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Topical Report
RETRAN Code
Transient Analysis Model Qualification

Florida Power & Light Company
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ABSTRACT

Qualification results are presented to the NRC for the purpose of gaining NRC approval for Florida Power & Light's plant-specific models based on the RETRAN code for non-LOCA licensing support analysis. RETRAN analysis results, benchmarked to actual plant data, generic vendor analyses, and plant-specific FSAR analyses demonstrate the adequacy of plant modeling techniques and FPL staff proficiency as required by the NRC (NRC Generic Letter 83-11, Licensee Qualification for Performing Safety Analyses in Support of Licensing Activities).

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SUMMARY

Current NRC practice requires licensees to validate the computer code models utilized for safety analysis of their nuclear plants and to demonstrate that the Licensee's Staff has the ability to set up the input, execute the code and properly interpret the results (Ref. 2). This report presents the validation of the models and methods for performing non-LOCA system safety analyses in support of licensing submittals for Florida Power & Light Company's four nuclear plants, Turkey Point Unit 3, Turkey Point Unit 4, St. Lucie Unit 1 and St. Lucie Unit 2.

FPL's system simulation is performed with the RETRAN code (Ref. 9) developed by the Electric Power Research Institute (EPRI) and approved for non-LOCA analyses by the NRC (Ref. 10). Verification of RETRAN models has been performed by benchmarking RETRAN calculations to actual plant test data, and to analyses of anticipated operational occurrences (AOOs) documented in plant-specific FSARs. Analyses are presented encompassing all four of FPL's plants as well as a range of events which include a spectrum of non-LOCA design basis events, unusual occurrences at the plants and startup tests. The design basis events which include Main Steam Line Break, Loss of Load, Inadvertent Power Operated Relief Valve Opening, Loss of Forced Flow, Uncontrolled Rod Control Cluster Assembly Withdrawal and Control Element Assembly Drop are benchmarked against FSAR transients. Plant occurrences and tests are benchmarked against available plant data. The validation effort thus covers both licensing and

best estimate type of analyses. As shown by a comparison of results, there is good agreement between the RETRAN analyses and the benchmarks.

Also included are Turkey Point RETRAN analysis results of three overcooling transients performed in support of the pressurized thermal shock issue resolution. These transients demonstrate FPL's capability of performing analyses to meet specific regulatory or licensing requirements.

The spectrum of analyses and benchmarks, ranging over four plants and a variety of transient events, validates the FPL, RETRAN based, system analysis models and also demonstrates the ability of FPL's staff to perform non-LOCA system safety analyses in accordance with NRC requirements and industry practices.

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LIST OF ACRONYMS

AFW.	AUXILIARY FEEDWATER
AOO	ANTICIPATED OPERATIONAL OCCURRENCE
CEA	CONTROL ELEMENT ASSEMBLY
DNB	DEPARTURE FROM NUCLEATE BOILING
ECCS	EMERGENCY CORE COOLING SYSTEM
EPRI	ELECTRIC POWER RESEARCH INSTITUTE
FPL	FLORIDA POWER & LIGHT COMPANY
FSAR	FINAL SAFETY ANALYSIS REPORT
HPSI	HIGH PRESSURE SAFETY INJECTION
INPO	INSTITUTE OF NUCLEAR POWER OPERATION
LCO	LIMITING CONDITIONS FOR OPERATION
LOAC	LOSS OF OFFSITE POWER
LOCA	LOSS OF COOLANT ACCIDENT
MFW	MAIN FEEDWATER
MSIV	MAIN STEAM ISOLATION VALVE
NSSS	NUCLEAR STEAM SUPPLY SYSTEM
NRC	NUCLEAR REGULATORY COMMISSION
PORV	POWER OPERATED RELIEF VALVE
PTS	PRESSURIZED THERMAL SHOCK
QA	QUALITY ASSURANCE
RCCA	ROD CLUSTER CONTROL ASSEMBLY
RCS	REACTOR COOLANT SYSTEM
RCP	REACTOR COOLANT PUMPS
SER	SAFETY EVALUATION REPORT

SG	STEAM GENERATOR
SGTR	STEAM GENERATOR TUBE RUPTURE
SI	SAFETY INJECTION
SIAS	SAFETY INJECTION ACTUATION SIGNAL
SRV	SAFETY RELIEF VALVE
TM/LP	THERMAL MARGIN/LOW PRESSURE
RWST	REFUELING WATER STORAGE TANK

1.0 INTRODUCTION

1.1 Background

Ever since FPL's first nuclear plant, Turkey Point Unit 3, came on line in 1972, and more so now with four nuclear plants in operation, there has been a requirement for reanalysis of those accidents that form the design basis of the operating plants. Among the major sources of analytical demand have been the issue of upper head voiding for the St. Lucie Unit 1 plant, the steam generator degradation of the Turkey Point units and the pressurized thermal shock (PTS) issue that arose in connection with the Turkey Point reactor vessels.

The increasing requirement for plant analytical support is further complicated when the fuel supplier is not the original designer of the plant, as is the case with the current fuel loadings of St. Lucie Unit 1. The models of two suppliers may differ substantially. In this case it is desirable for the utility to have their model in place, so that they can evaluate the conservatism of the vendor's model. The plant specific model allows timely evaluation of plant modifications, design changes, and special concerns.

The ultimate objective for developing and maintaining the plant models in accordance with current requirements and state-of-the-art techniques is to develop full reload capability. By developing full reload capability, the analytical basis supporting plant operation, including physics parameters, fuel management, and safety analysis is contained in a unified methodology package.

This volume addresses FPL's qualifications for performing plant system simulation by means of the RETRAN computer code (Appendix A) which has been reviewed and approved by the NRC for non-LOCA analysis. A subsequent report will cover FPL's capabilities in the area of core thermal-hydraulic analysis. The FPL lattice physics methodology has been described in a previous report (Ref. 1). Additional physics reports are to be submitted at a later date.

1.2 Objective

The objective of this topical report is to present, as qualification basis, RETRAN base model verification results for each of FPL's nuclear plants. It is requested that these models be approved by the NRC for non-LOCA licensing support analysis. RETRAN analyses ranging over a broad spectrum of transients are benchmarked to plant data, FSAR analyses, and generic vendor analyses to demonstrate the adequacy of these RETRAN base models for non-LOCA licensing analyses.

These analyses also demonstrate the proficiency of FPL personnel to perform credible system safety analyses, thereby satisfying NRC requirements documented in their letter on Qualification For Performing Safety Analyses in Support of Licensing Actions (Ref. 2).

1.3 Florida Power & Light Company's Nuclear Plants

Florida Power & Light Co. operates four nuclear power plants: Turkey Point Units 3 and 4, and St. Lucie Units 1 and 2. The Turkey Point units are geometrically similar, three-loop, Westinghouse built pressurized water reactors. Both units are licensed to operate at 2200 MW (thermal). Each primary coolant loop consists of a reactor coolant pump and a steam generator. The reactor core is composed of 157, 15X15 fuel assemblies with an active core height of 144 in.

The St. Lucie units are Combustion Engineering two-loop pressurized water reactors and, except for the fuel, are geometrically similar to each other. Each loop contains a steam generator and two reactor coolant pumps. St. Lucie Unit 1 is licensed to operate at 2700 MW (thermal). St. Lucie Unit 2 had been licensed at 2560 MW (thermal) until early 1985 when it was uprated to 2700 MW (thermal). Both reactor cores consist of 217 fuel assemblies with St. Lucie Unit 1 fuel rods arranged in a 14X14 matrix and St. Lucie Unit 2 fuel in a 16X16 matrix.

The principal characterics of these plants are summarized in Table 1.3.1.

1.4 FPL Modeling and Analysis Experience

FPL has gained considerable experience since the Mid1970's performing plant system simulations. Some of this early experience, principally based on the DYNODE code, (Ref. 12) has been documented in a topical report submitted to the NRC in 1978 (Ref. 3). The following reflects the breadth of FPL application experience:

1. RETRAN CODE VERIFICATION. Starting in 1977, FPL participated in the RETRAN-01 code verification effort at the request of the Electric Power Research Institute (EPRI). This work culminated with the documentation of the following transients benchmarked to Turkey Point data in the RETRAN Code Manual (Ref. 4): uncontrolled rod withdrawal, loss of flow, and RCS pump coastdown.

2. RESOLUTION OF SAFETY CONCERNS. Certain analyses required to meet specific regulatory or licensing requirements, such as those for the pressurized thermal shock (PTS) issue resolution, have been performed at FPL. Overcooling transients were analyzed with RETRAN to determine thermal-hydraulic boundary conditions for

thermal stress and fracture mechanics computer codes as a part of FPL's effort to address the PTS concern for the Turkey Point reactor vessels. Analysis results for some of these overcooling transients, compared to similar results for generic Westinghouse plants, are presented in section 7.0 of this report.

Another example of this type of analysis is the development at FPL of natural circulation cooldown curves for St. Lucie Unit 1 in response to NRC requirements to prevent boiling in the upper head of the reactor vessel following the St. Lucie Unit 1 Natural Circulation Cooldown incident of June 11, 1980. The analysis was found acceptable by the NRC (Ref. 8). Validation of the RETRAN model for this application was accomplished by simulating the incident and comparing results to plant recordings. The results of this simulation are presented in Section 2.1 of this report.

3. OPERATIONAL FLEXIBILITY IMPROVEMENTS. In 1983 a study was performed at FPL to determine the optimum pressurizer level setpoints for St. Lucie Unit 1 which would prevent uncovering the pressurizer heaters during transients.

Another example is the development of an algorithm using data calculated with RETRAN for converting measured

steam generator pressure drop into loop flow rate for St. Lucie Unit 1.

4. OPERATOR TRAINING. Best-estimate predictions of recorded plant parameters during transients have been performed to support operator training at the Turkey Point plant. An example of this type of work is a recent study to determine the time available for an operator to reduce the feedwater flow to match a turbine runback on a spurious control rod drop signal. Considered in this study were various combinations of control rod motion, availability of PORVs and steam dumps and moderator temperature coefficients.

5. VENDOR ANALYSIS EVALUATION. Check-type calculations have been made to assist in evaluating analysis results generated by vendors. FPL performed analyses of limiting transients such as Loss of Load, Main Steam Line Break and Loss of Coolant Flow for St. Lucie Unit 1 Cycle 6 to verify vendor (Exxon Nuclear Company) calculations.

1.5 Quality Assurance

The FPL Quality Assurance Program encompasses the full spectrum of activities involved in performing safety analyses. Requirements and responsibilities for the quality program have been delineated in the FPL Quality Assurance Report (Ref. 11)

submitted to the NRC and in internal FPL QA Manuals and Procedures. Implementation of these requirements are provided in detailed Quality Instructions. The following highlights some aspects of the quality program as it relates to safety analyses.

CONFIGURATION CONTROL - FPL analyses are performed with EPRI released versions of RETRAN. Changes to the code are precluded because only the "read only" compiled listing of the code can be accessed by a user. Desired changes to the code (error corrections, etc.) are currently made by EPRI.

Proper installation of the RETRAN code on the FPL IBM Computer System is verified by running the EPRI supplied test problems. The results of the test problems are compared to validated solutions to assure that the code has been properly installed.

RETRAN BASE MODEL - QA procedures were followed during the construction and verification of each plant's base model. Changes to the base models are controlled and verified. "Read only" copies of the base models are available for general use. Transient-specific changes to a base model are made by appending them to a copy of the base model.

CALCULATIONS - A calculation notebook is maintained for all safety related analyses in accordance with written instructions approved by the FPL QA Department. Calculations are checked

and verified by an independent reviewer and the records are maintained in accordance with established procedures. All QA related activities are audited periodically by the FPL QA Department.

1.6 RETRAN Base Model Formation and Verification

Three RETRAN base models (described in Appendix B), one for Turkey Point Units 3 and 4, one for St. Lucie Unit 1, and one for St. Lucie Unit 2 have been developed and are maintained by FPL. The development, modification, verification and maintenance of each model is in accordance with the FPL Quality Assurance program described in the previous section.

The RETRAN base models described in Appendix B have been developed as "best estimate" models with component configuration, system interaction and operating parameters simulating actual design and operating conditions as closely as possible. The models simulate the action of safety and non-safety grade equipment and do not account for effects of instrument uncertainties or measurement errors. Best estimate values of neutronics parameters, such as moderator temperature coefficient, Doppler coefficient and scram reactivity worths and best estimate values of heat transfer parameters, such as primary to secondary side heat transfer coefficients and clad to fuel gap heat transfer coefficients were used. The best estimate models have been utilized for the simulation of plant tests and

plant events where RETRAN analysis results have been compared with available plant data.

For the analysis of design basis events, where comparisons have been made with FSAR accident analysis results, the best estimate models were converted into licensing models. Conservatively bounding values were assigned to the operating parameters, such as core power, reactor coolant flow, temperatures, pressures, moderator temperature coefficients and scram reactivity worths. Only safety grade engineered safeguard systems were assumed to actuate during these events. For example, in the case of a Loss of Load transient, it was assumed that the turbine bypass valves do not open on demand and that the steam generator pressure builds up until the safety valves lift. The initial conditions, key parameters and safety systems that were assumed to operate during each of the transients are tabulated in the appropriate sections.

Based on results obtained by Northeast Utilities with their RETRAN models (Ref. 18), the effect of varying such model parameters as number of secondary side volumes, primary side steam generator tube noding, bubble rise velocity in the steam generator, blowdown back pressure, pump inertia and junction inertia on results mostly is minimal. Therefore, only a few sensitivity studies were conducted for the benchmark investigations reported here. Reasonable noding and parameter values were utilized. For analyses leading to actual licensing

submittals extensive sensitivity studies would be conducted to assure that results obtained are in the conservative direction.

Benchmarked RETRAN analyses of transients, divided into five major categories according to the initiating system response, are presented in the following sections.

1. INCREASE IN SECONDARY COOLANT SYSTEM HEAT REMOVAL

   St. Lucie Unit 1 Natural Circulation Cooldown Event

   St. Lucie Unit 1 Main Steam Line Break

2. DECREASE IN SECONDARY COOLANT SYSTEM HEAT REMOVAL

   Turkey Point Loss of One Main Feedwater Pump Event

   St. Lucie Unit 1 Loss of Load

   St. Lucie Unit 2 Generator Trip Test

   St. Lucie Unit 1 Main Steam Isolation Valve (MSIV) Closure Event

3. DECREASE IN PRIMARY COOLANT SYSTEM INVENTORY

St. Lucie Unit 2 Inadvertent Power Operated Relief Valve (PORV) Opening

4. LOSS OF REACTOR COOLANT SYSTEM (RCS) FLOW

Turkey Point Pump Coastdown Test

St. Lucie Unit 1 Loss of Forced Flow

5. REACTIVITY INSERTION

Turkey Point Uncontrolled Rod Control Cluster Assembly (RCCA) Withdrawal

St. Lucie Unit 1 Control Element Assembly (CEA) Drop

Also included are RETRAN analyses of three overcooling transients performed in support of the resolution of the pressurized thermal shock (PTS) issue for Turkey Point.

1. Turkey Point Small Break Loss of Coolant Accident (LOCA)

2. Turkey Point Stuck Open Steam Generator Relief Valve

3. Turkey Point Steam Generator Tube Rupture

Where possible the results of these analyses are compared to those of generic analyses performed by the Westinghouse Owners Group.* These transients demonstrate FPL's ability to perform analyses to meet specific regulatory or licensing requirements.

As demonstrated by the matrix shown in Table 1.6.1, the above RETRAN analyses collectively exercise all major components of the RETRAN models.

Presented in the appendices of this report are brief descriptions of the RETRAN computer code and the RETRAN base models.

*Generic analyses are performed for a particular plant, but results are generally applicable to other plants of similar design and operational characteristics.

TABLE 1.3.1

CHARACTERISTICS OF FPL'S NUCLEAR PLANTS

	TURKEY POINT UNITS 3 & 4	ST. LUCIE UNIT 1	ST. LUCIE UNIT 2
Core Power, MW (Thermal)	2200	2700	2560*
Nominal RCS Pressure, psia	2250	2250	2250
Nominal RCS Flowrate, Mlbm/hr	101.5	139.4	136.6
Core Inlet Temperature, °F	547	549	549
Steam Generator Pressure, psia	840	900	815
Number of Fuel Assemblies	157	217	217
Fuel Rod Matrix	15 X 15	14 X 14	16 X 16

*CORE POWER HAS BEEN INCREASED TO 2700 MW IN 1985, BUT ANALYSES PRESENTED HERE WERE PERFORMED PREVIOUSLY, AT LOWER POWER LEVEL.

Table 1.6.1 Actuation of RETRAN Model Components

	2.1 Natural Circulation Cooldown Event	2.2 Main Steam Line Break	3.1 Loss of One Feedwater Pump	3.2 Loss of Load	3.3 Generator Trip Test	3.4 MSIV Closure Event	4.1 Opening of PORVs	5.1 Pump Coastdown Test	5.2 Loss of Forced Flow	6.1 Uncontrolled RCCA Withdrawal	6.2 CEA Drop	7.1 Small Break LOCA	7.2 Open S.G. Relief Valve	7.3 S.G. Tube Rupture
Pressurizer Model	X	X	X		X	X	X			X	X	X	X	X
Pressurizer Level	X		X	X	X	X					X		X	X
Pressurizer Relief			X	X		X	X						X	
Primary Pressure & Temperature Response	X	X	X	X	X	X	X		X	X	X	X	X	X
Reactor Core Kinetics		X				X				X	X			
Reactivity Feedback		X	X			X				X	X			
Scram Worth & Insertion Rate			X	X	X	X			X	X	X			
RCS Pump Characteristics & Flow Resistance	X					[illegible]		X	X					
Reactor Protective System		X	X	X	X	X	X		X	X	X	X	X	X
Secondary Side Pressure & Temperature Response		X	X	X	X	X	X							
Secondary Side Pressure & Mass Relief			X	X	X		X					X		X
Steam Generator Level		X	X			X						X		X
Primary to Secondary Side Heat Transfer	X	X	X	X	X	X	X					X	X	X
Auxiliary Feedwater Actuation	X	X	X	X	X		X					X	X	X
Safety Injection		X					X					X	X	X
Decay Heat		X		X								X	X	X
Break Flow		X				[illegible]						X	X	X
Two Phase Flow	X											X		X
Upper Head Voiding	X													
Fuel Rod Heat Flux		X							X					
MSIV & Feedwater Isolation	X	X		X			X					X	X	X

2.0 INCREASE IN SECONDARY COOLANT SYSTEM HEAT REMOVAL

Events in this category result in a core coolant temperature decrease in response to an increase in heat removal by the secondary coolant system. Due to a negative moderator temperature coefficient, reactivity and therefore core power increases as the core inlet temperature decreases. The important modeling consideration is core reactivity change due to moderator and Doppler reactivity coefficients, and the secondary coolant system energy removal rate. The St. Lucie Unit 1 Main Steam Line Break Accident (Section 2.2) benchmarked to St. Lucie Unit 1 FSAR analysis results is presented in this category.

Cooldown of the RCS due to an increase in secondary coolant system heat removal may cause voiding of the fluid in the upper head of the reactor vessel if the fluid pressure is reduced below the saturation pressure. This condition occurred at St. Lucie Unit 1 on August 11, 1980 when the loss of component cooling water to the seals of the reactor coolant pumps (RCPs) necessitated tripping the reactor, stopping the RCPs, and attempting to bring the plant to cold shutdown by removing heat through the atmospheric dump valves (ADVs). Stored heat in the reactor vessel structures coupled with stagnation of fluid in the upper head during the natural circulation cooldown caused the upper head fluid to be hotter than the saturation temperature corresponding to the pressure in the region.

This resulted in void formation in the upper head. This event was simulated and benchmarked to measured plant parameters (Section 2.1) to assess modeling of the reactor vessel structures, heat transfer and flow circulation in the upper head region for the St. Lucie Unit 1 RETRAN base model. This benchmark served to demonstrate the adequacy of modeling such structures should they be needed in any of FPL's RETRAN base models.

2.1 St. Lucie Unit 1 Natural Circulation Cooldown Event

2.1.1 Transient Description

The St. Lucie Unit 1 Natural Circulation Cooldown Event occurred on August 11, 1980 (Ref. 6). An electrical failure caused a component cooling water isolation valve to shut off cooling water flow to the seals on all four reactor coolant pumps necessitating shutdown of the pumps to protect the pump seals. The reactor was cooled down with the steam generator atmospheric dump system with natural circulation providing reactor coolant flow. As the reactor was cooled down, steam formation in the upper head region coupled with charging system flow injection alternating between the normal cold leg connection and pressurizer spray, caused the pressurizer level and pressure to fluctuate. This procedure was continued until the reactor vessel upper head was cooled below the saturation

temperature of the fluid in this region.

Simulation of this event challenges, in particular, RETRAN's ability to calculate transient responses of long duration. This transient lasted approximately 4 hours before void formation occurred in the upper head and 9.5 hours before the anomolous pressurizer behavior ceased. Other noteworthy areas of code validation that are provided by simulating this event are the code's ability to calculate upper head flow stagnation under natural circulation conditions and the onset of void formation in the upper head during a cooldown. Accurate calculation of void formation in the upper head has led to the development of operating procedures which would prevent this occurrence in the future.

2.1.2 RETRAN Analysis Description

FPL simulated the transient with RETRAN01 to assess the modeling of the vessel upper head before using it to develop procedures for cooling down on natural circulation without upper head voiding (Ref. 8). (RETRAN01 was the only version of the code available when this analysis was performed). A simplified RETRAN01 model was developed which was fast-running, yet detailed enough to calculate upper head fluid temperature during natural circulation, and heat transfer from reactor vessel structures. A single reactor coolant system loop was adequate to simulate the symmetric cooldown of the RCS with the steam

generators. All manual operator actions, such as the operation of the atmospheric dump system to cool the reactor coolant system, were modeled. The RETRAN01 model nodalization diagram is shown in Figure 2.1.1 and a description of the control volumes, junctions, and heat slabs are presented, respectively, in Tables 2.1.1, 2.1.2 and 2.1.3.

Table 2.1.4 presents the initial conditions at the time of reactor trip. Table 2.1.5 presents the status of safety systems in the RETRAN model assumed for this analysis. The analysis is a prime example of a best estimate type of analysis where system action and system parameters were modeled to represent actual operating conditions as closely as possible.

2.1.3 Results

Figure 2.1.2 presents the measured and calculated pressurizer level (percent of indicated range). The calculated level agreed well with measured data for 3.5 hours after scram. This is an indication that changes in the volumetrically averaged RCS fluid density due to fluid temperature changes were reasonably calculated. The calculated and measured pressurizer pressures are presented in Figure 2.1.3. Cooling of the fluid in the pressurizer, coupled with level changes due to shrinkage of RCS fluid as it is cooled, cause the pressure to decrease. Good agreement was obtained between the measured and calculated data.

Flow stagnation and heat transfer from structures in the reactor vessel upper head caused the cooldown of fluid in this region to be less than that in the hot legs. As a result of cooling the fluid in the pressurizer substantially below that in the upper head, void formation was calculated to occur at approximately 3.5 hours after scram. The calculated onset. of void formation occurred approximately 1/2 hour before this was indicated by data. Voiding in the upper head caused fluid in this region to be displaced and consequently the pressurizer level to increase.

The onset of voiding in the upper head was calculated to occur sooner than indicated by the data because coolant in the upper head remained hotter. This is affected primarily by the amount of coolant circulation in the upper head during natural circulation assumed in the model.

Good overall agreement of trends and magnitudes of RETRAN calculation with plant data indicates that relevant phenomena leading to upper head voiding during a natural circulation cooldown were simulated. In addition, because the model assumptions result in calculating voiding in the upper head sooner than would actually occur in the plant, it is concluded that this model would conservatively establish operational limitations to prevent upper head voiding.

2.2 St. Lucie Unit 1 Main Steamline Break

2.2.1 Transient Description

The main steam line break (MSLB) event has been simulated for St. Lucie Unit 1 with the RETRAN02 computer code. The results have been benchmarked to a similar calculation described in the St. Lucie Unit 1 FSAR (Ref. 5) performed with the Combustion Engineering CESEC computer code. This calculation serves to check the modeling of the reactor kinetics, the break energy removal rate and the primary to secondary heat transfer mechanism.

2.2.2 RETRAN Analysis Description

This analysis was performed with the St. Lucie Unit 1 RETRAN Base Model (Appendix B). Modifications to the model were made to incorporate the break, and the end-of-cycle 4 reactor physics parameters and to achieve a conservative, licensing type, model.

Table 2.2.1 delineates the initial conditions and key parameters incorporated in the RETRAN model. These parameters were taken to be identical to the values given in the FSAR. The status of the plant safety systems in the RETRAN model is summarized in Table 2.2.2. The table shows which of the modeled safety systems actuated during the transient and which did not. Systems that were not modeled for this transient are also identified.

The scram worth assumed that the most reactive control element assembly (CEA) was stuck fully withdrawn. The reactivity insertion due to accumulation of boron, from the FSAR analysis was input as a function of time. The failure of one high pressure safety injection (HPSI) pump was conservatively assumed in this analysis. Safety injection (SI) flow was delayed an additional 30 seconds after receipt of the safety injection actuation signal (SIAS) to account for the time required to bring the pumps to full speed. The RCPs were manually tripped after receipt of the SIAS.

The break was modeled as a double ended guillotine rupture of the main steam line, located between the steam generator exit nozzle and an MSIV. This resulted in the greatest rate and magnitude of temperature reduction in the reactor core region. The break flow was based on the Henry-Fauske/isoenthalpic expansion critical flow models with a discharge coefficient of 1.0.

2.2.3 Results

Figure 2.2.1 shows the steam generator secondary pressure response. Good agreement with the steam generator pressure response presented in the FSAR was obtained. This indicated that the break mass and energy discharge rates were adequately modeled.

The pressurizer pressure (Figure 2.2.2) was in good agreement with the FSAR results.

The core inlet coolant temperature (Figure 2.2.3) was in fairly good agreement with the FSAR results, though the cooldown was slightly underpredicted for most of the transient. Because the cooldown of the RCS for RETRAN was slightly less than that for the FSAR results, the positive reactivity insertion due to moderator temperature feedback effects was less and total reactivity was somewhat less (Figure 2.2.4) than for the FSAR calculation. As a result, the core power (Figure 2.2.5) calculated by RETRAN was less than that in the FSAR.

The RETRAN analysis showed no return to power while the FSAR analysis showed a peak power of about 1% at 500 seconds.

The agreement of the timing of the sequence of events between RETRAN analysis and FSAR results (Table 2.2.3) is excellent.

The RETRAN analysis was continued for 600 seconds wl.ich corresponds to the range of data presented in the FSAR.

TABLE 2.1.1

ST. LUCIE 1 RETRAN01 MODEL VOLUME DESCRIPTION

FLUID VOLUME	DESCRIPTION
1	Combined hot leg volume.
2	Combined steam generator inlet plenum volume.
3	Combined steam generator tube volume from tube sheet to top of tube bundle.
4	Combined steam generator tube volume from top of tube bundle to tube sheet.
5	Combined steam generator outlet plenum volume.
6	Combined cold leg volume upstream of reactor coolant pump.
7	Combined reactor coolant pump volume.
8	Combined cold leg volume downstream of reactor coolant pump.
9	Reactor vessel downcomer volume.
10	Reactor vessel inlet plenum volume.
11	Core volume.
16	Volume from top of active core to fuel alignment plate.
12	Outlet plenum volume from fuel alignment plate to upper guide structure support plate.
13	CEA shroud volume.
17	Upper head volume from upper guide structure support plate to top of CEA shrouds.
14	Upper head volume above top of CEA shrouds.
32	Surge line volume.
34	Pressurizer volume.
51	Steam generator shell side volume.

TABLE 2.1.2

ST. LUCIE 1 RETRAN01 MODEL JUNCTION DESCRIPTION

FLOW JUNCTION	DESCRIPTION
1	Flow from volume 12 to volume 1.
2	Flow from volume 1 to volume 2.
3	Flow from volume 2 to volume 3.
4	Flow from volume 3 to volume 4.
5	Flow from volume 4 to volume 5.
6	Flow from volume 5 to volume 6.
7	Flow from volume 6 to volume 7.
8	Flow from volume 7 to volume 8.
9	Flow from volume 8 to volume 9.
10	Flow from volume 9 to volume 10.
11	Flow from volume 10 to volume 11.
17	Flow from volume 11 to volume 16.
12	Flow from volume 16 to volume 12.
13	Flow from volume 16 to volume 13.
14	Flow from volume 13 to volume 17.
15	Flow from volume 17 to volume 12.
18	Flow from volume 14 to volume 17.
35	Flow from volume 32 to volume 1.
36	Flow from volume 34 to volume 32.
37	Spray flow to volume 34.
38	Charging flow to volume 8.
39	Letdown flow from volume 6.

TABLE 2.1.2 (CONTINUED)

FLOW JUNCTION	DESCRIPTION
81	Feedwater flow to volume 51.
82	Atmospheric relief valve flow from volume 51.
83	Steam bypass valve flow to condenser from volume 51.
84	Steam dump valve flow to condenser from volume 51.
91	Steam flow to turbine from volume 51.

TABLE 2.1.3

ST. LUCIE 1 RETRAN01 MODEL HEAT CONDUCTOR DESCRIPTION

HEAT CONDUCTOR	DESCRIPTION
1	Fuel conductor connecting fuel to volume 11.
2	Steam generator tubes connecting volume 3 and volume 51.
3	Steam generator tubes connecting volume 4 and volume 51.
4	Metal in reactor vessel walls adjacent to volume 14.
5	Metal associated with upperhead drive shafts in volume 14.
6	Metal associated with CEA shrouds connecting volume 13 and volume 17.
7	Metal associated with CEA shrouds connecting volume 13 and volume 12.
8	Metal associated with CEA shrouds connecting volume 13 and volume 12.
9	Upper guide structure support plate connecting volume 17 and volume 12.
10	Metal associated with upper guide structure adjacent to volume 12.
11	Metal in reactor vessel wall adjacent to volume 17.
12	An effective conductor to allow axial heat conduction between volume 14 and volume 17.

TABLE 2.1.4

INITIAL AND BOUNDARY CONDITIONS

NATURAL CIRCULATION COOLDOWN EVENT

PARAMETER	VALUE
Core Power, MW (Thermal)	2560
Core Inlet Coolant Temperature, °F	539
Core Outlet Coolant Temperature, °F	588
Vessel Mass Flow Rate, 10^6 lbm/hr	136.8
Pressurizer Pressure, psia	2237
Steam Generator Pressure, psia	800

TABLE 2.1.5

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

NATURAL CIRCULATION COOLDOWN EVENT

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves			X
Pressurizer Safety Valves			X
Main Steam Isolation	X		
Main Feedwater Isolation	X		
Auxiliary Feedwater System	X		
Safety Injection System			
HPSI		X	
Accumulators			X
LPSI			X
Atmospheric Dump Valve Systems	X		
Steam Dump and Bypass System	X		
Pressurizer Level Control System	X		
Pressurizer Power Operated Relief Valves (PORV)			X
Chemical and Volume Control System	X		
S. G. Level Control System			X
Automatic Rod Motion			X

TABLE 2.2.1

INITIAL CONDITIONS AND KEY PARAMETERS

MAIN STEAM LINE BREAK

PARAMETER	VALUE
Core Power, MW (Thermal)	1
Core Inlet Coolant Temperature, °F	532
Pressurizer Pressure, psia	2300
Steam Generator Pressure, psia	900
Minimum CEA Worth Available at Trip, % Δk	-5

TABLE 2.2.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

MAIN STEAM LINE BREAK

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System		X	
Main Steam Safety Valves	X		
Pressurizer Safety Valves		X	
Main Steam Isolation	X		
Main Feedwater Isolation	X		
Auxiliary Feedwater System	X		
Safety Injection System			
HPSI	X		
Accumulators			X
LPSI	X		
Atmospheric Dump Valve Systems			X
Steam Dump and Bypass System			X
Pressurizer Level Control System			X
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System			X
S. G. Level Control System			X

TABLE 2.2.3

SEQUENCE OF EVENTS

MAIN STEAM LINE BREAK

EVENT	TIME(S) FSAR	RETRAN	SETPOINT
Steam Line Rupture	0.0	0.0	-
Low Steam Generator Press. Signal	3.8	2.8	578 psia
Main Steam Isolation Signal	3.8	2.8	578 psia
MSIV's Begin to Close	4.7	3.7	-
Trip Breakers Open	4.7	3.7	-
CEA's Begin to Drop	5.2	4.2	-
Pressurizer Empties	10.2	12.0	-
MSIV's Closed	10.7	9.7	-
SIAS on Low RCS Press.	12.0	14.3	1578 psia
RCP's Tripped	12.0	14.3	-
SI Actuation	42.0	44.3	-
MFW Isolation	86.0	86.0	-
AFW Initiated	180.1	180.1	253.6 lbm/sec
Transient Terminated	600.0	600.0	-

