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WESTINGHOUSE NUCLEAR SAFETY
SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S) : BYRON/BRAIDWOOD UNITS 1 & 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 FSAR?
(3.2) Yes ___ No X A change to procedures as described in the FSAR?
(3.3) Yes ___ No X A test or experiment not described in the FSAR?
(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 FSAR be increased?
(4.2) Yes ___ No X Will the consequences of an accident previously evaluated in the FSAR be increased?
(4.3) Yes ___ No X May the possibility of an accident which is different than any already evaluated in the FSAR be created?
(4.4) Yes ___ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.5) Yes ___ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR 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 FSAR 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 10CFR50.90.

5) REMARKS:

The following summarizes the justification based upon the written safety evaluation[1], for answers given in Part A 3.4 and Part B of this safety evaluation check list:

See Attached Evaluation

[1]Reference to documents containing written safety evaluation: ______

FOR FSAR UPDATE

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

Reason for/Description of Change:

Table 15.0-2, Table 15.0-5, Figure 15.0-1, and Section 15.2.3 were revised based on the new analyses (DT protection and LOL/TT).

6) SAFETY EVALUATION APPROVAL LADDER:

6.1) Prepared by (Nuclear Safety): B. Rarig Date: 7/22/91
B. E. Rarig

6.2) Nuclear Safety Group Manager: R. J. Sterdis Date: 7/27/91
R. J. Sterdis

BYRON/BRAIDWOOD UNITS 1 AND 2
INCREASED MAIN STEAM SAFETY VALVE SETPOINT TOLERANCE
SAFETY EVALUATION

1.0 BACKGROUND

Commonwealth Edison Company (CECo) has found that over an operating cycle, the setpoint of the Main Steam Safety Valves (MSSV) changes by more than 1% from the original set-pressure. As a result, the plant is placed in an ACTION statement and must take immediate steps to avoid a violation.

The Technical Specifications specify the setpoint at which the valves must open and the tolerance (in percent of the setpoint) within which the valves must begin to lift when calibrated and/or tested. The specified tolerance of ±1% of the setpoint, has proven to be difficult to meet when the valves are tested. Therefore, CECo has requested that Westinghouse perform an evaluation to support a relaxation in MSSV setpoint tolerances from ±1% to ±3% as defined in Technical Specification Section 3/4.7. This safety evaluation will address the effects of the ±3% tolerance on FSAR Accident analyses (non-LOCA, LOCA, SGTR), the primary component design transients, and the plant Overpressure Protection Report. The impact on the Main Steam System and the MSSVs is not within Westinghouse scope of supply and is not addressed in this evaluation.

During normal surveillance, if the valves are found to be within ±3%, they will be within the bases of the accident analyses. However, as required per Reference 4, it is strongly recommended that the valves be reset to the specified design tolerance (±1%) to prevent future accumulation of drift beyond ±3%. Resetting of the valves if the ±1% tolerance is exceeded is consistent with the existing Technical Specification requirements and the recommended Technical Specification modifications provided in Appendix D. Thus, this evaluation permits a ±3% setpoint tolerance to address as-found conditions.

The operation of the Class 2 main steam safety valves (MSSVs) is governed by the ASME Code (Reference 2). Commonwealth Edison 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. The Overpressure Protection Report (Reference 3) documents how the effects are accounted for in the accident analyses and the acceptability of the increase in the lift setpoint tolerance. 1

2.0 LICENSING BASIS

Title 10 of the Code of Federal Regulations, Section 50.59 (10 CFR 50.59) 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 Final Safety Analysis Report (FSAR). 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 Byron and Braidwood 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).

3.0 EVALUATIONS

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

3.1 Non-LOCA Evaluation

3.1.1 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 UFSAR Figure 15.0-1, 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. The results of this calculation are presented as Figure 1.

3.1.2 DNB Events

The transients identified in Table 1 are analyzed in the Byron/Braidwood 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, 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 Section 3.1.7 of this 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.

TABLE 1

DNB DESIGN BASIS TRANSIENTS

EVENT	UFSAR Section
Feedwater System Malfunction: Reduction in Temperature	15.1.1
Feedwater System Malfunction: Increase in Feedwater Flow	15.1.2
Excessive Increase in Secondary Steam Flow	15.1.3
Inadvertent Opening of a SG Relief or Safety Valve	15.1.4
Steam System Piping Failure (Double-Ended Rupture - Core Response)	15.1.5
Partial Loss of Forced Reactor Coolant Flow	15.3.1
Complete Loss of Forced Reactor Coolant Flow	15.3.2
Reactor Coolant Pump Shaft Seizure (DNB & Overpressurization Concerns)	15.3.3
Reactor Coolant Pump Shaft Break (DNB & Overpressurization Concerns)	15.3.4
Uncontrolled RCCA Bank Withdrawal From a Subcritical Condition	15.4.1
Uncontrolled RCCA Bank Withdrawal at Power	15.4.2
RCCA Misalignment	15.4.3
Inadvertent Operation of the ECCS	15.5.1
Inadvertent Opening of a Pressurizer Safety or Relief Valve	15.6.1
Startup of an Inactive Reactor Coolant Pump	15.4.4
CVCS Malfunction (Boron Dilution)	15.4.6

3.1.3 Dilution Events

The following dilution events are analyzed to demonstrate that the operators (or the automatic mitigation circuitry) have sufficient time to respond prior to reactor criticality once an alarm is generated. The secondary system is not modeled in the analysis of these events, and thus, changes to the MSSVs have no impact on these events. Therefore, the results and conclusions presented in the UFSAR remain valid.

DILUTION EVENTS	UFSAR Section
Startup of an Inactive Reactor Coolant Pump	15.4.4
CVCS Malfunction (Boron Dilution)	15.4.6

3.1.4 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 smaller break cases that might result in a heatup, based on the existing analyses one MSSV per steam generator is sufficient to provide adequate heat removal following reactor trip and is bounded by the MSSV assumption used in the current non-LOCA accident analyses. Thus, secondary pressures will be no greater than those presently calculated.

EVENT	UFSAR Section
Steamline Rupture Mass & Energy Releases Inside Containment	6.2.1.4
Steamline Rupture Mass & Energy Releases Outside Containment for Equipment Environmental Qualification	WCAP-10961-P-A

3.1.5 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/LOACP), and the core remains covered and in a coolable geometry (FLB). These transients are listed below.

EVENT	UFSAR Section
Loss of Non-Emergency AC Power to Plant Auxiliaries (LOACP)	15.2.6

Loss of Normal Feedwater (LONF)	15.2.7
Feedwater System Pipe Break (FLB)	15.2.8

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 backpressure increases, the amount of AFW delivered will be reduced. For the loss of non-emergency AC power and the loss of normal feedwater events, flow control valves in the AFW lines, designed to limit flow to a preset value, are assumed to operate since they conservatively minimize the amount of AFW available for cooling. These transients assume an AFW flow rate of 153 gpm per steam generator. If the valves were inoperable or failed during operation, they would do so in the open position resulting in higher AFW flow rates. The valves will function such that the 153 gpm accident analysis assumption will be met independent of the increase in the generator backpressure.

The feedline break event results in a faulted steam generator that depressurizes to atmospheric pressure. As a result, the AFW flow control valves are assumed to fail, minimizing the amount of AFW available for long-term cooling. This assumption results in the AFW flow being preferentially fed to the faulted steam generator where it is lost out the break. In order to ensure that some amount of AFW is supplied to the remaining intact steam generators, passive orifice plates, installed in each of the AFW lines, are used to limit the flow to the faulted loop. Since there is no method available to throttle AFW flow, the overall flow provided to the intact steam generators during a feedline break event will be reduced as the backpressure increases. Therefore, the effects of the MSSV setpoint tolerance relaxation on AFW performance during a feedline break accident must be considered.

A calculation was performed to determine the maximum steam pressure inside an intact steam generator during the long-term cooling portion of the transient (i.e., after steamline isolation occurs). The results showed that the maximum steam pressure at Byron and Braidwood is 1250 psia. Note that this value bounds both cases with and without offsite power available. Based on subsequent calculations, it was determined that the resultant AFW flow (458 gpm to the three intact steam generators) will remain greater than that currently assumed in the licensing-basis feedline break analysis (420 gpm). Therefore, the results and conclusions presented in the UFSAR (15.2.8) remain valid.

The final concern is the potential for steam generator overpressurization following reactor trip for the other long-term heatup events. Based on the existing UFSAR loss of non-emergency AC power and loss of normal feedwater analyses, long-term cooling requires a maximum of 1-3% of nominal plant steam flow from each steam generator or a plant total of 4-12% of nominal steam flow (depending on the transient). In order to pass the required flow, the two lowest set MSSVs would be required to lift. With a 3% lift tolerance, this condition would result in full open pressures for the two valves of 1249.5 and 1265.3 psia, respectively. The relief capacity of the first 2 MSSVs full open on each steam generator bounds 12% nominal steam flow. Thus, the steam flow requirement would be satisfied and resultant steam pressure of ~1265 psia would not exceed 110% of the secondary design pressure (1320 psia). As discussed above, the maximum expected pressure for a feedline break event is 1250 psia which is also less than the limit. Therefore, the proposed change does not adversely impact the long-term cooling overpressurization requirements.

Thus, based on the discussions presented above, only one UFSAR 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 remaining transients, the results and conclusions presented in the Byron/Braidwood UFSAR remain valid.

3.1.6 -3% Tolerance

Secondary steam releases are generated for the offsite dose calculations for the following non-LOCA transients: the steam system piping failure (UFSAR Table 15.1-3), the loss of external load (UFSAR Table 15.2-4), and the RCP shaft seizure (locked rotor - UFSAR Table 15.3-3). 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 (557°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 (588°F and 557°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.

The existing mass release calculations were performed using the temperatures previously identified (588°F and 557°F). Per the Byron/Braidwood Technical Specifications, the lowest set MSSV on each steam generator will open at 1190 psia (1175 psig) not including any tolerance. Based on the ASME Steam Tables (Reference 6) at saturated conditions, 557°F corresponds to 1106.4 psia and represents the lowest steam pressure considered in the mass calculations. Thus, the existing releases include a reseat pressure equal to 7% 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.

