ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES (UNIT 1)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-6	1 of 2 Intermediate range above setpoint (increasing power level)	1×10^{-10}	$<3 \times 10^{-10}$	Allows manual block of source range reactor trir
	2 of 2 Intermediate range below setpoint (decreasing power level)	5×10^{-11}	$>3 \times 10^{-11}$	Defeats the block of source range reactor trip
P-10	2 of 4 Power range above setpoint (increasing power level)	10%	<11%	Allows manual block of power range (low setpoint) and intermediate range reactor trips and intermediate range rod stop. Blocks source range reactor trip.
	3 of 4 Power range below setpoint (decreasing power level)	8%	>7%	Defeats the block of power range (low setpoint) and intermediate range reactor trips and intermediate range rod stop. Input to P-7.
P-7	2 of 4 Power range above setpoint or 1 of 2 Turbine Impulse chamber pressure above setpoint (Power level increasing)	10% Pressure equivalent to 10% rated turbine power	<11% <11%	Allows reactor trip when any of the following occur in more than one loop: low flow, reactor coolant pump breaker open, undervoltage (RCP busses) or underfrequency (RCP busses). Also allows reactor trip on: pressurizer low pressure or pressurizer high level.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-7 (Cont'd)	3 of 4 Power range below setpoint and	8%	>7%	Prevents reactor trip when any of the following occur: low flow, reactor coolant pump breakers open, under-voltage (RCP busses), under-frequency (RCP busses), pressurizer low pressure or pressurizer high level.
	2 of 2 Turbine Impulse chamber pressure below setpoint (Power level decreasing)	8%	>7%	
P-8	2 of 4 Power range above setpoint (Power level increasing)	30%	<31%	Allows reactor trip when any of the following occur: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip.
	3 of 4 Power range below setpoint (Power level decreasing)	28%	>27%	Prevents reactor trip when any of the following occur: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip.

of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by ≥ 1.616×10^6 lbs/hour of full steam flow at RATED THERMAL POWER. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The reactor trip due to the Undervoltage and Underfrequency on the Reactor Coolant Pump busses provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.5 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.1 seconds. The undervoltage and underfrequency trips are automatically blocked when reactor power is below the P-7 setpoint.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-8. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES (UNIT 2)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-6	1 of 2 Intermediate range above setpoint (increasing power level)	1×10^{-10}	$<3 \times 10^{-10}$	Allows manual block of source range reactor trip
	2 of 2 Intermediate range below setpoint (decreasing power level)	5×10^{-11}	$>3 \times 10^{-11}$	Defeats the block of source range reactor trip
P-10	2 of 4 Power range above setpoint (increasing power level)	10%	<11%	Allows manual block of power range (low setpoint) and intermediate range reactor trips and intermediate range rod stop. Blocks source range reactor trip.
	3 of 4 Power range below setpoint (decreasing power level)	8%	>7%	Defeats the block of power range (low setpoint) and intermediate range reactor trips and intermediate range rod stop. Input to P-7.
P-7	2 of 4 Power range above setpoint or 1 of 2 Turbine Impulse chamber pressure above setpoint (Power level increasing)	10% Pressure equivalent to 10% rated turbine power	<11% <11%	Allows reactor trip when any of the following occur in more than one loop: low flow, reactor coolant pump breaker open, undervoltage (RCP busses) or underfrequency (RCP busses). Also allows reactor trip on: pressurizer low pressure or pressurizer high level.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-7 (Cont'd)	3 of 4 Power range below setpoint and 2 of 2 Turbine Impulse chamber pressure below setpoint (Power level decreasing)	8% 8%	>7% >7%	Prevents reactor trip when any of the following occur: low flow, reactor coolant pump breakers open, under-voltage (RCP busses), under-frequency (RCP busses), pressurizer low pressure or pressurizer high level.
P-8	2 of 4 Power range above setpoint (Power level increasing)	30%	<31%	Allows reactor trip when any of the following occur: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip.
	3 of 4 Power range below setpoint (Power level decreasing)	28%	>27%	Prevents reactor trip when any of the following occur: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The reactor trip due to the Undervoltage and Underfrequency on the Reactor Coolant Pump busses provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.5 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.1 seconds. The undervoltage and underfrequency trips are automatically blocked when reactor power is below the P-7 setpoint.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-8. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of any one pump breaker above P-8 or the opening of two or more pump breakers below P-8. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip set point

ATTACHMENT 3

SAFETY EVALUATION IN SUPPORT OF
DELETION OF REACTOR TRIP ON TURBINE TRIP BELOW 30% OF RATED
THERMAL POWER
NORTH ANNA UNITS 1 AND 2

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1.0 DISCUSSION OF PROPOSED CHANGES

At present, for all power levels above 10% (the P-7 permissive setpoint) of Rated Thermal Power (RTP), the North Anna nuclear reactors are tripped directly on turbine trip from a signal derived from the turbine autostop oil pressure or turbine stop valve position. Historically, a number of reactor trips have been caused by turbine trips at low power, i.e. below the designed 50% load rejection capability. A direct reactor/turbine trip at low power is unnecessary, has inherent costs due to increased down time, and unduly stresses plant systems. Thus, Virginia Electric and Power Company (Virginia Power) is proposing a change which would allow for a block of the direct reactor trip on turbine trip below 30% of rated thermal power.

Currently, Permissive P-8 is used to enable and block reactor trip protection for low RCS flow conditions (low loop flow or reactor coolant pump breaker open) below 30% power level. The proposed modification would rewire the Solid State Protection System so that Permissive P-8 is also used to block the reactor trip on turbine trip instead of permissive P-7. A Virginia Power review of historical trip data shows that the most commonly occuring reactor trip on turbine trip events are well below 30% of rated thermal power. Thus it was concluded that the use of the existing P-8 bistable to block the direct reactor trip on turbine trip will be an effective means of eliminating unneeded low power reactor trips. Direct reactor trip on turbine trip will still be available above 30% power. The plant's designed load rejection capability is 50% of full load.

Virginia Power is proposing the following Technical Specification Changes which support blocking of reactor trip on turbine trip below 30% of rated thermal power. A safety evaluation of these changes is presented in subsequent sections. Each change is discussed separately in the following paragraphs. The Specifications that follow and the changes discussed are identical for Units 1 and 2.

Discussion of Technical Specification Changes

3/4.3.1 Table 3.3-1 REACTOR TRIP SYSTEM INTERLOCKS

Permissive P-7 is currently interlocked to block the direct reactor trip on turbine trip below 10% of RTP. Therefore, the proposed Technical Specification changes delete reactor trip on turbine trip from the list of trips associated with Permissive P-7. In addition, the function of P-7 has been reworded slightly for clarity.

Permissive P-8, at present, blocks the reactor trip on low flow or reactor coolant pump breaker open in a single loop below 30% of RTP. The proposed Technical Specification change allows a block of the reactor trip on turbine trip below 30% of RTP. Thus, the function column of Table 3.3.1 for Permissive P-8 is changed to read as follows:

> Allows reactor trip when any of the following occurs: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip (Power levels > 30% RTP increasing)
>
> Prevents reactor trip when any of the following occurs: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip (Power levels < 28% RTP decreasing)

Bases page B 2-7 is changed to reflect the above change in reactor trip system interlock P-8.

Methodology

The following three items have been considered from a safety analysis standpoint and are addressed in this evaluation:

a. It has been demonstrated that the loss of external load accident initiated from 30% power would be acceptable and meet ANS Condition II Criteria. Thus, this analysis was performed to demonstrate the adequacy of the primary and secondary side pressure relieving devices and to show that the minimum DNBR is above the limit value.

b. An analysis has been performed to show that the occurence of a loss of flow event during a loss of load, which could result from a failure of a fast bus transfer to offsite power after a 30 second turbine generator motoring delay or similar events, would be acceptable. Loss of load could lead to elevated inlet temperature at the time of the loss of flow; the impact on DNBR has therefore been analyzed.

c. An analysis has been performed to demonstrate on a best estimate basis that a turbine trip without reactor trip at reduced power will not challenge the pressurizer Power Operated Relief Valves (PORV). Virginia Power's response to NUREG-0737 Post-TMI requirements, submitted in Reference 1, committed to a program of reducing the probability of a small break LOCA due to a stuck open PORV such that it is not a significant contributor to the probability of a small break LOCA due to all causes. The results of this program were documented by the Westinghouse Owners Group in Reference 2. Based on the results of our analysis, the stuck open PORV will remain an insignificant contributor to the small break LOCA frequency.

Section 2 discusses the identification of causes and accident description. Section 3 presents the analysis assumptions and transient response of these accidents. Section 4 discusses the results of the analyses.

2.0 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

Major load loss on the plant can result from loss of external electrical load or from a turbine trip. For either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant is designed to accept a 50% step loss of load without actuating a reactor trip. The automatic steam bypass system (steam dump valves) with 40% steam dump capacity to the condenser is able to accommodate this load rejection by reducing the transient imposed upon the reactor coolant system. The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the rod control system. The pressurizer relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves do not lift for the 50% load rejection with steam dump.

Should the steam dump valves fail to open, or should their capacity be exceeded following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature delta T signal. The steam generator shell side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the reactor coolant system and steam generator against overpressure for all load losses without assuming the operation of the steam dump valves and steam generator PORVs, pressurizer spray, pressurizer power operated relief valves, automatic rod cluster control assembly control, or direct reactor trip resulting from turbine trip. This capability is demonstrated by the analysis presented in Section 15.2.7, Chapter 15 of the UFSAR.

The steam generator safety valve capacity is sized to remove the steam flow at the engineered safeguards design rating (2910 MWt) from the steam generator without exceeding 110% of the steam system design pressure. The pressurizer safety valve capacity is based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the reactor coolant system pressure within 110% of the reactor coolant system design pressure without taking credit for direct (i.e. on turbine trip) or immediate reactor trip action. Consequently, this incident is not sensitive to initial pressurizer level, therefore the programmed level versus power is assumed.

For a turbine trip, the reactor would be tripped directly (unless below approximately 10% power for the current design) from a signal derived from the turbine autostop oil pressure or turbine stop valves. The auto steam dump valves would accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser was not available, the excess steam generation would be dumped to atmosphere through the steam generator PORVs and safety valves. Additionally, main feedwater flow would be lost if the turbine condenser was not available. For this situation, feedwater would be maintained by the auxiliary feedwater system. Turbine trip initiation signals include:

1. Generator Trip
2. Low Condenser Vacuum
3. Loss of Lubricating Oil
4. Turbine Thrust Bearing Failure
5. Turbine Overspeed
6. Manual Trip
7. Reactor Trip

Normal power for the reactor coolant pumps is supplied through station service busses from station service transformers connected to the generator 22KV bus. When a generator trip occurs on Unit 2, these busses are automatically transferred to the reserve station service transformers supplied from offsite power, and the pumps continue to supply coolant flow to the core. When a generator trip occurs on Unit 1, the generator breaker opens and these busses remain powered from the station service transformers which receive offsite power. Should the generator breaker on Unit 1 fail or power be lost, these busses will automatically transfer to the reserve station service transformers. The case of a direct generator trip on turbine trip is bounded by the current analysis of a loss of offsite power to the station auxiliaries presented in Section 15.2.9 of the UFSAR.

Following any turbine trip where there are no faults which require immediate tripping of the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to this generator, thus ensuring flow for 30 seconds before any transfer is made. Should a reactor trip not occur in the first 30 seconds following the turbine trip (the period during which the generator is motored), a complete loss of forced reactor coolant flow should be assumed to occur due to a postulated failure in the fast bus transfer to offsite power for Unit 2. The immediate effect of loss of coolant flow is a rapid increase in the reactor coolant temperature superimposed on the already increased coolant temperature resulting from the turbine trip. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped within a short time period following the loss of flow.

The following signals provide the necessary protection during a complete loss of flow accident.

1. Reactor coolant pump power undervoltage and underfrequency.
2. Reactor coolant pump breakers open.
3. Low reactor coolant loop flow.

For the analysis of this event, the two important signals are reactor coolant pump undervoltage and low reactor coolant loop flow.

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps i.e. station blackout. This function is blocked below approximately 10 percent power (Permissive P-7).

The reactor trip on low reactor coolant loop flow is provided to protect against loss of flow conditions which affects one or more reactor coolant loop. Between approximately 10% power (Permissive P-7) and 30% power (Permissive P-8), low flow in any two loops will actuate a reactor trip. Above 30% power (Permissive P-8), low flow in any single loop will actuate a reactor trip.

3.0 ANALYSIS OF EFFECTS AND CONSEQUENCES

3.1 THE LOSS OF LOAD/TURBINE TRIP EVENT

This event is described in Section 15.2.7, Chapter 15 of the North Anna UFSAR. This transient was analyzed at 100% of rated thermal power, where it is limiting. The existing analysis shows acceptable results for the complete load rejection from 100% power without taking credit for the direct reactor trip on turbine trip, and will bound the results at 30% power. However, an explicit analysis at 30% power was performed to support this Technical Specification change.

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 40% of full power (30% of rated thermal power plus 10% instrumentation uncertainties) without a direct reactor trip on turbine trip, primarily to show the adequacy of the pressure-relieving devices and also to demonstrate core protection margins. The assumptions delay reactor trip until conditions in the RCS result in a trip due to a signal other than turbine trip. Thus, the analysis assumes a worst transient. In addition no credit is taken for condenser steam dumps or steam generator PORVs. Main feedwater is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

This accident is analyzed by using the computer program RETRAN (Reference 3). The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables, including temperatures, pressures, and power level. The

COBRA (Reference 4) code is then used to calculate the minimum DNBR during the transient based upon the heat flux, flow, temperature, and pressure from RETRAN. The WRB-1 CHF correlation is used (Reference 5).

Assumptions are:

1. Initial Operating Conditions - The initial reactor power is assumed at 40 percent of nominal full power. The initial reactor coolant system pressure and temperatures are assumed at their nominal value consistent with the steady-state 40% power level. Allowances for calibration and instrument errors are incorporated into the DNBR limit value as described in Reference 6. These assumptions result in the maximum power difference for the load loss, and the minimum margin to core protection limits at the initiation of the accident.

2. Moderator and Doppler Coefficients of Reactivity - The total loss of load is analyzed for both beginning-of-life and end-of-life conditions. A positive moderator temperature coefficient at beginning of life and a large (absolute value) negative value at end of life are used. Least negative and most negative Doppler power coefficients are used for minimum and maximum feedback, respectively.

3. Reactor Control - From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control.

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

setpoint value (the analysis assumes the valves open at the highest safety valve lift point of 1150 psia).

5. Pressurizer Spray and Power-Operated Relief Valves - Two cases for both the beginning and end of life are analyzed:

   a. Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure.

   b. No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure.

