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Niagara Mohawk Power Corporation Syracuse, New York 13202 T. J. Broshan	2-28-72	2-29-72	
	LTR. X	MEMO:	REPORT: OTHER:
TO:	ORIG.:	CC:	OTHER:
Dr. Peter A. Morris	3 signed & 67 conf'd		
	ACTION NECESSARY ☐ NO ACTION NECESSARY ☐	CONCURRENCE ☐ COMMENT ☐	DATE ANSWERED: BY:
CLASSIF: U	POST OFFICE REG. NO:	FILE CODE: 50-220	(INPUT)

DESCRIPTION: (Must Be Unclassified)

Ltr pursuant to Para 50.59 of 10CFR50, Request Change to Tech Specs (App A) for License DPR-17 w/attached Figures 1 thru 7 for Nine Mile Island...

ENCLOSURES:

REMARKS:

(Local PDR - Hold)

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U.S. ATOMIC ENERGY COMMISSION

MAIL CONTROL FORM FORM AEC-326S (8-60)

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NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

SYRACUSE, NEW YORK 13202

Regulatory File Cy.

February 28, 1972

Dr. Peter A. Morris
Division of Reactor Licensing
United States Atomic Energy
Commission
Washington, D. C. 20545

RE: Docket No. 50-220; License DPR-17

Dear Dr. Morris:

Pursuant to Paragraph 50.59 of 10 CFR Part 50, Niagara Mohawk Power Corporation proposes changes to Appendix A (Technical Specifications) of Provisional Operating License DPR-17 for the Nine Mile Point Nuclear Station.

Introduction

Improved analytical techniques now available at General Electric Company have caused us to adopt a revised basic scram reactivity curve for the calculation of plant response to operational transients. Figure 1 shows the change from the curve previously used by General Electric. This change may occur as the equilibrium fuel cycle is approached.

All transient analyses were reviewed and new calculations performed where necessary. The results reported herein indicate that a lowering of the set points for the solenoid-actuated relief valves is desirable to maintain the margin between the minimum set point for the safety valves and the peak pressure calculated for the limiting transient. A corresponding lowering of the initiation set point for the emergency cooling condensers is also proposed. These are the two changes proposed herein to the Station's Technical Specifications.

Transient Analyses and Core Dynamics

The following transients were reanalyzed:

Category	Event
Primary System Pressure Increase	1. Turbine Trip with Partial Bypass (1850 MWt) 2. Turbine Trip with Failure of Bypass

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Category	Event
Primary System Increase Pressure	3. Mainstream Line Isolation Valve Closure
Moderator temperature decrease	4. Feedwater Controller Malfunction (Increase Flow)
Decrease of coolant inventory	5. Loss of Auxiliary Power
Core coolant flow increase	6. Flow Controller Malfunction (Increase Flow)

Since the new scram reactivity curve represents an end-of-cycle condition, the new analyses were done with other inputs characteristic of end-of-cycle conditions to insure a realistic worst case analysis. Other conservative assumptions used in the previous transient analyses, such as rod insertion rates, were not changed.

Other transients and the specific reasons why these were not reanalyzed are shown below:

Category	Reasons
Safety Valve Sizing Transient	Reference transient is the Turbine Trip with Failure of Bypass and Scram. As scram does not occur the new scram reactivity curve does not affect the previous analysis.
Primary System Pressure Increase	All events were reanalyzed at full power and results are discussed below.
Moderator Temperature Decrease	The only event of significance in this category is the Feedwater Controller Malfunction, maximum demand, which was reanalyzed and is discussed below.
Reactivity Insertion	The limiting transient is the Rod Withdrawal Error transient which is terminated by the rod block and not a scram; the results of previous analysis will not be affected by the change in scram reactivity.

Category	Reasons
Decrease of Coolant Inventory	These transients result in a reactor pressure vessel depressurization and, in some cases, a low level scram. Power level drops due to void formation prior to the scram.
Core Coolant Flow (Increase)	Startup of idle recirculation loop is not a severe transient and is basically affected by the reactivity which is not significantly changed.
Core Coolant Flow (Decrease)	A scram does not occur as a direct result of these transients so the change in scram reactivity curve will not affect previous results.

1. Turbine Trip with Partial Bypass (1850 MWt)

Objectives

a. Demonstrate a safe abnormal operational transient within fuel limits.

b. Verify the adequacy of the 10 second time delay for emergency cooling system actuation at the design rating.

Assumptions and Initial Conditions

a. The reactor is at the 1850 MWt power condition with 1030 psig vessel pressure when the turbine stop valves close.

b. The turbine stop valve closure scram is initiated when the valves are $\leq$ 10 percent closed.

c. The trip point for the emergency cooling system is set at 1080 psig while the lowest set-point of the solenoid-actuated relief valves has been lowered to 1090 psig.

d. The bypass system capacity is 40 percent of the turbine steam flow at the 1850 MWt initial power level.

Comments

The solenoid relief valves are expected to open on this transient because of the lowered set points (1090 to 1100 psig).

Results

Figure 2 shows a turbine trip with partial bypass at the design rating power level (1850 MWt). The sudden closure of the turbine stop valves causes a trip scram to occur within 10 milliseconds. Neutron flux peaks at about 112 percent of the initial power level at about 0.48 seconds. The core average surface heat flux does not rise above its initial value because of the fast action of the trip scram.

Vessel dome pressure peaks at about 1116 psig while the mid-core peak is about 1131 psig. These are essentially the same pressure peaks that result from a generator trip transient. Vessel pressure exceeds the 1080 psig trip point of the emergency cooling system at about 1.3 seconds, indicating that the design 10 second time delay for actuation is adequate.

Full bypass flow is achieved within 0.3 seconds. The valves remain open until about 18 seconds when the pressure regulator demand becomes less than the combined valve capacity.

Pressure exceeds the 1090 psig set point of the solenoid-actuated relief valves at about 1.6 seconds and the valves remain open for 2.5 seconds.

2. Turbine Trip with Failure of Bypass

Objectives

Verify that the solenoid actuated relief valve capacities and setpoints satisfy the following conditions:

(1) Limit the pressure peak to at least 25 psi below the setpoint of the first safety valve.
(2) Limit the pressure peak to below the maximum fuel external design pressure.

Demonstrate a safe transient within the fuel thermal limits.

Assumptions and Initial Conditions

a. The reactor is at full design rating (1850 MWt) when the turbine stop valves close.

b. The bypass valves fail to open.

c. The turbine stop valve closure scram is initiated when the stop valves are $\leq$ 10 percent closed.

d. The action of the emergency cooling system is ignored.

e. The first set point of the relief valves is 1090 psig.

Comments

The solenoid actuated relief valves set point has been lowered to maintain the margin to the first safety valve set point.

Results

Figure 3 shows the transient resulting from the closure of the turbine stop valves with simultaneous failure of the bypass system. This accident causes a trip scram to occur within 10 milliseconds. Neutron flux peaks at about 163 percent of initial value at about 0.85 seconds. Because of the trip scram action, the core surface heat flux remains below the initial value. The relief valves open at about 1.6 seconds and then remain open for about 9 seconds. Trip scram and the solenoid-actuated relief valve operation turn the pressure transient and drop reactor power and pressure to levels at which the emergency cooling system can accomodate the remaining reactor decay heat.

The analysis indicates that the pressure peak at the location of the safety valves is approximately 44 psi below the set point of the first group of safety valves.

Pressure in the vessel dome peaks at 1174 psig and the mid-core pressure peaks at 1185 psig which is well below the fuel design pressure limits.

