BWR Transient Analysis Methods

NFSR-0111
Revision 0
June, 1995

ComEd

Nuclear Fuel Services Department
Chicago, Illinois

Commonwealth Edison Company

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Abstract

This topical report presents Commonwealth Edison Company's (ComEd's) BWR transient analysis methods. The report demonstrates the validity of the methods and the qualification of ComEd to perform transient analysis for reload licensing and operational support applications. This is accomplished by presenting the results of the benchmarking studies of ComEd BWR plant startup tests, Peach Bottom turbine trip tests, and the Peach Bottom NRC licensing basis transient. Related material including the core thermal limit, and reload application methodologies to qualify the reload licensing application will be provided in a separate report.

ComEd's transient analysis methods are primarily based on the computer codes developed by the Electric Power Research Institute (EPRI): RETRAN02/MOD005, ESCORE, FIBWR2 (EPRI/Scientech), and PETRA (Scandpower).

The ComEd benchmarking analysis of the plant startup tests was chosen to validate the ComEd transient analysis methods for a variety of plant transients. Comparison studies of the turbine trip tests performed at Peach Bottom Unit 2 Cycle 2 demonstrate the acceptability of the ComEd transient analysis methods for more challenging pressurization events similar to the licensing basis events. An NRC licensing basis transient case of Peach Bottom turbine trip without bypass was analyzed to demonstrate ComEd method's capability of predicting system response under conditions which challenge operating limits.

The comprehensive nature of the benchmarking scope and the good agreement of all the benchmarking results have fully demonstrated the capability of the ComEd transient analysis methods and the qualification of the ComEd staff to use the methods presented to perform transient analysis for reload licensing and operational support applications.

Acknowledgments

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1. Introduction

1.1 Purpose

This report describes transient analysis methods developed by Commonwealth Edison Company (ComEd) for performing safety related transient analyses for LaSalle Units 1 and 2, Quad Cities Units 1 and 2, and Dresden Units 2 and 3 Nuclear Power Stations (ComEd BWR plants). The purpose of this report and the information contained herein is to provide a technical basis of ComEd's qualification to perform safety related transient analysis for the ComEd BWR plants.

LaSalle, Quad Cities, and Dresden are General Electric (GE) Boiling Water Reactor (BWR) plants located in Marseilles, Cordova, and Morris, Illinois, respectively. The original architect engineer (AE) functions were performed by Sargent and Lundy Engineers of Chicago, Illinois. Commercial operation of LaSalle, Quad Cities, and Dresden started in 1982, 1972, and 1971 respectively. Key design features of the plants are shown in Table 1.1-1.

Table 1.1-1 Key Design Characteristics of LaSalle, Quad Cities and Dresden

Plant Name	LaSalle	Quad-Cities	Dresden
Reactor/Containment Type	BWR-5/Mark-II	BWR-3/Mark-I	BWR-3/Mark-I
Rated Thermal Power (MW)	3323	2511	2527
Rated Dome Pressure (psig)	1005	1005	1005
Steamline Pressure Drop (psid)	45	55	55
Rated Core Coolant Flow (Mlbm/hr)	108.5	98	98
Rated Feed Water Temperature (F)	420	340	340
Rated Feed Water/Steam Flow (Mlbm/hr)	14.3	9.76	9.8
Bypass Capacity (% of Rated Steam Flow)	25.0	40.0	40.0
Recirculation Flow Control Method	FCV	M/G	M/G
Number of Recirculation Pumps	2	2	2
Number of Jet Pumps	20	20	20
Number of Safety Valves	18(*)	8	8
Number of Electromatic Relief Valves	0	4(+)	4
Number of Target Rock Safety/Relief Valves	0	1	1
Number of Control Rods	185	177	177
Number of Fuel Bundles	764	724	724

* 18 dual function safety relief valves
+ To be replaced with Target Rock relief valves starting Q1C15 & Q2C14

1.2 Scope and Approach

This report describes the computer codes, plant models and analysis methods to be used to analyze BWR system transient events. These tools will be used in reload licensing, and operational support applications for the ComEd BWR plants. Analyses of loss of feed water heating (LOFWH), and control rod withdrawal error (RWE) are performed using steady-state neutronic methods previously approved in Reference 1. Design Basis Accidents (DBA) such as loss of coolant accidents (LOCA) and control rod drop accidents (RDA) analyses are not included in this report.

The ComEd transient analysis methods are qualified through benchmarking studies of plant test data for ComEd BWR plants. As a part of method qualification, the calculated results are also compared with the Philadelphia Electric's Peach Bottom Station Unit 2 (Peach Bottom) turbine trip tests. The ComEd methods are primarily based on the Electric Power Research Institute (EPRI) RETRAN code. Most of the model inputs are developed to produce a "best-estimate" model to accurately predict plant behavior and response. The Siemens Power Corporation steady-state core physics codes and models (CASMO/MICROBURN) which are used to produce input to the transient analysis models have been approved by the USNRC for ComEd design applications (Reference 1). The PETRA code is used to collapse the 3D neutronic data generated by MICROBURN to 1D neutronic input required by RETRAN. The fuel rod steady-state gap conductance calculations are performed by ESCORE. The FIBWR2 code calculates core steady-state thermal-hydraulic performance. The computer codes used for the analyses are validated and verified for adequacy in safety related applications as required per the ComEd software Quality Assurance (QA) program.

1.3 Report Summary

Section 2 presents a methodology flow chart, a list of computer codes used and a brief description of the calculations performed by the key codes used in the ComEd transient analysis methodology.

Section 3 describes the RETRAN plant system models for LaSalle, Quad Cities, and Dresden. Key features of all the RETRAN models and description of major plant specific input parameters are presented.

Section 4 presents the plant startup test data benchmarking results and analyses for ComEd BWR plants. Section 5 documents the benchmarking studies of the three Peach Bottom turbine trip tests. Section 6 provides a comparison study of ComEd and Brookhaven National Laboratory/NRC methods based on a Peach Bottom licensing basis transient case (turbine trip without bypass at rated conditions).

Section 7 summarizes the results of the benchmarking studies. These results demonstrate the capabilities of the ComEd transient analysis methods to analyze a broad spectrum of transient events for ComEd BWR plants .

Appendix A describes the 1D kinetics methodology. The 3D to 1D cross section is calculated based on SCANDPOWER's PETRA code and other utility codes developed by ComEd.

Appendix B provides the steady-state core thermal-hydraulic models and methodology based on FIBWR2 code.

Appendix C presents the fuel gap conductance calculation method for RETRAN system analyses using ESCORE.

1.4 Quality Assurance

The methodology and benchmarking calculations represent work performed by ComEd. The computer codes, plant models, and calculations supporting this work are prepared, reviewed, approved and controlled by formal procedures which conform to the ComEd nuclear quality assurance program.

1.5 Methodology Application

The methods described in this report will be used to perform system transient analyses for reload licensing and operational support applications for ComEd's LaSalle, Quad Cities, and Dresden plants. Additional documentation including the core thermal limit, uncertainty treatments and reload application methodologies to qualify the reload licensing application will be provided in a separate report.

2. Description of Computer Codes

The ComEd transient analysis methods developed for design applications for ComEd BWR plants consist of four major computer codes. The PETRA code collapses the 3D neutronic data generated by MICROBURN to 1D neutronic input required by RETRAN. The fuel rod steady-state, thermal-mechanical performance (gap conductance, etc.) calculations are performed by ESCORE. The FIBWR2 code calculates core steady-state thermal-hydraulic performance (core flow, void, and pressure distributions). The RETRAN02 code is used for reactor system transient response analysis. Figure 2.4-1 shows a flow diagram of the usage of these computer codes, including the major code functions and the transfer of major data variables.

2.1 PETRA

PETRA is used as a data process link between a 3D simulator (MICROBURN) and a corresponding 1D axial kinetic model (RETRAN). PETRA collapses the nodal distribution of cross-section data using appropriate averaging schemes. The averaging schemes in PETRA are based on the principle of preserving reactivity effects by importance weighting of diffusion equation parameters. To assure preservation of the axial power shape, PETRA uses a power/buckling correction algorithm to yield exact agreement between the 1D solution and the axial average of the 3D solution. PETRA uses least-square curve fitting of the collapsed data into polynomial forms.

The PETRA code package was received from SCANDPOWER in January 1993 and was installed on the ComEd HP UNIX workstation system in accordance with the ComEd installation procedures. A code certification project was then undertaken per the ComEd software QA program to assure the integrity of the code after the installation.

2.2 ESCORE

ESCORE is used to calculate steady-state, thermal-mechanical performance characteristics of a fuel rod. The performance parameters calculated by ESCORE include fuel temperatures, fuel-cladding gap conductance, fission gas release, and rod internal pressure. Currently, ESCORE is used in the ComEd methods to determine fuel-cladding gap conductance as an input to RETRAN for system transient analysis. ESCORE will be used in calculating the hot channel gap conductance required for transient CPR analysis in the ComEd thermal limit methodology.

ESCORE was developed under the sponsorship of EPRI . This code was reviewed by the NRC and received an SER in May 1990. The ESCORE code package was received from EPRI in January 1991 and was installed on the ComEd IBM mainframe system in accordance with the ComEd installation procedures. A code certification project was then undertaken per the ComEd software QA program to assure the integrity of the code after the installation.

2.3 FIBWR2

FIBWR2, an improved version of FIBWR (Reference 2), was developed to provide an accurate steady-state and transient core thermal-hydraulic simulator for BWR core reload design and licensing calculations. Currently, the ComEd methodology uses FIBWR2 to predict steady-state core pressure and flow distributions for use in the RETRAN system models. FIBWR2 will also be used in the hot channel transient CPR analysis in the ComEd thermal limit methodology.

FIBWR was initially released in 1981. It has been widely used by the US BWR utilities for steady-state core thermal-hydraulic applications in the system transient topical submittals and has received NRC approval.

FIBWR2 was developed by SCIENTECH, Inc. under the sponsorship of five BWR utilities. The FIBWR2 code package was received from SCIENTECH in January 1993 and was installed on the ComEd HP UNIX workstation system in accordance with the ComEd installation procedures. A code certification project was then undertaken per the ComEd software QA program to assure the integrity of the code after the initial installation.

2.4 RETRAN02/MOD005

The RETRAN02/MOD005 computer code is employed in performing one dimensional kinetics and thermal-hydraulic calculations for this study. This code was reviewed by NRC and received an SER in References 3 and 4.

The NRC SER was reviewed for the intended application of using RETRAN02/MOD005 for design analysis of BWR transient events. Special care was taken in regard to the restrictions in some of the models and applications as identified in the SER. It was concluded that the models and analyses presented in this report are in compliance with the general and specific limitations cited in the RETRAN SER, and that RETRAN02/MOD005 is an acceptable tool for ComEd BWR transient analysis applications.

The RETRAN02 code was corrected to obtain the Baroczy two phase friction multipliers based on flowing quality. The Baroczy correlation was not changed for this code correction. Only the input to the Baroczy correlation was changed. RETRAN02/MOD005 used the thermodynamic quality as input to the correlation, however, the correlation was developed based on flowing quality. This correction has been shown to provide improved comparisons to pressure drop test data (Reference 5).

The RETRAN02/MOD005 code package was received from the Electric Power Software Center in September 1993 and was installed on the ComEd HP UNIX workstation system in accordance with the ComEd installation procedures. A code certification project was then undertaken per the ComEd software QA program to assure the integrity of the code after the installation.

Use of RETRAN02/MOD005 for kinetics and thermal-hydraulics analysis for ComEd BWR transient analysis applications is supported by conformance with the NRC SER and the ComEd software QA program.

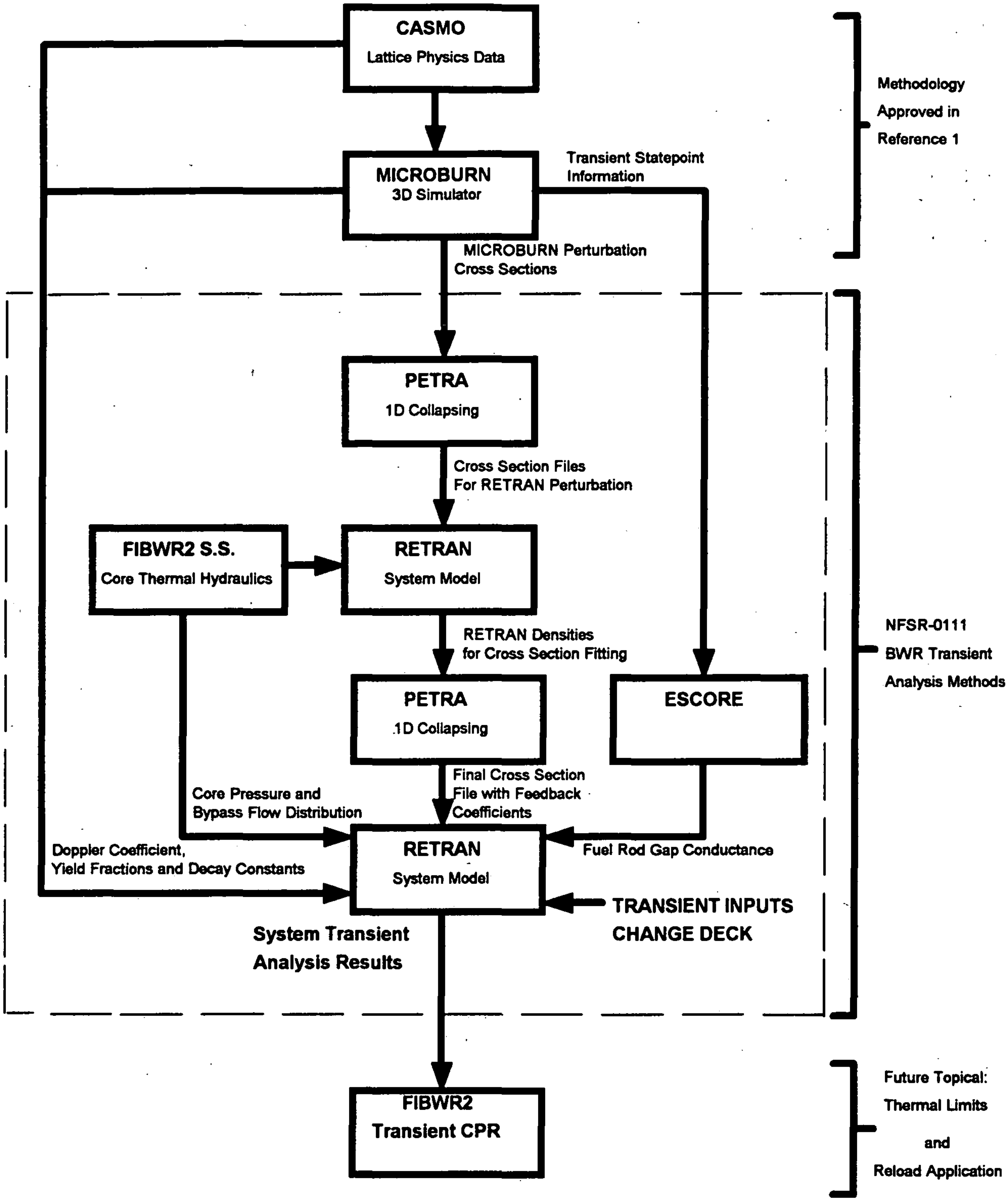

Figure 2.4-1, ComEd Transient Analysis Methods Computer Code Flowchart

3. Description of LaSalle, Quad-Cities and Dresden System Models

The RETRAN system models presented in this report were developed to analyze a wide range of transients; including startup tests and reload licensing analyses. They contain several specialized models for analysis. These specialized models include; one-dimensional core kinetics described in Appendix A, jet pump momentum mixing, Baroczy two-phase wall friction multiplier with flowing quality, and algebraic slip. The RETRAN options and model nodalization were chosen to accurately represent the plant phenomenon. This provides a reasonable match between RETRAN results and measured data. This is appropriate because conservative analysis assumptions and conditions are typically incorporated for licensing basis transients.

3.1 LaSalle RETRAN Model

3.1.1 System Model Nodalization and Geometry

The nodalization diagrams for the LaSalle RETRAN model shown in Figure 3.1-1 and Figure 3.1-2 represent the cycle 1 configuration. This nodalization scheme is very similar to many other utilities' BWR RETRAN model nodalizations which have proven acceptable for licensing applications. This nodalization is consistent with NRC approved models. These can be found in References 5 and 6.

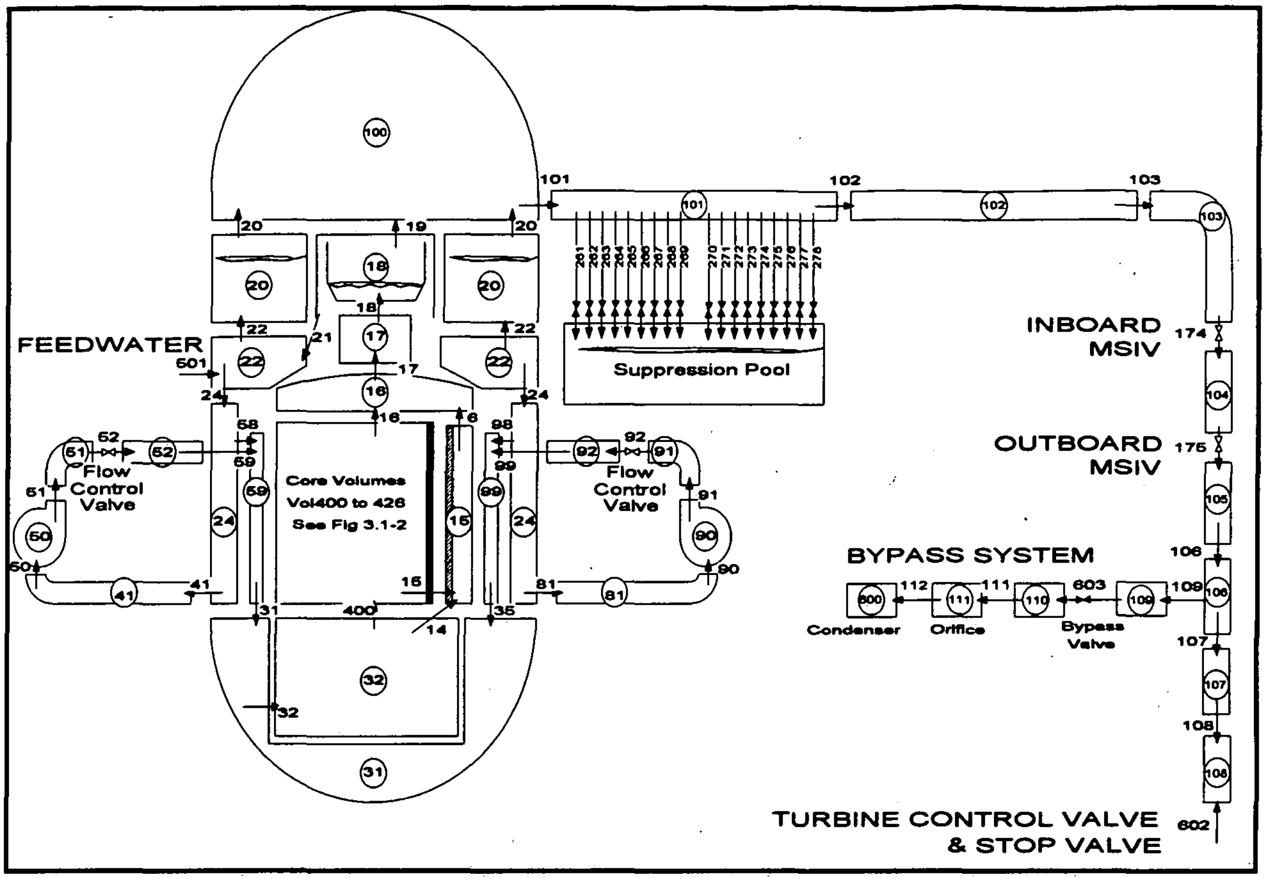

Figure 3.1-1, LaSalle RETRAN Model (Vessel, Steam Line and Bypass)

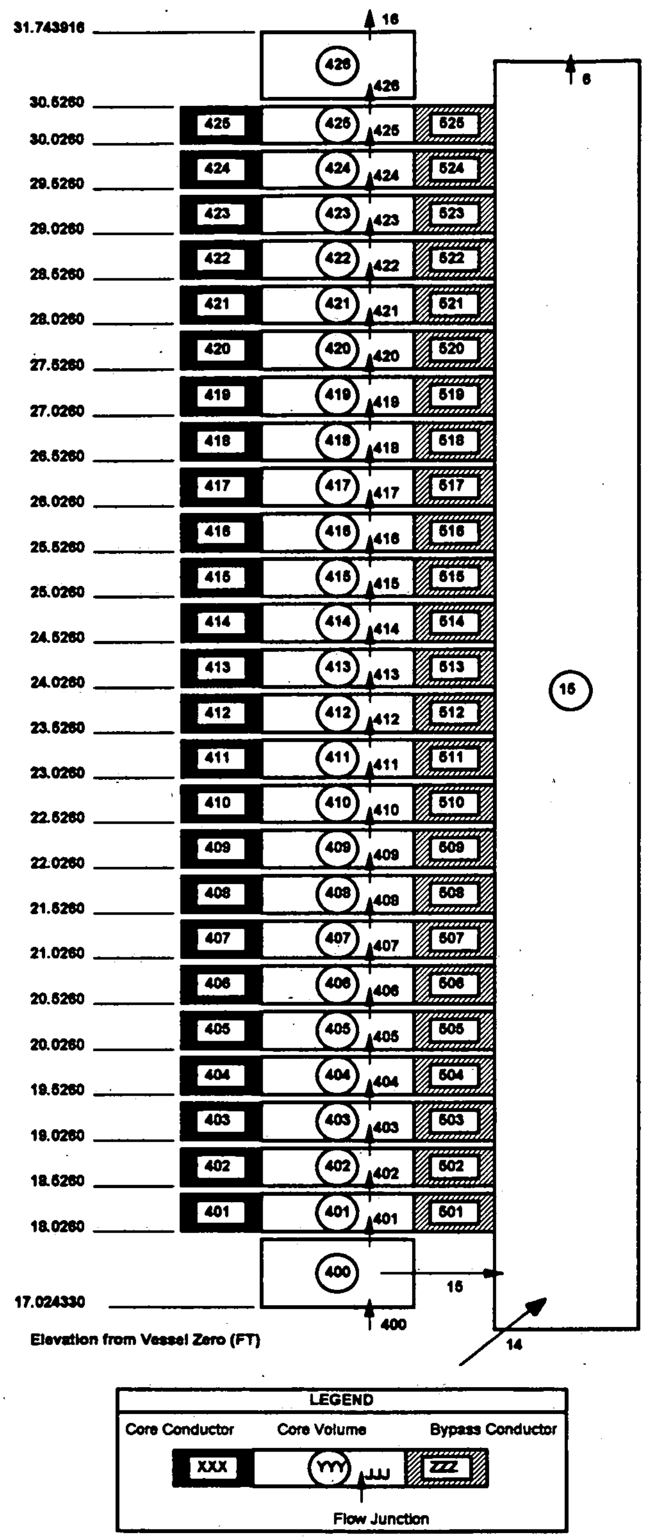

Figure 3.1-2, LaSalle RETRAN Model (Core)

3.1.1.1 Vessel Internals

The upper plenum region above the core was modeled as a single volume (volume 16). This volume receives flow from the active core as well as the core bypass volume. These flows mix and continue on into the steam separator standpipes. All 225 standpipes were modeled as a single volume (volume 17). The flow from volume 17 leads into the separators which were also modeled as a single volume (volume 18). The separator volume was modeled using the RETRAN separator component model and non-equilibrium pressurizer model. The separator model assumes a constant carryunder fraction. This assumption is acceptable since the fluid transport time from the separator to the core inlet is about 10 seconds. Therefore, using a constant carryunder fraction will not impact the core response for most transient events, especially the limiting pressurization events which are terminated within a few seconds. Changes in carryunder, which affect the core inlet enthalpy, will not be propagated to the core inlet until well after the time of minimum CPR. Since the dependence of carryunder fraction on water level is very weak, only an insignificant change in carryunder fraction will occur prior to the high water level turbine trip in the feedwater controller failure event. Detailed RETRAN volume and junction inputs can be found in Table 3.1-1 and Table 3.1-2.

Volume 100 was modeled as the steam dome plus the steam dryers. This volume receives flow from the steam separators (Volume 18) and in some cases the upper downcomer (volume 20) and discharges flow to the main steamline (volume 101).

The upper downcomer volume (volume 20) models the saturated water above the feedwater sparger and the steam water interface. The non-equilibrium pressurizer model was used for this volume to model the interaction between steam and water during pressurization events. The upper downcomer extends to allow for a reactor water level of -22.5 inches up to +80.0 inches. The reference 0.0 inches water level is located at the bottom of the dryer skirt. This range encompasses the valid plant narrow range level.

The middle downcomer (volume 22) is modeled as the portion of the downcomer above the upper plenum and below the upper downcomer. This region is extended to allow for a RETRAN junction between volume 22 and the steam separators (volume 18). The middle downcomer provides the mixing region for flow from the feedwater nozzle (fill junction 501) and recirculation flow from the steam separators. The middle downcomer discharges flow to the lower downcomer (volume 24). The lower downcomer (volume 24) models the narrow region outside the core shroud. This region receives flow from the middle downcomer, and discharges flow to the recirculation loop suction legs (volumes 41 and 81) as well as to the jet pumps (volumes 59 and 99).

The vessel lower plenum is modeled as two regions (volumes 31 and 32). Volume 31 is the region below the jet pump diffuser outside the shroud support legs as well as the lower head below the bottom of the control rod guide tubes. Volume 32 is the region inside the shroud support legs above the bottom of the control rod guide tubes up to the core support plate.

3.1.1.2 Core

The core is modeled as 27 volumes. These volumes represent the volume inside the shroud from the core support plate up to the bottom of the upper plenum. There are 27 axial nodes or volumes representing the core region. Of these 27 nodes, 25 represent the active core volumes, one node represents the lower unheated section, and one node represents the upper unheated section. This nodalization was chosen for consistency with the 3-D core simulator methods described in Reference 1. Benchmarking analyses have shown that this nodalization is adequate for predicting core thermal-hydraulic characteristics.

There is one volume modeling the core bypass region. The active core volumes are heated by powered heat conductors, which use information from the one-dimensional kinetics model described in Appendix A to obtain the power shape. There is also one conductor used for direct gamma heating of the core bypass.

The core conductors use a cylindrical geometry with three regions. The three regions represent the fuel pellet, gap and the cladding respectively. Six mesh steps are used in the fuel pellet to provide adequate detail on the radial temperature distribution, while the gap and cladding each use only one mesh step. The gap conductance is determined separately based on methods described in Appendix C.

3.1.1.3 Recirculation Loops

The two recirculation loops were modeled with four volumes per loop. Volumes 41 and 81 represent the recirculation suction legs from the lower downcomer to the recirculation pumps. Volumes 50 and 90 represent the recirculation pumps. Volumes 51 and 91 represent the recirculation pump discharge upstream of the recirculation flow control valves. Finally, volumes 52 and 92 represent the region from the Recirculation flow control valves to the jet pump nozzle. Volumes 52 and 92 model many components such as the manifolds, risers, and riser nozzles

Each recirculation loop was designed to drive ten jet pumps modeled as one volume in the RETRAN model. A stand-alone single loop submodel of the recirculation loop was developed to achieve the proper M-N characteristics of the jet pump as compared to the published jet pump performance and measured data. The RETRAN jet pump M-N characteristics were compared to the jet pump performance given in the LaSalle UFSAR. This comparison is shown in Figure 3.1-3. The RETRAN calculated drive flow vs. core flow was also compared to plant data. This comparison is shown in Figure 3.1-4. The comparison shows excellent agreement between RETRAN and the plant measurement. This comparison shows that the RETRAN jet pump and recirculation system have been appropriately modeled.

The recirculation flow in LaSalle is controlled by a flow control valve. This flow control valve is not modeled explicitly in RETRAN. The valve is modeled by a variable loss coefficient applied at the junction representing the valve. This loss coefficient is calculated based on desired valve position together with the valve's flow coefficient (C_v).

3.1.1.4 Steam Lines and Feedwater

All of the ComEd BWR RETRAN models have the condensate and feedwater systems as a positive fill junction injecting into the middle downcomer (volume 22) at the elevation of the feedwater sparger. Enthalpy of the feedwater fill junction was determined from heat balances verified to be consistent with actual operating data at various power and feed flow conditions. Feedwater flow control for the RETRAN model is calculated by the vessel water level control system. See Section 3.1.6.3 for more details on the feedwater controller.

The four main steam lines for the LaSalle plant were lumped into one hydraulically equivalent RETRAN steam line with 8 control volumes. The LaSalle RETRAN steam line was developed consistent with method used to develop the ComEd Peach Bottom RETRAN steam line model. This nodalization was validated through the Peach Bottom Turbine Trip benchmarking and the LaSalle startup benchmarking (see Sections 5 and 4 respectively). Main steam flow is removed from the end of the piping with a negative fill junction. Overall steam line pressure drop was verified against plant data and published data. The main steam bypass piping model was explicitly modeled from the equalization header to the bypass valve. The five main steam bypass lines were hydraulically lumped into one RETRAN volume which included a passive heat conductor. The main steam flow for the RETRAN model is calculated by the pressure control system. See Section 3.1.6.2 for more details on the pressure control system.

Table 3.1-1, LaSalle System Model Volume Geometric Data

Volume Node #	V	ZVOL	ZM	FLOWL	FLOWA	DIAMV	ELEV	DESCRIPTION
15	765	14.4627	14.4627	14.4627	52.8948	0.1411	17.2813	Core Bypass Flow Region
16	953	5.1415	5.1415	5.1415	183.3545	13.8387	31.7439	Shroud Head Region
17	400	7.5729	7.5729	7.5729	52.8198	0.6238	36.8854	Separator Standpipes
18	947.3	6.1667	3.5	6.1667	153.6154	0.818	44.4583	Separator Downcomer Region
20	1712.5846	8.5417	4.2917	8.5417	159.162	0.779	42.0833	Upper Annular Downcomer Region
22	1979.1154	10.15625	10.15625	10.15625	293.3128	2.776	34.30208	Middle Annular Downcomer Region
24	2471	23.34375	23.34375	23.34375	105.8527	2.275	10.95833	Lower Annular Downcomer Region
31	1084.1846	10.79167	10.79167	10.79167	100.465	5.219	0.000	Vessel Plenum Inlet Region
32	1098.63	13.0313	13.0313	13.0313	84.307	0.585	4.25	Vessel Plenum Outlet Region
41	138.6613	34.68242	34.68242	54.67265	2.53621	1.797	-19.4089	Loop A Recirculation Piping Suction Side
50	30.5	2.80475	2.80475	0.00	2.53621	1.797	-16.5104	Loop A Recirculation Pump
51	4.64971	1.797	1.797	1.83333	2.53621	1.797	-15.5027	Loop A Recirculation Piping Discharge
52	227.2891	44.02475	44.02475	79.8555	2.84625	0.98252	-15.5027	Loop A Recirculation Piping Discharge*
59	121.5	16.64583	16.64583	16.64583	19.6895	1.58333	10.79167	Loop A Jet Pump
81	138.6915	34.68242	34.68242	54.68455	2.53621	1.797	-19.4089	Loop B Recirculation Piping Suction Side
90	30.5	2.7839	2.7839	0.0000	2.53621	1.797	-16.4896	Loop B Recirculation Pump
91	4.64971	1.797	1.797	1.83333	2.53621	1.797	-15.5027	Loop B Recirculation Piping Discharge Side
92	227.2891	44.02475	44.02475	79.8555	2.84625	0.98252	-15.5027	Loop B Recirculation Piping Discharge Side
99	121.5	16.64583	16.64583	16.64583	19.6895	1.58333	10.79167	Loop B Jet Pump
100	6090.3	21.3175	21.3175	21.3175	285.695	21.1667	50.625	Steam Dome
101	893.9679	50.7813	50.7813	75.8034	11.7932	1.9375	4.1771	Steam Line
102	364.188	20.1042	20.1042	30.8811	11.7932	1.9375	-15.9271	Steam Line
103	326.4028	1.9375	1.9375	27.6771	11.7932	1.9375	-15.9271	Steam Line
104	851.6686	53.3854	53.3854	72.2166	11.7932	1.9375	-67.375	Steam Line
105	1423.919	1.9375	1.9375	120.7402	11.7932	1.9375	-67.375	Steam Line
106	697.4787	48.7188	48.7188	59.1423	11.7932	1.9375	-67.375	Steam Line
107	198.6498	19.0443	19.0443	14.5	13.7031	2.0885	-18.6563	Steam Line
108	130.1488	2.0885	2.0885	9.4978	13.7031	2.0885	-1.7005	Steam Line
109	238.4137	20.2462	20.2462	50.6352	2.836	1.3437	-20.004	Bypass Piping
110	620.9016	16.7968	16.7968	187.3677	3.528	0.9456	-16.5547	Bypass Piping
111	40.9425	1.0703	1.0703	8.7711	4.2611	0.7968	-16.8281	Bypass Piping
200	229538	100	100	0.00	4999	86.66	7.4193	Drywell
300	165100	33.67	26.5	0.00	4999	86.66	7.4193	Suppression pool

Table 3.1-1, LaSalle System Model Volume Geometric Data (Continued)

Volume Node #	V	ZVOL	ZM	FLOWL	FLOWA	DIAMV	ELEV	DESCRIPTION
400	51.5022	1.00167	1.00167	1.00167	25.5227	0.176433	17.02433	Core Inlet Reflector Volume
401	41.5903	0.5	0.5	0.5	83.1805	0.043967	18.026	1st Core Volume
402	41.5903	0.5	0.5	0.5	83.1805	0.043967	18.526	2nd Core Volume
403	41.5903	0.5	0.5	0.5	83.1805	0.043967	19.026	3rd Core Volume
404	41.5903	0.5	0.5	0.5	83.1805	0.043967	19.526	4th Core Volume
405	41.5903	0.5	0.5	0.5	83.1805	0.043967	20.026	5th Core Volume
406	41.5903	0.5	0.5	0.5	83.1805	0.043967	20.526	6th Core Volume
407	41.5903	0.5	0.5	0.5	83.1805	0.043967	21.026	7th Core Volume
408	41.5903	0.5	0.5	0.5	83.1805	0.043967	21.526	8th Core Volume
409	41.5903	0.5	0.5	0.5	83.1805	0.043967	22.026	9th Core Volume
410	41.5903	0.5	0.5	0.5	83.1805	0.043967	22.526	10th Core Volume
411	41.5903	0.5	0.5	0.5	83.1805	0.043967	23.026	11th Core Volume
412	41.5903	0.5	0.5	0.5	83.1805	0.043967	23.526	12th Core Volume
413	41.5903	0.5	0.5	0.5	83.1805	0.043967	24.026	13th Core Volume
414	41.5903	0.5	0.5	0.5	83.1805	0.043967	24.526	14th Core Volume
415	41.5903	0.5	0.5	0.5	83.1805	0.043967	25.026	15th Core Volume
416	41.5903	0.5	0.5	0.5	83.1805	0.043967	25.526	16th Core Volume
417	41.5903	0.5	0.5	0.5	83.1805	0.043967	26.026	17th Core Volume
418	41.5903	0.5	0.5	0.5	83.1805	0.043967	26.526	18th Core Volume
419	41.5903	0.5	0.5	0.5	83.1805	0.043967	27.026	19th Core Volume
420	41.5903	0.5	0.5	0.5	83.1805	0.043967	27.526	20th Core Volume
421	41.5903	0.5	0.5	0.5	83.1805	0.043967	28.026	21st Core Volume
422	41.5903	0.5	0.5	0.5	83.1805	0.043967	28.526	22nd Core Volume
423	41.5903	0.5	0.5	0.5	83.1805	0.043967	29.026	23rd Core Volume
424	41.5903	0.5	0.5	0.5	83.1805	0.043967	29.526	24th Core Volume
425	41.5903	0.5	0.5	0.5	83.1805	0.043967	30.026	25th Core Volume
426	111.477	1.2179	1.2179	1.2179	83.1805	0.040825	30.526	Core Outlet Unheated Volume
600	1E+20	50.0	50.0	0.0	1E+09	1E+09	-50.0	Condenser

Table 3.1-2, LaSalle System Model Junction Geometric Data

Junction Node #	WP	AJUN	ZJUN	INERTA	FJUNF	FJUNR	DIAMJ	CONCO	Description
6	0	52.8948	31.7439	0	0	0	0.1411	0.0000	Core Bypass Exit
14	0	10	17.2813	0	-1	0	0	0.0000	Core Plate Bypass Flow
15	0	10	17.5856	0	-1	0	0	0.0000	Core bypass flow region
16	0	139.5	31.7439	0	0.0725	0.0362	0.2857	0.0000	Shroud Head region
17	0	52.8198	36.8854	0.0918	0.2623	0.5246	0.6238	0.0000	Separator Standpipes
18	30138.88	52.8198	44.4583	0.2241	-1	0	0.6238	0.0000	Separator
19	3972.2222	60.132	50.625	0.0511	1.5119	1.5119	0.5833	0.0000	Dryer Region
20	0	159.162	50.625	0	0.1962	0.0981	0.779	0.0000	Upper Annular Downcomer region
21	0	135.1	44.45833	0.2241	-1	0	0.818	0.0000	Annular Downcomer region
22	0	159.162	42.0833	0	0.1046	0.2092	0.779	0.0000	Middle Annular Downcomer region
24	30138.88	105.8527	34.30208	0	-1	0	2.275	0.0000	Lower Annular Downcomer region
31	0	19.6895	10.79167	0	2.1544	0	1.5833	0.0000	Jet Pump A exit
32	30138.88	84.307	4.25	0	0	0	0.585	0.0000	Vessel Plenum Outlet region
35	0	19.6895	10.79167	0	2.1544	0	1.5833	0.0000	Jet Pump Exit
41	0	2.53621	14.375	0	5.4845	0	1.797	0.0000	Loop A Recirc Piping Suction Side
50	0	2.53621	-16.5104	0	0.737	1.697	1.797	0.0000	Loop A Recirc Pump Suction
51	0	2.53621	-14.6042	0	-1	0	1.797	0.0000	Loop A Recirc Pump Discharge
52	0	2.53621	-14.6042	8.0953	-504	-504	1.797	0.0000	Loop A Recirc Control Valve
58	10482.4266	1.95	27.4375	4.733	0.0742	0	0.21	0.0000	Loop A Jet Pump Suction
59	4587.0134	0.4673	27.4375	14.4509	0.1209	0	0.2758	0.0000	Loop A Jet Pump Nozzle
81	0	2.53621	14.375	0	5.4845	0	1.797	0.0000	Loop B Recirc Piping Suction Side
90	0	2.53621	-16.4896	0	0.737	1.697	1.797	0.0000	Loop B Recirc Pump Suction
91	0	2.53621	-14.6042	0	-1	0	1.797	0.0000	Loop B Recirc Pump Discharge
92	0	2.53621	-14.6042	8.0953	-514	-514	1.797	0.0000	Loop B Recirc Flow Control Valve
98	10482.4266	1.95	27.4375	4.733	0.0742	0	0.21	0.0000	Loop B Jet Pump Suction
99	4587.0134	0.4673	27.4375	14.4509	0.1209	0	0.2758	0.0000	Loop B Jet Pump Nozzle
101	3972.2222	11.7930	53.9896	0	0.4509	0	3.893	0.0000	Steamline
102	0	11.7930	4.1771	0	2.8356	0	3.893	0.0000	Steamline
103	0	11.7930	-14.9583	0	1.4109	0	3.893	0.0000	Steamline
106	0	11.7930	-66.4063	0	0.5681	0	3.99	0.0000	Steamline
107	0	11.7930	-18.6563	0	1.1140	0	2.72	0.0000	Steamline
108	0	13.7030	-0.6563	0	0.2137	0	2.101	0.0000	Steamline
109	0	2.83598	-18.6563	0	0.45946	0	0	0.0000	Bypass Header
111	0	0.66208	-16.1563	0	1.0118	0	0	0.0000	Steam Bypass
112	0	1.9000	-16.1563	1.0463	1.0461	0	0	0.0000	Steam Bypass
174	0	11.7930	-14.9583	0	1.3842	0	3.893	0.0000	MSIV inboard
175	0	11.7930	-66.4063	0	0.5679	0	3.893	0.0000	MSIV outboard
261	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
262	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
263	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
264	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
265	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
266	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
267	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
268	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
269	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV

Table 3.1-2, LaSalle System Model Junction Geometric Data (Continued)

Junction Node #	WP	AJUN	ZJUN	INERTA	FJUNF	FJUNR	DIAMJ	CONCO	Description
270	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
271	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
272	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
273	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
274	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
275	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
276	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
277	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
278	0	0.1119	23.1199	1.35	0	0	0.3775	0.8968	SRV
300	0	295.18	41.0893	0	5.2	0.5	1.95833	0.0000	Drywell/Suppression
400	28955.1875	83.1805	17.02433	0	-1	-1	0.043967	0.0000	Core Inlet Reflector Flow
401	26841.6865	83.1805	18.026	0	-1	-1	0.043967	0.0000	1st Core Flow
402	0	83.1805	18.526	0	-1	-1	0.043967	0.0000	2nd Core Flow
403	0	83.1805	19.026	0	-1	-1	0.043967	0.0000	3rd Core Flow
404	0	83.1805	19.526	0	-1	-1	0.043967	0.0000	4th Core Flow
405	0	83.1805	20.026	0	-1	-1	0.043967	0.0000	5th Core Flow
406	0	83.1805	20.526	0	-1	-1	0.043967	0.0000	6th Core Flow
407	0	83.1805	21.026	0	-1	-1	0.043967	0.0000	7th Core Flow
408	0	83.1805	21.526	0	-1	-1	0.043967	0.0000	8th Core Flow
409	0	83.1805	22.026	0	-1	-1	0.043967	0.0000	9th Core Flow
410	0	83.1805	22.526	0	-1	-1	0.043967	0.0000	10th Core Flow
411	0	83.1805	23.026	0	-1	-1	0.043967	0.0000	11th Core Flow
412	0	83.1805	23.526	0	-1	-1	0.043967	0.0000	12th Core Flow
413	0	83.1805	24.026	0	-1	-1	0.043967	0.0000	13th Core Flow
414	0	83.1805	24.526	0	-1	-1	0.043967	0.0000	14th Core Flow
415	0	83.1805	25.026	0	-1	-1	0.043967	0.0000	15th Core Flow
416	0	83.1805	25.526	0	-1	-1	0.043967	0.0000	16th Core Flow
417	0	83.1805	26.026	0	-1	-1	0.043967	0.0000	179th Core Flow
418	0	83.1805	26.526	0	-1	-1	0.043967	0.0000	18th Core Flow
419	0	83.1805	27.026	0	-1	-1	0.043967	0.0000	19th Core Flow
420	0	83.1805	27.526	0	-1	-1	0.043967	0.0000	20th Core Flow
421	0	83.1805	28.026	0	-1	-1	0.043967	0.0000	21st Core Flow
422	0	83.1805	28.526	0	-1	-1	0.043967	0.0000	22nd Core Flow
423	0	83.1805	29.026	0	-1	-1	0.043967	0.0000	23rd Core Flow
424	0	83.1805	29.526	0	-1	-1	0.043967	0.0000	24th Core Flow
425	0	83.1805	30.026	0	-1	-1	0.043967	0.0000	25th Core Flow
426	0	83.1805	30.526	0	-1	-1	0.043967	0.0000	Core Outlet Reflector Flow
501	3972.2222	1	41.104	0	0	0	1	0.0000	Feedwater Inlet
602	0	10	-1.7005	0	0	0	1	0.0000	Main Steam TCV Flow to Turbine
603	0	0.82494	-0.6563	0	3.3472	0	0	1.0000	Main Steam Bypass Valve Flow

Table 3.1-3, LaSalle System Model Heat Conductor Geometric Data

Heat Conductor #	ASUL	ASUR	VOLS	HDML	HDMR	DHEL	DHER	CHNL	CHNR	Description
110	2634.1525	2779.1305	73.8810	0.000000	0.000000	0.000000	0.000000	0	0	Steam Bypass Conductor
401	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	1st Core Conductor
402	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	2nd Core Conductor
403	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	3rd Core Conductor
404	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	4th Core Conductor
405	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	5th Core Conductor
406	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	6th Core Conductor
407	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	7th Core Conductor
408	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	8th Core Conductor
409	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	9th Core Conductor
410	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	10th Core Conductor
411	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	11th Core Conductor
412	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	12th Core Conductor
413	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	13th Core Conductor
414	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	14th Core Conductor
415	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	15th Core Conductor
416	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	16th Core Conductor
417	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	17th Core Conductor
418	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	18th Core Conductor
419	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	19th Core Conductor
420	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	20th Core Conductor
421	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	21st Core Conductor
422	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	22nd Core Conductor
423	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	23rd Core Conductor
424	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	24th Core Conductor
425	0	2994.84	30.14	0.000000	0.043967	0.000000	0.053942	0	0.5	25th Core Conductor
426	0	2994.84	1	0.000000	0.043967	0.000000	0.053942	0	12.5	Direct Bypass Heating Conductor

Table 3.1-3, LaSalle System Model Heat Conductor Geometric Data (Continued)

Heat Conductor #	ASUL	ASUR	VOLS	HDML	HDMR	DHEL	DHER	CHNL	CHNR	Description
501	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	1st Core Bypass Conductor
502	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	2nd Core Bypass Conductor
503	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	3rd Core Bypass Conductor
504	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	4th Core Bypass Conductor
505	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	5th Core Bypass Conductor
506	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	6th Core Bypass Conductor
507	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	7th Core Bypass Conductor
508	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	8th Core Bypass Conductor
509	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	9th Core Bypass Conductor
510	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	10th Core Bypass Conductor
511	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	11th Core Bypass Conductor
512	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	12th Core Bypass Conductor
513	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	13th Core Bypass Conductor
514	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	14th Core Bypass Conductor
515	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	15th Core Bypass Conductor
516	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	16th Core Bypass Conductor
517	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	17th Core Bypass Conductor
518	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	18th Core Bypass Conductor
519	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	19th Core Bypass Conductor
520	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	20th Core Bypass Conductor
521	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	21st Core Bypass Conductor
522	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	22nd Core Bypass Conductor
523	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	23rd Core Bypass Conductor
524	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	24th Core Bypass Conductor
525	649.52	669.52	5.5	0.043967	0.141346	0.256130	0.149576	0.5	0.5	25th Core Bypass Conductor

3.1.2 System Component Models

3.1.2.1 Recirculation Pumps

The LaSalle model used the RETRAN02 pump component model. LaSalle has two constant speed reactor recirculation pumps, core flow is regulated with recirculation flow control valves mounted on the discharge side of the reactor recirculation pumps. Manufacturer data was used to determine the pumps head, flow, speed and torque characteristics for the homologous pump curves in RETRAN.

3.1.2.2 Jet Pumps

Recirculation flow control valves mounted on the discharge side of the reactor recirculation pumps regulate drive flow through a header which curves horizontally around the reactor vessel and splits the discharge flow of each recirculation pump into 5 separate risers. Each riser has individual penetrations into the reactor vessel. Inside the vessel, the recirculation riser pipe takes the recirculation flow up to a rams head. Each ramshead drives two jet pump nozzles with suction coming from the surrounding downcomer fluid. Each jet pump nozzle has five holes. As described in sections 3.1.1 and 3.1.1.3, the LaSalle RETRAN model jet pump performance was tested and verified against measured data. Results can be seen in Figure 3.1-3 and Figure 3.1-4. Detailed RETRAN volume and junction inputs for the jet pumps can be found in Table 3.1-1 and Table 3.1-2.

3.1.2.3 Steam Separators and Dryer

LaSalle has 225 separators that sit atop the core shroud. General Electric has published in Reference 7 the separator inertias as a function of inlet quality. This information is used in the LaSalle RETRAN model. The separator inertia is split between the inlet and outlet junctions. ComEd calculates the case specific separator inertias based on the steady state initialized fluid quality in the separator standpipes (volume 17). ComEd used a bubble rise volume (volume 18) and the non-equilibrium separator component model to simulate the separators. Detailed RETRAN volume and junction inputs for the separators can be found in Table 3.1-1 and Table 3.1-2 respectively. An initial mixture level was chosen to prevent draining or filling of this volume during transient simulation to be consistent with NRC restrictions listed in the RETRAN SER.

LaSalle has the typical General Electric design of chevron type steam dryers. These dryers lower the moisture content to 0.01%. A cylindrical skirt extends down from the dryer below the separator exit elevation to approximately the elevation of the bottom of the separator. Its function is to ensure that fluid exiting the separators passes through the dryers for normal water levels. There is a pressure drop across the dryers equivalent to 7.0 inches of water at rated conditions. This is the water level difference between the water outside the dryer seal skirt and the water inside the dryer seal skirt. ComEd does match the total separator plus dryer pressure drop in the RETRAN separator volume (Volume 18) to preserve the pressure drop between the steam dome and core exit. ComEd also accounts for the water level difference inside and outside the dryer skirt with the water level indication portion of the feedwater control system described in section 3.1.6.3.

3.1.2.4 Safety/Relief Valves

The LaSalle RETRAN model has each valve modeled as a separate flow junction. LaSalle has eighteen safety/relief valves (SRVs) mounted on the four main steam lines upstream of the MSIVs. Each of these valves have their discharge piped directly to the suppression pool. These valves have dual safety and relief functions. The relief function has a controller actuated by a setpoint on reactor dome pressure. This function is used under normal operating conditions and requires non-safety grade power supplies to be available in order to operate. The safety function is used under accident conditions of extreme over-pressurization. This function actuates once the pressure at the valve reaches the preset setpoint. Several of these valves also perform the automatic depressurization system (ADS) function. ComEd models the characteristics of each valve individually along with each of these functions described above. Detailed RETRAN junction inputs for the SRVs can be found in Table 3.1-2. They are modeled as a valve junction from the steam line volume 101 to the suppression pool volume 300. ComEd sizes the contraction coefficients for these junctions number 261 through 278 to achieve the flow capacity at the rated pressure as specified in the Technical Specifications and ASME certification. Delay times for response and stroke times were set consistent with specifications listed in the Updated Final Safety Analysis Report (UFSAR) for LaSalle.

3.1.2.5 Turbine Stop, Control, and Bypass Valves

Turbine stop valves (TSVs) are closed rapidly upon any turbine trip signal. During normal operation, these valves are 100% open. These valves are not explicitly modeled as a separate flow junction. However, they are accounted for in the pressure control system described in section 3.1.6.2. Detailed RETRAN junction inputs for these valves can be found in Table 3.1-2. There is a reactor trip function when the TSVs reach 90% open which is also accounted for in the RETRAN Trip logic.

The LaSalle turbine control valves (TCVs) are operated in the full arc mode. TCVs regulate main steam flow rate in order to maintain the reactor at the preset programmed pressure. They are controlled by the pressure control system. During normal operation at full power, these valves typically are 50% open and are in the linear portion of their flow versus position performance curve. The TCVs are modeled as a negative fill junction in the LaSalle RETRAN model. The flow rate for this junction is determined from the pressure control system. TCV dynamic response, position, and resulting main steam flow rate are all calculated in this control system. TCV fast closure response to a load rejection is accounted for in this control system. Since the initial position of the TCVs are rarely full open, the pressure control system accounts for faster closure time caused by partially closed initial positions.

LaSalle's main steam bypass valves (BPVs) are sized to allow full flow capacity to be 25% of the reactor rated steam flow. The BPVs are fitted with fast acting solenoid controls such that in the event of a turbine trip signal, the BPVs will begin to open immediately on the turbine trip signal. Therefore, the BPVs are opening prior to full closure of the TSVs. This feature has the benefit of minimizing the core pressurization rate. The BPVs position are controlled by the pressure control system. The BPV will stroke open with a linear flow versus demand performance curve until the BPV is full open. The BPVs are modeled as a junction valve number 603 which has its area determined from the pressure control system. BPV dynamic response, position, and area are all calculated in that control system

3.1.2.6 Main Steam Isolation Valves

MSIVs are closed upon any Group I containment isolation signal. During normal operation, these valves are 100% open. These valves are modeled as separate flow junctions, one for the inboard containment penetration and one for the outboard containment penetration. Detailed RETRAN junction inputs for the MSIVs can be found in Table 3.1-2. The area as a function of time is prescribed for the MSIV in a data table. There is a reactor trip function when the MSIVs reach 90% open which is also accounted for in the RETRAN Trip logic.

3.1.2.7 Core Hydraulics

Figure 3.1-2 shows the details of the core nodalization. Detailed RETRAN volume and junction inputs for the LaSalle Cycle 1 core can be found in Table 3.1-1 and Table 3.1-2. The FIBWR2 core pressure distribution and core bypass flow distribution are input into RETRAN. Active core flow rate and bypass flow rates are also specified to match the FIBWR2 results. ComEd's RETRAN one-dimensional LaSalle core model has 25 axial nodes of 6 inches each with both, upper and lower unheated reflector nodes Since this Reference 1 methodology is used in conjunction with RETRAN02 models, as described in Appendix A, the number of RETRAN core conductors was set equal to the number of core nodes in the simulator code MICROBURN. The core volumes are calculated on a bundle weighted average for each fuel type. The core bypass volume has hydraulic characteristics assuming all control rods are out. Fuel types which have water rods in the assembly will have the water rods flow to the core bypass volume.

There are 25 core conductors and one bypass core conductor. Detailed RETRAN conductor inputs for the LaSalle Cycle 1 core can be found in Table 3.1-3. The core bypass conductor supplies direct heating for the core bypass fluid as a fraction of the total power generated in the fuel. This heating fraction to the core bypass is calculated by ComEd using the lattice physics code CASMO3G as described in Appendix A. Fuel assembly channels transfer heat from the fluid in the active fuel regions to the core bypass region through 25 passive heat conductors, one for each active core node.

3.1.3 Trip Logic

Trip actuation can be initiated by RETRAN calculated variables such as: pressure, mass flow rate, normalized power, control block output, or other trip signals. The following sections describe the pertinent trip logic included in the LaSalle RETRAN model. The RETRAN input cards require that each trip have an "ID" number associated with it. These trip ID number assignments to their various functions are strictly arbitrary just as volume node numbers and junction node numbers are arbitrary . The "Trip ID" numbers that appear in these summaries correspond to their assignments in the LaSalle RETRAN model.

3.1.3.1 Reactor Protection System Trips

Reactor protection system trips result in control rod insertion into the core. This is accomplished through a control system and a general data table in the RETRAN model. The model predicts a reactor scram (Trip ID 3) for the following:

1) High neutron flux	(Trip ID 10)
2) High steam dome pressure	(Trip ID 11)
3) Reactor vessel water level, low level 3	(Trip ID 19)
4) Main steamline isolation valve closure	(Trip ID 174 or 175)
5) High drywell pressure *	(Trip ID 20)
6) Turbine stop valve closure	(Trip ID 178)
7) Generator load rejection	(Trip ID 602)
8) Manual scram on elapsed time	(Trip ID 3)

* The high drywell pressure scram is included in the trip system, but will not be used since RETRAN will not necessarily predict a realistic drywell response.

3.1.3.2 Safety/Relief and Main Steam Isolation Valve Trips

All eighteen safety/relief valves for LaSalle are dual function valves. Both functions are modeled in RETRAN, but only one function will be credited during a given analysis. The relief valve function setpoint (Trip IDs 361 - 378) is taken off the dome pressure. The safety valve function setpoint (Trip Ids 261 - 278) is taken off the steamline pressure in the volume where the safety valves are located (Volume 101). The first seven relief valves will open coincident on an automatic depressurization system trip actuation (Trip ID 409). There is also the capability of opening up to four relief valves manually on elapsed time.

The LaSalle RETRAN model does not model the low-low setpoint relief function. The necessary logic will be added if this function is desired for future analysis purposes.

The MSIVs will close if the following trips are actuated:

1) High main steamline flow rate	(Trip ID 13)
2) Low main steamline pressure	(Trip ID 14)
3) Reactor water level, low level 1	(Trip ID 17)
4) Manual trip on elapsed time	(Trip IDs 174 and/or 175)

3.1.3.3 Recirculation Pump Trips

There are two recirculation pumps modeled for LaSalle. These pumps each have independent trip logic. A recirculation pump trip will actuate on any of the following:

1) Reactor vessel water level, low level 2	(Trip ID 18)
2) High steam dome pressure	(Trip ID 27)
3) Manual trip on elapsed time	(Trip IDs 50 and/or 90)

3.1.3.4 Turbine Trip, Generator Load Rejection Trips

The turbine trip (Trip ID 178 actuates on one of three signals: reactor water level, high level 8 (Trip ID 15), generator load rejection (Trip ID 602), and a manual trip on elapsed time (Trip ID 178). The generator load rejection actuates only by a trip on elapsed time (Trip ID 602).

3.1.3.5 LPCI, LPCS, HPCS and RCIC Trips

Both LPCI and LPCS are initiated on either a high drywell pressure (Trip ID 20) or a reactor water level, low level 1 (Trip ID 17). LPCI and LPCS cannot actuate until the low pressure interlock (Trip ID 12) has occurred. HPCS and RCIC both actuate on a reactor water level, low level 2 (Trip ID 18) signal. All four systems also allow for a manual initiation on elapsed time. These trips are included to indicate when the plant would experience ECCS initiation, although the systems are not modeled.

3.1.4 Direct Bypass Heating

Appendix A describes the methods for determining the nuclear characteristics for a particular core configuration. Lattice physics codes are used to determine the fractions of direct moderator heating for the core. These node specific fractions are input using the RETRAN direct moderator heating option. Power deposition to the core bypass region is also calculated using the fraction of total power generated in the fuel. This was described along with the core hydraulics in section 3.1.2.7

3.1.5 Fill Tables and Associated Valve Controls

The LaSalle RETRAN model has only two fill junctions, which are the feedwater positive fill and the main steam flow negative fill from the TCV. These were described in section 3.1.1.4.

The LaSalle RETRAN model has four types of valves, the characteristics of these valves are explicitly modeled. They are the recirculation flow control valves, the SRVs, the MSIVs and the BPVs. These are described in sections 3.1.2.1, 3.1.2.4, 3.1.2.6 and 3.1.2.5 respectively.

3.1.6 Control Logic

3.1.6.1 Sensor Response Models

ComEd used control systems to calculate the sensed plant variables which were typically used in the comparisons with startup test data. These control blocks have no direct impact on the model as they are used to provide edits to compare RETRAN variables to actual plant variables. Some of these variables include; dome pressure, core flow, and steam flow.

3.1.6.2 Pressure Regulator

The pressure control system input descriptions are documented in Table 3.1-4. A block diagram of the pressure control system is shown in Figure 3.1-8.

Each pressure regulator consists of a proportional gain which is defined in percent steam flow demand per unit pressure error. The proportional gain is chosen to provide good low frequency control over the basic reactor pressure dynamic mode. This proportional gain is included as a gain on the lag-lead compensation block. The presence of the lag-lead filter serves to stabilize the dominant pressure response mode in the middle frequency range by effectively reducing the gain at that frequency. The high frequency effects are accounted for by the addition of a lag and notch filter (commonly referred to as the steam line compensator) to the pressure regulators.

Once the pressure error signal is processed through the regulation filters, the pressure regulators signals are compared and the maximum error is used as the system flow demand. The pressure regulator demand is compared with the turbine load and the minimum demand is sent to the control valve actuator. The control valve demand signal is sent through a function generator . The function generator is set up to be "opposite" of the control valve flow characteristics. This results in a linear steam flow response to a given change in pressure regulator demand

The bypass valves are set to open in banks such that the resulting bypass flow changes linearly with a change in bypass demand. The bypass valve flow function generator uses a linearization function.

Trip logic is also included to actuate stop valve or control valve closure on a turbine trip (trip 178) or a turbine generator load rejection (trip 602). A function generator is provided to yield the stop valve position vs. time after a turbine trip. The integrator (-339) acts to close the control valve. This integrated signal is subtracted from the control valve demand position from the PRD to obtain the new valve position as it closes. A MIN block is used to determine lower valve position (TCV or TSV).

A pressure setpoint adjuster function is also included in the block diagram on Figure 3.1-8. The pressure setpoint adjuster operates off the load demand and operates only if the recirculation flow control is in automatic. The PRD signal is subtracted from the load demand with a -10% bias to obtain a total error signal. The setpoint is then adjusted to "anticipate" the increased load before the vessel pressure increases from the increased recirculation pump speed and steam generation.

3.1.6.3 Feedwater Controller

The feedwater/ level control system (FLCS) input descriptions are documented in Table 3.1-4. A block diagram of the feedwater/ level control system is shown on Figure 3.1-5.

The FLCS uses single or 3 inputs depending on if single element or 3 element control is used. The single element control mode only monitors liquid level and the controller adjusts feedwater flow based on the liquid level deviation from the setpoint. The single element mode is typically only used during startup. The 3 element control mode uses the liquid level deviation along with a steam flow / feed flow mismatch (level , steam flow and feed flow are the three inputs) to control feedwater flow. The 3 element control mode acts to anticipate a level change based on a mismatch between feed flow and steam flow. The 3 element control mode usually provides tighter level control due to its anticipatory behavior.

The RETRAN control block diagram, shown on Figure 3.1-5, is the RETRAN equivalent of the plant control logic. Liquid level is calculated by a function generator of level (in.) vs. downcomer liquid volume. The function generator will yield an accurate level change for a given change in downcomer liquid volume. This method is more accurate than simply monitoring the downcomer liquid level in a RETRAN control volume where the area is considered constant. The function generator accounts for the cross sectional area change throughout the downcomer. The sensed liquid level is lagged to account for sensor response. The sensed level is then compared (subtracted from) the level setpoint. When level reaches the setdown setpoint, the level controller setpoint signal is reduced by 50% to minimize level overshoot. The sensed level/ setpoint level comparison generates a level error signal, which is sent to the proportional + integrator (P+I) level controller. P+I controller processes the level error and sends % demand signal for the measured level error (inches).

The steam flow % and feed flow % mismatch is added to the flow demand (3 element control only) from the level controller to obtain an overall % flow demand. The overall % demand signal is sent to a feedwater controller for each turbine driven feedwater (TDFW) pump. The controller for each pump is a P+I controller which sends a demand signal to turbine control valve which governs the amount of steam flow to the TDFW pump. This steam flow rate then governs the pump speed and the pump flow rate. The simulation of the TDFW pumps and resultant flow response is modeled as a second order transfer function. This simulates the flow response to a changing flow demand.

3.1.6.4 Recirculation Controller

The recirculation flow control system (RFCS) input descriptions are documented in Table 3.1-4. A block diagram of the recirculation flow control system is shown in Figure 3.1-6.

The RETRAN control block diagram, shown on Figure 3.1-6, is the RETRAN equivalent of the plant control logic. The sensed drive flow, sensed neutron flux, and the neutron flux demand signal are all calculated by other portions of the LaSalle control system. The RETRAN control system allows the valve to move from 30.0% to 100.0% of the valve area. This is consistent with the plant.

In general LaSalle operates in manual flow control. All of the RETRAN control logic required to model automatic flow control was supplied.

Junctions 52 and 92 represent the flow control valves in the recirculation control system. These junctions in the RETRAN model do not consist of a RETRAN valve. These junctions provide the same function a valve in the actual plant would provide by varying the associated loss coefficient. By increasing the loss coefficient of the junction more resistance is applied at the junction and the flow is therefore reduced. The loss coefficient is calculated from a given valve position.

The valve loss coefficient is calculated by the control system shown in Figure 3.1-7. Figure 3.1-7 applies to both loops. The control block IDs shown are for Loop "A".

3.1.6.5 Normalized Neutron Flux Monitor

Figure 3.1-10 depicts the control systems used for modeling the normalized neutron flux that is measured by the nuclear instrumentation system. The high reactor power trip logic utilized the normalized neutron flux output of control block ID = -4. This control system simply subtracts the decay heat portion from the RETRAN PNRM variable to obtain the neutron flux response. Logic for the flow biased scram was not modeled since it is not credited for any of the plant events. But, neutron flux is required for the reactor protection system trips described in section 3.1.3.1.

3.1.6.6 Average and Local Power Range Monitor

Figure 3.1-11 depicts the control system used for modeling the LPRM level averaged neutron flux that is measured by the nuclear instrumentation system. Two neutronic region power fractions which are closest to the elevation of each LPRM were averaged together to obtain the relative response. Each of these A, B, C and D levels were averaged together, multiplied by the calculated normalized neutron Flux, to get the LPRM averaged response from control block ID=-986. Multiplying each LPRM response by the calculated normalized neutron flux yields the individual LPRM responses for A, B, C and D. These responses are calculated from control blocks ID=-979, ID=-980, ID=-981 and ID=-982 respectively.

3.1.6.7 Core Average Heat Flux

Figure 3.1-12 depicts ComEd's RETRAN control system used to calculate core average heat flux. It was designed to produce core averaged heat flux in percent from control block ID=-952.

3.1.6.8 Other Miscellaneous Control Systems

Figure 3.1-9 depicts the control systems used for calculating the total mass flow rate for the safety/relief valves. It sums up all of the flows in pounds mass per second in control block ID=-974.

Figure 3.1-13 depicts the control systems used for calculating reactivities in dollars for the total and component reactivities. It was developed to produce total reactivity, void reactivity, doppler reactivity and control rod reactivity in dollars from control blocks ID=-954, ID=-955, ID=-956 and ID=-957 respectively.

Figure 3.1-14 depicts the control systems used for modeling the control rod position. This control system calculates the length in feet of the control rod insertion into the active core length in control block ID=-702.

Table 3.1-4: LaSalle RETRAN Control Input Descriptions

Control Input ID	Description of Control Block Input	Control Input ID	Description of Control Block Input
1	Normalized Core Power	501	Constant to Square Value
2	Constant, Yield Fraction Groups 6-11	502	Constant to Raise Value to Forth Power
3	Steam Dome Pressure	777	Trip 777 To Activate Control Rod Positioning
4	Total Core Flow	900	Flow Regime
5	Turbine Inlet Pressure	910	Heat Transfer Regime
198	Unassigned	920	Constant Initial Core Thermal Power (Watts Th.)
199	Unassigned	921	Constant Conversion Factor to BTU/Hr
200	Sensed Liquid Volume in Downcomer Volume 20	922	Constant Fraction of Power Generated in Fuel
201	Sensed Liquid Volume in Downcomer Volume 22	923	Constant Single Core Conductor Area
202	Sensed Liquid Volume in Downcomer Volume 24	924	Right Surface Heat Flux Conductor 401
203	Sensed Liquid Volume in Separator Volume 18	925	Right Surface Heat Flux Conductor 402
204	Constant Initial Separator Liquid Volume	926	Right Surface Heat Flux Conductor 403
205	Steam Flow in Separator Junction 19	927	Right Surface Heat Flux Conductor 404
206	Constant Rated Separator Steam Flow	928	Right Surface Heat Flux Conductor 405
207	Trip ID 720 (Level Setpoint Setdown Trip)	929	Right Surface Heat Flux Conductor 406
208	Trip ID 721 (3 element / 1 element control)	930	Right Surface Heat Flux Conductor 407
209	Steam Flow Junction 103	931	Right Surface Heat Flux Conductor 408
210	Problem Simulation Time	932	Right Surface Heat Flux Conductor 409
211	Steam Dome Pressure	933	Right Surface Heat Flux Conductor 410
212	Constant Miniflow Value	934	Right Surface Heat Flux Conductor 411
213	Trip 722 (Flow Trip to Open FW Miniflow Valve)	935	Right Surface Heat Flux Conductor 412
214	Constant to Square Steam Flow Fraction	936	Right Surface Heat Flux Conductor 413
300	Turbine Inlet Pressure	937	Right Surface Heat Flux Conductor 414
301	Bias on Regulator Setpoint 1	938	Right Surface Heat Flux Conductor 415
302	Bias on Regulator Setpoint 2	939	Right Surface Heat Flux Conductor 416
303	Bypass Negative Bias (3%)	940	Right Surface Heat Flux Conductor 417
304	Constant (1.0)	941	Right Surface Heat Flux Conductor 418
305	Constant - Initial Specific Volume at Turbine Inlet	942	Right Surface Heat Flux Conductor 419
306	Trip Time - Turbine Trip	943	Right Surface Heat Flux Conductor 420
307	Load Reject Trip	944	Right Surface Heat Flux Conductor 421
308	Turbine Trip Time	945	Right Surface Heat Flux Conductor 422
309	Turbine Inlet Specific Volume	946	Right Surface Heat Flux Conductor 423
310	Elapsed simulation time	947	Right Surface Heat Flux Conductor 424
311	Negative Bias on Load Set (10%)	948	Right Surface Heat Flux Conductor 425
312	Initial Turbine Speed (%)	949	Total Reactivity
313	Auto Load Follow Trip	950	Void Reactivity
314	Bypass Specific Volume	951	Doppler Reactivity
400	Trip 740 Auto Load Follow Trip	952	Control Rod Reactivity
401	Constant	953	Beta
402	Simulation Time	954	Safety/Relief Valve Flow Junction 261
403	Multiplier Constant	955	Safety/Relief Valve Flow Junction 262
404	Loop "A" Drive Flow	956	Safety/Relief Valve Flow Junction 263
405	Loop "B" Drive Flow	957	Safety/Relief Valve Flow Junction 264
406	Constant (unity)	958	Safety/Relief Valve Flow Junction 265
407	Normalized Power	959	Safety/Relief Valve Flow Junction 266
500	Constant (Flow Control Valve Diameter)	960	Safety/Relief Valve Flow Junction 267

Table 3.1-5, LaSalle RETRAN Control Input Descriptions (Continued)

Control Input ID	Description of Control Block Input	Control Input ID	Description of Control Block Input
961	Safety/Relief Valve Flow Junction 268	971	Safety/Relief Valve Flow Junction 278
962	Safety/Relief Valve Flow Junction 269	972	Power Fraction Neutronic Region 4
963	Safety/Relief Valve Flow Junction 270	973	Power Fraction Neutronic Region 5
964	Safety/Relief Valve Flow Junction 271	974	Power Fraction Neutronic Region 10
965	Safety/Relief Valve Flow Junction 272	975	Power Fraction Neutronic Region 11
966	Safety/Relief Valve Flow Junction 273	976	Power Fraction Neutronic Region 16
967	Safety/Relief Valve Flow Junction 274	977	Power Fraction Neutronic Region 17
968	Safety/Relief Valve Flow Junction 275	978	Power Fraction Neutronic Region 22
969	Safety/Relief Valve Flow Junction 276	979	Power Fraction Neutronic Region 23
970	Safety/Relief Valve Flow Junction 277		

