DONALD C. COOK UNITS 1 & 2

MAIN STEAM SAFETY VALVE

LIFT SETPOINT TOLERANCE RELAXATION

Nuclear and Advanced Technology Division
Westinghouse Electric Corporation
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FIGURE

Customer Reference No(s).
PO: 04877-040-IN

Westinghouse Reference No(s).

WESTINGHOUSE NUCLEAR SAFETY
SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S): DONALD C. COOK UNITS 1 AND 2

2) SUBJECT (TITLE): RELAXATION OF MSSV SETPOINT TOLERANCE TO +/-3%

3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 (b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A 10CFR50.59(a)(1)

(3.1)	Yes X	No ___	A change to the plant as described in the UFSAR?
(3.2)	Yes ___	No X	A change to procedures as described in the UFSAR?
(3.3)	Yes ___	No X	A test or experiment not described in the UFSAR?
(3.4)	Yes X	No ___	A change to the plant technical specifications? (See note on Page 2.)

4) CHECK LIST - Part B 10CFR50.59(a)(2) (Justification for Part B answers must be included on Page 2.)

(4.1)	Yes ___	No X	Will the probability of an accident previously evaluated in the UFSAR be increased?
(4.2)	Yes ___	No X	Will the consequences of an accident previously evaluated in the UFSAR be increased?
(4.3)	Yes ___	No X	May the possibility of an accident which is different than any already evaluated in the UFSAR be created?
(4.4)	Yes ___	No X	Will the probability of a malfunction of equipment important to safety previously evaluated in the UFSAR be increased?
(4.5)	Yes ___	No X	Will the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR be increased?
(4.6)	Yes ___	No X	May the possibility of a malfunction of equipment important to safety different than any already evaluated in the UFSAR be created?
(4.7)	Yes ___	No X	Will the margin of safety as defined in the bases to any technical specifications be reduced?

NOTES:

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answers to any of the above questions in Part A 3.4 or Part B cannot be answered in the negative, based on the written safety evaluation, the change review would require an application for license amendment as required by 10CFR50.59(c) and submitted to the NRC pursuant to 1OCFR50.90.

5) REMARKS:

The attached safety evaluation summarizes the justification for answers given in Part A 3.4 and Part B of this safety evaluation check list:

[1]Reference to documents containing written safety evaluation:

____

FOR UFSAR UPDATE

Section: various Pages: ______ Tables: ______ Figures: ______

Reason for/Description of Change:

UFSAR Mark-ups to be provided by separate transmittal

____

____

6) SAFETY EVALUATION APPROVAL LADDER:

6.1) Prepared by (Nuclear Safety): BSlapidos Date: 12-10-93

6.2) Reviewed by (Nuclear Safety): [illegible] Date: 12-10-93

DONALD C. COOK UNITS 1 & 2
INCREASED MAIN STEAM SAFETY VALVE SETPOINT TOLERANCE
SAFETY EVALUATION

I. INTRODUCTION

American Electric Power Service Corporation (AEPSC) has found that over an operating cycle the setpoint of the Main Steam Safety Valves (MSSVs) can change by more than 1% from the original set-pressure. AEPSC has requested that Westinghouse perform an evaluation to increase the lift setpoint tolerance on the MSSVs at Donald C. Cook Units 1 & 2. The following safety evaluation is provided to support changing the as-found lift setpoint tolerance as stated by the Technical Specifications from ±1% to ±3%.

During normal surveillance, if the valves are found to be within ±3%, they will be within the bases of the accident analyses, however, the valves will be reset to ±1% to account for future accumulation of drift. Thus, this evaluation permits a ±3% setpoint tolerance to address as-found conditions.

The MSSVs are located outside containment upstream of the Main Steam Isolation Valves. The purpose of the valves is to prevent overpressurization of the steam generators. In order to accomplish this, a bank of five valves is located on each of the four steam generators, and the relief capacity is designed such that the total steam flow from the 20 valves will bound that produced by the limiting licensing-basis analysis. For Donald C. Cook, the total relief capacity of the 20 valves is 17.153 E6 lbm/hr at 1186.5 psia (1171.5 psig).

The lift setpoints of the individual valves on each steamline are staggered at different pressures to minimize chattering once the valves are actuated. Staggering the valves also minimizes the total number of valves actuated during those transients where less than the maximum relief capacity is required thereby reducing maintenance requirements on the valves. The actual setpoints are provided in Table 1 and are documented in Tables 4.7-1 and 3.7-4 of the Units 1 and 2 Technical Specifications, respectively (Reference 1).

The operation of the Class 2 main steam safety valves (MSSVs) is governed by the ASME Code (Reference 2). AEPSC will maintain the design basis of the MSSVs by ensuring that the valves, if outside the ±1% tolerance, will be recalibrated to within ±1%. The purpose of this evaluation is to provide a quantification of the effects of a higher as-found lift setpoint tolerance. This safety evaluation will address the effects of the ±3% as-found tolerance on UFSAR accident analyses (non-LOCA, LOCA, SGTR) and will document how the effects are accounted for within the accident analyses and the acceptability of the increase in the lift setpoint tolerance.

TABLE 1

MAIN STEAM SAFETY VALVE LIFT SETPOINT

Value Number	Lift Setpoint (±1%)
SV-1	1065 psig (1080 psia)
SV-1	1065 psig (1080 psia)
SV-2	1075 psig (1090 psia)
SV-2	1075 psig (1090 psia)
SV-3	1085 psig (1100 psia)

References: Table 4.7-1 of the Unit 1 Technical Specifications
and
Table 3.7-4 of the Unit 2 Technical Specifications

II. LICENSING BASIS

Title 10 of the Code of Federal Regulations, Section 50.59 (10CFR50.59) allows the holder of a license authorizing operation of a nuclear power facility the capacity to initiate certain changes, tests and experiments not described in the Updated Final Safety Analysis Report (UFSAR). Prior Nuclear Regulatory Commission (NRC) approval is not required to implement the modification provided that the proposed change, test or experiment does not involve an unreviewed safety question or result in a change to the plant technical specifications incorporated in the license. While the proposed change to the MSSV lift setpoint tolerances involves a change to the Donald C. Cook Technical Specifications and requires a licensing amendment request, this evaluation will be performed using the method outlined under 10CFR50.59 to provide the bases for the determination that the proposed change does not involve an unreviewed safety question. In addition, an evaluation will demonstrate that the proposed change does not represent a significant hazards consideration, as required by 10CFR50.91 (a) (1) and will address the three test factors required by 10CFR50.92 (c).

The non-LOCA safety analyses will be examined to determine the impact of the MSSV lift setpoint tolerance relaxation on the DNB design basis as well as the applicable primary and secondary system pressure limits. The long-term core cooling capability of the secondary side will also be considered. The LOCA evaluation will investigate the effects on the licensing basis small break analysis in terms of peak clad temperature, and any adverse effects on the steam generator tube rupture event and subsequent dose release calculations will also be determined.

III. EVALUATIONS

The results of the various evaluations from the Nuclear Safety related disciplines within Westinghouse scope are discussed in the following sections.

1. Non-LOCA Evaluation

The non-LOCA accident analyses that are currently presented in the UFSAR modelled the MSSVs as a bank of five valves, all of which having a lift setpoint equal to that of the highest set valve (1100 psia) plus 3% to account for accumulation. All of the analyses and evaluations performed for this report modelled the staggered behavior of the MSSVs. Specifically, each valve was assumed to operate individually. Moreover, the analyses/evaluations of this report modelled the flow rate of each valve to ramp linearly from no flow at its lift setpoint (nominal Technical Specification setpoint plus or minus the 3% tolerance value) to full open flow at its full open point (3% above the pressure at which the valves were assumed to pop open - i.e., accumulation effect).

+3% Tolerance:

For the purposes of this evaluation, all 20 MSSVs are assumed to lift 3% above the Technical Specification lift setpoint and achieve full rated flow (normally at 3% above the setpoint) 6% above the setpoint.

ΔT Protection

The increase in the MSSV lift setpoint tolerance has the potential to impact the Overtemperature ΔT and Overpower ΔT setpoint equations. Referring to Figure 1a for Unit 1 and Figures 1b and 1c (which are the most limiting case for each unit/core type), increasing the point at which the MSSVs lift will lower the steam generator safety valve line.

If the current OTΔT setpoint coefficients (K1 through K3) result in protection lines that just bound the thermal core limits, it is possible that by lowering the SG safety valve line to the right, a portion of the core limits will be uncovered.

In order to evaluate the effects of the increase in the setpoint tolerance, the Overtemperature ΔT and Overpower ΔT setpoint equations (K1 through K6) were examined to determine if the equations remained valid assuming that all 20 MSSVs opened with a +3% tolerance. The results of that evaluation showed that there was sufficient margin in the generation of the current setpoint equations to offset the lowering of the SG safety valve line. Thus, changes to the Overtemperature and Overpower Technical Specifications are not needed. The results of this evaluation are presented as Figures 1a, 1b, and 1c.

DNB Events

The transients identified in Table 2 are analyzed in the D. C. Cook UFSAR to demonstrate that the DNB design basis is satisfied. With one exception, these events are a) of such a short duration that they do not result in the actuation of the MSSVs, b) core-related analyses that focus on the active fuel region only (i.e., only the core is modelled), or c) cooldown events which result in a decrease in secondary steam pressure. The single exception is the loss of external load/turbine trip event which is addressed explicitly in the ANALYSIS section of this safety evaluation. Thus, based on the above, these non-LOCA DNB transients are not adversely impacted by the proposed change, and the results and conclusions presented in the UFSAR remain valid.

Boron Dilution Event

The boron dilution event (14.1.5) is analyzed to demonstrate that the operators (or the automatic mitigation circuitry) have sufficient time to respond prior to reactor criticality. The secondary system is not modeled in the analysis of this event, and thus, changes to the MSSVs have no impact on this event. Therefore, the results and conclusions presented in the UFSAR remain valid.

Steamline Break Mass & Energy Releases

For the steamline break mass and energy releases, the steam release calculations are insensitive to the changes in the MSSV lift setpoints since the vast majority of these calculations result in depressurizations of the secondary side such that the MSSVs are not actuated. For the

TABLE 2

DNB DESIGN BASIS TRANSIENTS
NOT AFFECTED BY MSSV LIFT SETPOINT TOLERANCE INCREASE

Event	UFSAR Section
Excessive Heat Removal Due to Feedwater System Malfunction	14.1.10
Excessive Load Increase Incident	14.1.11
Rupture of a Steam Pipe (Steamline Break - Core Response)	14.2.5
Loss of Reactor Coolant Flow (Includes Locked Rotor Analysis)	14.1.6
Uncontrolled RCCA Bank Withdrawal From a Subcritical Condition	14.1.1
Uncontrolled RCCA Bank Withdrawal at Power	14.1.2
RCCA Misalignment	14.1.3

smaller break cases that might result in a heatup, one MSSV per steam generator is sufficient (based on the existing analyses) to provide any required heat removal following reactor trip. The secondary pressures will be no greater than those presently calculated. Thus the existing steamline break mass and energy release calculations remain valid.

Event	UFSAR Section
Steamline Rupture Mass & Energy Releases Inside Containment	WCAP-11902 Supplement 1
Steamline Rupture Mass & Energy Releases Outside Containment for Equipment Environmental Qualification	WCAP-10961 Rev 1 (current) Submittal AEP:NRC:1140* (approved 11/20/91)

* Submittal AEP:NRC:1140 "Technical Specification Change Request, BIT Boron Concentration Reduction," March 26, 1991.
(included in WCAP-11902, Supplement 1)

Long-Term Heat Removal Events

The only non-LOCA transients remaining are the long-term heatup events. The long-term heat removal events are analyzed to determine if the auxiliary feedwater (AFW) heat removal capability is sufficient to ensure that the peak RCS and secondary pressures do not exceed allowable limits, the pressurizer does not fill (LONF/LOOP), and the core remains covered and in a coolable geometry (FLB). These transients are listed below.

Event	UFSAR Section
Loss of All AC Power to the Plant Auxiliaries (Loss of Offsite Power - LOOP)	14.1.12
Loss of Normal Feedwater (LONF)	14.1.9
Feedwater System Pipe Break (FLB)*	14.1.8

* The Feedwater System Pipe Break event is not part of the Unit 1 licensing basis and is presented in the Unit 1 UFSAR for information purposes only.

These transients are impacted by the increase in the MSSV lift setpoint tolerance because the calculations determining the amount of AFW flow available must assume a maximum given steam generator backpressure in order to determine the amount of AFW that can be delivered. As the steam

generator back pressure increases, the amount of AFW delivered will be reduced. For the loss of normal feedwater and the loss of all AC power to the Plant Auxiliaries events, evaluations were performed in which the staggered actuation of the MSSVs was taken into account.

The safety analysis presented in the current UFSAR assumed an AFW flow rate of 450 gpm, split evenly to all four steam generators. The evaluations done for this report concerning loss of normal feedwater (LONF) for Units 1 and 2, as well as loss of all AC power to the plant auxiliaries (LOOP) for Unit 1, demonstrated that the secondary side pressures will not exceed 1123 psia during the time AFW is delivered to the steam generators. Based on Reference 10, the AFW assumptions modeled in the safety analysis remain valid for steam generator backpressures up to 1123 psia. Since the evaluation, in which a +3% MSSV setpoint tolerance was assumed, showed that the secondary side pressure transient will not preclude the AFW flow rates assumed in the analysis from being supplied to the steam generators, the existing analyses remain valid for Unit 1 LONF/LOOP and Unit 2 LONF.

The Loss of Offsite Power event (LOOP) for Unit 2 was also evaluated for this report. The LOOP safety analysis presented in the current UFSAR for Unit 2 assumed an AFW flow rate of 430 gpm split evenly to all four steam generators. The recent evaluation done for this report took credit for the staggered actuation of the MSSVs as well as a +3% setpoint tolerance, as discussed earlier. The evaluation yielded results similar to those discussed above for Unit 1. The secondary side pressure for this Unit 2 evaluation was demonstrated not to exceed 1133 psia during the period AFW is supplied. Based on Reference 10, the secondary side pressure transient was found not to preclude the AFW flow rates assumed in the analysis from being delivered to the steam generators. Therefore, the existing Loss of Offsite Power analysis for Unit 2 remain valid.

The evaluations for the LONF/LOOP events for both Unit 1 and Unit 2, as discussed above, demonstrate that the respective analyses are still applicable even if a MSSV lift setpoint tolerance of +3% is assumed. Therefore the results and conclusions presented in the Donald C. Cook Unit 1 & 2 UFSAR remain valid.

The evaluation done for this report for the Unit 2 Feedline Break event demonstrated that the secondary side pressure will not exceed 1133 psia during the period when AFW is being delivered.

At 1133 psia, an AFW flow rate of 685 gpm with asymmetric flow splits to the three intact steam generators could be supplied based on information contained in Reference 10. The current analysis for this event assumed a total AFW flow rate of 600 gpm with an even split of 200 gpm to the three intact steam generators. Since the total AFW flow rate is more than sufficient to accommodate AFW flow split deviations of as much as 25 gpm per loop, the current Feedline Break analysis continue to be applicable and remain bounding for this evaluation. Therefore, the results and conclusions presented in the Unit 2 UFSAR (14.2.8) remain valid.

<u>-3% Tolerance</u>:

The secondary steam releases generated for the locked rotor offsite dose calculations for Unit 2 could be potentially affected by an increase in the MSSV setpoint tolerance from -1% to -3%. Reference 9 transmitted the most recent locked rotor dose analysis. Given that the radiological assumptions used in the Reference 9 analysis do not change with an increase in MSSV setpoint tolerance (i.e., rods-in-DNB and primary to secondary leakage remain at 11% and 1 gpm respectively) the only effect the tolerance increase would have would be on the mass release values. The methodology used to calculate these masses is based on determining the amount of secondary side inventory required to cool down the RCS. During the first two hours (0-2 hours), the operators are assumed to lower the RCS average temperature to no-load conditions (547°F) by bleeding steam. Over the next 6 hours (2-8 hours), the operators will cool the plant down such that Mode 4 operation (hot shutdown) can be entered.

The existing steam release calculations for the 0-2 hour period used enthalpies corresponding to saturated conditions at both the nominal full power RCS average temperature and the no-load temperature (581.3°F and 547°F, respectively). Thus, as long as the increased lift setpoint tolerance (-3%) does not result in the MSSVs remaining open at a saturation temperature outside of the range identified above, the existing mass releases remain valid (Reference 9).

The existing mass release calculations were performed using the temperatures previously identified (581.3°F and 547°F). Per the Donald C. Cook Technical Specifications, the lowest set MSSV on each steam generator will open at 1080 psia (1065 psig) not including any tolerance. Based on the ASME Steam Tables (Reference 6) at saturated conditions, 547°F corresponds to 1020.1 psia and

represents the lowest steam pressure considered in the mass calculations. Thus, the existing releases include a reseat pressure approximately 5.5% below the lowest Technical Specification lift setpoint. As long as the valves continue to reseat within this pressure range, the current mass releases remain valid.

The operating windows that are applicable for Unit 1 operation are bounded by the Unit 2 dose analysis. Therefore, the mass releases for Unit 2, as found in Reference 9, are applicable to Unit 1.

Evaluation Summary

Thus, based on the discussions presented above, only one UFSAR non-LOCA transient is impacted such that a new analysis must be performed in order to address the effects of the MSSV lift setpoint tolerance increase from ±1% to ±3%. This event is the loss of external load/turbine trip accident. For the other transients, the results and conclusions presented in the Donald C. Cook Unit 1 & 2 UFSAR remain valid.

Non-LOCA Analysis:

Loss of External Load/Turbine Trip

The loss of external load/turbine trip event is presented in Section 14.1.8 of the Donald C. Cook UFSAR. This transient is caused by a turbine-generator trip which results in the immediate termination of steam flow. Since no credit is taken for a direct reactor trip on turbine trip, primary and secondary pressure and temperature will begin to increase, actuating the pressurizer and steam generator safety valves. The reactor will eventually be tripped by one of the other reactor protection system (RPS) functions; specifically, overtemperature ΔT, high pressurizer pressure, or low-low steam generator water level.

The turbine trip event is the limiting non-LOCA event for potential overpressurization, i.e., this transient forms the design basis for the primary and secondary safety valves. Since the MSSVs will now potentially be opening at a higher pressure due to the increase in the lift setpoint tolerance, it is necessary to analyze this transient in order to demonstrate that all the applicable acceptance criteria

are satisfied. A turbine trip is classified as an ANS condition II event, a fault of moderate frequency. As such, the appropriate acceptance criteria are DNBR, peak primary pressure, and peak secondary pressure. The transient is described in greater detail in the UFSAR.

The turbine trip event is analyzed using a modified version of the LOFTRAN digital computer code (Reference 6). This modified version of LOFTRAN only differs from the standard code version in the way the MSSVs are modelled. The program simulates neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves. With the modified code, the MSSVs are explicitly modeled as a bank of 5 valves on each steam generator with staggered lift setpoints. Whereas the standard LOFTRAN version program conservatively models the MSSVs as a bank of five valves, all having one common lift setpoint. By modelling the staggered behavior of the MSSVs, a more accurate simulation of how the valves actually behave is achieved. Since higher steam pressures are conservative for this event, no blowdown or hysteresis behavior was assumed.

Consistent with the existing UFSAR analysis, all assumptions were the same as previously used unless specifically noted. The following assumptions were used in this analysis:

a. Initial power, temperature, and pressure were at their nominal values consistent with:

   1) ITDP methodology (WCAP-8567) for Unit 1, with the exception that a 2% conservatism on initial core power was assumed.

   2) RTDP methodology (WCAP-11397) for Unit 2, with no exceptions.

b. Turbine trip was analyzed with both minimum and maximum reactivity feedback.

c. Turbine trip was analyzed both with and without pressurizer pressure control. The PORVs and sprays were assumed operable in the cases with pressure control. The cases with pressure control minimize the increase in primary pressure which is conservative for the DNBR transient. The cases without pressure control maximize the increase in pressure which is conservative for the RCS overpressurization criterion.

d. The steam generator PORV and steam dump valves were not assumed operable. This assumption maximizes secondary pressure which in turn maximizes the primary temperature for DNBR and primary pressure for pressure cases.

e. Main feedwater flow was assumed to be lost coincident with the turbine trip. This assumption maximizes the heatup effects.

f. Only the overtemperature ΔT, high pressurizer pressure, and low-low steam generator water level reactor trips were assumed operable for the purposes of this analysis.

g. The flow rate for each MSSV was modelled to ramp linearly from no flow at its lift setpoint (3% above the nominal Technical Specification setpoint) to full open flow at its full open point (6% above the nominal setpoint). The full open flow rate is based on a reference full flow capacity of 238 lbm/sec at 1186.5 psia (based on the ASME rated flow for these valves). For secondary side pressures between the initial full open point for each valve and 1186.5 psia, the full open flow rate was modelled to vary proportionally with pressure. This assumption maximizes secondary pressure which in turn maximizes the primary temperature for DNBR and primary pressure for pressure cases.

<u>Results</u>

Four cases for each unit/core type (i.e. Unit 1, Unit 2 mixed core,and Unit 2 full V5 core) were analyzed: a) minimum feedback without pressure control, b) maximum feedback without pressure control, c) maximum feedback with pressure control, and d) minimum feedback with pressure control. The most limiting cases in the current UFSAR continue to be the most limiting cases. The calculated sequence of events for the four cases for each unit are presented in Tables 3 and 4.

UNIT 1

Case A:

Figures 2 through 6 show the transient response for the turbine trip event under minimum reactivity feedback conditions without pressure control. The reactor is tripped on high pressurizer pressure. The neutron flux remains essentially constant at full power until the reactor is tripped, and the DNBR remains above the initial value for the duration of the transient. The pressurizer safety valves are actuated and maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Case B:

Figures 7 through 11 show the transient response for the turbine trip event under maximum reactivity feedback conditions without pressure control. The core power is observed to undergo a momentary increase. This is due to positive reactivity being inserted as a result of the increase in coolant density caused by the increase in primary pressure. This affect is quickly countered by the subsequent temperature rise brought on by the abrupt loss of the heat sink. The reactor is tripped on high pressurizer pressure. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer safety valves are actuated and maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Case C:

Figures 12 through 16 show the transient response for the turbine trip event under maximum reactivity feedback conditions with pressure control. The core power is observed to undergo a momentary increase. This is due to positive reactivity being inserted as a result of the increase in coolant density caused by the rapid increase in primary pressure. This affect is quickly countered by the subsequent temperature rise brought on by the abrupt loss of the heat sink. The reactor is tripped on low-low steam generator water level. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer relief valves and sprays maintain primary pressure

below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Case D:

Figures 17 through 21 show the transient response for the turbine trip event under minimum reactivity feedback conditions with pressure control. The reactor is tripped on high pressurizer pressure. Although the DNBR value decreases below the initial value, it remains well above the limit throughout the entire transient. The pressurizer relief valves and sprays maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Analysis Conclusion (Unit 1)

Based on the results of these Unit 1 turbine trip analyses with a +3% tolerance on the MSSV lift setpoints, all of the applicable acceptance criteria are met. The minimum DNBR for each case is greater than the limit value. The peak primary and secondary pressures remain below 110% of design at all times.

UNIT 2: a) mixed and b) full V-5 cores

Case A:

Figures 22a through 26b ("a" designates mixed core figures and "b" denotes full V-5 core figures) show the transient response for the turbine trip event under minimum reactivity feedback conditions without pressure control for both core types. The reactor is tripped on high pressurizer pressure. The neutron flux remains essentially constant at full power until the reactor is tripped, and the DNBR remains above the initial value for the duration of the transient. The pressurizer safety valves are actuated and maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Case B:

Figures 27a through 31b show the transient response for the turbine trip event under maximum reactivity feedback conditions without pressure control for both mixed and full V-5 core types. The core power is observed to undergo a momentary increase. This is due to positive reactivity being inserted as a result of the increase in coolant density caused by the rapid increase in primary pressure. This affect is quickly countered by the subsequent temperature rise brought on by the abrupt loss of the heat sink. The reactor is tripped on high pressurizer pressure. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer safety valves are actuated and maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Case C:

Figures 32a through 36b show the transient response for the turbine trip event under maximum reactivity feedback conditions with pressure control for the two applicable Unit 2 core types. The core power is observed to undergo a momentary increase. This is due to positive reactivity being inserted as a result of the increase in coolant density caused by the rapid increase in primary pressure. This affect is quickly countered by the subsequent temperature rise brought on by the abrupt loss of the heat sink. The reactor is tripped on low-low steam generator water level. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer relief valves and sprays maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Case D:

Figures 37a through 41b show the transient response for the turbine trip event under minimum reactivity feedback conditions with pressure control for both the mixed and full V-5 cores. The reactor is tripped on high pressurizer pressure. Although the DNBR value decreases below the initial value, it remains well above the limit throughout the entire transient. The pressurizer relief valves and sprays maintain primary pressure below 110% of the design value. The

main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Analysis Conclusion (Unit 2)

Based on the results of these Unit 2 mixed and full core turbine trip analyses with a +3% tolerance on the MSSV lift setpoints, all of the applicable acceptance criteria are met. The minimum DNBR for each case is greater than the limit value. The peak primary and secondary pressures remain below 110% of design at all times.

Non-LOCA Conclusions

The effects of increasing the as-found lift setpoint tolerance on the main steam safety valves have been examined, and it has been determined that, with one exception, the current accident analyses as presented in the UFSAR remain valid. The loss of load/turbine trip event was analyzed in order to quantify the impact of the setpoint tolerance relaxation. As previously demonstrated in this safety evaluation, all applicable acceptance criteria for this event have been satisfied and the conclusions presented in the UFSAR are still valid. Thus, with respect to the non-LOCA transients, the proposed Technical Specification change does not constitute an unreviewed safety question, and the non-LOCA accident analyses, as presented in the report, support the proposed change.

2. LOCA and LOCA Related Evaluations

Large Break LOCA

The current large break LOCA analyses for Donald C. Cook Units 1 and 2 were performed with the NRC approved 1981 Evaluation Model plus BASH. After a postulated large break LOCA occurs, the heat transfer between the reactor coolant system (RCS) and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary system pressure increases and the MSSVs may actuate to limit the pressure. However, this does not occur in the large break evaluation model since no credit is taken for auxiliary feedwater actuation. Consequently, the secondary system acts as a heat source in the

postulated large break LOCA transient and the secondary pressure does not increase. Since the secondary system pressure does not increase, it is not necessary to model the MSSV setpoint in the large break evaluation model. Therefore, an increase in the allowable MSSV setpoint tolerance for Donald C. Cook Units 1 and 2 will not impact the current UFSAR large break LOCA analyses.

Small Break LOCA

The small break LOCA analyses for Donald C. Cook Units 1 and 2 were performed with the NRC approved Evaluation Model using the NOTRUMP code. After a postulated small break LOCA occurs, the heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary system pressure increases which leads to steam relief via the MSSVs. In the small break LOCA, the secondary flow aids in the reduction of RCS pressure. Subsequently, Donald C. Cook Units 1 and 2 were reanalyzed to determine the impact of an increased MSSV setpoint tolerance of 3%.

The licensing basis small break LOCA analysis for Donald C. Cook Unit 1 included a safety evaluation to address a 25 gpm charging pump flow imbalance and operation with the high head safety injection cross tie valve closed at 3250 MWt core power level. Also, a safety evaluation had been performed which modeled an increased auxiliary feedwater enthalpy delay time. These assumptions were incorporated in the increased MSSV setpoint tolerance NOTRUMP analysis of the limiting 3 inch break for Unit 1. However, in order to obtain a direct sensitivity for the increased MSSV setpoint tolerance, a NOTRUMP analysis was also performed incorporating these assumptions but modelling the original MSSV setpoints.

In addition, a 3 inch NOTRUMP analysis was performed for the low pressure, high temperature operating condition for Unit 1 since a safety evaluation had been originally performed as part of the licensing basis analysis. The increased MSSV setpoint tolerance, a core power level of 3250 MWt with the high head cross tie valve closed, and a 25 gpm charging pump flow imbalance were assumed for the analysis of the low pressure, high temperature case.

Donald C. Cook Unit 2 was reanalyzed for the limiting 3 inch break, low pressure and high temperature operating condition with the high head cross tie valve closed. The power shape axial offset was reduced from the licensing basis analysis of +30% to +13% for the MSSV increase analysis. An axial offset of +13% is equal to the value assumed in the licensing basis large break LOCA analysis. In addition, the licensing basis analysis conservatively assumed a maximum assembly average power (P_{HA}) of 1.519. The 3% increased MSSV setpoint tolerance analysis assumed a P_{HA} which was reduced to 1.46. In order to obtain a direct sensitivity for the increased MSSV setpoint tolerance, a NOTRUMP analysis was performed incorporating these assumptions but modelling the original MSSV setpoints.

Tables 5 and 6 summarize the MSSV setpoints used in the Donald C. Cook Units 1 and 2 current licensing basis small break LOCA analyses and the increased MSSV setpoint tolerance analyses, respectively. Tables 7 and 8 summarize the initial input assumptions used in the Unit 1 analysis. The Unit 2 initial input assumptions are summarized in Table 9.

The time sequence of events and results of the Unit 1 analysis are summarized in Tables 10 and 11, respectively. The limiting peak clad temperature calculated is 1879°F, including a 25°F burst and blockage penalty, for the 3% increased MSSV setpoint tolerance case at 3250 MWt and the low pressure, low temperature operating conditions*. This value is less than the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is 3.47%, which is well below the embrittlement limit of 17% as required by 10CFR50.46. The total core metal-water reaction is less than 1.0%, corresponding to less than 1.0 percent hydrogen generation, as compared to the 1% criterion of 10CFR50.46.

The time sequence of events and results of the Unit 2 analysis are summarized in Tables 12 and 13, respectively. The limiting peak clad temperature calculated is 2125°F, including a 12°F artificial leak-by penalty and 157°F burst and blockage penalty, for the 3% increased MSSV setpoint tolerance case at 3250 MWt and low pressure, high temperature operating condition. This value is less than the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is 4.26%, which is

* These results are from calculations using a nominal auxiliary feedwater flow. A subsequent analysis using a more conservative minimum auxiliary feedwater flow rate is presented in the next section.

well below the embrittlement limit of 17% as required by 10CFR50.46. The total core metal-water reaction is less than 1.0%, corresponding to less than 1.0 percent hydrogen generation, as compared to the 1% criterion of 10CFR50.46.

<u>Additional Small Break LOCA Analyses</u>

The small break LOCA analysis for Cook Unit 1, previously discussed, used nominal Auxiliary Feedwater (AFW) flow rates (1258 gpm total delivery), whereas minimum AFW flow rates were used for Cook Unit 2. Since minimum AFW flow rates are more limiting, the small break LOCA for Cook Unit 1 for ±3% MMSV setpoint tolerance was reanalyzed using lower auxiliary feedwater flow rates (750 gpm total delivery). The following presents the results of the revised small break LOCA analyses performed for Donald C. Cook Unit 1.

Based on the Cook Unit 1 analyses presented in the previous section, two additional small break LOCA cases were run to address a relaxation to ±3% for the MSSV setpoint tolerance. First, the original LPLT (Low Pressure, Low Temperature) case presented above, the results of which are shown in Tables 10 and 11, was rerun modeling 750 gpm total AFW system flow rate. As was demonstrated in References 11 and 12, the LPLT case is the limiting case for the pressure/temperature operating window for Cook Unit 1, and that will not change due to the reduction in AFW flow. In addition, since only the limiting break size (3 inch) was previously analyzed, a 2 inch break was also analyzed for the 750 gpm AFW flow rate to provide further assurance that the limiting break size has not shifted to a smaller break size due to the reduction in the AFW flow rate. Note that since both the reduction in AFW delivered flow and the increase in the setpoint tolerance to ±3% tend to shift the limiting break size to a smaller break, it is not necessary to consider that the limiting break could be larger than was presented in the current licensing basis analysis which demonstrated that the 3 inch break is limiting.

The MSSV performance assumed in these new cases is shown in Table 6. The initial input parameters assumed for these new cases are shown in Table 7a, and are compared with the original licensing basis in Reference 11. If the new analysis values from Table 7a are compared with the original evaluation cases shown in Table 7, very few differences are evident. Except for the auxiliary feedwater flow rate and a slight increase in the accumulator water temperature, the initial RCS

pressure was lowered to cover a safety evaluation that was performed for pressurizer pressure uncertainty. Incorporating this new RCS pressure had a negligible effect on the vessel inlet and outlet temperatures and the steam pressure assumed for reactor steady-state operation (prior to initiation of the transient). One final additional change is in the AFW enthalpy delay. The lower AFW flow rate would result in a longer delay. The current NOTRUMP model has been improved to model the volume of hot main feedwater that must be purged from the piping prior to cold AFW being delivered to the steam generator, and the delay is calculated by the model. Other than these minor differences, and the intended change (i.e., reduce AFW flow rate and increase accumulator water temperature), the initial conditions assumed for the additional runs are identical to the runs performed for the previous section.

The time sequence of events and results of the Unit 1 analyses are summarized in Tables 10a and 11a, respectively. The limiting Peak Clad Temperature (PCT) calculated is 2068°F, including a 117°F burst and blockage penalty, for the ±3% increased MSSV setpoint tolerance case at 3250 MWt and low pressure, low temperature operating condition. This value is less than the acceptance criteria limit of 2200°F, and is almost the same computed result that is seen for D. C. Cook Unit 2 (the pre-burst/blockage PCT of 1951°F versus 1956°F). The maximum local metal-water reaction is 5.06%, which is well below the embrittlement limit of 17% as required by 10 CFR 50.46. The total core metal-water reaction is less than 1.0%, corresponding to less than 1.0 percent hydrogen generation, as compared to the 1% criterion of 10 CFR 50.46.

The ±3% increased MSSV setpoint tolerance has been analyzed for the Donald C. Cook Nuclear Plant Unit 1 for the small break LOCA analyses performed by Westinghouse. The potential effect of this change on the FSAR analysis results for the small break LOCA analysis was examined via reanalysis and although the results are more limiting than previous analysis cases, it was shown that the effect of the increased MSSV setpoint tolerance did not result in exceeding any of the following design or regulatory limits:

1. The calculated peak fuel element cladding temperature is below the requirements of 2200°F.

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

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

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

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

Therefore, it is concluded that a relaxation to ±3% for the MSSV setpoint tolerance is acceptable from the standpoint of the small break LOCA FSAR accident analyses discussed in this safety evaluation.

Post-LOCA Long Term Core Cooling

The Westinghouse licensing position for satisfying the requirements of 10CFR50.46 Paragraph (b), Item (5), "Long Term Cooling," concludes that the reactor will remain shut down by borated ECCS water residing in the RCS/sump after a LOCA. Since credit for the control rods is not taken for a large break LOCA, the borated ECCS water provided by the accumulators and the RWST must have a boron concentration that, when mixed with other water sources, will result in the reactor core remaining subcritical assuming all control rods out. The calculation is based upon the reactor steady state conditions at the initiation of a LOCA and considers sources of both borated and unborated fluid in the post-LOCA containment sump. The steady state conditions are obtained from the large break LOCA analysis which, as stated above, does not take credit for MSSV actuation. Thus the post-LOCA long-term core cooling evaluation is independent of the MSSV setpoint tolerance, and there will be no change in the calculated RCS/sump boron concentration after a postulated LOCA for Donald C. Cook Units 1 and 2.

Hot Leg Switchover to Prevent Potential Boron Precipitation

Post-LOCA hot leg recirculation time is determined for inclusion in emergency operating procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This time is

dependent on power level and the RCS, RWST, and accumulator water volumes and with their associated boron concentrations. The proposed MSSV setpoint tolerance increase to 3% does not affect the power level or the boron concentrations assumed for the RCS, RWST, and accumulator in the hot leg switchover calculation for Unit 1. The proposed MSSV setpoint tolerance increase to 3% does not affect the boron concentrations assumed for the RCS, RWST, and accumulator in the hot leg switchover calculation for Unit 2. The current licensing basis hot leg switchover calculation for Unit 2 is at full power, 3413 MWt, with cross tie valve at closed position. With MSSV setpoint tolerance increased to 3%, Unit 2 LOCA analyses assumed a reduced core power, 3250 MWt, with cross tie valve at closed position. A reduction in power reduces the boil-off rate in the hot leg switchover calculation. A reduction in the boil-off rate results in the rate of boron build up also being reduced. Therefore, the licensing basis hot leg switchover calculation for the Donald C. Cook Units 1 and 2 remains bounding.

LOCA Hydraulic Forcing Functions

The peak hydraulic forcing functions on the reactor vessel and internals occur very early in the large break LOCA transient. Typically, the peak forcing functions occur between 10 and 50 milliseconds (0.01 and 0.05 seconds) and have subsided well before 500 milliseconds (0.50 seconds). Any change in time associated with an increased MSSV setpoint tolerance would occur several seconds into the transient. Since the LOCA hydraulic forcing functions have peaked and subsided before the time at which the MSSV may actuate, the increase in the MSSV setpoint tolerance to 3% will not impact the LOCA hydraulic forcing functions calculation for Donald C. Cook Units 1 and 2.

LOCA Conclusions

The effect of increasing the MSSV setpoint tolerance to 3% for Donald C. Cook Units 1 and 2 has been evaluated for each of the LOCA related analyses addressed in the UFSAR. For currently analyzed conditions, or for Unit 2 operation at a reduced power level of 3250 MWt when the high head cross tie valves are closed, it was shown that the 3% MSSV setpoint tolerance does not result in any design or Regulatory limit being exceeded. Therefore, with respect to the LOCA analyses, it can be concluded that increasing the MSSV setpoint tolerance to 3% for Donald C. Cook Units 1 and 2 will be acceptable from the standpoint of the UFSAR accident analyses discussed in the safety evaluation.

3. Containment Integrity Evaluation

Relaxation of the Donald C. Cook Units 1 & 2 Technical Specification Main Steam Safety Valve setpoint tolerances from ±1% to ±3% do not adversely affect the short term or long term LOCA mass and energy releases and, subsequently, the related containment analyses. Since there is no impact on the main steamline break mass and energy release calculations, there is also no impact on that associated containment response analysis. The proposed change does not affect the normal plant operating parameters, system actuations, accident mitigating capabilities or assumptions important to the mass and energy release and containment analyses, or create more limiting conditions than those already assumed in the current analyses. Therefore, the conclusions presented in the Donald C. Cook UFSAR remain valid with respect to containment.

4. Steam Generator Tube Rupture

To demonstrate that an unreviewed safety question does not exist for the steam generator tube rupture (SGTR) event, the increased MSSV setpoint tolerance was evaluated for Donald C. Cook Units 1 and 2. The analysis for uprating to 3600 MWT considered up to 15% steam generator tube plugging for both Units 1 and 2. The limiting cases from this analysis were reevaluated for the increased MSSV setpoint tolerance. An increased steam generator tube plugging level of 20% was also considered at power levels of 3262 MWT for Unit 1 and 3425 MWT for Unit 2. The criteria stated in the UFSAR analysis for Donald C. Cook were used in establishing the continued applicability of the SGTR licensing basis safety analysis by demonstrating that the conclusions for SGTR UFSAR analysis remain valid.

An evaluation has been performed to determine the impact on the Donald C. Cook Units' SGTR analysis of record for increased MSSV setpoint tolerance for all the cases with different steam generator tube plugging and power levels stated above. The primary thermal hydraulic parameters which affect the calculation of offsite radiation doses for a SGTR are the amount of radioactivity assumed to be present in the reactor coolant, the amount of reactor coolant transferred to the secondary side of the ruptured steam generator through the ruptured tube, and the amount of steam released from the ruptured steam generator to the atmosphere. Thus, the calculated offsite radiation doses for an SGTR for Donald C. Cook are dependent on these three factors.

For the UFSAR SGTR analysis, the activity in the reactor coolant is based on an assumption of 1% defective fuel, and this assumption will not be affected by the increased MSSV setpoint tolerance. The two remaining factors are affected by the increased MSSV setpoint tolerance, and the evaluation was performed to quantify this effect.

To evaluate the effect of the increased MSSV setpoint tolerance on the Donald C. Cook SGTR analysis, the revised SG safety valve set pressure was lowered by 3% from 1080 psia to 1047.6 psia. This resulted in a slightly higher equilibrium primary-to-secondary break flow (approximately 0.5%), since there was an increase in the pressure differential between the RCS and secondary side assumed in the analysis. The steam released to the atmosphere subsequently increased (by approximately 0.2%) because of the lower pressure assumed for the main steam safety valves. The limiting cases, for all power levels and steam generator tube plugging levels considered, were at 3600 MWt.