3.1.7 Analysis Summary

3.1.7.1 Loss of External Load/Turbine Trip

The loss of external load/turbine trip event is presented in Section 15.2.3 of the Byron/Braidwood 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 7). 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. Since higher steam pressures are conservative for this event, no blowdown or hysteresis behavior was assumed.

Consistent with the existing UFSAR analysis, the following assumptions were used in this analysis:

a. Initial power, temperature, and pressure were at their nominal values consistent with ITDP methodology (WCAP-8567).

b. Turbine trip was analyzed with both minimum and maximum reactivity feedback corresponding to beginning-of-life and end-of-life conditions, respectively.

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.

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 MSSVs were assumed to lift 3% above the Technical Specification setpoints and were assumed to be full open 5% above the setpoints. This is consistent with the 2% difference between lift and rated flow currently included in the code.

h. An individual MSSV was assumed to have a full flow capacity of 249 lbm/sec.

3.1.7.2 Analysis Results

Four cases 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 calculated sequence of events for the four cases is presented in Table 2.

Case A:

Figures 2 through 4 show the transient response for the turbine trip event under BOL 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 5 through 7 show the transient response for the turbine trip event under EOL conditions without pressure control. 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 8 through 10 show the transient response for the turbine trip event under EOL conditions with pressure control. The reactor is tripped on overtemperature ΔT. 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 11 through 13 show the transient response for the turbine trip event under BOL conditions with pressure control. The reactor is tripped on overtemperature ΔT. The neutron flux remains essentially constant at full power until the reactor is tripped, and 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.

3.1.7.3 Analysis Conclusions

Based on the results of these 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.

TABLE 2

TURBINE TRIP SEQUENCE OF EVENTS

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	4.3
	Rods begin to drop	6.3
	Initiation of steam release from the MSSVs	6.5
	Peak pressurizer pressure occurs	7.5
	Minimum DNBR occurs	(1)
Without pressurizer control (maximum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip setpoint reached	4.3
	Rods begin to drop	6.3
	Initiation of steam release from the MSSVs	6.5
	Peak pressurizer pressure occurs	7.0
	Minimum DNBR occurs	(1)

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

TABLE 2
(continued)

TURBINE TRIP SEQUENCE OF EVENTS

ACCIDENT	EVENT	TIME (sec)
With pressurizer control (maximum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	Initiation of steam release from the MSSVs	6.5
	Overtemperature ΔT reactor trip setpoint reached	7.4
	Peak pressurizer pressure occurs	7.5
	Rods begin to drop	9.4
	Minimum DNBR occurs	(1)
With pressurizer control (minimum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	Initiation of steam release from the MSSVs	6.5
	Overtemperature ΔT reactor trip setpoint reached	6.9
	Rods begin to drop	8.9
	Minimum DNBR occurs	10.0
	Peak pressurizer pressure occurs	10.5

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

3.1.8 Overpressure Protection Report

The Overpressure Protection Report (Reference 3) is published to demonstrate that the limiting ANS Condition II pressurization transient (loss of load/turbine trip) does not result in primary and secondary pressures in excess of 110% of the design values. The Overpressure Protection Report has been reviewed as part of this safety evaluation. In order to determine the effects of the increases in the lift setpoint tolerances, the loss of load/turbine trip transient presented in the Overpressure Protection Report was analyzed. The new analysis was performed consistent with the existing report with the exception of the explicit MSSV modeling described in the LOFTRAN description above. The results of this analysis demonstrated that the peak RCS pressure, assumed to be at the outlet of the reactor coolant pumps, was below the limit value (2750 psia).

With respect to the secondary steam system, the transient analysis resulted in approximately 60% of the total MSSV relief capacity being used. It also showed that the maximum secondary steam pressure was less than the limit (1320 psia). Thus, the conclusions presented in the Overpressure Protection Report remain valid. Changes to this report are included in this report.

3.1.9 Non-LOCA Conclusions

The effects of increasing the as-found lift setpoint tolerance on the main steam safety valve 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 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.

Changes to the UFSAR and the Overpressure Protection Report are included in this safety evaluation as appendices.

3.2 LOCA and LOCA Related Evaluations

The effects of increased tolerances for the Main Steam Safety Valve (MSSV) setpoints on the LOCA safety analyses has been previously performed for VANTAGE 5 fuel. The current Technical Specification setpoints with rated flow is given below for easy reference. The effect of either increasing or decreasing the setpoint by 3%, depending upon the direction of conservatism, has been evaluated for the LOCA analyses.

MSSV NUMBER	T/S SETPOINT	RATED FLOW	ACTUAL FLOW
MS017A,B,C,D	1190 (PSIA)	841,427	934,918

MS016A,B,C,D	1205 (PSIA)	852,039	946,710
MS015A,B,C,D	1220 (PSIA)	862,652	958,502
MS014A,B,C,D	1235 (PSIA)	873,265	970,294
MS013A,B,C,D	1250 (PSIA)	883,878	982,087

Rated flow should be used for heat-up accidents and actual flow should be used for cooldown accidents. The following presents the effect of the proposed setpoint revision from ±1% to ±3% on the LOCA-related analyses.

3.2.1 Large Break LOCA (FSAR Chapter 15.6.5)

Calculations performed to determine the response to a hypothetical large break LOCA do not model the MSSVs, since a large break LOCA is characterized by a rapid depressurization of the reactor coolant system primary below the pressure of the steam generator secondaries. Thus, the calculated consequences of a large break LOCA are not dependent upon assumptions of MSSV performance. Therefore, the large break LOCA analysis results are not adversely affected by the proposed revised MSSV setpoint tolerances.

3.2.2 Small Break LOCA (FSAR Chapter 15.6.5)

Small Break LOCAs are dependent upon heat transfer from the Reactor Coolant System (RCS) primary to the steam generator secondary in order to limit the consequences of the accident. A period exists when the RCS primary pressure hangs above the steam generator secondary pressure and excess decay heat is transferred to the steam generators. Since a loss of offsite power is assumed to occur coincident with the small break LOCA, the steam dump system and power operated relief valves are assumed to be inactive. Thus, steam relief from the steam generator secondaries takes place through the MSSVs.

The small break LOCA analyses presented in Appendix C of the Byron/Braidwood Stations Units 1 and 2 VANTAGE 5 Reload Transition Safety Report were performed using a 3% higher safety valve setpoint pressure. The standard 3% accumulation between valve actuation and full flow was also accounted for in the analyses. These analyses calculated peak cladding temperatures well below the allowed 2200^{0}F limit as specified in 10CFR50.46. demonstrating that the proposed change to the MSSV technical specification can be accommodated for small break LOCAs.

A reduction in the MSSV setpoint tolerance would act to lower the secondary pressure. Since the RCS pressure is controlled by the steam generator secondary pressure through the MSSVs, a decrease in secondary pressure would also result in a lower RCS pressure. A lower RCS pressure would result in more safety injection flow delivered to the RCS. As such, the -3% MSSV setpoint tolerance would provide increased safety injection water to the RCS, which would act to reduce the calculated peak clad temperature. Therefore, a -3% MSSV setpoint tolerance would not adversely affect the small break LOCA analysis results.

While the PCT has increased due to the revised +3% MSSV setpoint tolerance, the calculated PCT remains below 2200°F. Therefore, it is concluded that the increase in the MSSV setpoint tolerances limit to plus or minus 3 percent does not adversely affect the small break LOCA analysis results.

3.2.3 LOCA Blowdown Reactor Vessel and RCS Loop Forces (FSAR Chapter 3.9)

The licensing basis LOCA hydraulic forces analysis results found in the FSAR calculate that the peak loads occur within the first 500 milliseconds of the transient. This occurrence is well before any automatically operated safety feature has responded to the LOCA and before steam generator pressures could reach the set-pressures of the MSSVs. Therefore, changes in the MSSV Technical Specification set-pressures do not change the calculated consequences appearing in the FSAR.

3.2.4 LOCA Mass and Energy Releases for Containment Integrity Analyses (FSAR Chapter 6.2)

There is no effect due to increasing the tolerance of the steam generator Main Steam Safety Valve (MSSV) setpoints from ±1% to ±3% on short or long term LOCA mass and energy release and the resulting containment integrity response. Since a large break LOCA rapidly decreases the RCS pressure below that of the steam generator secondary pressure, the philosophy for long term LOCA considerations is to release all steam generator metal energy and primary coolant to containment. Therefore, only secondary to primary heat transfer is important in determining the amount of energy released to containment. Benefits from any mechanisms, such as MSSVs, that may possibly reduce the amount of available steam generator stored energy are small. Therefore, MSSVs are not modeled in the analysis performed to calculate the consequences for the long term design basis LOCA event.

The short term mass and energy release calculation is terminated after a few seconds. This time duration is so short as to preclude any appreciable effect due to either secondary to primary heat transfer or potential MSSV actuation.

3.2.5 Steam Generator Tube Rupture (FSAR Chapter 15.6.3)

For the steam generator tube rupture (SGTR) event, the FSAR analysis was performed to evaluate the radiological consequences. The major factors that affect the radiological doses are the amount of primary coolant transferred to the secondary side of the ruptured steam generator through the ruptured tube; the steam released from the ruptured steam generator to the atmosphere and the amount of radioactivity in the reactor coolant. The impact on these parameters of changing the main steam safety valve setpoint tolerance from ±1% to ±3% has been determined.

