Attachment E
PG&E Letter DCL-95-220

RELAXATION OF MAIN STEAM LINE SAFETY VALVE PRESSURE SETPOINT TOLERANCE FOR THE DIABLO CANYON NUCLEAR POWER PLANT UNITS 1 AND 2

TABLE OF CONTENTS

I. INTRODUCTION AND OVERVIEW

This report summarizes an analysis performed to justify an increase in the allowed tolerance for the main steam line safety valve setpoints. The present DCPP TS require that surveillance tests be performed to assure that all main steam line safety valves are operable and lift within +/-1 percent of the nominal setpoint pressure. This report will justify an increase of this tolerance to -3/+3 percent, except for the bank of main steam line safety valves with the lowest set pressure (RV-3, RV-7, RV-11, RV-58) which will be changed to -2/+3 percent. The proposed tolerance relaxation meets the NRC Standard Review Plan for Overpressure Protection and satisfies ASME Code requirements. In addition, this change has been shown to satisfy the acceptance criteria for all of the FSAR accidents, and results in no additional operational concerns such as excessive valve leakage problems.

This report first discusses the acceptance criteria that will be applied in justifying this tolerance relaxation. The RETRAN model is described, as is a benchmark of the model using plant data. The report then discusses the conservative assumptions used in the RETRAN calculation and compares RETRAN results to the FSAR Turbine Trip analysis as analyzed by Westinghouse using the LOFTRAN code. Results are presented for reanalysis of the Turbine Trip Event with increased main steam line safety valve setpoint tolerances. ASME Code and operational issues are discussed, followed by the conclusions and recommendations. Also provided is a discussion of the impact on other FSAR Chapter 15 accidents and a discussion of the RETRAN Safety Evaluation Report (SER).

The RETRAN input model used in this application has been independently reviewed by EI International, Inc.. In addition, Westinghouse performed an overall safety evaluation of the proposed tolerance change. This evaluation has been summarized in Section VII.

II. EVENT SELECTION AND ACCEPTANCE CRITERIA

NUREG-0800, USNRC Standard Review Plan Section 5.2.2, "Overpressure Protection," specifies that safety valves be designed with sufficient capacity to limit reactor coolant system (RCS) and main steam system (MSS) pressures to less than 110 percent of the design value. The design RCS pressure is 2500 psia and the design MSS pressure is 1100 psia. According to the Standard Review Plan, this can be satisfied by demonstrating, by analysis, that the RCS and MSS pressures for the most limiting abnormal operational transient (ASME Condition II events) do not exceed 110 percent of design value. In addition, the predicted peak pressures in all pressure build-up transients analyzed in Chapter 15 of the FSAR Update are required to be reviewed to assure that the existing FSAR Update conclusions remain valid.

There are five Condition II events which result in pressure increase of the RCS and MSS. They are:

(1) uncontrolled rod withdrawal from full power.

(2) loss of reactor coolant flow.

(3) loss of external electrical load/turbine trip.

(4) loss of normal feedwater.

(5) loss of all AC power to the station auxiliaries.

According to the FSAR Update, safety valve actuation is required to limit the system pressures in three of the five cases: loss of external electrical load/turbine trip, loss of normal feedwater, and loss of all AC power to the station auxiliaries. Of the three cases requiring safety valve actuation, the loss of external electrical load/turbine trip has the largest pressure increase and, therefore, is limiting and is the case requiring reanalysis.

In addition to the Standard Review Plan requirements, the change in main steam line safety valve tolerance is also reviewed against ASME Code and operational requirements, and is further shown to meet the appropriate acceptance criteria for all of the FSAR transients. Table II-1 shows the DCPP safety valve identification numbers, nominal setpoints and nominal flows.

TABLE II-1

DCPP SAFETY VALVE AND PORV DESIGN PRESSURES

Pressurizer Safety Valves	Nominal Setpoint (psig)	Nominal Flow (lbm/hr) Per Valve
8010A, 8010B, 8010C	2485	420,000

Main Steam Line Code Safety Valves				Nominal Setpoint (psig)	Nominal Flow (lbm/hr) Per Valve
Line 1	Line 2	Line 3	Line 4		
RV-3	RV-7	RV-11	RV-58	1065	803,789
RV-4	RV-8	RV-12	RV-59	1078	813,471
RV-5	RV-9	RV-13	RV-60	1090	822,408
RV-6	RV-10	RV-14	RV-61	1103	832,090
RV-222	RV-223	RV-224	RV-225	1115	841,021

Pressurizer PORVs	Nominal Setpoint (psig)	Nominal Flow (lbm/hr) Per Valve
PCV-455C, PCV-456, PCV-474	2335	210,000

Main Steam Line PORVs				Nominal Setpoint (psig)	Nominal Flow (lbm/hr) Per Valve
Line 1	Line 2	Line 3	Line 4		
PCV-19	PCV-20	PCV-21	PCV-22	1035	440,000

III. RETRAN MODEL DESCRIPTION

DESCRIPTION OF THE RETRAN CODE AND MODEL

RETRAN Code

The RETRAN Code is a best estimate thermal hydraulic code which can be used to perform system transient analysis. The RETRAN code solves equations that conserve mass, energy, and momentum. RETRAN02/MOD004 was used for the analysis presented in this report. This RETRAN version has been used by other utilities in licensing submittals.

The NRC staff has reviewed the RETRAN02/MOD004 computer code and issued a Safety Evaluation Report (letter, A. Thadani, NRR, to R. Furia, GPU Nuclear Corporation, dated October 19, 1988, "Acceptance for Referencing Topical Report EPRI-NP-1850 CCM-A, Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004"). In the SER, the NRC staff concluded that RETRAN02/MOD004 was acceptable for the use in transient analyses, provided that a series of restrictions were evaluated and shown not to apply for the analysis being performed. PG&E has reviewed these restrictions and determined that they either did not apply or were handled by appropriate conservatisms in the analysis. A point-by-point response to the restrictions is given in Attachment C.1.

RETRAN Model

The PG&E RETRAN model is displayed in Figure III-1. Figure III-2 is a schematic of the seventeen volume steam generator model. As may be seen from the figures, the PG&E RETRAN model uses two loops to simulate the four loop Diablo Canyon plant. One of the two loops in the RETRAN model is sized to represent the performance of three actual loops and is termed the "lumped" loop. The other loop is sized exactly as an actual flow loop and is called the "single" loop. A large volume simulating the containment is attached to the single loop and pressurizer.

The components in the lumped loop are sized so as to preserve the mass flow, flow velocities and fluid transient times of the actual plant loops. Thus the flow areas and fluid volumes are equal to three times that of a single loop, but the pipe hydraulic diameters and lengths are equal to the actual dimensions. Actual elevations are preserved in the RETRAN model on both flow loops. All control volumes and junctions on the single loop have numbers between 100 and 199. Similar control volumes on the pumped loop are numbered 300-399.

NODAL SCHEME OF THE DIABLO CANYON UNIT 2 MODEL

Units 1 and 2 are of similar design. Unit 2 has a higher thermal output and therefore will have the bounding pressure peak during postulated accidents.

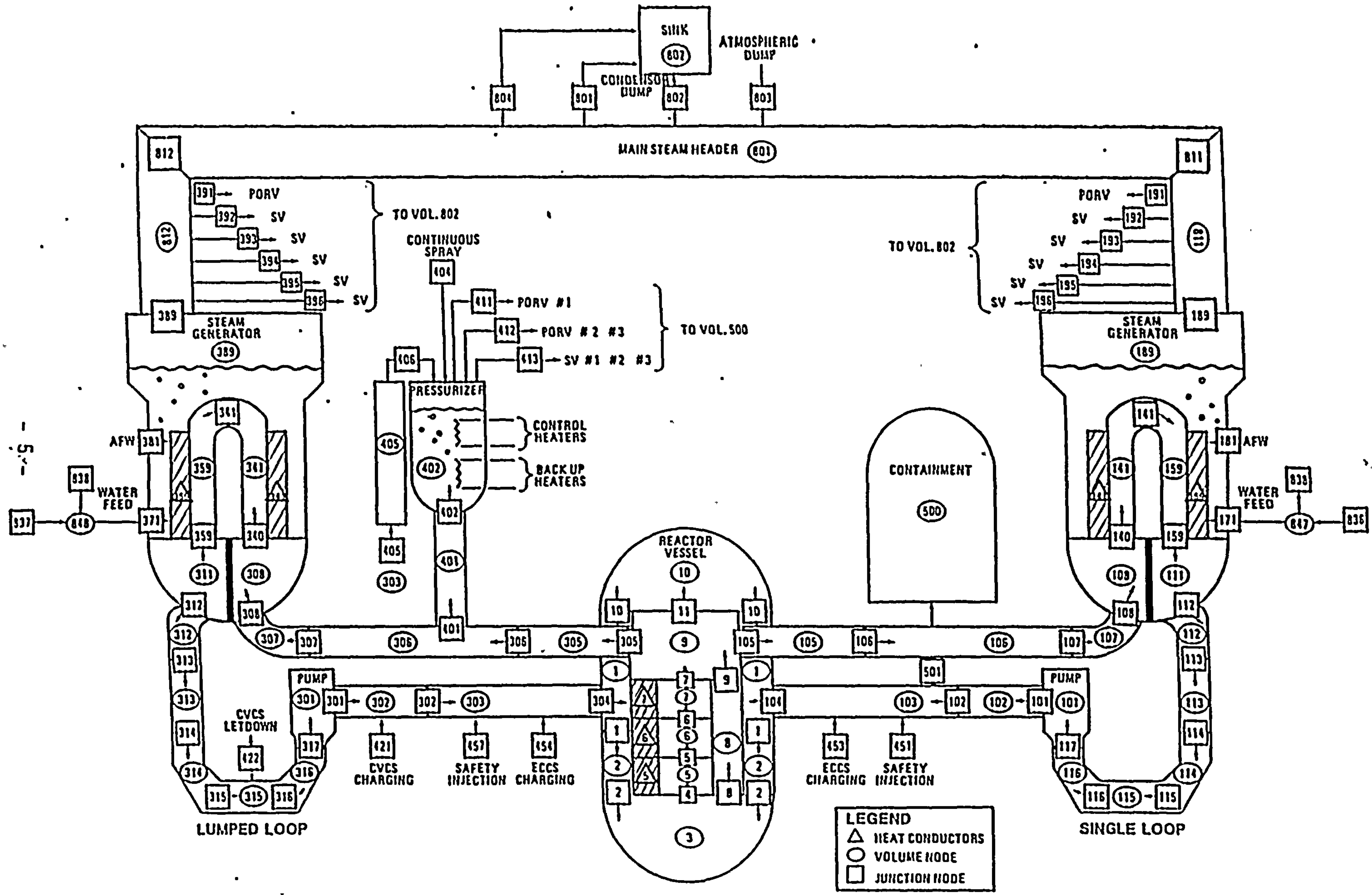

SINK
802
ATMOSPHERIC DUMP
CONDENSOR DUMP
804
801
802
803
MAIN STEAM HEADER
801
812
811
PORV
SV
TO VOL. 802
CONTINUOUS SPRAY
404
411
PORV #1
412
PORV # 2 #3
413
SV #1 #2 #3
TO VOL. 500
406
PRESSURIZER
405
402
CONTROL HEATERS
BACK UP HEATERS
401
303
STEAM GENERATOR
389
189
AFW
381
181
WATER FEED
838
837
848
371
836
847
171
833
CONTAINMENT
500
REACTOR VESSEL
10
9
3
PUMP
301
101
CVCS LETDOWN
422
421
CVCS CHARGING
457
SAFETY INJECTION
454
ECCS CHARGING
453
ECCS CHARGING
451
SAFETY INJECTION
501
LUMPED LOOP
SINGLE LOOP
LEGEND
HEAT CONDUCTORS
VOLUME NODE
JUNCTION NODE

Figure III-1

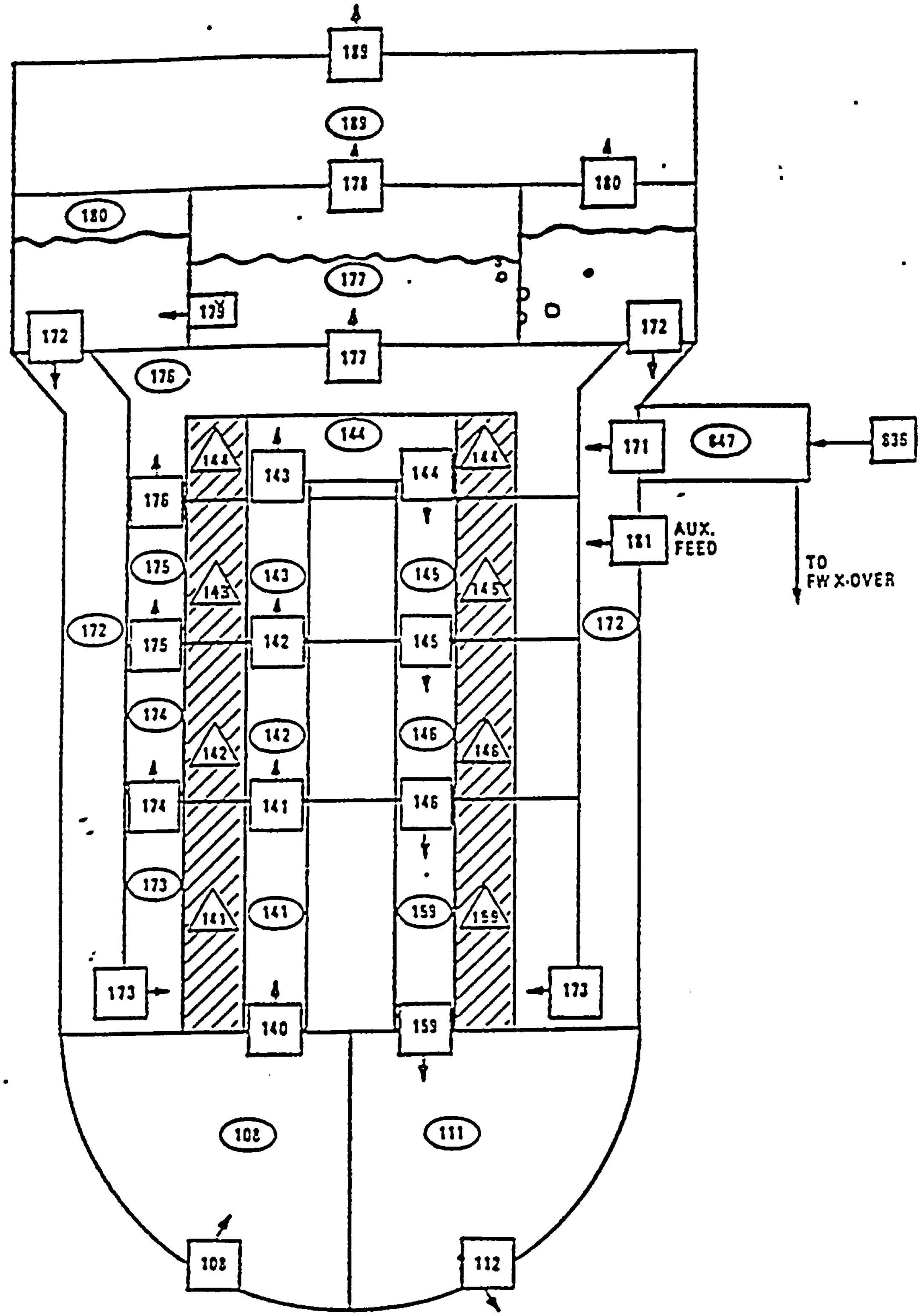


Figure III-2

Reactor Vessel

The reactor vessel is represented by nine control volumes (volumes 1 through 3 and volumes 5 through 10) connected together by ten junctions, with four additional junctions representing the inflow and outflow ports. Volumes 5, 6, and 7 represent the heated region of the core. These volumes are in contact with three heat slabs, numbered 5, 6, and 7, which represent the heat generating fuel rods within the core. Control volume 8 simulates the unheated core bypass flow, which is the fluid that flows upward through the core but is not directly heated by the fuel assemblies. This includes the fluid which flows between the fuel assemblies and the core baffle and the fluid which flows through control rod guide thimble tubes.

The heat generation in the core heat slabs is controlled by the RETRAN point kinetics model which uses six delayed neutron groups and eleven delayed gamma emitters. Reactivity inputs to the kinetics model were developed from the rod control system, boron associated with safety injection, reactor coolant density changes, and the fuel Doppler reactivity. The total core power is the sum of the direct fission power and the radioactive decay power. Pump power is also accounted for in the RETRAN model. The radioactive fission product decay is simulated by eleven decay groups plus a two-group actinide decay model.

Both the heated core flow and the core bypass fluid flow into the upper plenum region are represented by control volume 9. The hot legs for both the single and the lumped loops are supplied with fluid from volume 9 through junctions 105 and 305 respectively. The upper head region of the reactor vessel is represented by control volume 10. Cold leg fluid from the upper downcomer region is permitted to enter the upper head region through junction 10. However, this flow path is quite small relative to the other major flow paths inside the reactor vessel.

Two control volumes, numbered 1 and 2, are used to represent the downcomer. The junctions from both the single and the lumped loop cold legs (junctions 104 and 304 respectively) enter volume number 1. Flow is then downward from volume 1 through junction 1 to the lower downcomer (volume 2). Volume 2 is connected to the lower plenum (volume 3) by junction 2.

Primary System Piping

The modeling of the piping on each of the two loops is essentially the same. On each loop there are three control volumes representing the hot leg and nine control volumes including a pump representing the cold leg. Each loop also has a steam generator.

Reactor Coolant Pump

The four reactor coolant pumps (RCPs) are represented by volumes 101 and 301 for the single and lumped loop, respectively. These volumes utilize the RETRAN built-in centrifugal pump model to simulate the performance of an RCP. In this pump model, the behavior of the pump under various operating conditions is modeled using pump characteristics such as rated pump speed, torque, head, volumetric flow rate, initial fluid density, pump flywheel inertia and homologous curves.

In modeling a single pump on the lumped loop, the design flow rate at the rated speed and rated head was increased by a factor of three. With this specification the lumped loop pump will operate like three single loop pumps in parallel.

Pressurizer and Associated Piping

Three control volumes are used to model the pressurizer and associated piping. Volume 402 is a two-phase volume used to model the actual pressurizer tank while volume 405 models the spray line from the cold leg and volume 401 represents the surge line which connects the pressurizer to a hot leg. The bubble rise model is used in volume 402 to create a well defined liquid level in the pressurizer. Diablo Canyon has two spray lines which draw flow from two different cold legs, and then are joined into a single spray line which enters the pressurizer. The pressurizer spray lines are modeled by volume 405 as shown in Figure III-1.

The non-equilibrium model was selected to represent the pressurizer because subcooled liquid enters the volume through both the spray line and the surge line.

Containment Model

Volume 500 is a large volume of the same approximate size as the containment, and is initially full of saturated air at atmospheric pressure. This volume is used to provide a back pressure approximating that of the Diablo Canyon containment for those transients which involve a significant leak from the reactor system into the containment. For this analysis, this volume is connected via junction 501 to the cold leg of the single loop and junctions 411, 412 and 413 to the pressurizer.

Seventeen Volume Steam Generator Model

This steam generator model utilizes seventeen control volumes and twenty two junctions to model the primary and secondary sides of the steam generator.

Figure III-2 is a graphical representation of the control volume and junction structure used to model steam generator using the seventeen

volume model. The primary fluid enters the steam generator lower plenum (volume 108) through junction 108 from the hot leg of the primary system. Volumes 141 through 159 represent the primary side of the tube region of the steam generator, and volume 111 is the outlet plenum. Junctions 140 through 159 are used to connect the above volumes into a single flow path which models the tubes within the generator.

Seven volumes along the primary system flow path within the tube region are used to yield a more accurate prediction of the primary system temperature variation and the heat transfer rate through the steam generator. Separate heat slabs are used to model the primary to secondary heat transfer from each of the seven tube volumes. The two plenum volumes at the steam generator entrance and exit do not improve the accuracy of the temperature variation prediction, but do account for the time delay that the plena have on the steam generator exit temperature.

The primary side of the steam generator is modeled with a single flow path. The volume, length and flow area of this single flow path is set equal to the total volume, average length, and total flow area of the actual tubes. In order to model the frictional losses through the tubes correctly the hydraulic diameter of the single flow path of the steam generator model is set equal to the diameter of a single tube. The above assumptions allow the correct mass inventory, flow velocity, Reynolds number, and fluid transit time to be simulated in the model. The inlet and exit plenums of the model have the same volumes as those of the actual steam generator.

Volumes 173, 174, 175, and 176 represent the heated region of the steam generator secondary side as a single flow path vertically through the four volumes. During normal operation subcooled water enters volume 173 and flows vertically upward through this heated region and enters the separator region (volume 177) with a steam quality of approximately 38 percent. Thus, only 38 percent of the tube region mass flow goes directly out of the steam generator in the form of steam, the remaining liquid is recirculated through the downcomers (volume 180 & 172) and back into the tube region.

The steam generator recirculation is driven only by natural circulation, with the driving force being the density difference between the subcooled fluid in the downcomer and the boiling fluid in the heated region. Thus, in order to model the correct recirculation, frictional losses which exactly balance the natural circulation driving head at full power and flow conditions are modeled.

Volume 177 is used to represent the steam separators and the steam dryers, and preserves the steam flow rate and quality into the steam dome (volume 189). The bubble rise model is used here with maximum bubble rise velocity to ensure a complete separation of the liquid and steam phases. The liquid from this volume is discharged to the upper downcomer (volume 180) to complete the recirculation path. The high

quality steam is passed on to the steam dome (volume 189) via junction 178.

The annular region in the upper downcomer (volume 180) is simulated by a bubble rise volume to give a distinct mixture level, which is important as this level is used as a feedback to the steam generator level control system. In reality, feedwater is fed into the steam generator through this region, but since the volume contains a two phase mixture, numerical problems would result if subcooled liquid were introduced into this volume. Therefore, the feedwater inflow junction (junction 171) connects to the lower downcomer (volume 172) instead. This representation has negligible effects on the natural circulation driving force and the steam generator performance.

Main Steam Line

The main steam lines from the steam generators to the main steam isolation valves are represented by volumes 811 and 812 for the single and lumped loops respectively. Volumes 811 and 812 have inlet junctions 189 and 389 (for the single and lumped loop respectively) from the steam domes of the steam generator and outlet junctions 811 and 812 to the volume 801. The main steam safety valves are modeled as junction 192 to 196 and 392 to 396 for single loop and lumped loops, respectively. Junctions 191 and 391 represent the main steam PORVs for the single and lumped loops.

Volume 801 represents the rest of main steam lines and main steam header. This control volume has four outlet junctions representing the turbine inlet (801), the steam dump to condenser (802), the atmospheric steam dump (803), and the main steam bypass outlet to the moisture separator reheaters (804). Junction 804 models the moisture separator reheater (MSR) bypass after a turbine trip.

IV. RETRAN COMPARISON WITH PLANT TURBINE TRIP DATA

This section shows a comparison between RETRAN results and plant data. The plant data was obtained from a DCPP Unit 2 startup test performed at 3:19 AM on January 13, 1986 for Plant Trip from 100 percent power as specified in Unit 2 Test Procedure (TP) 43.4, "Plant Trip from 100 Percent Power." This test is considered to be a reasonably close match to the loss of external electrical load/turbine trip accident analyzed later in this report. The turbine was manually tripped from the control room. A reactor trip signal immediately followed the turbine trip and all automatic control systems were operable. The test data points were recorded at three second intervals.

