ENCLOSURE 1

POSITIVE MODERATOR TEMPERATURE COEFFICIENT AND RWST/ACCUMULATOR BORON CONCENTRATION INCREASE LICENSING REPORT FOR VOGTLE ELECTRIC GENERATING PLANT UNITS 1 AND 2

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1.0 INTRODUCTION

The present Vogtle 1 and proposed Vogtle 2 Technical Specifications require the moderator temperature coefficient (MTC) to be 0 pcm/°F* or less at all times while the reactor is critical. The implementation of a positive moderator temperature coefficient (PMTC) at reduced power levels at Beginning of Cycle Life (BOL) results in an increase in fuel cycle design flexibility while having only a minor effect on the safety analyses for the accident events presented in the FSAR.

The proposed Technical Specification change (3/4.1.1.3) would allow a +7 pcm/°F MTC below 70 percent of rated power, ramping down to 0 pcm/°F at 100 percent power. This MTC is shown in Figure 1.0-1. A power-level dependent MTC was chosen to minimize the effect of the specification on postulated accidents at high power levels. Moreover, as the power level is raised, the average core water temperature becomes higher as allowed by the programmed average temperature controller for the plant, producing a more negative moderator coefficient. Also, the boron concentration can be reduced as xenon builds into the core. Thus, there is less need to allow a positive coefficient as full power is approached. As fuel burnup is achieved, boron is further reduced and the moderator coefficient will eventually become negative over the entire operating power range.

As a result of planned reload cycle core designs using positive moderator temperature coefficient (PMTC) and high capacity factor 18-month core loading patterns, higher minimum boron concentration requirements for the Refueling Water Storage Tank (RWST) to meet post-LOCA Shutdown requirements are needed. An evaluation has been performed to demonstrate the acceptability of employing a PMTC and of increasing the boron concentration range to 2400-2600 ppm for the RWST and 1900-2600 ppm for the accumulators. The effects of the proposed changes on the FSAR Safety Analyses have been considered. (See Sections 2.0, 3.0 and 4.0)

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* 1 pcm = 10^{-5} $\Delta k/k$

The inclusion of a PMTC in the accident analyses presented in Chapter 15 of the Vogtle Final Safety Analysis Report (FSAR) has been assessed. Those events which were found to be sensitive to positive or near-zero moderator temperature coefficients were reanalyzed. In general, these events are limited to transients which cause reactor coolant temperature to increase.

The analyses presented in Section 3 that rely on the LOFTRAN computer code as a primary analytical tool were based on a +7 pcm/°F moderator temperature coefficient over the entire power range with the exception of the locked rotor accident. The locked rotor accident is discussed in Section 3.4. The coefficient was conservatively assumed to remain constant for variations in temperature for all transients. The Section 3 events which did not use LOFTRAN were rod ejection and rod withdrawal from subcritical, which are addressed in the next paragraph.

The control rod ejection and rod withdrawal from subcritical analyses were based on a coefficient which was at least +7 pcm/°F at zero power nominal average temperature, and which became less positive for higher temperatures. This was necessary since the TWINKLE computer code (Reference 2), on which the analyses are based, is a diffusion-theory rather than a point-kinetics approximation code and the moderator temperature feedback cannot be artificially held constant with temperature.

For all accidents which were reanalyzed, the assumption of a positive moderator temperature coefficient existing at 100% power is conservative since as diagrammed in Figure 1.0-1, the proposed Technical Specification (3/4.1.1.3) requires that the coefficient be linearly ramped from +7 pcm/°F to zero from 70 to 100 percent power.

In general, reanalysis was based on the identical analysis methods, computer codes, and assumptions employed in the FSAR; any exceptions are noted in the discussion of each event. Accidents not reanalyzed included those which were not affected by a PMTC or those where the assumption of a large negative moderator coefficient is conservative. Evaluations are presented in Section 2.0 for the accidents not reanalyzed. Table 1.0-1 gives a list of accidents presented in the Vogtle FSAR, and denotes those events reanalyzed or evaluated for a positive moderator coefficient. The following sections provide discussions for each of the FSAR events.

The non-LOCA analyses and evaluations for the PMTC are based upon the Vogtle Units 1 & 2 plant design. The core reactivity assumptions were based upon and confirmed for the Unit 1 Cycle 2 core design. The non-LOCA analyses described in this report do not apply to Unit 2 Cycle 1 (which will not have a PMTC). For subsequent Unit 1 and Unit 2 reload cores, the applicability of the PMTC analyses will be confirmed as part of the normal reload process.

The CVCS malfunction transient (see Section 3.9) for Modes 3, 4, and 5 was analyzed with setpoint changes specific to Unit 1. The high flux at shutdown alarm setpoint and the makeup control valve setpoint were revised as indicated in Section 3.9. Thus, the analysis presented is not applicable to the current Unit 2 Cycle 1 design. However, for subsequent Unit 2 reload cores the same setpoints identified in Section 3.9 will be utilized.

The effects of the higher boron concentrations in the RWST and accumulators has also been assessed (Section 4.0). The impacts of the boron concentration increases on the FSAR safety analyses, dose analysis, reactor vessel boron precipitation analysis, equipment qualification, etc., have been considered.

A note should be made at this point relative to the SGTR analysis discussed in Section 2.9.

The steam generator tube rupture analysis evaluated in Section 2.9 of this report is the tube rupture analysis currently discussed in Section 15.6.3 of the Vogtle FSAR. However, in order to comply with License Condition 2.C.(5) for Plant Vogtle Unit 1 and Confirmatory Items 48 and 49 of the Safety evaluation Report (SER), NUREG-1137, for Plant Vogtle Unit 2, Georgia Power Company has submitted a revised steam generator tube rupture (SGTR) analysis via Georgia Power letter SL-4149 dated 2/29/88. The effects of the PMTC and increased boron requirements have been evaluated on the FSAR chapter 15.6.3 SGTR analysis and are presented in Section 2.9. The revised PMTC and boron requirements have been incorporated into the recently submitted SGTR analysis and are also presented in Section 2.9. Please refer to Section 2.9 for further discussion.

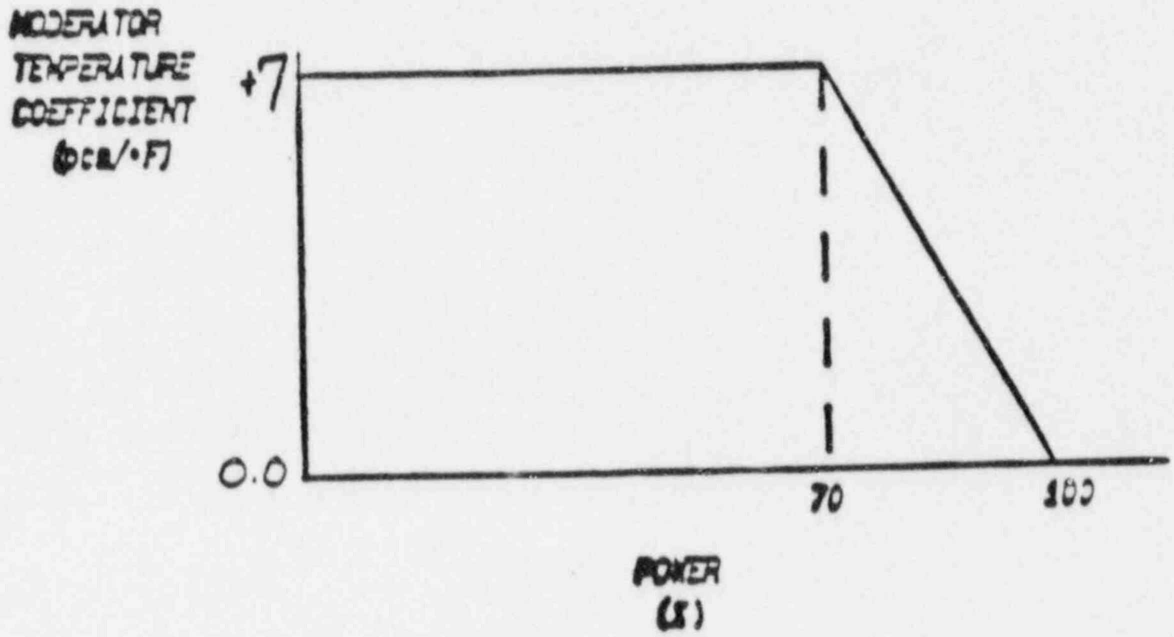


FIGURE 1.0-1

MODERATOR TEMPERATURE COEFFICIENT VS. RATED THERMAL POWER

TABLE 1.0-1

FSAR ACCIDENTS EVALUATED FOR
POSITIVE MODERATOR COEFFICIENT EFFECTS

FSAR	Accident	Discussion Section	Time[(1)] in Life
15.1.1/ 15.1.2	Feedwater System Malfunctions	2.1	EOC
15.1.3	Excessive Increase in Secondary Steam Flow	2.2	BOC/EOC
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	2.3	EOC
15.1.5/ 6.2.1.4	Steam System Piping Failure	2.3	EOC
15.2.2	Loss of External Electrical Load	(2)	
*15.2.3	Turbine Trip	3.5	BOC/EOC
15.2.4	Inadvertent Closure of Main Steam Isolation Valves	(2)	
15.2.5	Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	(2)	
*15.2.6	Loss of Nonemergency AC Power to the Plant Auxiliaries	3.7	BOC
*15.2.7	Loss of Normal Feedwater Flow	3.7	BOC
15.2.8	Feedwater System Pipe Break	2.4	EOC
*15.3.1	Partial Loss of Forced Reactor Coolant Flow	3.3	BOC
*15.3.2	Complete Loss of Forced Reactor Coolant Flow	3.3	BOC
*15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	3.4	BOC
15.3.4	Reactor Coolant Pump Shaft Break	(4)	

TABLE 1.0-1 (Continued)

FSAR	Accident	Discussion Section	Time(1) in Life
*15.4.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition	3.1	BOC
*15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	3.2	BOC/EOC
15.4.3	Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)	2.5	BOC
15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	2.6	EOC
*15.4.6	Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant	3.9	BOC
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	2.10	BOC
*15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	3.6	BOC/EOC
15.5.1	Inadvertent Operation of the Emergency Core Cooling System During Power Operation	2.7	BOC/EOC
15.5.2	Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	(3)	
*15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	3.8	BOC
15.6.3	Steam Generator Tube Failure or Rupture	2.9	BOC
15.6.5/6.2	Loss-of-Coolant Accidents	2.8	BOC

(1) BOC - Beginning of Cycle, EOC - End of Cycle
(2) Bounded by analysis in 15.2.3
(3) Covered by analyses in 15.4.6 and 15.5.1
(4) Bounded by analysis in 15.3.3

*Accidents Reanalyzed

2.0 TRANSIENTS EVALUATED FOR A POSITIVE MODERATOR TEMPERATURE COEFFICIENT

The following transients were not reanalyzed for the positive moderator temperature coefficient. A discussion of the effect of a PMTC, if any, is provided in this section for the transients not reanalyzed (refer to Table 1.0-1).

2.1 FEEDWATER SYSTEM MALFUNCTIONS (FSAR SECTIONS 15.1.1, 15.1 2)

The addition of excessive feedwater or the reduction of feedwater temperature are excessive heat removal incidents, and are consequently most sensitive to a negative moderator temperature coefficient. Results presented in Section 15.1.1 and 15.1.2 of the FSAR, based on a negative coefficient, represent the limiting case. Therefore, this incident was not reanalyzed and the conclusions of the FSAR remain valid.

2.2 EXCESSIVE INCREASE IN SECONDARY STEAM FLOW (FSAR SECTION 15.1.3)

An excessive increase in secondary system steam flow (excessive load increase event) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. This results in decreased reactor coolant system temperature. With the reactor in manual control, the analysis presented in Section 15.1.3 of the FSAR shows that the limiting manual control case assumes a large negative moderator coefficient. If the reactor is in automatic control, the control rods are withdrawn to increase power and restore the average temperature to the programmed value. The analysis of this case in the FSAR shows that the minimum DNBR is not sensitive to moderator temperature coefficient. Therefore, the results presented in the FSAR continue to be limiting and the conclusions presented remain valid.

2.3 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE/STEAM SYSTEM PIPING FAILURE (FSAR SECTIONS 15.1.4, 15.1.5 AND 6.2.1.4)

The inadvertent opening of a steam generator relief or safety valve and steam system piping failure events are transients that result in cooldown of the reactor coolant system. Therefore, the associated analyses initiated from hot zero power conditions (FSAR Sections 15.1.4 and 15.1.5) are performed assuming a strongly negative moderator temperature coefficient, which represents the limiting assumption for consideration of a post trip core power excursion. As a result these analyses are unaffected by the proposed Technical Specification change in the allowable positive moderator temperature coefficient. Therefore, the inadvertent opening of a steam generator relief or safety valve/steam system piping failure analyses of the FSAR remain limiting and the conclusions reached remain valid.

For the steamline break mass/energy release analyses (FSAR Section 6.2.1.4 - inside containment; WCAP-11285, "MSLB Information Used For Superheat Study For Vogtle Units 1 and 2" - outside containment) a negative moderator temperature coefficient produces the most limiting results. Therefore, the results of these analyses remain limiting and the conclusions remain valid.

2.4 FEEDWATER SYSTEM PIPE BREAK (FSAR SECTION 15.2.8)

The main feedwater pipe rupture accident is analyzed to demonstrate the long-term ability of the auxiliary feedwater system to remove decay heat from the reactor coolant system after reactor trip. For conservatism, the current FSAR analysis is performed at 102% of Engineered Safeguards power (104.5% NSSS rated) in order to significantly increase the total energy which eventually must be removed from the core, due to stored energy and decay heat. This analysis was performed using a large negative moderator temperature coefficient to provide positive reactivity feedback as the RCS temperature decreases in the long-term, and sensitivities have confirmed that this assumption continues to be appropriate. The event is not sensitive to a PMTC since the reactor trip occurs near the beginning of the transient limiting the RCS temperature to a small increase. Again, the focus of this transient is on long-term decay heat and stored energy heat removal capability of the auxiliary feedwater system.

Therefore, the results for this accident are unaffected by the incorporation of a positive moderator temperature coefficient Technical Specification.

Based on this, the event presented in the FSAR remains limiting and the associated conclusions remain valid.

2.5 RCCA MISALIGNMENT (FSAR SECTION 15.4.3)

The static misalignment cases discussed in FSAR Section 15.4.3 are not impacted by a PMTC. These cases are steady state analyses, and thus there is no transient response to be impacted by a PMTC. Therefore the conclusions of the FSAR remain valid.

For the single rod withdrawal at power case in Section 15.4.3, it is assumed that the withdrawal occurs until a reactor trip is generated by the overtemperature delta-T signal. The analysis is performed at the steady state power and coolant conditions which are expected to trip the plant. These conditions are not dependent on a PMTC, and therefore, this case was not reanalyzed. The conclusions of the FSAR remain valid.

The remaining cases discussed in FSAR Section 15.4.3 are the single or multiple dropped RCCAs. Use of a positive coefficient in the analysis would result in a larger reduction in core power level following the RCCA drop, thereby increasing the potential of a reactor trip. For the return to power with automatic rod control cases with lower worth dropped RCCAs, a positive coefficient would result in a small increase in the power overshoot. The limiting conditions for this transient occur at or near 100% power. Since the moderator temperature coefficient must be close to zero or negative at 100% power, the limiting case is unaffected by the proposed Technical Specification and reanalysis was not performed. The limiting analysis presented in the FSAR remains bounding and the associated conclusions remain valid.

2.6 STARTUP OF AN INACTIVE REACTOR COOLANT PUMP AT AN INCORRECT TEMPERATURE (FSAR SECTION 15.4.4)

An inadvertent startup of an idle reactor coolant loop at an incorrect temperature results in a decrease in core average temperature. As the most negative values of moderator reactivity coefficient produce the greatest reactivity addition, this accident is unaffected by the proposed Technical Specification and thus reanalysis was not required. Therefore, the analysis presented in FSAR Section 15.4.4 remains limiting and the conclusions reached remain valid.

2.7 INADVERTENT OPERATION OF THE ECCS DURING POWER OPERATION (FSAR SECTION 15.5.1)

This transient results in a decrease in average reactor coolant temperature and core power. Since tempe. ature is decreasing, a PMTC will result in the addition of negative reactivity. This will cause power to decrease even more rapidly, and margin to DNB will increase. Therefore, this incident was not reanalyzed with a positive moderator coefficient. The analysis presented in the FSAR continues to be limiting and the conclusions reached remain valid.

2.8 LOSS-OF-COOLANT ACCIDENTS (LOCA) (FSAR SECTIONS 6.2 AND 15.6.5)

Small Break LOCA

The current small break LOCA analysis for Vogtle Units 1 and 2 was performed using the WFLASH Evaluation Model, which assumes the reactor core is brought to a subcritical condition by the trip reactivity of the control rods. The influence of a PMTC on the calculated peak clad temperatures for a small break LOCA analysis has been evaluated from two different perspectives; first, the estimated effect of a PMTC on small break LOCA response, and second, the margin available to meet 10CFR50.46 ECCS acceptance criteria.

In the standard small break LOCA analysis methodology, core kinetics calculations are not explicitly performed. Instead, the core power is maintained at initial conditions (102% power) until the reactor trip setpoint is reached and a delay time has passed. The delay accounts for signal processing and the time it takes for the rods to reach the bottom of the core. This delay results in the generation of an additional few full-power-seconds of heat, by not accounting for partial rod worth while the rods are falling into the core and the shutdown effect of voiding for cores which have a negative moderator temperature coefficient. After trip, the decay heat power level is calculated by interpolation from a table of power versus time which has been conservatively derived to envelope all plants. This generic power decay curve is composed of three parts: residual fission heat, fission product decay, and actinide gamma decay. The residual fission term is based upon the exponential decay of the fission power for a low shutdown margin, and with the core full of hot water. This will be conservative for essentially all small break LOCAs, since some net voiding occurs coincident with reactor trip, due to the sudden depressurization.

An evaluation has determined that the excess core power generation which may be expected from explicitly modeling a PMTC of +7 pcm/°F would be much less than 1 full-power-second. This excess power could impact the transient in two areas; 1) the effect of increased power on the time of reactor trip and SI initiation signals, and 2) the influence of increased heat generation on peak clad temperature (PCT). By reference to applicable calculations and sensitivity studies it has been concluded that the PMTC has only a small, third order effect on PCT. The small core power excursion induced in the initial few seconds of the transient will slow the depressurization negligibly, delaying reactor trip and SI initiation only slightly. These delays, plus the small excess power, in turn will have a small influence on loop clearing and subsequently the core boiloff uncovery transient, hundreds of seconds into the accident, during which time the clad PCT occurs. However, there will be virtually no direct influence on decay heat generation during the clad temperature excursion.

The existing small break LOCA analysis of record, performed with the Westinghouse small break LOCA code, WFLASH, show that a very substantial margin exists (>600°F) between the results calculated in the small break LOCA analysis of record and the 10CFR50.46 limit of 2200°F. Any impact from operation with a PMTC as indicated previously will be very small compared to this margin. Thus the implementation of a PMTC of +7 pcm/°F from 0% power to

70% power and following a linear ramp down to 0 pcm/°F at 100% power does not alter the conclusions of the FSAR small break LOCA analysis, and meets the acceptance criteria in 10CFR50.46.

Large Break LOCA

For large break LOCA analyses, PMTC can currently only be modeled during blowdown. Once voids form during blowdown the negative reactivity added by the decrease in moderator density and the increase in neutron leakage from the core is substantial compared to the reactivity added during the brief period of blowdown where positive reactivity feedback occurs. Therefore the implementation of a positive moderator temperature coefficient will not alter the results or conclusions of the FSAR large break LOCA analysis and the requirements of 10CFR50.46 will still be met.

LOCA Forces

The primary factors affecting a LOCA forces analysis are pressure, temperature and density. The PMTC could potentially result in an increase in the system pressure or temperature resulting in changes in the fluid density. However, the transient is essentially over with the peak loads having been calculated before any feedback from the PMTC, which could alter the results of the forces analysis, could occur. Therefore the proposed positive moderator temperature coefficient will not alter the results of the forces analysis.

Short and Long Term LOCA Mass and Energy Releases (FSAR Section 6.2)

The containment integrity analyses are described in the FSAR Section 6.2. This section considers, Subcompartment Pressure Transient Analyses, Short Term and Long Term Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents (LOCA), and, Containment Response Analyses following a LOCA or Steamline Break Inside Containment.

For the containment integrity analyses, PMTC can currently only be modeled during the blowdown phase of the LOCA mass and energy analysis and in the steamline break mass and energy analysis. The LOCA mass and energy release analysis was performed based upon operation at 100% power. A proposed PMTC as identified in the introduction section will not affect the analysis since the PMTC is zero for 4-loop operation at 100% power. The steamline break mass and energy analyses are conservatively modeled using a negative Moderator Temperature Coefficient (MTC). The negative MTC is conservative because the steamline break transient is a cooldown of the RCS. It is therefore concluded, that the implementation of a PMTC will have no adverse effect on the Containment Integrity Analyses performed for Vogtle.

2.9 STEAM GENERATOR TUBE FAILURE OR RUPTURE (SGTR) (FSAR Section 15.6.3)

As mentioned in the Introduction section of this report (Section 1.0), Georgia Power has recently submitted a revised SGTR analysis in compliance with Plant Vogtle Unit 1 license conditions and Plant Vogtle Unit 2 confirmatory items. This revised SGTR analysis will ultimately replace the current SGTR analysis discussed in FSAR Section 15.6.3. Due to the existence of two SGTR analyses, the evaluation for the effect of PMTC on the analyses will be discussed separately.

SGTR Analysis (FSAR Section 15.6.3)

Since the PMTC will affect the mass release data for the FSAR SGTR analysis, an evaluation was performed to estimate the effects of PMTC. The Vogtle FSAR SGTR analysis was performed using the LOFTRAN program. The primary to secondary break flow was assumed terminated at 30 minutes after initiation of the SGTR event. The major factors that affect the radiological doses of an SGTR event are the amount of fuel failure, the amount of primary coolant transferred to the secondary side of the ruptured steam generator through the ruptured tube, the break flow flashing fraction, and the steam released from the ruptured steam generator to the atmosphere.

The minimum DNBR for the SGTR analysis with PMTC has been determined utilizing a LOFTRAN run. The results indicate that the minimum DNBR remains above the DNB limits. Therefore, fuel failure will not occur during the STGR accident employing PMTC.

Due to the increases in nuclear power and Tavg following an SGTR for a PMTC core, the time of reactor trip is reduced. The primary to secondary leakage after trip and the steam released from the ruptured steam generator to the atmosphere will increase as a result of an earlier reactor trip. It was conservatively determined that the integrated primary to secondary leakage (trip to 30 min) will increase by less than 34%, and the steam released from ruptured SG (trip to 30 min) will increase by less than 11.5%. The effects on the offsite doses for the increases in primary-to-secondary leakage, and the steam released from the ruptured steam generator to the atmosphere due to the PMTC have been evaluated based on the radiological analysis methodology presented in FSAR Section 15.6.3. The thyroid and whole body doses are estimated to increase by less than 19 and 22 percent.

The offsite doses considering the effects of the PMTC, while greater than currently reported in the FSAR, do not violate acceptance limits. This judgement is based on the fact that the dose increase is small and that the total dose is very low, being well below the NRC definition of a "small fraction" of the 10CFR100 exposure guideline. This "small fraction" limit, (defined as 10% of the guideline value) is 30 rem thyroid and 2.5 rem whole body, and is the smallest of the exposure limits defined by the NRC in NUREG-0800.

Revised SGTR Analysis (WCAP-11731)

As a result of License Condition 2.C.(5) for Plant Vogtle Unit 1 and Confirmatory Items 48 and 49 of the Safety Evaluation Report (SER), NUREG-1137, for Plant Vogtle Unit 2, a revised SGTR analysis was performed based on the Westinghouse Owner's Group (WOG) methodology which was approved by the NRC in a letter from C. Rossi (NRC) to A. Ladieu (WOG) dated March 30, 1987. The revised analysis employed Plant Vogtle specific PMTC values. The results of the revised SGTR analysis are documented in WCAP-11731 entitled" LOFTTR2 Analysis for a Steam Generator Tube Rupture Event for the Vogtle Electric Generating Plant Units 1 and 2, January 1988." The WCAP documents acceptable results.

2.10 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION (FSAR SECTION 15.4.7)

The analysis presented in the FSAR is a steady state analysis, and thus there is no transient response to be impacted by a PMTC. The conclusions of the FSAR remain valid.

3.0 TRANSIENTS ANALYZED FOR A POSITIVE MODERATOR TEMPERATURE COEFFICIENT

The principal computer codes used for the reanalyses documented in this section are LOFTRAN (Reference 5), TWINKLE (Reference 6), FACTRAN (Reference 7), and THINC (References 8 and 9). These codes are the same as those used in the Vogtle FSAR Analyses. Summaries of these computer codes are presented in Sections 15.0 and 4.4 of the FSAR.

For each event reanalyzed the basic assumptions regarding initial conditions, instrumentation errors, and setpoint errors remain essentially the same as those found in Chapter 15 of the FSAR. However, the current analyses do incorporate certain additional changes that should be noted. FSAR Section 15.0.3.2 specifies a $\pm$ 30 psi allowance on pressurizer pressure for steady state fluctuations and measurement penalty. The reanalyzed events include a more conservative $\pm$ 45 psi allowance for pressurizer pressure. Additionally, increased uncertainties have been applied to the pressurizer and steam generator water levels. These uncertainties have been increased from 5% to 6.6% for both the pressurizer and the steam generator. These increased uncertainties have been incorporated to bound calculated increases in the associated transmitter uncertainties. (Reference 14)

3.1 UNCONTROLLED RCCA BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW-POWER STARTUP CONDITION (FSAR SECTION 15.4.1)

Introduction

An RCCA bank withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (see Section 15.4.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the reactivity feedback of the negative fuel temperature coefficient and a reactor trip on source, intermediate or power range flux, or high positive nuclear flux rate. The power excursion causes a heatup of the moderator and fuel. A positive moderator coefficient causes an increase in the rate of reactivity addition, resulting in an increase in peak heat flux and peak fuel and clad temperature.

Method of Analysis

The analysis discussed in the FSAR assumed a reactivity insertion rate of 60 pcm/sec. The accident was reanalyzed with the same insertion rate. This insertion rate is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). The analysis used a moderator temperature coefficient of +7 pcm/°F at hot zero power initial conditions. The computer codes, initial conditions, and other assumptions remain as noted in the FSAR with the exception of initial RCS pressure which was assumed to be nominal minus 45 psi.

Results

Figures 3.1-1 through 3.1-3 show the transient behavior for the uncontrolled RCCA bank withdrawal incident, with the accident terminated by reactor trip at 35% of nominal power.

Figures 3.1-1 and 3.1-2 show the neutron flux and thermal flux transients. The neutron flux overshoots the nominal full power value; however, due to the beneficial effect of the inherent thermal lag in the fuel, the peak heat flux is much less than the full power nominal value.

Figure 3.1-3 shows the hot spot fuel average and clad temperature transients. The minimum DNBR at all times remains above the limiting value.

