TOPICAL REPORT A-85-11

RETRAN Computer Code Reactor System Transient Analysis Model Qualification

CALVERT CLIFFS

BALTIMORE GAS AND ELECTRIC COMPANY

NUCLEAR GENERATION ENGINEERING SECTION

NUCLEAR LICENSING AND ANALYSIS UNIT

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TOPICAL REPORT

RETRAN COMPUTER CODE
REACTOR SYSTEM TRANSIENT ANALYSIS
MODEL QUALIFICATION

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LIST OF ACRONYMS

ADV	Atmospheric Dump Valve
AFAS	Auxiliary Feedwater Automatic Signal
AFW	Auxiliary Feedwater
APS	Auxiliary Pressurizer Spray
BG&E	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
CE	Combustion Engineering
CEA	Control Element Assembly
CIS	Containment Isolation Signal
CRS	Containment Radiation Signal
CSAS	Containment Spray Actuation Signal
CVCIS	Charging and Volume Control Isolation Signal
DBA	Design Basis Accident
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
EI	Energy Incorporated
EPRI	Electric Power Research Institute
ESFAS	Engineered Safety Features Actuation System
FRV	Feedwater Regulating Valve
GPM	Gallons Per Minute
HPI	High Pressure Injection
HPSI	High Pressure Safety Injection
IBM	International Business Machines, Inc.
IHTC	Interface Heat Transfer Coefficient
KW	Kilowatt
LANL	Los Alamos National Laboratory
LOCA	Loss of Coolant Accident
LOFT	Loss of Fluid Test
LOSP	Loss of Offsite Power
LPSI	Low Pressure Safety Injection
MFRV	Main Feedwater Regulating Valves
MFW	Main Feedwater
MSFCV	Main Steam Flow Control Valve
MSIV	Main Steam Isolation Valve

LIST OF ACRONYMS (Cont'd.)

MW	Megawatt
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OL	Operating License
PCS	Primary Coolant System
PORV	Power Operated Relief Valve
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protective System
SER	Safety Evaluation Report
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SGFP	Steam Generator Feedwater Pump
SGIS	Steam Generator Isolation Signal
SIAS	Safety Injection Actuation Signal
SRV	Safety Relief Valve
TBV	Turbine Bypass Valve
TBCS	Turbine Bypass Control System
TSV	Turbine Stop Valve
UGRA	Utility Group for Regulatory Action
VBUB	Bubble Rise Velocity

1.0 INTRODUCTION

During the past three years, The Baltimore Gas and Electric Company (BG&E) has been developing system transient thermal-hydraulic analysis capability for the Calvert Cliffs Nuclear Power Plant using the RETRAN computer code. The first application of this capability was to perform independent audit analyses of the TRAC Pressurized Thermal Shock (PTS) Project (Unresolved Safety Issue A-49) funded by the U. S. Nuclear Regulatory Commission (NRC) (1, 2, 3). Since then, BG&E has performed other best estimate calculations with RETRAN (4).

The purpose of this topical report is to document the BG&E best estimate RETRAN thermal-hydraulic analysis capability. The NRC has stated in Generic Letter No. 83-11 (5) that:

> "NRC's experience with safety analyses using large, complex thermal-hydraulic computer codes such as RELAP and TRAC has shown that a large percentage of all errors or discrepancies discovered in safety analyses can be traced to the user rather than to the code itself. This realization has led NRR to place additional emphasis on assuring the capabilities of the code users as well as on the codes themselves. . . The information we look for includes comparisons performed by the user of the code results to experimental data, plant operational data, or other benchmarked analyses."

This topical report was developed to demonstrate the capability of BG&E analysts to properly develop RETRAN models of Calvert Cliffs, perform calculations with these models to simulate realistic plant transient response and compare the results to measured plant data, another best

estimate computer code (TRAC) and experimental data (LOFT). Furthermore, model enhancement and appropriate sensitivity studies were performed to gain a greater insight and a better understanding of modeling Calvert Cliffs with RETRAN. In this manner, BG&E has established both the competence of our analysts to perform best estimate RETRAN calculations of Calvert Cliffs and completed an acceptable code verification in accordance with NRC Generic Letter No. 83-11. The basis for demonstrating BG&E RETRAN best estimate analysis capability is the comparison of five plant transients, two TRAC code analyses and two LOFT tests with BG&E RETRAN calculations.

2.0 DESCRIPTION OF CALVERT CLIFFS NUCLEAR POWER PLANT

The Calvert Cliffs Nuclear Power Plant consists of two units located in Lusby, Maryland on the Chesapeake Bay. Unit 1 began commercial operation in 1975 and Unit 2 in 1977. Each unit is currently licensed to operate at a core thermal power of 2700 Megawatts.

Bechtel Power Corporation was the architect-engineer-constructor of the facility. Each Nuclear Steam Supply System (NSSS) is a Combustion Engineering (CE) design pressurized water reactor with a 2x4 loop arrangement: two hot legs and two U-tube steam generators with four cold legs and four reactor coolant pumps as illustrated in Figure 2-1. The core consists of 217 14x14 CE fuel assemblies. The core is enclosed in a core shroud and core support barrel within the reactor vessel. The reactor vessel is typical of other CE plants except that it does not have a thermal shield.

One hot leg of the primary coolant system is connected through a surge line to a cylindrical pressurizer. The pressurizer regulates primary coolant system pressure and provides overpressure protection. Pressure regulation is performed by the pressurizer heaters and spray to maintain saturated steam and water volumes at thermal equilibrium within the pressurizer. Two power-operated relief valves and two spring-loaded safety valves mounted on the top of the pressurizer can discharge steam to the quench tank for overpressure protection. A small continuous flow is maintained through the spray lines to keep the spray lines and surge line warm to minimize thermal shock during transients and to maintain the water chemistry in the pressurizer equivalent to that of the reactor coolant. Approximately 20% of the pressurizer heaters are continuously energized through proportional controllers to provide makeup for steady state heat losses. Auxiliary spray supplied by charging

pumps provides an alternate means of depressurizing the primary coolant system.

The two U-tube steam generators in each unit transfer heat generated in the reactor core via the primary coolant system to the secondary system. The steam generators are recirculating U-tube design. Each steam generator supplies saturated steam through flow-limiting orifices to the main steam line leading to the high pressure turbine. After passing through the high pressure and low pressure turbines, the steam is condensed in the main condensers. Two Atmospheric Dump Valves (ADVs) discharge steam from the main steam line directly to the atmosphere when condensers are not available and are also used in conjunction with the turbine bypass system to maintain a programmed average primary coolant temperature following turbine trip. Above 70% power, the ADVs are activated by a quick-open signal, while below 70% power they are opened by the control system. Sixteen spring-loaded steam line Safety Relief Valves (SRV) provide steam line overpressure protection. Four Turbine Bypass Valves (TBV) are normally used to rapidly remove eactor coolant system stored energy and limit steam line pressure following a turbine-reactor trip by exhausting steam directly to the main condensers. Each of the two main steam lines is equipped with one Main Steam Isolation Valve (MSIV) which closes in the event of an excessive steam demand event.

The main feedwater system for each unit consists of two turbine-driven main feedwater pumps, each of which could support 70% power operation. Auxiliary Feedwater (AFW) is supplied by up to three AFW pumps on each unit, drawing on water from the condensate storage tanks. Each unit has two turbine-driven and one motor-driven AFW pump. One safety grade and two non-safety grade condensate storage tanks are available as AFW sources for both units.

The Calvert Cliffs Reactor Protective System (RPS) functions to protect the core and reactor coolant system pressure boundary by tripping the reactor core. The parameters which initiate reactor trip are:

High Rate-of-Change of Power

High Power Level

Low Reactor Coolant Flow

Low Steam Generator Water Level

Low Steam Generator Pressure

High Pressurizer Pressure

Thermal Margin/Low Pressure

Loss of Load

High Containment Pressure

Axial Flux Offset

High Differential Steam Generator Pressure

The Engineered Safety Features Actuation System (ESFAS) initiates the start of equipment which protects the public and plant personnel by mitigating the effects of design basis accidents (DBAs). The ESFAS comprises the following functions:

Safety Injection Actuation Signal (SIAS)

Containment Spray Actuation Signal (CSAS)

Containment Isolation Signal (CIS)

Containment Radiation Signal (CRS)

Recirculation Actuation Signal (RAS)

Steam Generator Isolation Signal (SGIS)

Auxiliary Feedwater Automatic Start (AFAS)

Charging and Volume Control Isolation Signal (CVCIS)

The Calvert Cliffs design includes a number of regulating systems which control key primary and secondary parameters to allow safe operation and ensure high plant availability. These regulating systems control: Control Element Assembly (CEA) position, boric acid concentration, reactor coolant pressure, pressurizer level, feedwater flow, primary coolant average temperature, main steam line pressure, steam generator level, and steam low to the main turbine generator. Important Calvert Cliffs design parameters are delineated in Table 2-1.

Table 2-1

Calvert Cliffs Key Design Parameters

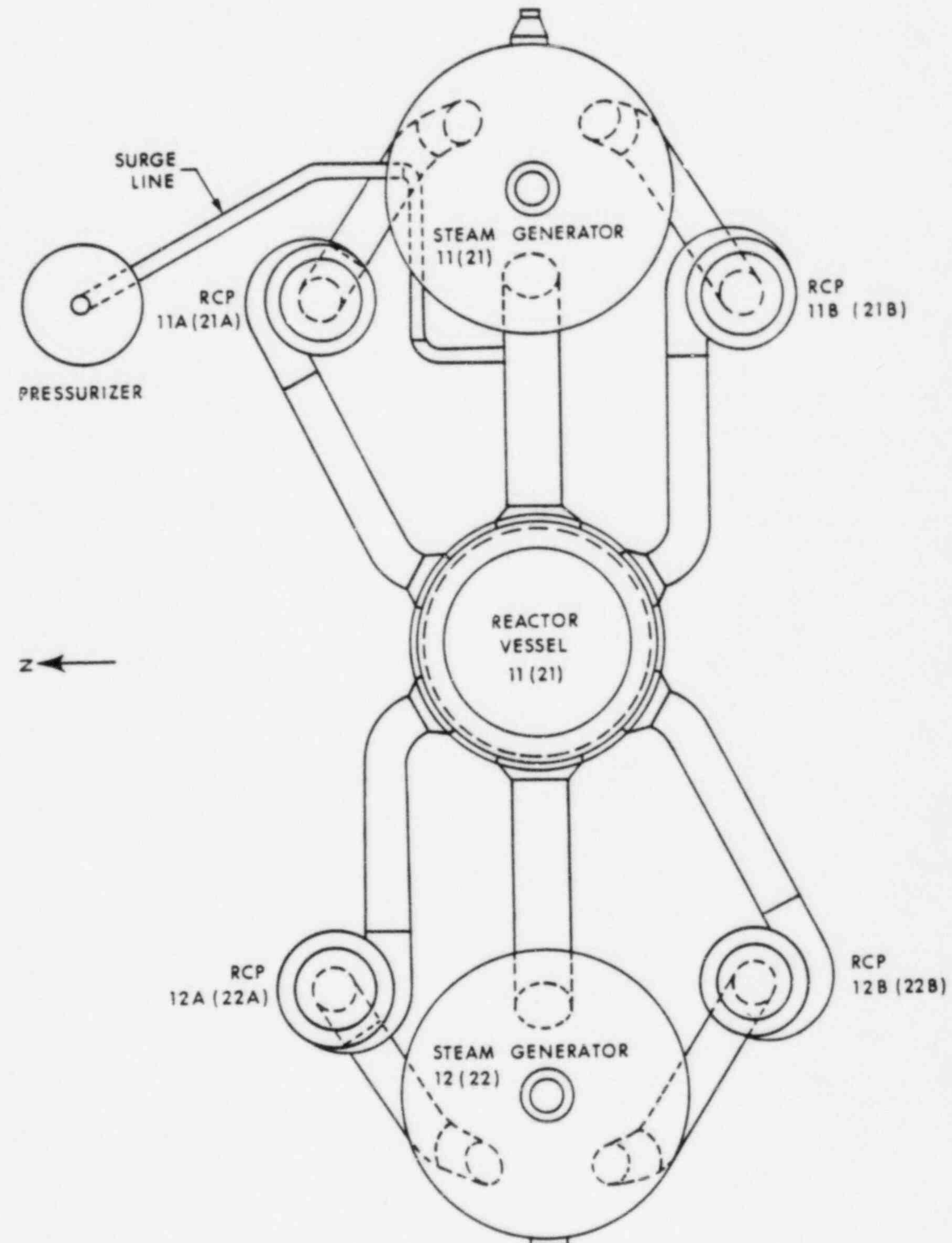
(each unit; except as noted)

Current Licensed Core Power:	2700 MW
Core:	217 CE 14x14 Fuel Assemblies
Primary Coolant System:	4 cold legs, 2 hot legs, 2 reactor coolant pumps, 2 U-tube steam generators, reactor vessel, pressurizer
Secondary System:	2 main steam lines, 2 main feedwater pumps, 3 auxiliary feedwater pumps (2 turbine-driven, 1 electric), 1 two-flow high pressure turbine, 3 two-flow low pressure turbines, one 3-shell condenser, 2 MSIVs
Primary Coolant System Volume (Excluding Pressurizer)	9601 ft^3
Normal Operating Pressure	2250 psia
Normal Operating Cold Leg Temperature	548^{o} F
Normal Operating Hot Leg Temperature	599.4^{o} F
Core Coolant Flow Rate	133.9×10^6 lb/hr
Pressurizer:	1500 $ft.^3$ volume (600-800 $ft.^3$ water, 700-900 $ft.^3$ steam)
	Proportional and Backup Electrical Heaters (1500 KW total)
	2 PORVs (1.354 in^2 flow area each)
	2 SRVs (1.872 in^2 flow area each)
	Normal Spray Maximum Flow - 375 GPM
	Auxiliary Spray Maximum Flow - 132 GPM

Steam Generator:	Number per Unit - 2
	Number of tubes per steam generator - 8519
	Normal Operating Secondary Size Pressure - 850 psia
	Steam flow per Steam Generator - 6×10^6 lb/hr
Main Steam Line:	1 Atmospheric Dump Valve (ADV) per line
	Total 5% Full Power Steam Flow - for both lines
	8 Safety Relief Valves (SRVs) per line
	Total 100% Full Power Steam Flow - for both lines
	4 Turbine Bypass Valves (TBVs)
	Total 40% Full Power Steam Flow
Safety Injection System:	3 High Pressure Safety Injection (HPSI) pumps (Shutoff head of 1275 psia)
	2 Low Pressure Safety Injection (LPSI) pumps (Shutoff head of 180 psia)
	4 Safety Injection Tanks (operating pressure of 215 psia)
	1 Refueling Water Tank (400,000 gallons)

FIGURE 2-1

CALVERT CLIFFS NUCLEAR STEAM SUPPLY SYSTEM LOOP ARRANGEMENT



3.0 RETRAN CALVERT CLIFFS MODELS

3.1 Description of RETRAN

RETRAN is a digital computer code designed to analyze nuclear power plant thermal-hydraulic transients. RETRAN was developed from the RELAP series of codes and sponsored by EPRI since 1975.

RETRAN simulates a nuclear power plant as a series of homogeneous volumes interconnected by flow path junctions. The conservation of mass, energy and momentum are maintained by RETRAN. Double-sided or adiabatic metal heat slabs and powered heat conductors (reactor core, passive pressurizer heaters) are explicitly modeled in RETRAN. A nonequilibrium thermal model and a bubble rise model are available for simulating the pressurizer, reactor vessel upper head and steam generator secondary side. Specific models for reactor coolant pumps, valves, control systems, reactor trips, critical (choked) flow, DNBR, transport delay, dynamic slip, steady state initialization, and reactor core neutronics are integral to RETRAN. RETRAN is described in greater detail in two EPRI reports (6, 7).

In October, 1979 and October, 1981, the Utility Group for Regulatory Action (UGRA) requested that the U.S. Nuclear Regulatory Commission (NRC) review two versions of the RETRAN code, RETRAN-01 Mod 003 and RETRAN-02 Mod 002 (8, 9). A Safety Evaluation Report (SER) for these codes was issued in September, 1984 (10), in which the NRC approved of the RETRAN code for performing transient thermal-hydraulics with certain restrictions. The Baltimore Gas and Electric Company uses RETRAN-02 Mod 003 to perform the analyses described in this document. The code modifications made to RETRAN-02 Mod 002 that resulted in RETRAN-02

Mod 003 are a direct result of the NRC and EPRI design review and user-found error correction. Modifications were performed in accordance with the EI/EPRI RETRAN project Quality Assurance Procedures which conforms with 10 CFR 50, Appendix B.

The Calvert Cliffs RETRAN models and calculations discussed in the following sections were developed, documented and verified in accordance with BG&E engineering calculation procedures.

3.2 RETRAN One Loop Model of Calvert Cliffs

The one loop RETRAN model of Calvert Cliffs was developed to analyze symmetrical transients (i.e., both steam generators, hot legs and all four cold legs/RCPs respond identically). This model, presented in Figure 3.2-1, consists of 32 volumes, 50 junctions and 41 heat slabs. Typical run times on the BG&E IBM 3081 mainframe computer for this model are equal to or faster than real (transient) time.

The one loop model explicitly simulates the entire primary coolant system and that portion of the secondary system from the steam generator to the turbine inlet. The reactor vessel comprises nine volumes which include a three-volume core, downcomer, lower plenum-head, bypass, upper plenum and upper head regions. A nonequilibrium volume is used for the reactor vessel upper head. All heat conductors within the vessel are modeled as RETRAN heat slabs.

The one loop model single hot leg, cold leg and reactor coolant pump (RCP) represent the combination of the two hot legs, four cold legs and four RCPs actually in the Calvert Cliffs design. While the hot leg is a single volume, the cold leg and RCP are represented as five RETRAN volumes. This detail allows for an accurate simulation of the loop seal elevation difference; charging, letdown and safety injection temporal effects; and proper pump performance characteristics. Detailed RCP performance curves are included in this model.

The one loop model single U-tube steam generator (representing two steam generators) consists of eight nodes for the primary side and one node for the secondary. The U-tubes are modeled with six volumes and the inlet and outlet plenum as one each. This allows for sufficient detail to better

predict degraded heat transfer if tube uncovery occurs. The single volume secondary side was found to produce acceptable results without the convergence problems and associated longer computer run times of a multi-noded recirculating steam generator secondary side RETRAN model.

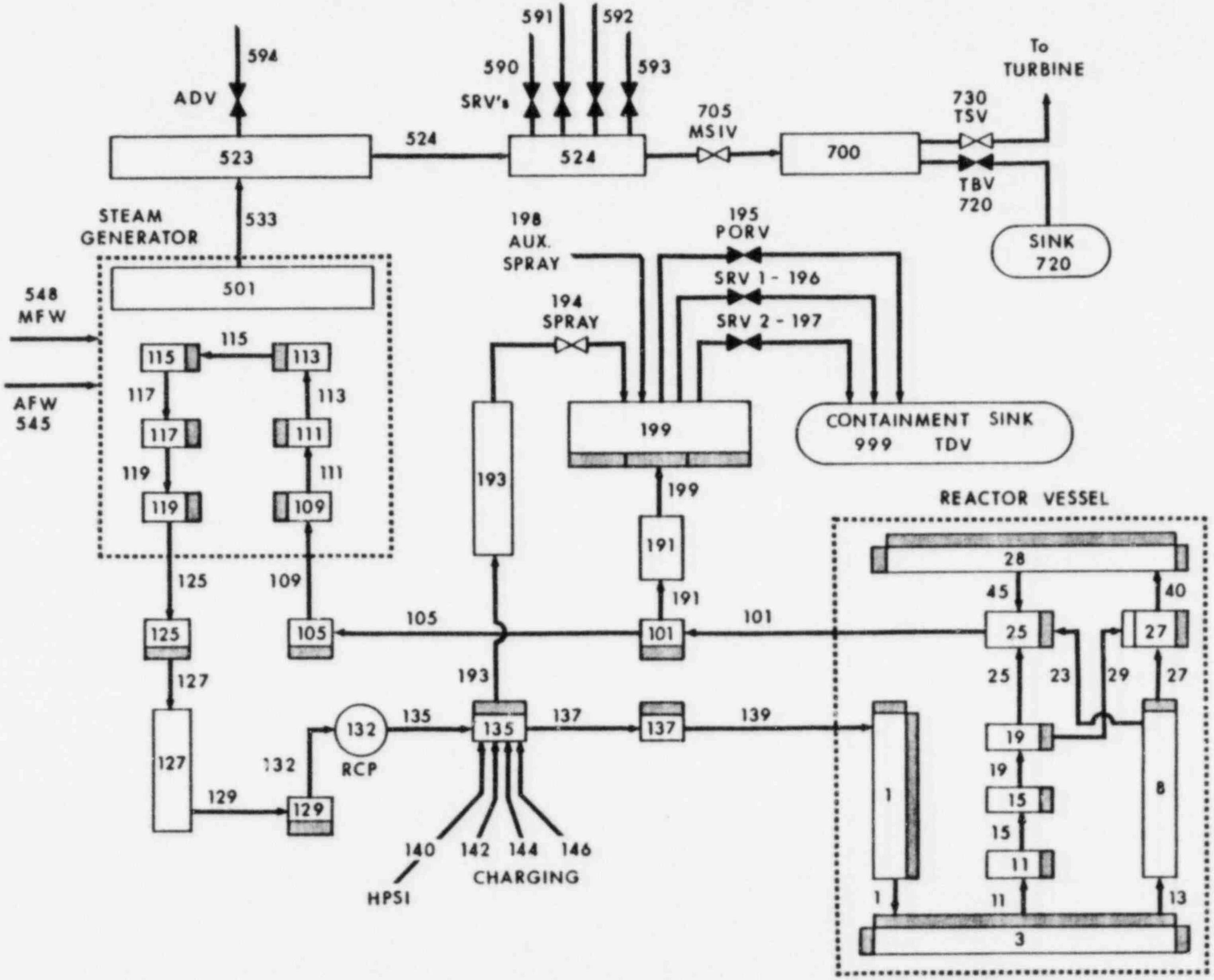
The pressurizer system is segmented into three volumes: one for the surge line, one for the spray line and one nonequilibrium volume for the pressurizer itself. All pressurizer pressure and level control systems are explicitly modeled. This includes normal and auxiliary sprays, power operated relief valves (PORVs), safety relief valves (SRVs) and pressurizer heaters.

The secondary side flow path begins with a fill junction boundary condition for the main and auxiliary feedwater flow into the steam generator shell. The steam generator's steam flows through a three volume steam line before entering either the turbine or condenser boundary conditions. The turbine boundary condition is a RETRAN negative fill junction which can be isolated by the turbine stop valve (TSV). The condenser boundary condition, a RETRAN time dependent volume, is normally isolated by a closed turbine bypass valve (TBV) which is actually representing four TBVs in the plant. One volume upstream of these two flow paths represents the steam header which would normally be connected to two main steam lines from their respective steam generators. This volume is isolable by closure of the main steam isolation valve (MSIV) modeled upstream in the RETRAN model. The next volume upstream of the MSIV is connected to four safety relief valve (SRV) flow boundary conditions. They simulate the sixteen SRVs at Calvert Cliffs. In actuality, there are eight SRVs on each main steam line with four different opening pressure setpoints for each pair of SRVs. Thus, each of the four SRVs in this RETRAN model accurately simulate the opening pressure setpoint and flow characteristics of four SRVs at the plant. The last main steam line

volume upstream of the SRV volume and downstream of the steam generator is used to provide the actual physical separation of the atmospheric dump valves (ADVs) from the SRVs as well as the flow restrictor at the steam outlet of the steam generator. The RETRAN model ADV flow boundary condition actually represents the two ADVs at Calvert Cliffs.

The RETRAN secondary side model includes control systems for steam generator level, TBV operation controlling pressure, ADV operation controlling average primary coolant temperature and trips controlling TSV, MSIV and SRV operation. Along with the aforementioned control systems and trips, other transient-specific trips have been incorporated into this model.

FIGURE 3.2-1
RETRAN ONE LOOP CALVERT CLIFFS MODEL



3.3 RETRAN Two Loop Model of Calvert Cliffs

The RETRAN two loop model of Calvert Cliffs was designed to analyze loop asymmetric transients (i.e., one steam generator and/or associated primary coolant piping responds differently than the other). The two loop model, depicted in Figure 3.3-1, used 71 volumes, 111 junctions and 92 heat slabs, and runs slower than real (transient) time on the BG&E IBM 3081 mainframe computer.

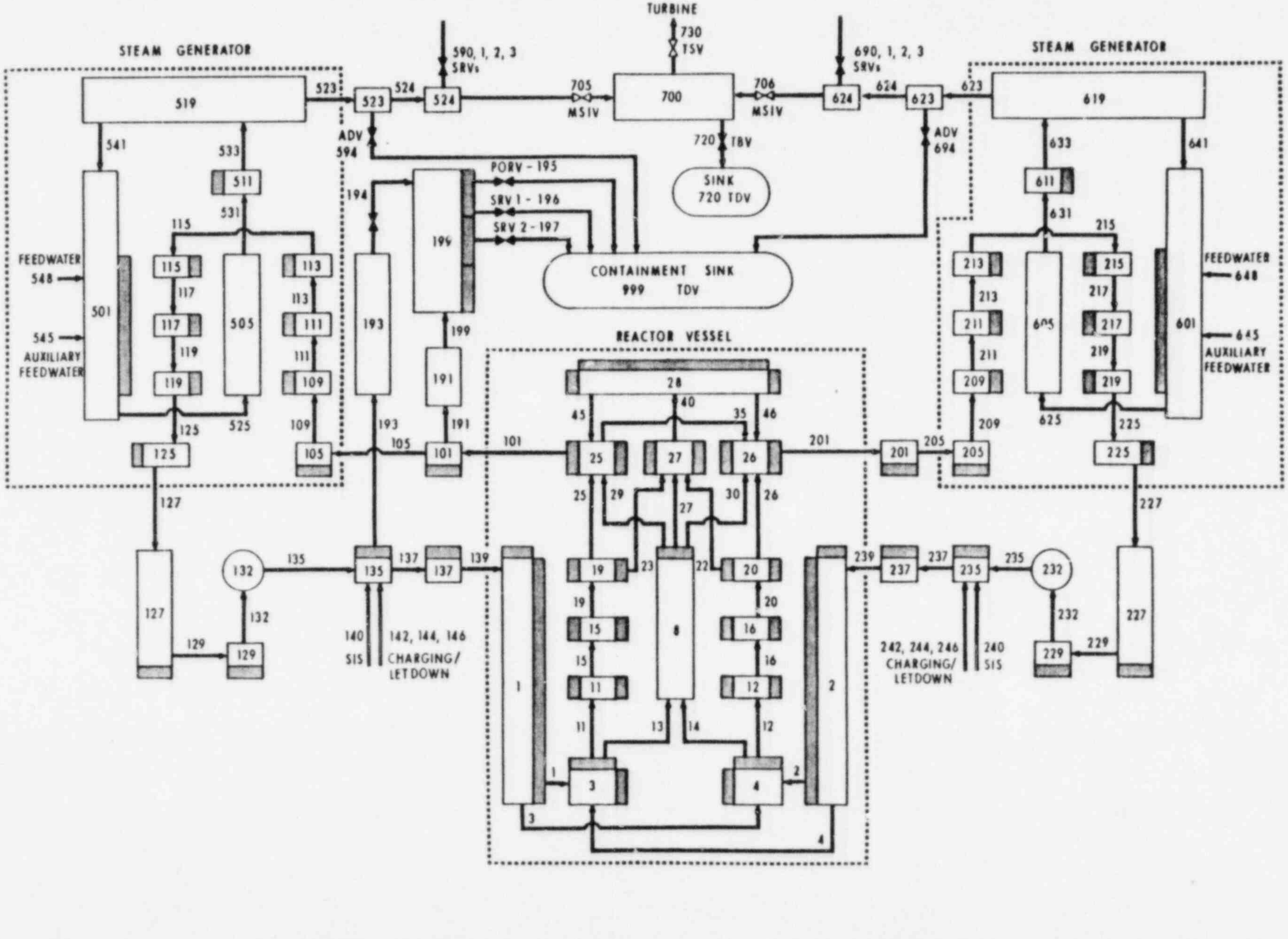
The two loop model explicitly models each steam generator, main steam line and hot leg at Calvert Cliffs. Two cold legs are modeled, one for each steam generator, each actually representing two cold legs and two RCPs. The number of volumes per hot leg, cold leg, main steam line, steam generator primary side and for the pressurizer are identical to that used in the one-loop model.

Increased volume nodalization and detail was incorporated into the reactor vessel and steam generator secondary side for the two loop model. The reactor vessel, made up of 15 volumes, was segmented into a split downcomer, lower plenum, core, and bypass model. Each loop (i.e., hot and cold leg from a common steam generator) is connected to its own downcomer - lower plenum - core - upper plenum chain of volumes. All realistic cross flow and bypass flow paths are included between the two parallel sets of volumes. This split-core reactor vessel is especially important in analyzing asymmetric loop effects on reactor vessel thermal-hydraulic response.

The steam generator secondary side is modeled as a four volume recirculating steam generator with a best estimate recirculation ratio. This model more accurately predicts steam generator performance, but requires considerably greater computer run time and modelling skills for a satisfactory

steady state and transient calculations. All other RETRAN model features (valves, control systems, trips, pump models, etc.) are identical to the one loop model described in Section 3.2.

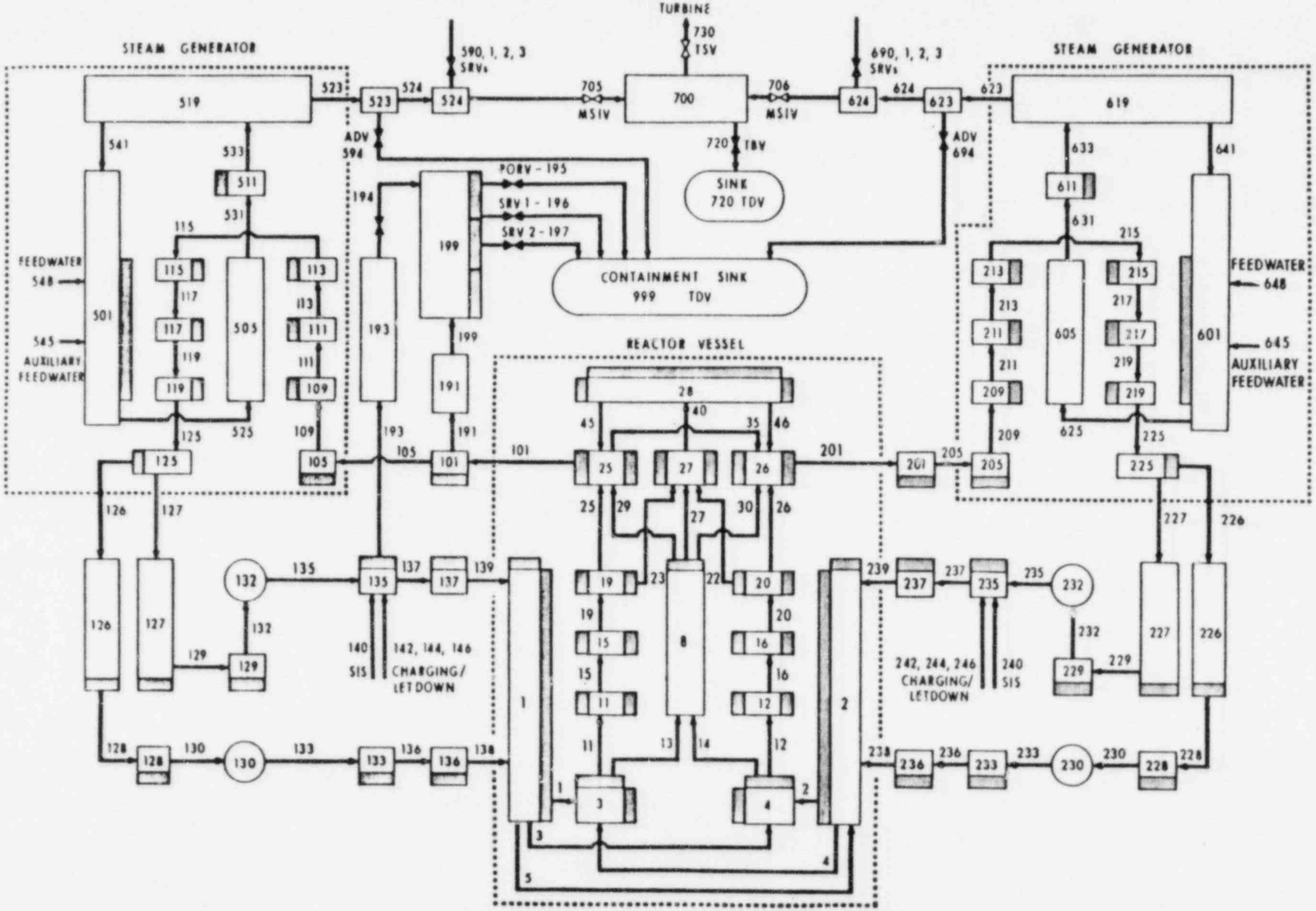
FIGURE 3.3-1
RETRAN TWO LOOP CALVERT CLIFFS MODEL



3.4 RETRAN Four Loop Model of Calvert Cliffs

The RETRAN four loop model of Calvert Cliffs, depicted in Figure 3.4-1, explicitly simulates the two hot leg and four cold leg geometry of Calvert Cliffs. This model utilizes 81 volumes, 123 junctions and 102 heat slabs, and as expected, is the slowest running model on the BG&E IBM 3081 mainframe computer. This model is only required when individual cold leg/RCP transients need to be analyzed. In some cases, simplification of this model may be warranted to reduce computer run times without affecting the accuracy of results. For example, the pump combination flow test simulation discussed in Section 4.3 of this report utilizes a steam generator secondary side energy removal boundary condition in lieu of the five steam generator volumes and three main steam line volumes per loop. All other features for the four loop model are identical to the two loop model described in Section 3.3.

FIGURE 3.4-1
RETRAN FOUR LOOP CALVERT CLIFFS MODEL



4.0 PLANT TRANSIENT COMPARISONS

4.1 RETRAN 02 Analysis of a Calvert Cliffs Multiple Secondary Side Malfunction Event

4.1.1 Description of Plant Event

On October 11, 1983 at 11:36:46 (time T), Unit 2 Steam Generator Feedwater Pump (SGFP) No. 22 tripped on low oil pressure caused by a loose lubricating oil connection and oil leakage (11, 12, 13). The operators began reducing reactor power by inserting control rods and borating while simultaneously reducing turbine power by partially closing the turbine governor valves. These actions were intended to bring reactor power down to a level (70%) that could be supported by one main feedwater pump without tripping the reactor or exceeding technical specification limits.

At T + 48 seconds, pressurizer pressure peaked at 2306 psia (159 MPa) due to a reactor power to steam flow mismatch caused by excessive throttling of the turbine governor valves. Subsequently, pressurizer level peaked at 254 inches (6.5 m) at T + 60 seconds. At T + 62 seconds, reactor trip occurred on low steam generator level (-50 inches/-1.3 m) resulting from the power to steam mismatch. At T + 66 seconds, SG 22 secondary pressure peaked at 1030 psia (71 MPa) because the reactor power exceeded steam flow due to the rapidly closing Turbine Stop Valve (TSV) and relatively slower opening TBVs and ADVs. The peak pressure was sufficient to lift some of the main steamline safety valves.

Following reactor trip, No. 21 FRV failed to ramp down to its programmed 5% flow position and remained fully open instead. Also, No. 21

SGFP governor valve stuck on high speed stop causing maximum 21 SGFP flow. These feedwater malfunctions caused SG 21 level to reach its maximum measurable value at T + 180 seconds while SG 22 level reached its minimum measurable value at T + 90 seconds. The operators isolated 21 SGFP at T + 340 seconds.

Upon reactor trip, the TSV rapidly closed, and the TBVs and ADVs opened. Although programmed to close at 870 psia (60 MPa), one of the TBVs failed to close at that pressure, remaining partially open until it was isolated by the operators at T + 300 seconds. This partially open TBV caused a continued cooldown and depressurization of the primary and secondary coolant systems. The operators later initiated Auxiliary Feedwater (AFW) to SG 22 and isolated AFW to SG 21 while using the Steam Generator Blowdown System (SGBS) to lower SG 21 level.

