TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

May 14, 1981

RECEIVED MAY 1 3 1981 U.S. NUCLEAR REGULATORY COMMISSION

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Schwencer:

In the Matter of the Application of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

My letter to you dated July 18, 1980 provided a response for Sequoyah Nuclear Plant to NUREG-0737 Item II.K.3.10, Anticipatory Trip Modification. In that response, we stated that TVA had not proposed the subject modification. TVA has now decided to propose this modification for Sequoyah to allow for deletion of reactor trip on turbine trip below 50 percent power. We are submitting a proposed technical specification change for Sequoyah unit 1 by separate letter.

The analysis for deletion of reactor trip on turbine trip below 50 percent power was performed for Sequoyah by Westinghouse Electric Corporation. The results are presented in the enclosed report. It was found that the reactor coolant system (RCS) does not over-pressurize, the pressurizer does not fill, and the departure of nucleate boiling ratio (DNBR) remains well above 1.30 throughout each transient.

In order to judge the relative probabilities in accordance with NUREG-0611 of opening the pressurizer power-operated relief valves (PORV's) during a loss of load from 50 percent power with and without an immediate reactor trip, it is necessary to define some common basis for comparison. In these cases, the basis used for analysis purposes assumed no instrument errors and assumed normal operation of the automatic rod control, pressurizer spray, and steam dump systems. These are best-estimate assumptions. In addition, the moderator temperature coefficient used was a conservative beginning-of-life value.

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The loss of load without immediate reactor trip analysis showed that the PORV's would not open and that a 26 lb/in^2 margin to opening would be available. This is a significant margin since it is beyond the pressurizer pressure control instrumentation error band. The assumption of error beyond the channel accuracy or of degradation of systems performance (i.e., reduced steam dump capability) will reduce this margin and may eventually lead to the opening of the PORV's; however, a departure from best-estimate conditions would put the plant in an assumed condition which is not characteristic of normal operation (i.e., the most likely condition) and therefore not appropriate for making judgements on probability.

If you have any questions, please get in touch with D. L. Lambert at FTS 857-2581.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

M R Wisenburg

M. R. Wisenburg
Nuclear Engineer

Sworn to and subscribed before me
this 14th day of May 1981

Bryant M. Lowery
Notary Public

My Commission Expires 4/4/82

Enclosure

ENCLOSURE

SEQUOYAH NUCLEAR PLANT

NUREG-0737 ITEM II.K.3.10
PROPOSED ANTICIPATORY TRIP MODIFICATION

REVISED RESPONSE

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.

Two safety consideration must be evaluated in order to implement a new system that eliminates reactor trip belcw 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.

The reported analyses show that both safety considerations are satisfied for the loss of electrical load at half power. The reactor coolant pressure does not exceed 110% of the ASME design limit and the DNBR never falls below 1.3 at any time during the transient. In fact, the DNBR remains well above the 1.3 limit in each of the reported cases.

2.0 IDENTIFICATION OF CAUSES AND 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 temperature and pressure do not significantly increase if the steam dump and pressurizer pressure control systems are functioning properly. If the turbine condenser were not available, the excess steam would be dumped to the atmosphere and main feedwater flow could 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 per 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 AND 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 adequancy 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, due to the assumed failure of the fast bus transfer, is 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.

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 four 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 and 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 set.)int value.

5. Pressurizer Spray and 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 available.

   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 momentarily 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.

REFERENCES

1. TWT Burnett, C. J. McIntyre, J. C. Buker, R. P. Rose, "LOFTRAN Code Description", WCAP-7907, June, 1972.

2. C. Hunin, "FACTRAN, A Fortan IV Code for Thermal Transients in UO_2 Fuel Rod", WCAP-7908, June, 1972.

3. H. Chelemer, J. Waisman, L. S. Tong, "Subchannel Thermal Analysis of Rod Bundle Core", WCAP-7015, January, 1969.

TABLE I

INITIAL CONDITIONS

	52% POWER
Core Power, Mwt	1780
Thermal Design Flow (TOTAL) GPM	354000
Reactor Coolant Temperature	
Vessel Outlet, °F	587
Vessel Inlet, °F	551
Steam Generator Steam	
Temperature, °F	542
Pressure, PSIA	977

