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THERMAL HYDRAULIC MODEL QUALIFICATION

DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM EVENTS

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LIST OF ACRONYMS

AFAS	Auxiliary Feedwater Automatic Signal
AFW	Auxiliary Feedwater
CE	Combustion Engineering
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
DEH	Digital Electronic Hydraulic
DNBR	Departure from Nucleate Boiling Ratio
EPRI	Electric Power Research Institute
FLB	Feedwater Line Break
FPL	Florida Power & Light Company
FRV	Feedwater Regulating Valve
FSAR	Final Safety Analysis Report
GPM	Gallons Per Minute
IHTC	Interface Heat Transfer Coefficient
LOAC	Loss of Non-Emergency Power to the Station Auxiliaries
LOF	Loss of Forced Coolant Flow
MFRV	Main Feedwater Regulating Valve
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
MW	Megawatt
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PORV	Power Operated Relief Valve
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System

LIST OF ACRONYMS (Continued)

RPS	Reactor Protective System
SER	Safety Evaluation Report
SDBS	Steam Dump and By-pass System
SG	Steam Generator
SI	Safety Injection
SLB	Steam Line Break
TBS	Turbine By-pass System
TBV	Turbine By-pass Valve
TSV	Turbine Stop Valve
W	Westinghouse

1.0 Introduction

1.1 Background

In 1986, Florida Power and Light Company (FPL) submitted a topical report to the NRC (Reference 1) which provided RETRAN analyses ranging over a broad spectrum of transients. FPL's goal with this submittal was to meet the requirements of the NRC as delineated in Generic Letter 83-11 (Reference 2) to demonstrate staff technical proficiency. This was a first step in the process of obtaining qualification of FPL to perform safety analyses using the RETRAN code to support licensing actions.

In 1988, the NRC safety evaluation (Reference 3) was obtained on the RETRAN topical report. The conclusion of this evaluation was that "...the topical report does demonstrate that FPL has the capability to use RETRAN computer code to perform systems transient calculations for the Turkey Point and St. Lucie plants and, therefore, fulfills the requirements of Generic Letter 83-11." The NRC safety evaluation also stated, "However, additional comparisons between the RETRAN computed results and plant operating data together with appropriate nodalization, sensitivity studies and licensing assumptions will be necessary in future reports before these models are acceptable for licensing submittals." The purpose of this document is to provide the additional information required by the NRC in one specific class of transient events, Decrease in Heat Removal by the Secondary System and obtain NRC approval for the use of RETRAN in licensing actions associated with transients within this category.

The sections that follow address the NRC requirement for additional information. Basic information related to the RETRAN models currently used at FPL will be discussed in Section 1.1. Models for St. Lucie Unit 1 and Unit 2 as well as that for Turkey Point Units 3 & 4 are provided.

Section 2.0 is a detailed description of a reactor transient at Turkey Point which produced a reduction in heat removal of the secondary side because of a turbine runback. The mismatch between primary power and secondary heat removal resulted in a reactor scram on high pressure. This section also provides sensitivity studies which will provide justification for the RETRAN modelling used in transients within the Decrease in Heat Removal by the Secondary System category. Prediction of this transient demonstrates the accuracy of two key RETRAN models, i.e., calculation of primary to secondary heat transfer and the pressurizer model, which are needed for the analysis of this class of event.

[illegible]

Section 3.0 presents a benchmark analysis for a partial loss of feedwater event that occurred at St. Lucie Unit 1. The RETRAN model used in this analysis is consistent with the Turkey Point model discussed in Section 2.0 with the exception of geometric and systems differences representative of Westinghouse versus Combustion Engineering Nuclear Steam Supply Systems (NSSS). This calculation demonstrates the use of the multi-node steam generator model and presents qualification of this model for future applications when accurate tracking of the steam generator level is important.

Section 4.0 will present the assumptions that will be utilized in the future for licensing actions within the category of Decrease in Heat Removal by the Secondary System events. A discussion of each event within this category will be provided along with the determination of the limiting event within the category through comparison of the key physical phenomena that impact the transients. The limiting transients are executed and presented in Section 4.0 using input parameters chosen to produce bounding results for those transients in RETRAN. Section 4.0. also includes the RETRAN modelling assumptions which will be used in the future for licensing actions. The modelling used is derived from the results of sensitivities performed in Section 2.0 using the basic plant models which will be discussed in Section 1.2.

1.2 FPL RETRAN Models

All the analyses presented in this report have been performed with RETRAN02 MOD004 (Reference 4). Approval of this code for use in licensing applications was obtained from the NRC in October 1988 (Reference 5).

The noding diagrams for the St. Lucie Unit 1, Unit 2 and the Turkey Point Units 3 & 4 RETRAN base models are shown in Figures 1.2-1, 1.2-2 and 1.2-3, respectively. A general description of these models is provided in Reference 1. A more detailed description of the Steam Bypass Control Systems and the Multi-node Steam Generator Models available for use in the respective base decks is provided in Sections 1.2.1 and 1.2.2. These two component models are described here because of their important roles in the prediction of several of the transients presented in this report. A summary matrix of the models used in the respective plant base decks is shown on Table 1.2-1. A discussion on the modelling approaches followed in the development of the plant base models is presented in Subsection 1.2.3.

1.2.1 Steam Bypass Control Systems

The discharge flows of the respective Steam Dump Bypass Systems (SDBS) for the St. Lucie Units and the Turbine By-pass System (TBS) for the Turkey Point Units are represented with fill tables of flow versus demand. Except for the cases where the demand is a function of pressure, allowance for pressure dependence of the discharge flow is ignored. The demand is computed by the respective Steam Bypass Control Systems (SBCS).

- ST. LUCIE STEAM DUMP BYPASS SYSTEM

The SDBS and their associated SBCS at both St. Lucie Units are identical. The total steam bypass flow capacity of the SDBS is 45% of the nominal steam flow that corresponds to 2560 Mwth (original plant rating). The system consists of three sequentially operated valve groups with one 5%, one 10% and three 10% capacity valves respectively. The system utilizes two modes of operation, the modulation and the Quick Opening (QO) modes. During load reductions and other transients, all groups are modulated to maintain steam header pressure at 910 psia. If a reactor trip occurs, valve groups 2 and 3 (40% total capacity) are switched from pressure to temperature modulation to reduce primary coolant average temperature to a value of 535.1 Degree F while valve group 1 continues to modulate based on the steam generator header pressure.

The pressure control mode of operation is accomplished by means of a proportional-plus-integral-plus-derivative (PID) controller operating on the difference between the steam header pressure and a constant setpoint of 910 psia. This mode of operation regulates the valves at a relatively low speed, which limits the maximum load changes that can be accommodated by this type of control alone.

For large load reductions and for unit trips from high power levels, the energy accumulated in the system would typically be large enough that the secondary safety valves would be required to relief secondary system pressure before the bypass valves would reach a fully open position under the pressure control mode of operation. To avoid this, the QO control mode is actuated in these situations to open all valves at a much faster rate. A QO signal is generated when a load reduction rate is greater than a given rate or if, after reactor trip, the primary coolant average temperature exceeds a preset threshold.

- TURKEY POINT TURBINE BY-PASS SYSTEM

At the Turkey Point Units, the relief capacity of the TBS is 40% of nominal steam flow. The system consists of two valve groups with two valves each and a total discharge capacity of 20% per group. The TBS at Turkey Point differs from that at St. Lucie in the manner in which the bypass flow is controlled. At Turkey Point, the only automatic control variable is primary coolant average temperature. The magnitude of the flow is based on the difference between actual coolant average temperature and a reference temperature. The reference temperature can be based either on a load reduction program or a temperature setpoint of 547 F. The first is used for transients not involving turbine trip and the second for situations where turbine trip has occurred. A QO logic is also available to open the valves at a faster rate than during temperature control. Its opening logic varies depending on whether or not a turbine trip has occurred.

1.2.2 Multi-node Steam Generator Models

Detailed steam generator models are available for cases where predictions of level, or inventory are important. The models for St. Lucie Units and Turkey Point are shown in Figures 1.2-4 and 1.2-5 respectively. Both models use the Non-Equilibrium code option to better represent the phenomena in the upper downcomer (outside the separators) region. The separators are represented with the RETRAN bubble rise model. Model parameters such as mixture levels and enthalpy of the liquid region in the non-equilibrium volumes are adjusted to yield design liquid mass inventories and good predictions of plant transient level responses.

1.2.3 Modelling Approaches

1.2.3.1 Pressurizer Modelling

The primary system pressure response in a PWR is largely determined by the pressurizer insurge or outsurge flows, the processes that take place within the pressurizer and by the response of the pressure control systems. It is important that all these effects be correctly modelled to ensure that the system pressure response is calculated adequately.

For heatup type transients, the pressurization or insurge phase of the transient is crucial because it affects pressure peak magnitude and timing and hence the time of the reactor protection system actuation.

The modelling approach for the pressurizer in the FPL RETRAN models has the following features:

- SINGLE NODE, NON-EQUILIBRIUM VOLUME

The non-equilibrium option is used to model the pressurizer volume in the FPL RETRAN models. This option allows the vapor and liquid regions in the pressurizer to have different temperatures while at the same pressure. This is a needed feature to model the phenomena taking place in the pressurizer especially during insurges. When an insurge occurs, the liquid region in the pressurizer becomes subcooled with the addition of cooler fluid from the hot leg while the vapor region, compressed by the addition of new fluid into the pressurizer, becomes superheated.

Another important process taking place in the pressurizer is thermal stratification within the liquid region. During fluid insurges into the pressurizer thermal stratification tends to delay the cooling of the vapor region and therefore results in higher pressurization rates. During outsurges, thermal stratification causes the upper layers of hotter liquid to stay hotter for a longer time thus causing the pressure to decrease less rapidly than in the cases where perfect mixing is allowed. The effects of the perfect mixing assumption used in the one node pressurizer model can be balanced with the use of a very low inter-region heat transfer coefficient (IHTC) value to better approximate the effects of thermal stratification. This is the approach taken in the FPL RETRAN base models.

- INTER-REGION HEAT TRANSFER COEFFICIENT

The IHTC has been shown to have little effect on the pressure response unless high values in the order of 30,000 to 50,000 Btu / hr-ft^2-F are used. A low value of 50 Btu / hr-ft^2-F is used in the FPL RETRAN models to compensate for the instantaneous mixing assumption in the liquid region.

- BUBBLE RISE AND RAINOUT VELOCITIES

These two velocities are important in de-pressurization situations where the saturation temperature in the pressurizer decreases. As the temperature decreases, liquid droplets start forming in the superheated vapor region and vapor bubbles form in the subcooled liquid region. The water droplets fall onto the interface at a certain velocity called rainout velocity. Vapor bubbles in

the liquid rise to the surface or interface between the liquid and the vapor at a certain velocity called the bubble rise velocity.

The rainout velocity is specified by the user and is kept constant by the code throughout the calculation. Typically, values between 1 and 5 ft/sec are recommended. This parameter has no impact during insurges into the pressurizer. The value of 5 ft/sec is used in the FPL RETRAN models.

The user has several options for the bubble rise model. In addition to the bubble velocity, the user can also select a bubble gradient to represent the increase in bubble concentration with elevation in the mixture region. The range of gradients is between 0 and 1 where 0 corresponds to the case with a homogeneous distribution of bubbles in the mixture. Bubble velocity and gradient define how quickly vapor in the liquid region moves into the vapor region. For regions that require the existence of a steam dome (e.g., pressurizer and steam generators) and well defined phase separation in the mixture region, values of 0.8 and 3.0 ft/sec for the gradient and initial bubble velocity are recommended (Reference 4) and have been incorporated into the RETRAN models.

- HEAT TRANSFER TO AND FROM PRESSURIZER WALL

During insurge transients, the colder metal in the pressurizer wall tends to absorb some of the heat from the superheated vapor region thus reducing the pressurization rate from the values that could be observed if wall heat transfer did not occur. Similarly during outsurges the presence of heat transfer from the hotter wall tends to decrease the de-pressurization rate.

Heat transfer to or from the pressurizer wall is not modelled in the FPL RETRAN models. In this case, it was determined that the conservatism in not modelling the effects of the pressurizer wall heat transfer (both for insurge and outsurge events) would be the appropriate choice for the base models.

- SPRAY OPTION

The RETRAN code offers two options to model the spray fluid into the pressurizer, one de-superheats the vapor region while the other does not. The second option tends to yield higher peak pressures on insurge transients and has been selected for the FPL RETRAN models.

1.2.3.2 Nodalization

- OVERALL NODALIZATION

The nodalization approach for the three FPL RETRAN base models is shown in Figures 1.2-1, 1.2-2 and 1.2-3. The three models are very similar in nodalization with the only differences being due to the geometric differences between CE and W NSSS. The nodalization for these models is discussed in some detail in Reference 1.

For analysis of transients within the Decrease in Heat Removal by the Secondary System category, the nodalization detail in regions such as the vessel and hot and cold legs is not important. The level of detail found in the three FPL RETRAN base models is more explicit than required for the analysis of Decrease in Heat Removal by the Secondary System events presented in this report. Simplified base models with combined volumes in the reactor vessel and other parts of the system could have been used with practically the same results. For consistency reasons, however, a single nodalization approach that can be expanded to include a multi-node steam generator when needed has been preferred for the three FPL RETRAN base models.

- STEAM GENERATOR NODALIZATION

The most important consideration in the modelling of the steam generators is whether or not a single node approach is adequate in the simulation of certain events. Based on industry experience with this issue, FPL has resolved to retain the single node approach for its base models. This approach is valid for most applications of the models where prediction of steam generator level response is not crucial. For situations where level is important (e.g., situations where reactor trip is on low level), the multi-node models for each plant (Figures 1.2-4 and 1.2-5) will be attached to their respective RETRAN base models.

1.2.3.3 Safety Valve Modelling

The safety valves in the existing RETRAN base decks are modelled in two parts: a fill junction to model the discharge flow and a valve to ensure initiation and termination of the fill flow at the selected setpoints. The discharge flow is entered as a table of pressure dependent values based on the Moody critical flow correlation. Design flow is assumed at the opening setpoint pressure and it is allowed to increase if the system pressure increases. Accumulation and hysteresis are not assumed. In the Turkey Point model one-second ramps are used to model the opening and closing of the pressurizer safety valve at the respective

setpoints and half-second ramps are used for the secondary safeties. In the St. Lucie base models the opening and closing of all safety valves is assumed instantaneous at the respective setpoints.

1.2.3.4 Steam Generator Tube Plugging

Current plant levels for steam generator tube plugging have been approximately incorporated into the respective RETRAN base models. Values of 0% for Turkey Point and 5% for the St. Lucie Units are used for the SG tube plugging in the base models.

1.2.3.5 Control Systems

Control systems are modeled in all the base models to represent as closely as possible the operation of the corresponding plant systems. Table 1.2-1 summarizes the control systems available in the respective FPL RETRAN models. The most elaborate controller in the base decks is that of the Steam Dump Bypass System developed for both the St. Lucie and the Turkey Point models. These systems have been described in detail in Section 1.2.1. A less complex system is the Feedwater Control System which currently is only available for the Turkey Point RETRAN model. Such a system facilitates the analyses of plant transients and will be developed for the St. Lucie models in the future.

In addition to that of the the feedwater controller, two additional differences can be noted between the control systems used in the different FPL RETRAN base models. One is in the modelling of the PORVs which at Turkey Point require the action of a controller to simulate the operation of one of the two available PORVs to generate an anticipatory opening signal earlier than at the pressure setpoint. The other difference is in the need for a controller for the Atmospheric Dump Valves at Turkey Point. Neither the PORV's or the ADV's operate in a manner which requires a control system at the St. Lucie Units.

TABLE 1.2-1

FPL RETRAN BASE MODELS SUMMARY

	TURKEY POINT	ST. LUCIE 1	ST. LUCIE 2
PRESSURIZER			
-Non-Equilibrium Volume	YES	YES	YES
-Metal Heat	NO	NO	NO
-Spray Option	YES	YES	YES
ENTHALPY TRANSPORT	YES	YES	YES
TEMPERATURE TRANSPORT	NO	NO	NO
STEAM GENERATOR			
-Multi-node	NO	NO	NO
-Tube Plugging	NO	YES	YES
CONTROL SYSTEMS			
-Pressurizer Heaters	YES	YES	YES
-PORV Opening	YES (1 out of 2)	NO	NO
-Feedwater Flow	YES	NO	NO
-Steam Bypass	YES	YES	YES
-Atmospheric Steam Dump	YES	NO	NO

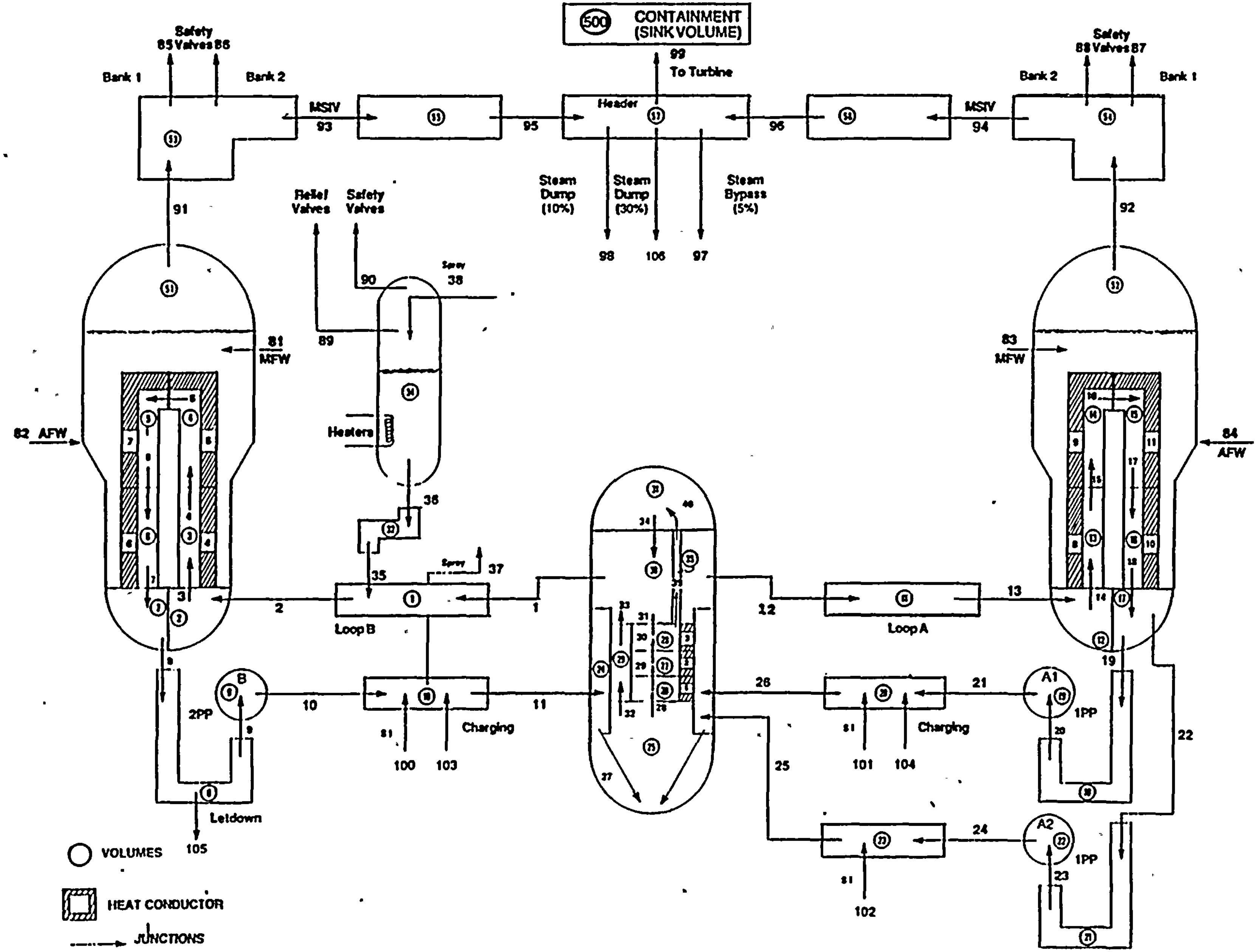

FIGURE 1.2-1

St. Lucie 1 RETRAN Base Model

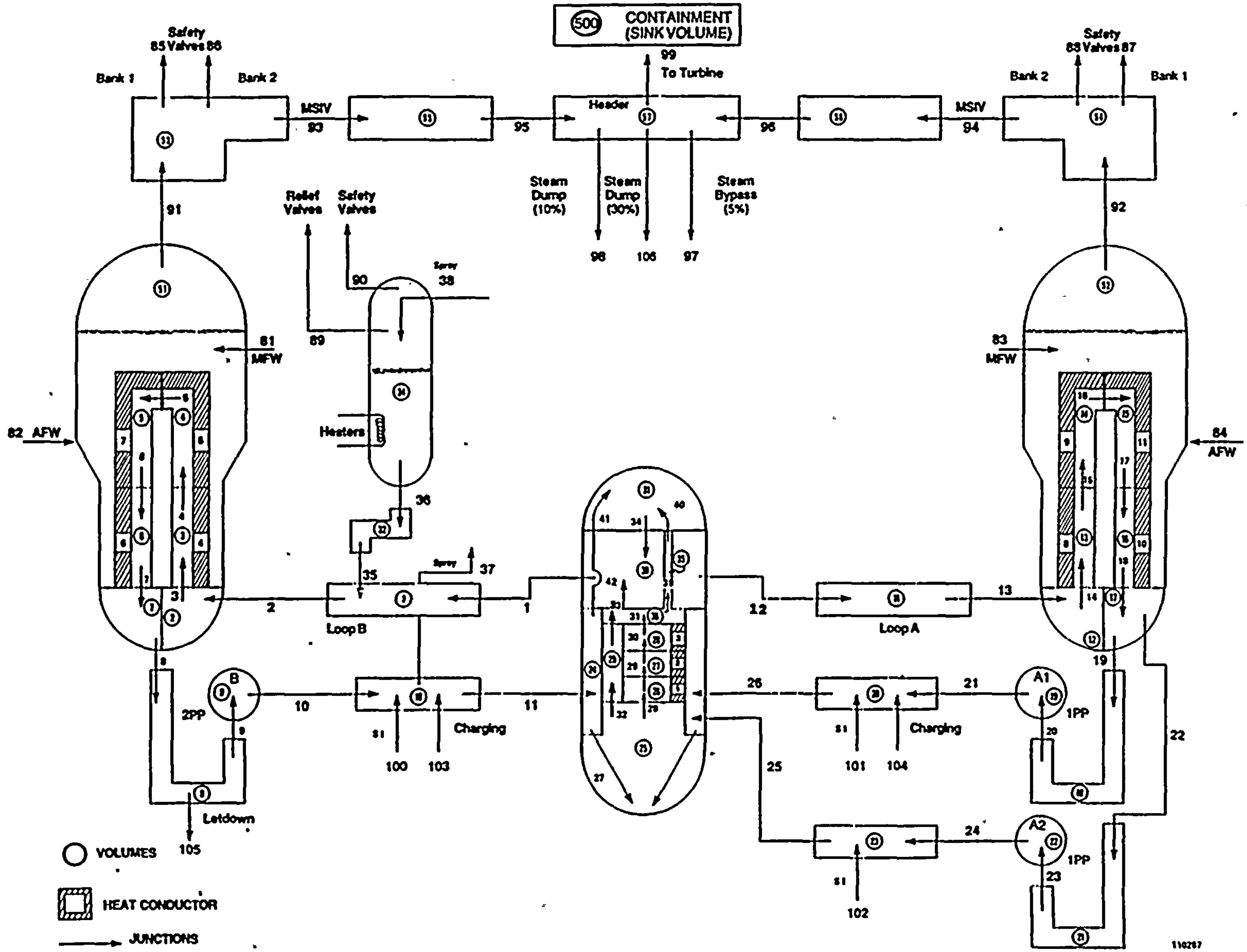

FIGURE 1.2-2
St. Lucie 2 RETRAN Base Model

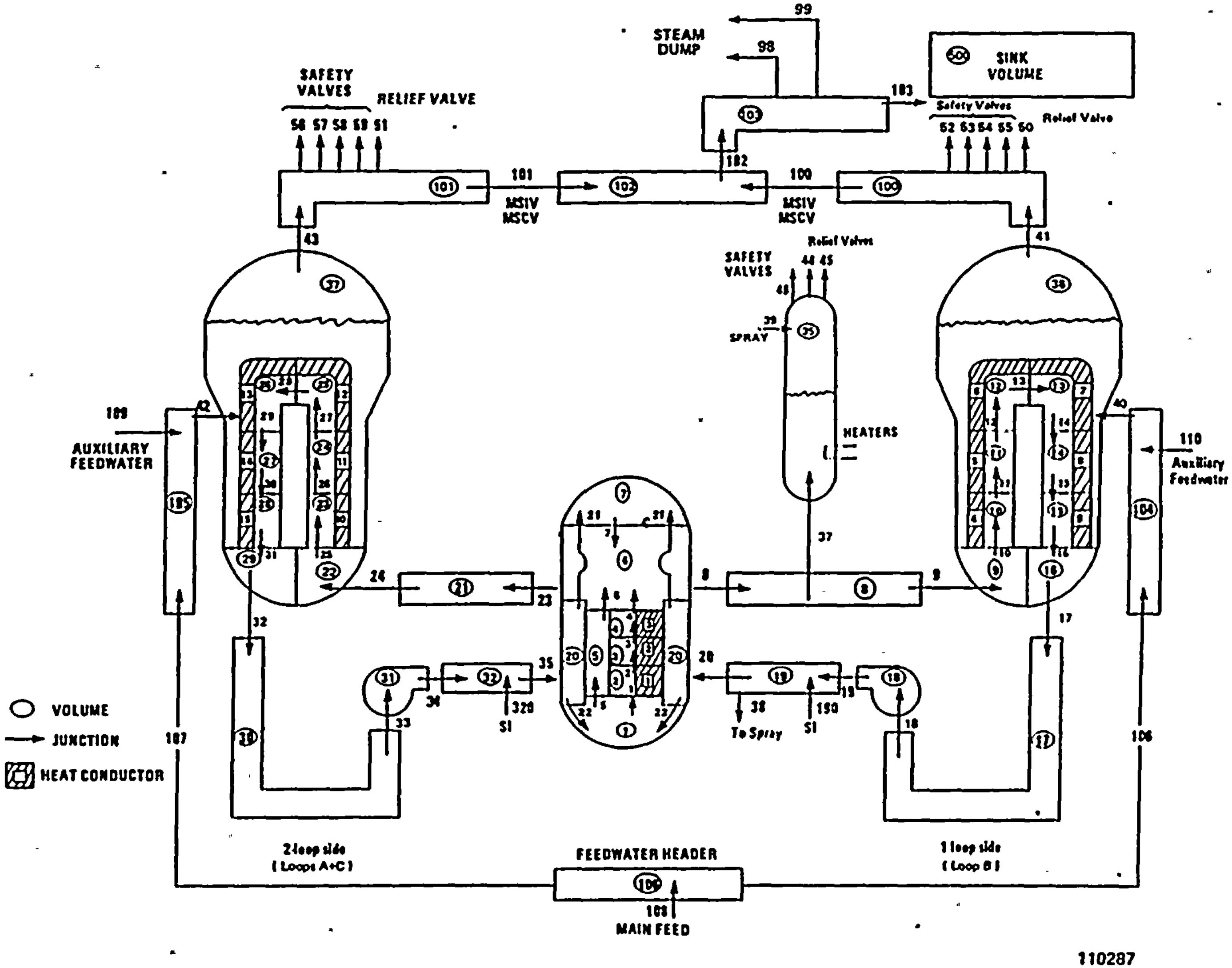

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FIGURE 1.2-3

Turkey Point RETRAN Base Model

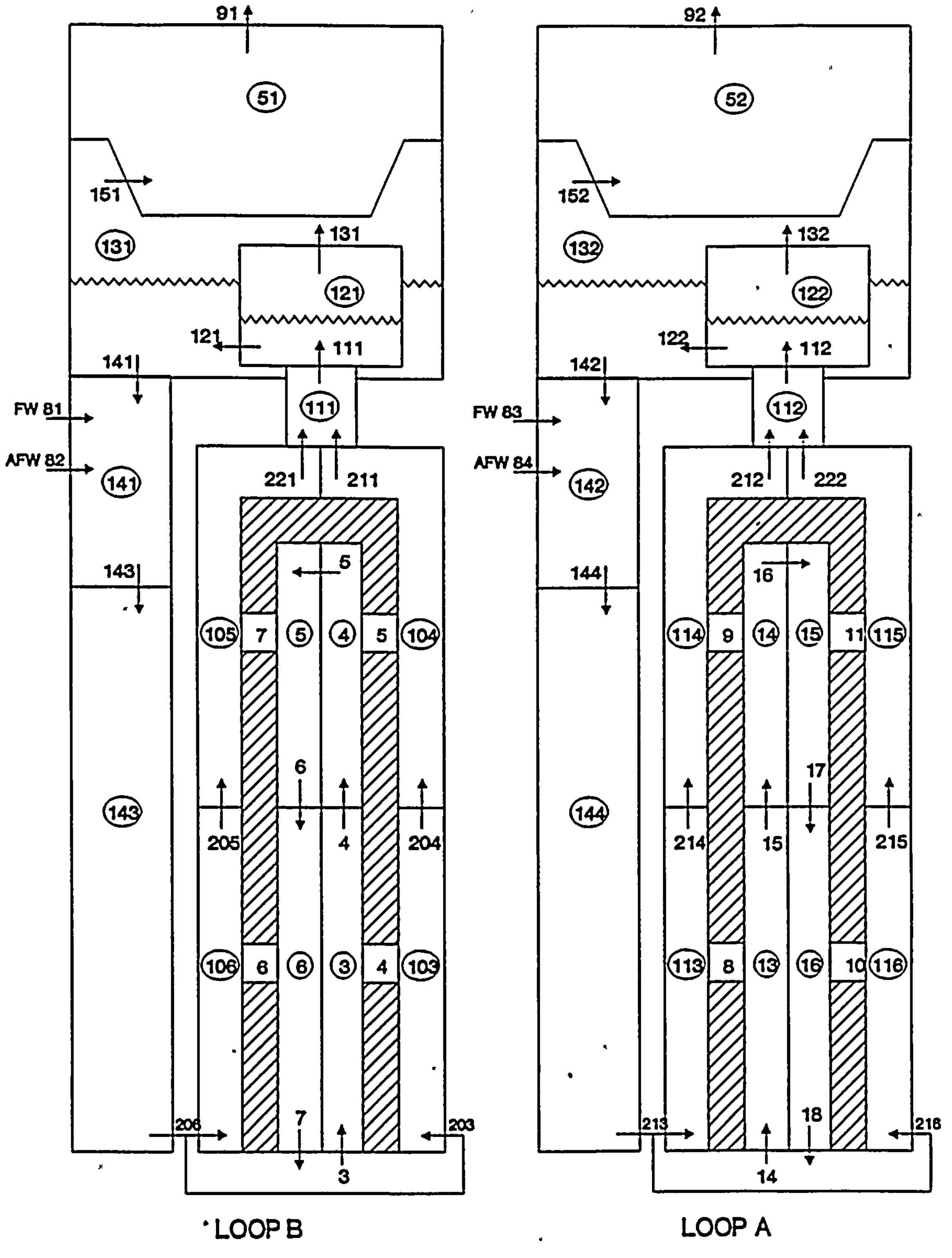


FIGURE 1.2-4

St. Lucie Multi-node Steam Generator Model

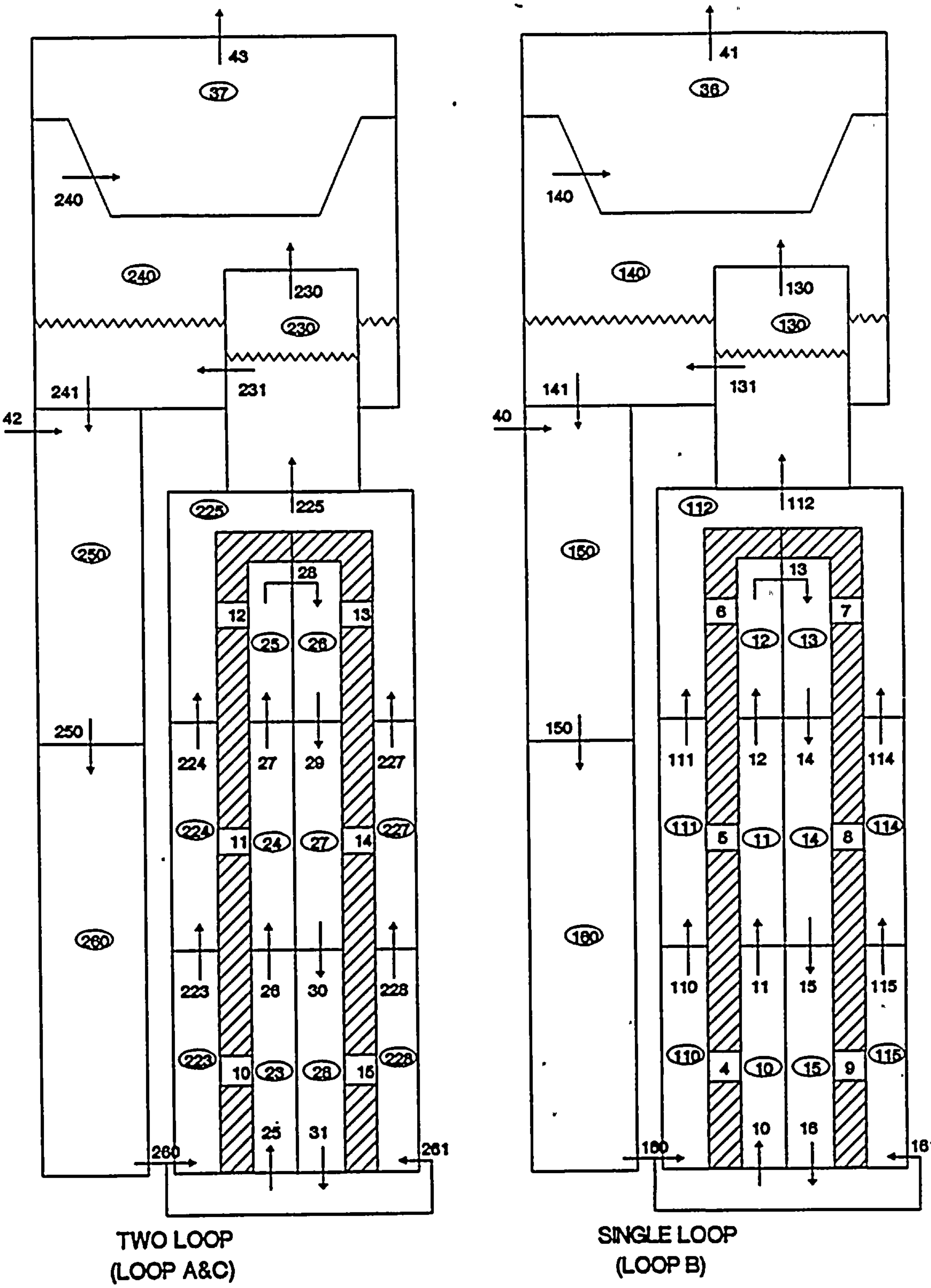


FIGURE 1.2-5

Turkey Point Multi-node Steam Generator Model

2.0 Turkey Point Unit 4 Loss of Inverter

2.1 Summary of Events

On June 20, 1985, Turkey Point Unit 4 experienced a reactor trip from 100% power. The initiating event was the tripping of the 4C inverter which was supplying power to the 120 volt vital instrument panel 4P06. The loss of the inverter initiated a turbine runback due to the loss of power to a nuclear instrumentation system channel. In addition, Loss of 4P06 de-energized the pressurizer level and spray valve controllers (causing the spray valve to remain at its last demand position). De-energizing of the level controller caused a false indication of low pressurizer level (less than 14%) which in turn de-energized the pressurizer heaters (control and backup) and initiated letdown isolation.

Loss of the 4C inverter also resulted in the loss of automatic operation of one of the two Power Operated Relief Valves (PORV), the other PORV was available but had its block valve closed due to leakage problems. These conditions resulted in the reactor coolant system pressure increasing until it reached the pressurizer high pressure reactor trip setpoint of 2385 psia which initiated an automatic reactor trip. It should be noted that the Technical Specification setpoint is 2385 psig for the high pressure trip. Plant procedures provide a 15 psia uncertainty allowance for instrument drift, therefore the setpoint applicable for this event is 2385 psia.

Pressurizer pressure decreased after the trip and continued to decrease because of the de-energized pressurizer spray valve controllers which maintained spray flow even as the pressure decreased.

Loss of the 4P06 panel also caused the "A" Steam Generator (SG) feedwater level controller to transfer from automatic to manual. Feedwater to SG "A" remained at the 100% power flow rate during the early stages of the transient. Loss of automatic level control along with continuous supply of 100% feedwater flow resulted in the "A" SG level increasing until it reached the Hi-Hi level setpoint (80%). Both SG feedwater pumps are tripped due to reaching the high SG level setpoint about one minute after the reactor trip. This resulted in a feedwater isolation signal and an automatic start of the auxiliary feedwater pumps.

A variety of countermeasures were taken by the operators to mitigate the cooldown caused by the feedwater transient. At about seventeen minutes into the event, power to the 4P06 panel was restored and the lost instrumentation on Unit 4 was regained.

2.2 Analysis

Analysis of plant events can be performed in order to validate the ability of the simulation model being used to accurately represent actual plant performance. The plant event described above in Section 2.1 has been chosen as a benchmark due to the many characteristics which are similar to the licensing type transients found in the Decrease in Secondary Heat Removal by the Secondary System category of events.

Specifically, this plant event demonstrates a primary to secondary heat generation/removal mismatch sufficient to result in a primary pressurization to the reactor trip setpoint. Another similarity was that the PORVs were not available to mitigate the primary pressurization, as in most licensing transients. Even though the over-filling of SG "A" and the possibility of operator action are not consistent with a licensing transient, this plant event, within the first 60 seconds, clearly offers a valid means to compare the predictive capability of the component models within RETRAN to the actual plant response to a decrease in the secondary system heat removal.

The plant event was analyzed with the initial conditions summarized in Table 2.2-1. A discussion of the assumptions and initial conditions used in the RETRAN analysis follows. The basis for this information is found in References 6-8.

1) Simulation Time

Only the first 60 seconds of the event are provided for comparison of the plant data to the RETRAN simulation. After the initial 60 seconds, possible operator actions to try to reduce the cooldown and de-pressurization caused by the excess feedwater flow and the continued spray flow are not well documented.

2) Feedwater and Auxiliary Feedwater Flows

Feedwater flow to the "A" SG switches to manual at the initiation of the event and the MFRV to SG "A" remains at the 100% power position throughout the first 100 seconds. Feedwater flow to SG's "B" and "C" remained in automatic for the duration of the event. Flow to the "B" SG was the only feedwater flow recorded on the System Parameter Display System (SPDS) and that information is provided at 10 second intervals.

For the RETRAN analysis, flow to the "C" SG is assumed to be the same as that to the "B" SG and is used as a boundary condition for the analysis. The flow for the "A" SG is kept constant at the initial value. After reactor trip, feedwater temperature is reduced based on the available measured data. These feedwater flow analysis assumptions have some uncertainty since no information is available for SG "A" and "C" feedwater flow and the potential exists for the operator to have taken action to reduce feedwater flow in order to mitigate the primary cooldown that was occurring during the event.

One auxiliary feedwater pump was undergoing routine testing at the time of the event. The total pump flow of 375 gpm. as indicated in the Test Procedure, is divided into 125 gpm per SG and kept constant for the duration of the RETRAN simulation.

3) Pressurizer Spray and Heaters

All pressurizer heaters were on at the time of the event initiation trying to compensate a faulty pressurizer low level indication. The pressurizer spray valve was 10% open to compensate for the heaters. At initiation of the simulation the heaters are lost while the spray valve is kept at the 10% open position.