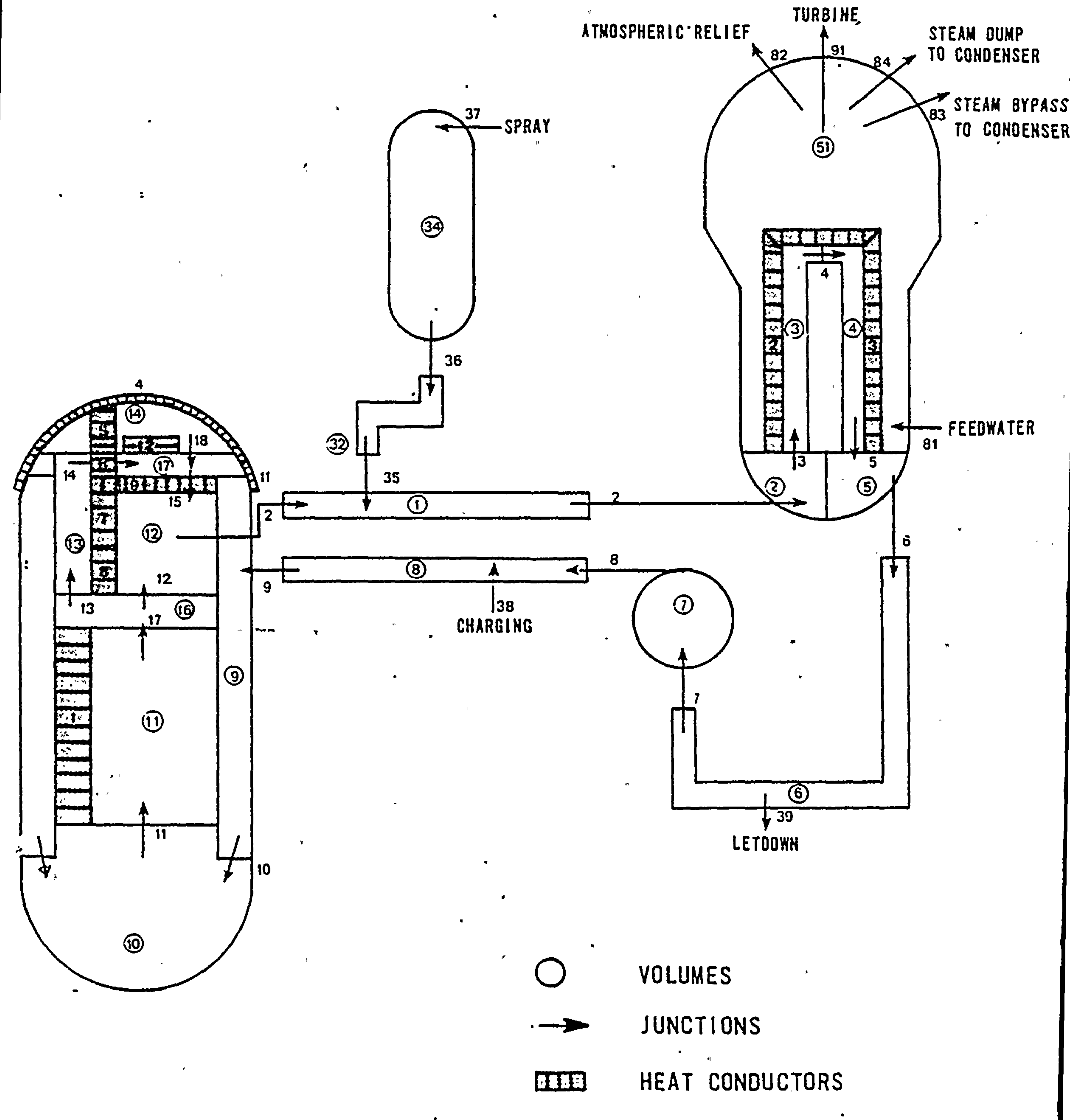

FIGURE 2.1.1 ST. LUCIE 1 RETRAN NODALIZATION DIAGRAM

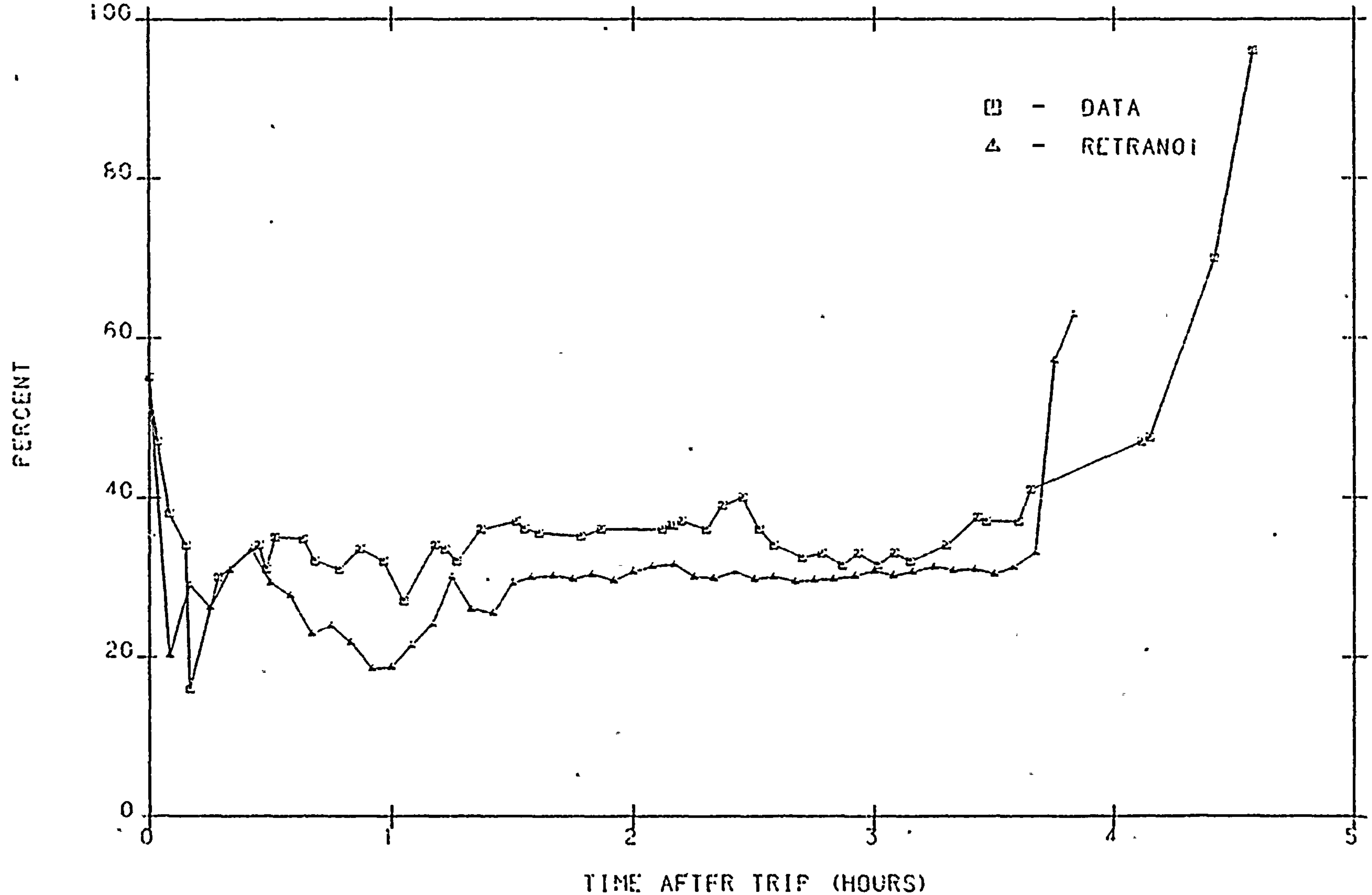

FIGURE 2.1.2 PRESSURIZER LEVEL
NATURAL CIRCULATION COOLDOWN EVENT

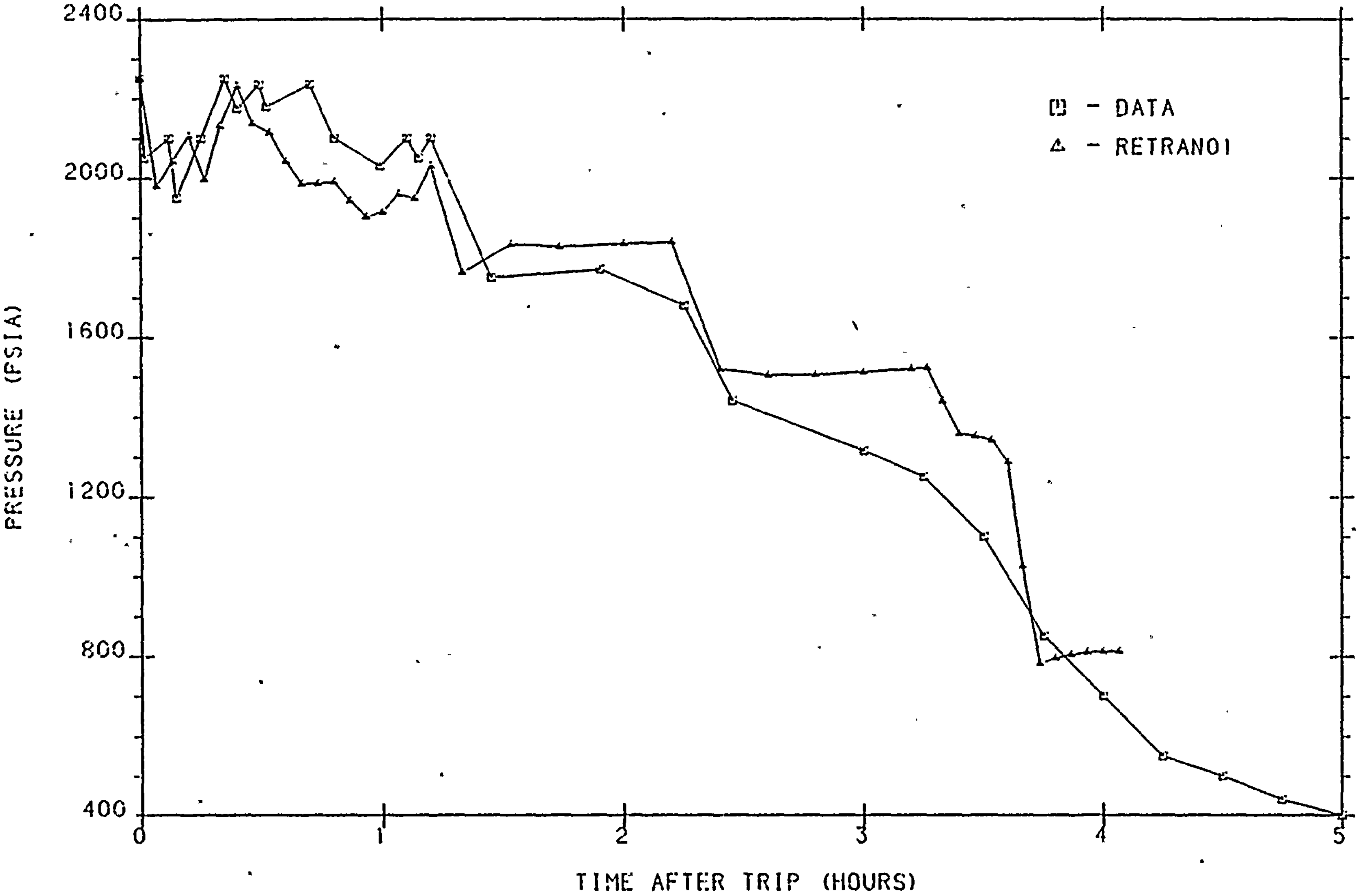

FIGURE 2.1.3 PRESSURIZER PRESSURE

NATURAL CIRCULATION COOLDOWN EVENT

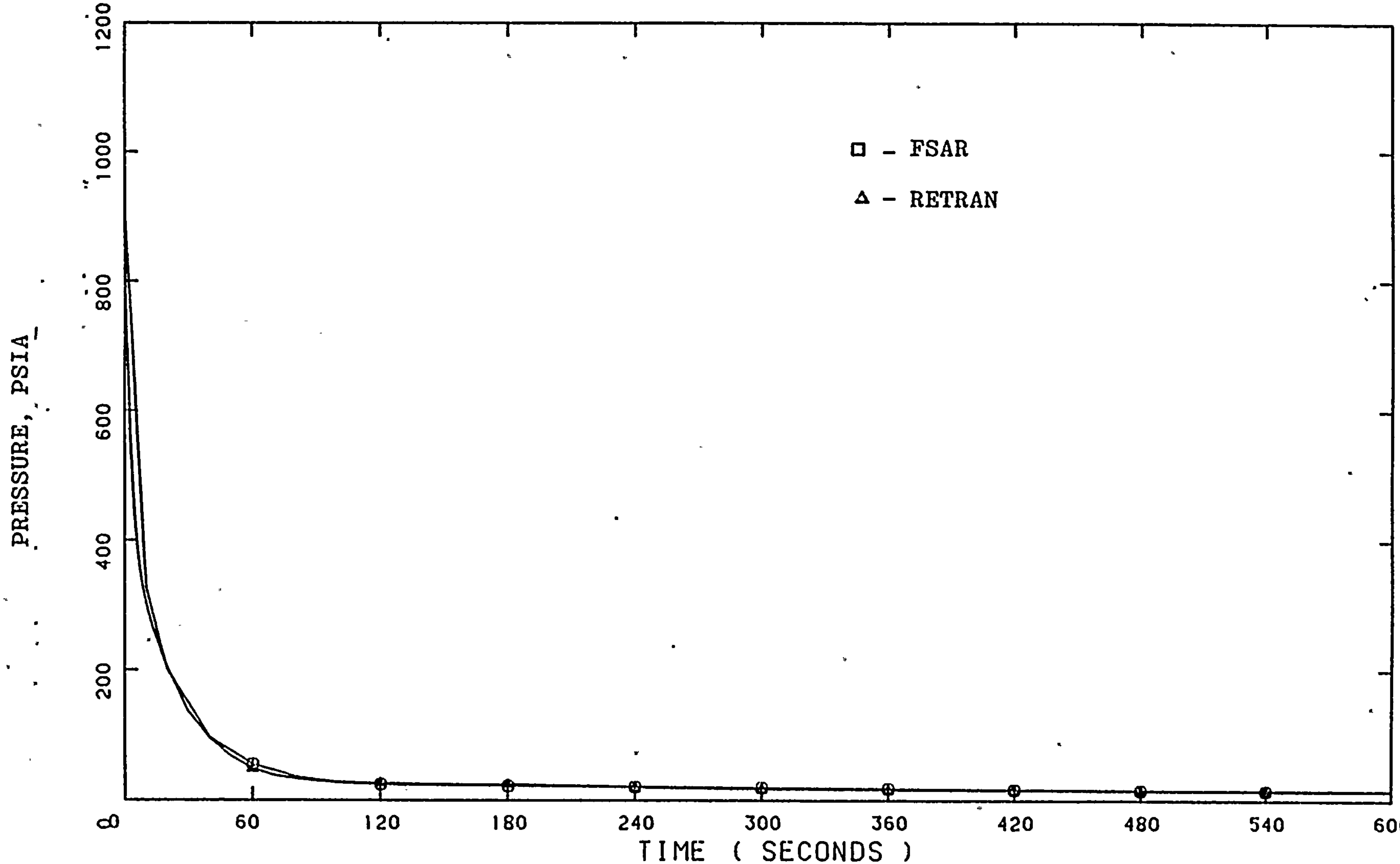

FIGURE 2.2.1 STEAM GENERATOR PRESSURE IN AFFECTED LOOP

MAIN STEAM LINE BREAK

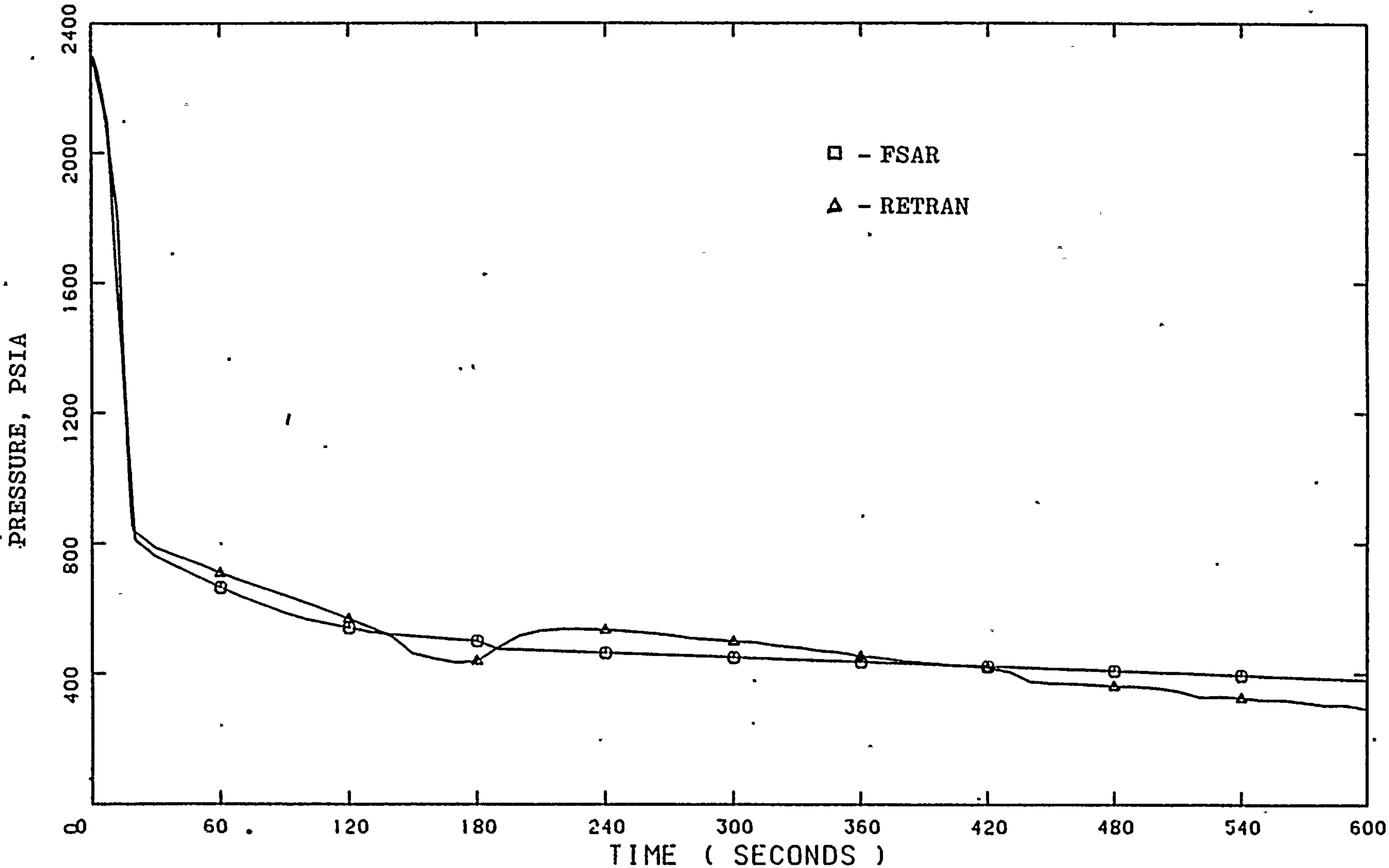

FIGURE 2.2.2 PRESSURIZER PRESSURE

MAIN STEAM LINE BREAK

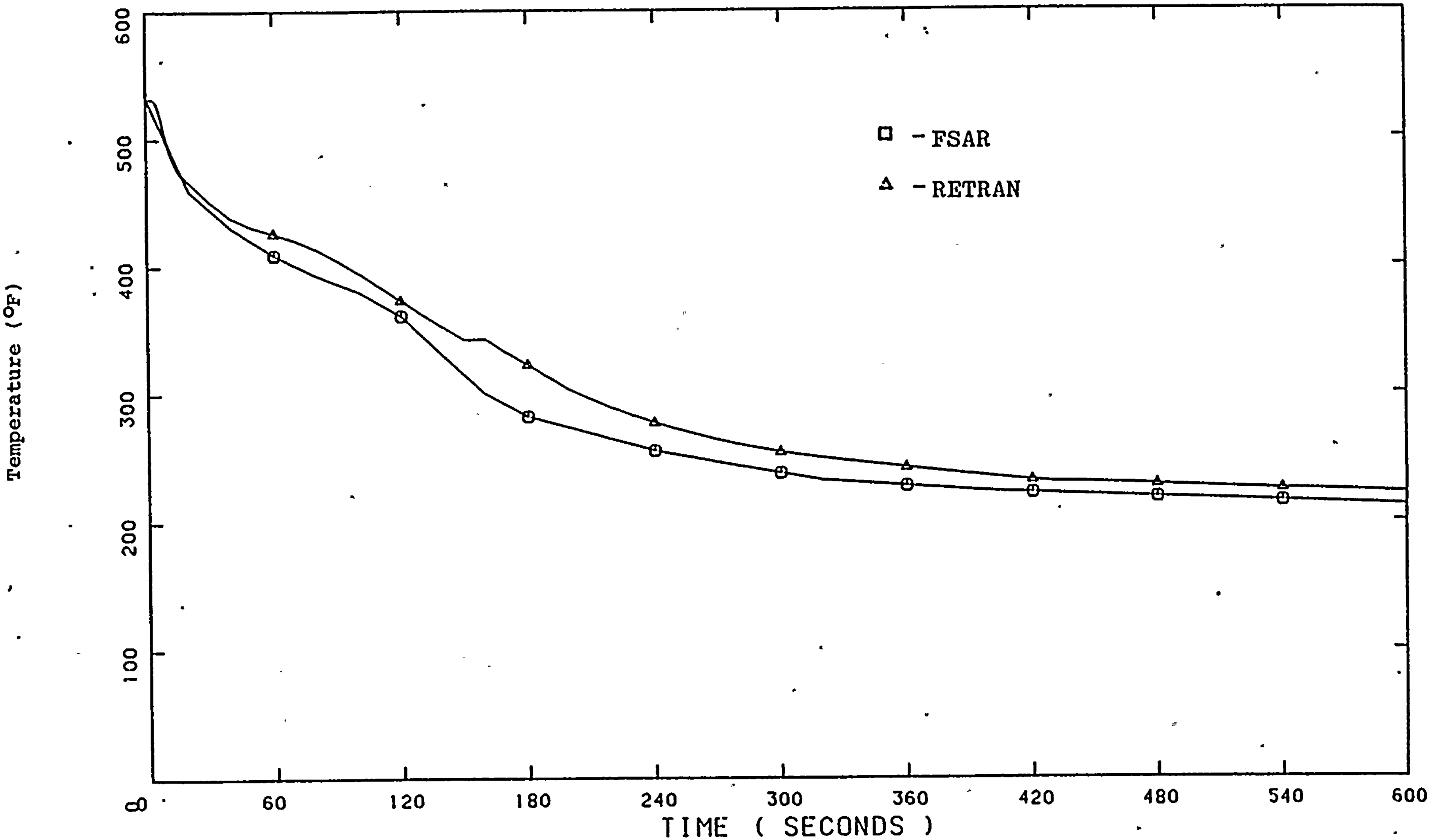

FIGURE 2.2.3 INLET CORE COOLANT TEMPERATURE

MAIN STEAM LINE BREAK

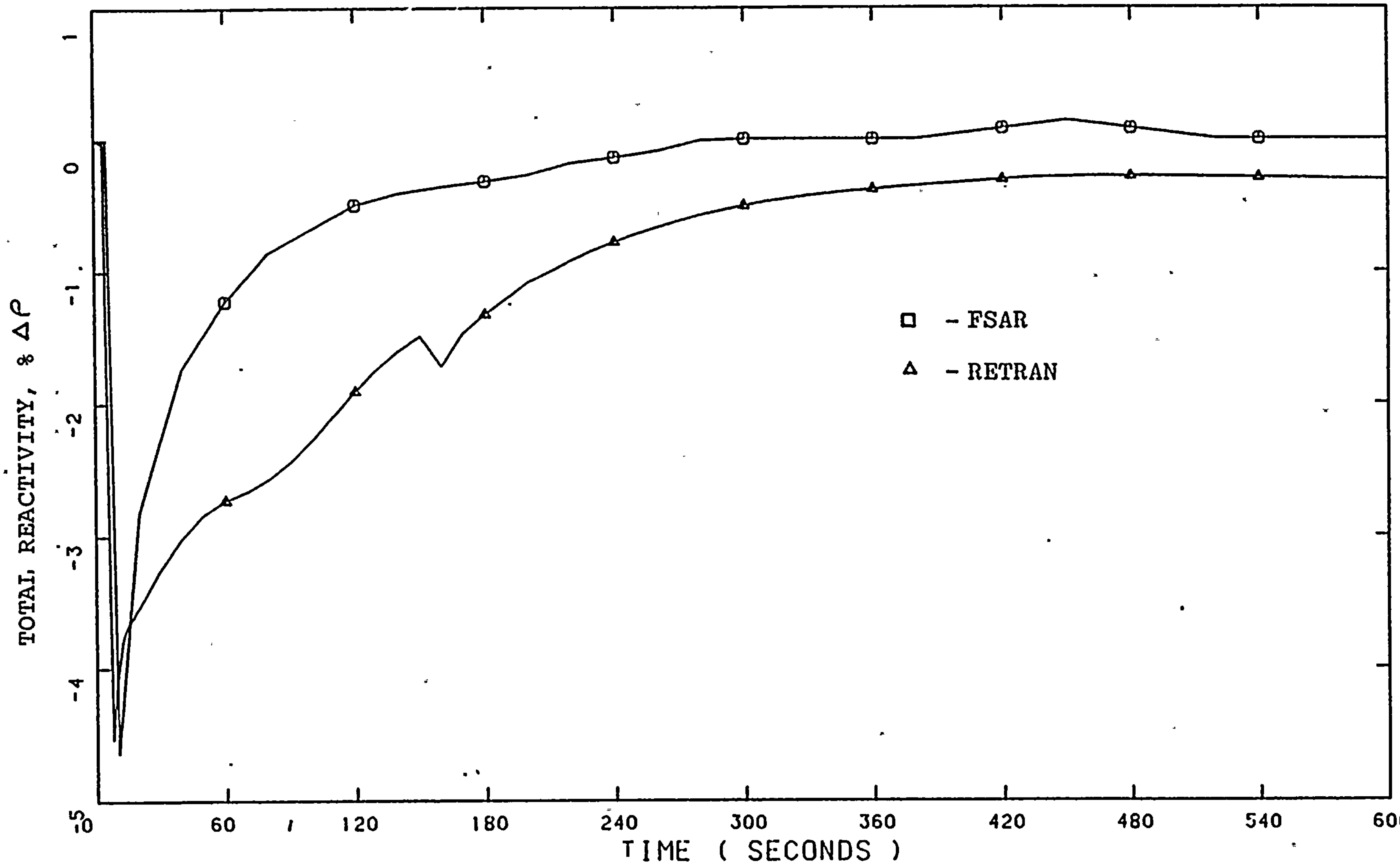

FIGURE 2.2.4 TOTAL REACTIVITY

MAIN STEAM LINE BREAK

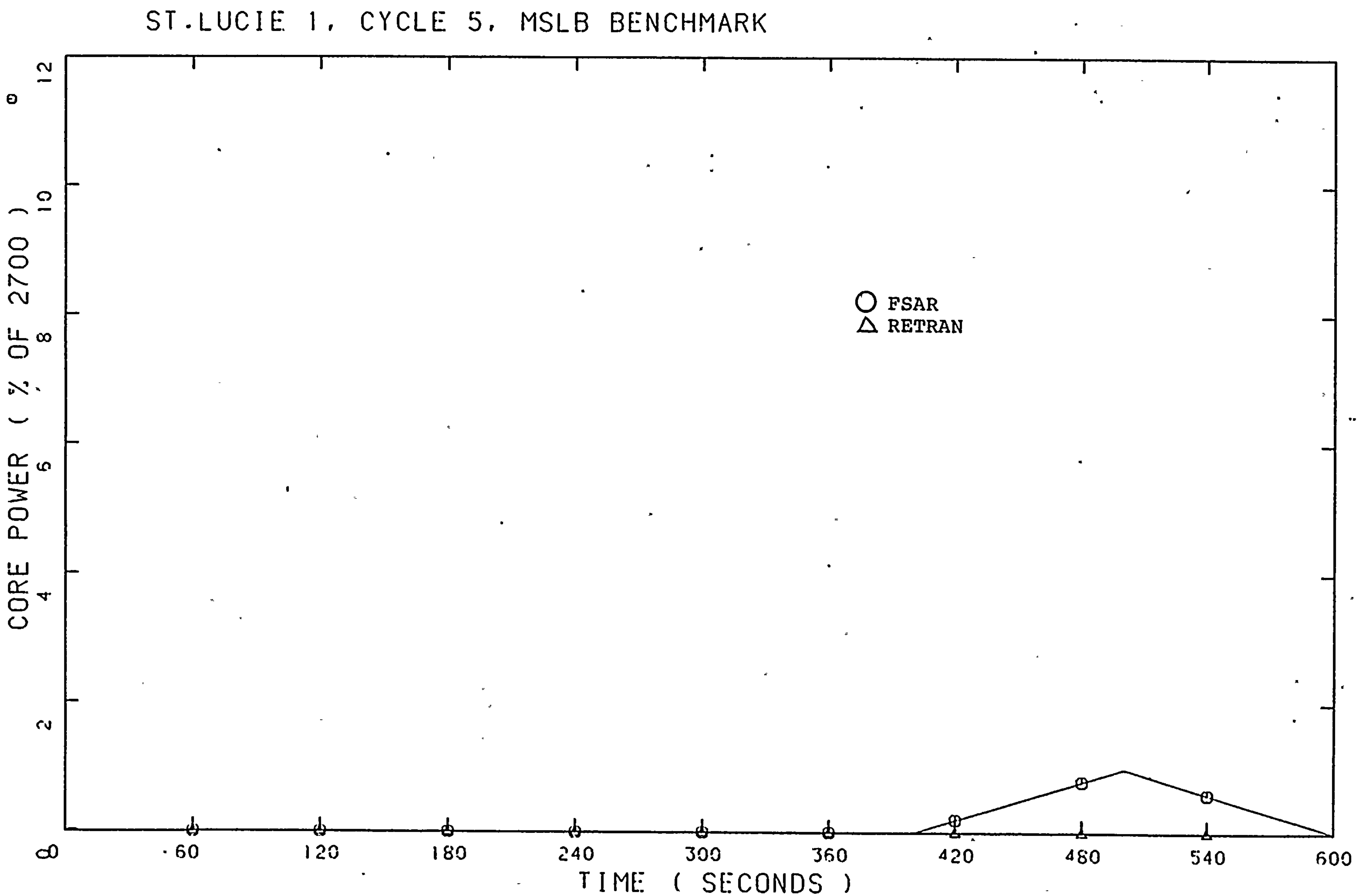

FIGURE 2.2.5 CORE POWER, MAIN STEAM LINE BREAK

3.0 DECREASE IN SECONDARY COOLANT SYSTEM HEAT REMOVAL

Events in this category result in significant RCS pressure increases due to a sudden reduction in heat removal by the secondary coolant system. These pressure increases may pose a direct challenge to the RCS boundary limits. A reduction in heat removal may be a result of either an increase in steam generator secondary side pressure or a decrease in steam generator secondary side fluid inventory. Important modeling considerations are the secondary coolant system modeling, reactivity coefficients, pressurizer level and pressure control systems, and reactor protection system functions.

Four benchmark analyses are presented in this category. The Turkey Point Loss of One Main Feedwater Pump Event (Section 3.1) benchmarked to plant data involves a decrease in the fluid inventory of the steam generators. The St. Lucie Unit 1 Loss of Load transient (Section 3.2) benchmarked to FSAR results, St. Lucie Unit 2 Generator Trip Test (Section 3.3) benchmarked to test data, and the St. Lucie Unit 1 MSIV Closure Event (Section 3.4) benchmarked to plant data involve increased steam generator pressures as the cause of the reduction in secondary coolant system heat removal.

3.1 Turkey Point Unit 4 Loss of One Main Feedwater Pump Event

3.1.1 <u>Transient Description</u>

On June 4, 1984, while operating at full power, Turkey Point

Unit 4 lost one of two main feedwater pumps due to a maintenance problem. As per design, the turbine ran back to 70%. The core power was reduced by inserting control rods manually at 72 steps per minute. In spite of this, the reactor tripped at 52 seconds into the transient. A peak primary pressure of 2345 psia, a peak secondary pressure of 1126 psia and a peak average primary temperature of 585°F were recorded. After the trip, the RCS experienced excessive cooldown with the average coolant temperature reaching 525°F.

The plant Technical Staff observed that the steam dump to the condenser failed to open; however, the atmospheric steam dump valves did open and one of the four steam generator safety relief valve groups also lifted. One of the four reheater steam isolation valves failed to close and continued to supply steam to one of the two high pressure feedwater heaters. The feedwater heater safety valve lifted at the setpoint of 465 psia and stayed open. The auxiliary feedwater initiated on a low-low level signal after reactor trip and was isolated manually by the plant operator.
The plant monitoring computer failed to record the exact sequence of events and the cause of the reactor trip. In order to reconstruct the sequence of events, determine the cause of the reactor trip, confirm the safety systems actuated, and to determine the cause of the post-trip cooldown, this event was simulated with the Turkey Point RETRAN base model.

Simulation of this event and comparing the results to the available plant data serves as a basis for assessing the adequacy of modeling of the plant protective systems, and reactivity feedback.

3.1.2 RETRAN Analysis Description

Simulation of this event was performed with the Turkey Point RETRAN base model (Appendix B) with modifications to explicitly account for the discharge of reheat steam to one of the feedwater heaters after turbine trip. This occurrence was simulated by opening a valve on the main steam line header which discharges steam into a constant pressure volume at 465 psia, the setpoint pressure for the heater relief valve.

The event was intiated by tripping one of the two MFW pumps. Steam dump to the condenser was deactivated. Reactivity due to insertion of control rods were input via tables. The auxiliary feedwater was initiated on reactor trip and isolated after two minutes. Feedwater enthalpy was varied as recorded by the plant computer.

Initial conditions for the analysis are given in Table 3.1.1. 250 seconds of the event were simulated which corresponds to the range covered by the available plant data. The safety systems

status assumed in the RETRAN model is presented in Table 3.1.2.

3.1.3 Results

Results of the RETRAN-simulation and their comparison with the plant data are presented in Figures 3.1.1 through 3.1.6. The sequence of events for the data and the RETRAN analysis is presented in Table 3.1.3. The analysis shows that the reactor tripped on low steam generator level coincident with steam/feed mismatch. The analysis confirms actuation of atmospheric steam dump valves, one of the four safety relief valve groups, pressurizer spray valve, and both pressurizer relief valves. The pressurizer pressure and level, the steam generator pressure and primary loop average temperature calculated by RETRAN compare well with the plant data.

3.2 St. Lucie Unit 1 Loss of Load

3.2.1 Transient Description

The St. Lucie Unit 1 Loss of Load event has been simulated with the RETRAN02 computer code. The results have been

benchmarked to a similar calculation described in the St. Lucie Unit 1 FSAR (Ref. 5) performed with the CESEC computer code. This benchmark demonstrates the technique of turning the RETRAN base model into a licensing type model and to simulate the system response to a decrease in secondary coolant system heat removal based on FSAR assumptions.

The transient is initiated from full power by the sudden closure of the turbine stop valves without a simultaneous reactor trip. After losing the secondary heat sink, the pressure in the primary and secondary side increases rapidly. A reactor trip signal is generated on a high pressurizer pressure.

Following reactor trip, the RCS pressure continues to increase, actuating the pressurizer safety valves. The rapid RCS cooldown associated with the reactor trip in conjunction with the mass and energy discharge through the safety valves reduces the RCS pressure. Likewise, the steam generator pressure is controlled by actuation of the main steam safety valves.

The average RCS coolant temperature and core inlet coolant temperature increase initially as the RCS pressure and core power increase. Following scram, the coolant temperatures decrease to hot zero power conditions, with the only energy sources the reactor coolant pumps and the core decay heat.

3.2.2 RETRAN Analysis Description

The St. Lucie Unit 1 RETRAN Base Model (Appendix B) was turned into a licensing type model for this simulation. Table 3.2.1, delineates the salient initial conditions incorporated in the RETRAN model. The status of the safety systems integrated in the RETRAN model are summarized in Table 3.2.2. The reactor physics parameters represent beginning of cycle conditions.

3.2.3 Results

The RETRAN02 results compares quite well with the FSAR predictions, as illustrated in Figures 3.2.1 - 3.2.4. The sequence of events is given in Table 3.2.3. The peak pressurizer pressure predicted by RETRAN is essentially the same as that of the FSAR.

The results substantiate the modeling techniques incorporated for predicting RCS pressure during over-pressure transients.

3.3 St. Lucie Unit 2 Generator Trip Test

3.3.1 Transient Description

The St. Lucie Unit 2 Generator Trip Test was performed on July 25, 1983 as part of the St. Lucie Unit 2 Power Ascension Test Program. During the ten minutes prior to initiating the test, initial conditions were established and maintained. The test was initiated by manually tripping the generator output oil circuit breakers which caused the turbine stop valve to close while operating at 100 % of rated power. There was no operator intervention for two minutes. The RETRAN02 simulation of this transient served to assess the adequacy of the steam bypass and dump system, the pressurizer and the steam generator modeling.

3.3.2 RETRAN Analysis Description

Simulation of the Generator Trip Test was performed with the St. Lucie Unit 2 RETRAN base model. The initial conditions are presented in Table 3.3.1. In accordance with observed behavior during the test, the auxiliary feedwater system was forced to start immediately following reactor trip. The benchmark effort in this analysis was limited to the first 120 seconds of the transient during which no operator action was taken.

3.3.3 Results

The sequence of events corresponding to the test and the RETRAN calculation appears in Table 3.3.3 and the status of the safety systems in Table 3.3.2. As a result of the turbine stop valve closure there was a rapid increase in the secondary pressure (Figure 3.3.1). Generally, good agreement between measured and calculated pressure responses in both steam generators was obtained. The inflection in the measured steam generator pressure response at approximately 10 seconds was not calculated because of small differences between the actual and calculated energy removal rates of the steam dump and bypass system. A maximum steam generator secondary pressure of 972 psia at 74 seconds was calculated. This compares well to the measured peak steam generator pressure of 953 psia.

Figure 3.3.2 depicts the calculated and measured pressurizer pressure. The initial heatup of the RCS caused by the loss of steam generator heat removal is reflected in the momentary pressure increase following the trip. RETRAN calculated a peak RCS pressure of 2263 psia at 2.0 seconds which, again, compares well to the measured pressure of 2262 psia at 2.0 seconds. Good agreement with data was achieved during the subsequent depressurization. A minimum pressure of 2057 psia was calculated which is in good agreement with the recorded value of 2078 psia. Following the closure of the steam bypass valves together with the activation of the pressurizer

backup heaters, the RCS started to slowly repressurize. This trend was calculated, but the amount of repressurization calculated was less than measured.

The calculated and measured hot leg temperatures are presented in Figure 3.3.3. The hot leg temperature remains essentially constant throughout the first five seconds of the transient due to the competing effects of the reactor trip and loss of steam generator load. Corresponding to the decrease in core power, together with the steam bypass action, the hot leg temperature then decreased rapidly. A maximum difference of $4^{\circ}F$ between measured and calculated temperatures exists.

As good agreement with data was calculated with the St. Lucie Unit 2 RETRAN Base Model, this model is judged to be adequate to simulate transients involving a decrease in secondary coolant system heat removal.

3.4 St. Lucie Unit 1 MSIV Closure Event

3.4.1 Transient Description

In December 1981, St. Lucie Unit 1 was operating at 2646 MW (thermal) when the MSIV closed on steam generator "A" due to a malfunction. A few seconds later the steam generator "B" MSIV closed due to increasing steam flow. The reactor tripped on high pressurizer pressure. The pressurizer PORVs responded to the

high pressure when it reached their setpoint, by opening automatically. At approximately the time of the PORV opening, a pressurizer safety relief valve (SRV) also lifted. The RCS pressure decreased as a result of the opening of the PORVs and safety valve.

As the pressure continued to decrease below the PORV reset setpoint, the operators shut the PORV block valves and the pressure stablized at 1650 psia (just above SI setpoint).

3.4.2 RETRAN Analysis Description

The MSIV closure transient has been simulated with the RETRAN02 computer code in conjunction with the St. Lucie 1 RETRAN base deck. The analysis was performed based upon available plant data and an INPO report (Ref. 7). The initial conditions are presented in Table 3.4.1. It was concluded from plant data that the PORVs and pressurizer SRV opened virtually at the same time. When the PORVs did not reset at their reset pressures, the operator closed the block valves upstream of the PORVs at 1650 psia. In order to benchmark the RETRAN results with actual transient results, the PORVs reset pressure was adjusted in the RETRAN model to correspond to the pressure at which the block valves were closed.

3.4.3 Results

The results of the RETRAN calculation and plant data for pressurizer pressure and level are compared in Figures 3.4.1 and 3.4.2. The status of the safety systems is given in Table 3.4.2. The comparison between the sequence of events is given in Table 3.4.3. The 'A' MSIV closed at the start of the transient and 'B' MSIV closed at 2.2 seconds because of increasing steam flow in steam generator B. The reactor tripped on high RCS pressure and PORVs reached their setpoint and opened. A pressurizer safety valve had lifted at the same time and closed when the pressure dropped to its reset value. The charging pumps were initiated according to the pressurizer level control system and the pressurizer level rose and stabilized at 600 seconds (Figure 3.4.1). The RETRAN02 predictions have been compared to available plant transient data and agree reasonably well.

