EXHIBIT B

This exhibit consists of the following pages revised to incorporate the proposed changes:

Page 16

Page 20

Page 21

Page 23

Page 24

Page 25

Page 26

Page 85

Page 118

Page 119

Page 134

Bases Continued:

2.1 During transient operation, the heat flux would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8-9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail (4,5,6,7). In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked each refueling outage to assure the insertion times are adequate. Exceeding a neutron flux scram setting and a delay in the control rod action to reduce neutron flux to less than the scram setting within 0.95 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed anytime a neutron flux scram setting of the APRM's is exceeded for longer than 0.95 seconds.

Analysis within the nominal uncertainty range of all appropriate significant parameters, show that if the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 0.95 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected.

The computer provided with Monticello has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram set point is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 2.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

(4) FSAR Volume I, Section III-2.2.3
(5) FSAR Volume III, Sections XIV-5
(6) Supplemenent on Transient Analyses submitted by NSP to the AEC February 13, 1973
(7) Letter from NSP to the AEC, "Planned Reactor Operation from 2,000 MWD/T to end of cycle 2", dated August 21,1973

Bases Continued:

2.3 For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 15% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps. During steady-state operation with one recirculation pump operating the equalizer line shall be open. Analysis of transients from this operating condition are less severe than the same transients from the two pump operation.

| The operator will set the APRM neutron flux trip setting no greater than that shown in Figure 2.3.1. However, the actual set point can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 18.

B. APRM Control Rod Block Trips - Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at a given recirculation flow rate, and thus protects against exceeding a MCHFR of 1.0. This rod block set point, which is automatically varied with recirculation flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The specified flow variable set point provides substantial margin from fuel damage, assuming steady state operation at the set point, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip point vs. flow relationship, therefore,

Bases Continued:

2.3 the worst case MCHFR during steady state operation is at 110% of rated power. Peaking factors as specified in Section 3.2 of the FSAR were considered. The total peaking factor was 3.08. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram setting, the APRM rod block setting is adjusted downward if peaking factors greater than 3.08 exist. This assures a rod block will occur before MCHFR becomes less than 1.0 even for this degraded case. The rod block setting is changed by changing the intercept point of the flow bias curve (keeping the slope constant); thus, the entire curve will be shifted downward.

The operator will set the APRM rod block trip settings no greater than that shown in Figure 2.3.1. However, the actual set point can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on Page 18.

C. Reactor Low Water Level Scram - The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained.

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual set point can be as much as 6 inches lower due to the deviations discussed on Page 18.

D. Reactor Low Low Water Level ECCS Initiation Trip Point - The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. The design of the ECCS components to meet the above criterion was dependent on three previously set parameters: the maximum break size, the low water level scram set point, and the ECCS initiation set point. To lower the set point for initiation of the ECCS could prevent the ECCS components from meeting their criterion. To raise the ECCS initiation set point would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor vessel pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel

2.4 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor Coolant High Pressure Scram shall be $\leq$ 1075 psig.

B. Reactor Coolant System Safety/Relief Valves Initiation shall be as follows:

4 valves at $\leq$ 1080 psig.

C. Reactor Coolant System Safety Valves Nominal Settings shall be as follows:

4 Valves at $\leq$ 1240 psig.

Bases:

2.2 The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured in the vessel steam space is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value was derived from the design pressures of the reactor pressure vessel, coolant piping, and recirculation pump casing. The respective design pressures are 1250 psig at 575°F, 1148 psig at 562°F, and 1400 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and the USAS Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10 percent over the vessel design pressure (110% x 1250 = 1375 psig) and the USAS Code permits pressure transients up to 20 percent over the piping design pressure (120% x 1148 = 1378 psig).

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig and temperature of 575°F; this is more than a factor of 1.5 below the yield strength of 42,300 psi at this temperature. At the pressure limit of 1375 psig, the general membrane stress increases to 29,400 psi, still safely below the yield strength.

The reactor coolant system piping provides a comparable margin of protection at the established pressure safety limit.

Bases Continued:

2.2 The normal operating pressure of the reactor coolant system is approximately 1025 psig. The turbine trip with failure of the bypass system represents the most severe primary system pressure increase resulting from an abnormal operational transient. The peak pressure in this transient is limited to 1214 psig. The safety valves are sized assuming no direct scram during MSIV closure. The only scram assumed is from an indirect means (high flux) and the pressure at the bottom of the vessel is limited to 1308 psig in this case. Reactor pressure is continuously monitored in the control room during operation on a 1500 psig full scale pressure recorder.

Bases:

2.4 The settings on the reactor high pressure scram, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

The reactor coolant system safety valves offer yet another protective feature for the reactor coolant system pressure safety limit. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, the safety valves must be set to open at a pressure no higher than 105 percent of design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety valves are sized according to the code for a condition of MSIV closure while operating at 1670 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety valves set as specified herein, the maximum vessel pressure (at the bottom of the pressure vessel) would be about 1308 psig. See FSAR Section 4.4.3 and supplemental information submitted February 13, 1973. Evaluations presented indicate that a total of eight valves (4 safety valves and 4 dual purpose safety/relief valves) set at the specified pressures maintain the peak pressure during the transient within the code of allowable and safety limit pressure.

