STEAM GENERATOR TUBE PLUGGING INCREASE LICENSING REPORT FOR GINNA NUCLEAR POWER STATION

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1.0 INTRODUCTION

Currently, safety analyses have been performed for Ginna with a uniform steam generator tube plugging (SGTP) level of 12% for the Loss-of-Coolant Accidents (LOCA) and a uniform level of 10% for the non-LOCA events (References 1,2). Based on the results of these analyses the maximum SGTP level for one steam generator (SG) is 12% while the maximum plant SGTP level is 10% (average of SGTP level in SG A and B). The actual SGTP level is 4.1% and 8.4% for steam generator A and B respectively. In anticipation of additional SG tube degradation in Cycle 17 operation and the subsequent need to sleeve and/or plug tubes, the safety analyses have been reviewed with respect to an increase in the SGTP level to 15%. This report will describe the impact of an increase in SGTP level to 15% on the Power Capability parameters (RCS flow, temperature, etc.), Thermal-Hydraulic Design, UFSAR Safety Analyses, and Technical Specifications.

In general, transients which were found to be sensitive to SGTP increases were reanalyzed and the results are presented in Section 5. Reanalysis was based on the identical analysis methods, computer codes, and assumptions as the Reload Transition Safety Report (RTSR); any exceptions are noted in the discussion of each event. For transients where SGTP increases have a small or negligible impact an evaluation was performed. The results of the evaluations are described in Section 4. In addition, those transients which are DNB limited and are impacted by the reduction in RCS flow associated with the increase in SGTP level were evaluated rather than reanalyzed with current DNB margin allocated to cover the DNB penalty associated with the increase in SGTP level (see Sections 3 and 4).

Table 1.0-1 gives a summary of the safety analyses and denotes those events reanalyzed or evaluated for an increase in SGTP.

TABLE 1.0-1

UFSAR ACCIDENTS ADDRESSED

FOR SGTP LEVEL INCREASE TO 15%

Section	Accident	Reanalysis
15.1.1	Excessive Heat Removal Due to Feedwater Temperature Decrease	
15.1.3	Excessive Load Increase	*
15.1.4	Inadvertent Opening of a SG Relief/Safety Valve	
15.1.5	Rupture of a Steam Pipe	
15.2.2	Loss of Electrical Load	*
15.3.1	Loss of Reactor Coolant Flow	
15.3.2	Locked Rotor	*
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15.4.2	Uncontrolled RCCA Withdrawal at Power	*
15.4.4	Chemical and Volume Control System Malfunction	*
15.4.5	Rupture of a Control Rod Mechanism Housing - RCCA Ejection	*
15.4.6	Rod Cluster Control Assembly (RCCA) Drop	
15.6.3	Steam Generator Tube Rupture	
15.6.4	Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)	*(1)

* Denotes that a reanalysis was performed

(1) Limiting large break case only

2.0 POWER CAPABILITY PARAMETERS

Design power capability parameters which were used as a basis for the evaluations and reanalyses for a uniform SGTP level of 15% are defined in Table 2.0-1. These parameters were calculated based on the assumption that the current vessel average temperature of 573.5°F is maintained. Column 1 lists the parameters assumed in the RTSR non-LOCA analyses corresponding to a SGTP level of 10%. Column 2 lists the same parameters with a SGTP level of 15%. As can be seen by a comparison, the increase in SGTP level results in an approximate 2.2% reduction in RCS Thermal Design flow. For conservatism, an approximate 3.0% reduction in Minimum Measured flow is used in analyses which employ the Improved Thermal Design Procedure (Reference 6). Since RCS average temperature is maintained the steam generator secondary pressure is calculated to drop. Vessel inlet temperatures drops and outlet temperature increases slightly resulting in an approximate 1.1°F increase in vessel ΔT.

TABLE 2.0-1
POWER CAPABILITY PARAMETERS
FOR LEVELS OF SGTP

Parameter	10% Plugging	15% Plugging
Thermal Output of NSSS (Mwt)	1520	1520
Core Inlet Temperature (°F)	543.7	543.1
Vessel Average Temperature (°F)	573.5	573.5
Vessel Outlet Temperature (°F)	603.4	603.9
RCS Pressure (PSIA)	2250	2250
Reactor Coolant Loop Flow (gpm)		
(Thermal Design)	87000	85100
(Minimum Measured Flow)	89600	86900
Steam Flow (10^6 LBM/HR)	6.58	6.61
SG Pressure (PSIA)	746.5	727

3.0 THERMAL HYDRAULIC DESIGN IMPACT

3.1 Introduction

The calculational methods presently employed for the R. E. Ginna thermal-hydraulic analysis are (1) the THINC-IV computer code (References 4 and 5), (2) the Improved Thermal Design Procedure (ITDP) (Reference 6), and, (3) the WRB-1 correlation (Reference 7) for the Westinghouse OFA fuel and the W-3 correlation for the Exxon fuel.

The THINC-IV computer program, approved by the NRC (Reference 8), is used to perform thermal-hydraulic calculations. The THINC-IV code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure, and DNBR distributions along flow channels within a reactor core under all expected operating conditions.

The design method employed to meet the DNB design basis is the ITDP, Reference 6. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR for the limiting power rod will be greater than or equal to the correlation limit DNBR. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. In addition, the limit DNBR values are increased to values designated as safety analysis limit DNBRs. The plant allowance available between the safety analysis limit DNBR values and the design limit DNBR is used for flexibility in the design and operation of the plant.

The WRB-1 DNB correlation is employed in the thermal-hydraulic design of the Westinghouse OFA fuel. The WRB-1 correlation provides a significant improvement in critical heat flux predictions over previous DNB correlations. The W-3 correlation is used in the design of the Exxon fuel.

The Exxon fuel assemblies currently in the core are at least thrice burned. The peak assembly power of Exxon fuel will be extremely low due to the fuel burndown effect. These assemblies cannot be limiting from a DNB standpoint. Therefore, the Exxon fuel is excluded in the evaluation of impact of steam generator tube plugging level increase to 15%.

3.2 Impact of SGTP Level Increase on Thermal-Hydraulic Design

The 15% plugging of steam generator tubes leads to a 3% reduction in the R. E. Ginna reactor coolant system flow. The flow reduction is offset by using available plant DNBR margin which is defined by the following equation.

$$\text{Safety Analysis DNBR Value} = \frac{\text{Design DNBR Value}}{1 - \text{Margin}}$$

The table below indicates the relationship between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values for Westinghouse 14x14 OFA fuel in R. E. Ginna.

	Typical	Thimble
Correlation Limit	1.17	1.17
Design Limit	1.34	1.33
Safety Analysis Limit	1.52	1.51
DNBR Margin	12%	12%

This DNBR margin is more than sufficient to accommodate the penalties described below including the penalty incurred due to flow reduction caused by the increase in steam generator tube plugging level.

The plant-specific DNBR sensitivity to the RCS flow rate is used to calculate the DNBR penalty to accommodate the flow reduction. By using the DNBR margin, the thermal and hydraulic design basis (References 1 and 2) and the core DNB limits remain unchanged. However, the vessel exit boiling limits are slightly reduced due to the flow reduction in order to maintain the same heat output. The corresponding setpoints change is made in Section 6 based on the new core limit lines. Figure 3.0-1 shows the revised core limits which are used in the safety analyses.

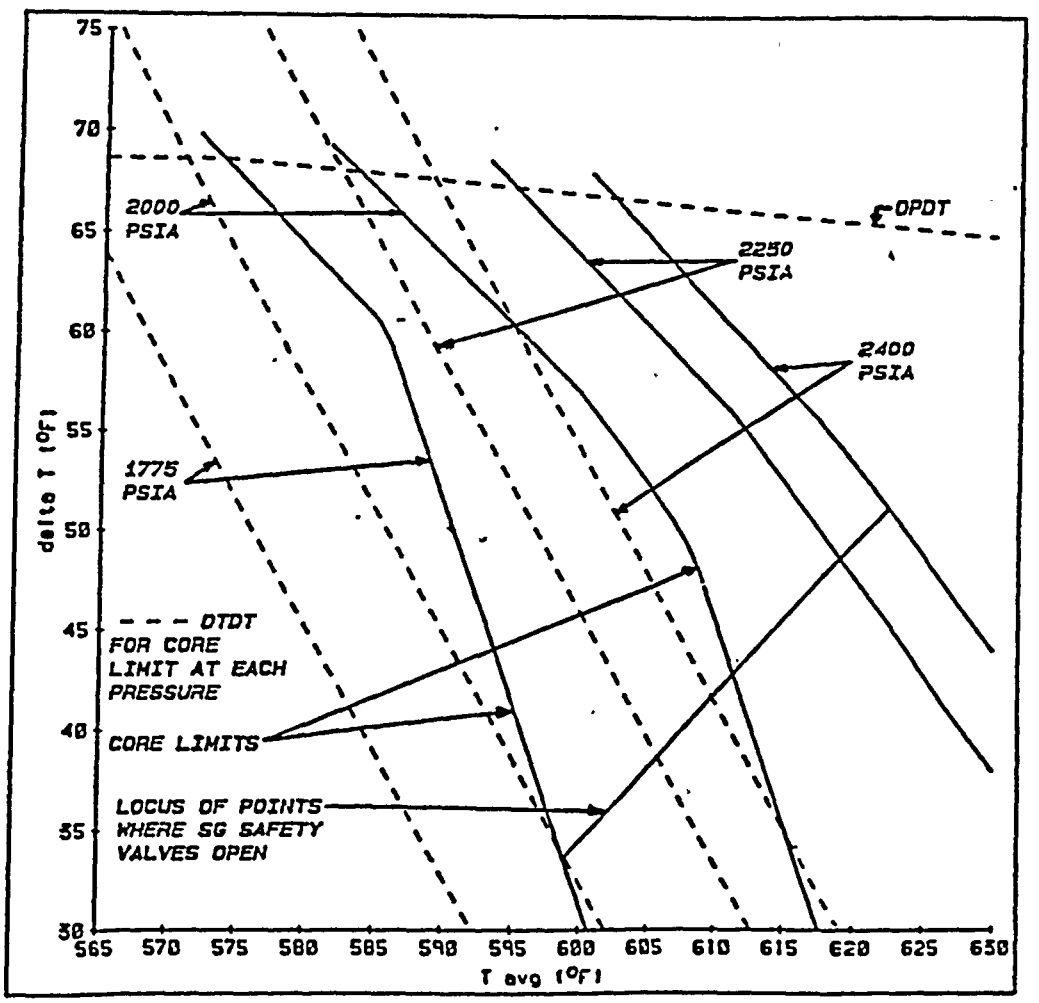
The new Westinghouse rod bow methodology described in Reference 9 is applied to calculate the rod bow penalty for R. E. Ginna. This methodology takes a statistical approach in calculating the DNBR penalty. The effect of rod bow on DNB is only considered up to a certain burnup. Beyond this burnup, credit is taken for an F-delta-H burndown effect in which the fuel is not capable of achieving limiting peaking factors due to the decrease in fissionable isotopes and the buildup of fission product inventory per Reference 10.

A 2% transition core DNBR penalty on the Westinghouse OFA fuel is still imposed since there will be a few Exxon fuel assemblies in the core.

FIGURE 3.0-1

GINNA CORE LIMITS AND

OVERPOWER - OVERTEMPERATURE ΔT SETPOINTS



4.0 ACCIDENTS EVALUATED FOR A 15% SGTP LEVEL

The transients described in the following sections were not reanalyzed for an increase in the SGTP level to 15%. All evaluations were performed for a uniform plugging level of 15%.

The small break LOCA event was not reanalyzed because the results are not sensitive to a SGTP level increase from 12% to 15%. The SGTR event evaluation is based on the limiting case currently presented in the UFSAR. For the non-LOCA events, available DNB margin was allocated to cover the DNB penalty associated with the reduction in flow. In addition, the non-LOCA events have been evaluated with respect to maximum asymmetric plugging considerations (ΔSGTP level of 11% between the A and B steam generators). The results of the evaluations show that this level of asymmetry is acceptable since the applicable safety criteria, as described in the RTSR, continue to be met.

4.1 Small Break LOCA

The small break Loss-Of-Coolant Accident (LOCA) analysis which currently forms the licensing basis was performed using the evaluation model in Reference 3. That analysis assumed a steam generator tube plugging level of 12%. Previous evaluations which have considered the effect of steam generator tube plugging on the small break analysis have considered three major effects:

1. The reduction in steam generator tube surface area and the subsequent effect this has on the ability of the secondary side to remove heat from the primary.

2. The effect of tube plugging on primary and secondary initial temperatures.

3. The effect that steam generator tube plugging has on the draining of the tubes, since this has a direct impact on the vessel water inventory.

The basis for evaluating these three effects is analyses which were performed and reported in References 22 and 23. The conclusion is that no effect would be expected in the small break analysis for tube plugging levels of up to 15%. For this particular case, the evaluation need only consider an increase in tube plugging from 12 to 15%. Based on the insensitivity demonstrated in the analyses in References 22 and 23 and the fact that the current limiting small break results for 12% tube plugging demonstrate 1108°F margin to the regulatory limits in 10CFR 50.46, the current analysis remains valid. In conclusion, an increase of 3% in steam generator tube plugging would have no effect on those small break LOCA results presented in the UFSAR.

4.2 Uncontrolled RCCA Withdrawal From a Subcritical Condition

A RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in a power excursion.

The rod cluster drive mechanisms are wired into preselected groups, and these group configurations are not altered during core life. The rods are therefore physically prevented from withdrawing in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming the simultaneous withdrawal of the combination of the two rod groups with maximum combined worth at maximum speed.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise, terminated by the reactivity feedback effect of the negative Doppler coefficient. This self limitation of the power excursion is of primary importance, since it limits the power to a tolerable level during the delay time for protective action.

An analysis is performed to verify that the minimum DNBR remains above the limit value at all times during the transient. A reduction in Thermal Design

flow of 2.2% will not cause the transient statepoint inlet temperature to change since the loop transport time is significantly longer than the time from which heat is added to the coolant and the time the minimum DNBR is reached. Any increase in core pressure (expected to be small) would be considered a DNB benefit. Additionally, the reduction in flow would cause the heat transfer film coefficient as calculated by FACTRAN to be reduced, which in turn would cause a slight reduction in the maximum heat flux used in the THINC DNB calculation. Thus, with the exception of the reduction in flow, the transient statepoints are not adversely affected due to the increase in SGTP level.

As discussed in Section 3.0, a review of the available DNB margin reveals that sufficient margin exists to accommodate the flow reduction associated with up to 15% steam generator tube plugging. Therefore, the minimum DNBR remains above the limit value and the conclusions presented in the RTSR remain valid.

4.3 Rod Cluster Control Assembly (RCCA) Drop

Dropping of a full length RCCA occurs when the drive mechanism is deenergized. This would cause a power reduction and an increase in the hot channel factor. If no protective action occurred, the Reactor Control System would restore the power to the level which existed before the incident. This would lead to a reduced safety margin or possible DNB, depending upon the magnitude of the resultant hot channel factor.

A rod drop signal from any rod position indication channel, or from one or more of the four power range channels, initiates the following protective action: reduction of the turbine load by a preset adjustable amount and blocking of further automatic rod withdrawal. The turbine runback is achieved by acting upon the turbine load limit and on the turbine load reference. The rod withdrawal is redundantly achieved.

Two cases are presented in Section 14.1.4 of the RTSR. The analysis is performed in which the load cutback very closely matches the power decrease from the negative reactivity for a dropped rod (800 pcm) and also for the case

in which the load cutback is greater than that required to match the worth of the dropped rod (100 pcm). In both cases the minimum DNBR statepoint Tavg and pressure are insensitive to small changes in flow. Heat flux is a function of core power, which is a function of the dropped rod worth, the turbine runback setpoint, and to a lesser extent, the moderator and Doppler feedback. Therefore, the heat flux will not change and there will be no adverse changes to the minimum DNBR statepoint except for the reduction in flow.

In addition, the minimum DNBR statepoints associated with a statically misaligned rod are not adversely impacted by the flow reduction.

As discussed in Section 3.0, a review of the available DNB margin reveals that sufficient margin exists to accommodate the 3.0% flow reduction associated with up to 15% steam generator tube plugging. Therefore, the minimum DNBR remains above the limit value and the conclusions presented in the RTSR remain valid.

4.4 Loss of Reactor Coolant Flow

A loss-of-coolant flow incident can result from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect is a rapid increase in coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

In section 14.1.6 of the RTSR two flow coastdown cases are analyzed:

1. Two reactor coolant pumps coasting down, with the reactor at full power,

2. One reactor coolant pump coasting down, with two loops operating and the reactor at full power.

Of the flow coastdown accidents case 1 is the most limiting. The analysis is performed using the Improved Thermal Design Procedure (ITDP) to verify that the minimum DNBR remains above the limit value at all times during the transient. As seen in Table 14.1.6-1 of the RTSR, the point of minimum DNBR occurs within 3 seconds after initiation of the flow coastdown. A reduction in flow of 3.0% will not cause the transient statepoint inlet temperature to change since the loop transport time is significantly longer than 3 seconds. Any increase in core pressure (expected to be small) would be considered a DNB benefit. Additionally, the reduction in flow would cause the film coefficient as calculated by FACTRAN to be reduced, which in turn would cause the statepoint heat flux used in the THINC DNB calculation to be decreased. Thus, with the exception of the reduction in flow, the transient statepoints would not be adversely effected due to the increase in SGTP level.

As discussed in Section 3.0, a review of the available DNB margin reveals that sufficient margin exists to accommodate the 3.0% flow reduction associated with up to 15% steam generator tube plugging. Therefore, the minimum DNBR remains above the limit value and the conclusions presented in the RTSR remain valid.

4.5 Excessive Heat Removal Due to Feedwater Temperature Decrease

The reduction in feedwater enthalpy is another means of increasing core power above full power. Such increases are attenuated by the thermal capacity in the secondary plant and in the Reactor Coolant System.

An extreme example of excess heat removal by the feedwater system is the transient associated with the accidental opening of the feedwater bypass valve which diverts flow around the low pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost.

In the event of an accidental opening there is a sudden reduction in inlet feedwater temperature to the steam generators. The increased subcooling will

create a greater load demand on the primary system which can lead to a reactor trip. The net effect on the RCS due to a reduction in feedwater enthalpy is similar to the effect of increasing secondary side steam flow, i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator delta-T.

Two cases are analyzed in Section 14.1.10 of the RTSR:

1. Reactor Control in manual with maximum moderator feedback
2. Reactor Control in automatic with maximum moderator feedback

In addition, the maximum reactivity insertion rate at no load for a step increase in feedwater flow to one SG was calculated.

The reduction in steam generator tube heat transfer surface area due to 15% tube plugging will reduce the rate of subcooling caused by this event. This means that the reactivity insertion rate will be lower. Thus, the transient will proceed at a slower rate until a steady state condition is reached. The steady state condition reached is determined primarily by the secondary side temperature reduction which in turn determines the reduction in the primary coolant average temperature. This is not sensitive to a small change in coolant flow. Thus, the transient statepoints will not be adversely effected by the small reduction in RCS flow.

As discussed in Section 3.0, a review of the available DNB margin reveals that sufficient margin exists to accommodate the 3.0% flow reduction associated with up to 15% steam generator tube plugging. The minimum DNBR remains above the limit value. In addition, the maximum reactivity insertion rate for zero power is less than the insertion rate analyzed for the Uncontrolled RCCA withdrawal from subcritical event. Therefore, the conclusions presented in the RTSR remain valid.

4.6 Rupture of A Steam Pipe (Core Response)

The steamline rupture is analyzed at hot zero power, end-of-life conditions for the following cases:

a) Double ended rupture of a steam pipe at the steam generator outlet, with offsite power, 2 loops operating. The equivalent break area is 4.6 sq. ft.

b) Case (a) above with loss of offsite power simultaneous with the steam break.

c) Case (a) above with one loop in service.

d) A break equivalent to steam release through one steam generator safety valve with offsite power, 2 loops operating.

e) Case (d) above with one loop in service.

The steamline rupture results in a rapid depressurization of the steam generators and a primary side cooldown. This causes a large reactivity insertion due to the presence of a negative moderator temperature coefficient. An increase in the steam generator plugging level to 15% will reduce the reactor coolant flow which will result in a reduction in heat transfer from the fuel to the coolant. Additionally, tube plugging will reduce the steam generator heat transfer surface area which will reduce the heat transfer from the primary side to the secondary side. Therefore, the reactivity insertion and return to power would be less limiting than the cases presented in Section 14.2.5 of the RTSR.

The most limiting of all cases reported in the RTSR is Case (a) above. For Case (a), the time of safety injection actuation is unaffected by an increase in the plugging level. This coupled with a slower return to power would result in a reduction in the peak core power from the RTSR results.

The analysis performed in Section 14.2.5 of the RTSR verifies that the minimum DNBR remains above the limit value at all times during the transient. As discussed in Section 3.0, a review of the available DNB margin in the steam line break event presented in the RTSR reveals that sufficient margin exists to accommodate up to 15% steam generator tube plugging taking no credit for the DNB benefit resulting from the peak core power reduction. Therefore, the minimum DNBR remains above the limit value and the conclusions presented in the RTSR remain valid.

4.7 Rupture of a Steam Pipe (Mass/Energy Release Inside Containment)

Steamline rupture mass and energy (M/E) release to containment analyses have been performed for R. E. Ginna. These analyses have been performed at hot zero and hot full power, end-of-life conditions with offsite power available for the following cases:

a) Full double ended rupture of a steam pipe at the steam generator outlet.

b) Small double ended rupture of a steamline.

c) Split steamline rupture.

Steamline ruptures occurring inside containment may result in significant releases of high energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. Primary factors in determining the M/E release inside containment are the secondary side mass available for release to containment and the primary side heat to be transferred to the released mass.

Steam generator tube plugging will result in a slight reduction of the secondary side mass which is available for release to containment and will reduce the reactor coolant flow which will result in a reduction in heat transfer from the fuel to the coolant. Additionally, tube plugging will

reduce the steam generator heat transfer surface area which will reduce the heat transfer from the primary side to the secondary side during the blowdown. Therefore, the rate of heat transfer from the primary to the secondary side with 15% steam generator tube plugging conditions will be less limiting than in the M/E cases previously analyzed.

Since the mass available for release inside containment is slightly reduced and the rate of heat transfer from primary to secondary is reduced, the M/E release rates previously calculated will bound those which would occur with 15% steam generator tube plugging. Thus, the containment pressure response previously calculated will not be adversely impacted by up to 15% steam generator tube plugging.

4.8 Steam Generator Tube Rupture (SGTR) Analysis

An evaluation was performed to estimate the effects of a 15% SG tube plugging level on the UFSAR SGTR mass releases data for Ginna. The Ginna UFSAR SGTR analysis was performed using the LOFTRAN program. The primary to secondary break flow was assumed terminated at 30 minutes (Case 1) and 60 minutes (Case 2) after initiation of the SGTR event. Since the offsite doses for Case 2 is significantly higher than the doses for Case 1, the impacts of a 15% tube plugging level are evaluated based on the Case 2. As a result of the reduced RCS cold leg and SG temperatures due to the increased SG tube plugging level, the primary-to-secondary break flow is slightly increased. The time of reactor trip and time of SG overfill also occur earlier, which in turn leads to a slight increase in water released through the ruptured SG safety valve. The increase in primary-to-secondary carryover and water release through the ruptured SG safety valve are estimated to result in a less than 1% increase in offsite radiation doses. Since the offsite doses with the effects of a 15% SG tube plugging level are still less than the 10CFR100 limits, no reanalysis or UFSAR changes are required.

5.0 ACCIDENTS REANALYZED FOR A 15% SGTP LEVEL

For each event reanalyzed the basic assumptions, methods, and computer codes remain the same as those found in the UFSAR and RTSR. All accidents reanalyzed were performed with a uniform SGTP level of 15%. In general, a higher SGTP level gives more conservative results.

The SGTP increase affects the initial conditions (T_{in}, flow, etc) as shown in Table 2.0-1. In addition primary tube flow area, volume, and heat transfer surface area values have been modified for a 15% SGTP level.

All reactor trip setpoints remain the same with the exception of the overtemperature ΔT and overpower ΔT functions which change as a result of the new core limit protection lines. See Figure 3.0-1.

5.1 Large Break Loss-Of-Coolant Accident (LOCA)

The analysis specified by 10 CFR 50.46, "Acceptance Criteria for, Emergency Core Cooling Systems for Light Water Power Reactors", Reference 14, is presented in this section. The analytical techniques used are in compliance with Appendix K of 10 CFR 50 and are described in the listed references. The basic assumptions, methods, and computer codes remain the same as those found in the RTSR.

Should a major break occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. Reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. A safety injection system signal is actuated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

2. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50. Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

When the reactor coolant system pressure falls below 715 psia the accumulators begin to inject borated water. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10 CFR 50.

Core Power Transient During Blowdown

The core power transient during blowdown for large breaks is evaluated using the SATAN-VI computer code. This code is discussed in detail in WCAP-8306, Reference 15.

Thermal Analysis

Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss of coolant accident including the double ended severance of the largest reactor cooling system cold leg pipe. The reactor core and internals together with the emergency core cooling system are designed so that the reactor can be safely shutdown and the essential heat transfer geometry of the core preserved following the accident.

The emergency core cooling system, even when operating during the injection mode with the most severe single failure, is designed to meet the Acceptance Criteria.

Method of Thermal Analysis

The description of the various aspects of the LOCA analysis is given in the listed references in UFSAR Section 15.6.4.2. This document describes the major phenomena modeled, the interfaces among the computer codes and features of the codes which maintain compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, and LOCTA-IV codes used in this analysis are described in detail in WCAP-8306, Reference 15, WCAP-8171, Reference 17 and WCAP-8305, Reference 16, respectively. The containment parameters used in the containment analysis code to determine the ECCS backpressure are the same as those presented in Table 14.3.2-3 of the RTSR. The containment pressure analysis code (COCO) is described in WCAP-8326, Reference 18.

The large break analysis was performed with the NRC Approved 1981 Version of the Evaluation Model, Reference 19, which includes modifications delineated in WCAP-9220-P-A and WCAP-9221-A (1981), and complies with the requirements of 10 CFR 50.46. The analysis was performed for an assumed steam generator tube plugging level of 15% and a reactor coolant system loop flow rate of 85,100 gpm on the current limiting case DECLG (C_D=0.4). For an increase in SGTP from 12% to 15% studies have shown that the limiting break size (discharge coefficient) will not change.

Results

Table 5.1-2 presents the peak clad temperatures and hot spot metal reaction for reanalysis of the DECLG (CD=0.4) case. Table 5.1-1 presents the sequence of events.

The analysis of the loss of coolant accident is performed at 102% of rated core power. The peak linear power, and core power used in the reanalysis are given in Table 5.1-2. The equivalent core parameters at the license application power level are also shown in Table 5.1-2. Since there is margin between the value of the peak linear power density used in this analysis and the value expected in operation, a low peak clad temperature would be obtained by using the peak linear power density expected during operation.

For the results discussed below, the hot spot is defined to be the location of maximum peak clad temperature. This location is given in Table 5.1-2.

Tables 5.1-4 through 5.1-6 present reflood mass and energy releases, blowdown mass and energy releases, and the broken loop accumulator mass and energy release to the containment. Table 5.1-3 presents the containment data used in the reanalysis, which is the same as that used in the RTSR. Figures 5.1-1 through 5.1-17 present the transients for the principal parameters for the case reanalyzed. The following items are noted:

Figures 5.1-1 through 5.1-3

Quality, mass velocity, and clad heat transfer coefficient for the hotspot and burst locations.

Figures 5.1-4 through 5.1-6

Core pressure, break flow, and core pressure drop. The break flow is the sum of the flowrates from both ends of the guillotine break. The core pressure drop is taken as the pressure just before the core inlet to the pressure just beyond the core outlet.

Figures 5.1-7 through 5.1-9

Clad temperature, fluid temperature, and core flow. The clad and fluid temperatures are for the hotspot and burst locations.

Figures 5.1-10

Reflood Transient - Core Inlet Velocity

Figure 5.1-11

Reflood Transient - Core and Downcomer Water Levels

Figures 5.1-12 and 5.1-13

Emergency core cooling system flowrates, for both accumulator and pumped safety injection.

Figure 5.1-14

Containment pressure.

Figure 5.1-15

Core power.

Figures 5.1-16 and 5.1-17

Break energy release during blowdown and the containment wall condensing heat transfer coefficient.

The clad temperature analysis is based on a total peaking factor of 2.32. The hot spot metal-water reaction reached is 2.8% which is well below the embrittlement limit of 17% as required by 10 CFR 50.46. In addition, the total core metal-water reaction is less than 0.3% for all breaks as compared with the 1% criterion of 10 CFR 50.46.

Conclusions

For breaks up to and including the double ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time as required by the long-lived radioactivity remaining in the core.

Based on the effect of upper plenum injection for Westinghouse design 2 loop plants, a 6°F increase in peak clad temperature results from assuming 14 x 14 OFA fuel. The methodology employed to develop this penalty was identical to that performed for previous LOCA analyses performed for the plant (References 20 and 21). In addition, a 10°F mixed core penalty is included since the core contains both Westinghouse and Exxon fuel assemblies. Utilizing the present Westinghouse ECCS evaluation models to analyze a postulated LOCA for a SGTP level of 15% results in a final peak clad temperature of 1887°F including both penalties. It can be seen from the results contained herein that this ECCS analysis for R. E. Ginna remains in compliance with 10CFR 50.46 and the conclusions in the RTSR remain valid.

5.2 Uncontrolled RCCA Withdrawal at Power

An uncontrolled RCCA withdrawal at power results in an increase in core heat flux. Since the heat extraction from the steam generator remains constant, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, this power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, to prevent the possibility of damage to the cladding, the Reactor Protection System is designed to terminate any such transient with an adequate margin to DNB.

The automatic features of the Reactor Protection System which prevent core damage during a rod withdrawal accident at power include the following:

1. Nuclear power range instrumentation actuates a reactor trip if two out of the four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of four delta-T channels exceed an overtemperature delta-T setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of four delta-T channels exceed an overpower delta-T setpoint This setpoint is automatically varied with axial power imbalance and coolant temperature to ensure that the allowable heat generation rate kw/ft is not exceeded.

4. A high pressure reactor trip actuated from any two out of three pressure channels is set at a fixed point. This set pressure will be less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip. actuated from any two out of three level channels, is actuated at a fixed setpoint. This affords additional protection for RCCA withdrawal accidents.

The manner in which the combination of overpower and overtemperature delta-T trips provides protection over the full range of reactor coolant system conditions is illustrated in Figure 3.0-1. Figure 3.0-1 presents allowable reactor loop average temperature and delta-T for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower delta-T trip and the overtemperature delta-T trip are represented as "protection lines" on this diagram. These protection lines are drawn to include all adverse instrumentation and setpoint errors, so that under nominal conditions trip would occur well within the area bounded by these lines. These protection lines bound a plant SGTP level of 15%. A maximum steady-state operating condition for the reactor is also shown on the Figure.

The utility of the diagram just described is in the fact that the operating limit imposed by any given DNB ratio can be represented as a line on this coordinate system. The DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.62 for the thimble cell and 1.54 for the typical cell). For conservatism, the DNB lines remain the same as those in Figure 15.0-1 of the UFSAR, even though the Exxon fuel will not longer be limiting due to its high burnup (see Section 3.2). All points below and to the left of this line have a DNB ratio greater than this value. The diagram shows that DNB is prevented for all cases if the area enclosed within the maximum protection lines is not traversed by the applicable DNB ratio line at any point.

The region of permissible operation (power pressure and temperature) is completely bounded by the combination of reactor trips: nuclear overpower (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature delta-T (variable setpoints). These trips are designed to prevent overpower and a DNB ratio of less than the limit value.

Method of Analysis

Uncontrolled rod cluster control assembly bank withdrawal is analyzed by the LOFTRAN (Reference 12) code. This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level. The core limits, as illustrated in Figure 3.0-1, are used as input to LOFTRAN to determine the minimum departure from nucleate boiling ratio during the transient. This accident is analyzed with the Improved Thermal Design Procedure as described in Reference 6.

In order to obtain conservative values of departure from nucleate boiling ratio, the following assumptions are made:

1. Initial Conditions - Initial reactor power, reactor coolant average temperatures, and reactor coolant pressure are assumed to be at their

nominal values consistent with a SGTP level of 15%. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 6.

2. A uniform SGTP level of 15% is assumed.

3. Reactivity Coefficients - Two cases are analyzed.

   a. Minimum Reactivity Feedback - A positive (5 pcm/°F) moderator coefficient of reactivity is assumed, corresponding to the beginning-of-core-life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed.

   b. Maximum Reactivity Feedback - A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

4. The rod cluster control assembly trip insertion characteristic is based or the assumption that the highest worth assembly is stuck in its fully withdrawn position.

5. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of nominal full power. The overtemperature delta-T trip includes all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values.

6. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at maximum speed.

The effect of rod cluster control assembly movement on the axial core power distribution is accounted for by causing a decrease in overtemperature and overpower delta-T trip setpoints proportional to a decrease in margin to DNB.

Results

Figures 5.2-1 through 5.2-6 show the plant response (including neutron flux, pressure, average coolant temperature, and departure from nucleate boiling ratio) to a rapid rod cluster control assembly withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in Tavg and pressure result, and a large margin to DNB is maintained.

The plant response for a slow control rod assembly withdrawal from full power is shown in Figures 5.2-7 through 5.2-12. Reactor trip on overtemperature delta-T occurs after a longer period, and the rise in temperature and pressure is consequently larger than for rapid rod cluster control assembly withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 5.2-13 shows the minimum departure from nucleate boiling ratio as a function of reactivity insertion rate from initial full-power operation for the minimum and maximum reactivity feedback cases. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature delta-T trip channels. The minimum DNBR is never less than the limit value.

Figures 15.2-14 and 5.2-15 show the minimum departure from nucleate boiling ratio as a function of reactivity insertion rate for rod cluster control assembly withdrawal incidents starting at 60% and 10% power respectively. The results are similar to the 100% power case, except that as the initial power is decreased, the range over which the overtemperature delta-T trip is effectively increased. In neither case does the departure from nucleate boiling ratio fall below the DNBR limit value.

In the referenced figures, the shape of the curves of minimum departure from nucleate boiling ratio versus reactivity insertion rate is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

In addition to trips on high neutron flux and overtemperature delta-T, credit is taken for the high pressurizer water level trip to prevent filling the pressurizer for slow reactivity insertion rates (< 2.0×10^{-4} delta-k/second). For these cases the high level trip is reached before the overtemperature delta-T trip. While the overtemperature delta-T trip protects the core against DNB for these insertion rates, it is not reached soon enough to prevent pressurizer filling. A conservatively high setpoint of 100% span with a trip delay time of 2.0 seconds is assumed for the high pressure level trip function.

Referring to Figure 5.2-15, for example, it is noted that:

1. For high reactivity insertion rates (i.e., between ~1×10^{-3} delta-k/second and ~3.0×10^{-4} delta-k/second), reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates, while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to a significant increase in heat flux or water temperature with resultant high minimum departure from nucleate boiling ratios during the transient. Within this range, as the reactivity insertion rate decreases core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNBR during the transient thus decreases with decreasing insertion rate.

2. With further decrease in reactivity insertion rate, the overtemperature delta-T and high neutron flux trips become equally effective in terminating the transient (e.g., at ~3.0×10^{-4} delta-k/second reactivity insertion rate).

   The overtemperature delta-T reactor trip circuit initiates a reactor trip when measured coolant loop delta-T exceeds a setpoint based on measured reactor coolant system average temperature and pressure. It is important in this context to note, however, that the average temperature contribution to the circuit is lead-lag compensated in

order to decrease the effect of the thermal capacity of the reactor coolant system in response to power increases.

For reactivity insertion rates between ~3.0×10^{-4} delta-k/second and ~6.0×10^{-6} delta-k/second, the effectiveness of the overtemperature delta-T trip increases (in terms of increased minimum departure from nucleate boiling ratio) due to the fact that, with slower insertion rates, the power increase rate is slower, the rate of rise of average coolant temperature is slower, and the lead-lag compensation provided can increasingly account for the coolant system thermal capacity lag.

3. For maximum reactivity feedback cases reactivity insertion rates less than ~5.0×10^{-4} delta-k/second, the rise in reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening these valves, which act as an additional heat load on the reactor coolant system, sharply decreases the rate of rise of reactor coolant system average temperature. This decrease in rate of rise of the average coolant system temperature during the transient is accentuated by the lead-lag compensation, causing the overtemperature delta-T trip setpoint to be reached later with resulting lower minimum departure from nucleate boiling ratios.

Figures 5.2-13, 5.2-14, and 5.2-15 illustrate minimum departure from nucleate boiling ratio calculated for minimum and maximum reactivity feedback. The calculated sequence of events for this accident is shown in Table 5.2-1.

Conclusions

In the unlikely event of a control rod withdrawal incident, from full power operation or lower power levels, the core and reactor coolant system are not adversely affected for an increase in SGTP level to 15% since the minimum value of DNB ratio reached is in excess of the DNB limit value for all rod reactivity rates. Protection is provided by nuclear flux overpower,

overtemperature delta-T, and high pressurizer level reactor trips. Additional protection would be provided by the overpower delta-T and the high pressure reactor trips. The preceding sections have described the effectiveness of these protection channels.

5.3 Chemical and Volume Control System Malfunction

Reactivity can be added to the core with the Chemical and Volume Control System by feeding reactor makeup water into the Reactor Coolant System via the reactor makeup control system. The normal dilution procedures call for a limit on the rate and magnitude for any individual dilution under strict administrative controls. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time. The Chemical and Volume Control System is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

There is only a single common source of reactor makeup water to the Reactor Coolant System from the reactor makeup water storage tank, an inadvertent dilution can be readily terminated by isolating this single source. The operation of the reactor makeup water pumps which take suction from this tank provides the only supply of makeup water to the Reactor Coolant System. In order for makeup water to be added to the Reactor Coolant System the charging pumps must be running in addition to the reactor makeup water pumps.

The rate of addition of unborated water makeup to the Reactor Coolant System is limited to the capacity of the makeup water pumps. This limiting addition rate is 120 gpm for two reactor makeup water pumps. For totally unborated water to be delivered at this rate to the Reactor Coolant System at pressure, two charging pumps must be operated at full speed. Normally, two charging pumps are operating at half speed, while the third pump is idle.

The boric acid from the boric acid tank is blended with the reactor makeup water in the blender and the composition is determined by the present flow rates of boric acid and reactor makeup water on the Reactor Makeup Control. Two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode. Second, the start button must be depressed. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very small.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

To cover all phases of plant operation, boron dilution during refueling, startup, and power operation are considered. An increase in SGTP level affects the boron dilution analyses which credit the primary side SG tube volume in calculating the time to lose shutdown margin. A reduction in volume results in a decrease in the available operator action time. The dilution during refueling case is not affected since the SG tube volume is not considered. Therefore only the cases during startup and power operation are reanalyzed.

Method of Analysis and Results

Dilution During Startup

Prior to refueling, the Reactor Coolant System is filled with borated water from the refueling water storage tank. Core monitoring is by external BF_3 detectors. Mixing of reactor coolant is accomplished by operation of the reactor coolant pumps. Again, the maximum dilution flow (120 gpm) is considered. The volume of reactor coolant is approximately 4687.8 ft^3 which is the volume of the Reactor Coolant System excluding the pressurizer. This volume accounts for 15 percent steam generator tube plugging. High source level and all reactor trip alarms are effective.

The minimum time required to reduce the reactor coolant boron concentration to 1500 ppm, where the reactor could go critical with all rods at the insertion limits, is about 63.2 minutes. Once again, this should be more than adequate time for operator action to the high count rate signal, and termination of dilution flow.

In any case, if continued dilution occurs, the reactivity insertion rate and consequences thereof are considerably less severe than those associated with the uncontrolled rod withdrawal from a subcritical condition.

Dilution at Power

For dilution at power, it is necessary that the time to lose shutdown margin be sufficient to allow identification of the problem and termination of the dilution. As in the dilution during startup case, the RCS volume reduction due to steam generator tube plugging is considered. The effective reactivity addition rate is a function of the reactor coolant temperature and boron concentration. The reactivity insertion rate calculated is based on a conservatively high value for the expected boron concentration at power (1500 ppm) as well as a conservatively high charging flow rate capacity (127 gpm). The reactor is assumed to have all rods out in either automatic or manual control. With the reactor in manual control and no operator action to terminate the transient, the power and temperature rise will cause the reactor to reach the reactor protection (i.e., Overtemperature Delta-T, high nuclear flux) trip setpoint, resulting in a reactor trip. After reactor trip there is at least 52.7 minutes for operator action prior to return to criticality. The boron dilution transient in this case is essentially the equivalent to an uncontrolled rod withdrawal at power. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be 1.6 pcm/sec and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. Prior to reaching the reactor protection system trip, the operator will have received an alarm on Overtemperature delta-T and turbine runback.

With the reactor in automatic control, a boron dilution will result in a power and temperature increase such that the rod controller will attempt to compensate by slow insertion of the control rods. This action by the controller will result in rod insertion limit and axial flux alarms. The minimum time to lose the shutdown margin from time of alarm at beginning of life would be greater than 43.2 minutes. The time would be significantly longer at end of life due to the low initial boron concentration.

Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered incredible. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

5.4 Locked Rotor Accident

A hypothetical transient analysis is performed for the postulated instantaneous seizure of a reactor coolant pump rotor. Flow through the reactor coolant system is rapidly reduced, leading to a reactor trip on a low-flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to heat up and expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two

power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the spray is not included in the analysis.

Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN (Reference 12) code is used to calculate the resulting loop core and flow transients following the pump seizure, the time of reactor trip based on loop flow transients, and the nuclear power following reactor trip, and to determine peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN (Reference 11) code which uses the core flow and nuclear power calculated by LOFTRAN. The FACTRAN code includes a film boiling heat transfer coefficient.

One case is analyzed with both loops operating and one locked rotor. At the beginning of the postulated locked rotor accident (i.e., at the time the shaft of one of the reactor coolant pumps is assumed to seize), the plant is assumed to be operating under the most adverse steady-state operating conditions with respect to the pressure. i.e.. maximum steady-state power level, maximum steady-state pressure (2280 psia), and maximum steady-state coolant average temperature.

The locked rotor event is not analyzed with a consequential loss of offsite power. At the R. E. Ginna plant, the generator breakers will not open until one minute after the loss of offsite power. Thus, power will be maintained to the intact reactor coolant pump throughout the limiting portion of the transient. This is within the first 10 seconds when the peak clad temperature occurs.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 30 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible coolant pressure during the

transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressurizer pressure response is shown in Figure 5.4-4. The point in the reactor coolant system having the maximum pressure is at the discharge of the reactor coolant pump and is conservatively estimated to be 50 psi higher than the pressurizer.

The locked rotor event is analyzed to show that the peak reactor coolant system pressure remains below 120 percent of design and that the peak clad temperature does not exceed 2700°F.

Evaluation of the Pressure Transient - After pump seizure, the neutron flux is rapidly reduced by control rod insertion effect. Rod motion is assumed to begin one second after the flow in the affected loop reaches 87% of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after plant trip.

Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are full open at 2575 psia, and their total capacity for steam relief is 20 ft^3/sec.

Evaluation of Departure from Nucleate Boiling in the Core During the Accident - For this accident, departure from the nucleate boiling is assumed to occur in the core, and therefore, an evaluation of the consequence with respect to fuel rod thermal transients is performed. Results obtained from analysis of this hot spot condition represent the upper limit with respect to clad temperature and zirconium-water reaction. In the evaluation, the rod power at the hot spot is assumed to be three times the average rod power (FQ=3) at the initial core power level.

Film Boiling Coefficient - The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation. The

fluid properties are evaluated at film temperature, which is the average between the wall and bulk temperatures. The program calculates the film coefficient at every time step, based on the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient, since they are the most conservative with respect to clad temperature response. For conservatism, departure from nucleate boiling was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient - The magnitude and the time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between the pellet and the clad. Based on investigations of the effect of the gap coefficient on the maximum clad temperature during the transient, the gap coefficient is assumed to increase from a steady-state value consistent with an initial fuel temperature to 10,000 Btu per hour-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial gap coefficent value is released to the clad at the initiation of the transient.

Zirconium-Steam Reaction - The zirconium-steam reaction can become significant above a clad temperature of 1800°F. The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction:

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp \frac{(45,000)}{1.986T}$$

where:

w = amount reacted (mg/cm**2)
t = time (seconds)
T = temperature (°F)

The reaction heat is 1,510 cal/gm.

Results

Figures 5.4-1 through 5.4-3 show the nuclear power, core flow, and loop flow transients and Figure 5.4-4 shows the pressurizer pressure transient. The heat flux and clad temperature transients are given in Figures 5.4-5 and 5.4-6. The results of these calculations are summarized in Table 5.4-1. The sequence of events is shown in Table 5.4-2.

Conclusions

Since the peak reactor coolant system pressure (2838 psia) reached during any of the transients is less than 120% of design pressure the integrity of the primary coolant system is not endangered. This value can be considered an upper limit, since the assumptions used in the analysis are conservative.

Since the peak clad surface temperature (2204°F) calculated for the hot spot during the more severe transient remains considerably less than 2700°F and the amount of zirconium-water reaction is small, the core remains in place and intact with no consequential loss of core cooling capability.

5.5 Loss of External Electrical Load

The plant is designed to accept a 50% loss of electrical load while operating at full power or a complete loss of load while operating below 50% power without actuating a reactor trip. The automatic steam bypass system 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. Should the reactor suffer a complete loss of load from full power, the reactor protection system would automatically actuate a reactor trip.