TABLE 3.1.1

INITIAL AND BOUNDARY CONDITIONS

LOSS OF ONE FEEDWATER PUMP

PARAMETER	VALUE
Core Power, MW (Thermal)	2200
Core Inlet Coolant Temperature, °F	546.2
Core Outlet Coolant Temperature, °F	602.2
Vessel Mass Flow Rate, 10^6 lbm/hr	101.
Pressurizer Pressure, psia	2250.
Steam Generator Pressure, psia	843

TABLE 3.1.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL
LOSS OF ONE FEEDWATER PUMP

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves	X		
Pressurizer Safety Valves		X	
Main Steam Isolation Valves		X	
Main Feedwater Isolation Valves		X	
Auxiliary Feedwater System	X		
Safety Injection System			
HPSI		X	
Accumulators			X
LPSI			X
Atmospheric Dump Valve Systems	X		
Steam Dump and Bypass System		X	
Pressurizer Level Control System	X		
Pressurizer Power Operated Relief Valves (PORV)	X		
Chemical and Volume Control System			X
S. G. Level Control System	X		
Manual Rod Motion	X		

TABLE 3.1.3

SEQUENCE OF EVENTS

LOSS OF ONE FEEDWATER PUMP

EVENT	TIME(S) PLANT	TIME(S) RETRAN	PARAMETER PLANT	PARAMETER RETRAN
Loss of One MFW Pump	0.0	0.0	--	--
Pressurizer Spray Valve Opened	--	13.0	--	2275. psia
Pressurizer Relief Valve-1 Opened	--	21.0	--	2350 psia
Pressurizer Relief Valve-2 Opened	--	21.1	--	2350 psia
Pressurizer Relief Valve-2 Closed	--	26.8	--	2330 psia
Pressurizer Relief Valve-1 Closed	--	43.3	--	2330 psia
Reactor Tripped	51.5	48.9	--	--SG Low-Level of 30% With Steam/Feed Mismatch
Turbine Tripped	51.9	49.9	--	Reactor Trip
MSR/MOV Stuck Open	--	49.9	--	Valve Failure
S.G. Relief Valve Opened	--	52.0	--	1050 psia
Pressurizer Spray valve closed	--	52.0	--	2275 psia
S.G. SRV Lifted	--	54.2	--	1100 psia
AFW Initiated	--	58.0	S.G. Low Level-15% Trip	G.S. Low Level-15% Trip
S.G. SRV Reseated	--	68.6	--	1089 psia

TABLE 3.1.3 (CONTINUED)

LOSS OF ONE FEEDWATER PUMP

EVENT	TIME(S) PLANT	TIME(S) RETRAN	PARAMETER PLANT	PARAMETER RETRAN
S.G. Relief Valve Closed	--	103.0	--	1100 psia
AFW Manually Terminated	--	105.0	Manual	Manual
MFW Isolated	135.5	124.0	--	Low T_{ave} of 554°F with Reactor Trip
Peak Pressurizer Pressure, psia	34.	21.-42.	2345.	2350.
Peak S.G. Pressure, psia	75.	58.	1126.	1124.

TABLE 3.2.1

INITIAL CONDITIONS AND KEY PARAMETERS

LOSS OF LOAD

PARAMETER	VALUE
Core Power MW (Thermal)	2754
Core Inlet Coolant Temperature (°F)	551
Core Coolant Flow (10^6 lbm/hr)	138.3
Pressurizer Pressure, psia	2200
Steam Generator Pressure, psia	820
Moderator Temperature Coefficient ($10^{-4} \Delta k$/°F)	+0.5
Fuel Temperature Coefficient ($10^{-5} \Delta k$/°F)	-1.07
CEA Worth at Trip (% Δk)	-4.7

TABLE 3.2.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

LOSS OF LOAD

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves	X		
Pressurizer Safety Valves	X		
Main Steam Isolation Valves	X		
Main Feedwater Isolation Valves	X		
Auxiliary Feedwater System		X	
Safety Injection System			
HPSI		X	
Accumulators			X
LPSI		X	
Atmospheric Dump Valve Systems			X
Steam Dump and Bypass System			X
Pressurizer Level Control System	X		
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System		X	
S. G. Level Control System			X
Automatic Rod Motion			X

TABLE 3.2.3

SEQUENCE OF EVENTS

LOSS OF LOAD

EVENT	TIME RETRAN	TIME FSAR	PARAMETER RETRAN	PARAMETER FSAR
Loss of Secondary Load	0.0	0.0	---	
Steam Generator SRVs Open	8.17	5.1	1010 psia	
High Pressurizer Pressure Trip Signal	8.00	7.8	2422 psia	
Pressurizer SRVs Open	9.10	9.0	2500 psia	
CEA's Begin to Drop Into Core	9.40	9.2	--	
Maximum Pressurizer Pressure	9.68	11.0	2574 psia	2572 psia
Maximum Steam Generator Pressure	12.4	11.2	1040 psia	1057 psia
Pressurizer SRVs Fully Close	14.8	13.5		

TABLE 3.3.1

INITIAL CONDITIONS

GENERATOR TRIP TEST

PARAMETER	VALUE
Core Power, MW (Thermal)	2560
Pressurizer Pressure, psia	2257.5
Steam Generator Pressure, psia	809.7
Core Inlet Coolant Temperature, °F	539.0
Core Outlet Coolant Temperature, °F	588.0
Pressurizer Level, %	55

TABLE 3.3.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

GENERATOR TRIP TEST

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves		X	
Pressurizer Safety Valves		X	
Main Steam Isolation Valves			
Main Feedwater Isolation Valves	X		
Auxiliary Feedwater System	X		
Safety Injection System			
HPSI		X	
Accumulators			X
LPSI			X
Atmospheric Dump Valve Systems		X	
Steam Dump and Bypass System	X		
Pressurizer Level Control System	X		
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System			X
S. G. Level Control System			X
Automatic Rod Motion			X

TABLE 3.3.3

SEQUENCE OF EVENTS

GENERATOR TRIP TEST

EVENT	TIME(S)		PARAMETER	
	TIME	RETRAN	TEST	RETRAN
Turbine Trip	0.	0.	--	--
Turbine Stop Valves Closed	0.22	0.22	--	--
Reactor Trip	0.1	0.1	--	--
Steam Dump and Bypass Activated	0.4	0.5	On High Average RCS Temperature After Reactor Trip	
Peak Pressurizer Pressure	2.0	2.0	2262 psia	2263 psia
Peak Steam Generator Pressure	65.0	74.0	953 psia	972 psia

TABLE 3.4.1

INITIAL CONDITIONS

MSIV CLOSURE EVENT

PARAMETER	VALUE
Core Power, MW (Thermal)	2646
Core Inlet Coolant Temperature, °F	535
Pressurizer Pressure, psia	2250
Pressurizer Liquid Volume, ft^3	743
Pressurizer Steam Volume, ft^3	757

TABLE 3.4.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

MSIV CLOSURE EVENT

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves	X		
Pressurizer Safety Valves		X	
Main Steam Isolation Valves	X		
Main Feedwater Isolation Valves	X		
Auxiliary Feedwater System		X	
Safety Injection System			
HPSI		X	
Accumulators			X
LPSI		X	
Atmospheric Dump Valve Systems			X
Steam Dump and Bypass System			X
Pressurizer Level Control System			X
Pressurizer Power Operated Relief Valves (PORV)	X		
Chemical and Volume Control System			X
S. G. Level Control System		X	
Automatic Rod Motion			X

TABLE 3.4.3

SEQUENCE OF EVENTS

MSIV CLOSURE EVENT

	PLANT DATA (SECONDS)	RETRAN (SECONDS)
MSIV (A) Valve Closed	0.0	0.0
MSIV (B) Valve Closed	2.6	2.6
PORV's Opened	6.7	5.8
Pressurized Safety Valve Opened	6.7	5.8
Reactor Tripped on HI Pressurizer Press.	6.8	6.7
Turbine Tripped on Reactor Trip.	7.0	6.9
Minimum Pressurizer Pressure	112 (1650 psia)	106 (1640 psia)