The thyroid and whole body doses estimated for Units 1 and 2, based on the analyses described above, are bounded by those previously determined for the rerating program. The actual estimated dose factors (compared to the results of the rerating calculation) are as follows:

Unit 1: thyroid 0.7, whole body 1.005
Unit 2: thyroid 0.99, whole body 0.98

Although the Unit 1 whole body dose exceeds the previous value by approximately 0.5%, this increase is well within the acceptable limit. Thus, the results and conclusion in the Donald C. Cook UFSAR that the offsite doses for an SGTR event would be within a small fraction of the 10CFR100 guidelines remains valid.

5. Component Performance

The relaxation of the lift setpoint tolerance for the MSSVs at Donald C. Cook does not directly or indirectly involve mechanical component hardware considerations. Direct effects as well as indirect effects on equipment important to safety (ITS) have been considered. Indirect effects include activities which involve non-safety related equipment which may affect ITS equipment. Component hardware considerations may include overall component integrity, sub-component integrity, and the

adequacy of component supports during all plant conditions. An evaluation is not required to determine whether the condition alters the design, material, construction standards, function or method of performing the function of any ITS equipment.

6. Systems Evaluation

The relaxation of the lift setpoint tolerance for the MSSVs at Donald C. Cook as described would not affect the integrity of a plant auxiliary fluid system or the ability of any auxiliary system to perform its intended safety function.

7. Radiological Evaluation

The relaxation of the lift setpoint tolerance for the MSSVs at Donald C. Cook as described do not affect radiological concerns other than those identified above in Section III.4 or post-LOCA hydrogen production. The evaluation in Sections III.1 and III.3 concluded that the existing mass releases used in the remaining offsite dose calculations (i.e., steamline break, rod ejection, locked rotor, and short-term & long-term LOCA) are still applicable.

8. Plant Risk Analyses (activities affecting IPE)

The relaxation of the lift setpoint tolerance for the MSSVs at Donald C. Cook does not adversely affect the Individual Plant Examination (IPE) for the plant. This test does not affect the normal plant operating parameters, system actuations, accident mitigating capabilities, operating procedures or assumptions important to the IPE analyses, or create conditions that would significantly affect core damage or plant damage frequency or the frequency of core damage initiating events. Therefore, the conclusions presented in the IPE remain valid.

9. Plant Risk Analyses (changes other than IPE-related)

The relaxation of the lift setpoint tolerance for the MSSVs does not result in an increase in the probability of occurrence of accidents previously evaluated in the UFSAR. This proposed change to the Technical Specifications does not result in an increase in the probability of occurrence of a

malfunction of equipment important to safety or of equipment that could indirectly affect equipment important to safety.

10. I&C Systems

The relaxation of the lift setpoint tolerance for the MSSVs does not directly or indirectly involve electrical systems, components, or instrumentation considerations. Direct effects as well as indirect effects on equipment important to safety have been considered. Indirect effects include conditions or activities which involve non-safety related electrical equipment which may affect Class 1E, post accident monitoring systems, or plant control electrical equipment. Consideration has been given to seismic and environmental qualification, design and performance criteria per IEEE standards, functional requirements, and plant technical specifications with respect to all plant conditions. An evaluation is not required to determine whether the MSSV setpoint tolerance relaxation alters the design, configuration, qualification, or performance of safety related electrical systems or components. The MSSV setpoint tolerance relaxation has no potential for impact to the identification of an unresolved safety question as it would relate to the safety related function of electrical systems of components.

11. Technical Specifications

A review of the Donald C. Cook Unit 1 and Unit 2 Technical Specifications was performed to address a change in the lift setpoint tolerance for the Main Steam Safety Valves. The Technical Specification review, inclusive of Amendments 157 and 141 for Units 1 and 2, respectively. Proposed markups are attached to this evaluation for both Unit 1 and Unit 2, and reflect changes to Table 4.7-1 and 3.7-4, respectively. A change to the basis for both units is also proposed and discusses the relationship between the $\pm 1\%$ and $\pm 3\%$ tolerances.

IV. ASSESSMENT OF NO UNREVIEWED SAFETY QUESTION

The relaxation in the lift setpoint tolerance for the MSSVs at Donald C. Cook Units 1 and 2 has been evaluated consistent with the requirements of 10CFR50.59 and does not involve an unreviewed safety question on the basis of the following justifications:

1. Will the probability of an accident previously evaluated in the SAR be increased?

   No. The ±3% tolerance on the MSSV setpoint does not increase the probability of an accident previously evaluated in the UFSAR. There are no hardware modifications to the valves and, therefore, there is no increase in the probability of a spurious opening of a MSSV. The MSSVs are actuated to protect the secondary systems from overpressurization after an accident is initiated. Sufficient margin exists between the normal steam system operating pressure and the valve setpoints with the increased tolerance to preclude an increase in the probability of actuating the valves. Therefore, the probability of an accident previously evaluated in the UFSAR would not be increased as a result of increasing the MSSV lift setpoint tolerance by 3% above or below the current Technical Specification setpoint value.

2. Will the consequences of an accident previously evaluated in the SAR be increased?

   No. Based on the discussions presented within, all of the applicable LOCA and non-LOCA design basis acceptance criteria remain valid both for the transients evaluated and the single event analyzed. Additionally, no new limiting single failure is introduced by the proposed change. The DNBR and PCT values remain within the specified limits of the licensing basis. Although increasing the valve setpoint will increase the steam release from the ruptured steam generator above the UFSAR value by approximately 0.2%, the SGTR analysis indicates that the calculated doses are bounded by those determined for the rerating program which, in turn, are within a small fraction of the 10CFR100 dose guidelines. The evaluation also concluded that the existing mass releases used in the offsite dose calculations for the remaining transients (i.e., steamline break, rod ejection) are still applicable. Therefore, based on the above, there is no increase in the dose consequences.

3. May the possibility of an accident which is different than any already evaluated in the SAR be created?

No. As previously indicated in Section III.1, the Inadvertent Opening of a SG Relief or Safety Valve event is currently presented in the Donald C. Cook UFSAR (Section 14.2.5) and is bounded by the Steamline Break analysis. Increasing the as-found lift setpoint tolerance on the MSSVs does not introduce a new accident initiator mechanism. No new failure modes have been defined for any system or component important to safety nor has any new limiting single failure been identified. No accident will be created that will increase the challenge to the MSSVs and result in increased actuation of the valves. Therefore, the possibility of an accident different than any already evaluated in the UFSAR is not created.

4. Will the probability of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

No. Although the proposed change takes place in equipment utilized to prevent overpressurization on the secondary side and to provide an additional heat removal path, increasing the as-found lift setpoint tolerance on the MSSVs will not adversely affect the operation of the reactor protection system, any of the protection setpoints, or any other device required for accident mitigation. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the UFSAR will not be increased.

5. Will the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

No. As discussed in the response to Questions 2 and 4, there is no increase in the dose release consequences as a result of increasing the as-found lift setpoint tolerance on the MSSVs as defined in the attached safety evaluation.

6. May the possibility of a malfunction of equipment important to safety different than any already evaluated in the SAR be created?

   No. As discussed in Question 4, an increase in the as-found lift setpoint tolerance on the MSSVs will not impact any other equipment important to safety. Therefore, the possibility of a malfunction of equipment important to safety different than any already evaluated in the UFSAR will not be created.

7. Will the margin of safety as defined in the bases to any technical specification be reduced?

   No. As discussed in the attached safety evaluation, the proposed increase in the as-found MSSV lift setpoint tolerance will not invalidate the LOCA or non-LOCA conclusions presented in the UFSAR accident analyses. The new loss of load/turbine trip analysis concluded that all applicable acceptance criteria are still satisfied. For all the UFSAR non-LOCA transients, the DNB design basis, primary and secondary pressure limits, and dose limits continue to be met. Peak cladding temperatures remain below the limits specified in 10CFR50.46. The calculated doses resulting from a steam generator tube rupture event remain within a small fraction of the 10CFR100 permissible releases. Thus, there is no reduction in the margin to safety. Note that, as identified earlier, changes will be required to the plant Technical Specifications in order to implement the proposed change.

V. CONCLUSIONS

The proposed change to main steam safety valve lift setpoint tolerances from ±1% to ±3% has been evaluated by Westinghouse. The preceding analyses and evaluations have determined that operation with the MSSV setpoints within a ±3% tolerance about the nominal values will have no adverse impact upon the licensing basis analyses, as well as the steamline break mass & energy release rates inside and outside of containment. In addition, it is concluded that the ±3% tolerance on the MSSV setpoint does not adversely affect the overpower or overtemperature protection system. As a result, adequate protection to the core limit lines continues to exists. Therefore, all licensing basis criteria continue to be satisfied and the conclusions in the UFSAR remain valid.

Thus, based on the information presented above, it can be concluded that the proposed increase of main steam safety valve lift setpoint tolerances from ±1% to ±3% does not represent an unreviewed safety question per the definition and requirements defined in 10CFR50.59.

The recommended Technical Specification changes, along with a no significant hazards evaluation, are presented as appendices to this evaluation.

VI. REFERENCES

1) Donald C. Cook Units 1 & 2 Technical Specifications through Amendments 157 and 141, respectively, 10/1/91.

2) ANSI/ASME BPV-III-1-NB, "ASME Boiler and Pressure Vessel Code - Section III Rules for Construction of Nuclear Power Plant Components," ASME, 1983.

3) ANSI/ASME OM-1-1981, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices," ASME, 1981.

4) "Donald C. Cook Units 1 & 2 Updated Final Safety Analysis Report (UFSAR), dated through July 1991.

5) ASME Steam Tables, Fifth Edition, 1983.

6) Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, June 1972.

7) Chelemer, H. et al.,"Improved Thermal Design Procedure," WCAP-8567-P-A, February 1989.

8) Butler, J.C. and D.S. Love,"Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," WCAP-10961-P, October 1985.

9) 90AE*-G-0126 W/AEP2-0098 Transmittal regarding "Locked Rotor Dose Analysis for Donald C. Cook Unit 2 Cycles 8 & 9," 7/19/90.

10) Letter regarding AFW flow rates from R. B. Bennett of American Electric Power to J. N. Steinmetz of Westinghouse Electric, 9/24/91.

11) WCAP-10054-P-A (Proprietary), WCAP-10081 (Non-Proprietary), Lee, H., et al., Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985.

12) WCAP-12135, Donald C. Cook Nuclear Plant Units 1 and 2 Rerating Engineering Report, Vol. 1, September 1989.

TABLE 3

UNIT 1
TURBINE TRIP SEQUENCE

Accident	Event	Time (sec)
Without pressurizer control (minimum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip setpoint reached	7.7
	Rods begin to drop	9.7
	Peak pressurizer pressure occurs	10.5
	Minimum DNBR occurs	*
Without pressurizer control (maximum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip setpoint reached	7.9
	Rods begin to drop	9.9
	Peak pressurizer pressure occurs	10.5
	Minimum DNBR occurs	*
With pressurizer control (maximum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	Peak pressurizer pressure occurs	10.0
	Low-low steam generator water level reactor trip setpoint reached	47.1
	Rods begin to drop	49.1
	Minimum DNBR occurs	*
With pressurizer control (minimum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip setpoint reached	12.4

TABLE 3 (continued)

UNIT 1
TURBINE TRIP SEQUENCE

Accident	Event	Time (sec)
	Rods begin to drop	14.4
	Peak pressurizer pressure occurs	16.0
	Minimum DNBR occurs	15.5

* DNBR does not decrease below its initial value.

TABLE 4

UNIT 2
TURBINE TRIP SEQUENCE OF EVENTS

Accident	Event	Time (sec) mixed core	Time (sec) full core
Without pressurizer control (minimum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0	0.0
	High pressurizer pressure reactor trip setpoint reached	5.5	7.5
	Rods begin to drop	7.5	9.5
	Peak pressurizer pressure occurs	9.5	11.0
	Minimum DNBR occurs	*	*
Without pressurizer control (maximum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0	0.0
	High pressurizer pressure reactor trip setpoint reached	5.5	7.6
	Rods begin to drop	7.5	9.6
	Peak pressurizer pressure occurs	9.0	10.0
	Minimum DNBR occurs	*	*

*DNBR does not decrease below its initial value.

TABLE 4 (continued)

UNIT 2
TURBINE TRIP SEQUENCE OF EVENTS

Accident	Event	Time (sec) mixed core	Time (sec) full core
With pressurizer control (maximum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0	0.0
	Peak pressurizer pressure occurs	7.0	7.5
	Low-low steam generator water level reactor trip setpoint reached	60.1	52.8
	Rods begin to drop	62.1	54.8
	Minimum DNBR occurs	*	*
With pressurizer control (minimum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0	0.0
	High pressurizer pressure reactor trip setpoint reached	10.6	11.2
	Rods begin to drop	12.6	13.2
	Peak pressurizer pressure occurs	13.5	14.5
	Minimum DNBR occurs	14.5	15.0

*DNBR does not decrease below its initial value.

TABLE 5

CURRENT LICENSING BASIS
STEAM LINE SAFETY VALVES PER LOOP

Safety Valve	Setpoint Pressure (psig)	Percent Accumulation	Accumulation Pressure (psig)	Flowrate @ Acc Pressure (lbs/hr)*
1A	1065	10.0	1171.5	857690
1B	1065	10.0	1171.5	857690
2A	1075	8.98	1171.5	857690
2B	1075	8.98	1171.5	857690
3	1085	7.97	1171.5	857690

* The rated valve capacity at full accumulation pressure was calculated as follows:

$$51.5 \times A \times K \times P = \text{Actual Flowrate}$$

where:

A = Valve orifice area = 16 in^2
K = Coefficient of discharge = 0.975
P = Pressure (psia) at accumulation pressure

The above actual flowrate is reduced by 0.9 to get the valve rated capacity.

TABLE 6

MSSV SETPOINT INCREASE
STEAM LINE SAFETY VALVES PER LOOP

Safety Valve	Setpoint Pressure @ +3% (psig)	Percent Accumulation	Accumulation Pressure (psig)	Flowrate @ Acc Pressure (lbs/hr)*
1A	1096.95	3.0	1129.86	827585.6
1B	1096.95	3.0	1129.86	827585.6
2A	1107.25	3.0	1140.47	835257.2
2B	1107.25	3.0	1140.47	835257.2
3	1117.55	3.0	1151.08	842928.9

* The rated valve capacity at full accumulation pressure was calculated as follows:

$$51.5 \times A \times K \times P = \text{Actual Flowrate}$$

where:

A = Valve orifice area = 16 in^2
K = Coefficient of discharge = 0.975
P = Pressure (psia) at accumulation pressure

The above actual flowrate is reduced by 0.9 to get the valve rated capacity.

TABLE 7

LOW PRESSURE, LOW TEMPERATURE

	Current Licensing <u>Basis</u>	MSSV Setpoint <u>Increase</u>
License Core Power[1] (MWt)	3588[2]	3250
Total Peaking Factor, F_Q	2.32	2.32
Axial Offset (%)	+30	+30
Hot Channel Enthalpy Rise Factor, F_H	1.55	1.55
Maximum Assembly Average Power, P_{HA}	1.433	1.433
Fuel Assembly Array	15 X 15 OFA	
Accumulator Water Volume (ft^3)	946	946
Accumulator Tank Volume (ft^3)	1350	1350
Minimum Accumulator Gas Pressure, (psia)	600	600
Loop Flow (gpm)	354000	354000
Vessel Inlet Temperature (F)[3]	509.89	513.23
Vessel Outlet Temperature (F)[3]	581.71	578.57
RCS Pressure (psia)	2100	2100
Steam Pressure (psia)[3]	564.36	596.48
Steam Generator Tube Plugging Level (%)	15	15
Maximum Refueling Water Storage Tank Temperature (F)	120	120
Maximum Condensate Storage Tank Temperature (F)	120	120
Fuel Backfill Pressure (psig)	275	275
Reactor Trip Setpoint (psia)	1860	1860
Safety Injection Signal Setpoint (psia)	1715	1715
Safety Injection Delay Time (sec)	27	27
Safety Injection Pump Degradation (%)	10	10
Charging Pump Flow Imbalance (gpm)	10	25
HHSI Cross Tie Valve Position	Closed	Closed
Signal Processing Delay and Rod Drop Time (sec)	2.0	4.4
Reactor Coolant Pump Delay Time (sec)	4.4	4.4
Main Feedwater Isolation Delay Time (sec)	0.0	0.0
Main Feedwater Valve Closure Time (sec)	8.0	8.0
Auxiliary Feedwater Enthalpy Delay Time (sec)	60[4]	272
Main Steam Safety Valve Setpoint (psia)	Table 5	Table 6

1 Two percent is added to this power to account for calorimetric error.
2 A safety evaluation for 25 gpm charging flow imbalance limits operation with HHSI cross tie valve closed to 3250 MWt.
3 Value is based on 102% core power, main coolant pump heat neglected, and best estimate Tavg.
4 A safety evaluation was performed to account for a auxiliary feedwater enthalpy delay of 272 seconds.

TABLE 7a

Initial Input Parameters for the Small Break LOCA Analysis

	Current Licensing Basis	MSSV Setpoint Increase
License Core Power[1] (MWt)	3588[2]	3250
Total Peaking Factor, F_Q^T	2.32	2.32
Axial Offset (%)	+30	+30
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}^N$	1.55	1.55
Maximum Assembly Average Power, P_{HA}	1.433	1.433
Fuel Assembly Array	15 X 15 OFA	
Accumulator Water Volume (ft^3)	946	946
Accumulator Tank Volume (ft^3)	1350	1350
Minimum Accumulator Gas Pressure, (psia)	600	600
Loop Flow (gpm)	354000	354000
Vessel Inlet Temperature (°F)[3]	509.89	513.20
Vessel Outlet Temperature (°F)[3]	581.71	578.44
RCS Pressure (psia)	2100	2033
Steam Pressure (psia)[3]	564.36	596.11
Steam Generator Tube Plugging Level (%)	15	15
Maximum Refueling Water Storage Tank Temperature (°F)	120	120
Maximum Condensate Storage Tank Temperature (°F)	120	120
Fuel Backfill Pressure (psig)	275	275
Reactor Trip Setpoint (psia)	1860	1860
Safety Injection Signal Setpoint (psia)	1715	1715
Safety Injection Delay Time (sec)	27	27
Safety Injection Pump Degradation (%)	10	10
Charging Pump Flow Imbalance (gpm)	10	25
HHSI Cross Tie Valve Position	Closed	Closed
Signal Processing Delay and Rod Drop Time (sec)	2.0	4.4
Reactor Coolant Pump Delay Time (sec)	4.4	4.4
Main Feedwater Isolation Delay Time (sec)	0.0	0.0
Main Feedwater Valve Closure Time (sec)	8.0	8.0
Auxiliary Feedwater Total Delivery (gpm)	1258	750
Auxiliary Feedwater Delivery Delay Time (sec)	60[4]	60[5]
Main Steam Safety Valve Setpoint (psia)	Table 1	Table 2
Accumulator Temperature (°F)	120	130

1 Two percent is added to this power to account for calorimetric error.

2 A safety evaluation for 25 gpm charging flow imbalance limits operation with HHSI cross tie valve closed to 3250 MWt.

3 Value is based on 102% core power, main coolant pump heat neglected, and best estimate T_{AVG}.

4 A safety evaluation was performed to account for an auxiliary feedwater enthalpy delay of 272 seconds.

5 Enthalpy delay computed internally based on AFW flow rate and 75 ft^3 purge volume.

TABLE 8

LOW PRESSURE, HIGH TEMPERATURE

	Current Licensing Basis[2]	MSSV Setpoint Increase
License Core Power[1] (MWt)	NA	3250
Total Peaking Factor, F_Q	NA	2.32
Axial Offset (%)	NA	+30
Hot Channel Enthalpy Rise Factor, F_H	NA	1.55
Maximum Assembly Average Power, P_{HA}	NA	1.433
Fuel Assembly Array	NA	15X15 OFA
Accumulator Water Volume (ft^3)	NA	946
Accumulator Tank Volume (ft^3)	NA	1350
Minimum Accumulator Gas Pressure, (psia)	NA	600
Loop Flow (gpm)	NA	354000
Vessel Inlet Temperature (F) [3]	NA	543.63
Vessel Outlet Temperature (F)[3]	NA	606.79
RCS Pressure (psia)	NA	2100
Steam Pressure (psia)[3]	NA	793.90
Steam Generator Tube Plugging Level (%)	NA	15
Maximum Refueling Water Storage Tank Temperature (F)	NA	120
Maximum Condensate Storage Tank Temperature (F)	NA	120
Fuel Backfill Pressure (psig)	NA	275
Reactor Trip Setpoint (psia)	NA	1860
Safety Injection Signal Setpoint (psia)	NA	1715
Safety Injection Delay Time (sec)	NA	27
Safety Injection Pump Degradation (%)	NA	10
Charging Pump Flow Imbalance (gpm)	NA	25
HHSI Cross Tie Valve Position	NA	Closed
Signal Processing Delay and Rod Drop Time (sec)	NA	4.4
Reactor Coolant Pump Delay Time (sec)	NA	4.4
Main Feedwater Isolation Delay Time (sec)	NA	0.0
Main Feedwater Valve Closure Time (sec)	NA	8.0
Auxiliary Feedwater Enthalpy Delay Time (sec)	NA	272
Main Steam Safety Valve Setpoint (psia)	NA	Table 6

1 Two percent is added to this power to account for calorimetric error.
2 A safety evaluation for the low pressure, high temperature operating condition was performed in the licensing basis analysis.
3 Value is based on 102% core power, main coolant pump heat neglected, and best estimate Tavg.

TABLE 9

LOW PRESSURE, HIGH TEMPERATURE

	Current Licensing Basis	MSSV Setpoint Increase
License Core Power[1] (MWt)	3413	3250
Total Peaking Factor, F_Q	2.34	2.357
Axial Offset (%)	+30	+13
Hot Channel Enthalpy Rise Factor, F_H	1.644	1.666
Maximum Assembly Average Power, P_{HA}	1.519	1.46
Fuel Assembly Array	17 X 17 V5	
Accumulator Water Volume (ft^3)	946	946
Accumulator Tank Volume (ft^3)	1350	1350
Minimum Accumulator Gas Pressure, (psia)	600	600
Loop Flow (gpm)	354000	354000
Vessel Inlet Temperature (F) [2]	544.41	544.41
Vessel Outlet Temperature (F)[2]	610.19	610.19
RCS Pressure Including Uncertainties (psia)	2100	2100
Steam Pressure (psia) [2]	807.03	807.03
Steam Generator Tube Plugging Level (%)	15	15
Maximum Refueling Water Storage Tank Temperature (F)	120	120
Maximum Condensate Storage Tank Temperature (F)	120	120
Fuel Backfill Pressure (psig)	275	275
Reactor Trip Setpoint (psia)	1860	1860
Safety Injection Signal Setpoint (psia)	1715	1715
Safety Injection Delay Time (sec)	27	27
Safety Injection Pump Degradation (%)	10	10
Charging Pump Flow Imbalance (gpm)	25	25
HHSI Cross Tie Valve Position	Closed	Closed
Signal Processing Delay and Rod Drop Time (sec)	4.7	4.7
Reactor Coolant Pump Delay Time (sec)	4.4	4.4
Main Feedwater Isolation Delay Time (sec)	0.0	2.0
Main Feedwater Valve Closure Time (sec)	8.0	6.0
Auxiliary Feedwater Enthalpy Delay Time (sec)	349	349
Main Steam Safety Valve Setpoint (psia)	Table 5	Table 6

1 Two percent is added to this power to account for calorimetric error.
2 Value is based on 102% core power, main coolant pump heat neglected, and best estimate Tavg.

TABLE 10

TIME SEQUENCE OF EVENTS

Event	LPLT w/ MSSV	LPLT w/o MSSV	LPHT w/ MSSV	LPHT w/o MSSV
Break Occurs	0	0	0	0
Reactor trip signal	11.23	11.23	13.54	13.54
Safety injection signal	19.28	19.28	22.42	22.42
Start of safety injection signal	46.28	46.28	49.42	49.42
Loop seal venting	643.4	644.7	601.8	608.3
Loop seal core uncovery	NA	NA	NA	NA
Loop seal core recovery	NA	NA	NA	NA
Boil-off core uncovery	1139.2	1077.3	1073.4	1057.8
Accumulator injection begins	1730.0	1751.0	1647.8	1695.8
Peak clad temperature occurs	1935.5	1831.4	1872.3	1824.7
Top of core covered	NA	NA	NA	NA
SI flow rate exceeds break flow rate	1988	2024	2293	2284

LPLT is low pressure, low temperature operating condition.
LPHT is low pressure, high temperature operating condition.
W/ MSSV is main steam safety valve setpoint tolerance increase case at 3250 MWt core power.
W/O MSSV is licensing basis main steam safety valve setpoint tolerance case at 3250 MWt core power.

TABLE 10a

TIME SEQUENCE OF EVENTS

Event	Time (seconds) LPLT w/MSSV 3 inch Break	LPLT w/MSSV 2 inch Break
Break Occurs	0.0	0.0
Reactor trip signal	8.64	19.03
Safety injection signal	17.13	37.11
Start of safety injection	44.13	64.11
Start of auxiliary feedwater delivery	68.6	79.1
Loop seal venting	592	1390
Loop seal core uncovery	N/A	N/A
Loop seal core recovery	N/A	N/A
Boil-off core uncovery	984	2312
Accumulator injection begins	1680	N/A
Peak clad temperature occurs	1890	4042
Top of core covered	N/A	N/A
SI flow rate exceeds break flow rate	1890	4091

LPLT is low pressure, low temperature operating condition.
W/ MSSV is main steam safety valve setpoint tolerance increase case at 3250 MWt core power.

TABLE 11

SUMMARY OF RESULTS

	LPLT w/ MSSV	LPLT w/o MSSV	LPHT w/ MSSV	LPHT w/o MSSV
NOTRUMP Peak Clad Temperature (°F)	1853.7	1772.9	1837.7	1710.3
Peak Clad Temperature Location (ft)	11.75	11.75	11.75	11.75
Peak Clad Temperature Time (sec)	1935.5	1831.4	1872.3	1824.7
Local Zr/H_20 Reaction Maximum (%)	3.47	2.47	3.13	1.82
Local Zr/H_20 Reaction Location (ft)	11.75	11.75	11.75	11.75
Total Zr/H_20 Reaction (%)	<1.0	<1.0	<1.0	<1.0
Rod Burst	None	None	None	None
Burst and Blockage Penalty (°F)	25	15	16	15
Total Peak Clad Temperature (°F)	1878.7	1787.9	1853.7	1725.3

LPLT is low pressure, low temperature operating condition.
LPHT is low pressure, high temperature operating condition.
W/ MSSV is main steam safety valve setpoint tolerance increase case at 3250 MWt core power.
W/O MSSV is licensing basis main steam safety valve setpoint tolerance case at 3250 MWt core power.

TABLE 11a

SUMMARY OF RESULTS

	LPLT w/MSSV 3 inch Break	LPLT w/MSSV 2 inch Break
NOTRUMP Peak Clad Temperature (°F)	1951	1833
Peak Clad Temperature Location (ft)	12.0	12.0
Peak Clad Temperature Time (sec)	1890	4042
Local Zr/H_20 Reaction Maximum (%)	5.06	3.75
Local Zr/H_20 Reaction Location (ft)	12.0	12.0
Total Zr/H_20 Reaction (%)	0.568	0.397
Rod Burst	None	None
Burst and Blockage Penalty (°F)	117	15
Total Peak Clad Temperature (°F)	2068	1848

LPLT is low pressure, low temperature operating condition.
W/ MSSV is main steam safety valve setpoint tolerance increase case at 3250 MWt core power.

TABLE 12

TIME SEQUENCE OF EVENTS

Event	Time LPHT w/ MSSV	Time LPHT w/o MSSV
Break Occurs	0	0
Reactor trip signal	11.01	11.01
Safety injection signal	20.92	20.92
Start of safety injection signal	47.92	47.92
Loop seal venting	620.0	627.2
Loop seal core uncovery	NA	NA
Loop seal core recovery	NA	NA
Boil-off core uncovery	620.0	627.2
Accumulator injection begins	1604.3	1631.7
Peak clad temperature occurs	1691.0	1720.6
Top of core covered	NA	NA
SI flow rate exceeds break flow rate	1683.0	1984.0

LPHT is low pressure, high temperature operating condition.
W/ MSSV is main steam safety valve setpoint tolerance increase case at 3250 MWt core power.
W/O MSSV is licensing basis main steam safety valve setpoint tolerance case at 3413 MWt core power.

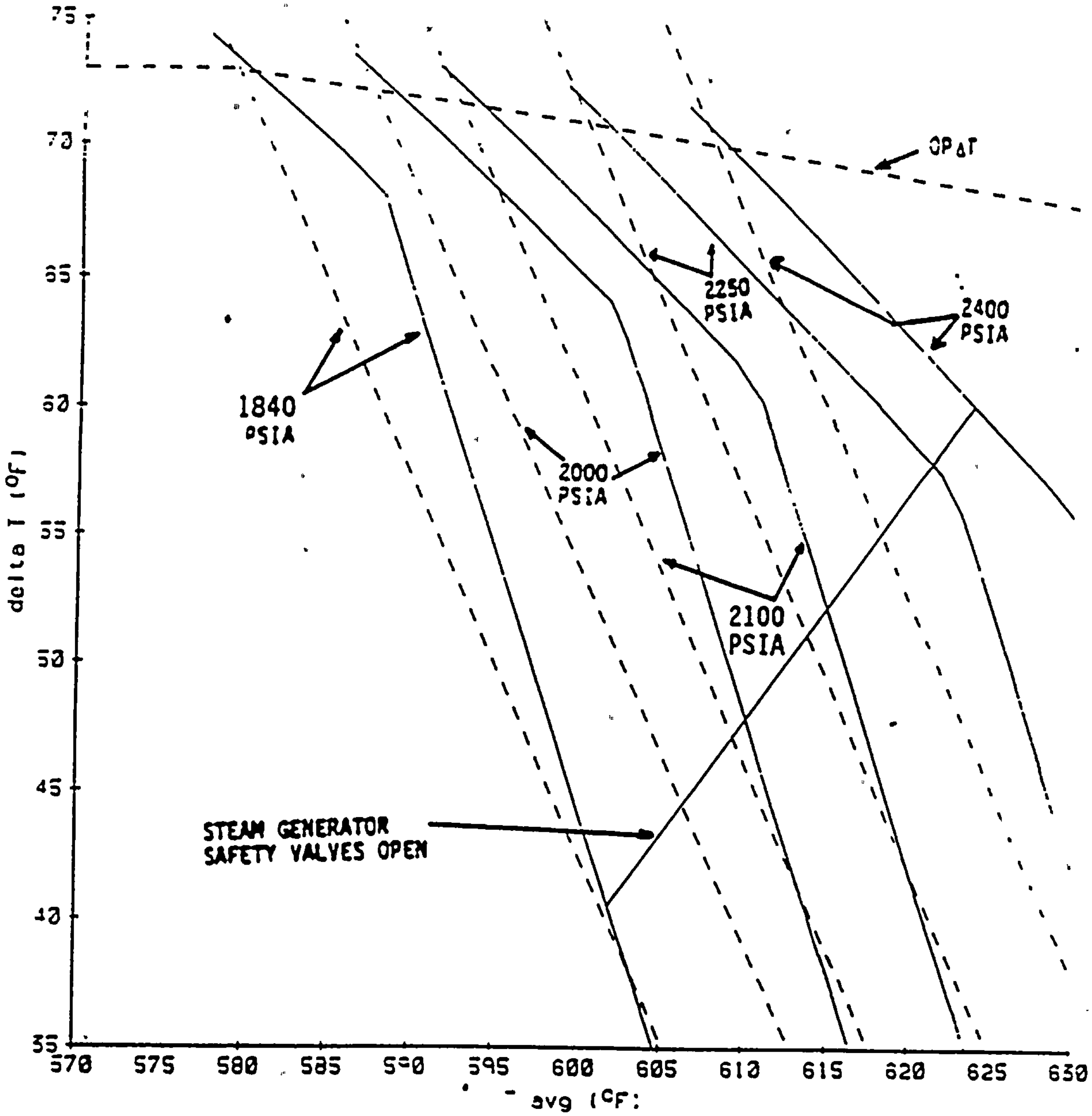
TABLE 13

SUMMARY OF RESULTS

	LPHT w/ MSSV	LPHT w/o MSSV
NOTRUMP Peak Clad Temperature (°F)	1955.9	1947.1
Peak Clad Temperature Location (ft)	11.75	11.75
Peak Clad Temperature Time (sec)	1691.0	1720.6
Local Zr/H_20 Reaction Maximum (%)	4.26	4.83
Local Zr/H_20 Reaction Location (ft)	11.75	11.75
Total Zr/H_20 Reaction (%)	<1.0	<1.0
Rod Burst	None	None
Artificial Leak-By Penalty (°F)	12	12
Burst and Blockage Penalty (°F)	157	143
Total Peak Clad Temperature (°F)	2124.9	2102.1

LPHT is low pressure, high temperature operating condition.
W/ MSSV is main steam safety valve setpoint tolerance increase case at 3250 MWt core power.
W/O MSSV is licensing basis main steam safety valve setpoint tolerance case at 3413 MWt core power.

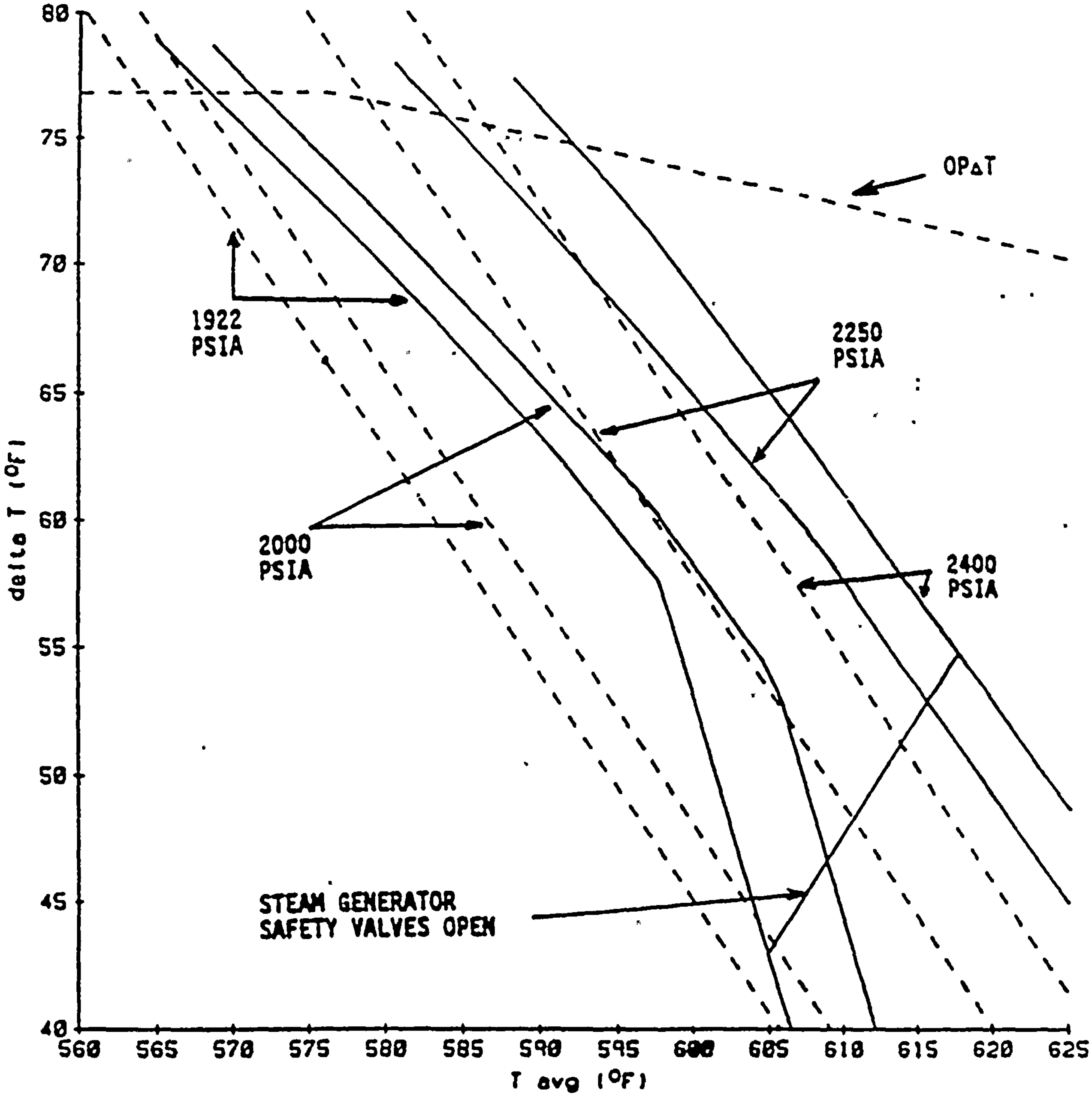
FIGURES



------ OTΔT
——— Core Limits

Nominal Tavg = 578.7°F
Nominal Pressure = 2100 psia

DONALD C. COOK UNIT 1
FIGURE 1a ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER DELTA T PROTECTION



------OT∆T Protection Lines

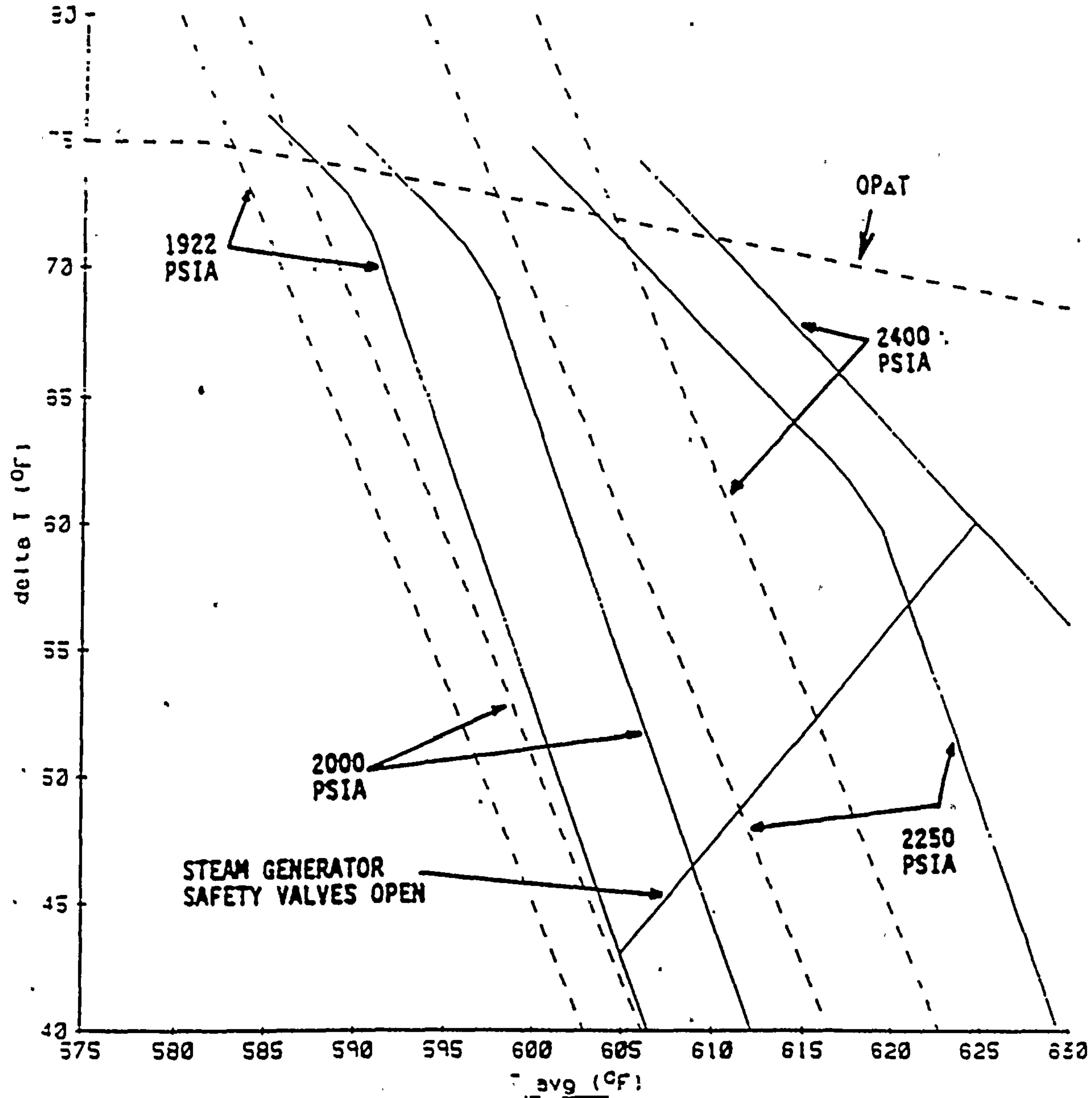
______Core Thermal Safety Limits

- Nominal Vessel Average Temperature = 576°F
- Nominal Pressurizer Pressure = 2250 psia

DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 1b

ILLUSTRATION OF OVERTEMPERATURE
AND OVERPOWER DELTA T PROTECTION

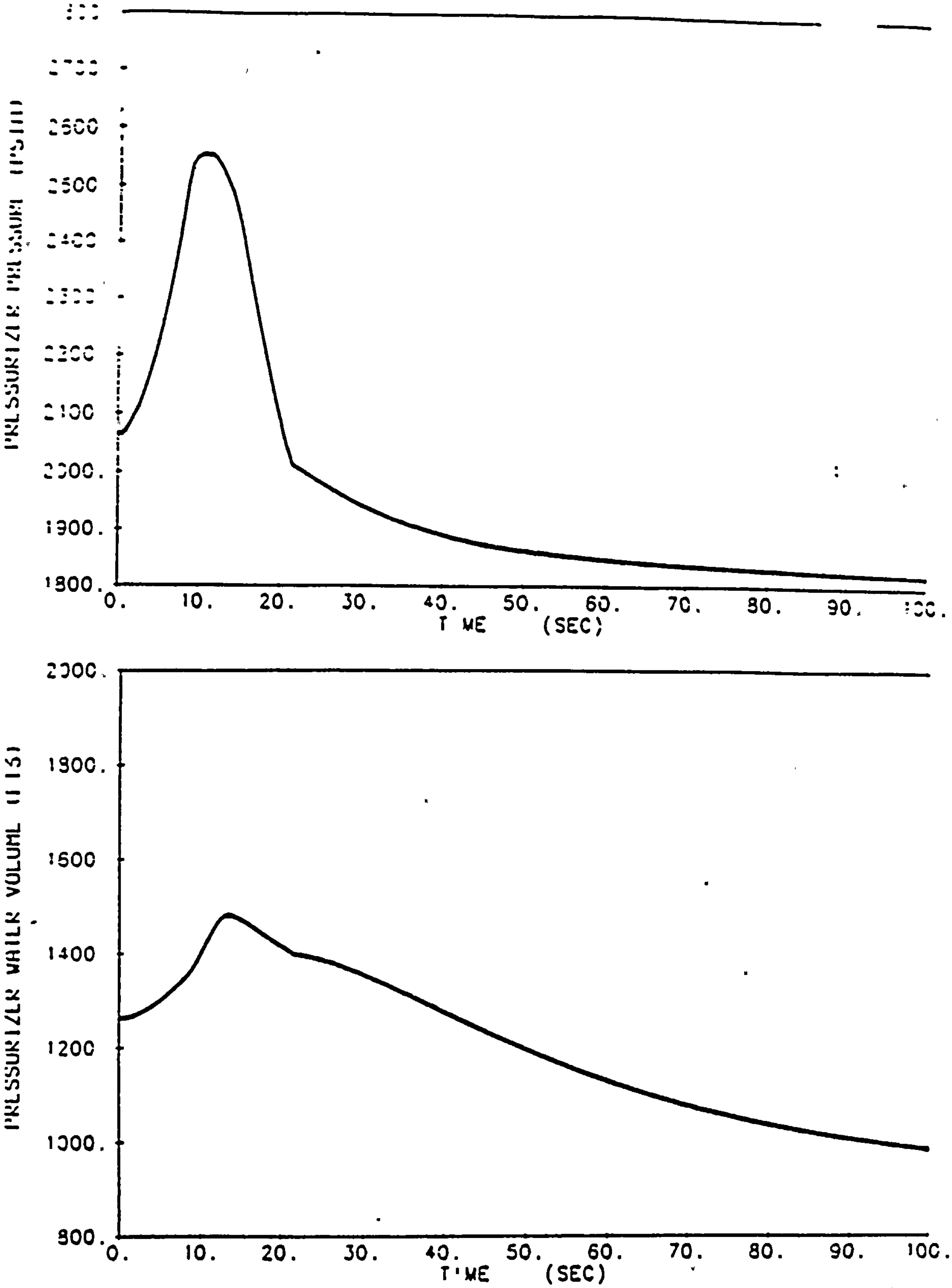


------OTΔT Protection Lines

______Core Thermal Safety Limits

- Nominal Vessel Average Temperature = 581.3°F
- Nominal Pressurizer Pressure = 2100 psia.