The operator will set the reactor coolant high pressure scram trip setting at 1075 psig or lower. However, the actual setpoint can be as much as 10 psi above the 1075 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1080 psig or lower. However, the actual set point can be as much as 11 psi above the 1080 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or when a sufficient number of devices have been affected by any means

Bases Continued 3.3 and 4.3:

consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10% of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The consequences of a rod block monitor failure have been evaluated and reported in the Dresden II SAR Amendments 17 and 19. These evaluations, equally applicable to Monticello, show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFR's less than 1.0. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Engineer, Nuclear, to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCHFR from becoming less than 1.0. This requires the negative reactivity insertion in any local region of the core and in the over-all core to be equivalent to at least one dollar within 0.75 second. The required average scram times for three control rods in all two by two arrays and the required average scram times for all control rods are based on inserting this amount of negative reactivity locally and in the overall core, respectively, within 0.75 second. Under these conditions, the thermal limits are never reached during the transients requiring control rod scram as presented in the FSAR. The limiting operational transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCHFR remains greater than 1.8. In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	(b) When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least once per shift and analyzed for conductivity and chloride ion content.
4. If Specification 3.6.C.1, 3.6.C.2, and 3.6.C.3 are not met, normal orderly shutdown shall be initiated.	
D. Coolant Leakage	D. Coolant Leakage
Any time irradiated fuel is in the reactor vessel, and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, initiate an orderly shutdown and have the reactor placed in the cold shutdown condition within 24 hours.	Reactor coolant system leakage into the drywell shall be checked and recorded at least once per day.
E. Safety and Relief Valves	E. Safety and Relief Valves
1. During power operating conditions and whenever the reactor coolant pressure is greater than 110 psig and temperature greater than 345°F, four safety valves and the safety valve func-	1. A minimum of two safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. All four valves shall be checked or replaced

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
tion of four safety/relief valves shall be operable. The solenoid activated relief function of the safety/relief valves shall be operable as required by Specification 3.5.E.	every two refueling outages. The nominal popping point of the four safety valves shall be set at ≤ 1240 psig.
2. If specification 3.6.E.1 is not met, initiate an orderly shutdown and have coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.	2. a. A minimum of two safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. All four valves shall be checked or replaced every two refueling outages. The popping point of the safety/relief valves shall be set as follows:

Number of Valves	Set Point (psig)
4	≤ 1080

b. At least one of the safety/relief valves shall be disassembled and inspected each refueling outage.

c. The integrity of the safety/relief valve bellows shall be continuously monitored.

d. The operability of the bellows monitoring

Bases Continued 3.6 and 4.6:

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such leakage was coming from the primary system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

The capacity of the drywell sump pumps is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

The performance of the reactor coolant leakage detection system, including an evaluation of the speed and sensitivity of detection, will be evaluated during the first 18 months of plant operating, and the conclusions of this evaluation will be reported to the AEC. Modifications, if required, will be performed during the first refueling outage after AEC review. In addition, other techniques for detecting leaks and the applicability of these techniques to the Monticello Plant will be the subject of continued study.

E. Safety and Relief Valves

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. A tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of the set pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the set pressure, the reactor coolant pressure safety limit of 1375 psig is not exceeded. Safety/relief valves are used to minimize activation of the safety vaives. The operator will set the pressure settings at or below the settings listed. However, the actual setpoints can vary as listed in the basis of Specification 2.4.

The required safety valve steam flow capacity is determined by analyzing the pressure rise accompanying the main steam flow stoppage resulting from a MSIV closure with the reactor at 1670 MWt. The analysis assumes no MSIV closure scram, but a reactor scram from indirect means (high flux). The relief and safety valve capacity is assumed to total 83.9% (47% relief and 36.9% safety) of the full power steam generator rate. This capacity corresponds to assuming that four safety/relief valves (47%) and four safety valves (36.9%) operated.

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MONTICELLO - SAFETY VALVE SETPOINT INCREASE

I. INTRODUCTION

Analysis of the recent change to the exposed core scram reactivity insertion curve (GE December 1972 curve, curve C, Fig. 1) has resulted in the inability of Monticello to satisfy, near the end of cycle, the GE recommended 25 psi margin between the "worst case" pressurization type transient (turbine trip without bypass, i.e., relief valve sizing transient) and the setpoint of the first spring safety valve (1210 psig).

Calculations performed previously have shown that the current scram reactivity insertion curve (GE Generic '72, curve B, Fig. 1) will remain valid until a core exposure (R-1 reload) of 2400 MWD/T. The transient analyses based on this curve and submitted to the AEC on February 13, 1973, adequately describes the effects of the transients.

Beyond an exposure of 2400 MWD/T, additional measures are required to ensure the margins of the February 13, 1973 analysis are met. Of the short range remedies studied, a reduction in reactor power to 84% will adequately meet the requirement. A second alternative, increasing the spring safety valve setpoints, could also assist in maintaining the transient margins.