3. Main Stream Line Isolation Valve Closure

Objective

Demonstrate the ability of the solenoid-actuated relief valves to keep vessel pressure within limits upon isolation.

Assumptions and Initial Conditions

a. The reactor at the 1850 MWt power conditions.

b. The minimum allowable normal valve closure time of 3 seconds is used.

c. The reactor scram is initiated by the valve position switches at the 90 percent open position on the isolation valves (at approximately 0.3 seconds).

d. The action of the emergency cooling system is ignored.

Comments

The solenoid-actuated relief valve capacities and set points are determined by the turbine trip with failure of bypass transient described in Analysis 2.

Results

Figure 4 shows the transients resulting from a three-second main steam line valve closure. It is apparent that at the 1850 MWt power level the solenoid-actuated relief valves remain adequate to prevent lifting of the safety valves.

The pressure at the location of the relief valves reaches the set point value of 1090 psig at about 3.7 seconds. The opening of the valves holds the vessel dome pressure peak to about 1121 psig which is well below the first safety valve set point. Mid-core pressure peaks at about 1135 psig.

Although the effects of the emergency cooling system are ignored in this analysis, the combined effect of the relief valves and the emergency cooling system is more than sufficient to turn the pressure transient remaining after the relief valves close for the first time.

4. Feedwater Controller Malfunction (Increase Flow)

Objective

Demonstrate a safe response to the malfunction which causes continuous maximum feedwater flow into the reactor.

Assumptions and Initial Conditions

The reactor is at a typical power level (reduced by flow control) when the malfunction occurs.

Comments

Feedwater could eventually be shut off by operator action, e.g., placing the controller on manual, closing the shut-off valves or tripping the feedwater pumps.

Results

Figure 5 shows the transient of a feedwater controller failing open. The transient is brought under control by the high water level turbine trip.

The transient is initiated with reactor power at about 54 percent of the 1850 MWt power level and about 35 percent of recirculation flow. Because of the mismatch between steam flow and feedwater flow, the vessel water level begins to rise and climbs at a maximum rate of about 3 inches per second after the maximum feedwater rate is achieved.

The high water level turbine trip is initiated at about 9.6 seconds. At the same time trip scram is also initiated. Neutron flux peaks at 90 percent of the 1850 MWt level at about 10 seconds and the core average surface heat flux peaks at about 61 percent shortly thereafter. Recirculation flow at this time is about 41.5 percent. The critical heat flux ratio at all times is greater than that at full power.

The pressure rise continues for a short time after the scram is initiated with the pressure peak occuring at about 12 seconds after the feedwater controller malfunction. The analysis indicates that peak pressure at mid-core is about 996 psig and about 992 psig in the vessel dome.

5. Loss of Auxiliary Power

Objective

Demonstrate that a safe shutdown can be made with loss of auxiliary power.

Assumptions and Initial Conditions

Loss of all auxiliary power causes loss of condenser cooling water, trip of feedwater pumps, and trip of recirculation pumps.

Comments

A worse case for this type of transient is simulated by a simultaneous loss of condenser vacuum with resulting scram, turbine trip without bypass, and trips of the feedwater and recirculation pumps.

Results

Figure 6 shows the results of the transient. Neutron flux peaks at 156.7 percent of the initial value at about 0.7 seconds. The relief valves open in about 1.6 seconds. Full flow capacity is achieved at 2.1 seconds later. The relief valves continue to function until about 12.6 seconds when the pressure at the location of the valves falls below the set point. However, heat continues to be generated after the relief valves are closed, building up pressure slowly such that the relief valves are again actuated at about 16.5 seconds.

The pressure drops quickly to the set point of the valves and again they close. The relief valves will continue to cycle as shown until heat generation has dropped to the capacity of the emergency cooling system.

Vessel dome pressure peaks at 1181 psig, well below the first set point of the safety valves. Mid-core pressure peaks at about 1187 psig.

6. Flow Controller Malfunction (Increase Flow)

Objective

Demonstrate a safe transient when one recirculation flow controller causes maximum increasing rate of change of pump speed.

Assumptions and Initial Conditions

The reactor is operating at partial power (reduced by flow control) when one speed controller malfunctions, causing the scoop tube positioner (for the fluid coupler in the motor-generator set) to move at maximum speed in the direction to increase flow.

Comments

This type of malfunction is not expected to occur simultaneously in another flow controller. A malfunction in the master flow controller would produce a less severe transient since the limit on flow rate for the five controllers is less than the rate limit for one scoop tube positioner.

Results

The transient resulting from this accident is shown in Figure 7. No scram trip level is reached during this transient.

The starting point for this transient analysis was about 54 percent of 1850 MWt and 36 percent recirculation flow which is considered a typical low-power operating point. The malfunction in the one speed controller causes total recirculation flow to increase to about 48 percent within 9 seconds. Neutron flux leads the flow and peaks at about 89 percent within 4 seconds. A second peak of the same magnitude occurs at about 7.5 seconds as a result of increasing feedwater. Peak thermal power is essentially reached at about 12 seconds with a value of 73 percent of 1850 MWt. The transient terminates at this time because of a model constraint. The transient is mild, not much more severe than the maneuver of running up in power on flow control. No thermal limits are approached during the transient. The flow in the loop in which the malfunction occurs increases rapidly to about 133.9 percent of rated loop flow. This flow increase from 36 percent to 133.9 percent occurs within 9 seconds.

Dwell Time Re-Determination

The dwell time was also reanalyzed because it is involved with the Station's Technical Specifications. The turbine trip with bypass transient was used to obtain the data for the dwell time calculations. To vary the time of scram initiation, a trip scram, flux scram, 0.5 second delayed pressure scram, and 1.0 second delayed pressure scram cases were analyzed.

The analyses show that neutron flux peaks are lower and broader than previously reported because the void coefficient is lower, thereby reducing the rate of reactivity insertion and slowing the progress of the transient. The recalculated dwell time is approximately 0.66 secs. longer. However, for conservatism, no change to the value presently incorporated in the Technical Specification is proposed.

In summary these results indicate that with the new relief valve settings (1090 to 1100 psig) a margin of much more than 25 psi is maintained between the peak pressure for the limiting transient and the first safety valve set point. For the less severe transients resulting from single errors, the lower setting of the relief valves may result in pressure relief and steam flow through the relief valves. Actuation of the relief valves in such instances is solely a change in the plant operational response for Nine Mile Point and reflects the normal operational response for BWR plants with small bypass capacity.

Proposed Changes to Appendix A (Technical Specifications)

As a result of the foregoing, the following specific changes to the Appendix A of License DPR-17 are proposed:

Item	Location	Change	Discussion
Basis statements for 2.1.2 & 2.1.2.a	page 11	Add a superscript 10 to 6, 7 & 9 on line 4 and the last line. At bottom of page add reference (10) to this letter.	The adequacy of the specifications is also confirmed with the new transient analyses described in this letter.
Basis statements for 2.1.2 g-h	page 14	Change the last sentence of the first paragraph by adding a reference to this letter in the parentheses.	The original peak pressure listed is still representative of the most severe case.
Basis statement for 2.1.2.i	page 15	Change the parenthetical reference to this letter.	The new transient analysis shows that the statement is still true.
Basis statement for 3.1.1.c.	page 24	In the last sentence of the first paragraph add a reference to this letter.	The new transient analyses were done with the same scram times as before.
Basis statement for 3.1.3	page 30	In the third paragraph, change the initiation setpoint from 1090 psig to 1080 psig. Also in the third sentence add Technical Supplement to Petition to Increase Power Level	The initiation point is moved down 10 psig in order to minimize the chances of automatic relief valve operation during slow transients involving pressure increases. This is solely an operational concern with the new relief valve setpoints at 1090 psig. The added references show that the time delay is still applicable.
Table 3.6.3.c.	page 102	Under emergency cooling initiation setpoint, change 1090 psig to 1080 psig.	Same as reason for change in Basis Statement for 3.1.3, noted above.
Basis statement for 2.2.2.a.	page 18	Change the second sentence of the second paragraph to read as follows: "Any five of these valves opening at 1090 to 1100 psig will keep the maximum vessel pressure below the lowest safety valve setting as demonstrated in Appendix E-I, 3.11*, The Technical	The new transient analyses show that the automatic relief valve set points must be lowered to prevent the safety valves from actuating on a worst case pressurization transient; i.e.: turbine trip without bypass. The change from four to five valves is made to maintain the margin from peak transient