4) Pressurizer PORVs

Of the two PORVs available at the plant, one was isolated for leakage problems at the time of the event, while the other was lost with the inverter failure. In the RETRAN simulation the two PORVs have been assumed unavailable.

5) Turbine Runback

As a result of the loss of inverter, the turbine experienced a runback to 70% power. This is modeled in RETRAN by decreasing the steam flow at the design runback rate of 200% /minute.

TABLE 2.2-1

TURKEY POINT UNIT 4

INITIAL CONDITIONS FOR LOSS OF INVERTER EVENT

PLANT CONDITIONS	VALUE
POWER LEVEL (% OF NOMINAL)	100
TIME IN CYCLE 11	MOC
PRESSURIZER PRESSURE (PSIA)	2255
PRESSURIZER LEVEL (% NR)	51.7
COLD LEG TEMPERATURE (DEG.F)	553
PRIMARY COOLANT AVERAGE TEMPERATURE (DEG.F)	572
CHARGING FLOW (GPM)	72
STEAM GENERATOR PRESSURE (PSIA)	825
STEAM GENERATOR LEVEL (% NR)	60.6
FEEDWATER TEMPERATURE (DEG.F)	427
AUXILIARY FEEDWATER FLOW / SG (GPM)	125

6) Plant Initial Conditions

Initial Pressurizer and SG pressures, cold leg and average temperatures, pressurizer level and charging flow for the RETRAN model have been obtained from Reference 6 and are shown in Table 2.2-1.

7) Turbine By-Pass

With the loss of the inverter, automatic operation of the Turbine By-Pass system is not available until after the reactor trip occurs.

2.3 Results

The results of the RETRAN analysis are presented in a listing of the Sequence of Events shown in Table 2.3-1. A detailed discussion of the RETRAN results is found in Section 2.3.1. Section 2.3.2 provides a discussion of the main differences between the results predicted by RETRAN and the plant response. Section 2.4 describes the results of a variety of parametric studies performed with RETRAN to better understand the code capabilities and limitations in modelling these type of events.

2.3.1 RETRAN Analysis

2.3.1.1 Secondary System Response

Following the loss of the 4C inverter, at the initiation of the event, the turbine admission valve closes to reduce steam flow from 100% to 70% in 9 seconds (turbine runback). Turbine runback is occurring since the loss of the inverter initiates actions as if there was a control rod drop event. The turbine runback is designed to reduce secondary heat removal in order to better match the reduced core power expected due to the insertion of a control rod. Since no control rod action actually happened, this produced a mismatch between the power generated by the core and the power removed by the secondary system which results in a heatup of the secondary system with the subsequent heatup of the primary. Figure 2.3-1 shows the ensuing secondary pressure response for SG's "B" and "C".

TABLE 2.3-1

SEQUENCE OF EVENTS

RETRAN ANALYSIS LOSS OF INVERTER EVENT

EVENT	TIME (seconds)	SETPOINT OR VALUE
Loss of Inverter	0.0	---
Turbine Runback	0.1	---
High Pressurizer Trip	17.7	2385 psia
Rods Begin to Drop	19.7	---
Peak Pressurizer Pressure	20.6	2428 psia
Turbine By-pass Actuates	20.8	---
Turbine Stop Valves Closed	21.2	---
Rods Fully Inserted	22.1	---
Peak SG Pressure	28.5	982 psia

The secondary pressure response in RETRAN shows a smooth rate of increase until the reactor/turbine trip. After the turbine stop valves are fully closed, 21.2 seconds after the initiation of the event, the secondary pressurization increases more rapidly until the Turbine By-pass system reaches full flow at 24 seconds. The Turbine By-pass system monitors the difference between primary temperature and the reference No-Load temperature. As primary temperature decreases after the reactor trip, the Turbine By-pass valves close (prior to the actual attainment of the No-Load Tavg due to the anticipatory action of the control system), with both groups fully closed at approximately 45 seconds into the transient. With the closure of the Turbine By-pass system, the rate of secondary system pressure decrease is reduced as shown on Figure 2.3-1 at 45 seconds.

The secondary pressure decrease seen from about 50 seconds to 60 seconds reflects the analysis assumptions made relating to the main feedwater addition to SG "A" and the reduction of feedwater temperature over time. Maintaining 100% feedwater flow into SG "A" at a reduced temperature along with normal main feedwater flow to SG's "B" and "C" is more than sufficient to remove decay heat from the primary without pressurizing the secondary system. This expected behavior of the secondary system pressure is shown on Figure 2.3-1 .

The secondary system pressure response for SG "A" is shown on Figure 2.3-2. No plant data is available for comparison with the RETRAN prediction of the SG "A" behavior. During the initial stage of the transient, the SG response is very similar to that shown on Figure 2.3-1. This is as expected since no real deviation between the three SG's was acknowledged at the plant and the RETRAN analysis assumes no differences until after the reactor trip when flow to SG "A" is assumed to be maintained at 100% flow. In fact, SG pressure response for the three SG's is basically the same until the Turbine By-pass valves close. At that point, the addition of main feedwater to SG "A" produces a faster de-pressurization of the SG. At the 60 second end point for this comparison, there is calculated to be a 12 psia reduction in pressure in SG "A" relative to that shown in Figure 2.3-1.

2.3.1.2 Primary System Response

As the secondary temperature and pressure increases, the primary coolant temperature also increases as shown in Figure 2.3-3. At approximately 23 seconds the primary heatup has been terminated by the reactor trip on high pressurizer pressure of 2385 psia and the action of the Turbine By-pass system. The RETRAN data shown in Figure 2.3-3 includes a time delay corresponding to the effect found in the Resistance Temperature Detectors (RTD's) present at the plant. The reactor trip causes the turbine to trip and the Turbine By-pass system to open and relieve the energy accumulated in the system. The Turbine Bypass valves operate based on the difference between actual primary temperature and a reference No-Load temperature.

As shown on Figure 2.3-3, the primary temperature decreases smoothly until the Turbine By-pass system closes. It can be seen that the rate of temperature decrease changes after the Turbine By-pass system closes. Primary temperature continues to decrease, however, due to the combination of continued addition of 100% feed flow into SG "A" and continued addition of charging with letdown isolation.

The primary system power is shown in Figure 2.3-4. The initial response to the decrease in heat removal by the secondary system is a slight power decrease due to the action of the negative moderator temperature coefficient. Power decreases rapidly after the reactor scram as the rods begin to be inserted at 19.7 seconds after the initiation of the event.

The pressurizer pressure as calculated by RETRAN is shown in Figure 2.3-5. As shown, the primary system heatup caused by the decrease in secondary system heat removal results in an insurge into the pressurizer. The pressurizer pressure increases and reaches a maximum of 2428 psia at 20.6 seconds into the event. Pressurizer pressure after that point decreases steadily throughout the rest of the simulation as primary temperature decreases and as the pressurizer sprays continue to operate at approximately 10% of full capacity.

The pressurizer level response calculated by RETRAN is shown in Figure 2.3-6. As the heatup progresses, the level increases due to the insurge into the pressurizer. A maximum level is reached at 20.8 seconds into the simulation. Similar to the pressurizer pressure, level thereafter decreases throughout the simulation. A change in slope of the decrease in pressurizer level occurs at 26.6 seconds. This change in slope in the level decrease corresponds to a reduction in the rate of de-pressurization calculated by RETRAN as shown in Figure 2.3-5.

2.3.2 RETRAN Comparison to Plant Data

The data calculated by RETRAN for the Loss of Inverter event has been described in Section 2.3. The sequence of events as calculated by RETRAN and measured at the plant is compared in Table 2.3-2. As previously mentioned, the plant data was only available at 10 second intervals in most cases. Therefore data presented as maximum only represents the maximum data point recorded.

In general, the RETRAN calculations show responses similar to the measured plant data. Specific comparisons show that the RETRAN model reacts slightly slower to the event than the actual plant. This can be observed most clearly by examining the response of SG "B" and "C".

As shown in Figure 2.3-1, the SG model in our base RETRAN calculation shows a slower pressurization during the period of turbine runback (0 - 21 seconds). After turbine trip however, the SG pressurization in RETRAN is faster and results in a higher peak SG pressure. The model then starts to de-pressurize due to the action of the Turbine By-pass system and the action of the feedwater, both the reduction in feedwater temperature as well as the continuation of full feedwater flow to SG "A".

The plant data demonstrates a de-pressurization after turbine by-pass system operation, however, the plant data after 40 seconds for SG "B" shows a stabilization of pressure not seen in RETRAN. The effect of a reduced feedwater temperature in the RETRAN calculation appears to be the cause of the difference. Instantaneous mixing of the fluid within the single node SG results in an overprediction of the impact the colder feedwater has in reducing secondary temperature and pressure.

Comparisons between the other key parameters and the RETRAN results show that RETRAN predicts the same general trends as the plant data. The RETRAN primary pressurization is larger and takes longer to be reduced than the plant data. This difference may be due to the fact that the RETRAN model ignores pressurizer wall heat transfer.

Primary temperature differences between RETRAN and the plant data show that the RCS temperature trends are very similar. RETRAN predicts a peak temperature somewhat later than the plant data, however the cooldown rate after the peak matches well between the plant and RETRAN with a relatively constant difference after 38 seconds.

Pressurizer level comparisons show a slightly larger level increase at the plant during the turbine runback with the long term level response showing almost exact agreement after about 38 seconds.

One key conclusion that can be drawn from these comparisons is that the single node SG model produces peak primary and secondary pressures which are conservative relative to the plant data. This is a key item when considering modelling applications for the licensing events.

TABLE 2.3.2-1

SEQUENCE OF EVENTS

TURKEY POINT UNIT 4 LOSS OF INVERTER EVENT

RETRAN COMPARISON TO PLANT DATA

EVENT	TIME (seconds) Plant Data	RETRAN
Loss of Inverter	0.0	0.0
Turbine Runback	0.1	0.1
High Pressurizer Trip	---	17.7
Rods Begin to Drop	19.9	19.7
Peak Pressurizer Pressure	20.0	20.6
Turbine By-pass Actuates	---	20.8
Turbine Stop Valves Closed	---	21.2
Rods Fully Inserted	21.8	22.1
Peak SG Pressure	28.7	28.5

2.4 SENSITIVITY STUDIES WITH RETRAN MODELS

The Loss of Inverter Event at Turkey Point has been selected to assess the impact of various RETRAN modelling techniques and options on the prediction of heatup events. These effects although investigated with the Turkey Point RETRAN model are also applicable to the St. Lucie models in the analysis of heatup events. Since, as described in Section 1.2, the three FPL RETRAN base models have a very similar nodalization approach and the physical phenomena involved in heatup events is the same, it is reasonable to expect applicability of relative effects from one model to the others. This is confirmed by the similarity of the results of the Loss of Condenser Vacuum analyses performed with the three FPL RETRAN models (see Section 4.0).

The relative impact of the modelling techniques and code options has been evaluated against the two following criteria:

- Fidelity of the RETRAN prediction to the plant data.

- Conservatism of the RETRAN prediction with respect to plant data.

The results of these sensitivity studies support the choices in modeling techniques made for the licensing methodology to analyze Decreases in Heat Removal by the Secondary System presented in Section 4.0.

The various sensitivity studies performed with the Loss of Inverter analysis are described below. A summary of these studies is presented in Table 2.4-1.

Pressurizer Inter-Region Heat Transfer Coefficient

The Inter-Region Heat Transfer Coefficient (IHTC) was changed from the value of 50. Btu/hr-ft^2-F used in the base model to a value of 20,000. Btu/hr-ft^2-F with no noticeable effect in the predicted peak pressure or its timing. Therefore it can be concluded that changes in the IHTC within the above range of values do not impact the key parameters of this event and that the value of 50. Btu/hr-ft^2-F used in the FPL RETRAN base models is adequate in preventing the two regions in the pressurizer from reaching equal temperatures.

Spray Option

The effects of having the spray option activated were investigated and found to cause an increase in predicted peak primary pressure as expected. The option, however, was not kept for the base case analysis of the Loss of Inverter event on the basis of fidelity to plant data. The effects

of the spray option are not applicable for analysis of licensing events involving Decrease in Heat Removal by the Secondary System because no credit for the use of the spray system is taken in such methodology.

Hydraulic Resistance

Higher hydraulic resistances in the surge line or at the pressurizer inlet will tend to yield higher peak pressures in insurge transients. This is because less cold fluid from the surge line can flow into the pressurizer. The impact of this effect has been investigated with the Loss of Inverter Event. The results show a very small sensitivity with hydraulic resistances. The increase of the hydraulic resistance at the entrance of the pressurizer from a value of 0 to a value of 10 resulted only in an increase of 0.9 psia in peak pressure. Based on this, it was determined that the FPL RETRAN models will use the nominal hydraulic resistances for both best estimate and licensing calculations.

Temperature Transport Delay

The temperature transport delay model is intended to simulate the displacement of a temperature front through a channel with little mixing such as in straight pipes. The effects of this option have been investigated with the Loss of Inverter Event by subdividing the hot and cold legs in the RETRAN model into 10 sections to more accurately represent the temperature variation throughout the system. The effects on predicted peak pressure and time of the peak are negligible. Therefore this option will not be utilized.

Courant Limit

The version of the code (MOD004) used in this report has a default value of 0.3 for the Courant time step control coefficient in the iterative numerics solution. This value can be changed by the user to try to improve running times as long as the accuracy or stability of the solution is not affected. Sensitivity studies have been performed by changing the base case value of 0.3 to 0.6 and 1.0. Table 2.4-1 shows that the results are insensitive to reasonable changes in the value of the Courant coefficient. Values of the coefficient above 1.0 are not considered realistic and could result in unstable solutions or convergence errors.

2.5 Conclusions

The Loss of Inverter Event has been analyzed with RETRAN using the known plant inital conditions and equipment actuations during the transient as boundary conditions. The results of the comparison shows the same trends for the parameters and general agreement in timing and magnitude. The key parameter for this type of event, primary system pressure, was calculated conservatively relative to the available data.

The sensitivity studies have reviewed the impact of varying inputs and the conclusions derived will be used in development of the licensing methodology discussed in Section 4.0

TABLE 2.4-1

RETRAN SENSITIVITY STUDIES

TURKEY POINT LOSS OF INVERTER EVENT

CASE	REACTOR TRIP TIME	PEAK PRIMARY PRESSURE	TIME	PEAK SECONDARY PRESSURE	TIME
Base Case	17.7	2428	20.6	981	28.5
IHTC Variation 20,000 BTU/hr-ft^2-F	17.7	2428	20.6	981	28.5
Pressurizer Spray Option	16.3	2430	19.1	975	27.0
Surge Line Hydraulic Resistance:					
$K_f = 0.0$	17.7	2428	20.6	981	28.5
$K_f = 10.0$	17.8	2429	20.6	982	28.5
Temperature Transport Delay	17.7	2428	20.6	982	28.5
Courant Coefficient (C_4)					
$C_4 = 0.6$	17.7	2428	20.6	981	28.5
$C_4 = 1.0$	17.7	2429	20.6	981	28.5

FIGURE 2.3-1

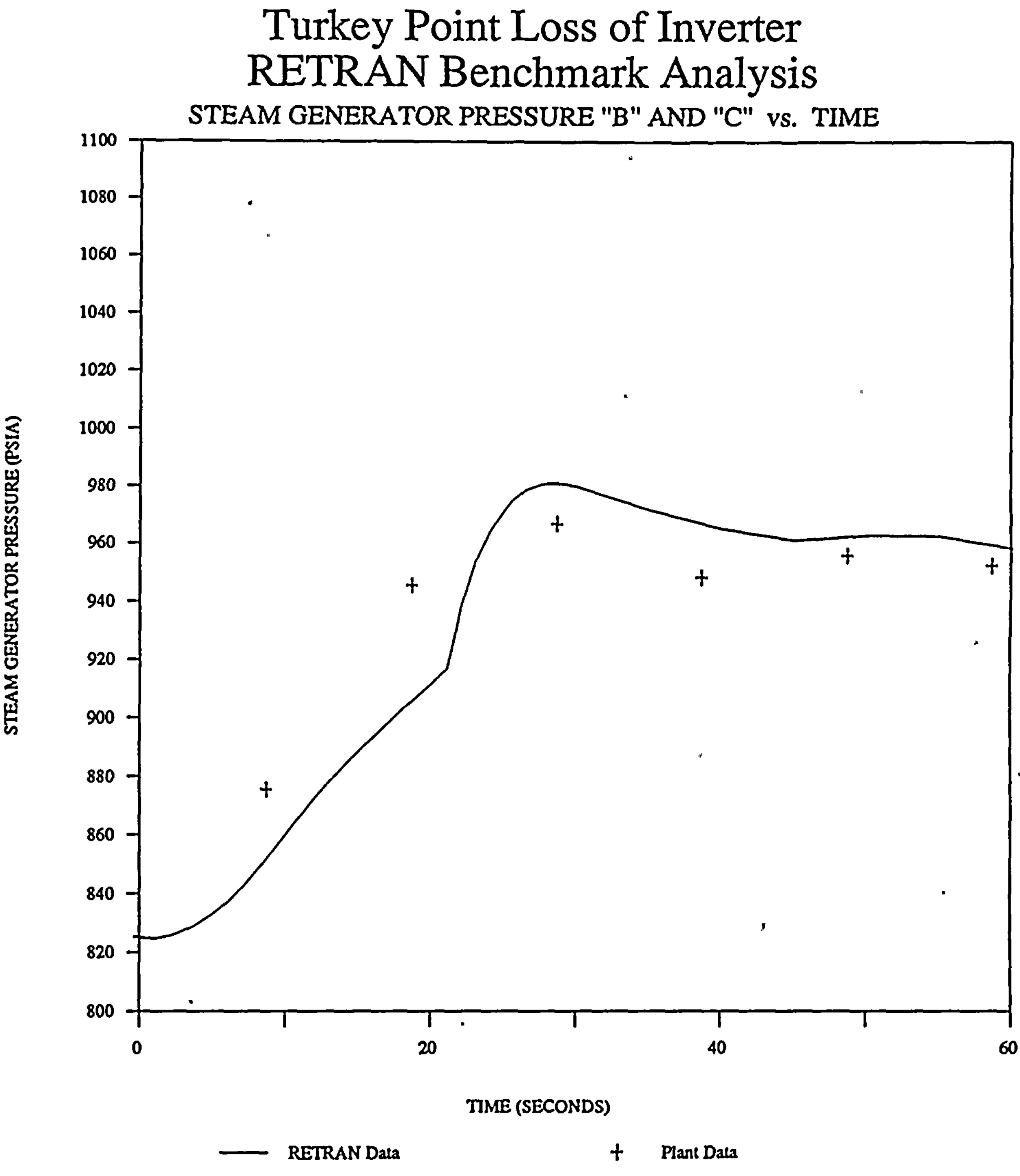

Turkey Point Loss of Inverter
RETRAN Benchmark Analysis
STEAM GENERATOR PRESSURE "B" AND "C" vs. TIME
STEAM GENERATOR PRESSURE (PSIA)
1100
1080
1060
1040
1020
1000
980
960
940
920
900
880
860
840
820
800
0
20
40
60
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 2.3-2

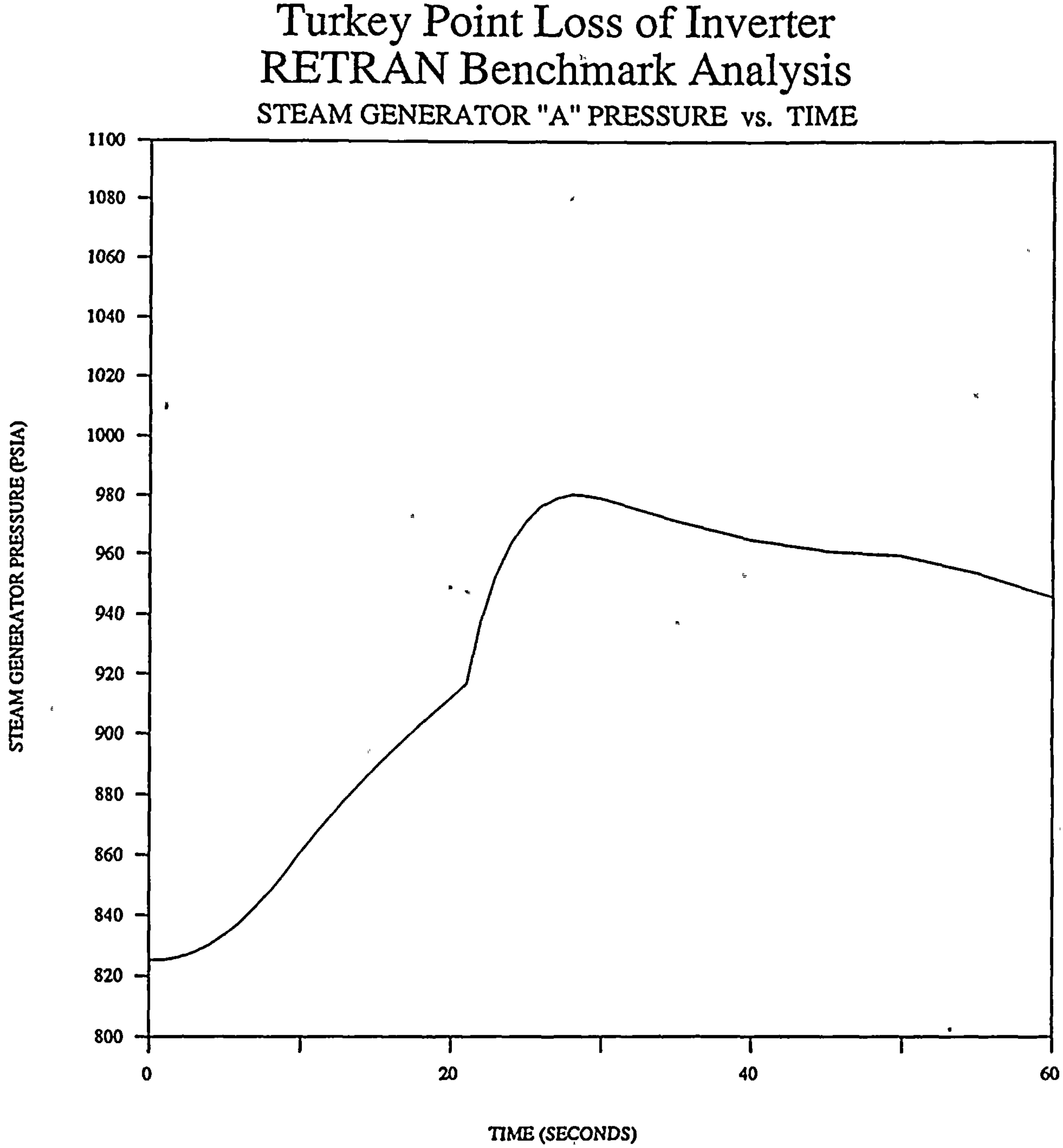

Turkey Point Loss of Inverter
RETRAN Benchmark Analysis
STEAM GENERATOR "A" PRESSURE vs. TIME
STEAM GENERATOR PRESSURE (PSIA)
1100
1080
1060
1040
1020
1000
980
960
940
920
900
880
860
840
820
800
0
20
40
60
TIME (SECONDS)

FIGURE 2.3-3

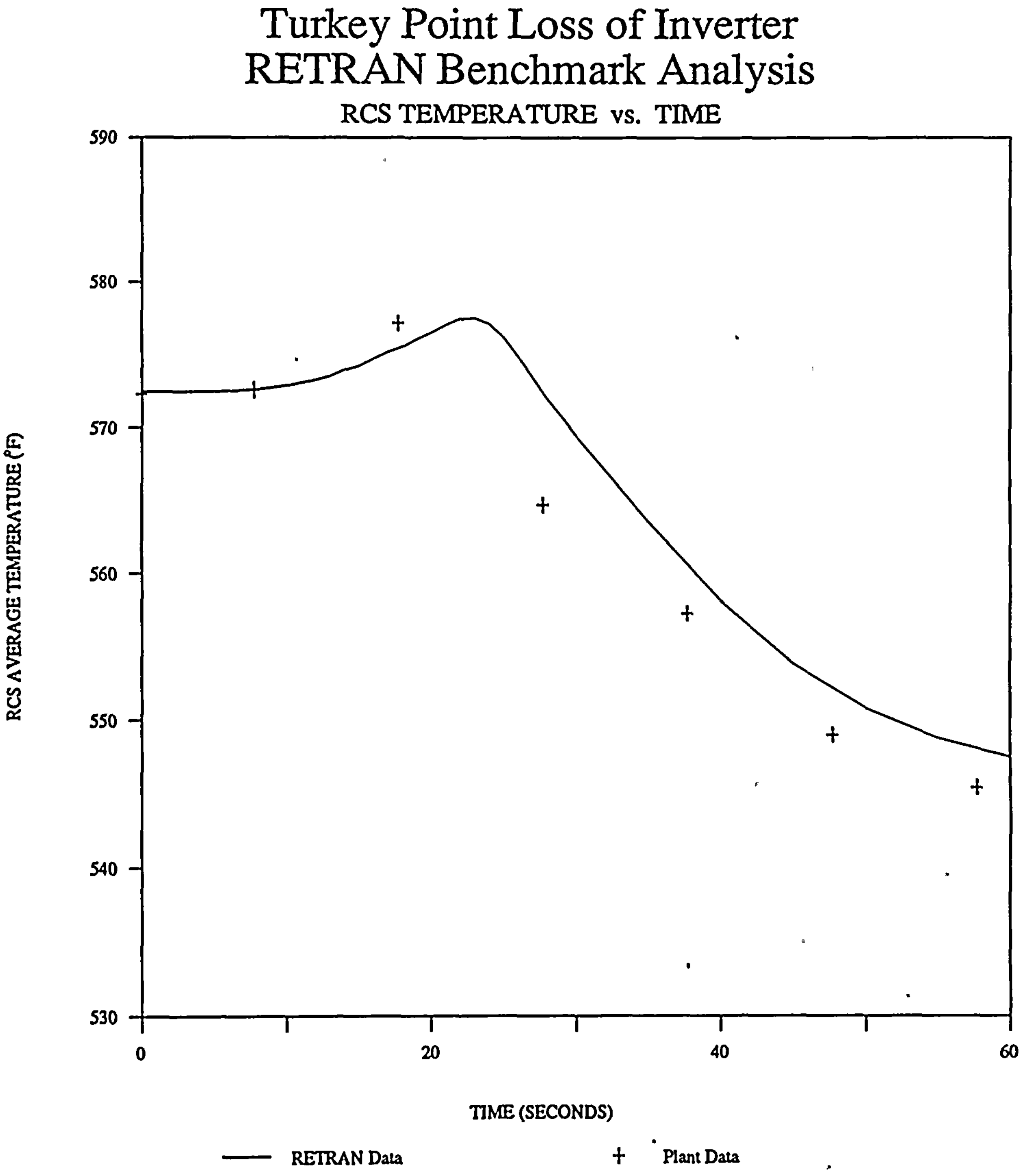

Turkey Point Loss of Inverter
RETRAN Benchmark Analysis
RCS TEMPERATURE vs. TIME
590
580
570
560
550
540
530
RCS AVERAGE TEMPERATURE (°F)
0
20
40
60
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 2.3-4

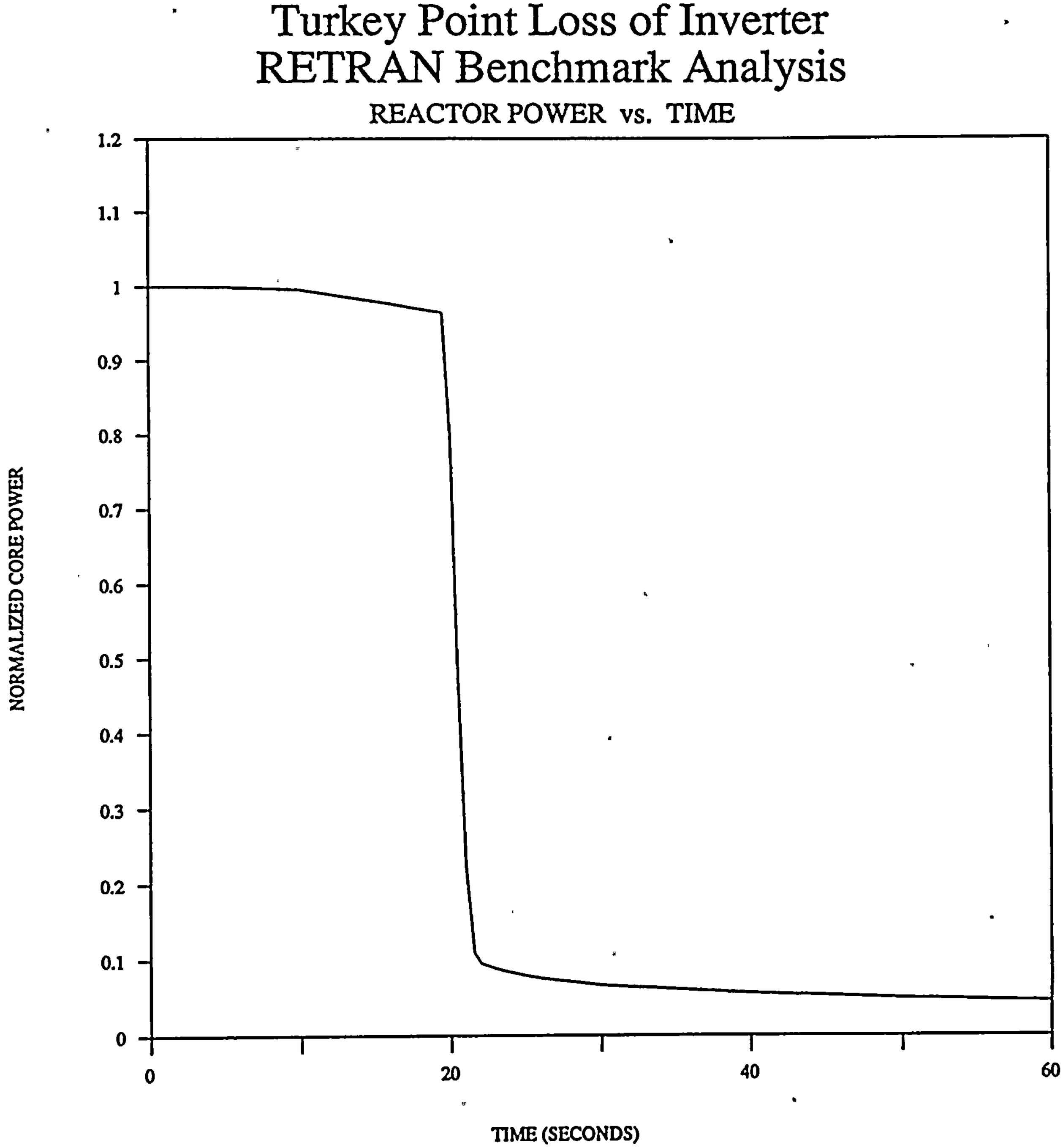

Turkey Point Loss of Inverter
RETRAN Benchmark Analysis
REACTOR POWER vs. TIME
NORMALIZED CORE POWER
1.2
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
20
40
60
TIME (SECONDS)

FIGURE 2.3-5

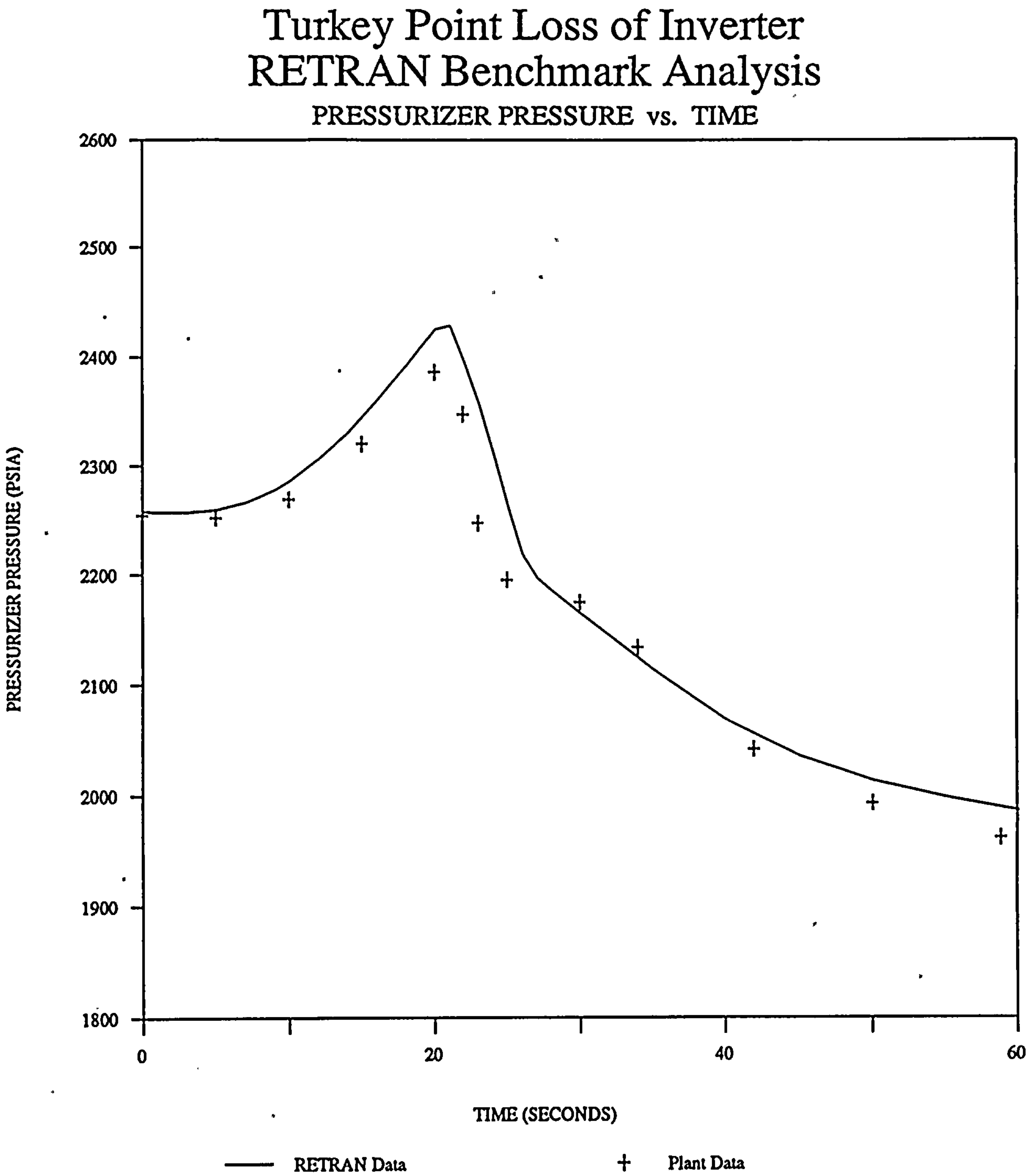
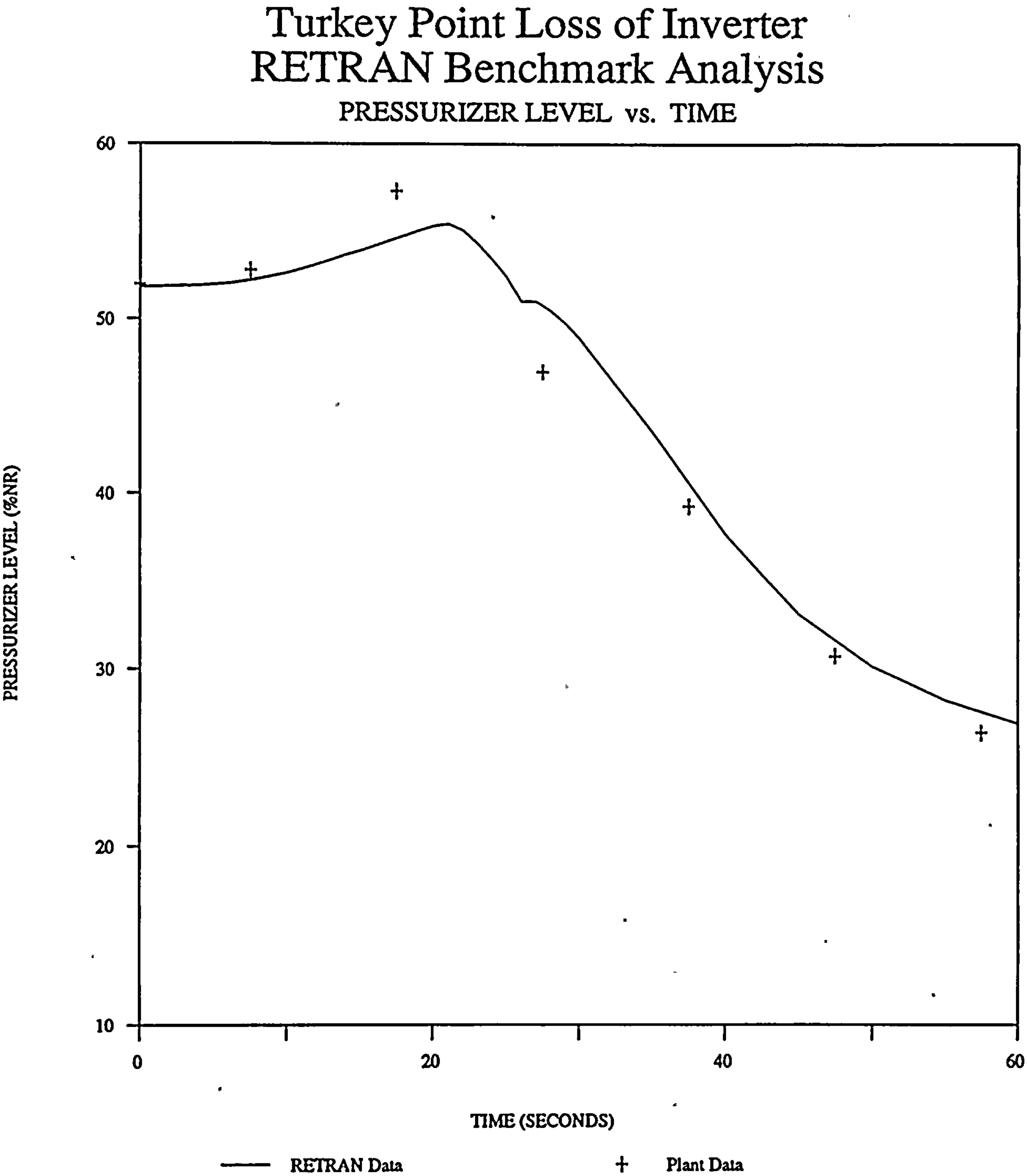

Turkey Point Loss of Inverter
RETRAN Benchmark Analysis
PRESSURIZER PRESSURE vs. TIME
PRESSURIZER PRESSURE (PSIA)
2600
2500
2400
2300
2200
2100
2000
1900
1800
0
20
40
60
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 2.3-6



Turkey Point Loss of Inverter
RETRAN Benchmark Analysis
PRESSURIZER LEVEL vs. TIME
PRESSURIZER LEVEL (%NR)
60
50
40
30
20
10
0
20
40
60
TIME (SECONDS)
RETRAN Data
Plant Data

[illegible]

3.0 St. Lucie Unit 1 Partial Loss of Feedwater Flow

3.1 Summary of Events

On September 20, 1988 a plant trip occurred at St. Lucie Unit 1. The initiating event was a loss of power to the "B" Main Feed Regulating Valve (MFRV) controller. The loss of power caused the "B" MFRV to shut and a loss of feedwater to the "B" Steam Generator to occur. The loss of feedwater resulted in a reactor trip due to reaching the low SG level setpoint, set at the plant at approximately 39.25% of narrow range. Following the trip, an Auxiliary Feedwater Actuation Signal (AFAS) was generated.

