RETRAN — A Program for One-Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems

CCM-5
Volume 4: Applications
Research Projects 342 and 889

Computer Code Manual, December 1978

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ABSTRACT

RETRAN represents a new computer code approach for analyzing the thermal-hydraulic response of Nuclear Steam Supply Systems (NSSS) to hypothetical Loss of Coolant Accidents (LOCA) and Operational Transients. In contrast to the "conservative" approach, RETRAN provides "best estimate" solutions to hypothetical LOCA's and Operational Transients. RETRAN is a computer code package developed from the RELAP series of codes, from reference data, and from extensive analytical and experimental work previously conducted relative to the thermal-hydraulic behavior of light-water reactor systems subjected to postulated accidents and operational transient conditions. The RETRAN computer code is constructed in a semimodular and dynamic dimensioned form where additions to the code can be easily carried out as new and improved models are developed. This report (the fourth of a four volume computer code manual) describes the extensive verification and qualification performed with RETRAN. The three companion volumes describe the theory and numerical algorithms, the programming details, and the user's input information.

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ACKNOWLEDGEMENT

The formal release of the RETRAN code represents the culmination of a three year effort to produce a verified and qualified computer code which can be used by the electric utility industry to evaluate nuclear power plants. This volume, "RETRAN Applications," presents the results of analyses which have been performed during the pre-release phase of the development project.

A number of organizations and individuals have made contributions to this project. Due to the team nature of this effort, it is sometimes difficult to give proper credit to the proper persons. The organizations and individuals who authored the reports summarized in this volume are listed in Table 1.

Volume IV provides documentation on the verification and qualification phase of the project. The actual code and accompanying documentation (Volume I - Theory Manual, Volume II - Programmer's Guide, and Volume III - User's Manual) are equally important parts of the RETRAN code package.

Many people have contributed to the overall effort of code shakedown, model development, and documentation. The assistance provided by members of the Utility Working Group, Intermountain Technologies Incorporated, and our colleagues at EPRI and EI is gratefully acknowledged. We are especially grateful for the contributions made by Ken Moore, the previous project manager at EI, and Jim Harrison and Rich Hentzen, who had the responsibility to coordinate the activities of the Utility Working Group.

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TABLE 1

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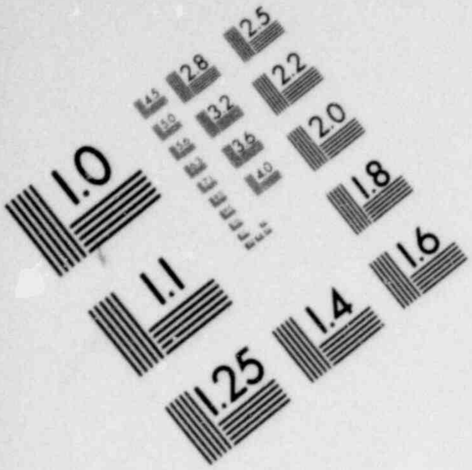

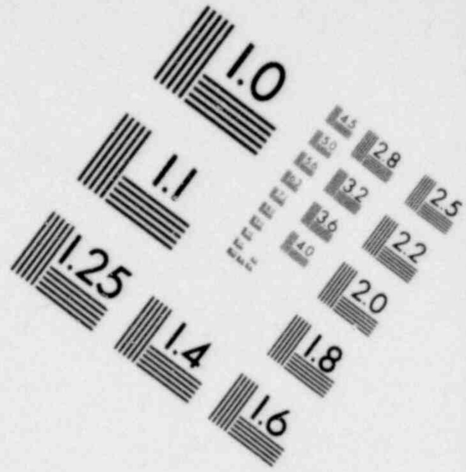

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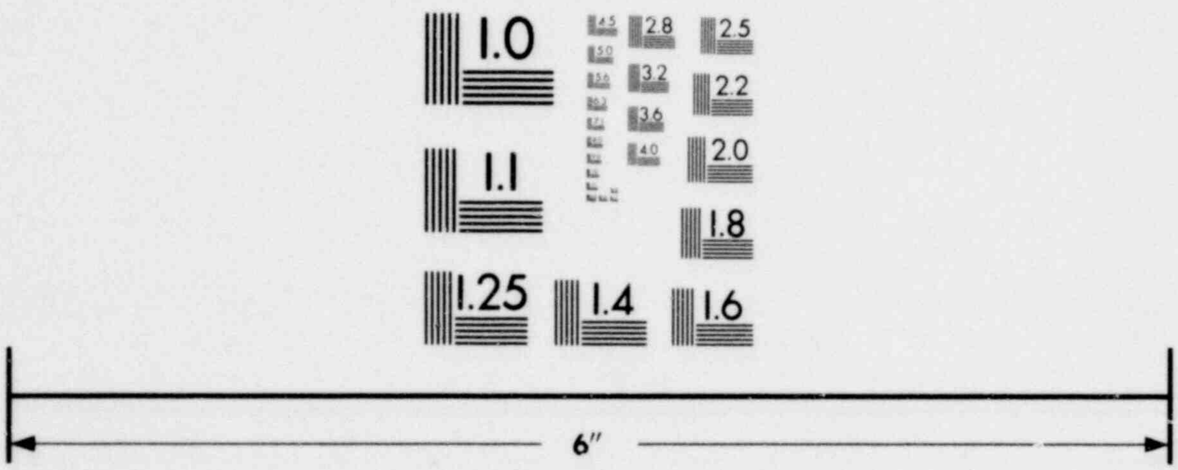

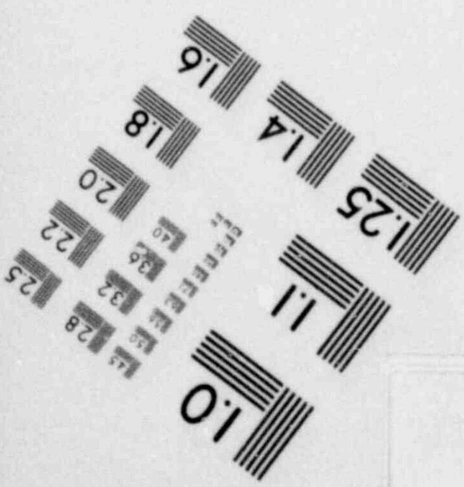

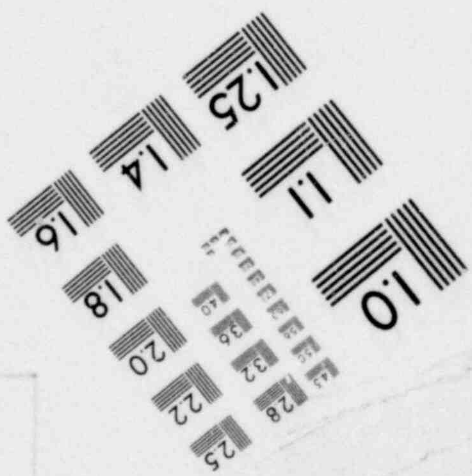

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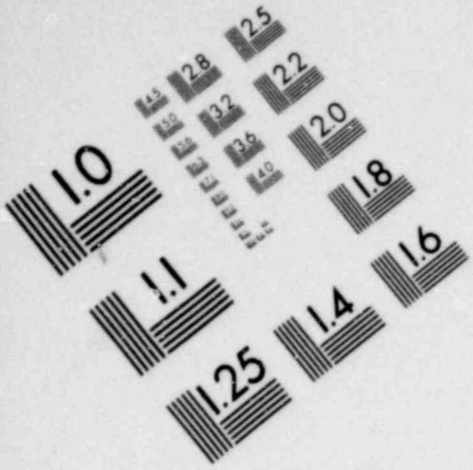

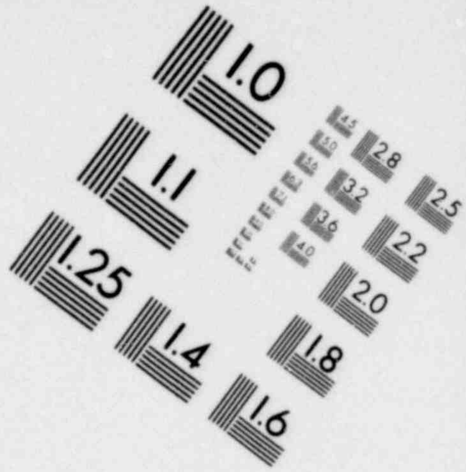

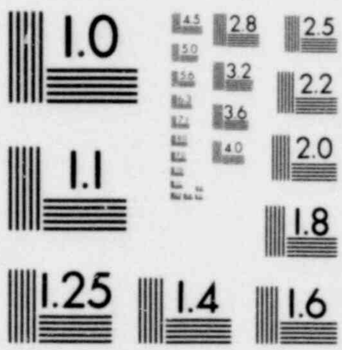

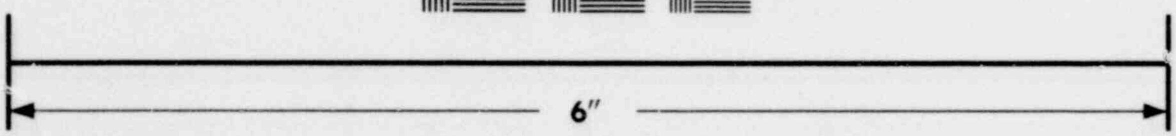

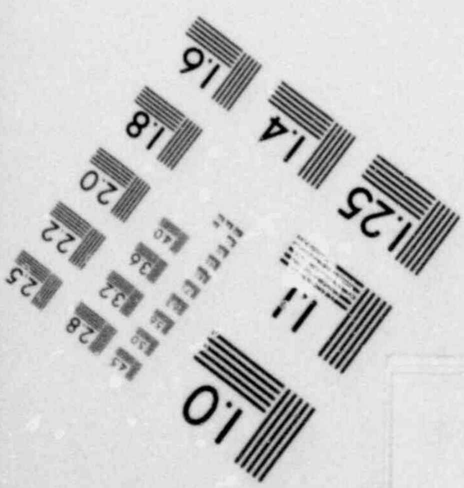

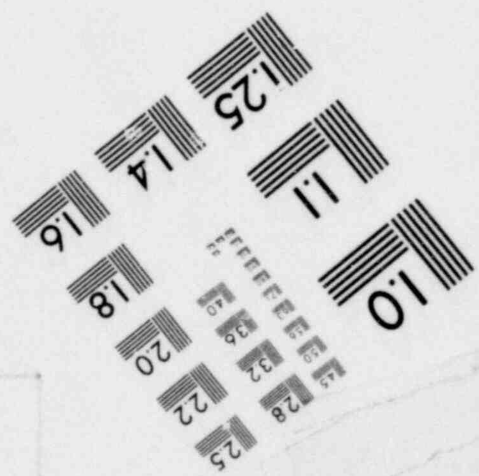

IMAGE EVALUATION TEST TARGET (MT-3)

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I. INTRODUCTION

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I. INTRODUCTION

The objective of this document is to describe the criteria and verification activity which preceded the formal EPRI release of the RETRAN Code. Specifically, the assembly verification process is carefully defined and details of the level of qualification against experimental data are shown.

1.0 OBJECTIVE OF RETRAN DEVELOPMENT

The overall objective of the RETRAN project (RP-342 and RP-889) was to develop an improved and reliable thermal-hydraulic program for analysis of light water reactor system transients. RETRAN was developed primarily for:

(1) Utitlity use in

- Evaluating and improving design and operation of reactor plants
- Evaluating safety considerations
- Support of licensing submittals

and for

(2) EPRI and EPRI contractors use in

- Interpretation of safety/operational related experiments and analysis
- Generic evaluation of
  - safety issues
  - proposed regulations
  - new concepts

and secondarily for

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- Defining research support

    - What research is to be performed

    - Support research implementation

    - Interpreting research results.

The minimum modeling requirements consistent with the above objectives are one-dimensional, homogeneous thermal-hydraulic models for the reactor cooling system, a point neutron kinetic model for the reactor core, and control system models. This level of sophistication permits analysis of most Safety Analysis Report (SAR) Chapter 15 incidents (except the reflood portion of a LOCA) for both boiling water reactor (BWR) and pressurized water reactor (PWR) plants. These analyses utilize the basic RETRAN models, for which a relatively large experience base exists. The minimum model requirements given above are satisfied by the current version of RETRAN.

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2.0 LIMITATIONS

In principle, the current version of RETRAN has the capability to model all SAR Chapter 15 incidents for both BWR and PWR applications. The accuracy of the BWR analysis may be limited by the homogeneous flow models and/or the point kinetics models. Because of these limitations, EPRI is in the process of developing an advanced version of RETRAN which has both a slip model and a one dimensional neutron kinetics model for BWR applications.

Likewise the analysis of the reflood portion of a LOCA requires additional thermal hydraulic capability which models

(1) slip between the liquid and gas phases of the coolant

(2) the non-equilibrium, emergency core cooling injection process

(3) the quenching process.

These applications require advanced models for which only a very limited experience base exists, and such models are not included in the present version of the code.

Because RETRAN is a very general thermal-hydraulic code, no assurance of accuracy can be made in general. Therefore, it is recommended that the application of RETRAN be limited to those plant transients for which some degree of confidence has been obtained. The results of the current qualification effort are summarized in Section 6.0, which lists most Chapter 15 incidents requiring system analysis and shows the current application levels of RETRAN. The RETRAN results are classified according to the following categories:

(1) direct comparison with experiments

(2) extrapolated from similar but different experiments

(3) results appear physically reasonable.

Application to other analysis not shown in Section 6.0 as of yet is uncertain.

3.0 RELEASIBILITY CRITERIA

Because RETRAN is a new computer program, the confidence in, and the limitations of, the program must be established for potential users of the program. The degree of confidence in any computer program stems from two different processes,

(1) Documentation must exist that

  (a) describes the theory and assumptions made in developing the code

  (b) describes the code models, logic and solution schemes

  (c) describes in detail how to use the code

(2) The code must be verified to assure

  (a) the coding is correct

  (b) the solution techniques are stable and convergent

  (c) the code is correctly solving the equation set programmed.

(3) The code must be qualified to perform the analyses to be performed by

  (a) comparison with relevant test data

  (b) comparison against other calculation techniques

  (c) assuring that all results are consistent with physical assumptions made.

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4.0 RETRAN DOCUMENTATION

The first criteria for the release of a code is adequate documentation. The RETRAN program is documented in this four volume Computer Code Manual (EPRI CCM-5, Volumes I-IV) entitled "RETRAN - A Program for One-Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems". Volume I - Equations and Numerics, satisfies requirement 1-a and part of 1-b as it gives a detailed description of the theory and nummerics used in RETRAN. (An earlier edition of this volume appeared as EPRI NP-408 and describes some models which are not included in the current version of RETRAN). The remaining part of requirement 1-b, the RETRAN code logic and detailed programming description, is given in Volume II - The Programmer's Manual. Requirement 1-c is satisfied by Volume III - User's Manual, which describes the code input, output and gives a series of sample problems to assist the user. The remaining two criteria (2-3) are the subject of this volume. The next two sections of this chapter summarize the RETRAN verification (criteria 2) and qualification (Criteria 3). The details of both the verification and qualification analyses performed to date are described in Chapters III-VIII. A summary of the experience of users of the code, along with specific modeling suggestions, is given in Chapter IX.

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5.0 RETRAN VERIFICATION

The verification process (Criteria 2, Section 3.0) was undertaken to assure that the computer program performed the calculation and processed the data and input as required by the documentation.

First, to assure that the models were documented properly, all the RETRAN models were listed (Section II.4) and the basic technical reference identified. Next the original reference was reviewed and compared with the RETRAN documentation to assure consistency.

The next step was to assure proper coding of each individual model. Four semi-independent methods of accomplishing this were identified. These are

(1) An independent visual check of the coding by someone other than the original programmer

(2) Specifying input (driving the model) that produces known results

(3) Editing all required information entering a model and hand calculating the output

(4) Comparing results with output from other codes.

A list of the RETRAN models was used to assure that each individual RETRAN model was verified against at least two of the four methods.

Finally a series of "basic" test problems were run for which either analytical solutions or detailed experimental results were available. Again, the list of RETRAN models was cross-correlated against this list of analyses to assure that all common mode models were correctly functioning. The results of some of this activity was useful in the qualification of the code, which is summarized in the next section.

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6.0 RETRAN QUALIFICATION

The qualification of a code involves comparisons of code results with experimental data to determine its applicability and sensitivity. The RETRAN code was qualified aga three different classes of data.

The first, and simplest, were separate effect experiments which, in general, are small scale tests in which complexities are held to a minimum and the governing parameters accurately measured. The results of these analyses produce some level of confidence in both the individual and combinations of models utilized. Chapter III describes the qualification of RETRAN against the separate effect tests.

The second category of experiments are System Effects Tests, in which the interaction of various components must be described. In general, these are intermediate size tests (i.e., Semiscale, Two-Loop Test Apparatus, LOFT) in which the assumption of one-dimensional streamline flow is reasonably approximated. The RETRAN qualification against experiments of this type is more demanding than for the separate effect tests, and gives confidence in, and implies limitations on, the basic theory used in RETRAN. The results of this class of experiments are given in Chapter IV.

The third and most important class of comparisons is against actual plant data. The data from large nuclear plants is generally obtained from operational instrumentation, and is usually limited in both quantity and quality. The actual comparisons are also made more difficult by the unavailability of all required input data, specifically data on items such as the initial conditions, times manual operator action was taken, and response characteristics of valves. This phase of the RETRAN qualification was performed mainly by the EPRI/Utility System Analysis Working Group.

A description of this Working Group is included in Chapter II. This group was primarily interested in the analysis of the operational transients generally addressed in Chapter 15 of most SARs. Each of the participating utilities identified both analysis of interest and some existing reactor start-up and/or operating test data against which the RETRAN results could be qualified. The results of this effort are presented in Tables I.6-1 and I.6-2, which also list

TABLE I.6-1

RETRAN QUALIFICATION FOR SOME SAR-CHAPTER 15 BWR TRANSIENTS

	Direct Comparison	Similar Phenomena	SAR-Other Calculations	See Footnote	Comments
PRESSURE INCREASE TRANSIENTS					
° Generator load rejection	X			1	
° Turbine trip with/without bypass	X		X	2	Peach Bottom Tests
° Steam Line isolation valve(s) closure					
° Pressure regulator failure (close)					
° Loss of condenser vacuum					
° Turbine control valve fast closure					
° Above incidents with delayed scram					
REACTOR VESSEL WATER TEMPERATURE DECREASE					
° Loss of feedwater heater					
° Inadvertent injection pump start					
REACTOR VESSEL COOLANT INVENTORY DECREASE					
° Loss of feedwater flow	X			1	
° Pressure regulator failure (open)					
° Relief or safety valve failure (open)					
° Loss of auxiliary power					
CORE COOLANT FLOW DECREASE					
° Recirculation pump seizure					
° Recirculation pump(s) trip	X			1	
° Recirculation flow control failure					
CORE COOLANT FLOW INCREASE					
° Recirculation flow control failure					
° Startup of idle recirculation pump					
POSITIVE REACTIVITY INSERTION					
° Continuous rod withdrawal					
° - from subcritical					
- from power					
° Rod ejection					
ANTICIPATED TRANSIENT WITHOUT SCRAM					
STEAM LINE BREAK					
RECIRCULATION LINE RUPTURE					
° Large break					
° Small break					

1. Startup test
2. SAR comparison
3. Other code comparison

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TABLE I.6-2

RETRAN QUALIFICATION FOR SOME PWR SAR-CHAPTER 15 TRANSIENTS

	Direct Comparison	Similar Phenomena	SAR-Other Calculations	See Footnote	Comments
UNPLANNED DECREASE IN SECONDARY HEAT REMOVAL					
° Loss of external load			X	2	
° Turbine trip					
° Loss of condenser vacuum					
° Steam pressure regulator failure					
° Loss of normal feedwater flow					
° Loss of A-C power to auxiliaries					
UNPLANNED INCREASE IN SECONDARY HEAT REMOVAL					
° Excessive load increase					
° Idle loop startup					
° Decrease in feedwater temperature					
° Increase in feedwater flow rate					
° Increase in steam flow rate					
° Inadvertent opening of steam generator relief or safety valve	X	X			TMI-2 Cooldown
CHANGES IN REACTOR COOLANT SYSTEM INVENTORY (PRIMARY SIDE INITIATED)					
° Inadvertent operation of ECCS					
° Accidental depressurization					
LOSS OF REACTOR COOLANT FLOW					
° Partial loss of flow		X		1	Pump Coastdown Test
° Complete loss of flow		X	X	1,2	
° Locked rotor					
REACTIVITY INSERTION (PRIMARY SIDE INITIATED)					
° Uncontrolled rod withdrawal					
- from subcritical					
- from power			X	2	
° Control rod misoperation					
° Chemical system malfunction					
ANTICIPATED TRANSIENTS WITHOUT SCRAM					
STEAM LINE BREAK			X	2	
RECIRCULATION LINE BREAK					
° Steam generator tube rupture					
° Small break					
° Loss of coolant accident			X	2	

1. Startup test
2. SAR comparison
3. Other code comparison

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all Chapter 15 incidents for BWRs and PWRs respectively. These tables also indicate those transients which were analyzed and the degree of qualification of the RETRAN analysis which was possible. The analyses are classified according to

(1) Direct comparison with experiments

(2) Extrapolated from similar but different experiments

(3) Results appear reasonable and agree with available SAR results or calculations from other codes.

The first category implies that the accuracy of the results were determined by direct comparison with experimental data for some (one or more) cases where such data exist. The second category includes cases where direct experimental confirmation does not exist, but confidence can be established in the calculation based on other transients which are governed by similar phenomena. The third category is one where no experimental data were available and the results were reviewed only to assure that they appeared physically correct.

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II. RETRAN VERIFICATION AND QUALIFICATION

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II. RETRAN VERIFICATION AND QUALIFICATION

Users of large computer codes expect

(1) The code to be free of errors to the greatest extent possible

(2) The code to be able, with appropriate input, to give reasonable agreement with experimental data when applied within theoretical limitations.

In the RETRAN project, a significant effort has been directed to meeting these expectations.

The first step, code verification, is best accomplished by very strict procedures during the actual coding. However, RETRAN is based on an existing RELAP code and, as a result, only complete model additions to the code were subjected to close scrutiny during the coding phase. Members of the EPRI/Utility Working Group have exercised the code with a great variety of problems, and have helped to identify errors in the code.

The qualification of the code is actually a measure of how well the code can analyze problems of interest. The EPRI/Working Group, EPRI and EPRI contractors have performed a number of analyses and compared results with experimental data. The code verification for RETRAN and the approach used to qualify the code are discussed in this section, along with some background information on the development of the code.

1.0 BACKGROUND OF RETRAN

The RETRAN computer program is the result of an extensive code development effort sponsored by EPRI since 1975. The effort was initiated as research project RP-342 in response to the utility need for a more realistic appraisal of the blowdown phase of the design basis Loss Of Coolant Accident (LOCA), and EPRI's need to evaluate relevant supporting experiments. During the term of this effort, the project was expanded such that the computer program, then denoted as RELAP/E, would be general enough to analyze both boiling water reactors (BWRs) and pressurized water reactors (PWRs) for either large or small break

LOCA's from the start of the accident through the REFILL and REFLOOD periods. During the first quarter of 1976, the Nuclear Safety Analysis Task Force identified the pressing need of the utilities to analyze the non-LOCA condition events for PWRs (Types I, II and III) and BWRs (Types I through VIII). In response to this request, EPRI obtained from Energy Incorporated the RETRAN system analysis submodules through RP-889. Thus the RETRAN code package stems from the development of two separate code packages, RELAP/E and RETRAN. Both of these codes were based upon RELAP4/003 update 85, released by the United States Nuclear Regulatory Commission (NRC) as a portion of the Water Reactor Evaluation Model (WREM). RELAP/E was developed to provide a "best estimate" thermal-hydraulic behavior of light-water reactor systems subjected to anticipated operational transients and normal startup and shutdown maneuvers. Since both codes were based on the thermal-hydraulic differential and state equations of RELAP4, and since RELAP/E was constructed for ease of model incorporation with its semimodular and dynamic structure, the operational transient models were added as options to RELAP/E and the code name RETRAN was retained.

During this time period (late 1976) the importance of the code verification effort first became apparent to EPRI. It was estimated that a minimum of 500 hours of CDC-6600 time would be required to verify this code package, in addition to an extensive amount of manpower to set up and execute the cases. It was also apparent that, when the verification phase was complete, there would be an additional delay in implementing RETRAN on the utility computers and in training their personnel in the use of this rather sophisticated computer code package. The EPRI/Utility System Analysis Working Group was established to attempt to combine these various tasks and shorten the overall time between development and application of RETRAN. Table II.1-1 gives a short summary of the overall intent of the RETRAN pre-release activity, while Table II.1-2 lists the current participants in the Working Group.

The first phase of this activity is now complete, and a version of RETRAN is to be released for EPRI members. This version of the code has been used to analyze separate effect experiments (Section III), small scale system effects (Section IV), operational transients (Sections V and VI) and PWR LOCA's (Section VIII). During the next year, additional analyses, including BWR LOCA's will be performed, and this document will be modified to include the additional work.

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TABLE II.1-1

A SUMMARY OF THE INTENT OF A PRE-RELEASE OF RETRAN

1. Provide participating utilities with RETRAN so they can become familiar with and competent in its use.

2. Obtain, via utility participation, a much more thorough "debugging" of RETRAN than EPRI can provide under conventional project effort.

3. Exercise RETRAN against a wide series of problems typical of utility application.

4. Qualify RETRAN against existing plant data and other analyses.

5. Reduce the confusion associated with implementing a large computer program by inexperienced users.

6. Accumulate results of RETRAN analysis from a wide-based application effort.

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TABLE II.1-2

EPRI/UTILITY SYSTEM ANALYSIS WORKING GROUP PARTICIPANTS

Consolidated Edison Company of New York
Consumers Power Company
Detroit Edison Company
Florida Power and Light Company
General Public Utilities Service Corporation
Northeast Utilities Service Company
Long Island Lighting Company
Pacific Gas and Electric Company
Philadelpha Electric Company
Portland General Electric
Power Authority of the State of New York
Public Service Electric and Gas Company
Southern California Edison Company
Tennessee Valley Authority
Washington Public Power Supply System
Virginia Electric and Power Company
Yankee Atomic Electric Company

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2.0 RELEASABILITY CRITERIA

Because RETRAN is a new computer program, confidence in, and the limitations of, the program must be established for potential code users. There are many individual factors necessary to establish confidence in a code. Defining what is a reasonable effort, coupled with the complex concept of a large computer code, make this a formidable problem. For simplicity, let us restrict our attention to one specific part of the computer code, a particular model. For any individual model one must address the following questions:

(1) What is the "design" function of the model?

(2) What are the general limitations of the model, i.e., the limiting theoretical assumptions and the range of applicability of the data base?

(3) What is the specific formulation of the model, i.e., what are the closure assumptions made and the constitutive models utilized?

(4) What solution technique was utilized and how dependent is this on time steps or spatial representation?

(5) What is past experience with this or similar models?

At this point the concept of "consistent application" and "extended application" must be made. An application of a model will be denoted as being "consistent" if the application does not violate any of the basic assumptions made in (1) through (4) above. An "extended" application is one that violates any of the basic restrictions given in (1) through (4) above. The fifth item above relates to previous experience associated with any model. Some models have been used extensively with satisfactory results while others are relatively new and have only been tested against a limited data base. The user must have some information regarding the confidence level of each model. For a particular model, one should expect reliable "consistent applications" and hopefully some limited "extended application". However, any "extended application" must be recognized as highly speculative and should not be expected to produce reasonable results.

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From this discussion, two requirements emerge, (1) the computer code must be completely described and (2) the confidence level of the code must be indicated. The degree of confidence in any computer program stems from two different processes, (1) the assembly verification process in which each submodel and the entire code are quality assured and (2) the qualification of the computer code against experimental data.

These considerations have led to the following releasability criteria:

(1) Documentation must exist that

   (a) Describes the theory and assumptions made in developing the code

   (b) Describes the code models, logic and solution schemes

   (c) Describes in detail how to use the code

(2) The code must be verified to assure

   (a) The coding is correct

   (b) The solution technique is stable and convergent

   (c) The code is correctly solving the equation set programmed

(3) The code must be qualified to perform the analysis required of it by

   (a) Comparison with relevant test data

   (b) Comparison against other calculation techniques

   (c) Assuring that all results are consistent with physical assumptions made.

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3.0 RETRAN DOCUMENTATION

The first criteria for the release of a code is adequate documentation. The RETRAN program is documented in this four volume Computer Code Manual (EPRI CCM-5, Volumes I-IV) entitled "RETRAN - A Program for One-Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems". Volume I - Equations and Numerics, satisfies requirement 1-a and part of 1-b as it gives a detailed description of the theory and numerics used in RETRAN. (An earlier edition of this volume appeared as EPRI NP-408 and decribes some models which were not released in the current version of RETRAN). The remaining part of requirement 1-b, the RETRAN code logic and detailed programming description, is given in Volume II - The Programmer's Manual. Requirement 1-c is satisfied by Volume III - User's Manual, which describes the code input, output and gives a series of sample problems to assist the user. The remaining two criteria (2-3) are directly addressed by this volume. Specifically, the next two sections of this chapter describe in detail the extent of the RETRAN verification (Criteria 2) and summarizes the qualification (Criteria 3). The remaining Chapters of this report detail the qualification analysis performed to date.

4.0 RETRAN VERIFICATION

In the development of any product the question inevitably arises as to how much quality assurance is required to produce an acceptable product. This question is especially difficult in the area of software products (computer codes). The basic consideration is not related to the usefulness or accuracy of the product, but more towards defining a meaningful and competent level at which to stop the verification process. In general, any computer code is designed to generate numbers by using a selected set of algorithms in some logical, predetermined manner. Initially, the computer code goes through a debug stage in which the programmer attempts to remove all coding errors and to establish that the model is working as required. Then the code is utilized by others for problem solving. Because of the extremely large number of logic paths possible in any modern computer program, the user inevitably discovers certain paths which produce non-physical or absurd answers. The main objective of this section is to describe the procedure used in the assembly verification process of RETRAN. This section describes why RETRAN can be reasonably expected to correctly generate numbers according to the prescribed logic, algorithms and formulas.

The RETRAN verification process was an extensive effort in attempting to assure that the computer program is performing the correct calculation and processing the data and input as required. The first step in this activity was to assure that all individual models were adequately documented. Table II.4-1 shows how this was accomplished. The first column tabulates the type of physical process represented while the second column lists the available RETRAN models. The next column identifies the Volume I reference which was used to confirm the particular models. The last column is a short description of the model to assist others in understanding and, if necessary, repeating this process. Note that the original references were reviewed and compared with the RETRAN documentation to assure consistency.

The next step was to assure proper coding of each individual model. Four semi-independent methods of accomplishing this were identified. These are

(1) An independent visual check of the coding by someone other than the original programmer

TABLE II.4-1

RETRAN MODEL DESCRIPTION

Type	Model	Volume I Page Reference	Description
Momentum Equation Volume Mass Flow	Average of Inlet and Outlet Flows	VIII-12	Approximate value of volume mass flow based on mass flow rates at adjacent junctions.
Momentum Equation Momentum Flux Term	Methods of Computing Momentum Flux	II-35 to II-43	Methods of treating momentum flux, including zero values, incompressible and compressible flow.
Momentum Equation Junction Area Change	Local Incompressible or Compressible Flow	II-35 to II-43	Option to calculate flow at junction with variable area assuming either incompressible or compressible flow at the junction.
Momentum Equation Wall Friction Force	Fanning Friction Factor	III-2 to III-5	Equations to compute friction for wall friction force term.
Momentum Equation Wall Friction Force	Homogeneous Correlation	II-35 to II-43 III-2 to III-6	Two-phase multiplier for Fanning friction factor based on assumption that flow is homogeneous.
	Baroczy Correlation	II-35 to II-43 III-2 to III-12	Empirically-determined two-phase multiplier for single mass flux with correction factor which is dependent on mass flux and a property index.
	Beattie Correlations	II-35 to II-43 III-2 to III-5 III-13 to III-17	Two-phase friction multipliers based on flow regime. The flow regime is determined from a modified version of the Bennett flow regime map.

TABLE II.4-1 (Cont'd)

Type	Model	Volume I Page Reference	Description
Momentum Equation Loss Coefficients	Loss Coefficients	II-35 to II-43 III-16	Momentum loss associated with area changes in a momentum equation cell.
Energy Equation Wall Heat Transfer	Dittus-Boelter Correlation	III-44 to III-45	Coefficient for single-phase, forced-convection heat transfer.
	Thom Correlation	III-45 to III-46	Heat flux for forced convection, fully developed, nucleate boiling.
	Schrock-Grossman Correlation	III-46 to III-49	Coefficient for forced convection vaporization heat transfer.
	McDonough, Milich and King Correlation	III-49 to III-50	Coefficient for forced convection transition boiling heat transfer.
	Groeneveld 5.7 and 5.9 Correlations	III-52 to III-54	Coefficient for forced convection stable film boiling heat transfer (user option).
	Dougall-Rohsenow Correlation	III-54	Coefficient for forced convection stable film boiling heat transfer (user option).
	Berenson Correlation	III-50 to III-52	Coefficient for pool transition boiling heat transfer.
Energy Equation Condensing Heat Transfer	Uchida Correlation	III-69 to III-70	Based on core spray experiments for vertical surfaces.

TABLE II.4-1 (Cont'd)

Type	Model	Volume I Page Reference	Description
	Tagami Correlation	III-70 to III-76	Based on steady-state and transient experiments in cylinders.
	CSB 6-1	III-76 to III-77	NRC equations used for containment pressure calculations.
Energy Equation Critical Heat Flux	B&W-2 Correlation	III-58	Critical heat flux for forced convection boiling when pressure is above 1500 psia.
Energy Equation Critical Heat Flux	Barnett Correlation	III-59 to III-61	Critical heat flux for forced convection boiling for pressures between 1000 and 1300 psia.
	Modified Barnett Correlation	III-61 to III-62	Critical heat flux for forced convection boiling when pressure is below 725 psia.
	Interpolation	III-62	Critical heat flux for $1300 \leq p \leq 1500$ and $725 \leq p \leq 1000$ based on above correlations.
Energy Equation Junction Enthalpy	Enthalpy Transport	II-44 to II-45 III-63 to III-69	Used to compute enthalpy at the junctions of conventional RETRAN volumes (cells for the energy equation).
Energy Equation Kinetic Energy	Volume-Averaged Flow	II-44 to II-45	Kinetic energy term based on volume average mass flows.

TABLE II 4-1 (Cont'd)

Type	Model	Volume I Page Reference	Description
State Equation Thermodynamic Properties	Tabulated Values	II-28 to II-29 II-45 to II-46 VIII-22 to VIII-32	Tabulated values of ASME equations.
State Equation Transport Properties	Viscosity	II-29	Tabulated values of ASME equation for vapor phase and saturation values for liquid phase.
	Conductivity	II-29	Tabulated values of ASME equations.
State Equation Void Fraction	Homogeneous Equilibrium	III-13	Assumes liquid and vapor phases have the same velocity and are in thermodynamic equilibrium.
State Equation Junction Pressure	Momentum Equation	II-48	Approximate solution of steady-state momentum equation to compute pressure at junctions.
State Equation Junction Enthalpy	Energy Transport or Donor Cell	II-48	Methods of determining enthalpy at junctions for STATE calls. Donor cell is the default model used.
Critical Flow	Sonic Choking	III-79 to III-83 III-91	Calculation of sound speed assuming homogeneous equilibrium.

TABLE II.4-1 (Cont'd)

Type	Model	Volume I Page Reference	Description
	Henry-Fauske and Extended Henry-Fauske	III-84 to III-89 III-91 to III-93	Tabulated values of stagnation enthalpy and critical flow based on equilibrium quality and stagnation pressure.
	Moody	III-89 to III-91 III-93 to III-94	Tabulated values of stagnation enthalpy and critical flow based on equilibrium quality and stagnation pressure.
Solution Techniques	FLASH-4	VIII-1 to VIII-22	Explicit solution of finite-differenced flow equations.
	Causal Volume	VIII-43	Linear solution of pressure equation where pressure is changing slowly in time.
	Steady-State	VIII-68 to VIII-100	Iterative solution of steady-state equations.
Heat Conduction	Two-Sided Heat Conduction	IV-1 to IV-2 VIII-47 to VIII-61	Solves conduction equation with fluid volumes on one or both sides of conductor based on surface boundary conditions.
	Causal Conductor	VIII-43 to VIII-46	Approximate solution of conduction equation for specific heat flux, fluid flow, wall temperatures, fluid temperature and power conditions.

TABLE II.4-1 (Cont'd)

Type	Model	Volume I Page Reference	Description
Energy Generation	Tabulated Values	None Required	User-supplied table of normalized power vs. time for the problem.
	Point Kinetics	V-1 to V-14 VIII-70 to VIII-75	Runge-Kutta solution of point kinetics model for user supplied delayed neutron values and decay heat values.
	Metal-Water	V-15 to V-18	Calculates energy-generation associated with zirconium-steam chemical reaction.
	Direct Moderator Heating	V-19 to V-20	Calculates energy associated with gamma and neutron heating.
Momentum Equation Component	Centrifugal Pumps	VI-1 to VI-17	Models used to compute momentum losses associated with flow through pumps.
Momentum Equation Component	Valves	V-18 to VI-21	Used to compute momentum losses associated with flow through valves.
Energy Equation Component	Non-Conducting Heat Exchangers	VI-22 to VI-24	Used to compute energy exchange between primary and secondary side of heat exchanger.
Reactor Trip System	Trips	VI-25 to VI-28	Simulated trip logic of reactor system

TABLE II.4-1 (Cont'd)

Type	Model	Volume I Page Reference	Description
Control System Control Blocks	Digital Control Blocks	VII-1 to VII-9 IX-10 to IX-11	Digital models of control elements which can be combined to model controllers.
Pressurizer	Non-equilibrium pressurizer	VII-10 to VII-20 IX-9	Solves energy equations in pressurizer, assumming non-equilibrium thermodynamic conditions.
Special Model	Bubble Rise	III-16 to III-35	Used to account for phase separation and to compute partial density of vapor phase for momentum equation gravity term.
Flow Regime Map	Modified Bennett	III-13 to III-15	Determines two-phase flow regime based on mass flux and void fraction.
Operational Transients Transport Delay	EI Model	VII-10 to VII-12 IX-8	Computes junction enthalpy for movement of a temperature change as a slug through a pipe.
Operational Transients Auxiliary DNB	EI Model	VII-21 to VII-26	Computes DNB based on hydraulic conditions from flow equation solution and core power response.

(2) Specifying input (driving the model) that produces known results

(3) Editing all required information entering a model and hand calculating the output

(4) Comparing results with output from other codes.

A list of the RETRAN models versus these four verification methods was used to assure that each individual RETRAN model was verified against at least two of the four methods. Table II.4-2 shows the details of this effort. The first two columns are the same as Table II.4-1 and give the physical process and model utilized. The last four columns identify the specific methods (described above) used to verify the individual model.

Finally a series of "basic" test problems were run for which either analytical solutions were available or self consistency checks could be made. Again the list of RETRAN models was cross-correlated against this list of problems to assure that all common mode models were correctly functioning. The results of this activity, which was useful in establishing that the basic models of the code were functioning correctly, are summarized in Tables II.4-3 and II.4-4. The analytical solution comparisons are listed in Table II.4-3. Some of the comparisons with experimental data, which are considered to be consistent applications, are given in Table II.4-4. Note that type and model herein refer to a combination of individual models, components, and numerical techniques, and the comparisons are with overall results of these groups.

A complete level of verification of the code logic is very difficult to establish. A major practical problem associated with RETRAN is associated with the fact that RETRAN is basically a one-dimensional, homogeneous equilibrium code. The set of problems thus classified as the consistent application type rigorously includes only simple geometries, like straight pipes, with well-mixed fluid conditions. If the definition of one-dimensional is relaxed to include a one-dimensional streamline, then simple closed loops are allowed. Even with this definition, almost all of the interesting reactor applications of the code fall into the extended class and hence should be carefully examined, with the results and limitations documented.