For the FSAR SGTR analysis, the loss of inventory due to the tube rupture results in a decrease in pressurizer pressure. Reactor trip plus SI actuation were assumed to occur on low pressurizer pressure. A loss of offsite power was also assumed to occur at the time of reactor trip, thus the steam dump system was assumed to be unavailable. The energy transfer from the primary system following reactor and turbine trip causes the secondary side pressure to increase rapidly after reactor trip until the steam generator power operated relief valves (PORVs) and/or safety valves lift to dissipate the energy. For the SGTR analysis, it was assumed that the secondary pressure is maintained at the lowest secondary safety valve setpoint following reactor trip. After reactor trip and SI initiation, the RCS pressure was assumed to reach equilibrium at the point where the incoming SI flowrate equals the outgoing break flowrate, and the equilibrium pressure and break flowrate were assumed to persist until 30 minutes after the accident. A change in the main steam safety valve setpoint tolerance to -3% will result in the secondary pressure being maintained at a lower pressure during this 30 minute period, thereby increasing the primary to secondary pressure differential. This will result in an increase to the primary to secondary break flow and the atmospheric steam release via the ruptured steam generator.

An evaluation was performed to determine the effect of decreasing the safety valve setpoint by -3% with respect the SGTR analysis in the FSAR. It is noted that this evaluation was performed in conjunction with the other changes associated with the VANTAGE-5 fuel upgrade, specifically 15% steam generator tube plugging and a hot leg temperature range of 618.4^{o}F to 600.0^{o}F. The results of the evaluation indicated that the break flow increases slightly but is still less than the conservative value reported in the FSAR for the SGTR event by approximately 2%. It is noted that the reactor coolant activity assumed for the SGTR analysis in the FSAR is based on 1% fuel defects and is assumed to be independent of the transient conditions. Therefore this assumption would not be affected by the aforementioned changes.

A radiological analysis using the revised mass releases was completed which indicates that the slight increase in the steam release is offset by the margin in the primary to secondary break flow (which exists in the FSAR report), such that the offsite radiation doses are less than the results reported in the FSAR. Therefore, it is concluded that a change in the MSSV setpoint tolerance from ±1% to ±3% will not increase the consequences of a SGTR as reported in the FSAR.

3.2.6 Hot Leg Switchover of the ECCS to Prevent Potential Boron Precipitation (FSAR Chapter 6.3.2.5)

The calculations performed to determine the time (post-LOCA) at which the boron concentration in the reactor vessel would exceed the solubility limit do not require modeling of the main steam safety valves. However, an evaluation is required to assure that adequate ECCS flow is provided to prevent boron precipitation following the switchover to hot leg recirculation. The minimum time for hot leg switchover for the Byror/Braidwood Stations was calculated to be I8 hours based on large break LOCA assumptions. The calculated core boil-off rate at 18 hours would be approximately 20 lbm/sec. The minimum ECCS flow required for delivery to the hot legs following switchover is 1.5 times the boil-off rate for a large break LOCA or approximately 30 lbm/sec. The RCS pressure for a small break LOCA at the hot leg switchover time of 18 hours can conceivably be as high as the highest steam generator safety valve setpoint (approximately 1250 psia plus 3%). Conditions for a small break LOCA differ significantly from those for a large break LOCA such that the requirements to prevent boron precipitation are much less restrictive than those for a large break LOCA. Thus, under small break LOCA conditions, ECCS flow to both the hot and cold legs can be considered in satisfying the boil-off requirement. Thus the charging and safety injection pumps must meet or exceed 30 lbs/sec at 1288 psia in order to satisfy the boil-off requirement for a small break LOCA. A review of the ECCS shows that the safety injection pumps, when aligned in the hot leg recirculation mode, can deliver more than the required 30 lbm/sec at an RCS pressure of 1288 psia. Thus, the proposed change to the MSSV Technical Specification setpoint pressure tolerance from $\pm$1% to $\pm$3% will not alter the results or conclusions appearing in the FSAR regarding the switchover of the ECCS to hot leg recirculation.

3.2.7 Post-LOCA Longterm Core Cooling (FSAR Chapter 15.6.5)

Since the post-LOCA subcriticality is based on large break requirements, deviations in MSSV set-pressures do not effect the boron concentration in the containment sump post-LOCA. Thus, the proposed change to the MSSV Technical Specification setpoint pressure tolerance from $\pm$1% to $\pm$3% will not alter the results or conclusions regarding the ability to keep the reactor cores subcritical on the boron provided by the ECCS.

3.2.8 LOCA Conclusions

The effect of a increase in the allowable Main Steam Safety Valve set pressure tolerance from $\pm$1% to $\pm$3% on the FSAR LOCA analysis 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 were well below the 10 CFR 50.46 2200°F limit.

3.3 Containment Integrity Evaluation

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 would have no impact on the containment integrity analysis. In addition, based on the conclusion of the transient analysis, the change to the MSSV tolerance will not affect the calculated steamline break mass and energy releases inside containment. Consequently, the main steam line break containment integrity analysis is not impact by the change to the MSSV setpoint tolerances.

3.4 EOP Evaluation

In the Emergency Operating Procedures (EOPs), the MSSV setpoint pressures are used to determine when to trip the reactor coolant pumps (RCPs). The determination is conservative, taking into account instrument uncertainties. The conservatism, along with the small difference between the MSSV pressure used to determine the RCP trip setpoint fr the EOPs and the in-plant first lift pressures of less than 5.6% leads to the conclusion that there is no significant impact on the EOPs in this area.

The MSSV pressures are also used in the EOPs on the heat sink status tree in determining which heat sink EOP is appropriate for implementation. These pressures are only involved in optional or yellow paths on the heat sink status tree. This means that the plant condition is such that the operator is not required to perform the heat sink EOPs called for by these yellow paths. Consequently, an inappropriate transition to these procedures would not cause the operator to forego an action required to maintain the plant in a safe condition. Thus, the variations found between the EOP MSSV setpoints and the MSSV in-plant lift pressures have negligible impact on the EOPs in this area. If the set pressures are within ±5%, use of these procedures will ensure that the secondary pressure remains within acceptable limits.

4.0 DETERMINATION OF UNREVIEWED SAFETY QUESTION

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

   The ±3% tolerance on the MSSV setpoint does not increase the probability of an accident previously evaluated in the FSAR. There are no hardware modifications to the valves. Therefore, there is not an increase in the spurious opening of a MSSV. The MSSVs are actuated after an accident is initiated to protect the secondary systems from overpressurization. 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 FSAR would not be increased as a result of increasing the MSSV lift setpoint tolerance by 3% above or below the current Technical Specification value.

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

   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 FSAR value by approximately 2%, the SGTR analysis indicates that the calculated break flow is still less than the value reported in the FSAR. Therefore, the radiological analysis indicates that the slight increase in the steam release is offset by the decrease in the break flow such that the offsite radiation doses are less than those reported in the FSAR. 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 releases.

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

   The $\pm 3\%$ tolerance on the MSSV setpoint does not create the possibility of an accident which is different than any already evaluated in the FSAR. 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 and result in increased actuation of the valves. Therefore, the possibility of an accident different than any already evaluated is not created.

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

   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.

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, there is no possibility of increasing the dose releases 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.

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 and 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 release limits continue to be met. Peak cladding temperatures remain well below the limits specified in 10CFR50.46. Thus, there is no reduction in the margin to safety.

5.0 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 FSAR remain valid.

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

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 10 CFR 50.59.

6.0 REFERENCES

1) Byron/Braidwood Technical Specifications through Amendments 37 and 23, respectively.

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

3) CAW-3581/CBW-3009,"Commonwealth Edison Company, Byron and Braidwood Stations - Units 1 and 2 Overpressure Protection Report," July 1981.

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

5) "Byron/Braidwood Stations Units 1 & 2 Updated Final Safety Analysis Report (UFSAR), Docket Numbers 50-454, 455, 456, and 457, December 1989.

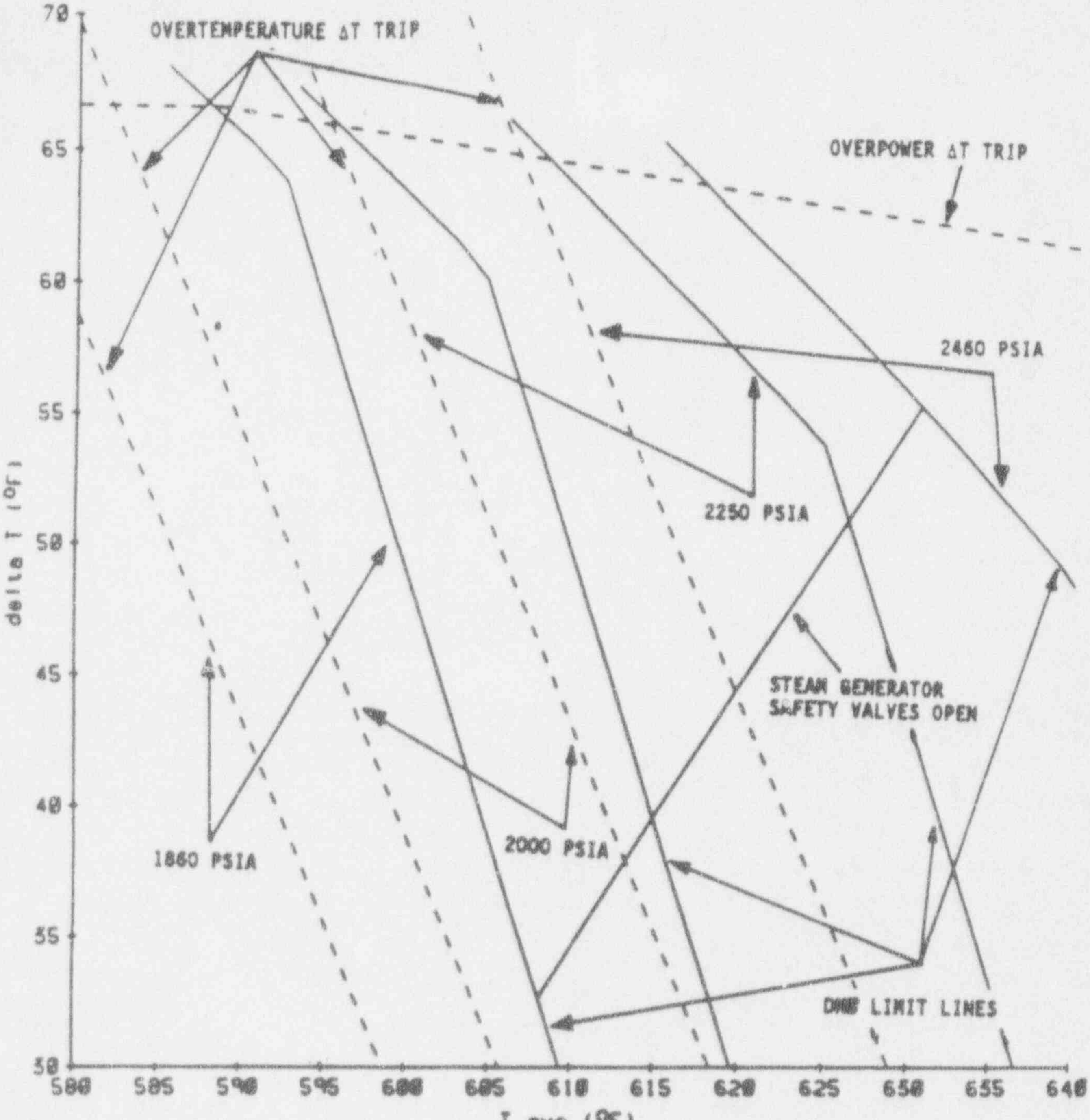
6) ASME Steam Tables, Fifth Edition, 1983.