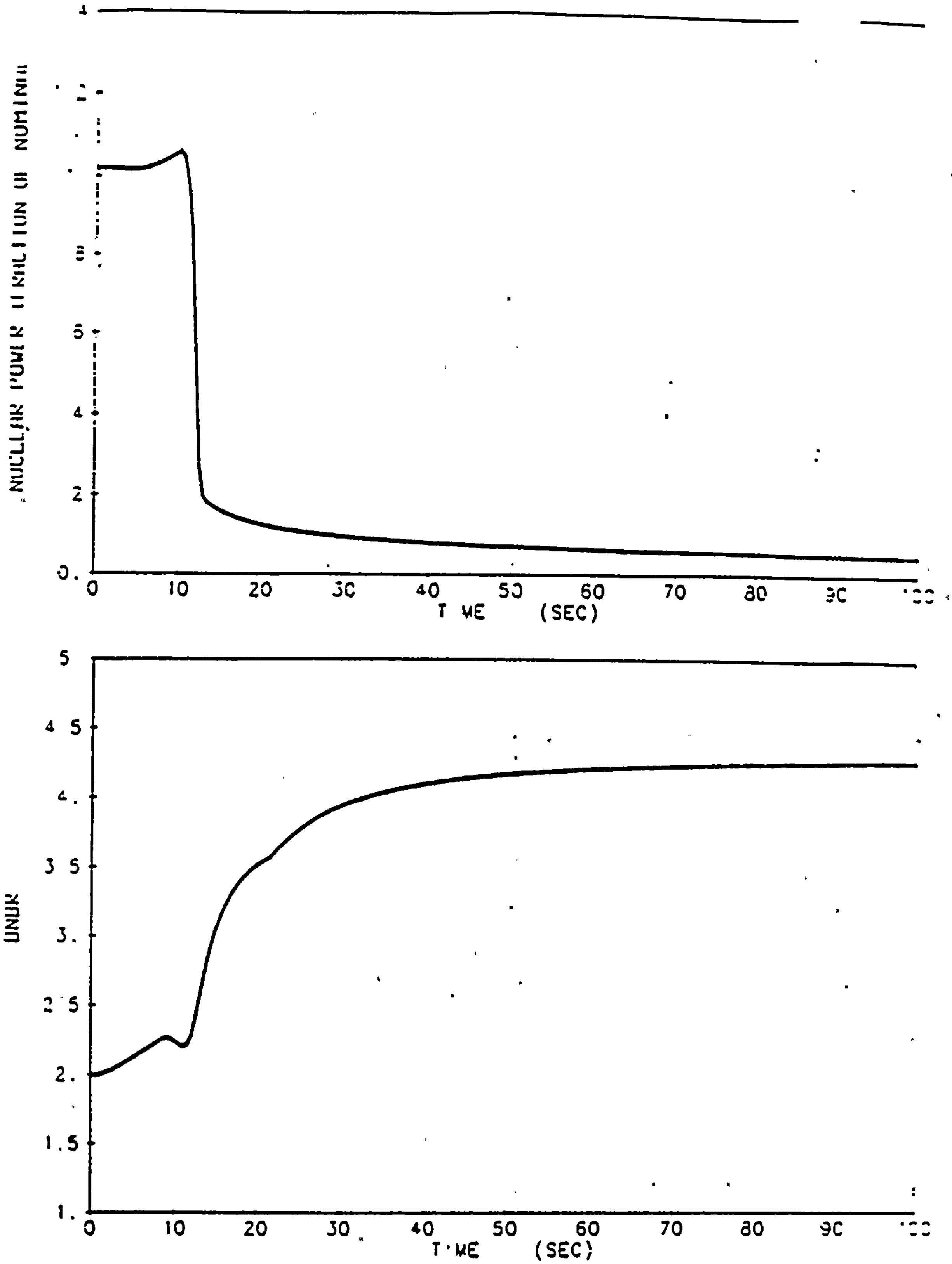
DONALD C. COOK UNIT 2 (FULL V5 CORE)
FIGURE 1c ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER DELTA T PROTECTION



DONALD C. COOK UNIT 1

FIGURE 2

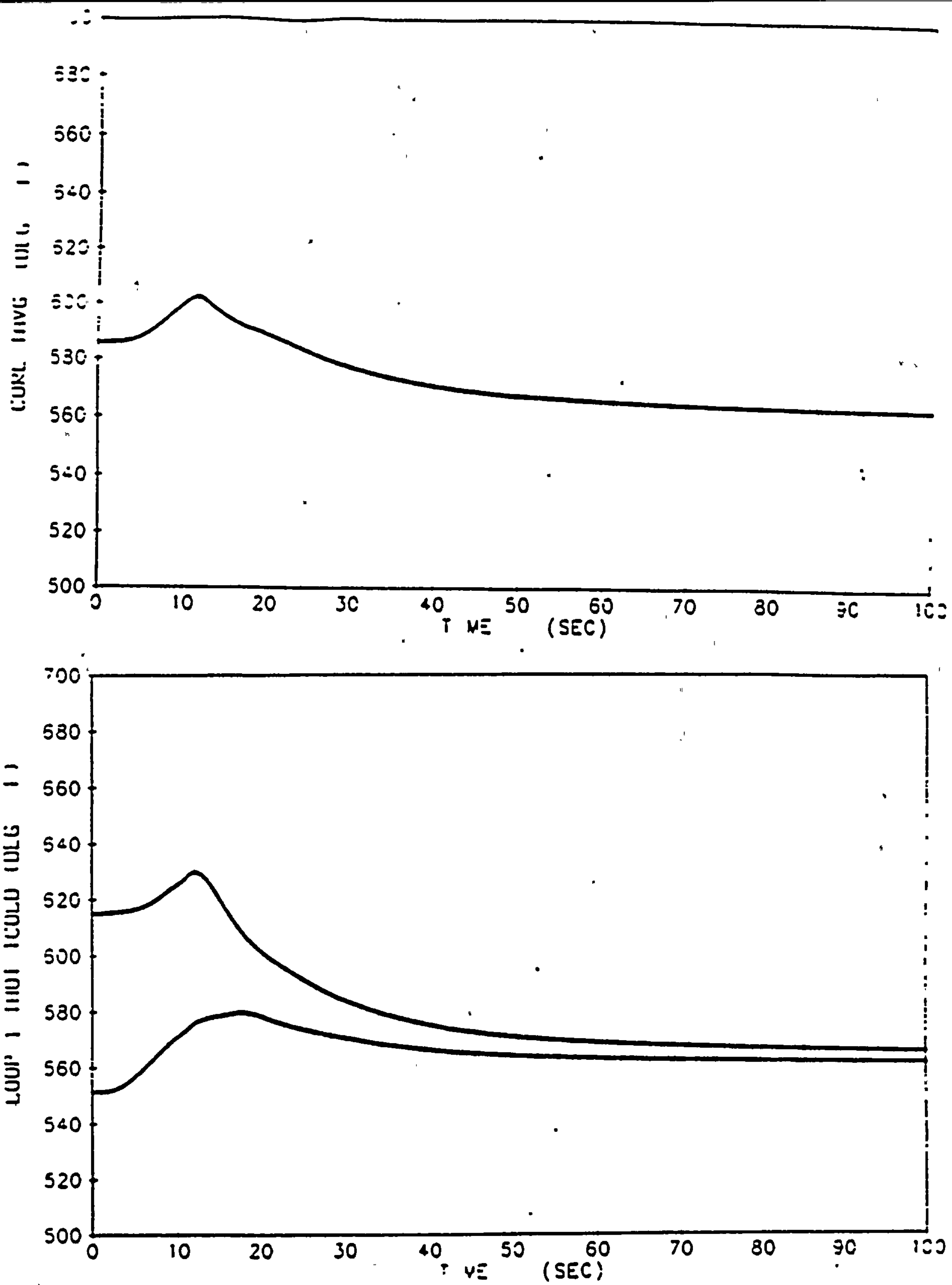
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 3

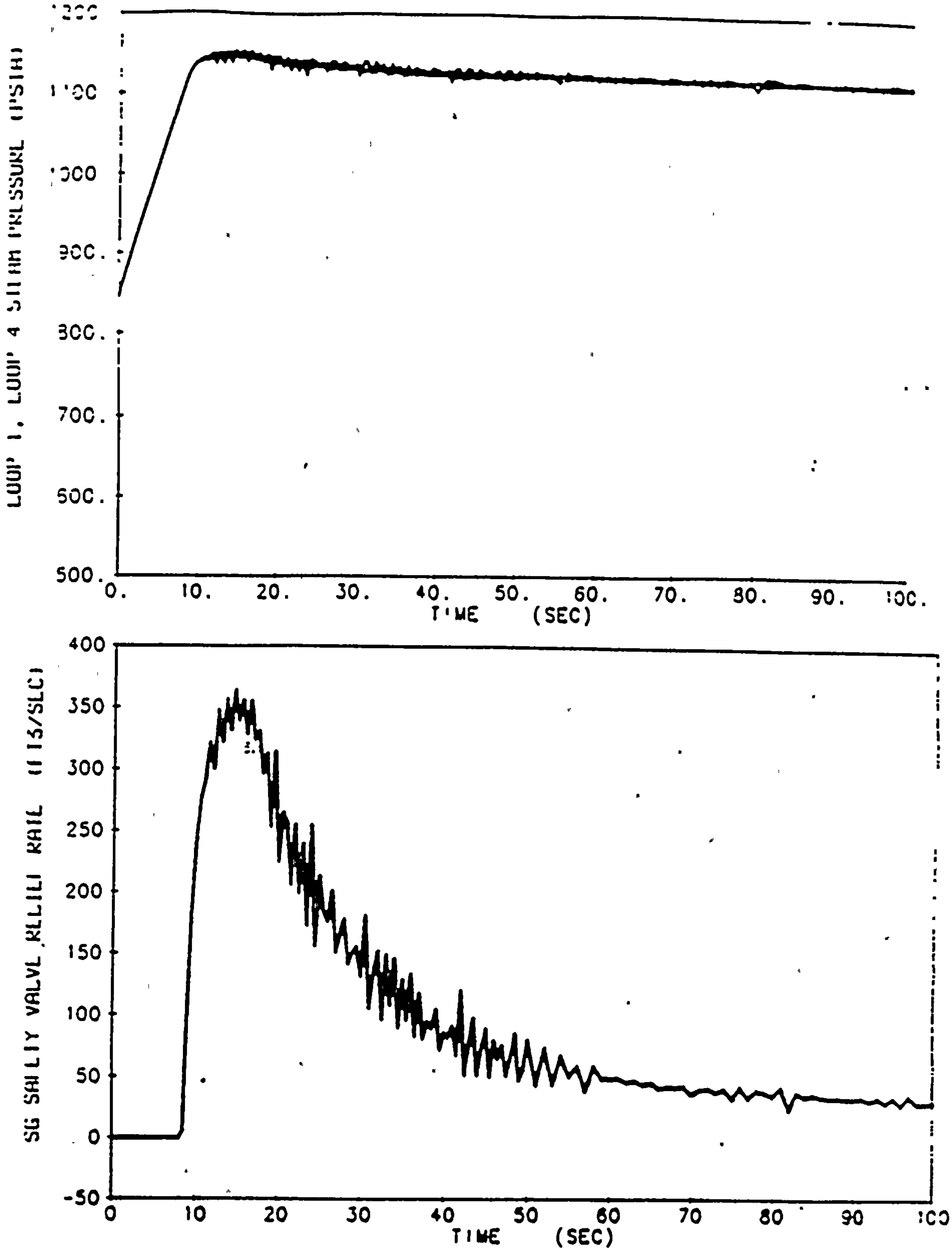
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 4

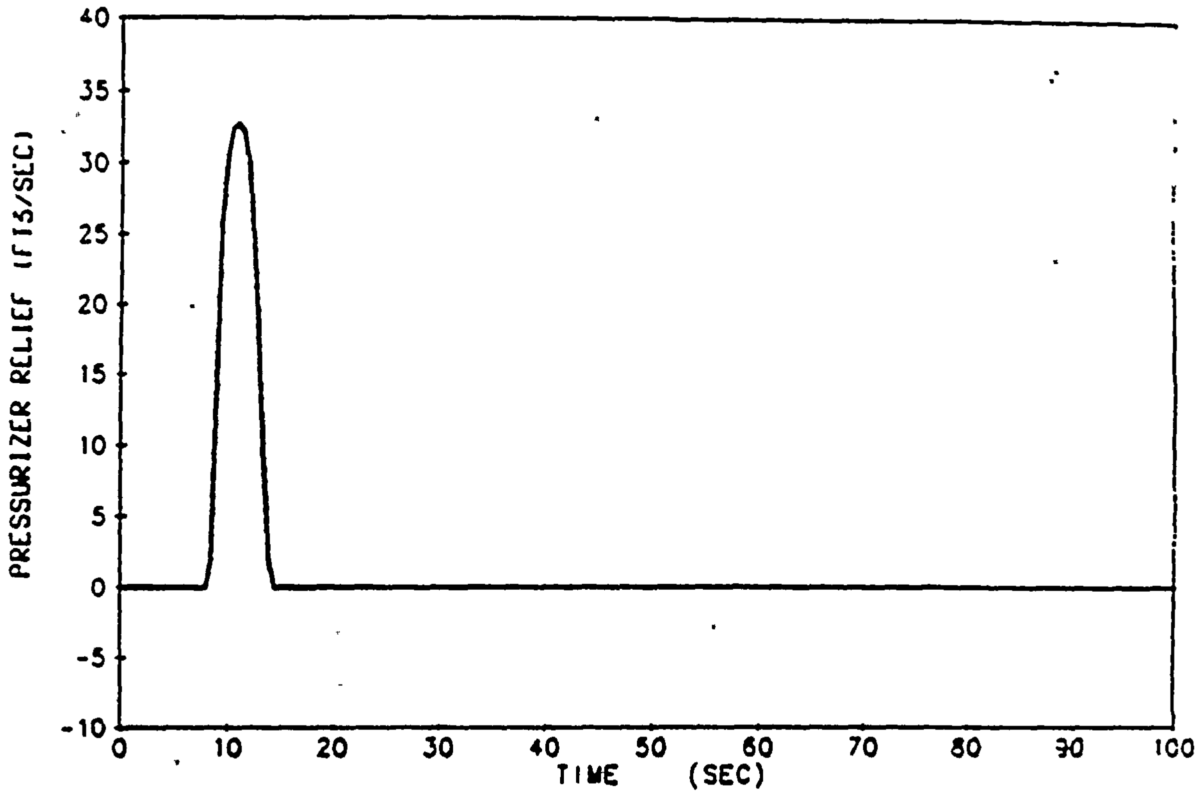
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 5

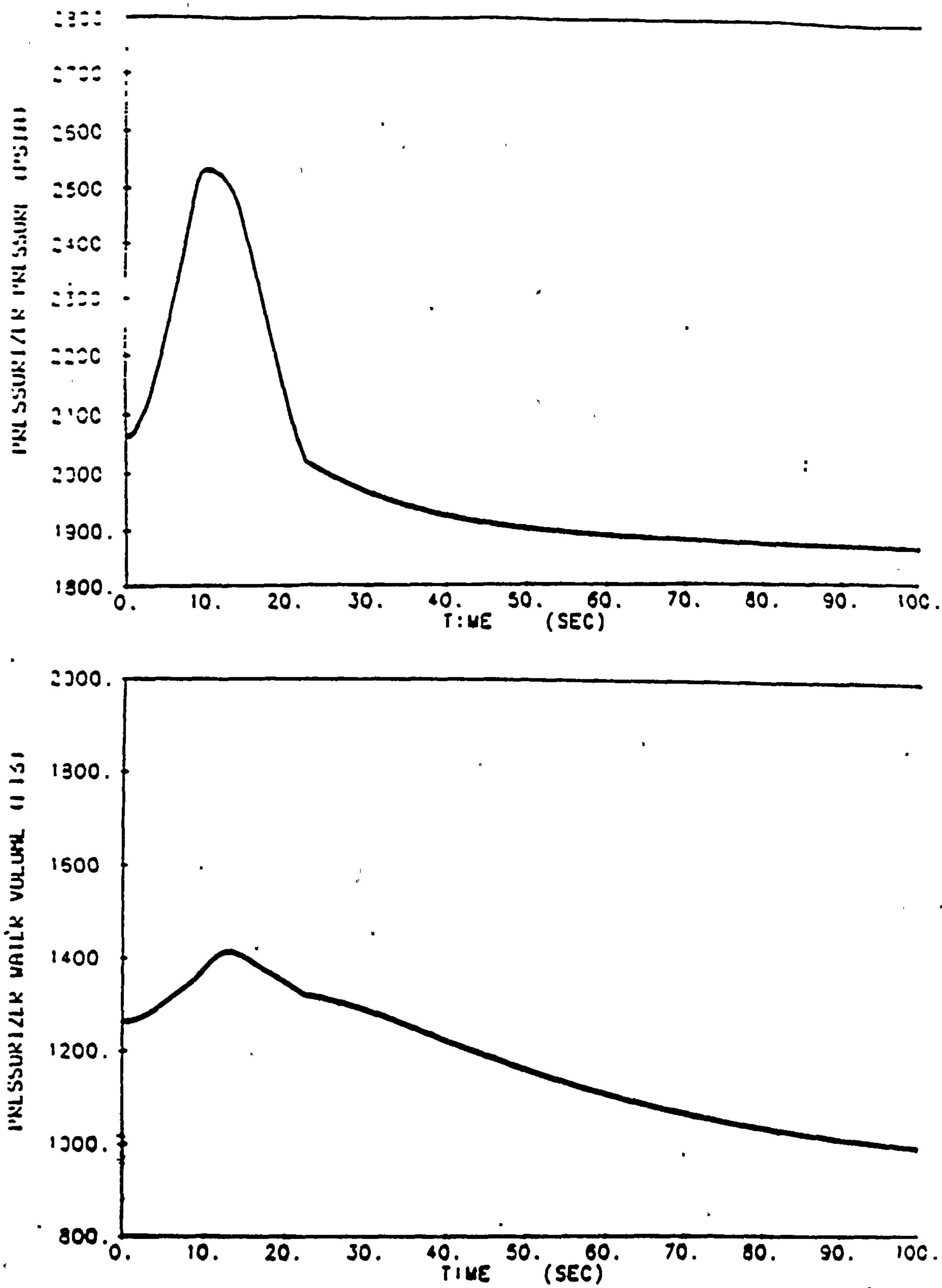
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MINIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 6

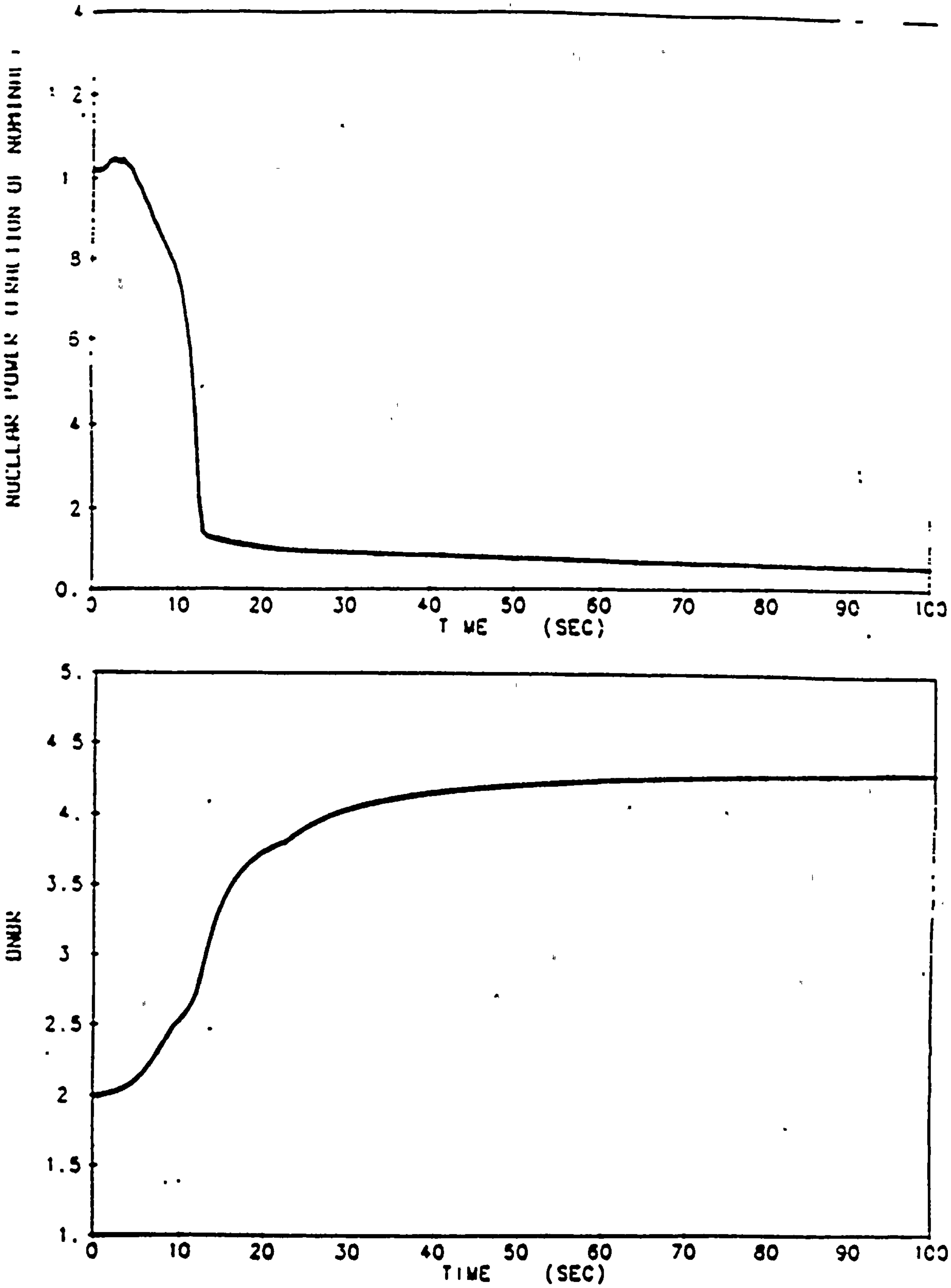
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MINIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 7

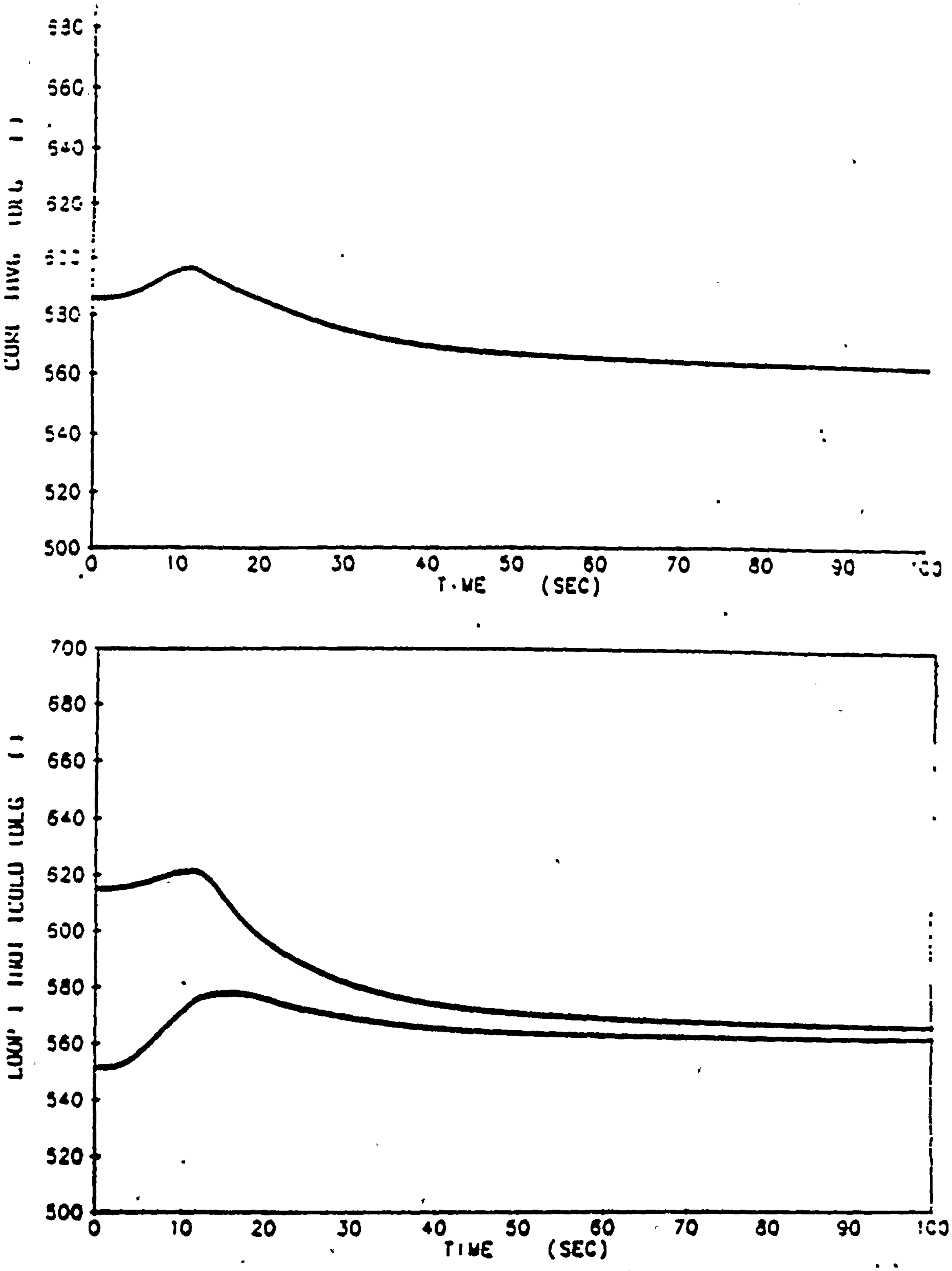
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 8

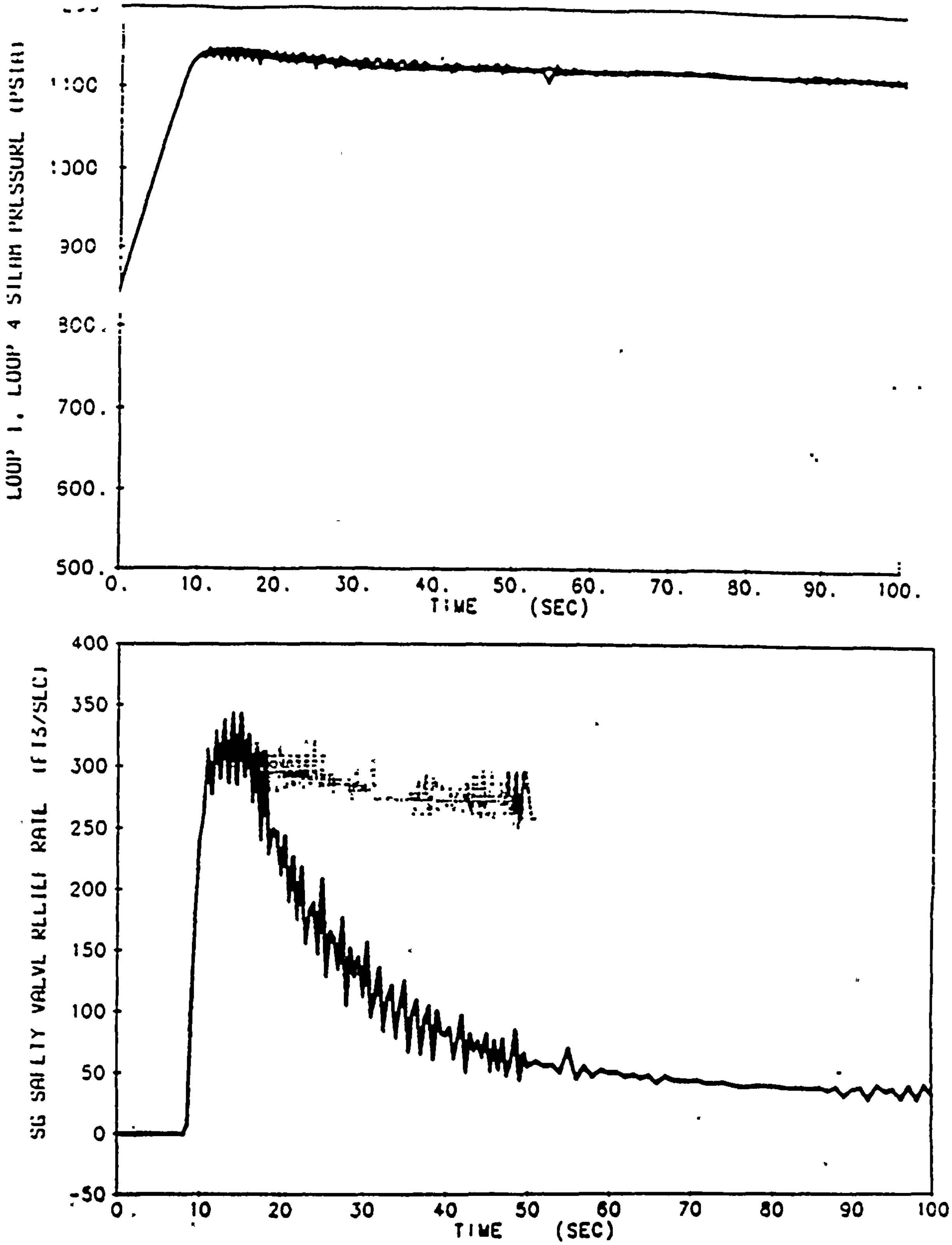
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 9

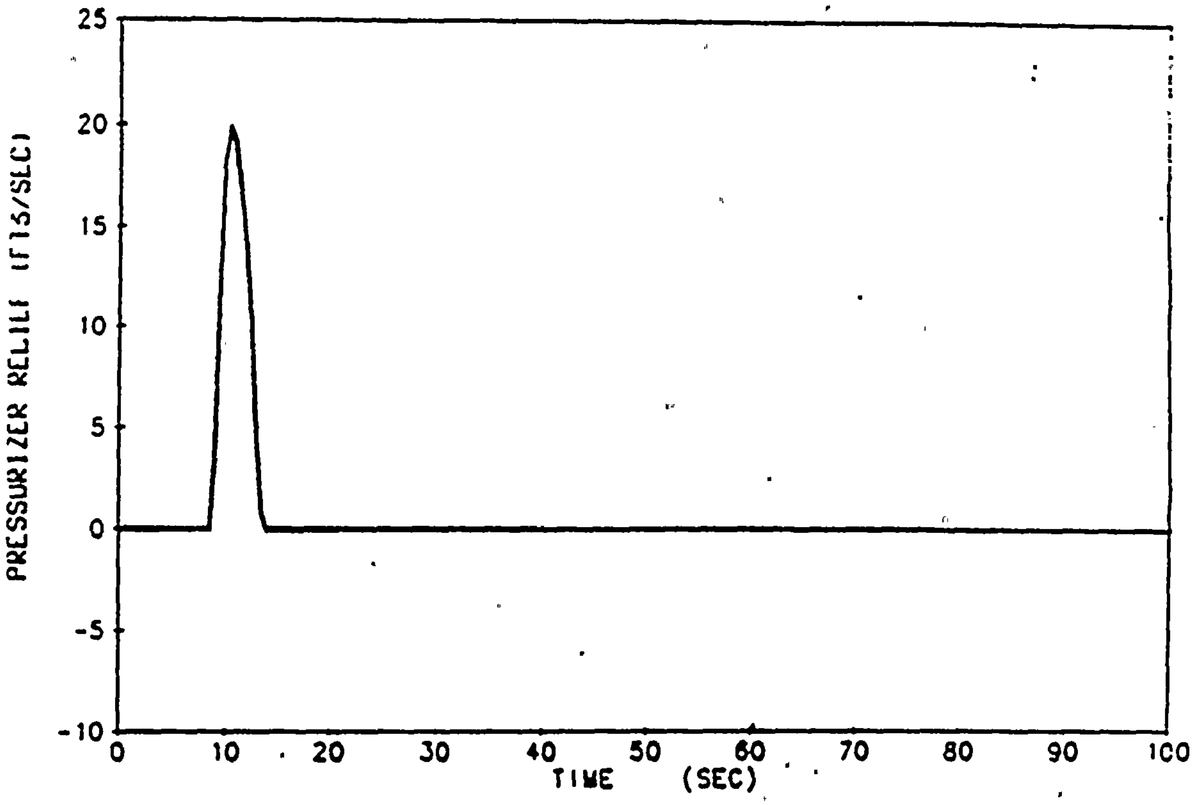
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 10

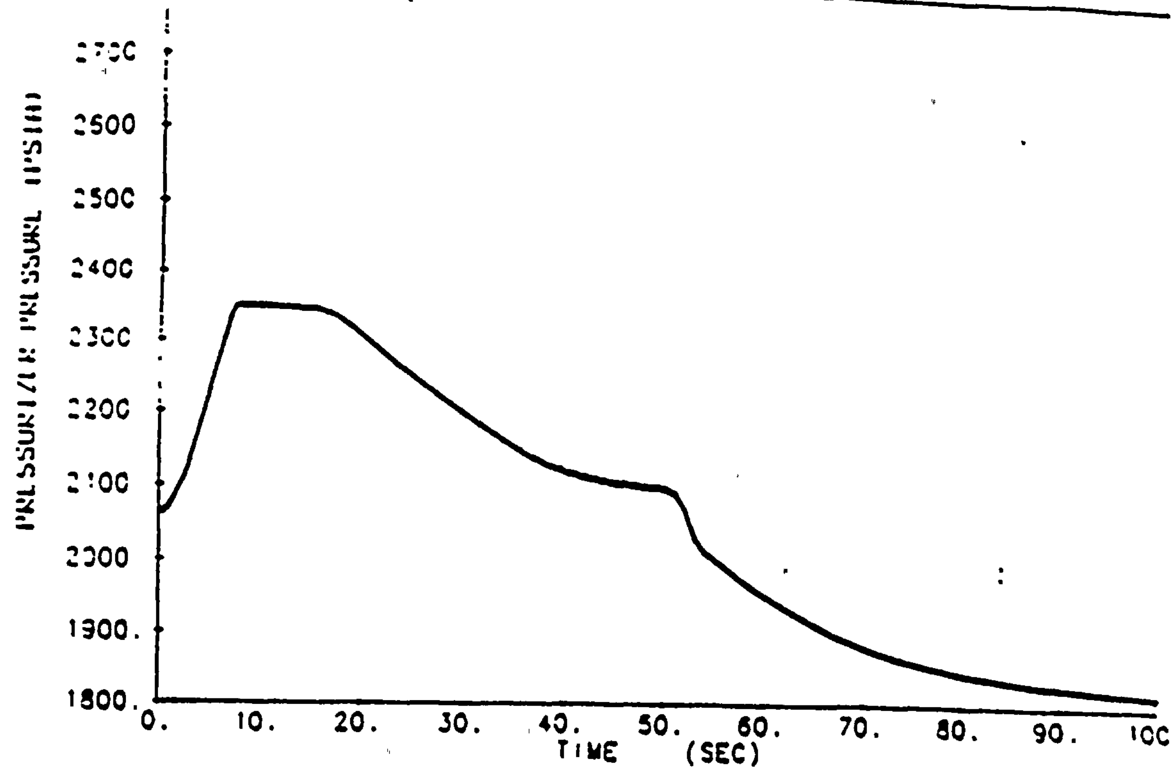
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK

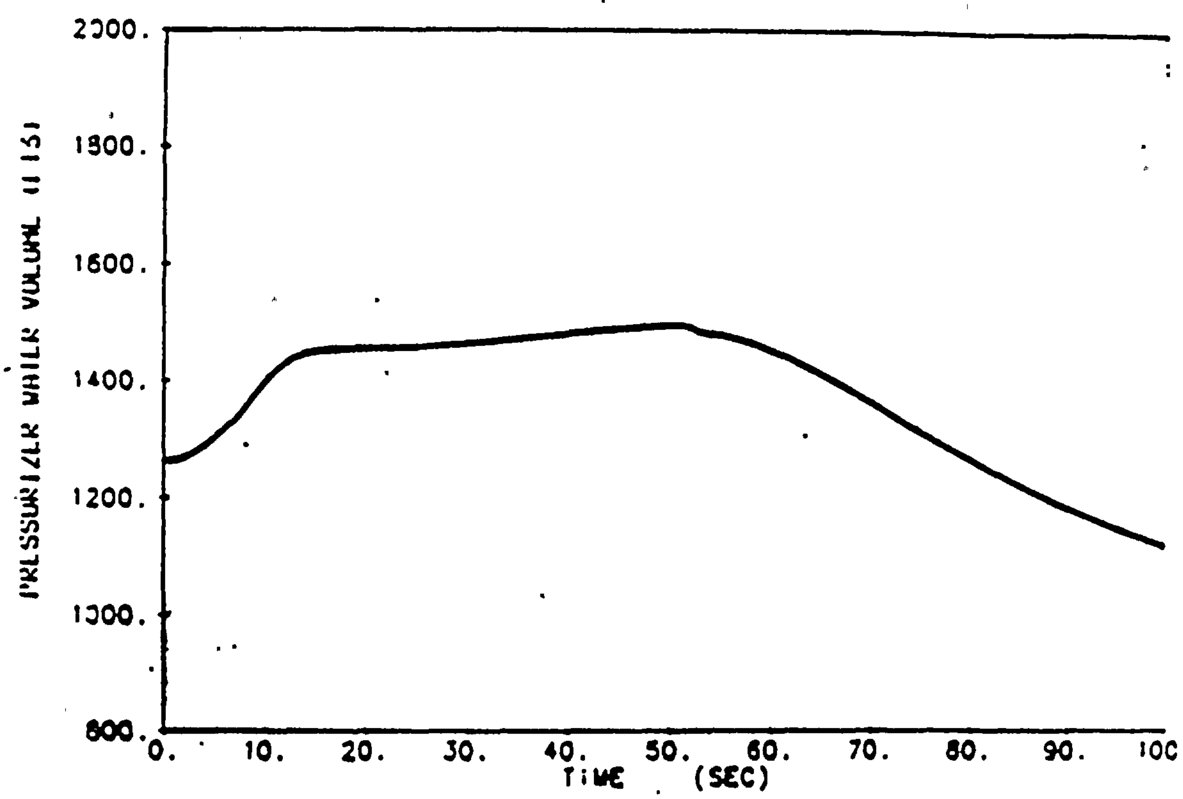


DONALD C. COOK UNIT 1

FIGURE 11

TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK





DONALD C. COOK UNIT 1

FIGURE 12

TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK

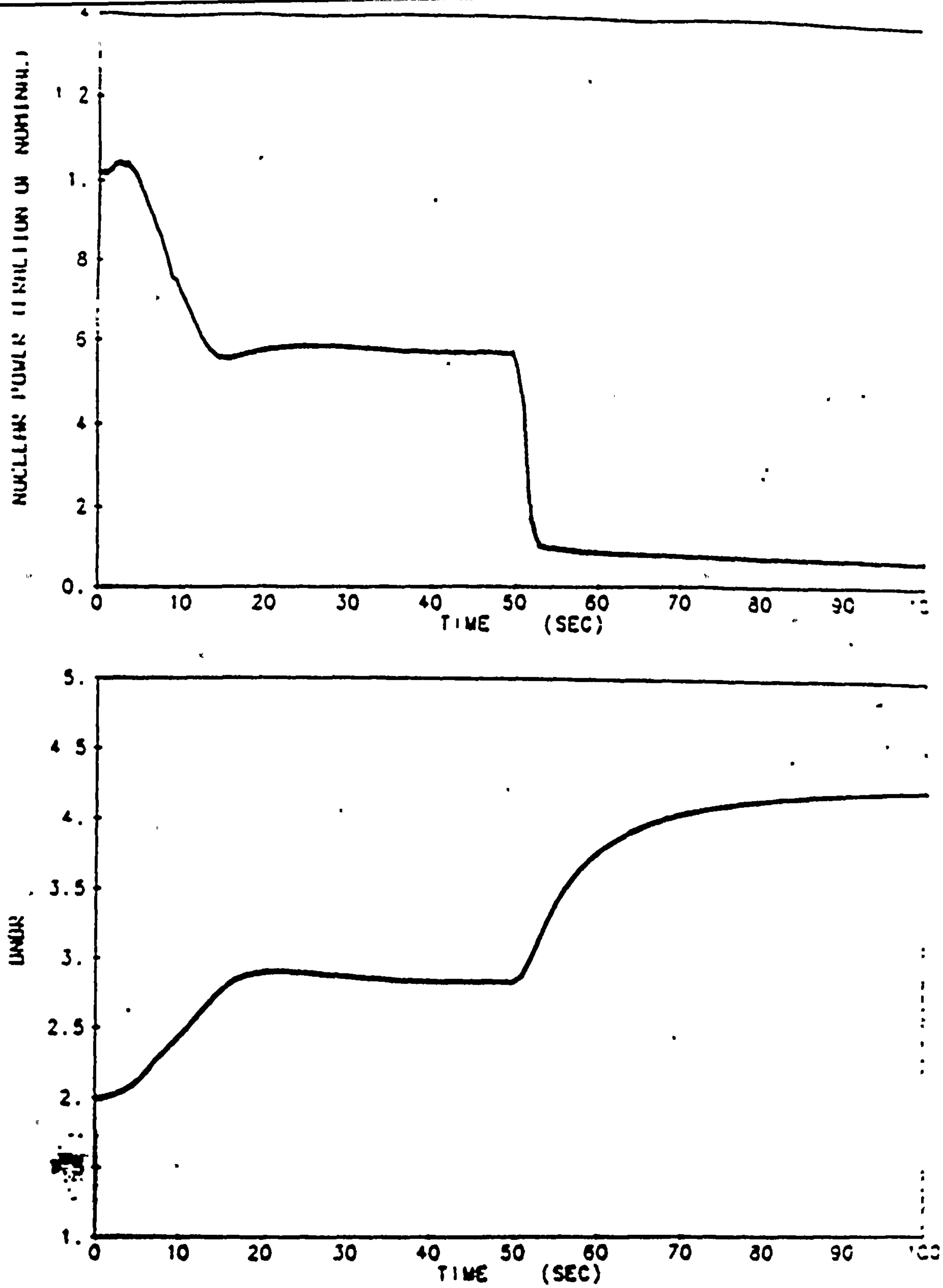


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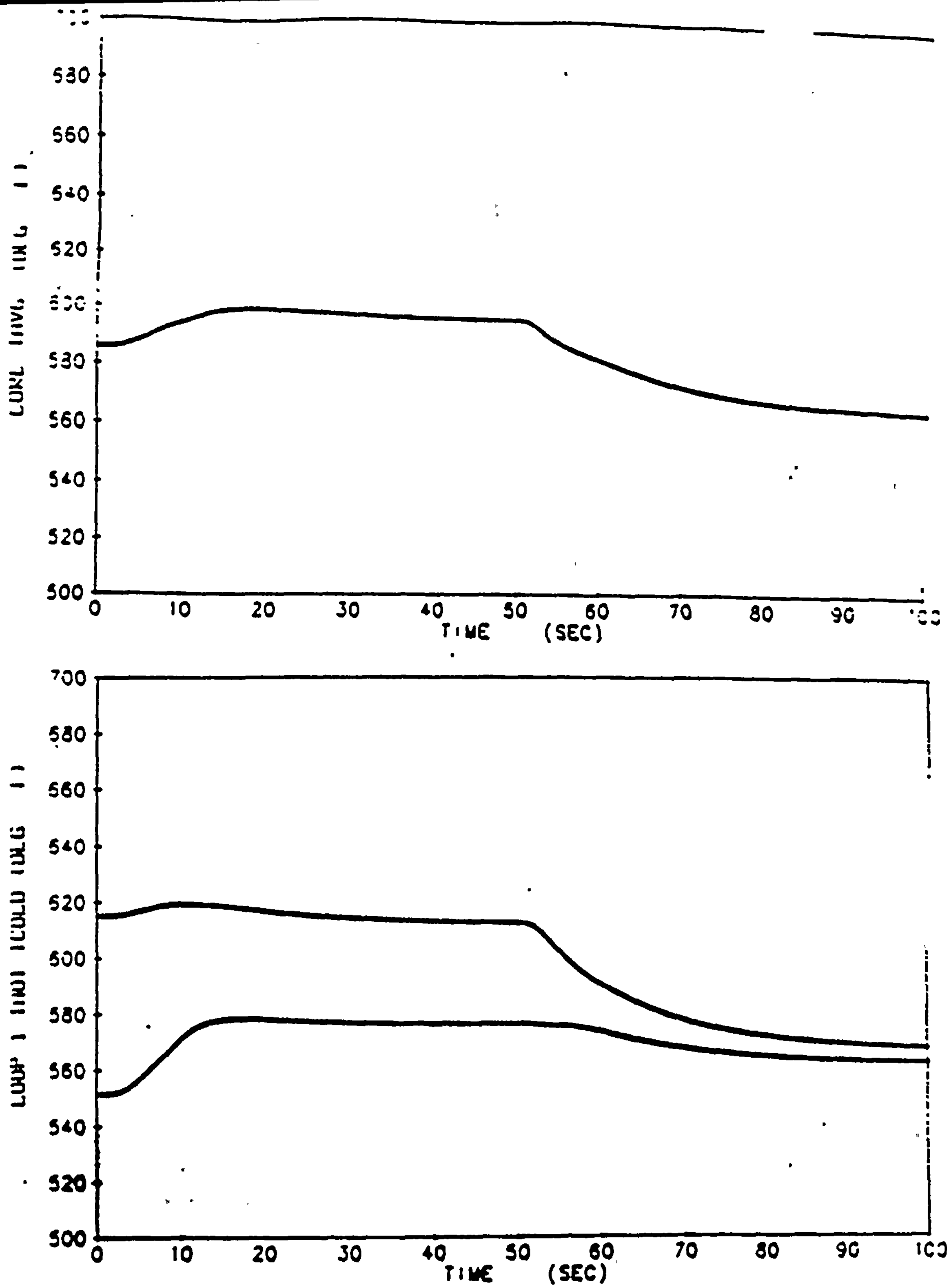
FIGURE 13

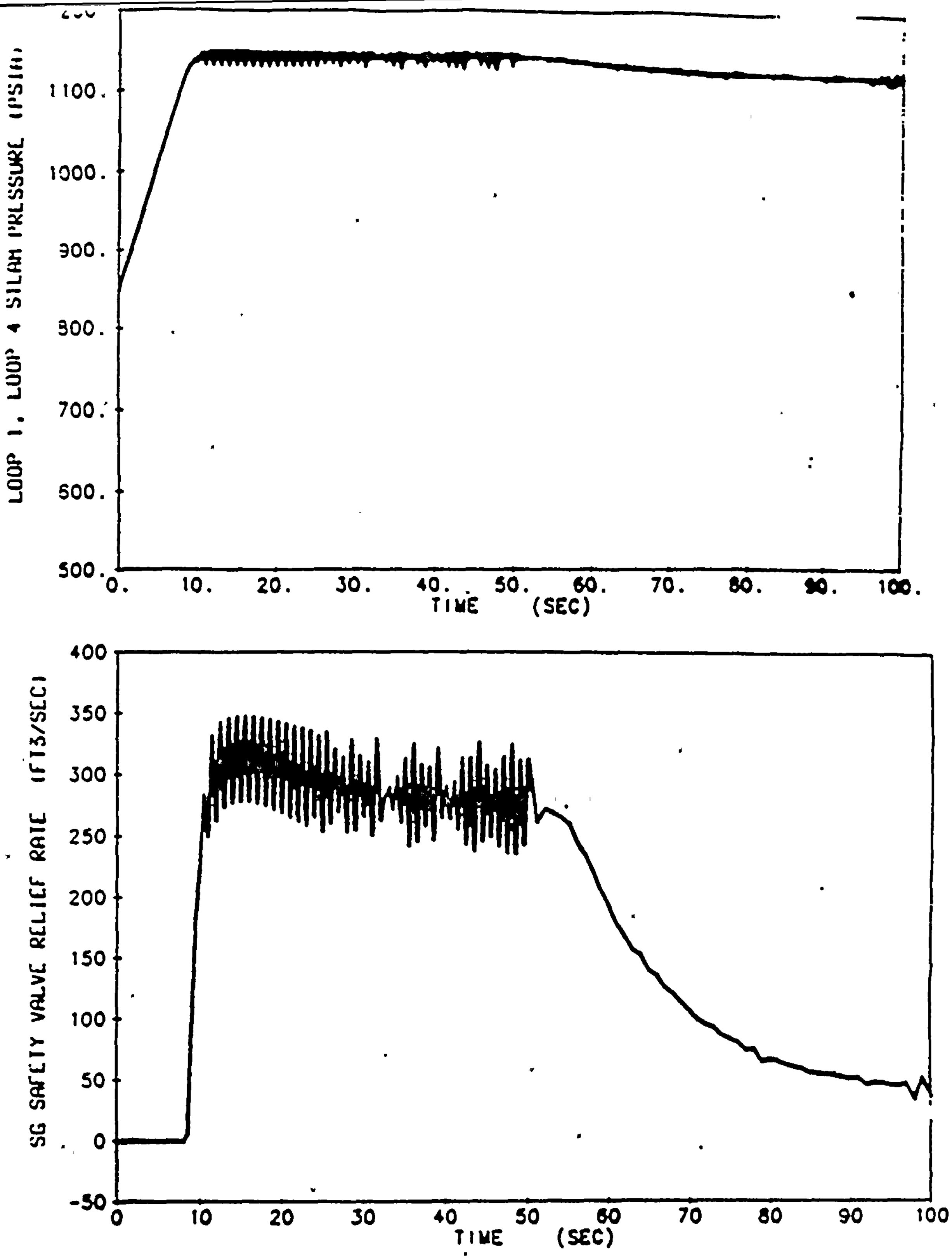
TURBINE TRIP EVENT WITH PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 14

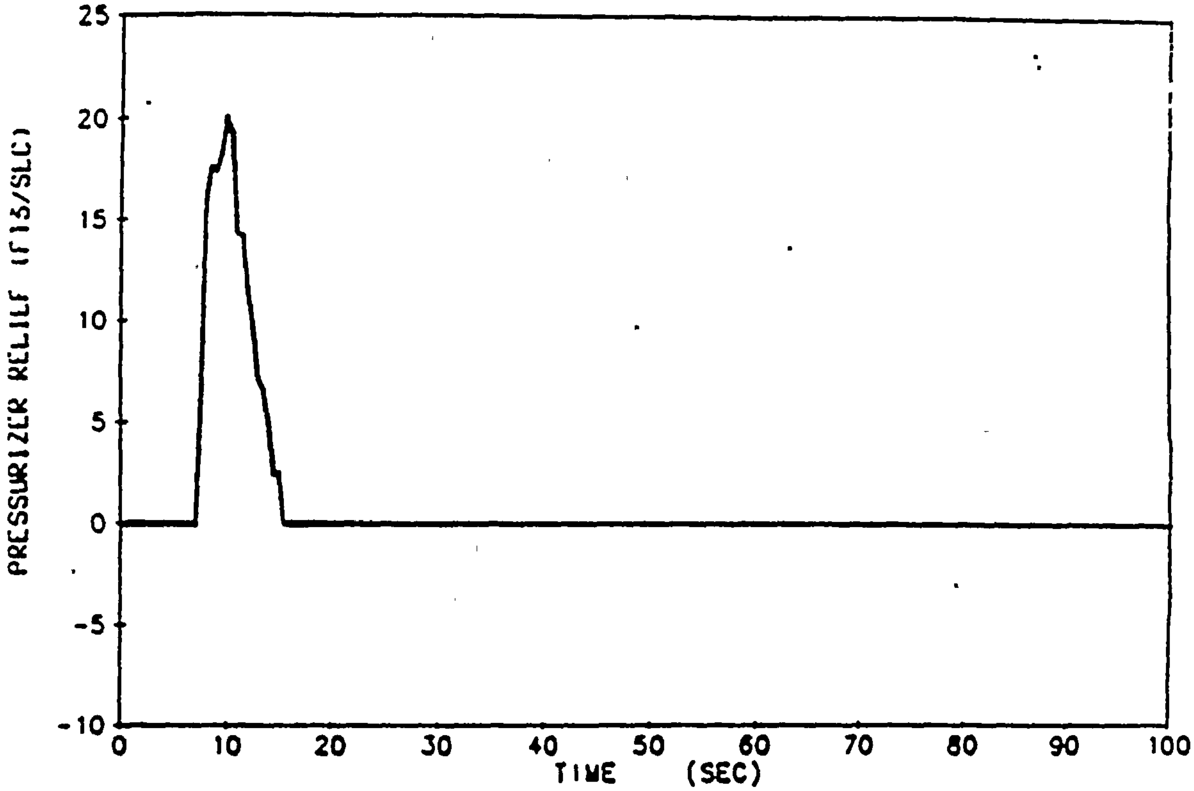
TURBINE TRIP EVENT WITH PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 15

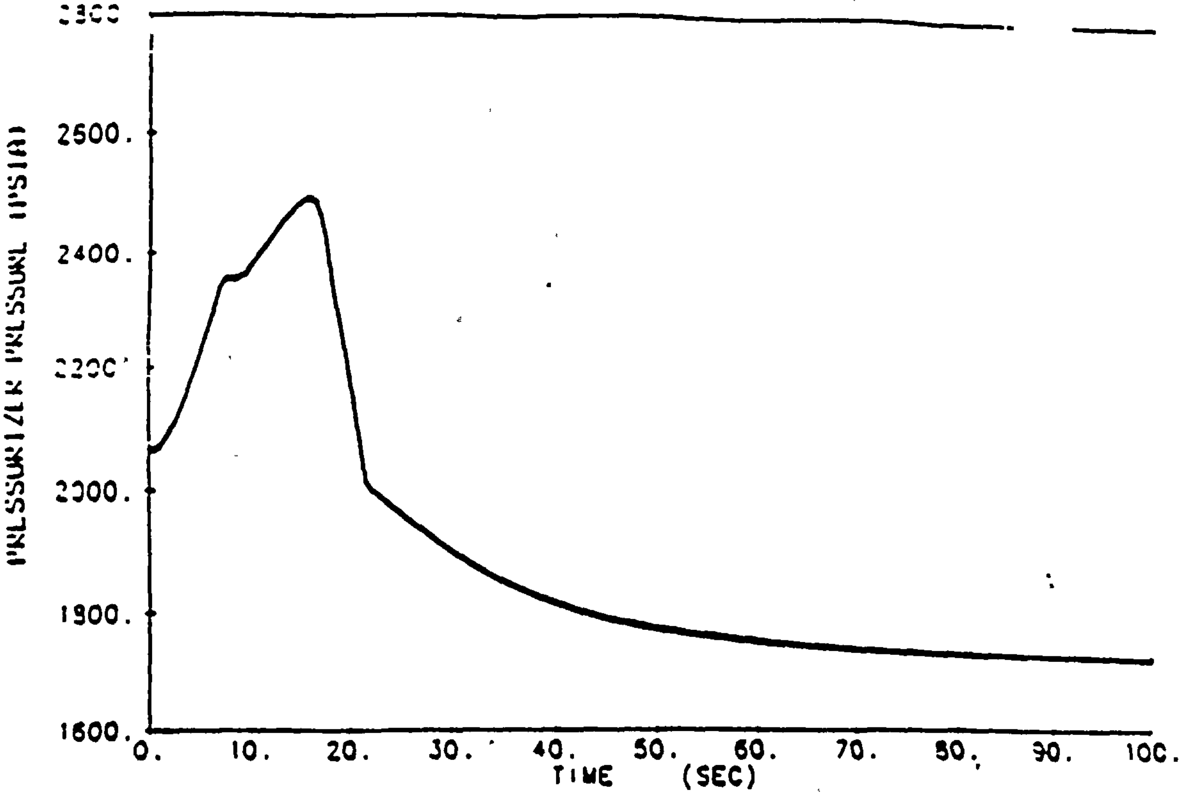
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK

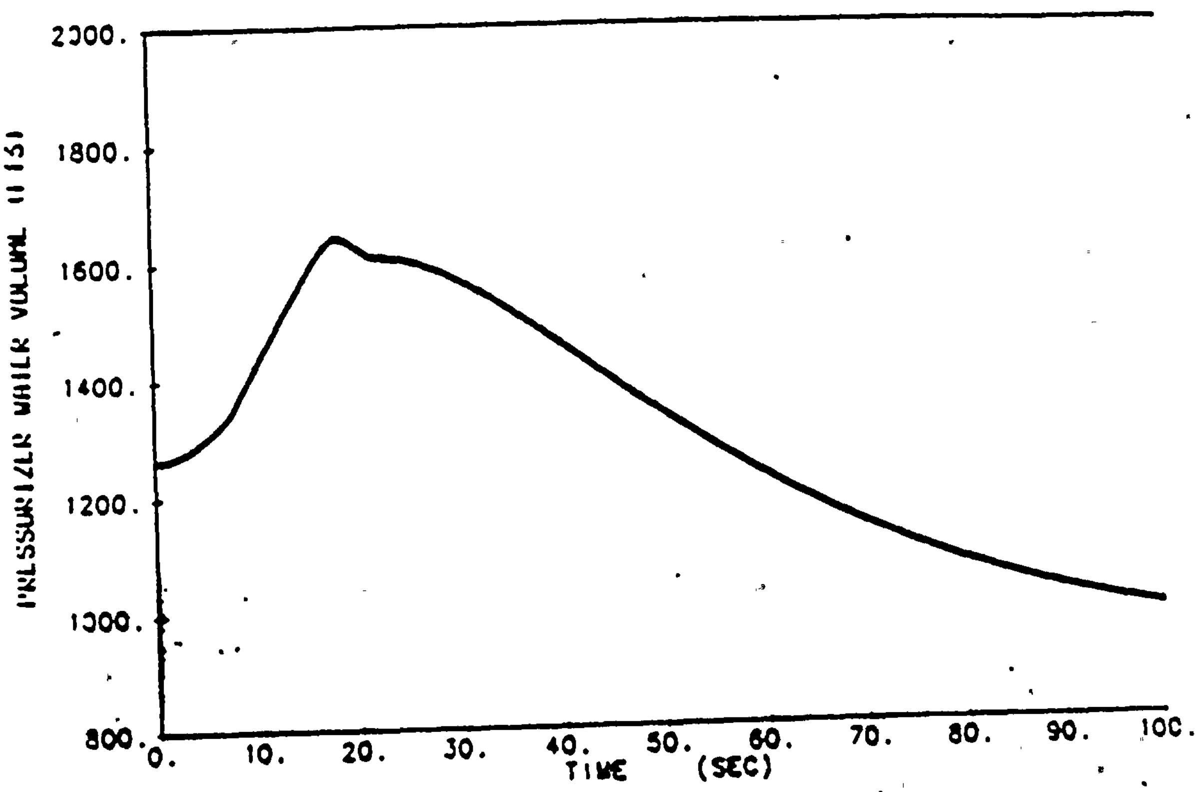


DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 16

TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK

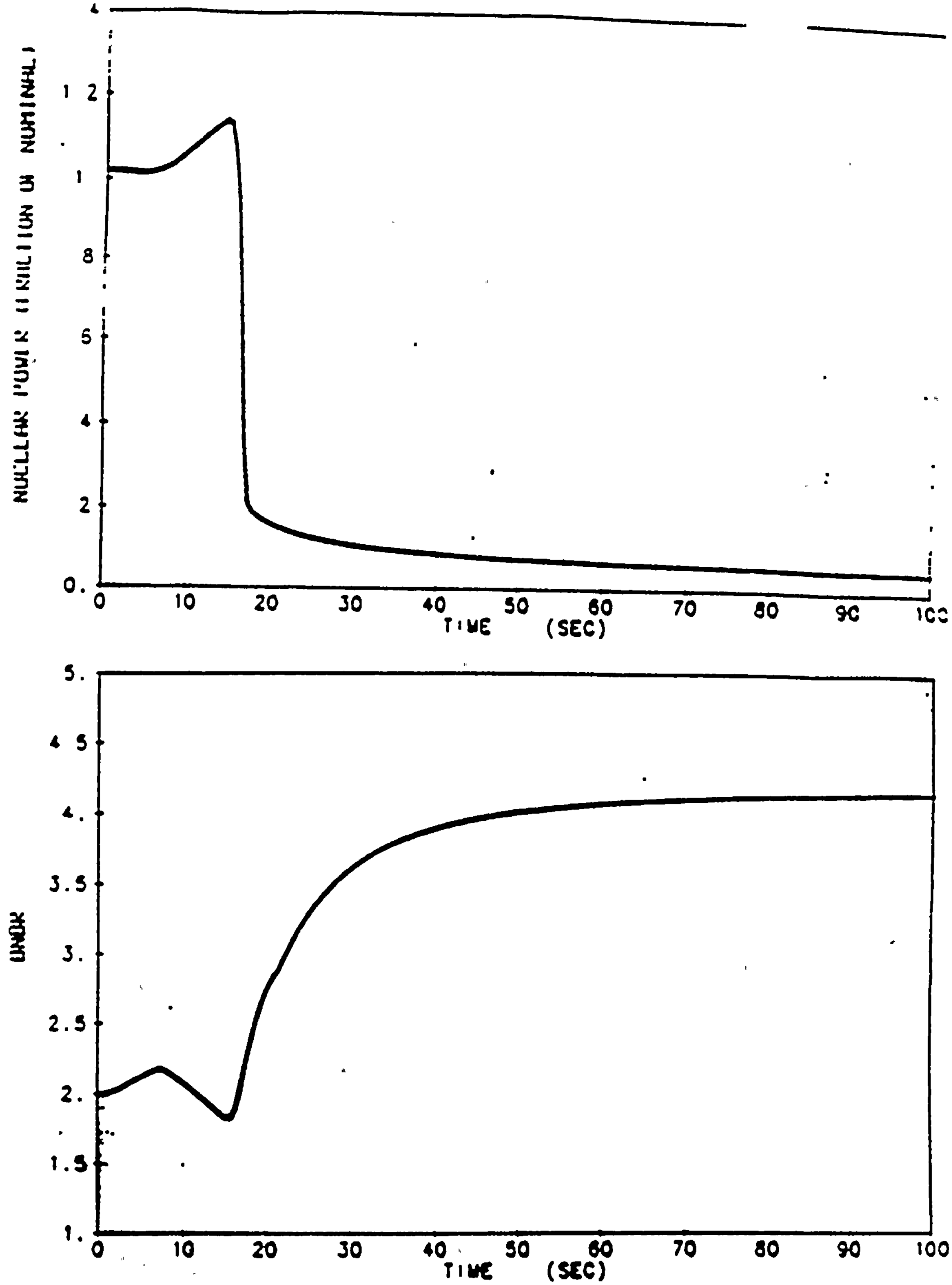




DONALD C. COOK UNIT I

FIGURE 17

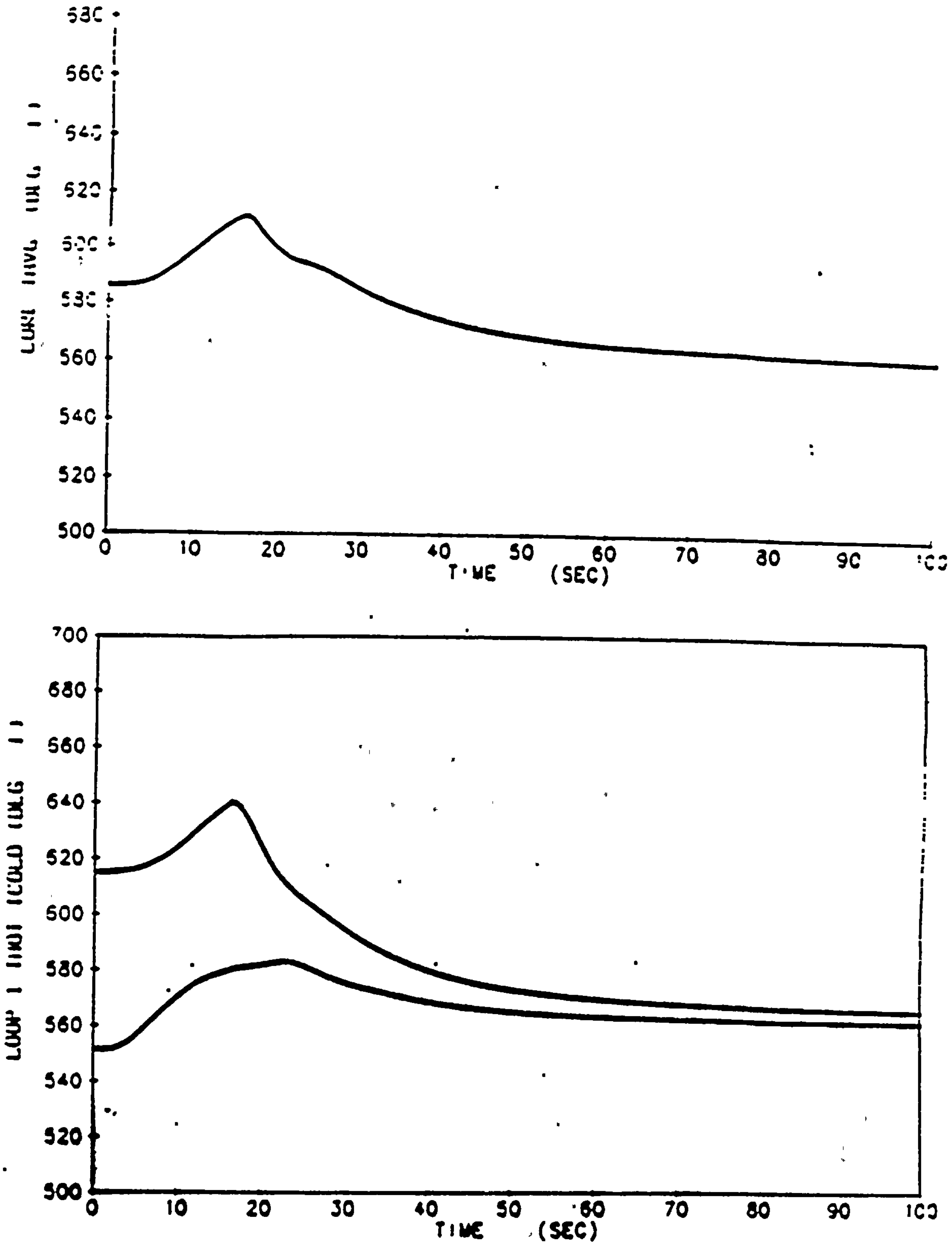
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 18

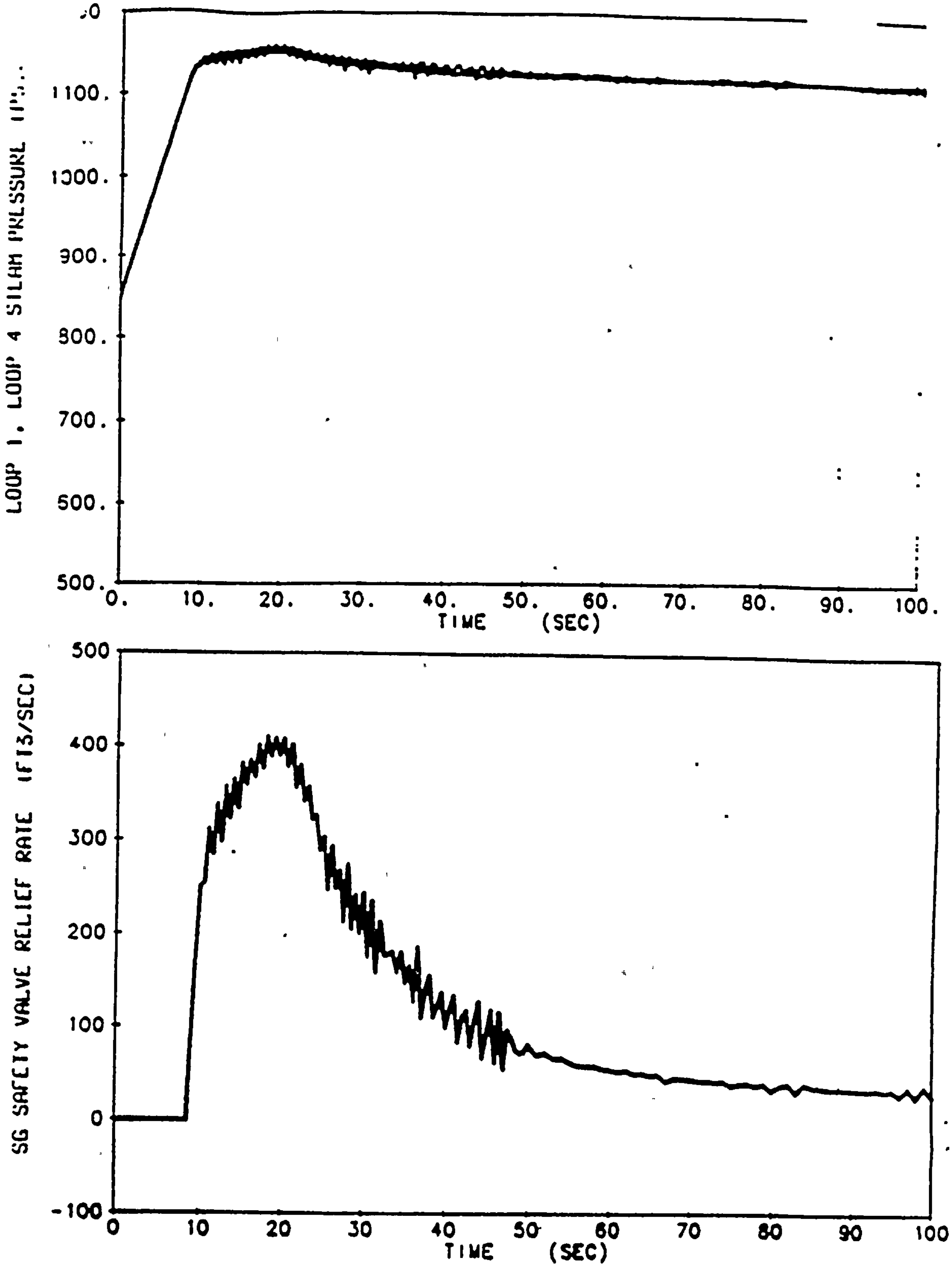
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 19

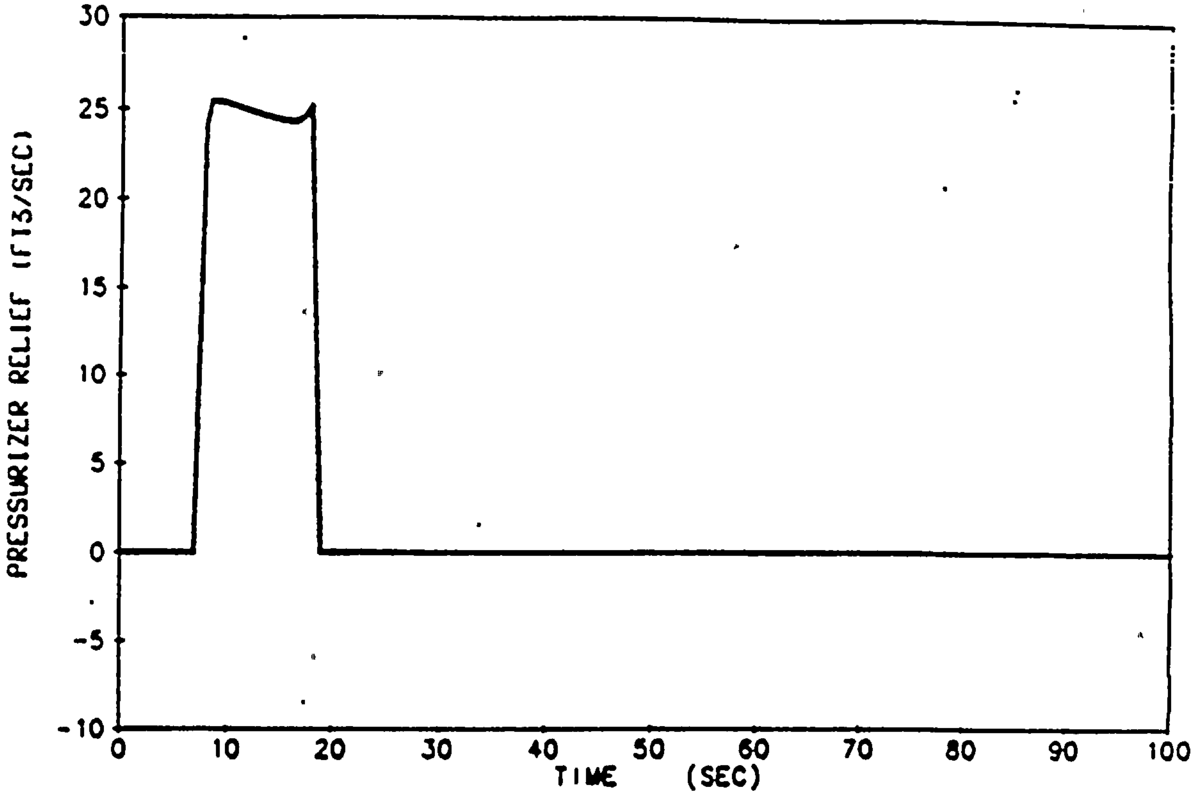
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 20

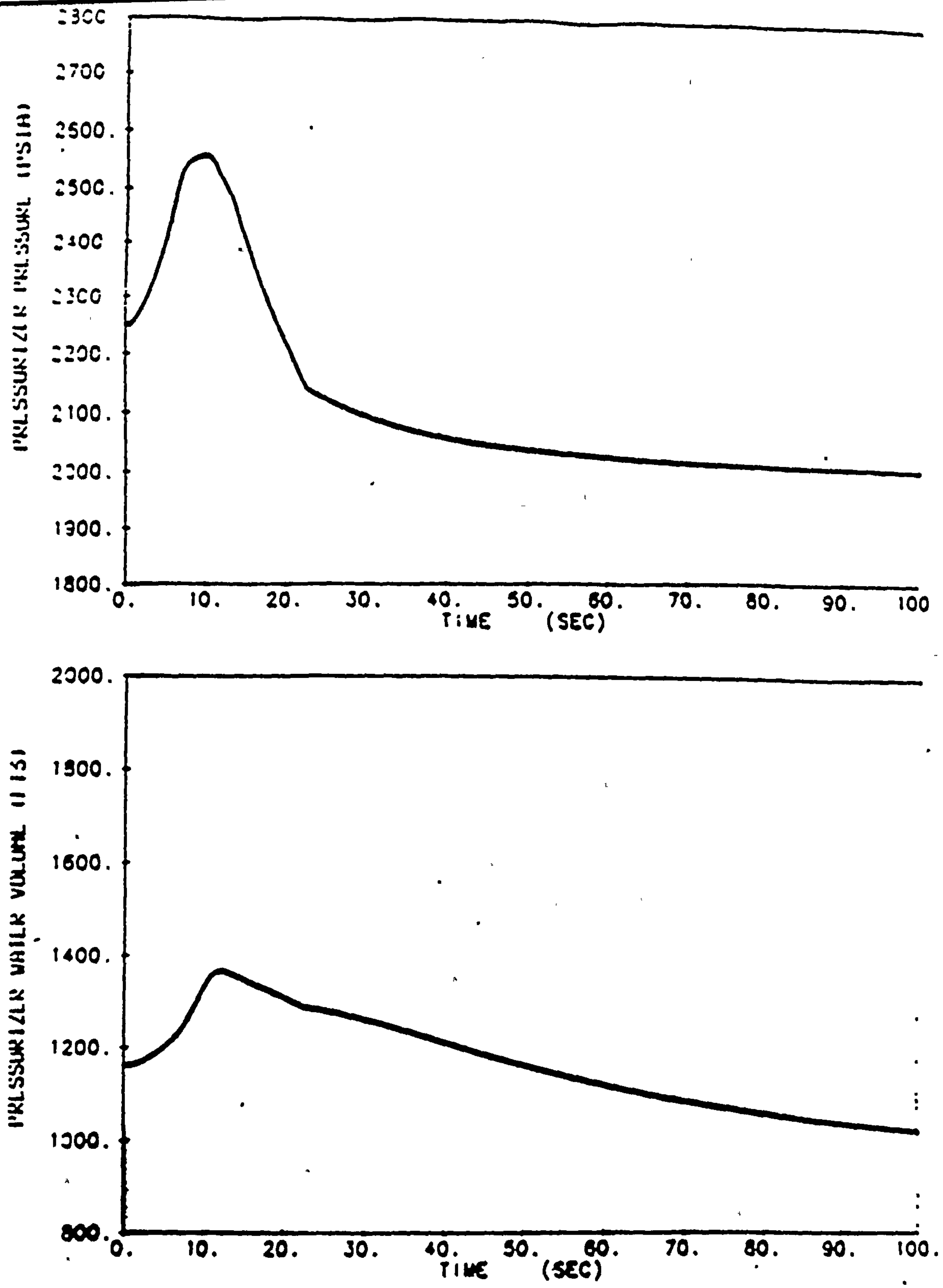
TURBINE TRIP EVENT WITH PRESSURE CONTROL, MINIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 1