6. Feedwater Flow - Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow following the reactor trip since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

Reactor trip is actuated by the first reactor protection system trip setpoint reached, with no credit taken for the direct reactor trip on the turbine trip.

In summary, four cases were analyzed;

Case A BOC Minimum Feedback with pressurizer control

Case B BOC Minimum Feedback without pressurizer control

Case C EOC Maximum Feedback with pressurizer control

Case D EOC Maximum Feedback without pressurizer control

3.2 THE LOSS OF LOAD/TURBINE TRIP WITH LOSS OF FLOW EVENT

This is not an original design basis accident. This scenario is analyzed in detail and documented in this evaluation.

An attempted fast bus transfer is assumed 30 seconds following the loss of steam load. The transfer to an external power source is assumed to fail which results in a complete loss of flow transient initiated from the loss of load condition.

The assumptions used in this analysis are the same as given in the previous section.

In summary, four cases were analyzed;

Case E BOC Minimum Feedback with pressurizer control

Case F BOC Minimum Feedback without pressurizer control

Case G EOC Maximum Feedback with pressurizer control

Case H EOC Maximum Feedback without pressurizer control

The initial conditions and key safety parameters used in the analysis are given in Table 3.1.1.

TABLE 3.1.1 INITIAL CONDITIONS AND KEY SAFETY PARAMETERS AND ASSUMPTIONS

Core Power	
40% of Rated Thermal Power	1157.2 Mwt
Thermal Design Flow (GPM)	289200
Reactor Coolant Temperature	
Vessel Outlet, °F	577
Vessel Inlet, °F	549.6
Steam Generator Steam	
Temperature, °F	537
Pressure, PSIA	924
Moderator Temperature Coefficients	+6 pcm/F (BOC)
	-59 pcm/F (EOC)
Doppler Temperature Coefficients	-1.4 pcm/F (BOC)
	-2.9 pcm/F (EOC)
Doppler Power Coefficients	
Least Negative (Hot Zero Power to Hot Full Power)	-10.2 to -6.7 pcm/% of power
Most Negative (Hot Zero Power to Hot Full Power)	-20 to -14 pcm/% of power
Beta-eff	.0075 (BOC)
	.0043 (EOC)
Normalized Trip Reactivity	4% dk/k
Generator Motoring Time	30.0 seconds
Undervoltage Trip Delay Time	1.2 seconds

3.3 PRESSURIZER PORV RESPONSE EVALUATION

To demonstrate that the pressurizer PORVs are not challenged during the loss of load transient without reactor trip, more of a best estimate analysis was performed.

NUREG-0737 required that the frequency of a small break LOCA caused by a stuck-open PORV be reduced and that it be demonstrated not to be a significant contributor to the probability of a small break LOCA. The loss of load and loss of flow both have the potential of causing the PORV to open (Reference 2). Thus, an analysis was also performed to demonstrate that the PORVs are not normally challenged during this event at reduced power. The following case was analyzed;

CASE I BOC Minimum Feedback with pressurizer control and credit taken for steam dump.

A most positive moderator temperature coefficient was used in the transient and no credit was taken for the heat transfer to the reactor coolant system metal. Main feedwater was terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient. Credit was taken for pressurizer spray, the condenser steam dumps and the steam generator PORVs.

In this analysis, loss of flow is not assumed. The reactor trips eventually on low steam generator mass. The transient is more severe than the case with subsequent loss of flow with respect to the PORV challenge in that the latter case would result in an early reactor trip. These assumptions make this a conservative analysis with respect to maximizing the pressure increase. Figure 4.3.2 shows that the PORVs are not challenged using these limiting parameters.

4.0 EVALUATION OF RESULTS

4.1 TURBINE TRIP/LOSS OF LOAD

The transient responses for a total loss of load from 40% power level were analyzed in detail for four cases: two cases for the beginning of core life and two cases for the end of core life. Two cases of these analyses (minimum feedback with and without pressurizer control) were found to be limiting with respect to the other two. Thus the results of these analyses are shown in Figures 4.1.1 through Figures 4.1.14

Figures 4.1.1 thru 4.1.6 show the transient response for the turbine trip/loss of load transient with minimum feedback. Full credit was taken for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the condenser steam dump and steam generator PORVs. The minimum DNBR remains well above the design limit value as shown in Figure 4.1.7 and is bounded by the loss of load analysis delineated in UFSAR Section 15.2.7.

Figures 4.1.8 thru 4.1.13 show the transient response due to the turbine trip/loss of load transient with minimum feedback. All the assumptions are the same as in the previous case except credit is not taken for the pressurizer spray and pressurizer power operated relief valves. The minimum DNBR is always above the design limit value as shown in Figure 4.1.14 and is bounded by the loss of load analysis delineated in UFSAR Section 15.2.7. The peak pressure remains well within the design limit.

The time sequences of events for these two cases are shown in Tables 4.1.1 and 4.1.2.

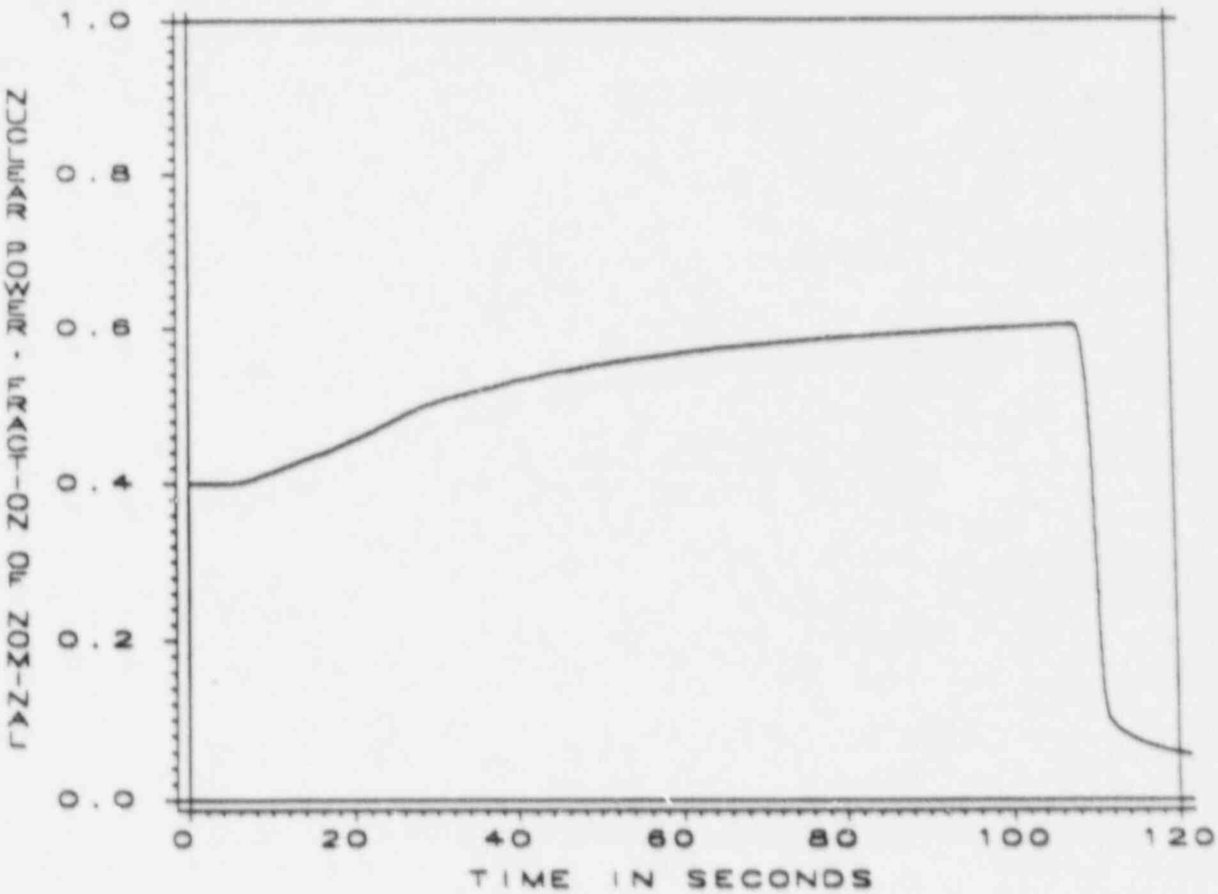
Table 4.1.1 Time Sequence of Events for Turbine Trip with Pressurizer Control

Minimum Feedback (BOC)	EVENT	TIME(s)
	Turbine Trip	0.0
	Peak pressurizer pressure occurs	14.5
	Initiation of steam release from steam generator safety valves	21.1
	Low Steam Generator mass trip (simulates low-low level)	106.3
	Rod begins to fall	108.3
	Minimum DNBR occurs	108.5

Table 4.1.2 Time Sequence of Events for a Turbine Trip w/o Pressurizer Control

Minimum Feedback (BOC)	EVENT	TIME(s)
	Turbine Trip	0.0
	Hi pressurizer pressure trip setpoint reached	12.3
	Rod begins to fall	14.3
	Minimum DNBR occurs	16.0
	Peak pressurizer pressure occurs	16.0
	Initiation of steam release from steam generator safety valves	21.0