In this transient, the primary pressure did not reach the Power Operated Relief Valve (PORV) setpoint of 2400 psia. The Main Steam Safety Valves (MSSV) opened to relieve secondary system pressure in conjunction with the operation of the Steam Dump and Bypass System (SDBS). After reset of the MSSV's, decay heat is removed through action of the SDBS. No unusual actuations or operator actions occurred.

3.2 Analysis

The RETRAN analysis of a plant transient can be performed to validate the ability of the model to accurately predict actual plant performance. The key RETRAN component model which is being examined with this transient is the multi-node model used for tracking the behavior of the water level in the steam generators. A noding diagram for the multi-node steam generator model is shown in Section 1.2. This transient was chosen since it demonstrates a loss of secondary system heat removal which results in a low steam generator trip rather that a high pressurizer pressure trip. The characteristics of the Loss of Normal Feedwater or Loss of AC type events within the category of Decrease in Heat Removal by the Secondary System are similar to the event that occurred at St. Lucie Unit 1.

One of the difficulties in examining the comparison of RETRAN to the plant results in this case is the lack of data from the plant instrumentation. Most of the parameters of interest were only recorded every 10 seconds. The data available is plotted as points on the comparison graphs rather than as lines since linear interpolation over such a large time interval is not representative of the way the plant responded.

This plant event was analysed using the initial conditions summarized in Table 3.2-1. A discussion of the assumptions and initial conditions follows. The basis for the plant data is found in References 9 and 10.

1) Simulation Time

Only the first 100 seconds of the event are provided for comparison of plant data to the RETRAN simulation. This event is an uncomplicated reactor trip on low level and data beyond 100 seconds provides no insights on the ability of the calculational models to accurately predict plant response during the time period important to licensing analysis.

2) Feedwater Assumptions

The main feedwater flow to SG "B" is ramped to zero in 10 seconds after initiation of the event. Main feedwater to SG "A" is assumed to be in automatic mode of control. After the reactor trip signal, main feedwater to SG "A" was also lost with flow being reduced to zero in 10 seconds. While data on feedwater flow to SG "A" was available for each second of the event, data for SG "B" was not available and the assumption of a linear rampdown was chosen.

3) Pressurizer Pressure Control System

The pressurizer pressure control system is assumed to be in automatic mode and is available when needed.

4) Plant Initial Conditions

Plant initial conditions are shown in Table 3.2-1. The initial conditions for the key parameters of interest in this event, i.e., initial SG pressure and level, were available from the plant data.

5) Reactor Protection Delay Times

The time delays associated with the action of the Reactor Protection System (RPS) is taken from the plant sequence of events recorder. The time delays recorded for opening reactor trip breakers and closing of the turbine stop valves are significantly shorter than the values typically assumed. For purposes of evaluating this event, the time delays as recorded at the plant will be assumed in the RETRAN calculation.

TABLE 3.2-1

ST. LUCIE UNIT 1

INITIAL CONDITIONS FOR PARTIAL LOSS OF FEEDWATER EVENT

PLANT CONDITIONS	VALUE
POWER LEVEL (% OF NOMINAL)	100
TIME IN CYCLE 9	BOC
PRESSURIZER PRESSURE (PSIA)	2250
SG "A" LEVEL (% NR)	65.9
SG "B" LEVEL (% NR)	69.9
SG PRESSURE (PSIA)	888
COLD LEG TEMPERATURE (DEG. F)	549
RCS AVERAGE TEMPERATURE (DEG. F)	574
FEEDWATER TEMPERATURE (DEG. F)	434

3.3 Results

The results of the RETRAN analysis are provided in Section 3.3.1. Section 3.3.1 discusses the RETRAN analysis results relative to the analysis assumptions and RETRAN modelling used for this event. Section 3.3.2 provides a discussion of the main differences between the results predicted by RETRAN and the plant data.

3.3.1 RETRAN Analysis

A sequence of events as calculated by RETRAN is provided in Table 3.3-1. As shown, the key actions during this transient are found in the secondary system, which is discussed in Section 3.3.1.1. The response of the secondary system is described over two time intervals due to the different actions occurring before and after reactor/turbine trip.

3.3.1.1 Secondary System Response

1) Time Interval 0 to 21 seconds

Following the loss of power to MFRV "B", main feedwater to SG "B" is assumed to ramp down to zero flow in 10 seconds. As the feedwater flow is reduced, the SG level for SG "B", Figure 3.3-1, shows a gradual reduction. When all feedwater is isolated, the level begins to reduce much more rapidly with the low level signal occurring at 21.6 seconds after the initiation of the event. The SG "B" pressure response is shown in Figure 3.3-2. The use of the multi-node SG model provides a much more sensitive response to small steam flow variations which are occurring between the two SG's than the use of a single node model. As shown in Figure 3.3-2, the SG pressure increases after feedwater flow has stopped. The pressure increases until action of the SDBS at 20 seconds begins to mitigate any further pressurization.

SG "A" behavior is shown in Figures 3.3-3 and Figures 3.3-4. For the first 21 seconds of the event, RETRAN calculates only a small increase in SG level. This increase is due to a slightly increased MFW flow to SG "A" that occurs when the "B" MFRV closes as both MFW pumps continue to operate. The drop in pressure that begins at approximately 20 seconds is due to the opening of the SDBS which has reacted to the pressurization of SG "B".

TABLE 3.3.1-1

SEQUENCE OF EVENTS

RETRAN ANALYSIS PARTIAL LOSS OF FEEDWATER

ST. LUCIE UNIT 1

EVENT	TIME (seconds)	SETPOINT OR VALUE
Loss of Power to MFRV	0.0	---
MFRV SG "B" Fully Closed	10.0	---
SDBS Begins to Open	20.0	---
Low SG "B" Level Trip	21.63	39.25 % NR
Reactor Trip Breakers Open	21.67	---
Turbine Stop Valves Closed	22.17	---
SDBS Full Capacity	23.0	---
Rampdown of Feedwater to SG "A"	25.0	AFAS Signal
CEA's Fully Inserted	25.1	---
SG "B" MSSV's Open	25.5	1000 psia
SG "A" MSSV's Open	26.0	1000 psia
SDBS Begins to Modulate Flow	41.0	---
MSSV's Close	44.0	920 psia

2) <u>Time Interval 21 to 100 seconds</u>

After the reactor trip and turbine trip, RETRAN calculates a swift increase in secondary pressure until the MSSV first bank setpoint is reached at 25.5 seconds for SG "B". The action of the MSSV's and the SDBS reduce the pressure until the MSSV's close. Plant data showed that the pressure for fully closing the MSSV's was at 920 psia. After the closure of the MSSV's, RETRAN predicts a slight re-pressurization which is turned around by the continued action of the SDBS. The changes in de-pressurization rates seen at 32.0 and 37.0 seconds are related to the sensitivity of the multi-node model. The changes in de-pressurization rates are tied directly to observed steam flow changes as both SG's compete to provide flow to the SDBS and the MSSV's. The increase in "B" SG level starting at approximately 27 seconds corresponds to the initiation of the secondary system pressure turnaround through action of the MSSV's and SDBS. This represents only a short term effect since there is a continuation of steam flow with no feedwater addition throughout this time interval. The level decrease is slowed when the MSSV's close and the SDBS begins to modulate and shows only a small rate of decrease until the end of the simulation.

The response of SG "A" is very similar in this time frame to SG "B" discussed above. The changes in SG "A" pressure after the turbine trip are associated with the changes in the steam flow. The SDBS acts as a constant demand in the RETRAN model which combined with the higher and earlier pressurization of SG "B" results in the behavior seen on Figure 3.3-3. In addition, the MSSV model used in the evaluation of this event is the simple flow versus pressure model discussed in Section 1.1. This model calculates oscillatory flows, that is, when pressure increases the flow will increase which in turn reduces the pressure which reduces the flow and so on. This results in the pressure spike behavior seen in the time interval of 23 to 30 seconds.

The level response shown in Figure 3.3-4 shows a decrease in level which occurs when the feedwater flow is lost to SG "A" and the pressure increases due to the turbine trip. The change in slope in the level decrease which is seen at approximately 32 seconds is related to the reduction in pressure and correspond to the changes in slope shown in Figure 3.3-3.

3.3.1.2 Primary System Response

The primary system response as calculated by RETRAN follows the effects seen in the secondary system. As the feedwater to SG "B" ramps to zero, RETRAN calculates a small insurge to the pressurizer and a consequent increase in primary system pressure. This behavior is shown in Figure 3.3-5. After the reactor trip, the steam relief from the SDBS and the MSSV's are sufficient to produce a de-pressurization. The rate of de-pressurization is slowed when the MSSV's close at 44 seconds into the event.

The pressurizer level is shown in Figure 3.3-6. The behavior of the level is consistent with that calculated for the pressurizer pressure.

The responses calculated by RETRAN for Tavg for loops "A" and "B" are shown in Figures 3.3-7 and 3.3-7 respectively. After the reactor trip, Tavg is calculated to decrease through the action of the MSSV's and the SDBS. A RTD delay time of 3.0 seconds is included in the values plotted for the RETRAN simulation.

3.3.2 RETRAN Comparison to Plant Data

A sequence of events comparing the RETRAN simulation described in Section 3.3.1 to the plant data is shown in Table 3.3.2-1. As shown the calculated time for reactor trip is earlier than the time inferred from the plant data. The basic trends of the data show excellent agreement. In particular, as shown on Figures 3.3-1 and 3.3-4, the SG level response as calculated by the multi-node SG model in RETRAN shows a close agreement to the plant data.

One source of the differences between RETRAN and the plant comparisons that was investigated was the short delay times measured by the plant and used in the RETRAN simulation. The response of the primary system indicates that the use of a longer delay time between reactor trip signal and rod motion would improve the timing of the primary system responses considerably. That is, the delay of the reactor scram would result in the primary cooldown being delayed in the RETRAN simulation and therefore the comparison would be better to the plant data. The Technical Specifications for low SG trip signal delay time is 1.15 seconds. Plant sequence of events recorder show completion of the low level trip logic and reactor trip breaker opening to take only 0.04 seconds. Discussion with Plant Staff after this benchmark was completed indicates that some additional time should have been credited for sensor delay times, however, this additional time would add only about 0.1 seconds to the reactor trip delay. This small additionally delay would not make a significant difference in the RETRAN results and the comparison to plant data.

TABLE 3.3.2-1

SEQUENCE OF EVENTS

ST. LUCIE UNIT 1 PARTIAL LOSS OF FEEDWATER

RETRAN COMPARISON TO PLANT DATA

EVENT	TIME (seconds) Plant Data	RETRAN
Loss of Power to MFRV	0.0	0.0
"B" MFRV Fully Closed	---	10.0
SDBS Begins to Open	---	20.0
Low SG "B" Level Trip	25.4	21.63
Reactor Trip Breakers Open	25.44	21.67
Turbine Stop Valves Closed	25.94	22.17
SDBS Full Capacity	---	23.0
Reduction Feedwater SG "A"	28.7	25.0
CEA's Fully Inserted	---	25.1
SG "B" MSSV's Open	29.0	25.5
SG "A" MSSV's Open	29.0	26.0
SDBS Begins to Modulate	---	41.0
MSSV's Close	53.0	44.0

As shown, it is clear that RETRAN is calculating a faster depressurization than the plant data indicates. Steam flow was not available in enough detail to determine the root cause of the differences, however, one possibility is the MSSV model used in the RETRAN simulation. The noding used in RETRAN places all the MSSV's in the same volume which is then represented by one composite valve for each of the two banks. Therefore, our RETRAN model opens all of the four MSSV's/SG that compose the first bank when the pressure setpoint is reached. This modelling would not be expected to simulate exactly the actual behavior at the plant since the valves are sure to have differences in opening setpoints as well as being physically separated. In addition, it would not be expected that all the MSSV's would close at the same setpoint as assumed in the RETRAN analysis.

Calculations, therefore, were made which varied the flow capacity allowed through the MSSV's in RETRAN. The secondary system response was closest to the plant data when it was assumed that only 75% of the MSSV capacity operated, i.e., only three of the four valves operated per SG. The results of that calculation are shown for SG pressure response in Figures 3.3-9 and 3.3-10. These results demonstrate a closer match to the measured data throughout the event for this parameter.

3.4 Conclusions

A partial Loss of Feedwater Flow event has been examined using the FPL RETRAN model. The results of the comparisons to plant data shows good agreement in trends and demonstrates the ability of the SG multi-node model to accurately predict level response.

FIGURE 3.3-1

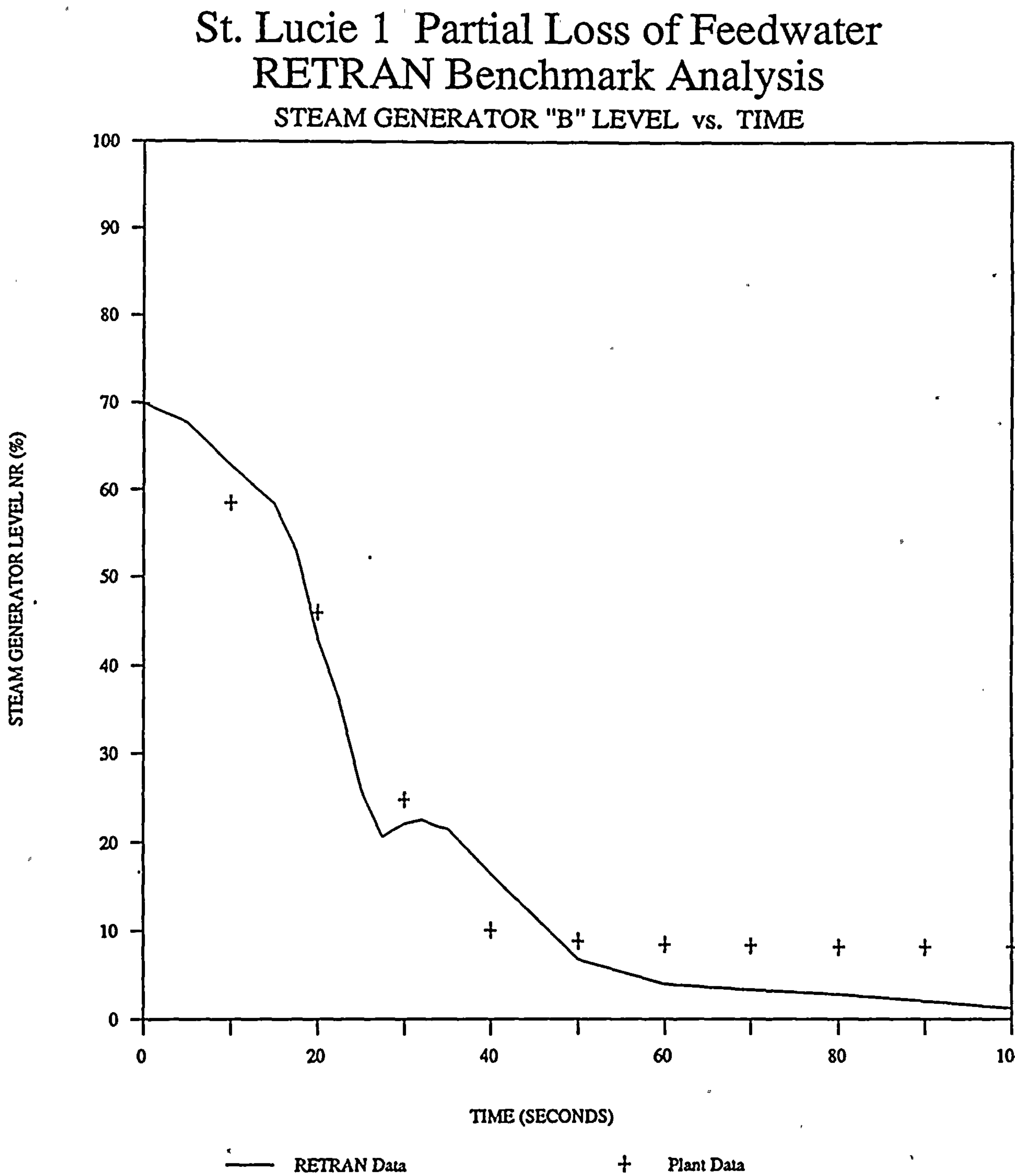

St. Lucie 1 Partial Loss of Feedwater
RETRAN Benchmark Analysis
STEAM GENERATOR "B" LEVEL vs. TIME
100
90
80
70
60
50
40
30
20
10
0
STEAM GENERATOR LEVEL NR (%)
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 3.3-2

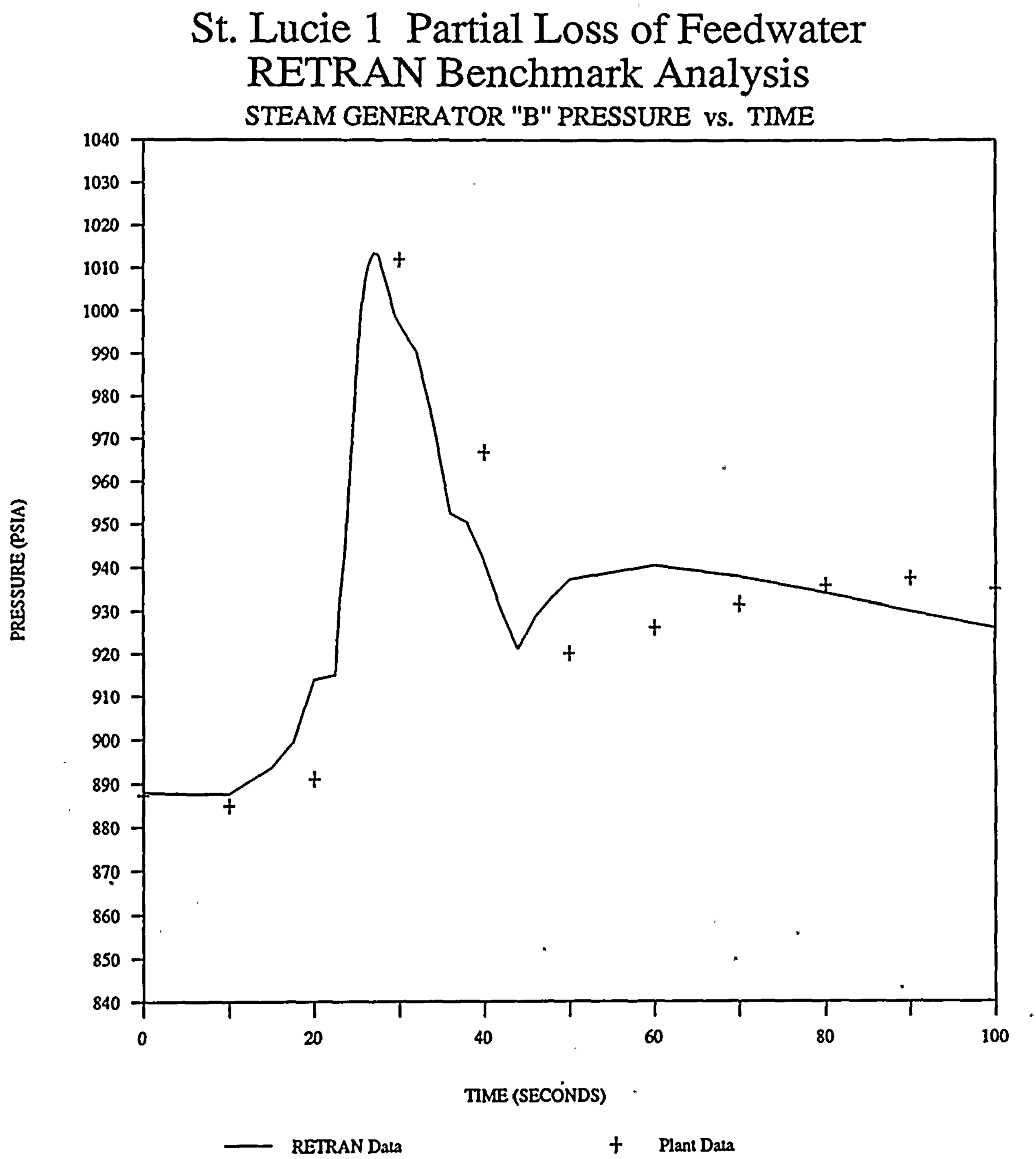

St. Lucie 1 Partial Loss of Feedwater
RETRAN Benchmark Analysis
STEAM GENERATOR "B" PRESSURE vs. TIME
PRESSURE (PSIA)
1040
1030
1020
1010
1000
990
980
970
960
950
940
930
920
910
900
890
880
870
860
850
840
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 3.3-3

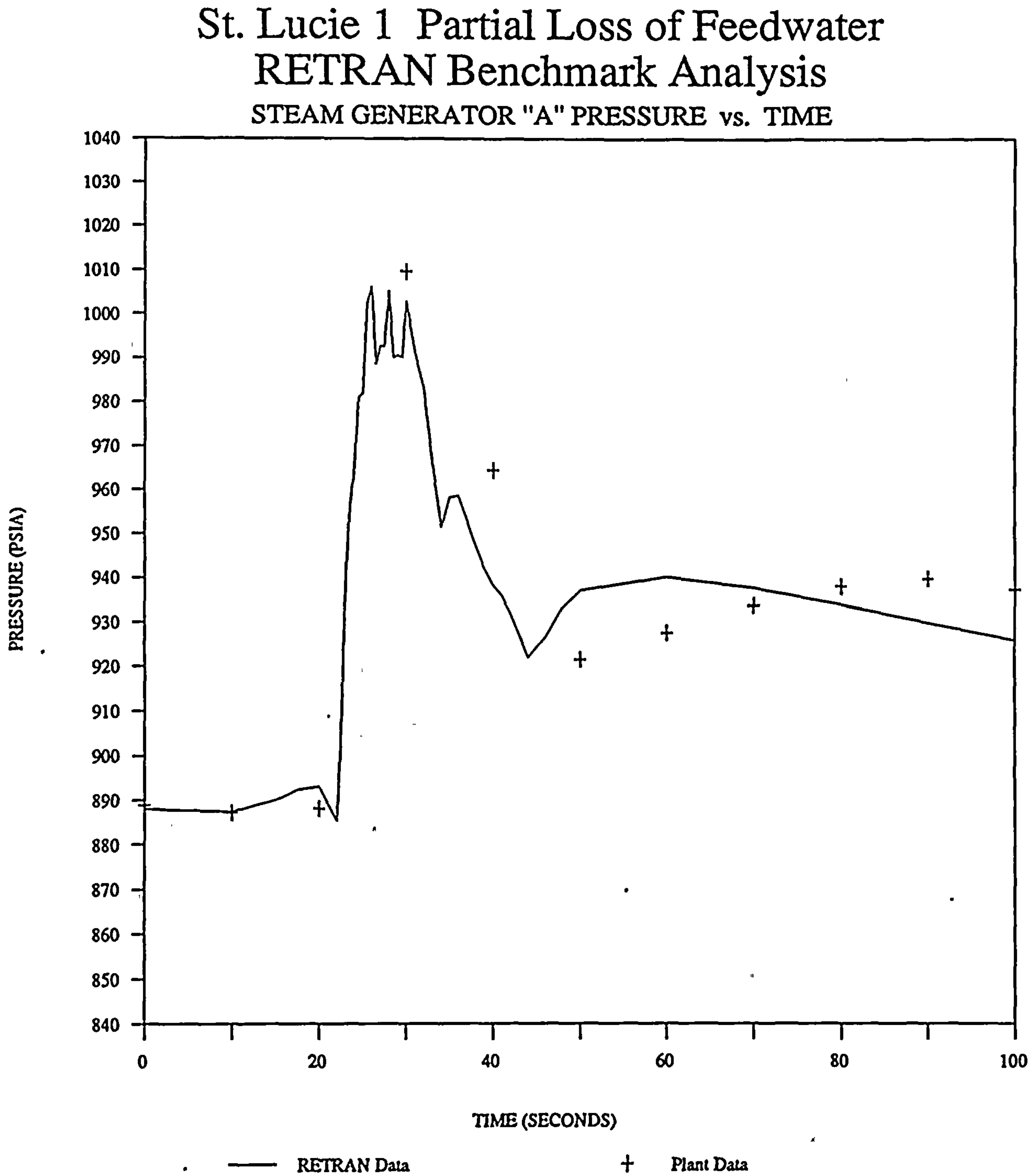

St. Lucie 1 Partial Loss of Feedwater
RETRAN Benchmark Analysis
STEAM GENERATOR "A" PRESSURE vs. TIME
1040
1030
1020
1010
1000
990
980
970
960
950
940
930
920
910
900
890
880
870
860
850
840
PRESSURE (PSIA)
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 3.3-4

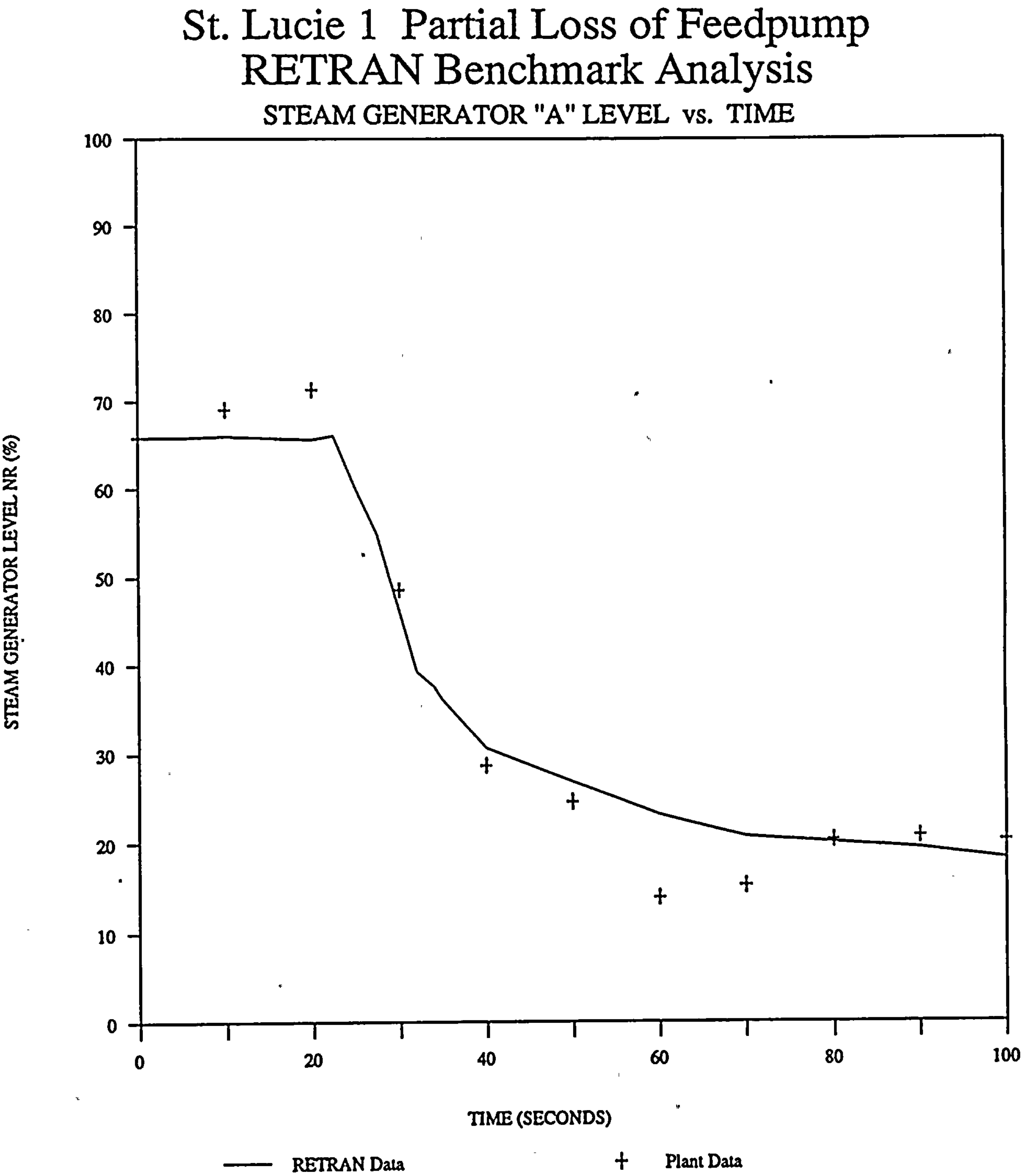

St. Lucie 1 Partial Loss of Feedpump
RETRAN Benchmark Analysis
STEAM GENERATOR "A" LEVEL vs. TIME
STEAM GENERATOR LEVEL NR (%)
100
90
80
70
60
50
40
30
20
10
0
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 3.3-5

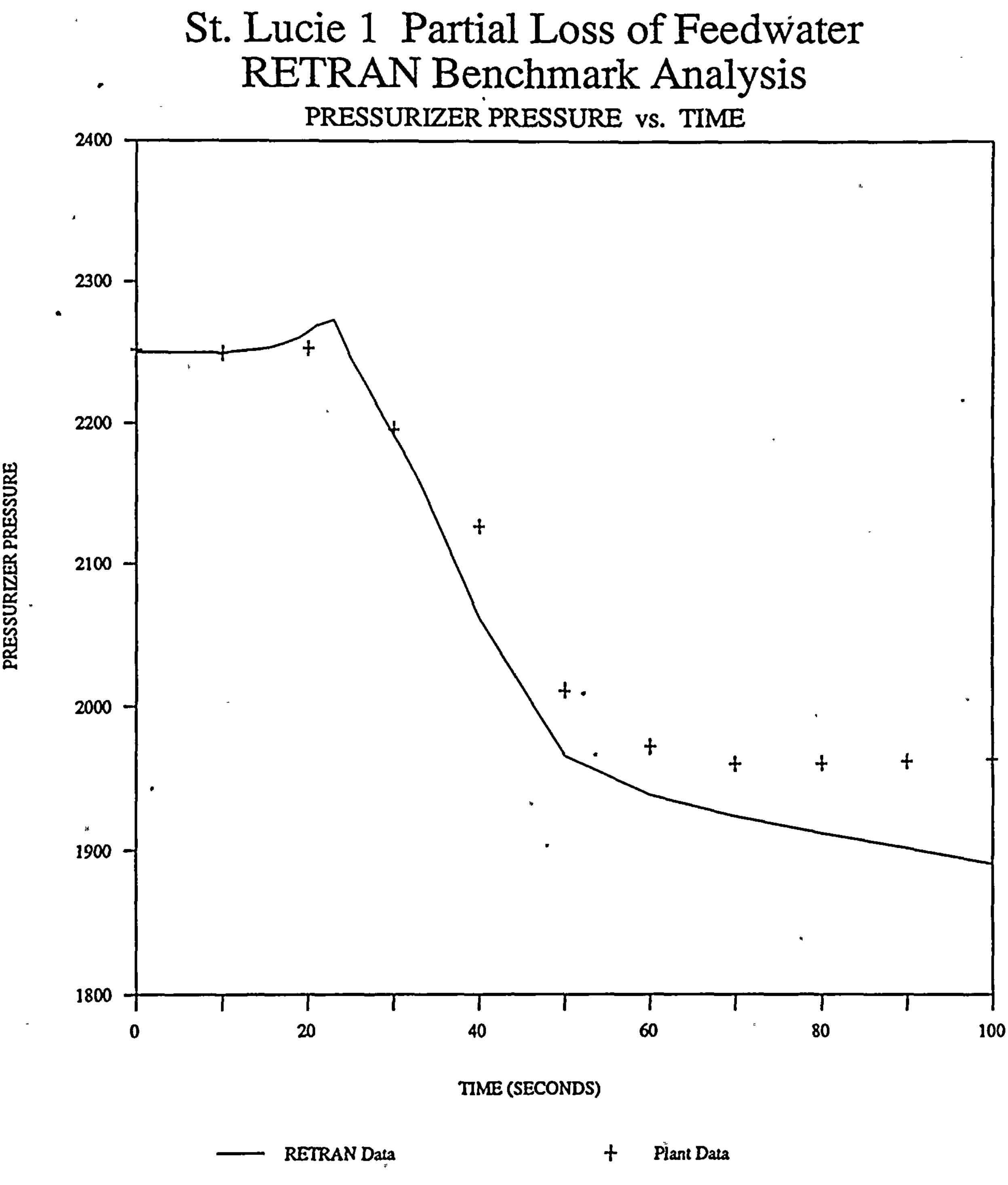
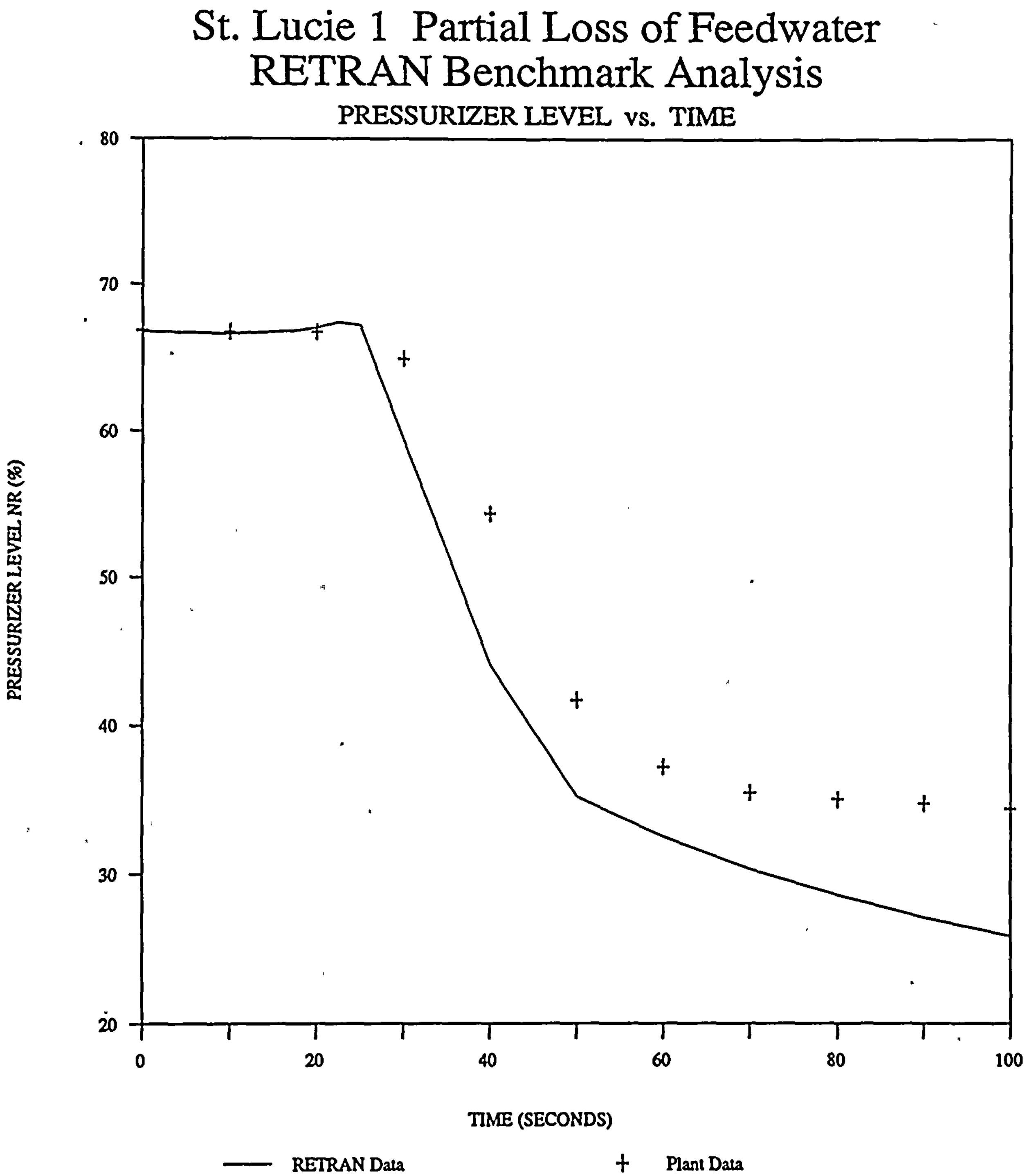

St. Lucie 1 Partial Loss of Feedwater
RETRAN Benchmark Analysis
PRESSURIZER PRESSURE vs. TIME
2400
2300
2200
2100
2000
1900
1800
PRESSURIZER PRESSURE
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 3.3-6



St. Lucie 1 Partial Loss of Feedwater
RETRAN Benchmark Analysis
PRESSURIZER LEVEL vs. TIME
PRESSURIZER LEVEL NR (%)
80
70
60
50
40
30
20
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 3.3-7

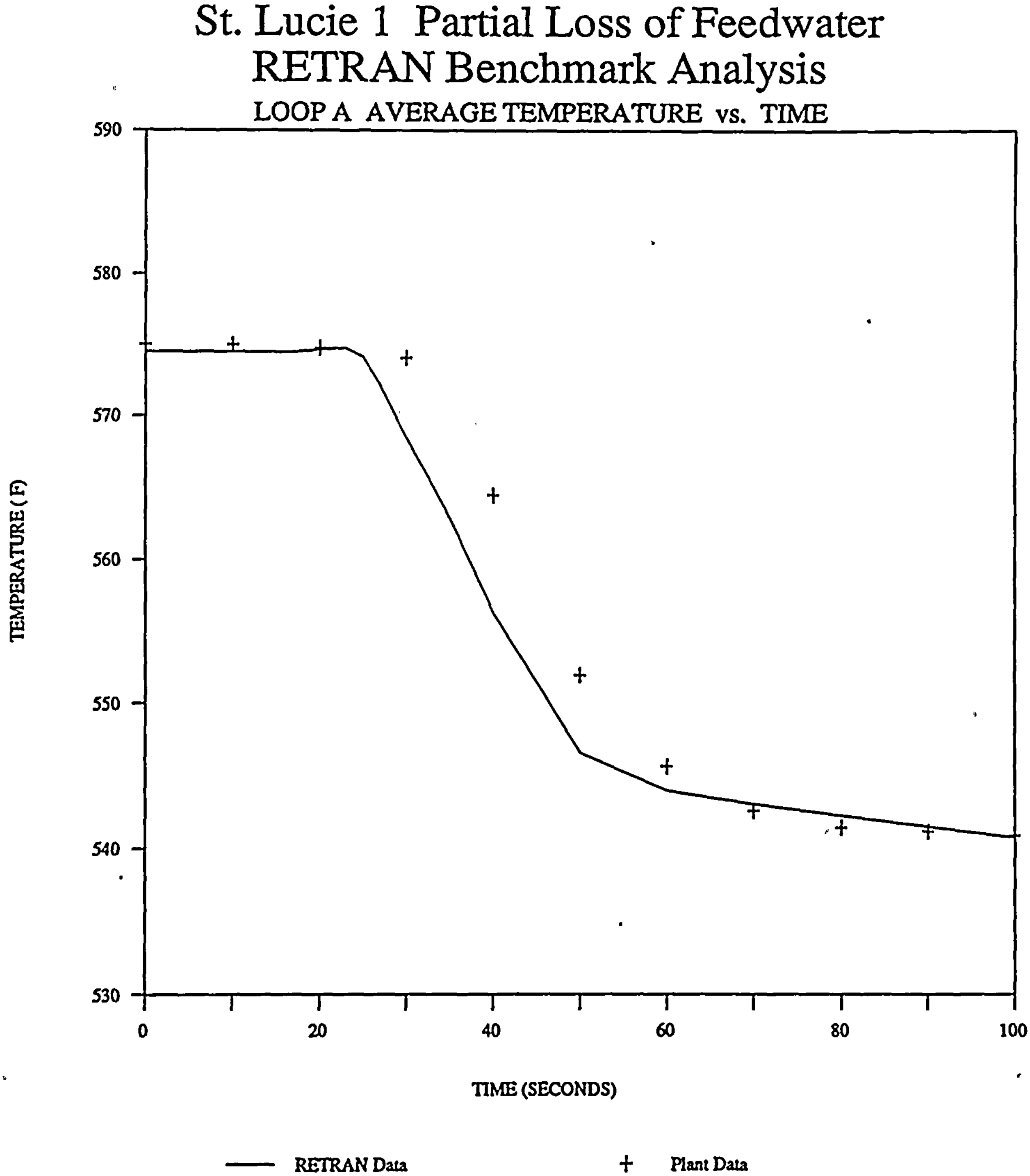

St. Lucie 1 Partial Loss of Feedwater
RETRAN Benchmark Analysis
LOOP A AVERAGE TEMPERATURE vs. TIME
590
580
570
560
550
540
530
TEMPERATURE (F)
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 3.3-8