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

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

9) DiTommaso, S.D. et al.,"Byron/Braidwood Thot Reduction Final Licensing Report," WCAP-11386-P, Revision 2, November 1987.

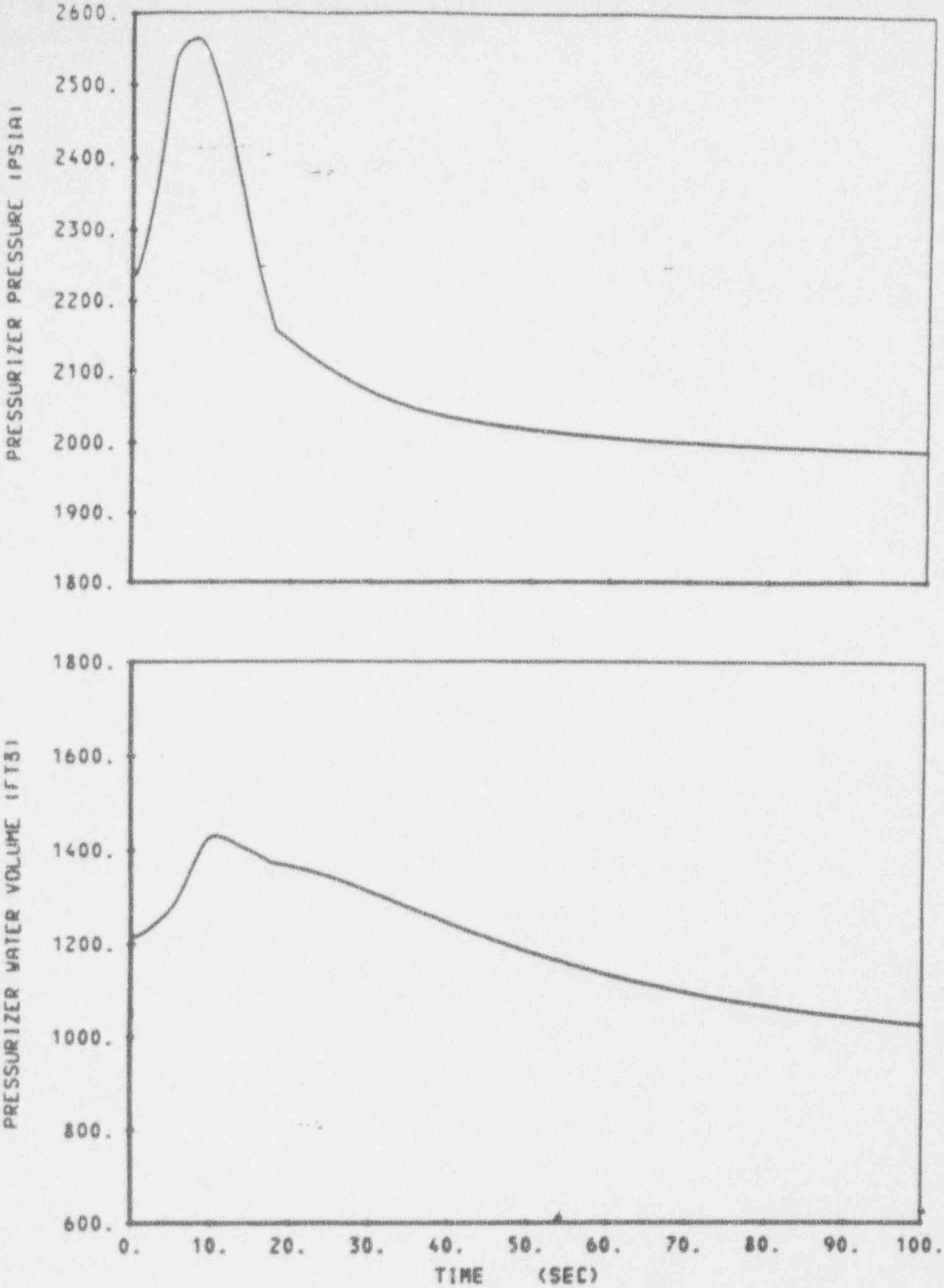
10) Butler, J.C. and D.S. Love,"Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," WCAP-10961-P, October 1985.



BYRON/BRAIDWOOD STATIONS

FIGURE 1

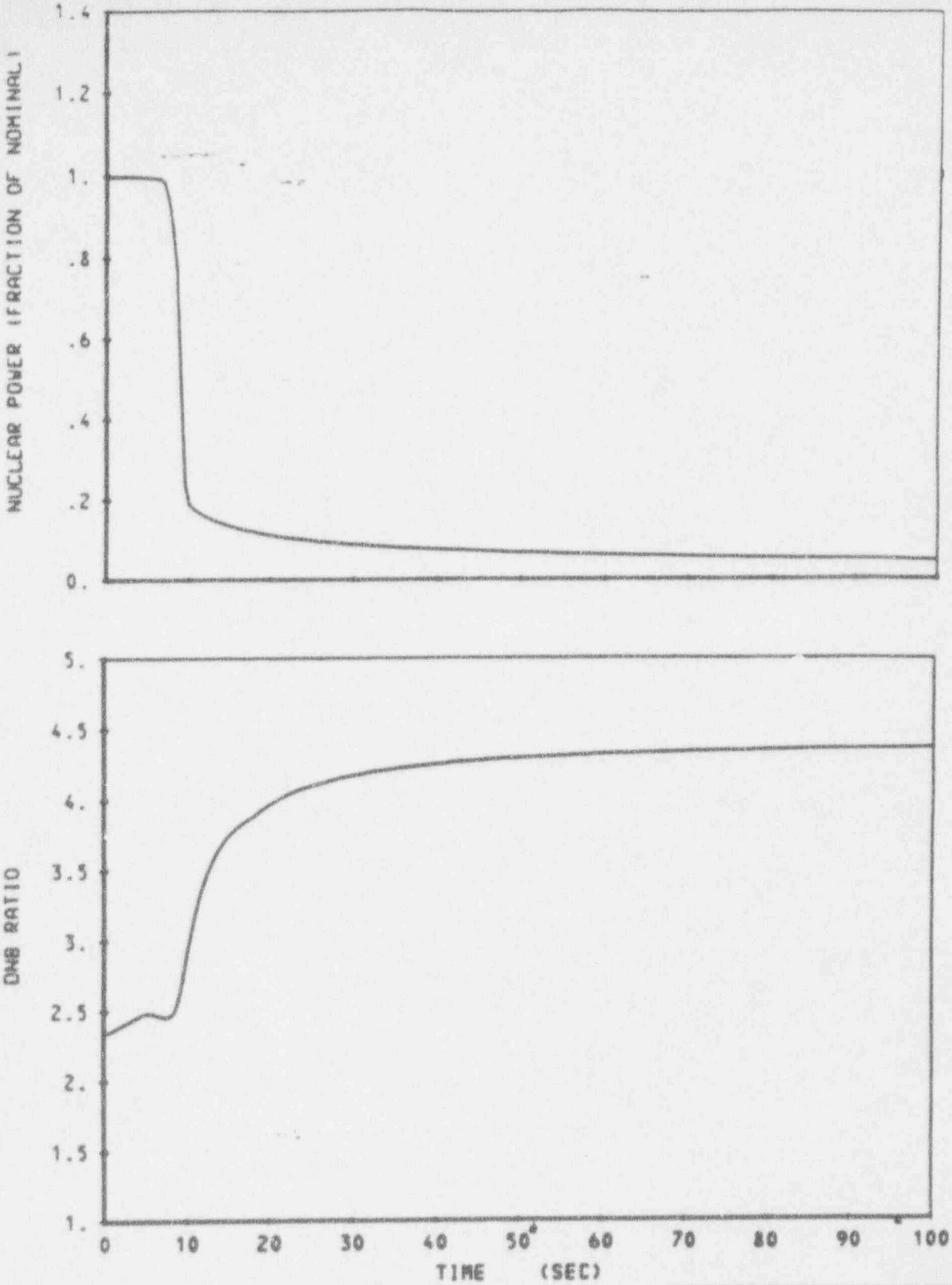
ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER DELTA T PROTECTION