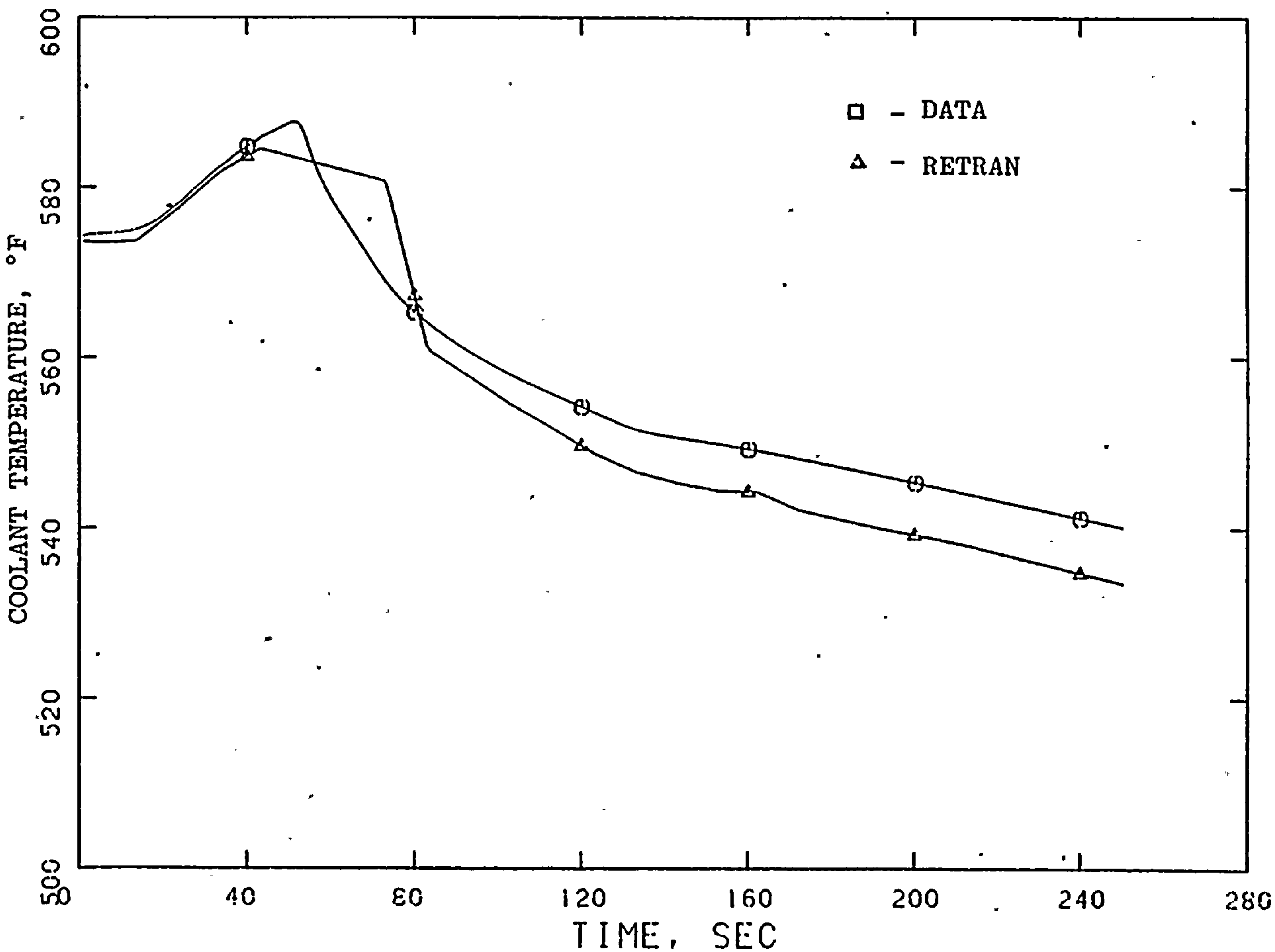

FIGURE 3.1.1 AVERAGE RCS TEMPERATURE

LOSS OF ONE FEEDWATER PUMP

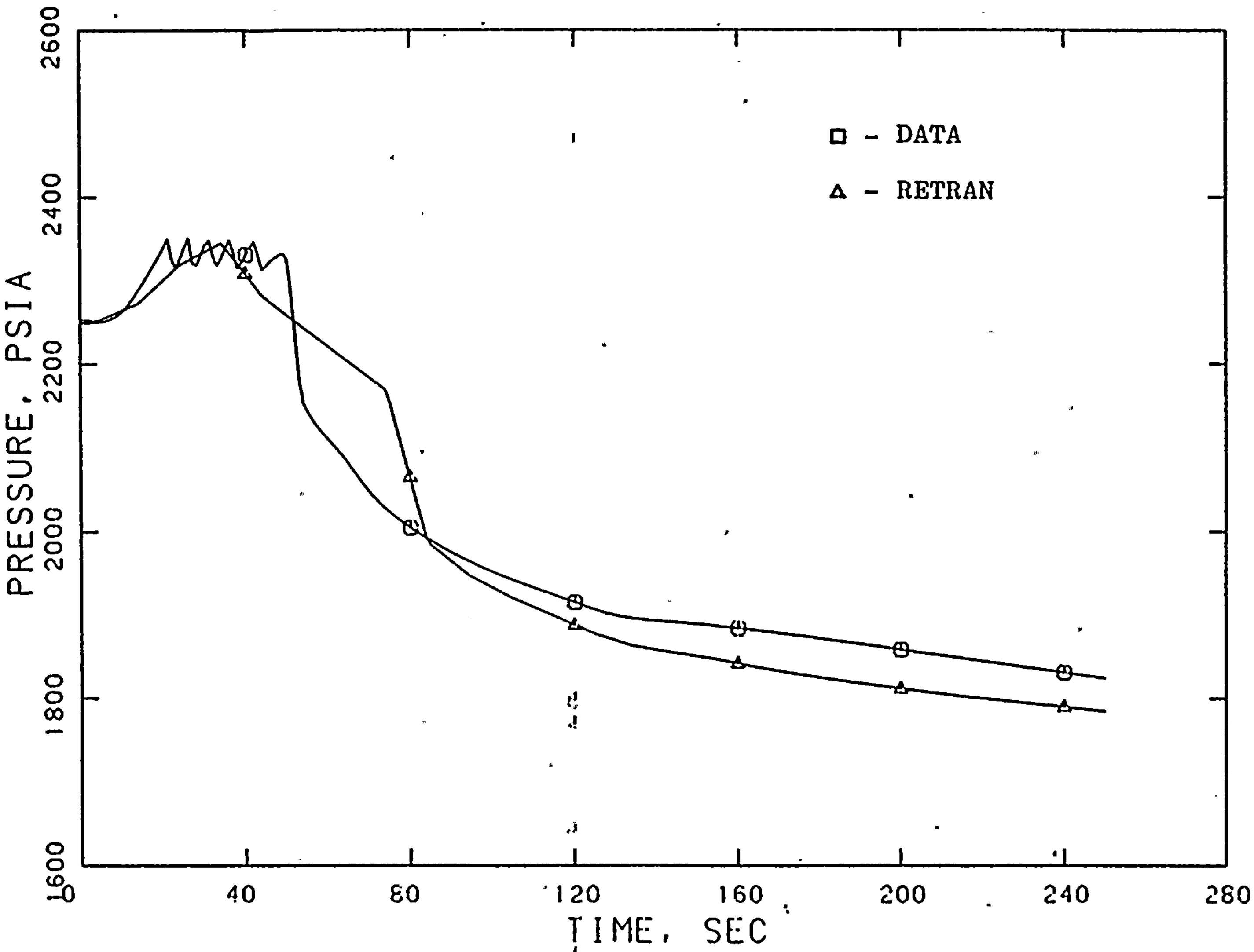

FIGURE 3.1.2 PRESSURIZER PRESSURE
LOSS OF ONE FEEDWATER PUMP

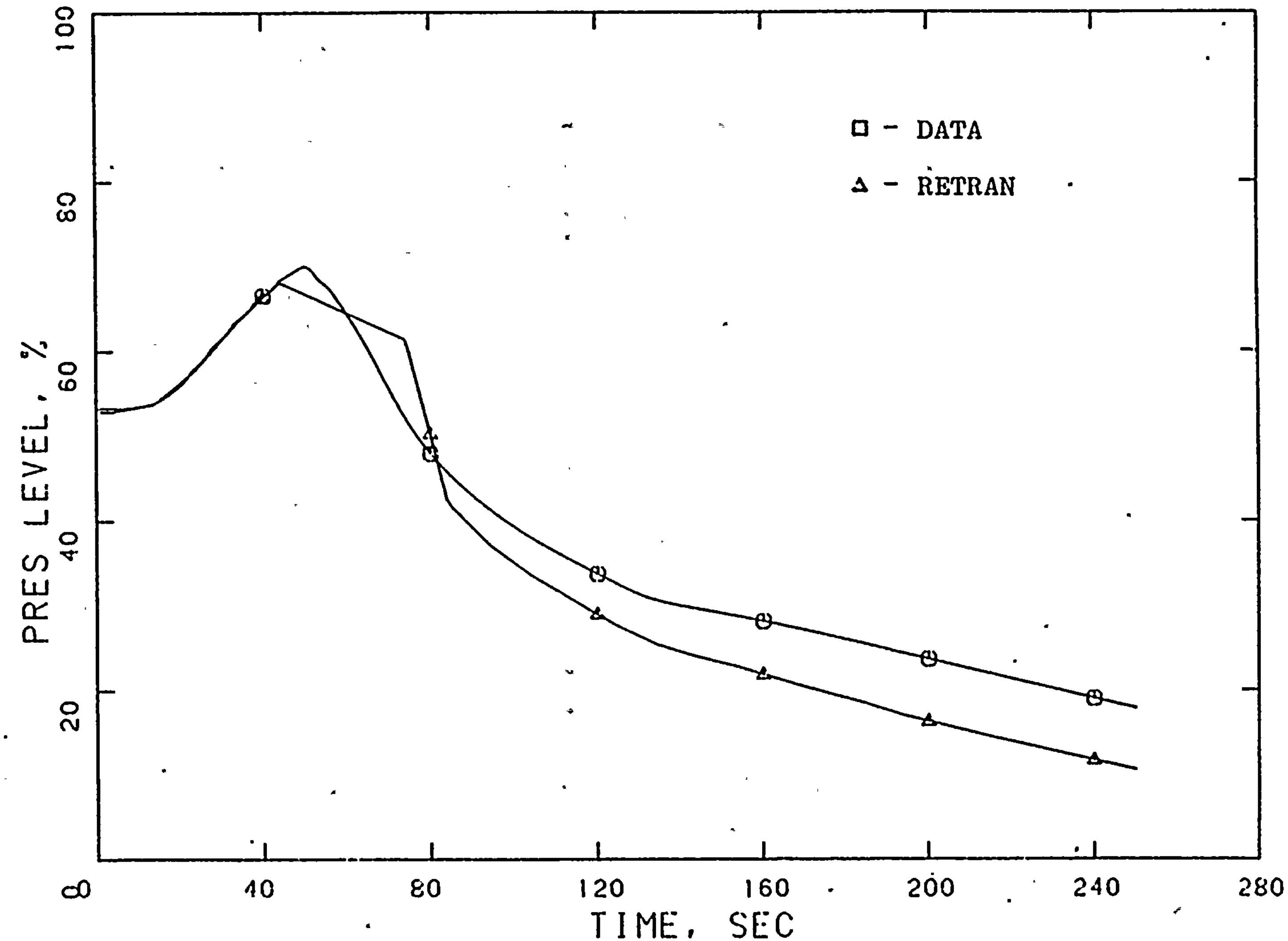

FIGURE 3.1.3 PRESSURIZER LEVEL
LOSS OF ONE FEEDWATER PUMP

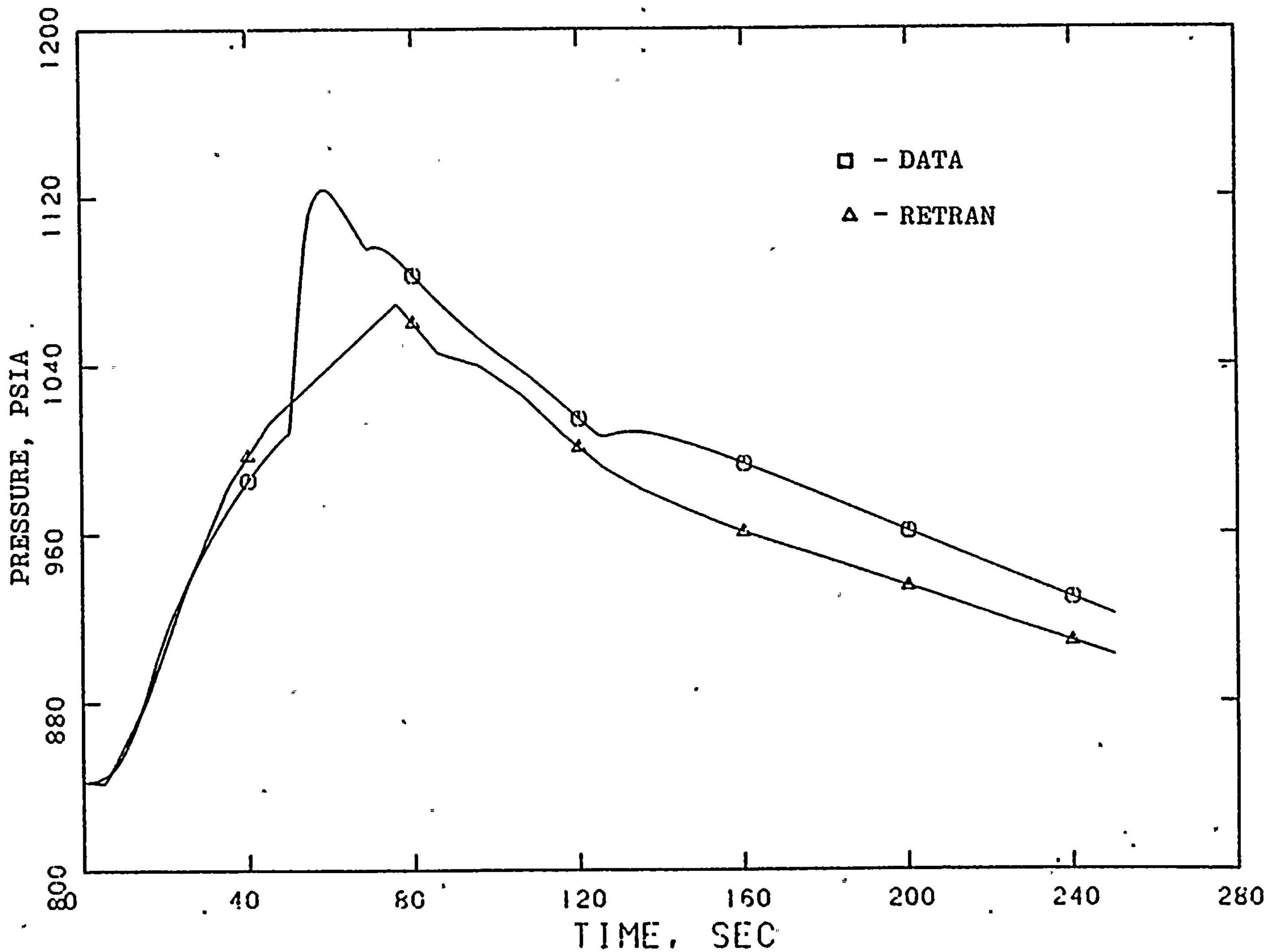

FIGURE 3.1.4 STEAM GENERATOR PRESSURE

LOSS OF ONE FEEDWATER PUMP

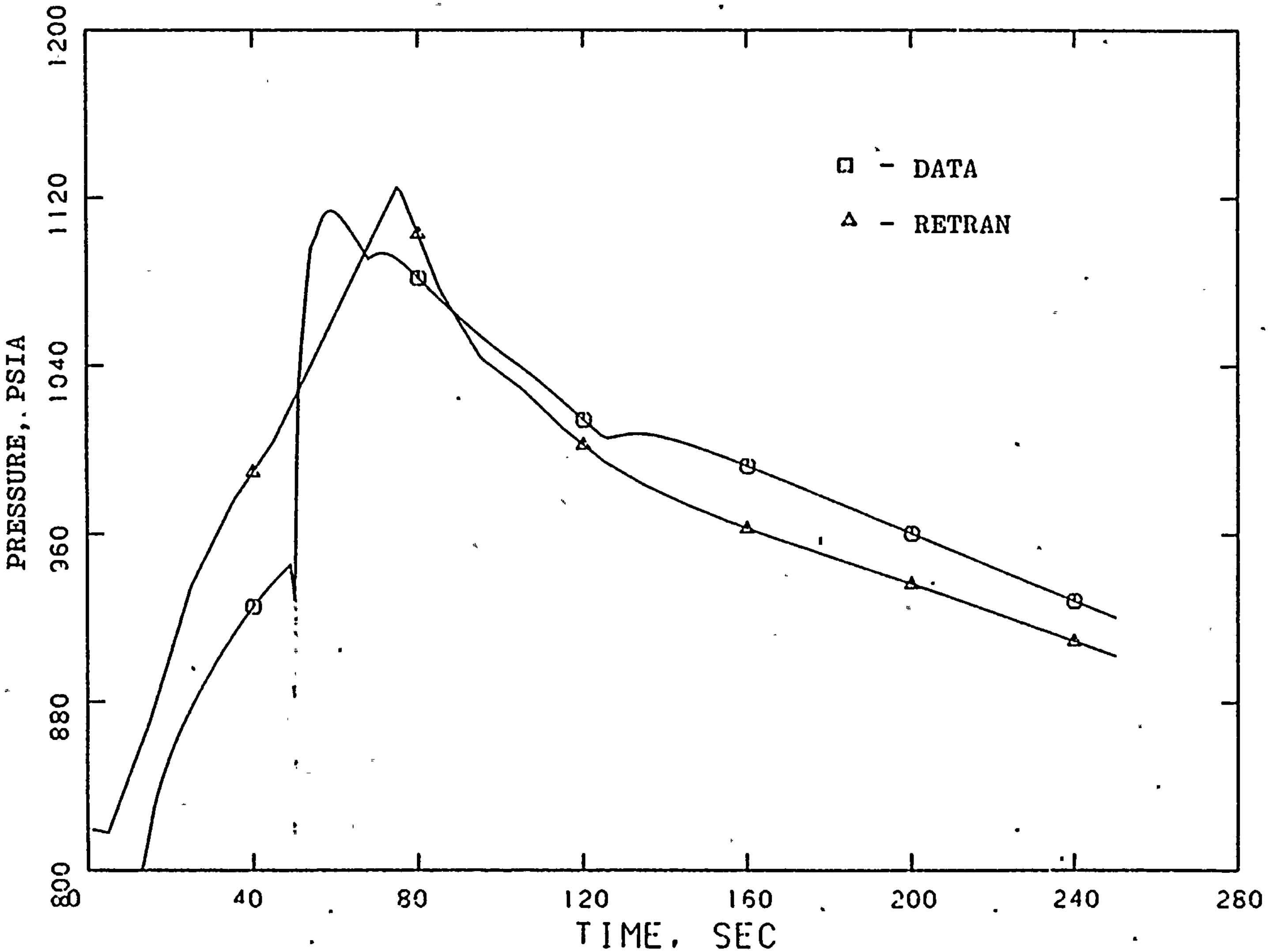

FIGURE 3.1.5 STEAM HEADER PRESSURE
LOSS OF ONE FEEDWATER PUMP

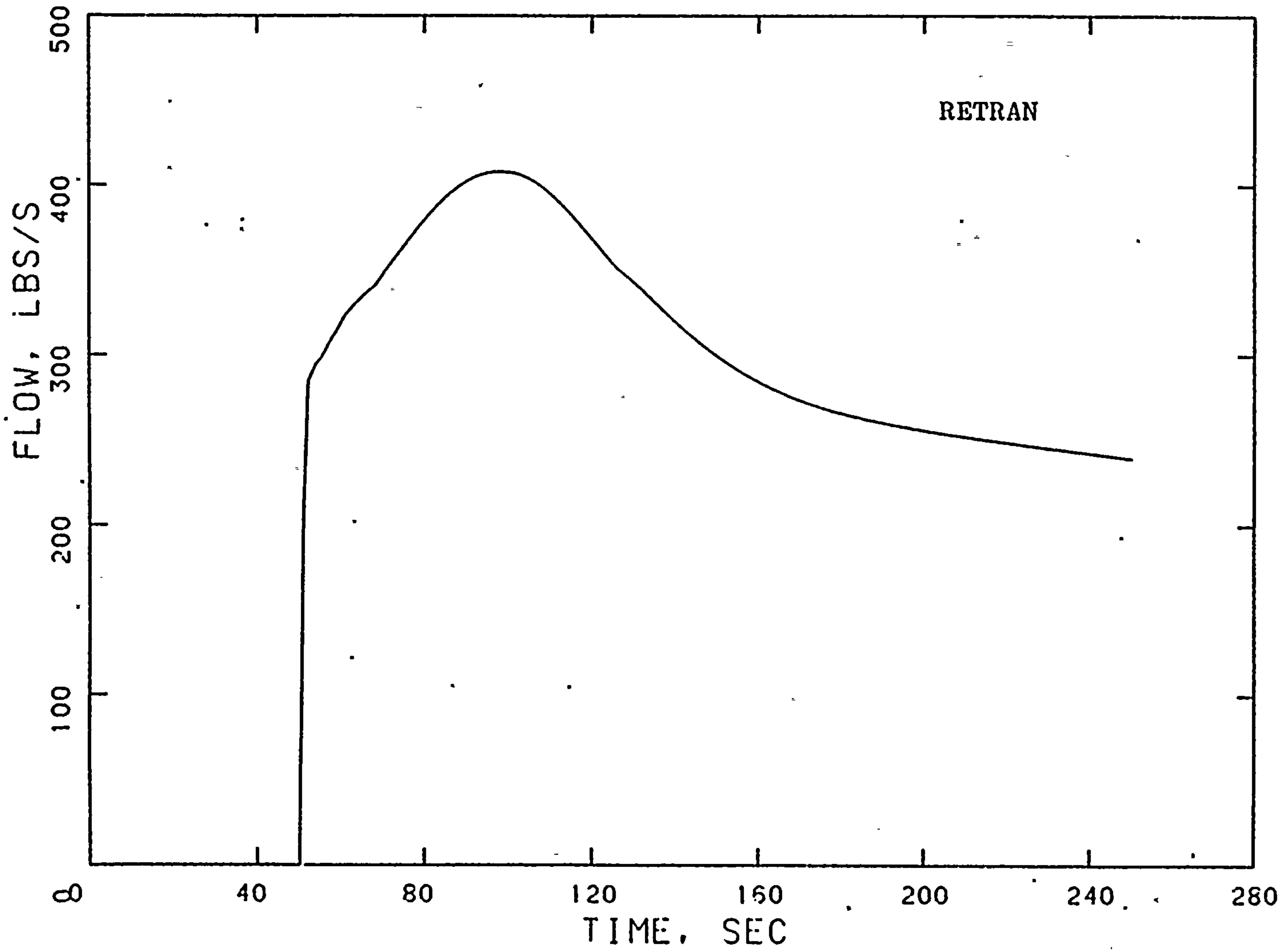

FIGURE 3.1.6 REHEAT STEAM FLOW

LOSS OF ONE FEEDWATER PUMP

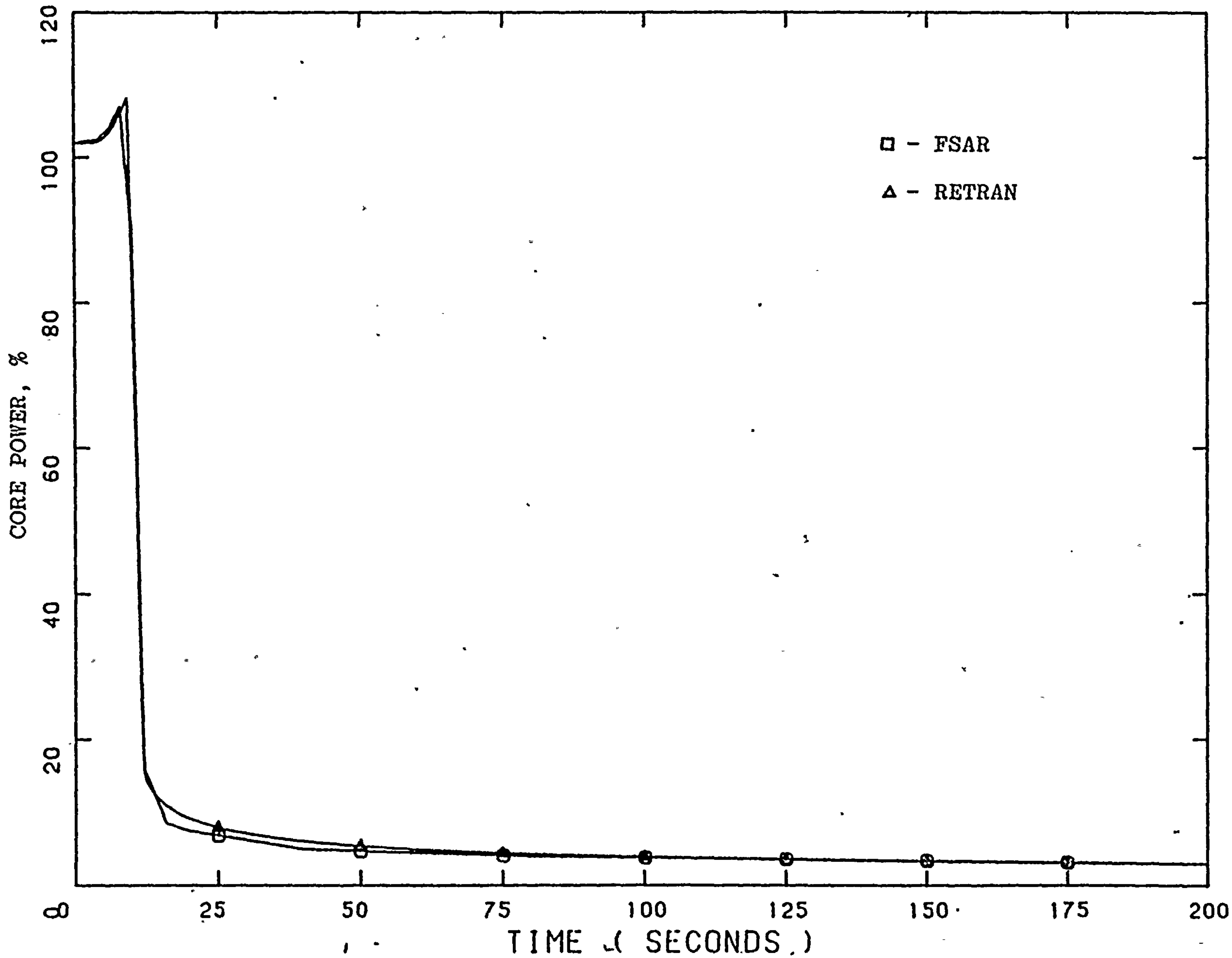

FIGURE 3.2.1 PERCENT CORE POWER

LOSS OF LOAD

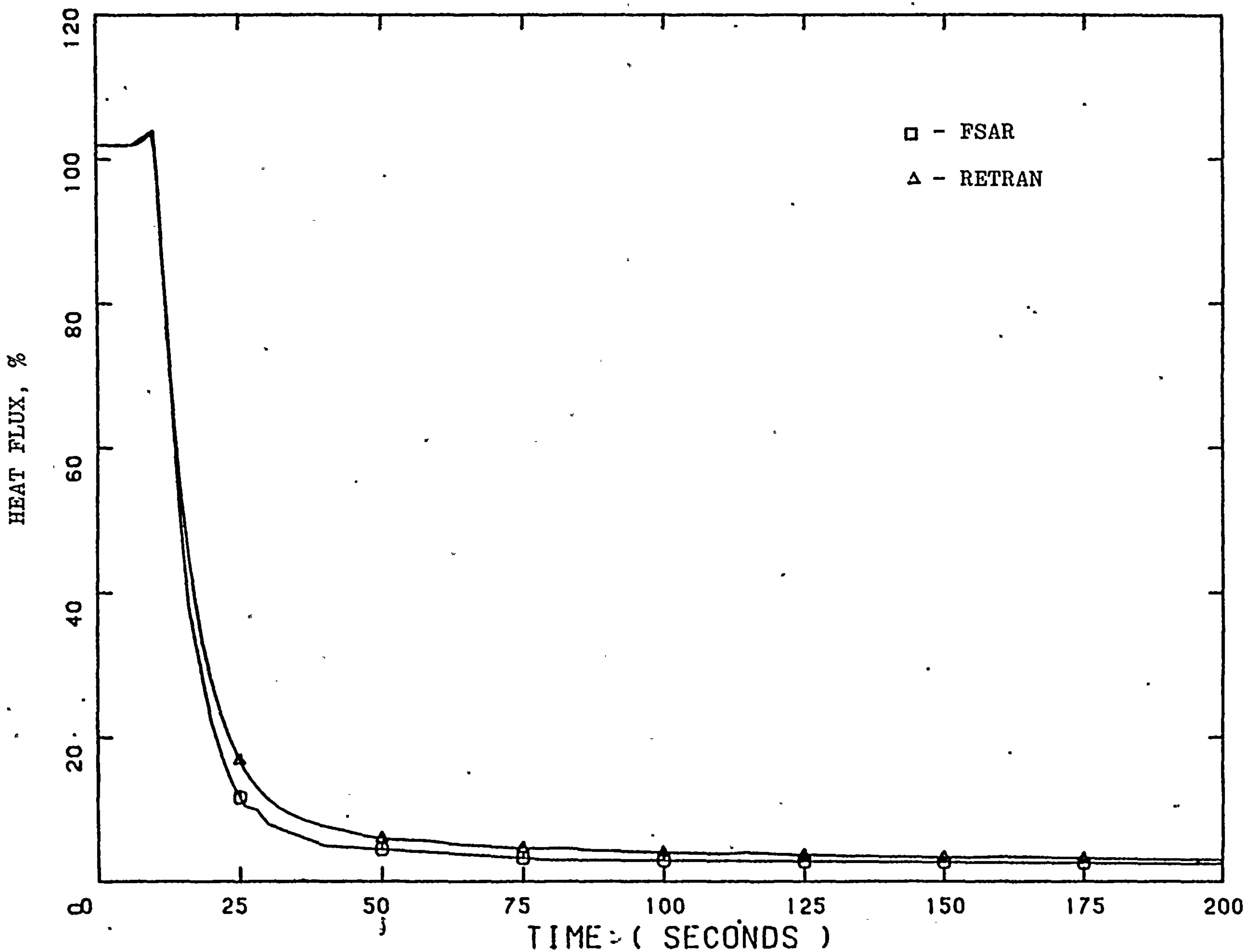

FIGURE 3.2.2 PERCENT CORE HEAT FLUX

LOSS OF LOAD

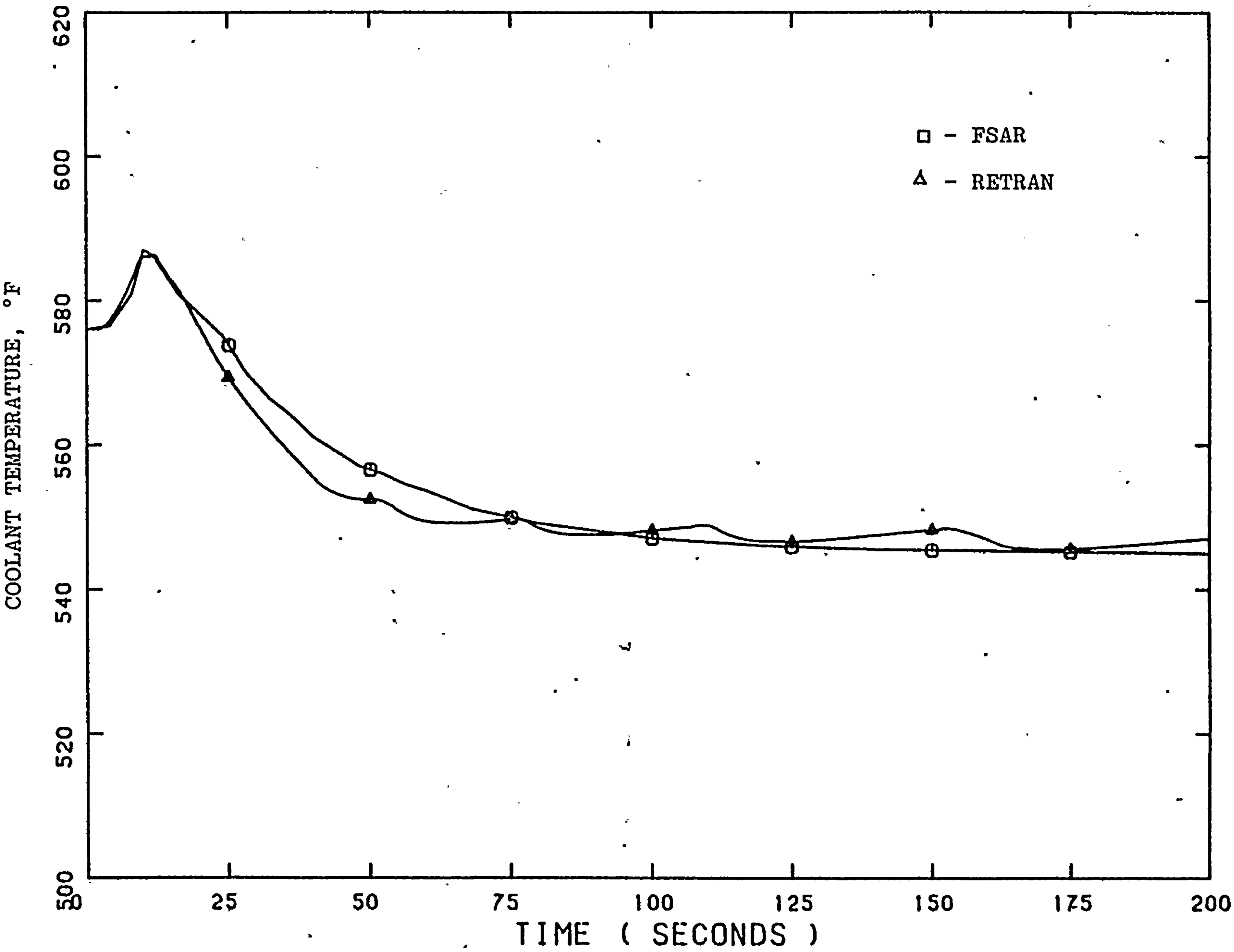

FIGURE 3.2.3 AVERAGE RCS TEMPERATURE

LOSS OF LOAD

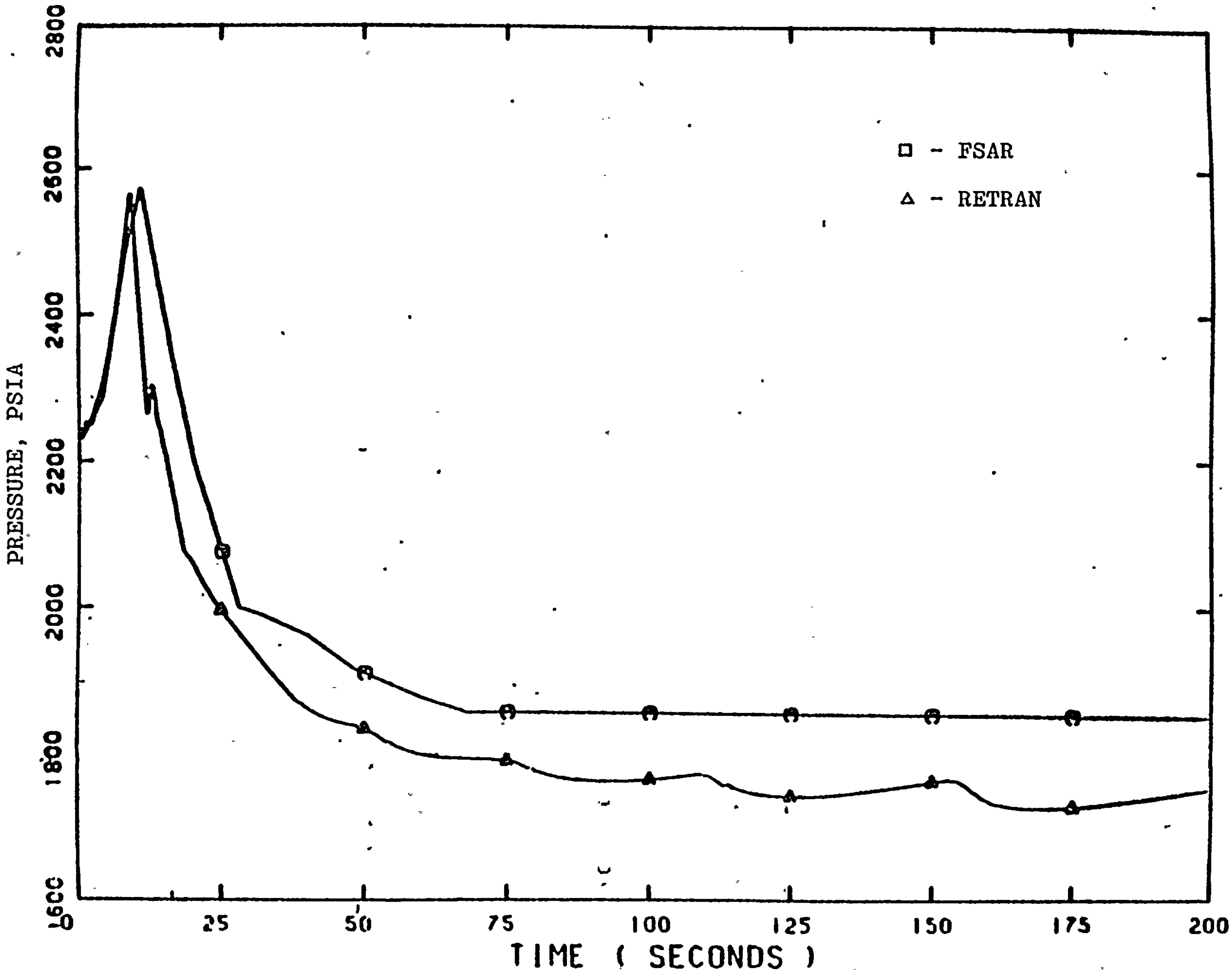

FIGURE 3.2.4 PRESSURIZER PRESSURE

LOSS OF LOAD

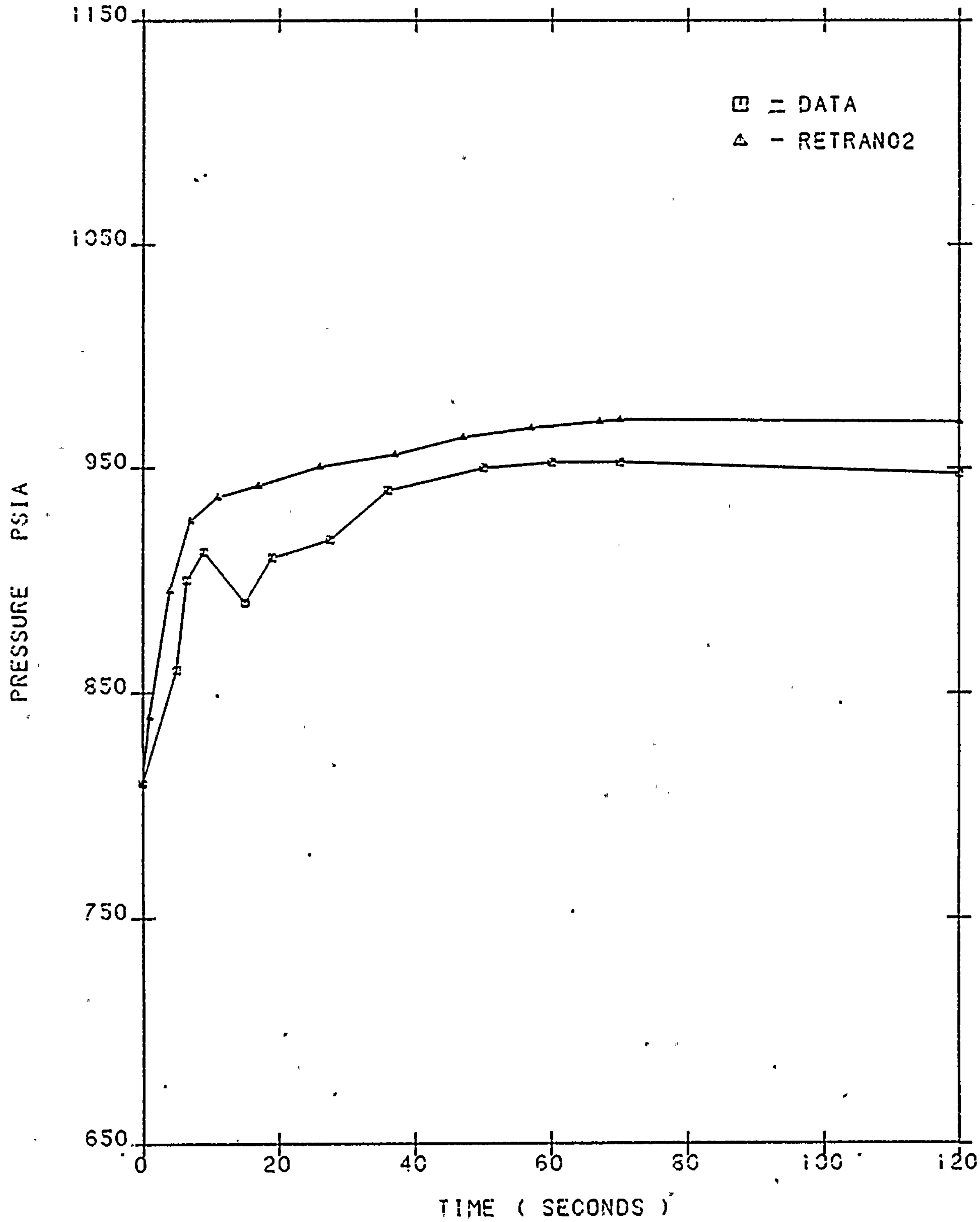

FIGURE 3.3.1 STEAM GENERATOR PRESSURE

GENERATOR TRIP TEST

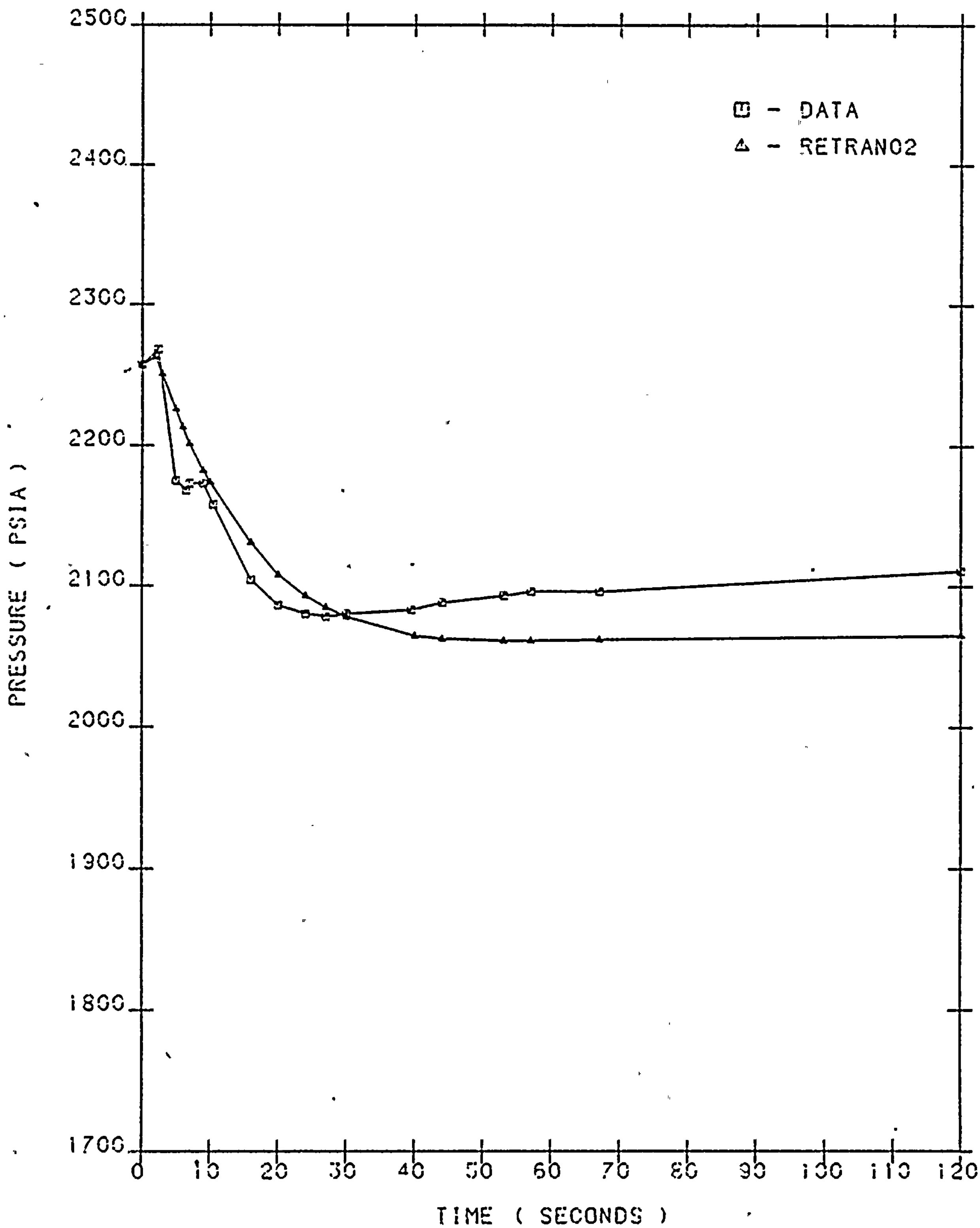

FIGURE 3.3.2 PRESSURIZER PRESSURE

GENERATOR TRIP TEST

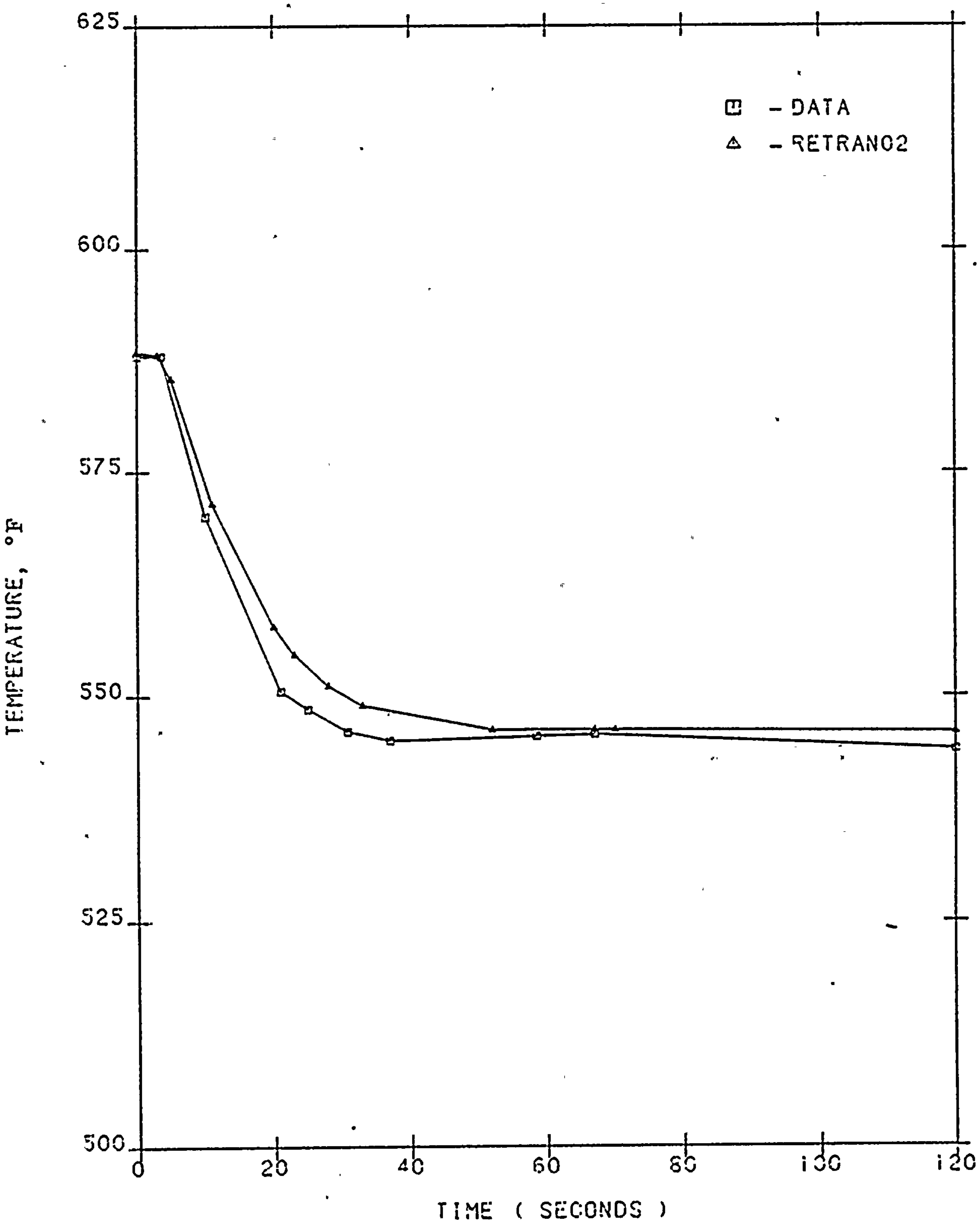

FIGURE 3.3.3 EXIT CORE COOLANT TEMPERATURE

GENERATOR TRIP TEST

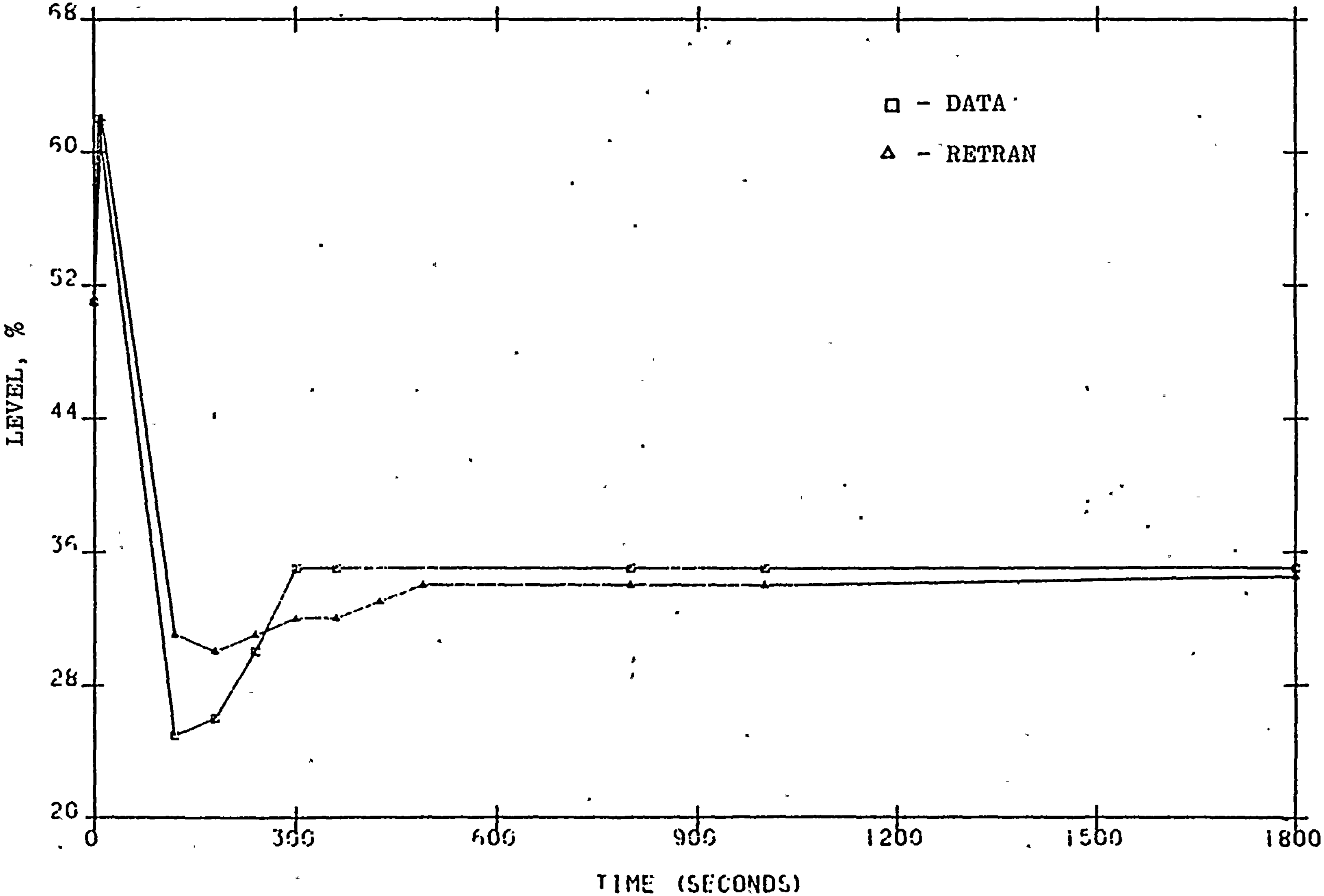

FIGURE 3.4.1 PERCENT PRESSURIZER LEVEL

MSIV CLOSURE EVENT

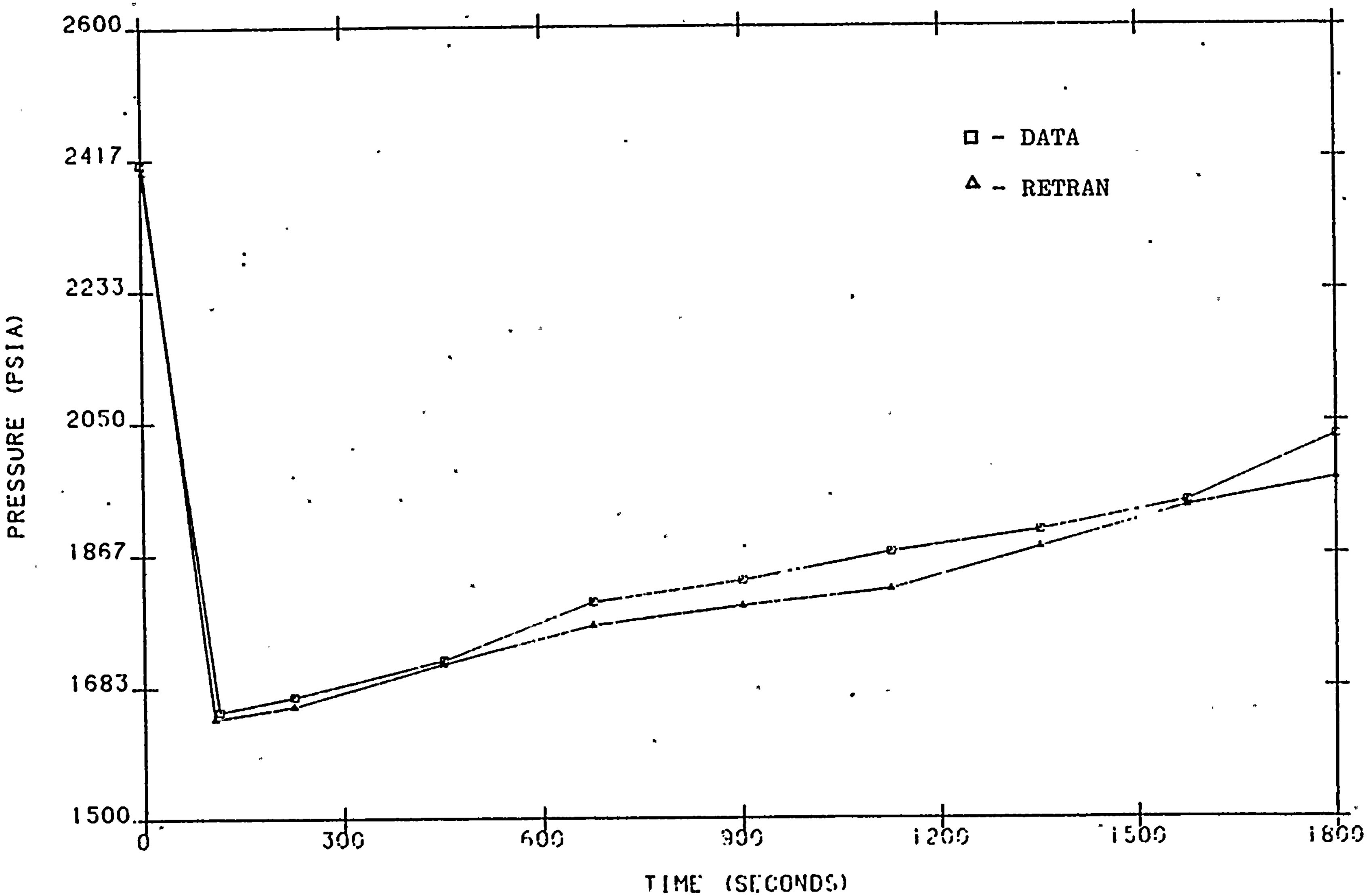

FIGURE 3.4.2 PRESSURIZER PRESSURE

MSIV CLOSURE EVENT

4.0 CHANGE IN PRIMARY COOLANT SYSTEM INVENTORY

A decrease in RCS inventory results from a failure of the RCS pressure boundary. Reductions in coolant inventory could lead to fuel damage due to a reduction in core heat removal capability. The St. Lucie Unit 2 Inadvertent PORV Opening transient (Section 4.1) benchmarked to FSAR data is presented in this category. The important modeling considerations are the rate of change of coolant inventory and plant protection system functions.

4.1 St. Lucie Unit 2 Inadvertent Opening of the PORVs

4.1.1 Transient Description

In this transient the two PORVs are assumed to open resulting in the loss of RCS coolant inventory to containment and the rapid depressurization of the RCS. The RETRAN analysis is benchmarked to the corresponding St. Lucie 2 FSAR analysis (Ref. 14). The reactor trips on a low pressurizer pressure signal. A loss of offsite power (LOAC) is assumed to occur following the turbine trip which results from the reactor scram. The reactor coolant pumps and main feedwater pumps therefore lose power and begin to coast down.

The steam bypass and steam dump valves are assumed to fail closed, such that the steam generator secondary pressure rises to

the main steam safety valve setpoints.

This transient was chosen to demonstrate the adequacy of the RETRAN non-equilibrium pressurizer model in the analysis of postulated depressurization events. It also serves to validate the modelling of the PORVs.

4.1.2 RETRAN Analysis Description

The transient was analyzed with the RETRAN02 computer code. Initial conditions for the RETRAN analysis are presented in Table 4.1.1. The status of the plant safety systems assumed in this transient are shown in Table 4.1.2.

The only significant change made to the RETRAN base model of St. Lucie Unit 2 was to adjust the discharge coefficients associated with the PORVs in order to obtain the manufacture's maximum flow rate through them at the rated core power. This was done in order to achieve the most rapid depressurization of the RCS.

4.1.3 Results

A comparison of the sequence of events in the FSAR and RETRAN analyses is presented in Table 4.1.3. Figures 4.1.1,

4.1.2 and 4.1.3 present respectively the RCS temperature, pressurizer pressure, and steam generator secondary pressure variation for the first 120 seconds of the transient. Good agreement is seen throughout. Small differences in the pressurizer pressure and RCS temperature after 65 seconds are due to small differences in the energy removal rate through the steam generator safety valves.

TABLE 4.1.1

INITIAL CONDITIONS AND KEY PARAMETERS

INADVERTENT OPENING OF THE PORVs

PARAMETER	VALUE
Core Power, MW (Thermal)	2754
Core Inlet Coolant Temperature, °F	552
Pressurizer Water Volume, ft^3	61.2
Pressurizer Pressure, psia	2300
Moderator Temperature Coefficient, $10^{-4}\Delta k/°F$	-2.7
CEA Worth for Trip, % Δk	-5.5

TABLE 4.1.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

INADVERTENT OPENING OF THE PORVs

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System		X	
Main Steam Safety Valves	X		
Pressurizer Safety Valves		X	
Main Steam Isolation	X		
Main Feedwater Isolation	X		
Auxiliary Feedwater System		X	
Safety Injection System			
HPSI	X		
Accumulators			X
LPSI			X
Atmospheric Dump Valve Systems		X	
Steam Dump and Bypass System		X	
Pressurizer Level Control System		X	
Pressurizer Power Operated Relief Valves (PORV)	X		
Chemical and Volume Control System			X
S. G. Level Control System			X
Automatic Rod Motion			X

TABLE 4.1.3

SEQUENCE OF EVENTS

INADVERTENT OPENING OF THE PORVs

EVENT	TIME		PARAMETER	
	VENDOR	RETRAN	FSAR	RETRAN
Inadvertent Opening of PORVs	0.0	0.0	--	--
Reactor Trip on TM/LP Signal	21.6	22.2	1799 psia	1799 psia
Turbine Trip	22.8	22.5	--	--
LOAC Assumed	24.8	24.7	--	--
Main Steam SRVs Open	26.2	24.9	990 psia	990 psia
SI Actuation Setpoint Reached	29.6	28.2	1648 psia	1648 psia
HPSI Pumps at Full Speed	59.6	58.2	--	--

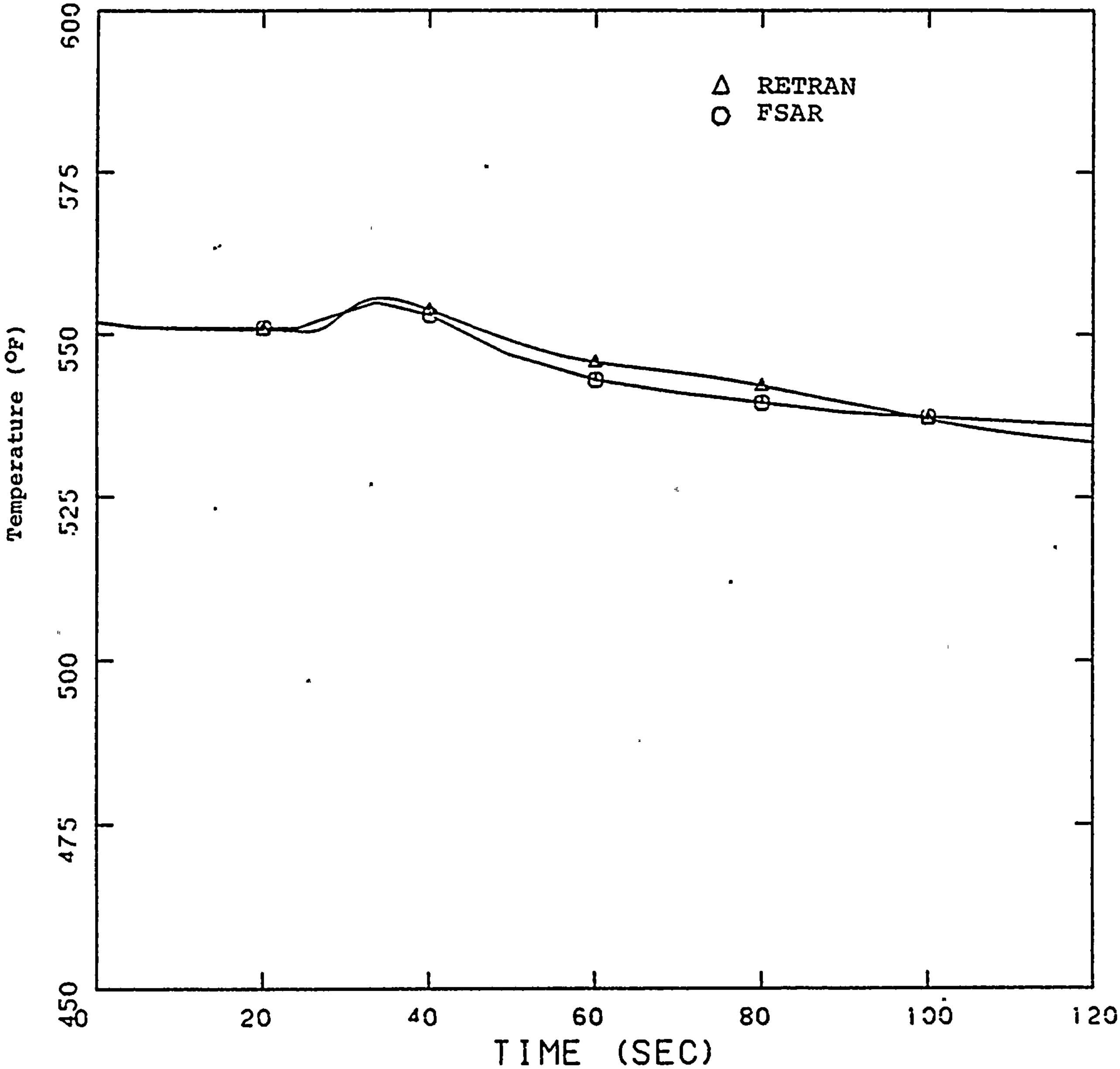

FIGURE 4.1.1 INLET CORE COOLANT TEMPERATURE

INADVERTENT OPENING OF THE PORVs

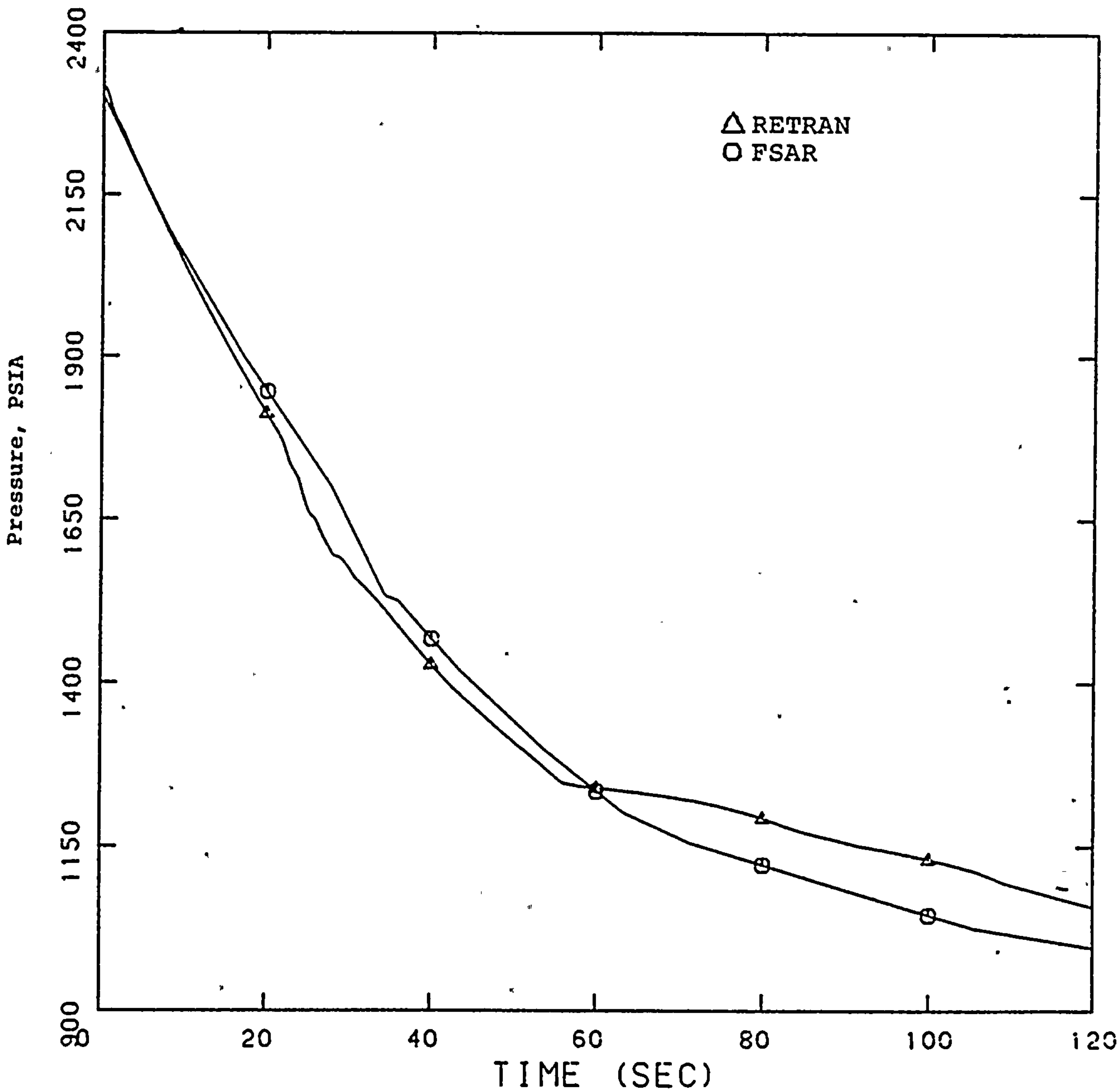

FIGURE 4.1.2 PRESSURIZER PRESSURE

INADVERTENT OPENING OF THE PORVs

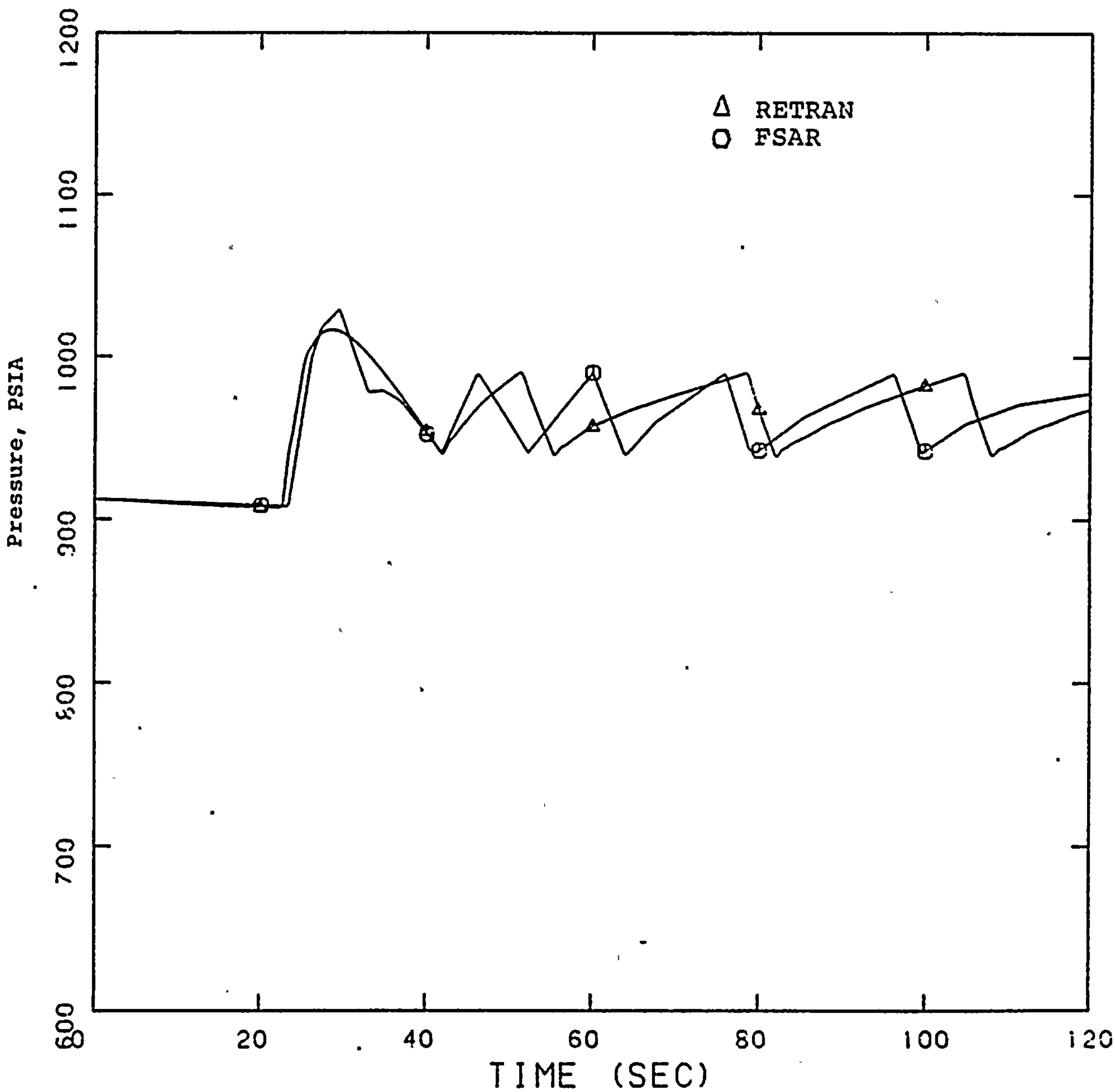

FIGURE 4.1.3 STEAM GENERATOR PRESSURE

INADVERTENT OPENING OF THE PORVs

5.0 LOSS OF REACTOR COOLANT SYSTEM FLOW

A reduction of core coolant flow rate reduces heat removal from the fuel and can result in exceeding specified acceptable fuel design limits. Transients may involve either a partial or complete loss of forced RCS flow. Important modeling considerations are reactivity coefficients, pump coastdown characteristics, and reactor protective system settings.