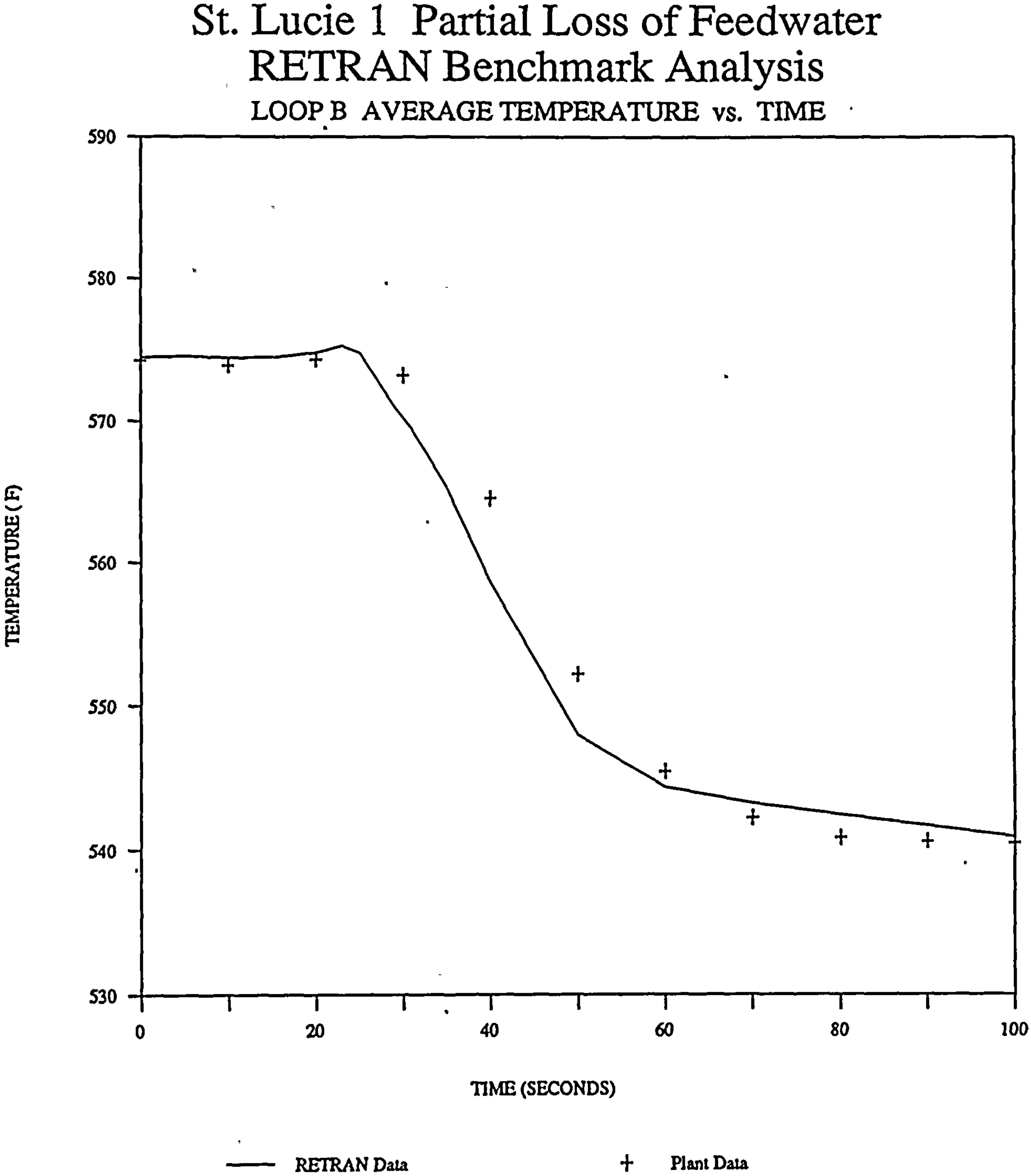
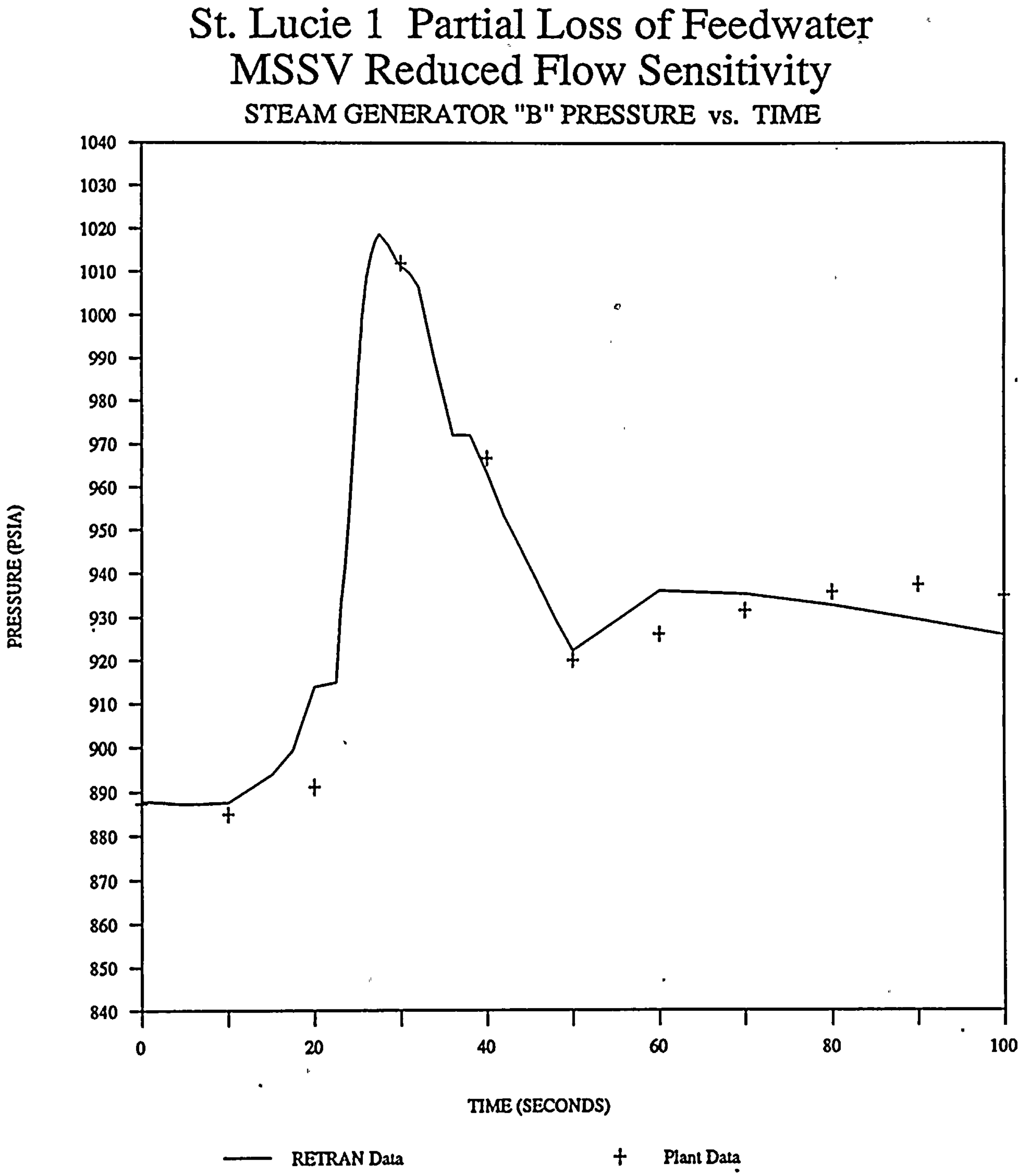
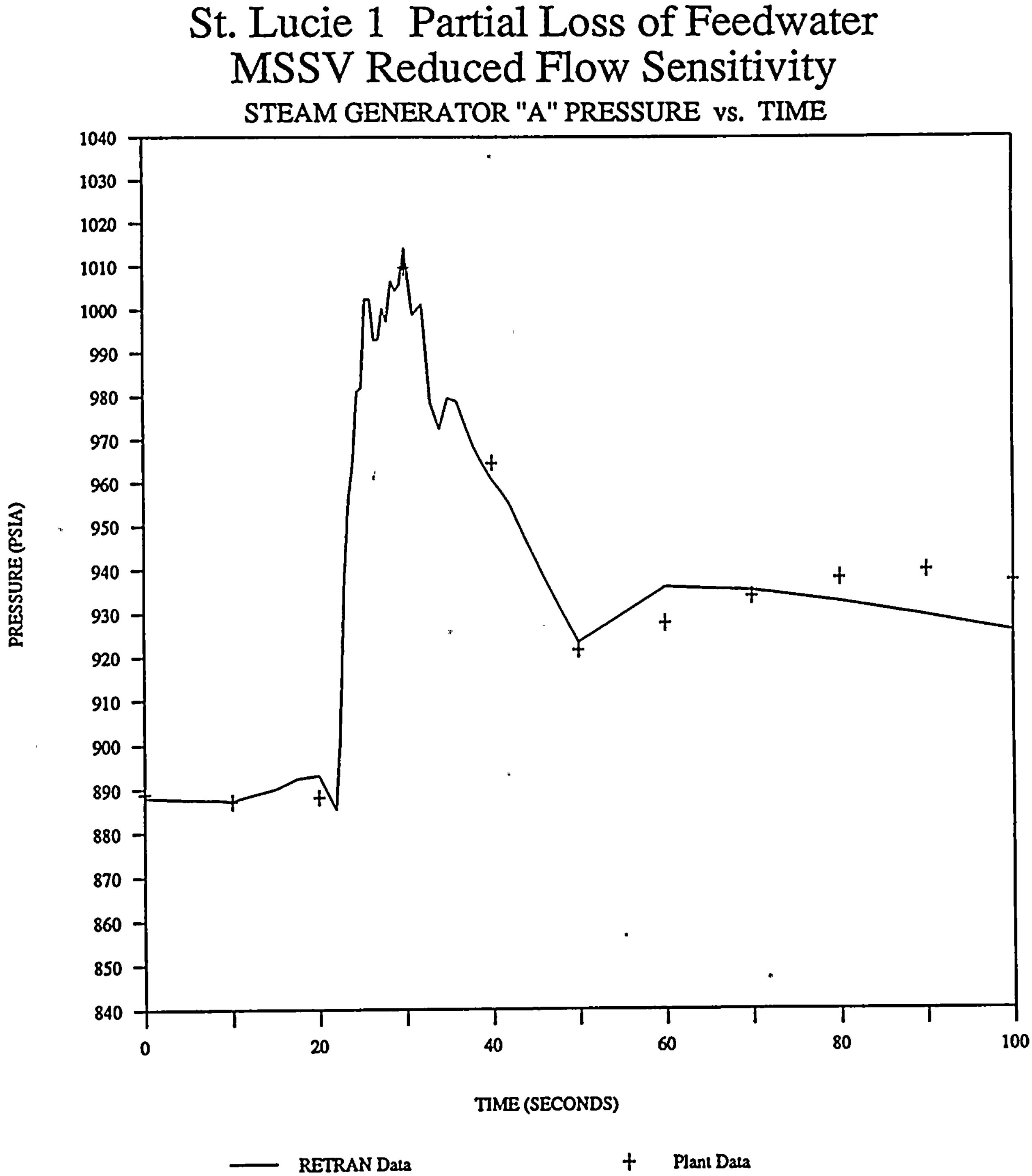

St. Lucie 1 Partial Loss of Feedwater
RETRAN Benchmark Analysis
LOOP B AVERAGE TEMPERATURE vs. TIME
590
580
570
560
550
540
530
TEMPERATURE (F)
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 3.3-9



St. Lucie 1 Partial Loss of Feedwater
MSSV Reduced Flow Sensitivity
STEAM GENERATOR "B" PRESSURE vs. TIME
1040
1030
1020
1010
1000
990
980
970
960
950
940
930
920
910
900
890
880
870
860
850
840
PRESSURE (PSIA)
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

FIGURE 3.3-10


St. Lucie 1 Partial Loss of Feedwater
MSSV Reduced Flow Sensitivity
STEAM GENERATOR "A" PRESSURE vs. TIME
PRESSURE (PSIA)
1040
1030
1020
1010
1000
990
980
970
960
950
940
930
920
910
900
890
880
870
860
850
840
0
20
40
60
80
100
TIME (SECONDS)
RETRAN Data
Plant Data

[illegible]

4.0 Safety Analysis Methods

4.1 Introduction

The events within the Decrease in Heat Removal from the Secondary System category are discussed in Standard Review Plan (SRP) Section 15.2 (Reference 11). Events in this category are listed in Table 4.1-1 with the numerical designation found in the SRP. The events in this category typically produce a reduction in secondary side heat removal which causes a heatup and pressurization of the primary side. The event and the associated pressurization is eventually terminated by a reactor trip.

The first task in developing the safety analysis methodology to be used with this category of events is to discuss each event and determine the limiting transients for each of FPL's plants. After the limiting transients have been identified, the key input parameters that will be used in the licensing analysis of the event will be discussed and justified. The analysis of the event will then be performed to provide the results of the limiting transient. This process is repeated for all three units in order to provide full and complete information in a format designed to facilitate FSAR updates when the FPL methods are utilized in the future. This approach has generated sections which are very similar from unit to unit since the parameters which impact the results of the events are in general the same, irrespective of the vendor.

It should be noted that the choice of the limiting event as well as the limiting set of input parameters is a straight-forward process. That is, these events are evaluated to ensure that under limiting conditions the appropriate acceptance criteria, as defined in the SRP will be met. For the events in the Decrease in Heat Removal from the Secondary System category, the key acceptance criteria is maintaining peak pressures, both primary and secondary, below 110% of design. Pressurizer level must also be examined since there is a potential for going water solid in the pressurizer during these events. In addition, for the Loss of Normal Feedwater type events an additional safety concern is maintaining an adequate water level in the Steam Generators to remove decay heat.

These criteria will form the basis of the determination of limiting transient and limiting input for the FPL safety methodology in the Decrease in Heat Removal by the Secondary System category of events. While transients within this category are analyzed primarily because of the potential to exceed the 110% design pressure limits on the primary and secondary system, other safety criteria (such as Departure From Nucleate Boiling Ratio (DNBR) behavior) will be evaluated in the detailed discussions for each transient. The major focus, however, in the limiting transient determination will be on assuring that the transient (with its associated input) will produce the limiting pressure excursion.

Table 4.1-1

Standard Review Plan Events

Decrease in Secondary Side Heat Removal

SRP Event Number	Event Title
15.2.1	Loss of External Load
15.2.2	Turbine Trip
15.2.3	Loss of Condenser Vacuum
15.2.4	Main Steam Isolation Valve Closure (BWR)
15.2.5	Steam Pressure Regulator Failure
15.2.6	Loss of Non-emergency AC Power to the Station Auxiliaries
15.2.7	Loss of Normal Feedwater Flow
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)

4.1.1 RETRAN MODELLING

Based on the results of the modelling sensitivity studies performed for the Loss of Inverter analysis (Section 2.0) and other licensing considerations, several changes to the modelling approaches used in the base models (see Section 1.2.3) have been evaluated for the methodology used to examine Decrease in Heat Removal by the Secondary System events. These changes to the base models along with the basis for either incorporating the change or maintaining the current models are described below. The main modelling approaches adopted in the FPL RETRAN models for the analysis of this category of events is summarized in Table 4.1-2.

4.1.1.1 Safety Valves

The safety valves model in the current RETRAN base decks used for best estimate evaluations, (Section 1.2.3.3) open and close at the respective setpoint pressures with small delays and start discharging full design flow at the opening setpoint pressure. This flow is allowed to increase as the pressure continues to increase. This model has been upgraded for licensing applications to include accumulation and hysteresis and also to limit the discharge flow at the design value. The changes affect only the fill part of the model where the critical flow table is now multiplied by a valve flow area fraction which is based on the hysteresis curves shown in Figure 4-1. Based on experimental data (Reference 13) on safety valves of very similar design that shows that the valves open and close nearly instantaneously, the time dependence for the valve opening in the models has been removed. The opening and closing of the valves is now modelled as pressure dependent (Figure 4-1) instead of time dependent. The discharge flow, in the new model, does not reach the design value until the accumulation pressure (opening setpoint pressure + 3%) is reached.

4.1.1.2 Pressurizer Model

The results of the sensitivity studies performed on some of the pressurizer parameters (Section 2.4) support the use of the modelling used in the current RETRAN base models for the events of interest. These studies also confirm that peak pressures could be slightly increased if the surge line hydraulic resistances were increased. However the calculated effect was very small and since the current method predicts a conservatively high pressure compared to plant data, changes to the resistances will not be included. Therefore no changes to the pressurizer modelling used in the current base decks as discussed in Section 1.2.3, are recommended. The modelling to be utilized is summarized in Table 4.1-2.

4.1.1.3 Temperature Transport Delay

The observed sensitivity of the peak pressure to the use of the temperature transport delay is small (Section 2.4) and will not be used for Decrease in Heat Removal by the Secondary System Events.

4.1.1.4 Multi-node Steam Generator Model

The results of the Loss of Inverter benchmark analysis presented in Section 2.0 confirm the use of the single node steam generator as a valid nodalization approach for situations where the prediction of level is not crucial. The single node steam generator model is justified because, in addition to its simplicity of use, it tends to over-predict the primary peak pressure. Based on this, the single node model will be used in analyses of Decrease in Heat Removal by the Secondary System events that do not require accurate level predictions or do not involve severe losses of steam generator inventories. The use of the single node SG model for evaluation of peak pressure for this category of event is consistent with the methods used by other RETRAN users (Reference 12) as well as that used by the fuel vendors (References 14-16).

For situations in which the evolution of the transient depends on the level prediction, such as in the Loss of Non-Emergency AC Power for Turkey Point or the Feedwater Line Break for St. Lucie Unit 2, the multi-node steam generator model will be used.

TABLE 4.1-2

FPL RETRAN METHODOLOGY

DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM EVENTS

PRESSURIZER MODEL

-Single Node
-Non-Equilibrium Option
-Inter-Region HTC = 50.0 Btu/hr-ft^2-F
-No Metal Heat
-Nominal Hydraulic Resistances

OTHER PRIMARY MODELLING OPTIONS

-No Temperature Transport Delay

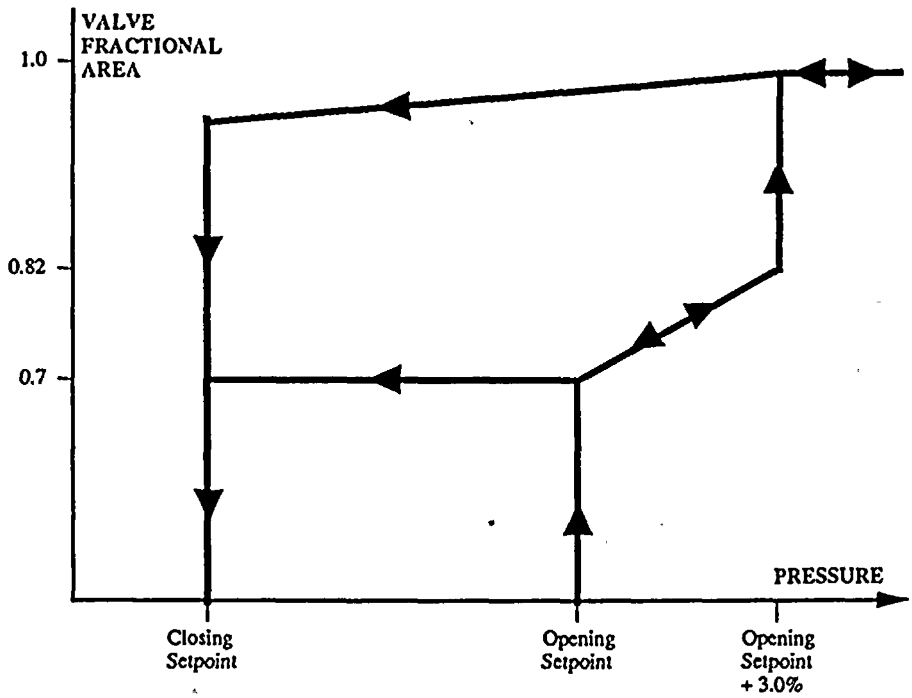
STEAM GENERATOR

-Single Node if Level is not Required
-Multi Node if Level is Required

SAFETY VALVES

-3% Accumulation and Hysteresis Included
-Design Flow Assumed at Accumulation
-Flow Limited to Design Value

Secondary Safety Hysteresis



Primary Safety Hysteresis

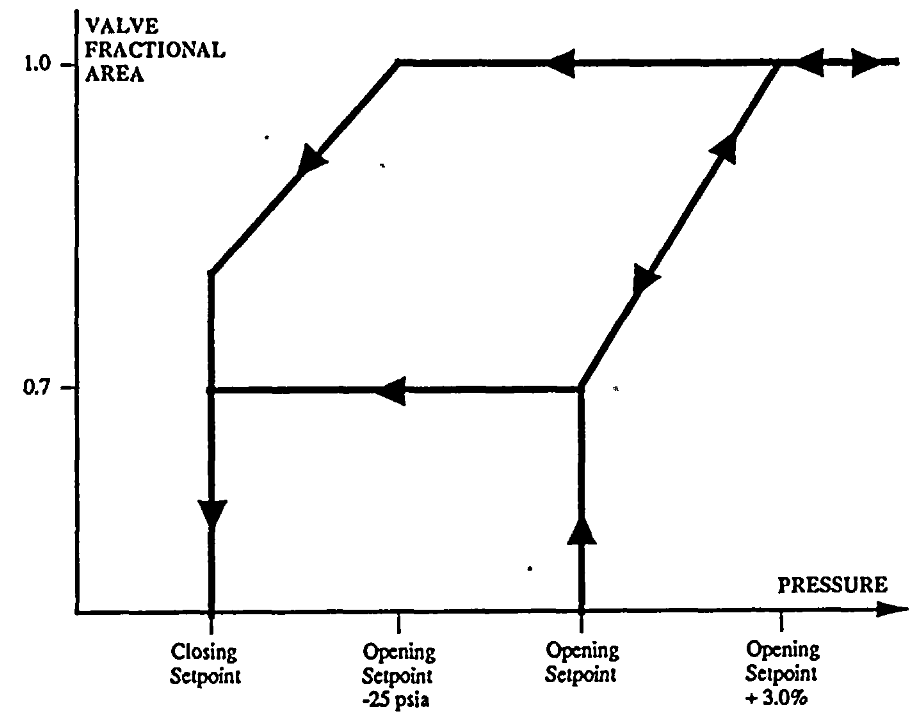


FIGURE 4-1
Primary and Secondary Safety Valves Hysteresis in the RETRAN licensing models.

4.2 LIMITING TRANSIENT DETERMINATION

4.2.1 ST. LUCIE UNIT 1

4.2.1.1 Loss of External Load

A loss of external load event is caused by abnormal events in the electrical distribution network. The turbine is protected from a complete loss of external load by two overspeed protection systems which act to trip the turbine if a complete loss of load occurs. The Overspeed Protection Controller and the mechanical overspeed protection system are both designed to trip the turbine at approximately 111 percent overspeed condition. More details of these two equipment protection systems are found in Section 10.2.2 of the St. Lucie Unit 1 Final Safety Analysis Report (FSAR). Upon occurrence of a turbine trip above 15% of full power at St. Lucie Unit 1 , a signal would be supplied to the reactor protective system to trip the reactor. Subsequent to the turbine trip, the main feedwater regulating valves would close and feedwater would be supplied to the steam generator through the feedwater bypass valve by the main feedwater pumps.

A fast pressurization transient will result from the closure of the turbine stop valves due to their fast response time of approximately 0.25 seconds. The primary side pressurization results from the reduced primary to secondary heat transfer which occurs with the stoppage of steam flow from the secondary side. However, the reactor trip on turbine trip that would occur would quickly reverse the pressurization event and bring the reactor to a safe shutdown condition before a reactor trip on high pressure could occur. In addition, action of the Steam Dump and Bypass System (SDBS) would mitigate the pressurization effects of an actual Loss of External Load event.

A bounding Loss of Load event is postulated if credit is not taken for the reactor trip on turbine trip or action of the SDBS. However, this transient is bounded by the Loss of Condenser Vacuum event discussed in Section 4.2.1.3 because of the assumption in that event of an instantaneous loss of main feedwater at the initiation of the event. The loss of main feedwater increases slightly the heatup of the primary and therefore increases the pressurization rate of the transient.

Evaluation of the DNBR response to a Loss of External Load Event shows qualitatively that DNBR will vary only slightly during this transient. DNBR is negatively effected by decreases in flow and pressure and increases in heat flux and temperature. Transients which can challenge the DNBR limit result from either significant decreases in flow, i.e., Loss of Flow events or significant increases in heat flux, i.e, Control Element Assembly (CEA) Drop. During the Loss of External Load, the

increase in temperature will be offset by the increase in pressure and only slight variations in DNBR would be expected. At all times in this transient, the Loss of Flow event as described in Section 15.2.5 of the FSAR would remain limiting with regard to DNBR results.

4.2.1.2 Turbine Trip

A turbine trip is an event in which the steam flow from the steam generators is stopped through closure of the turbine control and/or stop valves. The turbine is equipped with an automatic stop and emergency trip system which trips the stop and control valves to a closed position in the event of turbine overspeed, low bearing oil pressure, low condenser vacuum, or thrust bearing failure. Turbine protection devices which could cause a turbine trip are described in depth in Section 10.2.2 of the FSAR. Upon occurrence of a turbine trip above 15% of full power, a signal is supplied to the reactor protective system to trip the reactor. Subsequent to the turbine trip, the main feedwater regulating valves close and feedwater is supplied to the steam generator through the feedwater bypass valve by the main feedwater pumps.

A fast pressurization transient will result from the closure of the turbine stop valves due to their fast response time of approximately 0.25 seconds. The primary side pressurization results from the reduced primary to secondary heat transfer which occurs with the stoppage of steam flow from the secondary side. However, the reactor trip on turbine trip would quickly reverse the pressurization event and bring the reactor to a safe shutdown condition before a reactor trip on high pressure could occur. In addition, action of the SDBS would mitigate the pressurization effects of a Turbine Trip.

A bounding Turbine Trip could be postulated if credit is not taken for the reactor trip on turbine trip or action of the SDBS. However, this transient is bounded by the Loss of Condenser Vacuum event discussed in Section 4.2.1.3 because of the assumption in that event of an instantaneous loss of main feedwater. The loss of main feedwater increases the heatup of the primary and therefore increases the pressurization rate of the transient.

Evaluation of the DNBR response to a Turbine Trip Event shows qualitatively that DNBR will vary only slightly during this transient. DNBR is negatively effected by decreases in flow and pressure and increases in heat flux and temperature. As discussed in Section 4.2.1.1 transients such as Loss of Load and Turbine trip typically show no decrease in DNBR. Clearly, at all times in the transient, the results from the Loss of Flow event as described in Section 15.2.5 of the FSAR remain limiting with regard to DNBR results.

4.2.1.3 Loss of Condenser Vacuum

4.2.1.3.1 Limiting Event Development

A Loss of Condenser Vacuum event may occur due to the failure of the Circulation Water System, the failure of the Main Condenser Evacuation System to remove non-condensible gases, or the in-leakage of an excessive amount of air through a turbine gland. Low condenser pressure generates a turbine trip signal. The turbine trip signal causes the turbine stop and control valve to close. Upon occurrence of a turbine trip above 15% of full power, a signal is supplied to the reactor protective system to trip the reactor. The Loss of Condenser Vacuum event will disable the SDBS and will result in a gradual rampdown of main feedwater.

For purposes of developing an event that bounds the potential for primary pressurization, several conservative assumptions will be made relative to the Loss of Condenser Vacuum event. The following assumptions will be utilized in this analysis:

1) For this bounding event, coincident with the Loss of Condenser Vacuum, the turbine stop valves are assumed to instantly close on high condenser back pressure. This assumption assures the fastest possible pressurization rate.

2) A reactor trip on turbine trip signal will not be credited. Not allowing the reactor trip on turbine trip signal to operate will allow the high pressurizer pressure trip to activate to initiate reactor scram. Use of the high pressurizer pressure trip rather than any other system trip (such as steam generator low level) will insure that the highest possible primary pressure will be reached during the transient.

3) Main feedwater flow rate will be instantaneously set to zero to minimize the secondary side heat removal capacity.

4) Action of the Power Operated Relief Valves (PORV) and pressurizer spray will not be credited since it would mitigate the primary pressurization.

The choice for other key initial conditions are listed in Table 4.2.1.3-1. For evaluation of the peak primary pressure, initial conditions are chosen such that the time to reactor trip on high pressure is maximized and that a high rate of change of pressure is achieved. The basis of the key input parameters is discussed briefly in the following:

5) The maximum possible power (with uncertainties) will increase the heatup rate during the transient and result in a higher peak pressure.

6) A minimum initial primary pressure value is utilized to allow for a larger heatup and power increase prior to reaching the reactor trip setpoint.

7) In order to maximize the length of time prior to opening the secondary side safety valves, a minimum initial reactor temperature and a maximum steam generator tube plugging level are chosen. Opening of the secondary side safety valves will decrease secondary side pressure causing an increase in primary to secondary heat transfer which then produces a reduction in the primary system temperature. Delay in opening the secondary safety valves will, therefore, act to maximize the primary pressurization. In addition, the use of the maximum steam generator tube plugging value acts to reduce initial primary to secondary heat transfer and will cause a slightly faster primary pressurization.

8) The minimum flow allowed by Technical Specifications is chosen to reduce primary to secondary heat transfer during the transient.

9) A positive Moderator Temperature Coefficient (MTC) consistent with the limits in Technical Specifications is chosen in order to produce a power increase in conjunction with the primary coolant temperature increase. The least negative Doppler Coefficient is also chosen to allow the maximum possible power increase.

10) A bottom peaked axial shape corresponding to the limits of the full power operating band is used as the initial condition for the scram reactivity data. The use of a bottom peaked axial shape increases the power increase during the transient slightly since the negative reactivity due to the insertion of the control rods will be delayed compared to the reactivity insertion associated with a top peaked or cosine axial shape.

TABLE 4.2.1.3-1

KEY PARAMETERS ASSUMED FOR THE LOSS OF CONDENSER VACUUM EVENT

ST. LUCIE UNIT 1

Parameter	Units	Value
Total RCS Power (Core Thermal Power x 1.02)	MWth	2754
Initial Reactor Coolant System Pressure	psia	2203
Initial Core Coolant Inlet Temperature	°F	547
Tube Plugging	%	15
Initial RCS Vessel Flow Rate	gpm	370,000
Moderator Temperature Coefficient	pcm/°F	2.0
Doppler Coefficient	pcm/°F	-0.8
CEA Worth at Trip	%Δρ	-5.3

[illegible]

4.2.1.3.2 RESULTS

The Loss of Condenser Vacuum event initiated from the conditions given in Table 4.2.1.3-1, results in a high pressurizer pressure trip condition at 6.50 seconds. At 9.60 seconds, the primary pressure reaches its maximum value of 2603 psia. The maximum pressure point is calculated by RETRAN at the pump discharge. The maximum pressure value is shown to be well below the 2750 psia acceptance criteria.

The peak secondary side pressure occurs at 9.70 seconds and reaches a value of 1054 psia. This value is well below the 1100 psia acceptance criteria for the secondary side system.

Table 4.2.1.3-2 presents the sequence of events for this transient. Figures 4.2.1.3-1 to 4.2.1.3-6 show the power, heat flux, RCS pressure, RCS coolant temperatures, steam generator pressure, and pressurizer level response to this bounding Loss of Condenser Vacuum event.

Evaluation of the DNBR response to a Loss of Condenser Vacuum Event shows qualitatively that DNBR will vary only slightly during this transient. Previous discussion for the Loss of Load transient, Section 4.2.1.1 of DNBR response is also applicable to this event. The Loss of Flow event as described in Section 15.2.5 of the FSAR will remain the bounding event with regard to DNBR response.

TABLE 4.2.1.3-2

SEQUENCE OF EVENTS

LOSS OF CONDENSER VACUUM

ST. LUCIE UNIT 1

EVENT	TIME (seconds)	SETPOINT OR VALUE
Loss of Condenser Vacuum	0.0	---
High Pressurizer Pressure Trip	6.5	2422 psia
Trip Breakers Open	7.65	---
Pressurizer Safeties Open	7.8	2525 psia
CEAs Begin to Drop Into Core	8.15	---
Steam Generator Safeties Open	8.2	1010 psia
Peak RCS Pressure	9.6	2603 psia
Maximum Steam Generator Pressure	9.7	1054 psia
Pressurizer Safeties Close	12.7	2424 psia