This report is intended to provide the analytical and administrative justifications for the safety valve setpoint change.

II. CONCLUSION AND RECOMMENDATIONS

The setpoints of all four spring safety valves can and should be reset to 1240 psig prior to a core exposure (R-1 reload) of 2400 MWD/T. This setpoint increase can be effected within appropriate codes and regulations, and maintains the required and recommended margins described in earlier documents.

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With the four spring safety valves set at 1240 psig and the four combination relief/safety valves set at 1080 psig (present setpoint), the reactor can be operated at 91% power from which the margin between peak pressure resulting from the relief valve sizing transient and the first safety valve will be 26 psi. This satisfies the GE recommended design guideline margin (25 psi).

The safety valve sizing analysis assumed a power level of 100% and yields a margin of 67 psi from the 1375 psig limit; operation at 91% (limited by the RV sizing transient) would result in larger margins. In conjunction with the setpoint change, Tech Specs must also be modified to include the settings and their bases.

III. DISCUSSION

A. Basis for Change

On February 13, 1973, NSP submitted to the AEC a report prepared by General Electric entitled "Results of Transient Reanalysis for Monticello Nuclear Generating Plant with End-of-Cycle Core Dynamic Characteristics." This report described the changes to the abnormal operational transient analysis as described in the FSAR caused by a shift in the scram reactivity feedback curve for exposed core conditions. Also included in that report were the proposed changes necessary to meet the General Electric guideline which is to maintain a margin of 25 psi between the peak pressure resulting from the "worst case" pressurization type transient (turbine trip without bypass, i.e., relief valve sizing transient) and the setpoint of the first spring safety valve.

On June 1, 1973, NSP submitted a proposed change to the Technical Specifications incorporating the results and recommendations of the February 13 report.

General Electric, on the basis of refined analytical techniques, informed NSP earlier this year that the shift in the scram reactivity curve is a function of core exposure, i.e., the present curve (Fig. 1, GE's generic '72, Curve B) does not fully represent the final end-of-cycle core condition. GE has determined that Curve B will define the scram reactivity function to a core exposure (R-1 reload) of 2400 MWD/T. Postulated transients using the assumptions from the analyses and occurring up to that exposure would not result in peak pressures in excess of that described in previous submittals. Beyond an exposure of 2400 MWD/T, additional measures are required to ensure satisfaction of the GE recommended design objective of a 25 psi margin between peak transient pressure and the safety valve setpoints. The margin between peak pressure and the reactor vessel pressure limit following the safety valve sizing transient (MSIV closure) remains well above the 25 psi required design margin; therefore, no safety limit is affected whether or not additional measures are taken.

General Electric has determined an "all rods out" scram reactivity curve (Fig. 1, Curve C) to define the worst case core conditions between 2400 MWD/T and the end of the current cycle for Monticello. Although actual conditions do not reach Curve C until the all-rods-out end-of-cycle exposure, Curve C was applied to determine what measures were necessary to ensure maintenance of the effects of transients throughout the remainder of the cycle in conformance with the earlier analysis.

Evaluations have been made for a rod movement "freeze" until power coasts down to 84% at 2400 MWD/T; this is sufficient to meet the 25 psi margin in the relief valve sizing transient. This operational restriction maintains the validity of the earlier transient analyses. Other measures such as the change discussed in this

report, are being developed; their application may aid further in mitigating the overall effects of the newer curves.

B. <u>Assumptions Used in Analysis</u>

The same generic assumptions as those used in the February 1973 submittal were applied to the transients described in this report.

Conservative assumptions, such as a multiplier on the void coefficient, a multiplier on the scram reactivity curve, and average control rod scram times equivalent to the '67 Product Line BWR, were used.

Because the new scram reactivity curve (Curve C) represents an exposed core condition, the new analyses were done with end-of-cycle inputs for consistency and to ensure that a realistic worst case would be defined. For example, the void coefficient is reduced at the end of the cycle and this will tend to reduce the peak of the pressurization transients.

The scram reactivity curve (Curve C) is a function of core exposure and will not be approached until near the end of cycle; however, the curve is assumed to apply from 2400 MWD/T to the cycle end for the determination of 100% power transient effects.

In the analyses of this report, the four combination relief/safety valves are assumed operable as described in the February 1973 report. The four spring safety valves are also assumed operable with a setpoint of 1240 psig.

Because a complete set of transient analyses is not required, only the transients of most concern were redone. These were the turbine trip without bypass transient for checking relief valve adequacy and the MSLIV closure with indirect scram for checking safety valve adequacy.

C. Transients Not Reanalyzed

The FSAR included about 20 analyses of worst case abnormal transients in six categories of events. These categories are primary system pressure increases, moderator temperature decreases, reactivity insertions, core coolant inventory decreases, core coolant flow decreases, and core coolant flow decreases. These were all reviewed to determine those which might be significantly affected by the new end-of-cycle core characteristics assumptions. The breakdown of categories, events and logic for those in which only a review was deemed to be adequate, is described in the analysis submitted in February 1973. Reiteration of that discussion is unnecessary here.