The most likely source of a complete loss of load on the nuclear steam supply system is a trip of the turbine-generator. In this case, there is a direct reactor trip signal derived from either the turbine autostop oil pressure or a closure of the turbine stop valves, provided the reactor is operating above

50% power. Reactor temperature and pressure do not increase significantly if the steam bypass system and pressurizer pressure control system are functioning properly. However, the plant behavior is evaluated for a complete loss of load from full power without a direct reactor trip, primarily to show the adequacy of the pressure relieving devices and also to show that no core damage occurs. The reactor coolant system and steam system pressure relieving capacities are designed to ensure the safety of the plant without requiring the automatic rod control, pressurizer pressure control, and/or steam bypass control systems.

Method of Analysis

The total loss of load transients are analyzed by employing the detailed digital computer program LOFTRAN. 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.

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 100% of full power without direct reactor trip,primarily to show the adequacy of the pressure-relieving devices and also to demonstrate core protection margins.

This accident is analyzed with the Improved Thermal Design Procedure in WCAP-8567, Reference 6. A uniform SGTP level of 15% is assumed.

Initial Operating Conditions - The initial reactor power and reactor coolant system temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-8567.

Moderator and Doppler Coefficients of Reactivity - The loss-of-load accident is analyzed with both maximum and minimum reactivity feedback. The maximum feedback cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback cases

assume positive moderator temperature coefficient (+5 pcm/°F) and the least negative Doppler coefficient.

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

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 to the setpoint value.

Pressurizer Spray and Power-Operated Relief Valves - Two cases, for both maximum and minimum feedback, 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. Safety valves are operable.

Feedwater Flow - Main feedwater flow to the steam generators is assumed to be lost at the time of loss of external electrical load.

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

Results

The transient responses for a total loss of load from full-power operation are shown for four cases: two cases for minimum reactivity feedback and two cases for maximum reactivity feedback illustrated in Figures 5.5-1 through 5.5-24.

Figures 5.5-1 through 5.5-6 show the transient responses for the total loss-of-steam load with minimum reactivity feedback, assuming full credit for

the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for steam dump. The reactor is tripped by the high pressurizer pressure signal. The minimum departure from nucleate boiling ratio is well above the limit value.

Figures 5.5-7 through 5.5-12 show the response for the total loss-of-steam load with a large negative moderator temperature coefficient. As temperature increases nuclear power decreases due to negative reactivity feedback. Power then stabilizes at a lower power level until the low steam generator level trip setpoint is reached. The DNBR increases throughout the transient and never drops below its initial value. Pressurizer relief valves and steam generator safety valves prevent overpressurization in the primary and secondary systems, respectively. The pressurizer safety valves are not actuated for this case. Following the low steam generator water level reactor trip, auxiliary feedwater would be used to remove decay heat with the results less severe than those presented in Section 15.2.6 of the UFSAR, Loss of Normal Feedwater Flow.

The total loss of load accident was also studied assuming the plant to be initially operating at 100% 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. Figures 5.5-13 through 5.5-18 show the minimum feedback transients. The neutron flux increases slightly until the reactor is tripped. The departure from nucleate boiling ratio increases throughout the transient. In this case, the pressurizer safety valves are actuated.

Figures 5.5-19 through 5.5-24 show the transients with maximum feedback and all other assumptions being the same as those in Figures 5.5-13 through 5.5-18. Again, the departure from nucleate boiling ratio increases throughout the transient, and the pressurizer safety valves are actuated.

The calculated sequence of events for these four cases is shown in Table 5.5-1.

Conclusions

Results of the analyses show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the reactor coolant system or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures within the design limits.

The integrity of the core is maintained by operation of the reactor protection system; i.e., the departure from nucleate boiling ratio is maintained above the limit value. Therefore, the conclusions reached in the RTSR remain valid.

5.6 Excessive Load Increase Incident

An excessive load increase incident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step load increase and/or a 5% per minute ramp load increase (without a reactor trip) in the range of 15% to 100% full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. If the load increase exceeds the capability of the reactor control system, the transient is terminated in time to prevent a DNBR less than the limiting value by a combination of the nuclear overpower trip and the overpower-overtemperature delta-T trips. An excessive load increase incident could result from either an administrative violation, or an equipment malfunction in the steam bypass control or turbine speed control.

For excessive loading by the operator or by system demand, the turbine load limiter keeps maximum turbine load below 100% rated load.

During power operation, steam bypass to the condenser is controlled by reactor coolant condition signals, i.e., abnormally high reactor coolant temperature indicates a need for steam bypass. A single controller malfunction does not cause steam bypass; an interlock is provided which blocks the control signal to the valves unless a large turbine load decrease or a turbine trip has occurred.

Method of Analysis

This accident is analyzed using the LOFTRAN code. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables, including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate the plant behavior following a 10% step-load increase from rated load for a uniform SGTP level of 15%. These cases are as follows:

1. Reactor control in manual with minimum moderator reactivity feedback.

2. Reactor control in manual with maximum moderator reactivity feedback.

3. Reactor control in automatic with minimum moderator reactivity feedback.

4. Reactor control in automatic with maximum moderator reactivity feedback.

For the minimum moderator feedback cases, the core has a 0.0 pcm/°F moderator temperature coefficient (MTC) of reactivity and, therefore, the least inherent transient capability. For the maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its most negative value. This results in the largest amount of reactivity feedback due to charges in coolant temperature.

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters. This accident is analyzed with the Improved Thermal Design Procedure as described in Reference 6. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 6.

Results

Figures 5.6-1 through 5.6-12 illustrate the transient with the reactor in the manual control mode. For the minimum feedback case, nuclear power remains relatively constant through the transient, while RCS temperatures are reduced. This results in a departure from nucleate boiling ratio that increases above its initial value. For the maximum feedback, manually controlled case, there is an increase in reactor power due to moderator feedback. A reduction in DNBR is experienced, but the DNBR remains above the limit value.

Figures 5.6-13 through 5.6-24 illustrate the transient when the reactor is assumed to be in automatic control mode. Both the minimum and maximum feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both the minimum and maximum feedback cases, the minimum DNBR remains above the limit value. The calculated sequence of events is shown in Table 5.6-1.

The excessive load increase incident is an overpower transient for which the fuel temperatures rise. When a reactor trip does not occur, the plant reaches a new equilibrium condition at a higher power level corresponding to the new increase in steam flow.

Conclusion

It has been demonstrated that, for an excessive load increase, the minimum departure from nucleate boiling ratio during the transient will not be below the limit value. Therefore, the conclusions reached in the RTSR remain valid.

5.7 Rupture of a Control Rod Mechanism Housing-RCCA Ejection

In order for this accident to occur, a rupture of the control rod mechanism housing must be postulated creating a full system pressure differential acting on the drive shaft. The resultant core thermal power excursion is limited by Doppler reactivity effects of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

This event is classified as an ANS Condition IV incident. Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

a. Average fuel pellet enthalpy at the hot spot below 200 cal/gm.

b. Average clad temperature at the hot spot below the temperature at which clad embrittlement may be expected (2700°F).

c. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

d. Fuel melting will be limited to less than the innermost 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion (a) above.

Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculation without feedback is pessimistically assumed to persist throughout the transient.

The spatial kinetics code, TWINKLE (Reference 13), is used for the average core transient analysis. This code solves the two group neutron diffusion

theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one dimensional axial kinetics code, since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods of calculating the ejected rod worth and hot channel factor.

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 seconds, the time for full ejection of the rod. Therefore, the assumption is made that the hot spots before and after ejection are coincident. This is very conservative, since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN (Reference 11). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative pellet radial power distribution is used within the fuel rod.

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. Table 5.7-1 presents the parameters used in this analysis. These parameters are unchanged from those used in the RTSR.

Results

The most limiting full power and zero power cases presented in UFSAR Section 15.4.5 were reanalyzed assuming 15% steam generator tube plugging. The results are presented below.

1. Beginning of Cycle, Full Power

   Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 400 pcm and 5.61 respectively. The peak hot spot clad average temperature was 2511°F. The peak hot spot fuel center temperature reached melting, which was conservatively assumed at 4900°F. However, melting was restricted to less than 10% of the pellet.

2. Beginning of Cycle, Zero Power

   For this condition, control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 780 pcm and a hot channel factor of 7.80. The peak hot spot clad average temperature reached 2631°F, the fuel center temperature was 3862°F.

A summary of the cases presented above is given in Table 5.7-1. The nuclear power and hot spot fuel and clad temperature transients for the worst cases are presented in Figures 5.7-1 through 5.7-4 (beginning-of-life full power and beginning-of-life zero power). The sequence of events for these two cases are presented in Table 5.7-2.

For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously, the reactor will remain subcritical following reactor trip.

The pressure surge calculation, described in RTSR Section 14.2.6, is applicable to the case with 15% steam generator tube plugging. Therefore, no system overpressure reanalysis is necessary. The rods in DNB calculation

presented in the RTSR shows that less than 10% of the rods enter DNB. An increase in SGTP to 15% will not alter these conclusions.

Conclusions

Conservative analyses indicate that the described fuel and cladding limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that the fission product release, as a result of a number of fuel rods entering DNB, is limited to less than 10% of the fuel rods in the core. Therefore, the conclusion in the RTSR remain valid.

TABLE 5.1-1

LARGE BREAK LOCA TIME SEQUENCE OF EVENTS FOR 15% SGTP

Event	DECLG C_D=0.4 Time (sec)
Start	0.0
Reactor Trip Signal	0.477
Safety Injection Signal	0.63
Accumulator Injection	8.62
End-of-Blowdown	24.8
Pump Injection	25.63
Bottom of Core Recovery	43.999
Accumulator Empty	57.681

TABLE 5.1-2

LARGE BREAK LOCA - ANALYSIS INPUTS AND RESULTS FOR 15% SGTP

Results	DECLG C_D=0.4
Peak Clad Temperature (°F)	1871[ab]
Peak Clad Temperature Location (ft)	7.5
Local Zr/H_2O Reaction (max) (%)	2.8
Local Zr/H_2O Location (ft)	7.5
Total Zr/H_2O Reaction (%)	<0.3
Hot Rod Burst Time (sec)	66
Hot Rod Burst Location (ft)	6.25
Calculation	
Core Power, (Mwt)	102% of 1520
Peak Linear Power, (kw/ft)	102% of 13.241
Peaking Factor	2.32 (design rating)
Accumulator water volume, (ft^3 per tank)	1100
Accumulator pressure, (psia)	715
Number of safty injection pumps operating	2
Steam-generator tubes plugged	15%

a. A 6°F peak clad temperature penalty must be added to the analysis value to account for upper plenum injection penalty.

b. A 10°F penalty must be added to the analysis value to account for core crossflow penalty.

TABLE 5.1-3

LARGE BREAK - CONTAINMENT DATA

Dry Containment

Net free volume	1.066×10^6 ft^3
Initial conditions	
Pressure	14.7 psia
Temperature	90°F
Refueling water storage tank temperature	60°F
Service water temperature	35°F
Outside temperature	-10°F
Spray system	
Number of pumps operating	2
Runout flow rate	1800 gpm each
Actuation time	10 sec
Engineered safety features fan coolers	
Number of fan coolers operating	4
Fastest postaccident initiation of fan coolers	30 sec

TABLE 5.1-4

REFLOOD MASS AND ENERGY RELEASES
FOR 15% SGTP

Time (Sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)
43.999	0.000	0.00
44.6	0.01014	13.013
45.8	0.01043	13.387
50.2	36.360	46888.15
59.5	83.09	104249.48
71.4	142.37	119460.99
86.0	184.93	127032.15
102.8	194.97	124993.27
121.5	200.47	121626.23
164.8	210.78	114120.79
220.0	223.73	106087.73
399.0	242.92	92399.68
439.2	244.76	88631.49

TABLE 5.1-5

BLOWDOWN MASS AND ENERGY RELEASE
FOR 15% SGTP

Time (sec)	Break Mass Flow (lbm/sec)	Break Energy Flow ($x10^6$ Btu/sec)
0.00	0.0	0.0000
0.10	45068.4	24.0754
0.20	47338.3	25.2929
0.40	48199.2	25.7627
0.60	46400.8	24.8148
0.80	44638.9	23.9038
1.00	43472.5	23.3289
1.50	39804.5	21.5629
2.00	34675.9	19.0161
3.00	24774.8	13.9184
4.00	21661.7	12.2645
5.00	19522.8	11.3341
6.00	16010.7	9.9706
7.00	13386.7	8.7903
8.00	11002.1	7.5669
9.00	8671.1	6.3087
10.00	6410.2	5.0583
12.00	5692.1	4.0552
14.00	5048.8	3.3252
16.00	3747.4	2.3252
18.00	2425.9	1.2101
20.00	2152.5	0.7890
22.00	2250.4	0.5835
23.50	125.3	0.0447

TABLE 5.1-6

BROKEN LOOP ACCUMULATOR MASS AND
ENERGY RELEASE FOR 15% SGTP

Time (Sec)	Mass Flow (lb/sec)	Energy (Btu/sec)
0.000	2683.030	159962.227
1.010	2548.759	151957.005
2.010	2434.334	145135.006
3.010	2334.996	139212.473
4.010	2247.855	134017.115
5.010	2169.425	129341.147
6.010	2098.190	125094.083
7.010	2033.203	121219.534
8.010	1973.721	117673.233
9.010	1919.196	114422.446
10.010	1869.145	111438.454
11.010	1823.187	108698.386
12.010	1780.530	106155.212
13.010	1740.753	103783.712
14.010	1703.595	101568.346
15.010	1668.809	99494.374
16.010	1636.298	97556.106
17.010	1606.111	95756.326
18.010	1577.934	94076.438
19.010	1551.511	92501.101
20.010	1526.540	91012.290
21.010	1502.979	89607.624
22.010	1480.492	88266.940
23.010	1459.245	87000.199
24.010	1438.906	85787.595

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TABLE 5.2-1

TIME SEQUENCE OF EVENTS FOR
UNCONTROLLED RCCA WITHDRAWAL AT POWER
FOR A SGTP LEVEL OF 15%

Event	Time of Each Event (Seconds)
Case A:	
Initiation of uncontrolled rod cluster control assembly withdrawal at full power and maximum reactivity insertion rate (90 pcm/sec)	0.0
Power range high neutron flux high trip point reached	3.4
Rods begin to fall into core	3.9
Minimum departure from nucleate boiling ratio occurs	4.3
Case B:	
Initiation of uncontrolled rod cluster control assembly withdrawal at full power and at a small reactivity insertion rate (7 pcm/sec)	0.0
Power range high neutron flux high trip point reached	15.8
Rods begin to fall into core	16.3
Minimum departure from nucleate boiling ratio occurs	17.2

TABLE 5.4-1

SUMMARY OF LIMITING RESULTS FOR
LOCKED ROTOR ACCIDENT
15% SGTP LEVEL

Maximum Reactor Coolant System Pressure (psia)	2838
Maximum Cladding Temperature Core Hot Spot (°F)	2204
Zirconium-Water Reaction at Core Hot Spot (% by weight)	.9885

TABLE 5.4-2

TIME SEQUENCE OF EVENTS FOR
LOCKED ROTOR INCIDENT
15% SGTP LEVEL

Event	Time of Each Event (Seconds)
Rotor on one pump locks	0
Low flow trip setpoint reached	0.09
Rods begin to drop	1.09
Maximum RCS pressure occurs	3.20
Maximum clad temperature occurs	3.31

TABLE 5.5-1

TIME SEQUENCE OF EVENTS FOR
LOSS OF EXTERNAL ELECTRICAL LOAD
FOR A SGTP LEVEL OF 15%

Case	Event	Time of Each Event (Seconds)
a. With pressurizer control (minimum feedback)	Loss of electrical load	0
	High pressurizer pressure reactor trip setpoint reached	10.4
	Rod begins to drop	12.4
	Minimum departure from nucleate boiling ratio occurs	13.0
	Peak pressurizer pressure occurs	14.0
b. With pressurizer control (maximum feedback)	Loss of electrical load	0
	Peak pressurizer pressure occurs	15.0
	Low steam generator level reactor trip setpoint	84.2
	Rods begin to drop	86.2
	Minimum departure from nucleate boiling ratio occurs	*

* DNBR does not decrease below its initial value.

TABLE 5.5-1 (continued)

TIME SEQUENCE OF EVENTS FOR
LOSS OF EXTERNAL ELECTRICAL LOAD
FOR A SGTP LEVEL OF 15%

Case	Event	Time of Each Event (Seconds)
c. Without pressurizer control (minimum feedback)	Loss of electrical load	0
	High pressurizer pressure reactor trip setpoint reached	5.6
	Rods begin to drop	7.6
	Peak pressure occurs	9.0
	Initiation of release from S/G safety valves	13.0
	Minimum departure from nucleate boiling ratio occurs	*
d. Without pressurizer control (maximum feedback)	Loss of electrical load	0
	High pressurizer pressure reactor trip setpoint reached	5.4
	Rods begin to drop	7.4
	Peak pressure occurs	8.0
	Initiation of release from S/G safety valves	13.0
	Minimum departure from nucleate boiling ratio occurs	*

* DNBR does not decrease below its initial value.

TABLE 5.6-1

TIME SEQUENCE OF EVENTS FOR
EXCESSIVE LOAD INCREASE INCIDENT
FOR A SGTP LEVEL OF 15%

Case	Event	(Seconds)
a. Manual reactor control (minimum feedback)	10% step load increase	0
	Equilibrium conditions reached (approx. time only)	150.0
b. Manual reactor control (maximum feedback)	10% step load increase	0
	Equilibrium conditions reached (approx. time only)	50.0
c. Automatic reactor control (minimum feedback)	10% step load increase	0
	Equilibrium conditions reached (approx. time only)	100.0
d. Automatic reactor control (maximum feedback)	10% step load increase	0
	Equilibrium conditions reached (approx. time only)	60.0

TABLE 5.7-1

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT FOR A SGTP LEVEL OF 15%

Time of Life	Beginning	Beginning
Power level, percent	102	0
Ejected rod worth, pcm	400	780
Delayed neutron frac., %	.49	.49
Feedback reactivity weighting	1.3	1.417
Trip reactivity, % delta-k	4.0	2.0
Fq before rod ejection	2.5	-
Fq after rod ejection	5.61	7.80
Number of operational pumps	2	1
Maximum fuel pellet average temperature, °F	4194	3427
Maximum fuel center temperature, °F	4970	3862
Maximum clad average temperature, °F	2511	2631
Maximum fuel stored energy, cal/gm	185	145
Maximum fuel melt, %	< 10	0.0

TABLE 5.7-2

TIME SEQUENCE OF EVENTS RCCA EJECTION
FOR A SGTP LEVEL OF 15%

Case	Time of Each Event (Seconds)
a. Beginning-of-Life, Full Power	
Initiation of rod ejection	0.0
Power range high neutron flux setpoint reached	0.03
Peak nuclear power occurs	0.14
Rods begin to fall into core	0.53
Peak fuel average temperature occurs	1.88
Peak heat flux occurs	2.00
Peak clad temperature occurs	2.02
b. Beginning-of-Life, Zero Power	
Initiation of rod ejection	0.0
Power range high neutron flux low setpoint reached	0.23
Peak nuclear power occurs	0.29
Rods begin to fall into core	0.73
Peak heat flux occurs	2.10
Peak clad temperature occurs	2.11
Peak fuel average temperature occurs	2.23

