evaluate conformance to the requirements of IEEE-279-1971 criteria and associated 10CFR 50 General Design Criteria (GDC), Regulatory Guides, and other applicable industry standards. IEEE 279-1971 requires documentation of a design basis. Following is a discussion of design basis requirements in conformance to pertinent I&C criteria:

a. Single failure criterion continues to be satisfied by this change because the independence of redundant protection sets is maintained.

b. Quality components and modules being added is consistent with use in a Nuclear Generating Station Protection System. For the Westinghouse Quality Assurance program, refer to Chapter 17 of the FSAR.

c. Equipment seismic and environmental qualification will be to IEEE standards 344-1975 and 323-1974, respectively, as described in WCAP 8587, Rev. 5 "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment".

d. The changes will continue to maintain the capability of the protection system to initiate a reactor trip during and following natural phenomena credible to the plant site to the same extent as the existing system.

e. Channel independence and electrical separation is maintained because the Protection Set circuit assignments continue to be Loop 1 circuits input to Protection Set I; Loop 2, to Protection Set II; Loop 3, to Protection Set III; and Loop 4 to Protection Set IV, with appropriate observance of field wiring interface criteria to assure the independence. Output circuits are the same as before except that there will be one T_{cold} and 3 T_{hot} outputs to the computer sent through Class 1E isolators in each Protection Set.

f. The IEEE 279-1971 Section 4.7 and GDC 24 requirements concerning Control and Protection System interaction are satisfied because, even though control signals are derived from Protection Sets, the 2/4 voting coincidence logic of the Protection Sets is maintained.

Where a single random failure can cause a control system action that results in a condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining three redundant protection channels will be capable of providing the protective action even when degraded by a second random failure. This is because even though 1/4 channels failed without partially tripping, only 2 of the remaining 3 channels are necessary for a plant trip.

On the basis of the foregoing evaluation, it is concluded that these I&C modifications required for RTD bypass removal for the Byron/Braidwood units will meet IEEE 279-1971, applicable GDCs, and industry standards and regulatory guides.

4.6 MECHANICAL SAFETY EVALUATION

The presently installed RTD bypass system is to be replaced with fast acting narrow range RTD thermowells. This change requires modifications to the hot leg scoops, the crossover leg bypass return nozzle, the cold leg piping and the cold leg bypass manifold connection. All welding and NDE will be performed per ASME Code Section XI requirements. Each of these modifications is evaluated below.

<u>Byron 1 and Braidwood 1</u>

The original three scoops in each hot leg, which feed the bypass manifold, and the bypass manifold connection must be removed and the scoops modified to accept three fast response RTD thermowells. [ ]^{a,c} to provide the proper flow path. A thermowell design will be used such that the tip of the thermowell will be positioned to provide an average temperature reading. The thermowell will be fabricated in accordance with Section III Class 1 of the ASME Code. The installation of the thermowell into the scoop will be performed using GTAW for the root pass and finished out with either GTAW or SMAW. The welding will be examined by penetrant test (PT) per the ASME Code Section XI. Prior to welding, the surface of the scoop onto which welding will be performed will be examined as required by Section XI.

The cold leg RTD bypass line must also be removed. The nozzle must then be modified to accept the fast response RTD thermowell. Additionally, a spare fast response thermowell will be added to the cold leg in the length between the reactor coolant pump discharge and the loop isolation valve. This necessitates the creation of a new penetration into the piping. The boss for the new connection will be root welded by GTAW. Finish welding can be either GTAW or SMAW. Weld inspection by PT will be performed as required by Section XI. The thermowells will extend approximately [ ]^{a,c} inches into the flow stream. This depth has been justified based on [ ]^{a,c} analysis.

The root weld joining the thermowells to the modified nozzles or bosses will be deposited with GTAW and the remainder of the weld may be deposited with GTAW or SMAW. Penetrant testing will be performed in accordance with the ASME Code Section XI. The thermowells and installation bosses will be fabricated in accordance with the ASME Section III (Class 1). These two thermowells will be installed in the upper half of the piping.

Byron 2 and Braidwood 2

Three hot leg fast response RTDs will be installed into new penetrations between the reactor vessel and the loop isolation valves. This is done in order to facilitate initially operating the units with the bypass system before implementing the fast response RTDs. The design will be such that the thermowell will extend [ ]^{a,c} inches into the flow stream from the ID of the pipe. The installation boss for the new penetrations and the thermowell will be root welded by GTAW. Finish welding can be either GTAW or SMAW. Weld inspection by PT will be performed per Section XI. The installation bosses and thermowells are fabricated in accordance with Section III Class 1 of the ASME Code.

In the cold leg between the reactor coolant pump discharge and the loop isolation valves two fast response RTDs will be installed into new penetrations. The design will be such that the tip of the thermowells will extend approximately [ ]^{a,c} inches into the flow stream.

The installation bosses and thermowells will be fabricated in accordance with Section III Class 1 of the ASME Code. Root welding will be performed by GTAW. Finish welding can be either GTAW or SMAW. Weld inspection by PT will be performed per the ASME Code Section XI.

Upon removal of the RTD bypass piping, the hot leg scoops and the cold leg nozzles will be capped. The caps will be fabricated in accordance with Section III Class 1 of the ASME Code. The root weld joining the caps to either the scoops or cold leg nozzle will be done by GTAW. Finish welding will be done by either GTAW or SMAW. The welds will be inspected by PT per the ASME Code Section XI.

With the three thermowells in the hot leg and the two thermowells in the cold leg, a total of 20 thermowells will be utilized at each of the four-loop Byron/Braidwood units and they will perform the same function as the original bypass T_{hot} and T_{cold} signals.

The cross-over leg bypass return piping connection must be removed and the nozzles capped. The cap design, including materials, will meet the pressure boundary criteria and ASME Section III (Class 1). The cap will be root welded to the nozzles by GTAW and fill welded by either GTAW or SMAW. Non-destructive examinations (PT and radiographs) will be performed per ASME Section XI. Machining of the bypass return nozzle, as well as any machining performed during modification of the penetrations in the hot and cold legs, shall be performed such as to minimize debris escaping into the reactor coolant system.

In accordance with [ ]^{a,c} of the ASME Code, a hydrostatic test of new pressure boundary welds is required when the connection to the pressure boundary is [ ]^{a,c} in diameter. Since the cap for the crossover leg bypass return pipe is [ ]^{a,c} inches and the cold leg RTD connections are [ ]^{a,c} inches, a system hydrostatic test is required after bypass elimination. Paragraph [ ]^{a,c} defines this test pressure to be [ ]^{a,c} times the normal operating pressure at a temperature of [ ].^{a,c}

The integrity of the reactor coolant piping as a pressure boundary component, is maintained by adhering to the applicable ASME Code sections and Nuclear Regulatory Commission General Design Criteria. The pressure retaining capability and fracture prevention characteristics of the piping is not compromised by these modifications.

4.7 TECHNICAL SPECIFICATION EVALUATION

As a result of the calculations summarized in Section 3.1 on the impact of the fast response RTDs on flow measurement uncertainty, the Technical Specification modifications identified in Appendix A are necessary to achieve proper reactor trip and engineered safety features system operability.

TABLE 4.3-1
(page 1 of 2)

TIME SEQUENCE OF EVENTS FOR A
RCCA BANK WITHDRAWAL AT POWER

ACCIDENT	EVENT	TIME (SECS)
Case A	Initiation of uncontrolled RCCA withdrawal at a fast reactivity insertion rate (75 pcm/sec) with minimum reactivity feedback at full power	0.0
	Power range high neutron flux reactor trip signal generated	1.6
	Rods begin to drop	2.1
	Minimum DNBR occurs	3.2
	Peak water level in the pressurizer occurs	4.9
Case B	Initiation of uncontrolled RCCA withdrawal at a low reactivity insertion rate (3 pcm/sec) with minimum reactivity feedback at full power	0.0
	Overtemperature Delta-T reactor trip signal initiated	31.5
	Rods begin to drop	33.5
	Minimum DNBR occurs	33.7
	Peak water level in the pressurizer occurs	35.7

TABLE 4.3-1
(page 2 of 2)

TIME SEQUENCE OF EVENTS FOR A
RCCA BANK WITHDRAWAL AT POWER

ACCIDENT	EVENT	TIME (SECS)
Case C	Initiation of uncontrolled RCCA withdrawal at a fast reactivity insertion rate (75 pcm/sec) with maximum reactivity feedback at full power	0.0
	Power range high neutron flux reactor trip signal initiated	5.3
	Rods begin to drop	5.8
	Minimum DNBR occurs	6.4
	Peak water level in the pressurizer occurs	8.6
Case D	Initiation of uncontrolled RCCA withdrawal at a low reactivity insertion rate (3 pcm/sec) with maximum reactivity feedback at full power	0.0
	Overtemperature Delta-T reactor trip signal initiated	203
	Rods begin to drop	205
	Minimum DNBR occurs	203
	Peak water level in the pressurizer occurs	207

TABLE 4.3-2
(page 1 of 2)

TIME SEQUENCE OF EVENTS FOR A TURBINE TRIP

ACCIDENT	EVENT	TIME (SECS)
Case A	Initiation of turbine trip, loss of main feedwater flow, minimum reactivity feedback with pressure control	0.0
	High pressurizer pressure reactor trip signal generated	6.9
	Initiation of steam release from S/G safety valves	8.5
	Rods begin to drop	8.9
	Peak pressurizer pressure occurs	10.5
	Minimum DNBR occurs	(1)
Case B	Initiation of turbine trip, loss of main feedwater flow, maximum reactivity feedback with pressure control	0.0
	Initiation of steam release from S/G safety valves	8.1
	Overtemperature Delta-T reactor trip signal generated	8.5
	Peak pressurizer pressure occurs	9.4
	Rods begin to drop	10.5
	Minimum DNBR occurs	(1)

(1) DNBR does not decrease below its initial value.

TABLE 4.3-2
(page 2 of 2)

TIME SEQUENCE OF EVENTS FOR A TURBINE TRIP

ACCIDENT	EVENT	TIME (SECS)
Case C	Initiation of turbine trip, loss of main feedwater flow, minimum reactivity feedback without pressure control	0.0
	High pressurizer pressure reactor trip signal generated	4.1
	Rods begin to drop	6.1
	Initiation of steam release from S/G safety valves	8.0
	Peak pressurizer pressure occurs	8.0
	Minimum DNBR occurs	(1)
Case D	Initiation of turbine trip, loss of main feedwater flow, maximum reactivity feedback without pressure control	0.0
	High pressurizer pressure reactor trip signal generated	4.1
	Rods begin to drop	6.1
	Peak pressurizer pressure occurs	7.5
	Initiation of steam release from S/G safety valves	8.0
	Minimum DNBR occurs	(1)

(1) DNBR does not decrease below its initial value.