The calculated sequence of events for this transient is shown in Table 3.1-1.

Conclusion

In the event of a RCCA bank withdrawal accident from a subcritical condition, the core and the RCS are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR greater than the limit value and thus, no fuel or clad damage is predicted. Therefore, the conclusions presented in the FSAR remain valid.

3.2 UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER (FSAR SECTION 15.4.2)

Introduction

The Uncontrolled RCCA bank withdrawal at power event is described in Section 15.4.2 of the FSAR. An uncontrolled RCCA bank withdrawal at power causes a positive reactivity insertion which results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation, there is a net increase in the reactor coolant temperature. With a PMTC, this temperature increase will add positive reactivity. Unless terminated by manual or automatic action, the increase in coolant temperature and power could result in DNB. For this event, the Power Range High Neutron Flux and Overtemperature Delta-T reactor trips are assumed to provide protection against DNB. Therefore, the minimum reactivity feedback cases for this event were reanalyzed with a +7 pcm/°F moderator temperature coefficient to show that the DNBR limit is met. The maximum reactivity feedback cases presented in the FSAR assume a large positive moderator density coefficient (i.e., a large negative temperature coefficient), and therefore are not impacted by the PMTC.

Methods

With the exception of the items noted in Section 3.0, the assumptions used are consistent with the FSAR. The transient is analyzed at 10%, 60%, and 100% power assuming minimum reactivity feedback. A constant moderator coefficient of +7 pcm/°F was used in the analysis. The assumption that a positive moderator coefficient exists at full power is conservative since at full power the moderator coefficient will actually be zero or negative. The analysis was performed using the LOFTRAN computer code.

Results

The DNBR limit is met for the range of reactivity insertion rates analyzed at the various power levels. A calculated sequence of events for a fast and slow insertion rate from full power is presented in Table 3.2-1. The transient response for a fast insertion case and a slow insertion case from full power is shown in Figures 3.2-1 through 3.2-4. The plots of minimum DNBR versus reactivity insertion rate at the analyzed power levels are shown as Figures 3.2-5 through 3.2-7.

Conclusions

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

3.3 LOSS OF FORCED REACTOR COOLANT FLOW (FSAR SECTIONS 15.3.1, 15.3.2)

Introduction

The loss of flow events presented in FSAR Sections 15.3.1 and 15.3.2 were reanalyzed to determine the effect of a +7 pcm/°F moderator temperature coefficient on the nuclear power transient and the resultant minimum DNBR reached during the incident. The effect on the nuclear power transient would be limited to the initial stages of the incident during which reactor coolant temperature increases; this increase is terminated shortly after reactor trip.

Method of Analysis

With the exception of the moderator temperature coefficient and the items noted in Section 3.0, the methods and assumptions used in the reanalysis were consistent with the FSAR. Both of the cases presented in the FSAR, partial and complete loss of flow, were reanalyzed. The computer codes used in the reanalysis remained the same as those described in the FSAR, while a constant moderator temperature coefficient of +7 pcm/°F was used in the reanalysis to reflect the revised Technical Specification.

Results

Figures 3.3-1 through 3.3-4 show the transient response for the loss of two reactor coolant pumps with four loops in operation (partial loss of flow). Figure 3.3-4 shows the DNBR to be always greater than 1.30.

For the partial loss of flow case analyzed, since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the partial loss of flow case is shown in Table 3.3-1. The affected reactor coolant pumps will continue to coastdown, and the core flow will reach a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be obtained. Normal plant shutdown may then proceed.

Figures 3.3-5 through 3.3-8 show the transient response for the loss of power to all four reactor coolant pumps with four loops in operation. The reactor is assumed to be tripped on an RCP power supply bus undervoltage signal. Figure 3.3-8 shows the DNBR to be always greater than the limit value.

For the complete loss of flow case analyzed, since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the complete loss of flow case is shown in Table 3.3-1. The reactor coolant pumps will continue to coastdown, and natural circulation flow will eventually be established. As demonstrated in Section 15.2.6 of the FSAR, with the reactor tripped, a stable plant condition would be attained. Normal plant shutdown may then proceed.

Conclusions

The DNBR design basis is met for the partial and complete loss of flow cases. Therefore, the conclusions of the FSAR remain valid.

3.4 REACTOR COOLANT PUMP SHAFT SEIZURE (FSAR SECTION 15.3.3)

Introduction

The case presented in the FSAR (Section 15.3.3) for this transient was reanalyzed. Upon initiation of a locked rotor incident, reactor coolant system temperature rises until shortly after reactor trip. A positive moderator coefficient will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the incident. The transient was reanalyzed, however, due to the potential impact on the nuclear power transient which would affect the peak reactor coolant system pressure and clad temperatures.

The FSAR presents a radiological consequences evaluation for the locked rotor event. As part of the FSAR evaluation, a locked rotor analysis was performed to determine the percentage of fuel rods in DNB for the transient. For the radiological consequences, it is assumed that fuel rods in DNB will fail for this event. Existing sensitivity studies for this transient have shown that assuming 0 MTC at full power is more limiting than assuming a PMTC at a reduced power. The sensitivity results model the PMTC Technical Specification limits which do not allow a PMTC at full power. The sensitivity study modeled the same PMTC as that proposed for plant Vogtle. Based on the existing sensitivity results, the locked rotor analysis which determines the percentage of fuel rods in DNB was not reanalyzed. As a result, there is no effect on the radiological consequences evaluation.

Method of Analysis

With the exception of the items noted in Section 3.0, the methods and assumptions used in the reanalysis were consistent with the FSAR. The cases in the FSAR, one locked rotor for 4 loops in operation both with and without offsite power available, were reanalyzed using the same computer codes described in the FSAR. The reanalysis, which initiated the transient from full power conditions, employed a constant moderator temperature coefficient of +7.0 pcm/°F for peak pressure and clad temperature analyses.

Results

The transient results for the locked rotor event are shown on Figures 3.4-1 through 3.4-4. The results of these calculations are also summarized in Tables 3.3-1 and 3.4-1. The peak reactor coolant system pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than 2,700°F and the amount of zirconium-water reaction is small. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

Conclusions

Since the peak reactor coolant system pressure reached during both of the cases is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered. The peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2,700°F. Therefore, the conclusions presented in the FSAR with respect to peak pressure and clad temperature remain valid.

3.5 TURBINE TRIP (FSAR SECTION 15.2.3)

Introduction

A turbine trip event is more limiting than loss of external electrical load, loss of condenser vacuum, and other events which result in a turbine trip (FSAR Section 15.2.5). Inadvertent closure of the main steam isolation valves (FSAR Section 15.2.4) results in a turbine trip. Therefore, only turbine trip (FSAR Section 15.2.3) is reanalyzed. Only the minimum reactivity feedback cases (Beginning of Life) presented in the FSAR were reanalyzed because the maximum feedback cases (End of Life) assume a negative moderator temperature coefficient. The two minimum reactivity cases differ in the assumptions made for pressurizer pressure control. The two sets of pressure control assumptions are:

1. Full credit is taken for operation of the pressurizer spray and power operated relief valves. The pressurizer safety valves are also assumed to be available.

2. No credit is taken for the operation of the pressurizer spray and power operated relief valves. The pressurizer safety valves are assumed to be available.

<u>Method of Analysis</u>

The minimum reactivity feedback cases were run with a constant moderator temperature coefficient of +7 pcm/°F, which is conservative for full power. With the exception of the items noted in Section 3.0, the assumptions and the methodology employed were consistent with the FSAR.

<u>Results</u>

For the combinations of minimum reactivity feedback and pressure control, the applicable safety limits are met. The results of these cases are presented as Figures 3.5-1 through 3.5-4. A calculated sequence of events is shown in Table 3.5-1.

Figures 3.5-1 and 3.5-2 show the responses for a turbine trip event with minimum reactivity feedback assuming operability of pressurizer sprays and PORV's. The reactor is tripped by the High Pressurizer Pressure trip function. The DNBR remains above the design limit throughout the transient. The primary system pressure remains below the 110% design value.

Figures 3.5-3 and 3.5-4 show the responses for a turbine trip with minimum reactivity feedback and without pressure control. The reactor is tripped by the High Pressurizer Pressure trip function, and the DNBR does not drop below the initial value. The primary system pressure remains below the 110% design value.

Conclusions

The DNBR design basis is met and the system pressure remains below 110% of the design value in the minimum reactivity feedback cases and therefore, the conclusions presented in the FSAR remain valid. This transient remains the limiting Condition 2 transient with respect to peak pressure. Because it has been demonstrated that the system pressure remains below 110% of the design value, the conclusions of the Overpressure Protection Report remains valid.

3.6 Spectrum of RCCA Ejection Accidents (FSAR Section 15.4.8)

Introduction

The RCCA ejection transient is analyzed at full power and hot standby for both beginning and end of life conditions in the FSAR. Since the moderator temperature coefficient is negative at end of life, only the beginning of life cases are affected by a positive moderator temperature coefficient. The high nuclear power levels and hot spot fuel temperatures resulting from a rod ejection are increased by a positive moderator coefficient. Reactor trip occurs due to a high neutron flux signal. A discussion of this transient is presented in Section 15.4.8 of the FSAR.

Method of Analysis

The digital computer codes for analysis of the nuclear power transient and hot spot heat transfer are the same as those used in the FSAR. The ejected rod worths and transient peaking factors assumed are conservative with respect to the actual calculated values for the current fuel cycle. The analysis used a moderator temperature coefficient of +7 pcm/°F for hot zero power and full power conditions.

Results

The cases analyzed were beginning of life at hot full power and hot zero power. The peak hot spot clad average temperature was reached in the hot zero power case. However, the peak hot spot value of 2490°F was below the limit specified in the FSAR.

The maximum fuel temperature and fuel enthalpy were associated with the hot full power case. Although the peak fuel centerline temperature at the hot spot exceeded melting, the extent of the melting was restricted to less than the innermost 10% of the pellet. The peak fuel enthalpy in both cases was well below the limit specified in the FSAR.

A summary of the parameters and results is presented in Table 3.6-2. The calculated sequence of events for each case is shown in Table 3.6-1. The nuclear power and hot spot fuel and clad temperature transients for both cases, are shown in Figures 3.6-1 thru 3.6-4.

Conclusions

As fuel and clad temperatures do not exceed the fuel and clad limits specified in the FSAR, there is no danger of sudden fuel dispersal into the coolant, or consequential damage to the primary coolant loop. Therefore, the conclusions presented in the FSAR remain valid.

3.7 Loss of Normal Feedwater Flow/Loss of Nonemergency AC Power to the Plant Auxiliaries (FSAR Sections 15.2.7 and 15.2.6)

Introduction

This accident is described in Sections 15.2.6 and 15.2.7 of the FSAR. Section 15.2.6 represents the analysis which assumes offsite power is lost. Section 15.2.7 assumes offsite power is available. Since this transient is analyzed consistent with beginning of life conditions, it was reanalyzed with a positive moderator temperature coefficient.

Methods

A constant moderator temperature coefficient of +7 pcm/°F was assumed. A conservative core decay heat model based on the 1979 version of ANS 5.1 (Reference 10) was used. The pressurizer pressure control system (sprays and power operated relief valves) was assumed to be available since a lower pressure results in greater system expansion. With the exception of the items noted in Section 3.0, all remaining assumptions are consistent with the analysis presented in the FSAR. For the case without offsite power available, the uncertainties and errors on the initial average temperature were subtracted from the nominal value, and power is assumed to be lost to the reactor coolant pumps following rod motion. The reanalysis used LOFTRAN to obtain the plant transient following a loss of normal feedwater.

Results

The transient response of the RCS following a loss of normal feedwater is shown in Figures 3.7-1 through 3.7-5 with offsite power available and Figures 3.7-6 through 3.7-10 for the case without offsite power available. The calculated sequences of events are listed in Tables 3.7-1 and 3.7-2. The plots of pressurizer water volume clearly show that for all cases the pressurizer does not fill.

Conclusions

The reanalysis shows that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system, and the auxiliary feedwater system is sufficient to preclude water relief through the pressurizer relief or safety valves. For the case without offsite power available, the natural circulation capability of the RCS is sufficient to remove decay heat following a RCP coastdown to prevent fuel or clad damage. Therefore, since the pressurizer does not fill, the conclusions presented in the FSAR remain valid.

3.8 Inadvertent Opening of a Pressurizer Safety or Relief Valve (FSAR SECTION 15.6.1)

Introduction

This event is analyzed in Section 15.6.1 of the FSAR using a zero moderator coefficient in order to minimize negative reactivity feedback. A positive moderator temperature coefficient can also be considered as a negative density coefficient and therefore, the density reduction due to the RCS depressurization causes a positive reactivity insertion and an increase in nuclear power. Reactor trip is generated by the overtemperature delta-T function. The RCS depressurization incident is reanalyzed to determine the impact on the nuclear power transient and the minimum DNBR.

Methods

Assumptions made in the RCS Depressurization analysis include a constant moderator temperature coefficient (+7 pcm/°F) and a small (absolute value) Doppler coefficient of reactivity such that the resultant amount of positive feedback is conservatively high. The rod control system is assumed to be in the manual mode in order to prevent rod insertion due to an RCS temperature and power mismatch prior to reactor trip. Action of the automatic rod control system would tend to reduce nuclear power and temperature resulting in a higher DNBR. With the exception of the items noted here and in 3.0, the method of analysis and assumptions used were otherwise in accordance with those presented in the FSAR.

Results

A calculated sequence of events is presented in Table 3.8-1. Figures 3.8-1 and 3.8-2 show the nuclear power, average temperature, pressurizer pressure, and DNBR vs. time for the accidental depressurization of the RCS. The positive moderator coefficient causes nuclear power and temperature to increase as pressure decreases until reactor trip occurs on Overtemperature Delta-T. The DNBR decreases initially, but increases rapidly following the trip. The DNBR remains above 1.30 throughout the transient.

Conclusions

The analysis demonstrates that the DNBR remains above 1.30 and therefore, the conclusions presented in the FSAR remain valid.

3.9 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant. (FSAR Section 15.5.2, 15.4.6)

Introduction

This event is analyzed for Operational Modes 1, 2, 3, 4, and 5. For Mode 6, administrative procedures require that certain valves be locked closed to prevent a boron dilution. The analyses discussed in the FSAR use RCS volumes, critical boron concentrations, boron worths, and dilution flow rates to calculate the amount of time that the operator has to mitigate the event before losing shutdown margin. The analyses do not model a moderator temperature coefficient and thus are not directly impacted by the PMTC. There are indirect impacts which must be addressed, and these are discussed below.

The boron dilution for Mode 1, assuming manual rod control, is dependent on the results of the RCCA bank withdrawal at power analysis (see Section 3.2). A reactivity insertion rate is calculated for the boron dilution event and compared to those analyzed for RCCA bank withdrawal at power. This comparison gives the time of reactor trip which is used as the first indication for the operator that a transient is taking place. Because the RCCA bank withdrawal at power transient was reanalyzed, the boron dilution event for Mode 1 in manual rod control was also reanalyzed.

The boron dilution analyses for Modes 3, 4, and 5 result in curves of required shutdown margin as a function of RCS boron concentration. Meeting the requirements of these curves ensures that there are at least 15 minutes of operator action time. The operator action time is the time from the high flux at shutdown alarm to the time when shutdown margin is lost. The resulting shutdown margin curves are Technical Specification Figures 3.1-1 and 3.1-2.

Because greater RCS boron concentrations are part of the Unit 1 Cycle 2 design (and greater concentrations are expected for subsequent Unit 1 and 2 reloads), the current curves had to be revised to cover the higher concentrations.

Method of Analysis

The Mode 1 analysis was performed in the same manner as the current FSAR analysis. The dilution flow assumed was the same as the current FSAR analysis. The assumed RCS volume was increased slightly and is consistent with that used for Mode 3 and 4 (Mode 4 with at least on RCP running).

The analysis for Modes 3, 4, and 5 was performed in the same manner as the current FSAR analysis. Reload boron data was used in the analysis. In addition, the analysis assumed that the high flux at shutdown alarm setpoint will be revised to 2.3 times background, and that the makeup flow control valve setpoint will be revised to 100 gpm. The change to a 100 gpm setpoint allows a lower dilution flow value (110 gpm) to be assumed in the analysis. The alarm setpoint change and the revised dilution flow provide relief from excessively high shutdown margin requirements. The assumption for the dilution flow remains consistent with the probabilistic analysis described in the FSAR.

Results

For Mode 1 in manual rod control, a minimum of 16.9 minutes are available for operator action to terminate the event.

For Modes 3, 4, and 5, the required shutdown margin plotted as a function of RCS boron concentration are shown in Figures 3.9-1 and 3.9-2.

Conclusions

The boron dilution event for Modes 1 (manual rod control), 3, 4, and 5 have been reanalyzed to address the impacts of the PMTC and the resulting increases in boron concentration. The Mode 2 analysis is not impacted by PMTC. The results provide sufficient operator action time to terminate the dilution event and the conclusions in the FSAR remain valid.

TABLE 3.1-1

TIME SEQUENCE OF EVENTS FOR AN UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL CONDITION

EVENT	TIME (SEC)
[illegible] of uncontrolled rod withdrawal fr[illegible] of nominal power	0.0
Power range high neutron flux low setpoint reached	11.6
Peak nuclear power occurs	11.8
Rods begin to fall into core	12.1
Peak heat flux occurs	13.3
Minimum DNBR occurs	13.5
Peak average clad temperature occurs	13.7
Peak average fuel temperature occurs	13.9

TABLE 3.2-1

TIME SEQUENCE OF EVENTS FOR AN UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER

ACCIDENT	EVENT	TIME (SEC)
Case A	Initiation of Uncontrolled RCCA withdrawal at a high-reactivity insertion rate (80pcm/sec) with minimum reactivity feedback at full power	0.0
	Power range high neutron flux high trip setpoint reached	1.32
	Rods begin to fall into core	1.82
	Minimum DNBR occurs	2.8
Case B	Initiation of Uncontrolled RCCA withdrawal at a small reactivity insertion rate (3 pcm/sec) with minimum reactivity feedback at full power	0.0
	Overtemperature ΔT setpoint reached	24.5
	Rods begin to fall into core	26.5
	Minimum DNBR occurs	27.1

TABLE 3.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE

ACCIDENT	EVENT	TIME (SEC)
Partial Loss of forced reactor coolant flow		
Loss of two pumps with four loops in operation	Coastdown Begins	0.0
	Low flow reactor trip	1.5
	Rods begins to drop	2.5
	Minimum DNBR occurs	3.7
Complete loss of forced reactor coolant flow		
Loss of four pumps with four loops in operation	All operating pumps lose power and begin coasting down	0.0
	Reactor coolant pump undervoltage trip point reached	0.0
	Rods begin to drop	1.5
	Minimum DNBR occurs	3.0
Reactor coolant pump shaft seizure (locked rotor)		
One locked rotor with four loops in operation with offsite power available	Rotor on one pump locks	0.0
	Low flow trip point reached	0.05
	Rods begin to drop	1.05
	Maximum RCS pressure occurs	2.9
	Maximum clad temperature occurs	3.2

TABLE 3.3-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT
IN A DECREASE IN REACTOR COOLANT SYSTEM FLOWRATE

ACCIDENT	EVENT	TIME (SEC)
One locked rotor with four loops in operation without offsite power	Rotor on one pump locks	0.0
	Low-Flow trip point reached	.05
	Rods begin to drop	1.05
	Maximum RCS pressure occurs	2.9
	Maximum clad temperature occurs	3.2

TABLE 3.4-1

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS
(FOUR LOOPS OPERATING INITIALLY)

	WITH OFFSITE POWER AVAILABLE	WITHOUT OFFSITE POWER AVAILABLE
Maximum Reactor Coolant System Pressure (psia)	2605	2605
Maximum Clad Temperature (°F) Core Hot Spot	1742	1742
Zr-H_2O Reaction, Core Hot Spot	.219	.219

TABLE 3.5-1

TIME SEQUENCE OF EVENTS FOR A TURBINE TRIP

ACCIDENT	EVENT	TIME (SEC)
FSAR Case A with pressurizer control (minimum reactivity feedback)	Turbine Trip, loss of main feedwater flow,	0.0
	Initiation of Steam release from S/G safety valves	5.5
	High pressurizer pressure reactor trip point reached	8.7
	Rods begin to drop	10.7
	Minimum DNBR occurs	12.0
	Peak pressurizer pressure occurs	12.5
FSAR Case C without pressurizer control (minimum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip point reached	5.3
	Initiation of steam release from S/G safety valves	5.5
	Rods begin to drop	7.3
	Peak pressurizer pressure occurs	8.5
	Minimum DNBR occurs	(1)

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

TABLE 3.6-1

TIME SEQUENCE OF EVENTS FOR AN RCCA EJECTION ACCIDENT

ACCIDENT	EVENT	TIME (SEC)
1. Beginning of life, full power	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.04
	Peak nuclear power occurs	0.14
	Rods begin to fall into core	0.54
	Peak average fuel temperature occurs	1.85
	Peak clad temperature occurs	1.92
	Peak heat flux occurs	1.95
2. Beginning of life, zero power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.26
	Peak nuclear power occurs	0.31
	Rods begin to fall into core	0.76
	Peak average clad temperature occurs	1.96
	Peak heat flux occurs	1.99
	Peak average fuel temperature occurs	2.12

TABLE 3.6-2

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

TIME IN LIFE	HFP BEGINNING	HZP BEGINNING
Power Level, %	102	0
Ejected rod worth, % delta-k	0.24	0.75
Delayed neutron fraction, %	0.55	0.55
Feedback reactivity weighting	1.3	1.744
Trip reactivity, % delta-k	4.0	2.0
F_q before rod ejection	2.50	--
F_q after rod ejection	6.50	11.0
Number of operational pumps	4	2
Max. Average fuel pellet	3931	3396
Max. fuel center temperature (°F)	4900	3889
Max. average clad temperature	2159	2490
Max. fuel stored energy (cal/gm)	171	143
Percent of fuel melted	<10	0

TABLE 3.7-1

TIME SEQUENCE OF EVENTS FOR A LOSS OF NORMAL FEEDWATER

EVENT	TIME (SEC)
Main feedwater flow stops	10
Low-low steam generator water level trip setpoint reached	60.6
Rods begin to drop	62.6
Peak water level in the pressurizer occurs	63.0
Auxiliary feedwater reaches two of the four steam generators	120.6
Cold auxiliary feedwater is delivered to two of the steam generators	268.0
Core decay heat plus pump heat decreases to the auxiliary feedwater heat removal capacity	Approx. 3000

TABLE 3.7-2

TIME SEQUENCE OF EVENTS FOR A LOSS OF NONEMERGENCY AC POWER TO THE PLANT AUXILIARIES

(Loss of Offsite Power)

EVENT	TIME (SEC)
Main feedwater flow stops	10
Low-low steam generator water level trip setpoint reached	59.1
Rods begin to drop	61.1
Reactor coolant pumps begin to coastdown	63.1
Peak water level in the pressurizer occurs	64.0
Auxiliary feedwater reaches two of the four steam generators	119.1
Cold auxiliary feedwater is delivered to to two of the steam generators	268
Core decay heat decreases to the auxiliary feedwater heat removal capacity	Approx. 1600

TABLE 3.8.1

TIME SEQUENCE OF EVENTS FOR AN INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE

EVENT	TIME (SEC)
Pressurizer safety valve opens fully	0.0
Overtemperature ΔT reactor trip signal initiated	22.4
Rods begin to drop	24.4
Minimum DNBR occurs	25.0

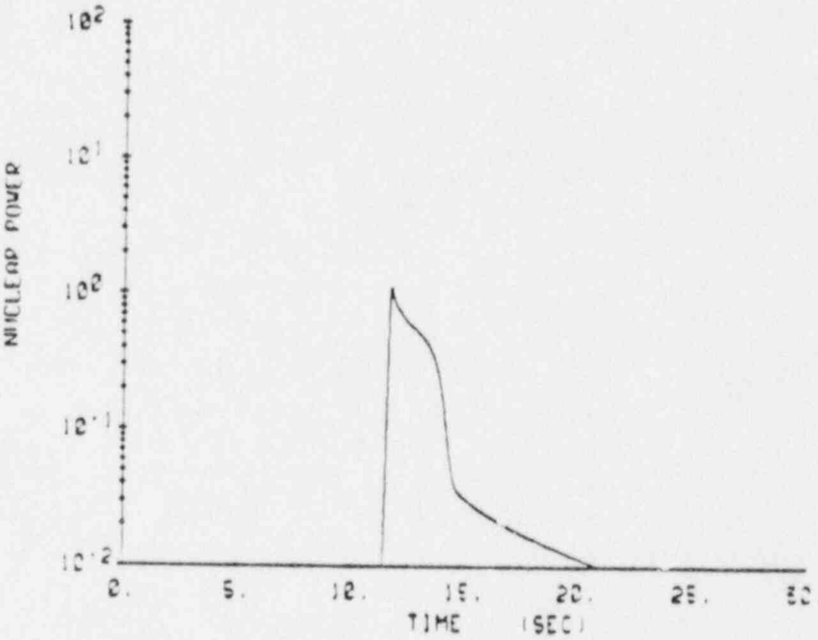


FIGURE 3.1-1
NEUTRON FLUX TRANSIENT FOR UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION

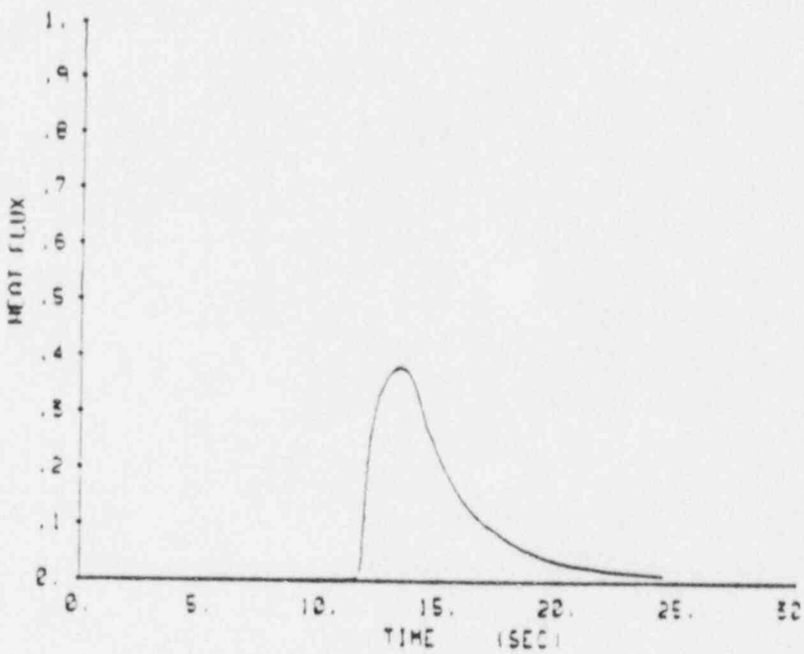
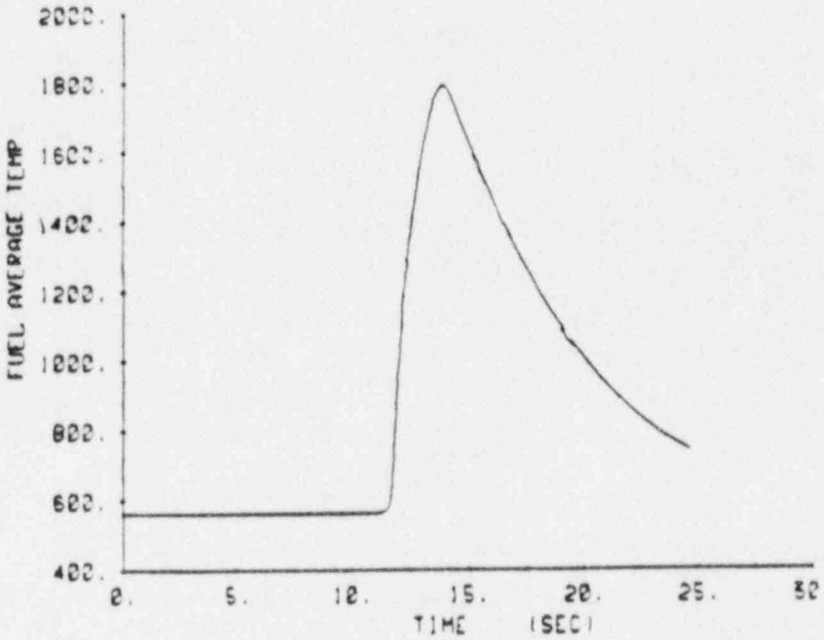