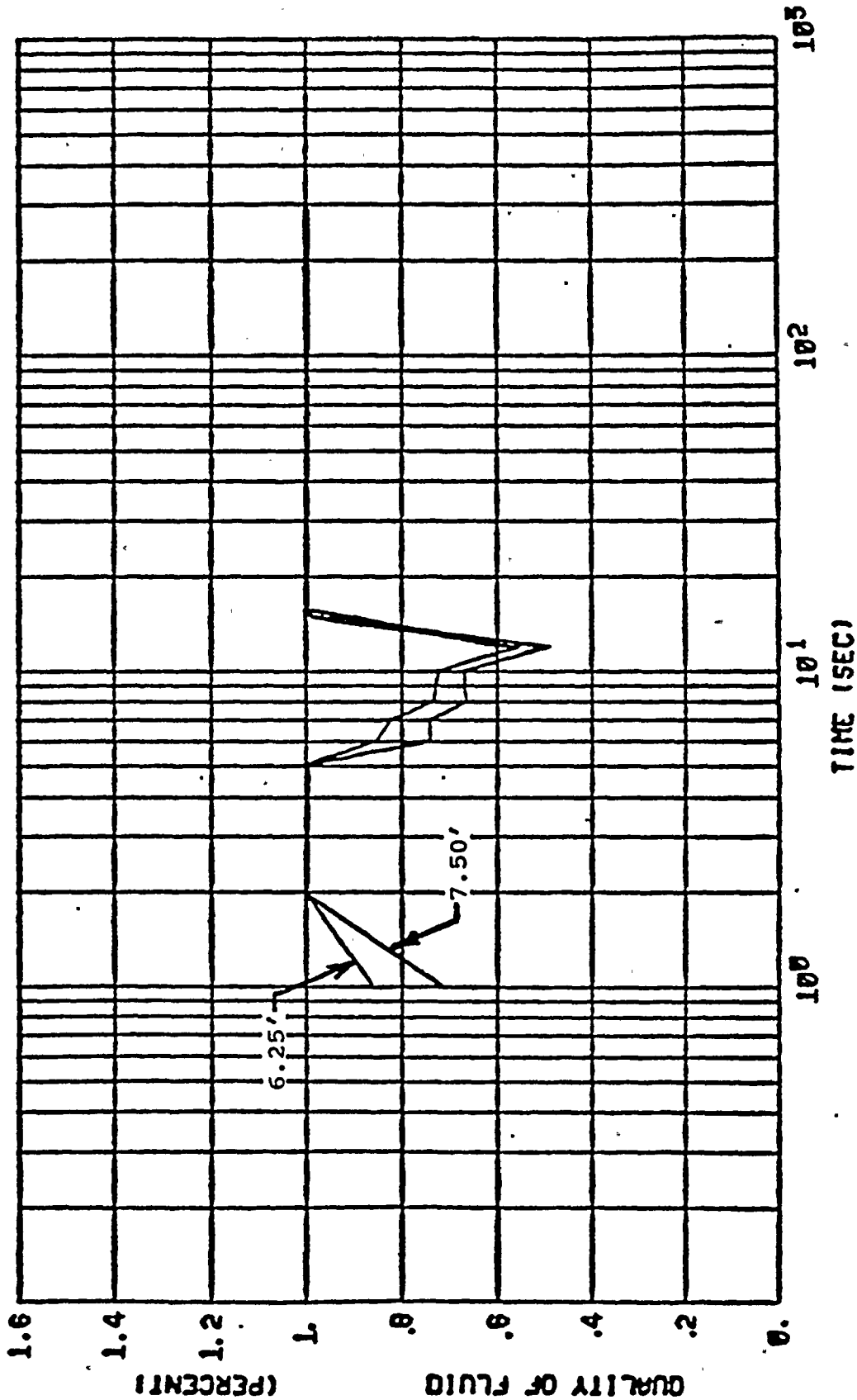


Figure 5.1-1

LOCA - DECLG, Fluid Quality Versus Time, CD = 0.4

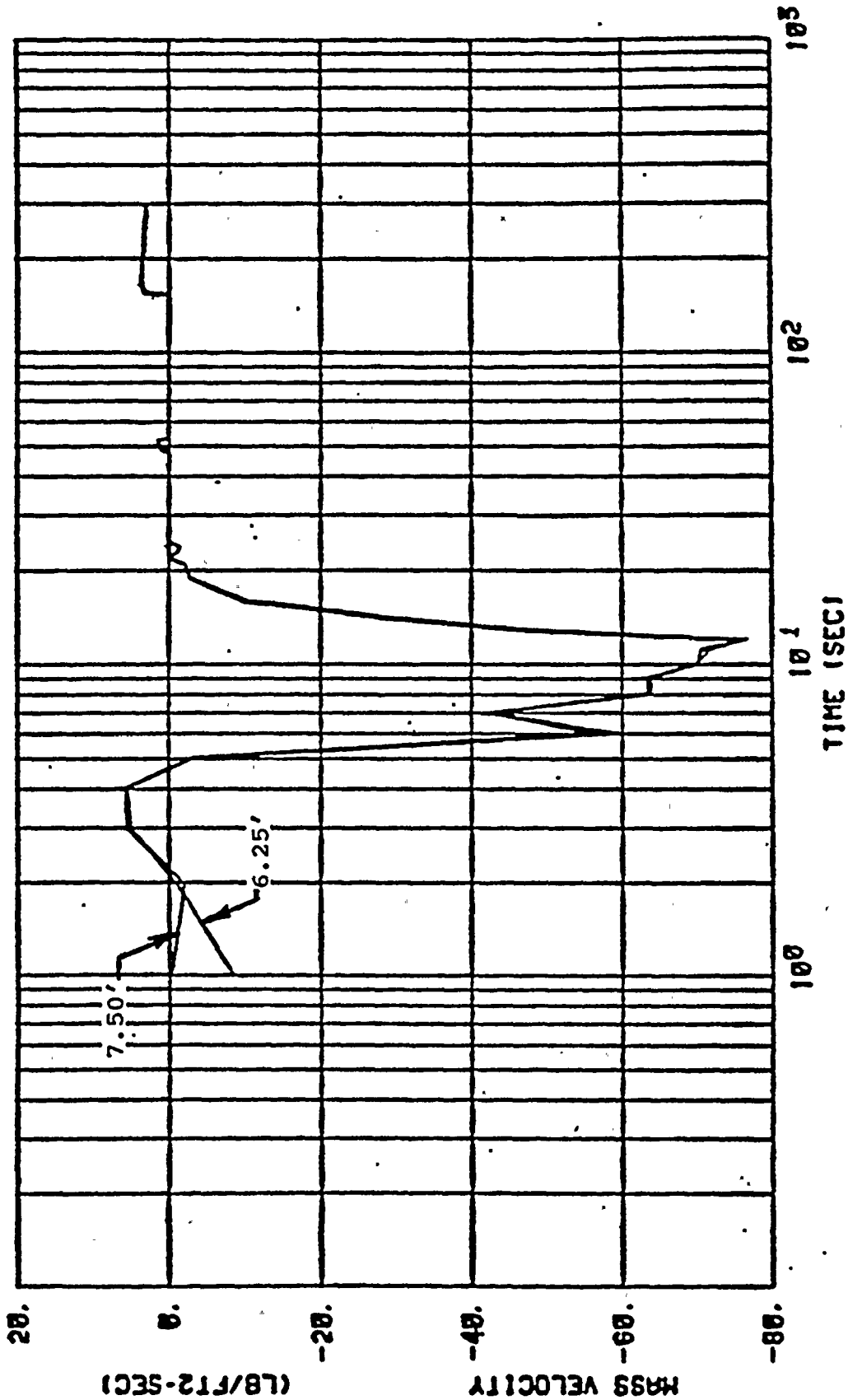


Figure 5.1-2

LOCA - DECLG, Mass Velocity Versus Time, CD = 0.4

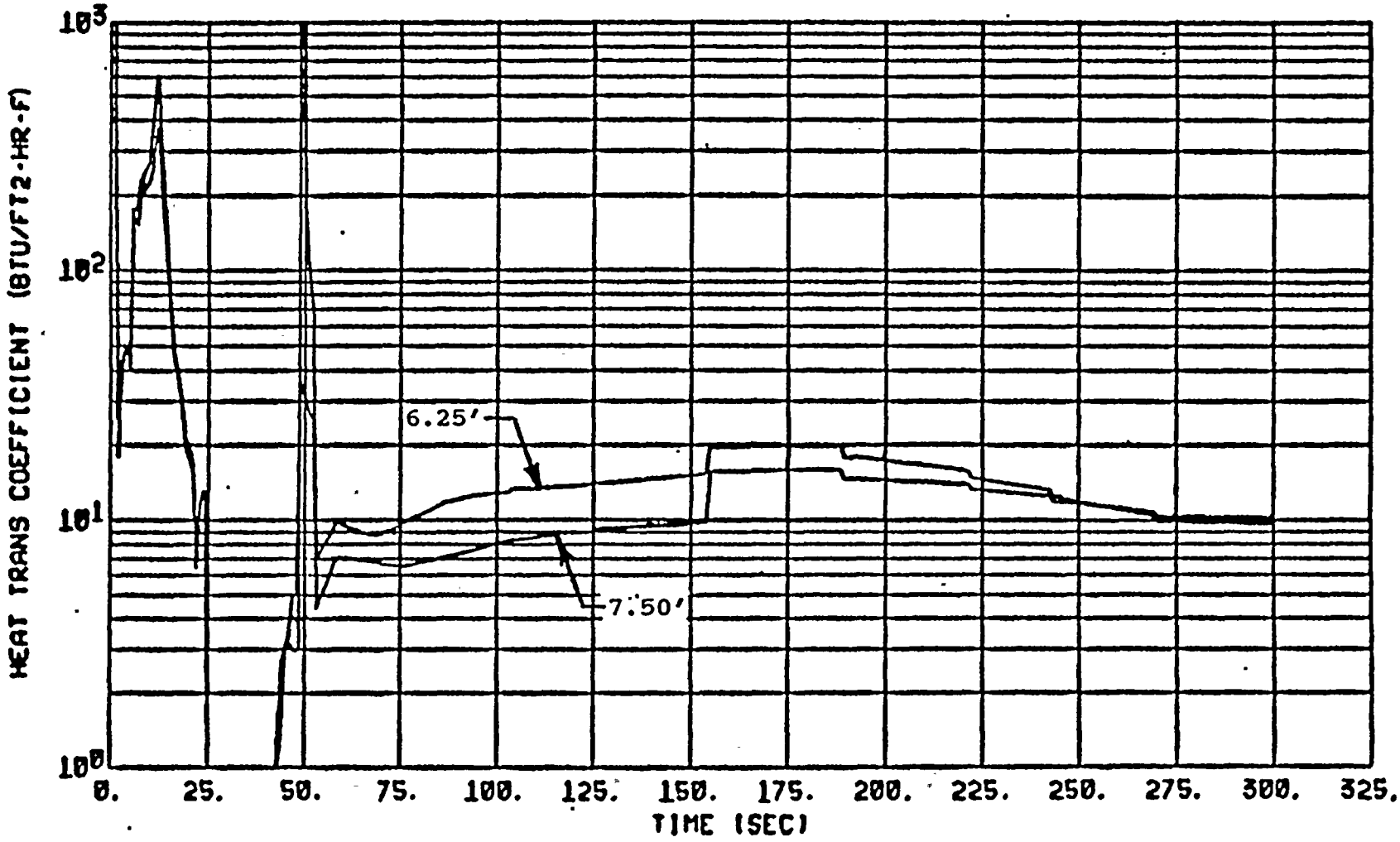


Figure 5.1-3
LOCA - DECLG, Heat Transfer
Coefficient Versus Time, CD = 0.4

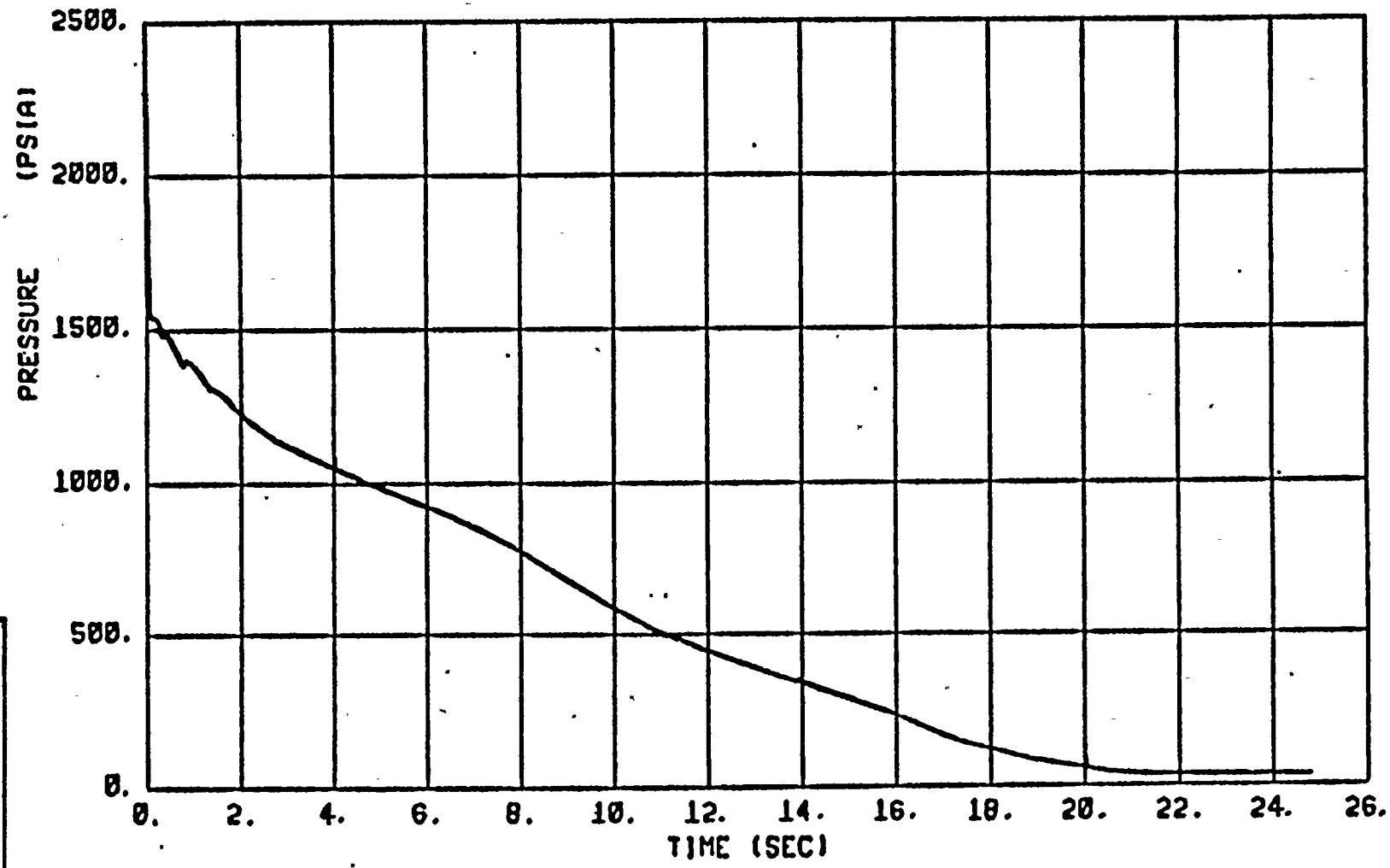


Figure 5.1-4
LOCA - DECLG, Core Pressure Versus Time, CD = 0.4

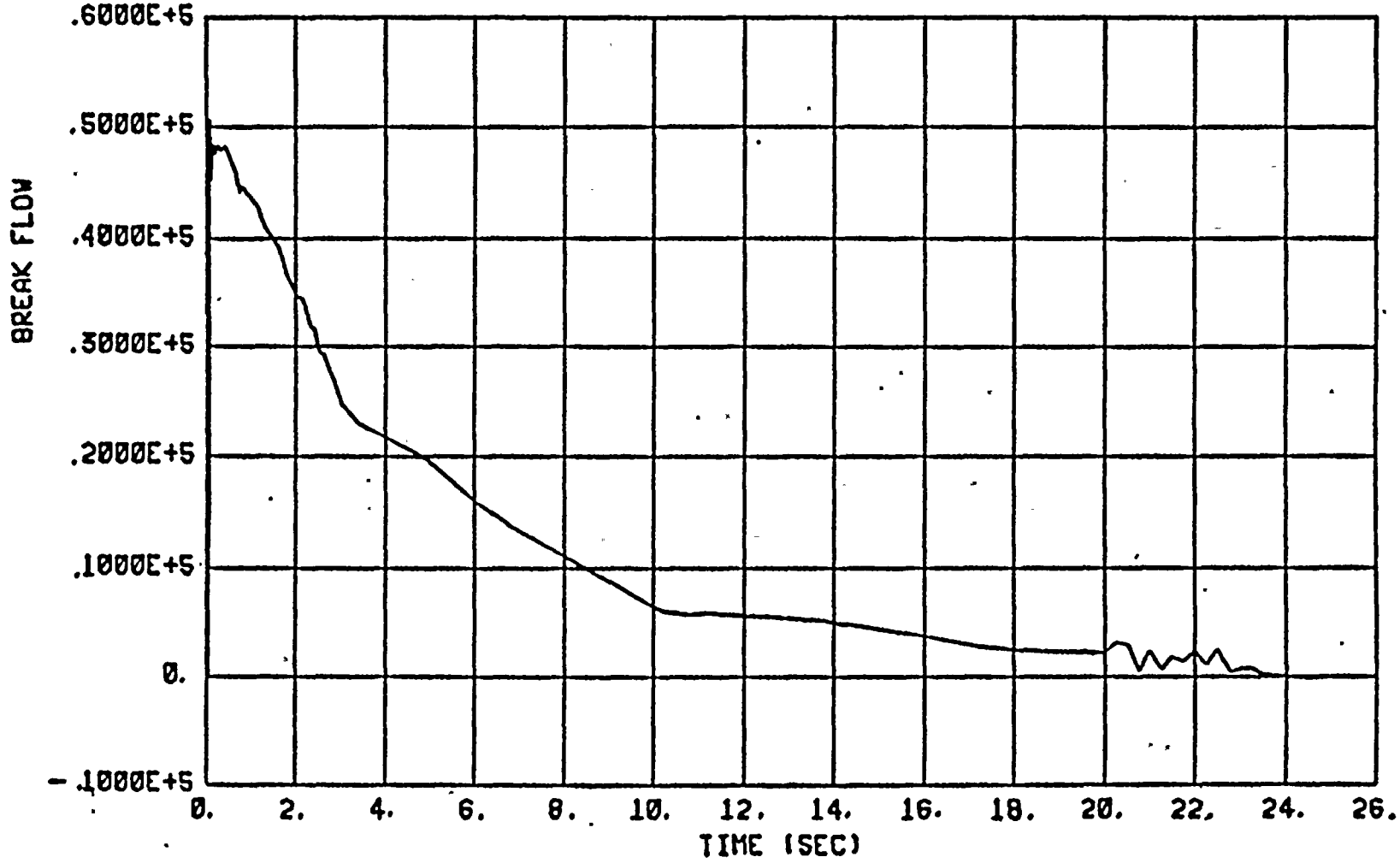


Figure 5.1-5
LOCA - DECLG, Break Flow Rate Versus Time, CD = 0.4

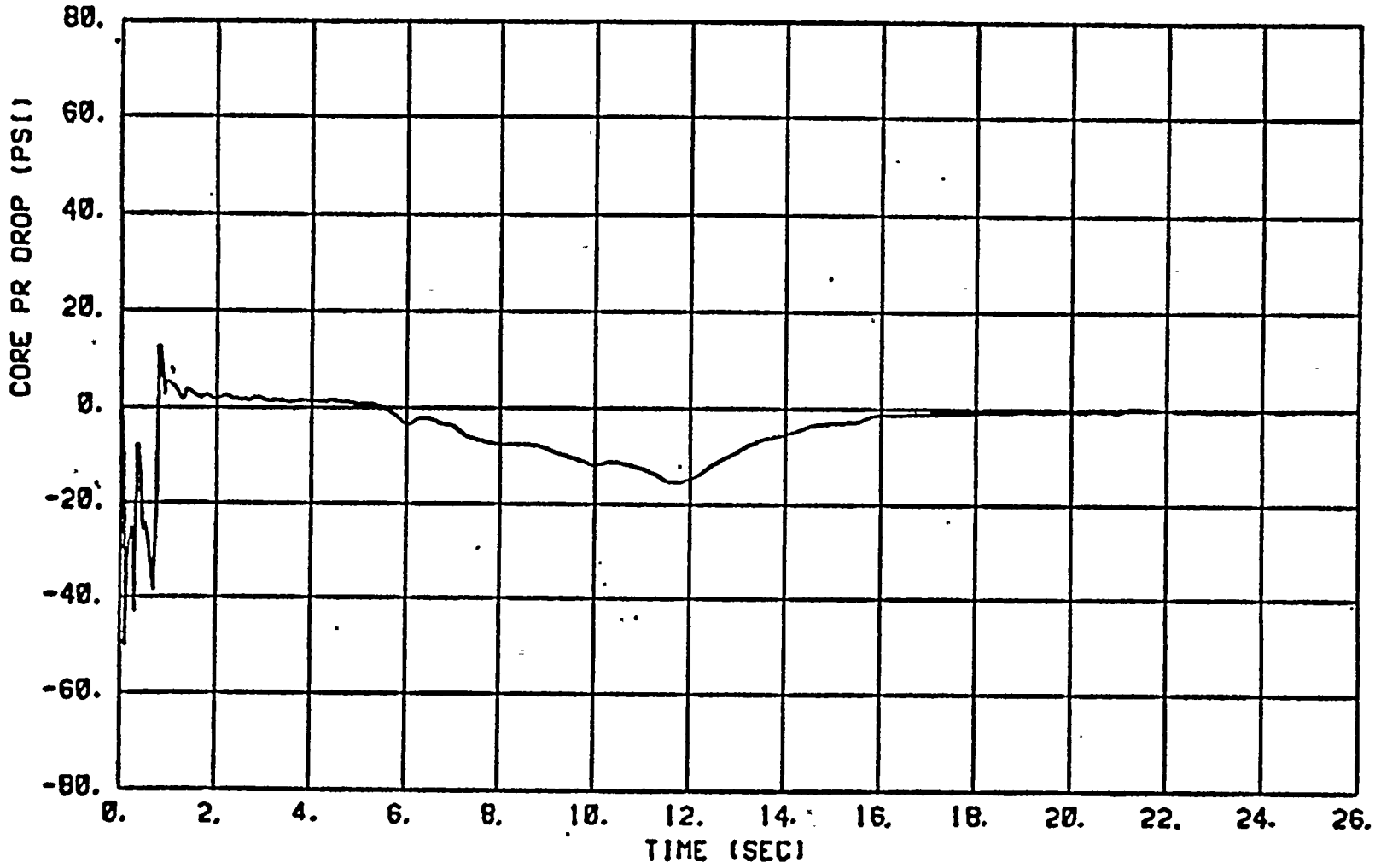


Figure 5.1-6
LOCA - DECLG, Core Pressure Drop
Versus Time, CD = 0.4

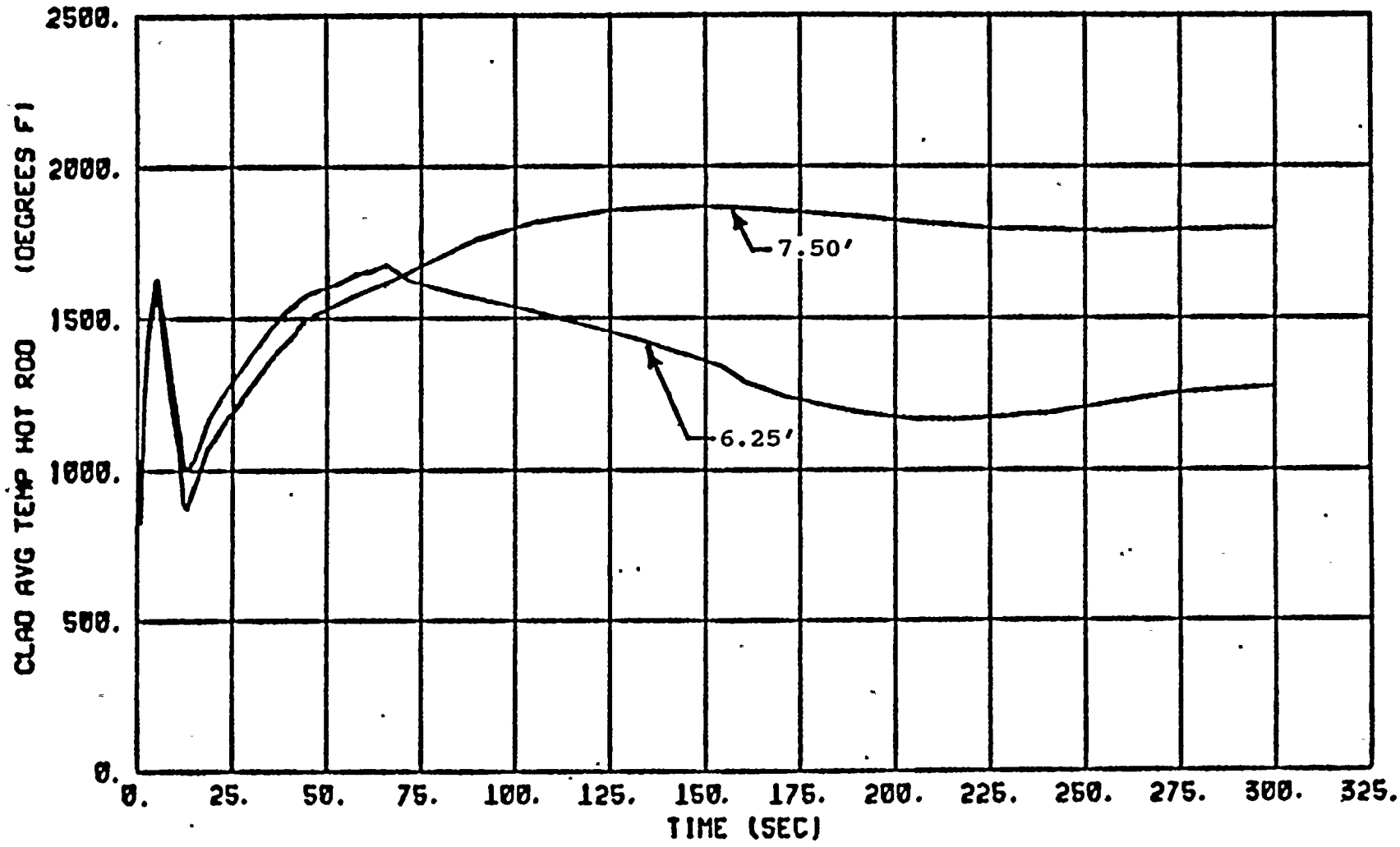


Figure 5.1-7
LOCA - DECLG, Peak Clad Temperature
Versus Time, CD = 0.4

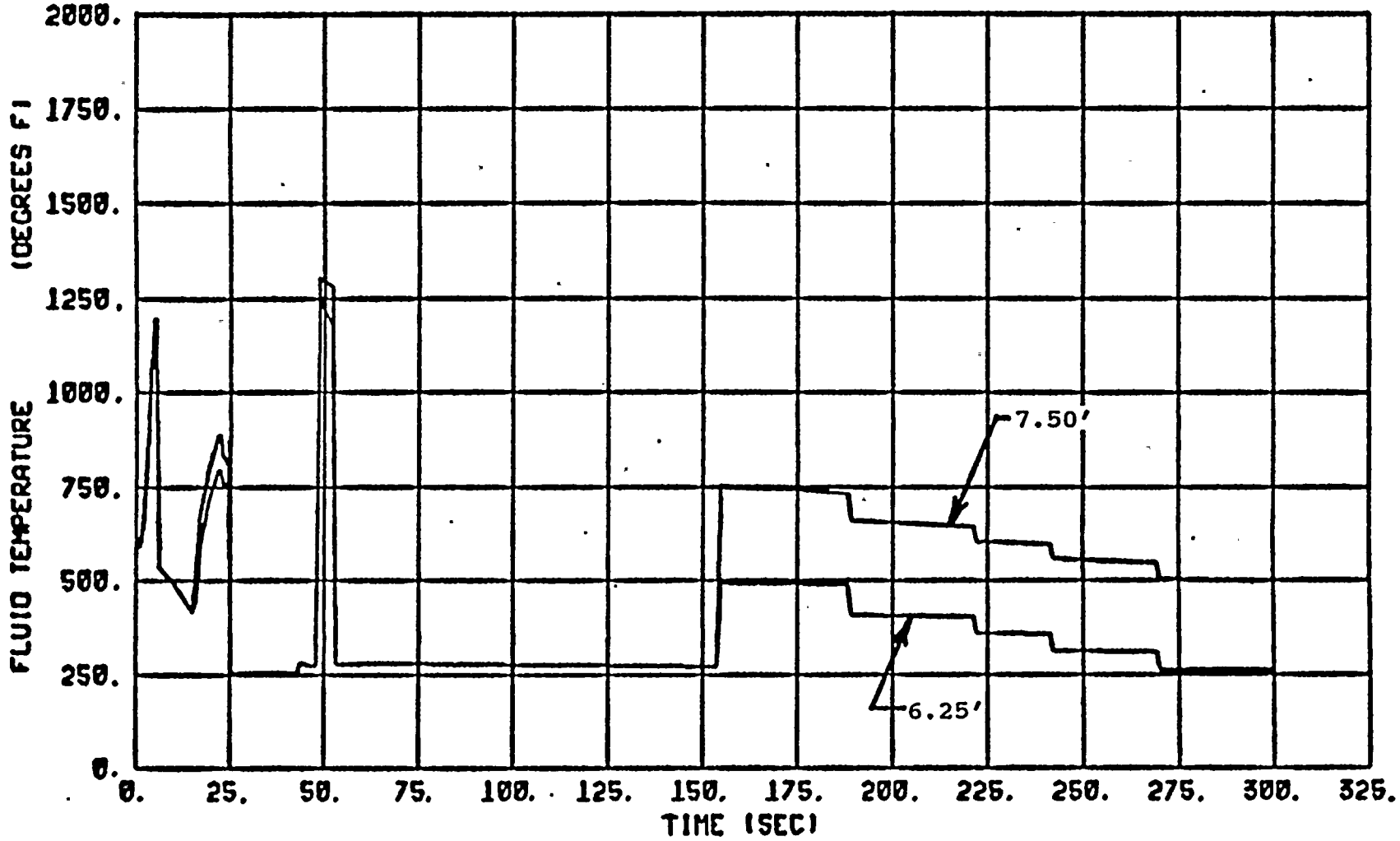


Figure 5.1-8
LOCA - DECLG, Fluid Temperature
Versus Time, CD = 0.4

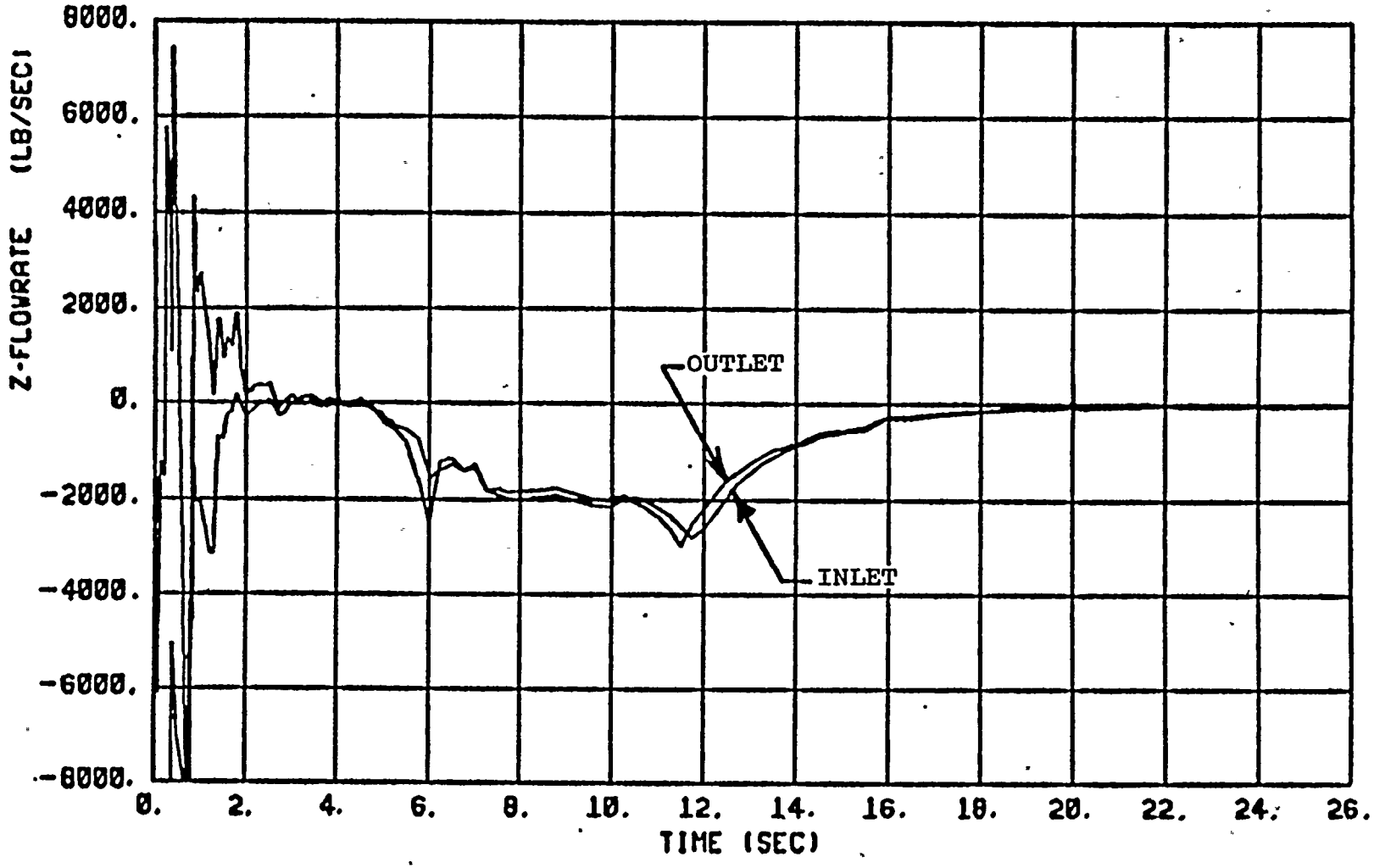


Figure 5.1-9
LOCA - DECLG, Core Flow Rate Versus Time, CD = 0.4

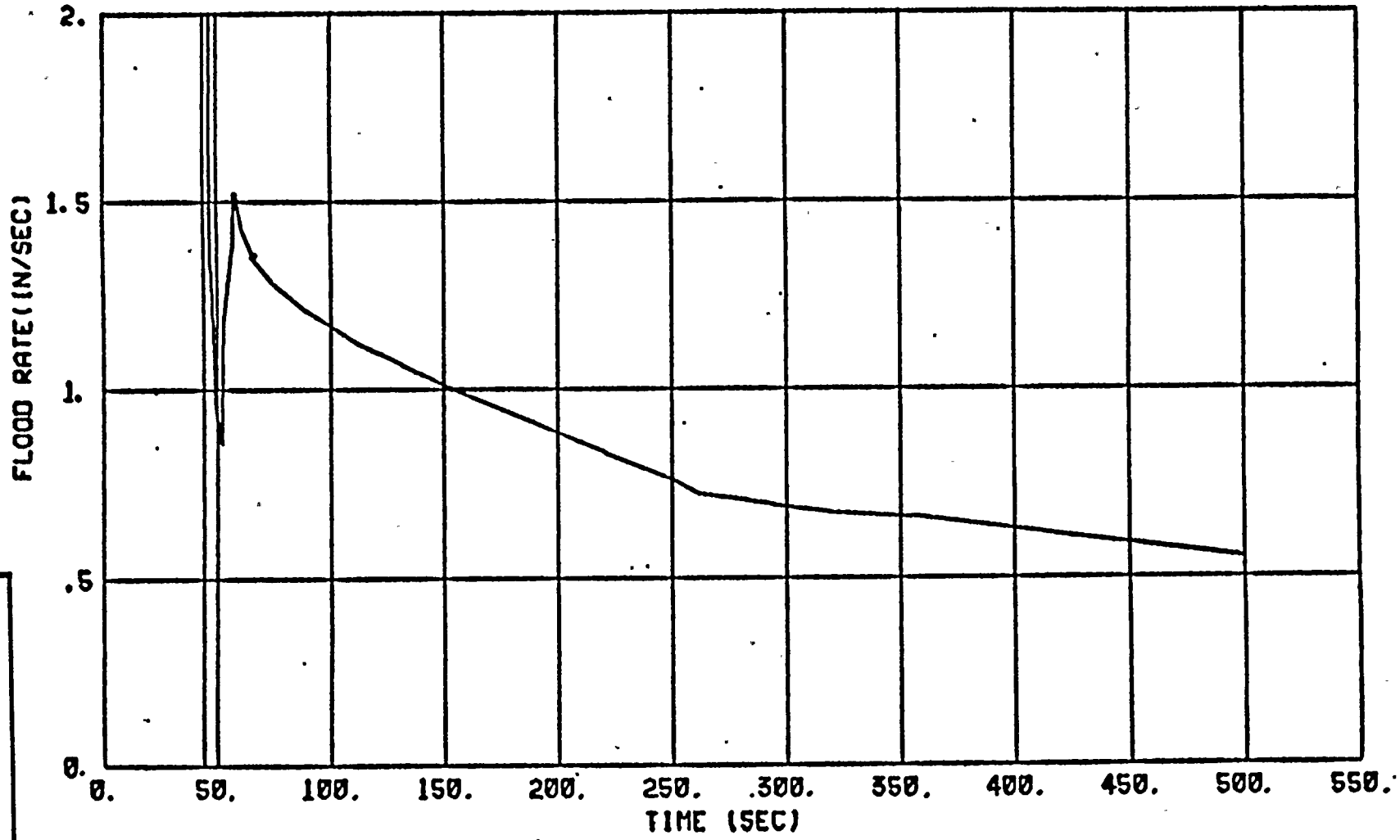


Figure 5.1-10
LOCA - DECLG Reflood Transient, Flood Rate Versus Time, CD = 0.4

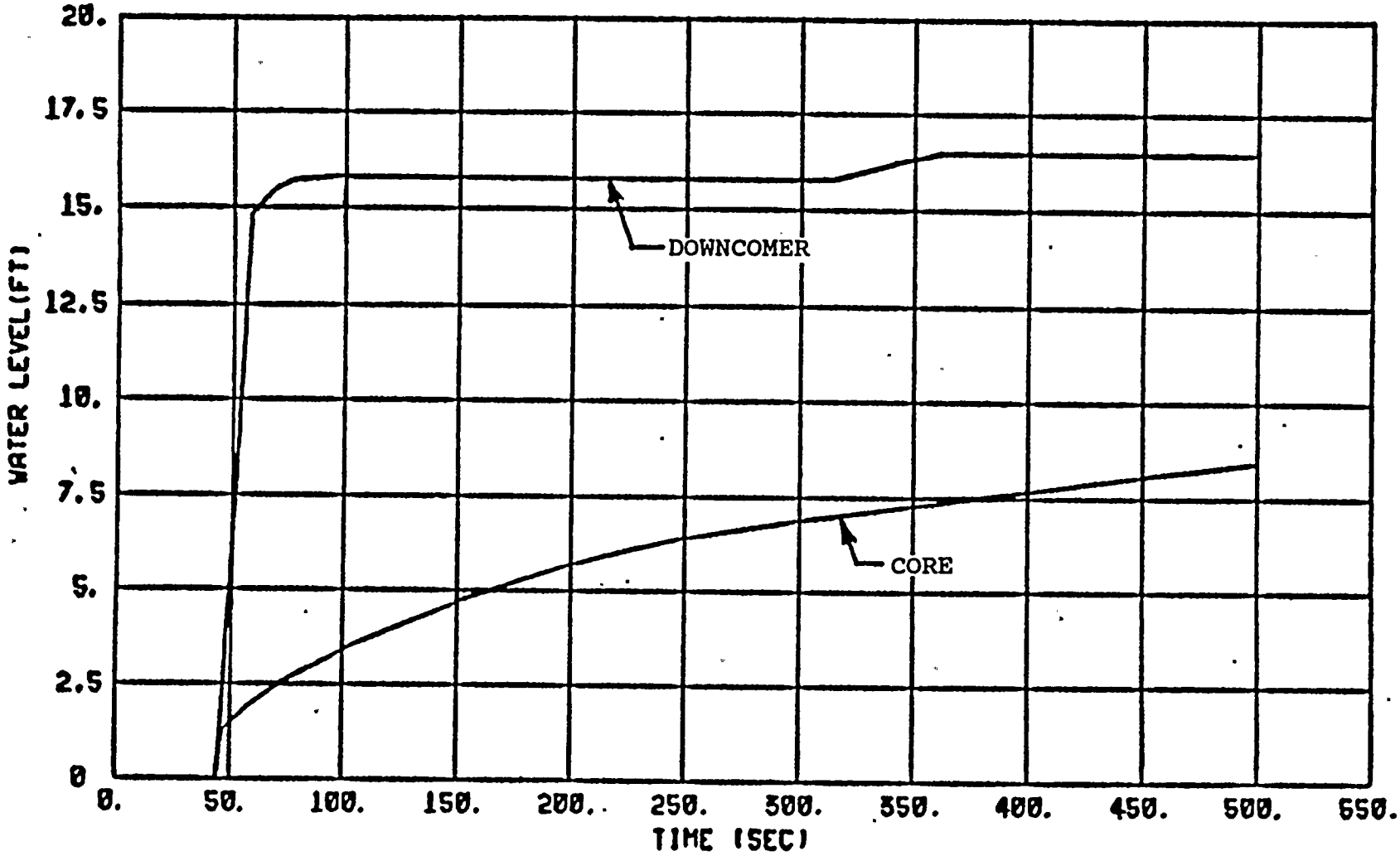


Figure 5.1-11
LOCA - DECLG Reflood Transient,
Core and Downcomer Water Levels
Versus Time, CD = 0.4

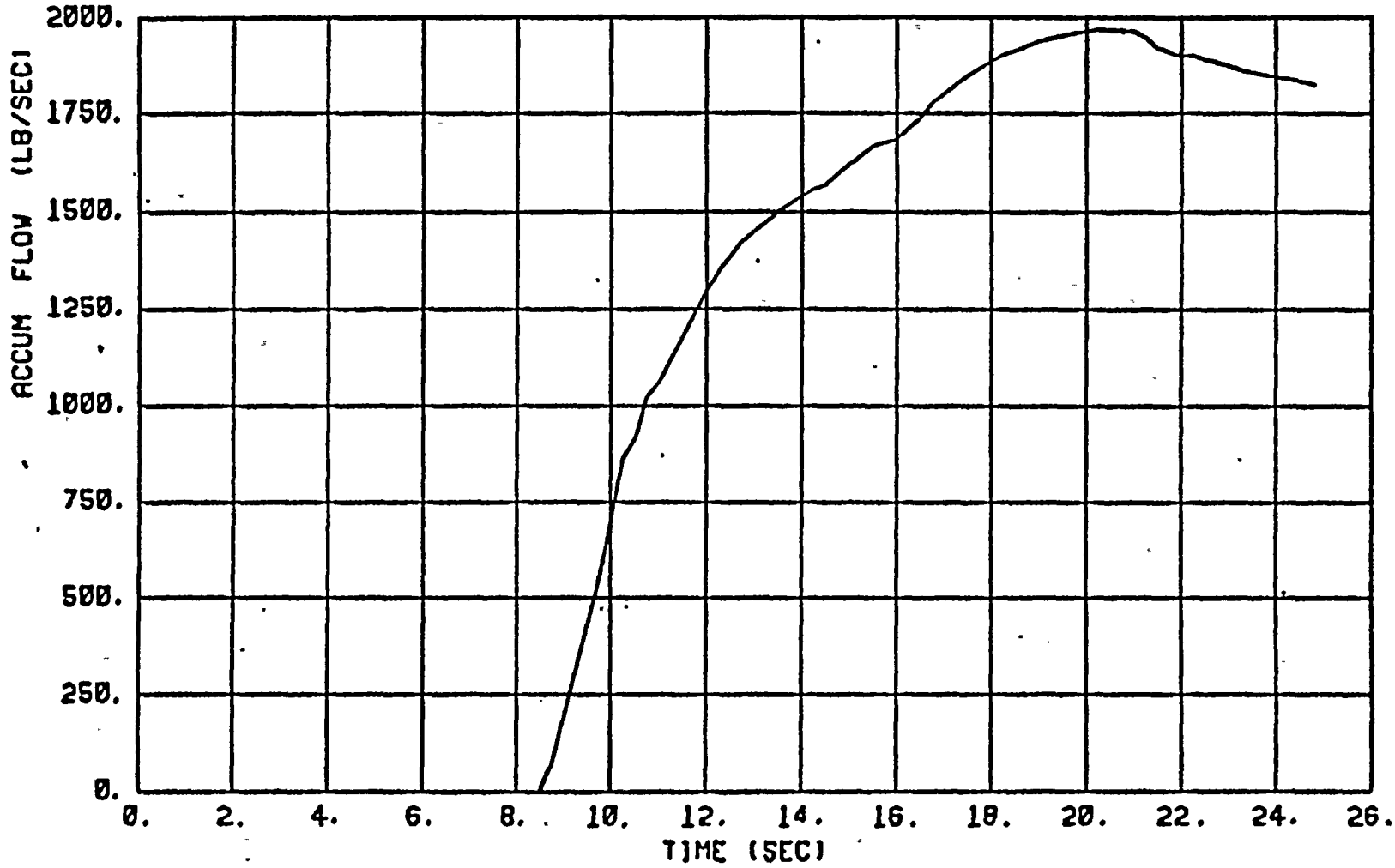


Figure 5.1-12
LOCA - DECLG, Accumulator Flow Versus Time, CD = 0.4

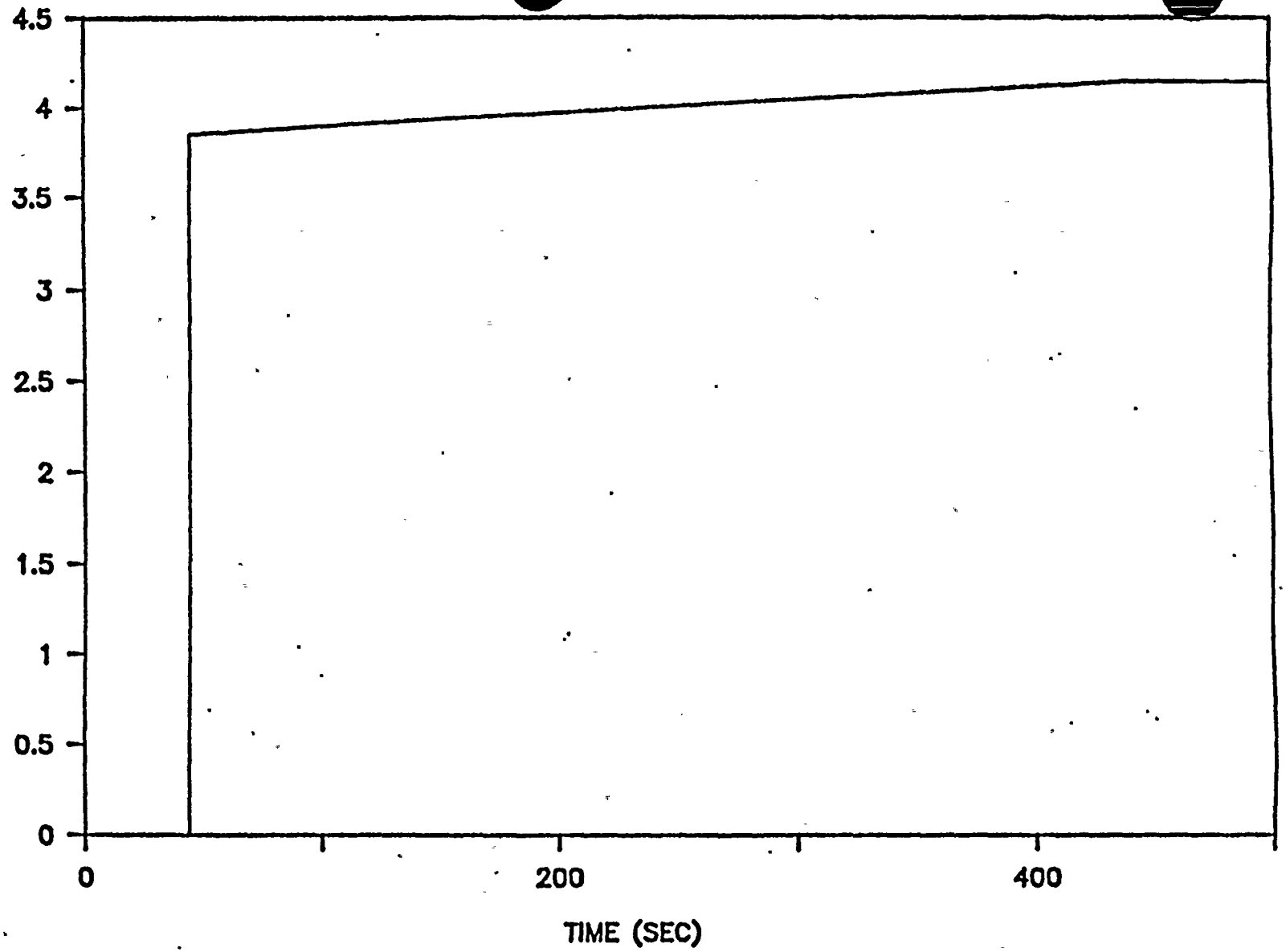


Figure 5.1-13
LOCA - DECLG, Safety Injection Flow
Versus Time, CD = 0.4

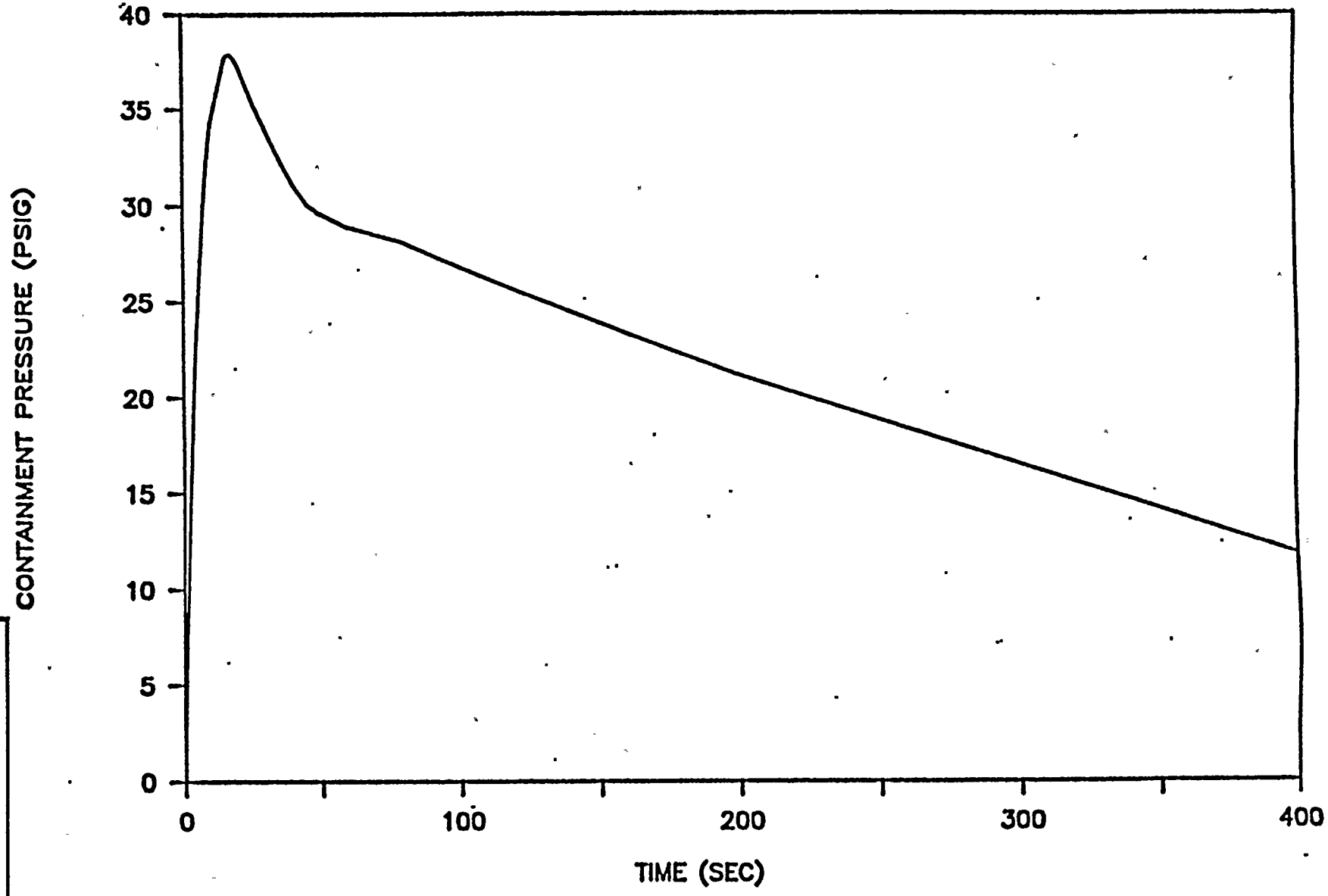


Figure 5.1-14
LOCA - DECLG, Containment Pressure
Versus Time, CD = 0.4

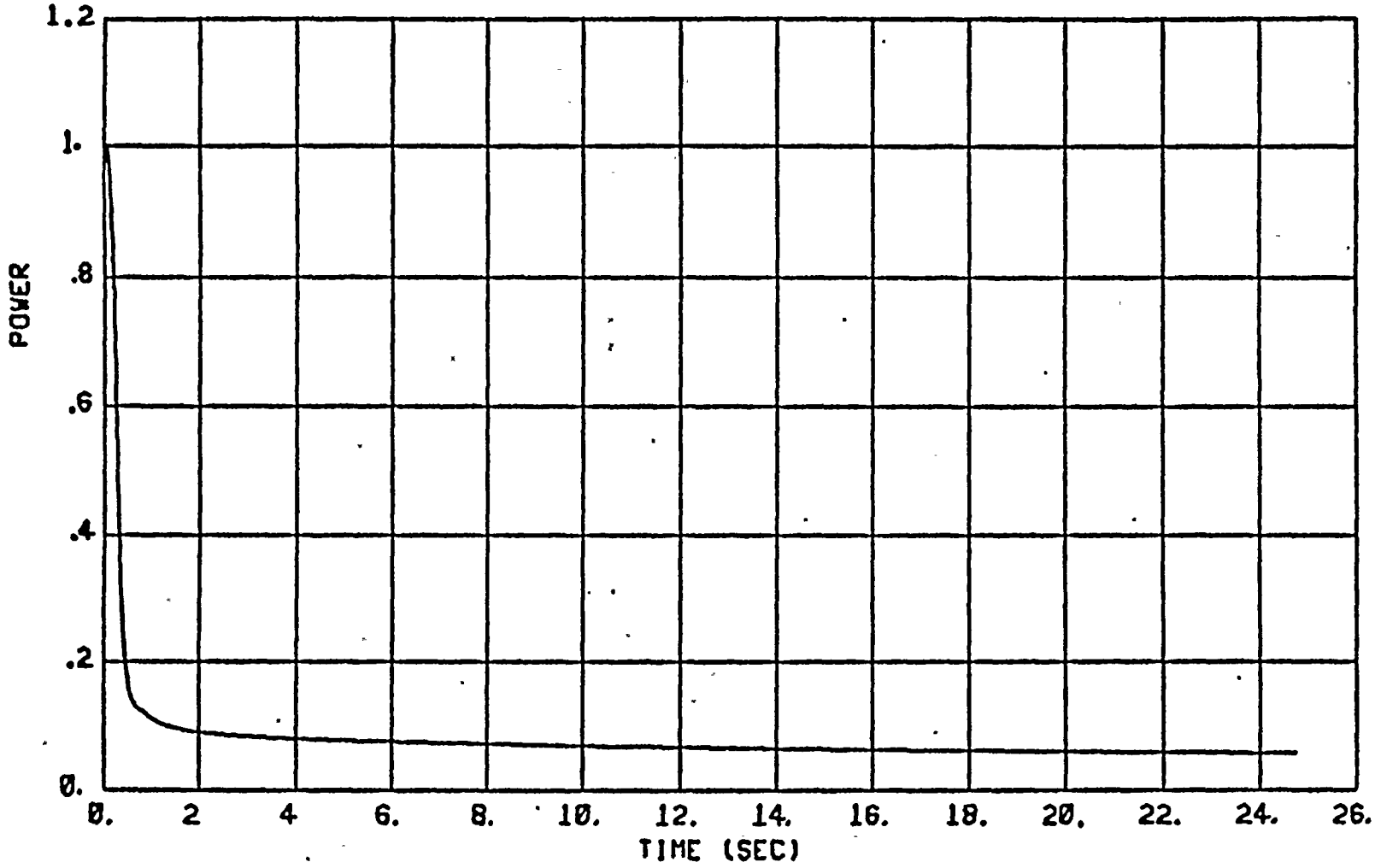


Figure 5.1-15
LOCA - DECLG, Core Power Versus Time,
CD = 0.4

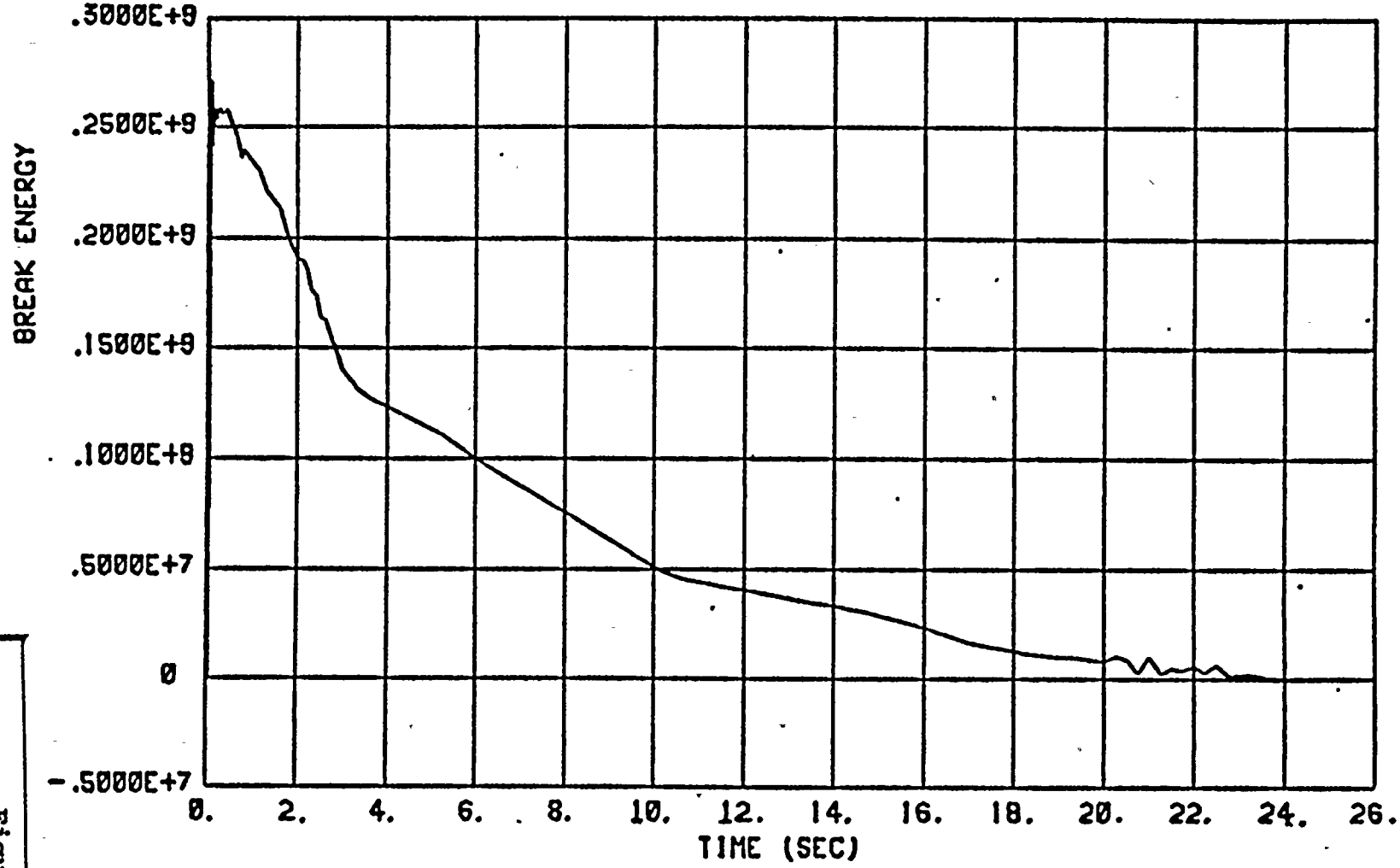


Figure 5.1-16
LOCA - DECLG, Break Energy Release to Containment Versus Time, CD = 0.4

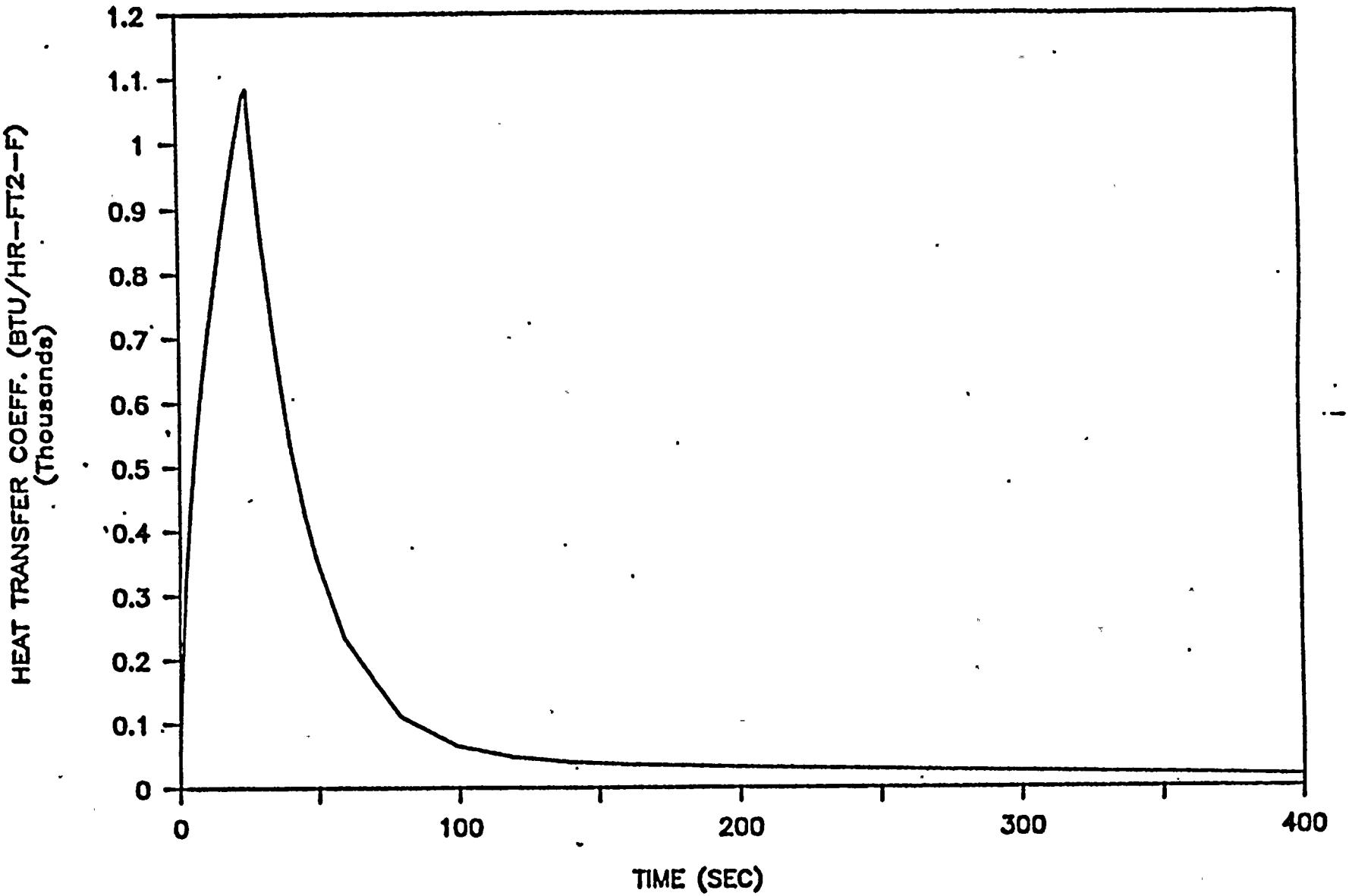
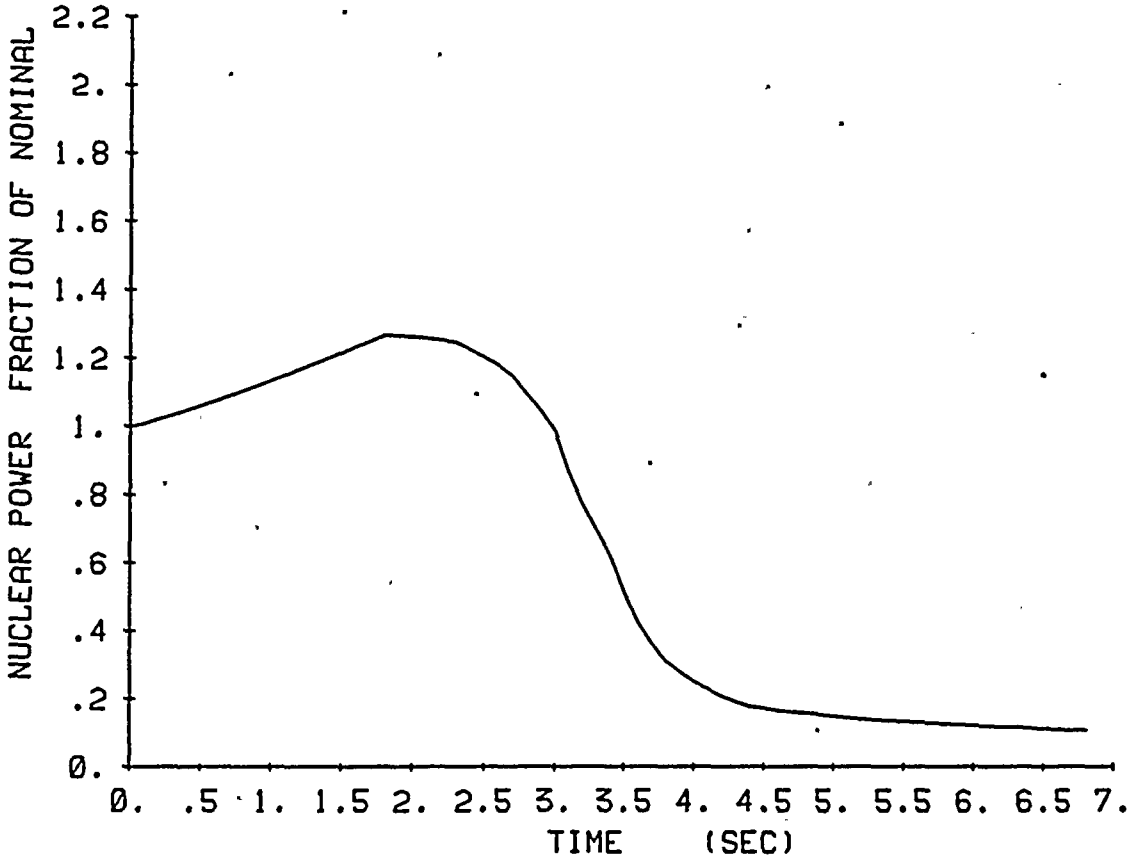
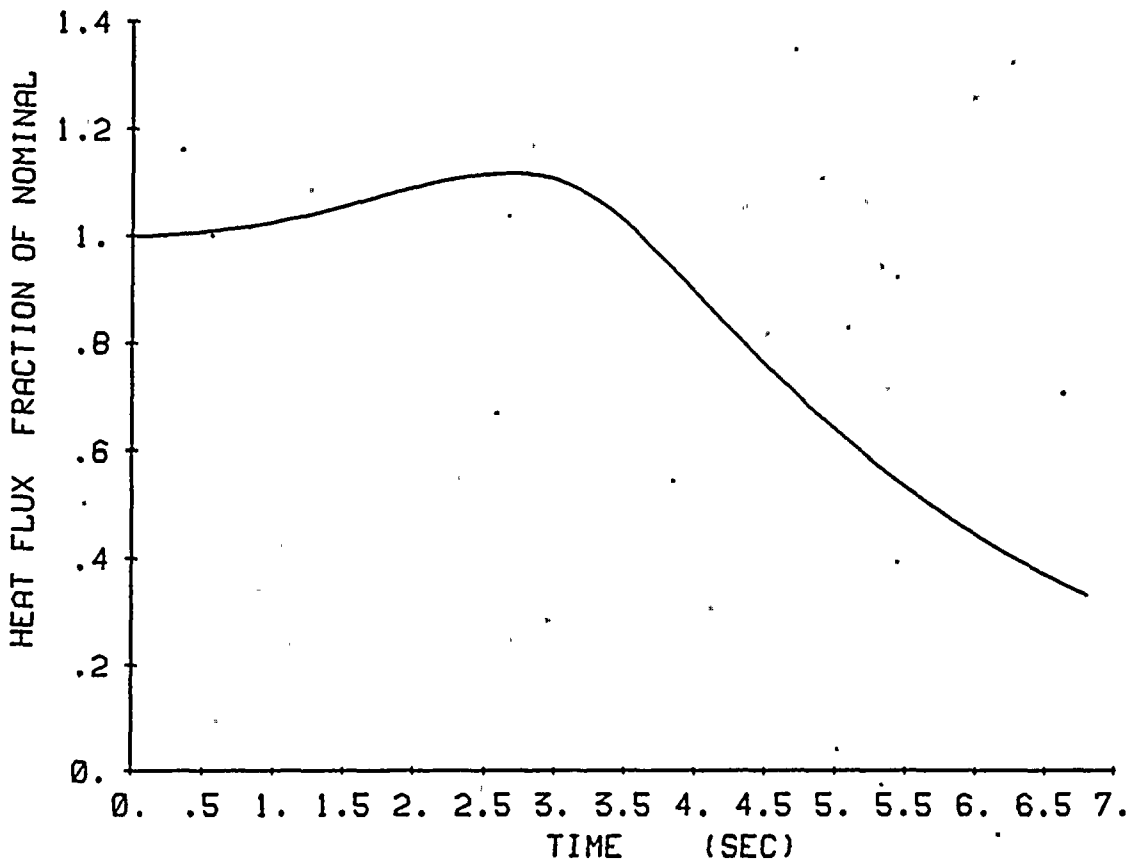


Figure 5.1-17
LOCA - DECLG, Containment Wall
Condensing Heat Transfer Coefficient
Versus Time, CD = 0.4