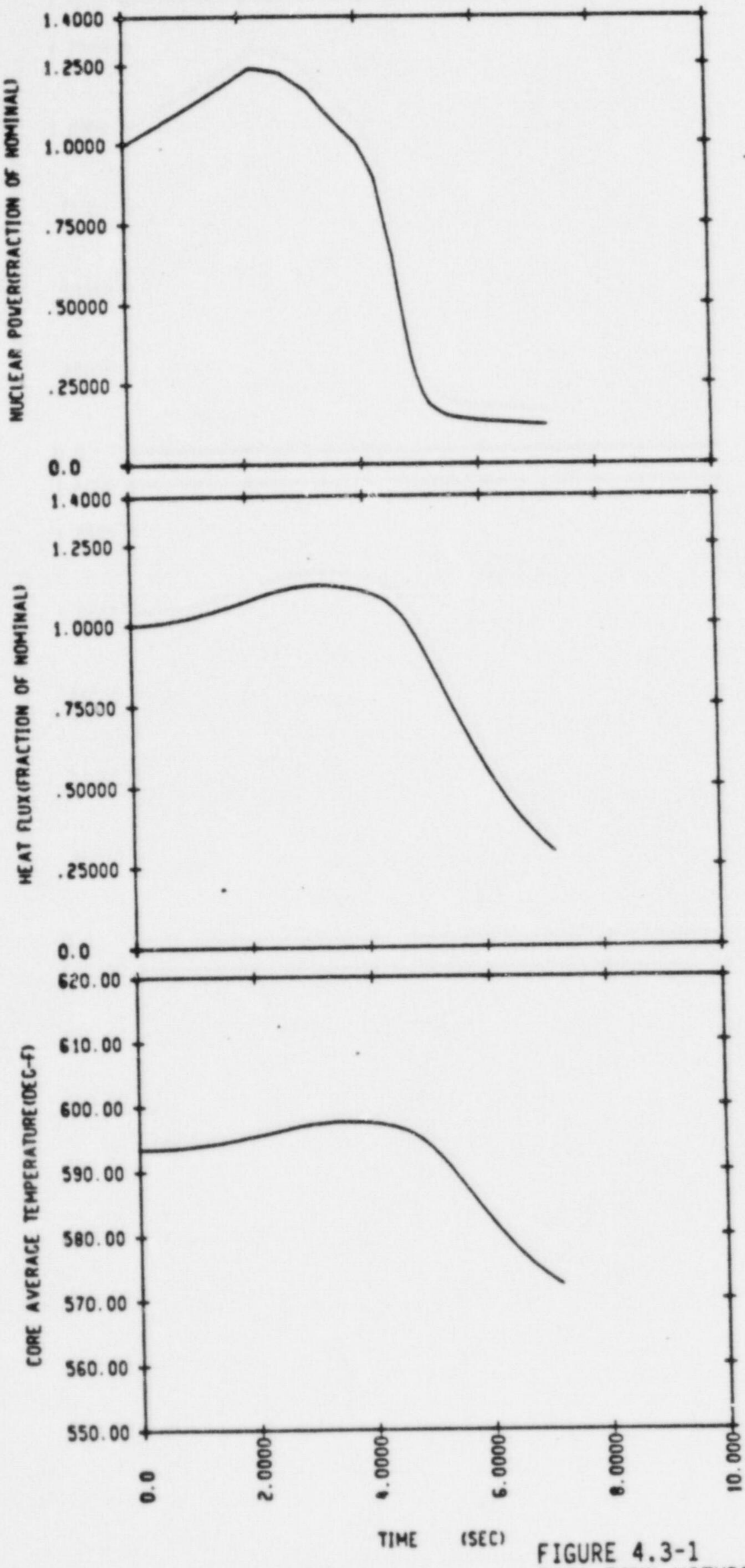

FIGURE 4.3-1
RCCA BANK WITHDRAWAL AT FULL
POWER WITH MINIMUM REACTIVITY
FEEDBACK (75 PCM/SEC RATE)

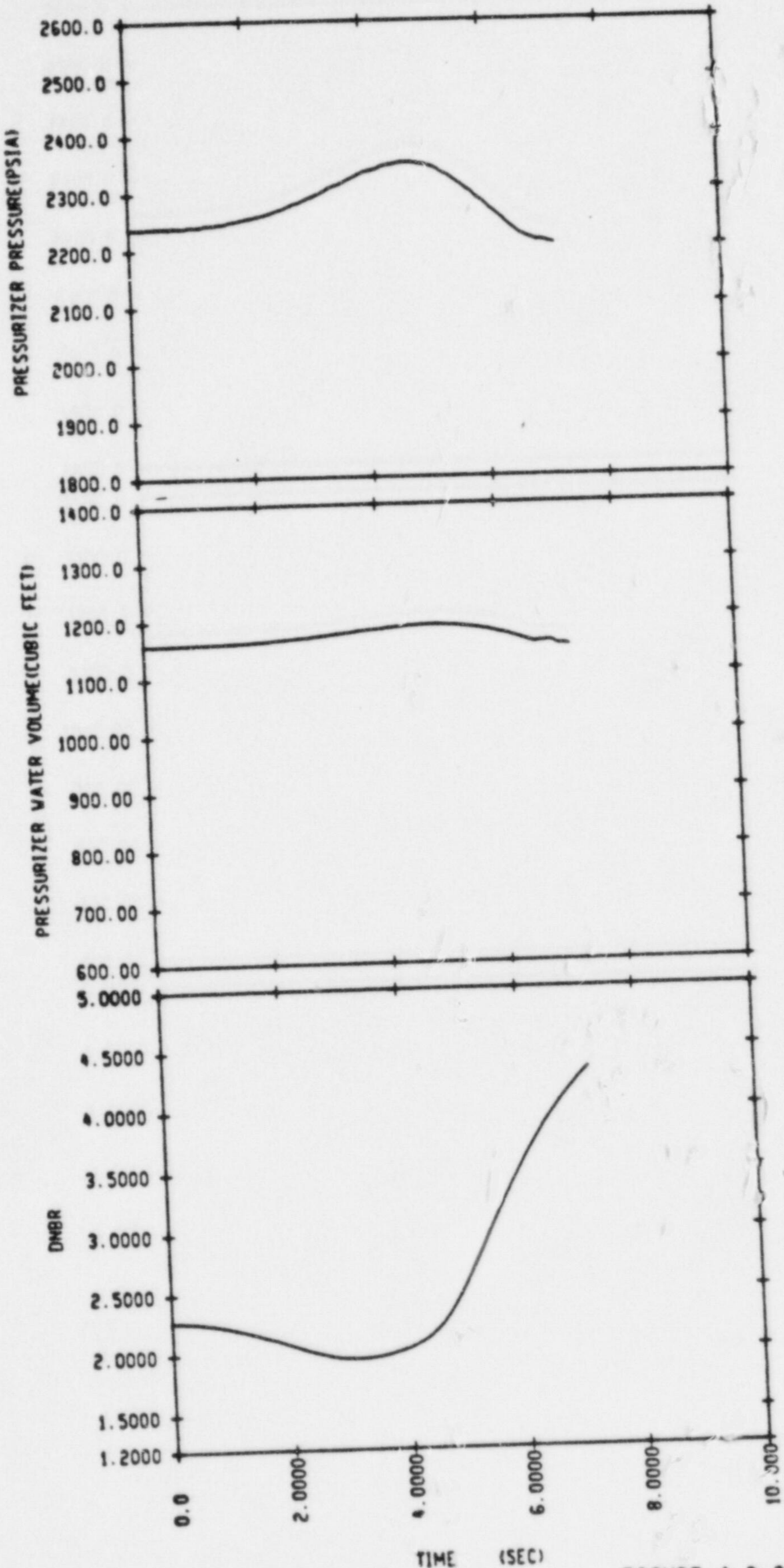

FIGURE 4.3-2
RCCA BANK WITHDRAWAL AT FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (75 PCM/SEC RATE)

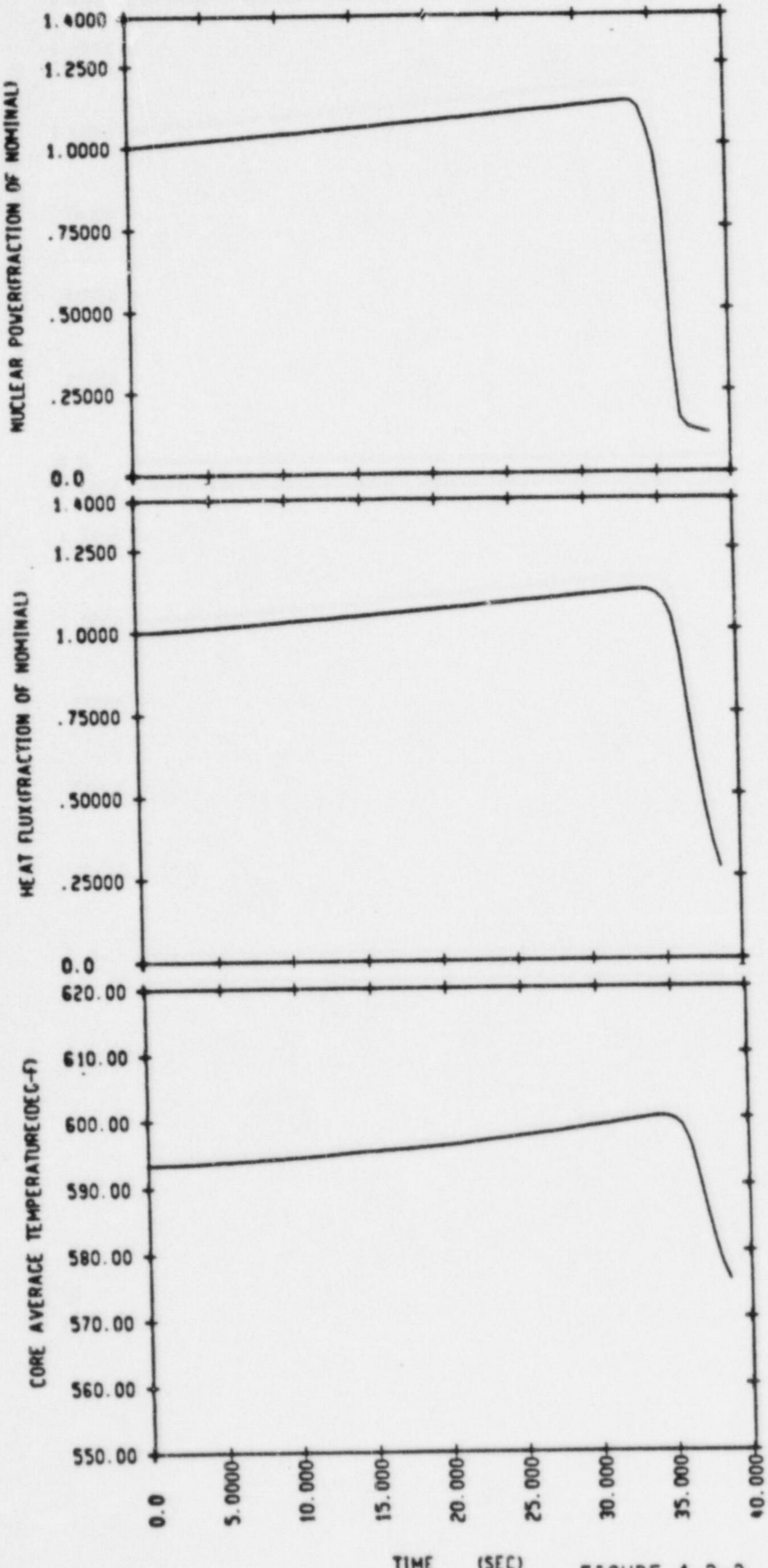

FIGURE 4.3-3
RCCA BANK WITHDRAWAL AT FULL
POWER WITH MINIMUM REACTIVITY
FEEDBACK (3 PCM/SEC RATE)