During the transient, a minimum pressurizer pressure of 1659 psia (114.4 MPa) and level of 18.5 inches (0.5 m) were reached at T + 244 and T + 258 seconds, respectively. Primary coolant loop 21 cold leg temperature dropped to a minimum value of 509°F (538 K) at T + 233 seconds representing a 49°F (27 K) drop in 184 seconds. Safety Injection Actuation Signal (SIAS) on 1725 psia (119 MPa) occurred at T + 175 seconds, but no High Pressure Safety Injection (HPSI) pump flow occurred because HPSI shutoff head is 1275 psia (88 MPa). After isolating 21 SGFP and the stuck open TBV, the operators restored normal pressurizer pressure and level, primary coolant system temperatures, and steam generator levels.

In summary, the October 11, 1983 event at Calvert Cliffs Unit 2 was initiated by the loss of one main feedwater pump which was later compounded by failures in the other feedwater pump. This condition caused the overfill of one steam generator and the underfill of the other steam generator following

a reactor trip on low steam generator level. This asymmetric cooldown-depressurization transient was further exacerbated by one turbine bypass valve failing to close.

4.1.2 Event Specific RETRAN Modeling Changes

The BG&E RETRAN model of Calvert Cliffs used for this analysis is the two loop model depicted in Figure 3.3-1. Several model refinements were added for this analysis. A stack of three heat conductors were incorporated into the pressurizer to account for heat transfer between the vapor-liquid volumes and the metal walls which has been shown to be important (14, 15). The spray line volume fluid was initialized at its actual measured temperature which is about $7^{\circ}F$ (4 K) less than the cold leg temperature due to ambient heat losses from this relatively stagnant pipe (16). A realistic time delay of two minutes was added to the pressurizer heater model to account for the thermal time lag constant of these electrical heating elements (17). Other model changes were made to simulate plant initial conditions and the transient specific malfunctions and operator actions.

RETRAN simulation of this event used measured main feedwater flow to each steam generator and core thermal power as temporal boundary conditions. The purpose of this analysis was to analyze the NSSS primary and secondary thermal-hydraulic response to these boundary conditions along with other systems actuating (e.g., TBV, pressurizer spray, etc.). The core thermal power and main feedwater flow boundary conditions are presented in Figures 4.1-1 and 4.1-2.

The TSV closing and TBV-ADV opening times were inferred from reactor trip time (plant data on reactor power and steam generator level). At

steam pressures below 870 psia (60 MPa) and until T + 300 seconds, the TBV flow area was modeled as one half of the area of one TBV. As previously discussed, operators reduced reactor power and turbine power to match the 70% feedwater flow available and avoid a reactor trip. Based on measured plant data on the first stage turbine pressure reduction prior to trip and turbine manufacturer's performance data (18), a reduction to 84% of full power turbine steam flow was used in this analysis. Auxiliary feedwater (AFW) simulation was accomplished by allowing AFW initiation on the low-low SG level of -170 inches (-4.3 m), but only allowing AFW flow to SG 22.

4.1.3 Comparison of RETRAN Results to Plant Data

A comparison of RETRAN results with measured plant data during the 360 second period after 22 SGFP trip for six key parameters is presented in Figures 4.1-3 to 4.1-8. RETRAN pressurizer pressure peaks at about the same time but to a somewhat higher value as shown in Figure 4.1-3. The subsequent drop in pressure due, at first, to reactor trip and later, to the open TBV compares well with the exception that RETRAN minimum pressure is again somewhat higher than the plant data. The pressurizer level comparison in Figure 4.1-4, shows RETRAN peaking at about the same time, but not as high as plant data. The drop in level compares well with RETRAN level higher than plant measured level. One would expect the overprediction in peak pressure to be accompanied by an underprediction in peak level.

The Loop 21 hot and cold leg temperature comparisons in Figures 4.1-5 and 4.1-6 show excellent agreement during the first 160 seconds of the transient analysis. This agreement substantiates the selection of reduced turbine steam flow prior to reactor trip since the rise in these temperatures

is due to this pre-trip power-steam flow mismatch. A good overall agreement exists after 160 seconds. Possibly higher run-out AFW pump flow rates and lower post-trip MFW temperatures could cause the more rapid cooldown after 160 seconds in plant data. The difference between RETRAN and plant data in this later period is the reason for RETRAN having higher minimum pressurizer pressure and level in Figures 4.1-3 and 4.1-4 which are due primarily to the lower temperature coolant density change.

Figures 4.1-7 and 4.1-8 show comparisons of steam generator secondary side pressure and narrow range level. Steam generator pressure response was identical for both generators with the exception that their pressures differ by their initial values throughout the transient, thus only SG 22 is shown in Figure 4.1-7. RETRAN predicts a peak pressure very close to plant data both in magnitude and time with a good agreement in the post-trip depressurization phase of the transient. Differences in the comparison of SG pressure are attributed to the exact timing of TSV closing, TBV/ADV opening and the subsequent characteristics of the one TBV that failed to close at 870 psia (60 MPa). The plant data in Figure 4.1-7 indicates that the stuck-open TBV did not simply maintain a half open position until isolated by the operators at 300 seconds, but rather varied its position. Also, the RETRAN model does not account for miscellaneous steam loads (e.g., MFW pump and AFW pump turbine). This can account for 1% of full power steam flow or 10% of a single TBV's flow. The difference in SG pressure response accounts for differences in primary coolant system cooldown after 160 seconds. Figure 4.1-8 shows close agreement between RETRAN and plant data for SG narrow range level. RETRAN results indicate similar timing for SG 22 underfill and SG 21 overfill as well as showing the effect of later AFW to SG 22 and the subsequent rising level.

RETRAN calculated hot and cold leg temperatures for the two loops demonstrate the asymmetry of this cooldown transient. The two legs differ by $3^{o}F$ (1.7 K) for the hot leg and $8^{o}F$ (4.4 K) for the cold leg. Data for the other loop was not available from the plant computer for this event.

Table 4.1-1 shows a comparison of sequence of event timing between plant data and RETRAN. RETRAN calculated timing agrees closely with plant data. Table 4.1-2 compares key thermal-hydraulic parameters which also show good agreement between RETRAN and plant data. The comparisons in Table 4.1-2 should be evaluated in the context of the accuracy of measured plant data (19) presented in Table 4.1-3. RETRAN calculated values fall within the measured plant data tolerances for peak pressurizer pressure, steam generator pressure and loop 21 peak hot and cold leg temperatures. Values are reasonably close for peak pressurizer level, minimum pressurizer pressure and minimum loop 21 cold leg temperature. The differences in minimum pressurizer level and minimum loop 21 hot leg temperature are significant. These parameters occur about 4-5 minutes into the transient and are a direct result of the stuck-open TBV-induced cooldown. As previously mentioned, the RETRAN simulation assumed this TBV at half its normally open area until isolation whereas a time varying flow area most probably existed.

4.1.4 RETRAN Model Sensitivity Study

A sensitivity study was performed with the RETRAN pressurizer model to evaluate the effects of model changes on calculated peak pressure and level during the insurge portion of the transient. The RETRAN pressurizer sensitivity study for this event evaluated the effects of the following

features: spray junction model, rainout velocity, interface heat transfer coefficient, heat conductors, initial spray line temperature, and number of pressurizer volume nodes. Based on this sensitivity study, the following conclusions are made regarding RETRAN pressurizer modeling for rapid insurge transients (0.8 inches/sec or 2.0 cm/sec):

1. Spray junction model, rainout velocity, and presence or noding of heat conductors have no significant effect on peak pressurizer pressure (1 psi or .07 MPa) or level.
2. Best estimate initial spray line fluid temperature will slightly reduce (2 psi or .14 MPa) peak pressurizer pressure.
3. Two node pressurizer results in slightly higher peak pressure (3.4 psi or .23 MPa) and higher peak level (5.2 inches or .13 m).
4. Large values of surge line loss factor significantly reduce peak pressure (28 psi or 1.90 MPa for a tenfold increase).
5. Large values of interface heat transfer coefficient (IHTC) (10,000 to 50,000 Btu/Hr-Ft^2-F or 56,780 to 283,900 W/M^2-K) significantly reduces peak pressure (26-56 psi or 1.8-3.8 MPa) and increases peak level. Such values of this parameter also reduce the time of peak pressure and level.

The final RETRAN pressurizer model used for this analysis utilized best estimate initial spray line fluid temperature, "spray" junction option, a stack of three vertical heat conductors each segmented into 100 equal width nodes, single volume node, and an interface heat transfer coefficient of 32,400 Btu/Hr-Ft^2-F (183,967 W/M^2-K) based on recommendations for the TOPS code (20). This very large value of IHTC, although not realistic, accounts for steam condensation which is not adequately modeled in RETRAN. The greater condensation causes lower peak pressure, higher level and moves both

of these peaks back to an earlier time which more closely corresponds to plant measured data.

Even with the high IHTC, RETRAN predicts about 72% of the pressurizer level swell from plant data. This points to the calculation of less steam condensation than actually occurred. Investigation of spray effectiveness with RETRAN involved a case in which the spray was deactivated. This resulted in a peak pressurizer pressure of 2411 psia (166.3 MPa) which was limited by PORV opening at 2400 psia (165.5 MPa). This spray effectiveness of 73 psia (5 MPa) in 33 seconds is within the range set by CE and confirmed during Calvert Cliffs startup tests (21).

The actual spray flow rate is not tested but has been shown to be as much as twice the design flow in some CE plants. A RETRAN case with twice design rated spray flow resulted in a peak pressurizer of 2300 psia. Another RETRAN case was run in which the spray was set to open to its fully rated flow at 2300 psia (158.6 MPa) rather than its programmed flow characteristic of opening at this pressure, but not reaching fully rated flow until 2350 psia (162 MPa). This case resulted in a peak pressure of 2314 psia (159.6 MPa) and a peak level of 246.3 in. (6.3 m). The performance of spray valves is not tested to ensure the linear flow increase between 2300 psia (158.6 MPa) and 2350 psia (162 MPa) that is assumed in the RETRAN model.

One process which may not be adequately simulated is the condensation of insurge induced superheated steam in the vapor space onto the pressurizer metal walls. These walls are modeled with RETRAN in fine heat conductor nodes (.05 in. or .0014 m thick). The surface heat transfer coefficient is internally calculated by RETRAN using a range of available correlations. During the insurge, RETRAN selected the Collier correlation (22, 23) which resulted in a heat transfer coefficient on the order of 300-400 Btu/Hr-Ft^2-F

(1703 to 2271 W/M^2-K). Steam condensation experimental data indicates values of 700-2500 Btu/Hr-Ft^2-F (3975 to 14195 W/M^2-K) (24, 25, 26). Saedi and Kim (14, 15) have shown that heat conduction to the pressurizer walls substantially reduces pressure during insurge transient. Calculations of the possible condensation effect of the ambient heat loss and balancing continuous spray show that they would only account for less than 1% of the difference between RETRAN peak level and plant data.

Finally, the presence and amount of non-condensible gases in the pressurizer vapor and liquid space were investigated. For oxygen scavenging purposes, a concentration of 25-50 cc hydrogen per Kg of primary coolant water is maintained. This along with smaller quantities of other dissolved gases corresponds to less than 1% mass fraction in the vapor space (27). The effect of this concentration of non-condensible gases in the pressurizer vapor space has been shown to be insignificant by Leonard (28).

4.1.5 Summary of Results

BG&E performed a RETRAN analysis of the asymmetric cooldown transient which occurred at Calvert Cliffs Unit 2 on October 11, 1983. Comparisons with measured plant data for pressurizer pressure and level, hot and cold leg temperature, and steam generator secondary side pressure and level all showed excellent agreement in both peak values and overall trends. The uncertainty in stuck open TBV position affected the RETRAN simulation of the later phase of the cooldown.

A sensitivity study was performed to evaluate RETRAN pressurizer models for rapid pressurizer insurge transients (0.8 inches/sec or 2.0 cm/sec). This study indicated that large interface heat transfer coefficients

or large values of surge line loss factor significantly reduce the magnitude and time of peak pressurizer pressure and increase peak level. Spray junction model, rainout velocity, presence and noding of heat conductors, realistic initial spray line fluid temperature, and a two node model of the pressurizer had a small or insignificant effect on pressure and level for this type of transient.

The RETRAN models for pressurizer spray heat transfer are a reasonable representation. The condensation heat transfer coefficients calculated by RETRAN for heat transfer from the vapor space to the walls appear to be low. The inability of RETRAN to model a steady state in which continuous spray flow and heater operation exist does not significantly affect its transient simulation capability.

Results of the pressurizer spray sensitivity analysis in Section 6.0 indicate that actual plant spray flow at 2300 psia (158.6 MPa) is greater than that assumed in the RETRAN model. The actual plant continuous spray flow is seventeen times larger than the value reported in the FSAR (29). This larger initial spray valve open area would result in full spray flow delivery at a lower pressure than the programmed 2350 psia (162 MPa). It should be noted that there are no tests performed to confirm spray flow vs pressurizer pressure. This higher spray flow at 2300 psia (158.6 MPa) would reduce RETRAN calculated peak pressurizer pressure and increase peak level to more closely agree with plant data.

Table 4.1-1

October 11, 1983 Calvert Cliffs 2 Event Timing Comparison

Event	Time, seconds (normalized to 11:36:46)	
	Plant Data	RETRAN
SGFP 22 Trip	0	0
Peak Pressurizer Pressure	48	53
Peak Pressurizer Level	60	66
Reactor Trip on Low SG Level	62	62
Peak Loop 21 T-Hot	64	64
Peak SG 22 Pressure	66	69
Peak Loop 21 T-Cold	70	70
SG 22 at Minimum Level	90	116
SIAS	175	191
SG 21 at Maximum Level	180	206
Minimum Pressurizer Pressure	244-256	261
Minimum Pressurizer Level	258	261
Open TBV Isolated	300	310
SGFP 21 Isolated	340	340

Table 4.1-2

Tolerance of Calvert Cliffs Plant Computer Data

Pressurizer Pressure	$\pm$ 2.0%
Pressurizer Level	$\pm$ 2.5%
Hot/Cold Leg	$\pm$ 2.5°F
Core Thermal Power	$\pm$ 3.0%
Steam Generator Level	$\pm$ 2.0%
Steam Generator Pressure	$\pm$ 2.5%
Main Feedwater Flow	$\pm$ 2.0%

Table 4.1-3

Comparison of Key Event Parameters

	Plant Data	RETRAN
Peak Pressurizer Pressure	2306 psia/159 MPa	2338 psia/161 MPa
Peak Pressurizer Level	254 in./6.5 M	244.4 in./6.2 M
Peak Loop 21 T-Hot	597.1°F/586.9 K	597.5°F/587.2 K
Peak Loop 21 T-Cold	558.7°F/566.0 K	561 °F/566.9 K
Peak SG 22 Pressure	1030 psia/71 MPa	1034 psia/71.3 MPa
Minimum Pressure	1659 psia/114.4 MPa	1717 psia/118.4 MPa
Minimum Level	18.5 in./0.47 M	61.2 in./1.55 M
Minimum Loop 21 T-Hot	513.7°F/540.6 K	526.2°F/547.6 K
Minimum Loop 21 T-Cold	509.5°F/538.6 K	513.6°F/540.6 K