FIGURE 3.1-2
THERMAL FLUX TRANSIENT FOR
UNCONTROLLED ROD WITHDRAWAL
FROM A SUBCRITICAL CONDITION



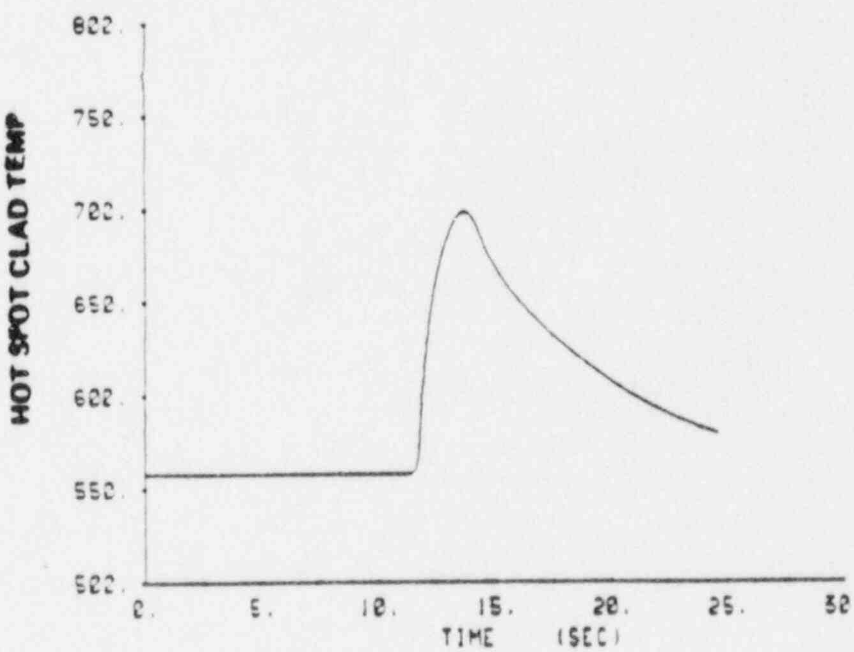


FIGURE 3.1-3
FUEL AND CLAD TEMPERATURE FOR UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION

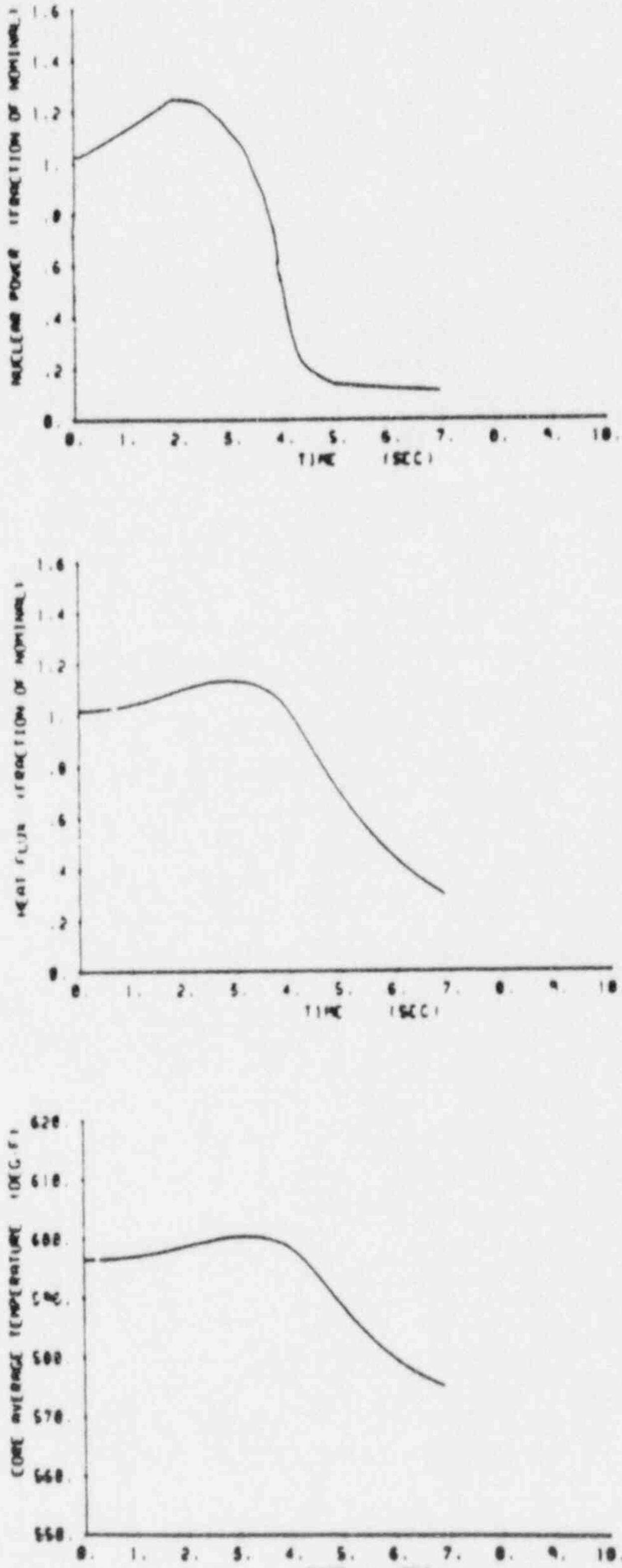


FIGURE 3.2-1
UNCONTROLLED RCCA BANK WITHDRAWAL
FROM FULL POWER WITH MINIMUM
REACTIVITY FEEDBACK
(80 PCM/SEC WITHDRAWAL RATE)

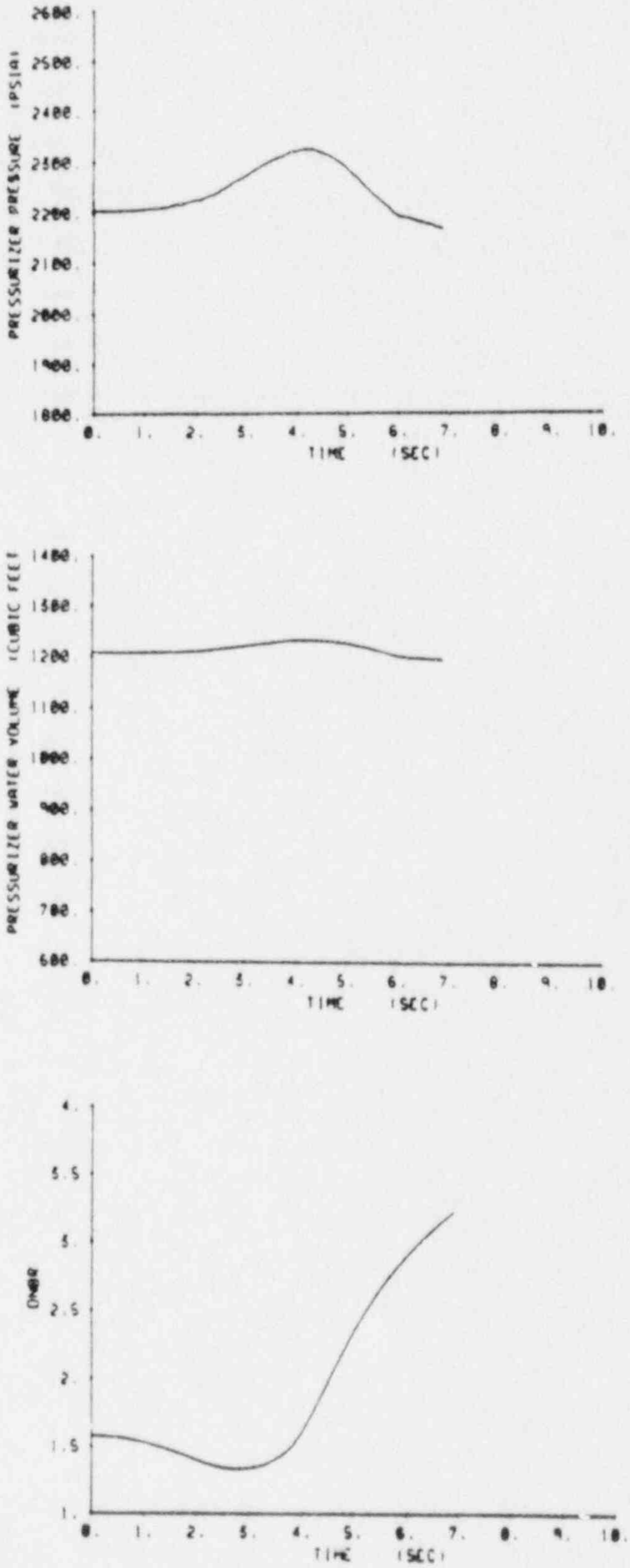


FIGURE 3.2-2
UNCONTROLLED RCCA BANK WITHDRAWAL FROM FULL POWER WITH MINIMUM REACTIVITY FEEDBACK (80 PCM/SEC WITHDRAWAL RATE)

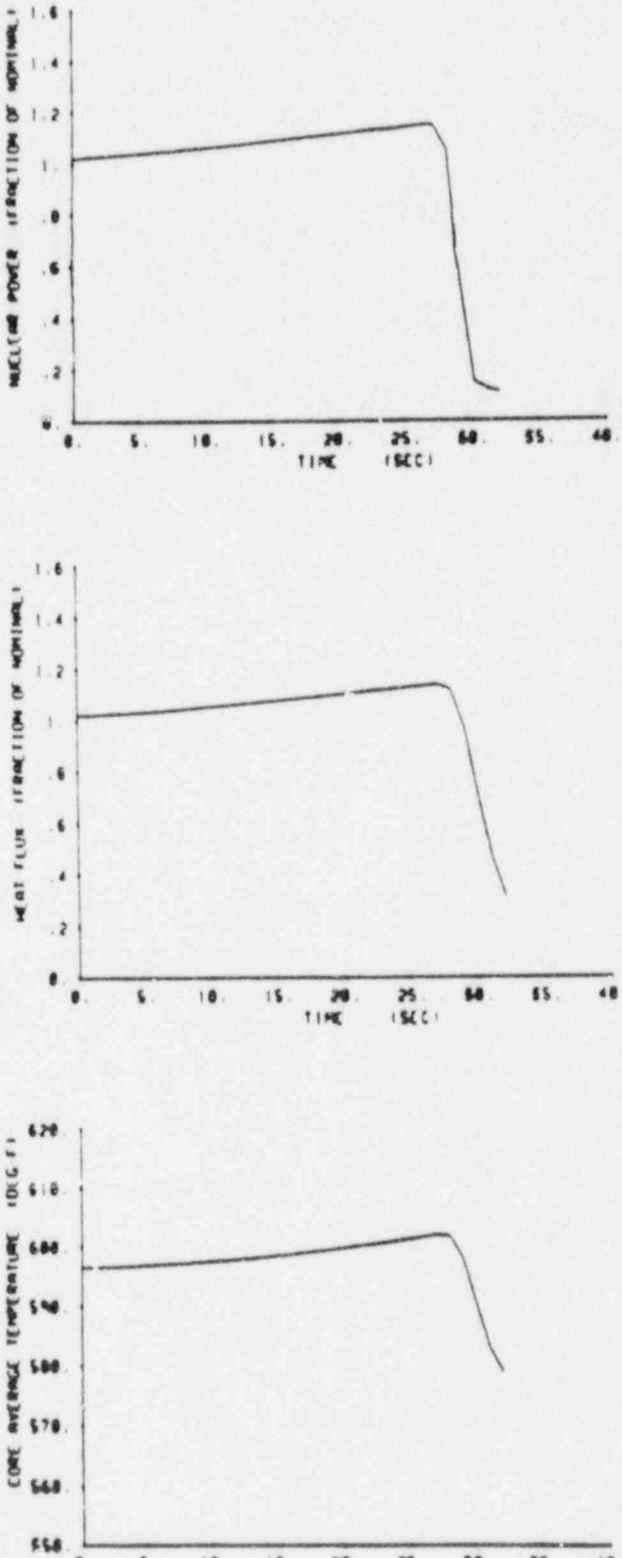


FIGURE 3.2-3
UNCONTROLLED RCCA BANK WITHDRAWAL
FROM FULL POWER WITH MINIMUM
REACTIVITY FEEDBACK
(3 PCM/SEC WITHDRAWAL RATE)

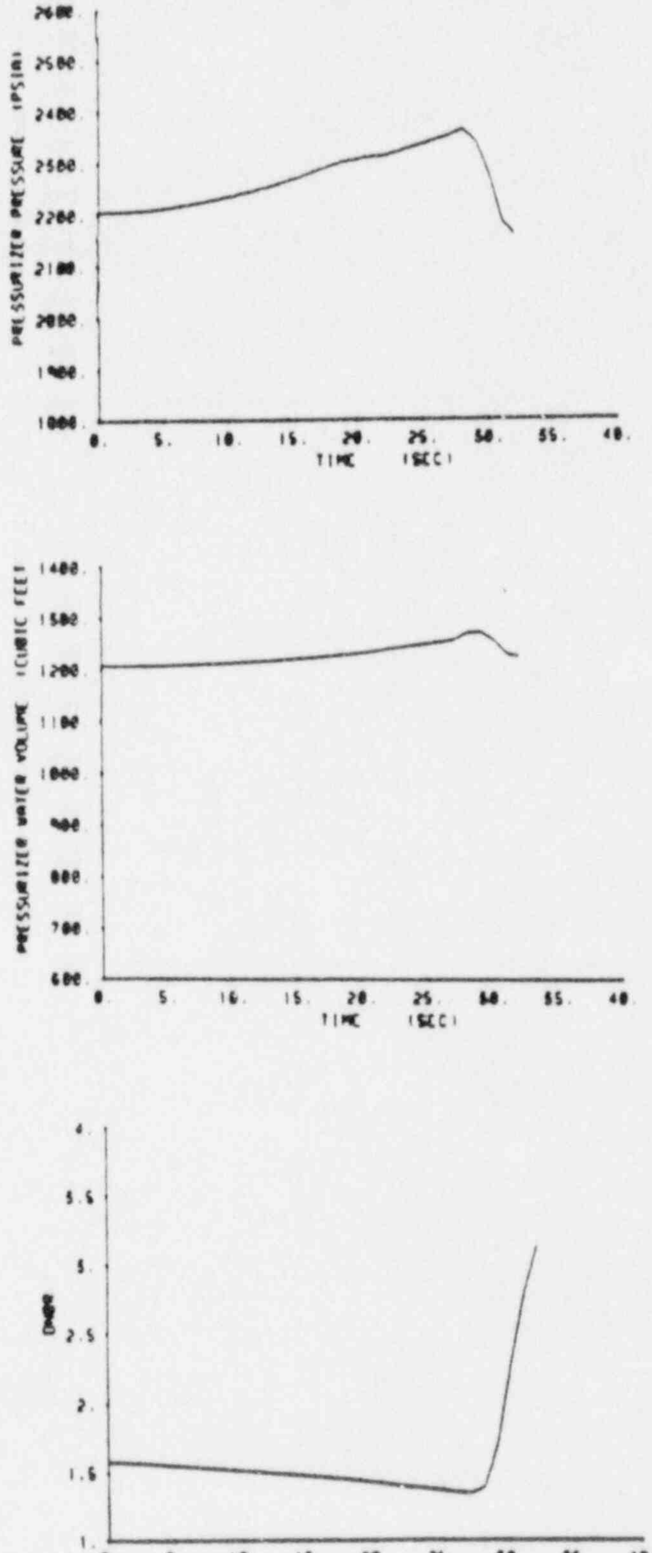


FIGURE 3.2-4
UNCONTROLLED RCCA BANK WITHDRAWAL
FROM FULL POWER WITH MINIMUM
REACTIVITY FEEDBACK
(3 PCM/SEC WITHDRAWAL RATE)

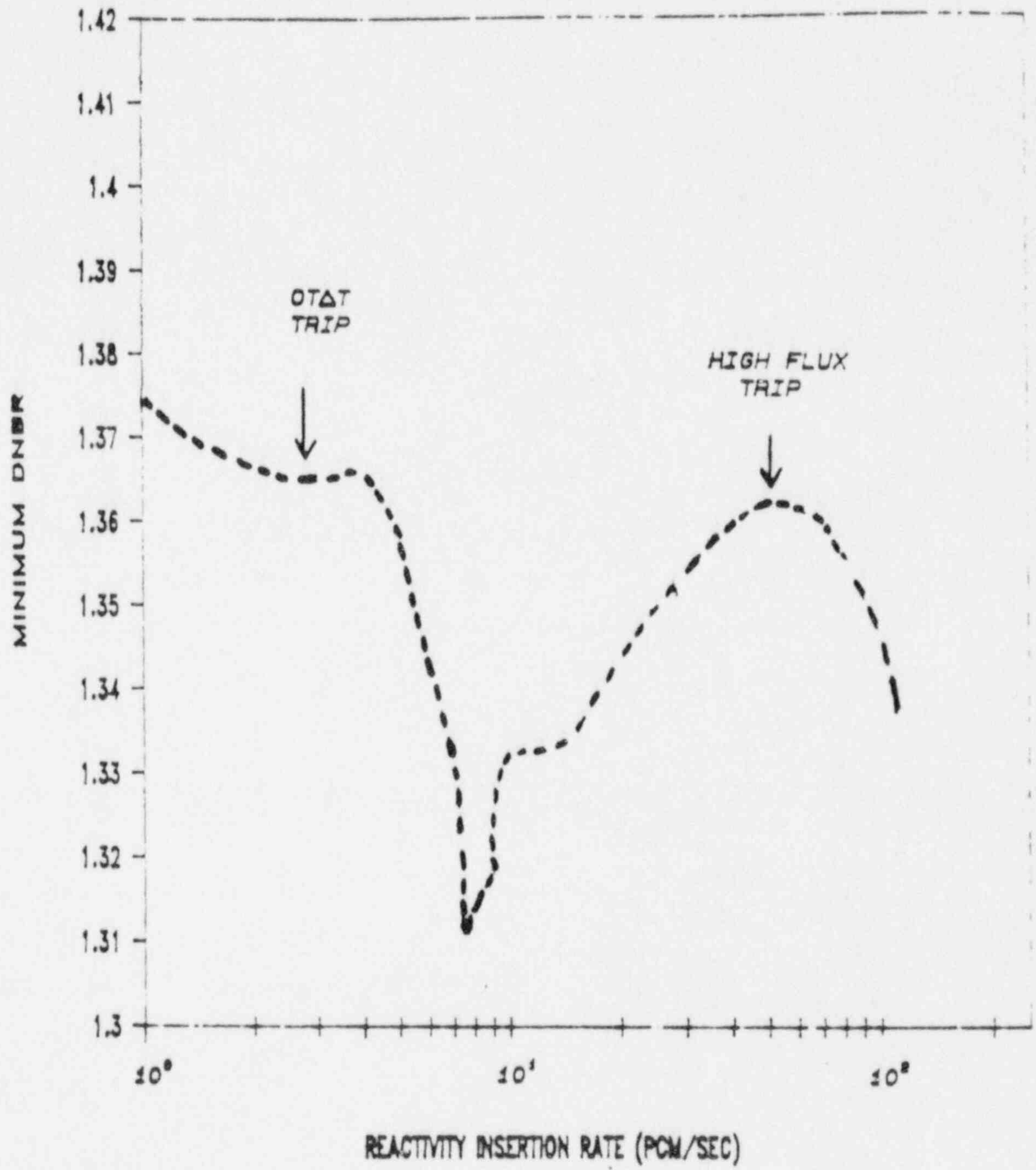


FIGURE 3.2-5
MINIMUM DNBR VS. REACTIVITY INSERTION RATE FOR ROD WITHDRAWAL FROM 100% POWER

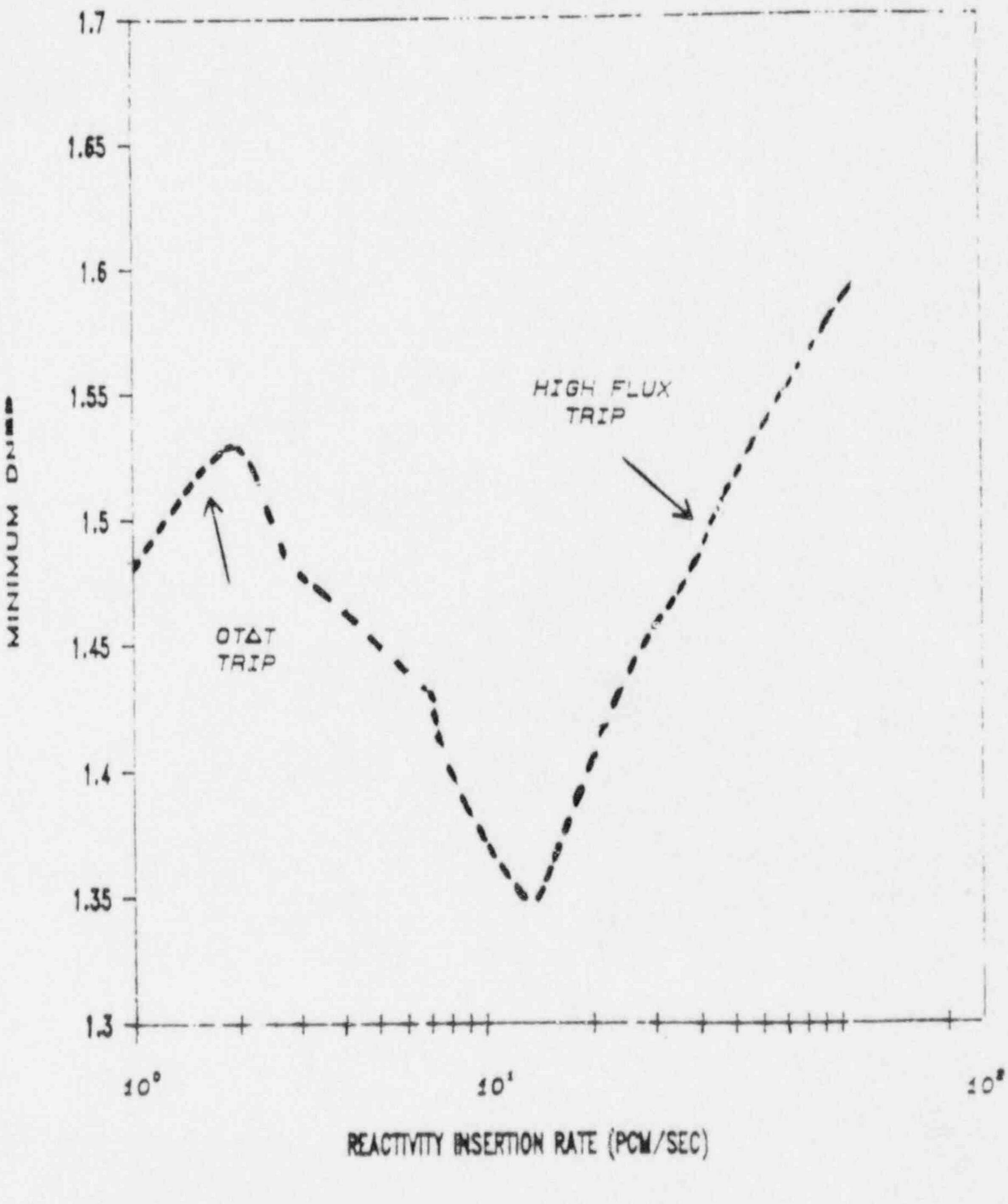


FIGURE 3.2-6
MINIMUM DNBR VS. REACTIVITY INSERTION RATE FOR ROD WITHDRAWAL FROM 60% POWER

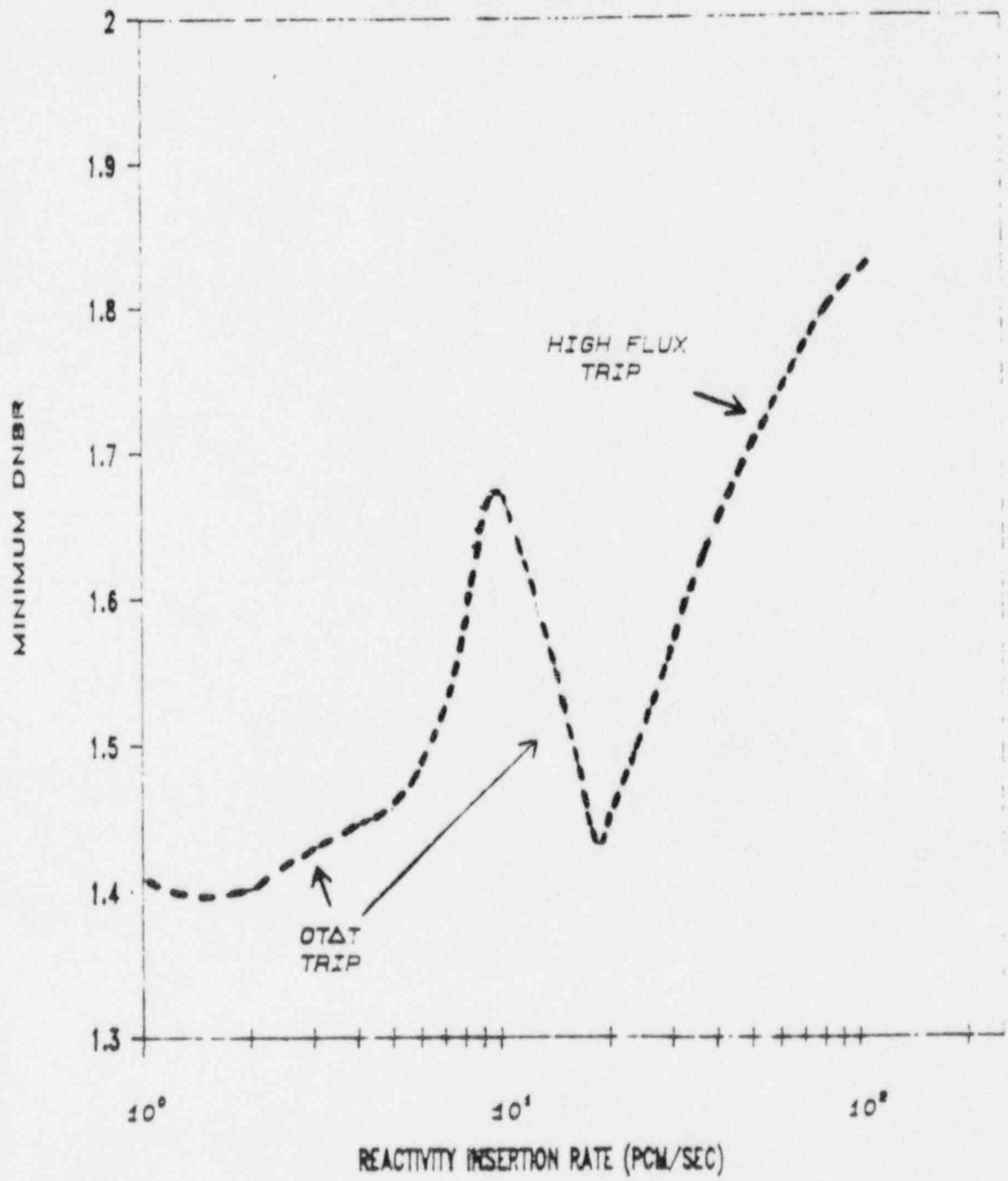
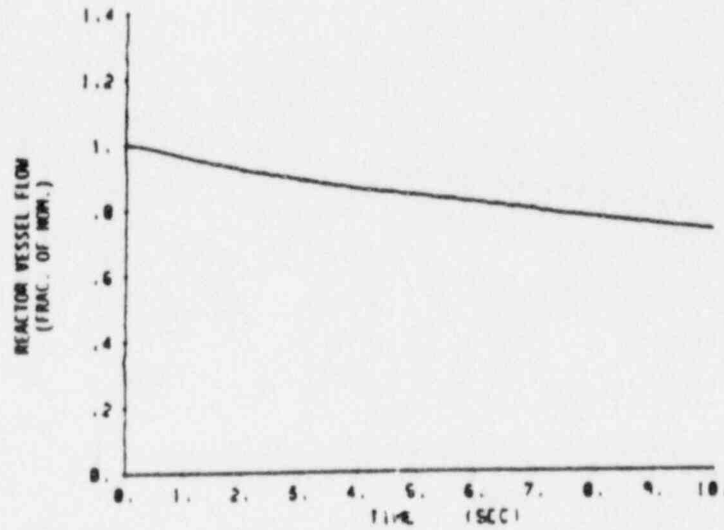