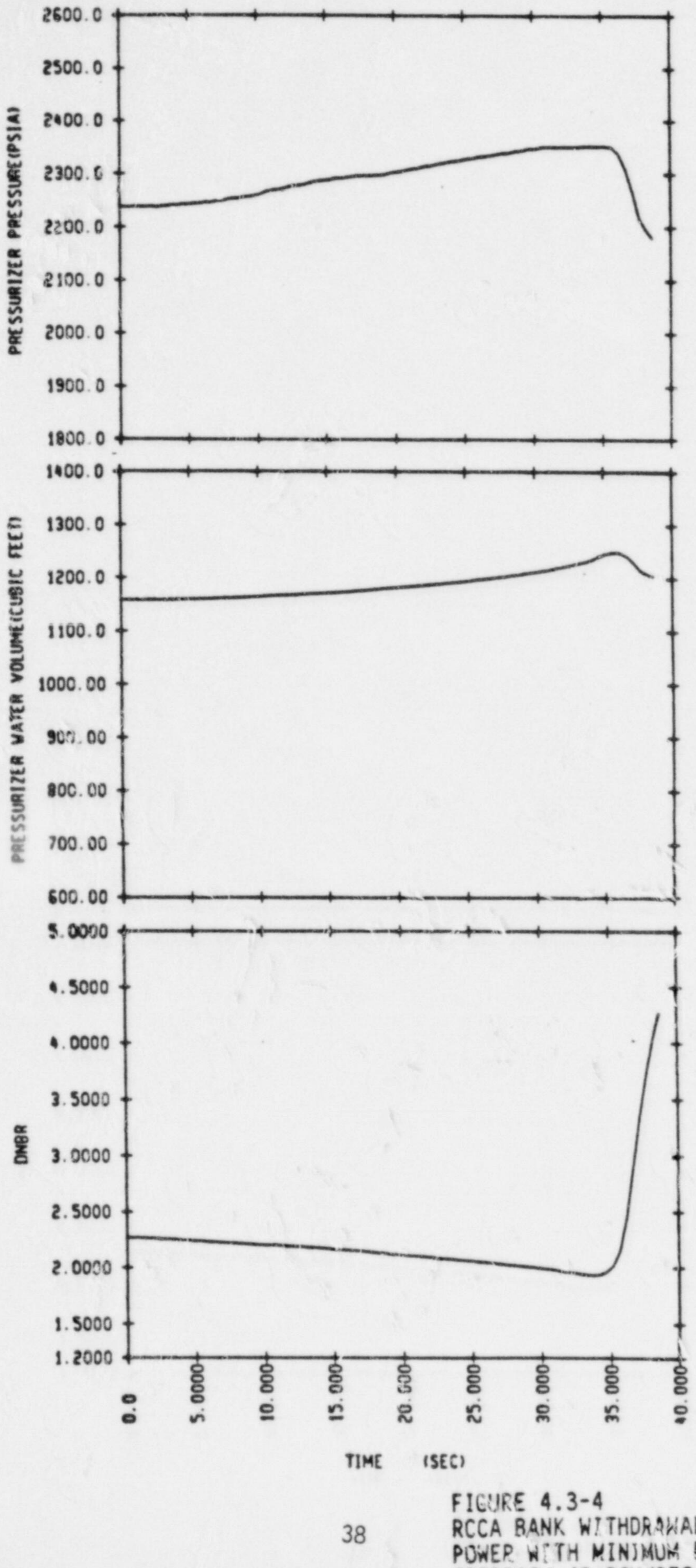

FIGURE 4.3-4
RCCA BANK WITHDRAWAL AT FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (3 PCM/SEC RATE)

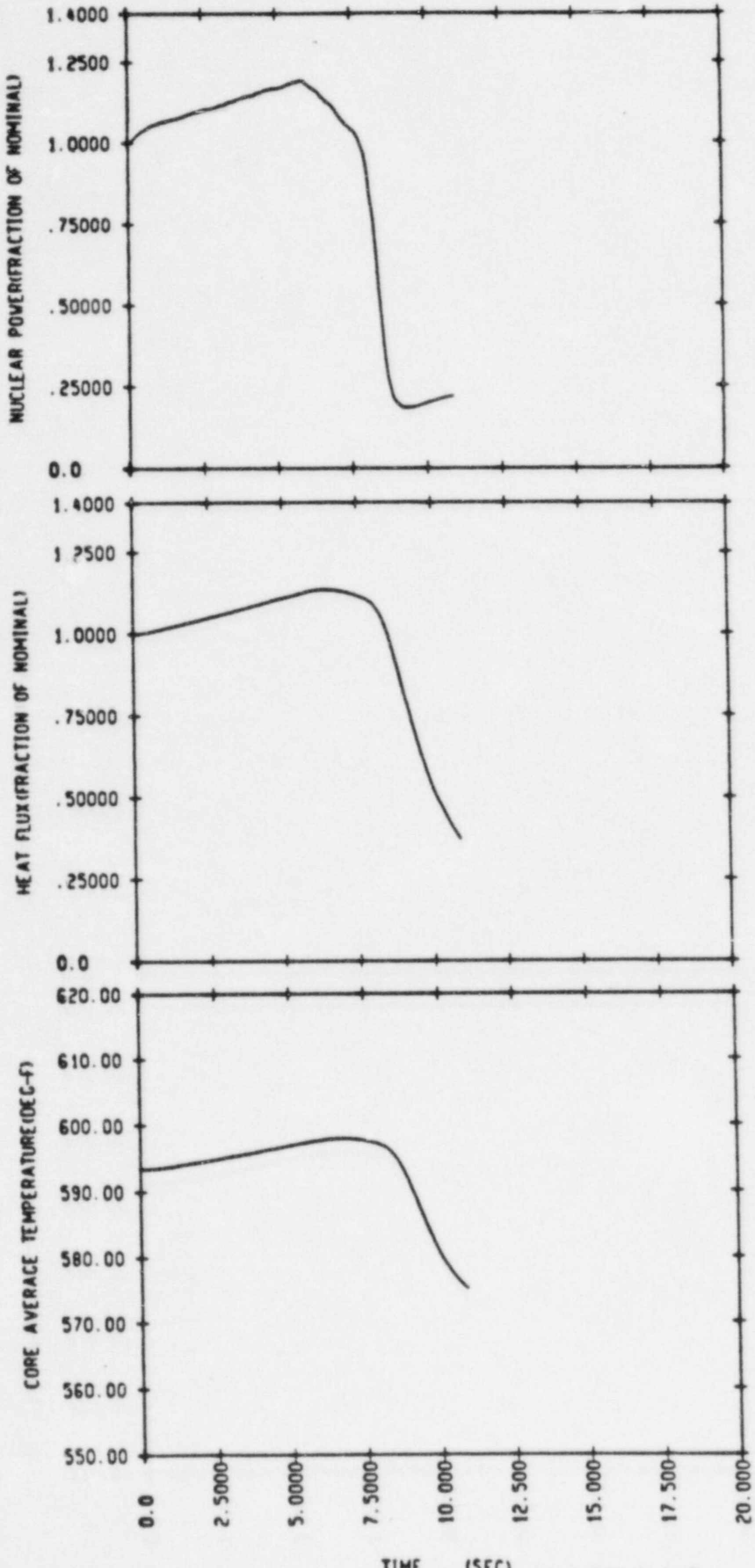

FIGURE 4.3-5
RCCA BANK WITHDRAWAL AT FULL
POWER WITH MAXIMUM REACTIVITY
FEEDBACK (75 PCM/SEC RATE)

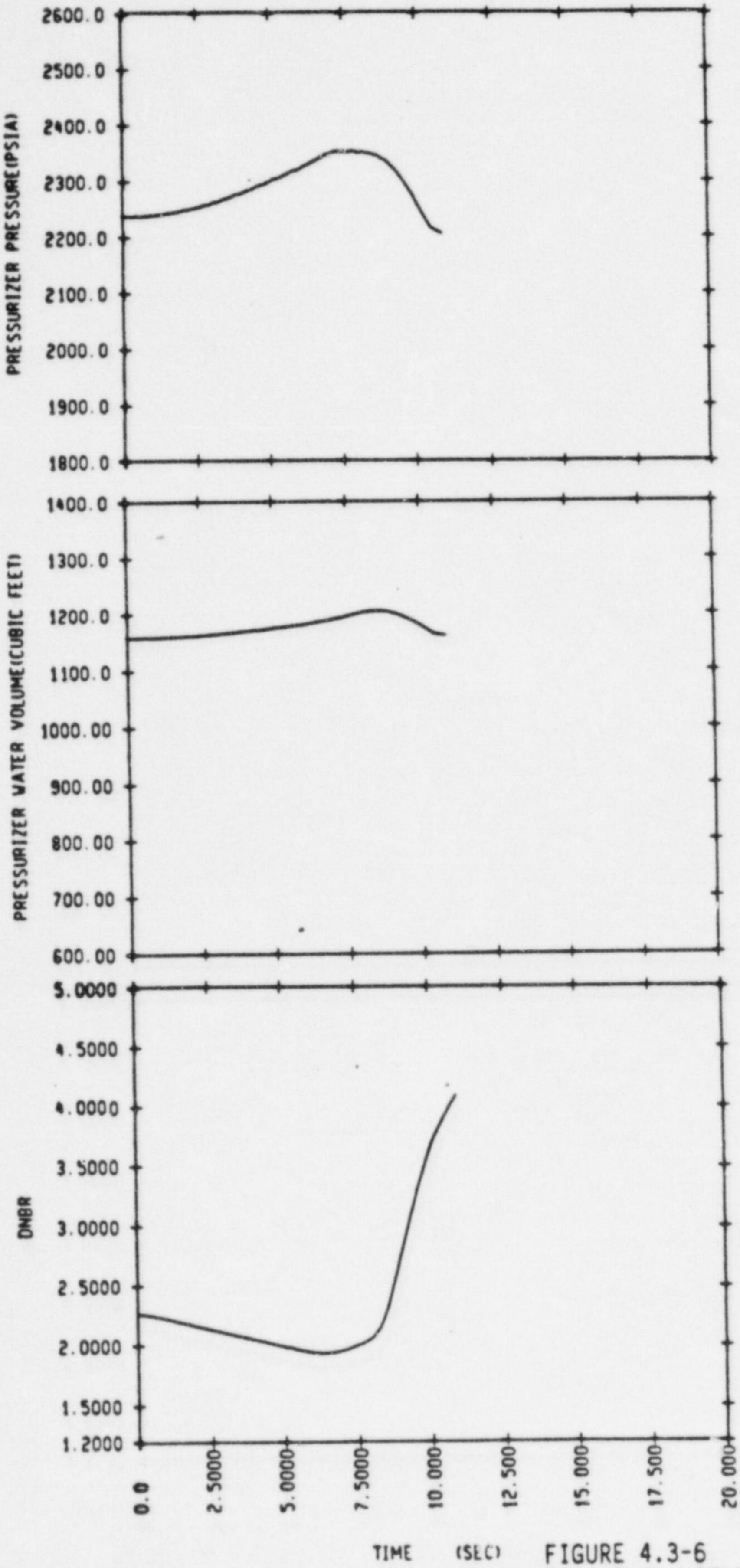

FIGURE 4.3-6
RCCA BANK WITHDRAWAL AT FULL POWER WITH MAXIMUM REACTIVITY FEEDBACK (75 PCM/SEC RATE)

