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PLANT NAMES: Monticello

ENCLOSURES:
SUPPL REPORT: "Results of Teansient Reanalysis for Monticello Nuclear Generating Plant with End-of-Cycle Core Dynamic Characteristics".

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Mr. A Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D C 20545

Dear Mr. Giambusso:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Supplemental Report of a Change in the
Transient Analysis as Described in the FSAR

On August 14, 1972 we informed you that results of an analysis of reactor transients for the end of cycle differed from those presented in the FSAR. We reported that the analysis assumed control rod scram times faster than required in our present Technical Specifications. In our letter we noted that recently measured scram times were approximately half of those used in the analysis; continued operation was therefore justified. We have since analyzed our control rod drive system and its past performance. We see no reason for significant increases in the attainable scram times in the future; therefore, we can accomodate revisions to our Technical Specifications to reflect the faster scram times used in the analysis.

Attached is a report prepared by General Electric entitled "Results of Transient Reanalysis for Monticello Nuclear Generating Plant with End-of-Cycle Core Dynamic Characteristics." Refinements in modeling the transient conditions have shown a shift in the scram reactivity feedback curve for the exposed core having all rods out. Since we do not plan to extend the present operating cycle to the all-rod-out condition, and because of the relatively low average core exposure, we do not expect the FSAR analysis results to be exceeded during the present cycle. The cycle subsequent to the spring, 1973 reload will begin with more control rods in the core; a scram from this condition will fall within the FSAR transient analysis results. The results of the transient analysis are, therefore, not representative until later in the second cycle.

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Technical Specification changes are recommended by General Electric in Section IV of the attached report. We will in the near future, submit a request for the recommended Technical Specification changes, but in a format and wording consistent with our current Technical Specifications. The changes will include revised scram times, required operability of the fourth safety/relief valve, and appropriate wording changes in the Bases section. Analytical studies of the effects of increased exposure on future cycles represents a continuing effort. We will advise you should these studies reflect a response in future cycles different from that described in the FSAR and the attached report.

Yours very truly,

L.O. Mayer

L O Mayer, P.E.
Director of Nuclear Support Services

LOM/MHV/br

cc: B H Grier
G Charnoff
Minnesota Pollution Control Agency
Attn. Ken Dzugan

RESULTS OF TRANSIENT REANALYSES FOR MONTICELLO NUCLEAR GENERATING PLANT WITH END-OF-CYCLE CORE DYNAMIC CHARACTERISTICS

I. Introduction

A recent reanalysis has shown that a significant change in the shape of the scram reactivity curve could occur by the end of a fuel cycle (see fig.1). As can be seen, even though the total scram reactivity has increased somewhat, the insertion rate is slower at the beginning of the stroke where the transient analyses described in the FSAR for single-event caused abnormal occurrences could be affected. This, in turn, could affect design, operational, and safety provisions derived from these analyses for such things as the relief and safety valve capacities, set points, and Technical Specification requirements. For this reason, all transient analyses previously performed for Monticello have been reviewed, those which affect relief and safety valve settings and capacity have been redone, and necessary Technical Specification revisions determined.

II. Conclusions and Recommendations

When the analyses were completed, all the design guidelines, margins, criteria, etc. were met with the exception of the relief valve adequacy transient (turbine trip without bypass). In this case, the margin from the peak of the pressure transient to the setting of the first safety valve was not sufficient to meet the General Electric recommended design guideline minimum of 25 psi. In order to meet this guideline, a slight improvement in the control rod drive scram time was assumed, to that being used on the '67 and later GE-BWR Product Line plants (see Fig. 2). With the improved scram time assumption, the calculated margin was 27 psi, which is acceptable.

In conjunction with this new assumption, the corresponding Technical Specification change is recommended and included later in this document. Other appropriate proposed Technical Specification changes based on the results of the reanalysis are included at the end of this document, but do not involve any major changes in safety philosophy.