Item	Location	Change	Discussion
Basis statement for 2.2.2.a. (Cont.)	page 18	Supplement to Petition to Increase Power Level, and letter: Niagara Mohawk Power Corporation (TJ Brosnan) to Atomic Energy Commission (PA Morris), dated February 28, 1972.	pressure to the safety valve set points derived by the new analysis.
Specification 3.2.9.a.	page 58	Change the wording from "....four of the six..." to "....five of the six ...."	Same as directly above
Basis Statement for 3.2.9.a.	page 58	In the first paragraph, change the set points to 1090, 1095 and 1100 psig. In the second sentence of the second paragraph, change the wording from "...only four valves ..." to "...only five valves ....". Also add a reference to this letter.	Same as directly above

Conclusions

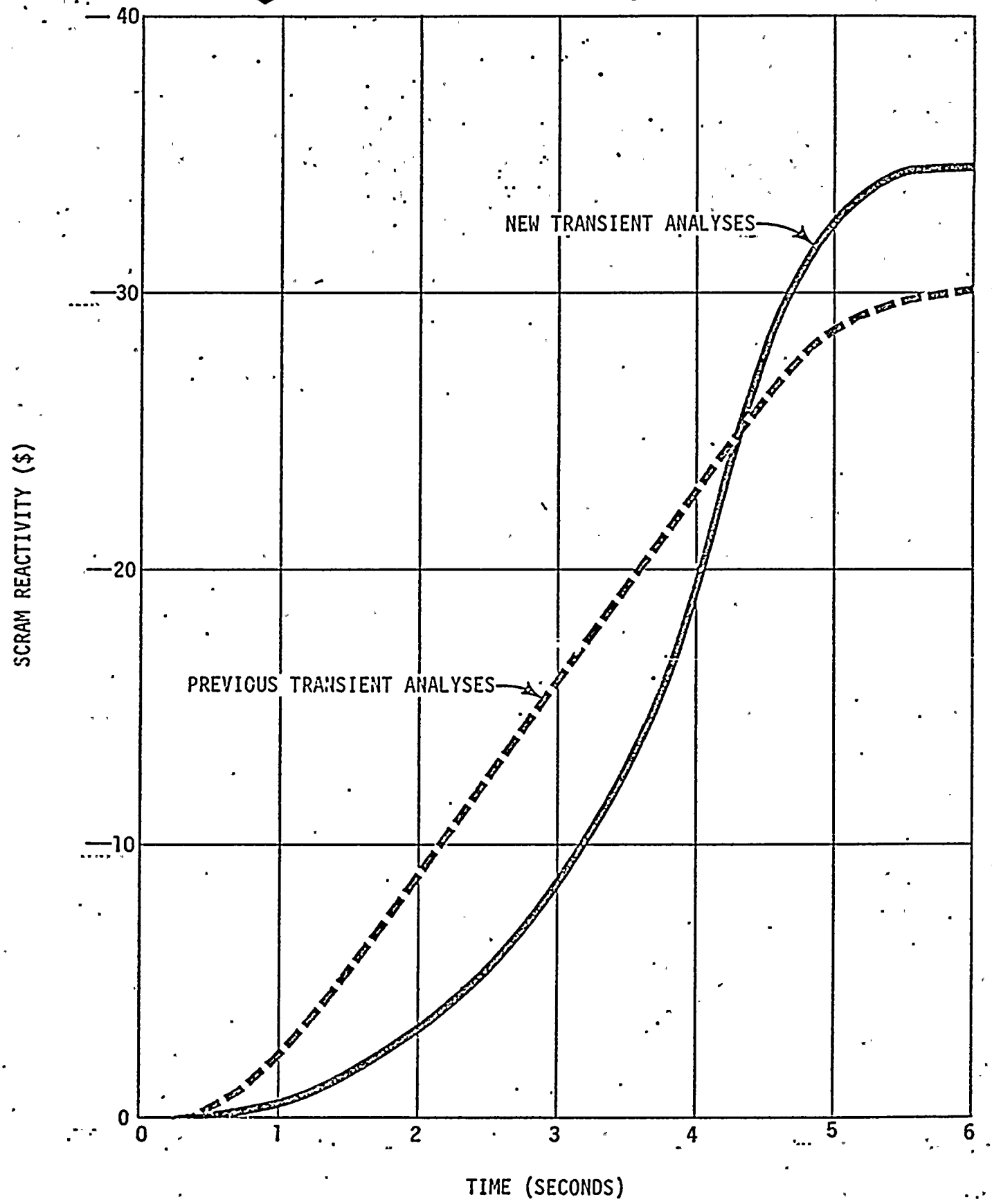
We conclude that these changes do not involve unreviewed safety questions as defined in 10 CFR 50.59, nor does it present significant hazards considerations not described or implicit in the Final Safety Analysis Report, the Petition to Increase Power Level and Supplements and Addendums thereto; and there is reasonable assurance that the health and safety of the public will not be endangered. This matter has been reviewed by the Safety Review and Audit Board which concurs with these findings.