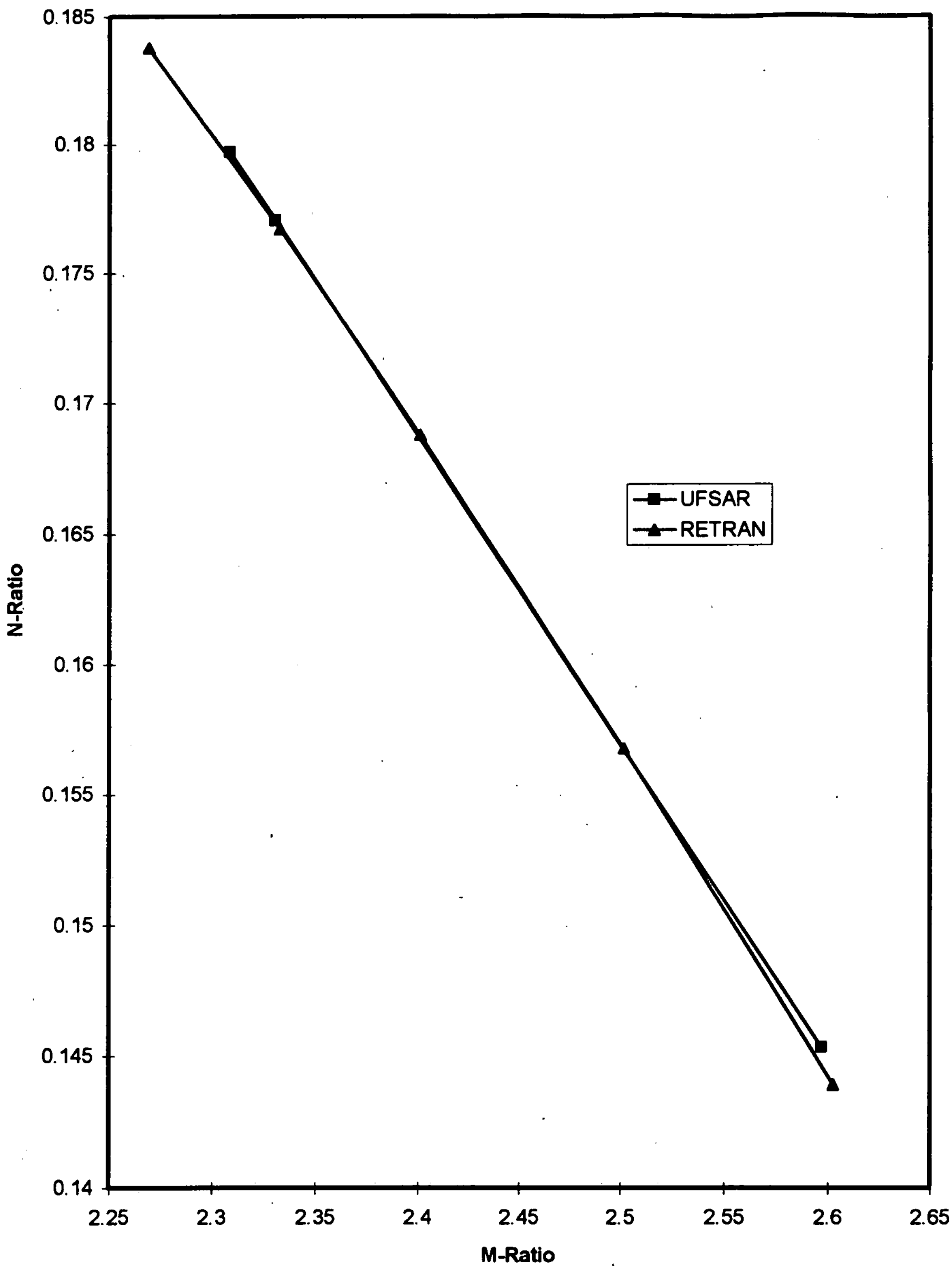

Figure 3.1-3, LaSalle Jet Pump M-N Characteristics

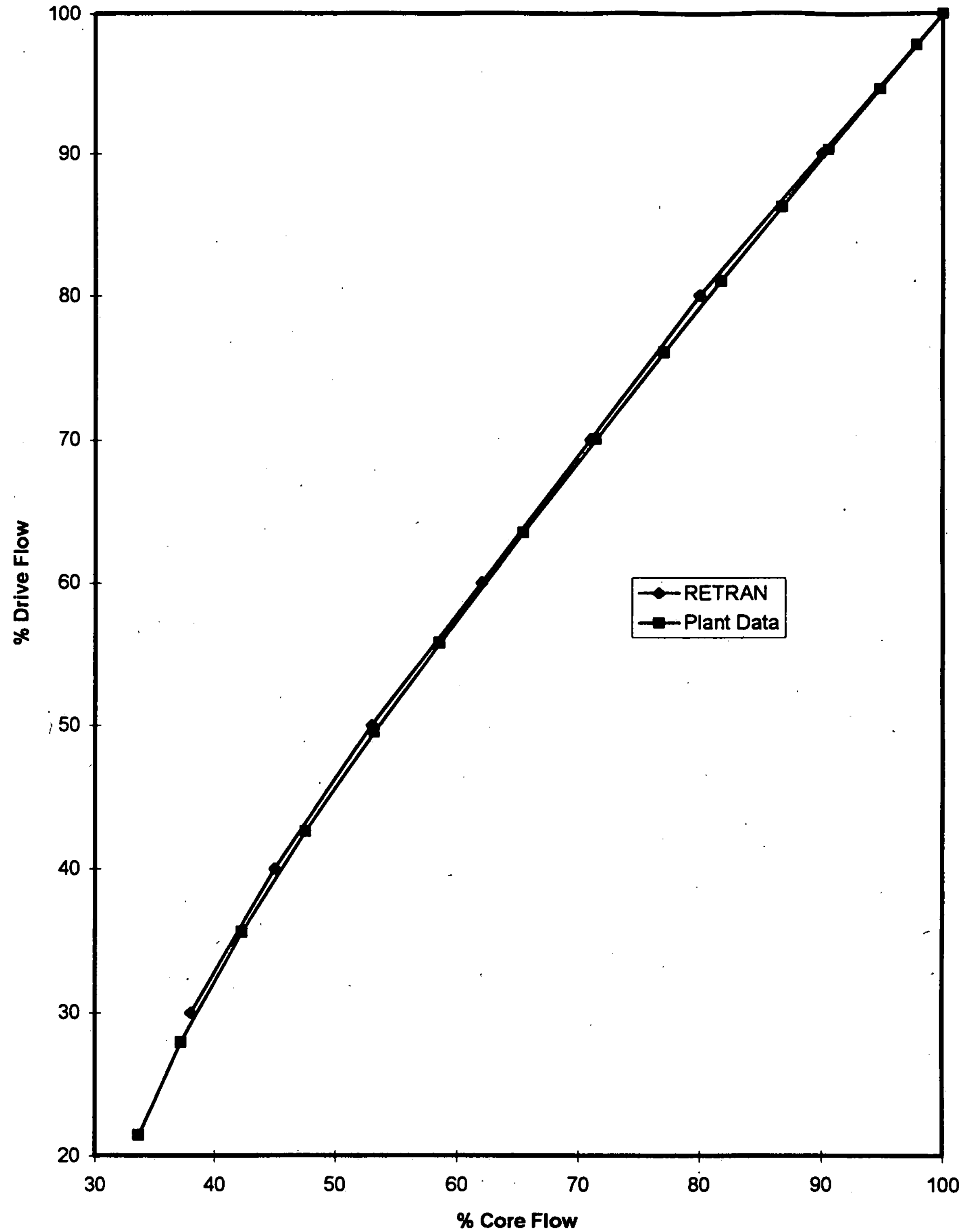

Figure 3.1-4, LaSalle Core Flow vs. Drive Flow Characteristics

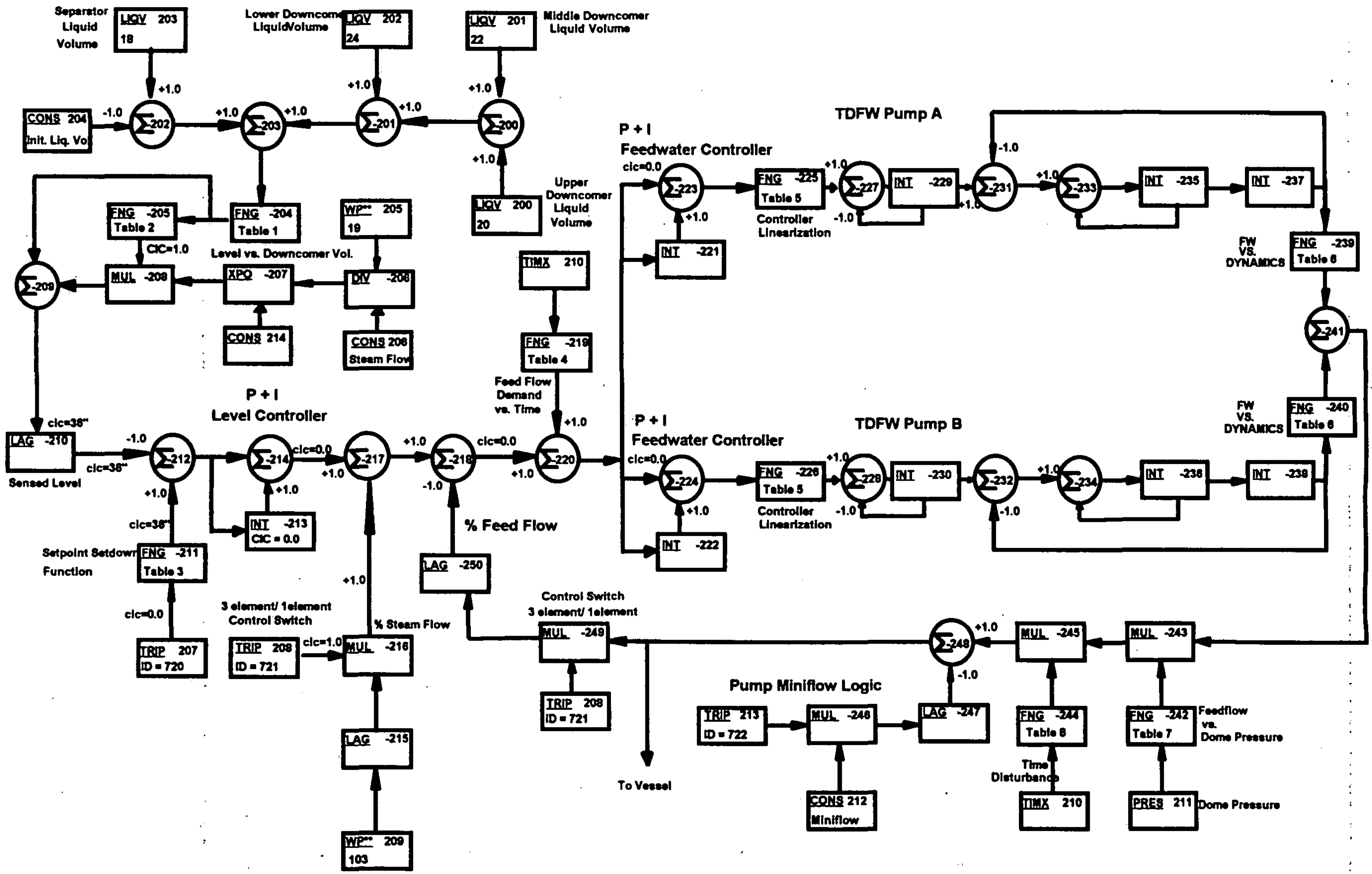

Figure 3.1-5, LaSalle Vessel Water Level Controller

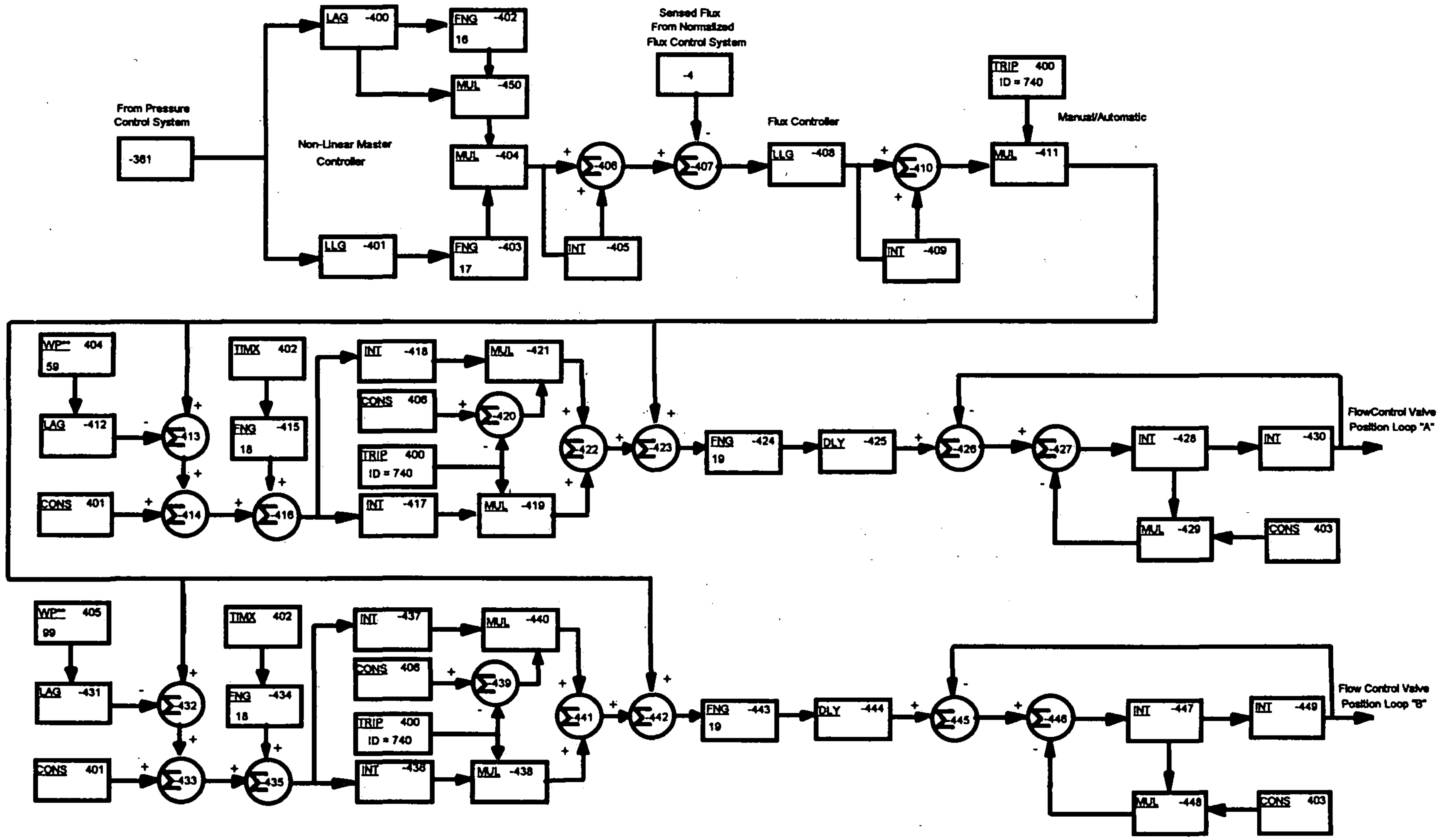

Figure 3.1-6, LaSalle Recirculation System Controller

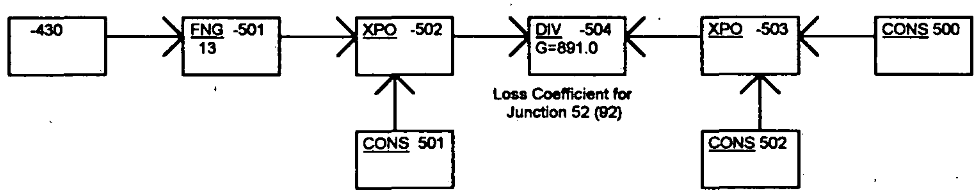

Figure 3.1-7, Recirculation Flow Control Valve Loss Coefficient Control System

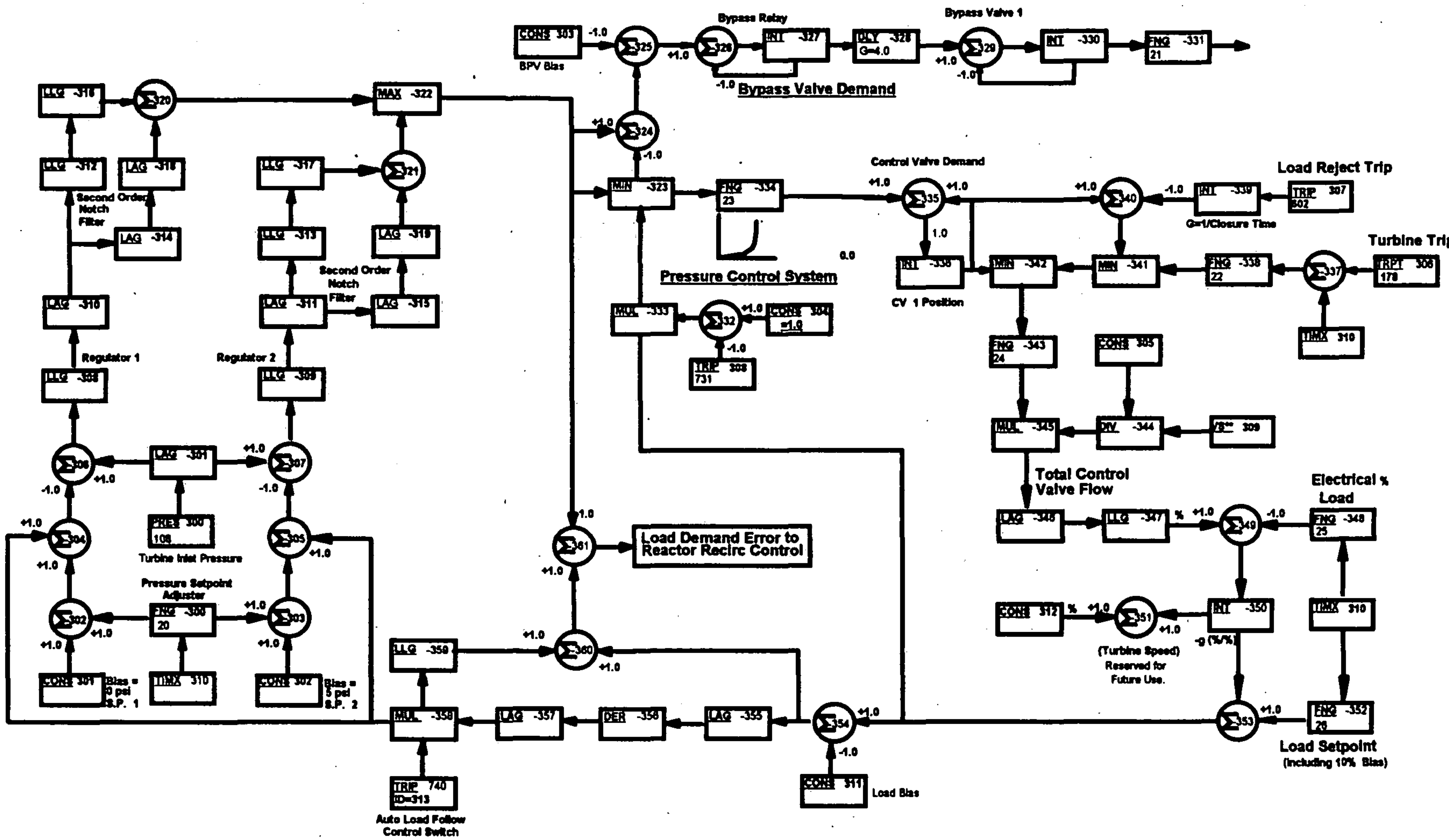

Figure 3.1-8, LaSalle Pressure Regulator

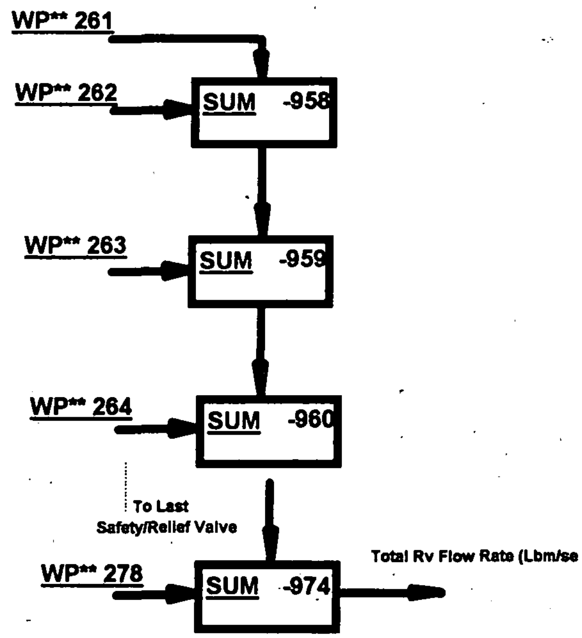

Figure 3.1-9, LaSalle Control Blocks for Calculating Total SRV Flow Rate

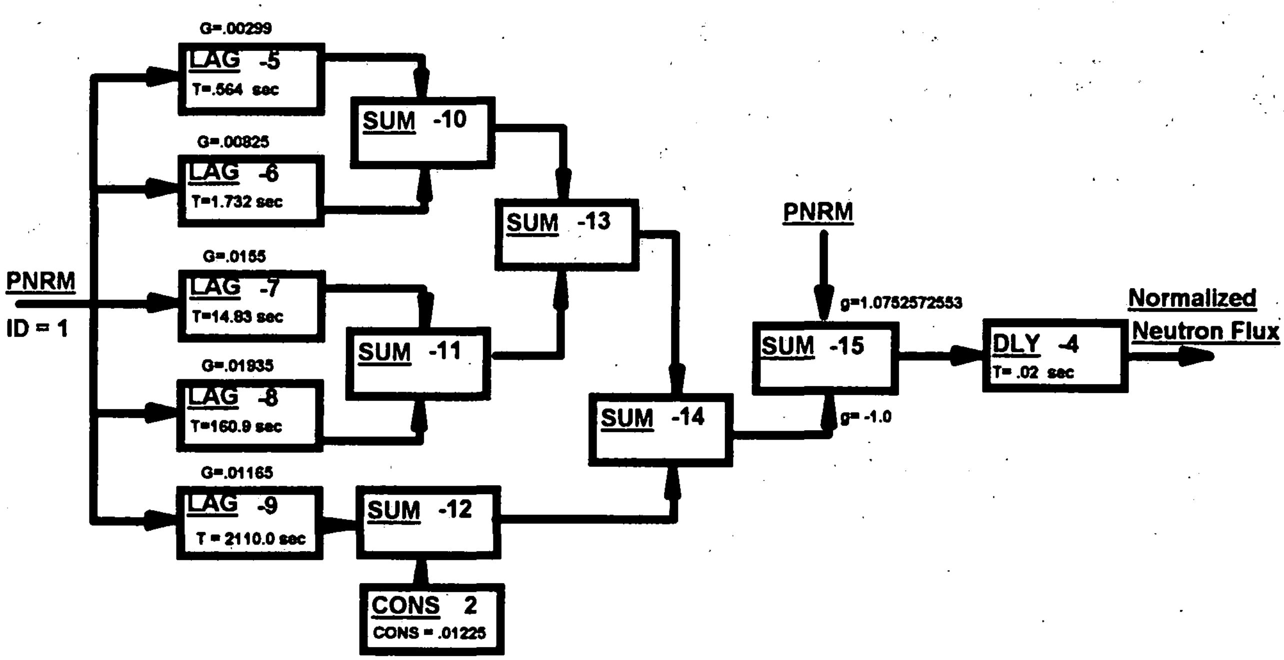

Figure 3.1-10, BWR System Model Calculation of Normalized Neutron Flux

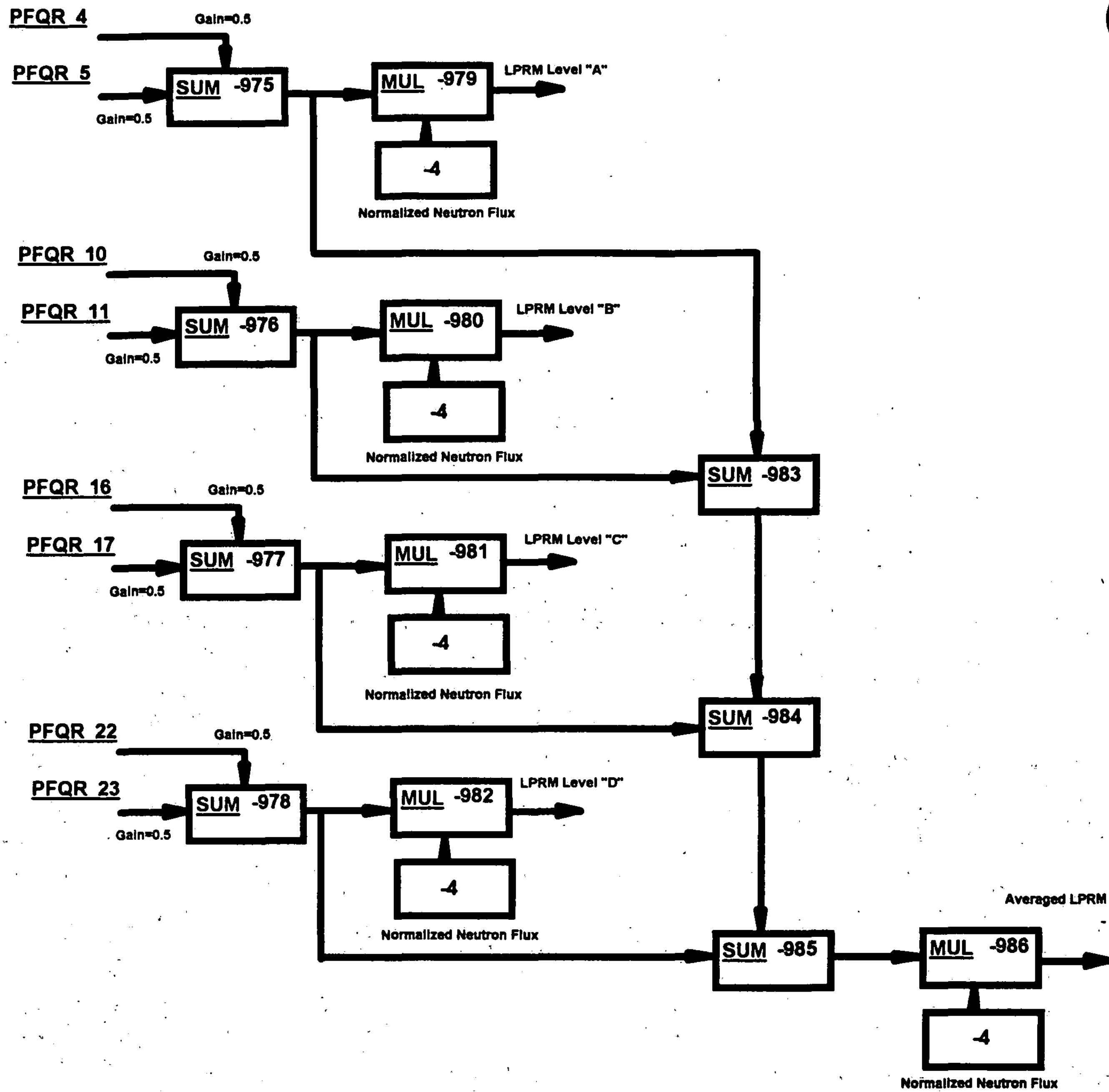

Figure 3.1-11, BWR System Model Averaged LPRM Power

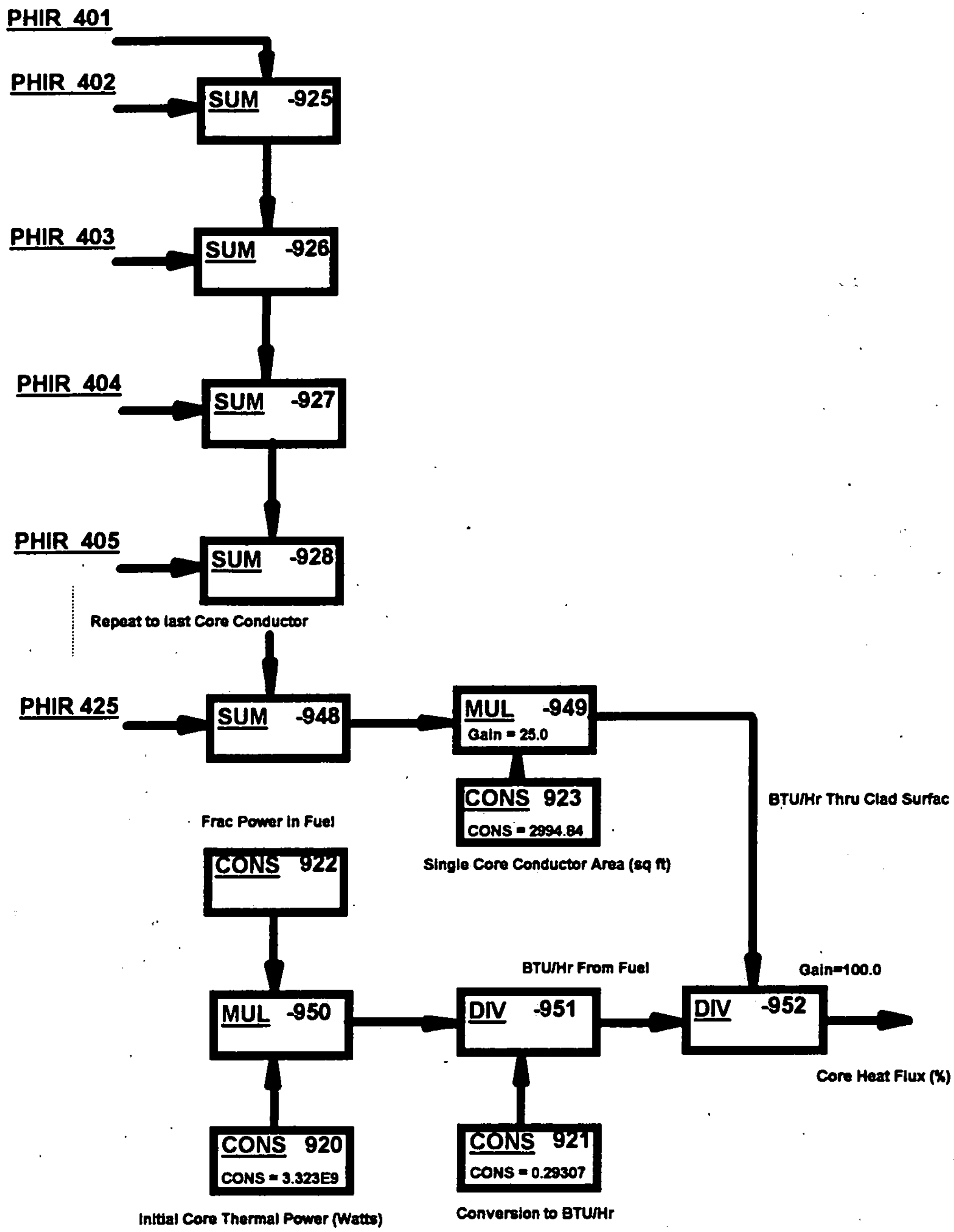

Figure 3.1-12, BWR System Model Simulated Thermal Power Monitor

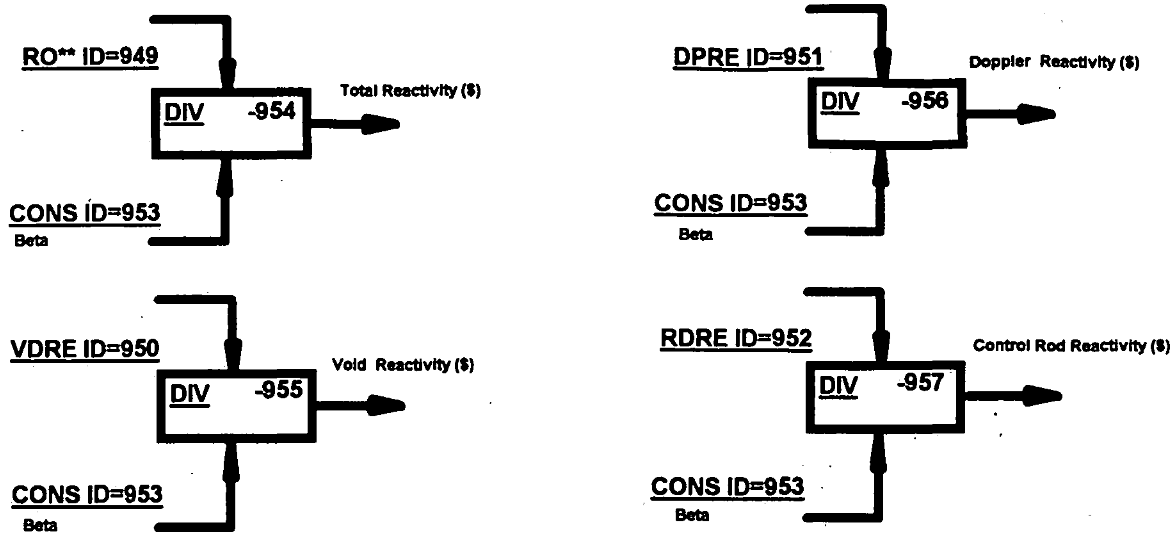

Figure 3.1-13, BWR System Model Core Reactivity Conversion to Dollars

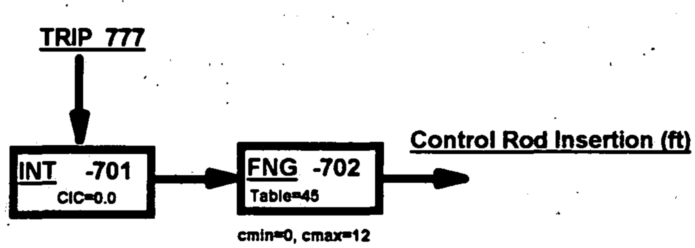

Figure 3.1-14, Control System for Calculating Control Rod Position

3.2 Quad-Cities RETRAN Model

3.2.1 System Model Nodalization and Geometry

Quad Cities used the same nodalization methodology as LaSalle and represents the Cycle 1 configuration. Quad-Cities is a General Electric BWR/3. Quad-Cities has several significant differences most notably the main steam line and the reactor recirculation system. A consistent methodology was used at ComEd to develop each of the RETRAN models starting with Peach Bottom, the LaSalle and finally Quad-Cities and Dresden. All models share the same RETRAN code options as described in Section 3.4.

Figure 3.2-1 shows the overall system model and Figure 3.2-2 shows the detailed core nodalization.

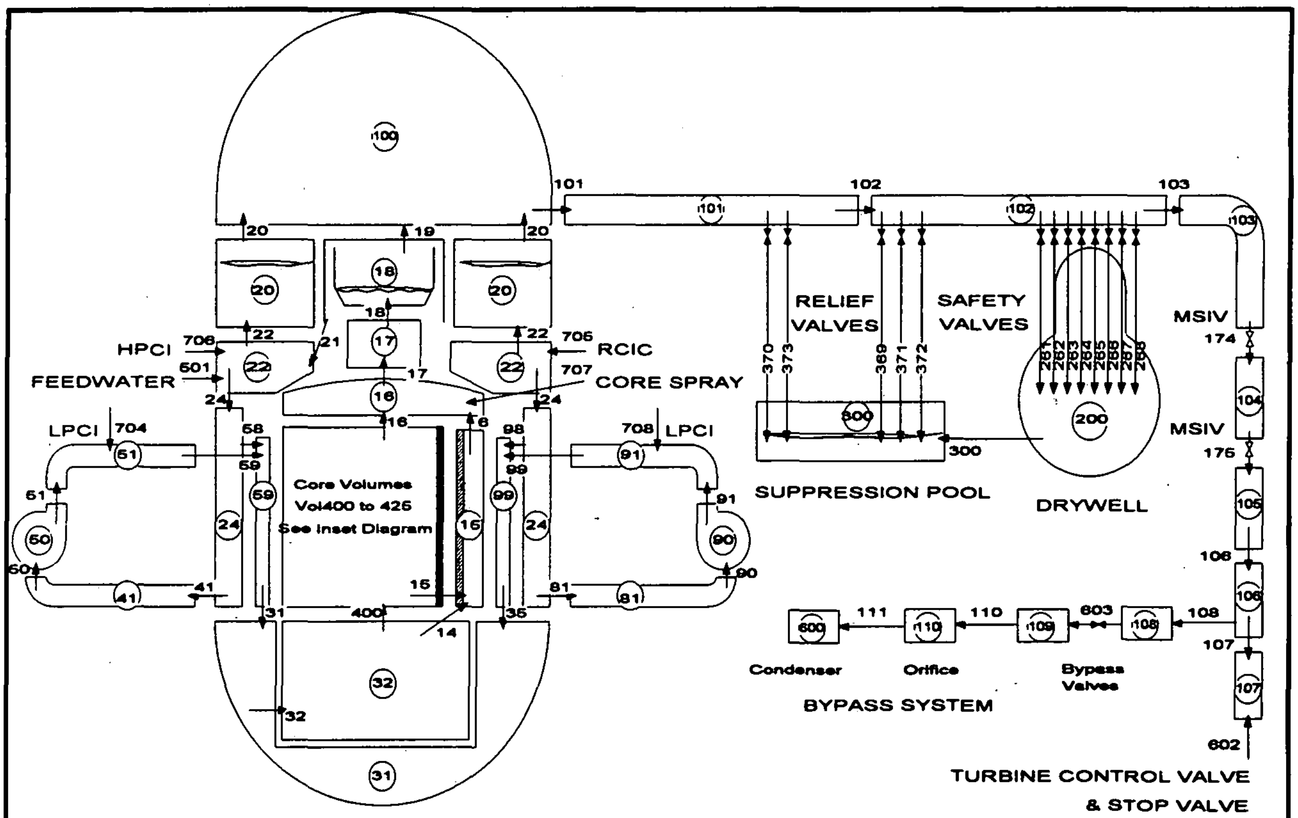

Figure 3.2-1, Quad-Cities and Dresden RETRAN Model

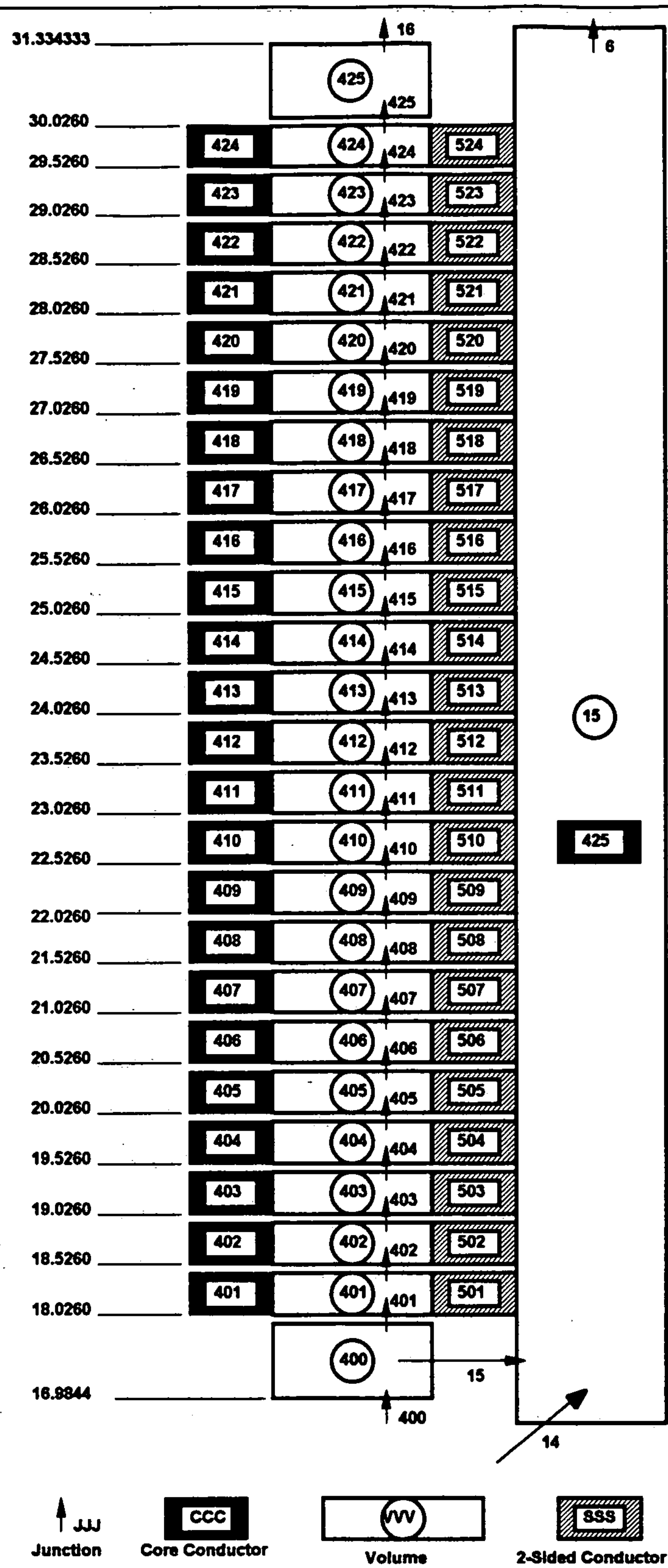

Figure 3.2-2, Quad-Cities and Dresden RETRAN Model (Core)

3.2.1.1 Vessel Internals

Quad-Cities Vessel internals were modeled consistent with the methods used for LaSalle in Section 3.1.1.1 except for the dimensional differences. Quad-Cities has 219 standpipes and separators (volumes 17 and 18 respectively) which is fewer than LaSalle. This difference slightly reduced the volume associated with these nodes. Detailed RETRAN volume and junction inputs can be found in Table 3.2-1 and Table 3.2-2.

The Quad-Cities upper downcomer node lower boundary, volume number 20, allows for a reactor water level of minus 20 inches below instrument zero up to plus 74 inches above instrument zero for narrow range reactor water level.

3.2.1.2 Core

The core is modeled as 26 volumes. These volumes represent the volume inside the shroud from the core support plate up to the bottom of the upper plenum. Where 26 axial nodes or volumes represent the core region. Of these 26 nodes, 24 represent the active core section, one node represents the lower unheated section, and one node represents the upper unheated section. The active core volumes are heated by powered heat conductors, which use information from the one-dimensional kinetics model described in Appendix A to obtain the power shape.

There is one volume modeling the core bypass region. There is also one conductor for direct gamma heating to the core bypass.

Quad-Cities conductors associated with the active core region were modeled consistent with the methods used for LaSalle in Section 3.1.1.2 except for the dimensional differences.

3.2.1.3 Recirculation Loops

Quad-Cities recirculation loops were modeled consistent with the methods used for LaSalle in Section 3.1.1.3 except for the dimensional differences and required control system changes to reflect that Quad-Cities uses a motor-generator (M-G) variable speed motor driven recirculation system. Similar to LaSalle, each recirculation loop was designed to drive ten jet pumps (which were collapsed into one volume in the RETRAN model). The Quad-Cities RETRAN jet pump was developed consistent with the methods used for LaSalle.

Unlike in LaSalle, the recirculation flow in the ComEd BWR/3s is controlled by a variable speed pump rather than a flow control valve. This M-G variable speed motor driven recirculation system is modeled in RETRAN. A RETRAN control system calculates the detailed dynamics of the M-G and the RETRAN pump options used by controlling its torque.

3.2.1.4 Steam Lines and Feedwater

The steam lines and feedwater description for Quad-Cities are similar to LaSalle (see Section 3.1.1.4). The only notable exception is that Quad-Cities has a seven node steam line model whereas LaSalle has an eight node model. Feedwater flow control for the RETRAN model is calculated by the vessel water level control system. See section 3.2.6.3 for more details on the feedwater controller.

Table 3.2-1, Quad-Cities System Model Volume Geometric Data

Volume Node #	V	ZVOL	ZM	FLOWL	FLOWA	DIAMV	ELEV	Description
15	1097.4	14.05308	14.053083	14.05308	78.0919	0.232383	17.2813	Core Bypass Flow Region
16	1043.0563	4.7073	4.7073	4.7073	232.5774	17.2083	31.3343	SHROUD HEAD RE
17	302.76	5.8907	5.8907	5.8907	43.9387	0.505425	36.0416	Separator Standpipes
18	819.936	6.2917	3.5	6.2917	108.54	0.794379	41.9323	Separator
20	1463.3153	7.974	3.50023248	7.974	127.6523	0.618629	40.25	UPPER ANNULAR
22	724.3991	5.724	5.724	5.724	309.1147	2.7751	36.2083	Middle Annular Downcomer Region
24	2719.65	26.25	26.25	26.25	103.6057	2.159442	9.9583	Lower Annular Downcomer Region
31	915.36	9.9999	9.9999	9.9999	91.5369	4.533865	0	Vessel Plenum Inlet Region
32	1200.86	13.0313	13.0313	13.0313	92.152	0.665298	4.25	Vessel Plenum outlet Region
41	194.9486	41.5937	41.5937	53.692	3.5814	2.135417	-27.0677	LOOP A Recirculation Piping Suction Side
50	39.656	4.1042	4.1042	4.1042	9.6623	2.104167	-23.6667	Loop A Recirculation Pump
51	291.3954	51.0342	51.0342	78.7413	3.7007	0.720086	-22.7188	Loop A Recirculation Piping Discharge Side
59	133.299	15.5858	15.5858	15.5858	22.3654	0.6771	9.9583	Loop A Jet Pump
81	187.5616	41.5937	41.5937	51.6294	3.5814	2.135417	-27.0677	LOOP B Recirculation Piping Suction Side
90	39.656	4.1042	4.1042	4.1042	9.6623	2.104167	-23.6667	Loop B Recirculation Pump
91	291.3954	51.0342	51.0342	78.7413	3.7007	0.720086	-22.7188	Loop B Recirculation Piping Discharge Side
99	133.299	15.5858	15.5858	15.5858	22.3654	0.6771	9.9583	Loop B Jet Pump
100	5602.1987	20.4114	20.4114	20.4114	274.4642	20.9167	48.224	Steam Dome
101	387.0154	45.0107	45.0107	55.0181	7.02	1.494833	7.4193	MAIN STEAM LINE
102	74.0259	1.4948	1.4948	10.5451	7.02	1.494833	7.4193	MAIN STEAM LINE
103	250.1334	22.8282	22.8282	35.6317	7.02	1.494833	-13.9141	MAIN STEAM LINE
104	165.3321	1.4948	1.4948	23.5517	7.02	1.494833	-13.9141	MAIN STEAM LINE
105	240.9131	20.3984	20.3984	24.7345	10.143	1.796833	-14.0651	MAIN STEAM LINE
106	217.6924	0.8984	0.8984	21.4624	10.143	1.796833	5.4349	MAIN STEAM LINE
107	188.3436	17.2968	17.2968	18.5689	10.143	1.796833	5.4349	MAIN STEAM LINE
108	165.94	13.8552	13.8552	38.2487	4.3385	1.91	5.1802	BYPS HDR
109	402.109	49.0997	49.0997	140.8945	2.854	0.635417	-30.0644	BYPS LINE
110	17.34	0.9	0.9	3.042	5.7	0.9	-30.1967	ORIFICE
200	158236	100	100	0	900	34	7.4193	DRYWELL
300	229448	30	14.67	0	7648.267	98.6817	7.4193	SUPP POOL

Table 3.2-1, Quad-Cities System Model Volume Geometric Data (Continued)

Volume Node #	V	ZVOL	ZM	FLOWL	FLOWA	DIAMV	ELEV	Description
400	51.6842	0.992192	0.992192	0.992192	24.1867	0.184354	17.03381	CORE INLET REFLE
401	38.7265	0.5	0.5	0.5	77.4529	0.047725	18.026	1ST CORE VOLUME
402	38.7265	0.5	0.5	0.5	77.4529	0.047725	18.526	2ND CORE VOLUME
403	38.7265	0.5	0.5	0.5	77.4529	0.047725	19.026	3RD CORE VOLUME
404	38.7265	0.5	0.5	0.5	77.4529	0.047725	19.526	4TH CORE VOLUME
405	38.7265	0.5	0.5	0.5	77.4529	0.047725	20.026	5TH CORE VOLUME
406	38.7265	0.5	0.5	0.5	77.4529	0.047725	20.526	6TH CORE VOLUME
407	38.7265	0.5	0.5	0.5	77.4529	0.047725	21.026	7TH CORE VOLUME
408	38.7265	0.5	0.5	0.5	77.4529	0.047725	21.526	8TH CORE VOLUME
409	38.7265	0.5	0.5	0.5	77.4529	0.047725	22.026	9TH CORE VOLUME
410	38.7265	0.5	0.5	0.5	77.4529	0.047725	22.526	10TH CORE VOLUME
411	38.7265	0.5	0.5	0.5	77.4529	0.047725	23.026	11TH CORE VOLUME
412	38.7265	0.5	0.5	0.5	77.4529	0.047725	23.526	12TH CORE VOLUME
413	38.7265	0.5	0.5	0.5	77.4529	0.047725	24.026	13TH CORE VOLUME
414	38.7265	0.5	0.5	0.5	77.4529	0.047725	24.526	14TH CORE VOLUME
415	38.7265	0.5	0.5	0.5	77.4529	0.047725	25.026	15TH CORE VOLUME
416	38.7265	0.5	0.5	0.5	77.4529	0.047725	25.526	16TH CORE VOLUME
417	38.7265	0.5	0.5	0.5	77.4529	0.047725	26.026	17TH CORE VOLUME
418	38.7265	0.5	0.5	0.5	77.4529	0.047725	26.526	18TH CORE VOLUME
419	38.7265	0.5	0.5	0.5	77.4529	0.047725	27.026	19TH CORE VOLUME
420	38.7265	0.5	0.5	0.5	77.4529	0.047725	27.526	20TH CORE VOLUME
421	38.7265	0.5	0.5	0.5	77.4529	0.047725	28.026	21ST CORE VOLUME
422	38.7265	0.5	0.5	0.5	77.4529	0.047725	28.526	22ND CORE VOLUME
423	38.7265	0.5	0.5	0.5	77.4529	0.047725	29.026	23RD CORE VOLUME
424	38.7265	0.5	0.5	0.5	77.4529	0.047725	29.526	24TH CORE VOLUME
425	110.9132	1.308333	1.308333	1.308333	77.4528	0.046617	30.026	CORE OUTLET RE
600	103000	100	10.68	0	900	34	-42	CONDENSER

Table 3.2-2, Quad-Cities System Model Junction Geometric Data

Junction Node #	WP	AJUN	ZJUN	INERTA	FJUNF	FJUNR	DIAMJ	CONCO	Description
6	2994.4444	78.0919	31.3343	0	0	0	0.000000	0.0000	CORE BYPASS EXIT
14	974.6916	10	17.2813	0	-1	0	0.000000	0.0000	CORE PLATE BYPASS FL
15	2019.7528	10	17.5341	0	-1	0	0.000000	0.0000	CORE BYPASS FLOW REG
16	24227.7778	139.4203	31.3343	0	0.952	0	0.283601	0.0000	SHROUD HEAD REGION
17	27222.2222	43.9387	36.0416	0.07719	0.5	1	0.505425	0.0000	Separator Standpipes
18	27222.2222	43.9387	41.9323	0.38218	-1	0	0.505425	0.0000	Separator
19	2710.8333	58.5285	48.224	0.06617	1.214	1.214	0.583333	0.0000	Dryer Region
20	0	127.6523	48.224	0.06842	0	0	0.618629	0.0000	Upper Annular Downcomer region
21	24511.3889	142.346	41.9323	0.38218	-1	0	0.228957	0.0000	Annular Downcomer region
22	0	309.1147	40.25	0.04049	0	0	2.775100	0.0000	Middle Annular Downcomer region
24	27222.2222	309.1147	36.2083	0.1414	-1	1	2.070186	0.0000	Lower Annular Downcomer region
31	0	22.3654	9.999	0	3.211875	0	1.687500	0.0000	Jet Pump A exit to Vess Plenum Inlet
32	27222.2222	200.8259	4.25	0	0	0	4.818500	0.0000	VESSEL PLENUM OUTLET
35	0	22.3654	9.999	0	3.211875	0	1.687500	0.0000	Jet Pump B exit to Vessel Plenum Inlet region
41	0	3.5814	13.4583	7.61562	0.23	1	2.135417	0.0000	LOOP A RECIRCUL
50	0	3.5814	-23.6667	0	0	0	2.135417	0.0000	LOOP A RECIRCUL
51	0	3.4774	-21.6667	0	-1	0	2.104167	0.0000	LOOP A RECIRCUL
58	8322.4811	2.9	25.5441	0.40212	0.06	1	0.363334	0.0000	Loop A Jet Pump Mixing Flow
59	5288.63	0.67	25.5441	10.9871	0.2	0.5	0.275833	0.0000	Loop A Jet Pump
81	0	3.5814	13.4583	7.61562	0.23	1	2.135417	0.0000	LOOP B RECIRCUL
90	0	3.5814	-23.6667	0	0	0	2.135417	0.0000	LOOP B RECIRCUL
91	0	3.4774	-21.6667	0	-1	0	2.104167	0.0000	LOOP B RECIRCUL
98	8322.4811	2.9	25.5441	0.40212	0.06	1	0.363334	0.0000	Loop B Jet Pump Mixing Flow
99	5288.63	0.67	25.5441	10.9871	0.2	0.5	0.275833	0.0000	Loop B Jet Pump
101	2710.8333	7.02	51.6667	0	0.321	0	0.000000	0.0000	REACTOR VESSEL STEAM
102	0	7.02	8.1667	0	0.5946	0	0.000000	0.0000	MAIN STEAM LINE UPST
103	0	7.02	8.1667	0	0.2776	0	0.000000	0.0000	MAIN STEAM LINE DOWN
106	0	10.143	6.3333	0	0.2542	0	0.000000	0.0000	MAIN STEAM LINE DOWN
107	0	10.143	6.3333	0	1.9037	0	0.000000	0.0000	MAIN STEAM LINE UPST
108	0	10.143	6.3333	0	4.396	0	0.000000	1.0000	BYPASS LINE
110	0	0.725	-29.7467	0	3.176	0	0.000000	0.0000	ORIFICE IN
111	0	1.46	-29.7467	0	6.3771	0	0.000000	0.0000	CONDENS IN
174	0	7.02	-13.1667	0	3.3063	0	0.000000	0.0000	INBOARD MAIN STEAM I
175	0	10.143	-13.1667	0	2.6476	0	0.000000	0.0000	OUTBOARD MAIN STEAM
261	0	0.1963	8.1666	0.04849	0	0	0.500000	0.3559	SAFETY VALVE (203-4A
262	0	0.1963	8.1666	0.40462	0	0	0.500000	0.3559	SAFETY VALVE (203-4B
263	0	0.1963	8.1666	1.11801	0	0	0.500000	0.3559	SAFETY VALVE (203-4C
264	0	0.1963	8.1666	1.47414	0	0	0.500000	0.3559	SAFETY VALVE (203-4D
265	0	0.1963	8.1666	0.93595	0	0	0.500000	0.3559	SAFETY VALVE (203-4E
266	0	0.1963	8.1666	1.29207	0	0	0.500000	0.3559	SAFETY VALVE (203-4F
267	0	0.1963	8.1666	0.04706	0	0	0.500000	0.3559	SAFETY VALVE (203-4G
268	0	0.1963	8.1666	0.40318	0	0	0.500000	0.3559	SAFETY VALVE (203-4H

Table 3.2-2, Quad-Cities System Model Junction Geometric Data (Continued)

Junction Node #	WP	AJUN	ZJUN	INERTA	FJUNF	FJUNR	DIAMJ	CONCO	Description
300	0	100	25	0	1	0.5	10.000000	0.0000	DRYWELL VENT TO SUPPL
369	0	0.1963	8.1666	0.76074	0	0	0.500000	0.3820	TARGET ROCK VALVE (2
370	0	0.1963	8.1666	3.82074	0	0	0.500000	0.3433	RELIEF VALVE (203-3B
371	0	0.1963	8.1666	0.42142	0	0	0.500000	0.3433	RELIEF VALVE (203-3C
372	0	0.1963	8.1666	0.11575	0	0	0.500000	0.3433	RELIEF VALVE (203-3D
373	0	0.1963	8.1666	3.85236	0	0	0.500000	0.3433	RELIEF VALVE (203-3E
400	26247.5306	77.4529	17.03381	0	-1	0	0.000000	0.0000	CORE INLET REFLECTOR
401	24227.7778	77.4529	18.026	0	-1	0	0.000000	0.0000	1ST CORE FLOW IN LOW
402	0	77.4529	18.526	0	-1	0	0.000000	0.0000	2ND CORE FLOW IN LOW
403	0	77.4529	19.026	0	-1	0	0.000000	0.0000	3RD CORE FLOW IN LOW
404	0	77.4529	19.526	0	-1	0	0.000000	0.0000	4TH CORE FLOW IN LOW
405	0	77.4529	20.026	0	-1	0	0.000000	0.0000	5TH CORE FLOW IN LOW
406	0	77.4529	20.526	0	-1	0	0.000000	0.0000	6TH CORE FLOW IN LOW
407	0	77.4529	21.026	0	-1	0	0.000000	0.0000	7TH CORE FLOW IN LOW
408	0	77.4529	21.526	0	-1	0	0.000000	0.0000	8TH CORE FLOW IN LOW
409	0	77.4529	22.026	0	-1	0	0.000000	0.0000	9TH CORE FLOW IN MID
410	0	77.4529	22.526	0	-1	0	0.000000	0.0000	10TH CORE FLOW IN MI
411	0	77.4529	23.026	0	-1	0	0.000000	0.0000	11TH CORE FLOW IN MI
412	0	77.4529	23.526	0	-1	0	0.000000	0.0000	12TH CORE FLOW IN MI
413	0	77.4529	24.026	0	-1	0	0.000000	0.0000	13TH CORE FLOW IN MI
414	0	77.4529	24.526	0	-1	0	0.000000	0.0000	14TH CORE FLOW IN MI
415	0	77.4529	25.026	0	-1	0	0.000000	0.0000	15TH CORE FLOW IN MI
416	0	77.4529	25.526	0	-1	0	0.000000	0.0000	16TH CORE FLOW IN MI
417	0	77.4529	26.026	0	-1	0	0.000000	0.0000	17TH CORE FLOW IN U
418	0	77.4529	26.526	0	-1	0	0.000000	0.0000	18TH CORE FLOW IN UP
419	0	77.4529	27.026	0	-1	0	0.000000	0.0000	19TH CORE FLOW IN UP
420	0	77.4529	27.526	0	-1	0	0.000000	0.0000	20TH CORE FLOW IN UP
421	0	77.4529	28.026	0	-1	0	0.000000	0.0000	21ST CORE FLOW IN UP
422	0	77.4529	28.526	0	-1	0	0.000000	0.0000	22ND CORE FLOW IN UP
423	0	77.4529	29.026	0	-1	0	0.000000	0.0000	23RD CORE FLOW IN UP
424	0	77.4529	29.526	0	-1	0	0.000000	0.0000	24TH CORE FLOW IN UP
425	0	77.4529	30.026	0	-1	0	0.000000	0.0000	CORE OUTLET REFLECTO
501	2710.8333	2.6827	40.5	0	0	0	1.000000	0.0000	FEED WATER FLOW
602	-2710.8333	1	21.8333	0	0	0	1.000000	0.0000	TURBINE STEAM FLOW
603	0	0.897	19.0353	0	3.4	0	0.000000	0.0000	BYPASS VALVE
610	0	1	8.1633	0	0	0	1.000000	0.0000	HPCI TURBINE STEAM SU
704	0	1.2272	-2.6667	0	0	0	1.250000	0.0000	LOW PRESSURE COOLAN
705	0	2.6827	40.5	0	0	0	0.924083	0.0000	REACTOR CORE ISOLATI
706	0	2.6827	40.5	0	0	0	0.924083	0.0000	HIGH PRESSURE COOLAN
707	0	0.6948	32.3958	0	0	0	0.665083	0.0000	CORE SPRAY FLOW
708	0	1.2272	-2.6667	0	0	0	1.250000	0.0000	LOW PRESSURE COOLAN

Table 3.2-3, Quad-Cities Heat Conductor Geometric Data

Heat Conductor #	ASUL	ASUR	VOLS	HDML	HDMR	DHEL	DHER	CHNL	CHNR	Description
109	2314.46	0	102.76	0	0	0	0	0	0	BYPASS LINE
401	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	1ST CORE CONDUCTOR IN
402	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	2ND CORE CONDUCTOR IN
403	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	3RD CORE CONDUCTOR IN
404	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	4TH CORE CONDUCTOR IN
405	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	5TH CORE CONDUCTOR IN
406	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	6TH CORE CONDUCTOR IN
407	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	7TH CORE CONDUCTOR IN
408	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	8TH CORE CONDUCTOR IN
409	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	9TH CORE CONDUCTOR IN
410	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	10TH CORE CONDUCTOR I
411	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	11TH CORE CONDUCTOR I
412	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	12TH CORE CONDUCTOR I
413	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	13TH CORE CONDUCTOR I
414	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	14TH CORE CONDUCTOR I
415	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	15TH CORE CONDUCTOR I
416	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	16TH CORE CONDUCTOR I
417	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	17TH CORE CONDUCTOR I
418	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	18TH CORE CONDUCTOR I
419	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	19TH CORE CONDUCTOR I
420	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	20TH CORE CONDUCTOR I
421	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	21ST CORE CONDUCTOR I
422	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	22ND CORE CONDUCTOR I
423	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	23RD CORE CONDUCTOR I
424	0	2614.46	30.67	0	0.047725	0	0.05975	0	0.5	24TH CORE CONDUCTOR I
425	0	2614.46	1	0	0.232383	0	0.263769	0	12	BYPASS CORE CONDUCTOR

Table 3.2-3, Quad-Cities Heat Conductor Geometric Data (Continued)

Heat Conductor #	ASUL	ASUR	VOLS	HDML	HDMR	DHEL	DHER	CHNL	CHNR	Description
501	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	1ST CORE BYPASS COND
502	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	2ND CORE BYPASS COND
503	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	3RD CORE BYPASS COND
504	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	4TH CORE BYPASS COND
505	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	5TH CORE BYPASS COND
506	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	6TH CORE BYPASS COND
507	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	7TH CORE BYPASS COND
508	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	8TH CORE BYPASS COND
509	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	9TH CORE BYPASS COND
510	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	10TH CORE BYPASS CON
511	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	11TH CORE BYPASS CON
512	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	12TH CORE BYPASS CON
513	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	13TH CORE BYPASS CON
514	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	14TH CORE BYPASS CON
515	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	15TH CORE BYPASS CON
516	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	16TH CORE BYPASS CON
517	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	17TH CORE BYPASS CON
518	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	18TH CORE BYPASS CON
519	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	19TH CORE BYPASS CON
520	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	20TH CORE BYPASS CON
521	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	21ST CORE BYPASS CON
522	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	22ND CORE BYPASS CON
523	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	23RD CORE BYPASS CON
524	617.2	632.35	4.17	0.047725	0.232383	0.250982	0.263769	0.5	0.5	24TH CORE BYPASS CON

3.2.2 System Component Models

3.2.2.1 Recirculation Pumps

The Quad-Cities model used the RETRAN02 pump component model with variable speed and torque for its two reactor recirculation pumps. Core flow is modulated with recirculation pump torque calculated by the recirculation control system. Manufacturer data was used to determine the pumps head, flow, speed and torque characteristics for the homologous pump curves in RETRAN.

3.2.2.2 Jet Pumps

Reactor recirculation pumps at their driven speeds regulate drive flow through a header which curves horizontally around the reactor vessel and splits the discharge flow of each recirculation pump into 5 separate risers. Each riser has individual penetrations into the reactor vessel. Inside the vessel, the recirculation riser pipe takes the recirculation flow up to a rams head. Each ramshead drives two jet pump nozzles with suction coming from the surrounding downcomer fluid. Quad-Cities and Dresden share identical jet pump and reactor recirculation designs. Their RETRAN jet pump performance was tested and verified against measured data. Results can be seen in Figure 3.2-3 and Figure 3.2-4 and apply equally to Quad-Cities and Dresden. Detailed RETRAN volume and junction inputs for the jet pumps can be found in Table 3.2-1 and Table 3.2-2. Cycle specific jet pump and recirculation system performance, as evidenced by rated condition motor-generator (MG) speeds and recirculation drive flows, with its effect on transient analysis will be addressed in the subsequent reload applications report.

3.2.2.3 Steam Separators and Dryer

Quad-Cities has 219 separators that sit atop the core shroud. Quad-Cities separators and dryers were modeled consistent with the methods used for LaSalle described in Section 3.1.2.3 except for the dimensional differences. Detailed RETRAN volume and junction inputs for the separators can be found in Table 3.2-1 and Table 3.2-2.

The dryer pressure drop for Quad-Cities is also 7.0 inches of water at rated steam flow and reactor pressure saturated conditions. The Quad-Cities model also accounts for the water level difference inside and outside the dryer skirt with the water level indication portion of the feedwater control system described in section 3.2.3.6.

3.2.2.4 Safety/Relief Valves

The Quad-Cities RETRAN model has each valve modeled as a separate flow junction using the valve component model. Quad-Cities has eight spring safety valves (SSV) mounted on the four main steam lines upstream of the MSIVs. Each of these valves discharge directly to the drywell. Each of the valves has only a safety function. The safety function is used under accident conditions of extreme over-pressurization. This RETRAN SSV model actuates when the local steam line volume pressure reaches the SSV's setpoint. The RETRAN model also has four relief valves (RV). The relief function is actuated by a trip setpoint on sensed reactor dome pressure consistent with the plant design. RVs at the plant actuate under normal operating conditions utilizing non-safety grade power supplies. The one Target-Rock dual function safety relief valve (SRV) functions in the same way as one of the LaSalle SRVs function as described in Section 3.1.2.4. The four RVs and the Target-Rock SRV also perform the safety related function of the automatic depressurization system (ADS). ComEd models each valve individually for each of these functions described above. Detailed RETRAN junction inputs for the SSVs, RVs and SRVs can be found in Table 3.2-2. They are modeled as a valve junction from the steam line volumes 101 and 102 to the suppression pool volume 300 for the piped Relief valves and to the drywell volume 200 for the Safety Valves. ComEd sizes the contraction coefficients for these junctions number 261 through 268 and 369 through 373 to achieve the flow capacity at the rated pressure as specified in the Technical Specifications and ASME certification. Delay times for response and stroke times were set to be consistent with available performance data.

3.2.2.5 Turbine Stop, Control, and Bypass Valves

The Quad-Cities turbine stop valve and turbine control valve are similar to LaSalle (see Section 3.1.2.5). Quad-Cities main steam bypass valves (BPVs) are sized to allow their full flow capacity to be 40% of the rated reactor steam flow. In the model, the BPVs junction number 603 area is controlled by the pressure control system. That control system is described in section 3.2.6.2.

3.2.2.6 Main Steam Isolation Valves

The Quad-Cities main steam isolation valves (MSIVs) are similar to LaSalle (see Section 3.1.2.6). Detailed RETRAN junction inputs for the MSIVs can be found in Table 3.2-2.

3.2.2.7 Core Hydraulics

Quad-Cities was modeled consistent with the methods used for LaSalle in Section 3.1.2.7 except for the dimensional differences. Figure 3.2-2 shows the details of the core nodalization. Detailed RETRAN volume and junction inputs for the Quad-Cities Cycle 1 core can be found in Table 3.2-1 and Table 3.2-2.

Quad-Cities has 24 core conductors and one core bypass conductor. This is different from LaSalle's 25 core conductors. Quad-Cities active fuel length is approximately 12.0 feet whereas LaSalle's active fuel length is 12.5 feet. ComEd's approved nuclear design methodology in Reference 1 models Quad-Cities active fuel regions as 24 nodes and LaSalle's active fuel region as 25 nodes. Since this Reference 1 methodology is used in conjunction with RETRAN02 models, as described in Appendix A, the number of RETRAN core conductors was set equal to the number of core nodes in the simulator code MICROBURN. Detailed RETRAN conductor inputs for the Quad-Cities Cycle 1 core can be found in Table 3.2-3. The core conductors were modeled consistent with LaSalle (see Section 3.1.2.7) with the exception that Quad-Cities has one less active core volume, active core junction, and core conductor.

3.2.3 Trip Logic

Each of the categories of RETRAN trip inputs was summarized here. The RETRAN input cards require that each trip have an 'ID"number associated with it. These trip ID number assignments to their various functions are strictly arbitrary just as volume node numbers and junction node are arbitrary . The "Trip ID" numbers that appear in these summaries correspond to their assignments in the Quad-Cities RETRAN model inputs may or may not be related to the control of its associated junction or volume node number.

3.2.3.1 Reactor Protection System Trips

Reactor protection system trips result in control rod insertion into the core. This is accomplished through a control system and a general data table in the RETRAN model. The RETRAN model predicts a reactor scram (Trip ID 3) for the following:

1) High neutron flux	(Trip ID 10)
2) High steam dome pressure	(Trip ID 11)
3) Reactor vessel water level, low level 3	(Trip ID 18)
4) Main steamline isolation valve closure	(Trip ID 174 or 175)
5) High drywell pressure *	(Trip ID 20)
6) Turbine stop valve closure	(Trip ID 178)
7) Generator load rejection	(Trip ID 602)
8) Manual scram on elapsed time	(Trip ID 3)

* The high drywell pressure scram is included in the trip system, but will not be used since RETRAN will not necessarily predict a realistic drywell response.

3.2.3.2 Safety/Relief and Main Steam Isolation Valve Trips

Thirteen valve junctions are used to model the relief, safety and Target Rock SRVs. The relief valve setpoint (Trip IDs 370 - 373) is taken off the dome pressure. The safety valve setpoint (Trip IDs 261 - 268) is taken off the steamline pressure in the volume where the safety valves are located. The Target Rock SRV setpoint (Trip ID 369) is taken off the steam dome pressure. The four relief valves and the Target Rock SRV will open coincident on an automatic depressurization system trip actuation (Trip ID 409). There is also the capability of opening up to four relief valves manually on elapsed time.

The MSIVs will close if the following trips are actuated:

1) High main steamline flow rate	(Trip ID 13)
2) Low main steamline pressure	(Trip ID 14)
3) Reactor water level, low level 1	(Trip ID 17)
4) Manual trip on elapsed time	(Trip IDS 174 and/or 175)

3.2.3.3 Recirculation Pump Trips

There are two recirculation pumps modeled for Quad-Cities. These pumps each have independent trip logic. A recirculation pump trip will actuate on receipt of any of the following:

1) Turbine trip	(Trip ID 178)
2) High steam dome pressure	(Trip ID 27)
3) Manual trip on elapsed time	(Trip IDs 50 and/or 90)
4) High drywell pressure	(Trip ID 20)

3.2.3.4 Turbine Trip, and Generator Load Rejection

The Turbine Trip (Trip ID 178) actuates on one of two signals: reactor water level, high level 8 (Trip ID 15) or a manual trip on elapsed time (Trip ID 178). In the model, the generator load rejection actuates only by a trip on elapsed time (Trip ID 602).

3.2.3.5 LPCI, CS, HPCI and RCIC Trips

Both LPCI (Trip ID 404) and CS (Trip ID 407) are initiated on a high drywell pressure (Trip ID 20). LPCI and CS cannot actuate until the reactor water level, low level 2 (Trip ID 17) and low pressure interlock (Trip ID 12) have occurred. HPCI (Trip ID 406) and RCIC (Trip ID 405) both actuate on a reactor water level, low level 2 (Trip ID 18) signal. HPCI also actuates on high drywell pressure (Trip ID 20). All four systems also allow for a manual initiation on elapsed time.

3.2.4 Direct Bypass Heating

The Quad-Cities modeling of direct bypass heating is consistent with the LaSalle modeling as described in Section 3.1.4.

3.2.5 Fill Tables and Associated Valve Controls

The Quad-Cities RETRAN model has only six fill junctions, the two most important for transient modeling are the feedwater positive fill and the main steam flow negative fill from the TCV. These were described in section 3.2.1.4 and is similar to the corresponding LaSalle section 3.1.1.4. The other positive fills represent the ECCS functions for LPCI, HPCI and RCIC. The ECCS fills do not actuate for any of the benchmark analyses in Section 4. However, their inclusion in the model could be used to analyze a non-limiting transient. Typically, inadvertent ECCS actuation is bounded by feed water controller failure licensing cases and are not required for reload transient analysis applications.

The Quad-Cities RETRAN model has four types of valves. The characteristics of all four valve types are modeled. They are the SSVs and RVs, the MSIVs and the BPVs. These were described in sections 3.2.2.4, 3.2.2.6 and 3.2.2.5 respectively.

3.2.6 Control Logic

3.2.6.1 Sensor Response Models

ComEd used control systems to calculated the sensed plant variables which were typically used in the comparisons with startup test data. These control blocks have no direct impact on the model as they are used to provide edits to compare RETRAN variables to actual plant variables. Some of these variables include; dome pressure, turbine inlet pressure, core flow, feedwater flow, and steam flow.

3.2.6.2 Pressure Regulator

The pressure control system inputs are listed in Table 3.2-4. A block diagram of the pressure control system is shown on Figure 3.2-7. The pressure control system is functionally the same as LaSalle (see Section 3.1.6.2).

3.2.6.3 Feedwater Controller

The RETRAN control block diagram is shown on Figure 3.2-5. This control system is functionally the same as LaSalle (see Section 3.1.6.3) with the following exception. The % feedwater demand signal is sent to actuate feed water regulating valve movement rather than turbine driven feedwater pump speed.

3.2.6.4 Recirculation Controller

The recirculation control (RC) system inputs are listed in Table 3.2-4. The RETRAN control system diagram of the whole recirculation control system is depicted on Figure 3.2-6. The RC system is designed along with the pressure and level control systems to ensure that the plant can meet maneuverability requirements. The variation in recirculation control is achieved through variation in the frequency of the power supplied to the pump motor. The pump motor power supply is taken from a generator driven by a constant speed motor, however, this motor and generator (M-G) are connected through a variable hydraulic coupler via the "scoop tube" positioning arm. By controlling the slip between the drive motor and the generator with the scoop tube, the output of the generator can be varied. The fluid coupler responds to demands placed on it by the operator or by the load error generated in the Pressure Control (see Figure 3.2-7) system. The demand to the fluid coupler changes the position of the scoop tube to vary coupling between the drive motor and the generator to produce the required pump speed and core flow.

The recirculation controller consists of a master controller and a speed controller. Both controllers are (P+I) controllers. The master controller takes a load error signal generated in the pressure control system. The load error signal is then processed through a P+I controller to obtain a % increase in speed demand. The speed demand signal is then sent to another summing block to obtain a % speed error between the speed demand and generator speed. The speed error is processed through the P+I speed controller to obtain the coupling demand. The coupling demand is processed through a function generator to linearize the control process. A delay is modeled to account for the scoop tube positioning process.

The output of the scoop tube dynamics model is a % coupling term. This term is used along with the slip between the drive motor and generator to obtain the coupler output torque. The coupler torque is then used in a torque balance (blocks -431 and -434) on the drive motor side as well as the generator side of the coupler. The differential torque is then integrated (blocks -432 and -435) to obtain a corrected drive motor and generator speeds.

The calculated drive motor and generator speeds are subtracted from the sync speed and pump speed respectively (see blocks -426 and -437) to obtain a slip term. The slip term is used as input to a function generator (blocks -427 and -438) to obtain a new drive motor or pump torque.

The calculated pump torque (block -440) is sent to the RETRAN pump model and the calculated pump torque is used to calculate the generator torque.

3.2.6.5 Normalized Neutron Flux Monitor

Both Quad-Cities and Dresden share the same control system configuration with LaSalle. Figure 3.1-10 describes the control systems used for modeling the normalized neutron flux that is measured by the nuclear instrumentation system. The high reactor power trip logic utilized the normalized neutron flux output of control block ID = -4. Logic for the flow biased scram was not modeled since it is not credited for any of the plant events. But, neutron flux is required for the reactor protection system trips described in Section 3.2.3.1.

3.2.6.6 Local Power Range Monitor

Quad-Cities shares the same control system configuration with LaSalle as described in Section 3.1.6.6. Figure 3.1-11 depicts the control systems used for modeling the LPRM. The control block ID's may be different than indicated in Figure 3.1.-11, as these control block IDs are LaSalle specific.

3.2.6.7 Core Average Heat Flux

Quad-Cities shares the same control system configuration with LaSalle as described in Section 3.1.6.7. Figure 3.1-12 depicts ComEd's RETRAN control system used to calculate core average heat flux. The control block IDs may be different than indicated in Figure 3.1-12, as these control block IDs are LaSalle specific.

3.2.6.8 Miscellaneous Control Systems

Figure 3.2-8 depicts the control systems used for calculating the total mass flow rate for the safety/relief valves. It sums up all of the flows in pounds mass per second.

Quad-Cities shares the same reactivity control system configuration with LaSalle. Figure 3.1-13 depicts the control systems used for calculating reactivities in dollars for the total and component reactivities. The control block IDs may be different than indicated in Figure 3.1-13, as these control block IDs are LaSalle specific.

Quad-Cities shares the same rod position control system configuration with LaSalle. Figure 3.1-14 depicts the control systems used for modeling the control rod position. This control system indicates the length in feet of the control rods insertion into the active core length.

Table 3.2-4, Quad-Cities and Dresden RETRAN Control Input Descriptions

Control Input ID	Description of Control Block Input
1	Steam Dome Volume 100 Pressure
2	Main Steam Line Volume 105 Pressure
3	Turbine Inlet Volume 107 Pressure
4	Normalized Power
5	Constant Decay Heat Fraction: I=6
200	Sensed Liquid Volume in Downcomer Volume 20
201	Sensed Liquid Volume in Downcomer Volume 22
202	Sensed Liquid Volume in Downcomer Volume 24
203	Sensed Steam Flow in Junction 103
204	Trip 3 (Scram Status)
205	Trip 650 (3 Element/ 1-Element Level Control)
206	Reactor Dome Pressure to which FW System Discharges
207	Sensed Liquid Volume in Separator Volume 18
208	Constant Initial Separator Liquid Volume
209	Steam Flow in Separator Junction 19
210	Constant Rated Separator Steam Flow
211	VALUE OF EXPONENT
300	Sensed pressure in junction 107
301	Pressure Regulator Setpoint 1
302	Pressure Regulator Setpoint 2
303	Load Set
304	Setpoint Adjuster/Recirc Control Bias
305	Trip 651 (Manual/Auto Recirc Control Switch)
306	Maximum Combined Flow Limit
307	Bypass Valve Negative Bias - (Deadband)
308	Problem Simulation Time
309	Trip Time of Trip 178
310	Trip Time of Trip 602
311	TURBINE CONTROL VALVE CLOSURE
312	TURBINE STOP VALVE CLOSURE
400	Trip 651, M/A Master Recirc. Control
401	Trip 652, M/A Speed Control
402	Elapsed simulation time
403	Constant used in empirical coupler torque equation
404	Constant used in empirical coupler torque equation
405	Trip 653 MG trip
406	Pump speed
407	Trip 50 or trip 90 for recirc pump trip
408	Rated Recirculation Pump Torque
409	Constant C2 in Coupler Torque Equation
410	Constant (unity)
701	Control Rod Drive Position Monitor
888	RETRAN CPU Timed Used
897	1 NODE HEAT TRANSFER AREA
898	TOTAL CORE POWER
899	CONVERSION: W/(BTU/HR)

Control Input ID	Description of Control Block Input
900	FRACTION OF POWER GENERATED IN FUEL
901	Heat Flux Core Conductor 401
902	Heat Flux Core Conductor 402
903	Heat Flux Core Conductor 403
904	Heat Flux Core Conductor 404
905	Heat Flux Core Conductor 405
906	Heat Flux Core Conductor 406
907	Heat Flux Core Conductor 407
908	Heat Flux Core Conductor 408
909	Heat Flux Core Conductor 409
910	Heat Flux Core Conductor 410
911	Heat Flux Core Conductor 411
912	Heat Flux Core Conductor 412
913	Heat Flux Core Conductor 413
914	Heat Flux Core Conductor 414
915	Heat Flux Core Conductor 415
916	Heat Flux Core Conductor 416
917	Heat Flux Core Conductor 417
918	Heat Flux Core Conductor 418
919	Heat Flux Core Conductor 419
920	Heat Flux Core Conductor 420
921	Heat Flux Core Conductor 421
922	Heat Flux Core Conductor 422
923	Heat Flux Core Conductor 423
924	Heat Flux Core Conductor 424
925	Total 1-D Reactivity (Rho)
926	Void 1-D Reactivity (Rho)
927	Doppler 1-D Reactivity (Rho)
928	Control Rod 1-D Reactivity (Rho)
929	Beta Delayed Neutron Fraction
930	Relief Valve Flow Junction 370
931	Relief Valve Flow Junction 371
932	Relief Valve Flow Junction 372
933	Relief Valve Flow Junction 373
934	Relief Valve Flow Junction 369
935	Reserved for Later Use
936	LPRM Level A, Lower half Power Fraction
937	LPRM Level A, Upper half Power Fraction
938	LPRM Level B, Lower half Power Fraction
939	LPRM Level B, Upper half Power Fraction
940	LPRM Level C, Lower half Power Fraction
941	LPRM Level C, Upper half Power Fraction
942	LPRM Level D, Lower half Power Fraction
943	LPRM Level D, Upper half Power Fraction
944	Normalized Power

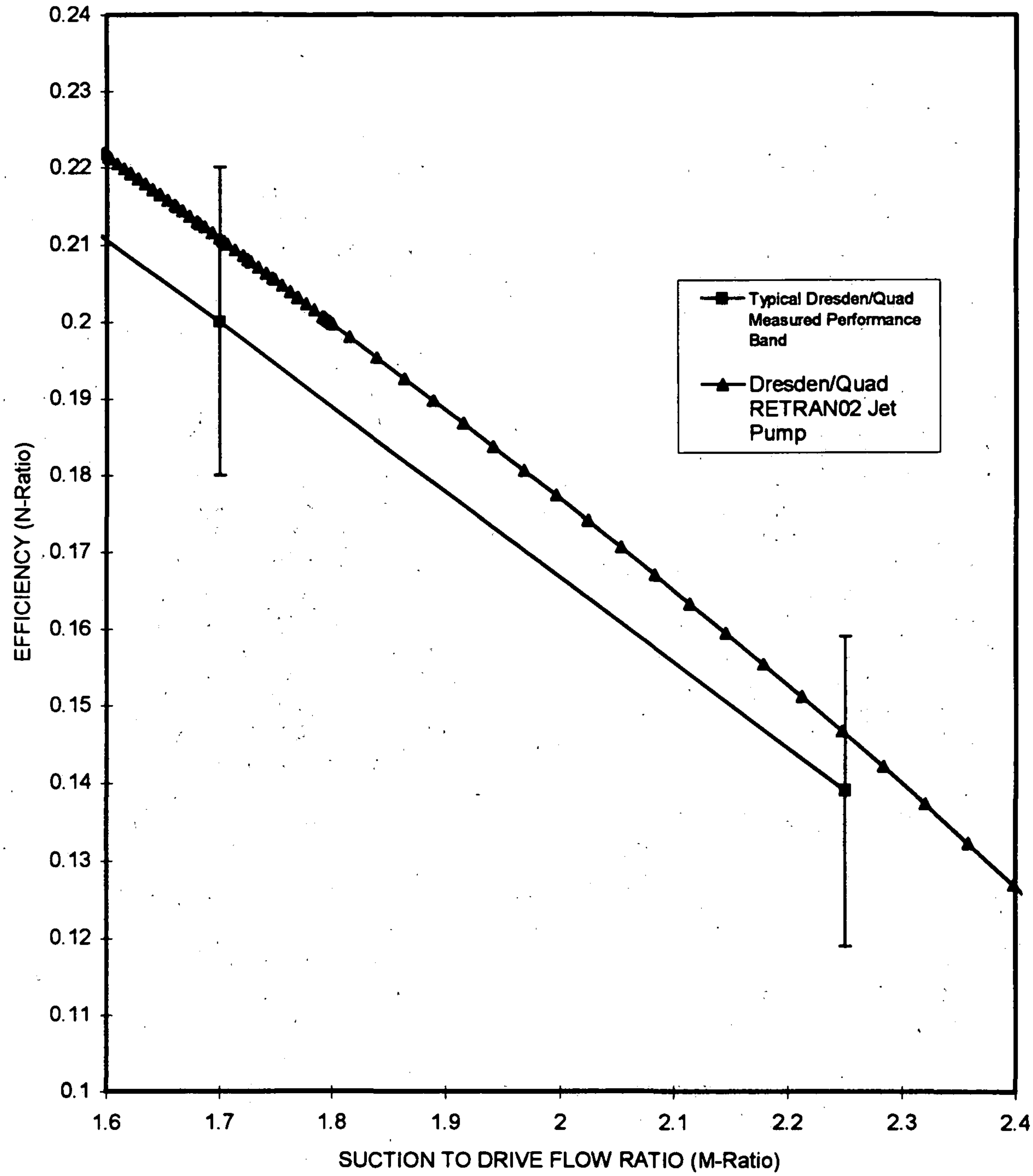

Figure 3.2-3, Quad-Cities and Dresden Jet Pump M-N Characteristics

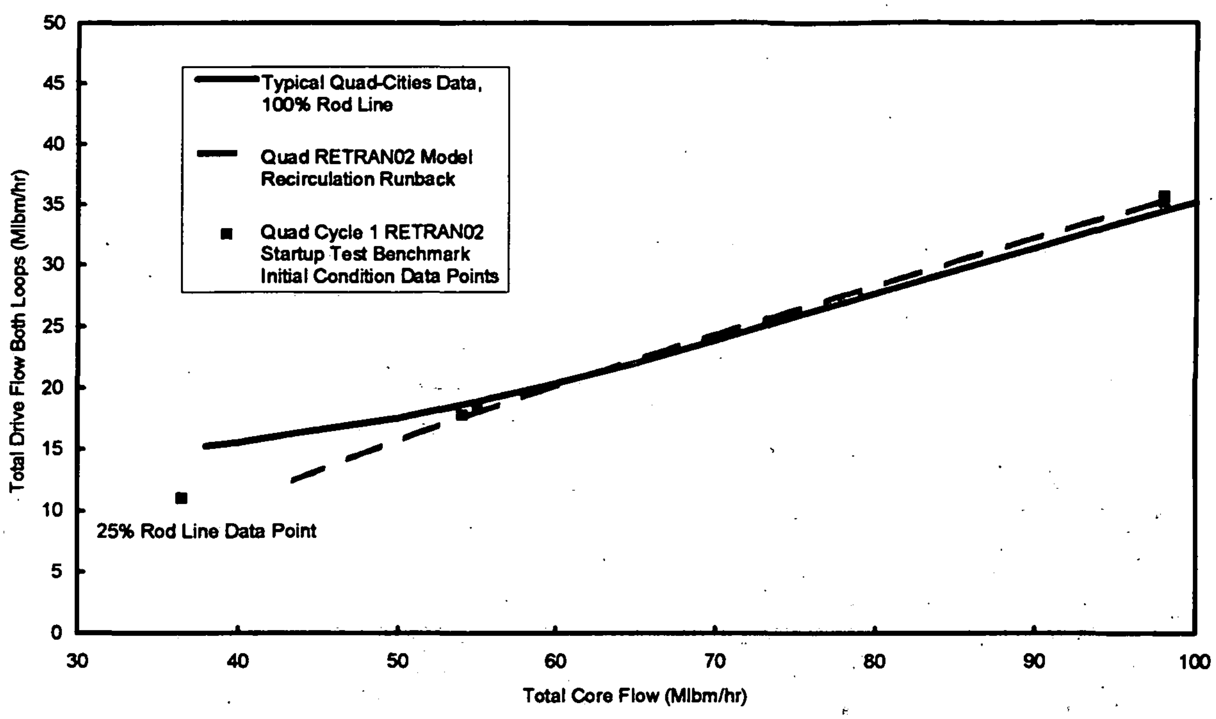

Figure 3.2-4, Quad-Cities Core Flow vs. Drive Flow Characteristics

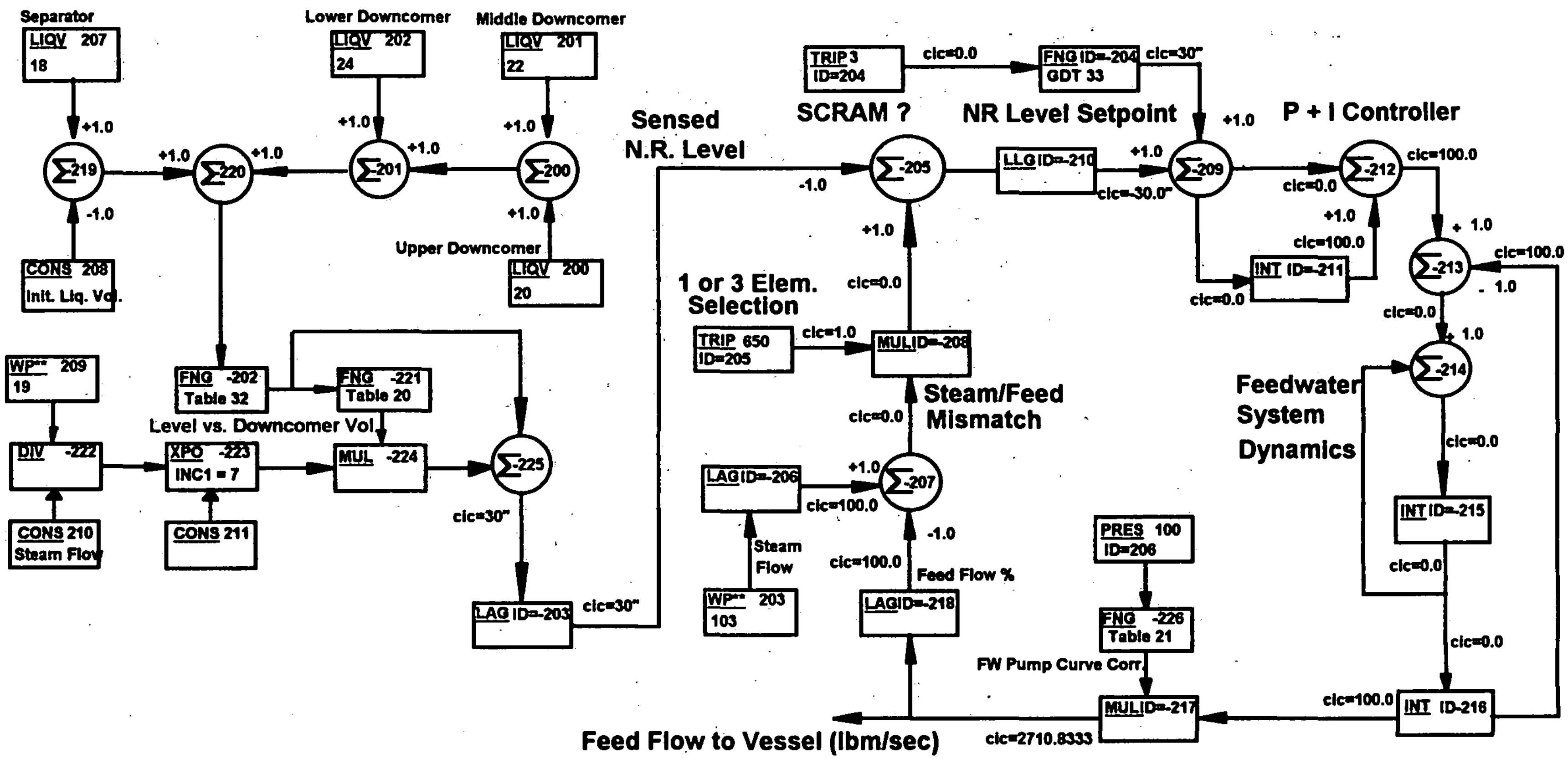

Figure 3.2-5, Quad-Cities and Dresden Vessel Water Level Controller

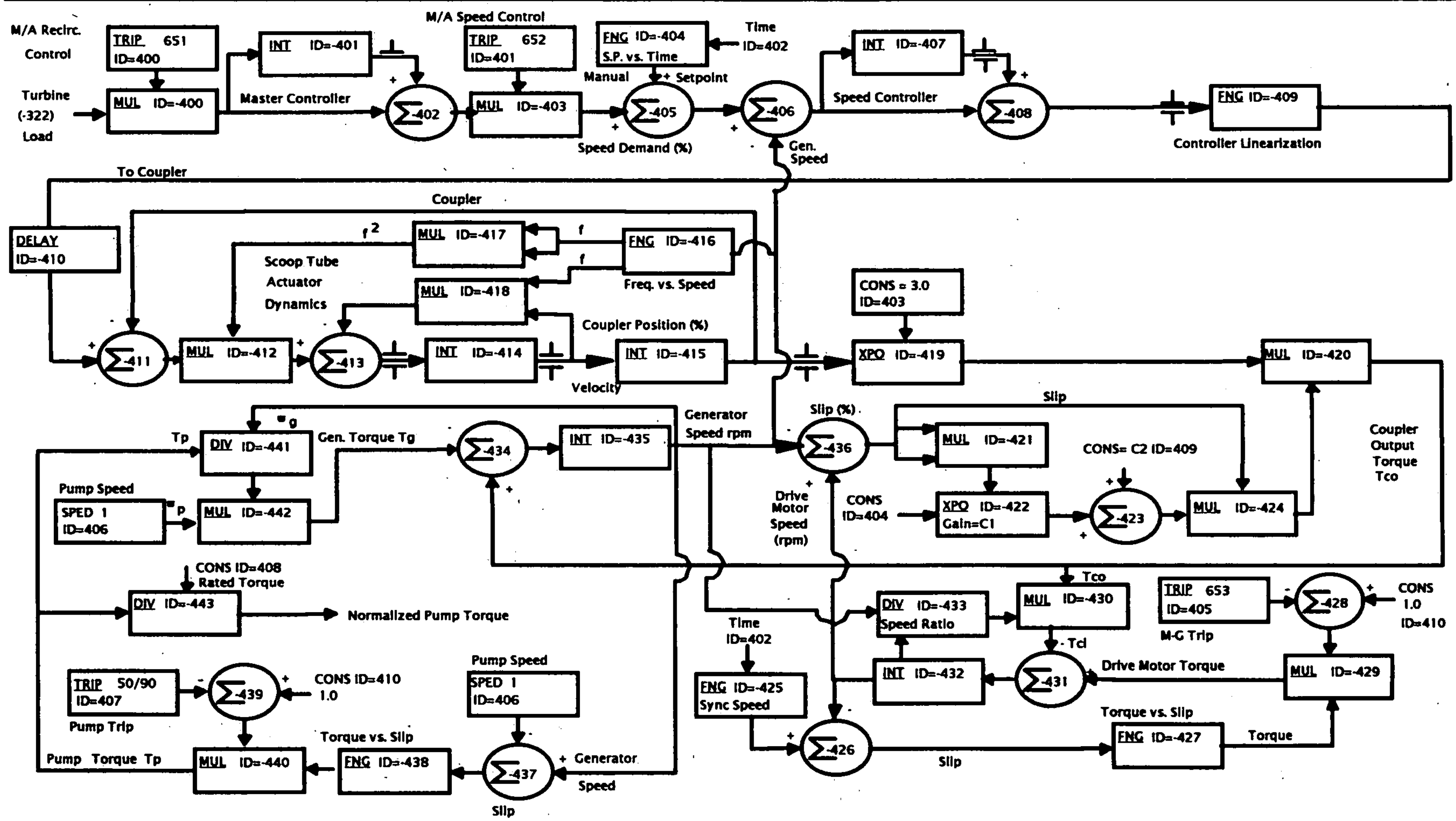

Figure 3.2-6, Quad-Cities and Dresden Recirculation System Controller

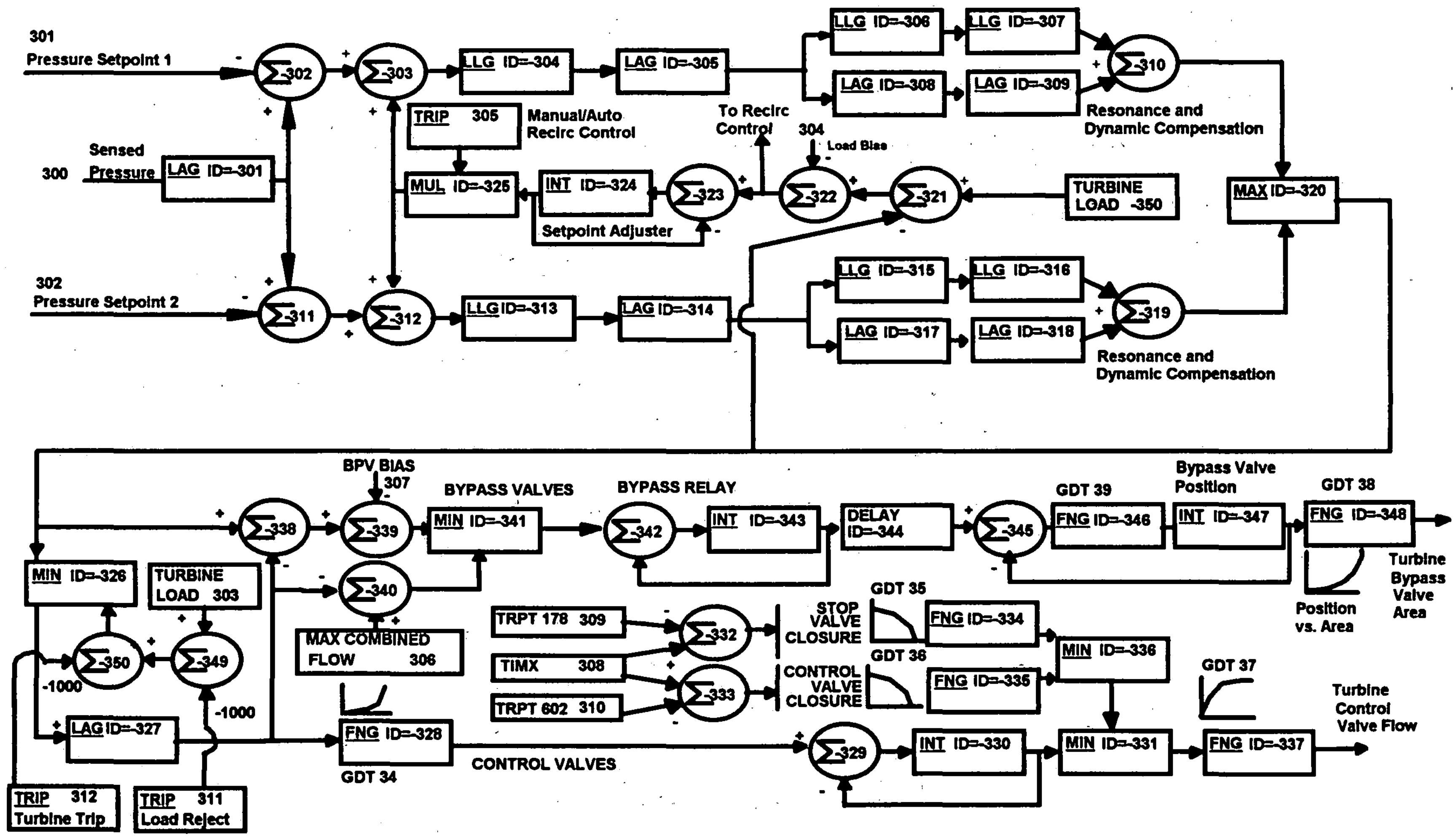

Figure 3.2-7, Quad-Cities and Dresden Pressure Regulator

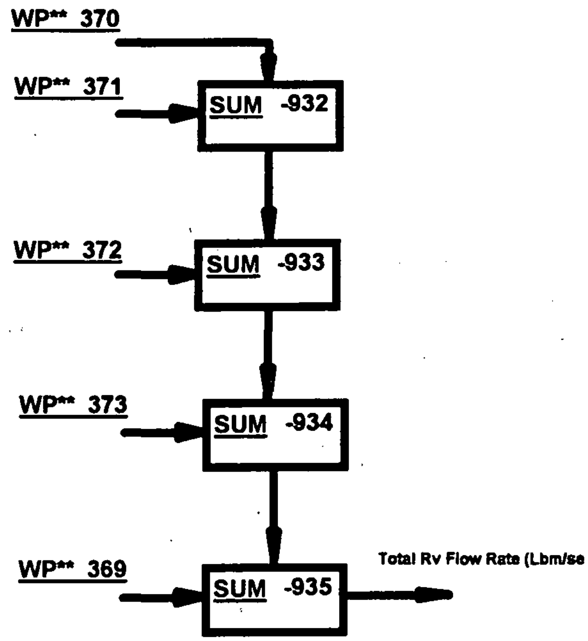

Figure 3.2-8, Quad-Cities and Dresden Control Total RV Flow Rate

3.3 Dresden RETRAN Model

3.3.1 System Model Nodalization and Geometry

3.3.1.1 Basis for Dresden Model

The Dresden RETRAN system model was developed with the Quad-Cities RETRAN model used as a basis. This section describes the design differences and the changes that were required to the Quad-Cities base RETRAN model to generate a Dresden model. These changes in the Quad-Cities model are to account for the design differences existing between the Dresden and Quad-Cities plant systems.