TABLE II.4-2

RETRAN MODEL VERIFICATION

Type	Model	Data Reference In Volume I	Visual[(1)] Check	Driver[(2)] Check	RETRAN[(3)] Check	Other[(4)] Code
Momentum Equation Volume Mass Flow	Average of Inlet and Outlet Flows	None	x		x	x
Momentum Equation Momentum Flux Term	Methods of Computing Momentum Flux	None	x			x
Momentum Equation Junction Area Change	Local Incompressible or Compressible Flow	None	x		x	x
Momentum Equation Wall Friction Force	Fanning Friction Correlation	III.1-1	x	x	x	x
Momentum Equation Wall Friction Force	Homogeneous Correlation	III.1-4	x	x	x	x
	Baroczy Correlation	III.1-6, III.1-7	x	x	x	x
	Beattie Correlations	III.1-14, III.1-15	x	x	x	
Momentum Equation	Loss Coefficients	None	x		x	x
Energy Equation Wall Heat Transfer	Dittus-Boelter Correlation	III.2-36	x		x	x
	Thom Correlation	III.2-37	x		x	x

TABLE II.4-2 (Cont'd)

Type	Model	Data Reference In Volume I	Visual[1] Check	Driver[2] Check	RETRAN[3] Check	Other[4] Code
Energy Equation Wall Heat Transfer	Schrock-Grossman Correlation	III.2-58	x		x	x
	McDonough, Milich and King Correlation	III.2-62	x		x	x
	Groeneveld 5.7 and 5.9 Correlations	III.2-35	x		x	x
	Dougall-Rohsenow Correlation	III.2-68	x		x	x
	Berenson Correlation	III.2-63	x		x	x
Energy Equation Condensing Heat Transfer	Uchida Correlation	III.2-75	x		x	x
	Tagami Correlation	III.2-76, III.2-77	x		x	x
	CSB 6.1	III.2-78, III.2-79	x		x	x
Energy Equation Critical Heat Flux	B&W-2 Correlation	III.2-23, III.2-24	x		x	x

TABLE II.4-2 (Cont'd)

Type	Model	Data Reference In Volume I	Visual Check (1)	Driver Check (2)	RETRAN Check (3)	Other Code (4)
	Barnett Correlation	III.2-25, III.2-69	x		x	x
	Modified Barnett Correlation	III.2-18	x		x	x
	Interpolation	None	x			x
Energy Equation Junction Enthalpy	Enthalpy Transport	None	x		x	x
Energy Equation Kinetic Energy	Volume Averaged Flow	None	x			x
State Equation Thermodynamic Properties	Tabulated Values	VIII.1-7 VIII.1-9	x		x	x
State Equation Transport Properties	Viscosity	VIII.1-7	x		x	x
	Conductivity	VIII.1-7	x		x	x
State Equation Void Fraction	Homogeneous Equilibrium	III.1-22	x		x	x
State Equation Junction Pressure	Momentum Equation	EI addition	x		x	

TABLE II.4-2 (Cont'd)

Type	Model	Data Reference In Volume I	Visual Check (1)	Driver Check (2)	RETRAN Check (3)	Other Code (4)
State Equation Junction Enthalpy	Energy Transport or Donor Cell	EI addition	x		x	
Critical Flow	Sonic Choking	III.3-1,III.3-2	x		x	
	Henry-Fauske	III.3-4	x		x	x
	Moody	III.3-6	x		x	x
Solution Techniques	FLASH-4	VIII.1-2,VII.1-11	x		x	x
	Causal Volume	EI addition	x		x	
	Steady-State	EI addition	x		x	
Heat Conduction	Two-sided Heat Conduction	EI addition	x		x	x
	Causal Conductor	EI addition	x		x	
Energy Generation	Tabulated Values	None	x		x	
	Point Kinetics	V.1-1	x		x	x
	Metal-Water	V.2-1	x		x	x
	Direct Moderator Heating	VI.1-2	x		x	

TABLE II.4-2 (Cont'd)

Type	Model	Data Reference In Volume I	Visual(1) Check	Driver(2) Check	RETRAN(3) Check	Other(4) Code
Momentum Equation Component	Centrifugal Pumps	VI.1-1,VI.1-2, VI.1-3	x		x	x
Momentum Equation Component	Valves	VI.1-2	x		x	x
Energy Equation Component	Non-Conducting Heat Exchangers	None	x		x	x
Reactor Trip System	Trips	VI.1-2, EI addition	x	x	x	x
Control System Control Blocks	Digital Control Blocks	EI addition	x	x	x	x
Pressurizer	Non-equilibrium Pressurizer	EI addition	x		x	x
Special Model	Bubble Rise	III.1-21	x		x	x
Flow Regime Map	Modified Bennett	III.1-22,III.1-16 III.1-24	x	x	x	
Operational Transients Transport Delay	EI Model	EI addition	x		x	
Operational Transients Auxiliary DNB	EI Model	EI addition	x		x	

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TABLE II.4-2 (Cont'd)

Type	Model	Data Reference In Volume I	Visual[1] Check	Driver[2] Check	RETRAN[3] Check	Other[4] Code

(1) Coding Check - Visual coding check by two people

(2) Driver Check - Model checked by a driver routine outside of RETRAN and results compared to hand calculation

(3) RETRAN Check - Model checked by hand calculation and compared to RETRAN computation

(4) Other Code - Results of RETRAN analysis of problem compared to results from another code

TABLE II.4-3

RETRAN CONSISTENT APPLICATIONS - ANALYTICAL

Type	Model	Analysis 1	2	3	4	5	6	7	8	9	10
Momentum Equation	Volume Mass Flow	X	X	X	X	X	X	X	X	X	X
	Momentum Flux	X	X	X	X	X	X	X	X		X
	Junction Area Change										
	Homogeneous Multiplier			X							
	Baroczy Multiplier			X		X				X	
	Beattie Multiplier			X							
	Loss Coefficients										
Energy Equation	Forced Convection Heat Transfer							X	X		
	Critical Heat Flux										
	Enthalpy Transport	X	X	X	X	X			X		
	Kinetic Energy	X	X	X	X	X		X	X	X	
State Equation	Thermodynamic Properties	X	X	X	X	X	X	X	X	X	X
	Viscosity			X	X	X		X	X		
	Conductivity							X	X		
	Void Fraction	X	X	X	X	X	X	X	X		X
Critical Flow	Sonic Choking										
	Henry-Fauske										
	Moody										

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TABLE II.4-3 (Cont'd)

Type	Model	Analysis 1	2	3	4	5	6	7	8	9	10
Solution Techniques	FLASH-4	X	X	X	X	X	X	X	X	X	X
	Causal Volume										
	Steady State	X									
Heat Conduction	Two-Sided Conduction							X	X		
	Causal Conductor										
Energy Generation	Tabulated Values										
	Point Kinetics										X
	Metal-Water										
	Direct Moderator Heating										
Components	Pumps										
	Valves										
	Heat Exchangers										
	Trips	X	X	X	X	X	X	X	X	X	X
	Control Blocks									X	
	Pressurizer										
Special Models	Bubble Rise										
	Bennett Map			X							
	Transport Delay										
	Auxiliary DNB										

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TABLE II.4-3 (Cont'd)

X - Indicates the model has been used for the analysis indicated.

1. Null problem
2. Symmetric perturbation
3. Single-phase friction
4. Zero friction
5. Junction enthalpy calculation
6. Momentum flux calculation
7. Conduction equation solution
8. Constant Energy addition
9. Solar controller
10. Neutron decay

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TABLE II.4-4

RETRAN CONSISTENT APPLICATIONS - EXPERIMENT

Type	Model	Analysis 1	2	3	4	5	6	7	8	9	10
Momentum Equation	Volume Mass Flow	X	X	X	X	X	X	X	X	X	X
	Momentum Flux	X	X	X	X		X	X	X	X	X
	Junction Area Change				X	X	X			X	X
	Homogeneous Multiplier		X		X				X		
	Baroczy Multiplier	X	X	X	X	X	X	X	X	X	X
	Beattie Multiplier				X						
	Loss Coefficients					X	X	X	X	X	X
Energy Equation	Forced Convection Heat Transfer		X	X						X	X
	Critical Heat Flux		X	X						X	X
	Enthalpy Transport	X	X	X	X	X	X	X	X	X	X
	Kinetic Energy	X	X	X	X	X	X	X	X	X	X
State Equation	Thermodynamic Properties	X	X	X	X	X	X	X	X	X	X
	Viscosity	X	X	X	X	X	X	X	X	X	X
	Conductivity		X	X						X	X
	Void Fraction	X	X	X	X	X	X	X	X	X	X

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TABLE II.4-4 (Cont'd)

Type	Model	Analysis 1	2	3	4	5	6	7	8	9	10
Critical Flow	Sonic Choking				X						
	Henry-Fauske				X				X	X	X
	Moody				X				X	X	X
Solution Techniques	FLASH-4	X	X	X	X	X	X	X	X	X	X
	Causal Volume								X		
	Steady State	X	X	X	X	X			X		X
Heat Conduction	Two-Sided Conduction		X	X							X
	Causal Conductor		X								
Energy Generation	Tabulated Values										X
	Point Kinetics										
	Metal-Water										X
	Direct Moderator Heating										
Components	Pumps										X
	Valves				X				X		X
	Heat Exchangers										
	Trips	X	X	X	X	X	X	X	X	X	X

TABLE II.4-4 (Cont'd)

Type	Model	Analysis 1	2	3	4	5	6	7	8	9	10
	Control Blocks									X	
	Pressurizer									X	
Special Models	Bubble Rise										X
	Bennett Map	X	X	X	X	X	X	X	X	X	X
	Transport Delay										
	Auxiliary DNB										

X - Has been run with the indicated model

1. Ferrell-McGee (pressure drop)
2. Bennett et al. (heat transfer, pressure drop)
3. Schrock-Grossman (heat transfer)
4. Fauske (critical flow)
5. Expansion/contraction
6. Flow in manifolds
7. Flow in tees
8. Edwards pipe (SP#1)
9. Shipping port pressurizer
10. Semiscale Test S-02-8 (SP#5)

5.0 TRANSIENTS OF INTEREST

As indicated in the preceding section, most applications of RETRAN to reactor analysis fall in the extended application category. This is because one or more of the basic assumptions used to develop the RETRAN models are in some way violated. The main assumptions which result in theoretical limitations can be classified as follows:

(1) One-dimensional assumptions

(2) Homogeneous flow assumptions

(3) Thermal equilibrium assumptions

(4) Steady-state correlations for most constitutive models, (e.g., heat transfer, critical flow, and friction factors).

Probably the most apparent limitation is that of one-dimensional streamline flow. Light water reactors (LWRs) have numerous regions where there are definitely multidimensional effects; for example the downcomer, steam generator, upper and lower plena, and the reactor core. However, if one is not interested in detailed distribution information in these regions, the multi-dimensionality may have only a minor influence on the bulk parameters of interest. Circumstances where this is most likely to occur are those cases in which only minor changes in system conditions occur and for those cases in which the change in a value, not the absolute value, is of interest.

It is thus necessary to categorize the various types of reactor analyses and to determine the code sensitivity and limitations in each case. Tables II.5-1 and II.5-2 list most SAR Chapter 15 transients of interest for BWRs and PWRs, respectively. Some of the normal operation and moderate frequency incidents produce mild transients in which the system variables are only slightly changed. For these cases, the initial conditions and the reactor control system can make a significant contribution to the plant response. Thus the steady-state and operational transient models in RETRAN are of great importance for these events. For other incidents and limiting fault events, the transients may produce large changes in the system conditions. In the case of a LOCA, these changes are

TABLE II.5-1

SOME BWR SAR-CHAPTER 15 TRANSIENTS

Transient	Condition	System Analysis Required
PRESSURE INCREASE TRANSIENTS		
° Generator load rejection	II	
° Turbine trip with/without bypass	II	Yes
° Steam line isolation valve(s) closure	II	Yes
° Pressure regulator failure (close)	II	Yes
° Loss of condenser vacuum	II	Yes
° Turbine control valve fast closure	III	Yes
° Above incidents with delayed scram	III	Yes
REACTOR VESSEL WATER TEMPERATURE DECREASE		
° Loss of feedwater heater	II	Yes
° Inadvertent injection pump start	II	Yes
REACTOR VESSEL COOLANT INVENTORY DECREASE		
° Loss of feedwater flow	II	Yes
° Pressure regulator failure (open)	II	Yes
° Relief or safety valve failure (open)	II III	Yes
° Loss of auxiliary power	II	Yes
CORE COOLANT FLOW DECREASE		
° Recirculation pump seizure	III	Yes
° Recirculation pump(s) trip	II	Yes
° Recirculation flow control failure	II	Yes
CORE COOLANT FLOW INCREASE		
° Recirculation flow control failure	II	Yes
° Startup of idle recirculation pump	III	Yes
POSITIVE REACTIVITY INSERTION		
° Continuous rod withdrawal		
- From subcritical	II	Yes
- From power	II	Yes
° Rod ejection	IV	Yes

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TABLE II.5-1 (cont'd)

SOME BWR SAR-CHAPTER 15 TRANSIENTS

Transient	Condition	System Analysis Required
ANTICIPATED TRANSIENT WITHOUT SCRAM	III	Yes
STEAM LINE BREAK	IV	Yes
RECIRCULATION LINE RUPTURE		
° Large break	IV	Yes
° Small break	II III	Yes
IMPROPER CORE ASSEMBLY	III	No
ROD REMOVAL ERROR AT REFUELING	IV	No
FUEL HANDLING ACCIDENT	IV	No

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TABLE II.5-2

SOME PWR SAR-CHAPTER 15 TRANSIENTS

Transient	Condition	System Analysis Required
UNPLANNED DECREASE IN SECONDARY HEAT REMOVAL		
o Loss of external load	II	Yes
o Turbine trip	II	Yes
o Loss of condenser vacuum	II	Yes
o Steam pressure regulator failure	II	Yes
o Loss of normal feedwater flow	II	Yes
o Loss if A-C power to auxiliaries	II	Yes
UNPLANNED INCREASE IN SECONDARY HEAT REMOVAL		
o Excessive load increase	II	Yes
o Idle loop startup	II	Yes
o Decrease in feedwater temperature	II	Yes
o Increase in feedwater flow rate	II	Yes
o Increase in steam flow rate	II	Yes
o Inadvertent opening of steam generator relief or safety valve	II	Yes
CHANGES IN REACTOR COOLANT SYSTEM INVENTORY (PRIMARY SIDE INITIATED)		
o Inadvertent operation of ECCS	II	Yes
o Accidental depressurization	II	Yes
LOSS OF REACTOR COOLANT FLOW		
o Partial loss of flow	II	Yes
o Complete loss of flow	III	Yes
o Locked rotor	IV	Yes
REACTIVITY INSERTION (PRIMARY SIDE INITIATED)		
o Uncontrolled rod withdrawal		Yes
- from subcritical	II	Yes
- from power	II	Yes
o Control rod misoperation	II	Yes
o Chemical system malfunction	II	Yes
ANTICIPATED TRANSIENTS WITHOUT SCRAM		Yes
STEAM LINE BREAK	IV	Yes

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TABLE II.5-2 (cont'd)

SOME PWR SAR-CHAPTER 15 TRANSIENTS

Transient	Condition	System Analysis Required
RECIRCULATION LINE BREAK		
° Steam generator tube rupture	II III	Yes
° Small break	III	Yes
° Loss of coolant accident	IV	Yes
FUEL ASSEMBLY INSERTION ERROR	III	No
CONTROL ROD EJECTION	IV	No
FUEL HANDLING ACCIDENT	IV	No
WASTE GAS DECAY TANK RUPTURE	IV	No

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usually the result of rapid depressurization, and/or ECCS injection, and may require use of blowdown and refill/reflood models which are not in the present version. Currently, RETRAN has been applied only to the blowdown portion of primary pipe break transients.

For those normal and moderate frequency events which only slightly perturb reactor conditions, it is expected that, even though RETRAN is being applied in the extended application range, reasonable results should be obtained. However, because this is an intuitive argument, the accuracy of the results should be determined by direct comparison with experimental data for those cases where such results exist. In this manner, one can establish confidence in the calculation based on other transients which are governed by similar phenomena.

Note that blowdown calculations produce large changes in the system variables, hence generalization of confidence in these cases is more demanding. For such cases, the direct comparison with some experimental results is required.

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6.0 RETRAN QUALIFICATION

The qualification of a code is the comparison of the code results with experimental data so as to determine its applicability and sensitivity. The RETRAN Code was qualified against three different classes of data.

The first and simplest types of comparisons are separate effect experiments which, in general, are small scale tests in which complexities are held to a minimum and the governing parameters accurately measured. The results of these analyses give confidence in the individual, and combinations of, models utilized. Chapter III describes the qualification of RETRAN against the Separate Effect Tests.

The second category of experiments are System Effects Tests in which the interaction of various components must be described. In general these are intermediate size tests (i.e., Semiscale, Two-Loop Test Apparatus, LOFT) in which the assumption of one-dimensional streamline flow is reasonably approximated. The RETRAN qualification against experiments of this type is much more demanding and gives confidence in, and implies limitations on, the basic theory used in RETRAN. The results of this class of experiments are given in Chapter IV.

The third and most important class of comparison is against actual plant data. The data from large nuclear plants is generally obtained from operational instrumentation and is usually limited in both quantity and quality. The actual comparisons are also made more difficult by the unavailability of all required input data, (e.g. information regarding the initial conditions, time manual operator action was taken, response characteristics of valves). This phase of the RETRAN qualification was performed mainly by the EPRI/Utility System Analysis Working Group.

A description of the Working Group is given in Section II.1. This group was primarily interested in the analysis of operational transients generally addressed in Chapter 15 of most Safety Analysis Reports. Each of the participating utilities identified both analyses of interest and some existing reactor start-up and/or operating tests against which to qualify the RETRAN results. The results of this effort are presented in Tables II.6-1 and II.6-2 which list all Chapter 15

TABLE II.6-1

RETRAN QUALIFICATION FOR SOME SAR-CHAPTER 15 BWR TRANSIENTS

	Direct Comparison	Similar Phenomena	SAR-Other Calculations	See Footnote	Comments
PRESSURE INCREASE TRANSIENTS					
° Generator load rejection	X			1	
° Turbine trip with/without bypass	X		X	2	Peach Bottom Tests
° Steam Line isolation valve(s) closure					
° Pressure regulator failure (close)					
° Loss of condenser vacuum					
° Turbine control valve fast closure					
° Above incidents with delayed scram					
REACTOR VESSEL WATER TEMPERATURE DECREASE					
° Loss of feedwater heater					
° Inadvertent injection pump start					
REACTOR VESSEL COOLANT INVENTORY DECREASE					
° Loss of feedwater flow	X			1	
° Pressure regulator failure (open)					
° Relief or safety valve failure (open)					
° Loss of auxiliary power					
CORE COOLANT FLOW DECREASE					
° Recirculation pump seizure					
° Recirculation pump(s) trip	X			1	
° Recirculation flow control failure					
CORE COOLANT FLOW INCREASE					
° Recirculation flow control failure					
° Startup of idle recirculation pump					
POSITIVE REACTIVITY INSERTION					
° Continuous rod withdrawal					
° - from subcritical					
- from power					
° Rod ejection					
ANTICIPATED TRANSIENT WITHOUT SCRAM					
STEAM LINE BREAK					
RECIRCULATION LINE RUPTURE					
° Large break					
° Small break					

1. Startup test
2. SAR comparison
3. Other code comparison

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TABLE II.6-2

RETRAN QUALIFICATION FOR SOME PWR SAR-CHAPTER 15 TRANSIENTS

	Direct Comparison	Similar Phenomena	SAR-Other Calculations	See Footnote	Comments
UNPLANNED DECREASE IN SECONDARY HEAT REMOVAL					
° Loss of external load			X	2	
° Turbine trip					
° Loss of condenser vacuum					
° Steam pressure regulator failure					
° Loss of normal feedwater flow					
° Loss of A-C power to auxiliaries					
UNPLANNED INCREASE IN SECONDARY HEAT REMOVAL					
° Excessive load increase					
° Idle loop startup					
° Decrease in feedwater temperature					
° Increase in feedwater flow rate					
° Increase in steam flow rate					
° Inadvertent opening of steam generator relief or safety valve	X	X			TMI-2 Cooldown
CHANGES IN REACTOR COOLANT SYSTEM INVENTORY (PRIMARY SIDE INITIATED)					
° Inadvertent operation of ECCS					
° Accidental depressurization					
LOSS OF REACTOR COOLANT FLOW					
° Partial loss of flow		X		1	Pump Coastdown Test
° Complete loss of flow		X	X	1,2	
° Locked rotor					
REACTIVITY INSERTION (PRIMARY SIDE INITIATED)					
° Uncontrolled rod withdrawal					
- from subcritical					
- from power			X	2	
° Control rod misoperation					
° Chemical system malfunction					
ANTICIPATED TRANSIENTS WITHOUT SCRAM					
STEAM LINE BREAK			X	2	
RECIRCULATION LINE BREAK					
° Steam generator tube rupture					
° Small break					
° Loss of coolant accident			X	2	

1. Startup test
2. SAR comparison
3. Other code comparison

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incidents for BWRs and PWRs respectively, and show which transients were analyzed and the degree of qualification of the RETRAN analysis which was possible. The analyses are classified as:

(1) Direct comparison with experiments

(2) Extrapolated from similar but different experiments

(3) Results appear reasonable and agree with SAR results or calculations performed with other codes.

The first category implies that the accuracy of the results were determined by direct comparison with experimental data for some (one or more) cases where such data exist. The second category includes cases where direct experimental confirmation does not exist, but where confidence can be established in the calculation based on other transients which are governed by similar phenomena. The third category is one where no experimental data was available and the results were reviewed only to assure that they appeared physically correct, and that they are consistent with results of other computer analysis, principally vendor SARs. Note however, that comparisons of RETRAN and SARs is of limited value unless the assumptions and models utilized by the vendor are known. In general RETRAN has to be used in a restricted manner to compare with these analyses.

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III. SEPARATE EFFECT ANALYSES

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III. SEPARATE EFFECT ANALYSES

All two-phase problems exhibit a combination of physical phenomena. Even the simplest of single-phase flow problems, flow in a straight pipe, requires a model(s) for representing the friction contribution to momentum change from one point to the next. When such a problem is analyzed with a code like RETRAN, additional uncertainties are introduced by the approximations to the differential equations, the numerical solution technique employed to solve the problem, and the actual modeling of the problem (e.g., boundary conditions, node sizes, time step sizes). Despite these uncertainties, it is possible to evaluate each of the above mentioned items with appropriate analyses.

When two-phase flow analyses are performed, further uncertainties arise due to the addition of more constitutive equations (e.g., heat transfer, friction losses for the mixture, separate phase velocities and/or energies). However, such analyses are required if confidence in specific models is to be achieved.

The first step in qualification of the RETRAN code was to perform analyses of simple problems, referred to as separate effect analyses. In these problems, complexities are held to a minimum in an effort to evaluate specific models. Assuming appropriate noding and time step studies are performed, analyzing flows in a heated pipe can yield information about the conduction solution, the momentum and energy constitutive equations, and the code logic used in the analyses.

In this section, the separate effect analyses performed with the RETRAN code are summarized. Where possible the RETRAN analyses are compared with experimental data and the results of similar analyses (generally from a RELAP4 code).

1.0 PRESSURE DROP

The simplest two-phase flow analyses performed were for steady-state flow in an unheated pipe. The data from these types of experiments, when compared to RETRAN solutions, are used to evaluate the friction terms in the momentum equation. Experimental data reported generally include flow rate, inlet and outlet pressure, and inlet and outlet enthalpy. These reported data should be sufficient to provide boundary conditions (expressed as a fill or time-dependent volume) so that the computed pressure drop across the pipe can be compared to the experimental data. The work was performed by Energy Incorporated.

1.1 Ferrell-McGee Pressure Drop Data

RETRAN comparisons were performed with experimental data obtained by Ferrell-McGee[III.1-1] for steady-state two-phase flow through constant and variable flow area test sections. The Ferrell-McGee data were obtained for vertical upflow of steam-water mixtures in test sections with constant flow areas and with abrupt expansions and contractions.

The test section containing the expansion or contraction had an area ratio of 0.608. The diameters ranged from 0.46 to 0.59 inches while the length of all the test sections was 72 inches. The experimental data consisted of axial pressure and void fraction measurements. The range of test variables encompassed pressures from 60 to 240 psia, mass velocities from 67 to 506 lbm/sec-ft^2, and qualities from subcooled conditions to 32 percent. Six test runs were selected to cover their range of data.

1.1.1 Description of the RETRAN Models

The RETRAN models used for these data comparisons were essentially the same as far as the number of volumes used and the method of applying the boundary conditions. All models had 9 control volumes and 9 flow junctions. The only difference between models were the flow areas and diameters for the expansion and contraction models.

The specification of the boundary conditions at the inlet was accomplished by a fill junction which set the flow rate and fluid energy. The exit pressure was established by a time dependent volume. The RETRAN steady-state option, the compressible flow form of the momentum equation, and the internal calculations of the applicable form loss coefficients were used for all the models.

1.1.2 Results and Data Comparisons

The comparisons of the RETRAN predictions with the experimental pressure data are given in Figures I.1-1 through I.1-3. The results of the comparisons are very good with the largest deviation being less than 4 percent for Test 7D-4.

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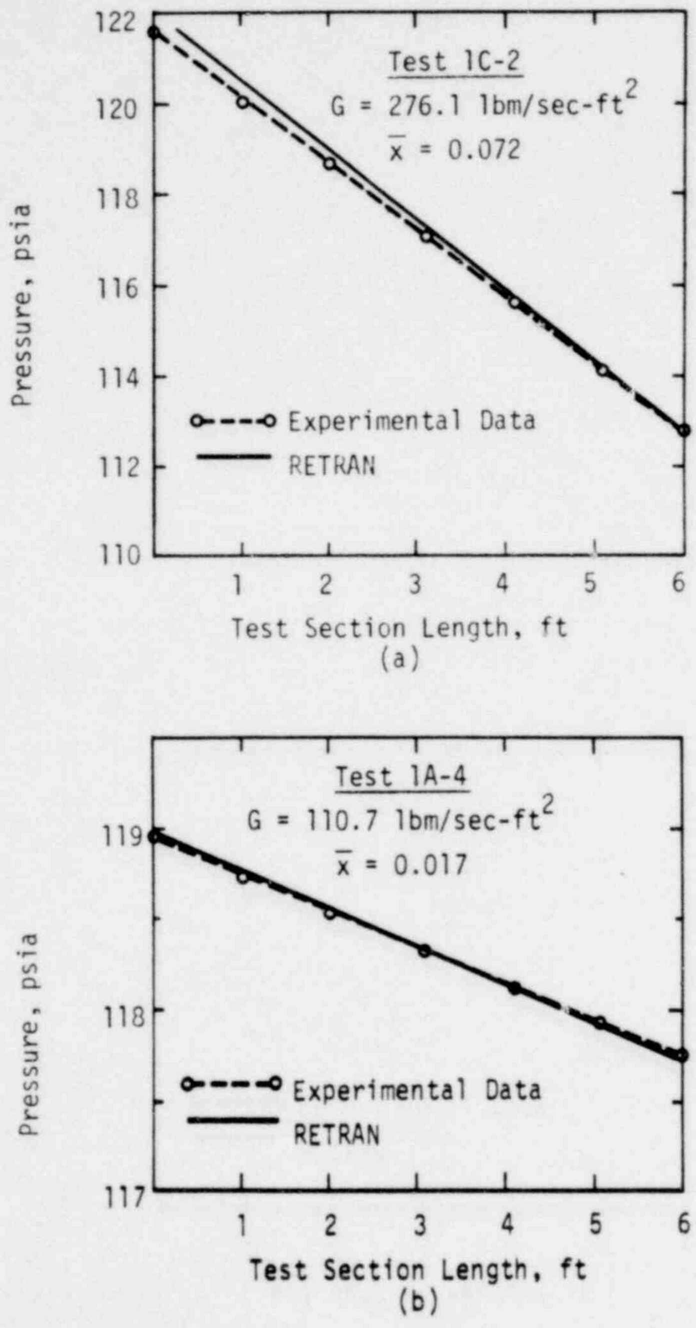

FIGURE III.1-1 RETRAN Predictions for Ferrell-McGee's Straight Test Section

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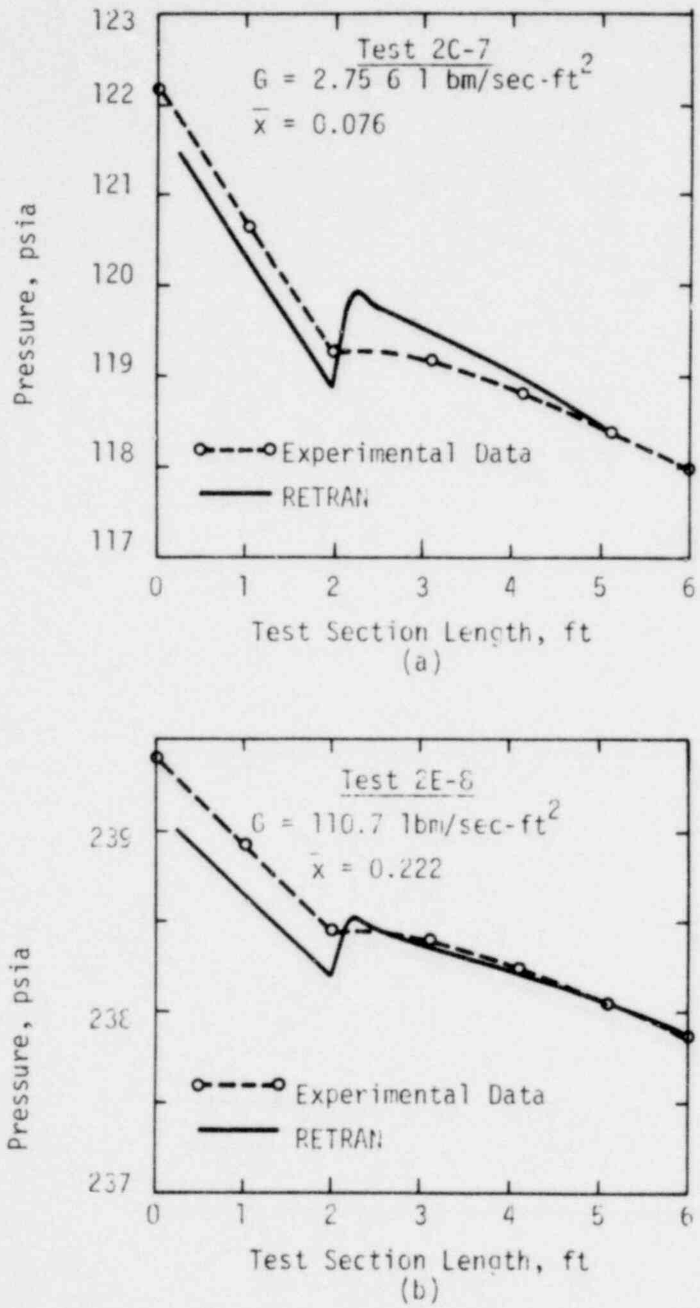

FIGURE III.1-2 RETRAN Predictions for Ferrell-McGee's Expansion Test Section

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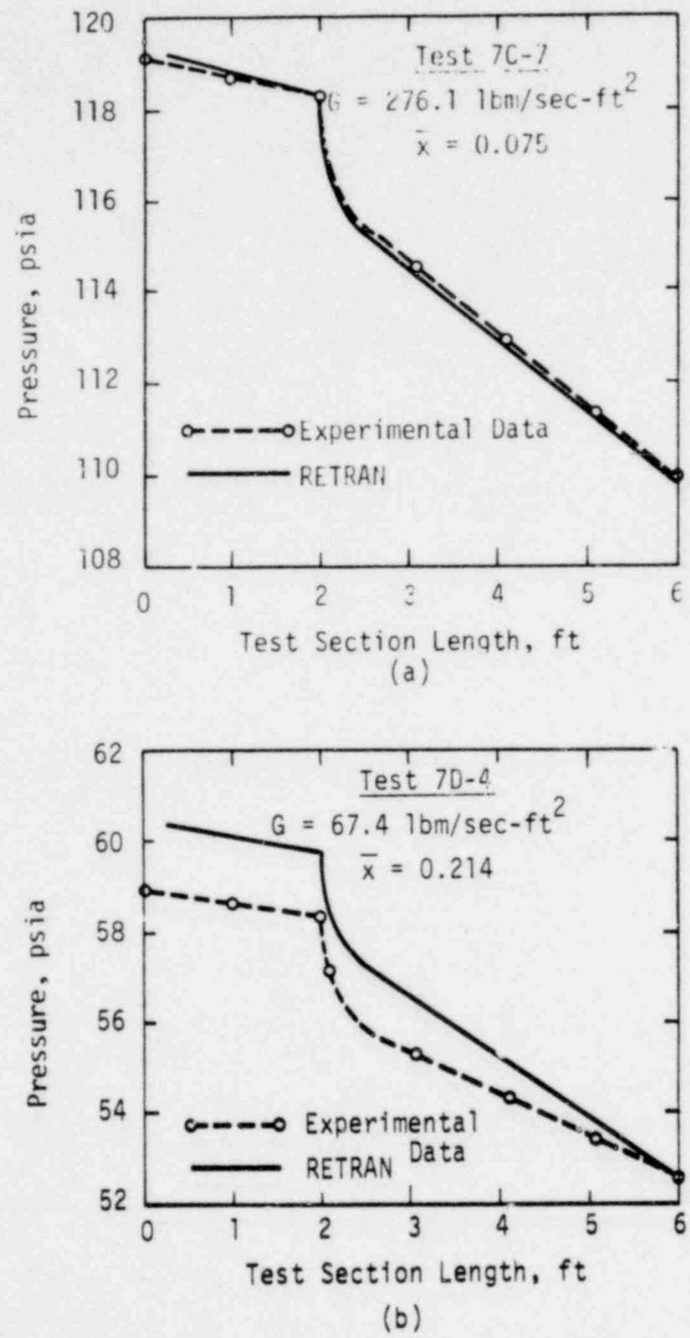

FIGURE III.1-3 RETRAN Predictions for Ferrell-McGee's Contraction Test Section

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The disagreement between the calculations and the experimental data may be due to the calculated form loss coefficients. RETRAN uses a constant value based on the up and downstream flow areas. In reality, the coefficients should vary with the flow rate. Also finer spatial nodalization of the test section in the immediate vicinity of the expansion or contraction could have produced better results.

[illegible]9 069

2.0 HEAT TRANSFER

Tests from three sets of heat transfer experiments have been modeled with RETRAN. The experiments involved steady-state flow in a heated test section, with heat transfer modes ranging from single-phase forced convection to stable film boiling. Experimental data reported include surface temperatures and test section pressure drop. These data are used, with the RETRAN analyses, to evaluate the RETRAN constitutive equations for energy exchange between the surface and the fluid. In addition, comparisons with RELAP4 analyses are presented for some cases. The work was performed by Hughes and Fujita[III.2-1] and additional information can be obtained from the referenced report.

2.1 Bennett, Hewitt, Kearsey and Keeys Round Tube Data

Steady-state, heat transfer experiments for vertical upflow in a round tube were conducted by A. W. Bennett et al.,[III.2-2]. Single-phase water was introduced at the inlet of a uniformly heated test section, and boiling conditions at the test section outlet varied from saturated nucleate boiling to stable film boiling.

The inside diameter of the test section was 0.497 inch and the total length was 19 feet. Heated test section lengths of 12 feet and 18.25 feet were obtained by the use of current clamps at three locations. Wall temperatures were recorded by thermocouples attached to the outer wall of the test section. The reported inner wall temperatures were determined by the experimenters from heat conduction calculations. In performing a given experiment, the system was brought to a pressure to provide 1000 psia at the outlet and the heater power was increased to provide the desired test condition. Conditions were then allowed to stabilize and the power input, inlet temperature, exit pressure, mass flow, and wall temperatures were recorded.

2.1.1 Description of Model

Six experimental runs were evaluated with both RETRAN and RELAP4. Since the heat transfer logic for these two codes is similar, the analyses provide an opportunity to check for coding errors in RETRAN which may have occurred during the development effort. The 12 foot test section was modeled with 12 volumes and 12 heat conductors, while the 18.25 foot test section was modeled with 20 volumes and 20 heat conductors. The boundary conditions applied were a fill (flow rate and fluid energy) at the test section inlet, and a time-dependent volume (pressure) at the test section outlet. A uniform heat flux was applied to the heat conductors.

The RETRAN analyses were performed using (1) the steady-state (self-initialization) option and (2) by running a pseudo-transient with initial conditions different from the steady-state experimental conditions. For the latter case, the transient is calculated until the computed results are constant, a steady-state condition. Since RELAP4 does not have the steady-state option, pseudo-transients were run in a manner similar to the second method described for RETRAN.

2.1.2 Results and Data Comparisons

The results of the RETRAN analyses using the steady-state option are shown in Figures III.2-1 to III.2-6. The Barnett correlation was used to evaluate CHF and two options for the stable film boiling regime were used, Groeneveld Equation 5.7 and Groeneveld Equation 5.9. The pre-CHF heat transfer correlations in RETRAN give good agreement with the experimental data. The dryout point however is predicted to occur earlier with the RETRAN analyses than is actually observed experimentally. This may be due to applying an empirical correlation (Barnett) based on rod bundle experiments to the round tube geometry. Hughes and Fujita [III.2-1] performed a limited number of analyses with the Bowring round tube correlation to investigate this possibility. In general, a better prediction of dryout was achieved with the Bowring correlation, but additional analyses are required to provide further qualification. The post-CHF heat transfer results show good agreement with the experimental data. In general, Groeneveld Equation 5.7 under-predicted the heat transfer coefficient.

Comparisons of the RETRAN and RELAP4 analyses are given in Tables III.2-1 to III.2-4. Agreement between the two codes is very good. In these tables, the RETRAN results were obtained using the steady-state option. A null transient was executed after achieving steady-state to assure agreement between the steady-state solution and the transient solution. The computational time indicates the savings which can be achieved with the RETRAN steady-state option.

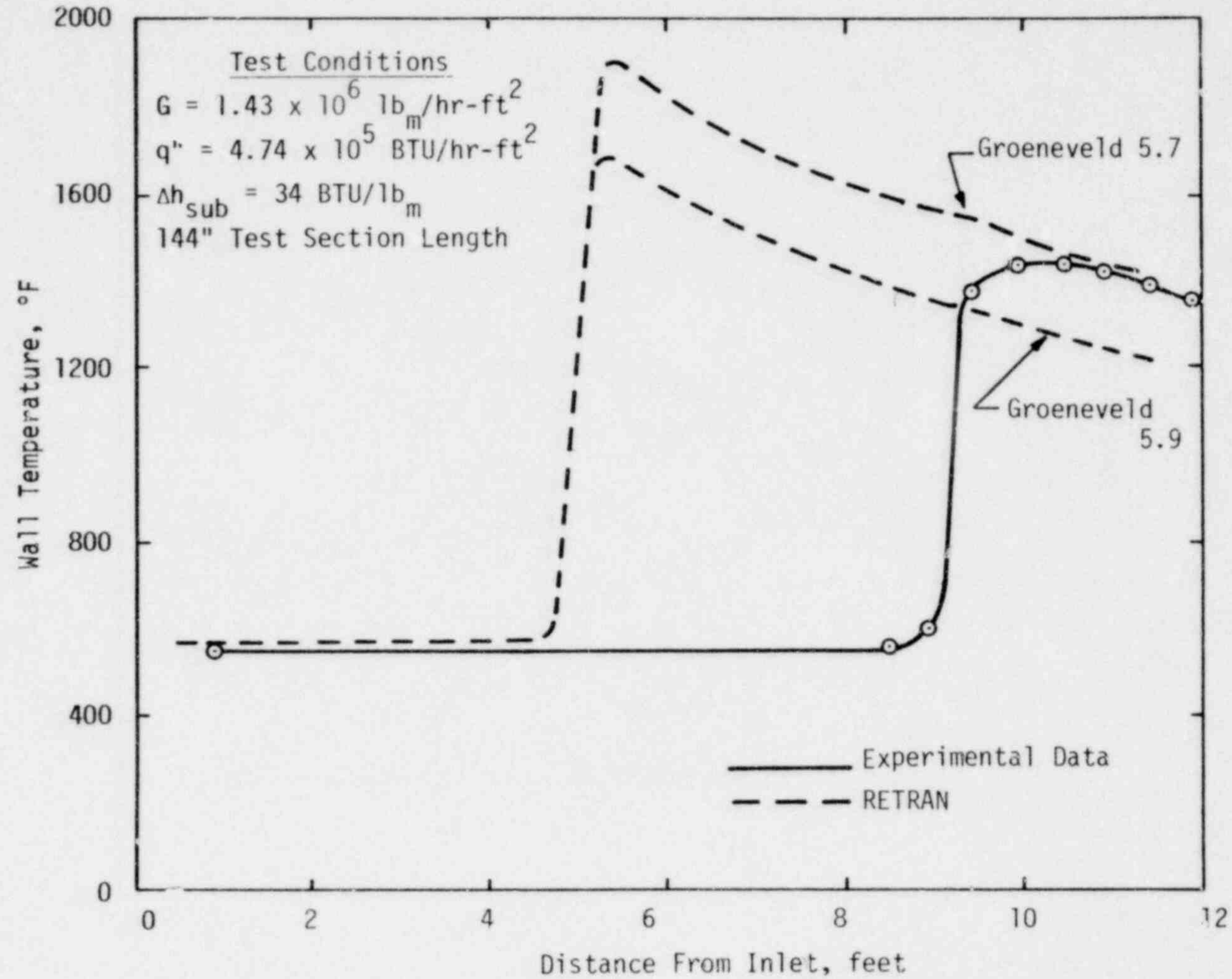

FIGURE III.2-1 Comparisons of Predicted and Experimental Wall Temperature - Run 5407 RETRAN Steady-State Initialization with "FILL" and "TDV".