The main purpose of that test was to verify the ability of the plant to sustain a unit trip from 100 percent power. Important parameters such as reactor power, reactor coolant temperature, pressurizer level and pressure, steam flow and pressure, steam generator level, and feedwater flow and pressure were monitored. During the test, the

pressurizer safety valves and main steam safety valves did not lift. No safety injection or reactor coolant pump trips occurred.

Table IV-1 shows the plant conditions at the start of the test. The first column gives the RETRAN initial conditions and the second column gives the recorded plant data. Note that in the plant data column, a few entries are designated (DD) for design data. For example, in the case of the steam and feed flows, design data was used because the plant data is not recorded with sufficient accuracy for computer code initialization purposes.

The boundary conditions and assumptions used in the RETRAN modeling of Turbine Trip Test are described below:

- o A manual turbine trip and reactor trip signal occur simultaneously 3 seconds after test initiation.
- o A control rod drop time of 1.3 seconds was used, which corresponds to measured plant control rod drop times.
- o The feedwater control valves close linearly within six seconds after the feedwater isolation signal was initiated.
- o The main steam bypass flow to the MSRs was modeled by creating a junction in the main steam header of sufficient area to allow the correct initial bypass flow at full power operation.
- o After the turbine trip signal, the turbine control valves close linearly over a 0.2 second interval.
- o The 1973 ANS standard decay heat curve is weighted by a factor of 0.2 to reflect the fact that the core was recently loaded at the time of the test, and there was very little burnup on the fuel.
- o A resistance temperature detector (RTD) overall delay time of 7.1 seconds was considered in the analysis. This is based upon startup test measurements at DCPP.

TABLE IV-1

INITIAL CONDITIONS FOR THE TURBINE TRIP TEST

	RETRAN	PLANT DATA
Reactor Power	3417 MWt	3417 Mwt
T-hot	600.3 degrees F	599.9 degrees F
T-cold	540.1 degrees F	540.2 degrees F
Tavg	570.2 degrees F	570.0 degrees F
Pressurizer Pressure	2250.0 psia	2247.4 psia
Pressurizer Level	54.2%	54.2%
Steam flow/loop	1032.1 lbm/sec	1032.1 lbm/sec (DD)
Feed flow/loop	1032.1 lbm/sec	1032.1 lbm/sec (DD)
Main Steam Pressure Header	742.5 psia	742.5 psia
Steam Generator Pressure	782.1 psia	782.1 psia
Steam Generator Level (NR)	44%	44%

Figures IV-1 through IV-10 compare the RETRAN calculations with plant data. Figure IV-1 shows reactor power as a function of time with the RETRAN power matching measured power very well. Figure IV-2 displays the RCS average temperature as a function of time. RETRAN results only slightly deviate from plant data near the end of the transient. Figures IV-3 and IV-4 show Loop 1 hot and cold leg temperatures respectively. RETRAN predicts slightly lower temperatures near the end of the transient.

Figure IV-5 shows the pressurizer pressure as a function of time. The RETRAN results show excellent agreement with the test data. RETRAN assumes normal automatic pressurizer spray and heater controls. The pressurizer level comparison is shown in Figure IV-6 and the agreement is also good.

Figure IV-7 shows steam header pressure. The peak pressure difference is approximately 50 psi, with RETRAN predicting a higher steam header peak pressure.

Figure IV-8 shows main steam flow during the transient for Loop 3. As expected, steam flow to the turbine diminishes quickly after turbine trip and the Group 1 steam dump valves open to provide sufficient cooling to RCS. The steam dump valves fully close after reaching Tavg-Low. RETRAN trends properly with the data.

Figure IV-9 shows feedwater flow as a function of time for Loop 3. The agreement in magnitude and trend is very good in that an initial drop in feed flow due to diminishing steam flow is followed by a rise due to rapid steam generator water level drop and then followed by a drop due to feedwater isolation. There remains a small but non-vanishing feed flow for both traces. Figure IV-10 shows the steam generator narrow range level for Loop 3 during the test. Both RETRAN and plant steam generator levels drop quickly after the turbine trip, and the trends are comparable.

In summary, overall agreement with the test data is very good, with all trends being properly predicted and good agreement between RETRAN and test data.

TURBINE TRIP TEST

DCPP UNIT 2

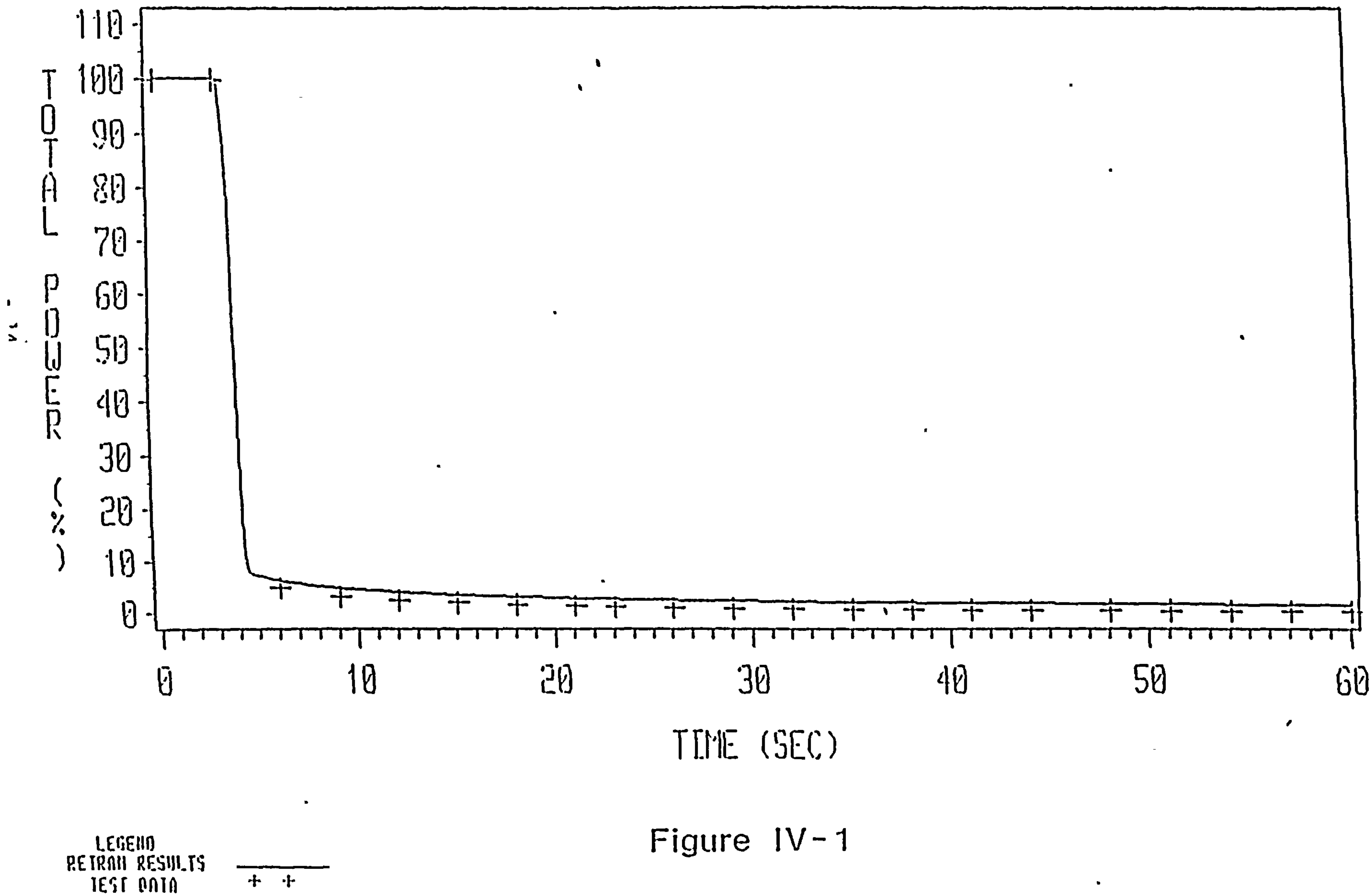


Figure IV-1

TURBINE TRIP TEST

DCPP UNIT 2

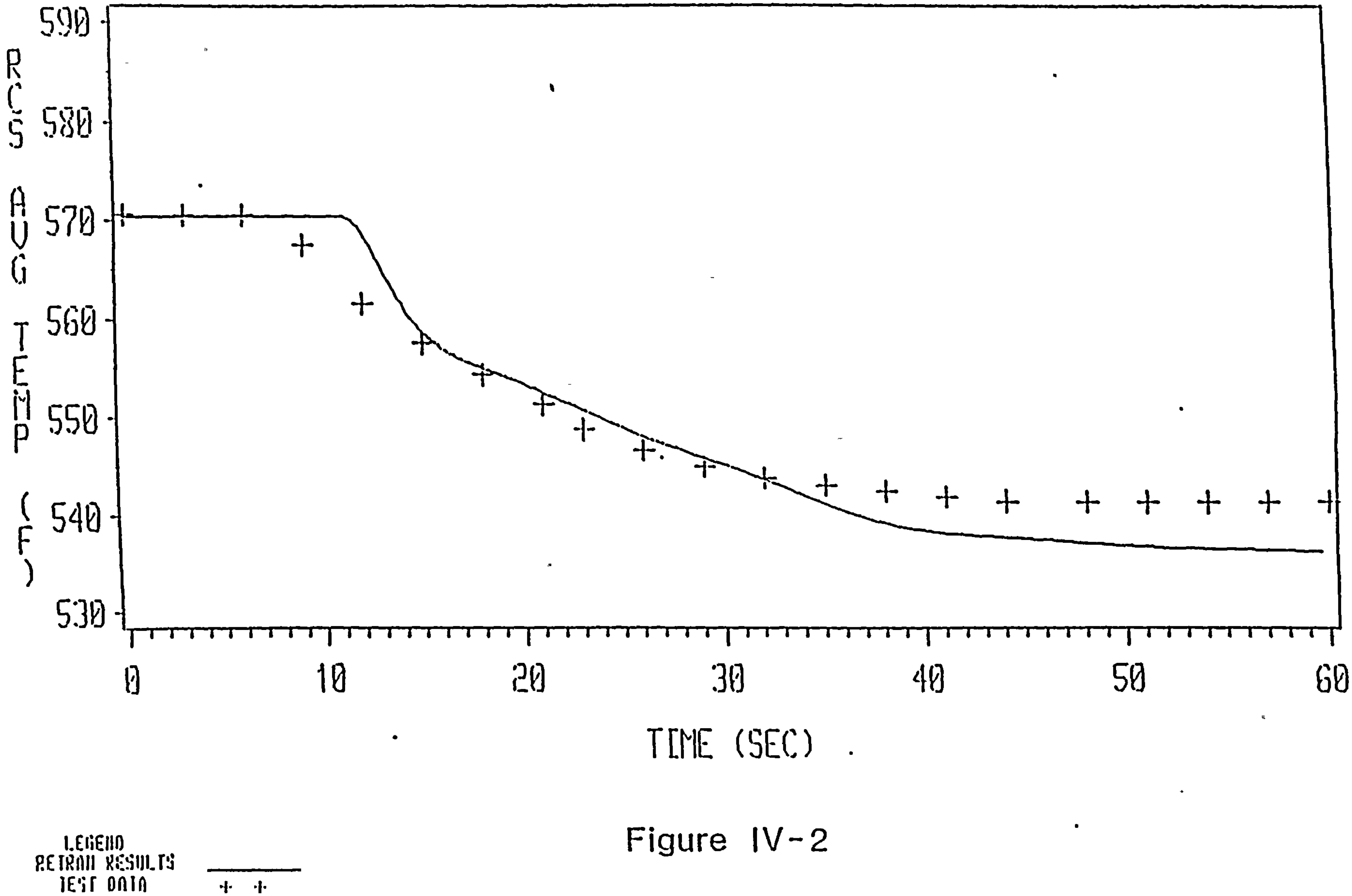


Figure IV-2

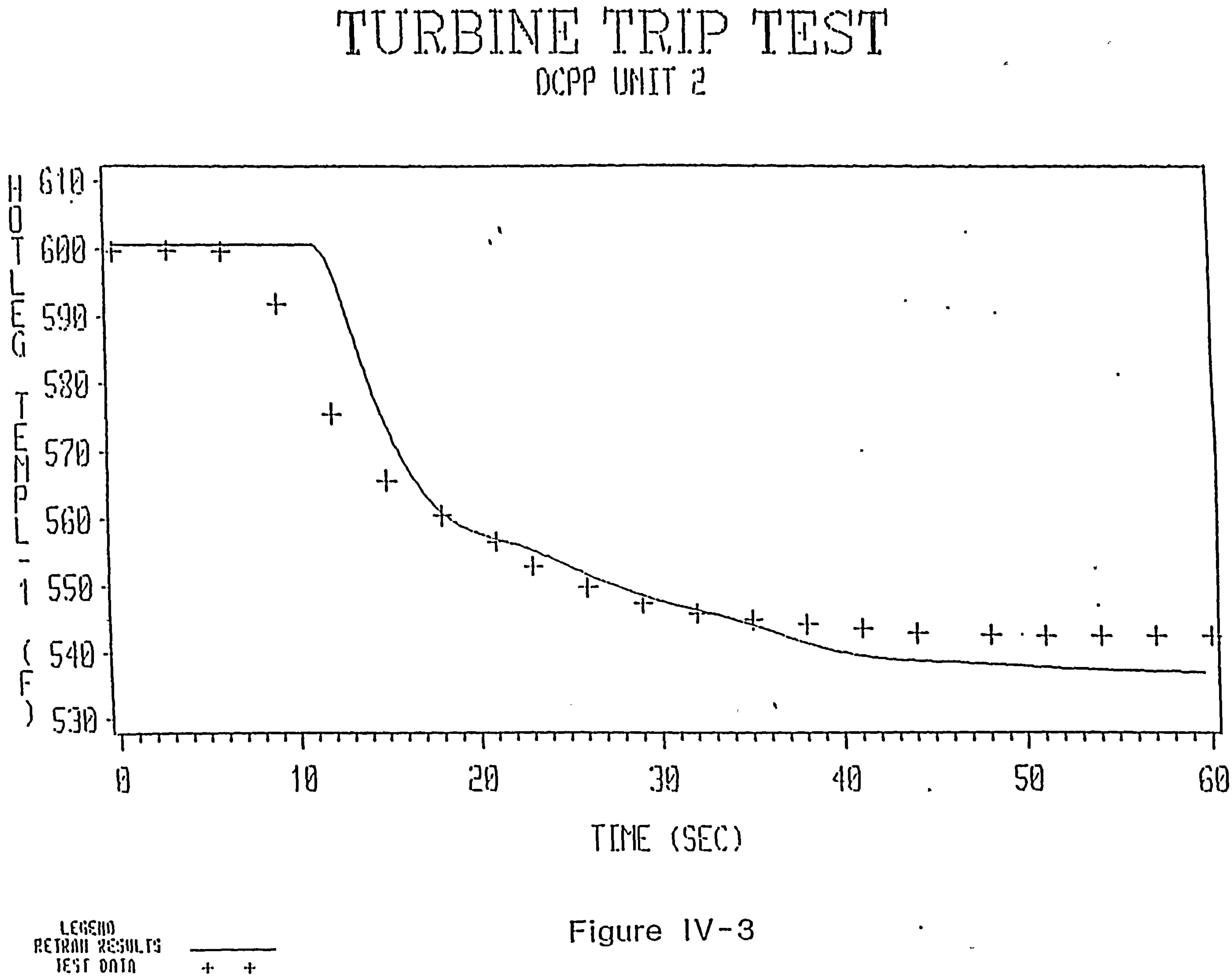

TURBINE TRIP TEST
DCPP UNIT 2
HOT LEG TEMP L-1 (F)
610
600
590
580
570
560
550
540
530
0
10
20
30
40
50
60
TIME (SEC)
LEGEND
RETRAN RESULTS
TEST DATA