FIGURE 4.1-1
OCTOBER 11, 1983 EVENT CORE THERMAL POWER RESPONSE

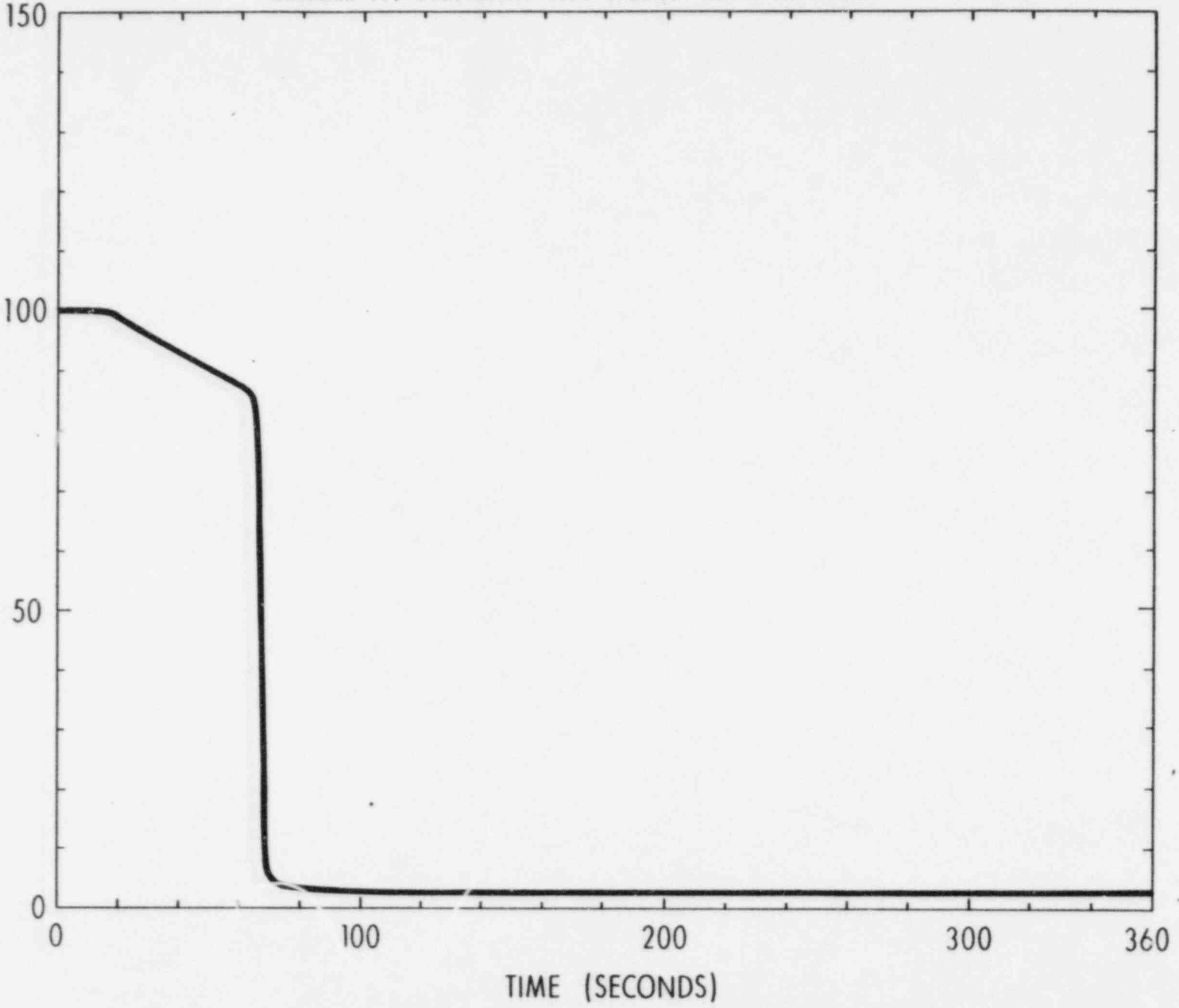


FIGURE 4.1-2
OCTOBER 11, 1983 EVENT MAIN FEEDWATER FLOW RESPONSE

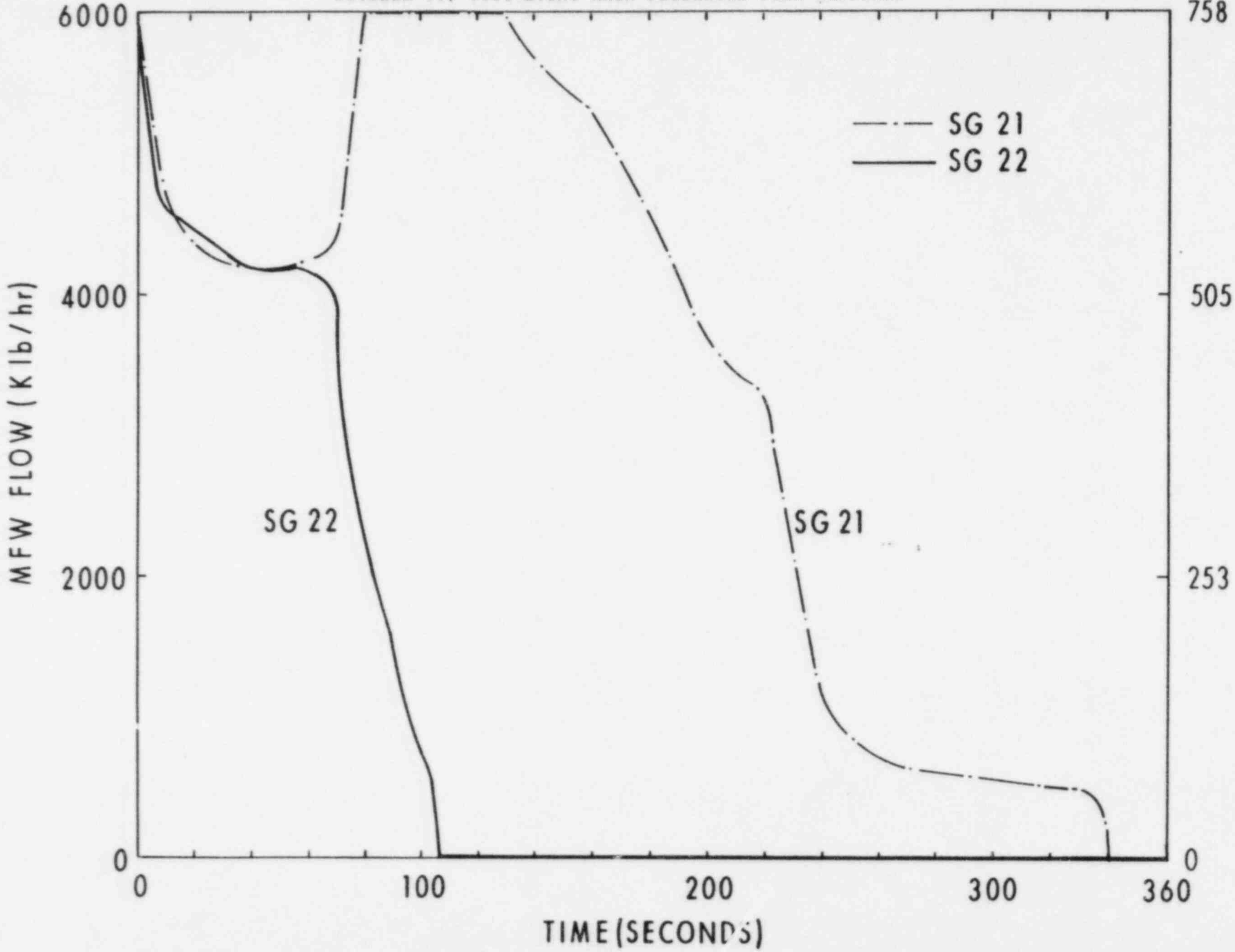


FIGURE 4.1-3
OCTOBER 11, 1983 EVENT PRESSURIZER PRESSURE RESPONSE

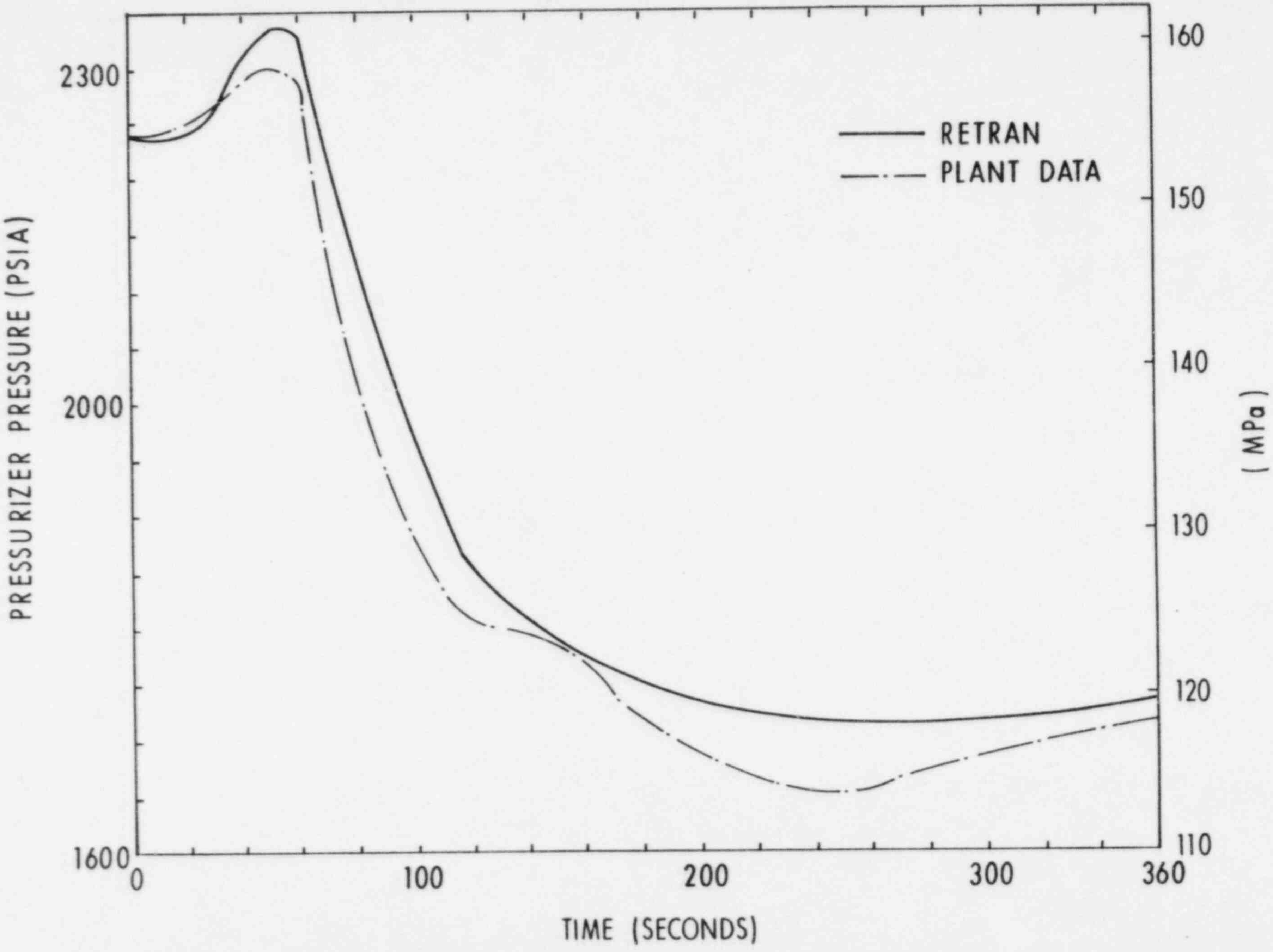


FIGURE 4.1-4
OCTOBER 11, 1983 EVENT PRESSURIZER LEVEL RESPONSE

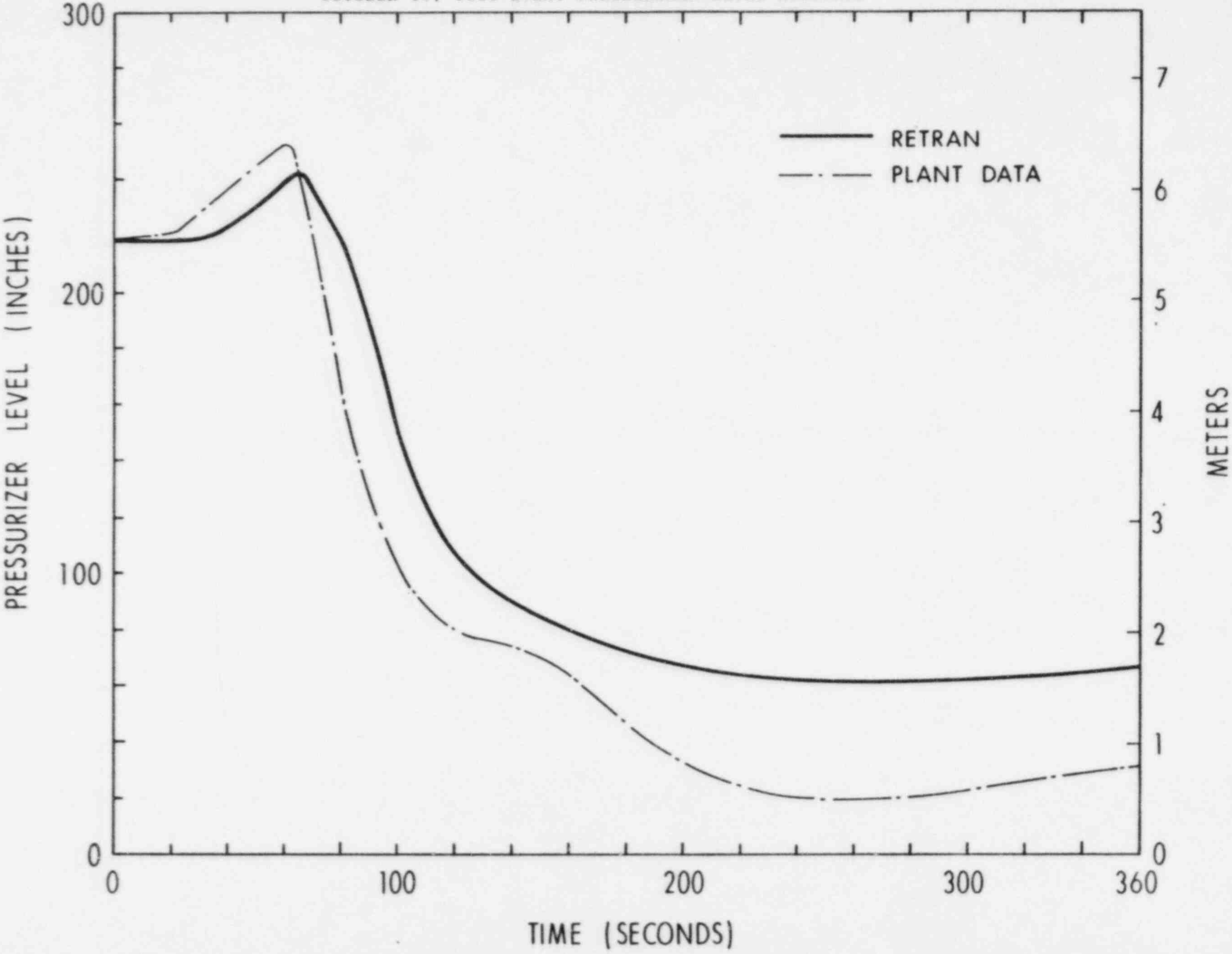


FIGURE 4.1-5
OCTOBER 11, 1983 EVENT LOOP 21 HOT LEG TEMPERATURE RESPONSE

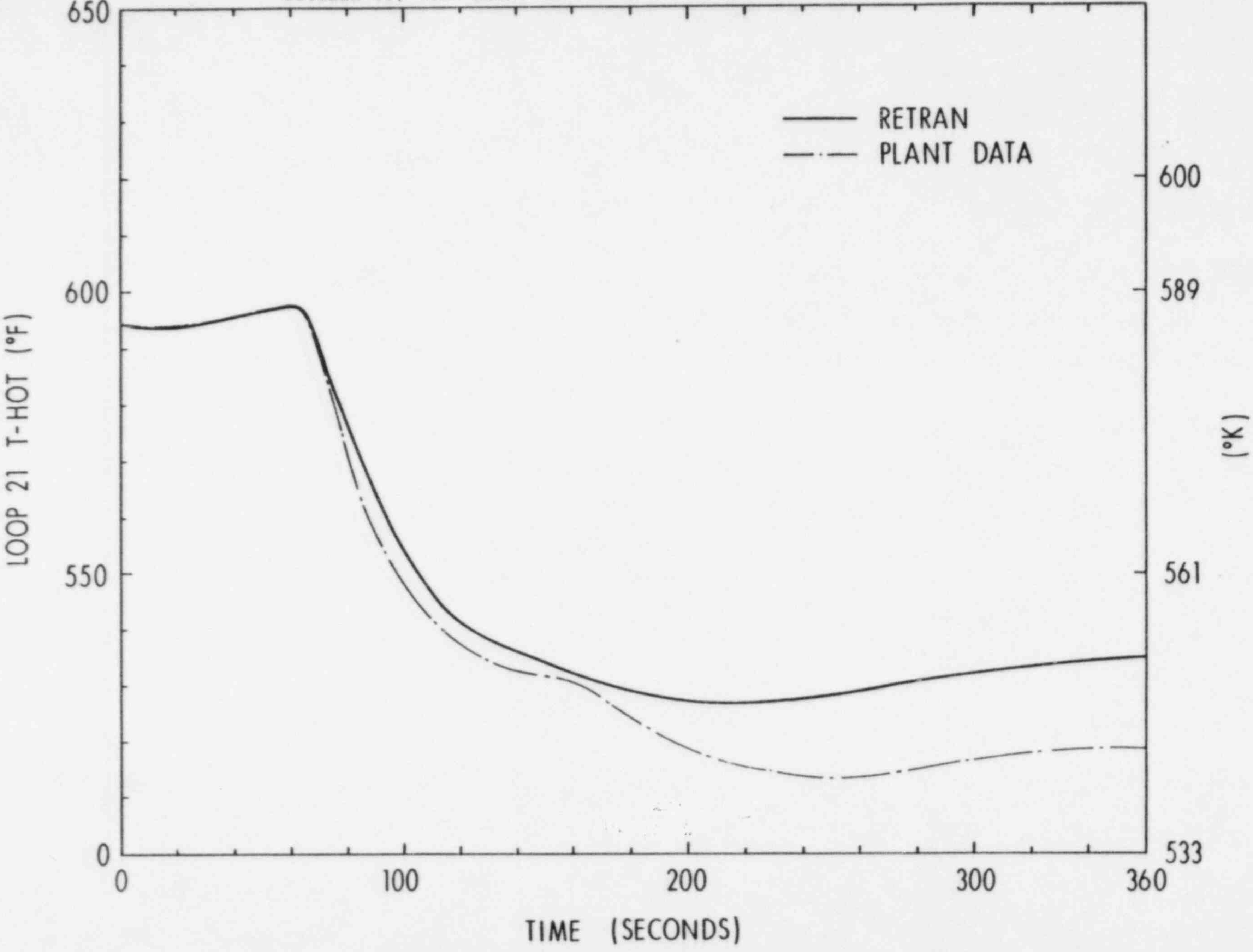


FIGURE 4.1-6
OCTOBER 11, 1983 EVENT LOOP 21 COLD LEG TEMPERATURE RESPONSE

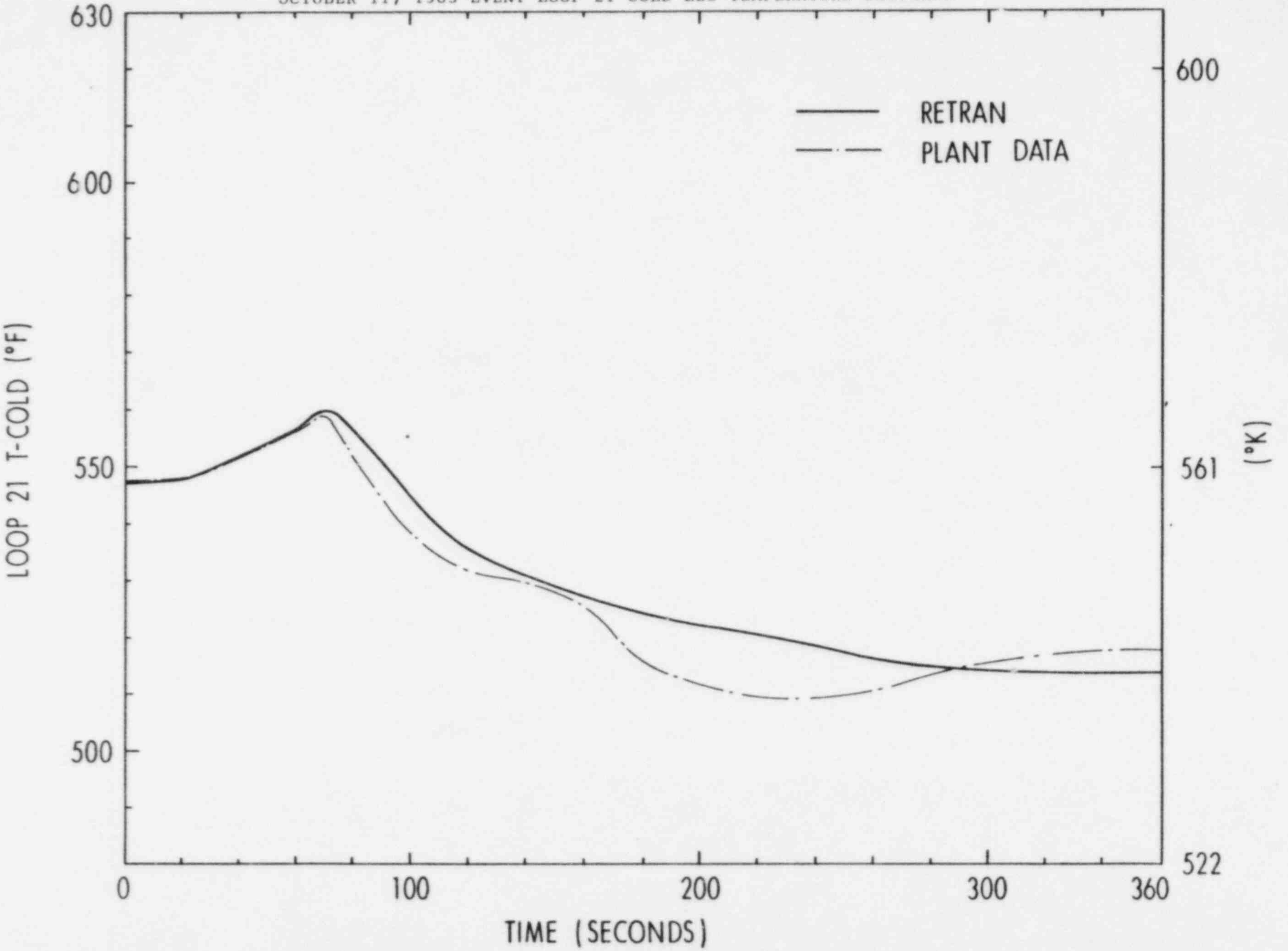


FIGURE 4.1-7
OCTOBER 11, 1983 EVENT STEAM GENERATOR 22 SECONDARY SIDE PRESSURE RESPONSE

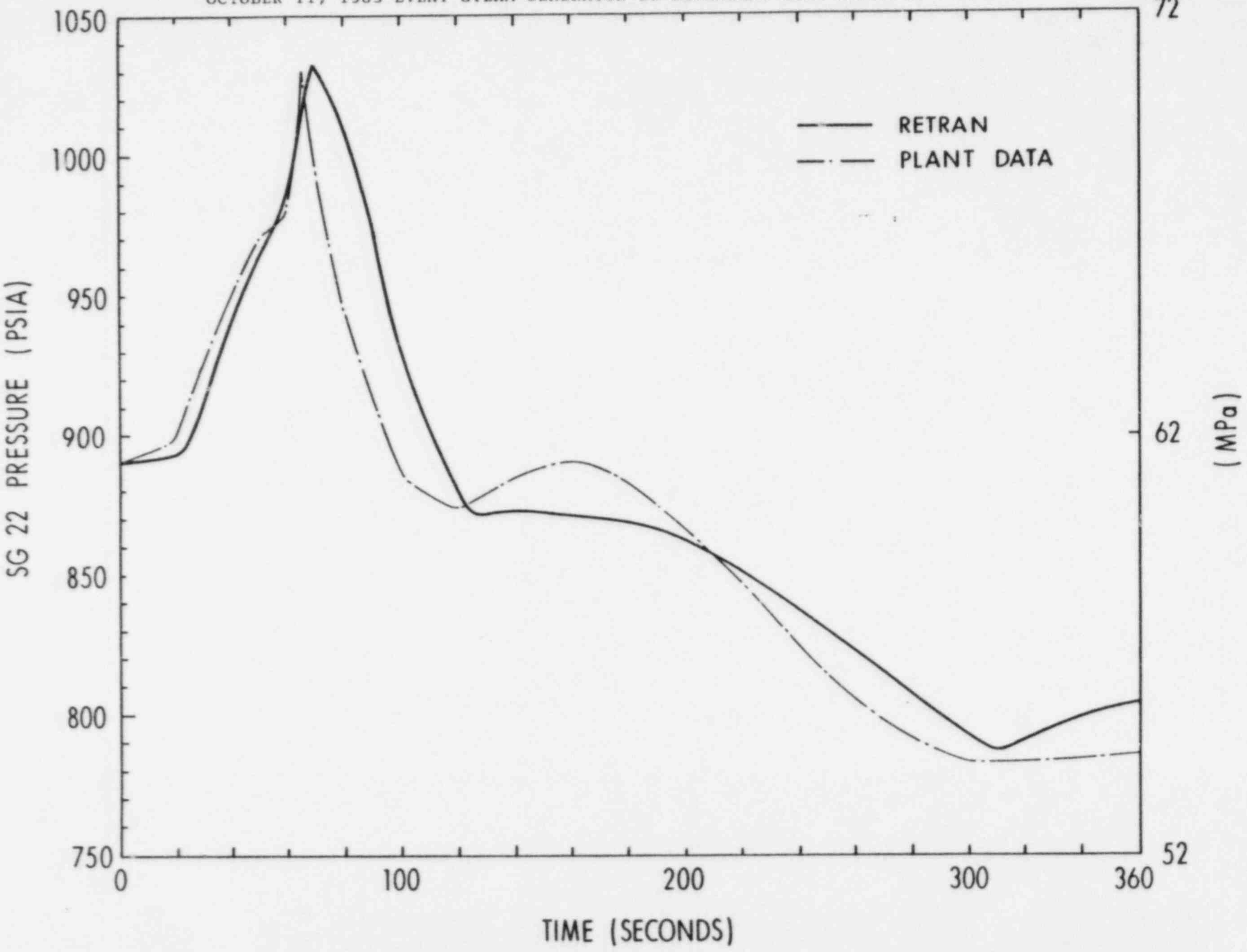
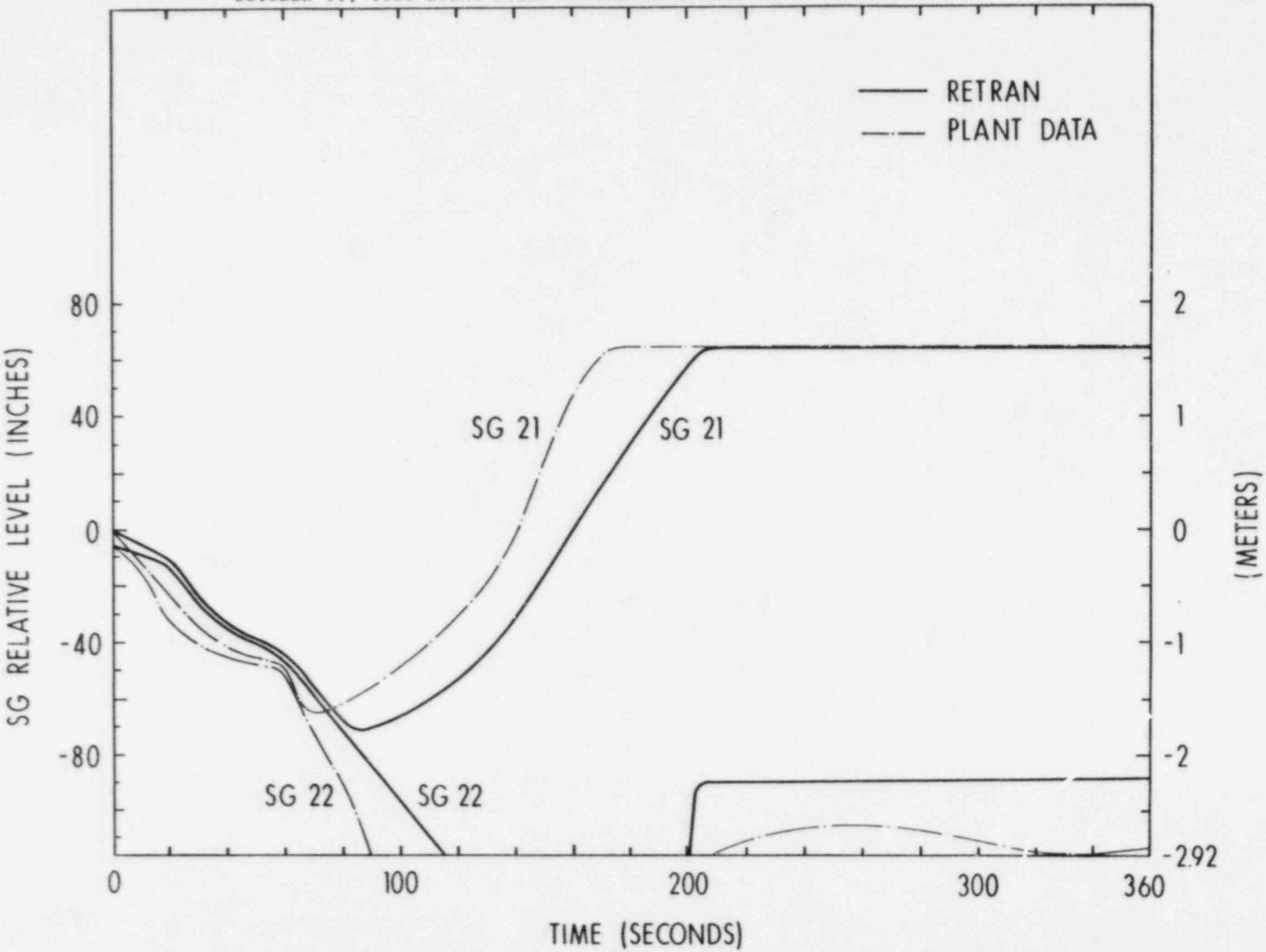


FIGURE 4.1-8
OCTOBER 11, 1983 EVENT STEAM GENERATOR NARROW RANGE LEVEL RESPONSE



4.2 Calvert Cliffs Unit 1 Startup Test - Four Pump Coastdown From 20% Power

4.2.1 Description of Event

This startup test was used to obtain reactor coolant system (RCS) flow data during a four pump coastdown at 20% of full power (30). The transient was initiated using the trip pushbuttons on the reactor protective system panels. Measurements were made of the total RCS flow for the first 55 seconds after reactor trip.

4.2.2 Event Specific RETRAN Modeling Changes

Both the one loop and four loop RETRAN models described in Sections 3.2 and 3.4 were used to simulate this four pump coastdown transient. The RETRAN model was initialized at 20% of 2560 MWt and with key thermal-hydraulic parameters at the measured plant values delineated in Table 4.2-1. These key parameters are: hot and cold leg temperature, pressurizer pressure and level, steam generator pressure and level.

4.2.3 Comparison of RETRAN Results to Plant Data

Figure 4.2-1 presents a comparison of RETRAN calculated total RCS flow with measured plant data. Both the one loop and four loop RETRAN model predicted flow coastdown closely following the measured plant data for the first 55 seconds after RCP trip. RETRAN calculated flow was within about 3% of measured flow for the first 35 seconds of the transient. This comparison is considered to be excellent because the flow measurement

accuracy is 2% and a time lag of up to 1.5 seconds is associated with this parameter.

Table 4.2-2 presents a comparison of RETRAN one loop model and four loop model with Calvert Cliffs measured total RCS flow for this transient. Although both RETRAN models calculate a similar flow coastdown, the four loop model calculated flow is closer to measured flow. This closer agreement is attributed to the finer nodalization and more accurate representation of RCS flow paths, pressure loss and temporal effects inherent in the four loop model.

This table also presents data from the CESEC Topical Report (31). CESEC is the Combustion Engineering (CE) system transient thermal-hydraulic code which CE uses for reload safety analysis licensing calculations and has been reviewed and approved by the USNRC. As part of the CESEC qualification, a comparison with measured four pump coastdown at hot zero power is presented in the CESEC topical report. The plant data and CESEC results are also presented in Table 4.2-2. The measured flow data is not identical to the Calvert Cliffs flow coastdown because it was taken from another CE plant which would be expected to have somewhat different characteristics (i.e., impeller trim, RCP motor performance, loop pressure drop, etc.) Also, this test was run at hot zero power which implies a slightly lower RCS temperature and higher fluid density which affects the flow coastdown. However, the relevant conclusion from this CESEC-plant data comparison is that CESEC predicts flow to within the same 3% as RETRAN does for the Calvert Cliffs data. This substantiates the acceptability of the RETRAN calculation of four pump coastdown when compared to measured plant data.

4.2.4 Summary of Results

A calculation of four RCP coastdown was performed with the one loop and four loop RETRAN models. Both models predicted a flow coastdown similar to plant data, but the four loop model resulted in a closer comparison. Both models calculated four pump coastdowns within instrumentation uncertainty and other comparisons performed by the CE CESEC licensing computer code.

Table 4.2-1

Four Pump Coastdown from 20% Power Initial Conditions

Power	512 MWt (20% of 2560 MWt)
Pressurizer Pressure	2242 psia
Pressurizer Level	150 inches
Hot Leg Temperature	544°F
Cold Leg Temperature	534°F
Steam Generator Pressure	855 psig
Steam Generator Level	0 inch Narrow Range

Table 4.2-2

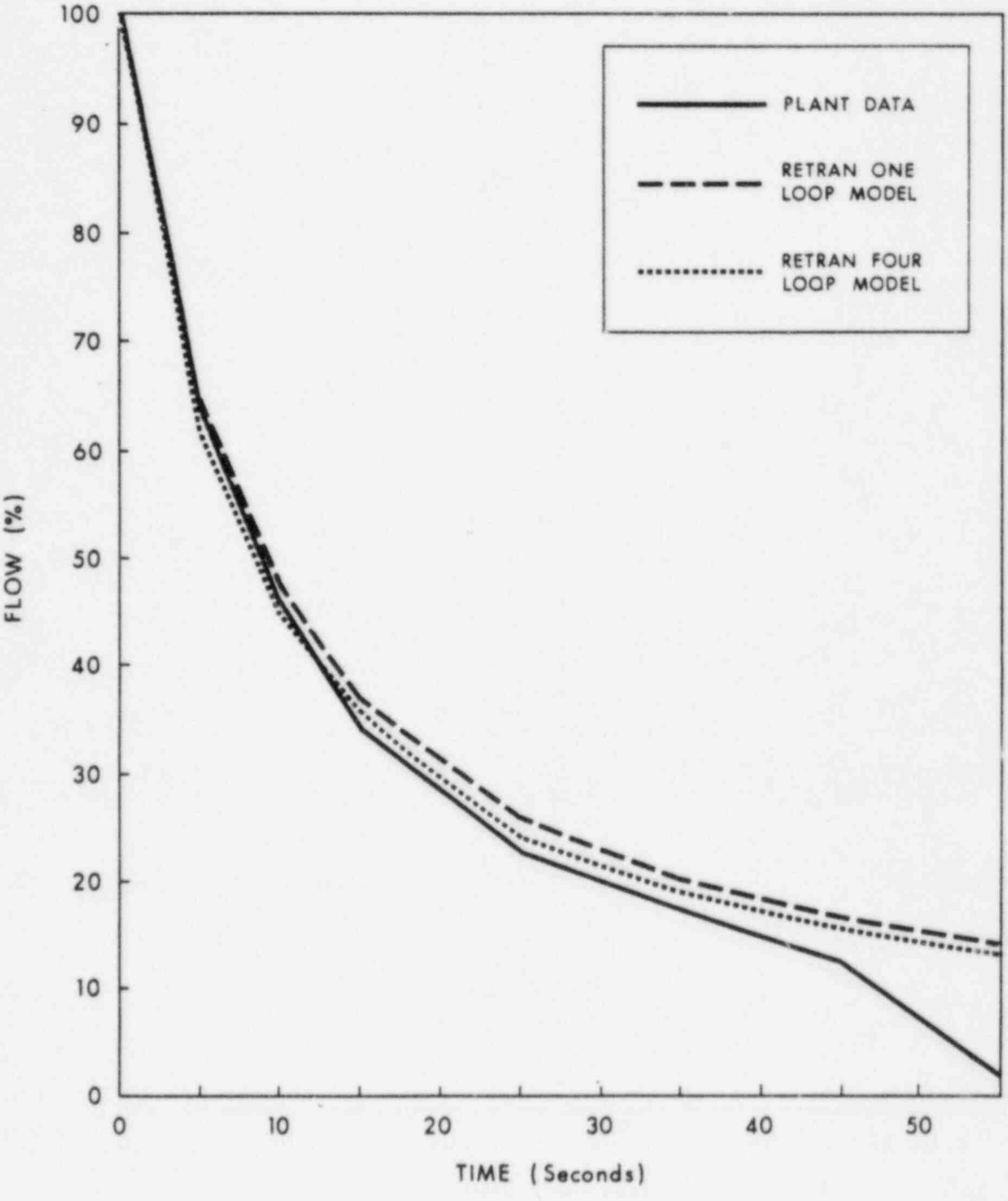
Comparison of RETRAN One Loop and Four Loop Model Calculated

Four Pump Coastdown Total RCS Flow with Plant Data

	Fraction of Total RCS Flow				
		RETRAN		CESEC (Hot Zero Power)	
Time After RCP Trip (sec)	Measured	One Loop Model	Four Loop Model	Data	CESEC
0	1.00	1.00	1.00	1.00	1.00
5	0.64	0.646	0.617	.65	.66
10	0.46	0.472	0.448	.50	.49
15	0.34	0.371	0.353	.39	.36
25	0.23	0.260	0.243	.28	.26
35	0.175	0.201	0.189	.21	.20
45	0.125	0.164	0.155	.16	.16
55	0.02	0.140	0.132	.13	.13

FIGURE 4.2-1

FOUR PUMP COASTDOWN TOTAL REACTOR COOLANT SYSTEM FLOW RESPONSE



4.3 Calvert Cliffs Unit 1 Startup Test Reactor Coolant Pump Operation Combination Flow Tests at Hot Zero Power

4.3.1 Description of Plant Event

A series of fifteen Reactor Coolant System (RCS) flow tests were conducted to determine total and individual Reactor Coolant Pump (RCP) flow rates with all possible combinations of RCPs running (30). The tests were conducted with the reactor at hot standby (i.e., zero core power but RCS at 2250 psia and 532^{o}F). The results of these tests were measured RCS flow at each of the four RCPs and total RCS flow. The fifteen RCS flow combination tests consisted of: one test with all four RCPs running, four tests with three different RCPs running, six tests with two different RCPs running (two of which were RCPs in the same loop; the remaining four being RCPs in opposite loops), and four tests with one RCP running. The measured plant data for each test, normalized to average RCP flow for four RCPs running, is presented in Table 4.3-1.

4.3.2 Event Specific RETRAN Modeling Changes

The RETRAN four loop model described in Section 3.4 was used for this analysis. Since the tests were conducted at hot standby, the steam generator secondary side and associated steam line volumes and junctions were replaced by a non-conducting heat exchanger which removed the nominal 10 Mwt core power. This small, but non-zero power was required to allow the use of the existing four loop primary side model without further modifications and has no effect on the results of this analysis.

During early RETRAN calculations of pump flow combinations, analysis of RETRAN results showed a deficiency in accurately modelling the flow paths from operating RCPs through shut down RCP loops. With one or more RCPs shut down some flow from operating RCPs will short circuit the core. This bypass flows circumferentially around the downcomer and out the cold leg nozzle(s) in the reverse direction through the shut down RCP(s) to the steam generator outlet plenum. The initial four loop RETRAN model was modified by adding a downcomer cross flow junction between the two identical downcomer volumes. Prior to this change, flow from one loop cold leg to the other loop cold let must first traverse the entire downcomer and then cross over to the other loop's lower plenum volume. The addition of this downcomer cross flow junction more accurately modelled the actual flow path available for reduced RCP operation configurations.

Another important RETRAN modelling change was the input of an explicit locked rotor reverse flow pressure loss coefficient. The RCPs at Calvert Cliffs are equipped with an "anti-reverse" rotation device on the motor that prevents reverse rotation of the impeller after pump shut down. Thus, after coastdown, the rotor will remain stationery and not rotate in the reverse direction even under the force of coolant flowing backwards. Calvert Cliffs RCP-specific locked rotor reverse flow pressure loss coefficients were obtained from the NSSS vendor, Combustion Engineering (32). This locked rotor reverse flow coefficient is, as expected, much larger than the loss coefficient previously used by RETRAN which was based solely on normal flow friction losses.

4.3.3 Comparison of RETRAN Results to Plant Data

RETRAN calculated results are compared to measured plant data for RCP operation combinations in Table 4.3-2. Five combinations of RCP operation are listed (4, 3, 2 same loop, 2 opposite loop, and 1 RCP) along with their corresponding cases from Table 4.4-1. A mean flow and associated standard deviation (also in terms of percent difference) is listed for each RCP flow path. Along with each measured normalized flow, the RETRAN calculated flows for these cases are listed with their respective deviation from plant data.

The standard deviation was calculated assuming a normal distribution of plant data. For a normal distribution, the mean value can be adjusted to encompass most of the data distribution by including one or more standard deviations as delineated below:

Number of Standard Deviations	Normal Distribution Data Inclusion
± 1	68%
± 2	95.5%
± 3	99.7%

The 3.2% standard deviation in RCP flow for four pumps running demonstrates the variation of individual pump flow characteristics and individual pipe loop hydraulic resistance. RETRAN pump and piping models are input with identical parameters for all four cold legs and thus show no flow asymmetry.

The case with 3 RCPs running resulted in 130.4% of normal RCP flow in the operating RCP on the loop with one RCP shut down. The shut down RCP

reverse flow is 36.9% of normal operating flow. The loop with two operating RCPs had 105.6% of normal flow in each cold leg. The three RETRAN cases, (initial four loop model, downcomer crossflow junction added, and realistic operating RCP locked rotor reversed flow loss factor) are compared to the three RCP case plant data. The final RETRAN model (includes downcomer crossflow and realistic RCP reverse flow loss factor) predicts individual and total RCP flows within two standard deviations of measured plant flow.

In a similar manner, two RCP and one RCP cases are presented and compared in Table 4.3-2. For almost all the RCP total flow comparisons, RETRAN calculated values are within two standard deviations of the measured plant data. The two exceptions are total RCS flow for two RCPs on in opposite loop and reverse flow for two RCPs on in the same loop. RETRAN predicted total RCS flow for two RCPs on in the same loops differs by 15.1% from mean measured plant data which has a 5.7% standard deviation. This RETRAN flow is within three standard deviations. This difference is not considered excessive in light of the relatively large standard deviation of measured data and the fact that only two cases were run for this data. In the comparison of total RCS flow for two RCPs running in opposite loops, plant measured data compilation showed only a 0.8% standard deviation whereas RETRAN calculated flow differs by 4.1%. Although, this number exceeds three standard deviations, its magnitude is well within the difference for other flow data point comparisons.

Figure 4.3-1 illustrates the normal and abnormal primary coolant system flow paths for Calvert Cliffs. Three abnormal flow paths were evident from the RETRAN analyses of less than four RCP operating configurations. These abnormal flow paths are:

1. Crossflow between the two downcomer volumes which receive cold leg flow from separate loops.
2. Crossflow in the steam generator outlet plenum between the two RCP suction cold leg pipe lines.
3. Backflow through the RCP, steam generator tubes, and hot leg to the reactor vessel.

Table 4.3-3 presents operating pump cases where these abnormal flow paths exist. Downcomer crossflow occurs for those pump combinations (one RCP on, two RCPs in the same loop on, or three RCPs on) where both loops do not have the same number of pumps operating thus creating a pressure differential between the two downcomer volumes. Reverse flow through cold legs, steam generator tubes and hot leg back to the reactor vessel occurs when neither pump on one loop is operating. This provides a second hydraulic loop for the operating pump(s). Steam generator outlet plenum crossflow exists when only one RCP in a loop is operating thus forcing reverse flow through that loop's other cold leg and back to the operating RCP's suction side.

4.3.4 RETRAN Sensitivity Study

The RETRAN sensitivity study performed for this analysis involved the variations of downcomer crossflow and RCP reverse flow loss coefficients as well as evaluating the importance of including downcomer crossflow and a realistic RCP reverse flow loss coefficient.

As shown in Table 4.3-2, inclusion of a downcomer crossflow junction significantly affected the flow split and total flow for asymmetric RCP operating configurations (i.e., all cases but four RCP and two RCP in opposite

loops). The downcomer crossflow junction generally increased back flow through shut down RCPs and reduced total RCS flow. It was also found that varying the downcomer flow loss coefficient over the range of .02 to 13.2 changed flows by less than 1%. This insensitivity to downcomer crossflow factor is due to the large flow area of this junction which resulted in low fluid velocities.

In Table 4.3-2, a realistic RCP reverse flow loss coefficient resulted in reduced reverse flow through stopped RCPs. Inclusion of this realistic higher loss coefficient dramatically reduced the difference between RETRAN predicted back flow and plant data. The back flow difference dropped from: 19% to 1.9% for three RCPs on, 32% to 15.5% for two RCPs on in the same loop, 28.5% to 1.6% for two RCP on in opposite loops, and 39.4% to 3.2%/34.2% to 10.1% for one RCP on. The realistic RCP locked rotor reverse flow loss coefficient provided by CE was actually a range of values which varied by about 14% for RCPs. Using values at both ends of this range resulted in generally small (less than 1%) changes in RCP flow.

4.3.5 Summary of Results

A RETRAN simulation of combination of operating reactor coolant pumps (RCP) was performed to compare to measured plant individual RCP flow and total reactor coolant system (RCS) flow. Tests with four, three, two same loop, two opposite loop, and one operating RCP were conducted at Calvert Cliffs. The measured test individual RCP and total RCS flow data was used to calculate a mean and standard deviation for each flow path.

A RETRAN four loop model of Calvert Cliffs was used to evaluate operating pump combination flow distribution. This model was modified by

adding a downcomer crossflow junction and accurate locked rotor reverse flow pressure loss coefficient which greatly improved RETRAN results as compared to plant data. RETRAN calculated flows were within two standard deviations (95.5% of data) of measured plant flows for 12 of the 14 flow paths in the four pump operation combinations. One of the remaining two flows differed by only 4.1%. The other flow that differed by more than two standard deviations was for a test in which only two data points were available with an inherent reduction in statistical significance. The absolute magnitude of the standard deviation for this particular flow was also much larger than for most of the other flow data. RETRAN predicted flow splits and total RCS flow was generally within about 4% of mean measured flow. Considering the fact that the standard deviation on individual RCP flow for the normal four pump configuration was 3.2%, this deviation is acceptable for reduced RCP operation conditions.

Along with the aforementioned RETRAN model enhancement, this analysis provided insights into the three abnormal RCS flow paths which occur when one or more RCPs are shutdown. These flow paths are: downcomer crossflow, steam generator outlet plenum cross flow and complete reverse flow through an RCS loop.

Table 4.3-1

Reactor Coolant Pump Combination Flow Test Data at Hot Standby*

		RCP Relative Flow				Relative Total
Case	RCPs*** Running	11A	11B	12A	12B	RCS Flow
1	11A, 11B, 12A, 12B**	.960	.987	1.030	1.022	1.000
2	11A, 11B, 12A	1.103	1.049	1.338	-.376	0.756
3	11A, 11B, 12B	1.019	1.041	-.368	1.319	0.753
4	11A, 12A, 12B	1.278	-.368	1.087	1.078	0.768
5	11B, 12A, 12B	-.363	1.282	1.087	1.070	0.769
6	11A, 11B	1.060	1.090	-.166	-.151	0.459
7	12A, 12B	-.151	-.168	1.123	1.112	0.479
8	11A, 12A	1.320	-.310	1.386	-.324	0.518
9	11A, 12B	1.338	-.312	-.305	1.368	0.521
10	11B, 12A	-.336	1.336	1.395	-.330	0.516
11	11B, 12B	-.327	1.323	-.310	1.359	0.511
12	11A 1.356	-.275	-.069	-.067	0.236	
13	11B -.292	1.359	-.074	-.057	0.234	
14	12A -.087	-.099	1.427	-.294	0.237	
15	12B -.087	-.089	-.276	1.392	0.235	

* RCS Temperature and pressure of 532°F and 2250 psia

** Flow is normalized to 25% of total flow for four RCPs running (Case 1)

*** RCPs 11A and 11B are both on Loop 11 while RCPs 12A and 12B are both on Loop 12

Table 4.3-2

Comparison of Measured Plant Data with RETRAN Calculated RCP Flow Split for RCP Operating Combinations

	*PLANT DATA	RETRAN CALCULATED RESULTS Flow (% Difference from plant data)								
Number of RCPs Running	Cases From Table 4.3-1	Mean Flow	Standard Deviation	(% difference)	Initial Model		Downcomer Crossflow Junction		Downcomer Crossflow and RCP Junction Reverse Flow Realistic Loss Factor	
4	1	1.000	.032	(3.2)	1.000		1.000		1.000	
3	2,3,4,5	1.304	.029	(2.2)	1.400	(7.4)	1.381	(5.9)	1.362	(4.4)
		-.396	.005	(1.4)	-.439	(19)	-.466	(26.3)	-.362	(1.9)
		1.056	.030	(2.8)	1.026	(2.8)	1.053	(0.3)	1.049	(0.7)
	Total RCS Flow	.761	.008	(1.1)	.753	(1.1)	.756	(0.7)	.775	(1.8)
2	6,7 (same loop)	1.096	.028	(2.6)	1.048	(4.4)	1.095	(0.1)	1.090	(0.5)
		-.159	.009	(5.7)	-.108	(32)	-.173	(8.8)	-.135	(15.1)
	Total RCS Flow	.469	.014	(3.0)	.470	(0.2)	.461	(1.7)	.477	(1.7)
2	8,9,10,11	1.353	.029	(2.1)	1.418	(4.8)	1.418	(4.8)	1.401	(3.5)
	(opposite loops)	-.319	.011	(3.4)	-.410	(28.5)	-.410	(28.5)	-.324	(1.6)
	Total RCS Flow	.517	.004	(0.8)	.504	(2.5)	.504	(2.5)	.538	(4.1)
1	12,13,14,15	1.384	.033	(2.4)	1.429	(3.3)	1.441	(4.1)	1.427	(3.1)
		-.284	.010	(3.5)	-.396	(39.4)	-.371	(30.6)	-.293	(3.2)
		-.079	.014	(17.7)	-.052	(34.2)	-.084	(6.3)	-.071	(10.1)
	Total RCS Flow	.236	.001	(0.4)	.234	(0.8)	.225	(4.7)	.248	(5.1)

* Instrumentation Accuracy $\pm 2\%$

Table 4.3-3

Reduced Reactor Coolant Pump Operation Abnormal Flow Paths

Number of Operating Reactor Coolant Pumps	Significant Downcomer Crossflow Between Loops	Outlet Plenum Crossflow (cold leg to cold leg flow)	Tube Side Reverse Flow
1	YES	YES	YES
2 (same loop)	YES	NO	YES
2 (opposite loop)	NO	YES	NO
3	YES	YES	NO
4	NO	NO	NO

Figure 4.3-1

CALVERT CLIFFS PRIMARY COOLANT SYSTEM FLOW PATHS

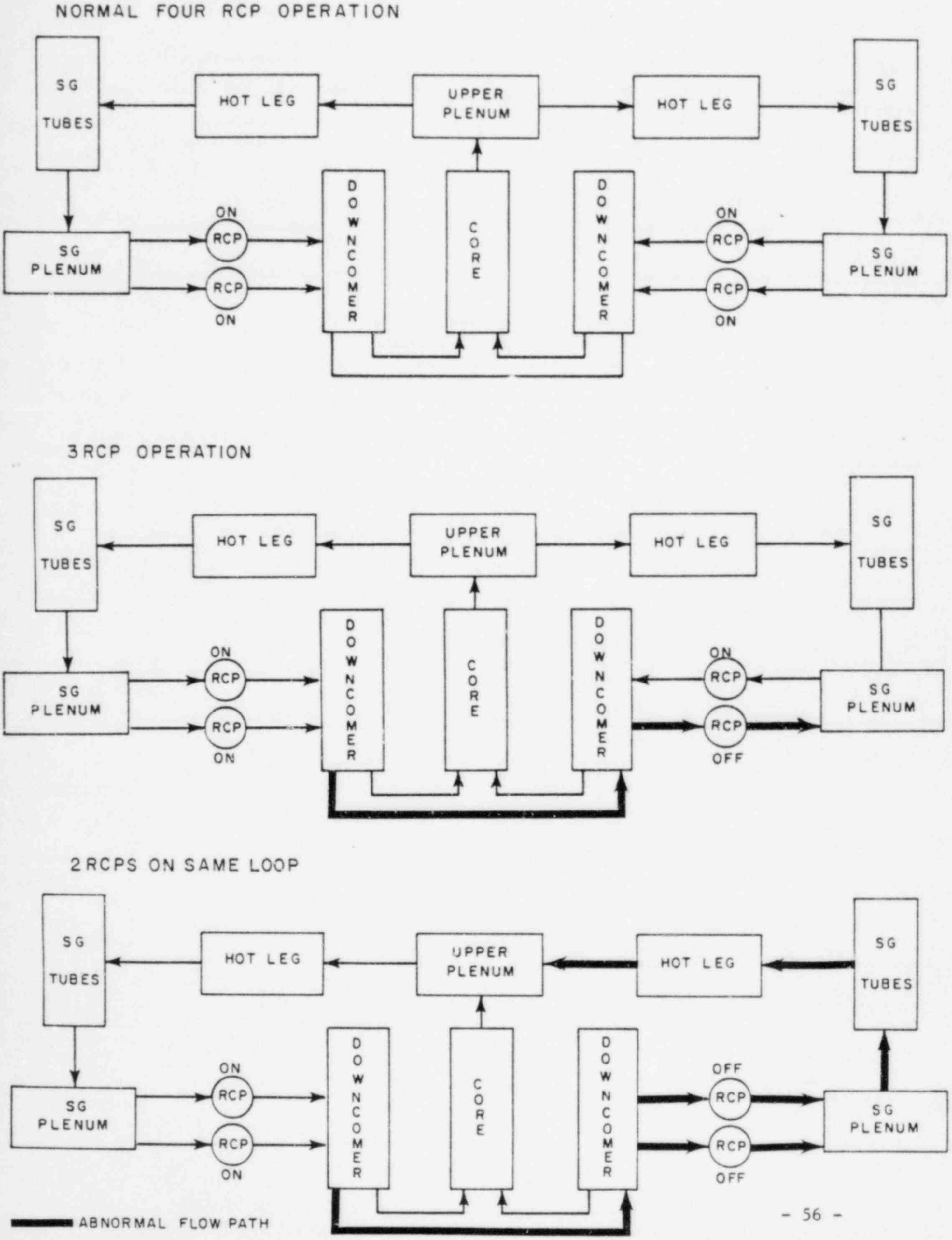
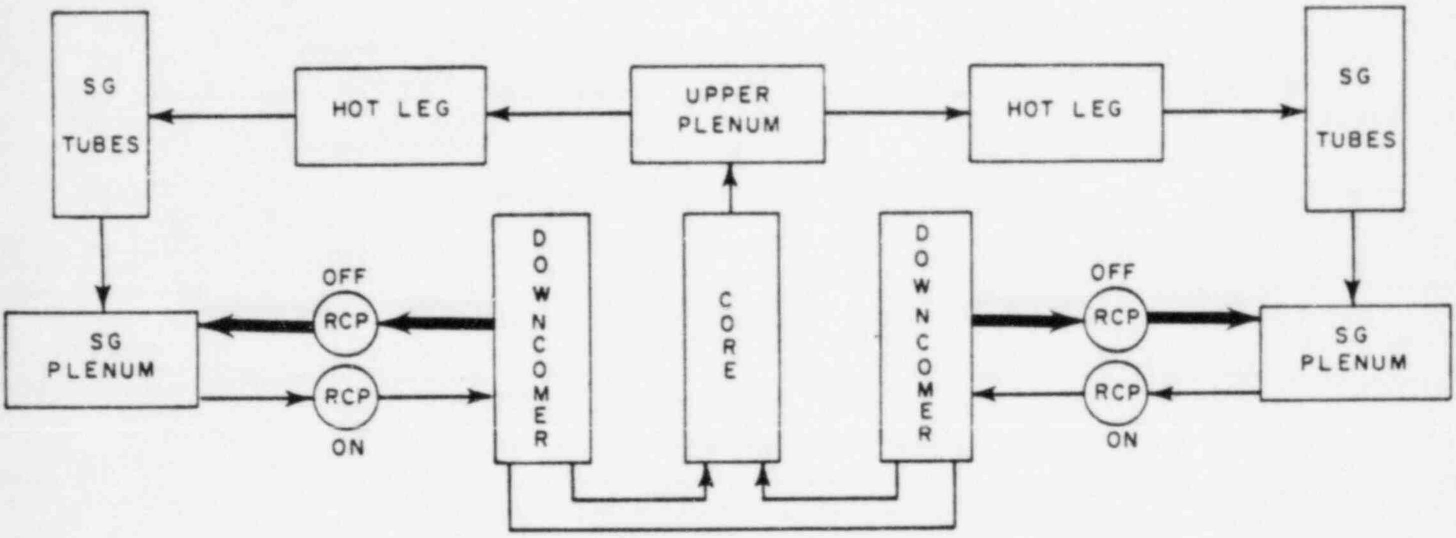
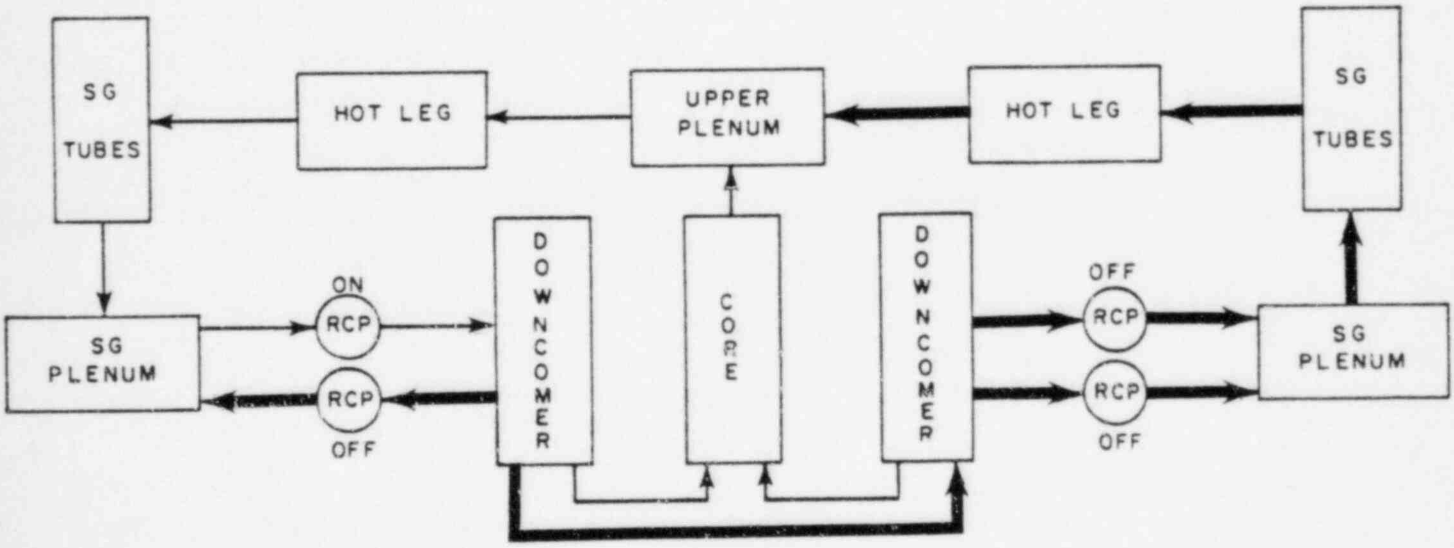


Figure 4.3-1 (Cont'd.)