FIGURE 3.2-7
MINIMUM DNBR VS. REACTIVITY INSERTION RATE FOR ROD WITHDRAWAL FROM 10% POWER



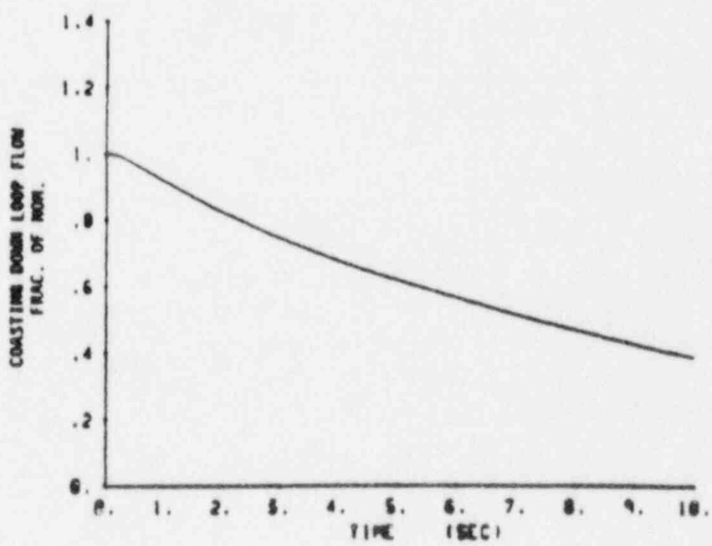
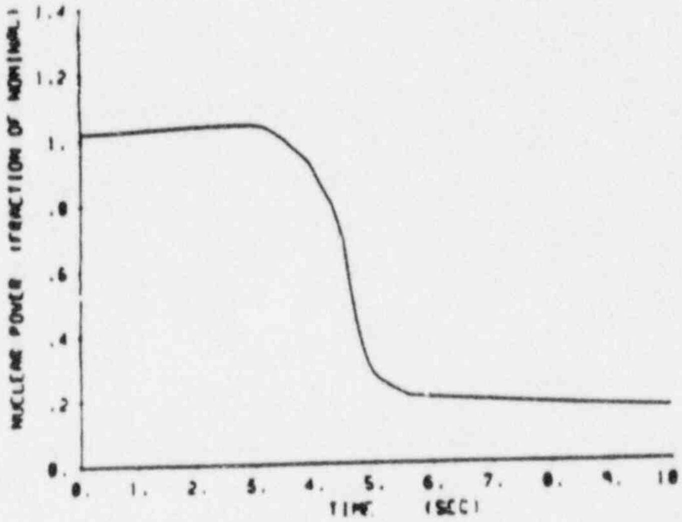


FIGURE 3.3-1
FLOW TRANSIENTS FOR 4 LOOPS IN
OPERATION, 2 PUMPS COASTING DOWN



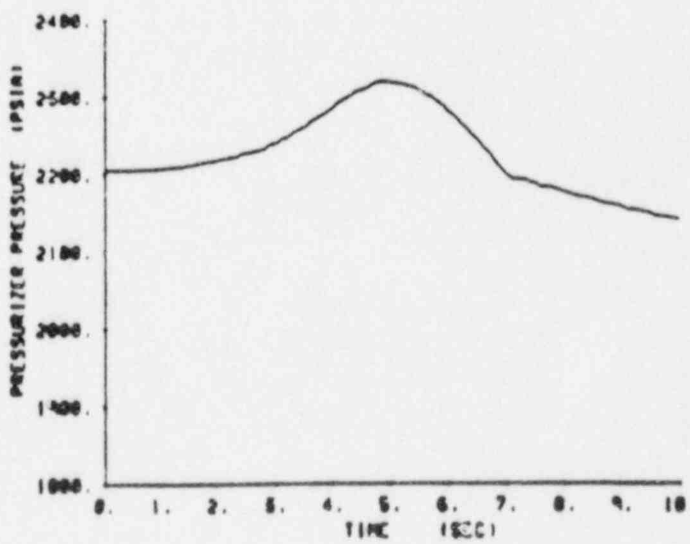


FIGURE 3.3-2
NUCLEAR POWER AND PRESSURIZER PRESSURE
TRANSIENTS FOR 4 LOOPS IN OPERATION,
2 PUMPS COASTING DOWN

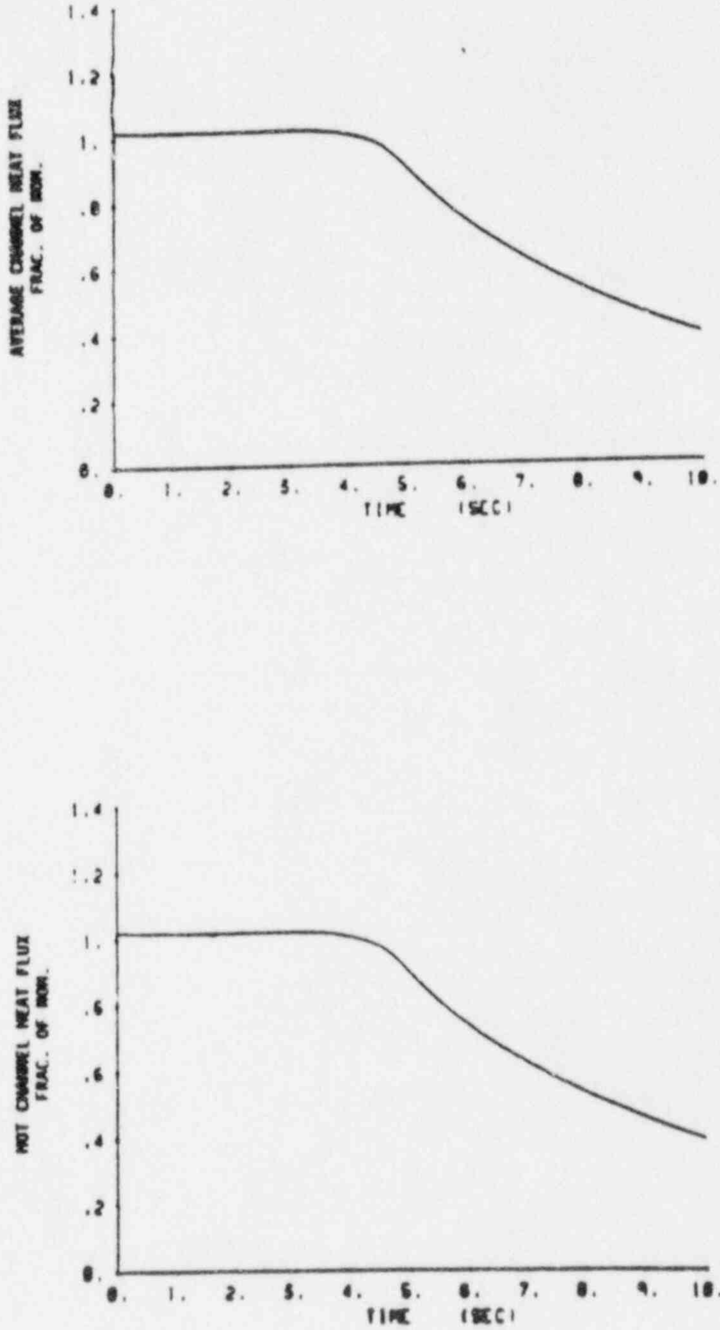


FIGURE 3.3-3
AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR 4 LOOPS IN OPERATION, 2 PUMPS COASTING DOWN

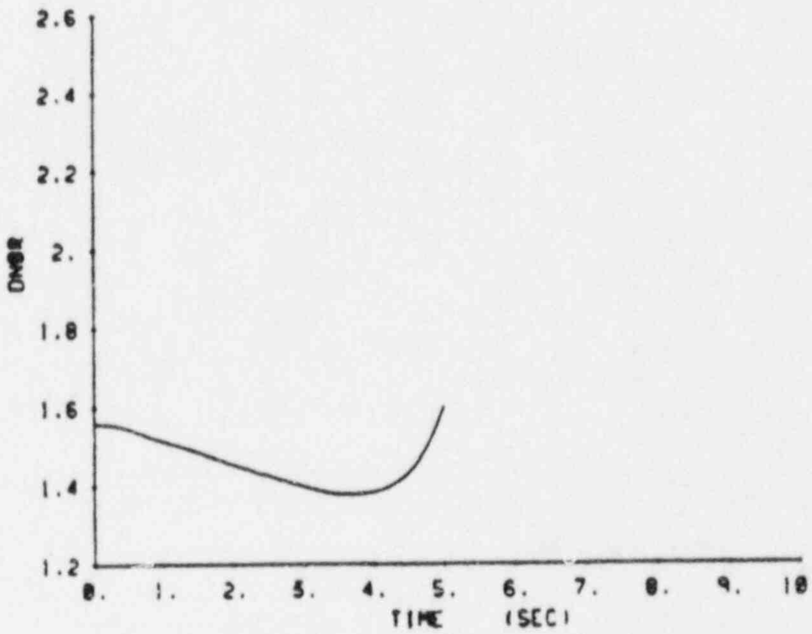


FIGURE 3.3-4
DNBR VS. TIME FOR 4 LOOPS IN OPERATION, 2 PUMPS COASTING DOWN

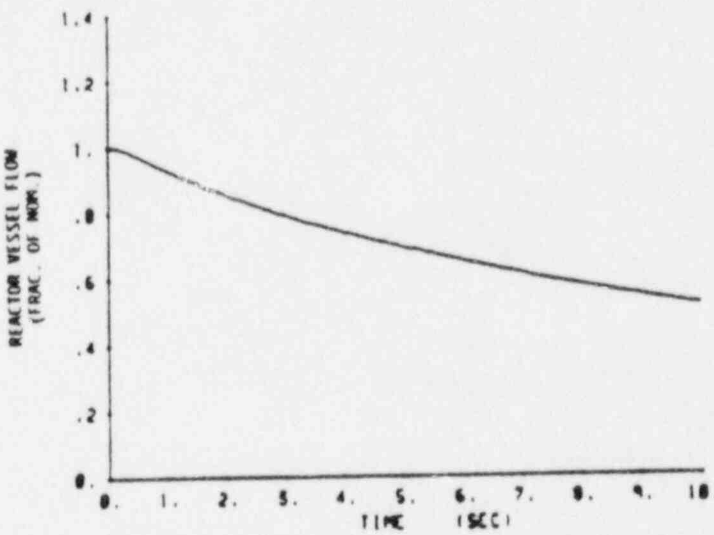
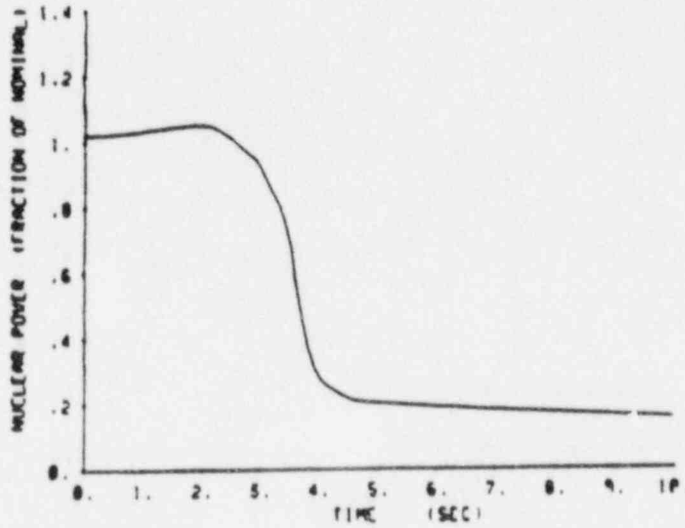


FIGURE 3.3-5
FLOW TRANSIENT FOR 4 LOOPS IN
OPERATION, 4 PUMPS COASTING DOWN



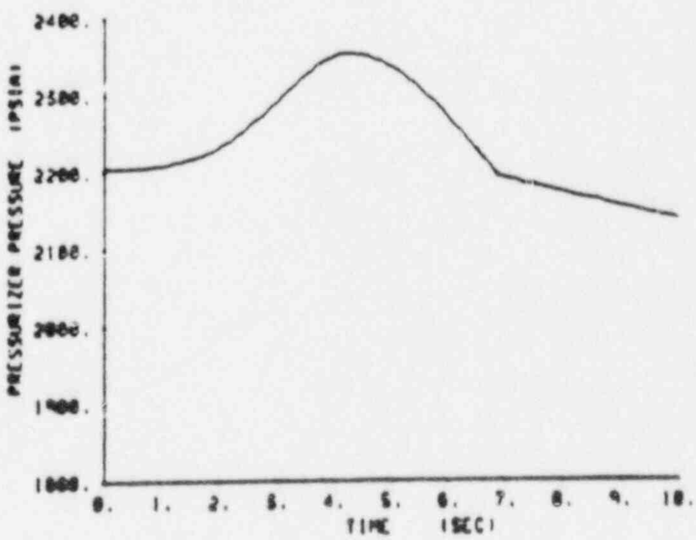


FIGURE 3.3-6
NUCLEAR POWER AND PRESSURIZER PRESSURE
TRANSIENTS FOR 4 LOOPS IN OPERATION,
4 PUMPS COASTING DOWN

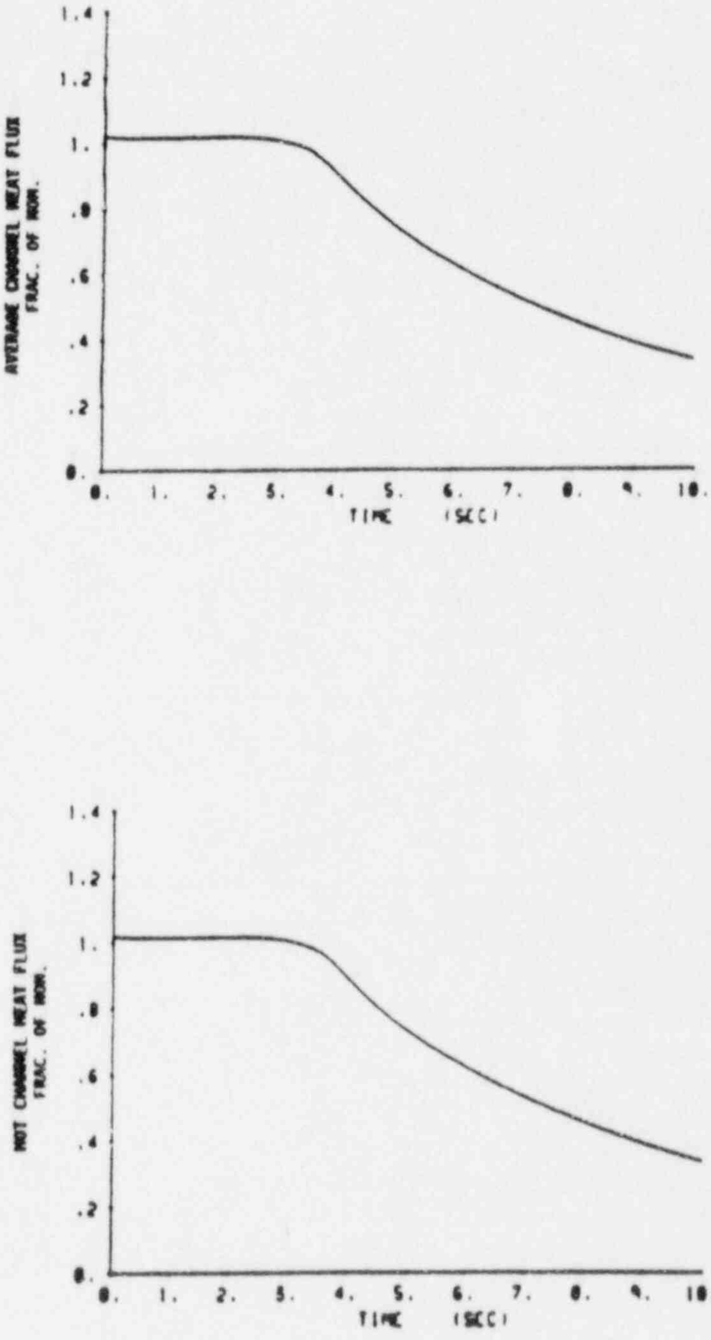


FIGURE 3.3-7
AVERAGE AND HOT CHANNEL HEAT FLUX
TRANSIENTS FOR 4 LOOPS IN OPERATION,
4 PUMPS COASTING DOWN

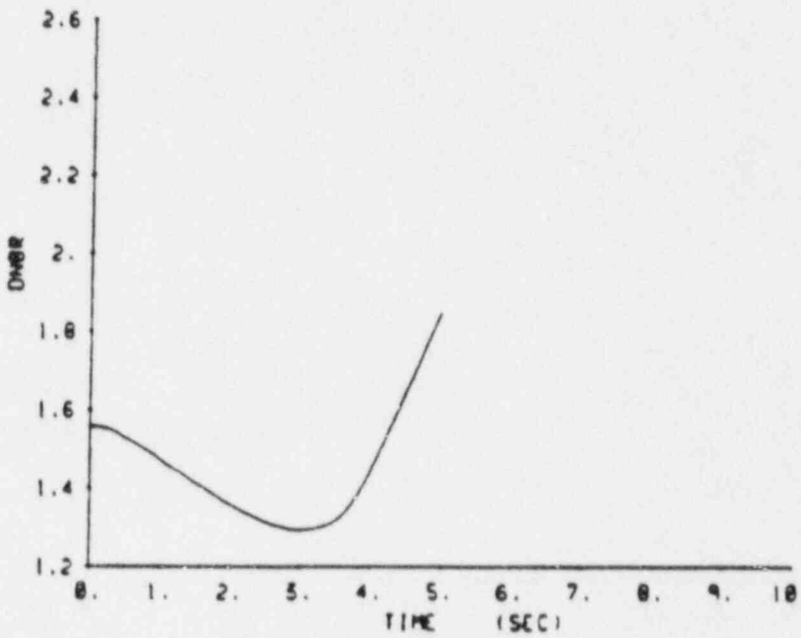
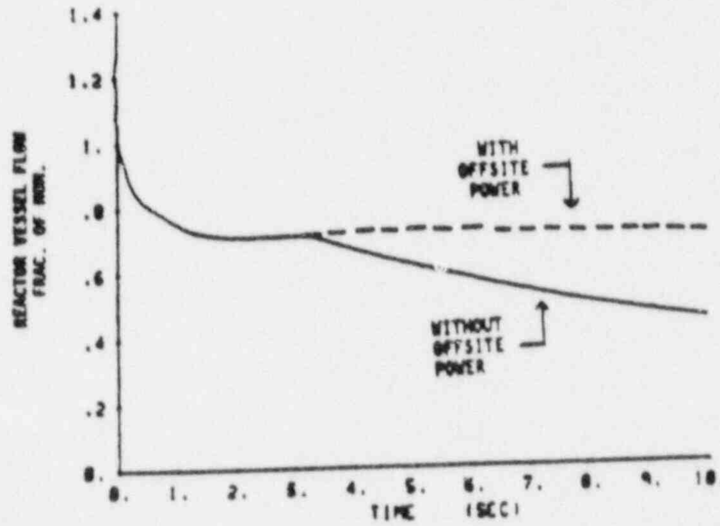


FIGURE 3.3-8
DNBR VS. TIME FOR 4 LOOPS IN OPERATION, 4 PUMPS COASTING DOWN



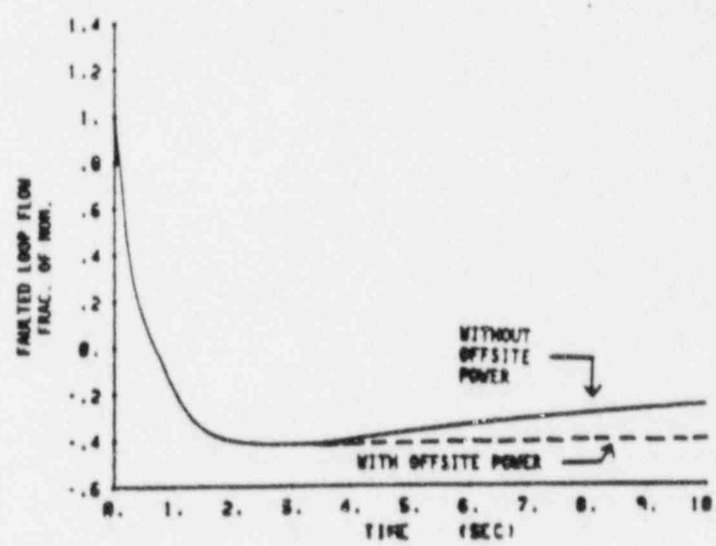


FIGURE 3.4-1
FLOW TRANSIENTS FOR 4 LOOPS IN
OPERATION, 1 LOCKED ROTOR

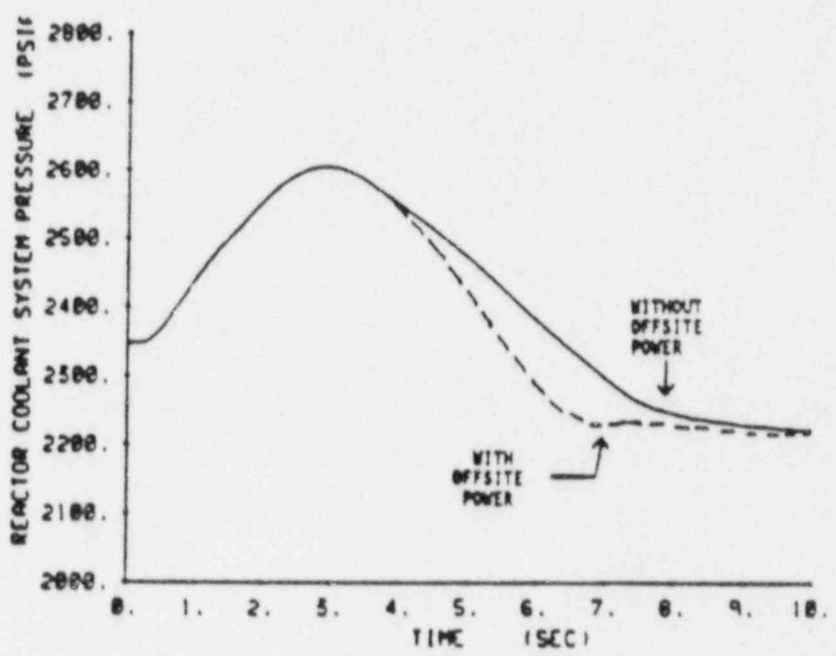
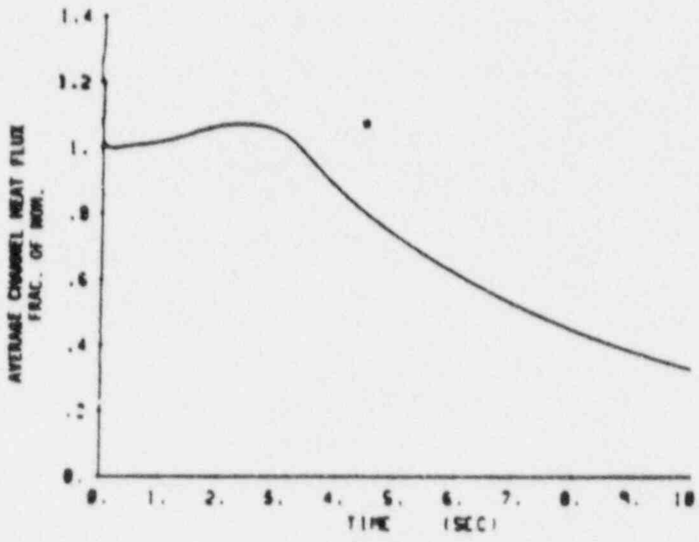
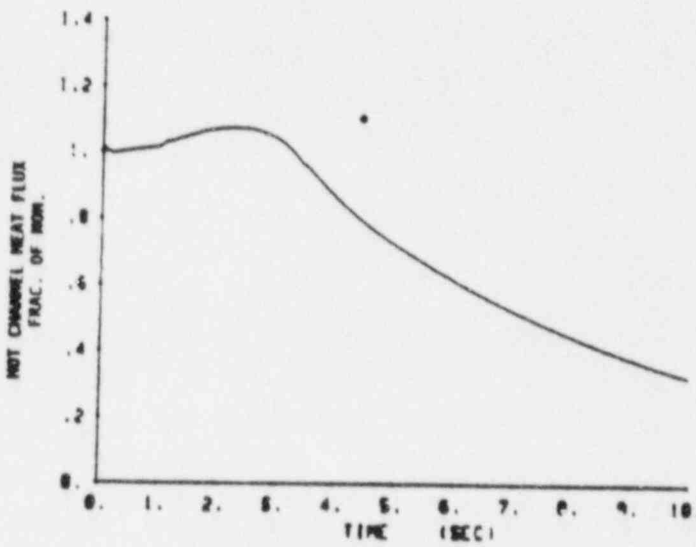