Figure IV-3

TURBINE TRIP TEST

DCPP UNIT 2

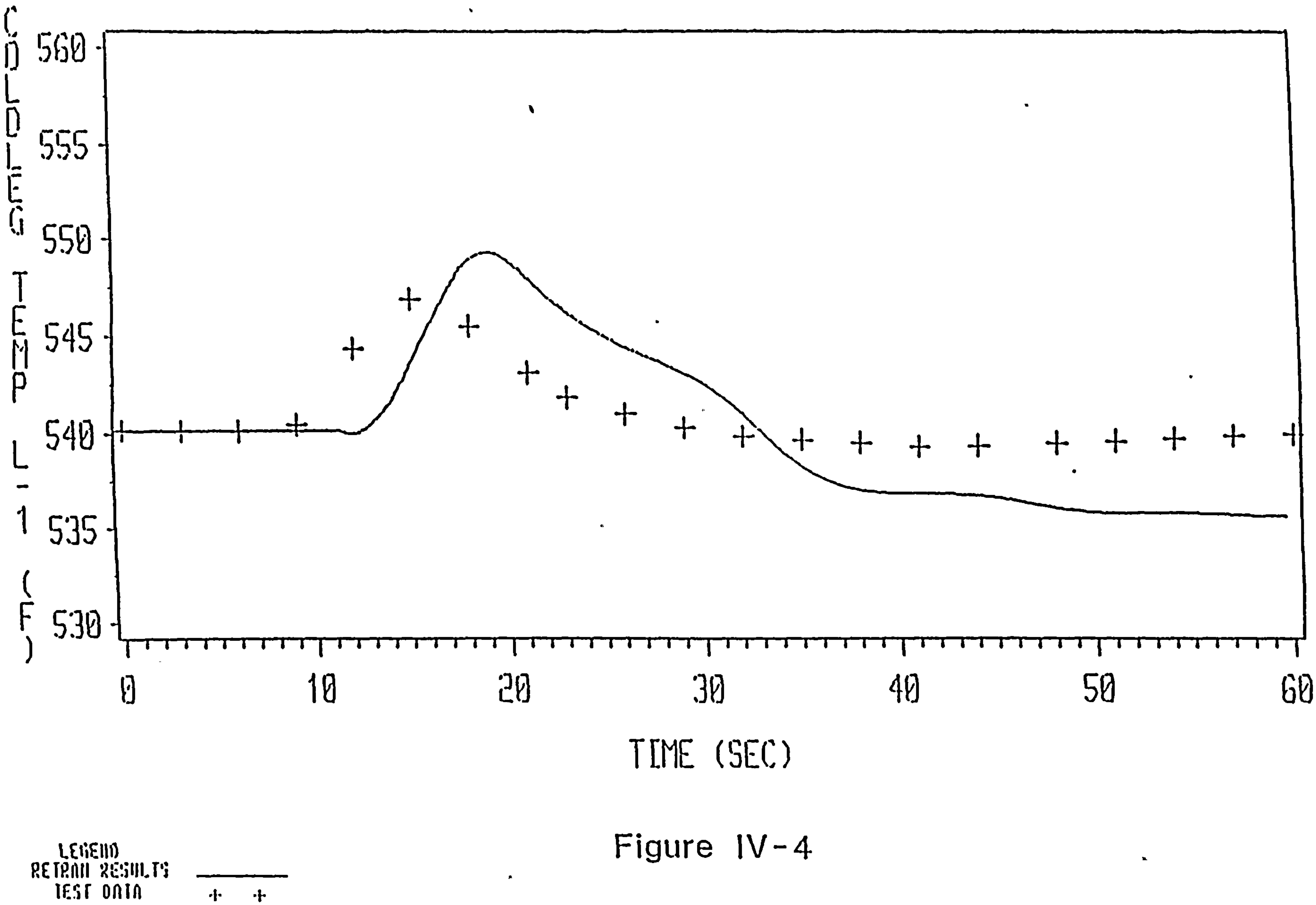


Figure IV-4

TURBINE TRIP TEST

DCPP UNIT 2

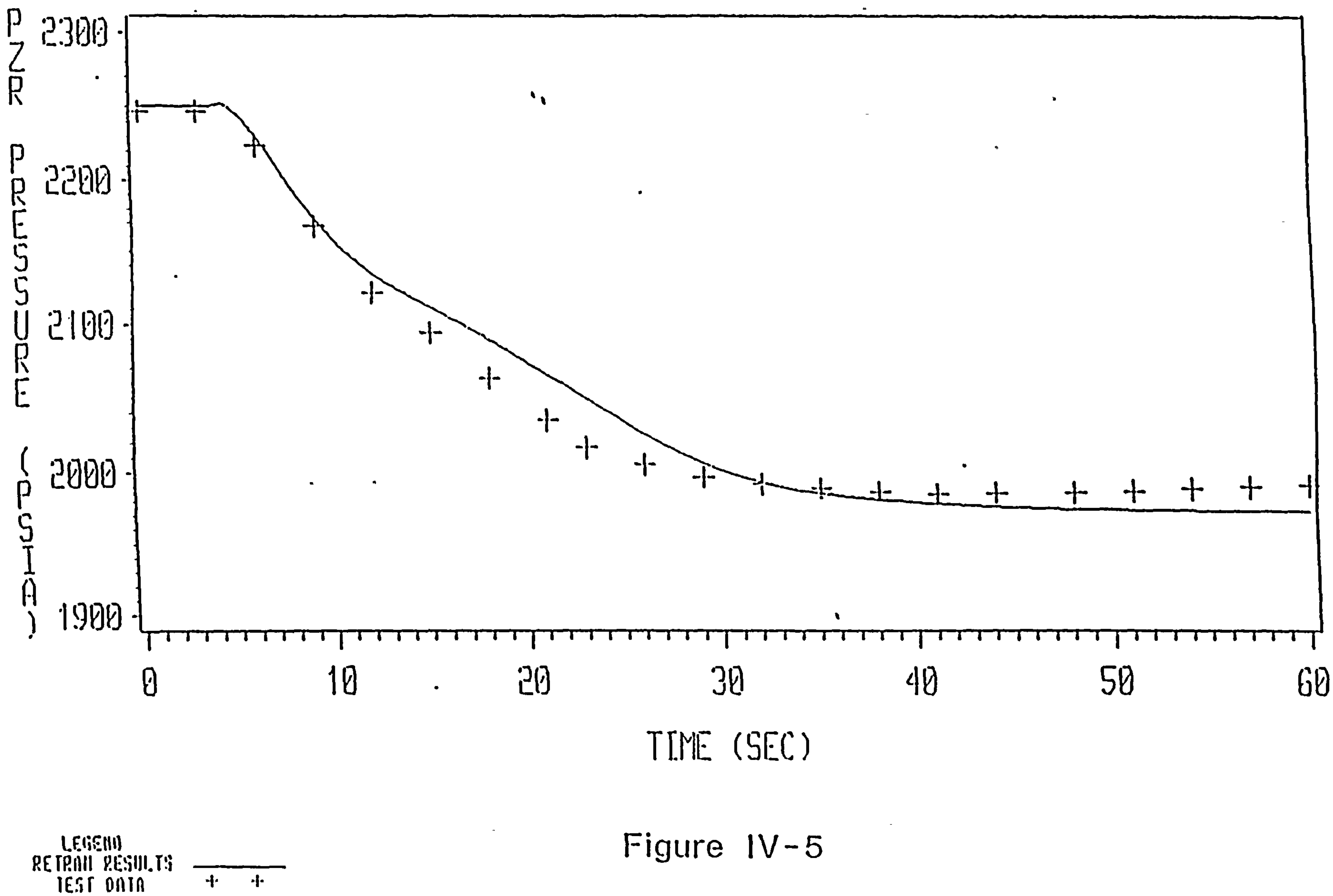
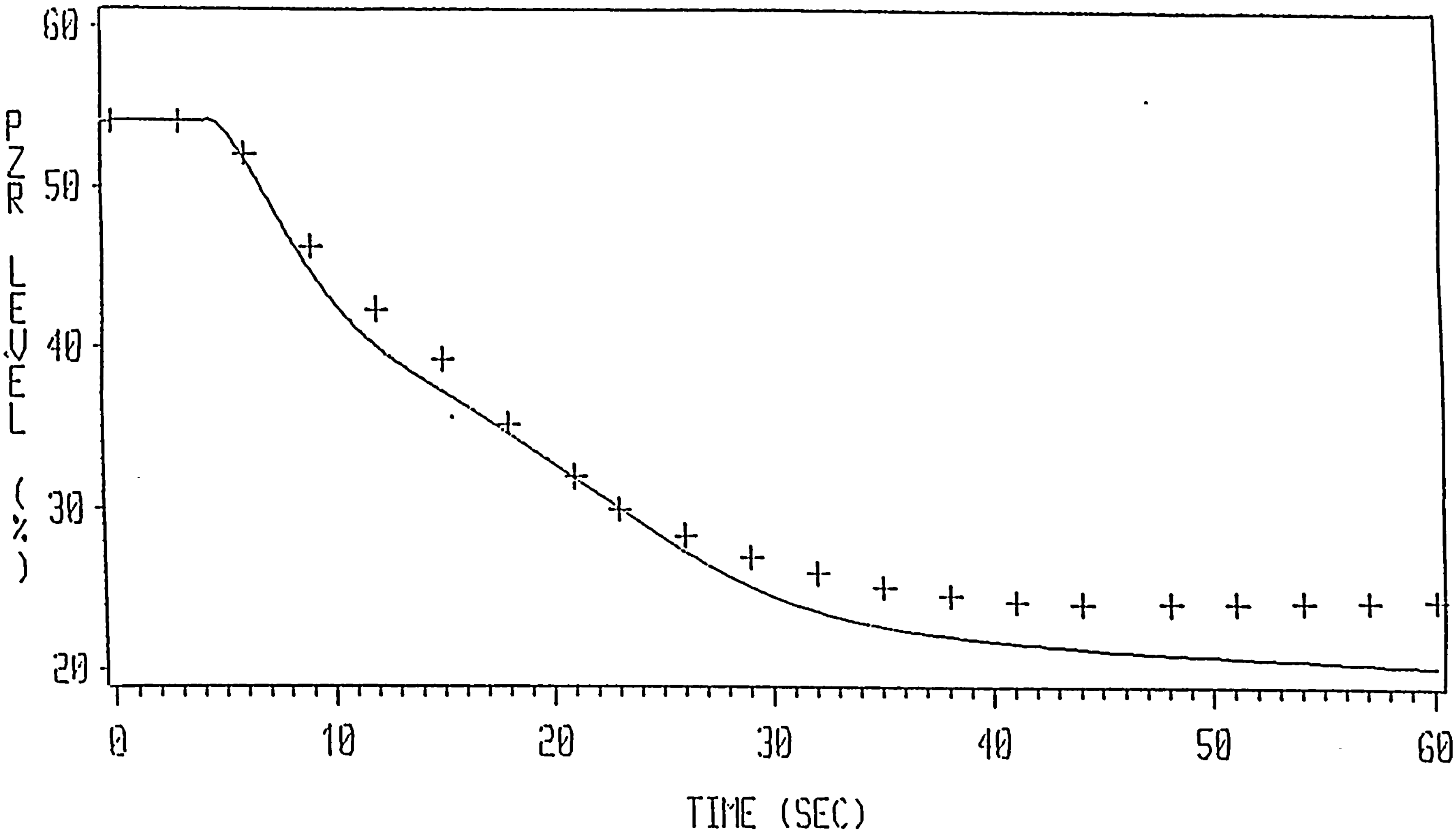


Figure IV-5

TURBINE TRIP TEST

DCPP UNIT 2



LEGEND
RETRAN RESULTS
TEST DATA + +

Figure IV-6

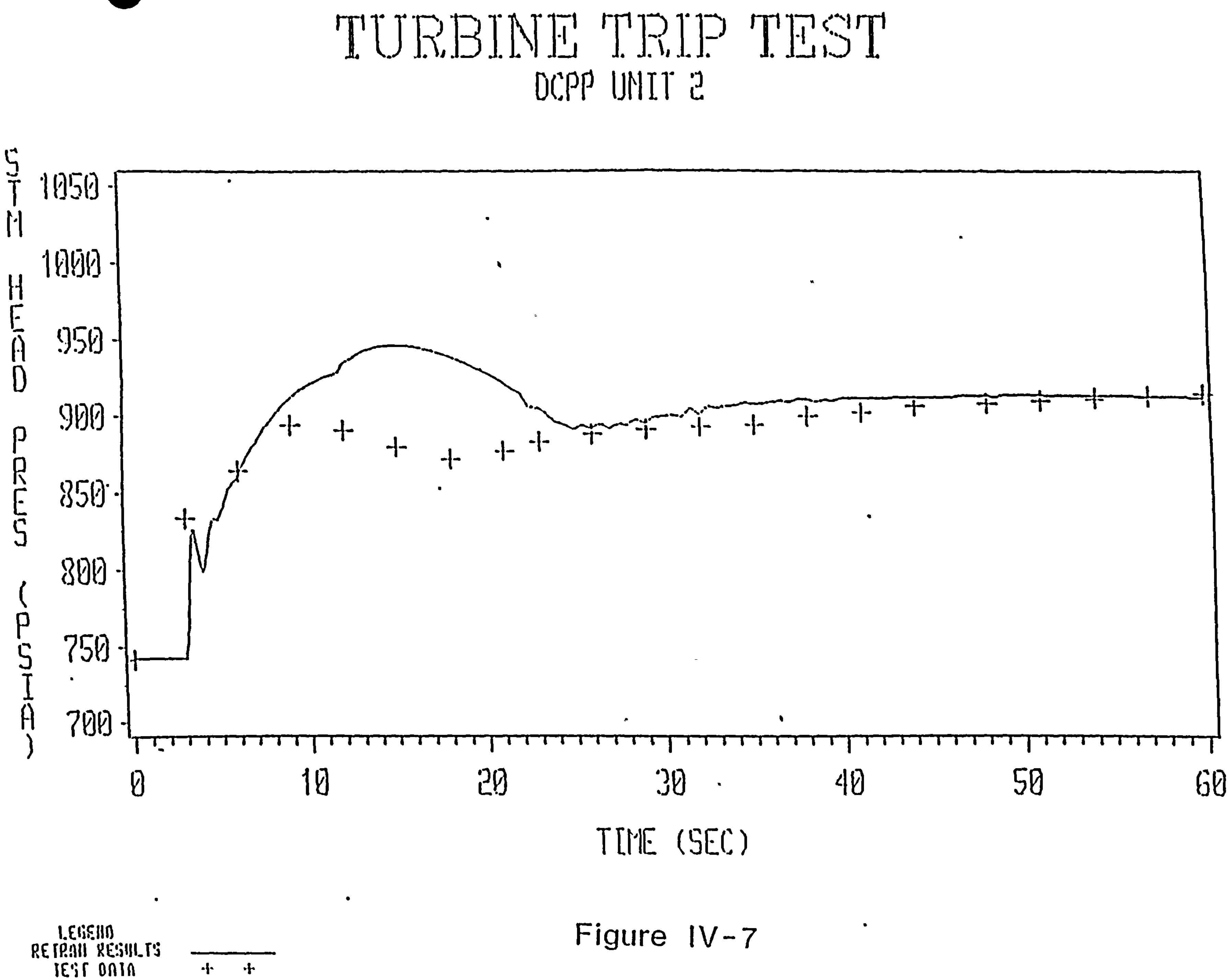

TURBINE TRIP TEST
DCPP UNIT 2
STM HEAD PRES (PSIA)
1050
1000
950
900
850
800
750
700
0
10
20
30
40
50
60
TIME (SEC)
LEGEND
RETRAN RESULTS
TEST DATA

Figure IV-7

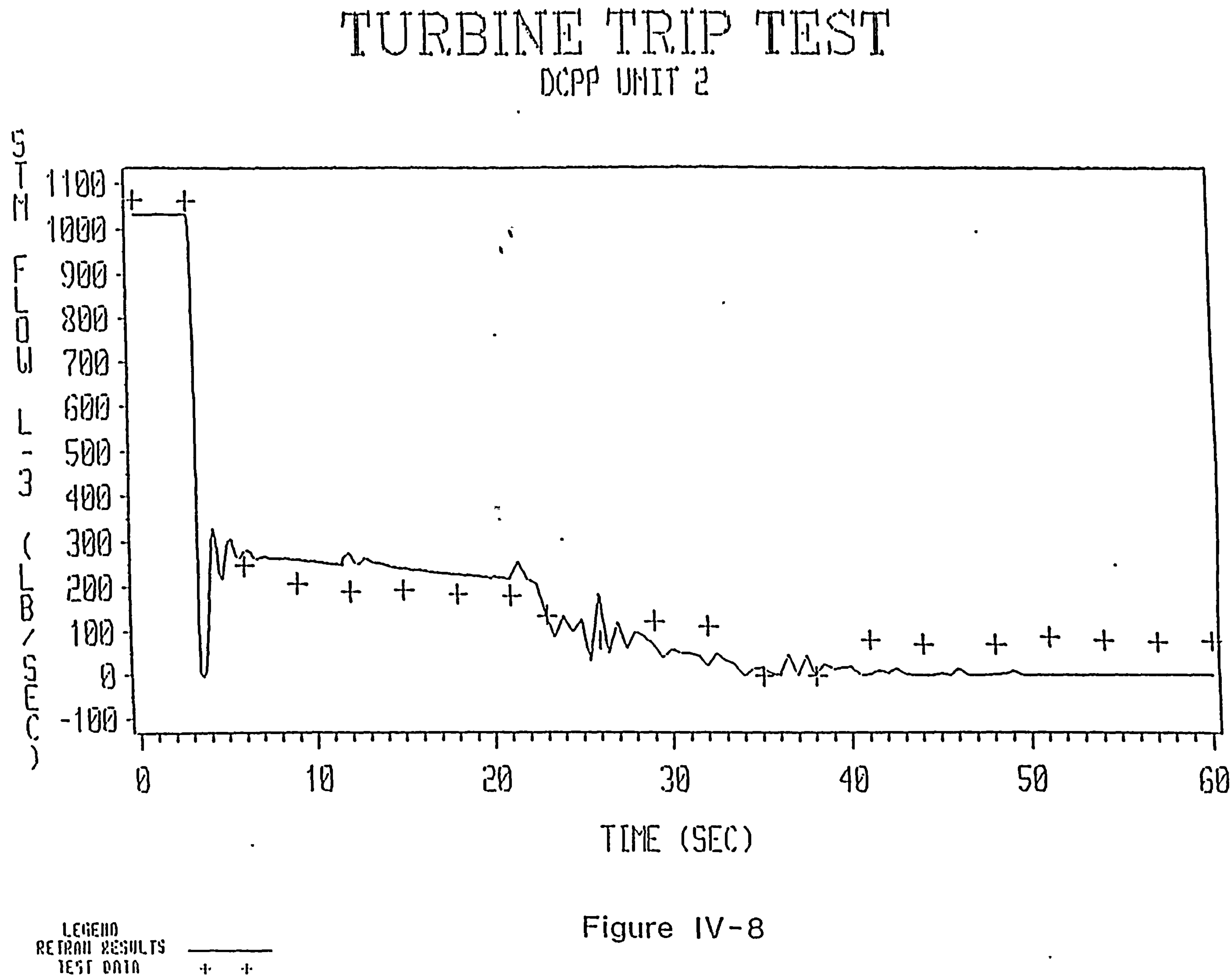


Figure IV-8

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TURBINE TRIP TEST

DCPP UNIT 2

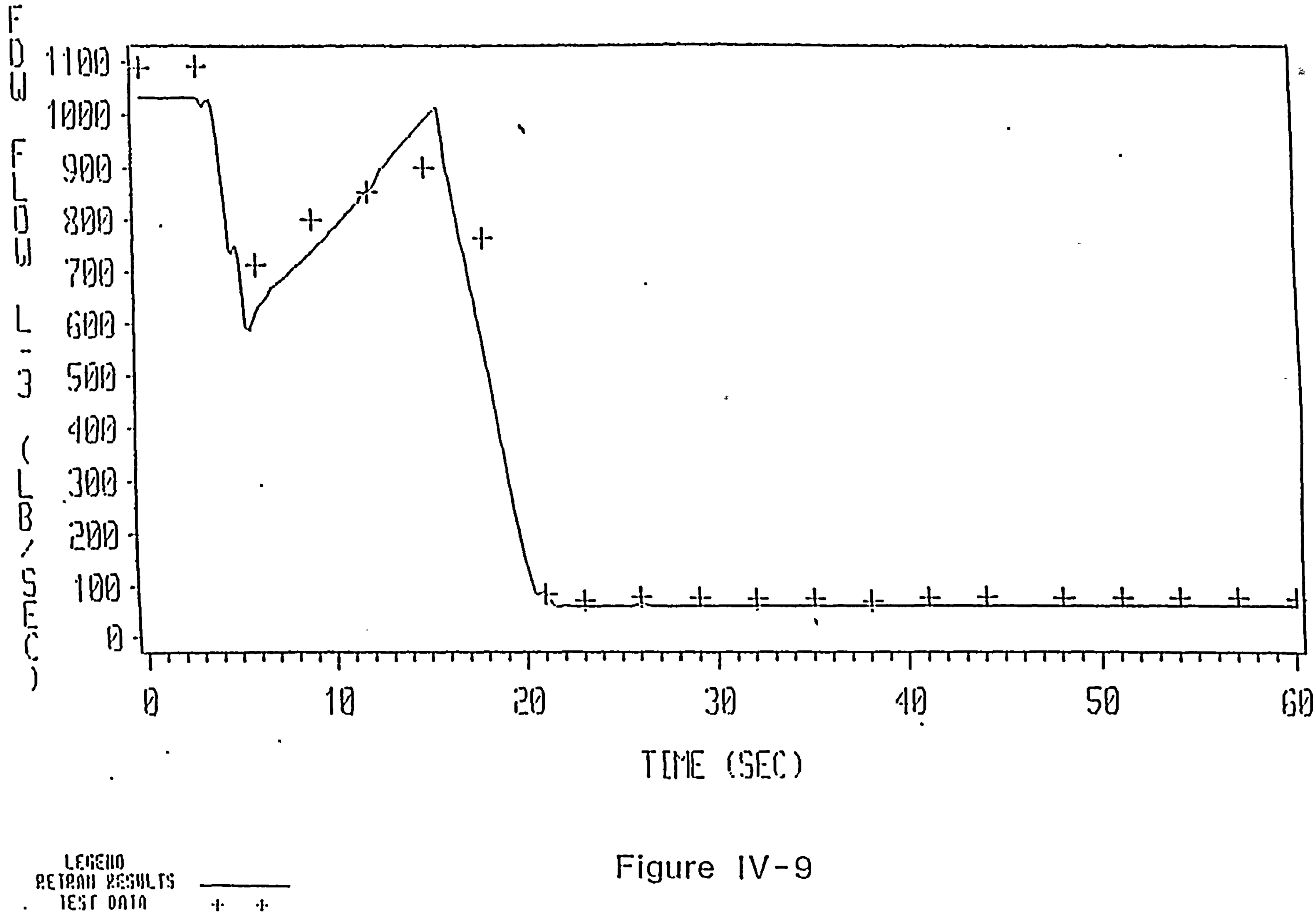


Figure IV-9

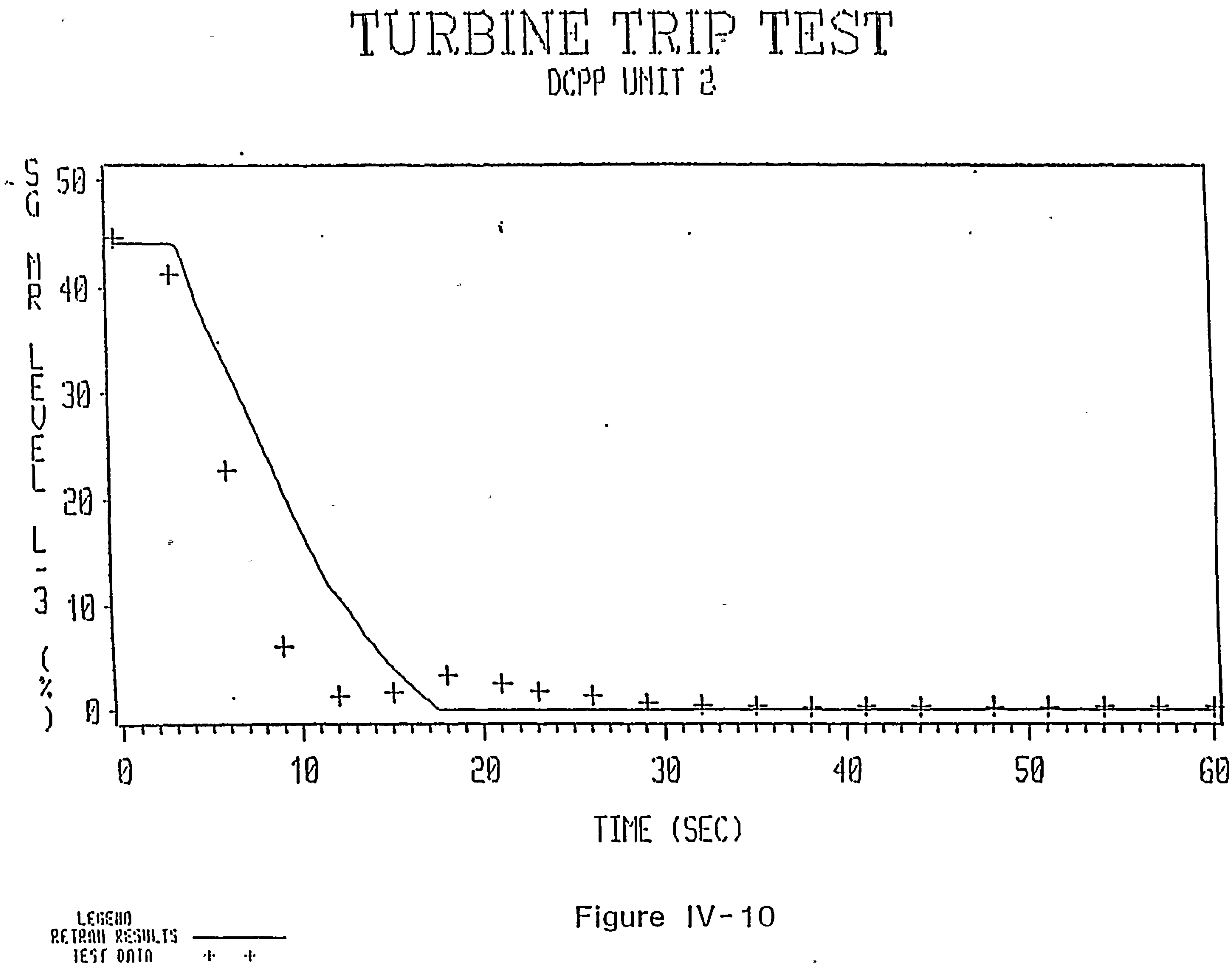

TURBINE TRIP TEST
DCPP UNIT 2
SG NR LEVEL L-3 (%)
50
40
30
20
10
0
0
10
20
30
40
50
60
TIME (SEC)
LEGEND
RETRAN RESULTS
TEST DATA

Figure IV-10

V. RETRAN COMPARISON WITH THE FSAR LOSS OF EXTERNAL ELECTRICAL LOAD/TURBINE TRIP TRANSIENT

The PG&E RETRAN model was used to rerun the most limiting RCS pressurization case of the four loss of external electrical load/turbine trip events found in the FSAR Update in order to demonstrate that the model is capable of conservatively analyzing this transient. The RETRAN results were compared with the FSAR Update results. The four loss of electrical load transients in the FSAR Update consider beginning of life (BOL), end of life (EOL), availability of pressurizer pressure control and no pressurizer pressure control. The most severe of the four cases as far as RCS pressurization is concerned is the case of no pressurizer pressure control at BOL.

The RETRAN analysis was performed with the following conservative assumptions, which are consistent with those in the FSAR Update and with the Standard Review Plan (NUREG-0800):

- The analysis is performed on Unit 2, which is the limiting of the two DCPP units for accident analysis.
- The initial core power is 102 percent of nominal, 3479 MWt, where nominal is 3411 MWt.
- The transient begins with the instantaneous closure of the turbine stop valves and the instantaneous loss of all steam generator feed water.
- The initial pressurizer pressure is 2225 psia, which is the setpoint of the low pressurizer pressure alarm.
- No credit is taken for the rod control system. It is conservatively assumed that the reactor is in manual control.
- No credit is taken for direct reactor trip after turbine trip. The high pressurizer pressure reactor trip setpoint has been conservatively set at 2425 psia, which is the nominal 2400 psia plus 25 psia for instrument uncertainty.
- There is a two-second delay assumed from the time when the high pressurizer pressure setpoint has been reached to when the control rods start to drop. The scram is modeled according to the scram table given in Figure 15.1-4 of the FSAR Update, with total negative reactivity insertion of 4 percent as assumed in Chapter 15 of the FSAR Update.
- The decay heat has been modeled according to the 1973 ANS decay heat table, with a 1.2 multiplier.

- A Moderator Temperature Coefficient of +5 pcm per degree F and a Doppler Reactivity Coefficient of -2.14 pcm per degree F were taken from Chapter 15 of the Diablo Canyon FSAR Update.

- No credit is taken for the operation of the pressurizer power operated relief valves (PORVs) or the pressurizer spray valves. The pressurizer heaters are assumed to operate during the transient.

- RCS pressure mitigation occurs only through the actuation of the pressurizer safety valves. These valves are assumed to start to open at the nominal setpoint (2485 psig), and then linearly open with pressure until fully open at 3 percent above the nominal setpoint (2560 psig).

- No credit is taken for the steam dump system. Secondary pressure is relieved only through the steam generator safety valves. Although these valves have setpoints which vary from 1065 to 1115 psig, all are assumed to fully open instantaneously at the highest nominal setpoint (1115 psig).

- The thermal design RCS flow of 88,500 gpm/loop was assumed.

Figures V-1 through V-5 compare the RETRAN and FSAR Update loss of external electrical load/turbine trip analyses. Figure V-1 shows reactor power as a function of time. The power calculated by RETRAN is slightly higher than the FSAR Update power in the first ten seconds of the transient, but the two become asymptotically identical. Figure V-2 compares pressurizer pressure and the figure shows good agreement between the two simulations with RETRAN giving a peak pressure nearly identical to that shown in the FSAR Update. Figure V-3 shows a comparison of the pressurizer water volume. They agree well for the first twenty seconds of the transient and then RETRAN predicts a smaller water volume. A simple calculation was performed which suggests that RETRAN predicts the more accurate pressurizer water volume. The shrinkage calculated is consistent with that seen in the RETRAN run. It should be noted that both the FSAR Update results and the RETRAN predict the same peak pressurizer water volume, which is the primary parameter in calculating peak pressure. Figure V-4 shows a comparison of average RCS temperature. RETRAN results agree well with those in the FSAR Update. Figure V-5 shows a comparison of the secondary side steam temperatures.