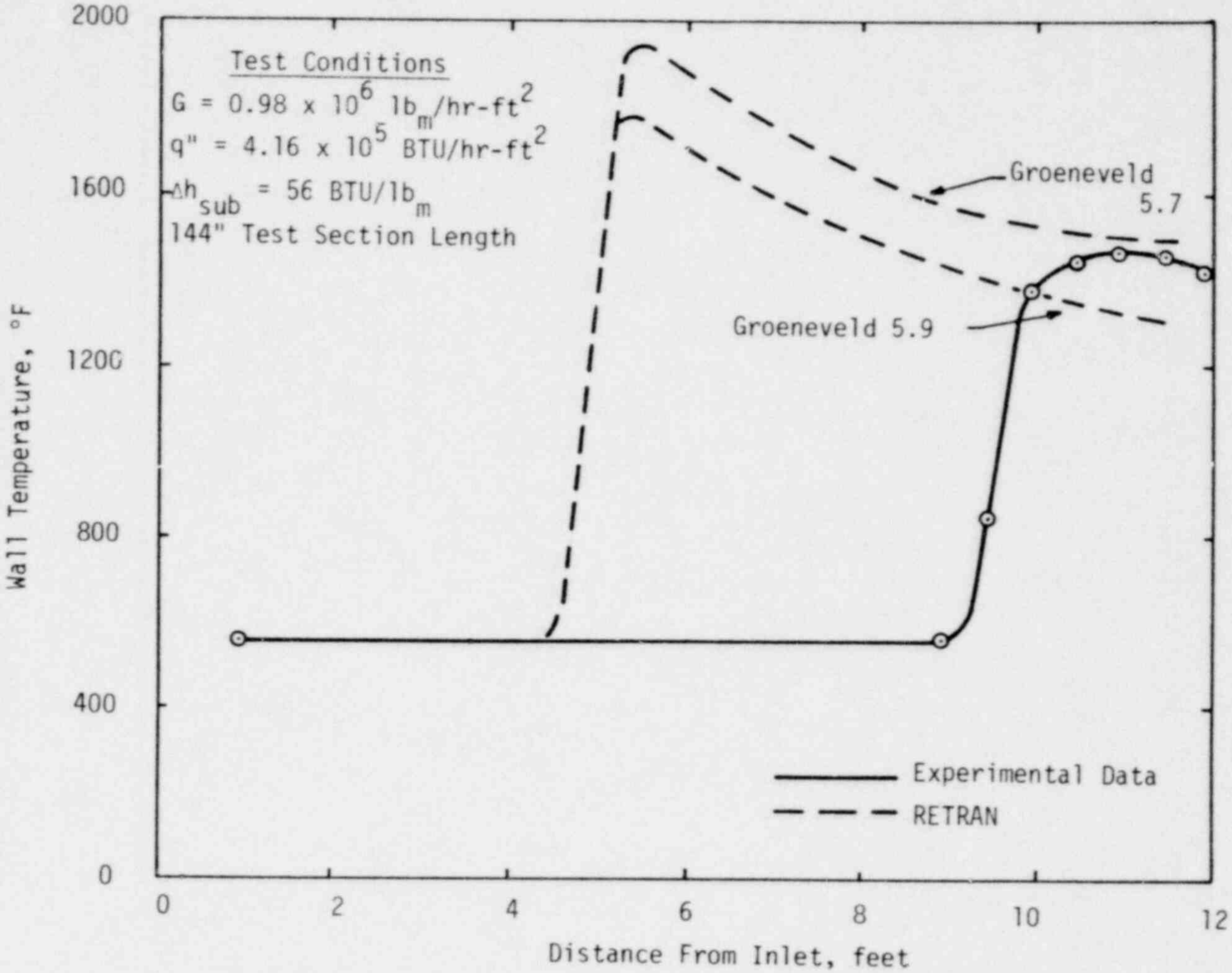

FIGURE III.2-2 Comparison of Predicted and Experimental Wall Temperature - Run 5456 RETRAN Steady-State Initialization with "FILL" and "TDV".

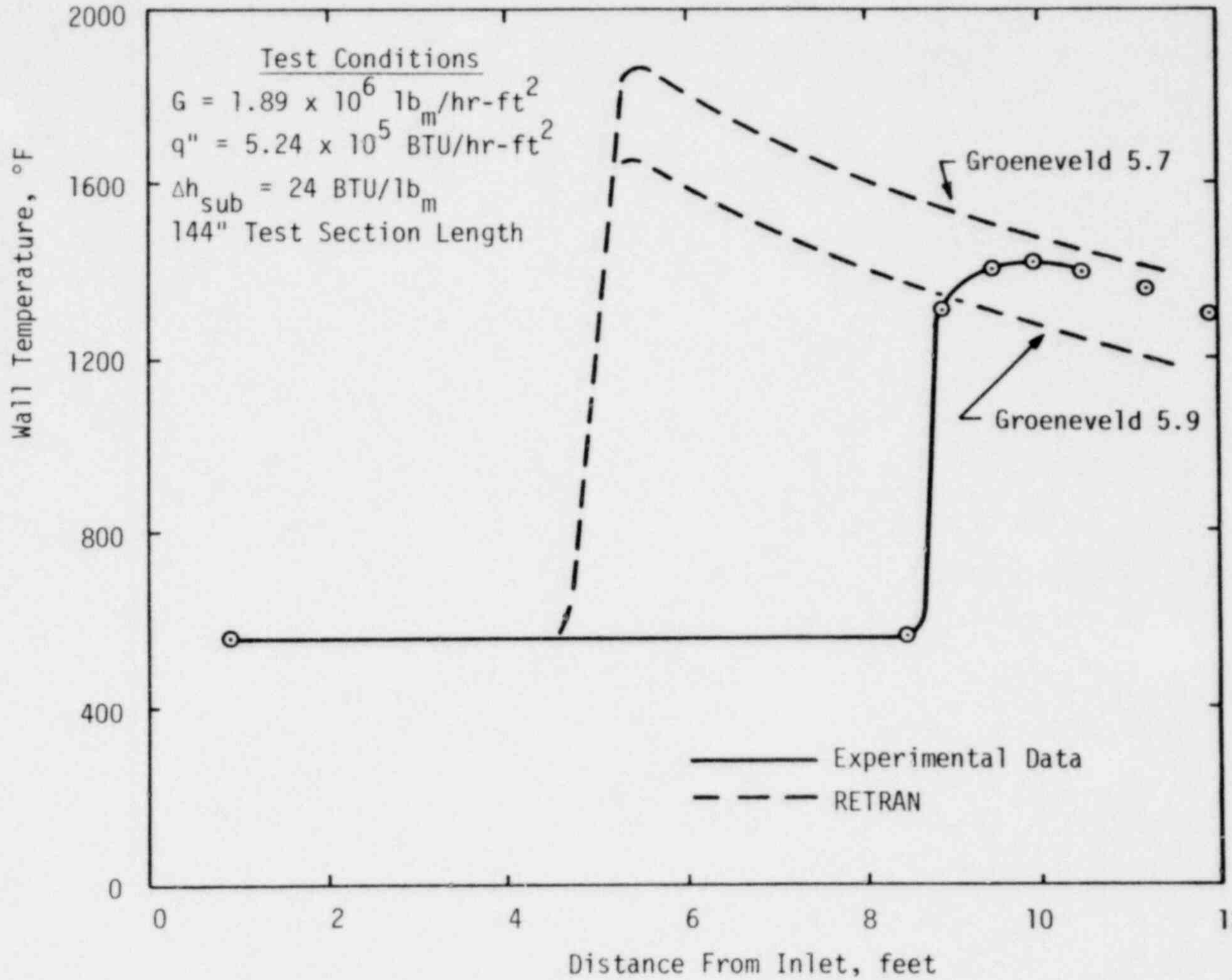

FIGURE III.2-3 Comparison of Predicted and Experimental Wall Temperature - Run 5424 RETRAN Steady-State Initialization with "FILL" and "TDV".

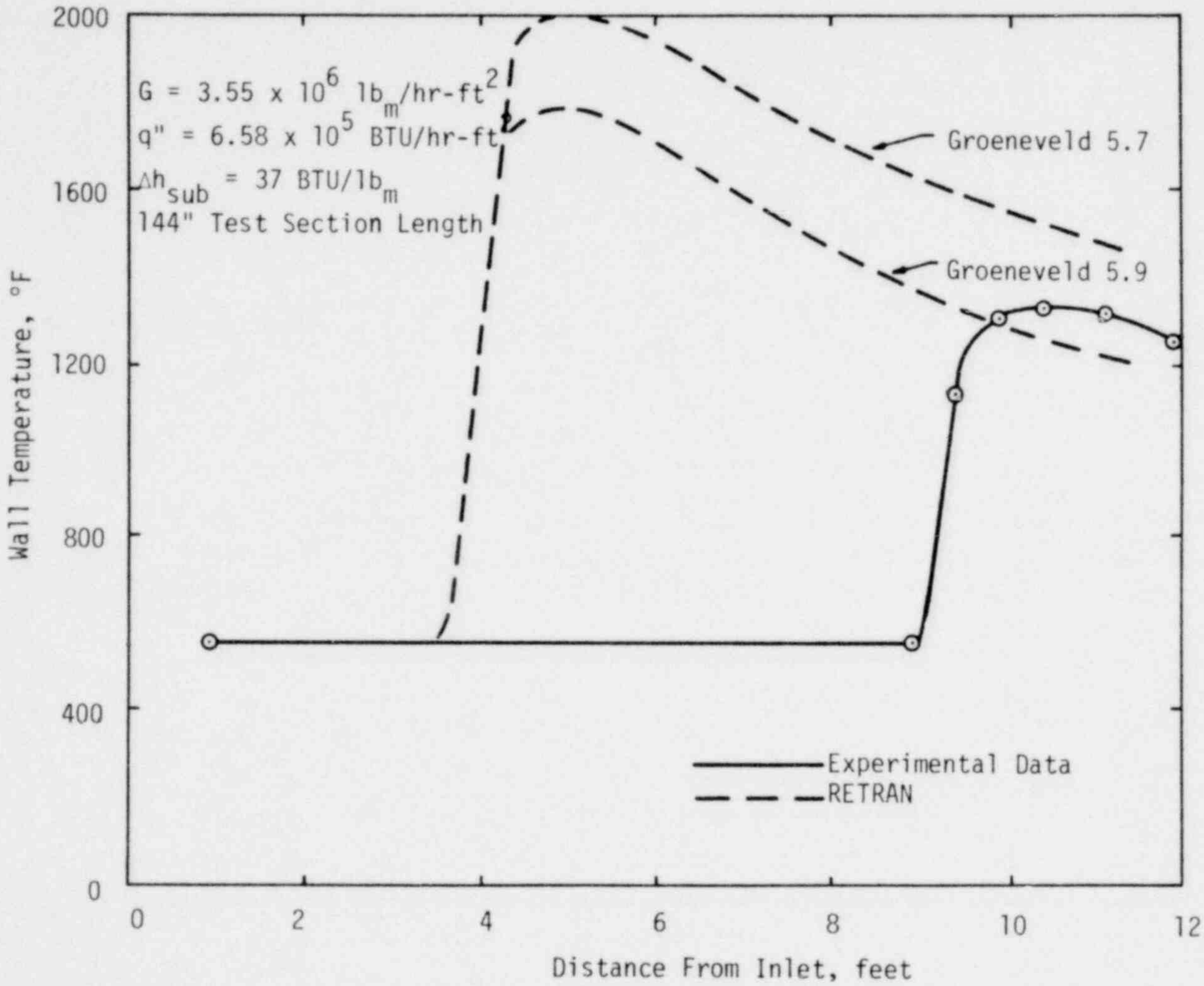

FIGURE III.2-4 Comparisons of Predicted and Experimental Wall Temperature - Run 5442 - RETRAN Steady-State Initialization with "FILL" and "TDV".

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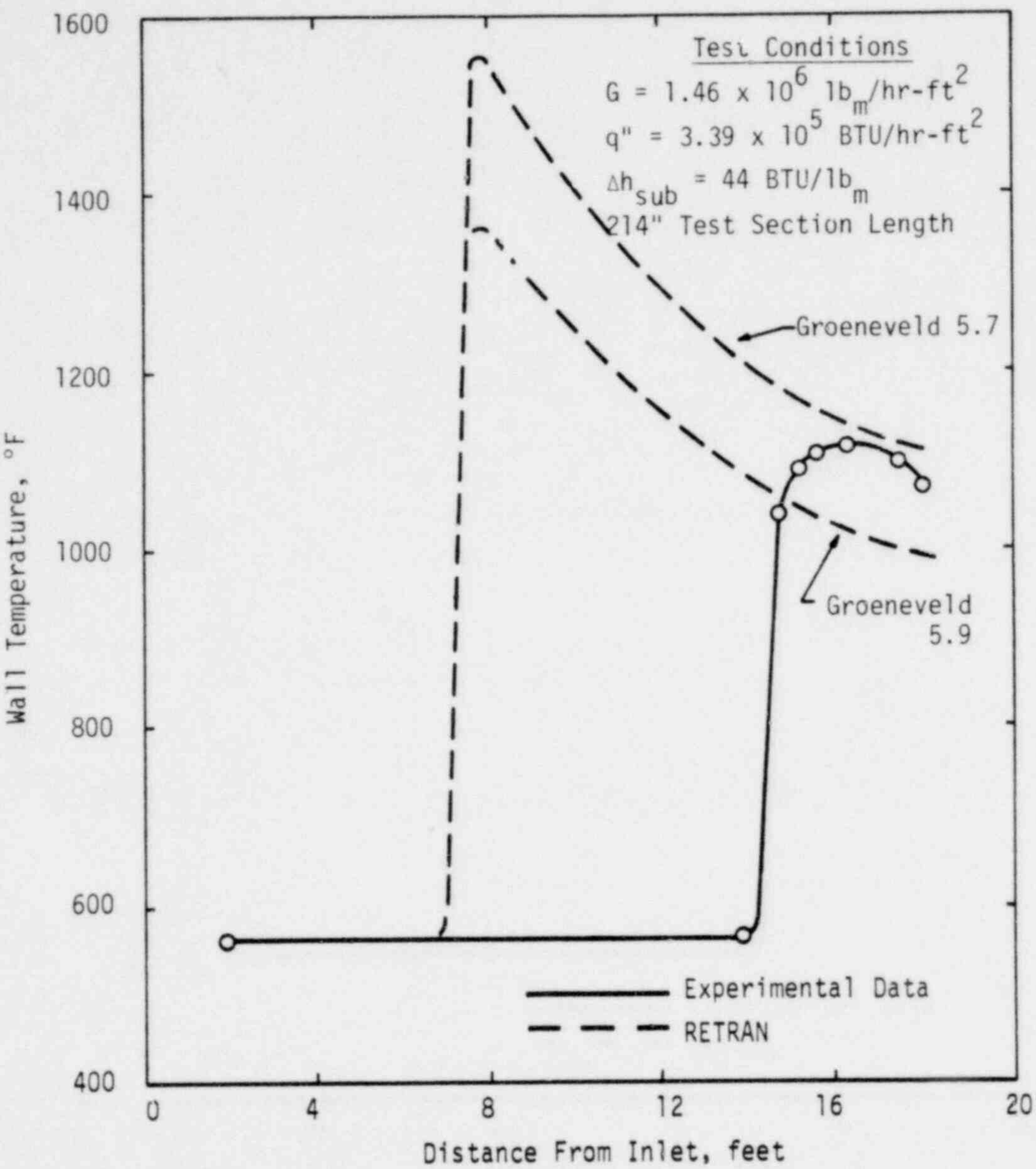

FIGURE III.2-5 Comparisons of Predicted and Experimental Wall Temperature - Run 5293 RETRAN Steady-State Initialization with "FILL" and "TDV".

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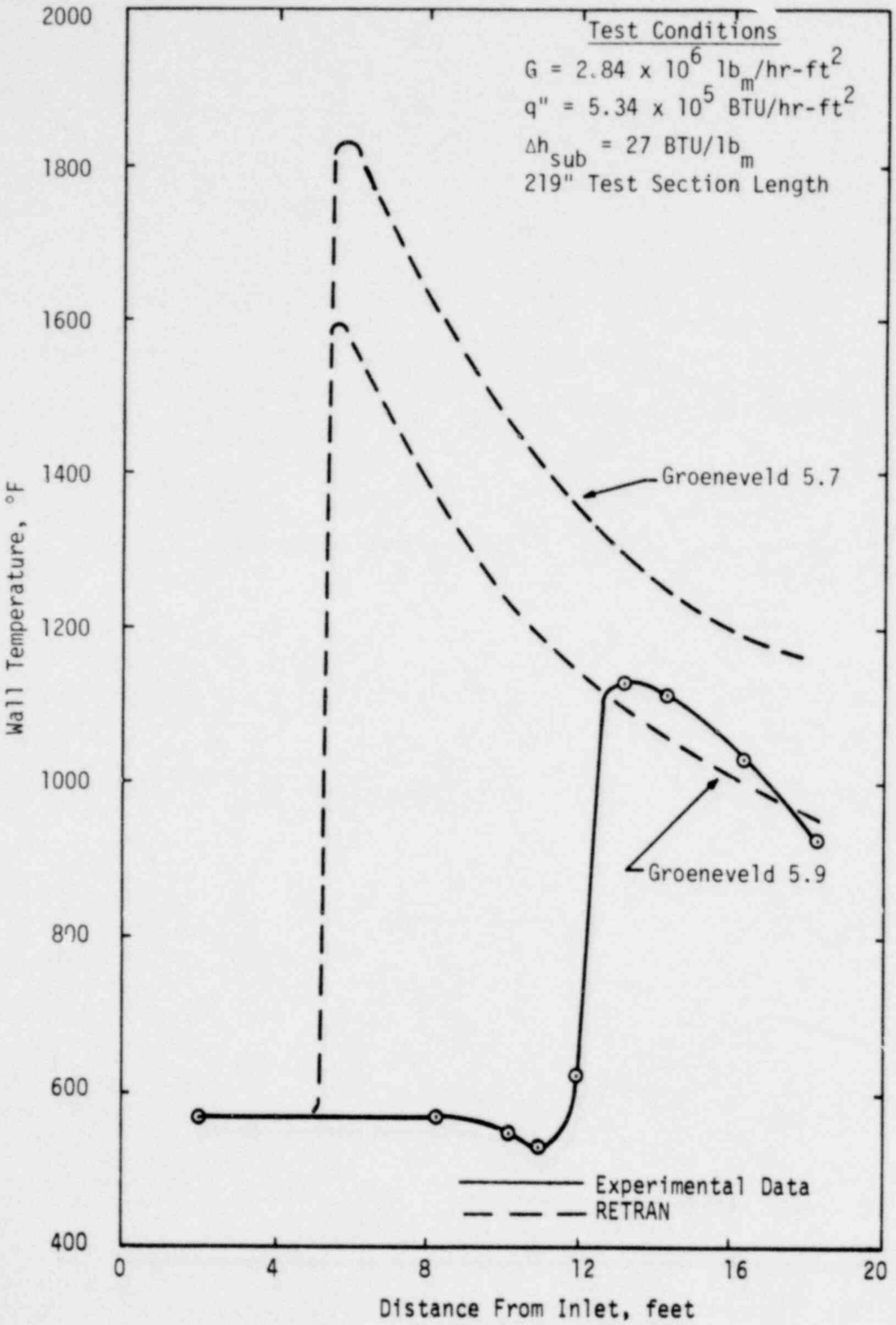

FIGURE III.2-6 Comparisons of Predicted and Experimental Wall Temperature Run 5380 - RETRAN Steady-State Initialization with "FILL" and "TDV".

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TABLE III.2-1

COMPARISONS OF PREDICTED PRESSURE (PSIA) FOR BENNETT'S 144-INCH VERTICAL, HEATED ROUND TUBE TEST SECTION

Run Number	5442		5424		5456		5407	
Volume Number	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4
1	1046.22	1046.36	1024.62	1024.61	1011.60	1011.71	1018.03	1018.04
2	1045.13	1045.27	1023.56	1023.55	1011.17	1011.25	1017.31	1017.32
3	1043.86	1043.99	1021.89	1021.88	1010.39	1010.47	1016.12	1016.13
4	1040.21	1040.32	1021.08	1020.09	1009.59	1009.68	1014.86	1014.86
5	1036.25	1036.35	1018.16	1018.15	1008.73	1008.81	1013.48	1013.49
6	1032.04	1032.13	1016.08	1016.07	1007.80	1007.87	1012.00	1012.00
7	1027.55	1027.63	1013.63	1013.86	1006.79	1006.85	1010.40	1010.40
8	1022.79	1022.85	1011.52	1011.51	1005.71	1005.76	1008.69	1008.70
9	1017.79	1017.84	1009.03	1009.03	1004.55	1004.59	1006.89	1006.89
10	1012.58	1012.61	1006.44	1006.44	1003.28	1003.30	1004.99	1004.99
11	1007.16	1007.18	1003.75	1003.75	1001.92	1001.94	1002.95	1002.96
12	1001.53	1001.54	1000.92	1000.92	1000.51	1000.51	1000.76	1000.77
CP time	8	155	8	131	8	143	8	129

TABLE III.2-2

COMPARISONS OF PREDICTED PRESSURE (PSIA) FOR BENNETT'S
219-INCH VERTICAL, HEATED ROUND TUBE TEST SECTION

Run Number	5293		5380	
Volume Number	RETRAN	RELAP4	RETRAN	RELAP4
1	1025.15	1025.25	1060.90	1061.04
2	1024.71	1024.81	1060.14	1060.27
3	1024.20	1024.29	1058.88	1059.02
4	1023.30	1023.39	1056.50	1056.73
5	1022.38	1022.47	1054.16	1054.28
6	1021.40	1021.49	1051.57	1051.69
7	1020.36	1020.44	1048.83	1048.94
8	1019.25	1019.33	1045.93	1046.03
9	1018.08	1018.15	1042.88	1042.98
10	1016.84	1016.91	1039.71	1039.80
11	1015.53	1015.59	1036.41	1036.49
12	1014.16	1014.21	1032.98	1033.06
13	1012.72	1012.77	1029.42	1029.49
14	1011.22	1011.27	1025.74	1025.80
15	1009.68	1009.72	1021.95	1022.00
16	1008.07	1008.11	1018.07	1018.12
17	1006.37	1006.40	1014.10	1014.14
18	1004.58	1004.60	1010.05	1010.08
19	1002.69	1002.70	1005.85	1005.87
20	1000.74	1000.74	1001.45	1001.46
CP Time	18	356	14	431

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TABLE III.2-3

COMPARISONS OF PREDICTED SURFACE TEMPERATURES (°F) FOR BENNET'S 144-INCH VERTICAL, HEATED ROUND TUBE TEST SECTION, GROENEVELD EQUATION 5.7

Run Number	5442		5424		5456		5407	
Heat Cond. Number	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4
1	575.55	575.62	570.67	570.68	566.78	566.86	568.86	568.88
2	575.45	575.52	570.56	570.57	566.73	566.80	568.79	568.79
3	575.33	575.39	570.39	570.40	566.65	566.72	568.67	568.67
4	574.97	575.03	570.21	570.22	566.57	566.63	568.54	568.54
5	574.58	574.64	570.02	570.02	579.07	583.41	569.11	569.34
6	574.17	574.23	575.60	576.39	592.58	594.77	582.25	583.63
7	573.73	573.79	581.76	582.54	594.96	594.70	587.55	588.17
8	1766.92	1768.75	1649.70	584.20	589.70	589.54	587.99	588.00
9	1666.86	1668.60	1571.52	1572.16	585.76	585.66	584.13	584.15
10	1586.43	1588.10	1507.87	1508.48	582.70	582.64	1517.61	1517.95
11	1520.25	1521.85	1446.28	1446.94	1495.35	1488.63	1456.89	1457.27
12	1457.01	1458.74	1396.97	1397.60	1436.28	1437.01	1408.02	1408.40

TABLE III.2-4

COMPARISONS OF PREDICTED SURFACE TEMPERATURES (°F) FOR BENNETT'S 219-INCH VERTICAL, HEATED ROUND TUBE TEST SECTION GROENEVELD EQUATION 5.7

Run Number	5293		5380	
Heat Cond. Number	RETRAN	RELAP4	RETRAN	RELAP4
1	566.19	566.25	574.49	574.54
2	566.14	566.21	574.41	574.47
3	566.09	566.14	574.29	574.34
4	565.99	566.05	574.06	574.11
5	565.90	566.95	573.82	573.87
6	565.80	565.85	573.57	573.62
7	565.67	565.74	573.30	573.34
8	565.57	565.62	573.01	573.06
9	571.44	572.51	572.71	572.75
10	574.76	575.65	1547.16	1548.42
11	576.26	576.81	1494.65	1487.29
12	576.61	576.86	1423.72	1425.01
13	575.47	575.51	1374.50	1375.74
14	573.65	573.68	1332.01	1333.21
15	572.06	572.09	1295.01	1296.17
16	1195.88	1197.19	1262.54	1263.68
17	1169.88	1171.19	1233.85	1234.97
18	1146.98	1148.30	1208.36	1209.45
19	1126.73	1128.06	1185.58	1186.65
20	1108.74	1110.08	1165.13	1166.19

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2.2 Bennett, Collier, Pratt and Thornton Annulus Data

The primary objectives of the experiments conducted by Bennett et al.[III.2-3] were to obtain wall temperature and pressure drop data in the annular flow regime. Steam and water were introduced into an annulus test section. After a suitable mixing length, the two-phase mixture entered a heated test section. The experiments analyzed by Hughes and Fujita [III.2-1] were for a test section 29 inches long. The inside diameter was 0.547 inch and the outside diameter was 0.623 inch.

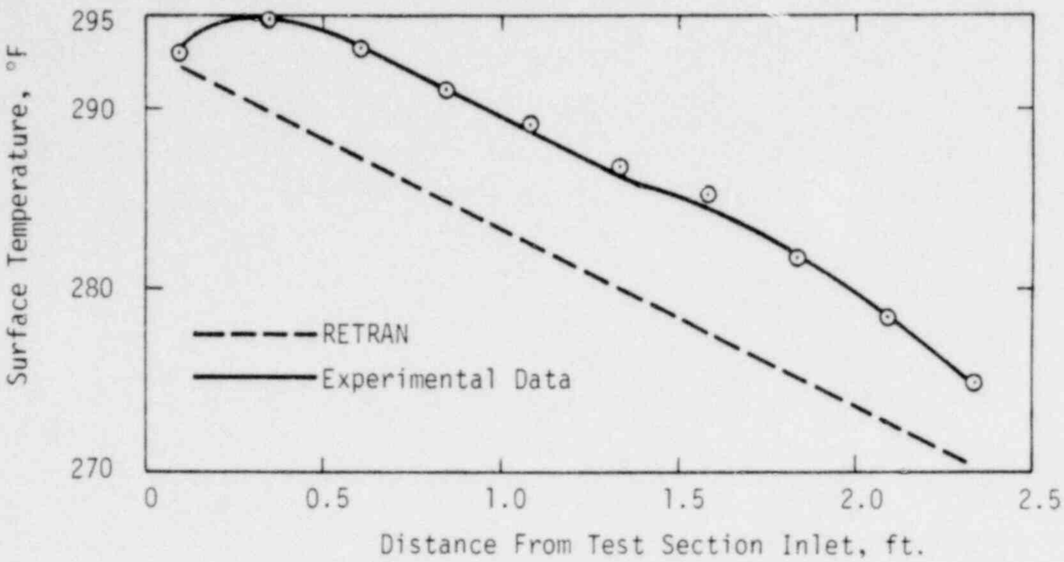
The experiments were conducted in a manner similar to that discussed in Section III.2-1, except that the test section pressure was close to atmospheric pressure. Ten wall temperature measurements and six pressure measurements were made in the test section. These data, in addition to inlet quality, flow rates and fluid temperatures, were reported.

2.2.1 Description of Model

Analyses were performed for six experimental runs with both RETRAN and RELAP4. The models used for the analyses are similar to those described for the round tube (Section III.2.1.1) except that 10 control volumes and heat conductors were used to model the test section. Boundary conditions were specified at the inlet with a fill junction and at the outlet with a time-dependent volume. The RETRAN analyses were performed using the steady-state option along with a null transient, while the RELAP4 analyses were conducted by running a pseudo-transient.

2.2.2 Results and Data Comparisons

Comparisons between the RETRAN analyses and experimental values for wall temperature and pressure are shown in Figures III.2-7 to III.2-12. Since the flow for these experiments was annular and dryout did not result, the data are good for evaluating the forced-convection vaporization heat transfer regime. In general, RETRAN underpredicted the wall temperature for this regime, although by a small (~5 °F) amount. The Baroczy two-phase multiplier was used to determine the friction losses, and the results shown in Figures III.2-7 to III.2-11 indicate good agreement with experiment. For the low mass flow case (Figure III.2-12), RETRAN showed a slightly larger pressure gradient than was observed experimentally.



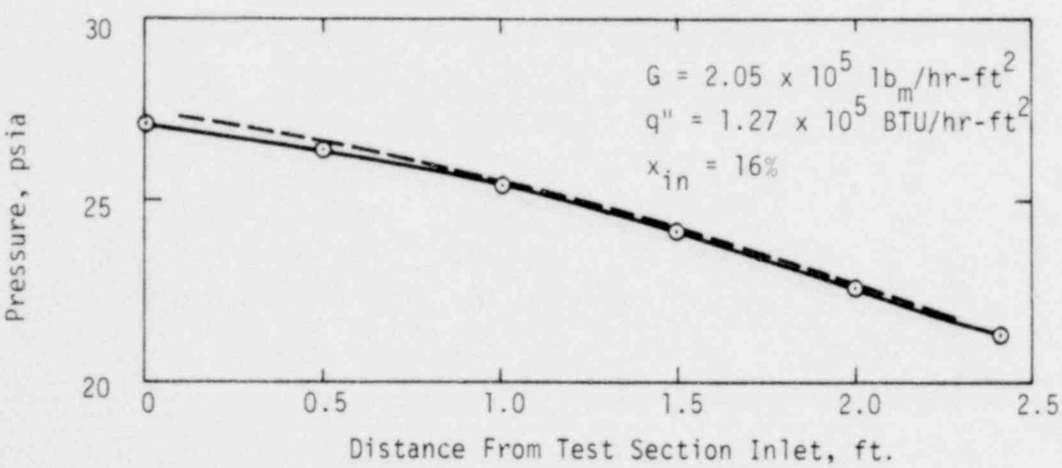


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FIGURE III.2-7 Comparison of Predicted Results with Experimental Data for Vertical Heated Annulus-Q-2 RETRAN Steady-State Initialization with "FILL" and "TDV".

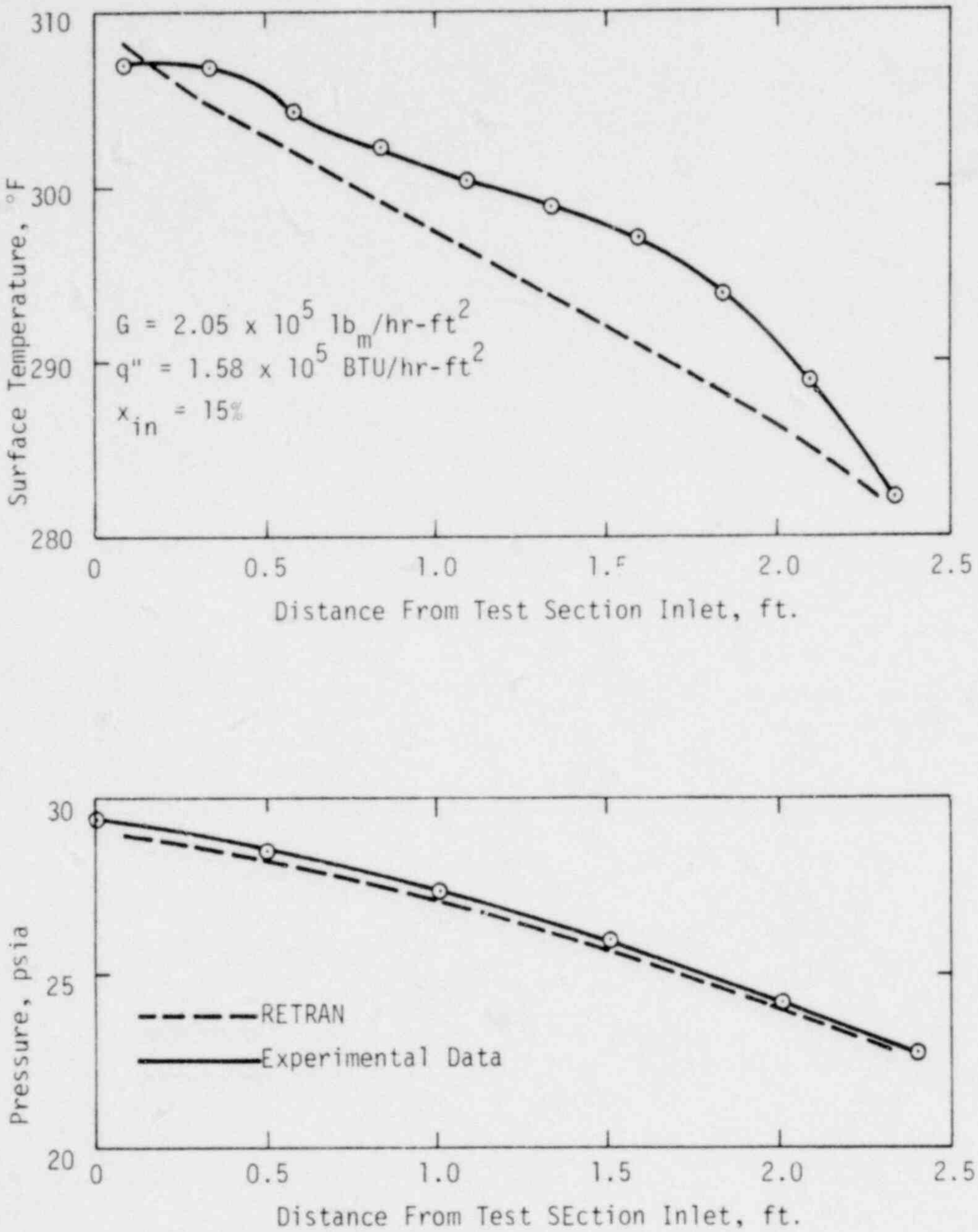

FIGURE III.2-8 Comparison of Predicted Results with Experimental Data for Vertical Heated Annulus - Run C-6 RETRAN Steady-State Initialization with "FILL" and "TDV".

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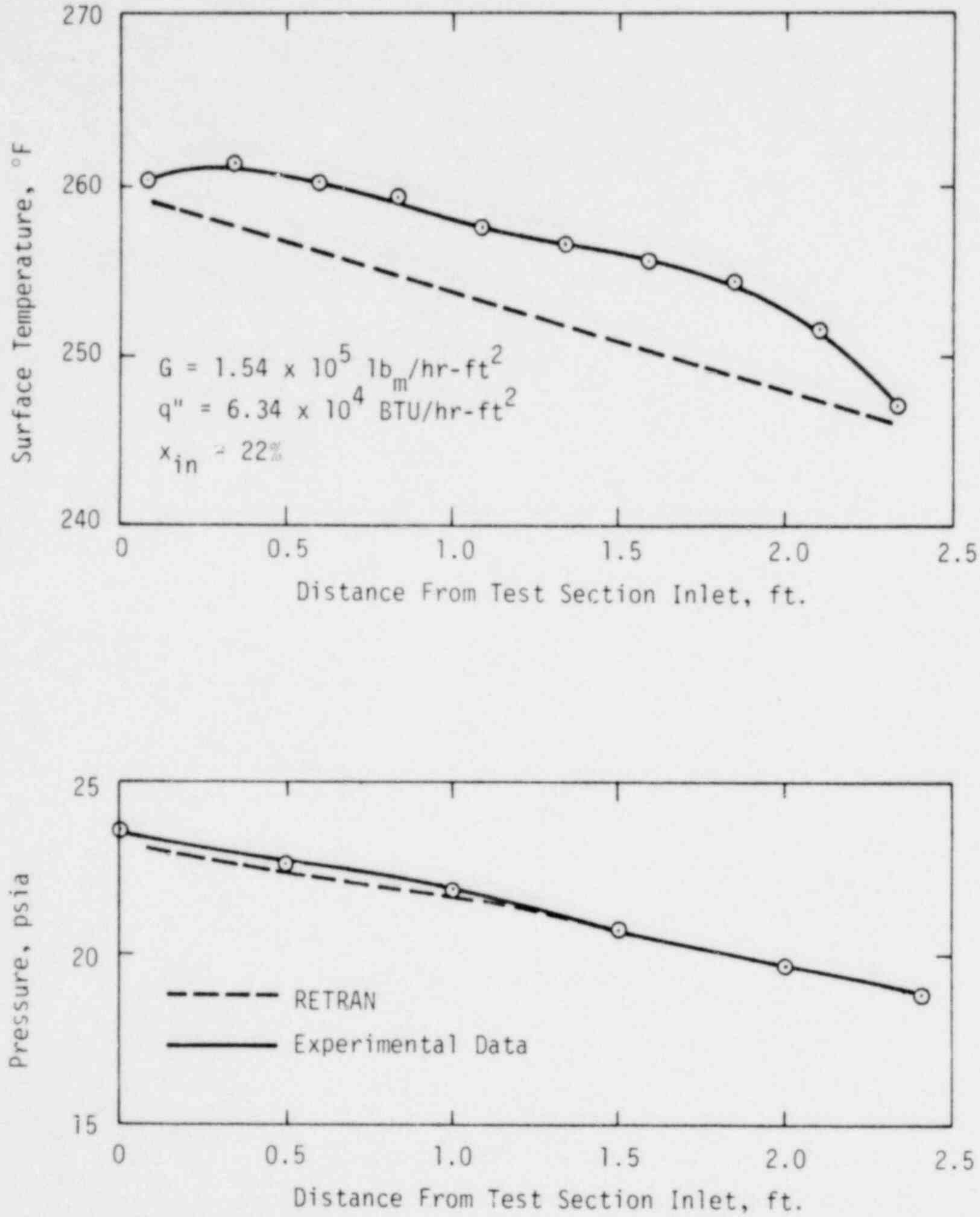

FIGURE III.2-9 Comparison of Predicted Results with Experimental Data for Vertical Heated Annulus - Run C-31 RETRAN Steady-State Initialization with "FILL" and "TDV".

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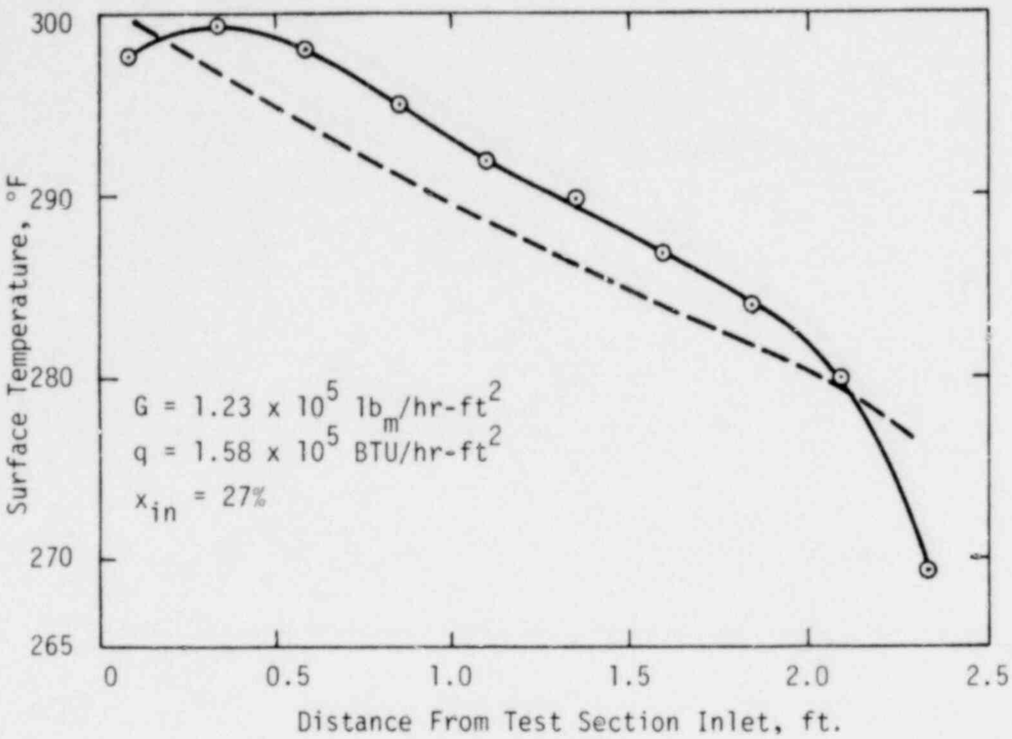

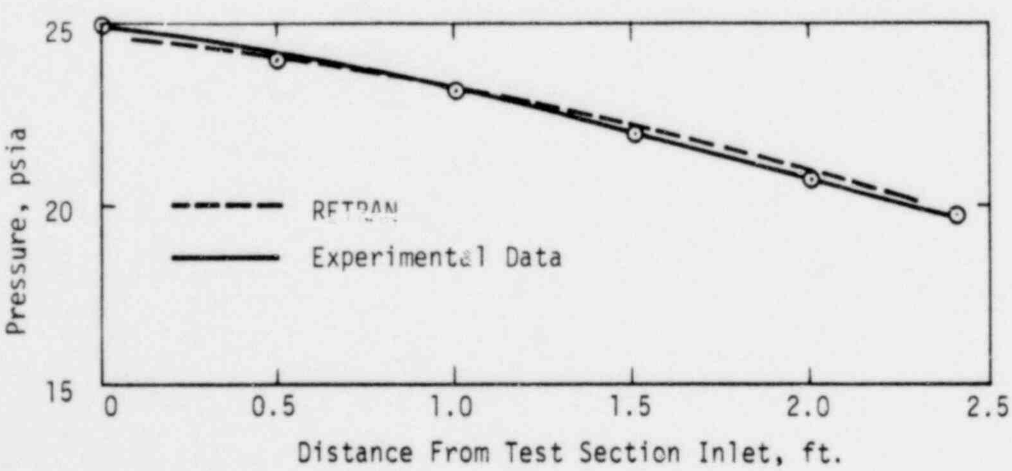

FIGURE III.2-10 Comparison of Predicted Results with Experimental Data for Vertical Heated Annulus - Run C-39 RETRAN Steady-State Initialization with "FILL" and "TDV".