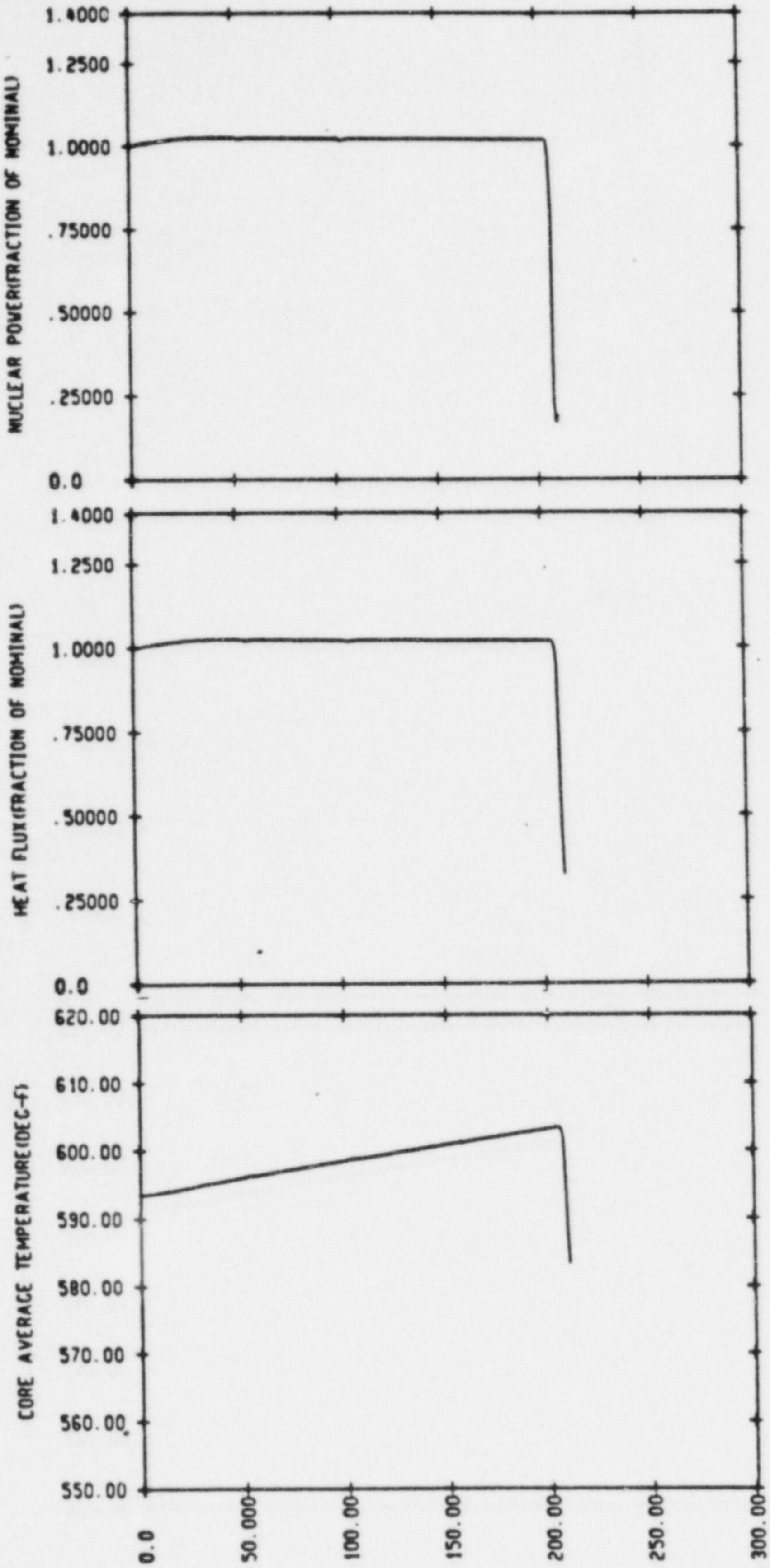

FIGURE 4.3-7
RCCA BANK WITHDRAWAL AT FULL
POWER WITH MAXIMUM REACTIVITY
FEEDBACK (3 PCM/SEC RATE)

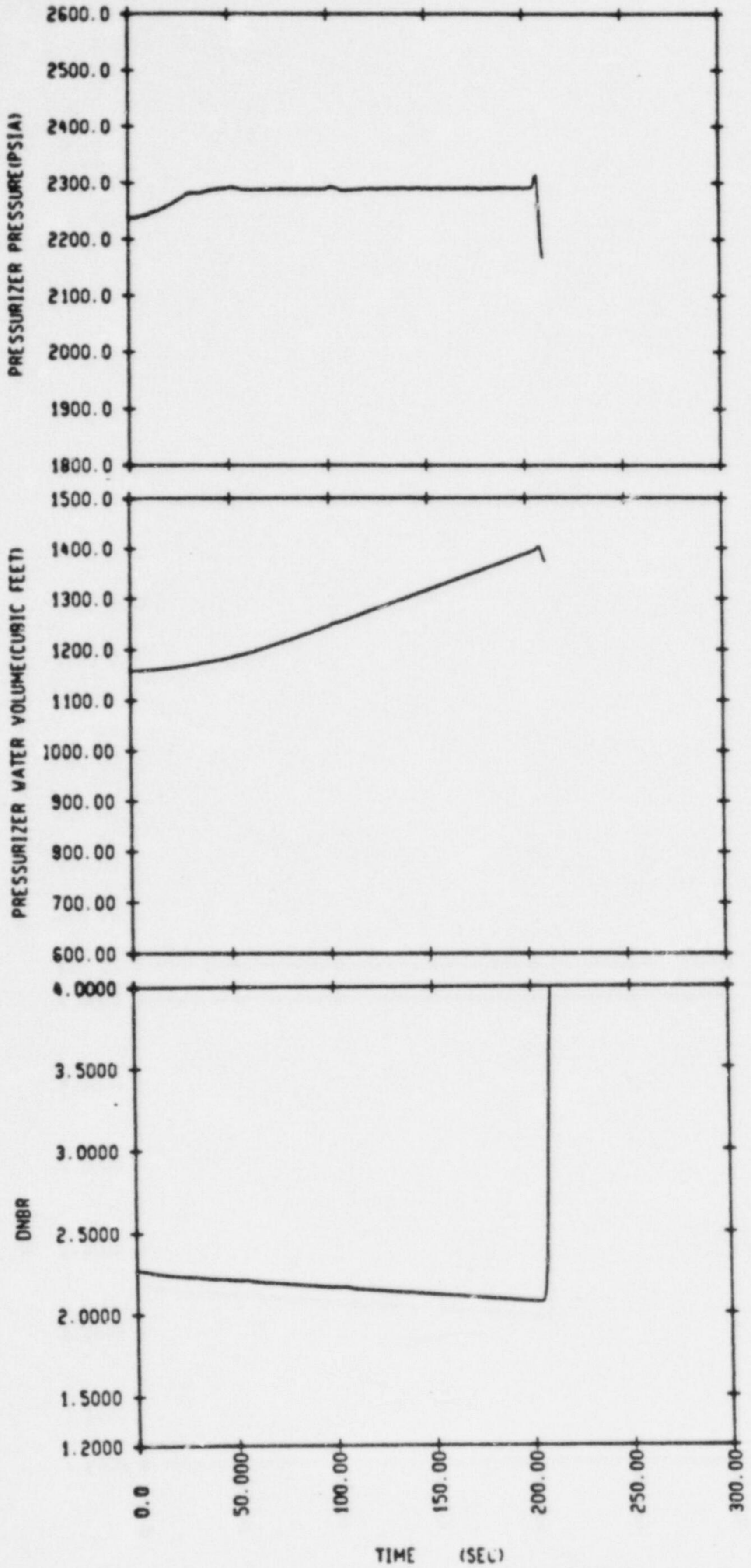

FIGURE 4.3-8
RCCA BANK WITHDRAWAL AT FULL
POWER WITH MAXIMUM REACTIVITY
FEEDBACK (3 PCM/SEC RATE)