FIGURE 4.1.1
NA TURBINE TRIP/LOSS OF LOAD
NUCLEAR POWER, FRACTION OF NOMINAL
MINIMUM FEEDBACK, PRESSURIZER CONTROL



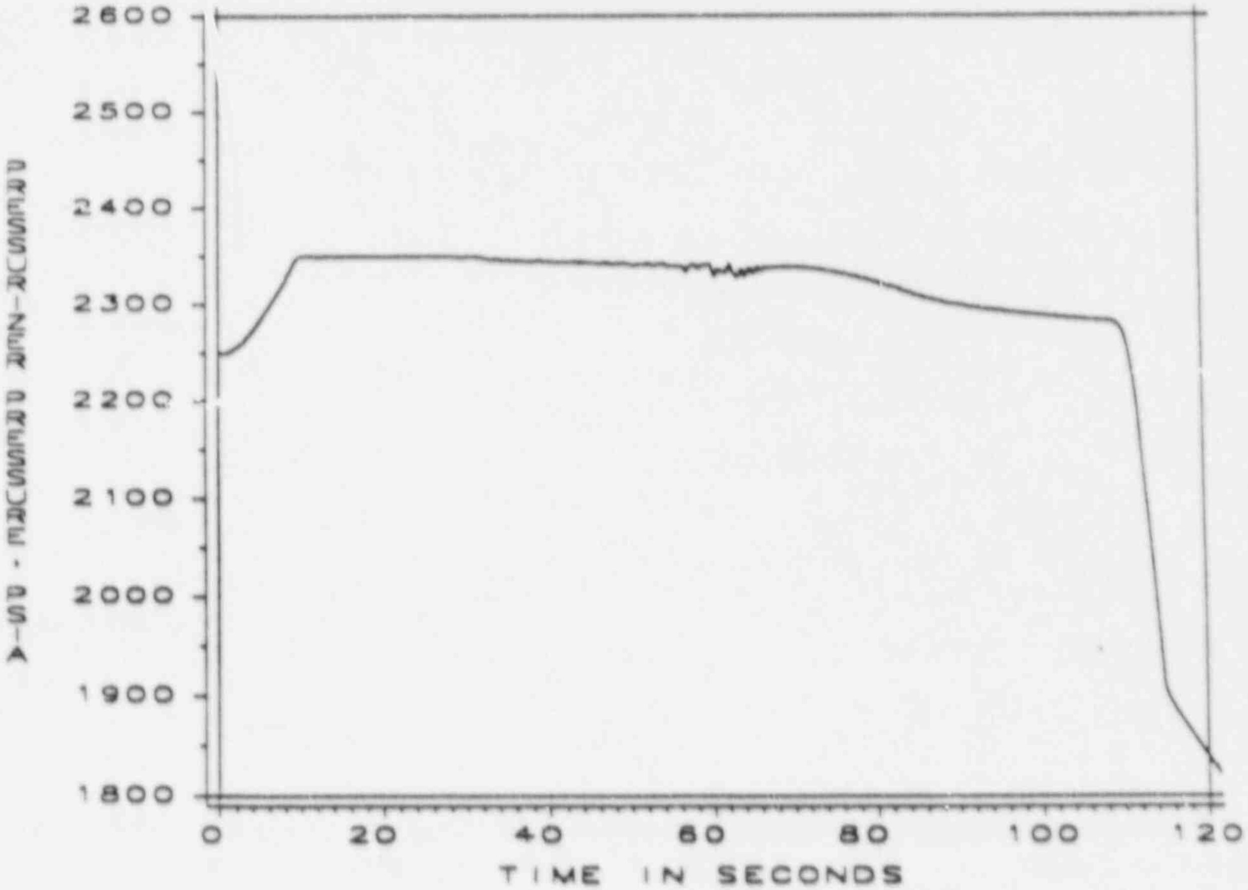
MINIMUM FEEDBACK

FIGURE 4.1.2

NA TURBINE TRIP/LOSS OF LOAD

PRESSURIZER PRESSURE, PSIA

MINIMUM FEEDBACK, PRESSURIZER CONTROL



MINIMUM FEEDBACK

FIGURE 4.1.3

NA TURBINE TRIP/LOSS OF LOAD

PRESSURIZER WATER VOLUME, FT3

MINIMUM FEEDBACK, PRESSURIZER CONTROL

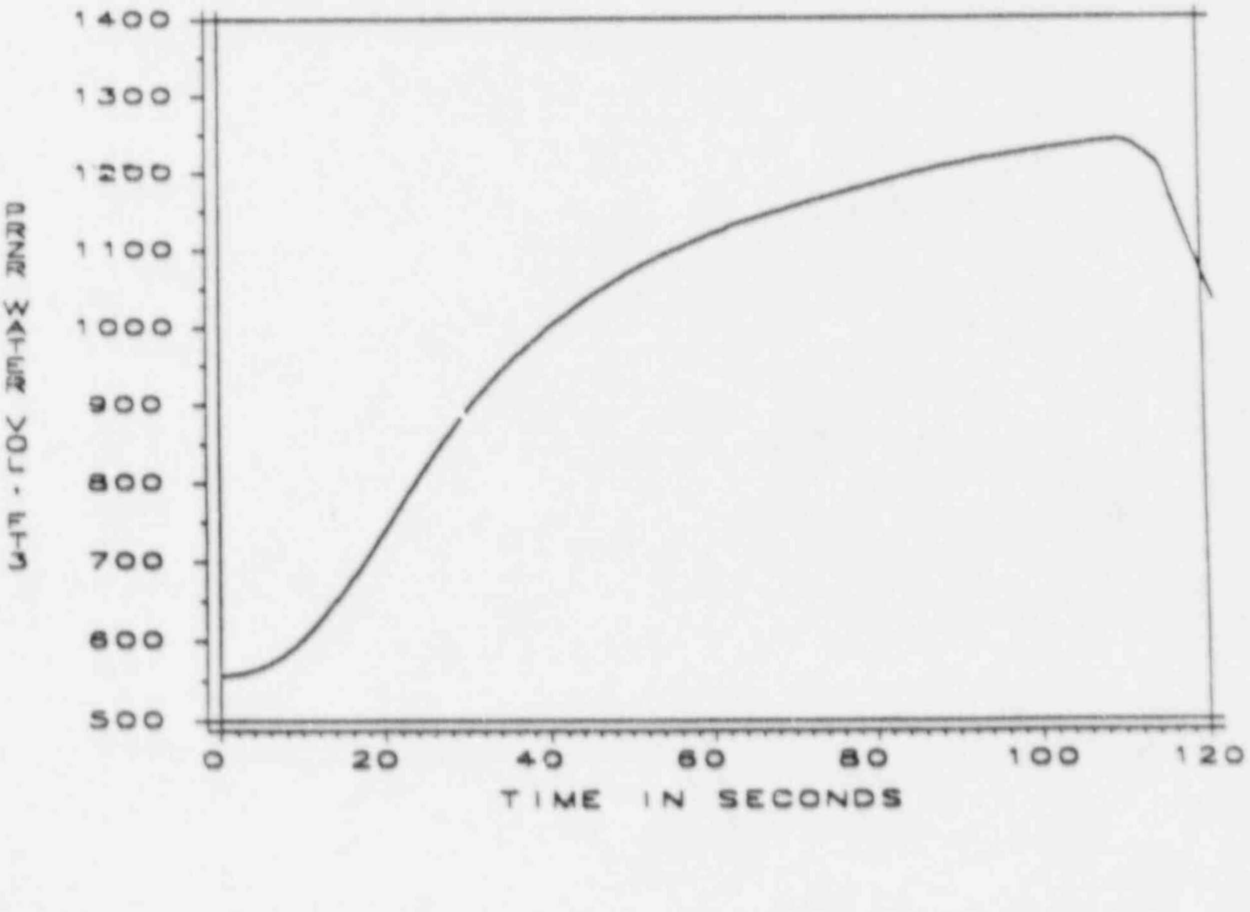
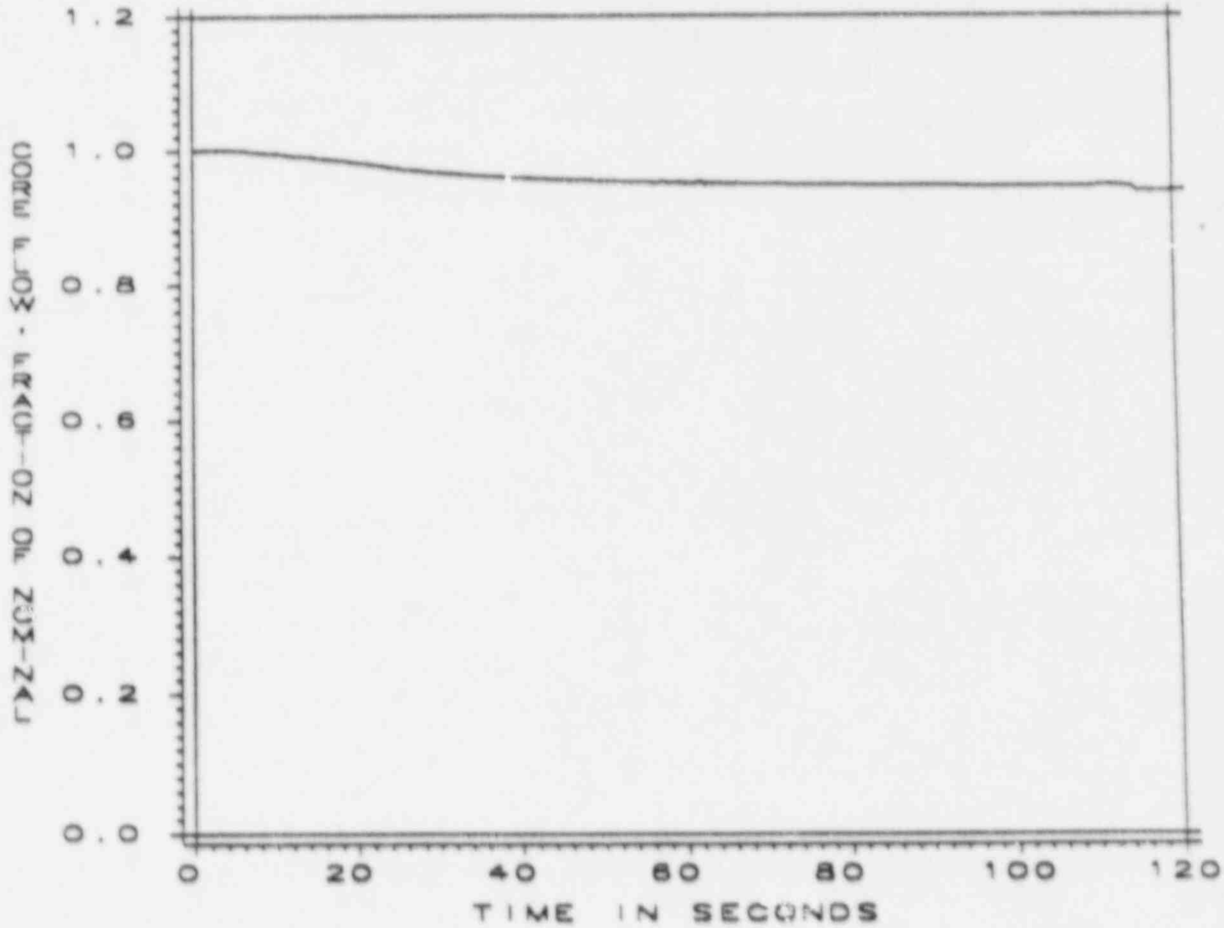


FIGURE 4.1.4

NA TURBINE TRIP/LOSS OF LOAD

CORE FLOW, FRACTION OF NOMINAL

MINIMUM FEEDBACK, PRESSURIZER CONTROL



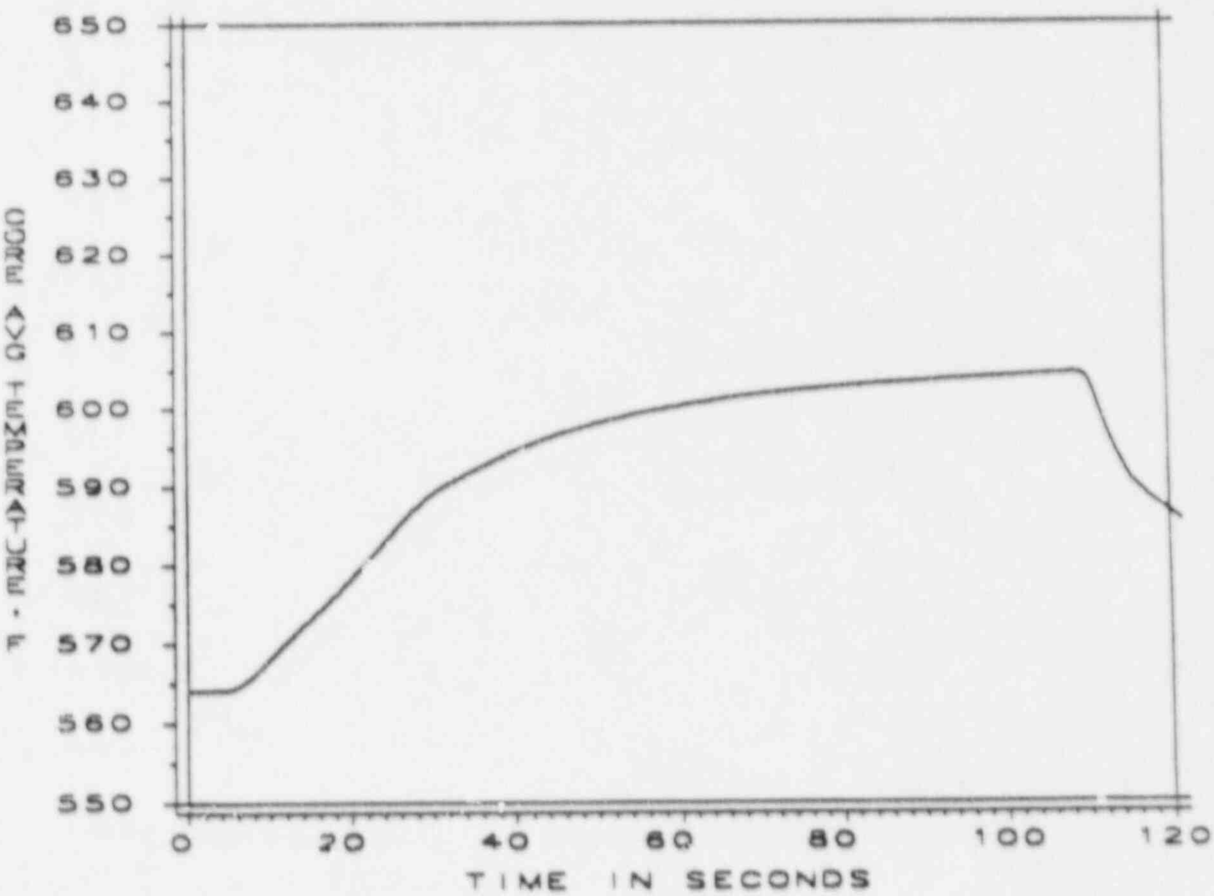
MINIMUM FEEDBACK

FIGURE 4.1.5

NA TURBINE TRIP/LOSS OF LOAD

CORE AVG TEMPERATURE, F

MINIMUM FEEDBACK, PRESSURIZER CONTROL



MINIMUM FEEDBACK

FIGURE 4.1.8

NA TURBINE TRIP/LOSS OF LOAD

INLET TEMPERATURE,F

MINIMUM FEEDBACK, PRESSURIZER CONTROL

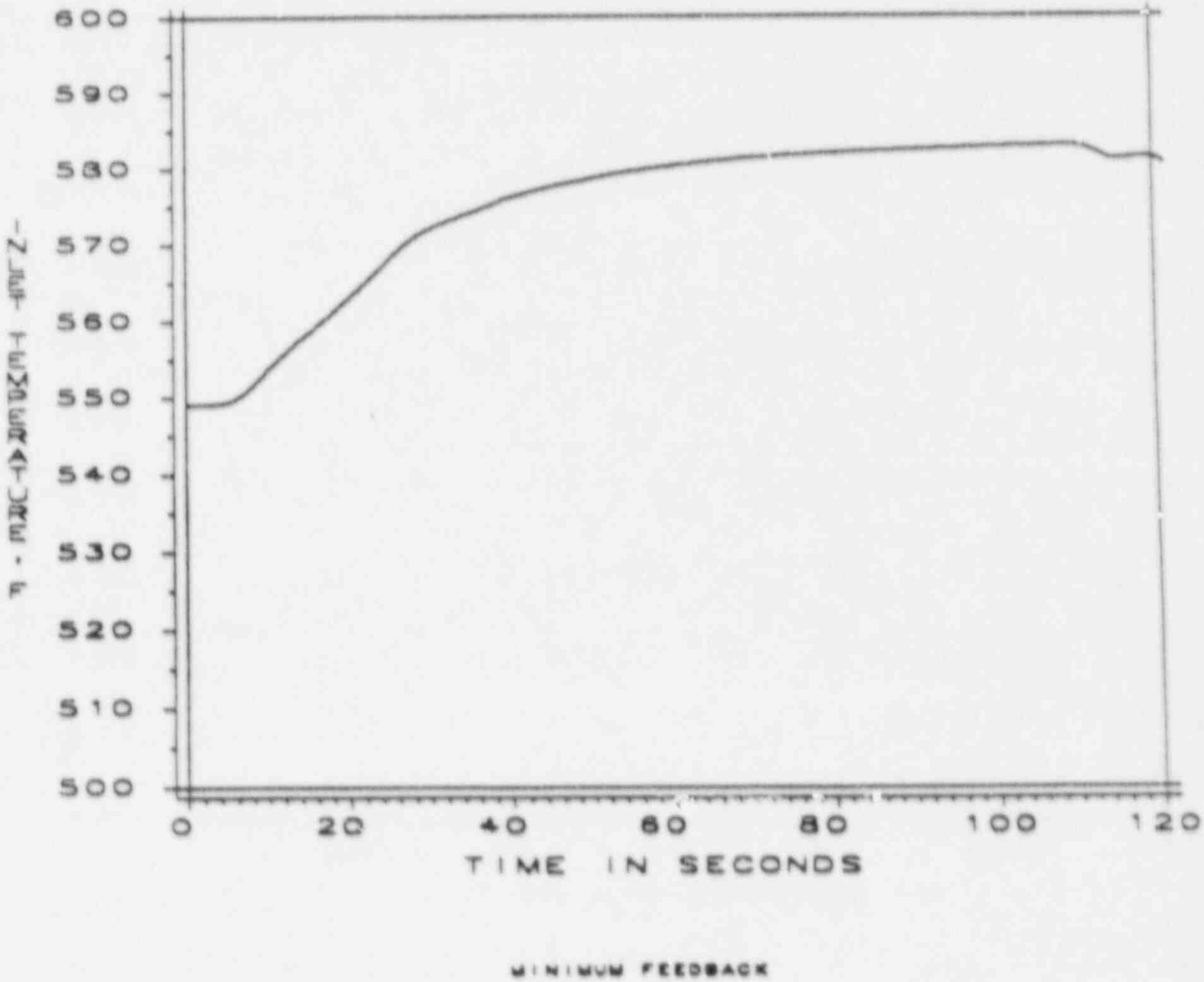
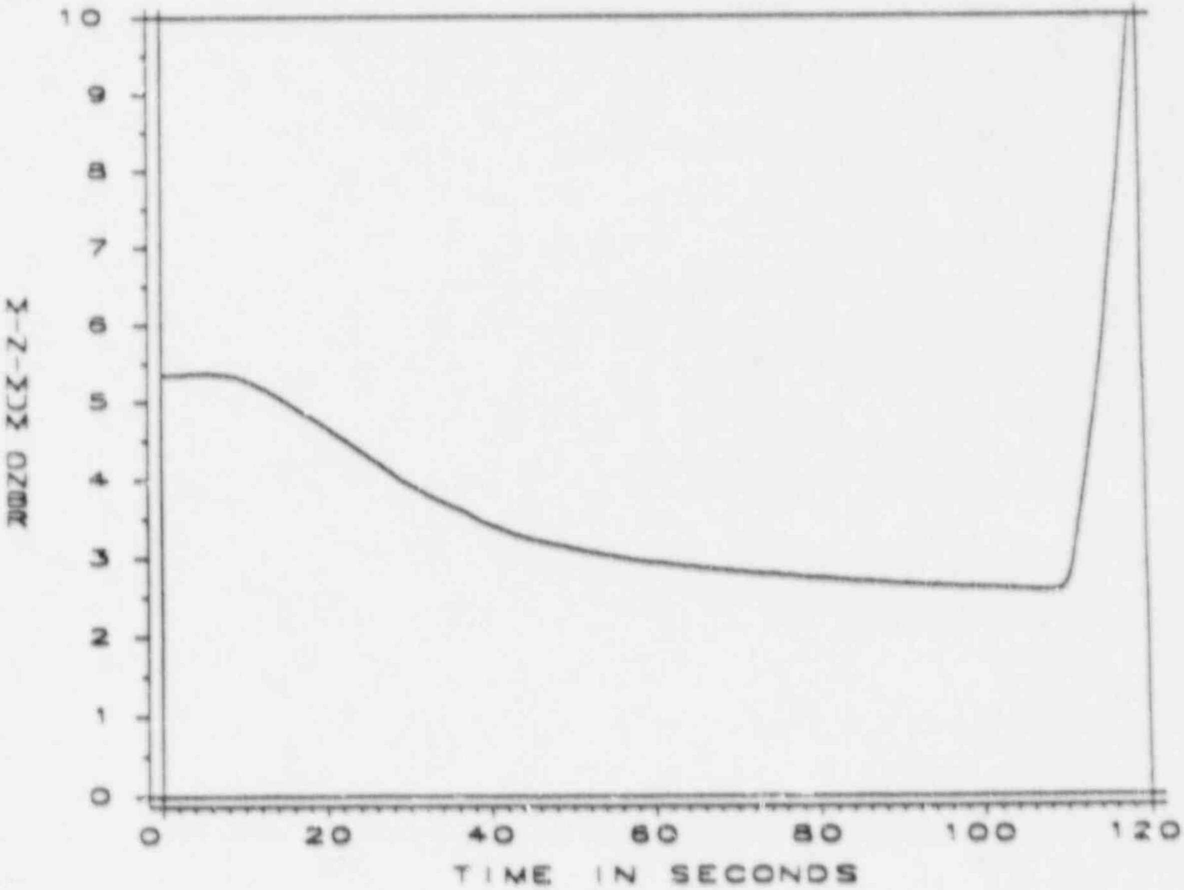