Very truly yours,

T. J. Brosnan
Vice President and Chief Engineer

Attachments

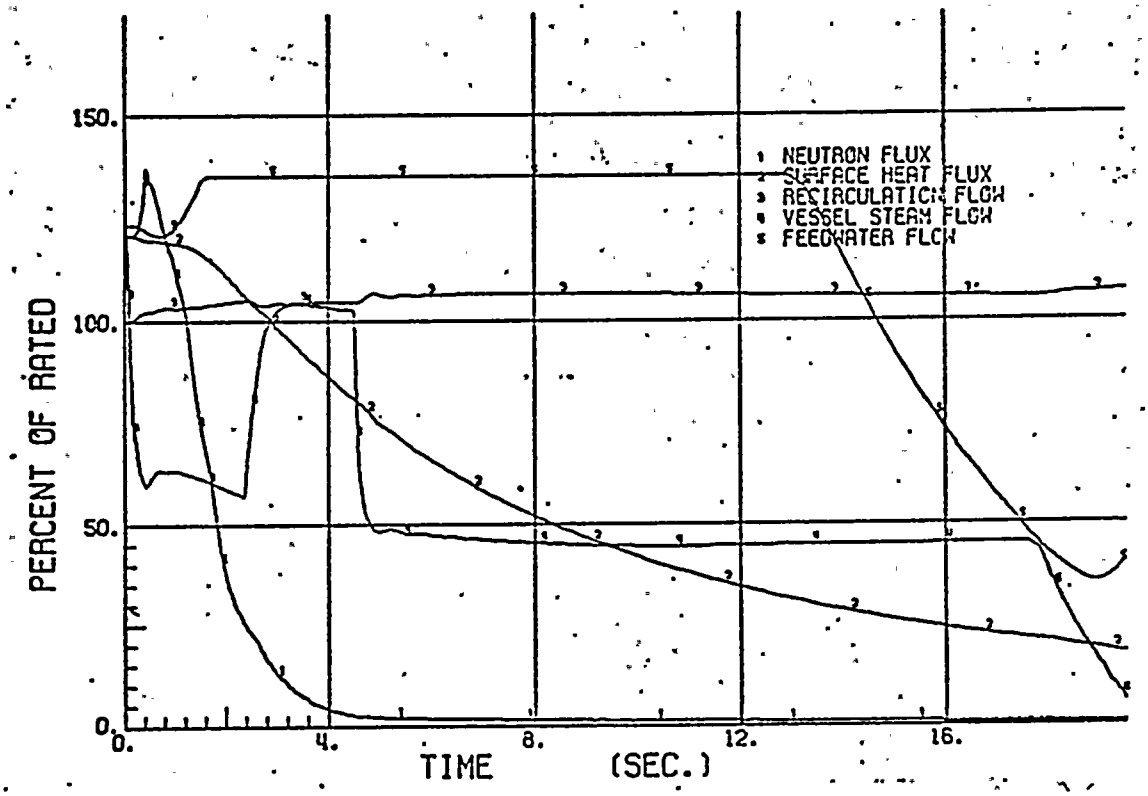
FIGURE 1

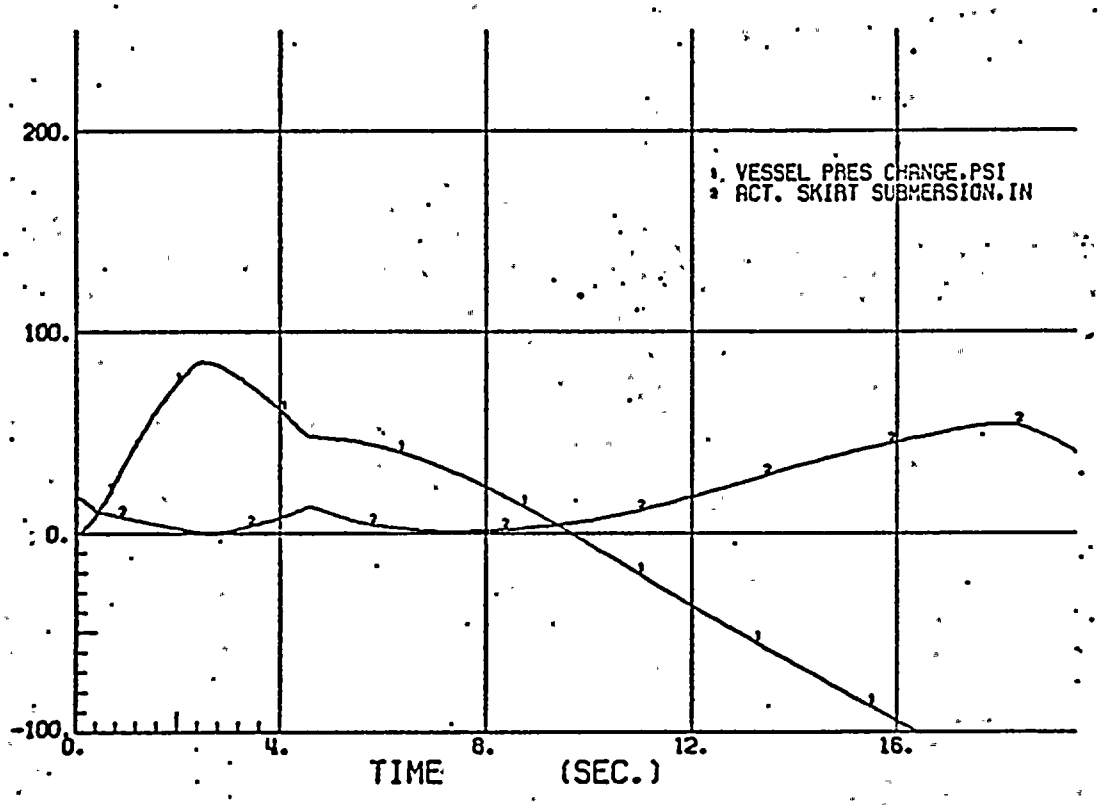


SCRAM REACTIVITY CURVES

FIGURE 2

-TURBINE TRIP WITH BYPASS-



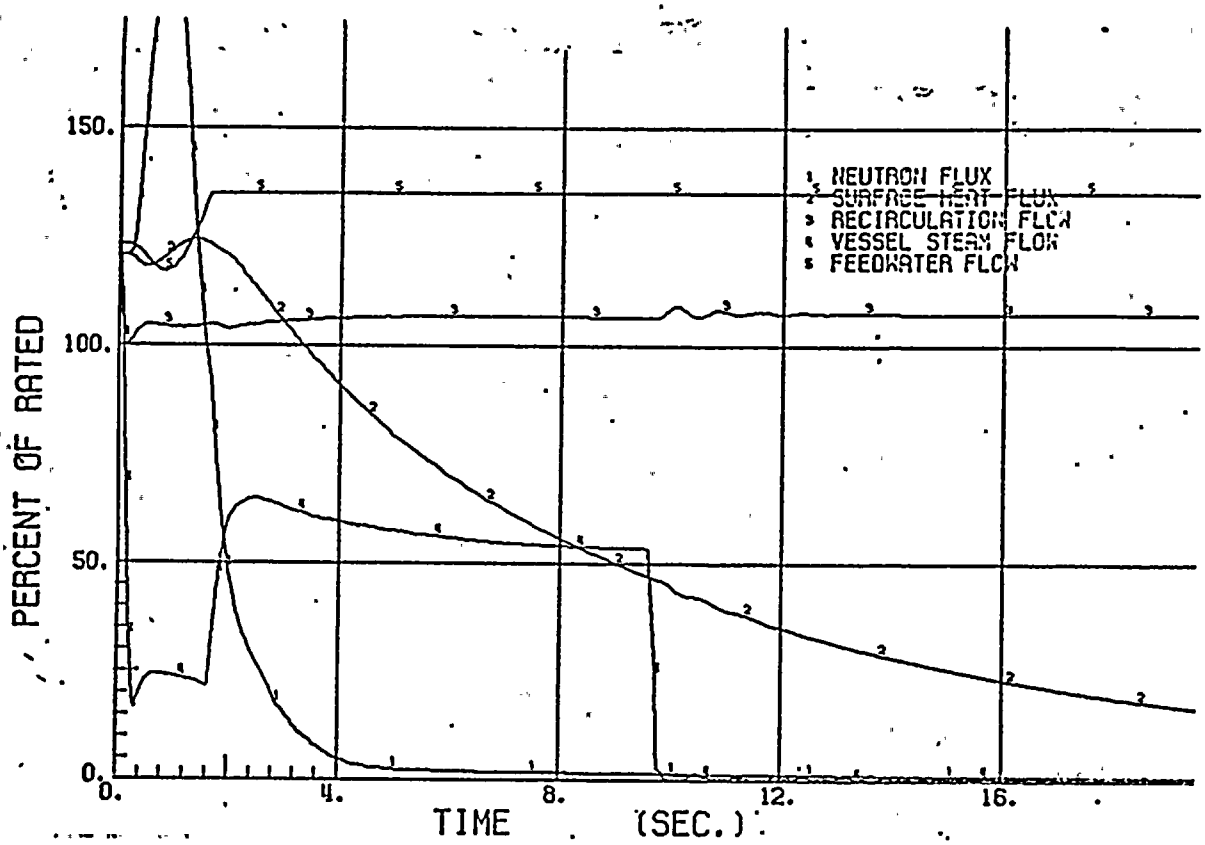


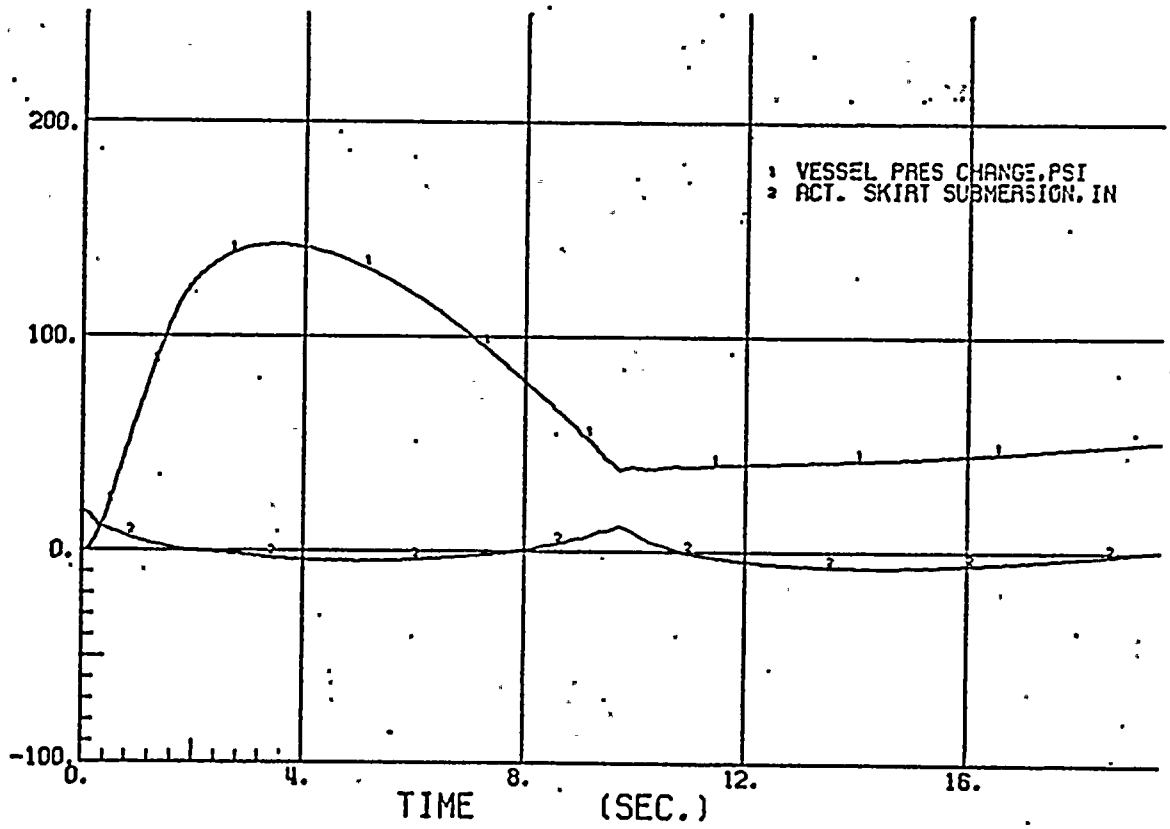
Note: 1. Text is based on 100 percent being equal to the design rating of 1850 MWt.