Analyses presented in this category are the Turkey Point Pump Coastdown Test benchmarked to plant data (Section 5.1), and the St. Lucie Unit 1 Loss of Flow Transient benchmarked to FSAR results (Section 5.2).

5.1 Turkey Point Pump Coastdown Test

5.1.1 Transient Description

During preoperational testing for Turkey Point Unit 4, a number of tests were conducted in which one, two and three reactor coolant pumps were simultaneously tripped and coolant flow measurements were made. These pump coastdown tests served to evaluate the response of the system to a total or partial loss of forced RCS flow. The benchmarking of the test results with the RETRAN model serves to validate the flow resistances (frictional and form losses) of the primary loop and the RCS pump characteristics of the model.

5.1.2 RETRAN Analysis Description

The analysis was performed with RETRAN01 before the 02 version became available. The Turkey Point RETRAN01 model is essentially identical to the RETRAN02 model. Best estimate parameters taken from test data and incorporated in the model are presented in Table 5.1.1. The status of safety systems in the RETRAN01 model are summarized in Table 5.1.2.

5.1.3 Results

The measured and calculated fraction of total RCS flow following a trip of all three pumps is shown in Figure 5.1.1. The RETRAN values compare well with the test data which shows that the flow resistance and pump characteristics are accurately modeled. In the two-of-three-pumps-tripped analysis, the two pumps that are tripped are represented in the model by Loop B (See Appendix B) and the remaining pump that continues to run is represented by Loop A. The fraction of flow in Loop A (Figure 5.1.2) increases due to the increased loop flow resistance in Loop B. There is good agreement between RETRAN results and the test data for core and loop flow. Figure 5.1.3 shows similar curves for the one of three pumps tripped analysis. The single pump in Loop A is tripped and the pumps in Loop B (representing two pumps) continue to run. Similarly, the flow in Loop B increases due to the increase in flow resistance in Loop A as the pump in this loop coasts down.

5.2 St. Lucie Unit 1 Loss of Forced Flow

5.2.1 Transient Description

The St. Lucie Unit 1 model was benchmarked against the Loss of Coolant Flow Event described in the St. Lucie Unit 1 FSAR (Ref. 5). The purpose of the benchmark was to validate pump characteristics, loop hydraulic resistances, trip characteristics and reactor protection features of the St. Lucie Unit 1 RETRAN base deck. Core flow, core heat flux, low-flow trip, and pressurizer pressure are key parameters in this transient which are compared with those of the FSAR analyses. The Loss of Coolant Flow Event assumes that all four reactor coolant pumps lose their power supply and the core is deprived of the required coolant flow to remove the heat generated in the fuel. The pump flywheels provide the necessary inertia to the rotary elements for the pumps to coast down slowly and the flow to decrease gradually over a period of several seconds. As the core flow decreases, the critical heat flux decreases which can result in departure from nucleate boiling (DNB) if the reactor is not tripped. The trip occurs on low coolant flow.

5.2.2 RETRAN Analysis Description

The plant's Reactor Protection System modelled in the

RETRAN02 base deck generates a reactor trip signal when the total pump flow decreases below 93% of the nominal value. A delay of 0.65 sec. is assumed between the time the signal is generated and the time the trip breakers open. This delay, together with a 0.5 sec. delay before the Control Element Assemblies (CEAs) enter the core, account for delays in signal actuation, opening time of the trip breakers and the release of the rods. The scram reactivity curve for the analysis was generated at FPL with the assumption of the most reactive CEA stuck in the fully withdrawn position. The reactivity worths of CEA banks were taken to be at their minimum allowable value. The scram reactivity curve was obtained by combining the reactivity worths of shutdown banks minus the most reactive CEA. The fuel gap conductivity was taken at a reasonably low value (approx. 500 BTU/hr ft^2 ^oF) to obtain conservatively high values of core heat flux. The FSAR and RETRAN calculated initial conditions and key parameters are presented in Table 5.2.1.

The charging and letdown systems, and the pressurizer heaters in the RETRAN base deck were assumed inactive for this analysis. The status of safety systems in the RETRAN model for this transient is summarized in Table 5.2.2.

5.2.3 Results

The results of the RETRAN analysis for the loss of flow

transient are shown in Figures 5.2.1 through 5.2.4 together with FSAR predictions. A comparative sequence of events is presented in Table 5.2.3. The RETRAN results for the key parameters such as core flow, core heat flux, time of reactor trip on low core flow signal and pressurizer pressure are in good agreement with those shown in the FSAR.

TABLE 5.1.1

INITIAL CONDITIONS

PUMP COASTDOWN TEST

PARAMETER	VALUE
Core Power, MW (Thermal)	0
Average RCS Temperature, °F	
Three of Three Pumps	550
Two of Three Pumps	543
One of Three Pumps	547
Vessel Mass Flow Rate, GPM	281000
Pressurizer Pressure, psia	2254
Steam Generator Pressure, psia	800

TABLE 5.1.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

PUMP COASTDOWN TEST

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)		X	
Pressurizer Pressure Control System		X	
Main Steam Safety Valves		X	
Pressurizer Safety Valves		X	
Main Steam Isolation Valves		X	
Main Feedwater Isolation Valves		X	
Auxiliary Feedwater System		X	
Safety Injection System			
HPSI		X	
Accumulators			X
LPSI		X	
Atmospheric Dump Valve Systems			X
Steam Dump and Bypass System			X
Pressurizer Level Control System		X	
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System			X
S. G. Level Control System			X

TABLE 5.2.1

INITIAL CONDITIONS AND KEY PARAMETERS

LOSS OF FORCED FLOW

PARAMETER	VALUE
Core Power, MW (Thermal)	2700.
Core Inlet Coolant Temperature, °F	549.
Core Flow (10^6 lbm/hr)	138.3
Pressurizer Pressure, psia	2225
Steam Generator Pressure, psia	900
Low Flow Signal (% of nominal)	93
CEA Worth at trip, % Δk	-5.60
Moderator Temperature Coefficient, $10^{-5}\Delta k/°F$	+5.0
Doppler Temperature Coefficient, $10^{-5}\Delta k/°F$	-1.33

TABLE 5.2.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL.

LOSS OF FORCED FLOW

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System		X	
Main Steam Safety Valves		X	
Pressurizer Safety Valves		X	
Main Steam Isolation Valves		X	
Main Feedwater Isolation Valves		X	
Auxiliary Feedwater System			X
Safety Injection System			
HPSI		X	
Accumulators			X
LPSI			X
Atmospheric Dump Valve Systems			X
Steam Dump and Bypass System			X
Pressurizer Level Control System			X
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System			X
S. G. Level Control System			X
Automatic Rod Motion			X

TABLE 5.2.3

SEQUENCE OF EVENTS

LOSS OF FORCED FLOW

EVENT	TIME		PARAMETER	
	FSAR	RETRAN	FSAR	RETRAN
Loss of Power to All Four Pumps	0.	0.	--	--
Low Flow Trip Signal	0.86	0.83	93%	93%
Trip Breakers Open	1.51	1.48	--	--
Maximum Core Power	1.91	1.97	102.5%	103.25%
CEA Begin to Drop Into Core	2.01	1.98	--	--
Maximum Core Outlet Temperature	4.35	3.68	610.9°F	608.7°F
Maximum Pressurizer Pressure	5.26	5.37	2326 psia	2314 psia
Turbine Trip	--	1.69	--	--

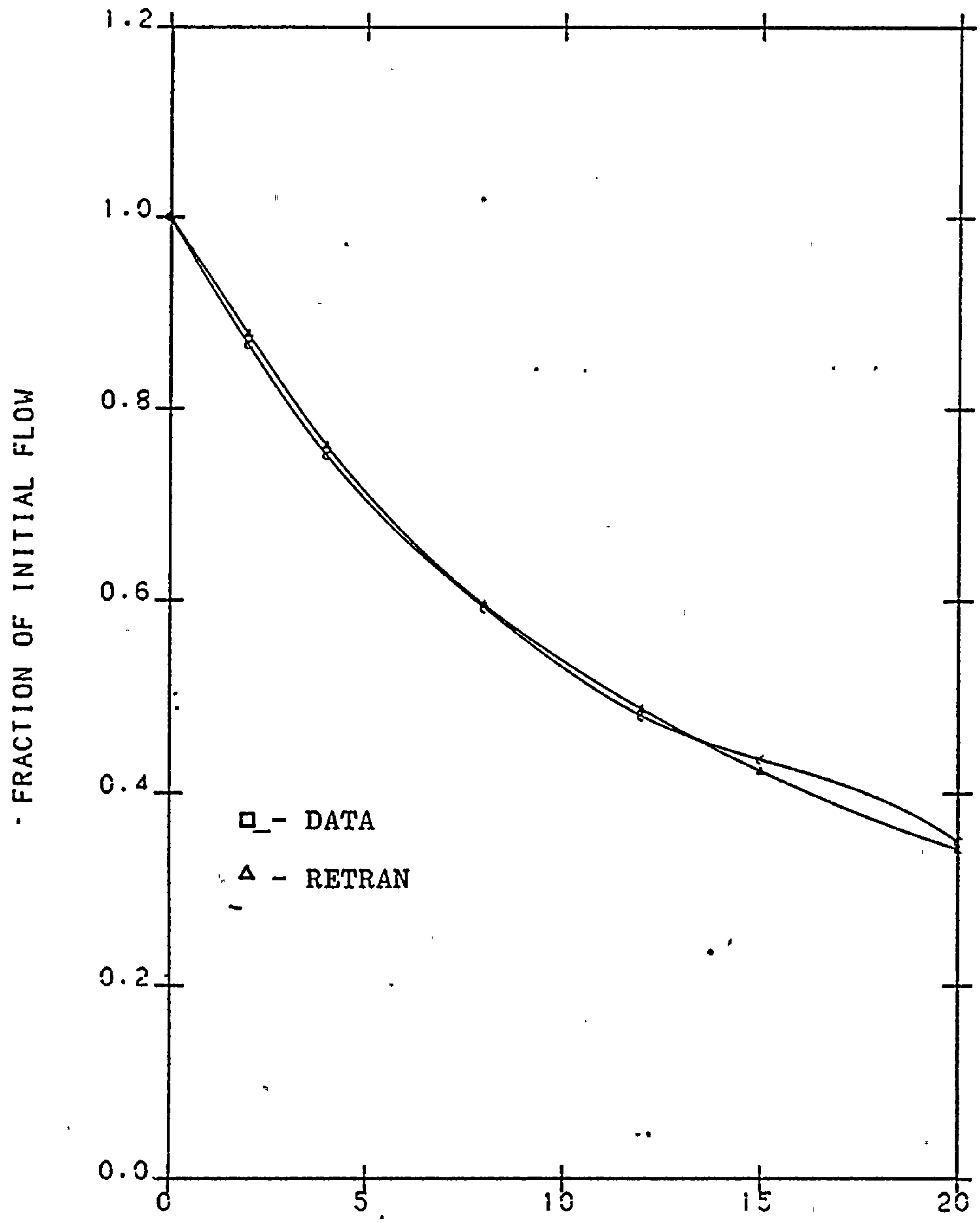

FIGURE 5.1.1 NORMALIZED RCS COOLANT FLOW RATE,

THREE PUMP COASTDOWN

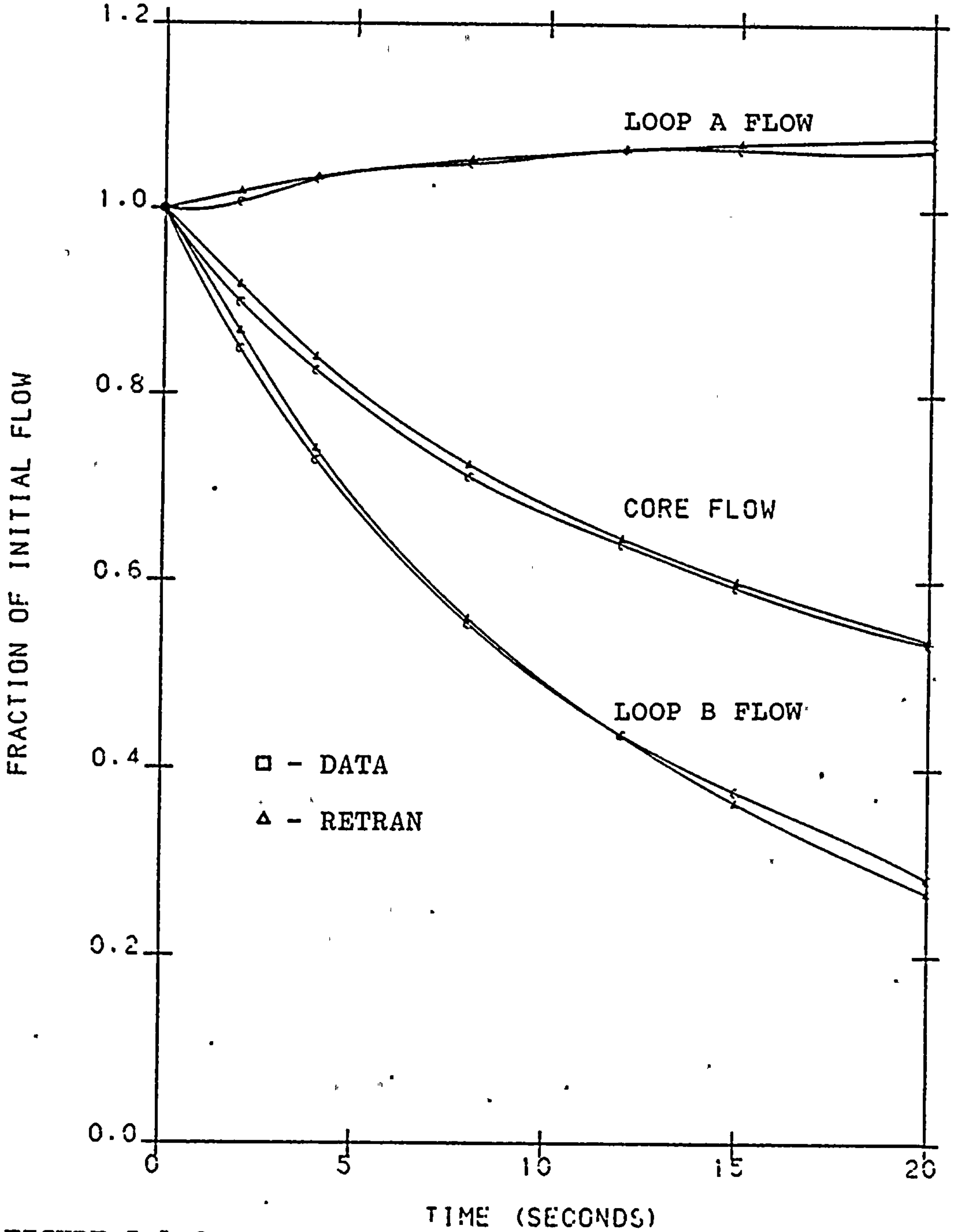

FIGURE 5.1.2 NORMALIZED FLOW RATES FOR LOOPS AND CORE, TWO PUMP COASTDOWN

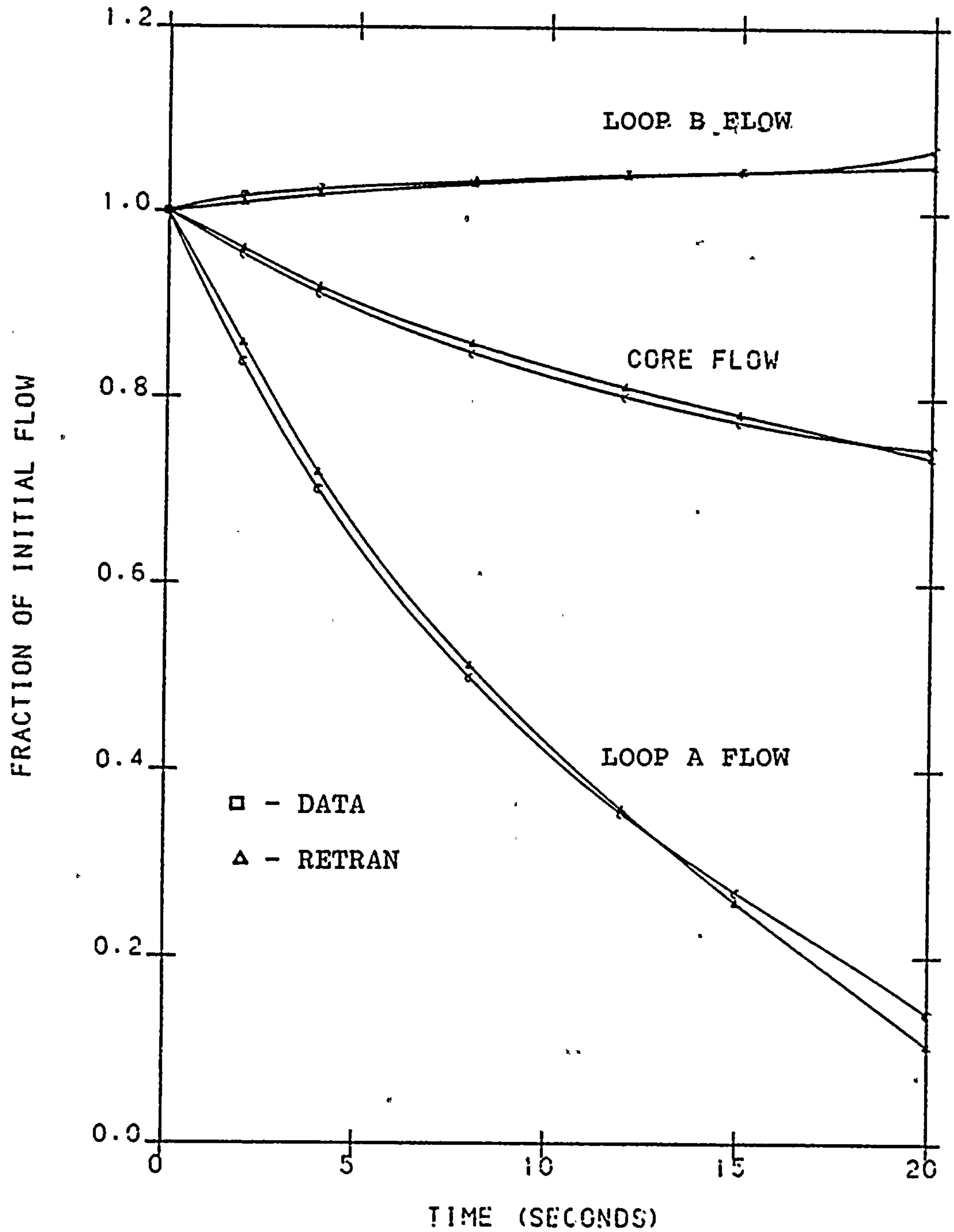

FIGURE 5.1.3 NORMALIZED FLOW RATES FOR LOOPS AND CORE, ONE PUMP COASTDOWN

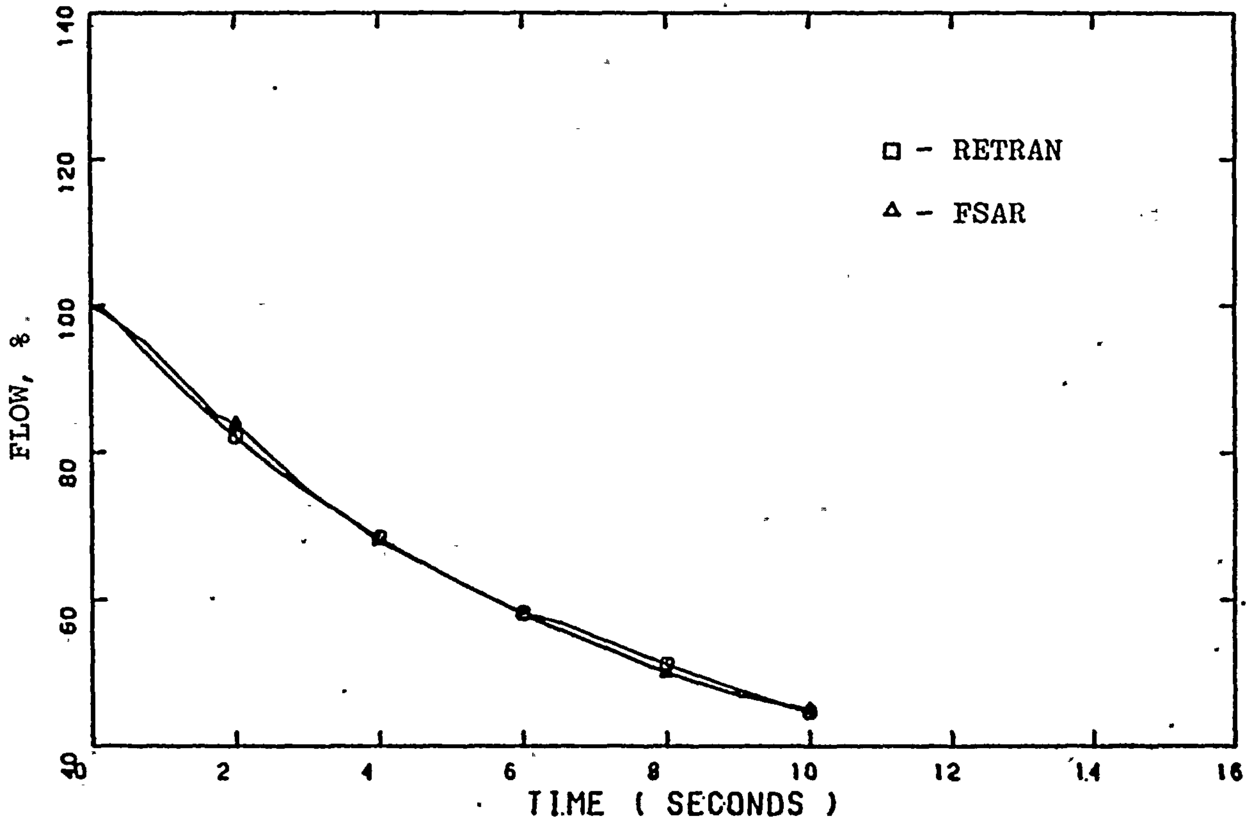

FIGURE 5.2.1 PERCENT CORE FLOW

LOSS OF FORCED FLOW

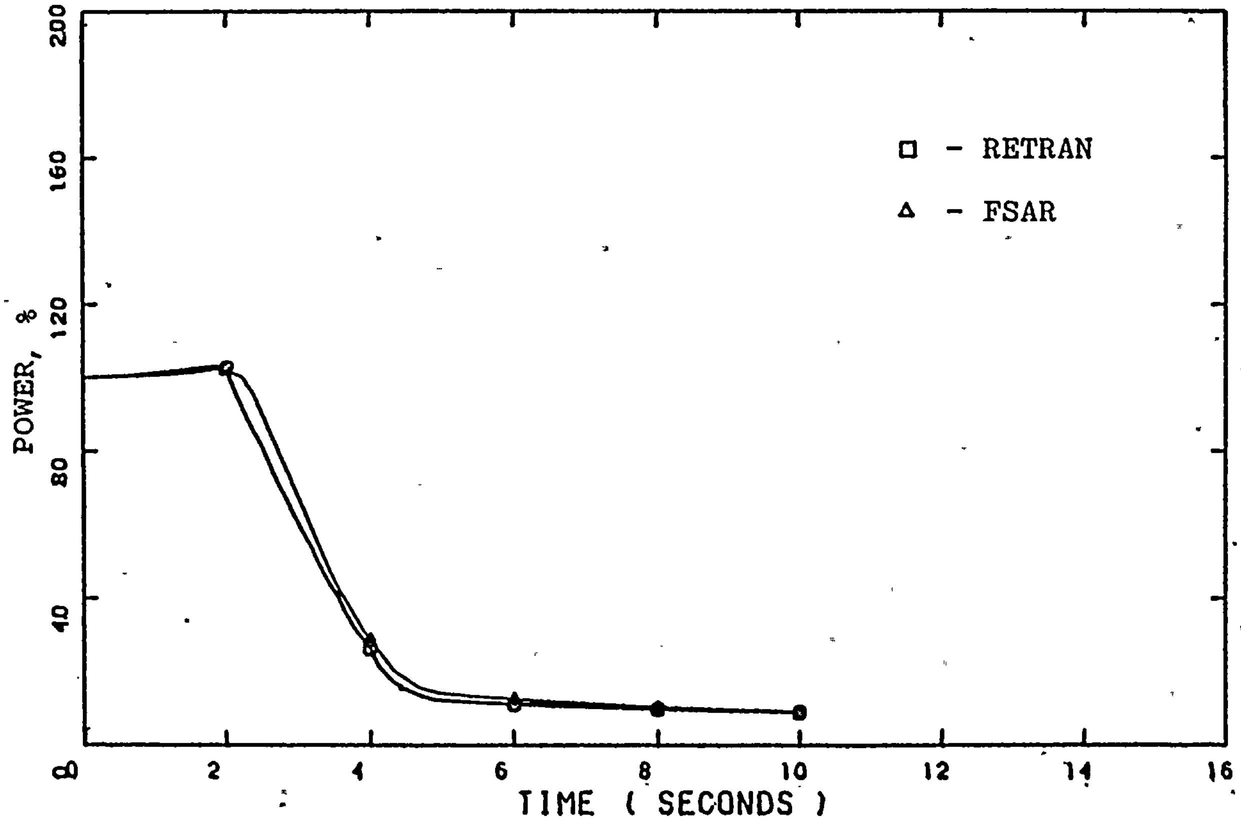

FIGURE 5.2.2 PERCENT CORE POWER

LOSS OF FORCED FLOW

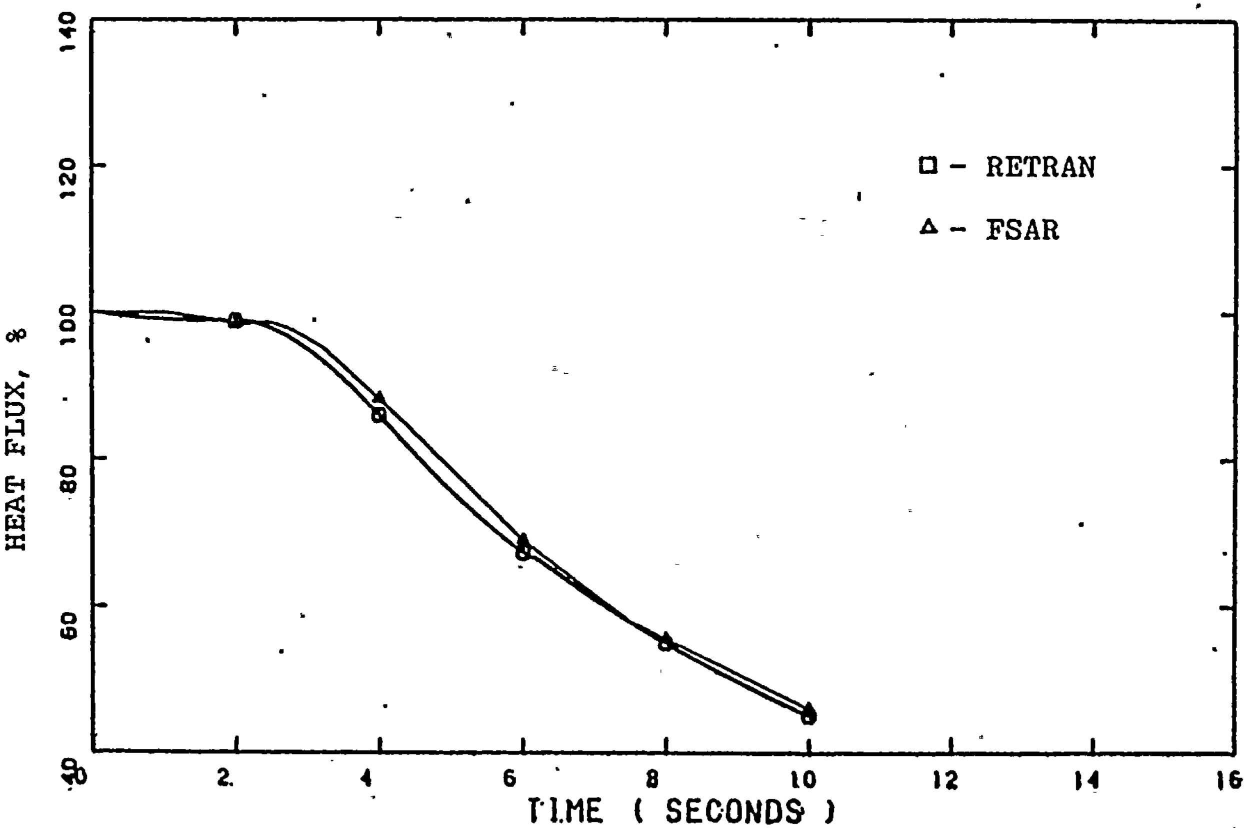

FIGURE 5.2.3 PERCENT CORE HEAT FLUX

LOSS OF FORCED FLOW

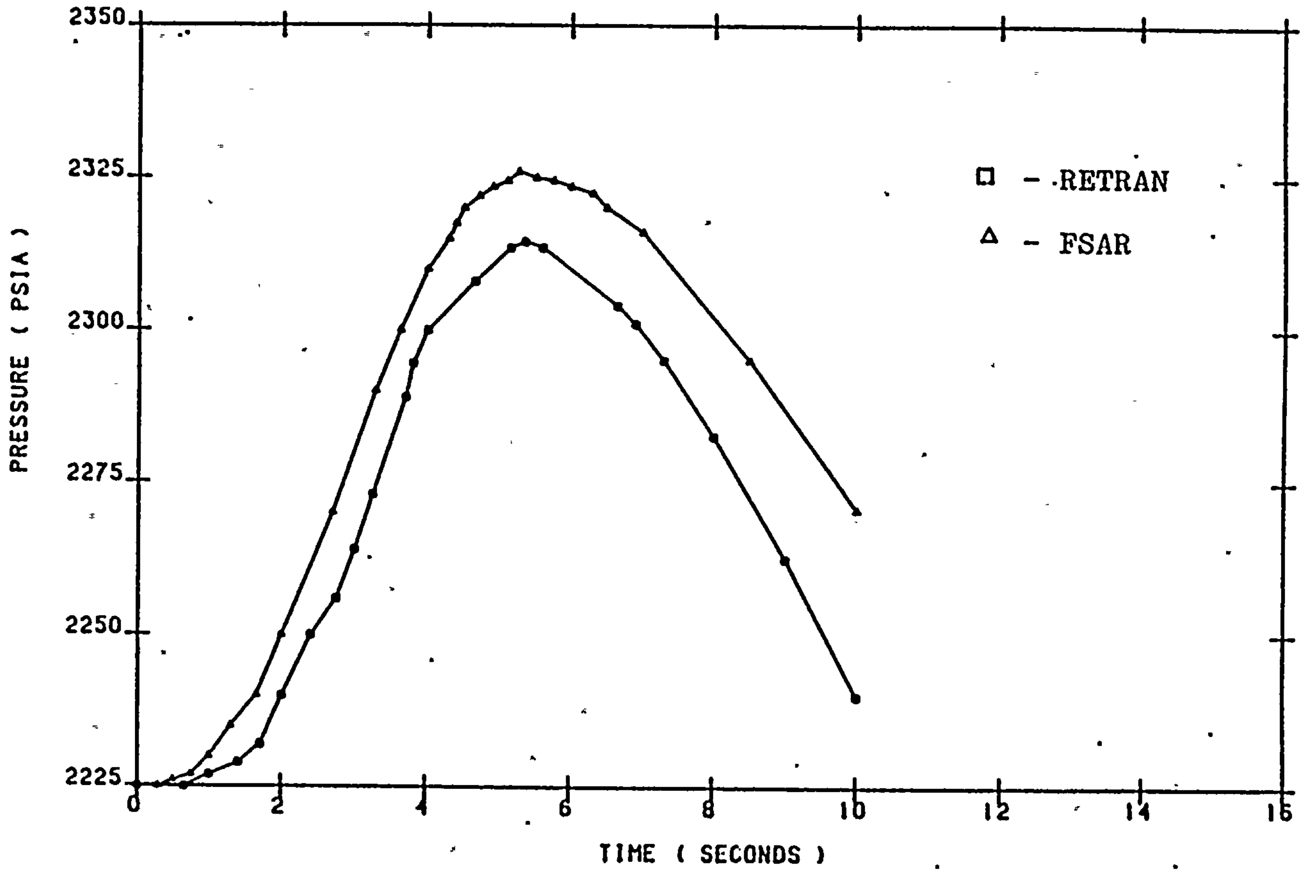

FIGURE 5.2.4 PRESSURIZER PRESSURE

LOSS OF FORCED FLOW

6.0 REACTIVITY INSERTION

Events in this category involve localized reactivity additions which cause anomalies in the core power distribution. Important modeling considerations are the reactor protection system, reactor kinetics and reactivity feedback coefficients. Analyses presented in this category are the Turkey Point Uncontrolled RCCA Withdrawal transient benchmarked to FSAR results (Section 6.1), and the St. Lucie Unit 2 CEA Drop transient benchmarked to FSAR results (Section 6.2).