The Dresden RETRAN model is identical to the Quad-Cities model except for the required changes to the Quad-Cities base model to address the following differences:

1. Differences in rated steam flow and core thermal power,
2. Differences in trip setpoints,
3. Dresden has the Isolation Condenser (IC) system rather than reactor core isolation cooling (RCIC).

The first two differences; 1 and 2 are accounted for by implementing the appropriate changes to the Quad-Cities RETRAN base model. Specifically, since Dresden has a slightly higher rated main steam flow rate than Quad-Cities, the pressure control system and the feedwater control system will have slightly different gains to account for this difference. Accordingly, Junctions 19, 101, 501 and 602 (see Figure 3.2-1) will have different flow rates as a result of this difference. In the Dresden RETRAN model, the IC system is not modeled, and the RCIC fill table from the Quad-Cities model is changed to insure no RCIC flow will initiate.

A neutronic comparison of the Quad-Cities and Dresden initial cores was made due to the differences in the cycle 1 cores. The bundle designs for the two reactors used the same lattice average enrichment of 2.13 w/o U^{235}. However, the method for the control of excess reactivity were different. The Quad-Cities bundles utilized a distribution of Gadolinia bearing fuel rods. This is the same practice as is done in current bundle designs. The Dresden bundles contained no Gadolinia, but the core design utilized an array of boron impregnated stainless steel "curtains" placed in the bypass region of the core to control the excess reactivity.

Comparison of the results for the Quad-Cities and Dresden lattices at the exposure near the time of the Dresden startup test indicates a small difference in hot excess reactivity, but the average void coefficients were found to be reasonably close.

Based on ComEd's evaluation, the Quad-Cities neutronic model is acceptable for use as the Dresden neutronic model for benchmarking because the difference in core average reactivity and void coefficient is small enough to be negligible.

Dresden Unit 3 had its reactor recirculation piping replaced in 1985, which slightly changed its physical dimensions from the Cycle 1 configuration. This modification in the reactor recirculation piping will be addressed when analyses are performed for the recent and future cycles.

3.3.1.2 Main Steam Lines

A new model was constructed for main steam lines. The main steam lines are modeled consistently with the methods used for LaSalle and Quad-Cities. The flow areas are consistent between the Quad-Cities and Dresden steam lines, however, the flow lengths are slightly different. Since the flow lengths are different, the loss coefficients must also be different to allow for an equivalent steam line pressure drop for Quad-Cities and Dresden. (See Table 3.3-1 and Table 3.3-2.) The Dresden RETRAN model uses the Quad-Cities turbine bypass model.

3.3.1.3 Safety/Relief Valves

Dresden has the same number of safety, relief and Target Rock valves as Quad-Cities. The only difference is that the safety valves as well as the safety function of the Target Rock valve have different rated flow rates than the corresponding Quad-Cities valves. The contraction coefficients at the RETRAN SRV model junctions (see Table 3.3-2) are selected appropriately to accommodate the different flow rates. The difference in the rated capacity results from differences listed in the Technical Specifications.

Table 3.3-1, Dresden System Model Volume Geometric Data

Volume Node #	X	V	ZVOL	FLOWL	FLOWA	DIAMV	ELEV	Description
101	407.9931	45.0107	45.0107	58.0064	7.02	1.494833	7.4193	MAIN STEAM LINE
102	60.1294	1.4948	1.4948	8.5655	7.02	1.494833	7.4193	MAIN STEAM LINE
103	247.3059	22.8282	22.8282	35.2289	7.02	1.494833	-13.9141	MAIN STEAM LINE
104	143.3244	1.4948	1.4948	20.4167	7.02	1.494833	-13.9141	MAIN STEAM LINE
105	270.0044	20.3984	20.3984	27.6027	10.143	1.796833	-14.0651	MAIN STEAM LINE
106	220.2283	0.8984	0.8984	21.7124	10.143	1.796833	5.4349	MAIN STEAM LINE
107	188.3436	17.2968	17.2968	18.5689	10.143	1.796833	5.4349	MAIN STEAM LINE

Table 3.3-2, Dresden System Model Junction Geometric Data

Junction Node #	WP	AJUN	ZJUN	INERTA	FJUNF	FJUNR	DIAMJ	CONCO	Description
101	2712.5	7.02	51.6667	0	0.302582	0	1.4948	0	MAIN STEAM LINE
102	0	7.02	8.1667	0	0.53075	0	1.4948	0	MAIN STEAM LINE
103	0	7.02	8.1667	0	0.259215	0	1.7968	0	MAIN STEAM LINE
106	0	10.143	6.3333	0	0.242798	0	1.7968	0	MAIN STEAM LINE
107	2712.5	10.143	6.3333	0	1.383742	0	1.4948	0	INBOARD MAIN STEAM
174	0	7.02	-13.1667	0	3.286017	0	1.4948	0	INBOARD MSIV
175	0	7.02	-13.1667	0	2.608019	0	1.4948	0	OUTBOARD MSIV
261	0	0.1963	8.1666	0.0485	0	0	.5	0	SAFETY VALVE (203 - 4A)
262	0	0.1963	8.1666	0.4046	0	0	.5	0	SAFETY VALVE (203 - 4B)
263	0	0.1963	8.1666	1.1180	0	0	.5	0	SAFETY VALVE (203 - 4C)
264	0	0.1963	8.1666	1.4741	0	0	.5	0	SAFETY VALVE (203 - 4D)
265	0	0.1963	8.1666	0.9359	0	0	.5	0	SAFETY VALVE (203 - 4E)
266	0	0.1963	8.1666	1.2921	0	0	.5	0	SAFETY VALVE (203 - 4F)
267	0	0.1963	8.1666	0.0471	0	0	.5	0	SAFETY VALVE (203 - 4G)
268	0	0.1963	8.1666	0.4032	0	0	.5	0	SAFETY VALVE (203 - 4H)
369	0	0.1963	8.1666	0.7607	0	0	.5	0	TARGET ROCK VALVE (203 - 3A)

3.3.2 Trip Logic

The Dresden relief valves (Trip IDs 370 - 373) have different setpoints from the Quad-Cities model. The main steam low pressure trip setpoint (Trip ID=14) and high drywell pressure setpoint (Trip ID=20) are different from Quad-Cities. The high main steam line flow rate setpoint (Trip ID 13) for Dresden has a different setpoint with respect to rated flow, as well as a different rated flow. These were set accordingly in the Dresden RETRAN model. The differences in these setpoints arise from station specific Technical Specification limits.

3.4 Correlations, Options, and Model Limitations

One very important aspect in RETRAN system modeling is the choice of RETRAN options. Sections 3.4.1 through 3.4.5 document the RETRAN options used in BWR RETRAN analysis. Table 3.4-1 summarizes the relevant options selected for ComEd BWR RETRAN analysis.

In Section 3.4.6 a summary of SER restrictions are listed and discussed to assure compliance in all applicable areas. Particular attention has been given to the subcooled void model and non-equilibrium pressurizer model. ComEd has the successful Peach Bottom Turbine Trip benchmark comparisons with LPRM data comparisons in the lower part of core which justify correct application and reasonableness of the subcooled void model applied in the one dimensional neutron kinetics. ComEd's approach to preserving core reactivity with the application of the methods described in Appendix A assure that proper neutronics initial conditions and feedback will be achieved in the RETRAN system model. With regard to the non-equilibrium pressurizer model, particular attention to nodalization of each RETRAN system model was given so that the interface boundary does not cross on top or bottom of the Bubble rise volume for the upper downcomer or separator. This satisfies the SER requirements.

3.4.1 General Options for Problem Control and Description Data Cards

The two stream momentum mixing option (NTMM =1) is employed in the BWR system models to model BWR jet pumps. This option is set and used in conjunction with MVMIX variable on the RETRAN junction inputs. The RETRAN jet pump model has been qualified, however, References 8 and 9 limit the application of the jet pump model to forward flow only. Further justification will be provided if reverse jet pump flow is expected for an analysis.

The space time kinetics option with the multiple control state rod model is used (NODEL = 5). This model offers more flexibility and provides more reasonable and accurate results. The multi-control state control rod model was approved by the NRC in Reference 9.

The metal water reaction model is not used in calculations (MWREAC=0). This model has not been qualified for cylindrical geometry (Reference 8).

The algebraic slip option (ISFLAG = 2) is used for BWR analysis. In the plant analysis qualification work, summarized in Reference 8, analyses were performed using 1-D kinetics along with the algebraic slip option and the subcooled boiling model. Reference 8 stated that the combination of models mentioned above can lead to estimates of peak power to within several percent of data for pressurization events. However Reference 8 also places some limitations on the algebraic slip model because of the lack of a separate effects comparison to FRIGG tests. Philadelphia Electric Company (PECo) performed the comparison to the FRIGG tests using algebraic slip and the subcooled boiling model with RETRAN02/MOD004 in Reference 6. The good comparison to the FRIGG test, along with the comparisons to Peach Bottom turbine trips with RETRAN02/MOD005 serve as justification for the use of algebraic slip with the subcooled boiling model.

The steady state initialization option is set to the default value (JSST=0) and the RETRAN steady state option is used.

The non-equilibrium option (IPRZR=1) is used in BWR system analysis in the separator and upper downcomer regions. The use of this model is of particular importance during pressurization events. With the homogeneous equilibrium model (HEM), the vapor temperature is always equal to the liquid temperature. In pressurization events, the HEM model would allow a more rapid transfer of thermal energy (condensation) from the vapor phase to the liquid phase. This would result in lower volume temperatures and an under-prediction of the pressurization rate. A limitation from Reference 8 discusses the lack of comparison data for a completely full or empty volume and there is no comparison data for fluid to boundary heat transfer. The nodalization is selected to avoid emptying or filling a non-equilibrium volume during normal operational transients. In addition a RETRAN trip is used to provide an indication of either filling of emptying a non-equilibrium volume. If an analysis is performed which violates the limitation and will have an impact on thermal margins, modeling studies will be performed to determine the conservative modeling approach.

The temperature transport delay option (ITRNS =1) is employed in the system model to simulate the movement of temperature fronts. This model may be used in the recirculation loop volumes. This model is not appropriate for plenum regions where considerable mixing occurs.

The combination forced and free convection map with condensation (IHTMAP=1) is used in the BWR RETRAN system models. Heat transfer is modeled in the system model via fuel rod to coolant, conduction through the channel wall and condensation in the steam bypass piping.

The iterative numerical solution technique (INEXPL = 1) is used in BWR RETRAN system models. The iterative solution method allows results of the time steps to be evaluated before the solution is accepted. This solution technique is likely to be more accurate and have fewer instabilities than the standard RETRAN solution method.

The neutron void reactivity model (IVOID = 1) is used in BWR system models. Qualification of the subcooled void model along with the algebraic slip option was presented in Reference 6. The use of this model with algebraic slip has been widely accepted for BWR analysis and comparisons to Peach Bottom turbine trip power response further serve as qualification for this model.

The profile fit option is not used (NFIT=0). The profile fit option allows the user to supply coefficients which affect the algebraic slip model and therefore changing the void profile. Changing the void profile also alters the subcooled void fit. This option may be used to perform sensitivity studies on the algebraic slip and subcooled void models.

The default method for calculating the volume flow for momentum flux (JFLAG = 0) is used in BWR system models.

The component steam separator model (NSEPR =1) is used to model the BWR steam separator. This option is set along with the input for the RETRAN component separator inputs to define the separator performance.

The generalized transport option is not used (IGNTR=0) since the transport of impurities within the coolant is not desired for the current RETRAN system model applications.

3.4.2 Volume Input Card Options

The relevant options on the RETRAN volume inputs correspond to the use of the non-equilibrium pressurizer option. The INEQ variable is set to -1 for the separator, volume 18, and the upper downcomer, volume 20, per Reference 10. The VRAIN and VLHTC values are both set to 0.0 to minimize the energy transfer between phases.

3.4.3 Junction Input Card Options

The junction choking index (JCHOKE) is set to -1 (no choking) at all junctions except at the main steam safety valves, relief valves, steam line flow restrictor and the main steam bypass system where choking is expected to occur. At junctions where choking occurs, the isoenthalpic expansion model is employed.

The two stream momentum mixing option (MVMIX) is normally set to zero. This variable is set to 2 for junctions 58, 59, 98 and 99 to model the jet pump mixing.

The junction flow regime flag (IFRJ) is set to use the default flow regime map (IFRJ=1) for all junctions except junctions 21 and 22. These two junctions represent the separator recirculation path and the junction between upper and lower downcomer. Since the RETRAN algebraic slip model is based upon modeling cocurrent upflow, this model is not appropriate for these junctions. Therefore, IFRJ is set to -99 for these junctions to turn off slip.

The two-phase wall friction index (JTPMJ) is set to 3 to use the Baroczy two-phase multiplier based on flow quality.

The enthalpy transport model (IHQCOR) is turned on for all junctions which connect "heated" volumes and it is turned off for all other junctions. The enthalpy transport model will more accurately predict the energy and mass distributions through heated sections like the core volumes.

3.4.4 Heat Conductor Input Card Options

The RETRAN heat conductor inputs contain an option for the critical heat flux heat transfer correlation (IMCL and IMCR). The default correlation, Groeneveld 5.9 (IMCR=IMCL=0), is selected for use and is valid over the range of expected conditions. This is a conservative correlation and critical heat flux is not expected to occur for normal BWR RETRAN analyses.

3.4.5 Conductor Geometry Input Card Options

The RETRAN conductor geometry inputs contain an option for the gap expansion model (IGP). This model is not used (IGP=0) and a constant, temperature independent, axially uniform gap conductance is used.

Table 3.4-1, RETRAN Code Options Used in All ComEd BWR Models

RETRAN Problem Description Inputs		
Variable	Value	Description
NTMM	1	**Two-stream momentum mixing**
NODEL	5	**Space time kinetics, multiple control state option**
MWREAC	0	**Metal water reaction**
ISFLAG	2	**Algebraic slip option**
JSST	0	**Steady state initialization option**
IPRZR	1	**Non-equilibrium pressurizer option**
ITRNS	1	**Temperature transport delay option**
IDNBC	0	**Auxiliary DNB Calculation**
ICF	1	**Control system options**
IHTMAP	1	**Combination forced and free convection map with condensation**
INEXPL	1	**Iterative numerical solution technique**
NSTK	0	**Local conditions heat transfer**
IVOID	1	**Neutron void reactivity option flag**
NFIT	0	**Profile fit coefficient index**
JFLAG	0	**Volume flow / momentum flux flag**
NSEPR	1	**Steam separator component option**
IGNTR	0	**General transport option**

RETRAN Volume Inputs		
Variable	Value	Description
INEQ	0 or -1	**Non-equilibrium model, 0 used for all volumes except 18 and 20 where -1 is input**
VRAIN	0.0	**Rain out velocity**
VLHTC	0.0	**Inter-region heat transfer coefficient**

RETRAN Junction Inputs		
Variable	Value	Description
JVERTL	0 or 1	**Vertical junction index, 0 input for all horizontal input junctions and 1 input for all vertically oriented junctions and junctions connected to bubble rise volumes**
JCHOKE	-1 or 1	**Junction choking index, -1 input for junctions where choking does not occur and option 1 (isoenthalpic expansion model) used for junctions where choking is expected.**
JCALCI	0 or 2	**Initial condition calculation index, value set to 0 for junctions except when RETRAN calculates the inertia based on the geometry**
MVMIX	0 or 2	**Momentum mixing option, 0 used for all junctions except at junctions 58, 59, 98 and 99 to model the jet pump.**
IFRJ	1 or -99	**Flow regime map (-99 is input for junctions 21 and 22 to avoid slip calculation)**
JTPMJ	3	**Two phase wall friction index (This option was added to obtain the Baroczy based on flowing quality)**
IHQCOR	0, 1, 2, or 3	**Enthalpy transport option (option used for the core and bypass volumes only)**
ISP	0	**Pressurizer spray model (not used)**

RETRAN Heat Conductor Inputs		
Variable	Value	Description
IMCL	0	**Critical heat flux heat transfer correlation (Groeneveld 5.9)**
IMCR	0	**Critical heat flux heat transfer correlation (Groeneveld 5.9)**

RETRAN Conductor Geometry Inputs		
Variable	Value	Description
IGP	0	**Gap expansion model (not used)**

3.4.6 RETRAN Model Limitations

Below is a set of limitations taken from various RETRAN SER's. They are addressed below in a chronological order, starting with the RETRAN02/MOD002 Technical Evaluation Report (TER) from Reference 8. In most cases, the limitations from earlier versions of RETRAN still apply, therefore the limitations from MOD002 through MOD005 are addressed below.

RETRAN02/MOD002

The RETRAN limitations listed below are taken directly from Reference 8. Below each listed limitation is a short discussion of how each limitation will be addressed in BWR RETRAN analysis.

"On the basis of our review of the analytical models, solution techniques and qualification work, the following evaluation can be made. The more significant approximations and assumptions of the models used are given below. Restrictions based on the examination of the qualification work, both separate effects and systems comparisons are given concurrently. Where not otherwise stated, it should be presumed that the approximations and assumptions have been judged acceptable as specifically applied for the BWR transients in question (given an appropriate selection of input) through the review of the qualification work. As limited material was submitted for the PWR qualification work, the use of the models for PWR systems require further justification. These additional requirements are detailed at the end of the summary of the approximations."

"The limitations of the models are:"

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

The one dimensional space-time kinetics option is used in BWR RETRAN analysis. A best-estimate use of this model is demonstrated in the Peach Bottom turbine trip comparisons. The Peach Bottom turbine trip tests resulted in large increases in nuclear power and therefore comparing to the measured data provides for an excellent benchmark of the RETRAN one dimensional kinetics model with ComEd methods. Conservative usage will be demonstrated in the Reload Licensing Application Topical report.

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

All RETRAN system model analyses are performed with non-zero fission power.

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

A generalized transport model is now available in RETRAN02/MOD005, however, this model is not currently used in BWR analyses.

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

Control rods are assumed to travel in banks in RETRAN analyses.

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

The metal water reaction calculation is not used in BWR RETRAN analyses.

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

Non equilibrium models are used for the separator and upper downcomer regions. Use of the non-equilibrium models in these regions provides more realistic and more conservative results than the HEM model. The HEM model sets the vapor phase temperature equal to the liquid phase temperature. Thus the HEM model instantly transfers thermal energy to the liquid phase (condensation) which results in lower volume temperatures and under-predictions in pressurization rates. The subcooled boiling model is used to more accurately simulate the void distribution for kinetics feedback.

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

Vector momentum models are not used in BWR RETRAN system models.

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

The homogeneous model is not used when slip occurs in BWR RETRAN system models. Homogenous slip is not really an option since there is no "slip" when the HEM model is used. Instead, the algebraic slip option is used. This option has been qualified by comparisons to FRIGG data (Reference 6) and has been accepted (References 5 and 6) for RETRAN BWR applications.

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

The algebraic slip and subcooled void models were compared to FRIGG test data in Reference 6 to provide further separate effects comparisons for the algebraic slip model. This comparison meets the conditions of n2.

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

The dynamic slip model is not used in BWR RETRAN system models.

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

The gap expansion model is not used in BWR RETRAN system models.

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

This option is not used in BWR RETRAN system models.

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

BWR RETRAN analyses are performed in the subcritical region, well below the critical pressure. The critical pressure is just above 3000 psia and all BWR RETRAN analyses are performed at pressures less than 1500 psia.

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

Pre-CHF heat transfer is used through out the RETRAN calculation.

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

The Bennett flow map is used within the conditions of the data base and the accepted Baroczy two phase friction multiplier based on flowing quality is used in BWR RETRAN analysis.

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

The algebraic slip and subcooled void models were compared to FRIGG test data in Reference 6 to provide further separate effects comparisons for the algebraic slip model. This comparison meets the conditions of h2.

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

The dynamic slip model is not used in BWR RETRAN system models.

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

The nodalization is selected to avoid emptying or filling a non-equilibrium volume during normal operational transients. In addition a RETRAN trip is used to provide an indication of filling or emptying a non-equilibrium volume. If an analysis is performed which violates the limitation and will have an impact on thermal margins, modeling studies will be performed to determine the conservative modeling approach. The lack of fluid boundary heat transfer is conservative since the lack of transfer of heat from the fluid to the surrounding walls would result in increases in pressurization.

"p. The non-mechanistic separator model assumes quasi-static (time constant-few tenths seconds) and uses GE BWR6 carryover/carryunder curves for default values. Use of the default curves has to be justified for specific applications. As with the pressurizer a constant L/A is used. The treatment in the off normal flow quadrants is limited and those quadrants should be avoided. Attenuation of pressure waves at low flow/ low quality conditions are not simulated well. Specific application to BWR pressurization transients under those conditions should be justified."

Constant carryover/carryunder values are incorporated in to the BWR system models. The separator L/A as a function of inlet quality is used as a case specific input based on vendor data. The low flow/ low quality conditions are not required to be modeled during relevant portions of reload licensing cases, however, the Peach Bottom Turbine Trip Test 1 experienced the low quality conditions and the RETRAN model was shown to provide a <u>conservative</u> pressurization and power response for this condition.

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

The BWR RETRAN models use vendor supplied pump characteristics that were converted to RETRAN homologous pump curves. Single phase conditions always exist in the recirculation lines and pumps during licensing and benchmarking calculations.

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

For most all licensing and benchmarking calculations, the jet pump remains in the forward flow quadrant for the relevant portion of the analyses. If reverse jet pump flow is required, further jet pump qualification will be provided.

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

The nonmechanistic turbine model is not used in BWR RETRAN system models.

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

The subcooled void model is used in BWR RETRAN analyses for void reactivity computations and it has been shown to be applicable to BWR situations. Additional separate effects qualification of the subcooled void model has been submitted in Reference 6. The comparisons to the Peach Bottom turbine trip tests also serve as further qualification of the subcooled void model. This model has been widely accepted for BWR applications.

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

The bubble rise model is primarily used in the separator and the upper downcomer. Sensitivity studies were performed for the Peach Bottom turbine trip test 3 to determine a conservative bubble rise velocity and void profile for pressurization events. Slip is also set to zero in the separator recirculation junction and the upper downcomer to lower downcomer junction.

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

The transport delay option may only be used in the recirculation loops. These regions will experience one dominant flow direction during licensing and benchmarking calculations.

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

The auxiliary DNBR model is not used in BWR RETRAN system models.

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

The enthalpy transport model is used in the heated core and bypass regions to more accurately predict the energy and mass distributions through heated sections. The model is only applied at volumes where there is only one outlet junction. This model has been widely used and accepted for BWR applications and the manual for the MOD03 modifications has been submitted.

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

The local conditions heat transfer model is not used in BWR RETRAN analyses.

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

The steady state initialization option is used and a null transient is run for to assure that a steady state is maintained. The RETRAN calculated values for core heat transfer areas, bubble rise velocities, loss coefficients, control system initial conditions and the feedwater fill enthalpy bias are examined to ensure good engineering and design practices.

RETRAN02/MOD003 and MOD004

The RETRAN limitations listed below are taken directly from Reference 9, the Safety Evaluation Report for RETRAN02/MOD003 and MOD004. Below is a short discussion of how each limitation will be addressed in BWR RETRAN analysis.

"We have reviewed (1) the lists of corrections and changes to the RETRAN02 code, (2) the identified modeling changes implemented in the RETRAN02 MOD003 and MOD004 versions, and (3) the responses to our review questions. Based on the results of our review, we conclude that RETRAN02/MOD003 and MOD004 are acceptable for use in transient analyses with the restrictions as follows."

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

One RETRAN nodalization is used throughout the Peach Bottom turbine trip benchmarking and the plant startup benchmarking. The models and options are chosen to address RETRAN limitations and to conform to previously accepted BWR RETRAN modeling techniques. The benchmarking work included pressurization events, depressurization events, changes in reactor water inventory and changes in recirculation flow. The use of one nodalization throughout the benchmarking serves to qualify the adequacy of the nodalization. Altering the nodalization for special applications is acceptable provided proper qualification, benchmarking, or sensitivities are performed.

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

The limitations of the original SER have been discussed in the RETRAN02/MOD002 above.

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

This limitation applies to the dynamic slip model in non-bubbly flow regimes. The BWR RETRAN system models use the algebraic slip option. The algebraic slip option is not flow regime dependent (Reference 10).

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

The limitations of the original SER have been discussed in the RETRAN02/MOD002 above.

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

The BWR RETRAN system models do not use heat conductors connected to the non-equilibrium volumes (separator and upper downcomer). It is assumed that there is no heat transfer to the metal walls of the separator or upper downcomer. Modeling heat transfer and energy to the surrounding metal would tend to attenuate the pressure wave in limiting pressurization transients.

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

The default Courant time step control is used in BWR RETRAN system analysis. Time step sensitivities are performed to demonstrate the adequacy of the time step selection.

RETRAN02/MOD005

The RETRAN limitations listed below are taken directly from Reference 3, the Safety Evaluation Report for RETRAN02/MOD005. Below each listed limitation is a short discussion of how each limitation will be addressed in BWR RETRAN analysis.

"We have reviewed (1) the lists of corrections and changes to the RETRAN02 code provided by RETRAN02 Maintenance Group, (2) the identified modeling changes implemented in the RETRAN02 MOD005.0 version, together with (3) the responses to questions provided by Computer Simulation Analysts. Based upon the foregoing, and <u>subject to the limitations and restrictions contained in the original SER, and the SERs related to RETRANO2 MOD003 and MOD004 and as set forth above with respect to RETRAN02 MOD005.0</u>, we conclude that there are reasonable assurances that the RETRAN02 MOD005.0 computer code version is acceptable for use.'

"In addition, with respect to MOD005.0:"

"1. With respect to each transient for which the general transport model is used, the user should be required to justify the selected degree of mixing;"

The generalized transport model is not currently used in BWR RETRAN system model analysis for current applications.

"2. When using the 1979 standard decay heat model, the user should be required to justify the associated parameter selection as presented in Section 2.2;"

The 1979 decay heat model is not currently used for BWR analysis applications. However, if special RETRAN analyses are performed with the 1979 decay heat model, additional justification will be provided.

"3. Each user should be cautioned that the reactivity components provided by the new edit feature are somewhat inexact and may be used only as a qualitative indicator rather than a quantitative indicator of transient reactivity feedback."

This limitation is noted and the component reactivity information is only used qualitatively and is not used for input to any other analyses.

RETRAN02/MOD005.1

The RETRAN limitations listed below taken directly from Reference 4.

There are no new limitations associated with using RETRAN02/MOD005.1. MOD005.1 consists of error corrections to 5.0 and modifications to allow MOD 5.1 to be used on UNIX workstations. No new models were introduced in MOD 5.1.

3.4.7 Time Step Selection

ComEd analyzed the time step selection impact on the RETRAN calculated key results. The maximum time step was varied from 0.001 to 0.0005 and the results show almost no impact, demonstrating adequate selection of time steps. Table 3.4-2 through Table 3.4-4 show the results of some of the time step studies.

Table 3.4-2, Peach Bottom Turbine Trip 3 Time Step Study

	Normalized Flux		Normalized Heat Flux		Peak Dome Pressure
Time Step Sensitivity	Peak	Time (sec)	Peak	Time (sec)	(psia)
DTMAX = .001	5.27	0.71	1.121	1.10	1072.2
DTMAX = .0005	5.34	0.71	1.122	1.11	1072.3

Table 3.4-3, Dresden Two Recirculation Pump Trip Time Step Study

	Core Flow (%)		Core Power (%)	
Time Step Sensitivity	Minimum	Time (sec)	Minimum	Time (sec)
DTMAX = .001	31.37	34.6	24.99	10.1
DTMAX = .0005	31.40	34.3	25.05	10.1

Table 3.4-4, LaSalle Cycle 1 MSIV Closure Time Step Study

	Reactor Pressure (psia)		Core Flow (lbm/sec)	
Time Step Sensitivity	Peak	Time (sec)	Peak	Time (sec)
DTMAX = .001	1102.20	7.85	30342.9	4.46
DTMAX = .0005	1102.21	7.86	30342.7	4.46

3.5 Model Initialization

3.5.1 RETRAN Initialization Method

Once the key system parameters have been determined, the RETRAN model may be initialized. Table 3.5-1 summarizes the key RETRAN inputs for initialization. The starting point of the RETRAN initialization is a FIBWR2 run. The FIBWR2 analysis requires core power, core flow, core exit pressure, and core inlet enthalpy as inputs. The core power, core flow and reactor dome pressure are typically known for a given analysis condition. The inlet enthalpy is determined from a heat balance calculation. The core exit pressure is determined from the dome pressure and separator pressure drop information from vendor equations that describe the separator pressure drop. This information along with the power distribution from MICROBURN are input to FIBWR2. The FIBWR2 analysis generates values for the core bypass flow and the axial pressures for a single equivalent channel as discussed in Appendix B. This output is used to initialize the RETRAN model.

The FIBWR2 bypass flow information is used to determine the flow rates for junctions 400, 14 and 15. Junction 14 and 15 represent the total bypass flow with junction 15 representing the bypass fraction that is channel dependent. The FIBWR2 axial pressure values are input to the RETRAN core volume pressures for volumes 400 through 425. The core inlet pressure and enthalpy from FIBWR2 are also used to determine the lower plenum pressure and enthalpy input for the RETRAN model (volume 32). With these pressures used as input, the loss coefficients for junctions 14, 15 and 400 through 425 are left unspecified and RETRAN calculates the junction loss coefficients that balance the input pressure drops with the flow rates. The other system pressure that is input is the dome pressure because it is usually a known value.

The mixture level in the upper downcomer (volume 20) is set to achieve the correct narrow range sensed level. The mixture level in the separator (volume 18) is typically set near the middle of the region to avoid filling or emptying the bubble rise volume. The mixture level in the upper downcomer establishes the elevation head in the downcomer and the pressures in volumes 20 and 22. Thus the vessel pressure distribution has been established.

Once the vessel pressure distribution is determined, the jet pump suction and drive flows must be determined to properly balance the downcomer to lower plenum pressure drop. Since the vessel pressure distribution is specified, the junction loss coefficient in the lower downcomer (junction 24) is unspecified for the RETRAN steady state initialization. The pressure drop from the downcomer to lower plenum is a strong function of the jet pump M-ratio (suction flow / drive flow). Thus the M-ratio is set to balance the required pressure drop and to obtain the correct loss coefficient for junction 24. Experience has shown that only a small adjustment in M-ratio is required to achieve pressure balance. This "fine-tuning" adjustment leaves the drive flow well within the uncertainty range of the core flow versus drive flow relationship.

Once the correct M-ratio is determined, the LaSalle recirculation flow control valve position or the Dresden/Quad-Cities pump speed and torque are adjusted. This adjustment is made to balance the recirculation system pressure drop with the recirculation drive flow and to obtain the correct loss coefficient for the recirculation junctions 51 and 91. The steam flow and feed water flow are set consistent with the heat balance for a specific power and flow condition. At this point the RETRAN model is balanced with the correct thermal-hydraulic conditions.

Once all of the model has been balanced with the correct thermal-hydraulic initial conditions, the control systems are initialized. For example, once the recirculation valve control position is known, the recirculation control system initial conditions are set to obtain the correct valve position. This process will assure control system convergence at steady state initialization. Table 3.5-2 summarizes some of the key steps in the control system initialization. The RETRAN initialization is then checked for a proper steady state initialization. The RETRAN calculated values for core heat transfer areas, bubble rise velocities, loss coefficients, control system initial conditions and the feedwater fill enthalpy bias are examined to ensure good engineering and design practices.

After a proper initialization is obtained, a null transient is run. This assures that RETRAN does reach a thermal-hydraulic steady state and it shows that the control system initial conditions (CIC's) are set properly such that the control systems do not drive the system away from the initialized values.

Additional inputs are also required on a case specific basis to accurately simulate a transient. Some of these inputs include direct moderator heating, direct bypass heating, yield fraction and decay constants, gap conductance and separator inertia.

Table 3.5-1, Parameters Directly Input for Model Initialization

Parameter	RETRAN Model	Source of input
Core Power	System Input	Measured or Desired Analysis Value
Dome Pressure	Volume 100	Measured or Desired Analysis Value
Core Pressure Distribution	Volumes 32, 400-425	Calculated from FIBWR2 code
Core Inlet Enthalpy	Volume 32	Heat Balance Calculation, Consistent with FIBWR2
Core & Bypass Flow distribution	Junctions 32, 400 and 401	Calculated from FIBWR2 Code
Sensed Level	Volume 20	Measured or Desired Analysis Value
Recirculation Drive and Suction Flow	Junctions 59, 99, 58, and 98	Calculated based on Jet Pump Performance
Steam Flow/Feed Flow	Junctions 101, 501	Measured, Heat Balance, or Desired Analysis here.

Table 3.5-2 Control System Initialization Steps

Control System	Description
Sensed Variables	The LAG control block Control System Initial Conditions (CIC's) are set to the initial sensed variable values
Recirculation Control: (LaSalle)	With the recirculation control valve loss coefficient known, the valve position, drive flow demand and drive flow setpoint are back calculated and the CIC's are set to balance the system for steady state.
(Dresden/Quad Cities)	With the recirculation pump speed and torque known from the recirculation loop pump balance, the generator speed, drive motor torque, coupler position and scoop tube position demand may be calculated and the CIC's are set to balance the system for steady state.
Pressure Control	The pressure control system initialization is similar for all ComEd BWR plants. With the initial steam flow and turbine inlet pressure known, the initial turbine control valve position, pressure regulator demand and the pressure setpoint may be calculated and the CIC's are set to balance the system for steady state.
Feedwater Control	The feedwater control system initialization is similar for all ComEd BWR plants. With the initial feed water known, the initial feed water demand may be calculated. With the initial steam flow known, the dryer pressure drop may be determined. Once the dryer pressure drop is calculated, the downcomer mixture level may be calculated to obtain the desired sensed level at steady state initialization.
Heat Flux and APRM Control Systems	Constants in the heat flux control system may be adjusted to obtain the correct calculated heat flux % at time =0.0. Gains in the APRM control system may be adjusted on a case to case basis to obtain a proper normalization for the individual and averaged LPRM levels. This change is required to renormalize for different initial power shapes.

3.6 Model Sensitivity Study

A sensitivity study was performed in order to determine the relative importance of key RETRAN input parameters. The sensitivity studies also serve as justify the adequacy of the steam line nodalization, gap conductance modeling, and the choice of RETRAN bubble rise velocities. All sensitivities were performed for the Peach Bottom Turbine Trip Test 3 (TT3) event, since the TT3 is the closest to licensing type conditions of the three Peach Bottom turbine trip tests. A detailed discussion of the Peach Bottom turbine trip tests is in Section 5. The Peach Bottom benchmarking analyses showed very good agreement between RETRAN and the measured data. A summary of key results from the model sensitivity is shown in Table 3.6-1.

Core Model Sensitivities

Sensitivities were performed on several core input parameters. One of the more sensitive parameters was the scram delay time. The scram delay time was varied by ± 20 msec. The results show change in normalized neutron flux increased about 12% from the base case with an increased scram delay time, but the change in normalized heat flux was about 1.5%. The change in direct moderator heating had a smaller impact. With less direct moderator heating, the trend shows larger neutron flux results as expected. The amount of bypass heating was varied from 1% to 2% and this showed almost no sensitivity. Decreasing the core pressure drop by 2 psi had a small impact on normalized neutron flux and a negligible impact on normalized heat flux.

The gap conductance was varied over a wide range from 200 to 1200 BTU/ft^2-°F-hr. These values were arbitrarily chosen and are considered bounding. The gap conductance had a moderate impact on nuclear flux, but it had a big impact on peak heat flux and the timing of the peak heat flux, as expected. The peak vessel pressure was also greatly impacted by the gap conductance. There was nearly a 30 psi difference in peak dome pressure between the 200 to 1200 BTU/ft^2-°F-hr cases. This was due to the fact that the lower gap conductance resulted in higher initial fuel temperatures and stored energy. This energy was eventually released to the coolant and produced a higher peak pressure. An axial varying gap conductance was input. The axial varying gap conductance resulted in nearly identical nuclear and heat flux responses. Since, the overall peak heat flux was conserved, using an axially uniform gap conductance is acceptable

Jet Pump Model Sensitivities

Some of the jet pump parameters were varied. The suction and diffuser loss coefficients were reduced by 20% and the recirculation system was reconverged to obtain the required jet pump pressure drop. This change had almost no impact on the key results. The suction and diffuser inertia were varied by ±30 %. This had about a 4% impact on normalized neutron flux and almost no impact on normalized heat flux.

Separator Model Sensitivities

Several of the separator parameters were varied. The separator inlet inertia was varied by ±30 %. This study showed that this parameter is one of the more sensitive model inputs. This change produced about a 16% change in normalized neutron flux but less than a 1% change in normalized heat flux. The separator flow area and volume were also decreased by 10% and the results show very little impact. The separator and downcomer bubble rise velocities were widely varied from values of 1.0 to 1.0E6 ft/sec and there was essentially no change in the results.

Main Steam System Sensitivities

Some of the steam line system parameters were varied. The steam line loss coefficients were decreased by 20%, producing a lower steam line pressure drop. The lower steam line pressure drop resulted in an 9% increase in normalized neutron flux and less than 1% change in normalized heat flux. The steam dome volume was decreased by 10% and showed about the same impact as the steam pressure drop decrease. The steam line length was increased by 5% and there was a very small impact on normalized neutron flux. The number of steam line nodes was increased from 7 to 14. The 14 node steam line resulted in a small decrease of about 3% in normalized neutron flux and less than 0.5% change in normalized heat flux. This sensitivity shows the adequacy of the 7 node steam line.

Time Step Sensitivities

The maximum time step was varied from .0005 to .001 and the results show almost no impact (about 1% in peak normalized neutron flux), demonstrating adequate selection of time steps.

In summary, many aspects of the RETRAN model were investigated. The sensitivities demonstrate the adequacy of the gap conductance modeling, the steam line nodalization and the choice of RETRAN bubble rise parameters for pressurization events.

Table 3.6-1, Sensitivity Study Results for Peach Bottom Turbine Trip Test 3

	Normalized Neutron Flux		Normalized Heat Flux		Peak Dome Pressure
Core Sensitivities	Peak	Time (sec)	Peak	Time (sec)	(psia)
Base	5.34	0.71	1.122	1.11	1072.3
Scram Delay -20 msec	4.63	0.69	1.108	1.11	1070.4
Scram Delay +20 msec	5.99	0.72	1.139	1.10	1074.4
2.5% Direct Mod. Heating	4.99	0.71	1.111	1.11	1069.9
1% Direct Mod. Heating	5.44	0.71	1.127	1.10	1073.3
2% Bypass Heating	5.37	0.71	1.123	1.11	1072.1
1% Bypass Heating	5.30	0.71	1.123	1.10	1072.3
HGAP = 1200 BTU/ft^2-F-hr	5.11	0.71	1.164	0.83	1064.5
HGAP = 200 BTU/ft^2-F-hr	5.77	0.71	1.081	1.14	1094.5
Axial Varying Gap Cond.	5.33	0.71	1.121	1.11	1072.6
Core Pressure Drop -2 psi	5.54	0.71	1.126	1.11	1072.5
Jet Pump Sensitivities	Peak	Time (sec)	Peak	Time (sec)	(psia)
Suction/Diffuser K -20%	5.36	0.71	1.123	1.11	1072.3
Inertia +30%	5.12	0.71	1.119	1.11	1071.9
Inertia -30%	5.57	0.70	1.125	1.10	1072.7
Separator Sensitivities	Peak	Time (sec)	Peak	Time (sec)	(psia)
Inlet Inertia +30%	6.19	0.71	1.132	1.12	1073.9
Inlet Inertia -30%	4.52	0.70	1.113	1.08	1070.9
Separator Area -10%	5.33	0.71	1.122	1.11	1072.3
Separator Volume -10%	5.38	0.71	1.123	1.10	1072.6
Downcomer VBUB set to 1.0E6 from 3.0	5.34	0.71	1.122	1.11	1072.3
Sep. VBUB set to 1.0 from 1.0E6	5.34	0.71	1.122	1.11	1072.3
Steam Line Sensitivities	Peak	Time (sec)	Peak	Time (sec)	(psia)
Loss Coefficients -20%	5.80	0.71	1.132	1.10	1073.8
Steam Line Length +5%	5.40	0.71	1.123	1.12	1072.8
Steam Dome Volume-10%	5.78	0.71	1.132	1.10	1074.9
14 steam line nodes	5.17	0.71	1.129	1.09	1072.3
Time Step Sensitivity	Peak	Time (sec)	Peak	Time (sec)	(psia)
DTMAX = .0005	5.34	0.71	1.122	1.11	1072.3
DTMAX = .001	5.27	0.71	1.121	1.10	1072.2

4. Comparisons to LaSalle, Quad-Cities and Dresden Startup Tests

To accomplish the purpose described in Section 1.1, comparisons of plant startup test data and RETRAN predictions for ComEd BWR/5 (LaSalle) and BWR/3 (Quad-Cities and Dresden) models were performed. These comparisons serve as one of the bases for the validity of the ComEd models and methods to perform transient analysis. The key features of each model was validated by a comparison of the RETRAN analysis to the startup test data. These tests further demonstrated the applicability of the one-dimensional kinetics model in addition to the Peach Bottom Turbine Trip tests benchmarking. These benchmark comparisons are qualified through use of a prescribed set of acceptance criteria. (See Section 4.1 for details.)

The LaSalle startup data was taken from the Startup Transient Recording (STARTREC) system. The Quad-Cities and Dresden Startup test results were reported in References 11 and 12, respectively. The STARTREC data was available in digital form. The Quad-Cities and Dresden startup test results were taken from strip charts and data sheets.

The reactor water level setpoint change (RWLSC) transient was analyzed to verify the RETRAN reactor water level predictions under small level demand variations and to validate the feed water control system with respect to feed pump and feed water flow response. This transient is applicable for qualifying the models representing the vessel internals. Startup test data from LaSalle and Quad-Cities were used for comparison.

The pressure regulator setpoint change (PRSC) transient was analyzed to verify the RETRAN pressure control system and the performance of the bypass valves. Startup test data from LaSalle and Quad-Cities were used for comparison.

The dual recirculation pump trip (DRPT) transient was analyzed to validate the recirculation pump, recirculation loop, and jet pump model behavior. This test also provided evidence for the performance of the core thermal-hydraulics models. Startup test data from LaSalle and Dresden were used for comparison.

The generator load rejection with bypass/turbine trip (LRWB/TT) transient was analyzed to validate the RETRAN core thermal-hydraulics and models. The pressure control system, feed water level control system, and protective trip systems models are also verified by this test. Startup test data from LaSalle and Quad-Cities were used for comparison.

The main steam line isolation valve closure (MSIVC) transient was analyzed to validate the performance of the main steam line isolation and safety/relief valve models. This transient also serves as a validation of the recirculation pump models in the LaSalle RETRAN model. Startup test data from LaSalle was used for comparison.

Each RETRAN analysis of a startup test was initialized using case specific initial conditions. The initial conditions used for analyzing the LaSalle and Quad-Cities and Dresden startup tests are presented in Table 4.0-1 and Table 4.0-2. In addition, the jet pump M-ratio, flow control valve position (LaSalle), recirculation pump speed (Quad-Cities and Dresden), turbine inlet pressure, and separator inertia input are required for model initialization.

The comparisons of the RETRAN predictions to the measured data are discussed in Sections 4.2 and 4.3.

Table 4.0-1: LaSalle Startup Test Initial Conditions

BWR/5 LaSalle Startup Test Initial Conditions						
Initial Condition	**Units**	**RWLSC**	**PRSC**	**DRPT**	**MSIVC**	**TT/LRWB**
Reactor Power	%	95.3	99.0	68.0	94.3	95.0
Core Flow	Mlb/hr	93.6	97.7	92.2	91.0	91.0
Steam Dome Pressure	psig	1002.5	985.7	978.0	999.6	997.5
Feed Water Flow	Mlb/hr	14.0	13.9	9.9	14.1	13.7
Reactor Water Level	In.N.R.	39.8	37.9	35.5	35.0	35.1
Steam Flow	Mlb/hr	13.1	13.5	9.2	13.1	12.9

Table 4.0-2: Quad-Cities and Dresden Startup Test Initial Conditions

BWR/3 Quad-Cities & Dresden Startup Test Initial Conditions					
Initial Condition	**Units**	**RWLSC**	**PRSC**	**DRPT**	**TT**
Reactor Power	%	91.5	22.5	91.5	67.0
Core Flow	Mlb/hr	98.0	36.5	99.1	54.0
Steam Dome Pressure	psig	998.0	958.0	1002.0	991.0
Feed Water Flow	Mlb/hr	8.8	2.21	9.3	6.20
Reactor Water Level	In.N.R.	32.5	29.0	35.0	29.0
Steam Flow	Mlb/hr	*	2.64	*	*

* Initial steam flow was not recorded.

4.1 Screening Criterion for Startup Test Benchmarking

A two-step approach was used to assess the benchmark predictions. The first step of the assessment process was a screening based solely on the magnitude of key parameters. The differences between the measured and predicted maximum, minimum, and/or steady state values were typically used. These differences were compared solely to numerical criteria identified in Table 4.1-1, which are based on the method described in Reference 13. The values in this table take into account measurement uncertainties and are based on plant rated conditions from Table 1.1-1. One of three ratings was assigned to the differences. These ratings are acceptable, generally acceptable, and unacceptable, represented by the symbols (+), (0), and (-). Benchmark predictions which met the acceptable or generally acceptable rating were considered successful and the second step was not required. Those which fell outside this screening criteria were identified for further assessment. This second assessment step relied on additional review of both the model and the test data to determine the cause of the benchmark analysis discrepancy and to determine the validity of the benchmark analysis.

All values listed in Table 4.1-1 are from Reference 13 with the exception of bypass valve (BPV) position. The criterion for the BPV position was based on the criterion for steam flow rate. The BPV flow rate is linear with position; a 5% change in BPV position will result in 5% change in steam flow rate. Therefore, the criteria for the BPV is set at <5%, <10%, and >10%.

Table 4.1-1, Ratings for RETRAN Comparisons to Data

	Rating		
Parameter	**(+)**	**(0)**	**(-)**
Steam Dome Pressure	<10psi	<20psi	>20psi
Downcomer Level (Reactor Water Level)	<5in	<10in	>10in
Steam Flow Rate	<5%	<10%	>10%
Feedwater Flow Rate	<5%	<10%	>10%
Recirculation Loop Flow Rate	<5%	<10%	>10%
Core Flow Rate	<5%	<10%	>10%
Reactor Power	<3%	<6%	>6%
Bypass Valve Position	<5%	<10%	>10%

4.2 LaSalle BWR/5 Startup Test Comparisons

4.2.1 Reactor Water Level Setpoint Change (RWLSC)

4.2.1.1 Test Description

The reactor water level setpoint change startup test was conducted at LaSalle Unit 2 on September 4, 1984.

The purpose of the reactor water level setpoint change was to demonstrate satisfactory reactor water level and feed water flow rate control.

The transient was initiated by rapidly (1-2 seconds) changing the setpoint to the position determined to result in a negative 5 inch reactor water level change. Changes in level were demanded via the startup level controller manual/auto station. The test was performed with the "A" and "B" turbine driven feed pumps and with water level controller operating in 3-element control.

4.2.1.2 RETRAN Modeling of Test

The LaSalle RETRAN base model was initialized to the power and flow rates described in Table 4.0-1. The reactor water level control system was slightly modified to allow the water level setpoint to be a function of time. Water level setpoint was reduced from 39.8 inches narrow range to 34.8 inches narrow range over a period of 2 seconds. The feed water level control system was set up with the appropriate dynamic compensation to capture the settings used for that specific test. Plotted results were from the LaSalle RETRAN control blocks for the "Sensed" indications.

Some adjustments were made to the RETRAN model in order to benchmark this startup test. One control block and one general data table were changed in order to model this transient. General data table 3 which is the setpoint setdown table was changed to ramp the initial level setpoint down 5 inches over 2 seconds. A function generator was changed to take information from simulation time rather than a reactor protection trip. This had the effect of changing the function generator from functioning on a trip signal to elapsed time. The steam flow and feed water flow indicated from the startup test data as listed in Table 4.0-1 were not equal. Because RETRAN does not allow a mass flow imbalance for steady state initialization, these values must be equal. Based on a reactor core heat balance, a value of 13.6 Mlb/hr was used for the steam and feed water flow.

4.2.1.3 Results

Plant Response:

When the reactor water level setpoint is reduced, the feed water controller begins a steady decrease to reduce the reactor water level. The noise in the water level signal produces small oscillations except when the level is dropping rapidly. The feed water controller allows feed water flow to decrease until water level begins to decrease. At this point, the feed water control attempts to stabilize the reactor water level.

Model Response:

The parameters of interest for this test are the reactor narrow range water level and the feed water flow. The narrow range level and feed water flow are the only system parameters that experience any significant change and these are the key parameters for benchmarking the feed water level control system. Core power, core flow, reactor pressure and steam flow show very small changes.

Figure 4.2-1 shows the comparison of feed water flow change from initial value. The results show acceptable agreement. There is reasonable agreement for timing. Figure 4.2-2 shows the comparison of the water level response to the setpoint change. The results show acceptable agreement. It is judged that the RETRAN results would be acceptable when compared to the data without the noise. The RETRAN model predicts a slightly slower decrease in level than the startup test data, but the overall trend is preserved. Core power, steam flow rate, and reactor pressure remain relatively constant as expected over the course of the transient.

This transient was mild and had magnitude changes which were close to the magnitude changes for a (+) rating. Although the screening criterion was used to rate the parameter magnitude changes, engineering judgment was used to determine the validity of the benchmark analysis. RWLSC requires that level will change by five inches. The analysis results were examined to assure that the model achieved the prescribed change. The other parameter of interest for this test was also examined for reasonableness with respect to the magnitude changes prescribed by the test.

In summary, the RETRAN model accurately simulated the feed water flow and the reactor water level responses. The model's feed water level control system was shown to be accurate over the range presented. All pertinent RETRAN calculated variables behave as anticipated. Table 4.2-1 summarizes the ratings for the RWLSC benchmark.

Table 4.2-1: Ratings for the LaSalle BWR/5 RWLSC Benchmark

Parameter	Rating
Reactor Water Level	+
Feed Water Flow	+

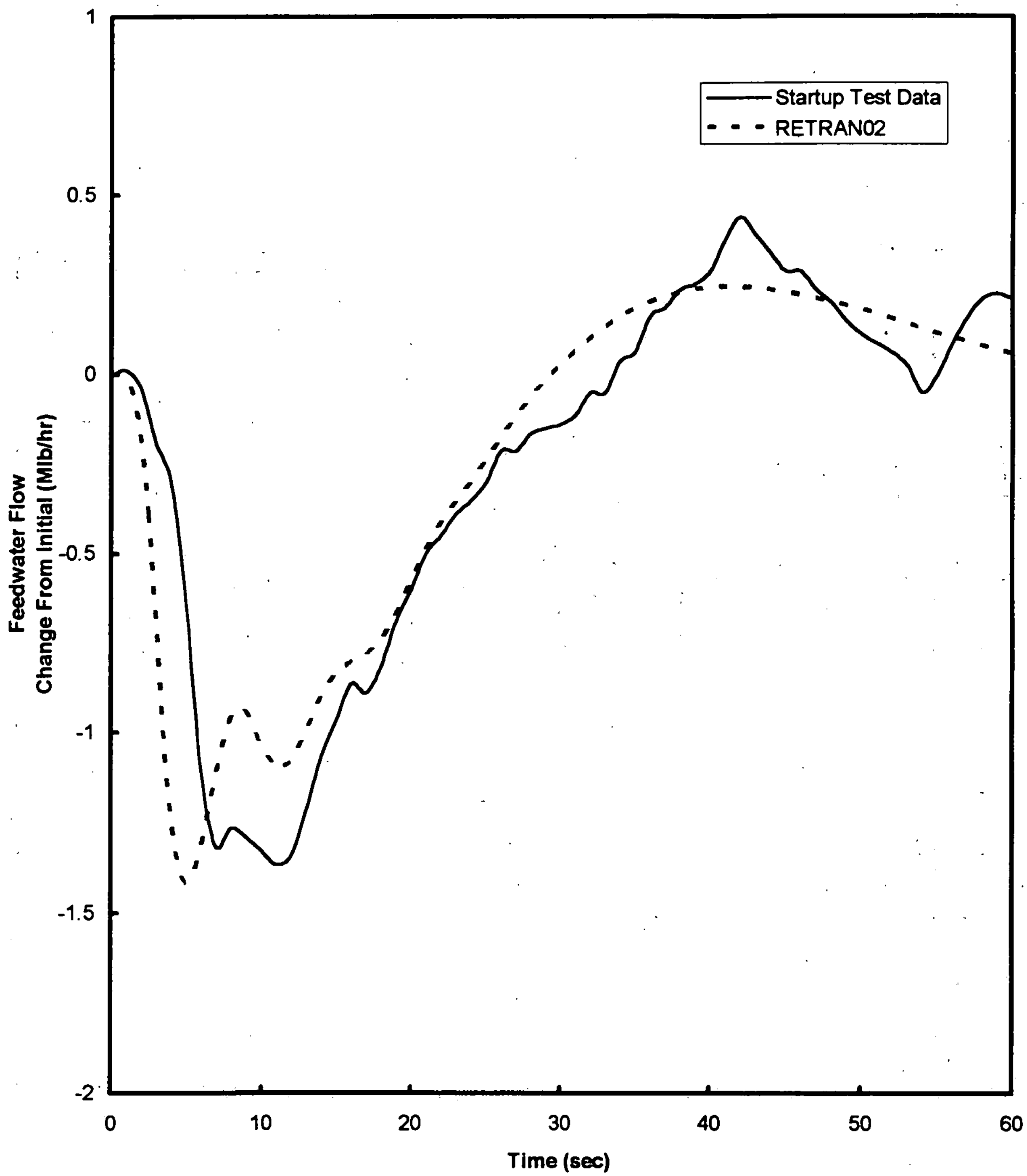

Figure 4.2-1: LaSalle BWR/5 RWLSC Feed water Flow

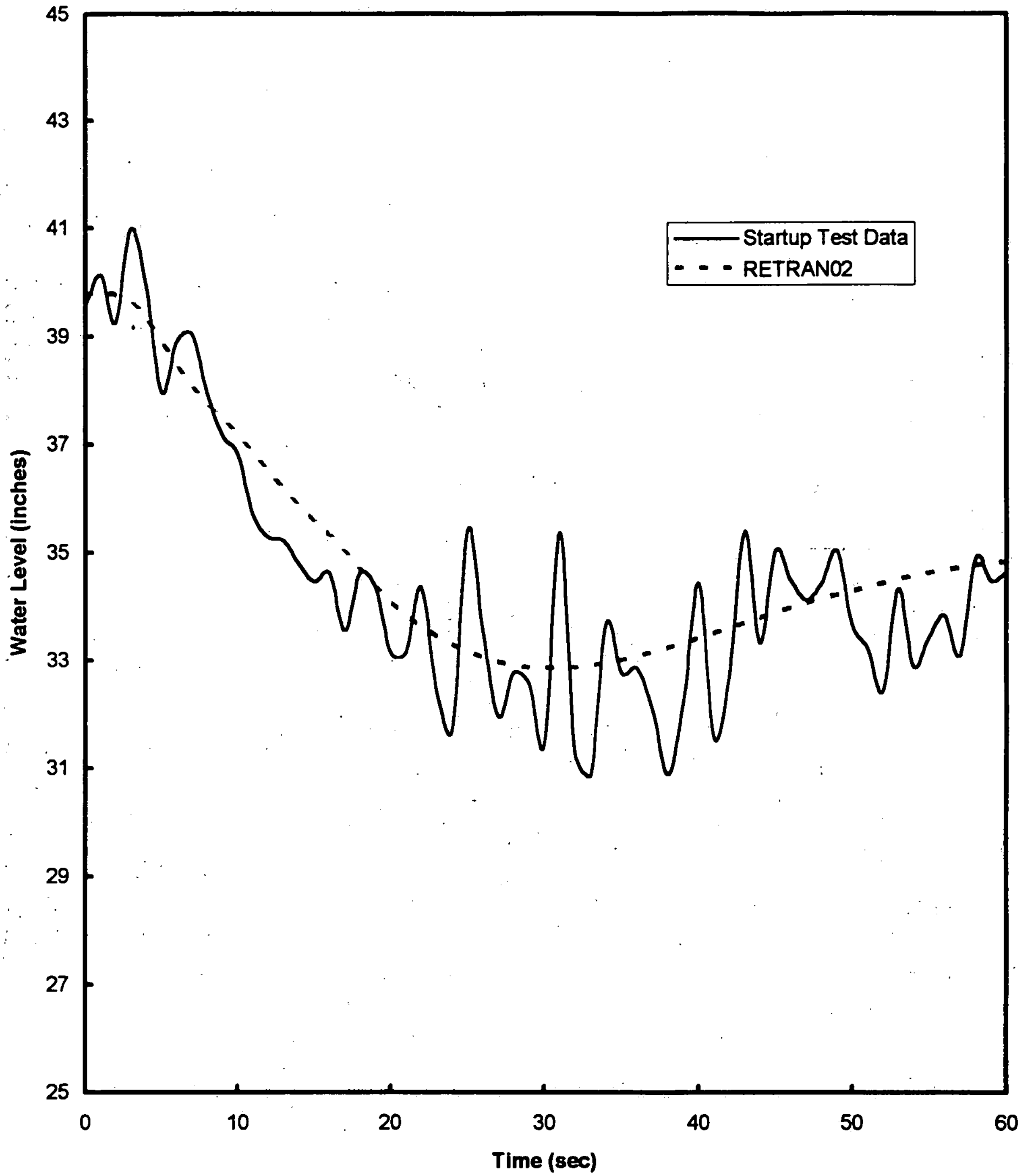

Figure 4.2-2: LaSalle BWR/5 RWLSC Water Level

4.2.2 Pressure Regulator Setpoint Changes (PRSC)

4.2.2.1 Test Description

The pressure regulator setpoint change startup test was conducted at LaSalle Unit 2 on August 4, 1984.

The purpose of the pressure regulator setpoint change was to determine the optimum settings for the pressure control loop by analysis of the transients. Also, the test should demonstrate the smooth transition between the turbine control valves and bypass valves when the reactor steam generation exceeds the steam flow through the turbine.

The pressure regulator setpoint change startup test was initiated by a 10 psi downward step in the "A" pressure regulator facilitated by a temporary test installation in the electrohydraulic control cabinet. Prior to the initiation of this event the load setpoint was reduced until bypass valves 1 and 2 of the 5 were fully open. This transient was performed with the load limit set to obtain bypass valve actuation.

4.2.2.2 RETRAN Modeling of Test

Some adjustments were made to the RETRAN model in order to benchmark this startup test. To model a negative 10 psi change, the pressure regulator setpoint versus time table in RETRAN had to be changed. The duration over which the pressure setpoint was changed was not specified in the documentation and was assumed to be 0.1 seconds. The steam flow and feed water flow from the startup test data as listed in Table 4.0-1 were not equal. Because RETRAN does not allow a mass flow imbalance for steady state initialization, these values must be set equal. Based on a reactor core heat balance, a value of 13.9 Mlb/hr was used for the steam and feed water flow. The time constants in the model's pressure regulators were changed to reflect the settings at the time of the startup test. A change was required because the total bypass valve flow was initially 42.6% at the start of the transient. To achieve this initial condition, the load setpoint in the pressure control system was ramped down such that the bypass valve opened to the desired position. Opening the bypass valve slightly changes some of the initial conditions listed in Table 4.0-1. Any variable that did not match the initial value was plotted as change from initial.

4.2.2.3 Results

<u>Plant Response:</u>

In this test, the pressure regulator setpoint is reduced by 10 psi. Since the load limiter has been reduced, the bypass valves open further to lower the turbine inlet pressure. As the pressure decreases, the bypass valves close partially and cause the pressure to increase slightly. The bypass valves will adjust to stabilize the pressure at its setpoint value. The change in main steam flow follows the change in bypass valve position. As a result of the pressure decrease, the core voiding increases which lowers the core power and raises the reactor water level.

Model Response:

The parameters of interest for this test are the reactor pressure, core power, steam flow and the bypass valve position. The model should stabilize at the new pressure setpoint and the model should predict the bypass valve opening and the small variations in core power and steam flow. These are the key parameters for benchmarking the pressure control system and the reactor kinetics response to small pressure changes.

Figure 4.2-3 shows the comparison of the change in steam flow. The data shows acceptable agreement. Figure 4.2-4 shows the comparison for reactor core power. The data shows acceptable agreement. It is judged that the RETRAN results would be acceptable when compared to the data without the noise. Figure 4.2-5 shows the comparison for the reactor pressure. The RETRAN results follow the data and shows acceptable agreement. The reactor pressure matches the data initially but the final pressure differs by approximately 1.5 psi. The RETRAN results show a 10 psi change which is consistent with the 10 psi change in pressure regulator setpoint. The plant data shows only an 8.5 psi change. Figure 4.2-6 shows the comparison of bypass valve position. The RETRAN results follow the data and results show generally acceptable results.

This transient was mild and had certain magnitude changes which were close to the magnitude changes for a (+) rating. Although the screening criterion was used to rate the parameter magnitude changes, engineering judgment was used to determine the validity of the benchmark analysis. PRSC requires that pressure will change by ten psi for the test. The analysis results were examined to assure that the model achieved the prescribed change. The other parameters of interest for this test were also examined for reasonableness with respect to the magnitude changes prescribed by the test.

In summary, the RETRAN model accurately simulated the pressure regulator setpoint change. The pressure control system functions properly for a change in demand over the range presented. The bypass valve performance is modeled correctly. All pertinent RETRAN calculated variables behave as anticipated. Table 4.2-2 summarizes the ratings for the PRSC benchmark.

Table 4.2-2: Ratings for the LaSalle BWR/5 PRSC Benchmark

Parameter	Rating
Bypass Valve Position	0
Steam Flow Rate	+
Steam Dome Pressure	+
Core Power	+

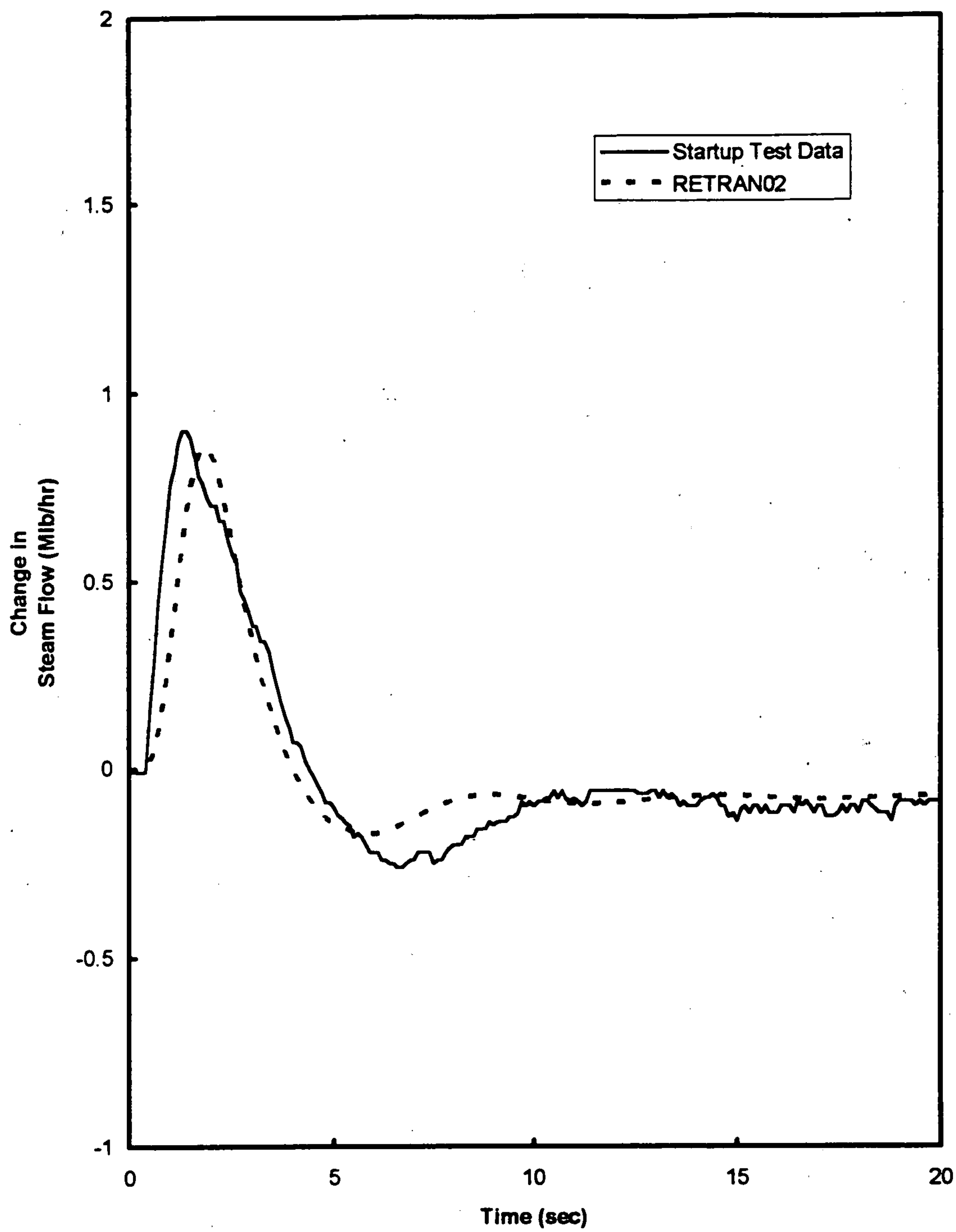

Figure 4.2-3: LaSalle BWR/5 PRSC Steam Flow

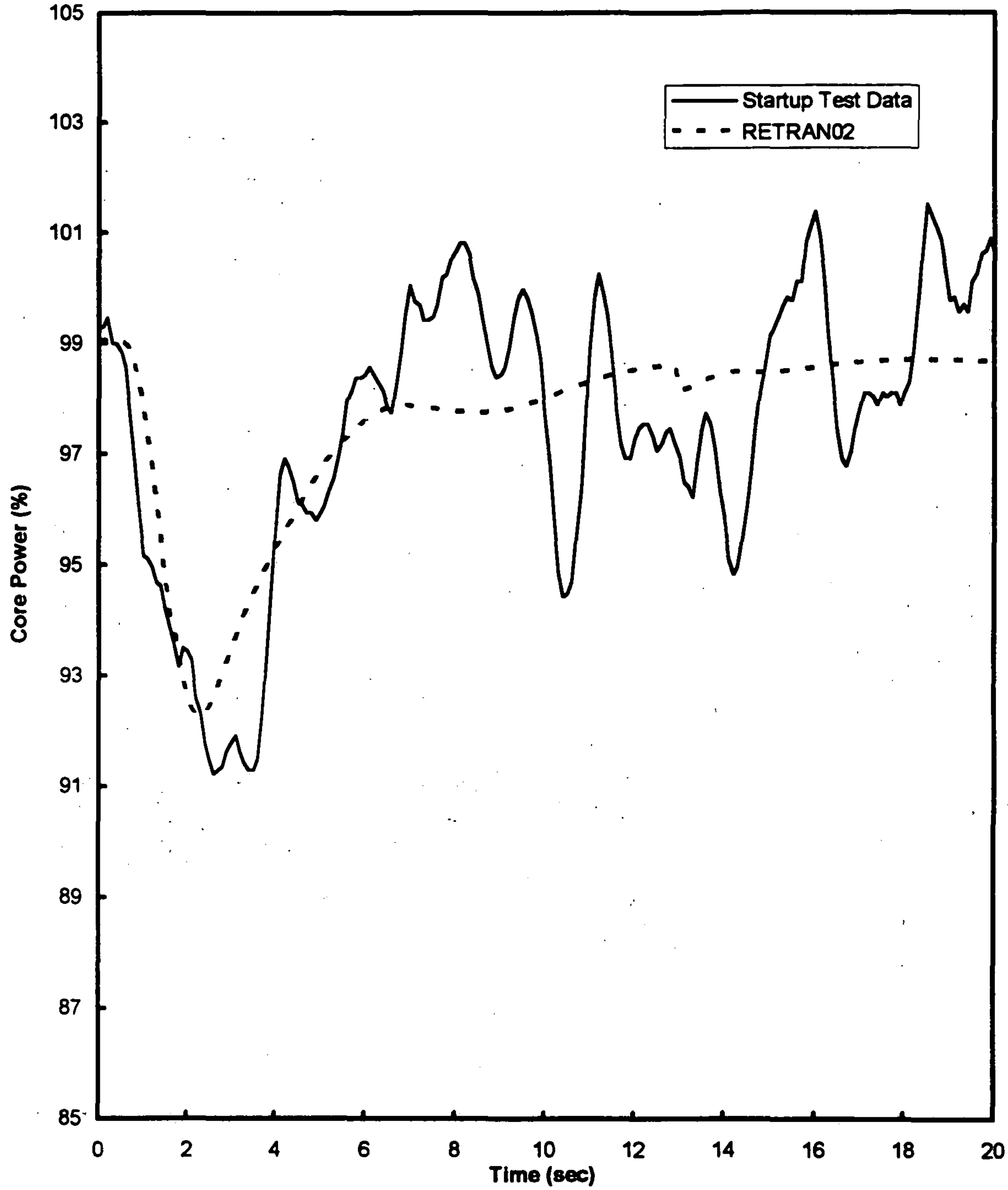

Figure 4.2-4: LaSalle BWR/5 PRSC Core Power

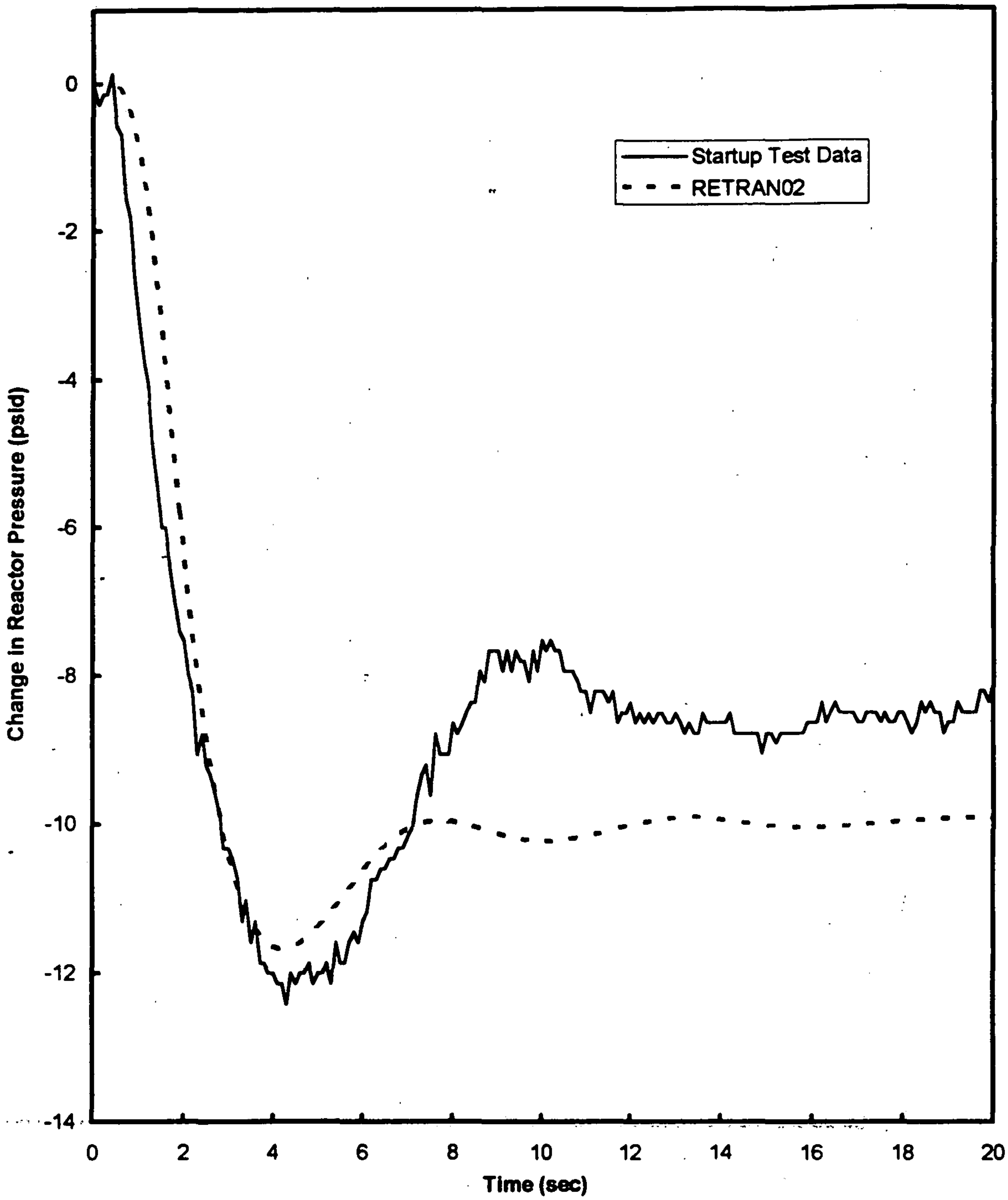

Figure 4.2-5: LaSalle BWR/5 PRSC Dome Pressure

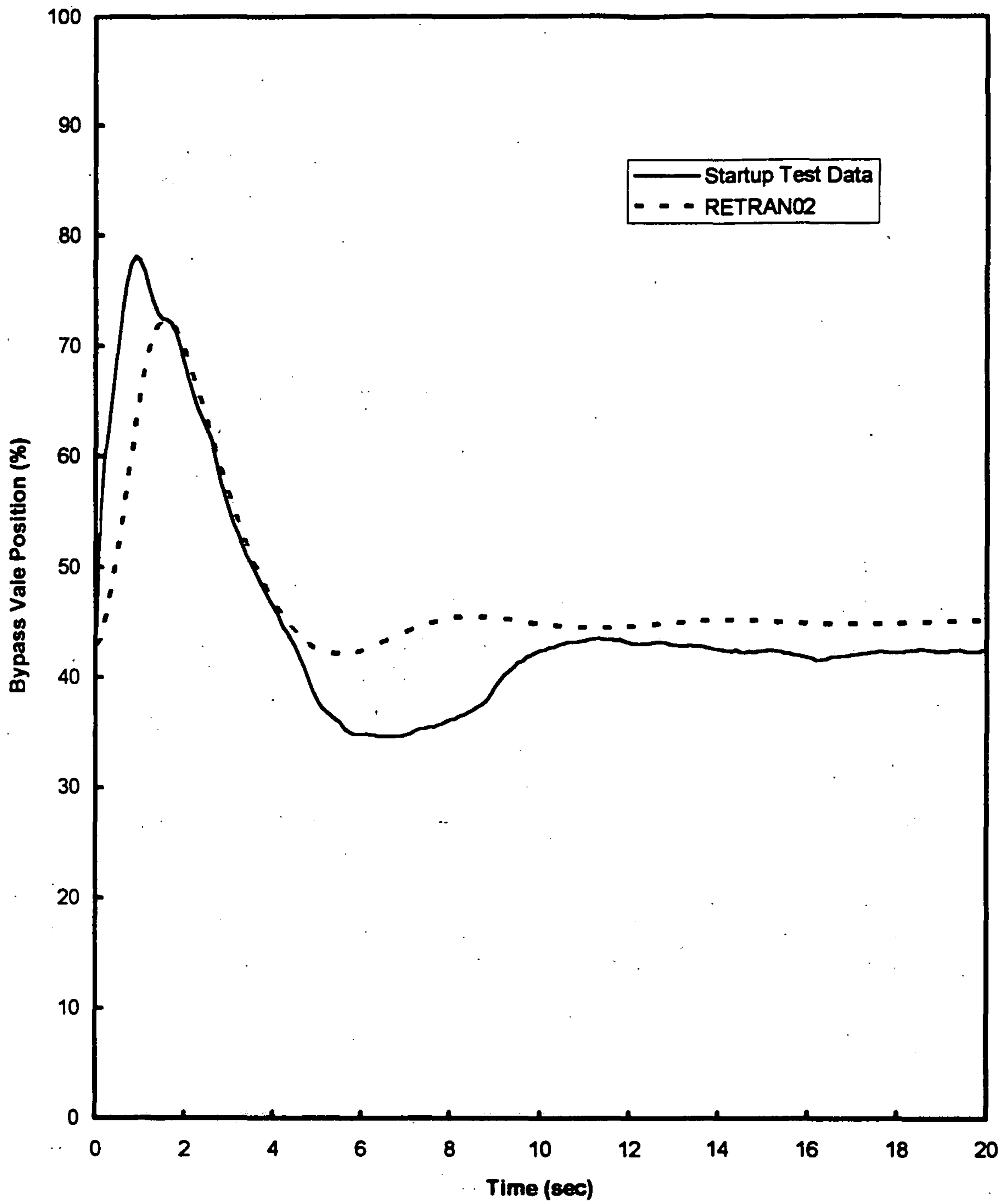

Figure 4.2-6: LaSalle BWR/5 PRSC Bypass Valve Position

4.2.3 Dual Recirculation Pump Trip (DRPT)

4.2.3.1 Test Description

The dual recirculation pump trip startup test was conducted at LaSalle Unit 2 on June 5, 1984.