2 RCPS ON OPPOSITE LOOP



I RCP



4.4 Calvert Cliffs Unit 1 Startup Test - One Pump Coastdown From 80% Power

4.4.1 Description of Event

The objective of this startup test was to measure the plant response to a partial loss of Reactor Coolant System (RCS) flow at 80% full power (30). This transient was initiated by securing 12A Reactor Coolant Pump (RCP) with the reactor operating at 80% of its full power which was 2560 Mwt for the original Operating License (OL). This test provided single RCP coastdown data and validated the low RCS flow trip from 80% power with one RCP secured.

4.4.2 Event Specific RETRAN Modeling Changes

The Calvert Cliffs RETRAN model used for this analysis is the four loop model presented in Figure 3.4-1 and discussed in Section 3.4. The two loop model described in Sections 3.3 and 4.1.2 was expanded to a four loop model in which the two cold leg loops were converted into four individual cold legs and reactor coolant pumps. This model explicitly simulates the exact primary coolant system geometry which was required to analyze a single pump coastdown for the Calvert Cliffs design. Model enhancements (i.e., downcomer crossflow and realistic RCP reverse flow pressure loss coefficient) from Section 4.3 were also used.

The other major event specific RETRAN model change for this analysis was to initialize the model at 80% of 2560 Mwt with all its associated thermal-hydraulic conditions. Along with core power, total RCS flow was adjusted to properly match the measured initial hot and cold leg temperatures. In addition pressurizer pressure and level and steam generator secondary side pressure were

changed to reflect initial conditions at this power level. Finally, the RCS level control system was adjusted to maintain initial level.

4.4.3 Comparison of RETRAN Results to Plant Data

Figures 4.4-1 to 4.4-5 present a comparison of RETRAN calculated key thermal-hydraulic parameters with plant data. In Figure 4.4-1, RETRAN calculated total RCS flow conservatively underpredicts plant data by 1% to 4% over the 60 second period. The difference is bounded by the flow instrumentation accuracy of $\pm 2\%$ and time lag of up to 1.5 seconds. Another flow comparison of interest for this transient is the time of reactor trip on low (95% of full) total RCS flow. RETRAN predicts this trip at 2.37 seconds after RCP trip which compares closely to the plant measured 2.2 seconds.

Figure 4.4-2 presents a comparison of hot and cold leg temperatures between plant measurements and RETRAN calculated results. RETRAN results follow the same general trend as plant data and slightly overpredicts measured temperatures. This difference is due to the lower RCS flow coastdown and the unknown exact timing of Turbine Stop Valve (TSV) closure/Turbine Bypass Valve (TBV) opening following reactor trip. The small cold leg temperature rise reflects the temporary core power-steam flow mismatch that typically follows a reactor trip.

Figures 4.4-3 and 4.4-4 compare measured pressurizer pressure and level with that calculated by RETRAN. These parameters slowly decrease in this transient in response to the primary coolant system cooldown. This same trend is calculated by RETRAN which predicts a slightly higher pressure and level with the higher RETRAN calculated loop temperatures. It should be noted that no information on the status or operation of the pressurizer pressure and level

control systems (i.e., spray, heaters, charging, and letdown) was available for this event. The RETRAN model assumed normal operation of these systems normalized to the initial values of pressure and level.

The comparison of Steam Generator #11 secondary side pressure with RETRAN in Figure 4.4-5 shows close agreement. Both show the initial post-trip characteristic core power-steam flow mismatch induced pressure rise. The subsequent pressure drop to about 875 psig reflects the Turbine Bypass Control System (TBCS) which maintains a pressure of 875 psig with the TBVs after a trip. RETRAN calculated peak pressure and later decrease compares well with measured plant data.

4.4.4 RETRAN Sensitivity Study

A RETRAN sensitivity study was performed for this transient in which RCP moment of inertia and rated torque were varied to assess their effect on the RCS flow coastdown. Moment of inertia and torque were each varied by ±10% in four separate runs. The RCS flow coastdown for each sensitivity case as well as the base case and plant data is presented in Table 4.4-1. By either increasing pump moment of inertia by 10% or decreasing pump torque by 10%, the temporal flow rate increased closer to measured plant data. Conversely, lower inertia or higher torque resulted in lower coastdown flow rates.

In addition, Table 4.4-1 presents coastdown flow data for the base case modified by adding the downcomer cross flow path and RCP reverse flow loss factor discussed in Section 4.3. These model changes have no significant effect on flow coastdown until 18 seconds into the transient, after 18 seconds, the new model calculated RCS flow is 1% to 3% higher than the initial model.

4.4.5 Summary of Results

A RETRAN four loop model of Calvert Cliffs was developed to simulate a single pump coastdown transient from 80% power which was conducted as part of the original startup tests. Several key thermal-hydraulic parameters were compared between RETRAN and measured plant data for sixty seconds following the RCP trip. A comparison of total RCS flow coastdown and time of low flow reactor trip showed excellent agreement. RETRAN underpredicted flow by less than 4% which is both conservative and within instrumentation measurement accuracy/time lag. A comparison of hot and cold leg temperatures, pressurizer pressure and level, and steam generator secondary side pressure all showed good agreement both in peak values (if any) and trends. Finally, sensitivity study on RCP inertia and torque indicated that these inputs can significantly affect the flow coastdown. Larger values of inertia have the same effect as smaller values of torque. Incorporating downcomer crossflow and RCP reverse flow model enhancements developed from the analysis described in Section 4.3 had a small (3%) effect on the total RCS flow coastdown in the first sixty seconds.

Table 4.4-1

Calvert Cliffs One RCP Trip at 80% Power Sensitivity Study Results

RETRAN Normalized Total Reactor Coolant System Flow

Time after Trip of RCP 12A	Plant Data	Base Case	Inertia +10%	Inertia -10%	Torque +10%	Torque -10%	Base Case with Downcomer Cross Flow and RCP Reverse Flow Loss Coefficient
4 seconds	0.925	0.914	0.923	0.905	0.906	0.923	0.913
6.5 seconds	0.900	0.867	0.867	0.858	0.859	0.877	0.865
8.5 seconds	0.875	0.843	0.851	0.835	0.836	0.851	0.842
10.5 seconds	0.850	0.826	0.833	0.817	0.818	0.834	0.825
12.5 seconds	0.825	0.810	0.817	0.802	0.803	0.818	0.812
18 seconds	0.800	0.780	0.787	0.772	0.772	0.787	0.791
29 seconds	0.775	0.755	0.758	0.753	0.753	0.758	0.778
33-60 seconds	0.765	0.760	0.760	0.760	0.760	0.760	0.778
Time of low RCS flow trip (seconds)	2.2	2.37	2.59	2.17	2.18	2.61	2.37

FIGURE 4.4-1

ONE PUMP COASTDOWN TOTAL REACTOR COOLANT SYSTEM FLOW RESPONSE

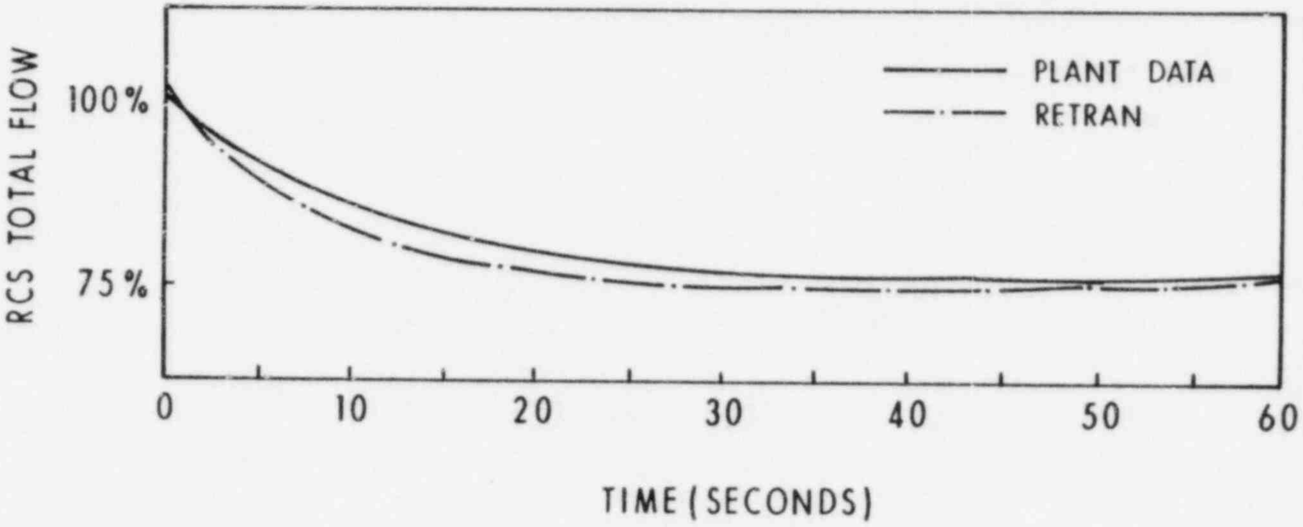


FIGURE 4.4-2
ONE PUMP COASTDOWN HOT AND COLD LEG TEMPERATURE RESPONSE

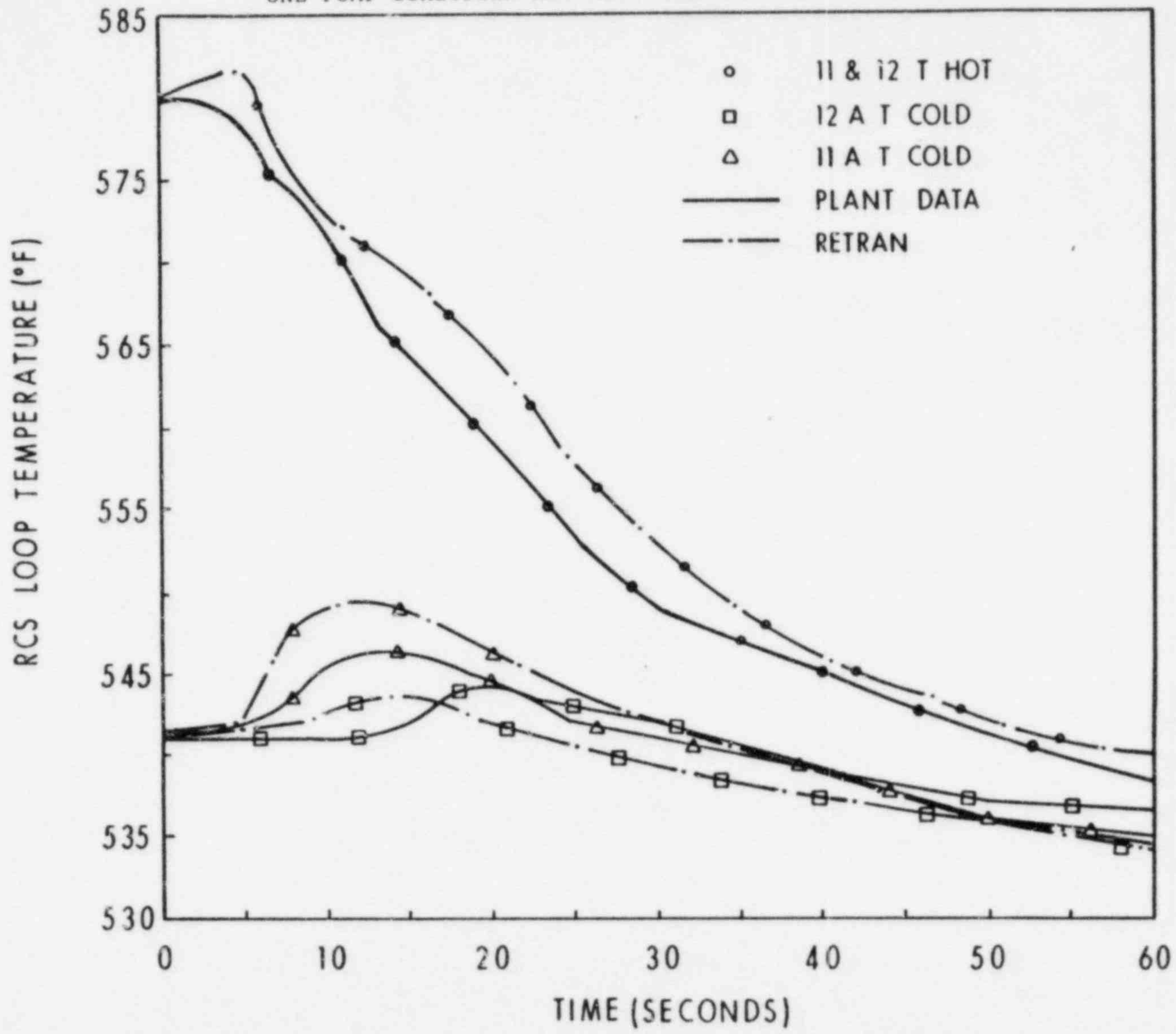


FIGURE 4.4-3

ONE PUMP COASTDOWN PRESSURIZER PRESSURE RESPONSE

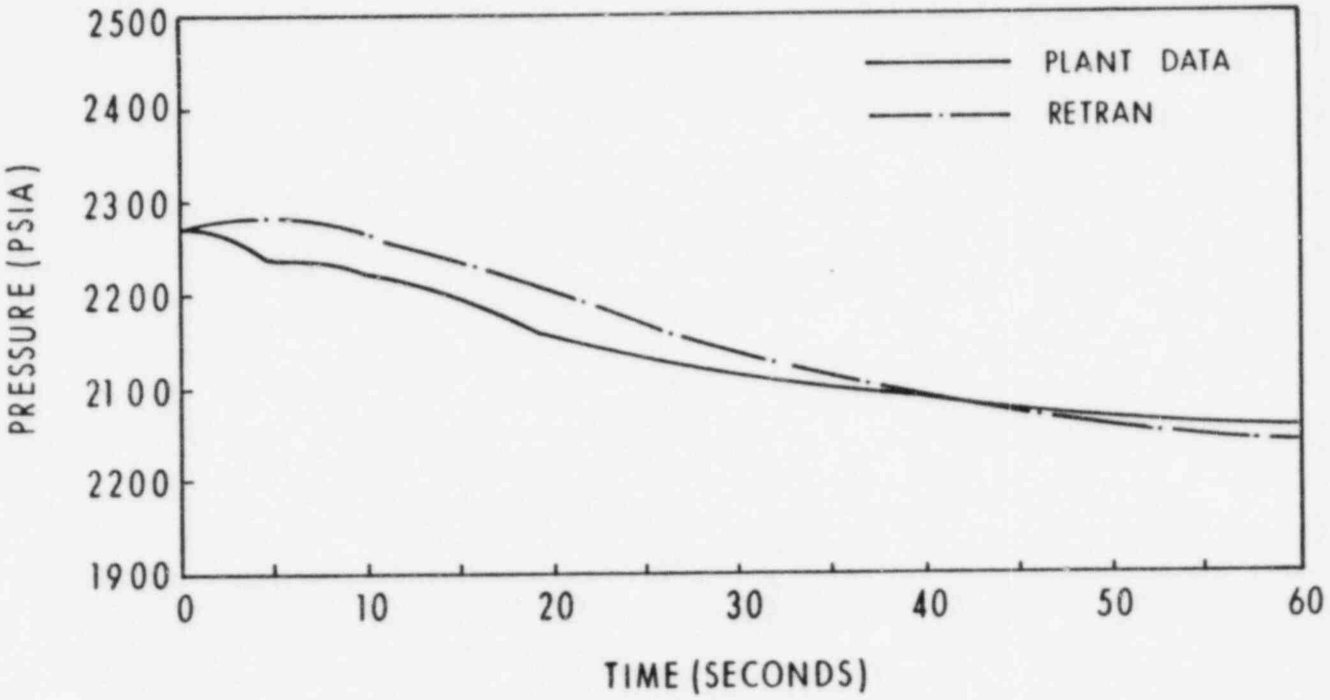


FIGURE 4.4-4
ONE PUMP COASTDOWN PRESSURIZER LEVEL RESPONSE

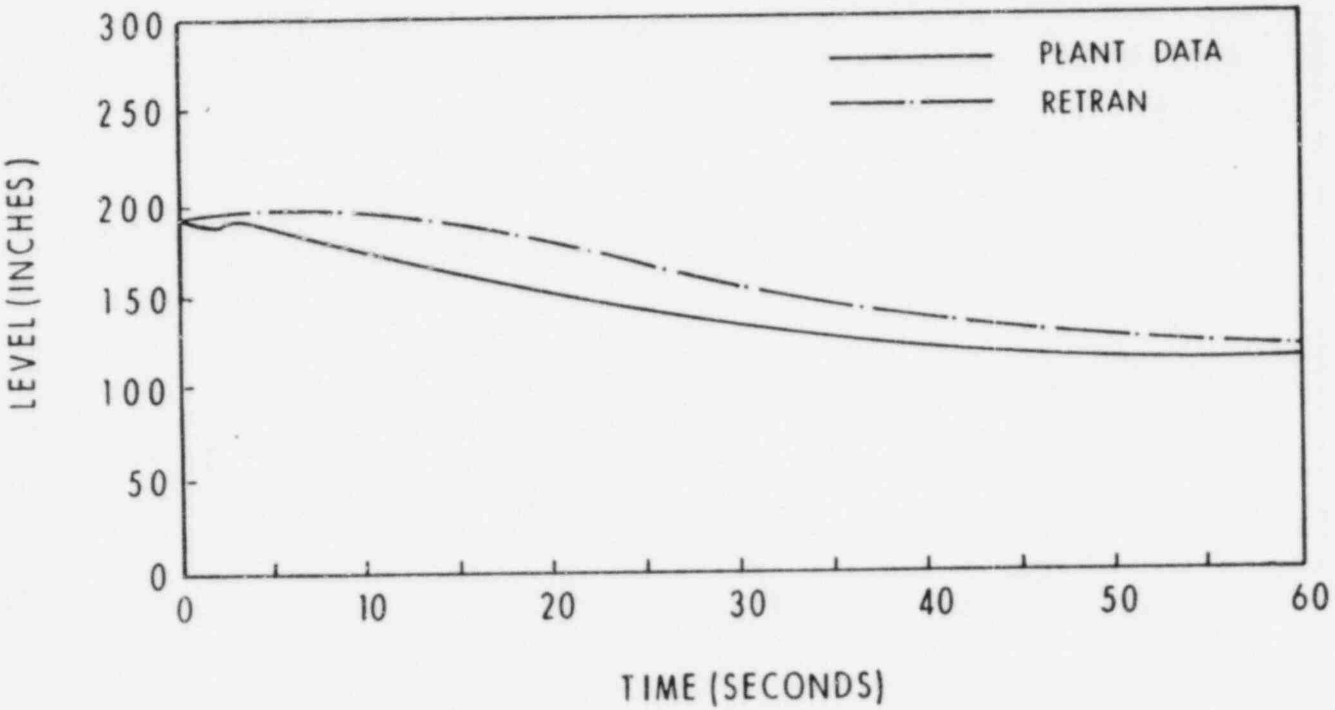
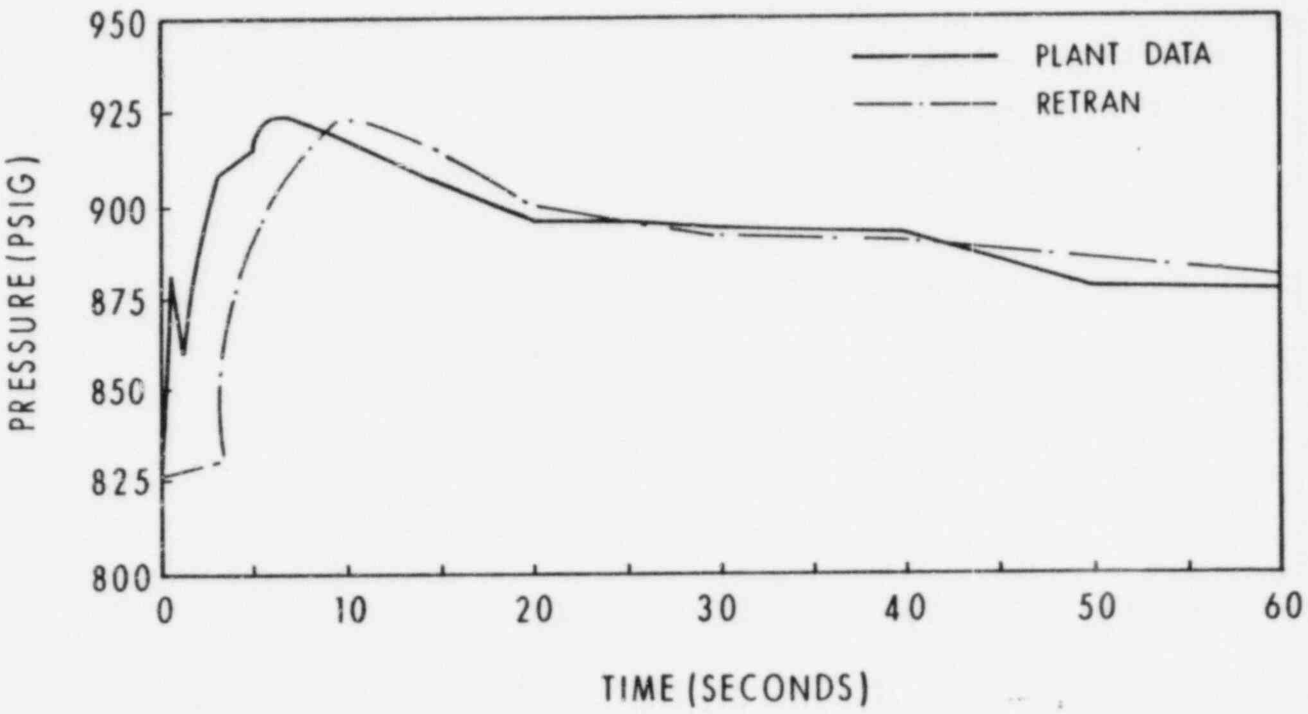


FIGURE 4.4-5

ONE PUMP COASTDOWN STEAM GENERATOR SECONDARY SIDE PRESSURE RESPONSE



4.5 Calvert Cliffs Unit 1 Startup Test - Total Loss of Flow/Natural Circulation Test From 40% Power

4.5.1 Description of Plant Event

This test was conducted to monitor plant response to a total loss of flow from 40% of full power and to determine the power-to-flow ratio during natural circulation (30). The test was conducted by tripping the breaker supplying power to all four reactor coolant pumps. Key plant parameters were recorded for the first sixty seconds of this transient.

Natural circulation was verified at five minutes after the trip based on reactor coolant system hot and cold leg temperature variations. The principal indication of natural circulation was the decreasing hot leg temperature at this time.

The power-to-flow ratio was obtained by two different methods. The first method involved securing feedwater to the steam generators and adjusting the turbine bypass valve controller to provide a constant cold leg temperature for one hour while steaming down the steam generators. This method was utilized from 70 to 130 minutes after RCP and reactor trip. The data indicated that 13.09 MWth decay heat (0.51% of rated power) was being removed with 9200 GPM of primary coolant flow (2.29% of full flow). This results in a power-to-flow ratio of 0.22. The second method, using the ΔT power readout, indicated a power-to-flow ratio of 0.207. These power-to-flow ratios (0.22 and 0.207) are well within the acceptance criterion of less than 1.0.

4.5.2 Event Specific RETRAN Modeling Changes

In order to model this total loss of flow/natural circulation test, the RETRAN two loop model was modified to initialize at the 40% power hot and cold leg temperatures, feedwater and main steam flow rates. Also, post-trip feedwater flow rate was adjusted to balance the temporal decay heat rate during the transient time period modeled by RETRAN.

4.5.3 Comparison of RETRAN Results to Plant Data

Natural circulation was confirmed in the RETRAN analysis of this event at about four minutes into the transient by decreasing hot leg temperature. Another indication of the natural circulation in the RETRAN analysis was the primary coolant system loop flow rates. These flows drop rapidly until about 12 minutes into the transient after which the loop flows become relatively steady. From 12 minutes until the end of the RETRAN calculation at 87 minutes into the transient, the total loop natural circulation flow remained at about 2.7% of initial forced circulation flow. The RETRAN calculation was run out to 87 minutes.

The key result of the Calvert Cliffs natural circulation test is the measured power-to-flow ratio during the period from 70 to 130 minutes after RCP and reactor trip. Using two different methods, plant data for this parameter was determined to be 0.220 and 0.207. The RETRAN calculated power-to-flow ratio at 75 minutes after RCP and reactor trip was 0.211. This closely agrees with plant data. Both the plant test natural circulation test power-to-flow data and similar RETRAN results were compared to other nuclear power plant data and theoretical calculations presented by Zvirin (33)

and were found to fall within the range of predicted performance for natural circulation.

Figures 4.5-1, 4.5-2 and 4.5-3 compare measured plant data with RETRAN results for pressurizer pressure and level and total RCS flow for the first 60 seconds after reactor/RCP trip. RETRAN very closely follows plant data.

Figures 4.5-4 and 4.5-5 compare RETRAN with measured plant data for RCS loop hot-cold leg temperature and steam generator secondary side pressure, respectively, for the first 60 seconds. RETRAN calculated hot and cold leg temperatures follow the same trend as plant data during this time period. RETRAN overpredicts hot leg temperature and underpredicts cold leg temperature by one to five degrees fahrenheit during the sixty seconds. This difference is attributed in TBV and ADV opening times and rates after reactor trip and does not affect the long-term onset of natural circulation. The RETRAN underprediction of peak steam generator secondary pressure is due to the same difference in TBV and ADV performance.

4.5.4 Summary of Results

The RETRAN Calvert Cliffs two loop model was used to simulate a total loss of flow/natural circulation test performance from 40% power at CCNPP Unit 1. For the first 60 seconds after reactor/turbine trip, RETRAN calculations were compared to measured plant data for pressurizer pressure and level, total RCS flow, hot and cold leg temperature and steam generator secondary side pressure. RETRAN closely matched measured RCS flow, pressurizer pressure and level. RETRAN results for hot and cold leg temperature and steam generator secondary side pressure differ slightly from

plant data, but follow the important trends. The most important result of this test was the measured natural circulation flow to reactor thermal power ratio from 70 to 130 minutes after reactor/turbine trip. Using two different methods, plant data for this ratio was determined to be 0.220 and 0.207. The RETRAN calculated ratio at 75 minutes after reactor-turbine trip was 0.211, which closely agrees with plant data and theoretical predictions.