In summary, all five comparisons show good agreement between the RETRAN and FSAR Update results.

TURBINE TRIP ANALYSIS

DCPP UNIT 2

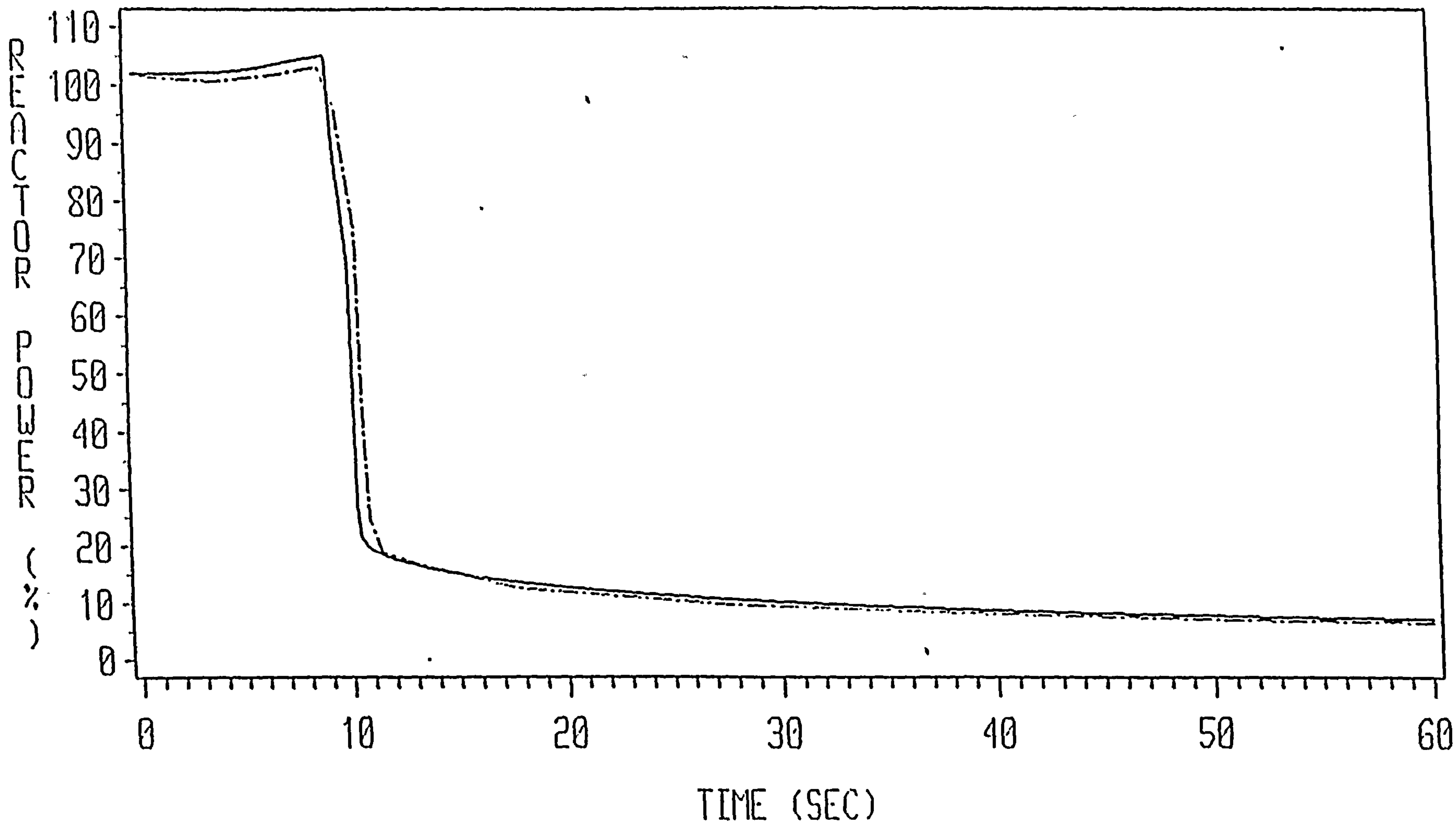
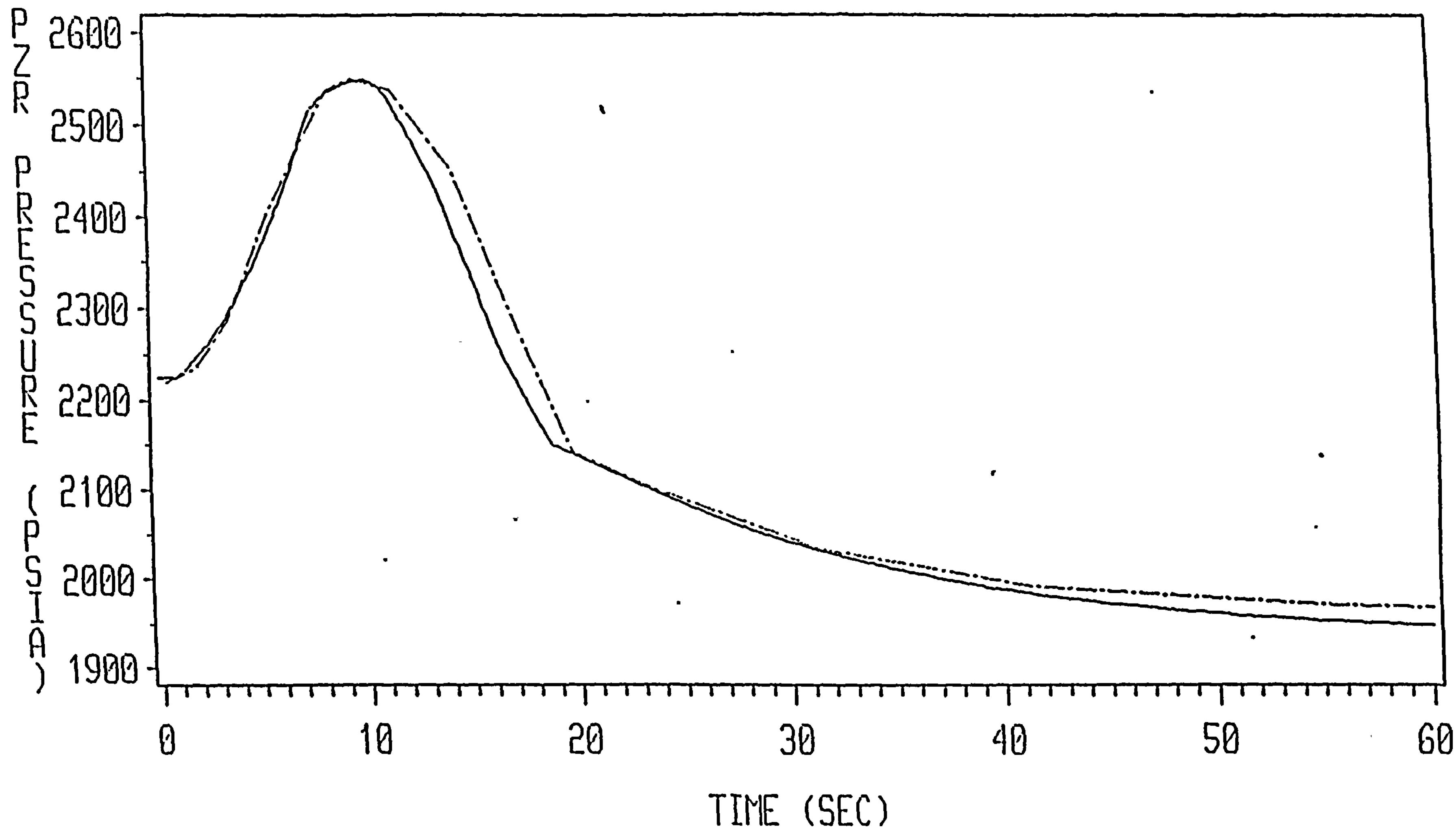


Figure V-1

LEGEND
RETRAN RESULT
FSAR DATA

TURBINE TRIP ANALYSIS
DCPP UNIT 2



LEGEND
RETRAN RESULT ————
FSAR DATA —·—·—

Figure V-2

TURBINE TRIP ANALYSIS

DCPP UNIT 2

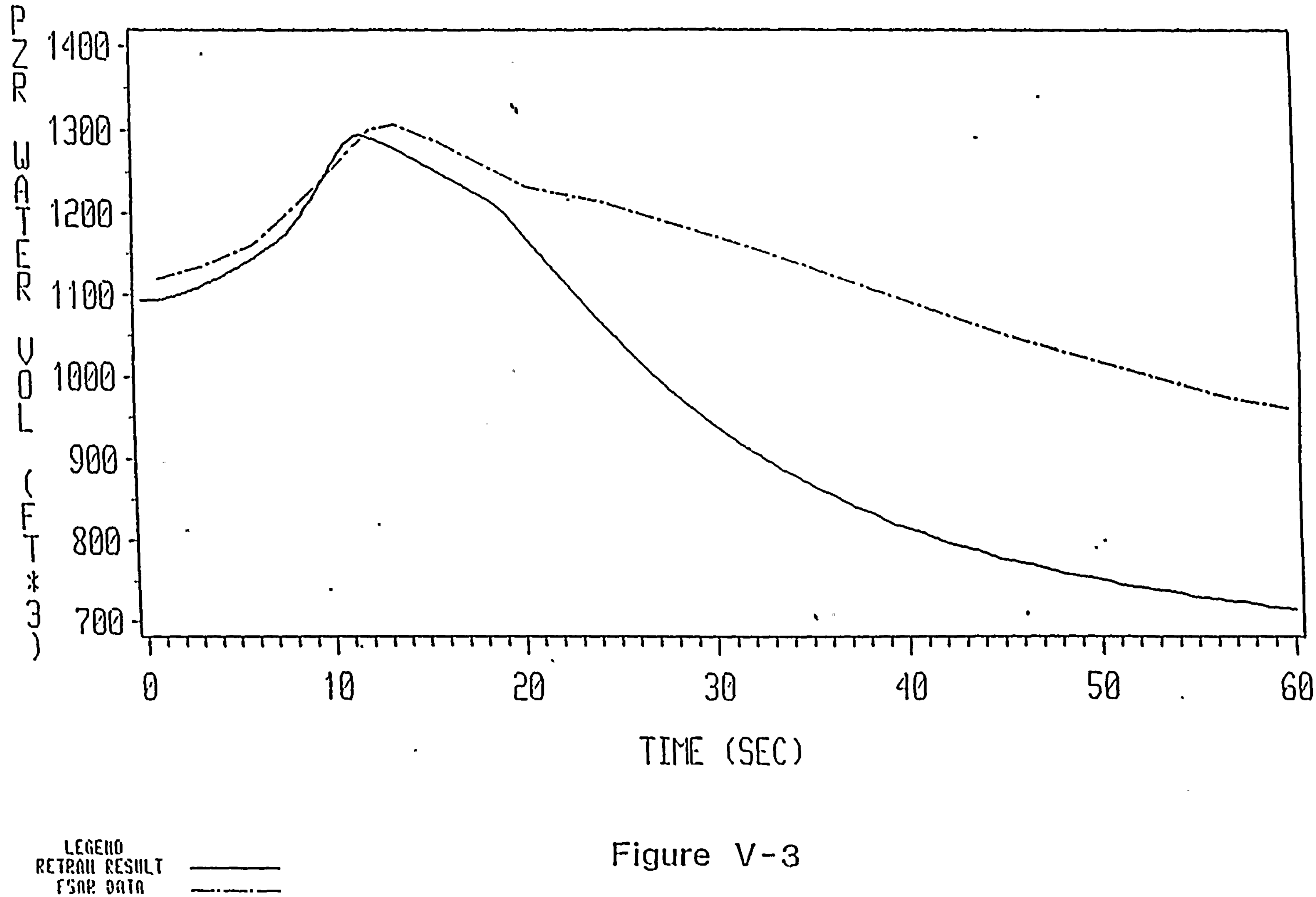


Figure V-3

TURBINE TRIP ANALYSIS

DCPP UNIT 2

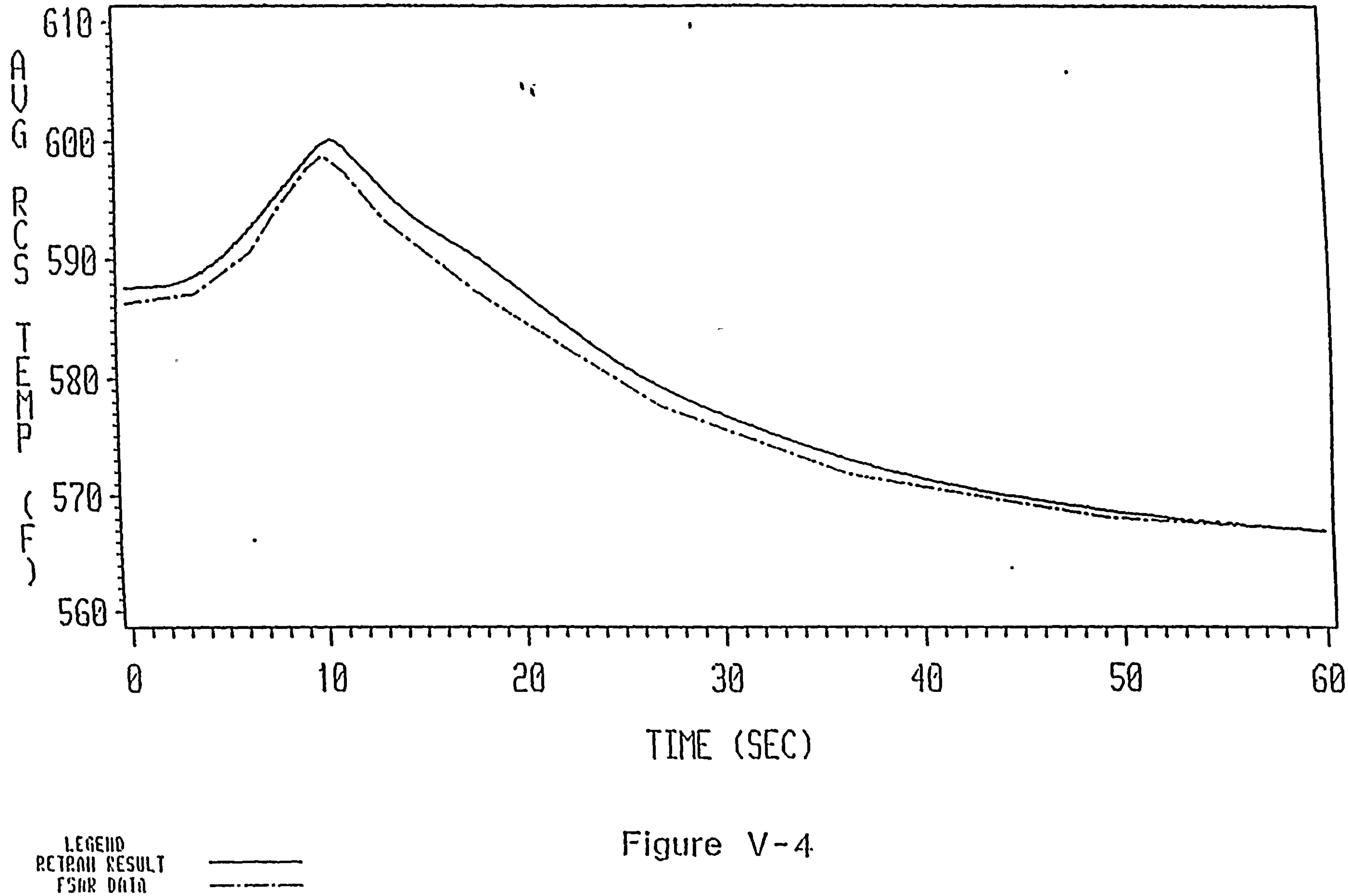
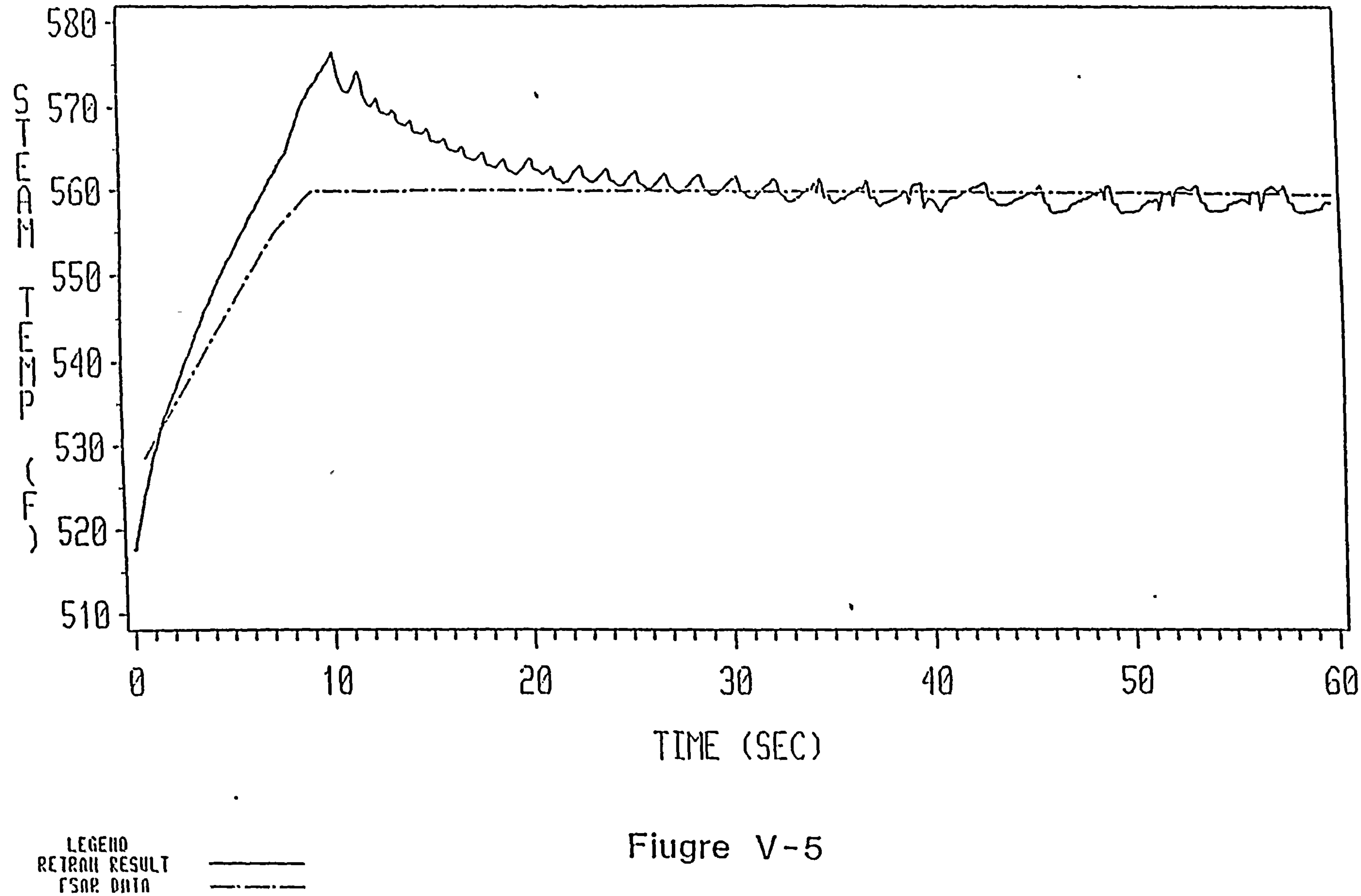


Figure V-4

TURBINE TRIP ANALYSIS
DCPP UNIT 2



Fiugre V-5

VI. ANALYSIS OF LOSS OF EXTERNAL ELECTRICAL LOAD/TURBINE TRIP EVENT WITH RELAXED MAIN STEAM LINE SAFETY VALVE SETPOINTS

Among the four cases analyzed in DCPP FSAR for the loss of external electrical load/turbine trip event, the case for BOL without pressurizer pressure control results in the highest peak RCS pressure. While, the case for BOL with pressurizer pressure control generates the limiting peak steam generator pressure. The PG&E RETRAN model was used to analyze both of the cases for +3 percent main steam safety valve setpoint tolerance. The following PG&E RETRAN model enhancements were made for the analysis.

- o To enhance the main steam line safety valve model, the individual nominal setpoint plus 3 percent tolerance for each safety valve is modeled in the analysis. In other words, the main steam safety valves start to open when the steam pressure reaches their nominal setpoints plus 3 percent. Then, the main steam safety valves are assumed to linearly open with the pressure until fully open at the 3% pressure accumulation (3% above the initial opening pressure) as recommended by ASME code. The setpoints for the main steam line safety valves are presented in Table VI-1.

- o Recent study of the impact of pressurizer safety valve loop seal identified that the presence of loop seal delays the opening of pressurizer safety valve. The loop seal water starts to leak out from the safety valve when the safety valve setpoint is reached. However, no pressure is relieved from the pressurizer until the loop seal water is completely purged, after which the safety valve pops full open in less than 0.1 second. The loop seal water purge time for Diablo Canyon Power Plant Units 1 and 2 is calculated to be 1.272 seconds using the formula developed by Westinghouse Owner's Group. Therefore, in the analysis, the opening of pressurizer safety valve is delayed by 1.272 seconds after the pressurizer pressure reaches the nominal setpoint of the safety valves plus 1 percent tolerance. After the delay, the pressurizer safety valves pop open in 0.1 second. The summary of pressurizer safety valve control is given in Table VI-1.

- o The pressurizer pressure control uncertainty was revised to 58.1 psi. Conservatively, the initial pressurizer pressure of 2176.9 psig is used in the analysis.

The following conservative assumptions were used in the analysis which were consistent with both the FSAR Update and the NRC Standard Review Plan on Overpressure Protection.

- o The initial core power is 102 percent of nominal Unit 2 core power.

- o The turbine stop valves close and all steam generator feedwater is lost at the beginning of the transient.

- o The reactor is in manual control.

- No credit is taken for a direct reactor trip after the turbine trip.

- The scram table is modeled according to the FSAR Update scram table with total negative reactivity insertion of 4 percent.

- The 1973 ANS decay heat table with 1.2 multiplier is used.

- The Moderator Temperature Coefficient is +5 pcm per degree F and the Doppler Reactivity Coefficient is -0.842 pcm per degree F.

- No credit is taken for the steam dump system and the steam generator PORVs. Secondary pressure is relieved only by the safety valves.

The following additional assumption is used for the case without pressurizer pressure control.

- Pressurizer PORVs and sprays do not operate, whereas pressurizer heaters do operate.

The following additional assumptions are used for the case with pressurizer pressure control.

- Due to the slow pressurizer pressure increase for the case with pressurizer pressure control, the reactor trip setpoints for low-low steam generator water level and over-temperature Delta-T may be reached before the high pressurizer pressure reactor trip setpoint. Conservatively, the reactor trips for low-low steam generator water level and over-temperature Delta-T are assumed unavailable. Then, the reactor will be tripped by the high pressurizer pressure.