III. Discussion

A. Basis for Changes

It has been recognized in the past that there could be substantial changes in axial reactivity characteristics with increased exposure which could affect the shape of the scram reactivity curve. However, it had been assumed that during most of the fuel cycle, enough stubbed (partially inserted) rods were available to effect a fast scram reactivity rate at the beginning of rod stroke. On the other hand, even though all rods could be out at the end of the fuel cycle, flux would be peaked at the bottom of the core and this, combined with other exposed core characteristics was expected to be

sufficient to still obtain a relatively fast scram reactivity rate at the beginning of rod stroke. Thus, the old curve was previously judged to be adequate to cover the worst of these cases without being so extreme as to unduly penalize the plant.

Later generic development work was based on following the Haling principle which establishes an axial flux distribution which peaks flux toward the bottom of core early in the fuel cycle to reduce reactivity at that location while the control rods were still there to control it. This procedure prevents limiting flux peaks from developing towards the bottom of the core when all rods are out and voids are reducing reactivity at the top of the core. This principle has been successfully applied to core management of earlier plants and reload applications.

With the rods all the way out (or the resulting tendency to all rods out/in patterns) and a reduced flux peak at the bottom of the core, there was concern that the early part of the scram stroke might not be as effective, i.e., the scram reactivity insertion might be slower at the beginning of the rod stroke. Information from operating plants has confirmed this tendency. Further, improved analytical capability allowed a more refined calculation of scram reactivity characteristics for exposed cores.

B. Assumptions Used in Reanalyses

Because the new scram reactivity curve represents an end-of-cycle condition, the new analyses were also done with other inputs at end-of-cycle conditions for consistency and to ensure that a realistic worst case would be found between the two sets of conditions. 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.

Other conservative assumptions used in the original transient analyses, such as a multiplier on the void coefficient, and average control rod scram times equivalent to the Technical Specification limit, were also used in the reanalysis. However, when the reanalyses were completed, the minimum acceptable design margin from the peak of the pressure transient to the setting of the first safety valve could not be met on the turbine trip without bypass transient, (i.e., the General Electric recommended minimum design margin for such transients is 25 psi). To meet this margin, it is proposed that slightly improved control rod drive scram times be assumed, the same as those used for later plants, such as Vermont Yankee and Browns Ferry.

The control rod drive equipment is the same at Monticello as at later plants and is easily capable of meeting the improved scram time requirement, as illustrated in Figure 2. As can be seen, the improvement is in the early part of the scram stroke where it can be of most benefit to the results of the transient analyses.

Because it was undesirable to await another complete set of transient analyses, only the transients of most concern were redone. These were the Turbine trip without bypass transient (identical to instantaneous loss of condenser vacuum transient) for checking relief valve adequacy and the MSLIV valve closure with indirect scram for checking safety valve adequacy.) The resulting set of curves based on the different assumptions have been appropriately labeled and are included herein. However, the discussion only includes a statement

about the margins they demonstrate.

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 increase, 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 shown below.

Category	Event	Reasons Reanalysis Not Needed
Nuclear System Pressure Increase	All except those identified in IIIB	These events are less severe than those analyzed.
Moderator Temperature Decrease	All	All these events are less severe than those analyzed. The only event of significance is the feedwater controller failure, maximum demand, which is terminated by the high level turbine trip resulting in a pressurization transient which is less severe than the turbine trip without bypass transient analyzed herein.
Reactivity Insertion	Rod Withdrawal error	These transients are terminated by the rod block monitor and not a scram, so they will not be affected by the scram reactivity curve change. Other core characteristics will not change sufficiently to significantly affect the outcome of the analysis or the block setpoint.
Decrease of Coolant Inventory	All	These transients result in a RPV depressurization and, in some cases, a low level scram. Power level drops due to void formation before the scram, and MCHFR effects are minimal. A mild repressurization on MSLIV closure at 850 psig occurs on some. RPV temperature transients are the only concern on some.