FIGURE 4.5-1

TOTAL LOSS OF FLOW/NATURAL CIRCULATION TEST
TOTAL RCS FLOW RESPONSE

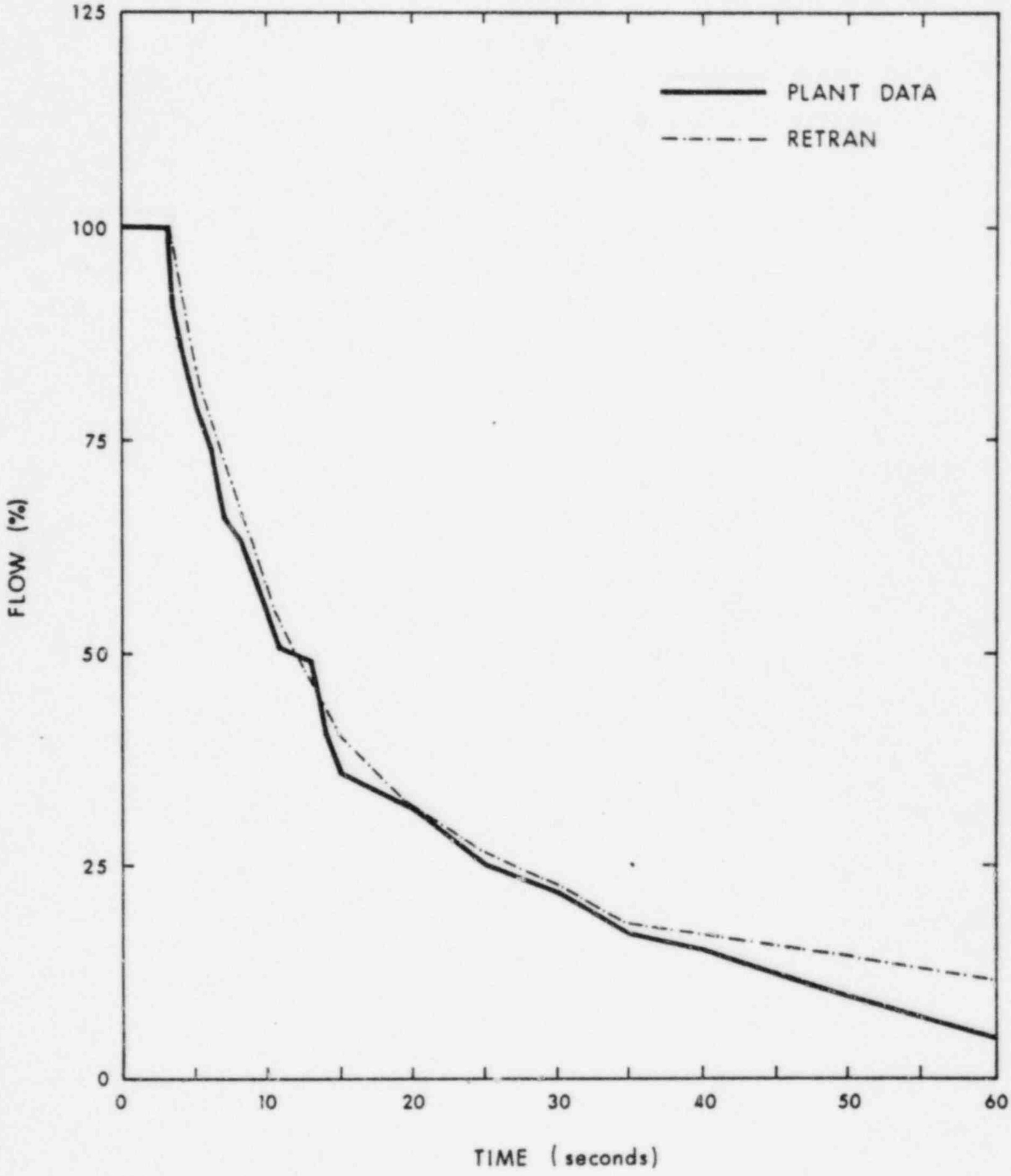


FIGURE 4.5-2

TOTAL LOSS OF FLOW/NATURAL CIRCULATION TEST
PRESSURIZER PRESSURE RESPONSE

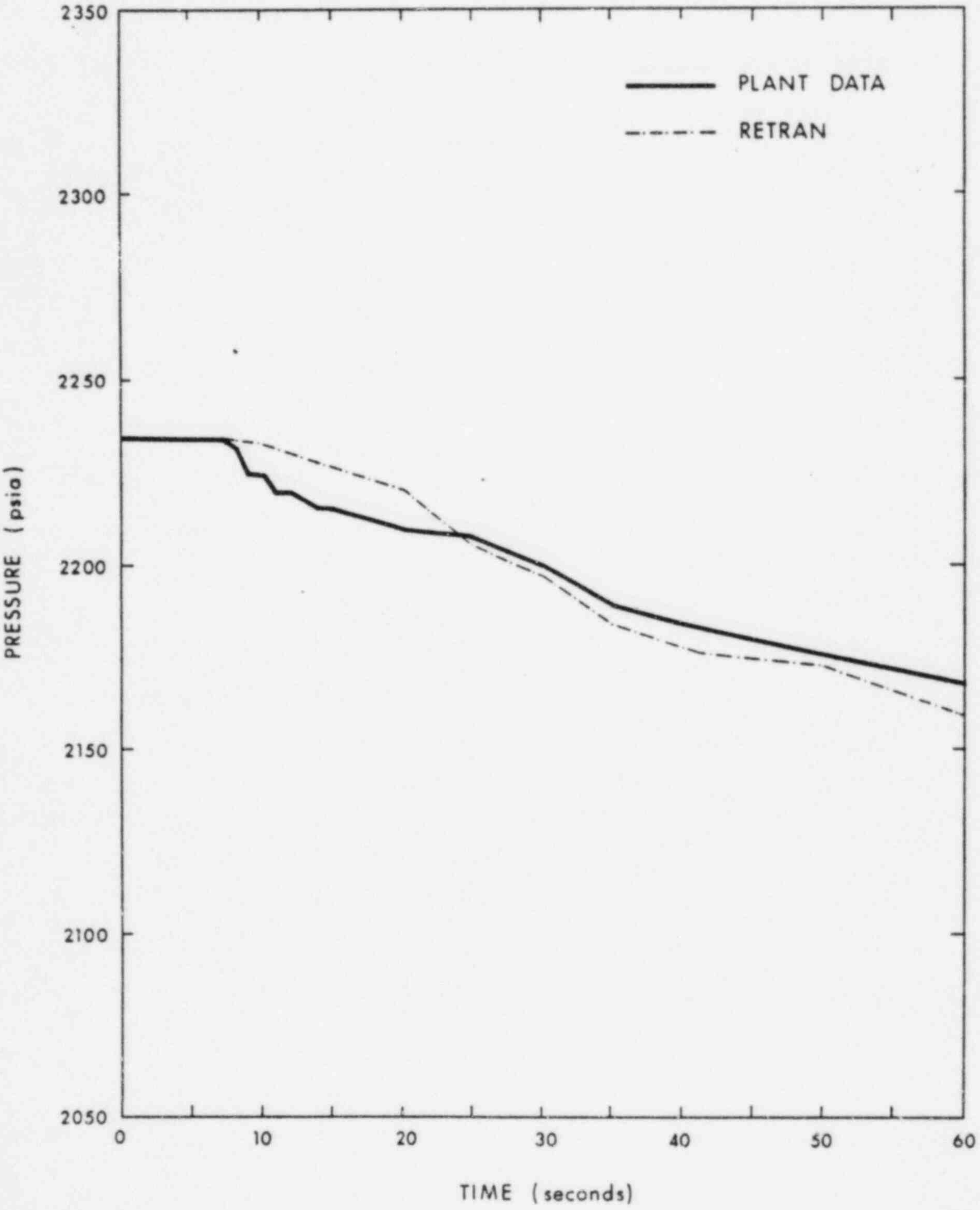


FIGURE 4.5-3

TOTAL LOSS OF FLOW/NATURAL CIRCULATION TEST
PRESSURIZER LEVEL RESPONSE

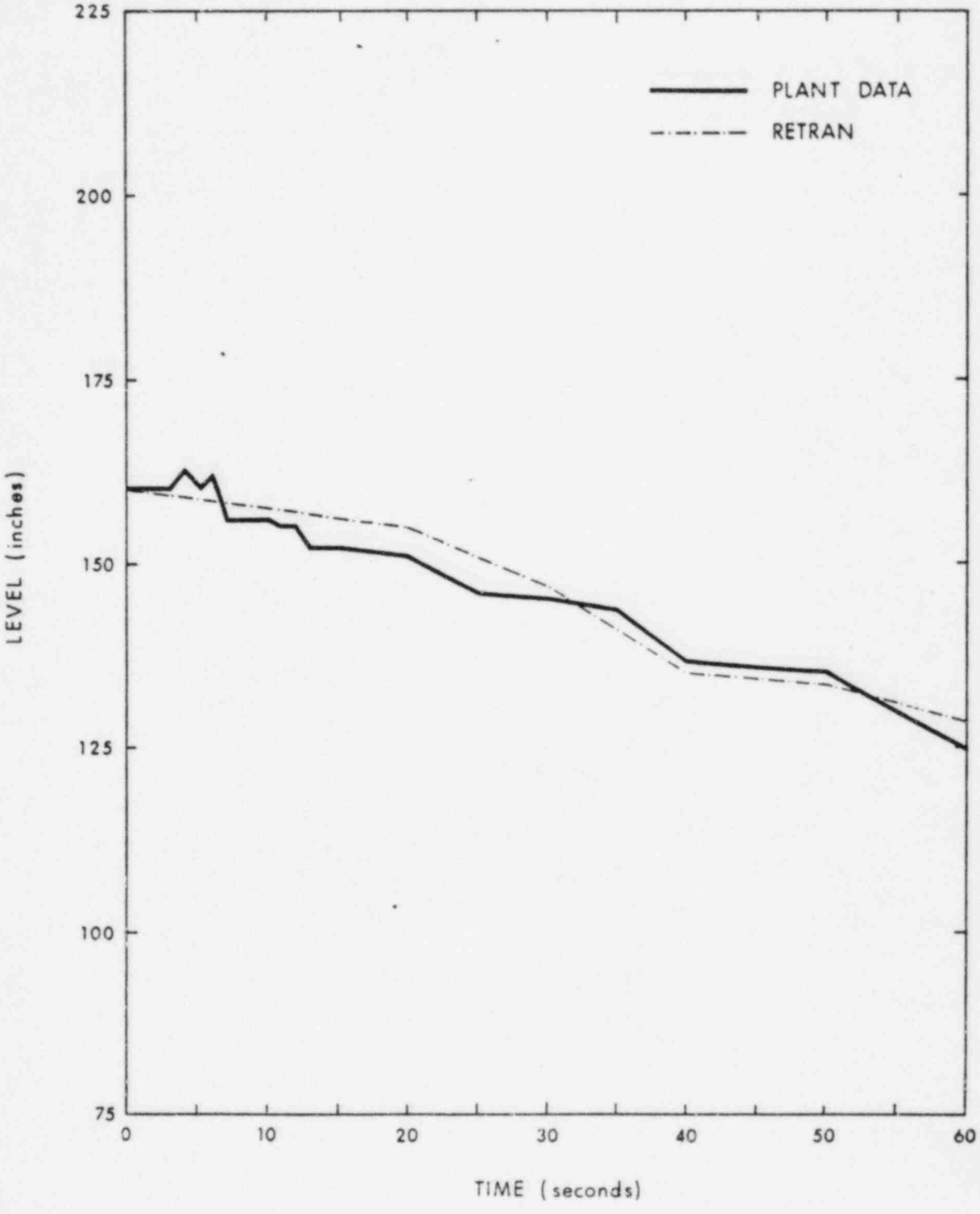


FIGURE 4.5-4

TOTAL LOSS OF FLOW/NATURAL CIRCULATION TEST
HOT AND COLD LEG TEMPERATURE RESPONSE

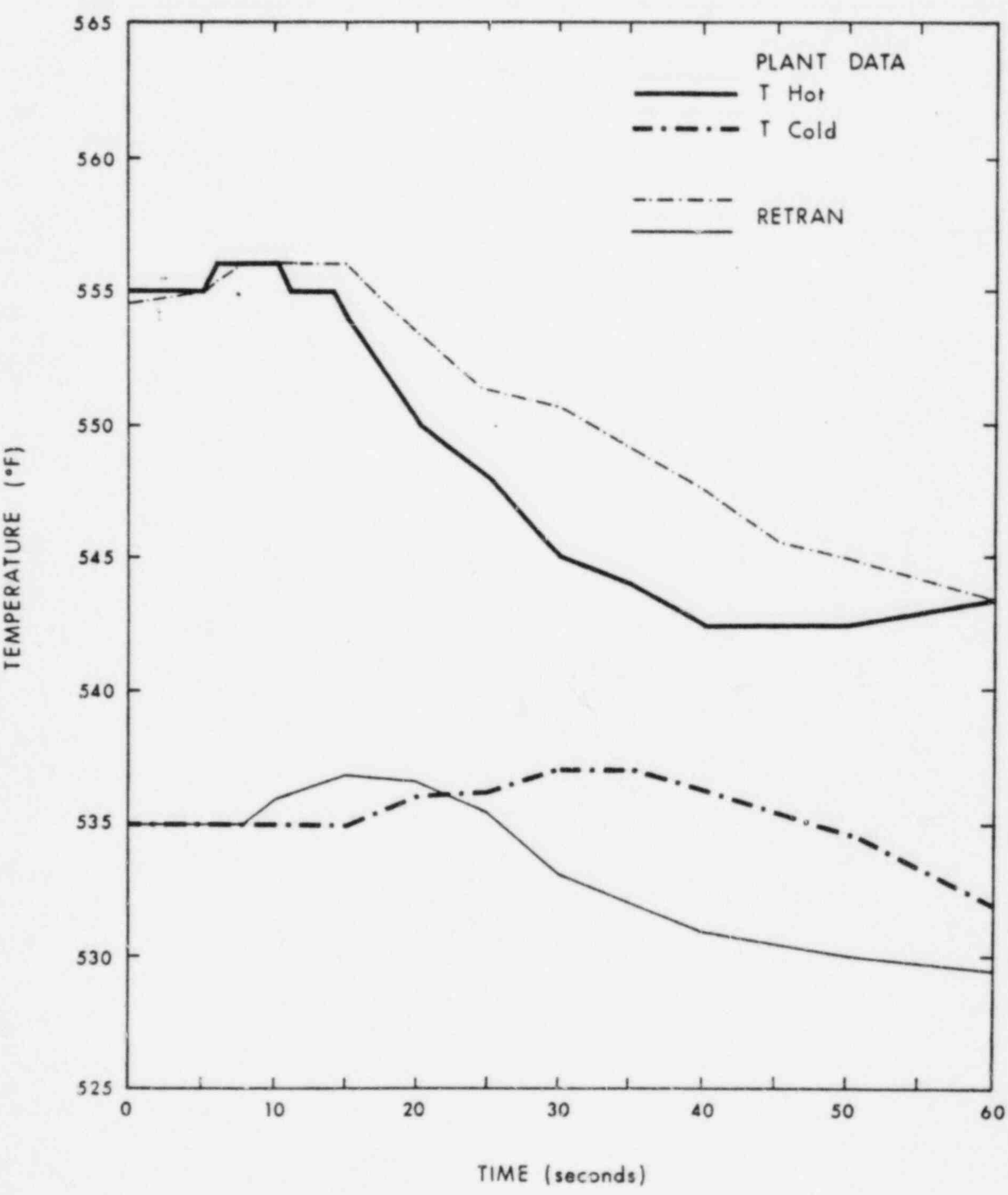
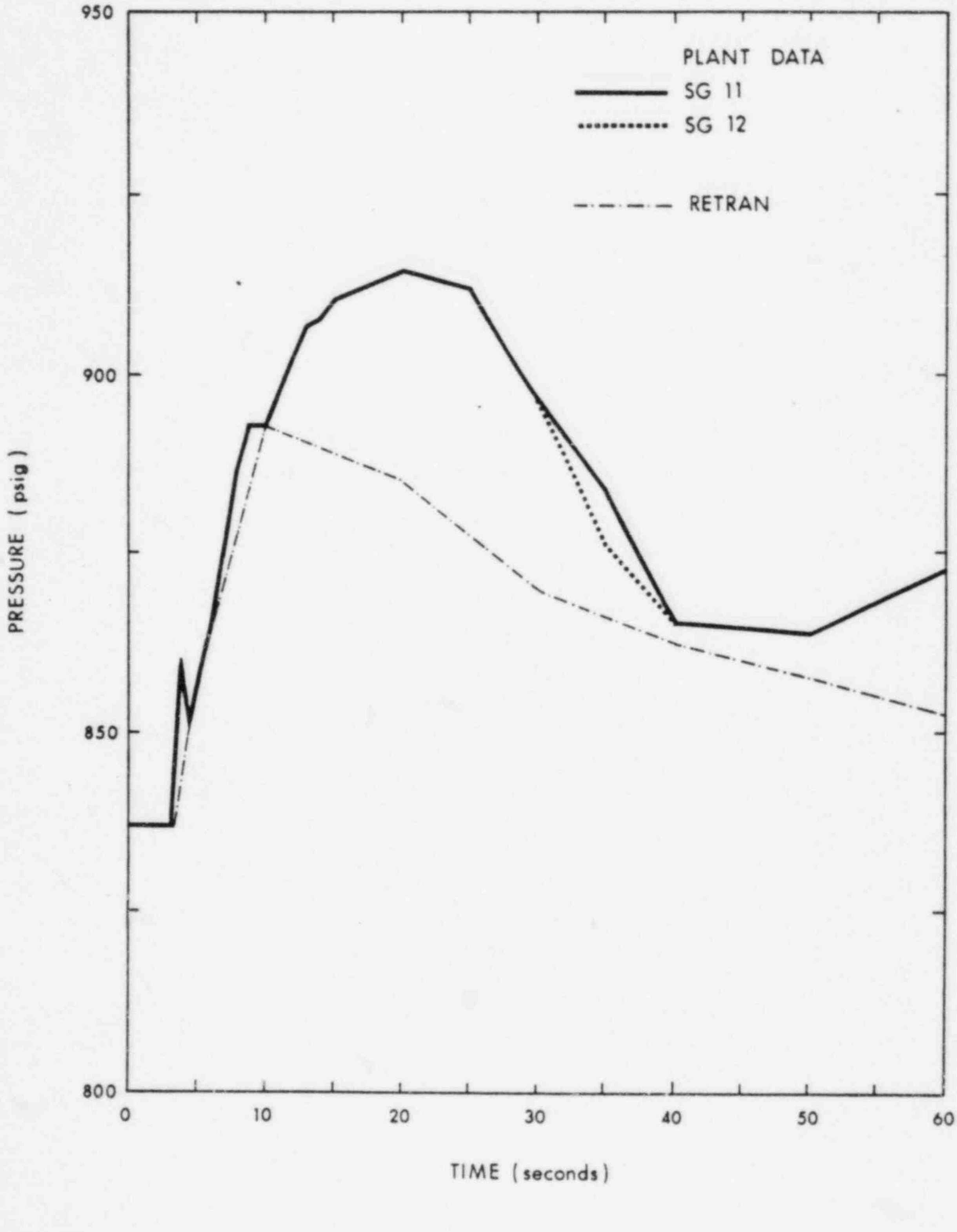


FIGURE 4.5-5

TOTAL LOSS OF FLOW/NATURAL CIRCULATION TEST
STEAM GENERATOR SECONDARY SIDE PRESSURE RESPONSE



5.0 TRAC ANALYSIS COMPARISON

5.1 Description of the TRAC Calvert Cliffs Model

TRAC-PF1 is an advanced best estimate finite-difference systems code capable of performing small, intermediate, and large-break loss of coolant accident (LOCA) analyses, and non-LOCA thermal-hydraulic transients in both one and three dimensions (31). TRAC solves the full set of field equations for mass, energy and momentum conservation for both steam and liquid.

Figure 5.1-1 shows the TRAC noding diagram for the primary side of Calvert Cliffs (35). The reactor vessel of Calvert Cliffs - 1 is modeled three dimensionally with twelve axial levels, two radial rings - the inner ring represented the core region located within the core-support barrel, the outer ring represented the annular downcomer region, located between the vessel wall and the core support barrel and six symmetric azimuthal segments - one for each penetration (four cold legs and two hot legs). The vessel model totaled 144 calculational-mesh cells, and was divided into 12 axial levels, using the bottom of the vessel as a reference point. The top of the first level corresponded to the bottom of the core support barrel. The top of the second level corresponded to the bottom of the fuel column. The bottom end fitting of each fuel assembly was located in level 2. The active core height, 11.4 ft. (3.5 m) was divided into five axial sections. The gas plenum of each fuel rod and the top end-fitting of each fuel assembly were located in level 8. The top of level 8 was at the same height as the bottom of the hot-leg penetrations and the top of level 9 corresponded to the top of the hot-leg penetrations.

In the reactor vessel, a small portion of the total flow (1.9%) goes through the CEA shrouds located in the upper plenum into the upper-head region and is referred to as bypass flow. The flow recirculates back into the upper plenum through 19 small holes located in the CEA grid-support plate.

Hot legs were modeled with an inner diameter of 48 in. (1.2 m) at the core barrel converging to 42 in. (1.0 m) outside the vessel. The surge line to the pressurizer was connected to the hot leg in Loop B.

The pressurizer was represented by three TRAC components. The first, PIPE component 10, simulated the part of the pressurizer containing the proportional and backup heaters. TEE component 47 was the major part of the pressurizer with a connection to the power operated relief valves (PORVs) and the primary safety relief valves (SRVs). This component contained six cells, which were found to be adequate for modeling the liquid/steam interface. PRIZER component 9 fixed the pressure at 2250 psia (15.51 MPa) during a steady state calculation.

There are two U-tube SGs with 8519 tubes, modeled in TRAC as a single flow path, with heat transfer area adjusted 10% to match stead state conditions supplied by BG&E. A typical study would use 20 primary cells, and 26 secondary cells.

The pump-suction end of the cold leg is a 6.5 ft. (2 m) U-shaped pipe leading to the RCPs. The rest of the piping is horizontal, with the HPI and charging flow injecting downstream of the RCPs. Each cold leg is modeled separately and represented the piping from the SG to the vessel.

HPI was modeled as a mass flow vs. pressure boundary condition with the fluid at a temperature of 55° F (286° K). HPI is injected into all four cold legs based on delivery curves supplied by C-E. HPI lines inside (120° F (322° K)), and outside (85° F (302° K) of containment was also taken into account.

Parts of the secondary system were included into the TRAC model of Calvert Cliffs. These parts included the steam lines up to the turbine-stop valves (TSVs) and turbine-bypass valves (TBVs) and about half of the feedwater train.

Figure 5.1-2 shows the major components of the complete main feedwater train. The high-pressure heaters were modeled as one heater as were the low-pressure heaters. Some 3280 ft. (1000 m) of pipe length was modeled with 170 fluid cells. Main feedwater pumps were modeled separately allowing one to be run in manual and the other in automatic.

Figure 5.1-3 shows the TRAC model noding diagram of the steam lines. The model did not include the steam lines that supply the MFW- and AFW-pump turbines. Furthermore, some liberty was taken with the arrangement of the line to the TBV. The line to the TBV is actually between the Loop-A main steam isolation valve (MSIV) and the lines to the high pressure (HP) turbines.

FIGURE 5.1-1
TRAC NODING OF CALVERT CLIFFS PRIMARY SIDE.

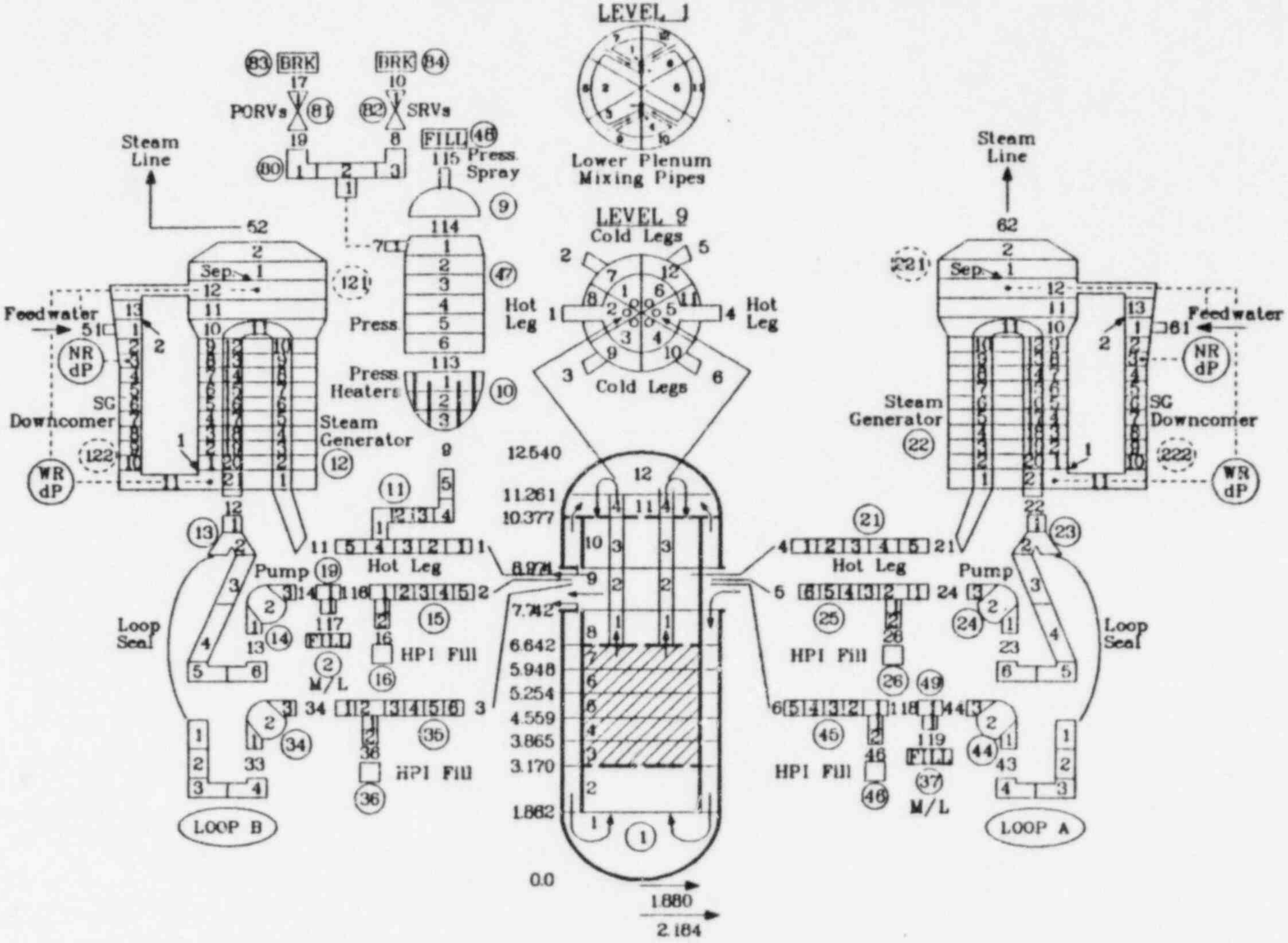


FIGURE 5.1-1 (Cont'd.)
TRAC NODING OF CALVERT CLIFFS REACTOR VESSEL

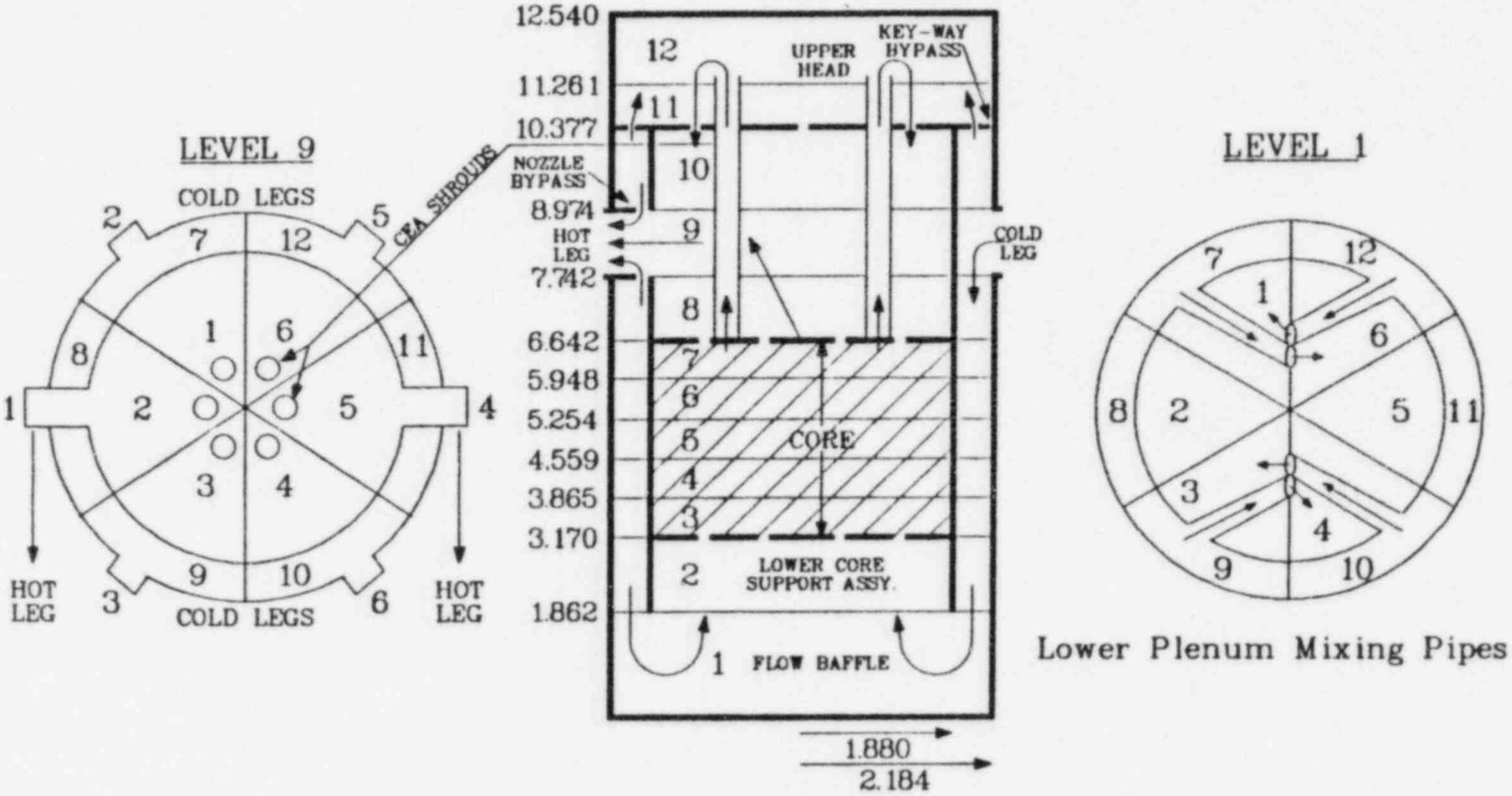


FIGURE 5.1-2

TRAC NODING OF CALVERT CLIFFS MAIN FEEDWATER TRANSIENT

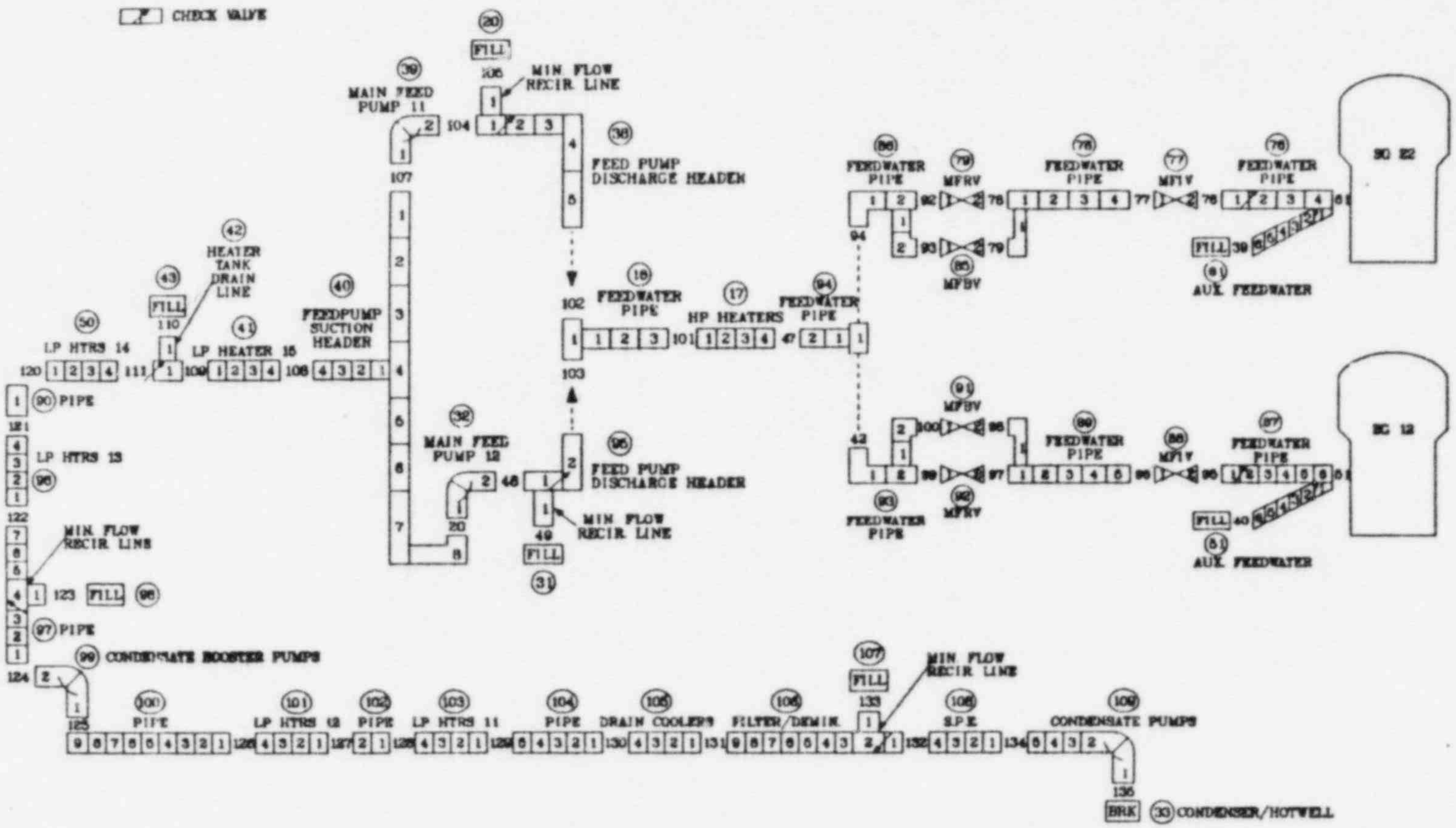
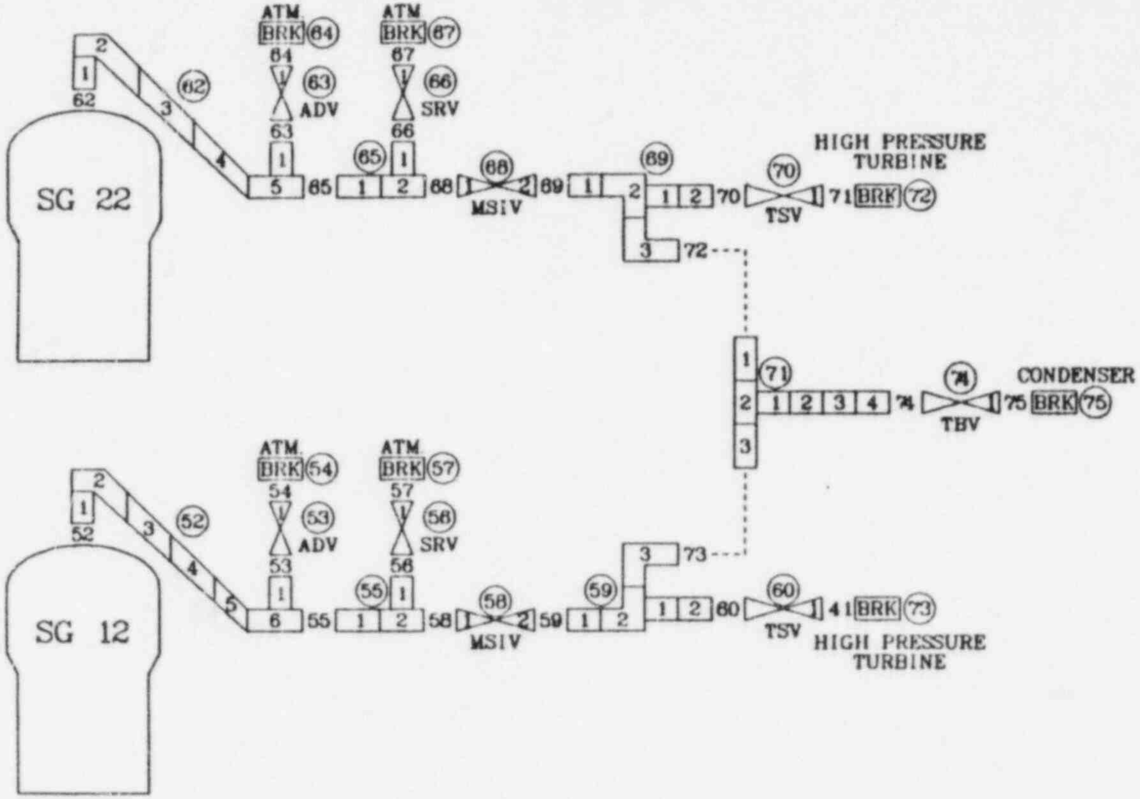


FIGURE 5.1-3
TRAC NODING OF CALVERT CLIFFS MAIN STEAM LINES



5.2 TRAC Analysis of Cooldown to RHR Entry Using ADV and APS

5.2.1 Description of TRAC Transient

The Los Alamos National Laboratory (LANL) performed a TRAC-PF1 analysis of the Baltimore Gas and Electric Company's Calvert Cliffs Nuclear Power Plant for a Loss of Off-Site Power (LOSP) natural circulation cooldown (36).

A loss of off-site power causes the reactor and turbine to trip immediately. The turbine stop valves and turbine bypass valves shut to protect the turbine and condenser respectively. The main steam safety relief valves (SRVs) begin lifting at their set point of 985 psig. The plant diesel generators start up and provide power to the vital busses, allowing the atmospheric dump valves to operate. The four reactor coolant pumps coast down and a natural circulation condition is established. Temperature is maintained by control of the atmospheric dump valves. Auxiliary feedwater is delivered by one steam driven and one electric driven Aux feed pump. Shutdown boron concentration is established and a 100°F per hour cooldown to RHR entry condition is begun.

The LANL calculation commenced a cooldown immediately after reactor trip, not accounting for the delay necessary to ensure proper shutdown boron concentration. For consistency, this omission was duplicated in the RETRAN analysis. For the LOSP transient, LANL employed a fill boundary condition for the steam generator feedwater.

5.2.2 RETRAN Models for TRAC Comparison

The RETRAN model used for this analysis was the simplified one loop primary coolant system and a non-recirculating steam generator described in Section 3.2. As a result, this model runs in real time on the BG&E IBM mainframe computer.

5.2.3 Comparison of RETRAN and TRAC Results

Table 5.2-1 lists the sequence of events for the LOSP transient. No operator action is taken until 10 minutes have passed. At that time, auxiliary feedwater is delivered. One minute later the operators take control of the ADVs to begin the 100°F per hour cooldown. After another minute, the operator realigns the charging pumps to provide auxiliary pressurizer spray (also disabling the makeup flow) and controls primary depressurization to maintain 30°F subcooling.

As can be seen in Figure 5.2-1, primary pressure compares very well. Sensitivity analyses show that the rate of depressurization depends strongly on APS flow rate and is unaffected by the interphase heat transfer coefficient. The sudden pressure drop predicted by RETRAN at 3200 seconds corresponds to an emptying of the pressurizer. This is consistent with a previous comparison to TRAC for an overcooling event (37) in which RETRAN predicted pressurizer emptying due to primary contraction and TRAC did not.

Figure 5.2-2 shows the hot leg temperature response. Neither RETRAN nor TRAC predicted a cooldown to 300°F. RETRAN predicted a more limited cooldown due to a more restrictive ADV model which takes into account the geometry of all piping associated with the ADVs. The TRAC operator ADV

control system oscillates to maintain the $100^{\circ}F$ per hour cooldown average. The RETRAN control is much finer, producing no noticeable oscillations.

Figure 5.2-1 shows the good agreement in calculated secondary pressure. RETRAN predicts slightly higher pressure than TRAC due to the ADV modeling. Again, the oscillating ADV control is noticeable in the TRAC results. Figure 5.2-2 once again demonstrates the importance of the ADV model to this calculation. The best estimate RETRAN model limits the steaming rate and slows the secondary liquid temperature cooldown relative to TRAC.

5.2.4 Summary and Conclusion

For a LOSP natural circulation cooldown at Calvert Cliffs, a simplified fast-running RETRAN model can duplicate the results of a sophisticated TRAC-PF1 model. In this case, the RETRAN model takes full advantage of the symmetry of the event. The results are most sensitive to APS flowrate and ADV modeling. Although not investigated by sensitivity analyses, the primary system heat capacity and decay heat curves are also important in long term cooling transients.

Table 5.2-1

LOSP COOLDOWN SEQUENCE OF EVENTS FOR TRAC AND RETRA

Time(s)	Event
0.0	LOSP event
0.1	Reactor trip
	MFW trip*
	Turbine trip*
	Reactor coolant pumps trip*
	ADV trip signal but action delayed 14.5 s
0.52	TBV trip
14.73	ADVs quick-open after 14.5 s delay
600.9	AFW initiated
723.	APS on (cycling thereafter)
11500	END OF CALCULATION

* These events occurred at time zero in the RETRAN analysis.