TABLE 2

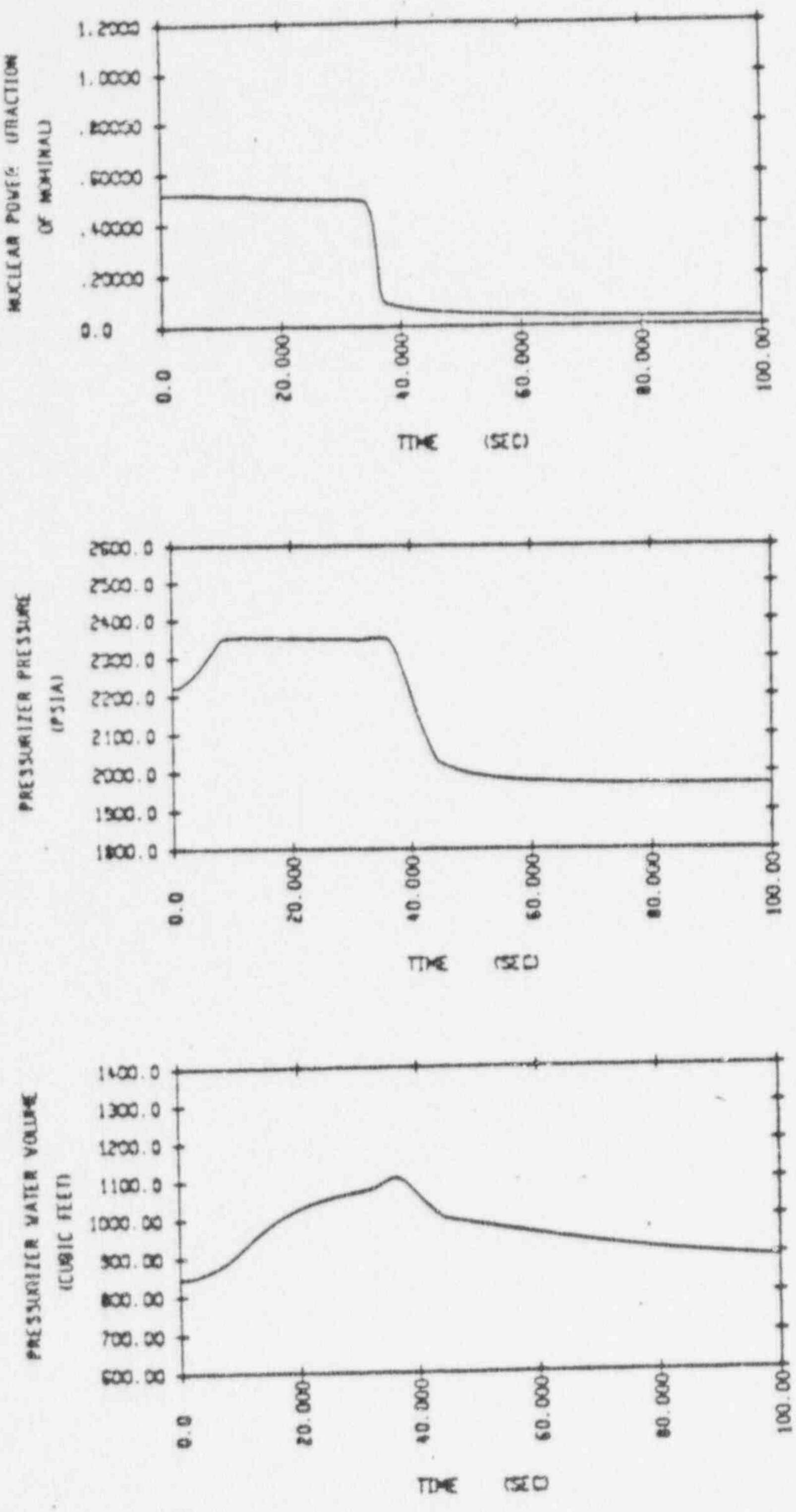
TIME SEQUENCE OF EVENTS FOR A TURBINE TRIP WITH PRESSURIZER PRESSURE CONTROL

	EVENT	52% POWER
1. Minimum Feedback (BOL)		
	Turbine Trip	0
	Initiation of steam release from steam generator safety valves	11
	Peak pressurizer pressure occurs	12
	Fast bus transfer failure, flow coastdown begins	30
	Low flow reactor trip occurs and rods begin to fall	33
2. Maximum Feedback (EOL)		
	Turbine Trip	0
	Initiation of steam release from steam generator safety valves	11
	Peak pressurizer pressure occurs	12
	Fast bus transfer failure, flow coastdown begins	30
	Low flow reactor trip occurs and rods begin to fall	33

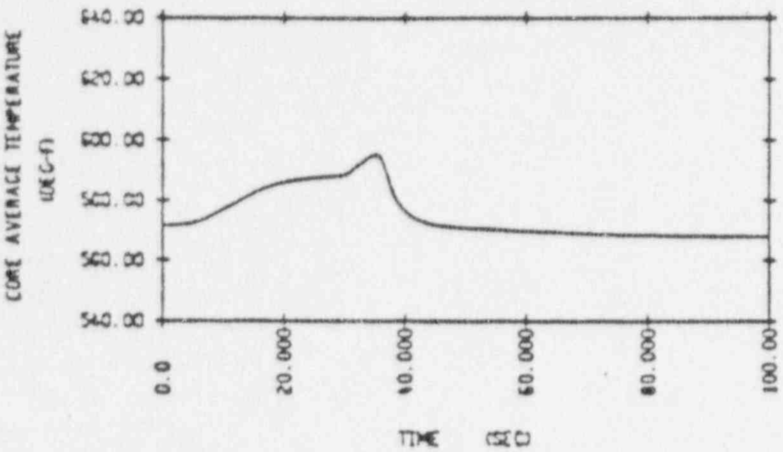