Category	Event	Reasons Reanalysis Not Needed
Core Coolant Flow Increase	All	These transients are not severe and are affected by the reactivity increase due to sweeping out voids from the core and the addition of colder water to the core. The lower void coefficient at the end of life conditions makes these transients less severe than as previously analyzed.
Core Coolant Flow Decrease	All	A scram does not occur as a direct result of this transient, so the void coefficient change will be the principal effect changing the results of these transients. The change in void coefficient is not sufficient to significantly affect the the results. Start up test results where actual recirculation pump trips were conducted, demonstrate that these transients are of a mild nature.
Others		Any other transient analyses conducted ancillary to the standard ones or for other special purposes, such as a DC power interruption, do not include considerations pertinent to this discussion and are therefore not included.

D. Results of Transient 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, together with the transient curves obtained, are included herein. Curves are shown for the turbine trip without bypass, relief valve adequacy transient and the MSLIV closure transient, safety valve adequacy transient. This corresponds with the extra analysis done with the improved control rod drive scram time assumption to show adequate design margins.

It should be noted that the original FSAR analysis used for the safety valve sizing transient was the turbine trip without bypass with flux scram. However, it was determined with later plants that the main steam line isolation

with flux scram could be more severe. During the reanalysis work reported herein, this possibility was checked by performing both analyses and the results showed a somewhat higher peak pressure with main steam isolation valve closure. Hence, this analysis is used for checking safety valve adequacy in this report.

The dwell time is a special Technical Specification requirement to take account of an AEC concern about the possibility of a delayed scram. It was arbitrarily assumed that one of the more probable of the transients, the turbine trip with bypass, occurred and a neutron flux initiated scram was delayed until the safety limit was reached. The amount of the delay, 0.95 secs. for Monticello, is a Technical Specification. Analyses performed for all other plants (e.g. Millstone) have shown that the neutron flux peaks are lower and broader than previous analyses have shown (Turbine Trip with bypass transient). This is as a result of the lower void coefficient which reduces the rate of reactivity insertion and slows the progress of the transient. The dwell time for the Monticello plant calculated for end of life conditions will therefore be longer than that calculated by previous analysis. Because the previous analysis represents a worst case condition, those results are retained as the technical specification requirement.

2. Turbine Trip Without Bypass - Relief Valve Adequacy Transient

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. The initial 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 safety valve set point. Hence, the transient was reanalyzed using the improved control rod drive scram time assumption discussed previously and illustrated in Figure 2. The results are shown in Figure 3. The peak pressure in the steam line at the safety valve location was 1183 psig, which provided an adequate margin of 27 psi to the first safety valve set point. 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 to prevent this pressure transient from exceeding the safety valve set point. 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 reaches 241 percent of design (Figure 3) before the scram shuts down the reactor. Peak surface heat flux is less than 110 percent (Figure 3) thus adequate thermal margins are maintained.

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 set points of the safety valves comply with the ASME pressure vessel code taking into account static heads and dynamic losses.

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 set points.

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 set points are $\leq$ 1080 psig and the spring safety valve set points are at 1210 psig (2 valves) and 1220 psig (2 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 nameplate capacity greater than 35 percent of turbine design flow.

Figure 4 shows the resulting transient assuming the capacity of only 3 of the 4 relief/safety valves (35% of main steam generation rate) and only 2 of the 4 safety valves (18% of main steam generation rate). An abrupt pressure and power rise occur as soon as the isolation becomes effective. Neutron flux reaches scram at approximately 1.8 seconds initiating reactor shutdown. It peaks at a value of 610 percent. Peak fuel surface heat flux is slower, reaching a peak of 127 percent at about 27 seconds. The assumed safety valve capacity (Target Rock plus spring safety capacities) keeps the peak vessel pressure 92 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 a large margin, because of the reduced capacities assumed for this analysis.

IV Technical Specification Changes

A. Scope of Changes

The principle changes of interest concern the slightly improved control rod scram times. This is needed to be consistent with the new assumptions used in the transient reanalyses and is discussed in detail in Section IIIB. Other changes are those associated with the results of the transient reanalyses discussed in Section IIID. None of these are of a crucial safety nature and mostly affect statements about margins for various pressurization transients.