FIGURE 21

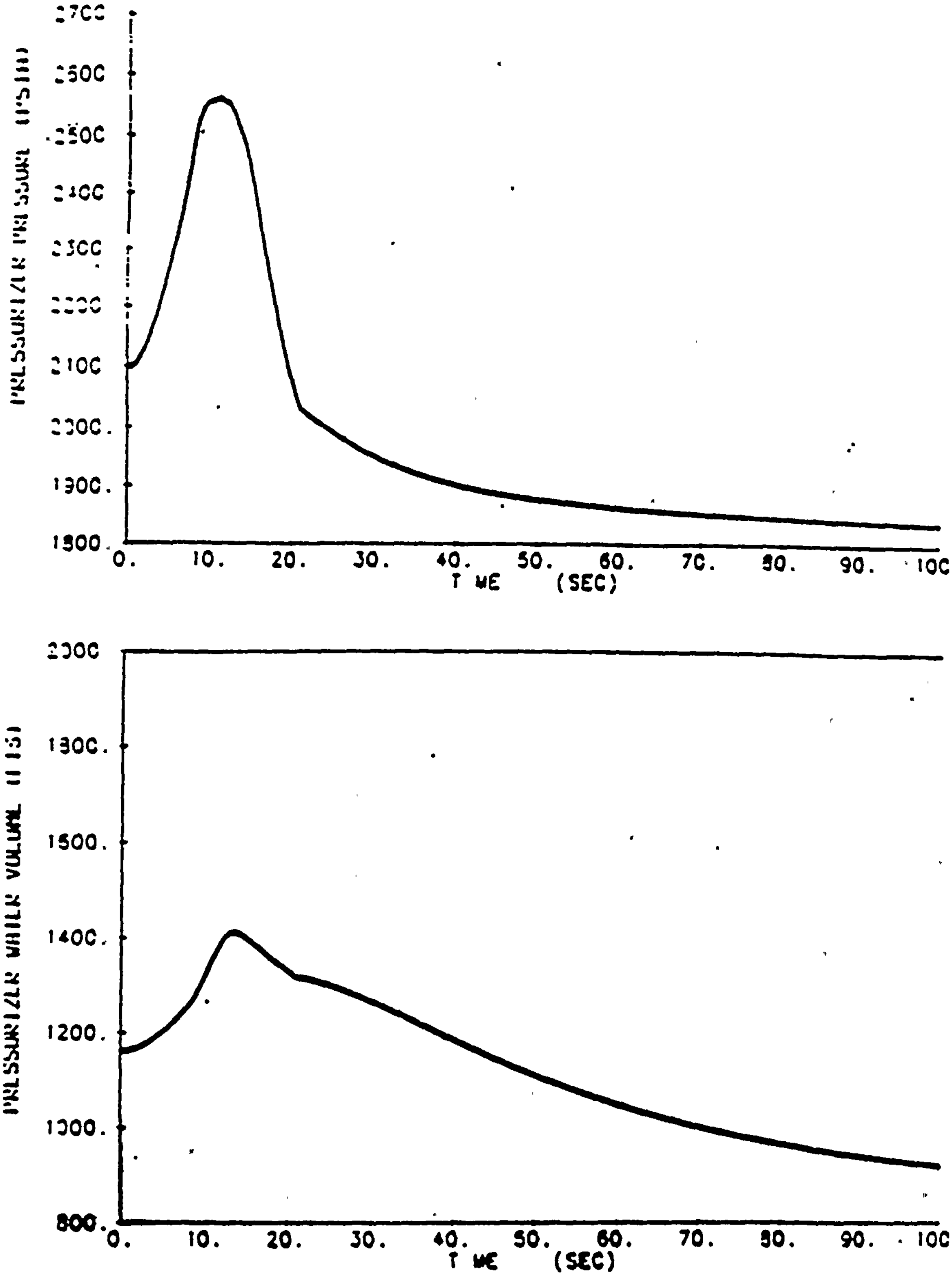
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 22a

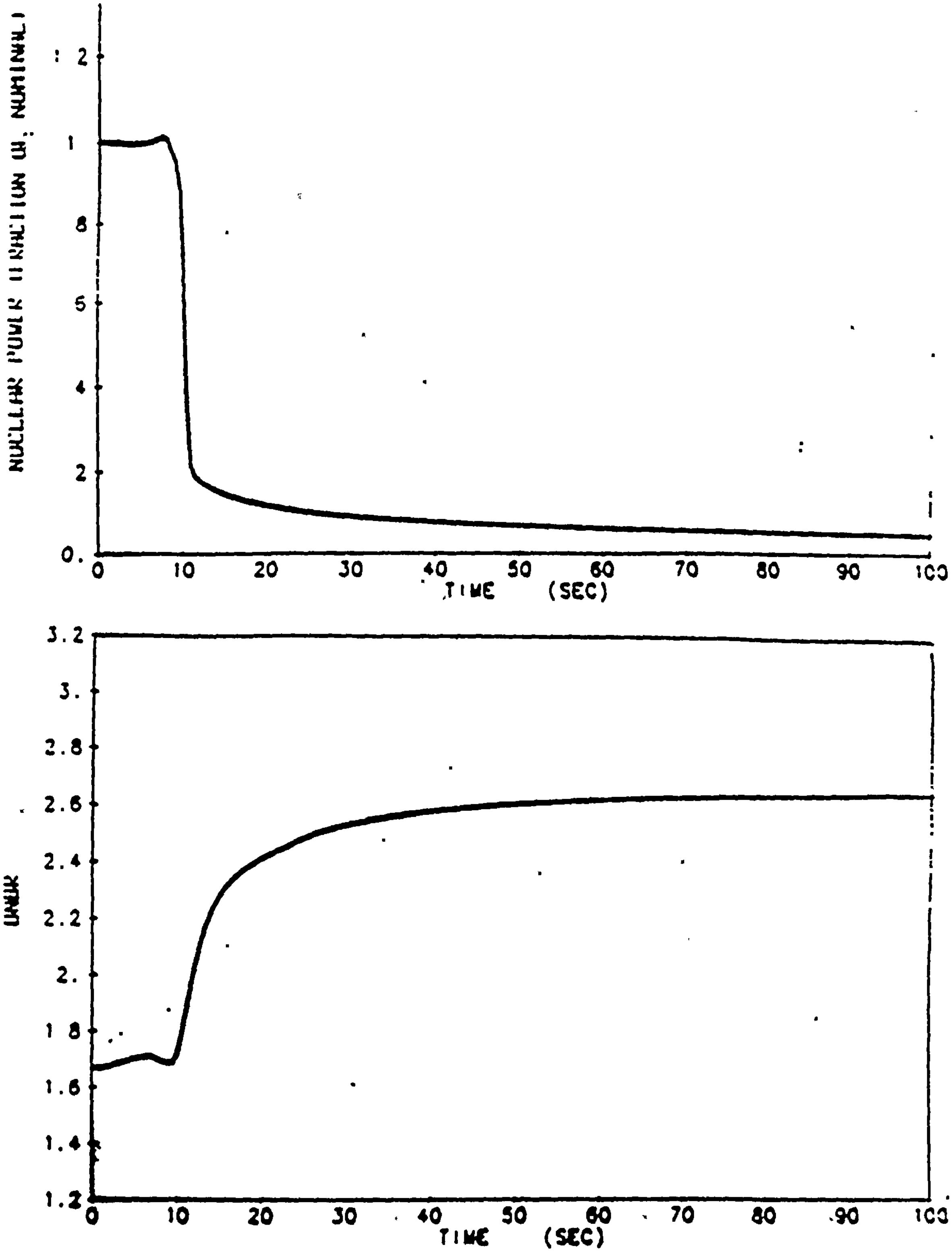
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 22b

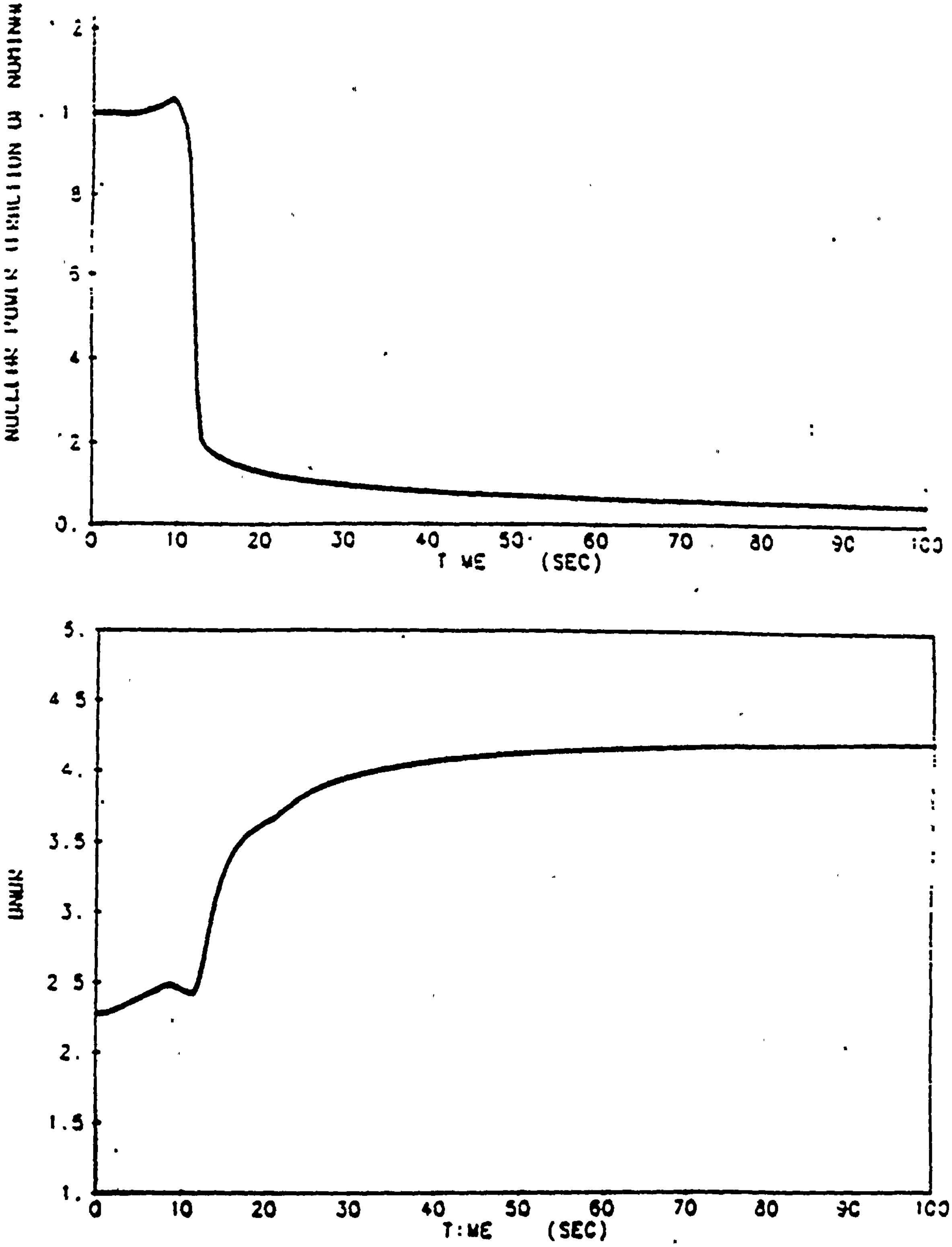
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 23a

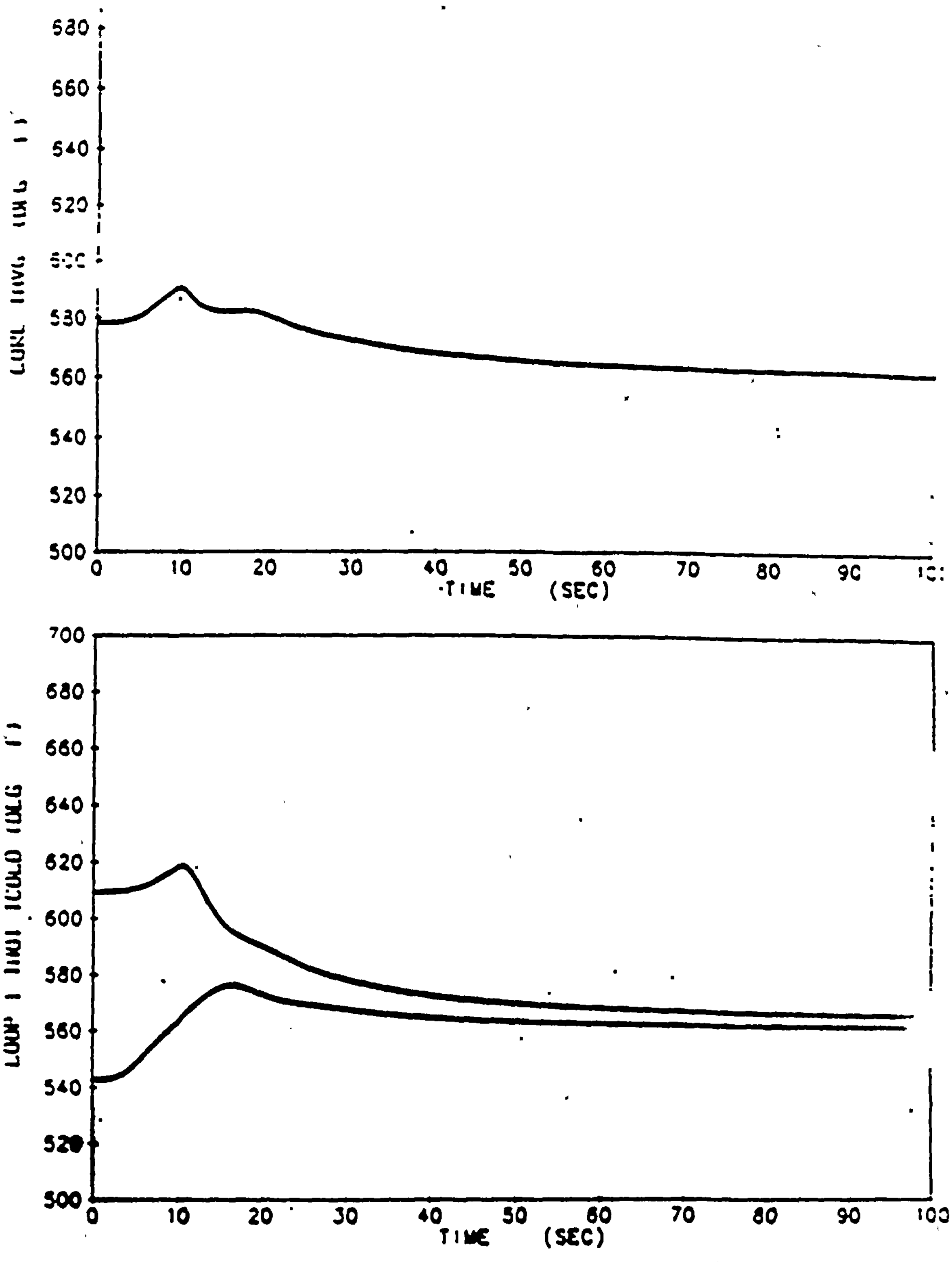
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 23b

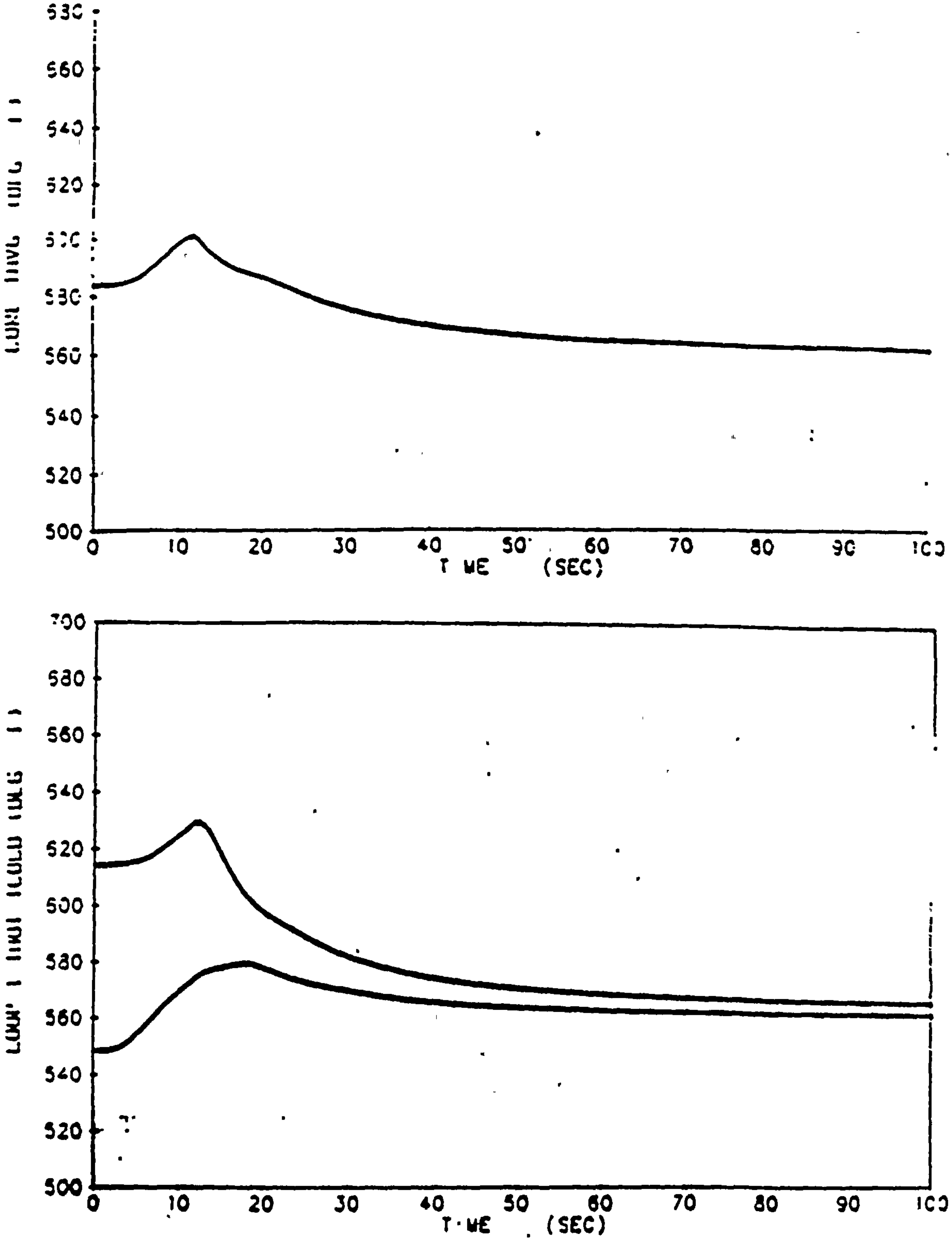
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 24a

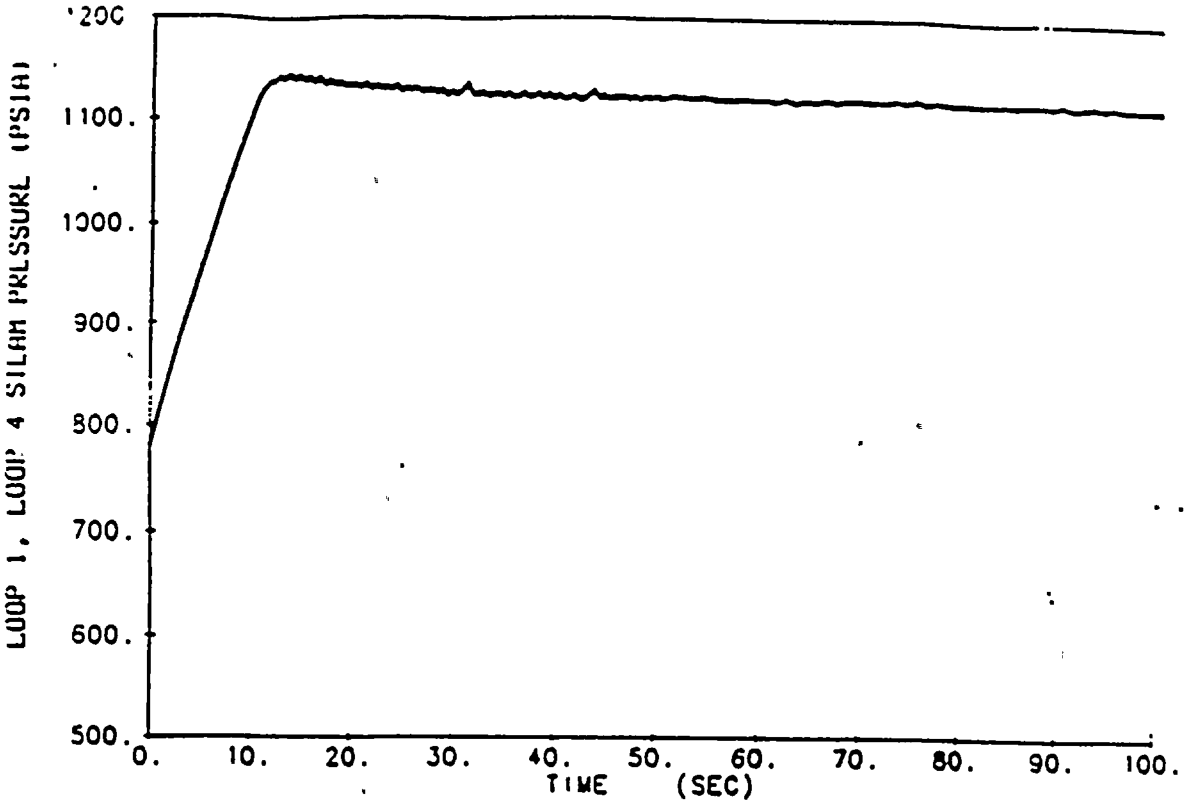
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK

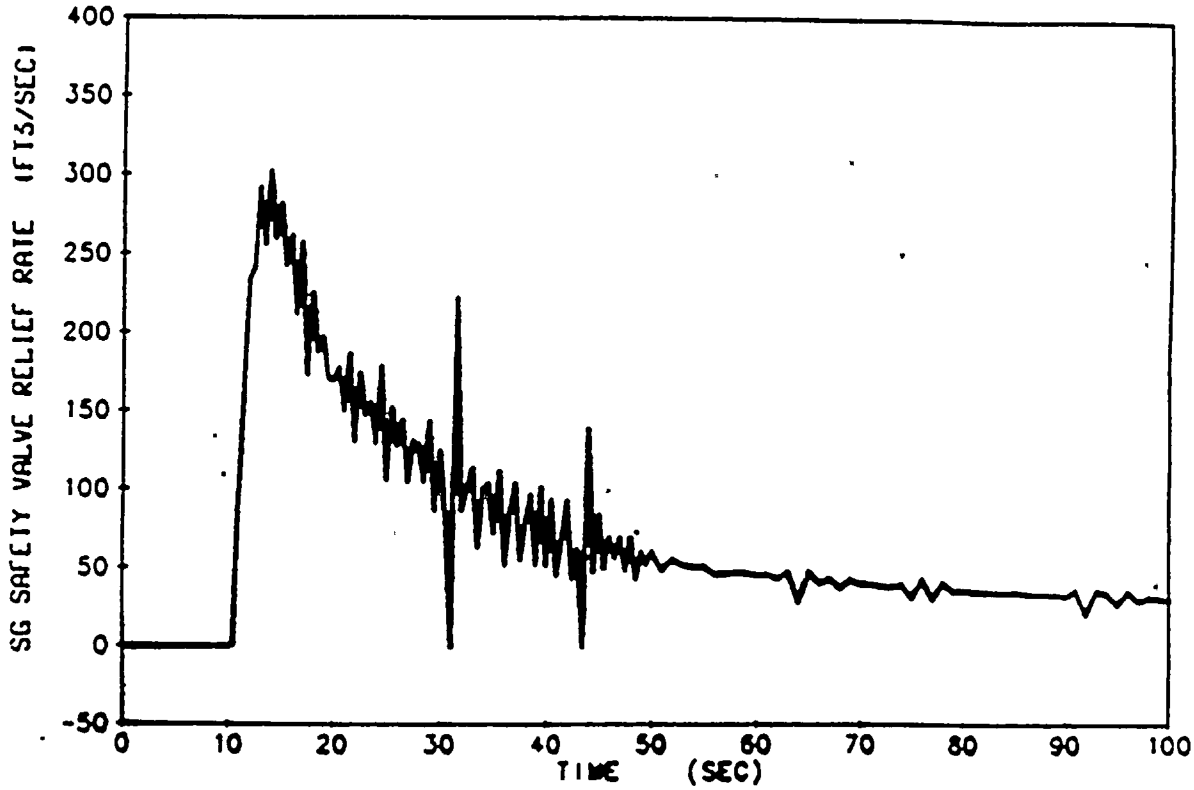


DONALD C. COOK UNIT 2
(FULL V5 CORE)

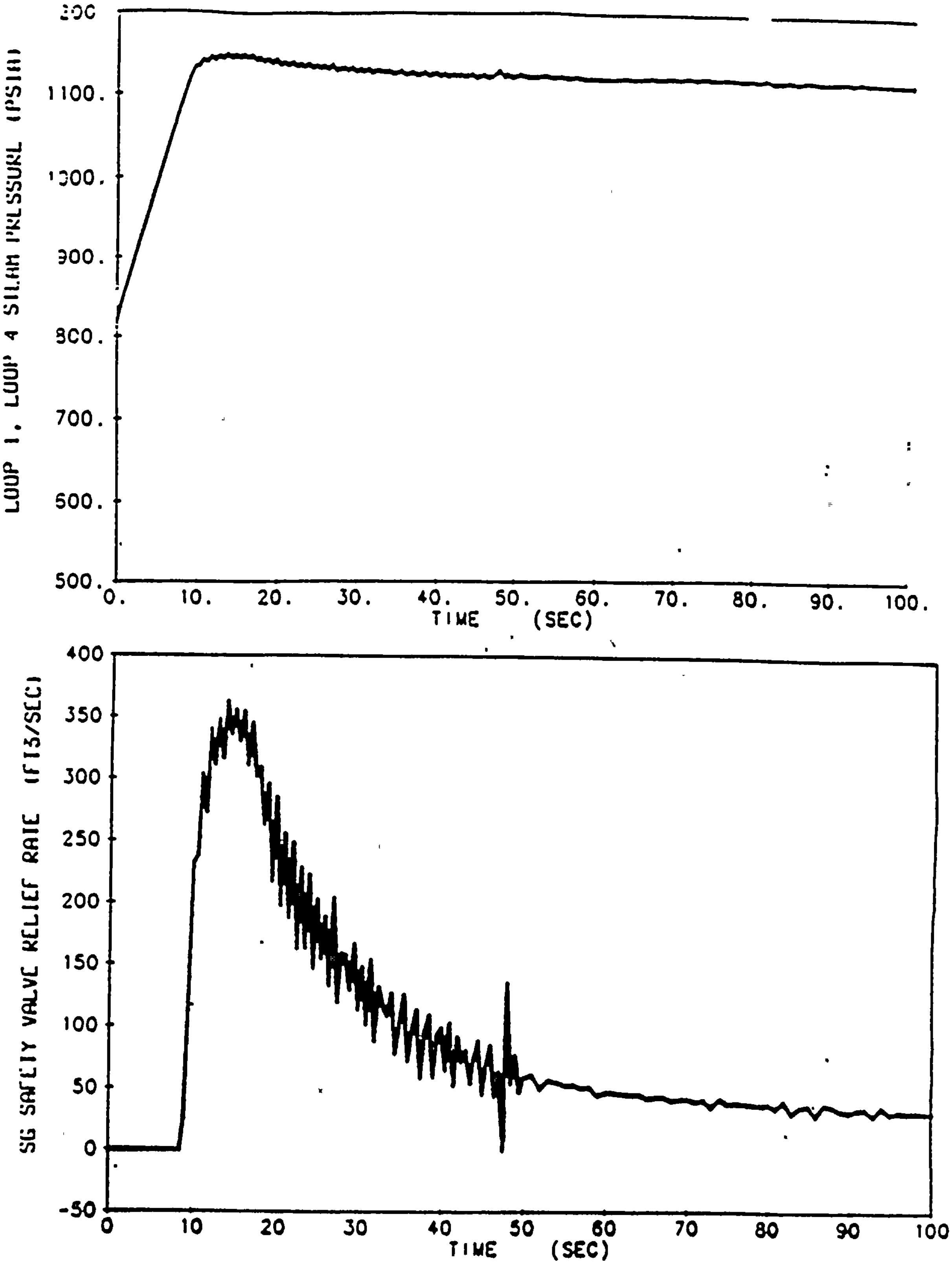
FIGURE 24b

TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK





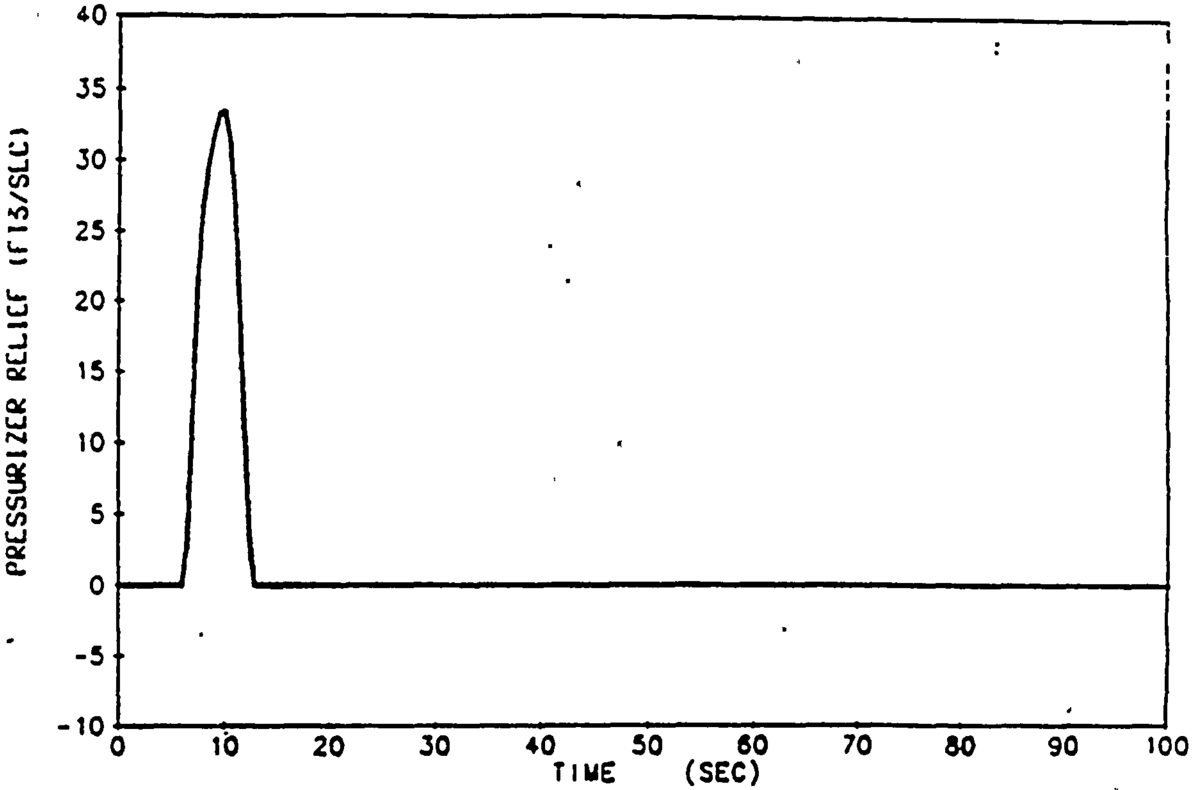
DONALD C. COOK UNIT 2 (MIXED CORE)
FIGURE 25a
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 25b

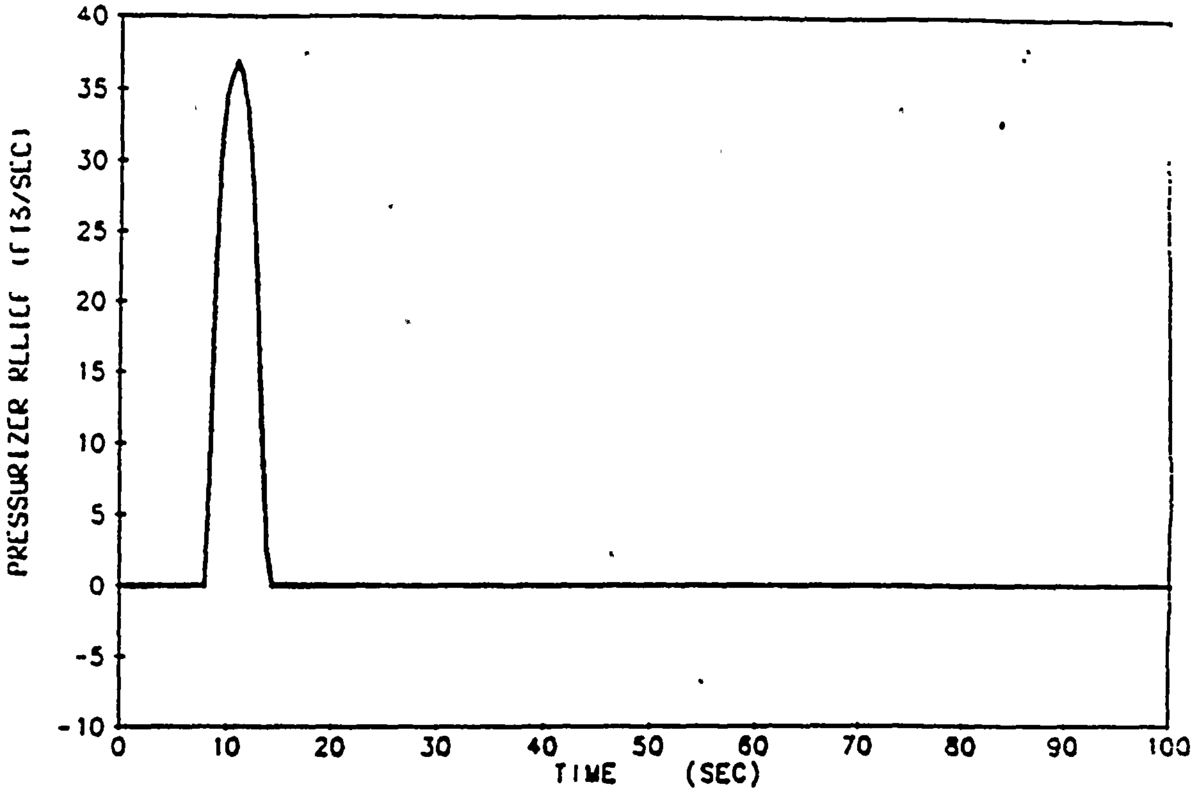
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 26a

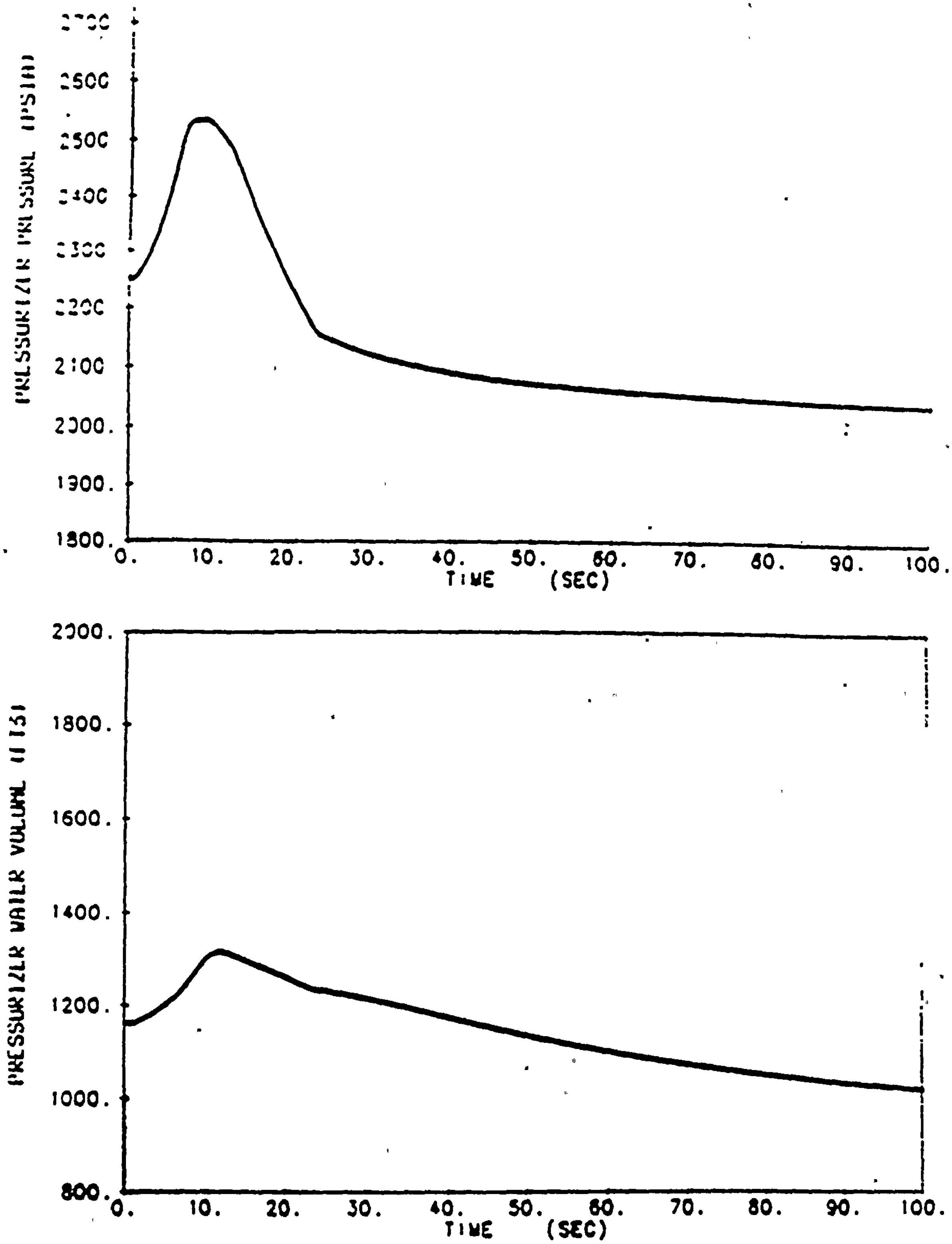
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 26b

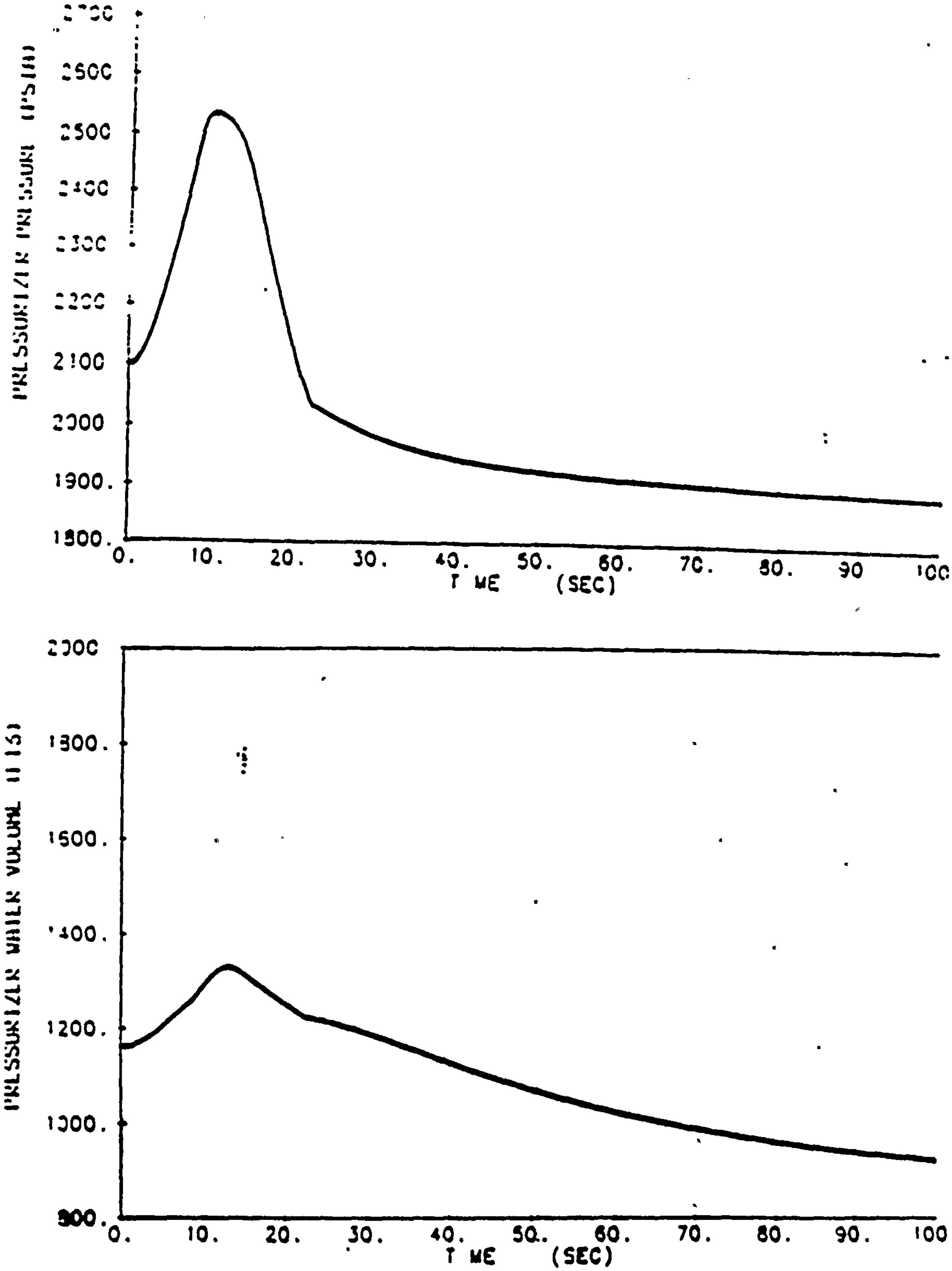
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 27a