TABLE 3

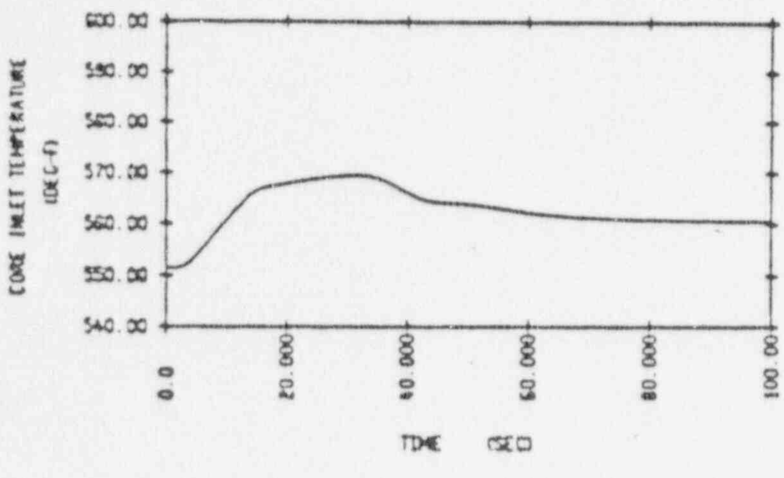
TIME SEQUENCE OF EVENTS FOR A TURBINE TRIP WITHOUT PRESSURIZER PRESSURE CONTROL

	EVENT	52% POWER
1. Minimum Feedback (BOL)		
	Turbine Trip	0
	Initiation of steam release from steam generator safety valves	11
	High pressurizer pressure trip occurs, and rods begin to fall	12.4
	Peak pressurizer pressure occurs	14
	Fast bus transfer failure, flow coastdown begins	30
Maximum Feedback (EOL)		
	Turbine Trip	0
	Initiation of steam release from steam generator safety valves	11
	High pressurizer pressure trip occurs, and rods begin to fall	13
	Peak pressurizer pressure occurs	15.5
	Fast bus transfer failure, flow coastdown begins	30

