DELETION OF REACTOR TRIP ON TURBINE TRIP BELOW 50 PERCENT POWER

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1.0 INTRODUCTION

The present protection for a turbine trip automatically results in a reactor trip. However for plants with a 50 percent load rejection capability this trip is unnecessary if the cause of the turbine trip is readily correctable. Deletion of the reactor trip following turbine trip for these cases would significantly reduce the down time required to restart the plant. Thereby an increase in plant availability could be achieved.

Several safety considerations must be evaluated in order to implement a new system that eliminates reactor trip below 50 percent power. First the results of a loss-of-external-electrical-load transient initiated from 50 percent power must be shown to be acceptable. Second the results of a loss of reactor coolant flow occurring 30 seconds after a turbine trip must be shown to be acceptable. This analysis is required to demonstrate reactor safety should the fast bus transfer fail following the generator motoring delay on turbine trip.

Section 2 discusses the plant transient behavior following a loss of external electrical load without a subsequent turbine trip and following a loss of load resulting from a turbine trip.

In section 3 an analyses is presented for a loss of load from 52% power resulting from a turbine trip but without a direct reactor trip.

In section 6 an analyses is presented for a loss of load from 72% power resulting from a turbine trip without a direct reactor. It is concluded in this report that with respect to core integrity deletion of reactor trip on turbine trip is acceptable below 70% power. However the plant must have 70% load injection capability or the reactor protection system would trip the reactor following every turbine trip.

2.0 IDENTIFICATION OF CAUSES & ACCIDENT DESCRIPTION

A major load loss on the plant can result from loss of external electrical load due to some electrical system disturbance or from a turbine trip. Offsite a-c power remains available to operate plant components such as the reactor coolant pumps. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant with less than full load rejection capability would be expected to trip from the Reactor Protection System. In the event that a safety limit is approached, protection would be provided by the high pressurizer pressure and overtemperature ΔT trips.

In the event the steam dump valves fail to open following a loss of load or turbine trip, 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 ΔT 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 (RCS) and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, automatic rod cluster control assembly control or direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow at the Engineered Safety Features Rating (105 percent of steam flow at rated power) from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized 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 relieve sufficient steam to maintain the RCS pressure within 110 percent of the RCS design pressure.

For a turbine trip event, the reactor would be tripped directly (unless below approximately 50 percent power) from a signal derived from the turbine auto steam emergency trip fluid pressure and turbine stop valves. The turbine stop valves close rapidly (typically .1 second) on loss of trip-fluid pressure actuated by one of a number of possible turbine trip signals. 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

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate steam dump and if above 50% power, a reactor trip. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure with a resultant primary system transient.

The automatic steam dump system would normally 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 the atmosphere and main feedwater flow would be lost. For this situation feedwater flow would be maintained by the Auxiliary Feedwater System to insure adequate residual and decay heat removal capability. Should the steam dump system fail to operate, the steam generator safety valves may lift to provide pressure control, as discussed previously.

Normal power for the reactor coolant pumps is supplied through busses from a transformer connected to the generator. When a generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator, thus ensuring flow for 30 seconds before any transfer is made.

Should the network bus transfer fail at 30 seconds, a complete loss of forced reactor coolant flow would result. The immediate effect of loss of coolant flow is a rapid increase in the coolant temperature in addition to the increased coolant temperature as a result of the turbine trip. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly.

The following signals provide the necessary protection against a complete loss of flow accident:

1. Reactor coolant pump power supply undervoltage.
2. 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 7).

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals pwer reactor coolant loop. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip.

3.0 ANALYSIS OF EFFECTS & CONSEQUENCES

METHOD OF ANALYSIS

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 52 percent of full power without direct reactor trip. This shows the adequacy of the pressure relieving devices and also demonstrates the core protection margins; that is, the turbine is assumed to trip without actuating all the sensors for reactor trip on the turbine stop valves. The assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

A fast bus transfer is attempted 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 conditions.

The loss of flow transient coincident with turbine trip transients are analyzed by employing the detailed digital computer codes LOFTRAN[1], FACTRAN[2], and THINC[3]. The LOFTRAN Code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level from LOFTRAN. The FACTRAN Code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC Code is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

Major assumptions are summarized below:

1. Initial Operating Conditions - the initial reactor power and RCS temperatures are assumed at their maximum values consistent with the steady state 52 percent power operation including allowances for calibration and instrument errors.

The initial RCS pressure is assumed at a minimum value consistent with the steady state 52 percent power operation including allowances for calibration and instrument errors. The initial RCS flow is assumed to be consistent with thermal design flow for three loop operation. This results in minimum margin to core protection limits at the initiation of the accident. Table 1 summarizes the initial conditions assumed.

2. Moderator & Doppler Coefficients of Reactivity - the turbine trip is analyzed with both a least negative moderator temperature coefficient and a large negative moderator temperature coefficient. Doppler power coefficients are adjusted to provide consistent minimum and maximum reactivity feedback cases.

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

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

5. Pressurizer Spray & Power-Operated Relief Valves - two cases for both the minimum and maximum reactivity feedback cases 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. Safety valves are also avilable.

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

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 since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

7. 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. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high pressurizer water level, low reactor coolant loop flow, and reactor coolant pump power supply undervoltage.

Except as discussed above, normal reactor control system and Engineered Safety Systems are not required to function.

The Reactor Protection System may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function.

4.0 RESULTS

The transient responses for a turbine trip from 52 percent of full power operation are shown for four cases: two cases for minimum reactivity feedback and two cases for maximum reactivity feedback (Figures 1 through 8). The calculated sequence of events for the accident is shown in Tables 2 and 3.

Figures 1 and 2 show the transient responses for the total loss of steam load with a least negative moderator temperature coefficient assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The fast bus transfer is attempted and assumed to fail 30 seconds after the total loss of steam load. The transfer failure results in an undervoltage trip of the reactor and the initiation of the loss of flow transient. The minimum DNBR remains well above the 1.3 limit. The steam generator safety valves limit the secondary steam conditions to saturation at the safety valve setpoint.

Figures 3 and 4 show the responses for the total loss of steam load with a large negative moderator temperature coefficient. All other plant parameters are the same as the above. The minimum DNBR remains well above the 1.3 limit throughout the transient. Pressurizer relief valves and steam generator safety valves prevent overpressurization in primary and secondary systems, respectively.

The turbine trip accident was also studied assuming the plant to be initially operating at 52 percent of full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. The fast bus transfer for this case is assumed to fail at 30 seconds after the total loss of load. Figures 5 and 6 show the transients with a least negative moderator coefficient. The neutron flux remains essentially constant at 52 percent of full power until the reactor is tripped. The DNBR remains above 1.3 throughout the transient. In this case the pressurizer safety valves are actuated and maintain system pressure below 110% of the design value.

Figures 7 and 8 are the transients with maximum reactivity feedback with the other assumptions being the same as in the preceding case. Again, the minimum DNBR remains above 1.30 throughout the transient. In this case the pressurizer safety valves are not actuated.

5.0 CONCLUSIONS

Results of the analyses show that the plant design is such that a turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits. The analysis also demonstrates that for a complete loss of forced reactor coolant flow initiated from the most adverse preconditions of a turbine trip, the DNBR does not decrease below 1.3 at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

6.0 DELETION OF REACTOR TRIP ON TURBINE TRIP BELOW 70 PERCENT POWER

The behavior of the unit was also evaluated for a complete loss of steam load from 72 percent of full power without a direct reactor trip. The cases described in section 3.0 were analyzed at 72 percent of full power and the results are shown on figures 9 through 16. These figures include a plot of DNBR versus time. As is seen from the figures the minimum DNBR does remain above 2.3 in all cases. Therefore deletion of reactor trip on turbine trip below 70 percent power presents no hazards to the integrity of the RCS. No fuel or clad damage is predicted and all applicable acceptance criteria are met. However if deletion of reactor trip on turbine trip below 70 percent power is to be implemented, Duquesne Light Company must demonstrate that the unit has condenser capacity to reject 70 percent power through the steam dump system.