- All of the three pressurizer PORVs and the pressurizer spray are assumed available to slow down the pressurizer pressure increase rate.

Analysis Results for The Case Without Pressurizer Pressure Control

Figures VI-1 through VI-5 show the RETRAN results for the case without pressurizer pressure control. The summary of various event timings as well as the peak RCS and steam generator pressures is listed in Table VI-2.

As shown in Figure VI-2, the peak pressurizer pressure during the transient is 2648 psia. As can be seen in Figure VI-3, the peak RCS pressure is 2743 psia. This peak pressure occurs in the lower plenum of the reactor vessel. Both of these pressures are lower than the RCS design pressure limit of 2750 psia.

The peak main steam pressure is 1162 psia as shown in Figure VI-4. As shown in Figure VI-5, the peak steam generator water pressure which is

located at the bottom of the steam generator secondary side is 1172 psia. Both of these pressures are well below the steam generator design pressure limit of 1210 psia.

Analysis Results for The Case With Pressurizer Pressure Control

Figures VI-6 through VI-10 show the RETRAN results for the case with pressurizer pressure control. The summary of various event timings as well as the peak RCS and steam generator pressures is listed in Table VI-3.

As shown in Figure VI-7, the peak pressurizer pressure during the transient is 2586 psia. As can be seen in Figure VI-8, the peak RCS pressure is 2681 psia. Both of these pressures are bounded by the case without pressurizer pressure control.

The peak main steam pressure is 1183 psia as shown in Figure VI-9. As shown in Figure VI-5, the peak steam generator water pressure which is located at the bottom of the steam generator secondary side is 1192 psia. Both of these pressures are higher than those for the case without pressurizer pressure control. However, the peak steam generator pressure is still below the steam generator design pressure limit of 1210 psia.

Based on the loss of external electrical load/turbine trip analysis, the main steam line safety valve tolerance can be changed from +1 percent to +3 percent. Therefore, it is concluded that with respect to overpressure protection, the main steam line safety valve tolerance limit may be relaxed to +3 percent. Relaxing the tolerance in negative direction aids in overpressure protection.

TABLE VI-1

PRESSURIZER AND MAIN STEAM LINE SAFETY VALVE OPENING SETPOINTS

Safety Valve	Setpoints (psia) Nominal	Start Open	Full Open	Time (Second) Delay	Opening
Pressurizer 8010A, 8010B, 8010C	2499.7	2524.7	N/A	1.272	0.1
Main Steam Line RV-3, RV-7, RV-11, RV-58	1079.7	1112.1	1145.5	N/A	N/A
RV-4, RV-8, RV-12, RV-59	1092.7	1125.5	1159.3	N/A	N/A
RV-5, RV-9, RV-13, RV-60	1104.7	1137.8	1172.0	N/A	N/A
RV-6, RV-10, RV-14, RV-61	1117.7	1151.2	1185.8	N/A	N/A
RV-222, RV-223, RV-224, RV-225	1129.7	1163.6	1198.5	N/A	N/A

TABLE VI-2

SUMMARY OF EVENTS
for
The Case Without Pressurizer Pressure Control

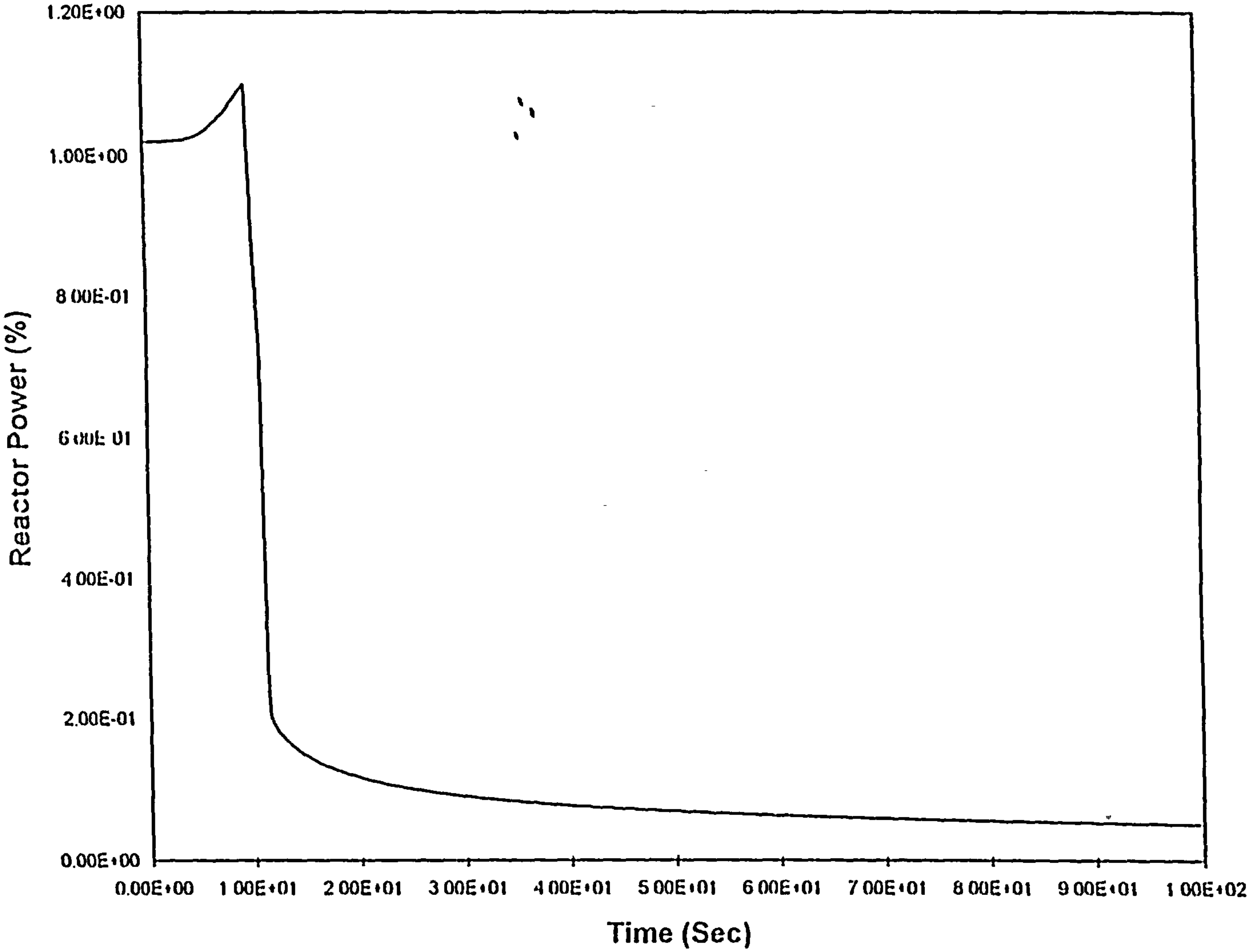
High Pressurizer Pressure Trip Setpoint Reached (sec.)	6.6
Reactor Trip (sec.)	8.6
Pressurizer Safety Valves Open (sec.)	9.2
Peak Primary Pressure Occurs (sec.)	9.7
Main Steam Line Safety Valves With Lowest Setpoint Open (sec.)	9.8
Peak Steam Generator Pressure Occurs (sec.)	16.2
Peak Primary Pressure (psia)	2743
Peak Main Steam Pressure (psia)	1162
Peak steam Generator Water Pressure (psia)	1172

TABLE VI-3

SUMMARY OF EVENTS
for
The Case With Pressurizer Pressure Control

Pressurizer PORVs Open (Sec.)	5.5
Main Steam Line Safety Valves With Lowest Setpoint Open (sec.)	9.9
High Pressurizer Pressure Trip Setpoint Reached (sec.)	12.0
Reactor Trip (sec.)	14.0
Pressurizer Safety Valves Open (sec.)	15.5
Peak Primary Pressure Occurs (sec.)	15.7
Peak Steam Generator Pressure Occurs (sec.)	19.3
Peak Primary Pressure (psia)	2681
Peak Main Steam Pressure (psia)	1183
Peak steam Generator Water Pressure (psia)	1192

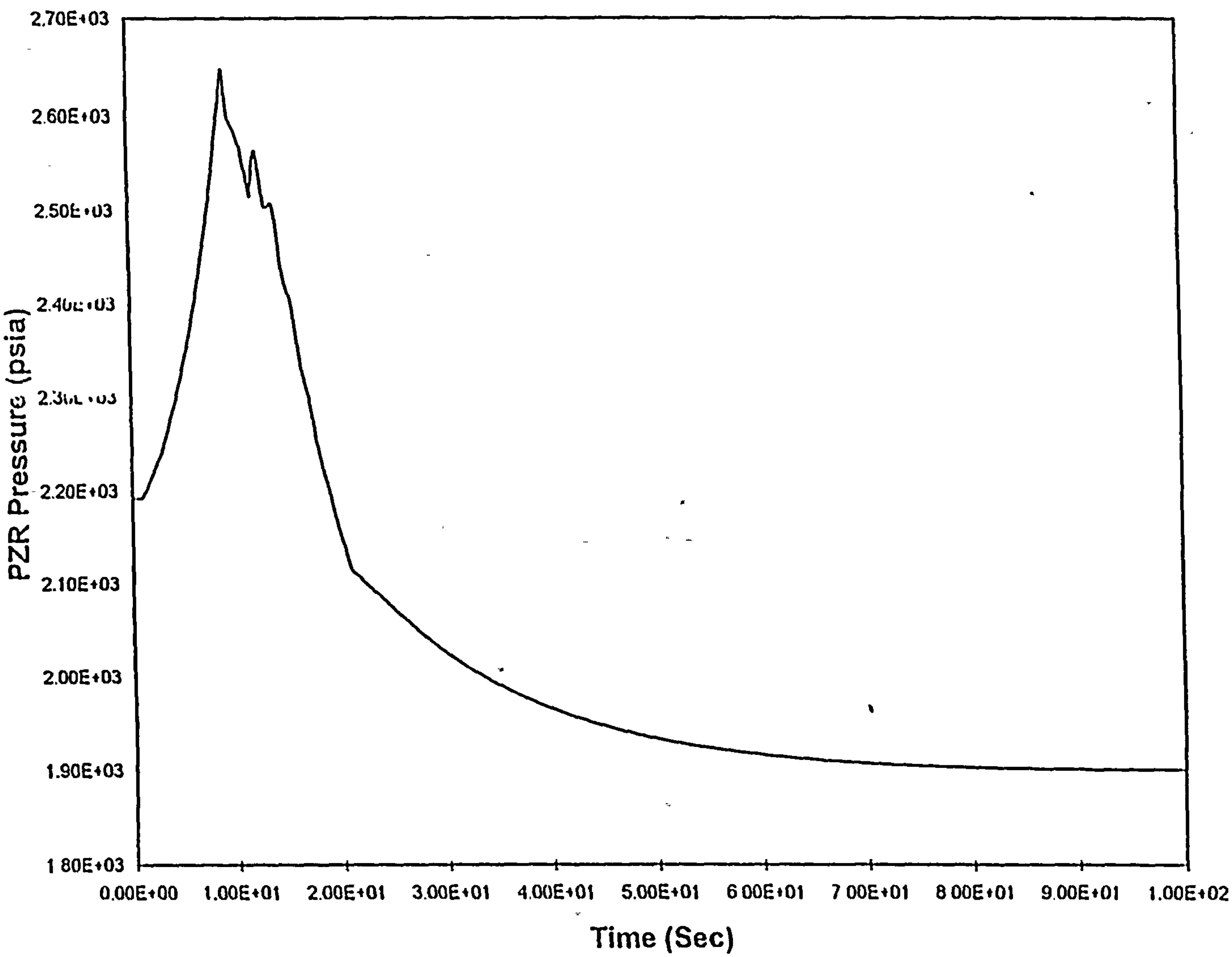
Loss of Load Analysis W/O PZR Pressure Control



LEGEND
NSSV SETPOINT 3% ABOVE NOMINAL

Figure VI-1

Loss of Load Analysis W/O PZR Pressure Control



LEGEND
NSSV SETPOINT 3% ABOVE NOMINAL

Figure VI-2

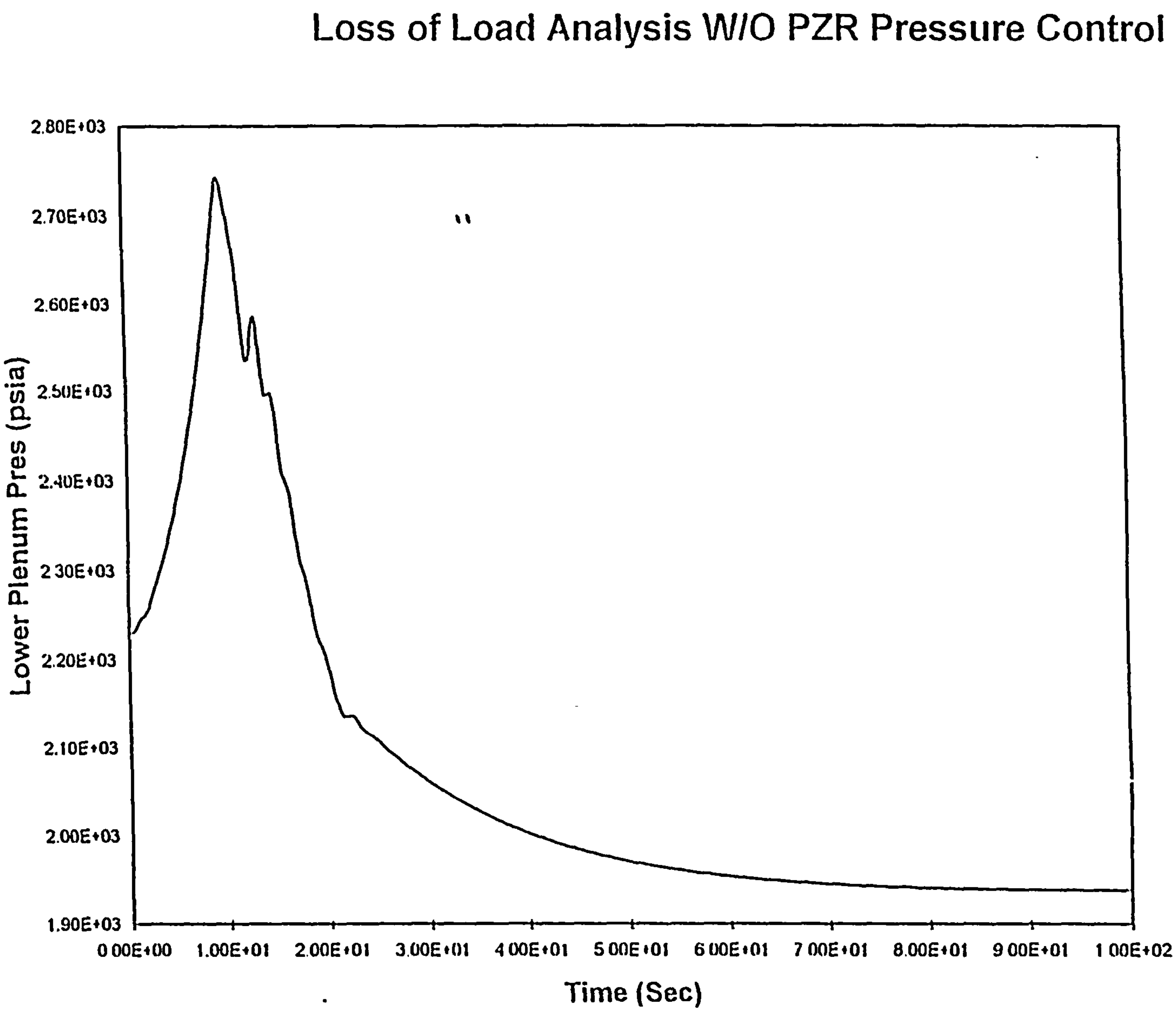
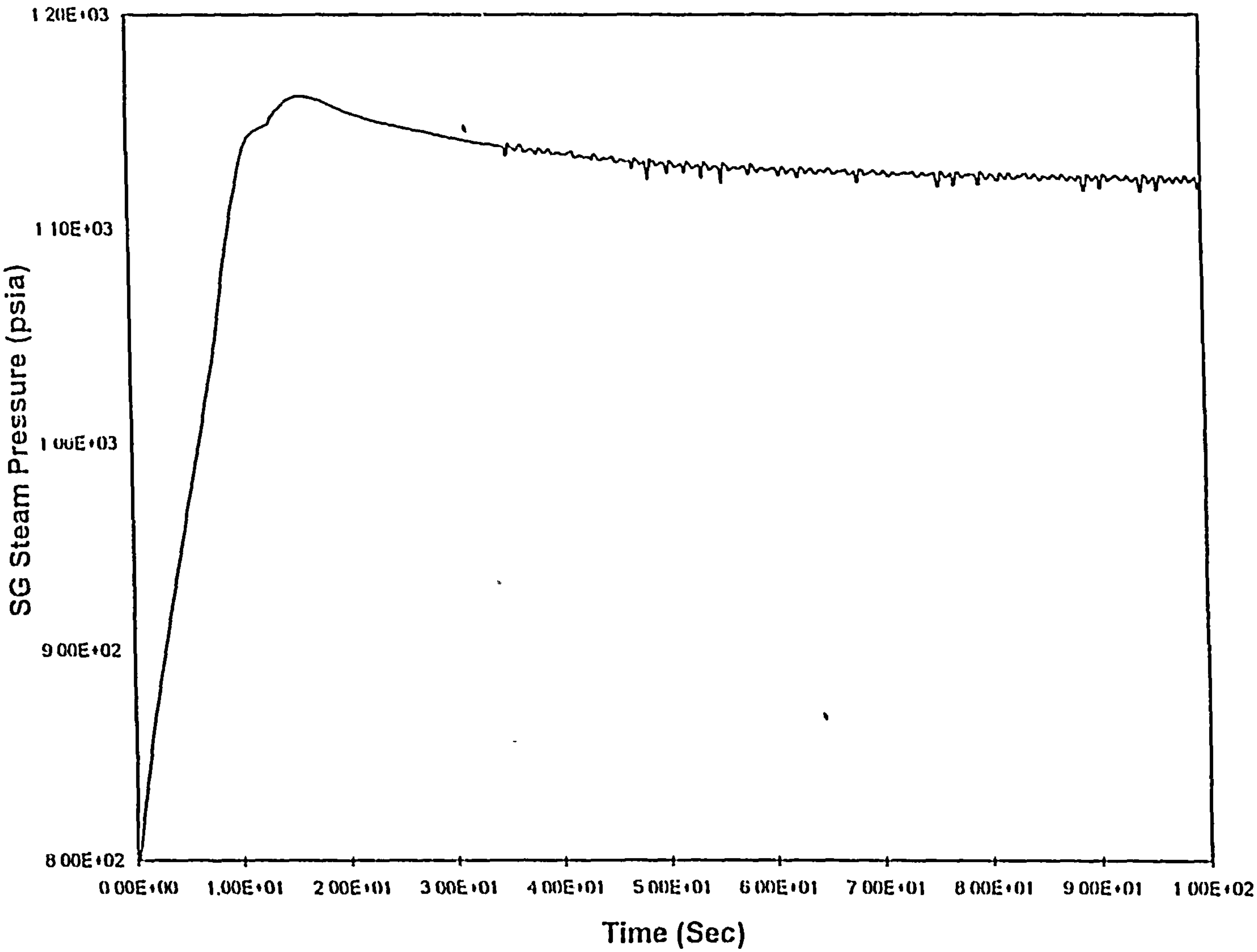


Figure VI-3

LEGEND
NSSV SETPOINT 3% ABOVE NOMINAL

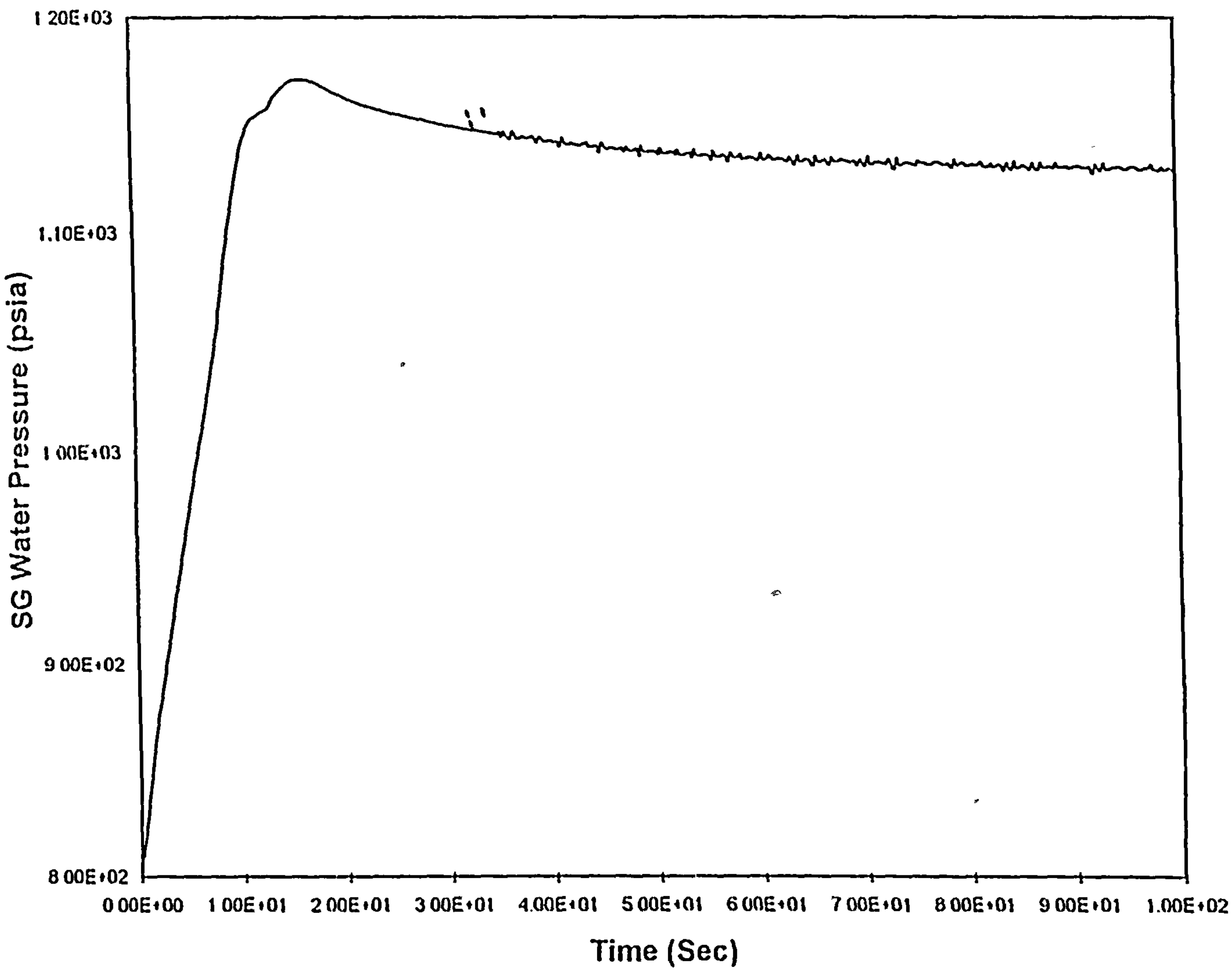
Loss of Load Analysis W/O PZR Pressure Control