D. Results of Transients Reanalyses

1. Scope of Reanalyses

The following transients were reanalyzed in order to determine the specific changes that might occur to the previous analytical results:

Turbine trip without bypass (Relief valve adequacy check)

Main Steam Isolation Valve Closure, (includes delayed scram case for safety valve adequacy check)

Specific write-ups for these analyses are included herein.

The original FSAR analysis used the turbine trip without bypass with flux scram for the safety valve sizing transient. However, analyses of later plants revealed that the main steam line isolation with flux scram could be more severe. During the reanalyses work reported in February 1973, this possibility was checked by performing both analyses and the results showed a somewhat higher peak pressure with main steam isolation valve closure. This analysis is used for checking safety valve adequacy in this report as well.

2. Turbine Trip Without Bypass - Relief Valve Adequacy Transient

Reactor operating at 91% of rated, core flow 100%, 67 product line scram times (data interpolated from several cases run from 85 to 100% power):

A scram signal is initiated at the same time a turbine trip occurs by position switches on the turbine stop valves. This transient causes a rapid pressure increase in the reactor pressure vessel. Primary system relief valves are provided to remove sufficient energy from the reactor to prevent safety valves from lifting. Reanalysis showed that peak pressure in the steam line at the safety valve location did not meet the GE margin of 25 psi to the first safety valve setpoint (1210 psig). However, a peak pressure in the steam line of 1214 psig at the safety valve location provides an adequate margin of 26 psi to 1240 psig, the recommended first safety valve setpoint. Thus, the adequacy of the four relief valves was confirmed for these conditions. Using the parameters associated with the end of life conditions, four relief/safety valves are required to operate as described in the February 1973 report to prevent this pressure transient from exceeding the safety valve setpoint. The rapid pressure rise due to

rapid closure (0.10 sec.) of the turbine stop valve without bypass operation causes core voids to collapse and neutron flux peaks at 262 percent of design in 0.92 sec. (Figure 3) before the scram shuts down the reactor. Peak surface heat flux is 100.3% at 1.36 sec. MCHFR and other pertinent parameters remain within acceptable limits.

3. Closure of All Main Steam Line Isolation Valves (Flux Scram) - Safety Valve Adequacy Transient

The ASME Nuclear Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from the consequence of pressure and temperature in excess of design conditions. The ASA Code for Pressure Piping also requires overpressure protection. The setpoints of the safety valves comply with the ASME pressure vessel code taking into account static heads and dynamic losses.

This was discussed at length in the February 1973 report and is included here for completeness.

The change in safety valve setpoints described in this report will meet appropriate codes and regulations (ASME NB & PV Code Sect. XI).

To determine the required flow capacity of the safety valves, it is assumed that:

a. The reactor is at 1670 MWt,

b. The reactor experiences its worst main steam isolation transient,

c. Direct reactor scram is neglected (based on isolation valve position switches),

d. The backup scram due to high neutron flux shuts down the reactor,

e. The Target Rock relief valves act as safety valves with low setpoints.

Both a turbine trip without bypass and closure of all main steam line isolation valves produce severe overpressure transients. Analyses for these two events have shown that the 3 second closure of the isolation valves is slightly more severe for the final plant configuration when direct reactor scram is neglected. This results because the longer steam lines, allowing more volume for steam compression, more than compensates for the faster acting turbine stop valves in the former transient when compared with MSLIV closure. The latter transient is therefore provided here as the basis for determining the adequacy of the safety valves.

Pressure increases follow this reactor isolation until limited by the opening of the safety valves. The peak allowable pressure is 1375 psig (according to ASME Section III, equal to 110 percent of the vessel design pressure of 1250 psig). The Target Rock setpoints are $\leq$ 1080 psig and the spring safety valve setpoints are at 1240 psig (4 valves). Thus the ASME code specifications that the lowest safety valve be set at or below vessel design

pressure, and the highest safety valve be set to open at or below 105 percent of vessel design pressure are satisfied. The four spring valves together have a capacity of greater than 35 percent of turbine design flow.

The resulting transient assuming the capacity of the 4 safety/relief valves (47% of main steam generation rate) and the 4 safety valves (36.9% of main steam generation rate) is shown in Figure 4. An abrupt pressure and power rise occurs as soon as the isolation becomes effective. Neutron flux causes the scram at approximately 1.8 seconds thereby initiating reactor shutdown. Flux peaks at a value of 644 percent in 2.14 sec. The assumed safety valve capacity (Target Rock plus spring safety capacities) keeps the peak vessel pressure 67 psi below the peak allowable ASME overpressure of 1375 psig. Therefore, the relief valves plus the spring safety valves provide adequate protection against excessive overpressurization of the nuclear system process barrier with an adequate margin.