TABLE I
INITIAL CONDITIONS

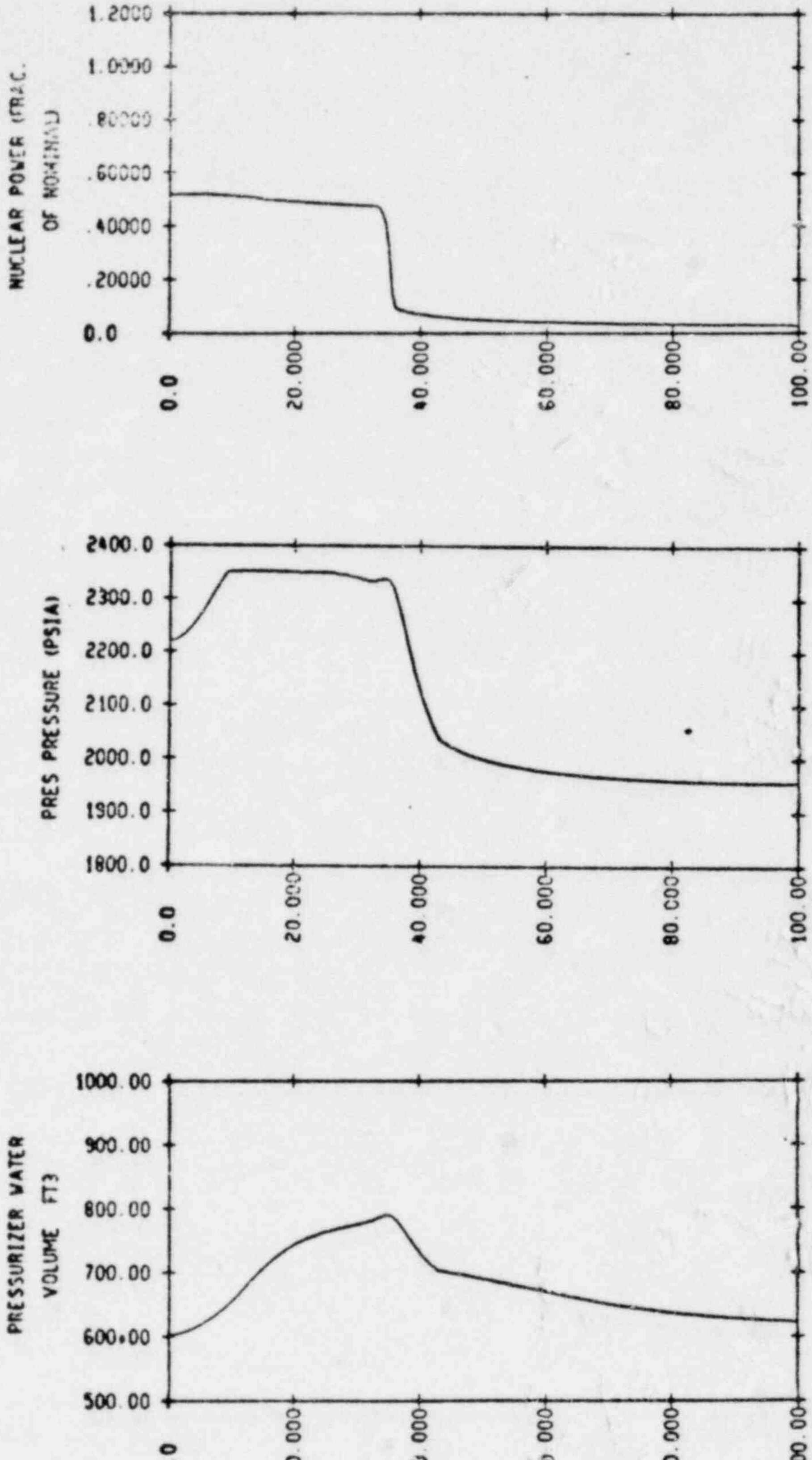
	52% POWER	72% POWER
Core Power, Mwt	1383	1915
Thermal Design Flow (TOTAL) GPM	265500	265500
Reactor Coolant Temperature		
Vessel Outlet, ^{o}F	588.4	600.2
Vessel Inlet, ^{o}F	552	550.6
Steam Generator Steam		
Temperature, ^{o}F	537.8	531.7
Pressure, PSIA	945	898

TABLE 2
TIME SEQUENCE OF EVENTS FOR A TURBINE
TRIP WITH PRESSURIZER PRESSURE CONTROL

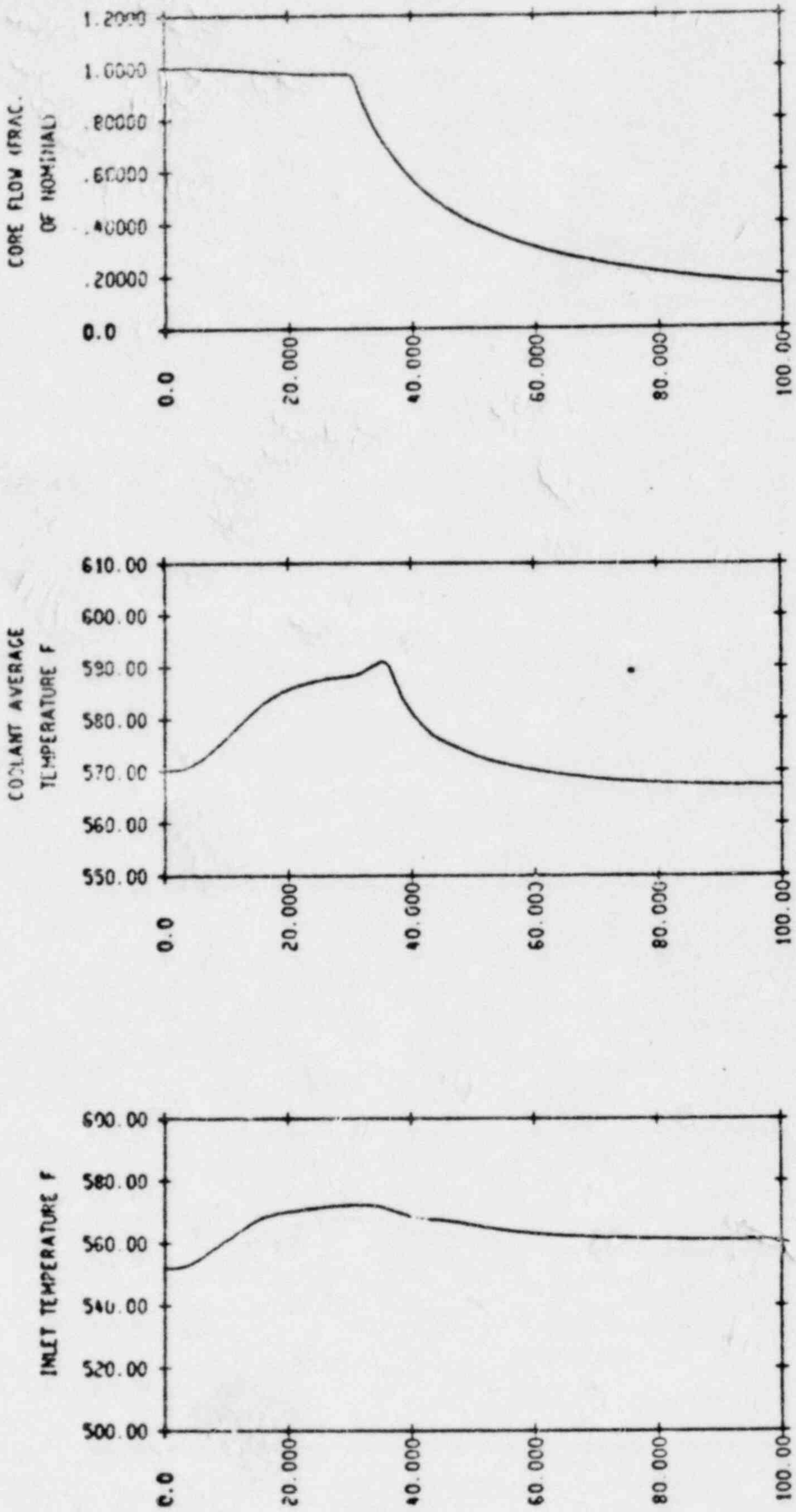
	EVENT	52% POWER	72% POWER
1. Minimum Feedback (BOL)			
	Turbine Trip	0	0
	Initiation of steam release from steam generator safety valves	13	11
	Fast bus transfer failure, flow coastdown begins	30	30
	Low flow reactor trip occurs	32.1	32.1
	Rods begin to fall	33.2	33.2
	Minimum DNBR occurs		34
	Peak pressurizer pressure occurs	10	10
2. Maximum Feedback (EOL)			
	Turbine Trip	0	0
	Initiation of steam release from steam generator safety valves	13	11.5
	Fast bus transfer failure, flow coastdown begins	30	30
	Low flow reactor trip occurs	32.1	32.05
	Rods begin to fall	33.1	33.05
	Peak pressurizer pressure occurs	10.5	9.0
	Minimum DNBR occurs	----	4.0

TABLE 3
TIME SEQUENCE OF EVENTS FOR A TURBINE TRIP WITHOUT PRESSURIZER PRESSURE CONTROL

	EVENT	52% POWER	72% POWER
1. Minimum Feedback (BOL)			
	Turbine Trip	0	0
	Fast bus transfer failure, flow coastdown begins	30	30
	Rods begin to fall and high pressurizer pressure trip occurs	13.1	9.8
	Minimum DNBR occurs		0.0
	Peak pressurizer pressure occurs	14.5	11
	Initiation of steam release from steam generator safety valves	13	11
2. Maximum Feedback (EOL)			
	Turbine Trip	0	0
	Fast bus transfer failure, flow coastdown begins	30	30
	Rods begin to fall and high pressurizer pressure trip occurs	13.9	9.9
	Minimum DNBR occurs		4
	Peak pressurizer pressure occurs	16	11
	Initiation of steam release from steam generator safety valves	13	11