The purpose of this test was to obtain recirculation system performance under different conditions such as pump trip and flow coastdown, to verify that no recirculation system cavitation will occur in the operating region of the power-flow map, and to record and verify acceptable performance of the recirculation two pump circuit trip system.

The startup test transient was initiated by simulating a turbine control valve fast closure. As a result, the circuit breakers in the high frequency power supply automatically opened, thereby tripping the recirculation pumps.

4.2.3.2 RETRAN Modeling of Test

The RETRAN model makes provisions for manual recirculation pump trips (trips on time) in the base model. To analyze the two recirculation pump trips, two trip cards were altered. The trip setpoints were changed to obtain a manual recirculation pump trip.

Some adjustments to the RETRAN model were made in order to benchmark this startup test. The steam flow and feed water flow from the startup test data as listed in Table 4.0-1 were not equal. Because RETRAN does not allow a mass imbalance for steady state initialization, these values must be set equal. Based on a reactor core heat balance, a value of 9.4 Mlb/hr was used for the steam and feed water flow. The recirculation control system was set to manual control and the recirculation flow control valves were not allowed to change position to be consistent with the plant test. Since the flow control valves did not change position during this transient, the entire pump coastdown response was based on the homologous pump curves, the specified pump parameters, and the junction inertias of the recirculation system.

4.2.3.3 Results

<u>Plant Response:</u>

The recirculation pump speed and the recirculation drive flow decrease. The flow control valve does not change position during this transient. Since the recirculation drive flow decreases, core flow decreases. This causes increased core voiding and a reduction in core power. Since less power is being produced with less steam production, the reactor pressure decreases. The pressure control system attempts to compensate for this effect by closing the turbine control valves. Increased core voiding and decreasing reactor pressure cause the water level to swell, or rise. The feed water controller, in turn, reduces flow until the level stops increasing. Main steam line flow decreases steadily throughout this event.

Model Response:

The parameters of interest for this test are the core flow, core power, reactor pressure and steam flow. The model should predict the large decrease in core flow and core power. The pressure control system should also respond to the changes in steam flow and stabilize reactor pressure at the new conditions. These are the key parameters for benchmarking the RETRAN thermal hydraulics during a rapid core flow change, the recirculation pump model, the recirculation control system, the reactor kinetics and the pressure control system.

Figure 4.2-7 shows the comparison for core flow. The RETRAN calculation follows the same trend as the test data. For the first 20 seconds, the magnitude and timing are excellent (the dip in core flow at 16.7 seconds appears to be an anomalous recording of the data). After 20 seconds, the RETRAN flow levels out to a slightly lower value than the measured data which shows acceptable agreement. Figure 4.2-8 shows the steam flow comparisons. The results show acceptable agreement. Figure 4.2-9 shows the comparison of reactor core power. The results show acceptable agreement. Figure 4.2-10 shows the comparison of reactor dome pressure. The predicted pressure follows the data and shows acceptable agreement

The reasons for the minor differences between the model and the data are as follows. The total core flow measurement is based on a summation of the jet pump flows. These jet pump flows are calculated using pressure differentials. RETRAN calculates the mass flow rate at the junction in the model. This difference could account for the small difference between the predicted and measured data. Also, the time constant associated with the measured core flow, pressure, and drive flow is uncertain. Hence, the timing of the two curves may not be accurately represented. A reasonable value for this time constant was chosen based on time constants for similar systems. The reason for the dome pressure predicted by RETRAN being lower than the measured data is a result of the pressure control system. This system is modeled using the actual system gains where possible. When the actual plant controller settings were not listed, representative values were used. This is the reason that the pressure settles out at a slightly lower pressure with less oscillations than the measured data.

In summary, the results of the RETRAN model for the dual recirculation pump trip demonstrate that the recirculation system components are properly modeled and can adequately predict the transient behavior of the plant. Table 4.2-3 summarizes the ratings for the DRPT benchmark.

Table 4.2-3: Ratings for the LaSalle BWR/5 DRPT Benchmark

Parameter	Rating
Core Flow	+
Steam Flow	+
Steam Dome Pressure	+
Core Power	+

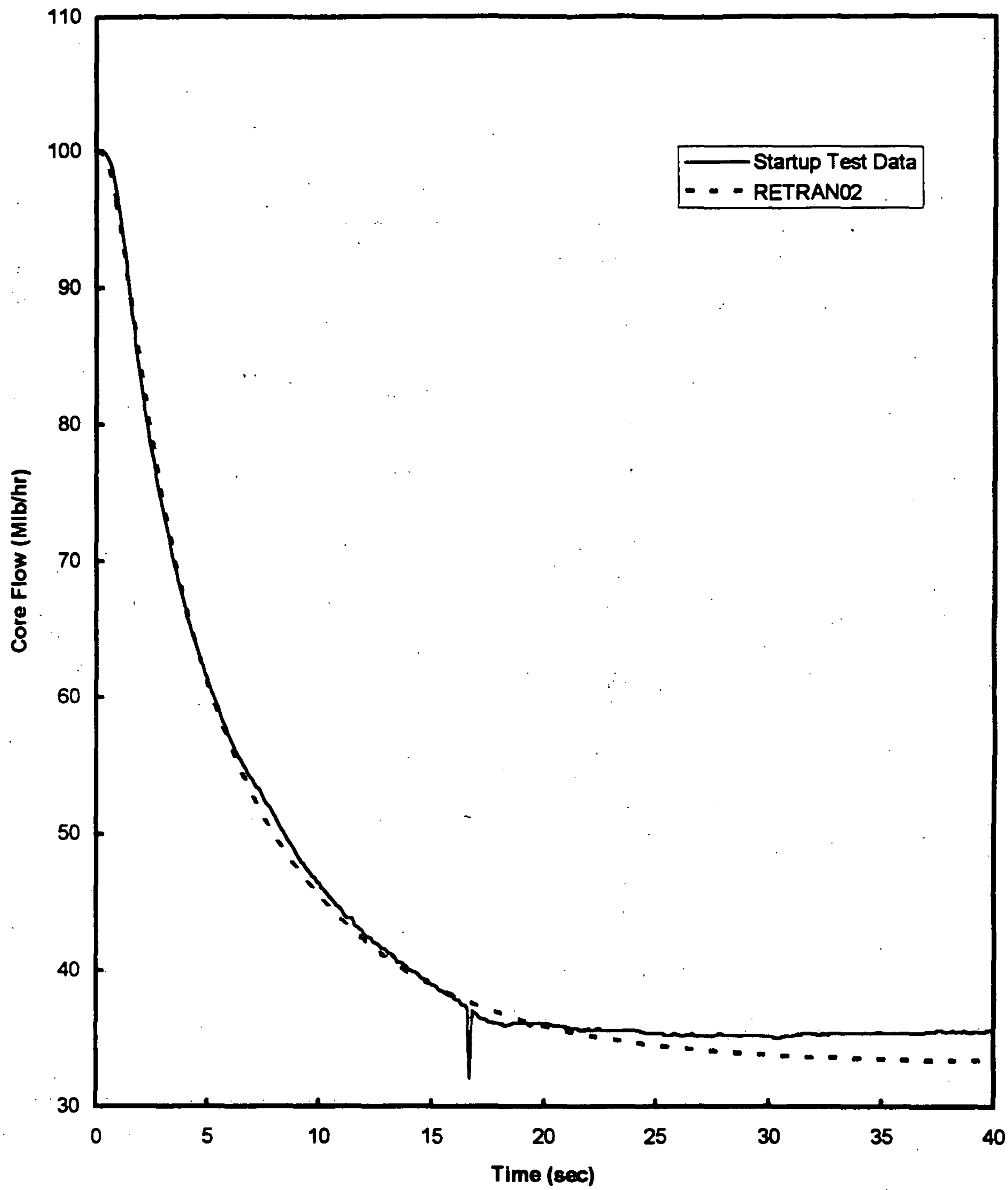

Figure 4.2-7: LaSalle BWR/5 DRPT Core Flow

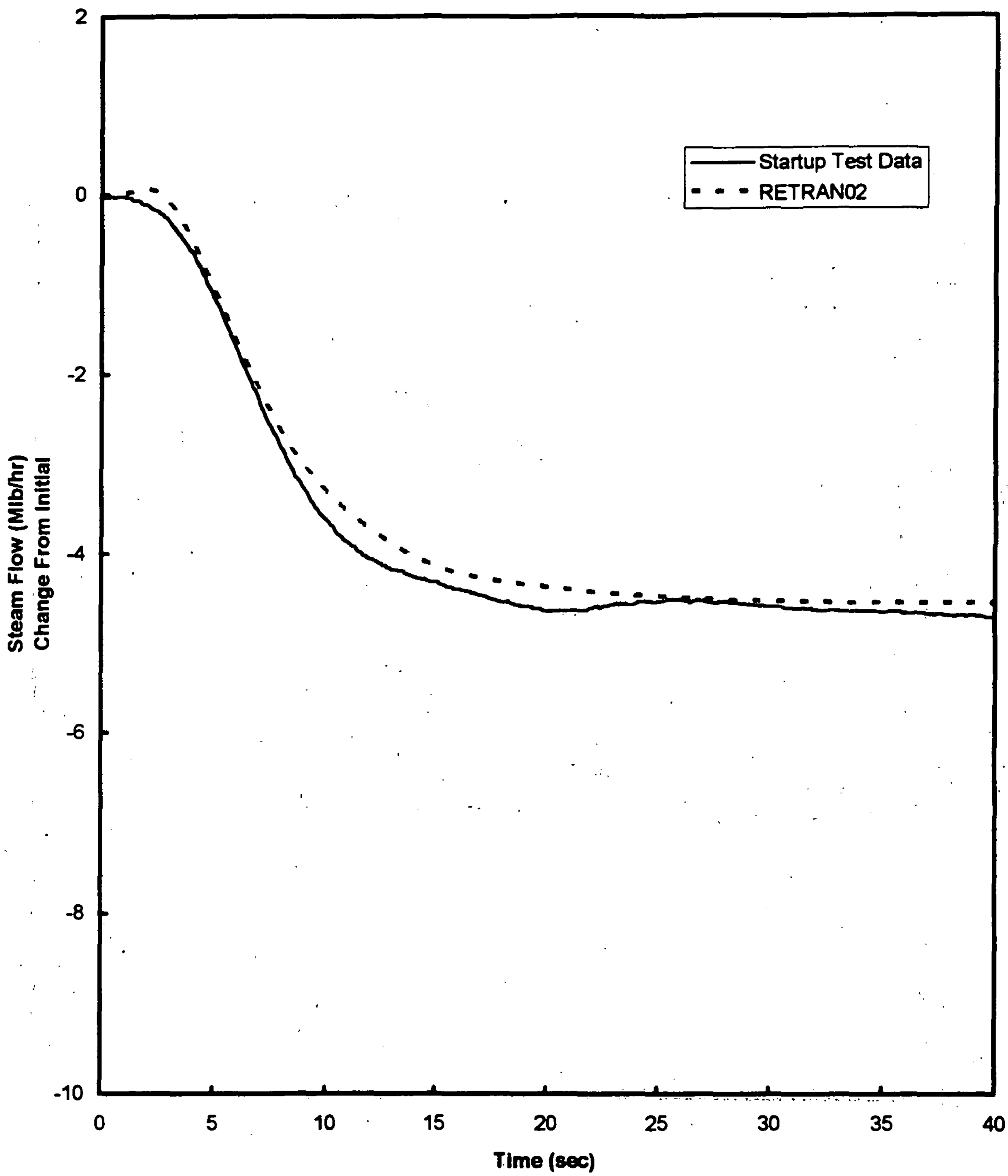

Figure 4.2-8: LaSalle BWR/5 DRPT Steam Flow

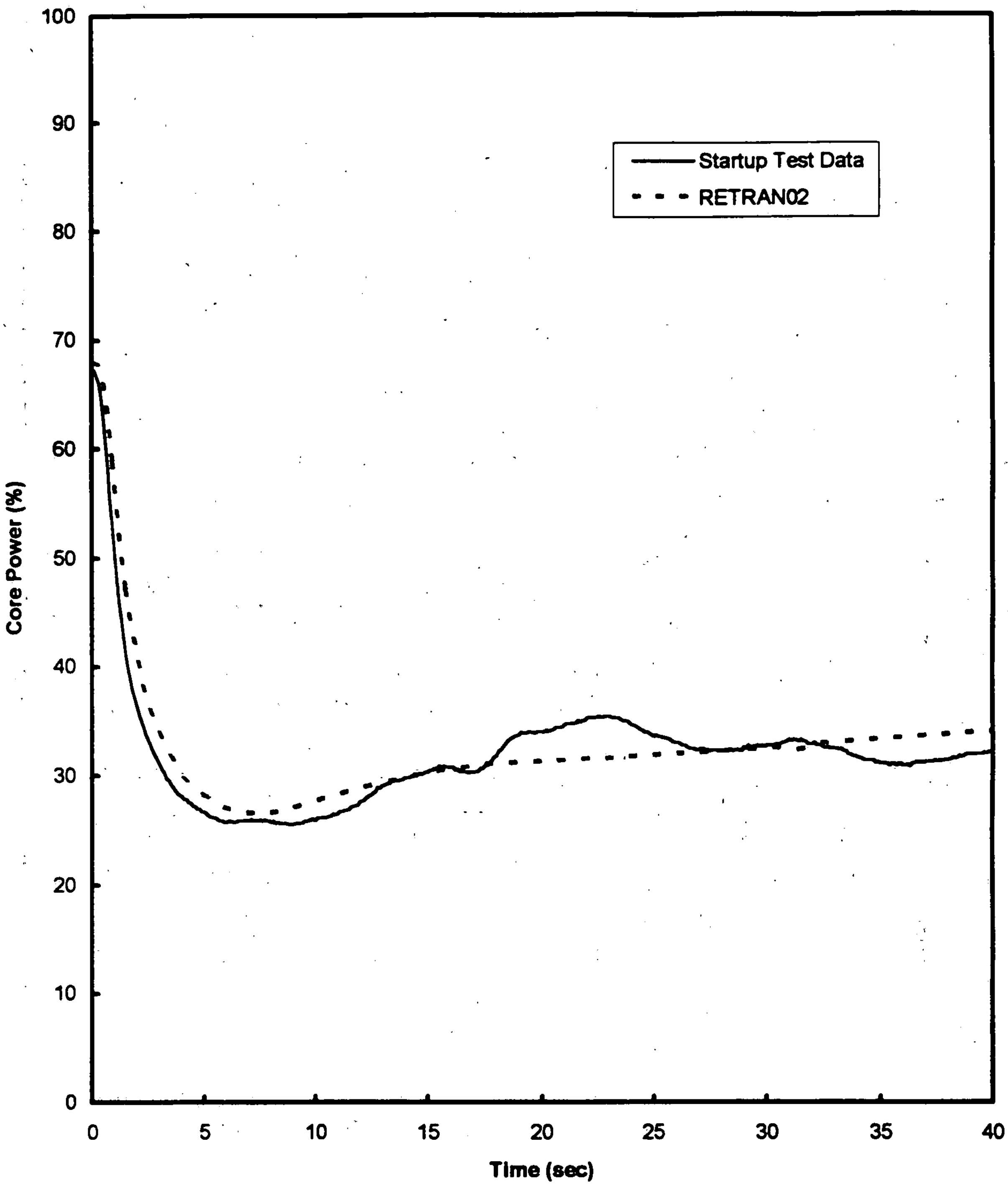

Figure 4.2-9: LaSalle BWR/5 DRPT Core Power

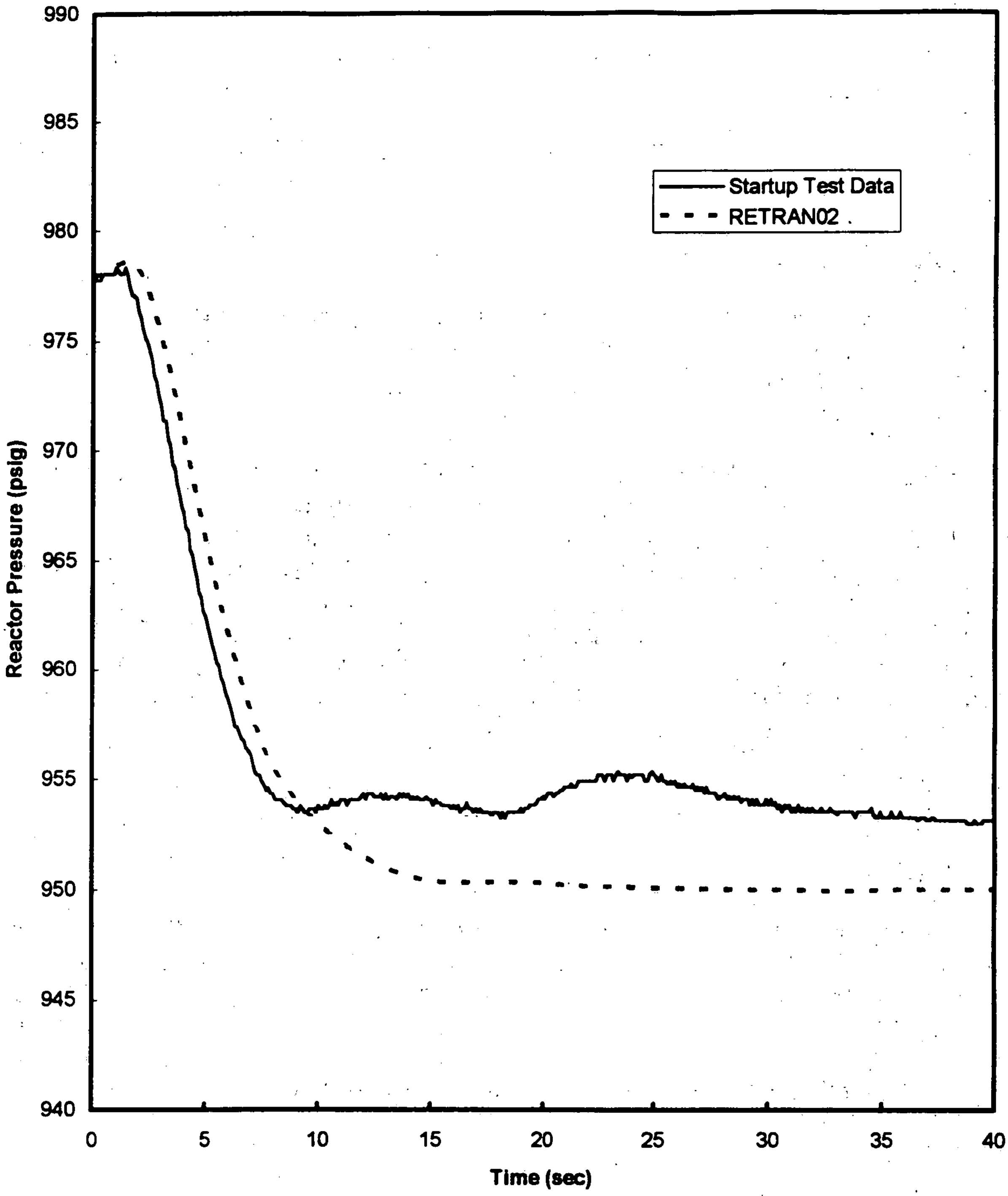

Figure 4.2-10: LaSalle BWR/5 DRPT Dome Pressure

4.2.4 Load Rejection/Turbine Trip With Bypass (LRWBP)

4.2.4.1 Test Description

The generator load rejection with bypass startup test was conducted at LaSalle Unit 1 on September 3, 1983.

The purpose of performing this startup test was to demonstrate the response of the reactor, its control systems, and protective trips in the turbine and the generator.

4.2.4.2 RETRAN Modeling of Test

The manual load rejection trip was included in the base model. This trip was initiated at the first time step. The initiation times of the recirculation pump trips and three relief valve trips were taken directly from Reference 14. These changes were incorporated to obtain the best estimate results for the benchmarking calculations.

Some adjustments to the RETRAN model were made in order to benchmark this startup test. The steam flow and feed water flow from the startup test data as listed in Table 4.0-1 were not equal. Because RETRAN does not allow a mass flow imbalance for steady state initialization, these values must be equal. Based on a reactor core heat balance, a value of 13.3 Mlb/hr was used for the steam and feed water flow. Steam flow will be plotted in terms of change from initial because the initial RETRAN value is different from the initial STARTREC value. One trip was modified to facilitate the generator load rejection. Nominal relief valve setpoints were used in order to better match the startup test conditions.

4.2.4.3 Results

Plant Response:

As the steam flow stops, pressure builds in the steam line. The bypass valves open promptly to the 100% position in less than 0.3 seconds. The build up in pressure creates a pressure wave that travels back to the core. Increased pressure in the core collapses voids causing a decrease in reactor water level. The pressure also impacts the core flow and causes a brief increase before the recirculation pumps trip to low speed. The reactor scram occurs on a generator load rejection signal.

Model Response:

The parameters of interest for this test are the core power, core flow, reactor pressure, steam flow and bypass valve position. The model should predict the core power, core flow and reactor pressure well since these are key variables for calculating critical power. The model should predict the bypass opening and the change in steam flow.

These are the key parameters for benchmarking, the RETRAN thermal hydraulics during pressurization events, the reactor kinetics, the pressure control system and the bypass valve model.

Figure 4.2-11 shows the comparison of reactor dome pressure. The results show acceptable agreement for the pressurization. The RETRAN model slightly overpredicts the dome pressure. This overprediction is acceptable and conservative. Figure 4.2-12 shows the comparison of reactor power. The results show acceptable agreement. Figure 4.2-13 presents the comparison of steam flow rate change from initial value. The results show acceptable agreement. The magnitude of the first oscillation is not within the acceptable criteria. The magnitude of this oscillation is not important compared to the stabilized bypass flow response, which shows acceptable agreement. Figure 4.2-14 shows the comparison of bypass valve position. The results show acceptable agreement. Figure 4.2-15 presents the comparison of core flow rates. The results show acceptable agreement.

This test uses the cross section file generated for the LaSalle Unit 2 startup test initial conditions. This was considered acceptable since this test was conducted at about the same core power and exposure. This assumption is validated by the core power comparison presented in Figure 4.2-12.

In summary, the RETRAN model accurately simulated the plant response for the load rejection/turbine trip with bypass. All pertinent RETRAN calculated variables behave as anticipated. Table 4.2-4 summarizes the ratings for the LRWB benchmark.

Table 4.2-4: Ratings for the LaSalle BWR/5 LRWBP Benchmark

Parameter	Rating
Steam Dome Pressure	+
Core Power	+
Core Flow Rate	+
Steam Flow Rate	+
Bypass Valve Position	+

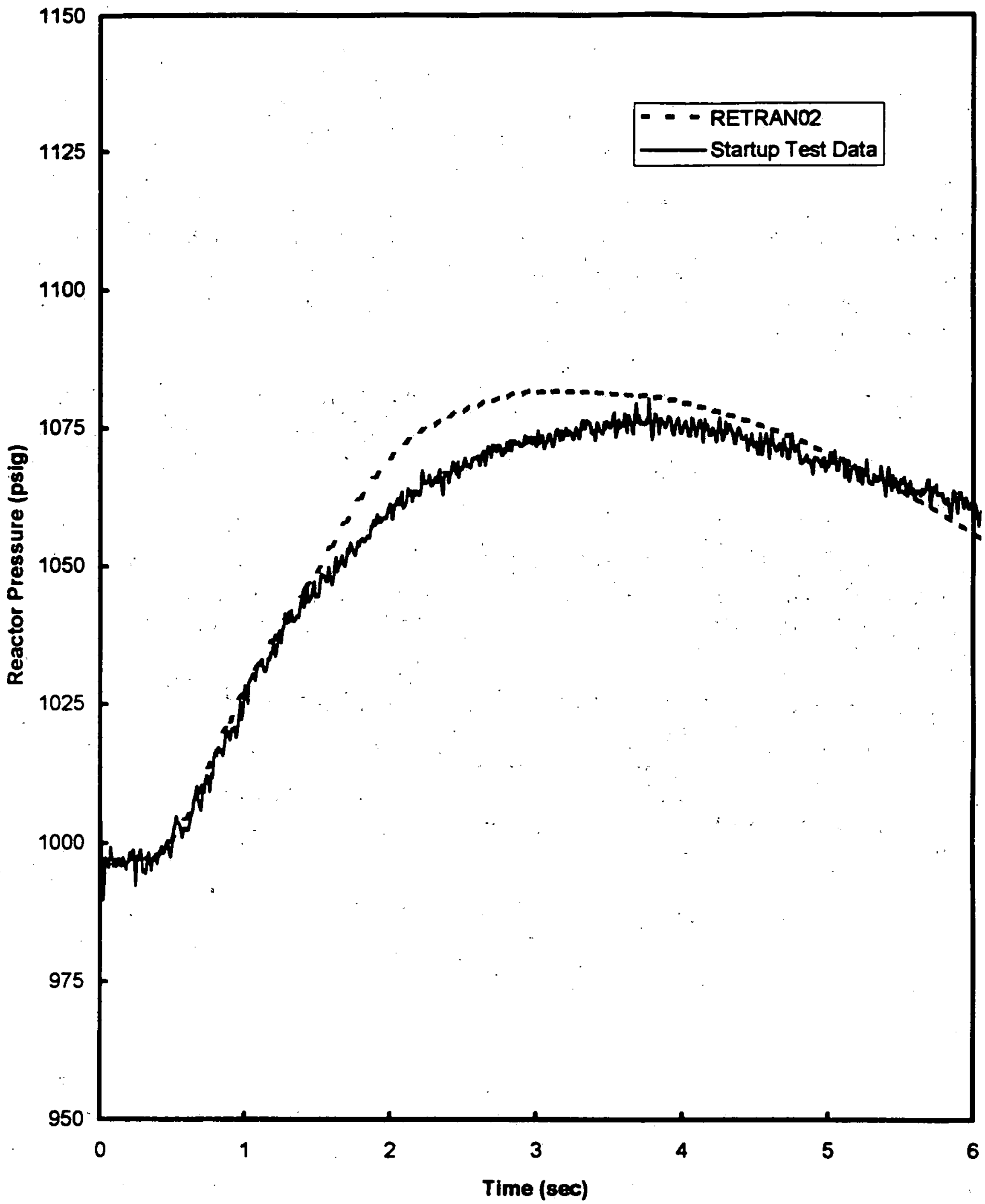

Figure 4.2-11: LaSalle BWR/5 LRWBP Dome Pressure

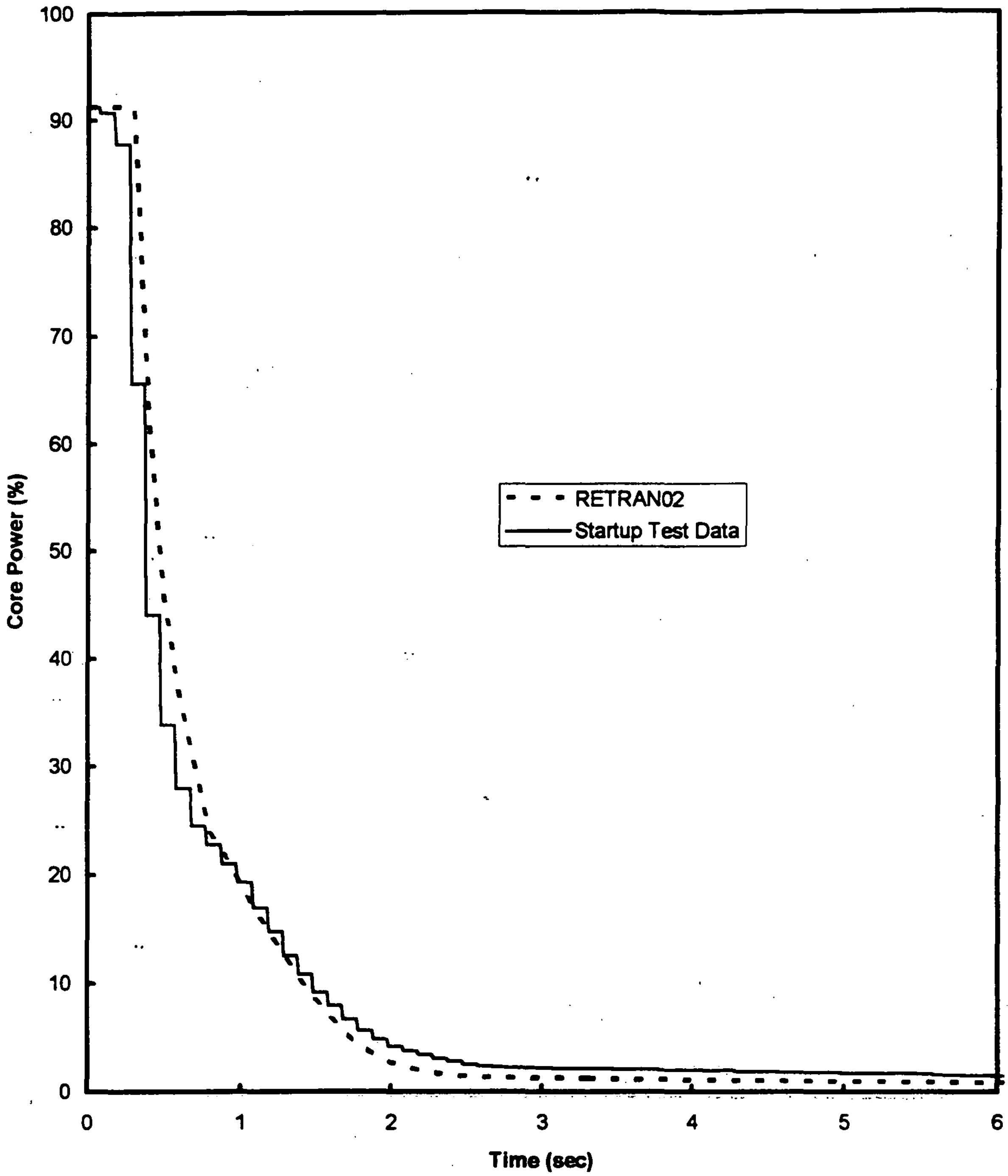

Figure 4.2-12: LaSalle BWR/5 LRWBP Core Power

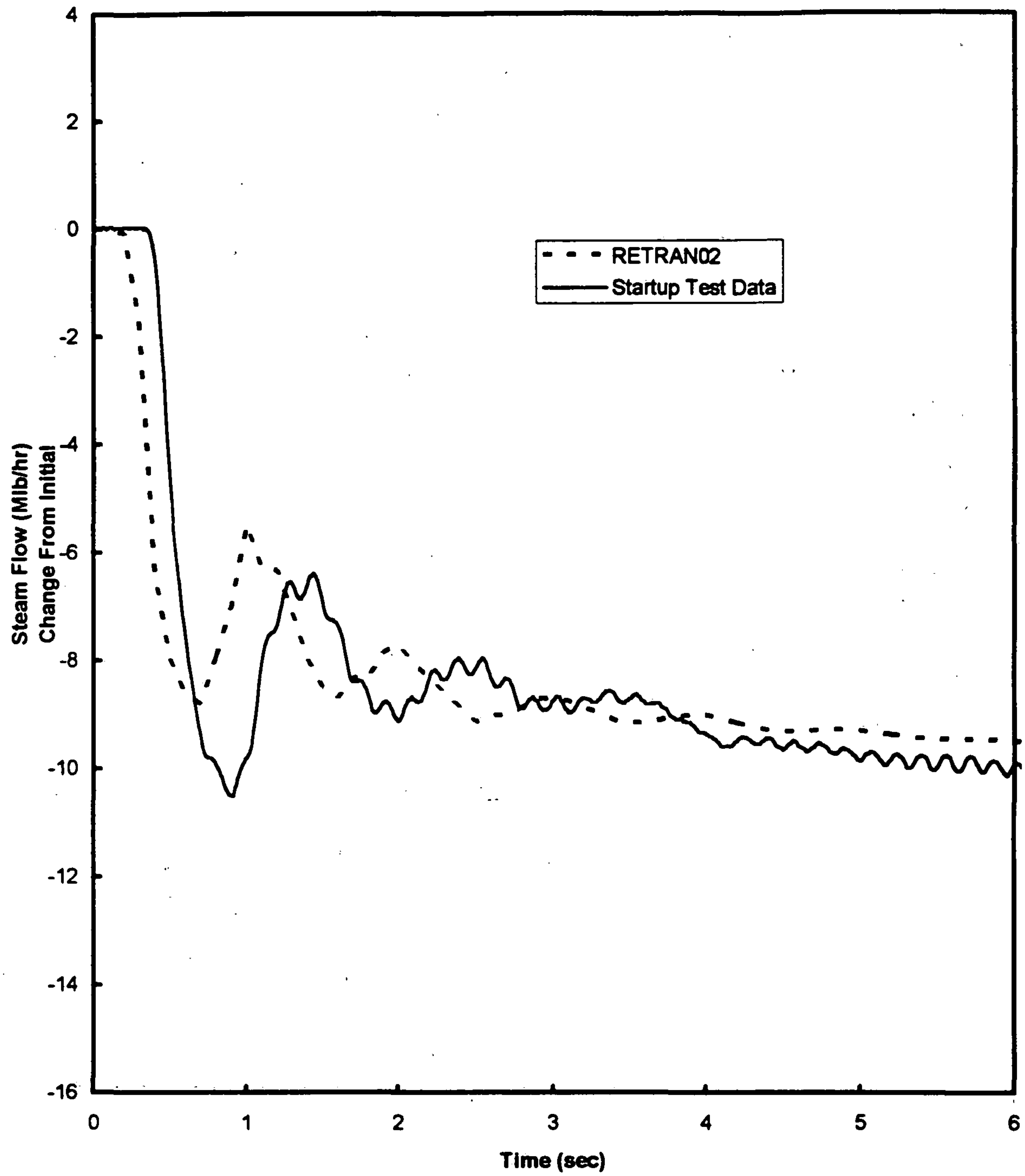

Figure 4.2-13: LaSalle BWR/5 LRWBP Steam Flow

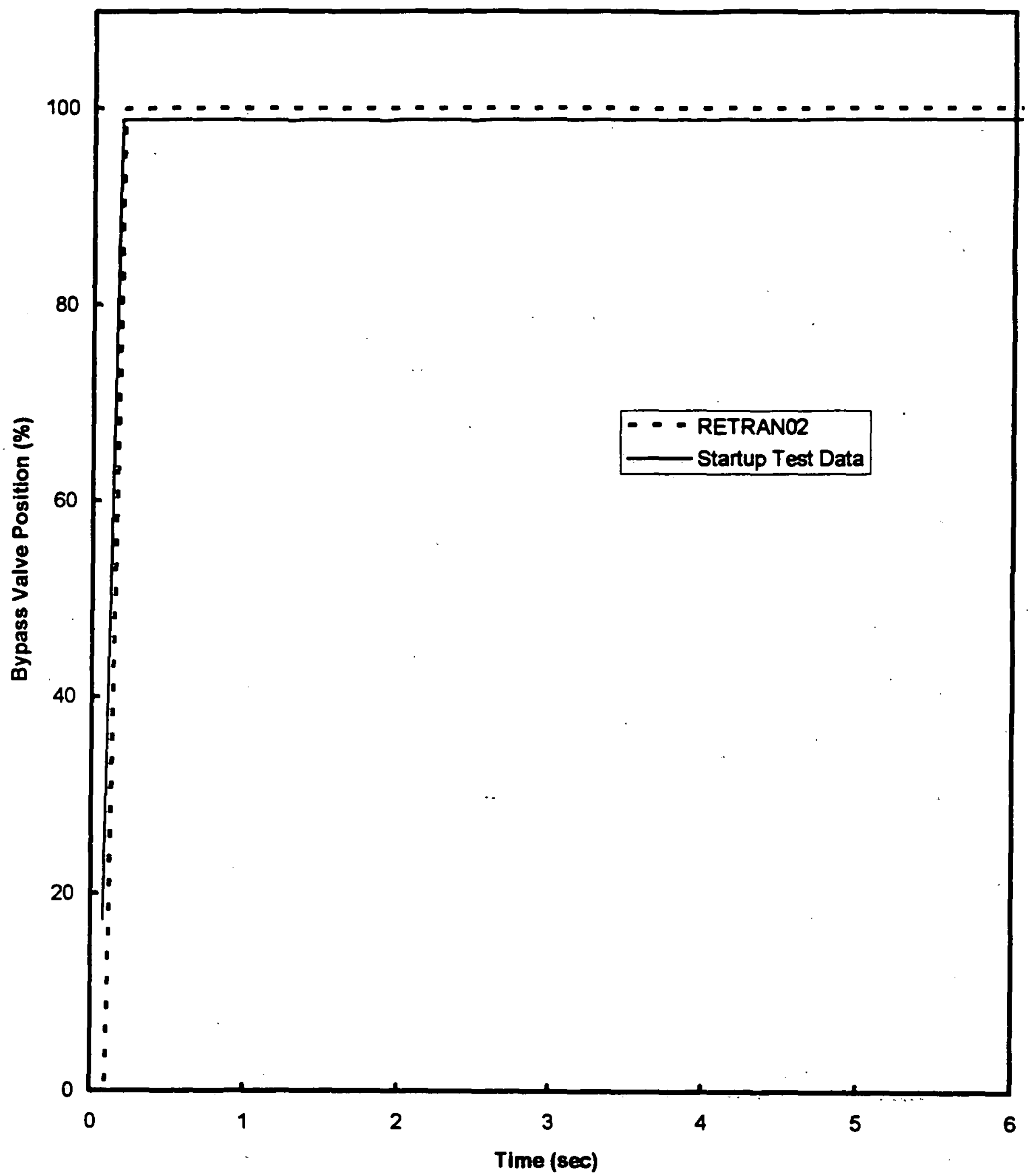

Figure 4.2-14: LaSalle BWR/5 LRWBP Bypass Valve Position

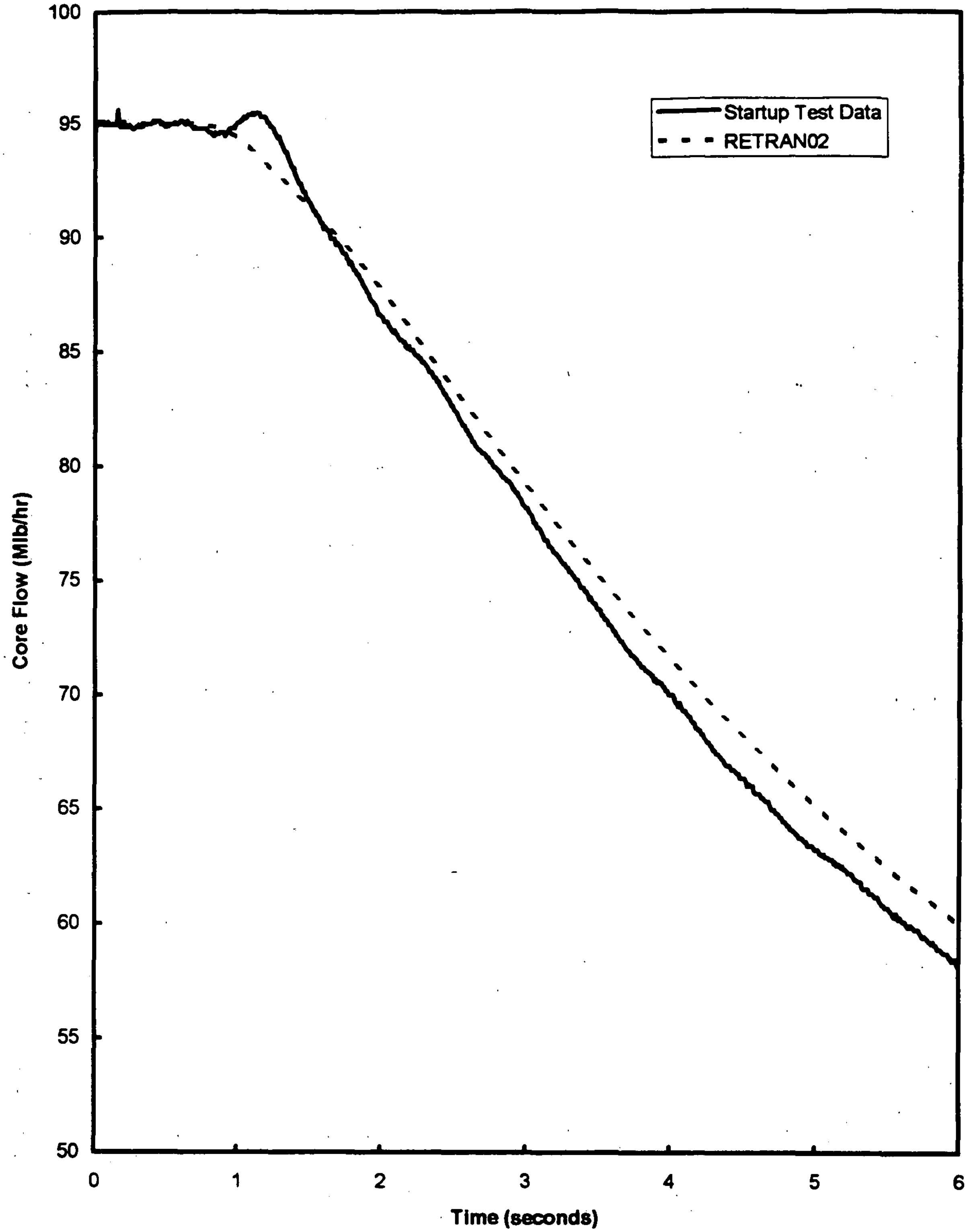

Figure 4.2-15 : LaSalle BWR/5 LRWBP Core Flow

4.2.5 MSIV Closure (MSIVC)

4.2.5.1 Test Description

The main steam isolation valve closure (full isolation) startup test was conducted at LaSalle Unit 1 on October 23, 1983.

The purpose of this startup test was to check the main steam line isolation valves (MSIV's) for proper operation at selected power and to determine isolation valve closure times. Also, the transient behavior resulting from the simultaneous full closure of all MSIV's is tested.

This test was initiated by removing two fuses in the containment isolation control circuits which caused a Group I isolation and full closure of the main steam line isolation valves.

4.2.5.2 RETRAN Modeling of Test

Some adjustments to the RETRAN model were made in order to benchmark this startup test. The steam flow and feed water flow from the startup test data as listed in Table 4.0-1 were not equal. RETRAN does not allow a mass flow imbalance for steady state initialization. Therefore, these values must be equal. The value used was 13.8 Mlb/hr. It was obtained from heat balance calculations. Steam flow was plotted in terms of change from initial value because the initial value was different for RETRAN and the test data. The general data tables containing the valve position versus time for the inboard and outboard MSIV's were changed to achieve a profile that matched the startup data. In addition, two trips were altered to facilitate the full closure of all MSIV's. Nominal relief valve trip setpoints were used in order to better match the startup test conditions. The recirculation pumps were tripped off consistent with data from Reference 14.

4.2.5.3 Results

<u>Plant Response</u>:

A reactor scram signal is generated upon 10% closure of the MSIV's. This rapidly reduces the steam produced. The scram due to the isolation acted faster than the increase in void reactivity caused by void collapse due to increased reactor pressure. Hence, no immediate increase in reactor power was observed. Eventually, the reactor pressure increases until the first relief valve setpoint is reached opening the relief valve. This pressurization together with the reactor scram causes the water level to fall off substantially. As the water level drops, recirculation pumps trip on a low reactor water level (Level 3 signal). The water level does not fall below the -40 inch mark. Therefore, the reactor core isolation cooling (RCIC) does not activate. During this transient, the core flow increases as the pressure increases. Then, it begins to level out as the pressure begins to drop. Finally, it decreases once the recirculation pump trip occurs.

Model Response:

The parameters of interest for this test are the core power, core flow, reactor pressure, and steam flow. The model should predict the core power, core flow and reactor pressure well since these are key variables for calculating critical power. The model should accurately predict the change in steam flow. These are the key parameters for benchmarking, the RETRAN thermal hydraulics during pressurization events, the reactor kinetics, and the MSIV model.

Figure 4.2-16 shows the comparison for reactor steam dome pressure. The RETRAN model exhibits an acceptable agreement. Close agreement is met at the peak where the pressure stabilizes. The RETRAN model predicts pressure conservatively. Figure 4.2-17 shows the comparison of percent reactor power. Acceptable agreement is shown. Figure 4.2-18 shows the comparison for steam flow change from initial value. The model predicts acceptable agreement. Figure 4.2-19 shows the comparison for core flow with acceptable agreement.

This test uses the cross section file generated for the LaSalle Unit 2 startup test initial conditions. This was considered acceptable since this test was conducted at about the same core power and exposure. This assumption is validated by the comparison presented in Figure 4.2-17.

In summary, The RETRAN model accurately simulated the MSIV closure transient. The modeling was shown to be accurate over the range presented. All pertinent RETRAN calculated variables performed as anticipated. Table 4.2-5 summarizes the ratings for the MSIVC benchmark.

Table 4.2-5: Ratings for the LaSalle BWR/5 MSIVC Benchmark

Parameter	Rating
Steam Dome Pressure	+
Core Power	+
Core Flow Rate	+
Steam Flow Rate	+

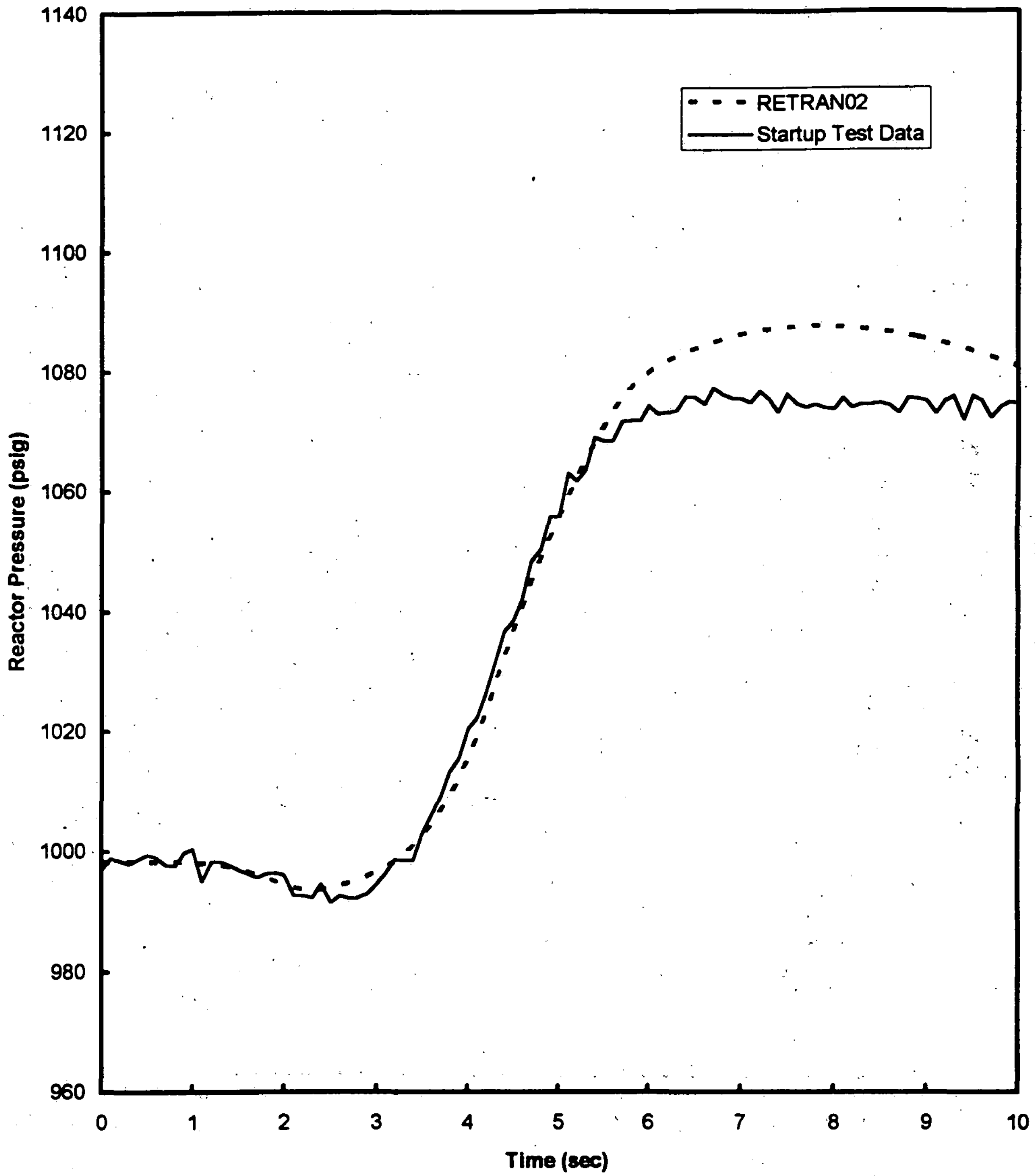

Figure 4.2-16: LaSalle BWR/5 MSIVC Reactor Pressure

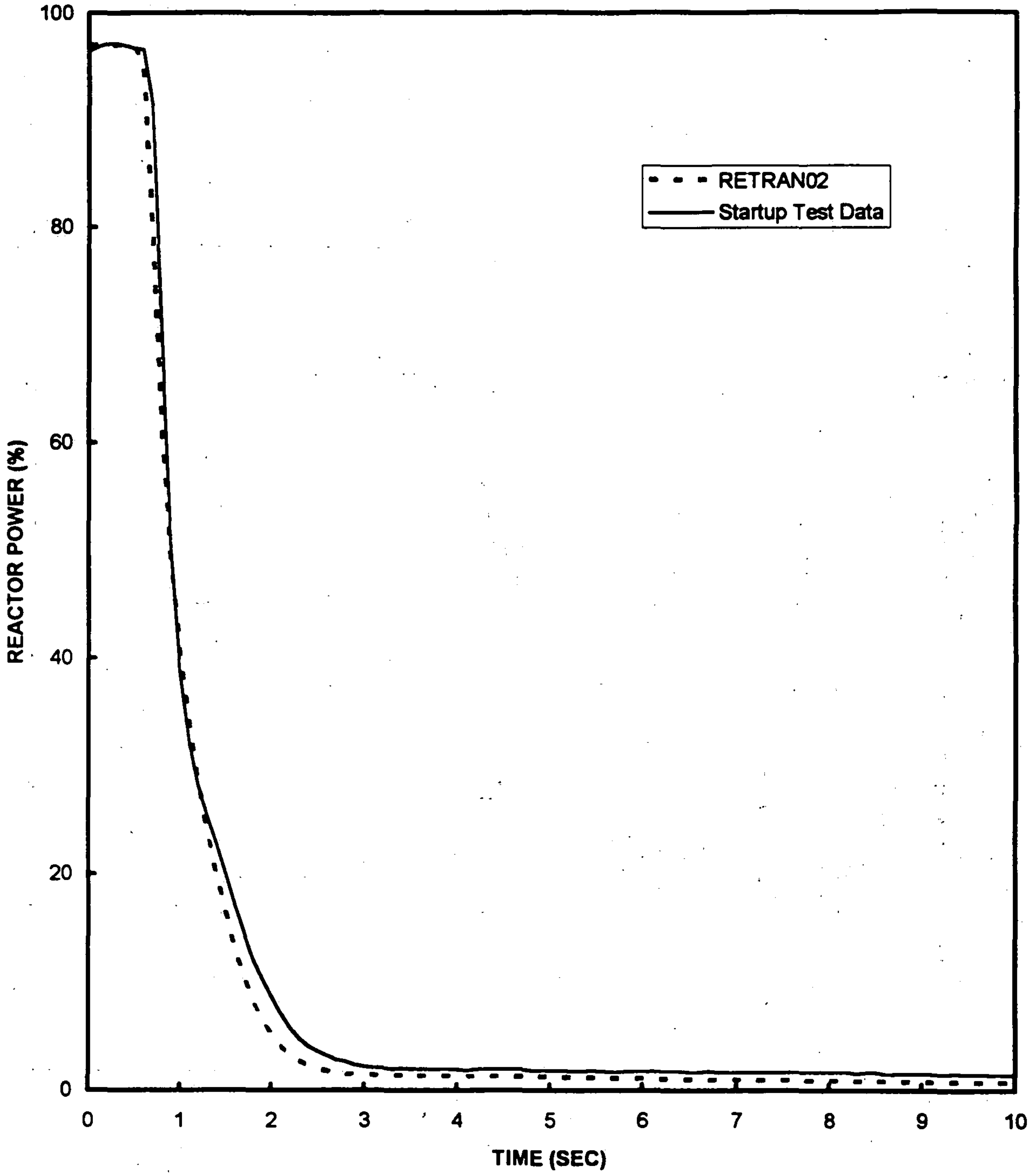

Figure 4.2-17: LaSalle BWR/5 MSIVC Reactor Power

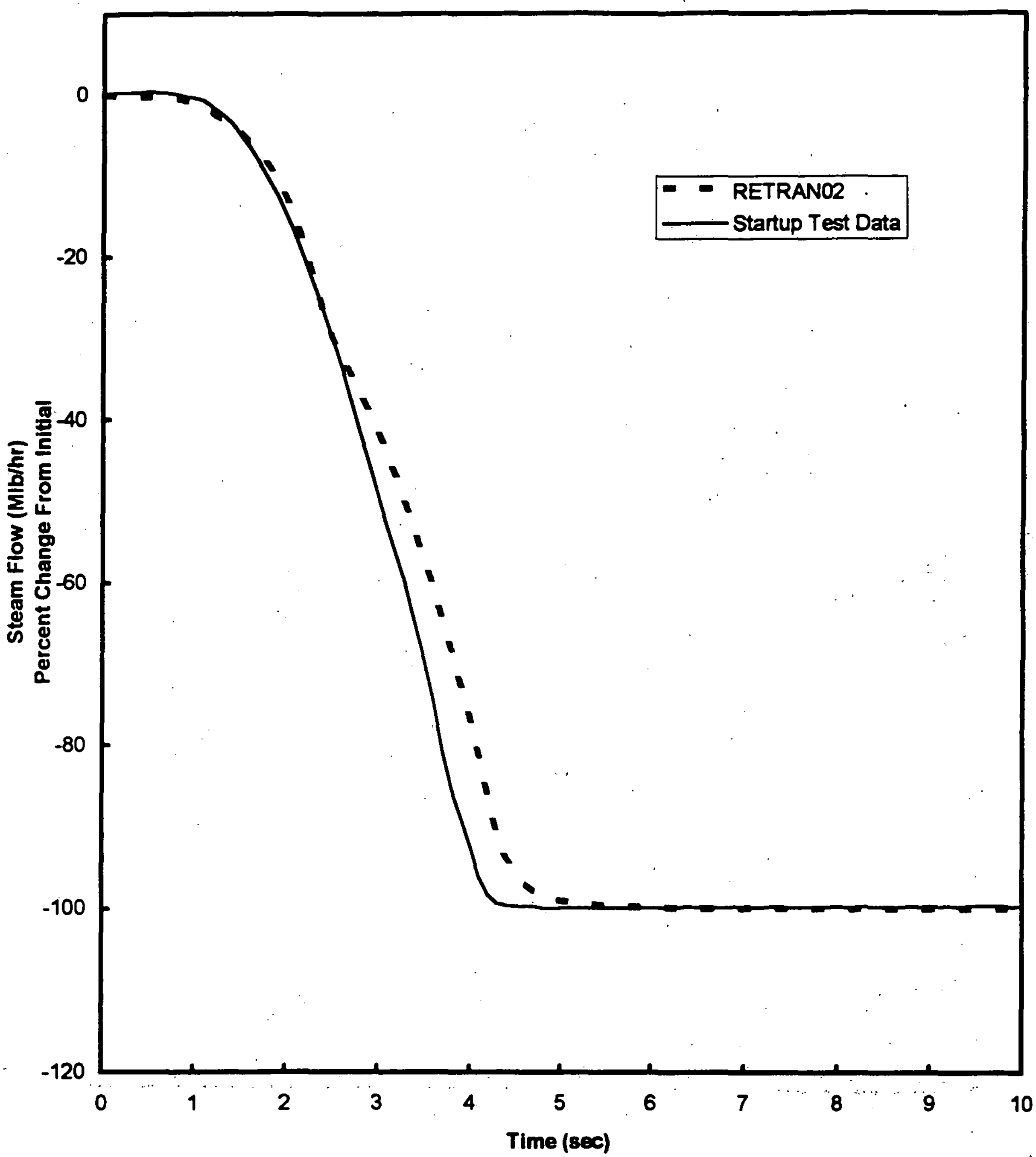

Figure 4.2-18: LaSalle BWR/5 MSIVC Steam Flow

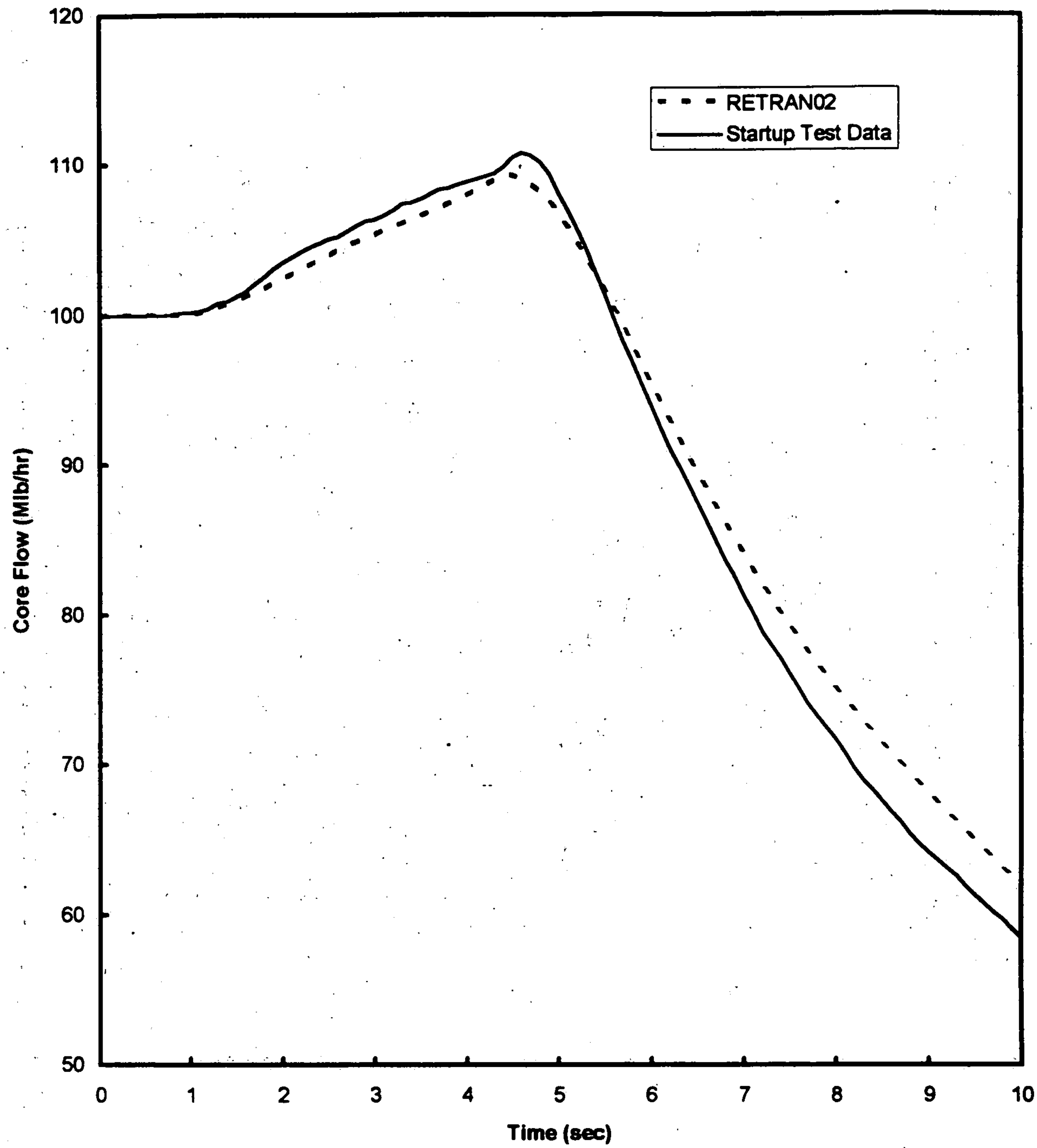

Figure 4.2-19: LaSalle BWR/5 MSIVC Core Flow

4.3 BWR/3 Quad-Cities and Dresden Startup Test Comparisons

4.3.1 Reactor Water Level Setpoint Change (RWLSC)

4.3.1.1 Test Description

The Reactor Water Level Setpoint Change startup test was conducted at Quad-Cities Unit 1 on June 28, 1972.

The purpose of the reactor water level setpoint change was to demonstrate that the reactor water level control system was adequate and stable in response to small changes.

The transient was initiated by stepping up the level controller setpoint by 6 inches. Tests were performed for the "A" and "B" feed water control valves in both single element and in 3-element control. This specific test was performed in 1-element control. Only one valve was allowed to be in automatic at a time with the other valve throttled back in manual at the operators' discretion.

4.3.1.2 RETRAN Modeling of Test

Some adjustments to the RETRAN model were made in order to benchmark this startup test. The feed water control system was changed to reflect single element control. The feed water control system blocks that control the reactor water level setpoint after the reactor trip were modified to initiate this water level setpoint change benchmark test. One control input block and one general data table were modified to allow the reactor water level setpoint to be changed after a delay of 2.0 seconds. This delay was selected due to uncertainty in the initiation of the recorded startup test data. Adjustment to the RETRAN model was made on the feed water control P-I gains. Reference 11 lists the final controller settings for Quad-Cities station, but does not specify the controller settings used for the specific plotted test data and confirms that the plotted data was not taken with the final settings. Because documentation for the startup tests did not specify ramp rate for the setpoint change, it was assumed that the level setpoint was changed over 2 seconds. The recirculation control system was assumed to be in manual control. Based on a reactor core heat balance, a value of 8.9 Mlb/hr was used for the steam and feed water flow.

4.3.1.3 Results

Plant Response:

When the reactor water level setpoint is increased, the feed water controller output immediately begins to rise causing feed water flow to increase. This initiates a mismatch in feed water and steam flow rates and causes the reactor water level to rise. Since the level is rising, the feed water flow peaks at a higher flow rate, and slowly decreases as water level reaches the new setpoint until feed water flow returns to essentially the same flow rate as initial. The pressure controller opens the turbine control valve slightly (less than 1%) as a result of a pressure increase of about 1 psi.

Model Response:

The parameters of interest for this test are the reactor narrow range water level and the feed water flow. The narrow range level and feed water flow are the only system parameters that experience any significant change and these are the key parameters for benchmarking the feed water level control system. Core power, core flow, reactor pressure and steam flow show very small changes.

Figure 4.3-1 shows the comparison of change in feed water flow rate. The results show acceptable agreement. Figure 4.3-2 shows the comparison of the water level response to the setpoint change. The results also show acceptable agreement. Core power, steam flow rate, and reactor pressure remain relatively constant as expected over the course of the transient.

This transient was mild and had magnitude changes which were close to the magnitude changes for a (+) rating. Although the screening criterion was used to rate the parameter magnitude changes, engineering judgment was used to determine the validity of the benchmark analysis. RWLSC requires that level will change by six inches for the test. The analysis results were examined to assure that the model achieved the prescribed change. The other parameter of interest for this test was also examined for reasonableness with respect to the magnitude changes prescribed by the test.

In summary, the RETRAN model accurately simulated the feed water flow and the reactor water level response. The feed water level control system was shown to be accurate over the range presented. All pertinent RETRAN calculated variables behave as anticipated. Table 4.3-1 summarizes the ratings for the RWLSC benchmark.

Table 4.3-1 Ratings for the Quad-Cities BWR/3 RWLSC Benchmark

Parameter	Rating
Reactor Water Level (Narrow Range)	+
Feed Water Flow	+

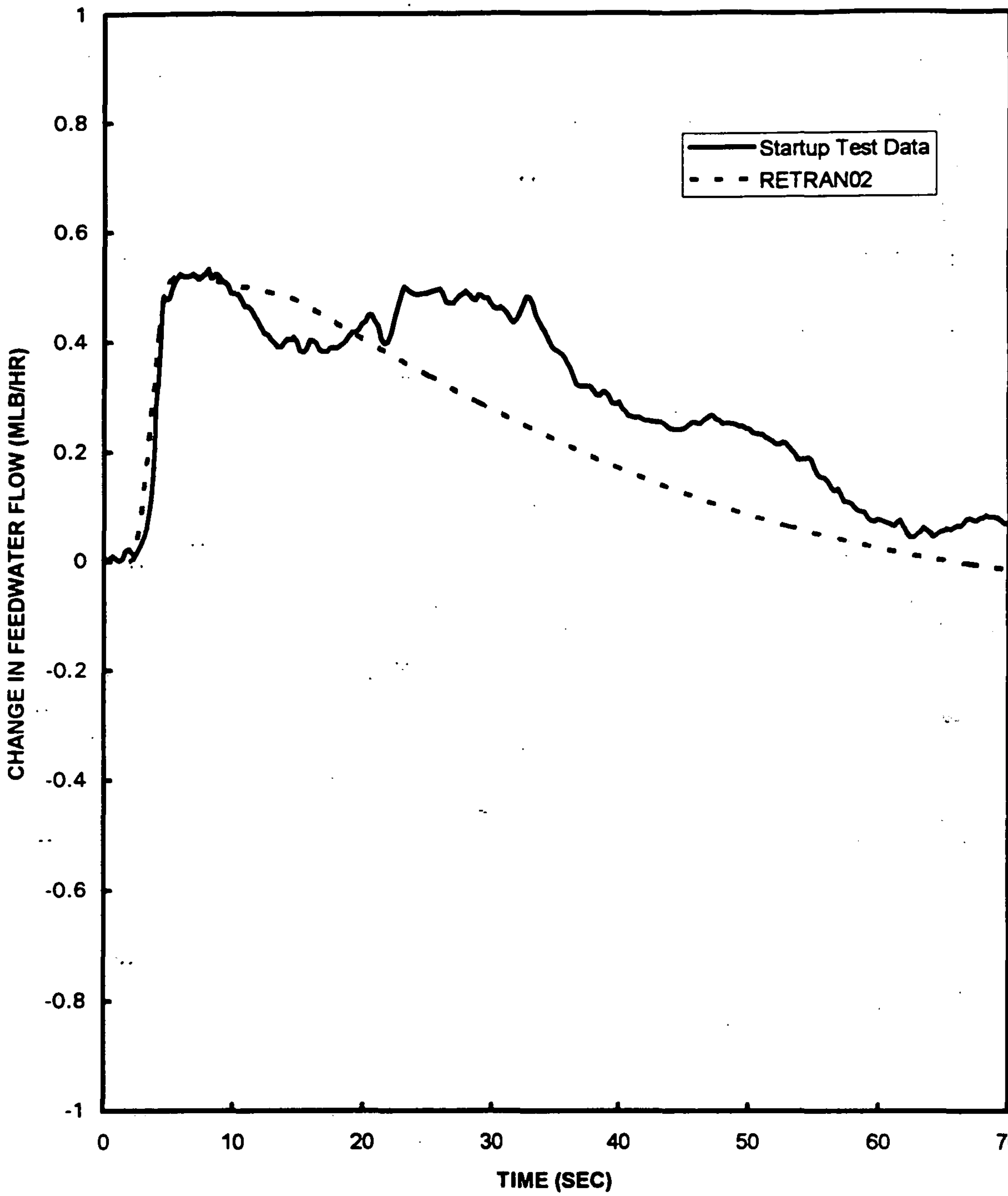

Figure 4.3-1: Quad-Cities BWR/3 RWLSC Feed Water Flow

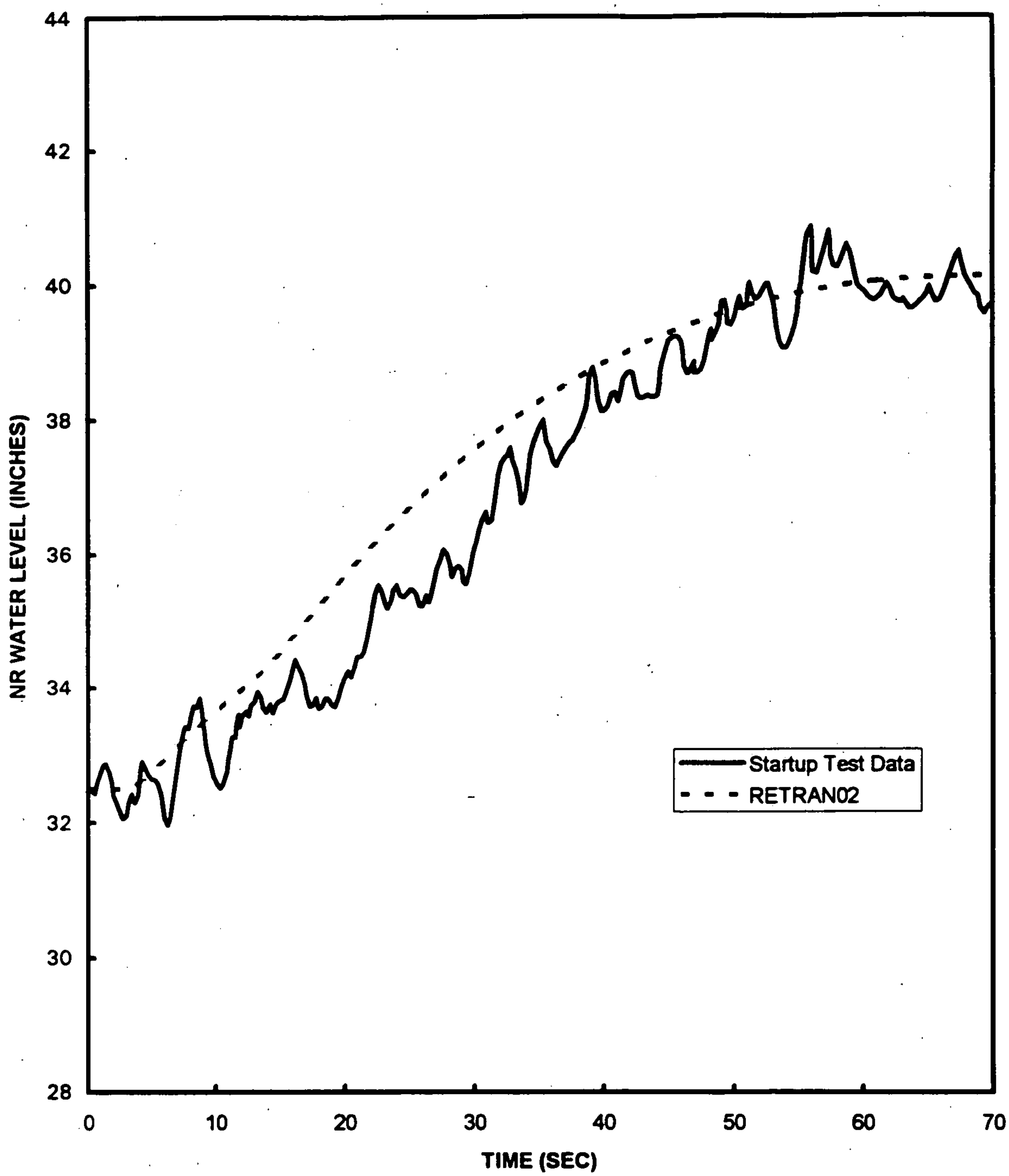

Figure 4.3-2: Quad-Cities BWR/3 RWLSC Water Level

4.3.2 Pressure Regulator Setpoint Change (PRSC)

4.3.2.1 Test Description

The pressure regulator setpoint change startup test was conducted at Quad-Cities Unit 1 on April 21, 1972.

The purpose of the pressure regulator setpoint change startup test was to determine the response of the reactor and pressure regulator to disturbances and to demonstrate the stability of the reactor pressure control system. It was also to test Bypass Valve actuation when demand exceeds the load limiter setpoint.

The pressure regulator setpoint change startup test was initiated by reducing the turbine inlet pressure setpoint in a stepwise fashion. Also the load limiter was set down to ensure that the bypass valves would open to make the adjustment in pressure. The data shown was for a -10 psi setpoint change.

4.3.2.2 RETRAN Modeling of Test

Some adjustments to the RETRAN model were made in order to benchmark this startup test. To model a setpoint change, the pressure regulator setpoint versus time table in RETRAN had to be changed. The pressure setpoint was changed by 10 psi during the first 0.1 seconds. An adjustment to the RETRAN model was made on the pressure control system lead-lag compensator time constants. Reference 11 lists the final controller settings for Quad-Cities station, but does not specify the controller settings used for the specific plotted test data and confirms that the plotted data was not taken with the final settings. The steam flow and feed water flow from the startup test data as listed in Table 4.0-2 were not equal. Because RETRAN does not allow a mass flow imbalance for steady state initialization, these values must be set equal. Based on a reactor core heat balance, a value of 1.964 Mlb/hr was used for the steam and feed water flow. The turbine load limiter setpoint was lowered, consistent with the startup test, to ensure that the bypass valves open. The recirculation control system was assumed to be in the manual mode which was confirmed by Reference 11 data.

4.3.2.3 Results

<u>Plant Response:</u>

The operator manually reduces the turbine inlet pressure setpoint from 970 to 960 psig. Since the load limiter has been reduced to a setting just above the steam demand, the bypass valves open to reduce the turbine inlet pressure. Pressure decreases, closing the bypass valves. Then pressure increases slightly and stabilizes due to a small throttling of the turbine control valve. The measured main steam flow initially increases because the main steam flow transmitter is located upstream of the bypass valves. After the bypass valves close, the steam flow decreases slightly. As a result of the pressure decrease, the core voiding increases which lowers reactor power and simultaneously raises water level.

Model Response:

The parameters of interest for this test are the reactor pressure, core power, steam flow and the bypass valve position. The model should stabilize at the new pressure setpoint and should predict the bypass valve opening and the small variations in core power and steam flow. These are the key parameters for benchmarking the pressure control system and the reactor kinetics response to small pressure changes.

Figure 4.3-3, Figure 4.3-4 and Figure 4.3-5 show the comparison of the change in steam flow, reactor core power and reactor dome pressure. The RETRAN results follow the data and show acceptable agreement. Figure 4.3-6 shows the comparison of bypass valve position. The RETRAN results follow the data and show acceptable agreement during the valve opening part of the transient. The results show generally acceptable agreement with the data during the valve closing part of the transient.

This transient was mild and had certain magnitude changes which were close to the magnitude changes for a (+) rating. Although the screening criterion was used to rate the parameter magnitude changes, engineering judgment was used to determine the validity of the benchmark analysis. PRSC requires that pressure will change by ten psi for the test. The analysis results were examined to assure that the model achieved the prescribed change. The other parameters of interest for this test were also examined for reasonableness with respect to the magnitude changes prescribed by the test.

In summary, the RETRAN model accurately simulated the pressure regulator setpoint change. The pressure control system functions properly for a change in demand over the range presented. The bypass valve performance is modeled correctly. All pertinent RETRAN calculated variables behave as anticipated. Table 4.3-2 summarizes the ratings for the PRSC benchmark.