FIGURE 4.1.7

NA TURBINE TRIP/LOSS OF LOAD

MINIMUM DNBR

MINIMUM FEEDBACK, PRESSURIZER CONTROL



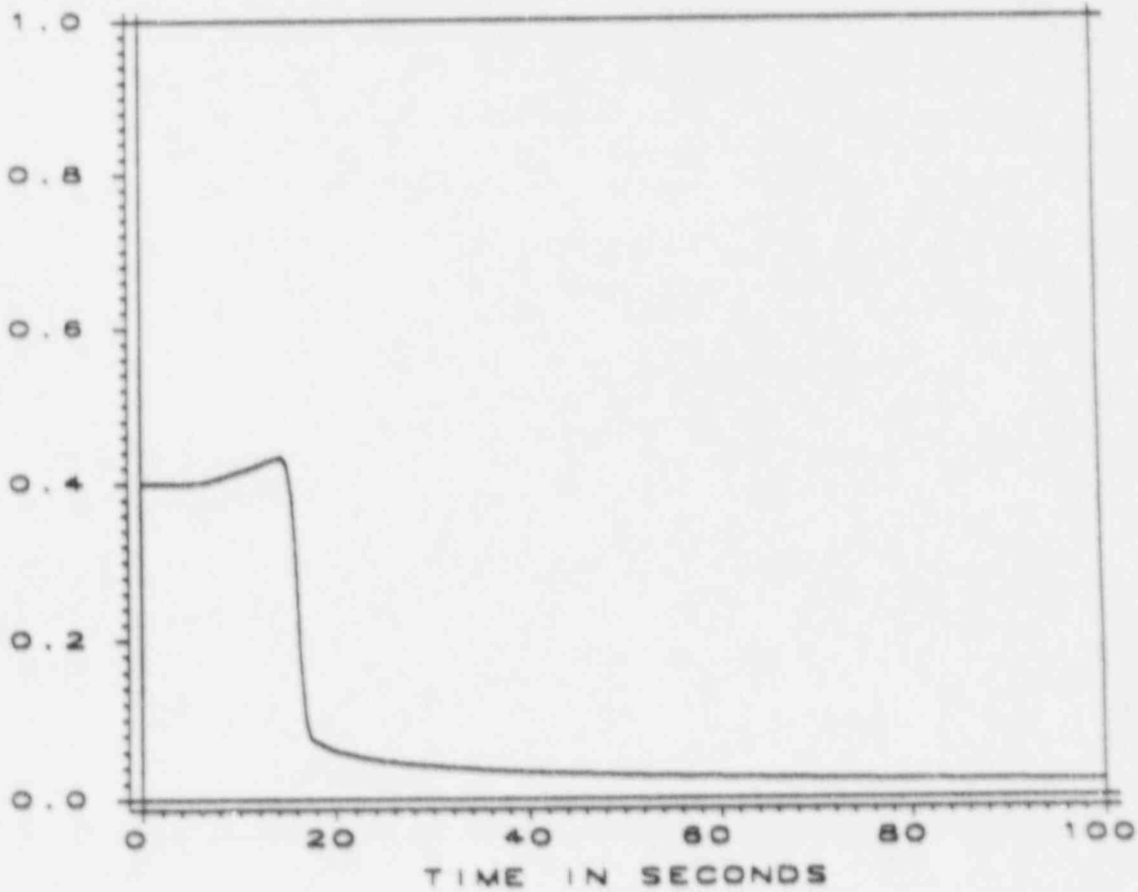
MINIMUM FEEDBACK

FIGURE 4.1.8

NA TURBINE TRIP/LOSS OF LOAD

NUCLEAR POWER, FRACTION OF NOMINAL

MINIMUM FEEDBACK, NO PRESSURIZER CONTROL



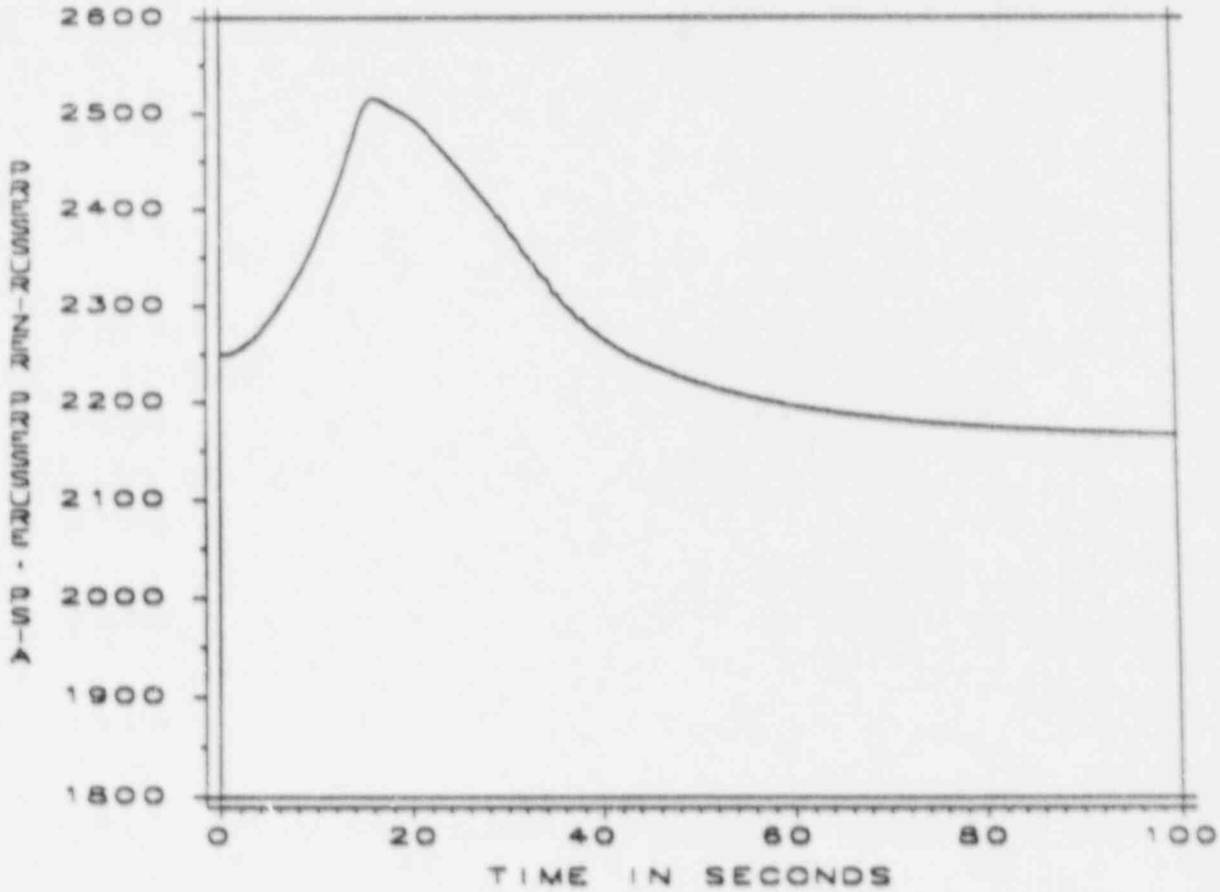
MINIMUM FEEDBACK

FIGURE 4.1.9

NA TURBINE TRIP/LOSS OF LOAD

PRESSURIZER PRESSURE, PSIA

MINIMUM FEEDBACK, NO PRESSURIZER CONTROL



MINIMUM FEEDBACK

FIGURE 4.1.10
NA TURBINE TRIP/LOSS OF LOAD
PRESSURIZER WATER VOLUME, FT3
MINIMUM FEEDBACK, NO PRESSURIZER CONTROL

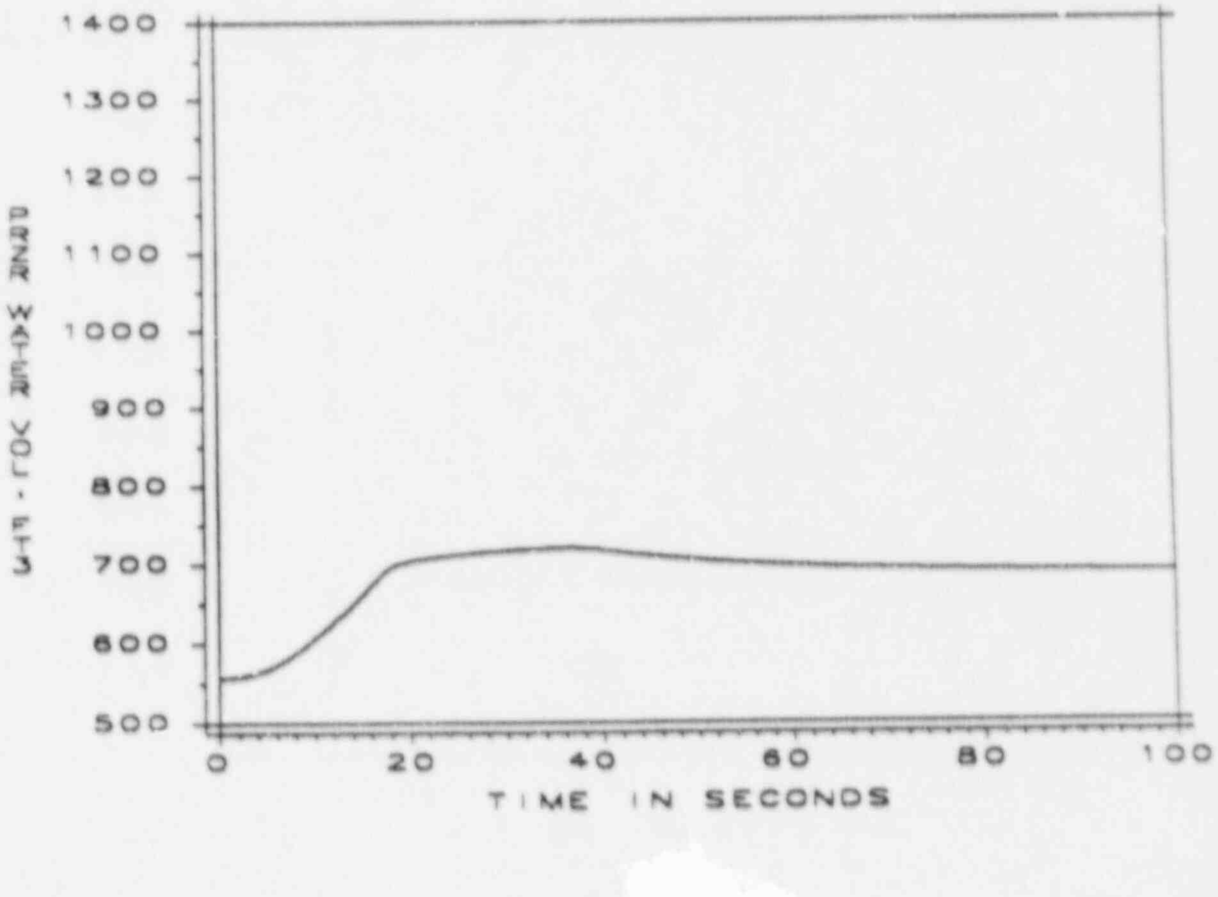


FIGURE 4.1.11

NA TURBINE TRIP/LOSS OF LOAD

CORE FLOW, FRACTION OF NOMINAL

MINIMUM FEEDBACK, NO PRESSURIZER CONTROL