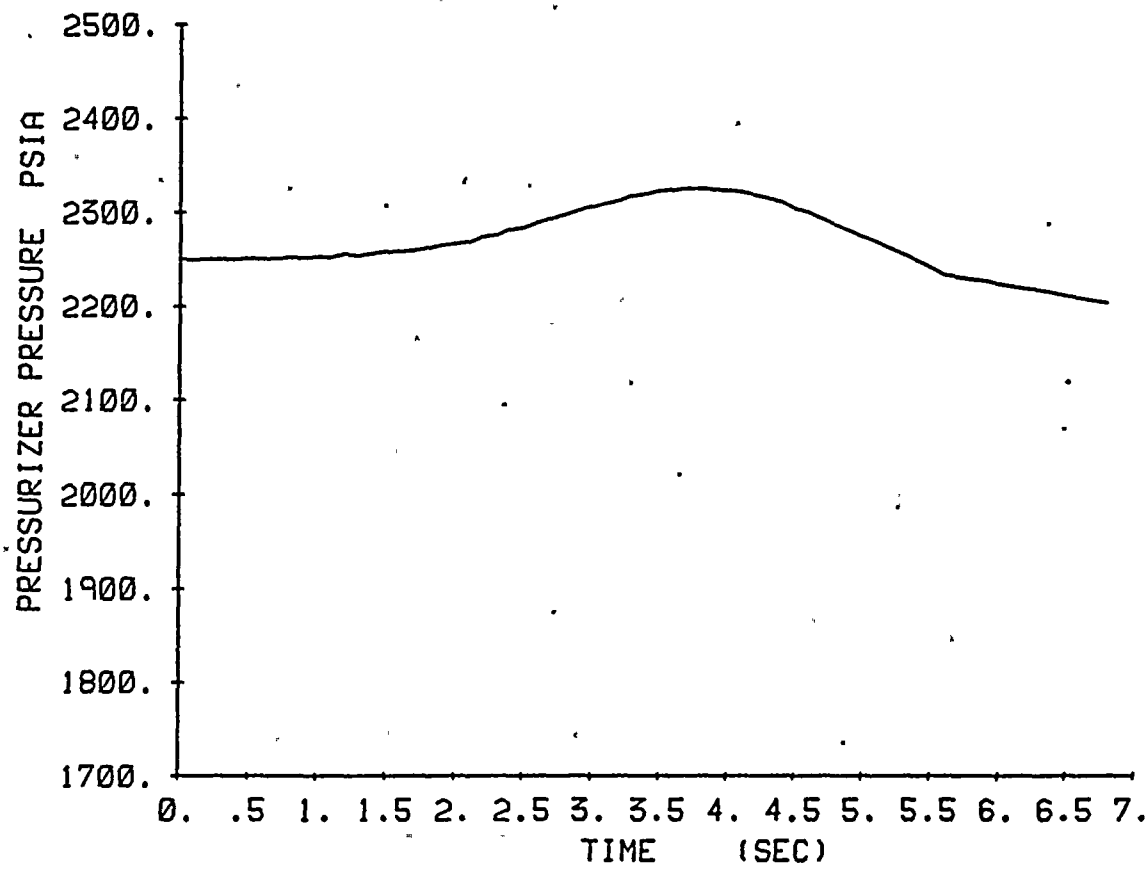
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Maximum Feedback
100% Power, 90 pcm/sec

Figure 5.2-1

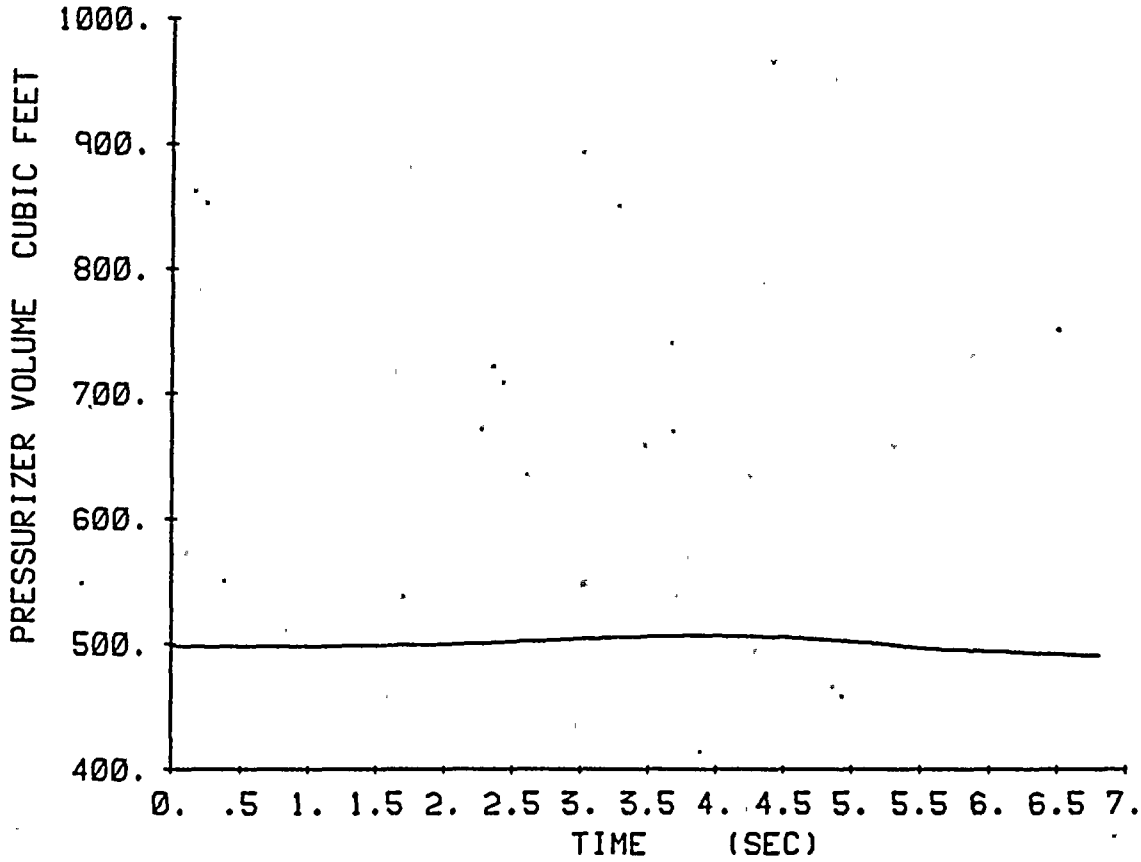


R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Maximum Feedback
100% Power, 90 pcm/sec

Figure 5.2-2

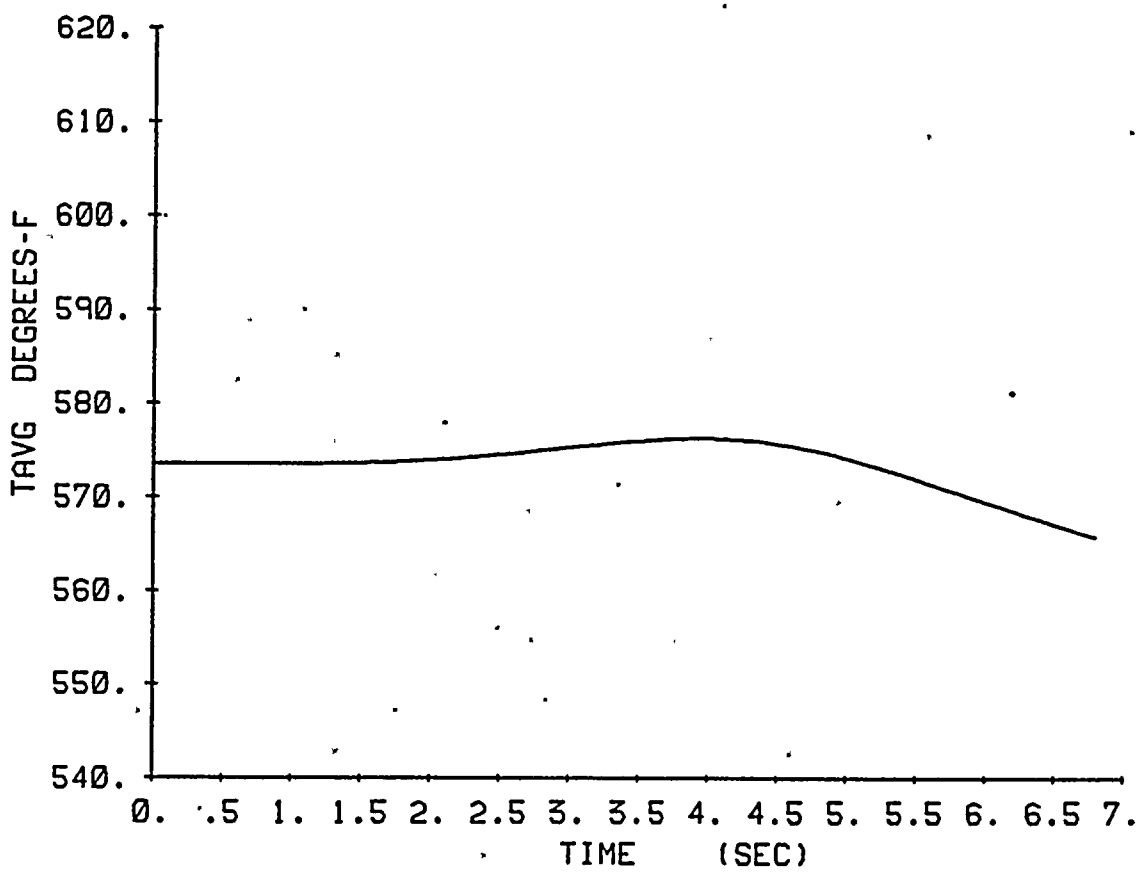


R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Maximum Feedback
100% Power, 90 pcm/sec
Figure 5.2-3



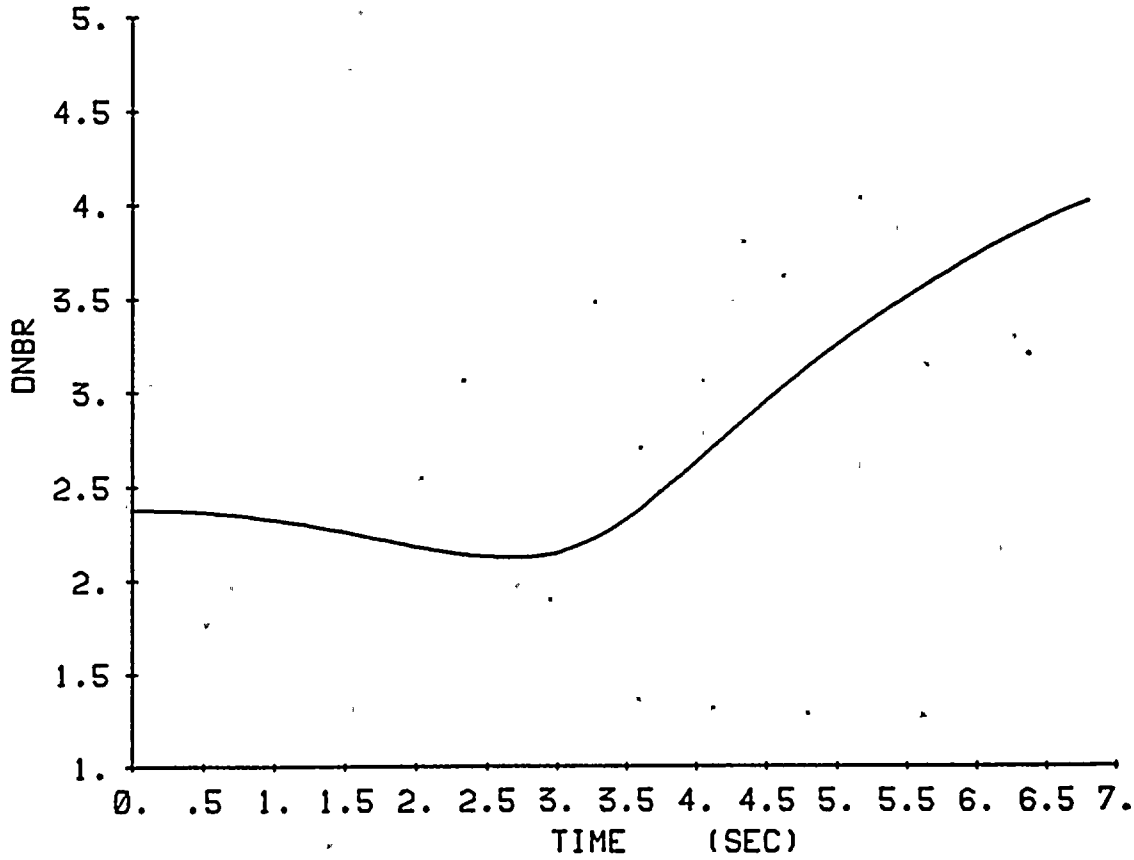
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Maximum Feedback
100% Power, 90 pcm/sec

Figure 5.2-4



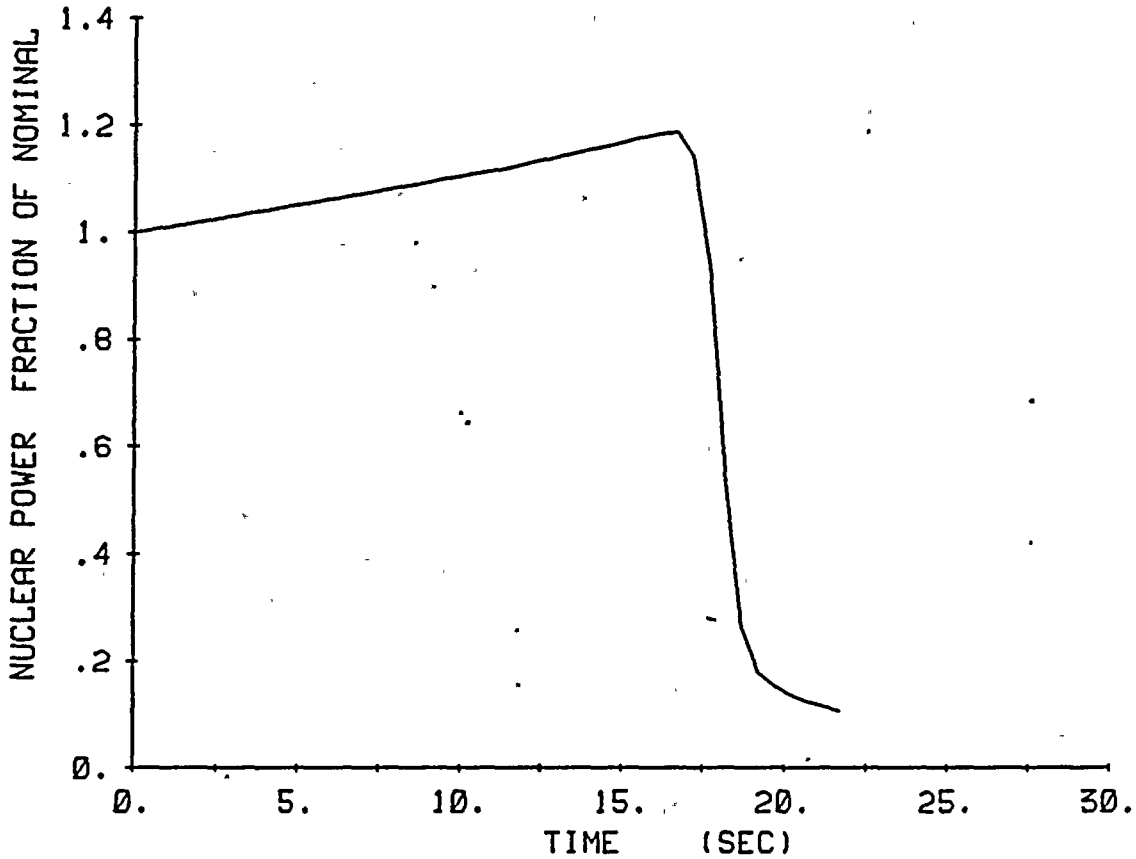
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Maximum Feedback
100% Power, 90 pcm/sec

Figure 5.2-5



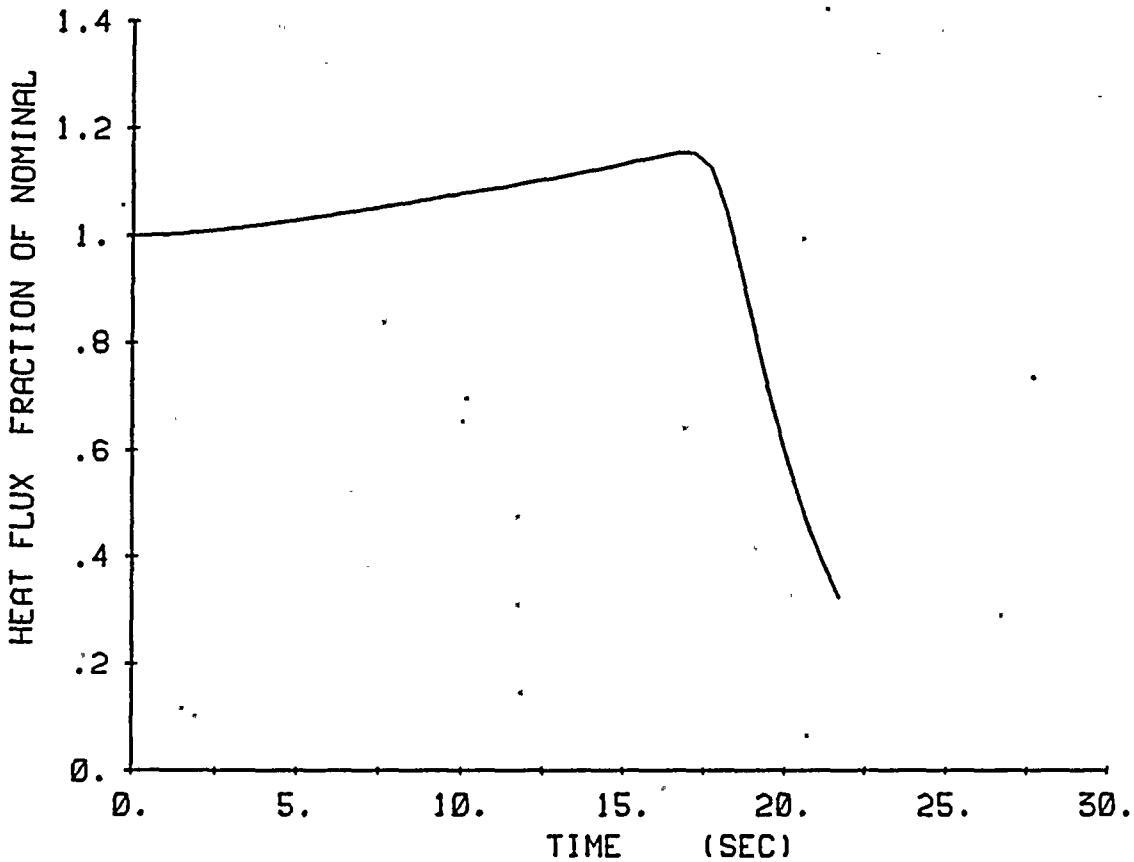
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Maximum Feedback
100% Power, 90 pcm/sec

Figure 5.2-6



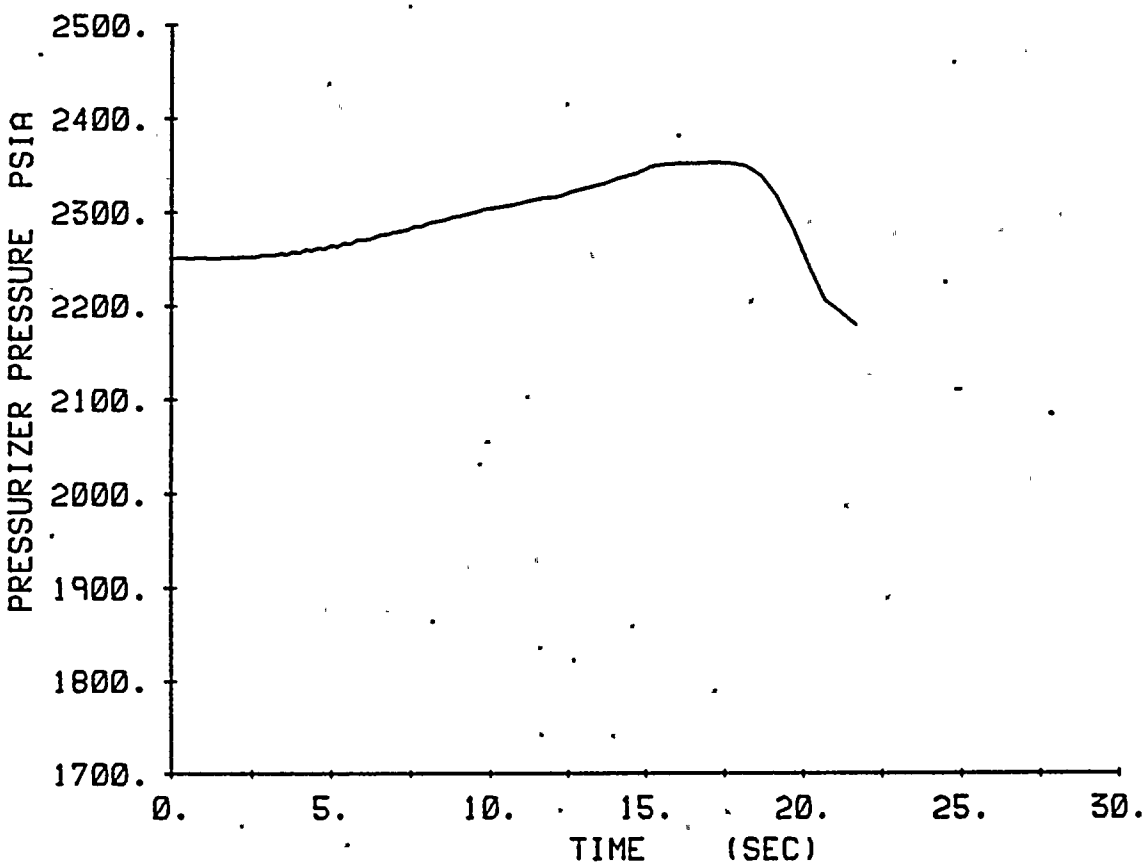
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Minimum Feedback
100% Power, 7 pcm/sec

Figure 5.2-7



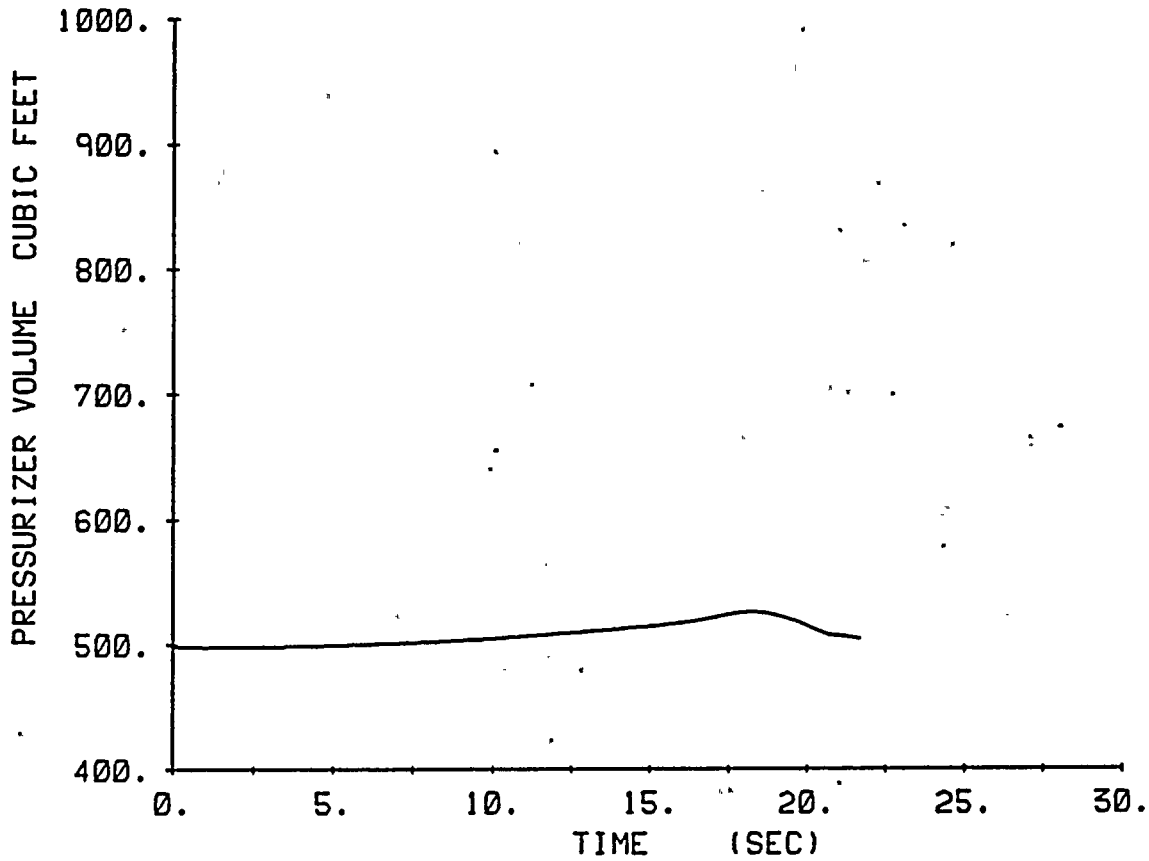
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Minimum Feedback
100% Power, 7 pcm/sec

Figure 5.2-8



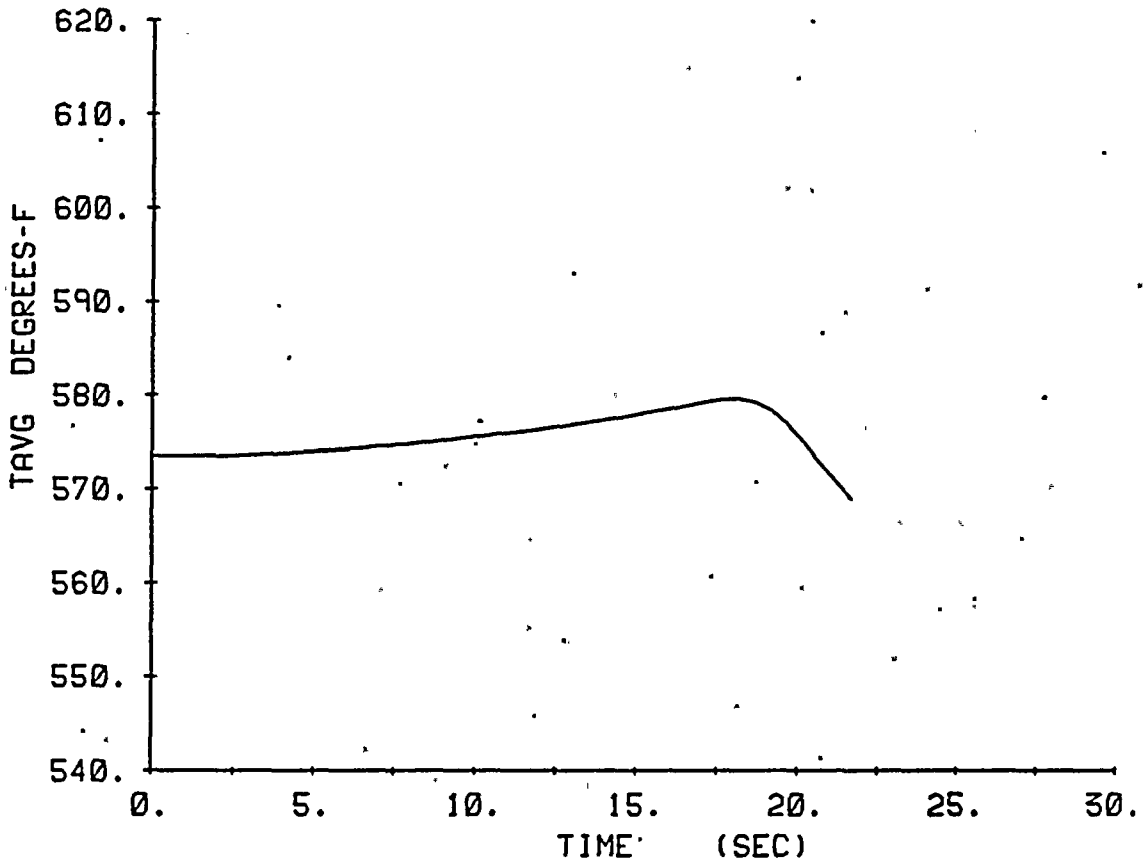
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Minimum Feedback
100% Power, 7 pcm/sec

Figure 5.2-9



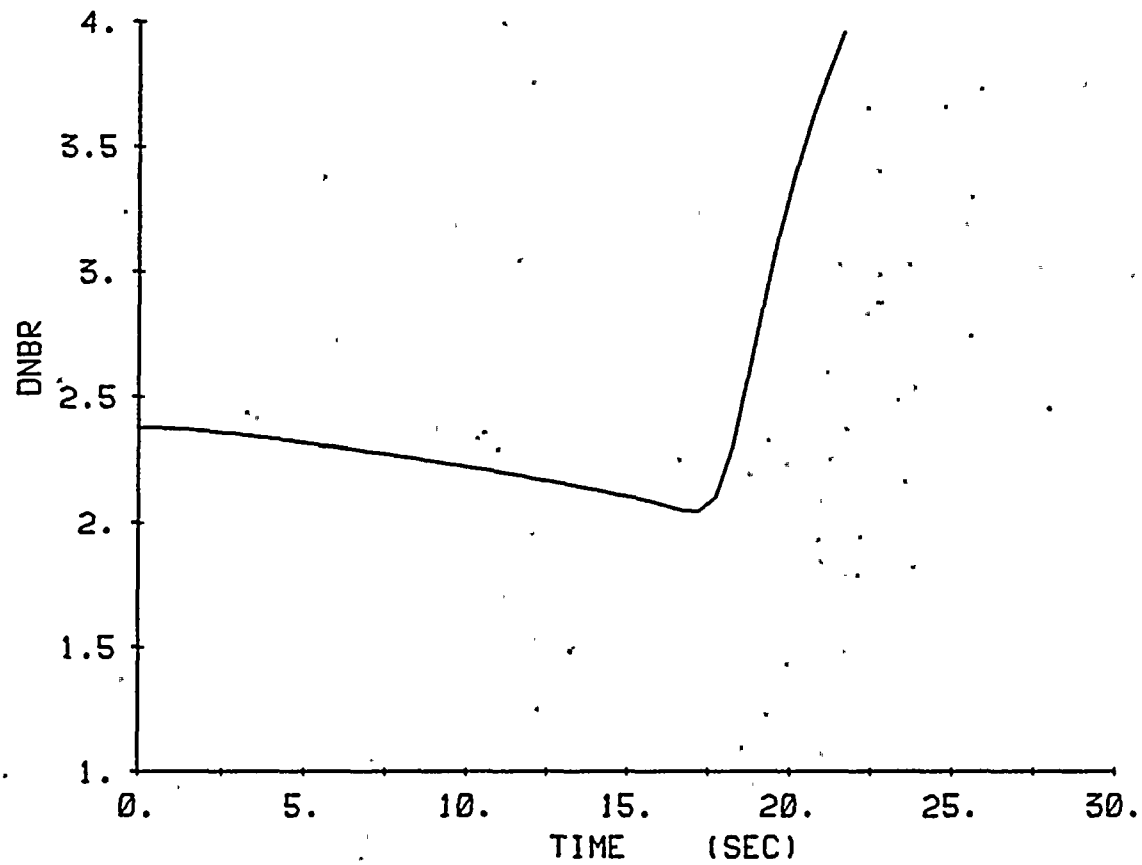
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Minimum Feedback
100% Power, 7 pcm/sec

Figure 5.2-10



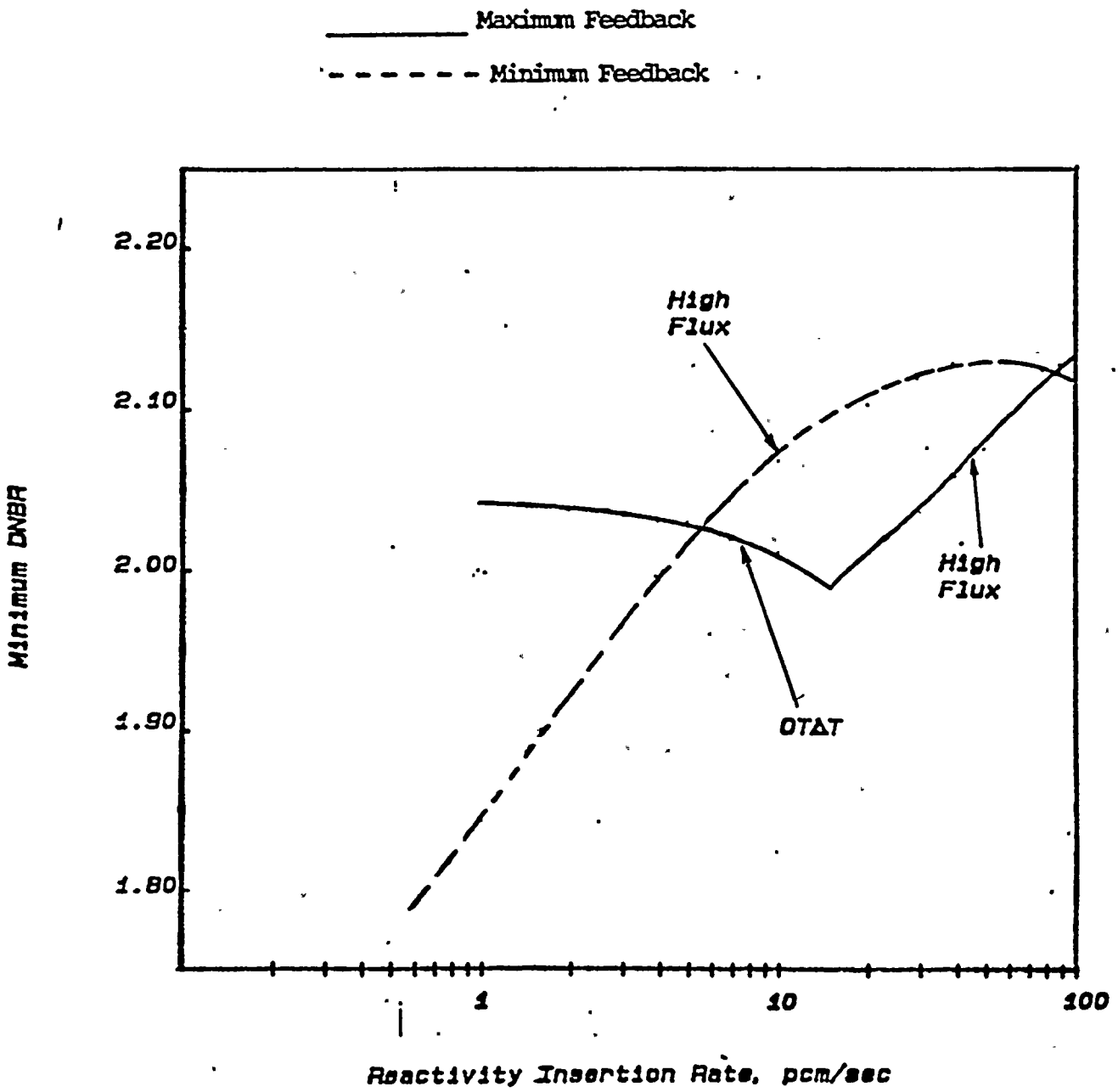
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Minimum Feedback
100% Power, 7 pcm/sec

Figure 5.2-11



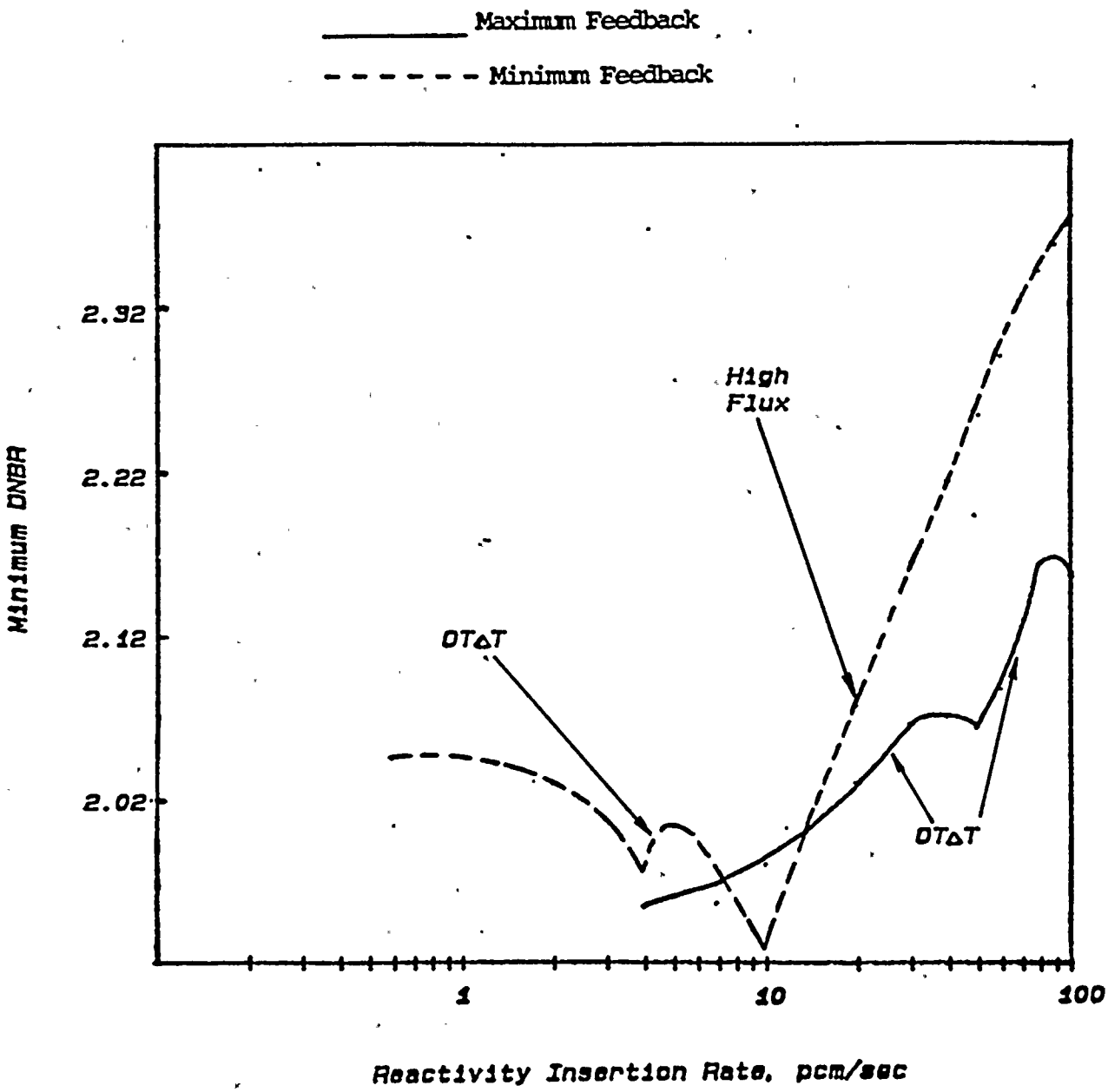
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
Minimum Feedback
100% Power, 7 pcm/sec

Figure 5.2.12



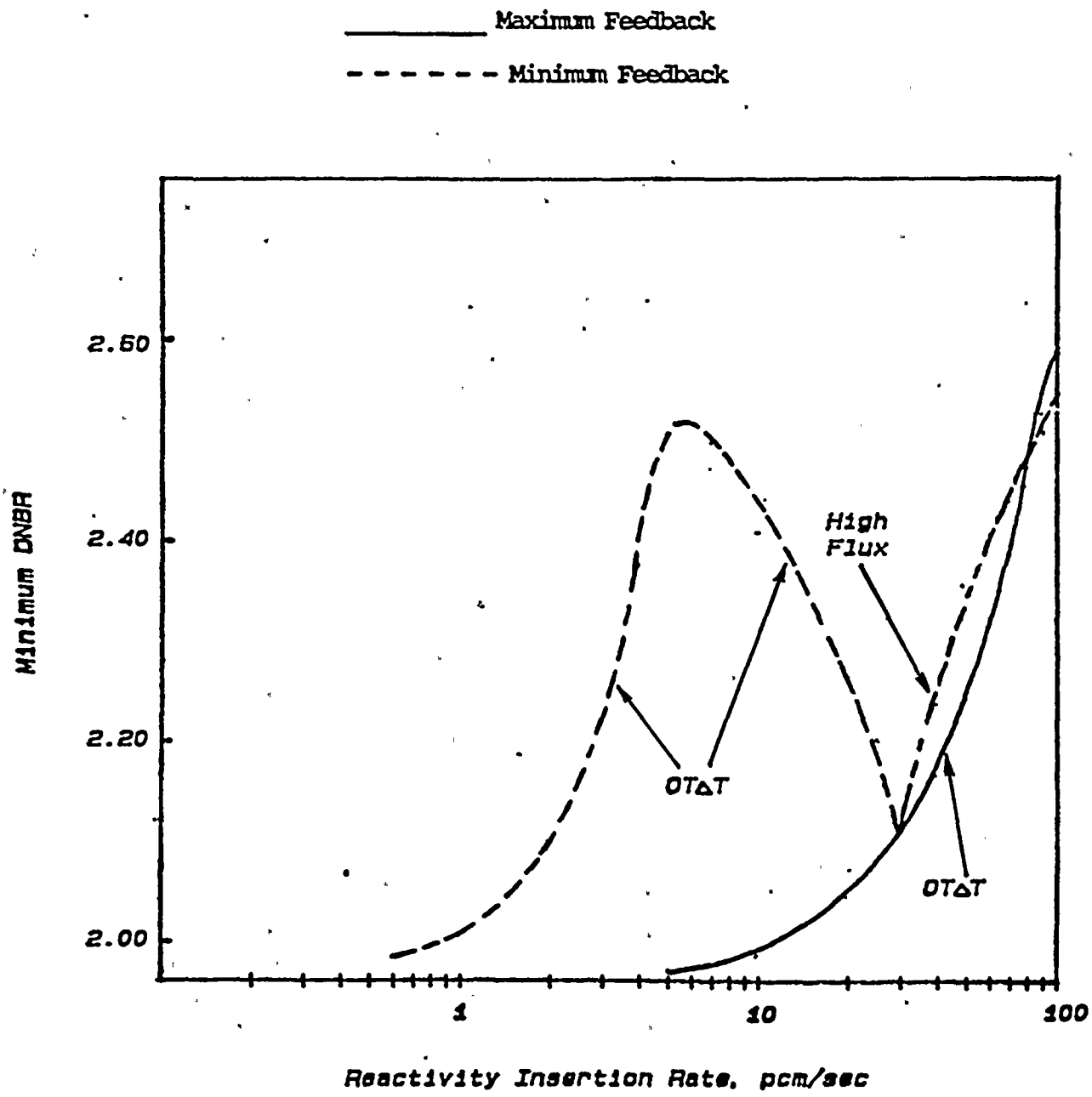
R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
100% Power