BYRON/BRAIDWOOD STATIONS

FIGURE 2

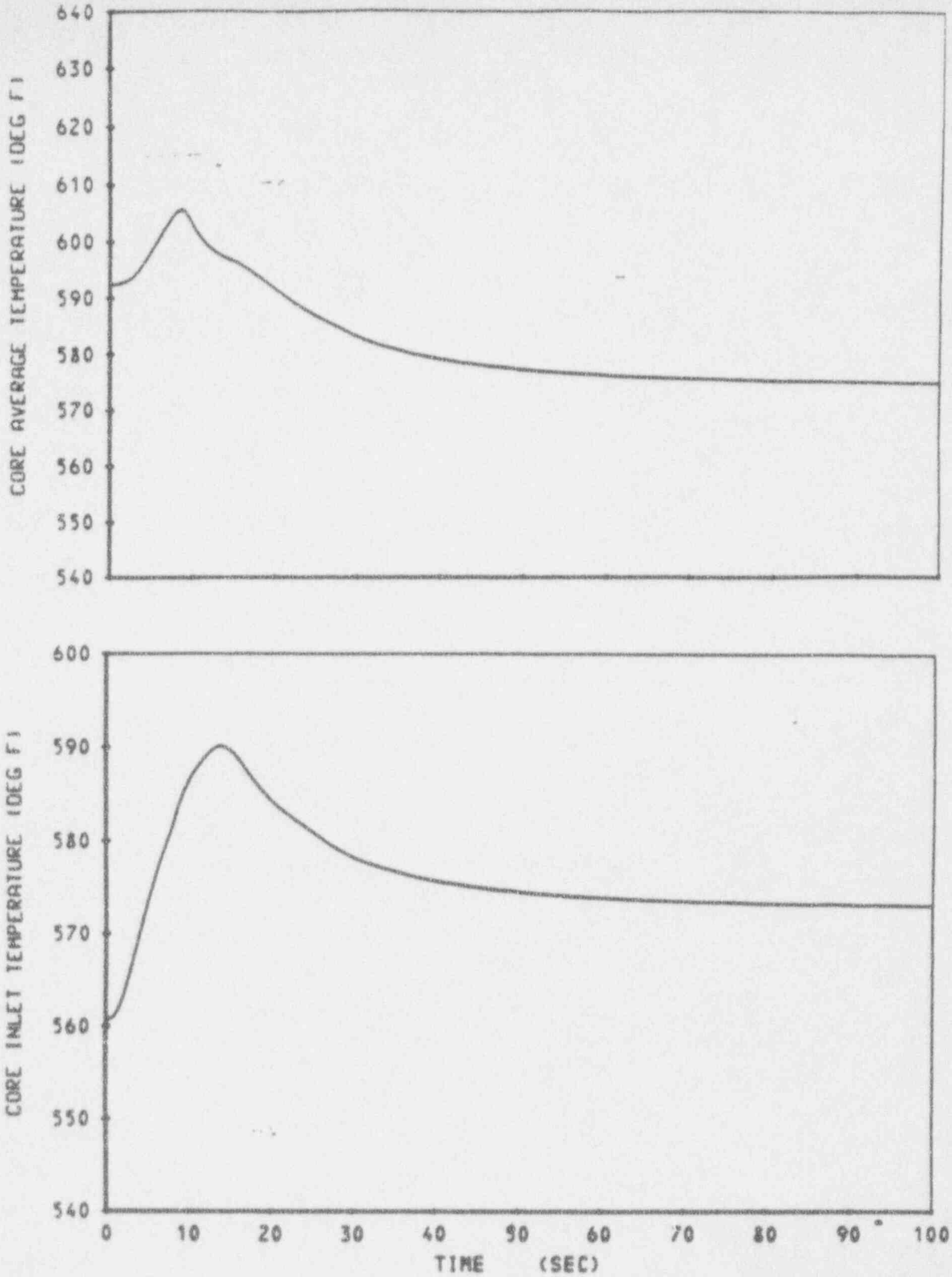
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MINIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 3

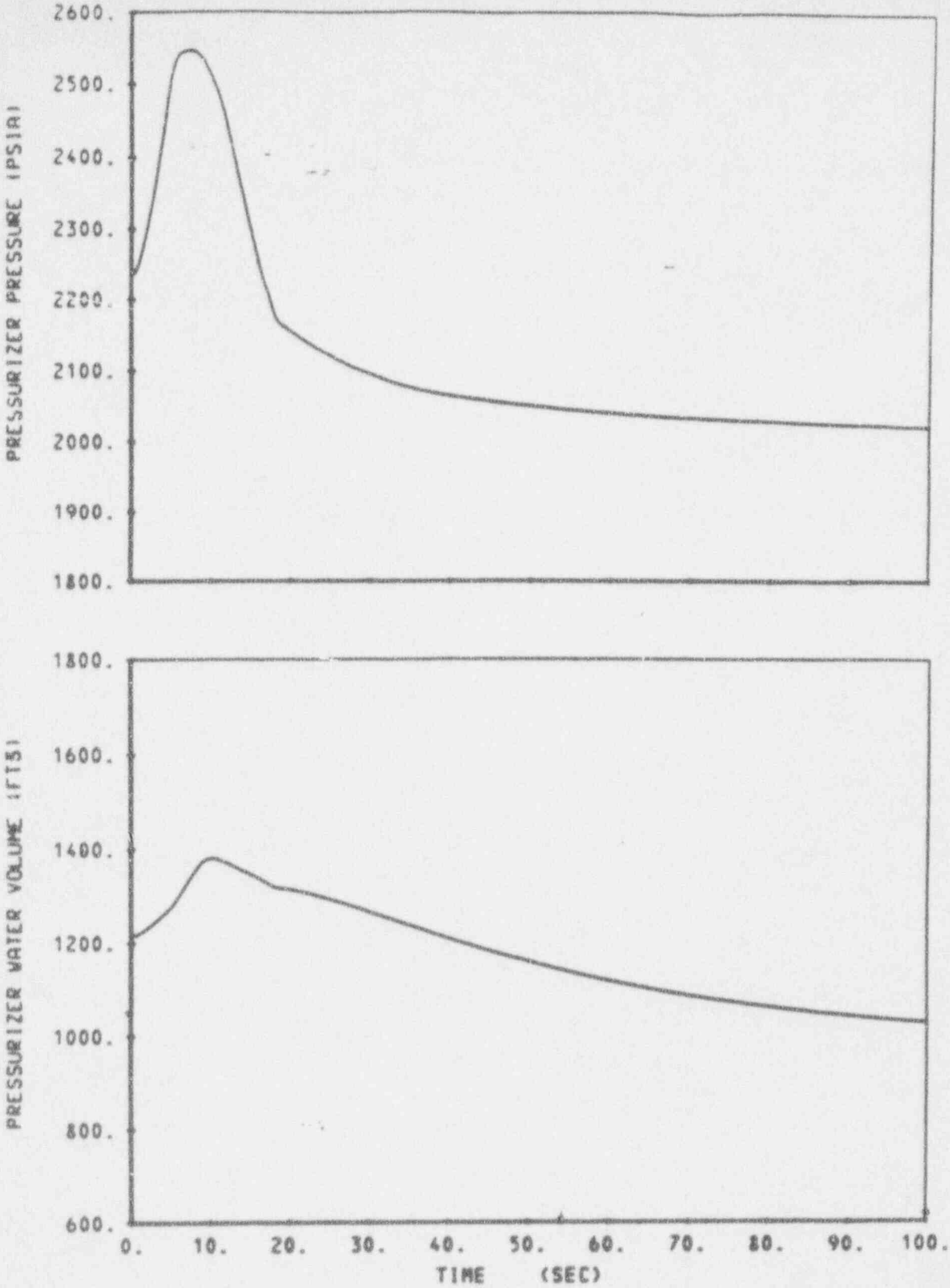
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MINIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 4

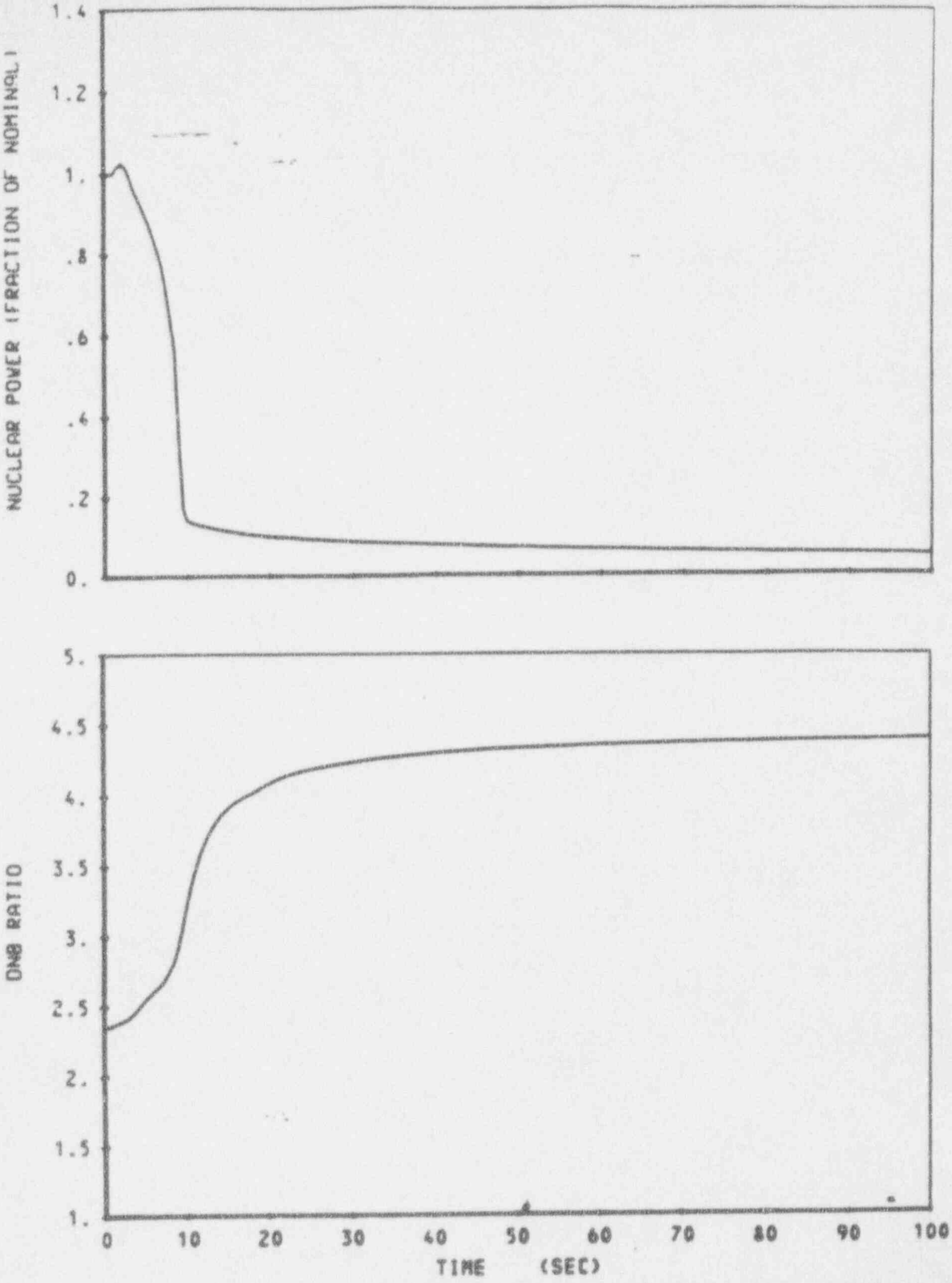
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MINIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 5