FIGURE 4.2.1.3-1

St. Lucie Unit 1 Loss of Condenser Vacuum

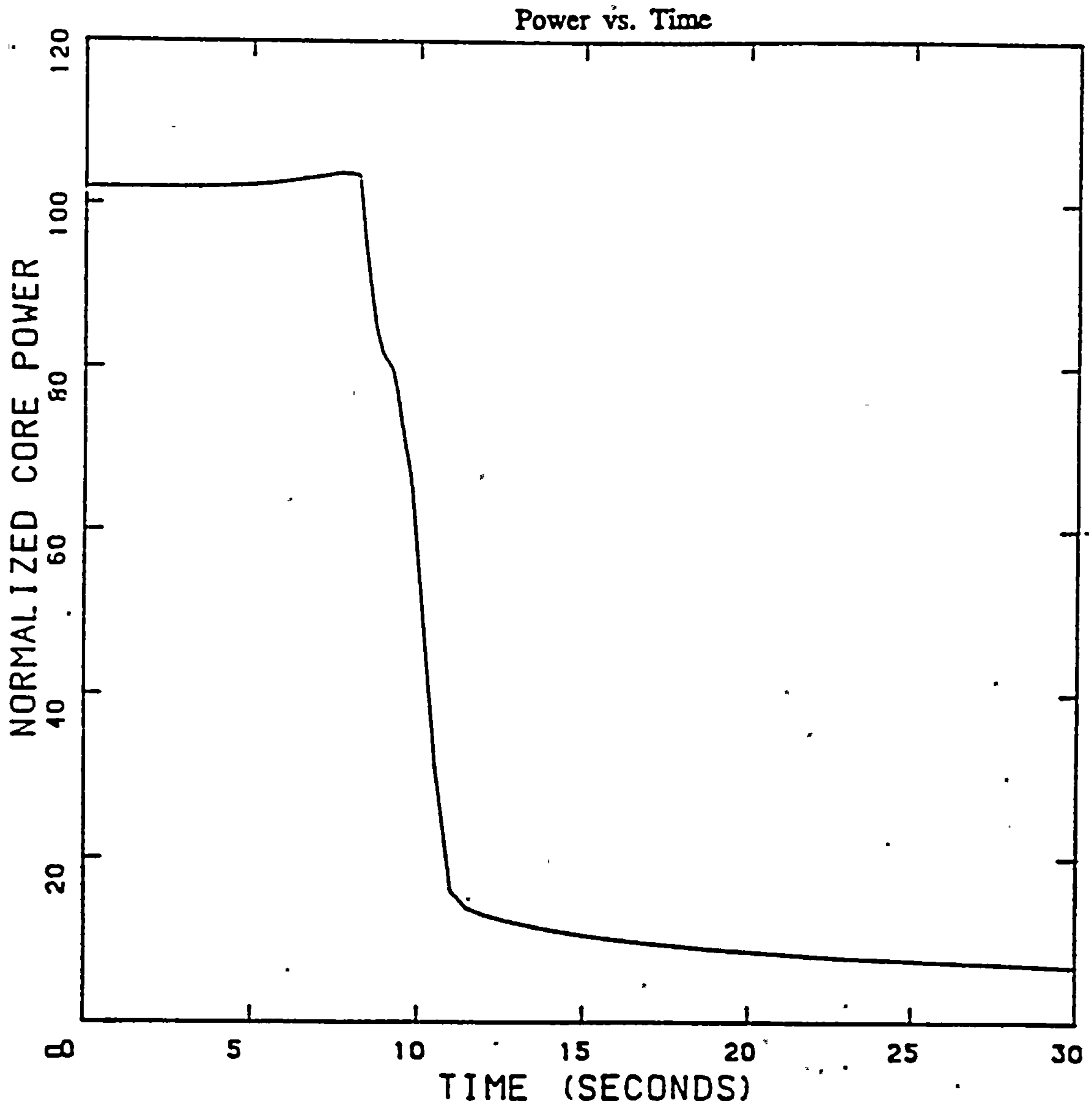


FIGURE 4.2.1.3-2

St. Lucie Unit 1 Loss of Condenser Vacuum

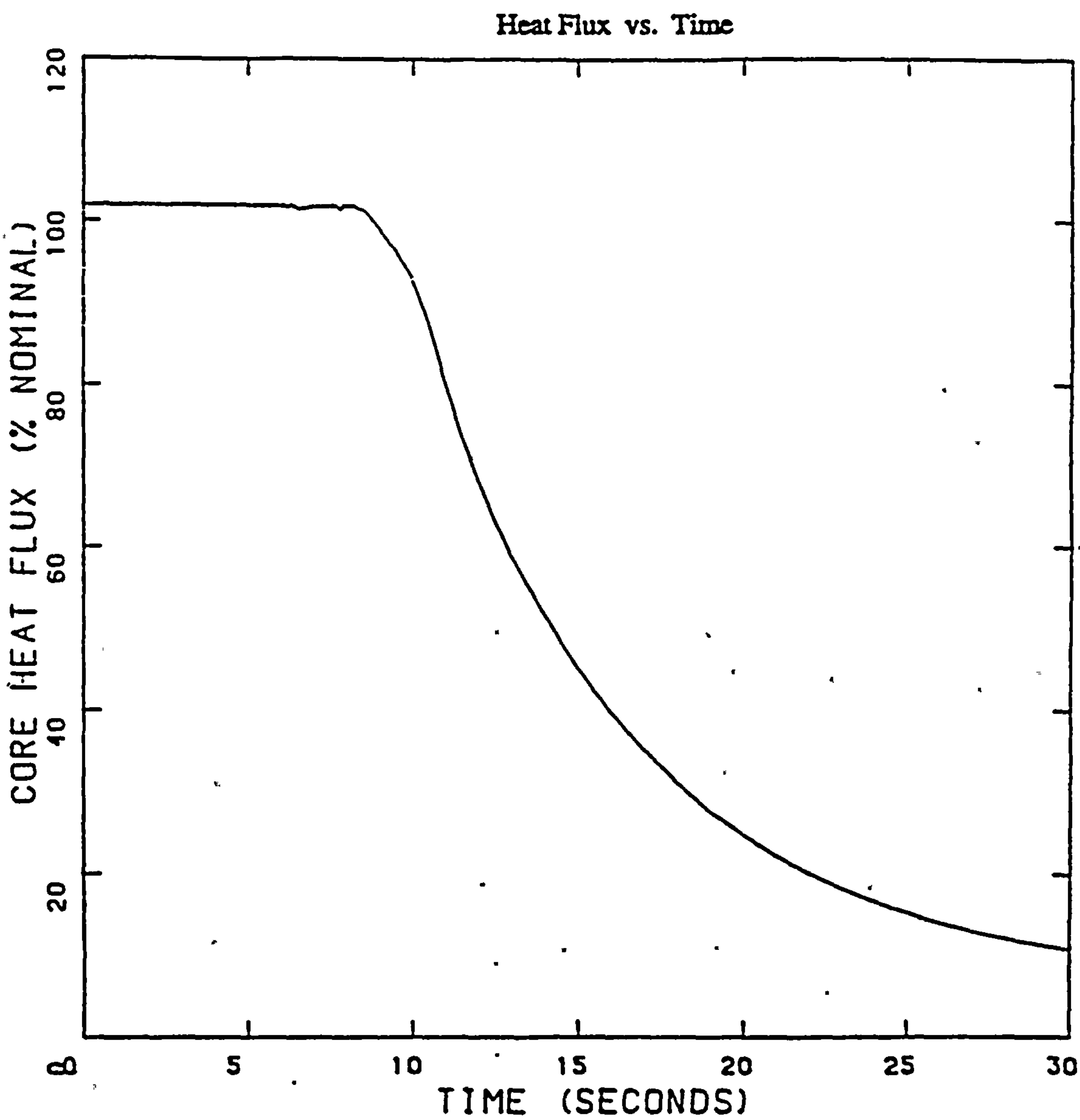


FIGURE 4.2.1.3-3

St. Lucie Unit 1 Loss of Condenser Vacuum

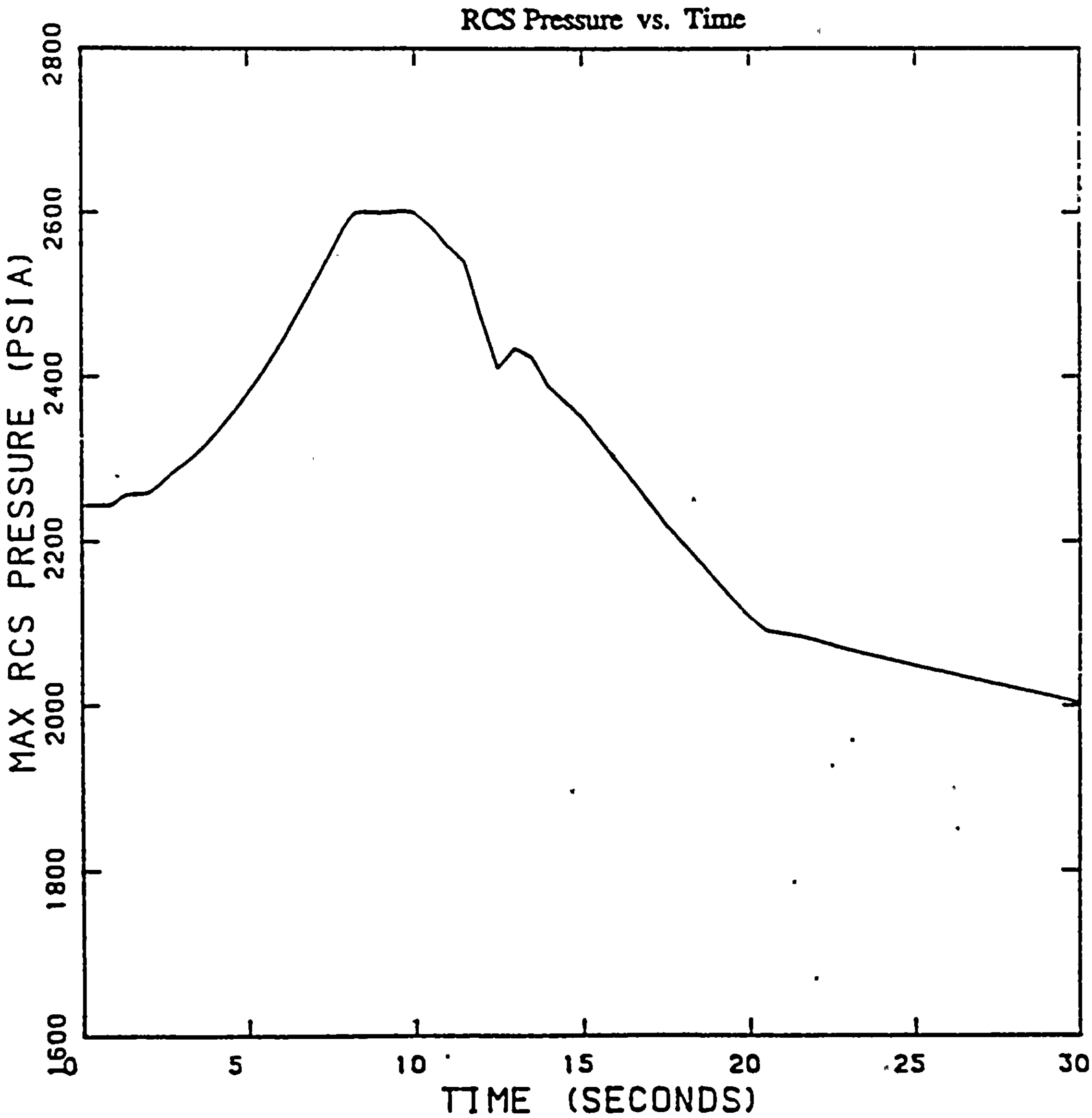


FIGURE 4.2.1.3-4

St. Lucie Unit 1 Loss of Condenser Vacuum

RCS Coolant Temperatures vs. Time

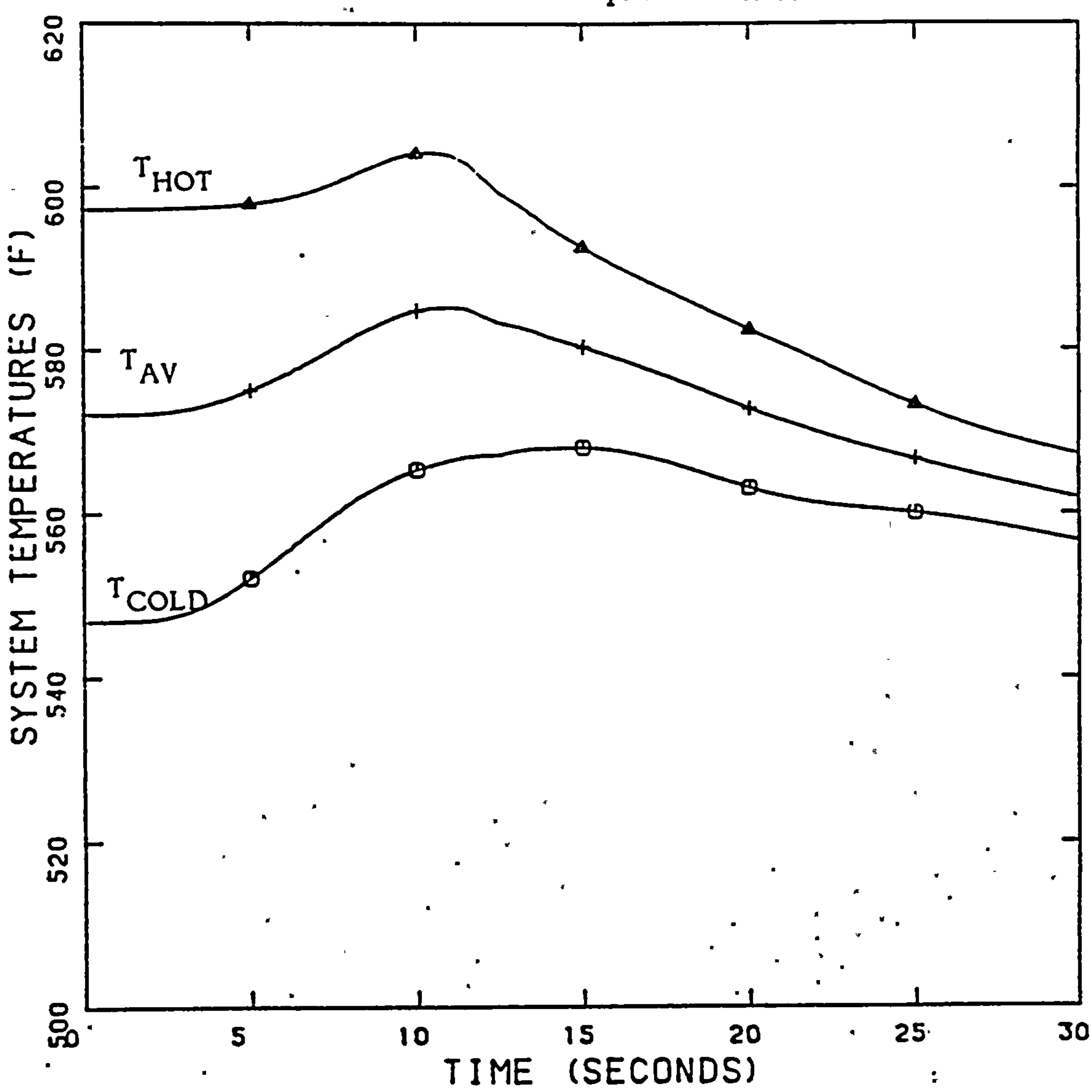


FIGURE 4.2.1.3-5

St. Lucie Unit 1 Loss of Condenser Vacuum

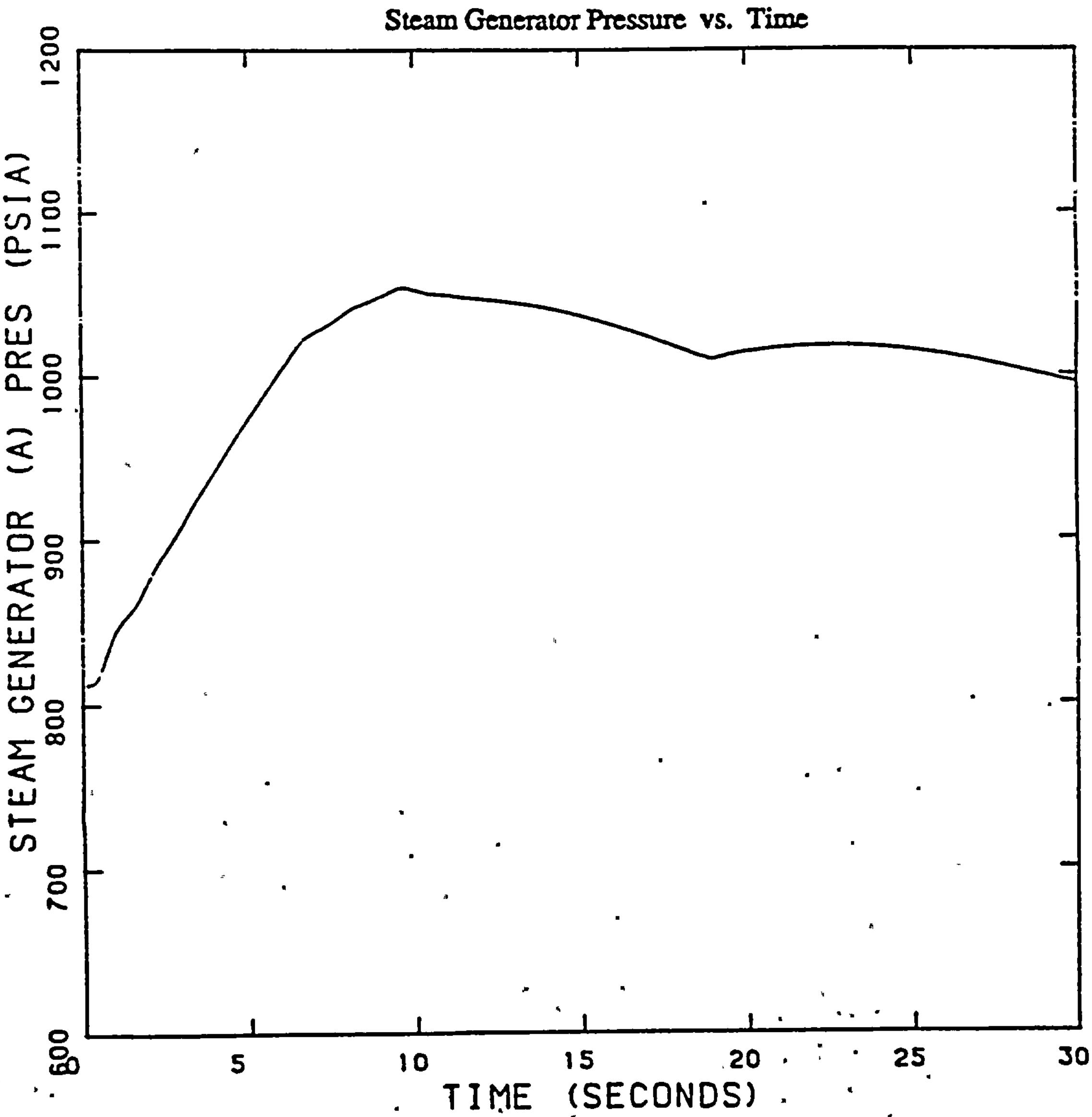
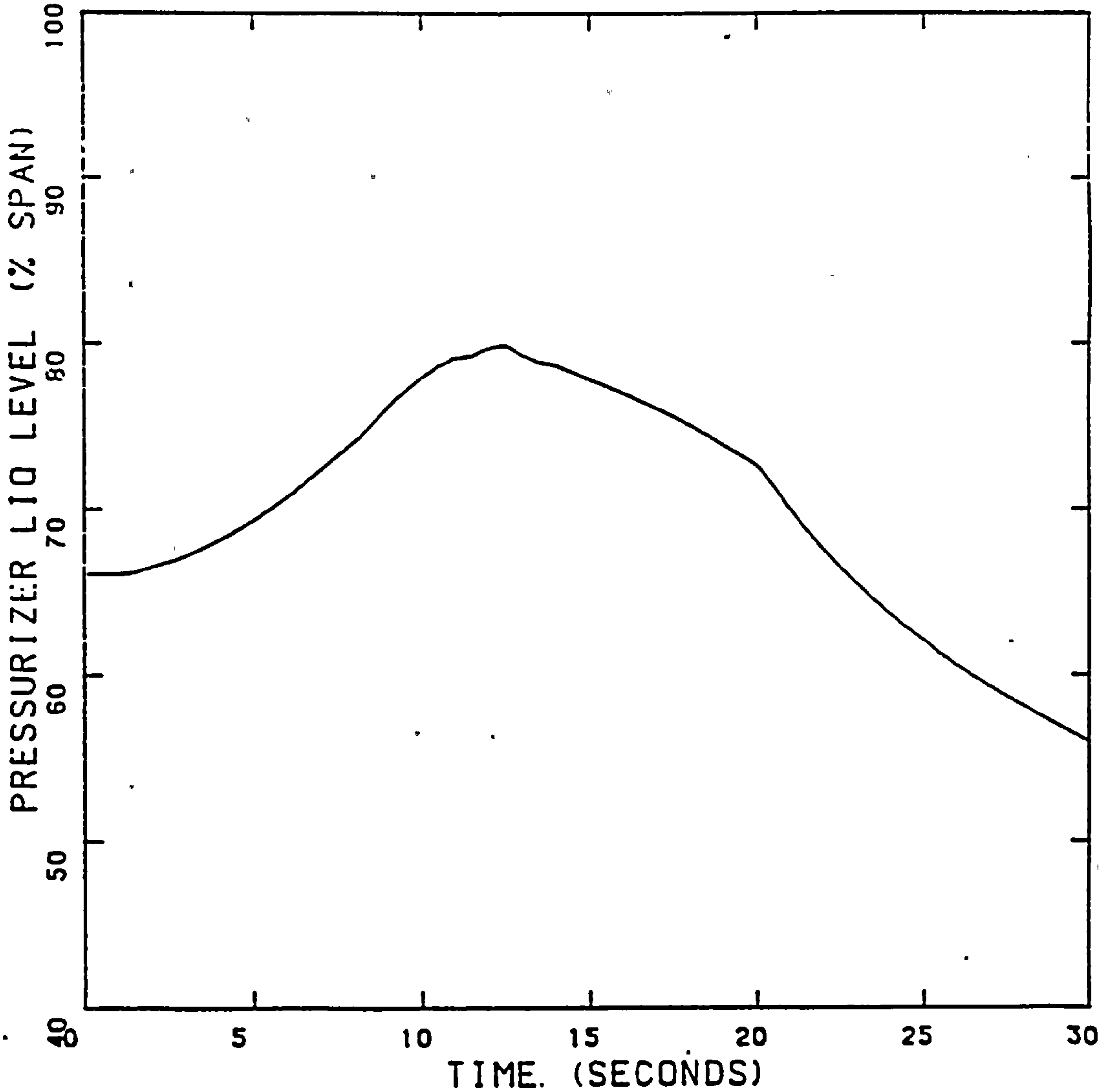


FIGURE 4.2.1.3-6

St. Lucie Unit 1 Loss of Condenser Vacuum

Pressurizer Level vs. Time



[illegible]

4.2.1.4 Main Steam Isolation Valve Closure

The Main Steam Isolation Valve (MSIV) Closure event is evaluated by assuming that one or both of the MSIV's close and stop steam flow from either one or both SG's. The MSIV closure time is much greater than that of the turbine stop valves and therefore the resultant heatup and pressurization of the secondary and primary systems would be less than that produced by the Loss of Condenser Vacuum event. The closure of only a single MSIV would produce a asymmetric pressure difference between steam lines. This asymmetric event would be quickly terminated by the asymmetric steam generator pressure trip.

DNBR response to this event is less limiting than that experienced in the Loss of Flow event described in Section 15.2.5 of the FSAR.

4.2.1.5 Steam Pressure Regulator Failure

A Steam Pressure Regulator Failure for St. Lucie Unit 1 would be similar to a failure of the turbine control system. The turbine control system is a digital electronic hydraulic (DEH) system which controls the turbine automatically using a process control computer, servo-mechanism and hydraulic valve actuators. The computer represents the digital portion of the system, the servo hardware represents the electronic portion of the system and the valve actuators represent the hydraulic part of the system. During automatic operation the DEH control system sends output signals to the servo system which in turn positions the hydraulic valve actuators and controls turbine speed or load.

A failure in this system which would close the turbine valves would be bounded by the Loss of Condenser Vacuum event due to the much more rapid action of the turbine stop valves. This rapid action of the stop valves increases the primary heatup and maximizes primary pressurization.

DNBR response to this event is less limiting than that experienced in the Loss of Flow event described in Section 15.2.5 of the FSAR.

4.2.1.6 Loss of Non-emergency AC Power to the Station Auxiliaries

The Loss of Non-emergency AC Power to the Station Auxiliaries (LOAC) is defined as a complete loss of offsite electrical power and concurrent turbine trip. As a result of this event, electrical power would be unavailable for the reactor coolant pumps and main feedwater pumps. The plant would therefore experience a simultaneous loss of load, a loss of feedwater and a loss of forced reactor coolant flow.

The LOAC is followed by automatic startup of the emergency diesel generators. The power output of each is sufficient to supply electrical power to all engineered safety features and to ensure the capability of establishing and maintaining the plant in a safe shutdown condition. Since power is not available for the control element assembly drive mechanisms (CEDM's), the de-energization of the CEDM magnetic holding coils would release the CEA's and initiate reactor shutdown.

The early part of this transient (0-10 seconds) would be bounded by the Loss of Forced Reactor Coolant Flow (LOF) event since any change in primary to secondary heat transfer resulting from the loss of load would not be experienced by the core during this time period of the transient due to loop transit time effects. Even if reactor scram on LOAC is not credited, reactor trip on low reactor coolant flow would quickly occur and the transient DNBR response would be the same as the LOF.

For the remainder of the transient, the ability to maintain a secondary side heat sink is assured by the action of the auxiliary feedwater system. Evaluation of the auxiliary feedwater system ability to maintain the secondary side heat sink during transient conditions is presented in Chapter 10 of the FSAR and bounds the results of the LOAC.

Radiological consequences of this event are bounded by the results of the Inadvertent Opening of a Steam Generator Safety Valve presented in Section 15.2.11.3.2 of the FSAR. The Opening of the Safety Valve is bounding since that postulated event will result in the complete blowdown of one steam generator and partial blowdown of the other. Since the mass of steam released is much larger, the radiological releases are also much larger than those that result from the LOAC. It should be noted that the calculation of radiological doses from a new analysis, if needed, would be performed by using a simple ratio of the steam masses released (New Analysis/FSAR Analysis) to multiply the FSAR calculated doses, since the basic assumptions related to initial concentrations of radioisotopes, steam generator partition factor, atmospheric dispersion coefficient, breathing rate and dose conversion factor would all remain constant.

4.2.1.7 Loss of Normal Feedwater Flow

The Loss of Normal Feedwater Flow event is defined as a reduction in feedwater flow to the steam generators when operating at power, without a corresponding reduction in steam flow from the steam generators. The result of this mismatch is a reduction in the water inventory in the steam generators, which will cause a reduction in primary to secondary heat transfer and subsequent pressurization of the primary system.

A complete loss of feedwater flow to the steam generators can occur when;

1) a malfunction in the feedwater regulating system for both steam generators causes all feedwater regulating valves to close,or

2) a loss of all feedwater or condensate pumps occurs,or

3) in manual feedwater control, the operator either closes the feedwater regulating valves or closes the feedwater stop valves,or

4) a main feedwater header ruptures.

Action of the low steam generator level trip will initiate reactor scram and end the primary pressurization. Auxiliary feedwater will actuate to provide sufficient feedwater flow to remove decay heat from the RCS following the reactor trip. The auxiliary feedwater system consists of one non-condensing steam turbine driven auxiliary feed water pump and two motor driven auxiliary feedwater pumps. The steam generators are designed to withstand the thermal shock and loading imposed by a total loss of feedwater and subsequent refill transient. Evaluation of the auxiliary feedwater system performance to provide decay heat removal capability is discussed in detail in Chapter 10 of the FSAR.

Primary pressure will increase during this transient as the water level in the steam generator is reduced from nominal values to the low level setpoint. However, the pressurization rate will be significantly lower than that experienced during the Loss of Condenser Vacuum transient described in Section 4.2.1.3 of this document. The reason the pressurization rate is much lower for the Loss of Normal Feedwater Flow event is that the continuation of the steam flow until reactor trip will maintain the primary temperatures at near constant conditions. Therefore, the bounding event for this category remains the Loss of Condenser Vacuum event.

DNBR response during this transient would remain bounded by the Loss of Flow event described in Section 15.2.5 of the FSAR.

[illegible]

4.2.1.8 Feedwater Line Break

This event is a cooldown event in the licensing basis for St. Lucie Unit 1. It should be noted that while this event can be made to be a heatup type event by artificially placing the feedwater entry into the steam generator at the bottom of the steam generator (as done for St. Lucie Unit 2), this does not result in a realistic representation of the physical effects of this type of transient. Analysis in this artificial way results in potential misunderstanding of the transient response by operation personnel and may also result in plant changes being examined incorrectly when they are evaluated relative to the results in the FSAR.

When the actual physical configuration of the steam generators and feedwater system is examined, it can be seen clearly that steam, not liquid would be released from the steam generators during the major portion of a feedwater line break event. As such, the feedwater pipe break event is bounded by the steamline break event since the area for flow in a broken feedwater pipe is less than that of a severed steamline. The smaller area for flow results in lower steam relief rate which produces a more benign event.

4.2.2 ST. LUCIE UNIT 2

4.2.2.1 Loss of External Load

A loss of external load event is caused by abnormal events in the electrical distribution network. The turbine is protected from a complete loss of external load by two overspeed protection systems which act to trip the turbine if a complete loss of load occurs. The Overspeed Protection Controller and the Mechanical Overspeed Protection system are both designed to trip the turbine at approximately 111 percent overspeed condition. More details of these two equipment protection systems are found in Section 10.2.2.2 of the St. Lucie Unit 2 FSAR. In addition to the overspeed trip mechanism, the turbine is provided protection also for the following:

1) Low Condenser Vacuum. This tripping device is designed to trip the turbine in case of a serious rise in exhaust pressure. The turbine will trip when the vacuum decreases to 19-22 inches of mercury.

2) Low Turbine Bearing Oil Pressure. The bearing oil pressure setting at normal speed is approximately 14-18 psig. However, should this pressure reach 6 psig the low bearing oil pressure protective device will trip the turbine.

3) Thrust Bearing Trip. The thrust bearing trip device is designed to shut down the turbine when the thrust bearing trip control pressure rises to above 75-80 psig.

Any turbine trip causes the hydraulic trip fluid header pressure to decrease and close steam to the turbine. Upon occurrence of the turbine trip above 15% of full power, a signal is supplied to the reactor protective system to trip the reactor. Subsequent to the turbine trip, the main feedwater regulating valves close and feedwater is supplied to the steam generator through the feedwater bypass valve by the main feedwater pumps.

A fast pressurization transient will result from the closure of the turbine stop valves due to their fast response time, approximately 0.25 seconds. The primary side pressurization results from the reduction in primary to secondary heat transfer after event initiation. The increased secondary temperature and pressure which occurs with the stoppage of steam flow from the secondary side is the driving force in the heat transfer degradation. However, the reactor trip on turbine trip would quickly reverse the pressurization event and bring the reactor to a safe shutdown condition before a reactor trip on high pressure could occur. In addition, action of the SDBS will mitigate the pressurization effects of a Loss of External Load event.

A bounding Loss of Load event is postulated if credit is not taken for the reactor trip on turbine trip or action of the SDBS, this transient is bounded by the Loss of Condenser Vacuum event discussed in Section 4.2.2.3 because of the assumption in the Loss of Condenser Vacuum of loss of main feedwater. The loss of main feedwater increases the heatup of the primary and therefore increases the pressurization rate of the transient.

Evaluation of the DNBR response to a Loss of External Load Event shows qualitatively that DNBR will vary only slightly during this transient as discussed in Section 4.2.1.1 above. The Loss of Flow event as described in Section 15.3.2.2.6 of the FSAR will remain limiting relative to DNBR results.

4.2.2.2 Turbine Trip

A turbine trip is an event in which the steam flow from the steam generators is stopped through closure of the turbine control and/or stop valves. The turbine is equipped with an automatic stop and emergency trip system as discussed in Section 4.2.2.1 above. Turbine protection devices which could cause a turbine trip are described in depth in section 10.2.2 of the FSAR. Upon occurrence of a turbine trip above 15% of full power, a signal is supplied to the reactor protective system to trip the reactor. Subsequent to the turbine trip, the main feedwater regulating valves close and feedwater is supplied to the steam generator through the feedwater bypass valve by the main feedwater pumps.

A fast pressurization transient will result from the closure of the turbine stop valves due to their fast response time, approximately 0.25 seconds. The primary side pressurization results from the reduced primary to secondary heat transfer which occurs with the stoppage of steam flow from the secondary side. However, the reactor trip on turbine trip would quickly reverse the pressurization event and bring the reactor to a safe shutdown condition before a reactor trip on high pressure could occur. In addition, action of the SDBS will mitigate the pressurization effects of a Turbine Trip.

A bounding Turbine Trip event is postulated if credit is not taken for the reactor trip on turbine trip or action of the SDBS, this transient is bounded by the Loss of Condenser Vacuum event discussed in Section 4.2.2.3 because of the assumption, in that event, of an instantaneous loss of main feedwater. The loss of main feedwater increases the heatup of the primary and therefore increases the pressurization rate of the transient.

Evaluation of the DNBR response to a Turbine Trip Event shows qualitatively that DNBR will vary only slightly during this transient as discussed in Section 4.2.1.1 above. The Loss of Flow event as described in Section 15.3.2.2.6 of the FSAR will remain the limiting event with regard to DNBR results.

4.2.2.3 Loss of Condenser Vacuum

4.2.2.3.1 Limiting Event Development

A Loss of Condenser Vacuum event may occur due to the failure of the Circulation Water System, the failure of the Main Condenser Evacuation System to remove non-condensible gases, or the in-leakage of an excessive amount of air through a turbine gland. Low condenser pressure generates a turbine trip signal. The turbine trip signal causes the turbine stop and control valve to close. Upon occurrence of a turbine trip above 15% of full power, a signal is supplied to the reactor protective system to trip the reactor. The Loss of Condenser Vacuum event will disable the SDBS and will result in a gradual rampdown of main feedwater.

For purposes of developing an event that bounds the potential for primary pressurization, several conservative assumptions will be made relative to the Loss of Condenser Vacuum event. The following assumptions will be utilized in this analysis:

1) For this bounding event, coincident with the Loss of Condenser Vacuum, the turbine stop valves are assumed to instantly close on high condenser back pressure. This assumption assures the fastest possible pressurization rate.

2) A reactor trip on turbine trip signal will not be credited. Not allowing the reactor trip on turbine trip signal to operate will allow the high pressurizer pressure trip to activate to initiate reactor scram. Use of the high pressurizer pressure trip rather than any other system trip (such as steam generator low level) will insure that the highest possible primary pressure will be reached during the transient.

3) Main feedwater flow rate will be instantaneously set to zero to maximize the decrease in primary to secondary heat transfer.

4) Action of the Power Operated Relief Valves (PORV)and the pressurizer sprays will not be credited since this would mitigate the pressurization of the primary.

5) A Loss of AC will be initiated such that the trips on High Pressurizer Pressure and Low RCS Flow will occur approximately coincident. The loss of forced coolant flow will decrease the primary to secondary heat transfer and produce a higher peak primary pressure. Coincident trips will produce the maximum pressurization rate since the duration of the loss of flow is maximized while maintaining the requirement for tripping on high pressure.

The choice for other key initial conditions are listed in Table 4.2.2.3-1. For evaluation of the peak primary pressure, initial conditions are chosen such that the time to reactor trip on high pressure is maximized and that a high rate of change of pressure is achieved. The basis of the key initial conditions is discussed briefly in the following:

6) The maximum possible power (with uncertainties) will increase the heatup rate during the transient and result in a higher peak pressure.

7) A minimum initial primary pressure value is utilized to allow for a larger heatup and power increase prior to reaching the reactor trip setpoint.

8) In order to maximize the length of time prior to opening the secondary side safety valves, a minimum initial reactor temperature value and a maximum steam generator tube plugging level are chosen. A minimum primary temperature initiates the transient with the minimum secondary side pressure which maximizes the pressure increase required before the secondary safeties open. In addition, the use of the maximum steam generator tube plugging value acts to reduce initial primary to secondary heat transfer and will cause a slightly faster primary pressurization.

9) The minimum flow value consistent with Technical Specifications is chosen to minimize primary to secondary heat transfer during the event.

10) A positive Moderator Temperature Coefficient (MTC) consistent with the limits in Technical Specifications is chosen in order to produce a power increase in conjunction with the primary coolant temperature increase. The least negative Doppler Coefficient is also chosen to allow the maximum possible power increase.

11) A bottom peaked axial shape corresponding to the limits of the full power operating band is used as the initial condition for the scram reactivity data. The use of a bottom peaked axial shape maximizes the power increase, during the transient, slightly because the effects of the negative control reactivity are delayed with respect to those from top peaked or cosine axial shapes.

TABLE 4.2.2.3-1

KEY PARAMETERS ASSUMED FOR THE LOSS OF CONDENSER VACUUM EVENT

ST. LUCIE UNIT 2

Parameter	Units	Value
Total RCS Power (Core Thermal Power x 1.02)	MWth	2754
Initial Reactor Coolant System Pressure	psia	2167
Initial Core Coolant Inlet Temperature	^{o}F	532
Steam Generator Tube Plugging	%	15
Initial RCS Vessel Flow Rate	gpm	363,000
Moderator Temperature Coefficient	pcm/^{o}F	3.0
Doppler Coefficient	pcm/^{o}F	-0.8
CEA Worth at Trip	%$\Delta\rho$	-5.5

4.2.2.3.2 Results

The Loss of Condenser Vacuum event initiated from the conditions given in Table 4.2.2.3-1, results in a high pressurizer pressure trip condition at 7.14 seconds. A loss of AC is chosen such that the reactor trip on low flow occurs coincident with the reactor trip on high pressure. At 11.0 seconds, the maximum pressure point is calculated by RETRAN, as 2675 psia, located in the lower plenum. The maximum pressure value is shown to be well below the 2750 psia acceptance criteria.

The peak secondary side pressure occurs at 16.0 seconds and reaches a value of 1041 psia. This value is well below the 1100 psia acceptance criteria for the secondary side system.

Table 4.2.2.3-2 presents the sequence of events for this transient. Figures 4.2.2.3-1 to 4.2.2.3-6 present the power, heat flux, RCS pressure, RCS coolant temperatures, steam generator pressure, and pressurizer level response to this bounding Loss of Condenser Vacuum event.

A Loss of Condenser Vacuum event was also evaluated using best-estimate parameters. In this case, all initial conditions were nominal in an attempt to represent the actual plant response if a real Loss of Condenser Vacuum event were to occur. In order to obtain some pressurization in the primary during the event, even this case did not take credit for action of the reactor protective system reactor trip on turbine trip. For this case, the peak pressure reached a maximum of 2441 psia. The difference between the peak pressure calculated using our licensing methods and the best-estimate results provides a measure of the conservatism applied in the design inputs.

Evaluation of the DNBR response to a Loss of Condenser Vacuum Event shows qualitatively that DNBR will vary only slightly during this transient. Previous discussion for the Loss of Load transient, Section 4.2.1.1 of DNBR response is also applicable to this event. The Loss of Flow event as described in Section 15.3.2.2.6 of the FSAR will remain the bounding event with regard to DNBR response.

TABLE 4.2.2.3-2

SEQUENCE OF EVENTS

LOSS OF CONDENSER VACUUM

ST. LUCIE UNIT 2

EVENT	TIME (seconds)	SETPOINT OR VALUE
Loss of Condenser Vacuum	0.0	---
Loss of Offsite Power	6.7	---
High Pressurizer Pressure Trip	7.14	2428 psia
Low Flow Trip	7.57	337,590 gpm
Trip Breakers Open	8.22	---
Pressurizer Safeties Open	8.26	2525 psia
CEAs Begin to Drop Into Core	8.96	---
Peak RCS Pressure	11.0	2675 psia
Steam Generator Safeties Open	11.05	1010 psia
Pressurizer Safeties Close	14.5	2424 psia
Maximum Steam Generator Pressure	16.0	1041 psia