B. Specific Changes

ITEM	LOCATION	CHANGE	REASON
Basis statement for 2.3.F	Pg.,22	in last sentence-105% to 110% and 1.9 to 1.8	This reflects the results of this new analysis.
Basis statement for 2.2	last para. Pg.24	in the second sentence change "turbine trip" to closure of all the main steam line isolation valves. (MSLIV).	This reflects the results of the new analysis
		in the third sentence change 1187 psig to-1183 psig	This reflects the results of the new analysis
		in the last sentence change "turbine trip valve" to-MSLIV position	This reflects the results of the new analysis
Basis statement for 2.2	Pg. 25	change 1293 psig to-1283 psig	This reflects the results of this new analysis
Basis statement for 2.4	second para. Pg. 26	in the sixth line change "turbine trip" to-MSLIV position.	This reflects the results of this new analysis
		in the ninth line change 1293 psig to-1283 psig. Delete "Section 4.4.3 FSAR" and reference this analysis. Delete" in the FSAR" and reference this analysis.	The adequacy of the safety valves is verified and presented in this analysis

ITEM	LOCATION	CHANGE	REASON
Basis statement for 3.1	Pg.39 first para.	in the sixth and seventh line delete reference to the FSAR and reference this analysis	This transient has been reanalyzed and the results are presented in this analysis
	Pg.39 Third para.	in the last line delete reference to the FSAR and reference this analysis	This transient has been reanalyzed and the results are presented in this analysis
Spec 3.3.C.1	Table On Pg. 79	Change to the following:	The transient reanalyses were done with these new scram time requirements

% Inserted From Fully Withdrawn	Average Scram Insertion Time (secs)
5	0.375
20	0.900
50	2.00
90	5.00

ITEM	LOCATION	CHANGE	REASON
Spec. 3.3.C.2	Table on Pg. 79	Change to the following:	The transient reanalyses were done with these new scram time requirements.

% Inserted From Fully Withdrawn	Average Scram Insertion Time (secs)
5	0.398
20	0.954
50	2.120
90	5.300

ITEM	LOCATION	CHANGE	REASON
Basis statement for 3.3 and 4.3	Pg 82, A.1 First para.	in the eleventh line, change the reference to refer to this analysis	This is in conjunction with the new scram reactivity curve used in this analysis
	Pg 85 para. C	starting in the eighth line, change "turbine stop valve closure" to closure of the main steam isolation valves	This is consistent with the results of this new analysis
		in the eleventh line, change 1.9 to 1.8	This is consistent with the results of this new analysis

ITEM	LOCATION	CHANGE	REASON
		in the twelfth line, change 390 to 290	This is in conjunction with the changed scram time requirement and is consistent with current design practic on Transient analyses completed on other plants
		in the thirteenth line delete all remaining after. "This is-----" through the end of the paragraph on pg. 86 and replace with - This is adequate and conservative when compared with the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of the time interval results from the sensor and circuit delays; at this point the pilot scram solenoid deenergized. Approximately 120 milliseconds later control rod motion is estimated to begin. However, to be conservative, control rod motion is not assumed to start until 200 milliseconds later. This value was included in the transient analyses and is included in the allowable scram insertion times of specification 3.3.C.1 and 3.3.C.2.	This is in conjunction with the changes in the scram time and to be consistent with this analysis.
Basis statement for 3.6 and 4.6	Pg 134 Para E.	In the second line of the third paragraph delete "turbine trip initiated" and replace with-MSLIV closure.	All changes on the page are to be consistent with this analysis

ITEM	LOCATION	CHANGE	REASON
		In the third line delete "no steam bypass system flow and change "turbine" to-MSLIV and "trip" to-position In the fifth line change (35.4%) to -(35%) and in the sixth line change (18-5%) to -(18%).	