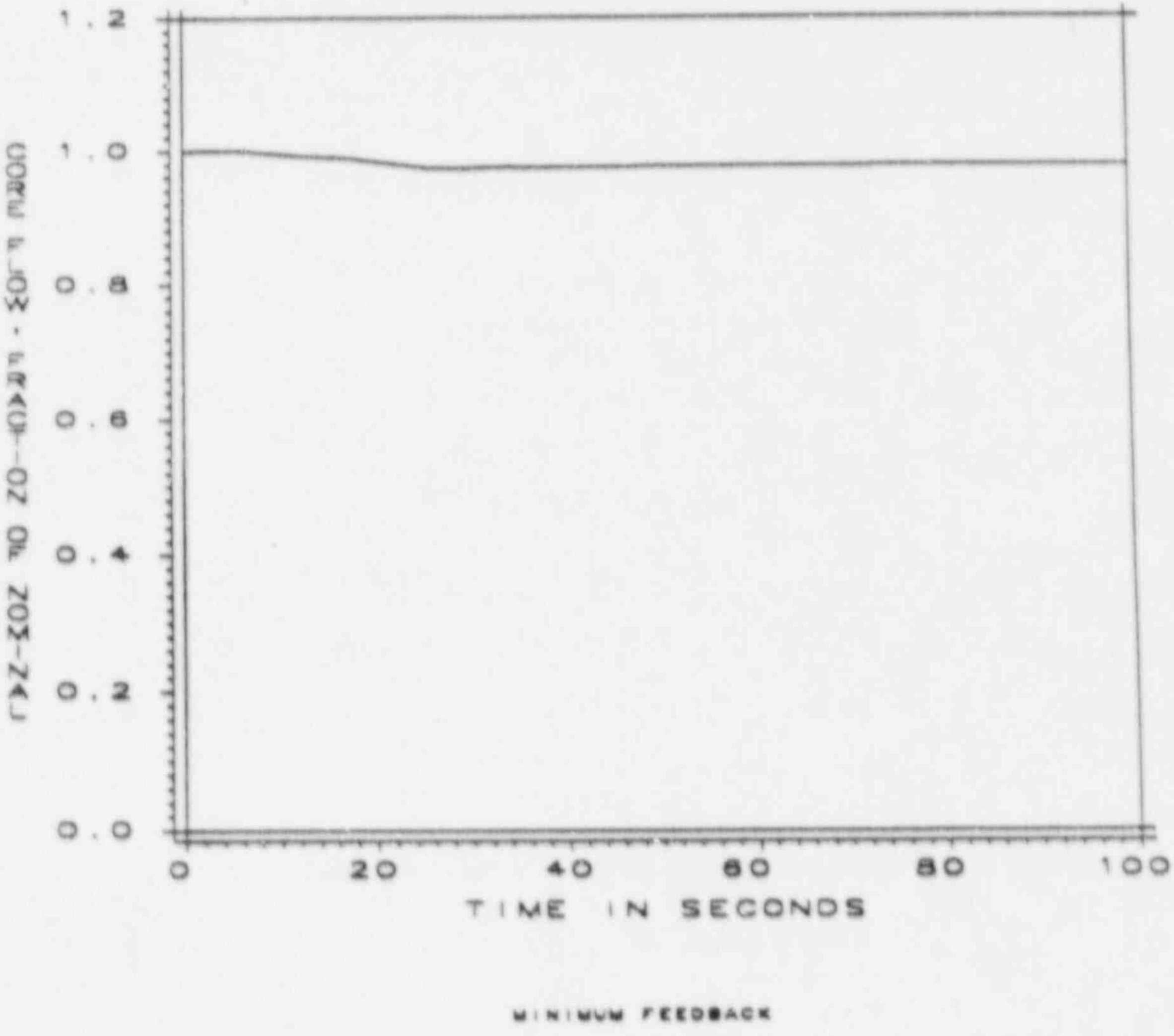


FIGURE 4.1.12
NA TURBINE TRIP/LOSS OF LOAD
CORE AVG TEMPERATURE, F
MINIMUM FEEDBACK, NO PRESSURIZER CONTROL

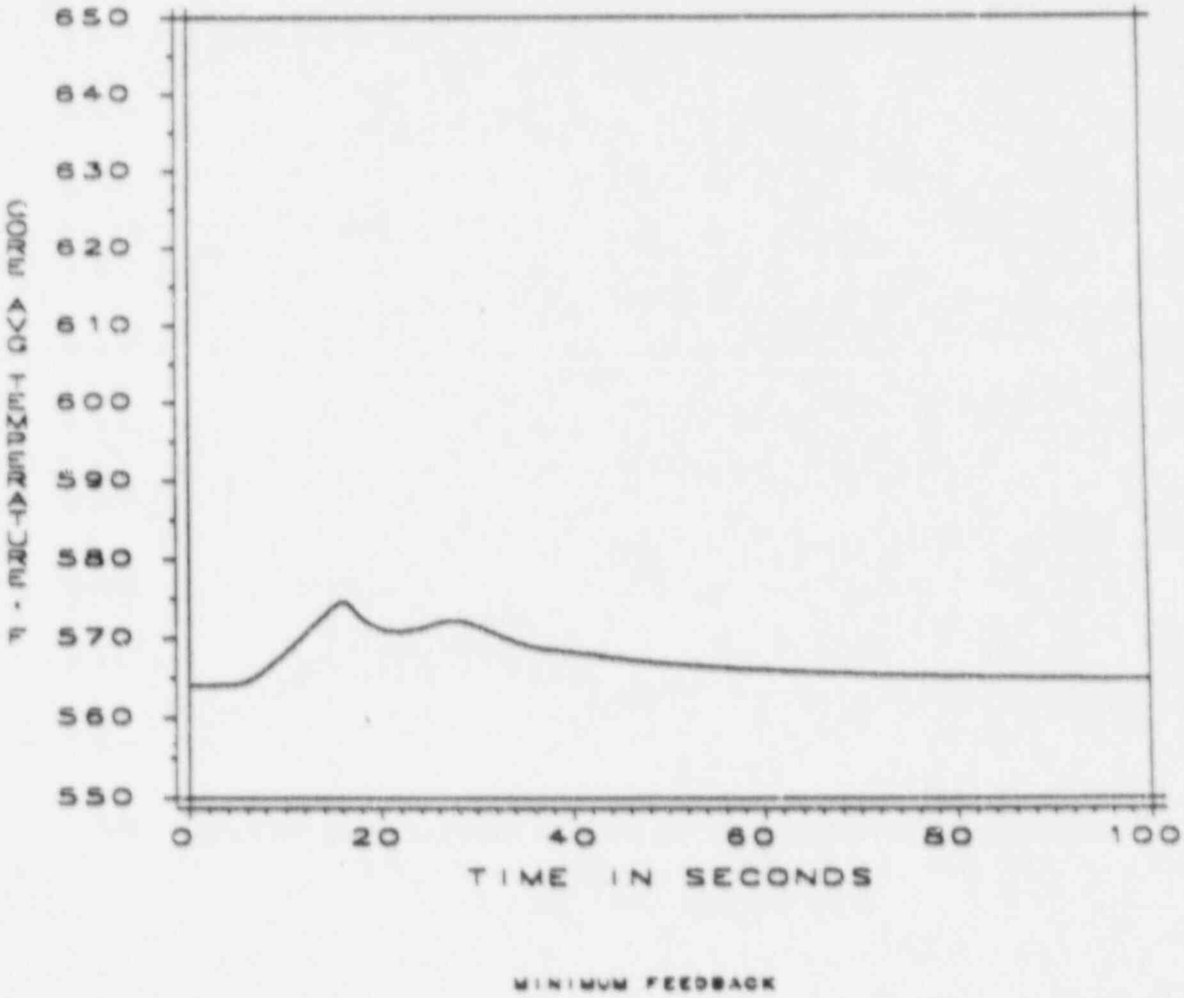
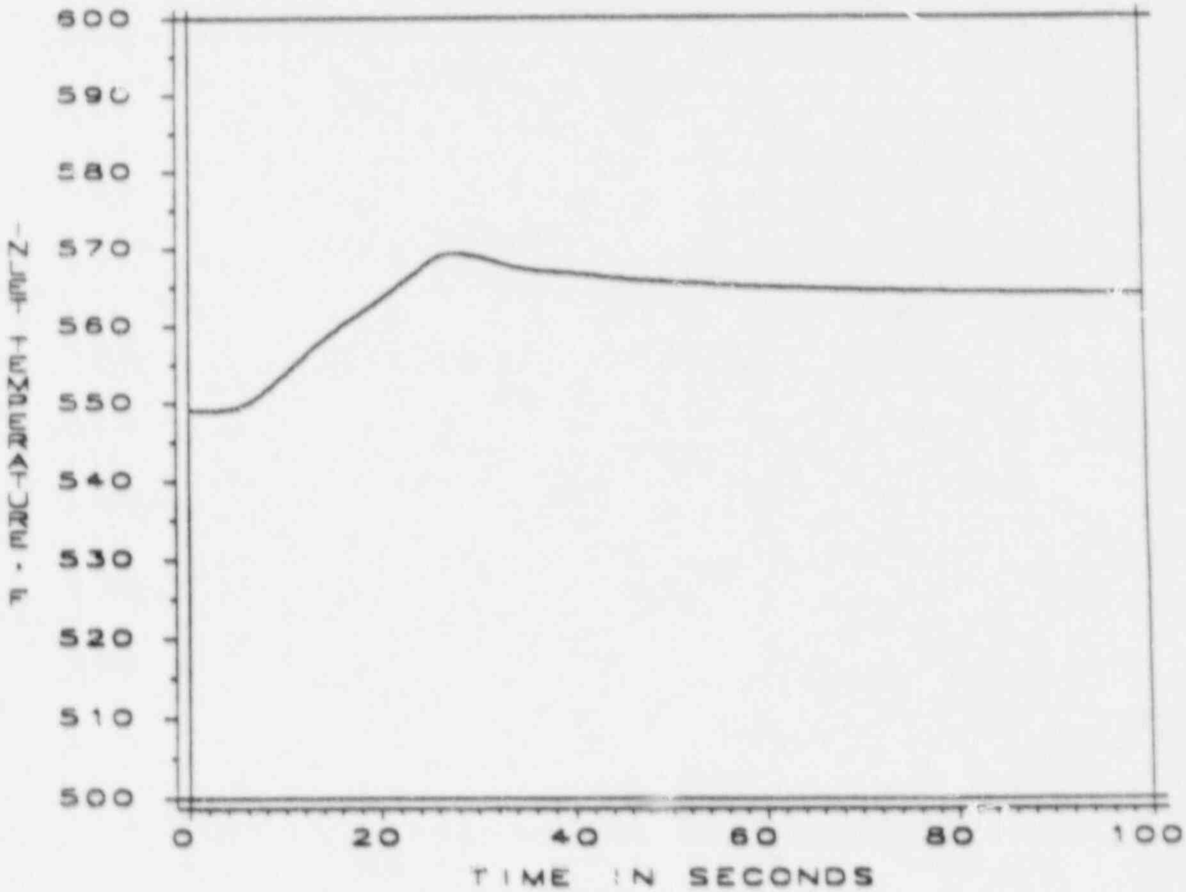
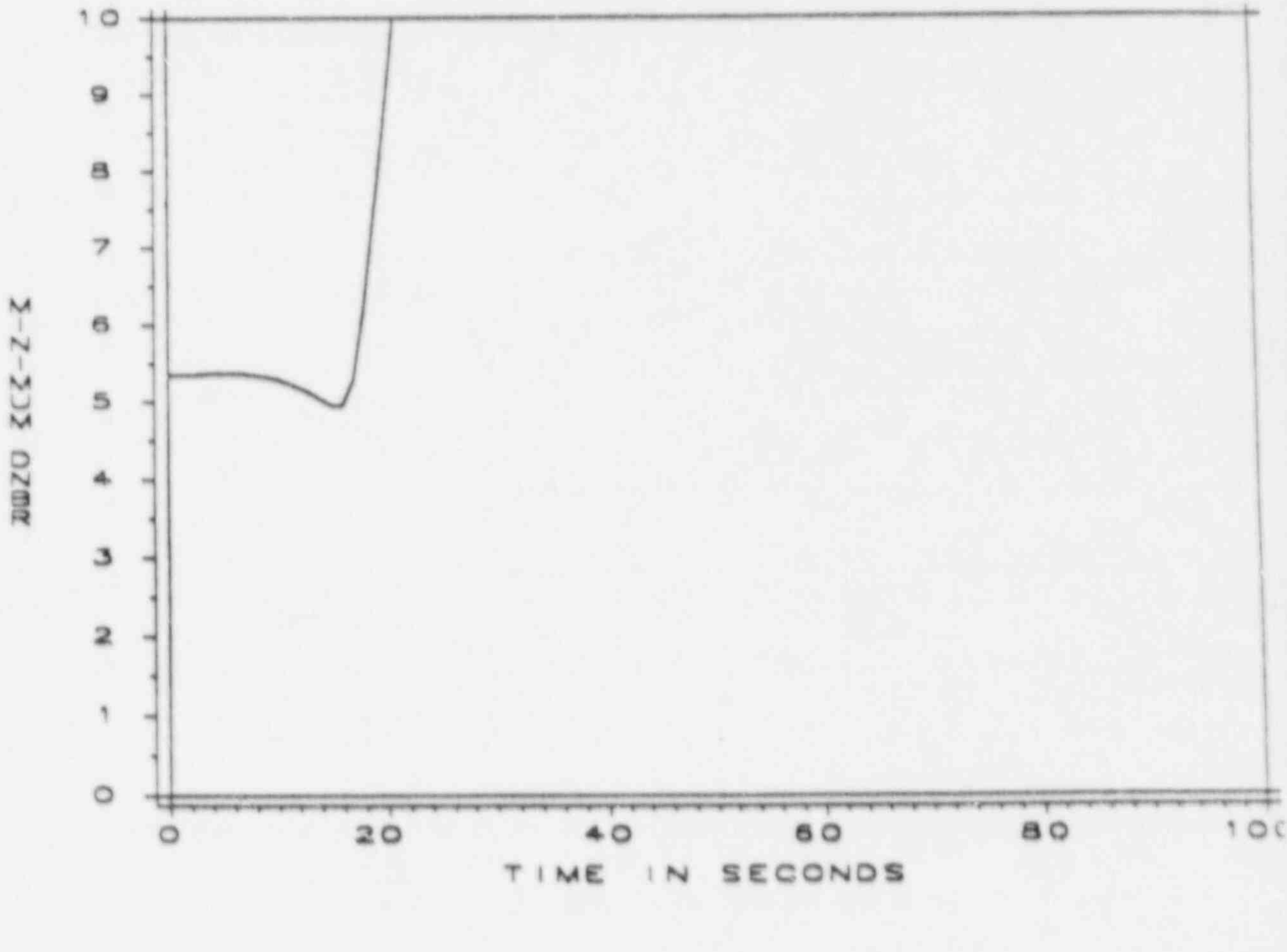


FIGURE 4.1.13
NA TURBINE TRIP/LOSS OF LOAD
INLET TEMPERATURE,F
MINIMUM FEEDBACK, NO PRESSURIZER CONTROL



MINIMUM FEEDBACK

FIGURE 4.1.14
NA TURBINE TRIP/LOSS OF LOAD
MINIMUM DNBR
MINIMUM FEEDBACK, NO PRESSURIZER CONTROL



4.2 TURBINE TRIP/LOSS OF LOAD WITH LOSS OF FLOW

The transient responses for a total loss of load with subsequent loss of flow from 40% power level were analyzed in detail: two cases for the beginning of core life and two cases for the end of core life.