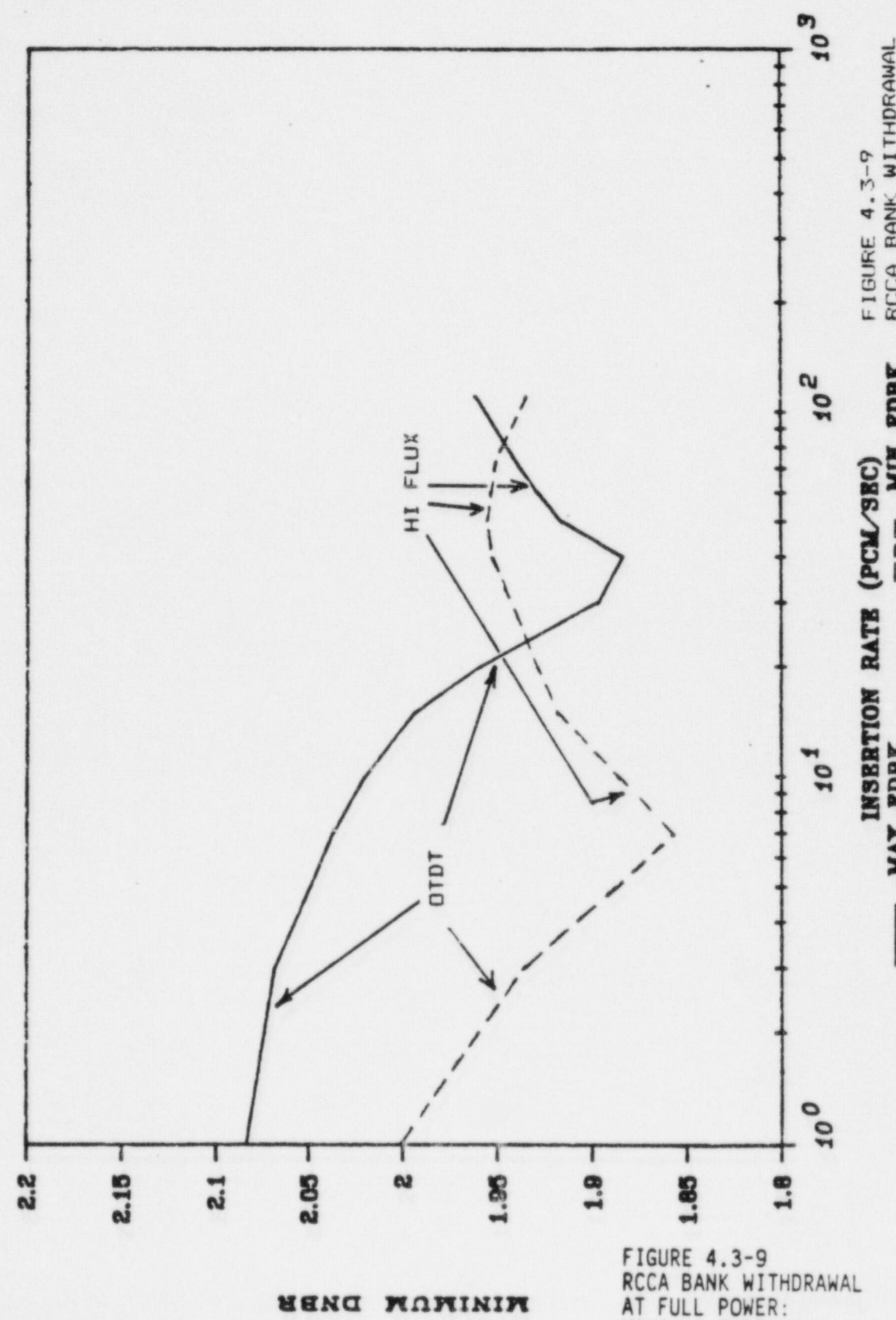

FIGURE 4.3-9
RCCA BANK WITHDRAWAL
AT FULL POWER:
MINIMUM DNBR VS
REACTIVITY INSERTION RATE

FIGURE 4.3-9
RCCA BANK WITHDRAWAL
AT FULL POWER:
MINIMUM DNBR VS
REACTIVITY INSERTION RATE

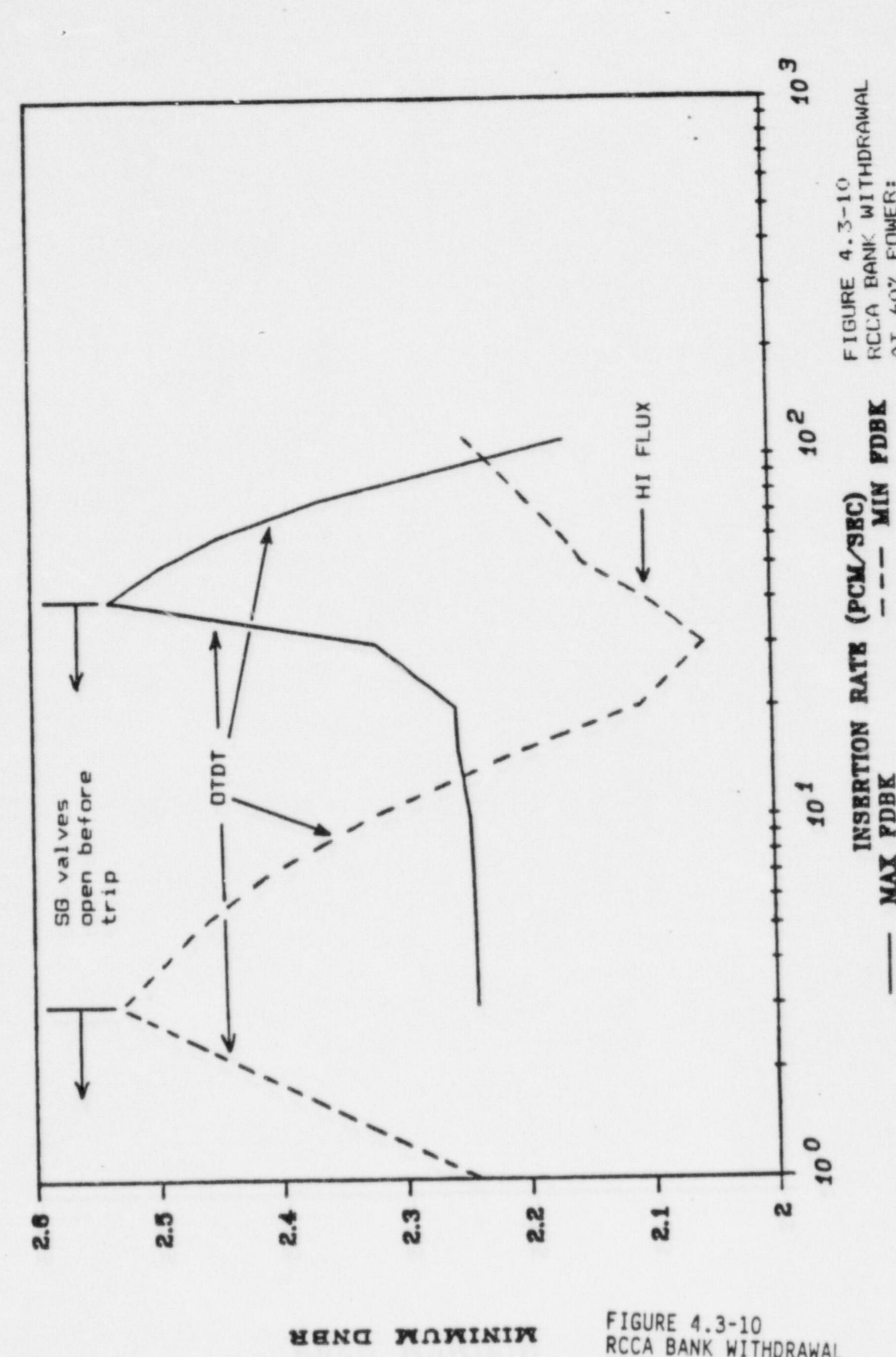

FIGURE 4.3-10
RCCA BANK WITHDRAWAL
AT 60% POWER:
MINIMUM DNBR VS
REACTIVITY INSERTION RATE

FIGURE 4.3-10
RCCA BANK WITHDRAWAL
AT 60% POWER:
MINIMUM DNBR VS
REACTIVITY INSERTION RATE

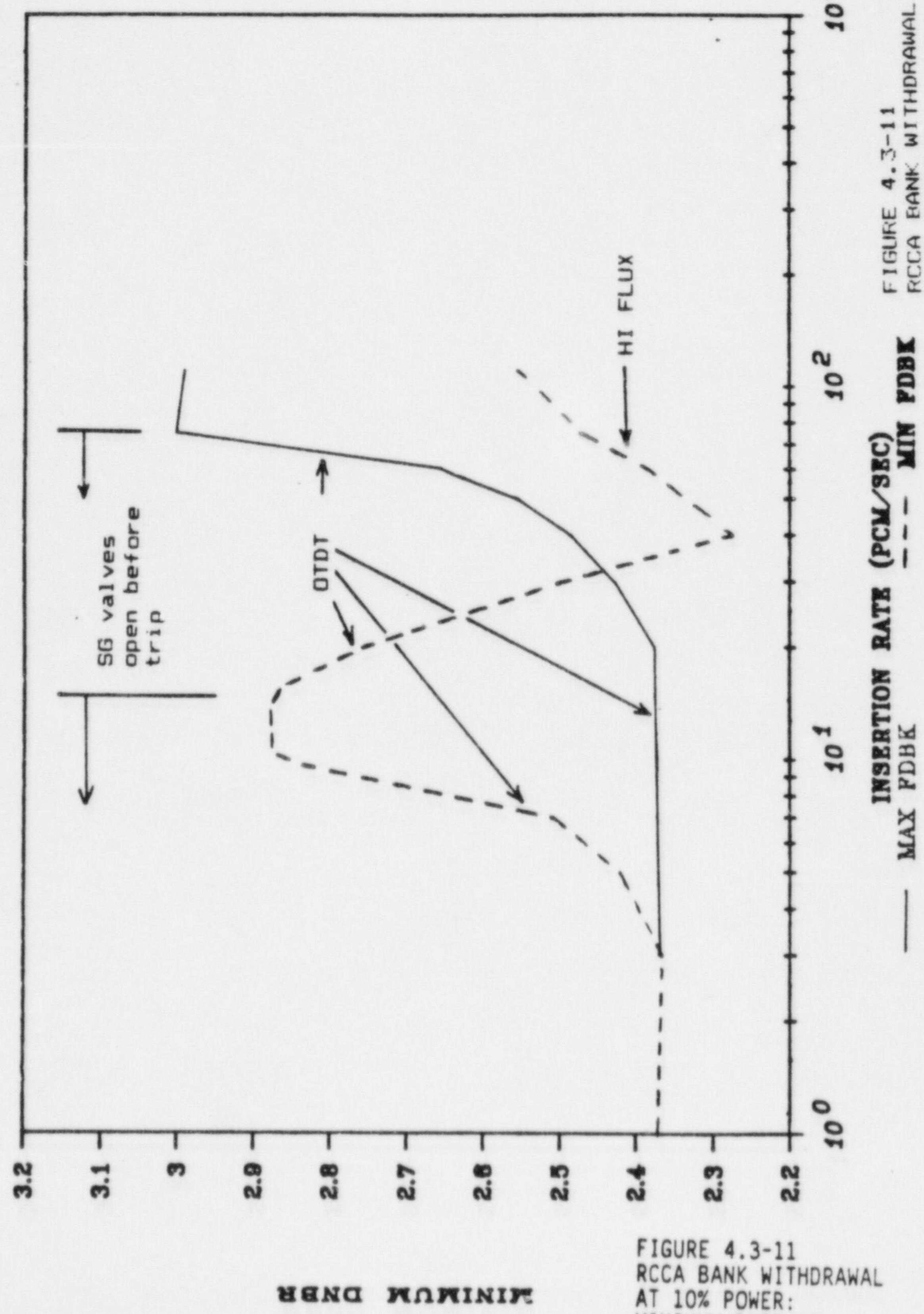

FIGURE 4.3-11
RCCA BANK WITHDRAWAL
AT 10% POWER:
MINIMUM DNBR VS
REACTIVITY INSERTION RATE

FIGURE 4.3-11
RCCA BANK WITHDRAWAL
AT 10% POWER:
MINIMUM DNBR VS
REACTIVITY INSERTION RATE

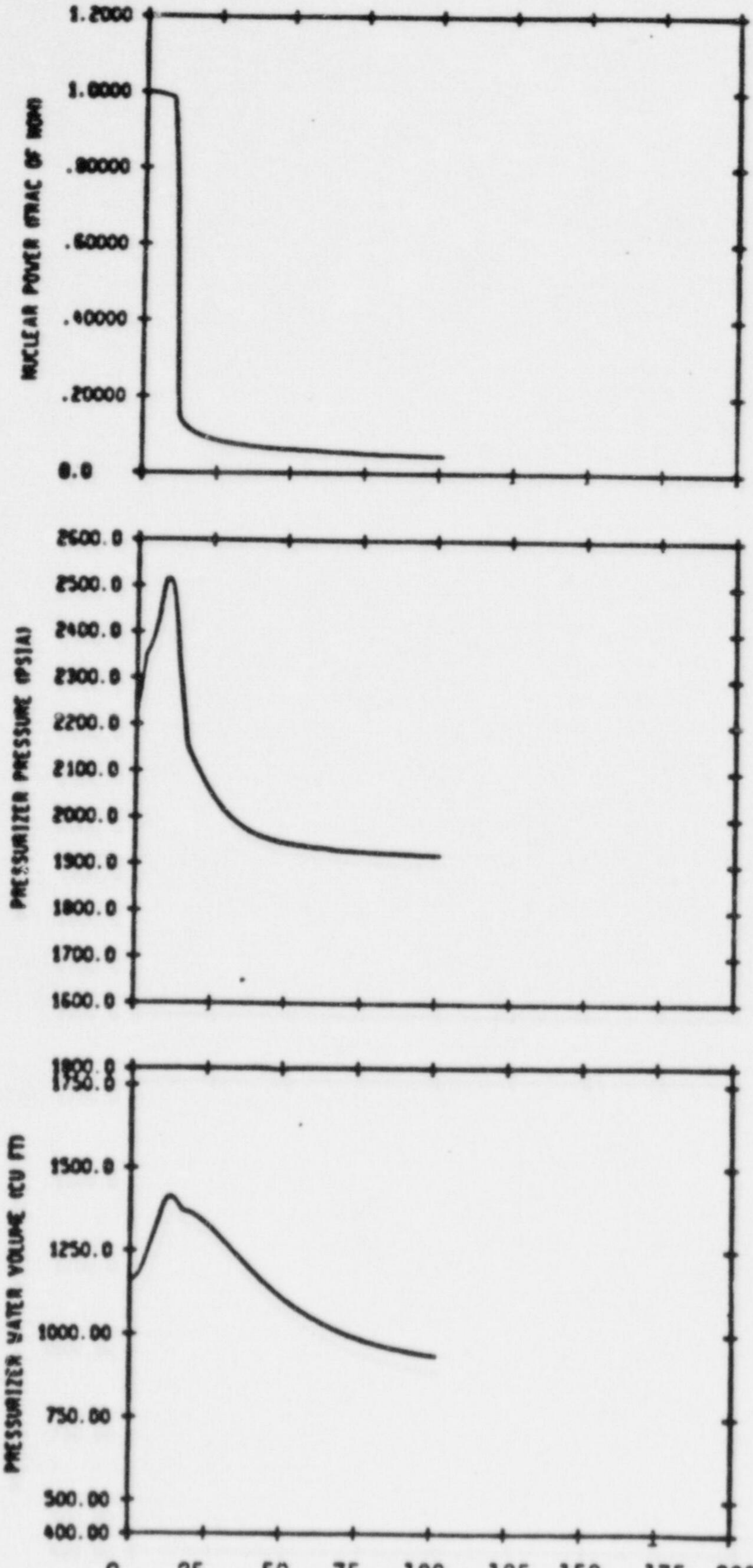

FIGURE 4.3-12
TURBINE TRIP EVENT
WITH PRESSURE CONTROL
MINIMUM REACTIVITY FEEDBACK

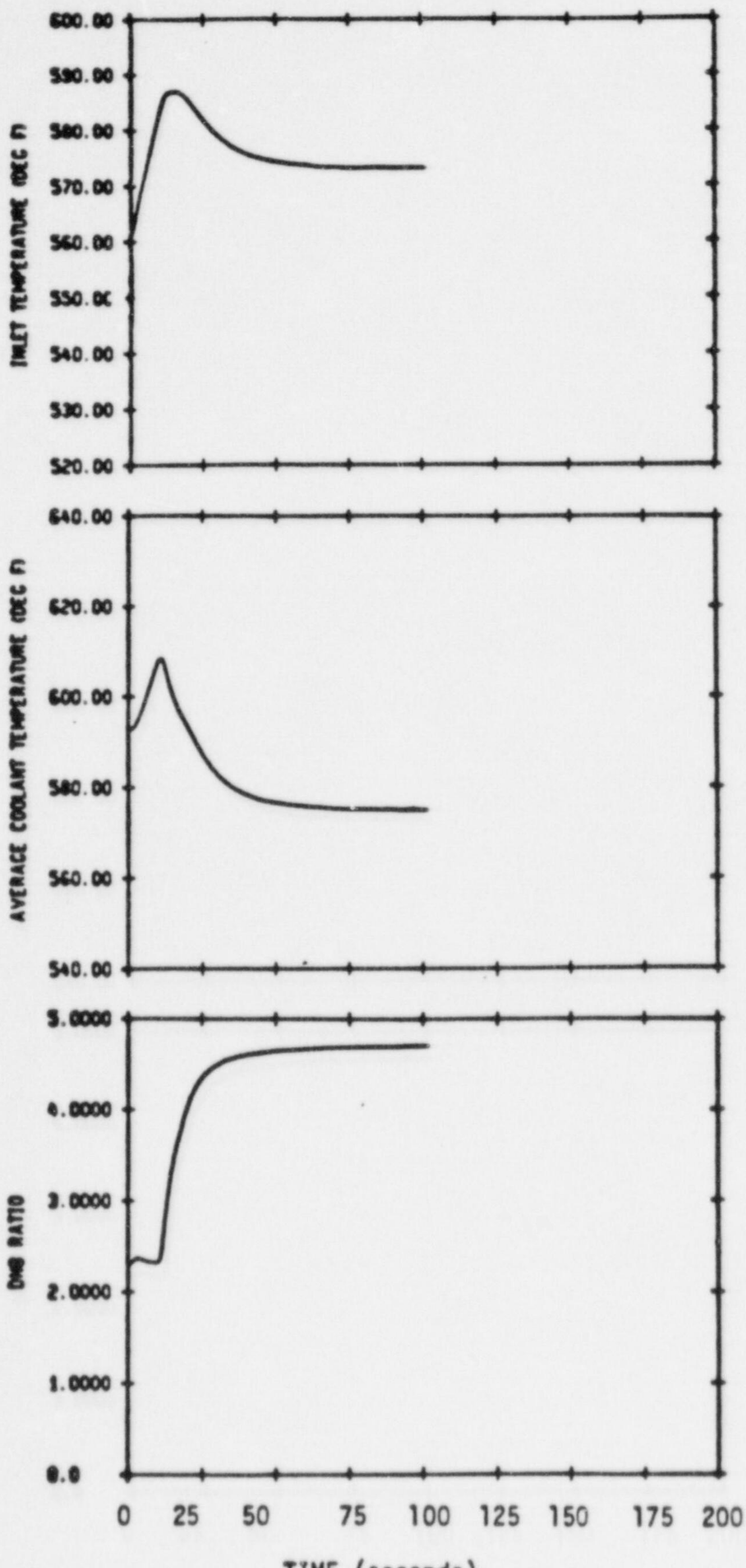

FIGURE 4.3-13
TURBINE TRIP EVENT
WITH PRESSURE CONTROL
MINIMUM REACTIVITY FEEDBACK

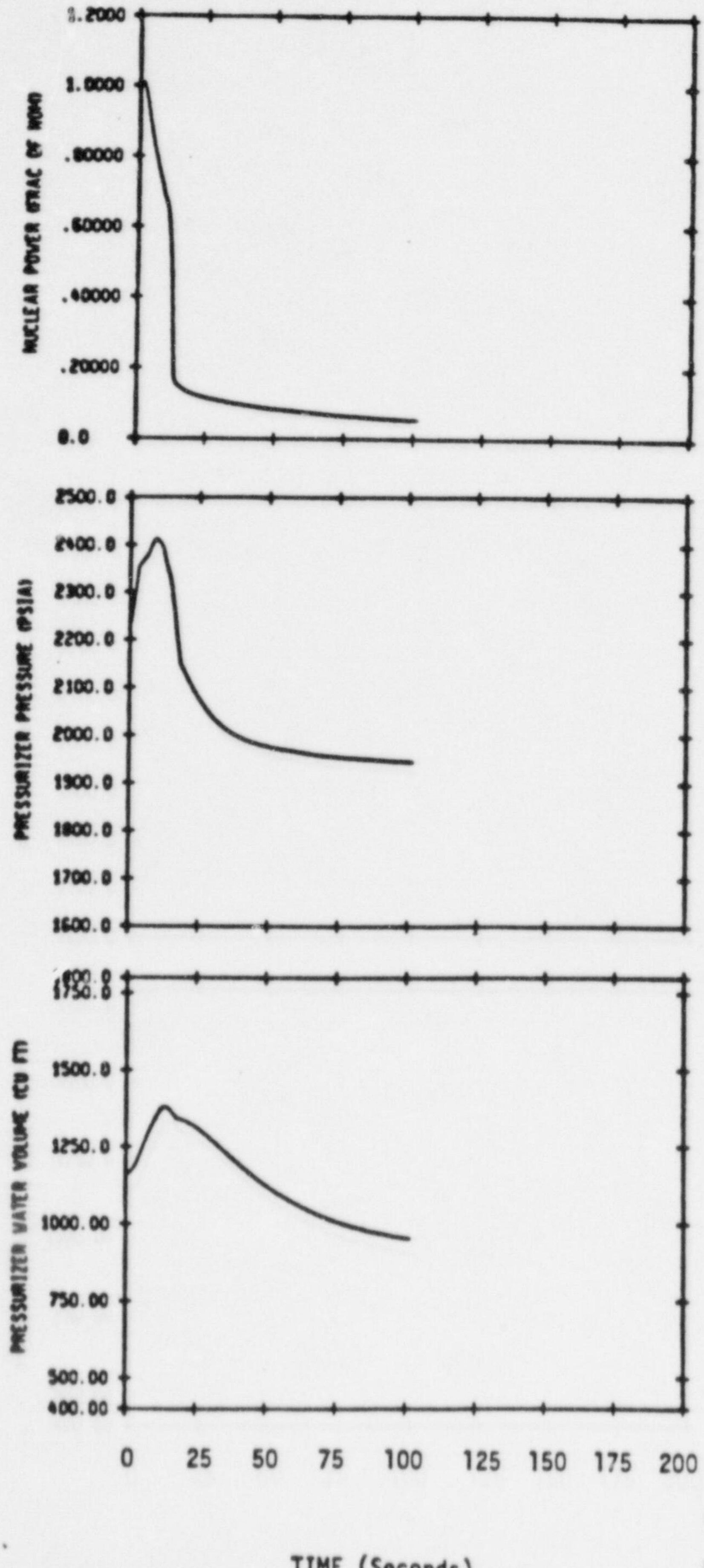

FIGURE 4.3-14
TURBINE TRIP EVENT
WITH PRESSURE CONTROL
MAXIMUM REACTIVITY FEEDBACK

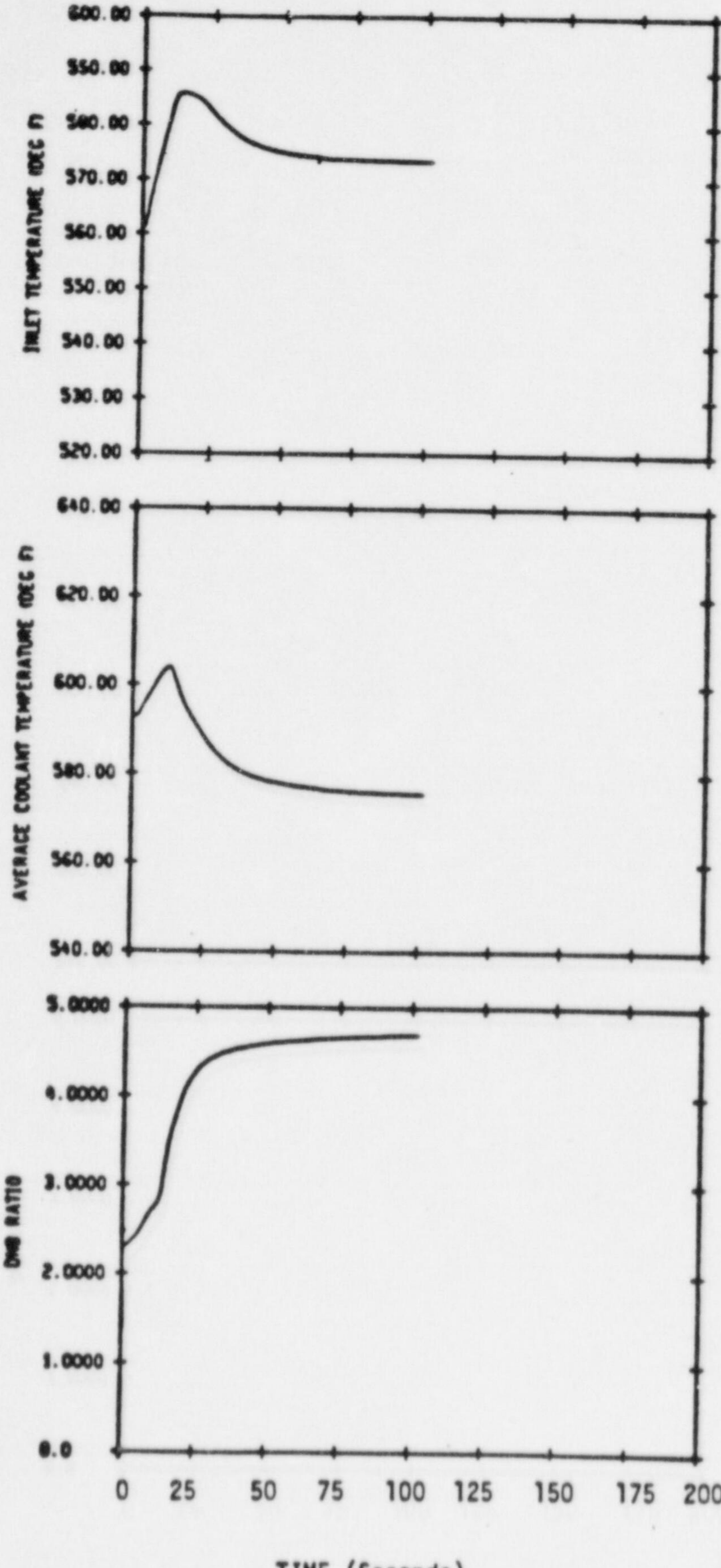

FIGURE 4.3-15
TURBINE TRIP EVENT
WITH PRESSURE CONTROL
MAXIMUM REACTIVITY FEEDBACK

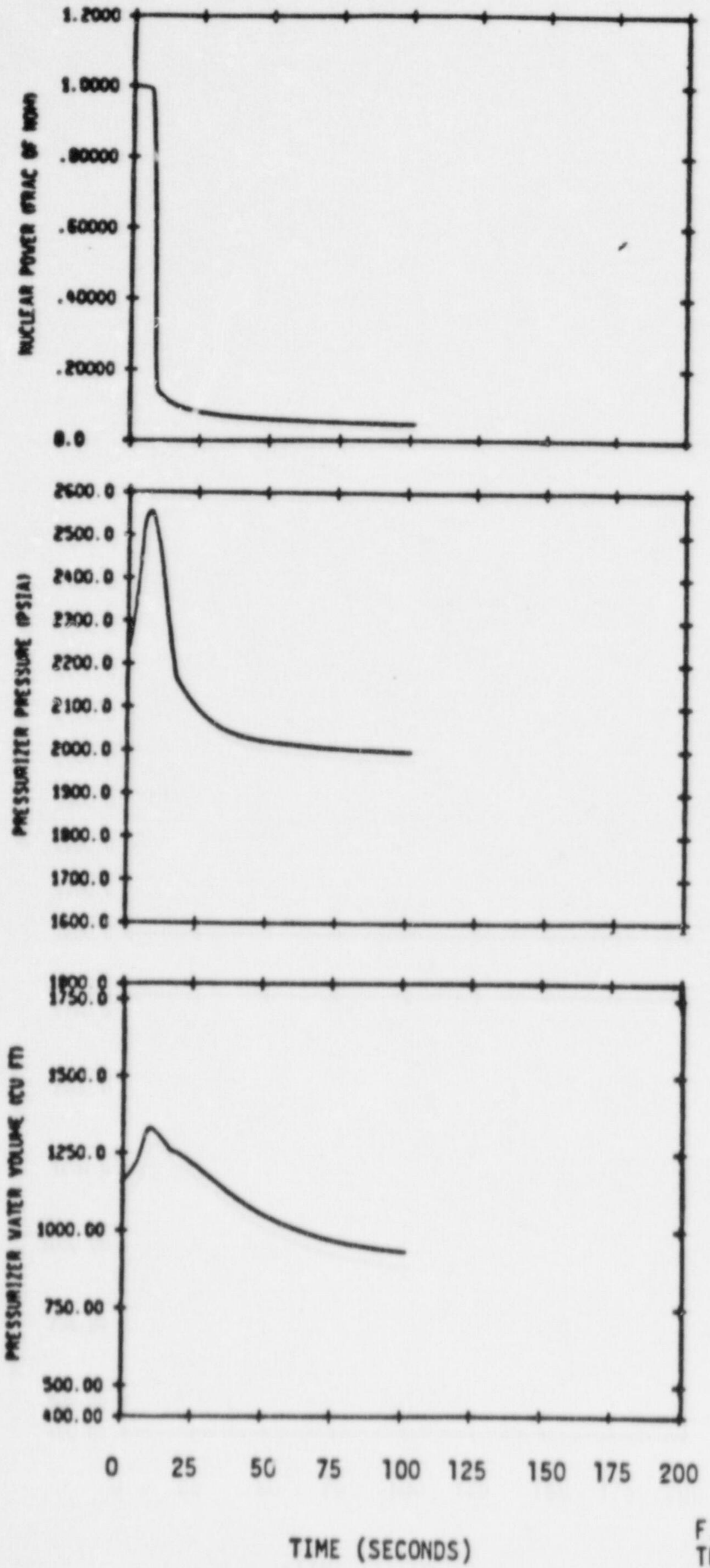

FIGURE 4.3-16
TURBINE TRIP EVENT
WITHOUT PRESSURE CONTROL
MINIMUM REACTIVITY FEEDBACK

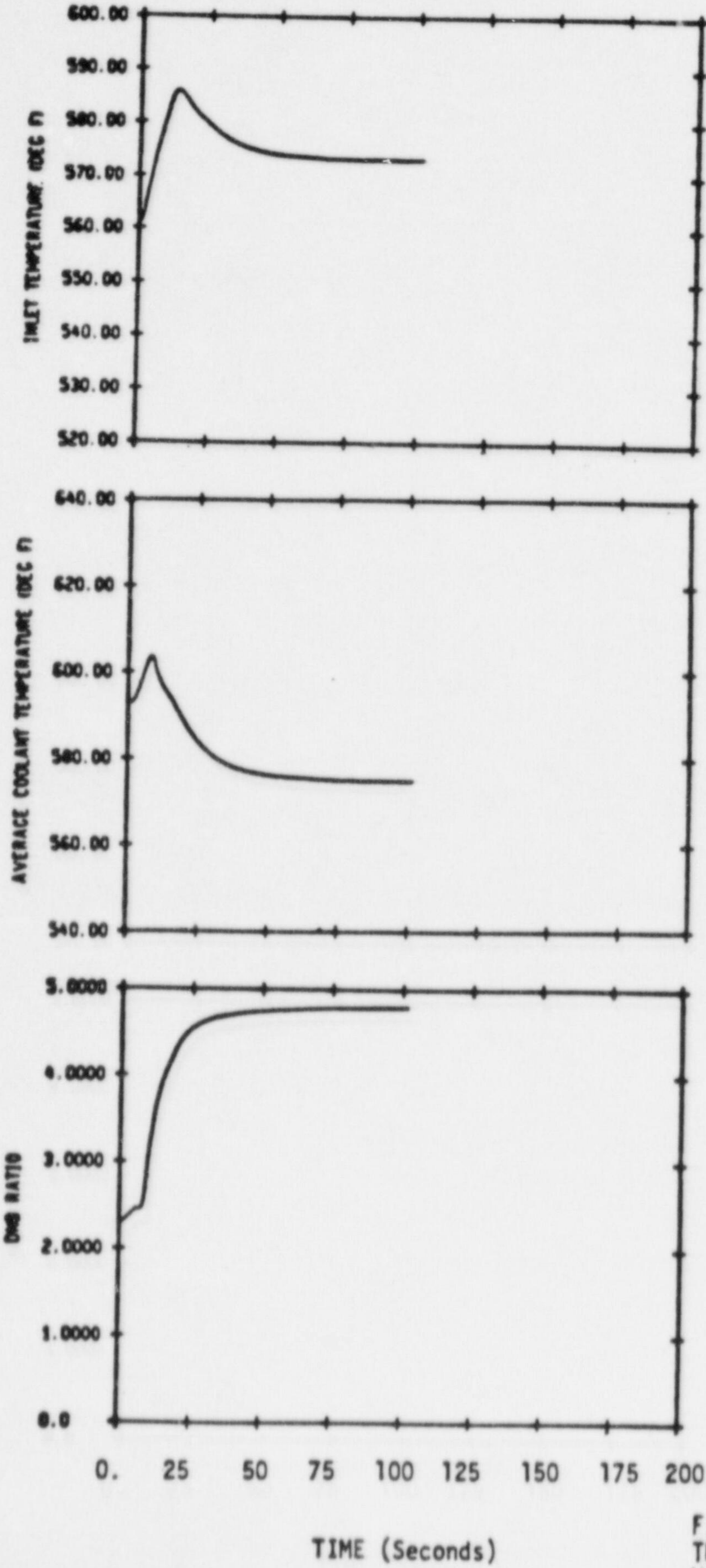

FIGURE 4.3-17
TURBINE TRIP EVENT
WITHOUT PRESSURE CONTROL
MINIMUM REACTIVITY FEEDBACK

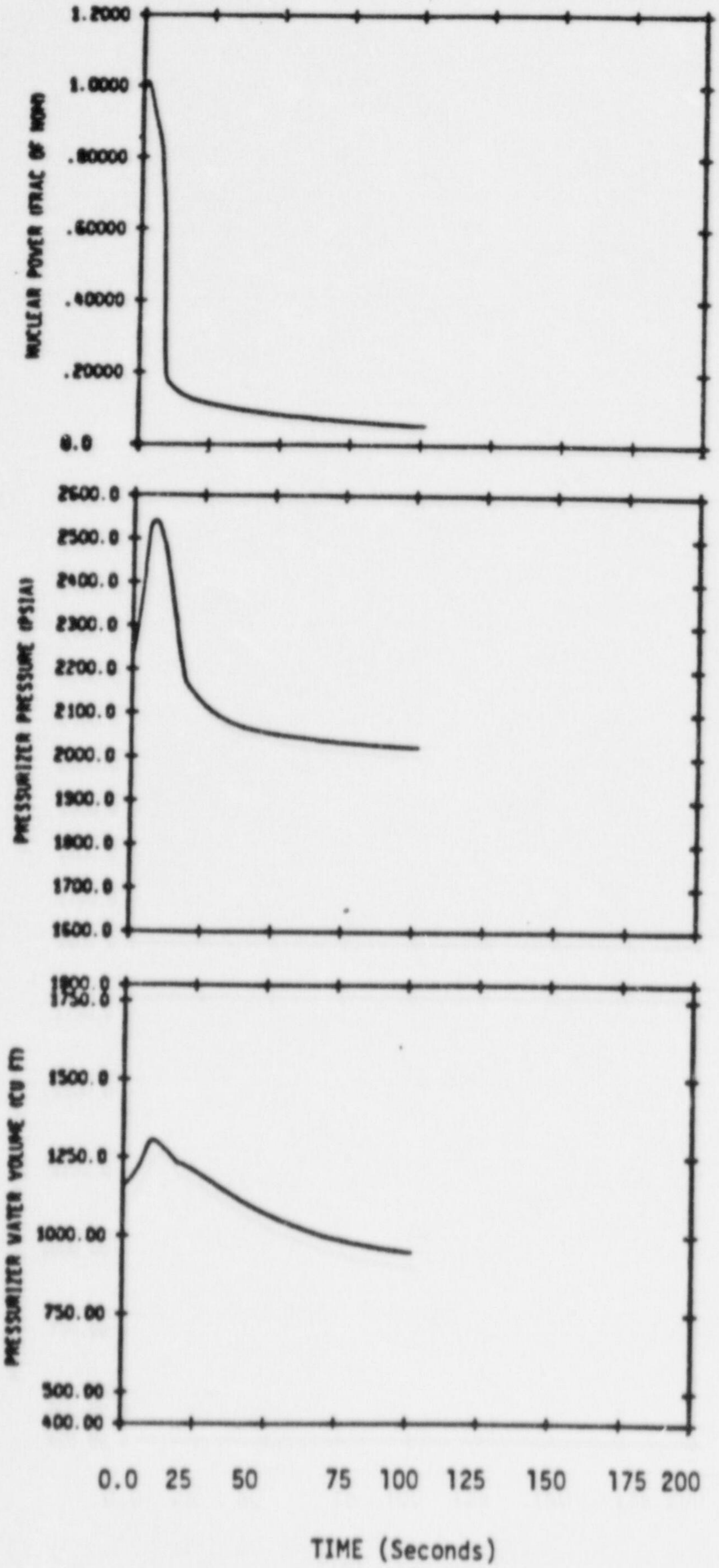

FIGURE 4.3-18
TURBINE TRIP EVENT
WITHOUT PRESSURE CONTROL
MAXIMUM REACTIVITY FEEDBACK

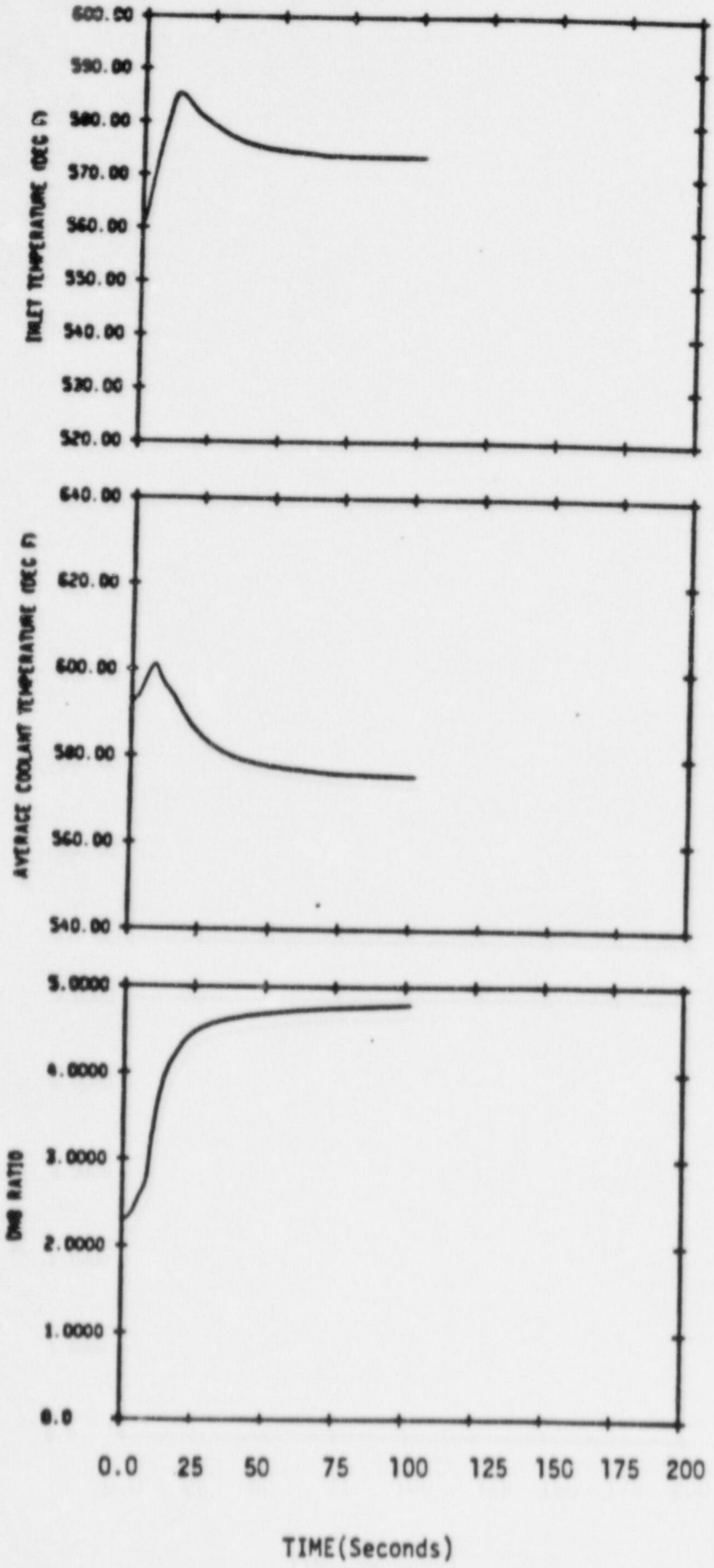

FIGURE 4.3-19
TURBINE TRIP EVENT
WITHOUT PRESSURE CONTROL
MAXIMUM REACTIVITY FEEDBACK

5.0 CONTROL SYSTEM EVALUATION

A prime input signal to the various NSSS control systems is the RCS average temperature (T_{avg}). This is calculated electronically as the average of the measured hot leg and cold leg temperatures in each loop.

The major control systems affected are [ ]^{a,c} The effect of the new RTD is to potentially change the time response of the T_{avg} channels in the various loops. However, as noted in Section 2.1, Table 2.1-1, the new RTD system will have a time response close to that of the present system. There will therefore be no significant effect on