2. Figure is based on 100 percent being equal to initial rating 1538 MWt.

FIGURE 3

TURBINE TRIP-FAILURE OF BYPASS

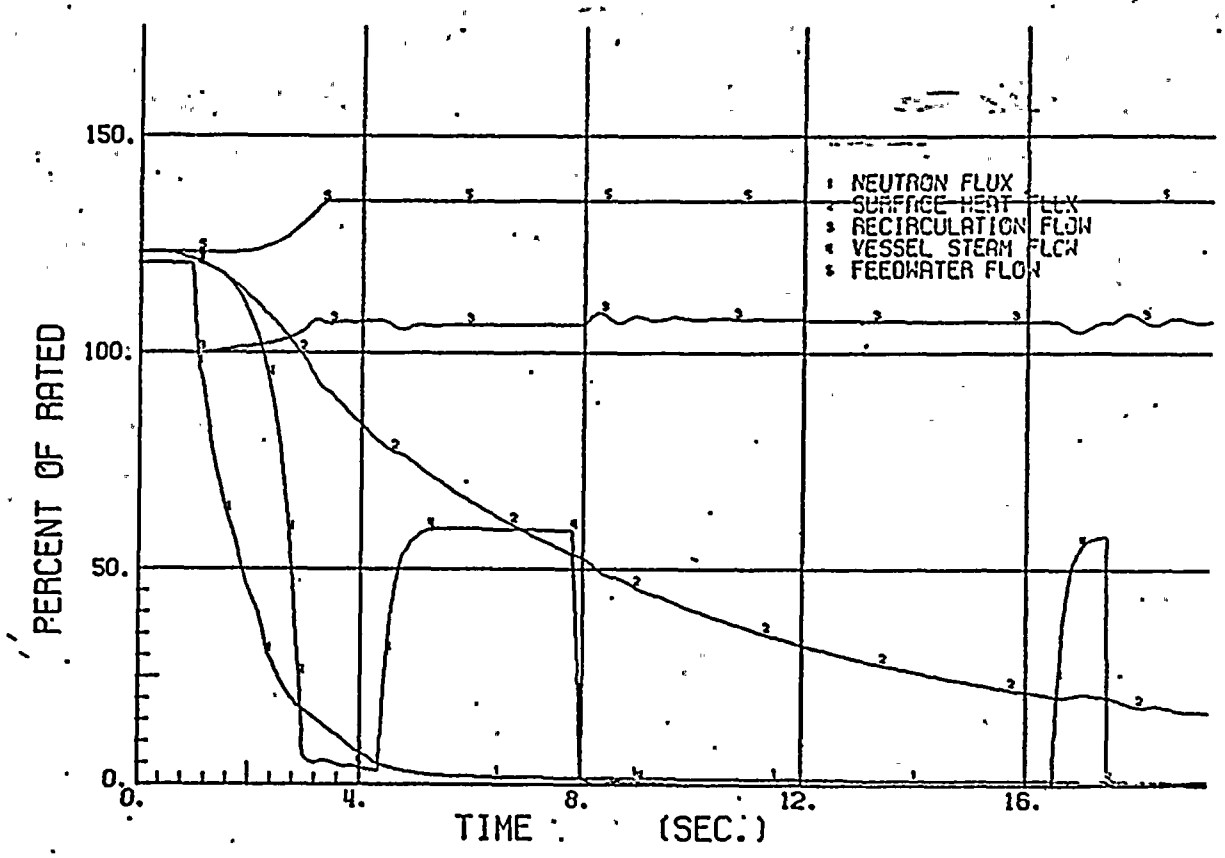


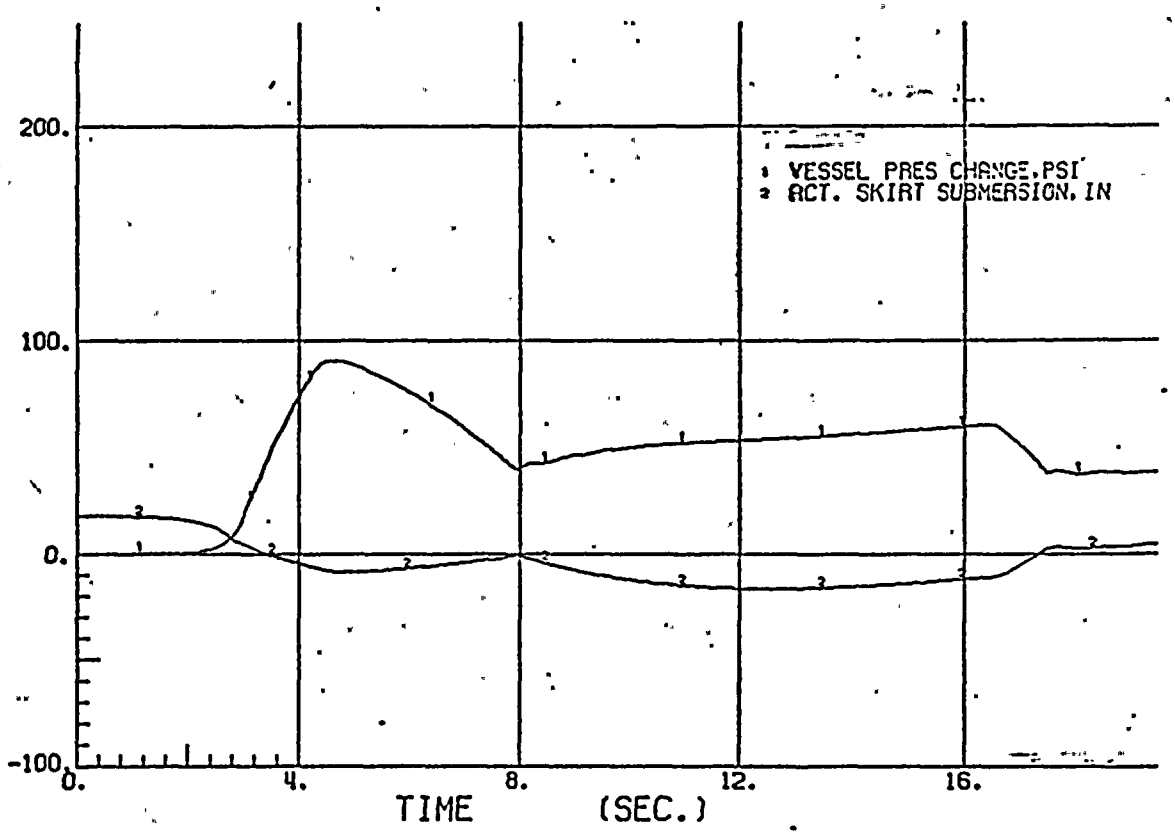


Note: 1. Text is based on 100 percent being equal to the design rating of 1850 MWt.