The case with minimum reactivity feedback and with pressurizer control is most limiting with respect to DNBR. The results of that case are presented here.

Figures 4.2.1 through 4.2.6 show the transient response. For this case, full credit was taken for the pressurizer spray and pressurizer power-operated relief valves. No credit was taken for the condenser steam dump and steam generator PORVs. It was also assumed that a fast bus transfer fails and a loss of flow event is initiated 30 seconds after turbine trip. The minimum DNBR remains well above the limit value as shown in Figure 4.2.7 and is bounded by the loss of load analysis delineated in UFSAR Section 15.2.7.

The time sequence of events is shown in Table 4.2.1.

Table 4.2.1 Time Sequence of Events for Turbine Trip with Pressurizer Control

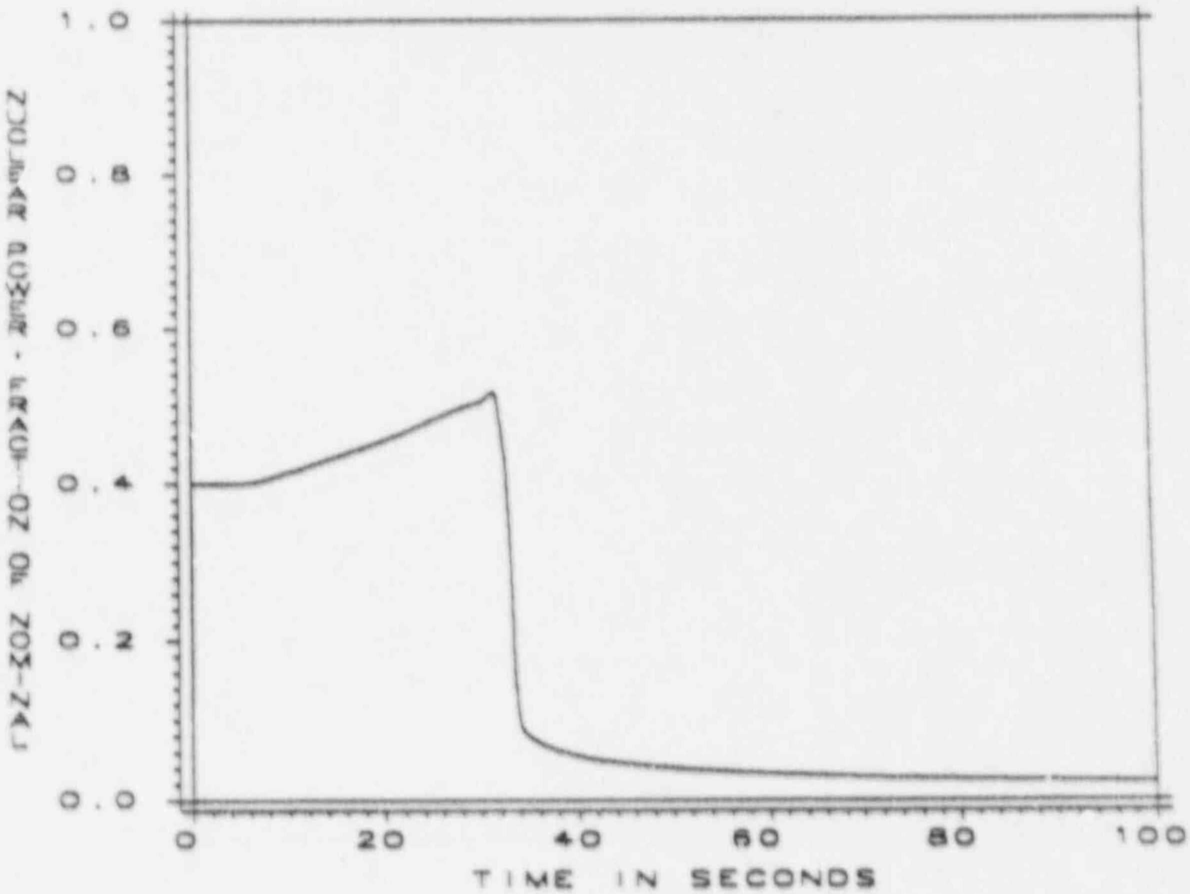
	EVENT	TIME(s)
Minimum Feedback (BOC)	Turbine Trip	0.0
	Peak pressurizer pressure occurs	11.0
	Initiation of steam release from steam generator safety valves	21.1
	Fast bus transfer failure, flow coastdown begins	30.0
	Undervoltage sensor reactor trip	30.025
	Rod begins to fall	31.225
	Minimum DNBR occurs	33.0