the T_{avg} channel response, and no apparent need to revise any of the control system setpoints from those presently installed in the plant. The need to modify control system setpoints will be determined during the plant startup following the installation of the new RTD system by observing control system behavior. In addition, the Cold Overpressurization Mitigation System (COMS) will be unaffected by the RTD bypass elimination since the COMS utilizes the wide range RTDs which are unaffected by this program.

6.0 CONCLUSIONS

The method of utilizing fast-response RTDs installed in the reactor coolant loop piping as a means for RCS temperature indication has undergone extensive analyses, evaluation and testing as described in this report. The incorporation of this system into the Byron/Braidwood design meets all safety, licensing and control requirements necessary for safe operation of these units. The analytical evaluation has been supplemented with in-plant and laboratory testing to further verify system performance. The fast-response RTDs installed in the reactor coolant loop piping adequately replace the present hot and cold leg temperature measurement system and enhances ALARA efforts and improved plant reliability.

7.0 REFERENCES

1. Tuley, C. R., Moomau, W. H., "RCS Flow Uncertainty for Shearon Harris Unit 1", WCAP-11168 Rev. 1, Proprietary, WCAP-11169 Rev. 1, Non-Proprietary, October, 1986.

2. Byron/Braidwood Final Safety Analysis Report, Amendment #47, April 1986.

3. Chelemer, H., et al., "Improved Thermal Design Procedure," WCAP-8567-P (Proprietary, WCAP-8568 (Non-Proprietary), July 1975.

4. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

APPENDIX A

TECHNICAL SPECIFICATION MODIFICATIONS

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤30.9% of RTP*
6. Source Range, Neutron Flux	17.0	10.0	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	9.7 ~~22.1~~	5.85 ~~5.38~~	See Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.8 ~~4.5~~	1.22 ~~1.5~~	1.2	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	2.21	1.5	≥1885 psig	≥1871 psig
10. Pressurizer Pressure-High	3.1	0.71	1.5	≤2385 psig	≤2396 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	2.5 1.42	~~1.77~~	0.6	≥90% of loop minimum measured flow*	88.8% ≥~~89.2~~% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low					