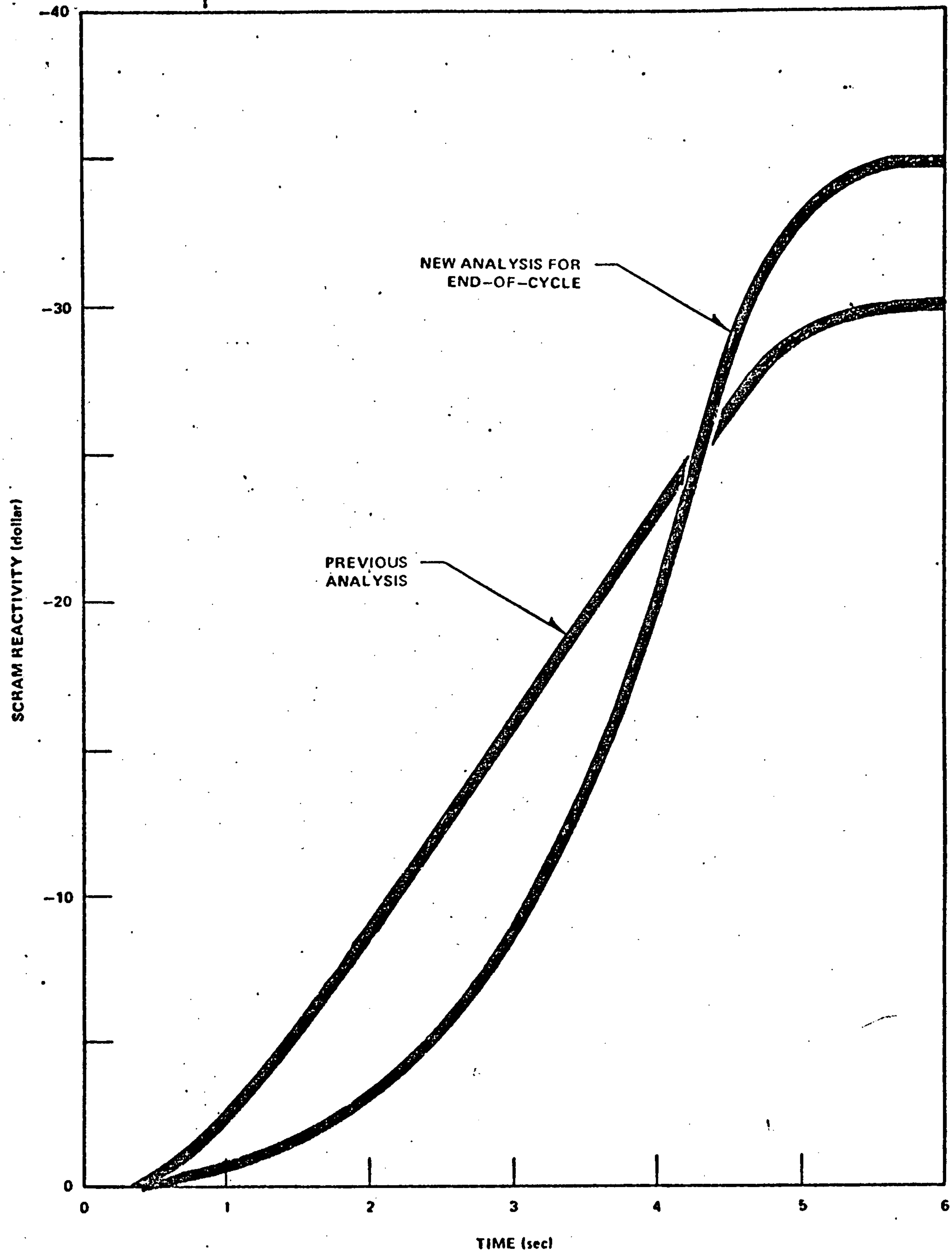

FIGURE 1. SCRAM REACTIVITY CURVES — MONTICELLO