FIGURE 4.2.2.3-1

St. Lucie Unit 2 Loss of Condenser Vacuum

Power vs. Time

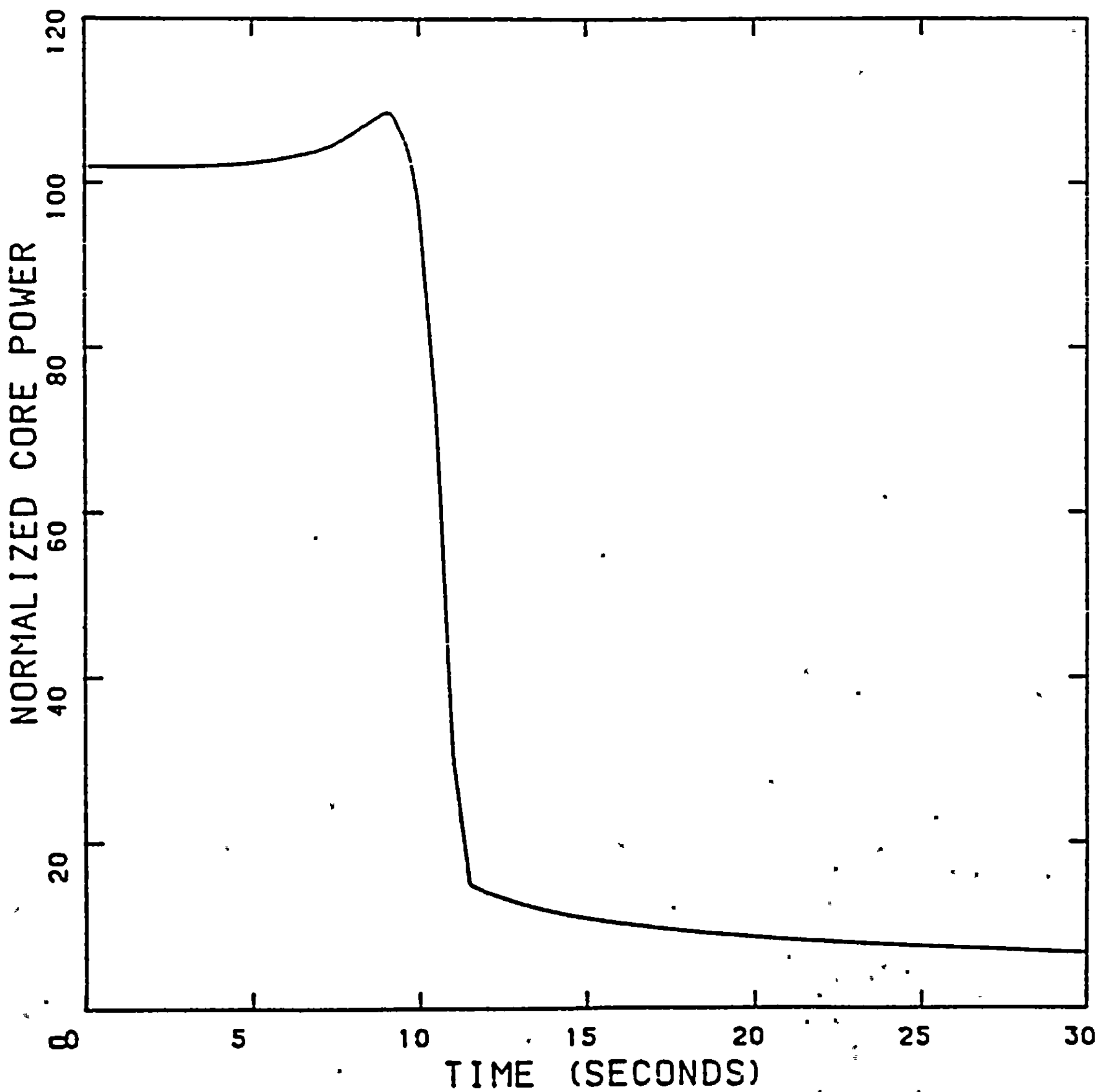


FIGURE 4.2.2.3-2

St. Lucie Unit 2 Loss of Condenser Vacuum

Heat Flux vs. Time

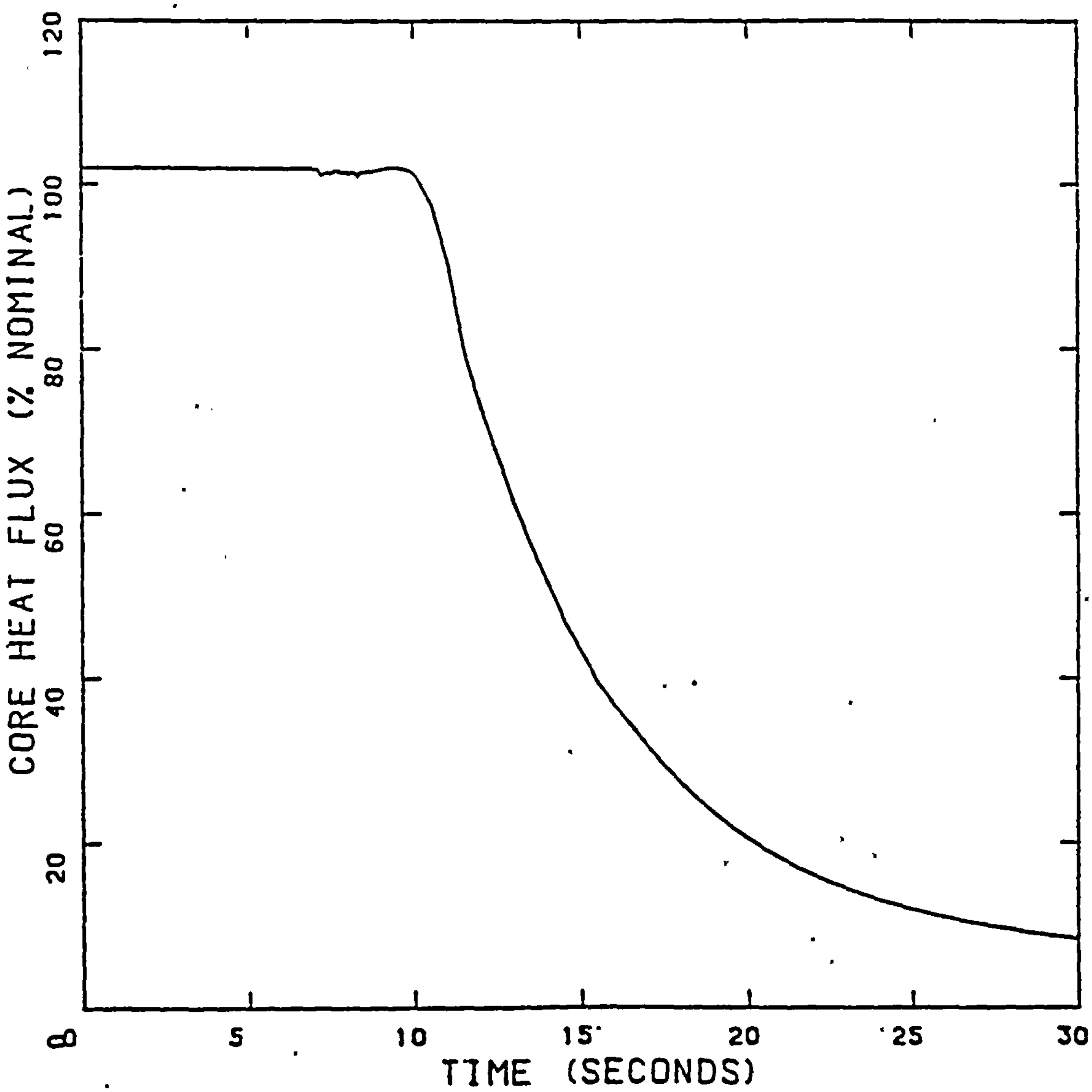


FIGURE 4.2.2.3-3

St. Lucie Unit 2 Loss of Condenser Vacuum

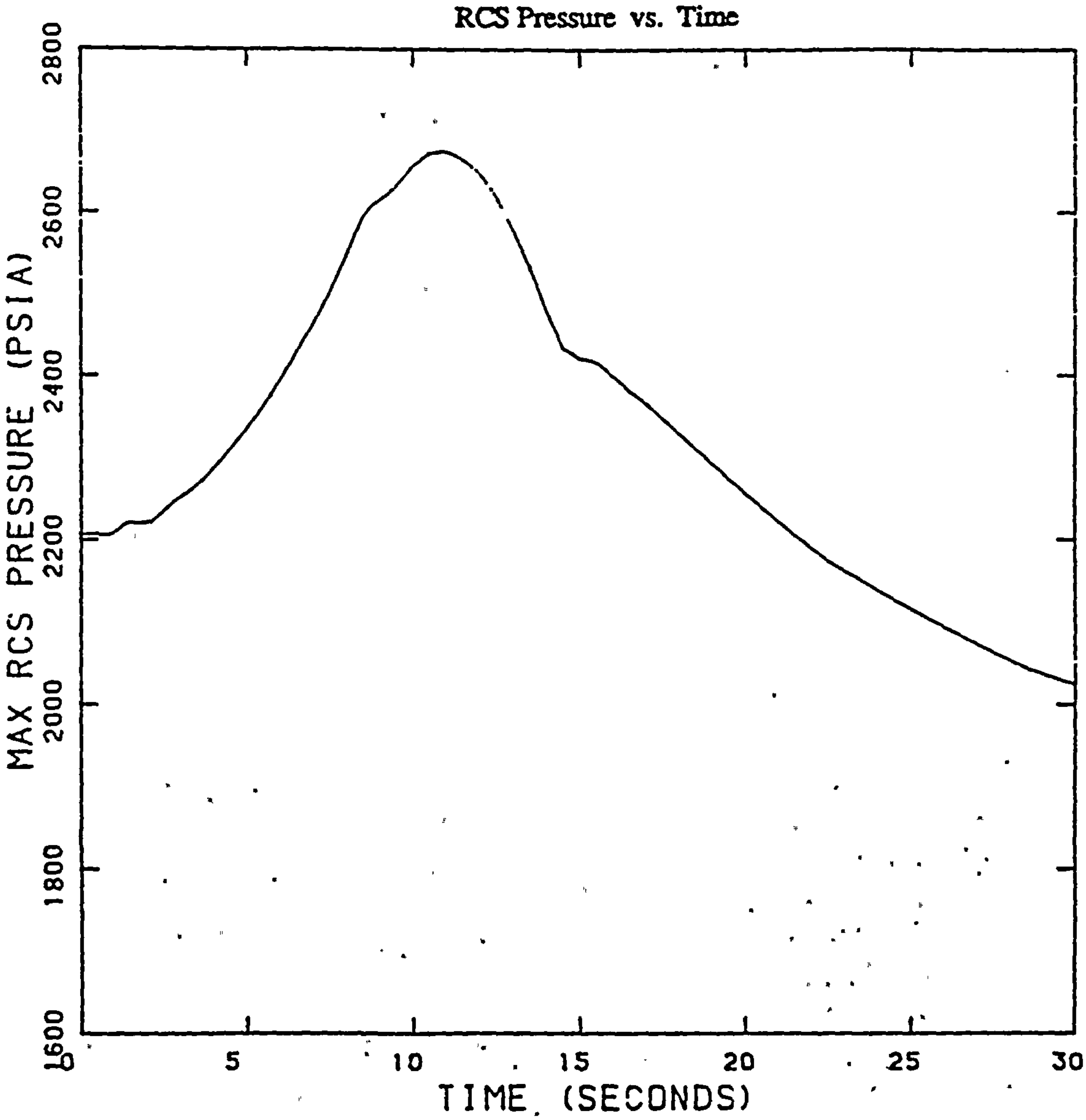


FIGURE 4.2.2.3-4

St. Lucie Unit 2 Loss of Condenser Vacuum

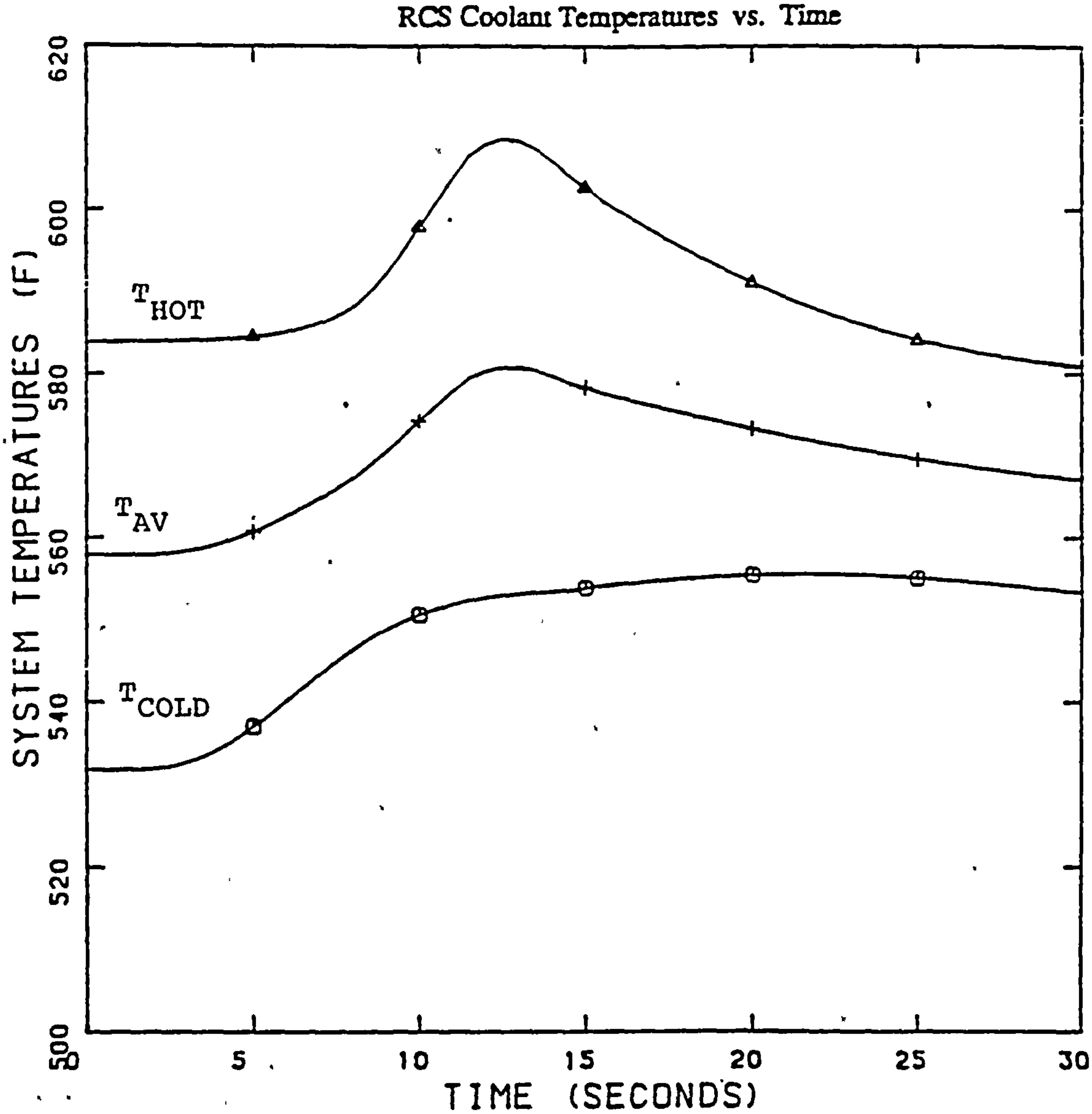


FIGURE 4.2.2.3-5

St. Lucie Unit 2 Loss of Condenser Vacuum

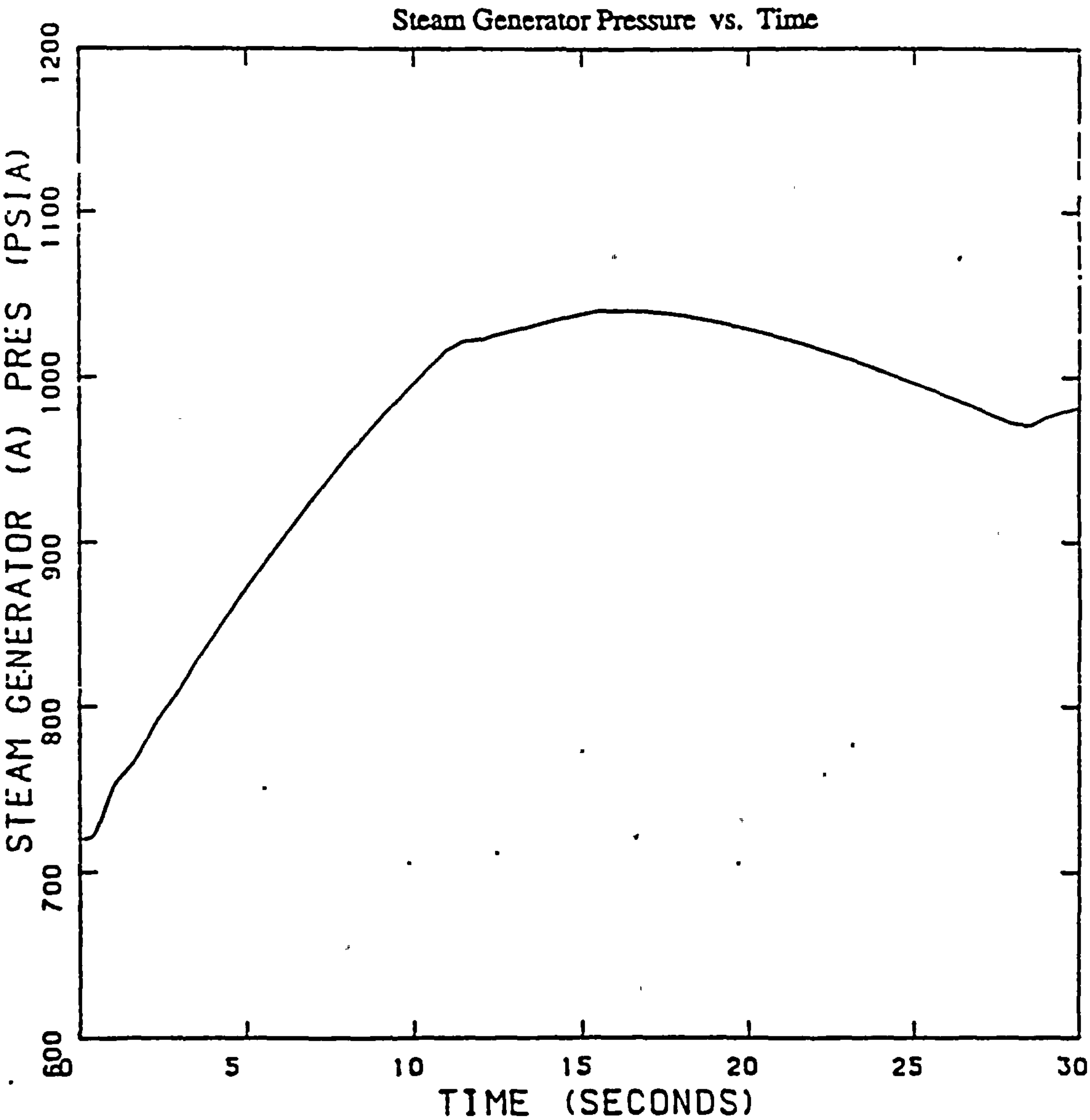
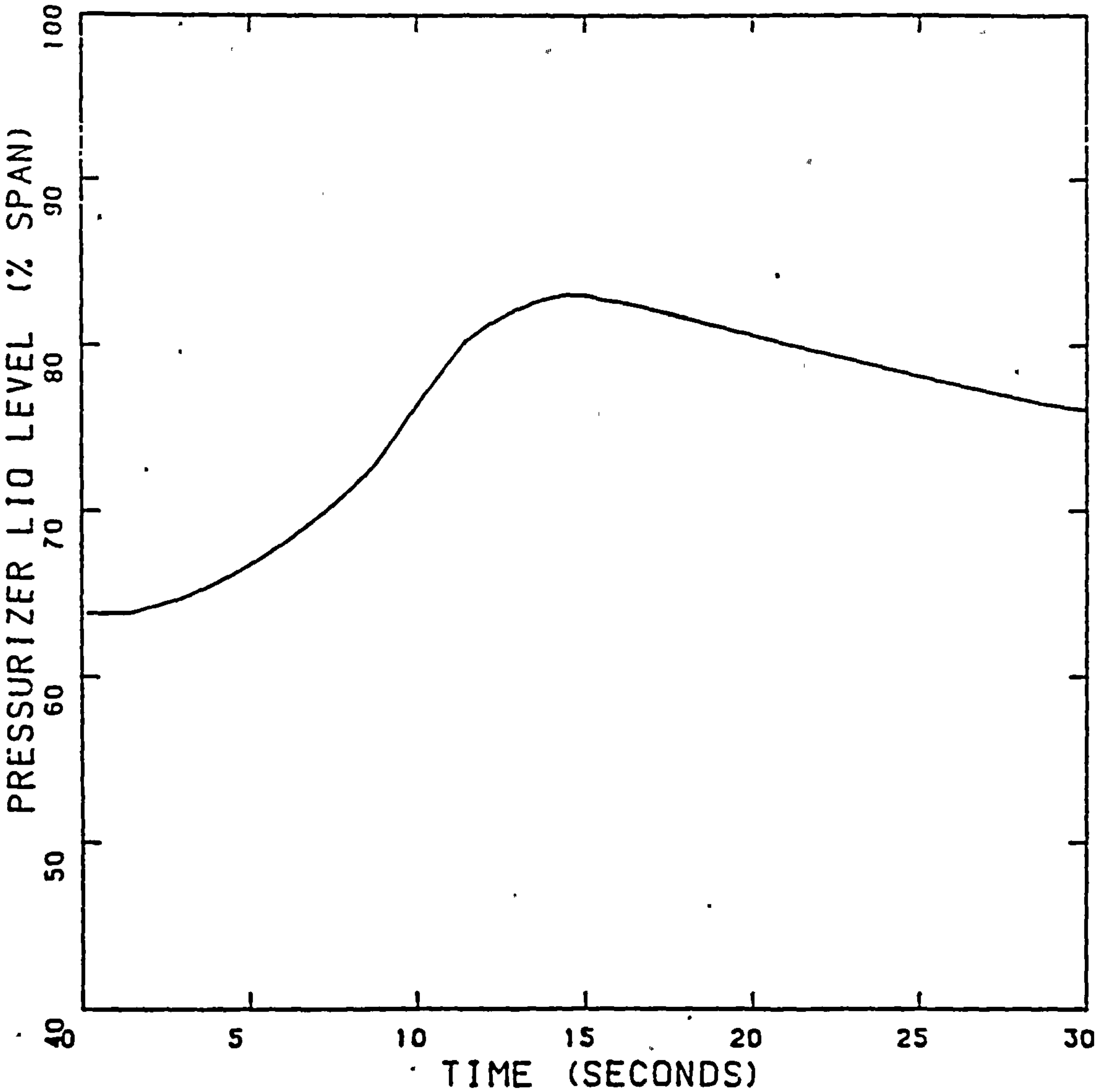


FIGURE 4.2.2.3-6

St. Lucie Unit 2 Loss of Condenser Vacuum

Pressurizer Level vs. Time



4.2.2.4 Main Steam Isolation Valve Closure

The MSIV Closure event is evaluated by assuming that one or both of the MSIV's close and stop steam flow from one or both SG's. The MSIV closure time is much greater than that of the turbine stop valves and therefore the resultant heatup and pressurization of the secondary and primary systems would be less than that produced by the Loss of Condenser Vacuum event. The closure of only a single MSIV would produce a asymmetric pressure difference between steam lines. This asymmetric event would be quickly terminated by the asymmetric steam generator pressure trip.

DNBR response during this event is bounded by the Loss of Flow event described in Section 15.3.2.2.6 of the FSAR.

4.2.2.5 Steam Pressure Regulator Failure

A Steam Pressure Regulator Failure for St. Lucie Unit 2 would be similar to a failure of the turbine control system. The turbine control system is a DEH system which controls the turbine automatically using a process control computer, servo-mechanism and hydraulic valve actuators. The computer represents the digital portion of the system, the servo hardware represents the electronic portion of the system and the valve actuators represent the hydraulic part of the system. The turbine control system is designed to:

a) control automatically the turbine-generator output power during all phases of normal operation,

b) trip the turbine to guard the equipment from exposure to hazardous conditions,

c) provide an automatic reactor trip signal when the turbine is tripped,

d) provide a turbine runback signal upon privation of one main feedwater pump or upon loss of service of two heater drain pumps.

A failure in this system which would close the turbine valves would be bounded by the Loss of Condenser Vacuum event due to the much more rapid action of the turbine stop valves. This rapid action of the stop valves increases the primary heatup and maximizes primary pressurization.

DNBR response to this event is bounded by the Loss of Flow event described in Section 15.3.2.2.6 of the FSAR.

4.2.2.6 Loss of Non-emergency AC Power to the Station Auxiliaries

The Loss of Non-emergency AC Power to the Station Auxiliaries (LOAC) is defined as a complete loss of offsite electrical power and concurrent turbine trip. As a result of this event, electrical power would be unavailable for the reactor coolant pumps and main feedwater pumps. The plant would therefore experience a simultaneous loss of load, a loss of feedwater and a loss of forced reactor coolant flow.

The LOAC is followed by automatic startup of the emergency diesel generators. The power output of each is sufficient to supply electrical power to all engineered safety features and to ensure the capability of establishing and maintaining the plant in a safe shutdown condition. Since power is not available for the control element assembly drive mechanisms (CEDM's), the de-energization of the CEDM magnetic holding coils would release the CEA's and initiate reactor shutdown.

The early part of this transient (0-10 seconds) would be bounded by the Loss of Forced Reactor Coolant Flow (LOF) event since any change in primary to secondary heat transfer resulting from the loss of load would not be experienced by the core during this time period of the transient due to loop transit time effects. Even if reactor scram on LOAC is not credited, reactor trip on low reactor coolant flow would quickly occur and the transient DNBR response would be the same as the LOF.

For the remainder of the transient, the ability to maintain a secondary side heat sink is assured by the action of the auxiliary feedwater system. Evaluation of the auxiliary feedwater system ability to maintain the secondary side heat sink during transient conditions is presented in Chapter 10 of the FSAR and bounds the results of the LOAC.

Radiological consequences of this event are bounded by the results of the Inadvertent Opening of a Steam Generator Safety Valve presented in Section 15.1.3.1.1 of the FSAR. The Opening of the Safety Valve is bounding since that postulated event will result in the complete blowdown of one steam generator and partial blowdown of the other. Since the mass of steam released is much larger, the radiological releases are also much larger than those that result from the LOAC. It should be noted that the calculation of radiological doses from a new analysis, if needed, would be performed by using a simple ratio of the steam masses released (New Analysis/FSAR Analysis) to multiply the FSAR calculated doses, since the basic assumptions related to initial concentrations of radioisotopes, steam generator partition factor, atmospheric dispersion coefficient, breathing rate and dose conversion factor would all remain constant.

4.2.2.7 Loss of Normal Feedwater Flow

The Loss of Normal Feedwater Flow event is defined as a reduction in feedwater flow to the steam generators when operating at power, without a corresponding reduction in steam flow from the steam generators. The result of this mismatch is a reduction in the water inventory in the steam generators, which will cause a reduction in primary to secondary heat transfer and subsequent pressurization of the primary system.

A complete loss of feedwater flow to the steam generators can occur when:

1) a malfunction in the feedwater regulating system for both steam generators causes all feedwater regulating valves to close,or

2) a loss of all feedwater or condensate pumps occurs,or

3) in manual feedwater control, the operator either closes the feedwater regulating valves or closes the feedwater stop valves,or

4) a main feedwater header rupture.

Action of the low steam generator level trip will initiate reactor scram and end the primary pressurization. Auxiliary feedwater will actuate to provide sufficient feedwater flow to remove decay heat from the RCS following the reactor trip. The auxiliary feedwater system consists of one non-condensing steam turbine driven auxiliary feedwater pump and two motor driven auxiliary feedwater pumps. The steam generators are designed to withstand the thermal shock and loading imposed by a total loss of feedwater and subsequent refill transient. Evaluation of the auxiliary feedwater system performance to provide decay heat removal capability is discussed in detail in Chapter 10 of the FSAR.

Primary pressure will increase during this transient as the water level in the steam generator is reduced from nominal values to the low level setpoint. However, the pressurization rate will be significantly lower than that experienced during the Loss of Condenser Vacuum transient described in Section 4.2.2.3. The reason the pressurization rate is much lower for the Loss of Normal Feedwater Flow event is due to the continuation of the steam flow until reactor trip which maintains the primary temperatures at near constant conditions prior to reactor trip. Therefore, the bounding event for this category remains the Loss of Condenser Vacuum event.

DNBR response during this transient would remain bounded by the Loss of Flow event described in Section 15.3.2.2.6 of the FSAR.

4.2.2.8 Feedwater Line Break

4.2.2.8.1 Limiting Event Development

A Feedwater Line Break (FLB) is defined as the failure of a main feedwater system pipe during plant operation. A rupture in the main feedwater system rapidly reduces the steam generator secondary inventory as first a two phase mixture and then steam exits the break. The limiting location for a feedwater line break occurs between the steam generator and the feedwater line check valves, since blowdown of the affected steam generator would continue until the steam generator loses all inventory. An adequate secondary heat sink is provided by auxiliary feedwater actuation on steam generator low level and by initiation of the auxiliary feedwater to the unaffected SG after the expiration of the AFAS time delay.

Another potential FLB event could assume that the break occurs between the main feed header and the check valve. This break would essentially be a loss of main feedwater event with reactor trip occurring on low steam generator level, therefore, all conclusions reached in Section 4.2.2.7 would be valid for this type of FLB event.

Previous analysis of the FLB events for St. Lucie Unit 2 have been performed assuming that the ruptured feedline was physically located at the bottom of the Steam Generators. This assumption results in a FLB event that produces a primary system heatup with a large primary pressurization. Even for this physically non-realistic event, previous analyses by the reactor vendor demonstrated that the Loss of Condenser Vacuum event was still bounding relative to peak primary pressure.

The analysis presented here evaluates the FLB event as it would occur at St. Lucie Unit 2, i.e., as a cooldown event when the break occurs at the limiting location between the SG and the feedwater line check valves. The purpose of this analysis is to demonstrate that the FLB meets all design criteria using conservative analysis inputs. It will also demonstrate that the limiting cooldown event for St. Lucie Unit 2 remains the Steam Line Break (SLB) presented in Section 15.1.4.3.5.3 of the FSAR. The SLB event with a break size of 6.358 ft^2 produces a much quicker cooldown since the break size is so much larger than the largest possible FLB (1.44 ft^2) at the limiting location.

The following assumptions will be utilized in this analysis:

1) The largest feedline break size (1.44 ft^2) is assumed in order to maximize the cooldown in the primary system.

2) Maximum time delays and worst case setpoints for Main Steam Isolation and Auxiliary Feedwater Isolation are chosen to account for the potential environmental factors which

could occur with a FLB within containment. Auxiliary Feedwater actuation is assumed to act with the minimum time delay in order to maximize the amount of feedwater to the affected SG.

3) Main feedwater will continue to the unaffected steam generator until a Main Feedwater Isolation signal is generated and the valve has fully closed.

4) Consistent with the SLB methodology to maximize the potential to exceed the DNBR limits following a return to power, a Loss of AC is assumed at time of reactor trip. The Loss of AC reduces the reactor coolant flow which may negatively impact the minimum DNBR value. Of course, if the reactor never returns to power following a FLB there will be no approach to the DNBR limit as long as natural circulation flow is maintained.

The choice for other key initial conditions are listed in Table 4.2.2.8-1. The basis of the key initial conditions is discussed briefly in the following:

5) Reactor power is assumed to be at 102% of rated power. Having the initial power high increases the reactivity increase associated with the cooldown.

6) Initial coolant temperature and pressure assumptions are maximized, consistent with the values used in the limiting SLB event. High initial temperature and pressure increases the reactivity feedback associated with the primary system cooldown during the feedwater line break event. The initial coolant average temperature utilized is slightly less than that used in the previous evaluation of the SLB event. Use of the maximum coolant flow to maximize primary to secondary heat transfer prior to reactor coolant pump trip results in the FLB calculation being initialized at the lower value.

7) Initial reactor coolant flow is assumed to be at the maximum value as described above.

8) The MTC and Doppler reactivity coefficients are assumed to be representative of End of Cycle (EOC) conditions and are the most negative possible. A curve of moderator reactivity versus temperature is used assuming all rods in with the exception of the worst stuck rod.

9) The control rod worth at trip is the minimum rod worth with the worst stuck rod.

10) The multi-node SG model will be utilized in analyzing this event. This model provides the more accurate assessment of SG level needed for this evaluation.

TABLE 4.2.2.8-1

KEY PARAMETERS ASSUMED FOR THE FEEDWATER LINE BREAK EVENT

ST. LUCIE UNIT 2

Parameter	Units	Value
Total RCS Power (Core Thermal Power x 1.02)	MWth	2754
Initial Reactor Coolant System Pressure	psia	2408
Initial Core Coolant Inlet Temperature	°F	552
Initial RCS Vessel Flow Rate	gpm	400,000
Moderator Temperature Coefficient	pcm/°F	-27.
Doppler Coefficient	pcm/°F	-1.8
CEA Worth at Trip	%Δρ	-7.52

[illegible]

4.2.2.8.2 Results

The Feedwater Line Break event initiated from the conditions given in Table 4.2.2.8-1 with the assumptions listed above results in a low SG level trip at 3.7 seconds. A Main Feedwater Isolation Signal is generated from the AFAS initiation at 3.8 seconds. The loss of AC occurs at 4.9 seconds. A Main Steam Isolation signal is generated on low SG pressure at 41.7 seconds into the event. The minimum primary average temperature prior to the actuation of Safety Injection (SI) occurs at 120 seconds with a value of 472 Degrees F. SI flow occurs at 130 seconds after the initiation of the event. After the affected SG goes dry, the unaffected SG is available to remove decay heat and bring the plant to a safe shutdown condition.

Table 4.2.2.8-2 presents the sequence of events for this transient. Figures 4.2.2.8-1 and 4.2.2.8-2 present the comparison between the results of the Feedwater Line Break performed using RETRAN and the results obtained by CE for the limiting SLB in the FSAR for pressurizer pressure and average RCS temperatures, respectively. It can be seen that the SLB produces a more limiting cooldown transient as expected. The difference between the SLB and the FLB is the more rapid cooldown in the SLB due to the larger break area associated with the SLB. The FLB response is representative of the results which would occur by examining a non-limiting SLB with the same break area (1.44 ft^2).

Figures 4.2.2.8-3 to 4.2.2.8-7 present the response of primary power, pressurizer level, RCS coolant temperatures for the unaffected loop and the affected loop, and the calculated SG pressures for the two loops for this Feedwater Line Break event.

The results clearly show that the Feedwater Line Break event is a cooldown which is bounded by the results of the Steamline Break event found in Section 15.1.4.3.5.3 of the FSAR.

The DNBR response to the FLB would be bounded by the Loss of Flow event described in Section 15.3.2.2.6 of the FSAR. Since there is no return to power for the FLB, and natural circulation flow would be maintained, this event would not approach the DNBR limit.

TABLE 4.2.2.8-2

SEQUENCE OF EVENTS

FEEDWATER LINE BREAK

ST. LUCIE UNIT 2

EVENT	TIME (seconds)	SETPOINT OR VALUE
Feedwater Line Break	0.0	1.44 ft^2
Low SG Level Trip	3.7	20.5 % NR
AFAS Signal Generated Affected SG	3.8	19.0 % NR
MFIS Generated	3.8	AFAS signal
Trip Breakers Open	4.8	---
Loss of AC Power	4.9	---
Turbine Trip Signal	5.8	---
MSIS Signal	42.0	460 psia
AFAS Signal Generated Unaffected SG	48.0	19.0 % NR
Auxiliary Feedwater Isolation Signal, Affected SG on High Differential Pressure	81.0	281 psia
SI Signal Initiated	100.0	1578 psia
Minimum Average Temperature Prior to SI	120.0	472 Degree F
SI Flow Begins	130.0	---