Table 4.3-2 Ratings for the Quad-Cities BWR/3 PRSC Benchmark

Parameter	Rating
Reactor Pressure	+
Core Power	+
Steam Flow	+
Bypass Valve Position	+

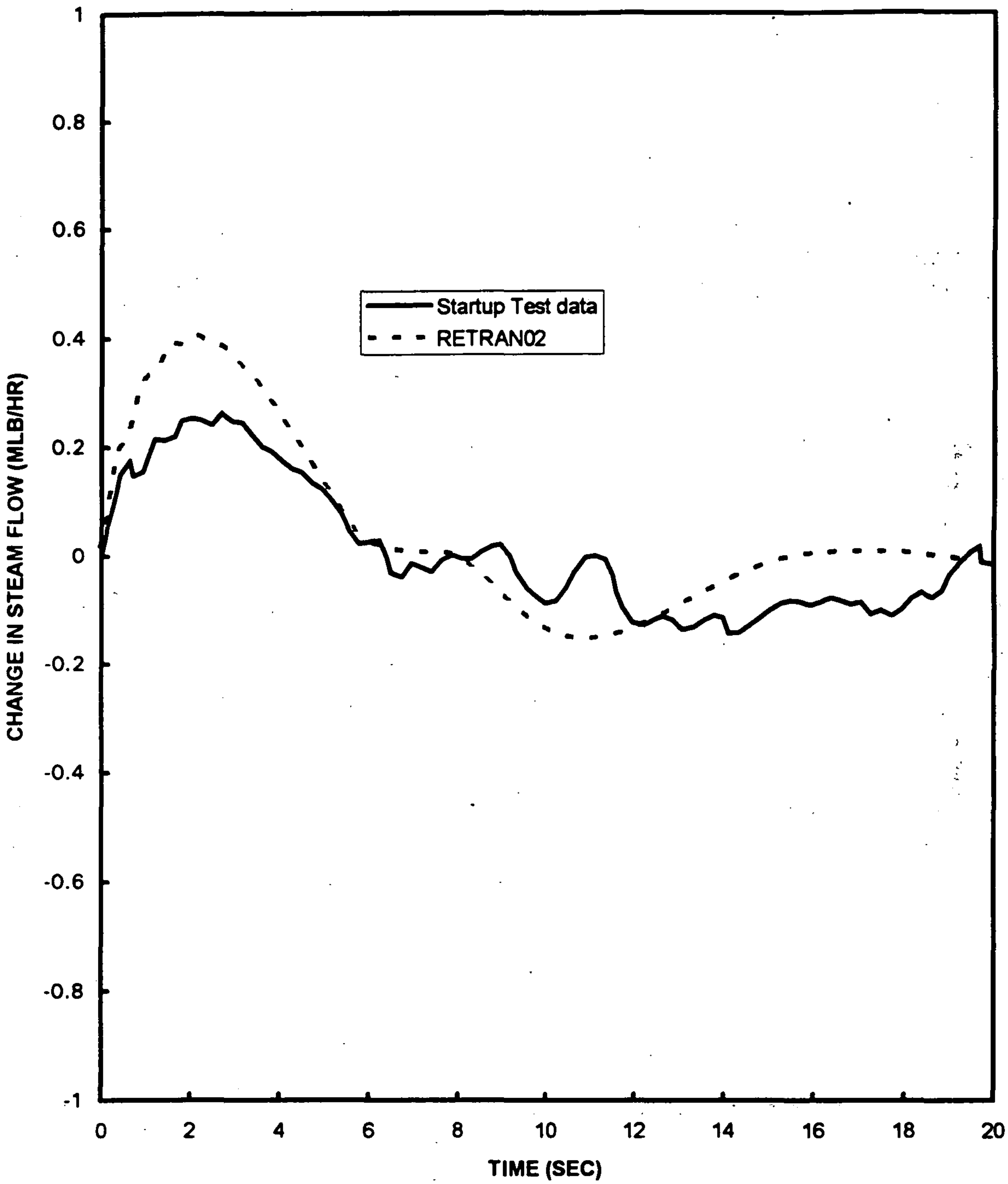

Figure 4.3-3: Quad-Cities BWR/3 PRSC Steam Flow

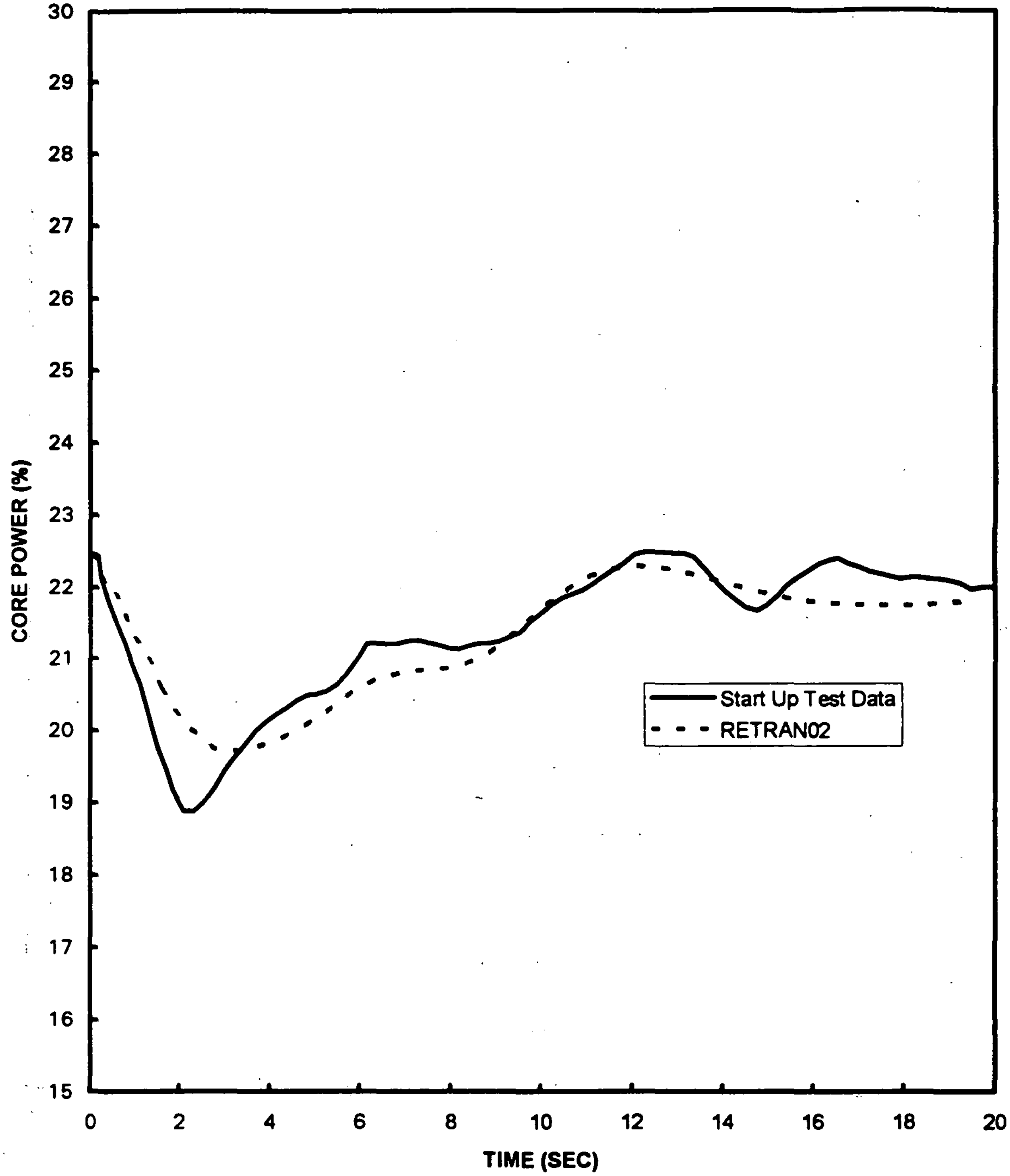

Figure 4.3-4: Quad-Cities BWR/3 PRSC Core Power

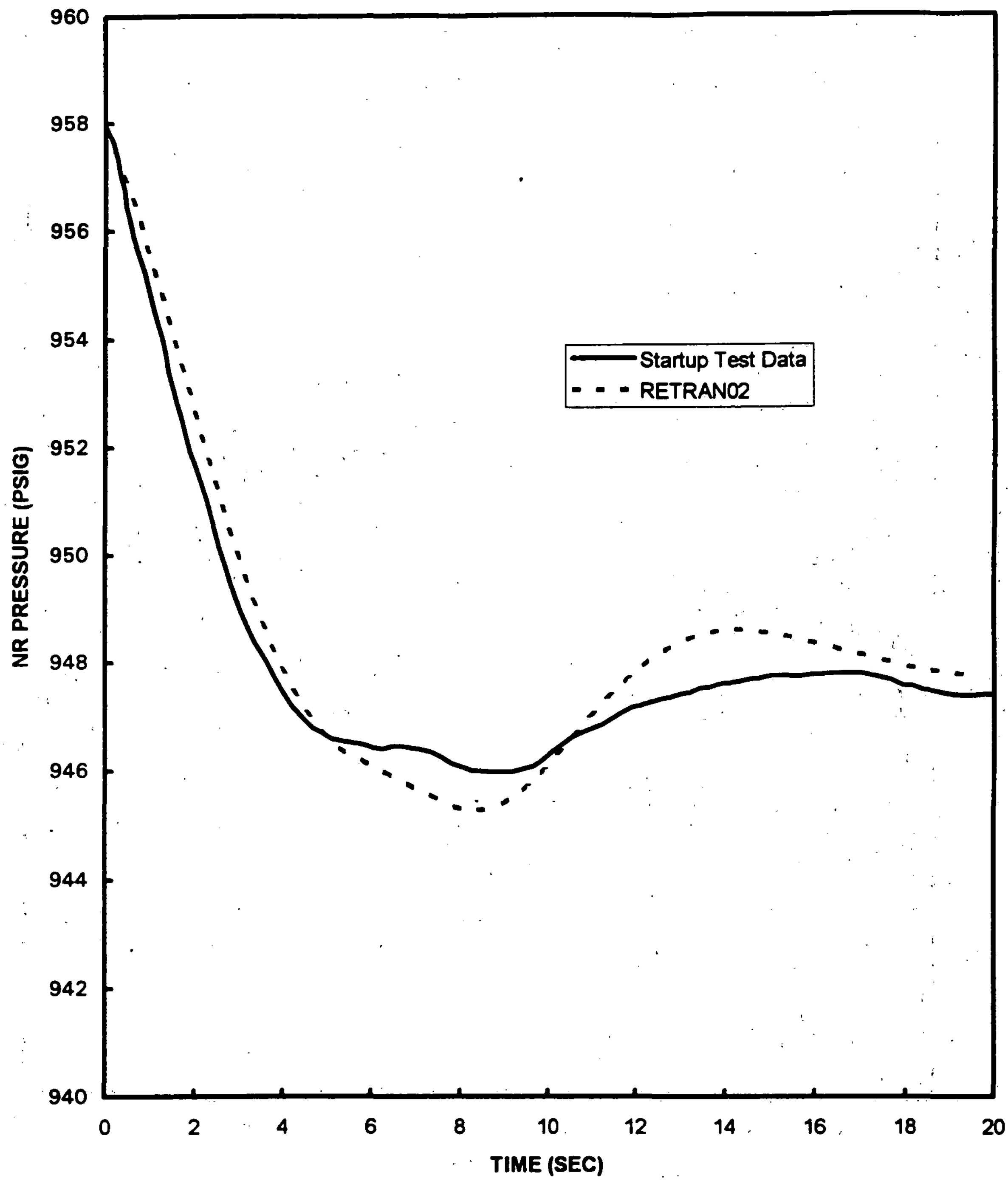

Figure 4.3-5: Quad-Cities BWR/3 PRSC Dome Pressure

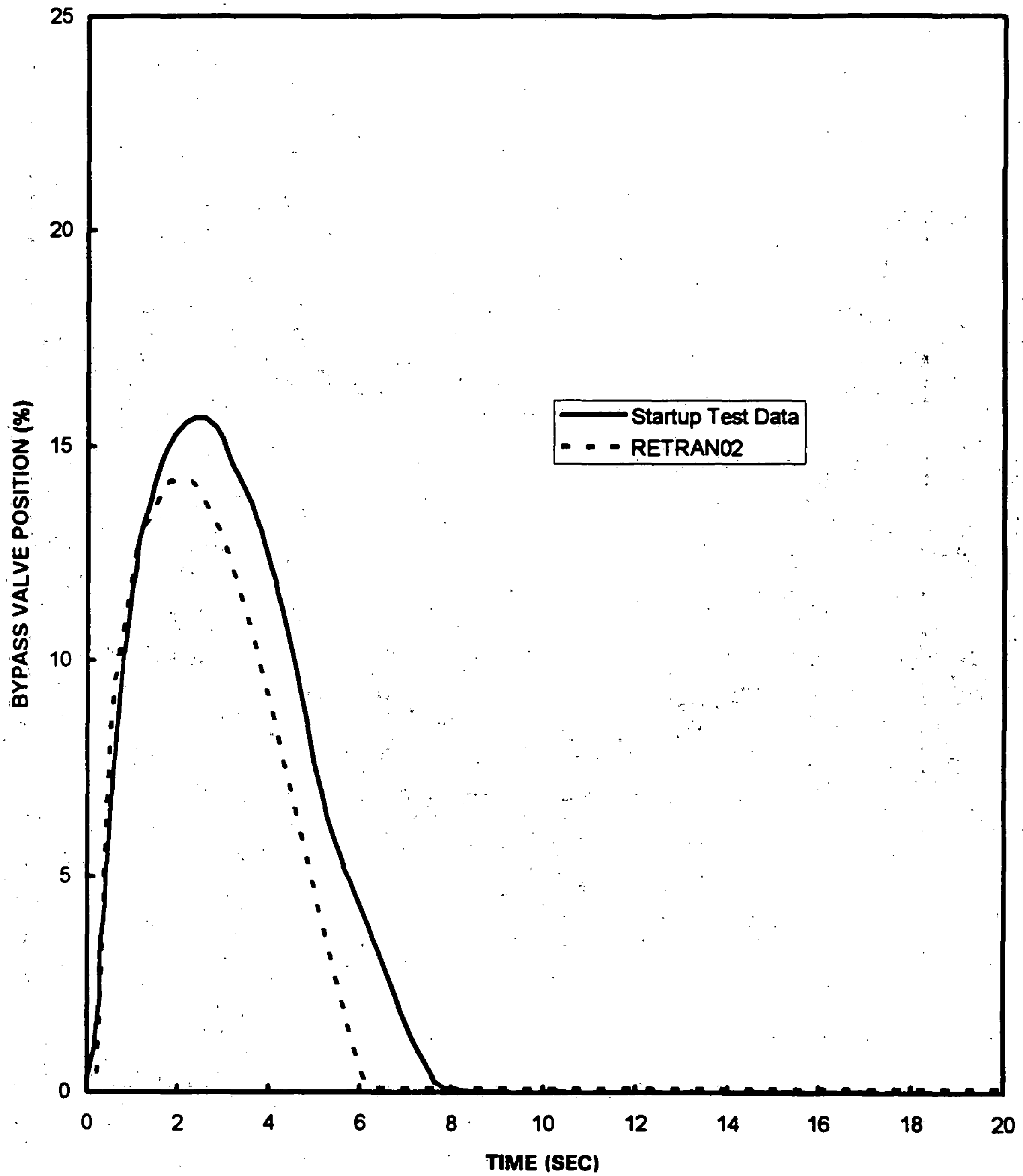

Figure 4.3-6: Quad-Cities BWR/3 PRSC Bypass Valve Position

4.3.3 Dual Recirculation Pump Trip (DRPT)

4.3.3.1 Test Description

The dual recirculation pump trip startup test was conducted at Dresden Unit 3 on October 13, 1971.

The purpose of performing this startup test was to evaluate the thermal-hydraulic transients following trips of one or both recirculation pumps.

The startup test transient was initiated by tripping both M-G set drive motor breakers. After tripping the breakers, the recirculation pumps slowly reduce speed and flow resulting in a core flow reduction.

4.3.3.2 RETRAN Modeling of Test

Minor adjustments to the RETRAN model were made in order to benchmark this startup test. The recirculation control system was set to manual control, as confirmed in Reference 12. Two trips were modified to allow the M-G sets to trip at the first RETRAN time step. Based on a reactor core heat balance, a value of 9 Mlb/hr was used for the steam and feed water flow.

4.3.3.3 Results

<u>Plant Response:</u>

The recirculation pump speed and recirculation drive flow decrease. The rate of speed decrease is a function of the effective inertia in the M-G set and pump. Since the jet pump drive flow decreases, core flow decreases. This causes increased core voiding and a decrease in core power. With less steam production, the reactor pressure decreases. The pressure control system tries to compensate by closing the turbine control valves. Increased core voiding and decreasing reactor pressure causes the water level to rise. The feed water controller reduces flow until the level stops increasing. Main steam flow in this event decreases steadily.

<u>Model Response:</u>

The parameters of interest for this test are the core flow, core power, reactor pressure and steam flow. The model should predict the large decrease in core flow and core power. The pressure control system should also respond to the changes in steam flow and stabilize reactor pressure at the new conditions. These are the key parameters for benchmarking the RETRAN thermal hydraulics during a rapid core flow change, the recirculation pump model, the recirculation control system, the reactor kinetics and the pressure control system.

Figure 4.3-7 shows the comparison for core flow. The RETRAN calculation follows the data. The results show acceptable agreement. Figure 4.3-8 and Figure 4.3-10 show the RETRAN core power and main steam flow comparisons to the test data respectively. The RETRAN results in these figures follow the data and shows acceptable agreement. Figure 4.3-9 shows the results for dome pressure. The RETRAN calculation has the same trend and shows acceptable agreement.

In summary, the results of the RETRAN model for the dual recirculation pump trip demonstrate that the recirculation system components are properly modeled and can adequately predict the transient behavior for core flow, power, steam flow and pressure in response to a dual recirculation pump trip. Table 4.3-3 summarizes the ratings for the DRPT benchmark.

Table 4.3-3 Ratings for the Dresden BWR/3 DRPT Benchmark

Parameter	Rating
Core Flow	+
Core Power	+
Reactor Pressure	+
Steam Flow	+

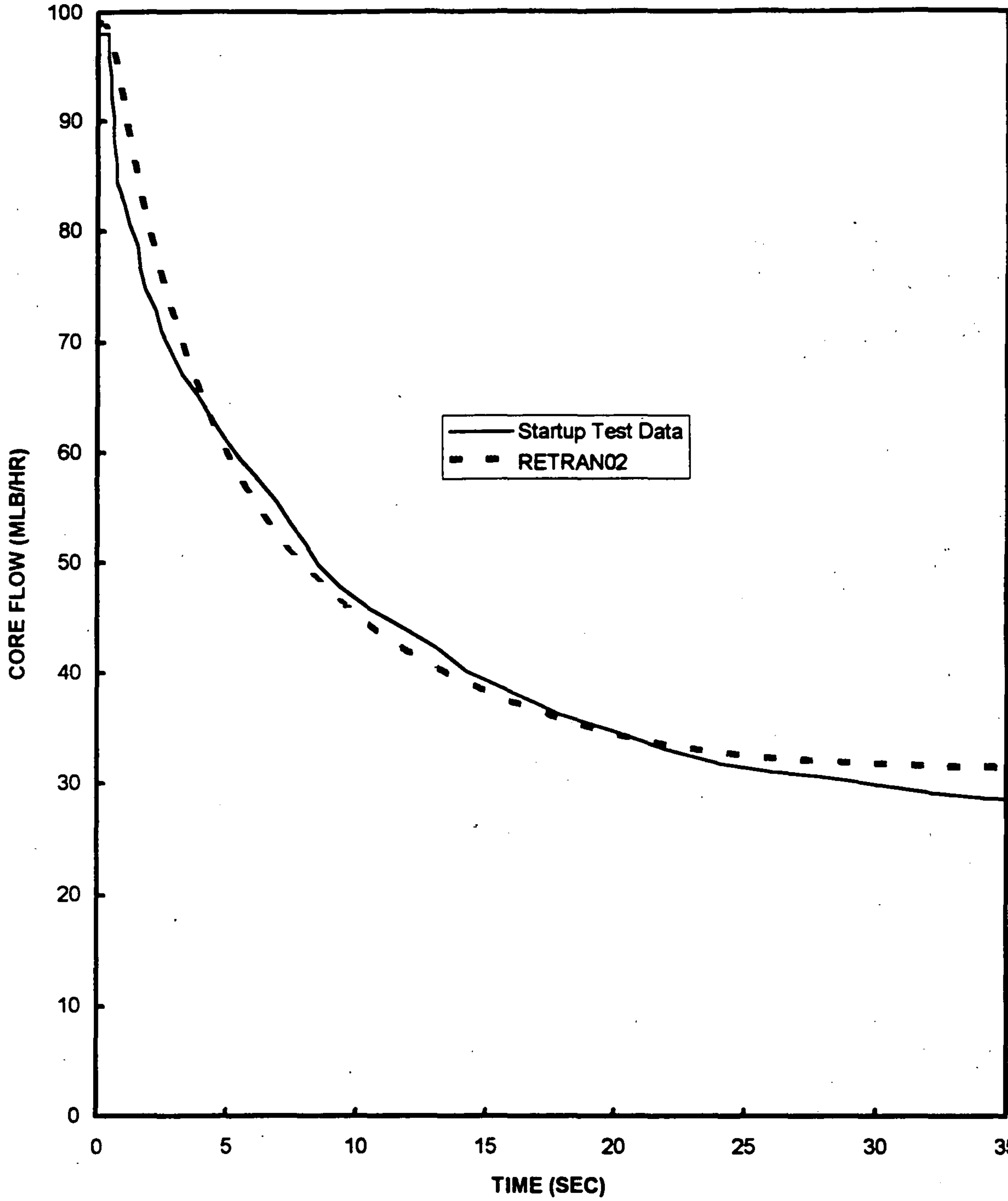

Figure 4.3-7: Dresden BWR/3 DRPT Core Flow

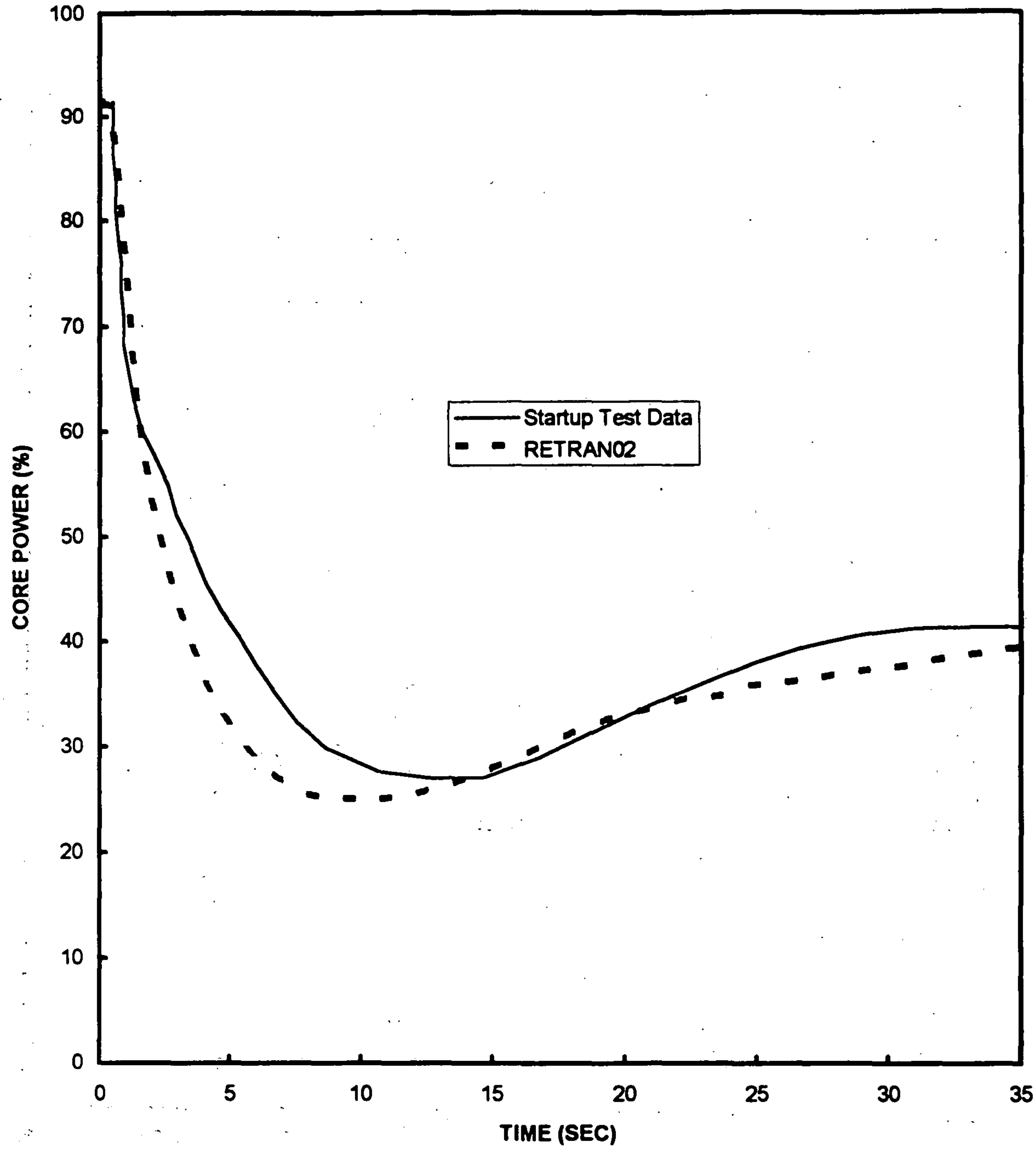

Figure 4.3-8: Dresden BWR/3 DRPT Core Power

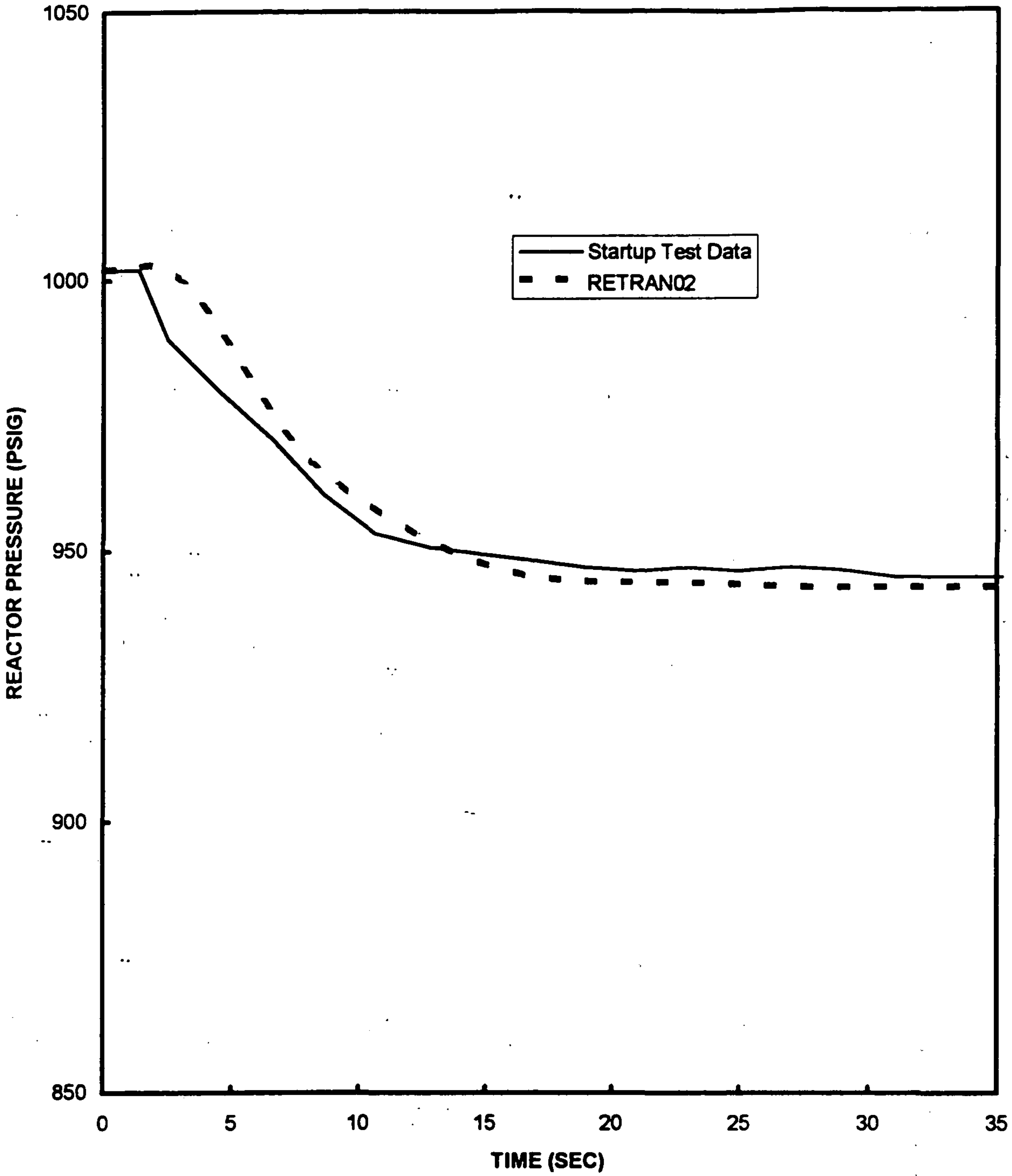

Figure 4.3-9: Dresden BWR/3 DRPT Dome Pressure

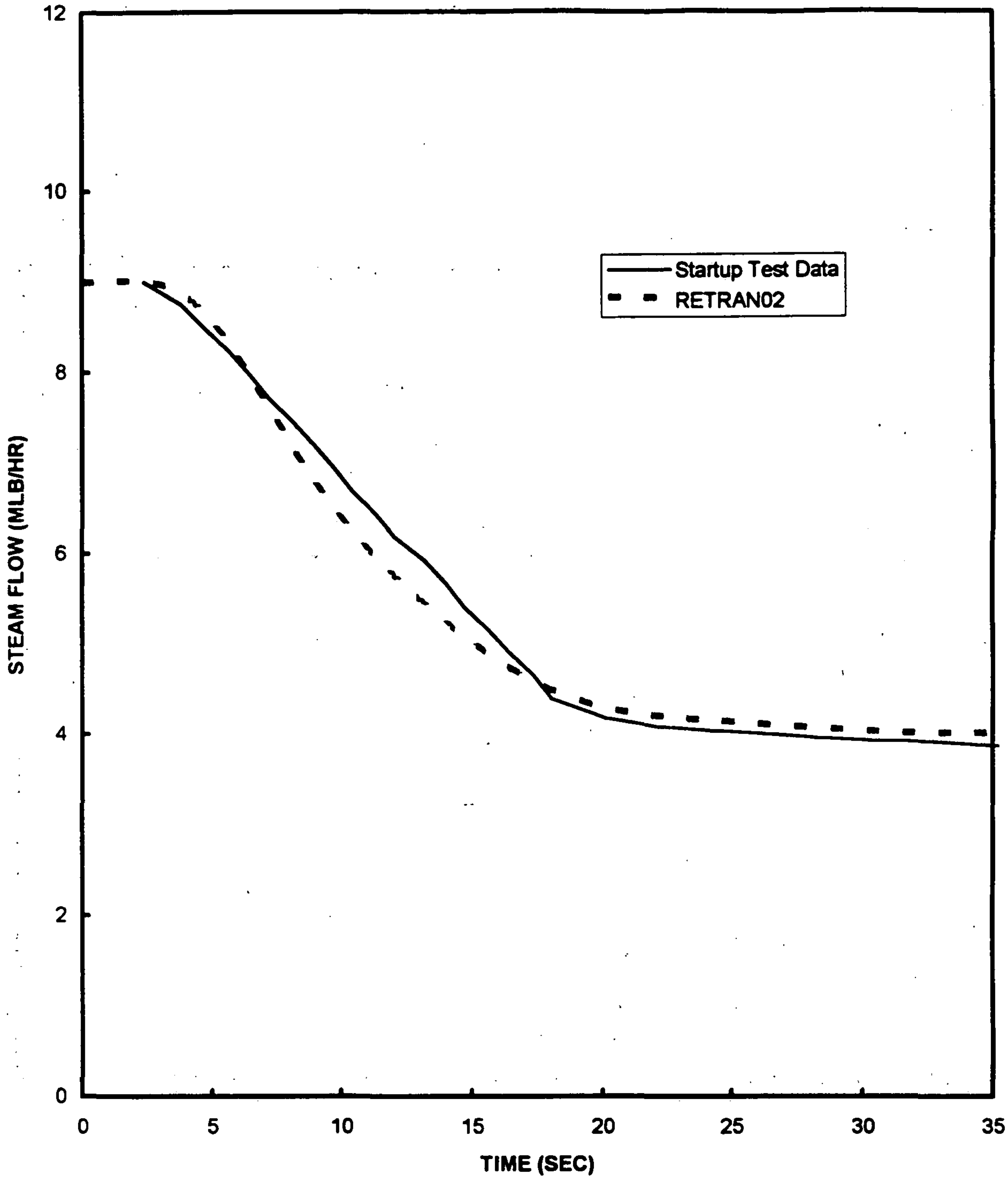

Figure 4.3-10: Dresden BWR/3 DRPT Vessel Steam Flow

4.3.4 Turbine Trip With Bypass(TTWBP)

4.3.4.1 Test Description

The Turbine Trip startup test was conducted at Quad-Cities Unit 2 on July 3, 1972.

The purpose of this startup test was to demonstrate the response of the reactor and its control systems to a turbine trip.

4.3.4.2 RETRAN Modeling of Test

Some adjustments to the RETRAN model were made in order to benchmark this startup test. One trip was modified to obtain the turbine stop valve closure at the first time step. The turbine stop valve closure time was set to 225 milliseconds. The bypass valve position as a function of time was not available in the startup test data. Precise measurements of the bypass valve stroke time were made during cycle 13 at Quad-Cities. This recorded time was the basis for adding a 240 millisecond delay to the bypass valve actuation logic in the pressure control system. Stroke time for the bypass was not altered but calculated from the pressure control system. Based on a reactor core heat balance, a value of 6.3 Mlb/hr was used for the steam and feed water flow.

4.3.4.3 Results

Plant Response:

This transient was initiated by manually tripping the turbine from the control room panel. As the steam flow stops, pressure builds in the steam line. For this case, the main steam bypass system is operational. Neither relief valve setpoints nor safety valve setpoints were reached. The bypass valves open promptly and limit the pressure excursion. The build up in pressure creates a pressure wave that travels back to the core. Increased pressure in the core collapses voids causing the core reactivity to increase rapidly. It also causes a large drop in the reactor water level. The pressure wave also impacts the core flow. For this case a rapid scram on turbine stop valve 10% closure prevents a power spike. Feed water flow compensates for the reduction in indicated water level. Since the recirculation system is in manual control, pump speed does not change. Measured core flow increases slowly because of the pressurization dynamics.

Model Response:

The parameters of interest for this test are the core power (neutron flux), core flow, reactor pressure, steam flow and bypass valve position. The model should predict the core power, core flow and reactor pressure well since these are key variables for calculating critical power. The model should predict the bypass opening and the change in steam flow. These are the key parameters for benchmarking, the RETRAN thermal hydraulics during pressurization events, the reactor kinetics, the pressure control system and the bypass valve model.

Figure 4.3-11 shows the comparison of reactor power. The measured data is clearly in error as the power was measured to level off around 10% after the reactor scram. The RETRAN prediction is not inaccurate or unacceptable. The results show that the RETRAN prediction has acceptable agreement. Figure 4.3-12 shows the comparison of narrow range dome pressure. The results show acceptable agreement. Figure 4.3-13 shows the comparison of core flow. Plant data shows an initial dip in core flow which is not predicted by the model. This dip in the plant data was small, it was within the noise of the original data and is not believed to be physical. The core flow results show acceptable agreement.

Figure 4.3-14 shows the comparison of vessel steam flow. The results show acceptable agreement since the RETRAN prediction stabilizes close to the plant data beyond 2.0 seconds. RETRAN does not predict the measured initial rise in the main steam flow. But this rise in flow is not believed to reflect the physical process and represents a temporary error in the flow measurement. A comparison to the LaSalle LRWBP steam flow in Figure 4.2-13 demonstrates this.

Figure 4.3-15 shows the bypass valve response. The results show acceptable agreement. Note that the Startup Test data was incomplete because the channel recording the main steam bypass valves total position was miscalibrated. This caused the trace to go off the multichannel recorder's paper. However, the results for the first 0.45 seconds were shown. The accurate prediction of pressure from Figure 4.3-12 also confirms acceptable main steam bypass position response.

In summary, the RETRAN model accurately simulated the plant response for this transient. All pertinent RETRAN calculated variables behave as anticipated. Table 4.3-4 summarizes the ratings for the TTWBP benchmarking. The bypass valve position was not rated since the measured data was incomplete.

Table 4.3-4 Ratings for the Quad-Cities BWR/3 TTWBP Benchmark

Parameter	Rating
Core Power	+
Core Flow	+
Reactor Pressure	+
Main Steam Flow	+
Main Steam Bypass Valve Position	+

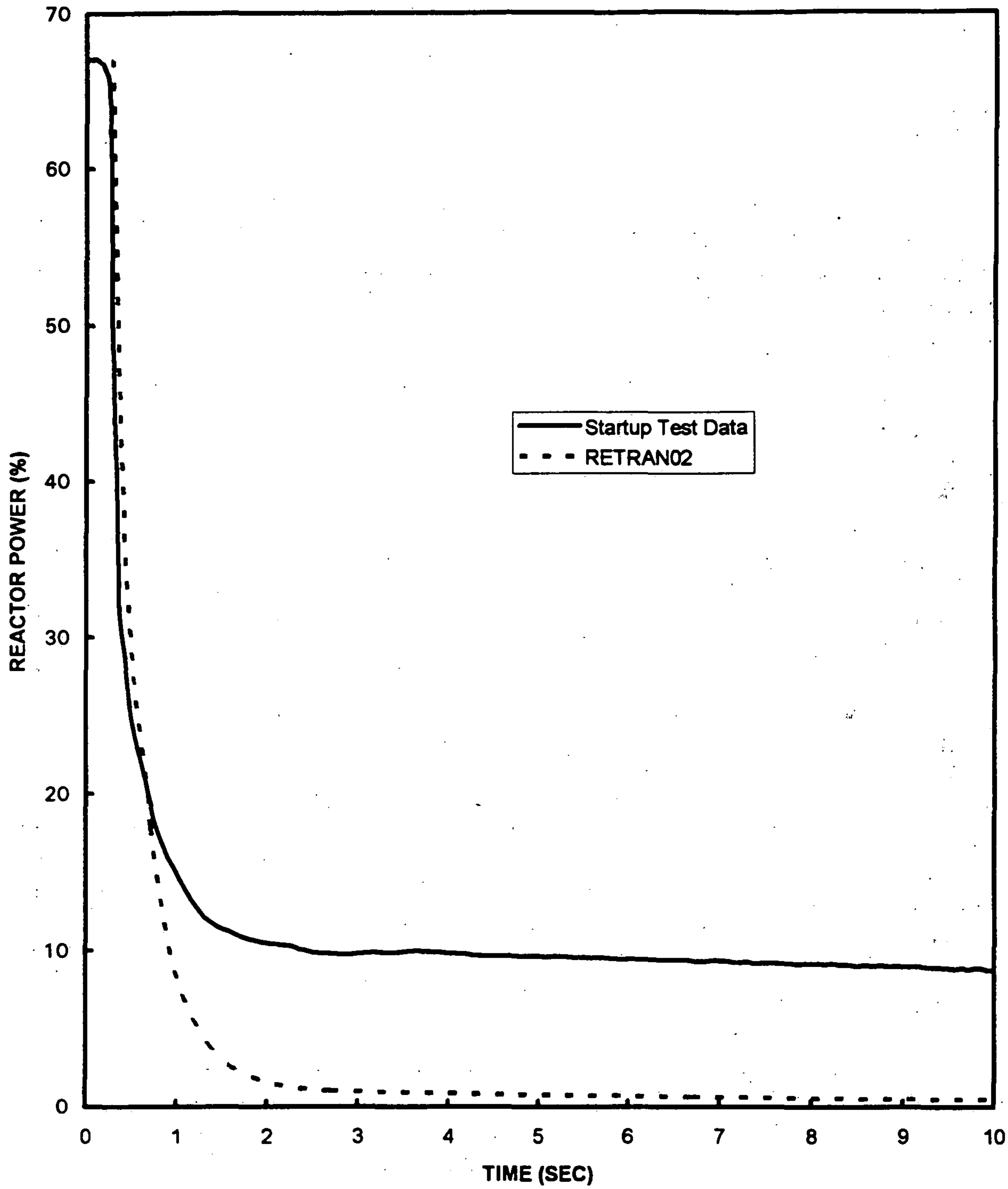

Figure 4.3-11: Quad-Cities BWR/3 TTWBP Core Power

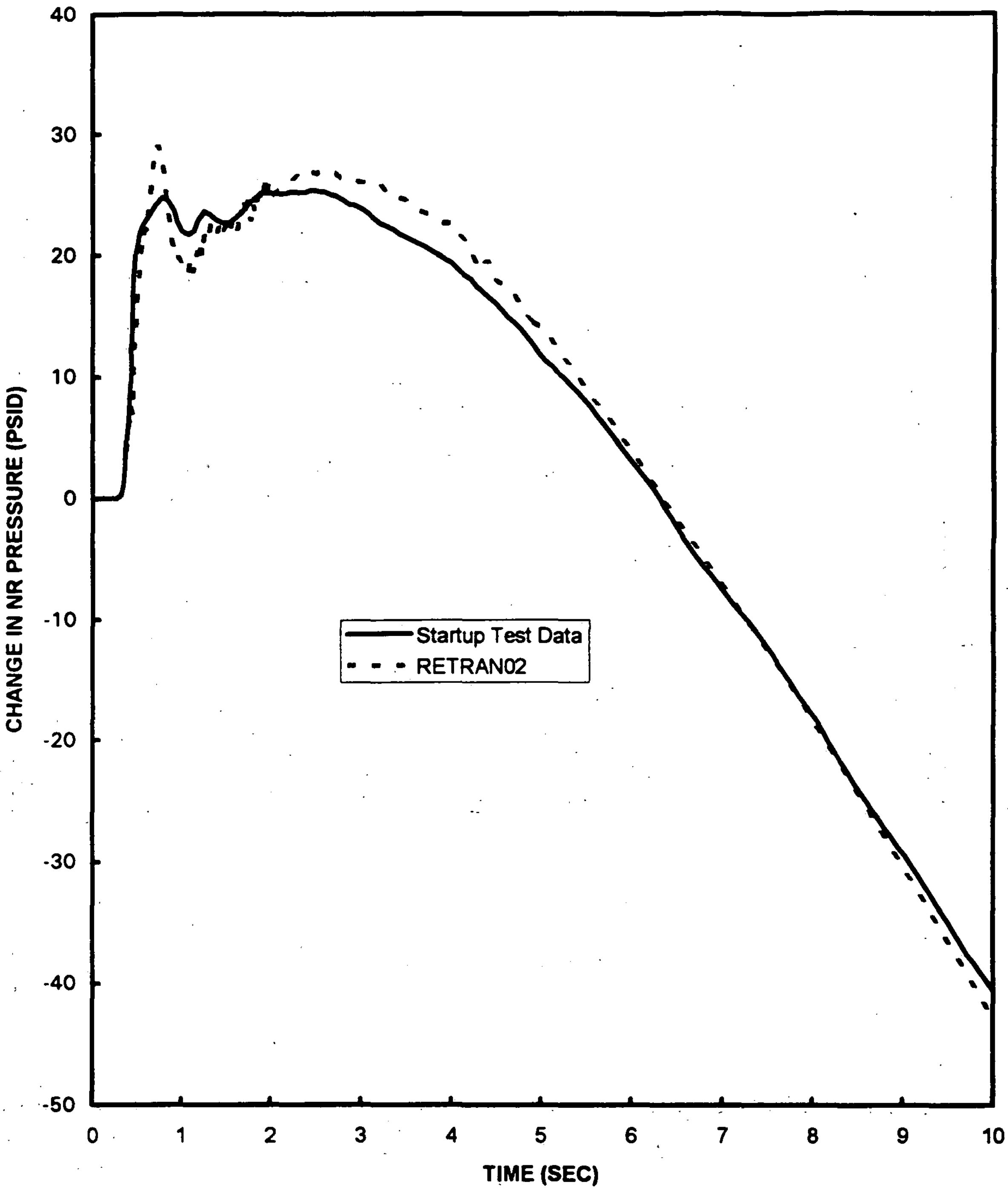

Figure 4.3-12: Quad-Cities BWR/3 TTWBP Narrow Range Dome Pressure

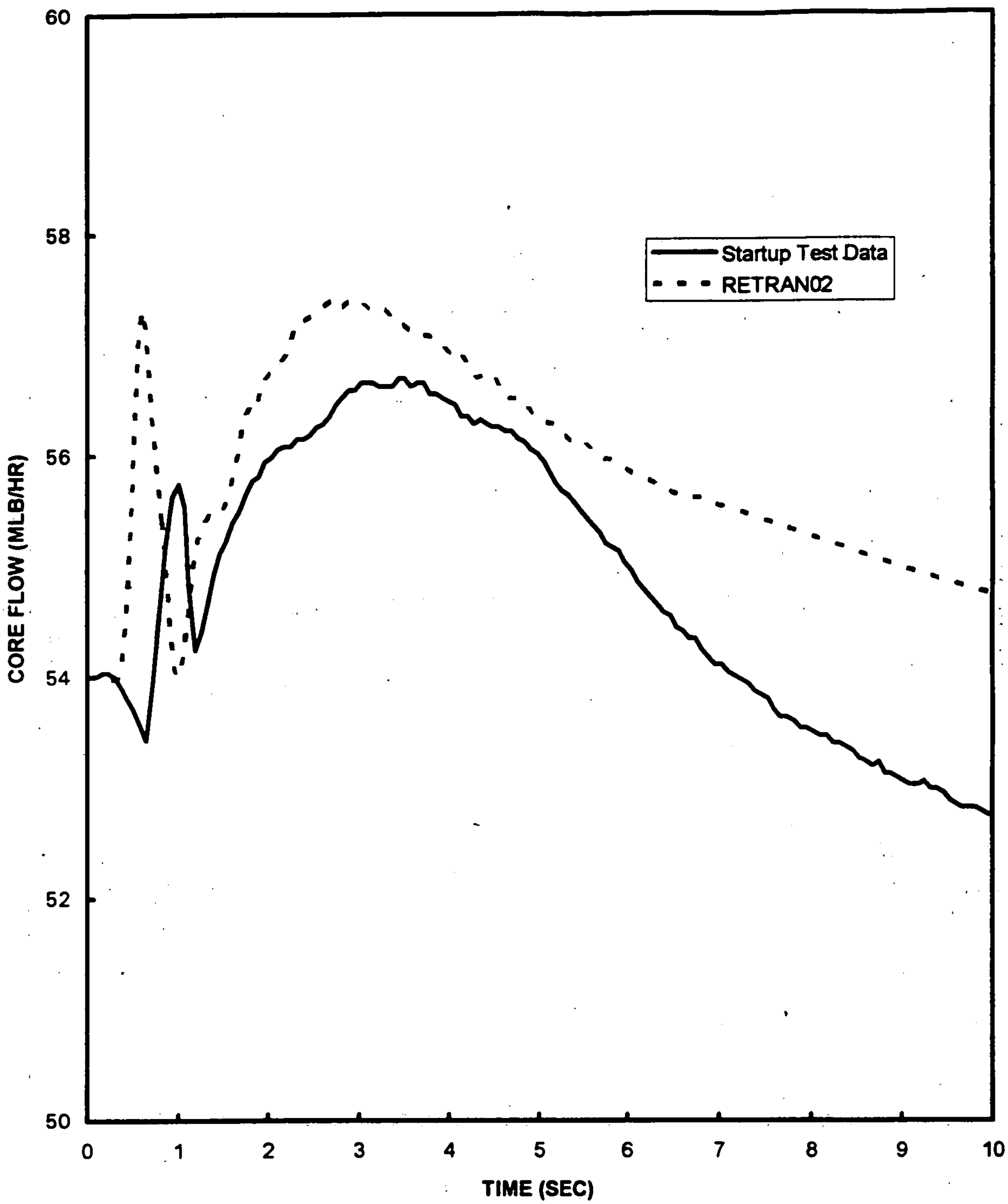

Figure 4.3-13: Quad-Cities BWR/3 TTWBP Core Flow

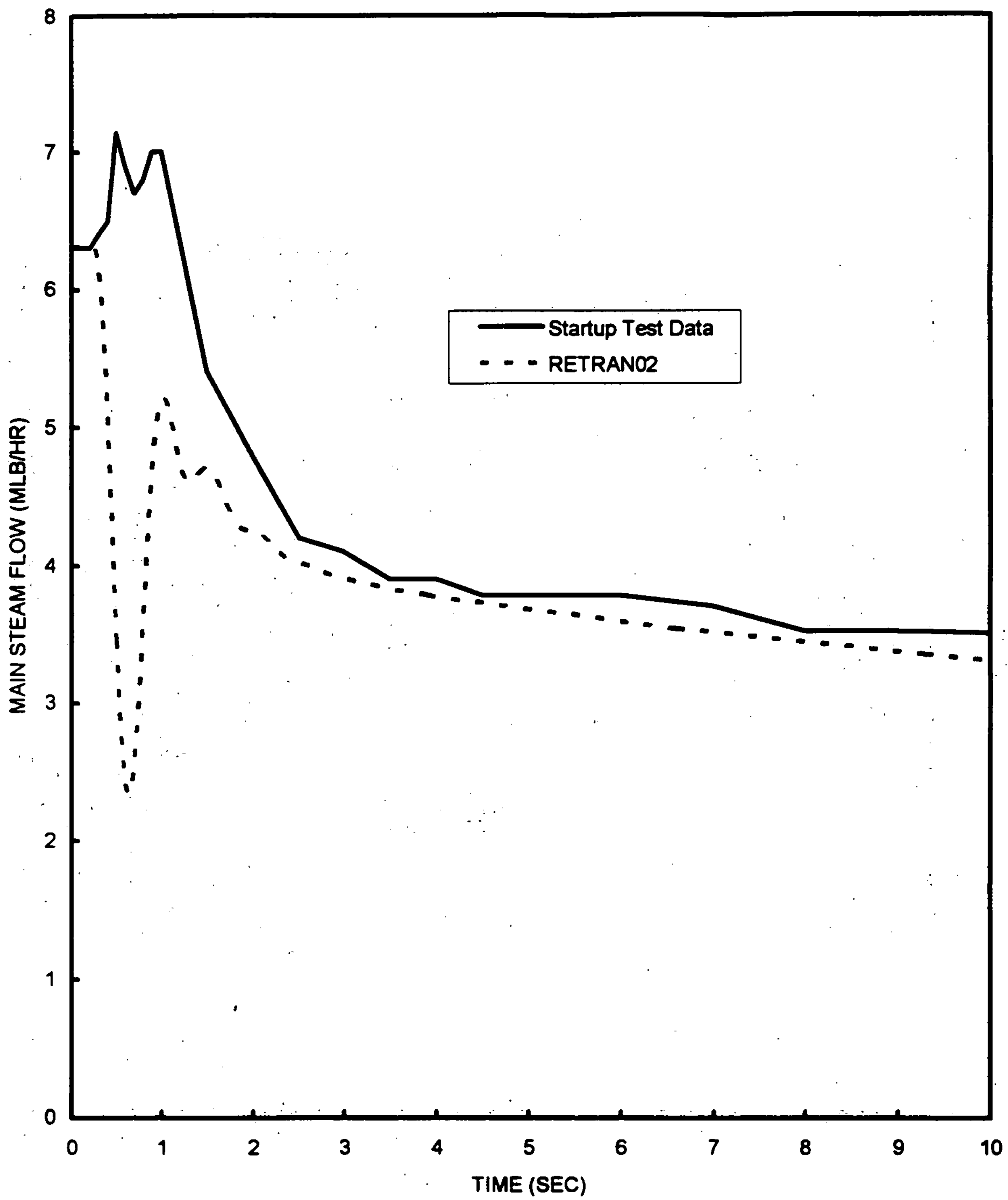

Figure 4.3-14 Quad-Cities BWR/3 TTWBP Steam Flow

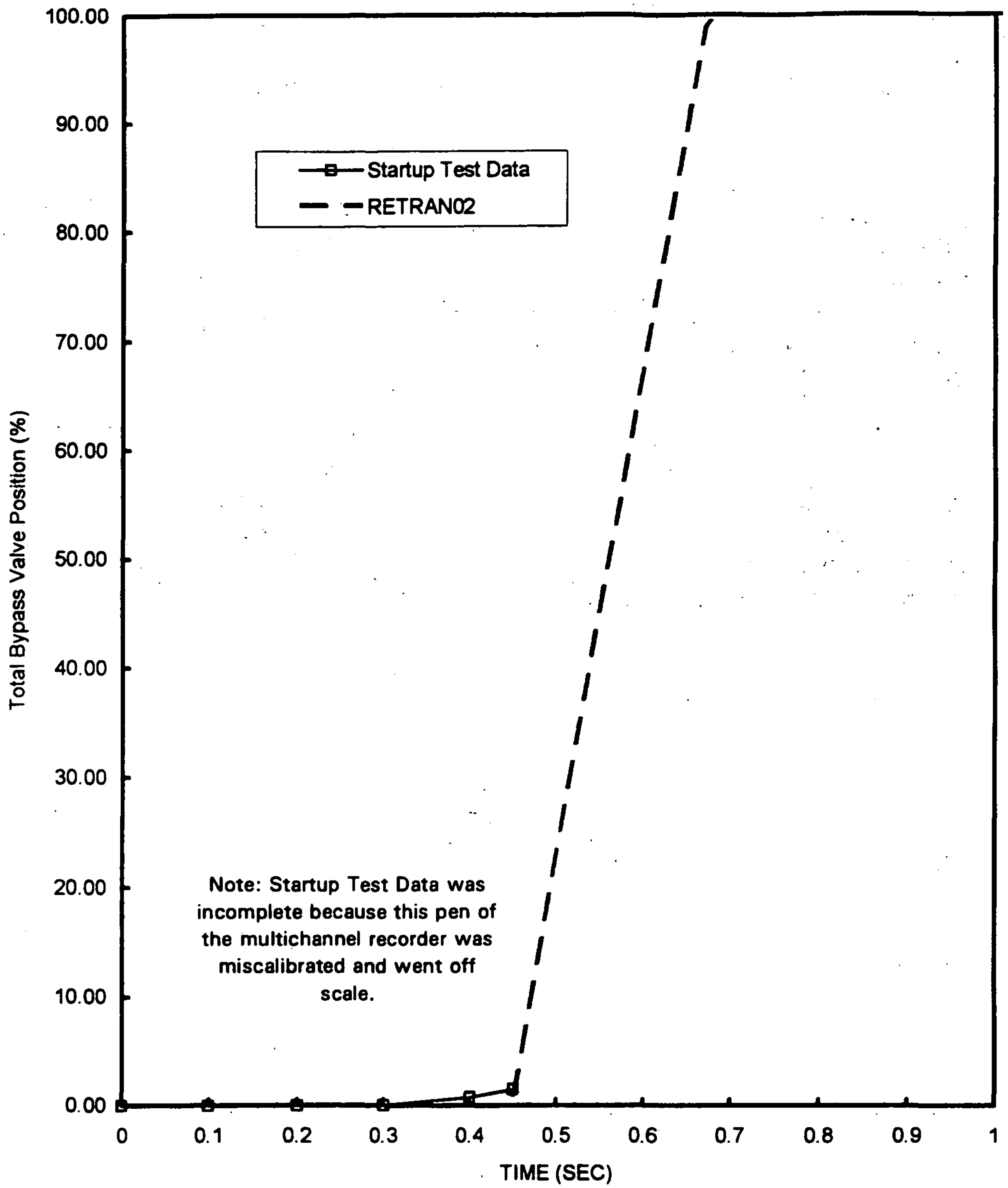

Figure 4.3-15 Quad-Cities BWR/3 TTWBP Bypass Valve Position

4.4 Startup Test Benchmark Summary

The analysis predicted powers, pressures, flows, levels and valve positions indicated in this report are acceptable and within measurable error ranges for the ComEd BWR RETRAN models. Table 4.4-1 and Table 4.4-2 show the ratings for the BWR/5 (LaSalle) and the BWR/3 (Quad-Cities and Dresden) respectively for each of the startup tests. Most parameters were predicted with a high degree of accuracy using the RETRAN models; however, some parameters fell within the generally acceptable range. Discussions of the comparisons of the RETRAN predictions against the test data can be found under "Model Response" in Sections 4.2 and 4.3. The RETRAN analyses had small differences compared to the plant data in reactor pressure, core power, core flow, steam flow, feed water flow, recirculation drive flow and bypass valve position results but was within the screening criterion. The RETRAN models accurately predicted the plant responses. All parameters of interest which RETRAN calculated behaved as anticipated.

The ComEd BWR RETRAN models are acceptable for best estimate predictions of plant behavior under pressure changes, core flow changes, water level changes and steam line isolations. It is validated for a wide range of plant transients at rated and off-rated initial conditions. These ComEd BWR plant startup test benchmarks demonstrated the validity of the models, methods and the qualification of ComEd to perform transient analysis for reload licensing and operational support applications.

Table 4.4-1 Ratings for LaSalle RETRAN Model Benchmark

Startup Test	Parameter of Interest	Rating*
Reactor Water Level Setpoint Change	Reactor Water Level	+
	Feedwater Flow	+
Pressure Regulator Setpoint Change	Reactor Pressure	+
	Core Power	+
	Steam Flow	+
	Bypass Valve Position	0
Dual Recirculation Pump Trip	Core Flow	+
	Core Power	+
	Reactor Pressure	+
	Steam Flow	+
Load Rejection With Bypass	Reactor Pressure	+
	Core Power	+
	Core Flow	+
	Steam Flow	+
	Bypass Valve Position	+
Main Steam Isolation Valves Closure	Steam Dome Pressure	+
	Core Power	+
	Core Flow	+
	Steam Flow	+

*Ratings Defined: (+) is acceptable, (0) is generally acceptable, (-) is unacceptable

Table 4.4-2 Ratings for Quad-Cities and Dresden RETRAN Model Benchmark

Startup Test	Parameter of Interest	Rating*
Reactor Water Level Setpoint Change	Reactor Water Level	+
	Feedwater Flow	+
Pressure Regulator Setpoint Change	Reactor Pressure	+
	Core Power	+
	Steam Flow	+
	Bypass Valve Position	+
Dual Recirculation Pump Trip	Core Flow	+
	Core Power	+
	Reactor Pressure	+
	Steam Flow	+
Turbine Trip With Bypass	Reactor Pressure	+
	Core Power	+
	Core Flow	+
	Steam Flow	+
	Bypass Valve Position	+

*Ratings Defined: (+) is acceptable, (0) is generally acceptable, (-) is unacceptable

5. Comparisons to Peach Bottom Turbine Trip Tests

5.1 Test Description

5.1.1 Introduction

The purpose of this section is to document the benchmark of the Peach Bottom RETRAN model in order to qualify Commonwealth Edison's methodology for modeling BWR pressurization events and reload licensing analysis applications. Peach Bottom turbine trip benchmarking was important because the turbine trip tests resulted in significant neutron flux peaks, similar to reload licensing transients. Special changes to the Peach Bottom plant's reactor protection system were made to allow a significant neutron flux spike without fuel damage. The direct scram on the turbine stop valve position 10% closed was disabled to produce the neutron flux spikes.

The goal is to benchmark the RETRAN thermal hydraulics and kinetics models to the three turbine trip tests described in References 15 and 16 and to show that the RETRAN modeling techniques are capable of predicting the neutronic and thermal hydraulic response during pressurization events. This neutronic and thermal hydraulic response is not usually seen during plant startup testing. Benchmarks have also been performed for the Dresden, Quad-Cities and LaSalle startup tests. Successful comparisons to the plant startup tests also serve as qualification of the RETRAN thermal hydraulics and control system models.

Each of the three turbine trip tests were benchmarked. These turbine trip tests provide excellent data for analytical code comparison. A Peach Bottom RETRAN Base Model was developed to match the Unit 2, Cycle 2 configuration of the core and NSSS per Reference 15 and 16. The Peach Bottom base model was used for each specific turbine trip case with initial condition changes to simulate the different power and flow conditions of each test.

One dimensional core kinetics were used with the methodology as described in Appendix A. The cross sections were developed for each turbine trip case to describe core feedback for each case. The MICROBURN code (Reference 17) was used with exposure accounting models of Peach Bottom Unit 2 for cycles 1 and 2 to allow nodal cross sections to be calculated for each of the three turbine trip test conditions.

5.1.2 Turbine Trip Test Description

A detailed description of the Peach Bottom turbine trip tests may be found in Reference 16. Below is a brief description of the tests.

There were three turbine trip tests performed at Peach Bottom 2 (PB2) at the end of cycle 2 in April of 1977. These tests are commonly referred to as TT1, TT2 and TT3. These tests were performed in a joint effort by Philadelphia Electric Company, General Electric and EPRI. The main purpose of the test was to investigate the neutron kinetic and thermal hydraulic effect of pressurization transients following turbine trips in a boiling water reactor. The tests also serve as qualification data for BWR safety analysis methods. The tests were set up to measure the major system variables such as neutron flux, core pressure and dome pressure. The data acquisition system was set up to measure the process variables every .006 seconds.

The turbine trip tests were initiated by manually tripping the turbine, which causes the turbine stop valve to close. Normally, the position switches on the turbine stop valves cause a reactor scram when the position switch is 10% closed. This reactor scram function was disabled for these tests to allow a significant neutron flux transient to occur. Instead, a reactor scram was initiated on the APRM high-high signal. The APRM scram clamps were used to lower the high flux scram setpoints. Water level was increased up to the high level alarm signal to prevent a low reactor water level isolation after the turbine trip. The reactor was allowed to reach equilibrium Xenon before each test. Extensive special instrumentation to record the tests results were installed as described in Reference 16 and a description will not be repeated here. Each test was performed at different power levels and at near rated core flow. Table 5.3-1 summarizes the initial conditions and APRM scram setpoint for each case.

5.1.3. Measured Data

Turbine trip transient data was edited and copied into files by EPRI. These files were set up in ASCII format. This raw unfiltered data was acquired by ComEd from EPRI. This data was used to benchmark the ComEd Peach Bottom RETRAN model. Reference 18 contained point by point data in increments of 0.006 seconds of all LPRM detectors, all APRM signals, pressures, temperatures, water level and positions for the various points in the reactor. Data from Reference 18 began at a steady state condition prior to the turbine trip. ComEd used the beginning of turbine stop valve motion to determine the transient initiation time for the benchmarking effort.

Normalization of the core averaged LPRM and A, B, C and D level averaged LPRM's were calculated with respect to the initial data point. This was verified to be consistent with Reference 16 plotted data.

5.2 Peach Bottom RETRAN System Model

The Peach Bottom model was constructed with the same modeling assumptions, options and techniques used in the construction of ComEd's BWR RETRAN system models, i.e., plant nodalization schemes are consistent even though there may be differences in physical plant parameters. Table 5.2-1 summarizes the comparison between ComEd Nuclear units and Peach Bottom Unit 2. The main concept was to keep as much RETRAN modeling consistency as possible between the Peach Bottom model and the ComEd BWR system models. This will assure that the modeling techniques used in benchmarking the pressurization events at Peach Bottom are also used in ComEd's BWR RETRAN system models. A nodalization diagram of the Peach Bottom RETRAN model is shown in Figure 5.2-1. The RETRAN model consists of 24 heated core nodes. The Main Steam line had 7 volumes. Each Recirculation loop was modeled separately but the ten jet pumps for each loop were lumped into a single jet pump. The jet pump M-N performance was benchmarked to published data (Reference 6). Each recirculation loop contained 3 nodes representing the suction piping, recirculation pump and the discharge piping. The upper plenum is represented by 1 volume, while the standpipes and separators were each lumped into 1 RETRAN volume. The steam dryer and dome region were lumped into 1 RETRAN volume. The downcomer was split into three volumes and represents the region outside the separators, standpipes and core shroud. The main steam bypass lines and orifice sections were lumped into two RETRAN volumes.

Table 5.2-1, Comparison of ComEd Units and Peach Bottom Unit 2

Plant Name	Dresden Units 2 & 3	Quad-Cities Units 1 & 2	LaSalle Units 1 & 2	Peach Bottom Unit 2
Reactor Type	GE BWR/3	GE BWR/3	GE BWR/5	GE BWR/4
Containment Type	Mark-I	Mark-I	Mark-II	Mark-I
Rated Thermal Power (MW)	2527	2511	3323	3293
Rated Dome Pressure (psig)	1005	1005	1005	1005
Steamline Pressure Drop (psid)	55	55	45	55
Rated Core Coolant Flow (Mlbm/hr)	98.0	98.0	108.5	102.5
Rated Feedwater Temperature (F)	340	340	420	376
Rated Feedwater/Steam Flow (Mlbm/hr)	9.8	9.76	14.3	13.4
Recirculation Flow Control Method	M/G	M/G	FCV	M/G
Number of Recirculation Pumps	2	2	2	2
Number of Jet Pumps	20	20	20	20
Number of Safety Valves	8	8	*18	**
Number of Relief Valves	4	4	*	11
Number of T/R Safety/Relief Valve	1	1	0	0
Active Fuel Length (Inches)	144	144	150	144
Number of Control Rods	177	177	185	185
Number of Fuel Bundles	724	724	764	764

* Each valve has dual function of safety and relief

** Safety Valves were not used in Peach Bottom modeling

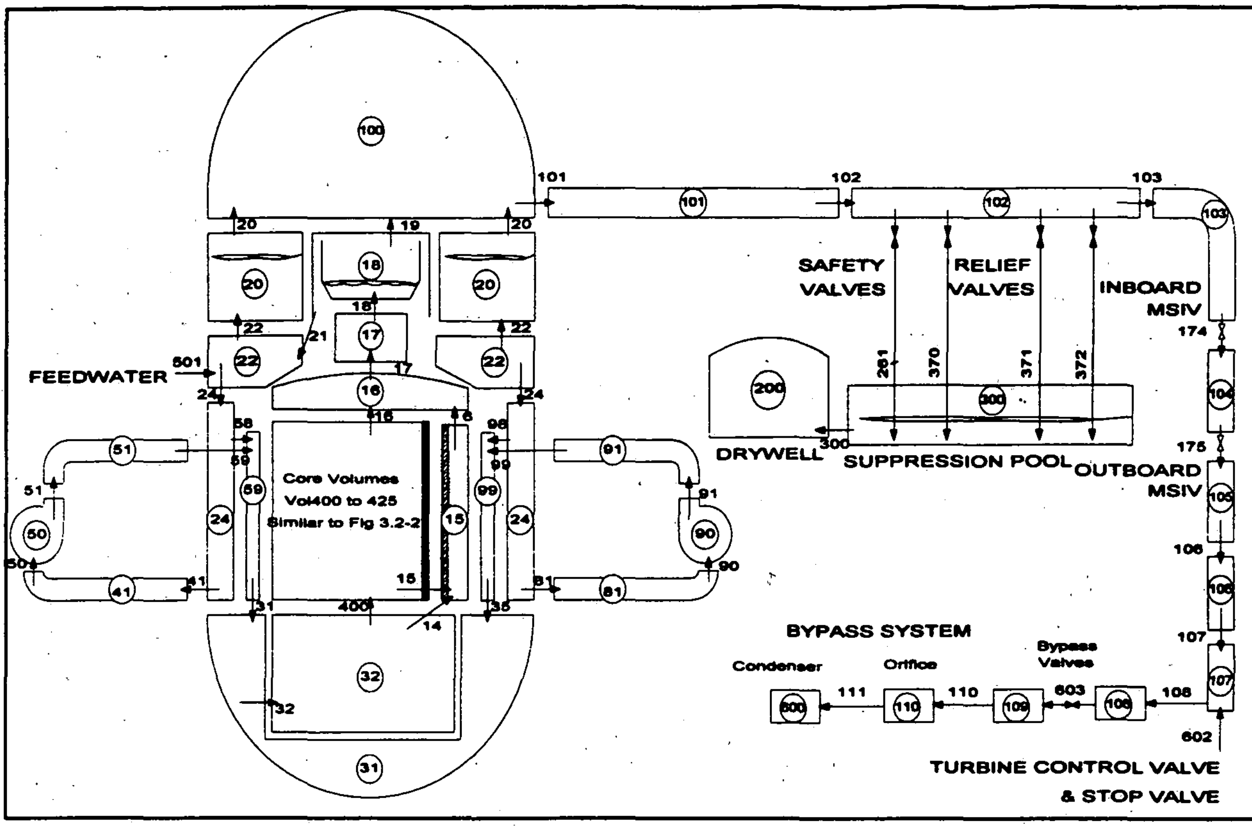

Figure 5.2-1, ComEd Peach Bottom RETRAN Model

5.3 Initial Conditions and Transient Modeling

For the benchmarking effort, the RETRAN model was initialized at the measured initial conditions. Table 5.3-1 and Table 5.3-2 summarize the key initialization parameters for the RETRAN model. The core power, core flow and core inlet enthalpy were obtained from the Reference 16 EPRI report. The core bypass flow and core pressure distribution were calculated by FIBWR2 and these values were used as RETRAN initialization inputs consistent with the methods outlined in Appendix B. There were some inconsistencies noted in Reference 16 in regard to system pressures. For this reason, the system pressures documented in Reference 6 were used. The steam flows were calculated and set to a value which yielded the proper heat balance and resulted in a RETRAN feedwater enthalpy that is consistent with the feedwater temperatures reported in Reference 16. The recirculation flow was adjusted for each case to balance the downcomer to lower plenum pressure distribution.

After a proper initialization is obtained, a null transient is run. This assures that RETRAN obtained a thermal-hydraulic steady state and it shows that the control system initial conditions (CIC's) are set properly such that the control systems do not drive the system away from the initialized values. The feedwater control system from the Quad Cities RETRAN model was used in the simulations. The feedwater flow response has very little impact on the calculated neutron flux and pressure for the Peach Bottom turbine trip tests.

Each turbine trip transient case was set up to scram on high neutron flux consistent with the test. The RETRAN trip setpoints were set to values consistent with the APRM setpoints listed in Table 5.3-1. A recirculation pump trip at 3.0 seconds was modeled for the TT3 test per Reference 16. The TT1 and TT2 tests did not have the recirculation pump trip.

The separator inertia value was a case specific input. Per Reference 7, the separator inertia is a function of inlet quality. Therefore, the total separator inertia was determined based on the inlet quality and applied at RETRAN junctions 18 and 21.

The bypass valve position as a function of time and the turbine stop valve closure times were also case specific inputs. The bypass valve and turbine stop valve positions were recorded during each turbine trip test and were then used to describe the valve motion during the RETRAN calculation. Since the TSV signal for TT1 failed, the TSV closure time from the TT2 test was assumed for TT1.

The scram delay time (time from high neutron flux trip to rod movement) was incorporated into a scram speed table. The scram delay time was obtained from the recorded data and input for each case, while the average of the measured scram speeds from Reference 16 was assumed for the three turbine trip test simulations.

Each TT transient also had a case specific gap conductance. The gap conductance mainly varies with exposure and initial power level. The gap conductance were calculated consistent with the method outlined in Appendix C. Constant gap conductance values were implemented in RETRAN. The gap expansion model was not used.

The measurement of system variables at the plant inherently includes some delays. These delays were estimated in Reference 16 for the pressure measurement signals. The delays for the core exit pressure and reactor pressure were estimated to be ~30 milliseconds. This delay was also assumed for the turbine inlet pressure. For all 3 transient change decks, control logic was added to "lag" RETRAN system variables (core exit pressure, reactor pressure and turbine inlet pressure) for comparison to measured data.

Figure 3.1-10 represents a control system to simulate the neutron flux. The RETRAN variable PNRM is typically used to show the relative change in neutron power, however, this change in power includes decay heat power. Therefore the PNRM variable was modified with a control system to subtract out the decay heat. This will yield a closer approximation to neutron flux for comparison to the measured Peach Bottom neutron flux. The decay heat fractions and corresponding λ's were obtained from Reference 19. The normalized neutron flux value is then delayed by .02 seconds to model the LPRM response time (Reference 20). The individual LPRM level signals (A, B, C and D) and the core averaged LPRM was calculated consistent with Figure 3.1-11 for comparison to the measured data.

Table 5.3-1 Peach Bottom Turbine Trip Test Conditions

Test	Reactor Power		Core Flow Rate		APRM scram setpoint
	(MWt)	(% Rated)	(10^6 lbm/hr)	(% Rated)	(% Rated)
TT1	1562	47.4	101.3	98.8	85
TT2	2030	61.6	82.9	80.9	95
TT3	2275	69.1	101.9	99.4	77

Table 5.3-2 RETRAN Model Initial Conditions

	TT1	TT2	TT3
Core Thermal Power (MWt)	1562.0	2030.0	2275.0
Core Flow (lbm/sec)	28138.9	23027.8	28305.6
Core Bypass Flow (lbm/sec)	1927.5	1605.0	2060.6
Steam Dome Pressure (psia)	994.0	986.0	993.0
Core Exit Pressure (psia)	999.9	991.2	999.4
Core Inlet Enthalpy (BTU/lbm)	528.4	519.8	523.6
Steam Flow (lbm/sec)	1628.0	2176.0	2475.6
Recirculation Flow (lbm/sec)	8666.2	7101.0	8911.3

5.4 Results

5.4.1 Pressure Comparisons

The Peach Bottom turbine trip measured pressures were compared to the RETRAN02 system model calculated pressures. Three pressure comparisons were made for each test. The three pressures were: turbine inlet pressure, reactor dome pressure and core exit pressure. The RETRAN02 pressures were lagged by 30 msec per Reference 16. The time constant for the turbine inlet sensor was not known, however, a 30 msec time constant was also assumed for the turbine inlet pressure. Comparisons were made to unfiltered measured data as obtained from Reference 18. All pressure comparisons are shown in difference from the initial value.

Table 5.4-1 summarizes the key results of the pressure comparisons for the three turbine trip tests. Figure 5.4-1 through Figure 5.4-3 show the pressure comparisons for TT1. Figure 5.4-4 through Figure 5.4-6 show the pressure comparisons for TT2. And Figure 5.4-7 through Figure 5.4-9 show the pressure comparisons for TT3.

Figure 5.4-1, Figure 5.4-4 and Figure 5.4-7 show the comparison of the measured turbine inlet pressure response to the calculated RETRAN02 response for TT1, TT2 and TT3 respectively. The calculated response shows excellent agreement in both timing and magnitude. This comparison indicates that the pressure oscillations near the turbine stop valve are accurately calculated by RETRAN.

Figure 5.4-2, Figure 5.4-5 and Figure 5.4-8 show the comparison of the measured reactor dome pressure response to the calculated RETRAN02 response. The calculated response shows excellent agreement in both timing and magnitude. Figure 5.4-10, Figure 5.4-11 and Figure 5.4-12 show the dome pressure response for TT1, TT2 and TT3 respectively for the first 1.0 second of the transient. The accurate prediction of the steam dome response indicates that the steam line dynamic characteristics are accurately represented by the RETRAN02 steam line model. The initial steam dome pressure pulse for all three cases is accurately predicted to within about 1.5 psi. The TT1 pressure response is slightly over-predicted, while TT2 and TT3 are slightly under-predicted. The maximum dome pressure rise is also accurately calculated. The peak rise in dome pressure is within about 4.0 psi for each TT test. This comparison indicates that the steam bypass and gap conductance have been accurately modeled. The gap conductance essentially sets the initial fuel average temperature, which then sets the initial stored energy in the fuel. The gap conductance values for the three turbine trip tests were calculated using the core conditions at which the tests were initiated and based on the method in Appendix C. The stored energy is eventually released to the coolant, which then affects the pressure response. Also, the pressure dip after the first pressure pulse shows excellent agreement to the RETRAN02 calculation. This also shows that the steam line and bypass dynamics are accurately represented by the RETRAN02 steam line.

Figure 5.4-3, Figure 5.4-6, and Figure 5.4-9 show the comparison of the measured core exit pressure response to the calculated RETRAN02 response. All three comparisons show excellent agreement between the measured data and the RETRAN02 results. The calculated time of the core exit pressure rise and the rate of pressure increase both show very close agreement to the measured data. The measured core exit pressure was filtered to obtain a clearer comparison. A 5 Hz low pass filter was used to filter out the noise observed for the core exit pressure. Figure 5.4-13, Figure 5.4-14 and Figure 5.4-15 show the filtered core exit pressure response for TT1, TT2 and TT3, respectively, for the first 1.5 seconds of the transient. Table 5.4-1 shows the comparison of the core exit pressure at the first pressure oscillation. The TT1 calculated core exit pressure rise is about 4.2 psi higher than the measured response while the TT2 core exit response is just slightly under-predicted by RETRAN. The TT3 core exit pressure response is just slightly over-predicted by RETRAN. The accurate prediction of the core exit pressure response indicates that the separator dynamic characteristics are accurately represented by the RETRAN02 separator model.

The slight over-prediction of the TT1 core exit and dome pressure is likely due to the different bypass response observed for TT1 or the turbine stop valve closure time. The observed bypass response for TT1 indicated a rather large delay in the bypass valve response (over 300 msec) and there is some uncertainty in the turbine stop valve closure time per Reference 16. There also may be some uncertainty in the time at which the transient was initiated for TT1. Also, The TT1 initial conditions represent a low power high flow condition with a low separator inlet quality. The RETRAN02/MOD002 SER also stated that the attenuation of pressure waves through the RETRAN02 non-mechanistic separator model may not be simulated well at low flow/low quality conditions. The TT1 comparison shows that at the low quality conditions, the core exit pressure is over-predicted and conservative.

In summary, the RETRAN02 predicted pressure responses are very close to the measured data. The accurate prediction of system pressures indicates that the dynamic characteristics of pressurization events have been correctly modeled with RETRAN02.

5.4.2 Power Comparisons

Comparisons to the measured core averaged LPRM and the four axial LPRM signals are shown in Figure 5.4-21 through Figure 5.4-35 for the three turbine trip tests. In general, the RETRAN02 predictions show good agreement to the measured data. Table 5.4-2 summarizes the comparison of the measured vs. RETRAN02 calculated peak neutron flux results, Table 5.4-3 summarizes the peak neutron flux timing, Table 5.4-4 summarizes the integral power comparisons and Table 5.4-5 summarizes the net reactivity results. The measured reactivity and timing results were obtained from Reference 6. The RETRAN02 calculated reactivity in delta-k/k was converted to $ based on the case specific total β value. The integrated power was obtained by calculating the area under the calculated and measured APRM response curves from 0.0 to 1.25 seconds. Preliminary CPR studies show that the Minimum CPR will occur just before 1.25 seconds.

Figure 5.4-21 through Figure 5.4-25 show the flux responses for TT1. The core averaged LPRM flux response calculated by RETRAN02 over-predicts the measured response by about 17%. The integrated power at 1.25 seconds is about 7.2% higher than measured. The initial rise in flux slightly lags behind the measured response until the normalized peak neutron flux of 5.71 is reached at 0.78 seconds. The TT1 net reactivity is turned over by the combination of moderator and Doppler reactivity. The RETRAN02 LPRM axial level response shows a good comparison to the measured data for timing with an over-prediction in magnitude. RETRAN02 properly represents the axial shift as shown in the calculated D level response. The over-prediction in the TT1 response is most likely due to the slight over-prediction of the core exit pressure as discussed under the pressure comparisons. As discussed above, the over-prediction of the TT1 pressure is likely due to the different bypass response observed for TT1 or the turbine stop valve closure time. It is also noted that Reference 16 contained inconsistencies for the heat balance on TT1. The heat balance and initial vessel steam flow could have an impact on pressurization events.

Figure 5.4-26 through Figure 5.4-30 show the flux responses for TT2. The core averaged LPRM flux response calculated by RETRAN02 just slightly under-predicts the measured response by about 1.6%. The integrated power at 1.25 seconds is within approximately 2% of the measured integrated power. The initial rise in flux and the rate of flux change is very close to the measured response. An LPRM averaged normalized peak neutron flux of 4.46 is reached at 0.74 seconds. The TT2 net reactivity is turned over by the combination of moderator and Doppler reactivity. The timing and magnitude of all of the LPRM responses show good agreement to the measured results. The peak D level response is under-predicted by about 7%, however, the overall shift in flux to the top of the core is accurately predicted. At 1.25 seconds, just after the flux peak, the A level flux is a very small fraction of the initial value while the D level flux is still roughly 50% of the initial value. The under-prediction of the peak D level response is a common trend for all three turbine trip tests, but the D level still matches the overall trend and shift of flux to the top of the core.

Figure 5.4-31 through Figure 5.4-35 show the flux responses for TT3. The core averaged LPRM flux response calculated by RETRAN02 slightly over-predicts the measured response by about 5.5%. The integrated power at 1.25 seconds is within 1% of the measured integrated power. The initial rise in flux and the rate of flux change is very close to the measured response. A normalized peak neutron flux of 5.18 is reached at 0.71 seconds. The TT3 net reactivity is turned over by the initiation of scram reactivity. The timing and magnitude of all of the LPRM responses show good agreement to the measured results. As discussed above, the overall shift in flux to the top of the core is accurately predicted.

In summary, the timing and magnitude of the calculated flux responses show good agreement to the measured response. The calculated net reactivities are all within about ±2% of measured. This is due to the accurate prediction in core exit pressure and the accuracy of the one dimensional kinetics prediction of the void reactivity. The neutron flux peaks are very close considering that the neutron flux is very sensitive to small changes in reactivity as the net reactivity approaches prompt critical. Therefore, the one-dimensional kinetics methods and RETRAN02 modeling techniques can accurately predict the dynamic behavior of BWR pressurization events.

5.4.3 TT1 Bypass Valve Response

As a sensitivity calculation, the TT1 transient was rerun assuming a different bypass response. The new bypass response was input with an opening time about 90 milli-seconds sooner than the previously assumed bypass delay.

The pressure comparisons for this case are shown in Figure 5.4-16 through Figure 5.4-20. The dome pressure at the first oscillation is slightly under-predicted by less than 1 psi (see Figure 5.4-19) and the core exit pressure at the first oscillation is just slightly over-predicted by just over 1 psi. Overall the core exit pressure response for this case matches the measured data much better than the base case. The pressure response is then under-predicted for the remainder of the transient, after the first pressure oscillation. It is speculated that the loss of anticipatory full-open bypass valve signal due to the TSV signal failure may have resulted in less than 100% bypass valve position depending on the load limit setting and the maximum combined flow at the time of the test.

The neutron flux comparisons are shown in Figure 5.4-36 through Figure 5.4-40. The APRM response (Figure 5.4-36) shows a much better comparison to the measured data. The peak normalized neutron flux was 5.04, which was about 4.0% higher than the measured data. The individual A,B,C and D level responses also compare very well with the measured data.

This analysis shows the adequacy of the cross section file (void and Doppler coefficients) developed with the methods described in Appendix A since the neutron flux peak was very accurately calculated when the core exit pressure response closely follows the measured data.