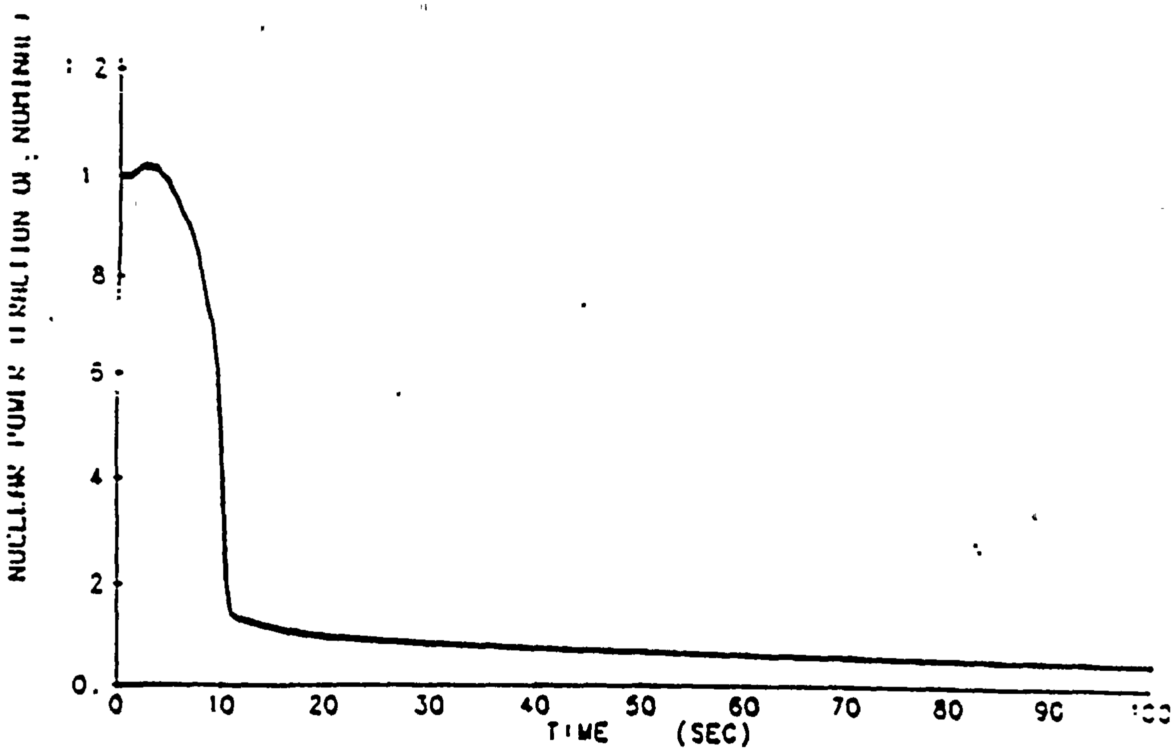
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK

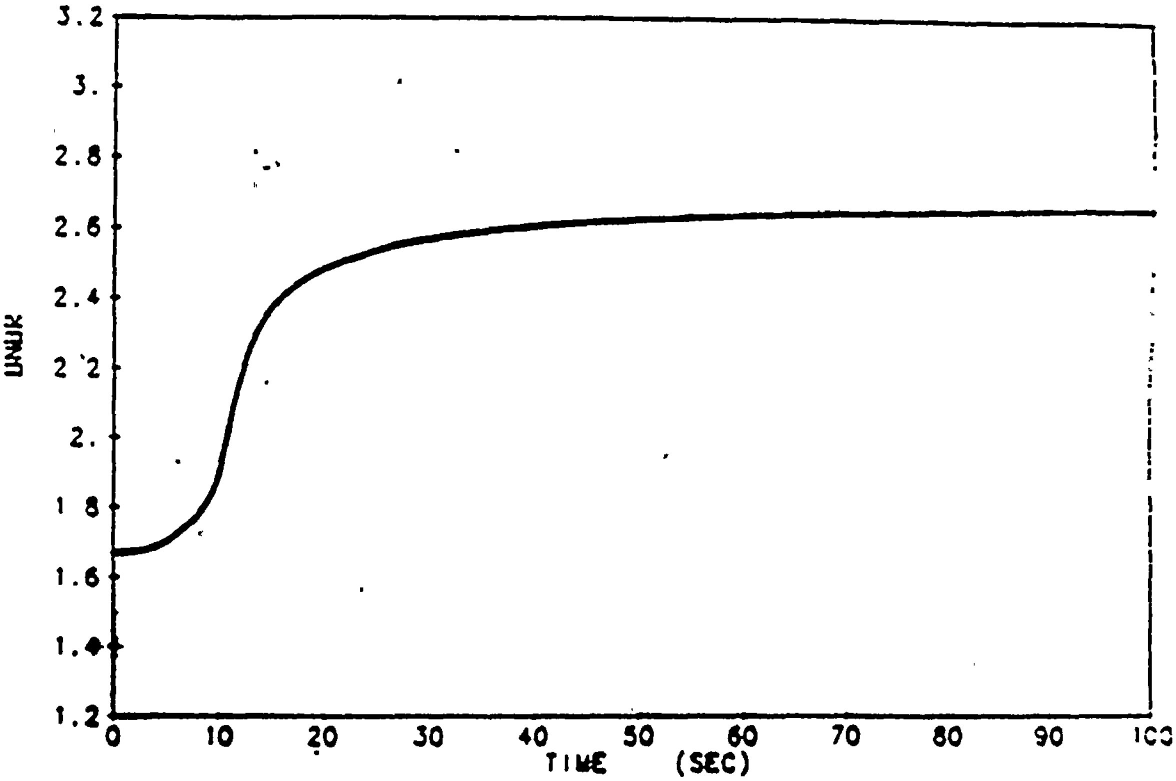


DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 27b

TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK

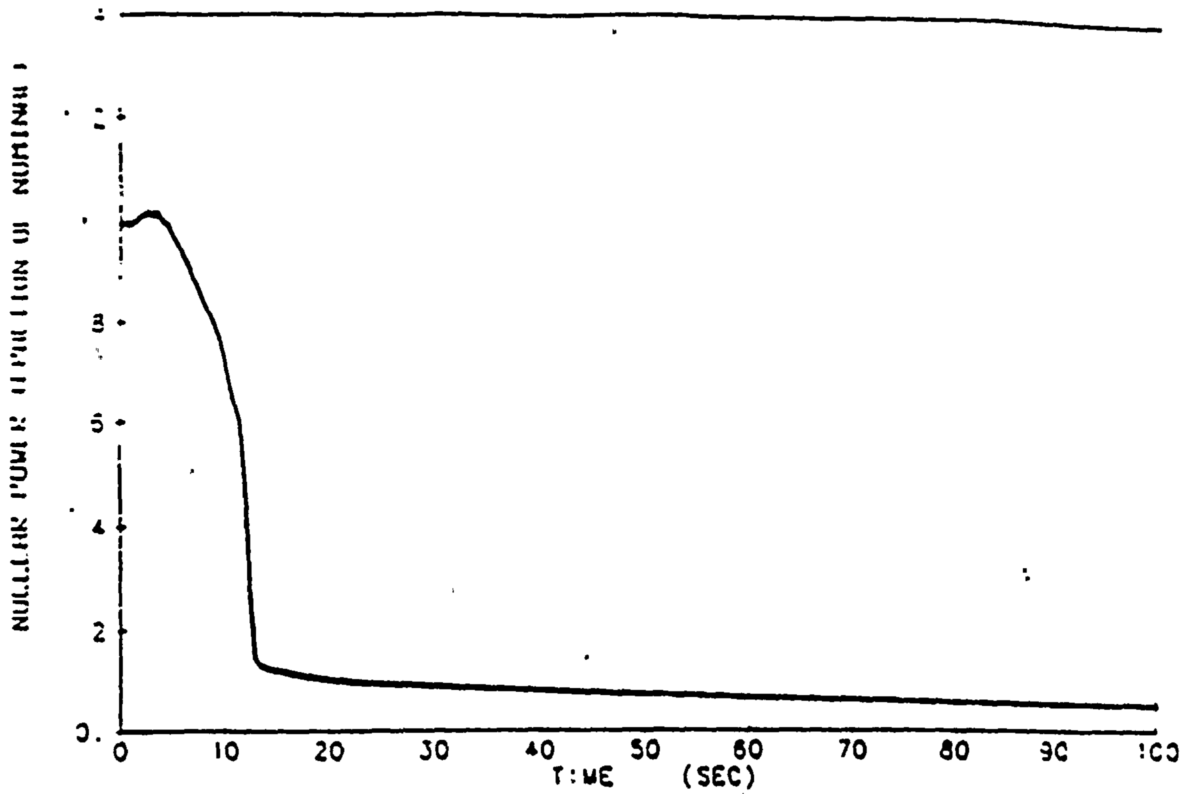


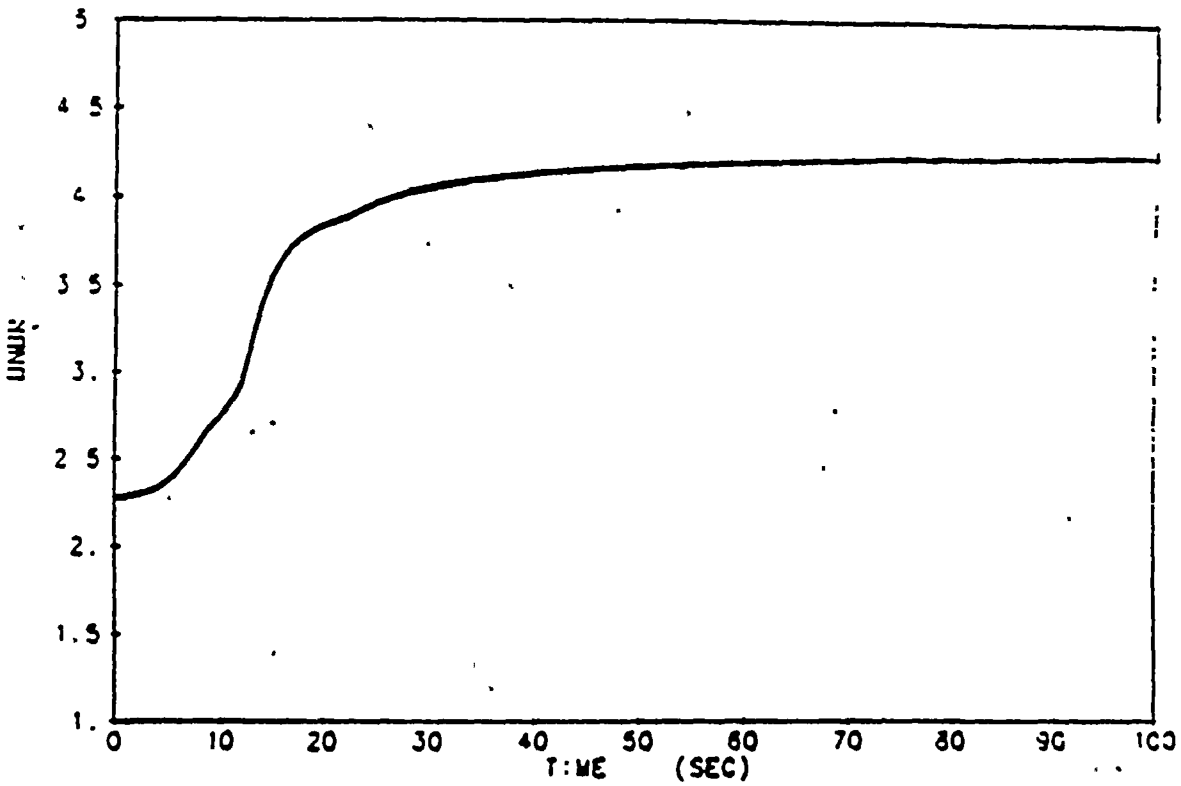


DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 28a

TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK

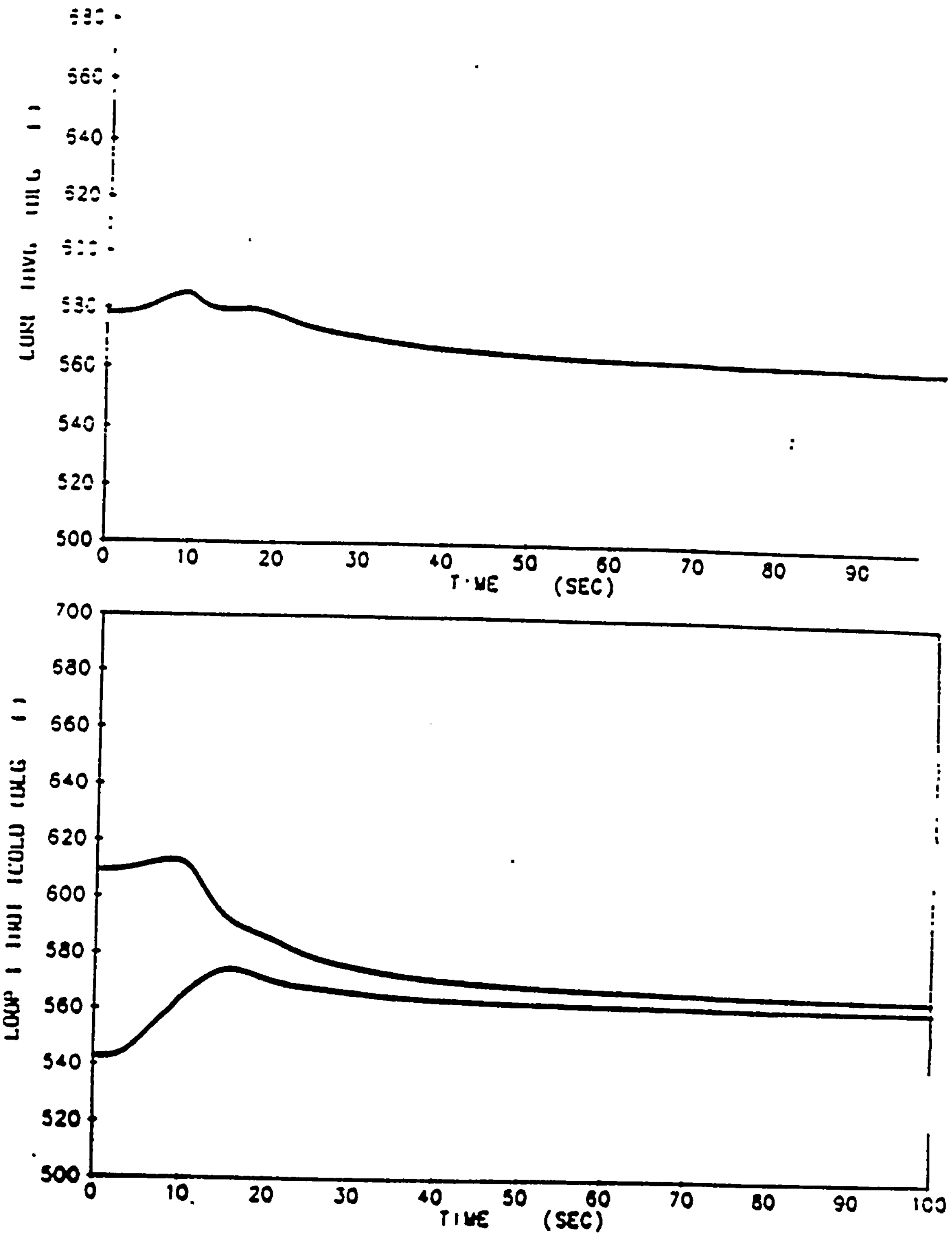




DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 28b

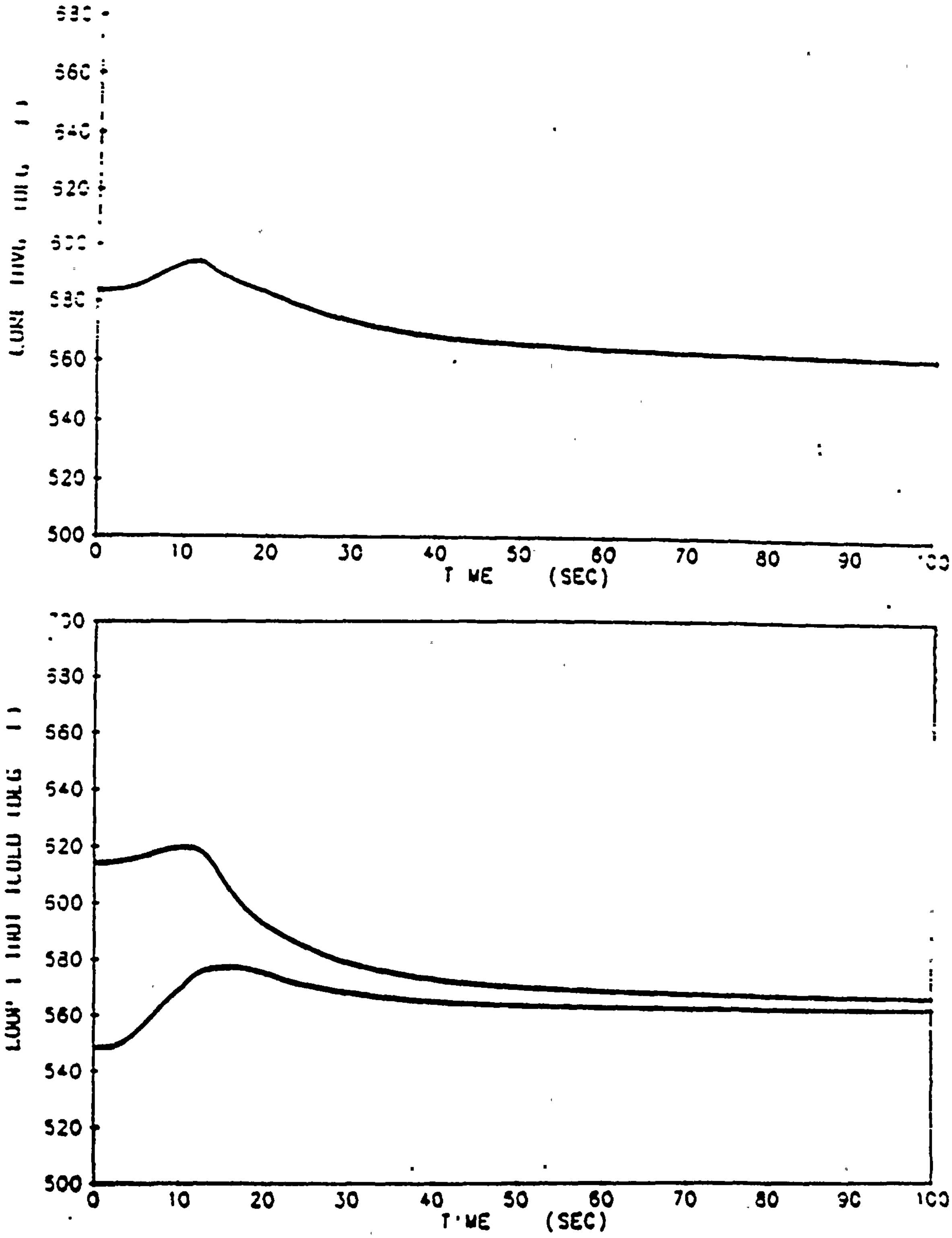
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 29a

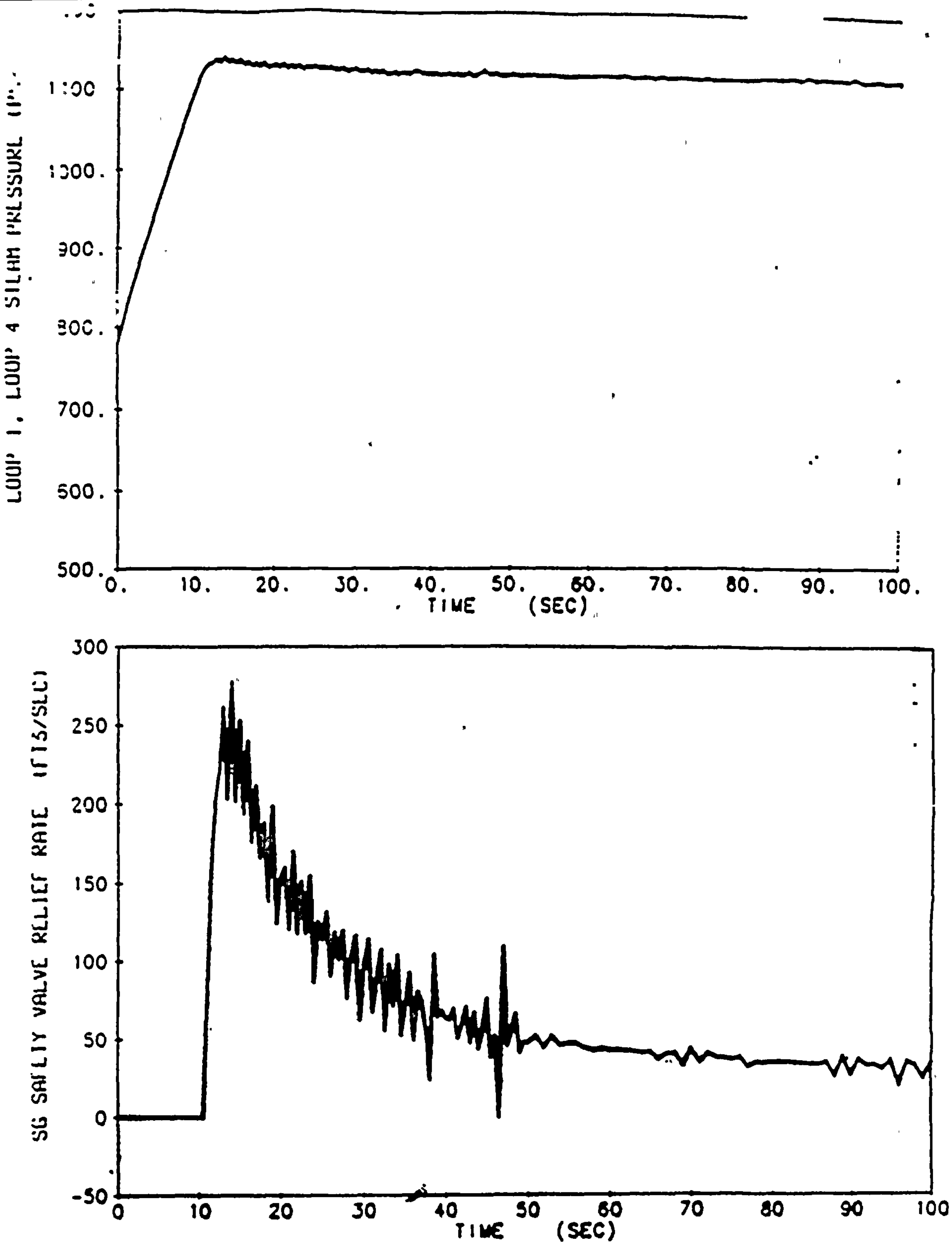
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK.



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 29b

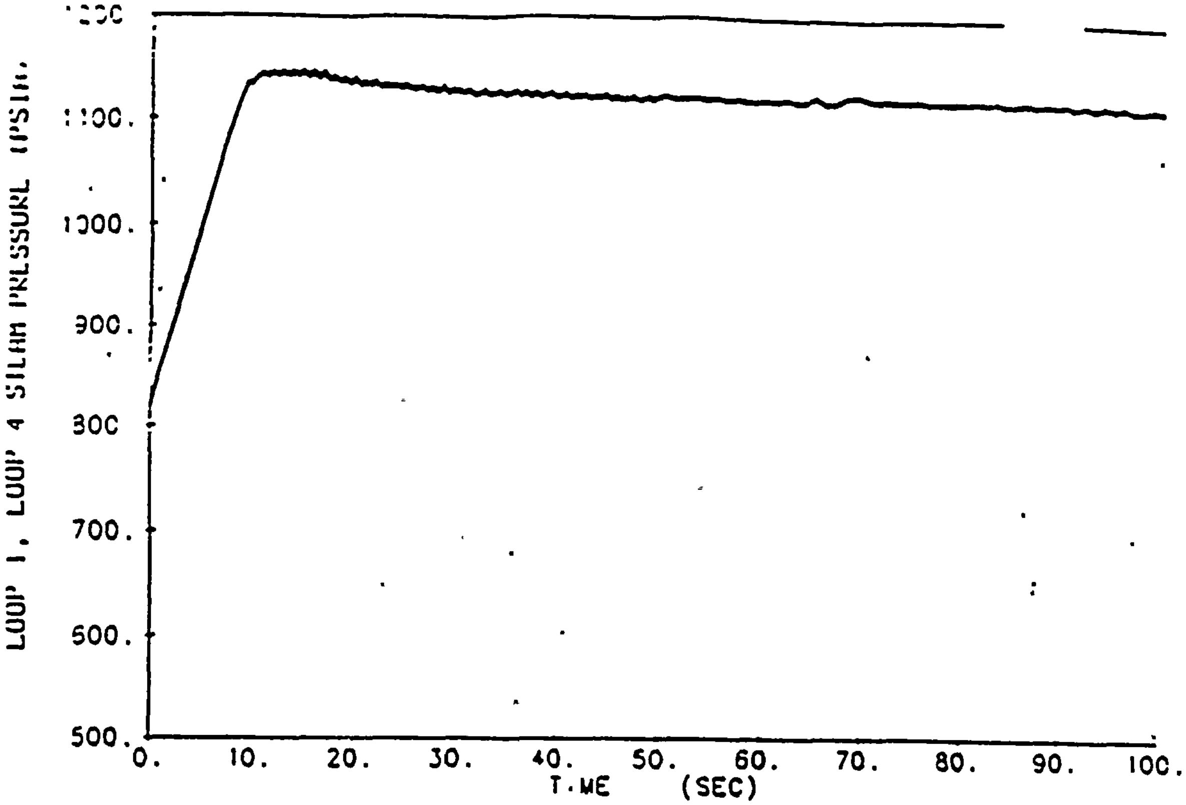
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK

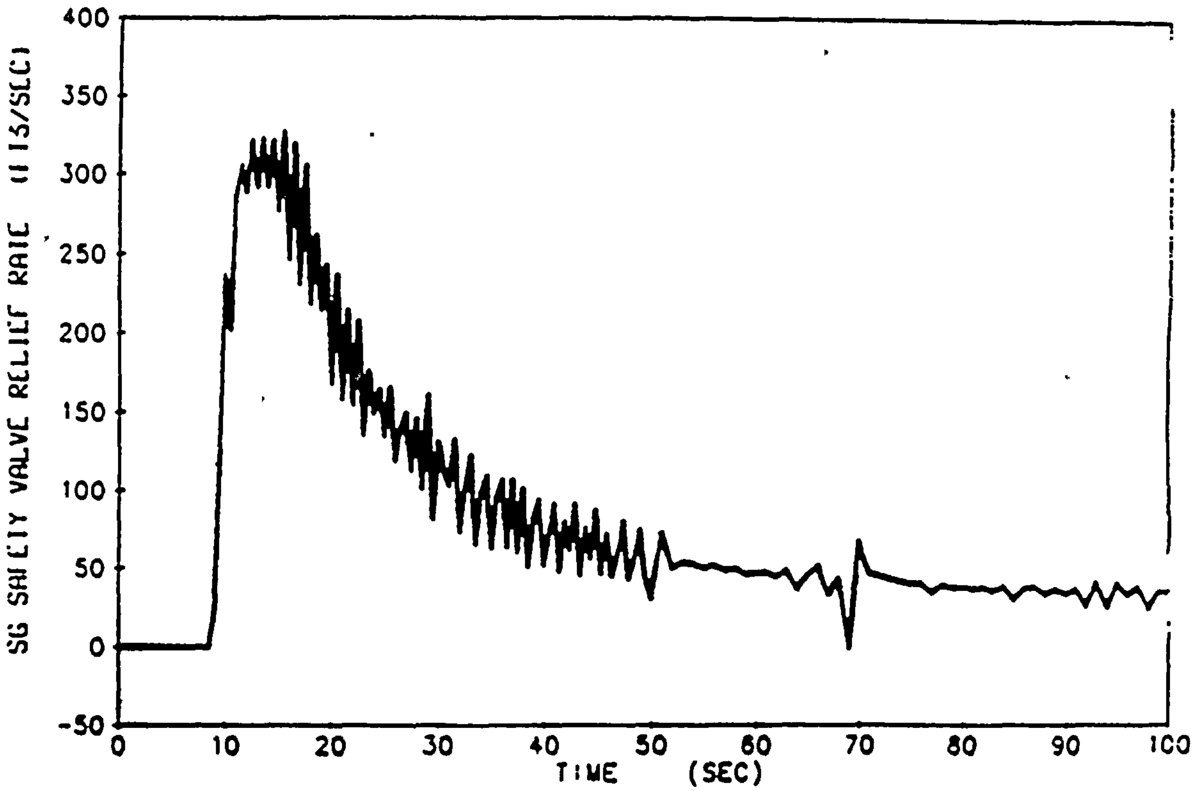


DONALD C. COOK UNIT 2
(MIXED CORE)

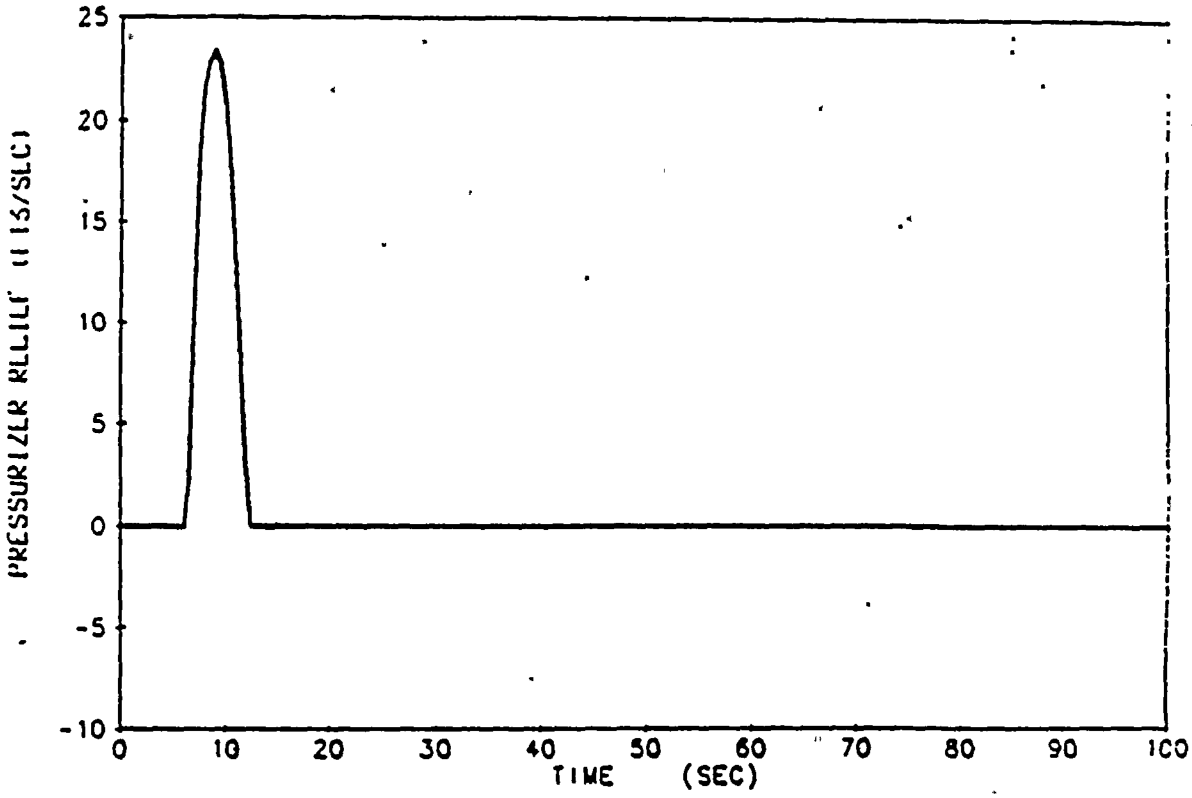
FIGURE 30a

TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK





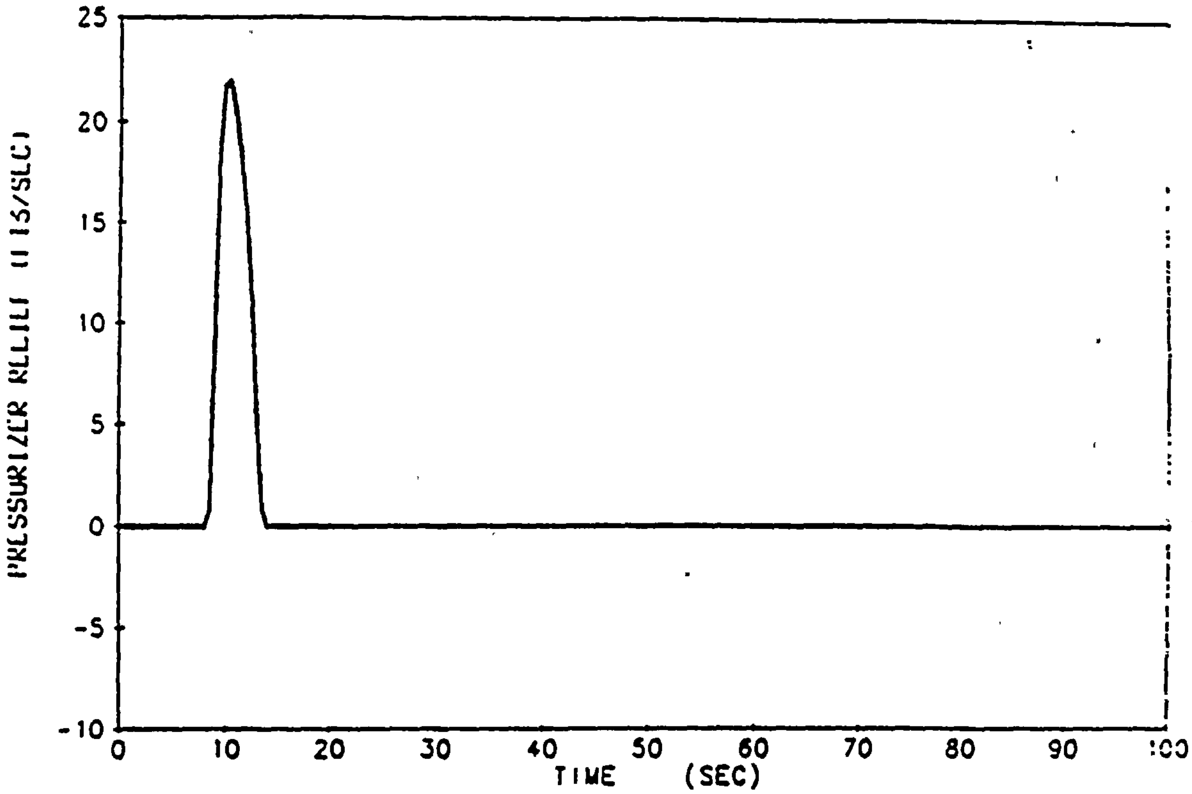
DONALD C. COOK UNIT 2 (FULL V5 CORE)
FIGURE 30b TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 31a

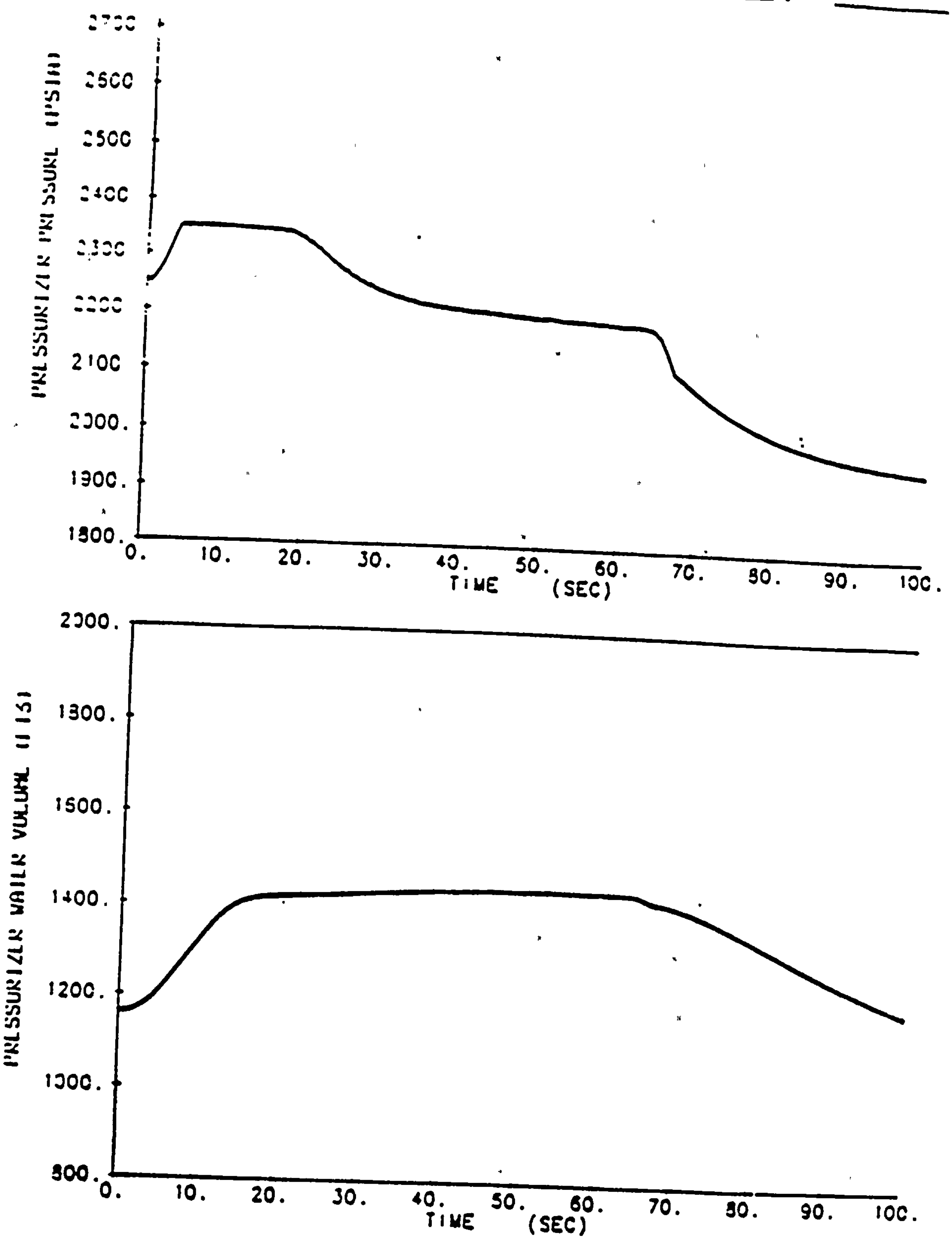
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK.