a. Unit 1	27.1	18.28	1.5	≥40.8% of narrow range instrument span	≥39.1% of narrow range instrument span
b. Unit 2	17.0	14.78	1.5	≥17% of narrow range instrument span	≥15.3% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	12.0	0.7	0	≥5268 volts - each bus	≥4728 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	14.4	13.3	0	≥57.0 Hz	≥56.5 Hz
16. Turbine Trip					
a. Emergency Trip Header Pressure	N.A.	N.A.	N.A.	≥540 psig	≥520 psig
b. Turbine Throttle Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	N.A.	N.A.

*Minimum measured flow = 97,600 gpm

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s,

T' $\leq$ 588.4°F (Nominal T_{avg} at RATED THERMAL POWER),

K_3 = 0.00134,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, s^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between $-\infty$% and +10%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds +10%, the ΔT Trip Setpoint shall be automatically reduced by 2.0% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~3.9%~~ of ΔT span.
3.2%

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00170/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.4°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.7 ~~2.6%~~ of ΔT span.

NOTE 5: The sensor error for temperature is 1.7 ~~1.2~~ and for pressure is 1.1 ~~1.0~~.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power					
a. ESF Bus Undervoltage	N.A.	N.A.	N.A.	2870 volts w/1.8s delay	$\geq$2730 volts w/$\leq$1.9s delay
b. Grid Degraded Voltage	N.A.	N.A.	N.A.	3804 volts w/310s delay	$\geq$3728 volts w/310 ± 30s delay
9. Engineered Safety Feature Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	$\leq$1930 psig	$\leq$1936 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
c. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$550°F	$\geq$~~547.6~~ 547.0°F
d. Steam Generator Water Level, P-14 (High-High)	See Item 5.b. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

POWER DISTRIBUTION LIMITS

BASES

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided the Conditions a. through d. above are maintained. The combination of the RCS flow requirement (390,400 gpm) and the requirement on $F^N_{\Delta H}$ guarantee that the DNBR used in the safety analysis will be met.