2. Figure is based on 100 percent being equal to initial rating 1538 MWt.

FIGURE 4
MAIN STEAM ISOLATION VALVE
CLOSURE

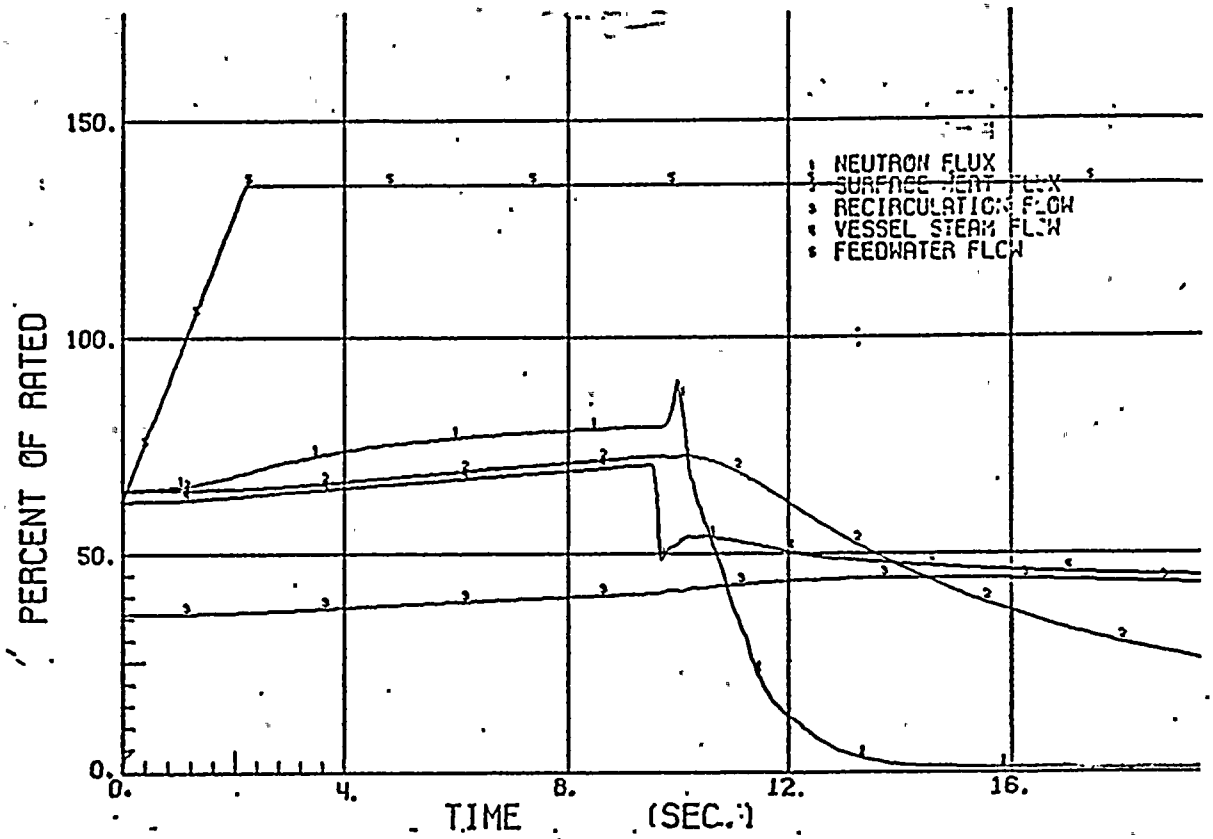


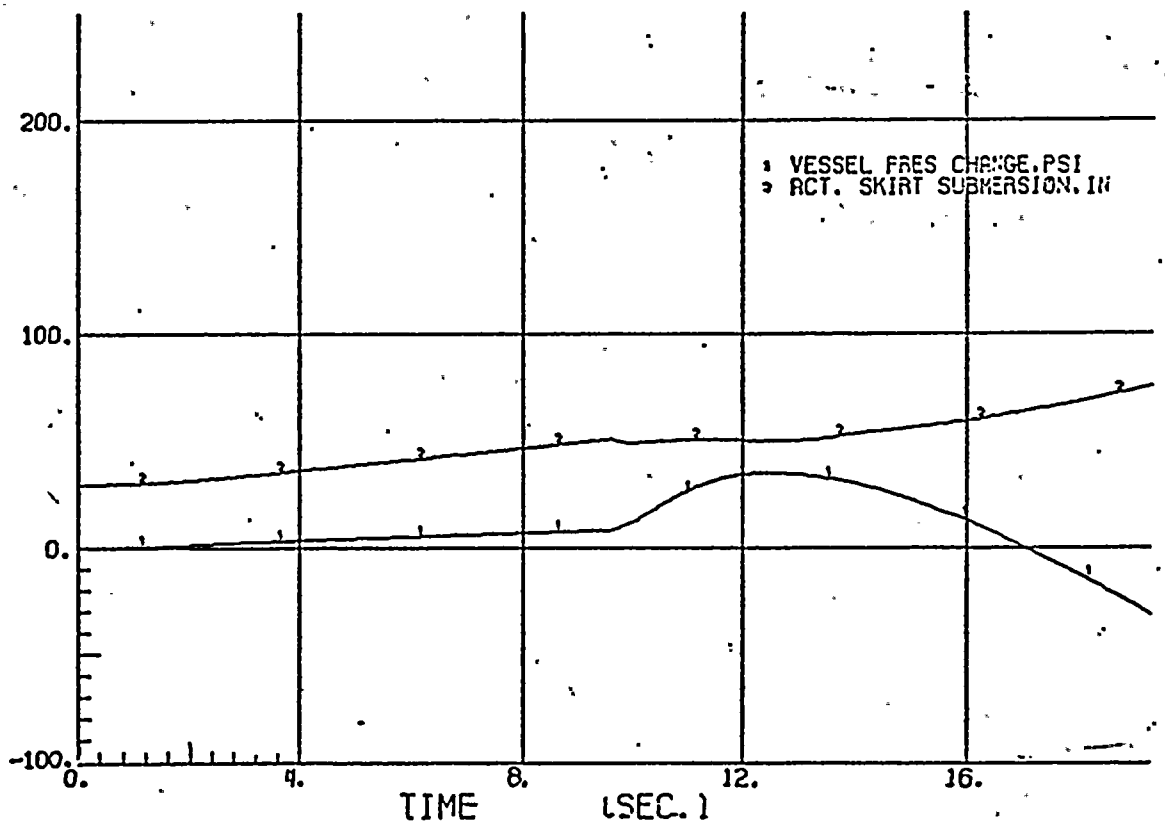


Note: 1. Text is based on 100 percent being equal to the design rating of 1850 MWt.