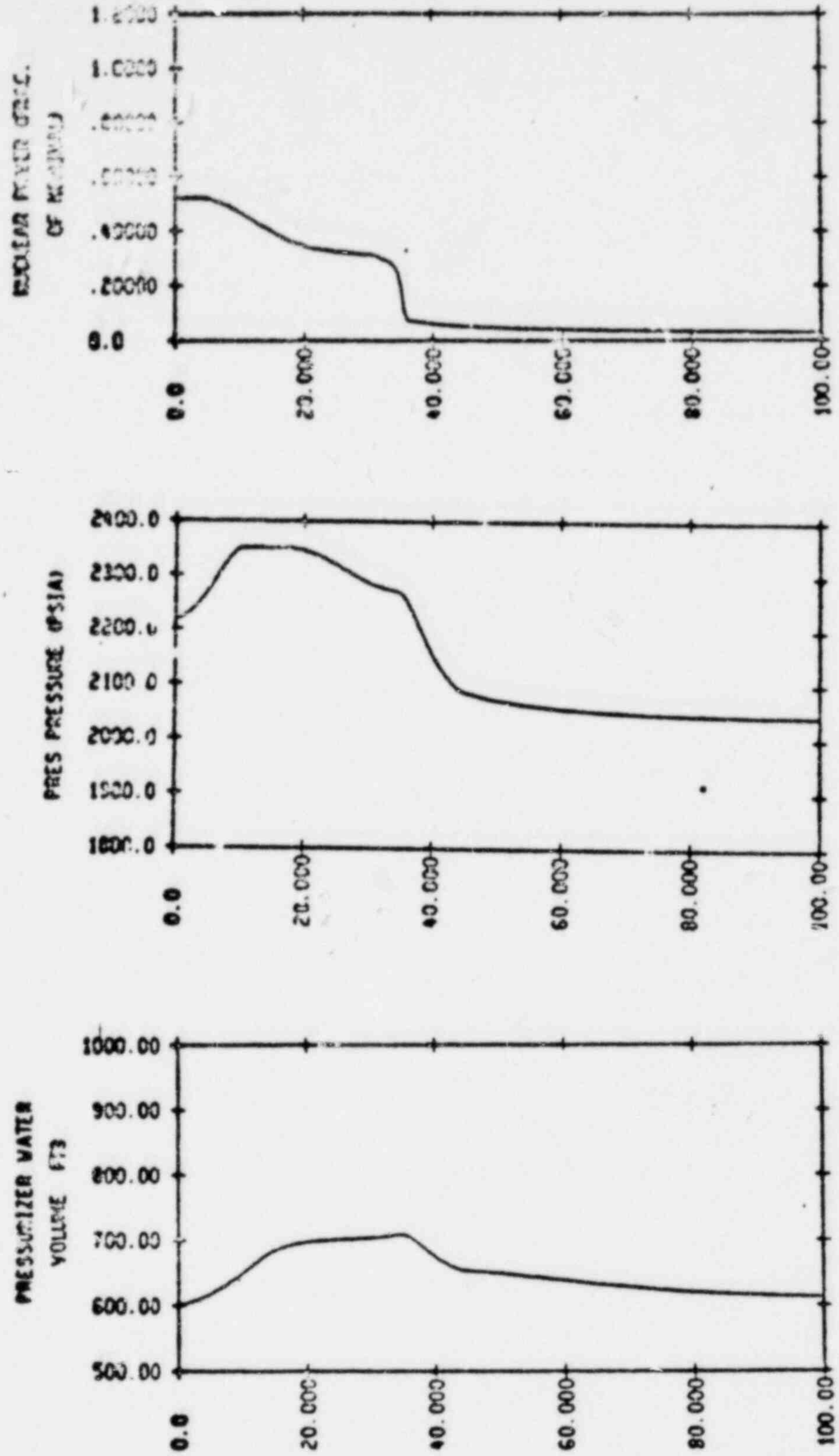
LOSS OF LOAD WITH PRESSURE CONTROL, MINIMUM FEEDBACK
FIGURE 1



TIME

LOSS OF LOAD WITH PRESSURE CONTROL, MINIMUM FEEDBACK

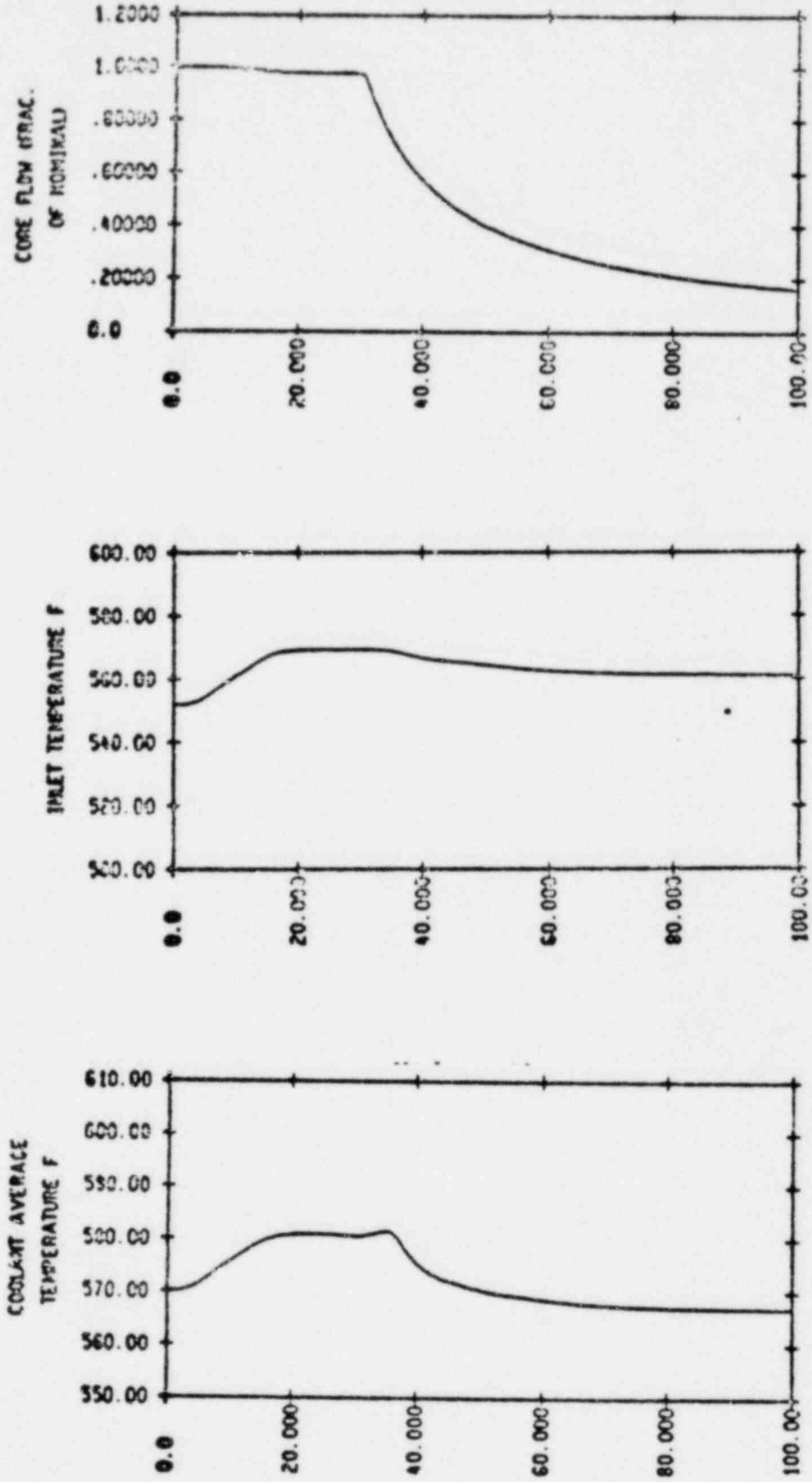
FIGURE 2



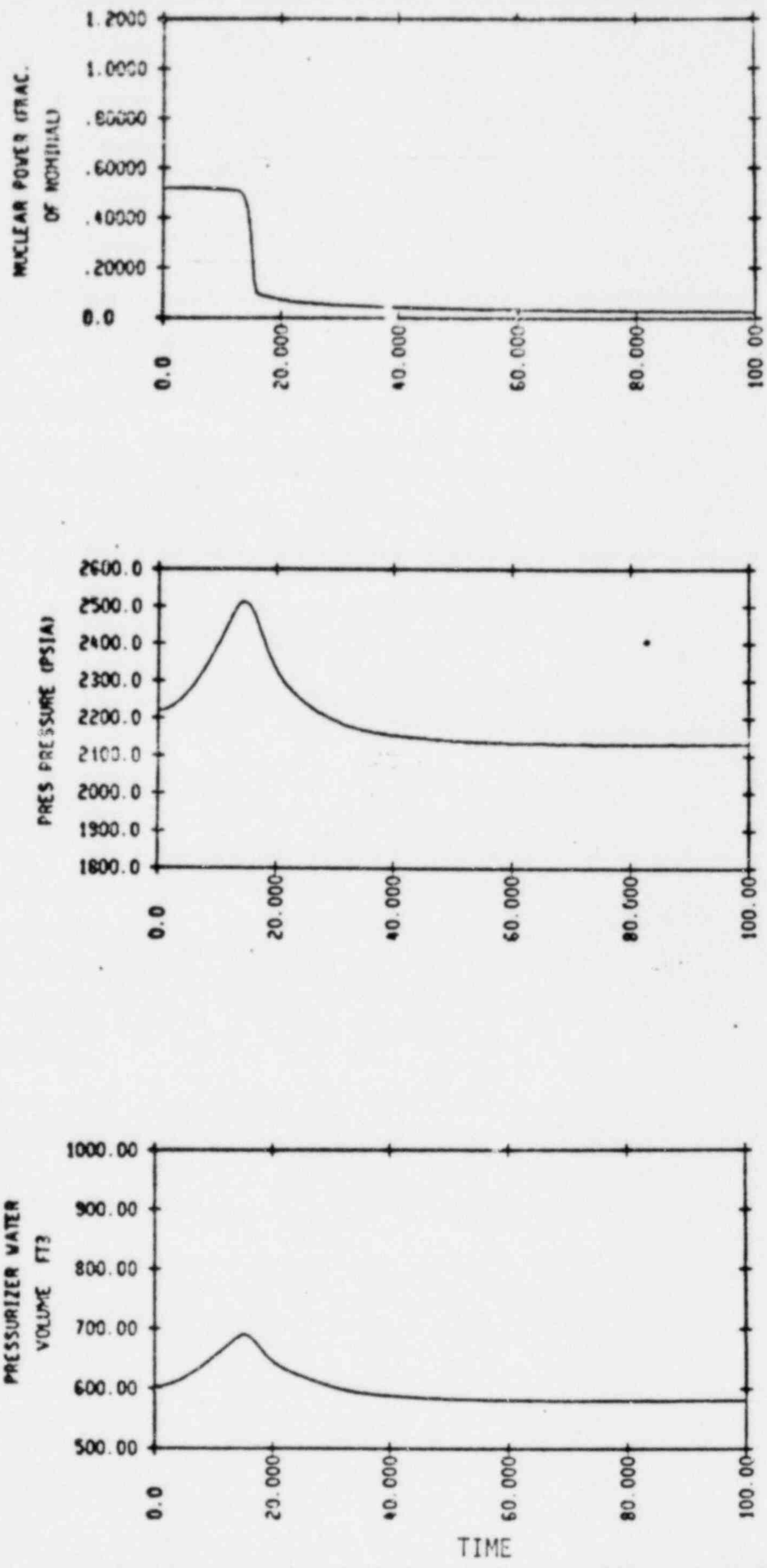
TIME

LOSS OF LOAD WITH PRESSURE CONTROL, MAXIMUM FEEDBACK

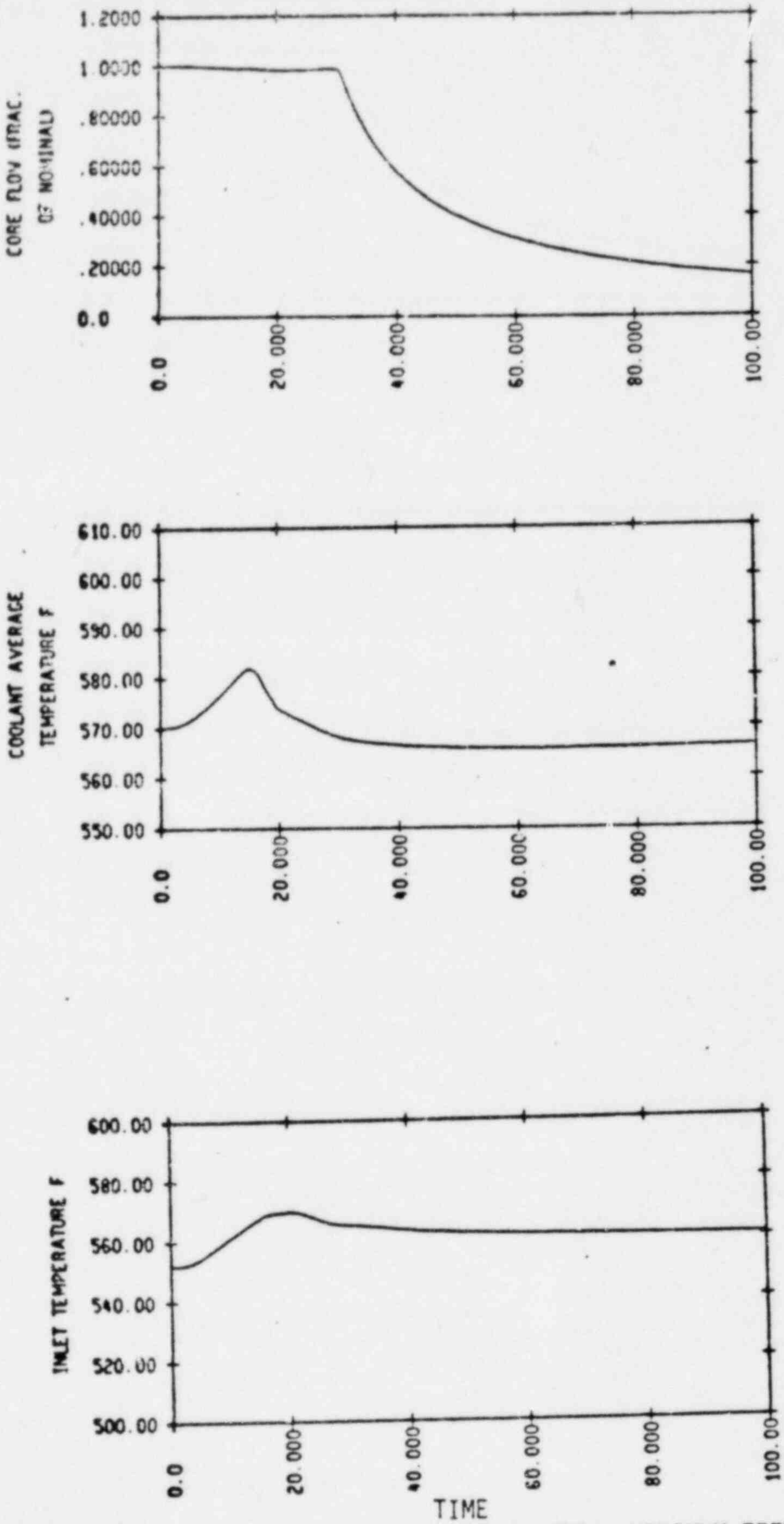
FIGURE 3



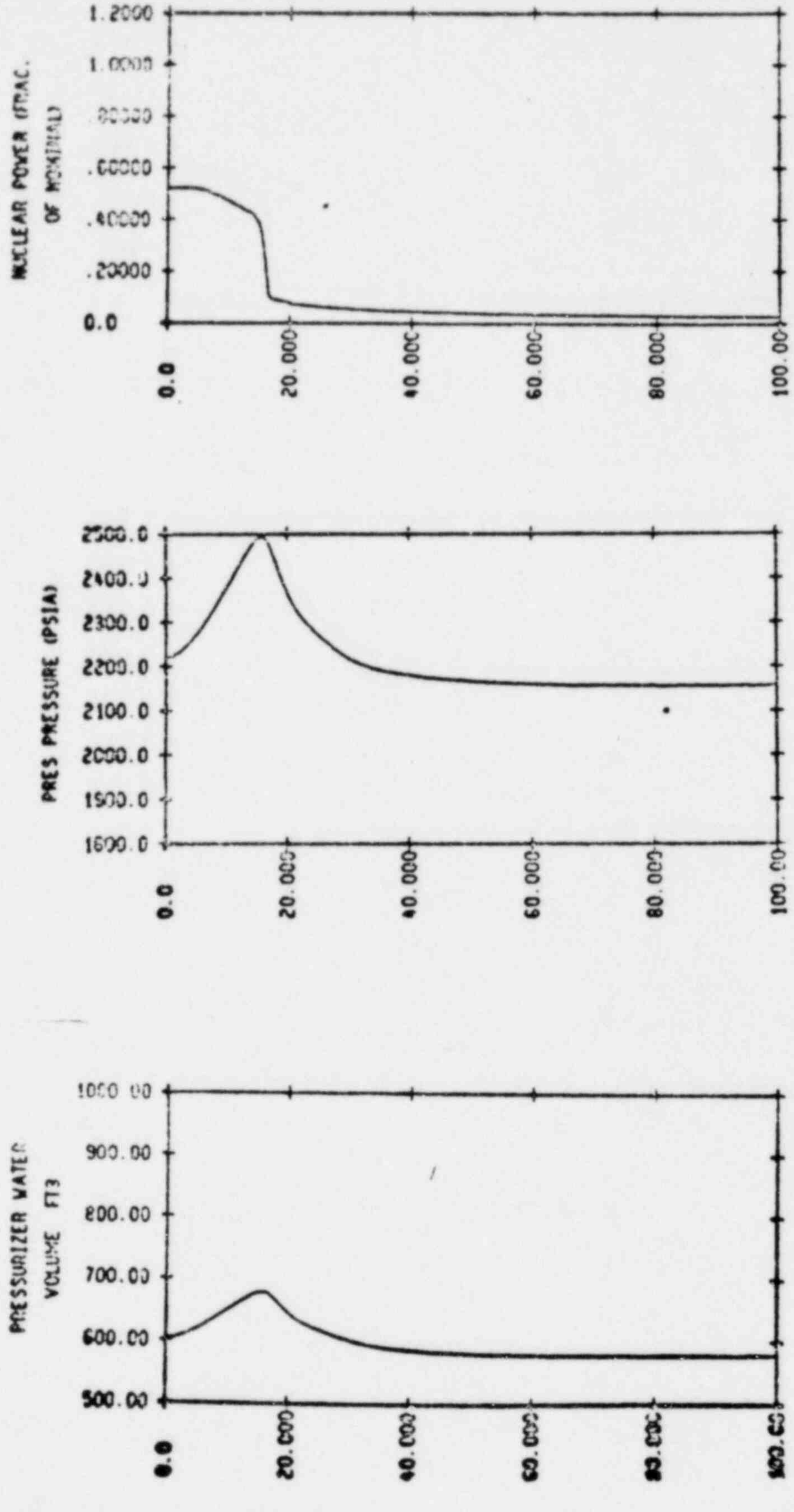
TIME
LOSS OF LOAD WITH PRESSURE CONTROL, MAXIMUM FEEDBACK
FIGURE 4



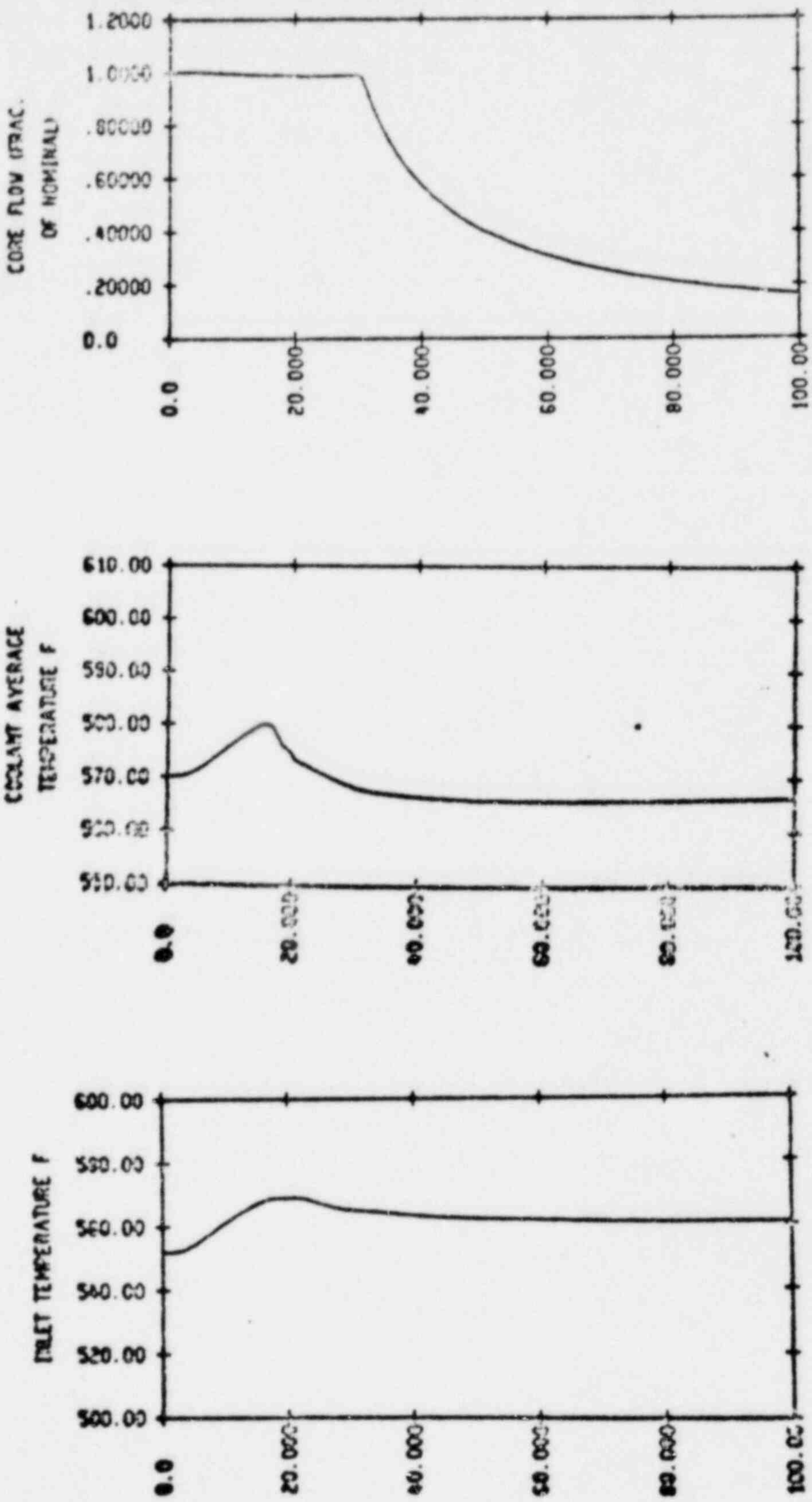
LOSS OF LOAD WITHOUT PRESSURE CONTROL, MINIMUM FEEDBACK
FIGURE 5