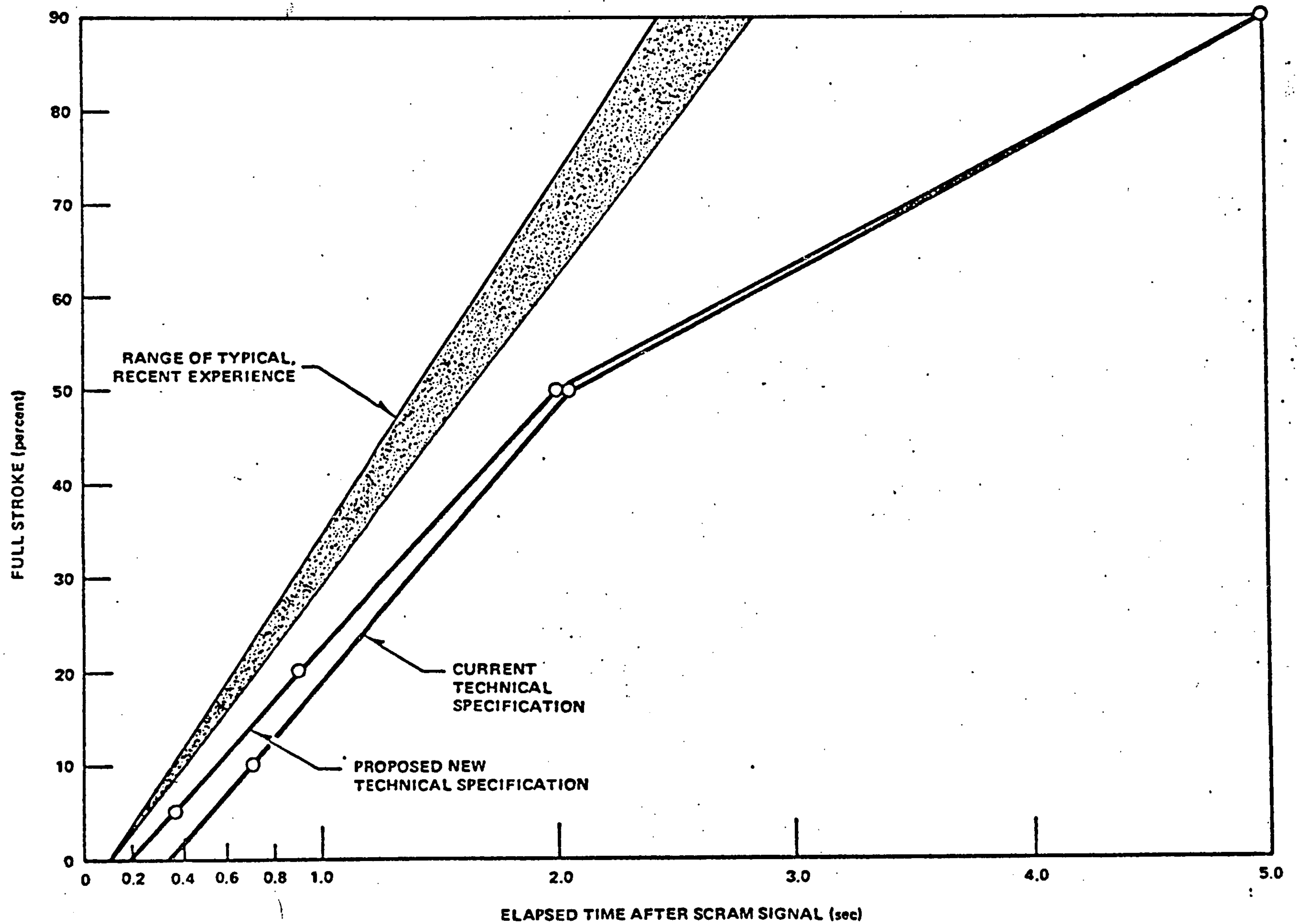

FIGURE 2. CONTROL ROD DRIVE