LEGEND
MSSV SETPOINT 3% ABOVE NOMINAL

Figure VI-4

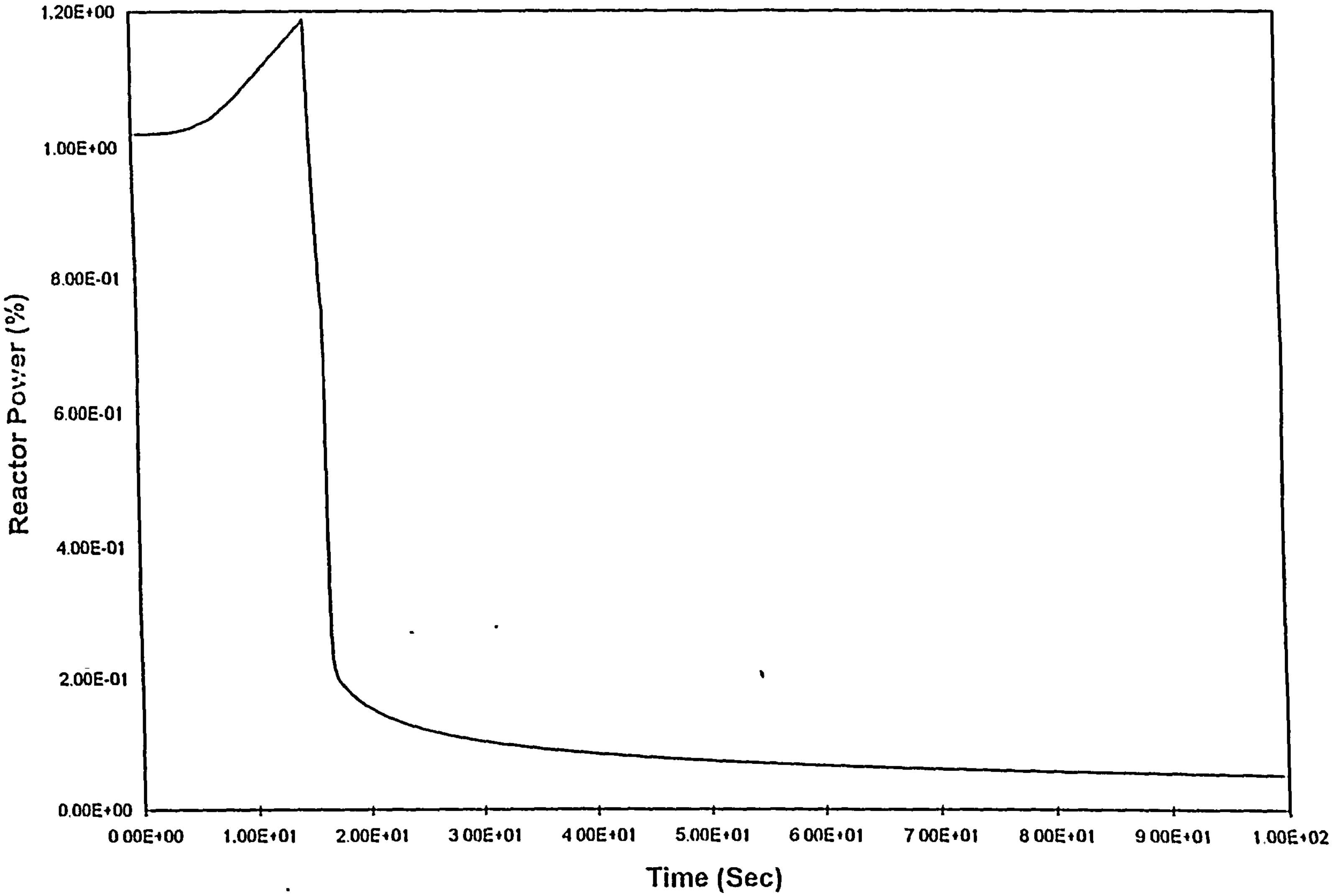
Loss of Load Analysis W/O PZR Pressure Control



LEGEND
MSSV SETPOINT 3% ABOVE NOMINAL

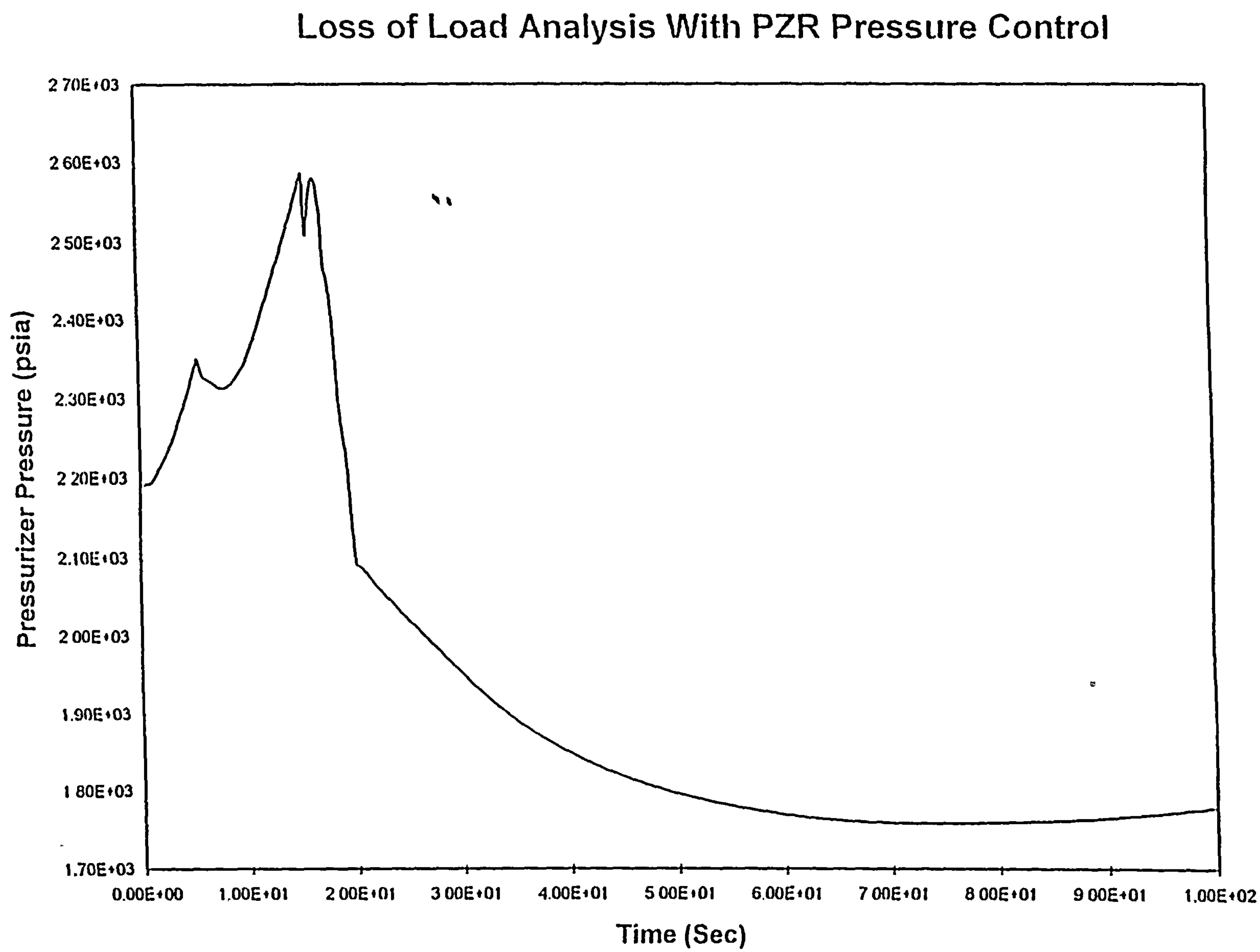
Figure VI-5

Loss of Load Analysis With PZR Pressure Control



LEGEND
NSSV SETPOINT 3% ABOVE NOMINAL

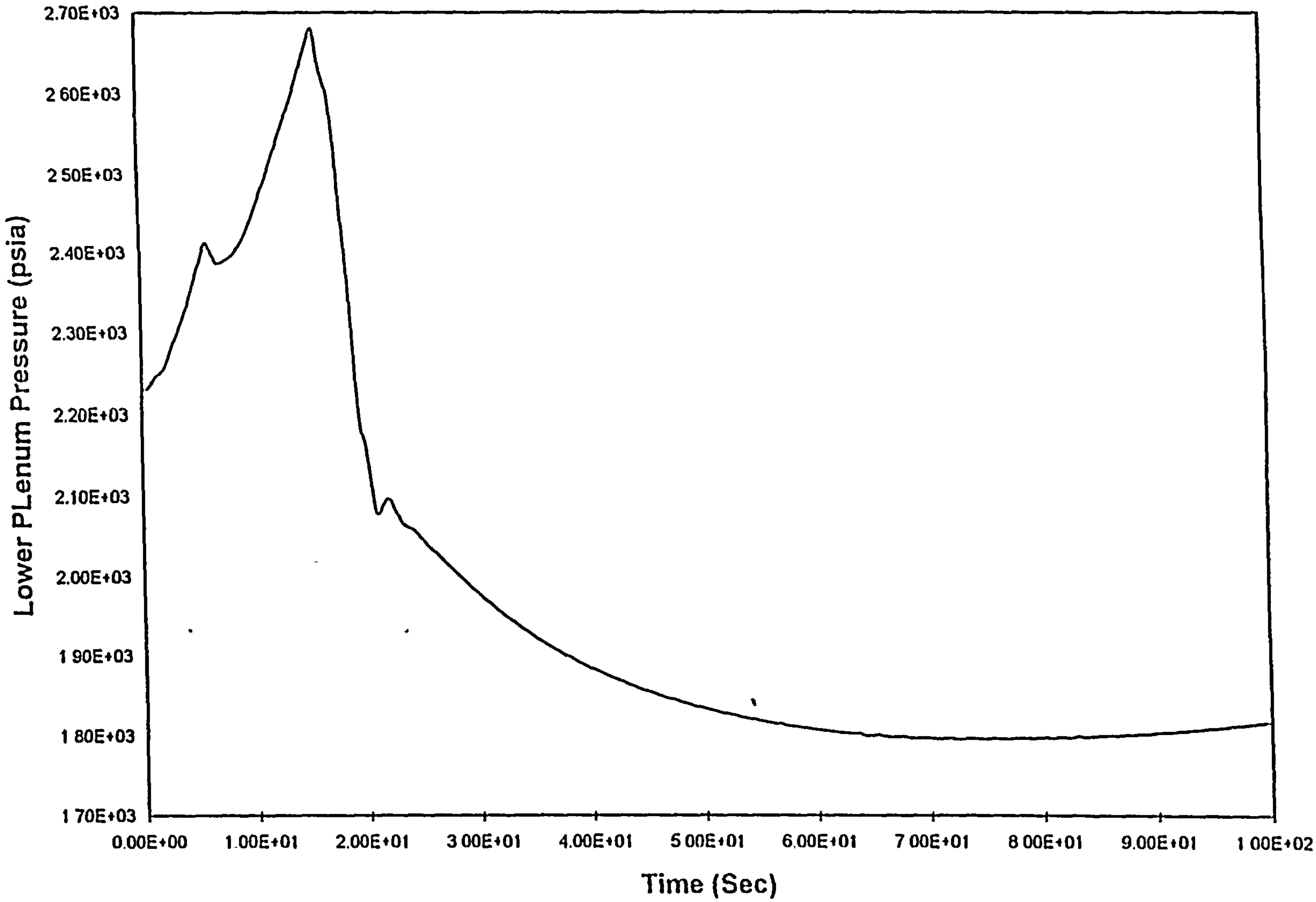
Figure VI-6



LEGEND
NSSV SETPOINT 3% ABOVE NOMINAL

Figure VI-7

Loss of Load Analysis With PZR Pressure Control



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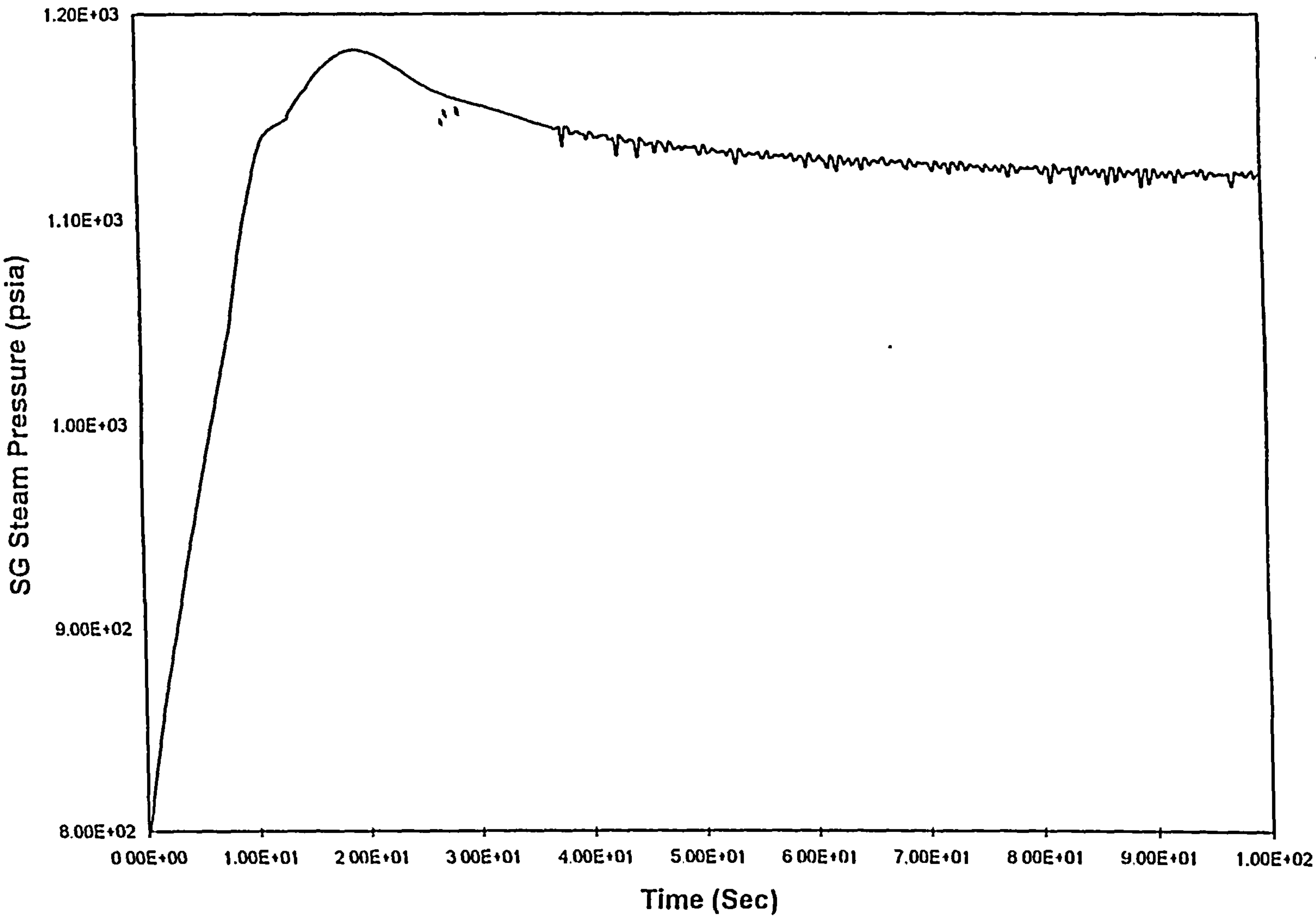
LEGEND
NSSV SETPOINT 3% ABOVE NOMINAL

Figure VI-8

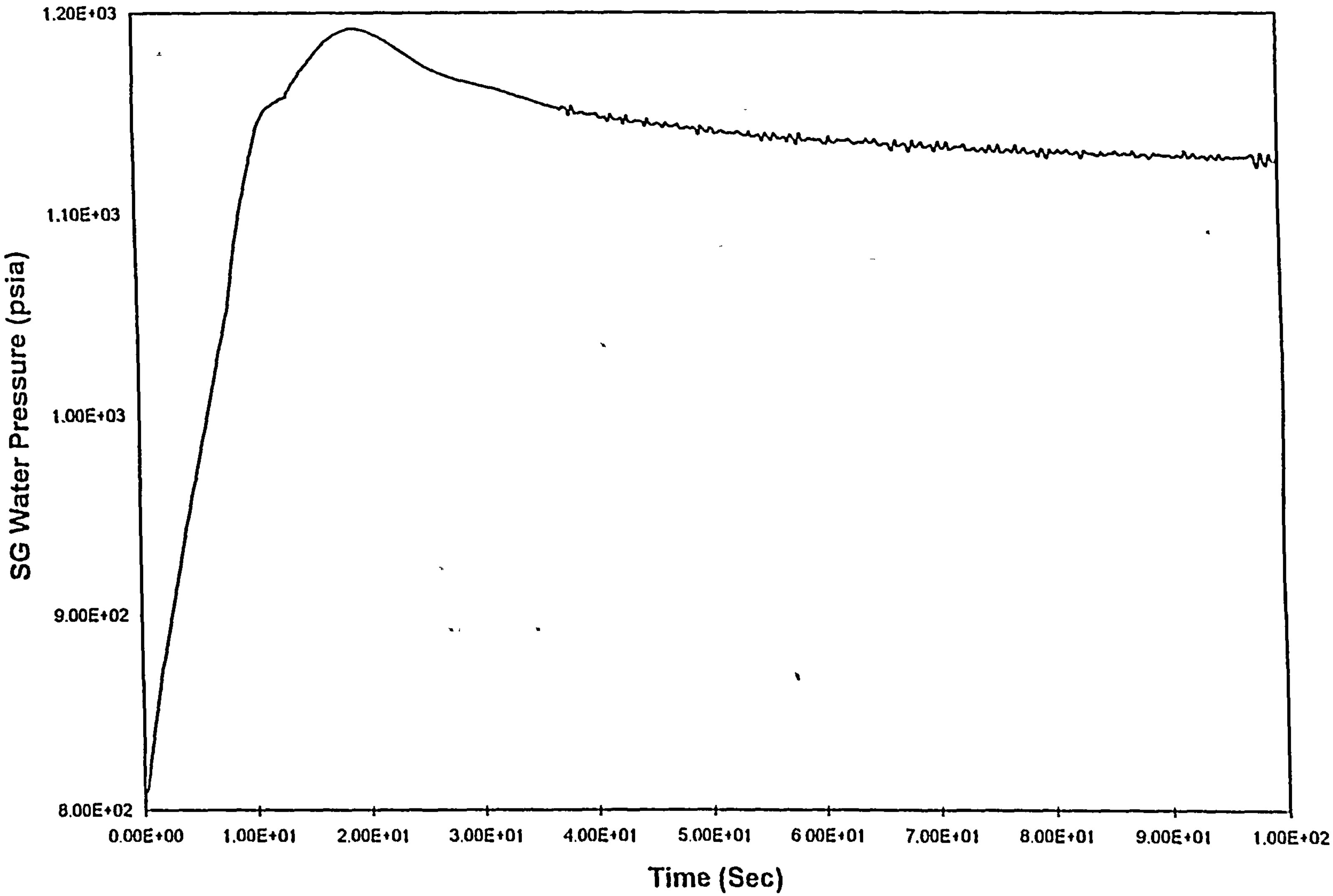
Loss of Load Analysis With PZR Pressure Control



LEGEND
NSSV SETPOINT 3% ABOVE NOMINAL

Figure VI-9

Loss of Load Analysis With PZR Pressure Control



LEGEND
MSSV SETPOINT 3% ABOVE NOMINAL

Figure VI-10

VII. EVALUATION OF FSAR CHAPTER 15 ACCIDENT ANALYSIS

An evaluation of the FSAR Update accident analyses was performed to determine if any adverse impact would result from increased safety valve tolerances. The following is a summary of the FSAR Update accidents evaluation.

Historically, the 1% tolerance of the Main Steam Safety Valve (MSSV) setpoints has been included with respect to the safety analyses. An increase in the tolerance to ±3% for the four highest banks of MSSVs and to -2% and +3% for the lowest bank of MSSVs is considered to be sufficiently significant such that its impact on the safety analyses should be considered.

Modifying either side of the tolerance band potentially affects the safety analyses. The MSSVs provide protection from over-pressurization of the primary and secondary systems. By increasing the positive side of the tolerance band, the pressure at which the safety valves potentially lift and thus the potential maximum pressure attained is increased. By increasing the negative side of the tolerance band, the pressure at which the safety valves potentially lift is decreased.

This safety evaluation assumes that the accumulation point for the MSSVs occurs at a pressure 3% above the actual valve lift setpoint (i.e. stamped pressure plus (minus) 3% plus 3% accumulation pressure to achieve full rated flow). This is more conservative than the ASME code requirement which states that the accumulation point occur within 3% above the nominal valve lift setpoint for the valve.

Many of the accidents are not affected by the assumption of minimum and maximum tolerance on the main steam line safety valve setpoints. This is because the steam generator pressures reached in the transient never approach the safety valve setpoint. The pressure in a given transient would have to reach 1058 psia on the secondary side for the transient to begin to be affected by the increased tolerance. The 1058 psia setpoint reflects the lowest main steam line safety valve setpoint minus the proposed tolerance relaxation.

The secondary side pressure has very little effect on the DNBR. The minimum DNBR is reached before the mean steam safety valve lift for the majority of the transients in the FSAR. For the remaining transients, the impact of the increase of MSSV setpoint tolerances is also insignificant.

Listed below is a detailed assessment of the impact of the increased MSSV's setpoint tolerances on the FSAR analyses:

1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (FSAR Update 15.2.1)

   For this condition II event, rod withdrawal results in a rapid reactivity insertion and increase in core power potentially leading to high local fuel temperatures and heat fluxes and a reduction in the minimum DNBR. The MSSVs are not actuated during the limiting portions of this event since reactor trip and minimum DNBR occur in the first few seconds of the transient prior to any significant changes in secondary conditions. Therefore, the results of this analysis are unaffected by increasing the tolerance on the MSSVs as described above and the conclusions in the FSAR remain valid.

2. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Update 15.2.2)

   For this condition II event, various initial power levels and reactivity insertion rates for both minimum and maximum feedback assumptions are analyzed. The resulting power excursion may lead to high local fuel temperatures and heat fluxes and a reduction in the minimum DNBR. This event is analyzed to demonstrate that the limiting DNBR remains above the limit value.

   The secondary pressure does not reach the reduced MSSV's setpoint during the transient except for slow reactivity insertion rates (approximately less than 7 pcm/sec) at low reactor power. Therefore, the increase of MSSV's setpoint tolerance has no effects on the analysis for high initial reactor power cases. For low initial reactor power cases, the limiting minimum DNBR occurs at the reactor insertion rates much greater than 7 pcm/sec. The effect of the MSSV's setpoint tolerance increase on the limiting minimum DNBR has been evaluated and found to be insignificant. Therefore, the conclusions in the FSAR remain valid.

3. Rod Cluster Control Assembly Misoperation (FSAR Update 15.2.3)

   This condition II event is analyzed to demonstrate that the minimum DNBR remains above the limit value.

   RCCA misoperation accidents include:

   (1) One or more dropped RCCAs within the same group

   (2) A dropped RCCA bank

   (3) Statically misaligned RCCA.