FIGURE 5.2-1

COOLDOWN TO RHR ENTRY - PRIMARY AND SECONDARY PRESSURE RESPONSE

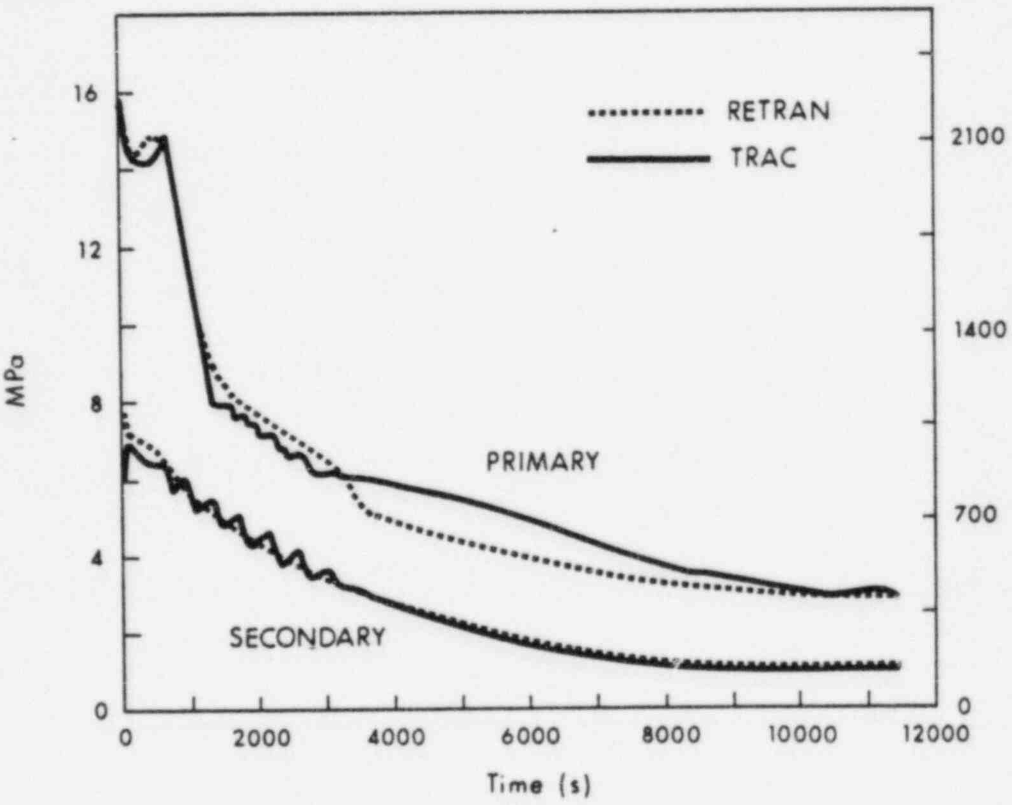
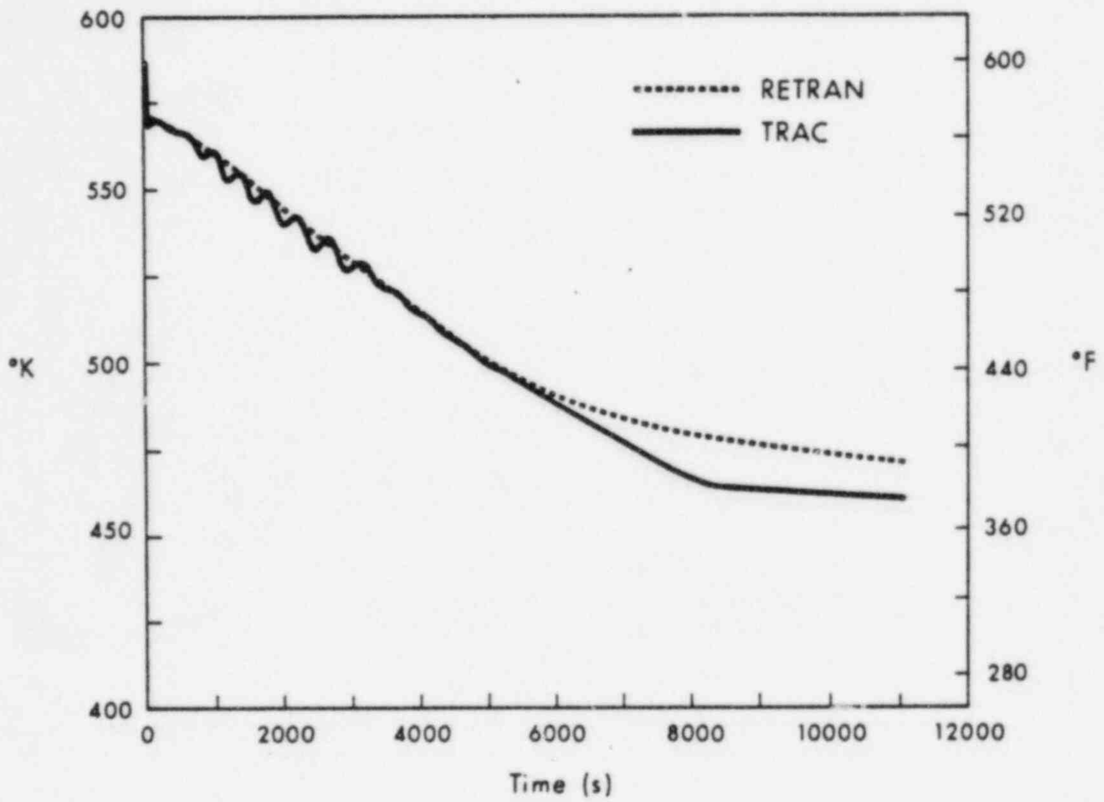


FIGURE 5.2-2

COOLDOWN TO RHR ENTRY - HOT LEG TEMPERATURE RESPONSE



5.3 RETRAN Comparison to TRAC Simulation of Runaway Feedwater to One Steam Generator

5.3.1 Description of TRAC Transient

As part of the NRC sponsored pressurized thermal shock (PTS) project (1), the Los Alamos National Laboratory (LANL) was contracted to develop a detailed model of Calvert Cliffs Unit 1 and perform a set of PTS-limiting system transient analyses (35). These analyses were all performed as best estimate calculations without the imposition of the usual licensing analysis conservative assumptions.

One of the categories of Calvert Cliffs PTS transient analyses analyzed by LANL was runaway feedwater. It was postulated that, following a reactor/turbine trip, the main feedwater regulating valves (MFRV) fail to close and MFW continues to be pumped into the steam generator. This injection of colder MFW with a reduced heat source on the primary side of the steam generator (i.e., core at decay heat) results in an overcooling PTS event. The specific TRAC runaway feedwater analysis chosen for this RETRAN comparison was the runaway main feedwater to one steam generator. This transient results in both an overcooling and asymmetric response of the primary cooling system.

The key PTS parameters calculated by TRAC are downcomer temperature and primary system pressure since these are the thermal-hydraulic factors that influence high neutron fluence induced weld embrittlement to allow a preexisting weld crack to grow in PTS. TRAC calculated a minimum downcomer temperature of 424° F at 363 seconds. Primary system pressure initially drops due to the overcooling but at 1850

seconds rises and cycles around the PORV setpoint due to operator failure to shut off the charging pumps.

The TRAC analysis showed several interesting phenomena during this postulated transient. The transient response was divided into the following five phases:

1. Runaway main feedwater to one steam generator (0 to 363 seconds)
2. Reheat following termination of excess feedwater (363 to 3200 seconds)
3. Quasi-equilibrium (3200 to 4800 seconds)
4. Auxiliary feedwater to both steam generators (4800 to 5800 seconds)
5. Extrapolation (5800 to 7000 seconds)

During Phase 1, the steam generator receiving normal post-trip feedwater enters a reverse heat transfer mode in which it heats the primary coolant system. This phenomena is due to the overfed steam generator cooling down the primary coolant system below the normal post-trip saturation temperature being maintained by the other steam generator. This effect only lasts until the primary coolant temperature rises back above the steam generator saturation temperature after the runaway MFW is terminated by depletion of the condenser hotwell liquid inventory. SIAS and HPSI flow injection also occur during Phase I in which primary coolant pressure drops to 1000 psia.

Phase 2 reflects the return to normal post-trip system thermal-hydraulic conditions after the runaway MFW train depleted condenser hotwell inventory. Phase 3 is the equilibrium state where normal decay heat removal via steam generator inventory steaming occurs. In Phase 4, low steam generator level initiates auxiliary feedwater which results in a slight cooldown as the colder AFW fluid enters the steam generator. Phase 5 is an extrapolation of TRAC results out to 7000 seconds for PTS purposes.

The TRAC calculated sequence of events and plots of TRAC calculated system parameters are presented and compared to RETRAN results in Section 5.3.3.

5.3.2 RETRAN Model for TRAC Comparison

The two-loop model decribed in Section 3.3 was used to simulate the runaway feedwater transient. This model was modified in several areas to better simulate the transient scenario evaluated by TRAC. The Auxiliary Feedwater Control System was isolated to prevent AFW injection in accordance with the TRAC simulation. In addition, the charging pump control system was modified to allow continuous operation after initiation even after pressurizer level returned to normal levels. This failure to isolate charging flow was also assumed in the TRAC calculation.

A major modification of the tube volumes in the steam generator isolated from the runaway main feedwater was necessary for this RETRAN calculation. These modifications were: Consolidation of the four highest elevation steam generator tube volumes (nodes 111, 113, 115 and 117 in Figure 3.3-1) into a single volume (new node 111); inclusion of a bubble rise model for two phase thermodynamic conditions in the new consolidated volume (node 111); and elimination of the enthalpy transport model from the two junctions connected to this new consolidated volume (node 111). These changes were required because of the Phase I phenomena of steam generator reverse heat transfer discussed in Section 5.3.1. This reverse heat transfer causes void formation in the upper tube volumes. The two loop RETRAN model without the aforementioned modifications experienced large void formation in the relatively small upper tube volumes which causes RETRAN

to abort when calculating fluid properties. These modifications reduce the magnitude of void formation since the upper tube volume is four times larger. Also, by including the bubble rise model and deleting the enthalpy transport model, RETRAN can calculate fluid properties.

The driving boundary condition for this transient is the temporal main feedwater flow rate and enthalpy to each steam generator. Since the RETRAN model does not include an explicit simulation of the feedwater train back to the condenser, the flow and enthalpy used in the TRAC analysis was used as a fill junction boundary condition in the RETRAN calculation.

5.3.3 Comparison of RETRAN Results to TRAC

Key RETRAN calculated results for pressurizer pressure and level, downcomer temperature, hot leg temperature, and hot leg flow rate are compared to TRAC calculations in Figures 5.3-1, 5.3-2, 5.3-3, 5.3-4 and 5.3-5, respectively. The RETRAN calculation was terminated at 3600 seconds due to excessive computer run time, whereas TRAC results were presented out to 5800 seconds. All important trends and phenomena occur prior to 3600 seconds for this transient.

In Figure 5.3-1, RETRAN pressurizer pressure drops to a minimum slightly lower than TRAC, but at the same time. RETRAN's pressure recovers and rises to the 2400 psia PORV setpoint slightly earlier than TRAC. The initial drop in pressure is due to the overcooling effect of runaway main feedwater. The subsequent pressure rise to 2400 psia reflects termination of main feedwater flow at 303 seconds and uninsolated continuous charging pump flow postulated as part of the basis for this PTS event. In the pressurizer level comparison in Figure 5.3-2, both RETRAN and

TRAC predict a rapid drop in level to an empty pressurizer by about 200 seconds. This level drop corresponds to the pressurizer pressure drop caused by the runaway MFW. Both codes calculate level recovery after cessation of MFW flow although RETRAN shows a more rapid level rise to the top of the pressurizer (30 foot level). The level rise is due to the termination of the runaway MFW overcooling source which is later exacerbated by the runaway charging pump flow.

Figures 5.3-3 and 5.3-4, RETRAN calculated downcomer and hot leg temperatures follow the same trends as TRAC results. In both figures, RETRAN predicts a lower minimum temperature occurring at the same time and the same trend of increasing temperature throughout the balance of the event. The lower RETRAN temperatures can be attributed to the finer nodalization in TRAC. TRAC utilizes six circumferential cells (nodes) and nine axial levels for a total of 54 nodes to represent the downcomer as compared to RETRAN's two node downcomer. The greater number of nodes and concomitant crossflow paths in the TRAC model allows greater mixing of the colder fluid from the overfed steam generator with the warmer fluid from the unaffected steam generator.

Figure 5.3-5 presents the hot leg flow rates for both loops as calculated by TRAC and RETRAN. The loop A flow drops to about 750 lbm/sec whereas the loop B flow drops close to zero and subsequently recovers to match the loop A flow at about 3600 seconds. The higher loop A flow in the early phase (500 to 3000 seconds) of this transient was driven by the higher natural circulation induced by the runaway feedwater to steam generator A and resulting larger temperature drop across this steam generator. After MFW isolation at 303 seconds and the gradual establishment of thermal equilibrium between the two loops, the natural circulation flow becomes loop-

symmetrical. RETRAN calculated flow rates for each loop closely follow TRAC results including the large difference between loops until about 3000 seconds when both loop flows converge. This loop hot leg convergence is predicted by both RETRAN and TRAC.

Table 5.3-1 presents a comparison of key sequence of events for this transient as calculated by both TRAC and RETRAN. RETRAN predicts almost identical timing for initiation of steam generator B reverse heat transfer, SIAS, and minimum average downcomer temperature. The event initiation and cessation of main feedwater flow times are dictated by the transient specific assumptions.

5.3.4 RETRAN Sensitivity Study

Due to the long computer run times associated with this analysis, a limited sensitivity study was performed for the RETRAN simulation of this transient. The nonequilibrium pressurizer model interface heat transfer coefficient (IHTC) was varied from the base case value of 32,400 BTU/HR-FT^2-F down to 1000 and 10 BTU/HR-FT^2-F. The base case value of IHTC was substantiated in Section 4.1, which evaluated a rising pressure transient. Since the runaway main feedwater event is a decreasing pressure event, these two additional calculations with lower IHTCs were performed to evaluate the effect on subsequent pressure response. Both cases of lower IHTC resulted in no significant effect on pressurizer pressure or level response for this transient.

5.3.5 Summary of Results

Using a modified version of the CCNPP two loop model described in Section 3.3, RETRAN calculations of a postulated runaway main feedwater to one steam generator event were performed. This scenario had also been analyzed with the TRAC PF1 computer code as part of the PTS project. RETRAN results followed the same trends and timing for key thermal-hydraulic parameters such as pressurizer pressure and level, downcomer and hot leg temperature and hot leg flow rates. RETRAN slightly underpredicted the minimum pressure and temperatures. Important asymmetric phenomena including different loop natural circulation flow rates and steam generator B reverse heat transfer were predicted by RETRAN in agreement with TRAC. Also, capability for analyzing steam generator tube voiding and pressurizer level from empty to water solid with RETRAN was demonstrated.

Table 5.3-1

RUNAWAY MAIN FEEDWATER TO ONE STEAM GENERATOR
SEQUENCE OF EVENTS COMPARISON

TIME (SECONDS)		EVENT
TRAC	RETRAN	
0	0	Reactor/Turbine Trip - MFRV to Steam Generator A failed to open - ADVs and TBVs open on "quick-open" logic
102	106	Steam Generator B enters into reverse heat transfer mode
123	122	Safety Injection Activation Signal (SIAS) on RCS pressure of 1740 psia
153	152	Reactor Coolant Pumps tripped off by operator 30 seconds after SIAS
303	303	Main Feedwater pumps trip off due to loss of liquid inventory in condenser/hotwells
363	368	Minimum average downcomer temperature reached

FIGURE 5.3-1

RUNAWAY MAIN FEEDWATER TO ONE STEAM GENERATOR
PRESSURIZER PRESSURE RESPONSE

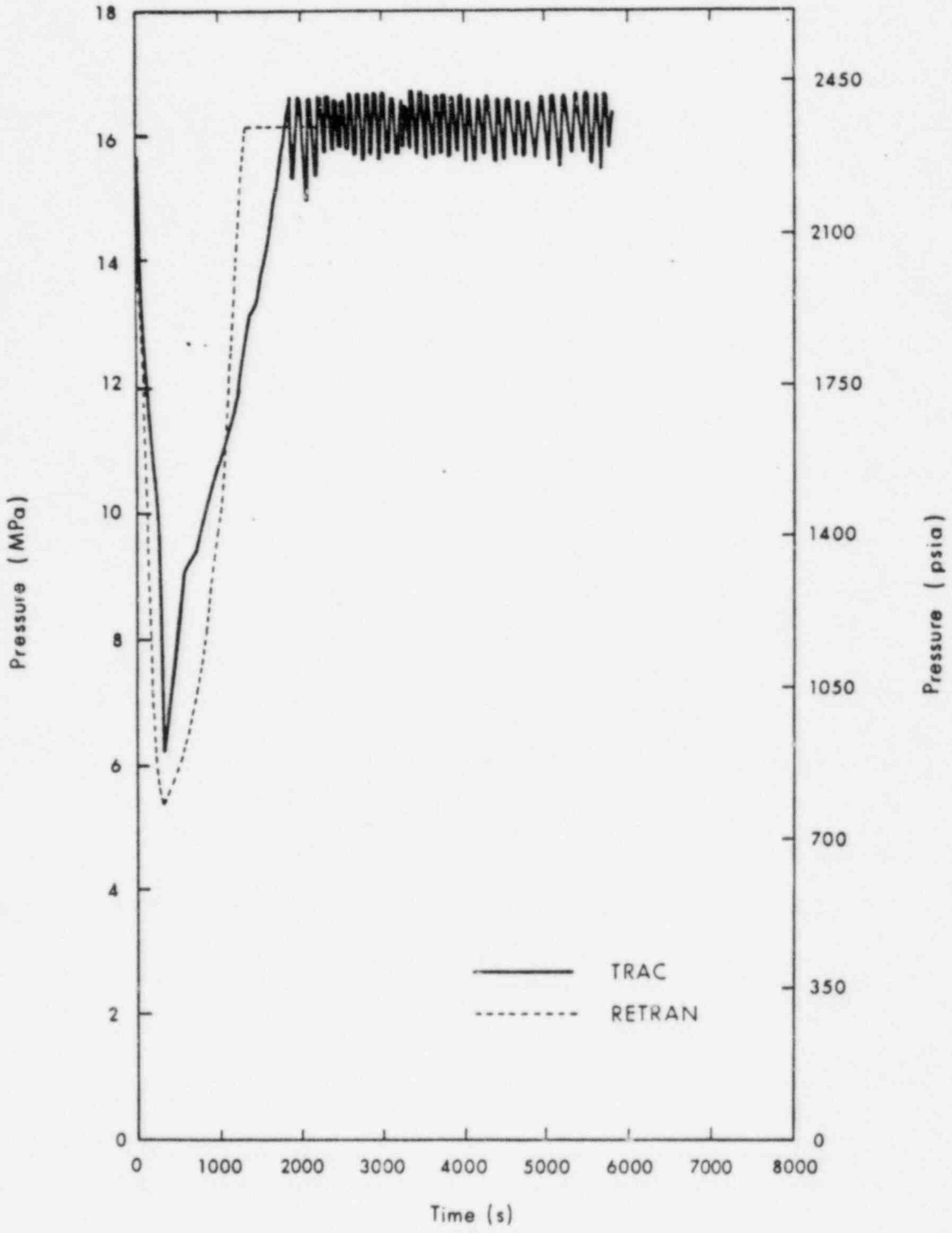


FIGURE 5.3-2

RUNAWAY MAIN FEEDWATER TO ONE STEAM GENERATOR
PRESSURIZER LEVEL RESPONSE

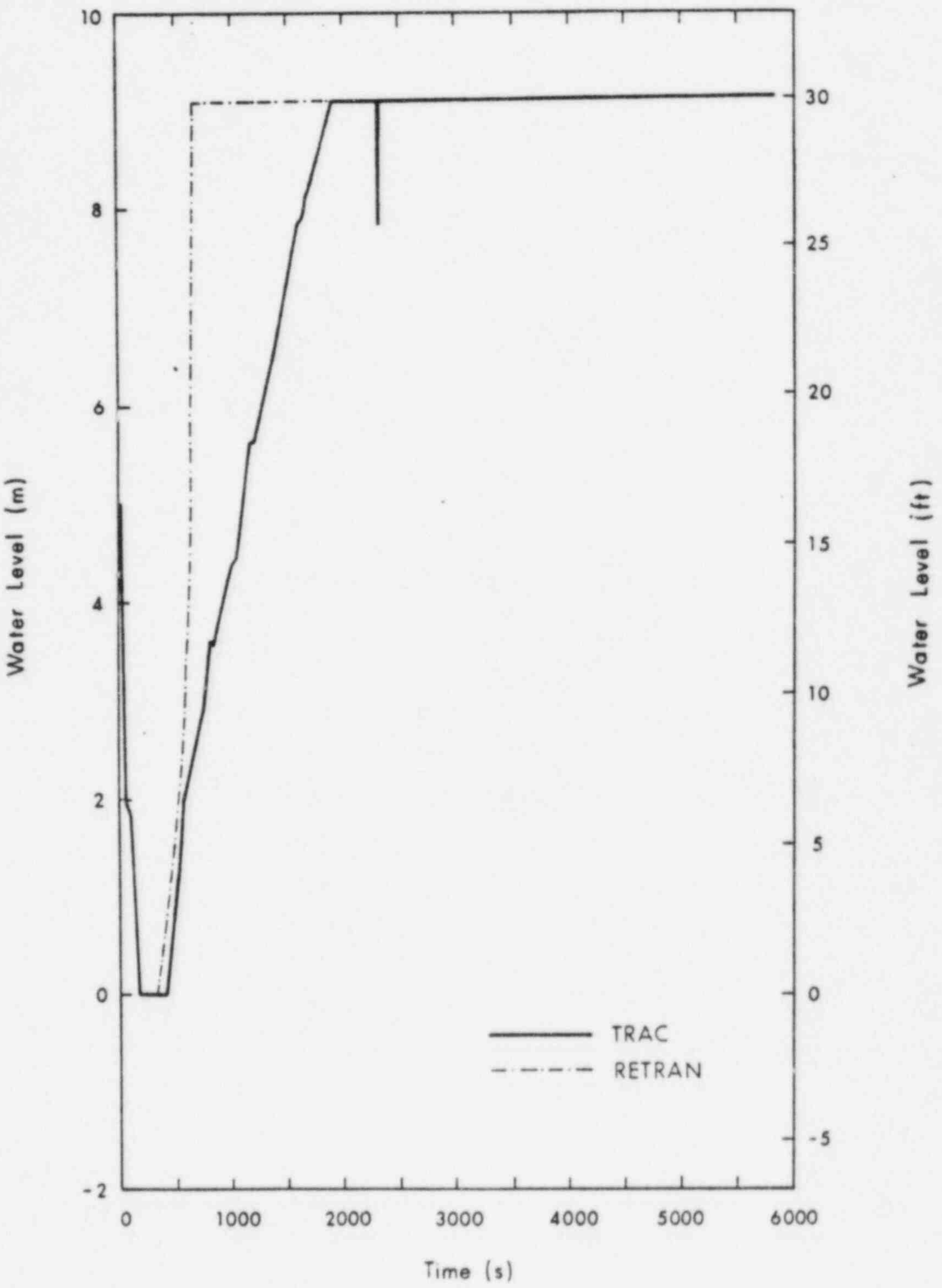


FIGURE 5.3-3

RUNAWAY MAIN FEEDWATER TO ONE STEAM GENERATOR
DOWNCOMER TEMPERATURE RESPONSE

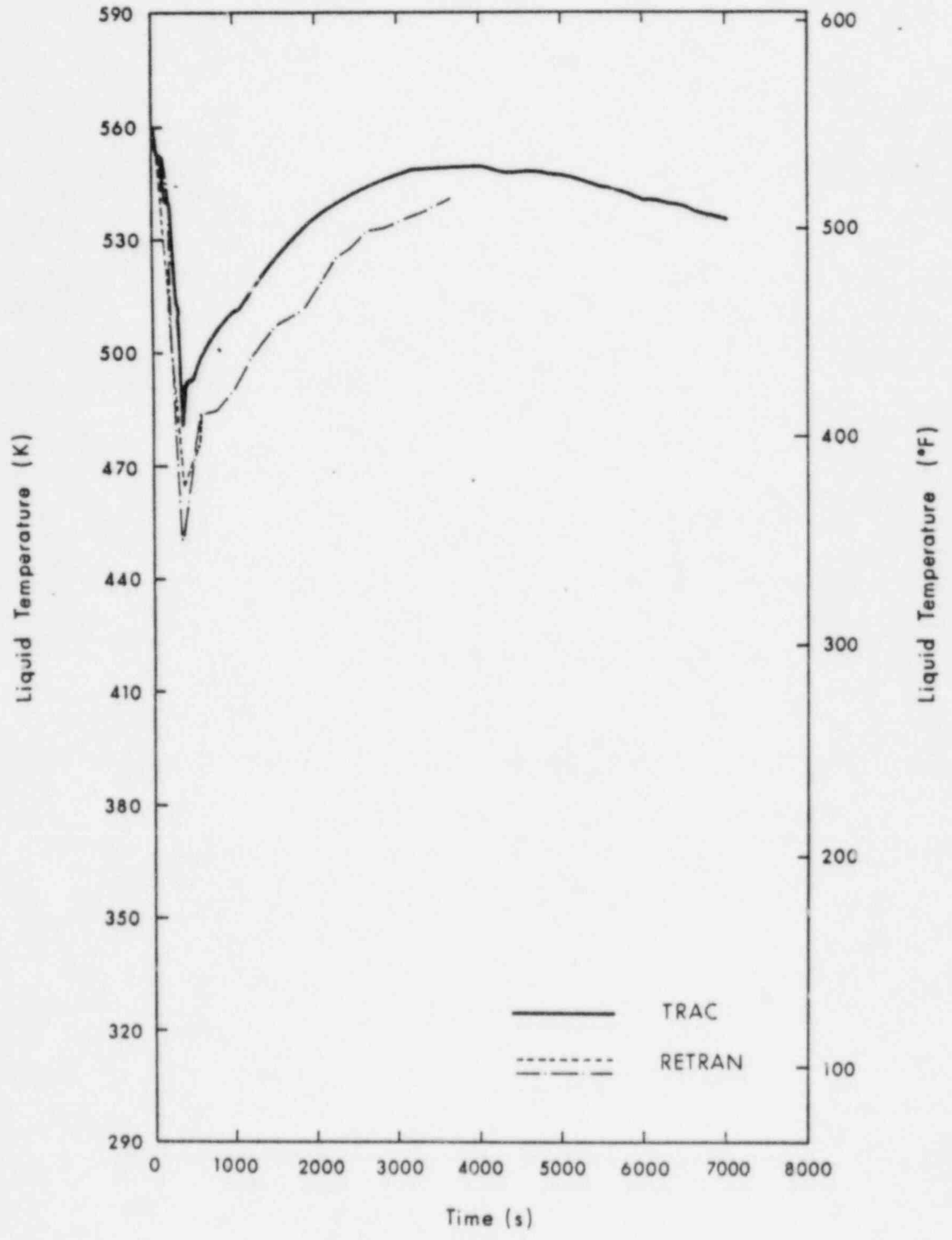


FIGURE 5.3-4

RUNAWAY MAIN FEEDWATER TO ONE STEAM GENERATOR
HOT LEG TEMPERATURE RESPONSE

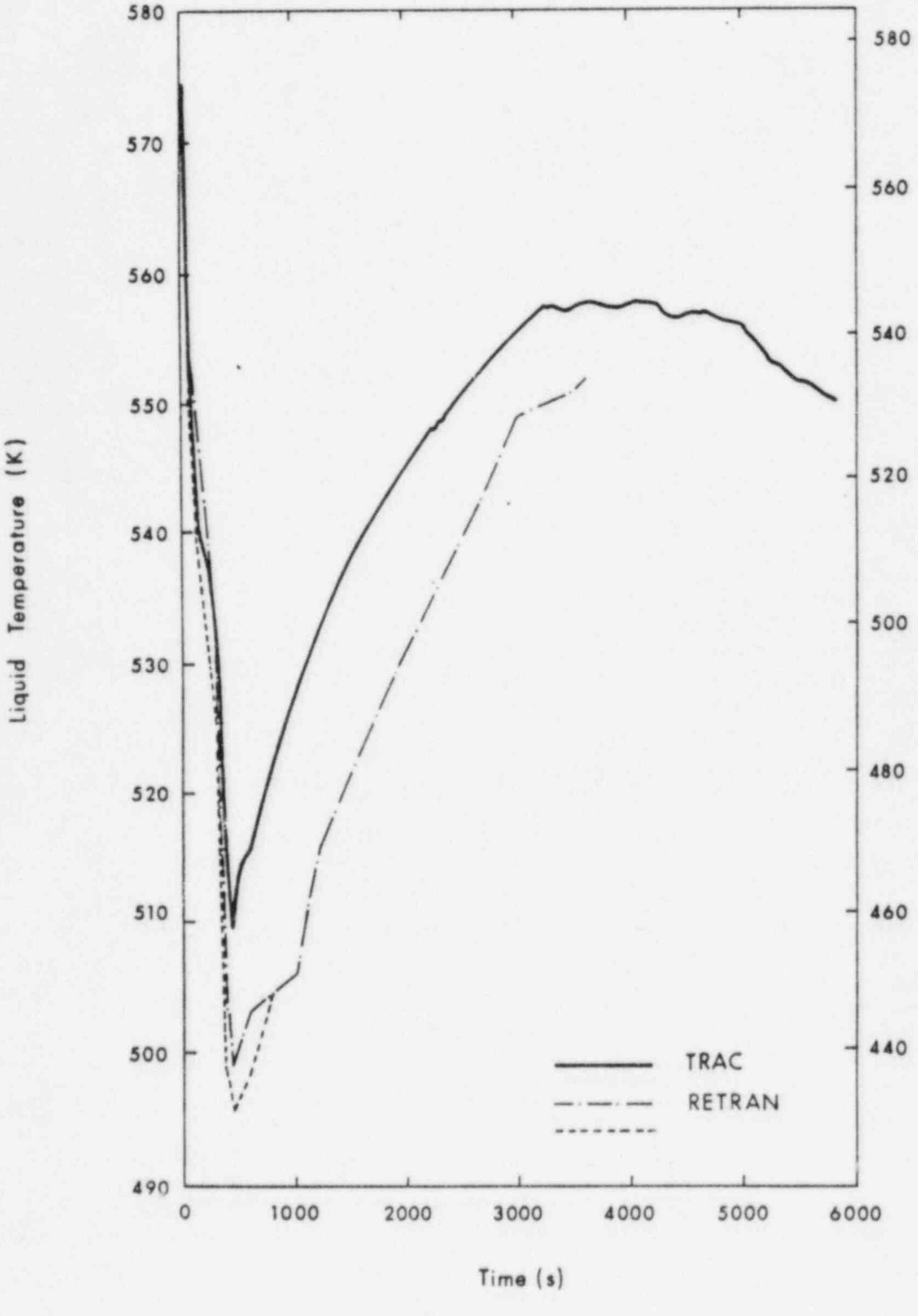
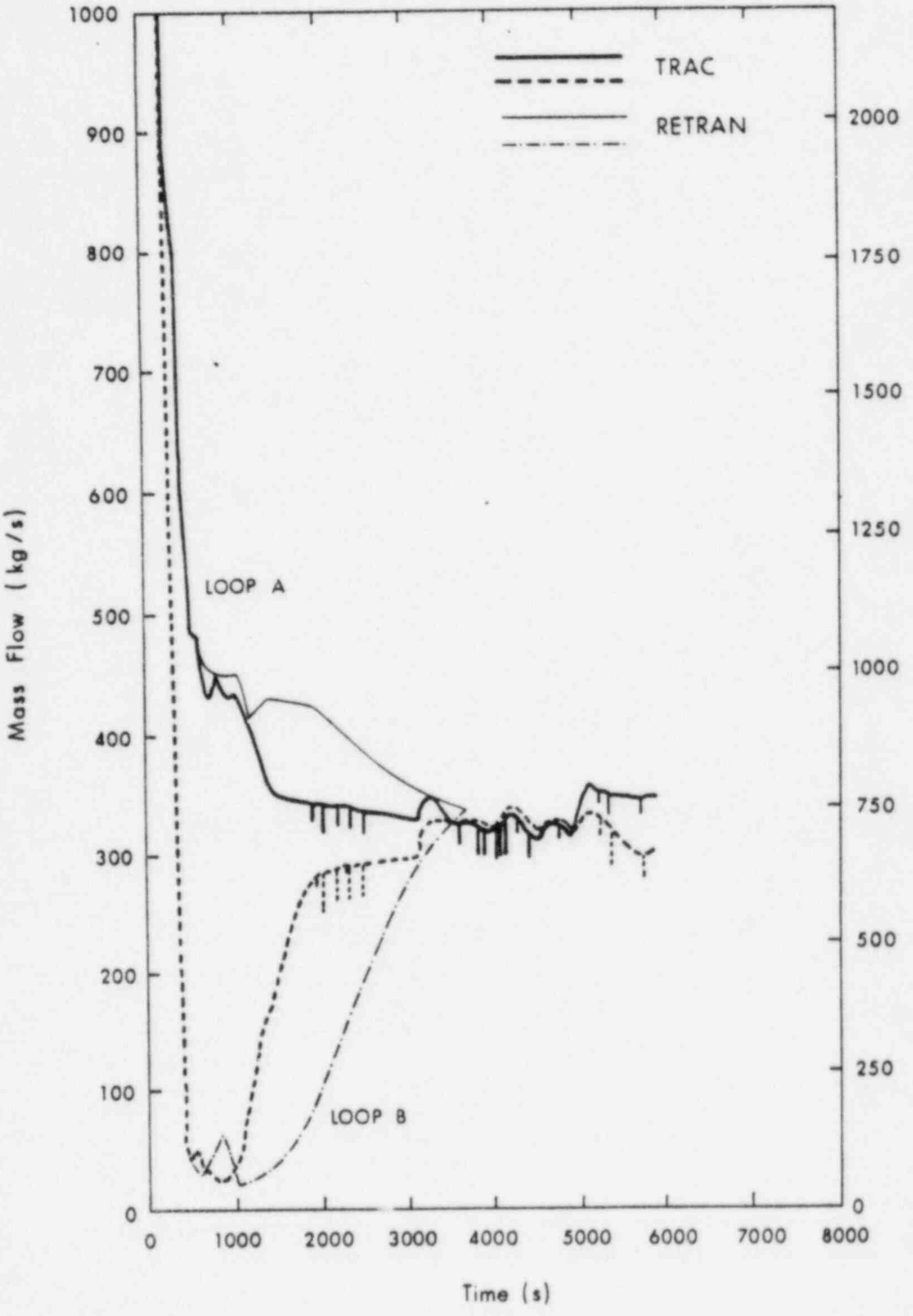


FIGURE 5.3-5

RUNAWAY MAIN FEEDWATER TO ONE STEAM GENERATOR
HOT LEG MASS FLOW RESPONSE



6.0 LOFT TEST SIMULATION

6.1 Description of LOFT Facility and Tests

The LOFT (Loss of Fluid Test) facility was used in a series of experiments administered by the Department of Energy to provide data required to evaluate the adequacy and improve the analytical methods currently used to predict the loss of coolant accident (LOCA) response and anticipated transients caused by abnormal operation of pressurized water reactors (38, 39). A schematic of LOFT is presented in Figure 6.1-1.

The primary objective of this facility was to test the performance of the engineered safety features with particular emphasis on emergency core cooling system and the quantitative margins of safety inherent in the performance of the engineered safety features. Another objective was to identify and investigate an unexpected event in the response of the plant or the engineered safety features and develop analytic techniques to adequately describe such behavior. A third objective was to evaluate and develop methods to prepare, operate and recover systems and plant for and from reactor accident conditions. The last objective was to identify and investigate methods by which reactor safety can be enhanced, with emphasis on the interaction of the operator with the plant.

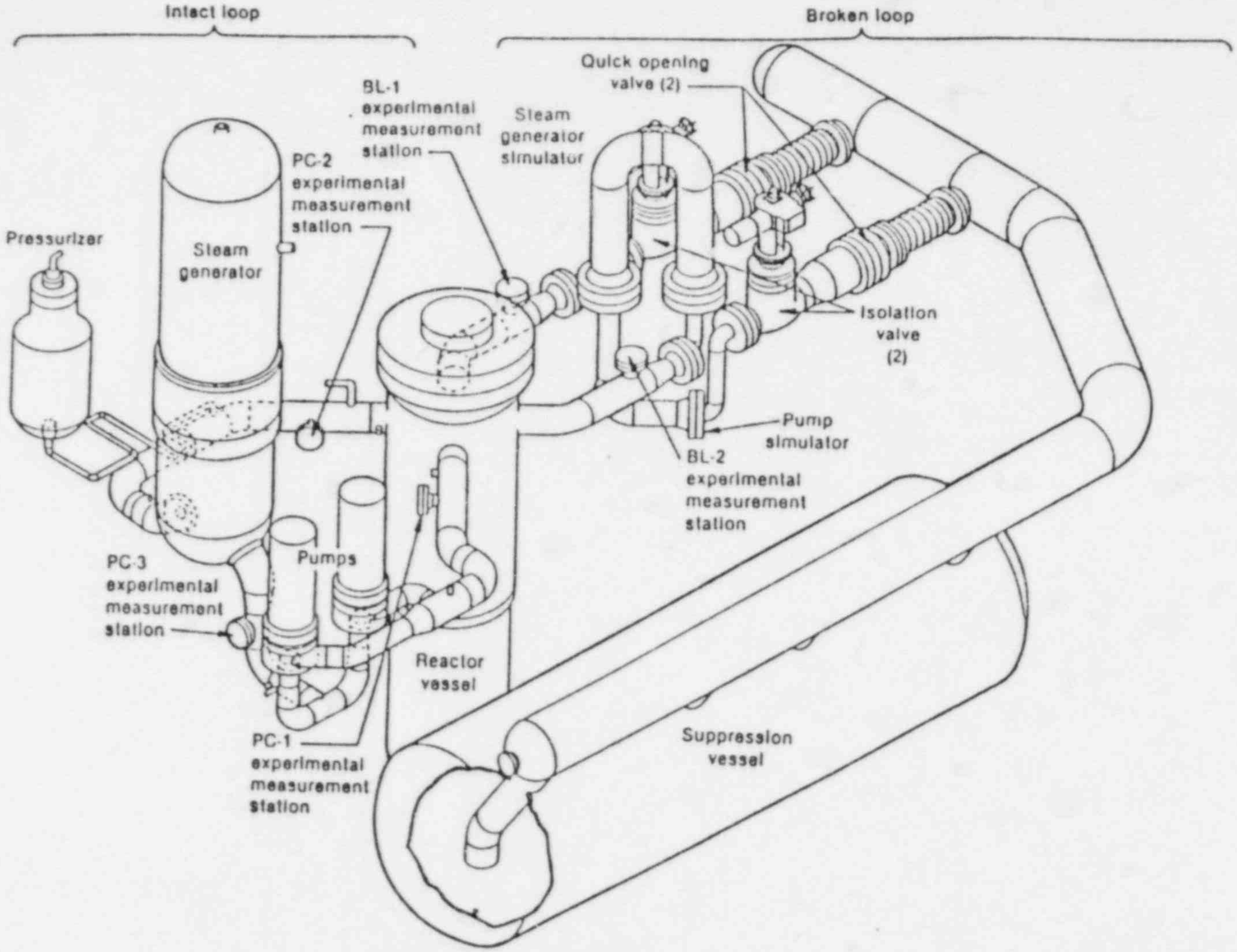
LOFT is a 50 MW thermal scale model which closely models a 3000 MW thermal PWR, major emphasis being on the reactor system, primary coolant system, blowdown system, blowdown suppression system, emergency core cooling system, and secondary cooling system.