6.1 Turkey Point Uncontrolled RCCA Withdrawal

6.1.1 <u>Transient Description</u>

A slow, uncontrolled rod cluster control assembly (RCCA) withdrawal transient from 100% power was simulated with the RETRAN02 computer code and benchmarked to the analogous transient documented in the Turkey Point FSAR. (Ref. 13). In this transient the rod withdrawal causes an increase in core power and heat flux which result in increases in RCS temperature and pressure. Reactor trip can occur on high RCS pressure, high pressurizer level or on exceeding the high power, overpower ΔT or overtemperature ΔT setpoints. This transient assesses, the adequacy of the RETRAN reactor kinetics modeling and the modeling of the reactor protection system.

6.1.2 RETRAN Analysis Description

The initial conditions of the benchmark and RETRAN02 analysis, presented in Table 6.1.1, were incorporated into the Turkey Point RETRAN base model (see Appendix B). These initial conditions represent beginning of cycle conditions as listed in the Turkey Point FSAR. The analysis was performed for a rod withdrawel rate of $2.5\times10^{-5}\Delta k$/sec. For this case the reactor trips on overtemperature ΔT. Presented in Table 6.1.2 is the status of safety systems included in the RETRAN simulation of this transient.

6.1.3 Results

Results of the RETRAN02 calculation and the FSAR are presented in Figures 6.1.1, 6.1.2 and 6.1.3. A sequence of events for both the RETRAN calculation and the FSAR results is shown in Table 6.1.3. As the core power increases the sensed temperature difference between the hot leg and cold leg reaches the dynamic overtemperature ΔT setpoint, when the scram signal is generated and the reactor trips. The turbine trips on the reactor trip. RETRAN predicts the reactor trip about 3 seconds later than the FSAR analysis but maximum core power, maximum RCS pressure and temperature calculated for the RETRAN and FSAR analyses are essentially identical. Overall, the RETRAN simulation shows good agreement with the FSAR analysis results.

6.2 St. Lucie Unit 2 CEA Drop

6.2.1 Transient Description

The CEA drop event performed with the RETRAN02 code has been benchmarked against the CEA drop analyzed in the St. Lucie 2 FSAR (Ref. 14).

The CEA Drop Event is defined as the inadvertent release of a single or subgroup CEAs causing it to drop into the core. The occurrence of an electrical or mechanical failure in a CEA drive mechanism would result in a CEA drop.

The CEA drop event causes an initial decrease in reactor power. Since the heat extraction remains relatively constant, the average reactor coolant temperature decreases. The effect of the decrease in temperature in conjunction with a large negative moderator temperature coefficient, is to return the reactor to its initial power level at a slightly reduced core inlet temperature. Additionally, the core power distribution is distorted due to the CEA insertion.

The Limiting Conditions for Operation (LCO) are designed to maintain a DNB ratio sufficiently above the design limit without the necessity for a reactor trip during a CEA drop event.

Operation of the detection systems which are designed to sense a CEA drop event and to reduce turbine load to a preset value are not assumed in this analysis. The action of the protection system to inhibit CEA withdrawal during a CEA drop event has been credited.

6.2.2 RETRAN Analysis Description

The single full length CEA drop transient was simulated with the RETRAN02 computer code. The assumptions and initial conditions were based on the St. Lucie Unit 2 FSAR (Ref. 14) parameters as shown in Table 6.2.1. The St. Lucie Unit 2 RETRAN base model was initialized at 90% of rated power. The reactively worth of the dropped rod was taken to be $-0.074 \times 10^{-2} \Delta k$ in accordance with the FSAR analysis. The safety systems actuated during this transient are shown in Table 6.2.2.

6.2.3 Results

The results of the RETRAN calculation compared to the FSAR results are presented in Figures 6.2.1 to 6.2.4. A comparison of results is shown in Table 6.2.3.

The single CEA drop results in a negative reactivity insertion, initially causing a drop in core power. Doppler and moderator feedback eventually bring the core power back to its original level. The RCS pressure and temperature decline slightly as a result of the drop in core power. The RETRAN02 results are reasonable and agree well with FSAR results.

TABLE 6.1.1

INITIAL CONDITIONS AND KEY PARAMETERS

UNCONTROLLED RCCA WITHDRAWAL

PARAMETER	VALUE
Core Power, MW (Thermal)	2244
Core Inlet Coolant Temperature, °F	550.2
Core Mass Flow Rate, 10^6 lbm/hr	101.5
Pressurizer Pressure, psia	2220
Doppler Coefficient, $10^{-4}\Delta k/°F$	-.12
Moderator Temperature Coefficient, $10^{-4}\Delta k/°F$	-.4
Over-Temperature ΔT Above Nominal ΔT Trip setpoint (%)	4
Rod Withdrawal Rate Δk/sec	$2.5X10^{-5}$

TABLE 6.1.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

UNCONTROLLED RCCA WITHDRAWAL

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves		X	
Pressurizer Safety Valves		X	
Main Steam Isolation Valves		X	
Main Feedwater Isolation Valves	X		
Auxiliary Feedwater System		X	
Safety Injection System			
HPSI		X	
Accumulators			X
LPSI		X	
Atmospheric Dump Valve Systems			X
Steam Dump and Bypass System			X
Pressurizer Level Control System			X
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System			X
S. G. Level Control System		X	
Automatic Rod Motion			X

TABLE 6.1.3

SEQUENCE OF EVENTS

UNCONTROLLED RCCA WITHDRAWAL

EVENT	TIME(S) FSAR	TIME(S) RETRAN	PARAMETER FSAR	PARAMETER RETRAN
Rod Withdrawn	0	0	--	--
Reactor Tripped on Over-Temp. ΔT	50.5	55.6	--	--
Turbine Tripped on Reactor Trip	--	56.6	--	--
Maximum Core Power	51	55	113.6%	113.4%
Maximum Pressurizer Pressure	51	55	2332 psia	2322 psia
Maximum Core Average Temperature	52	57	585.8°F	585.6°F

TABLE 6.2.1

INITIAL CONDITIONS AND KEY PARAMETERS

CEA DROP

PARAMETER	VALUE
Power Level, MW (Thermal)	2322
Core Inlet Coolant Temperature, °F	550
Core Mass Flow Rate, 10^6 lbm/hr	139
Pressurizer Pressure, psia	2150
Pressurized Water Level, %	52
Doppler Coefficient, $10^{-4}\Delta k/°F$	-7.56
Moderator Temperature Coefficient, $10^{-4}\Delta k/°F$	-4
Steam Generator Water Level, % of Narrow Range Top Span	70

TABLE 6.2.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

CEA DROP

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves		X	
Pressurizer Safety Valves		X	
Main Steam Isolation Valves		X	
Main Feedwater Isolation Valves		X	
Auxiliary Feedwater System		X	
Safety Injection System			
HPSI			X
Accumulators			X
LPSI			X
Atmospheric Dump Valve Systems			X
Steam Dump and Bypass System			X
Pressurizer Level Control System	X		
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System			X
S. G. Level Control System		X	
Automatic Rod Motion			X

TABLE 6.2.3

COMPARISON OF KEY PARAMETERS

CEA DROP

PARAMETER	RETRAN	FSAR
Minimum Power, %	84.1	84
Minimum RCS Pressure, psia	2126	2128
Maximum Power, %	90.7	93

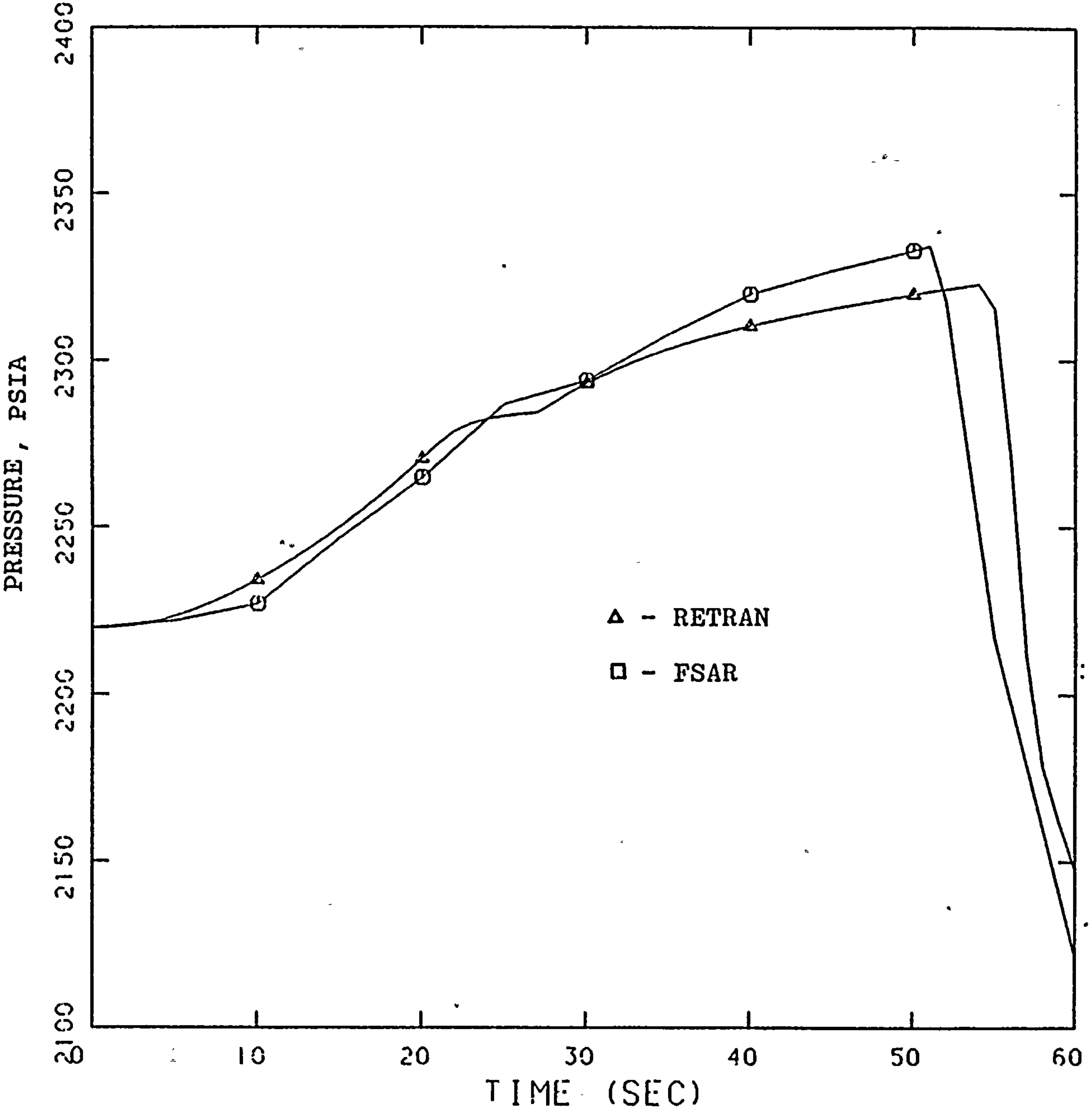

FIGURE 6.1.1 PRESSURIZER PRESSURE
UNCONTROLLED RCCA WITHDRAWAL

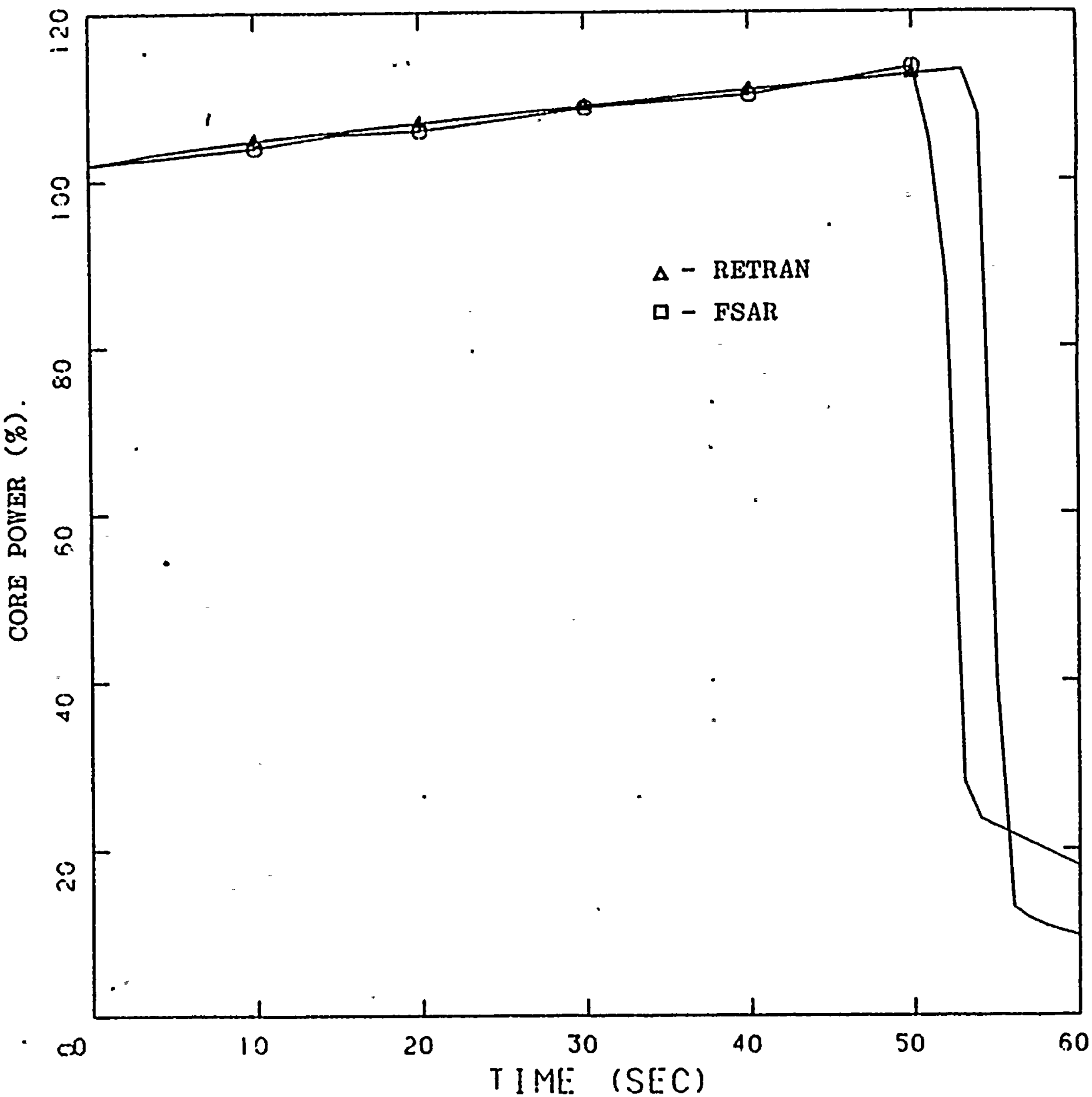

FIGURE 6.1.2 PERCENT CORE POWER
UNCONTROLLED RCCA WITHDRAWAL

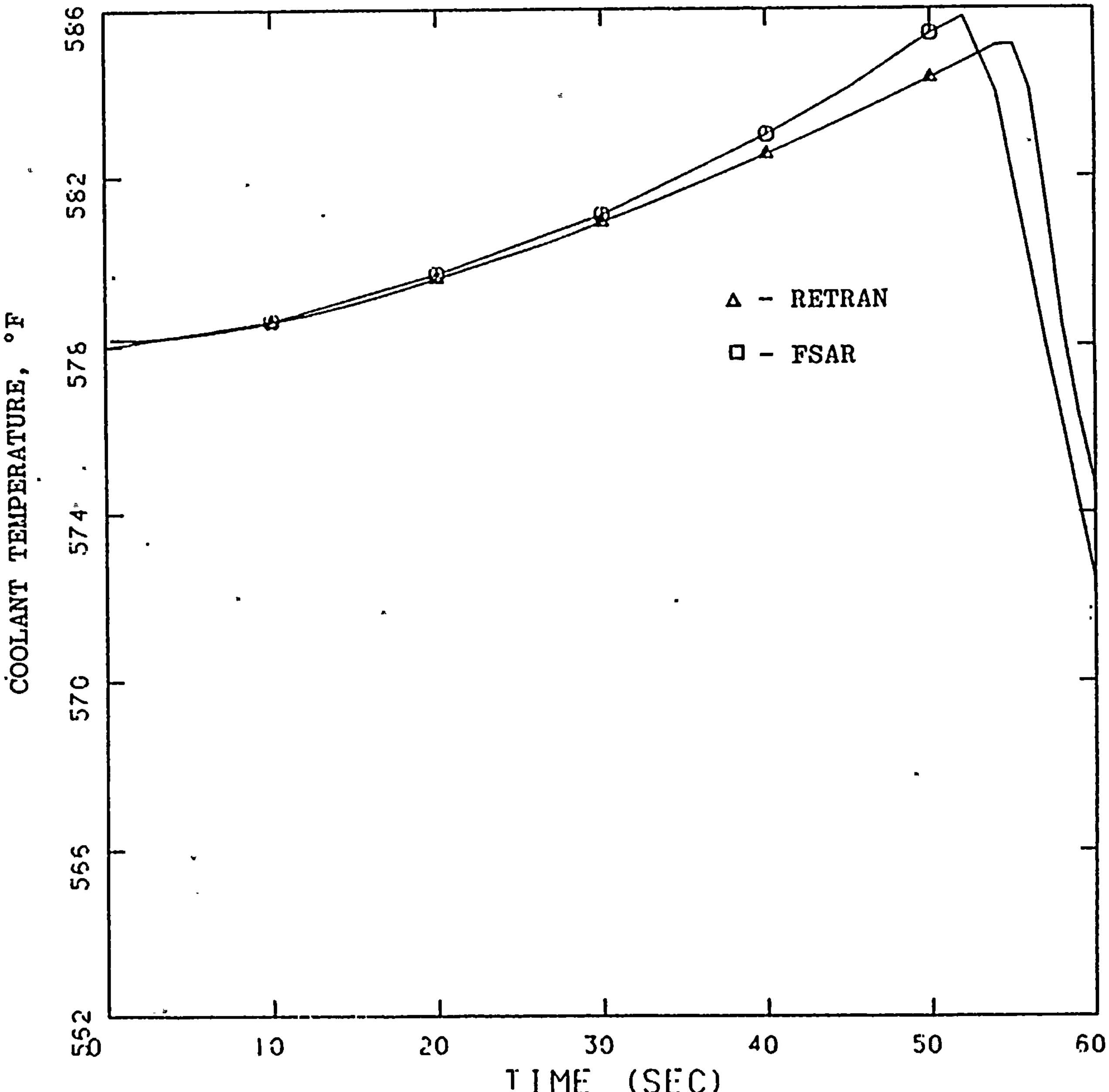

FIGURE 6.1.3 AVERAGE CORE COOLANT TEMPERATURE
UNCONTROLLED RCCA WITHDRAWAL

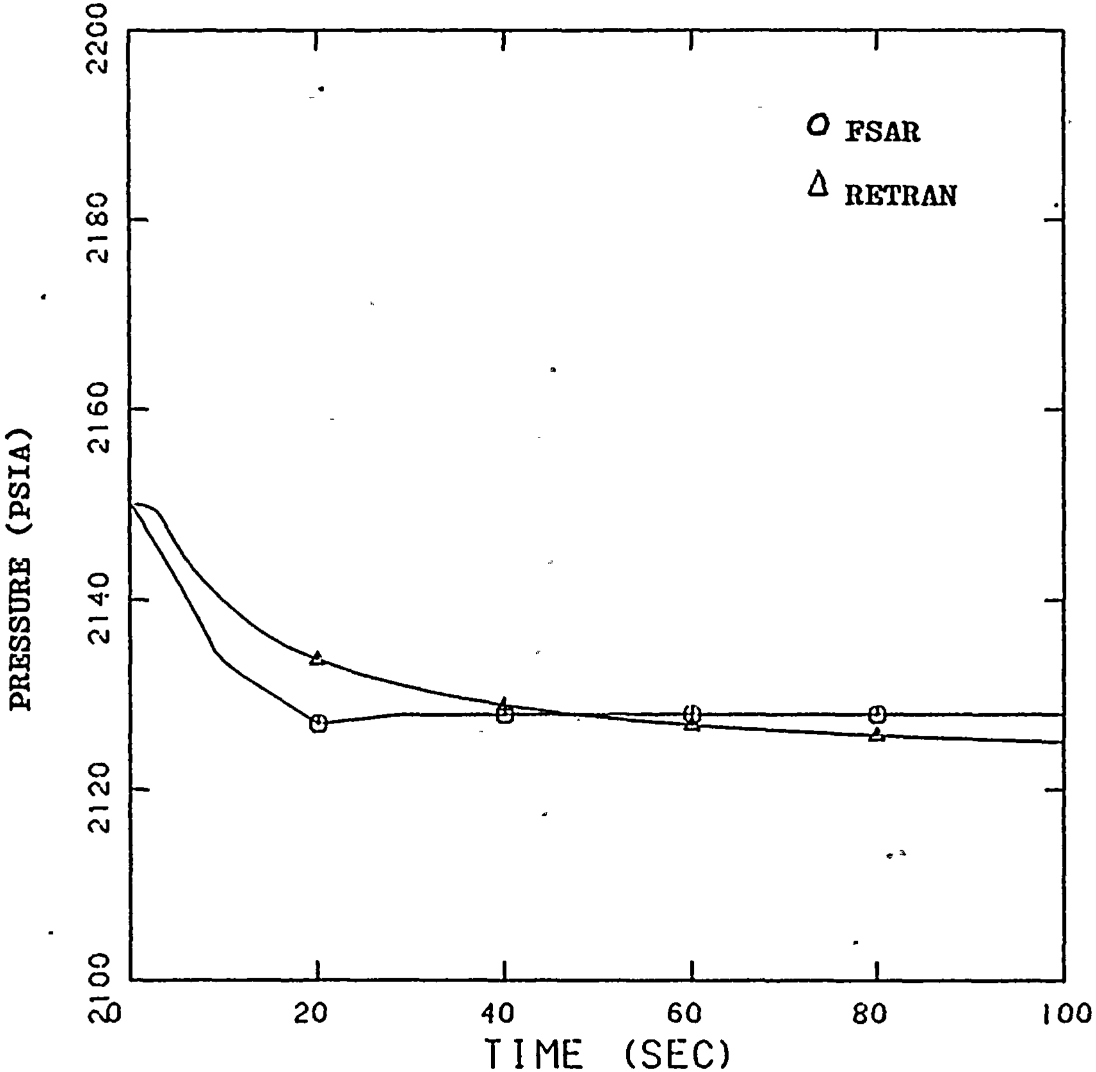

FIGURE 6.2.1 PRESSURIZER PRESSURE

CEA DROP

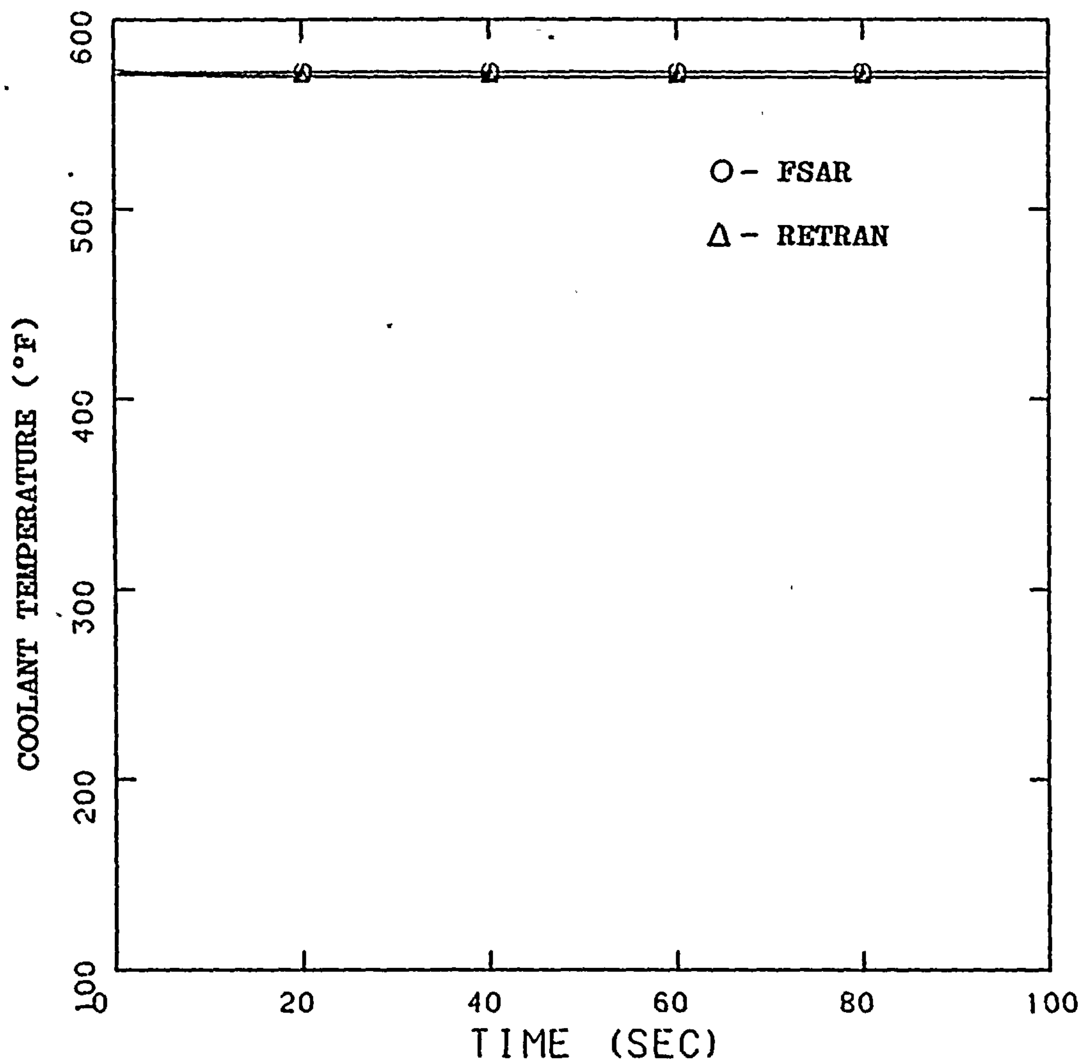

FIGURE 6.2.2 AVERAGE CORE COOLANT TEMPERATURE

CEA DROP

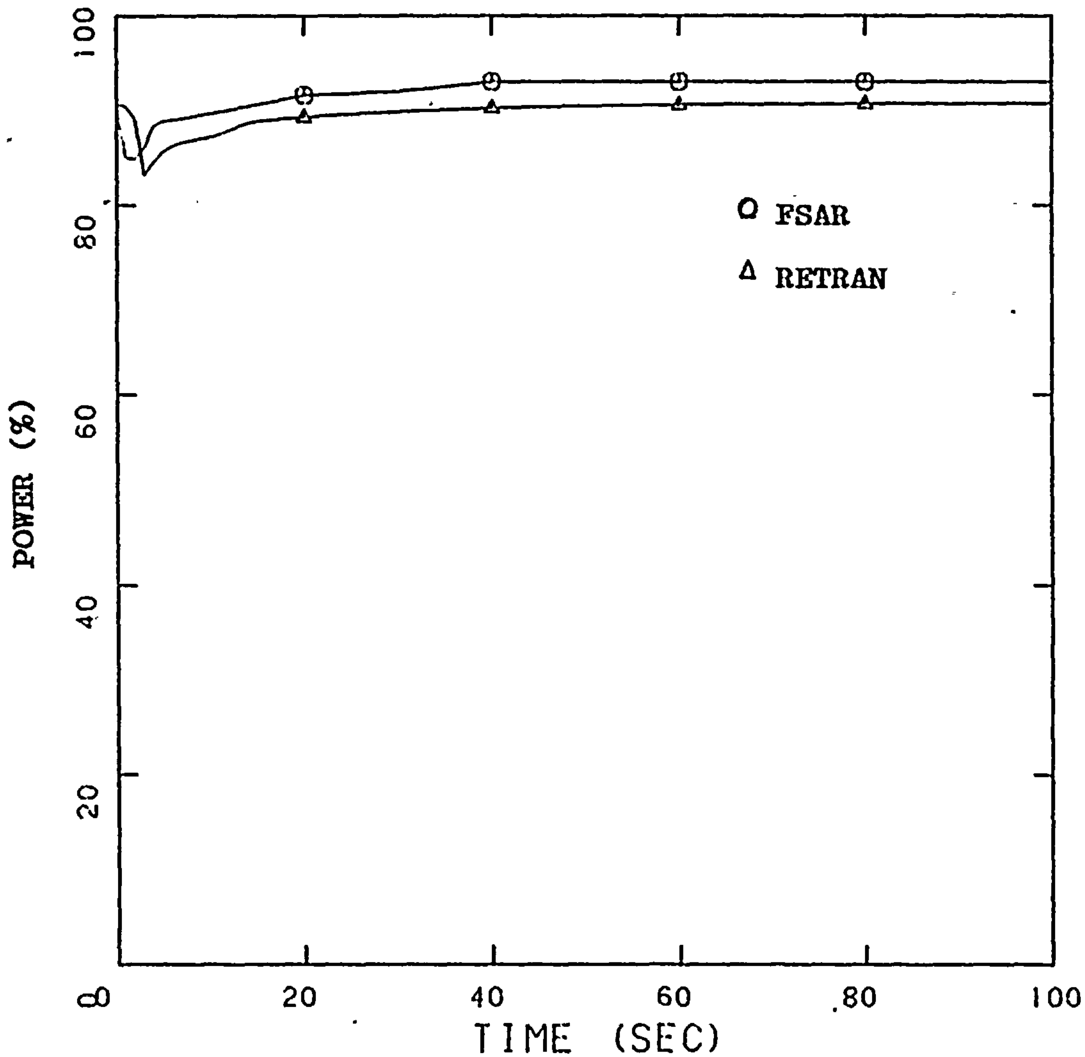

FIGURE 6.2.3 PERCENT CORE POWER
CEA DROP

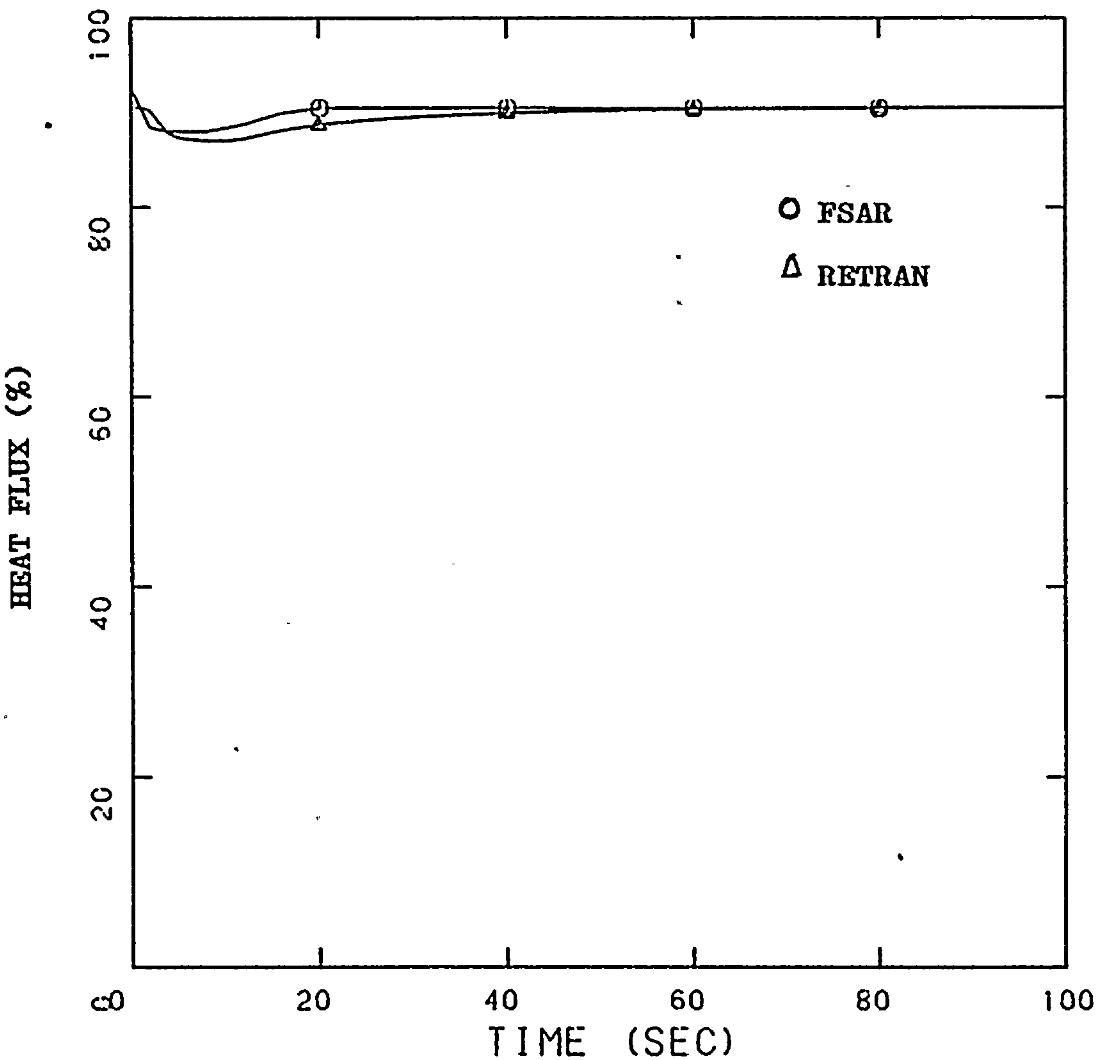

FIGURE 6.2.4 PERCENT CORE HEAT FLUX

CEA DROP

7.0 TURKEY POINT PRESSURIZED THERMAL SHOCK TRANSIENTS

These are overcooling transients which involve a rapid and severe cooldown of fluid in the RCS with a high RCS pressure. Reduced reactor vessel fracture resistance in the beltline region due to neutron irradiation coupled with the effects of an overcooling transient may induce propagation of a flaw, thereby potentially affecting the integrity of the vessel. Several of these type transients for the Turkey Point units were simulated by FPL with the Turkey Point RETRAN02 base model. The RETRAN calculated results were compared to similar results produced by Westinghouse with four generic type plants. The comparison was to ensure that important phenomena were being represented, and that analysis assumptions and their influence on transient responses were reasonable. Three of these transients are presented in this report:

Small Break Loss of Coolant (Section 7.1)

Stuck Open Steam Generator Relief Valve (Section 7.2)

Steam Generator Tube Rupture (Section 7.3)

These transients demonstrate FPL's capability of performing analyses required to meet specific regulatory or licensing requirements.

7.1 Small Break Loss of Coolant Accident

7.1.1 Transient Description

The loss of coolant transient falls into the category of a decrease in primary system coolant inventory. The transient is characterized by a rapid cooldown and depressurization of the RCS.

The transient is initiated by a two inch diameter break, located in the hot leg. Breaks of this size tend to be limiting in terms of pressurized thermal shock. Larger breaks have more rapid depressurization as the SI flow cannot keep up with the break flow. Smaller breaks cause less severe cooldowns because SI, if at all required, is small. After the break, a loss of subcooled fluid occurs which results in a rapid depressurization until the saturation pressure of the reactor coolant is reached. In accordance with plant emergency procedures, the charging pumps are not activated and the reactor coolant pumps are tripped manually after RCS pressure falls below 1400 psia.

Safety injection flow continues to cool the fluid entering the downcomer, while maintaining the RCS pressure. After pump coastdown, natural circulation develops in the system.

7.1.2 RETRAN Analysis Description

The RETRAN model utilized the Turkey Point RETRAN Base Model (Appendix B) to which the initial and boundary conditions (Table 7.1.1) for the small break LOCA analysis were incorporated. The slip option in RETRAN was exercised to perform this calculation. The status of the safety systems integrated in the RETRAN model are summarized in Table 7.1.2. The model is initialized at full power. To obtain greater cooldown, stored energy in the metal structures is neglected. The 2 in. diameter hot leg break is modeled with a critical flow based on the Extended Henry model for subcooled flow and the Isoenthalpic Expansion model in RETRAN for two phase flow, with a discharge coefficent of 1.0.

The maximum safety injection flow capacity is assumed to be delivered to the RCS at a temperature of 40°F. The auxiliary feedwater flow is modeled to deliver maximum capacity to each steam generator at 40°F. The Steam Dump System in the No-load Tav. control mode is modeled to relieve the pressure on the secondary side. It provides another mechanism for heat removal in addition to the break.

7.1.3 Results

The results of the analysis are shown in Figures 7.1.1, 7.1.2 and 7.1.3, together with results of a generic analysis (Ref. 15)

performed by Westinghouse for a plant similar to Turkey Point. A sequence of events for the RETRAN analysis is provided in Table 7.1.3.