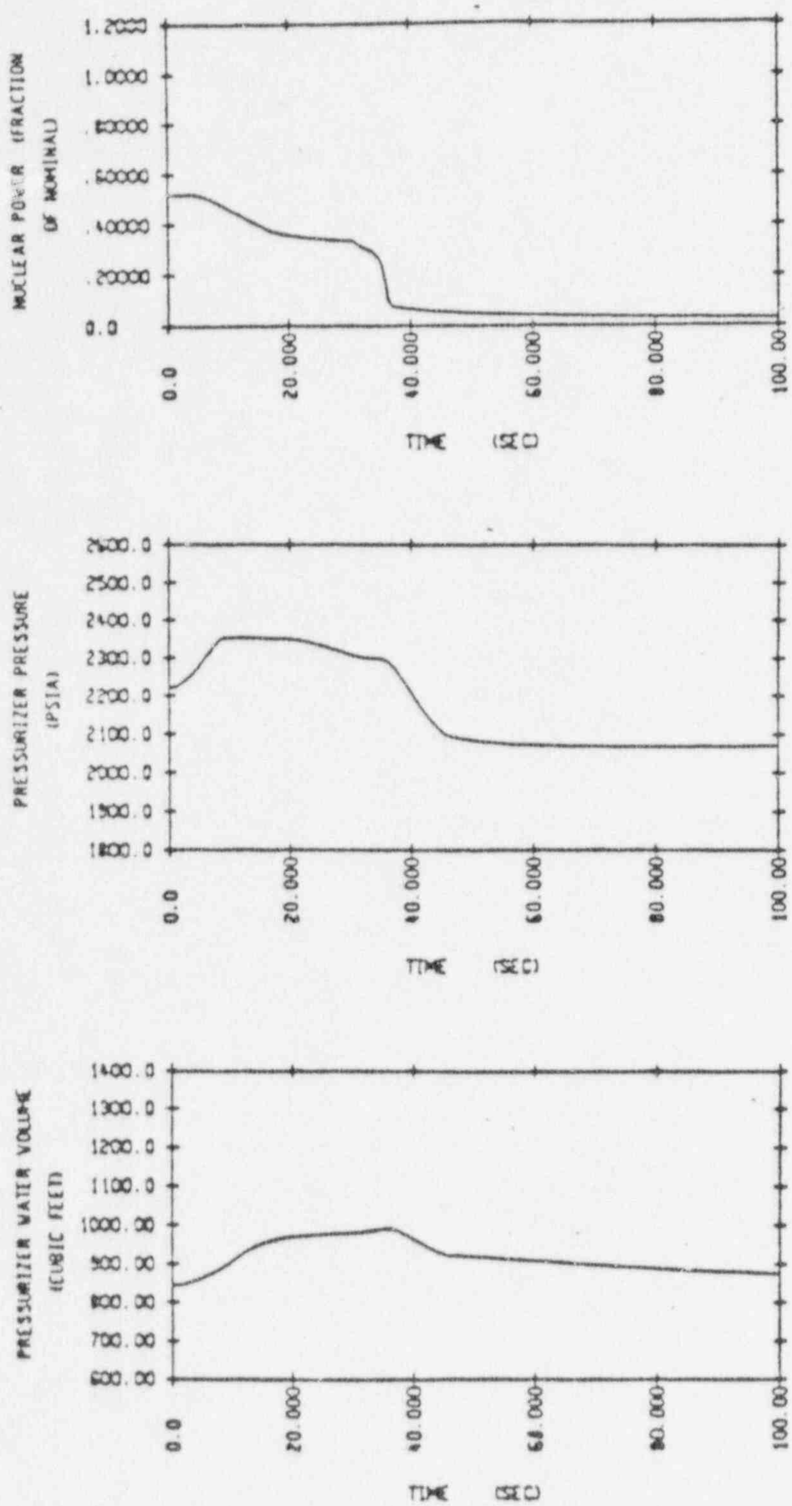


LOSS OF LOAD WITH PRESSURE CONTROL, MINIMUM FEEDBACK
FIGURE 1

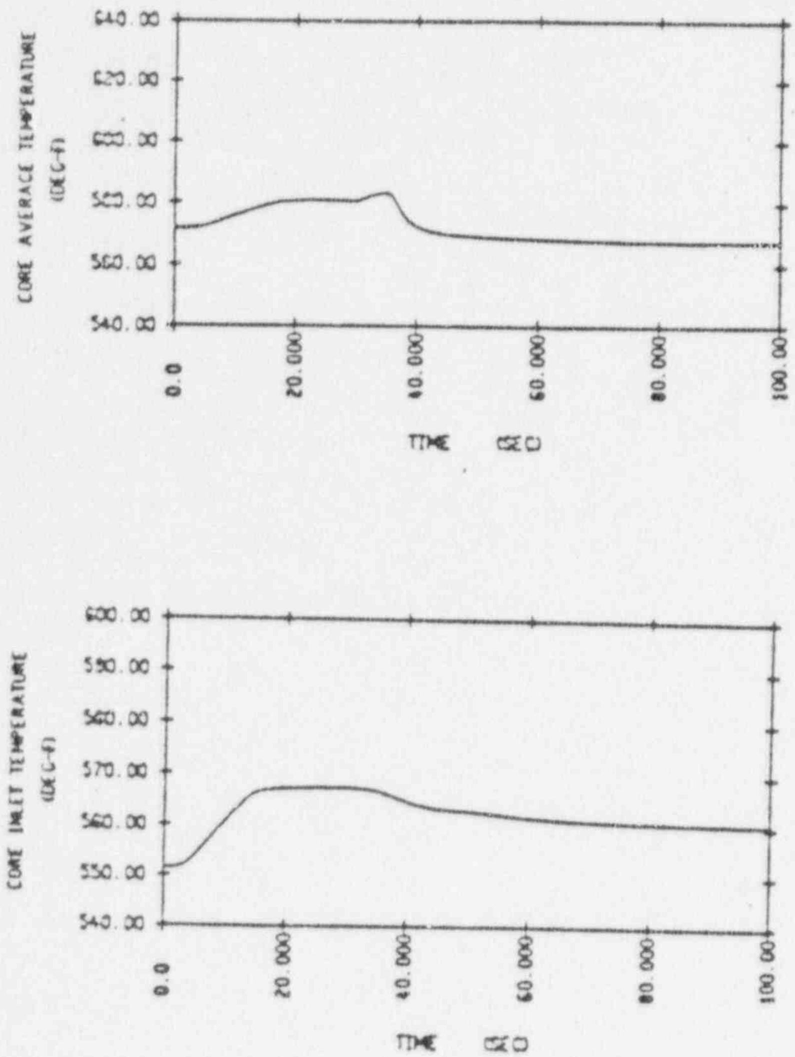