FIGURE 5.2-13



R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
60% Power

FIGURE 5.2-14



R. E. GINNA 15% SGTP
Uncontrolled RCCA Bank Withdrawal at Power
10% Power

FIGURE 5.2-15

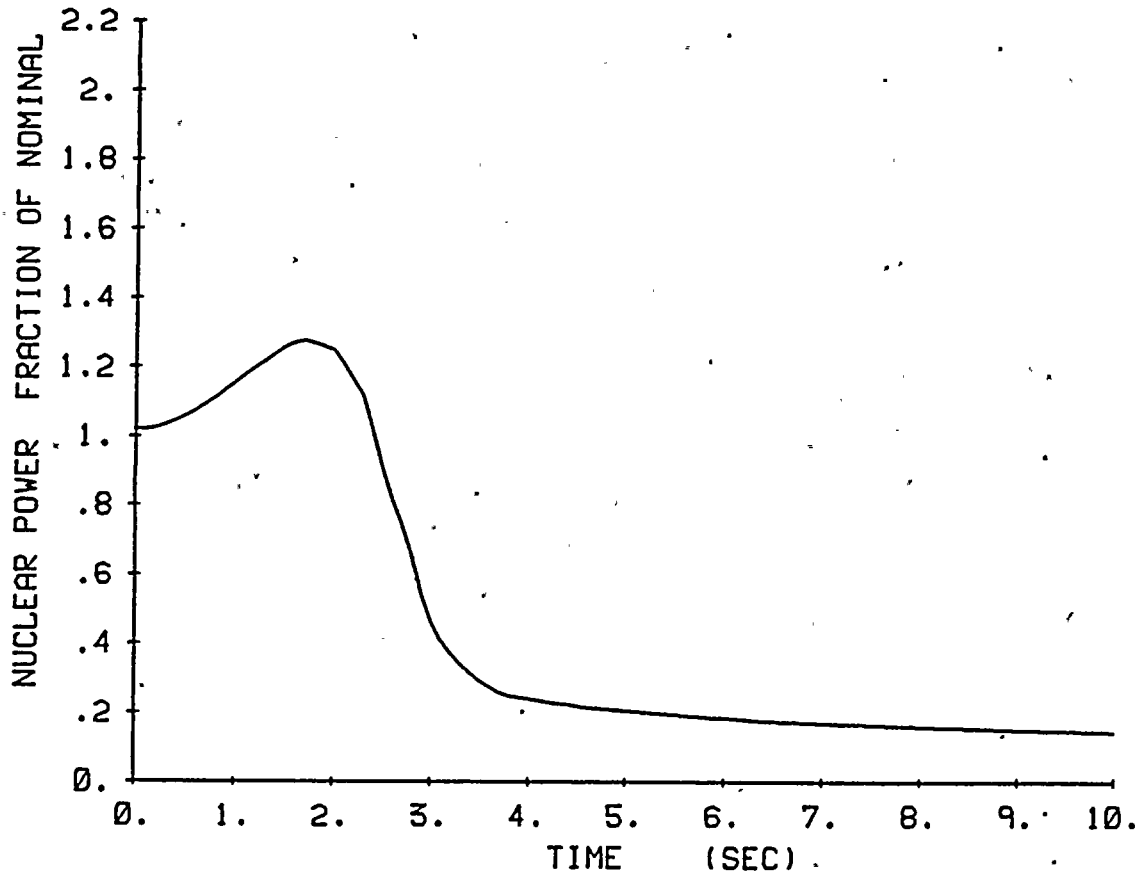


FIGURE 5.4-1
R. E. Ginna 15% SGTP
Locked Rotor

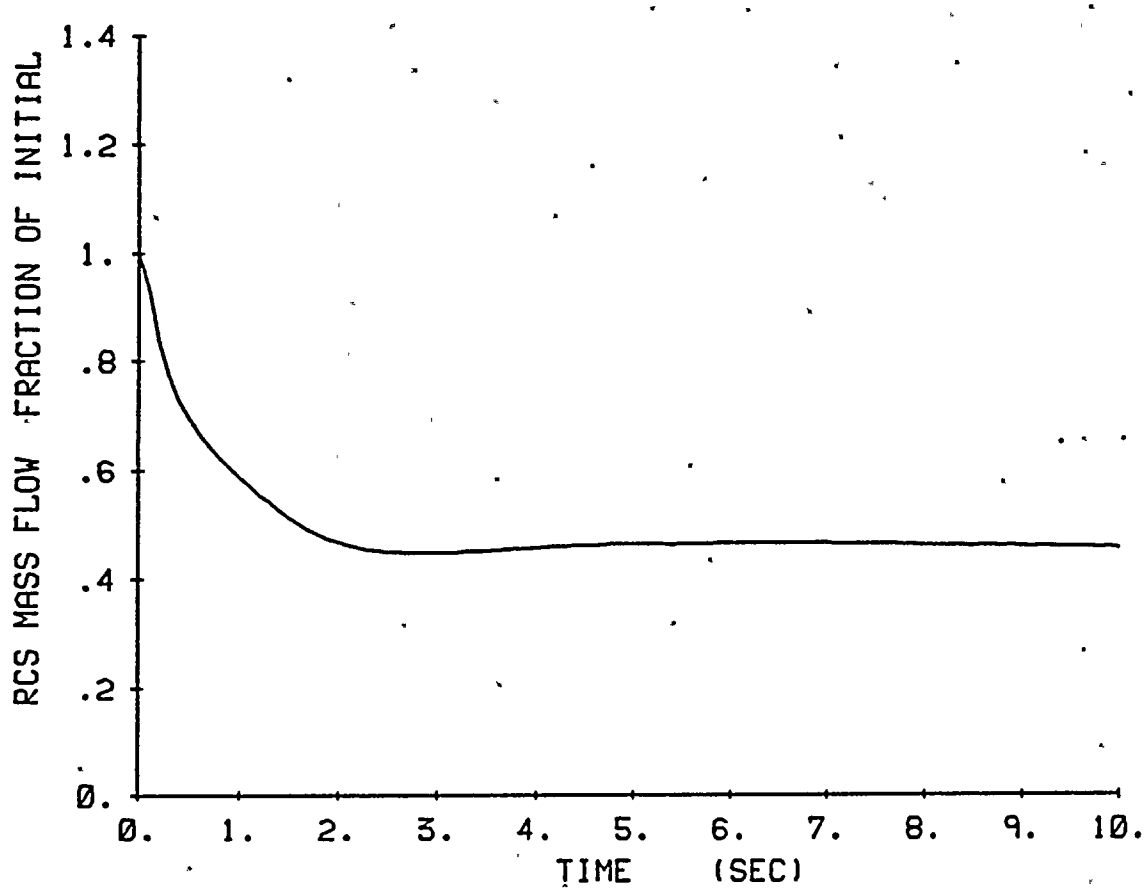


FIGURE 5.4-2
R. E. Ginna 15% SGTP
Locked Rotor

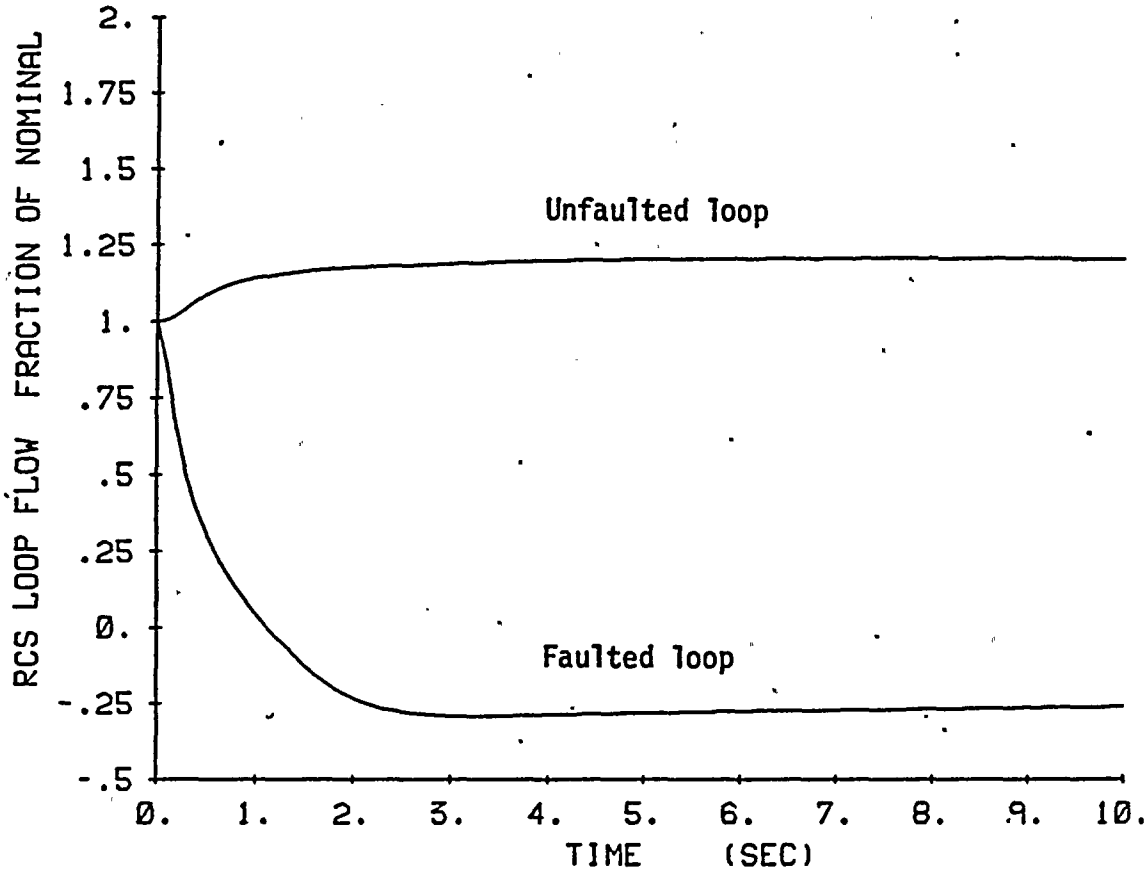


FIGURE 5.4-3
R. E. Ginna 15% SGTP
Locked Rotor

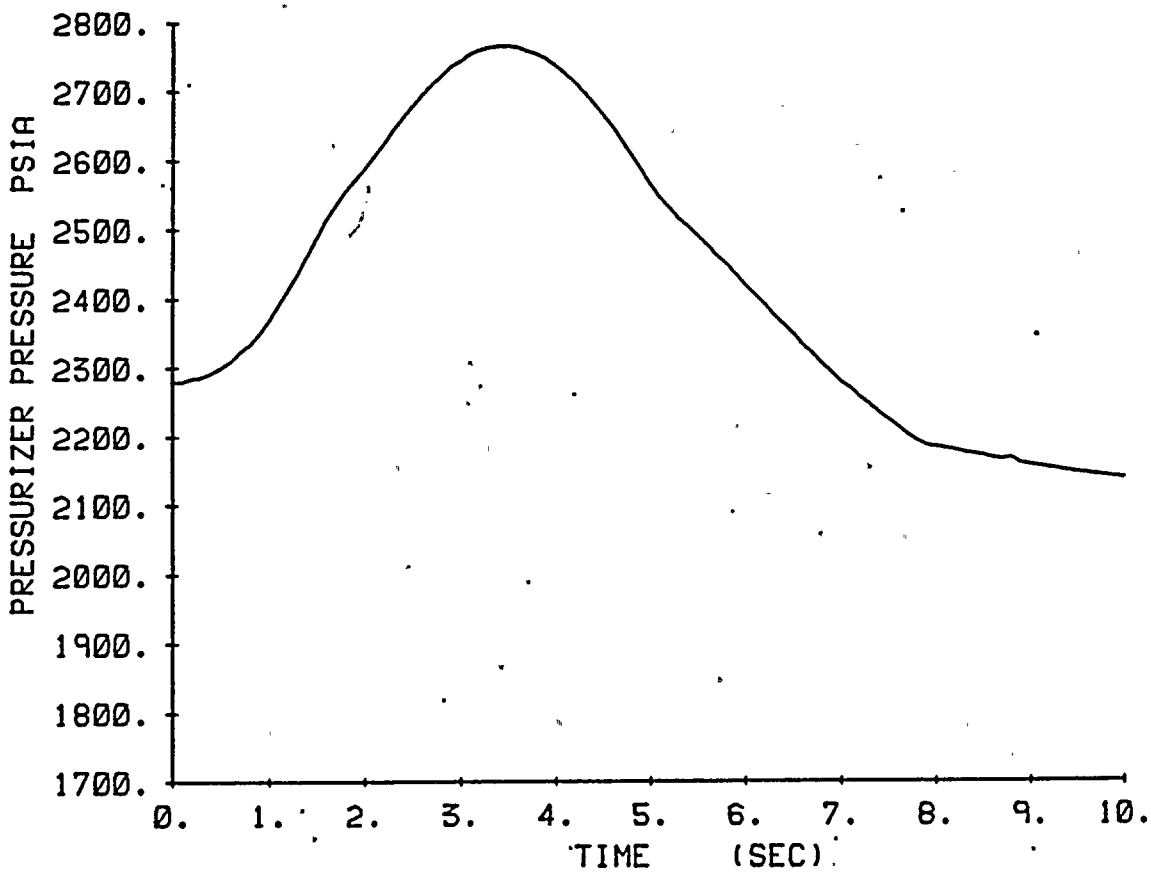


FIGURE 5.4-4
R. E. Ginna 15% SGTP
Locked Rotor

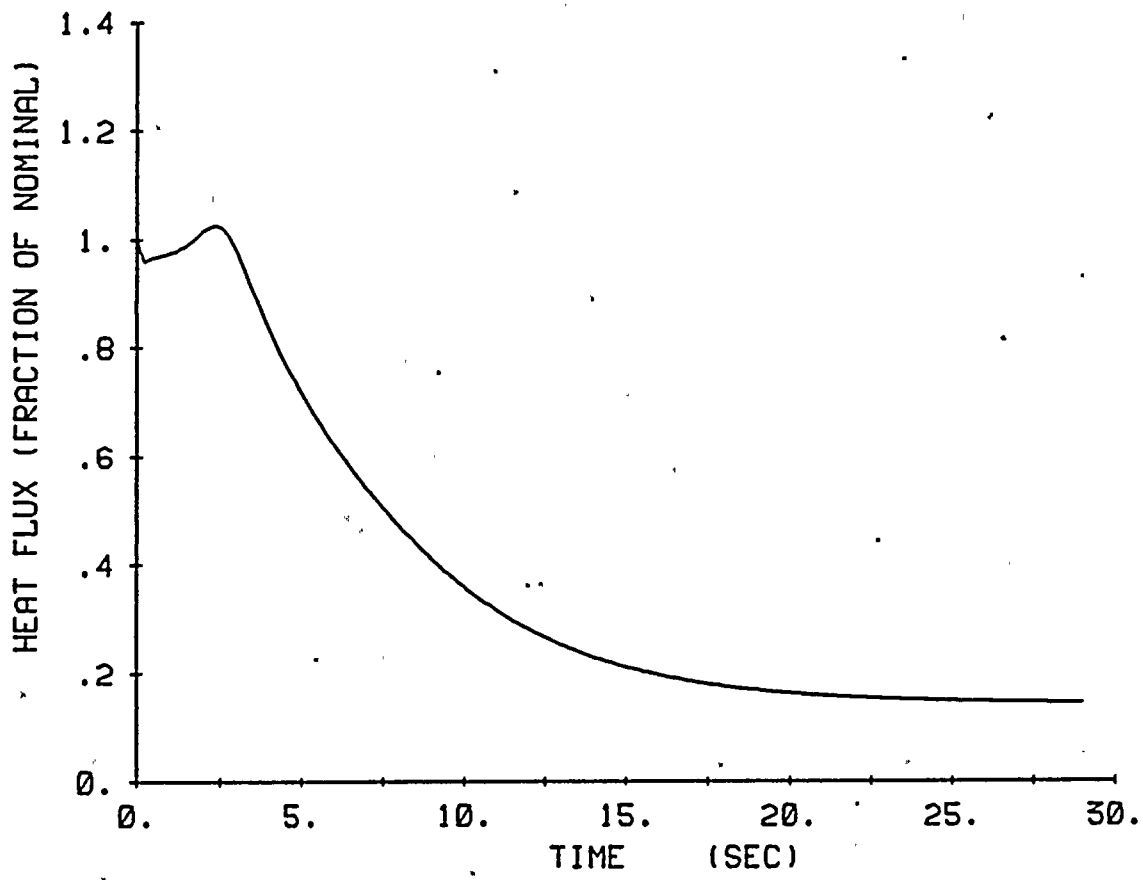


FIGURE 5.4-5
R. E. Ginna 15% SGTP
Locked Rotor

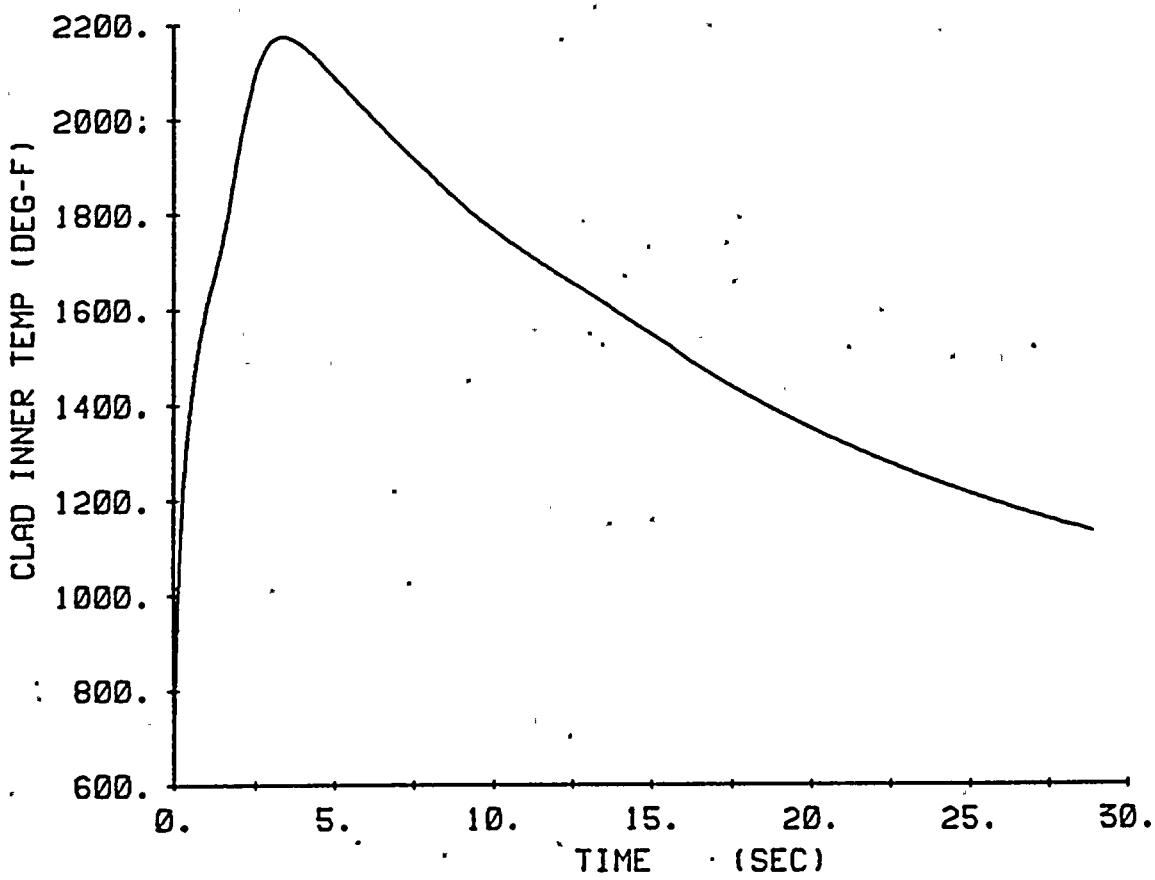


FIGURE 5.4-6
R. E. Ginna 15% SGTP
Locked Rotor

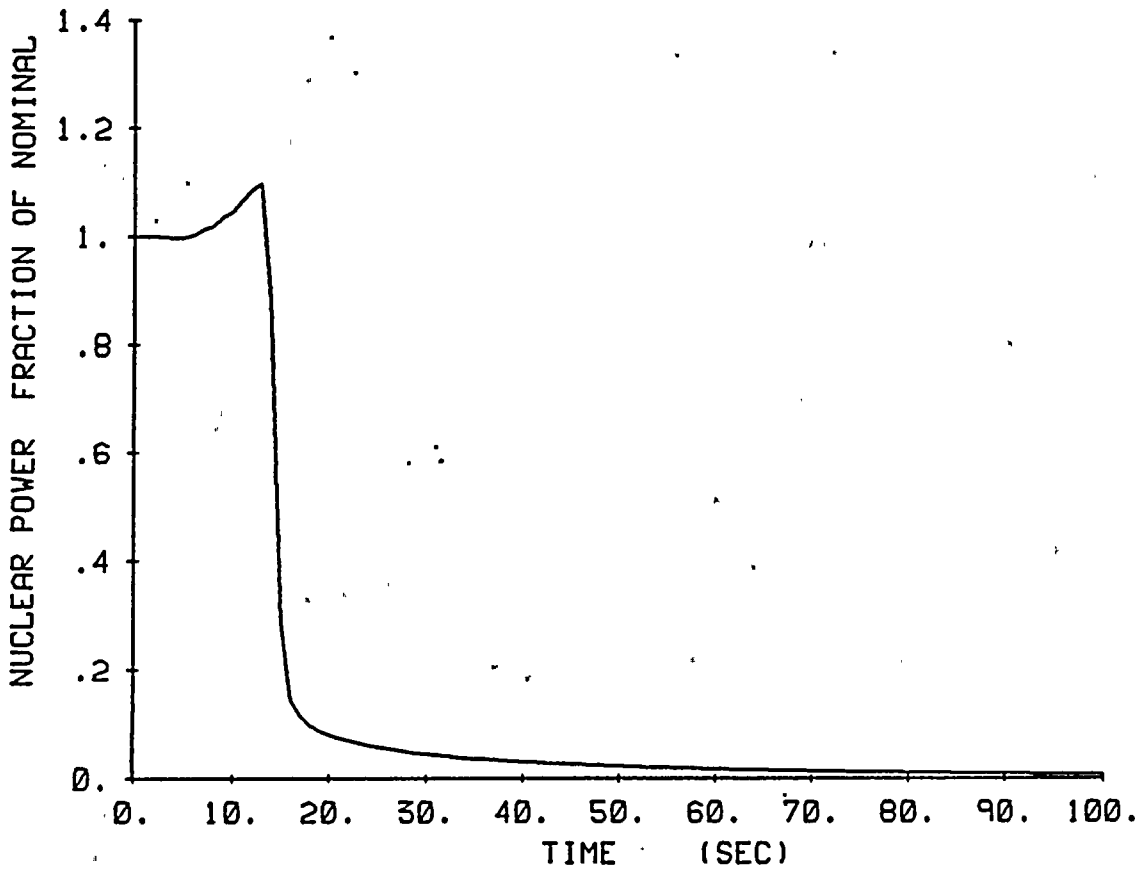


FIGURE 5.5-1

R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback with Automatic Pressure Control

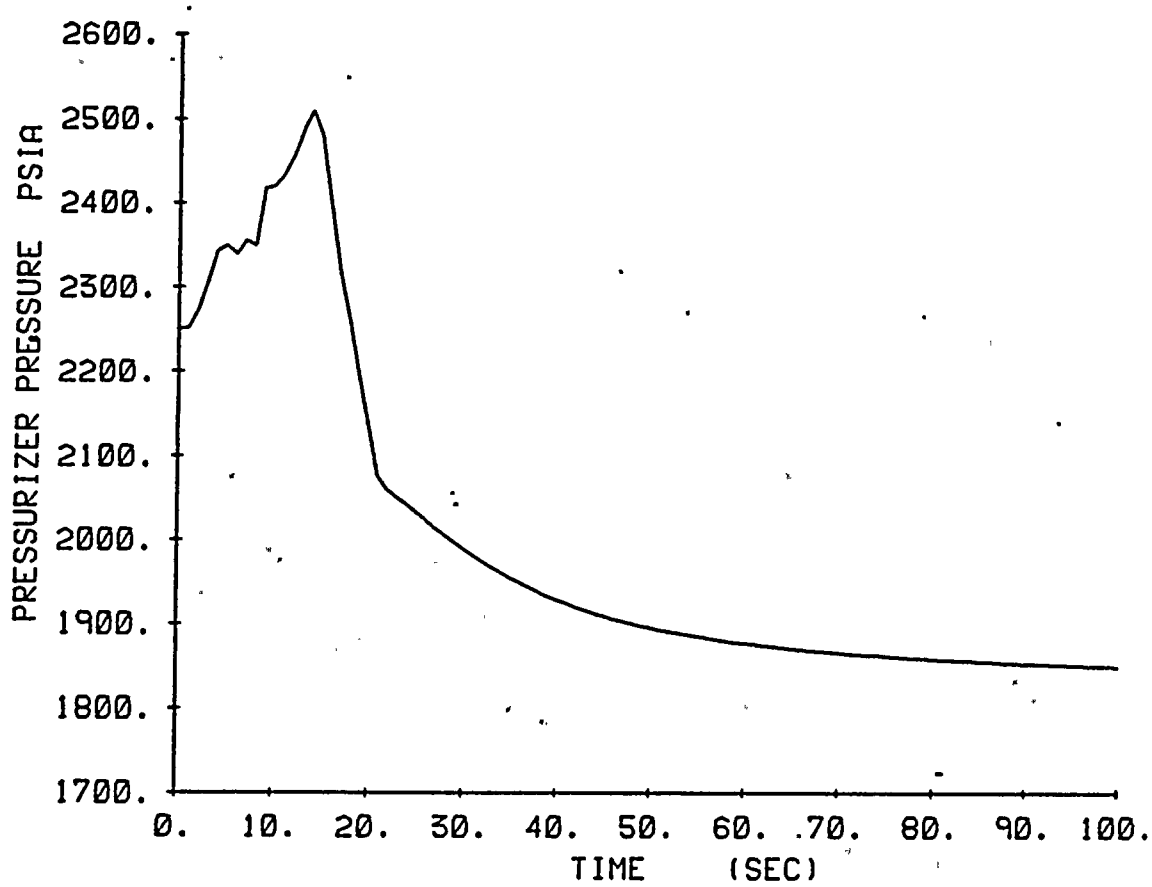


FIGURE 5.5-2
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback with Automatic Pressure Control

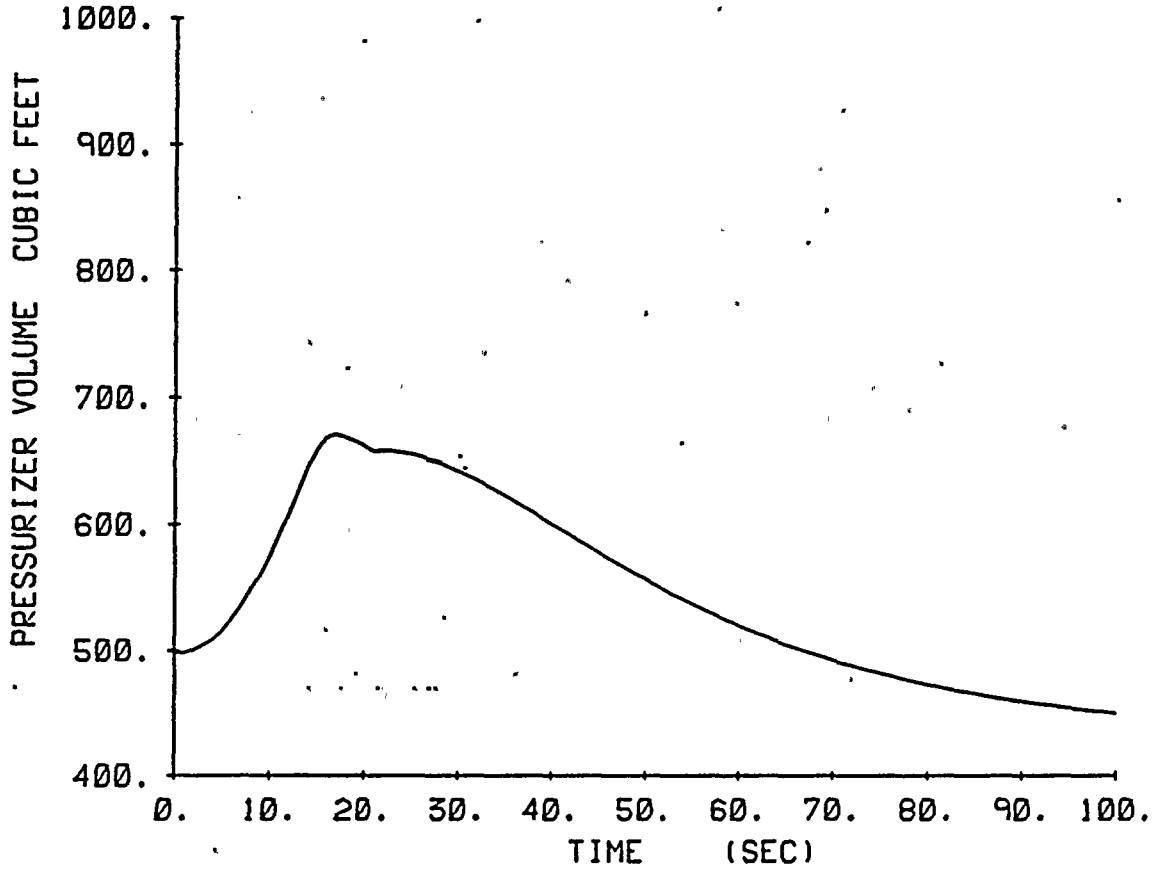


FIGURE 5.5-3
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback with Automatic Pressure Control

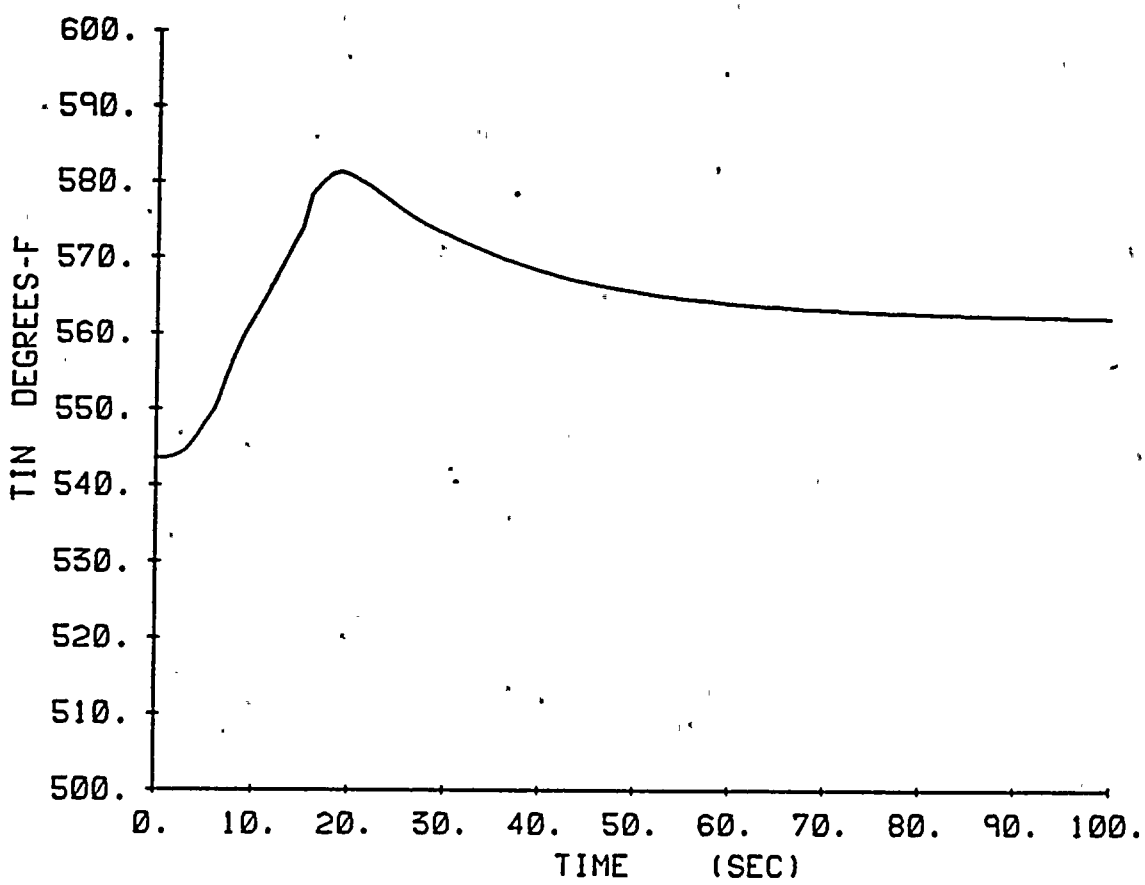


FIGURE 5.5-4
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback with Automatic Pressure Control

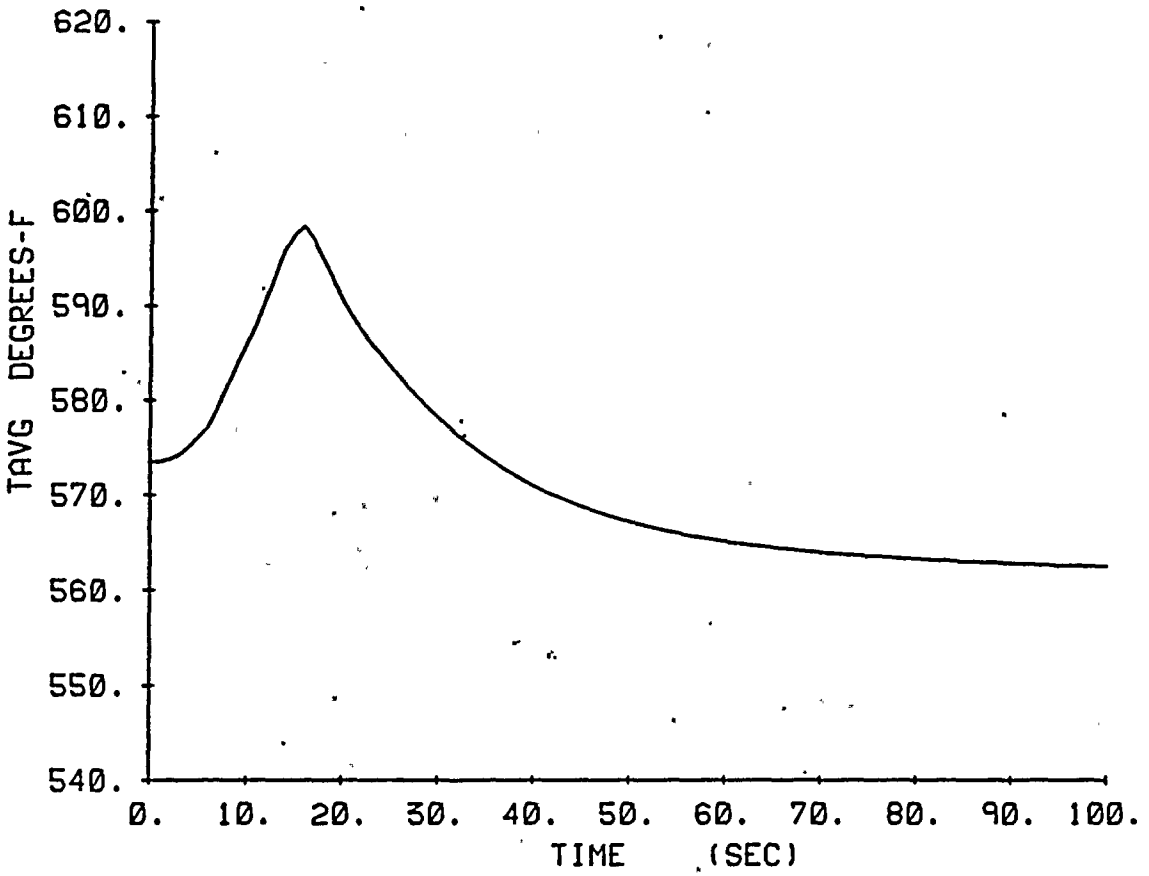


FIGURE 5.5-5
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback with Automatic Pressure Control

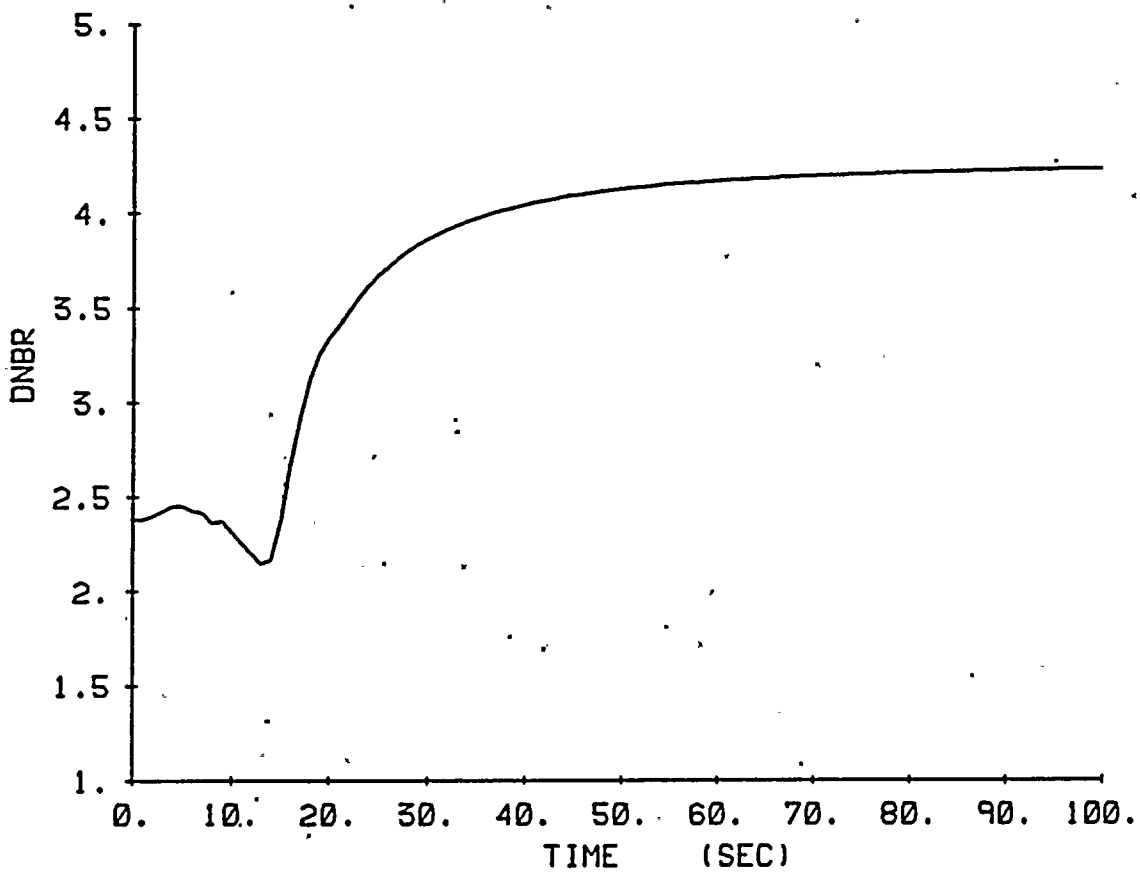


FIGURE 5.5-6

R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback with Automatic Pressure Control

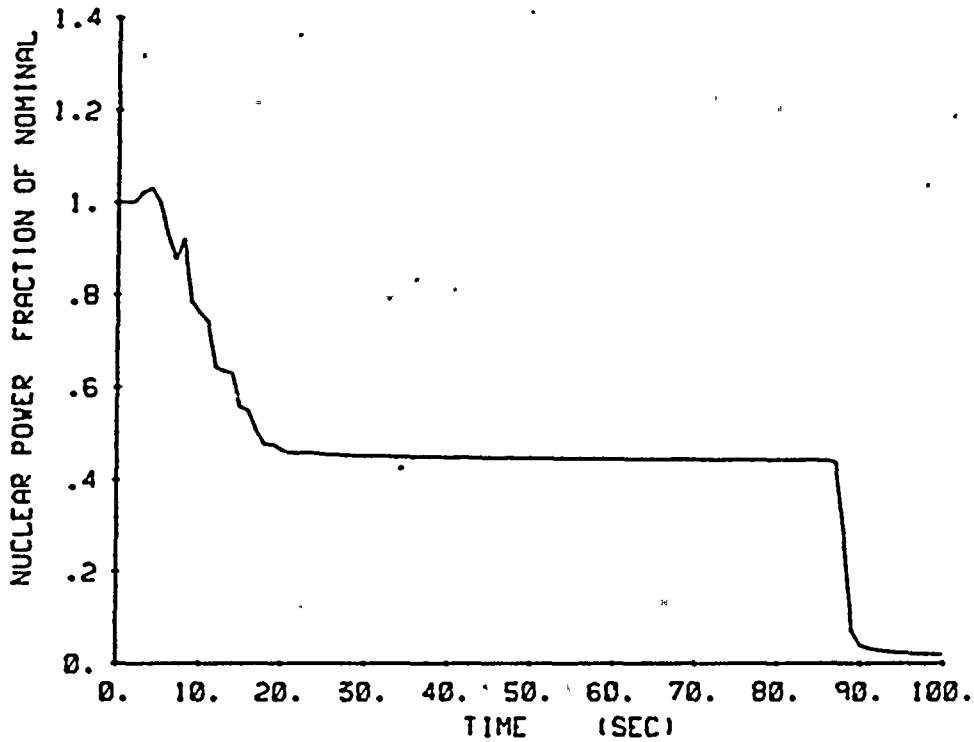


FIGURE 5.5-7
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback with Automatic Pressure Control

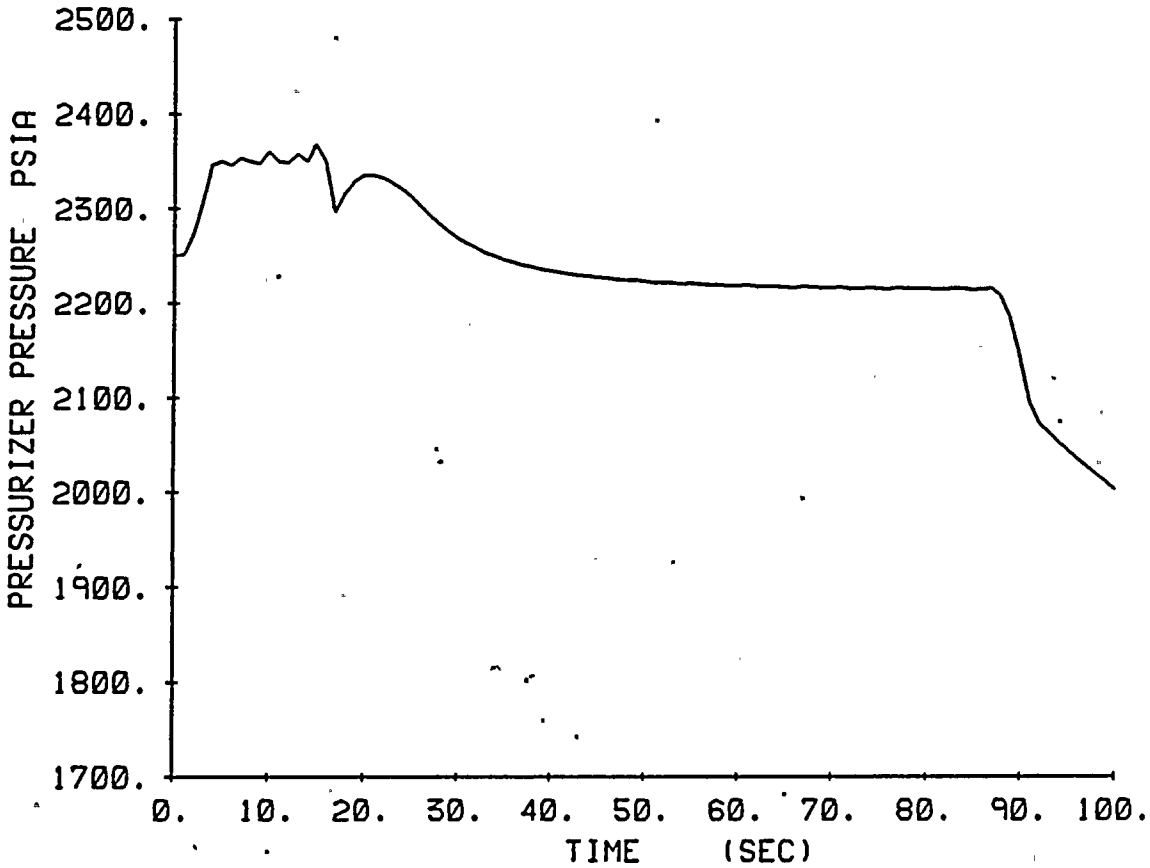


FIGURE 5.5-8
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback with Automatic Pressure Control

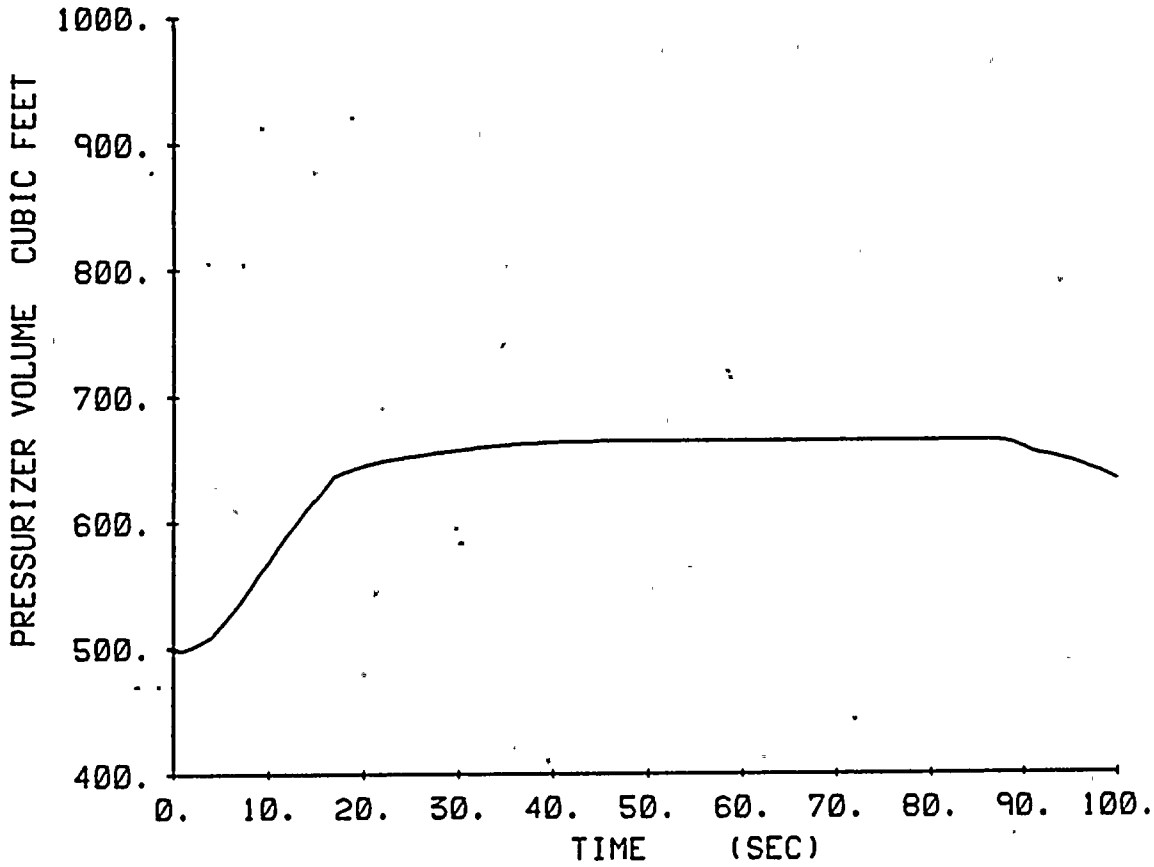


FIGURE 5.5-9
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback with Automatic Pressure Control