LOSS OF LOAD WITHOUT PRESSURE CONTROL, MINIMUM FEEDBACK
FIGURE 6



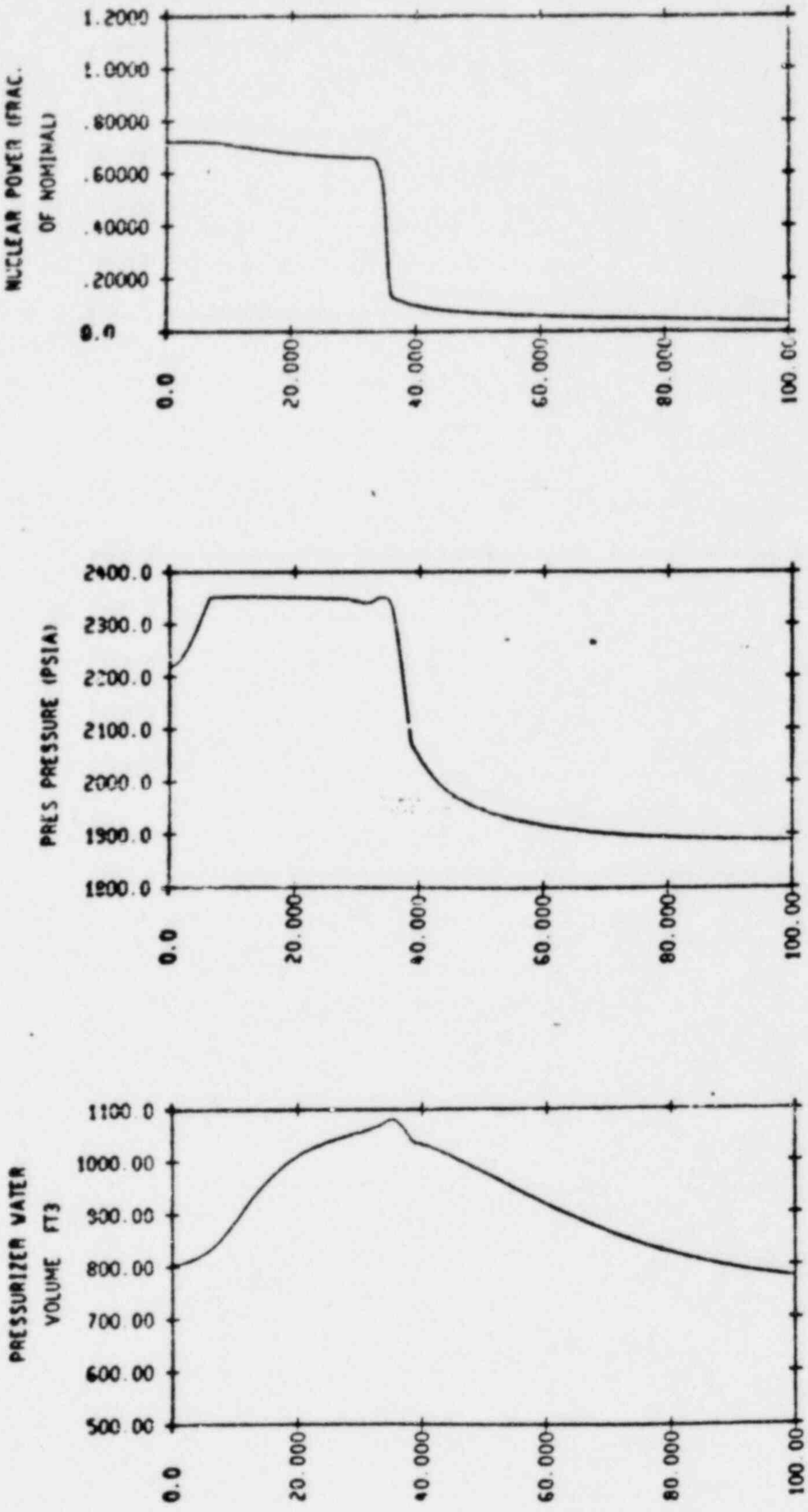
LOSS OF LOAD WITHOUT PRESSURE CONTROL, MAXIMUM FEEDBACK
FIGURE 7



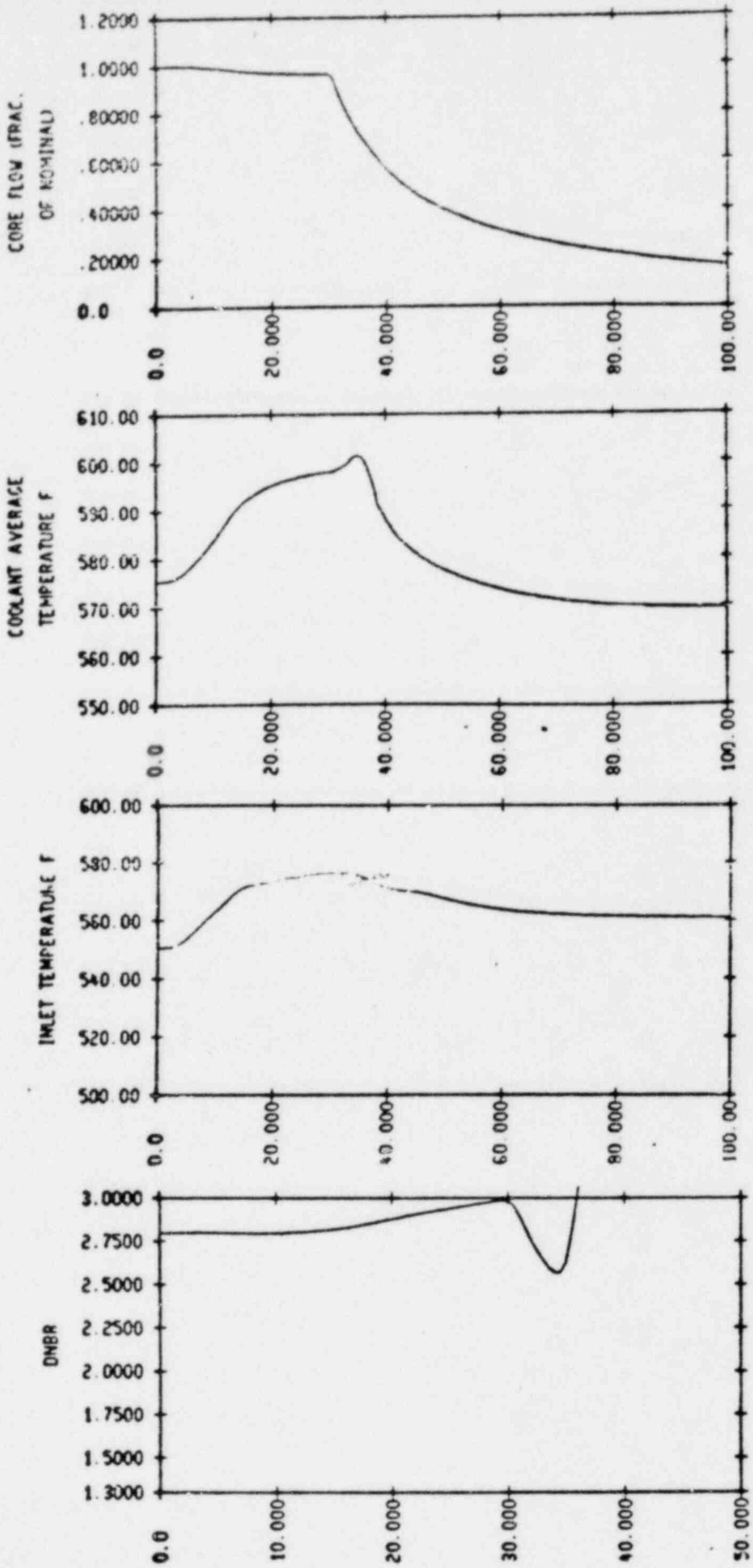
TIME

LOSS OF LOAD WITHOUT PRESSURE CONTROL, MAXIMUM FEEDBACK

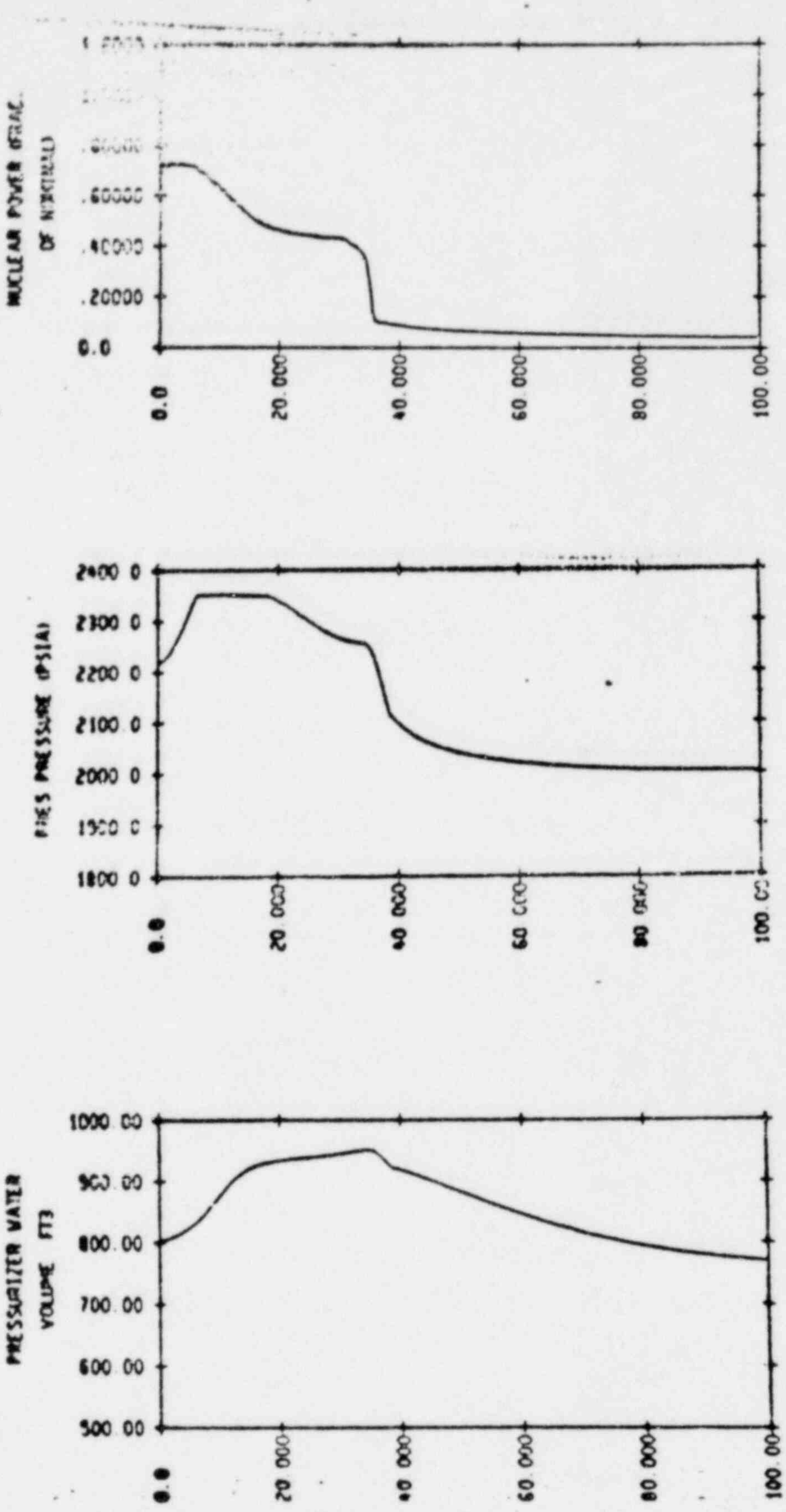
FIGURE 8



TIME
LOSS OF LOAD FROM 70 PERCENT POWER TO PRESSURE CONTROL
MINIMUM FEEDBACK
FIGURE 9



TIME
LOSS OF LOAD FROM 70 PERCENT POWER WITH PRESSURE CONTROL
MINIMUM FEEDBACK
FIGURE 10

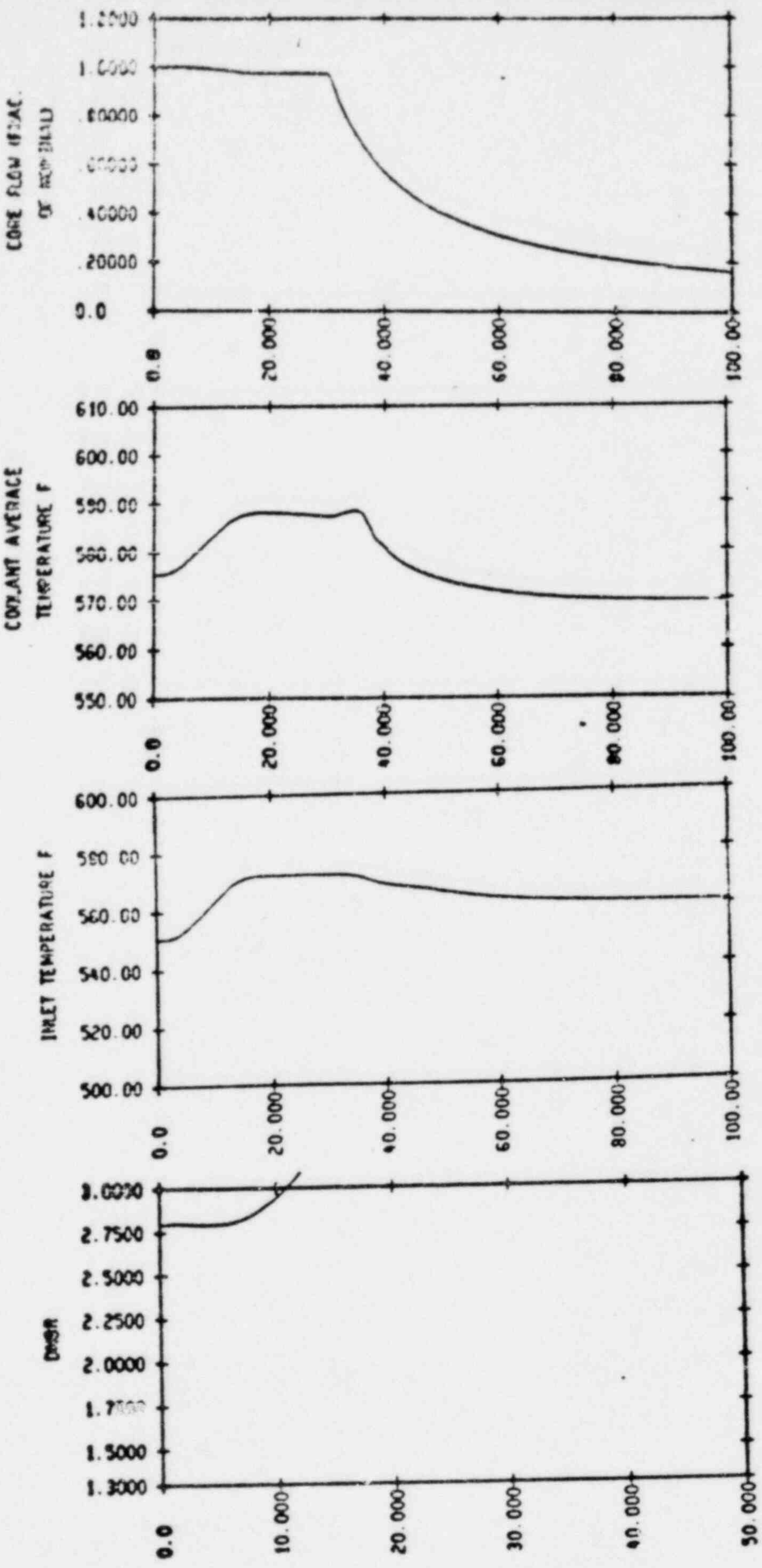


TIME

LOSS OF LOAD FROM 70 PERCENT POWER WITH PRESSURE CONTROL

MAXIMUM FEEDBACK

FIGURE 11