The secondary pressure does not reach the reduced MSSV's setpoint during the event. Therefore, the results of this analysis are not affected by increasing the tolerance on the MSSV's setpoint to ±3% and the conclusions in FSAR remain valid.

4. Uncontrolled Boron Dilution (FSAR Update 15.2.4)

   To cover all phases of the plant operation for this Condition II event, boron dilution during refueling, startup, and power operation is considered in this analysis.

   During refueling and startup adequate operator action time is assured prior to reaching criticality, so no additional heat is added to the core and no pressurization of the primary or secondary systems occurs. Therefore, the increase of the MSSV setpoint tolerances has no impact on RCS or SG conditions.

   For dilution during full power operation with the reactor in automatic control, the power and temperature increase from boron dilution results in the insertion of the RCCAs. The reactivity insertion from the boron dilution is slow enough that the automatic rod control maintains RCS and SG conditions close to nominal. Therefore, SG pressure does not approach the MSSV setpoint and there is no impact on the analysis due to the MSSV setpoint tolerance relaxation.

   Boron dilution with the reactor in manual control and no operator action is essentially identical to a RCCA withdrawal accident at power. The impact of the MSSV setpoint tolerance relaxation on this case is bounded by the discussion presented above for Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power.

   Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR remain valid.

5. Partial Loss of Forced Reactor Coolant Flow (FSAR Update 15.2.5)

   This condition II event is analyzed under full power conditions assuming that 2 of 4 operating reactor coolant pumps coasts down. The reactor is promptly tripped on low reactor coolant loop flow within a few seconds of the event initiation. The MSSVs are not actuated during the limiting portions of this event since reactor trip and minimum DNBR occur in the first few seconds of the transient prior to any significant changes in secondary conditions. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR remain valid.

6. Startup of an Inactive Reactor Coolant Loop (FSAR Update 15.2.6)

This condition II event is analyzed assuming a maximum initial power level consistent with 3 loop operation. The startup of an inactive loop results in a reactivity insertion since the inactive loop fluid is at a lower temperature than the rest of the core. The analysis demonstrates that the minimum DNBR remains above the limit value. The MSSVs are not actuated during the limiting portions of this event since reactor trip and minimum DNBR occur in the first few seconds of the transient prior to any significant changes in secondary conditions. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR remain valid.

7. Loss of External Load and/or Turbine Trip (FSAR Update 15.2.7)

A major load loss on the plant can result from either a loss of external electrical load or from a turbine trip. For either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps.

The analysis presented in the FSAR represents a complete loss of steam load from full power without a direct reactor trip. Four cases are analyzed, Beginning of Life (BOL) and End of Life (EOL) core conditions, with and without pressure control. The minimum DNBRs presented in the FSAR for all 4 cases are well above the DNBR limit. The increase of MSSV lift setpoint tolerances to $\pm 3\%$ has insignificant impact on the DNBR and will not reduce the minimum DNBRs below the FSAR Update limit.

This event is the limiting Condition II event for primary and secondary systems overpressure concern. Among the four cases analyzed in the FSAR Update, the case without pressure control at BOL results in the highest primary system pressure, while the case with pressure control at BOL results in the highest secondary pressure. PG&E reanalyzed these two cases with a +3% MSSV lift setpoint tolerance (plus 3% accumulation pressure) and found the peak primary and secondary pressures for both cases are below 110% of their design pressures. Therefore, it can be concluded that MSSVs with a +3% lift setpoint tolerance can provide adequate overpressure protection and the conclusions in the UFSAR remain valid.

8. Loss of Normal Feedwater (FSAR Update 15.2.8)

The analysis presented in the FSAR represents a complete loss of feedwater from full power. The loss of the secondary side heat sink results in a heatup and pressurization of the primary and secondary systems. The analysis demonstrates that adequate auxiliary feedwater flow is delivered to the steam generators to remove decay heat such that over-pressurization of the primary and secondary systems will not occur and the pressurizer does not

fill. Should the MSSVs actuate at a lift setpoint up to 3% below nominal, the maximum secondary and primary side temperatures and pressures will be beneficially reduced.

Should the MSSVs actuate at a lift setpoint up to 3% above nominal, the peak secondary and primary pressures are bounded by the discussion presented above (7.) for the Loss of External Electrical Load and/or Turbine Trip event. In addition, Westinghouse reanalyzed the event with a +3% MSSV lift setpoint and found all the acceptance criteria are met. Thus, the conclusions in the FSAR remain valid.

9. Loss of Offsite Power to the Station Auxiliaries (Station Blackout) (FSAR Update 15.2.9)

   The analysis presented in the FSAR represents a complete loss of offsite power and a turbine trip resulting in loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power results in a heatup and pressurization of the primary and secondary systems. The analysis demonstrates that adequate auxiliary feedwater flow is delivered to the steam generators to remove decay heat such that DNB will not occur and the peak primary and secondary pressures remain below the design limits.

   Should the MSSVs actuate at a lift setpoint up to 3% below nominal, the maximum secondary and primary side temperatures and pressures will be beneficially reduced. Should the MSSVs actuate at a lift setpoint up to 3% above nominal, the peak secondary and primary pressures are bounded by the discussion presented above (7.) for the Loss of External Electrical Load and/or Turbine Trip event. In addition, Westinghouse reanalyzed the event with a +3% MSSV lift setpoint and found all the acceptance criteria are met. Thus, the conclusions in the FSAR remain valid.

10. Excessive Heat Removal Due To Feedwater System Malfunctions (FSAR Update 15.2.10)

    Reductions in feedwater temperature or excessive feedwater additions are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower-overtemperature protection (neutron high flux, overtemperature Delta-T, and overpower Delta-T trips) prevent any power increase that could lead to a DNBR that is less than the DNBR limit. The analysis demonstrates that for the zero power case, the minimum DNBR is bounded by the rod withdrawal from subcritical event. For the full power case, the minimum DNBR is shown to remain above the limit value.

    Although Tavg increases as a result of increased core power, the increase is not sufficiently large to actuate the MSSVs during

this event even if the MSSV lift setpoints are reduced by up to 3%. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs to ±3% and the conclusions in the FSAR Update remain valid.

11. Sudden Feedwater Temperature Reduction (FSAR Update 15.2.10A)

A reduction in feedwater temperature may be caused by an inadvertent actuation of the load transient bypass relay (LTBR). This would cause the feedwater bypass valve to open, diverting flow around the low pressure feedwater heaters. A consequent reduction in feedwater temperature to the steam generators would occur.

Feedwater temperature may also be reduced during a load rejection trip. The feedwater transient data taken from a 100% net load trip test showed that a maximum feedwater temperature decrease of 230 F occurred over a 400-second time period.

The MSSVs are not actuated during the event even if the MSSVs lift setpoints are reduced by up to 3%. Thus, the results of the analysis are unaffected by increasing the tolerance on the MSSVs to ±3% and the conclusions in the FSAR remain valid.

12. Excessive Load Increase Accident (FSAR Update 15.2.11)

The analysis presented in the FSAR Update describes plant response to a 10% step increase in load. Four different cases are analyzed: minimum and maximum feedback, with and without reactor control. For each case it is shown that the minimum DNBR remains above the limit value. Since an increase in load results in a secondary side pressure reduction the MSSVs are not actuated. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs to ±3% and the conclusions in the FSAR Update remain valid.

13. Accidental Depressurization of the Reactor Coolant System (FSAR Update 15.2.12)

For this Condition II event, the transient is initiated by the opening of a single pressurizer relief or safety valve at full power. Initially, the RCS pressure drops rapidly until pressure reaches the hot leg saturation pressure. At this time the pressure decrease continues but at a slower rate. The analysis demonstrates that the minimum DNBR remains above the limit value. This event is analyzed for 60 seconds in the FSAR. The MSSVs are not actuated during the limiting portion of this event since reactor trip and minimum DNBR occur within the first minute of the transient prior to any significant changes in secondary conditions. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

14. Accidental Depressurization of the Main Steam System (FSAR Update 15.2.13)

For this Condition II event, the transient is initiated by the full opening of a single steam dump, relief, or safety valve at zero power. The analysis confirms that the minimum DNBR remains above the limit value. Since the secondary side pressures drop immediately following initiation of the event, the MSSVs are not actuated. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

15. Spurious Operation of the Safety Injection System at Power (FSAR Update 15.2.14)

For this Condition II event, a spurious Safety Injection signal is assumed to be generated at full power. The injection of borated water into the RCS reduces core power, temperature, and pressure until the reactor trips on low pressurizer pressure. The power and temperature reduction causes a similar reduction in pressure on the secondary side. Since the secondary side pressures drop immediately following initiation of the event, the MSSVs are not actuated. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

16. Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes that Actuate Emergency Core Cooling System (Small Break LOCA) (FSAR Update 15.3.1)

The current small break LOCA analysis forming the licensing basis for Diablo Canyon Units 1 and 2 was performed with the approved Westinghouse Small Break Evaluation Model with NOTRUMP. A small break jn the RCS primary causes the system to depressurize to a pressure slightly above that of the steam generator secondary side pressure relief. The main steam line safety valves provide a significant path for RCS energy release until steam venting through the break occurs. The primary pressure and the duration of time that the RCS primary remains above the secondary side pressure is governed by the rate of decay energy removal through the break and the amount of heat transferred to the steam generator secondary side. A slight increase in RCS pressure is computed to occur during this portion of the transient due to the higher secondary pressure as a result of the relaxed tolerances.

Analysis has shown that increasing the secondary pressure, and as a result, the RCS pressure, results in higher peak cladding temperatures (PCT). The current Diablo Canyon small break analysis assumes nominal main steam line safety valve setpoints. A +3% tolerance on the main steam line safety valve opening pressures will increase secondary pressure and PCT. This increase was evaluated by Westinghouse as a PCT penalty of 117 degrees F.

The current PCTs for Diablo Canyon small break LOCA analysis is substantial below the 10 CFR 50.46 limit of 2200 degrees F. With this PCT increase, the new small break LOCA PCTs for both Diablo Canyon Units 1 and 2 remain below the 10 CFR 50.46 limit. Therefore, the relaxed MSSV setpoint tolerance relaxation is acceptable with respect to the small break LOCA analysis.

17. Minor Secondary System Pipe Breaks (core response) (FSAR Update 15.3.2)

    This Condition III event is bounded by the evaluation presented in item 22 below.

18. Inadvertent Loading of a Fuel Assembly into Improper Position (FSAR Update 15.3.3)

    For the event presented in the FSAR Update, the loading of a fuel assembly into an improper position would affect the core power shape. Since the power shape and not the total power generated would be affected, the steam system conditions will remain unaffected such that the MSSVs would not be affected. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

19. Complete Loss of Forced Reactor Coolant Flow (FSAR Update 15.3.4)

    This Condition III event is analyzed under full power conditions assuming 4 of 4 operating reactor coolant pumps coast down. The reactor is assumed to trip on an undervoltage signal. The analysis demonstrates that the minimum DNBR remains above the limit value. The MSSVs are not actuated during the limiting portion of this event since reactor trip and minimum DNBR occur in the first few seconds of the transient prior to any significant changes in secondary conditions. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

20. Single Rod Cluster Control Assembly Withdrawal at Full Power (FSAR Update 15.3.5)

    For this Condition III event, two cases are analyzed and presented in the FSAR Update: automatic and manual reactor control modes. In both cases an increase in core power, coolant temperature, and hot channel factor result in a reduction in the minimum DNBR. The analysis demonstrates that, although it is not possible for all cases to ensure that DNB will not occur, an upper bound on the number of fuel rods experiencing DNB is less than or equal to 5%.

    Since this event is a limiting DNB event and not peak pressure limiting, credit is not taken for any pressure increase associated with this event. The MSSVs are not actuated during this event.

Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

21. Major Reactor Coolant System Pipe Ruptures (Large Break LOCA) (FSAR Update 15.4.1)

The current large break LOCA analysis forming the licensing basis for Diablo Canyon Units 1 and 2 was performed with the approved Westinghouse Large Break Evaluation Model with BART, including those revisions specified in WCAP-9561-P-A, Addendum 3. A large break LOCA (break size greater than or equal to 1.0 sq. ft.) results in a very rapid (approximately 30 seconds) depressurization of the RCS from the operating pressure to a pressure slightly above that of the containment. The shorter the blowdown and subsequent depressurization, the sooner accumulator flow can begin to refill and cool the core. The duration of the transient is dependent on the rate RCS energy is released through the break and transferred to the steam generator secondary. Because of the rapid primary depressurization, the secondary side of the steam generators quickly becomes a heat source rather than a heat sink such that the MSSVs are not challenged. Therefore, the proposed MSSV setpoint tolerance relaxation will have no effect on the large break LOCA analysis.

Post-LOCA Long-Term Cooling Subcriticality Requirements

The Westinghouse licensing position for satisfying the requirements of 10CFR Part 50 Section 50.46 Paragraph (b) Item (5) "Long Term Cooling" is defined in WCAP-8339. The Westinghouse Evaluation Model commitment is that the reactor will remain shutdown indefinitely by borated emergency core cooling system (ECCS) water residing in the sump following the postulated LOCA and when safety injection (SI) switchover is accomplished. Since credit for the control rods is not taken for large break LOCA, the borated emergency core cooling system (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.

An increase in MSSV setpoint tolerances would not result in any change in the sump boron concentration. Sump boron concentration is determined by the accumulation of all potential water sources in the containment, based on each respective source boron concentration. MSSV operation does not result in spilling additional non-borated water, reduce the inventory of borated water, or limit component boron concentration as used in the mass average calculation used in the evaluation. It is concluded that there would be no change to the long-term cooling capability of the ECCS system as a result of increased MSSV setpoint tolerances.

22. Major Secondary System Pipe Rupture (FSAR Update 15.4.2)

Rupture of a Main Steam Line (FSAR Update 15.4.2.1)

For this Condition IV event, the transient is assumed to be initiated by the instantaneous double-ended rupture of a main steam line. Since the secondary side pressures drop immediately following initiation of the event, the MSSVs are not actuated. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

Environmental Consequences of a Major Steam Pipe Rupture (FSAR Update 15.5.18)

As reported in Section 15.4.2, a major steam line rupture is not expected to cause cladding damage and thus no release of fission products to the coolant is expected following this accident. If significant radioactivity exists in the secondary system prior to the accident, however, some of this activity will be released to the environment with the steam escaping from the pipe rupture. In addition, if an atmospheric steam dump from the unaffected steam generators is necessitated by unavailability of condenser capacity, additional activity will be released.

The amounts of steam released following a major steam line break depend on the size of the break, the time MSSV and atmospheric steam dump valves remain open, and the availability of condenser bypass cooling capacity. All of these factors are not impacted by the increased MSSV setpoint tolerances. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

Major Rupture of a Main Feedwater Pipe (FSAR Update 15.4.2.2)

For this Condition IV event, the double-ended rupture of a main feedwater pipe initially results in a cool down of the RCS due to the heat removal of the steam generator blowdown. This cool down period is followed by a heat up as the high levels of decay heat and the lack of inventory on the secondary side results in inadequate heat transfer. The event is analyzed to show that adequate heat removal capability exists to remove core decay heat and stored energy following a reactor trip from full power and that the core remains in a coolable geometry. This is accomplished by applying the strict criterion that no hot leg boiling occurs during the transient.

For this event, the MSSVs are actuated during the heatup phase following reactor trip. Should the MSSVs actuate at a lift setpoint up to 3% below nominal, the maximum secondary and primary side temperatures and pressures will be beneficially reduced. Should the MSSVs actuate at a lift setpoint up to 3% above

nominal, the peak secondary and primary pressures are bounded by the discussion presented above (7.) for the Loss of External Electrical Load and/or Turbine Trip event. In addition, Westinghouse reanalyzed the event with a +3% MSSV lift setpoint and found all the acceptance criteria are met. Thus, the conclusions in the FSAR remain valid.

23. Steam Generator Tube Rupture (SGTR) (FSAR Update 15.4.3)

The FSAR Update Steam Generator Tube Rupture (SGTR) analysis is performed to evaluate the margin to overfill and radiological consequences of an SGTR accident.

The major factors that affect the resultant offsite doses are the amount of radioactivity in the reactor coolant, the total amount of primary coolant transferred to the secondary side of the ruptured steam generator through the ruptured tube, and the steam released from the ruptured steam generator to the atmosphere. The amount of radioactivity in the reactor coolant assumed in the FSAR Update SGTR analysis is not affected by the changes in the MSSV setpoint tolerances.

An SGTR event results in a depressurization of the RCS due to the continued primary to secondary leakage, and reactor trip and SI actuation occur automatically as a result of low pressurizer pressure. Following reactor trip, the steam dump system is designed to maintain the steam generator secondary pressure approximately at the no-load value. However, if a loss of offsite power occurs, the steam dump system would not be available for removal of the energy transferred from the primary to the secondary. In this event, the steam generator secondary pressure would increase rapidly following reactor trip until the SG ADVs and/or MSSVs lift to dissipate the energy.

The radiological consequences analysis for Diablo Canyon Units 1 and 2 conservatively assumes operation of the SG ADVs to control the secondary pressure. Following reactor trip, the SG ADVs will lift earlier than the MSSVs since the setpoint pressure for the SG ADVs is lower. The safety valves will also lift to remove the primary system stored energy shortly after reactor trip since the steam release rate exceeds the capacity of the SG ADVs. The MSSVs will then blow down to pressures typically 5% below their setpoints. After the first cycle of the safety valves operation, the capacity of SG ADVs alone are sufficient to remove the decay heat. Thus, the secondary pressure is assumed to be maintained at the SG ADV setpoint. The major effect of a decreased safety valve setpoint is a slightly earlier safety valve lift/blowdown operation. The steam released from the ruptured SG is expected to be approximately the same since the primary system stored energy and the core decay heat are not affected. Thus, relaxing the lowest main steam line safety valve setpoint tolerances to -2% and

+3% would have an insignificant effect on the SGTR radiological consequences analysis.

For margin to overfill analysis, a review of the analysis indicates that the steam release rate following reactor trip is less than the steam generator PORV capacity. If the tolerance for the lowest main steam safety valve setpoint is changed to -2% and +3%, the steam generator PORV setpoint pressure remains below the lowest safety valve setpoint with the setpoint tolerance in the negative direction. The MSSVs would not open for the margin to overfill analysis. Therefore, it is concluded that there would be no change in the SGTR margin to overfill with the increased MSSV setpoint tolerances and the conclusions in the FSAR remain valid.

24. Single Reactor Coolant Pump Locked Rotor (FSAR Update 15.4.4)

This Condition IV event is analyzed under full power conditions assuming the instantaneous seizure of one Reactor Coolant Pump motor. This results in a rapid RCS flow reduction and pressure rise with possible DNB. The reactor is promptly tripped on a low flow signal. The analysis demonstrates that less than 10% of the rods experience DNB, that the RCS peak pressure remains below that which would cause stresses to exceed the faulted condition stress limits, and the peak cladding surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700 F and the amount of zirconium-water reaction is small, the core will remain in place and intact with no consequential loss of core cooling capability.

The MSSVs are not actuated during the limiting portions of this event since reactor trip, peak RCS pressure, and maximum fuel and cladding temperatures occur in the first few seconds of the transient prior to any significant changes in secondary conditions. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

25. Fuel Handling Accident (FSAR Update 15.4.5)

The reactor coolant and secondary systems are not involved in the analysis of a fuel handling accident. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

26. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (FSAR Update 15.4.6)

For this Condition IV event, a rapid reactivity insertion and increase in core power leads to high local fuel and clad temperatures and possible fuel and/or clad damage. Four cases are analyzed: beginning of life (BOL), end of life (EOL), hot zero power, and hot full power. The analysis shows that the fuel and

clad limits discussed in FSAR Update Section 15.4.6 are not exceeded and that RCS pressure does not exceed the faulted condition stress limits. The MSSVs are not modeled as part of this analysis and would not actuate during the limiting portions of this event since reactor trip, peak RCS pressure, and maximum fuel and cladding temperatures occur in the first few seconds of the transient prior to any significant changes in secondary conditions. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

27. Rupture of a Waste Gas Decay Tank (FSAR Update 15.4.7)

    The reactor coolant and secondary systems are not involved in the analysis of a waste gas decay tank rupture accident. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

28. Rupture of a Liquid Holdup Tank (FSAR Update 15.4.8)

    The reactor coolant and secondary systems are not involved in the analysis of a liquid holdup tank rupture accident. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

29. Rupture of Volume Control Tank (FSAR Update 15.4.9)

    The reactor coolant and secondary systems are not involved in the analysis of a volume control tank rupture accident. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

30. Other FSAR Events

    a. CONTAINMENT ANALYSES

       Steamline Break Mass/Energy Releases - Inside/outside Containment

       Various steam line break cases are analyzed for the purposes of generating mass and energy release rates which are then applied to containment response or compartment environmental analyses. Cases are performed assuming various break sizes and initial power levels. For small breaks occurring at high power levels, it is possible that pressurization of the primary and secondary systems may occur. Specifically, if the energy release through the break is less than the decay heat addition into the RCS, pressurization may occur to the point of safety valve actuation. However, since the relief capacity of the MSSVs is

undiminished, there is sufficient capacity to prevent over pressurization of the secondary systems. Raising the MSSV setpoints will have an insignificant impact upon the mass and energy releases previously calculated. A small reduction in the MSSV setpoints will serve to reduce the primary and secondary side temperatures and energy release rates. Thus, the results of the calculated mass and energy releases are not adversely affected by increasing the tolerance on the MSSVs to +/- 3% and the conclusions in the FSAR remain valid.

b. Hot Leg Switchover to Prevent Potential Boron Precipitation (FSAR Update Chapter 6.3.2.5)

Post-LOCA hot leg recirculation switchover time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level, boron concentrations, and water volumes of the RCS, RWST, and accumulators. Since the secondary safety valves affect neither the maximum boron concentrations nor the volumes assumed for the RCS, RWST, and accumulators, there is no effect on the post-LOCA hot leg switchover time.

c. Blowdown Reactor Vessel and Loop Forcing Functions (FSAR Update 3.9.3)

The LOCA hydraulic forcing functions acting upon the vessel, internals, and loop are a function of the primary system geometry and primary operating conditions. The peak forces are generated within the first seconds after break initiation. For this reason, the forces model does not consider the effects of the secondary side conditions. As such, the relaxation in MSSV setpoint tolerances will have no effect on the magnitude or frequency of the LOCA hydraulic forcing functions as given in the Diablo Canyon FSAR Update.

d. Evaluation of Auxiliary Feedwater Flow (AFW)

The minimum required AFW flows from the turbine driven and motor driven AFW pumps are 820 gpm and 410 gpm, respectively. The AFW pumps have enough pump head to deliver the minimum required AFW flows with the additional steam generator back pressure resulting from an +3 percent MSSV setpoint tolerance. Therefore, the increase of the MSSV setpoint tolerance does not affect the AFW system's ability to meet its required performance. Should the MSSVs actuate at a lift setpoint below the nominal value, the steam generator back pressure will be reduced and the AFW flow will be beneficially increased.

e. Environmental Consequences of Condition II Faults (FSAR Update 15.5.10)

As reported in FSAR Update Section 15.2, none of the Condition II faults are expected to cause breach of any of the barriers preventing fission product release from the core or plant. Under some conditions, however, small amounts of radioactive isotopes could be released to the atmosphere following Condition II events as a result of atmospheric steam dumps required for plant cooldown.