Figure 7.1.1, the reactor vessel downcomer pressure response, illustrates that the RETRAN prediction matches the generic analysis quite well. Both analyses predict a minimum pressure of approximately 600 psia at 4000 seconds.

Figure 7.1.2, the downcomer temperature prediction, shows that RETRAN agrees well with the generic analysis. The 40°F safety injection water tends to accentuate the density gradients throughout the system which produce the fluctuating temperature response observed. At 4000 seconds, both analyses predict downcomer temperatures of approximately 230°F.

Figure 7.1.3, shows the RETRAN calculated break and total safety injection flow rates. The break and safety injection flow rates are nearly equal throughout the transient. This maintains the RCS pressure, while cooling the fluid entering the downcomer. Only 5% of the initial fluid mass inventory is lost during the first 4000 seconds of the transient.

It can be concluded that the Turkey Point RETRAN model does reasonably well in predicting the trends and magnitudes of complicated phenomena such as a small break LOCA. The good agreement with the results of the vendor's generic analysis provides additional confidence of the validity of the model.

These RETRAN results also demonstrate that the current slip option, placed in the code to better predict two-phase phenomena, works well. The only drawback experienced with the option is the relatively high computer processing time required. It was found that a similar calculation without slip ran about six times faster than with slip but exhibited unrealistic oscillations in the RCS pressure and temperature predictions.

7.2 Stuck Open Steam Generator Relief Valve

7.2.1 Transient Description

The stuck open steam generator relief valve transient is characterized by a rapid cooldown of the RCS and a rapid depressurization until charging is actuated and a rapid repressurization occurs. The pressurizer empties but refills shortly after the charging system is actuated. The transient is initiated by the opening of the secondary relief valve and the concurrent loss of AC power which entails the loss of reactor coolant pumps and the startup of the auxilliary feedwater system. The RCS cools down due to energy removal through the affected steam generator. The charging system repressurizes the primary. RCS pressure is controlled by PORV actuation. Natural circulation drives flow through the broken steam

generator for the duration of the transient. The intact loop flow stagnates after isolation of the intact steam generators.

7.2.2 RETRAN Analysis Description

The RETRAN02 calculation utilized the Turkey Point RETRAN Base Model (Appendix B) to which the initial and boundary conditions (Table 7.2.1) for the stuck open steam generator relief valve analysis were incorporated. The status of the safety systems integrated in the RETRAN model are summarized in Table 7.2.2.

In order to obtain maximum cooldown the maximum charging flow capacity is assumed to be delivered to the RCS at a temperature of 40°F. One charging pump is started at the time of SI signal, the other two pumps are started at 10 minutes. The auxiliary feedwater flow is modeled to deliver 40°F water to each steam generator. To maximize break energy removal, no moisture carryover is assumed. The break flow, consisting of all steam, is computed with the Moody correlation. To maximize the cooldown no decay heat and no metal heat storage are assumed.

7.2.3 Results

Results of the RETRAN analysis are shown in Figures 7.2.1 and 7.2.2, along with results from a generic analysis (Ref. 16)

performed by Westinghouse for a plant similar to Turkey Point. A RETRAN sequence of events is provided in Table 7.2.3.

Figure 7.2.1, the pressurizer pressure response, shows that the RETRAN prediction of the minimum pressure as well as the rates of depressurization and repressurization agree well with the trends predicted by the generic analysis.

Figure 7.2.2, the cold leg temperature response, illustrates that RETRAN predicts the general trend of the cooldown quite well. At 4000 seconds RETRAN conservatively underpredicts the generic analysis by approximately 30°F.

It can be concluded that the Turkey Point RETRAN model does reasonably well in predicting the trends and magnitudes of the cooldown due to secondary side depressurization. Comparisons with vendor generic results show good agreement.

7.3 Steam Generator Tube Rupture

7.3.1 Transient Description

The steam generator tube rupture event falls into the category of a decrease in primary coolant inventory. The transient is initiated by a guillotine rupture of a single steam generator tube slightly above the steam generator tube sheet on the cold leg side. The RCS pressure decreases as break flow, in excess of

charging pump capacity, depletes the primary coolant inventory. A loss of AC is assumed to occur simultaneously with a low pressurizer pressure reactor trip. Main feedwater is automatically terminated on reactor trip, actuating the auxiliary feedwater system.

Following the reactor trip, the rate of RCS depressurization increases, as the RCS shrinkage associated with the scram and the loss of primary coolant inventory via the ruptured tube continues. As the post-trip RCS cooldown subsides, safety injection and charging flow begin to refill the pressurizer, terminating the RCS depressurization.

7.3.2 RETRAN Analysis Description

The RETRAN02 calculation utilized the Turkey Point RETRAN Base model (Appendix B) to which the initial and boundary conditions (Table 7.3.1) for the steam generator tube rupture analysis were incorporated. The status of the safety systems integrated in the RETRAN model are summarized in Table 7.3.2.

The tube leak was simulated by a junction with flow area equivalent to a double ended rupture of a single tube. The Extended-Henry-Fauske Isoenthalpic Expansion critical flow model is incorporated. The break is located slightly above the steam generator tube sheet on the cold side of the tube bundle. This provides for the maximum rate of RCS depressurization

prior to reactor trip.

The maximum safety injection and charging flow capacities are assumed to be delivered to the RCS at a temperature of 70°F. The auxiliary feedwater flow is modeled to deliver maximum flow capacity to each steam generator at 70°F.

The reactor physics parameters represent beginning-of-cycle conditions.

7.3.3 Results

Results of the RETRAN analysis are shown graphically in Figures 7.3.1 and 7.3.2, along with results from a generic analysis (Ref. 17) performed by Westinghouse for a 2770 MWth 3 loop plant. A RETRAN sequence of events is provided in Table 7.3.3.

Figure 7.3.1, the pressurizer pressure response, indicates that the RETRAN analysis predicts the pre-trip depressurization quite well. Both analyses agree well at time of reactor trip. The minimum RCS pressure calculated by RETRAN is approximately 1590 psia at 500 seconds, compared to 1658 psia at 430 seconds by the generic analysis.

Figure 7.3.2, illustrates the affected loop cold leg inlet temperature history for both analyses. RETRAN predicts the general trend of the generic analysis quite well. At 600 seconds,

RETRAN calculates a temperature of 555.6°F, agreeing well with a temperature of 556.3°F for the generic analysis.

It can be concluded that the Turkey Point RETRAN model predicted reasonably well the trends and magnitudes of the steam generator tube rupture. Comparison with generic vendor results show good agreement.

TABLE 7.1.1

INITIAL AND BOUNDARY CONDITIONS

SMALL BREAK LOCA

PARAMETER	VALUE
Core Power, MW (Thermal)	2200
Core Inlet Coolant Temperature, °F	547
Core Outlet Coolant Temperature, °F	602
Vessel Mass Flow Rate, 10^6 lbm/hr	101
Pressurizer Pressure, psia	2250
Feedwater Flow per Steam Generator, 10^6 lbm/hr	3.16
Steam Generator Pressure, psia	785
Feedwater Temperature, °F	434
Steam Generator Level, % NR	52

TABLE 7.1.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

SMALL BREAK LOCA

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves		X	
Pressurizer Safety Valves		X	
Main Steam Isolation Valves	X		
Main Feedwater Isolation Valves	X		
Auxiliary Feedwater System	X		
Safety Injection System			
HPSI	X		
Accumulators			X
LPSI			X
Atmospheric Dump Valve Systems		X	
Steam Dump and Bypass System	X		
Pressurizer Level Control System			X
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System			X
S. G. Level Control System	X		
Automatic Rod Motion			X

TABLE 7.1.3
SEQUENCE OF EVENTS
FOR SMALL BREAK LOCA

EVENTS	TIME (SEC.)	SETPOINT
Break	0	
Reactor Trip	33	Overtemp. ΔT Signal
Turbine Trip	34	On Reactor Trip
Steam Dump Opens	34	On Turbine Trip
Safety Injection (SI) Signal	56	Low Press. (1723 psig)
Auxiliary Feedwater Signal	56	On SI Signal
Feedwater Isolation Signal	57	On SI Signal
Low Primary Press. Signal	68	1400 psig
Reactor Coolant Pumps Stopped	98	30 sec after 1400 psig
Low Tave Signal	150	543°F
Steamline Isolation	151	See Footnote
Steam Dump Closes	151	On Steam Isolation
Hi-Hi Steam Gen. Level Signal (Intact Loop)	231	80% of level span
Auxiliary Feedwater Off (Intact Loop)	231	Indirect on Hi-Hi level signal
Hi-Hi Steam Generator Level Signal (Affected Loop)	247	80% of level span
Auxiliary Feedwater Off (Affected Loop)	247	Indirect on Hi-Hi level signal

Footnote

The MSIV trip logic in the RETRAN model is based on a High Steam Flow Signal coincident with either low Tave (543°F) or a low steamline pressure. In this analysis a spurious High Steam Flow Signal at the time of turbine trip combined with a low Tave produced the MSIV closure at 151 seconds.

TABLE 7.2.1

INITIAL & BOUNDARY CONDITIONS

OPEN S.G. RELIEF VALVE

PARAMETER	VALUE
Core Power, MW (Thermal)	1
Average RCS Coolant Temperature, °F	547
Vessel Mass Flow Rate, 10^6 lbm/hr	101
Pressurizer Pressure, psia	2250
Steam Generator Pressure, psia	1015
Steam Generator Level, % NR	39
Charging Flow per Pump, gpm	77

TABLE 7.2.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

OPEN S.G. RELIEF VALVE

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves		X	
Pressurizer Safety Valves		X	
Main Steam Isolation Valves		X	
Main Feedwater Isolation Valves	X		
Auxiliary Feedwater System	X		
Safety Injection System			
HPSI	X		
Accumulators			X
LPSI			X
Atmospheric Dump Valve Systems		X	
Steam Dump and Bypass System			X
Pressurizer Level Control System			X
Pressurizer Power Operated Relief Valves (PORV)	X		
Chemical and Volume Control System	X	-	
S. G. Level Control System			X
Automatic Rod Motion			X

TABLE 7.2.3
SEQUENCE OF EVENTS, OPEN S.G. RELIEF VALVE

EVENT	SETPOINT OR VALUE REACHED	TIME (SEC.)
Break		0
R.C. Pumps Trip	Loss of AC Power	1
Feedwater Pumps Trip	"	1
Aux. Feedwater Pumps Start	"	1
Reactor Trip	Low Flow	3
Turbine Trip	Indirect on Reactor Trip	4
T_{av}	554°F	6
SI Signal	High Steamline ΔP	20
T_{av}	543°F	52
Aux. Feedwater Off	Ten Minutes	600
Low Level in SG	Broken SG	890
Low-Low Level in SG	"	1070

TABLE 7.3.1

INITIAL AND BOUNDARY CONDITIONS

S.G. TUBE RUPTURE

PARAMETER	VALUE
Core Power, MW (Thermal)	2200
Pressurizer Pressure, psia	2250
Core Inlet Coolant Temperature, °F	547
Reactor Trip Pressure Setpoint, psia	1860
SI Initiation Setpoint, psia	1700
Main Feedwater Isolation, psia	1860
Auxiliary Feedwater Flowrate (3SG's), gpm	1520
RWST Temperature, °F	70
Condensate Storage Tank Temperature, °F	70

TABLE 7.3.2

SAFETY SYSTEMS STATUS ASSUMED IN MODEL

S.G. TUBE RUPTURE

SYSTEM	ACTUATED	AVAILABLE BUT NOT ACTUATED	NOT SIMULATED
Reactor Protection System (SCRAM)	X		
Pressurizer Pressure Control System	X		
Main Steam Safety Valves	X		
Pressurizer Safety Valves		X	
Main Steam Isolation Valves	X		
Main Feedwater Isolation Valves	X		
Auxiliary Feedwater System	X		
Safety Injection System			
HPSI	X		
Accumulators			X
LPSI		X	
Atmospheric Dump Valve Systems		X	
Steam Dump and Bypass System	X		
Pressurizer Level Control System	X		
Pressurizer Power Operated Relief Valves (PORV)		X	
Chemical and Volume Control System	X		
S. G. Level Control System	X		

TABLE 7.3.3
SEQUENCE OF EVENTS
STEAM GENERATOR TUBE RUPTURE

EVENTS	TIME (SEC.)	SETPOINT
Tube rupture	0.0	-
Reactor Trip on low Primary Pressure	187.0	1860 psia
Main Feedwater Isolation Valves Begin to Close	187.0	1860 psia
Loss of Offsite Power (RCF's Tripped)	188.0	-
Turbine Trip Turbine Throttle Valves Close	188.0	Reactor Trip
Safety Injection Actuated	201.0	1700 psia
Auxiliary Feedwater Delivered to Non-affected SG	367.0	-
Main Steamline Isolation	522.0	Hi-SG level alarm (Affected SG)
Analysis Terminated	600.0	-

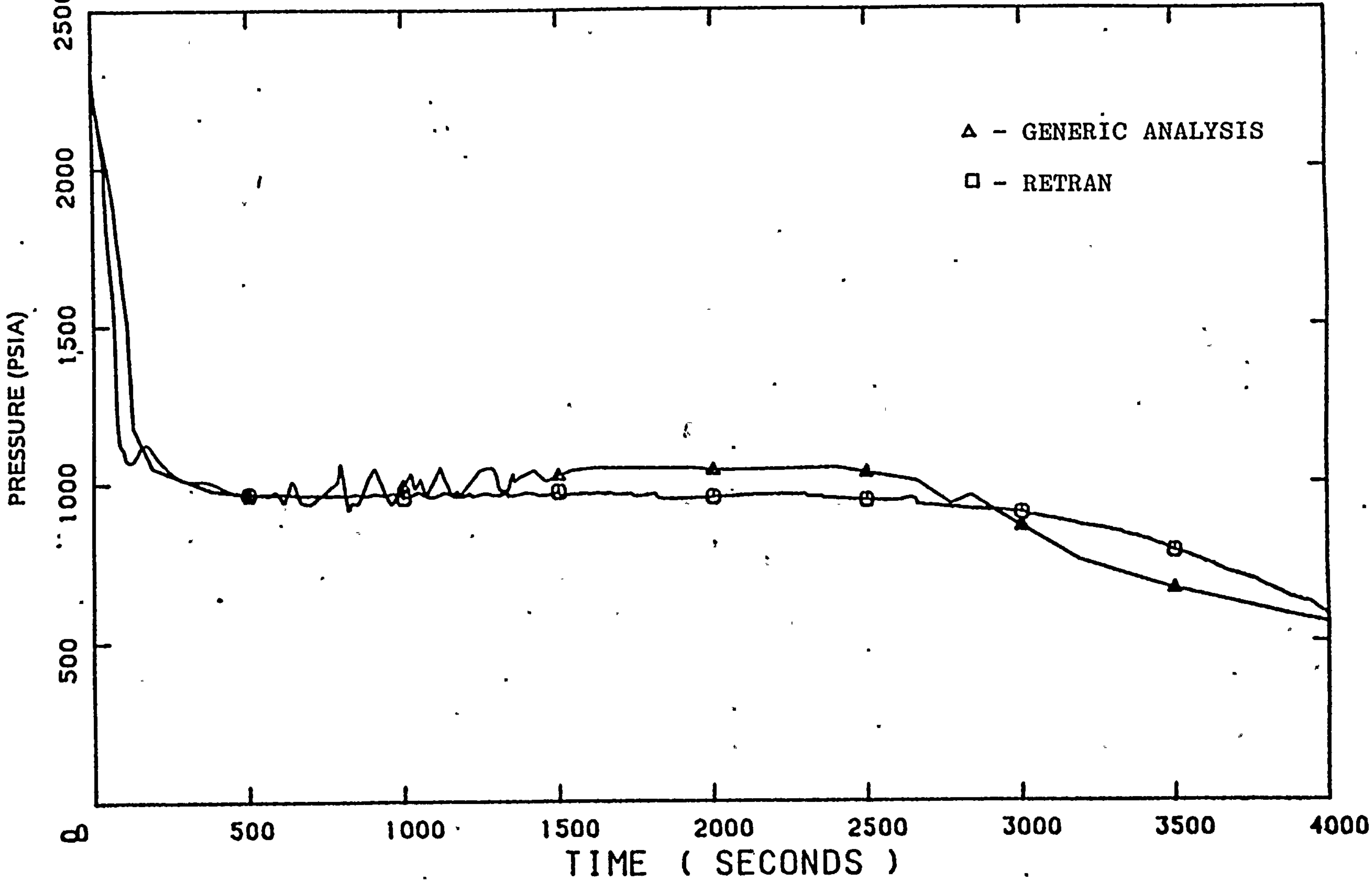

FIGURE 7.1.1 DOWNCOMER PRESSURE

SMALL BREAK LOCA

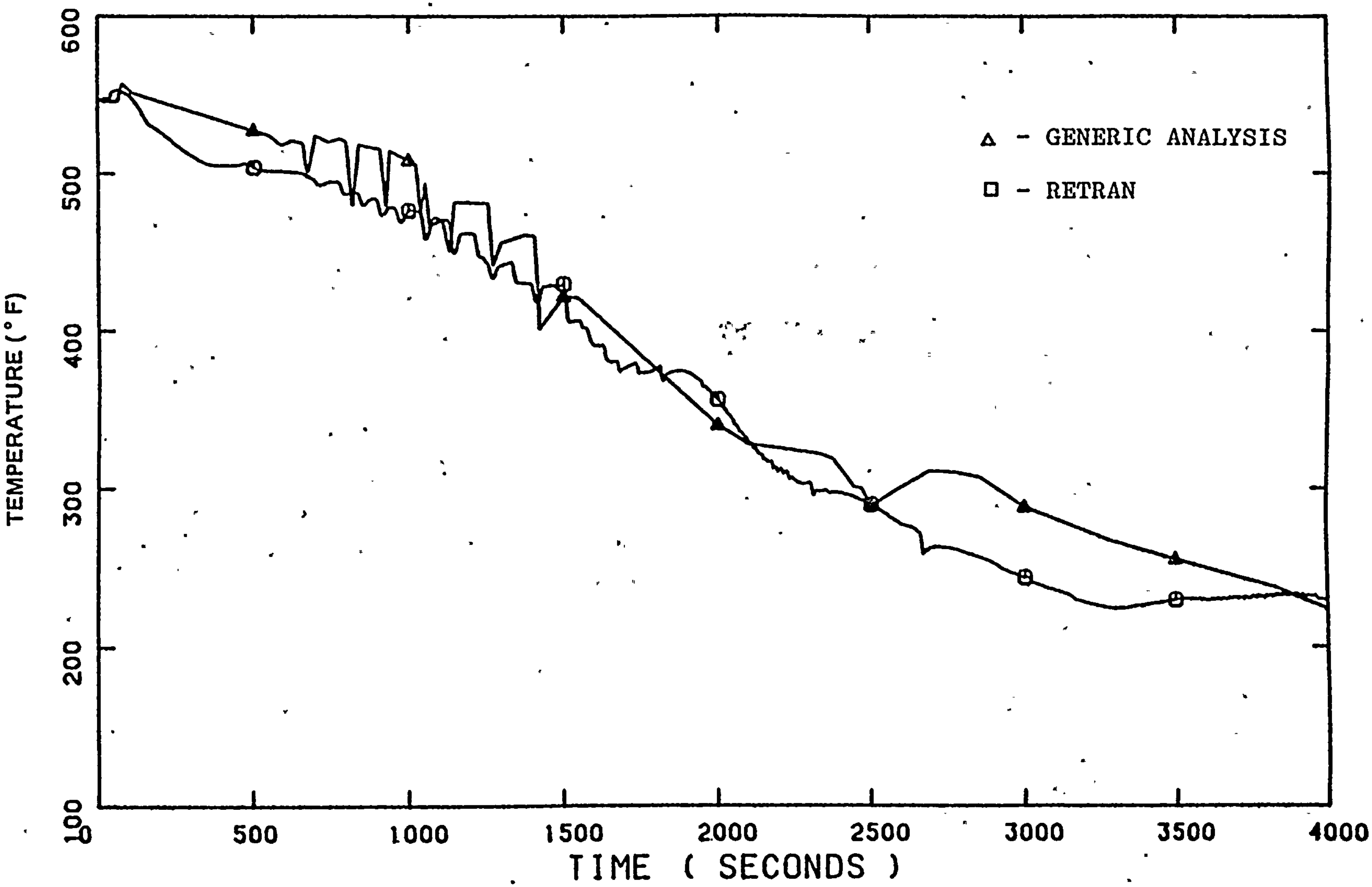

FIGURE 7.1.2 DOWNCOMER COOLANT TEMPERATURE

SMALL BREAK LOCA

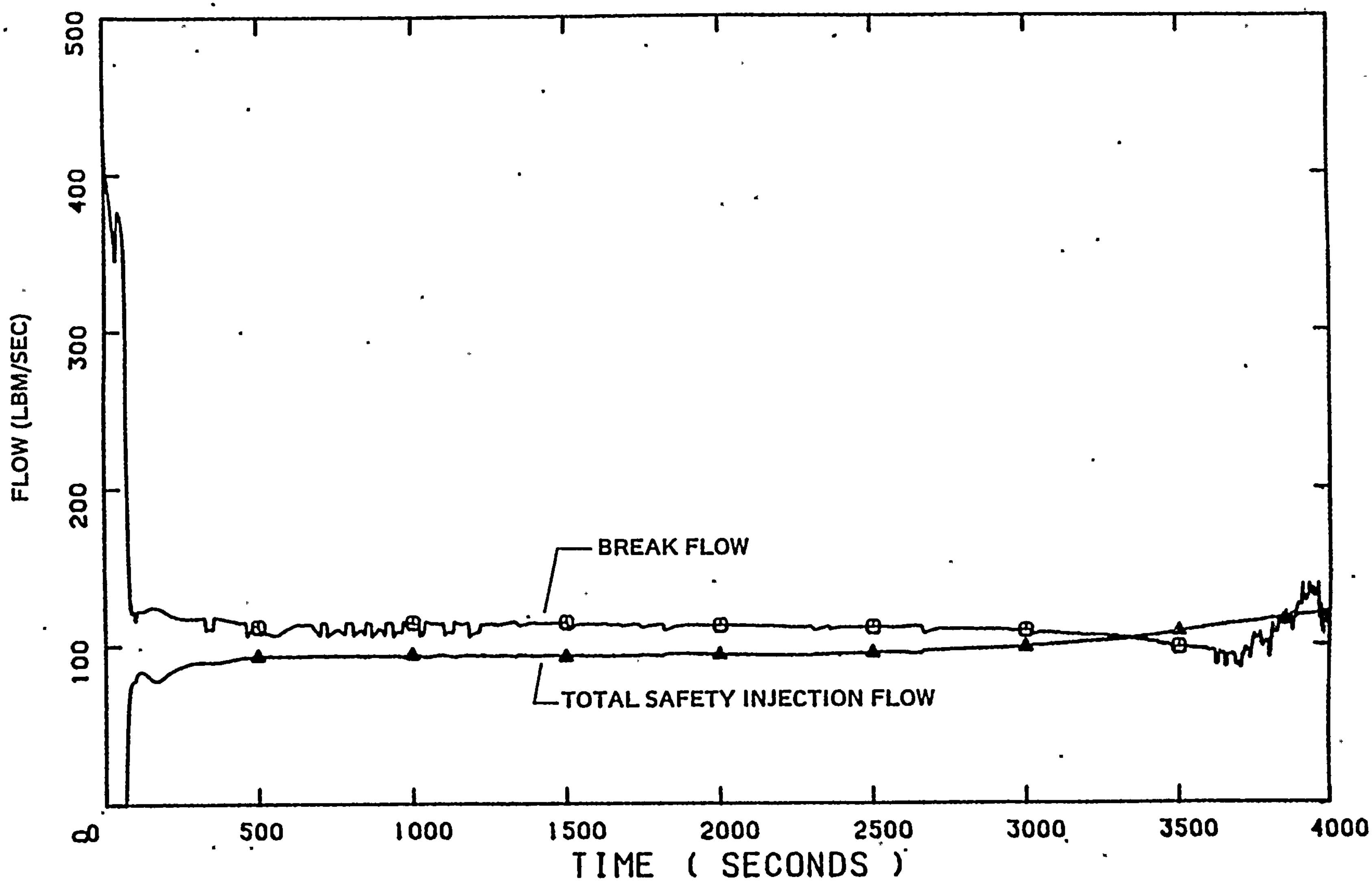

FIGURE 7.1.3 RETRAN BREAK AND SAFETY INJECTION FLOW RATES

SMALL BREAK LOCA

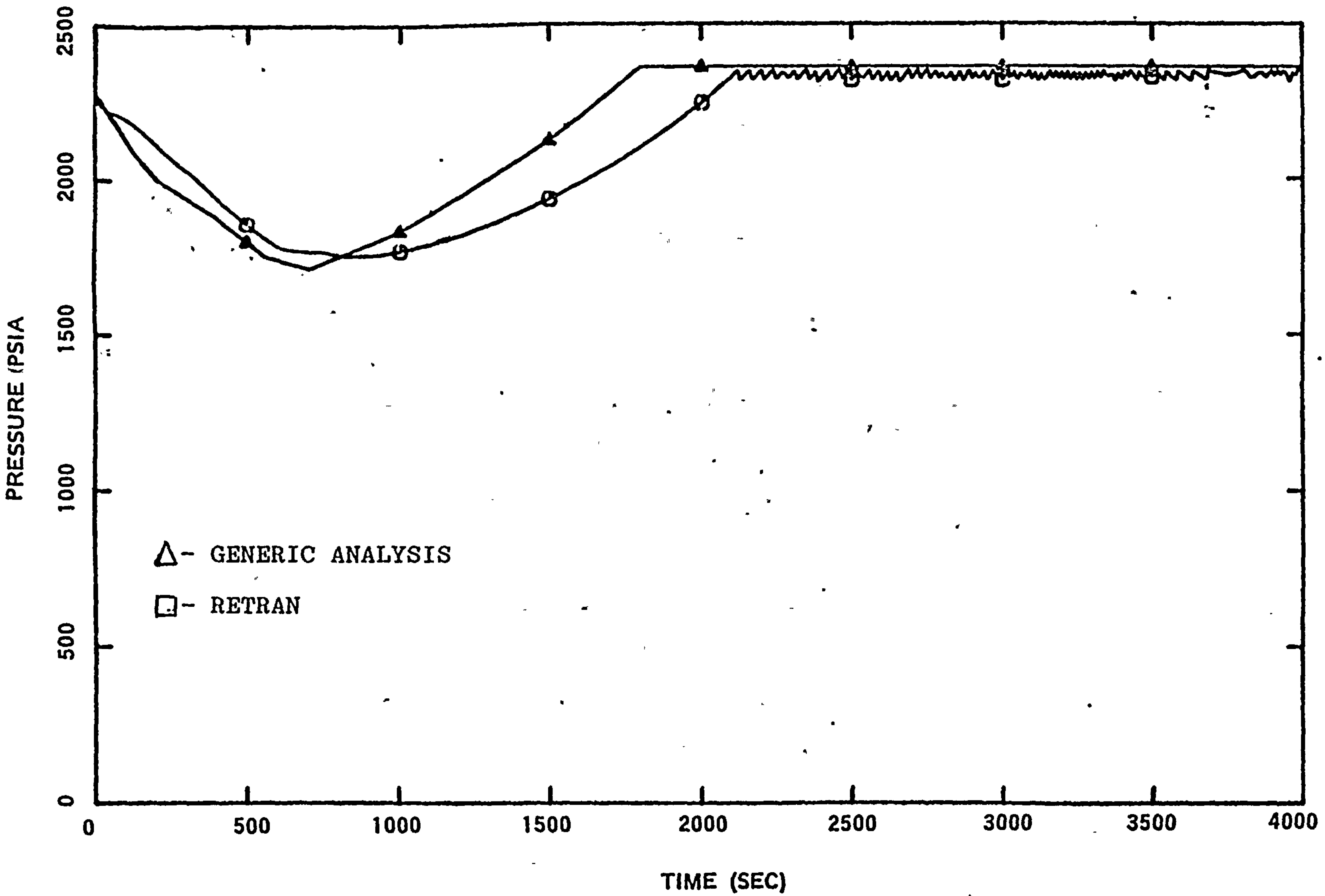

FIGURE 7.2.1 PRESSURIZER PRESSURE

OPEN S.G. RELIEF VALVE

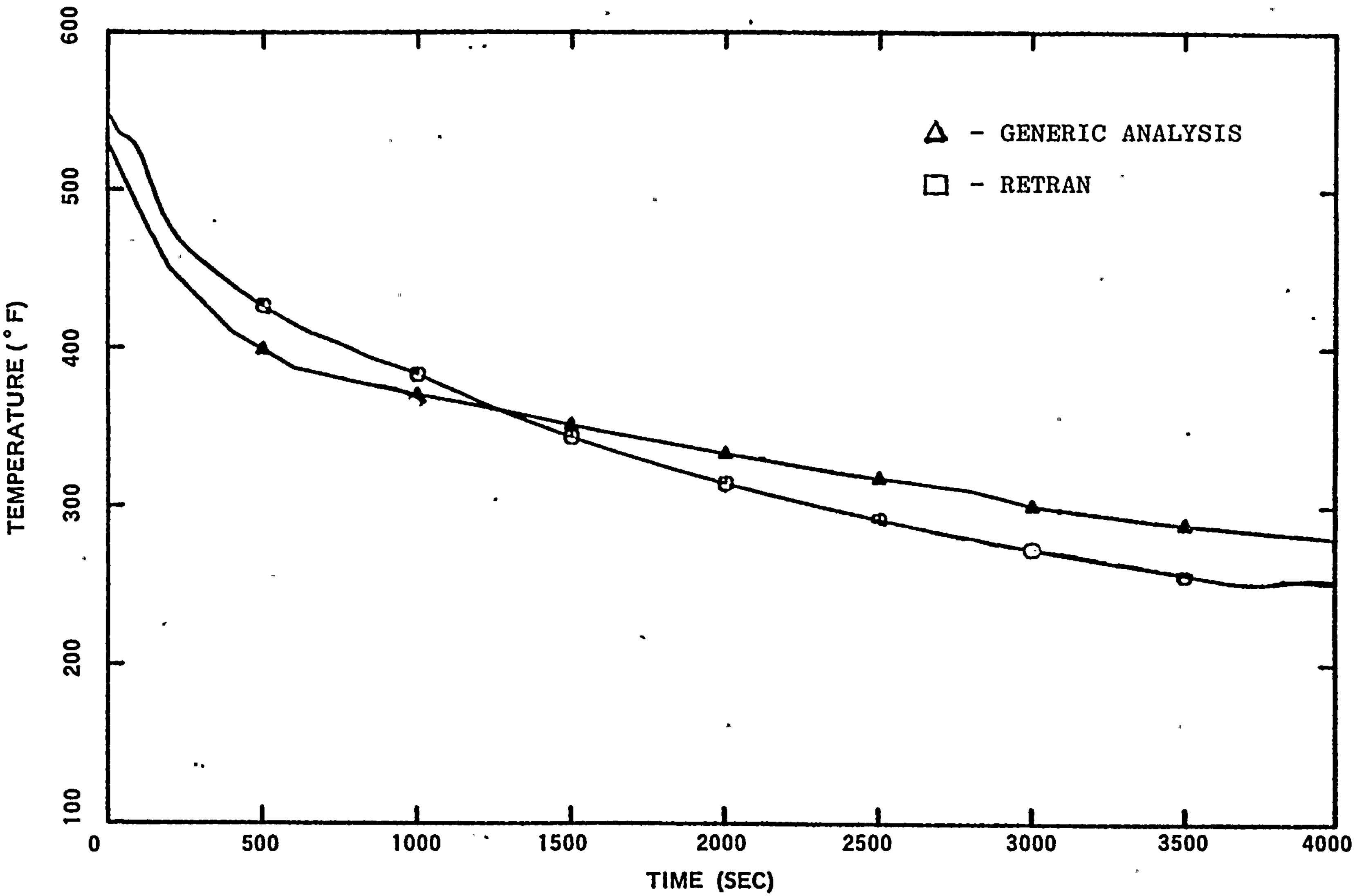

FIGURE 7.2.2 INLET CORE COOLANT TEMPERATURE

OPEN S.G. RELIEF VALVE

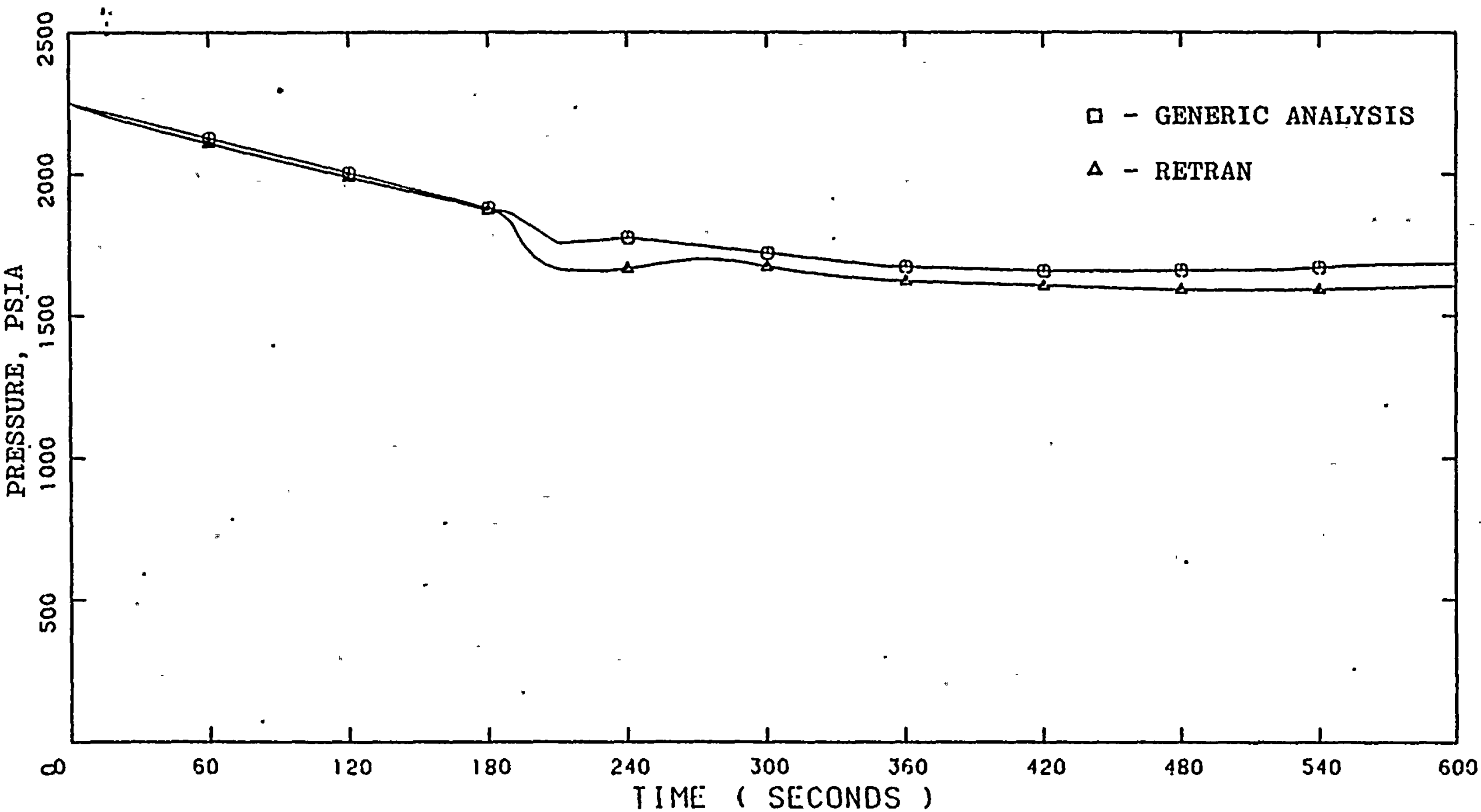

FIGURE 7.3.1 PRESSURIZER PRESSURE

S.G. TUBE RUPTURE

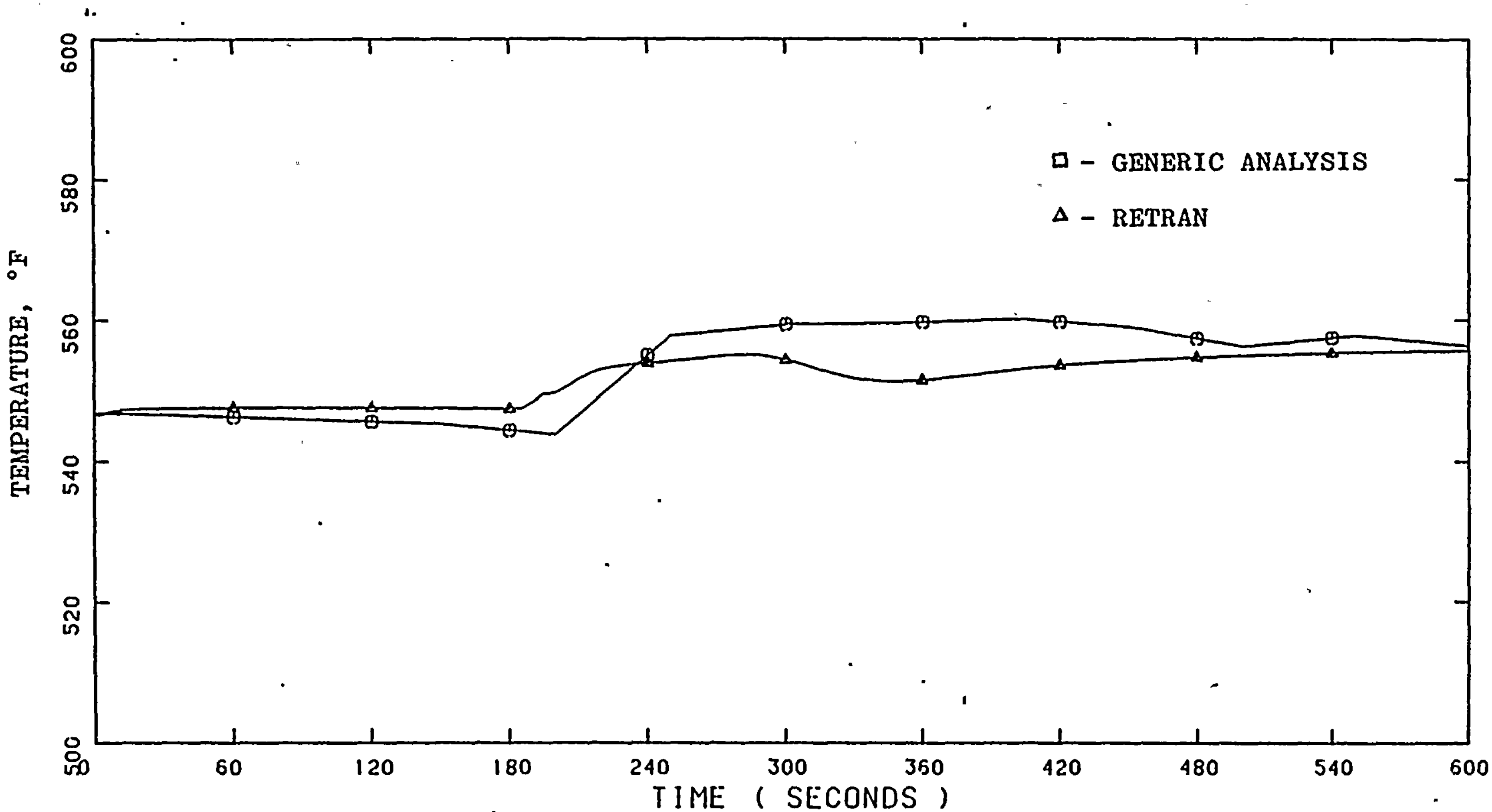

FIGURE 7.3.2 INLET CORE COOLANT TEMPERATURE

S.G. TUBE RUPTURE

8.0 CONCLUSIONS

Verification results were presented for the Turkey Point Units 3 and 4, St. Lucie Unit 1, and St. Lucie Unit 2 RETRAN base models. These results were presented as qualification basis for receiving an SER to use these models for transient and non-LOCA accident analyses in support of licensing actions.