The major series of experiments included nonnuclear mini-blowdown experiments, nuclear small-break experiments, nonnuclear large-break

experiments, nuclear intermediate-break experiments, and finally anticipated transients and transients with multiple failures experiments.

LOFT tests L6-1 and L6-3 were analyzed with RETRAN with results being discussed in Sections 6.3 and 6.4. The L6-1 and L6-3 tests are from the Anticipated Transient Experiment Series L6 (40). The objectives of these experiments were to: 1) evaluate the automatic recovery methods in bringing the plant to a hot-standby condition; to 2) provide data to evaluate computer code capabilities to predict secondary system initiated events; for L6-1 to 3) investigate plan response to a transient in which the heat removal capabilities to the secondary system are significantly reduced, and for L6-3 to 4) investigate plant response to a transient in which the heat removal capability of the secondary system is significantly increased.

FIGURE 6.1-1
LOFT FACILITY SCHEMATIC



6.2 LOFT RETRAN Model

The BG&E RETRAN model of the LOFT facility was based on a RETRAN-01 deck used by Energy Incorporated (EI) to simulate the L6-5 test (39). The EI deck was revised to execute under the RETRAN-02 Mod 3 code version and was then modified to simulate the L6-1 and L6-3 tests. A noding diagram is presented in Figure 6.2-1.

Both the unbroken and broken coolant loops are modeled. Since neither the L6-1 nor the L6-3 tests involved a pipe break, the broken loop was modeled simply by using two 1-volume piping stubs connected to the unbroken loop by small circulation lines. In contrast, multiple volumes were used to model the unbroken hot and cold legs.

The hot leg is modeled using four volumes. There is one volume for the hot leg nozzle, two volumes for the 14 inch piping, and one volume for the 16 inch piping. The primary side of the steam generator is modeled using four volumes. One volume is used for the steam generator inlet plenum. There are two volumes representing the steam generator tubes and one volume for the steam generator outlet plenum. The cold leg is modeled using nine volumes. There are three volumes leading to the reactor coolant pump inlet and individual volumes for each pump and its outlet. Two further volumes combine the pump outlets and connect to the reactor vessel.

The reactor vessel model includes a 5 volume core, volumes for the upper and lower plena, and a 2 volume downcomer. The pressurizer vessel is modeled using one volume and the non-equilibrium option. Pressurizer spray is provided by a one volume connection to the cold leg. There is a one volume surge line which connects to the 14 inch piping in the hot leg. The steam generator secondary is modeled to recirculate and uses four volumes. The

feedwater piping has one volume and is connected to a fill junction. The steam line is modeled as with the Main Steam Flow Control Valve (MSFCV) in between two volumes. Steam is exhausted to a time dependent volume acting as the air cooled condenser.

There are a total of 42 volumes and 50 junctions in the system. There are 28 heat slabs used to simulate the metal mass of the system and 3 non-conducting heat exchangers used to simulate the pressurizer heaters. Control systems and trips are provided to simulate the Reactor Protective System logic, the pressurizer pressure control logic, the steam generator pressure control logic, and the ECCS actuation logic. Special controls are provided for each transient to simulate its particular initiator.

FIGURE 6.2-1

LOFT RETRAN NODING DIAGRAM

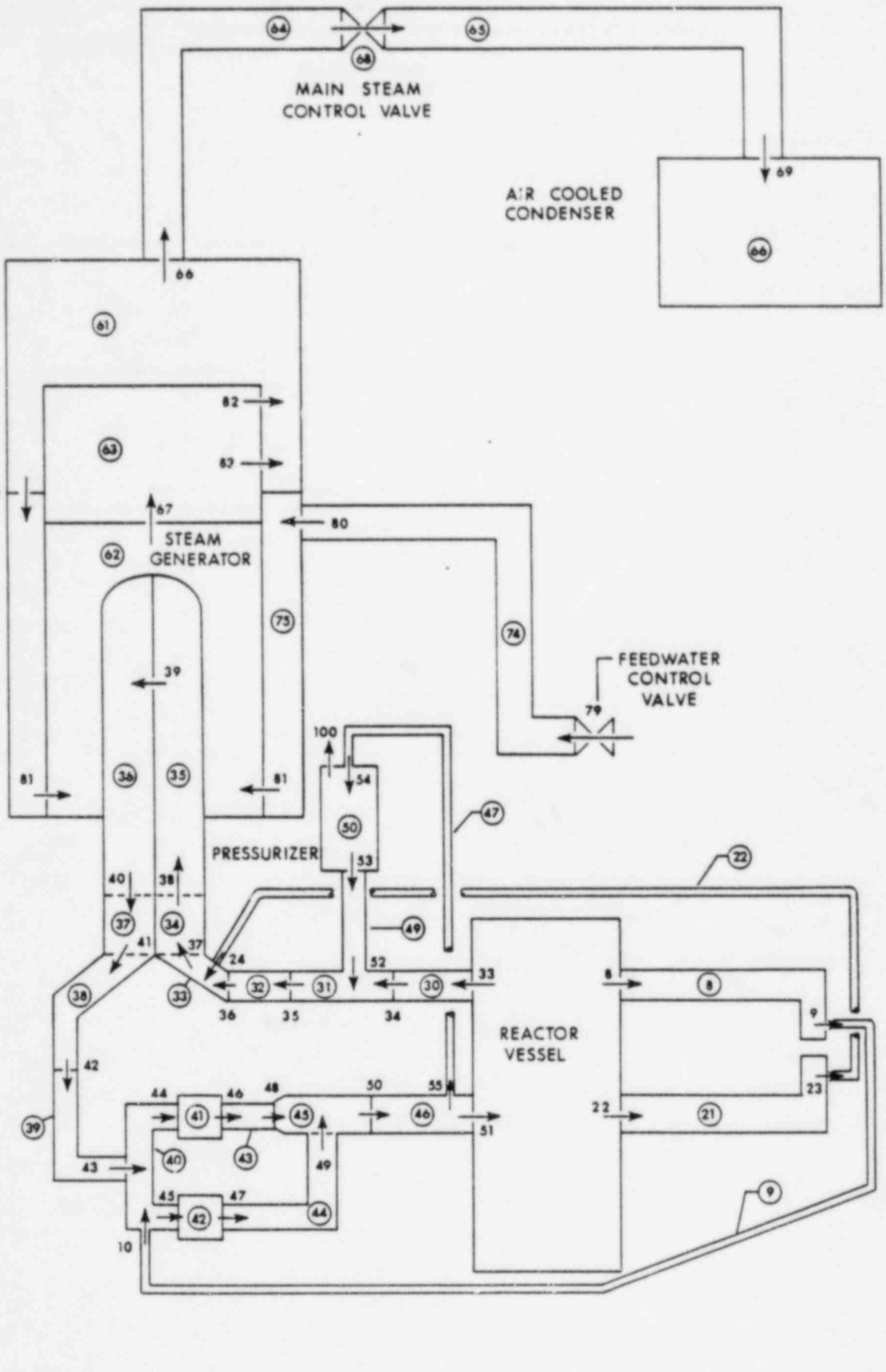
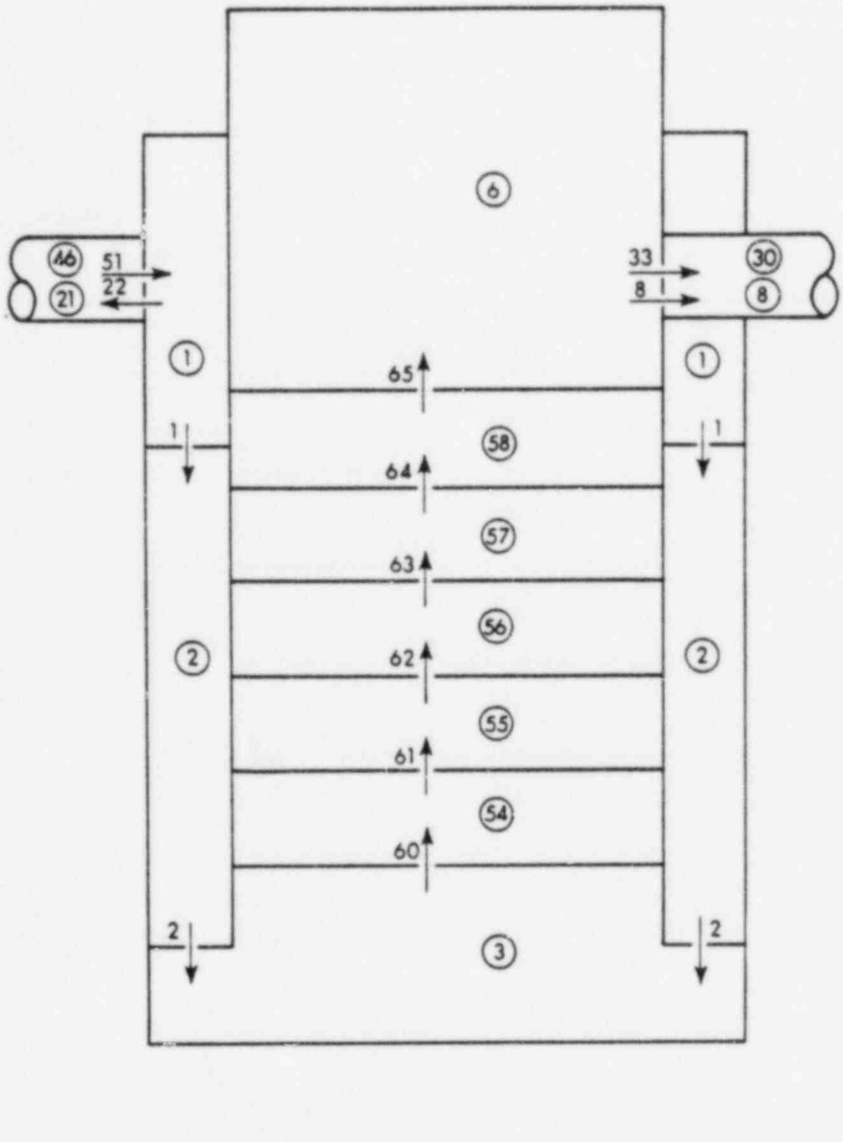


FIGURE 6.2-1 (Cont'd)

LOFT RETRAN NODING DIAGRAM



6.3 LOFT Test L6-1 Comparison

6.3.1 LOFT Test L6-1 Transient Description

Experiment L6-1 was a loss of steam load anticipated transient and was performed on October 8, 1980 (40). The experiment was initiated from a power level of 36.9 MW by closing the Main Steam Flow Control Valve (MSFCV) at its maximum rate, resulting in reduced steam flow and degraded heat transfer between the primary and secondary. The primary pressure increased until 9.1 seconds when pressurizer spray was initiated resulting in a small and short-lived pressure decrease. The primary pressure began to increase again, causing the reactor to scram at 21.8 seconds. The MSFCV then began to open to reduce secondary side pressure. As primary pressure followed the decreasing core power, spray was terminated at 30.4 seconds and the pressurizer backup heaters were activated at 32.5 seconds. The MSFCV opened and closed automatically at 91.2 seconds and 104.4 seconds, and was manually opened and closed at 312.6 seconds and 339 seconds to regulate secondary pressure. The pressurizer backup heaters were shut off at 415.4 seconds. The experiment was terminated at 700 seconds (40).

6.3.2 RETRAN Modeling for L6-1 Transient Simulation

The model described in Section 6.2 was used to analyze LOFT Test L6-1 without modifications.

6.3.3 Comparison of RETRAN Results to LOFT Test Data

The response of the LOFT facility to the L6-1 test initiator is described in detail in Reference 40. Data from that document have been plotted with our RETRAN results and are presented here as Figures 6.3-1 through 6.3-5. Only the first 200 seconds of the transient were simulated. Beyond 200 seconds, the plant was in a recovery mode which was dominated by the operators. All important thermal-hydraulic phenomena occurred during the first 200 seconds.

The RETRAN and test recorded event sequences have been tabulated and are presented as Table 6.3-1. There is generally very good agreement in both the sequence and timing of events. RETRAN predicts an earlier reactor trip, causing some RETRAN events to occur earlier than the corresponding L6-1 events. More detail on the significance of the event sequence is given in the discussions of the Figures.

Figure 6.3-1 shows the pressure response in the steam generator dome. The RETRAN predicted pressure compares well with measured data. The RETRAN peak pressure is somewhat lower because less energy is deposited in the secondary due to the early scram. The second pressure peak is lower for the same reason. RETRAN did not predict the reopening of the MSFCV at 91.2 seconds because not enough energy had been transferred to the secondary side to cause the pressure to exceed the reopening setpoint. It appears from the slowly rising pressure that, further out in time, RETRAN will calculate a reopening of the MSFCV and exhibit the second dip and climb in steam dome pressure.

Figure 6.3-2 shows the steam generator level response (the LOFT wide range indication was used). The RETRAN predicted level response falls

somewhere between that of the LOFT wide range (shown) and narrow range (not shown) measurements. While RETRAN does predict the trend and magnitude of the falling level fairly well, whenever the MSFCV is closed, RETRAN does a poor job. This is an artifact of the secondary model. The level is calculated as an equivalent liquid level summing volumes 61 (the steam dome) and 75 (the downcomer). After the MSFCV closes, the pressure builds up in the steam line (volume 64) and a reverse flow is calculated from the steam line to the steam dome. This results in a surge in the equivalent liquid level. As the pressure equilibrates, the level returns to normal. Thus, during the secondary transients caused by the cycling of the MSFCV at 11.6 and 43.2 seconds, RETRAN tends to swing about the true level. Once the secondary pressure stabilized, the RETRAN predicted level closely follows test data.

Figure 6.3-3 shows the pressurizer pressure response. RETRAN correctly predicts the initial pressure increase to the pressurizer spray setpoint. Once initiated, RETRAN calculated a slight depressurization due to the spray flow, but the full effectiveness of the spray in reducing the pressure was not seen. Ultimately, the spray could not check the increasing pressure due to level swell, and a scram on high pressure was reached. The scram on high pressure resulted in decreased core heat production and the reopening of the MSFCV. The increased heat transfer to the steam generator along with the decrease in core heat production lead to a swift depressurization of the primary that did not stop until the MSFCV was again shut. RETRAN predicted a greater depressurization than was measured for two related reasons. First, the RETRAN scram is 4 seconds earlier than the LOFT scram, depriving the primary of a significant amount of energy needed to maintain pressure. Second, RETRAN calculates the MSFCV to remain open more than

7 seconds longer than was measured - resulting in a significant amount of energy removal from the primary. Once the MSFCV was closed, RETRAN calculated an increase in pressure consistent with that seen in the LOFT data, but again with the exception of the depressurization caused by the unpredicted cycling of the MSFCV at 91.2 seconds.

Figure 6.3-4 shows the pressurizer level response (the LOFT data is not density compensated). The RETRAN pressurizer level response agrees well with the LOFT data. After scram, the RETRAN level decreases more than the LOFT measured level because, as discussed for Figure 6.3-3, there is less energy in the primary system to keep temperature and specific volume up.

Figure 6.3-5 shows the primary temperature response at the inlet plenum to the steam generator. The RETRAN results are well within the uncertainty band of the data ($\pm$ 10.62 ^{o}F). The RETRAN calculated temperature falls off faster than the measured data due to two factors: the early reactor trip calculated by RETRAN, and the response delay time of the LOFT temperature measurements.

6.3.4 RETRAN Sensitivity Study for LOFT Test L6-1

Two parameters were varied to study their effect on overall pressurizer pressure response in the simulation of L6-1. These parameters were the bubble rise velocity (VBUB) and the interface heat transfer coefficient (IHTC).

The overall pressurizer performance was found to be more predictable and stable when IHTC was the parameter being varied rather than VBUB. For the L6-1 transient, a pressurizer insurge followed by an outsurge with backup heaters on at time zero, it was critical that the IHTC be a very high value.

The IHTC actually used in the RETRAN simulation was 27,500 BTU/HR-FT^2-F. Lower values for IHTC resulted in either a too rapid pressurization during the insurge portion of the transient or an underprediction of the value of the pressurizer spray or both. When a low value of IHTC, like 10 BTU/HR-FT^2-F, is used in concert with a low value of VBUB like 0.6 F/sec (3.0 F/sec is normally recommended), there is some compensating effect on the pressure response to the insurging level. This may be due to the slower transport of the steam bubbles generated by the backup heaters to the pressurizer steam/liquid interfacing region. Once the backup heaters are turned off, the IHTC dominates. The more accurate LOFT RETRAN model uses a high value for IHTC and the normal value of 3.0 FT/sec. for VBUB. The values used for this transient, 27,500 BTU/HR-FT^2-F and 3.0 FT/sec., provide good agreement with the LOFT data.

6.3.5 Summary of Results

A simulation of the LOFT test L6-1 was performed using the RETRAN 02 Mod 3 computer code. Figures showing comparisons of RETRAN calculated and LOFT test measured data have been presented for the pressure in the steam generator dome, the steam generator level, the pressure in the pressurizer, the level in the pressurizer, and the temperature of the primary coolant in the steam generator outlet plenum. In every case, RETRAN calculated both the general trends and the relative magnitudes of the data within acceptable bounds. A sensitivity study was conducted to determine the best value of IHTC and VBUB in the pressurizer for this transient.

Table 6.3-1
Sequence of Events for Experiment L6-1*

Event	Time After Experiment Initiation (seconds)	RETRAN Time (seconds)
MSFCV closing initiated	0.0	0.0
Pressurizer backup heaters off	6.1 ± 0.1	7.07
Pressurizer spray on	9.1 ± 0.1	9.47
MSFCV closed	11.6 ± 0.2	11.61
Reactor scrammed	21.8 ± 0.2	17.67
Maximum PCS pressure reached	22.0 ± 0.2	18
MSFCV opened	22.2 ± 0.2	18.61
$Q_{SG} > Q_{core}$**	26.5 ± 0.5	25
Pressurizer spray off	30.4 ± 0.1	21.46
Pressurizer backup heaters on	32.5 ± 0.1	23.8
MSFCV closed	40.6 ± 0.2	43.23
MSFCV opened	91.2 ± 0.2	***
MSFCV closed	104.4 ± 0.2	***

* Only the first 200 seconds are presented.

** Pressurizer liquid level at maximum.

*** This event not predicted by RETRAN.

FIGURE 6.3-1

LOFT TEST L6-1

PRESSURE IN STEAM GENERATOR DOME

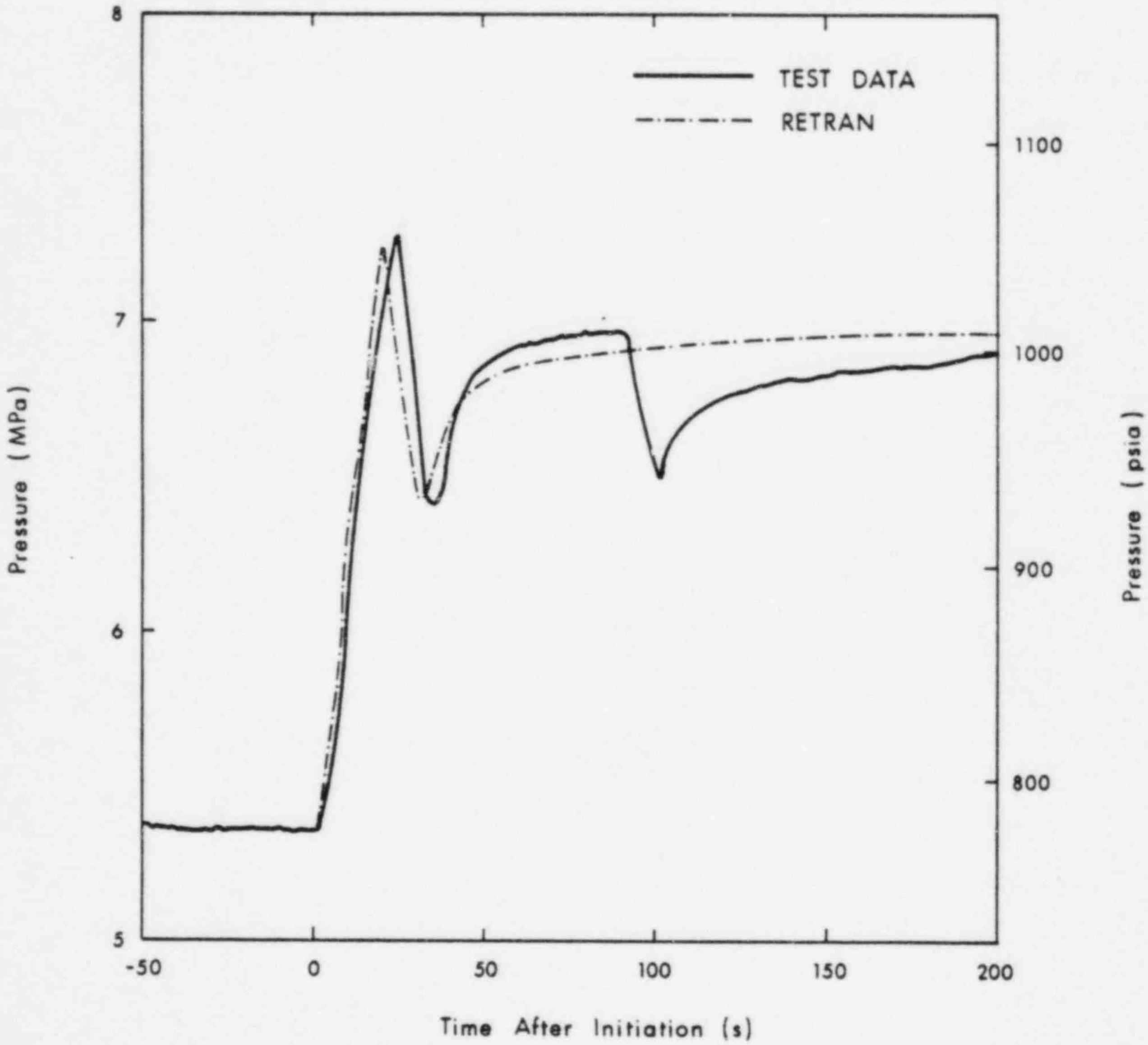


FIGURE 6.3-2

LOFT TEST L6-1

LIQUID LEVEL IN STEAM GENERATOR SECONDARY SIDE

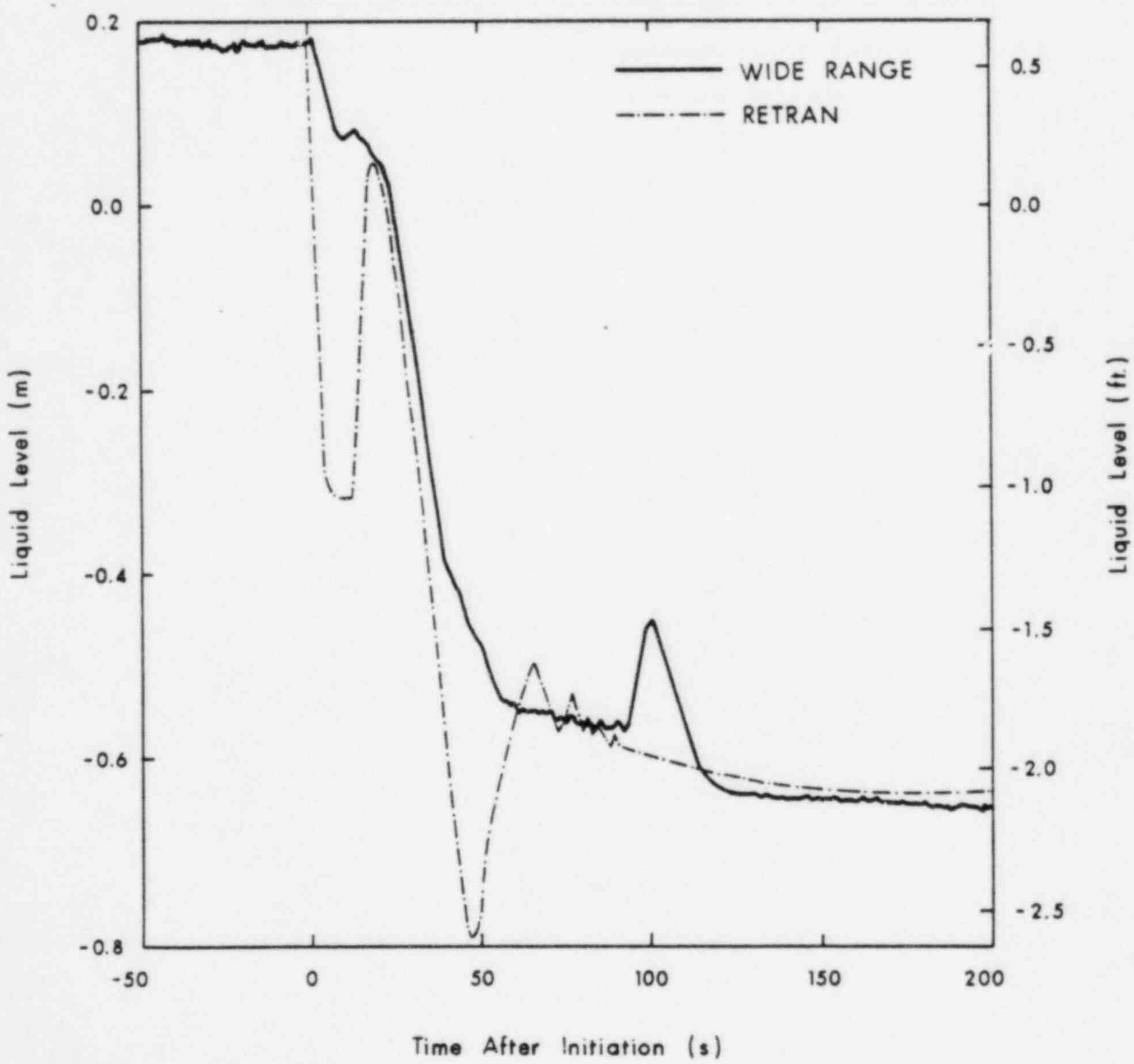


FIGURE 6.3-3

LOFT TEST L6-1

PRESSURE IN PRESSURIZER

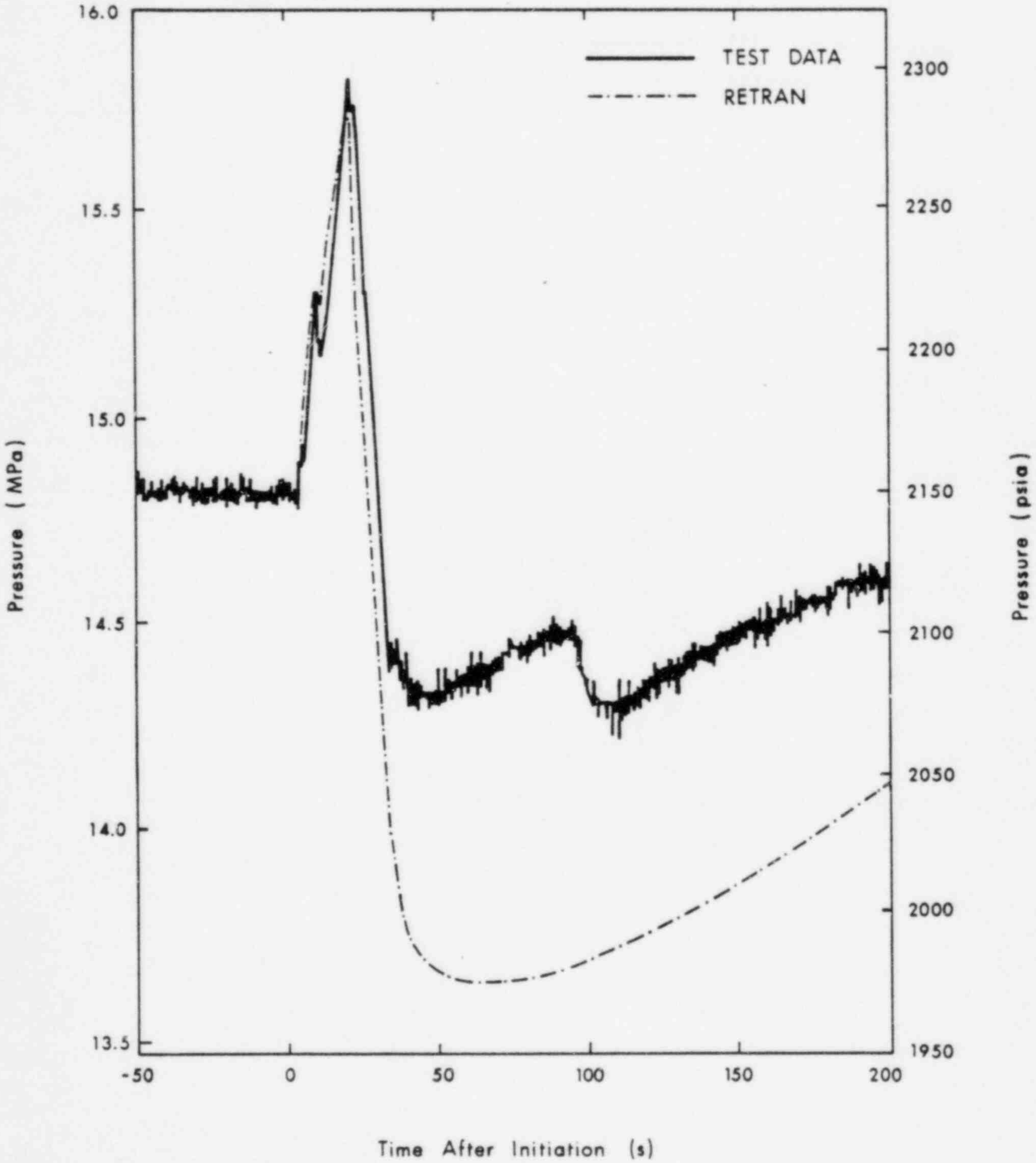


FIGURE 6.3-4

LOFT TEST L6-1

LIQUID LEVEL IN PRESSURIZER

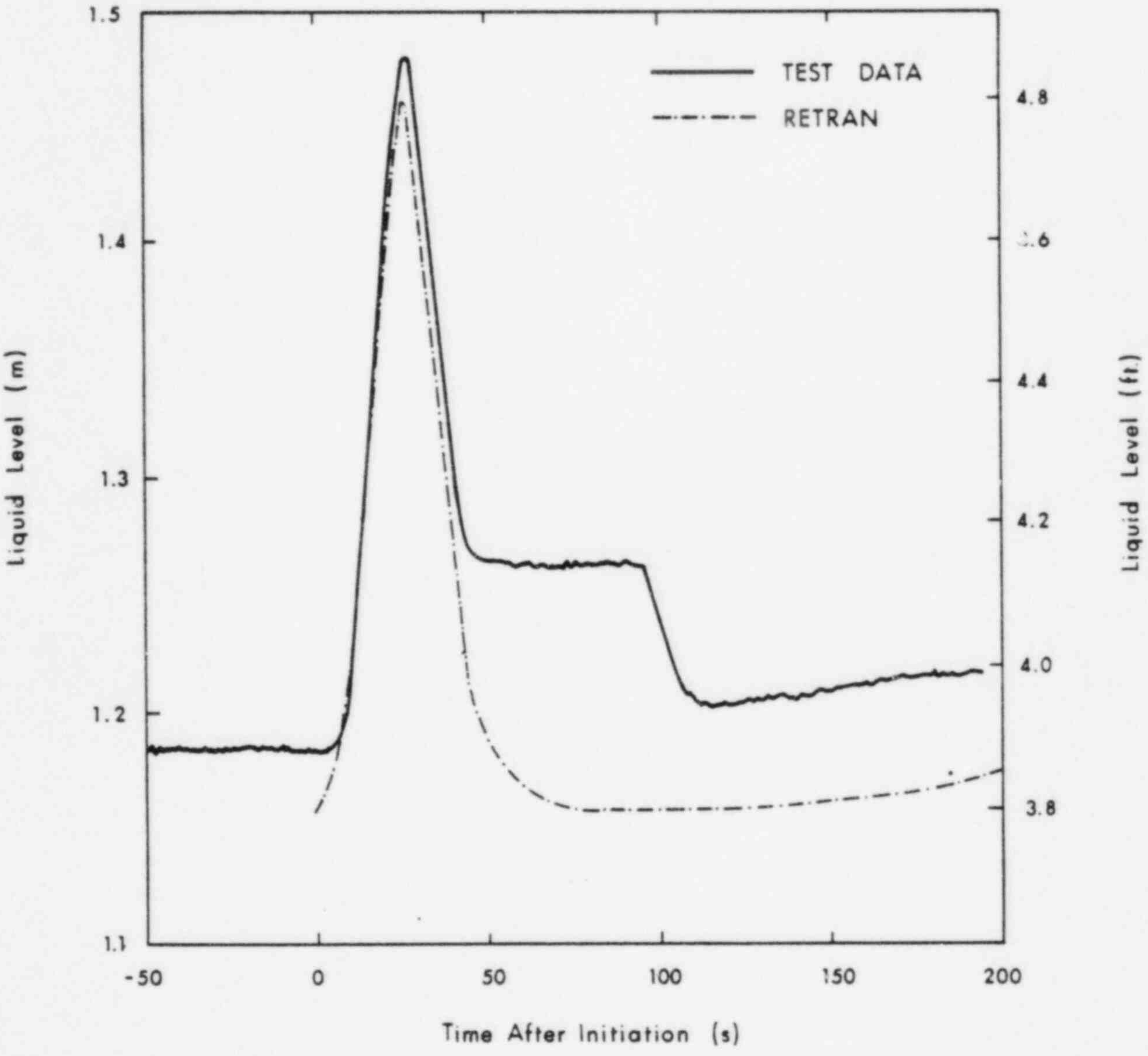
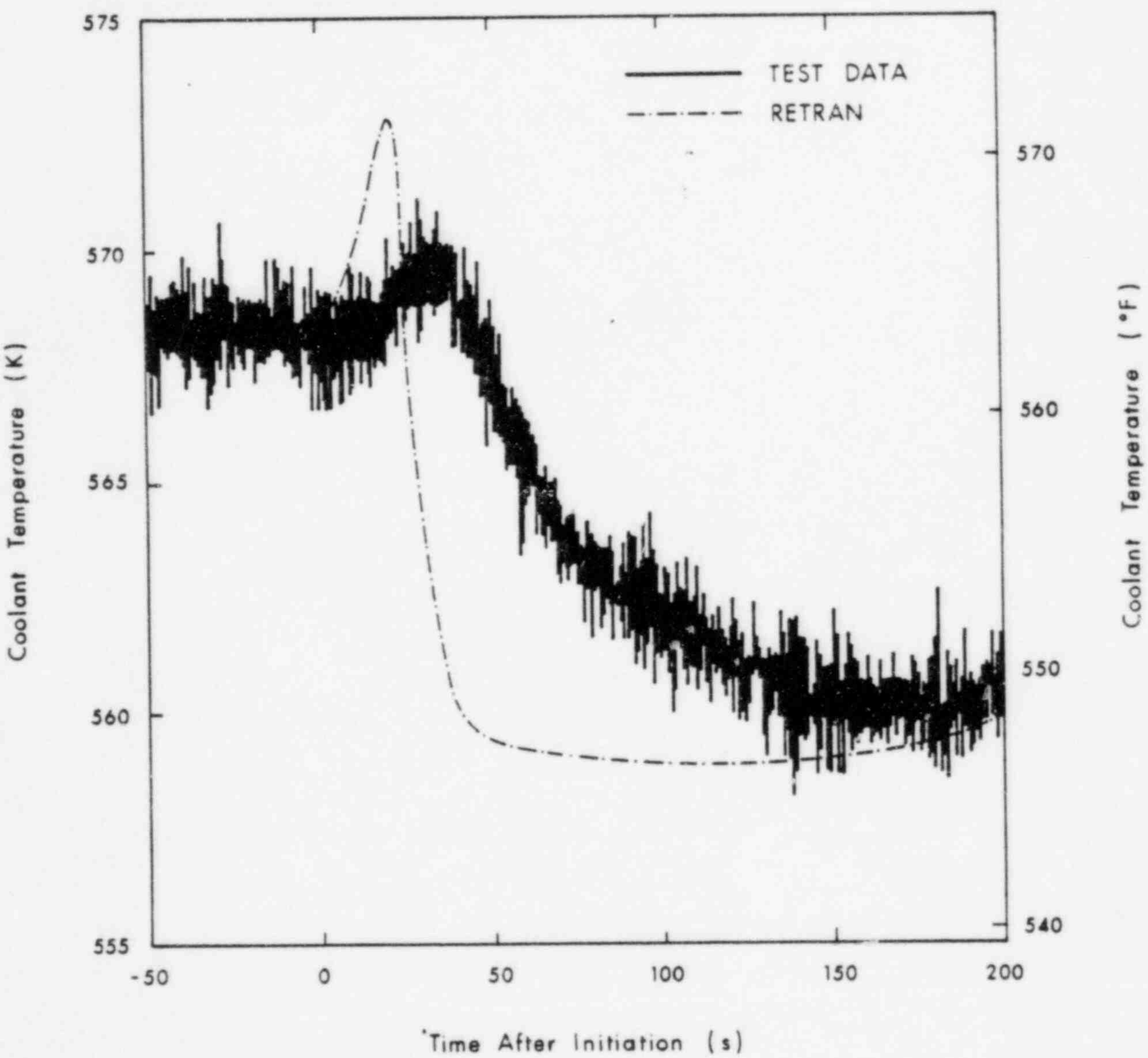


FIGURE 6.3-5

LOFT TEST L6-1

COOLANT TEMPERATURE IN INTACT LOOP

STEAM GENERATOR PRIMARY SIDE INLET PLENUM



6.4 LOFT Test L6-3 Comparison

6.4.1 LOFT Test L6-3 Transient Description

Experiment L6-3 was an excessive load increase anticipated transient and was performed on October 9, 1980 (40). The experiment was initiated from a power level of 36.9 MW by ramping the MSFCV open at its maximum rate. Secondary steam flow increased and caused a cooldown in the primary coolant system. The cooldown and its associated primary coolant density increase caused reactor power to increase and pressurizer level to decrease. The decreasing level caused Primary Coolant System (PCS) pressure to drop and the reactor to trip on low pressure at 15.6 seconds. Pressure continued to decrease and, at about 26.5 seconds, the HPSI pumps began to inject emergency core coolant into the cold leg. The PCS pressure reached a minimum at 26.8 seconds. The MSFCV was closed at 36.2 seconds. With core heat production exceeding secondary heat removal at 33 seconds, the PCS temperature and pressure began to rise. At about 48.6 seconds, the HPSI pumps were shut off by the operators. Pressurizer cycling heaters shut off at 154.9 seconds.