FIGURE 4.2.2.8-1

St. Lucie Unit 2 Feedline Break

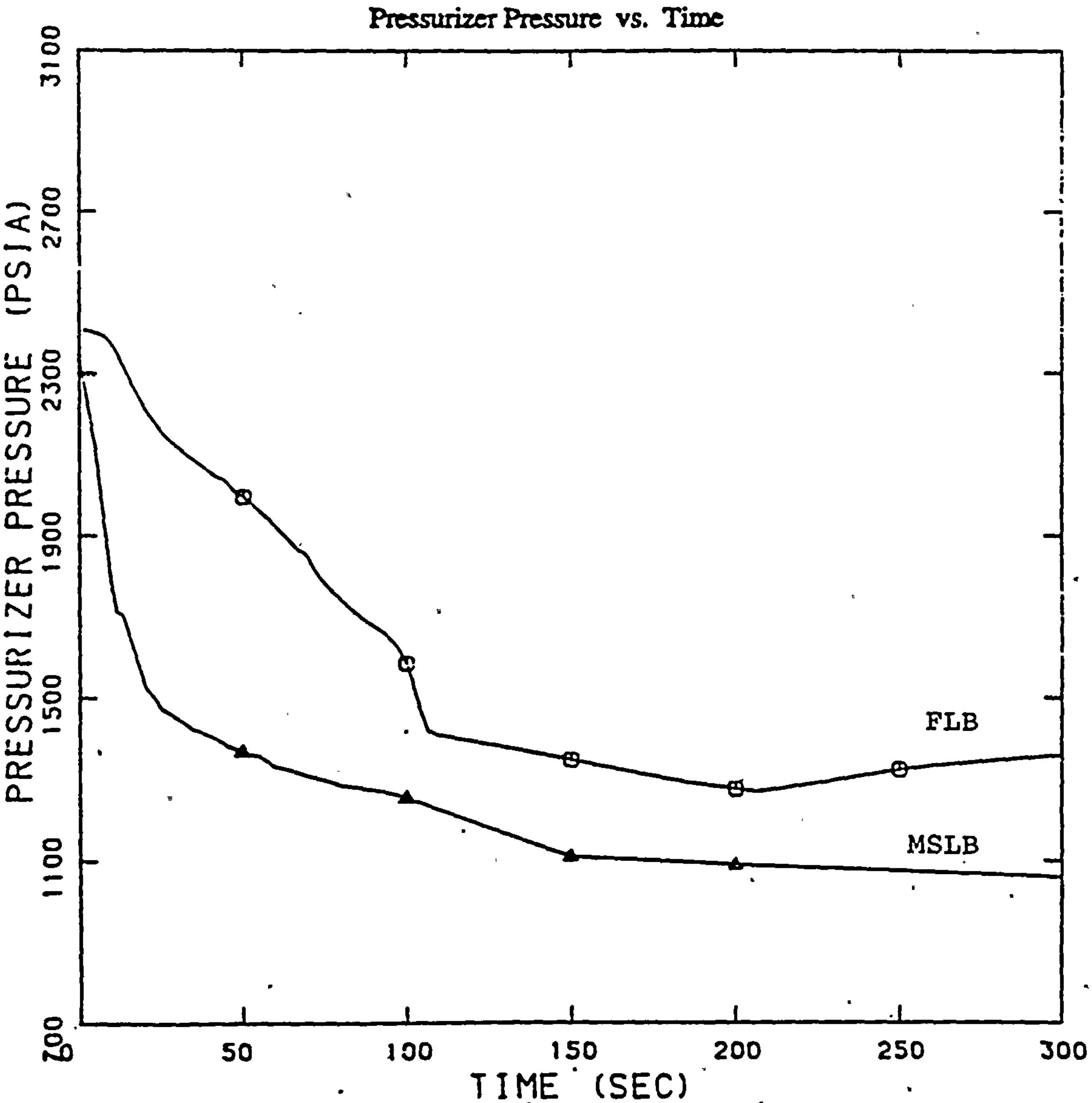


FIGURE 4.2.2.8-2

St. Lucie Unit 2 Feedline Break

RCS Average Temperature vs. Time

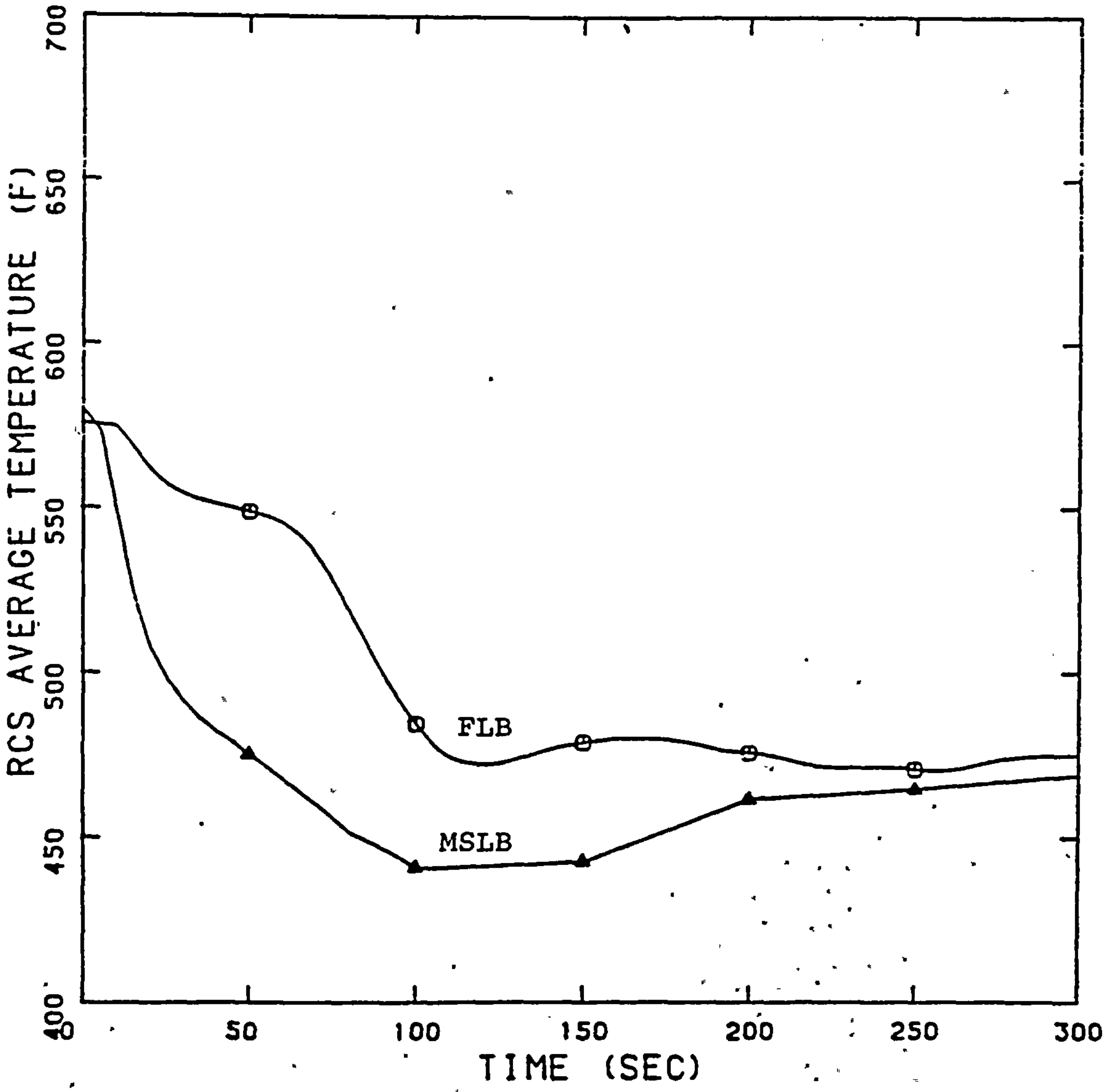


FIGURE 4.2.2.8-3

St. Lucie Unit 2 Feedline Break

Core Power vs. Time

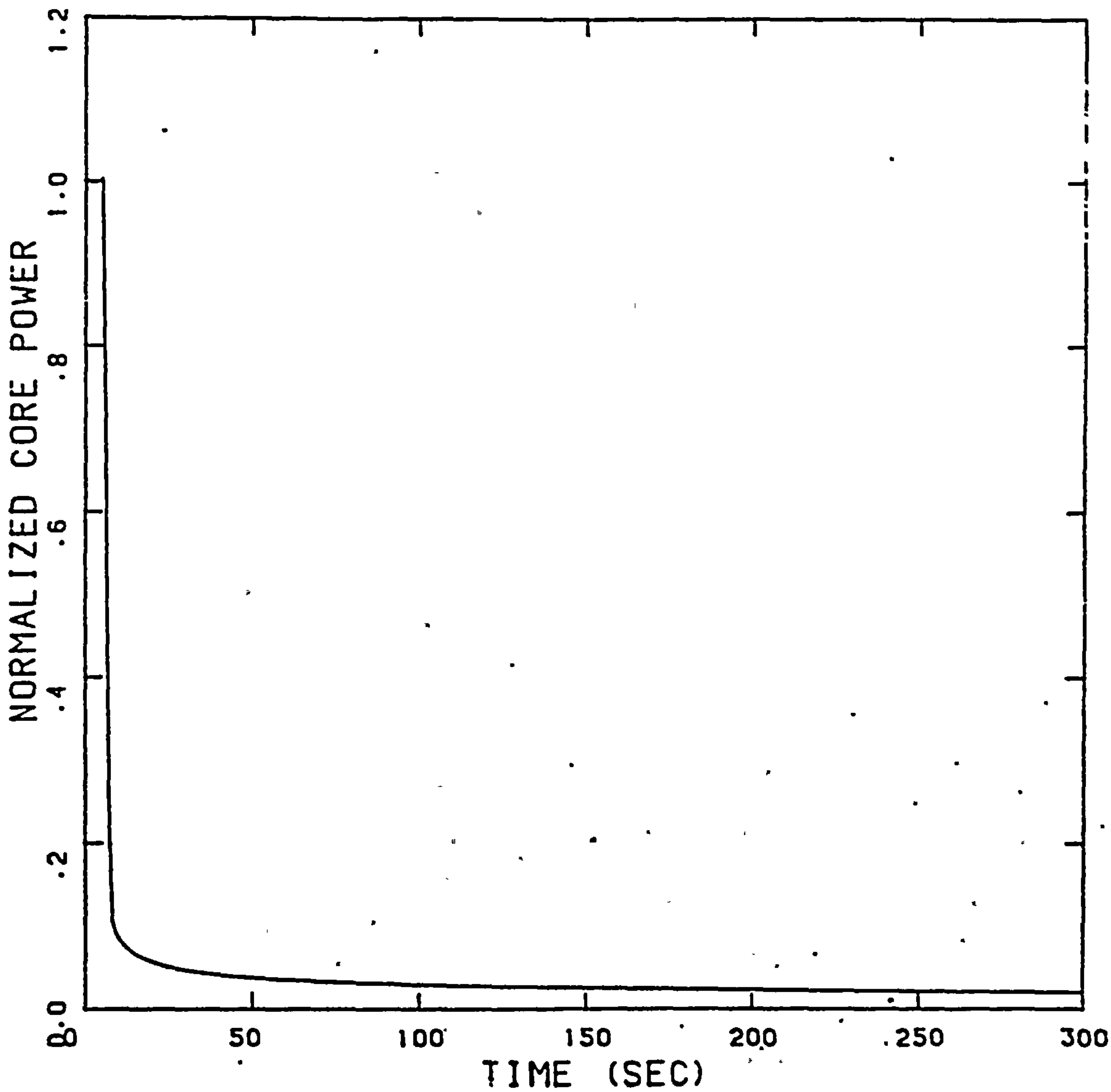


FIGURE 4.2.2.8-4

St. Lucie Unit 2 Feedline Break

Pressurizer Level vs. Time

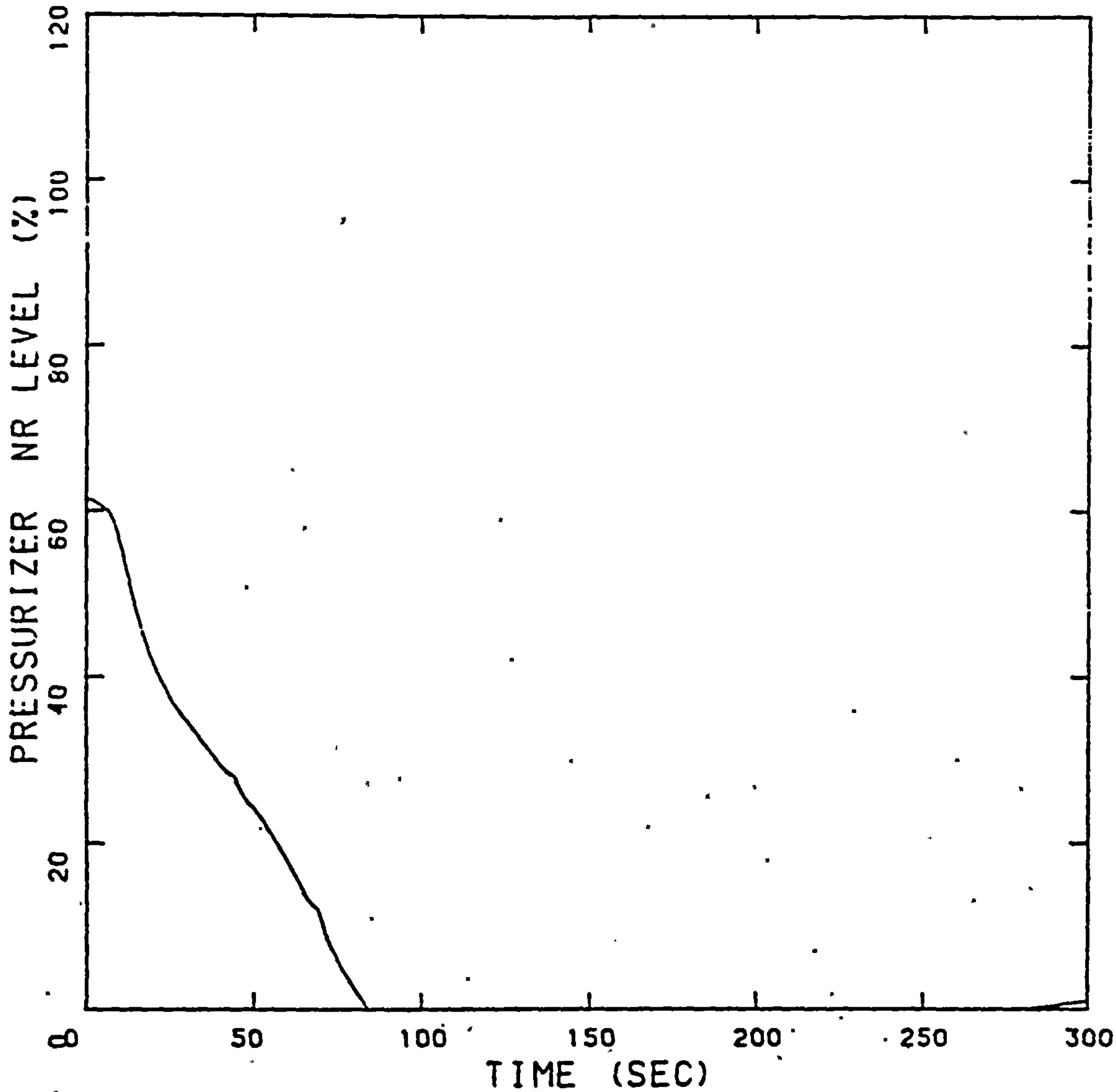


FIGURE 4.2.2.8-5

St. Lucie Unit 2 Feedline Break

Unaffected Loop Temperatures vs. Time

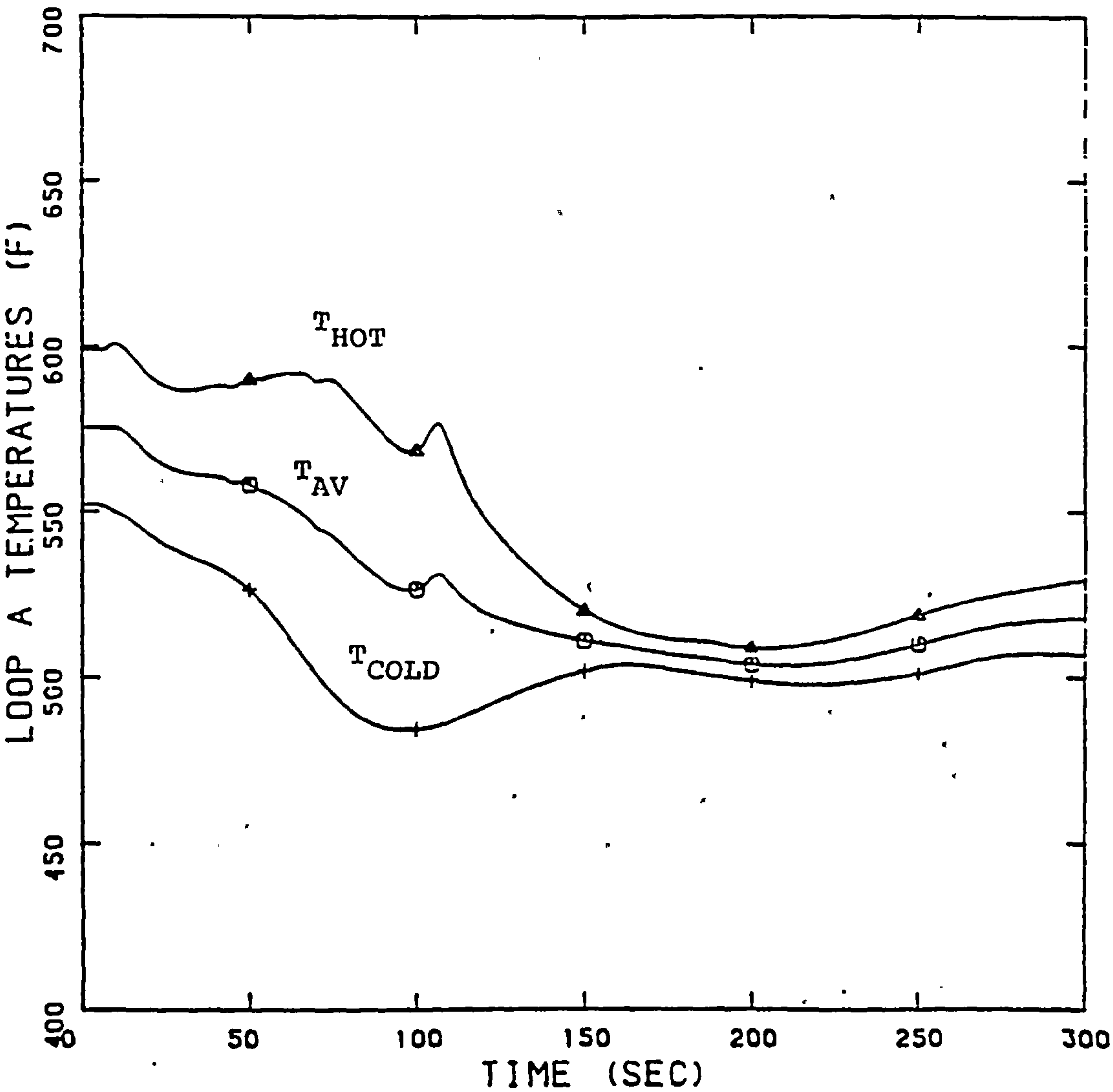


FIGURE 4.2.2.8-6

St. Lucie Unit 2 Feedline Break

Affected Loop Temperatures vs. Time

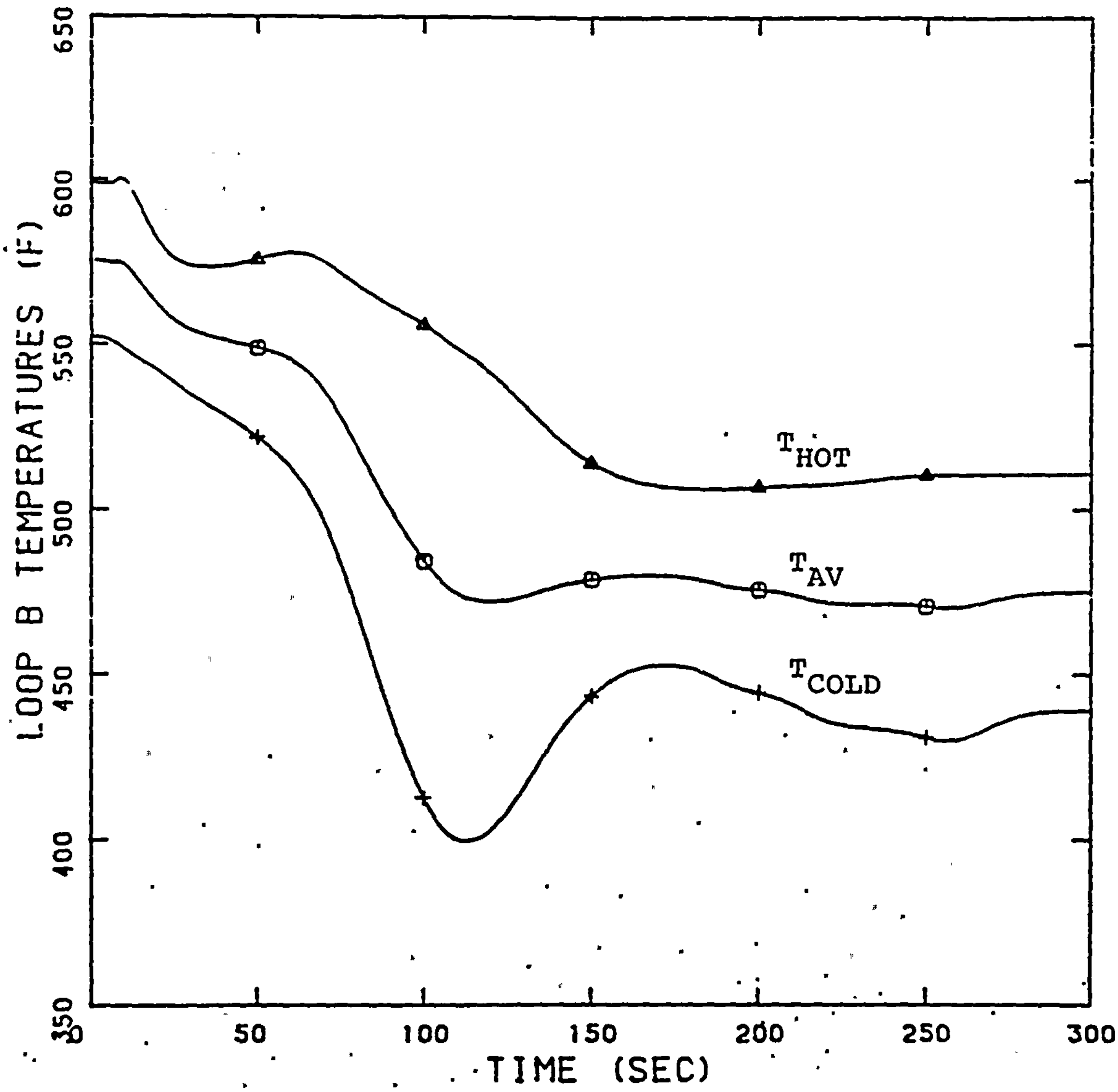
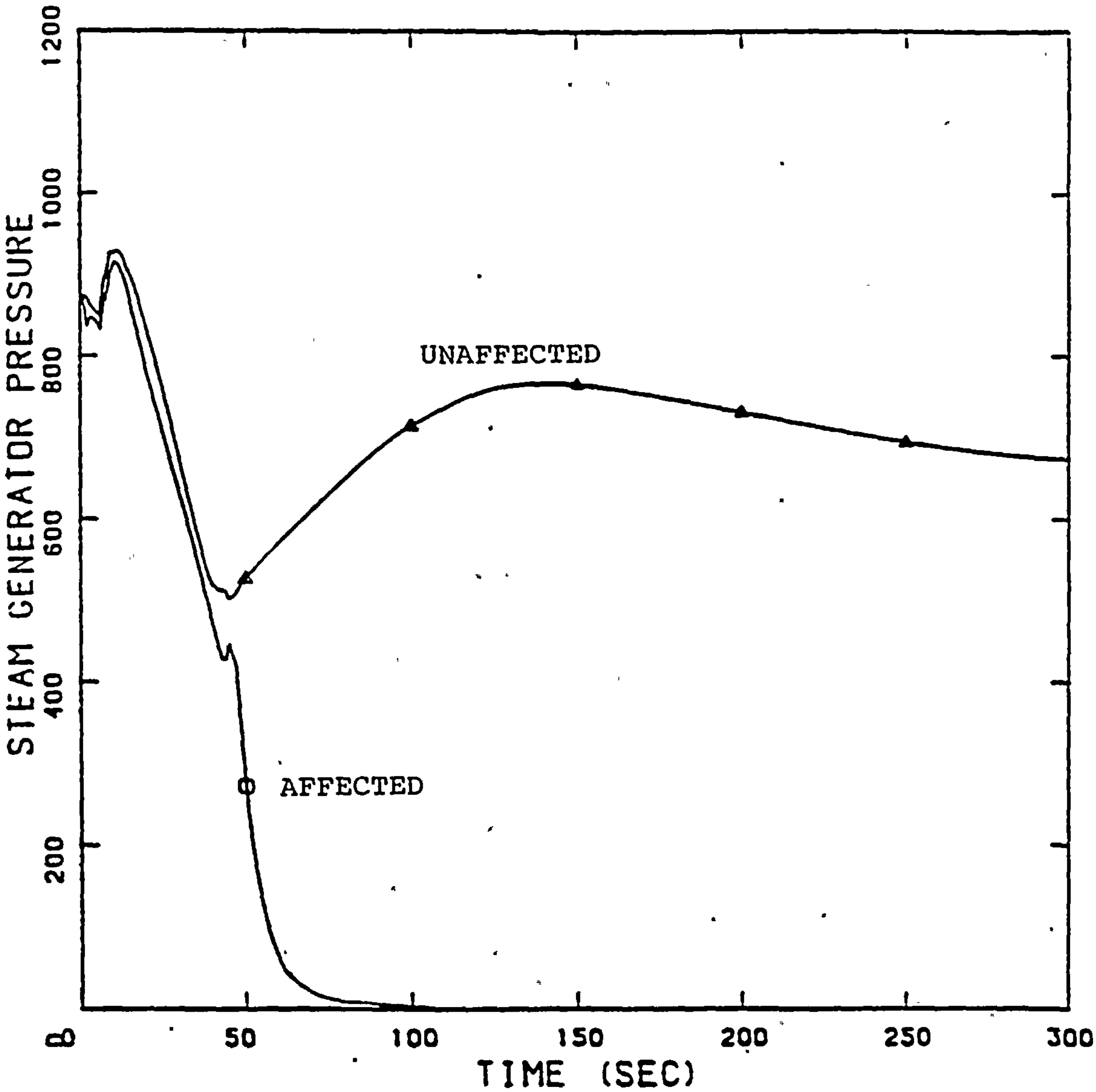


FIGURE 4.2.2.8-7

St. Lucie Unit 2 Feedline Break

Affected and Unaffected Steam Generater Pressures vs. Time



4.2.3 Turkey Point Units 3 and 4

4.2.3.1 Loss of External Load

A loss of external load event is caused by abnormal events in the electrical distribution network. The turbine is protected from a complete loss of external load by a mechanical overspeed trip mechanism as well as a electro-magnetic pickup on the governing channel which acts to trip the turbine if a complete loss of load occurs. More details of these equipment protection systems are found in Section 10.2.2 and Section 14.1.13 of the Turkey Point FSAR. Upon occurrence of a turbine trip above 10% of full power, a signal is supplied to the reactor protective system to trip the reactor.

A fast pressurization of the primary will result from the closure of the turbine stop valves due to their fast response time of approximately 0.1 seconds. The primary side pressurization results from the reduced primary to secondary heat transfer which is produced by the increase in secondary temperature and pressure that occurs with the stoppage of steam flow from the secondary side. However, the expected reactor trip on turbine trip would quickly reverse the pressurization event and bring the reactor to a safe shutdown condition prior to the primary pressure reaching the high pressure trip setpoint. In addition, action of the turbine bypass system will mitigate the pressurization effects of a Loss of External Load. Even if credit is not taken for the reactor trip on turbine trip or action of the turbine bypass system, this transient is bounded relative to primary pressurization by the Loss of Condenser Vacuum event discussed in Section 4.2.3.3 because of the assumption in the Loss of Condenser Vacuum that an instantaneous loss of feedwater will occur. The loss of main feedwater increases the heatup of the primary and therefore increases the pressurization rate of the transient.

Evaluation of the DNBR response to a Loss of External Load Event shows qualitatively that DNBR will vary only slightly during this transient. DNBR is negatively affected by decreases in flow and pressure and increases in heat flux and temperature. Transients which can challenge the DNBR limit result from either significant decreases in flow, i.e., Loss of Flow events or significant increases in heat flux, i.e., Rod Withdrawal at Power. During the Loss of External Load, the increase in temperature will be offset by the increase in pressure and only slight variations in DNBR would be expected. For all reasonable combinations of input parameters for the Loss of External Load, other events reported in Chapter 14 of the FSAR would be limiting with respect to DNBR response.

4.2.3.2 Turbine Trip

A turbine trip is an event in which the steam flow from the steam generators is stopped through closure of the turbine stop valves and governing control valves. Each stop valve is an oil operated, spring closing valve controlled primarily by the turbine over speed trip devices. The turbine overspeed trip pilot is actuated by one of the following signals to close the stop valves:

- Turbine thrust bearing trip
- Low bearing oil pressure trip
- Low condenser vacuum
- Solenoid trip
- Overspeed trip
- Manual trip
- Any generator fault
- Generator lockout
- Hi-hi SG level
- Safety Injection signal
- Reactor Trip (above P-7)

Upon occurrence of a turbine trip above 10% of full power, a signal is supplied to the reactor protective system to trip the reactor.

A fast pressurization of the primary will result from the closure of the turbine stop valves due to their fast response time of approximately 0.1 seconds. The primary side pressurization results from the reduced primary to secondary heat transfer which occurs with the stoppage of steam flow from the secondary side. However, the expected reactor trip on turbine trip would quickly reverse the pressurization event and bring the reactor to a safe shutdown condition prior to the primary pressure reaching the high pressure trip setpoint. In addition, action of the turbine bypass system will mitigate the pressurization effects of a Turbine Trip. Even if credit is not taken for the reactor trip on turbine trip or action of the turbine bypass system, this transient is bounded relative to primary pressurization by the Loss of Condenser Vacuum event discussed in Section 4.2.3.3 because of the assumption in the Loss of Condenser Vacuum that a instantaneous loss of feedwater will occur. The loss of main feedwater increases the heatup of the primary and therefore increases the pressurization rate of the transient.

Evaluation of the DNBR response to a Turbine Trip shows qualitatively that DNBR will vary only slightly during this transient. As discussed in Section 4.2.3.1, the bounding events for DNBR response would remain the Loss of Flow transients as described in Chapter 14.1.9 of the FSAR.

4.2.3.3 Loss of Condenser Vacuum

4.2.3.3.1 Limiting Event Development

A Loss of Condenser Vacuum may occur due to the failure of the Circulation Water System, the failure of the Main Condenser Evacuation System to remove non-condensible gases, or the in-leakage of an excessive amount of air through a turbine gland or condenser seal. Low condenser pressure will generate a turbine trip signal. The turbine trip signal causes the turbine stop valves to close and stops steam flow out of the steam generators. Upon occurrence of a turbine trip above 10% of full power, a signal is supplied to the reactor protective system to trip the reactor. The Loss of Condenser Vacuum will disable the turbine bypass system and will result in a gradual rampdown of main feedwater.

For purposes of developing an event that bounds the potential for primary pressurization, several conservative assumptions will be made relative to the Loss of Condenser Vacuum event. The following assumptions will be utilized in this analysis:

1) For this bounding event, coincident with the Loss of Condenser Vacuum, the turbine stop valves are assumed to instantly close on high condenser back pressure. This assumption assures the fastest possible pressurization rate.

2) A reactor trip on turbine trip signal will not be credited. Not allowing the reactor trip on turbine trip signal to operate will allow the high pressurizer pressure trip to activate to initiate reactor scram. Use of the high pressurizer pressure trip rather than any other system trip (such as steam generator low level) will insure that the highest possible primary pressure will be reached during the transient.

3) Main feedwater flow rate will be instantaneously set to zero to maximize the decrease in primary to secondary heat transfer.

4) Action of the PORV's and pressurizer sprays will not be credited since this would mitigate the pressurization of the primary system.

[illegible]

The choices for other key initial conditions are listed in Table 4.2.3.3-1. For evaluation of the peak primary pressure, initial conditions are chosen such that the time to reactor trip on high pressure is maximized and that a high rate of change of pressure is achieved. The basis of the key initial conditions is provided briefly in the following discussion:

5) The maximum possible power (with uncertainties) will increase the heatup rate during the transient and result in a higher peak pressure.

6) A minimum initial primary pressure value is utilized to allow for a larger heatup and power increase prior to reaching the reactor trip setpoint.

7) In order to maximize the length of time prior to opening the secondary side safety valves, a minimum initial reactor temperature value and a maximum steam generator tube plugging level are chosen. Opening of the secondary side safety valves will increase primary to secondary heat transfer and result in a reduction in the primary temperature. Delay in opening the secondary safety valves will, therefore, act to maximize the primary pressurization. In addition, the use of the maximum steam generator tube plugging value acts to reduce initial primary to secondary heat transfer and will cause a slightly faster primary pressurization and a higher final peak pressure.

8) A positive MTC, conservative relative to the limits in Technical Specifications is chosen in order to produce a power increase in conjunction with the primary coolant temperature increase. The least negative Doppler Coefficient is also chosen to allow the maximum possible power increase.

9) A bottom peaked axial shape corresponding to the limits of the full power operating band is used as the initial condition for the scram reactivity data. The use of a bottom peaked axial shape maximizes the power increase, during the transient, slightly because the effects of the negative control reactivity are delayed with respect to those from top peaked or cosine axial shapes.

TABLE 4.2.3.3-1

KEY PARAMETERS ASSUMED FOR THE LOSS OF CONDENSER VACUUM EVENT

TURKEY POINT UNITS 3 & 4

Parameter	Units	Value
Total RCS Power (Core Thermal Power x 1.02)	MWth	2244
Initial Reactor Coolant System Pressure	psia	2200
Initial Core Coolant Inlet Temperature	^{o}F	542
Steam Generator Tube Plugging	%	5
Initial RCS Vessel Flow Rate	gpm	268,500
Moderator Temperature Coefficient	pcm/^{o}F	+2.0
Doppler Coefficient	pcm/^{o}F	-1.0
Rod Worth at Trip	% $\Delta\rho$	4.0

4.2.3.3.2 Results

The Loss of Condenser Vacuum event initiated from the conditions given in Table 4.2.3.3-1, results in a high pressurizer pressure trip condition at 7.1 seconds after initiation of the event. At 8.3 seconds, the primary pressure reaches its maximum value of 2601 psia. The maximum pressure point is calculated by RETRAN at the pump discharge. The maximum pressure value is shown to be well below the 2750 psia acceptance criteria.

The peak secondary side pressure occurs at 15.7 seconds and reaches a value of 1151 psia. This value is well below the 1210 psia acceptance criteria for the secondary side system.

Table 4.2.3.3-2 presents the sequence of events for this transient. Figures 4.2.3.3-1 to 4.2.3.3-6 show the power, heat flux, RCS pressure, RCS coolant temperatures, steam generator pressure, and pressurizer level response to this bounding Loss of Condenser Vacuum event.

A Loss of Condenser Vacuum event was also evaluated using best-estimate parameters. In this case, all initial conditions were nominal in an attempt to represent the actual plant response if a real Loss of Condenser Vacuum event were to occur. In order to obtain some pressurization of the primary system, this case did not take credit for the action of the reactor protective system to initiate a reactor trip on the turbine trip signal. For this case, the peak pressure reached a maximum of 2488 psia. The difference between the peak pressure calculated using our licensing methods and the best-estimate results provides a measure of the conservatism applied in the design inputs.

Evaluation of the DNBR response to a Loss of Condenser Vacuum Event show qualitatively that DNBR will vary only slightly during this transient. The Loss of Flow events as described in Section 14.1.9 of the FSAR will remain bounding relative to DNBR response.

TABLE 4.2.3.3-2

SEQUENCE OF EVENTS

LOSS OF CONDENSER VACUUM

TURKEY POINT UNITS 3 & 4

EVENT	TIME (seconds)	SETPOINT OR VALUE
Loss of Condenser Vacuum	0.0	---
High Pressurizer Pressure	7.10	2455 psia
Pressurizer Safeties Open	7.98	2525 psia
Peak RCS Pressure	8.3	2601 psia
Rods Begin to Drop Into Core	9.1	---
Steam Generator Safeties Open	10.15	1111 psia
Pressurizer Safeties Close	12.8	2400 psia
Maximum Steam Generator Pressure	15.7	1151 psia