FIGURE 3.4-2
PEAK REACTOR COOLANT PRESSURE FOR
4 LOOPS IN OPERATION, 1 LOCKED ROTOR

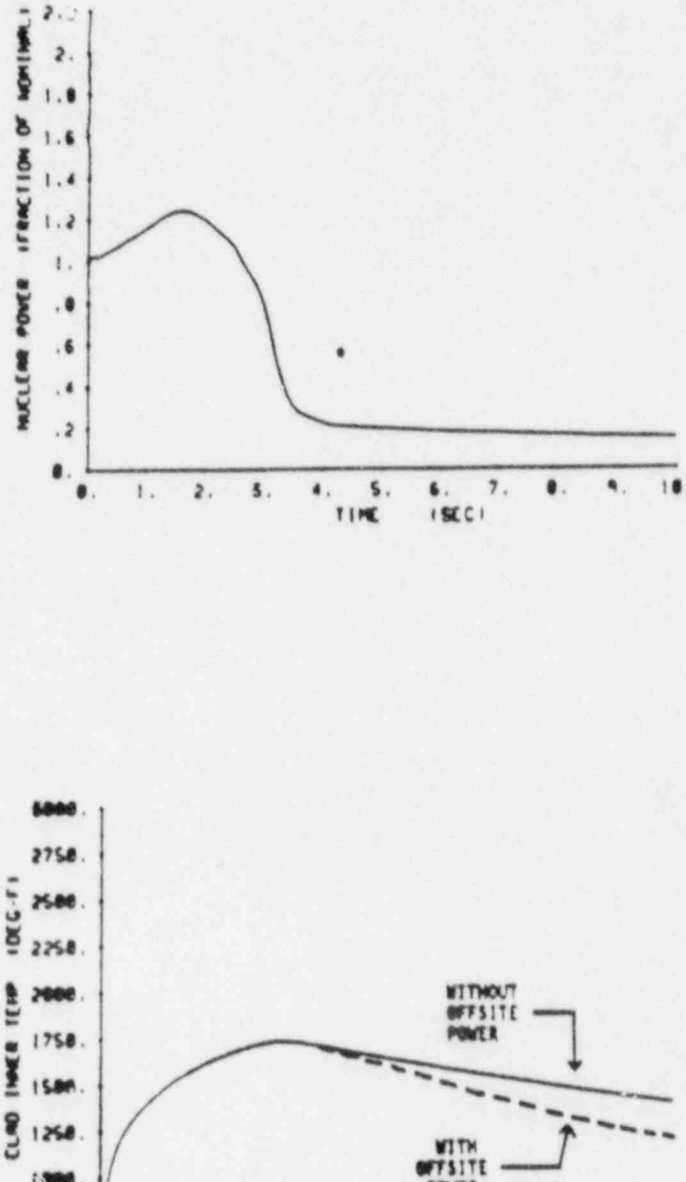




NOTE:

* WITH AND WITHOUT OFFSITE POWER AVAILABLE

FIGURE 3.4-3
AVERAGE AND HOT CHANNEL HEAT FLUX
TRANSIENTS FOR 4 LOOPS IN OPERATION,
1 LOCKED ROTOR



NOTE:

* WITH AND WITHOUT OFFSITE POWER AVAILABLE

FIGURE 3.4-4
NUCLEAR POWER AND MAXIMUM CLAD TEMPERATURE AT HOT SPOT TRANSIENTS FOR 4 LOOPS IN OPERATION, 1 LOCKED ROTOR

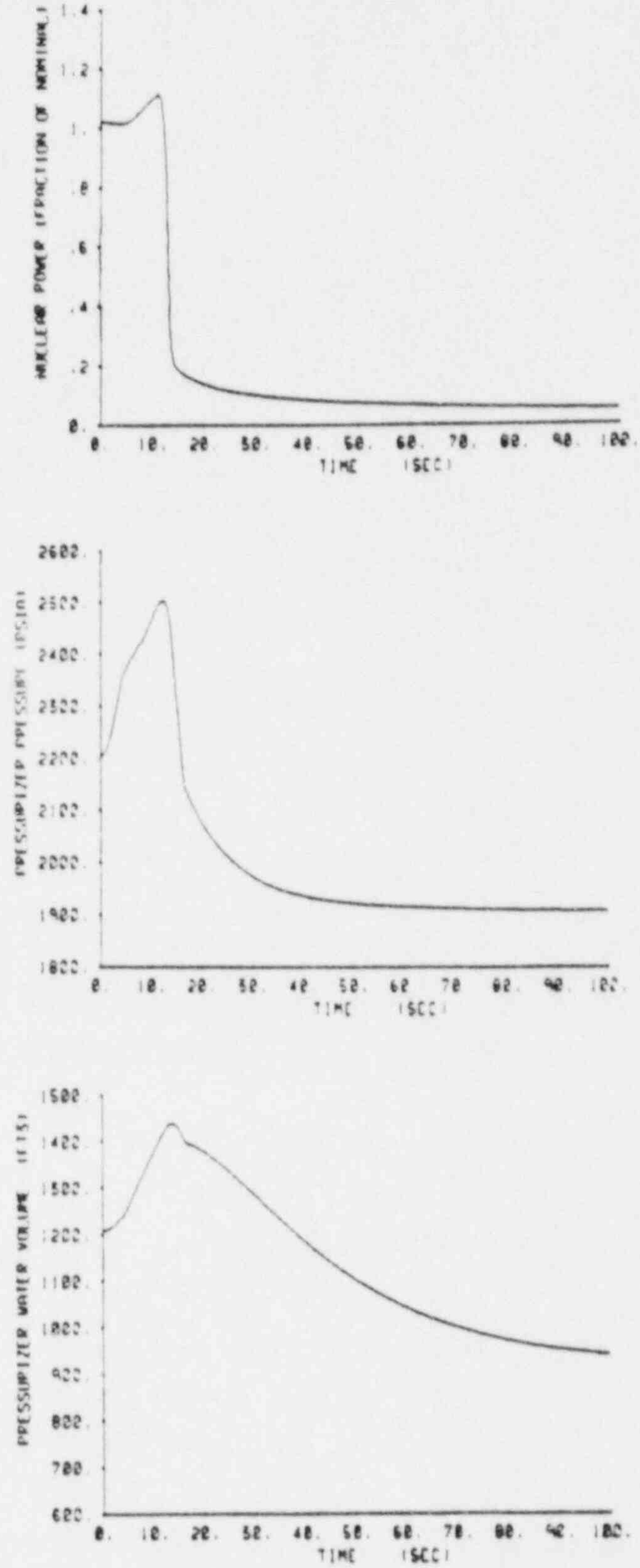


FIGURE 3.5-1
TURBINE TRIP WITH PRESSURIZER
SPRAY AND POWER-OPERATED RELIEF
VALVES, MIN. MODERATOR FEEDBACK

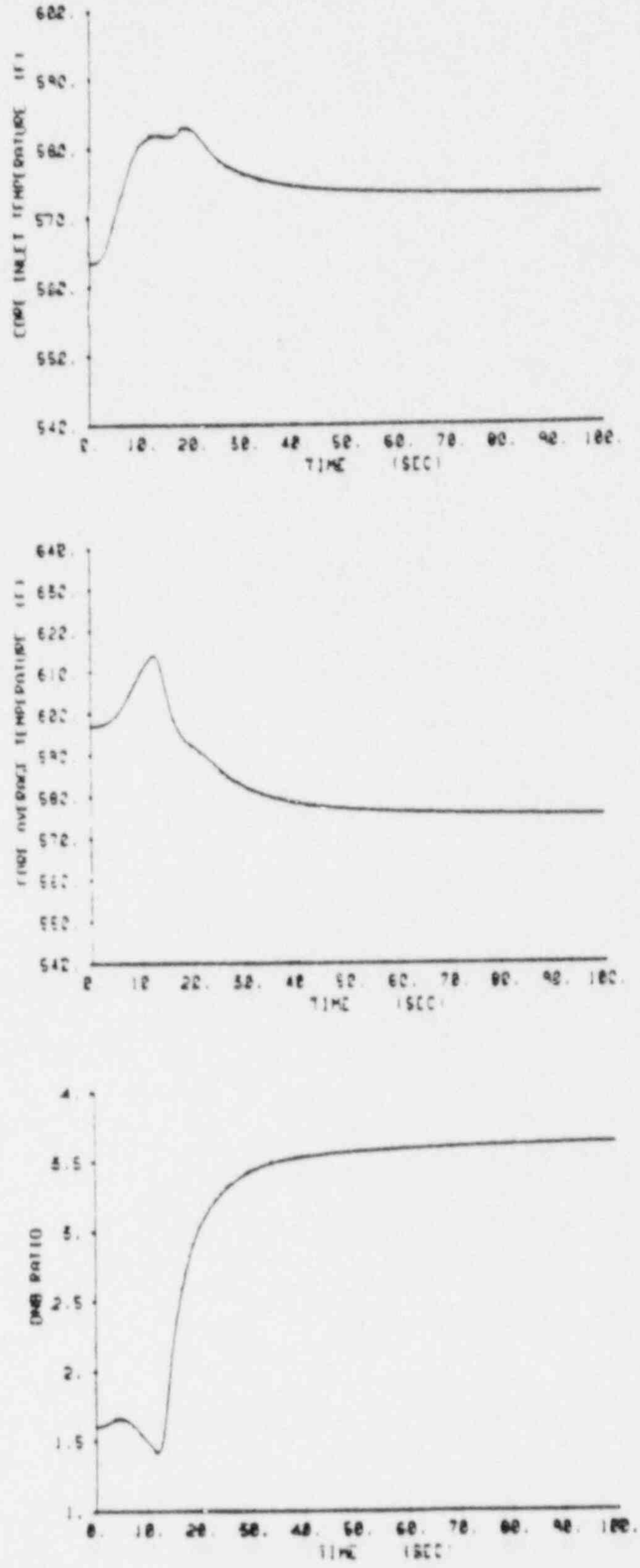


FIGURE 3.5-2
TURBINE TRIP WITH PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MIN. MODERATOR FEEDBACK

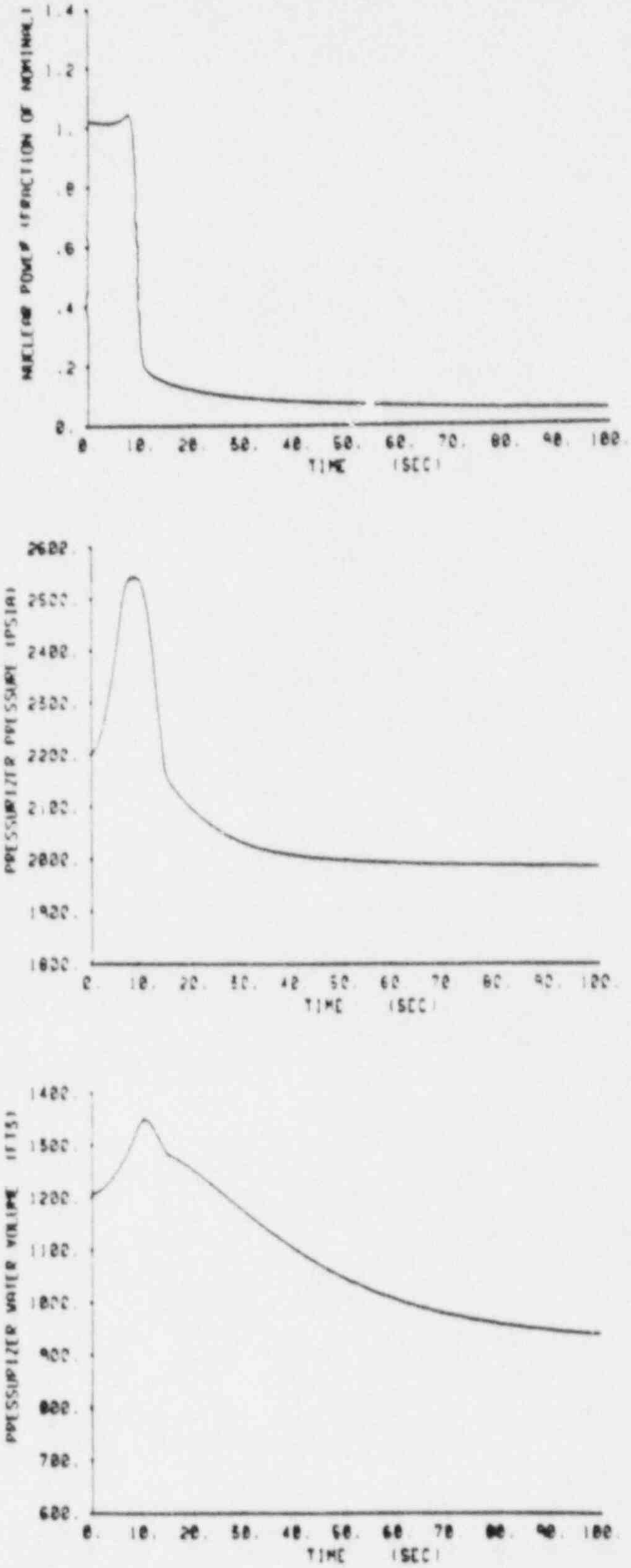


FIGURE 3.5-3
TURBINE TRIP WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MIN. MODERATOR FEEDBACK

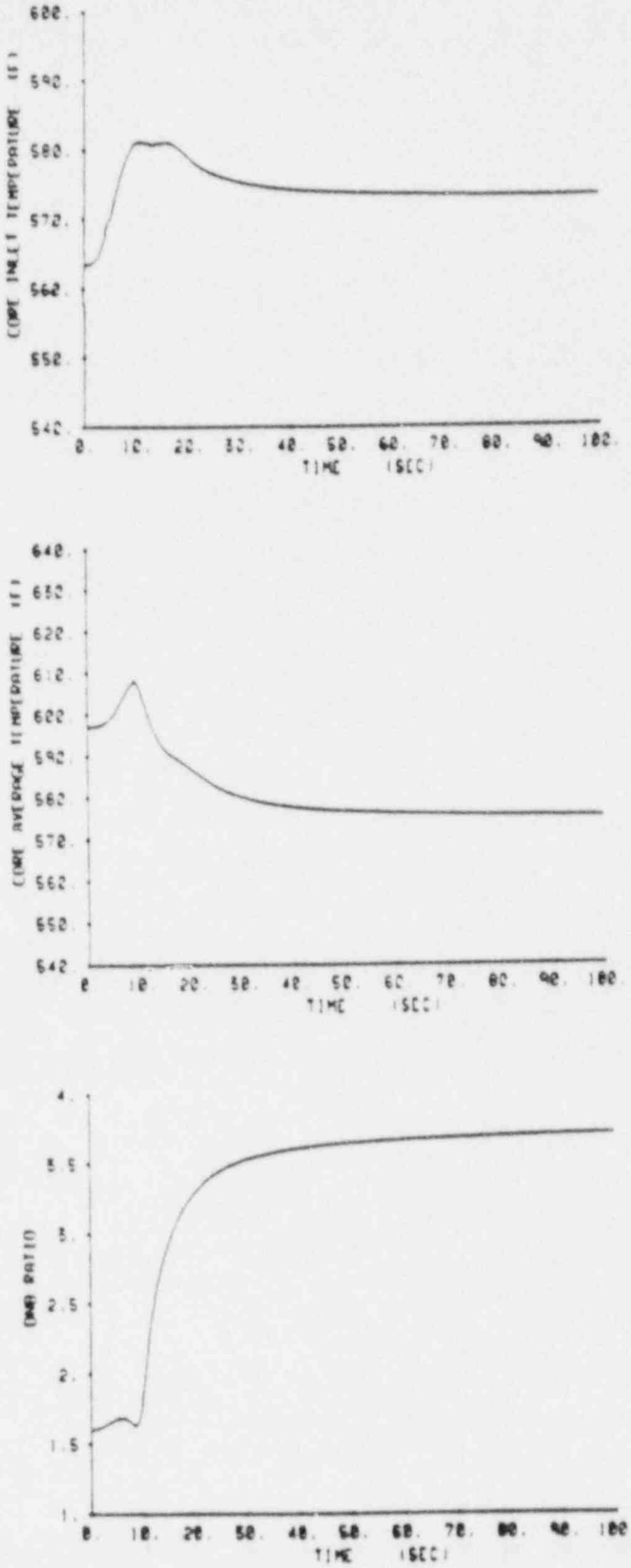


FIGURE 3.5-4
TURBINE TRIP WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES, MIN. MODERATOR FEEDBACK

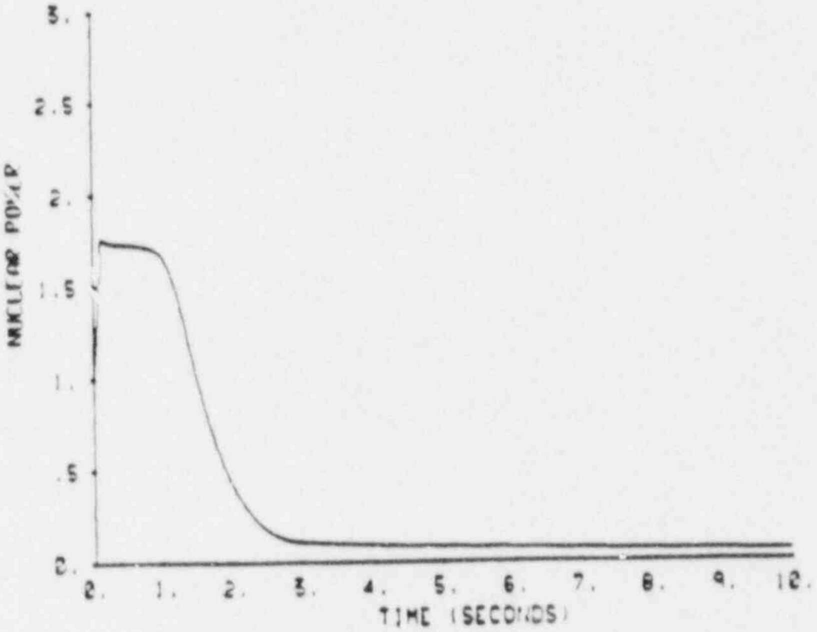


FIGURE 3.6-1
NUCLEAR POWER TRANSIENT FOR
BOL HFP RCCA EJECTION ACCIDENT

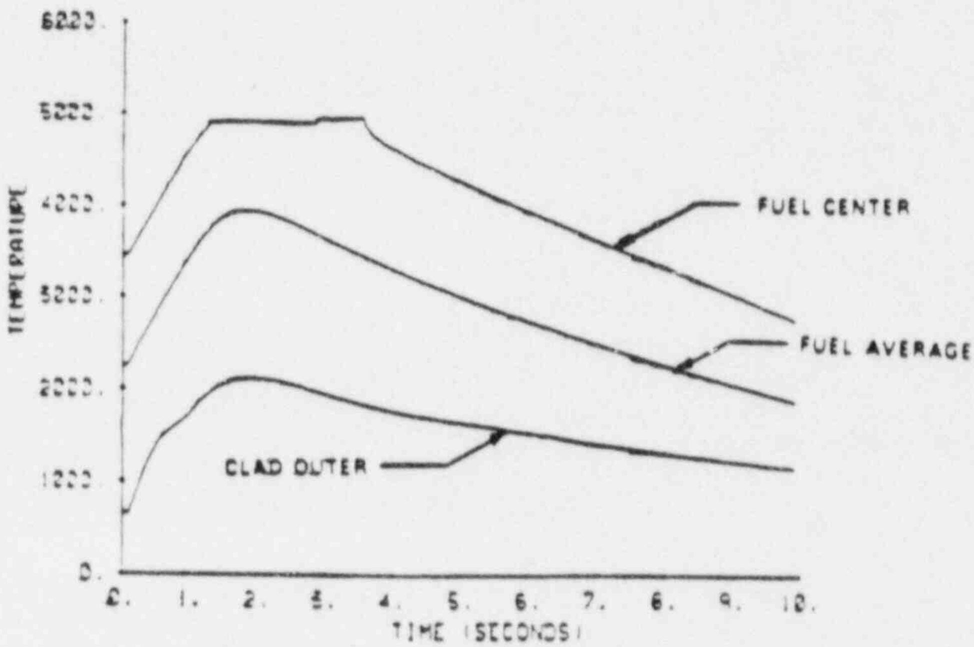


FIGURE 3.6-2
HOT SPOT FUEL & CLAD TEMPERATURE
VS. TIME FOR BOL HFP RCCA EJECTION
ACCIDIENT

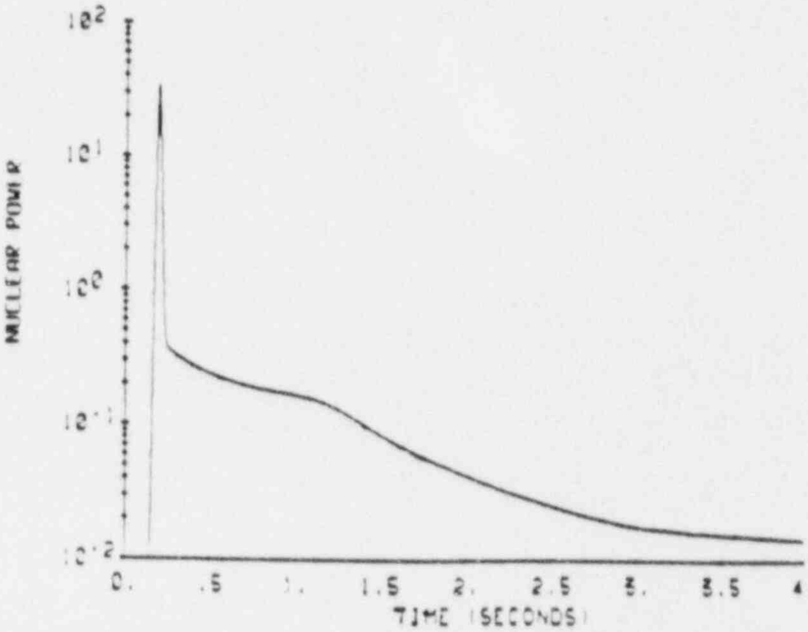


FIGURE 3.6-3
NUCLEAR POWER TRANSIENT FOR
BOL HZP RCCA EJECTION ACCIDENT

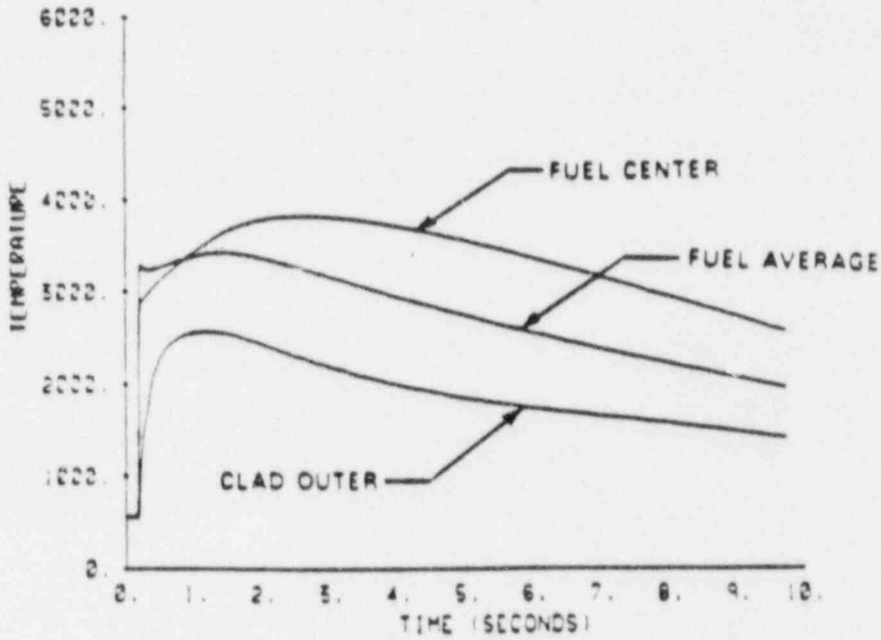


FIGURE 3.6-4
HOT SPOT FUEL & CLAD TEMPERATURE
VS. TIME FOR BOL HZP RCCA EJECTION
ACCIDENT

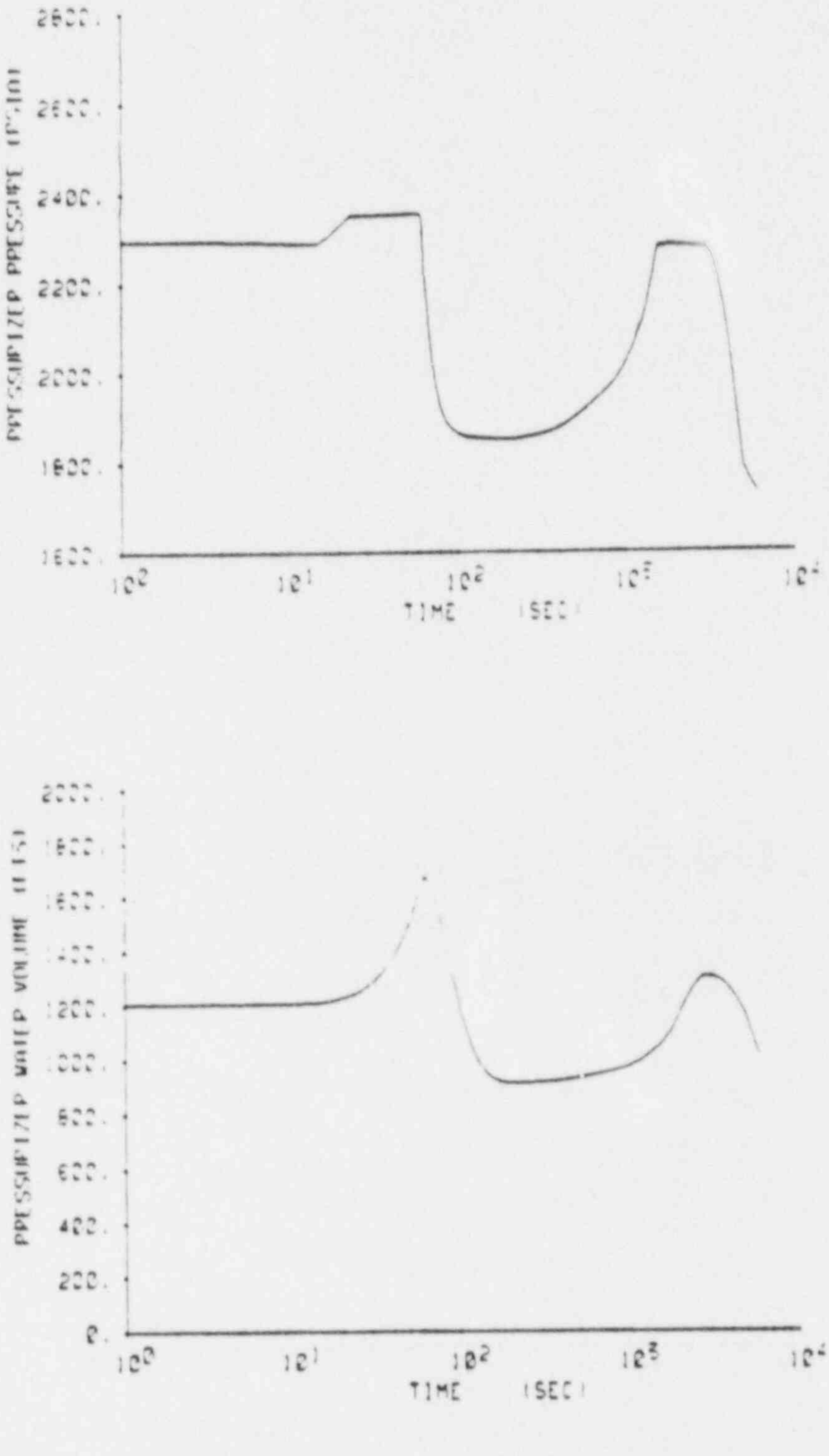
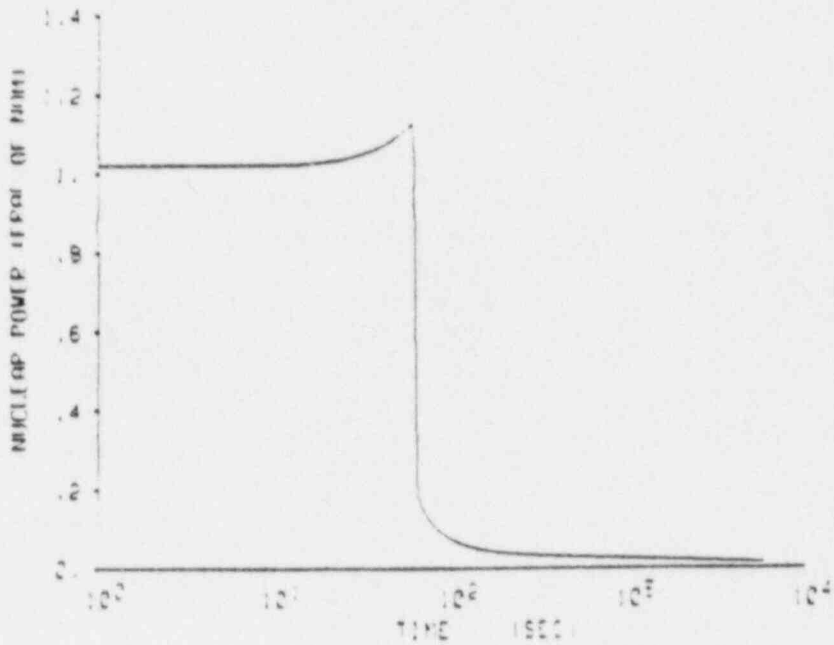