To maintain pressurizer liquid temperature, the pressurizer backup heaters were turned on by the operators at 536.8 seconds. The operators turned the cycling heaters on at 554.0 seconds. The backup heaters and cycling heaters were turned off by the operators at 564.6 seconds and 565.4 seconds, respectively. The operators turned the backup heaters on again at 600.2 seconds. The experiment was terminated at 700 seconds.

6.4.2 RETRAN Modeling for L6-3 Transient Simulation

The base LOFT deck was set up for the L6-1 transient. Few modifications were required to simulate the L6-3 test. The most prominent modification was the addition of a High Pressure Safety Injection (HPSI) system. This was modeled simply by connecting two fill junctions to the cold leg and regulating their flow with a pressure dependent trip signal.

6.4.3 Comparison of RETRAN Results to LOFT Test Data

The response of the LOFT facility to the L6-3 test initiator is detailed in Reference 40. Data from that document have been plotted with our RETRAN results and are presented here as Figures 6.4-1 through 6.4-5. Only the first 200 seconds of the transient were simulated. All important thermal-hydraulic phenomena occurred during that time period.

The RETRAN calculated and test recorded event sequences have been tabulated and are presented as Table 6.4-1. There is generally good agreement in both the sequence and timing of significant events. RETRAN predicts an earlier reactor trip on low pressure in the hot leg, causing all subsequent RETRAN events to occur earlier than the corresponding L6-3 events. More detail on the significance of the event sequence is given in the discussion of the Figures.

Figure 6.4-1 shows the pressure response in the steam generator dome. RETRAN correctly calculated both the rate and timing of the depressurization due to the opening of the MSFCV. The magnitude of the depressurization was underpredicted, but that can be attributed to the earlier RETRAN reactor trip signal and the resulting earlier closure of the MSFCV.

As the MSFCV closes, the steam flow decreases and the secondary pressure increases. The RETRAN repressurization curve rises more gently and steadies at a lower pressure than the LOFT data. This behavior is partly because less energy is being deposited in the secondary due to the early scram, but mostly it is due to a somewhat higher terminal steam flow and feedwater flow in the PETRAN calculation.

Figure 6.4-2 shows the steam generator level response (the LOFT wide range indication was used). RETRAN calculated both the trends and magnitudes of the level response reasonably well with the exception of a large level swing calculated to occur at 48.5 seconds. As explained in Section 6.3.3, this swing is an artifact of the secondary model. The level is calculated as an equivalent liquid level summing volumes 61 (the steam dome) and 75 (the downcomer). After the MSFCV closes, the pressure builds up in the steam line (volume 64) and a reverse flow is calculated from the steam line to the steam dome. This results in a surge in the equivalent liquid level. As the pressure equilibrates, the level returns to normal.

Figure 6.4-3 shows the pressurizer pressure response. RETRAN correctly predicts the initial pressure decrease due to the rapid cooldown and level shrink. The rate of depressurization is somewhat less than in the LOFT test because the RETRAN pressurizer was initialized at a slightly lower pressure (2143 psi). This pressure is below the setpoint for the backup heaters and so they came on at time zero. The energy delivered by the heaters in the first 10 seconds before the heaters came on in the LOFT test helped keep the pressure up. The additional heater input also contributed to the RETRAN minimum pressure being higher than the test minimum. Another reason for the higher minimum pressure is that the RETRAN HPSI pumps deliver flow (on low pressure in the cold leg) about 8 seconds earlier. The added flow

helps to maintain pressurizer level and keep the pressure high. The last effect worth noting is that the pressure does not recover to the same level as found in the test data. In fact, the pressure does not quite reach the setpoint to reset the pressurizer heaters. The reason is the same as that given for the steam generator pressure recovery. A combination of decreased core power input and higher energy removal through the secondary (caused by increased steam flow and feedwater flow over that measured in the test) retards the peak pressure level.

Figure 6.4-4 shows the pressurizer level response. The RETRAN pressurizer level response decreased at the same rate and reached a minimum at about the same time as measured in the experiment. The RETRAN minimum level is greater than the LOFT minimum because of the additional HPSI flow (as discussed for the pressurizer pressure). The RETRAN level does not recover to the same level as in the test because there is less expansion of the primary coolant due to a lessened primary side heatup. This is due to more energy being transferred to the secondary side and less overall heat input from the core.

Figure 6.4.-5 shows the primary temperature response at the inlet plenum to the steam generator. RETRAN predicts the correct trend and the magnitude is within the uncertainty band of the data ($\pm$ 10.62 ^{O}F). The steeper slope of the RETRAN cooldown can be attributed primarily to the response delay time of the LOFT temperature measurements.

6.4.4 RETRAN Sensitivity Study for LOFT Test L6-3

The two sensitivity studies performed for LOFT Test L6-3 examined various values of the IHTC, and the effect of HPSI delivery to both the cold leg and the hot leg as opposed to delivery to just the cold leg.

For the IHTC investigation, the IHTC was varied from 10 to 30,000 BTU/HR-FT^2-F with a broad range of effects on both the pressurizer pressure and temperature. The value used for the final analysis was 400 BTU/HR-FT^2-F. For large values of IHTC, greater than about 6000 BTU/HR-FT^2-F, there is excessive heat transfer in the pressurizer and temperatures are depressed. A small value of IHTC will reduce the heat transfer in the pressurizer, thus leading to repressurization rates and pressures that are consistent with the observed data for a transient in which there is an initial outsurge, followed by an insurge into the pressurizer. Values which are too small, less than about 10 BTU/HR-FT^2-F, lead to an overprediction of the repressurization rate and pressures.

For the HPSI delivery location study, it was found that injecting cold HPSI flow into the hot leg volume upstream of the pressurizer caused excessive depressurization because some of the cold HPSI water entered the pressurizer. Injecting both the A and B HPSI pumps into the cold leg allowed the core to heat the water before entering the hot leg. This latter method was used in both the LOFT test and the RETRAN simulation.

6.4.5 Summary of Results

A simulation of the LOFT test L6-3 was performed using the RETRAN 02 Mod 3 computer code. Figures showing comparisons of RETRAN calculated and LOFT test measured data have been presented for the pressure in the steam generator dome, the steam generator level, the pressure in the pressurizer, the level in the pressurizer, and the temperature of the primary coolant in the steam generator and outlet plenum. In every case, RETRAN calculated both the general trends and the relative magnitudes of the data

within acceptable bounds. A sensitivity study was conducted to determine the best value of pressurizer IHTC and HPSI injection location for this transient.

Table 6.4-1

Sequence of Events for Experiment L6-3*

Event	LOFT Time	RETRAN Time
MSFCV Opening	0.0	0.0
Feedwater Flow Increased	1.4 ± .2	0.0**
Pressurizer Backup Heaters On	10.2 ± .1	0.0
Maximum Reactor Power	15.6 ± .2	14.18
Reactor Scrammed	15.6 ± .2	14.18
Feedwater Flow Terminated	16.6 ± .2	15.0
MSFCV Start Closing	17.8 ± .2	15.0
HPSI Pump A On	26.4 ± .2	18.5
HPSI Pump B On	26.6 ± .2	18.5
Minimum PCS Pressure Reached	26.8 ± .2	30.5
$Q_{SG} < Q_{core}$ ***	33 ± 1	32.25
MSFCV Closed	36.2 ± .2	34.00
HPSI Pump A Off	48.6 ± .2	44.14
HPSI Pump B Off	50 ± 2	44.14
Pressurizer Backup Heaters Off	105.4 ± .1	****
Pressurizer Cycling Heaters Off	154.9 ± .1	****

* Only events occuring in the first 200 seconds of the LOFT Test are presented.

** RETRAN Feedwater was delivered by a fill junction designed to match the LOFT data.

*** Pressurizer liquid level at minimum.

**** RETRAN did not predict this event.

FIGURE 6.4-1

LOFT TEST L6-3

PRESSURE IN STEAM GENERATOR DOME

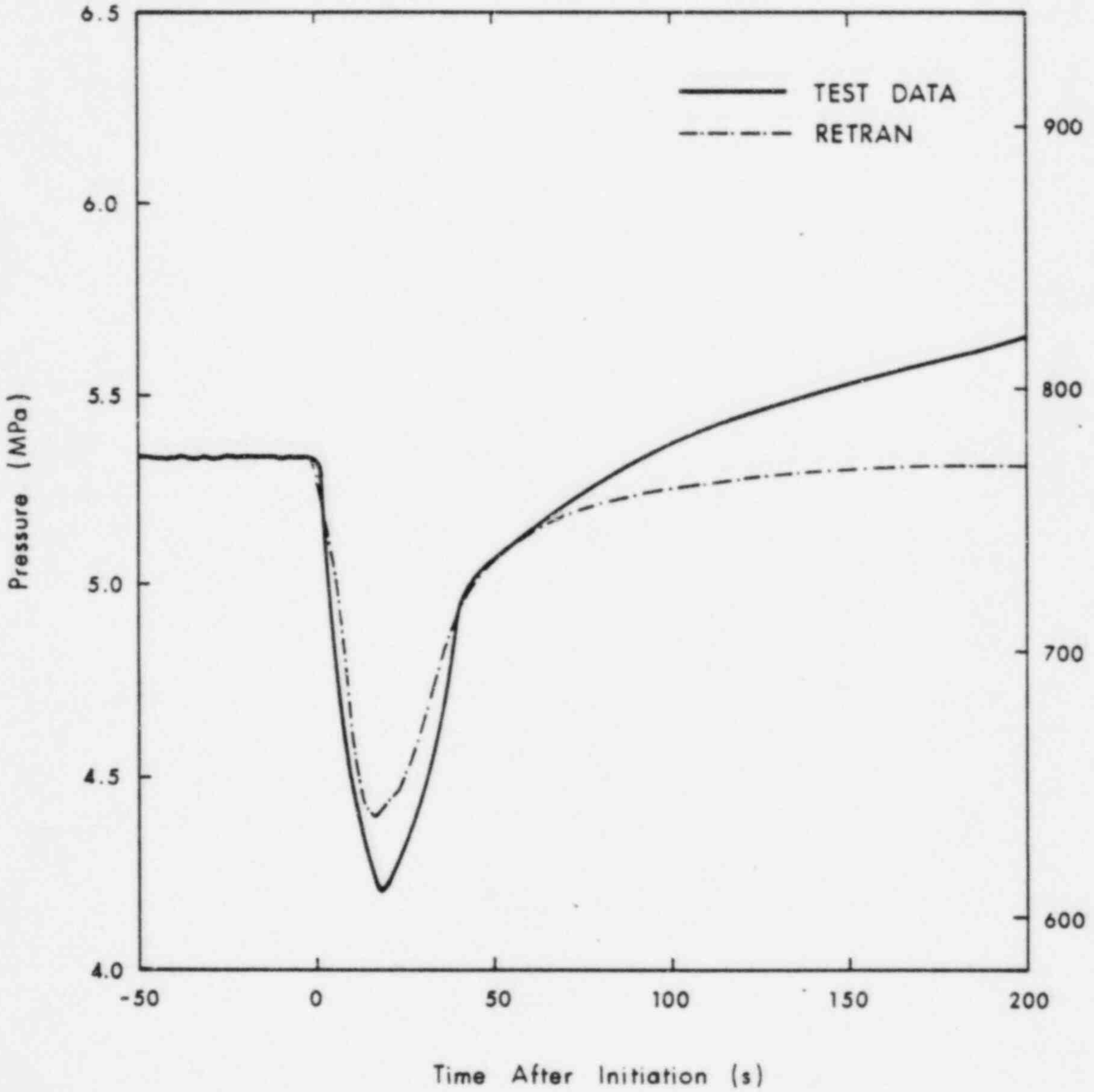


FIGURE 6.4-2

LOFT TEST L6-3

LIQUID LEVEL IN STEAM GENERATOR SECONDARY SIDE

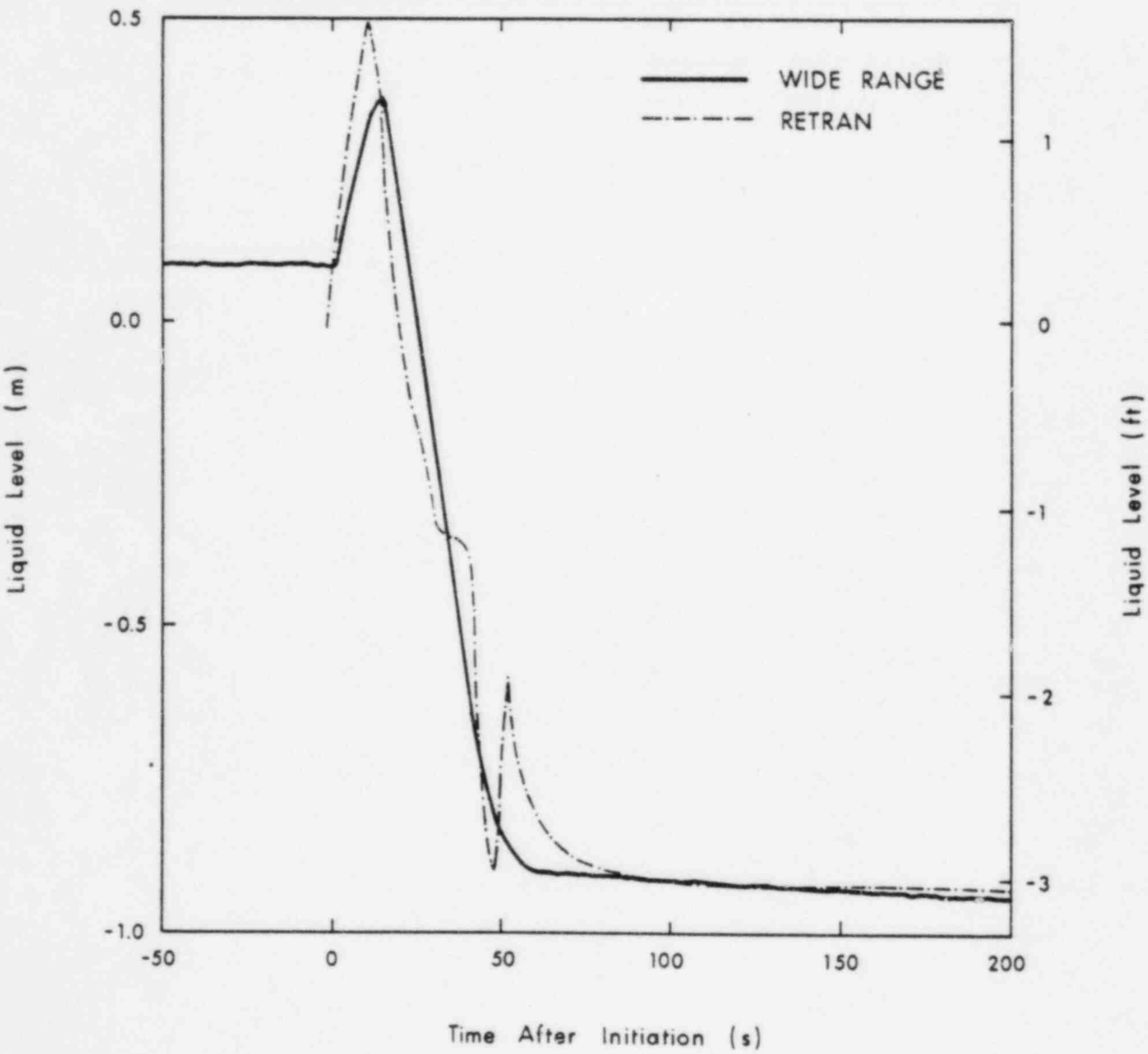


FIGURE 6.4-3

LOFT TEST L6-3

PRESSURE IN PRESSURIZER

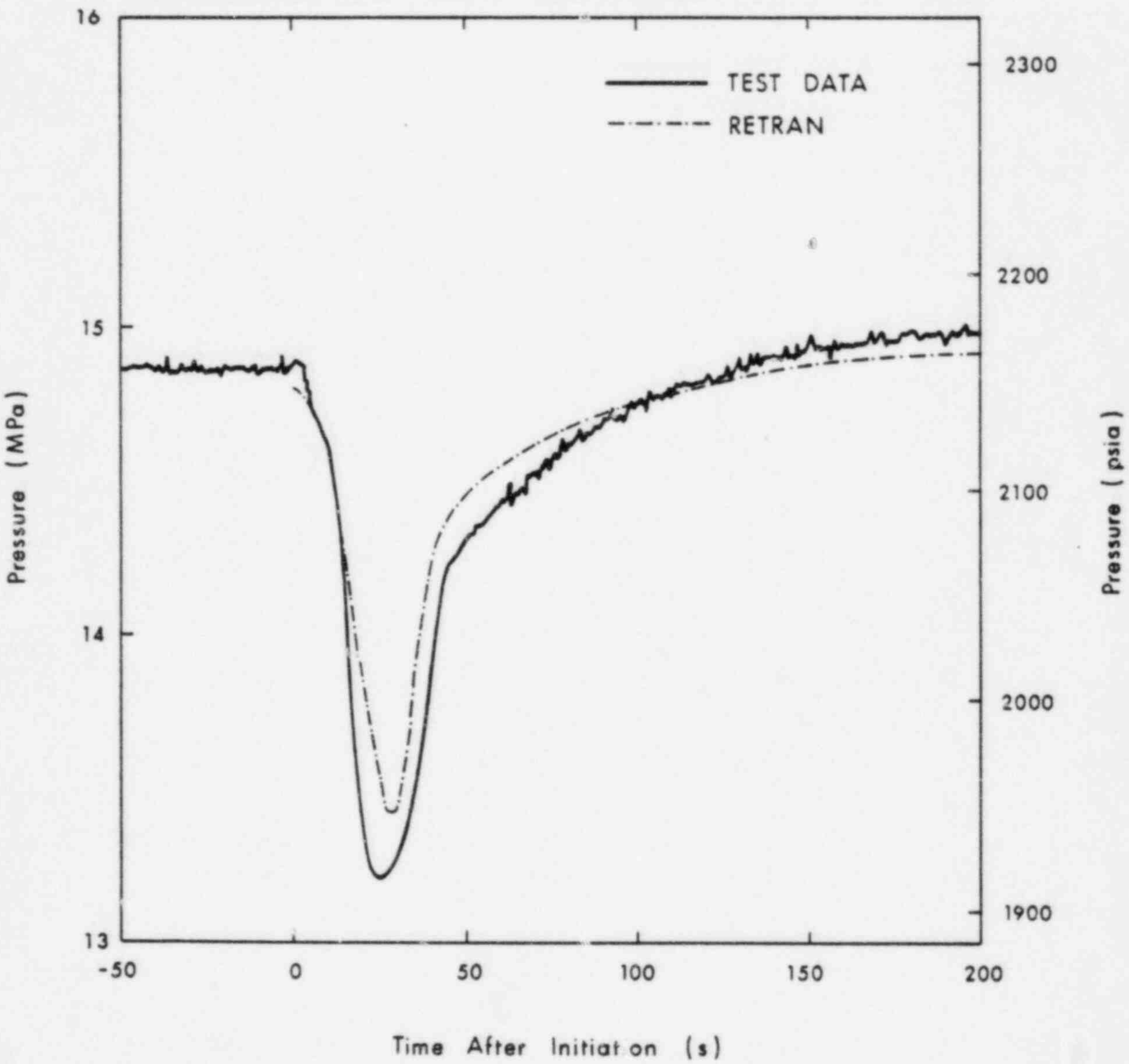


FIGURE 6.4-4

LOFT TEST L6-3

LIQUID LEVEL IN PRESSURIZER

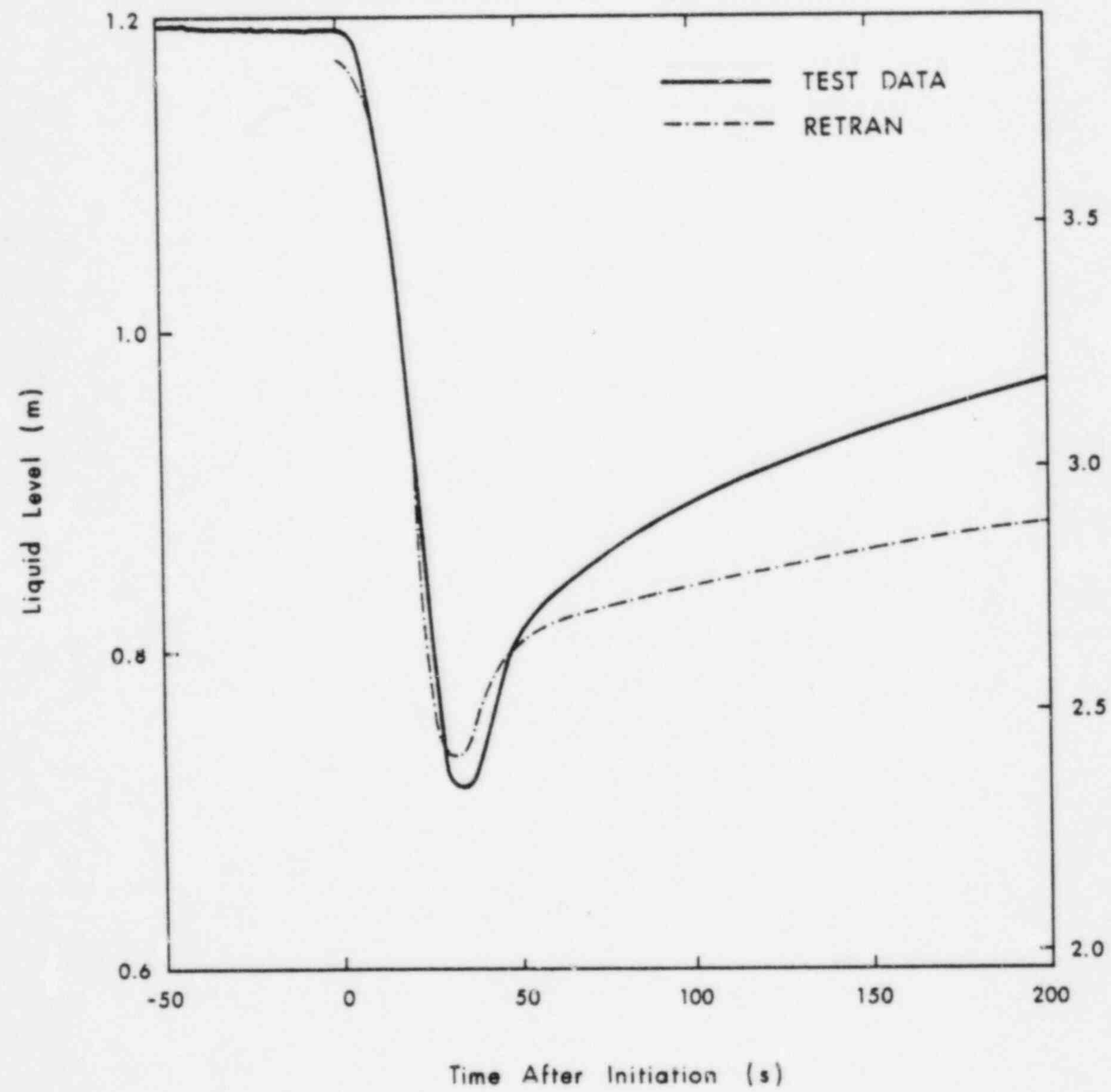
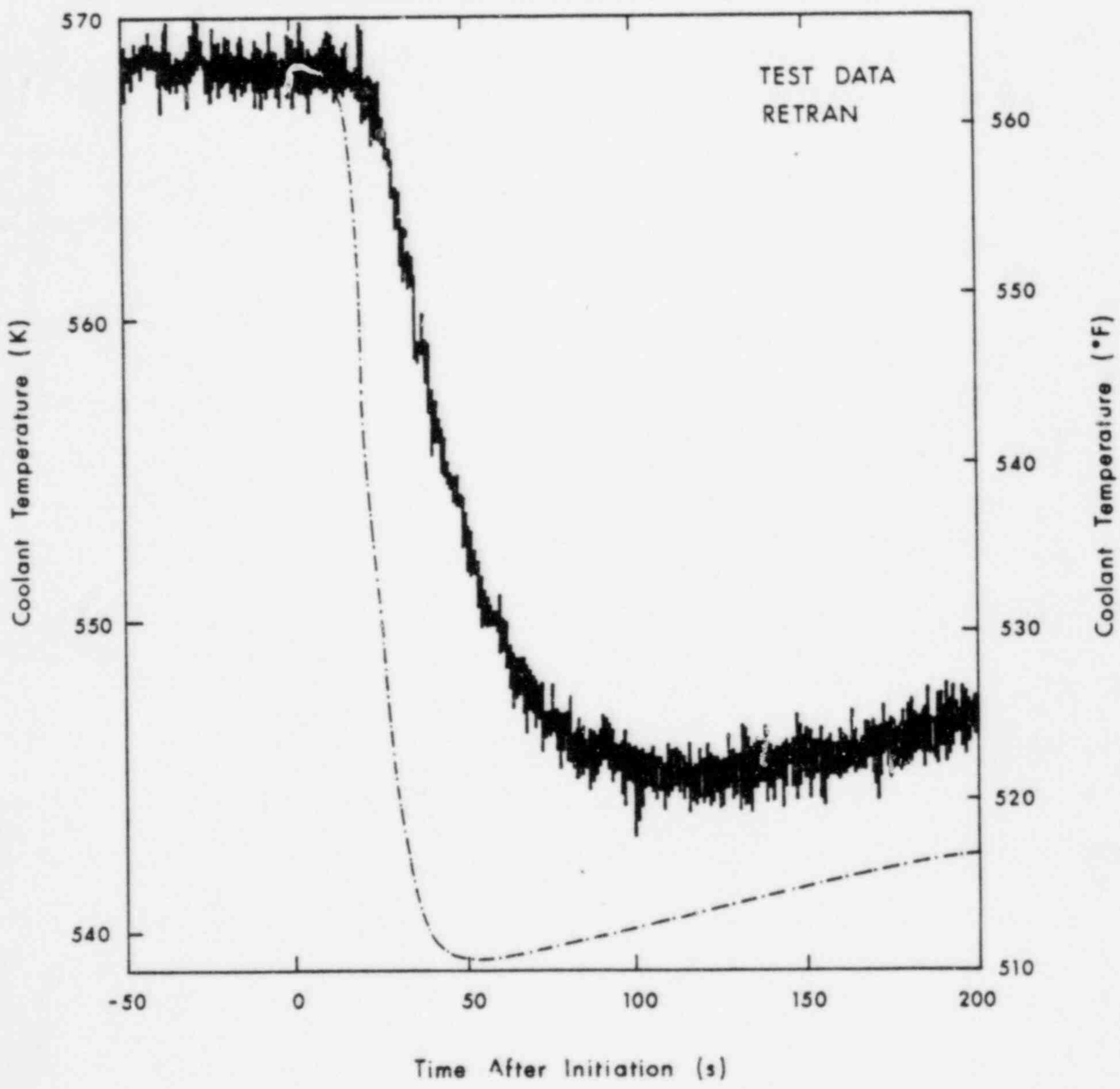


FIGURE 6.4-5

LOFT TEST L6-3

COOLANT TEMPERATURE IN INTACT LOOP
STEAM GENERATOR PRIMARY SIDE INLET PLENUM



7.0 SUMMARY AND CONCLUSIONS

In response to NRC Generic Letter 83-11 and as part of the Inhouse Safety Analysis Capability Development Project, BG&E has presented the results of nine analyses of Calvert Cliffs and the LOFT Facility with the RETRAN computer code in this topical report. These RETRAN calculations were compared to measured plant transient data, calculations performed with the TRAC advanced best estimate computer code, and LOFT test data.

The matrix of the nine RETRAN analyses is delineated and cross-referenced in Table 7-1 to the applicable report section. The key simulation parameters; i.e, type of model used, transient time, and initial power level are presented for these nine analyses in Table 7-2. This table shows that each of the three Calvert Cliffs RETRAN models and the LOFT RETRAN model are utilized in at least two different analyses. Furthermore, simulation times of zero (steady state) to 192 minutes were required for the calculations in this report. Finally, these RETRAN models were initiated from 0%, 20%, 40%, 74%, 80% and 100% of full power.

Table 7-3 delineates specific thermal-hydraulic phenomenology evaluated and tested in the nine RETRAN analyses of this report. The BG&E CCNPP RETRAN model components verified in these analyses include: Reactor Coolant Pump, Steam Generator, Primary Coolant Loop, Pressurizer, ADV, TBV, PORV, SRV, HPSI, and Reactor Vessel. The RETRAN Calvert Cliffs model enhancements, refinements and insights which resulted from the analyses documented in this report are presented in Table 7-4.

This topical report has demonstrated BG&E's capability to model the Calvert Cliffs Nuclear Power Plant with the RETRAN computer code. BG&E has verified the accuracy of these RETRAN models by performing best

estimate analyses of Calvert Cliffs and LOFT transient thermal-hydraulic behavior. This process has served to validate our models while substantiating the capabilities of BG&E RETRAN code users.

Table 7-1

RETRAN ANALYSIS MATRIX

REPORT SECTION	TRANSIENT MODELED WITH RETRAN
4.1	CCNPP 2 Multiple secondary side malfunction event
4.2	CCNPP1 Four reactor coolant pump coastdown test
4.3	CCNPP1 Reactor coolant pump combination flow test
4.4	CCNPP1 One reactor coolant pump coastdown test
4.5	CCNPP1 Total loss of flow - natural circulation test
5.2	CCNPP Cooldown to RHR entry with ADVs and APS TRAC calculation
5.3	CCNPP Runaway main feedwater to one steam generator TRAC calculation
6.3	LOFT test L6-1 reduced secondary side heat removal
6.4	LOFT test L6-3 excessive secondary side heat removal

Table 7-2

RETRAN ANALYSIS KEY MODELING PARAMETERS

Report Section	RETRAN Model Used	Transient Time	Initial Power Level Megawatts	Percent of Full Power
4.1	Two Loop CCNPP[1]	360 seconds	2700	100
4.2	One Loop CCNPP[1]	55 seconds	512	20[3]
	Four Loop CCNPP[1]	55 seconds	512	20[3]
4.3	Four Loop CCNPP[1]	Steady State	0	0
4.4	Four Loop CCNPP[1]	60 seconds	2048	80[3]
4.5	Two Loop CCNPP[1]	87 minutes	1024	40[3]
5.2	One Loop CCNPP[1]	192 minutes	2700	100
5.3	Two Loop CCNPP[1]	60 minutes	2700	100
6.3	Two Loop LOFT[2]	200 seconds	36.9	74
6.4	Two Loop LOFT[2]	200 seconds	36.9	74

[1] CCNPP = Calvert Cliffs Nuclear Power Plant Model

[2] LOFT = Loss of Fluid Test Model

[3] These tests were performed when the CCNPP licensed power level was 2560 MWt.

Table 7-3

RETRAN ANALYSIS PHENOMENOLOGY

Report Section	*Event Symmetry	Phenomenology Simulated and Evaluated
4.1	A	Pressurizer thermal-hydraulics, steam generator heat transfer, moderate primary coolant system (PCS) cooldown and depressurization
4.2	S	Four reactor coolant pump (RCP) coastdown, loop flow resistance
4.3	A/S	Combinations of RCP operating loop resistance, abnormal, short-circuit and reverse flow paths in the primary coolant system
4.4	A	Single RCP coastdown and loop flow resistance
4.5	S	Natural circulation flow and heat transfer in the primary and secondary system
5.2	S	Steam generator heat transfer, atmospheric dump valve thermal-hydraulic performance, auxiliary pressurizer spray thermal-hydraulic performance, slow PCS cooldown and depressurization
5.3	A	Normal and reverse steam generator heat transfer, rapid asymmetric PCS cooldown and depressurization, steam full and water solid pressurizer, steam generator upper tube voiding
6.3	S	Pressurizer spray flow interaction, pressurizer thermodynamics, primary and secondary response to changing steam demands, primary to secondary heat transfer, reactor kinetics
6.4	S	HPSI injection location, pressurizer level effect on pressure, primary and secondary response to changing steam demands, reactor kinetics

* A = Loop Asymmetric S = Loop Symmetric

Table 7-4

Calvert Cliffs RETRAN Model Improvements and Lessons Learned

REPORT SECTION	REFINEMENTS, ENHANCEMENTS OR INSIGHTS
4.1	For a rapid pressurizer insurge transient, High IHTC, Surgeline Loss Factor, Initial Spray Line Temperature and Two Node Pressurizer significantly affect pressure and level response. Spray junction model, rainout velocity and pressurizer heat slabs have no significant effect. Non-condensible gases and normal operational spray flow, heater operation and ambient heat losses don't affect pressurizer performance. Uncertainties in actual spray flow rate and performance can significantly affect pressurizer response.
4.2	The RETRAN Four Loop model predicts a slightly closer four RCP coastdown than the One Loop model in comparison to plant data.
4.3	Downcomer crossflow junction and accurate RCP locked rotor reverse flow pressure loss coefficient are important for less than four RCP operation steady state loop flow calculations. The value of downcomer crossflow junction loss coefficient does not significantly affect (4-N) RCP operation. Three abnormal loop flow paths were identified for (4-N) RCP operation.
4.4	Varying RCP moment of inertia and rated torque significantly affect flow coastdown. Larger values of moment of inertia have the same effect as smaller values of torque. Downcomer crossflow and RCP locked rotor reverse flow loss coefficient do not significantly affect flow coastdown.
4.5	Early post trip primary and secondary thermal-hydraulic response is affected by the timing of TSV closure and ADV/TBV opening.
5.2	The rate of pressurizer depressurization is strongly dependent on pressurizer spray flow. For cooldowns using the ADVs, it is important to model the ADV mass flow accurately. For long cooldown and depressurization transients, the value of IHTC has no effect.
5.3	A large SG tube volume with the bubble rise model and enthalpy transport turned off in the SG allows simulation of SG tube voiding. Pressurizer model IHTC does not affect pressurizer outsurge transients.
6.3	A high value for the IHTC is needed to properly model the pressure and level response of a pressurizer undergoing an insurge transient (See 4.1).
6.4	For transients in which there is an initial outsurge followed quickly by an insurge into the pressurizer, the value of IHTC must be small to properly calculate the pressure and level response.

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