The amount of steam released following these events depends on the time the relief valves remain open and the availability of condenser bypass cooling capacity.

Approximately 1.6×10^6 lbm of steam is the maximum steam release expected for a full cooldown without the condenser available, and a steam release of approximately 1×10^5 lbm would result from releasing the contents of one steam generator due to a safety valve failure to close or steam line break with condenser cooling available. This bounds the steam releases for all of the Condition II events. Since the total energy available for release is not significantly changed by the increased MSSV setpoint tolerance relaxation, these values are not impacted. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSVs and the conclusions in the FSAR Update remain valid.

f. Overtemperature and Overpower Delta-T Setpoints

Increasing the MSSV safety valve tolerance will impact the core limits and the Overpower and Overtemperature protection. Figure 15.1-1 of the FSAR Update, shows the Overtemperature Delta-T and Overpower Delta-T protection terminated at the MSSV setpoint. This is because the temperature drop across the steam generator, primary to secondary, is approximately proportional to power. The secondary temperature is approximately constant at the saturation temperature corresponding to the MSSV setpoint. Therefore, the primary temperature cannot rise above the MSSV setpoint saturation temperature plus the temperature drop across the steam generator. This temperature limit serves as one of the boundaries on power and temperature in addition to the bounds imposed by the Overpower and Overtemperature trip setpoints.

By increasing the MSSV setpoint by 3%, the saturation temperature is increased by less than 4 degrees F. By decreasing the MSSV setpoint by 3%, the saturation temperature is decreased by less than 4 degrees F. Examination of Figure 15.1-1 reveals that the Overtemperature Delta-T provides the primary DNBR protection at high RCS average temperature and the margin between the DNBR limit and the Overtemperature Delta-T

protection increases as the MSSV setpoint increases. A reduction in the saturation temperature will reduce the operational margin in which the delta-T protection system must provide DNB protection (i.e. actuate earlier in an operational transient). Thus, the increase of the MSSV setpoint tolerances does not impact the Overpower and Overtemperature Delta-T setpoints to provide adequate protection from core limits.

In conclusion, PG&E believes there is reasonable assurance that the health and safety of the public will not be adversely affected by the proposed MSSV setpoint tolerance increase.

VIII. OTHER ISSUES

ASME Code Issues

The Diablo Canyon main steam line safety valves were procured and constructed to specifications and design codes which did not specify tolerances for the set pressure.

Subsequent to this (and in agreement with then-current construction Code requirements), the DCPP TS were written with a tolerance on these valves' set pressures of +/-1 percent.

The DCPP IST Program is based on ASME Section XI, 1977 Edition through the Summer of 1978 Addenda. ASME Section XI, IWV-3500, in turn requires that safety valve setpoints be tested in accordance with ASME Power Test Code (PTC) 25.3-1976. However, neither Section XI nor the PTC provides any requirements or guidance for tolerances on the set pressure for periodic testing. PG&E has continued to use the TS tolerance of +/-1 percent for the IST Program testing.

More recent editions of Section XI (beginning with the Winter of 1985 Addenda) have endorsed ANSI/ASME OM-1-1981 requirements for IST of such safety valves. OM-1-1981 acknowledges that a drift in the setpoint of up to 3 percent would not be considered an unreasonable deviation from the stamped set pressure.

It is PG&E's position that a setpoint tolerance of +/-3 percent is consistent with the requirements of OM-1-1981 and is compatible with the Code requirements of the present Diablo Canyon IST Program and valve design documents.

Further, ASME Section XI, IWV-3500, requires that if any tested valve fails to meet the 3 percent criteria, additional valves shall be tested. DCPP's present IST Program complies with these requirements and will continue unchanged with a setpoint acceptance criterion revised as discussed previously. In addition, PG&E will assure that all valves tested will be reset to within +/-1 percent of nominal setpoint.

Operational Issues

The proposed tolerance relaxation would decrease the margin for the lowest main steam line safety valve set point to the normal operation steam generator pressure from 264 psi to 253 psi. This slight reduction of pressure margin is not judged to be a problem in terms of valve leakage. With the proposed tolerance in the negative side, the lowest main steam line safety valve set point would be 1043.7 psig which remains higher than the steam generator pressure of 1005 psig at no load condition and the nominal set point of 1035 psig for MSPORV. Therefore, the proposed tolerance relaxation does not affect the plant operation.

IX. CONCLUSIONS AND RECOMMENDATIONS

An overpressure analysis has been performed by using the RETRAN computer code with the same set of conservative assumptions used in the FSAR Update. The ASME Code requirements for overpressure protection and safety valve testing have been reviewed. A comprehensive safety evaluation of all of the FSAR Update accidents with relaxed mean steam line safety valve tolerances was performed. Potential operational concerns due to main steam line safety valve setpoint tolerance changes have been examined. Based on these efforts, the following conclusions are reached:

The loss of external electrical load/turbine trip is the limiting FSAR Update transient for DCPP with respect to overpressure protection. A safety analysis performed on the loss of external electrical load/turbine trip for DCPP using FSAR Update assumptions does not exceed this limiting pressure (110 percent of design) and therefore, by itself, justifies relaxation of the main steam line safety valve tolerances to +3 percent.

ASME Section XI Boiler and Pressure Vessel Code dictates that the upper end of the pressure setpoint tolerances on the safety valves be +3 percent above the setpoint.

A Safety Evaluation on all FSAR Update Chapter 15 accidents justifies relaxation of the four lowest setpoint bank main steam line safety valves to -2/+3 percent and all other main steam line safety valves to +/-3 percent.

Operational considerations permit the lower limit of main steam line safety valve tolerances as proposed.

Based on the aforementioned Safety Evaluation, ASME Code issues, and operational concerns, it is recommended that the following valve tolerance relaxations are implemented:

Main Steam Line Safety Valves

First Stage Main Steam Line Safety Valves	+3%, -2%
Second Stage Main Steam Line Safety Valves	+/-3%
Third Stage Main Steam Line Safety Valves	+/-3%
Fourth Stage Main Steam Line Safety Valves	+/-3%
Fifth Stage Main Steam Line Safety Valves	+/-3%

Attachment E.1

PG&E RESPONSE TO RETRAN SAFETY EVALUATION REPORT RESTRICTIONS

ATTACHMENT E.1
PG&E RESPONSE TO RETRAN SAFETY EVALUATION REPORT RESTRICTIONS

As noted above, the Nuclear Regulatory Commission (NRC) staff reviewed the RETRAN02/MOD004 computer code and issued a SER. In the SER, the NRC staff concluded that RETRAN02/MOD004 is acceptable for the use in transient analyses with the following restrictions. The PG&E response to each of the restrictions is listed after a restatement of the restriction.

Restriction 1. "The RETRAN code is a generically flexible computer code requiring users to develop their own nodalization and select from optional models in order to represent the plant and transients being examined. Thus, as specified in the original SER (Ref. 1), RETRAN users should include a discussion in their submittals as to why the specific nodalization scheme and optional models chosen are adequate. These should be performed on a transient by transient basis."

Response: The DCPP Unit 2 RETRAN results were compared to the DCPP Unit 2 startup test data acquired during the turbine trip from 100 percent power as described in Section IV of this report. Furthermore, the RETRAN model was used to reproduce the loss of load transient analysis contained in Chapter 15 of the DCPP FSAR Update. This is described in Section V of this report. These analyses provide sufficient confidence that the nodalization and optional models chosen are adequate to perform a safety analysis for the loss of external electrical load/turbine trip transient with relaxed safety valve tolerances.

Restriction 2. "Restrictions imposed on the use of RETRAN02 models (including the separator model, boron transport, jump pump and range of applicability, etc.) in the original SER (Ref. 1) have not been addressed in the GPU submittal therefore remain in force for both MOD003 and MOD004."

Response: The limitations imposed on the use of RETRAN02 models in the original SER are listed as follows. The responses to each of the limitations are presented immediately following the limitations.

Limitations:

a. "Multi-dimensional neutronic space time effects cannot be simulated as the maximum number of dimensions is one. Conservative usage has to be demonstrated."

Response: The multi-dimensional neutronic space time model is not used in the DCPP\Unit\2 RETRAN02 model. Therefore, this concern is not applicable.

b. "There is no source term in the neutronics models and the maximum number of energy groups is two. The space time option assumes an initially critical system. Initial conditions with zero fission power cannot be simulated by the kinetics. The neutronic models should not be started from subcritical or with zero fission power without further justification."

Response: The loss of load transient is initiated from full power. Therefore, this concern is not applicable.

c. "A boron transport model is unavailable. User input models will have to be reviewed on an individual basis."

Response: The boron transport model is not used in the DCPP Unit 2 RETRAN02 model because neither boration or dilution occurs in the loss of load transient. Therefore, this concern is not applicable.

d. "Moving control rod banks are assumed to travel together. The BWR plant qualification work shows that this is an acceptable approximation."

Response: Control rod movement is not credited in the loss of load transient analysis. Therefore, this concern is not applicable.

e. "The metal-water heat generation model is for slab geometry. The reaction rate is therefore underpredicted for cylindrical cladding. Justification will have to be provided for specific analyses."

Response: There is no metal-water heat generation in the loss of load transient. Therefore, this concern is not applicable.

f. "Equilibrium thermodynamics is assumed for the thermal hydraulics field equations although there are nonequilibrium models for the pressurizer and the subcooled boiling region."

Response: The RETRAN02 model was used to simulate the DCPP Unit 2 turbine trip test. Excellent agreement between the measured data and RETRAN results was obtained for the pressurizer pressure and level. This demonstrates that the RETRAN02 model for DCPP can accurately simulate the pressurizer during a loss of external electrical load/turbine trip transient. Subcooled boiling does not occur during the entire loss of load transient.

g. "While the vector momentum model allows the simulation of some vector momentum flux effects in complex geometry the thermal hydraulics are basically one dimensional."

Response: The vector momentum model is not used in the RETRAN02 model. Therefore, this concern is not applicable.

h1. "Further justification is required for the use of the homogeneous slip option with BWRs."

Response: DCPP is a PWR plant. Therefore, this concern is not applicable.

h2. "The drift flux correlation used was originally calibrated to BWR situations and the qualification work for both this option and for the dynamic slip option only cover BWRs. The drift flux option can be approved for BWR bundle geometry if the conditions of (n2) are met."

Response: The drift flux correlation is not used in the RETRAN02 model. Therefore, this concern is not applicable.

i. "The profile effect on the interphase drag (among all the profile effects) is neglected in the dynamic slip option. Form loss is also neglected for the slip velocity. For the acceptability of these approximations refer to (n3)."

Response: The dynamic slip option is not used in the DCPP2 RETRAN02 model. Therefore, this concern is not applicable.

j. "Only one dimensional heat conduction is modeled. The use of the optional gap linear thermal expansion model requires further justification."

Response: The multiple dimensional heat conduction is not modeled; the RETRAN02 model uses a conservative, constant gap conductance.

k. "Air is assumed to be an ideal gas with a constant specific heat representative of that at containment conditions. It is restricted to separated and single phase vapor volumes. There are no other non-condensibles."

Response: In the RETRAN02 model, air exists only in the volume modeling the containment. Since the flows through the safety valves are choked and no other leak flow goes to the containment during the entire loss of external electrical load/turbine trip transient, the containment condition does not affect the RETRAN results for the transient.

l. "The use of water properties polynomials should be restricted to the subcritical region. Further justification is required for other regions."

Response: The water properties stay within the subcritical region during the entire loss of load transient. Therefore, the concern is not applicable.

m. "A number of regime dependent minimum and maximum heat fluxes are hardwired. The use of the heat transfer correlations should be restricted to situations where the pre-CHF heat transfer or single phase heat transfer dominates."

Response: The heat fluxes of the entire fluid system never reach critical heat fluxes during the entire loss of load transient. Therefore, the concern is not applicable.

n1. "The Bennett flow map should only be used for vertical flow within the conditions of the data base and the Beattie two-phase multiplier option requires qualification work."

Response: The Bennett flow map and the Beattie two-phase multiplier option are not used in the RETRAN02 model. Therefore, this concern is not applicable.

n2. "No separate effects comparisons have been presented for the algebraic slip option and it would be prudent to request comparisons with the FRIGG tests (5) before the approval of the algebraic slip option."

Response: The algebraic slip option is not used in the RETRAN02 model. Therefore, this concern is not applicable.

n3. "While FRIGG tests (5) comparisons have been presented for the dynamic slip option the issues concerning the Shrock-Grossman round tube data comparisons should be resolved before the dynamic slip option is approved. Plant comparisons using the option should also be required."

Response: The dynamic slip option is not used in the RETRAN02 model. Therefore, this concern is not applicable.

o. "The nonequilibrium pressurizer model has no fluid boundary heat losses, cannot treat thermal stratification in the liquid region and assumes instantaneous spray effectiveness and a constant rainout velocity. A constant L/A is used and flow detail within the component cannot be simulated. There will be a numerical drift in energy due to the inconsistency between the two region and the mixture energy equations but it should be small. No comparisons were presented involving a full or empty pressurizer. Specific application of this model should justify the lack of fluid boundary heat transfer on a conservative basis."

Response: The pressurizer is never full or empty during the entire loss of load transient. Since the loss of load transient is

a heat up transient, the assumption of no heat loss to the pressurizer wall provides a more conservative result for an overpressure transient.

p. "The nonmechanistic separator model assumes quasi-statics (time constant - few tenths seconds) and uses GE BWR 6 carryover/carryunder curves for default values. Use of the default curves has to be justified for specific applications. As with the pressurizer a constant L/A is used. The treatment in the off normal flow quadrants is limited and those quadrants should be avoided. Attenuation of pressure waves at low flow/low quality conditions are not simulated well. Specific application to BWR pressurization transients under those conditions should be justified."

Response: DCPP is a PWR plant. Therefore, this concern is not applicable.

q. "The centrifugal pump head is divided equally between the two junctions of the pump volume. Bingham pump and Westinghouse pump data are used for the default single phase homologous curves. The SEMISCALE MOD-1 pump and Westinghouse Canada data are used for the degradation multiplier approach in the two phase regime. Use of the default curves has to be justified for specific applications. Pump simulation should be restricted to single phase conditions."

Response: The homologous curves for Westinghouse model 93A-1 7000 HP RCP are used in the RETRAN02 model. During the entire loss of load transient, only single-phase water flows through the reactor coolant pumps. Therefore, justifying the two-phase pump model is not applicable for the loss of load transient. No pump other than the RCPs are modeled in the RETRAN02 model.

r. "The jet pump model should be restricted to the forward flow quadrant as the treatment in the other quadrants is conceptually not well founded. Specific modeling of the pump in terms of volumes and junction is at the user's discretion and should therefore be reviewed with the specific application."

Response: DCPP does not have jet pumps. Therefore, this concern is not applicable.

s. "The nonmechanistic turbine model assumes symmetrical reaction staging, maximum stage efficiency at design conditions, a constant L/A and a pressure behavior dictated by a constant loss coefficient. It should only be used for quasistatic conditions and in the normal operating quadrant."

Response: The turbine model is not used in the DCPP2 RETRAN02 model. Therefore, this concern is not applicable.

t. "The subcooled void model is a nonmechanistic profile fit using a modification of EPRI recommendations (4) for the bubble departure point. It is used only for the void reactivity computation and has no direct effect on the thermal hydraulics. Comparisons have only been presented for BWR situations. The model should be restricted to the conditions of the qualification data base. Sensitivity studies should be requested for specific applications. The profile blending algorithm used will be reviewed when submitted as part of the new manual (MOD03) modifications."

Response: The subcooled void model is not used in the RETRAN02 simulation of loss of load events. Therefore, this concern is not applicable.

u. "The bubble rise model assumes a linear void profile; a constant rise velocity (but adjustable through the control system); a constant L/A; thermodynamic equilibrium and makes no attempt to mitigate layering effects. The bubble mass equation assume zero junction slip which is contrary to the dynamic and algebraic slip model. The model has limited application and each application must be separately justified."

Response: A sensitivity study was performed for bubble rise velocity. The results showed that the effects of bubble rise velocity are not significant for the loss of load/turbine trip event.

v. "The transport delay model should be restricted to situations with a dominant flow direction."

Response: The transport delay model is not used in the RETRAN02 simulation of the loss of load transient. Therefore, the concern is not applicable.

w. "The stand alone auxiliary DNBR model is very approximate and is limited to solving a one dimensional steady state simplified HEM energy equation. It should be restricted to indicating trends."

Response: The auxiliary DNBR model is not used in the RETRAN02 model. Therefore, this concern is not applicable.

x. "Phase separation and heat addition cannot be treated simultaneously in the enthalpy transport model. For heat addition with multidirectional, multijunction volumes the enthalpy transport model should not be used without further justification. Approval of this model will require submittal of the new manual (MOD03) modifications."

Response: The enthalpy transport model is not used for multi-directional, multi-junction volumes in the RETRAN02 model. Therefore, this concern is not applicable.

y. "The local conditions heat transfer model assumes saturated fluid conditions, one dimensional heat conduction and a linear void profile. If the heat transfer is from a local conditions volume to another fluid volume, that fluid volume should be restricted to a nonseparated volume. There is no qualification work for this model and its use will therefore require further justification."

Response: The local conditions heat transfer model is not used in the RETRAN02 model. Therefore, this concern is not applicable.

z. "The initializer does not absolutely eliminate all ill-posed data and could have differences with the algorithm used for transient calculations. A null transient computation is recommended. A heat transfer surface area adjustment is made and biases are added to feedwater inlet enthalpies in order to satisfy steady state heat balances. These adjustments should be reviewed on a specific application basis."

Response: A null transient run was carried out using the same RETRAN initial conditions as those used for the FSAR Update loss of load transient analysis. The steady state conditions were held throughout the entire null transient run of 30 seconds. Since the feedwater is assumed to be isolated during the entire loss of load transient, the biases imposed on the feedwater inlet enthalpy do not affect the RETRAN simulation of the FSAR Update loss of load transient.

"In addition, for PWR systems analysis the following items require further justification:

i) Justification of the extrapolation of FRIGG data or other data to secondary side conditions for PWRs should be provided. Transient analysis of the secondary side must be substantiated. For any transient in which two phase flow is encountered in the primary all the two phase flow models must be justified."

Response: The extrapolation of FRIGG data is not used in the RETRAN02 model. The results of the RETRAN simulation for the DCPP Unit 2 turbine trip test showed that the RETRAN02 model can accurately predict the transient behavior of the secondary system for the loss of load transient. Two-phase flow never occurs in the primary system during the entire loss of external electrical load/turbine trip transient. The concern regarding the two-phase flow model is therefore not applicable.

ii) "The pressurizer model requires qualification work for the situations where the pressurizer either goes solid or completely empties."

Response: The pressurizer never becomes water solid nor is it completely drained during the entire loss of load transient. Therefore, this concern is not applicable.

Restriction 3. "The countercurrent flow logic was modified, but continues to use the constitutive equations for bubbly flow; i.e., the code does not contain constitutive models for stratified flow. Therefore, use of the hydrodynamic models for any transient which involves a flow regime which would not be reasonably expected to be in bubbly flow will require additional justification."

Response: The RETRAN02 model does not use constitutive equations in any of the RETRAN02 models. Therefore, this restriction is not applicable.

Restriction 4. "Certain changes were made in the momentum mixing for use in the jet pump model. These changes are acceptable. However, those limitations on the use of the jet pump momentum mixing model which are stated in the original SER (Ref. 1) remain in force."

Response: DCPP is a PWR plant and does not have jet pumps. Therefore, this restriction is not applicable.

Restriction 5. "If licensees choose to use MOD004 for transient analysis, the conservatism of the heat transfer model for metal walls in non-equilibrium volumes should be demonstrated in their plant specific submittals."

Response: A non-equilibrium volume is only used to model the pressurizer in the RETRAN model. The loss of external electrical load/turbine trip transient is a heatup-and-pressure-increase transient. Therefore, it is more conservative to exclude the heat transfer to the pressurizer metal wall in modeling the pressurizer since metal wall heat transfer would only slow down the pressure rise. Thus, this restriction is conservative for this application.

Restriction 6. "The default Courant time step control for the implicit numerical solution scheme was modified to 0.3. No guidance is given to the user in use of default value or any other values. In the plant specific submittals, the licensees should justify the adequacy of the selected value for the Courant parameter."

Response: A parametric study was performed for Courant time step control coefficient. It was found that the effect of Curant time step control coefficient on the RETRAN results for loss of external electrical load/turbine trip transient is insignificant.

The above review shows that all the restrictions on the use of RETRAN02/MOD004 do not apply to or are conservatively covered by the loss of external electrical load/turbine trip transient analyses performed in this report. Therefore, the RETRAN02/MOD004 code is acceptable to carry out the safety analyses supporting the relaxation of safety valve setpoint tolerance.

Attachment E.2
COMPLIANCE WITH STANDARD REVIEW PLAN 5.2.2 ON OVERPRESSURE PROTECTION

U.S. NRC Standard Review Plan (NUREG-0800), Section 5.2.2, "Overpressure Protection," specifies that safety valves shall be designed with sufficient capacity to limit the pressure to less than 110 percent of the Reactor Coolant Pressure Boundary (RCPB) design pressure during the most severe abnormal operational transient with the reactor scrammed. To account for uncertainties in the design and operation of the plant, the following operating conditions need to be included in the determination of peak pressure. The assumptions used in PG&E loss of external electrical load/turbine trip analysis, which are in response to the plant operation conditions required by NUREG-0800, are listed after each of the plant operating conditions.

i. The reactor is operating at a power level that will produce the most severe overpressurization transient.

The initial core power is 102 percent of nominal, which accounts for full reactor power plus 2 percent uncertainty for power measurement.

The reactor is in manual control.

A two-second delay is used from when the high pressurizer pressure trip setpoint is reached to when control rods start to drop.

A conservatively low total Rod Control Cluster Assembly (RCCA) bank reactivity of -4% WK is used, which includes the effect of the highest worth rod stuck out of the core.

The Moderator Temperature Coefficient of +5 PCM/degrees F and the Doppler Reactivity Coefficient of -2.14 PCM/degrees F are used. The combination of these two coefficients provide conservative power history.

The 1973 ANS decay heat table with a 1.2 multiplier is used.

ii. All system and core parameters are at values within normal operating range, including uncertainties and TS limits that produce the highest anticipated pressure.

The initial pressure is 2210 psig, which is greater than the setpoint of the low pressurizer low pressure alarm. The low initial pressure delays the reactor trip, and results in higher peak reactor coolant temperature and pressure.

Pressurizer PORVs and sprays do not operate, whereas pressurizer heaters do operate.

No credit is taken for the steam dump system and the PORVs. Secondary pressure is relieved only by the safety valves.

The high pressurizer pressure reactor trip setpoint has been conservatively set at 2425 psia, which is the nominal 2400 psia plus 25 psi for instrument uncertainty.

iii. The reactor scram is initiated by the second safety-grade signal from the reactor protection system.

No credit is taken for direct reactor trip after the turbine trip.

iv. The discharge flow is based on the rated capacities specified in the ASME Boiler and Pressure Vessel Code for each type of valve.

The discharge flow rates of both pressurizer and main steam safety valves at fully open are equal to or less than the rated flow rates given in FSAR Update.

The above review shows that the plant operations conditions required by NUREG-0800 to account for uncertainties in the design and operation of the plant are fully met in the PG&E loss of external electrical load/turbine trip analyses. Therefore, PG&E loss of external electrical load/turbine trip analyses meet the requirements of NUREG-0800.