Table 5.4-1 Pressure Comparison Summary

TT1	Measured	Calculated	Difference
Increase In Dome Pressure (psi)*	33.0	34.4	1.4
Increase in Core Exit Pressure (psi)*	33.0	37.2	4.2
Maximum Dome Pressure Rise (psi)	39.0	40.4	1.4

TT2	Measured	Calculated	Difference
Increase In Dome Pressure (psi)*	41.5	40.4	-1.1
Increase in Core Exit Pressure (psi)*	44.9	44.5	-0.4
Maximum Dome Pressure Rise (psi)	64.5	66.5	2.0

TT3	Measured	Calculated	Difference
Increase In Dome Pressure (psi)*	48.0	46.9	-1.1
Increase in Core Exit Pressure (psi)*	50.9	51.1	0.2
Maximum Dome Pressure Rise (psi)	75.0	79.3	4.3

*Values are for the first pressure pulse

Table 5.4-2 Normalized Power Comparison Summary

Peak Normalized Power		Level A	Level B	Level C	Level D	Core Avg
TT1	Calculation	4.27	5.70	6.46	6.49	**5.71**
	Data	3.51	4.49	5.26	5.63	**4.86**
	% Difference	21.7	26.9	22.8	15.3	**17.5**
TT2	Calculation	3.66	4.68	4.92	4.64	**4.46**
	Data	3.52	4.49	4.88	4.98	**4.53**
	% Difference	4.0	4.2	0.8	-6.8	**-1.6**
TT3	Calculation	3.74	5.41	6.04	5.63	**5.18**
	Data	3.68	4.85	5.44	5.52	**4.91**
	% Difference	1.6	11.5	11.0	2.0	**5.5**

Table 5.4-3 Peak Core Average Neutron Flux Timing

Test Number	Calculated (sec)	Measured (sec)
TT1	0.78	0.77
TT2	0.74	0.73
TT3	0.71	0.70

Table 5.4-4 Integrated Power Comparison (t=0.0 to 1.25 sec)

Test Number	Calculated	Measured	% Difference
TT1	2.22	2.07	7.2
TT2	1.88	1.84	2.2
TT3	1.81	1.80	0.6

Table 5.4-5 Net Reactivity Summary

	Peak Reactivity ($)			Time of Peak (sec)	
	Calculated	Measured	% Difference	Calculated	Measured
TT1	0.785	0.770	1.9	0.72	0.73
TT2	0.759	0.767	-1.0	0.67	0.69
TT3	0.830	0.816	1.7	0.67	0.67

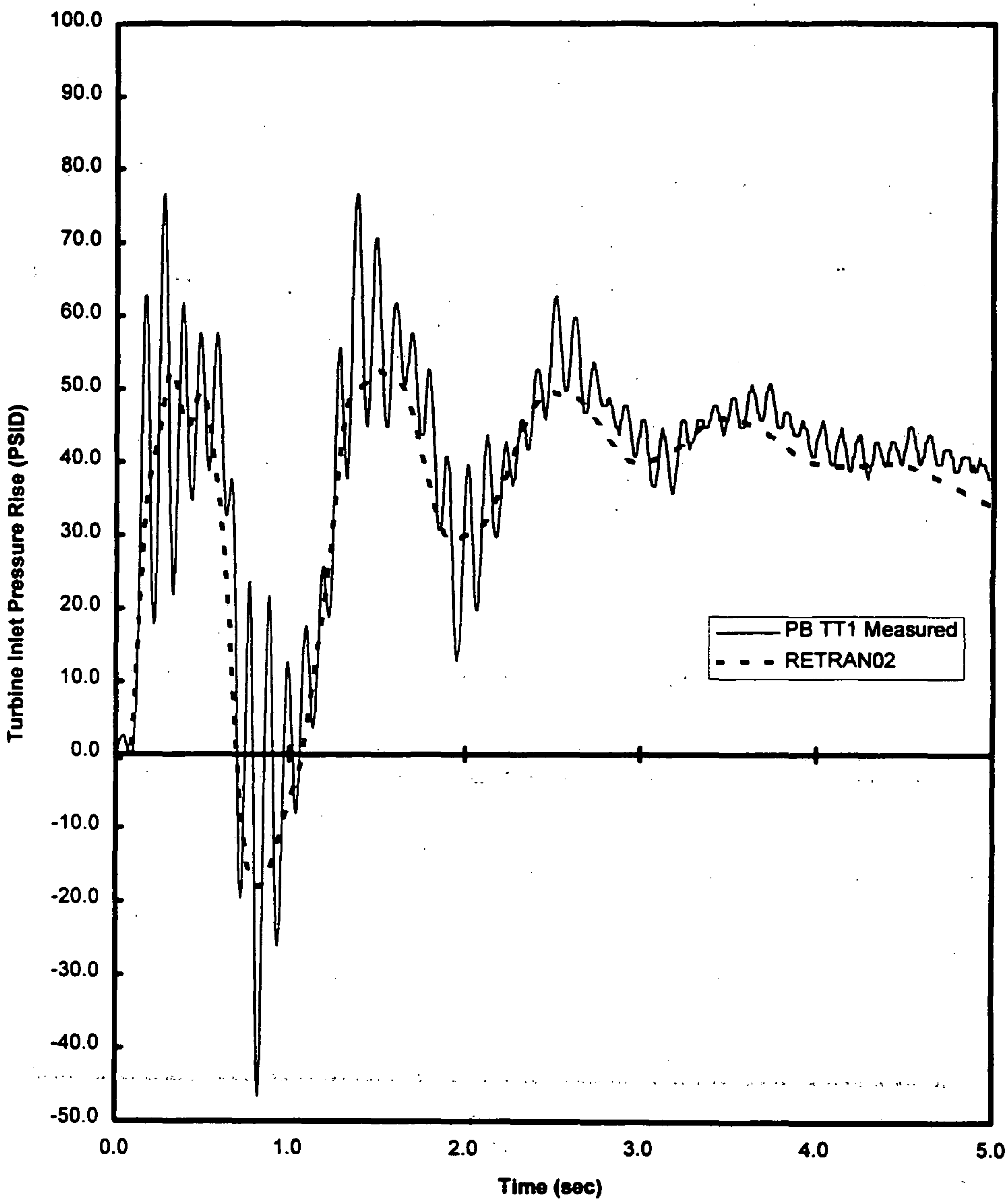

Figure 5.4-1 Turbine Trip Test 1 Turbine Inlet Pressure Rise

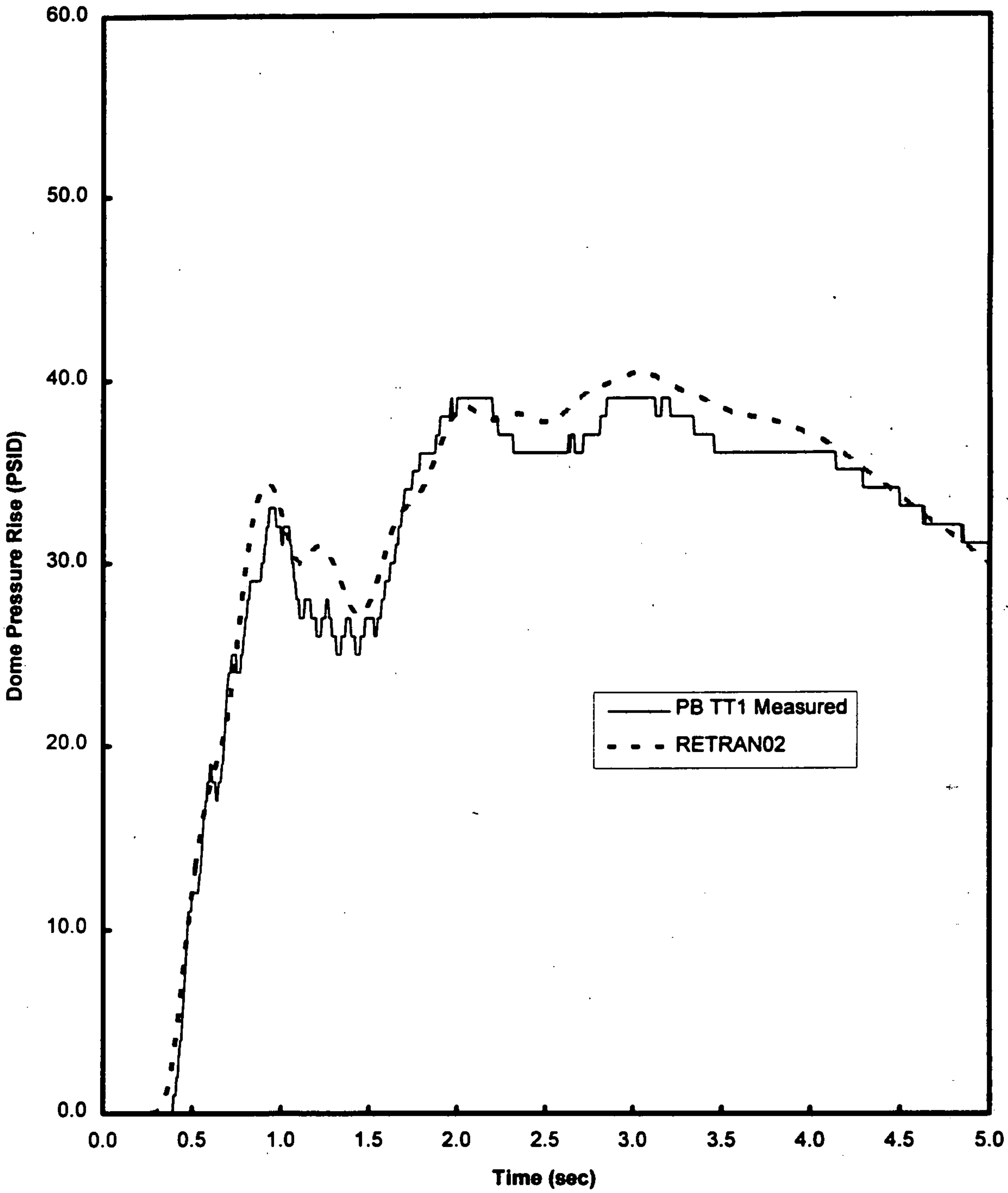

Figure 5.4-2 Turbine Trip Test 1 Reactor Dome Pressure Rise

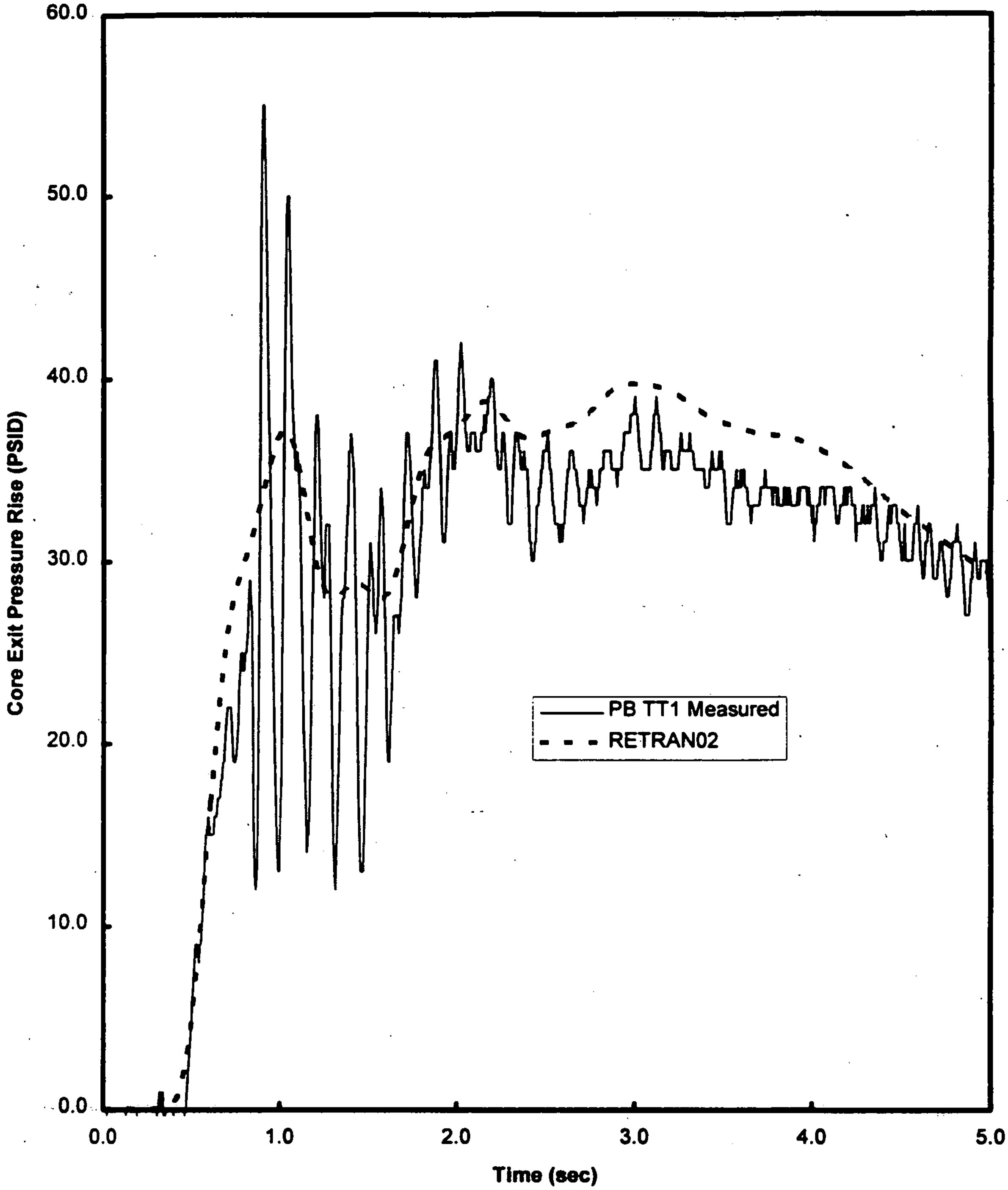

Figure 5.4-3 Turbine Trip Test 1 Core Exit Pressure Rise

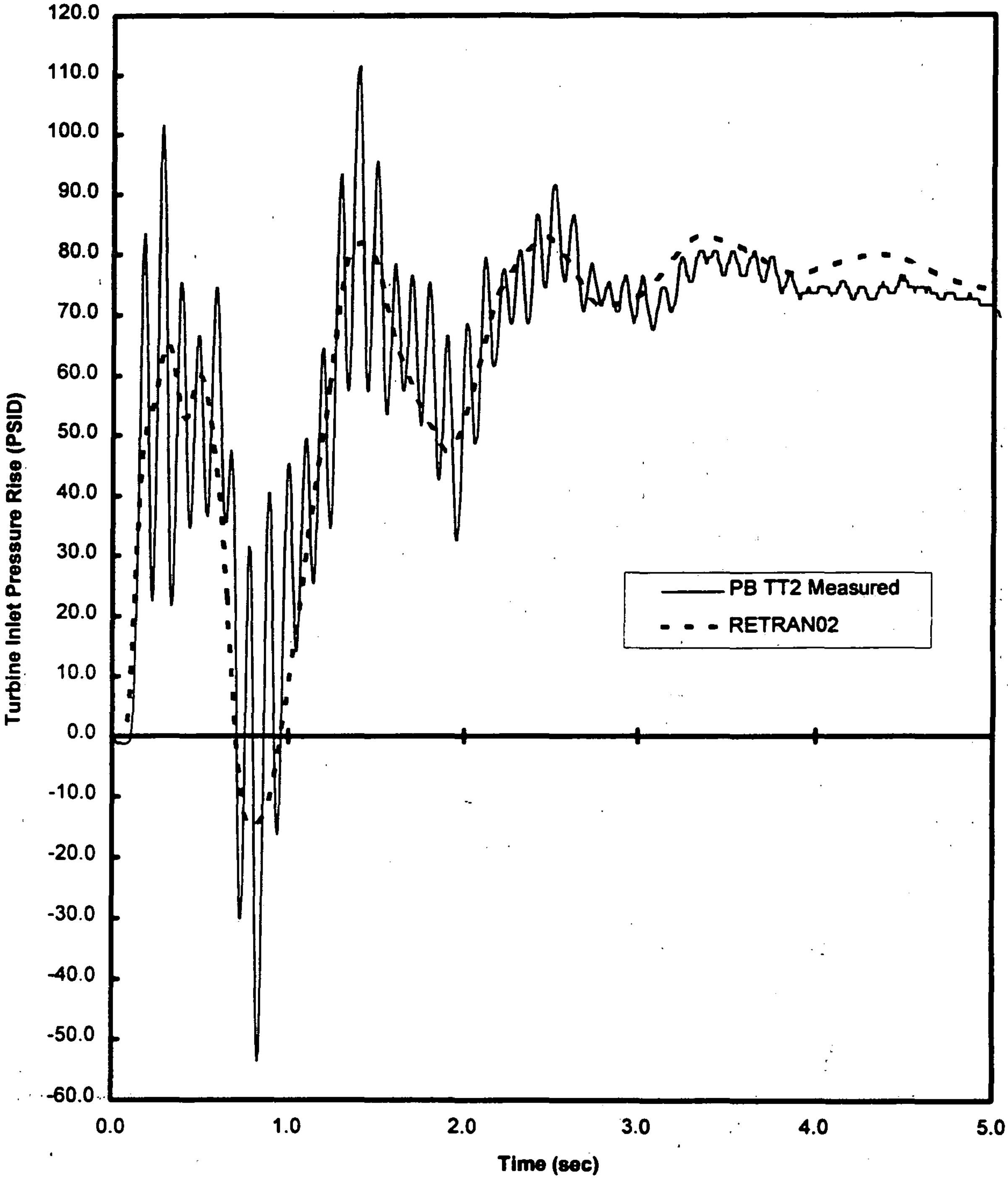

Figure 5.4-4 Turbine Trip Test 2 Turbine Inlet Pressure Rise

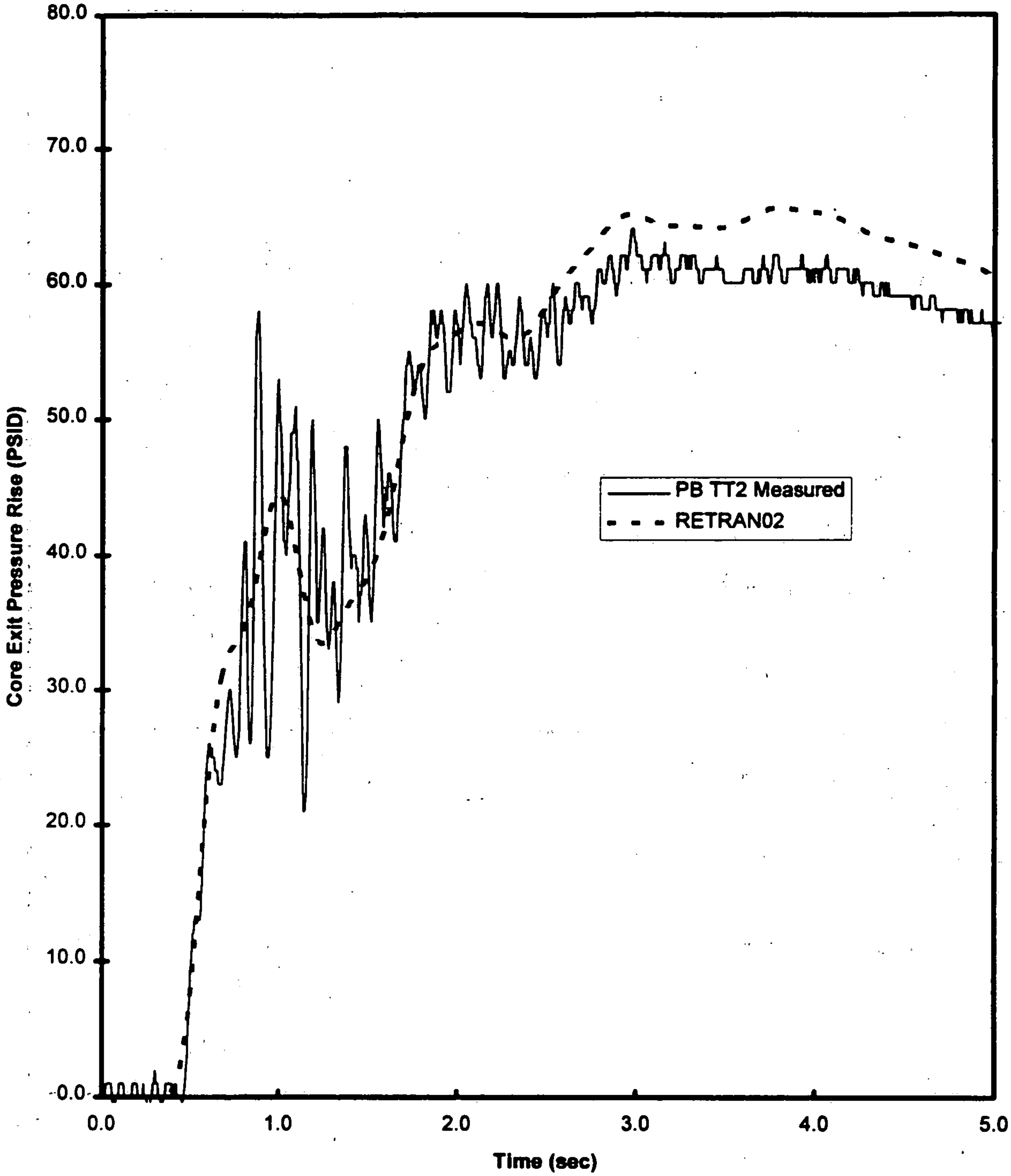

Figure 5.4-5 Turbine Trip Test 2 Reactor Dome Pressure Rise

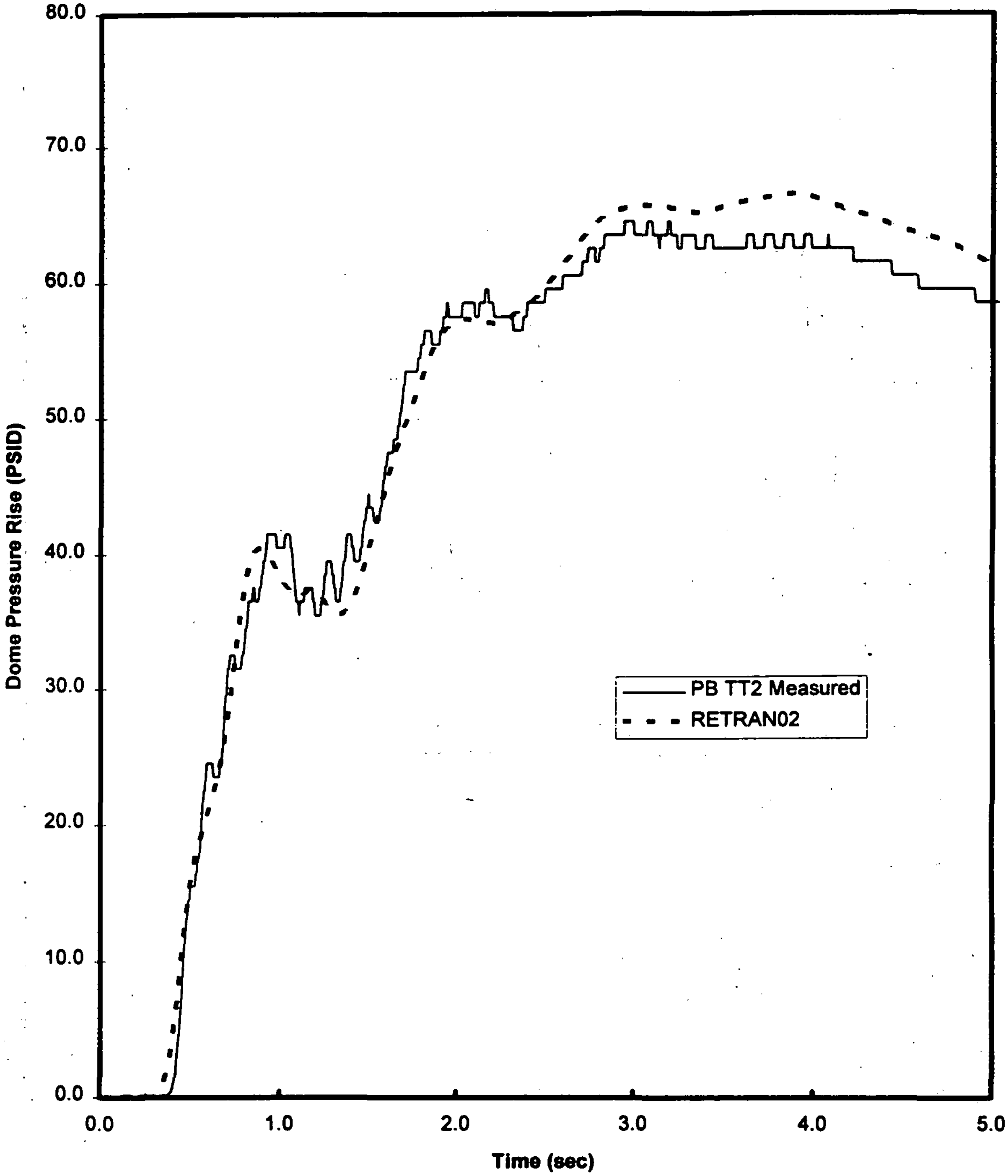

Figure 5.4-6 Turbine Trip Test 2 Core Exit Pressure Rise

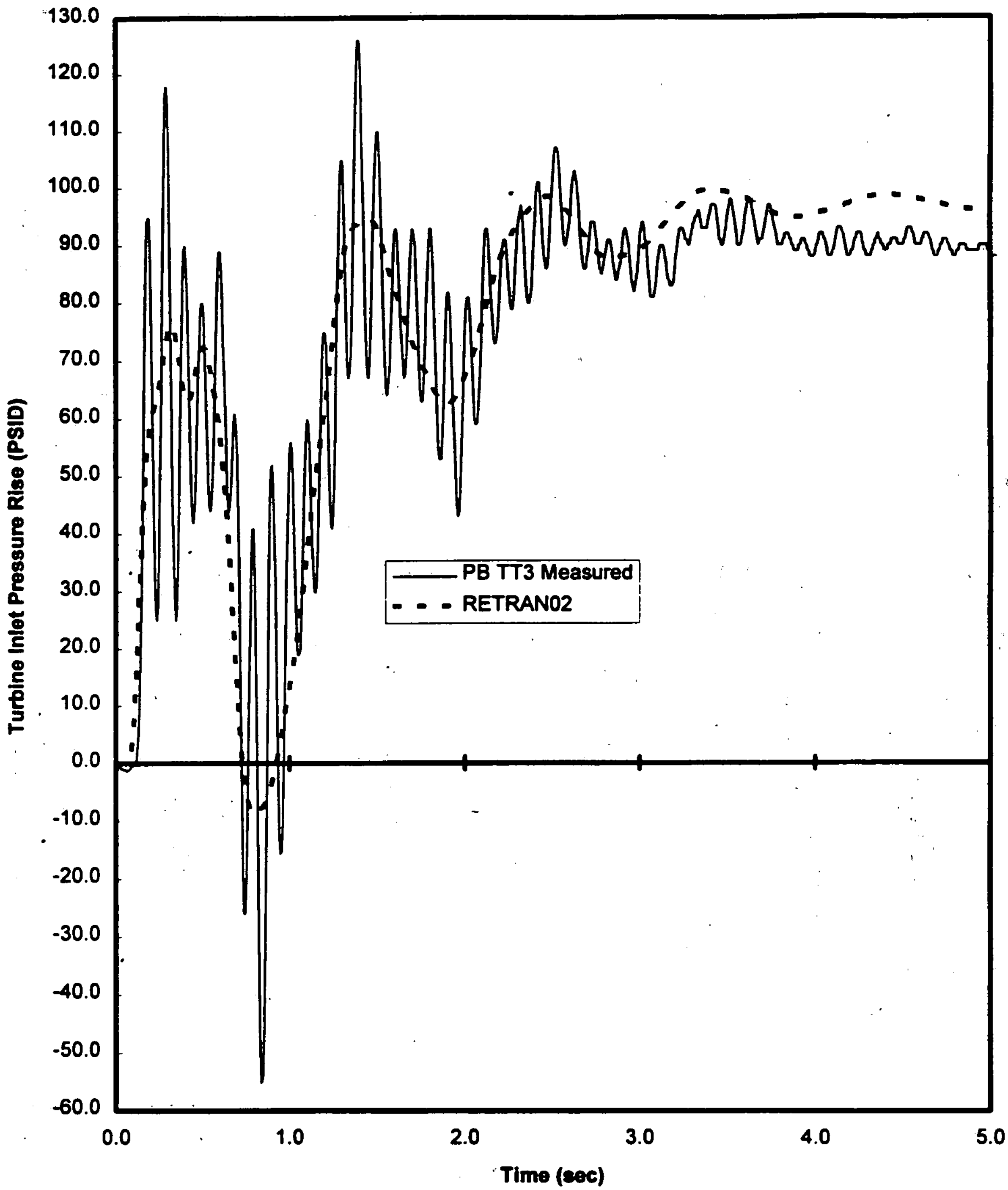

Figure 5.4-7 Turbine Trip Test 3 Turbine Inlet Pressure Rise

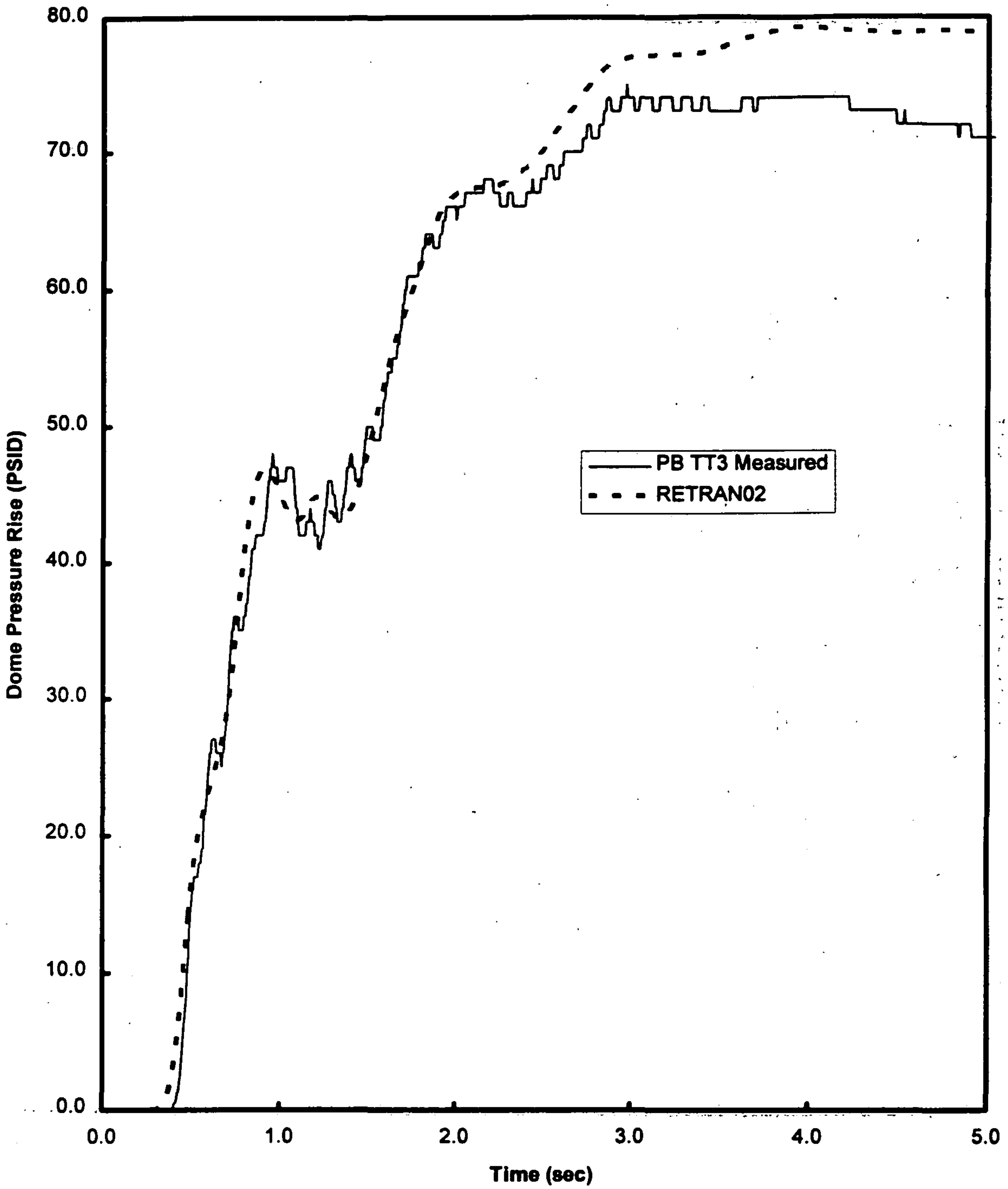

Figure 5.4-8 Turbine Trip Test 3 Reactor Dome Pressure Rise

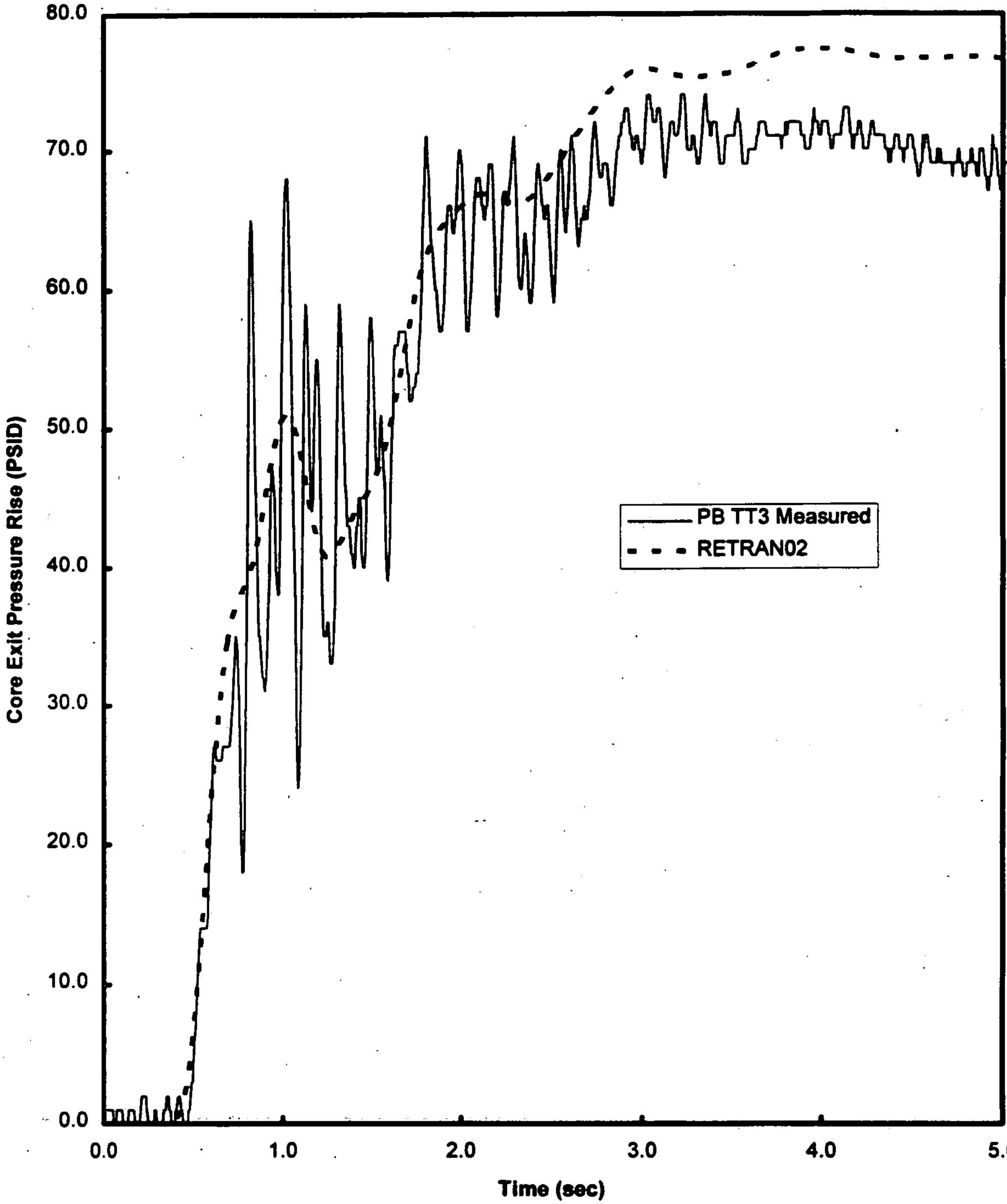

Figure 5.4-9 Turbine Trip Test 3 Core Exit Pressure Rise

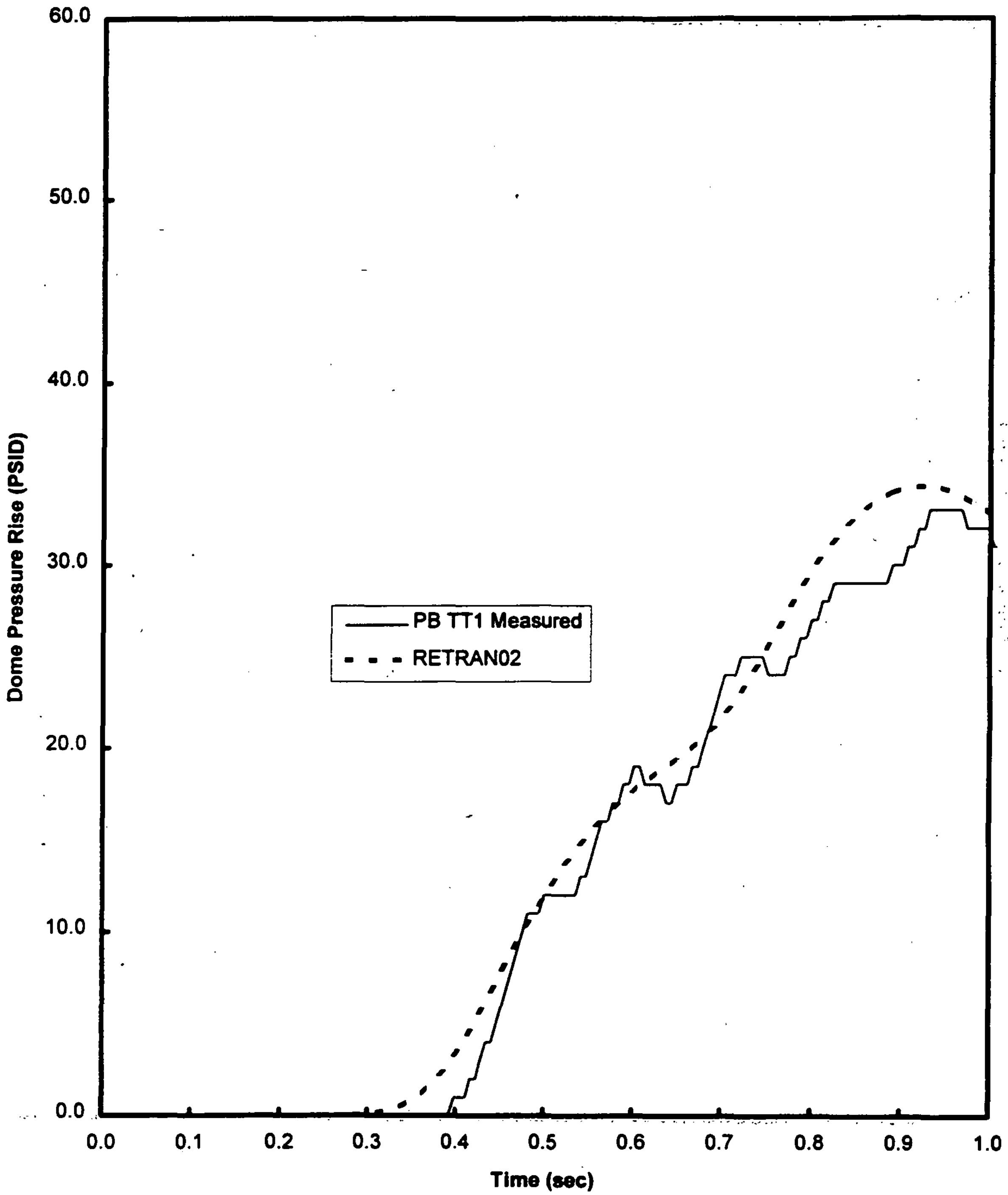

Figure 5.4-10 Turbine Trip Test 1 Reactor Dome Pressure Rise

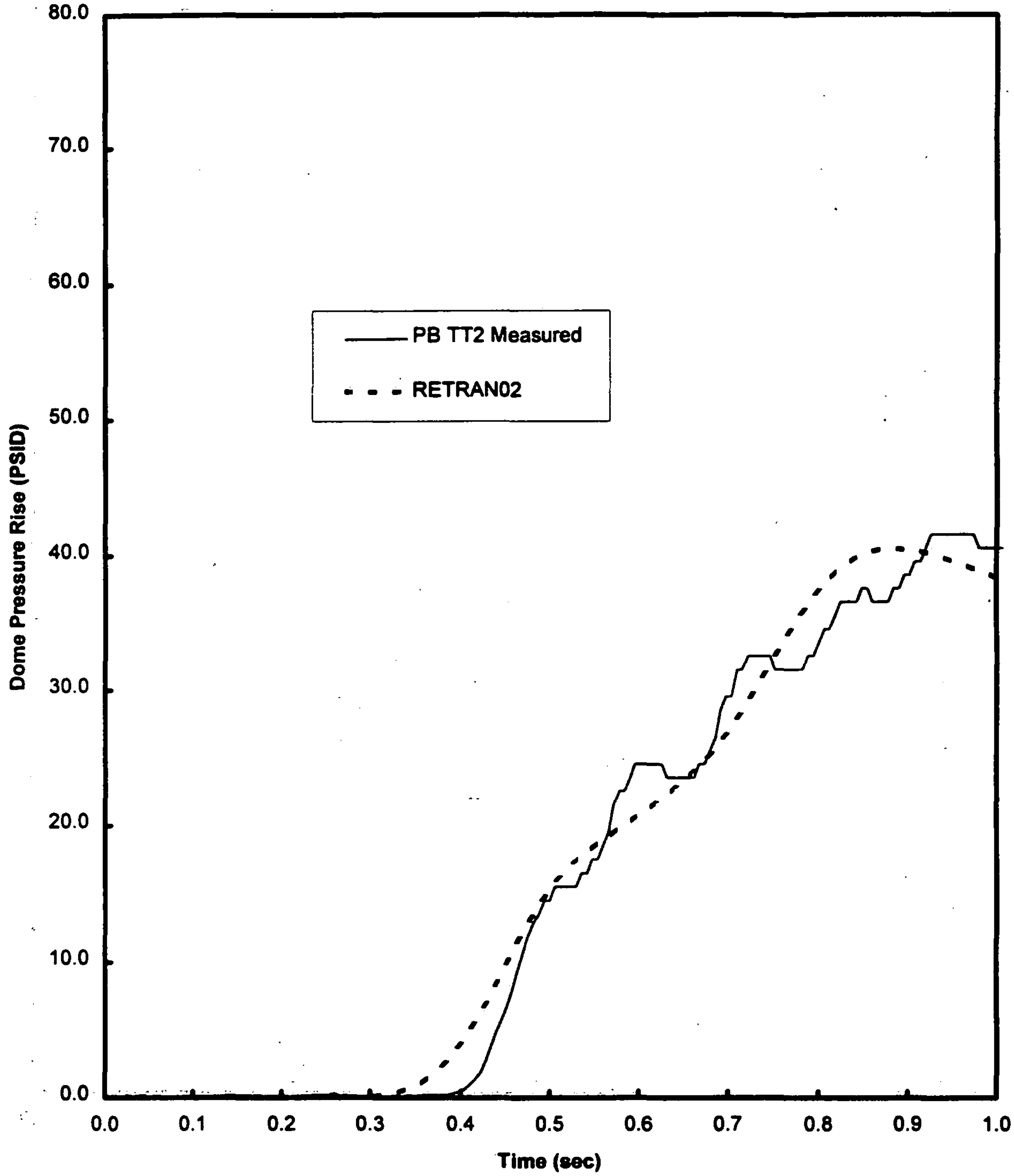

Figure 5.4-11 Turbine Trip Test 2 Reactor Dome Pressure Rise

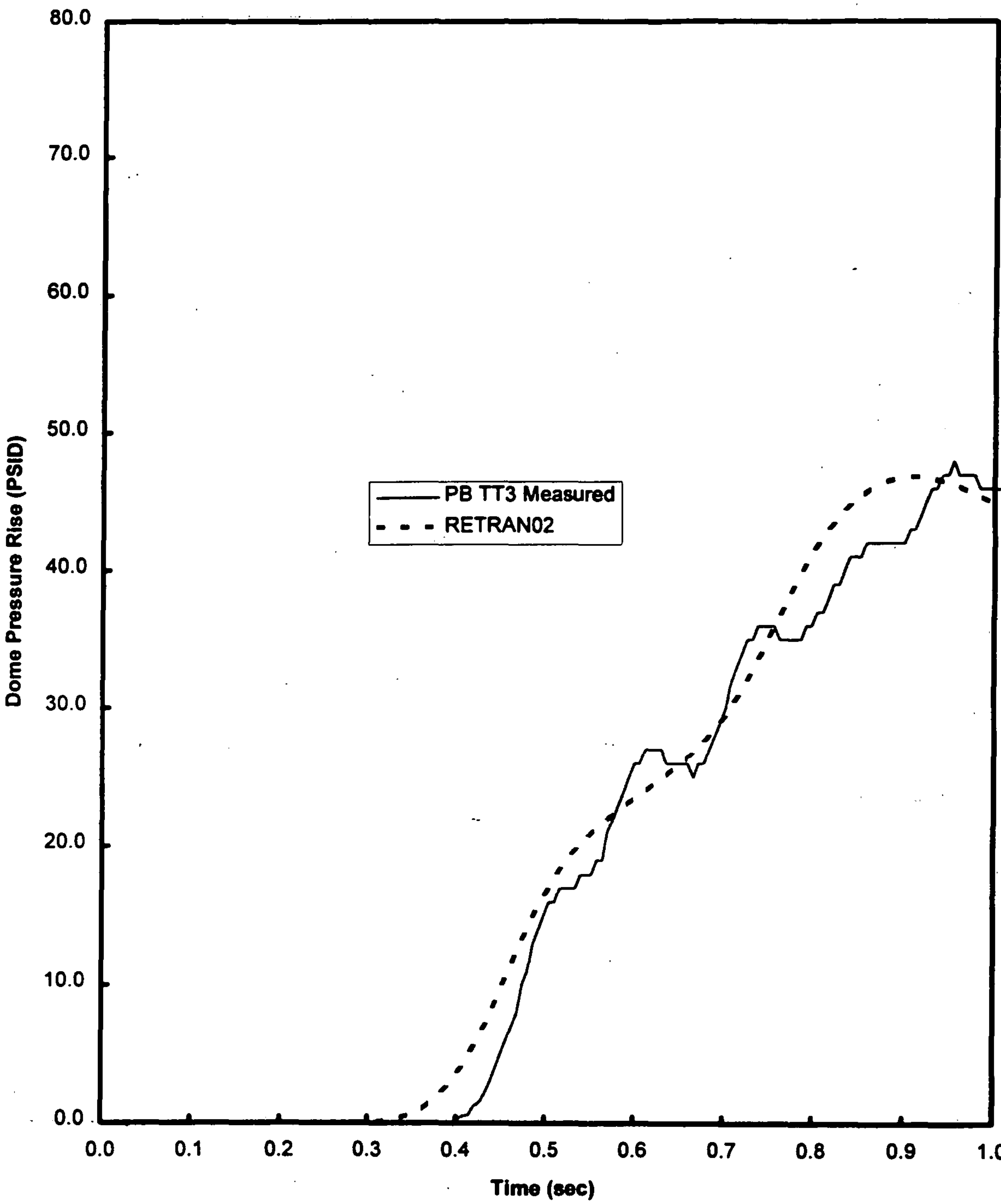

Figure 5.4-12 Turbine Trip Test 3 Reactor Dome Pressure Rise

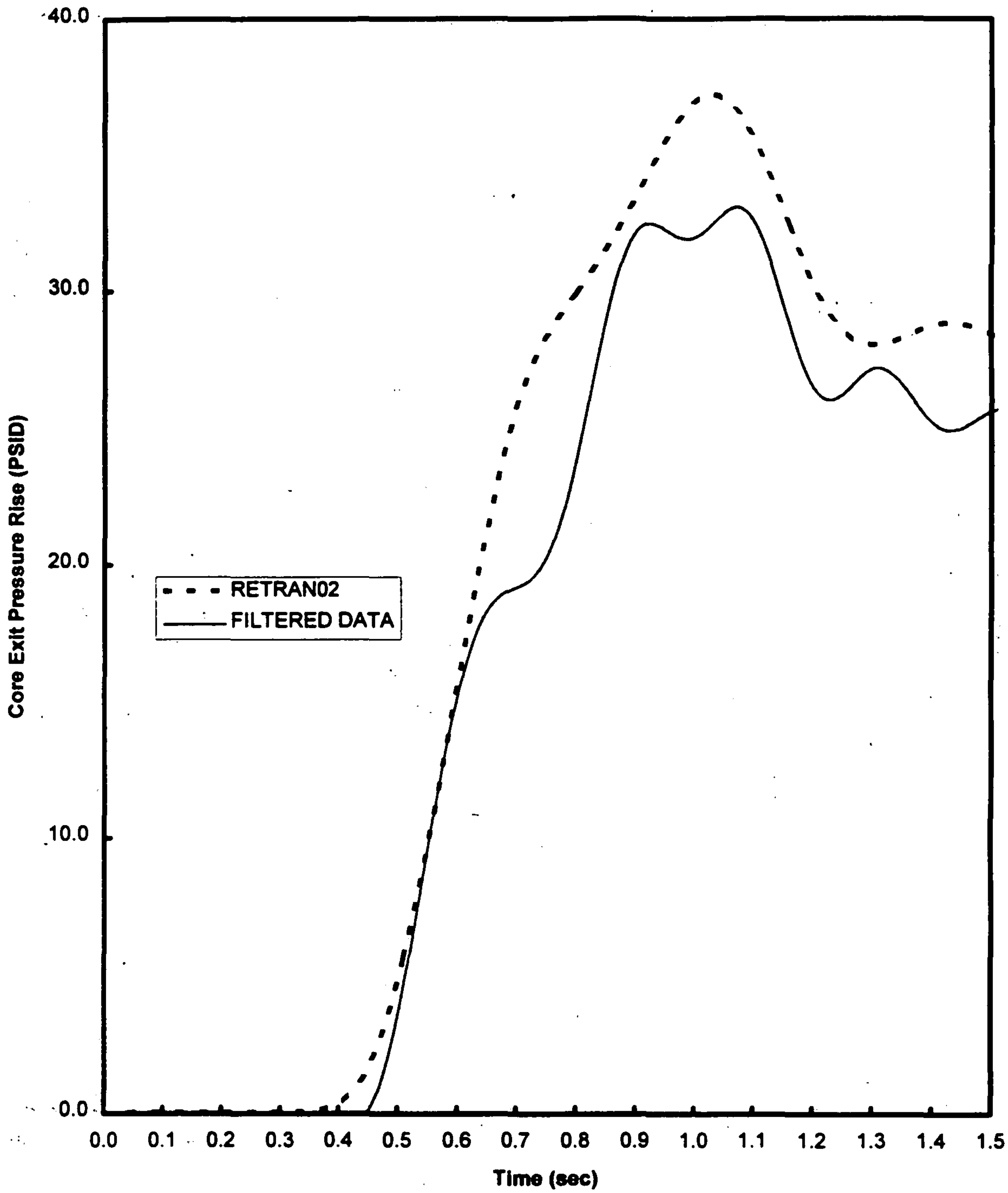

Figure 5.4-13 Turbine Trip Test 1 Core Exit Pressure Rise

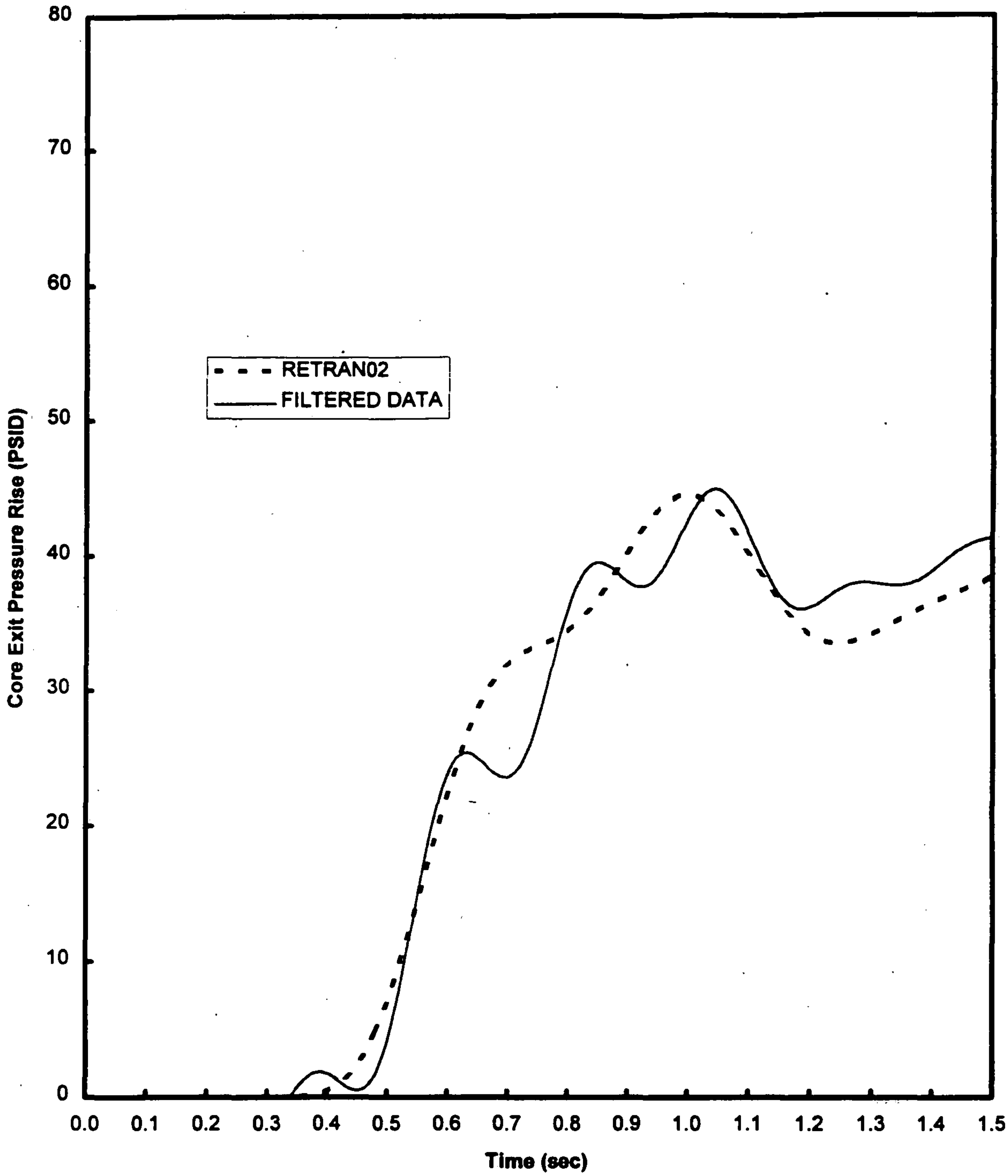

Figure 5.4-14 Turbine Trip Test 2 Core Exit Pressure Rise

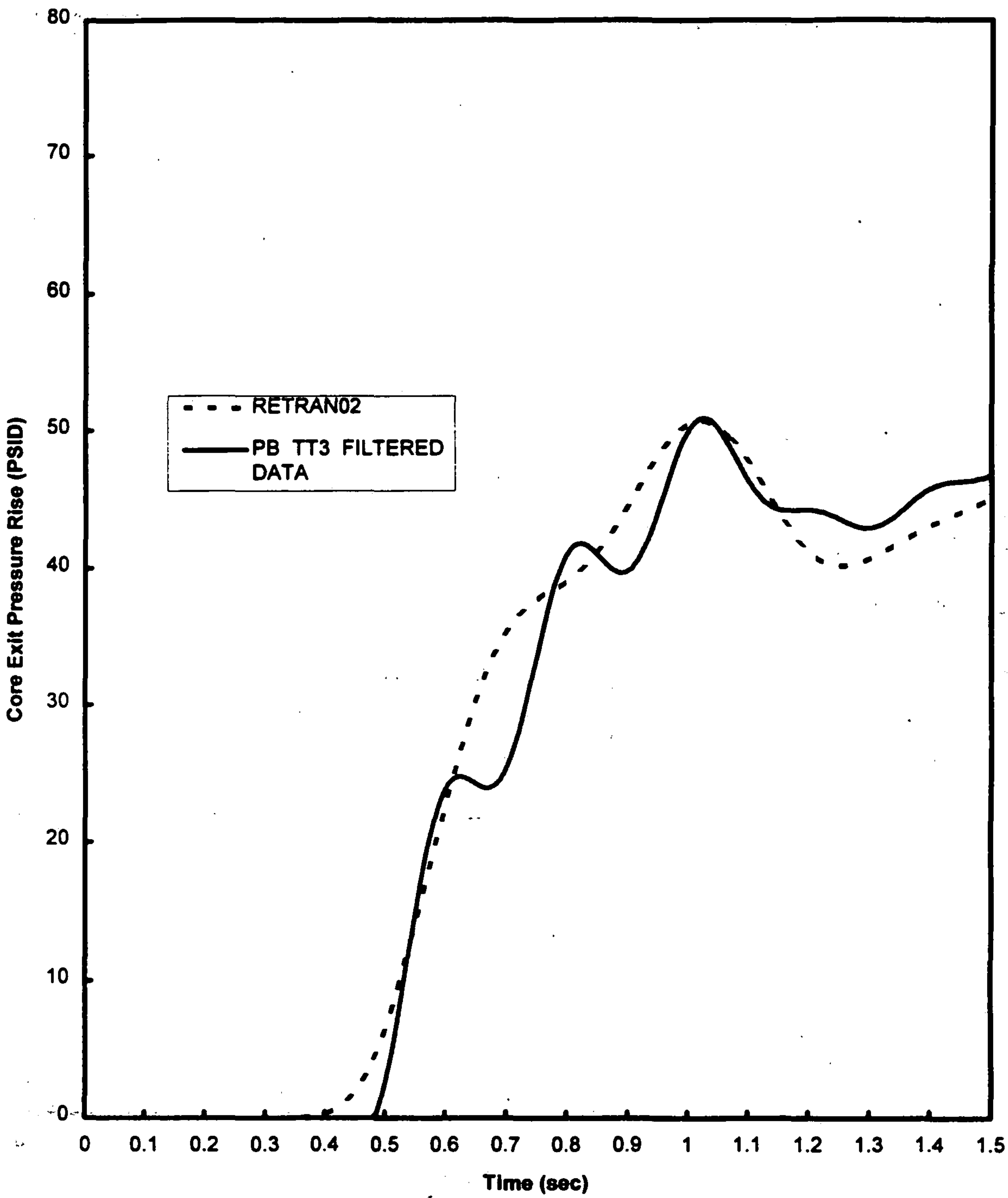

Figure 5.4-15 Turbine Trip Test 3 Core Exit Pressure Rise

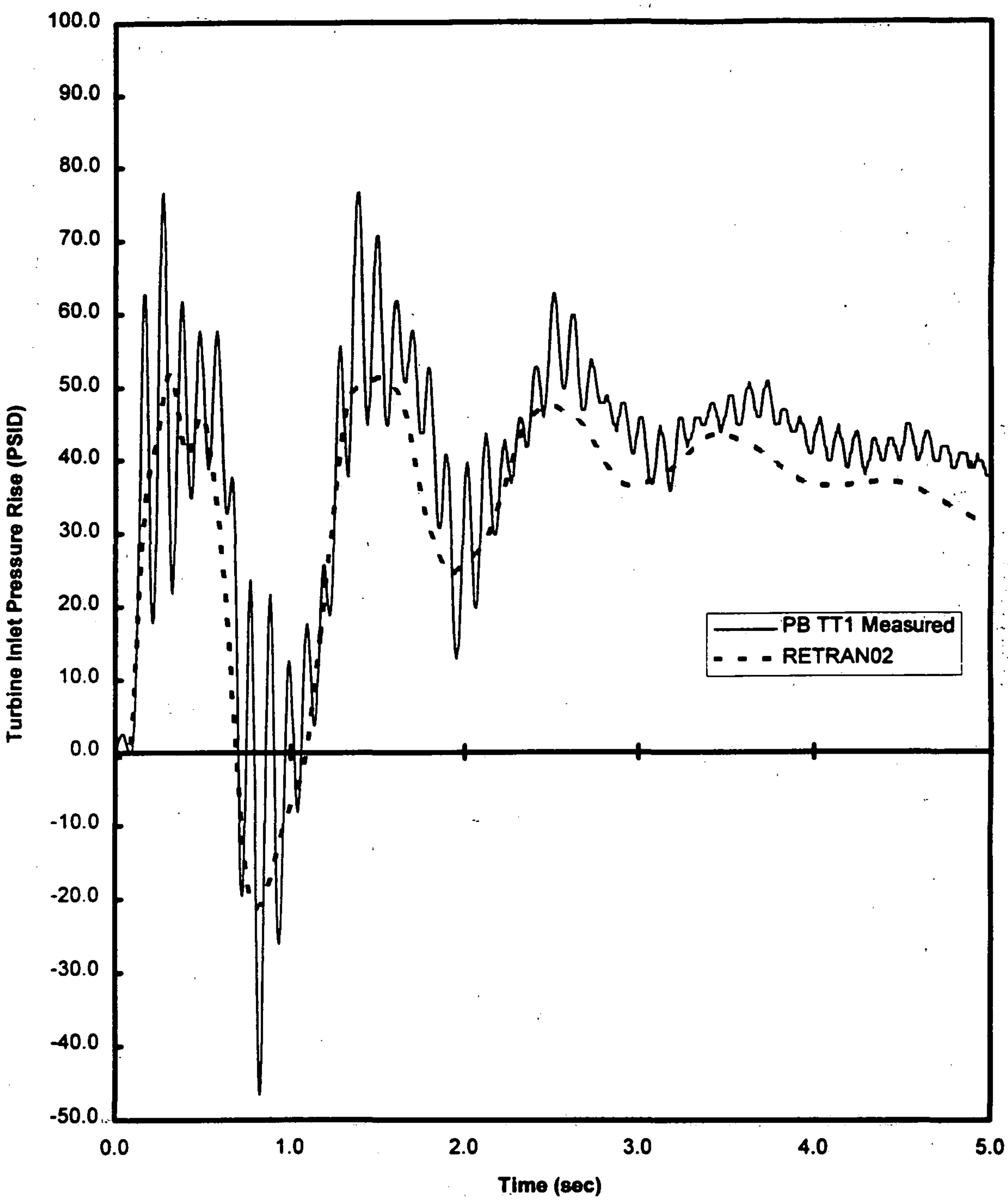

Figure 5.4-16 Turbine Trip Test 1 Turbine Inlet Pressure Rise (BPV Sensitivity)

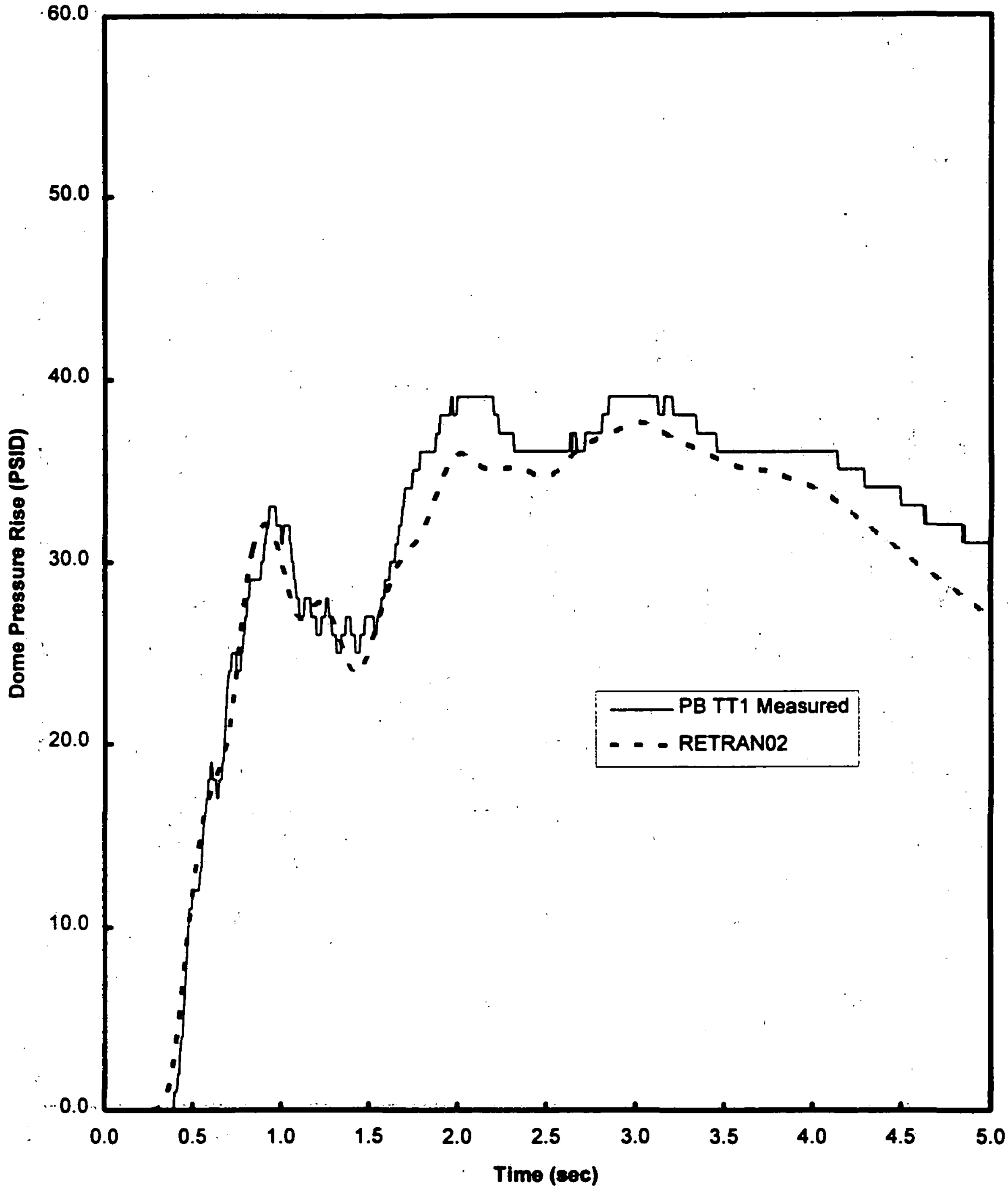

Figure 5.4-17 Turbine Trip Test 1 Reactor Dome Pressure Rise (BPV Sensitivity)

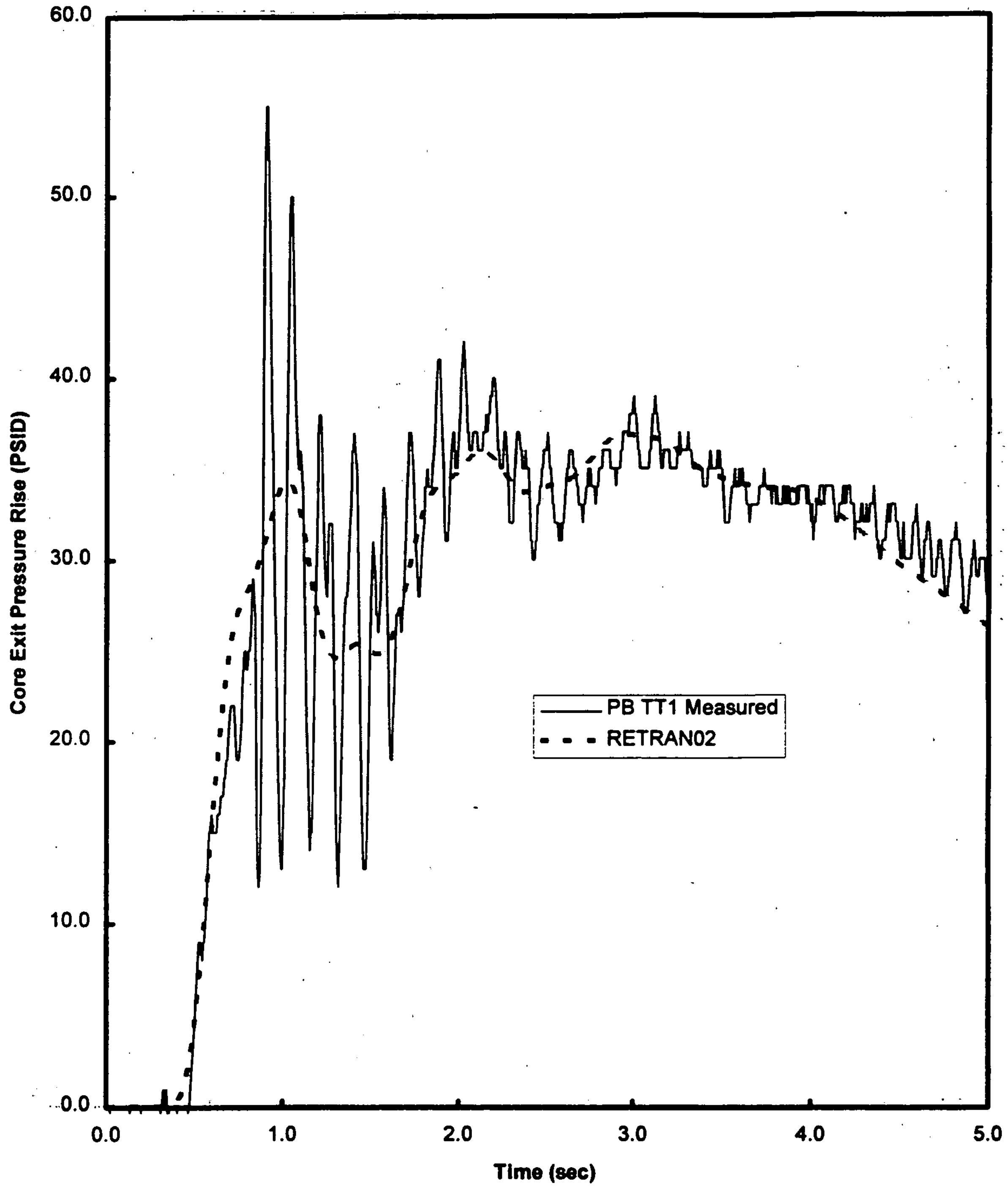

Figure 5.4-18 Turbine Trip Test 1 Core Exit Pressure Rise (BPV Sensitivity)

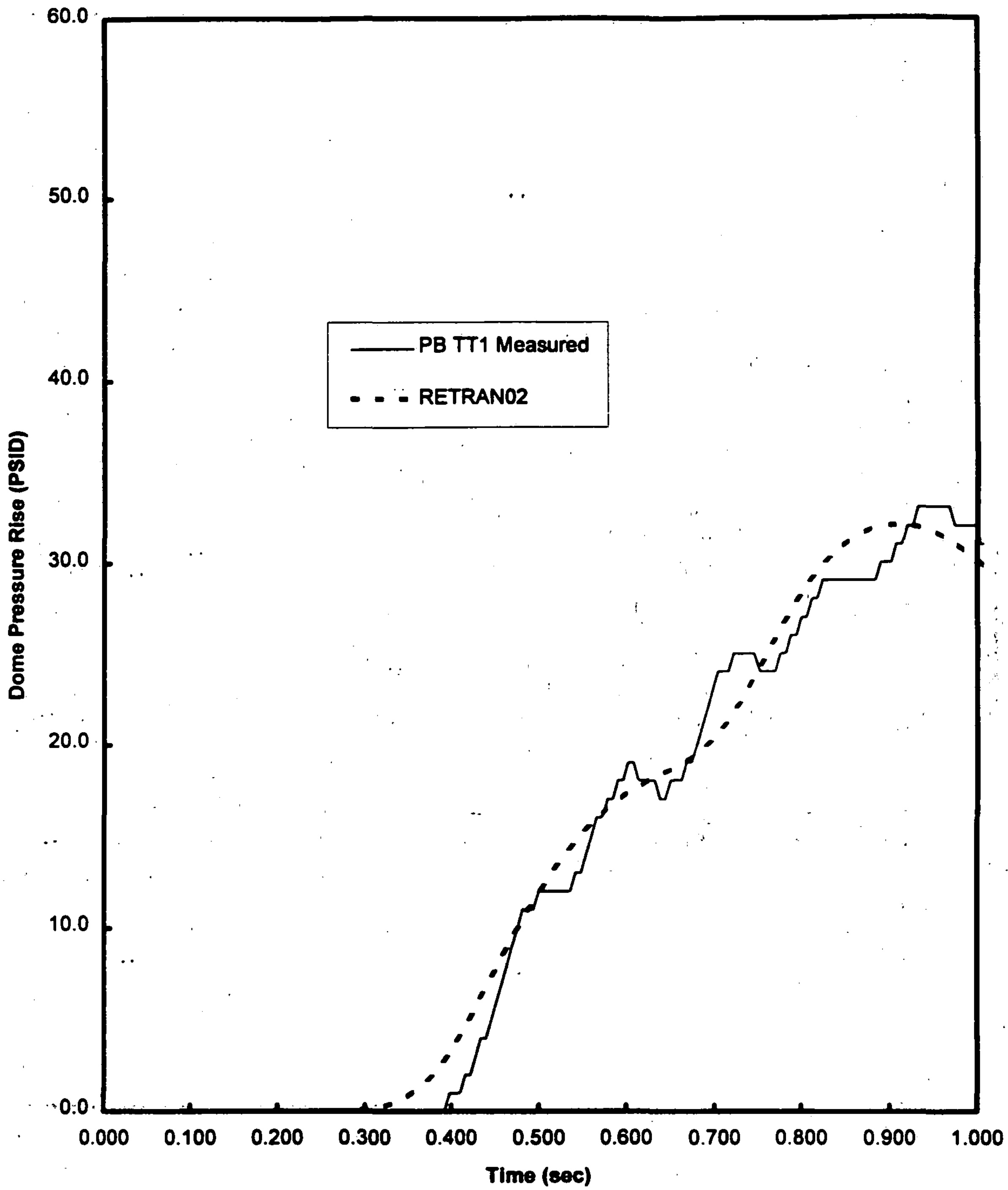

Figure 5.4-19 Turbine Trip Test 1 Dome Pressure Rise (BPV Sensitivity)

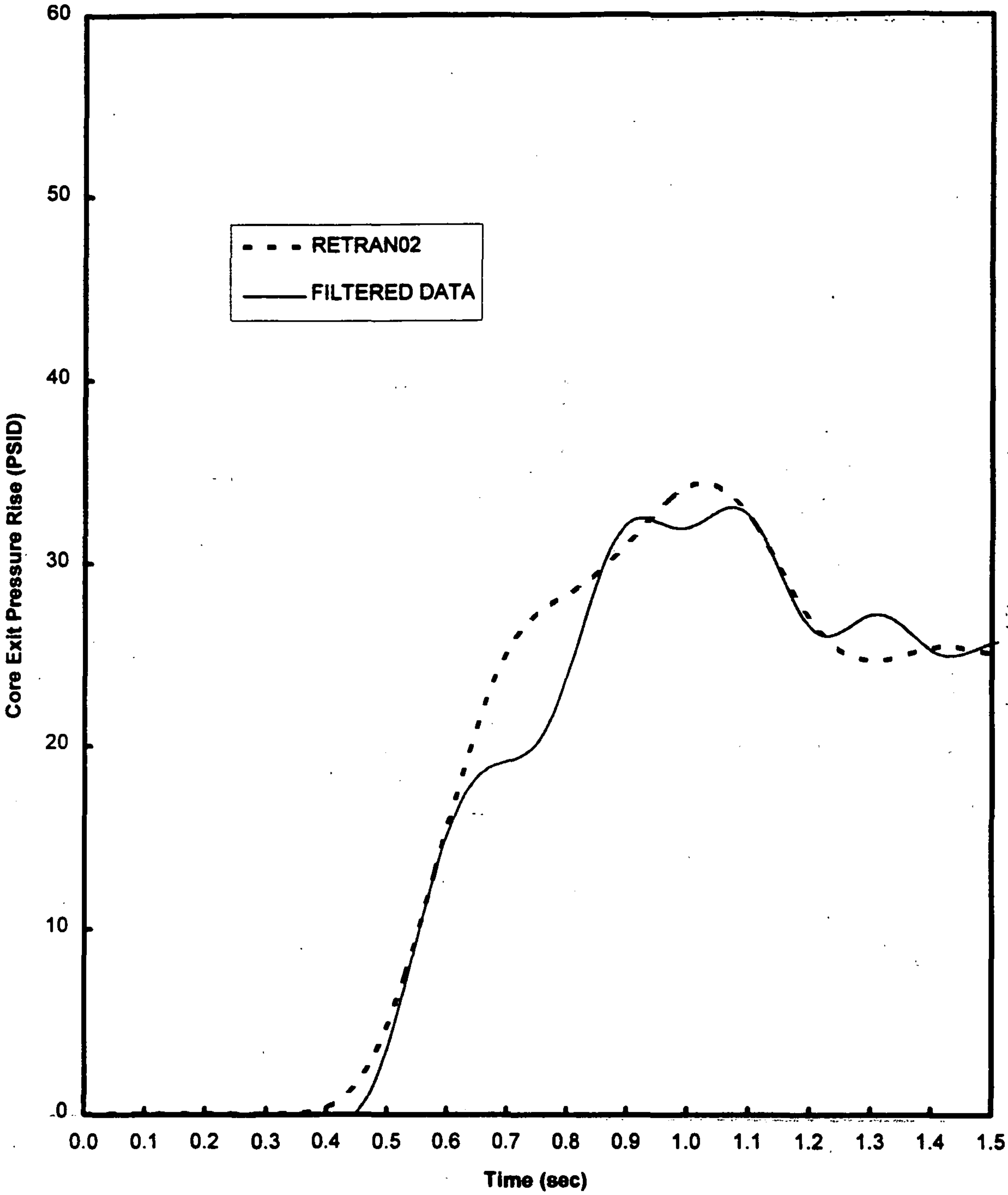

Figure 5.4-20 Turbine Trip Test 1 Core Exit Pressure (BPV Sensitivity)