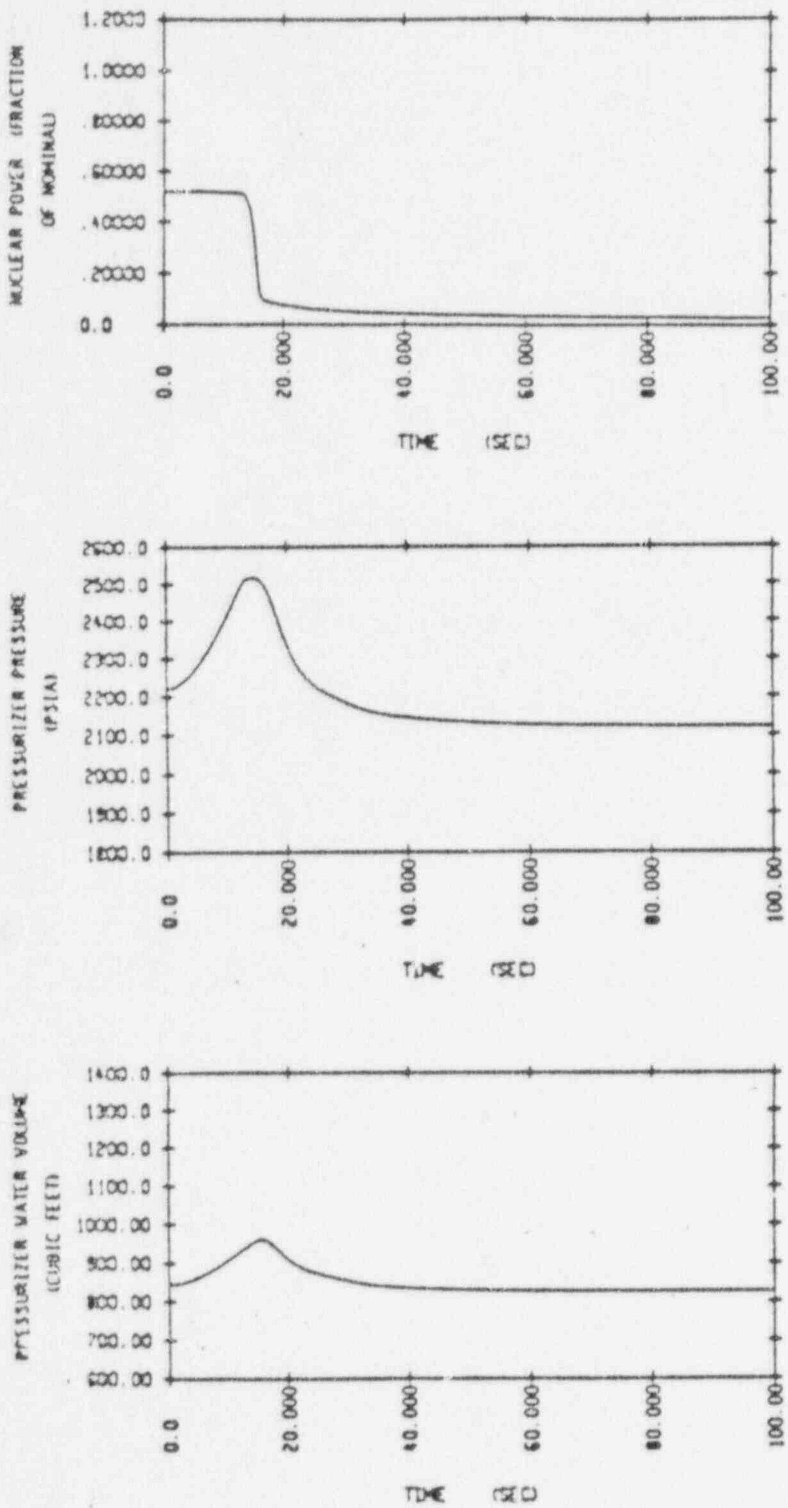
LOSS OF LOAD WITH PRESSURE CONTROL, MINIMUM FEEDBACK
FIGURE 2



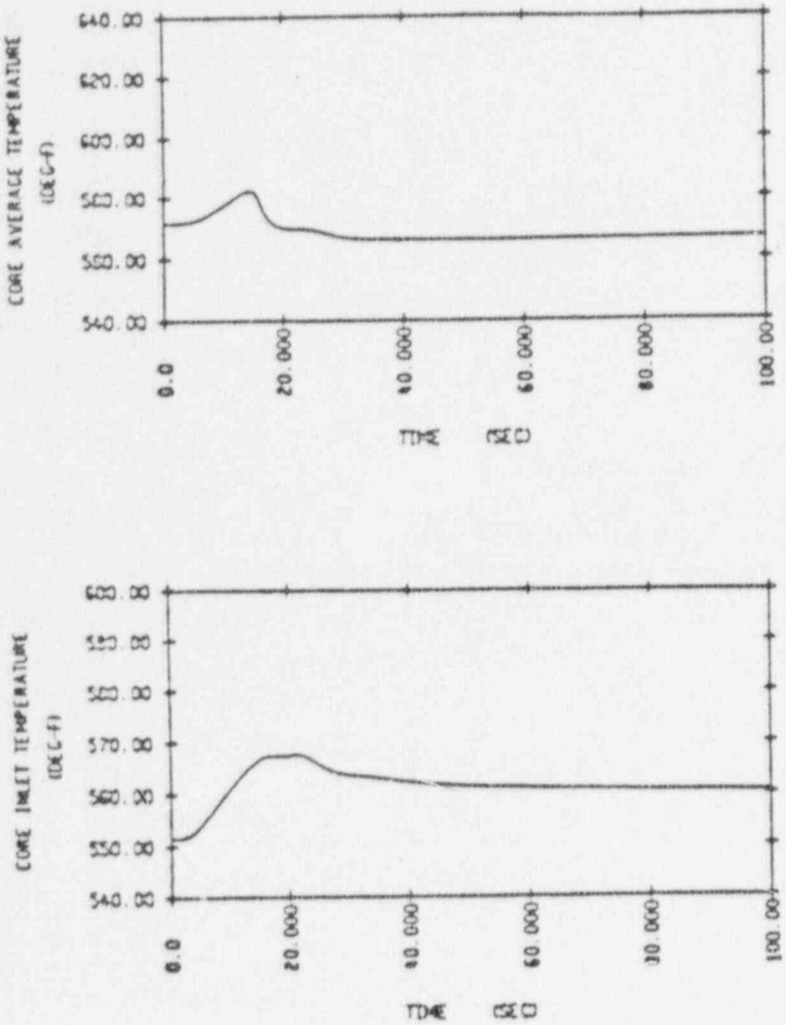
LOSS OF LOAD WITH PRESSURE CONTROL, MAXIMUM FEEDBACK