FIGURE3.7-1
PRESSURIZER PRESSURE AND WATER VOLUME
TRANSIENTS FOR LOSS OF NORMAL FEEDWATER



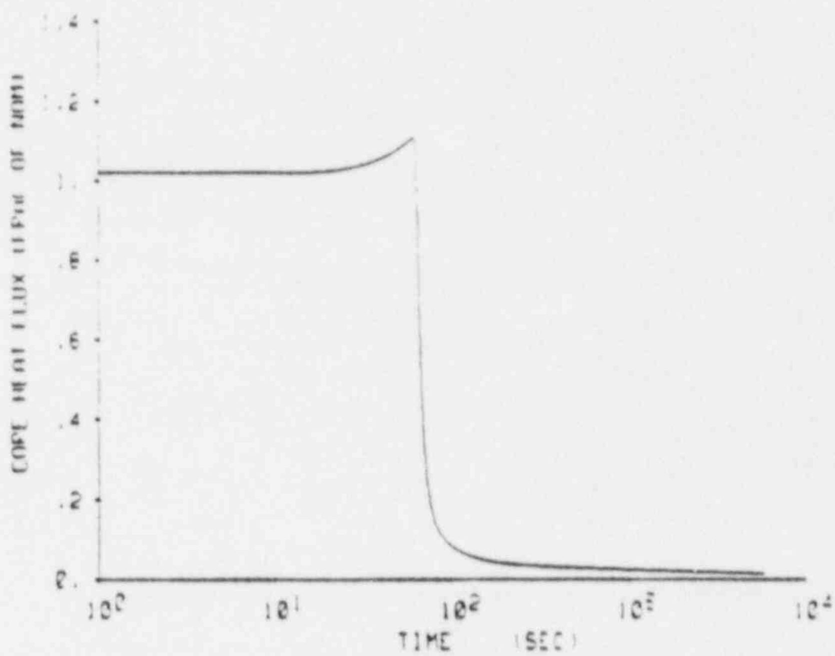


FIGURE 3.7-2
NUCLEAR POWER AND CORE HEAT
FLUX FOR LOSS OF NORMAL FEEDWATER

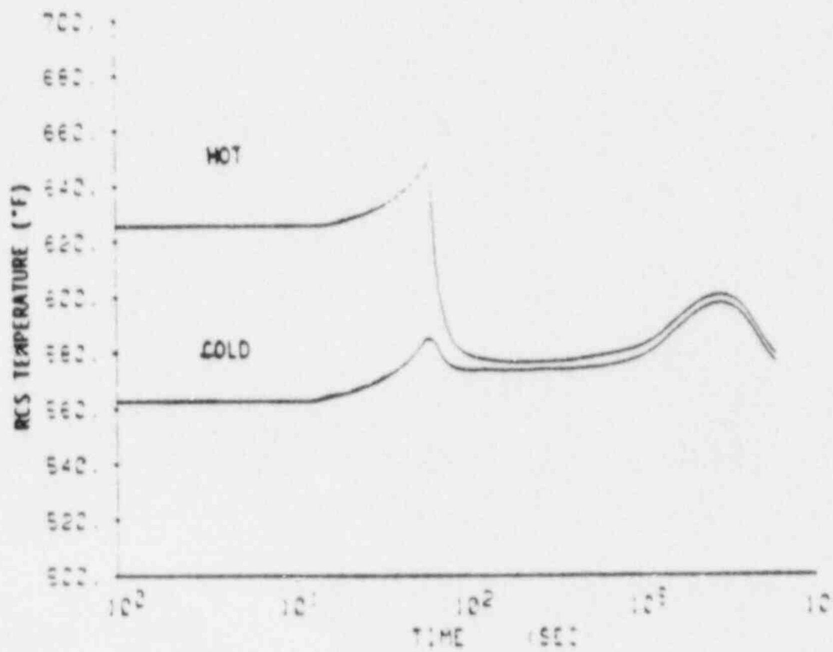
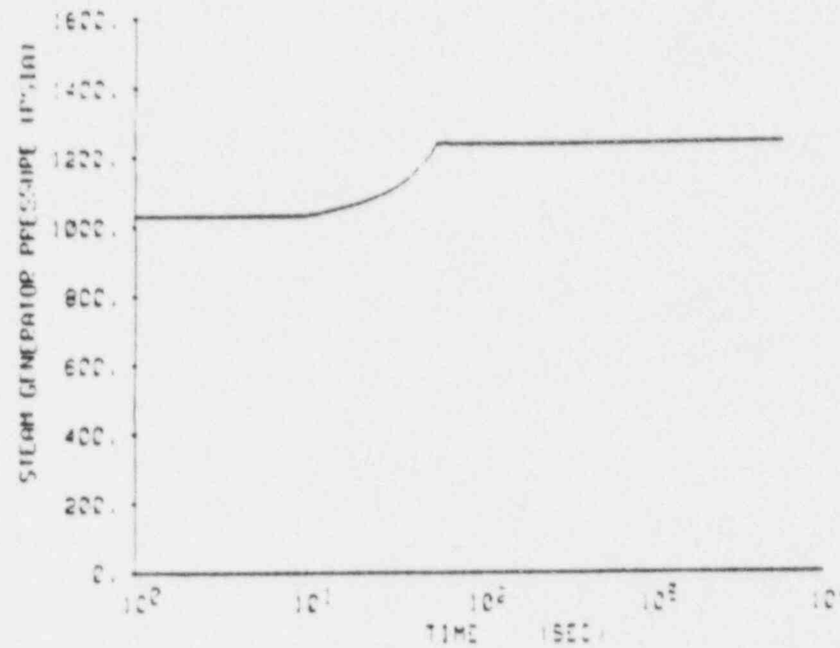


FIGURE 3.7-3
LOOP TEMPERATURE FOR LOSS
OF NORMAL FEEDWATER



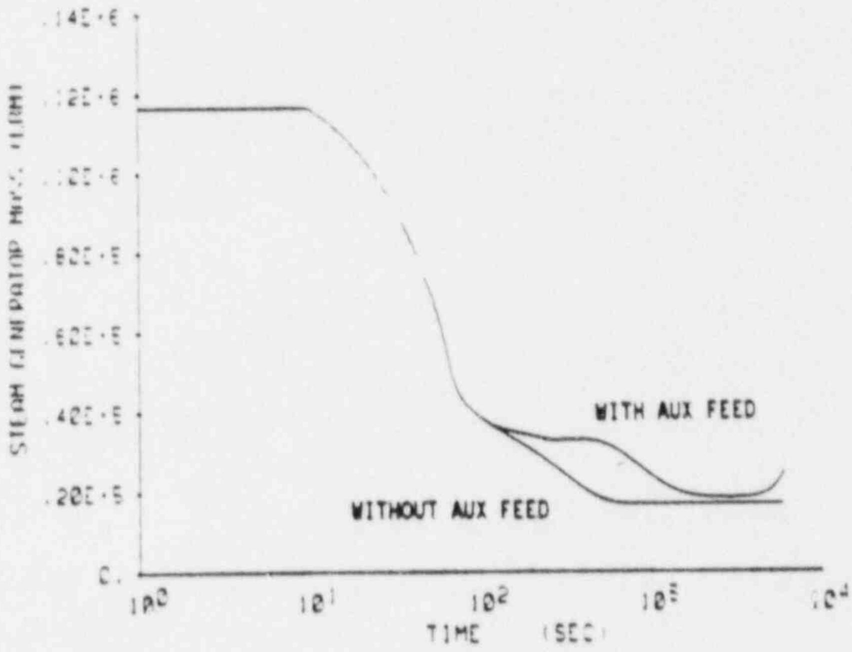
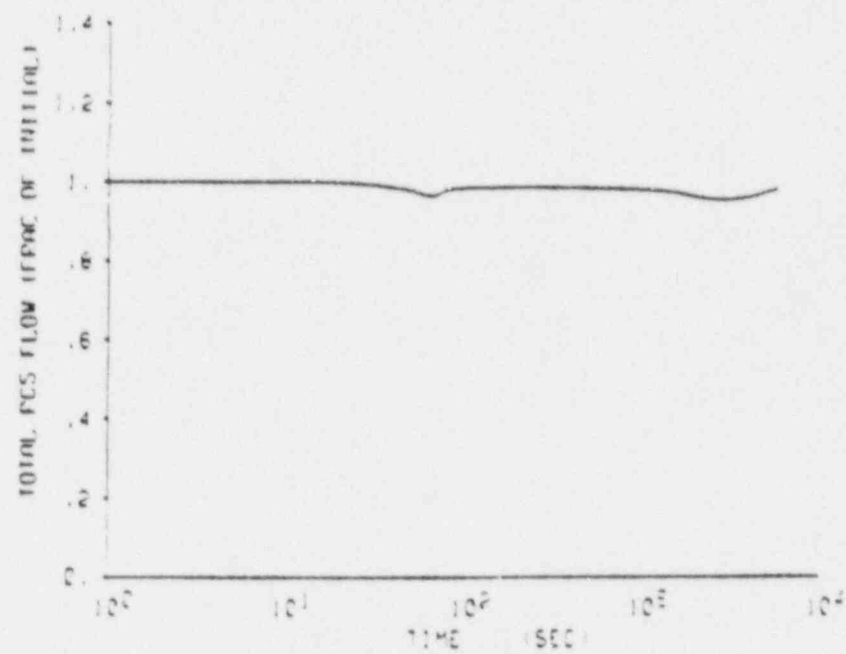


FIGURE 3.7-4
STEAM GENERATOR PRESSURE AND MASS
FOR LOSS OF NORMAL FEEDWATER



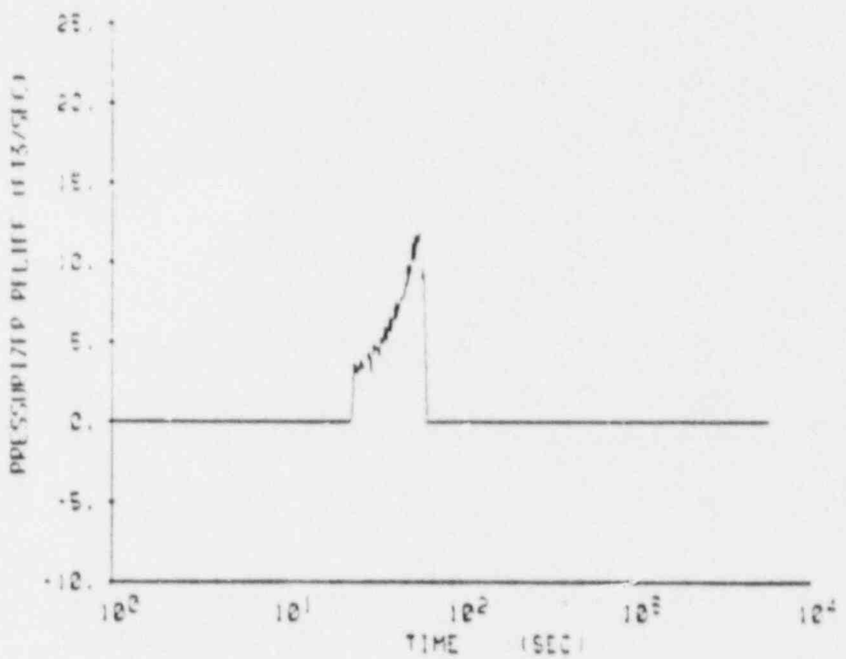


FIGURE 3.7-5
RCS FLOW AND PRESSURIZER RELIEF
FOR LOSS OF NORMAL FEEDWATER

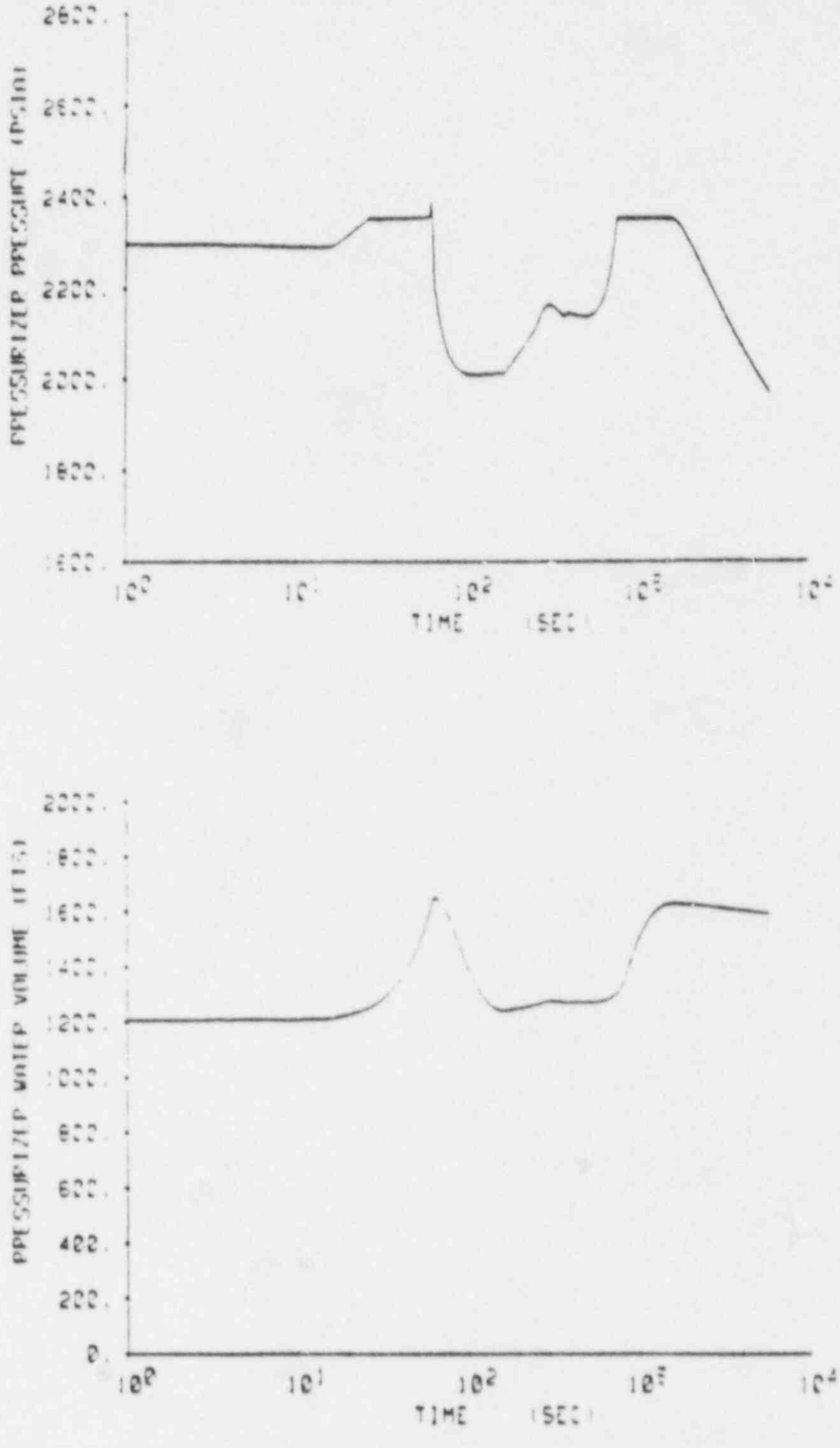
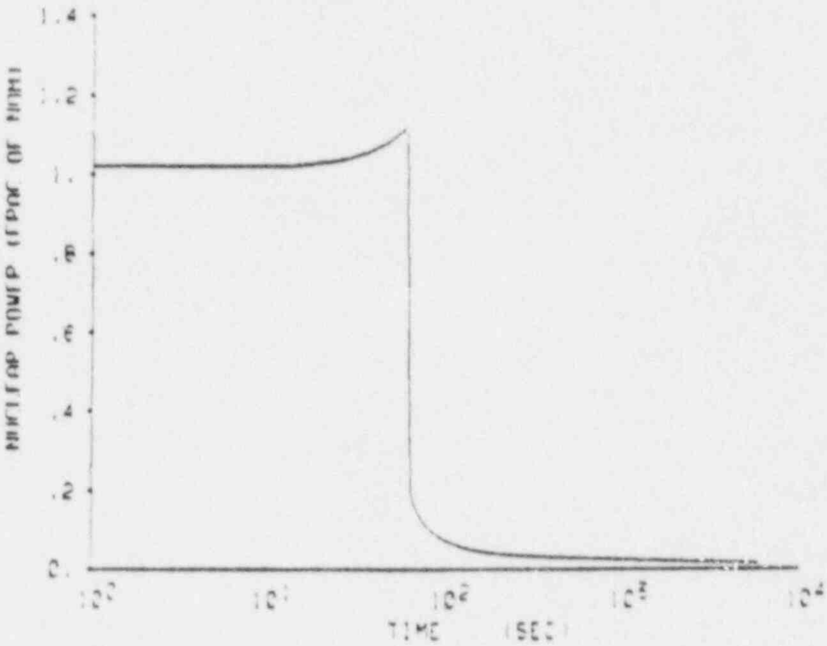
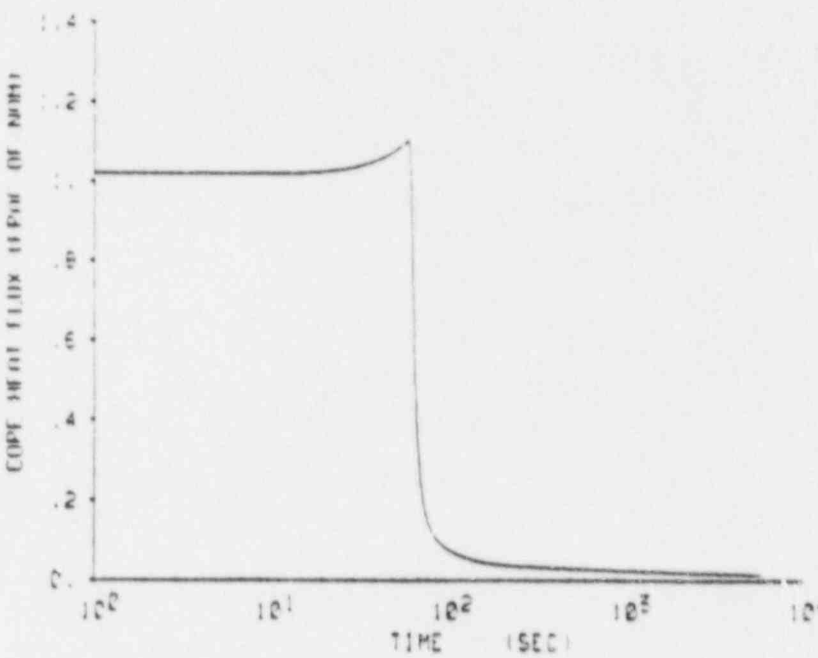


FIGURE 3.7-6
PRESSURIZER PRESSURE AND
WATER VOLUME TRANSIENTS FOR
LOSS OF OFFSITE POWER





7
AND CORE HEAT FLUX
SITE POWER

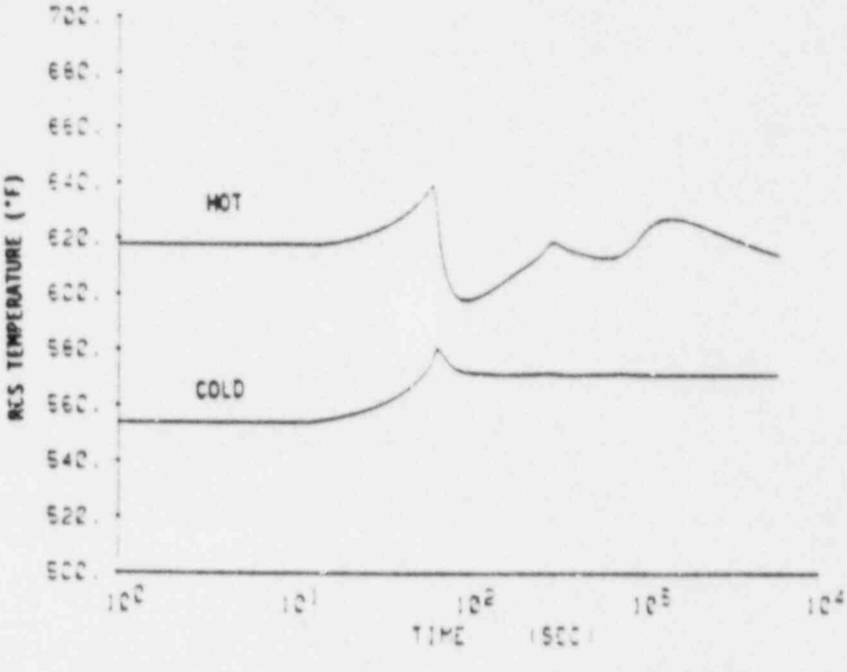


FIGURE 3.7-8
LOOP TEMPERATURES FOR
LOSS OF OFFSITE POWER

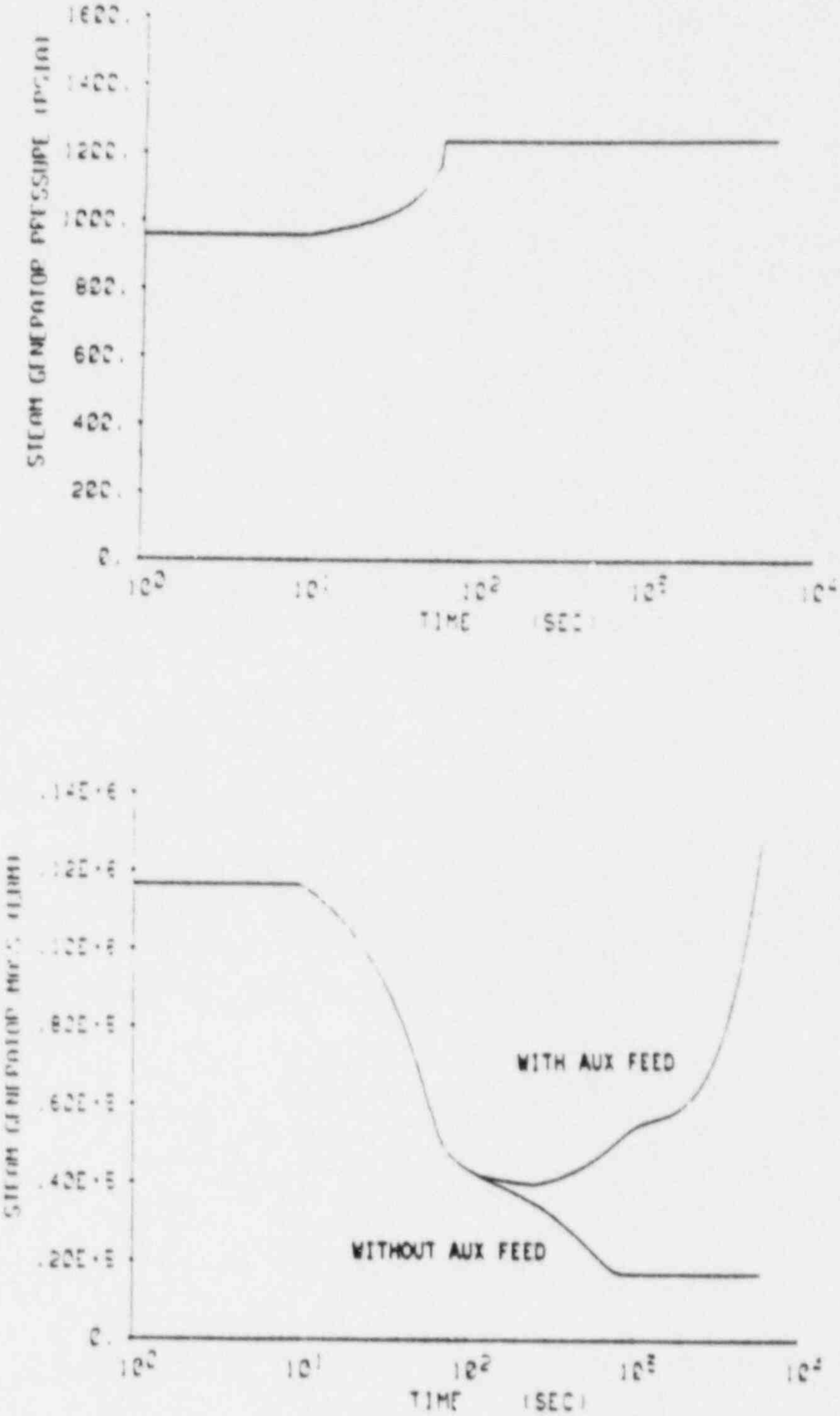
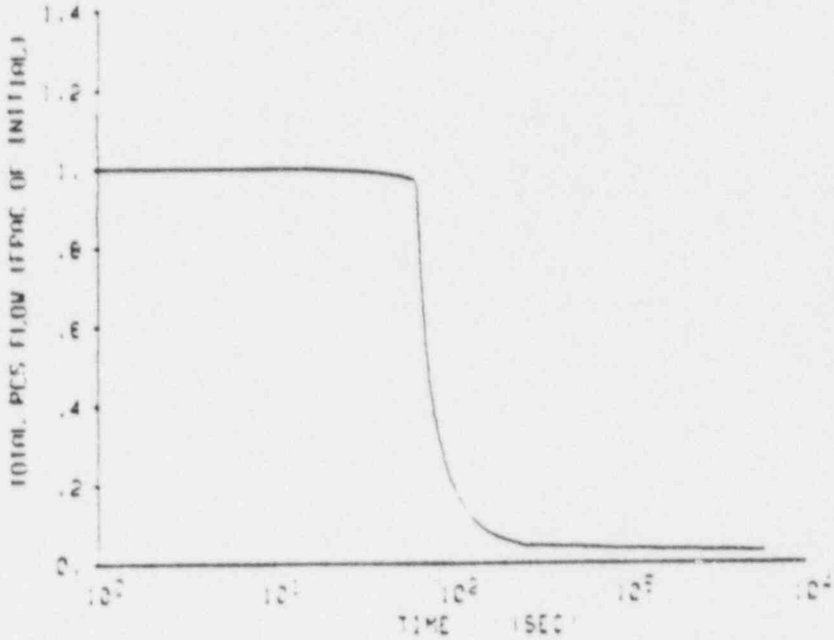