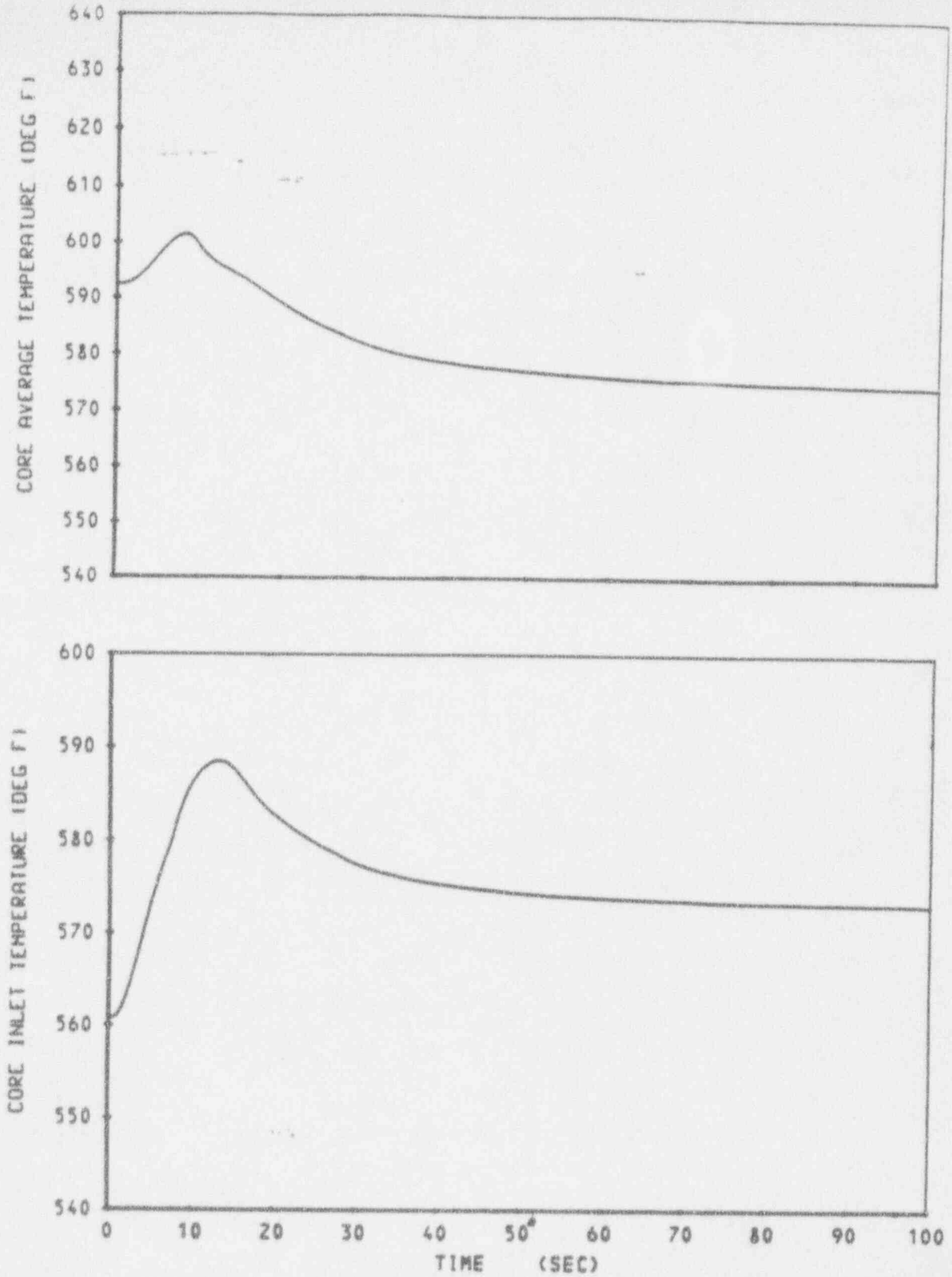
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 6

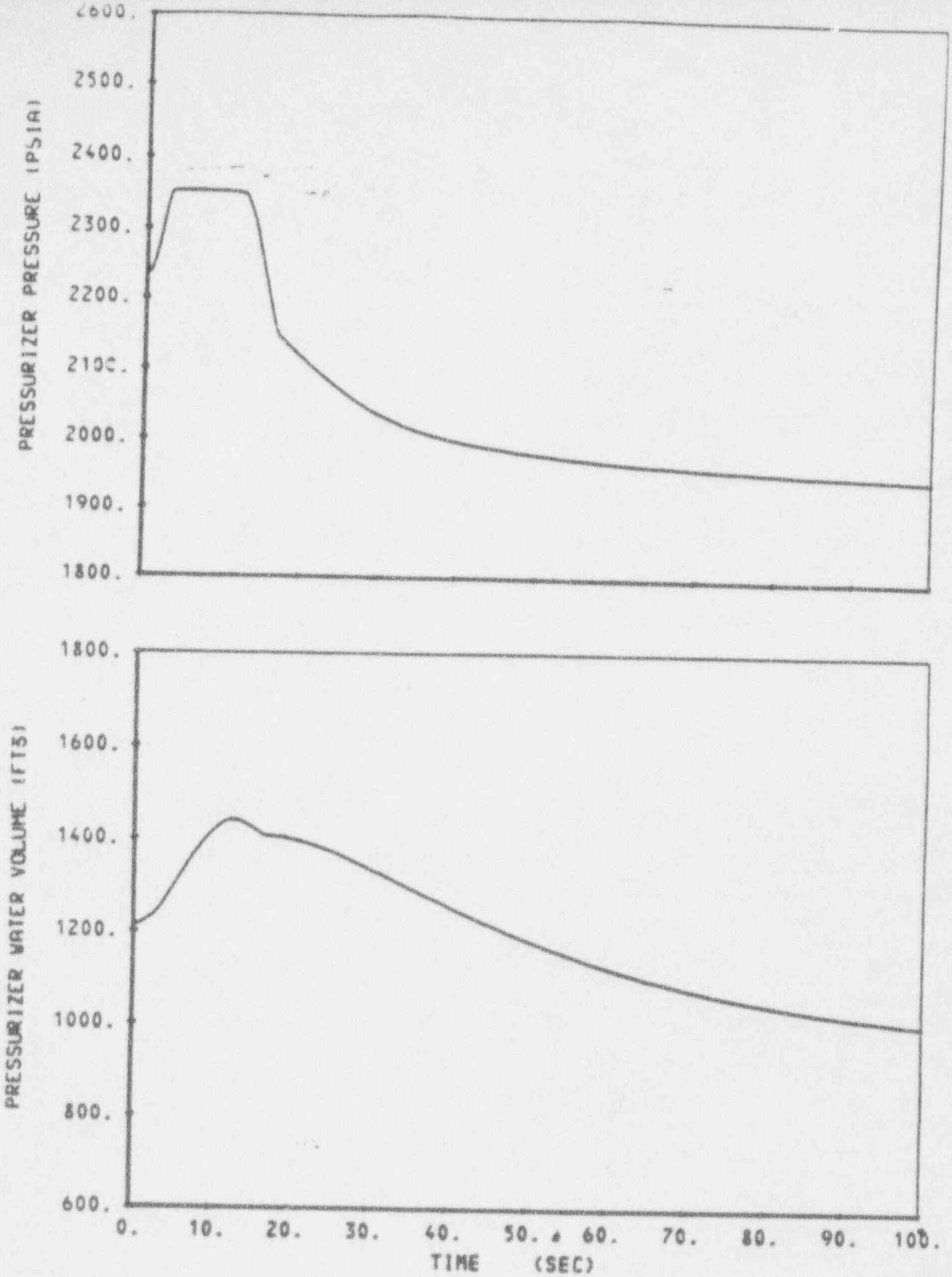
TURBINE TRIP EVENT WITHOUT
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 7