IV TECHNICAL SPECIFICATION CHANGES

A. Scope of Changes

The principle changes of interest concern the safety valve setpoints. This is needed to be consistent with the new assumptions used in the transient reanalyses and is discussed in detail in Section III. B. Other changes are those associated with the results of the transient reanalyses discussed in Section III. D. None of these are of a crucial safety nature and mostly affect statements about margins for various pressurization transients. Tech Spec changes submitted to the AEC June 1, 1973, are considered to be in effect; errors or omissions related to the February 1973 report and June 1, 1973 submittal have been corrected or added.

B. Specific Changes

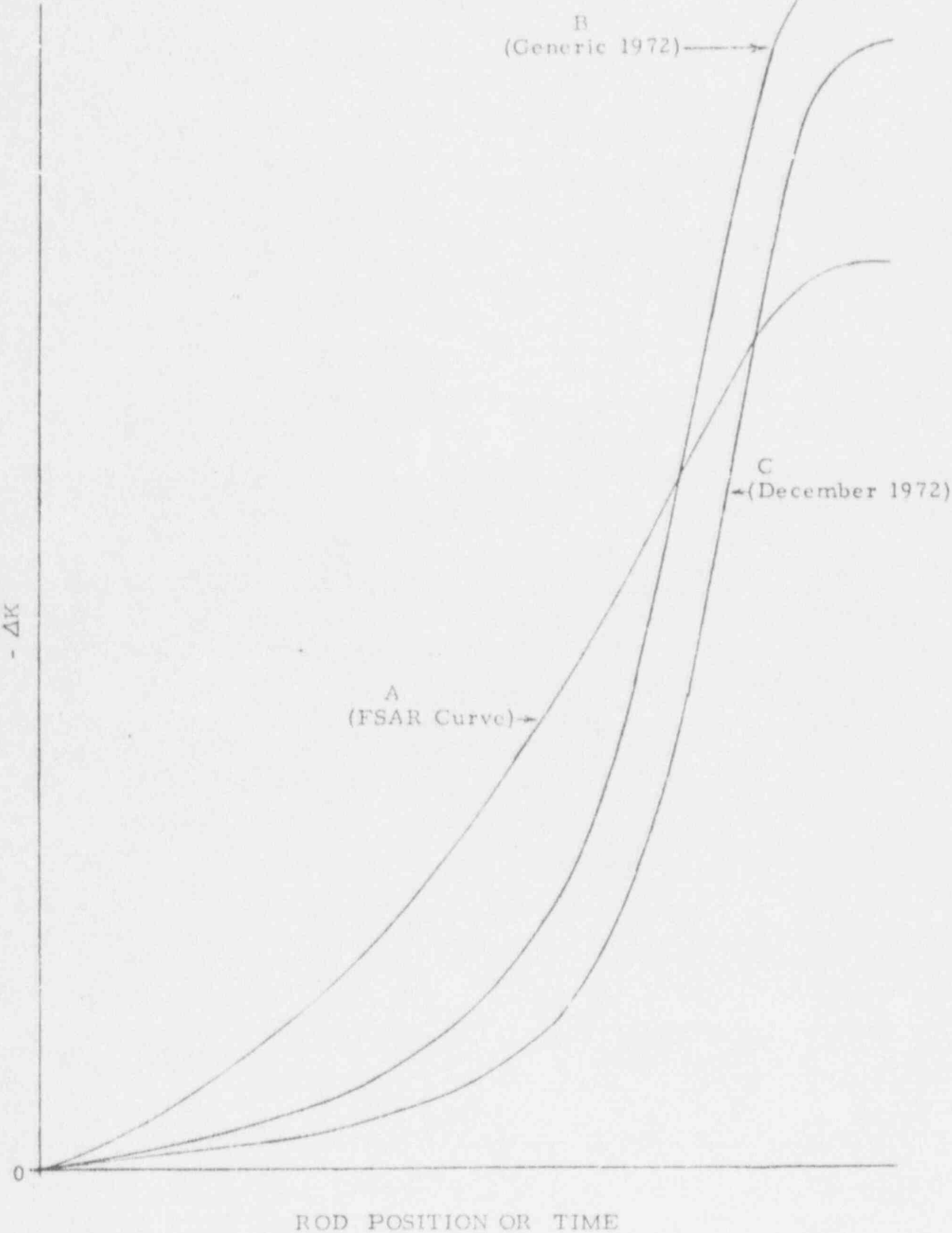
Item	Location	Change	Reason
Bases statement for 2.1	Pg. 16	Add reference to the February 1973 submittal and this analysis	Indicates documentation of discussions on this topic.
Bases statement for 2.3.A	Pg. 20 - end of third para.	Change "...Pg. 22." to "... Pg. 18."	Corrects typographical error
Bases statement for 2.3.B	Pg. 21 - end of second para.	Change "... Pg. 22." to "... Pg. 18."	Corrects typographical error
Bases statement for 2.3.C	Pg. 21 - end of second para.	Change "...Pg. 22." to "... Pg. 18."	Corrects typographical error
Tech. Spec. 2.4.C	Pg. 23	Change "2 valves at ≤ 1210 psig." and "2 valves at ≤ 1220 psig" to "4 valves at ≤ 1240 psig."	This change reflects an assumption of this analysis.
Bases statement for 2.2	Pg. 24 last para. Pg. 25 top of pg.	Change to read as follows: "The normal operating pressure of the reactor coolant system is approximately 1025 psig. The turbine trip from 91% power with failure of the bypass system represents the most severe primary system pressure increase resulting from an abnormal operational transient. The peak pressure in this transient is 1214 psig.	This change provides the basi for the valve configurati used in this analysis