2. Figure is based on 100 percent being equal to initial rating 1538 MWt.

FIGURE 5
FEEDWATER CONTROLLER MALFUNCTION
(INCREASE FLOW)

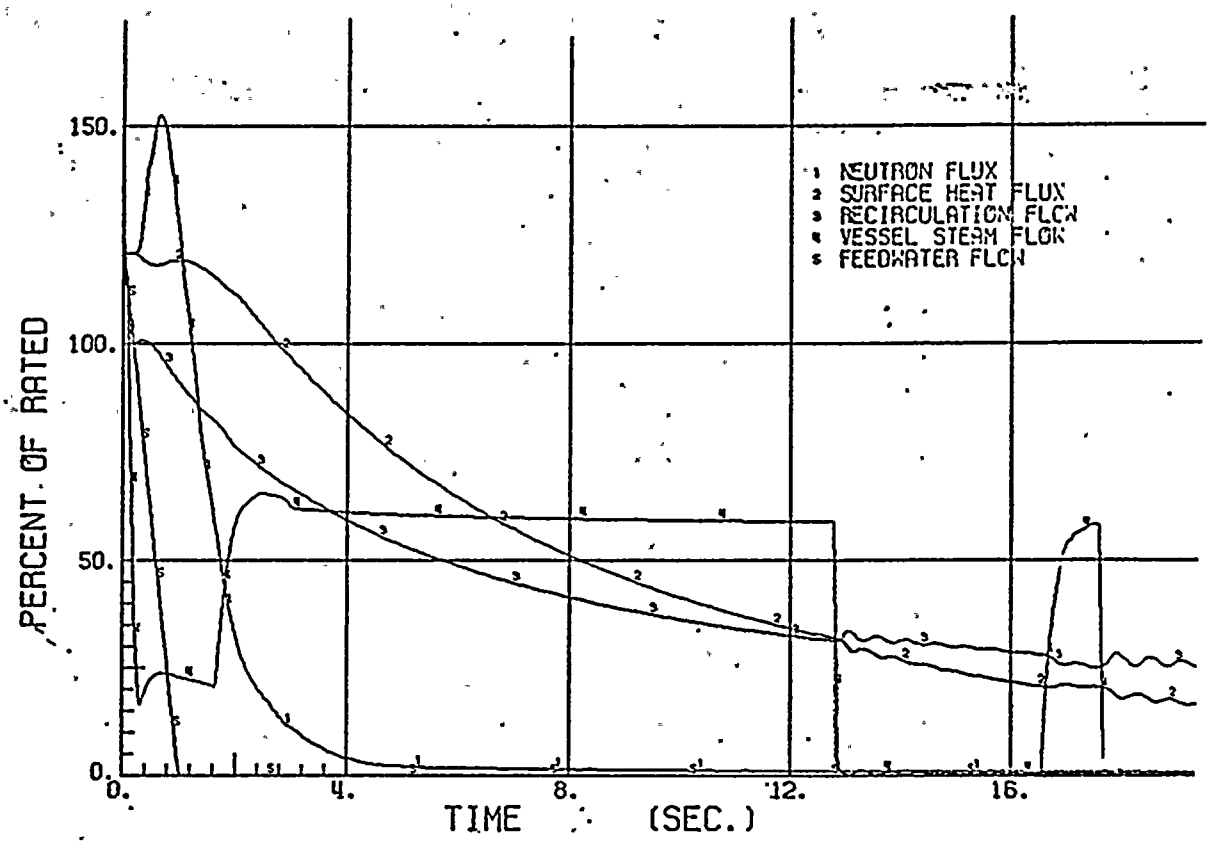


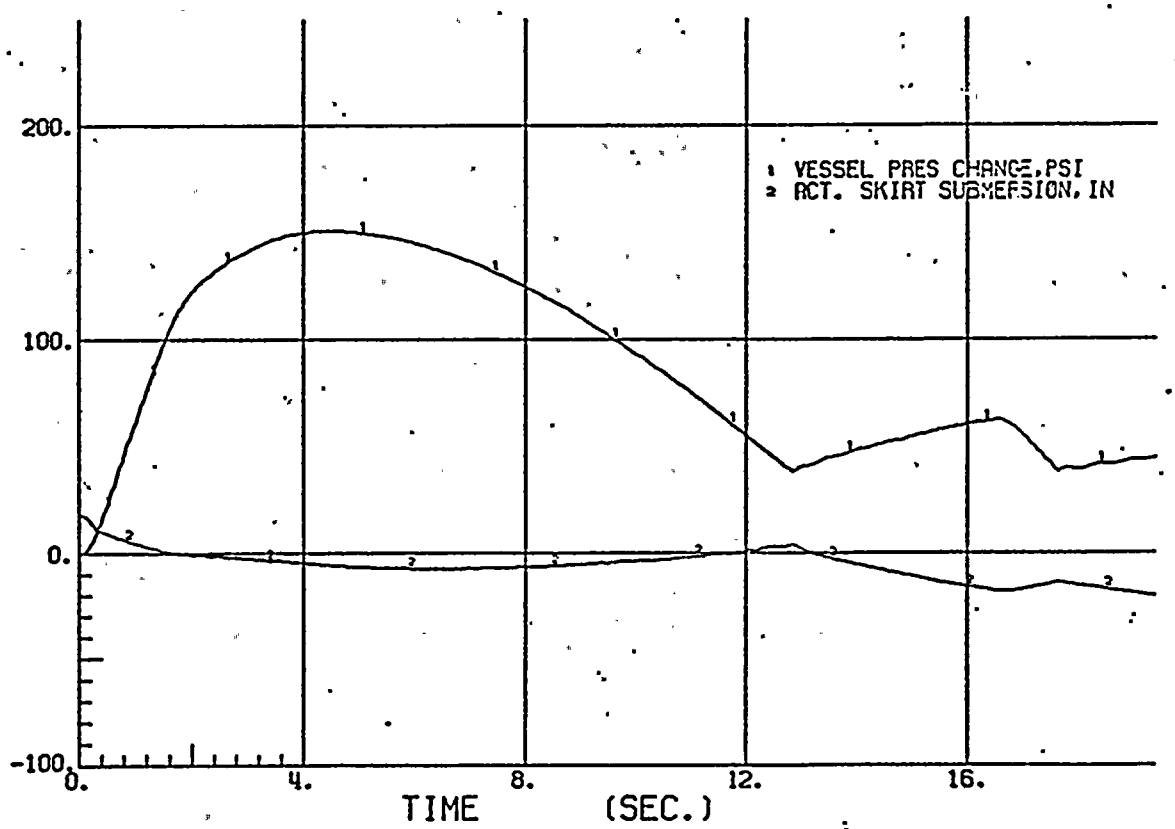


Note: 1. Text is based on 100 percent being equal to the design rating of 1850 MWt.