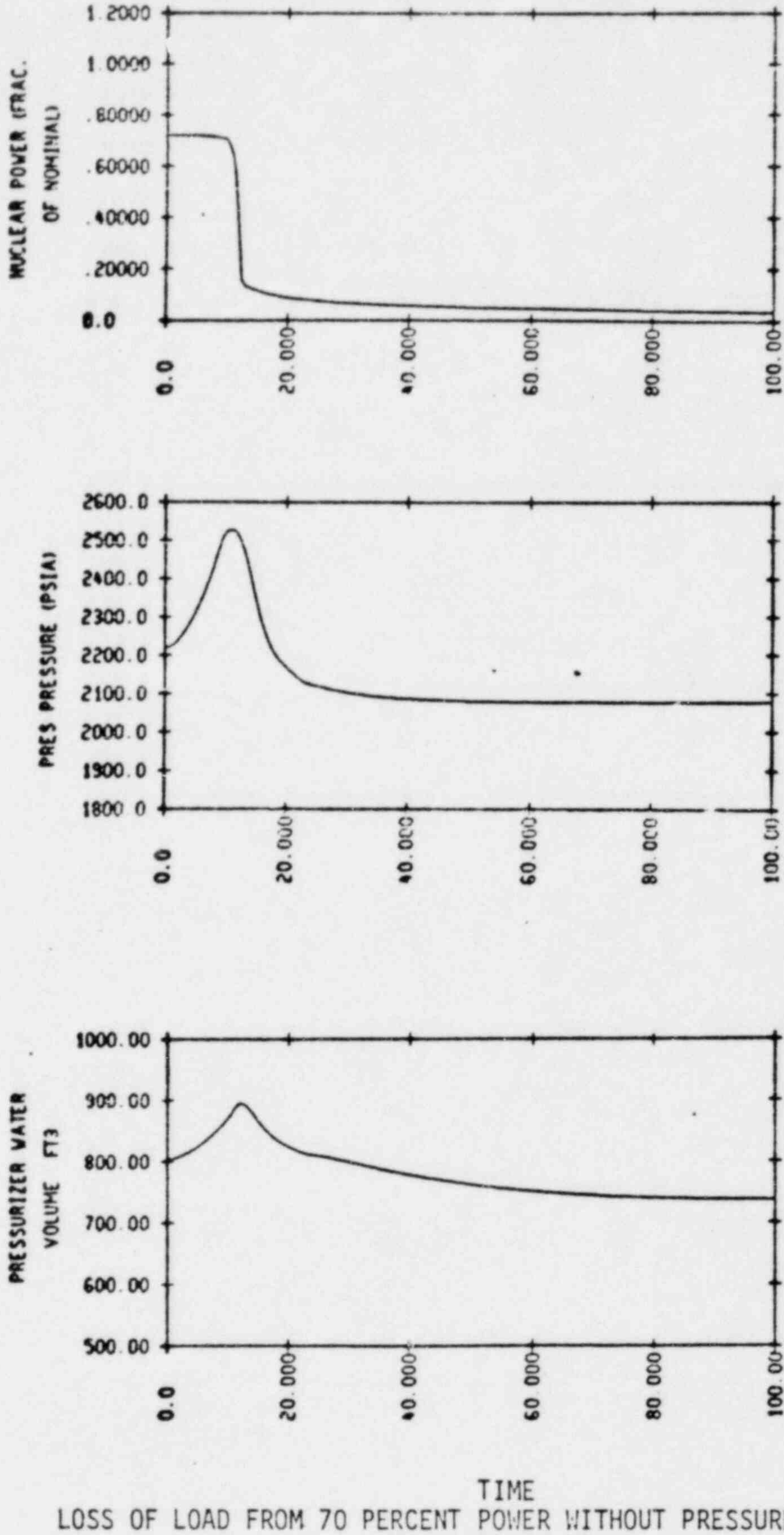


TIME

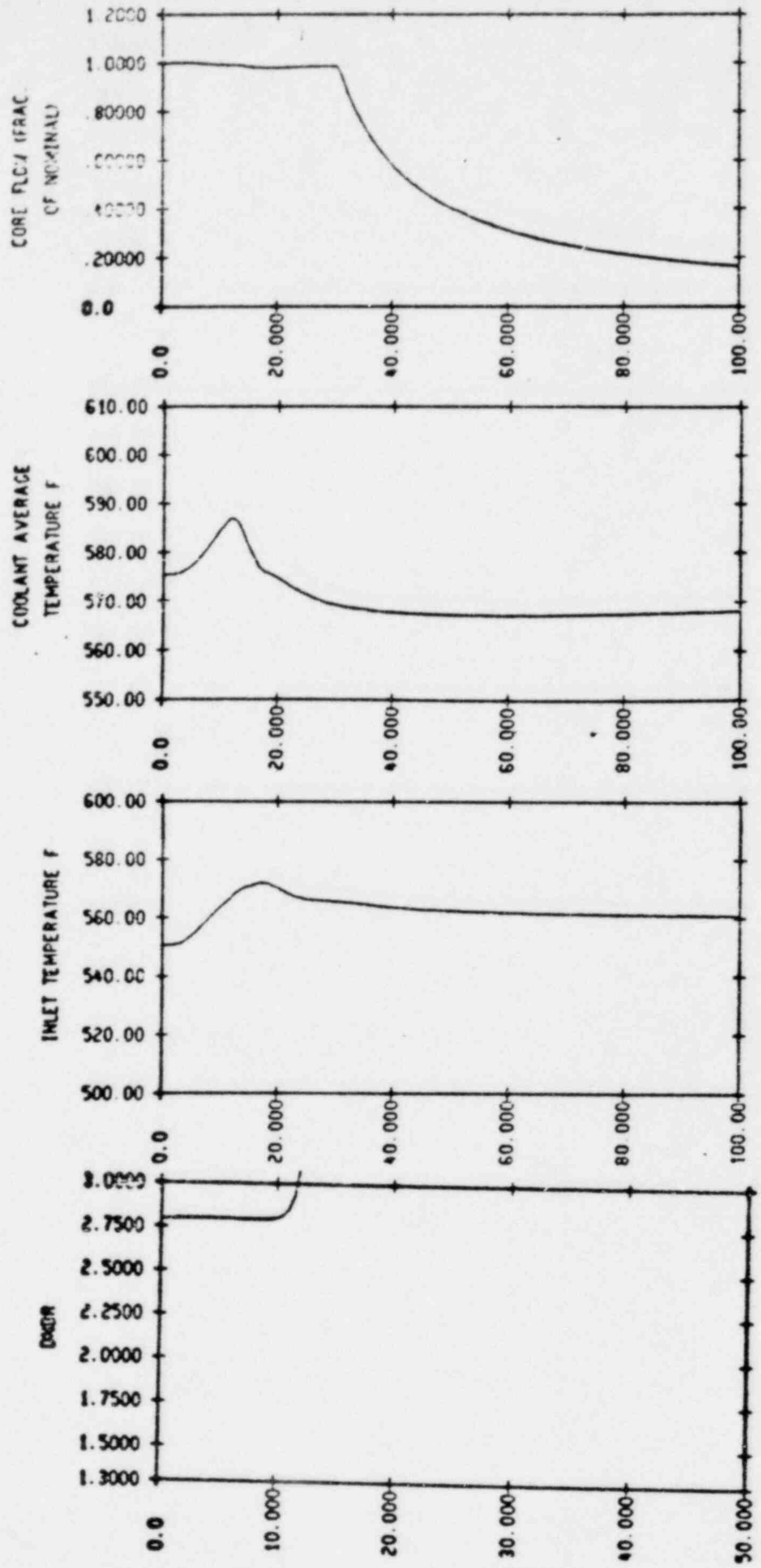
LOSS OF LOAD FROM 70 PERCENT POWER WITH PRESSURE CONTROL

MAXIMUM FEEDBACK

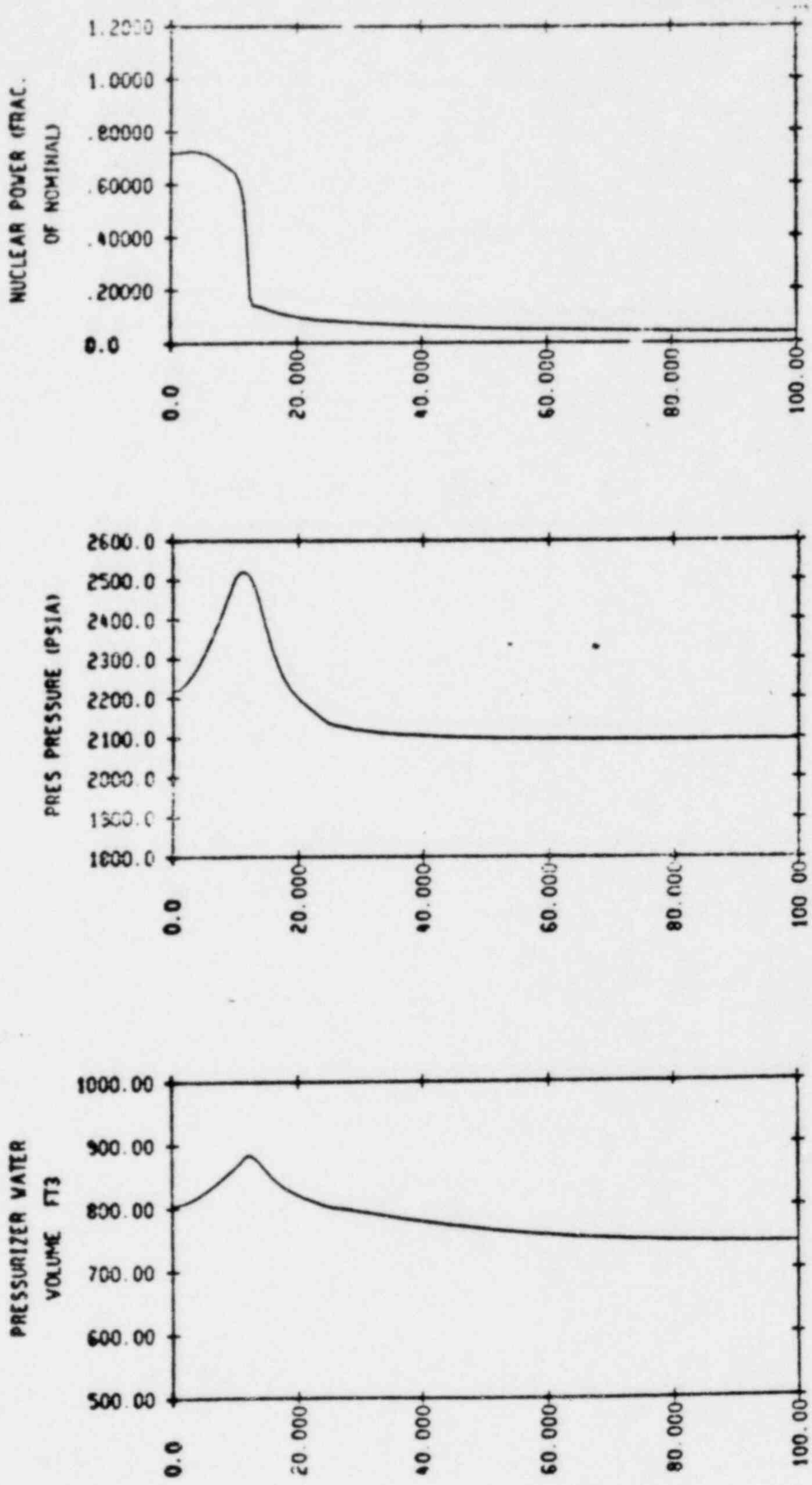
FIGURE 12



LOSS OF LOAD FROM 70 PERCENT POWER WITHOUT PRESSURE CONTROL
MINIMUM FEEDBACK
FIGURE 13

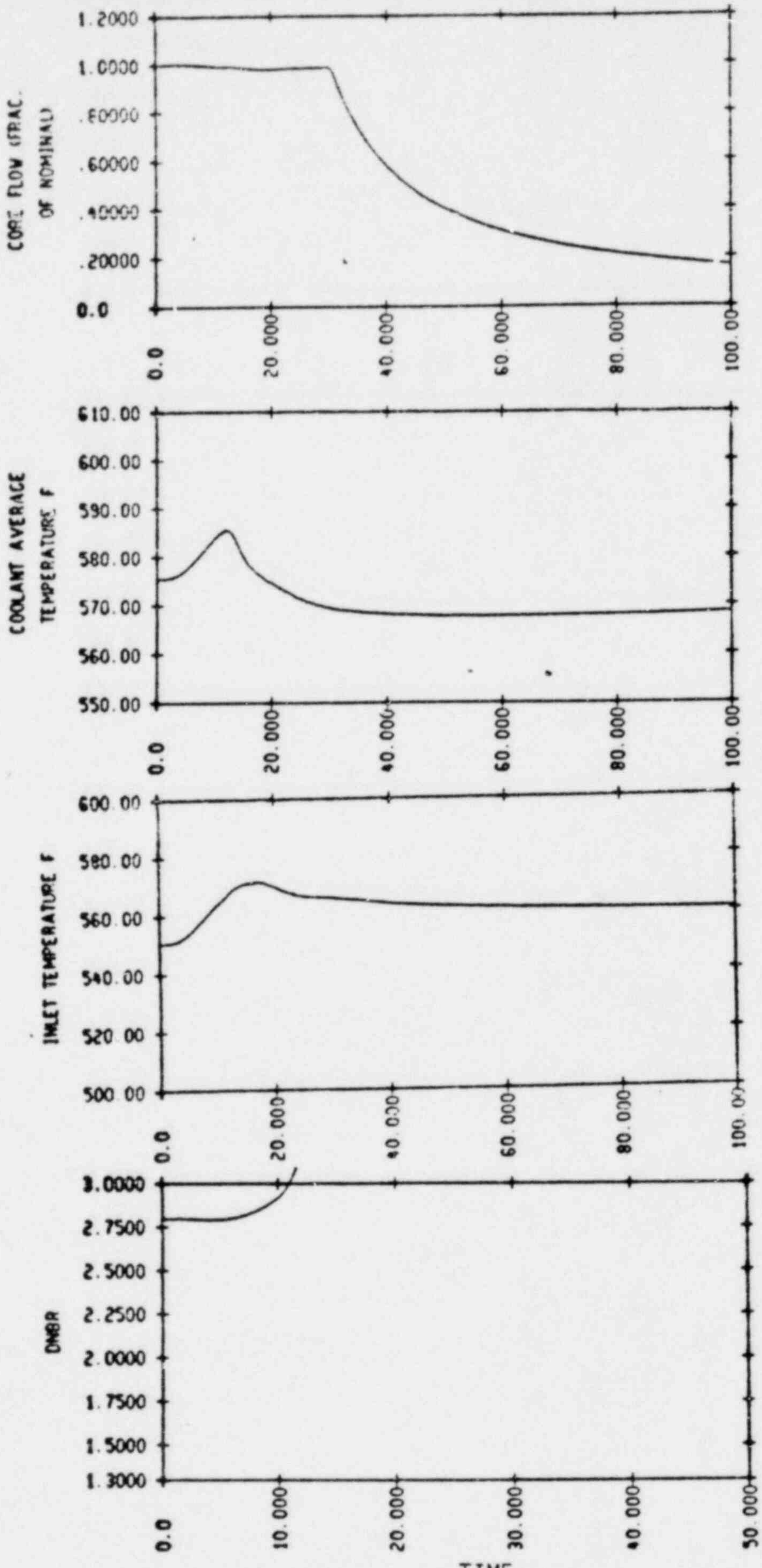


LOSS OF LOAD FROM 70 PERCENT POWER WITHOUT PRESSURE CONTROL
MINIMUM FEEDBACK
FIGURE 14



TIME

LOSS OF LOAD FROM 70 PERCENT POWER WITHOUT PRESSURE CONTROL
MAXIMUM FEEDBACK
FIGURE 15



LOSS OF LOAD FROM 70 PERCENT POWER WITHOUT PRESSURE CONTROL
MAXIMUM FEEDBACK