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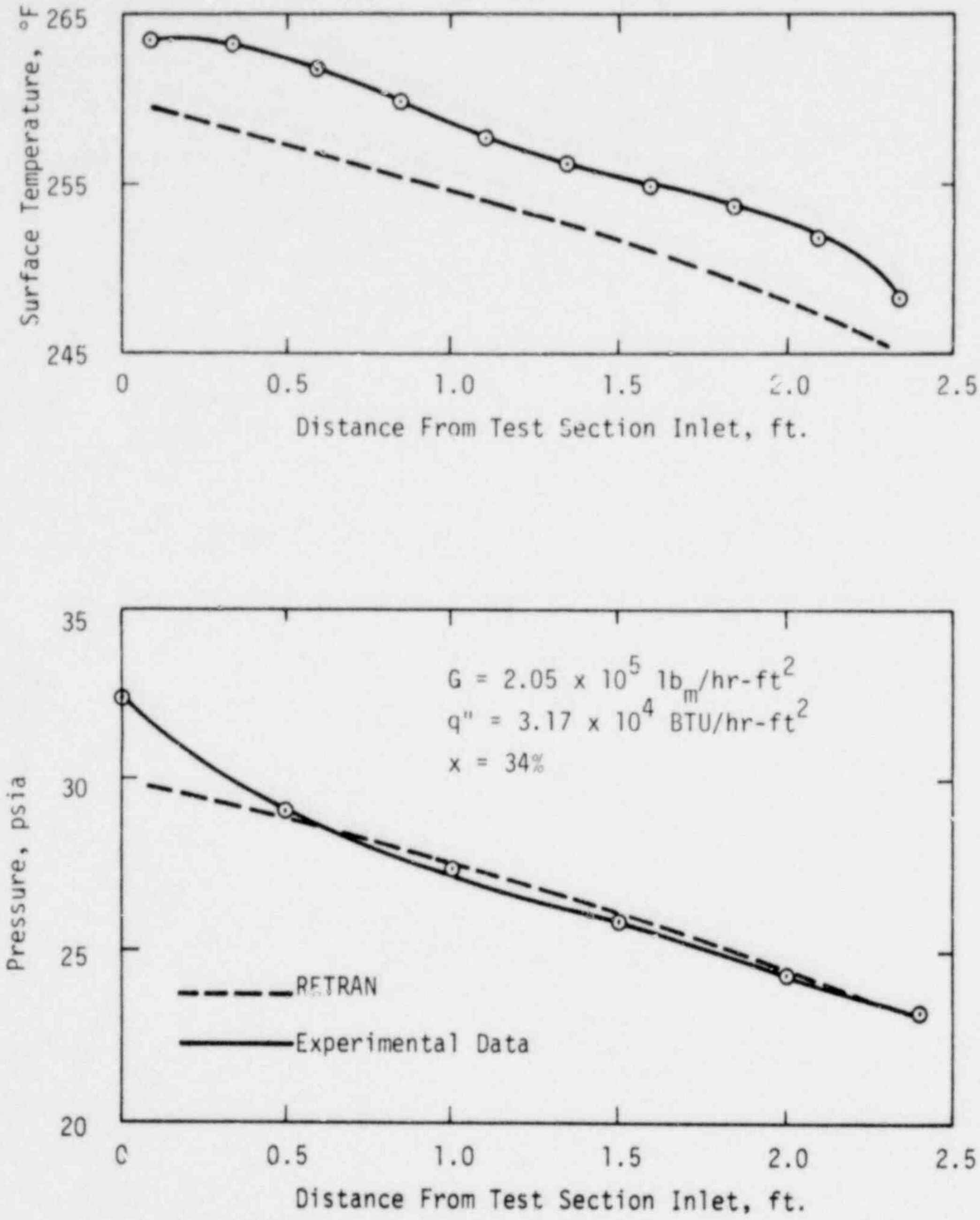

FIGURE III.2-11 Comparison of Predicted Results with Experimental Data for Vertical Heated Annulus - Run C-63 RETRAN Steady-State Initialization with "FILL" and "TDV".

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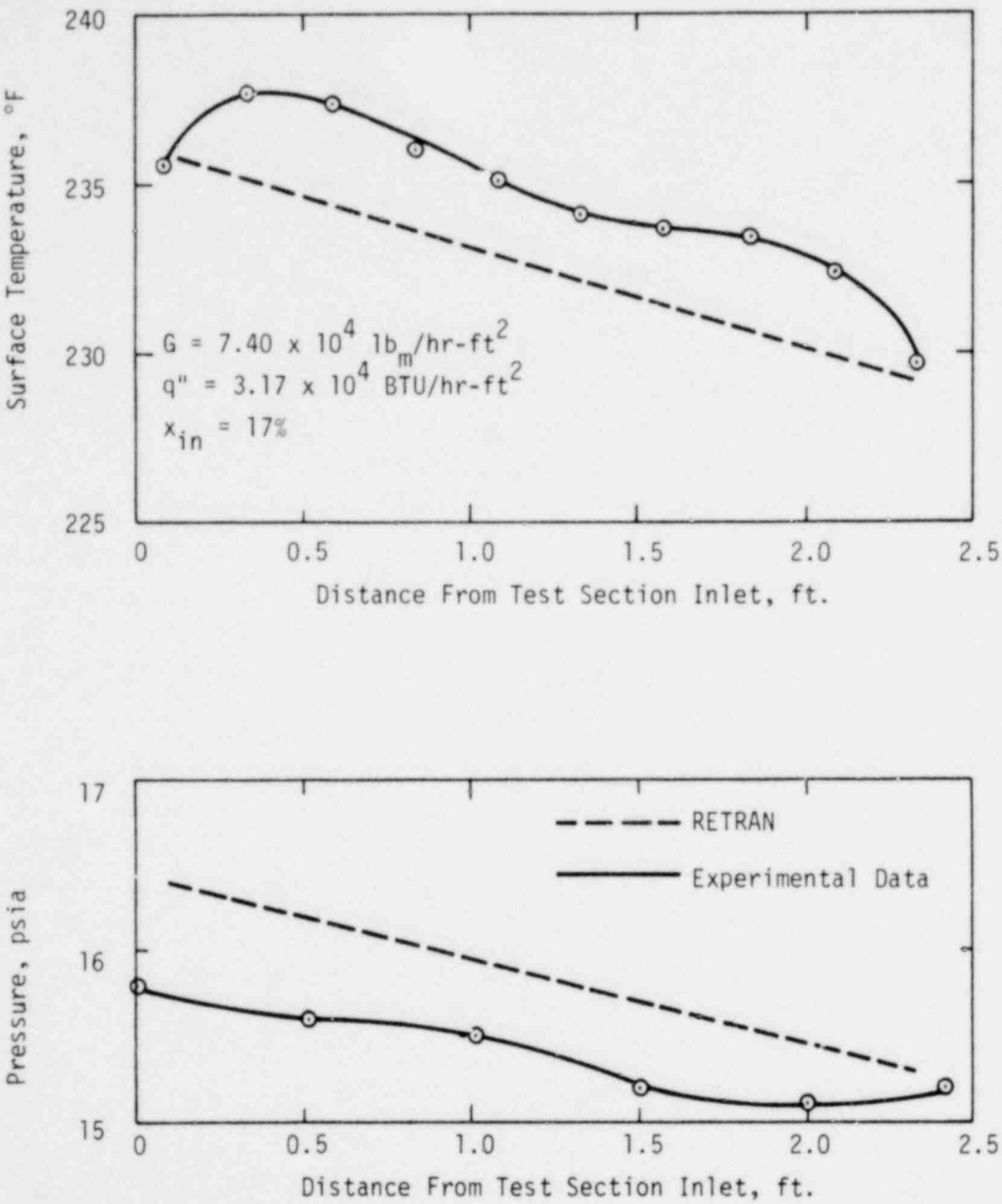

FIGURE III.2-12 Comparison of Predicted Results with Experimental Data for Vertical Heated Annulus - Run C-85 RETRAN Steady-State Initialization with "FILL" and "TDV".

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Tables III.2-5 and III.2-6 show that RETRAN and RELAP4 give essentially the same values for these experiments. This result indicates the steady-state initialization option in RETRAN is coded correctly for the logic used in analyzing these experiments.

2.3 Schrock-Grossman Round Tube Data

A third set of steady-state heat transfer experiments which have been analyzed with RETRAN are those conducted by Schrock and Grossman[III.2-4] In these experiments, subcooled water was introduced at the inlet of a heated round tube, vertically oriented. The experimental conditions are such that forced convection vaporization is the heat transfer mode for most of the test section. This occurs due to the high surface heat flux imposed on the test section.

Experimental data reported include inlet subcooling and pressure, exit quality, and pressure and wall temperature profiles in the test section. The inside diameter of the test section for these analyses was 0.118 inches, and two test section lengths were evaluated, 30 inches and 40 inches.

2.3.1 Description of Model

Hughes and Fujita[III.2-1] evaluated four experimental runs with RETRAN, RELAP4, and UVUT, an unequal velocity, unequal temperature code. Only RETRAN and RELAP4 results are reported here. The test section was modeled with 10 volumes and 10 conductors. As in the previous analyses, the boundary condition at the inlet was modeled with a fill junction while a time-dependent volume was used for the outlet boundary condition. The steady-state option was used for RETRAN while a pseudo-transient was executed to achieve the RELAP4 analyses.

2.3.2 Results and Data Comparisons

Comparisons of the RETRAN prediction with the experimental data for test section pressures and wall temperatures are given in Figures III.2-13 to III.2-16. The computed wall temperatures are within 2 percent of the experimental data except for points close to the test section inlet. The code results for this region indicated nucleate boiling heat transfer. However, Hughes and Fujita[III.2-1] showed that the heat transfer mode was actually forced con-

TABLE III.2-5

COMPARISONS OF PREDICTED PRESSURES (PSIA) FOR BENNETT'S VERTICAL, HEATED ANNULAR TEST SECTION

Run Number	C85-Core 6R'		C63-Core 8R		C39-Core 8	
Volume Number	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4
1	16.42	16.40	29.97	29.91	24.60	24.62
2	16.34	16.31	29.45	29.39	24.29	24.32
3	16.22	16.20	28.82	28.76	23.90	23.93
4	16.11	16.08	28.16	28.10	23.49	23.52
5	15.99	15.96	27.48	27.42	23.04	23.07
6	15.86	15.83	26.78	26.72	22.55	22.59
7	15.72	15.70	26.04	25.99	22.01	22.06
8	15.59	15.56	25.27	25.22	21.42	21.46
9	15.44	15.42	24.45	24.39	20.75	20.80
10	15.29	15.26	23.59	23.54	20.02	20.07
CP Time	5	109	5	312	5	242

Run Number	C31-Core 8		C6-Core 8		C2-Core 6	
Volume Number	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4
1	23.01	23.07	28.98	28.94	27.68	27.34
2	22.70	22.76	28.57	28.54	27.29	26.94
3	22.31	22.36	28.06	28.02	26.79	26.44
4	21.90	21.96	27.50	27.46	26.25	25.89
5	21.47	21.53	26.89	26.85	25.67	25.30
6	21.02	21.08	26.24	26.20	25.05	24.66
7	20.54	20.60	25.54	25.50	24.38	23.98
8	20.04	20.11	24.78	24.73	23.66	23.23
9	19.52	19.58	23.95	23.90	22.88	22.44
10	18.96	19.01	23.04	22.99	22.03	21.60
CP Time	5	209	5	202	5	203

TABLE III.2-6

COMPARISONS OF PREDICTED SURFACE TEMPERATURES (°F) FOR BENNETT'S VERTICAL, ANNULAR TEST SECTION

Run Number	C85-Core 6R'		C63-Core 8R		C39-Core 8	
Heat Cond. Number	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4
1	235.92	235.94	259.52	259.50	298.74	299.27
2	235.41	235.27	258.58	258.48	296.80	296.83
3	234.62	234.48	257.29	257.19	294.04	294.08
4	233.84	233.71	255.94	255.84	291.46	291.51
5	233.07	232.94	254.51	254.42	289.00	289.05
6	232.31	232.18	253.01	252.91	286.59	286.64
7	231.55	231.42	251.41	251.31	284.16	284.22
8	230.79	230.66	249.68	249.58	281.66	281.73
9	230.03	229.90	247.82	247.70	279.70	279.13
10	229.26	229.12	245.78	245.66	276.27	276.36

Run Number	C31-Core 8		C6-Core 8		C2-Core 6	
Heat Cond. Number	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4
1	258.74	259.01	307.76	308.10	292.78	292.31
2	257.88	257.96	305.45	305.29	291.09	290.22
3	256.56	256.65	302.30	302.14	288.66	287.78
4	255.21	255.30	299.34	299.19	286.29	285.39
5	253.81	253.90	296.51	296.36	283.95	283.01
6	252.35	252.45	293.75	293.59	281.59	280.61
7	250.82	250.93	290.99	290.83	279.18	278.16
8	249.22	249.34	288.18	288.02	276.69	275.62
9	247.53	247.65	285.28	285.11	274.08	272.94
10	245.72	245.85	282.22	282.04	271.30	270.06

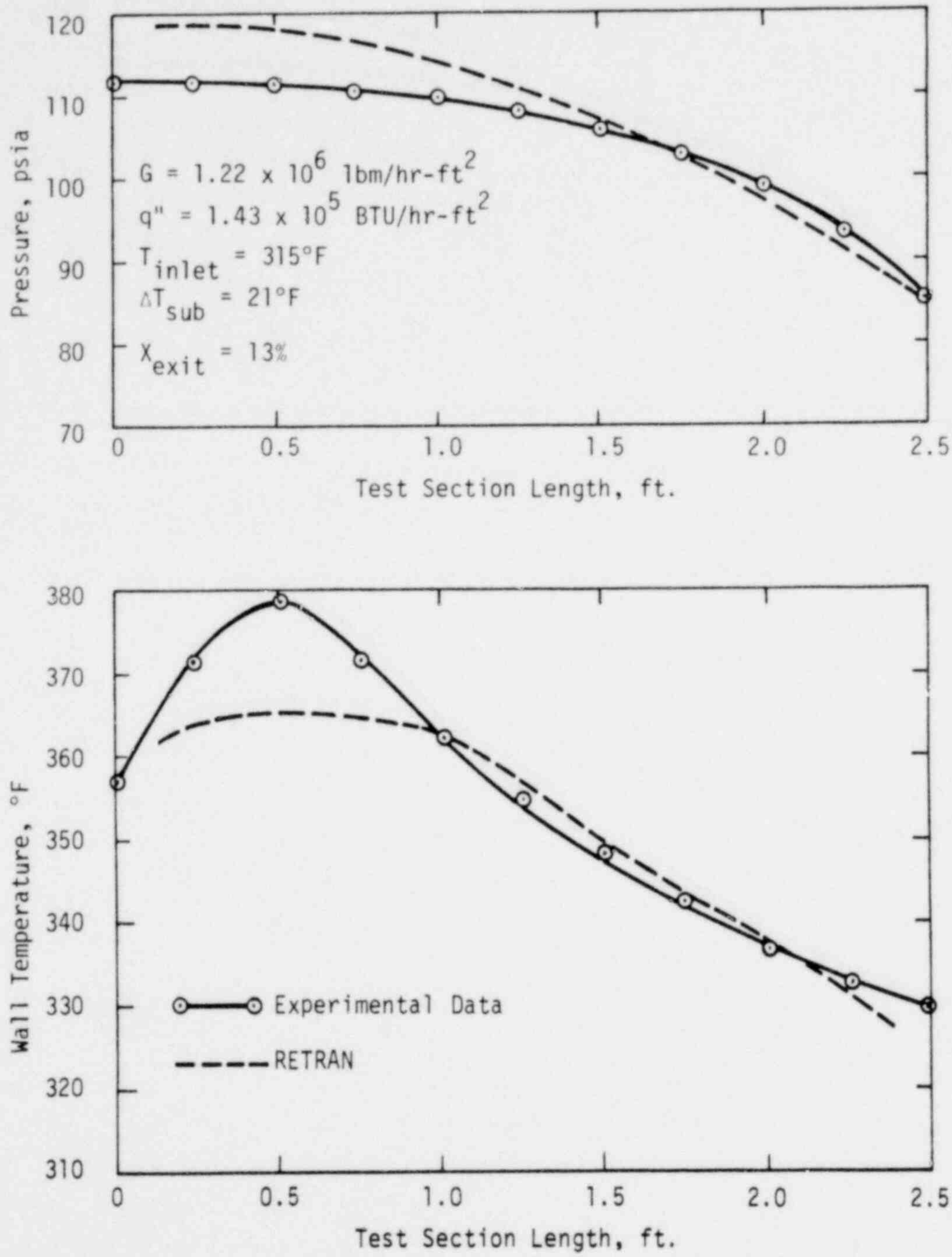

FIGURE III.2-13 RETRAN Comparisons of Schrock-Grossman Test 271. Steady-State Initialization with "FILL" and "TDV".

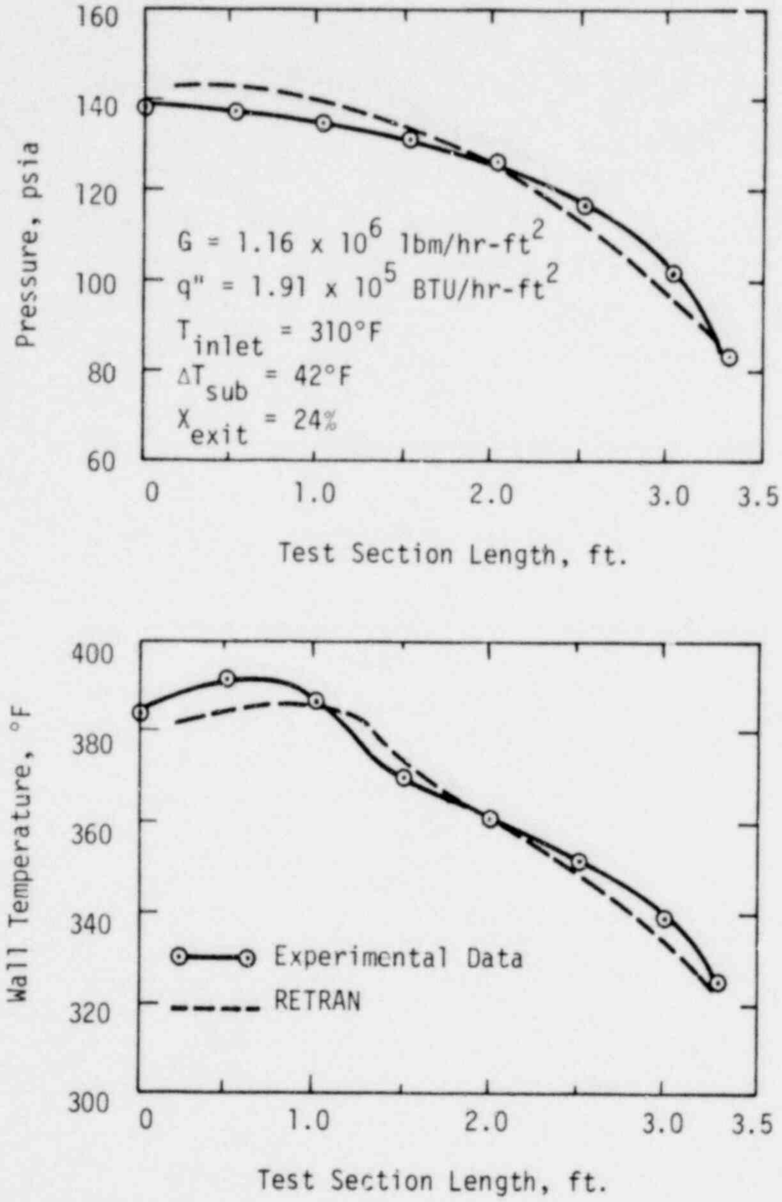

FIGURE III.2-14 RETRAN Comparisons of Schrock-Grossman Test 279. Steady-State Initialization with "FILL" and "TDV".

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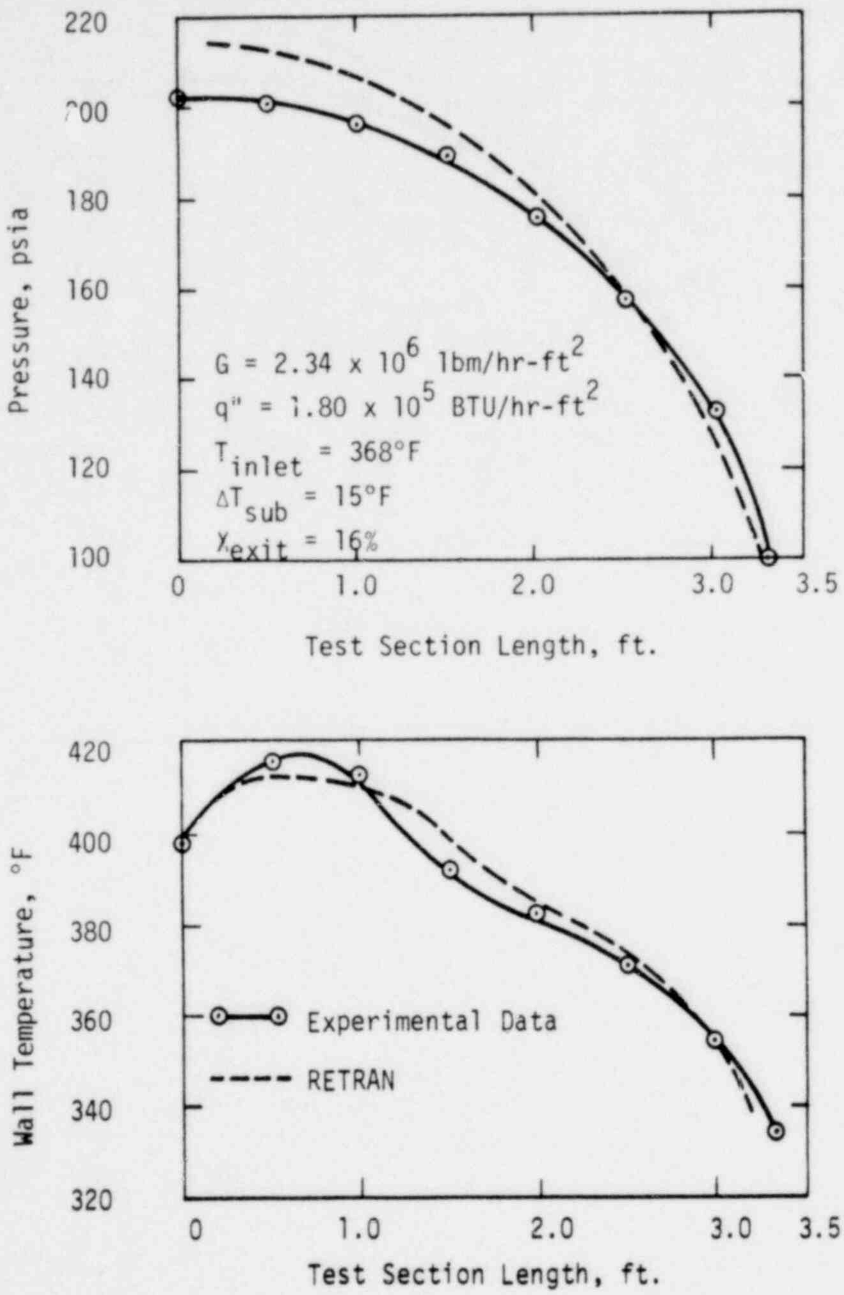

FIGURE III.2-15 RETRAN Comparisons of Schrock-Grossman Test 281. Steady-State Initialization with "FILL" and "TDV".

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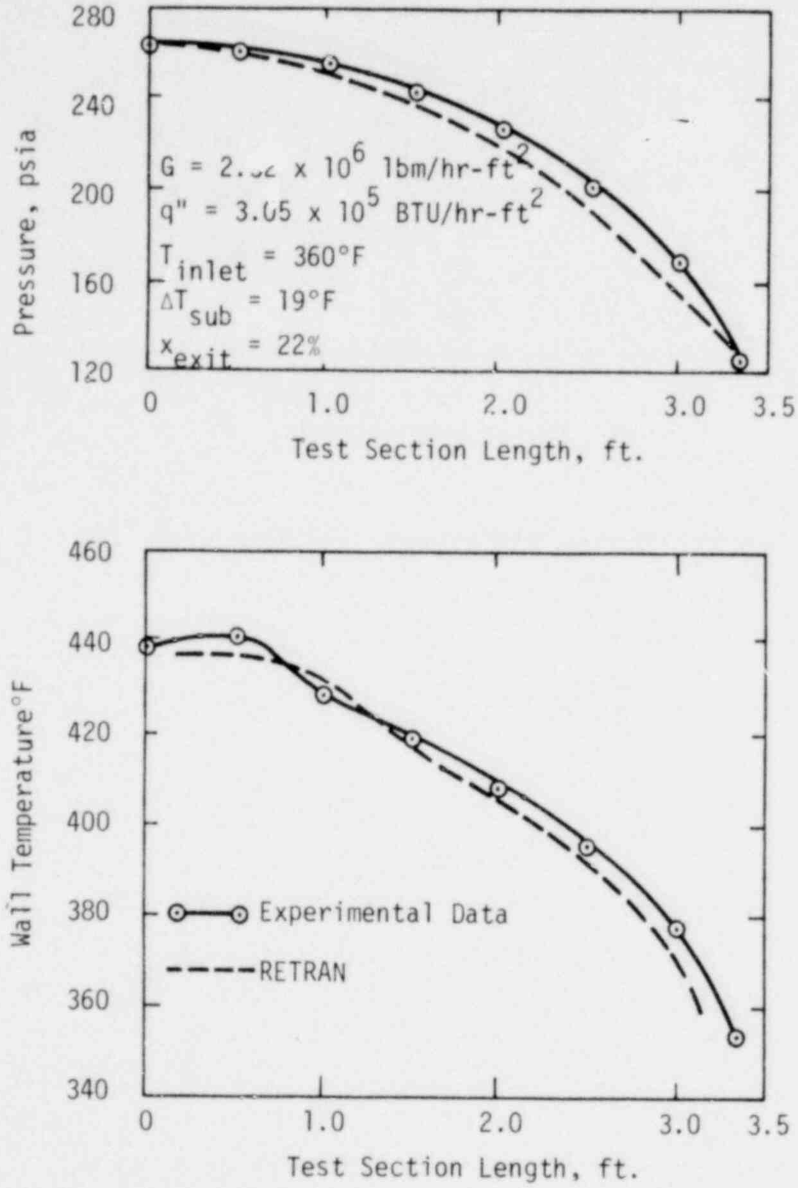

FIGURE III.2-16 RETRAN Comparisons of Schrock-Grossman Test 282. Steady-State Initialization with "FILL" and "TDV".

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vection vaporization, which results in a lower heat transfer (higher wall temperatures) than does nucleate boiling. The pressure distribution, based on use of the Baroczy two-phase multiplier, is generally above the experimental data, although the differences are not large. RELAP4 calculations gave essentially the same results as RETRAN for wall temperatures and pressure (Table III.2-7) in the test section.

2.4 Summary of Results

The comparisons presented in the previous sections between RETRAN calculations and experimental data lead to the following conclusions for the RETRAN heat transfer relationships and the Baroczy two-phase multiplier:

(1) The pre-CHF correlations give good agreement with experimental data for steady-state experiments.

(2) The Groeneveld stable film boiling correlations provide an accurate representation of this type of heat transfer for the experiments.

(3) The Barnett CHF correlation predicts CHF to occur sooner than was observed experimentally. Additional analyses with bundle data as well as round tube data should be performed to further qualify the Barnett correlation.

(4) The Baroczy two-phase multiplier gave good agreement with data with the possible exception of low mass flux. Analyses should be performed for low mass flux experiments with and without external heat sources.

The comparisons between RETRAN and RELAP4 calculations indicate that:

(1) The conversion of the RELAP4 code to RETRAN was correct for heat transfer and the solution of the basic equations.

(2) The steady-state option in RETRAN provides an inexpensive method for evaluating consitutive equations for the code.

9 096

TABLE III.2-7

COMPARISONS OF PREDICTED PRESSURES (PSIA) FOR SCHROCK-GROSSMAN FORCED CONVECTION EXPERIMENTS

Run Number	271		279		281		282	
Volume Number	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4	RETRAN	RELAP4
1	119.51	118.83	145.99	142.18	216.19	214.09	263.11	262.10
2	119.27	118.59	145.70	141.88	214.86	212.83	260.60	259.69
3	118.54	117.82	144.43	140.37	210.25	208.26	254.26	253.70
4	116.17	115.57	138.94	136.86	203.93	202.27	246.74	246.18
5	113.19	112.63	134.24	132.28	196.28	194.85	237.49	236.96
6	109.55	108.99	128.31	126.67	187.19	185.81	226.18	225.65
7	105.18	104.68	121.37	119.81	176.01	174.85	212.12	211.61
8	100.19	99.69	112.42	111.33	161.39	161.20	194.39	193.91
9	94.32	93.83	101 [illegible]8	100.73	142.78	142.77	171.13	170.64
10	86.66	86.69	86.72	86.58	107.03	107.17	131.43	131.26
CP Time	134	6	172	6	214	7	252	7

3.0 CRITICAL FLOW

The RETRAN code is designed primarily for the analyses of transient thermal-hydraulic conditions in complex systems. The constitutive equations, including the critical flow models, are based on steady-state conditions and applied in transient calculations with local, instantaneous values of the appropriate independent variables. The purpose of this section is to compare RETRAN analyses with simple problems.

Three analyses were performed which involved an evaluation of the critical flow models in RETRAN. Kmetyk and Ginsberg[III.3-1] modeled a two-volume system to study the junction choking options. They also investigated critical flow in the subcooled, saturated and superheated regions for various representations of the momentum equation. Cross[III.3-2] and Hughes and Fujita[III.3-3] analyzed experimental steady-state critical flow experiments conducted by Fauske[III.3-4]. In addition to evaluating the critical flow options, the two-phase multiplier options, noding schemes, boundary conditions and momentum flux representations were investigated.

3.1 Two-Volume Critical Flow Problem

The two-volume model of Kmetyk and Ginsberg[III.3-1] is shown in Figure III.3-1 along with the initial thermodynamic conditions. The purpose of the analyses was to evaluate the critical flow option and critical flow tables in RETRAN and to investigate some of the possible momentum equation representations when applied to a critical flow problem.

The results are summarized in Table III.3-1 and the accompanying notes. The computed flow rate gave correct values for the appropriate table choking models in the saturated and superheated regions (Figure III.3-2). However, in the subcooled liquid region, Kmetyk and Ginsberg showed the Moody and Henry-Fauske values for critical flow are not correct. This is due to the fact that these models are not based on subcooled conditions, although the code input allows the user to specify either of these models for such a case. The extended Henry-Fauske model, which is based on subcooled conditions, produced correct values for critical flow in this region.

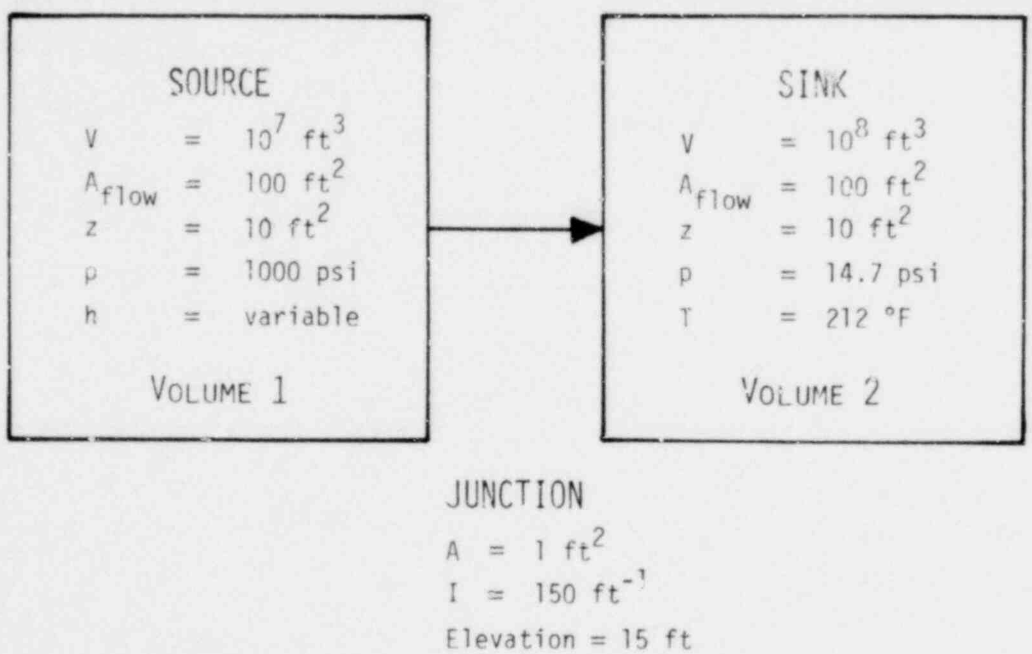

FIGURE III.3-1 Model for Two-Volume Critical Flow Study

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TABLE III.3-1

SUMMARY OF TWO VOLUME CRITICAL FLOW STUDY

Model	Subcooled Region	Saturated Region	Superheated Region
Henry * Fauske	extrapolation flow error - note 3 pressure error	tabulated flow rates reproduced correctly - note 1 frictional pressure differential error	extrapolated flow rates look reasonable - note 2 pressure error
Moody *	extrapolation flow error - note 3 pressure error	tabulated flow rates reproduced correctly - note 1 frictional pressure differential error	tabulated flow rates reproduced correctly - note 1 frictional pressure differential error
extended * Henry-Fauske	tabulated flow rates reproduced correctly - note 1 pressure error	not applicable	not applicable
Incompressible	flow rates and pressures look reasonable - note 4	flow rates and pressures look reasonable - note 4	flow rates and pressures look reasonable - note 4
Sonic ----------			
Compressible	instability - note 5	instability - note 5	instability - note 5

*for table choking, MVMIX=0 (compressible) and MVMIX=3 (incompressible) give identical results - see note 1

1079 100

NOTES FOR TABLE III.3-1 (MODIFIED FROM REFERENCE III.3-1)

Note 1: Moody and Henry-Fauske models were checked for the following cases -

a) saturated water for source pressures (i.e., stagnation pressures) of 100, 400, 600, 1000, 1400, 1800 and 2200 psi,

b) saturated steam for pressures of 100, 400, 600, 1000, 1400, 1800 and 2200 psi.

c) saturated steam/water mixture for a source pressure of 1000 psi and qualities of 0.2, 0.4, 0.6 and 0.8, and

d) superheated steam (Moody model only) for a source pressure of 1000 psi and source enthalpies of 1390.6, 1590.06 and 1790.2 (Btu/lb).

Extended Henry-Fauske model was checked for a source pressure of 1000 psi for subcooled water at enthalpies of 243.1, 355.9 and 476.2 Btu/lb.

In all cases, the tabulated mass flow rate was reproduced correctly. Since setting JCHØKE=1 gives incompressible flow whether MVMIX=0 or MVMIX=3, the results for both values of MVMIX were identical.

Some of these results are shown in Table III.3-2 and Figure III.3-2.

Note 2: Setting ICHØKE=1 or 2 gives the corresponding table choking in the superheated as well as saturated regions. While the Moody tables extend into the superheated region, the Henry model extrapolates. Enthalpies of 1390.6 and 1590.6 Btu/lb for a source pressure of 1000 psi were run. The extrapolated Henry-Fauske results agree well with the tabulated Moody flow values.

Note 3: The RETRAN manual (Volume III) allows the option (ICHØKE > 0) of using the Moody and/or Henry-Fauske tables for choking in the subcooled region. Cases for enthalpies of 243.1, 355.9 and 476.1 Btu/lb at a pressure of 1000 psi were run. The flow rates obtained indicate that this option, if not totally eliminated, should be flagged in the manual as a risky procedure. In particular, using ICHØKE=+2 (Henry in saturated and subcooled regions) does not give extended Henry-Fauske in a subcooled region, as might be expected.

The only options that we believe should be used in the subcooled region are sonic choking and/or extended Henry-Fauske. It is unfortunate that these two options are mutually exclusive (not only is sonic or extended Henry in the subcooled region forbidden, but so is the combination of sonic in the saturated region and extended Henry in the subcooled region). Extended Henry-Fauske is used only if ICHØKE $\geq$ 10, but sonic choking is always suppressed (even in the saturated region) if ICHØKE $\geq$ 10.

Note 4: Incompressible (MVMIX=3) sonic choking (JCHØKE=2) cases were run for the following cases -

a) super heated steam at a pressure of 1000 psi and enthalpies of 1390.6 and 1590.6 Btu/lb,

b) saturated steam/water mixtures at a pressure of 1000 psi and qualities of 0, 0.2, 0.4, 0.6, 0.8, and 1, and

c) subcooled water at a pressure of 1000 psi and enthalpies of 243.1, 355.9 and 476.1 Btu/lb.

The resulting critical mass flow rates are shown in Table III.3-3. Although there was no way for us to calculate the sound speed RETRAN was using (which is why we feel that the sound speed CMACHJ should be added to the junction minor edit variables), Table III.3-3 shows the choked flows obtained from RETRAN compared to choked flows calculated using the choking velocity from the 1967 ASME steam tables. Those numbers which can be obtained from the steam tables (high-quality saturated mixtures and superheated steam) show very good agreement.

For the subcooled liquid cases, sonic choking never occurred, increasing the source mass by a factor of 10 did not significantly alter the maximum flow rate obtained. This combined with the fact that the sonic choked flow for saturated water seems low when compared to table choked flow for saturated water suggests that the calculation of sound speed for saturated and subcooled water should be rechecked.

Note 5; Compressible (MVMIX=0) sonic (JCHØKE=2) choking run for the same cases as incompressible sonic choking exhibits numerical instabilities in junction mass flow rate, specific volume and pressure differentials. The size of the oscillations increase as the volume fraction of steam increases and subcooled water gave essentially no difference between compressible and incompressible models since liquid water is not particularly compressible. As shown in Figure III.3-3 the oscillations are completely damped as quality drops to 0.2, but a problem with saturated water is clearly visible.

TABLE III.3-2

CRITICAL FLOW CASES (p=1000 psia)

DESCRIPTION	ENTHALPY (Btu/lb)	FLOW (lb/sec) MOODY	HENRY (ex. H.)	SONIC-INC.	COMP. SONIC
Subcooled	243.2	1613	4163 (22751)	(20203) *	(20209) *
Subcooled	355.9	3572	(20219)	(19625) *	(19599) *
Subcooled	476.2	6286	7419 (14147)	(18703) *	(18728) *
Saturated Water	542.6	7883.7	9669.8	7012	6[illegible]21-4250
X=0.2	672.7	5277.3	4311.8	5396	3370
X=0.4	802.7	3818.1	3278.5	4547	2776-2751
X=0.6	932.8	2983.3	2718.4	4000	2597-2395
X=0.8	1062.8	2446.9	2369.9	3618	[illegible]47-2163
Saturated Steam	1192.9	2073.8	2124.9	3326	2157-1955
Superheated	1390.6	1715.8	1776	2890	1860-1675
Superheated	1590.6	1479.3	1425	2439	1614-1412
Superheated	1790.2	1323.6	-	-	-

* choking index not set - maximum of inertial solution (even multiplying the source volume by 10 - and hence the available source mass - did not produce choked flow before the maximum inertial solution was reached)

TABLE III.3-3

CRITICAL FLOW RATES FOR SONIC CHOKING (INCOMPRESSIBLE)

$p = 1000$ psi
$Aj = 1\ ft^2$
$Wcrit = \rho\ Aj\ V$

Enthalpy (Btu/lb)	V (ft/sec)	(lb/ft^3)	Wcrit (lb/sec)	W RETRAN (lb/sec)
802.7	850	5.23	4443.5	4547
932.8	1100	3.62	3985.5	4000
1062.8	1305	2.77	3615.0	3618
1192.9	1480	2.24	3318.7	3326
1390.6	2020	1.46	2940.3	2890
1590.6	2340	1.08	2517.5	2439

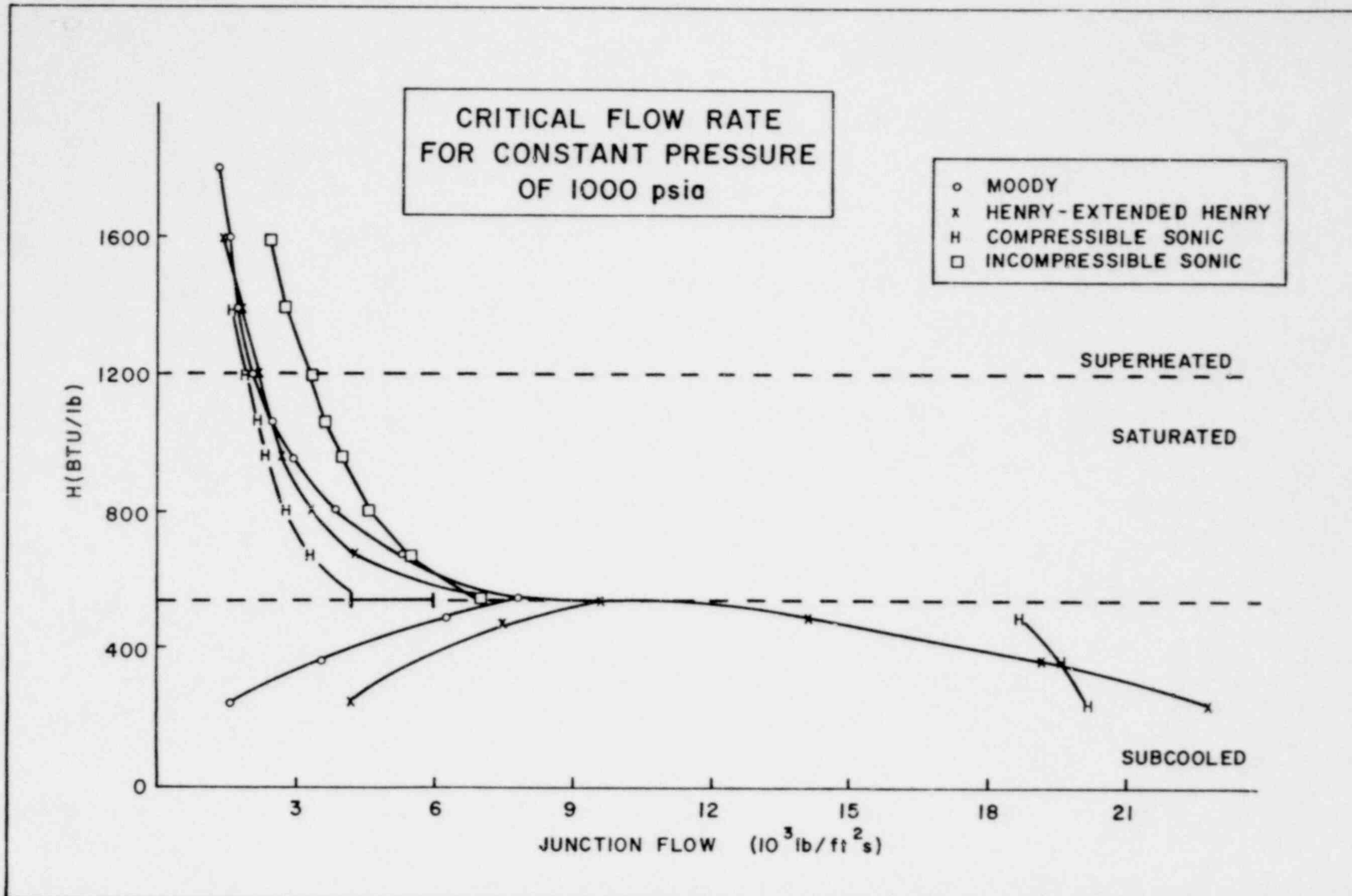

FIGURE III.3-2 Critical Flow Rate for Constant Pressure of 1000 psia.