FIGURE 3



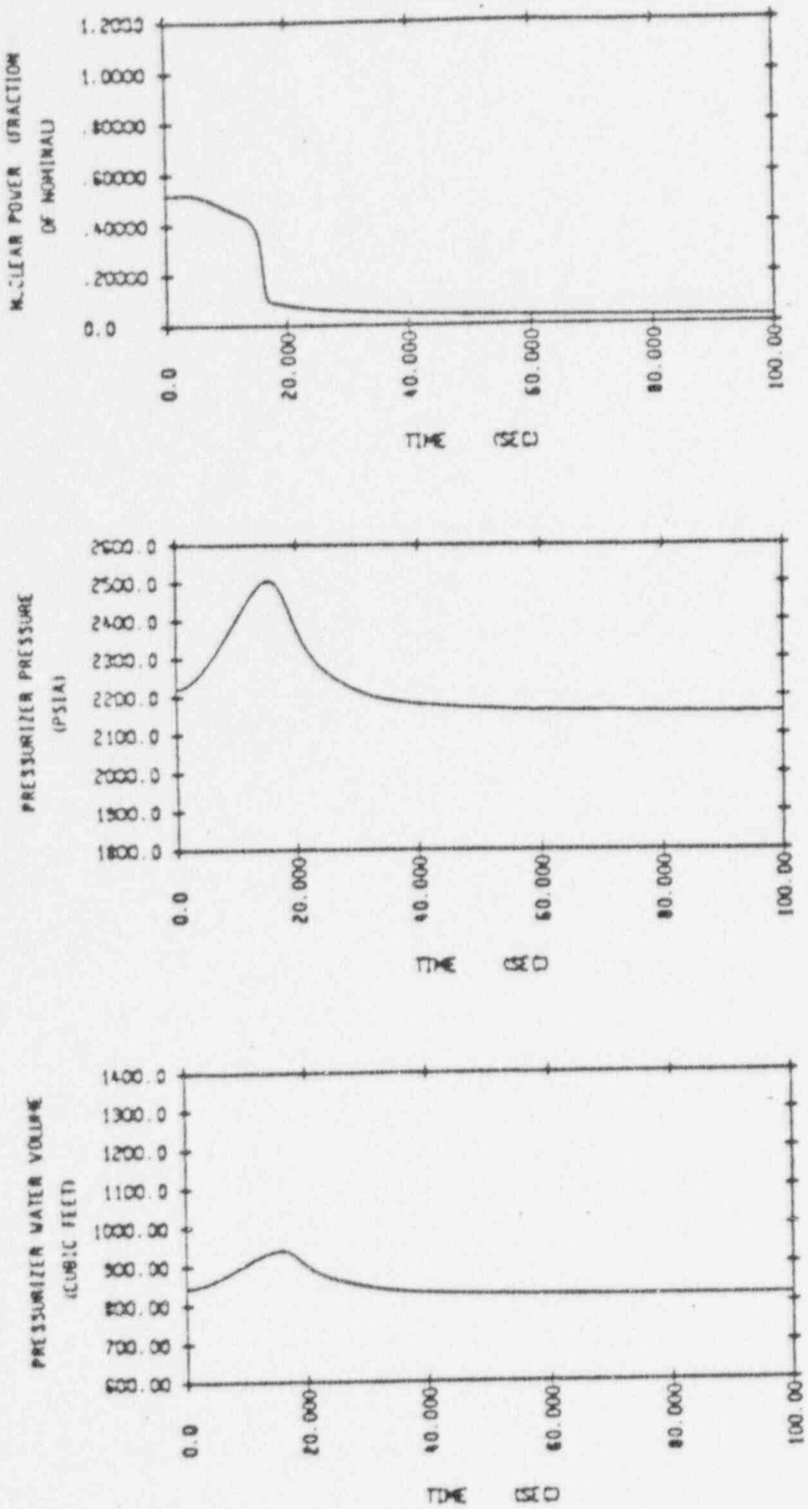
LOSS OF LOAD WITH PRESSURE CONTROL, MAXIMUM FEEDBACK
FIGURE 4



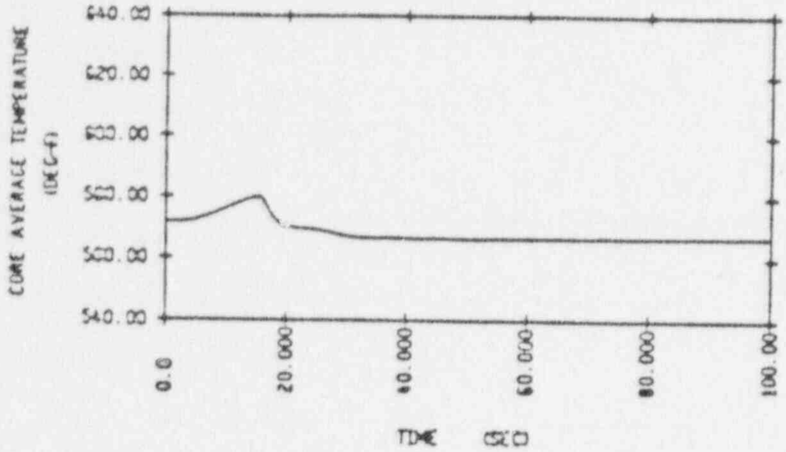