FIGURE 3.7-9
STEAM GENERATOR PRESSURE AND MASS TRANSIENTS FOR LOSS OF OFFSITE POWER



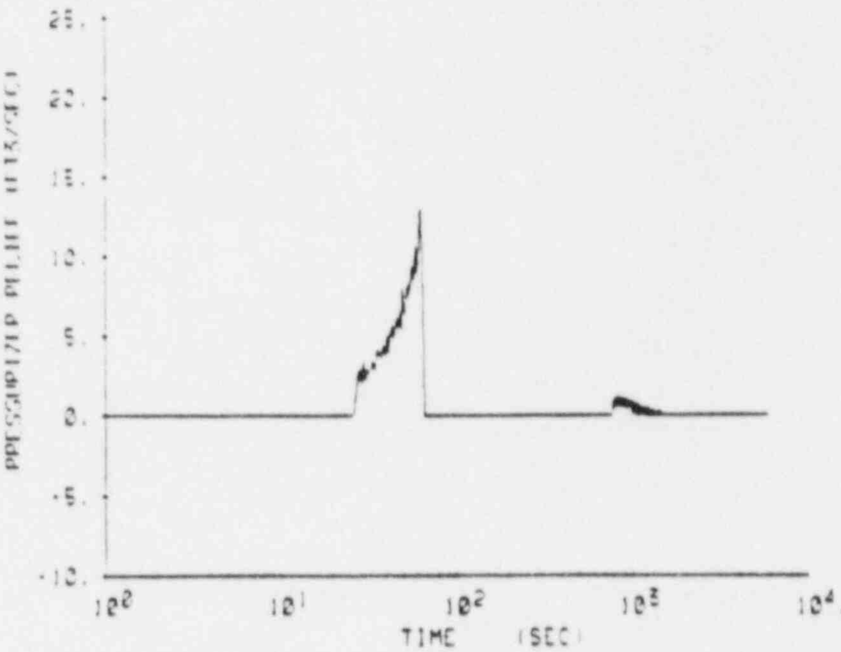


FIGURE 3.7-10
RCS FLOW AND PRESSURIZER RELIEF
FOR LOSS OF OFFSITE POWER

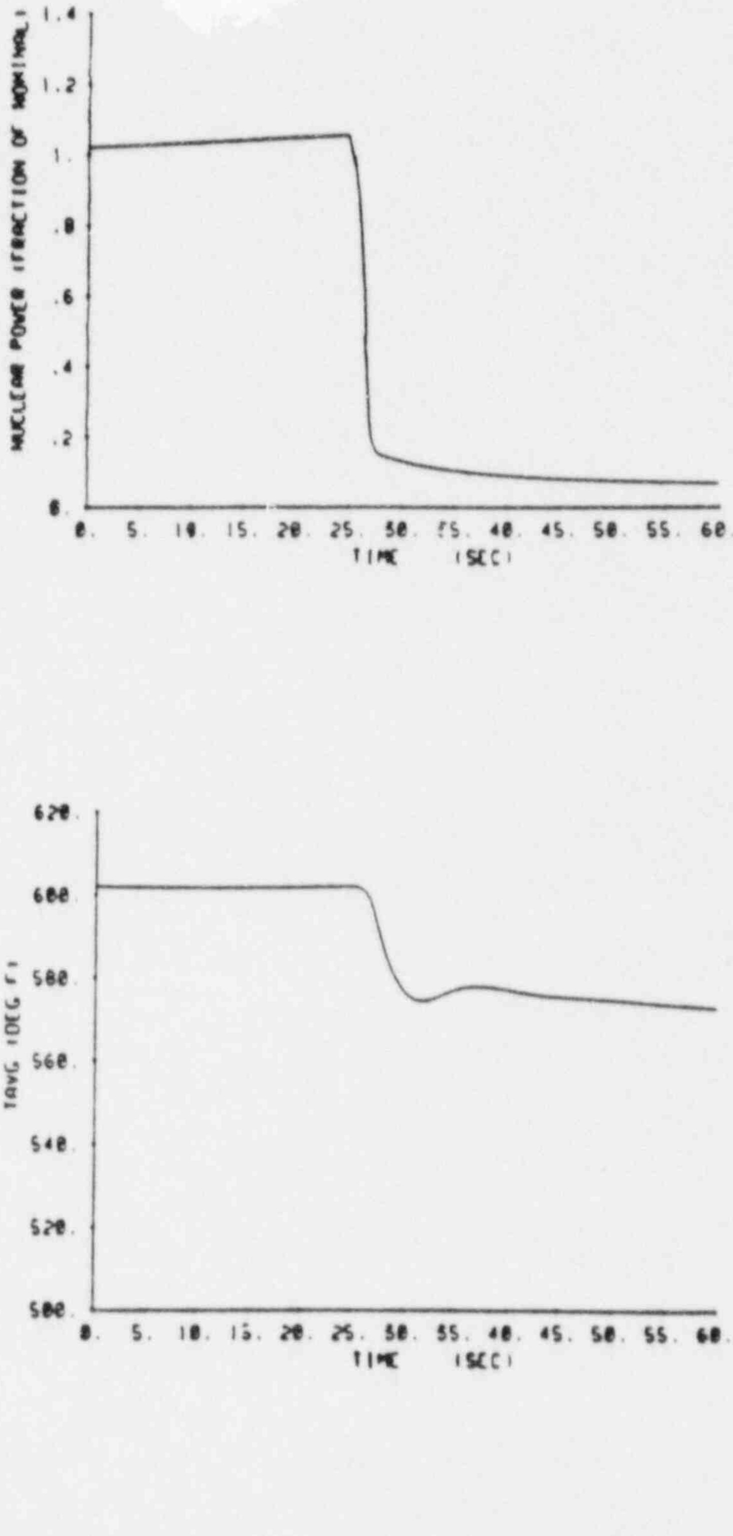
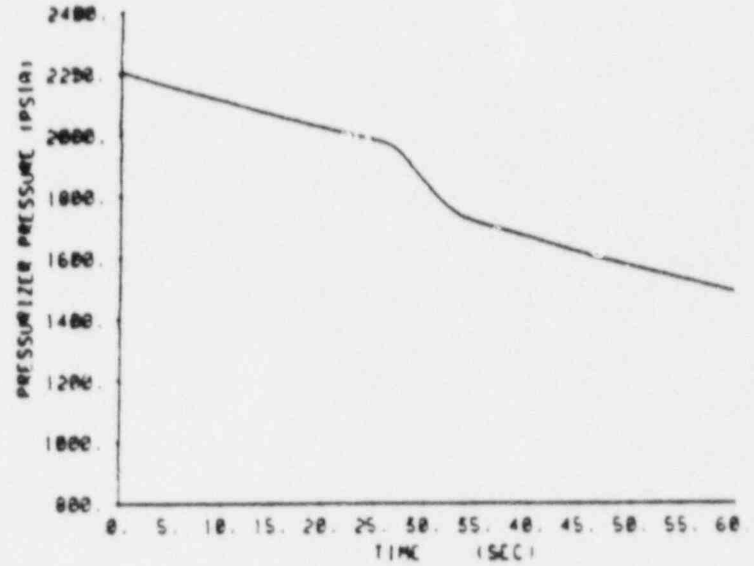


FIGURE 3.8-1
NUCLEAR POWER AND CORE AVERAGE
TEMPERATURE FOR INADVERTENT OPENING
OF A PRESSURIZER SAFETY VALVE



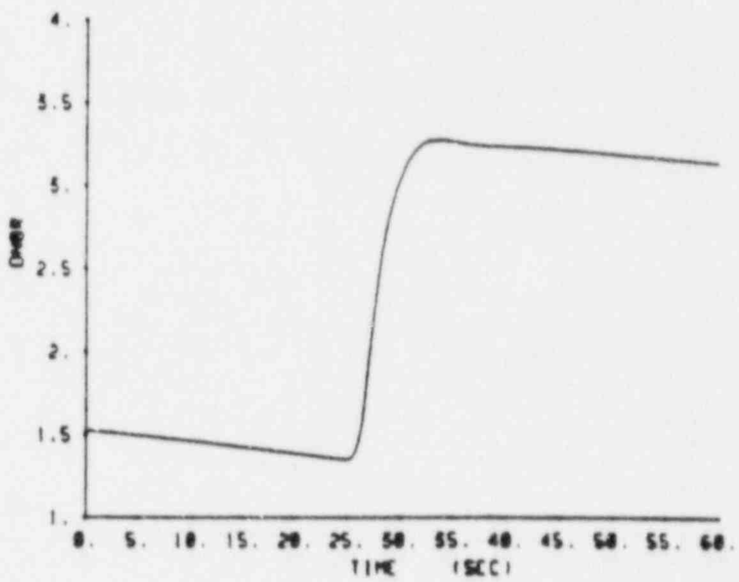


FIGURE 3.8-2
PRESSURIZER PRESSURE AND DNBR
FOR INADVERTENT OPENING OF A
PRESSURIZER SAFETY VALVE

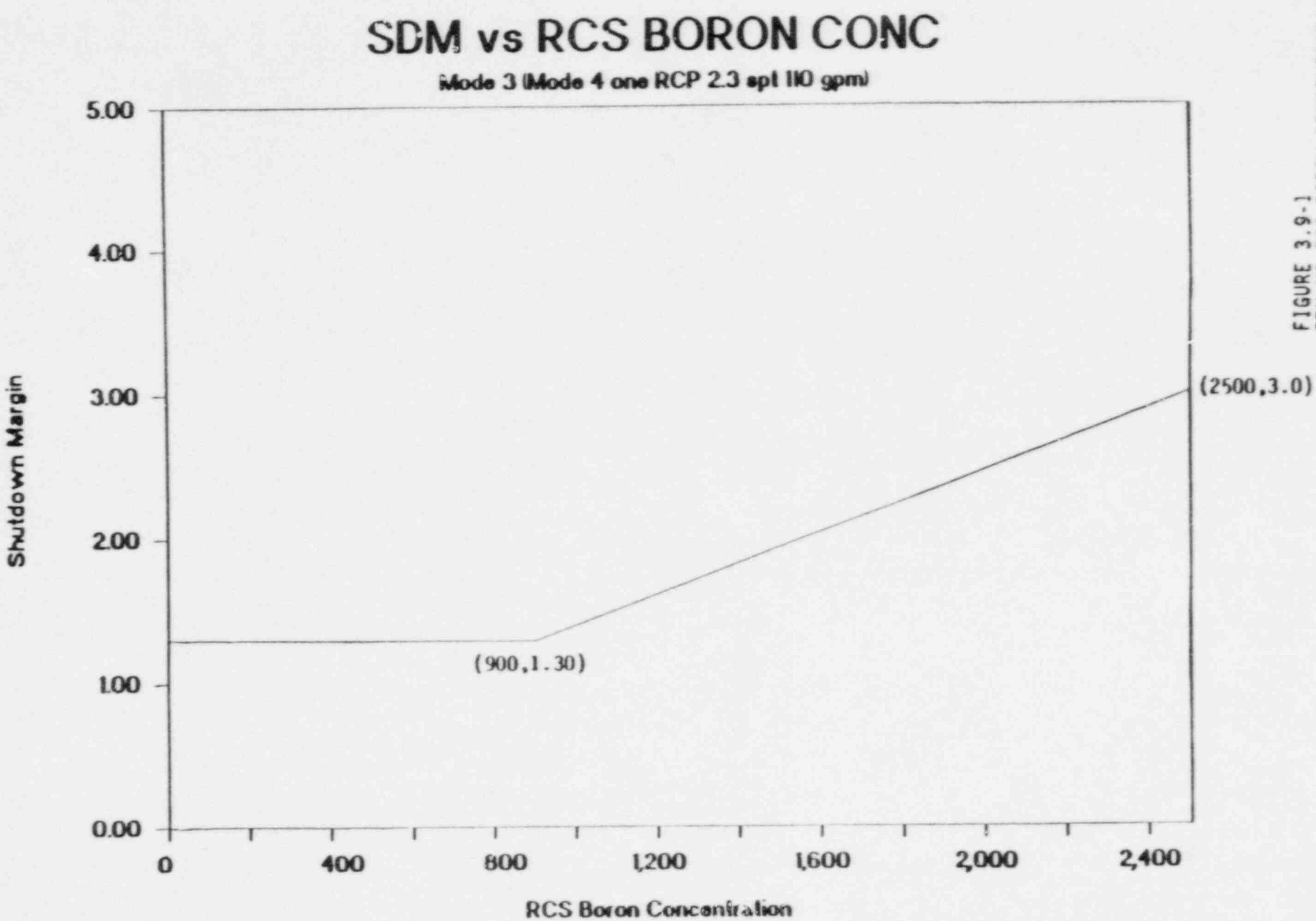


FIGURE 3.9-1
SDM VS RCS BORON CONCENTRATION -
MODE 3 (MODE 4 WITH RCP)

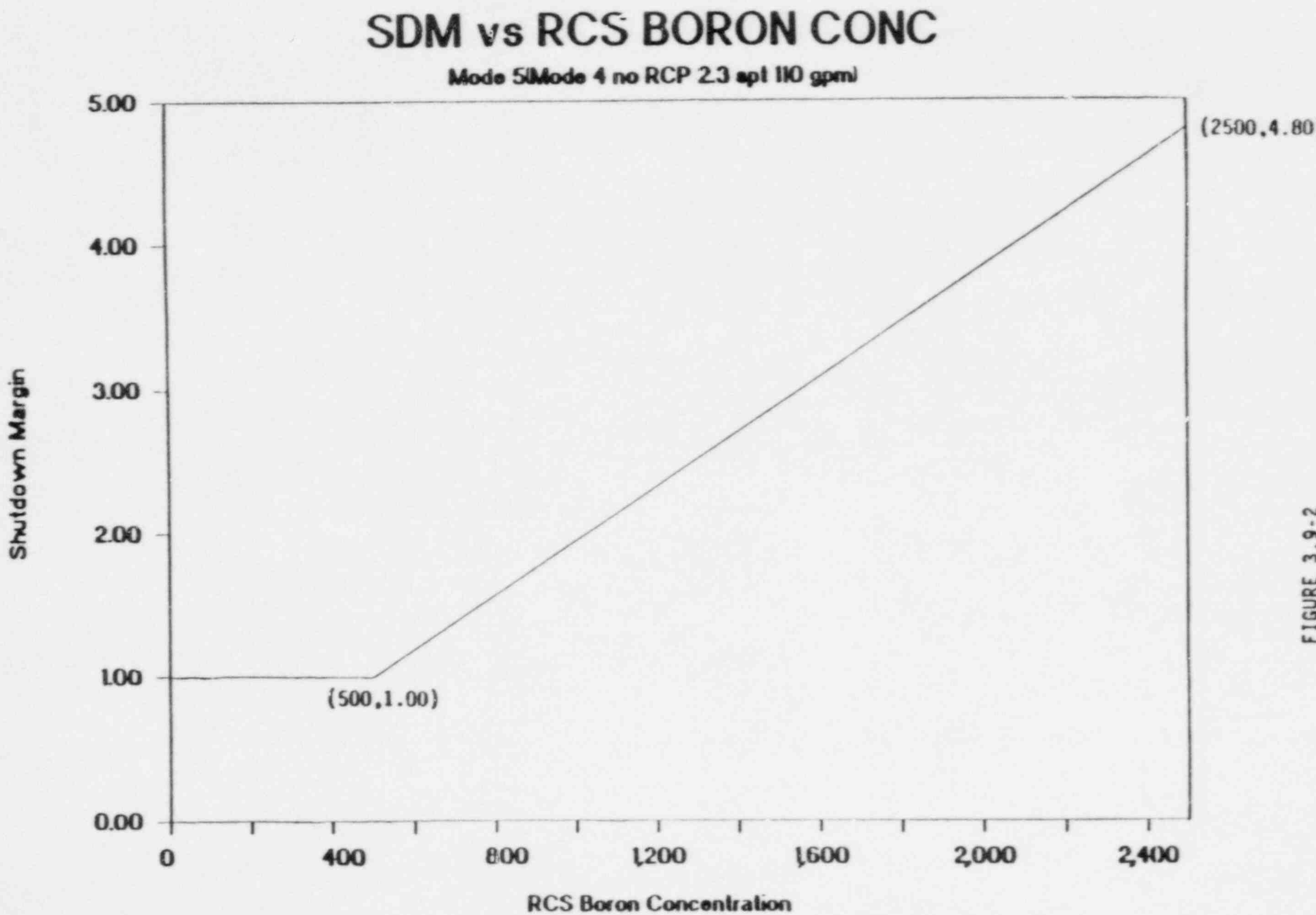


FIGURE 3.9-2
SDM VS RCS BORON CONCENTRATION -
MODE 5 (MODE 4 WITHOUT RCP)

4.0 RWST AND ACCUMULATOR BORON INCREASE

4.1 INTRODUCTION

As a result of the implementing the positive moderator temperature coefficient and extending operating cycles to 18 months, a change to the boron requirements is required to meet long term core cooling requirements outlined in 10CFR50.46. In doing this, the maximum boron concentration of the RWST and accumulators has been increased to 2600 ppm, the minimum boron concentration of the RWST has been increased to 2400 ppm, and the minimum boron concentration of the accumulators remained at 1900 ppm.

Having modified the boron concentration limits, a review of the pertinent LOCA and non-LOCA transients which employ boron concentration as an input is necessary.

These include the following areas of consideration.

- Non-LOCA transient and accident analyses
- Post-LOCA precipitation of boron due to long term boiling in the core
- Post-LOCA subcriticality
- Containment spray pH and long term equilibrium sump pH
- Borated water volume requirements for boric acid storage tank and refueling water storage tank technical specifications and setpoints
- Operation of the Reactor Makeup Control System (Boric Acid Blending System)

The above issues are discussed in subsequent sections of this report.

4.2 EVALUATION OF RWST AND ACCUMULATOR BORON CONCENTRATION INCREASE ON NON-LOCA TRANSIENTS

In conjunction with licensing a positive moderator temperature coefficient for Vogtle Units 1 and 2, the RWST boron concentration range (as defined in the Technical Specifications 3/4.1.2.5, 3/4.1.2.6, 3/4.5.4) is being revised to 2400-2600 ppm. The current range is 2000-2100 ppm (Modes 1, 2, 3, 4) and 2000-2200 ppm (Modes 5 and 6). For the non-LOCA FSAR transients discussed below, an increase in the RWST boron concentration results in an operational state which is less limiting than that assumed in the current licensing basis analysis. The impact of the boron concentration increase is discussed below for each of the non-LOCA transients which model the RWST. A brief discussion of the impact of the corresponding change in accumulator boron concentration from 1900-2100 ppm to 1900-2600 ppm follows the RWST discussion.

1. Inadvertent Opening of a Steam Generator Relief or Safety Valve (FSAR Section 15.1.4), Steam System Piping Failure (FSAR Section 15.1.5), Steamline Break Mass and Energy Release Inside Containment (FSAR Section 6.2.1.4), and Steamline Break Mass and Energy Release Outside Containment (WCAP-11285).

   All of the above steamline break transients take credit for the RWST boron (injected into the RCS via the SI system) to help mitigate the positive reactivity insertion caused by the excessive cooldown of the RCS. The minimum RWST boron concentration currently allowed by the Technical Specifications (2000 ppm), or a value conservatively less, is assumed in the analyses. Increasing the minimum boron concentration of the RWST will insert more negative reactivity into the core for these transients and provide less limiting results. The analyses in the FSAR will remain bounding and the associated conclusions will remain valid for the increased boron concentration.

2. Feedwater System Pipe Break (FSAR Section 15.2.8).

   The SI system is modeled for this transient with the minimum RWST boron concentration. This transient is not sensitive to boron concentration; however, increasing the boron concentration would tend to produce slightly less limiting results. For this transient, the SI serves primarily as a source of cool water which aids in cooling the primary system and helps ensure that the core remains covered. Therefore, the conclusions of the FSAR analysis remains valid.

3. CVCS Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (FSAR Section 15.4.6).

   The minimum RWST boron concentration is considered for the boron dilution events in Modes 3, 4, and 5. The minimum concentration is used to help define an upper limit on the RCS boron concentration plotted in Tech Spec Figures 3.1-1 and 3.1-2. These figures define the required shutdown margin, as a function of RCS boron concentration, to ensure sufficient operator action time for a boron dilution event. To cover possible higher RCS boron concentrations allowed by the increased RWST concentration, the Technical Specification Figures 3.1-1 and 3.1-2 have been revised.

4. Inadvertent Operation of the ECCS During Power Operation (FSAR Section 15.5.1).

   The RWST boron concentration is used to define the boron concentration of the SI water inadvertently injected into the RCS. The increased RWST concentration will affect this transient by decreasing the nuclear power (and consequently the core average temperature and RCS pressure) at a slightly faster rate than in the current analysis. The decreasing power and temperature would be of sufficient magnitude to offset the DNBR penalty of decreasing RCS pressure, such that the trend of increasing DNBR for this event will not change and there will continue to be a large margin to the DNBR limit. The conclusions of the FSAR analysis will remain valid for an increased RWST boron concentration.

All of the non-LOCA transients impacted have been reviewed, and it has been shown that the conclusions of the FSAR analyses will remain valid for an increase in RWST boron concentration. Except for revising the shutdown margin curves for Technical Specification Figures 3.1-1 and 3.1-2, no other reanalysis is necessary.

4.2.1 Evaluation of the Accumulator Boron Increase on Non-LOCA Transients

The accumulator boron concentration range, defined in Technical Specification (3/4.5.1), is currently 1900-2100 ppm. The proposed range is 1900-2600 ppm. The only non-LOCA transient which take credit for the accumulators are the steamline break events. The minimum concentration is modeled for conservatism. Because there is no change to the minimum concentration, there is no change to the steamline break events. If, in the future, the minimum concentration is increased, there would be no adverse impact on the analyses for the same reasons as presented in item 1 above.

4.3 EVALUATION OF THE RWST AND ACCUMULATOR BORON CONCENTRATION INCREASE ON THE LOCA TRANSIENTS

1. Small Break LOCA Analysis (FSAR Section 15.6.5)

   The small break LOCA analyses performed by Westinghouse assume that the reactor core is brought to a subcritical condition by the trip reactivity associated with the control rods and is complemented by borated water from the safety injection. The computer calculation models used to calculate small break LOCA PCT do not have explicit models to calculate core reactivity or account for the boron provided by the Emergency Core Coolling System (ECCS). An assumption used in these calculational models is that the reactor remains subcritical following control rod insertion. Therefore, an underlying assumption is that the boron provided by the Accumulators and ECCS pumps is sufficient to keep the reactor subcritical All Rods In -2 (ARI-2). The assumption of ARI-2 is based on a postulated small break LOCA as a result of a control ejection accident. In this case, one control rod would be assumed lost to the ejection event and another is assumed not to insert. Thus, an increase in the RWST and accumulator boron concentrations do not alter the conclusions of the FSAR small break LOCA analysis.

2. Large Break LOCA Analysis (FSAR Section 15.6.5)

   Large break LOCA analyses performed by Westinghouse do not take credit for the negative reactivity introduced by the soluble boron in the ECCS water in determining reactor power level during the early phases of the hypothetical large break LOCA. The traditional large break LOCA analyses performed by Westinghouse analyzes the LOCA transient to a time just beyond the time at which Peak Cladding Temperature (PCT) is calculated to occur. During this time period the reactor is kept subcritical by the voids present in the core. Thus, the changes to the RWST and accumulator boron concentration do not alter the conclusions of the FSAR large break LOCA analyses.

3. Post-LOCA Long-Term Core Cooling (FSAR Section 15.6.5)

A Westinghouse licensing position for satisfying the requirements of 10CFR Part 50 Section 50.46 Paragraph (b) Item (5) "Long-Term cooling" is defined in WCAP-8339 (page 4-22). The Westinghouse commitment is that the reactor remain shutdown by the borated ECCS water. Since credit for the control rods is not taken for large break LOCA, the borated ECCS water provided by the RWST and Accumulators must have a concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical assuming all control rods out. The result of increasing the RWST minimum boron concentration limit to 2400 ppm yields an increase of about 280 ppm in the post LOCA RCS/sump boron concentration. This increase in the RCS/sump boron concentration should provide enough additional negative reactivity to offset the effects introduced by PMTC and longer cycles. Plant Vogtle compliance with the long term cooling commitment is verified through the RSAC process for each reload. The RSAC effort will ensure compliance.

4. LOCA Hydraulic Vessel and Loop Forces (FSAR Section 3.6)

The blowdown hydraulic loads resulting from a loss of coolant accident are considered in the FSAR Section 3.6.2. The increase in the RWST minimum boron concentration will have no effect on the LOCA blowdown hydraulic loads since the maximum loads are generated within the first few seconds after break initiation. For this reason the ECCS, including the RWST, is not considered in the LOCA hydraulic forces modeling and thus the increase in RWST boron concentration will have no effect on the results of the LOCA hydraulic forces calculations.

5. Steam Generator Tube Failure or Rupture (FSAR Section 15.6.3)

As noted in the introduction of this report (Section 1.0) Georgia Power Company has recently submitted a revised SGTR analysis using NRC approved methodology to replace the current SGTR analysis of FSAR Section 15.6.3. As such, the effects of the RWST/Accumulator boron increase are discussed for both accidents.