The incompressible form of the momentum equation, when used with the sonic choking option, produced correct values of flow. Solution of the compressible form however resulted in numerical instabilities. The sonic choking is correctly calculated initially, however the instability occurs due to the compressibility caclulation in subroutine MACH. As shown in Figure III.3-3, this problem occurs for saturated water and for saturated mixtures with quality greater than 0.2.

Results of specific analyses for subcooled water and superheated steam are shown in Figures III.3-4 and III.3-5. Figure III.3-4 clearly shows that the values for the Henry and Moody models are incorrect for subcooled conditions. When these options are used in plant analyses, the critical flow multiplier is frequently changed to yield break flow values which agree with data. Critical flow modeling is discussed further in Section IX.1.7. The complete report[III.3-1] includes pressure response results along with sample input and output.

3.2 Fauske Critical Flow Experiments

Experimental results of two-phase critical flow have been reported by Fauske[III.3-4 and III.3-5]. The analyses summarized in this section were for flow in a horizontal tube with an inside diameter of 0.269 inches (6.83×10^{-3} m). The test section was 110 inches long from the inlet (end of the mixing section) to the outlet and was instrumented with 6 pressure taps.

The experiments were conducted by mixing water-vapor from a high-pressure boiler with water from a high pressure pump. This mixture entered the test section after flowing through a mixing section and exited from the test section into a 4 inch expansion. Fauske reported experimental values of pressure ($\pm$10 psi) and steam and water flow. The accuracy of the flow measurements was reported to be $\pm$5 lbm/sec for the vapor and $\pm$25 lbm/sec for the liquid.

Cross[III.3-2] based his study on Fauske's experimental run TS-II-9, and investigated the critical flow options, the two-phase friction multipliers, and the

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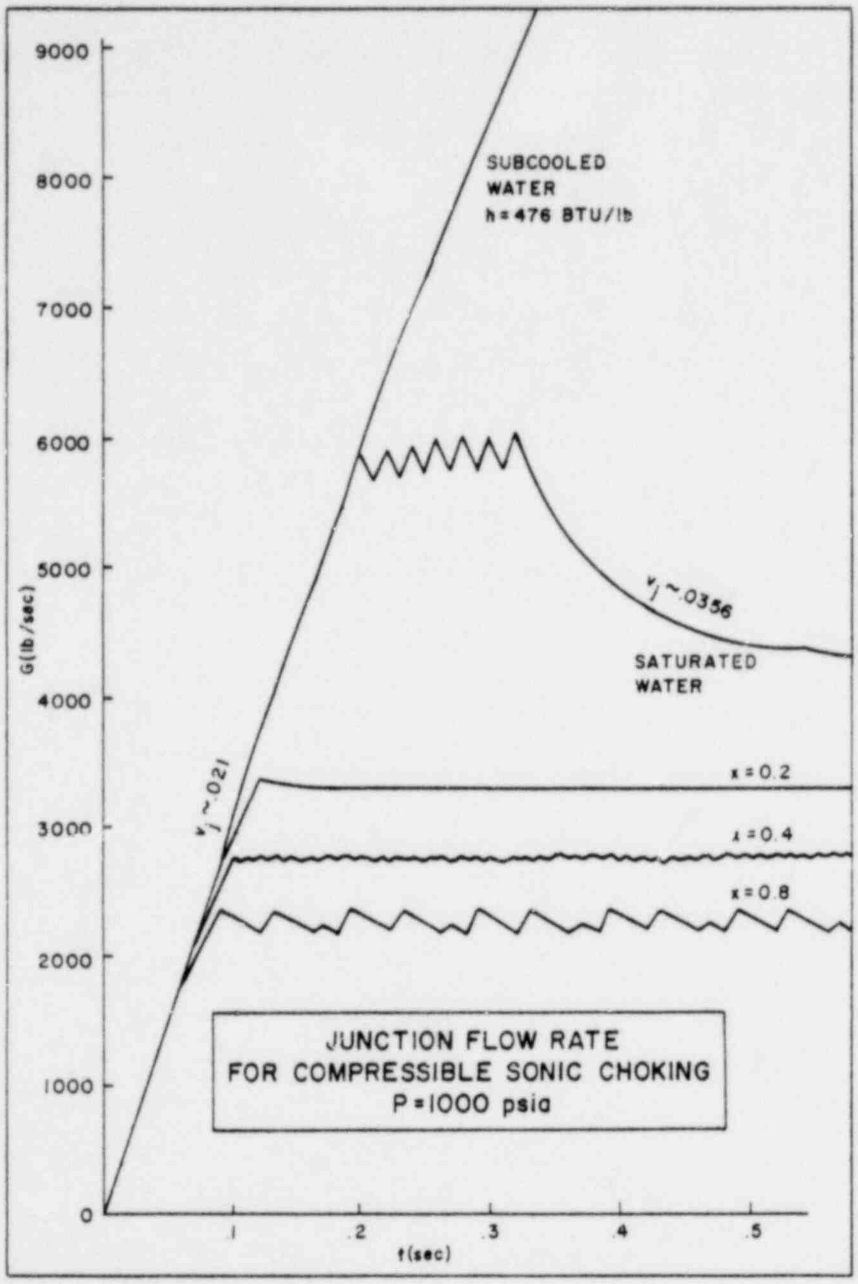

FIGURE III.3-3 Junction Flow Rate for Compressible Sonic Choking.

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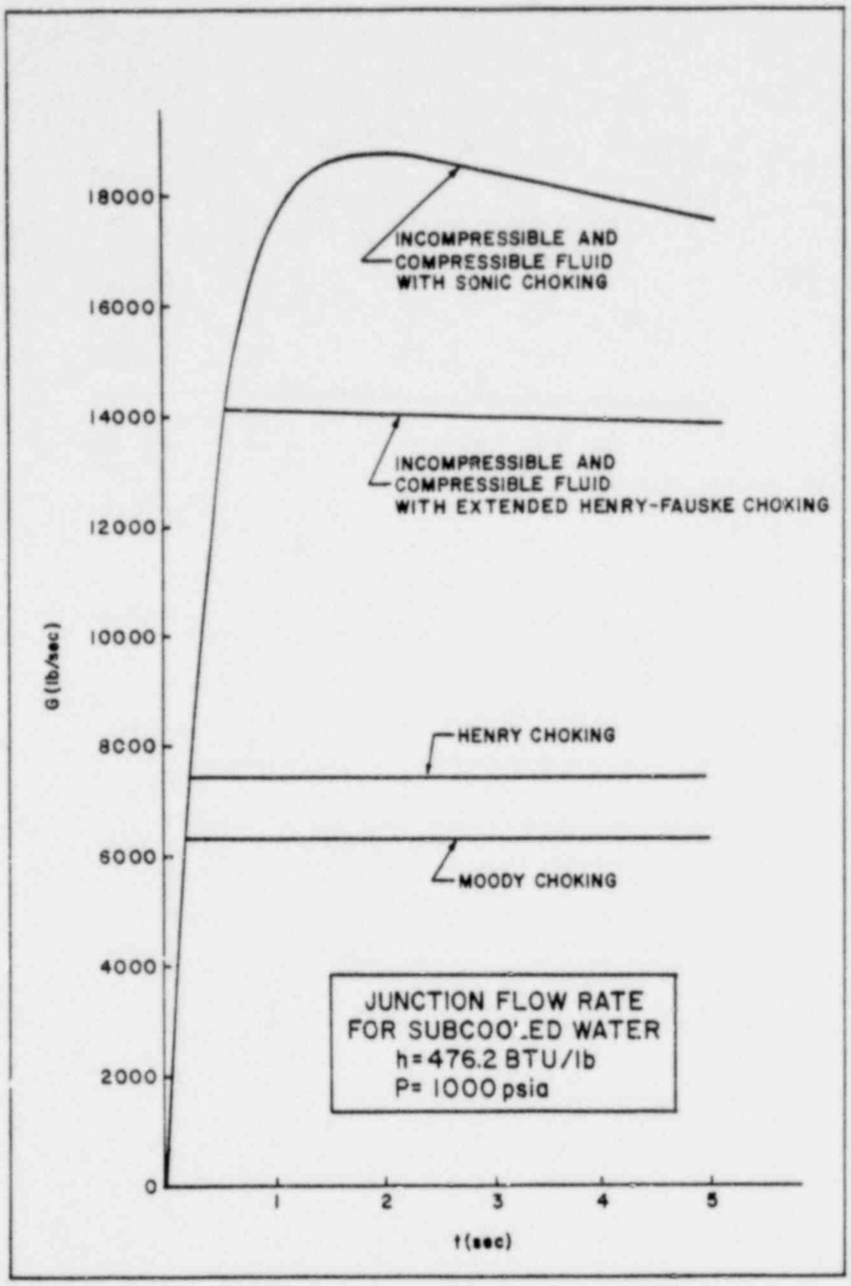

FIGURE III.3-4 Junction Flow Rate for Subcooled Water.

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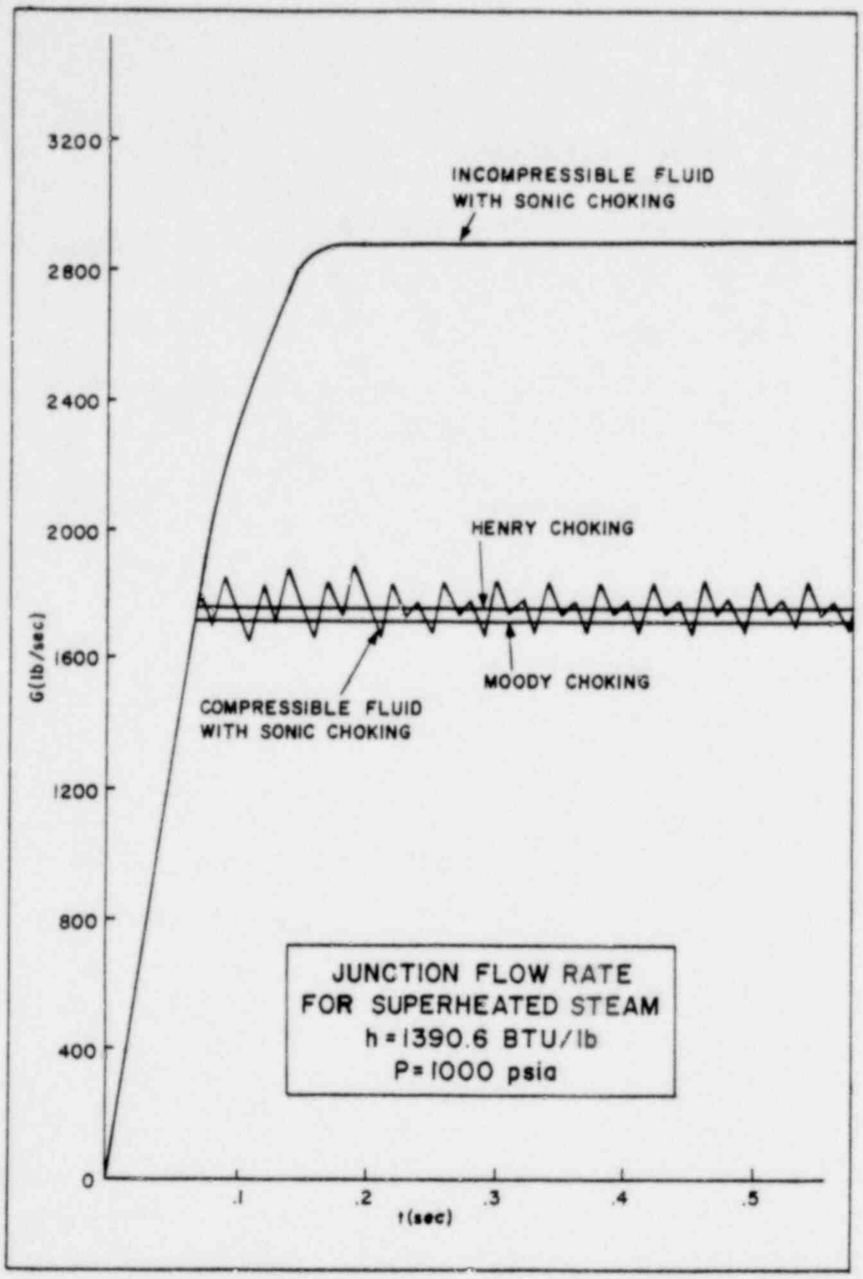

FIGURE III.3-5 Junction Flow Rate for Superheated Steam.

9 109

boundary condition options in RETRAN. Hughes and Fujita [III.3-3] analyzed three of Fauske's experimental runs and studied the critical flow options and the effect of changing the volume size.

3.2.1 Analysis by Cross

3.2.1.1 Description of Model

In the analyses performed by Cross, the last 48.125 inches of the test section were modeled with 9 volumes of equal length. Boundary conditions for the base case were applied through a fill junction at the test section inlet and a time-dependent volume at the test section outlet. For this base case, Cross investigated the Moody and Henry critical flow models, and the two-phase friction multipliers in RETRAN. In addition, he studied two other boundary condition cases;

(1) a positive and a negative fill, and

(2) two time-dependent volumes.

These two cases were run with the Moody critical flow model and the Baroczy two-phase multiplier. The momentum equation for all the runs assumed compressible flow with momentum flux.

3.2.1.2 Results and Data Comparisons

Comparisons of the experimental data with RETRAN solutions for the Moody and the Henry critical flow models are given in Figure III.3-6. In both cases, the pressure in the test section was overpredicted, and this is attributed primarily to the two-phase multiplier. The differences between the Moody prediction and the calculation with the Henry model are due to different values of the derivative of mass flow with respect to pressure at critical junctions for the two models.

With the Moody critical flow option, Cross then evaluated the three options for representing the two-phase multiplier. As can be seen in Figure III.3-7, the test section pressures for the homogeneous and Beattie multipliers were slightly above the values calculated using the Baroczy multiplier.

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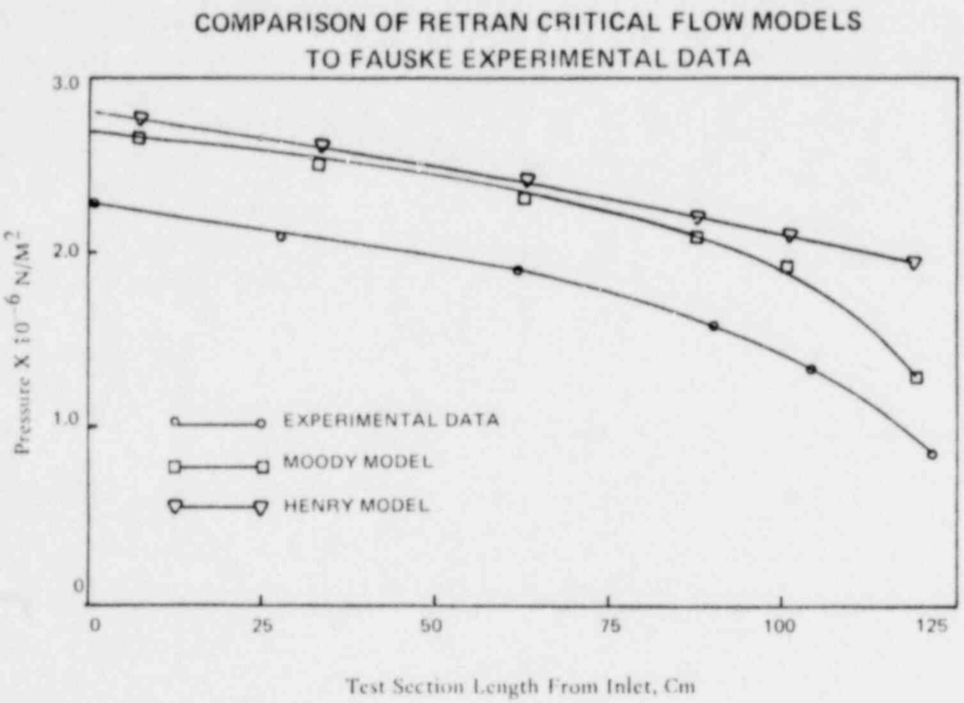

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FIGURE III.3-6 Comparison of RETRAN Critical Flow Models to Fauske Experimental Data

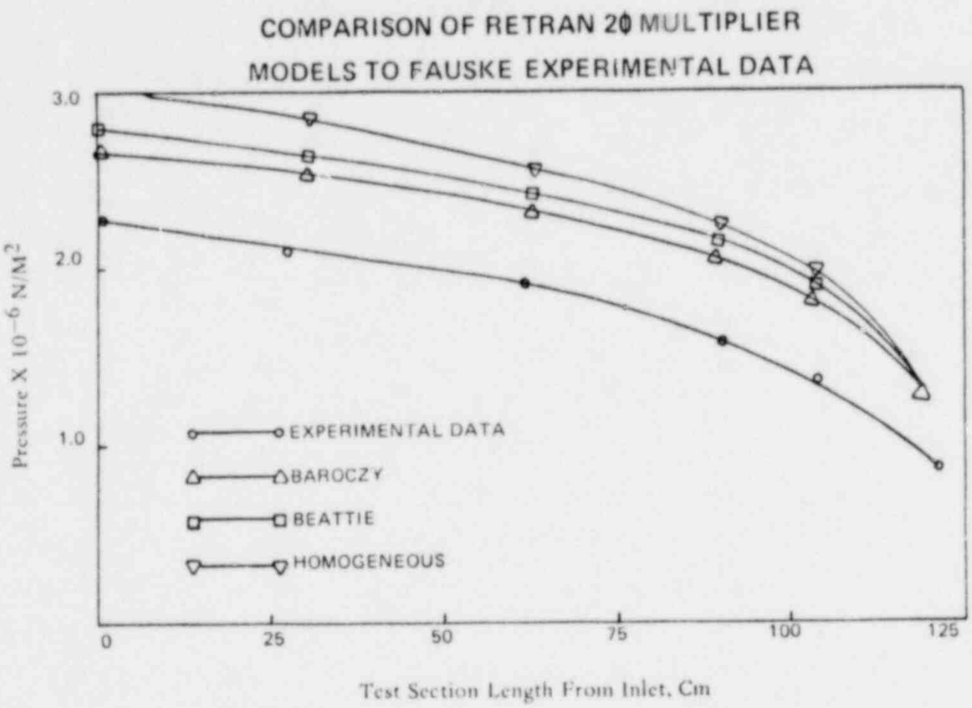

FIGURE III.3-7 Comparison of RETRAN 2Ø Multiplier Models To Fauske Experimental Data

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To study other boundary condition representations, the outlet junction was modeled with a negative fill to approximate the LEAK representation which is available in RELAP4 but is not in RETRAN. Figure III.3-8 shows that the negative fill is not an appropriate representation of the outlet boundary condition for this problem. This results from the fact that RETRAN does not solve for critical flow at negative fill junctions. The final run reported here is for the case with large, time-dependent volumes at the inlet and outlet (Figure III.3-9). This yields good agreement with the data, since the pressure is specified at both the inlet and outlet volumes.

3.2.2 Analysis by Hughes and Fujita

Three experiments were analyzed by Hughes and Fujita. As designated by Fauske, these were runs TS-II-9, TS-II-36 and TS-II-76. These three runs cover a wide range of flow rates, pressure gradients and operating pressure level.

3.2.2.1 Description of Model

The base model for this study included 9 equal size volumes representing the last 48.125 inches of the test section. Boundary conditions were applied with a fill junction at the inlet and a time dependent volume at the test section outlet. To illustrate the effect of noding, run TS-II-9 was also modeled with 4 and 18 volumes. In all the analyses, the Baroczy two-phase multiplier was used. The Moody choking model option was selected for the noding study.

3.2.2.2 Results and Data Comparisons

Comparisons of the RETRAN analyses with the experimental data are shown in Figures III.3-10 to III.3-12 for the critical flow study. All the analyses resulted in an overprediction of the pressure. This is attributed to the two-phase flow multiplier and the critical flow solution.

The sonic choking model gave the best overall agreement with the data for the analyses performed, although all the options produced reasonable results. The Moody model agreed quite well with the results from the sonic choking model except for run TS-II-76. The exit quality for this run is about 50 percent of the values for the other two tests.

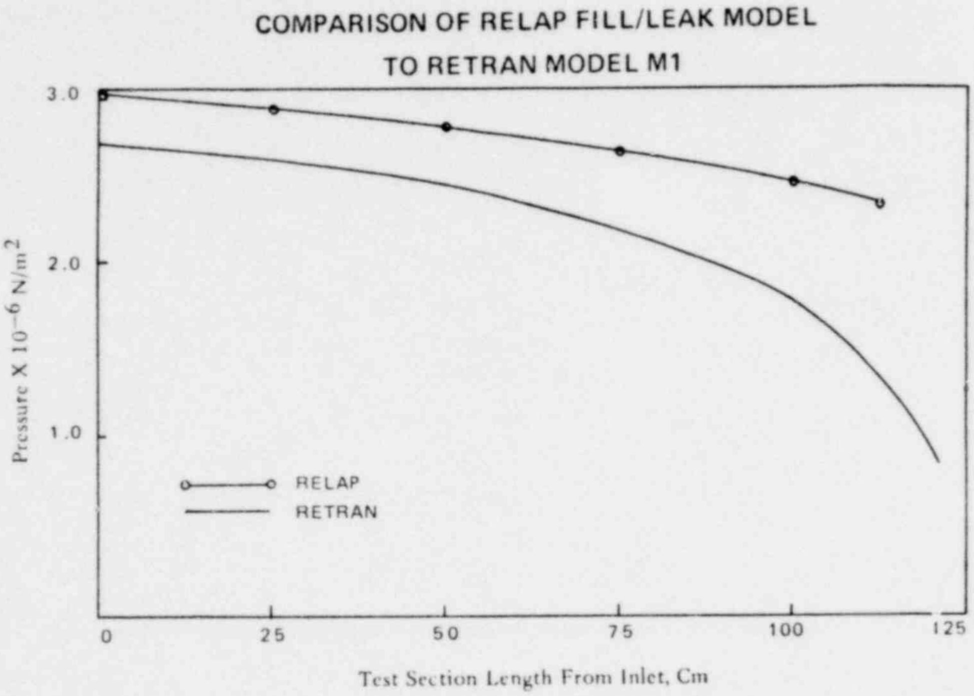

FIGURE III.3-8 Comparison of RELAP FILL/LEAK Model to RETRAN Model M1

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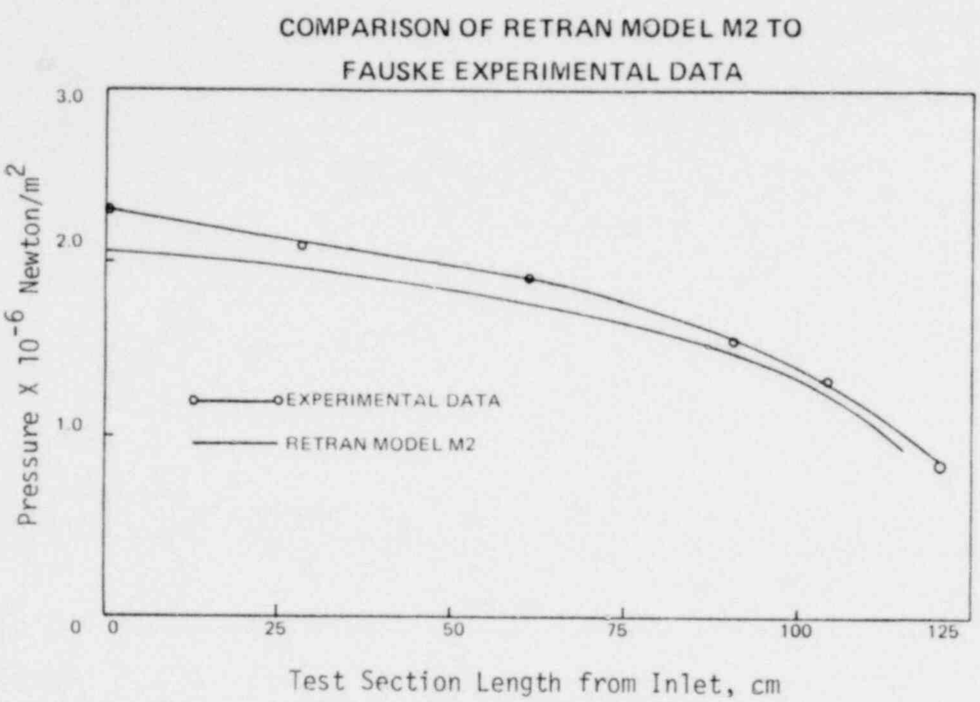

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FIGURE III.3-9 Comparison of RETRAN Model M2 to Fauske Experimental Data

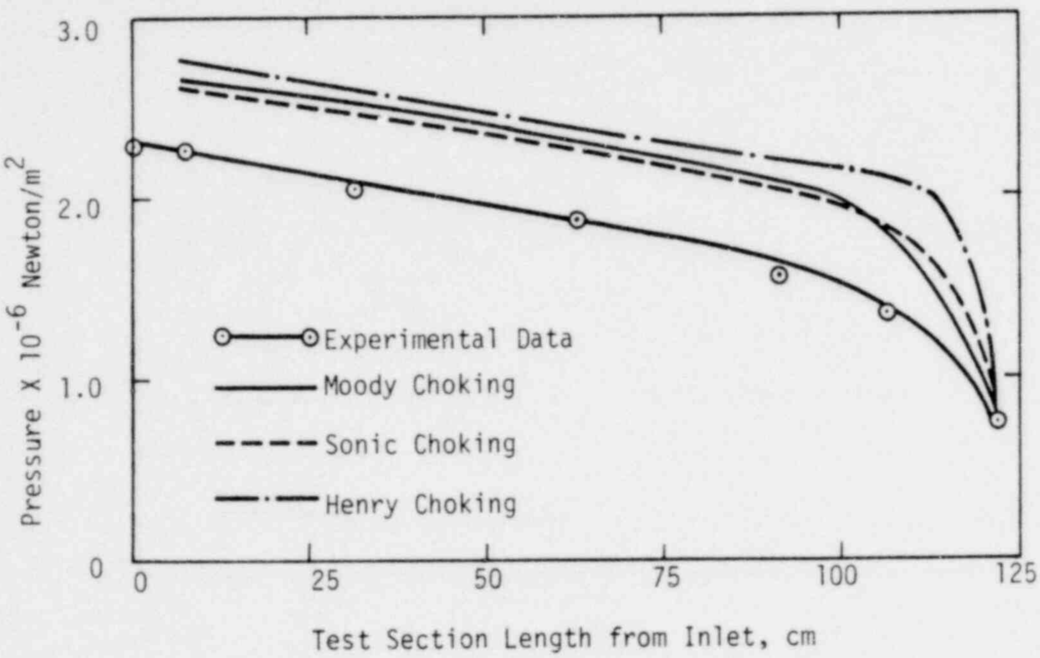

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FIGURE III.3-10 Comparisons of RETRAN Predicted Pressure Gradients with Experimental Data for Fauske Critial Flow Test TSII-9.

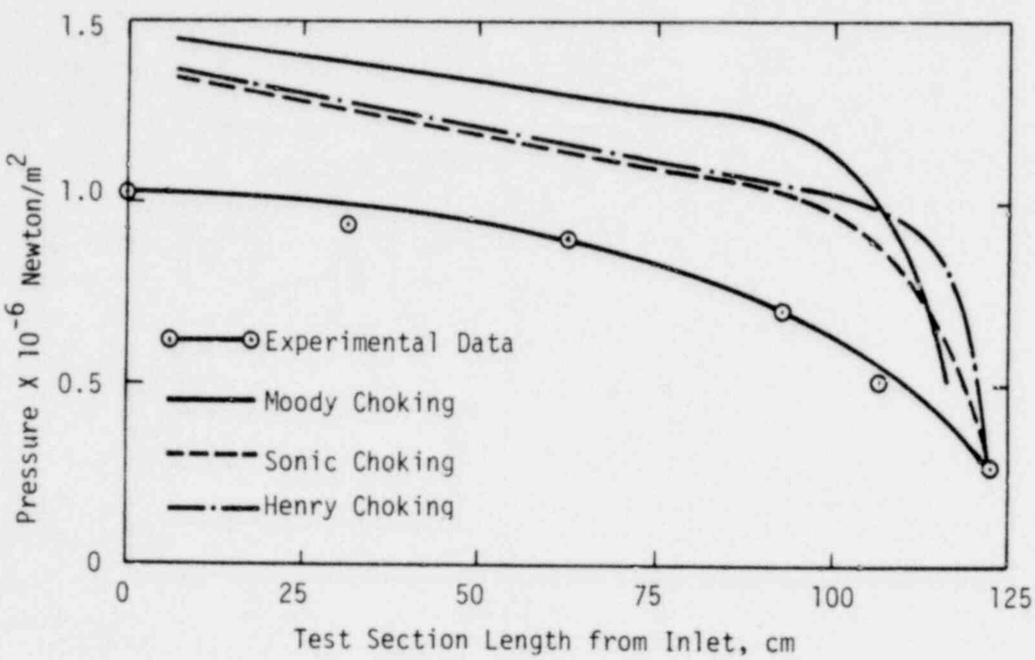

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FIGURE III.3-11 Comparisons of RETRAN Predicted Pressure Gradients with Experimental Data for Fauske Critical Flow Test TSII-76.

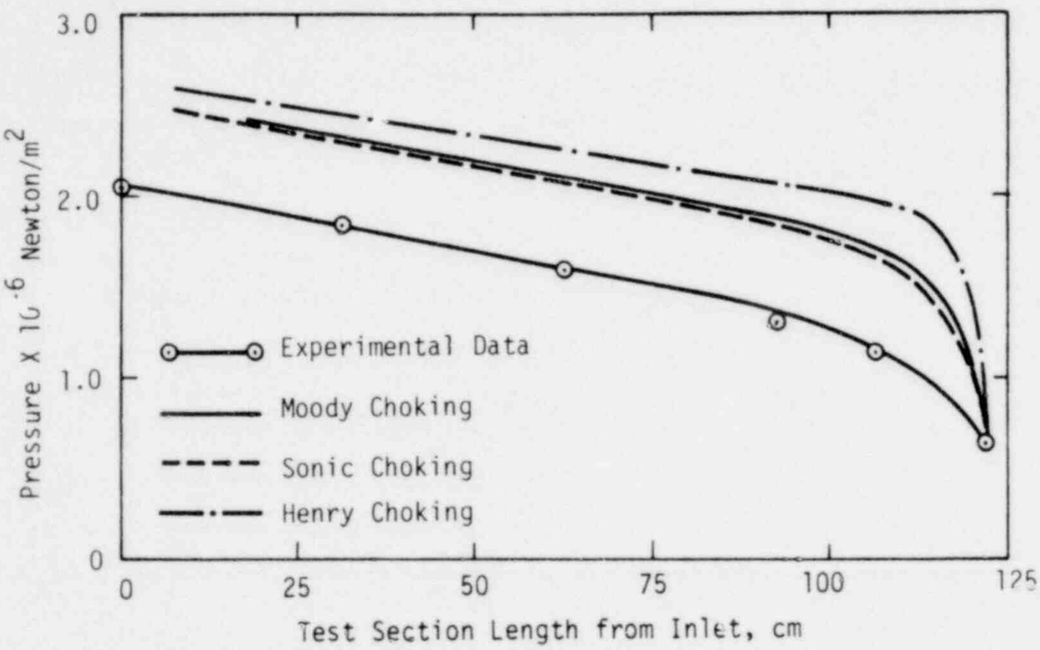

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FIGURE III.3-12 Comparisons of RETRAN Predicted Pressure Gradients with Experimental Data for Fauske Critical Flow Test TSII-36.

The results of a spatial nodalization study are presented in Figure III.3-13. This study shows that variations in nodalization can produce significant differences in the computed results. In this particular comparison, a large difference in pressure gradient exists near the throat area. The slope of the pressure gradient at the throat increased as the spatial nodalization was increased.

3.3 Summary of Results

The analyses by Kmetyk and Ginsberg[III.3-1] showed that the RETRAN critical flow options yielded correct values for saturated conditions and superheated conditions. However, the application of the Moody and Henry-Fauske models to subcooled water produced incorrect critical flow values. The user is cautioned against using either for these models for subcooled conditions. Rather, the extended Henry-Fauske option is recommended. This subject is also discussed in Section IX.1.7.

The modeling of the Fauske critical flow experiments indicated that the Baroczy two-phase multiplier was more appropriate than the other options, even though the experimental pressure was overpredicted. The sonic choking model gave the best overall agreement with the data, although the results obtained with the Moody model were generally good. These studies also demonstrated the need to apply correct boundary conditions and to use sufficient spatial detail in modeling a problem of this type.

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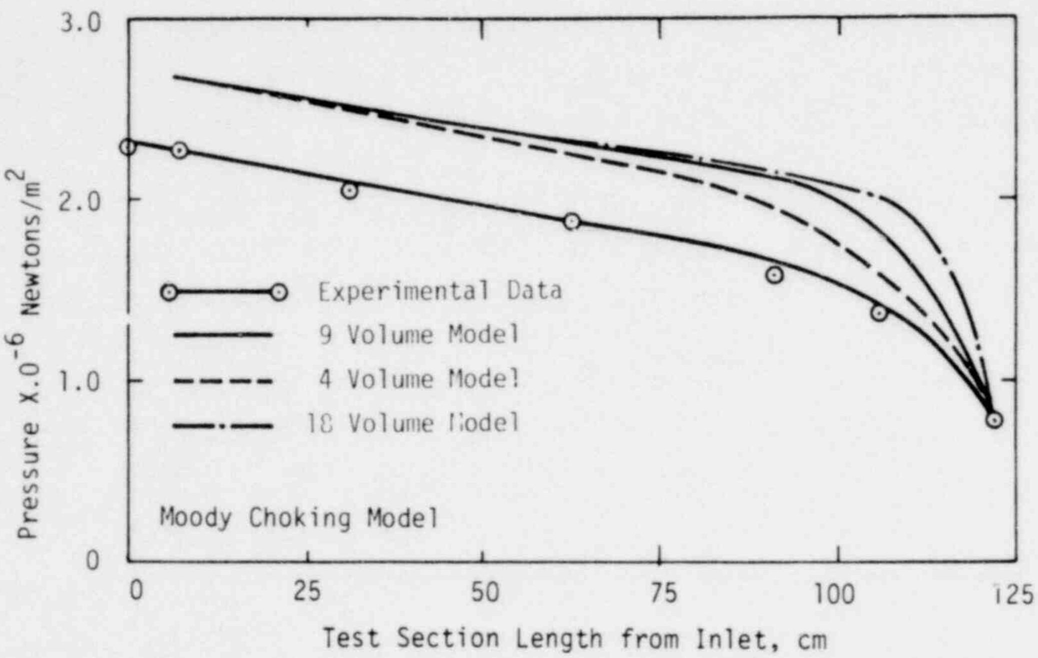

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FIGURE III.3-13 Comparison of Pressure Gradients for RETRAN Convergence Study of Fauske Critical Flow Test TSII-9.

4.0 MULTI-DIMENSIONAL FLOW

As discussed in Section II of this document and in Volume I of this report, the theoretical basis for the RETRAN code is one-dimensional flows. There are methods (e.g. neglecting momentum flux) of approximating these equations for multi-dimensional flows, and these are commonly used in regions such as the downcomer and plenum in reactor analyses.

Results of two studies of RETRAN applied to multi-dimensional flow situations are discussed in this section. Carlson[III.4-1] analyzed flow in a manifold and compared the results to hand calculations. Mirsky[III.4-2] evaluated a reactor downcomer and lower plenum for both a one-dimensional and a three-dimensional model.

4.1 Simple Multi-dimensional Regions

The model analyzed by Carlson[III.4-1] is shown in Figure III.4-1. Two vertical pipes are connected to a horizontal pipe which is closed at one end. A large, constant pressure volume is at the exit of each vertical pipe. The objective was to introduce a constant flow and determine if RETRAN would compute the correct flow split and pressure distribution. The principal area of investigation was the form of the momentum equation.

4.1.1 Description of RETRAN Model

The RETRAN model for this problem is shown in Figure III.4-1. The flow into the system was controlled by a fill, while two time-dependent volumes were used for the outlet boundary conditions. In addition to checking the Fanning friction losses, two forms of the momentum equation were evaluated for the tee junctions; (1) compressible flow with momentum flux (MVMIX = 0), and (2) incompressible flow without momentum flux (MVMIX = 3).

4.1.2 Results of Analysis

Carlson performed a number of analyses with RETRAN and compared these, in part, to engineering calculations. The latter computations were performed to estimate the flow split at the tees. Based on these calculations, Carlson estimated that

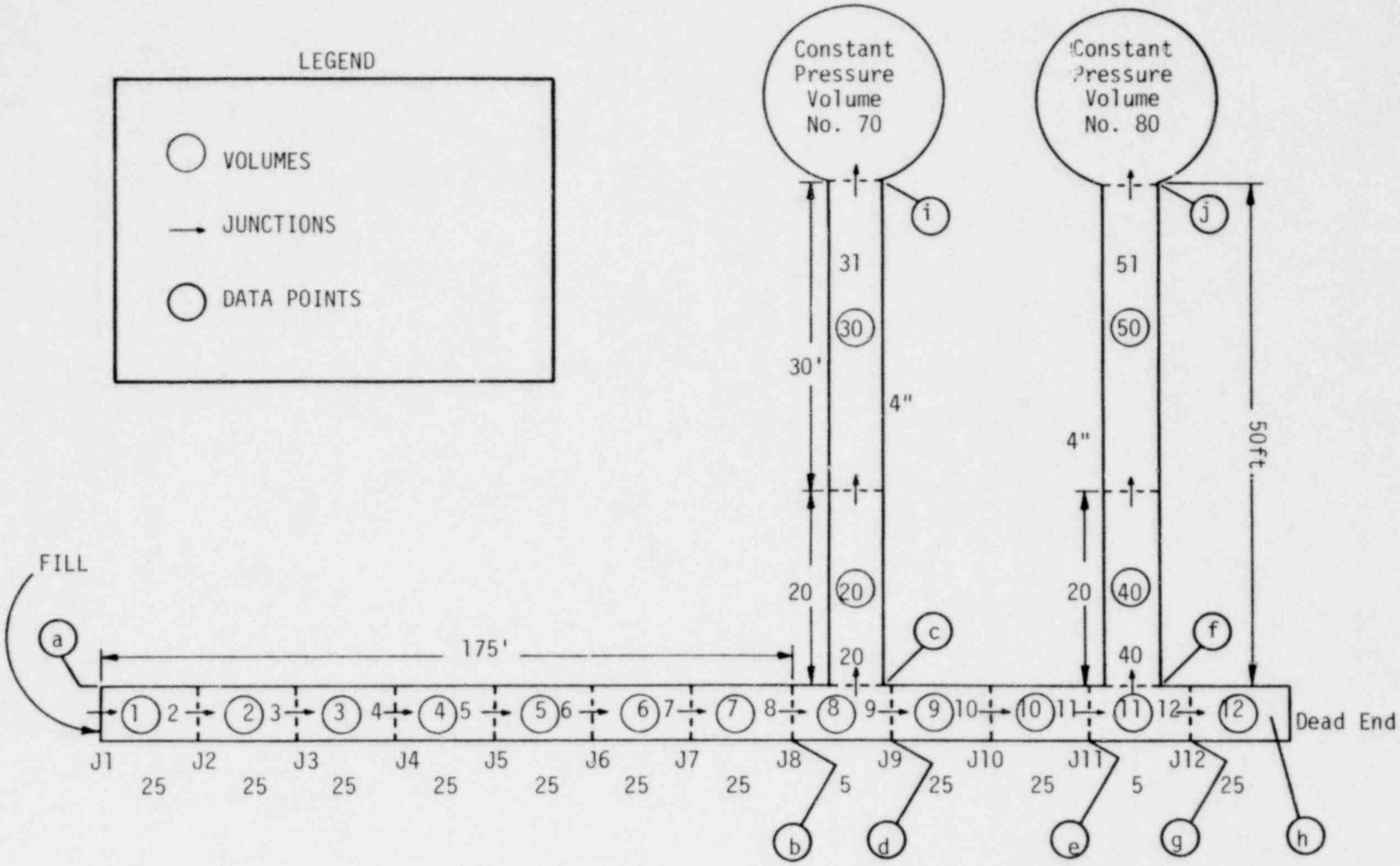

FIGURE III.4-1 Separate Effects Problem. Simple Multidimensional Region Six & Four Inch Water Mains.

1779 122

49.3 percent of the inlet flow went into the first tee with the remainder entering the second tee.

The RETRAN results which Carlson determined to be the best solution for this problem are shown in Figure III.4-2. For this case, the momentum equation at the inlet to the vertical pipes was solved by neglecting the momentum flux term (MVMIX = 3).

4.2 Multi-dimensional Flow in a Downcomer

The second analysis of multi-dimensional flow conducted with RETRAN was performed by Mirsky[III.4-2]. Mirsky modeled the downcomer-lower plenum region of the reactor vessel with both a one-dimensional and a three-dimensional representation. These models are shown in Figures III.4-3 and III.4-4. Flow into the system was controlled by fills at the downcomer entrance. The results were compared to vendor flow mixing results. Mirsky also evaluated the forms of the momentum equation required to achieve a steady-state solution and performed some time step size sensitivity studies. The results presented below are summarized from Reference III.4-2.