LOSS OF LOAD WITHOUT PRESSURE CONTROL, MINIMUM FEEDBACK
FIGURE 5

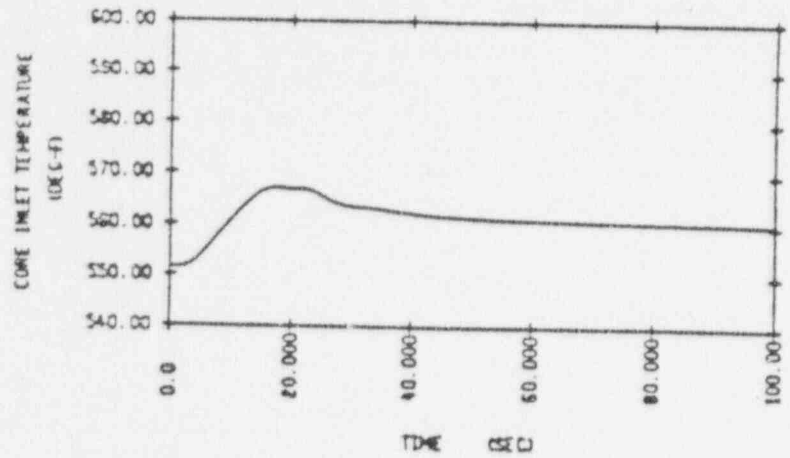


LOSS OF LOAD WITHOUT PRESSURE CONTROL, MINIMUM FEEDBACK
FIGURE 6



LOSS OF LOAD WITHOUT PRESSURE CONTROL, MAXIMUM FEEDBACK
FIGURE 7





LOSS OF LOAD WITHOUT PRESSURE CONTROL, MAXIMUM FEEDBACK
FIGURE 8