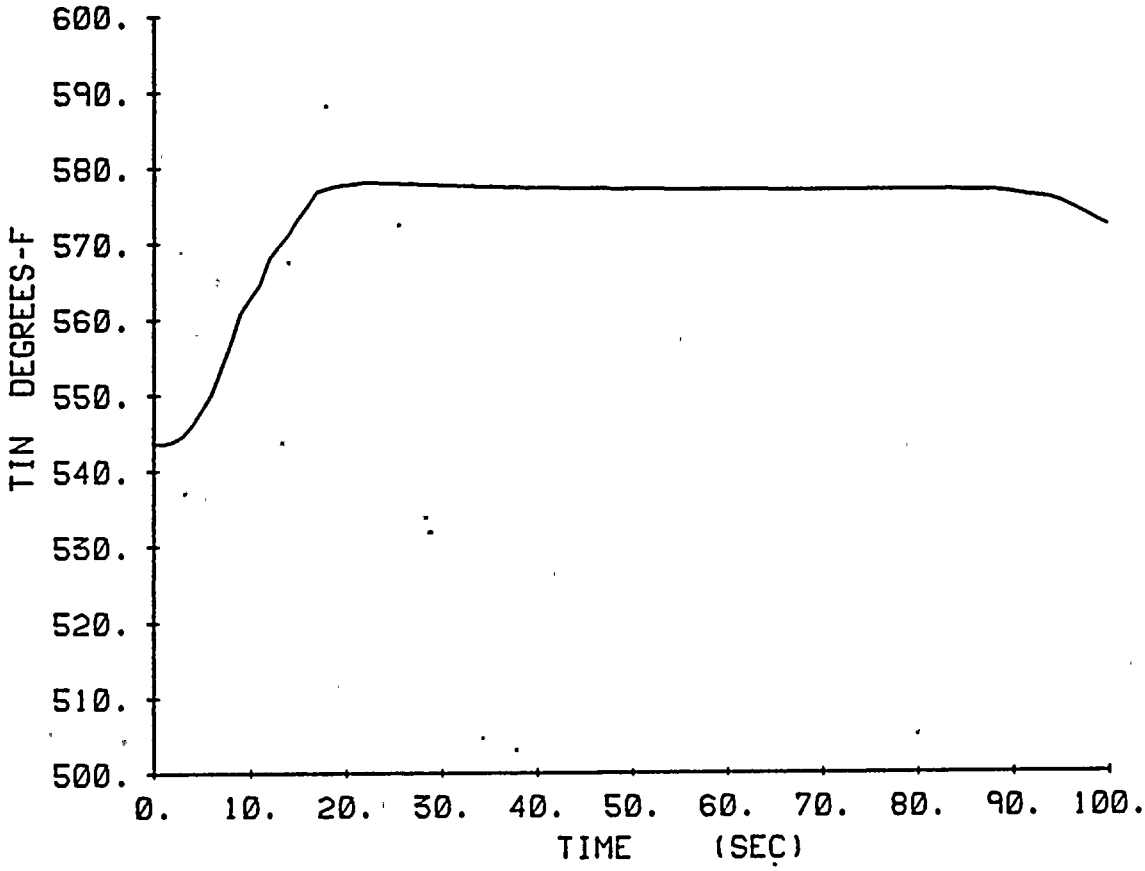


FIGURE 5.5-10
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback with Automatic Pressure Control

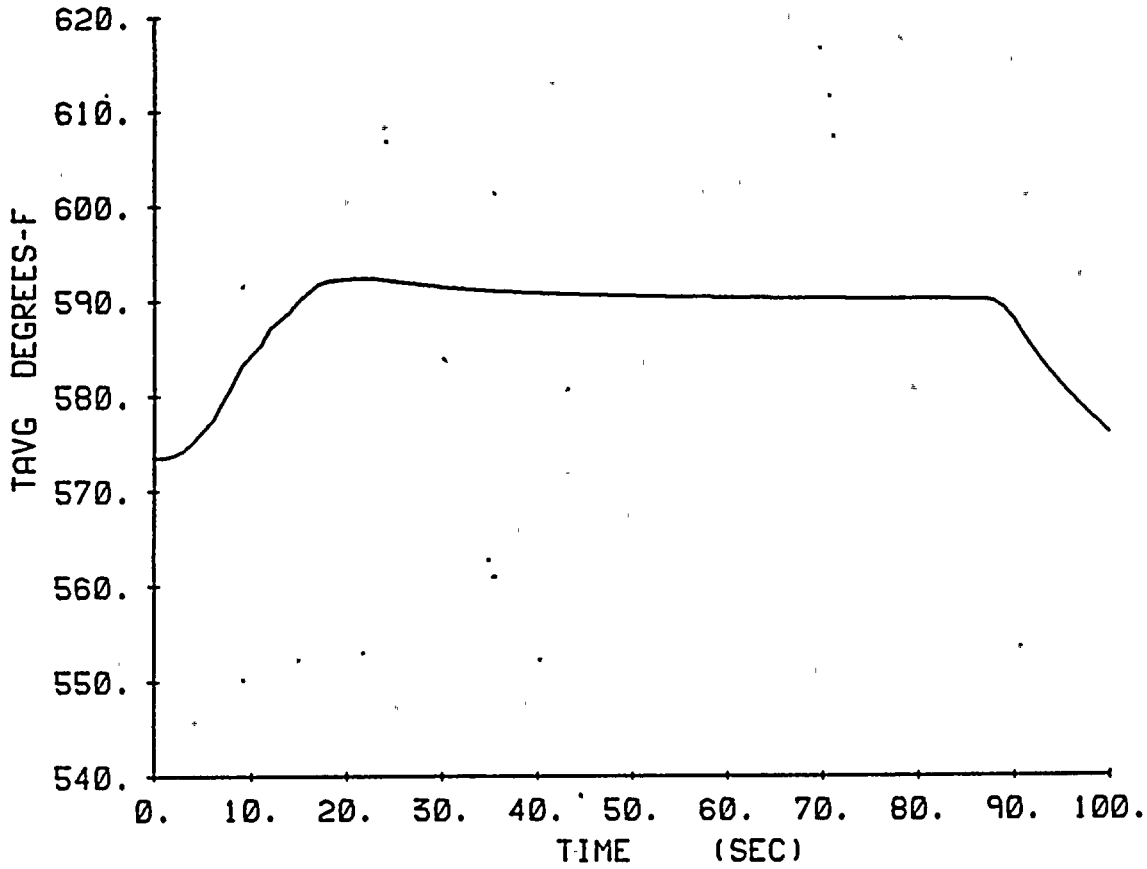


FIGURE 5.5-11
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback with Automatic Pressure Control

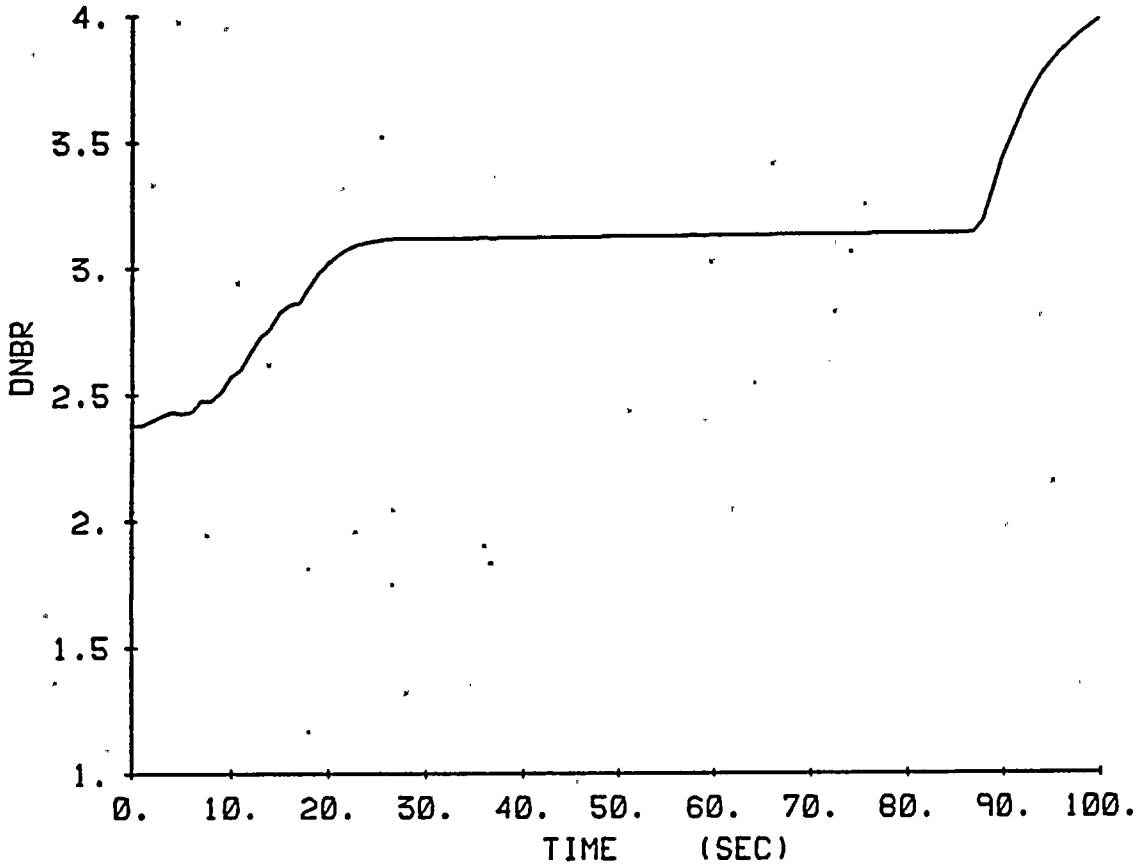


FIGURE 5.5-12
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback with Automatic Pressure Control

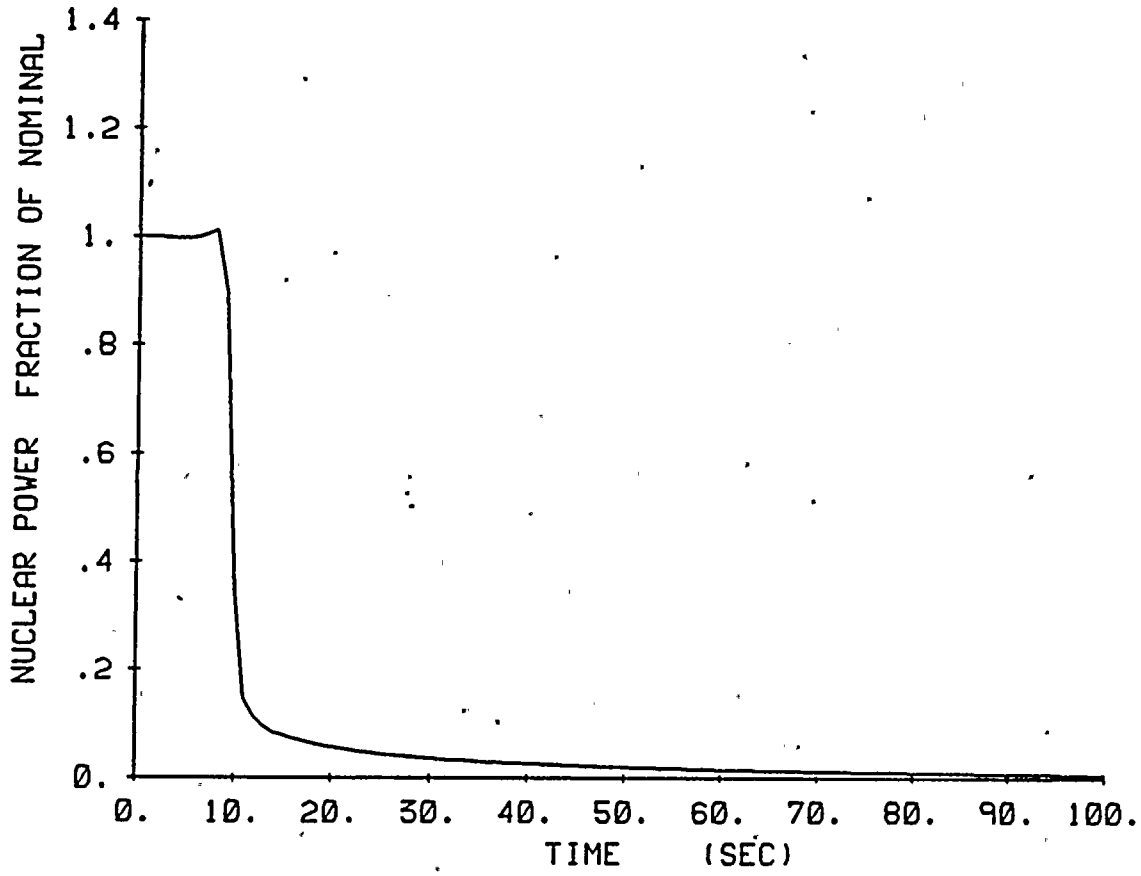


FIGURE 5.5-13
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback without Pressure Control

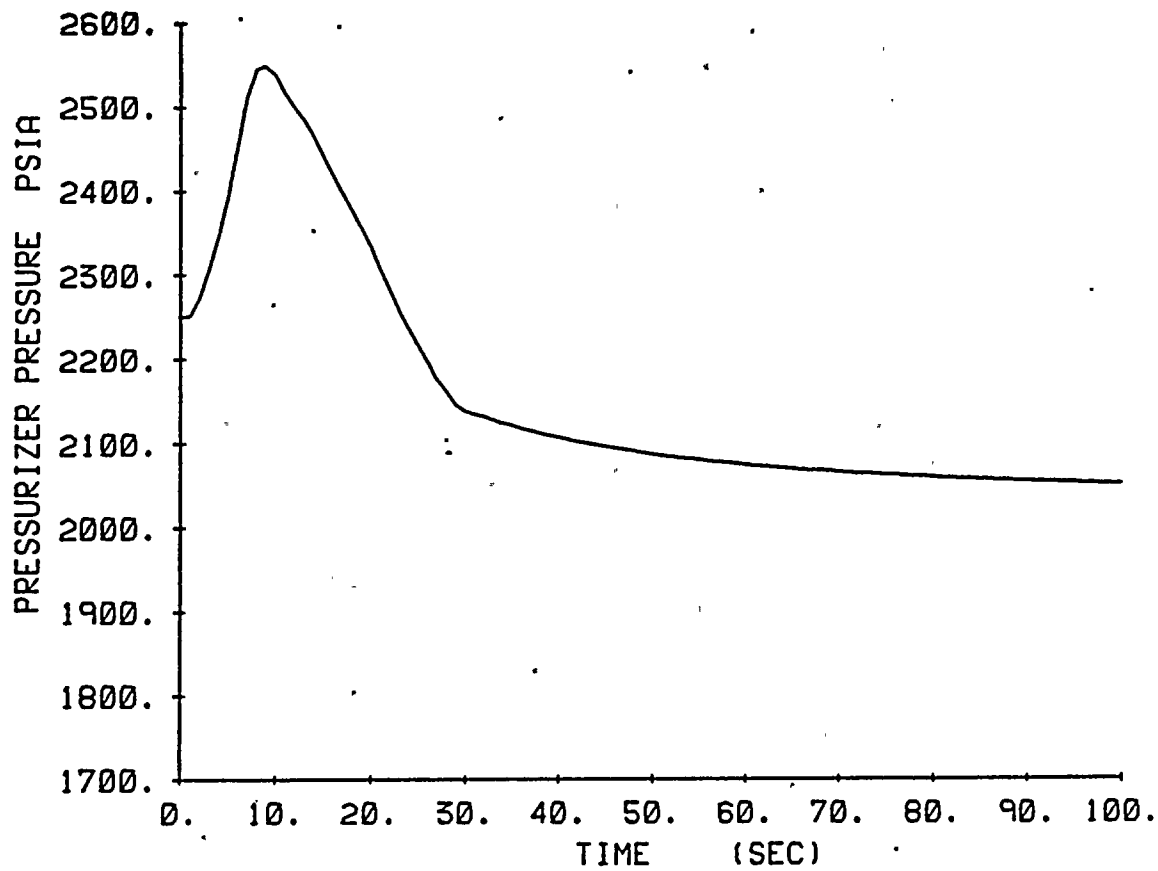


FIGURE 5.5-14
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback without Pressure Control

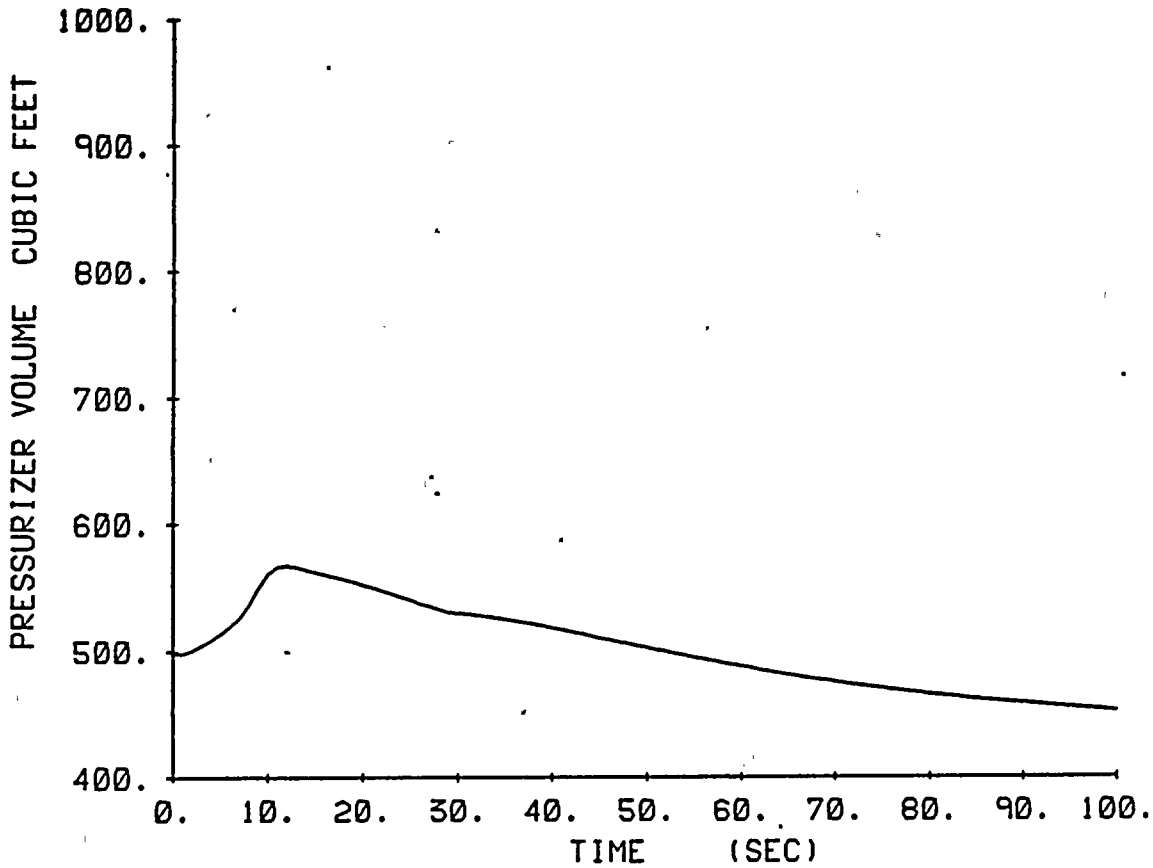


FIGURE 5.5-15
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback without Pressure Control

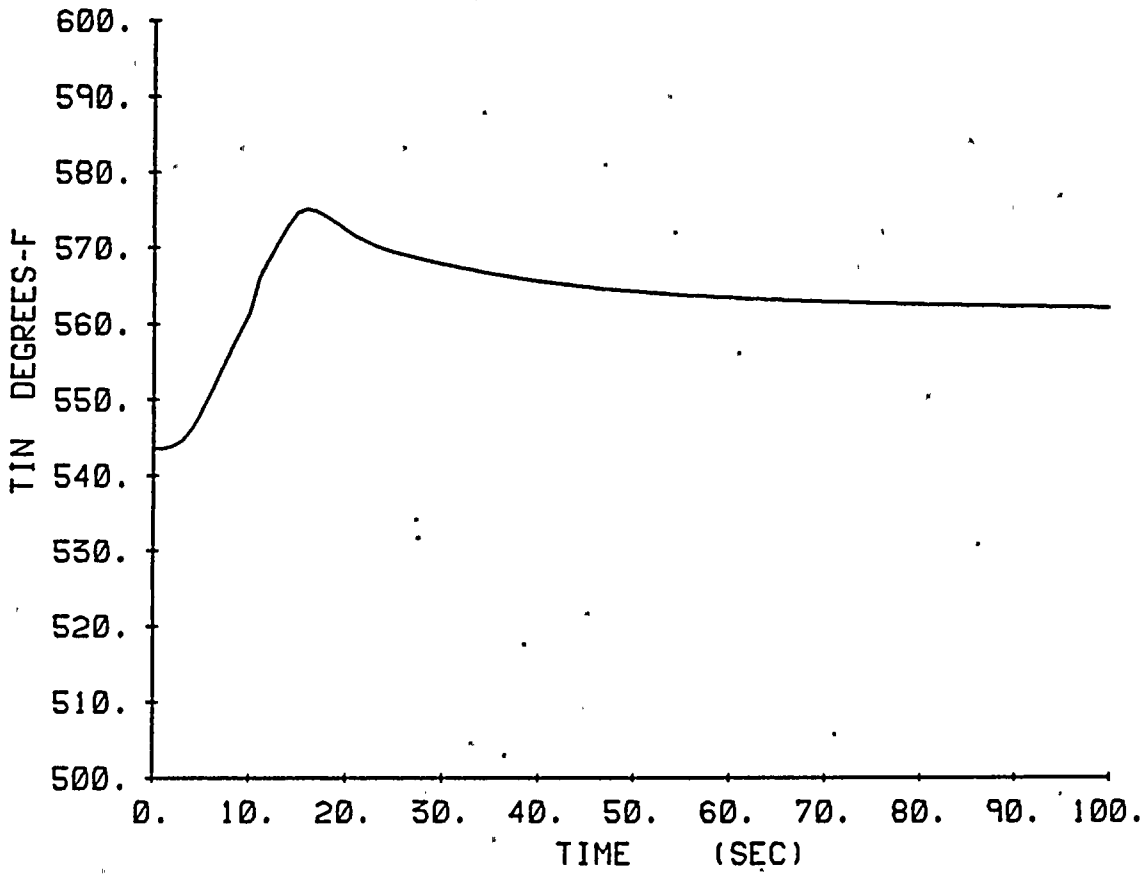


FIGURE 5.5-16
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback without Pressure Control

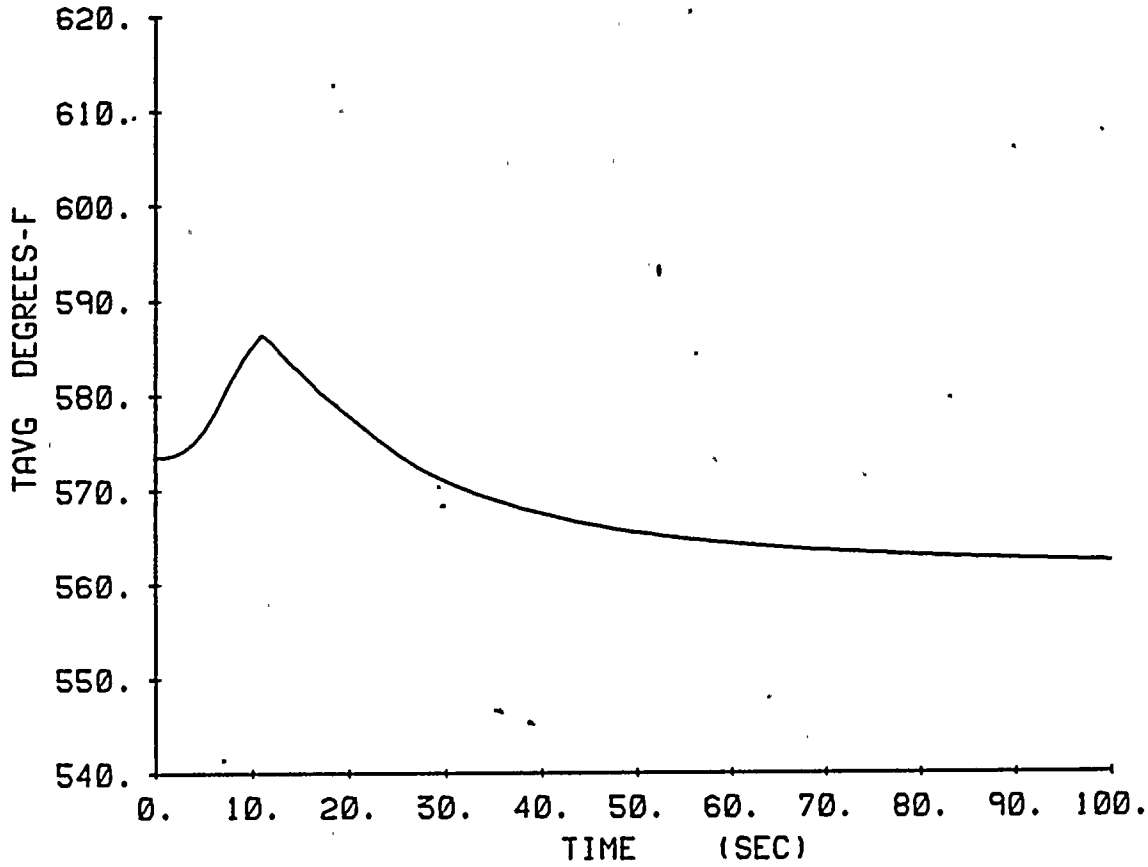


FIGURE 5.5-17
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback without Pressure Control

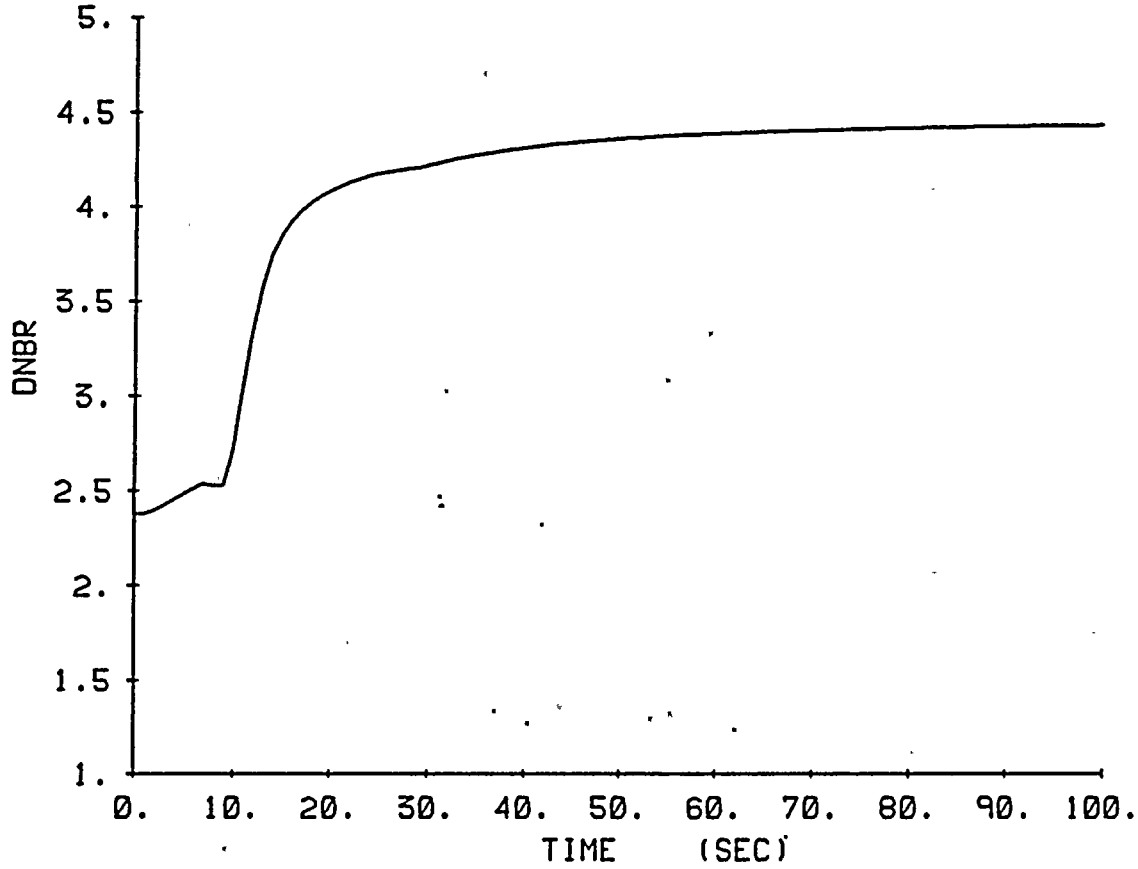


FIGURE 5.5-18
R. E. GINNA 15% SGTP
Loss of Load
Minimum Feedback without Pressure Control

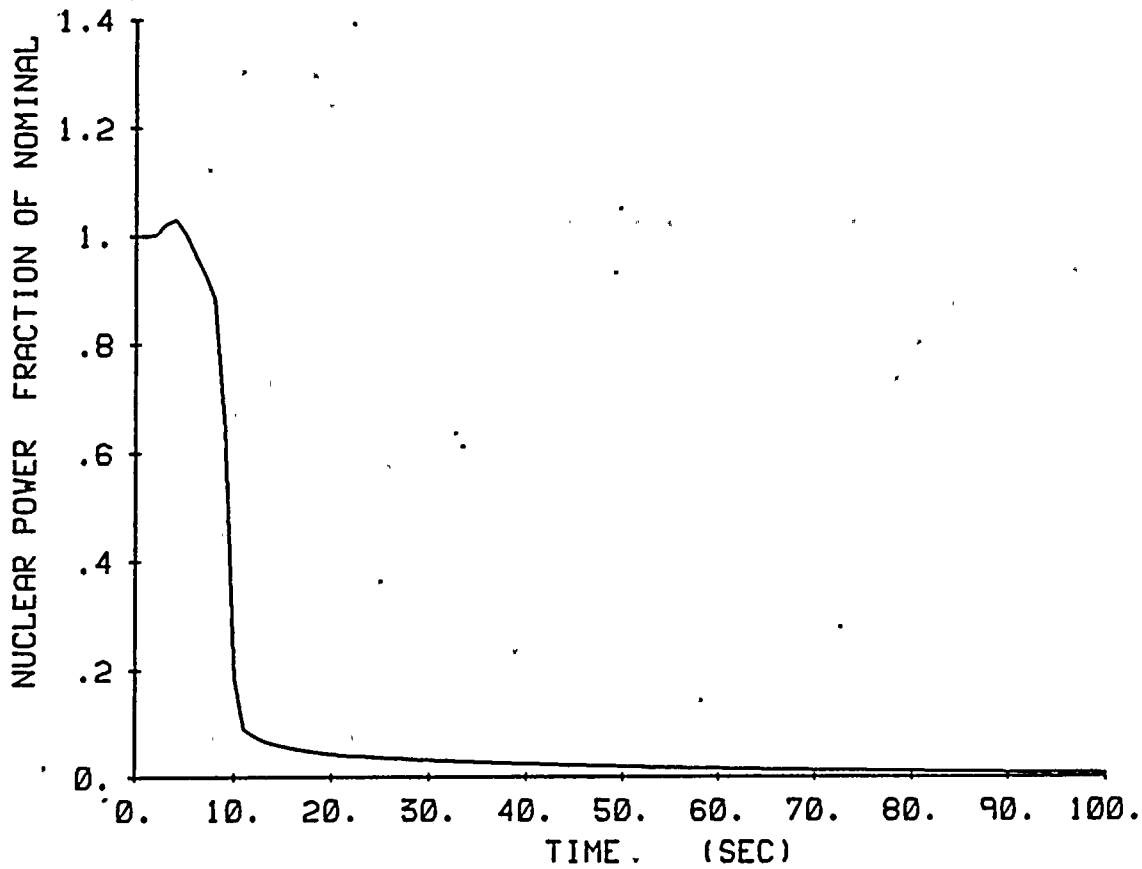


FIGURE 5.5-19
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback without Pressure Control

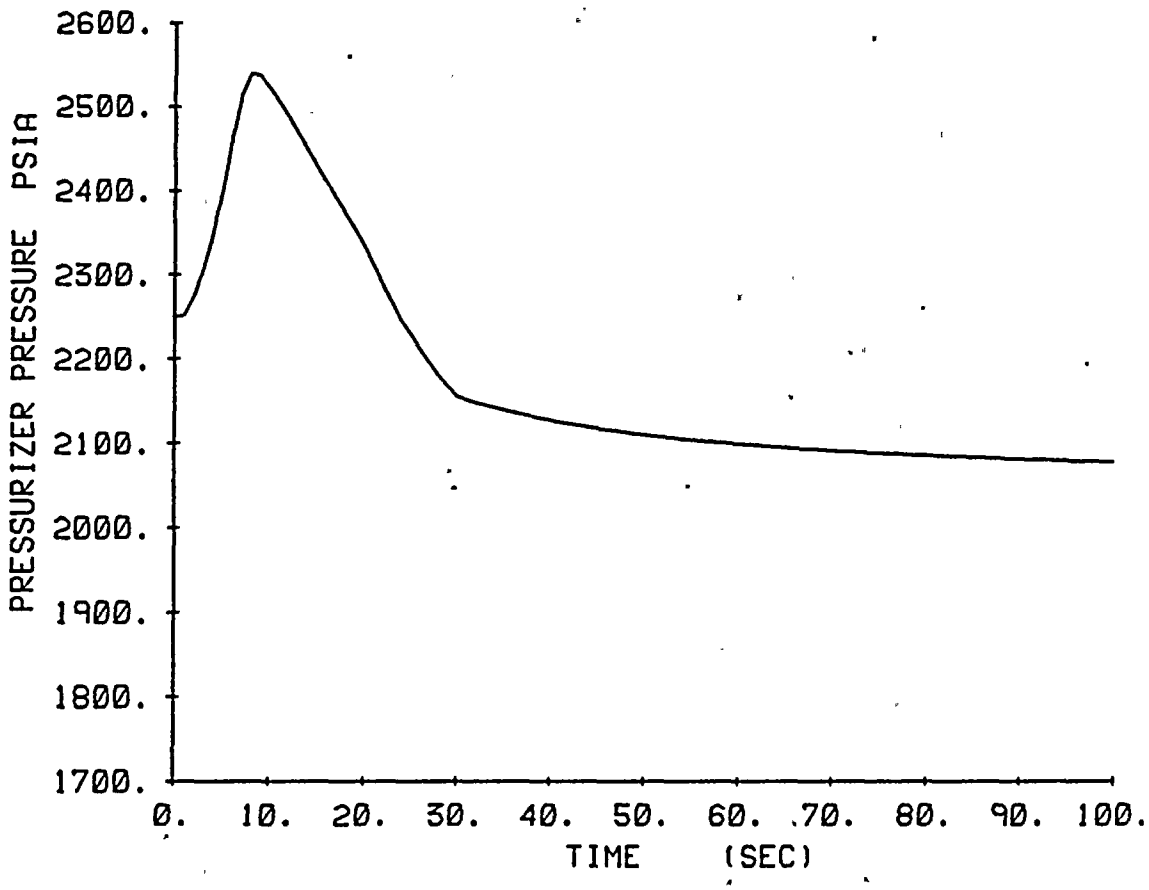


FIGURE 5.5-20
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback without Pressure Control

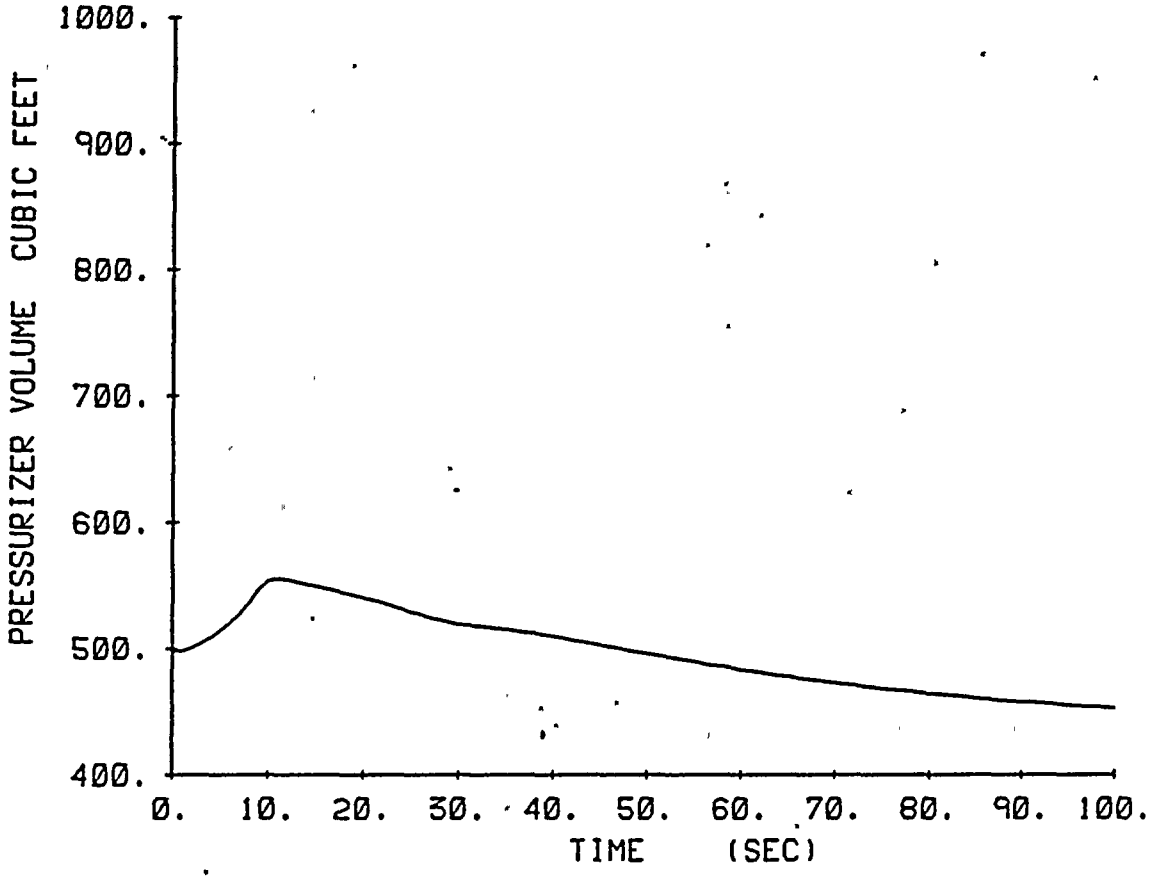


FIGURE 5.5-21

R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback without Pressure Control

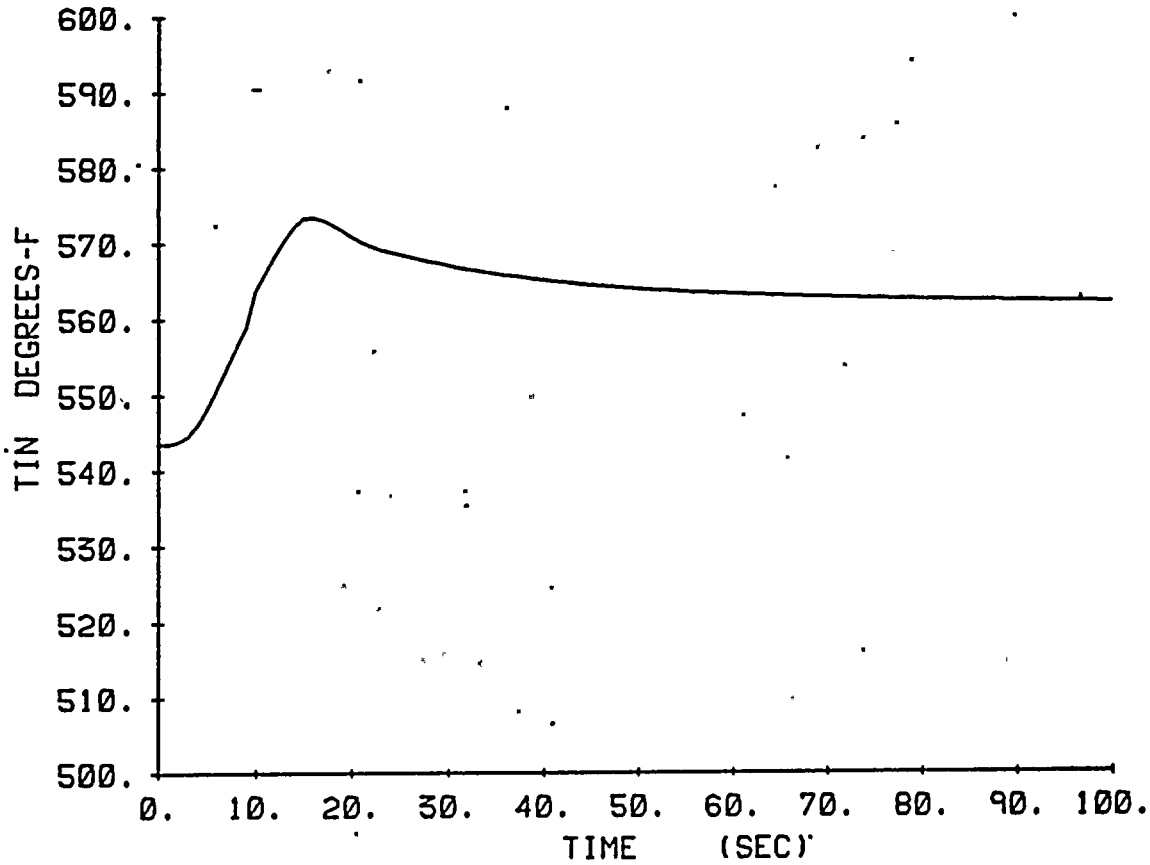


FIGURE 5.5-22
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback without Pressure Control

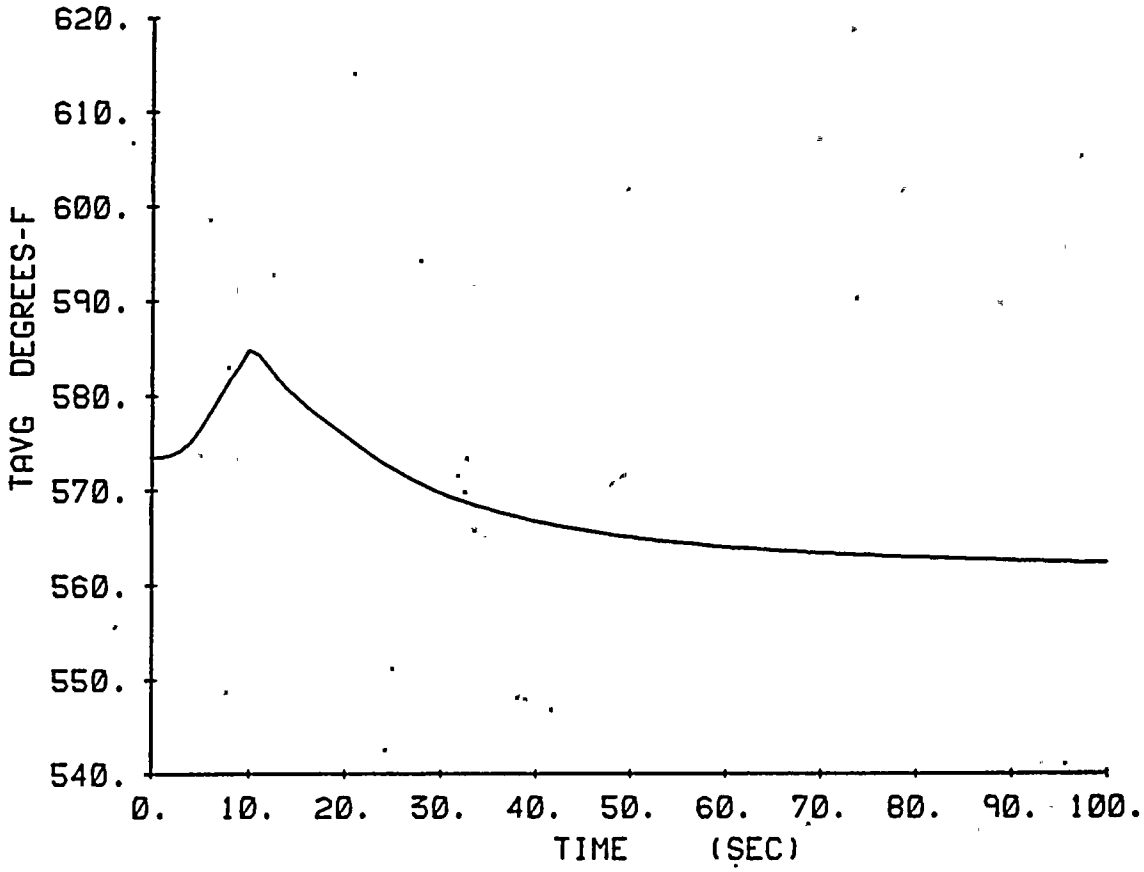


FIGURE 5.5-23
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback without Pressure Control