FIGURE 4.2.1

NA TURBINE TRIP/LOSS OF FLOW

NUCLEAR POWER, FRACTION OF NOMINAL

MINIMUM FEEDBACK, PRESSURIZER CONTROL



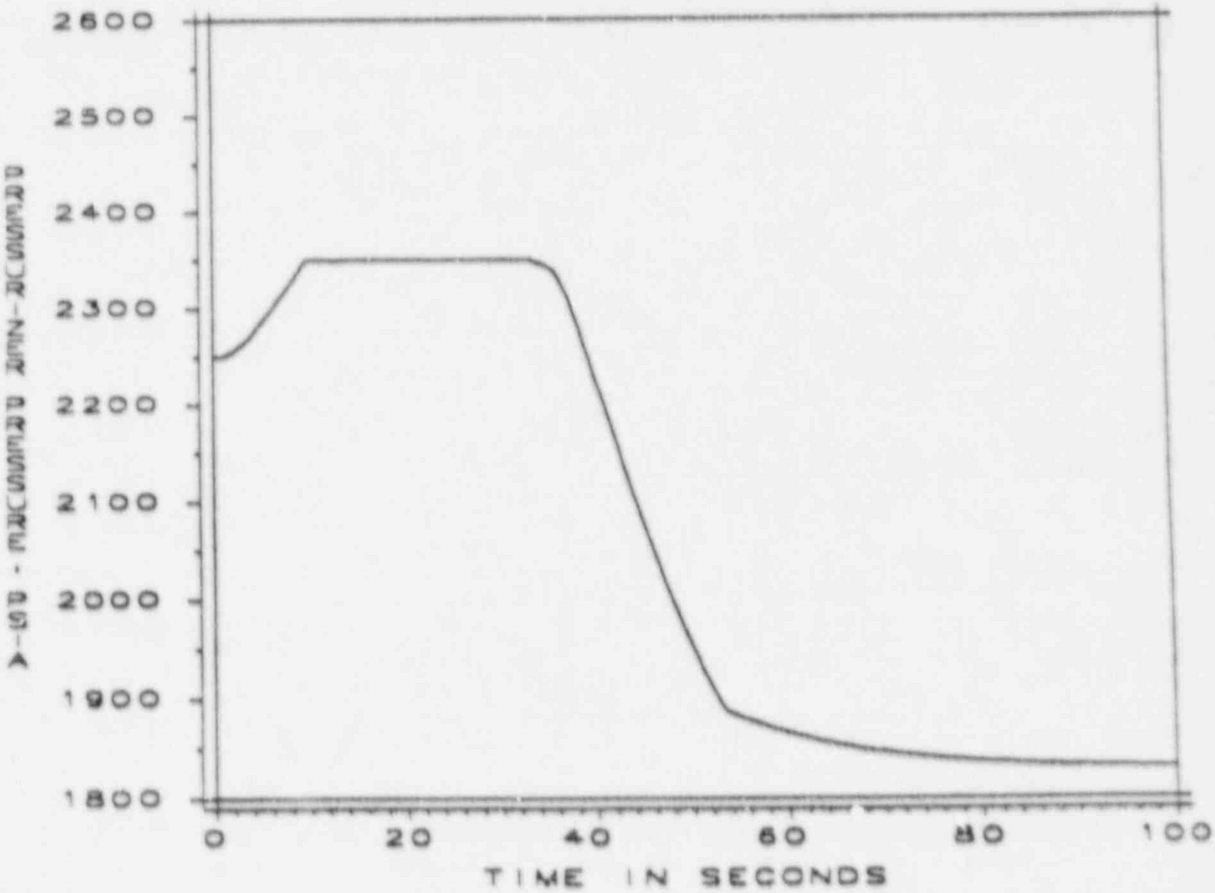
MINIMUM FEEDBACK

FIGURE 4.2.2

NA TURBINE TRIP/LOSS OF FLOW

PRESSURIZER PRESSURE, PSIA

MINIMUM FEEDBACK, PRESSURIZER CONTROL



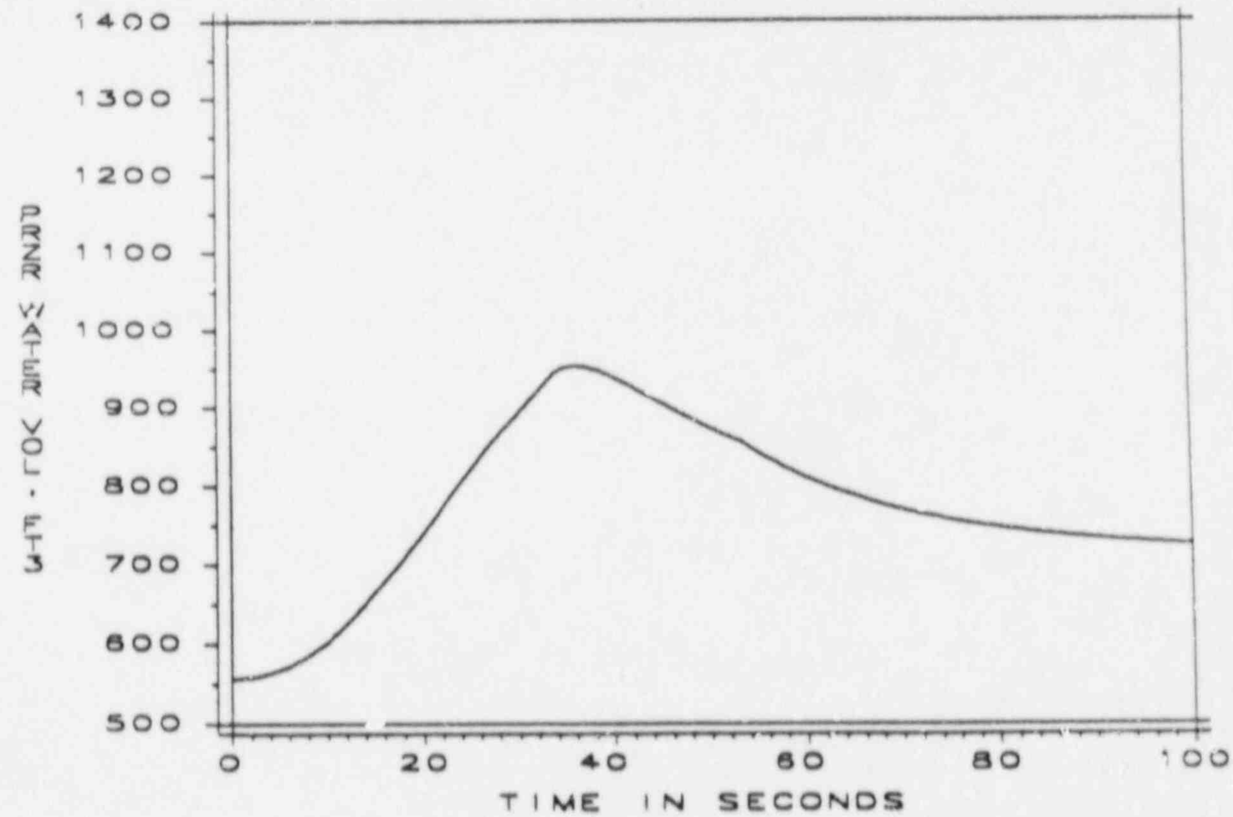
MINIMUM FEEDBACK

FIGURE 4.2.3

NA TURBINE TRIP/LOSS OF FLOW

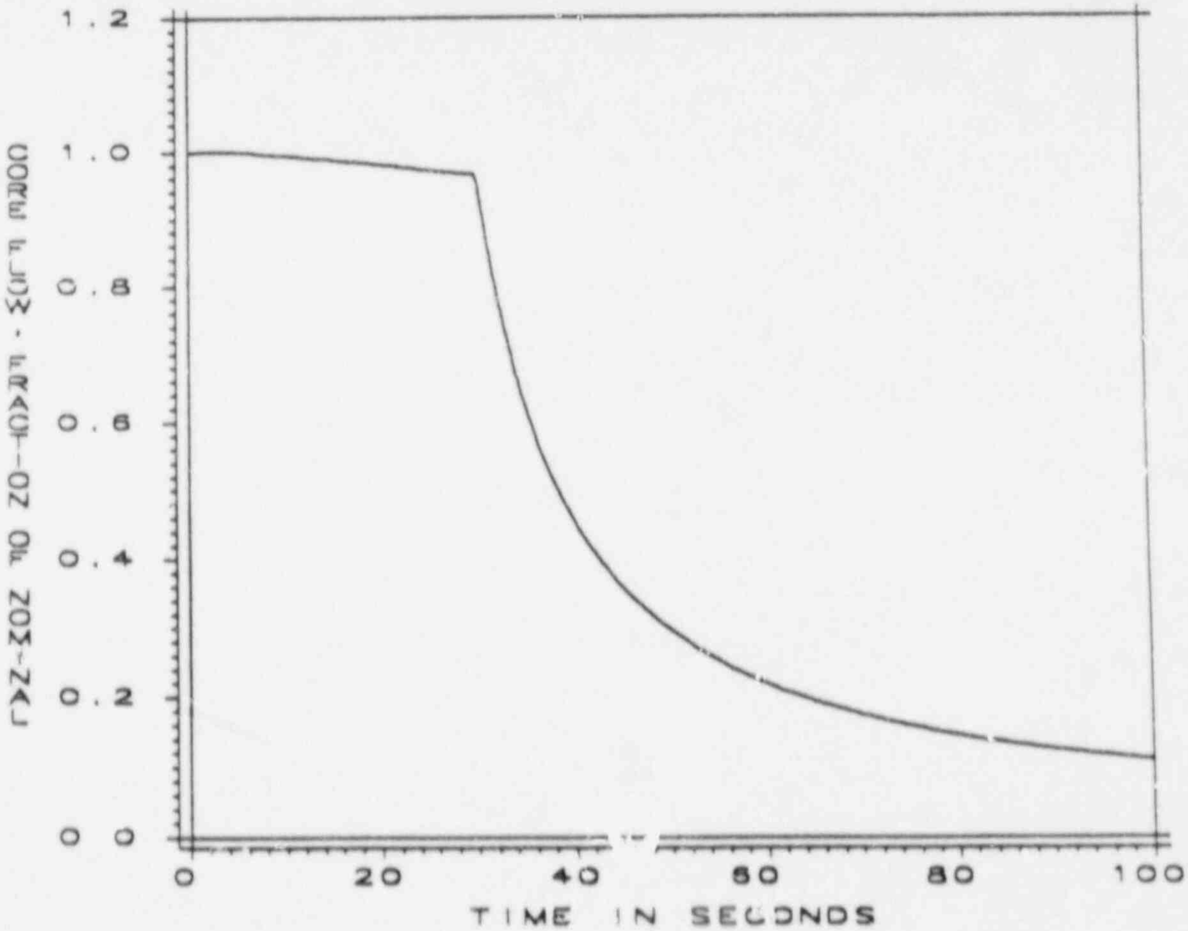
PRESSURIZER WATER VOLUME, FT3

MINIMUM FEEDBACK, PRESSURIZER CONTROL



MINIMUM FEEDBACK

FIGURE 4.2.4
NA TURBINE TRIP/LOSS OF FLOW
CORE FLOW, FRACTION OF NOMINAL
MINIMUM FEEDBACK, PRESSURIZER CONTROL



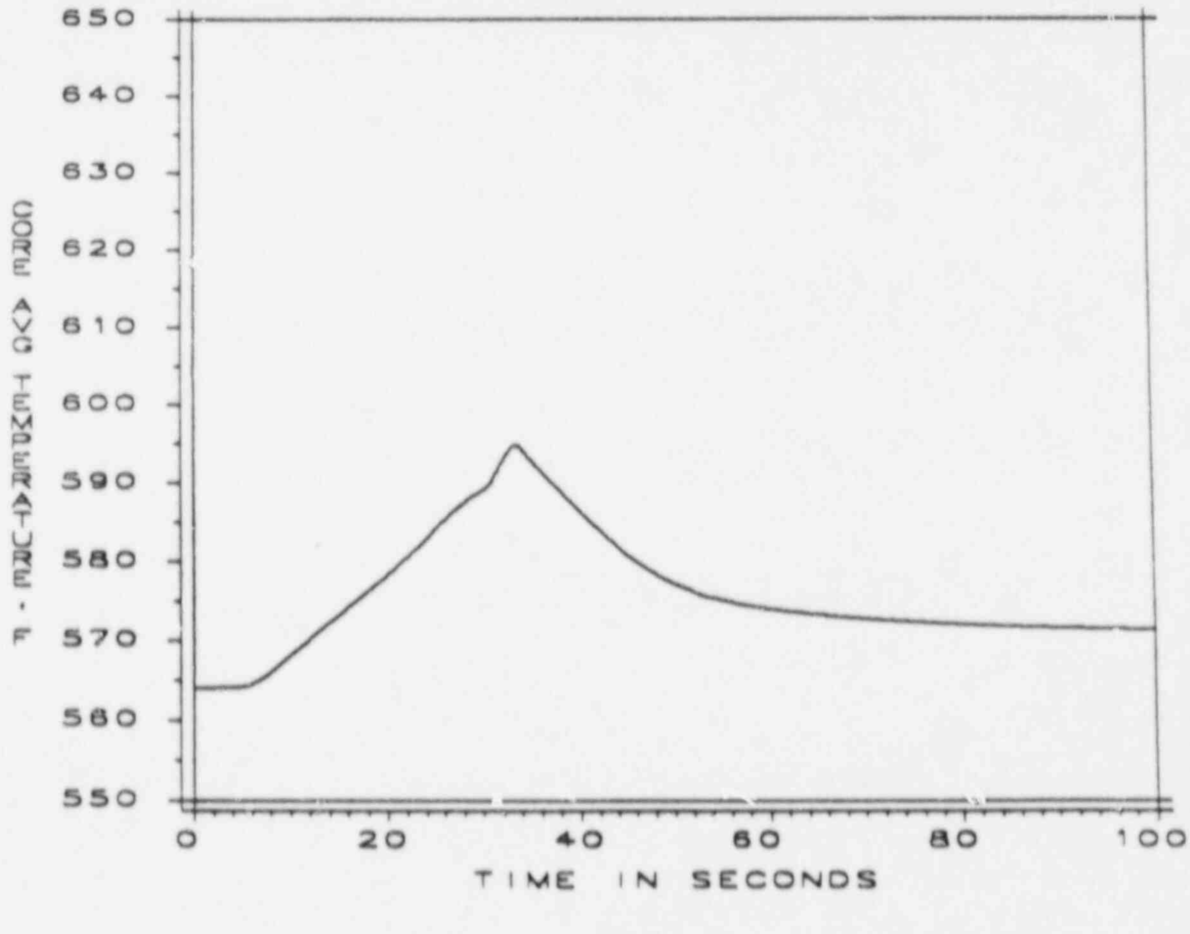
MINIMUM FEEDBACK

FIGURE 4.2.5

NA TURBINE TRIP/LOSS OF FLOW

CORE AVG TEMPERATURE, F

MINIMUM FEEDBACK, PRESSURIZER CONTROL



MINIMUM FEEDBACK

FIGURE 4.2.8
NA TURBINE TRIP/LOSS OF FLOW
INLET TEMPERATURE, F
MINIMUM FEEDBACK, PRESSURIZER CONTROL

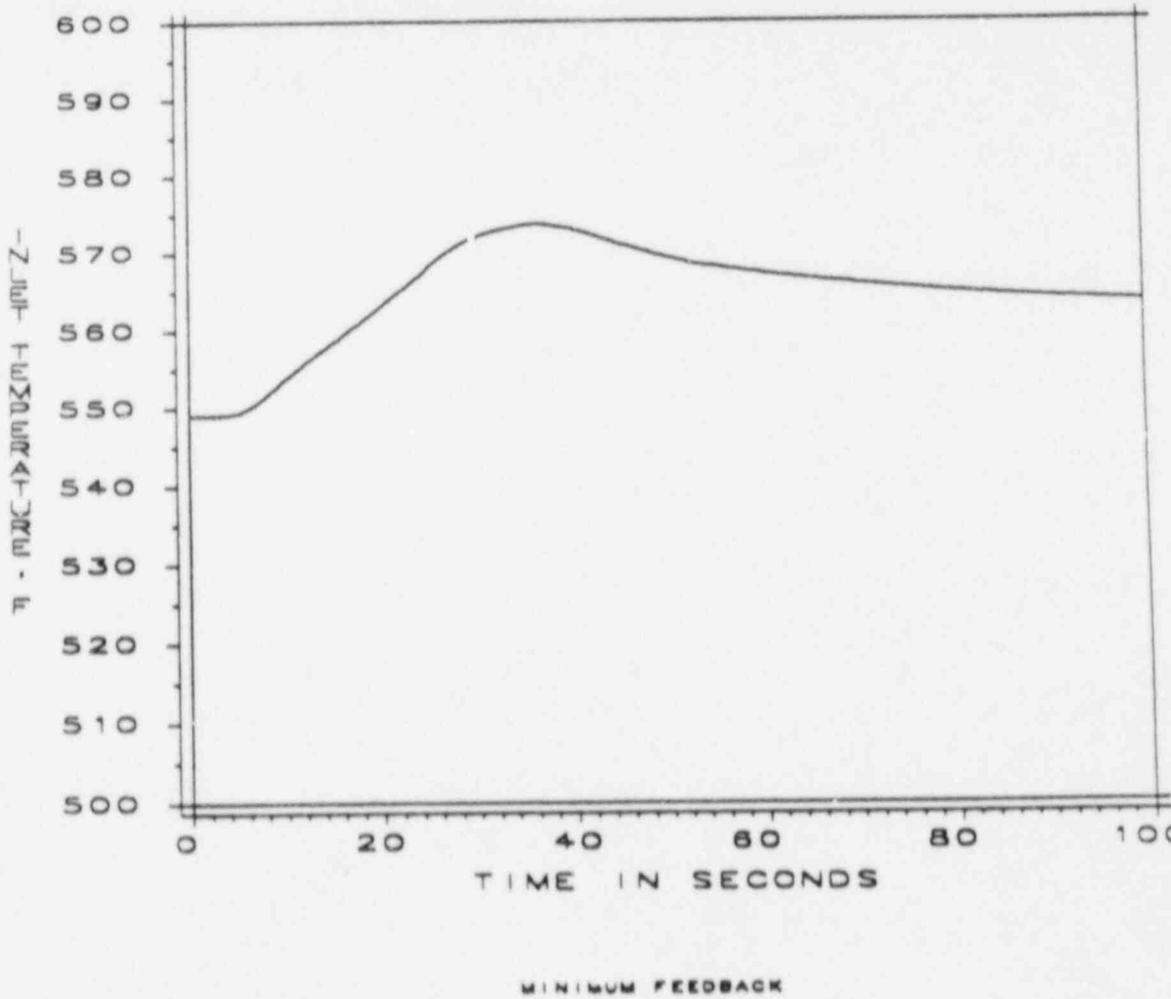
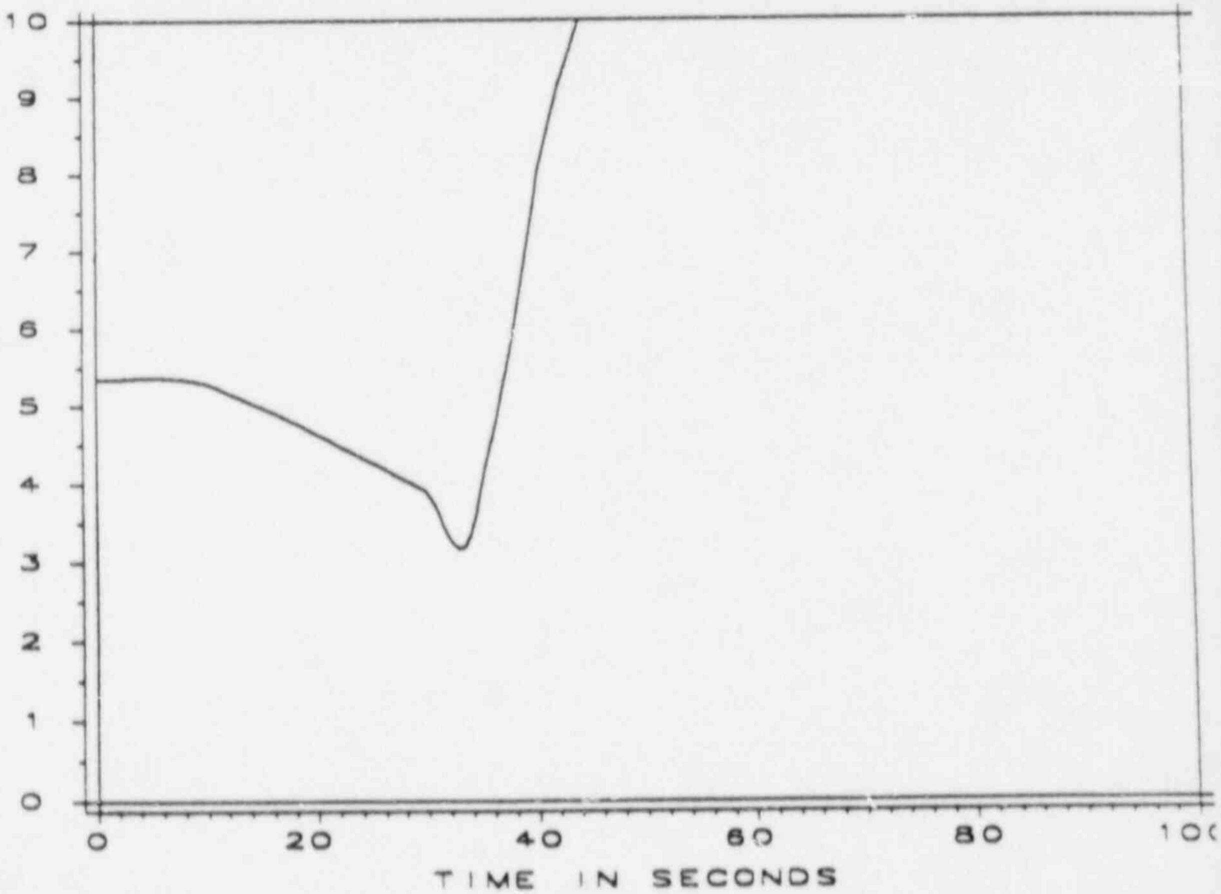


FIGURE 4.2.7

NA TURBINE TRIP/LOSS OF FLOW

MINIMUM DNBR

MINIMUM FEEDBACK, PRESSURIZER CONTROL



MINIMUM FEEDBACK

4.3 PRESSURIZER PORV RESPONSE EVALUATION

The transient was analyzed to confirm that on a best estimate basis, the pressurizer PORVs are not challenged during this transient. Figures 4.3.1 through 4.3.3 shows the results of this transient. The time sequence of events during the transient is given in Table 4.3.1.