2. Figure is based on 100 percent being equal to initial rating 1538 MWt.

FIGURE 6
LOSS OF AUXILIARY POWER

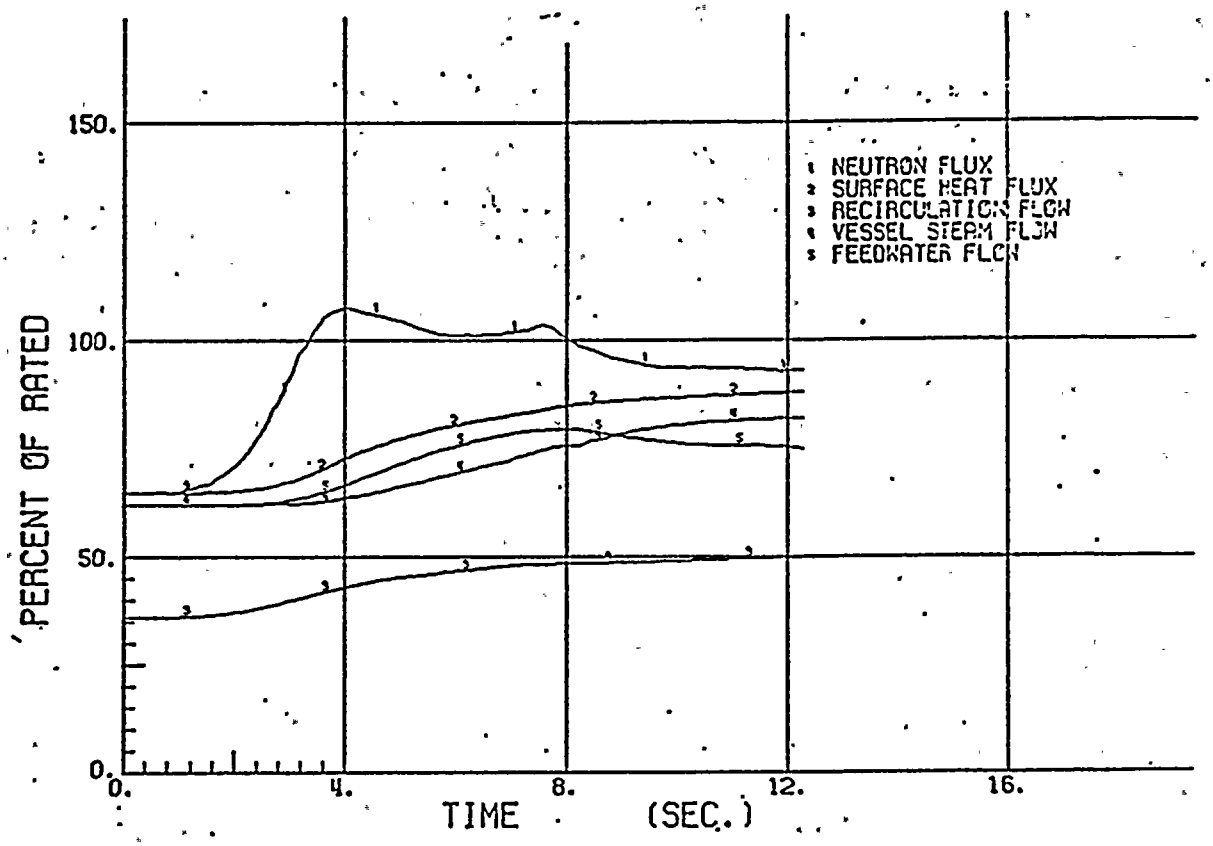


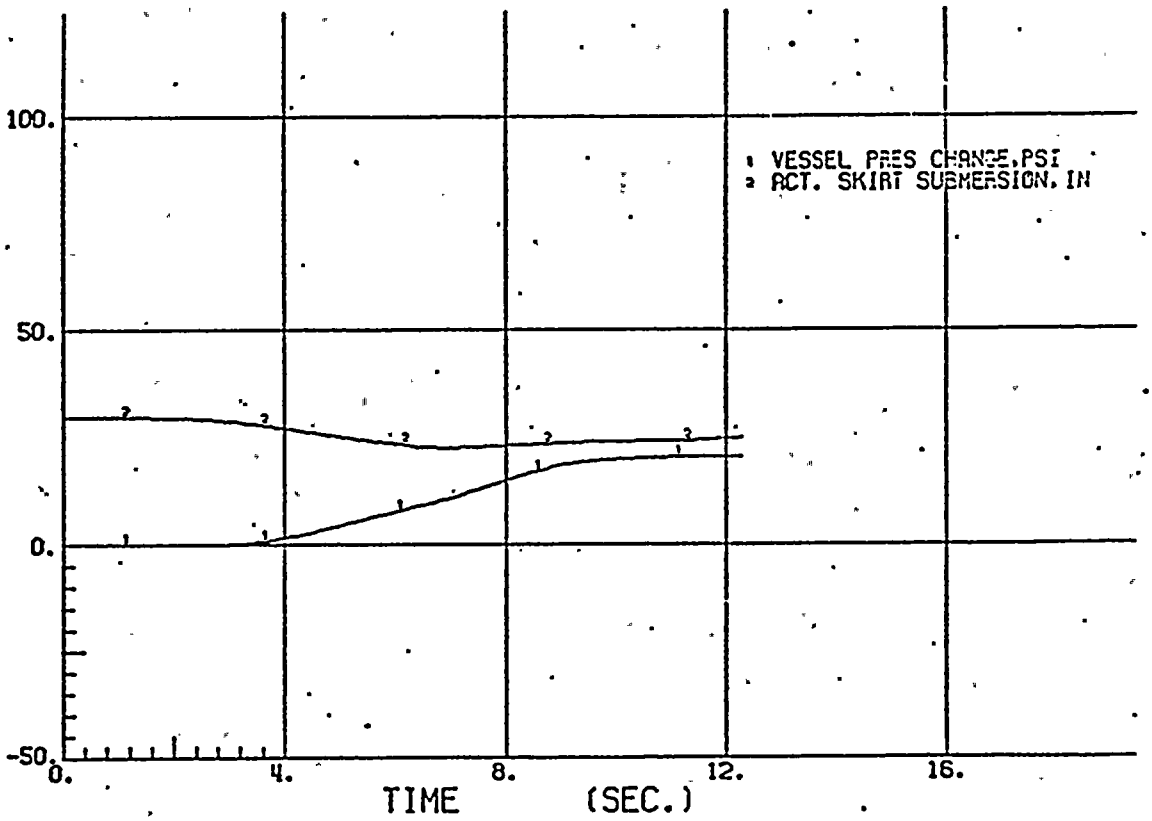


Note: 1. Text is based on 100 percent being equal to the design rating of 1850 MWt.

2. Figure is based on 100 percent being equal to initial rating 1538 MWt.

FIGURE 7
FLOW CONTROLLER MALFUNCTION
(INCREASED FLOW)





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