SCRAM TIMES – MONTICELLO

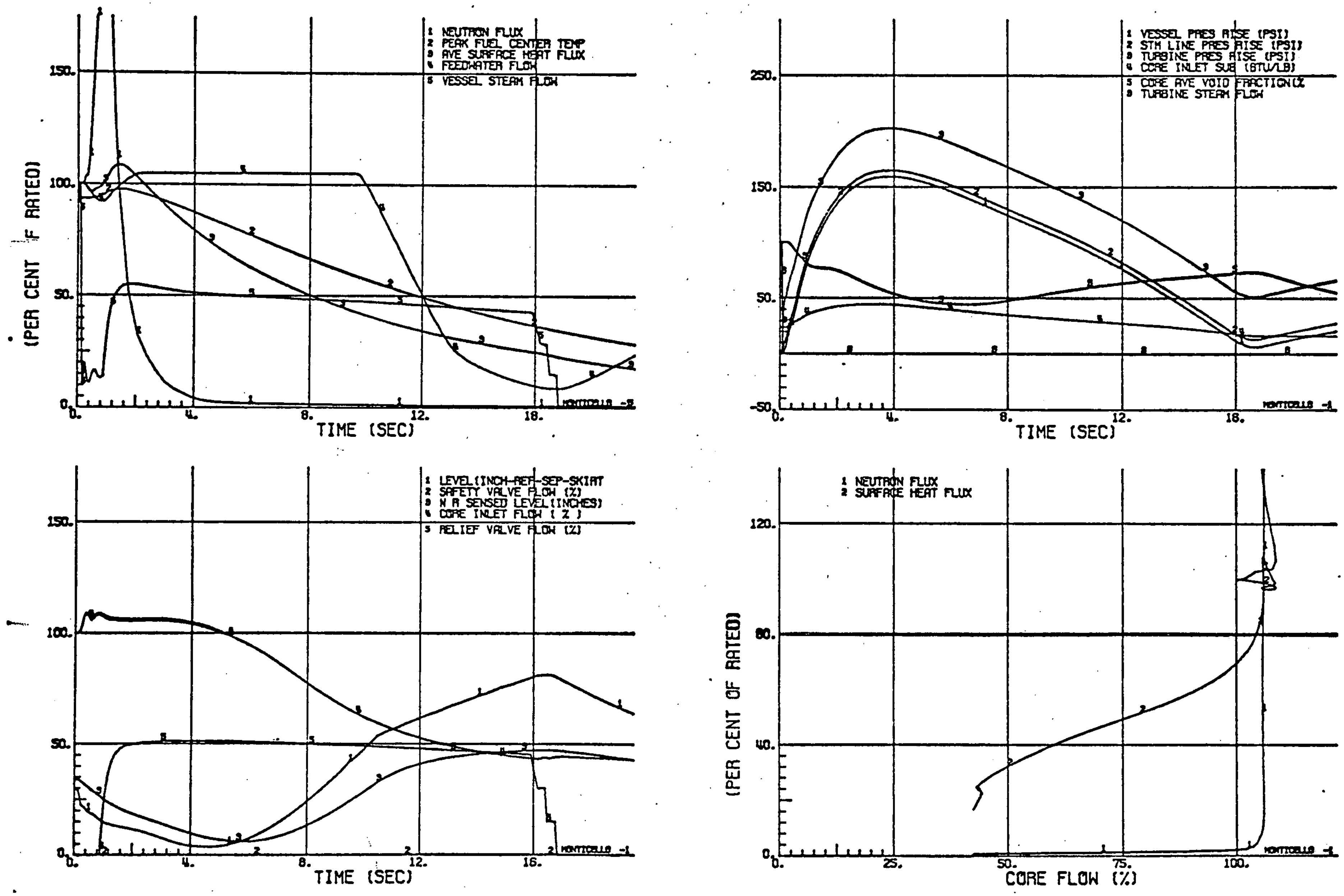

FIGURE 3.