4.2.1 Description of RETRAN Model

The two models used in this study are shown in Figures III.4-3 and III.4-4. In the one-dimensional model, the downcomer and lower plenum are each represented by a single volume. For the three-dimensional model, the downcomer has three axial levels, with each level represented by three volumes. The lower plenum also is modeled with three volumes. It is noted that the model allows both horizontal and vertical flow to occur.

The transient analyzed was initiated by flow from the three fills which represent flow from the inlet nozzles. Two fills were modeled with equal pressure and enthalpy values, while the third had lower values. The input values were taken from FSAR calculations for a main steamline break. The outlet (reactor core) for both the one-dimensional and three-dimensional cases was modeled as a time-dependent volume.

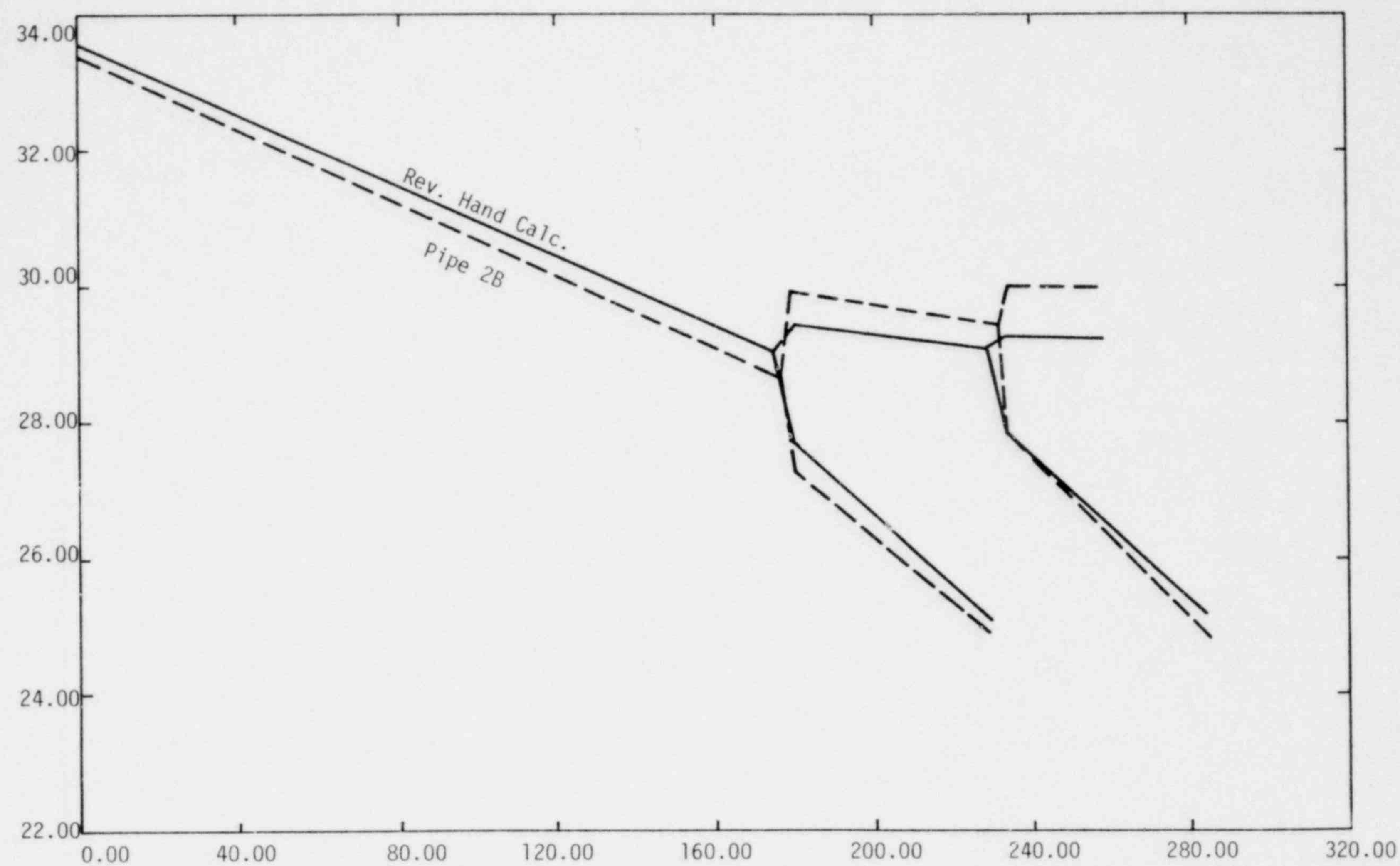

FIGURE III.4-2 Pressure Distribution Throughout System-Separate Effects Problem Four and Six Inch Main

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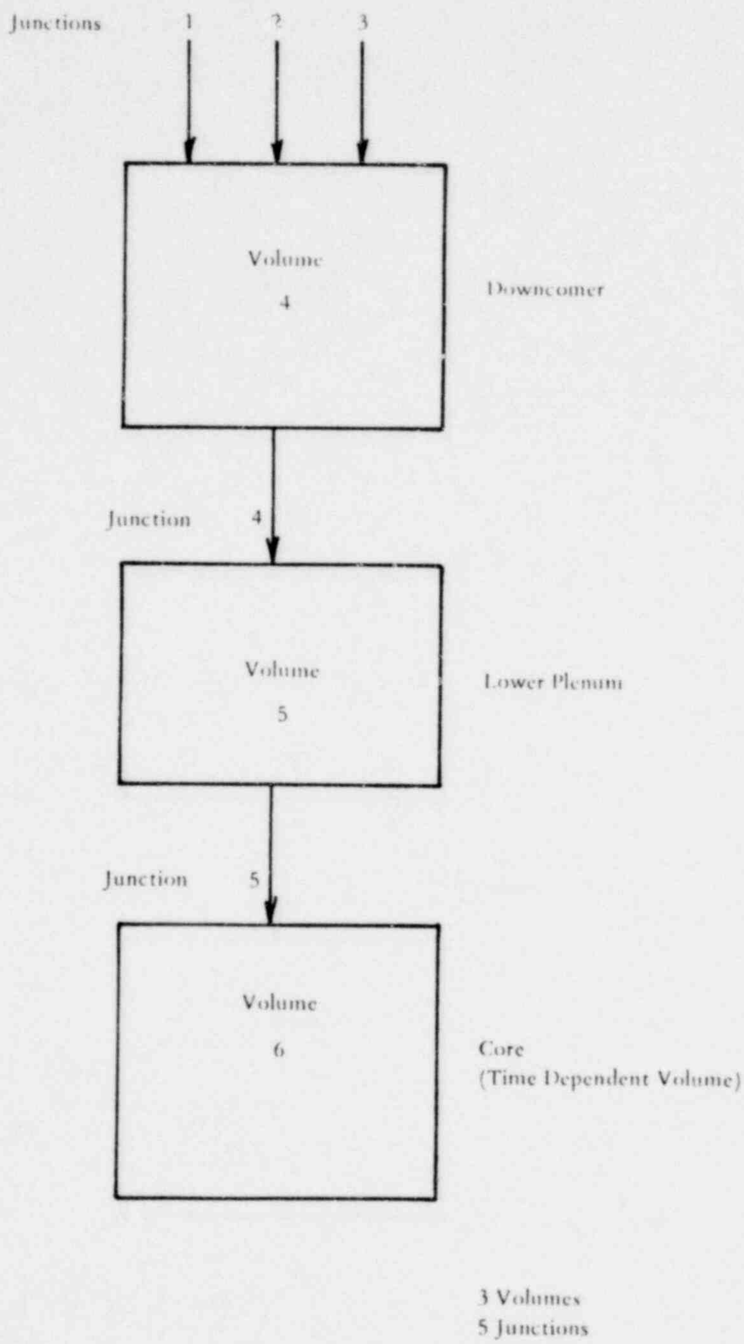

FIGURE III.4-3 One Dimensional Model Representation.

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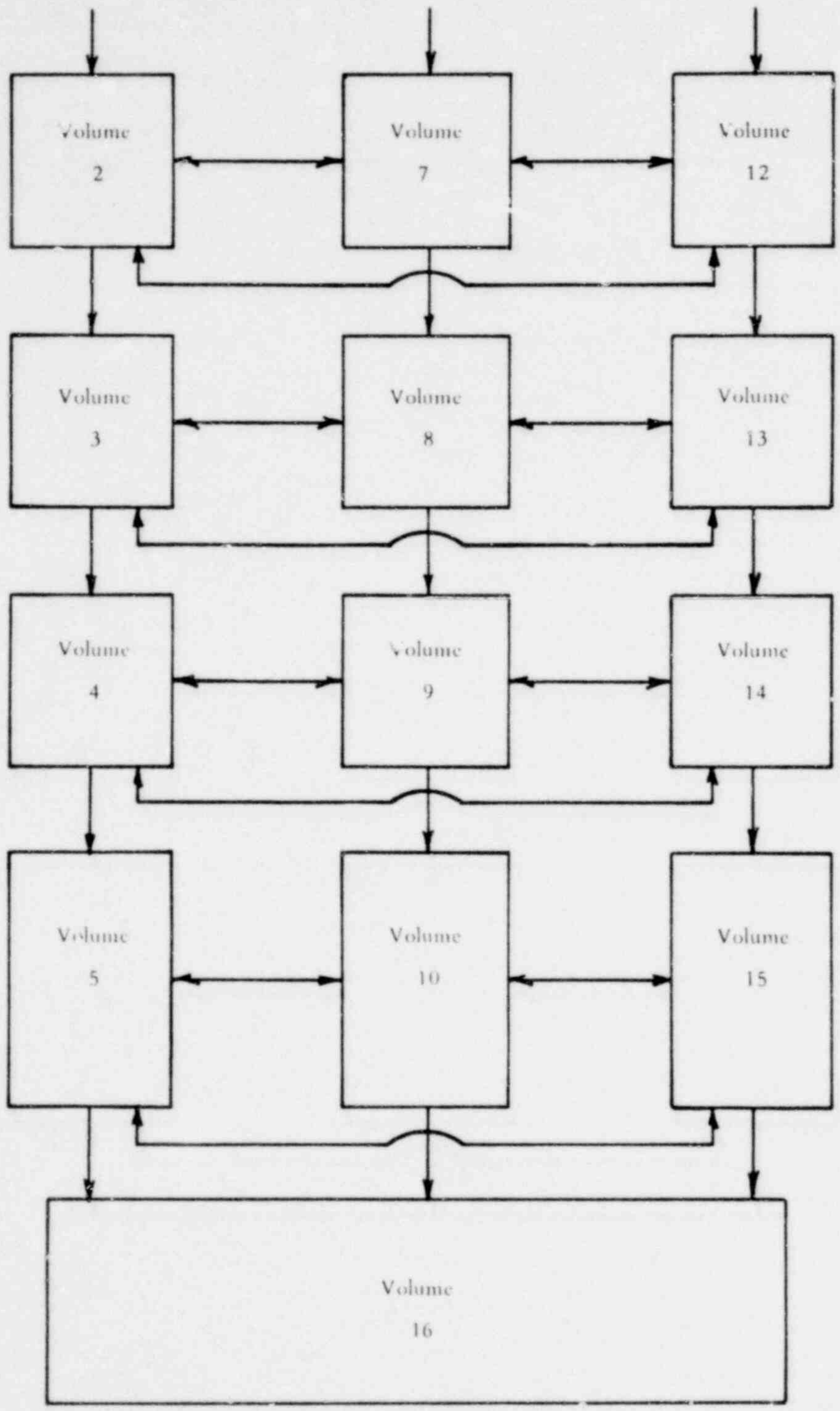

Downcomer

Lower
Plenum

Core
(Time Dependent Volume)

FIGURE III.4-4 Three Dimensional Model Representation.

13 Volumes
27 Junctions

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4.2.2 Results of Analyses

One aspect of this study was related to modeling friction and momentum flux terms in the momentum equation. With equal, constant values of pressure and enthalpy for all three fills, sensitivity studies were performed. The objective was to determine the representation which would result in a steady-state solution for the three-dimensional model. Mirsky found that a steady-state solution could be achieved by neglecting friction and momentum flux for the crossflow junctions. These effects were included for the junctions with vertical flow. This was the basic model used for the three-dimensional model.

A comparison of the lower plenum temperatures for the one- and three-dimensional models is shown in Figure III.4-5. The effect of spatial mixing for the three-dimensional model is shown in Figure III.4-6, with the fluid temperature in the cold lower plenum volume approximately 15°F warmer than the temperature for the corresponding top downcomer volume. The opposite effect is observed for the volumes with the highest temperatures.

Mirsky evaluated these results with those presented in Reference III.4-3. Tests conducted by a vendor indicated that 65 percent to 75 percent of the flow emerging from an inlet nozzle will flow up the same radial section of the core. Mirsky estimated that RETRAN calculated, for the three-dimensional model, 75 percent of the fluid introduced from the cold nozzle reached the cold lower plenum volume. Additional RETRAN analyses, along with comparisons to multi-dimensional code calculations and/or experimental data are desirable for further evaluation of this modeling approach.

Additional studies performed during this investigation were directed to evaluating the effects of enthalpy transport, the outlet boundary condition value for pressure, and time step size. No signficant changes in the results occurred for these cases.

4.3 Summary of Results

The analyses discussed in the preceding sections are important because they are applications of RETRAN outside of the basic theoretical limitations of a one-dimensional system. Carlson found that the best representation of a momentum

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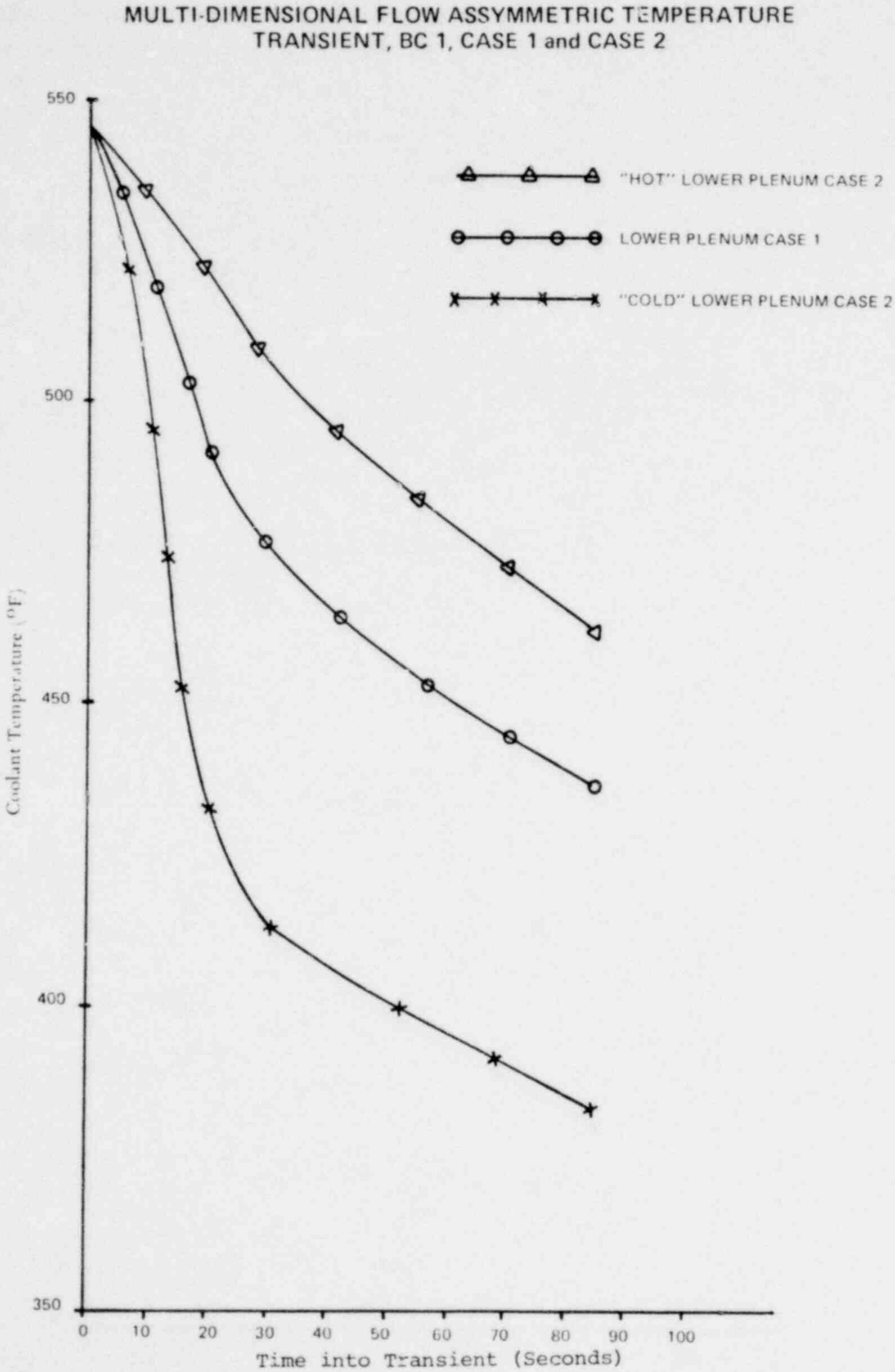

FIGURE III.4-5 Multi-Dimensional Flow Assymmetric Temperature Transient, BC 1, Case 1 and Case 2.

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MULTI-DIMENSIONAL FLOW - DOWNCOMER TO LOWER PLENUM
BC1, CASE 2

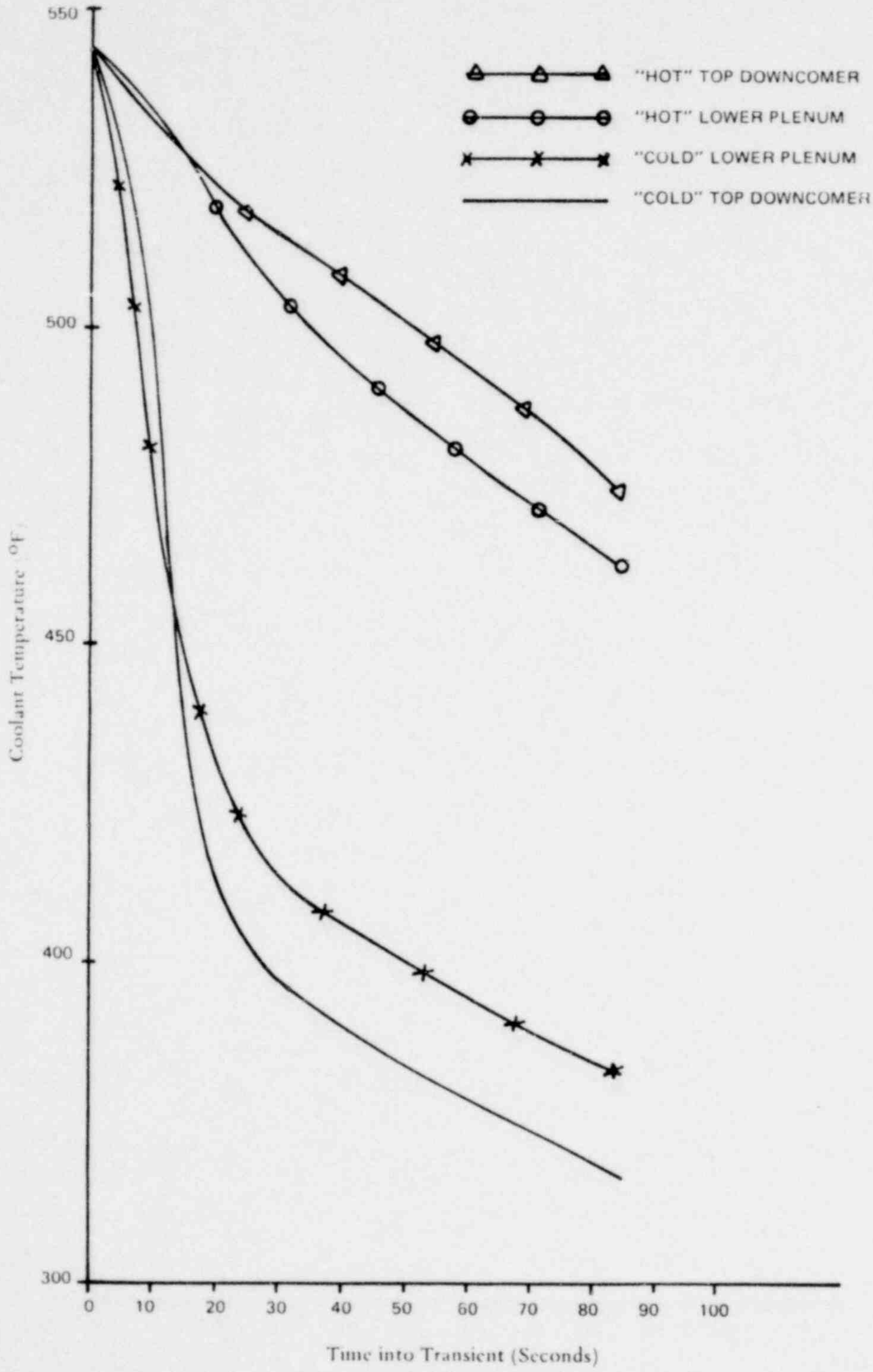

FIGURE III.4-6 Multi-Dimensional Flow-Downcomer to Lower Plenum BC1, Case 2.

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equation at tees is to neglect the momentum flux terms. This assumption is frequently used with RETRAN type codes. The studies by Mirsky indicate that, under some conditions, RETRAN can be used to represent three-dimensional flow such as occurs in a downcomer and lower plenum. It is important to note that such problems must be approached with caution. The use of a null transient, as was done by Mirsky, is necessary to verify that the model is reasonable representation of the problem. Additional analyses, with comparisons to multi- dimensional calculations and/or experimental data, are desired for further evaluation of this modeling method.

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5.0 EDWARD'S PIPE (STANDARD PROBLEM ONE)

The Nuclear Regulatory Commission has developed a program to evaluate codes used for thermal hydraulic analyses. This program is titled "Comparative Analysis of Standard Problems" (CASP), and the first problem in the series of analyses was based on experiments conducted by Edwards and O'Brien[III.5-1].

Standard Problem One was a blowdown of a straight, pressurized, horizontal pipe, 13.44 feet long. The pipe was filled with water and brought to an initial pressure of 1000 psig for this particular experiment. A glass disc at one end of the pipe was broken to start the transient. Experimental data included fluid temperature and pressure (7 locations) and void fraction (2 locations). The locations of these data measurements are shown in Figure III.5-1.

Standard Problem One has been analyzed by many investigators as it is the first problem of the CASP series. One such analysis was performed by Energy Incorporated with a RELAP4 code[III.5-2]. In addition to modeling the experiment, sensitivity studies were conducted to investigate the effects of space noding, time step sizes, integration solution techniques, and momentum equation forms.

Two RETRAN studies of Standard Problem One are discussed in the following sections. These studies represent various parts of the work reported in Reference III.5-2 for a RELAP4 code. McLaren[III.5-3] used the basic model from Reference III.5-2 and performed sensitivity studies for time step size, nodalization and momentum equation terms (momentum flux, friction, inertia and critical flow models). Smith[III.5-4] also used the base case model from Reference III.5-2. Sensitivity studies were performed by varying the critical flow options, the loss coefficient at the break, and the initial enthalpy distribution in the pipe. The RETRAN results from both of these studies agree quite well with both the RELAP4 calculations and the experimental data. Only a limited number of results from the many runs are presented here. Additional comparisons can be found in References III.5-2, III.5-3, and III.5-4.

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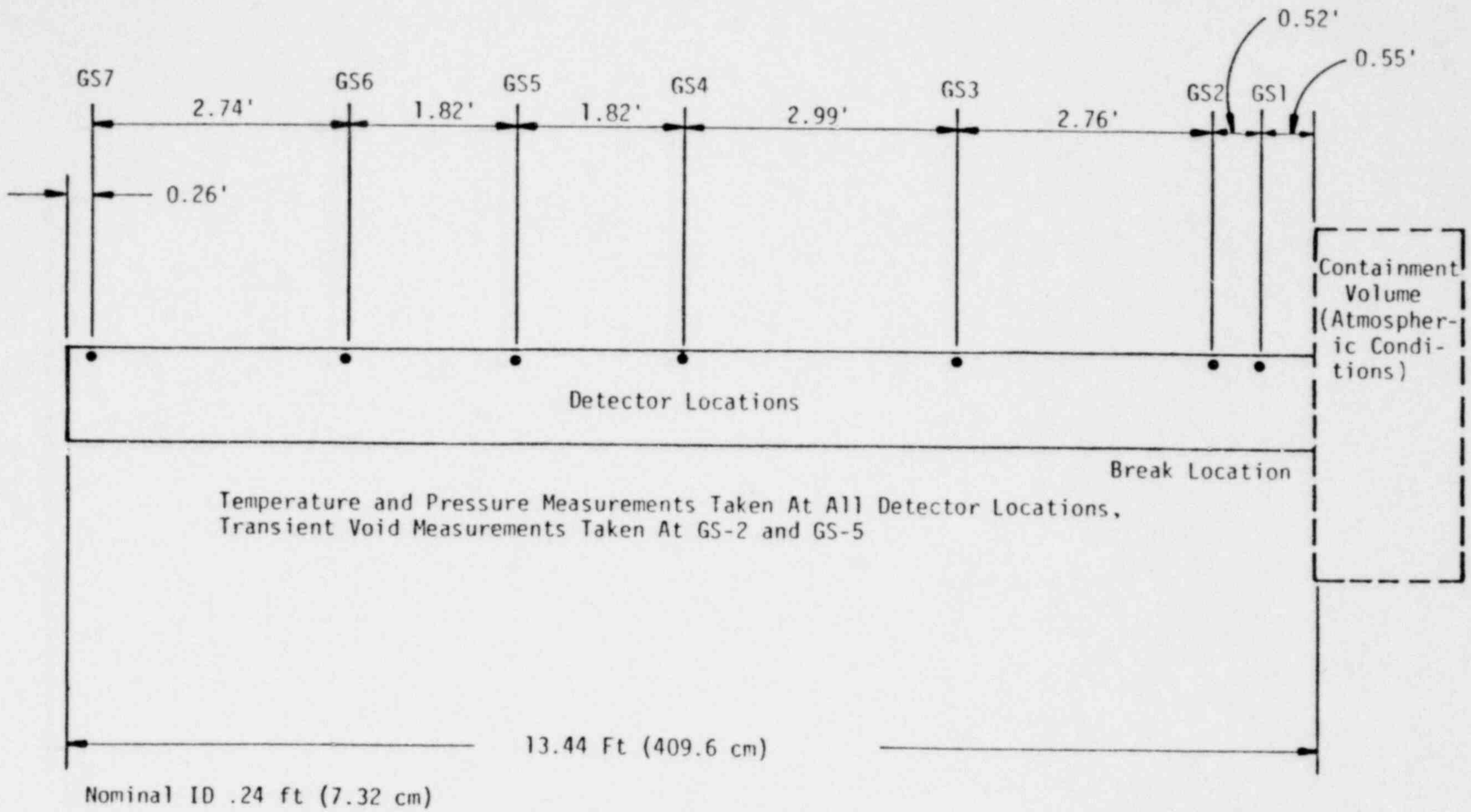

FIGURE III.5-1 Standard Problem 1 - Edwards Pipe Experiment

5.1 Pacific Gas and Electric Analyses

5.1.1 Description of RETRAN Model

The basic RETRAN model used in this study included 11 control volumes (Figure III.5-2). This particular model was chosen so that all detector locations would correspond to the centers of control volumes, and was the only difference between this model and the 9 volume model of References III.5-2 and III.5-4. In addition to this 11 volume (unequal length) model, a 24 volume model and a 48 volume representation of the pipe were studied. Choking options investigated included the sonic critical flow solution as well as the Moody and Henry-Fauske models.

5.1.2 Results of Analyses

The computed and experimental pressure values for three locations are shown in Figures III.5-3 to III.5-5. For the models chosen, the effect of noding is not significant and the 11 volume model gives a converged solution. Long term (to 600 msec) solutions showed similar agreement. Other studies showed that the effect of noding on flow rate was not significant. Although the values varied at some points, the overall values and trends were not affected by node size. The Henry and Moody critical flow models were found to give better agreement with data than does the sonic model. The effect of time step size is illustrated in Figure III.5-6. McLaren noted that the use of the variable time step size option would be quite helpful if the user did not have a good prior estimate of time step size. Although this option may result in a slightly longer running time, the number of analyses required to demonstrate a converged solution is reduced. As shown in Figure III.5-6, the analysis with the variable time step size option yielded a converged solution in one run, whereas 2 runs were required when a time step size was specified.

5.2 VEPCO Analyses

5.2.1 Description of RETRAN Model

Smith[III.5-4] used the 9 volume model shown in Figure III.5-7 for his investigation. Sensitivity studies were conducted for the critical flow options, the initial enthalpy distribution, and the loss coefficient at the break junction. The complete momentum equation was used for all the analyses.

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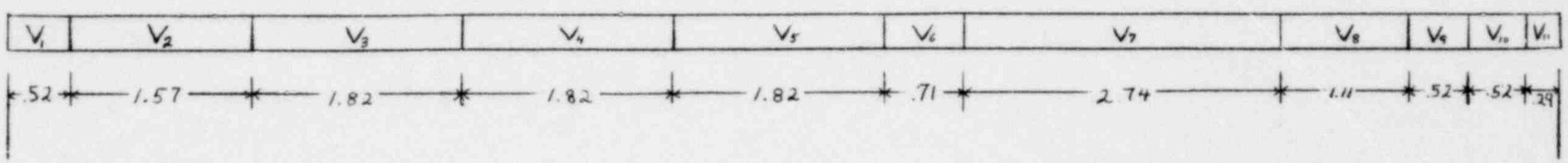

FIGURE III.5-2   Volume Divisions for Case 1.

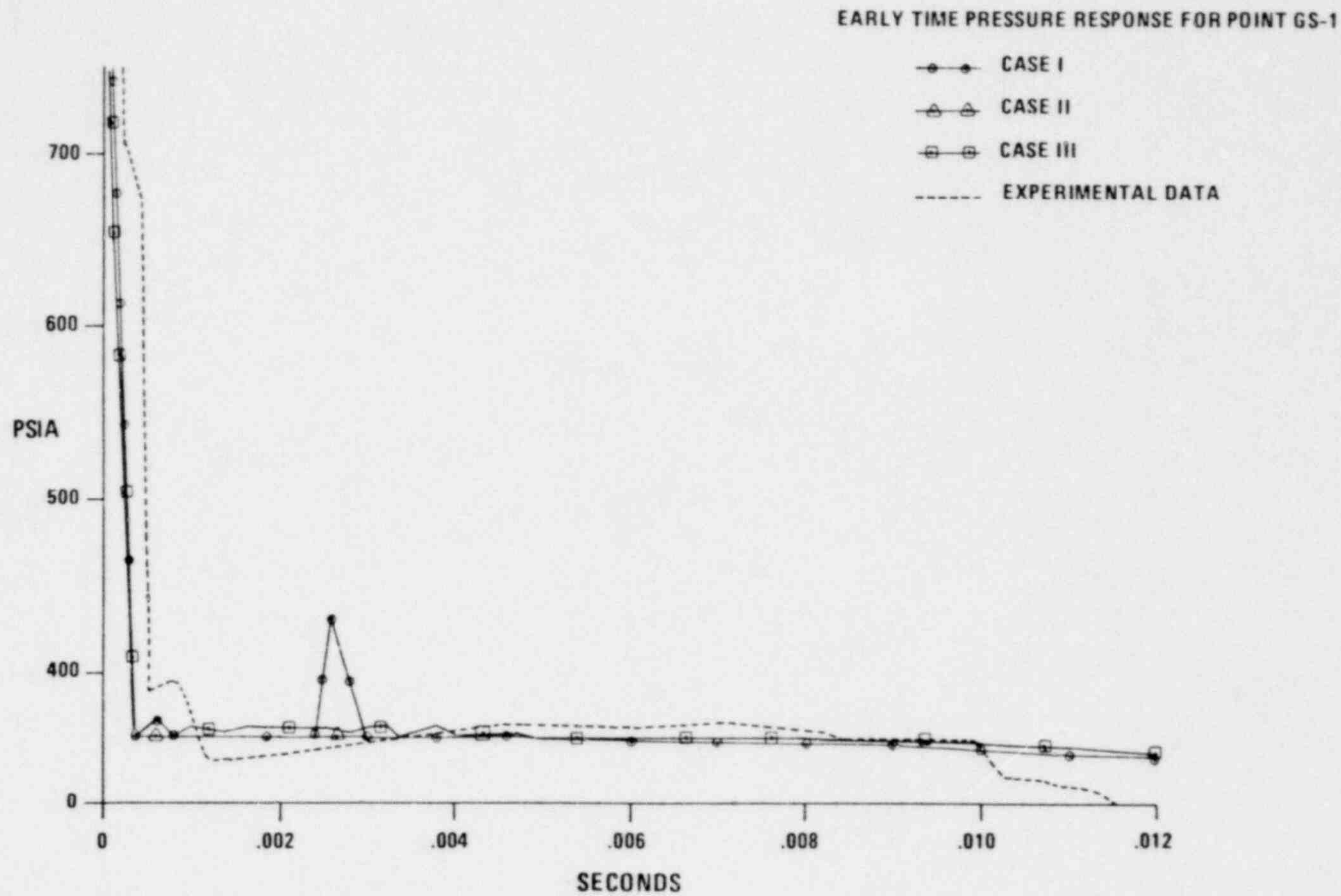

FIGURE III.5-3 Early Time Pressure Response for Point GS-1.

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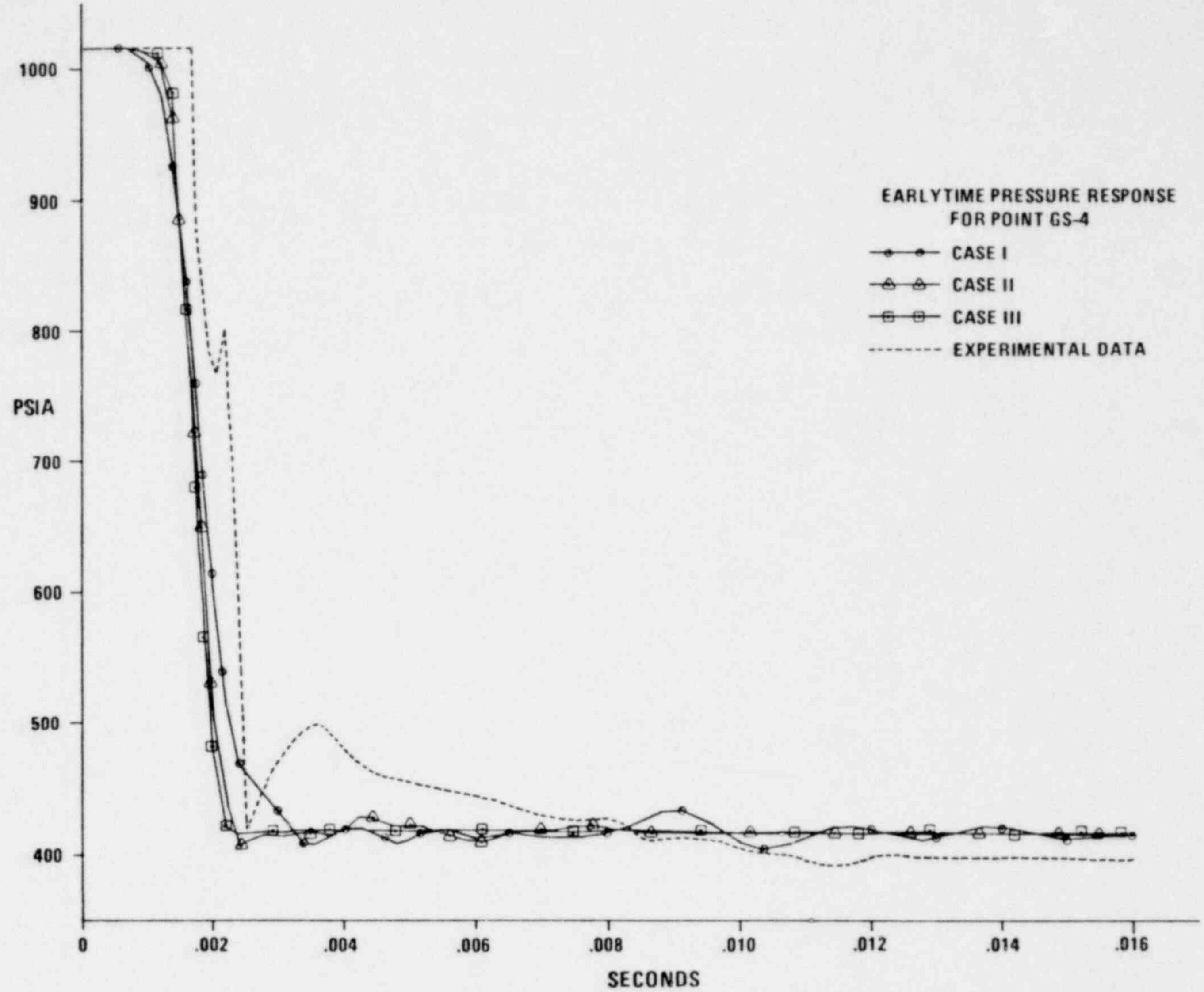

FIGURE III.5-4 Early Time Pressure Response for Point GS-4.

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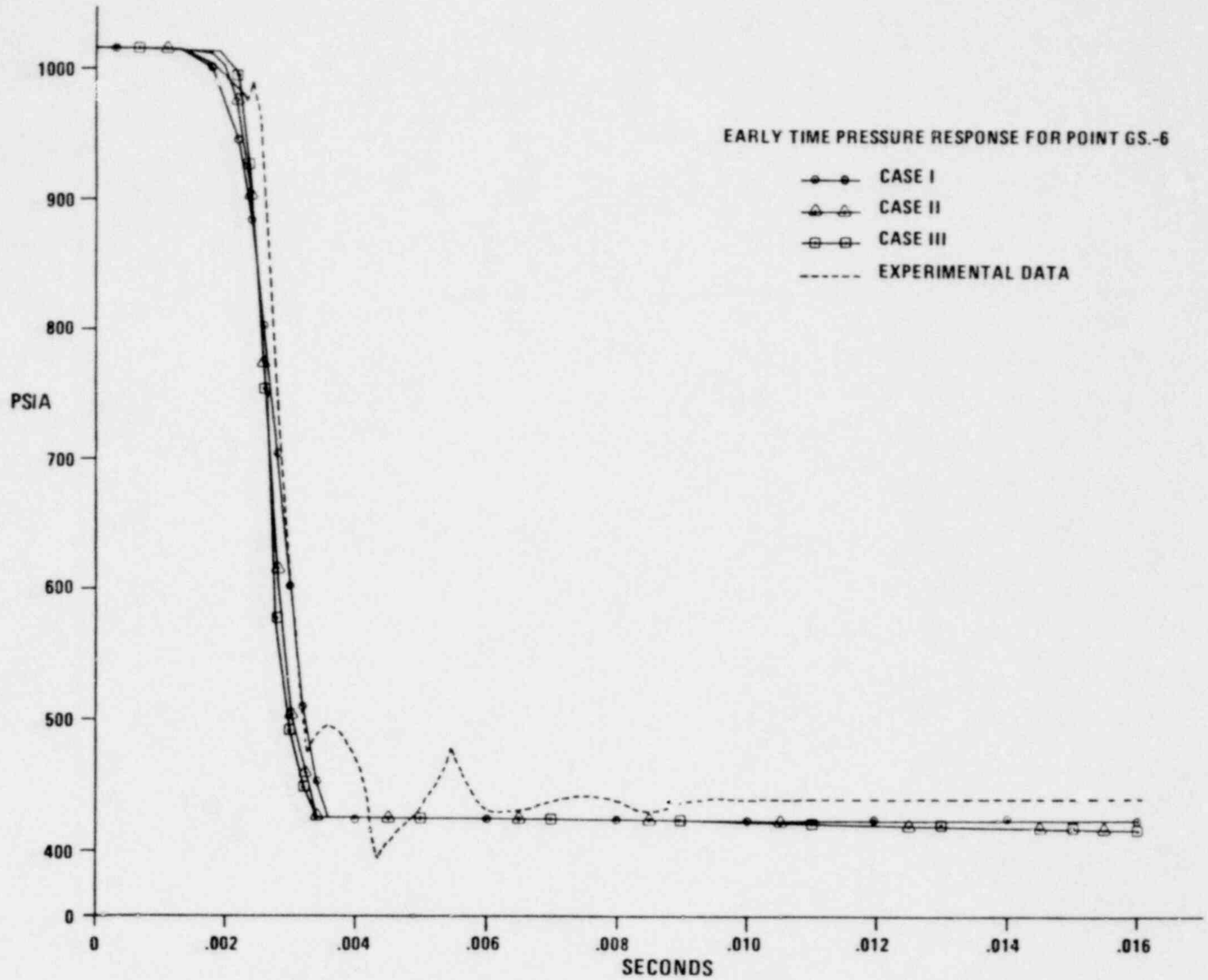

FIGURE III.5-5 Early Time Pressure Response for Point GS-6.