SGTR (FSAR Section 15.6.3)

The steam generator tube rupture (SGTR) accident for Vogtle Units 1 and 2 is presented in FSAR Chapter 15.6.3. For the SGTR accident, the low pressurizer pressure SI signal is actuated due to the decrease in the reactor coolant inventory shortly after reactor coolant system. (No. Accumulator actuation occurs in this analysis.) For the FSAR SGTR analysis, the primary to secondary break flow was assumed to be terminated at 30 minutes after the initiation of the event. However, the operator actions required to terminate the break flow, including the initial RCS cooldown to provide subcooling margin, were not modeled in the analysis. Although the RCS cooldown is not modeled, sufficient shutdown margin is assumed to be available initially due to insertion of the control rods following reactor trip, and adequate shutdown margin is assumed to be maintained for the long term by the borated safety injection water. An increased RWST minimum boron concentration will result in more negative reactivity insertion in the SGTR accident. Therefore, the higher RWST boron concentration will have no adverse effect on the FSAR SGTR analysis.

Revised SGTR Analysis (WCAP-11731)

For the SGTR accident, the low pressurizer pressure SI signal is actuated due to the decrease in the reactor coolant inventory shortly after reactor trip and borated water from the RWST is delivered to the reactor coolant system. (No Accumulator actuation occurs in this analysis).

For the revised SGTR analysis in WCAP-11731, the operator actions required to terminate the primary to secondary break flow are modeled in the analysis. The operator actions are based on the recovery operations specified in the Vogtle Emergency Operating Procedures (EOPs) which were developed from the Westinghouse Owners Group Emergency Response Guidelines (ERGs). Although the results of the revised SGTR analyses in WCAP-11731 indicate that adequate shutdown margin is maintained during the initial RCS cooldown to provide subcooling margin, the analysis is not designed to provide a conservative evaluation of the shutdown margin for the conditions which may be encountered for recovery from an SGTR. Rather the analyses in WCAP-11731 are designed to provide a conservative evaluation of the margin to overfill and the offsite radiological consequences for an SGTR. Since the ERG actions and setpoints are designed to prevent a return to criticality during the initial RCS cooldown, it is assumed that application of the Vogtle EOPs will assure that subcriticality is maintained for the conditions encountered during the recovery operations. On this basis, it is assumed that sufficient shutdown margin will be provided initially due to insertion of the control rods and that adequate shutdown margin will be maintained for the initial RCS cooldown by the borated safety injection water. Since the higher RWST boron concentration will result in an increase in the negative reactivity insertion due to the addition of the safety injection water, there will also be no adverse effect on the revised SGTR analysis in WCAP-11731.

6. Containment Integrity Analyses (FSAR Section 6.2)

The containment integrity analyses are described in the FSAR Chapter 6.2. This section considers, Inadvertent Spray Actuation and Subcompartment Pressure Transient Analyses, Short Term and Long Term Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents (LOCA), and, Containment Response Analysis following a LOCA or Steamline Break Inside Containment.

For Short Term Mass and Energy and Subcompartment Pressure Analyses an increase in the RWST boron concentration would have no effect on the calculated results, since the short duration of the transient ($\leq$3 seconds) does not consider any safety injection flow taken from the RWST. The long term mass and energy release and containment response calculations following a LOCA do not take credit for the soluble boron present in the safety injection from the RWST supplied to the RCS. This is similar to the ECCS LOCA analyses assumptions, and therefore an increase in ECCS water boron concentration, would have no effect on these analyses.

The minimum RWST boron concentration allowed by the Technical Specifications is modeled in the mass and energy release analysis for postulated secondary system pipe ruptures inside containment. An increase in the boron concentration will insert more negative reactivity into the core and result in less limiting mass and energy releases, and therefore will lessen the consequences of adverse containment conditions. Thus, the conclusions presented in the current Vogtle FSAR will remain valid.

7. <u>Hot Leg Switchover to Prevent Boron Precipitation (FSAR Section 6.3.2)</u>

The hot leg recirculation switchover time analysis has been performed for Vogtle Units 1 and 2 to determine the time following a LOCA that hot leg recirculation should be initiated. This analysis addresses the concern of boron precipitation in the reactor vessel following a LOCA and has been performed to support the increase in Technical Specification RWST and Accumulator maximum boron concentration limits to 2600 ppm.

During a large break LOCA the plant switches to cold leg recirculation after the RWST switchover setpoint has been reached. If the break is in the cold leg there is a concern that the cold leg injection water will fail to establish flow through the core. Safety injection entering the broken loop will spill out the break, while SI entering the intact cold legs will circulate around the downcomer and out the break. With no flow path established through the core the fluid in the core remains stagnant. As steam is produced in the core from decay heat the analysis conservatively assumes that the boron associated with the steam remains in the vessel. Thus, as water is boiled off and with no recirculation present in the core, the boric acid concentration increases to a level where boron will begin to precipitate. As the boron precipitates, it may plate out on the fuel rods which would adversely affect their heat transfer characteristics. The purpose of the hot leg recirculation switchover time analysis is to provide a time at which hot leg recirculation can be established such that boron precipitation in the core can be prevented.

The calculation considers the increase in boric acid concentration in the vessel during the long term cooling phase of a LOCA. The analysis assumes that following a LOCA the steam boil-off from the core does not carry any boron. A constant volume of liquid in the vessel is assumed so that as steam is boiled off and the boron is left behind, the boric acid concentration of the vessel increases. The time when the boric acid solution reaches the solubility limit less 4 weight percent is when hot leg recirculation should be initiated. The solubility limit less 4 weight percent at a solution temperature of 212°F has been established as 23.53%. Thus when the boric acid solution concentration reaches 23.53%, hot leg recirculation should be initiated. Hot leg recirculation provides an injection path into the core which dilutes the high concentration boric acid solution in the core and prevents the build-up of boron on the fuel rods.

Results

An analysis has been performed to determine the time following a LOCA that switchover to hot leg recirculation should be initiated to prevent boron precipitation in the reactor vessel.

This time has been determined to be 11 hours.

The analysis considers the increase in boric acid concentration in the reactor vessel during the long term cooling phase of a LOCA, assuming a conservatively small effective vessel volume. This volume includes only the free volumes of the reactor core and upper plenum below the bottom of the hot leg nozzles. This assumption conservatively neglects the mixing of boric acid solution with directly connected volumes, such as the reactor vessel lower plenum. The calculation of boric acid concentration in the reactor vessel considers a cold leg break of the reactor coolant system in which steam is generated in the core from decay heat while the boron associated with the boric acid solution is completely separated from the steam and remains in the effective vessel volume.

The results of the analysis show that the maximum allowable boric acid concentration of 23.53 weight percent established by the NRC, which is the boric acid solubility limit less 4 weight percent, will not be exceeded in the vessel if hot leg recirculation is initiated 11 hours after the LOCA inception. The operator should reference this switchover time against the reactor trip/SI signal. The typical time interval between the accident inception and the reactor trip/SI actuation signal is negligible when compared to the switchover time.

Procedure philosophy assumes that it would be very difficult for the operator to differentiate between break sizes and locations. Therefore one hot leg switchover time is used to cover the complete break spectrum.

The implementation of the PMTC for Vogtle requires that the RWST boron concentration be increased. The results of increasing the boron concentration have been evaluated above. Since this is the primary means in which PMTC affects the time for hot leg switchover and the switchover time has been recalculated to reflect the increased boron concentration, the implementation of a PMTC is acceptable with respect to the time determined for hot leg switchover.

8. Rod Ejection Mass and Energy Releases for Dose Calculations (FSAR Section 15.4.8)

The rod ejection mass and energy releases for Vogtle Units 1 and 2 are presented in FSAR Section 15.4.8. The effect of the increase in the RWST boron concentration will be negligible on the rod ejection accident analysis. Since the SI flow taken from the RWST is modeled under similar assumptions as in the large break and small break LOCA analyses, there will be no adverse effect on the FSAR rod ejection accident.

4.4 EVALUATION OF THE RWST AND ACCUMULATOR BORON INCREASE ON FLUID SYSTEMS AND ON LOCA RADIOLOGICAL CONSEQUENCES

Refueling Water Storage Tank

In conjunction with the licensing of the Vogtle units with a positive moderator temperature coefficient, the minimum boron concentration specified for the Refueling Water Storage Tank (RWST) has been increased from 2000 ppm to 2400 ppm in order to assure that core subcriticality is maintained in the event of a postulated Loss-of-Coolant Accident (LOCA) . In order to accommodate expected variation in RWST boron concentrations, the maximum allowable boron concentration in the RWST is established at 2600 ppm. This provides a window of 200 ppm between the minimum and maximum limits. This operating window is twice the current 100 ppm window (MODES 1,2,3,4) in order to give flexibility in increased operating margin with respect to RWST boron concentration limits.

Considering the requirements for 18 month reload design for Vogtle, the volume of boric acid solution required in the RWST during MODES 5 and 6 (Technical Specification 3.1.2.5) is increased despite the higher boron concentration of the solution. Likewise, the volume of boric acid solution in the RWST required to bring the plant to cold shutdown conditions is increased from the value reported presently in the Technical Specification Bases (Section 3/4.1.2) but does not result in a change in Technical Specification 3.1.2.6 which addresses the volume of solution to be maintained in the RWST during MODES 1,2,3, & 4.

Accumulators

When establishing boron concentration limits for the safety injection accumulators, it is customary to set the maximum boron concentration of the accumulator tanks equal to the maximum RWST boron concentration since the accumulator is filled and the water level is maintained by pumping in water from the RWST with the safety injection pump. This practice is continued for Plant Vogtle so that the maximum boron concentration of the accumulator water is set at 2600 ppm - equal to that in the RWST.

The minimum boron concentration of the accumulator water is typically set at least 100 ppm lower than the minimum boron concentration for the RWST. This allows for some small amount of dilution of the accumulator tank water due to possible back leakage of low boron concentration water from the Reactor Coolant System during normal operation. This practice sets the minimum boron concentration assumed for the accumulators in calculations performed to determine the sump boron concentration post-LOCA necessary to show that the reactor core remains subcritical. If this practice were continued, the minimum boron concentration for the accumulators would be increased from the current value of 1900 ppm up to a value of 2300 ppm (i.e., 2400 ppm RWST less 100 ppm). However, the minimum accumulator boron concentration will be retained at 1900 ppm to aid in implementation of the RWST Technical Specification change. An additional benefit of this approach is to provide as large a window as possible (1900 ppm to 2600 ppm) to accommodate possible variations in accumulator boron concentration. A minimum accumulator boron concentration of 1900 ppm will be assumed in calculations performed to determine if future reload cores meet the post-LOCA subcriticality requirement.

Boric Acid Storage Tank

There are no changes necessary in the boron concentration limits for the boric acid storage tanks. These limits remain at the current values ranging from 7000 ppm minimum to 7700 ppm maximum.

Considering the requirements for 18 month reload designs for Vogtle, the volumes of boric acid solution required in the boric acid storage tanks during MODES 1,2,3, & 4, as well as during MODES 5 and 6 remain below the values given in the Technical Specification Bases (Section 3/4.1.2) and below the limits used in Technical Specifications 3.1.2.5 and 3.1.2.6. Thus, there is not change to the operating limits for the boric acid storage tanks.

Reactor Makeup Control System

The Reactor Makeup Control System was originally designed to blend 120 gpm of makeup water at a nominal boron concentration of 2000 ppm. This 2000 ppm boron concentration was based upon the refueling requirement and was achieved by blending approximately 35 gpm of nominal 4 weight percent boric acid from the boric acid storage tank at 7000 ppm with 85 gpm of reactor grade makeup water. This makeup system performance requirement is utilized for filling and maintaining levels in the refueling water storage tank and the spent fuel pit. It is also used to provide automatic makeup to the volume control tank when the CVCS is used to provide makeup for RCS contraction during a refueling shutdown.

The desired blended boron concentration achieved by the reactor makeup control system may be as high, for example, as 2500 ppm in order to accommodate the anticipated RCS boron concentrations required by long reload cycles and in order to accommodate the 2400 ppm - 2600 ppm RWST boron concentration limits.

In order to blend 120 gpm of 2500 ppm boric acid, the required flowrate of concentrated boric acid (7000 ppm) from the boric acid storage tank is approximately 43 gpm. This exceeds the capacity of the boric acid portion of the blending system.

The recommended course of action is to lower the rack setpoint for the automatic makeup flowrate from the current 120 gpm value down to 100 gpm. In this way, the makeup control system will be able to blend approximately 35 gpm of 7000 ppm boron liquid from the boric acid storage tank with 65 gpm of reactor makeup water to deliver 100 gpm of 2500 ppm liquid.

Containment Spray and Equilibrium Sump pH

The Containment Spray System is automatically actuated on a Hi-Hi containment pressure signal. Sodium hydroxide solution (30-32 weight percent) contained in the spray additive tank is automatically entrained into the spray pump flow drawn from the RWST in order to remove iodine from the containment atmosphere during the injection phase and to adjust the pH of the sump solution from an acid condition to a specified level of alkalinity. The current licensing basis for the Vogtle Containment Spray System is to limit long term sump pH to between 8.5 and 10.5 and to limit spray pH to less than 11.0.

The increase in RWST boron concentration to 2400-2600 ppm boron and the related increase in boron concentration for the accumulators and for the reactor coolant system itself have the combined effect of reducing the minimum long term equilibrium sump pH post LOCA. For purposes of this evaluation, the liquid in the RWST and in the accumulators was assumed to be at the maximum proposed boron concentration (i.e., 2600 ppm). The borated water volumes in these two sources were maximized relative to existing setpoints for high alarms. The sodium hydroxide solution was assumed to be at the minimum concentration permitted by the Technical Specifications (30 weight percent). The RCS boron concentration was also maximized consistent with the possible initial boron concentrations associated with future postulated core designs. Both trains of emergency core cooling are assumed to operate, but one of the two containment spray pumps is assumed to fail. This set of assumptions conservatively minimizes the sump pH at the time that the RWST empty alarm is reached by minimizing the ratio of sodium hydroxide to boron in the sump. Determined in this fashion, the minimum sump pH is established at 8.15. Since this value is less than the current licensing basis minimum equilibrium sump solution pH of 8.5, the Vogtle licensing basis should be revised to reflect a minimum long term sump pH of 8.0 to provide some margin between the calculated value and the limit.

The maximum sump pH remains less than 10.5 since the increase in the RWST boron concentration range would decrease the sump pH. In addition, the maximum spray pH during injection is conservatively estimated to be less than 10.5 and above 8.5.

The reduction in the minimum equilibrium sump solution pH from 8.5 to 8.0 potentially impacts a number of areas which are addressed as follows:

Post-LOCA Hydrogen Generation

Following a large-break LOCA, one of the sources of hydrogen is the corrosion of construction materials in the containment; specifically aluminum and zinc. The corrosion rate for aluminum is pH dependent and decreases monotonically with decreasing pH. Thus the adoption of a reduction in the equilibrium sump solution pH would slow the the rate of hydrogen production due to aluminum corrosion. The corrosion of zinc is a function of pH and temperature, with temperature being the dominant factor. The zinc corrosion rate increases with decreasing pH; however, the increase in corrosion rate between pH of 8.5 and pH of 8.0 is insignificant.

Equipment Qualification

The pH envelope used as a basis for equipment qualification for Plant Vogtle is 8.5 to 10.5 (FSAR Section 3.11). The reduction in the minimum pH from 8.5 to 8.0 is a change to a less aggressive environmental limit (closer to neutral) and thus would not adversely impact the the existing equipment qualification. It is noted that the environmental testing of Westinghouse supplied equipment is performed at a pH of 10.7 (to provide some margin beyond the upper limit of 10.5) and the lower pH of 8.5 is not included in the testing as it is much less aggressive. The same argument can be applied to pH reductions down to 7.0 (neutral). Components qualified at higher pH may actually have a longer post-accident service life in a lower pH environment (in the caustic range) although the primary cause of component failure is elevated temperature, not pH.

The above discussion, is specifically applicable to Westinghouse supplied equipment.

Like electrical equipment, environmental testing of coatings uses a high pH solution to maximize the severity of the test environment. The coatings may show better resistance to lower pH solutions; although, in the post accident environment, degradation of coatings, just like equipment, is most likely to result from the elevated temperatures and not from pH.

Chloride Induced Stress Corrosion Cracking of Stainless Steel

Initiation of chloride induced stress corrosion cracking is a function of both the pH of the environment and the time that the stainless steel is exposed to the environment. Test results show that at a pH of 8.0, cracks do not appear even after sixteen months of exposure. At a pH of 7.0, tests show that cracking is initiated at between seven and eight months. Based on these test results, the Westinghouse recommendation for minimum sump solution pH is 7.5 (Reference 1). The NRC recommendation for sump pH after a LOCA is contained in Reference 2 which specifies that the minimum equilibrium sump pH should be no less than 7.0 and states that the higher the pH, in the range of 7 to 9.5, the greater the assurance that no stress corrosion cracking will occur. With the reduction in minimum equilibrium sump solution pH from 8.5 to 8.0, the solution pH remains consistent with the recommendations of Westinghouse and of the NRC and the change is judged to involve no significant impact insofar as chloride induced stress corrosion cracking of stainless steel is concerned.

LOCA Thyroid Doses

With the reduction in minimum sump solution pH from 8.5 to 8.0, the rate of removal of elemental iodine by the sprays is unaffected since the injection spray pH is still within the originally specified range of 8.5 to 11.0. However, the fraction of the airborne elemental iodine that can be assumed to be retained in the sump solution is dependent on the sump solution pH. Using the pH vs. Partition Coefficient curve from Reference 3 (as was used in the original analysis), the reduction from pH of 8.5 to 8.0 results in a reduction in the Decontamination Factor (e.g., the ratio of the elemental iodine initially airborne to the amount of elemental iodine remaining in the air after spray removal is complete) from 200 down to 60.0 (i.e., at the end of spray removal there is 1.7 percent of the original elemental iodine airborne instead of the 0.5 percent previously determined).

It is recognized that the radiological consequences analysis of the postulated LOCA contains many significant conservatisms which, if removed, would result in calculated doses being a small fraction of those reported in the Vogtle FSAR (Table 15.6.5-6). *In order to support the increase in the RWST boron concentration, the LOCA doses have been determined using reduced conservatisms in regard to:

1. Deposition removal of elemental iodine from the containment atmosphere (Reference 11),
2. Spray removal of particulate iodine from the containment atmosphere (Reference 11), and
3. Rate of unfiltered inleakage into the control room (Reference 12, 13).

The changes made in these areas are reductions in conservatism but do not constitute a "best estimate". The values utilized for the analysis remain conservative by still being within the quidelines of the above referenced documents. Also, there are many other conservatisms which have not been revised.

In addition to revising the LOCA radiological consequences analysis to reflect the increase in RWST boron concentration (including the above discussed reduction in the level of conservatism), the reanalysis also reflects revised performance identified for the Control Room Emergency HVAC, which impacts all control room doses (thyroid, beta-skin, and gamma-body), and for the Piping Penetration Area Emergency HVAC, which impacts thyroid doses both in the control room and off site.

The reanalysis provides the following doses:

Site Boundary Thyroid Dose

Containment Leakage	50.6 rem
Containment Purge	0.32 rem (not recalculated)
Recirculation Leakage	2.4 rem
Total	53.3 rem

Low Population Zone Boundary Thyroid Dose

Containment Leakage	57.2 rem
Containment Purge	0.13 rem (not recalculation)
Recirculation Leakage	2.9 rem
Total	60.2 rem

Control Room Thyroid Dose

Containment Leakage	28.2 rem
Containment Purge	0.01 rem (not recalculated)
Recirculation Leakage	0.7 rem
Total	28.9 rem

Control Room Gamma-Body Dose

Containment Leakage	4.8 rem
Containment Purge	4.2×10^{-5} rem (not recalculated)
Total	4.8 rem

Control Room Beta-Skin Dose

Containment Leakage	65.3 rem
Containment Purge	8.8×10^{-4} rem (not recalculated)
Total	65.3 rem

The control room doses are within the acceptance limits of 30 rem thyroid, 5 rem gamma-body, and 75 rem beta-skin (limit assuming use of appropriate action as mentioned in FSAR Table 15.6.5-6) while the off site thyroid doses are reduced significantly from the reported values in the FSAR. Thus, based on this revised determination of the LOCA doses, the increase in the RWST boron concentration does not result in violation of the acceptance limits.

5.0 CONTROL SYSTEM EVALUATION

One of the more important parameters in defining the NSSS response to a temperature and/or power transient is the moderator temperature reactivity feedback coefficient to nuclear power. With the reload core design there is a potential for this parameter to be positive. The effect of a positive (or at least less negative) moderator temperature coefficient (PMTC) is to potentially change the response of the NSSS and in turn the response of the steam dump, feedwater, and rod control systems towards a less stable configuration. However, the extent and time during which the new core design is expected to have a PMTC is limited and should not significantly affect the response of the control systems. Therefore, there is no apparent need to revise any of the control system setpoints.

The need to modify control system setpoints will be determined during the plant startup following the installation of the new core by observing the response of the control systems. If necessary, signal compensators and function generators in the control systems could be adjusted to obtain a more optimum system response. Since control system responses are not assumed in the PMTC transient reanalyses, changing control system setpoints will not impact the results reported in Section 3.0 of this report.

6.0 TECHNICAL SPECIFICATION CHANGES

Technical Specification changes as a result of PMTC and the increase in the boron concentration requirements for the RWST and the RCS accumulators are contained in Appendix A. The sections below summarize these changes.

6.1 SPECIFICATION 3/4.1.1.2

The variable shutdown margin curves for Modes 3, 4, and 5 (Figures 3.1-1 and 3.1-2) were replaced with revised curves. These curves were revised to reflect higher RCS boron concentrations (due to PMTC and high capacity factor 18 month core loading patterns), a reduction in the assumed dilution flow rate, and a reduction in high flux at shutdown alarm setpoint.

6.2 SPECIFICATION & BASES 3/4.1.1.3

The Moderator Temperature Coefficient (MTC) for beginning of cycle life was changed to allow for a PMTC. The limiting condition for operation (LCO) statement for beginning of core life (BOL) was revised to allow for MTC to be less positive than $+ 0.7 \times 10^{-4}$ Δk/k/°F for power levels up to 70% Rated Thermal Power with a linear ramp to 0 Δk/k/°F at 100% Rated Thermal Power.

The applicable ACTION statement was also changed to be consistent with the above allowable range of PMTC.

6.3 SPECIFICATION & BASES 3/4.1.2.5 & 3/4.1.2.6

The RWST minimum boron concentration was increased from 2000 to 2400 ppm and the RWST maximum boron concentration was increased from 2100 or 2200 ppm to 2600 ppm. The minimum contained RWST borated water volume (and corresponding level for Modes 5 and 6 was changed from 70,832 to 99,404 gallons.

The BASES maximum expected boration capability requirements for the RWST were revised. For Modes 1 through 4, the usable volume and boron concentration changed from 87,720 gallons and 2000 ppm boron to 178,182 gallons and 2400 ppm boron. For Modes 5 and 6, the usable volume and boron concentration changed from 12,630 gallons and 2000 ppm boron to 41,202 gallons and 2400 ppm boron.

The BASES lower pH limit for the solution recirculated within containment after a LOCA was also changed from 8.5 to 8.0.

6.4 SPECIFICATION & BASES 3/4.3.1

The Source Range High Flux at Shutdown Alarm Setpoint identified in Table 4.3-1, Footnote 9 was changed from 3.16 to 2.30 times background.

A statement was added to the BASES section to identify that the Source Range High Flux at Shutdown Alarm is modeled in the Boron Dilution Accident Analyses.

6.5 SPECIFICATION & BASES 3/4.5.1

The maximum boron concentration in the RCS Accumulators was changed from 2100 to 2600 ppm.

A statement was added to the BASES section to identify the basis for minimum boron concentration in the RCS accumulators.

6.6 SPECIFICATION & BASES 3/4.5.4

The minimum and maximum boron concentration for the RWST was changed from 2000 and 2100 ppm to 2400 and 2600 ppm, respectively.

The BASES lower pH limit for the solution recirculated within containment after a LOCA was changed from 8.5 to 8.0.

6.7 BASES 3/4.6.2.2

The BASES lower pH limit for the solution recirculated within containment after a LOCA was changed from 8.5 to 8.0.

6.8 BASES 3/4.9.1

The BASES discussion was revised to clarify the basis of the minimum boron concentration.

7.0 FSAR CHANGES

The FSAR changes associated with the implementation of the PMTC and associated boron concentration increases will be provided under a separate cover.

8.0 CONCLUSION

To assess the accident analysis effect of operation of Vogtle Units 1 & 2 with a positive moderator temperature coefficient and increased boron concentration in the RWST and accumulators, evaluations and transient reanalyses were performed. Discussions of the transient evaluations and reanalyses are discussed in Sections 2.0, 3.0 and 4.0 of this report. These evaluations/reanalyses indicate that the proposed technical specification of Figure 1.0-1 does not result in the violation of safety limits for any of the analyzed transients.

Except as noted, the analyses employed a constant moderator coefficient of +7 pcm/°F, independent of power level. The results of this study are conservative for the accidents investigated at full power, since the proposed Technical Specification diagrammed in Figure 1.0-1 requires that the coefficient linearly decrease from +7 pcm/°F to 0 pcm/°F from 70 percent to 100 percent of rated power.

In addition, evaluations are performed to assess the impact of an increase in the boron concentration range for the RWST and accumulators. The results of these evaluations show that the increases in allowable borcn concentration have no adverse impact on safety.

The proposed technical specification changes as a result of PMTC and the RWST and accumulator boron concentration are contained in Section 6.0.

9.0 REFERENCES

1. "Stress Corrosion Testing", WCAP-7628, non-proprietary, December, 1970.

2. Branch Technical Position MTEB 6-1, "pH for Emergency Coolant Water for PWR's"

3. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-0800, Section 6.5.2, Rev. 1, July, 1981.

4. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-0800, Section 6.4, Rev. 2, July, 1981.

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

6. Risher, D. H., Jr. and Barry, R. F., "TWINKLE - A Multidimensional Neutron Kinetics Computer Code, "WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), 1975.

7. Hargrove, H. G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod," 1972.

8. Hochreiter, L. E.; Chelemer, H.; and Chu, P. T., "Subchannel Thermal Analysis of Rod Bundle Cores, "WCAP-7015, Revision 1, 1969.

9. Chelemer, H.; Weisman J.; and Tong L. S., "Subchannel Thermal Analysis of Rod Bundle Cores, "WCAP-7015, Revision 1, 1969.

10. "American National Standard for Decay Heat Power in Light Water Reactors, "ANSI/ANS-5.1-1979, August, 1979.

11. NUREG/CR-009, Technological Bases for Model of Spray Washout of Airborne Contaminants in Containment Vessels, October 1978.

12. Georgia Power Letter to NRC, Bailey to Youngblood, GN-808, 2/19/86.

13. Regulatory Guide 1.78, June 1974.

14. Westinghouse Project Letter to GPC, L. Vota to F. B. Marsh, GP-11438, 8/25/86.