Item	Location	Change	Reason
		In addition, the safety valves are sized on the basis of a closure of all Main Steam Isolation Valves (MSIV Closure) where scram is assumed to be indirect (high flux) rather than from the MSIV position switches. In this transient, assuming rated power, the pressure at the bottom of the vessel is 1308 psig. Reactor pressure is continuously monitored in the control room during operation on a 1500 psig full-scale pressure recorder.	
Basis statement for 2.4	Pg. 26, Para 2, Line 9	Change 1283 psig to 1308 psig	This change reflects the results of this analysis.
	Pg. 26, Para 2, Line 10	Change to read: "... a total of eight valves (4 safety valves and 4 dual purpose safety relief valves) set at..."	This change reflects an assumption of this analysis.
	Pg. 26, Para 3, Lines 3 and 6	Change "...Page 22" to "...Page 18."	Corrects typographical error

Item	Location	Change	Reason
Basis statement for 3.3.C/4.3.C	Pg. 85, Lines 8 and 9	Change to read: "The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system."	Restores original statement erroneously changed in February 1973 submittal.
Tech. Spec. 3.6.E.1	Page 118, Last Line	Change "... three safety valves ..." to "... four safety valves..."	This change reflects an assumption in this analyses
Tech Spec 4.6.E.1	Pg. 119	Change to read as follows: "... every two refueling outages. The nominal popping point of the four safety valves shall be set at $\leq$ 1240 psig."	This change reflects an assumption in this analysis
Bases statement for 3.6.E/4.6.E	Pg. 134, Last Para., Line 4	Change RV/SV capacity as follows: "...to total 83.9% (47% relief and 36.9% safety) of ..."	These changes reflect assumptions in this analysis
	Line 5	"...assuming that four relief safety valves (47%) and four safety valves (36.9%) operated."	
	Lines 6 and 7	Delete entire last sentence.	