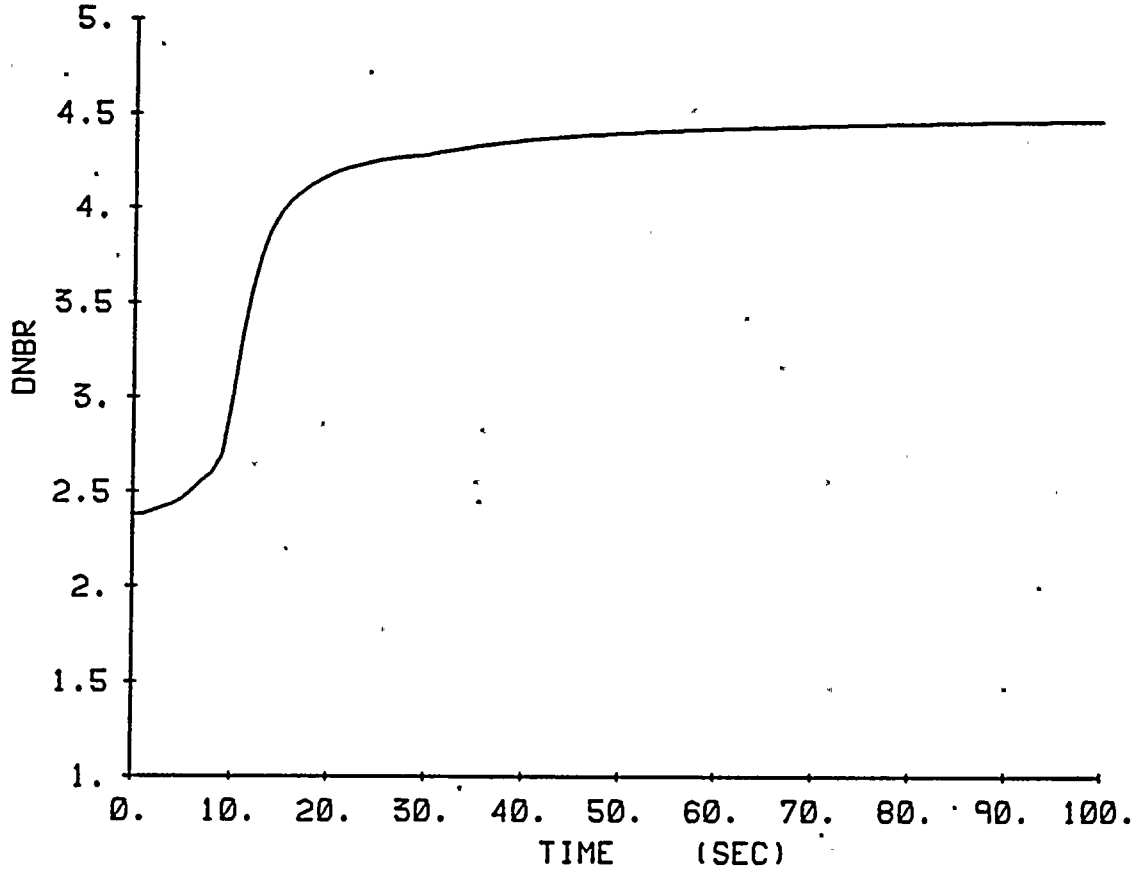


FIGURE 5.5-24
R. E. GINNA 15% SGTP
Loss of Load
Maximum Feedback without Pressure Control

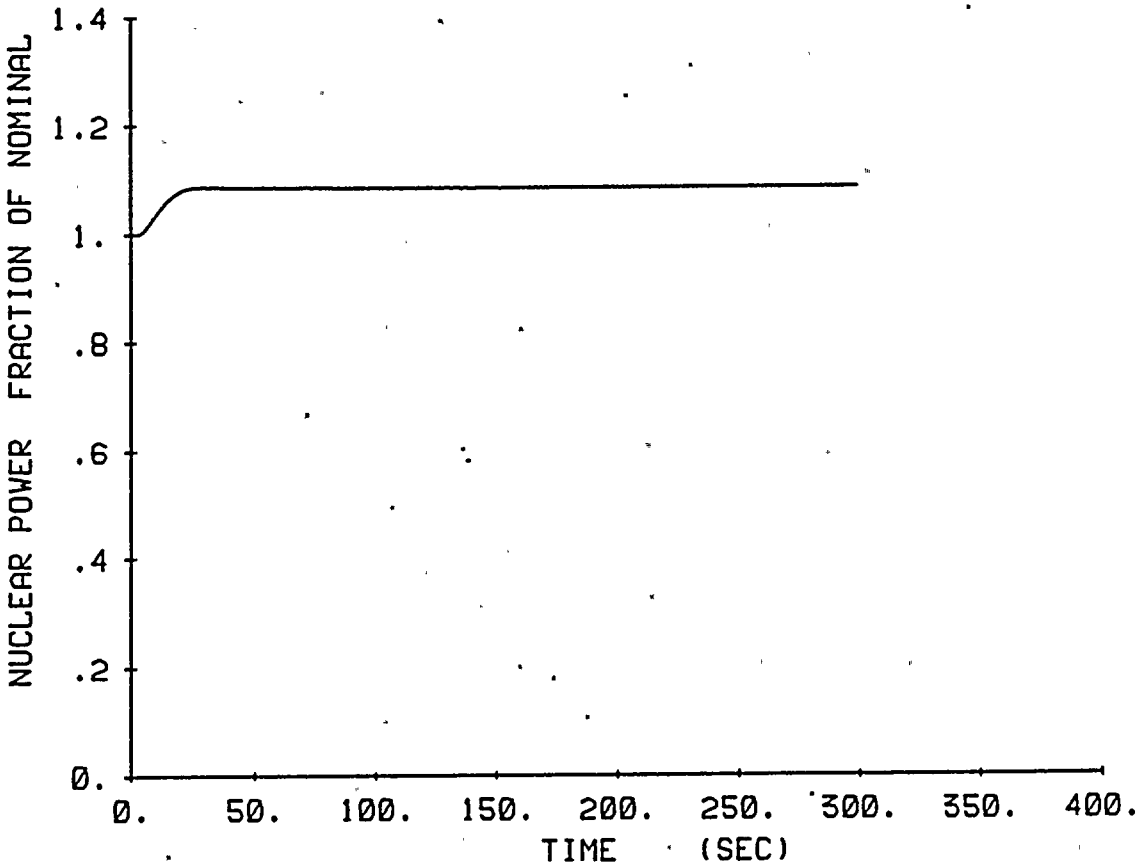


FIGURE 5.6-1
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback without Rod Control

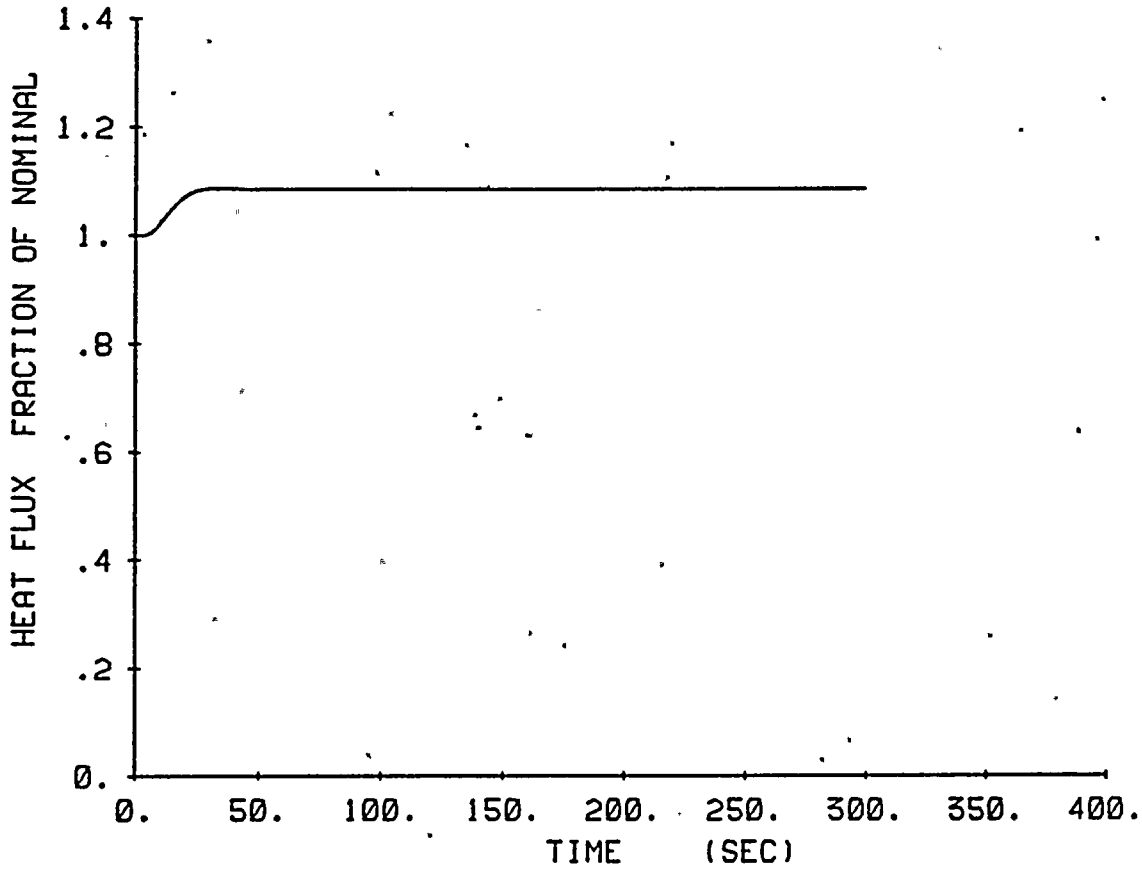


FIGURE 5.6-2
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback without Rod Control

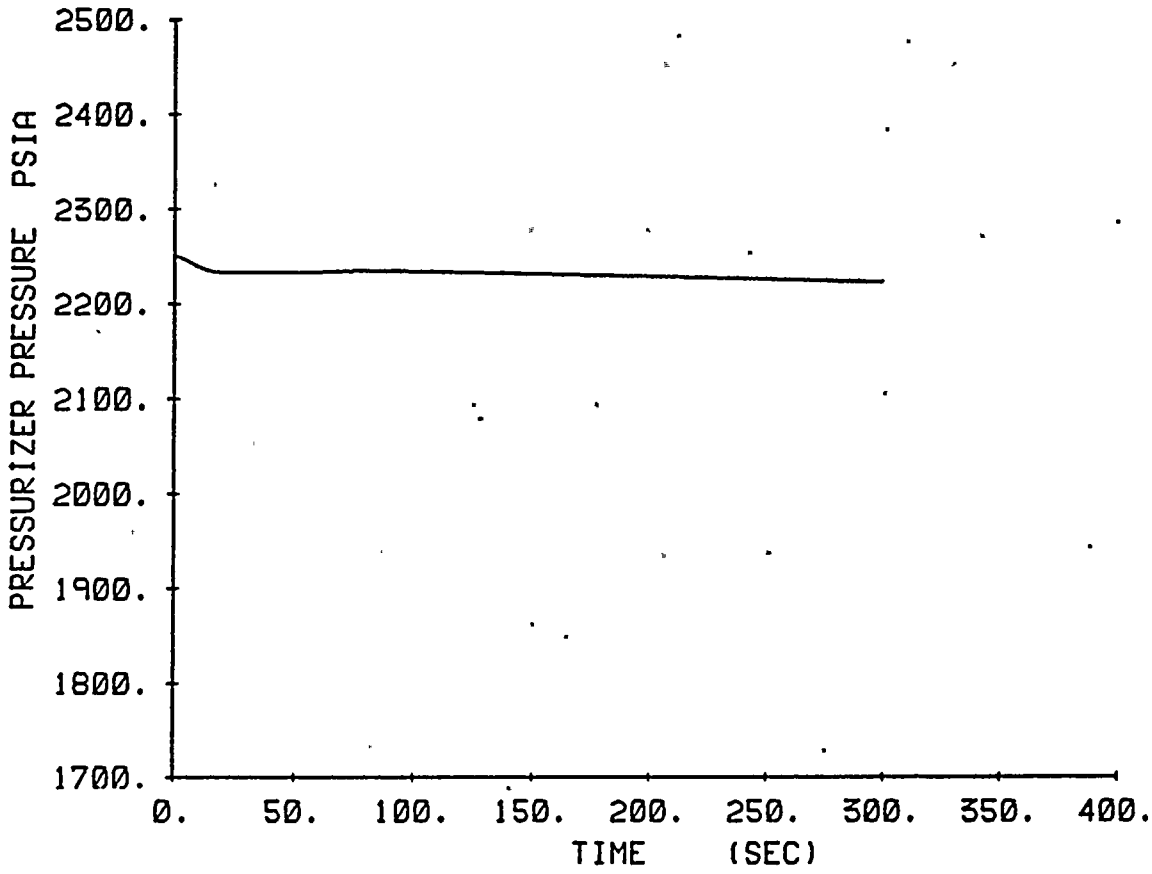


FIGURE 5.6-3
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback without Rod Control

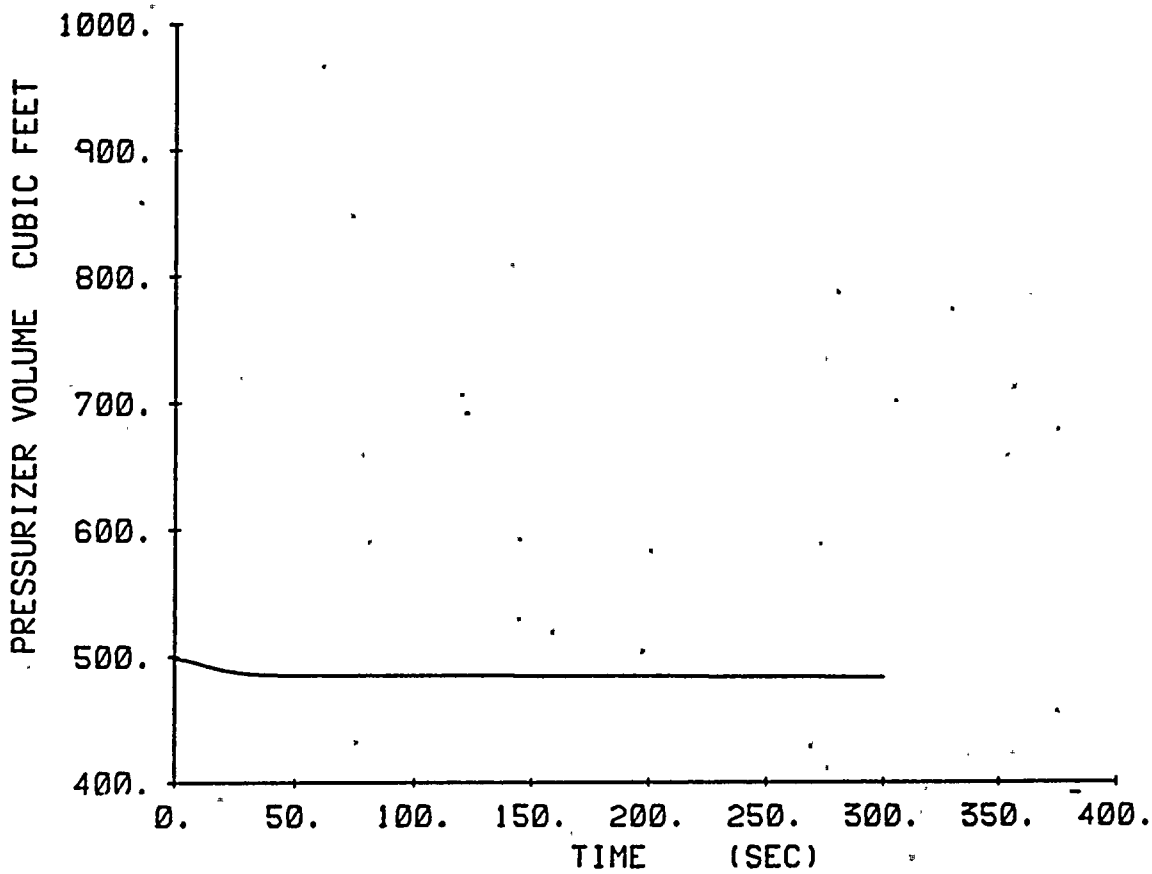
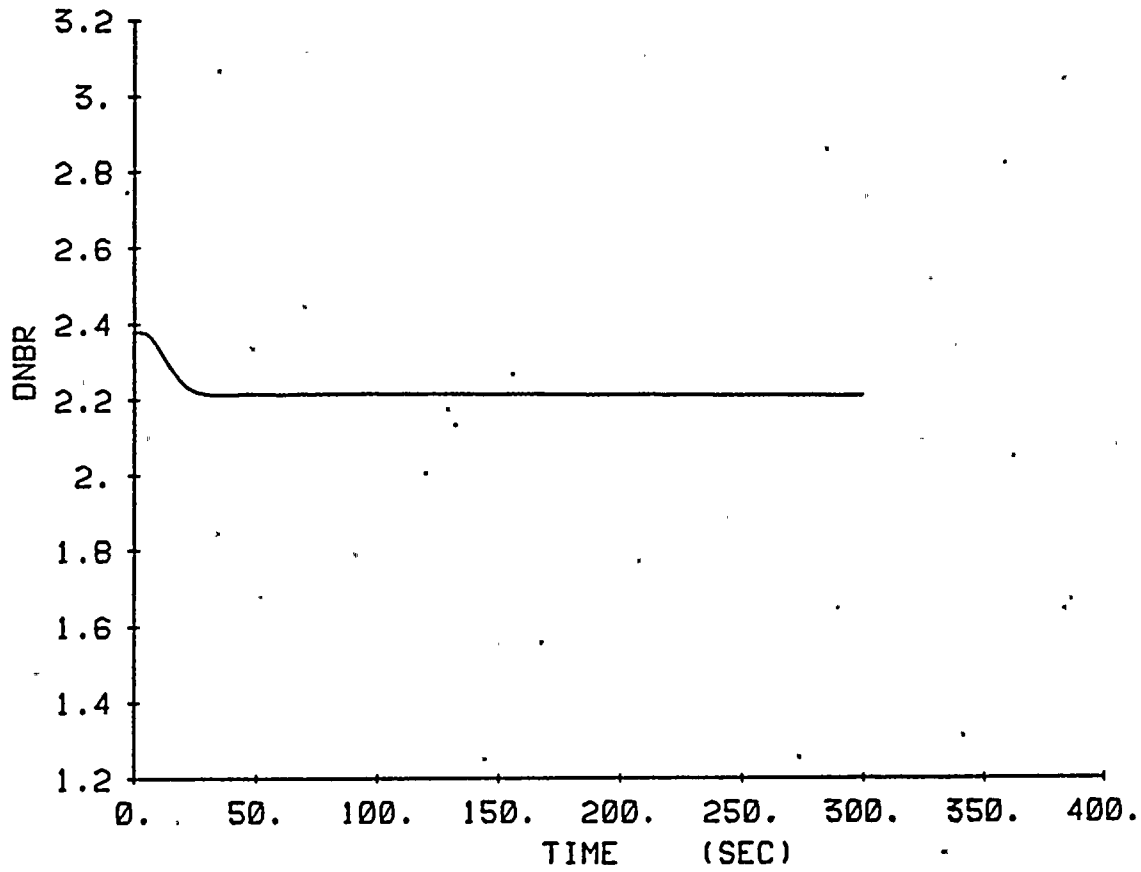
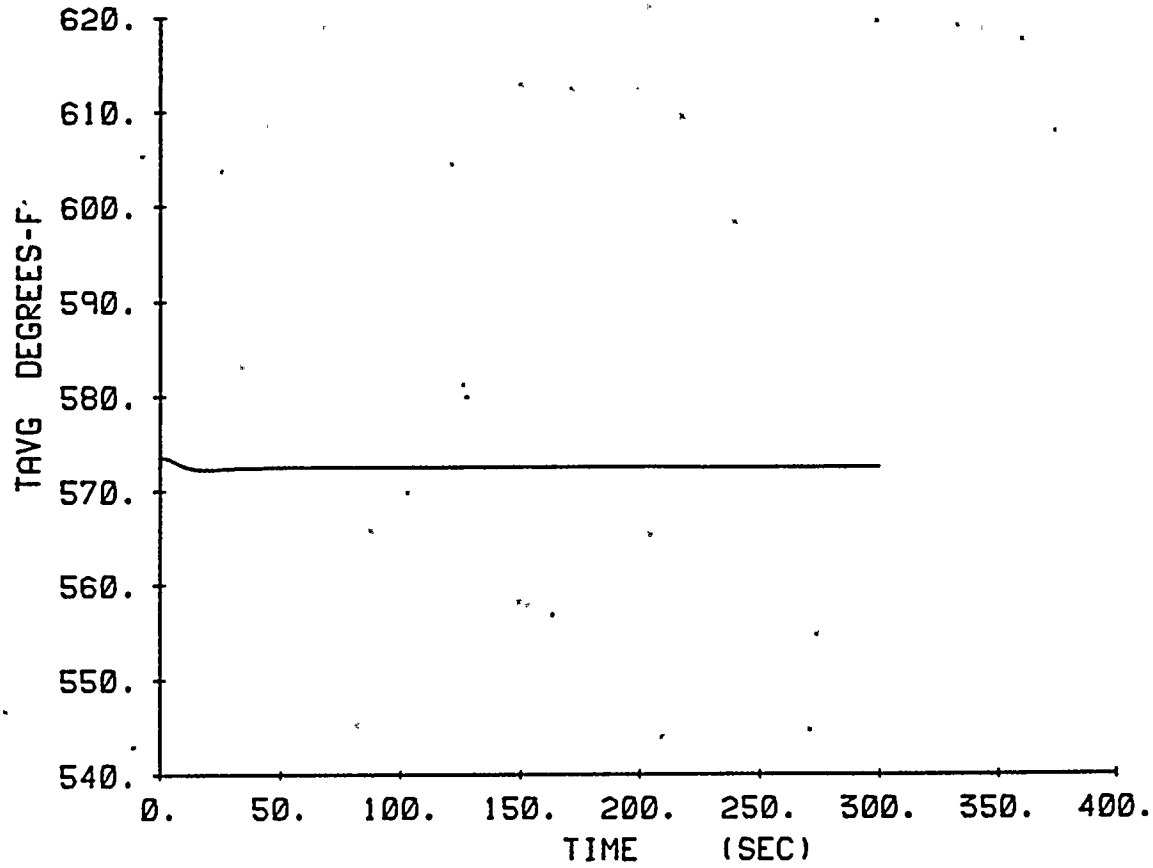


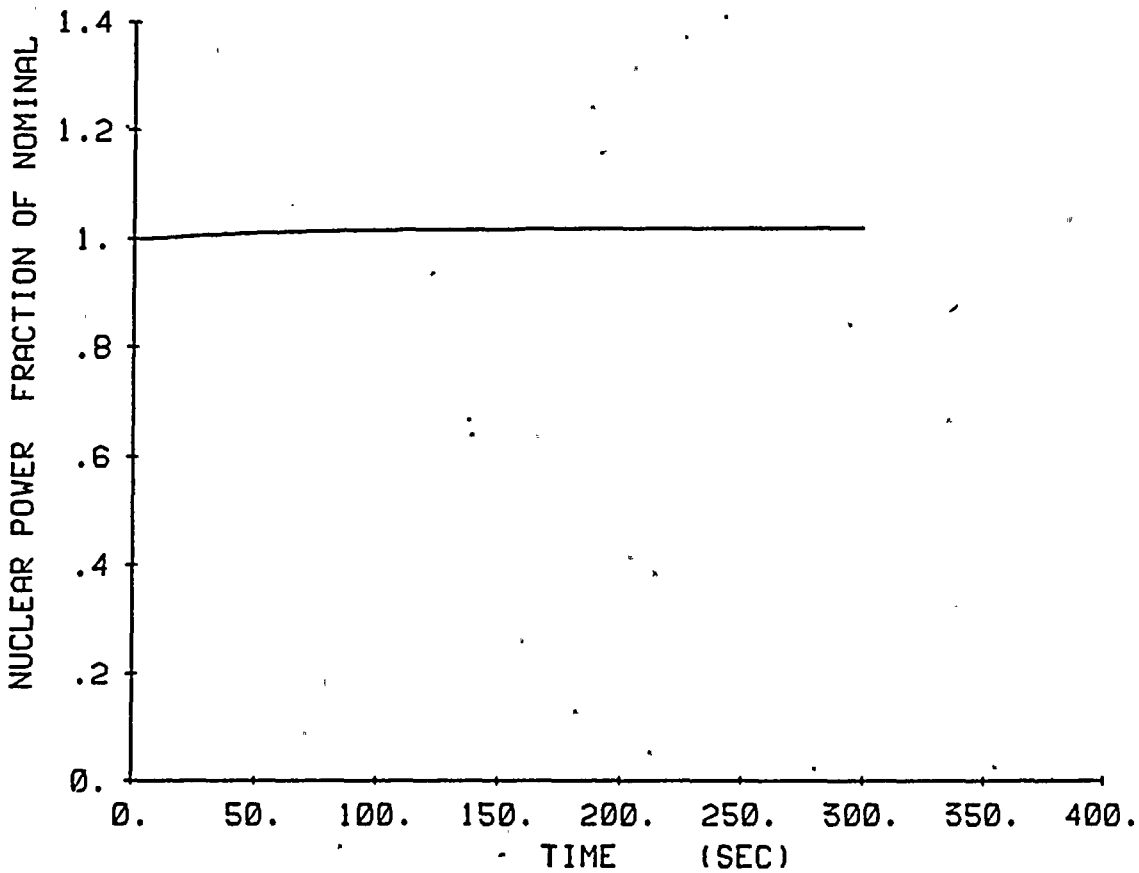
FIGURE 5.6-4
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback without Rod Control



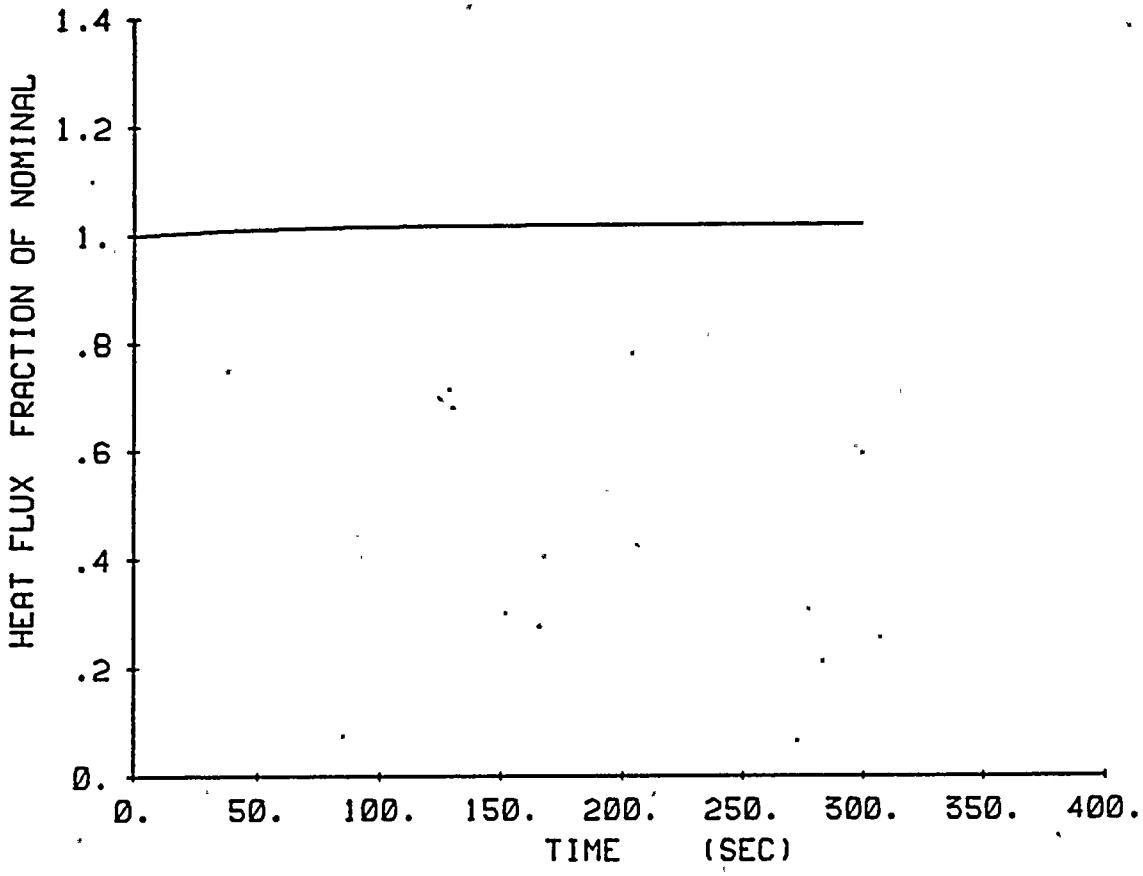
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback without Rod Control
Figure 5.6-5



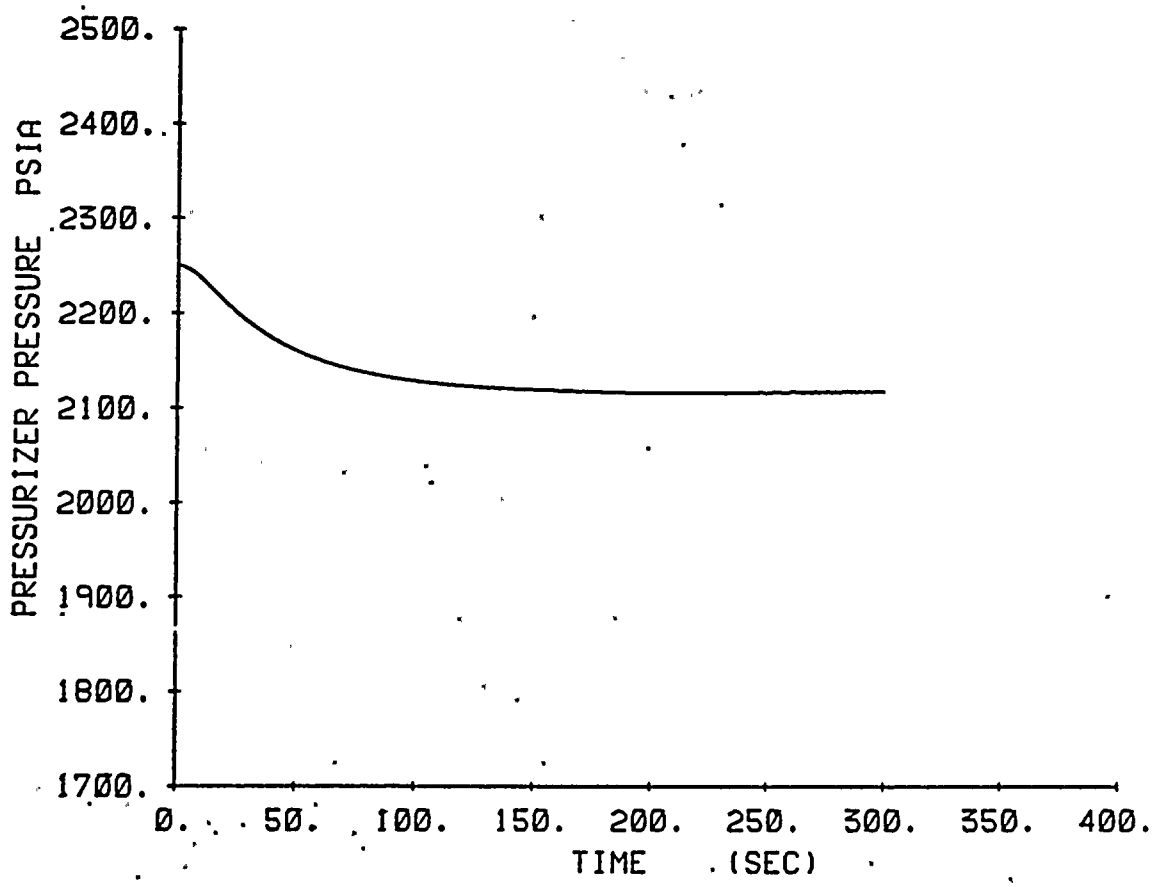
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback without Rod Control
Figure 5.6-6



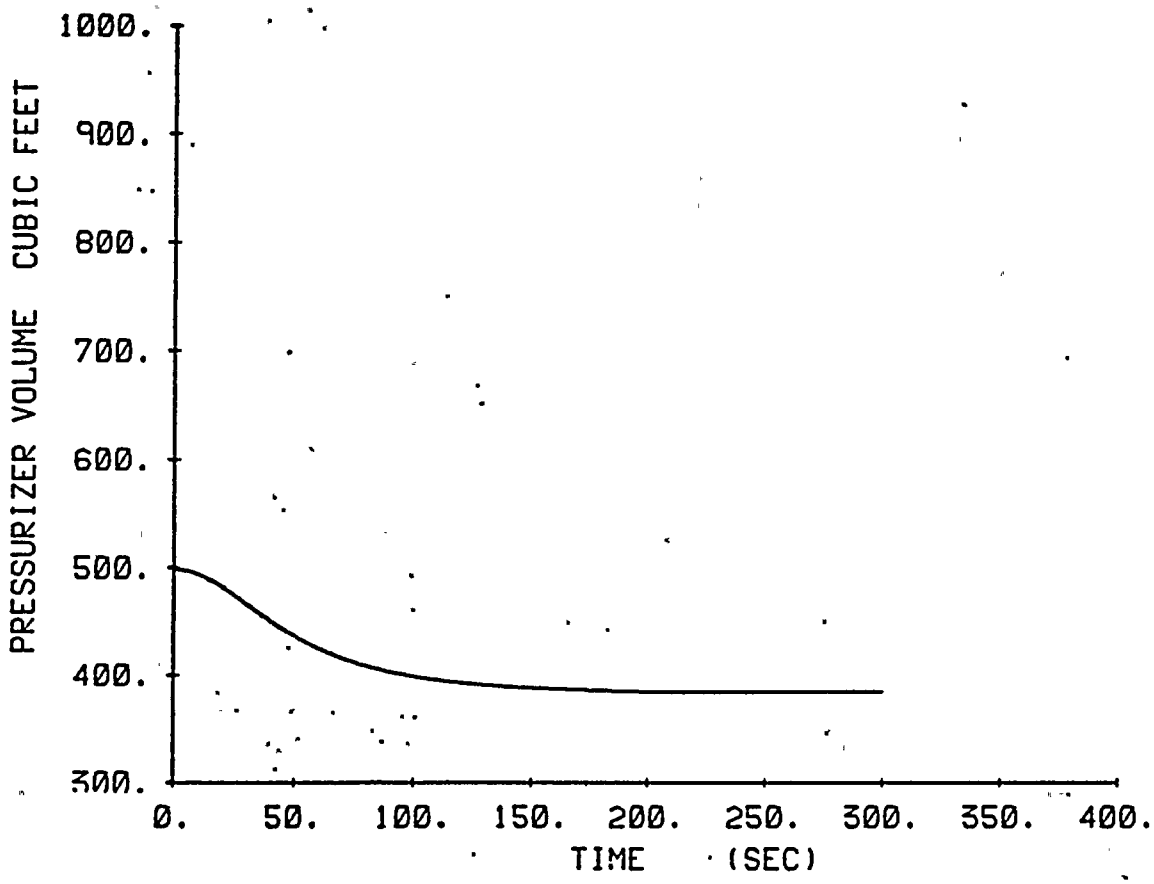
R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback without Rod Control
Figure 5.6-7



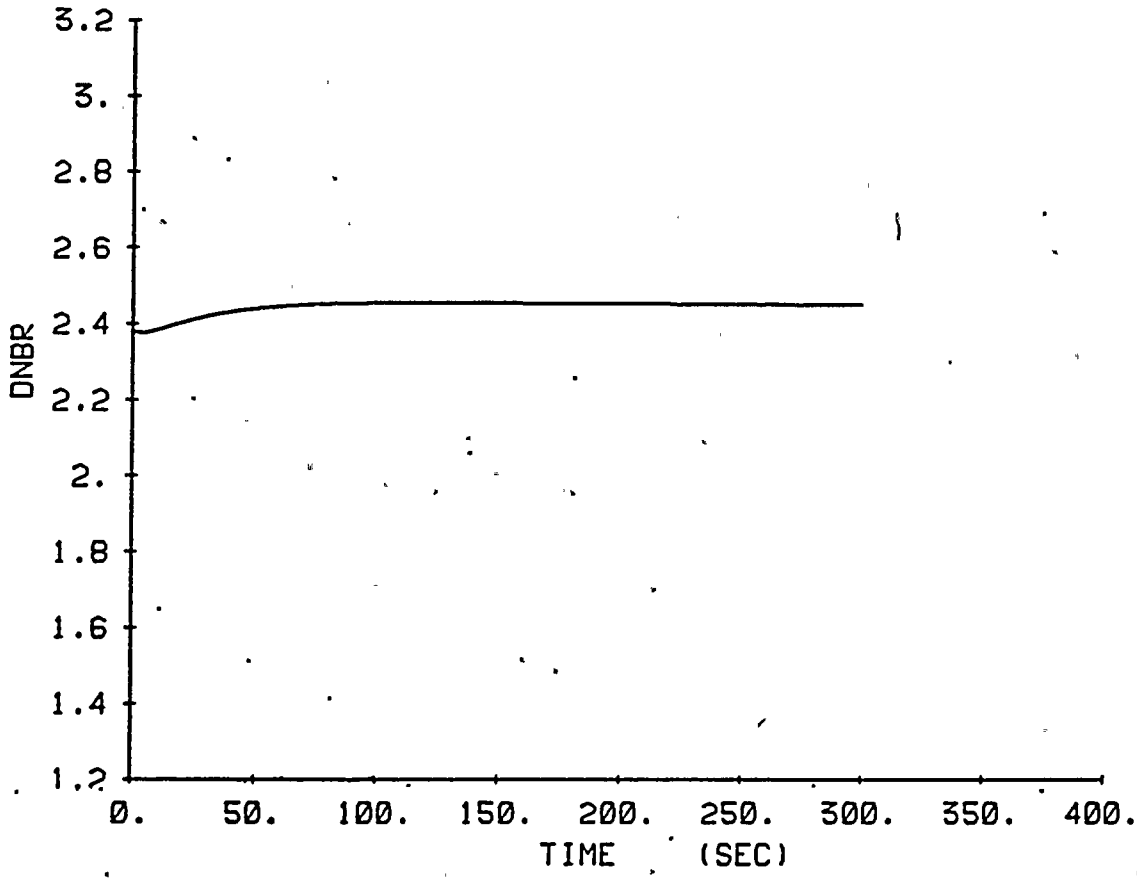
R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback without Rod Control
Figure 5.6-8



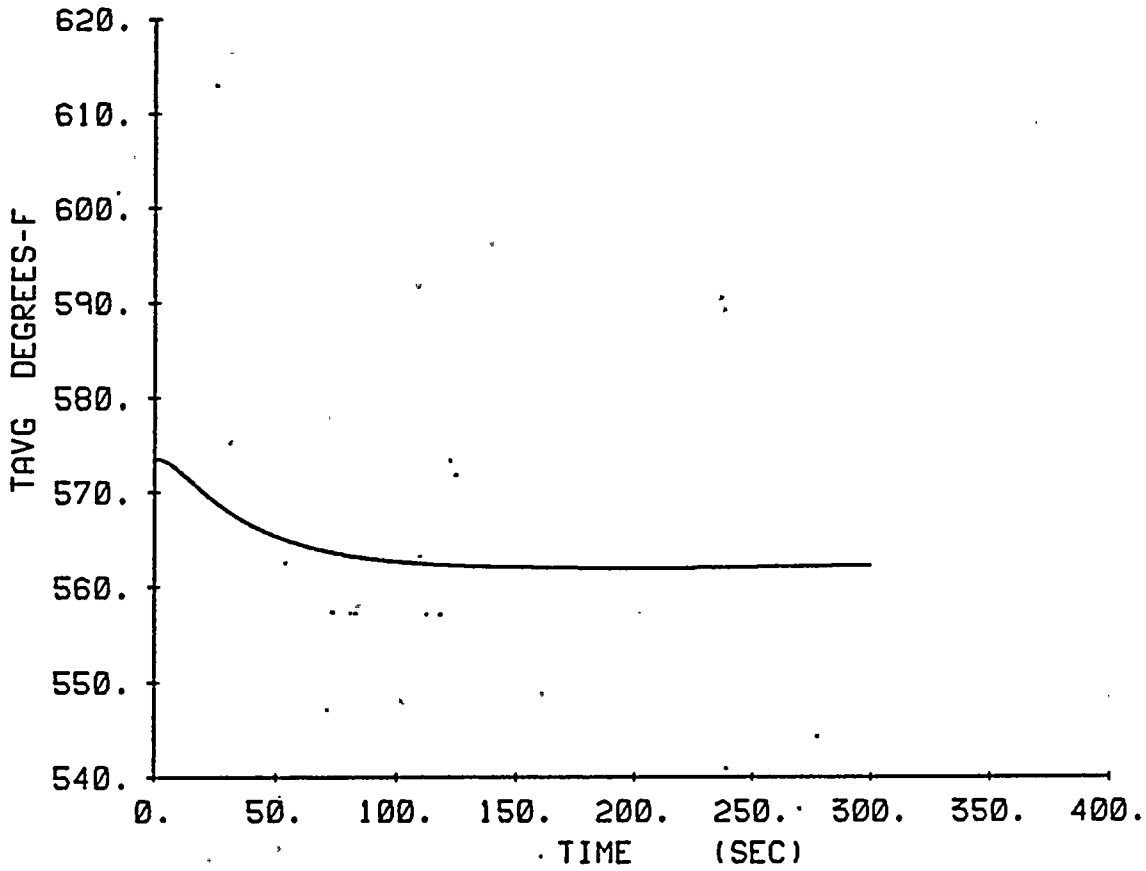
R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback without Rod Control
Figure 5.6-9



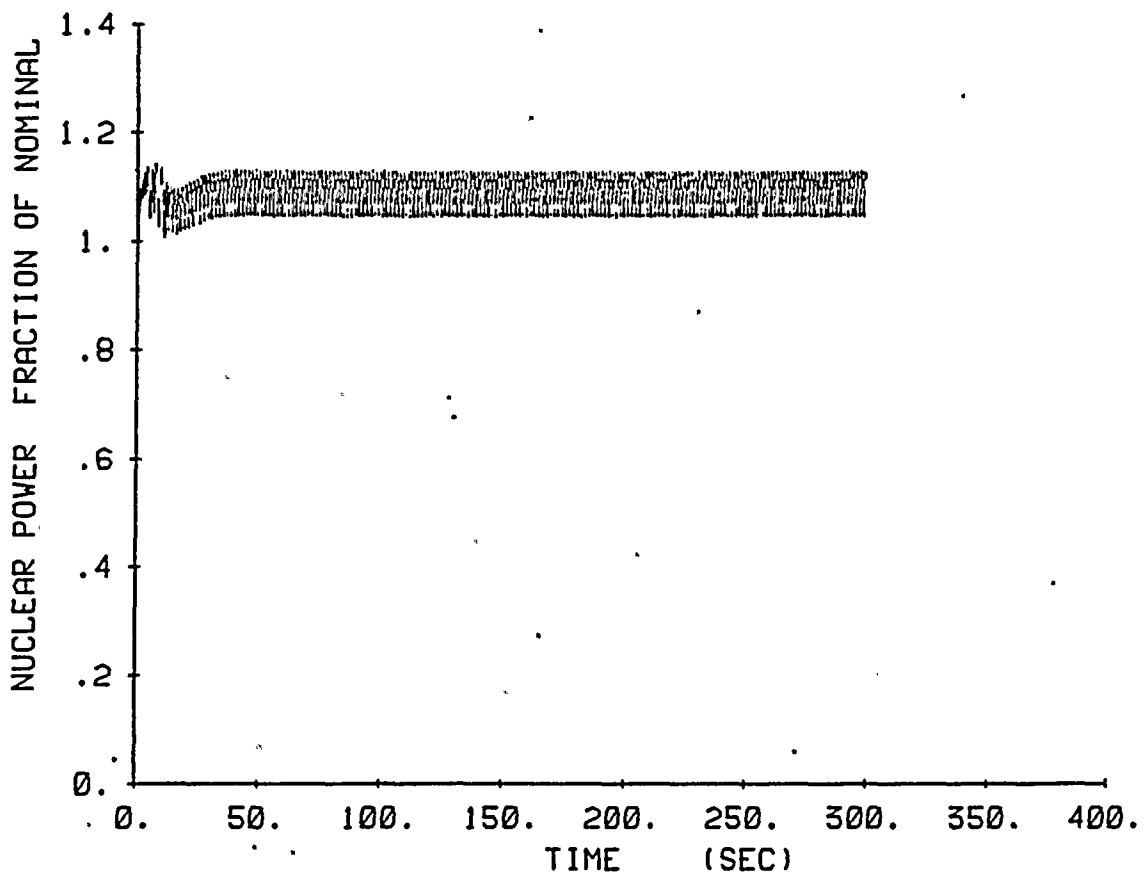
R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback without Rod Control
Figure 5.6-10



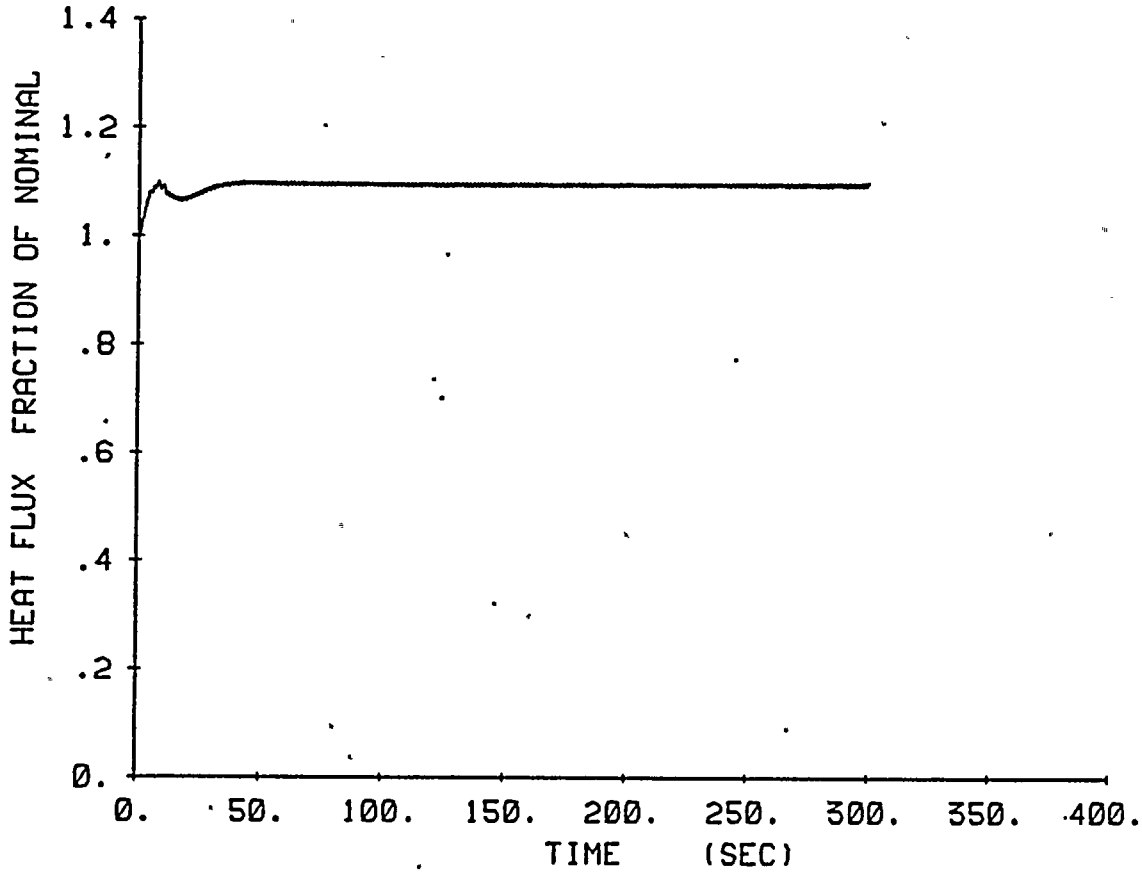
R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback without Rod Control
Figure 5.6-11