FIGURE 16

SECTION 7
CHANGES TO THE TECHNICAL SPECIFICATIONS

TABLE 3.3-1 (CONTINUED)

DESIGNATION		
P-9	With 2 of 4 power range neutron flux channels ≥51% of rated thermal power	P-8 prevents or defeats the automatic block of reactor trip on turbine trip
P-10	With 3 of 4 power range neutron flux channels <9% of rated thermal power	P-10 prevents or defeats the manual block of: Power Range low setpoint reactor trip, intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7

SECTION 8
PRECAUTIONS, LIMITATION AND SETPOINTS
DOCUMENT

3. Reactor Trip Interlocks

A. Permissive circuit P-6 (Permissive for manual block of source range high level reactor trip by intermediate range flux)

(NC-35D)
(NC-36D) 10^{-10} amperes (2)

B. Permissive circuit P-7 (Block of "at-power" reactor trips)

1. Nuclear power

(NC-41M)
(NC-42M)
(NC-43M)
(NC-44M) 10% of full power

2. Turbine load

(PC-446A)
(PC-447E) 10% of full load

C. Permissive circuit P-8 (Block of single-loop loss of coolant flow reactor trip)

Nuclear power

(NC-41N)
(NC-42N)
(NC-43N)
(NC-44N) 30% of full power

The Permissive 8 setpoint should be set at 30% during all times when the three loop setpoints for the overtemperature ΔT trip are being used. After the overtemperature ΔT trip has been reset to the value specified for two loop operation, the Permissive 8 setpoint may be reset to 65% of full power with the inactive loop stop valves open and to 70% of full power with the inactive loop stop valves closed.

D. Permissive circuit P-9 (Block reactor trip on turbine trip)

Nuclear Power

(NC-41S)
(NC-42S)
(NC-43S)
(NC-44S) 50% of full power

E. Permissive circuit P-10 (Permissive for manual block of reactor trip on intermediate and power range, low setpoint

Nuclear Power

(NC-41M)
(NC-42M)
(NC-43M)
(NC-44M) 10% of full power

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