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 31b

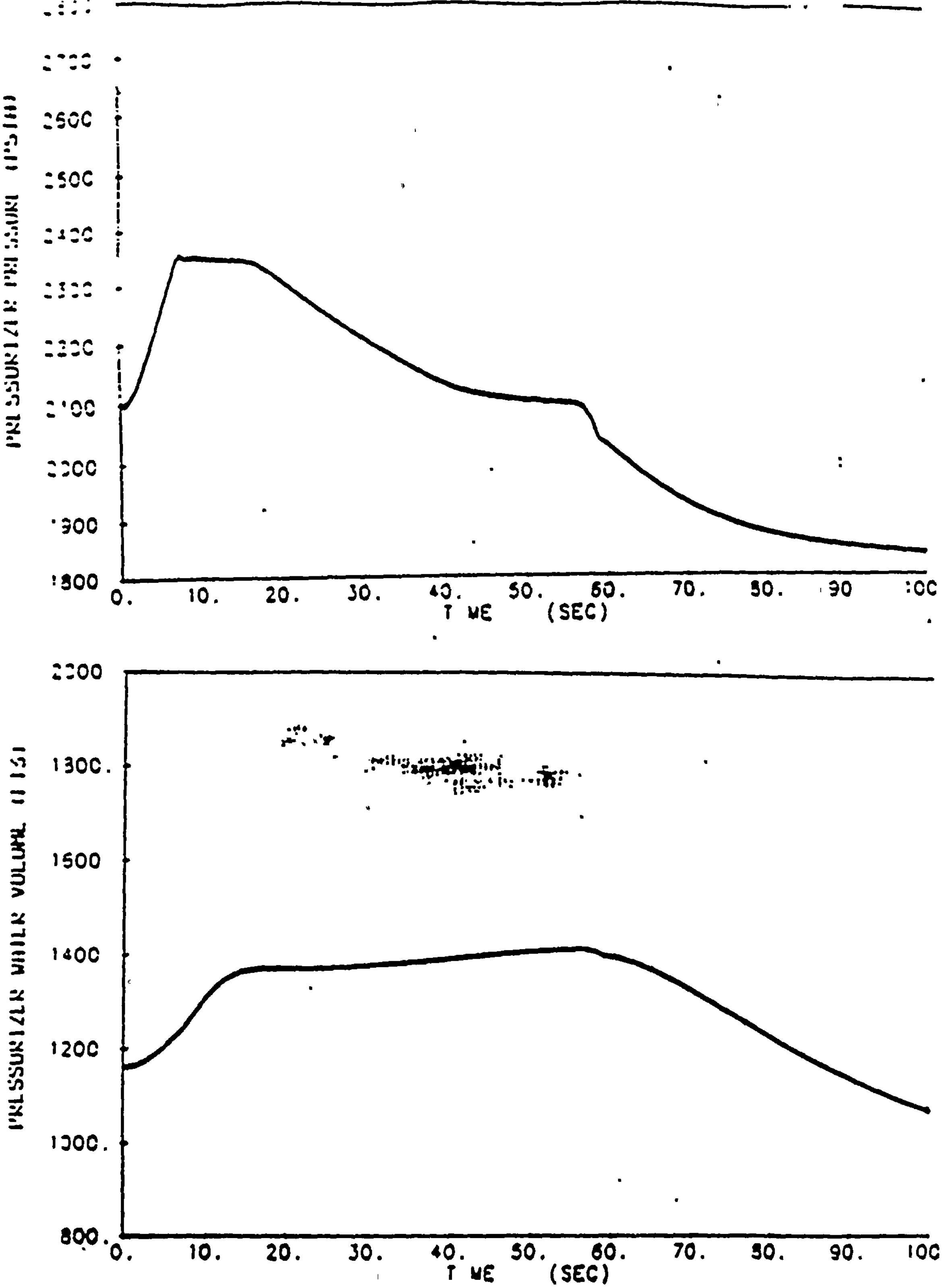
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 32a

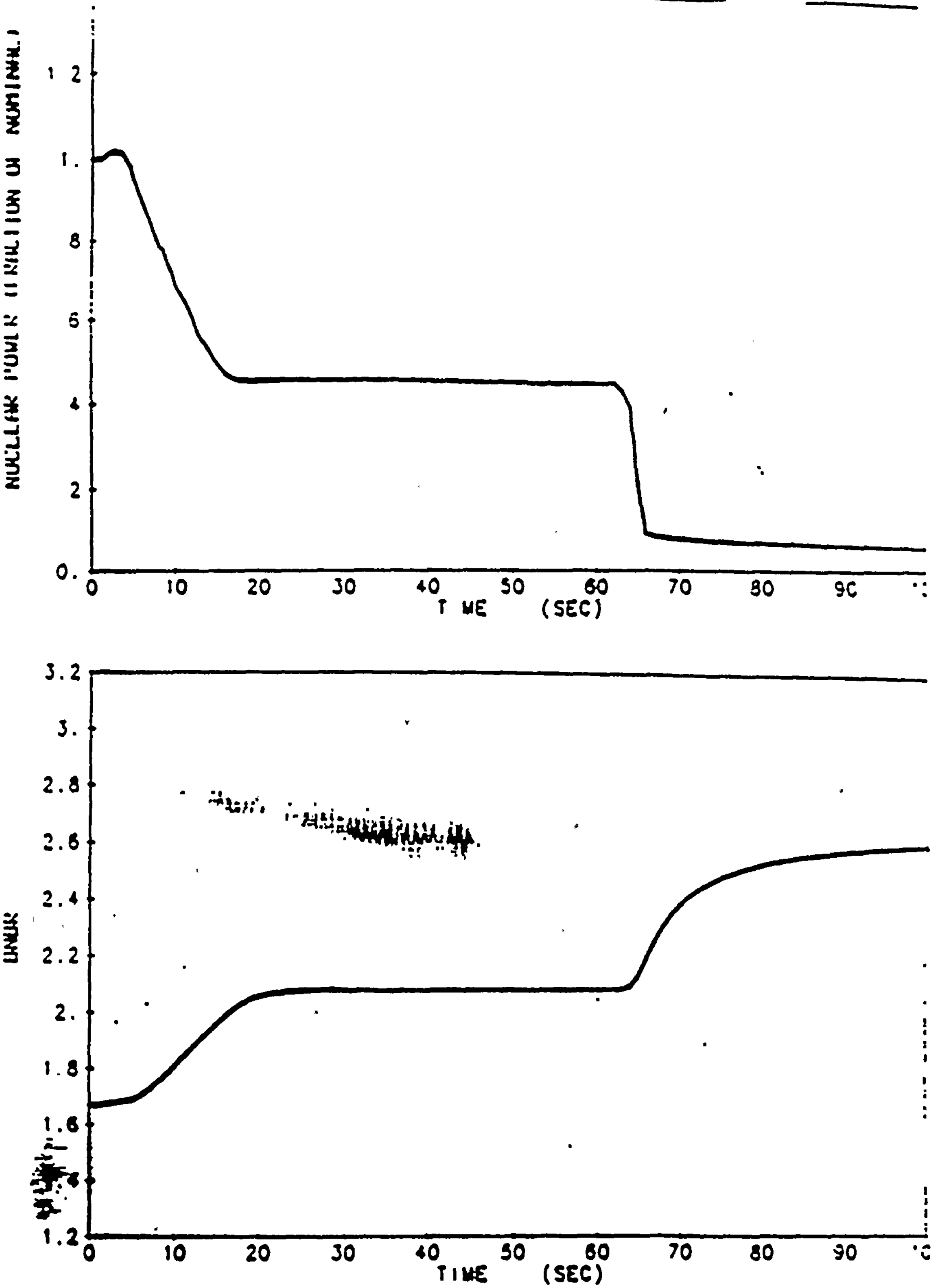
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL VS CORE)

FIGURE 32b

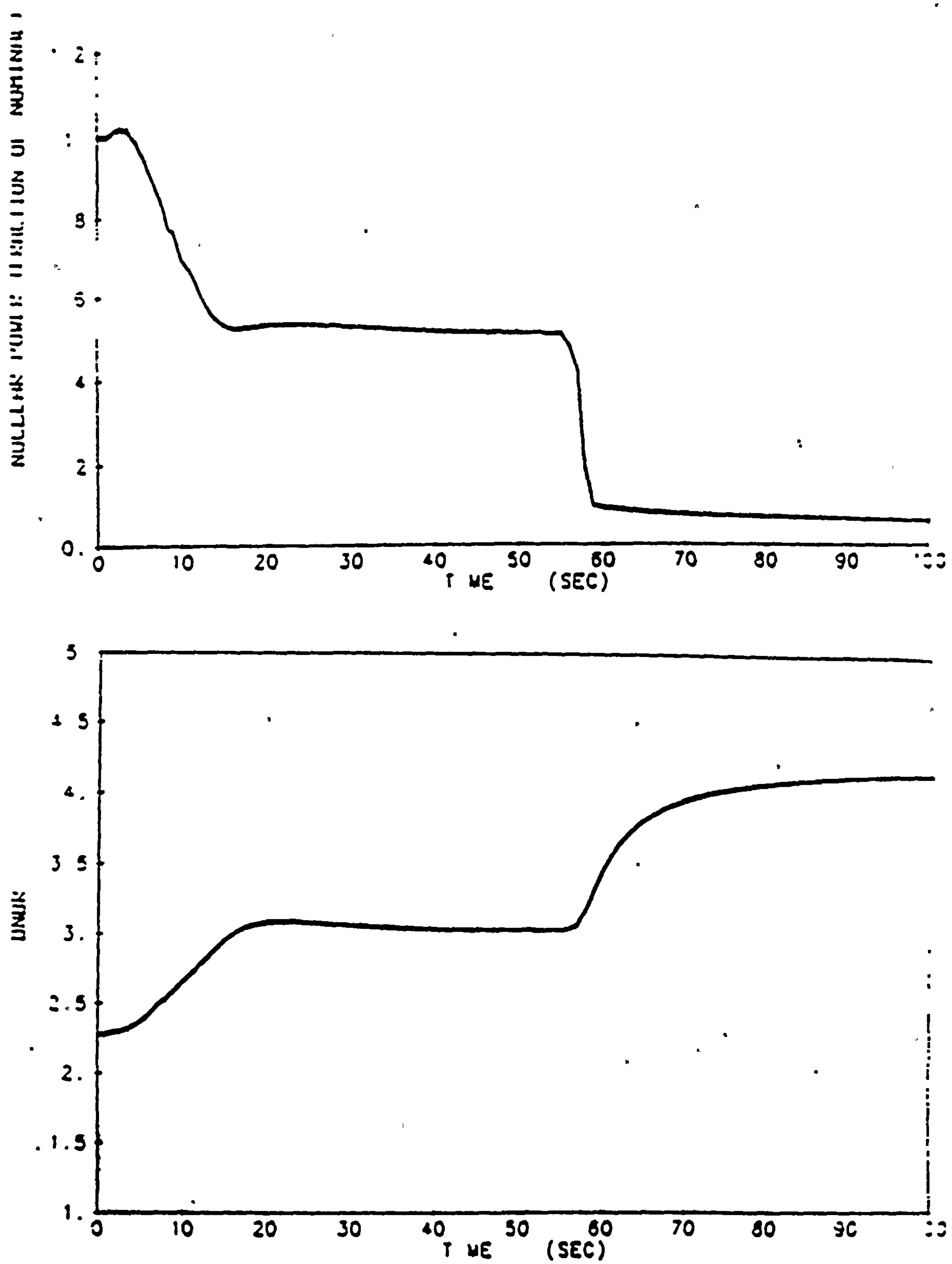
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 33a

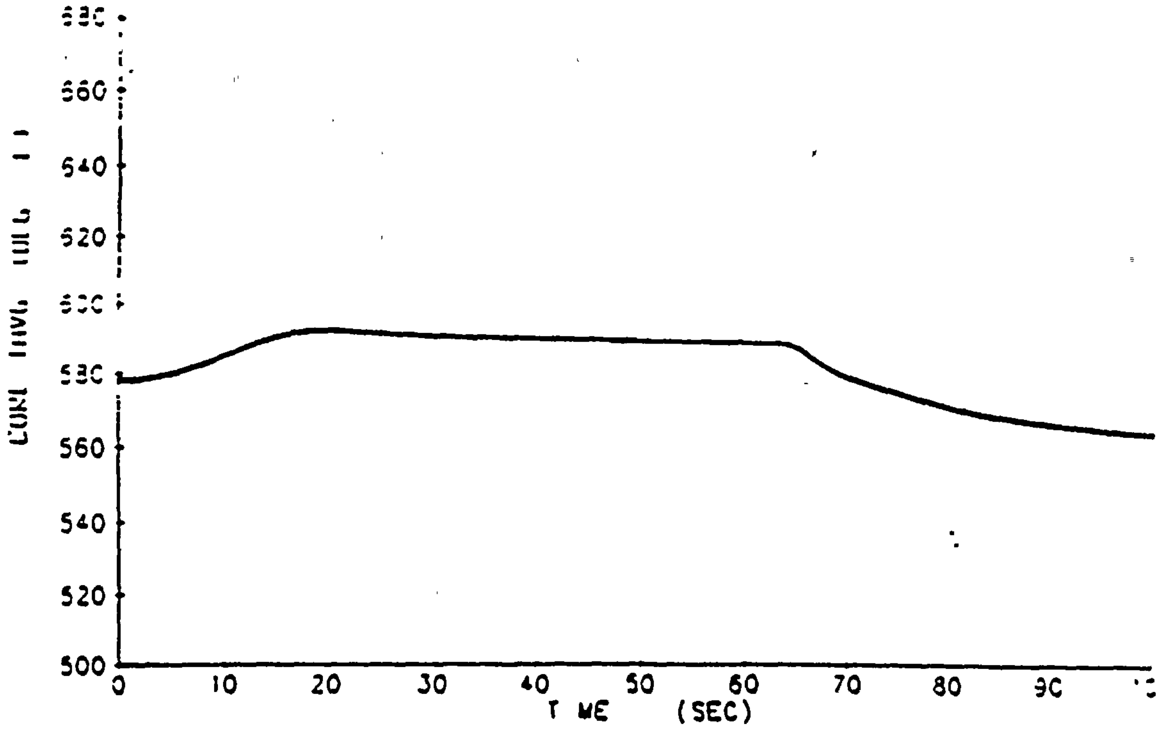
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK

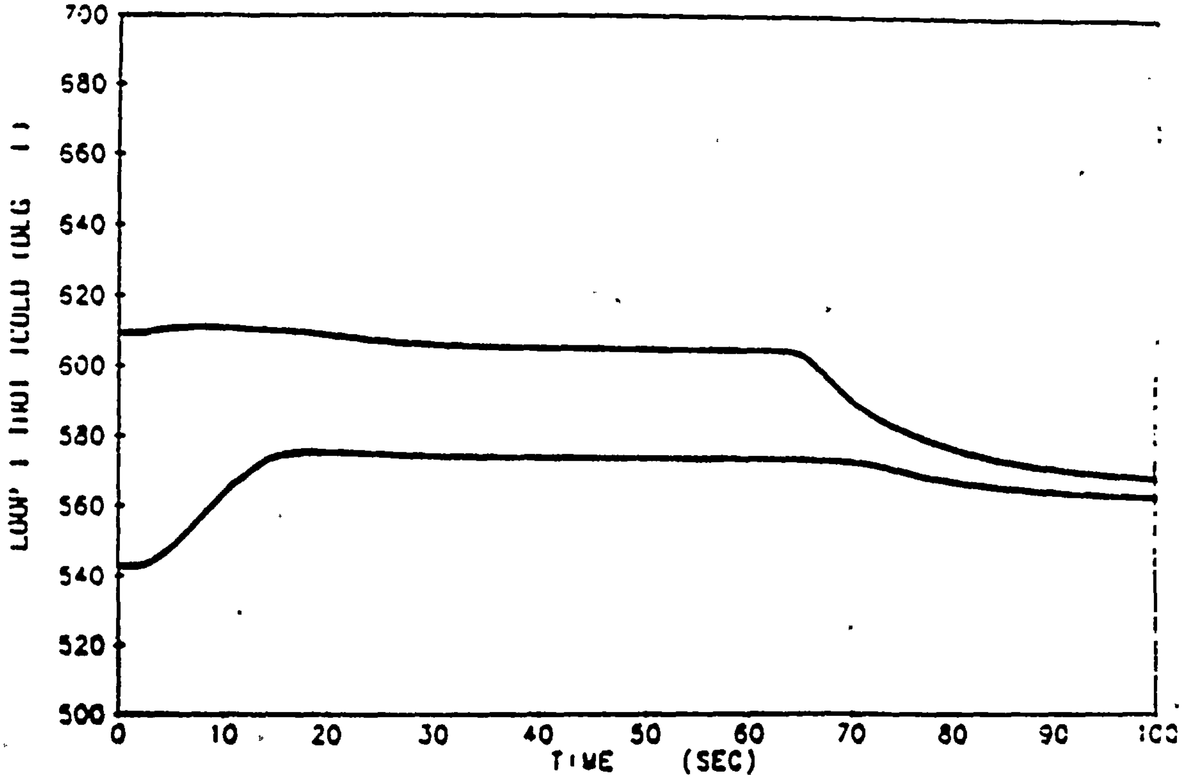


DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 33b

TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK

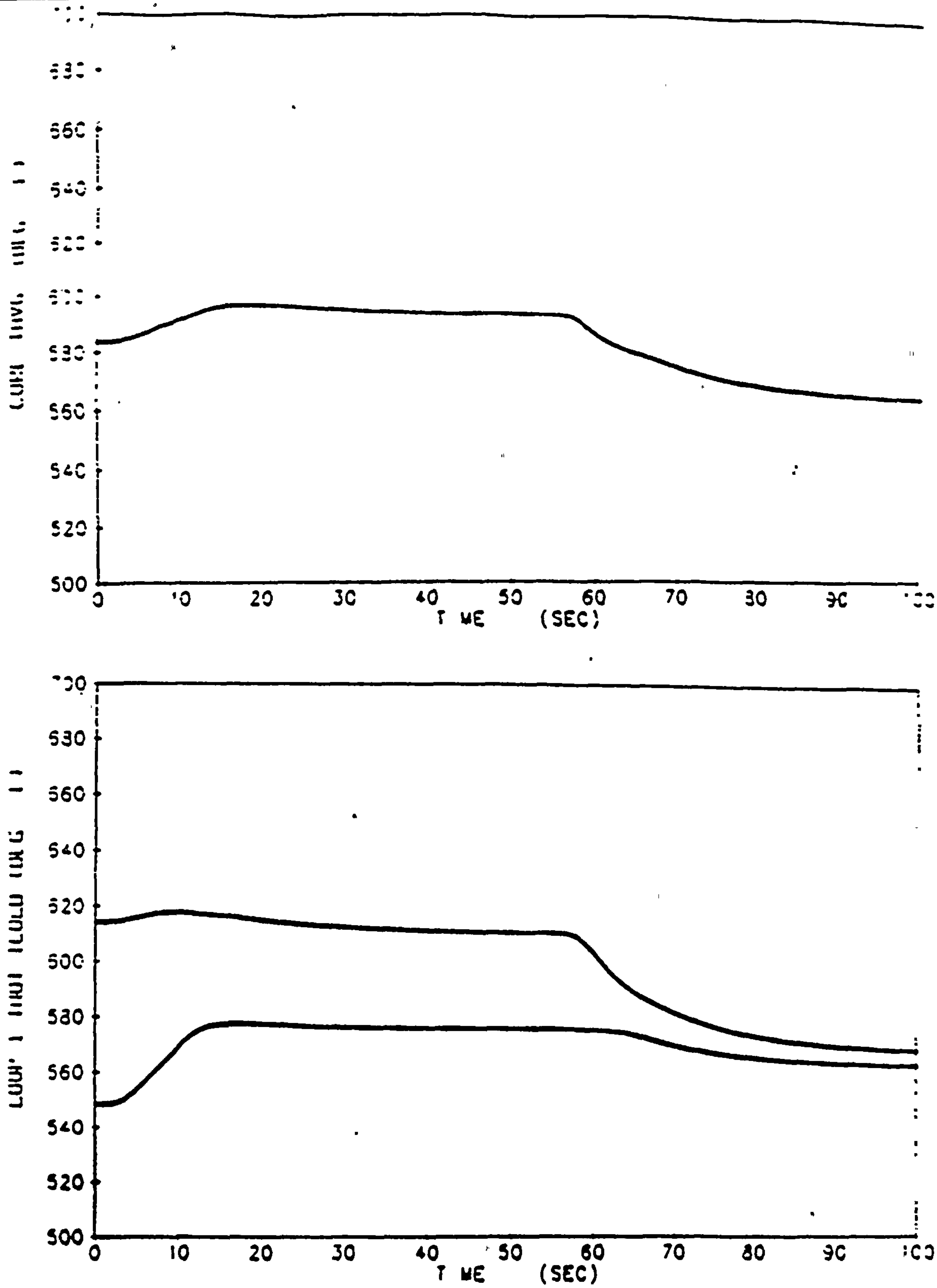




DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 34a

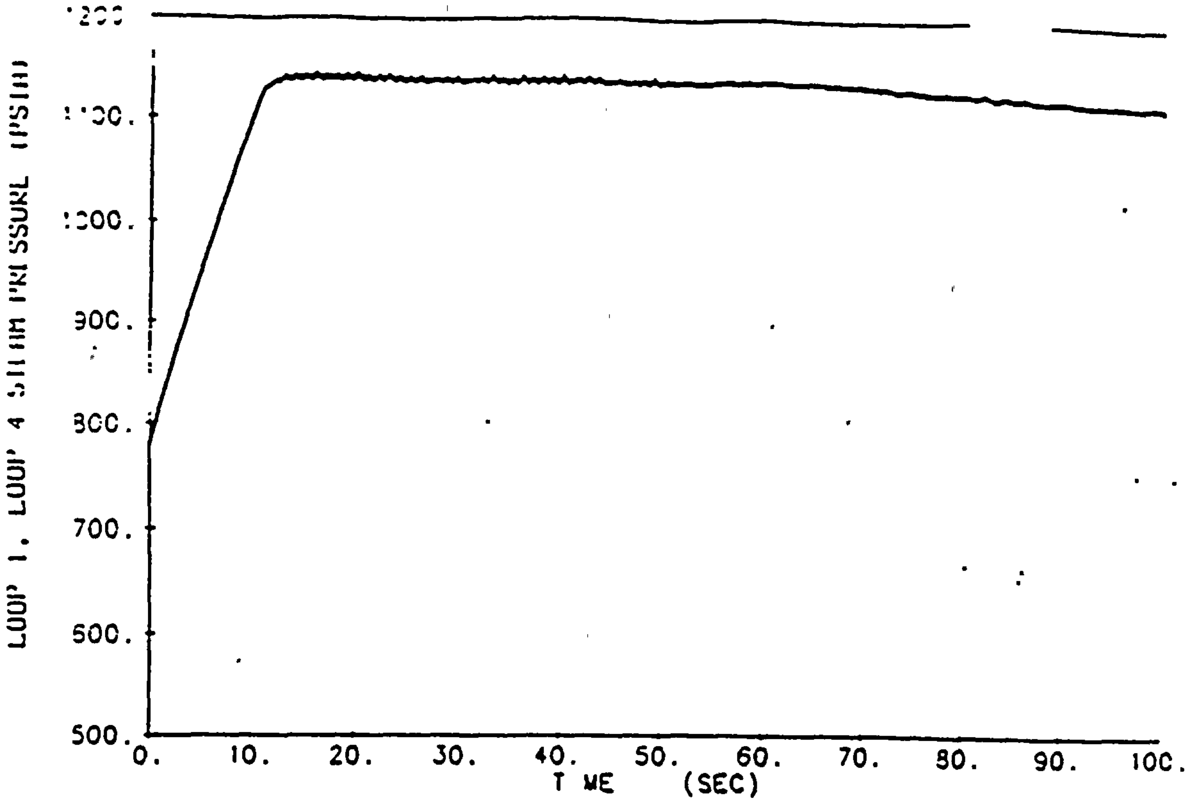
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK

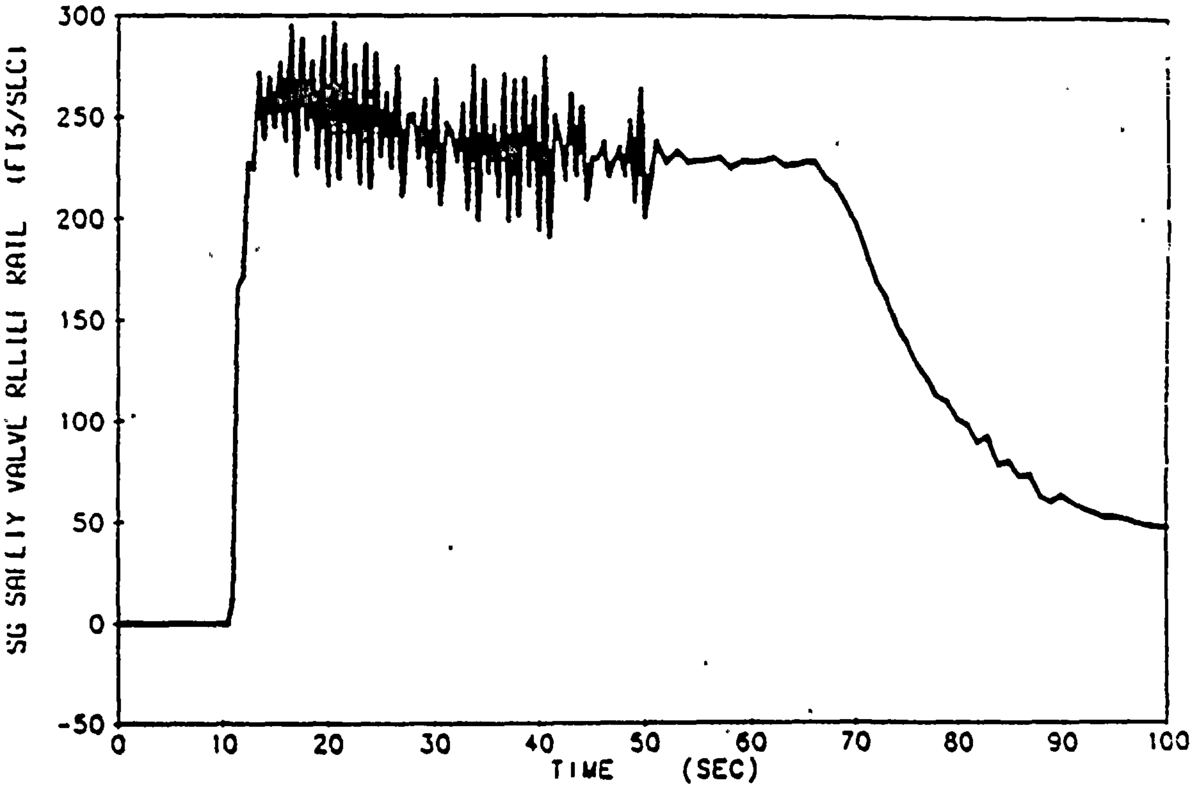


DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 34b

TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK

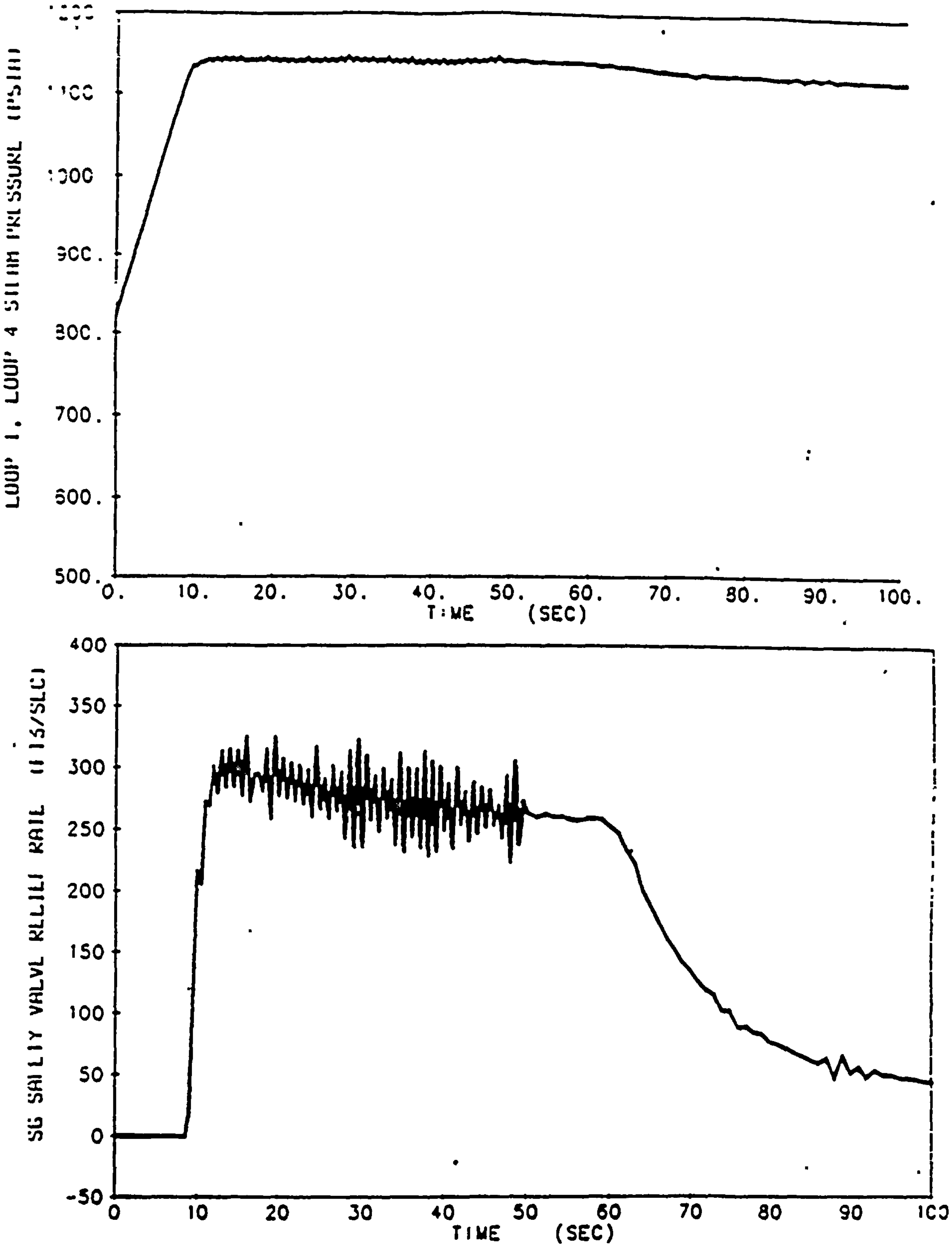




DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 35a

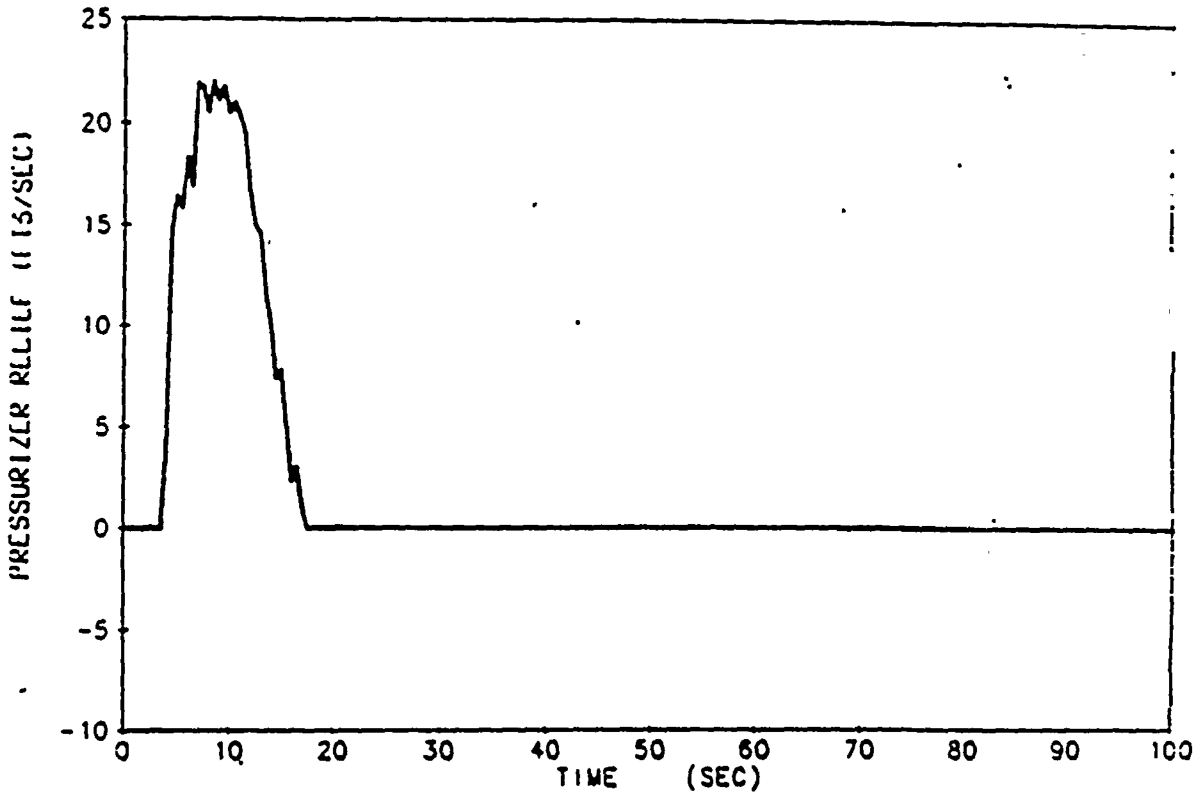
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 35b

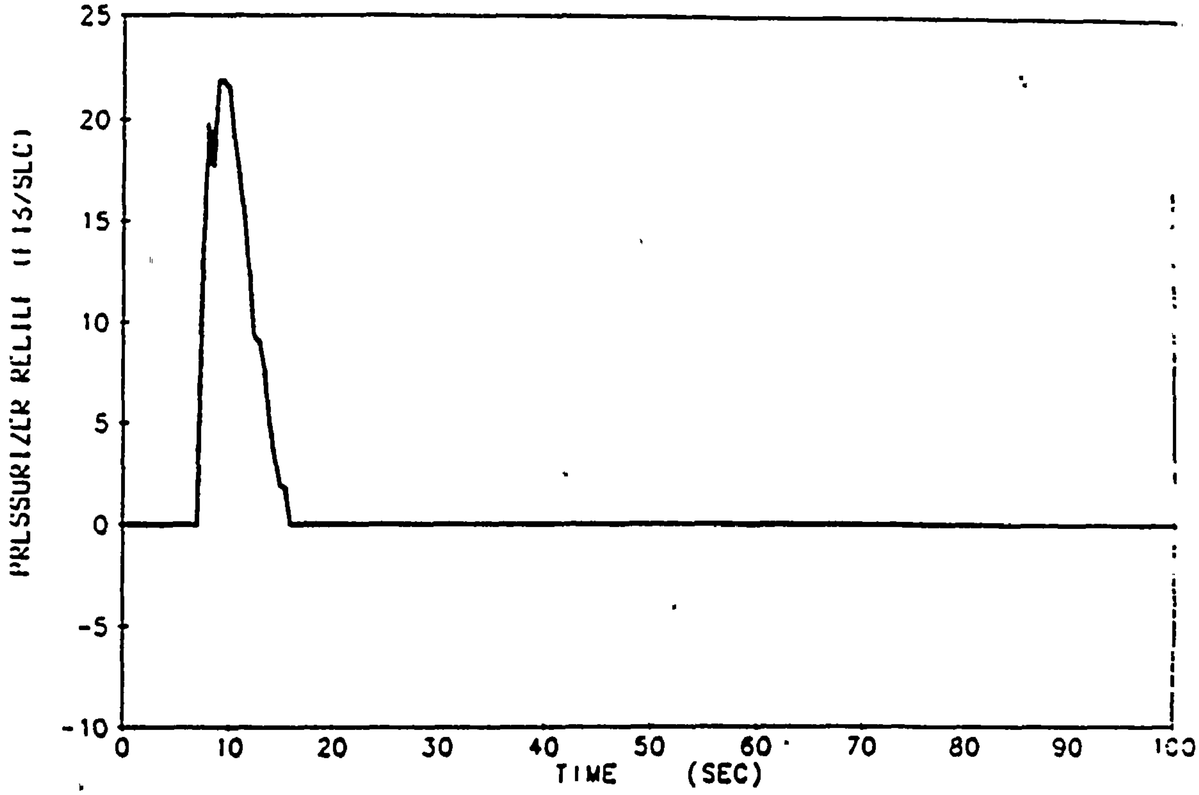
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 36a

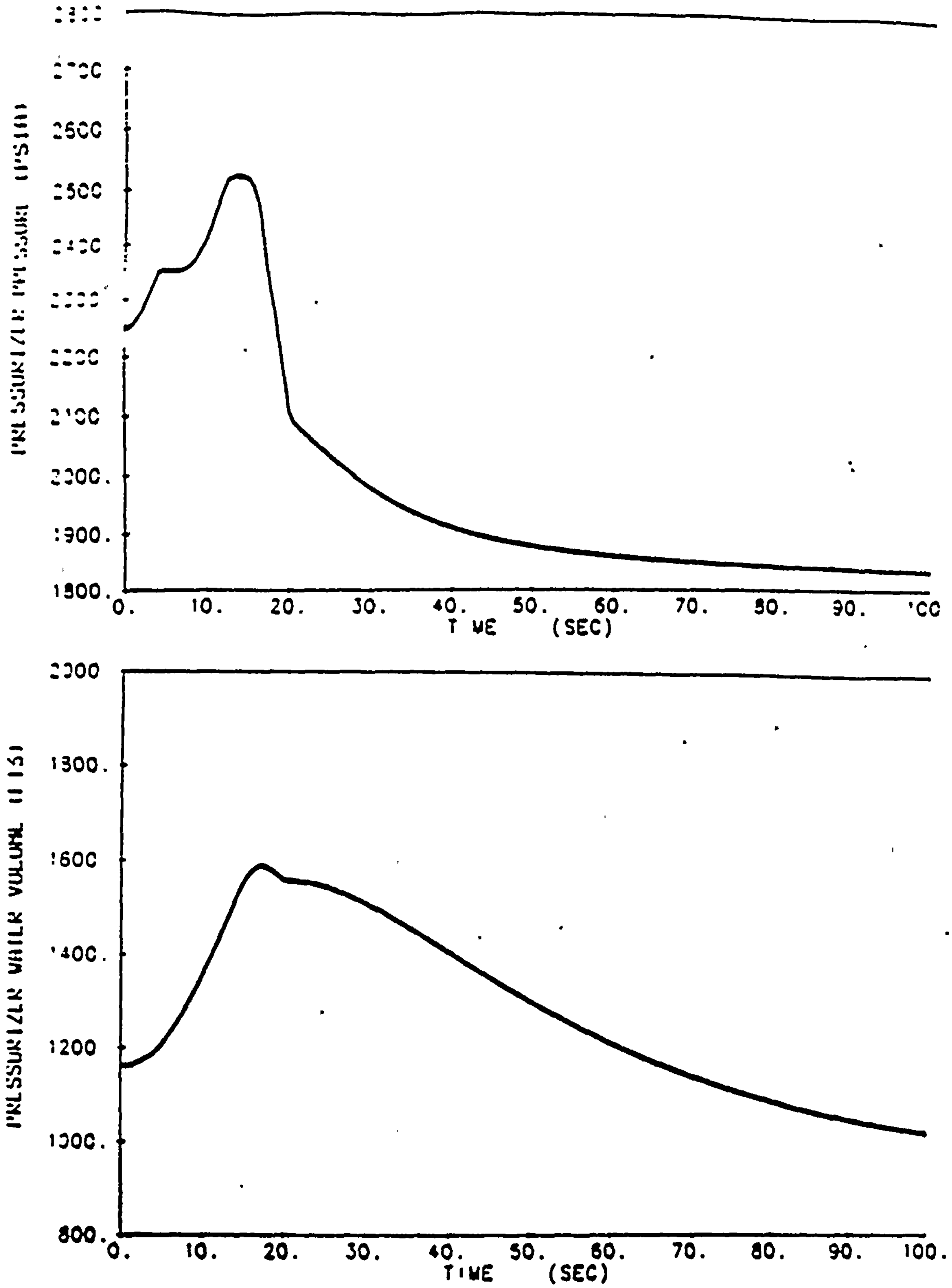
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 36b