MONT ALD4E
TURBINE TRIP WITHOUT BYPASS, TRIP SCRAM 47% RELIEF .2 SEC RLF DLY 67 PL

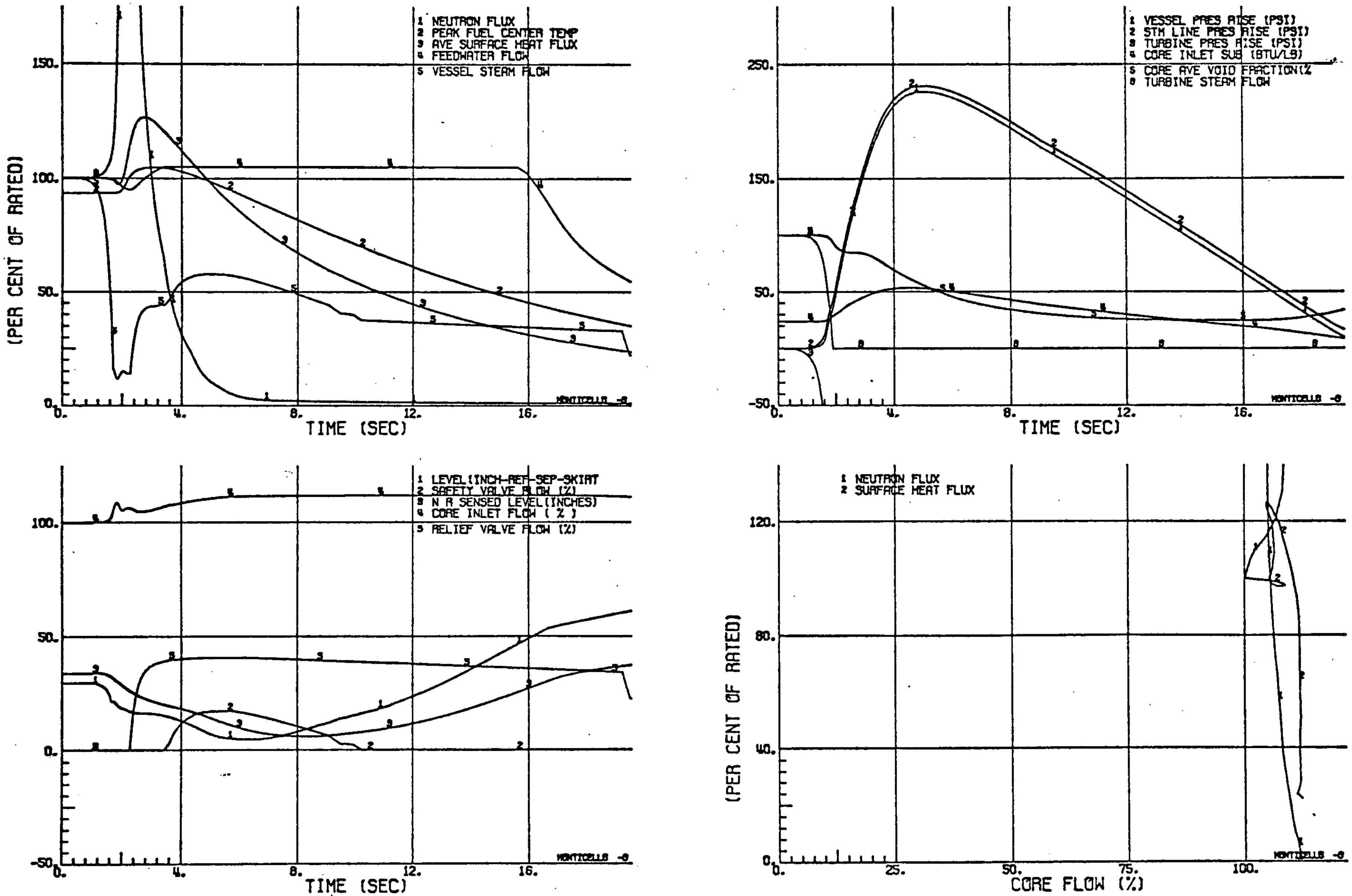

FIGURE 4

MONT RLD9B
3 SEC CLOSURE ALL ISOL VLVS, FLUX SCRAM 35% RELIEF .2 SEC RLF DLY 67 PL
18% SAFETY