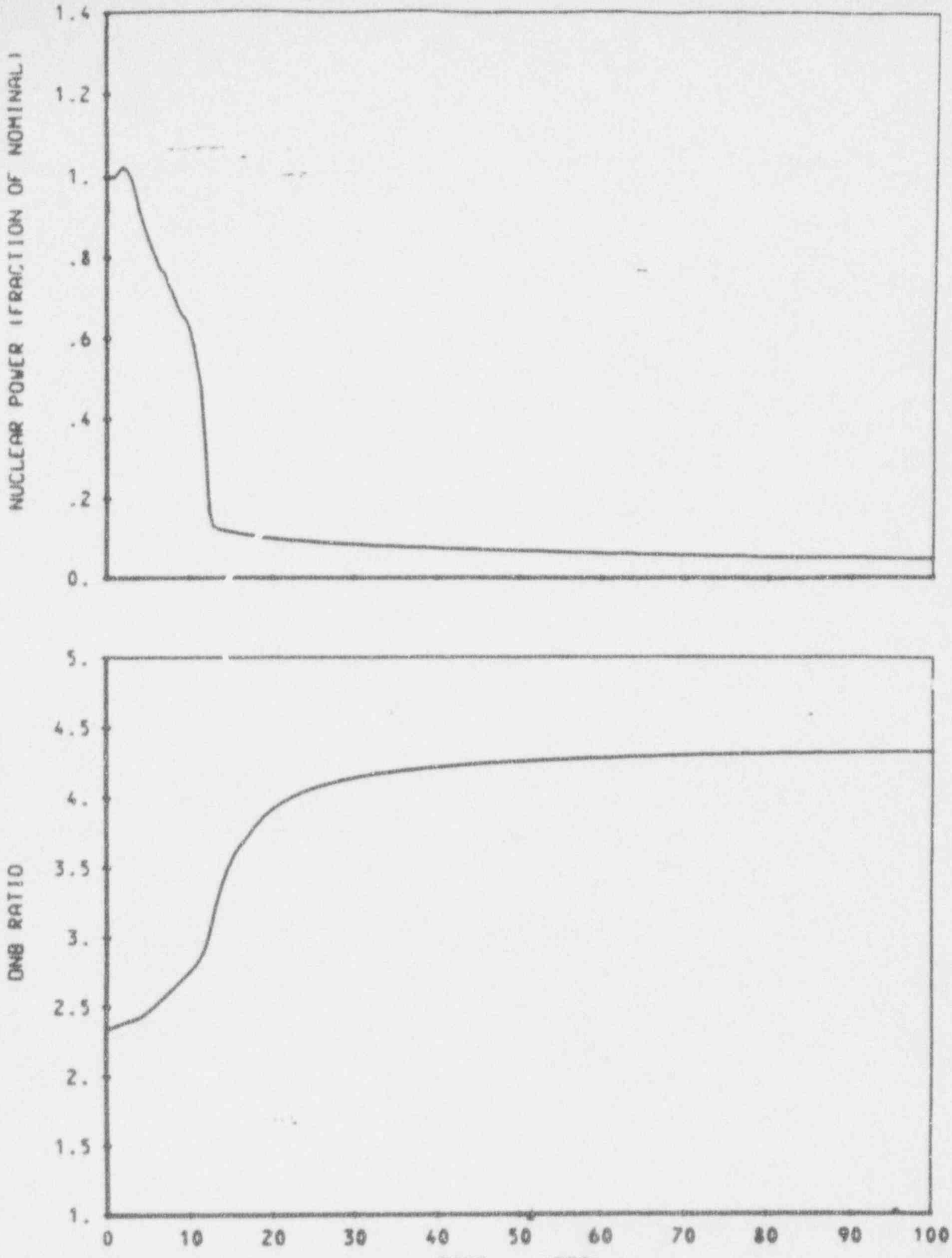
TURBINE TRIP EVENT WITHOUT PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 8

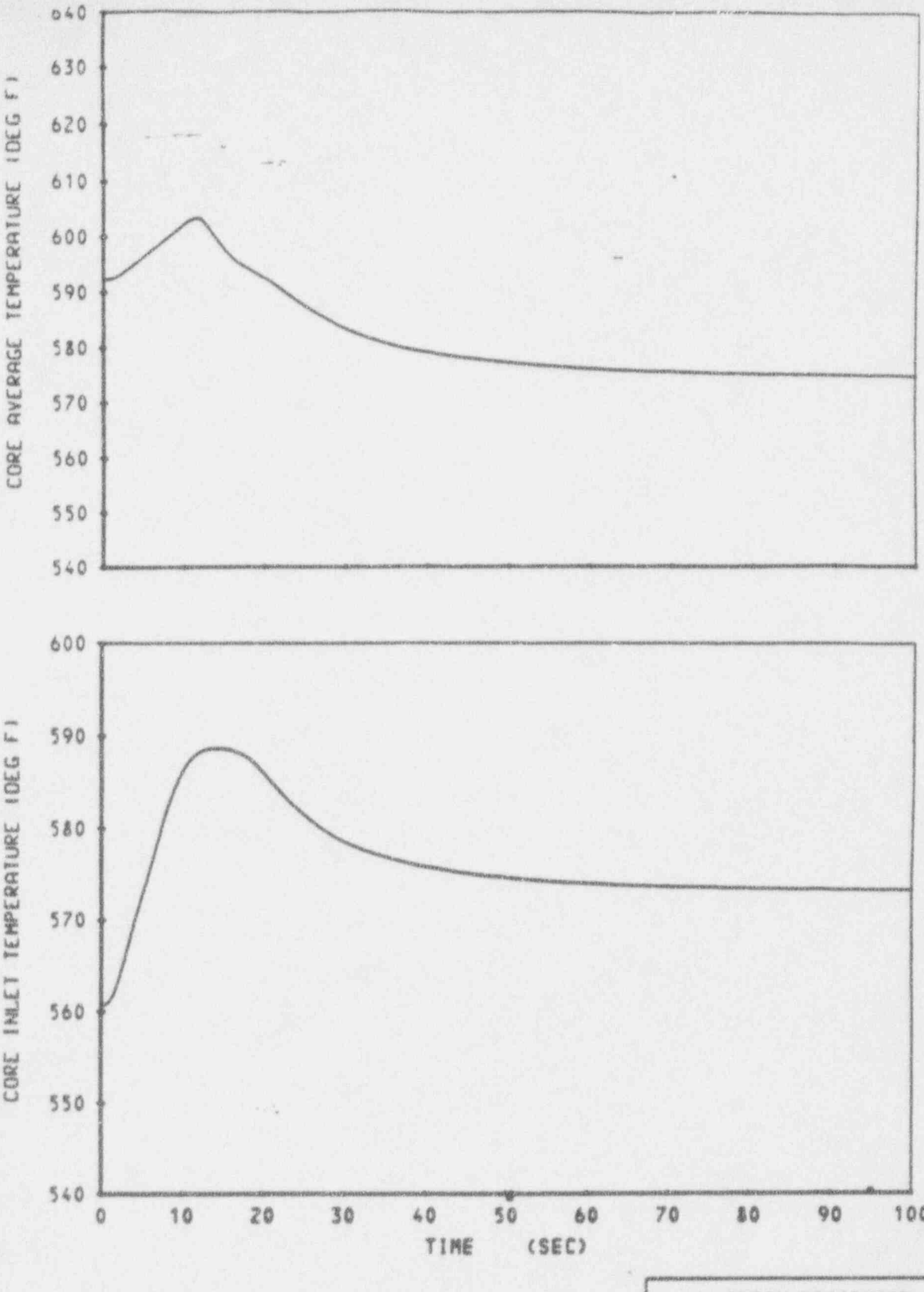
TURBIME TRIP EVENT WITH
PRESSURE CONTROL,
MAXIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 9

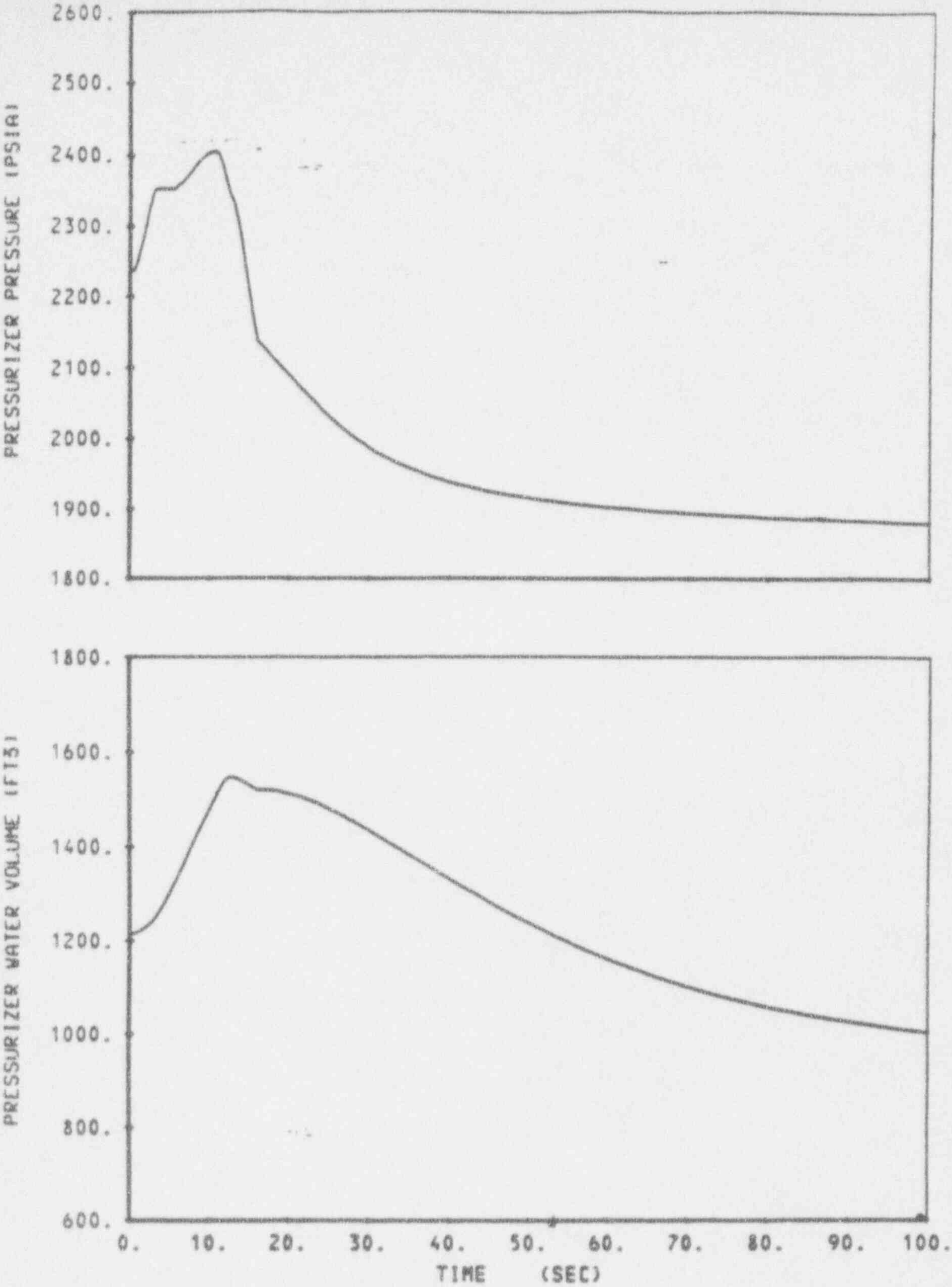
TURBINE TRIP EVENT WITH PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 10