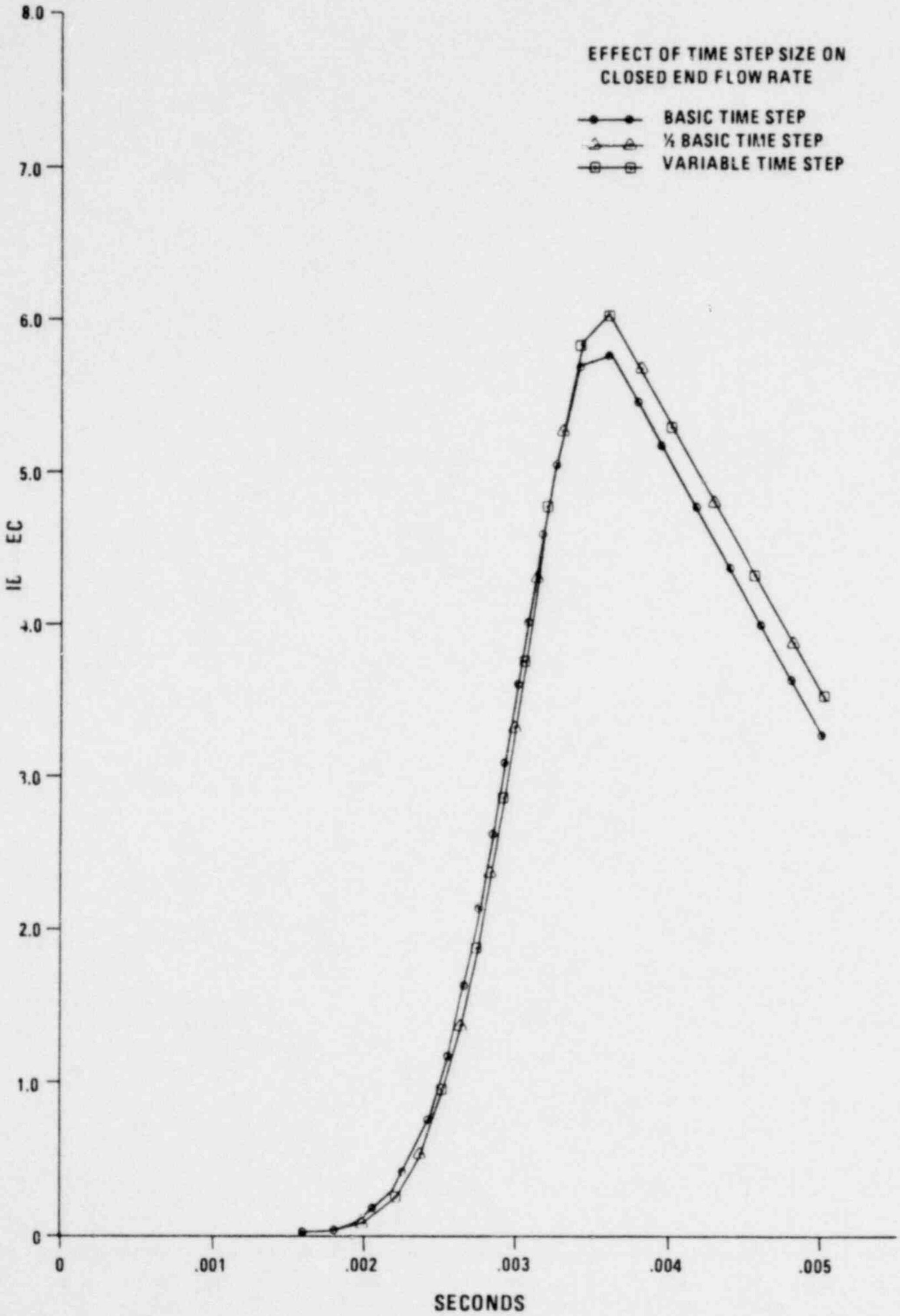

FIGURE III.5-6 Effect of Time Step Size on Closed End Flow Rate.

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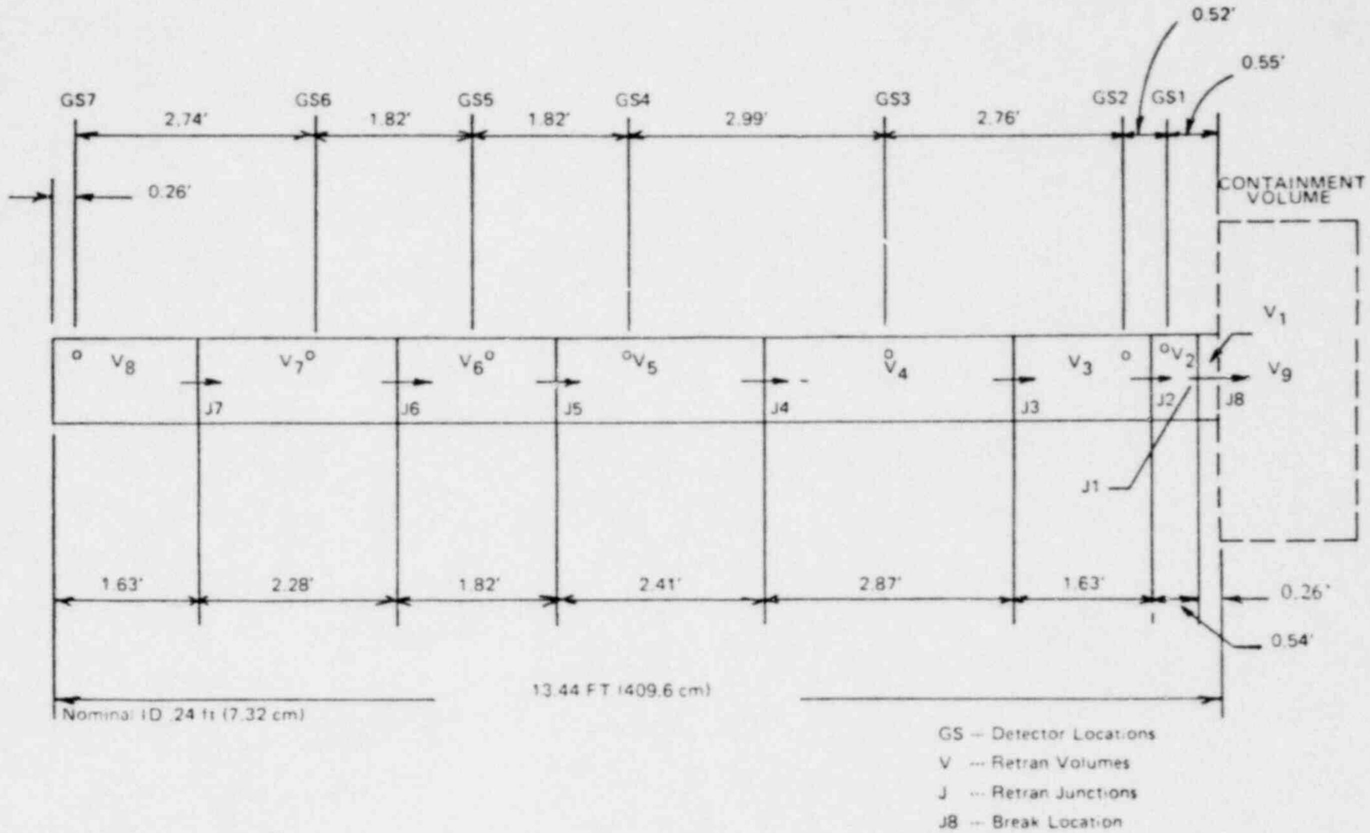

FIGURE III.5-7 RETRAN Model, Standard Problem 1 - Edward's Pipe Exeperiment.

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5.2.2 Results of Analyses

The RETRAN and RELAP4 values of pressure at the detector closest to the break are compared with experimental data in Figure III.5-8. Void fraction and temperature values are shown with experimental data in Figures III.5-9 and III.5-10. In general, the RETRAN results agree favorably with the RELAP4 calculations, and the calculated values of pressure and void fraction agree well with the data. However, the temperature values are below the data. This could be due to non-equilibrium conditions after 200 msec (the void fraction is greater than 0.7 for this time increment) or errors in the measurement.

5.3 Summary of Results

The results of these studies demonstrate that RETRAN accurately computes the depressurization of the pipe for this experiment. Increasing the number of control volumes showed that spatial convergence can be achieved with 11 volumes. The variable time step size option was shown by McLaren[III.5-3] to be working correctly, and its value as an analysis aid was demonstrated.

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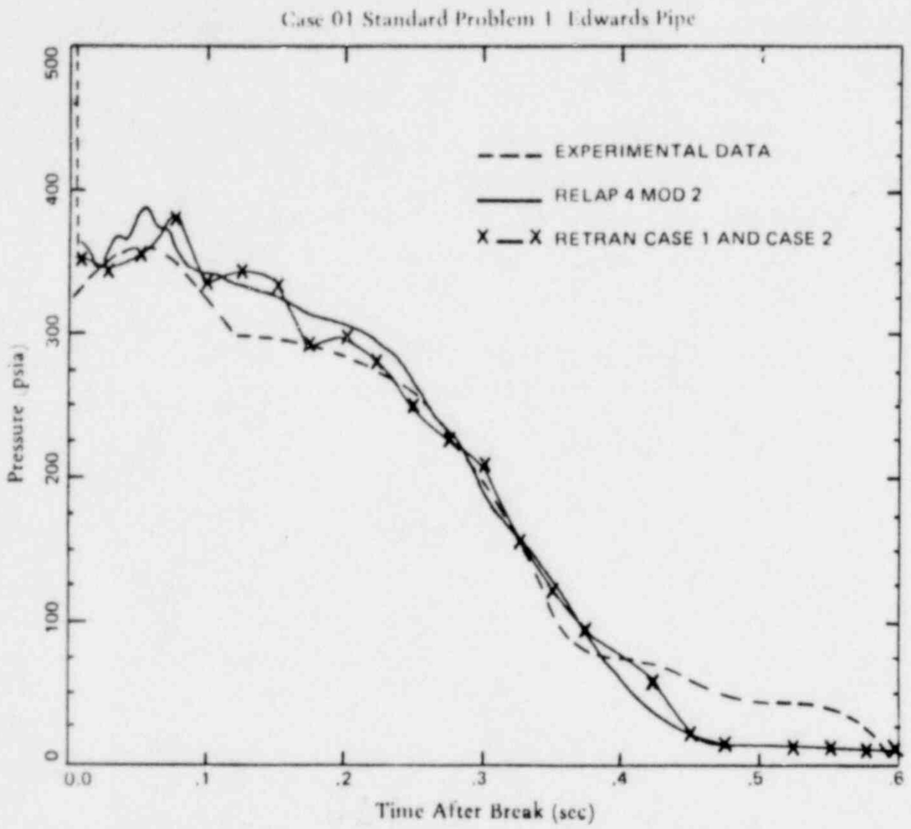

FIGURE III.5-8 Pressure Comparison for Location GS-1.

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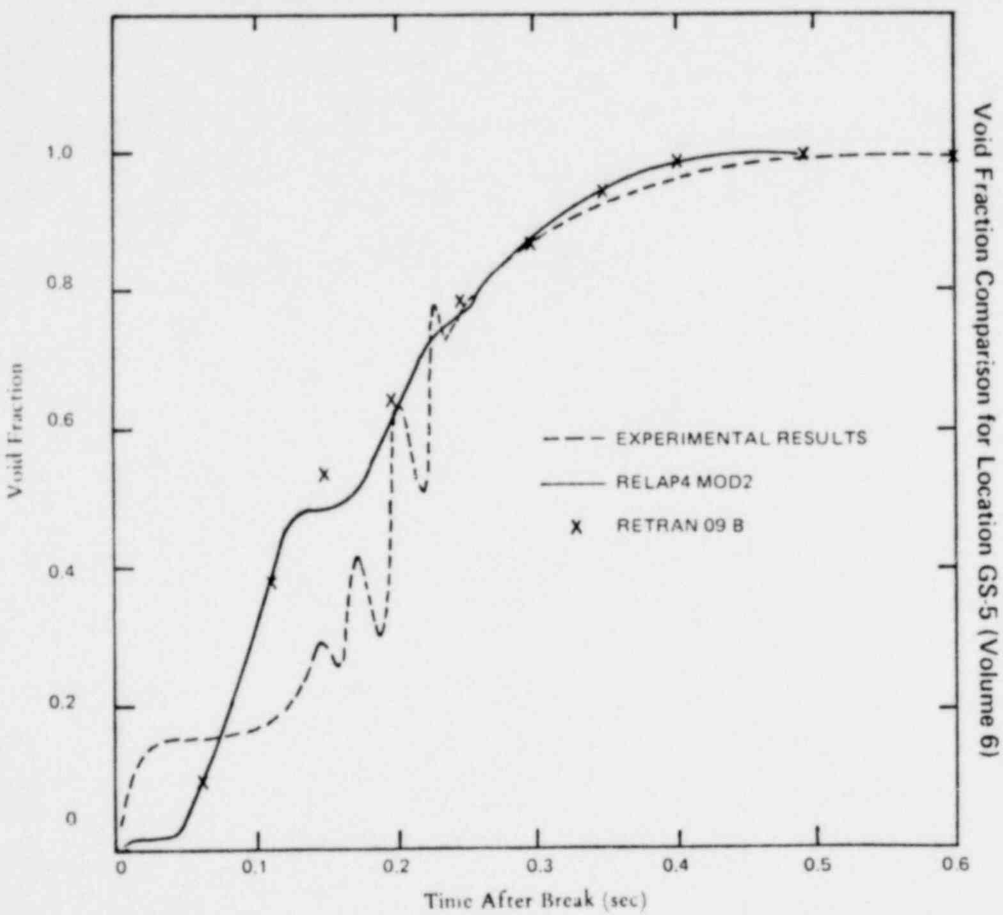

FIGURE III.5-9 Void Fraction Comparison for Location GS-5.

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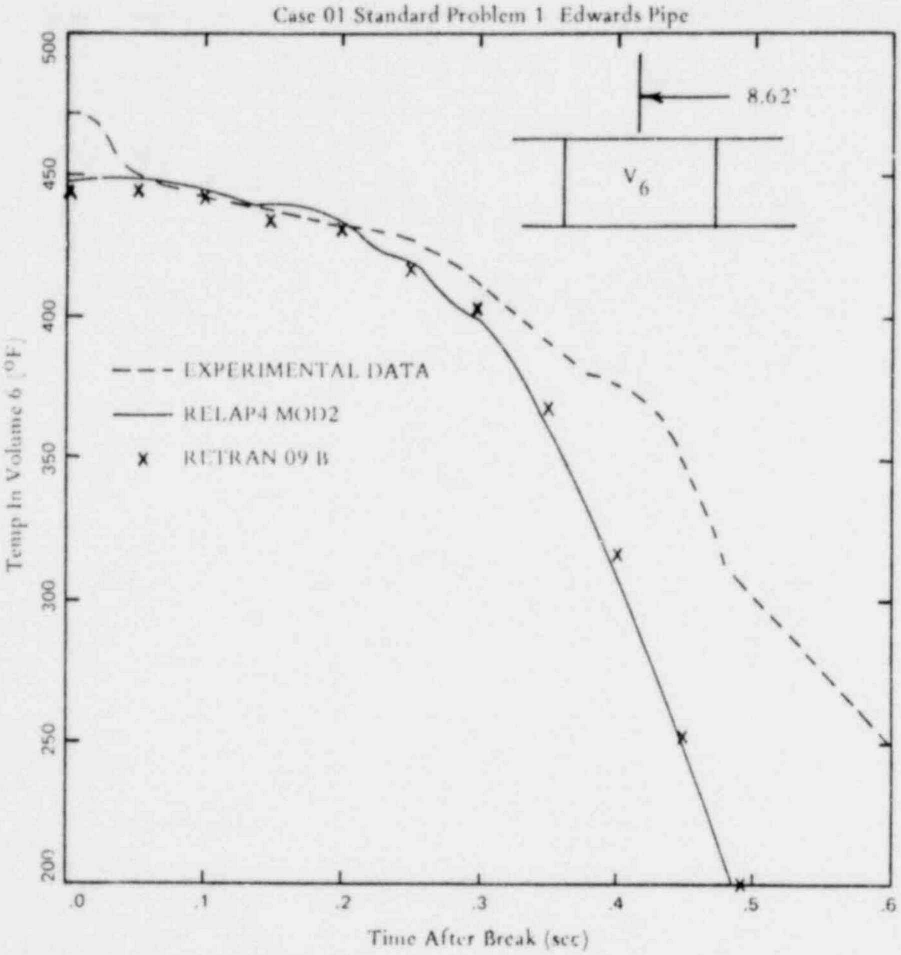

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FIGURE III.5-10 Temperature Comparison for Location GS-5.

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IV. SYSTEM EFFECTS
ANALYSES

IV. SYSTEM EFFECTS ANALYSES

In modeling two-phase flow problems, the first level of complexity is encountered with separate effect problems as discussed in Section III. After individual models have been evaluated with simple experiments, the next step is to apply the code to systems which are more complex than separate effect tests, but less complex than power reactors. These analyses are termed "system effects analyses," and are designed to evaluate the ability of the code to calculate the overall response of a transient for a small scale system.

Two points are particularly relevant to the analyses discussed in this section. First, we are interested in modeling simple systems, those for which we have an intuitive feeling for the nature of the analysis. In general, these applications should either

(1) be consistent with the theoretical limitations of the code or

(2) be limited in the deviation from the theory.

The second point to note is that we are interested in evaluating the code for overall response. While the need for accuracy is desired, the agreement sought is not as rigorous for all variables as is the case for separate effect analyses.

The analyses discussed in this section include a wide range of applications, ranging from rather simple problems such as an accumulator blowdown to small scale systems such as Semiscale and TLTA.

1.0 TROJAN ANALYSES

Four applications of RETRAN to components of the Trojan reactor have been performed. During preoperational testing, an accumulator discharge test was performed. RETRAN and RELAP4 analyses of this test are compared to data in this section. The overpressure mitigating system was modeled with RETRAN and the results were compared with vendor calculations. The response of the steam generator secondary following a loss of feedwater was analyzed with RETRAN and the results also were compared to a vendor analysis. The fourth RETRAN calculation was a subcompartment analysis of the reactor cavity.

1.1 Analysis of Accumulator Discharge

An essential part of the PWR Emergency Core Cooling System is the passive injection of borated water provided by the accumulator tanks (Figure IV.1-1). These tanks, pressurized with nitrogen gas, automatically discharge when depressurization of the RCS causes a reversal of the pressure drop across the accumulator check valves. In this way, the accumulators help provide rapid cooling of the reactor core following large RCS piping breaks. The objective of this analysis was to model the accumulator discharge following simulated RCS depressurization using the RETRAN code, and validate these results with data taken during preoperational testing.

Prior to the preoperational test, accumulator No. 3 was filled to the 100-percent level and pressurized to 99.2 psig. The system temperature was estimated to be 80°F, the condition that prevailed in the containment during the earlier Containment Integrated Leak Rate Test. The reactor vessel was empty, the upper and lower internals removed, and the head of the vessel removed. Precautions were taken to ensure that the discharged water did not interfere with the test or overflow the reactor vessel, thus creating an increase in the backpressure. Accumulator discharge was initiated by opening the motor-operated isolation valve in the 10-in. safety injection line. Pressure readings were taken using a temporary gauge attached to the normal accumulator pressure measuring instrumentation taps. The test was terminated by closing the isolation valve when the indicated accumulator pressure reached 10 psig. Two analyses were performed for this test, one by Portland General Electric[IV.1-1] and the second by Pacific Gas and Electric Company[IV.1-2].

1.1.1 Portland General Electric Analyses

In addition to modeling the test with a base model, sensitivity of the calculation with respect to form loss coefficients and time step size was investigated. The results were compared with RELAP4 calculations as well as the data.

1.1.1.1 Description of Model

A description of the RETRAN and RELAP4 models for this analysis is shown in Figure IV.1-2. It has 7 volumes and 6 junctions. The effective opening time of

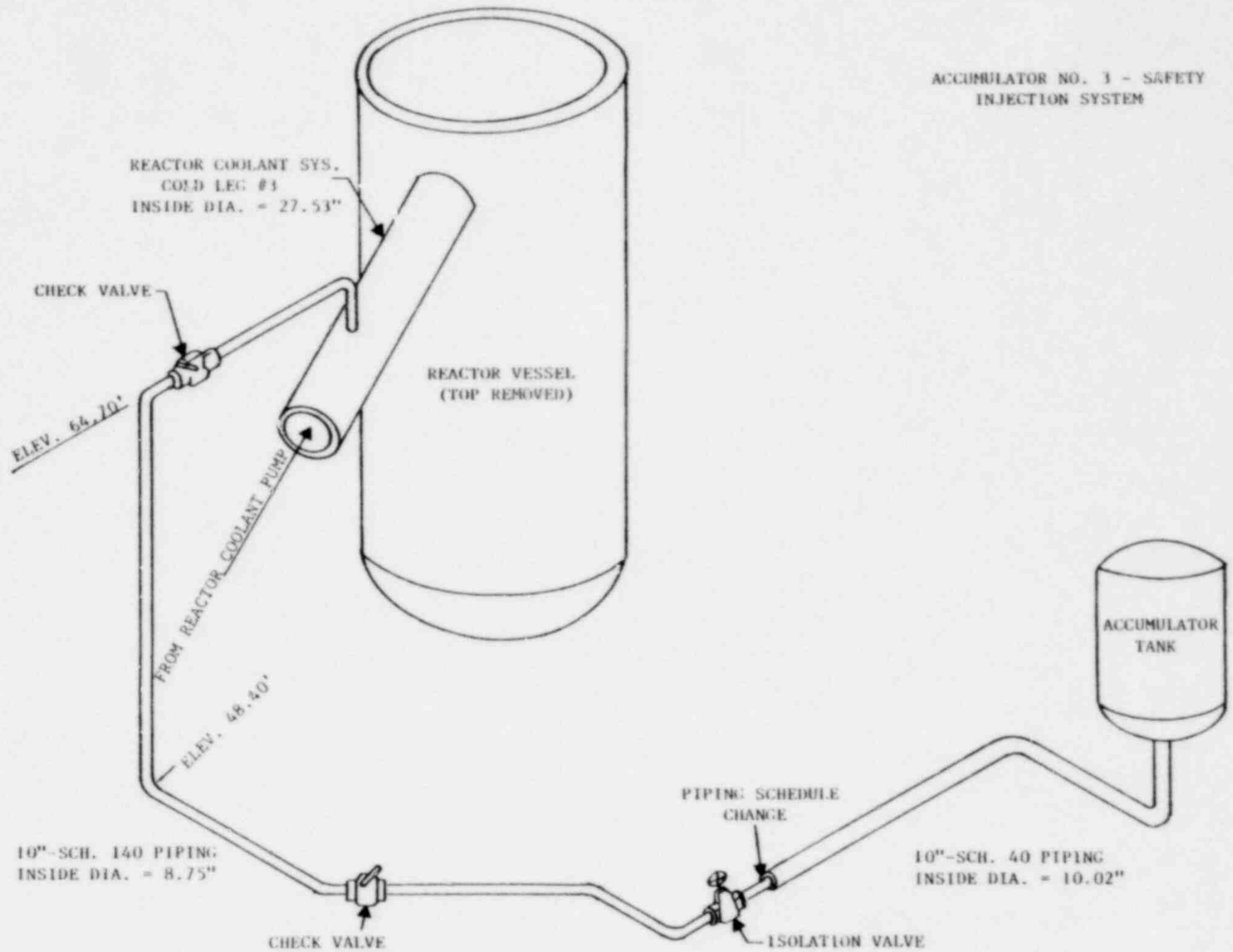

FIGURE IV.1-1 Accumulator No. 3 - Safety Injection System.

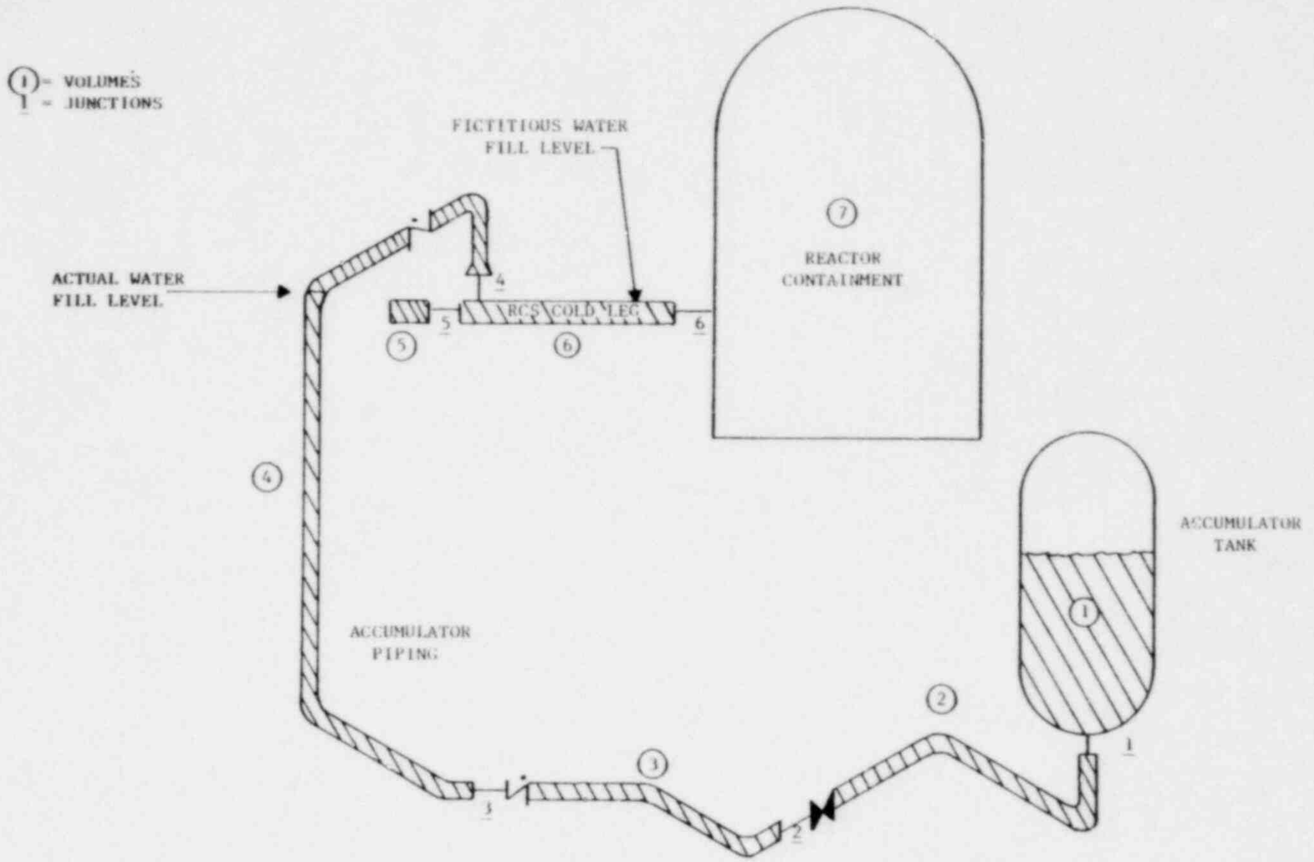

FIGURE IV.1-2 Accumulator Blowdown Model.

the isolation valve was determined from test results, and the time-dependent valve flow area was calculated based on this opening time and the valve flow characteristics. The handbook values for form loss were reduced by 20 percent. This value was determined to be representative of the conservatism associated with design values.

In evaluating the code sensitivity to time step variation, both RETRAN and RELAP4 calculations were performed with: a) automatic time step control (minimum and maximum time steps at .00001 and .01 sec, respectively); b) a manual maximum time step of 0.0005 sec; and c) a manual maximum time step of 0.005 sec. All other parameters remained the same.

1.1.1.2 Results of Analysis and Comparison With Data

The RETRAN and RELAP4 predicted accumulator depressurizations are compared to the actual test blowdown in Figure IV.1-3. The code results agree with the data to within ±10 percent for the first 20 sec. During this time, the accumulator had discharged about 40 percent of its initial mass. The results agree with the depressurization data to within 18 percent to about 30 sec, at which time 53 percent of the liquid mass has been discharged. The agreement continues to diverge until, at the end of the test period, the predicted pressure is approximately 30 percent greater than the test value. The depressurization predictions from the RETRAN and RELAP4 codes were in close agreement. The largest deviation was approximately 2 percent, with RETRAN giving slightly higher values.

Figure IV.1-4 illustrates the effect of the 20-percent reduction in the form loss coefficients on the predicted accumulator depressurization. In evaluating the code sensitivity to time step variations, it was found that with automatic time step control, RELAP4 used considerably smaller time steps (approximately 64 percent smaller) than RETRAN. This resulted in higher magnitude pressure peaks calculated by the RELAP4 code (Figures IV.1-5 and IV.1-6). The results obtained with manual time step control are shown in Figure IV.1-7. As expected, the longer time step damps out the initial pressure peaks for both codes.

1.1.2 Pacific Gas and Electric Analyses

The analyses performed by Pacific Gas and Electric[IV.1-2] were based on the same test as discussed above. The basic difference was associated with an

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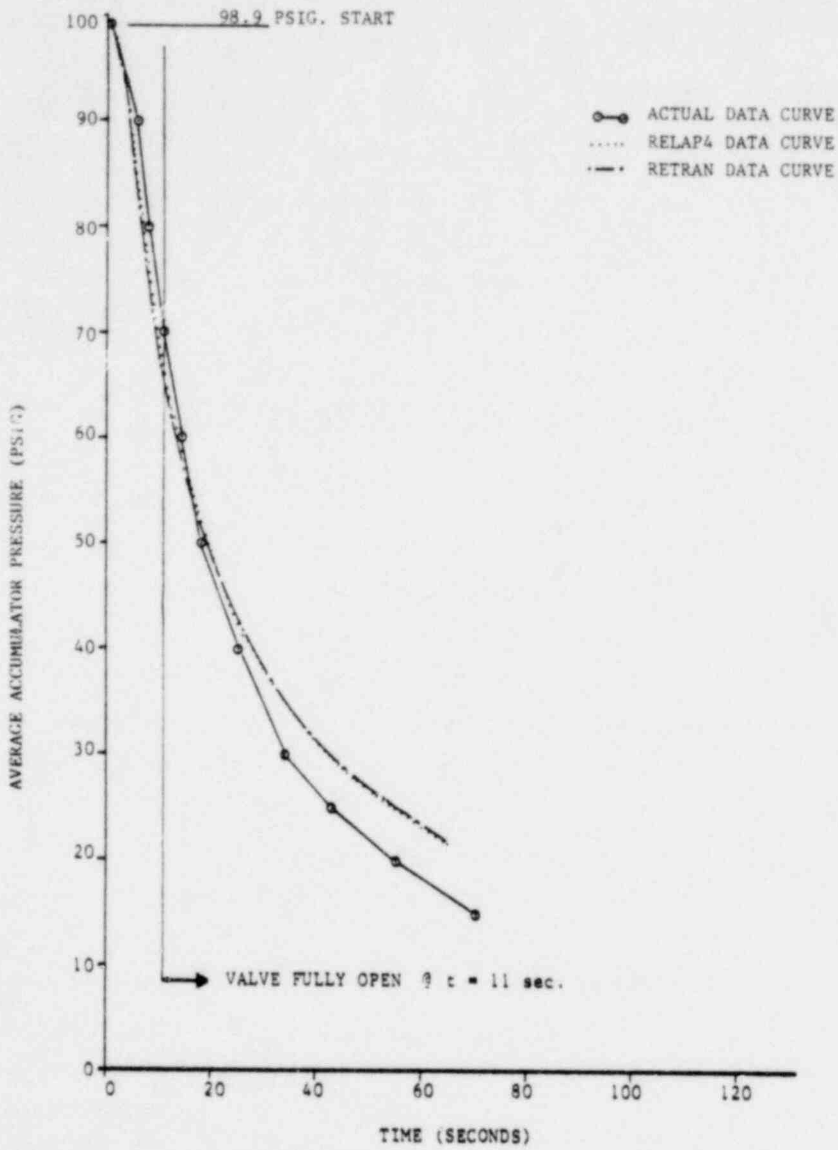

FIGURE IV.1-3 Average Pressure in Accumulator No. 3.

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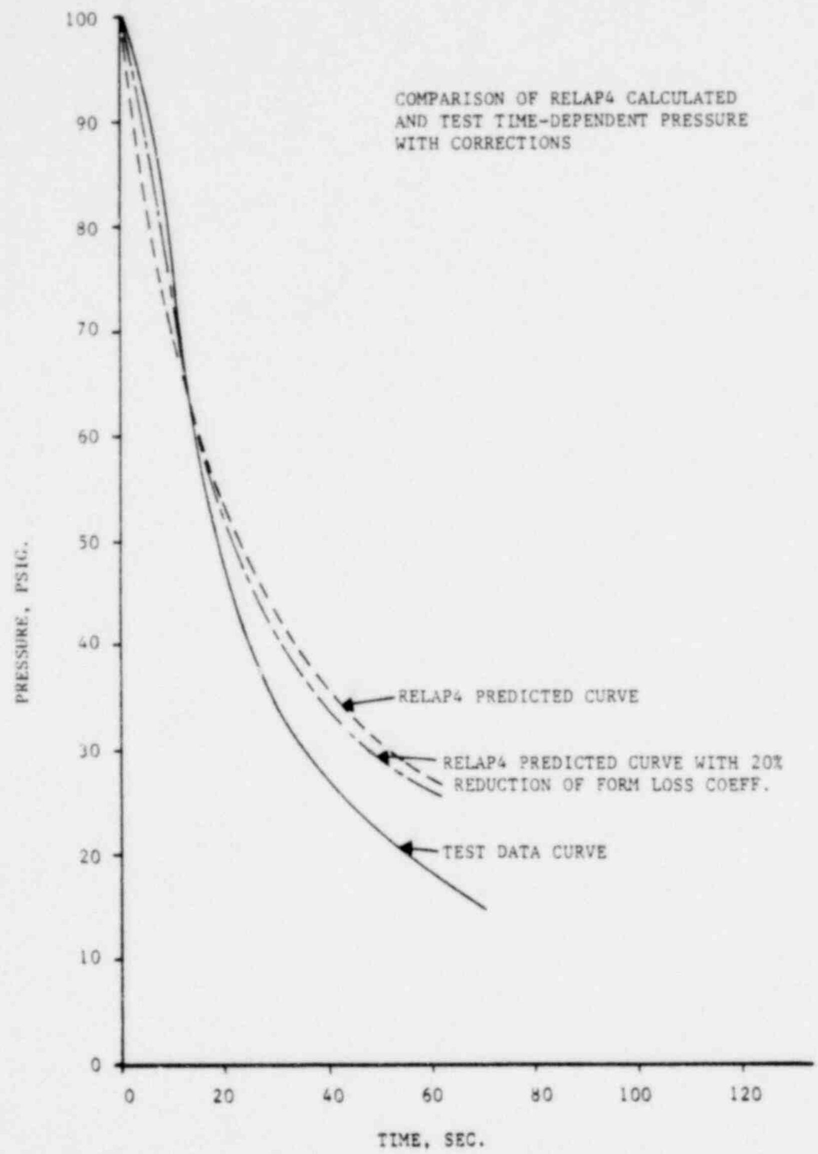

FIGURE IV.1-4 Accumulator No. 3 Depressurization.

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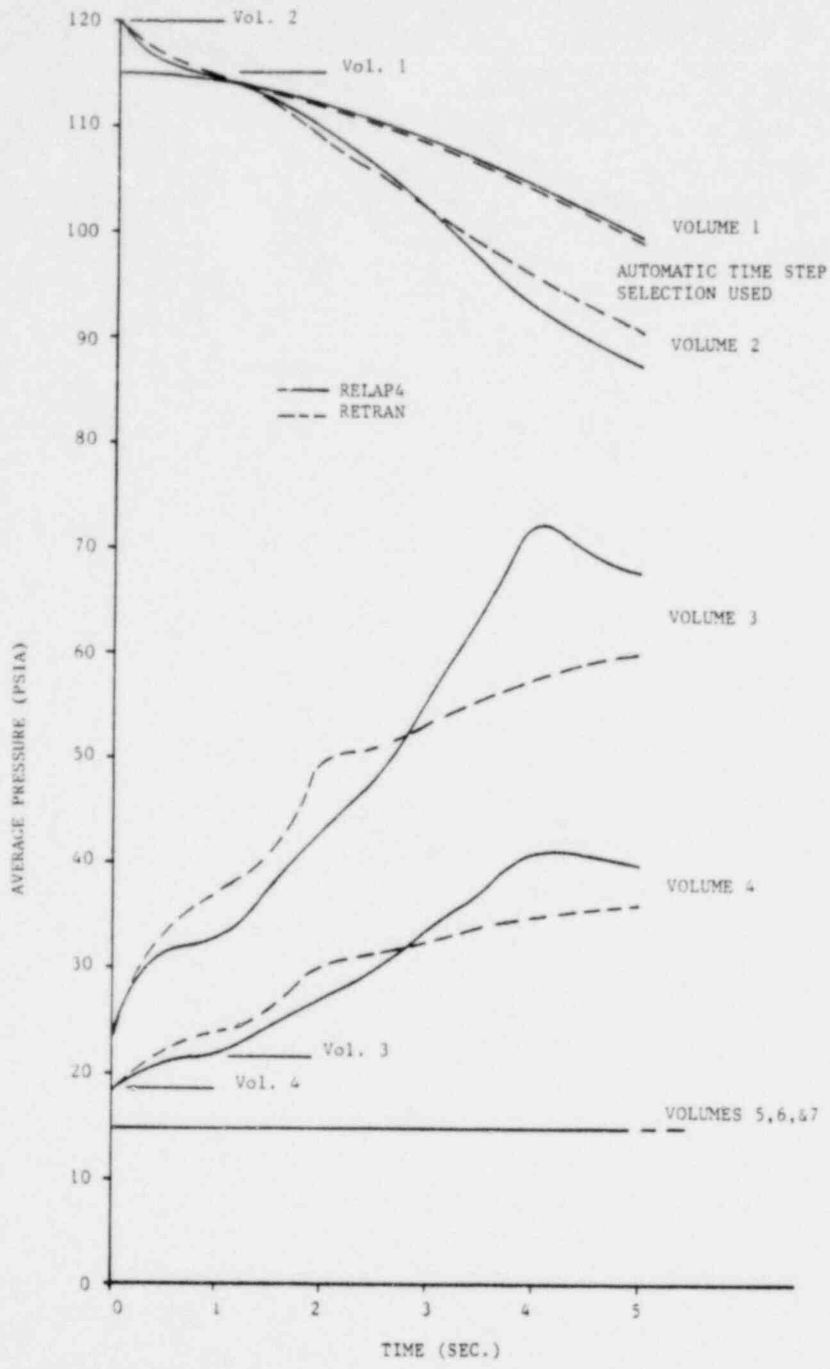

FIGURE IV.1-5 Predicted Average Pressure for Entire System.

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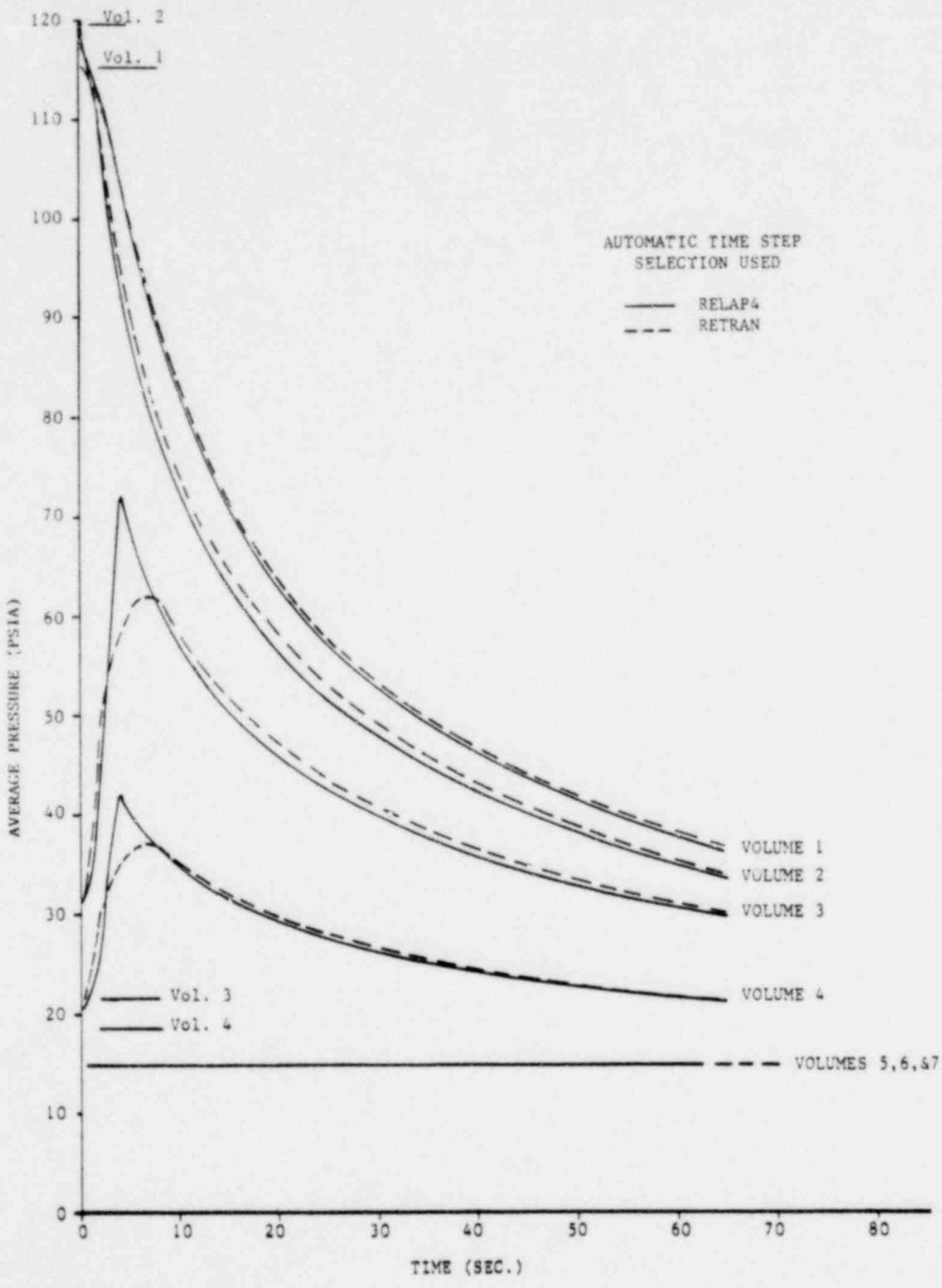

FIGURE IV.1-6 Predicted Average Pressure for Entire System.