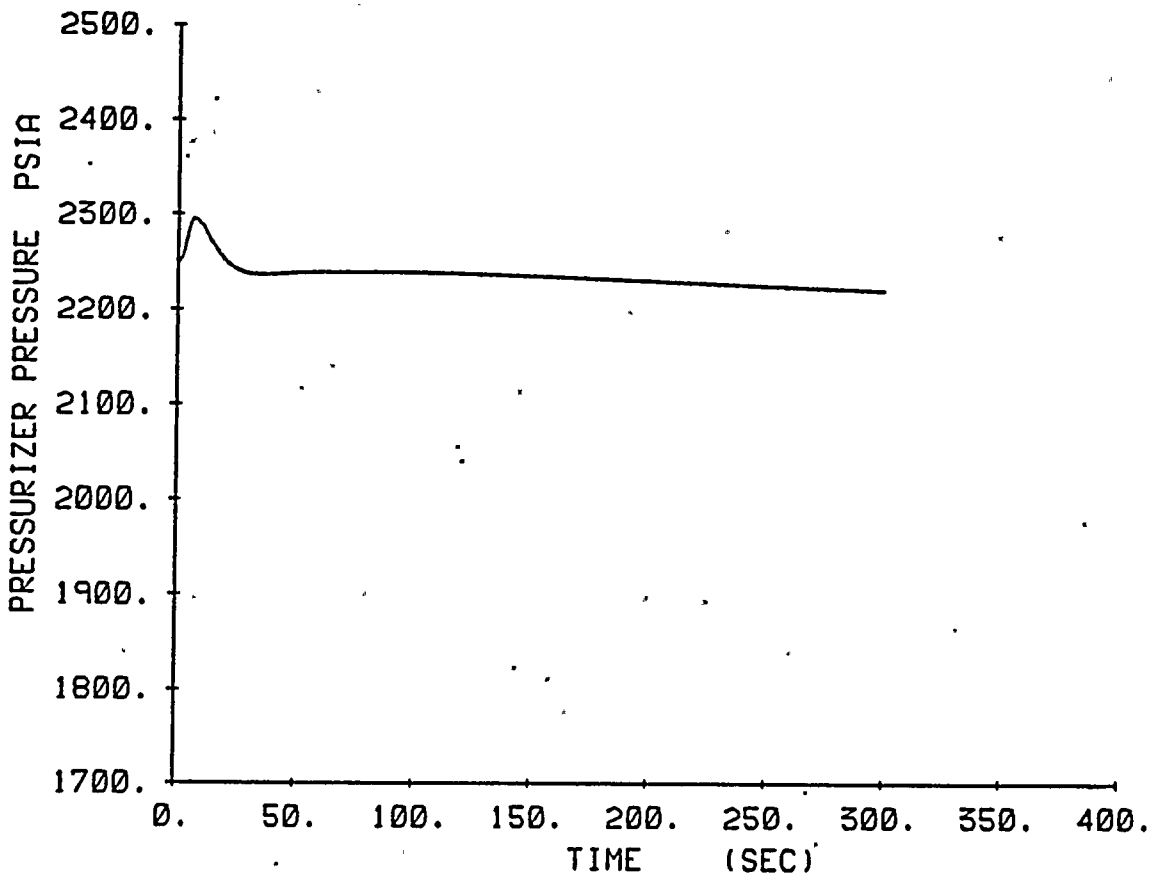
R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback without Rod Control
Figure 5.6-12



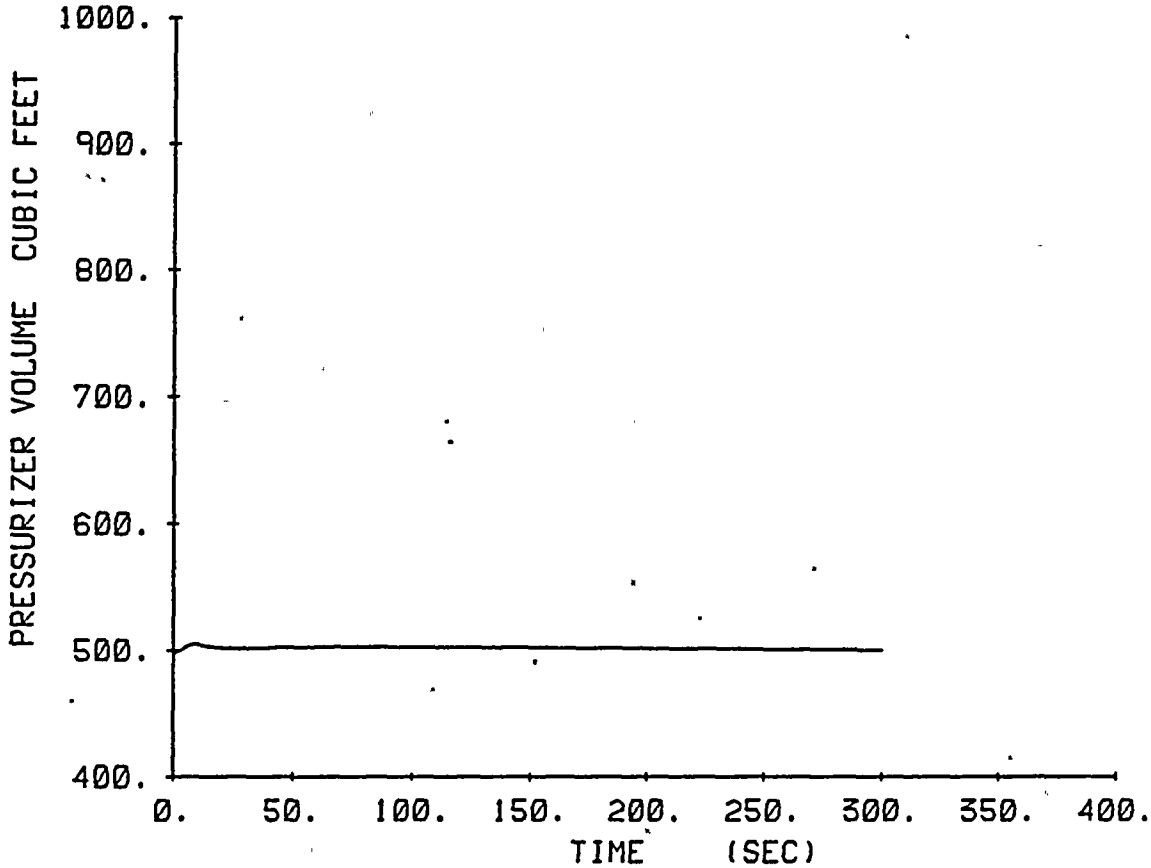
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback with Automatic Rod Control
Figure 5.6-13



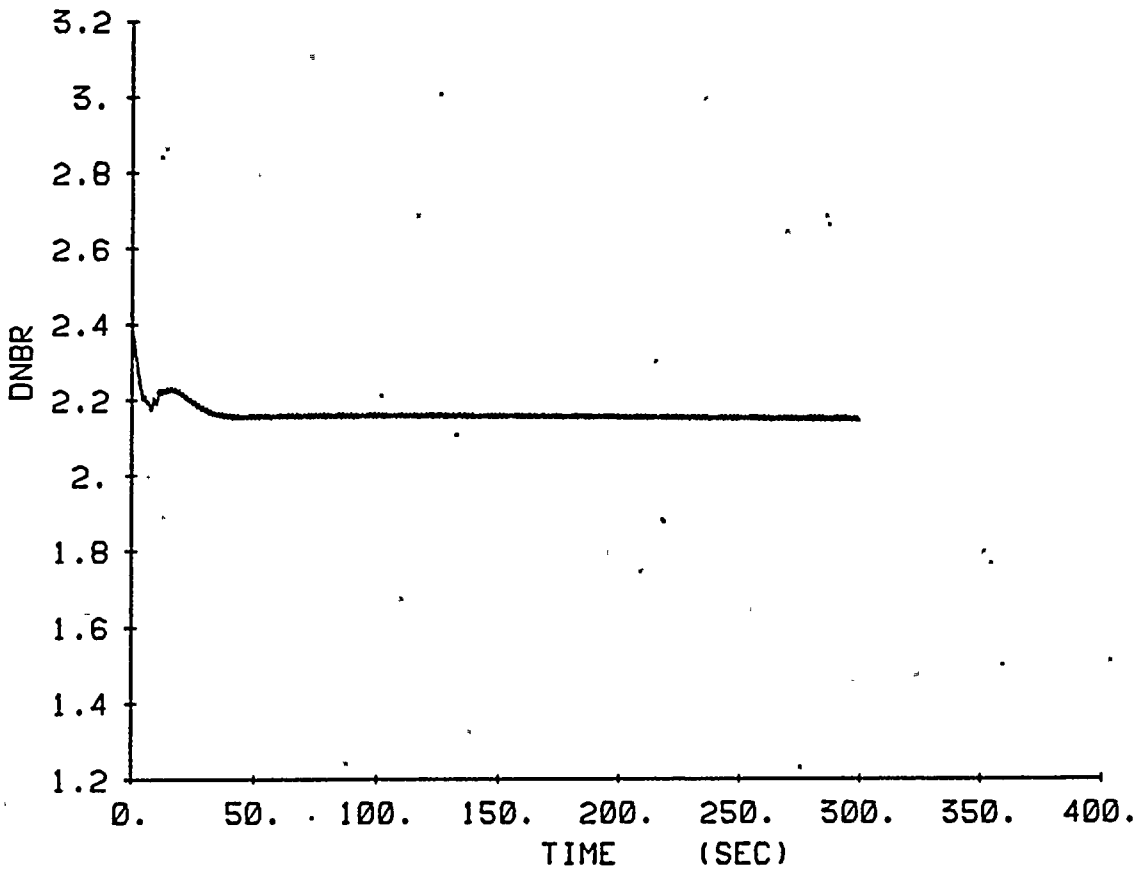
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback with Automatic Rod Control
Figure 5.6-14



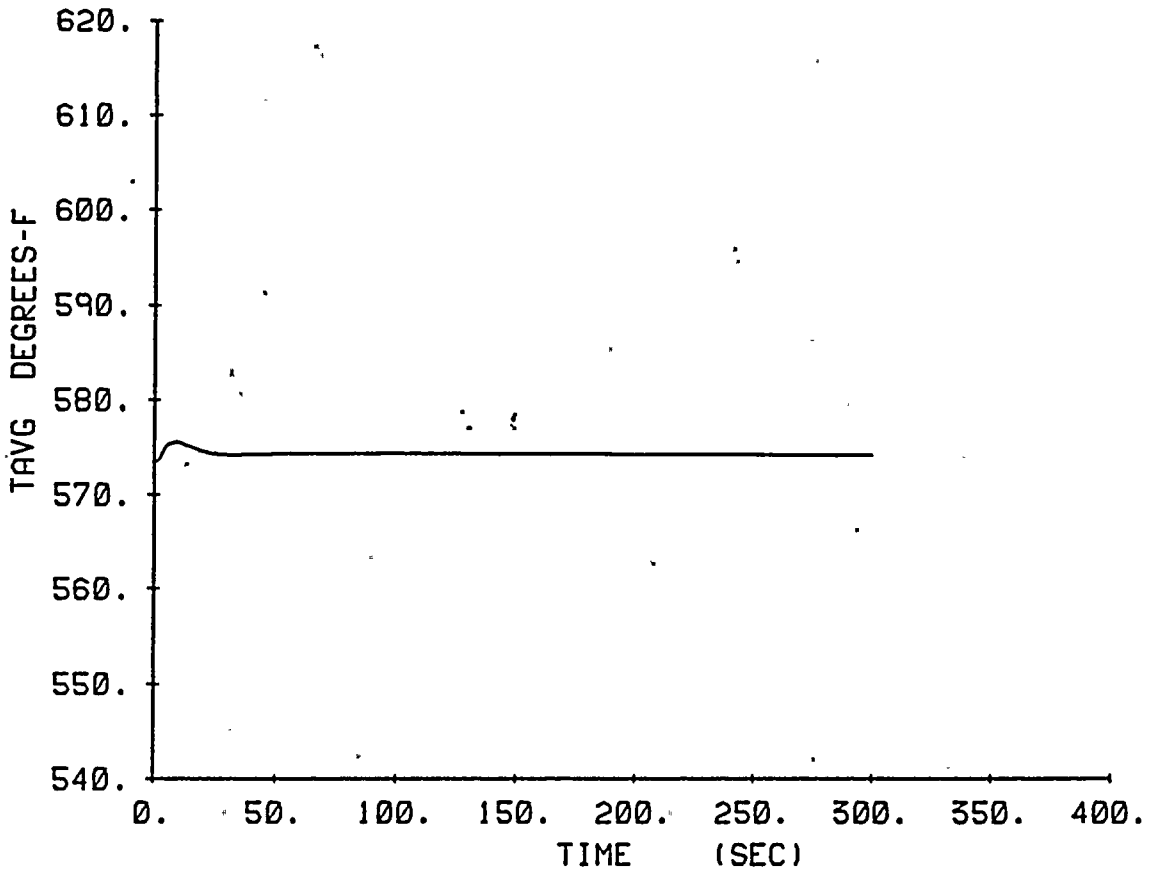
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback with Automatic Rod Control
Figure 5.6-15



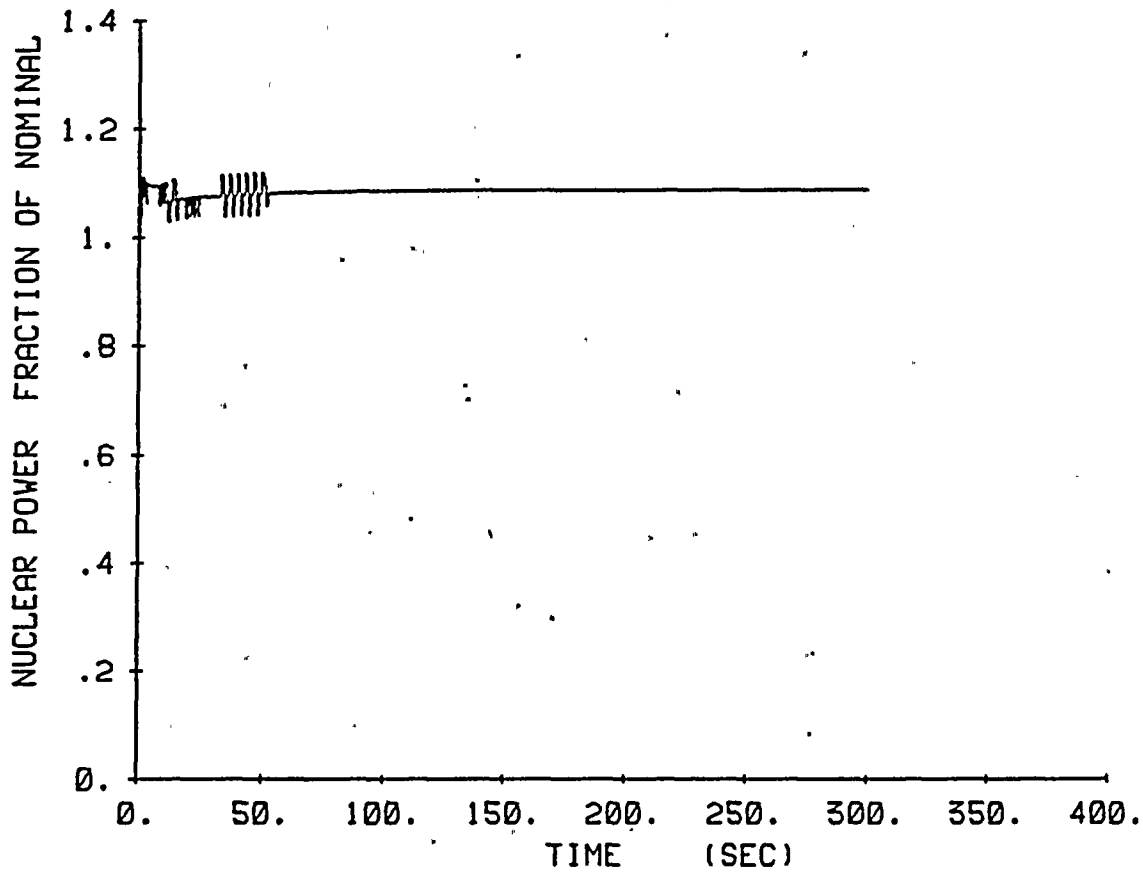
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback with Automatic Rod Control
Figure 5.6-16



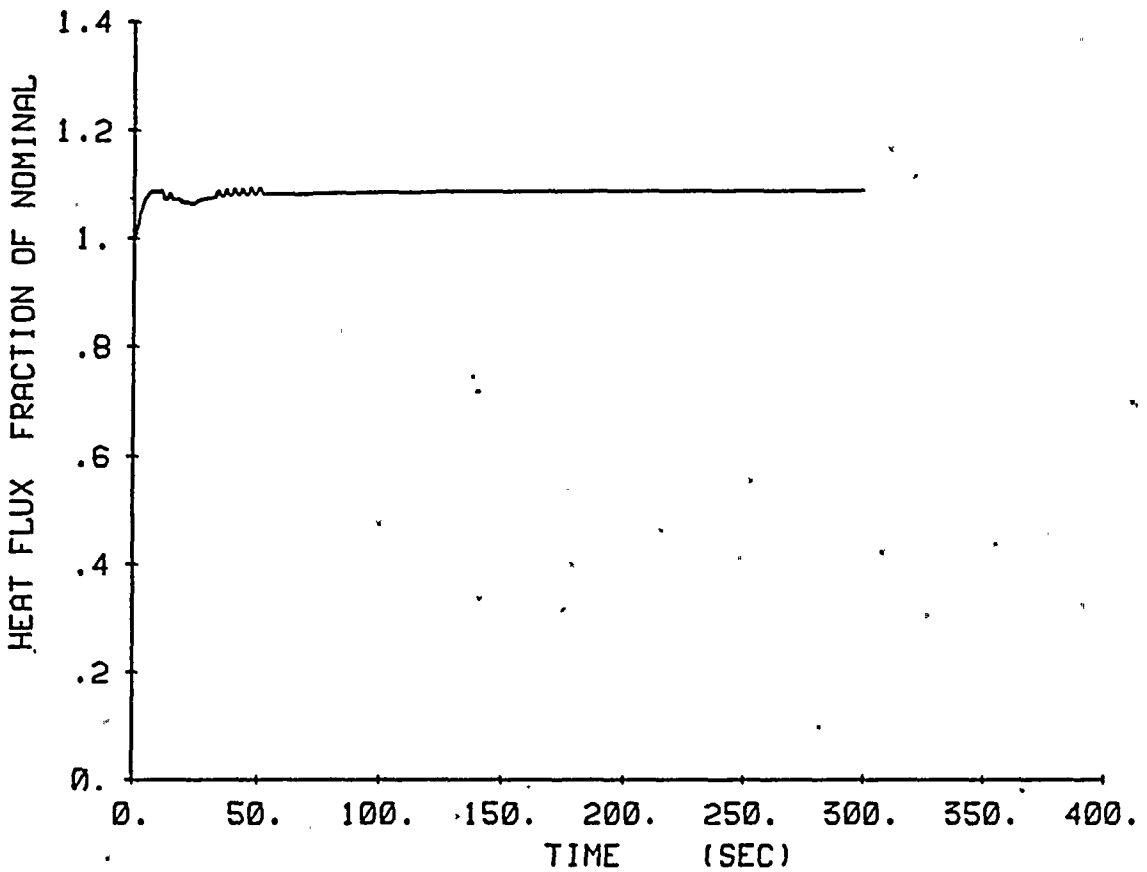
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback with Automatic Rod Control
Figure 5.6-17



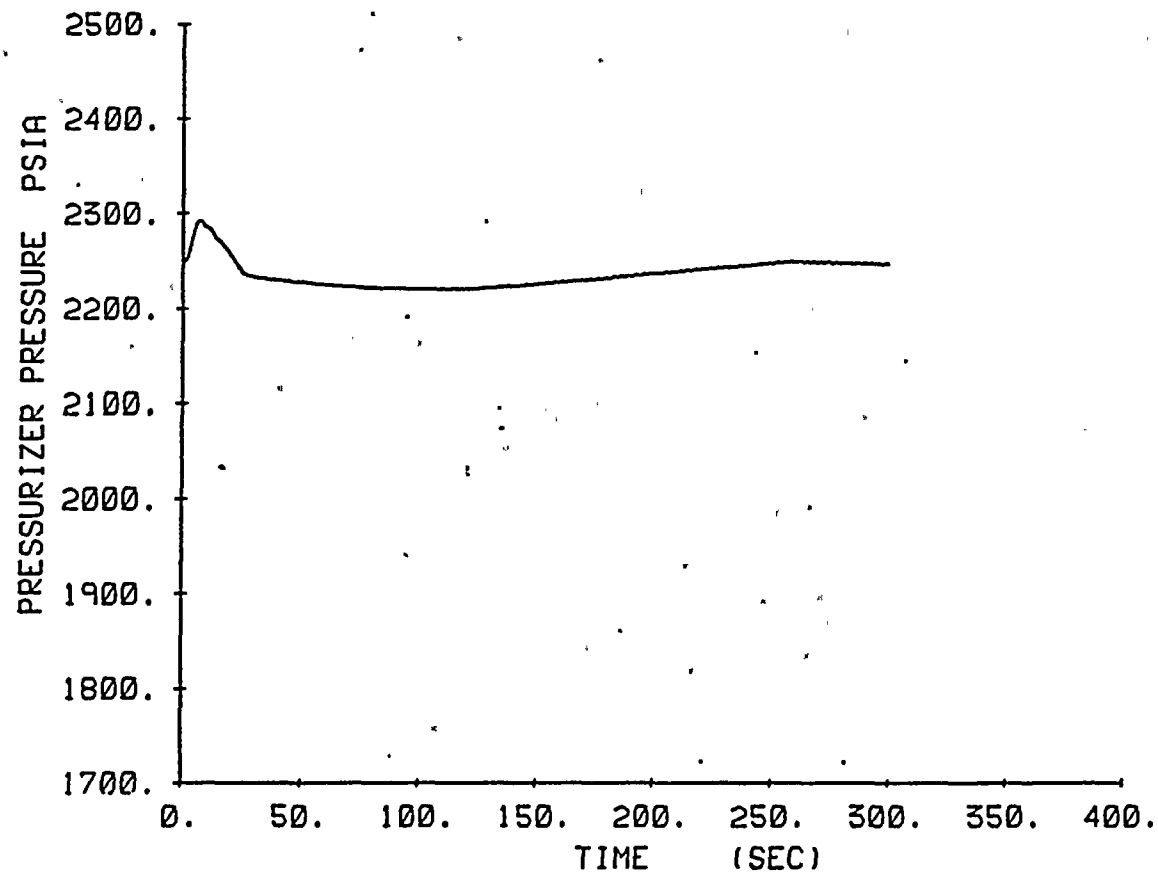
R. E. Ginna 15% SGTP
Excess Load Increase
Maximum Feedback with Automatic Rod Control
Figure 5.6-18



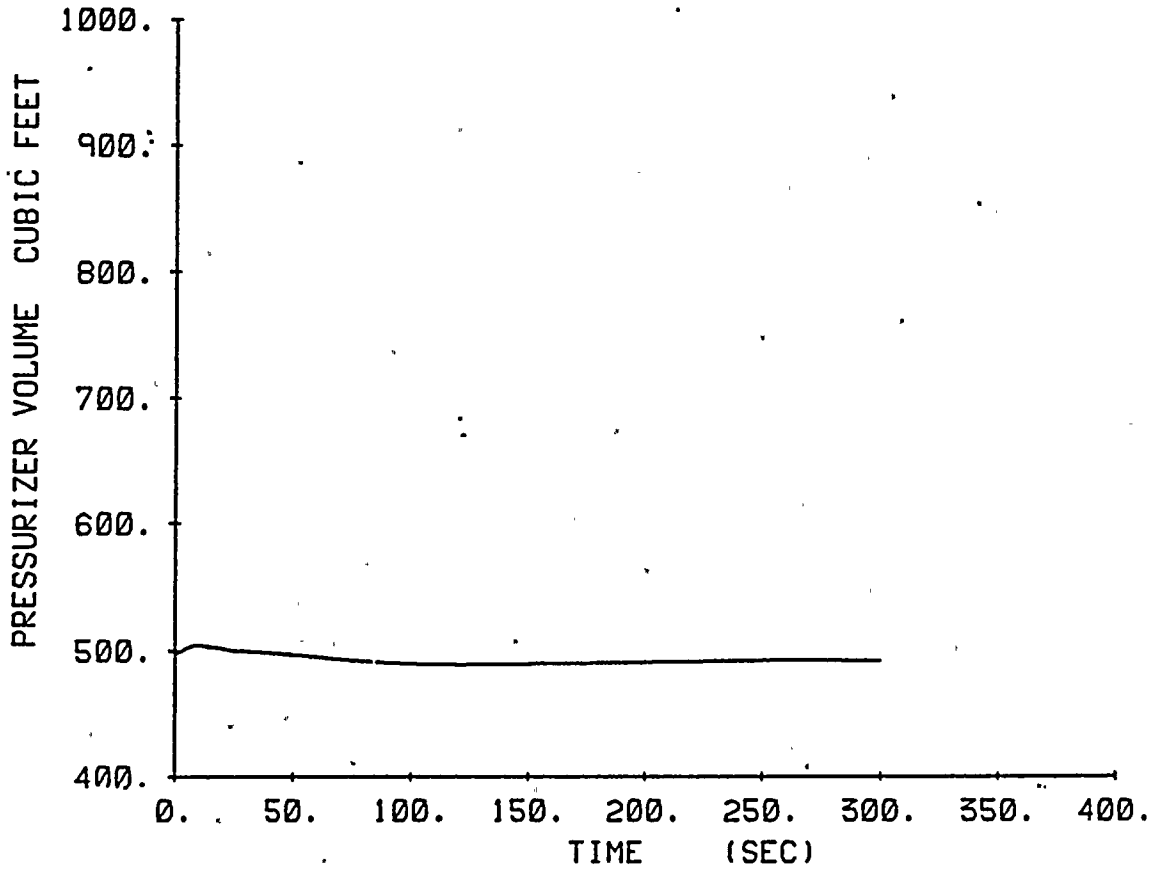
R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback with Automatic Rod Control
Figure 5.6-19



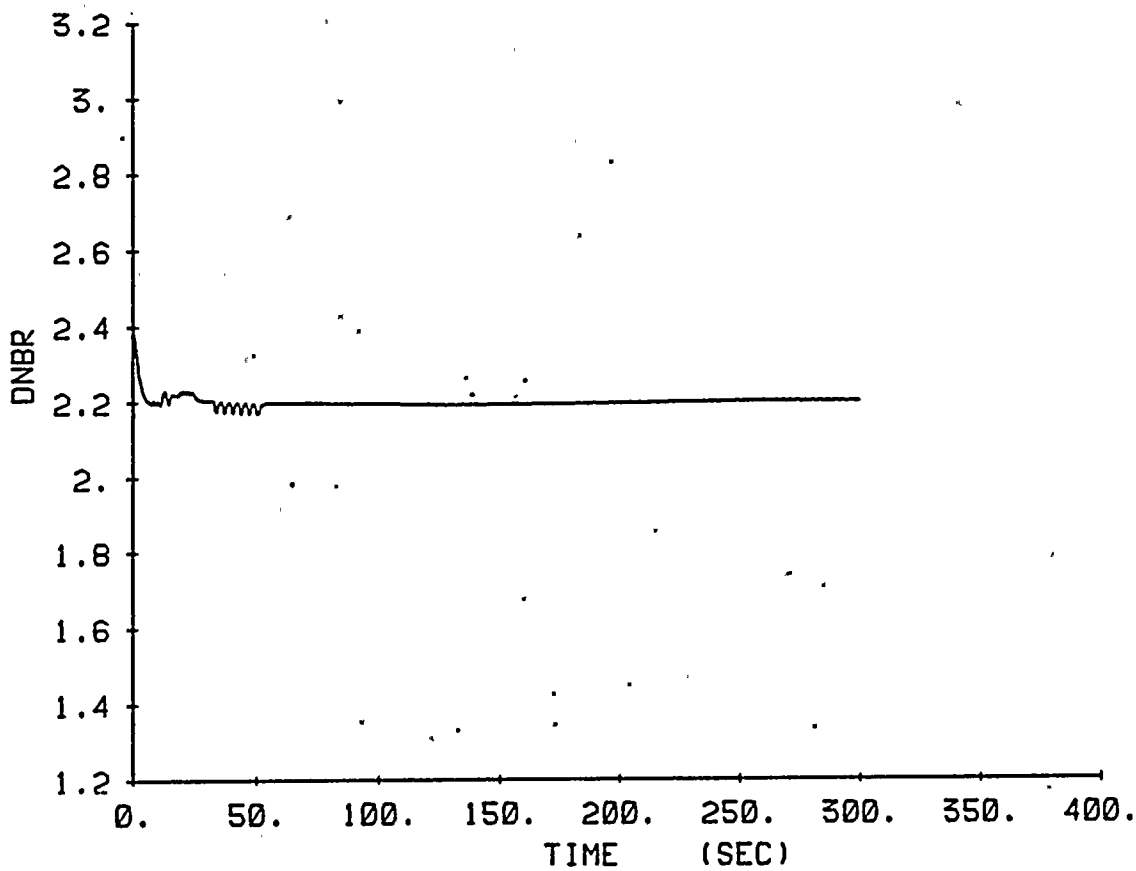
R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback with Automatic Rod Control
Figure 5.6-20



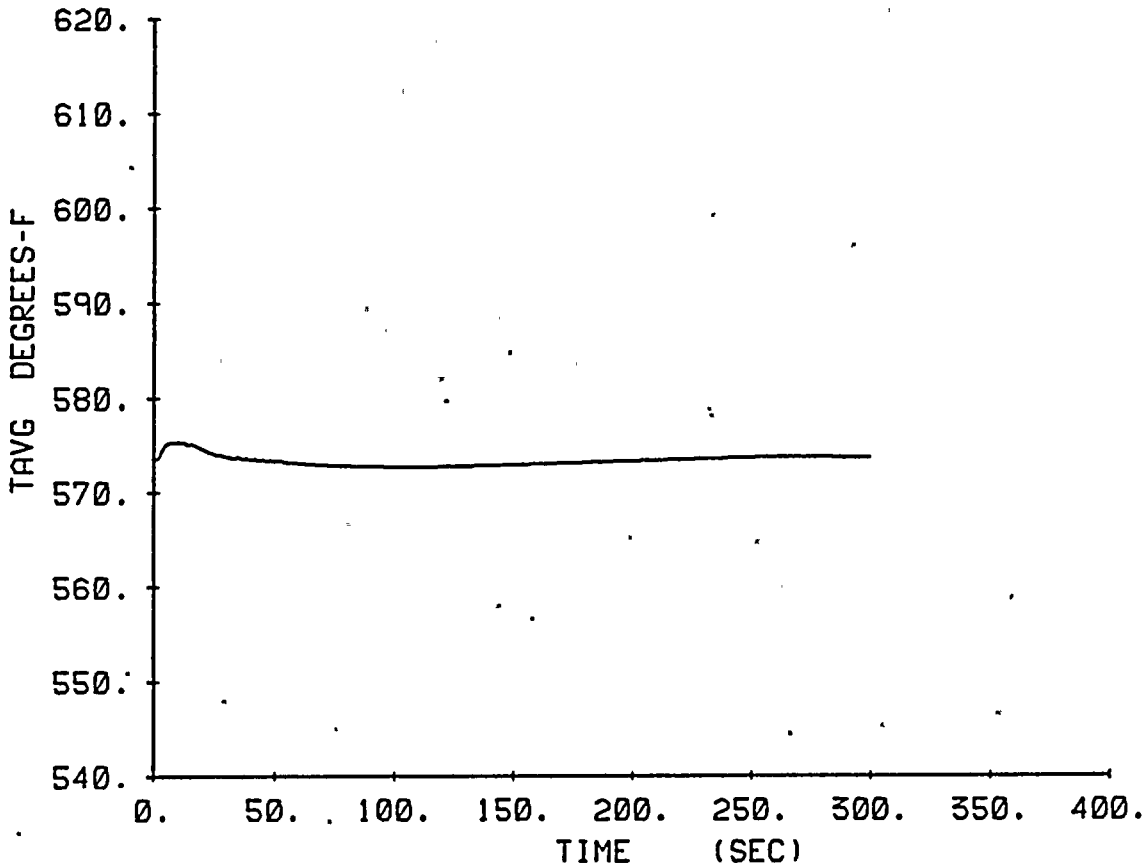
R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback with Automatic Rod Control
Figure 5.6-21



R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback with Automatic Rod Control
Figure 5.6-22



R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback with Automatic Rod Control
Figure 5.6-23



R. E. Ginna 15% SGTP
Excess Load Increase
Minimum Feedback with Automatic Rod Control
Figure 5.6-24

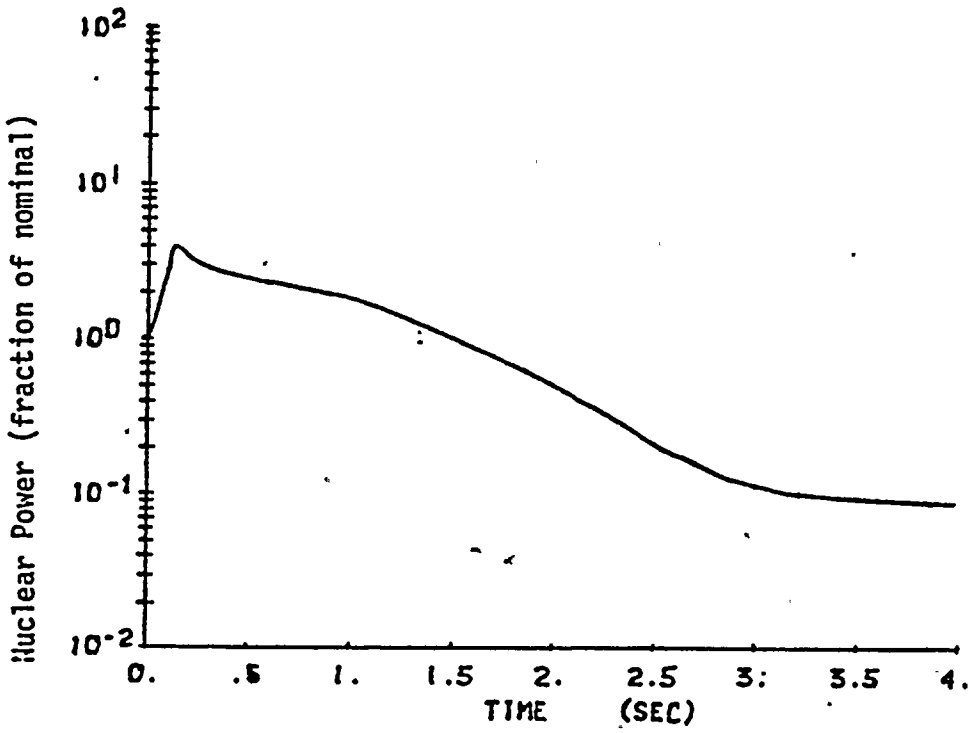


Figure 5.7-1
R. E. Ginna 15% SGTP
RCCA Ejection
Beginning of Life, Full Power

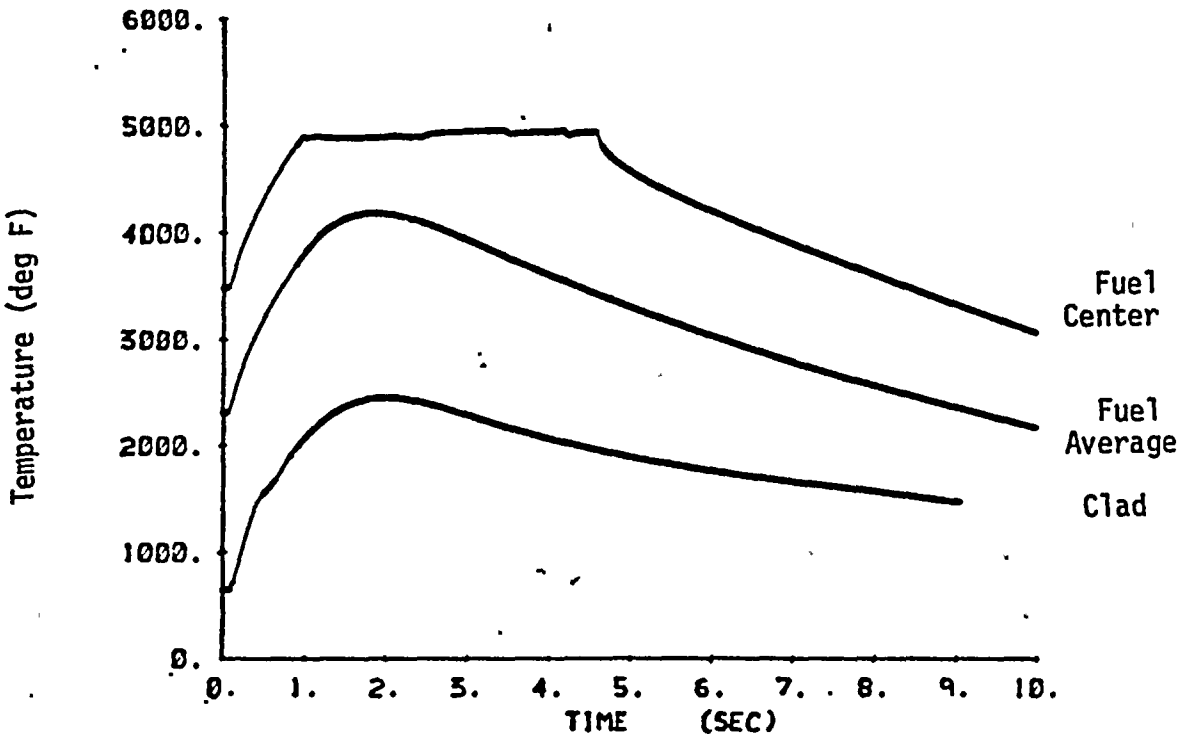


Figure 5.7-2
R. E. Ginna 15% SGTP
RCCA Ejection
Beginning of Life, Full Power

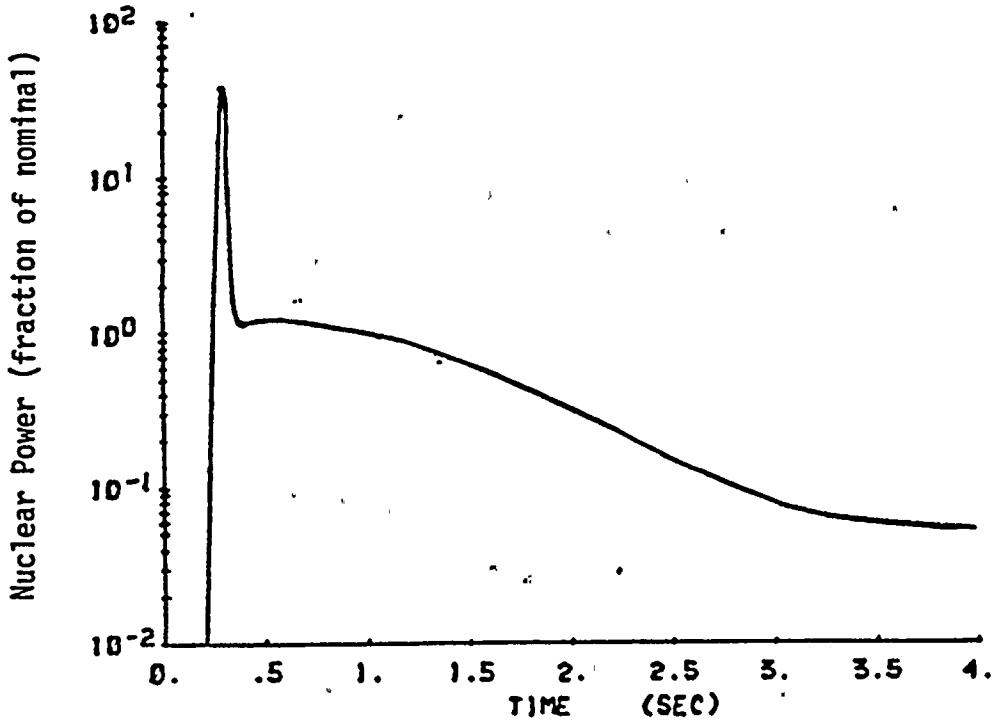


Figure 5.7-3
R. E. Ginna 15% SGTP
RCCA Ejection
Beginning of Life, Zero Power

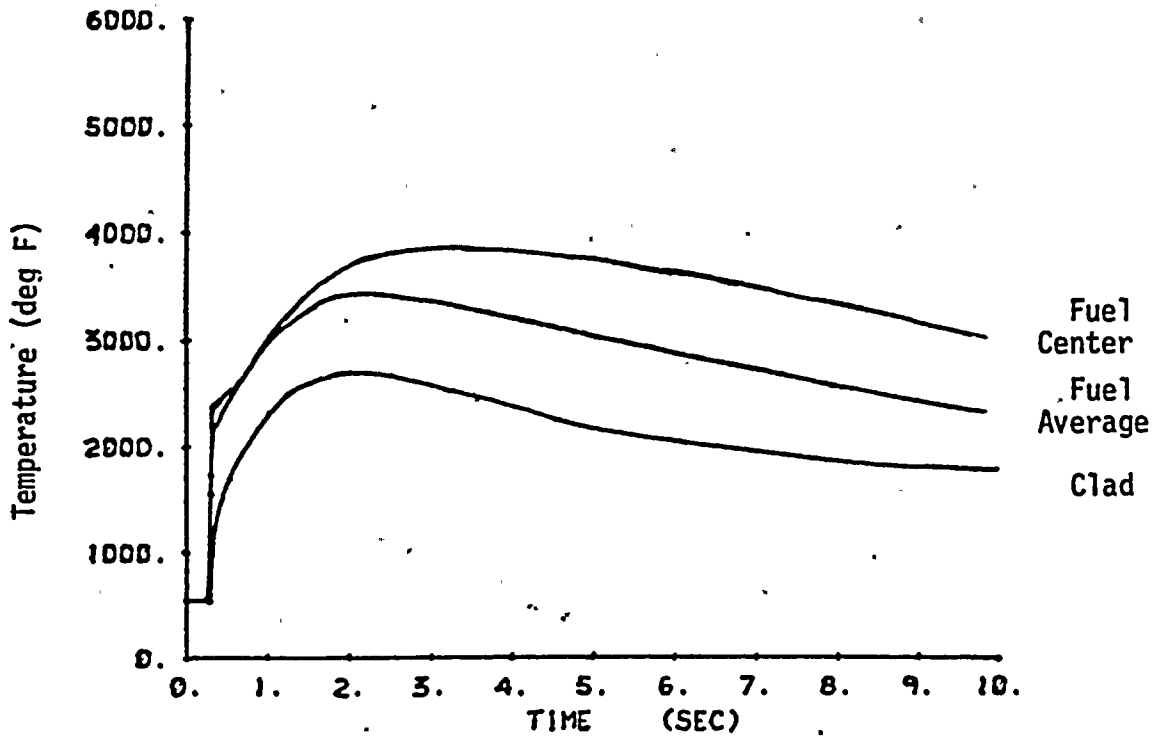


Figure 5.7-4
R. E. Ginna 15% SGTP
RCCA Ejection
Beginning of Life, Zero Power

6.0 TECHNICAL SPECIFICATION IMPACT

Technical Specification Changes as a result of the increase in SGTP level to 15% are contained in Attachment A. The changes are minor affecting only the core limits (Figure 2.1-1) and the overtemperature and overpower ΔT $f(\Delta I)$ function (2.3.1.2d).

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8.0 CONCLUSIONS

To assess the effect on accident analysis of operation of the Ginna Nuclear Power Plant with a steam generator tube plugging level of 15% evaluation and transient reanalysis was performed. Discussions of the transient evaluations are presented in Section 4. The results of the transients reanalyzed are presented in Section 5. These evaluations/reanalyses indicate that the increase in SGTP level to 15% does not result in violation of safety limits for any of the transients addressed. Therefore, the conclusions in the UFSAR and RTSR remain valid. Resulting Technical Specification changes are contained in Attachment A.