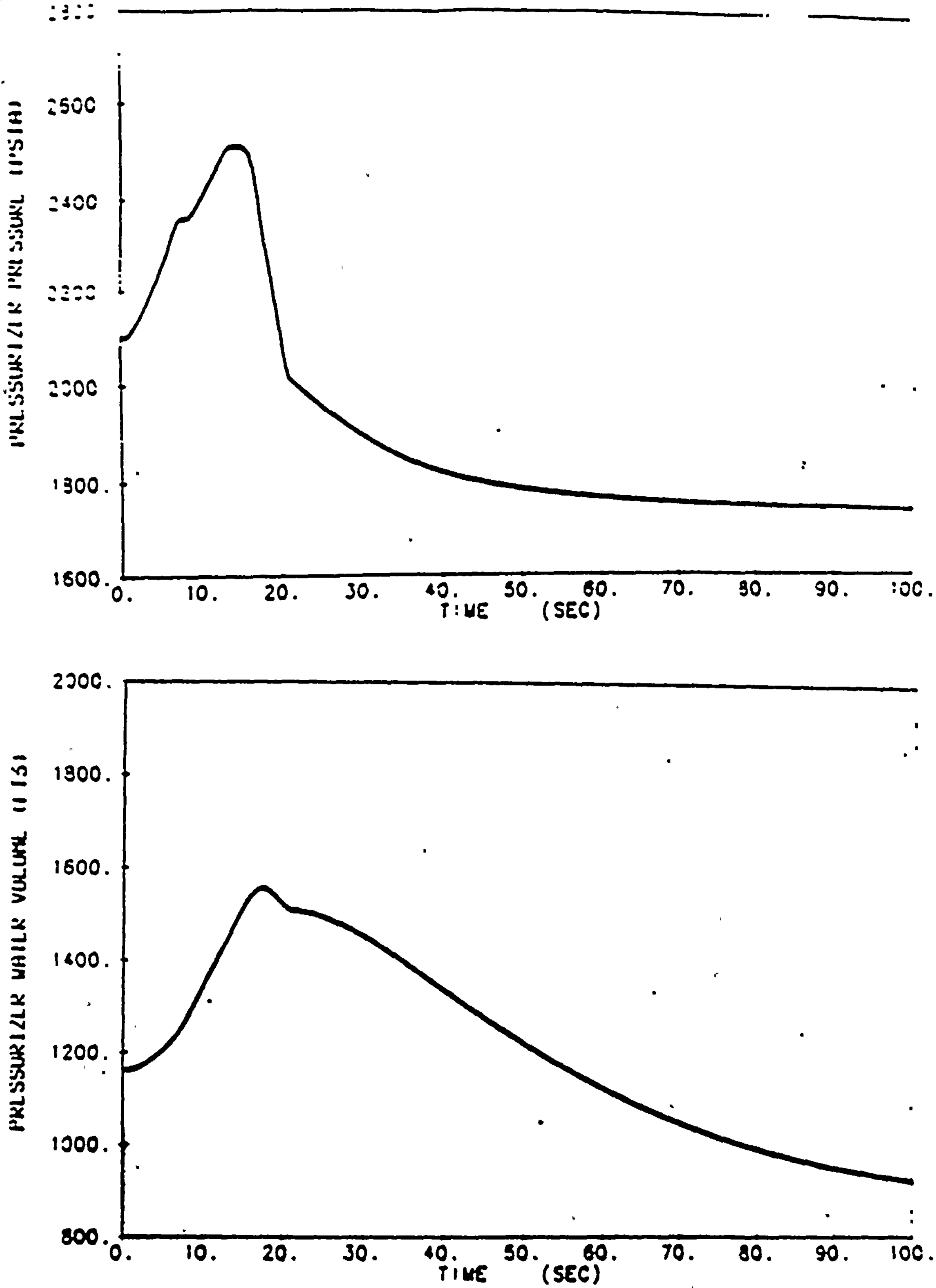
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 37a

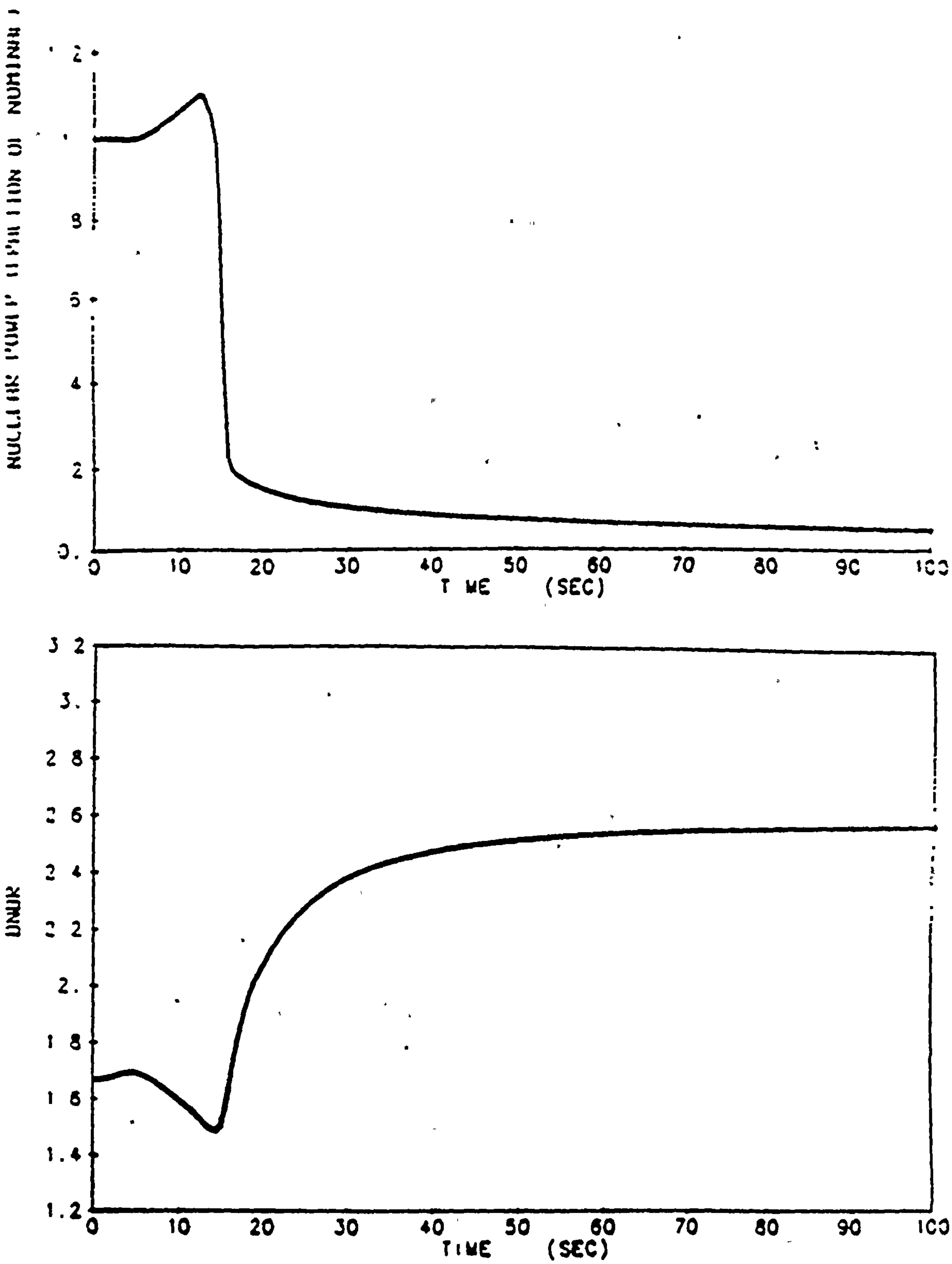
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 37b

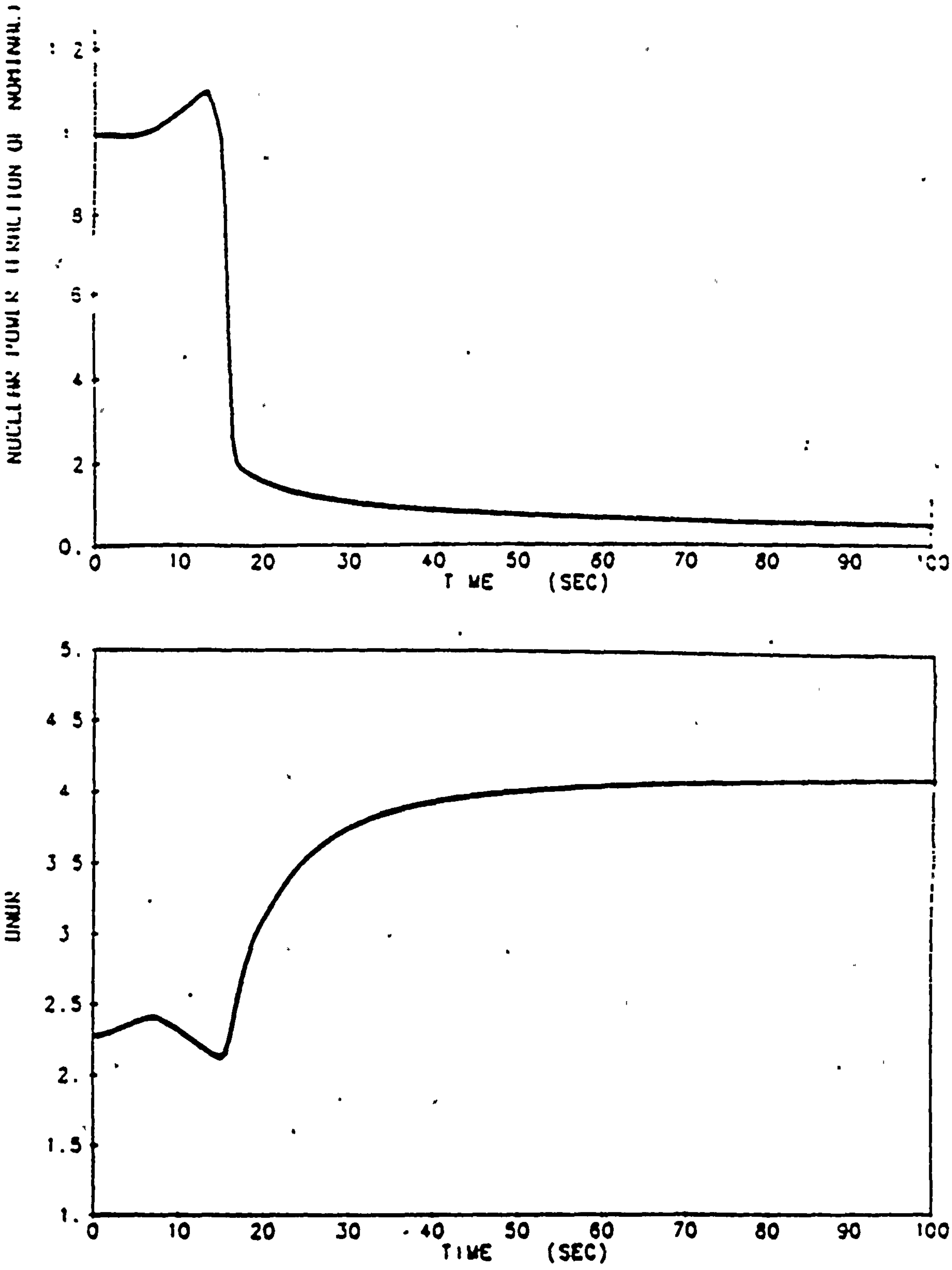
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 38a

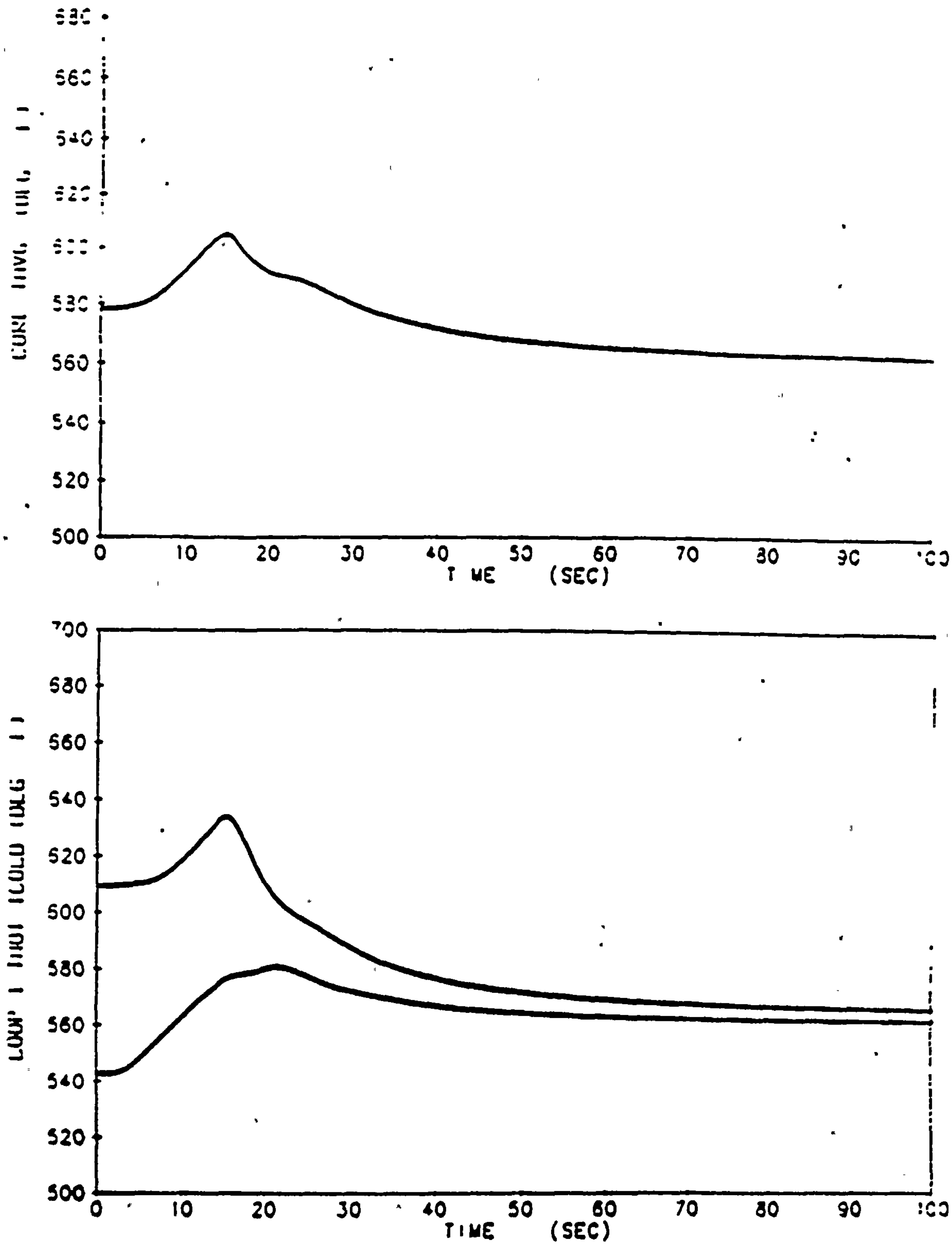
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 38b

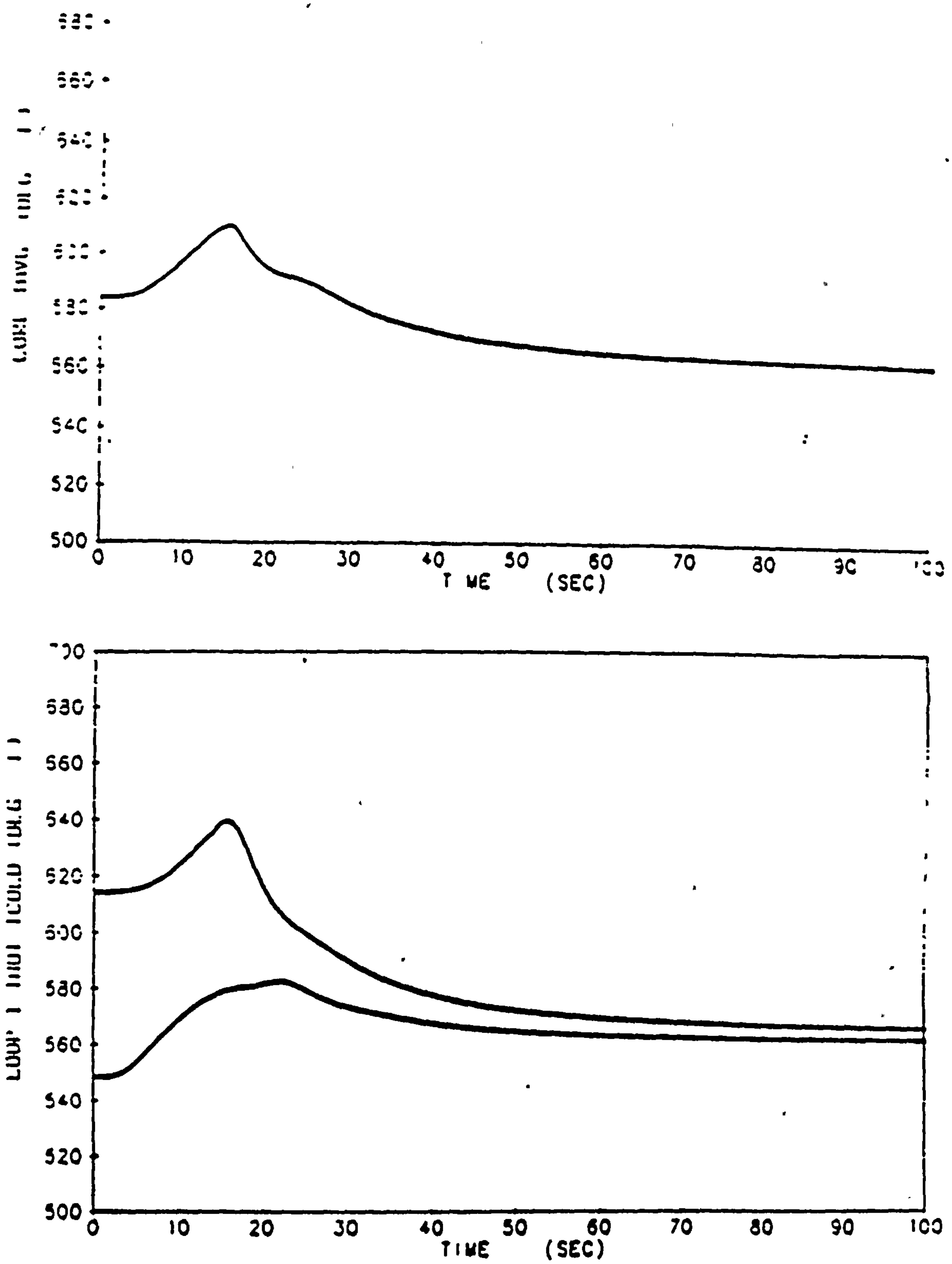
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 39a

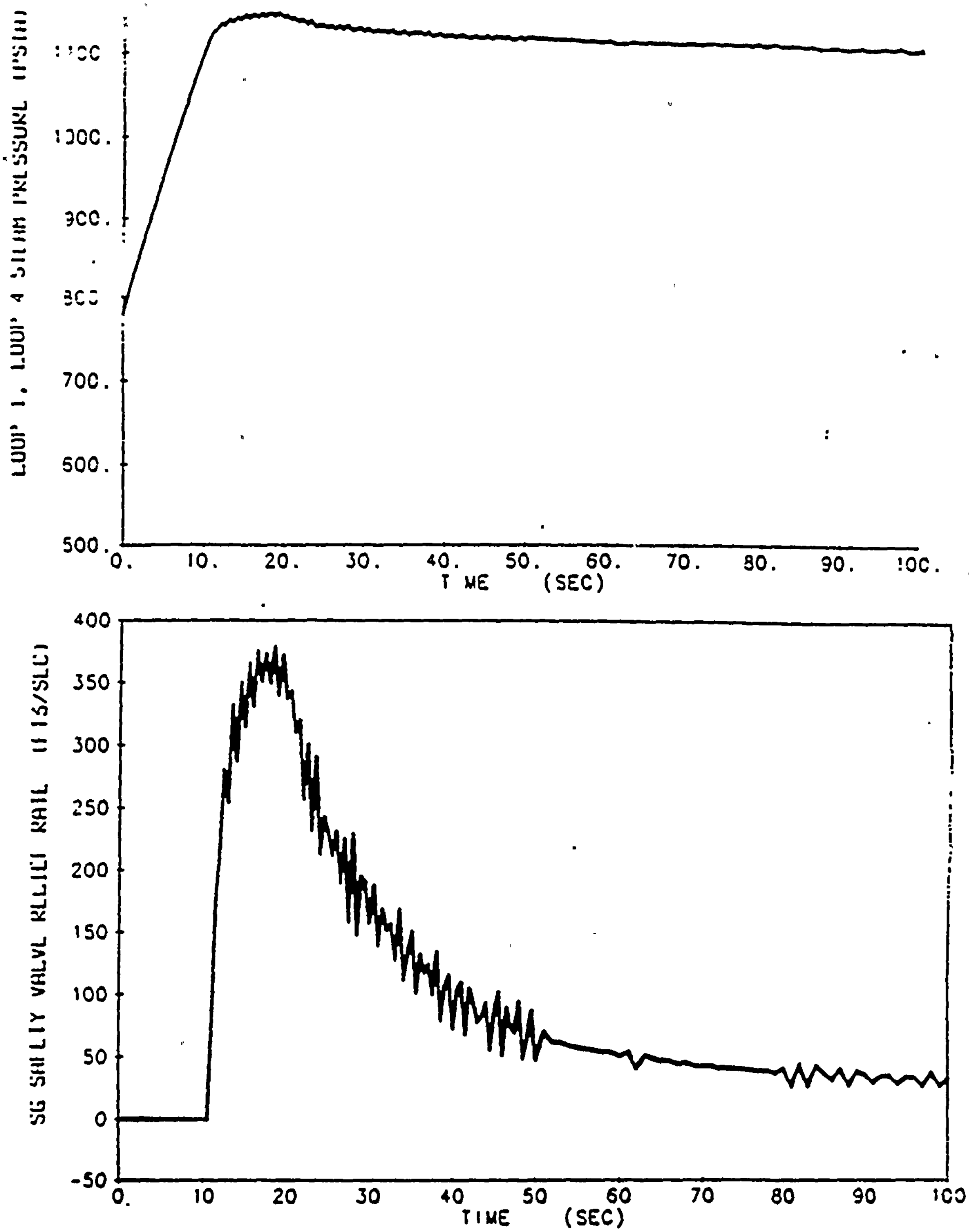
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 39b

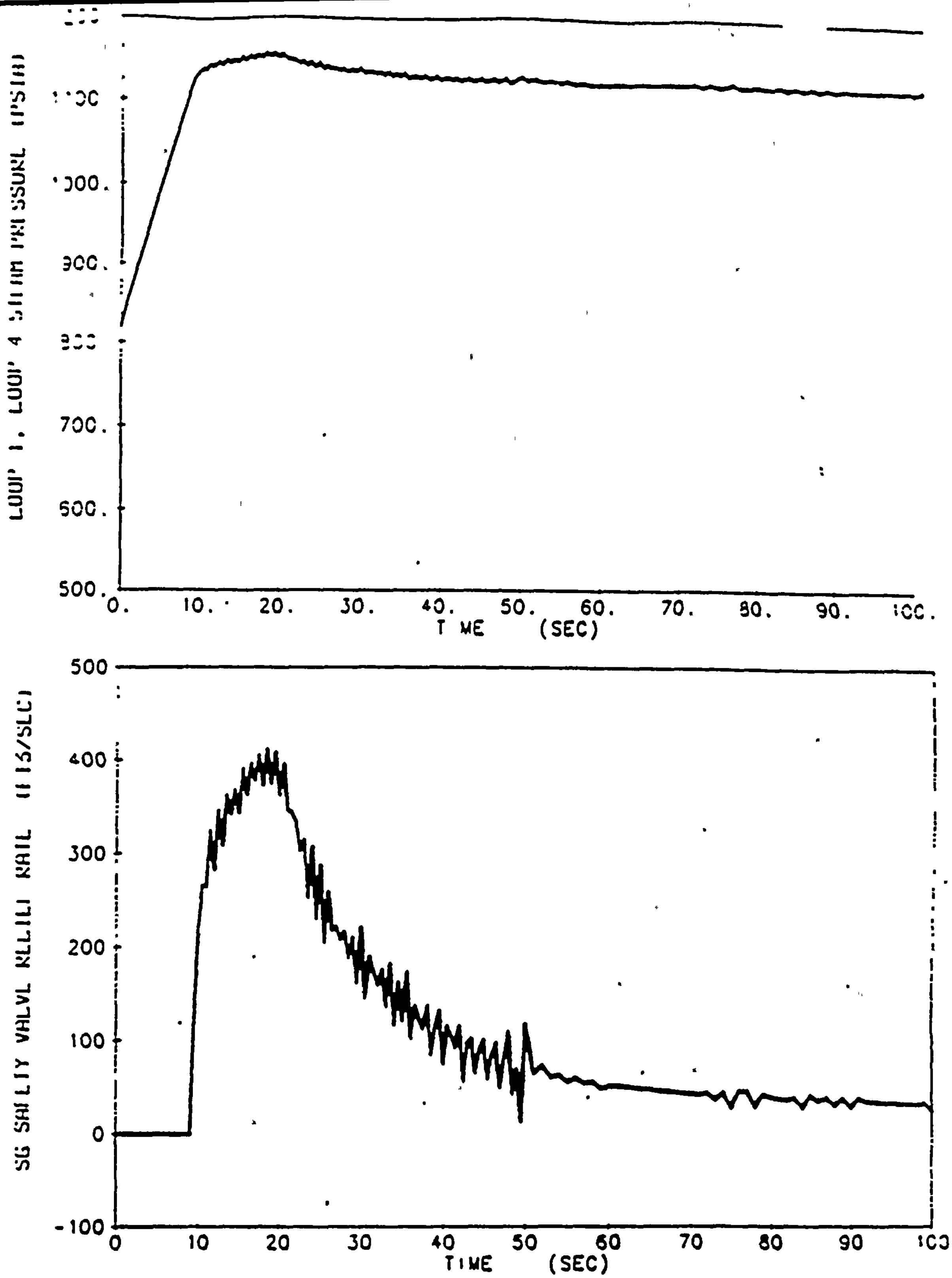
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINUMUM REACTIVITY FEEDBACK



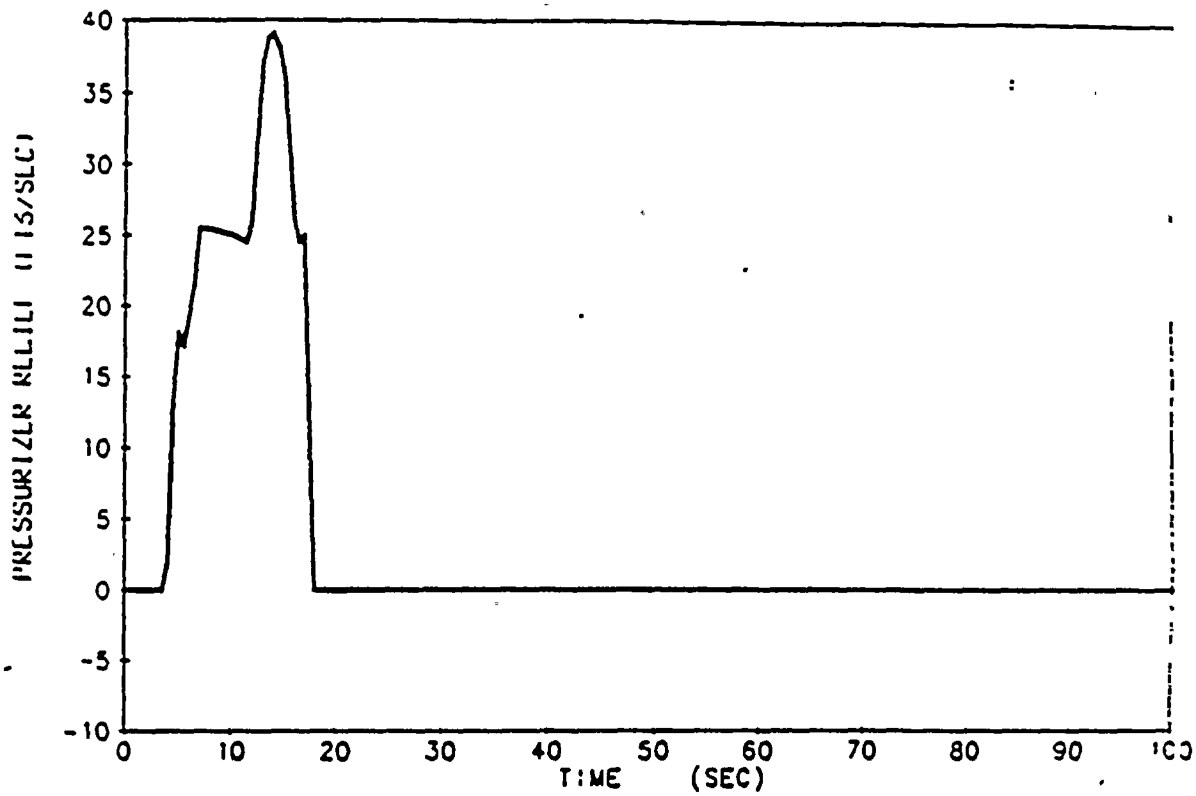
DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 40a

TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINIMUM REACTIVITY FEEDBACK



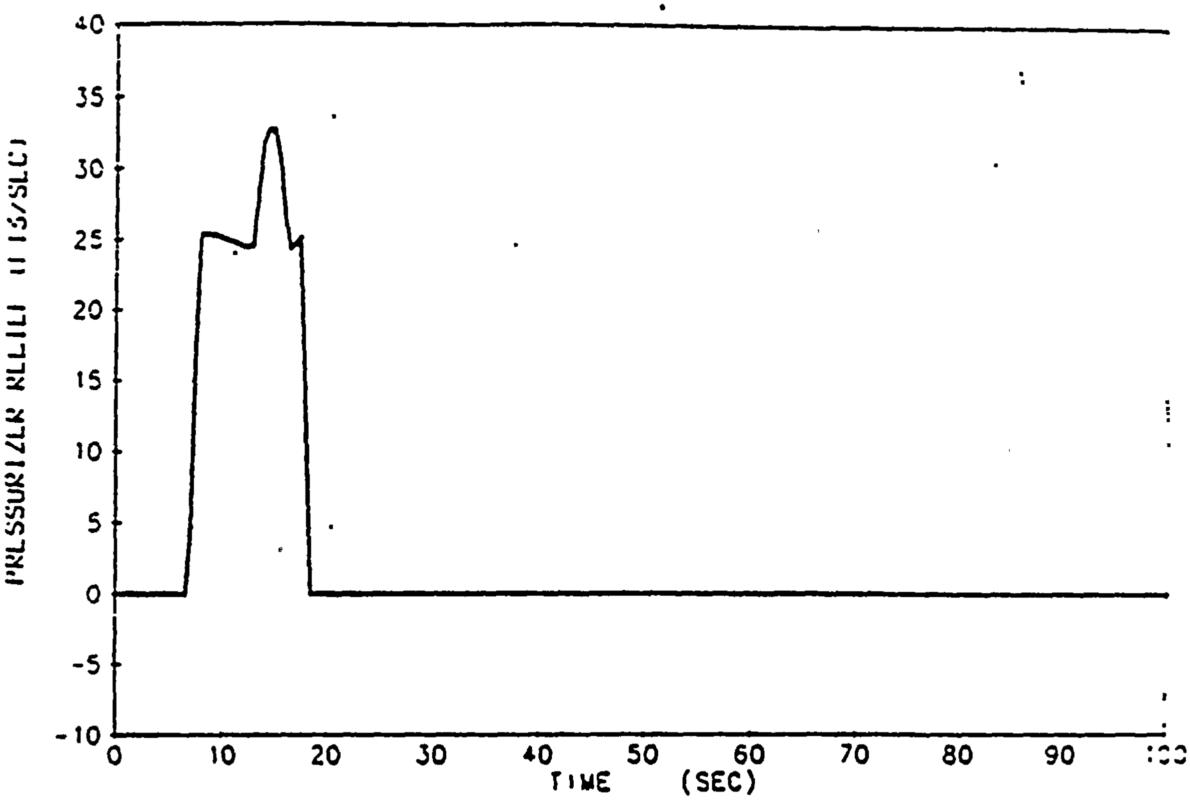
DONALD C. COOK UNIT 2 (FULL V5 CORE)
FIGURE 40b TURBINE TRIP EVENT WITH PRESSURE CONTROL, MINIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(MIXED CORE)

FIGURE 41a

TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINIMUM REACTIVITY FEEDBACK



DONALD C. COOK UNIT 2
(FULL V5 CORE)

FIGURE 41b

TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINIMUM REACTIVITY FEEDBACK

APPENDIX A

SIGNIFICANT HAZARDS EVALUATION

SIGNIFICANT HAZARDS EVALUATION

DONALD C. COOK UNITS 1 & 2 MSSV LIFT SETPOINT TOLERANCE

TECHNICAL SPECIFICATION CHANGE

INTRODUCTION:

Pursuant to 10CFR50.92, each application for amendment to an operating license must be reviewed to determine if the proposed change involves a significant hazards consideration. The Commission has provided standards for determining whether a significant hazards consideration exists (10CFR50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated, or 2) create the possibility of a new or different kind of accident from any accident previously evaluated, or 3) involve a significant reduction in a margin of safety.

DESCRIPTION Of AMENDMENT REQUEST:

The purpose of this amendment request is to revise Technical Specification Section 3/4.7 for both Donald C. Cook units in order to relax the main steam safety valve (MSSV) lift setpoint tolerance from $\pm 1\%$ to $\pm 3\%$. The currently specified tolerance of $\pm 1\%$ of the lift setpoint can be difficult to meet when the valves are tested due to setpoint drift over the duration of the operating cycle. This evaluation will provide margin for American Electric Power Service Corporation (AEPSC) when they perform their surveillance testing.

The ASME Code requires that the valves lift within 1% of the specified setpoint (NB-7512.2). The code also states that the valves must attain rated lift (i.e., full flow) within 3% of the specified setpoint (NB-7512.1). This evaluation will form the basis for taking exception to the ASME Code with respect to the lift setpoint tolerances. As defined in NB-7512.2, exceptions can be made to the code providing the effects are accounted for in the accident analyses.

BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION:

The effects of increasing the as-found lift setpoint tolerance on the main steam safety valve have been examined for the non-LOCA accidents, and it has been determined that, with one exception, the current accident analyses as presented in the UFSAR remain valid. The loss of load/turbine trip event was analyzed in order to quantify the impact of the setpoint tolerance relaxation. As previously demonstrated in this evaluation, all applicable acceptance criteria for this event have been satisfied and the conclusions presented in the UFSAR are still valid. Thus, the proposed Technical Specification change does not constitute an unreviewed safety question, and the non-LOCA accident analyses, as presented in the report, support the proposed change.

The effect of an increase in the allowable Main Steam Safety Valve set pressure tolerance from ±1% to ±3% on the UFSAR LOCA analyses has been evaluated. In each case the applicable regulatory or design limit was satisfied. Specific analyses were performed for small break LOCA assuming the current MSSV Technical Specification set pressures plus the proposed additional 3% uncertainty. The calculated peak cladding temperatures remained below the 10CFR50.46 2200°F limit.

The steam generator tube rupture event was also analyzed to determine the effects of the lift setpoint tolerance increase. The results of the analysis concluded that there was a very slight increase in the whole body dose release for Unit 1, but the magnitude of the increase was SECL-91-429, Revision 1 within the uncertainty associated with the calculation itself, and that the releases generated for the Donald C. Cook Rerating Program bound those calculated for this evaluation. The evaluation also determined that the current Unit 2 doses remain bounding. Thus, the conclusions presented in the Donald C. Cook UFSAR remain valid.

Neither the mass and energy release to the containment following a postulated loss of coolant accident (LOCA), nor the containment response following the LOCA analysis, credit the MSSV in mitigating the consequences of an accident. Therefore, changing the MSSV lift setpoint tolerances will have no impact on the containment integrity analysis. In addition, based on the conclusion of the transient analyses, the change to the MSSV tolerance will not affect the calculated steamline break mass and energy releases inside containment.

The proposed change has been evaluated in accordance with the Significant Hazards criteria of 10CFR50.92. The results of the evaluation demonstrate that the change does not involve any significant hazards as described below.

1. A significant increase in the probability or consequences of an accident previously evaluated.

   Relaxation of the MSSV setpoint tolerance from $\pm 1\%$ to $\pm 3\%$ does not increase the probability or consequences of an accident previously evaluated. Component and system performance will not be adversely affected since equipment and system design criteria continue to be met. The MSSVs do not initiate any accident not already discussed in the UFSAR. Neither the mass and energy release to the containment following a postulated loss of coolant accident (LOCA), nor the containment response following the LOCA analysis, credit the MSSV in mitigating the consequences of an accident. For the events analyzed, all applicable acceptance criteria were satisfied, and there was no increase in the doses over those previously generated. As a result, the conclusions presented in the Donald C. Cook UFSAR are unaffected by the proposed change. Therefore, changing the MSSV lift setpoint tolerances would have no impact on the consequences of an accident.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The possibility for an accident or malfunction of a different type than evaluated previously in the safety analysis report is not created. Increasing the lift setpoint tolerance on the MSSVs does not introduce a new accident initiator mechanism. No new failure modes have been defined for any system or component important to safety nor has any new limiting single failure been identified. No accident will be created that will increase the challenge to the MSSVs or result in increased actuation of the valves. Therefore, the possibility of an accident different than previously evaluated is not created.

3. Involve a significant reduction in a margin of safety.

   The margin of safety as defined in the basis of the Technical Specifications is not reduced by the change in the MSSV lift setpoint tolerance. The proposed Increase in the as-found MSSV lift

setpoint tolerance will not Invalidate the LOCA or non-LOCA conclusions presented in the UFSAR accident analyses. The new loss of load/turbine trip analysis concluded that all applicable acceptance criteria are still satisfied. For all the UFSAR non-LOCA transients, the DNB design basis, primary and secondary pressure limits, and dose limits continue to be met. Peak cladding temperatures remain below the limits specified in 10CFR50.46. The calculated doses resulting from a steam generator tube rupture event remain within a small fraction of the 10CFR100 permissible releases. Thus, there is no reduction in the margin to safety. Note, however, in order to implement the proposed change, the Technical Specifications will have to be changed.

* NO CHANGE TO THIS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Setpoint trip is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODE 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the reactor trip breakers are opened; otherwise, be in COLD SHUTDOWN within the next 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each main steam line code safety valve shall be demonstrated OPERABLE, with lift settings and orifice sizes as shown in Table 4.7-1, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87.2
2	65.4
3	43.6

* NO CHANGE TO THIS

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER		LIFT SETTING (± 3%)*	ORIFICE SIZE
a.	SV-1	1065 psig	16 in.^{2}
b.	SV-1	1065 psig	16 in.^{2}
c.	SV-2	1075 psig	16 in.^{2}
d.	SV-2	1075 psig	16 in.^{2}
e.	SV-3	1085 psig	16 in.^{2}

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. INSERT A The total relieving capacity for all valves on all of the steam lines is 17,153,800 lbs/hr which is approximately 12[illegible] percent of the total secondary steam flow of 14,120,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per operable steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER req[illegible] by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line 1, 2 or 3.

X = Total relieving capacity of all safety valves per steam line = 4,288,450 lbs/hour.

Y = Maximum relieving capacity of any one safety valve = 857,690 lbs/hour.

(109) = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation.

* NO CHANGE TO THIS PA.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-4.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODE 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the reactor trip breakers are opened; otherwise, be in COLD SHUTDOWN within the next 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87.2
2	65.4
3	43.6

* NO CHANGE TO THIS PAGE

TABLE 3.7-4

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (± $\boxed{1\%}$)* 3%	ORIFICE SIZE
a. SV-1	1065 psig	16 in.^{2}
b. SV-1	1065 psig	16 in.^{2}
c. SV-2	1075 psig	16 in.^{2}
d. SV-2	1075 psig	16 in.^{2}
e. SV-3	1085 psig	16 in.^{2}

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and INSERT A Pressure Code, 1971 Edition. The total relieving capacity of all safety valves on all of the steam lines is 17,153,800 lbs/hr which is at least 105 percent of the maximum secondary steam flow rate at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

X = total relieving capacity of all safety valves per steam line in lbs./hours = 4,288,450

Y = maximum relieving capacity of any one safety valve in lbs./hour = 857,690

109 = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

INSERT A

The safety valve is OPERABLE with a lift setting of ±3% about the nominal value. However, the safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.