A rod bow penalty is not applied to the final value of $F^N_{\Delta H}$ for the following reason:

Fuel rod bowing does reduce the value of the DNBR. However, predictions with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979 for the 17x17 Optimized Fuel Assemblies indicate that the fuel rod bow reduction on DNBR will be less than 3% at 33,000 MWD/MTU assembly average burnup. At higher burnups, the decrease in fissionable isotopes and the buildup of fission product inventory more than compensate for the rod bow reduction in DNBR.

There is a 11% margin available between the 1.32 and 1.34 design DNBR limits and the 1.47 and 1.49 safety analysis DNBR limit. Use of the 3% fuel rod bow DNBR margin reduction still leaves a 8% margin in DNBR between design limits and safety analysis limits.

The RCS flow requirement is based on the loop minimum measured flow rate of 97,600 gpm which is used in the Improved Thermal Design Procedure described in FSAR 4.4.1 and 15.0.3. A precision heat balance is performed once each cycle and is used to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% is assessed for potential feedwater venturi fouling. A maximum measurement
1.9% uncertainty of ~~2.2%~~ has been included in the loop minimum measured flow rate to account for potential undetected feedwater venturi fouling and the use of the RCS flow indicators for flow rate verification. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken, before performing subsequent precision heat balance measurements, i.e., either the effect of fouling shall be quantified and compensated for in the RCS flow rate measurement, or the venturi shall be cleaned to eliminate the fouling.

Surveillance Requirement 4.2.3.4 provides adequate monitoring to detect possible flow reductions due to any rapid core crud buildup.

Surveillance Requirement 4.2.3.5 specifies that the measurement instrumentation shall be calibrated within seven days prior to the performance of the calorimetric flow measurement. This requirement is due to the fact that the drift effects of this instrumentation are not included in the flow measurement uncertainty analysis. This requirement does not apply for the instrumentation whose drift effects have been included in the uncertainty analysis.