FIGURE 4.2.3.3-1

Turkey Point Units 3 & 4 Loss of Condenser Vacuum

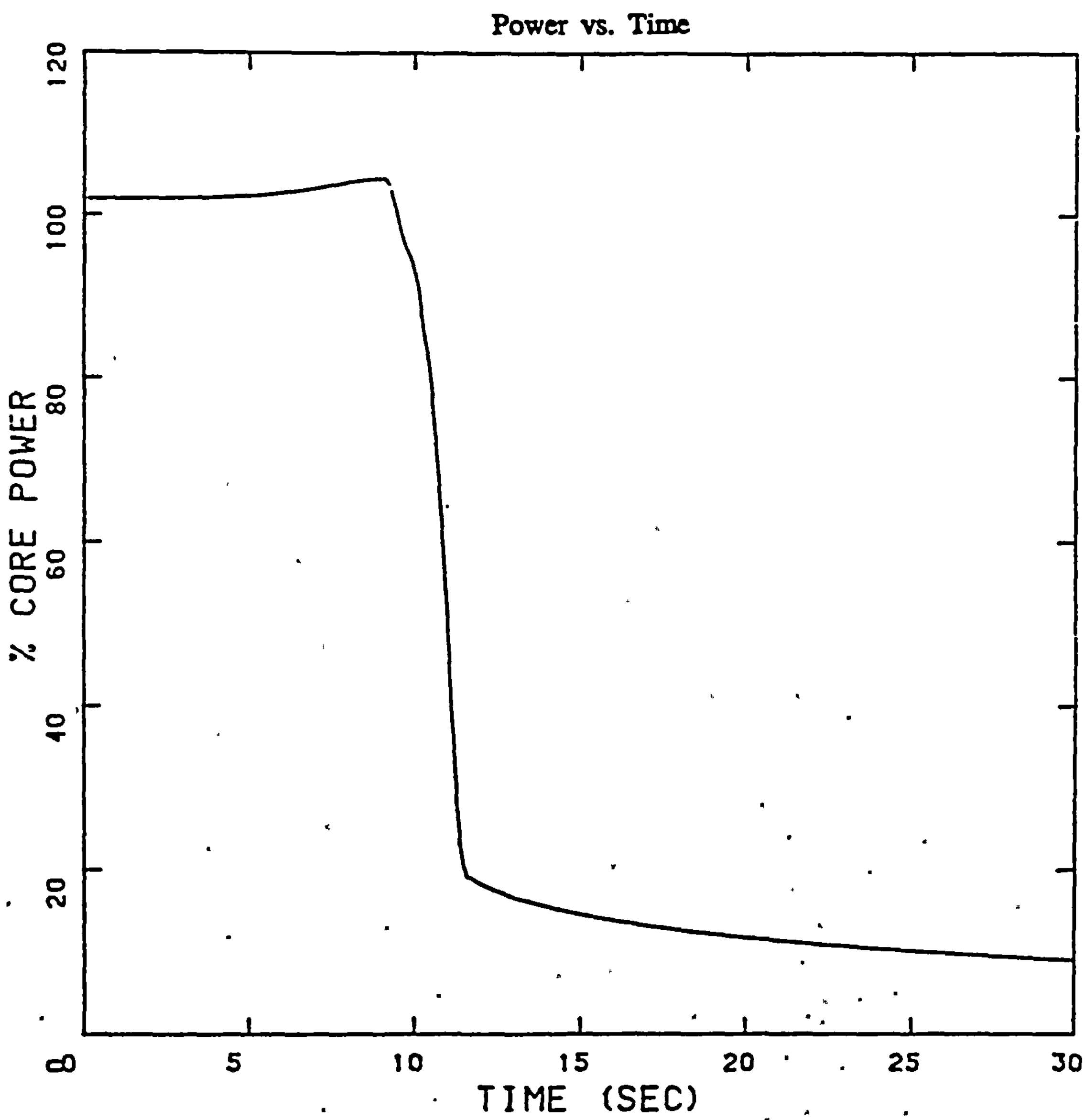


FIGURE 4.2.3.3-2

Turkey Point Units 3 & 4 Loss of Condenser Vacuum

Heat Flux vs. Time

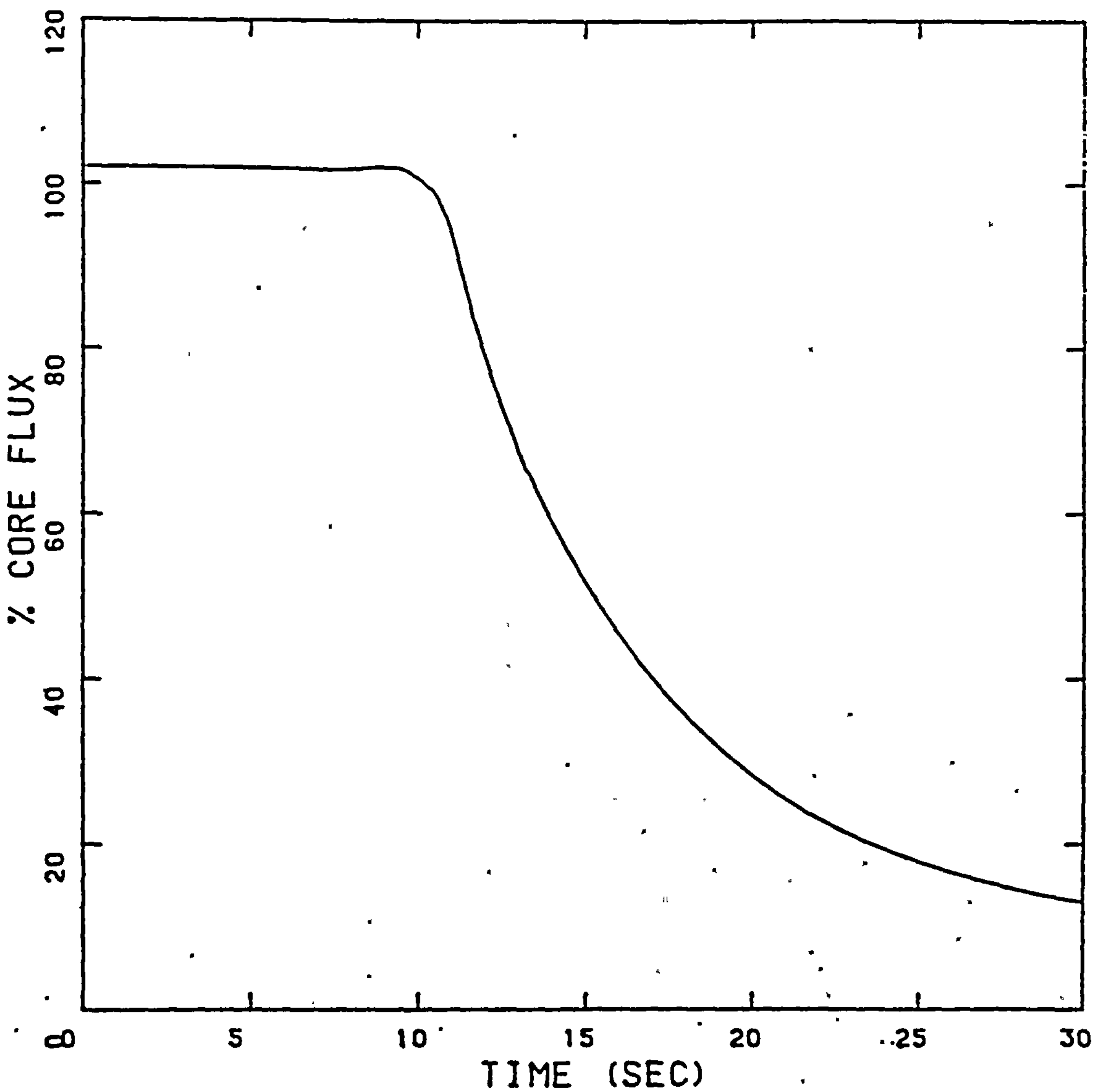


FIGURE 4.2.3.3-3

Turkey Point Units 3 & 4 Loss of Condenser Vacuum

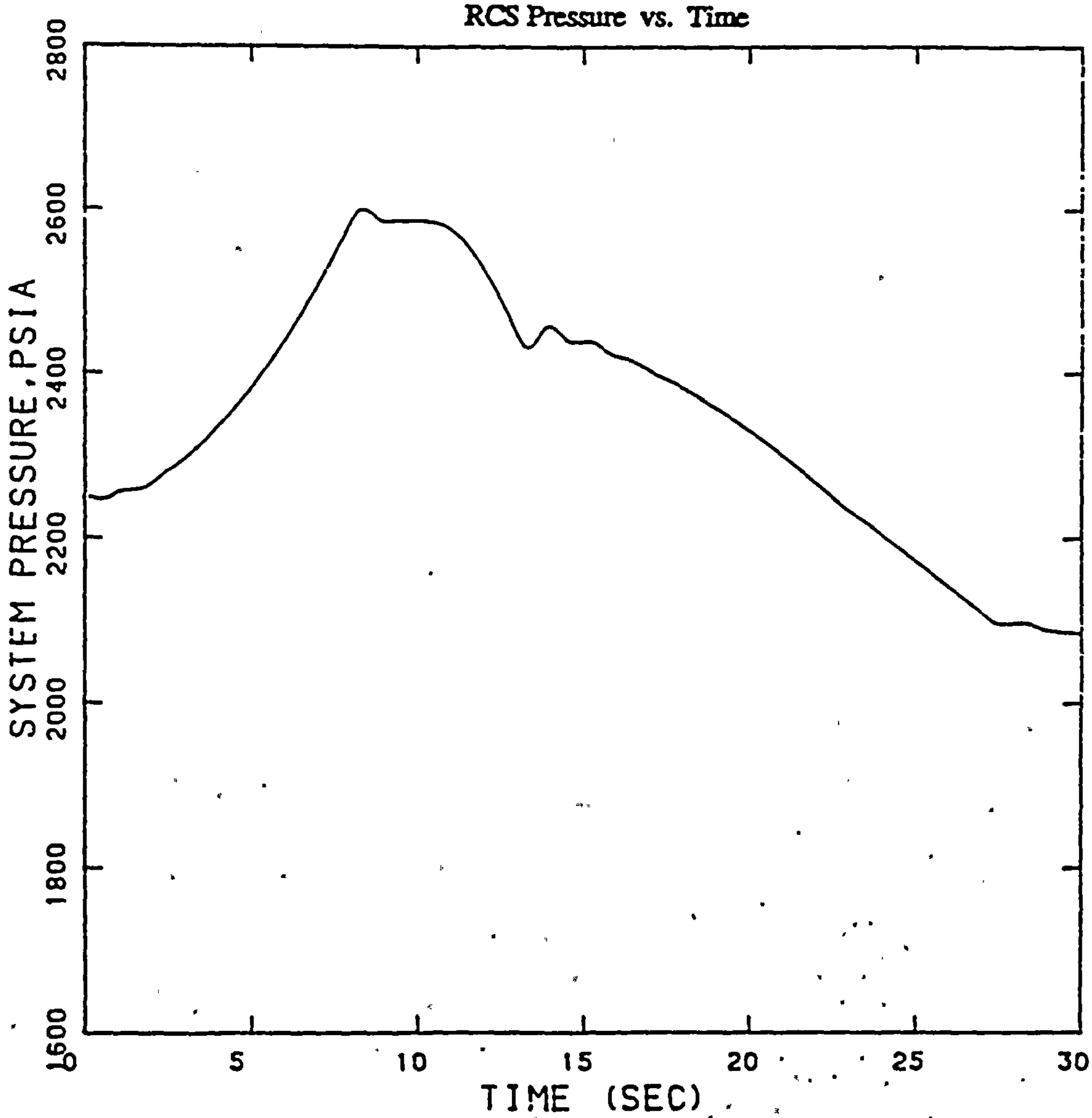


FIGURE 4.2.3.3-4

Turkey Point Units 3 & 4 Loss of Condenser Vacuum

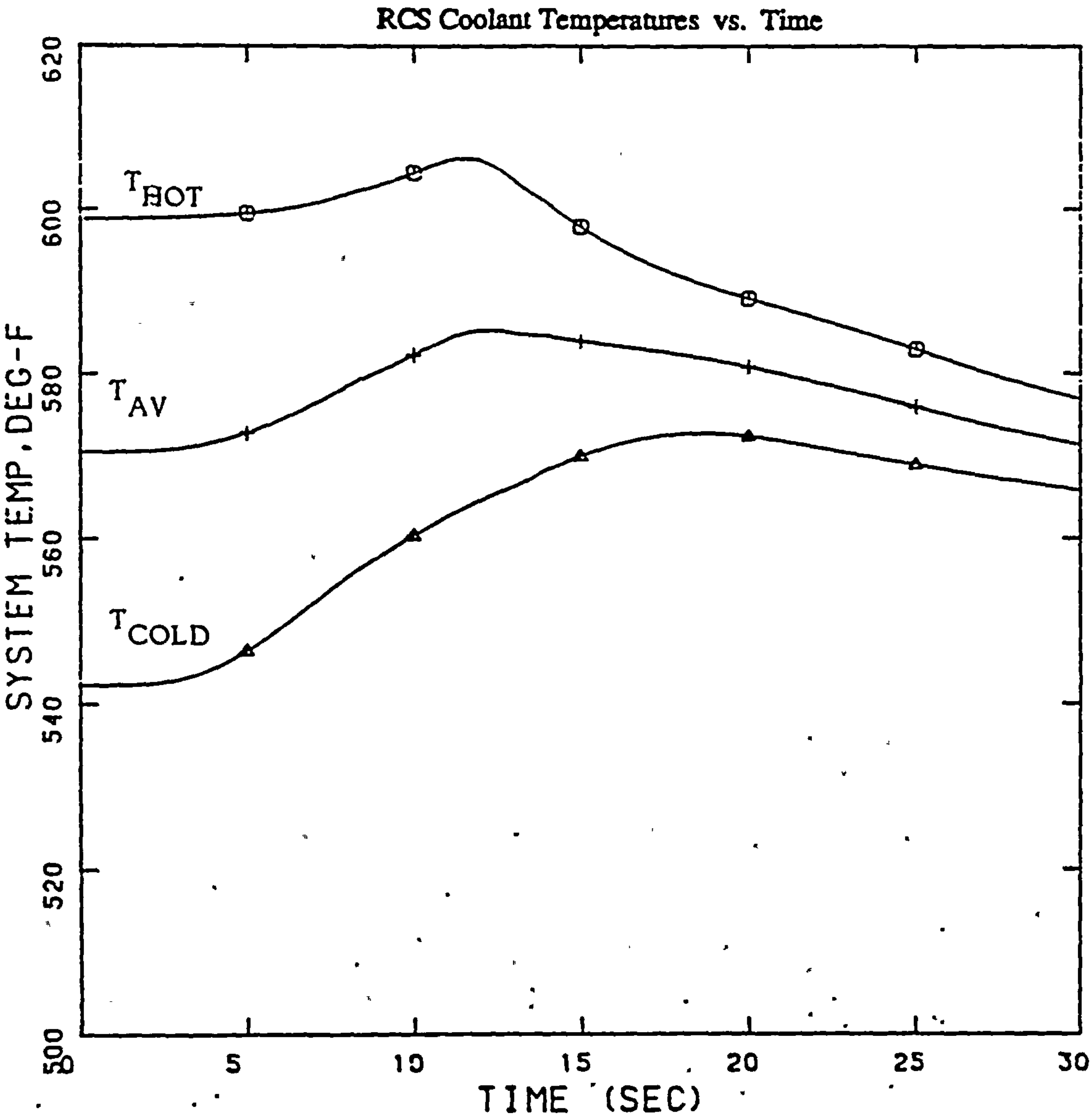


FIGURE 4.2.3.3-5

Turkey Point Units 3 & 4 Loss of Condenser Vacuum

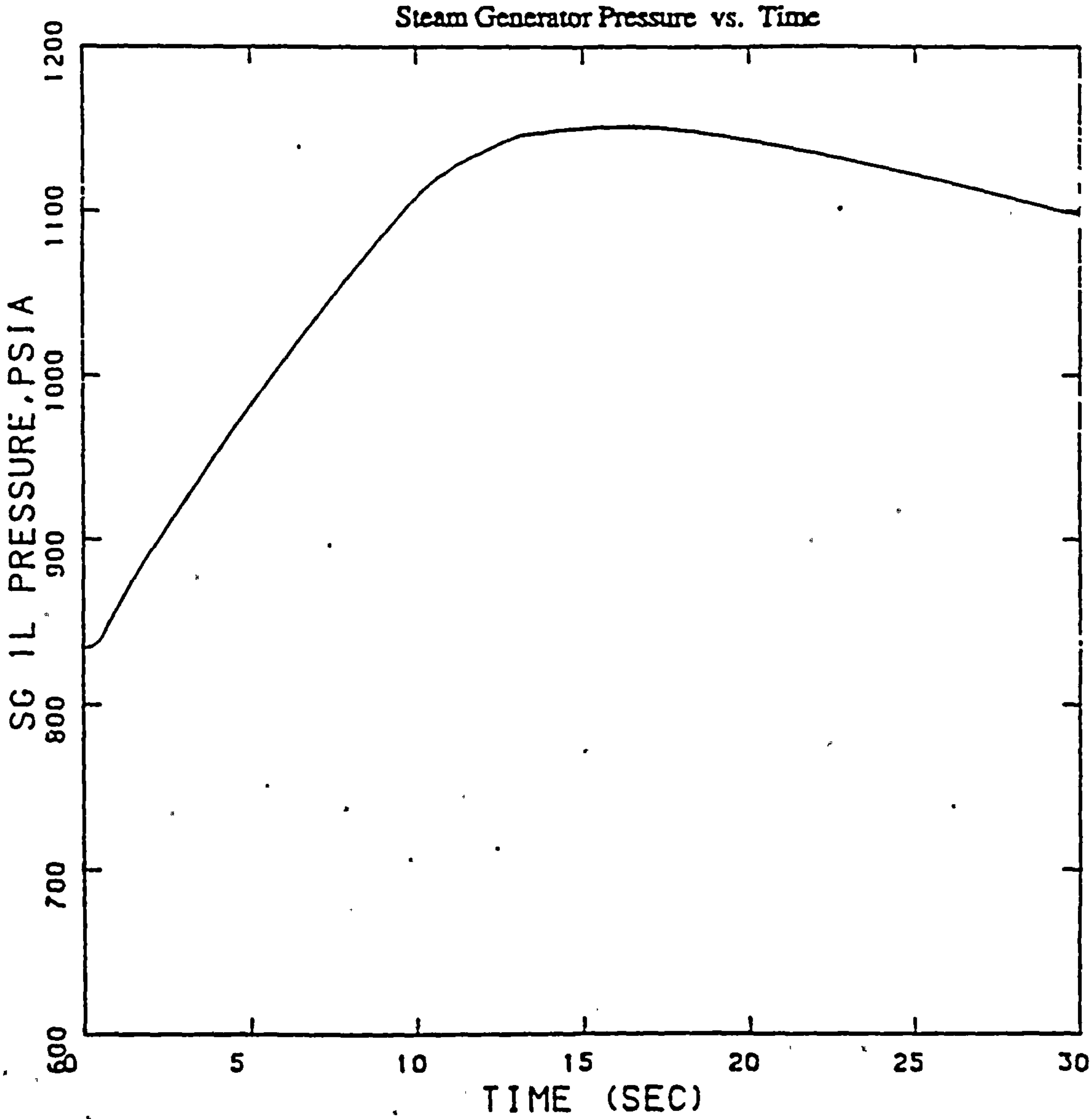
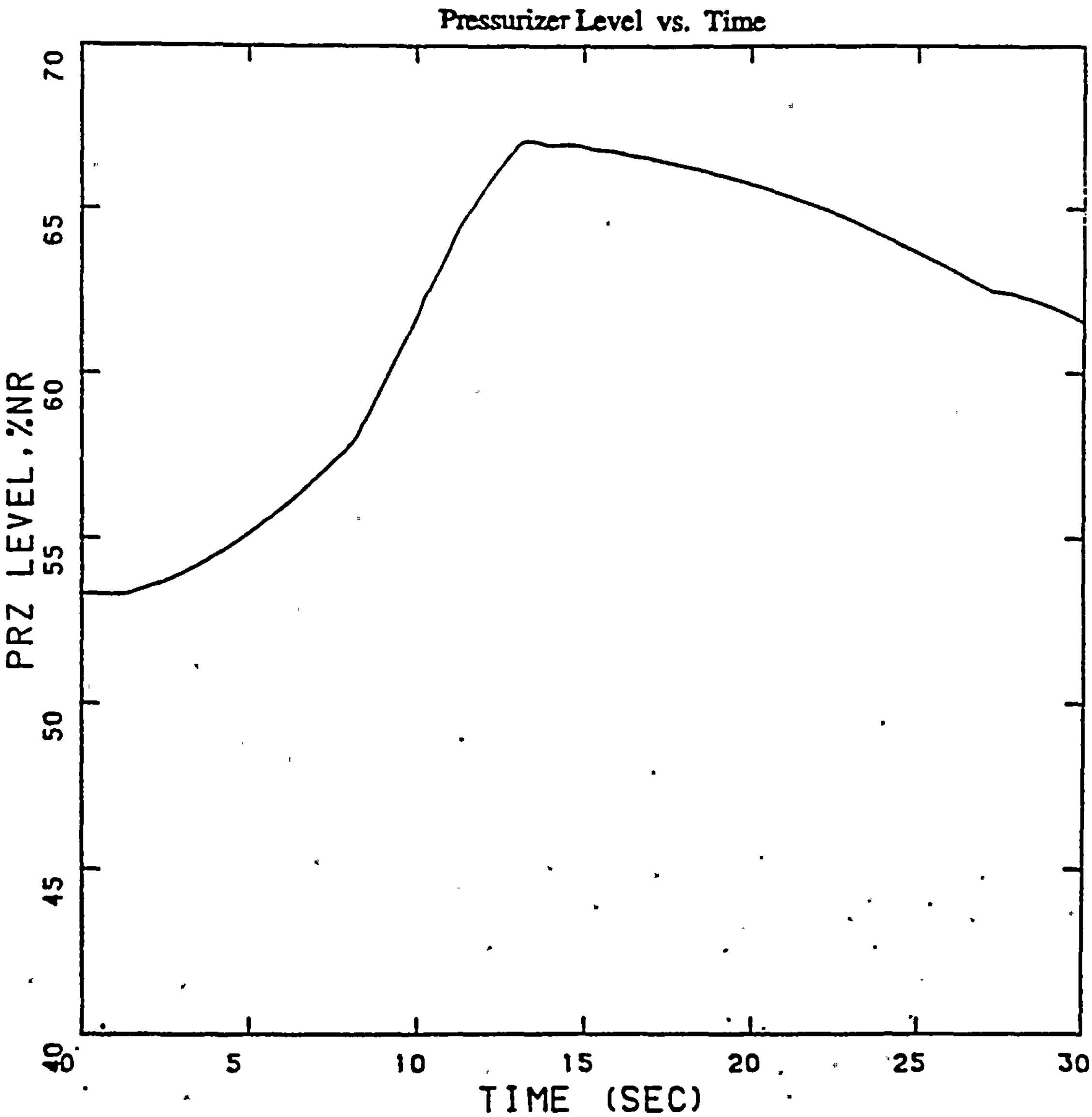


FIGURE 4.2.3.3-6

Turkey Point Units 3 & 4 Loss of Condenser Vacuum



4.2.3.4 Main Steam Isolation Valve Closure

A MSIV Closure event which resulted in the closure of the three MSIV's for the Turkey Point Units would be bounded by the Loss of Condenser Vacuum event due to the slower response time of the MSIV compared to the turbine stop valves. Closure of less than all three MSIV's would produce only a limited primary heat production to secondary heat removal mismatch which would also be bounded by the Loss of Condenser Vacuum results.

DNBR response to this event is less limiting than the Loss of Flow events described in Section 14.1.9 of the FSAR.

4.2.3.5 Steam Pressure Regulator Failure

A Steam Pressure Regulator Failure for Turkey Point Units 3 & 4 would be similar to a failure of the turbine control system. A failure in this system which would close the turbine valves would be bounded by the Loss of Condenser Vacuum event due to the much more rapid action of the turbine stop valves. The rapid action of the stop valves relative to the control valves increases the primary heatup and maximizes primary pressurization.

DNBR response to this event is less limiting than the Loss of Flow events described in Section 14.1.9 of the FSAR.

[illegible]

4.2.3.6 Loss of Non-emergency AC Power to the Station Auxiliaries

4.2.3.6.1 Limiting Event Development

The Loss of Non-emergency AC Power to the Station Auxiliaries (LOAC) is defined as a complete loss of offsite electrical power and concurrent turbine trip. As a result of this event, electrical power would be unavailable for the reactor coolant pumps and main feedwater pumps. The units would therefore experience a simultaneous loss of load, a loss of feedwater and a loss of forced reactor coolant flow.

The LOAC is followed by automatic startup of the emergency diesel generators. The power output of each is sufficient to supply electrical power to all engineered safety features and to ensure the capability of establishing and maintaining the units in a safe shutdown condition. Since power is not available for the control rod drive mechanisms, an immediate reactor shutdown will occur.

The three turbine driven auxiliary feedwater pumps would be started on the loss of offsite power and would begin providing auxiliary feedwater to the two units. The turbines utilize steam from the main steam line to drive the auxiliary feedwater pumps to deliver feedwater to the steam generators. The turbine driver exhausts the steam to the atmosphere. The pumps take suction directly from the condensate storage tanks for delivery to the steam generators.

Following the reactor trip, decay heat removal would be provided by the action of the auxiliary feedwater pumps and the steam generator safety valves.

The evaluation of this transient is performed in order to verify that sufficient heat removal capacity is available to ensure that the pressurizer does not become water solid.

In order to develop a bounding analysis of the response of the Turkey Point units to a LOAC, several conservative assumptions will be made relative to the LOAC event. The following assumptions will be utilized in this analysis:

1) Reactor trip due to loss of power to the control rod drive mechanisms at the initiation of the LOAC is not credited.

2) Reactor coolant pump coastdown will be delayed until the reactor trip signal is generated. This assumption ensures that a quick reactor trip on low flow will not be generated. An early reactor trip would reduce the severity of the event since the Steam Generator water mass inventory after reactor trip would be larger and consequentially the primary system long term heatup would be reduced.

3) A single failure will occur in the auxiliary feedwater system. This results in the availability of only one auxiliary feedwater pump supplying water to the two units. Auxiliary feedwater for this analysis will be assumed to be only 110 gpm until operator action, 10 minutes after reactor trip, raises the flow to 230 gpm. It is also assumed that a feedwater piping volume of 110 cubic feet contains liquid at normal feedwater temperatures which must be emptied into the SG's prior to cold auxiliary feedwater reaching the SG's.

4) The pressurizer PORV's are available and assumed to operate. The action of the PORV's will maximize the pressurizer level when they open to relief primary system pressure. Pressurizer sprays are also assumed available to aid in the maximization of the pressurizer level.

5) Core decay heat generation is based on the 1971 version ANS-5.1. This standard is a very conservative representation of the decay heat release rates.

The choice for the key initial conditions are listed in Table 4.2.3.6-1. For evaluation of the heat removal capacity, initial conditions are chosen to maximize the primary heat content. The basis of the key input parameters is discussed briefly in the following:

6) The maximum possible power (with uncertainties) will increase the primary heat source during the transient and maximize the primary heatup after trip.

7) The maximum initial primary pressure and temperature will be assumed to maximize the initial water density within the primary system.

8) Minimum RCS flow and maximum steam generator tube plugging are assumed to minimize the primary to secondary heat transfer during the transient.

9) Conservative reactivity parameters are assumed in order to increase primary power prior to the reactor trip.

10) The multi-node SG model will be used in this analysis. This model provides a more accurate assessment of SG level than our standard single node model and is needed for this evaluation since the reactor trip occurs due to SG low level.

TABLE 4.2.3.6-1

KEY PARAMETERS ASSUMED FOR THE LOSS OF NON-EMERGENCY AC POWER

TURKEY POINT UNITS 3 & 4

Parameter	Units	Value
Total RCS Power (Core Thermal Power x 1.02)	MWth	2244
Initial Reactor Coolant System Pressure	psia	2300
Initial Core Coolant Inlet Temperature	^{o}F	550.2
Steam Generator Tube Plugging	%	5
Initial RCS Vessel Flow Rate	gpm	268,500
Moderator Temperature Coefficient	pcm/^{o}F	+2.0
Doppler Coefficient	pcm/^{o}F	-1.0
Control Rod Worth at Trip	%$\Delta\rho$	-4.0

4.2.3.6.2 Results

The LOAC event initiated from the conditions given in Table 4.2.3.6-1, is presented in Figure 4.2.3.6-1 to Figure 4.2.3.6-5. The sequence of events is shown in Table 4.2.3.6-2.

After the reactor trip and the subsequent coastdown of the RCP's, energy removal from the reactor core is through natural circulation to the steam generators and is ultimately removed through the secondary safety valves. With an auxiliary feedwater flow of only 110 gpm to feed all three SG's, the primary system will not initially cooldown after the reactor trip due to the high level of decay heat. As shown in Figure 4.2.3.6-3, primary pressure is maintained at a high value and reaches the PORV setpoint approximately 400 seconds into the transient. The PORV's cycle until the auxiliary feedwater begins to cool the primary system. Pressurizer liquid level is shown on Figure 4.2.3.6-4. As shown, the maximum level occurs at 720 seconds after the initiation of the event. After that peak, liquid level decreases as secondary cooling becomes more effective as the operator acts to increase auxiliary feedwater flow.

The results of this transient show that for this bounding event, liquid is never released from the primary relief valves and that the primary system can be cooled through use of one auxiliary feedwater pump.

TABLE 4.2.3.6-2

SEQUENCE OF EVENTS

LOSS OF NON-EMERGENCY AC POWER

TURKEY POINT UNITS 3 & 4

EVENT	TIME (seconds)	SETPOINT OR VALUE
Loss of Non-emergency AC	0.0	---
Low SG Level Trip	52.0	10% NR
Rods Begin to Drop	54.0	---
RCP's Coastdown Initiation	58.0	---
Flow From one Auxiliary Feedwater pump started	232.0	110 gpm
Operator Realigns System to Increase Auxiliary Feedwater Flow	652.0	230 gpm
Peak Water Level in Pressurizer	720.0	1020 cubic feet
Feedwater Lines Purged and Cold Auxiliary Feedwater is Delivered	1095.0	---

FIGURE 4.2.3.6-1

Turkey Point Units 3 & 4 Loss of Non-Emergency AC Power

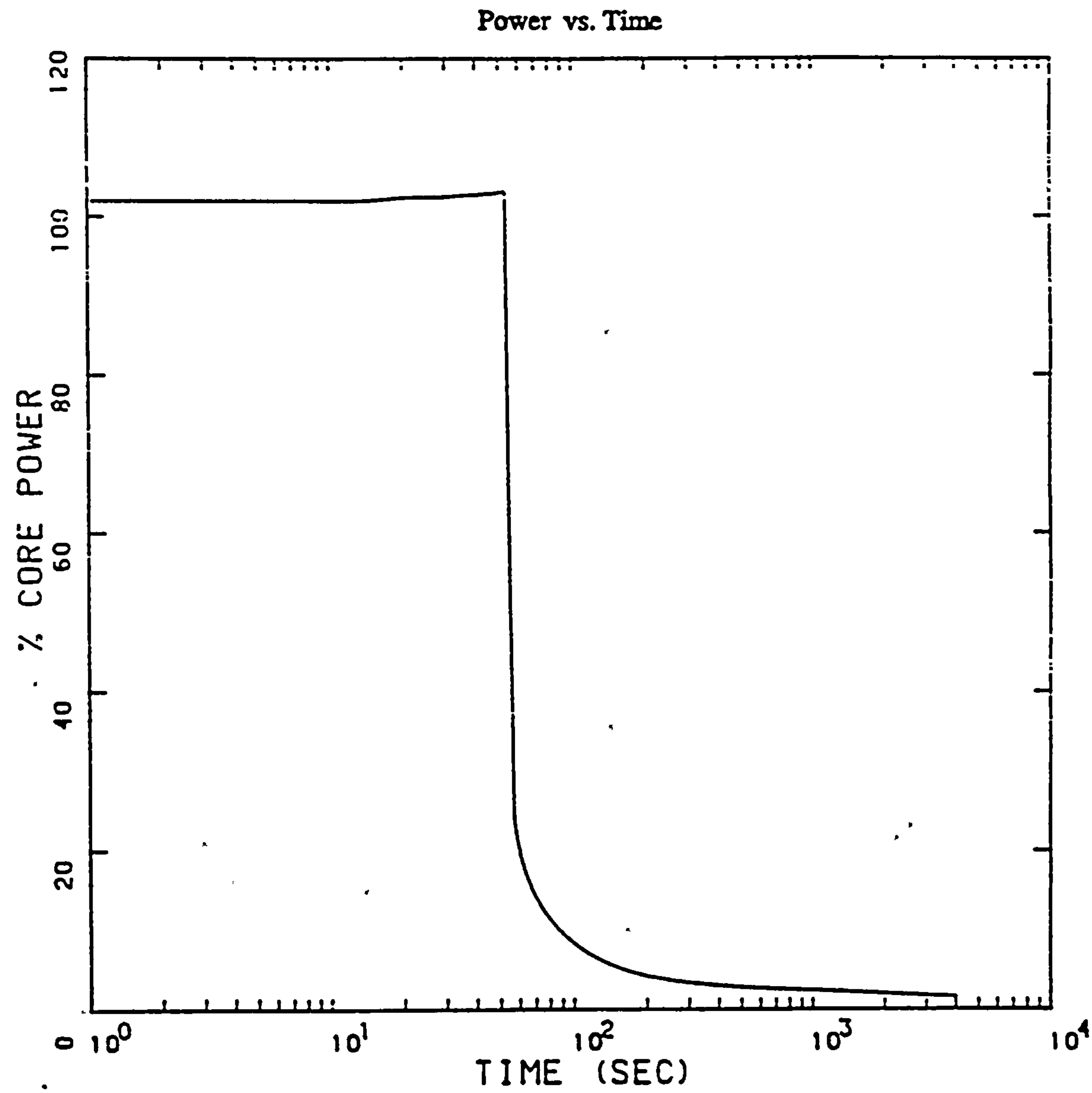


FIGURE 4.2.3.6-2

Turkey Point Units 3 & 4 Loss of Non-Emergency AC Power

Heat Flux vs. Time

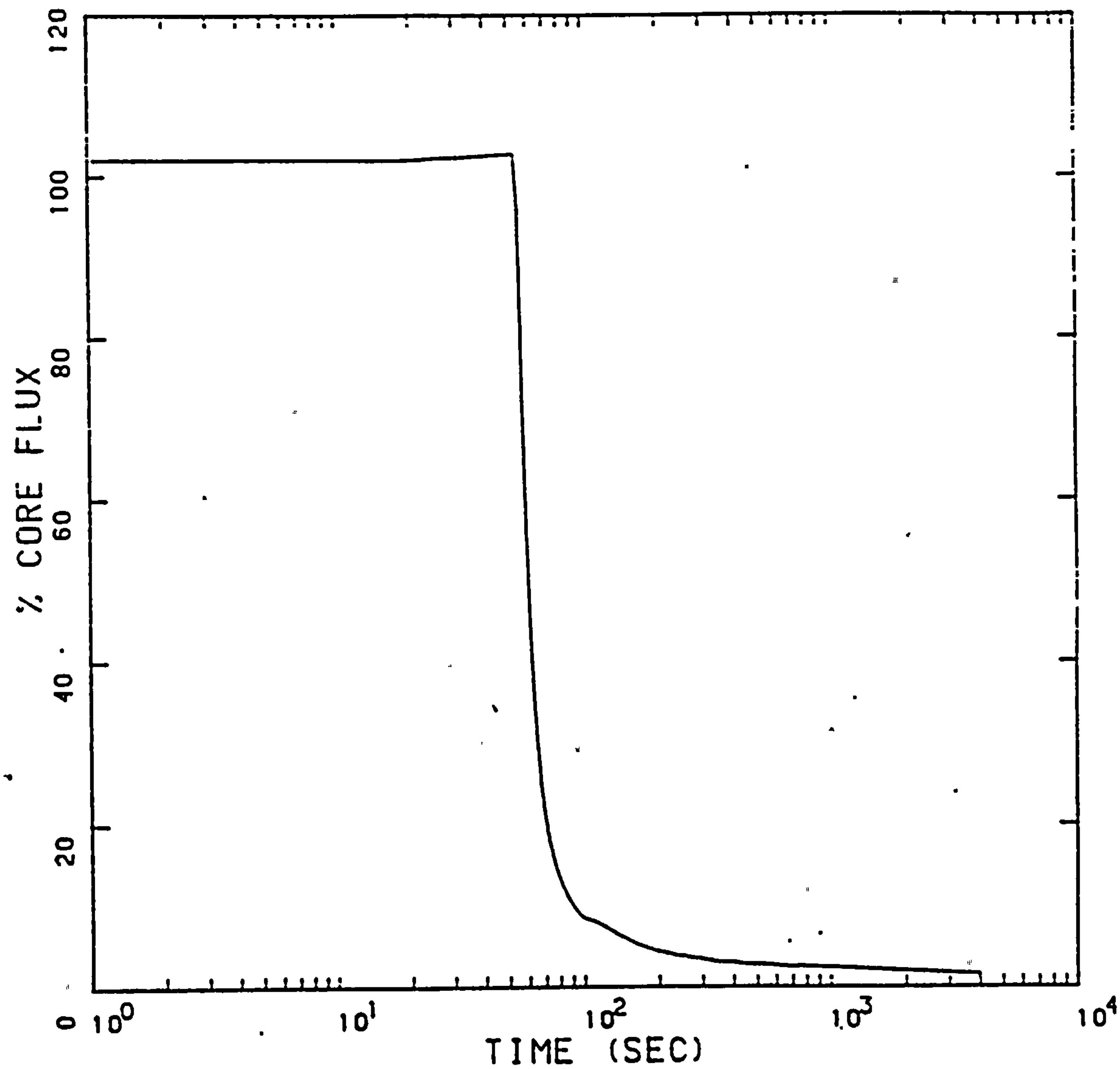


FIGURE 4.2.3.6-3

Turkey Point Units 3 & 4 Loss of Non-Emergency AC Power

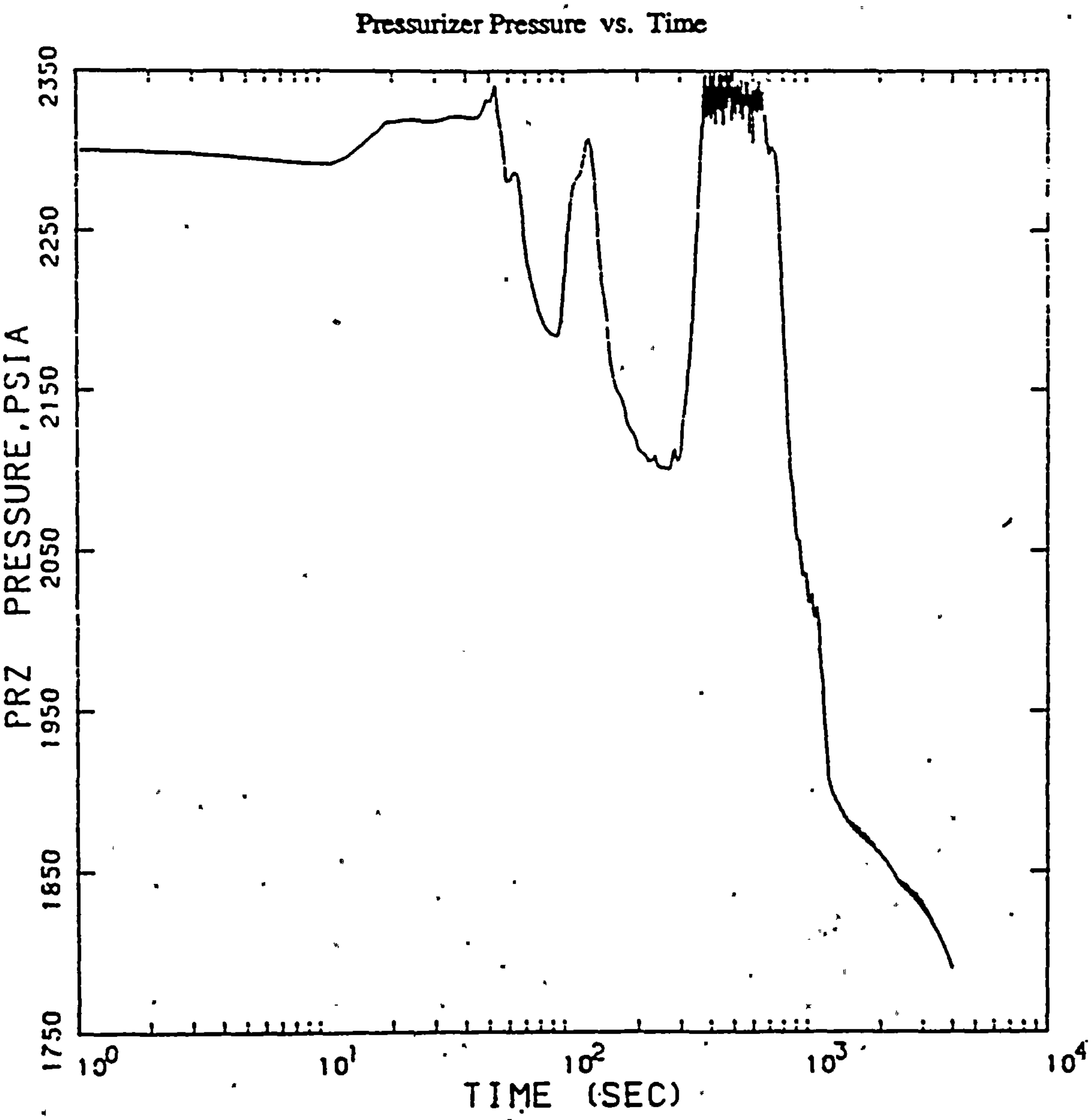


FIGURE 4.2.3.6-4

Turkey Point Units 3 & 4 Loss of Non-Emergency AC Power

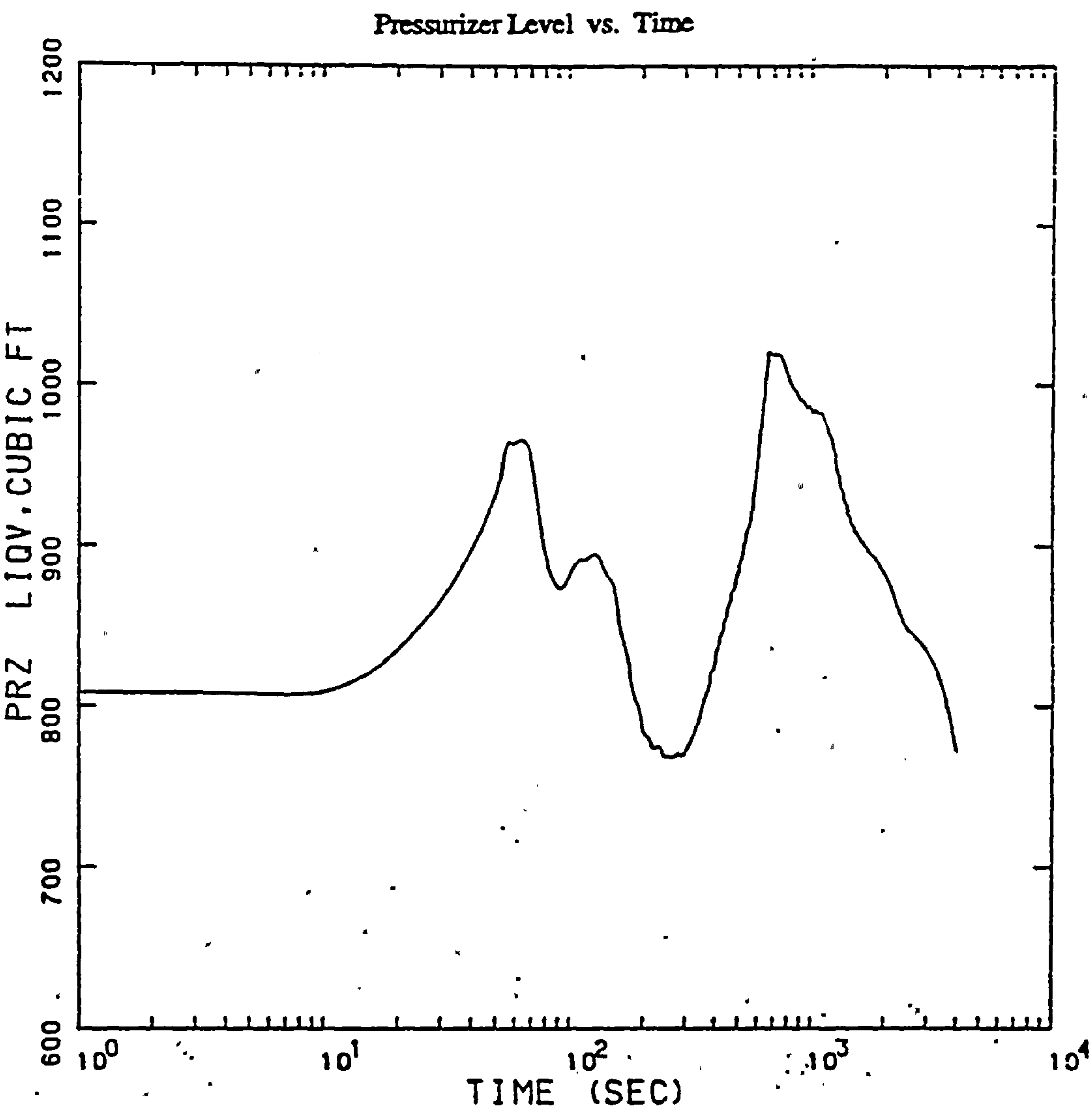
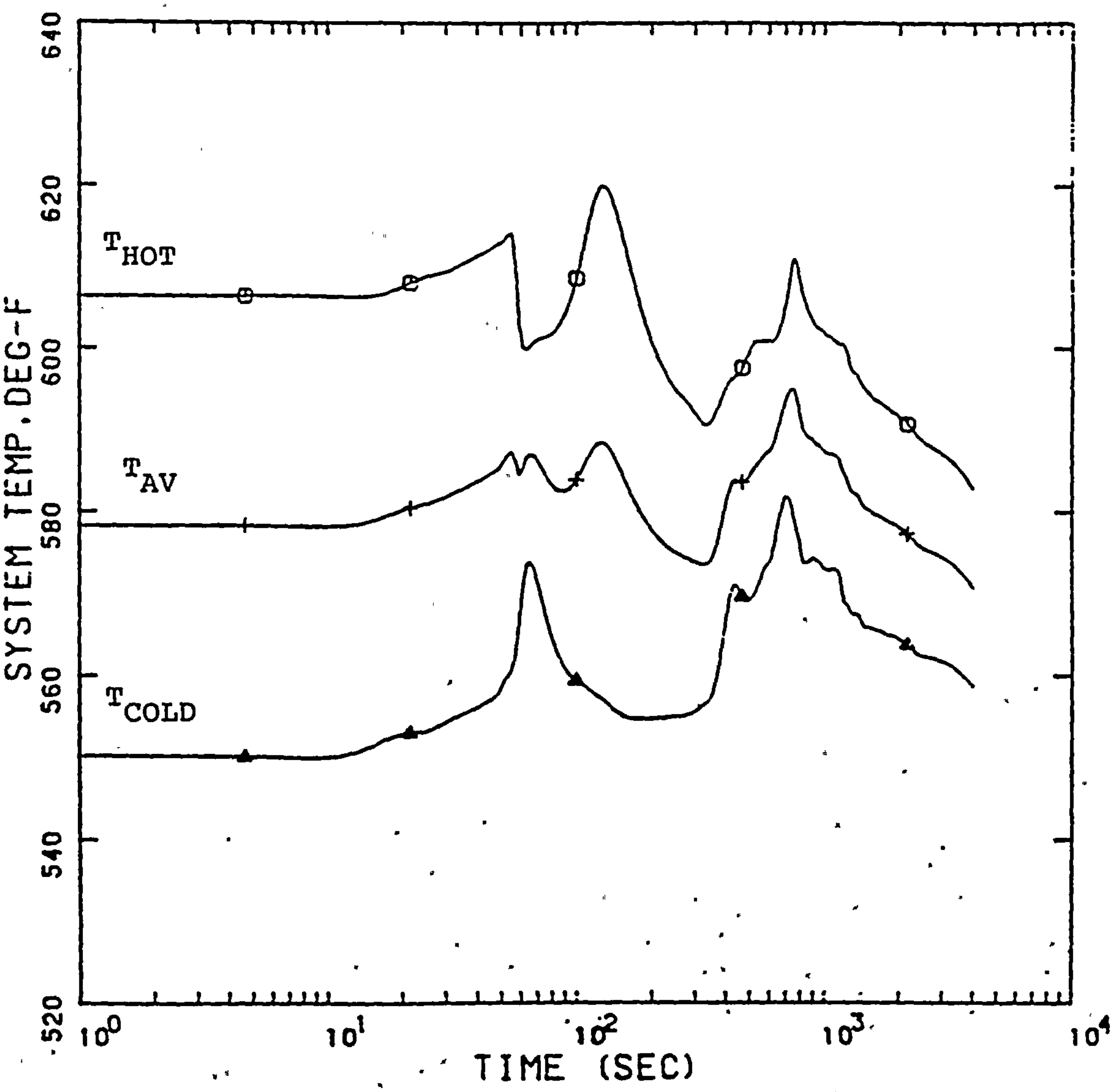


FIGURE 4.2.3.6-5

Turkey Point Units 3 & 4 Loss of Non-Emergency AC Power

RCS Temperatures vs. Time



4.2.3.7 Loss of Normal Feedwater Flow

The Loss of Normal Feedwater Flow event is defined as a reduction in feedwater flow to the steam generators when operating at power without a corresponding reduction in steam flow from the steam generators. The result of this mismatch is a reduction in the water inventory in the steam generators, which will cause a reduction in primary to secondary heat transfer and subsequent pressurization of the primary system.

The loss of normal feedwater flow event would be similar to the LOAC Power event described in the proceeding section, however, the auxiliary feedwater flow would be larger in this event since all the auxiliary feedwater would go to the affected unit rather than being split between the two units. Assuming a single failure of an auxiliary feedwater train, the minimum flow to the secondary side would be assumed as 315 gpm. The additional flow is sufficient to remove decay heat as well as the heat generated in the reactor coolant pumps without producing the amount of pressurizer level increase seen in the LOAC power event. Therefore, the results of the LOAC power will bound this event.

DNBR response during this transient would remain bounded by the Loss of Flow events described in Section 14.1.9 of the FSAR.

4.2.3.8 Feedwater Line Break

This event is a cooldown event in the licensing basis for Turkey Point Units 3 & 4. As such, the feedwater pipe break event is bounded by the steamline break event since the area for flow in a broken feedwater pipe is less than that of a severed steamline. The smaller area for flow results in lower steam relief rate which produces a more benign event.

[illegible]

5.0 Conclusions

FPL has presented in the preceding sections information on comparisons between the RETRAN computed results and plant operating data together with sensitivity studies for key parameters for two plant events. In addition, the licensing assumptions which will be utilized for events within the category of Decrease in Heat Removal by the Secondary System have been discussed in depth and the limiting transients for each of the units has been presented.

FPL believes that the information that the NRC requested in Reference 3 has been supplied for this one category of events. The work presented here has firmly tied the RETRAN model to actual plant operation through the comparison to two plant transients. It should also be noted that in the 1986 Topical Report, three other plant transients were presented in this category of event. This extensive plant benchmarking has demonstrated the adequacy of the FPL RETRAN modelling.

The methodology that will be used in future licensing analyses has been described in detail in Section 4.0. It is FPL's belief that the necessary conservatisms required for licensing applications have been utilized in the analysis of these limiting events.

FPL believes that this combination of a benchmarked, accurate RETRAN model, with conservative licensing assumptions provides sufficient basis for a determination that FPL can support licensing actions within the Decrease in Heat Removal from the Secondary System event category in a manner consistent with maintaining the health and safety of the public.

6.0 References

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