Table 4.3.1 Time Sequence of Events for a Turbine Trip with Pressurizer Control

	EVENT	TIME(s)
Minimum Feedback (BOL) Credit is taken for the condenser steam dump and steam generator PORVs		
	Turbine Trip	0.0
	Peak pressurizer pressure occurs (2331.68 psia)	22.7
	Lo Steam Generator mass trip (simulates low-low level)	150.4
	Rod begins to fall	152.4

FIGURE 4.3.1
NA TURBINE TRIP/ PORV
NUCLEAR POWER, FRACTION OF NOMINAL
MINIMUM FEEDBACK, PRESSURIZER CONTROL

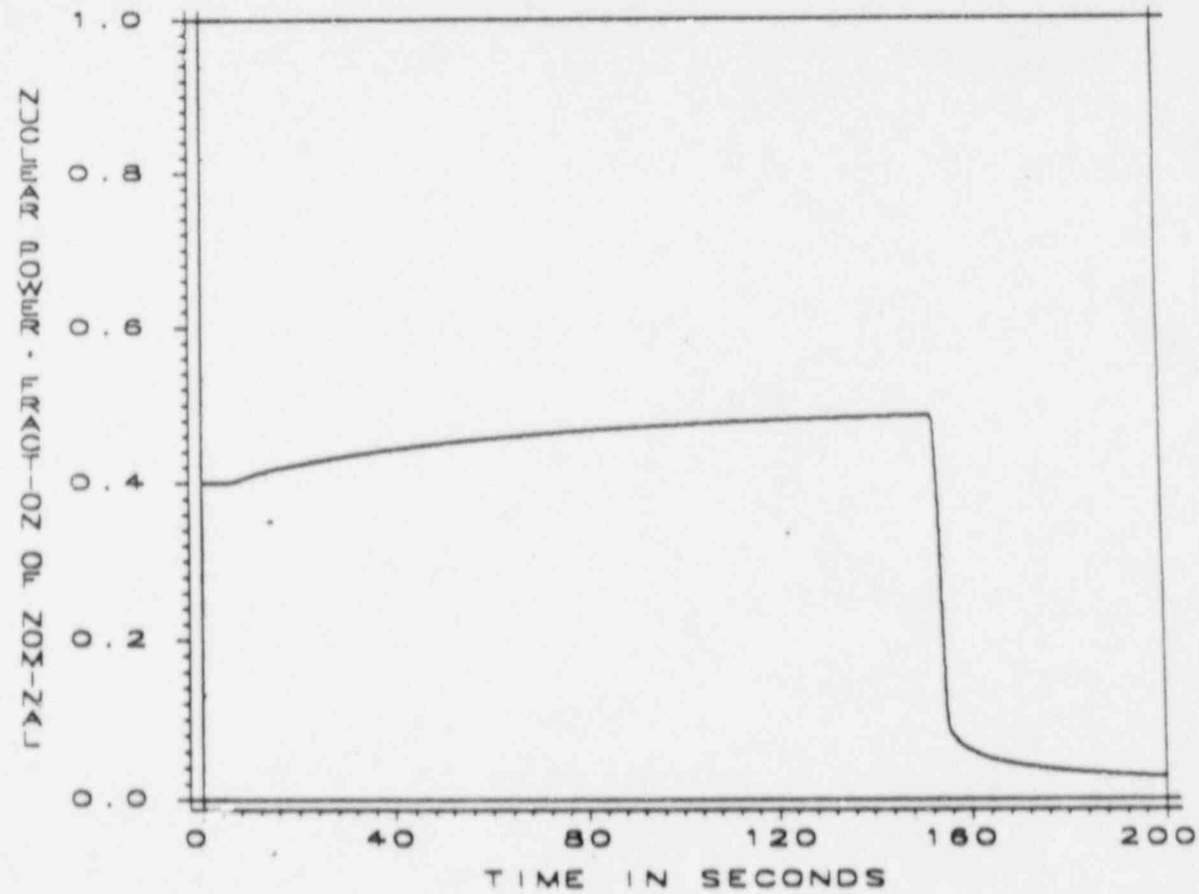
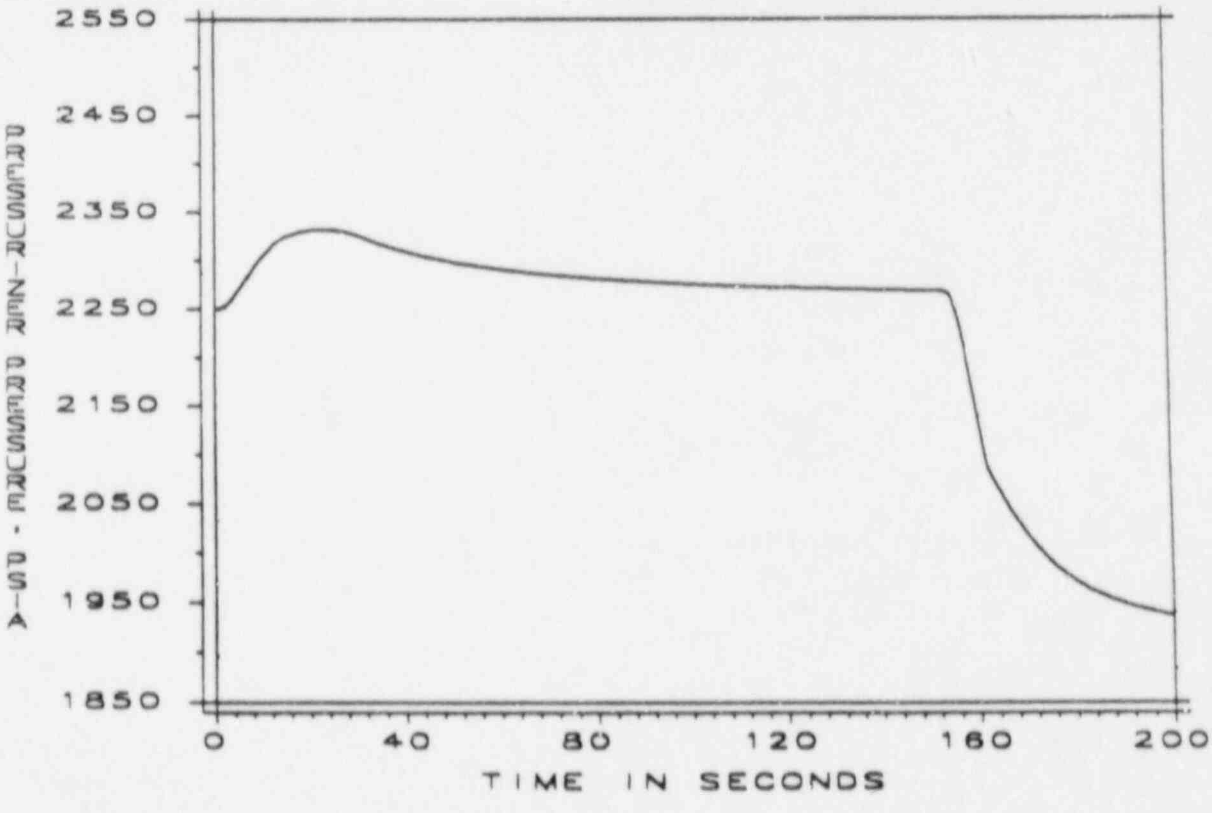
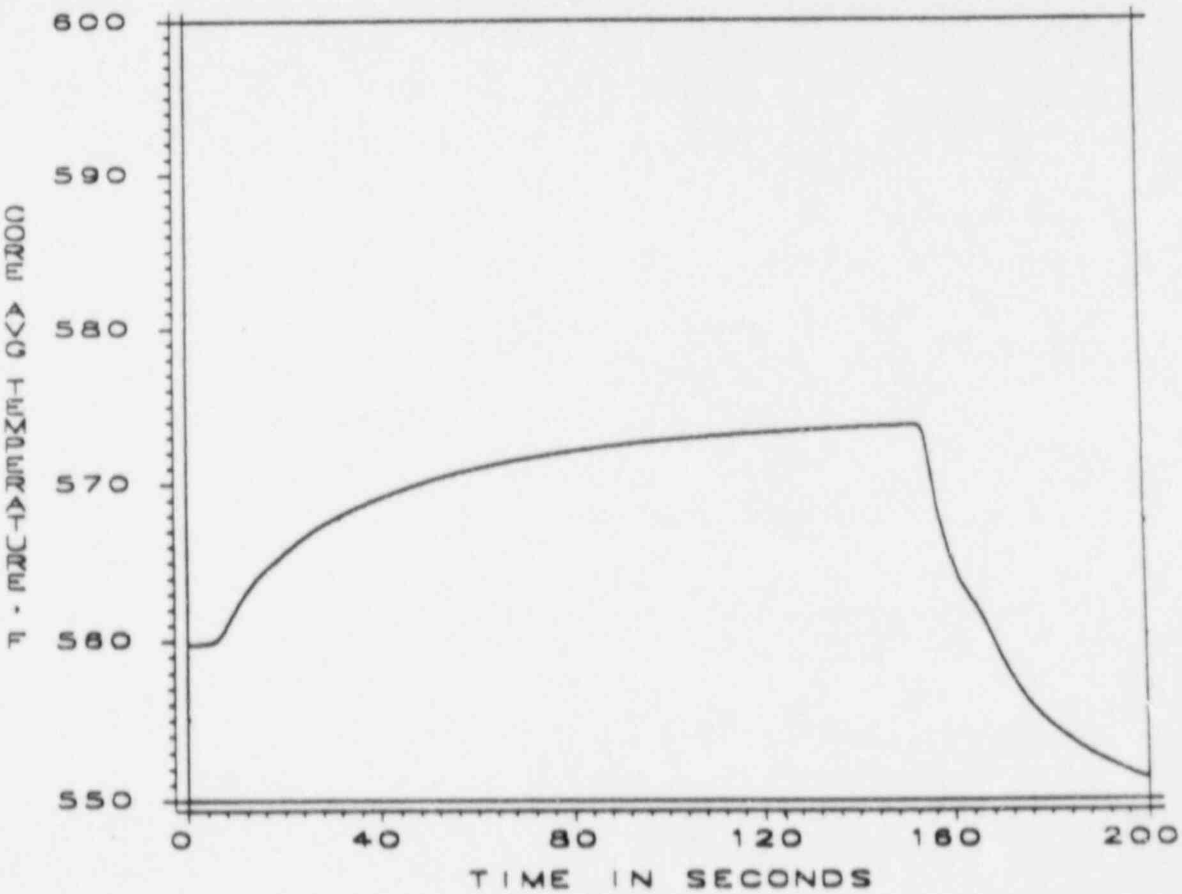


FIGURE 4.3.2
NA TURBINE TRIP/PORV
PRESSURIZER PRESSURE, PSIA
MINIMUM FEEDBACK, PRESSURIZER CONTROL



MINIMUM FEEDBACK

FIGURE 4.3.3
NA TURBINE TRIP/PORV
CORE AVG TEMPERATURE, F
MINIMUM FEEDBACK, PRESSURIZER CONTROL



MINIMUM FEEDBACK

5.0 CONCLUSIONS

Results of the analyses show that a total loss of external electrical load without a direct or immediate reactor trip below 30% of Rated Thermal Power presents no hazards to the integrity of the reactor coolant system or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to keep the maximum pressure within the design limits.

The analysis demonstrates that for a complete loss of forced reactor coolant flow initiated from the most adverse preconditions of a turbine trip, the integrity of the core is maintained by operation of the reactor protection system, i.e., the DNBR will be maintained above the design limit value. Thus, there will be no cladding damage and no release of fission products to the reactor coolant system.

The analysis also demonstrates that on a better estimate basis, the pressurizer PORVs are not challenged at any time during the transient, although conservative assumptions were used to demonstrate the pressurizer response. The proposed changes will not have a significant impact on the frequency of a small break LOCA caused by a stuck-open PORV.

5.1 10 CFR 50.59 EVALUATION

From the analyses presented in this report, it is concluded that the results of this analyses, i.e., minimum DNBR and peak pressurizer pressure are bounded by the design limit values and consequently that no unreviewed safety questions as defined in 10CFR50.59 exists as a result of the proposed change. The results of this evaluation can be stated as follows:

1. No increase in the probability of occurrence or consequences of an accident analyzed in the UFSAR will result from elimination of reactor trip on turbine trip below 30% of Rated Thermal Power (RTP). The analyses results shows that the DNBR does not decrease below the design limit at any time. The analysis also shows that, except under conservative assumptions, the pressurizer PORVs are not challenged during the transient.Pressure relieving devices incorporated in the primary and the secondary systems are adequate to keep the maximum pressure within the design limit. Since the predicted results are within the range of existing safety analysis values, it is concluded that operation with the proposed Technical Specification changes will neither increase the probability of occurrence nor the consequences of initiating events for any known accident.
2. The possibility of a new or different accident type not previously considered in the UFSAR is not created by this proposed change. The complete loss of unit load without a direct reactor trip on turbine trip is a design event and is addressed in Section 15.2.7 of the UFSAR. The results for a loss of flow due to fast bus transfer failure after a turbine trip are bounded by the results for a complete loss of flow from full power, which is discussed in Section 15.3.4 of the UFSAR. Thus, the results of all the relevant accident analyses show that operation with this modification does not create the possibility of an accident of a different type than any evaluated previously in the UFSAR.
3. The margin of safety is not reduced. The proposed Technical Specifications changes have been incorporated in the safety analyses. These analyses have demonstrated that calculated results meet all design acceptance criteria as stated in the UFSAR.

6.0 REFERENCES

1. Letter from B. R. Sylvia (Vepco) to H. R. Denton (NRC),"Response to NUREG-0737 TMI Action Items", December 15, 1980.

2. WCAP-9804,"Probabilistic Analysis and Operational Data in Response to NUREG-0737, Item II.K.3.2 for Westinghouse NSSS Plants", Feb. 1981.

3. Smith, N. A.,"VEPCO Reactor System Transient Analysis using the RETRAN Computer Code", VEP-FRD-41A, May, 1985.

4. Sliz, F. W. and Basehore, K. L.,"VEPCO Reactor Core Thermal-Hydraulic Analysis Using the COBRA IIIC/MIT Computer Code", VEP-FRD-33-A, October, 1983.

5. Anderson, R. C. and Wolfhope, N. P.,"Qualification of WRB-1 CHF Correlation in the Virginia Power COBRA Code", VEP-NE-3, Nov., 1986.

6. Anderson,"Statistical DNBR Evaluation Methodology", VEP-NE-2A, June, 1987.

ATTACHMENT 4

10 CFR 50.92 Evaluation

10 CFR 50.92 EVALUATION

Virginia Power is proposing the following Technical Specification Changes which support blocking of reactor trip on turbine trip below 30% of rated thermal power. The following three items have been considered from a safety analysis standpoint and are addressed in this evaluation: The results of this evaluation can be stated as follows:

1. No significant increase in the probability of occurrence or consequences of an accident analyzed in the UFSAR will result from elimination of reactor trip on turbine trip below 30% of Rated Thermal Power (RTP). The analyses results show that the DNBR does not decrease below the design limit at any time. The analysis also shows that, except under conservative assumptions, the pressurizer PORVs are not challenged during the transient. Pressure relieving devices incorporated in the primary and the secondary systems are adequate to keep the maximum pressure within the design limit. Since the predicted results are within the range of existing safety analysis values, it is concluded that operation with the proposed Technical Specification changes will neither significantly increase the probability of occurrence nor the consequences of initiating events for any known accident.

2. No new or different accident type not previously considered in the UFSAR is created by this proposed change. The complete loss of unit load without a direct reactor trip on turbine trip is a design event and is addressed in Section 15.2.7 of the UFSAR. The results for a loss of flow due to fast bus transfer failure after a turbine trip are bounded by the results for a complete loss of flow from full power, which is discussed in Section 15.3.4 of the UFSAR. Thus, the results of all the relevant accident analyses show that operation with this modification does not create a new or different accident type than any evaluated previously in the UFSAR.

3. The margin of safety is not reduced. The proposed Technical Specification changes have been incorporated in the safety analyses. These analyses have demonstrated that calculated results meet all design acceptance criteria as stated in the UFSAR.