Figure 1

MONTICELLO



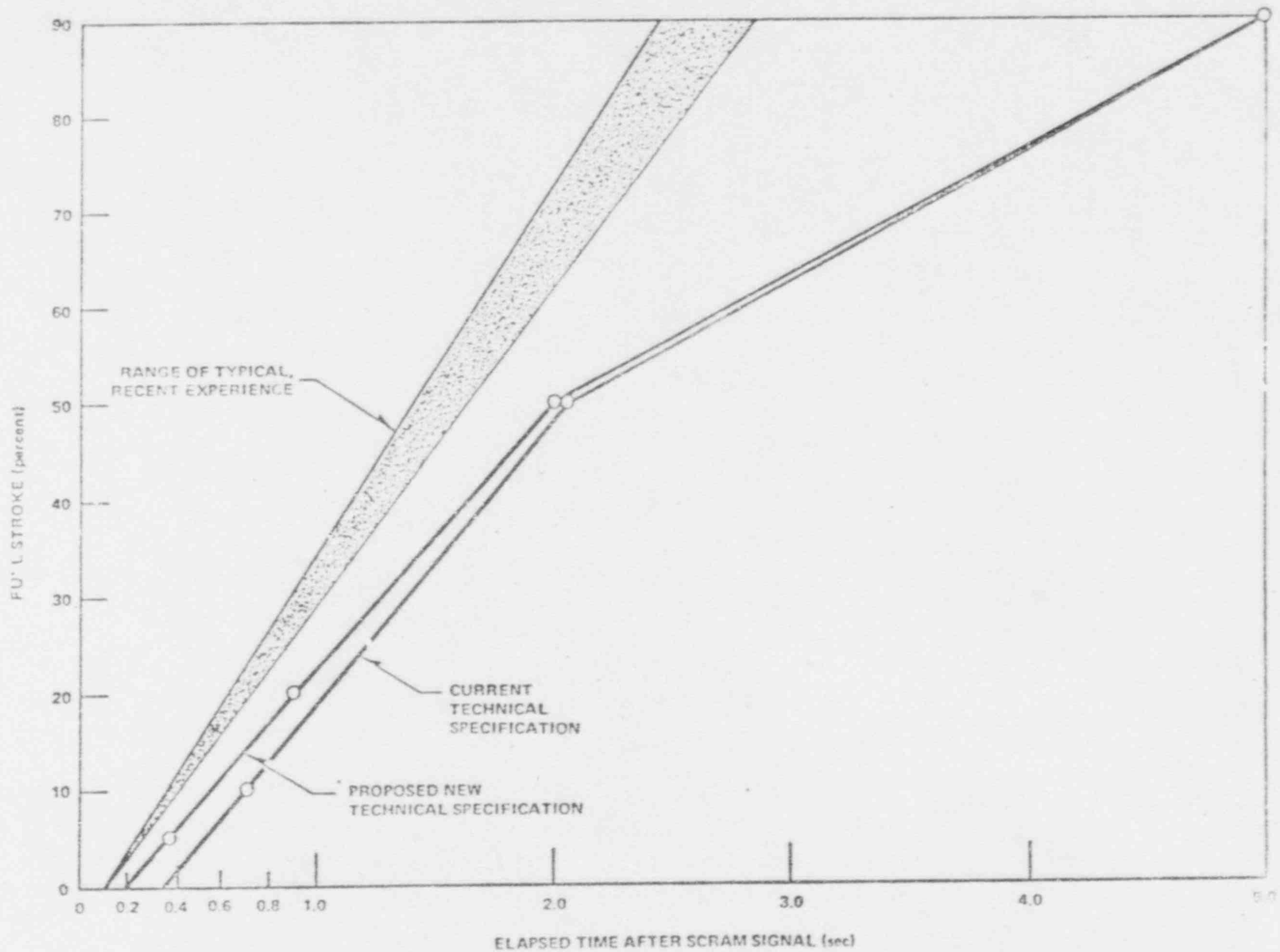


FIGURE 2. CONTROL ROD DRIVE SCRAM TIMES — MONTICELLO

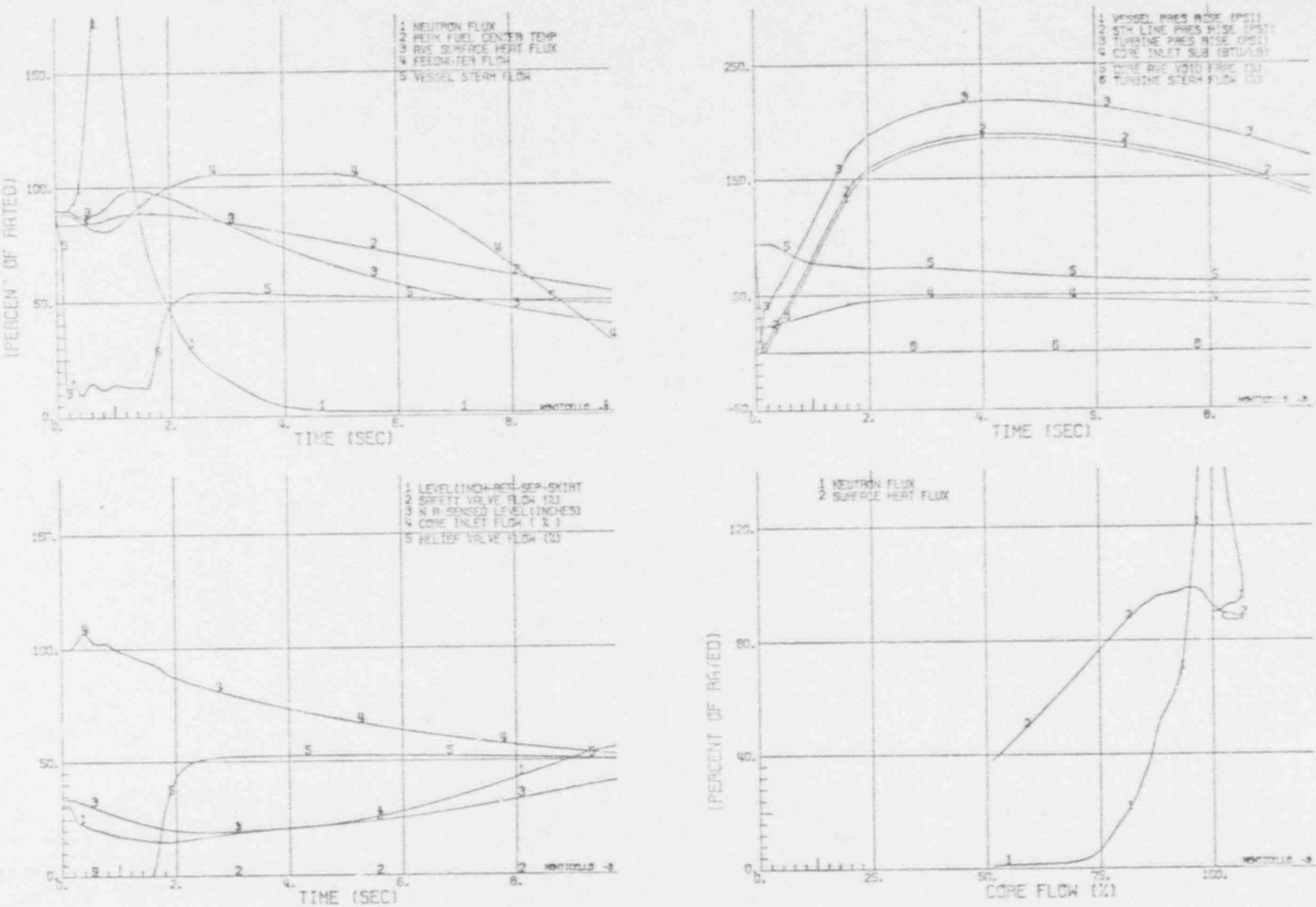


Figure 3

MONT RLD7C
TT W/O BP, 80% 67PL SCRM, 47% RV .80LY, 1.28 VOID MULT W/2 PUMP TRIP

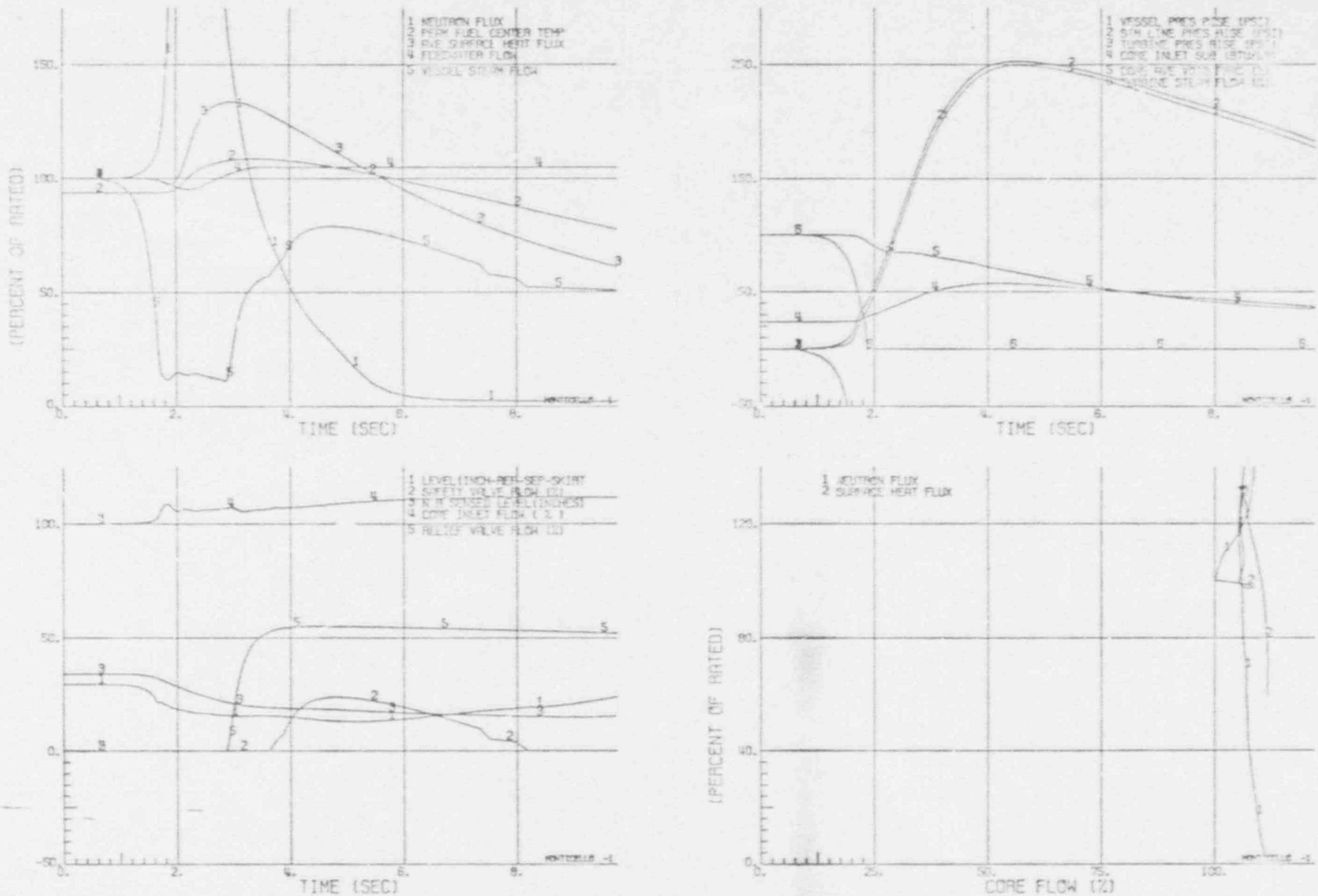


Figure 4

MONT RLD100
ALL MSIV CLS FLUX SCRM 47% RLF 1096SP 37% SV 1268SP .8SEC DLY 67PL
37% SAFETY

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(TEMPORARY FORM)

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Mr. O'Leary	3 signed			SENT AEC PDR X SENT LOCAL PDR X

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ENCLOSURES:
Request for Authorization to Change Appendix A of the Tech Specs, notarized 9-13-73.......W/Attached Rpt, "Monticello - Safety Valve Setpoint Increase".

(40 cys rec'd)

ACKNOWLEDGED
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BOYD	STELLO		SHEPPARD (E)	
MOORE (L)(BWR)	HOUSTON	ENVIRO	SMITH (L)	INFO
DEYOUNG(L)(PWR)	NOVAK	MULLER	TEETS (L)	C. MILES
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1-GERALD ULRIKSON...ORNL

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