The following are specific conclusions based on the contents of this report:

1. The adequacy of the geometry and system configuration representation in the RETRAN base models was demonstrated by good agreement of RETRAN calculated results with data from preoperational and power ascension plant tests, and off-normal events that have occurred at the plants.

2. Analysis capability for each of FPL's four nuclear plants was shown by presenting benchmarked transient analyses for each plant.

3. Ability to perform licensing type analyses was demonstrated by benchmarking to results presented in FSARs. Best-estimate analysis capability was demonstrated by benchmarking to plant tests and off-normal events.

4. Ability to analyze the full spectrum of transients and non-LOCA

accidents was shown by benchmarking to transients characterized by : a) decrease in secondary coolant system heat removal, b) increase in secondary coolant system heat removal, c) decrease in reactor coolant system inventory, d) reduction in core flowrate, and e) reactivity insertion.

5. Capability of performing analyses such as those performed in support of the Pressurized Thermal Shock issue resolution, in response to specific regulatory and licensing requirements.

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APPENDIX A

RETRAN COMPUTER CODE DESCRIPTION

RETRAN Computer Code Description

The RETRAN computer code was developed under the Electric Power Research Institute (EPRI) sponsorship by Energy, Inc. The code is an offshoot of the RELAP code and provided the utility industry with a versatile and reliable thermal-hydraulic code for the analysis of light water reactor systems. An ongoing code development effort since 1975 has produced two major versions of the code, RETRAN01 and RETRAN02. Both versions of the code have been extensively validated and qualified. The RETRAN01/MOD003 and RETRAN02/MOD002 codes have been reviewed by the Nuclear Regulatory Commission with the assistance of the Argonne National Laboratory, and approved for non-LOCA transient safety analysis application (Ref. A-1). Both versions of RETRAN have been installed on the FPL computer system. Currently, all analyses at FPL are performed with the RETRAN02 code.

The main features of RETRAN01 are:

1. A one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the reactor cooling system.

2. A point neutron kinetics model for the reactor core.

3. Special auxiliary or component models

   (nonequilibrium pressurizer, temperature transport delay).

4. Control system models.

5. A consistent steady-state initialization technique.

6. A model for steam separators based on separator efficiency curves.

7. A local conditions heat transfer model.

In order to remove some of the limitations of RETRAN01 and extend its capability the RETRAN02 code was released in May, 1981. New models added to the code include:

1. A dynamic slip equation and an algebraic slip equation.

2. A one-dimensional neutron kinetics model.

3. A set of two-phase natural convection heat transfer correlations.

With the new and improved models in the RETRAN02 version of the code (Ref. A-2), most of the objectives of the development effort are satisfied. The one-dimensional kinetics, dynamic slip, vector momentum, separator, and the auxiliary neutron void-fraction models allow the analyses of most PWR transients. The revised pressurizer solution techniques, the local

conditions heat transfer model, and other modifications were used to analyze a number of anticipated transients without scram (ATWS) events. Both pretest and posttest analyses of loss-of-fluid tests and semiscale small-break experiments demonstrated the capability of RETRAN02 for these transients.

The NRC concluded that RETRAN is an acceptable computer program for calculating non-loca transients and can be used in licensing applications.

REFERENCES

A-1 Cecil O. Thomas (USNRC) letter to Dr. Thomas W. Schnatz, Acceptance for Referencing of Licensing Topical Reports EPRI CCM5, RETRAN-A Program for One-Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems, and EPRI NP1850-CCM, RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, September 4, 1984.

A-2 RETRAN02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCMA, Vol. 1, Rev. 2, Computer Code Manual, November, 1984.

APPENDIX B

RETRAN MODEL DESCRIPTIONS

CONTENTS

B 1.0 TURKEY POINT RETRAN MODEL DESCRIPTION

B 1.1 General

The Turkey Point units are identical Westinghouse 3-loop nuclear steam supply systems (NSSS). As such, the same base model serves for both Turkey Point Unit 3 and Unit 4. The base model consists of 42 control volumes, 66 junctions, and 15 conductors. The model includes the major components of the NSSS and control systems necessary to simulate most transients and thus should be considered a best estimate type model. For those transients which require a more detailed representation of the system, special modeling considerations or conversion to licensing type models the base model is modified as necessary.

A nodalization diagram for the base model is presented in Figure B 1.1. A description of control volumes and junctions is presented in Tables B 1.1 and B 1.2, respectively. Fission energy is calculated by solution of the point kinetics reactivity model coupled with six delayed neutron precursor groups. Feedback effects due to moderator density and fuel temperature changes are accounted for with reactivity coefficients. CEA scram reactivity is represented by a tabular function of inserted

reactivity worth versus time after reactor trip. The reactivity associated with the presence of soluble boron is incorporated via tabular input or control system model. Reactor physics parameters are input to reflect cycle-specific conditions or FSAR analysis assumptions. Decay heat is accounted for with the ANS standard decay heat curve inherent in RETRAN coupled with a multiplier to conservatively adjust the decay heat when warranted.

The enthalpy transport option is activated at junctions associated with core volumes in order to represent the axial fluid temperature distribution more accurately. With the exception of the core and the steam generator tubes, the metal mass associated with NSSS structures are not modeled.

B 1.2 Reactor Vessel

Core

The Turkey Point reactor cores consists of 157, 15X15 fuel assemblies. The active core region is modeled as three equal length volumes, Volumes 2, 3 and 4 (Figure B 1.1). Heat slabs 1, 2 and 3 are associated with the core volumes and three radial regions, representing the UO_2 fuel, the fuel-to-clad gap and the cladding. These regions are subdivided into 4, 1 and 3 mesh intervals, respectively. Thermal conductivity and volumetric heat capacity data tables are provided by the code for the fuel, clad material and helium in the gap.

Downcomer, Lower and Upper Plenums, Upper Head

Volume 20 models the downcomer region. Flow from the cold legs of both loops enter this volume.

Volume 1 represents the region below the active core. Flow enters this volume from the downcomer and exits to the core and core bypass volumes.

Volumes 6 and 7 represents the upper plenum and upper head volumes, respectively.

B 1.3 Reactor Coolant Loops

The Turkey Point RETRAN model consists of two loops. Loop A, models one of the 3 loops of the plant. The hot leg, the pump suction leg, the reactor coolant pump, and the cold leg are represented by Volumes 8, 17, 18 and 19, respectively.

Loop B, thermal hydraulically lumps the hot legs, the pump suction legs, the reactor coolant pumps, and the cold legs of the remaining two loops. These components are modeled with Volumes 21, 30, 31 and 32, respectively.

The pressurizer (Volume 35) has been modeled utilizing the RETRAN non-equilibrium volume option. The volume occupied by the surge line has been included in the pressurizer volume. The surge line hydraulic

resistance has been incorporated in Junction 37, which connects the pressurizer to the hot leg (Volume 8). Junction 38 represents the hydraulic resistance of the pressurizer spray line connecting the cold leg (Volume 19) to the pressurizer.

The pressurizer heaters are represented in the model and controlled by pressurizer pressure and level.

The three pressurizer safety valves have been lumped together as Junction 46. The power operated relief valves (PORVs) are modeled as Junctions 44 and 45.

B 1.4 Steam Generators and Main Steam Piping

The steam generator tube bundles have been modeled as six equal length regions (Volumes 10, 11, 12, 13, 14 and 15 for the steam generator in Loop A; and Volumes 23, 24, 25, 26, 27 and 28 for the steam generator in Loop B). The enthalpy transport option is activated at the junctions associated with these volumes.

The tube bundle mass in each steam generator is modeled with four passive conductors. In Loop A, these consist of conductors 4, 5, 6, 7, 8 and 9. Conductors 10, 11, 12, 13, 14 and 15 are used in Loop B. Heat transfer coefficients are calculated by the RETRAN code from correlations that account for local flow conditions inside and outside of the tubes.

The steam generator inlet plenums and outlet plenums are modeled with Volumes 9 and 22, and Volumes 16 and 29, respectively.

The secondary side consists of a single volume for each of the two steam generators (Volumes 36 and 37). A distinct mixture level exists within each of the steam generators by use of the RETRAN phase separation option. Main feedwater (Junction 108) enters a feedwater header (Volume 106), which regulates flow to the feedwater piping line (Volumes 105 and 104). Auxiliary feedwater is modeled as a fill junction (Junctions 109 and 110) connected to the feedwater piping volumes.. Junctions 40 and 42 provide the feedwater to the loop A and B steam generators, respectively.

The steam line is represented by two control volumes. Volumes 100 and 101 represent that portion of the steam line upstream of the MSIVs represented by Junctions 100 and 101. Immediatley downstream of the MSIVs, a common main steam header (Volume 102) is connected to a single turbine header volume (Volume 103). Junction 103, acts as the turbine stop valve, providing a flow path to the turbine, modeled with Volume 500.

The code safety valves and the relief valves are located on the steam line upstream of the MSIVs. They are represented by negative fill junctions 51, 52, 53, 54, 55, 56, 57, 58, 59 for the code safety valves; and 50 and 51 for the relief valves. The steam dump system is represented as Junction 104, a negative fill junction off the turbine header (Volume 103).

B 1.5 Safety Systems

Ninety trips are included in the RETRAN base model of Turkey Point. These trips provide for the complete set of initiating conditions for reactor trip, safety injection, main steam isolation, main feedwater isolation and auxiliary feedwater startup as given in Table 7.2.1 of the updated FSAR for Turkey Point (Ref. 13). Additional trips are included for actuating pressurizer heaters, safety and relief valves and main steam safety and relief valves.

Control system models are included in the base deck for the following control and protection features:

1. Overtemperature Delta-T Trip Setpoint
2. Overpower Delta-T Trip Setpoint
3. Low Pressurizer Pressure Trip Setpoint
4. Pressurizer Pressure Control
5. High Steam Flow Safety Injection Trip Setpoint
6. Steam Generator Water Level Control
7. Pressurizer Water Level Control
8. Main feedwater Flow Control
9. Auxiliary Feedwater Flow Control
10. Steam Generator Relief Valve Flow Control

The status of the modeled safety systems for each one of the analyses described in this report is provided in a separate table in the corresponding section.

B 2.0 ST. LUCIE RETRAN MODEL DESCRIPTION

B 2.1 General

St. Lucie Unit 1 and Unit 2 are geometrically similar, except for the fuel, which is 14 x 14 for Unit 1 and 16 x 16 for Unit 2. The modeling approach is indentical for each; and as such, only one of the base models, St. Lucie Unit 1, is described here.

Represented in the model are all major components of the NSSS and control systems necessary to simulate most transients. For those transients which require a more detailed representation of the system, special modeling considerations or licensing type parameters, the base model is modified as necessary. The base model consists of 42 control volumes, 64 junctions, and 11 heat conductors.

The following sections describe the major areas of the St. Lucie 1 RETRAN base model. A nodalization diagram is presented in Figure B 2.1 and a description of control volumes and junctions is presented in Tables B 2.1 and B 2.2, respectively.

With the exception of the core and the steam generator tubes, the metal mass associated with NSSS structures were not modeled.

B 2.2 Reactor Vessel

Core

The St. Lucie Unit 1 reactor core consists of 217, 14X14 fuel assemblies. The active core region is modeled as three equal length volumes, Volumes 26, 27 and 28 (Figure B 2.1). Heat slabs 1, 2 and 3 are associated with the core volumes. These heat slabs have been modeled as three radial regions, representing the UO_2 fuel, the fuel-to-clad gap and the cladding. These regions are subdivided into 4, 1 and 3 mesh intervals, respectively. Thermal conductivity and volumetric heat capacity data tables are provided for the fuel, clad material and helium in the gap.

Fission energy is calculated by solution of the point kinetics reactivity model coupled with six delayed neutron precursor groups. Feedback effects due to moderator density and fuel temperature changes are taken into account via reactivity coefficients. CEA scram reactivity is represented by a tabular function of reactivity worth inserted versus time after reactor trip. Reactivity associated with the presence of soluble boron is incorporated via tabular input or control system modeling. Reactor physics parameters are input to reflect cycle-specific conditions or FSAR analysis assumptions. Decay heat is accounted for with the ANS standard decay heat curve inherent in RETRAN, coupled with a multiplier to conservatively adjust the decay heat when warranted.

The enthalpy transport option is activated at junctions associated with core volumes in order to represent the axial fluid temperature distribution more accurately.

Flow through the CEA guide tubes and core shroud annulus, which allows coolant to traverse the core region without contacting the heated fuel cladding, is represented in the model by Volume 29. The hydraulic resistance in this path is set to result in the nominal steady state core bypass flow of 2.5% of total loop flow.

Downcomer, Lower and Upper Plenums, Upper Head

Volume 24 models the downcomer region. Flow from the cold legs in both loops enter this volume.

Volume 25 represents the region below the active core. Flow enters this volume from the downcomer and exits to the core and core bypass volumes.

Volumes 30 and 31 represents the upper plenum and upper head volumes, respectively. The guide tubes in the upper plenum are modeled with Volume 35.

B 2.3 Reactor Coolant Loops

The RETRAN model is divided into two loops. Loop A, a combined loop, lumps the two pump suction legs (Volume 8), the two reactor

coolant pumps (Volume 9), and the two cold legs (Volume 10). The hot leg is modeled with Volume 1.

Loop B models each of the NSSS components uniquely. The hot leg is modeled with Volume 11. The pump suction legs are modeled as Volumes 18 and 21. The reactor coolant pumps are modeled as Volumes 19 and 22. The cold legs are modeled as Volumes 20 and 23.

The pressurizer (Volume 34) has been modeled utilizing the RETRAN non-equilibrium volume option. The surge line (Volume 32) connects the pressurizer to the hot leg of loop A (Volume 1). Volume 33 represents the pressurizer spray line connecting the lumped cold leg (Volume 10) to the pressurizer.

The pressurizer heaters are represented in the model and controlled by pressurizer pressure and level.

The pressurizer level controller determines the normal charging and letdown flow rates. The charging flow is modeled as positive fill Junctions 103 and 104. The letdown flow is modeled as negative fill Junction 105.

The three pressurizer safety valves have been lumped together as Junction 90. The power operated relief valves (PORVs) are modeled as Junction 89.

B 2.4 Steam Generators and Main Steam Piping

The steam generator tube bundles have been modeled as four equal length regions (Volumes 3, 4, 5 and 6 for loop A; and Volumes 13, 14, 15 and 16 for loop B). The enthalpy transport option is activated at the junctions associated with these volumes.

The tube bundle metal mass in each steam generator is modeled with four passive conductors consisting in Loop A of conductors 4, 5, 6 and 7 and in Loop B of conductors 8, 9, 10 and 11. Heat transfer coefficients are calculated by the RETRAN code from correlations that account for local flow conditions inside and outside of the tubes.

The steam generator inlet plenums and outlet plenums are modeled with Volumes 2 and 12, and Volumes 7 and 17, respectively, for loops A and B.

The secondary side of the base model has a single volume for each of the two steam generators (Volumes 51 and 52). A distinct mixture level is allowed to exist within each of the steam generators by use of the RETRAN phase separation option. Fill Junctions 81 and 82 for Volume 51 and Junctions 83 and 84 for Volume 52 represent the main and auxiliary feedwater.

The steam line piping from the steam generator outlet nozzles to the turbine stop valves comprise the main steam system. Volumes 53 and 54 model the piping upstream of the MSIVs in both loops. The code

safety valves are simulated by Junctions 85, 86, 87 and 88. Volumes 55 and 56 model the piping immediately downstream of the MSIVs.

The main steam header up to the turbine throttle valve is modeled as a single Volume (Volume 57). The steam dump and bypass to condenser is modeled by negative fill Junctions 96 and 97, respectively, from the main steam header.

B 2.5 Safety Systems

Sixty two trips are included in the RETRAN base model of ST. Lucie. These trips provide the required conditions to model initiating signals for reactor trip, safety injection, main steam isolation and main feedwater isolation. Additional trips are included for actuating pressurizer heaters, safety and relief valves, and main steam safety valves.

Control system models are included in the base deck for the pressurizer pressure and level control functions. Additional trips and controls such as for the auxilary feedwater and the steam dump systems are added to the base model on an as needed basis depending on the type of transient being modeled. The status of the modeled safety systems for each one of the analyses described in this report is provided in a separate table in the corresponding section.

TABLE B1.1

TURKEY POINT RETRAN MODEL VOLUME DESCRIPTION

VOLUME NO.	DESCRIPTION
	VESSEL
1	Lower Plenum
2	Lower Core
3	Middle Core
4	Upper Core
5	Core Bypass
6	Upper Plenum
7	Upper Head
20	Downcomer
	SINGLE LOOP (Loop A)
8	Hot Leg Piping
9	Steam Generator Inlet Plenum
10	Steam Generator Tubes, Hotter Side, First Section
11	Steam Generator Tubes, Hotter Side, Second Section
12	Steam Generator Tubes, Hotter Side, Top
13	Steam Generator Tubes, Cooler Side, Top
14	Steam Generator Tubes, Cooler Side, First Section
15	Steam Generator Tubes, Cooler Side, Second Section
16	Steam Generator Outlet Plenum
17	Pump Section Piping
18	Pump
19	Cold Leg, Pump Discharge Piping
35	Pressurizer
	COMBINED OR DOUBLE LOOP (Loop B)
21	Hot Leg Piping
22	Steam Generator Inlet Plenum
23	Steam Generator Tubes, Hotter Side, First Section
24	Steam Generator Tubes, Hotter Side, Second Section
25	Steam Generator Tubes, Hotter Side, Top
26	Steam Generator Tubes, Cooler Side, Top
27	Steam Generator Tubes, Cooler Side, First Section
28	Steam Generator Tubes, Cooler Side, Second Section
29	Steam Generator Outlet Plenum

TABLE B1.1 (CONTINUED)

TURKEY POINT RETRAN MODEL VOLUME DESCRIPTION

VOLUME NO.	DESCRIPTION
	COMBINED OR DOUBLE LOOP
30	Pump Section Piping
31	Pump
32	Cold Leg, Pump Discharge Piping
	STEAM GENERATORS (SECONDARY)
36	Single Steam Generator
37	Double Steam Generator (Combined)
	STEAMLINE
100	Single Steamline Upstream of MSIV
101	Double Steamline (Combined) Upstream of MSIV
102	Steamline Downstream of MSIV's
103	Turbine Header
	FEEDWATER
104	Feedwater Piping From Header to Single S.G.
105	Feedwater Piping From Header to Combined S.G.
106	Feedwater Header
	CONTAINMENT
500	Sink Volume (Infinite Volume)

TABLE B1.2

TURKEY POINT RETRAN MODEL JUNCTION DESCRIPTION

JUNCTION NO.	DESCRIPTION
	VESSEL
20	Vessel Inlet from Single Loop
35	Vessel Inlet from Double Loop
21	Leakage Path from Downcomer to Upper Head
22	Downcomer to Lower Plenum
1	Core Inlet
2	Middle Core Inlet
3	Middle Core Outlet
4	Core Outlet
5	Core Bypass Inlet
6	Core Bypass Outlet
7	Upper Plenum to Upper Head
8	Vessel Outlet to Single Loop
23	Vessel Outlet to Combined Loop
	SINGLE LOOP
9	Steam Generator Plenum Inlet
10	Steam Generator, Inlet Tubes
11	Steam Generator Tubes
12	Steam Generator Tubes
13	Steam Generator Tubes
14	Steam Generator Tubes
15	Steam Generator Tubes
16	Steam Generator, Outlet Tubes
17	Steam Generator Plenum Outlet
18	Pump Section
19	Pump Discharge
37	Surge Line
38	Spray Line Inlet
39	Spray Line Outlet
190	Safety Injection
191	Charging
	COMBINED OR DOUBLE LOOP
24	Steam Generator Plenum Inlet
25	Steam Generator Inlet, Tubes
26	Steam Generator Tubes
27	Steam Generator Tubes
28	Steam Generator Tubes

TABLE B1.2 (CONTINUED)

TURKEY POINT RETRAN MODEL JUNCTION DESCRIPTION

JUNCTION NO.	DESCRIPTION
	COMBINED OR DOUBLE LOOP
29	Steam Generator Tubes
30	Steam Generator Tubes
31	Steam Generator Outlet, Tubes
32	Steam Generator Plenum Outlet
33	Pump Section
34	Pump Discharge
320	Safety Injection
321	Charging
	STEAM GENERATORS (SECONDARY)
40	Single SG Inlet
41	Single SG Outlet
400	Auxiliary Feedwater Inlet to Single SG
42	Combined SG Inlet
43	Combined SG Outlet
420	Auxiliary Feedwater Inlet to Combined SG
	PRESSURIZER VALVES
44	Relief Valve (Signal from P. Controller)
45	Relief Valve (2350 psia)
46	Safety Valve (2500 psia)
	STEAMLINE
100	MSIV, Single Loop
101	MSIV, Combined Loop
102	Steamline to Turbine Header
103	Turbine Inlet
50	Atmospheric Steam Relief. Single Loop (1050 psia)
51	Atmospheric Steam Relief. Combined Loop (1050 psia)
52	SG Single Loop Safety Valve 1 (1100., 1089. psia)
53	SG Single Loop Safety Valve 2 (1115., 1104. psia)
54	SG Single Loop Safety Valve 3 (1130., 1119. psia); Break (Steamline Break)

TABLE B1.2 (CONTINUED)

TURKEY POINT RETRAN MODEL JUNCTION DESCRIPTION

JUNCTION NO.	DESCRIPTION
	STEAMLINE (CONT'D)
55	SG Single Loop Safety Valve 4 (1145., 1134. psia)
56	SG Double Loop Safety Valve 1 (1100., 1089. psia)
57	SG Double Loop Safety Valve 2 (1115., 1104. psia)
58	SG Double Loop Safety Valve 3 (1130., 1119. psia)
59	SG Double Loop Safety Valve 4 (1145., 1134. psia)
	STEAM DUMP SYSTEM
104	Steam Dump to Condenser
	FEEDWATER
106	Feedwater Header to Single Loop Piping
107	Feedwater Header to Double Loop Piping
108	Feedwater Header Inlet

TABLE B2.1

ST. LUCIE UNIT 1 RETRAN MODEL VOLUME DESCRIPTION

VOLUME NO.	DESCRIPTION	
	Unaffected Loop (W/Pressurizer)	**Loop-A**
1	Hot Leg Piping	
2	Steam Generator Inlet Plenum	
3	Steam Generator Tubes, Hotter Side, First Section	
4	Steam Generator Tubes, Hotter Side, Top	
5	Steam Generator Tubes, Cooler Side, Top	
6	Steam Generator Tubes, Cooler Side, First Section	
7	Steam Generator Outlet Plenum	
8	Pump Section Piping	
9	Pumps (1-A and 1-B)	
10	Cold Leg, Pump Discharge Piping	
32	Pressurizer Surge Line	
33	Pressurizer Spray Line	
34	Pressurizer	
	Affected Loop	**Loop-B**
11	Hot Leg Piping	
12	Steam Generator Inlet Plenum	
13	Steam Generator Tubes, Hotter Side, First Section	
14	Steam Generator Tubes, Hotter Side, Top	
15	Steam Generator Tubes, Cooler Side, Top	
16	Steam Generator Tubes, Cooler Side, Second Section	
17	Steam Generator Outlet Plenum	
18	Pump Suction Piping	
19	Pump 2-A	
20	Cold Leg, Pump Discharge Piping	
21	Pump Suction Piping	
22	Pump 2-B	
23	Cold Leg, Pump Discharge Piping	
	Vessel	
25	Lower Plenum	
26	Lower Core	
27	Middle Core	
28	Upper Core	
29	Core Bypass	
30	Upper Plenum	
31	Upper Head	
24	Downcomer	
35	CEA Guide Tube	

TABLE B2.1 (CONTINUED)

ST. LUCIE UNIT 1 RETRAN MODEL VOLUME DESCRIPTION

JUNCTION NO	DESCRIPTION
	Steam Generators (Secondary)
51	Unaffected Loop Steam Generator
52	Affected Loop Steam Generator
	Steamline
53	Unaffected Loop Steamline Upstream of MSIV
54	Unaffected Loop Steamline Upstream of MSIV
55	Unaffected Loop Steamline Downstream of MSIV
56	Affected Loop Steamline Downstream of MSIV
57	Turbine Header

TABLE B2.2

ST. LUCIE UNIT 1 RETRAN MODEL JUNCTION DESCRIPTION

JUNCTION NO.	DESCRIPTION
	Vessel
11	Vessel Inlet from Unaffected Loop
25,26	Vessel Inlet from Affected Loop
27	Downcomer to Lower Plenum
28	Core Inlet
29	Middle Core Inlet
30	Middle Core Outlet
31	Core Outlet
32	Core Bypass Inlet
33	Core Bypass Outlet
34	Upper Plenum to Upper Head
1	Vessel Outlet to Unaffected Loop
12	Vessel Outlet to Affected Loop
39	CEA Guide Tube Inlet
40	CEA Guide Tube Outlet
	Unaffected Loop
2	Steam Generator Plenum Inlet
3	Steam Generator, Inlet Tubes
4,5,6	Steam Generator Tubes
7	Steam Generator, Outlet Tubes
8	Steam Generator Plenum Outlet
9	Pump Section
10	Pump Discharge
35	Surge Line Inlet
36	Pressurizer Inlet
37	Spray Line Inlet
38	Spray Line Outlet
100	Safety Injection
103	Charging
105	Letdown
	Affected Loop
13	Steam Generator Plenum Inlet
14	Steam Generator Inlet, Tubes
15,16,17	Steam Genertaor Tubes
18	Steam Generator Outlet, Tubes
19,22	Steam Generator Plenum Outlet
20	Pump Suction
21	Pump Discharge
101	Safety Injection
104	Charging
23	Pump Suction
24	Pump Discharge

TABLE B2.2 (CONTINUED)

ST. LUCIE UNIT 1 RETRAN MODEL JUNCTION DESCRIPTION

JUNCTION NO	DESCRIPTION
	Steam Generators (Secondary)
81	Affected SG Main Feedwater
91	Single SG Outlet
82	Auxiliary Feedwater Inlet to Affected Loop
83	Unaffected SG Main Feedwater
92	Unaffected SG Outlet
84	Auxiliary Feedwater Inlet to Unaffected Loop
SG	
	Pressurizer Valves
89	PORVs (2350 psia)
90	Safety Valve (2500 psia)
	Steamline
93	MSIV, Affected Loop
94	MSIV, Unaffected Loop
95,96	Steamline to Turbine Header
99	Turbine Inlet
85	SG Unaffected Loop Safety Valve Bank 1
86	SG Unaffected Loop Safety Valve Bank 2
87	SG Affected Loop Safety Valve Bank 1
88	SG Affected Loop Safety Valve Bank 2
	Steam Dump System
97	Steam Bypass to Condenser
98	Steam Dump to Condenser

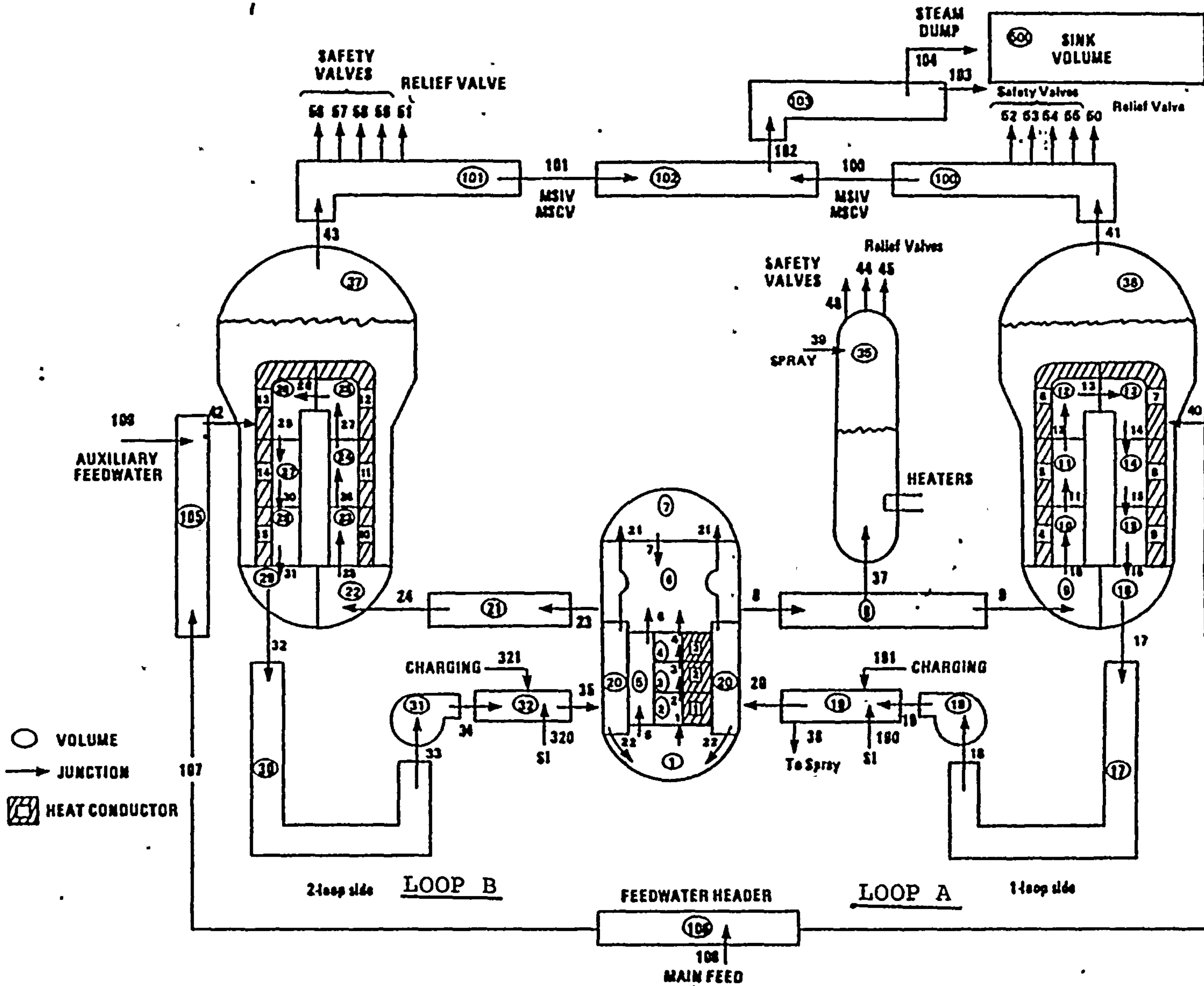

FIGURE B1.1 TURKEY POINT RETRAN MODEL NODALIZATION DIAGRAM

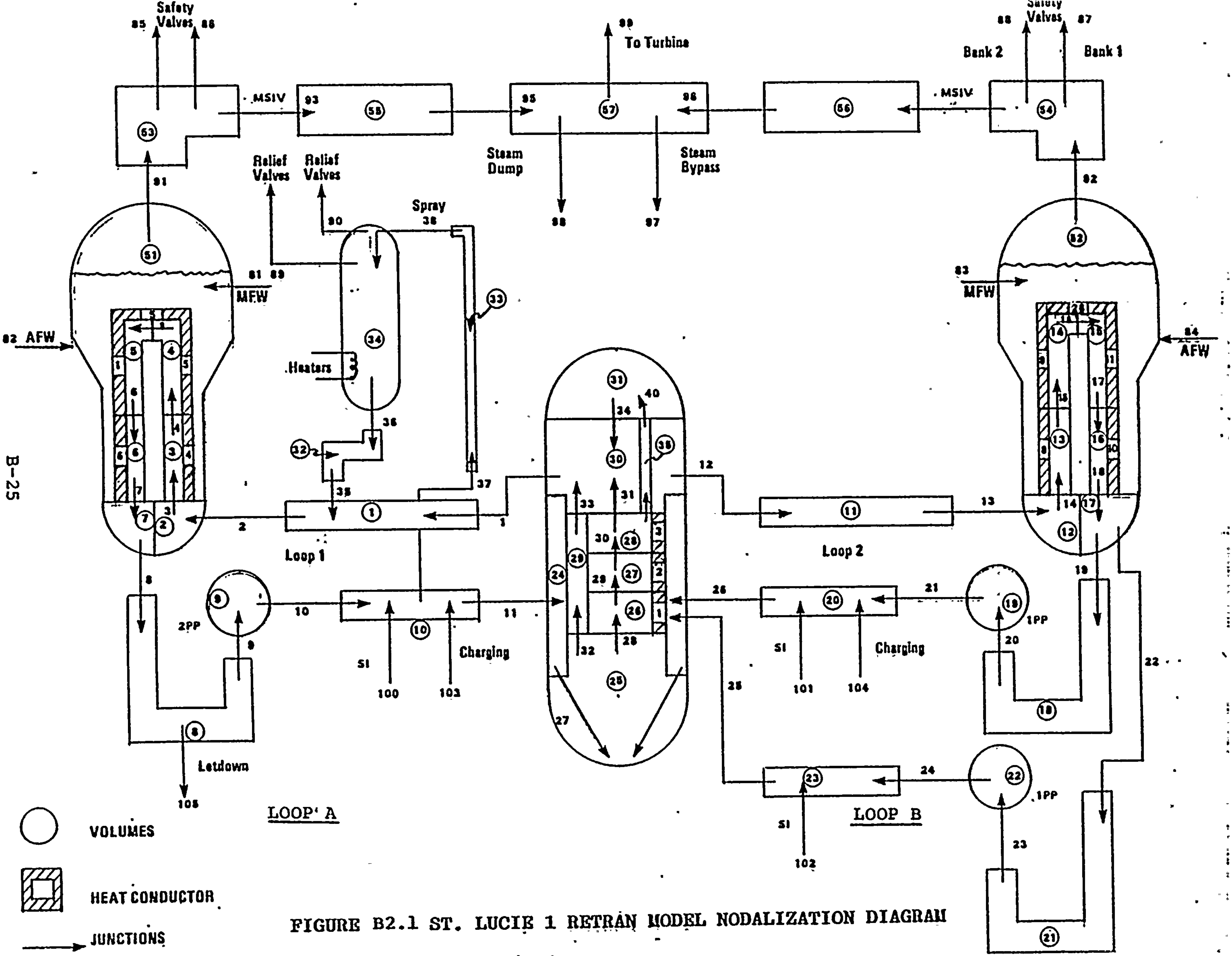

FIGURE B2.1 ST. LUCIE 1 RETRAN MODEL NODALIZATION DIAGRAM

APPENDIX C

STAFF EXPERIENCE

The Thermal Hydraulics and System Analysis Group is part of the Fuel Resources Department of FPL and has overall responsibility for performing reload safety evaluations, plant transient and safety analyses. All thermal hydraulic system models are maintained by this group. The group currently includes a supervisor, and seven full-time engineers. A number of supporting personnel such as consultants, programmers, technicians, and a co-op student are available. Present nuclear and thermal hydraulic analysis related experience in the group totals approximately 60 engineer-years. Degree levels include PhD, MS, and BS degrees in Engineering and related disciplines. Typically, each member of the group attends two industry related meetings per year, such as user group meetings or topical meetings sponsored by professional societies. Overall, members of the group have presented about 20 papers at industry related meetings. It is expected that the level of experience will be maintained at or above the present level.

The Thermal Hydraulics and System Analysis Group routinely consults with Dr. Joel Weisman of the University of Cincinnati.