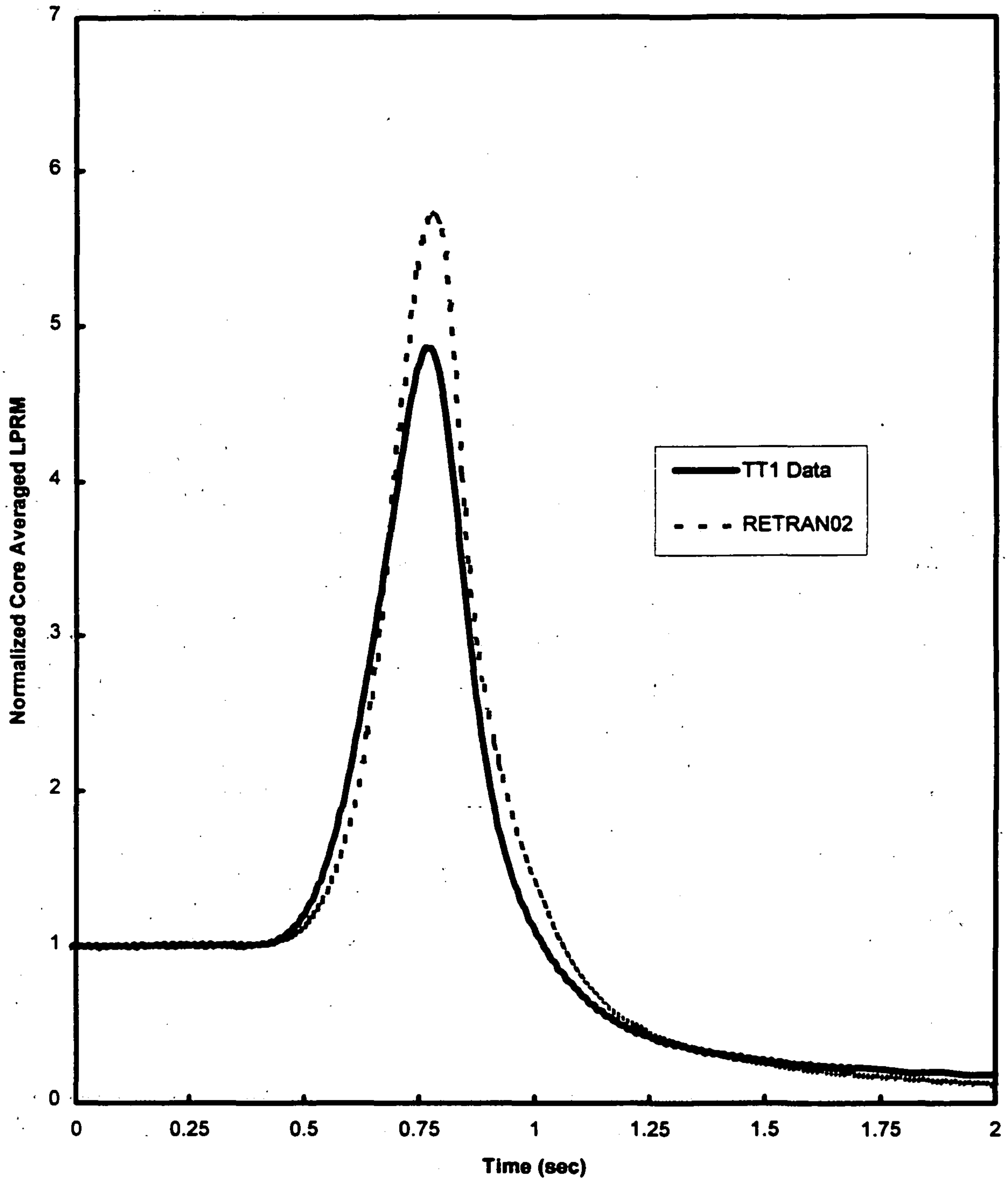

Figure 5.4-21 Turbine Trip Test 1 Normalized Core Averaged LPRM

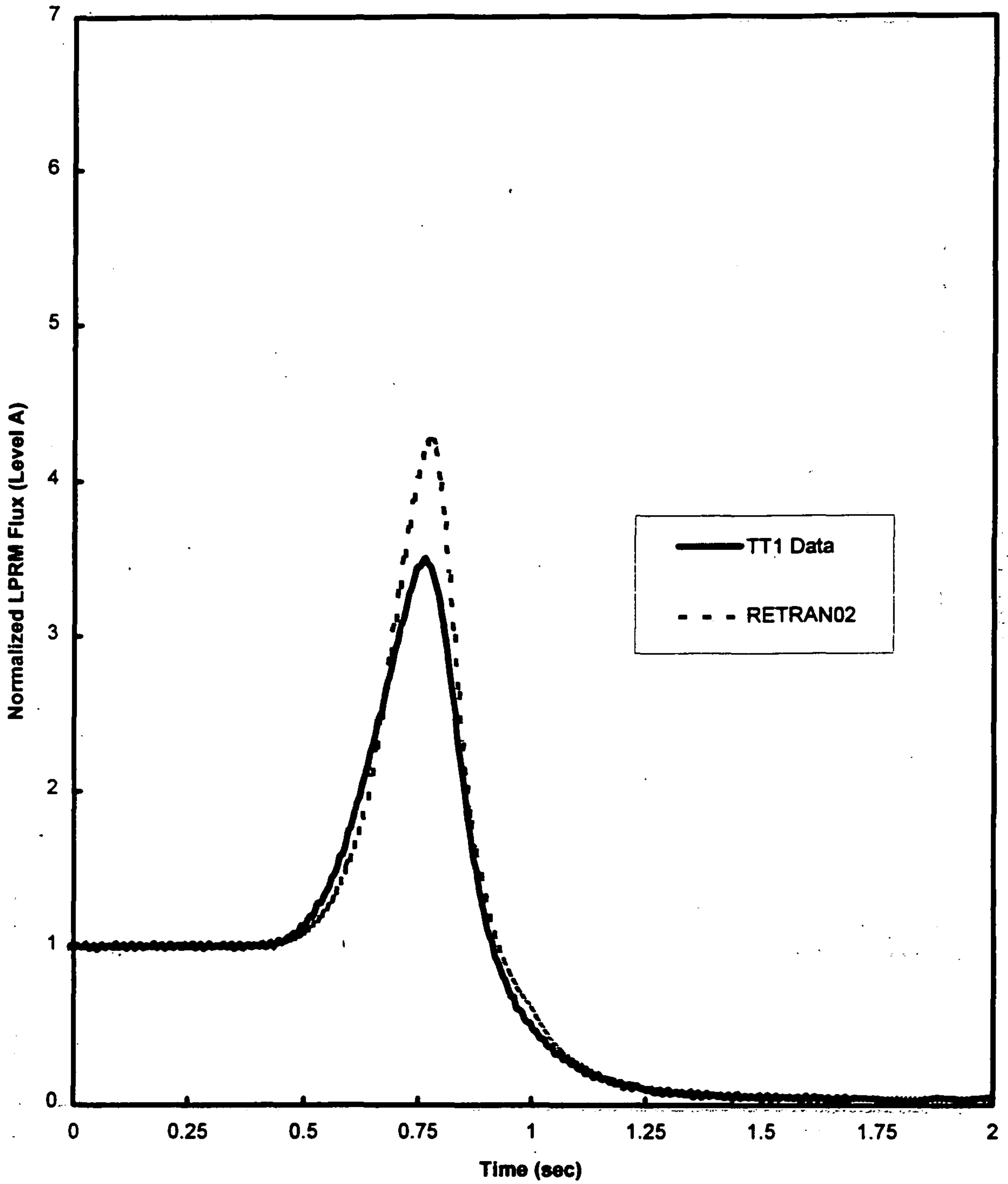

Figure 5.4-22 Turbine Trip Test 1 LPRM Level A

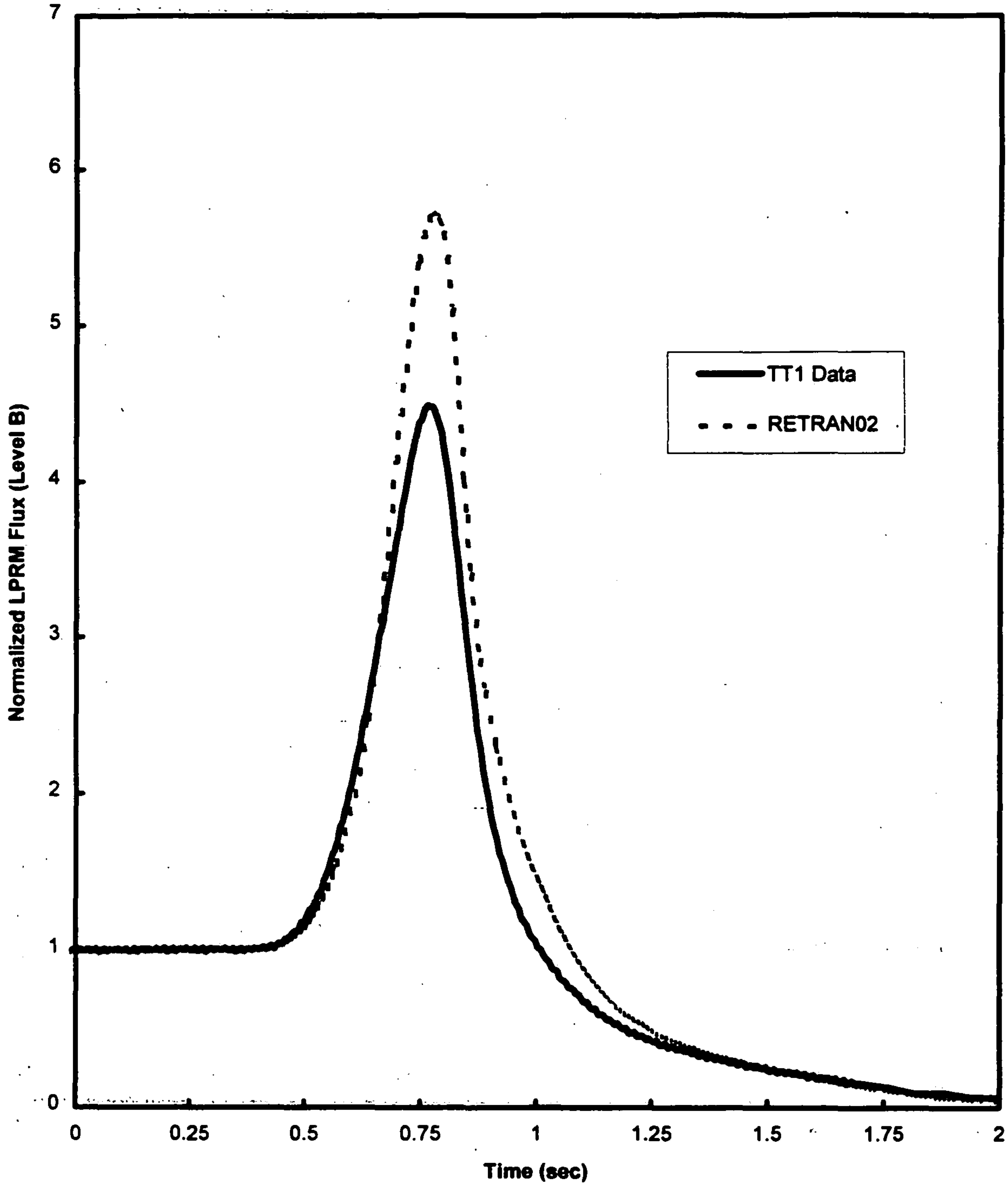

Figure 5.4-23 Turbine Trip Test 1 LPRM Level B

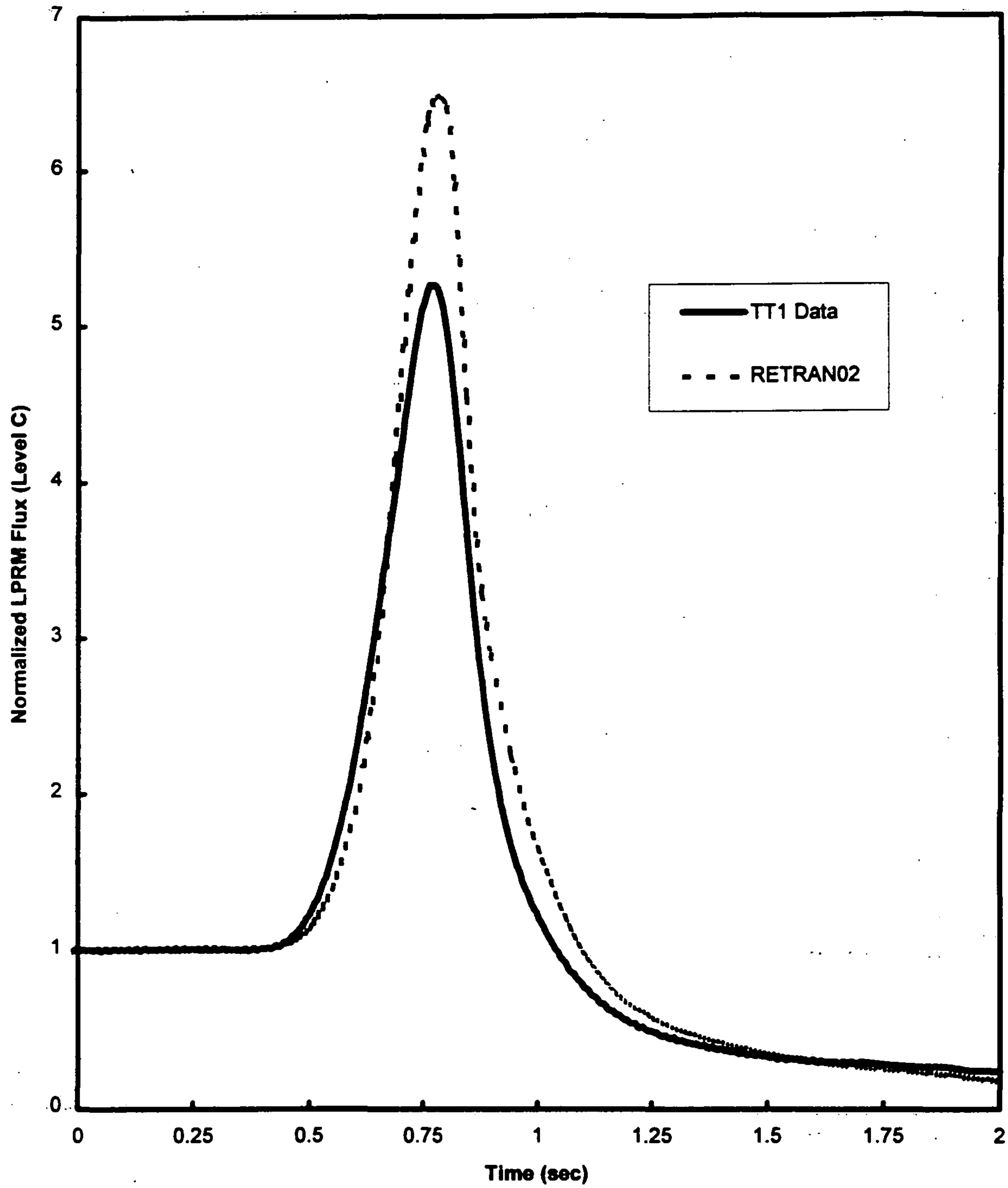

Figure 5.4-24 Turbine Trip Test 1 LPRM Level C

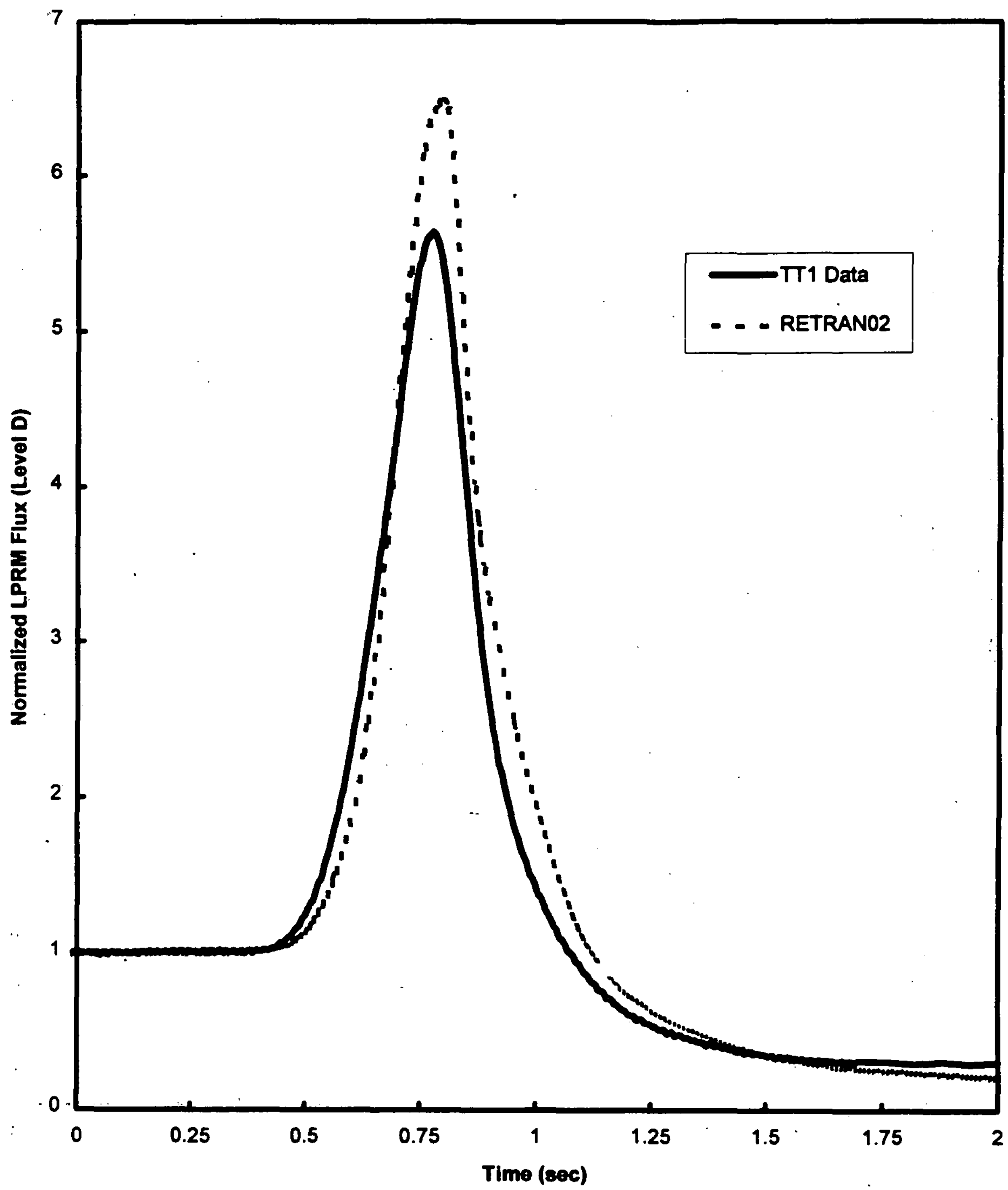

Figure 5.4-25 Turbine Trip Test 1 LPRM Level D

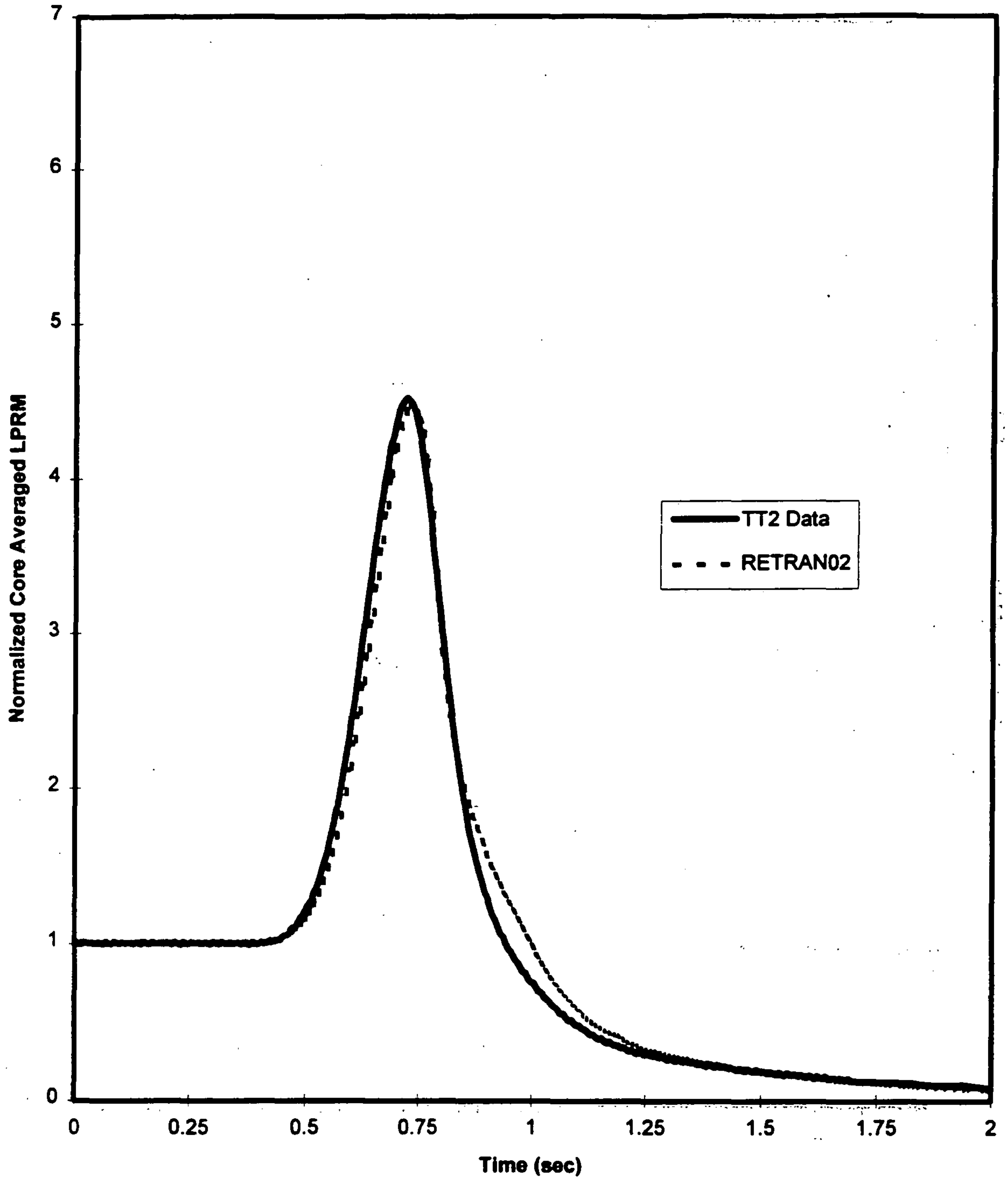

Figure 5.4-26 Turbine Trip Test 2 Normalized Core Averaged LPRM

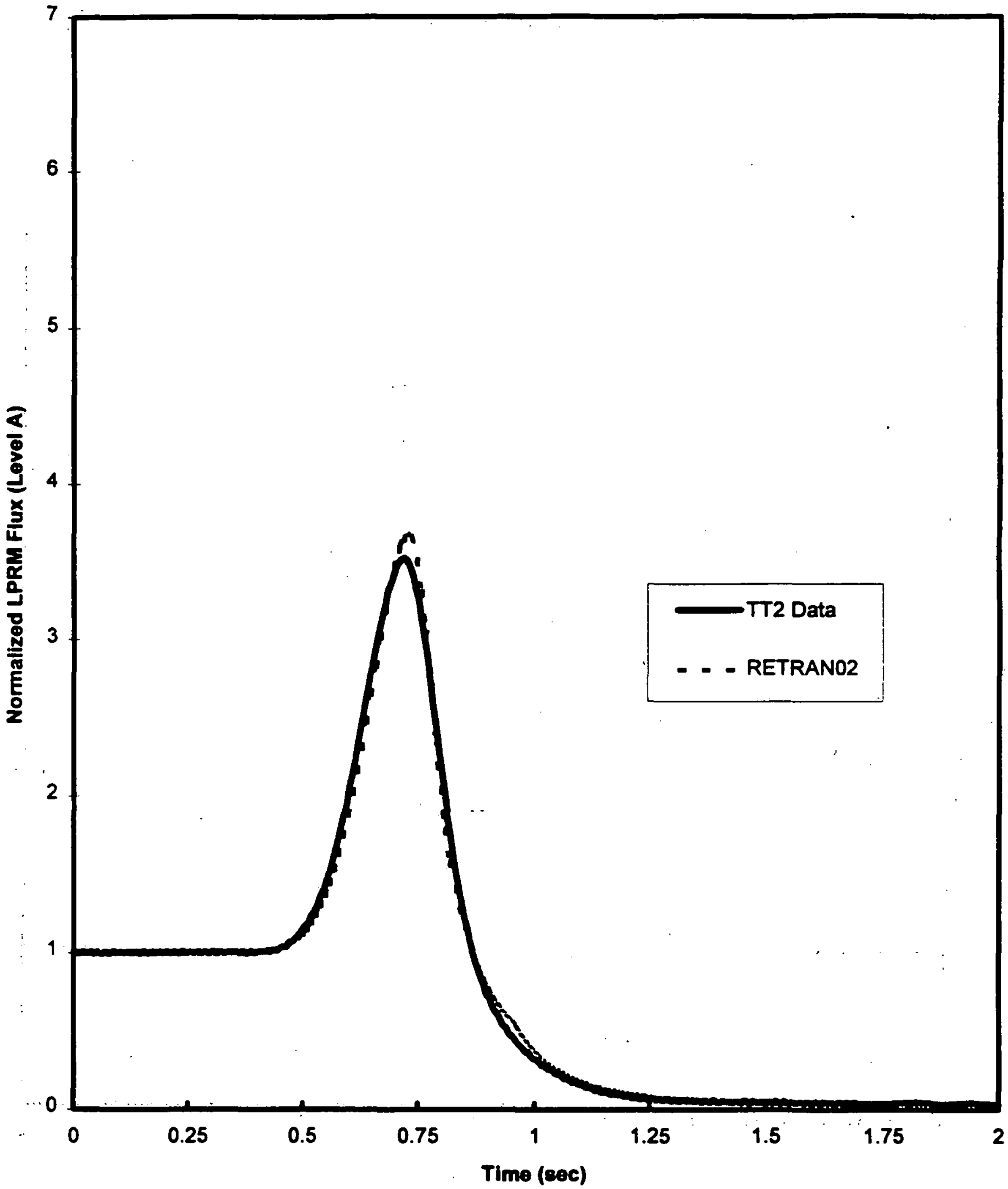

Figure 5.4-27 Turbine Trip Test 2 LPRM Level A

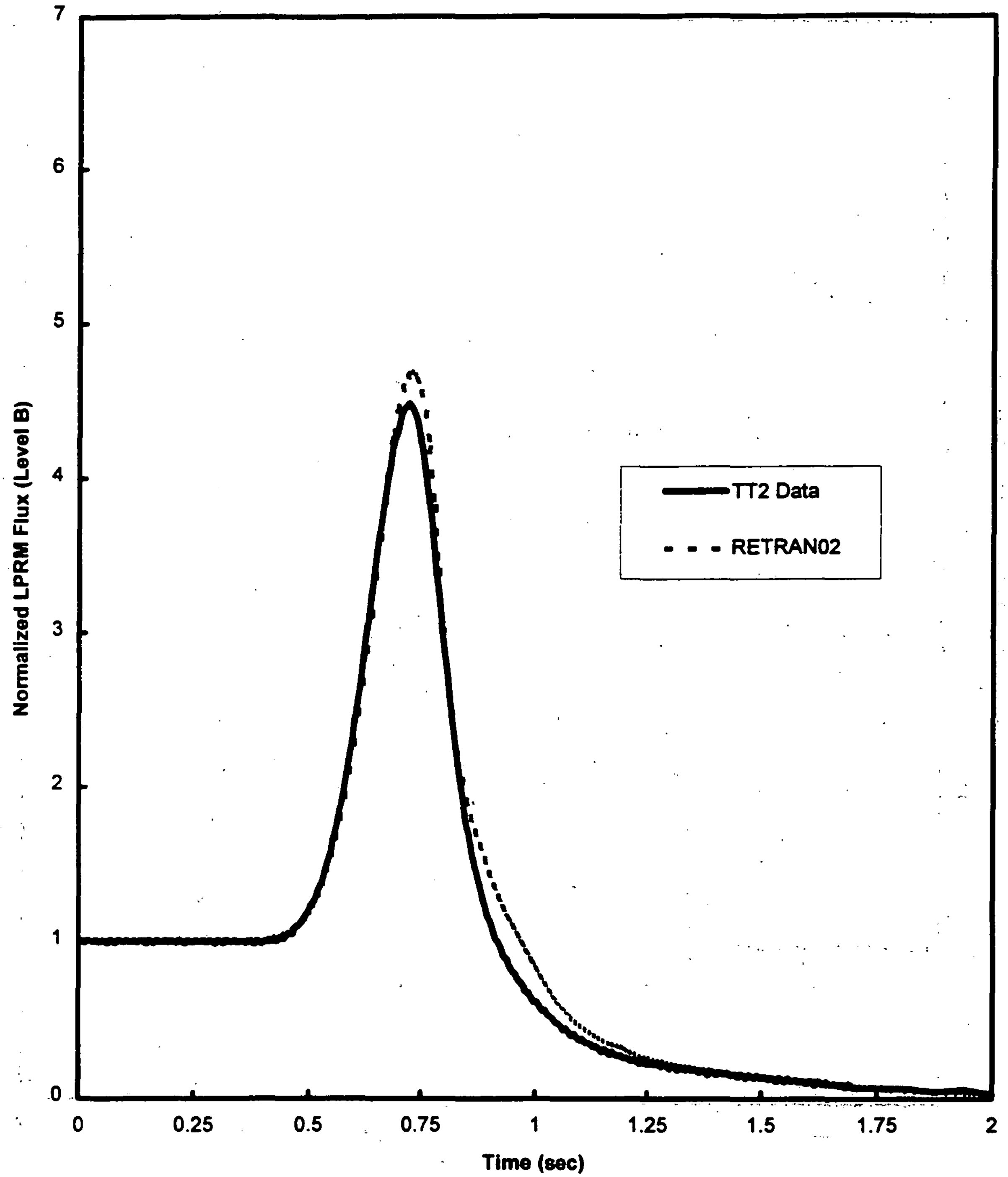

Figure 5.4-28 Turbine Trip Test 2 LPRM Level B

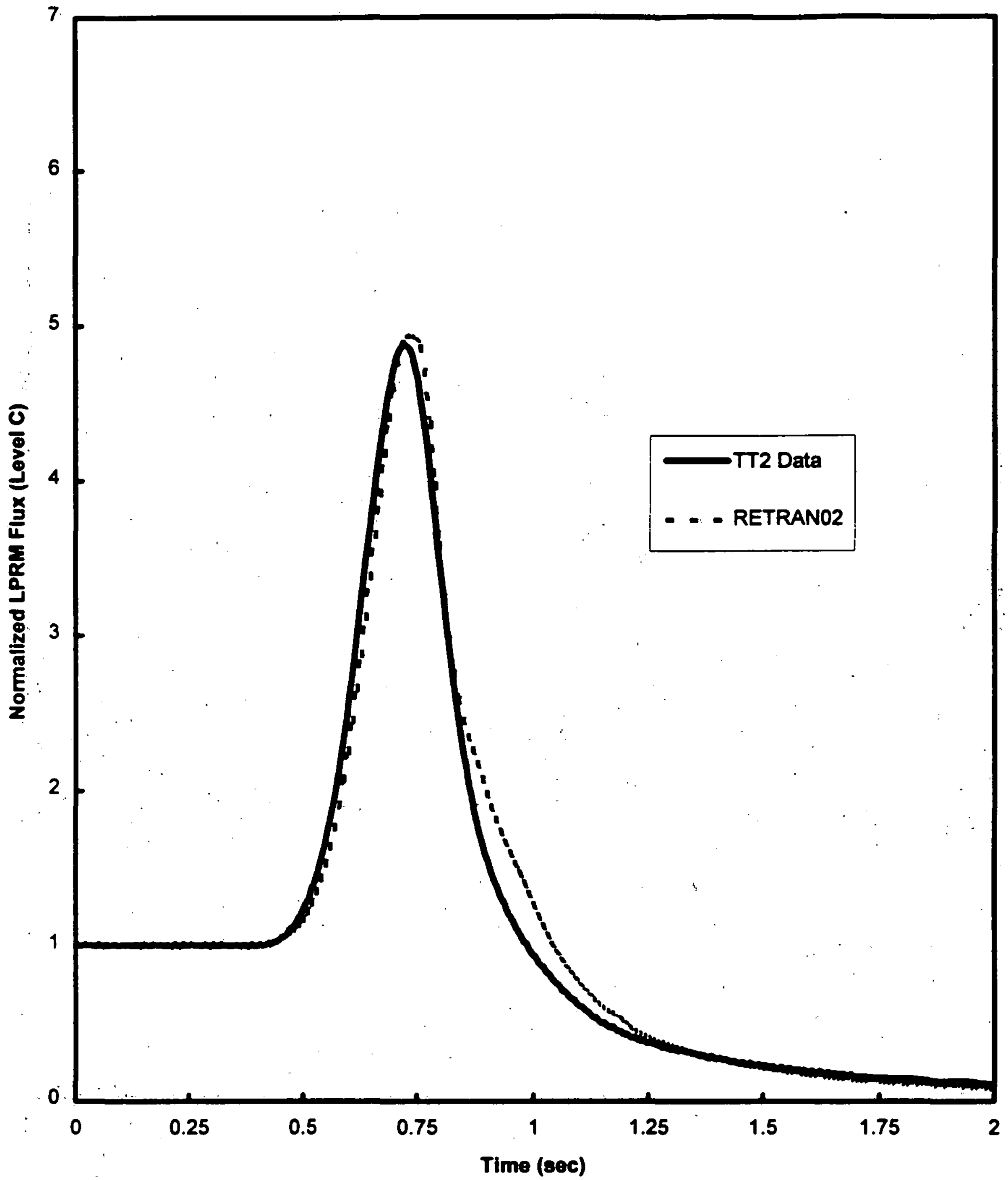

Figure 5.4-29 Turbine Trip Test 2 LPRM Level C

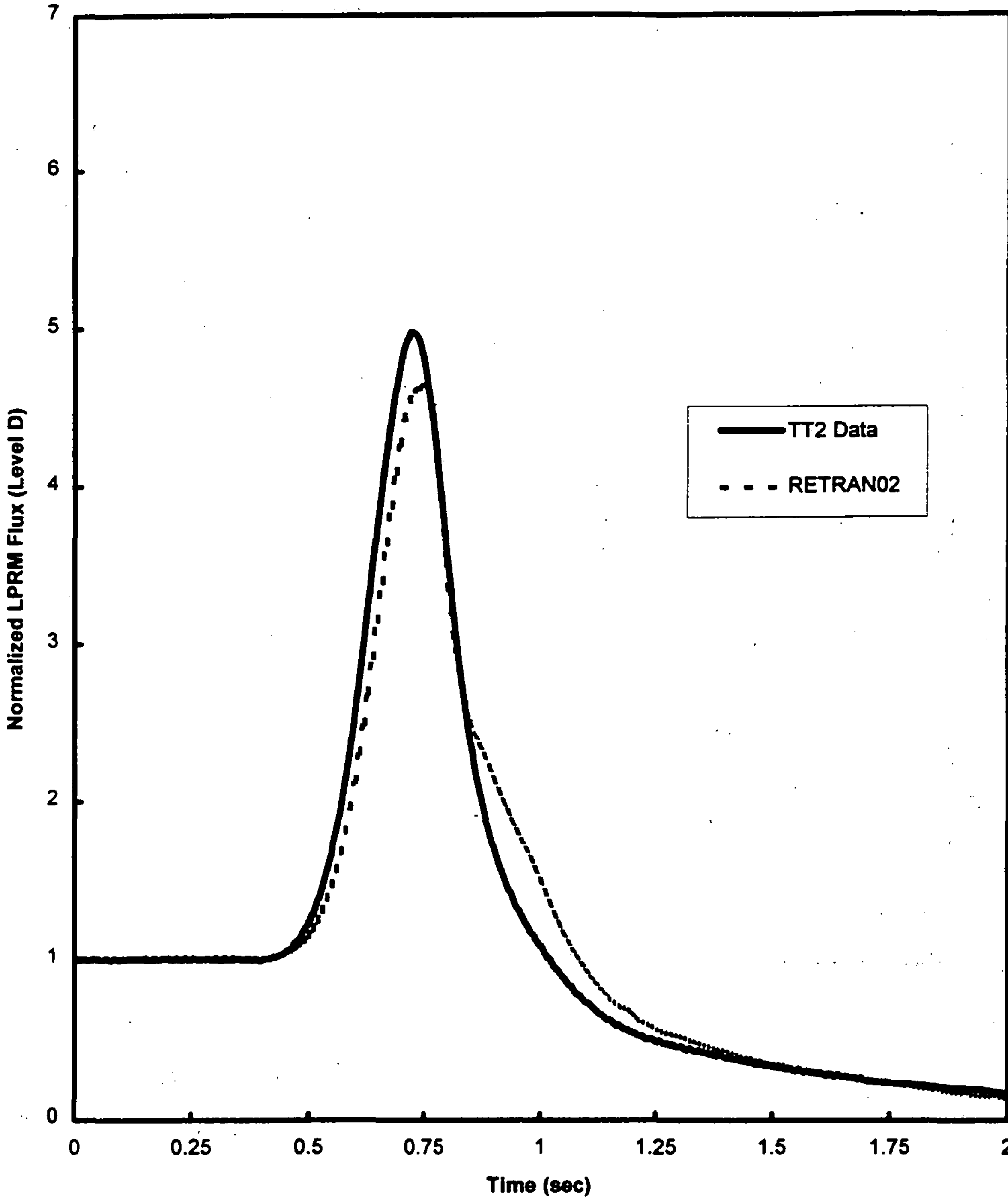

Figure 5.4-30 Turbine Trip Test 2 LPRM Level D

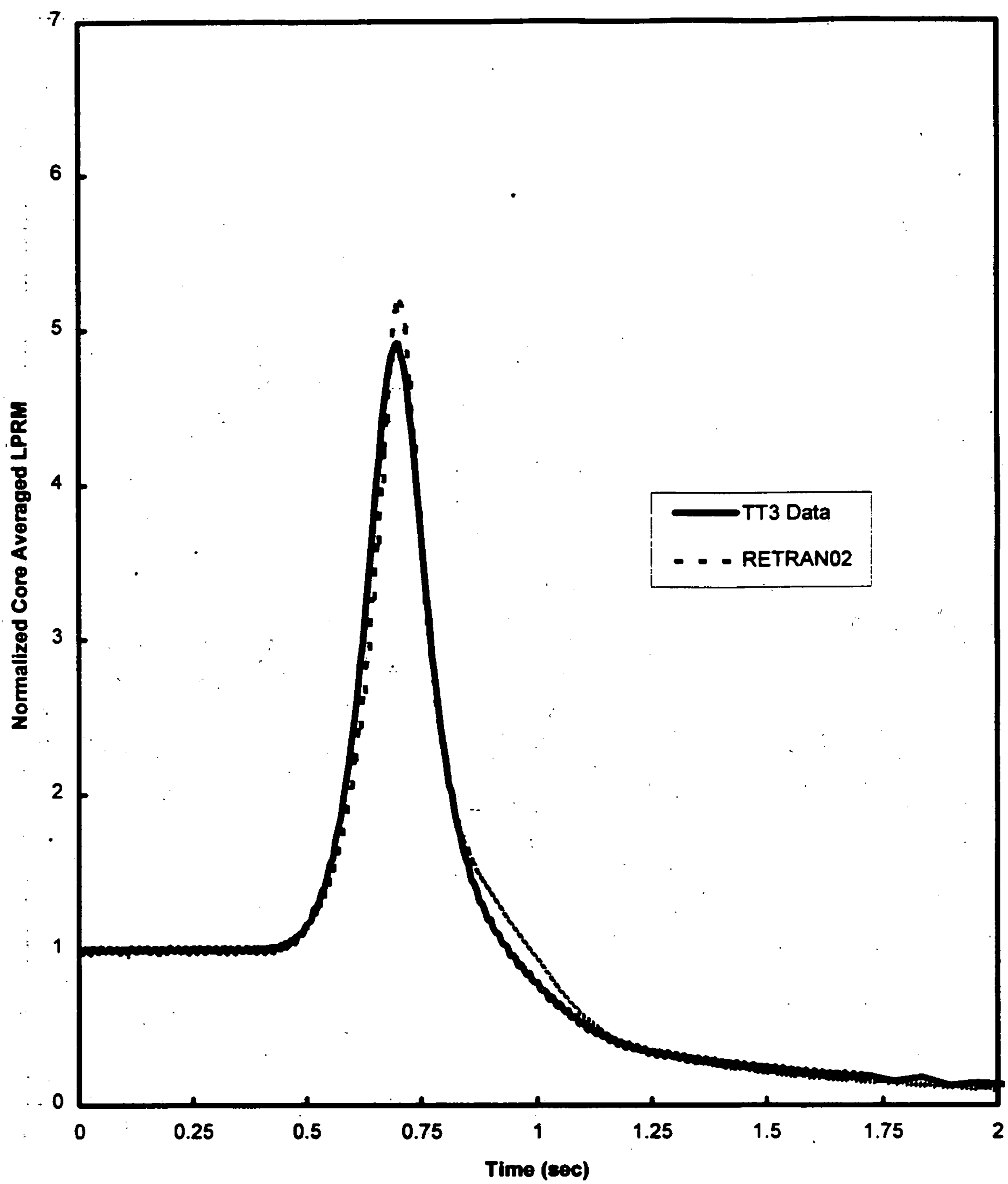

Figure 5.4-31 Turbine Trip Test 3 Normalized Core Averaged LPRM

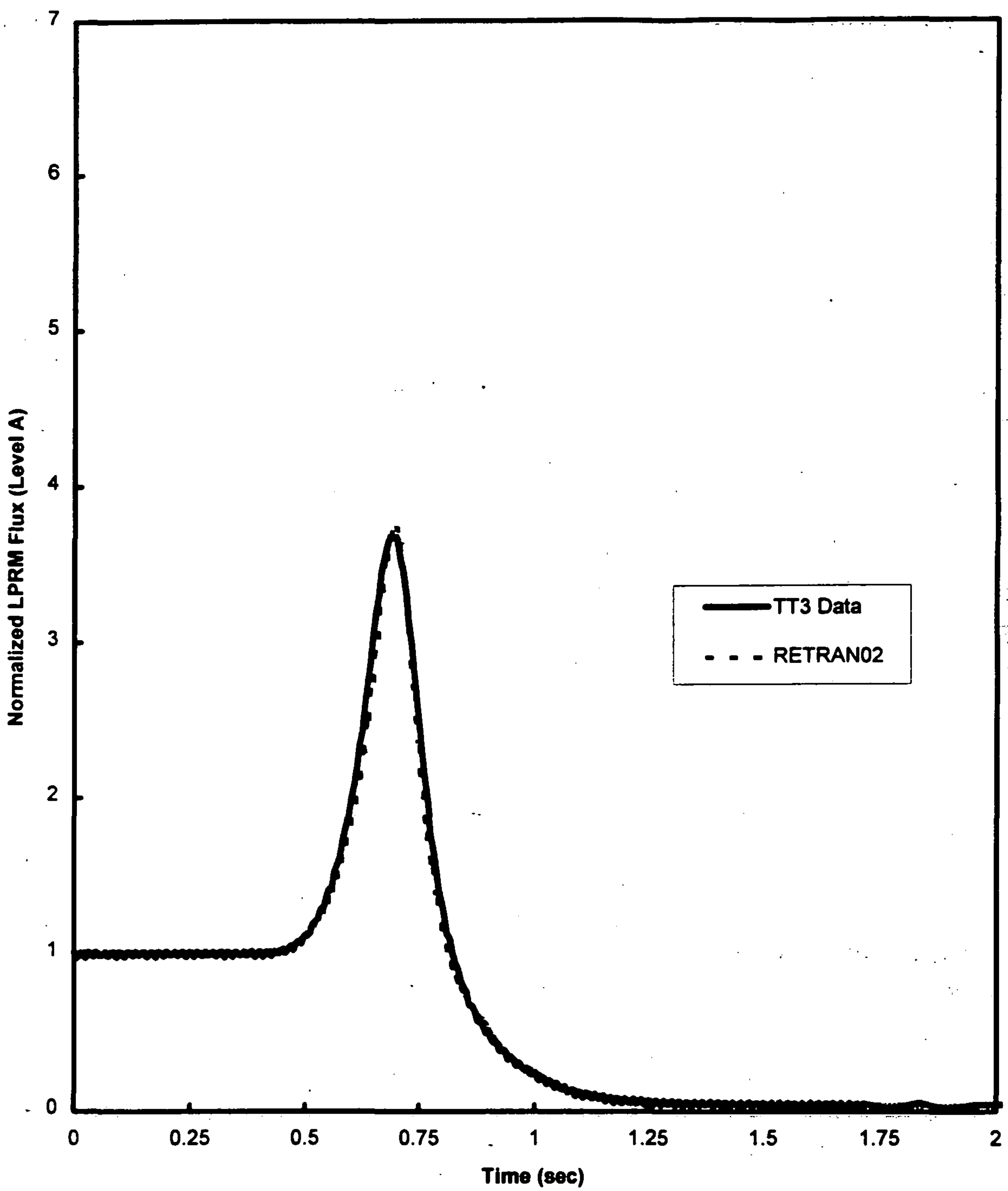

Figure 5.4-32 Turbine Trip Test 3 LPRM Level A

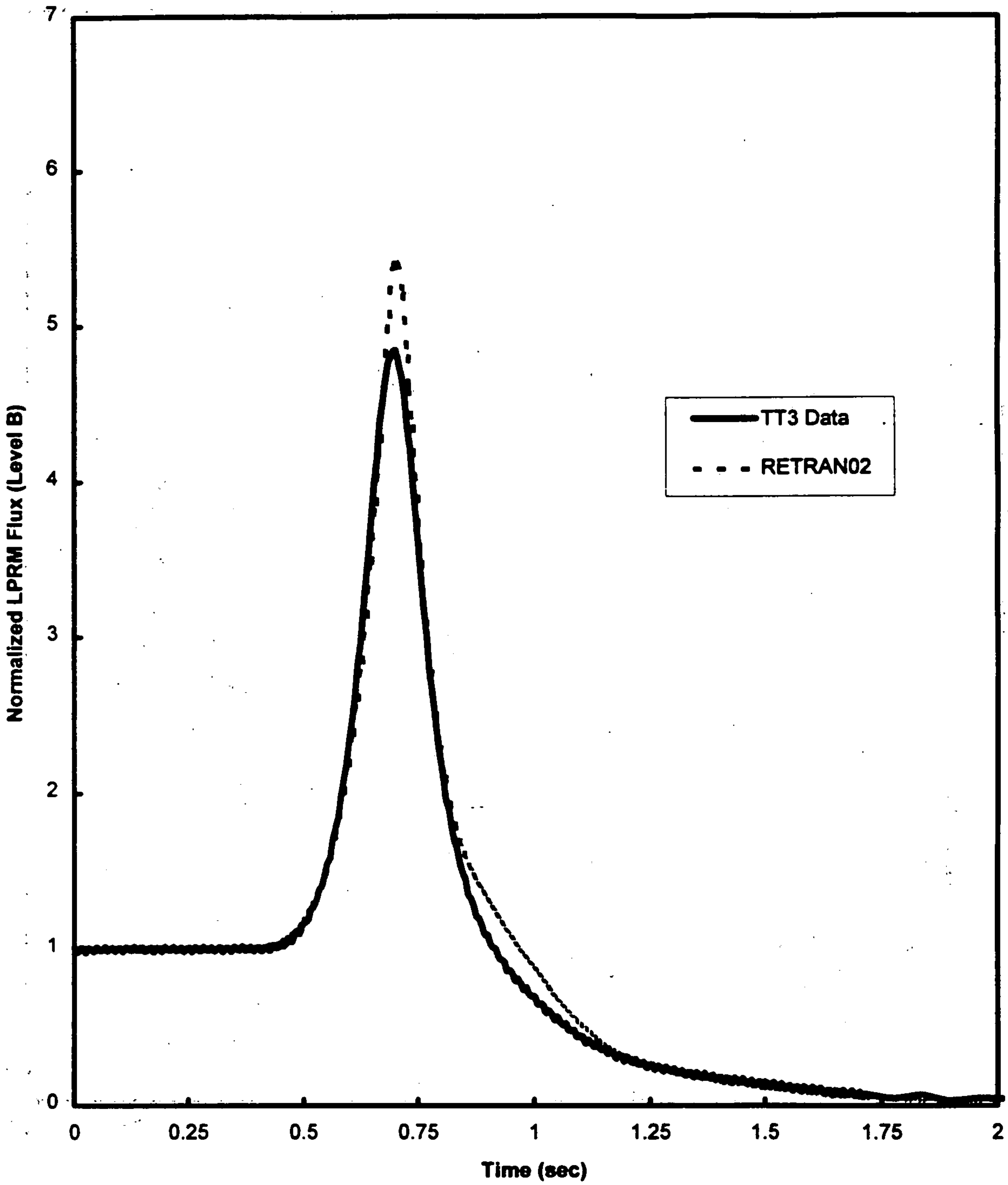

Figure 5.4-33 Turbine Trip Test 3 LPRM Level B

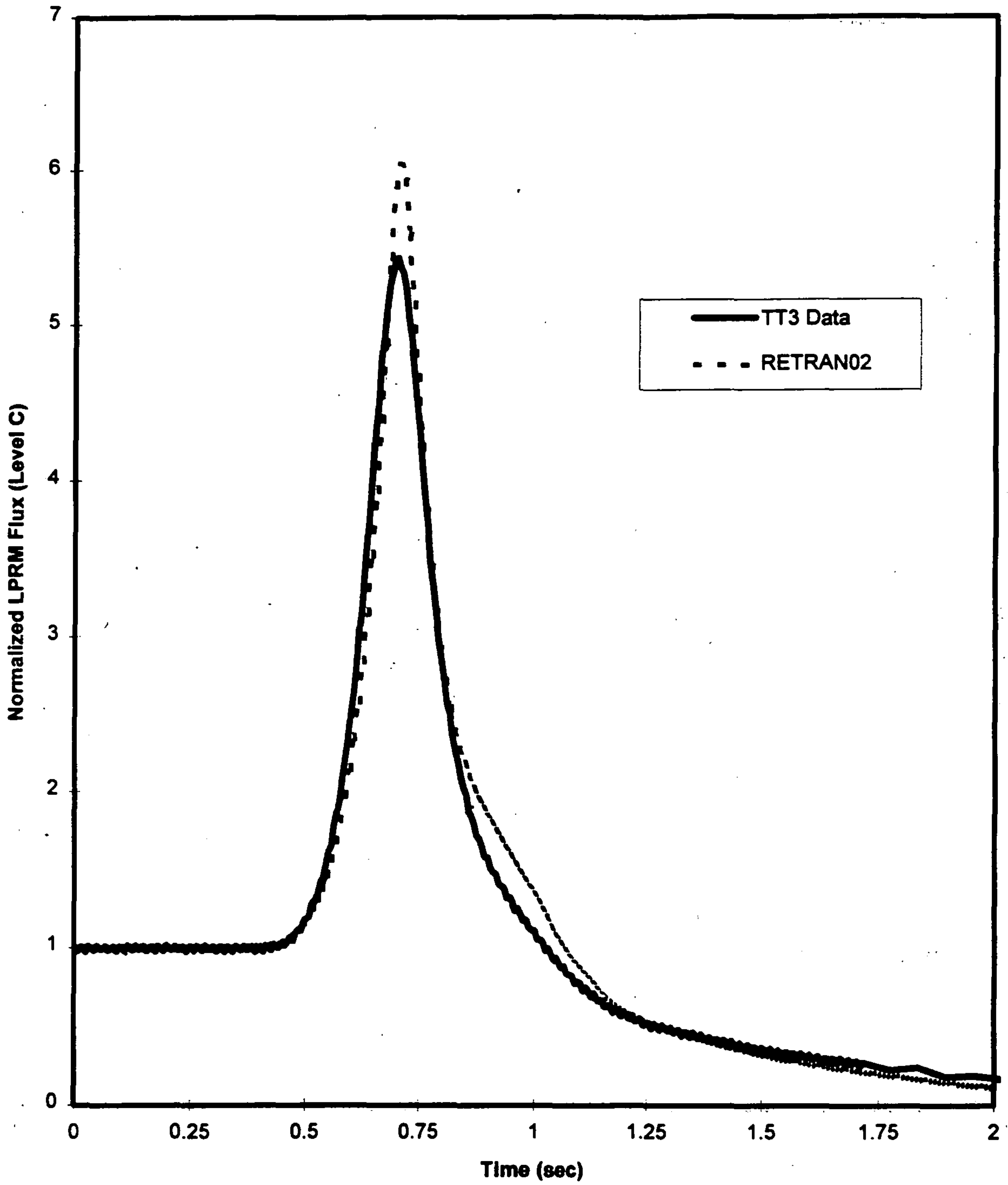

Figure 5.4-34 Turbine Trip Test 3 LPRM Level C

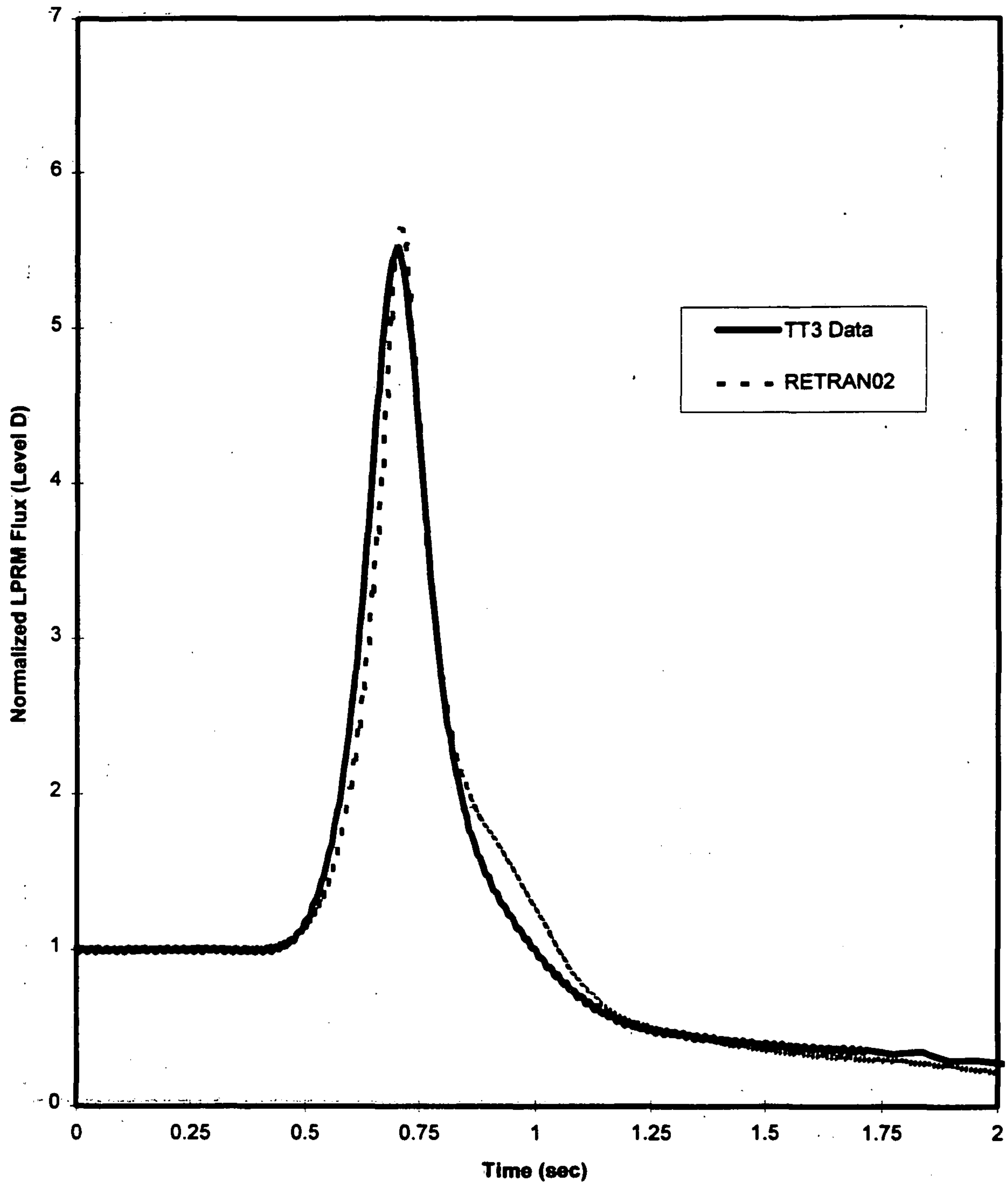

Figure 5.4-35 Turbine Trip Test 3 LPRM Level D

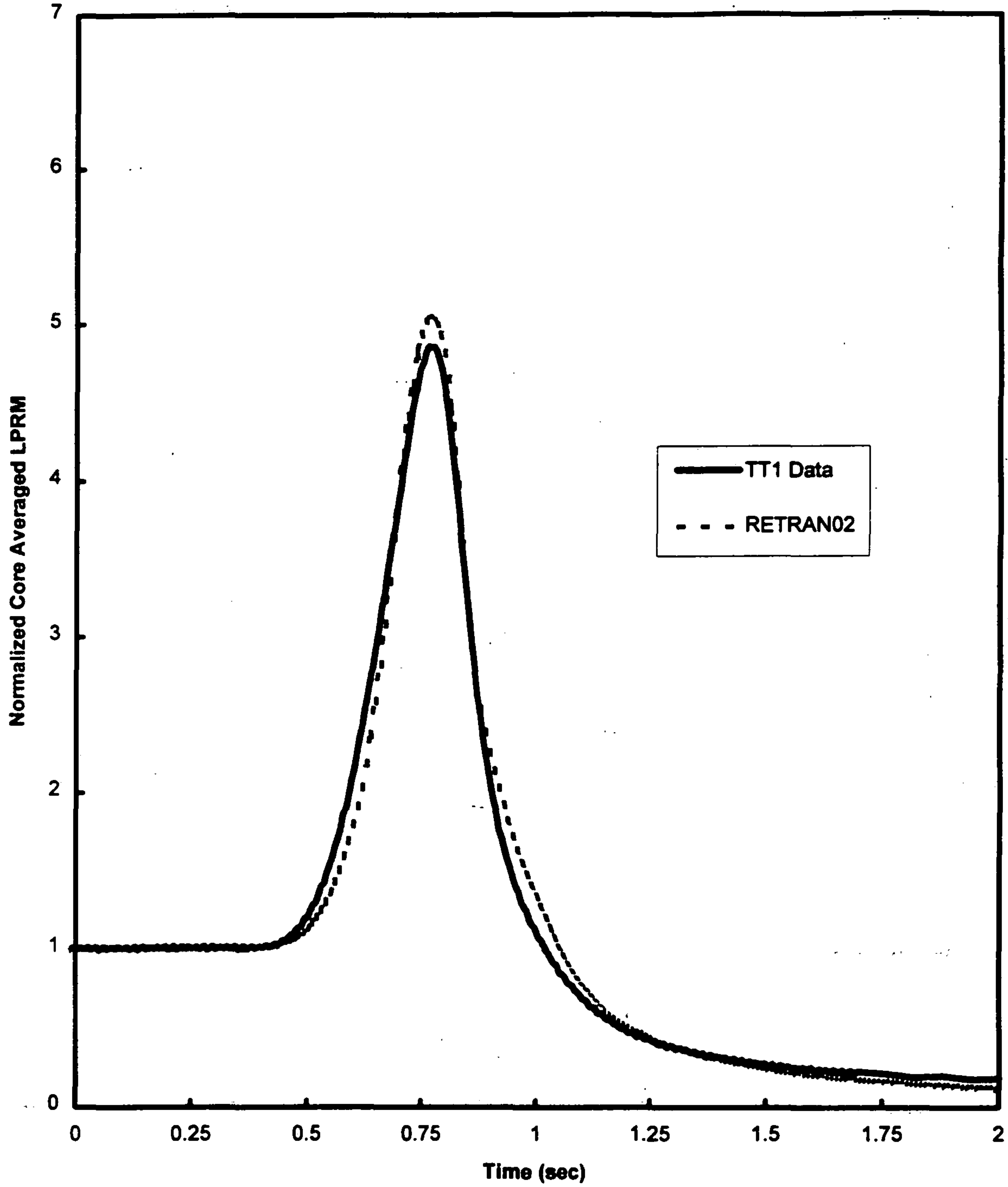

Figure 5.4-36 Turbine Trip Test 1 Norm. Core Averaged LPRM (BPV Sensitivity)

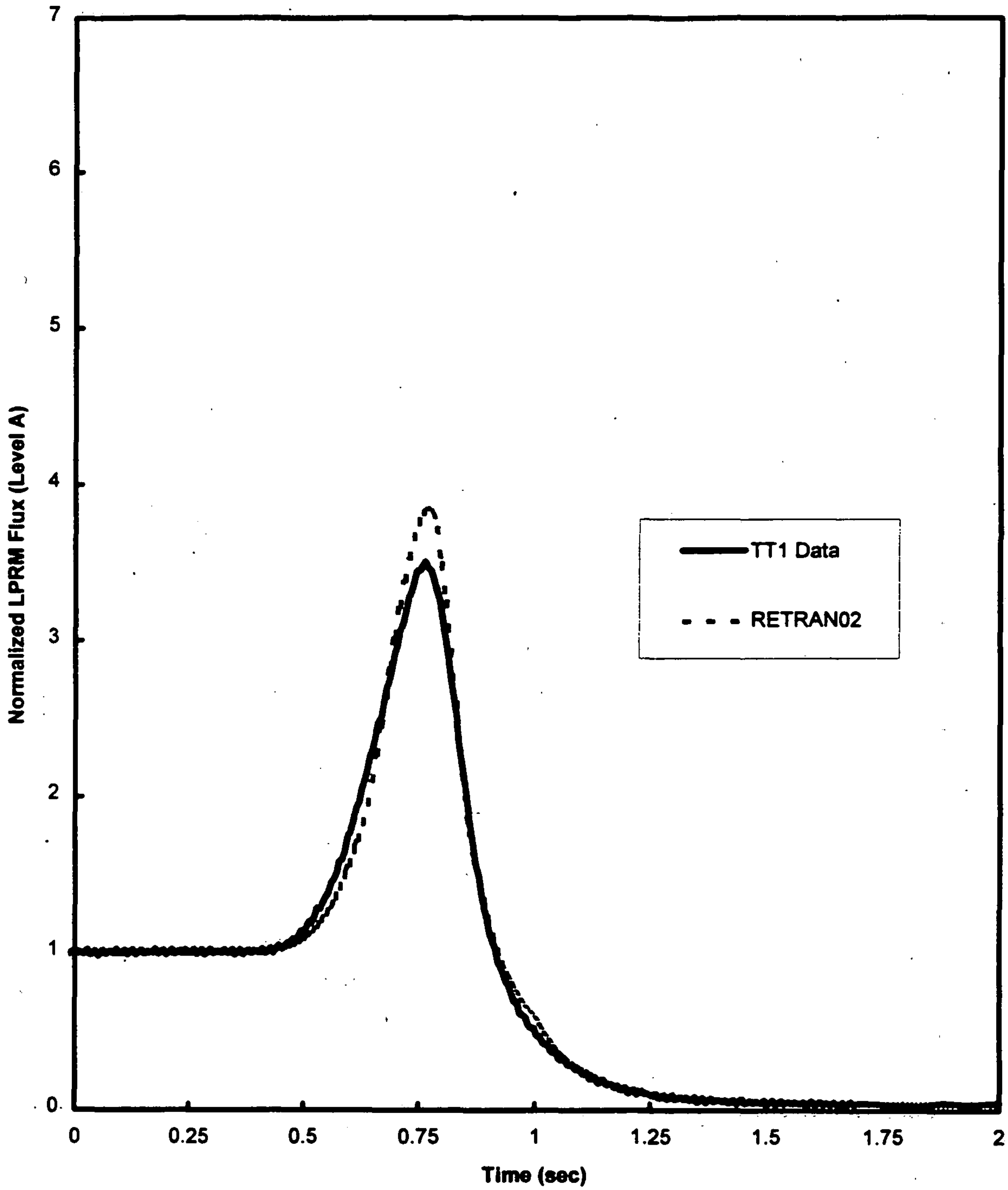

Figure 5.4-37 Turbine Trip Test 1 LPRM Level A (BPV Sensitivity)

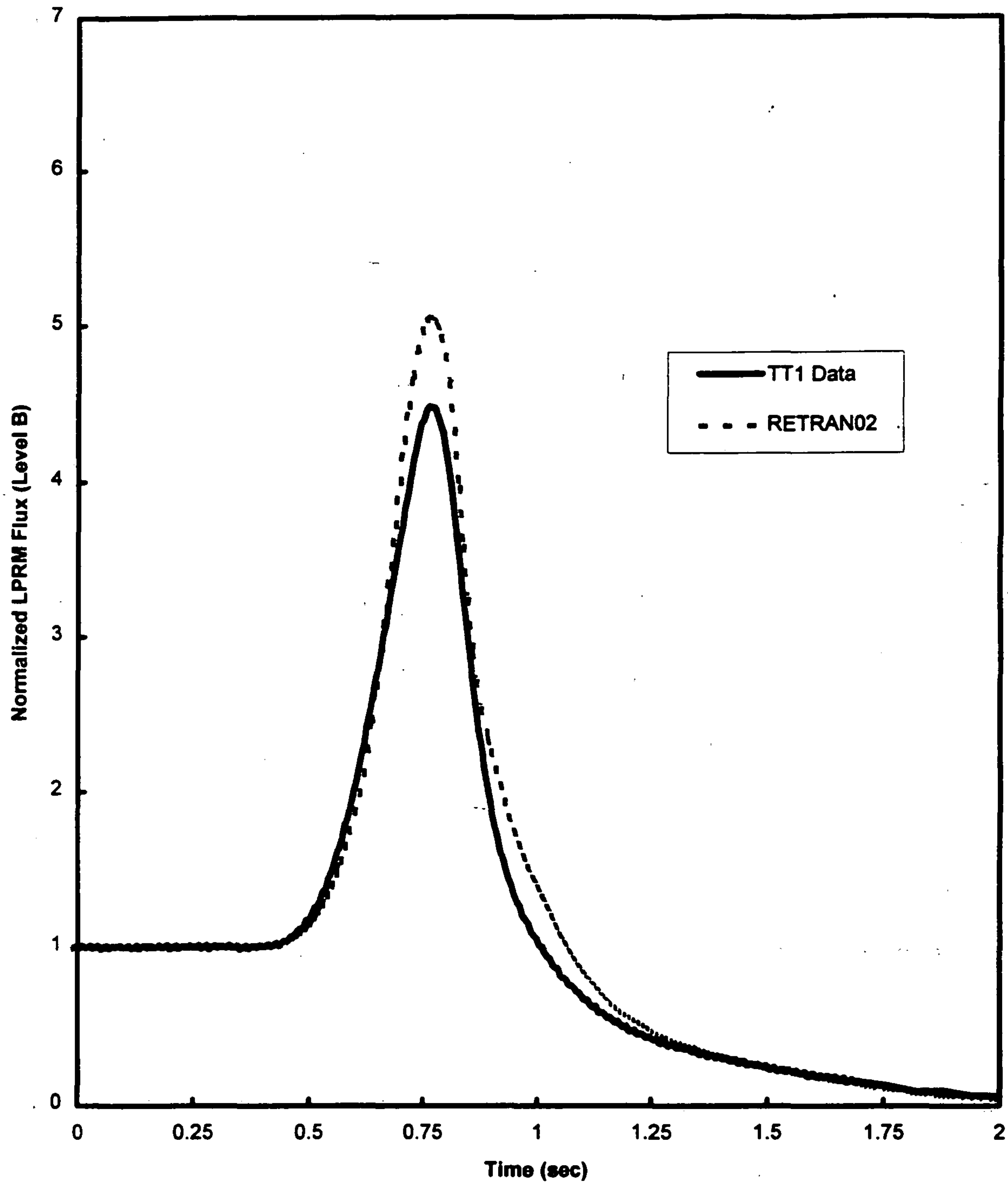

Figure 5.4-38 Turbine Trip Test 1 LPRM Level B (BPV Sensitivity)

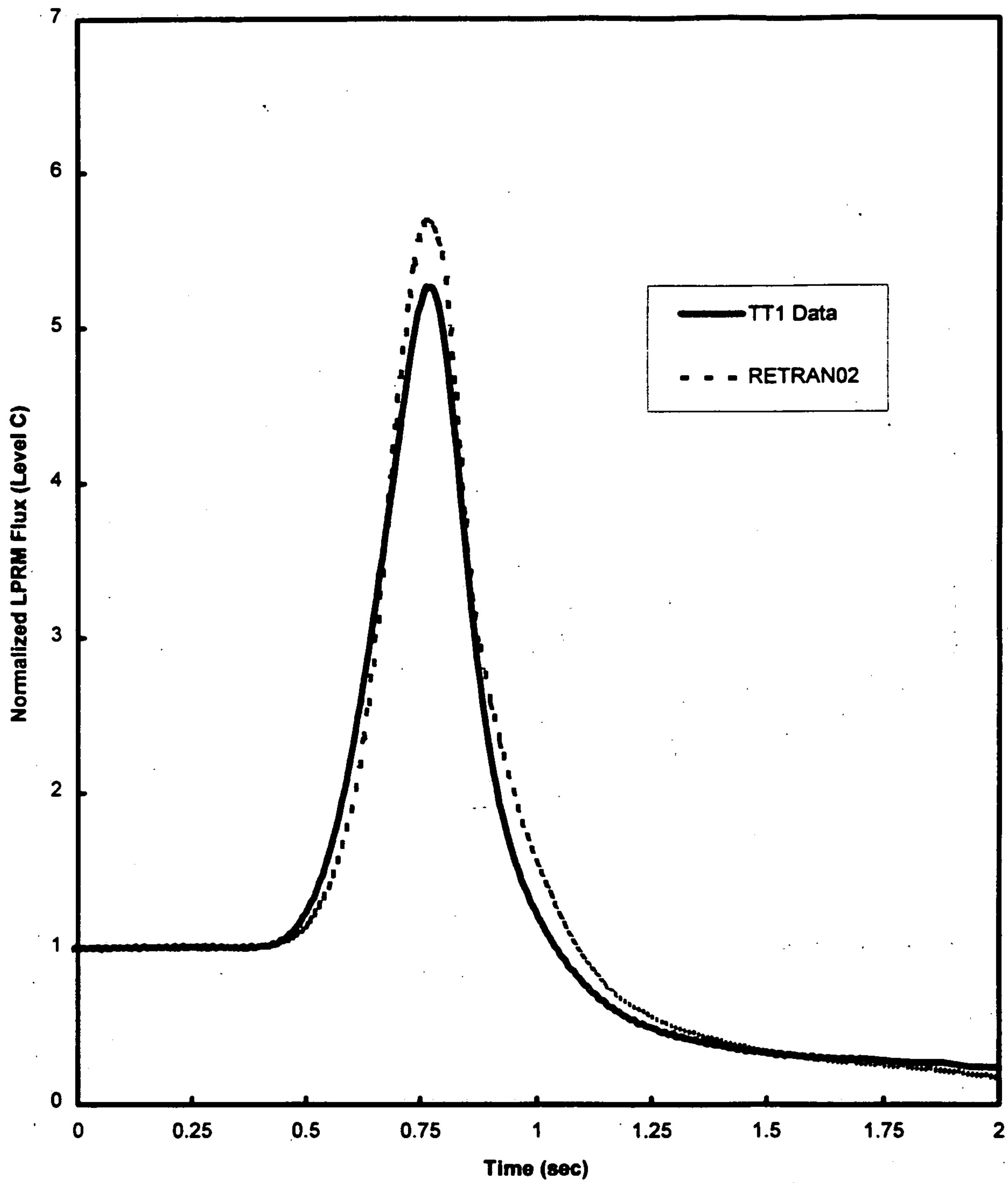

Figure 5.4-39 Turbine Trip Test 1 LPRM Level C (BPV Sensitivity)

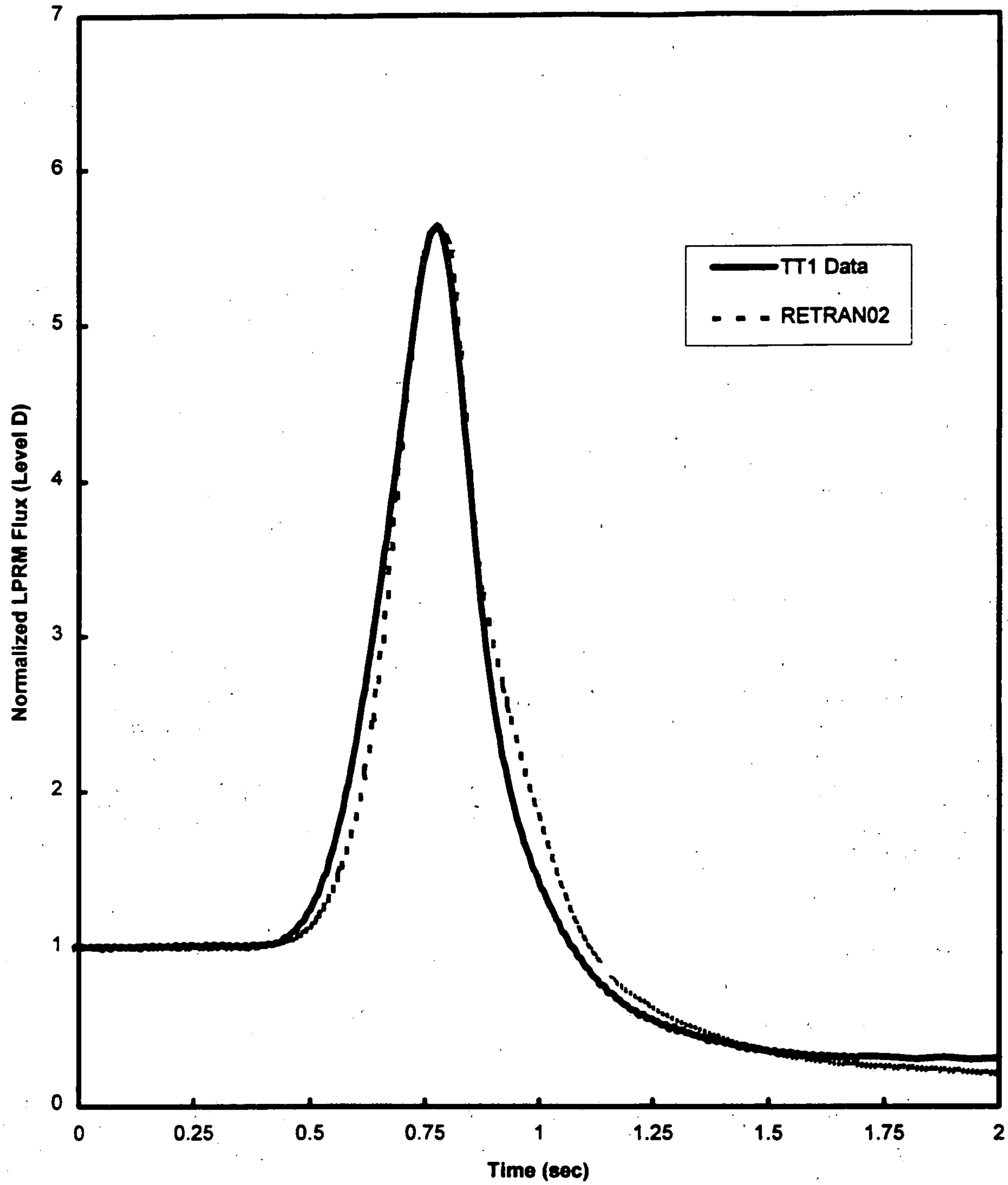

Figure 5.4-40 Turbine Trip Test 1 LPRM Level D (BPV Sensitivity)

6. Peach Bottom NRC Licensing Basis Transient

6.1 Transient Description

The NRC test problem is a limiting pressurization licensing transient for Peach Bottom. The licensing transient analyzed was a turbine trip without bypass. This analysis was originally prepared by Brookhaven National Laboratory (BNL) (Reference 21), to independently assess GE's analytical results. Since then, many utilities have been comparing the results of their Licensing Basis Transient (LBT) for Peach Bottom to the BNL and GE analytical results.

The LBT for Peach Bottom was performed with the same models and methods as described in Section 5. The model was initialized consistent with the methods discussed previously to the values given in Table 6.2-1, consistent with the Reference 21 documentation. The inlet enthalpy was calculated with a heat balance because the value listed in the reference documentation was lower than achievable based on the given power, pressure and steam flow. The LBT also requires the actuation of relief valves. The relief valves were modeled consistent with Reference 21 and all comparisons were made to figures given in Reference 21.

6.2 RETRAN Modeling of Transient

The turbine trip transient was initiated at the first time step and the trip logic was set up to obtain a scram on TSV position. This was set up as a trip at time 0.0 with a delay of .07 seconds. Twenty milliseconds of this delay is the time for the TSV to reach the trip position and generate the signal. The remaining 0.05 seconds accounts for the RPS delay logic. This is a standard delay assumed by GE for turbine trip analyses. The TSV position was assumed to be linear as a function of time with full closure in 100 msec which is typical for GE licensing analyses. The turbine bypass was disabled during this simulation.

The scram speed (position vs. time) was assumed to be the scram speed required by Technical Specifications for the control rod drive operability. This scram speed is commonly known as the "Option A" scram speed, which GE typically uses in their analyses.

The relief valve trips were all set to sense high dome pressure and provide a trip consistent with the setpoints and delays in Table 6.2-2. The relief valve contraction coefficients were adjusted to obtain the reference flow rate listed in Table 6.2-2. The safety valves were not modeled since pressure did not reach the safety valve opening setpoint during the turbine trip simulation.

The direct moderator heating was set to 2.0% at all core nodes, and an average bypass heating value of 1.5% was also assumed for this analysis. A constant gap conductance of 1000.0 BTU/hr-ft2-°F was input, consistent with Reference 21.

The yield fractions, β_i/β, and the corresponding decay constants were calculated consistent with the methods documented in Appendix A. These values are listed in Table 6.2-3.

Table 6.2-1 Licensing Basis Transient Initial Conditions

Parameter	Initial Value
Core Power (MWt)	3440.0
Total Core Flow (Mlbm/hr)	102.5
Core Inlet Enthalpy (BTU/lbm)	522.87
Turbine Steam Flow (Mlbm/hr)	14.04
Dome Pressure (psia)	1034.0
Gap Conductance (BTU/hr-ft^2-°F)	1000.0

Table 6.2-2 LBT Relief Valve Modeling Assumptions

# Relief Valves	Setpoint (psia) (open/close)	Flow/Valve (lbm/sec)	Delay (sec)	Stroke Time (sec)
4	1090.8/1070.8	218.0	0.4	.15
4	1100.9/1080.8	218.0	0.4	.15
3	1111.0/1091.0	218.0	0.4	.15

Table 6.2-3 LBT Delayed Neutron Data

Group	Yield Fraction	λ (sec^{-1})
1	0.030557	0.01280
2	0.207060	0.03153
3	0.185500	0.12424
4	0.389420	0.32719
5	0.149630	1.40520
6	0.378260	3.83680
Total β:	0.005376	

6.3 Results

The RETRAN02 calculated power and heat flux distributions (Figure 6.3-1 and Figure 6.3-2) closely match both GE and BNL. The RETRAN02 axial average temperature (Figure 6.3-3) is in close agreement with GE, but it is less than the BNL result. The initial void fraction profile (Figure 6.3-4) is in good agreement with both GE and BNL. The initiation of subcooled boiling is in closer agreement with the BNL calculation and the exit void fraction is only a few percent higher than GE and BNL. The good agreement between the power shape, heat flux and void fraction shows that the RETRAN02 initial core steady state configuration is fairly consistent with GE and BNL calculations. The observed differences are expected due to the differences in the analytical models, void correlations, etc.

Figure 6.3-5 shows the comparison of the RETRAN02 peak normalized power to GE and BNL. The RETRAN power peak is higher and narrower than the GE calculation. The magnitude and width of the peak power are very sensitive to the delayed neutron fractions. The higher RETRAN peak power results in a higher heat flux as shown in Figure 6.3-6.

The core midplane pressure increase is shown in Figure 6.3-9. The pressure begins to decrease just after 1.0 second due to the actuation of relief valves and the pressure continues to increase shortly after the relief valves have opened. The initial core pressurization rate and magnitude are higher than both GE and BNL. The higher core pressure increase is reflected in the void fraction change as a function of time shown in Figure 6.3-7. This figure shows a larger decrease in the core average void fraction. The larger change in void fraction is reflected in the normalized power response and the reactivity response shown in Figure 6.3-13. The reactivity trend between BNL and RETRAN is quite different. The BNL result shows that reactivity momentarily levels off at about 0.6 seconds. This is probably due to the large inflection in the BNL core pressure shown in Figure 6.3-9.

Figure 6.3-8 shows the comparison of the core averaged fuel temperature as a function of time. The comparison shows that the BNL initial value is about 150°F higher than the RETRAN value. The higher average BNL temperature is consistent with the initial axial temperature distribution shown in Figure 6.3-3. However, there is close agreement between BNL and RETRAN for the change in average fuel temperature during the transient as shown in Figure 6.3-8.

Figure 6.3-10 shows the comparison of the core flow response as a function of time. The comparison shows close agreement between the RETRAN and the GE results for both timing and magnitudes. The trend of the BNL result is very similar but the flow is much lower.

Figure 6.3-11 and Figure 6.3-12 show the comparison of the axial heat flux response for 0.8 seconds and 1.2 seconds respectively. The two figures show the shift in power toward the top of the core as a function of time. The calculated RETRAN axial heat flux follows the GE calculated trend. The RETRAN axial heat flux at 1.2 seconds is higher, consistent with the change in average heat flux shown in Figure 6.3-6.

In summary, with the same methods as applied to the Peach Bottom turbine trip tests, the Peach Bottom LBT was simulated. The model was set up to match the initial conditions from published material and conservative licensing type input assumptions were assumed for the analysis. The results show that the RETRAN model would be more conservative than GE or BNL for the Peach Bottom LBT.