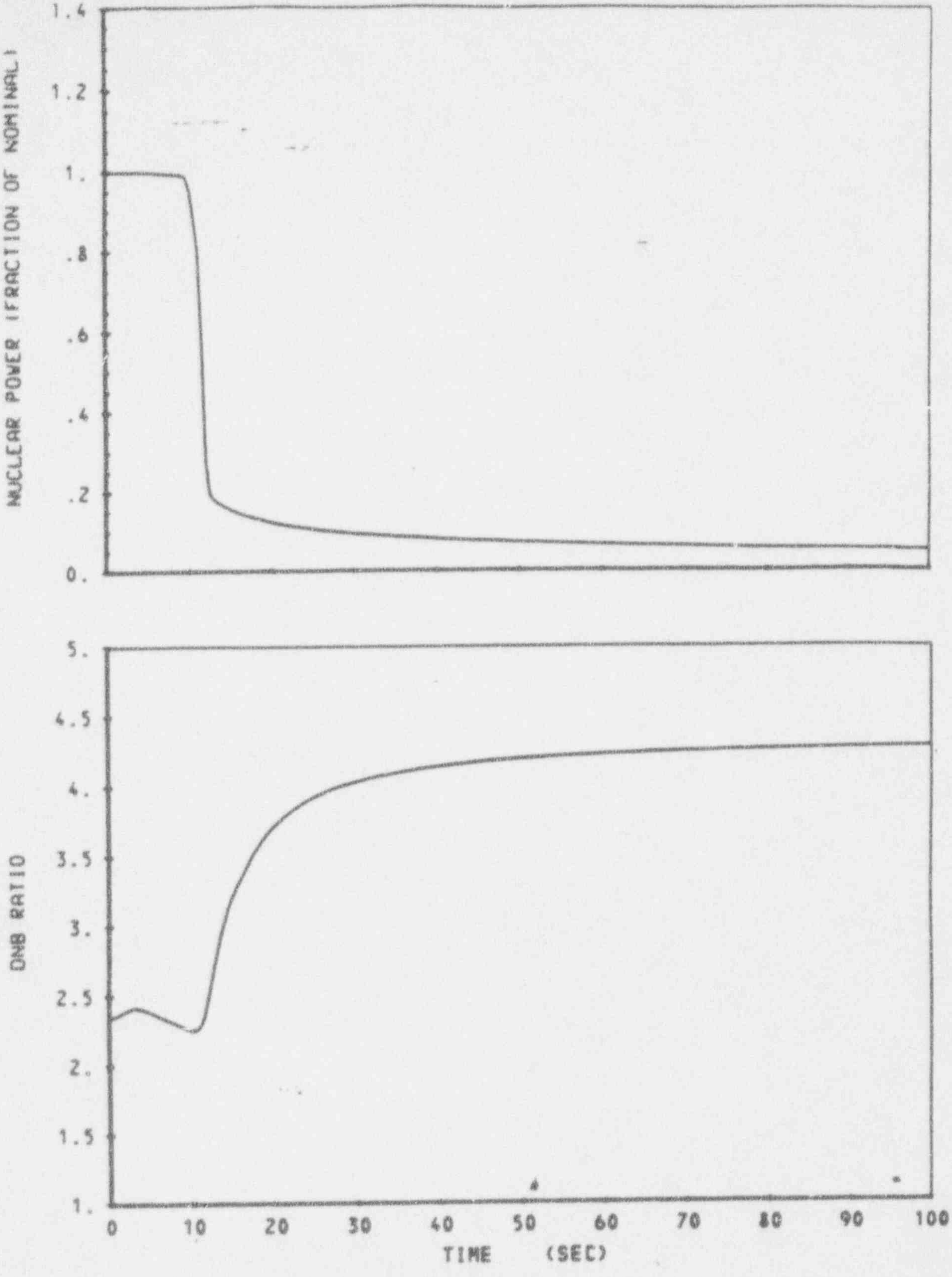
TURBINE TRIP EVENT WITH PRESSURE CONTROL, MAXIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 11

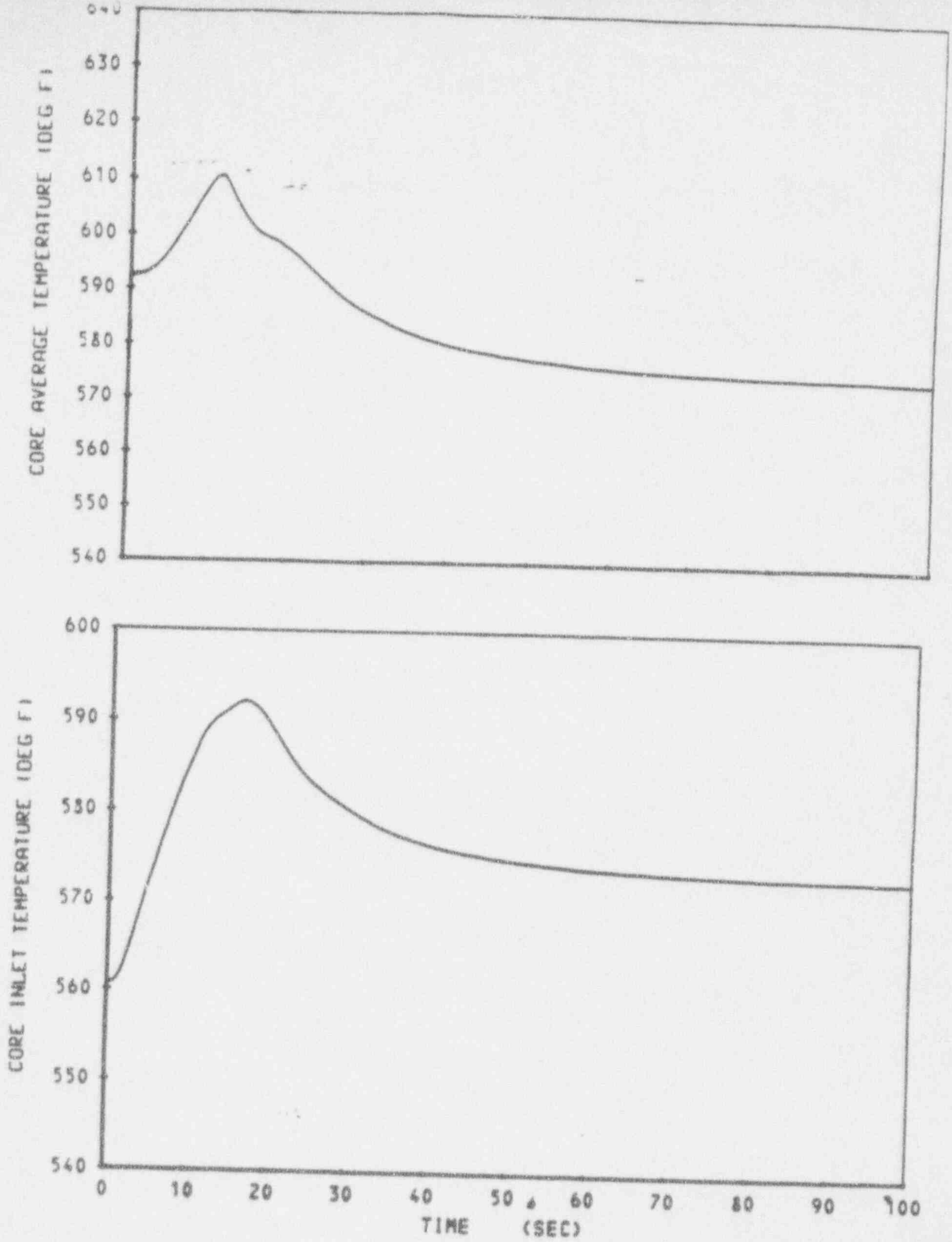
TURBINE TRIP EVENT WITH
PRESSURE CONTROL,
MINIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 12

TURBINE TRIP EVENT WITH PRESSURE CONTROL, MINIMUM REACTIVITY FEEDBACK



BYRON/BRAIDWOOD STATIONS

FIGURE 13

TURBINE TRIP EVENT WITH PRESSURE CONTROL, MINIMUM REACTIVITY FEEDBACK

Westinghouse Electric Corporation
Energy Systems

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CAE-90-315
CCE-90-310
NS-OPLS-OPL-I-90-619
October 31, 1990

Commonwealth Edison Company
Byron/Braidwood Units 1 and 2
Relaxation of MSSV Setpoint Tolerance to +/-3%

Dear Mr. Pleniewicz:

Enclosed please find the safety evaluation performed by Westinghouse in accordance with the criteria of 10 CFR 50.59 which concludes that the proposed relaxation of main steam safety valve lift setpoint tolerances from ±1% to ±3% does not involve an unreviewed safety question.

The recommended Technical Specification revisions are provided as Appendix D to the safety evaluation. A significant hazards evaluation to support the recommended changes is included as Appendix C.

Westinghouse recommends resetting the valves to within the design tolerance of ±1% consistent with ASME Code, Section 11 requirements. In order to relax this requirement, the valve manufacturer must concur with the change to the design basis tolerance specified for the valves in question.

This submittal represents the final version of the safety evaluation and supporting documentation. However, Westinghouse will resolve any additional comments or questions from Commonwealth Edison concerning the enclosed evaluation.

Please take a few minutes to complete the enclosed Quality Survey Form. If you have any questions, do not hesitate to call.

Very truly yours,

TF Woodhouse for

G. P. Toth, Manager
Commonwealth Edison Projects
Customer Projects Department

B. E. Rarig
Enclosure