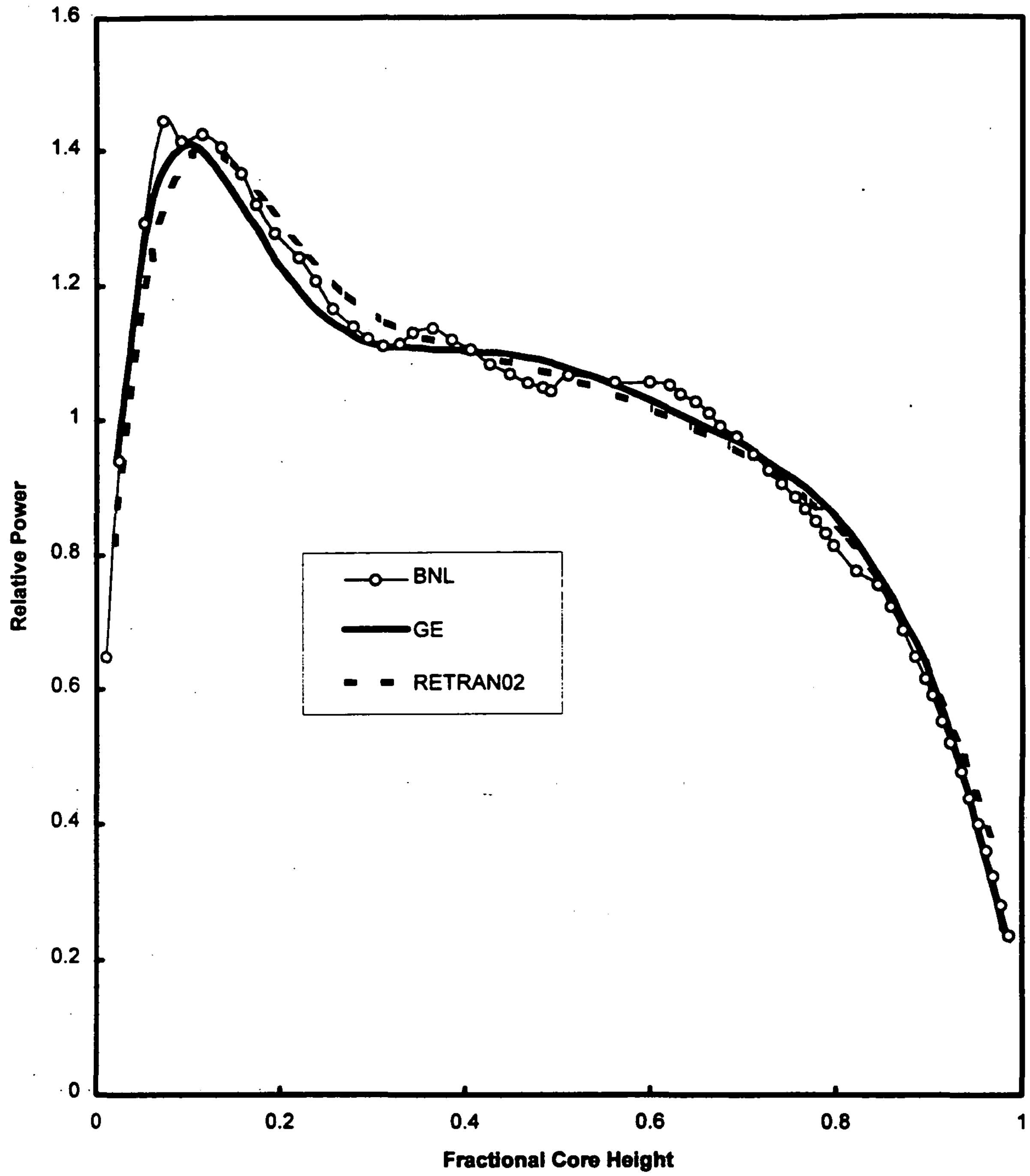

Figure 6.3-1 Peach Bottom LBT Initial Power Shape

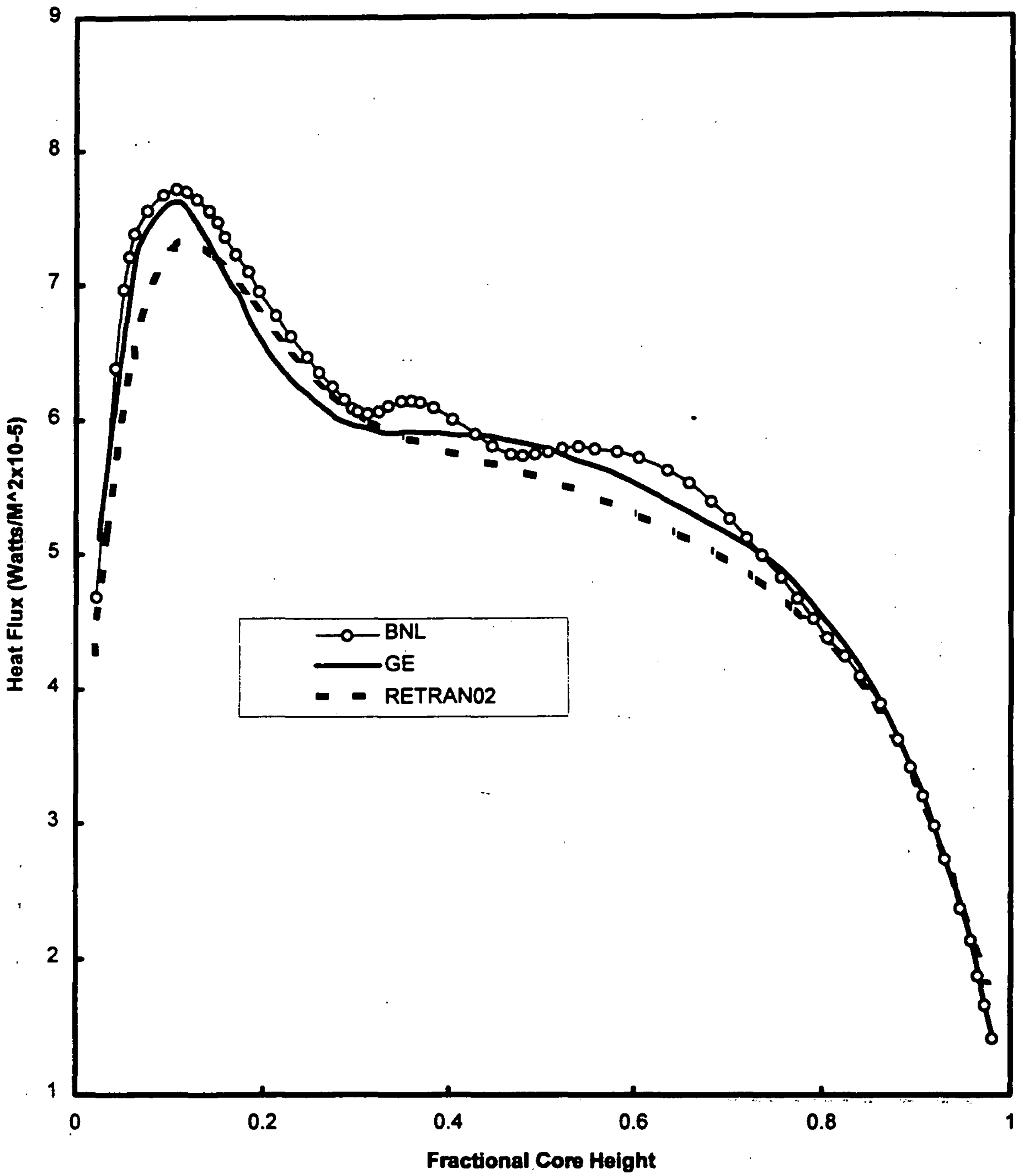

Figure 6.3-2 Peach Bottom LBT Initial Heat Flux

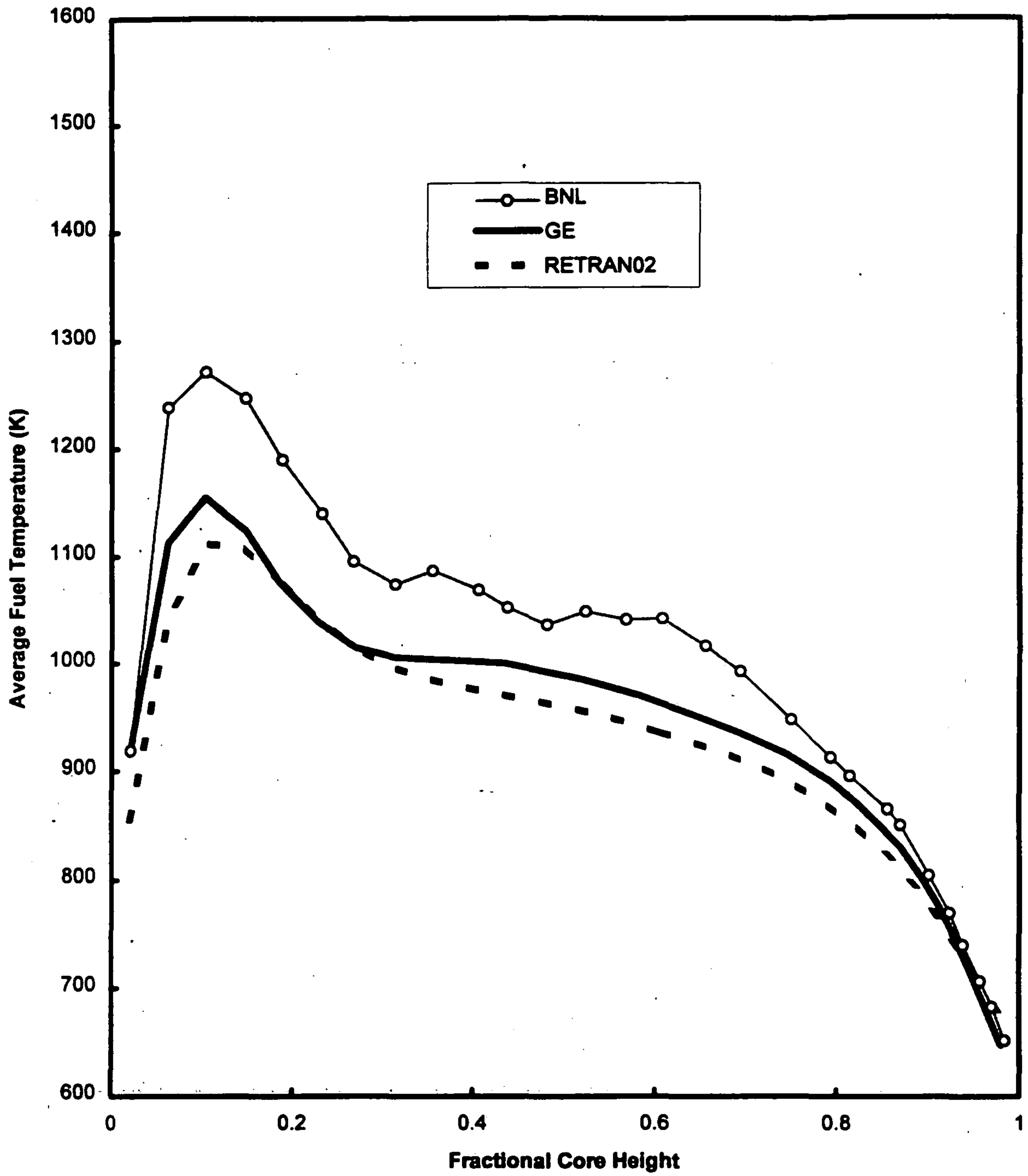

Figure 6.3-3 Peach Bottom LBT Initial Fuel Temperatures

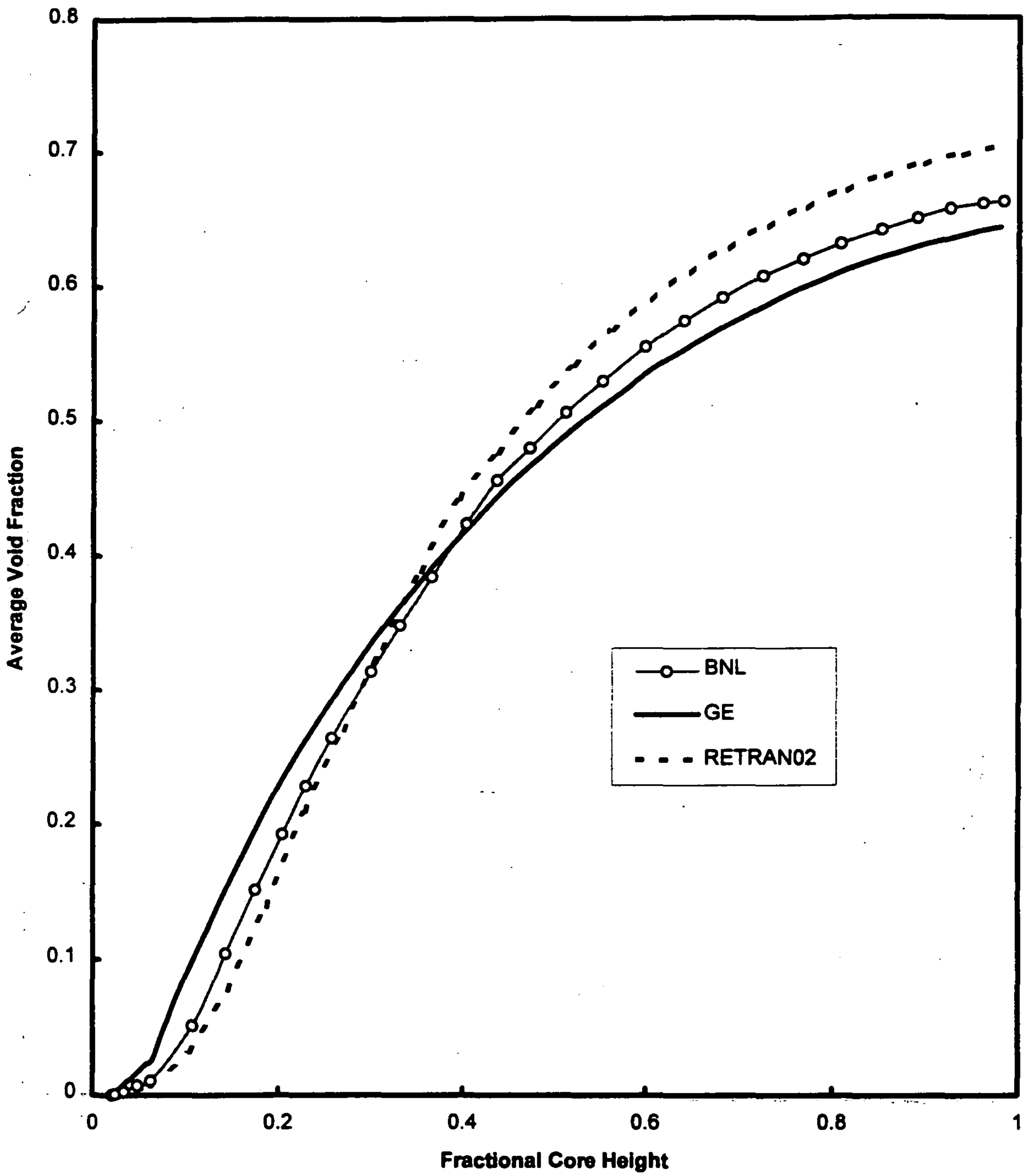

Figure 6.3-4 Peach Bottom LBT Initial Void Distribution

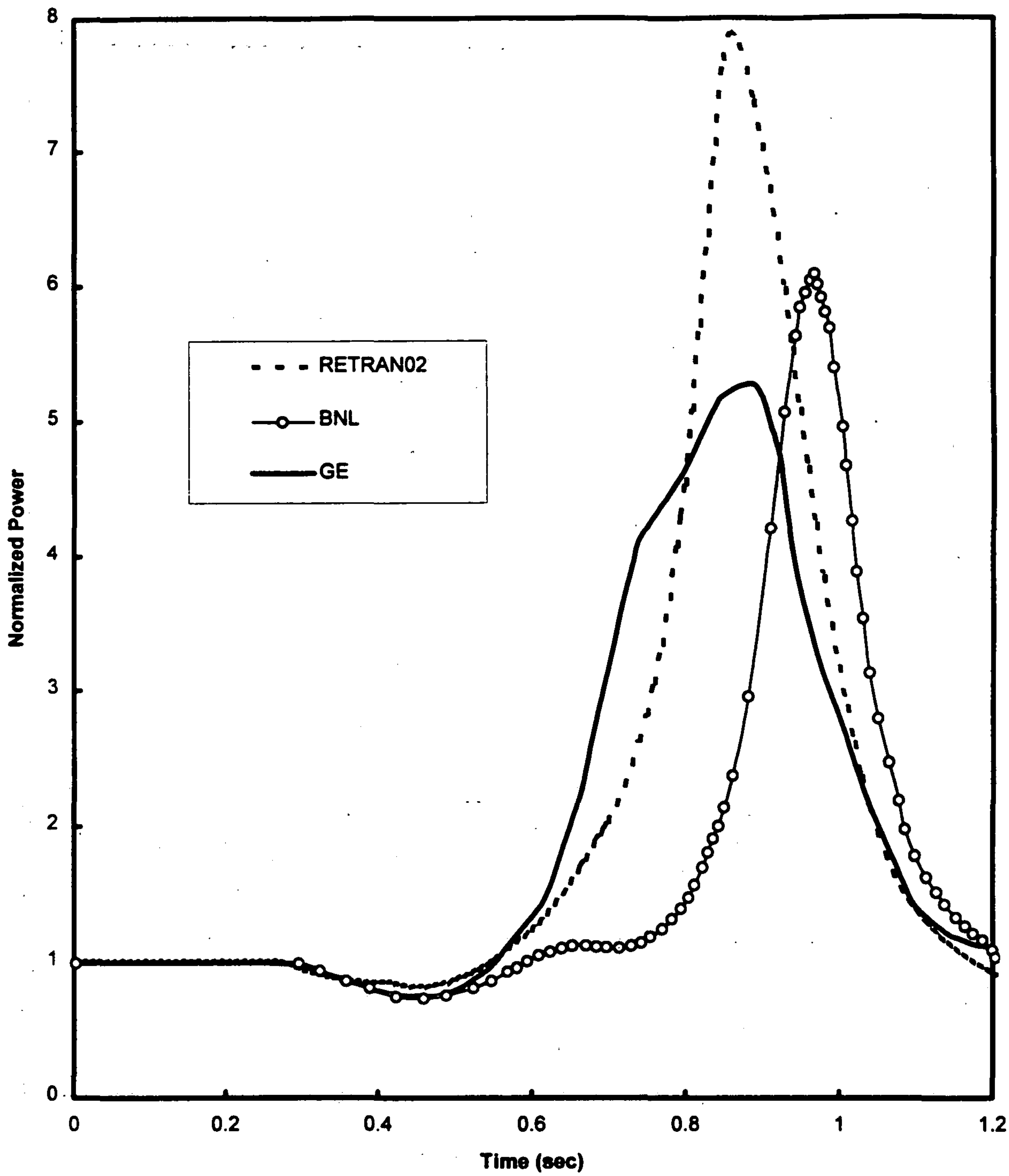

Figure 6.3-5 Peach Bottom LBT Normalized Power

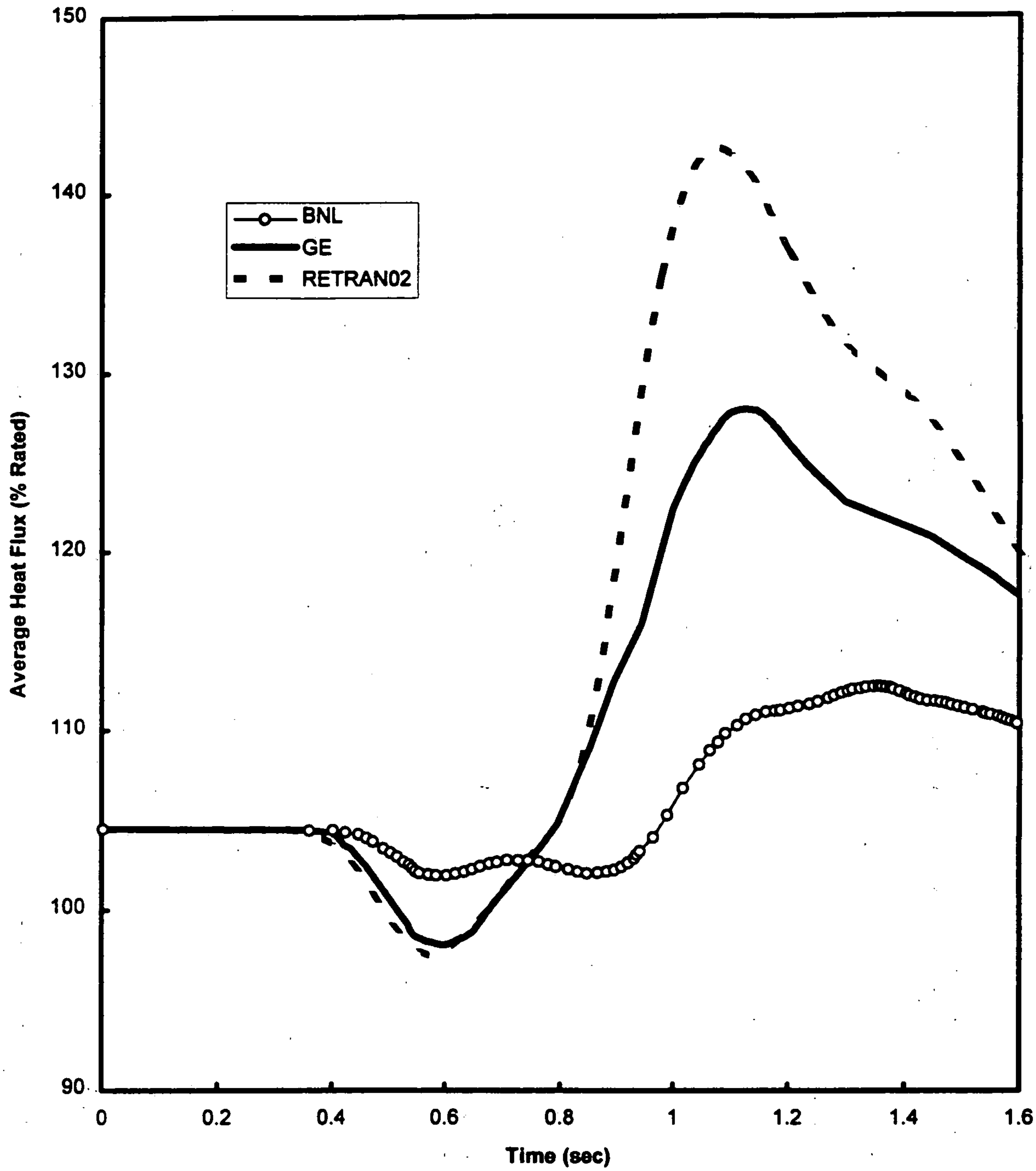

Figure 6.3-6 Peach Bottom LBT Core Heat Flux

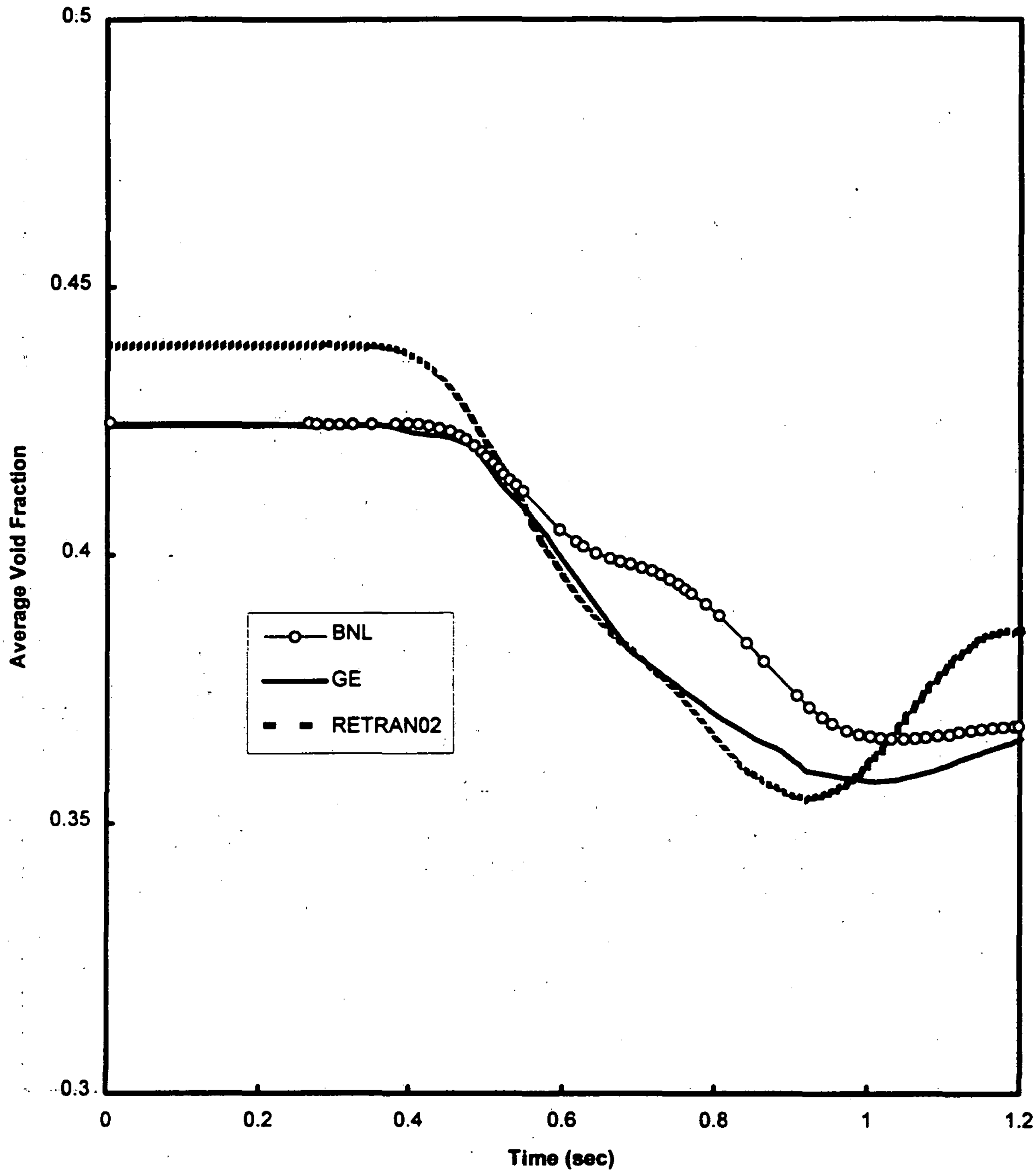

Figure 6.3-7 Peach Bottom LBT Core Average Void Fraction

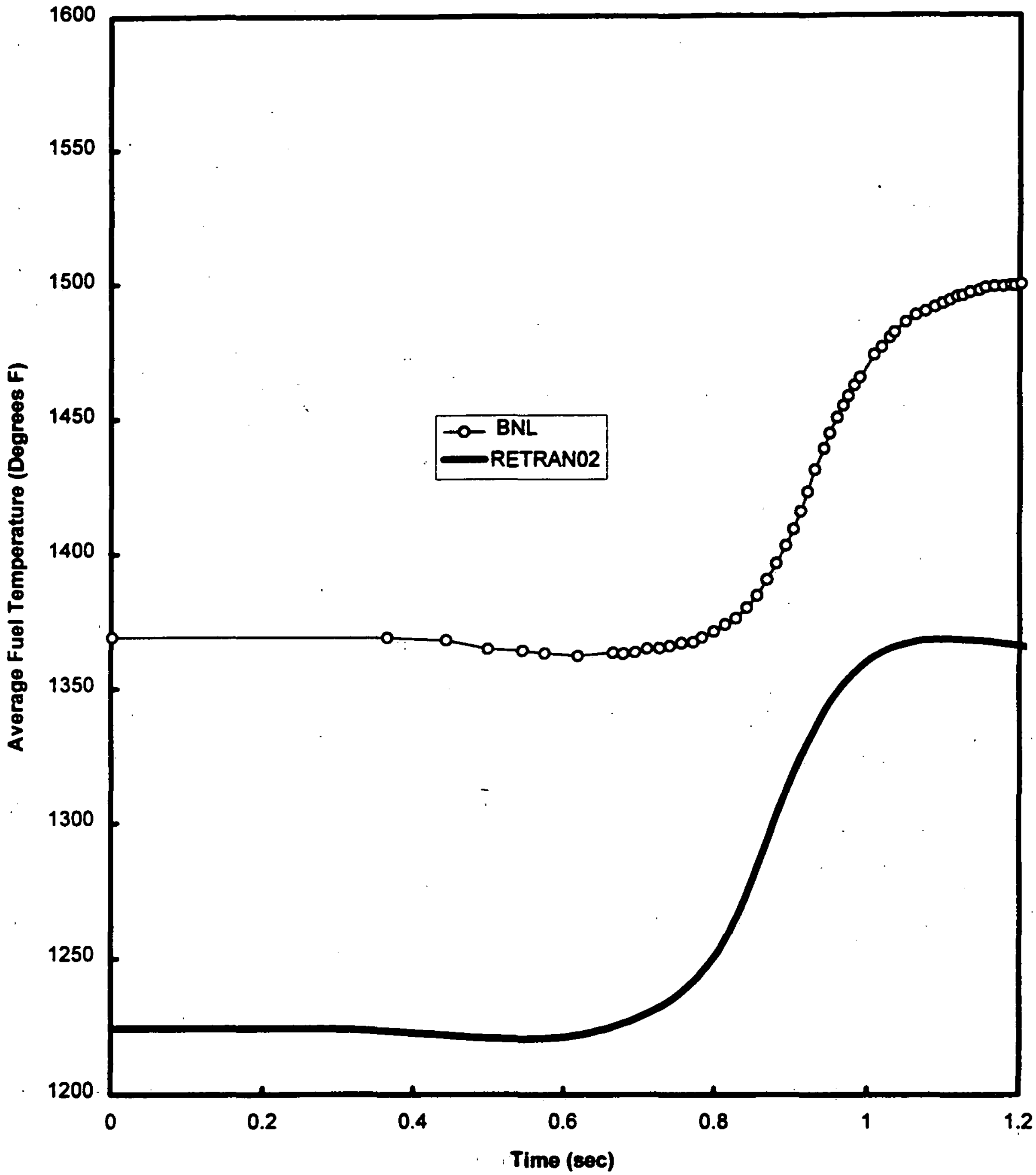

Figure 6.3-8 Peach Bottom LBT Average Fuel Temperature

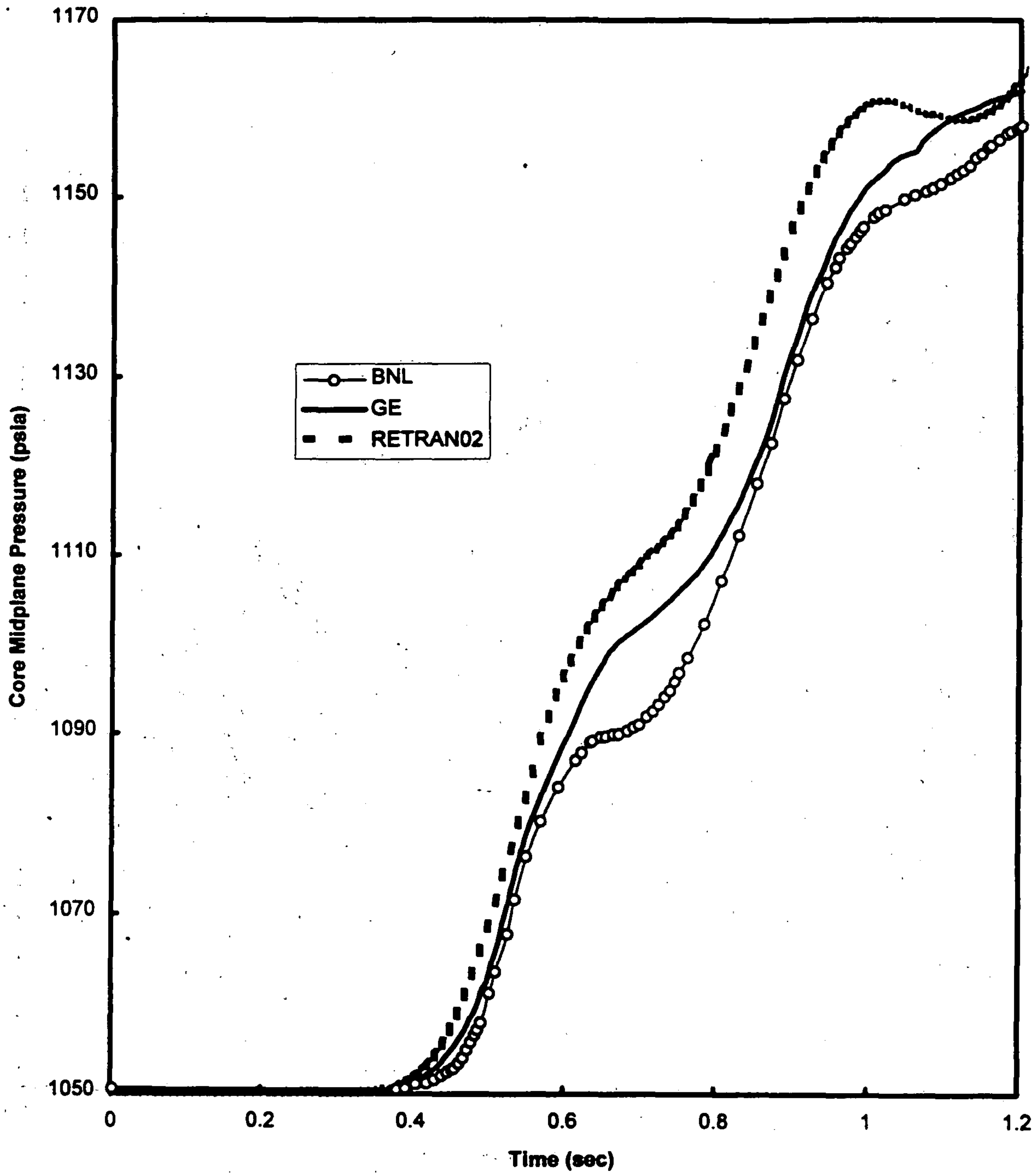

Figure 6.3-9 Peach Bottom LBT Core Midplane Pressure

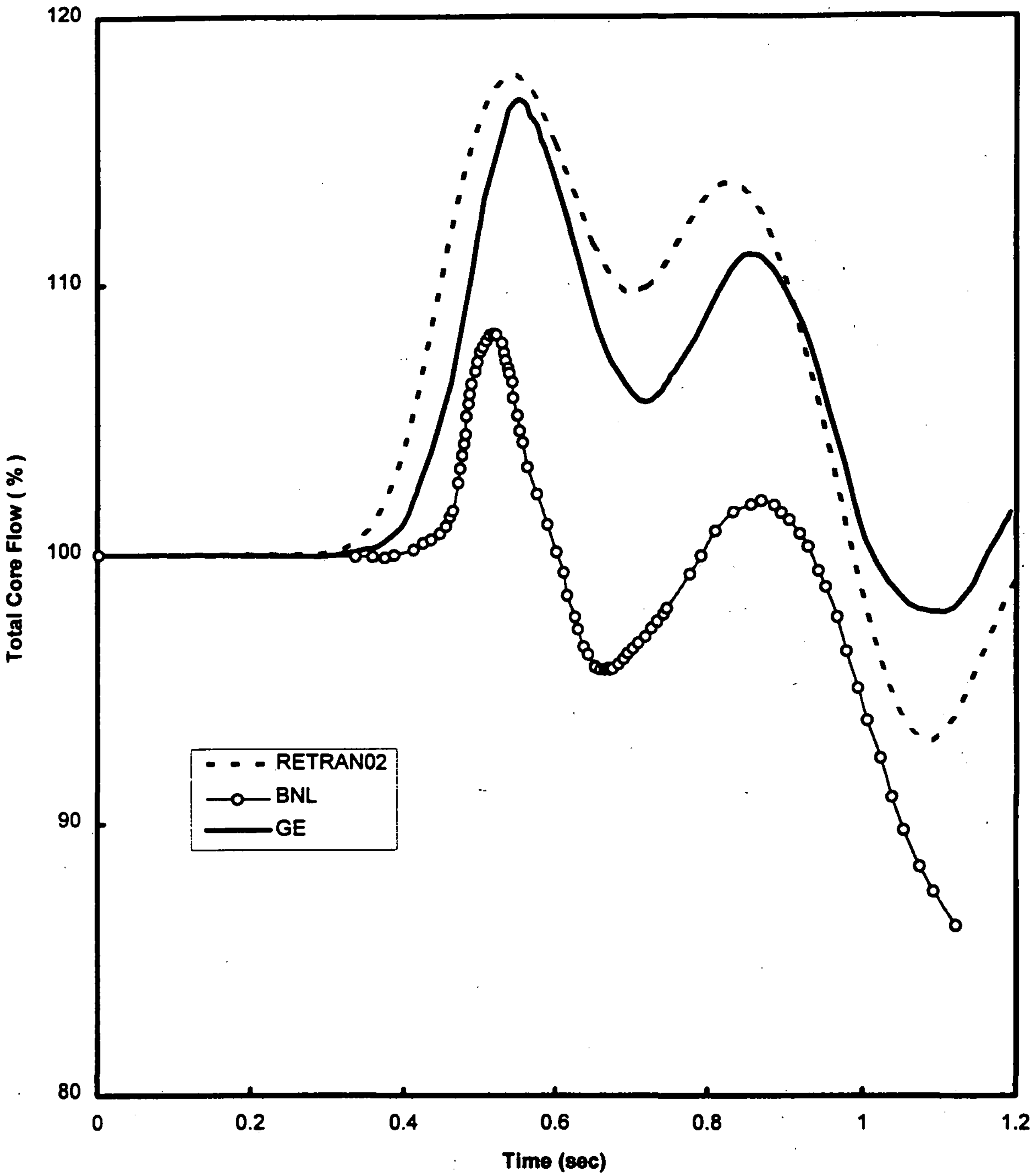

Figure 6.3-10 Peach Bottom LBT Total Core Flow

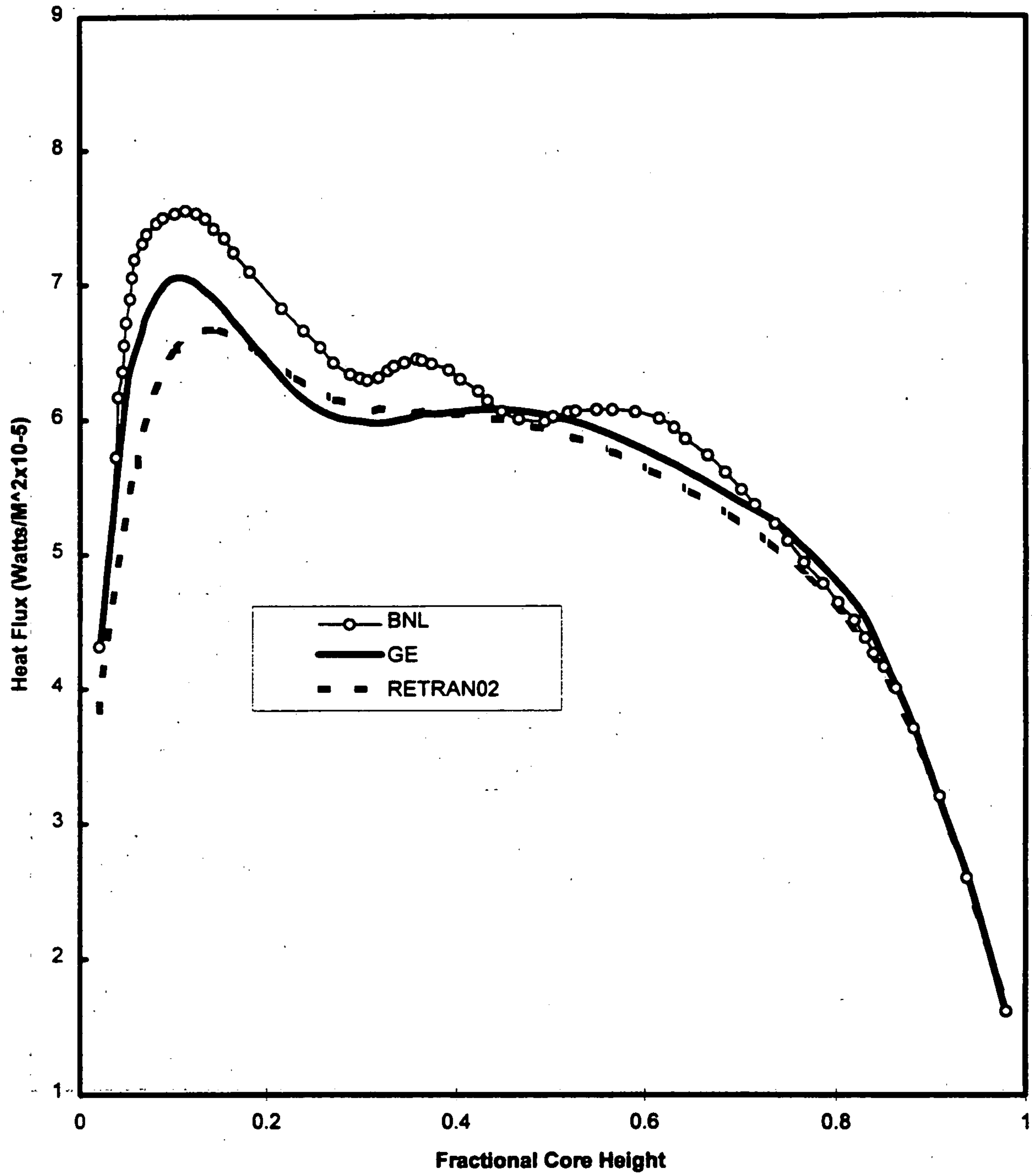

Figure 6.3-11 Peach Bottom LBT Heat Flux Distribution at 0.8 Seconds

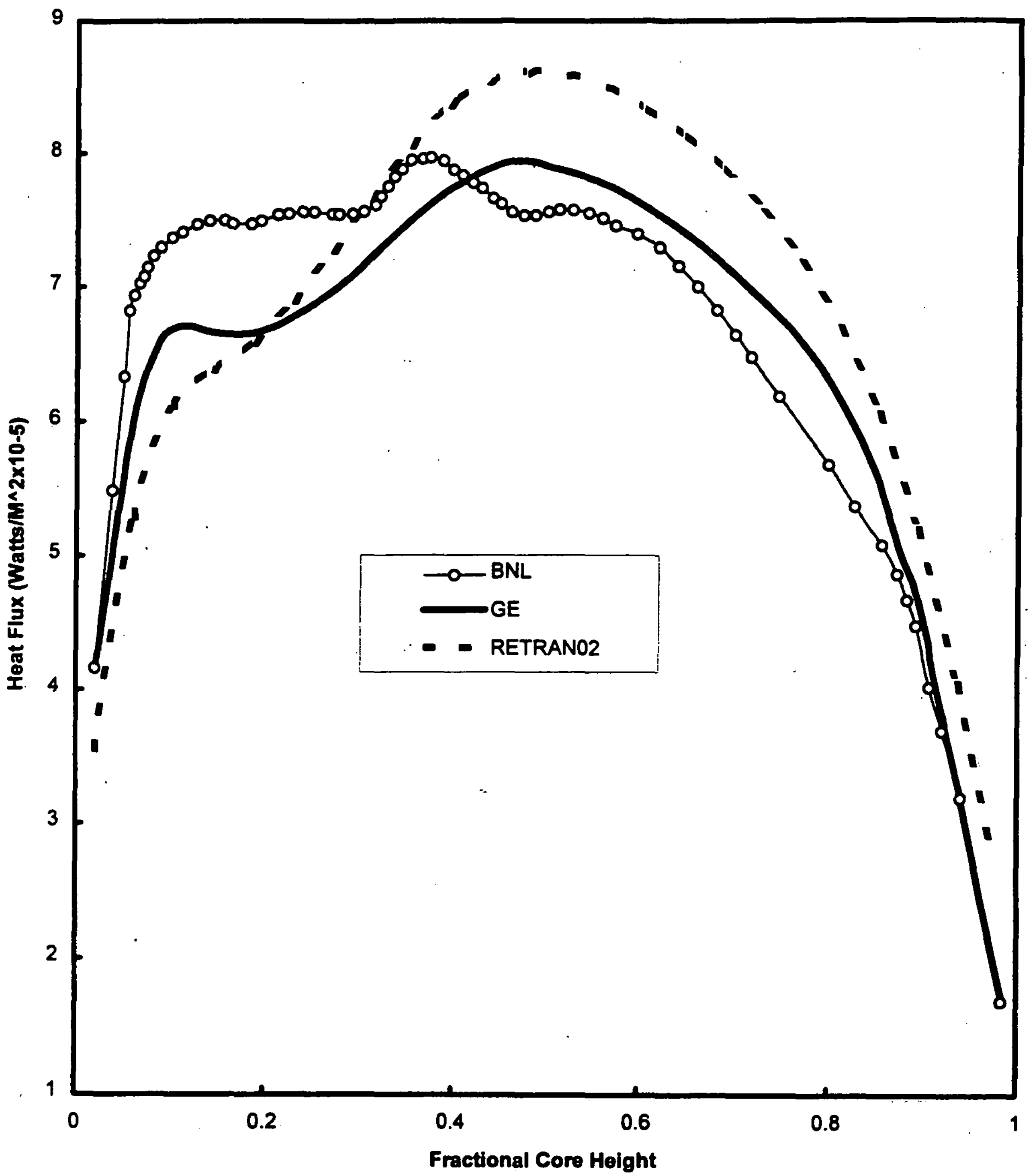

Figure 6.3-12 Peach Bottom LBT Heat Flux Distribution at 1.2 Seconds

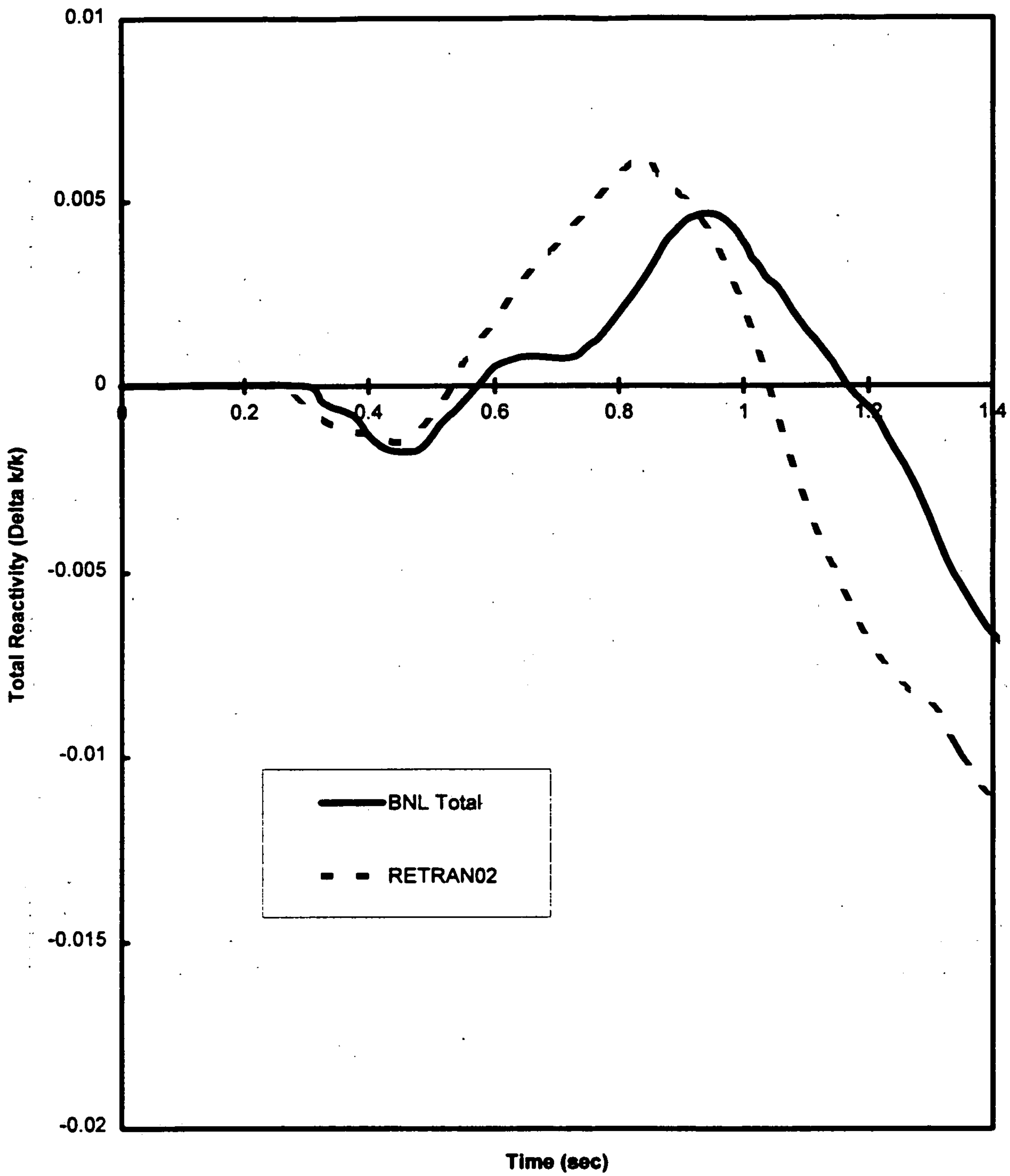

Figure 6.3-13 Peach Bottom LBT Total Reactivity

7. Summary and Conclusions

The analyses presented in this report demonstrate that ComEd's transient analysis methods and plant models accurately predict actual core and system transient behavior. ComEd intends to use the methods and models for both reload licensing and operational support applications. Details of the thermal limit and reload application methodologies to produce conservative licensing transient analyses will be described in a separate report.

The benchmarking analysis of the plant startup tests was chosen to validate the ComEd transient analysis methods for a variety of plant transients. The startup tests analyzed were: reactor water level setpoint change, pressure regulator setpoint change, dual recirculation pump trip, load rejection/turbine trip with bypass, and MSIV closure. All the key core and system transient parameters predicted by the RETRAN plant models match the measured data well and show acceptable agreements.

Benchmarking analyses of the turbine trip tests performed at Peach Bottom Unit 2 Cycle 2 demonstrate the validity of ComEd transient analysis methods for more challenging pressurization events similar to licensing basis events. The calculated and measured results for the key parameters agree well for all three cases.

An NRC licensing basis transient case (with a set of standard assumptions) of Peach Bottom turbine trip without bypass was analyzed to demonstrate the ComEd Method's capability of predicting system response under conditions which challenge operating limits. The ComEd calculations were consistently conservative compared to GE and BNL results.

The acceptability of ComEd 1D kinetics method and RETRAN plant models for design applications of the rapid pressurization transient event analyses has been fully demonstrated by the benchmarking studies of the Peach Bottom turbine trip tests and the NRC licensing basis transient case. A broad spectrum of plant startup test data are also used to verify the ComEd plant-specific RETRAN models. The analyses and results presented in this report demonstrate the capability of the ComEd transient analysis methods and the qualification of the ComEd safety analysis staff to perform transient analyses for reload licensing and operational support applications.

8. References

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Appendix A - One-Dimensional Kinetics Methodology

The diagram in Figure A-1 outlines the 1-D kinetic methodology associated with the RETRAN one dimensional space-time kinetics model. This methodology requires the generation of sets of polynomials which correlate changes in RETRAN feedback parameters (relative moderator density and square root of fuel temperature) with 3-D core simulator (Reference 1) calculated kinetics parameters. This is accomplished in a six step process which principally utilizes the MICROBURN and PETRA computer codes:

1. MICROBURN - FIBWR2 Iteration establishes a MICROBURN axial power shape consistent with a FIBWR2 axial pressure distribution (core inlet subcooling agreement).

2. MICROBURN base, perturbation and scram cases generate sets of nodal cross sections.

3. PETRA creates "steady state" 1-D kinetic files (no feedback coefficients) for the base and each of the two perturbation cases from the sets of nodal cross sections.

4. RETRAN uses each "steady state" 1-D kinetic file to calculate the axial moderator density distribution for the base case and each perturbation case core thermal hydraulic conditions.

5. PETRA uses the RETRAN axial moderator density distributions and the MICROBURN nodal cross sections to create a "transient" 1-D kinetic file containing moderator density feedback polynomial coefficients.

6. Fuel temperature (Doppler) feedback coefficients are calculated from CASMO and MICROBURN data and inserted into the "transient" 1-D kinetic file.

The core inlet enthalpy input to FIBWR2 is calculated from a heat balance. The inlet subcooling used in MICROBURN is calculated from the core inlet enthalpy input to FIBWR2 and the core inlet pressure calculated by FIBWR2. The methodology for application of FIBWR2 was described in Appendix B. The core axial pressure distribution calculated by FIBWR2 is influenced by the input axial power distribution calculated by MICROBURN. The MICROBURN axial power distribution is influenced by the inlet subcooling. An iterative process continues until the axial power and pressure distributions are consistent with each other. The FIBWR2 axial pressure distribution will be used to initialize the RETRAN system model.

Two MICROBURN inlet enthalpy perturbation cases are restarted from the base case and depleted with Doppler feedback and xenon (~4 seconds time interval in time-dependent xenon mode) frozen. These three cases form the initial control state.

To represent reactor scram conditions MICROBURN cases are restarted from each of the initial control state MICROBURN cases. These MICROBURN scram cases are depleted with Doppler feedback, void feedback and xenon frozen. Figure A-2 shows a good agreement between RETRAN and MICROBURN calculated scram worth curves.

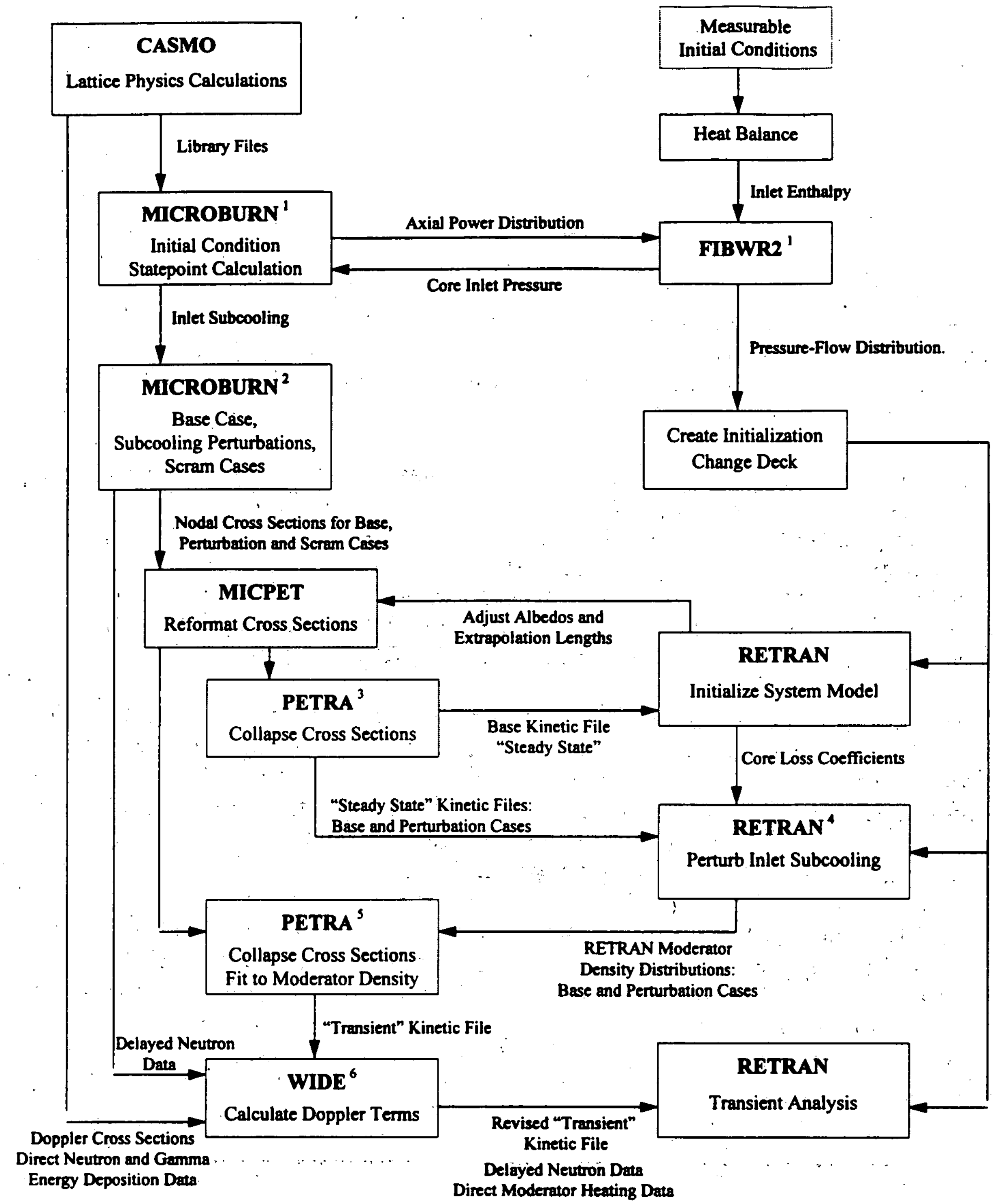
The ComEd computer code MICPET is used to reformat the nodal cross section, power and void data in each MICROBURN output file into the PRESTO (Scandpower 3-D core simulator code) restart file format. Albedo and extrapolation length values for the top and bottom reflector boundary conditions are placed into the PRESTO restart files. These values are selected to be equivalent in PETRA to the material cross sections used in MICROBURN for the top and bottom reflector boundary conditions. The accuracy of the selected reflector conditions is demonstrated by comparing the MICROBURN and RETRAN axial power distributions. Figures A-3, A-4, A-5, and A-6 show a good agreement between the RETRAN and MICROBURN calculated axial power shapes for the Peach Bottom transients.

PETRA uses the three dimensional, two-group flux and cross section distributions calculated by MICROBURN to perform an adjoint flux weighted averaging of these cross sections in each reactor axial plane. This process generates a set of average one dimensional cross sections, which input to RETRAN with the MICROBURN reflector cross sections, will reproduce the MICROBURN core average axial flux and power distributions to a high degree of precision. The cross section collapse is performed by PETRA using the equations presented in Reference 22.

PETRA is used to create a "steady state" 1-D kinetic file without any feedback terms for each of the initial control state base and perturbation cases. These are used along with the FIBWR2 axial pressure distribution to initialize the RETRAN model and calculate an axial moderator density distribution for each case. The RETRAN axial moderator density distributions are then used as input to a PETRA case which creates a "transient" 1-D kinetic file containing moderator density feedback terms.

The ComEd computer code WIDE is used to calculate linear Doppler feedback terms using the base case (initial control state) MICROBURN output file, CASMO lattice physics data files, and the "transient" 1-D kinetic file as input. WIDE then inserts these Doppler feedback coefficients into the (revised) transient 1-D kinetic file. WIDE also calculates axial varying neutron velocities, axial varying total delayed neutron fractions, axial varying moderator and bypass direct heating values, delayed neutron group-wise yield fractions, and delayed neutron six group lambdas.

The Peach Bottom benchmark results in Sections 5.4 and 6.3 provide qualification for applying this methodology to BWR transient analysis.



Note: Superscripts correspond to step numbers described on P. A-1

Figure A-1, Flowchart of 1-D Kinetic Methodology

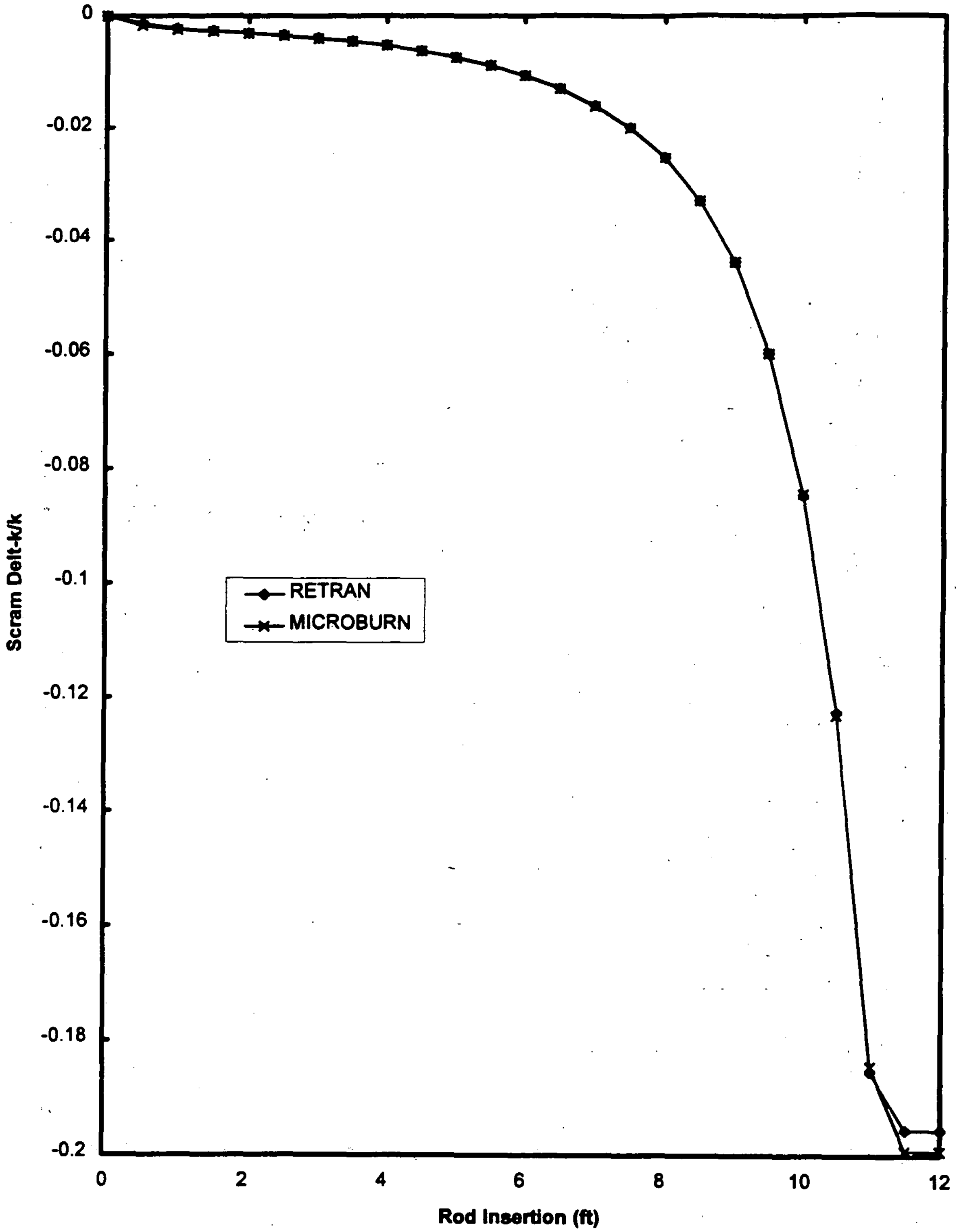


Figure A-2 MICROBURN and RETRAN Control Rod Worth (Peach Bottom TTL)

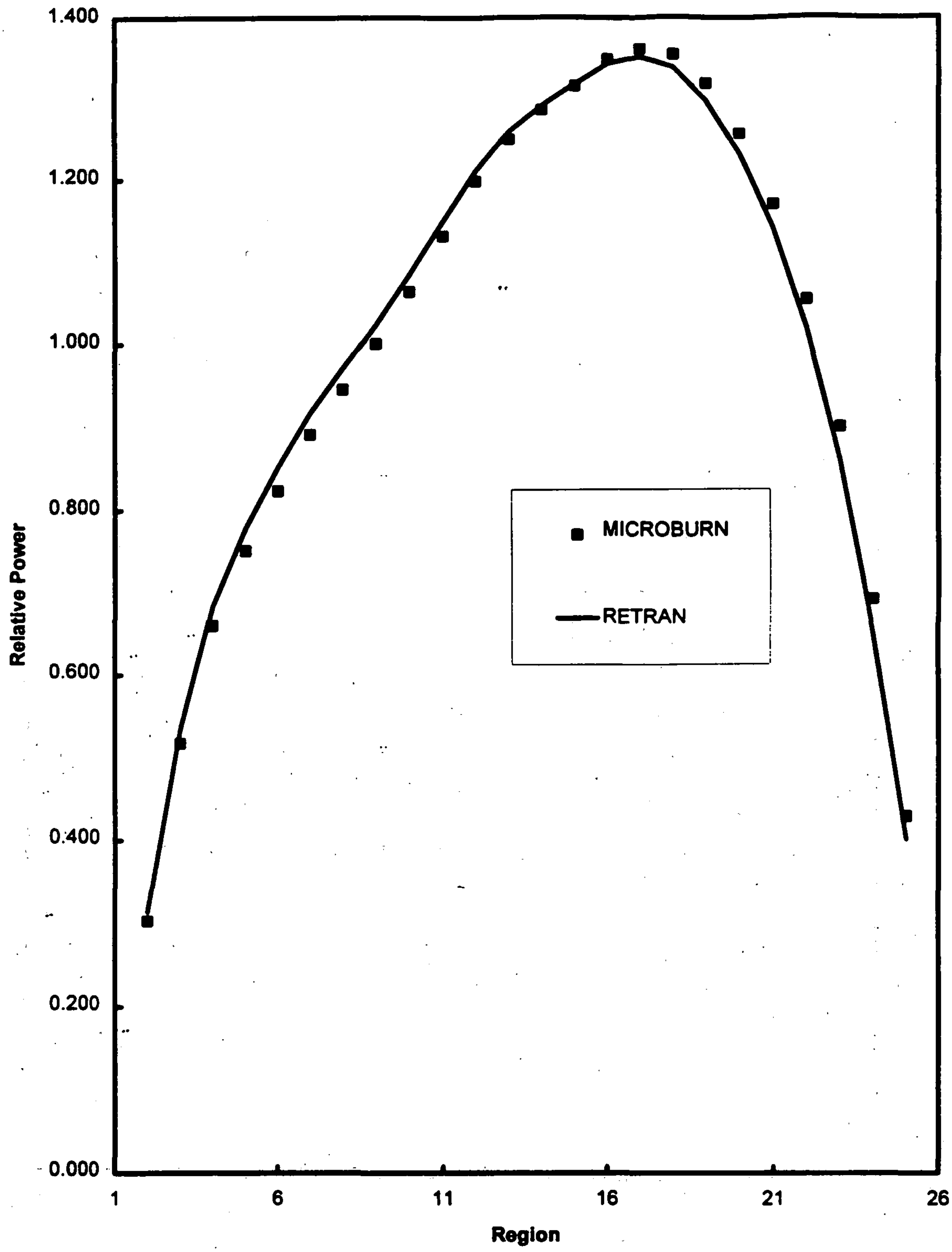


Figure A-3 MICROBURN and RETRAN Axial Power Distribution for PB TT1

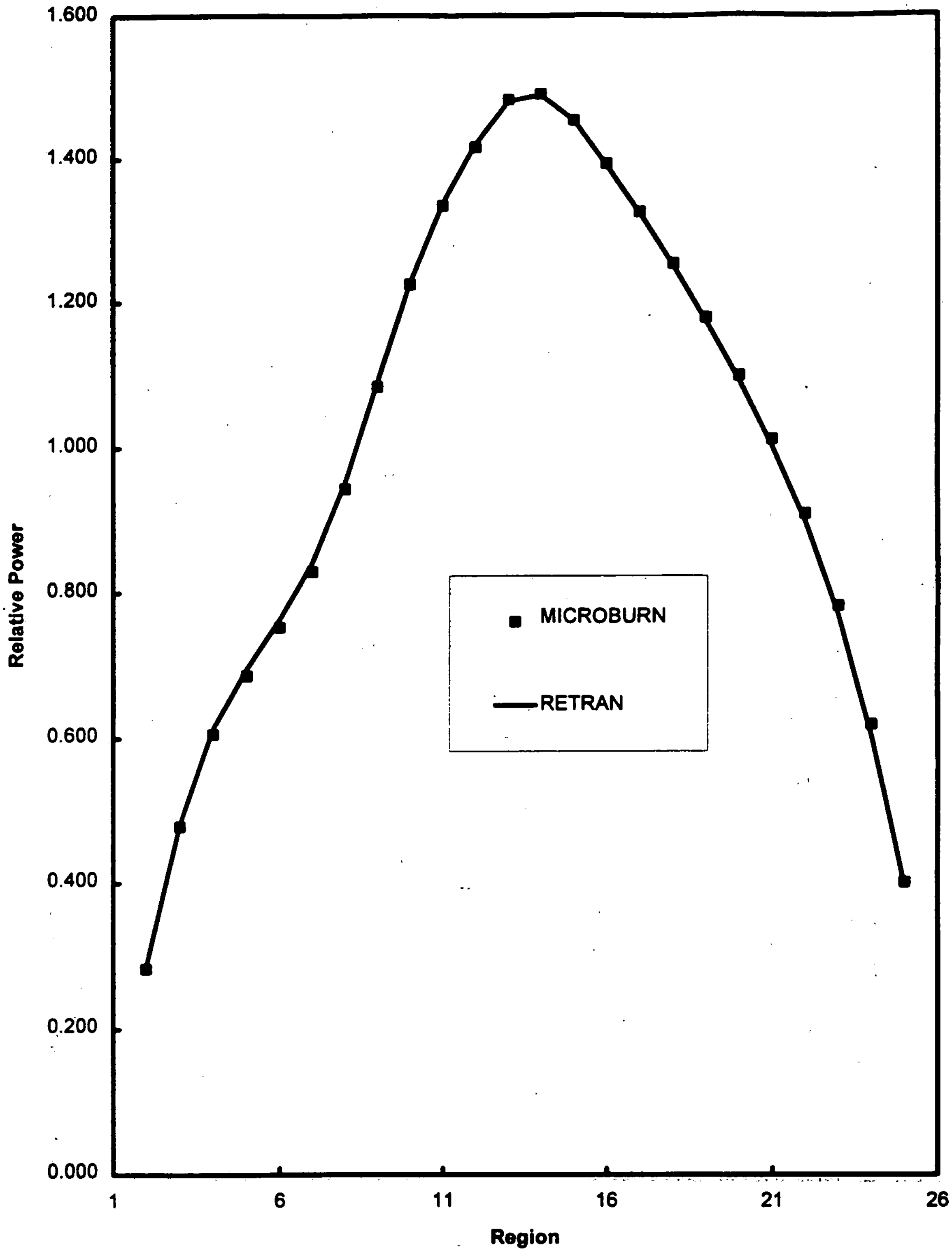


Figure A-4 MICROBURN and RETRAN Axial Power Distribution for PB TT2

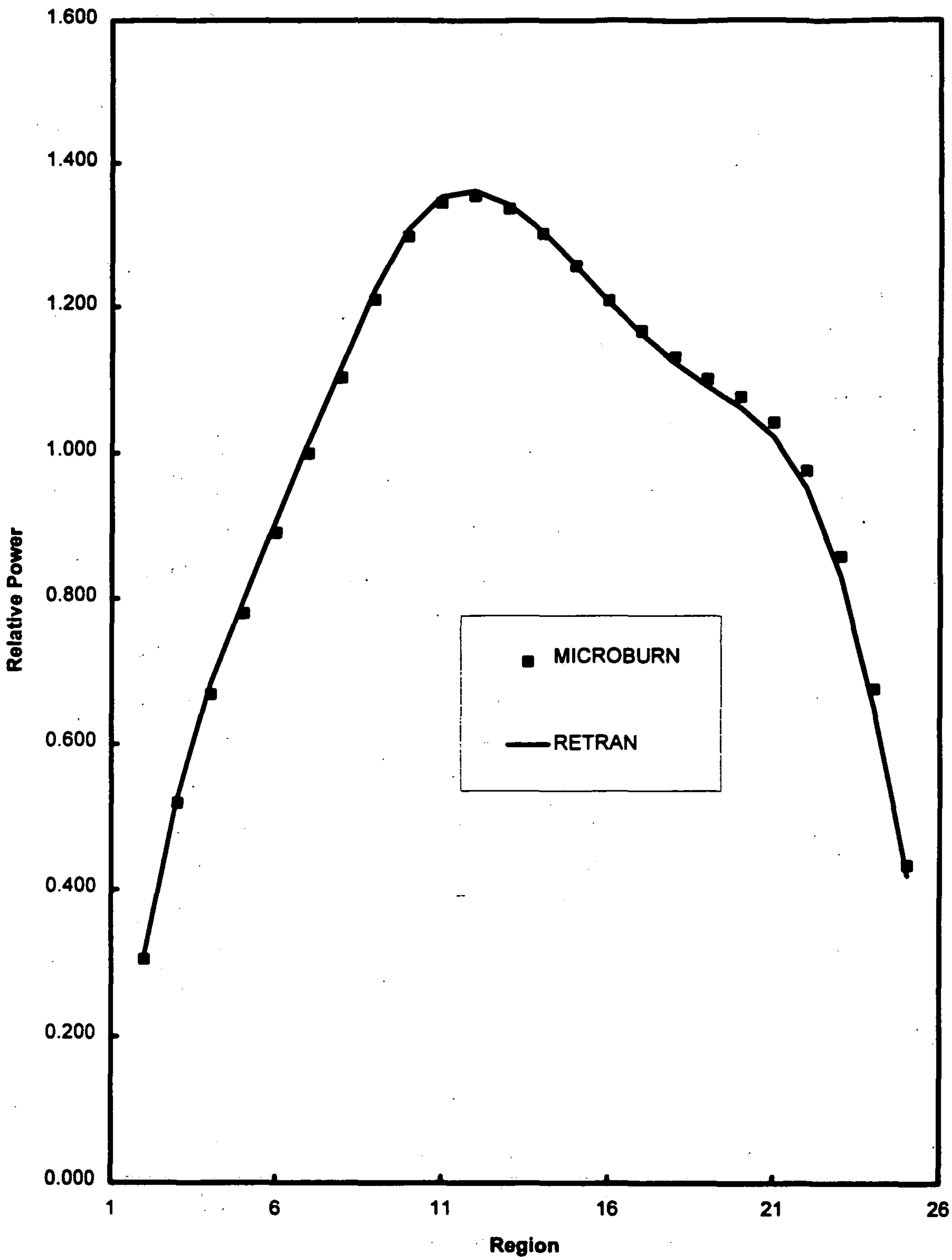


Figure A-5 MICROBURN and RETRAN Axial Power Distribution for PB TT3

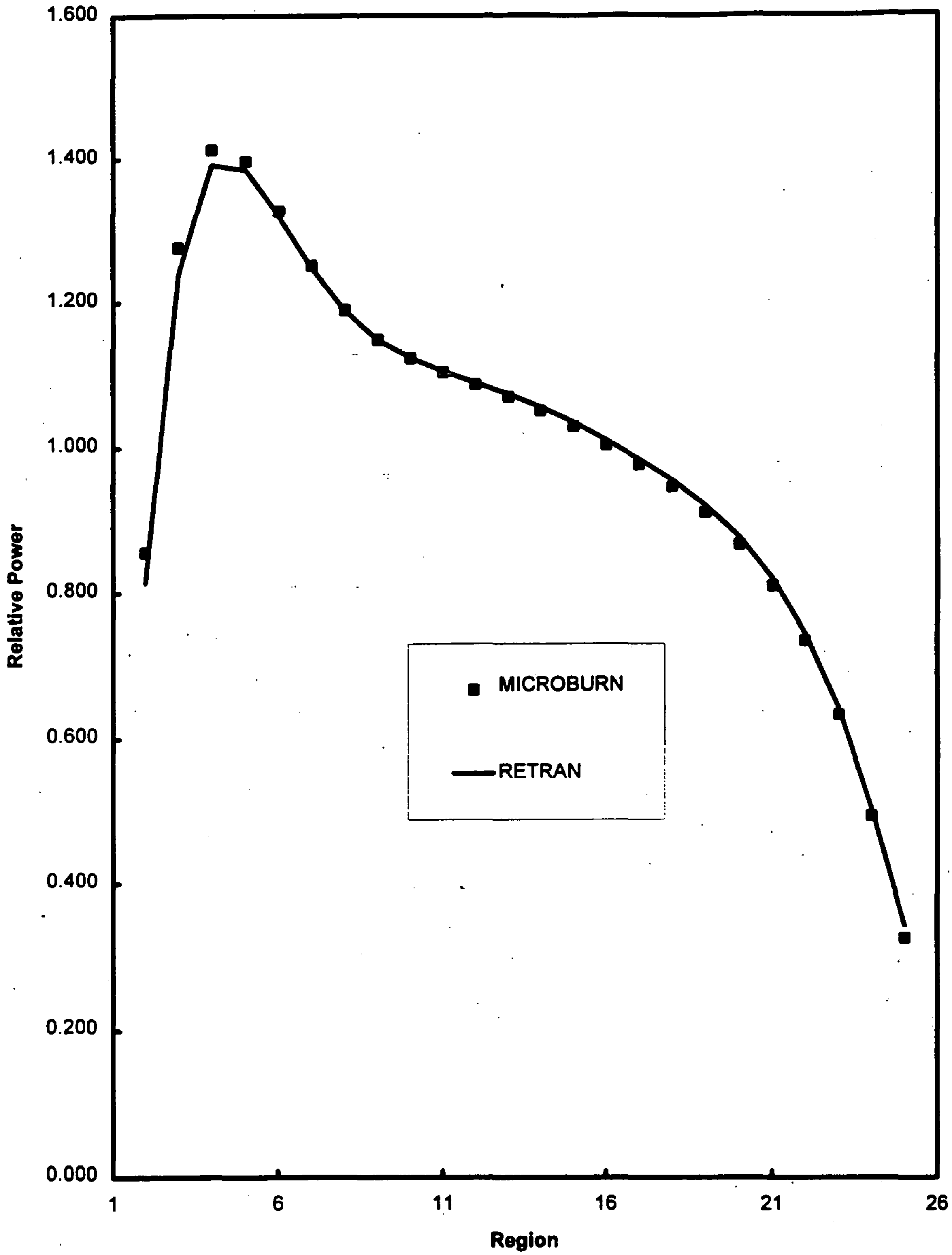


Figure A-6, MICROBURN and RETRAN Axial Power Distribution for PB TTL

Appendix B - Steady State Core Thermal-Hydraulic Methodology

B.1 Introduction

The detailed modeling of a Boiling Water Reactor (BWR) core requires accurate prediction of pressure losses and void distribution. Evaluation of coolant pressure drop is of primary importance in BWRs because of coupling between channel power generation, moderator void content, and channel coolant flow. This appendix provides a summary description of the ComEd steady state thermal-hydraulic methodology.

The ComEd steady state thermal-hydraulic methodology employs FIBWR2, which accurately predicts the pressure drop, flow, and void distributions over a range of power and flow operating conditions in a BWR core. FIBWR2 has its origin in the FIBWR code developed by Yankee Atomic Electric Company (YAEC) (Reference 23). It has the added capability to model new fuel designs that call for axially varying flow geometry and multi-channel thermal-hydraulic transient analysis.

The FIBWR code was reviewed by the NRC and was determined to be acceptable for steady state core flow distribution calculations for Vermont Yankee reload analysis. The NRC review included the FIBWR method of solution, constitutive relationships, data comparison and overall benchmark evaluation (Reference 2). Also, the NRC reviewed the FIBWR topical report submitted by Carolina Power and Light and determined it to be acceptable for reference for the Brunswick reload thermal-hydraulic analysis (Reference 24).

The FIBWR2 development work was sponsored by five utilities: Detroit Edison, Gulf States Utilities, New York Power Authority, Public Service Electric & Gas Company, and Yankee Atomic Electric Company (Reference 25). SCIENTECH, Inc. has acted as the main contractor on the FIBWR2 development work. ComEd purchased the right to use FIBWR2 under a license agreement with YAEC acting as the agent to the owners. The ComEd FIBWR2 code is a version (V1.04) released by SCIENTECH, Inc. that has been qualified for steady state controlled analysis in accordance with ComEd Quality Assurance procedures.

B.2 FIBWR2 Steady State Analysis

The FIBWR2 steady state solution was designed to match that of FIBWR. ComEd has verified the FIBWR2 steady state calculation against the FIBWR code results. The comparison of the FIBWR2 results for Quad-Cities and Dresden Cycle 1 cores show an almost exact match with the results from the FIBWR models of the same cores. The FIBWR2 code results have also been verified against the General Electric (GE) steady state core thermal-hydraulic code ISCOR (Reference 26). The comparison shows very reasonable agreement between the results of the two codes.

The FIBWR2 model of a flow channel accounts for the effects introduced by the inlet orifice, fuel support piece, lower and upper tie plates, lower and upper unheated fuel regions, grid spacers, water tubes, and channel exit. There are three leakage flow paths that are located after the orifice that would allow the flow to bypass the channel. There are up to eight leakage paths that are independent of the channel and are dependent on the core support plate pressure differential. Important to the FIBWR2 modeling are core geometry data, form loss coefficients, bypass leakage pressure drop coefficients, and thermal and hydraulic models.

The ComEd method uses a compressed core representation for steady state thermal-hydraulic calculation. Fuel bundles of the same geometry and hydraulic characteristics (either core central or peripheral bundles) are represented as an average bundle. The bundle axial power profiles are represented by a core average profile. The radial power profiles are represented by nominal values.

The hydraulic model options used in the FIBWR2 are selected to be consistent with RETRAN where applicable. The actual physical dimensions for a specific bundle design model are taken from the fuel vendors' design drawings. The vendor's data is utilized to determine the loss coefficients that are used in the FIBWR2 core modeling of local losses for each bundle design. FIBWR2 calculates the bypass flow by an empirical correlation that determines the flow as a function of the pressure differential, which is driving the flow through the leakage path. These coefficients are calculated by using the vendor's data for a given flow rate and pressure drop.

System parameters such as core exit pressure and inlet enthalpy (or inlet subcooling) are determined through a heat balance calculation. The axial power shape data is case specific and is obtained from MICROBURN-B as discussed in Appendix A. The number of axial nodes used in FIBWR2 is consistent with MICROBURN-B.

B.3 Application

The FIBWR2 steady state analysis has its application in the RETRAN initialization. The parameters that are used for the RETRAN initialization include the single equivalent channel core axial pressure profile, channel-dependent bypass, total bypass flow, and the inlet core pressure. (See Section 3.5.1 for details.)

The FIBWR2 steady state calculations have been qualified and used for initializing the RETRAN Peach Bottom Unit 2 Cycle 2 core model and the models for the initial cores of Quad-Cities, Dresden, and LaSalle. Based on the results of the model benchmarking and comparison to the FIBWR and ISCOR calculations, the FIBWR2 predictions provide the correct data for initialization of the RETRAN model.

The ComEd method, which uses one channel to represent one fuel geometry, is adequate for steady state modeling. It is sufficient to represent fuel bundles of the same geometry by one FIBWR2 channel to reduce computing time.

Appendix C - Gap Conductance Methodology

C.1 Introduction

The pellet to cladding gap conductance is important to the system transient response since it affects the surface heat flux. One of the parameters required as input to the RETRAN core model is the thermal conductivity of the fuel pellet to cladding gap. This parameter is obtained from the pellet to cladding gap conductance analysis. This appendix describes the methodology and the computer code used to generate the gap conductance data for the benchmarking analyses. The methodology to generate gap conductance data for use in licensing analyses will be described in the ComEd thermal limits and application topical.

The ComEd fuel pellet to cladding gap heat transfer analysis uses the ESCORE computer code. ESCORE provides an analytical method of predicting a best-estimate, steady-state thermal-mechanical performance of the fuel rods in Light Water Reactors (LWR). ESCORE has been benchmarked to an extensive fuel data base as described in Reference 27.

ESCORE has been reviewed and evaluated by the NRC as a licensing tool for determining fuel rod pressure, centerline temperature, and input to transient and fuel thermal limits analyses. Based on this review, ESCORE is acceptable for performing steady state LWR fuel performance licensing analyses under the stated conditions in the NRC evaluation. For this benchmarking analyses, ComEd has used ESCORE in accordance with the SER application limits.

C.2 ESCORE Gap Conductance Analysis

The ESCORE analysis is performed using a fuel pin model representing a rod with certain physical characteristics and power. The ComEd RETRAN benchmarking effort involved the Quad-Cities, Dresden, and LaSalle initial cores and Peach Bottom Unit 2 Cycle 2 core. The fuel rod models were set up for GE 7x7 and GE 8x8 in the Peach Bottom Unit 2 Cycle 2 core, GE 7x7 for Quad-Cities and Dresden initial cores, and GE 8x8 for LaSalle initial core.

The primary inputs to ESCORE are fuel rod parameters and the MICROBURN-B predicted core power histories. The gap conductance analysis is based on a best estimate representation of the different fuel types and of fuel bundle average power and exposure histories. The ESCORE input uses 24 axial nodes. Table C.1 shows some key parameters used in the ESCORE fuel pin models.

The gap conductance is calculated for a fuel rod type at a power level over a specific exposure range. Since the nodes with higher stored energy and higher gap conductance contribute to a relatively greater extent in adding thermal energy to the coolant, the individual nodal gap conductance values are weighted by the corresponding stored energy and the gap conductance value itself. These axially varying gap conductance values are further weighted by the number of rods at a given power. The final result is a constant gap conductance that is calculated by averaging the axially varying gap conductance.

C.3 Results

The ESCORE calculations are performed using the Peach Bottom Unit 2 Cycle 2 power history predicted by MICROBURN including the state point at which a turbine trip test is initiated. Similar calculations are performed using the MICROBURN-B predicted power histories including the state point at which a startup test event is initiated for the Quad-Cities, Dresden, and LaSalle initial cores.

From the ESCORE analysis an average gap conductance is determined consistent with the core power and exposure where the Peach Bottom Turbine Trip Tests and the startup tests for Quad-Cities, Dresden, and LaSalle are conducted. The ComEd method generates a constant gap conductance appropriate for use in the RETRAN core model. Table C.2 shows the average gap conductance values calculated for the three turbine trip tests conducted during the Peach Bottom Unit 2 Cycle 2 core operation. Tables C.3 and C.4 show the average gap conductance values for the selected startup tests performed on the initial cores of Quad-Cities, Dresden, and LaSalle.

A study was performed to demonstrate the applicability of an axially uniform gap conductance for the RETRAN system model analyses. This study, conducted for the RETRAN Peach Bottom core model, showed that the rate of heat deposited to the coolant for a transient like Turbine Trip Test 3 would be the same whether an axially varying or an axially uniform gap conductance was used. (See Section 3.6 for details.)

Table C.1 Key Parameters in ESCORE Fuel Pin Models

	Peach Bottom GE7x7	Peach Bottom GE8x8	Quad-Cities/ Dresden GE7x7	LaSalle GE8x8
Fuel Height (inch)	144.	144.	144.	150.
Clad Length (inch)	160.	160.	156.	162.1
Clad OD (inch)	0.563	0.493	0.563	0.483
Clad ID (inch)	0.489	0.425	0.499	0.419
Pellet OD (inch)	0.477	0.416	0.488	0.410

Table C.2 Peach Bottom Unit 2 Cycle 2 Core Gap Conductance Data

Test Description	Gap Conductance (BTU/HR-FT^2-F)
Turbine Trip Test 1	519
Turbine Trip Test 2	600
Turbine Trip Test 3	619

Table C.3 Gap Conductance Data for Quad-Cities and Dresden Initial Cores

Startup Test	Unit	Burnup (GWD/MTU)	Gap Conductance (BTU/HR-FT^2-F)
Turbine Trip With Bypass	QC2	0.29	464
Pressure Regulator Setpoint Change	QC1	0.29	400
Two Recirculation Pump Trip	DR3	0.29	590
Reactor Water Level Setpoint Change	QC1	0.29	587

Table C.4 LaSalle Initial Core Gap Conductance Data

Startup Test	Unit	Burnup (GWD/MTU)	Gap Conductance (BTU/HR-FT^2-F)
Pressure Regulator Setpoint Change	LS2	0.85	751
Generator Load Rejection	LS2	1.08	723
Two Recirculation Pump Trip	LS2	0.50	522
Reactor Water Level Setpoint Change	LS2	1.05	720
Main Steam Line Isolation Closure	LS2	1.05	726