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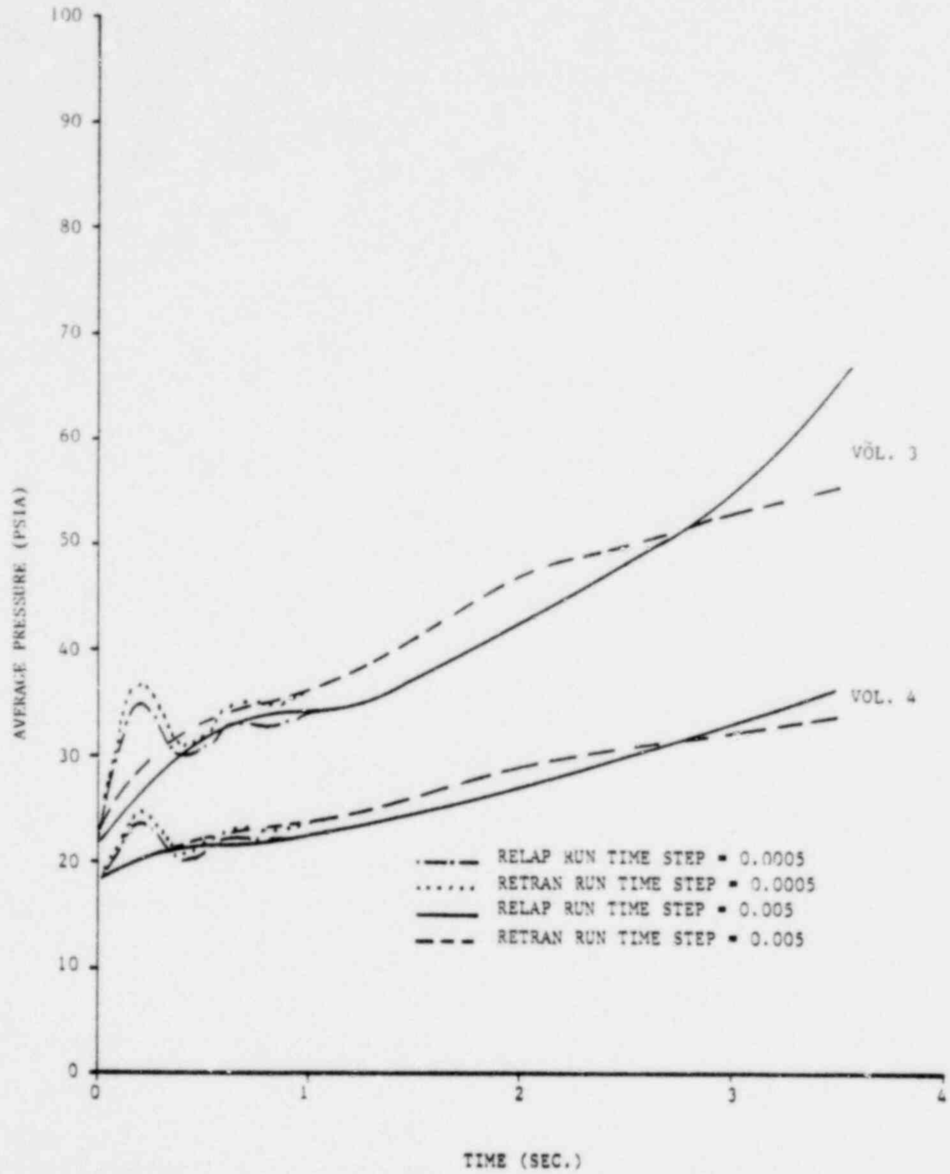

FIGURE IV.1-7 Average Pressure in Volumes 3 and 4 Using Manual Time-Step Control.

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analysis which included temperature effects caused by the expansion of the accumulator gas.

1.1.2.1 Description of Model

The geometric representation of the model is very similar to that depicted in Figure IV.1-2, except that the piping was represented with 3 volumes instead of 5 volumes. The basic model is like the one discussed in Section 1.1.1.1, except that a modification of the accumulator gas expansion was made for one run.

1.1.2.2 Results of Analyses

A comparison between the calculated and measured values of accumulator pressure is given in Figure IV.1-8. The results are very similar to those shown in Figure IV.1-3. The second analysis involved modifying the accumulator pressure to include the effect of temperature changes in the accumulator gas during depressurization. The results shown in Figure IV.1-9 show the effect of this modification.

1.1.3 Summary of Analyses

The results presented in the previous section indicate that RETRAN can adequately represent an accumulator blowdown. The computed values for pressure late in the transient were above the test data, but not by a significant amount. In an effort to more closely agree with the data, each investigator ran sensitivity studies. It is not clear if one (the loss coefficient approach or the accumulator gas modeling) or the other method is more correct based on the analysis. More data comparisons are required to establish this, although the RETRAN model for gas expansion is definitely an approximation and additional modeling work is required in this area.

1.2 Overpressure Mitigating System

In some PWRs, an Overpressure Mitigating System (OMS), using the power-operating pressurizer relief valve (PORV) with a low-pressure setpoint, has been installed to provide a means of pressure relief during periods of low pressure and low temperature Reactor Coolant System (RCS) operation. The OMS will mitigate

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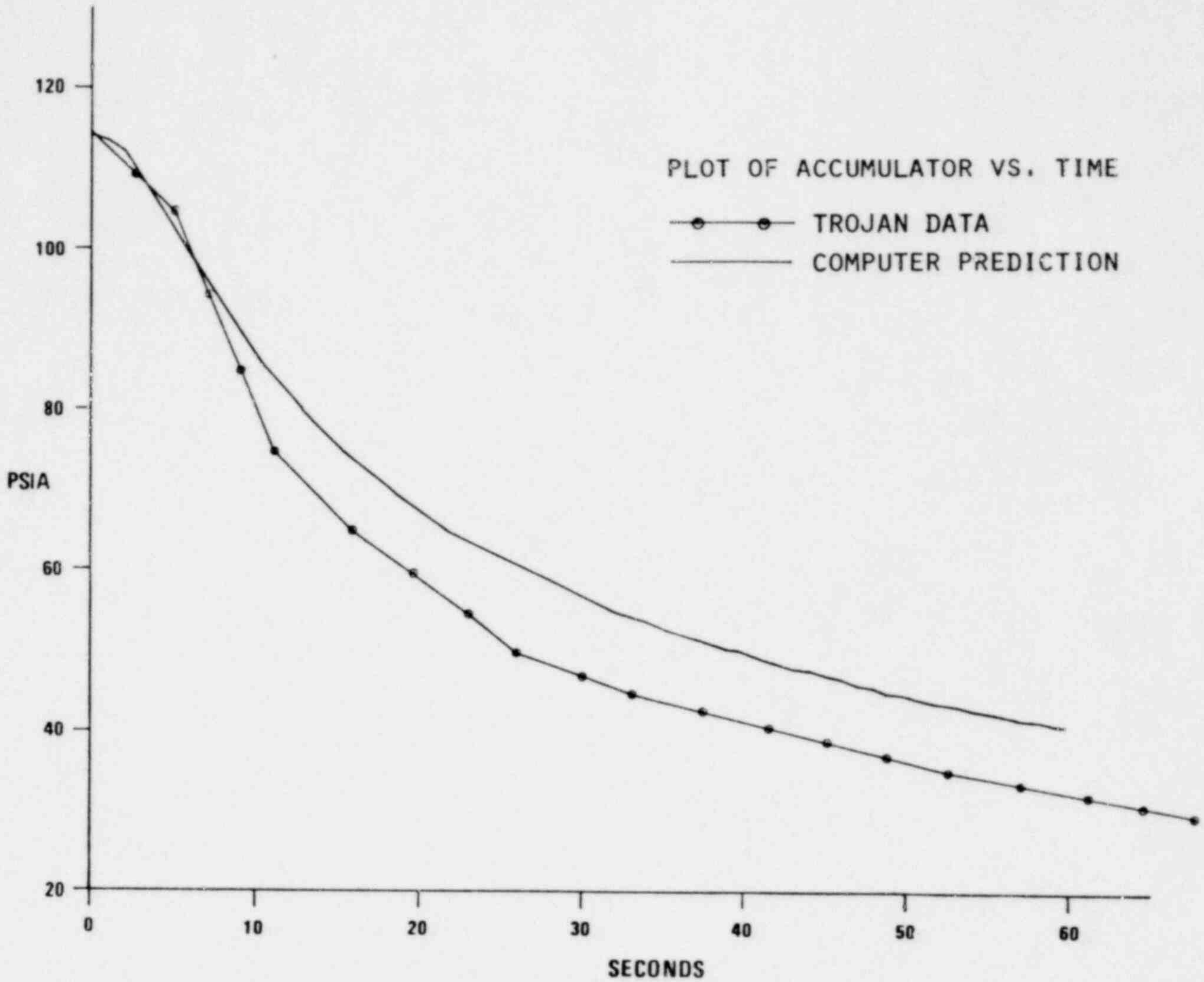

FIGURE IV.1-8 Uncorrected Accumulator Pressure.

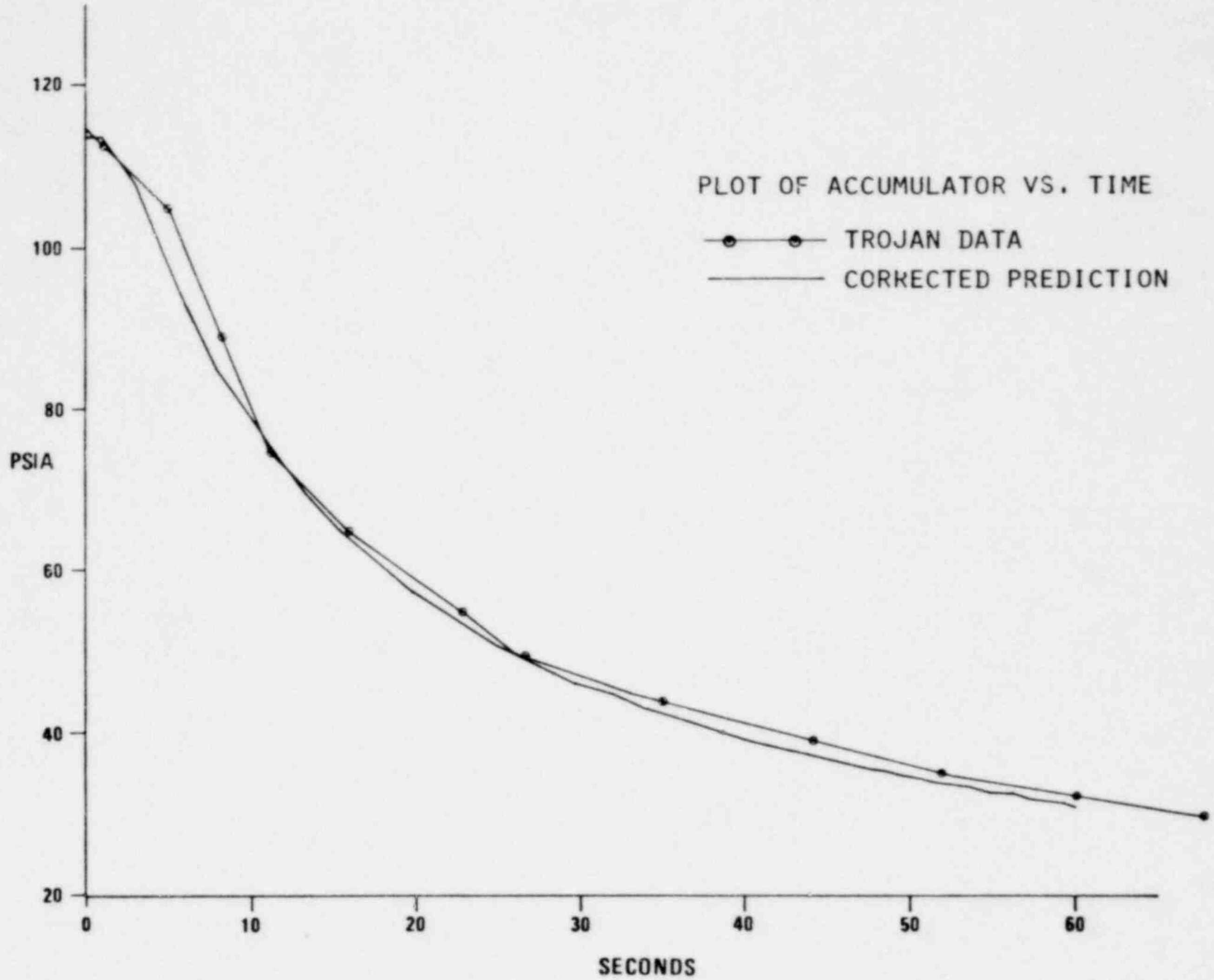

FIGURE IV.1-9 Corrected Accumulator Pressure.

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possible pressure transients that can occur during startup and shutdown without the need for immediate operator intervention. RETRAN analyses[IV.1-3] of mass input pressure transients were performed to determine the OMS performance for a safety injection (SI) pump startup incident and for the centrifugal charging pump operating with the letdown isolated.

1.2.1 Description of Model

A simple 1-volume, 2-junction RETRAN model was used: the volume representing the RCS, 1 junction representing the mass injection, and the other junction representing mass relief through a PORV. The SI pump startup incident was evaluated for a reference case since it was desired to compare the RETRAN calculated results to those computed with a standard vendor code. The assumptions used in this reference analysis are given in Table IV.1-1. The incident of a centrifugal pump charging with letdown isolation was analyzed using the pl... specific assumptions listed in Table IV.1-2.

The transients were initiated by a time-zero trip of the fill junction. The PORV was modeled to open following attainment of the relief pressure setpoint taking into account the appropriate delay time. Similarly, the PORV closes as a result of reaching a reset pressure setpoint also taking into account the appropriate delay time. Instantaneous instrument response times were assumed.

1.2.2 Results of Analyses

The RETRAN code and model were shown to produce favorable results compared to a vendor analysis of this reference pressure transient, as shown in Figure IV.1-10. In the vendor analysis, the startup characteristics of the SI pump were modeled while in the RETRAN analysis, full pump speed was assumed during the entire transient. This accounts for the differences in results during the first few seconds. The small difference in the overshoot and undershoot pressures may be due to the computational differences between the two codes. The cycle time from one PORV relief setpoint (615 psia) to another is nearly the same for both analyses.

An incident of a centrifugal pump charging with letdown isolation was analyzed to determine the criteria for PORV cycling for use in the design of air accumula-

TABLE IV.1-1

ASSUMPTIONS USED IN REFERENCE SI PUMP STARTUP ANALYSIS

Parameter	Value
Initial RCS pressure	65 psia
RCS volume	6000 ft^3
RCS temperature	100°F
PORV relief setpoint	615 psia
PORV reseat setpoint	595 psia
SI pump delivery	Curve C of Figure 1(a), Reference IV.1-3
PORV relief flow	10 psig backpressure curve of Figure 1(b), Reference IV.1-3
PORV opening characteristic	Figure 1(c), Reference IV.1-3, with 3 sec stroke
PORV closing characteristic	Figure 1(d), Reference IV.1-3, with 5 sec stroke

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TABLE IV.1-2

ASSUMPTIONS USED IN CENTRIFUGAL CHARGING PUMP ANALYSIS

Parameter	Value
Initial RCS pressure	65 psia
RCS volume	12,500 ft^3
RCS temperature	100°F
PORV relief setpoint	425 psia
PORV reseat setpoint	405 psia
PORV open characteristic	0.3 sec delay time, 0.05 sec stroke time
PORV closing characteristic	1.62 sec delay time, 0.58 sec stroke time
PORV relief flow	10 psig backpressure curve of Figure 1(b), Reference IV.1-3
Centrifugal charging pump delivery	50 psig initial pressure curve of Figure 2, Reference IV.1-3

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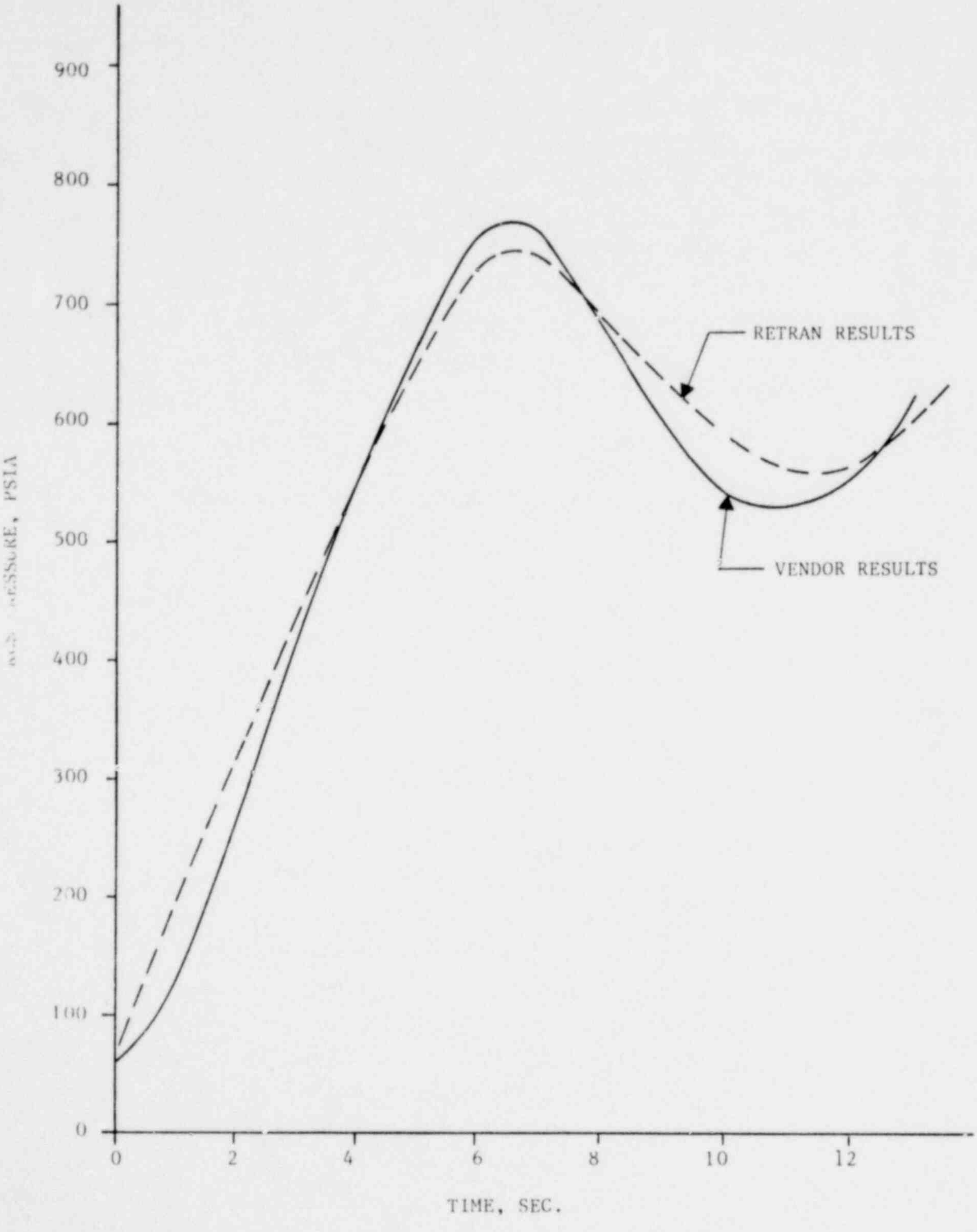

FIGURE IV.1-10 RCS Pressure Transient for One Cycle of PORV Opening and Closing for Reference SI Pump Startup Incident.

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tors in the OMS. Air accumulators are provided to preclude the situation where a loss of instrument air causes both letdown isolation and an overpressure incident. The results of the plant-specific RETRAN analysis of this incident are given in Figure IV.1-11. The PORV is calculated to mitigate this transient and the results indicate a minimum PORV cycle time of 19.1 sec. If it is assumed that no operator action can be taken for 10 min, then the air accumulator design must assure that 32 PORV cycles can be accomplished. The maximum pressure overshoot is only about 2 psi above the setpoint, which is insignificant. The maximum undershoot yields an RCS pressure of 318 psia, which is high enough to assure that damage to the reactor coolant pump seals is precluded.

1.2.3 Summary of Results

From the analyses of the reference SI pump startup incident, the good agreement between the RETRAN and vendor code results provides a reasonable confidence in the RETRAN model and code for this type of analyses. Only about a 5 percent difference in overshoot and undershoot pressures was observed. The plant-specific RETRAN analysis of centrifugal charging with letdown isolation formed the basis for sizing the air accumulators in the OMS and demonstrated the adequacy of the system.

1.3 Steam Generator Secondary Side Dryout

Provisions of long-term core cooling capability following a variety of operational transients and accident situations, other than LOCAs, is an area that is receiving increased attention. Central to a long-term cooling evaluation of the complete loss of feedwater, resulting from a loss of all a-c power, is the availability of steam generator secondary water inventory to provide primary system heat removal. To analyze this situation, a RETRAN nonconduction heat exchanger was used to model steam generator behavior. Provisions were made for time-dependent mass addition and pressure-dependent mass relief. Junctions were specified for the normal feedwater inlet line, the steam line, and the five main steam line safety valves. Two analyses were performed using this model; a bounding analysis to duplicate a vendor calculation, and a conservative calculation to obtain a more realistic rate of steam generator dryout. The results summarized below are documented in Reference IV.1-4.

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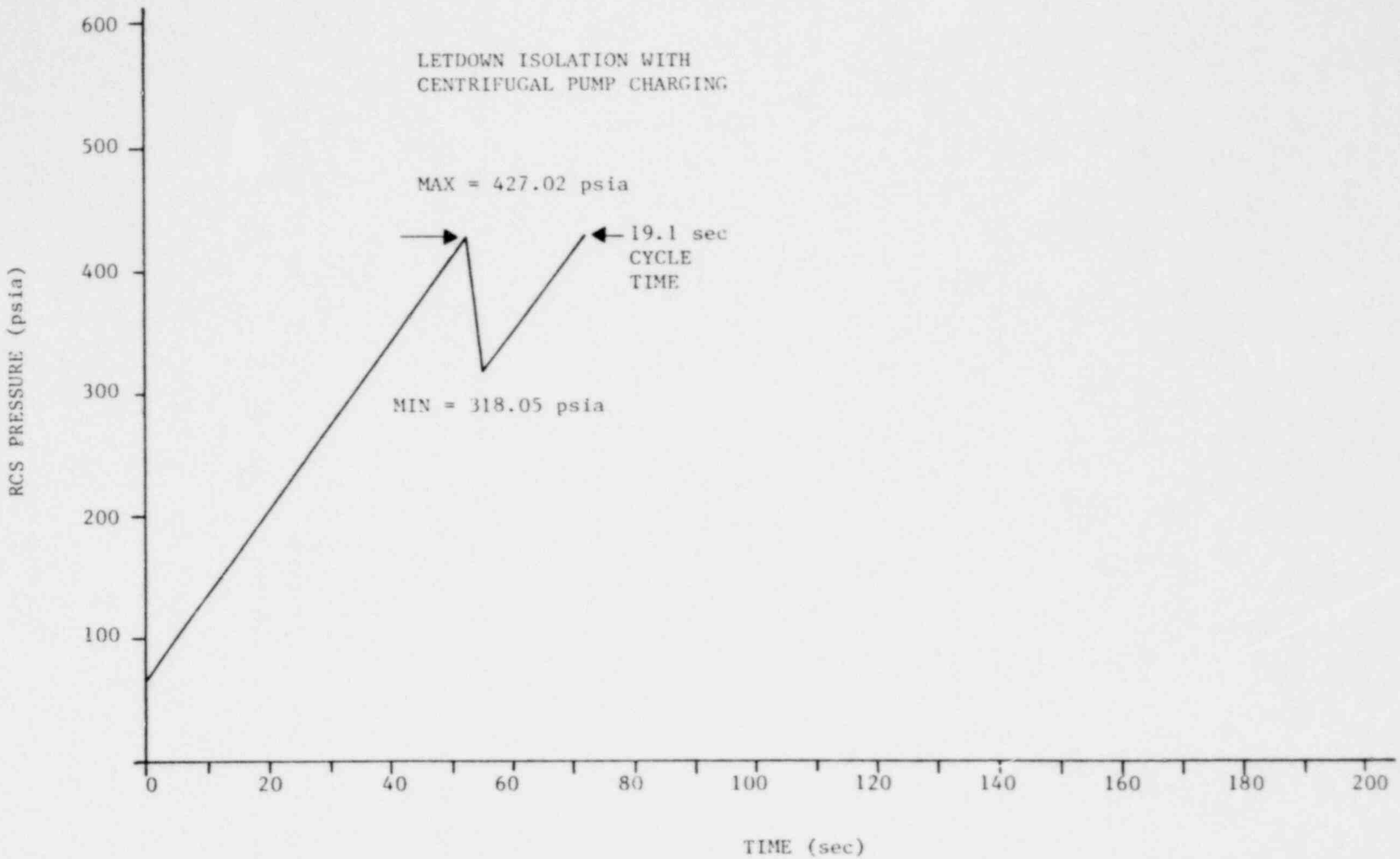

FIGURE IV.1-11 RETRAN Code Calculated RCS Pressures for Letdown Isolation with Centrifugal Pump Charging Incident.

1.3.1 Description of Model

The RETRAN model consisted of a 1-volume representation of the steam generator secondary side with one positive and six negative fill junctions as shown in Figure IV.1-12. Pertinent information on the volumes, flow areas, and other geometric data is given in Table IV.1-3. The major differences between the conservative and bounding analyses involve the assumptions regarding the initial water level, power inputs, and variation of the valve closing times. The sequence of events for these analyses is summarized in Table IV.1-4.

In order to model a time-dependent heat addition to the secondary system without calculating primary system behavior, the nonconduction heat exchanger option was used. For this option, a curve of power versus time was obtained from SAR values of nuclear power and total residual heat following reactor trip. The heat addition rate used is quite conservative with respect to what would be expected following an actual reactor shutdown. To monitor the change in water inventory as dryout proceeds, complete separation of the liquid and vapor phases was modeled by use of an artifically large bubble velocity (10^6 ft/sec). Phase separation under this assumption causes the mixture level to be lower than would actually be expected during plant operation. To compensate for this, trip setpoints and normal operating levels were adjusted accordingly.

1.3.2 Results of Analyses

Two analyses were performed, a bounding analysis to approximate the vendor calculation and a conservative analysis which is expected to approximate actual conditions.

The bounding analysis was initiated with a steam generator secondary side mixture level of 0 percent of the instrument span which is below the low-low level trip setpoint. The initial reactor power was specified as 102 percent of engineered safeguards design rating. The results of this analysis compare within 5 percent of the dryout time predicted by the vendor (Figure IV.1-13). The major difference in the predicted results was the variation of rate of water loss during the transient. The RETRAN curve is the result of not modeling the primary system behavior and of using a heat exchanger model that does not provide for changing surface areas. The very rapid initial drop in coolant inventory in the vendor's

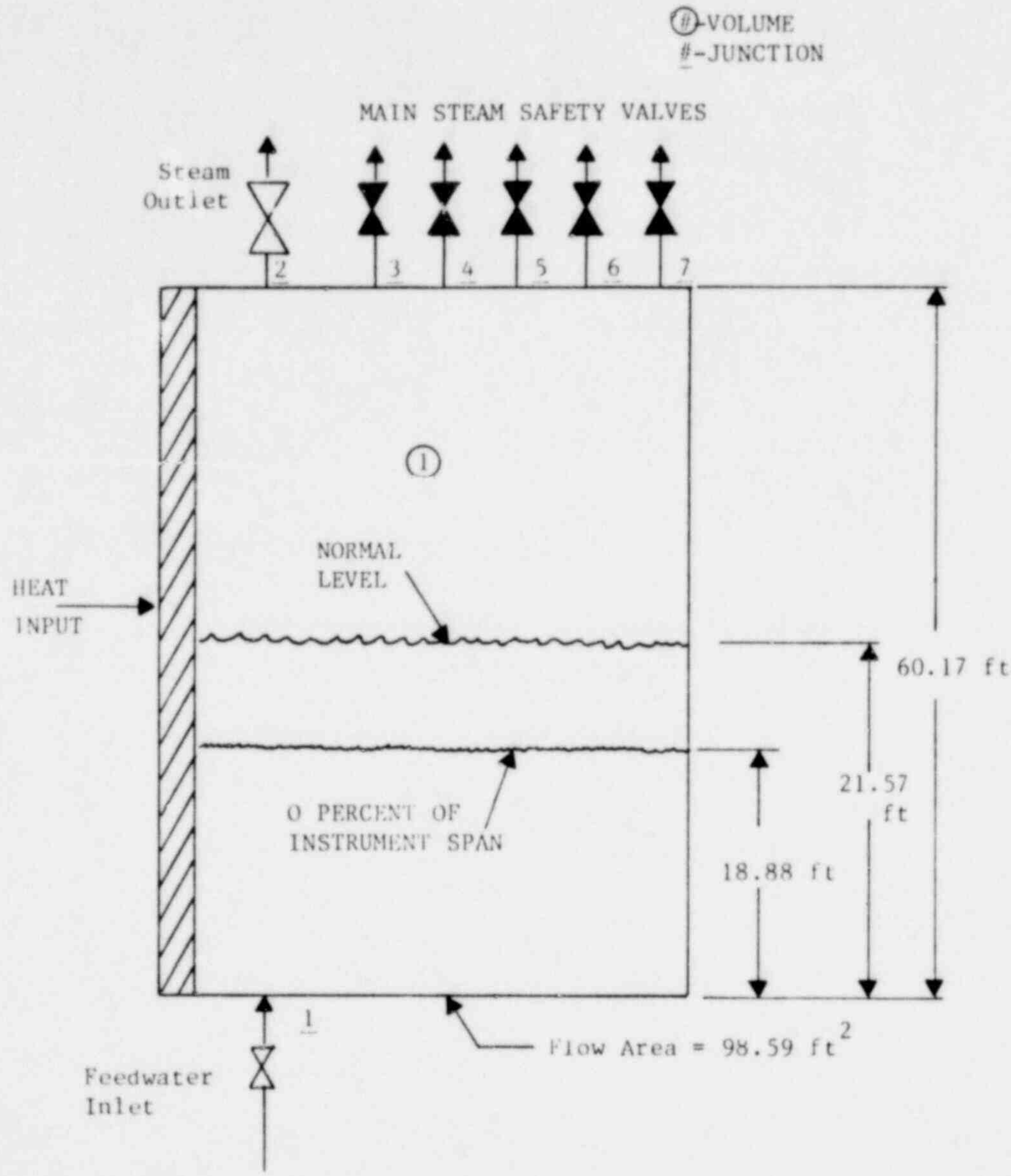

FIGURE IV.1-12 Schematic of RETRAN Steam Generator Dryout Model.

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TABLE IV.1-3

RETRAN STEAM GENERATOR SECONDARY SIDE DRYOUT ANALYSIS INPUT ASSUMPTIONS

Parameter	Value	
	Conservative	Bounding
Steam generator secondary side volume (ft^3)	5,931.8	5,931.8
Secondary side pressure (psia)	910	910
Initial secondary side water mass (lb_m)	99,900	88,483
Initial feedwater flow rate (lb/sec)	1,046.9	1,046.9
Initial heat addition rate (MWt)[a]	852.75	910.34
Main steam line safety valve relief setpoints (psia) Valves No. 1-5, Junctions No. 3-7	1,184.7 1,214.7 1,224.7 1,237.7 1,244.7	1,184.7 1,214.7 1,224.7 1,234.7 1,244.7
Main steam line safety valve reset pressures	1,125.5 1,153.97 1,163.47 1,172.97 1,182.47	1,125.5 1,153.97 1,163.47 1,172.97 1,182.47
Flow through main steam line safety valve (lb/sec)	186.41 191.13 253.27 255.34 257.41	186.41 191.13 253.27 255.34 257.41
Main steam line isolation time (sec)	5.0	5.0
Feedwater isolation time (sec)	5.0	0.0

[a] Represents heat addition to one steam generator in a four-loop plant.

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TABLE IV.1-4

SEQUENCE OF EVENTS FOR STEAM GENERATOR SECONDARY SIDE DRYOUT ANALYSES

Analysis	Event	Time (Sec)
Conservative	Loss of a-c power	0.0
	Reactor trip	1.0
	Feedwater and steam line isolation begins	1.0
	Feedwater isolation complete	5.0
	Steam line isolation complete	5.0
Bounding	Loss of a-c power	0.0
	Feedwater and steam line isolation begins	1.0
	Feedwater isolation complete	1.0001
	Reactor trip	3.0
	Steam line isolation complete	5.0

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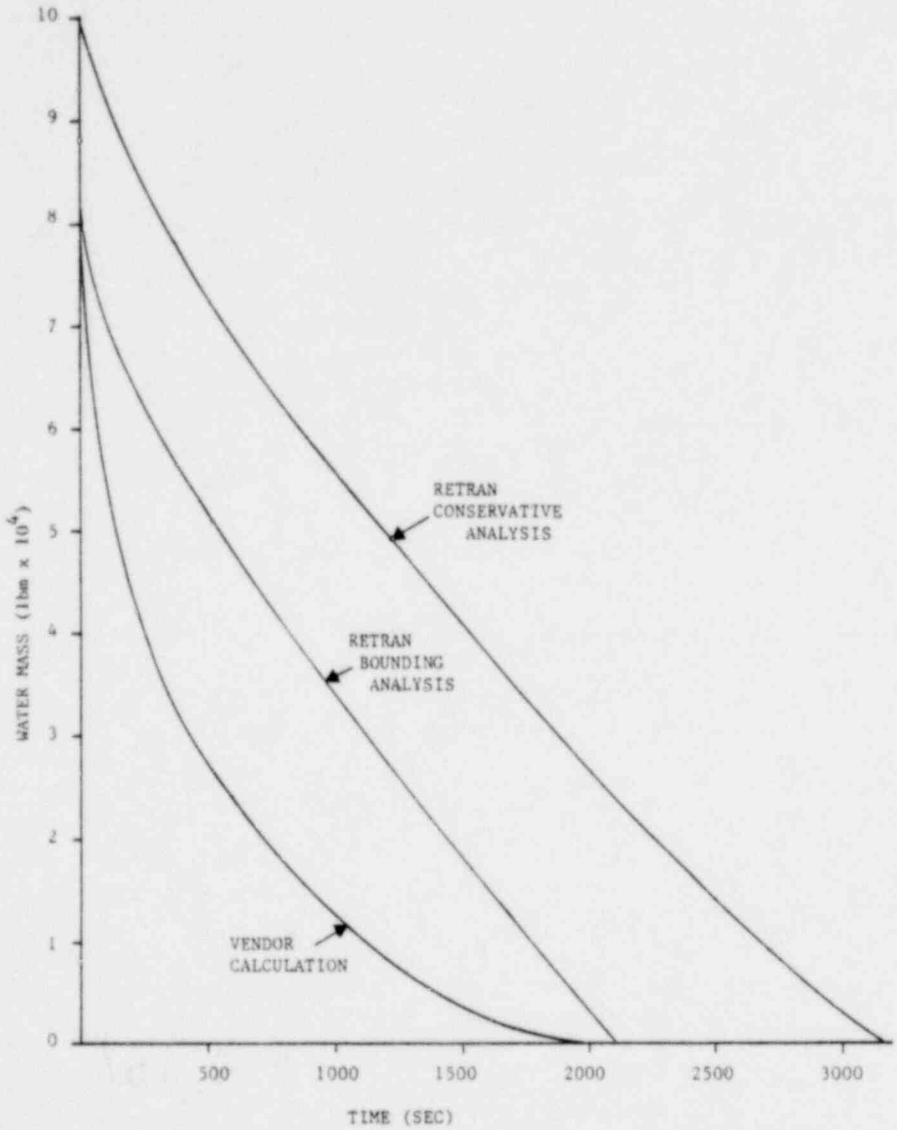

FIGURE IV.1-13 Steam Generator Secondary Side Water Inventory as a Function of Time After Loss of Feedwater.

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curve is the result of the high temperature difference between primary and secondary systems during the initial primary system transient. This causes a much more rapid energy transfer to the secondary side during the first 300 sec than is predicted by the RETRAN model. This rapid initial change in coolant inventory causes a corresponding decrease in heat transfer area to the water during the latter stages of the transient. From about 900 sec until the steam generator has been boiled dry at 2000 sec, the vendor model predicts a comparatively slower loss of water.

RETRAN results for the steam generator secondary side water mass and average pressure as functions of time are given in Figures IV.1-14 and IV.1-15 for the bounding analysis. The secondary side water dries out in about 2100 sec (35 min). Figure IV.1-15 shows the predicted mass relief through the steam line safety valves. Intermittent actuation of Valve Number 1, which has an opening setpoint of 1184.7 psia and a resetting setpoint of 1125.5 psia, was sufficient to provide pressure relief for the isolated steam generator. The valve was assumed to open linearly in 0.5 sec after the opening setpoint is reached, and to close instantaneously following attainment of the reset setpoint.

The initial conditions for the conservative analysis are closer to what would be expected to occur during normal operations. The initial water level was input as 44 percent of the instrument span, which represents the normal level, and the power level was specified as 100 percent of rated power. Additionally, feedwater flow was assumed to coast down from the full to no-flow conditions over a 5 sec ramp to simulate the closing time of the feedwater isolation valves. The results of this analysis are very similar in shape to the bounding analysis done with RETRAN, however, the dryout time is shifted to the right an additional 1040 sec. The conservative analysis indicates that the residual heat removal capability is predicted to be essentially lost around 52 min, since the secondary side water has boiled off.

1.3.3 Summary of Results

From these calculations it can be seen that, using a simple RETRAN model, scoping calculations can be performed that yield meaningful steam generator secondary water dryout times. The calculations are also relatively inexpensive to run considering the long real times involved.

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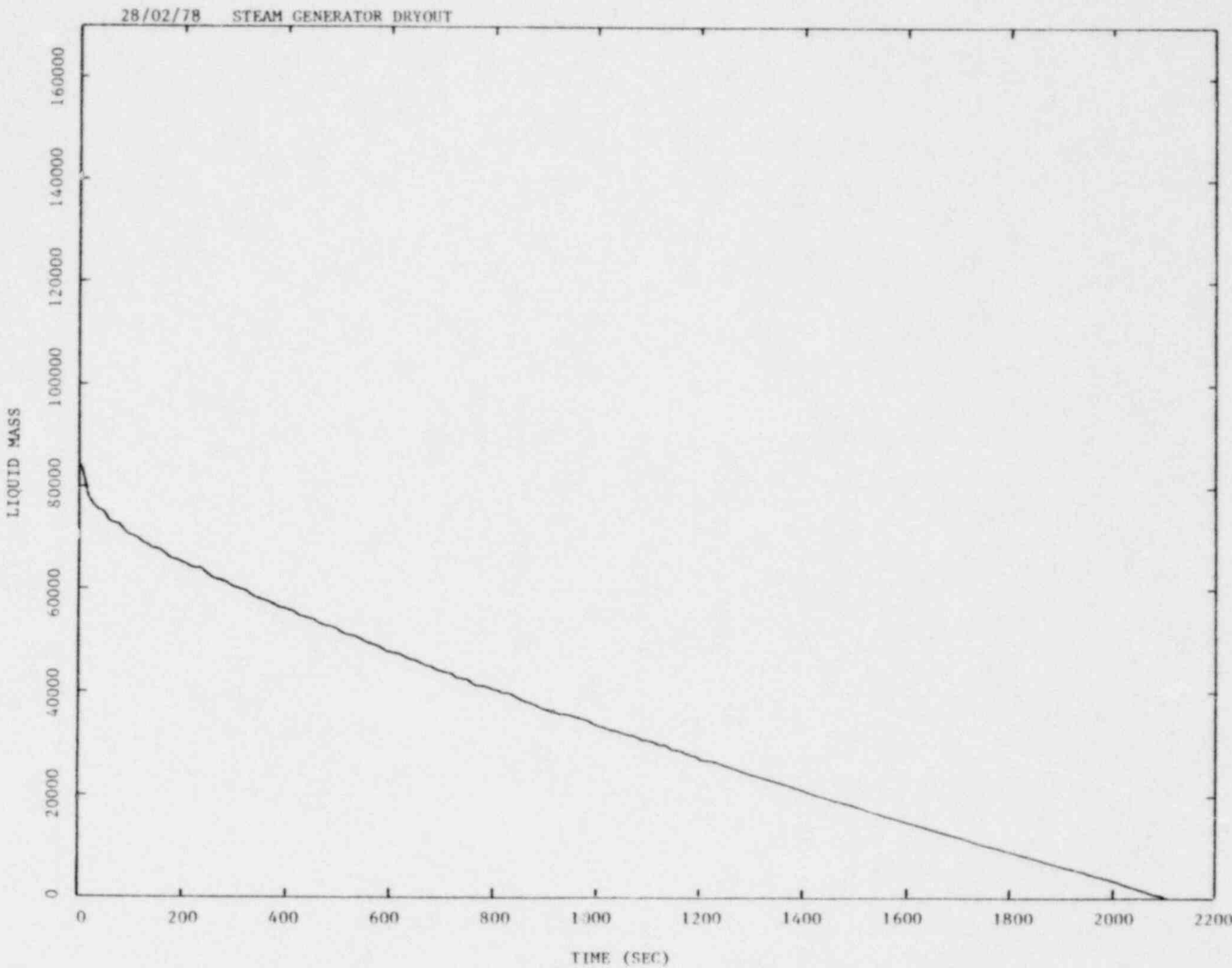

FIGURE IV.1-14 RETRAN Time-Dependent Steam Generator Secondary Side Liquid Mass (Bounding Analysis).

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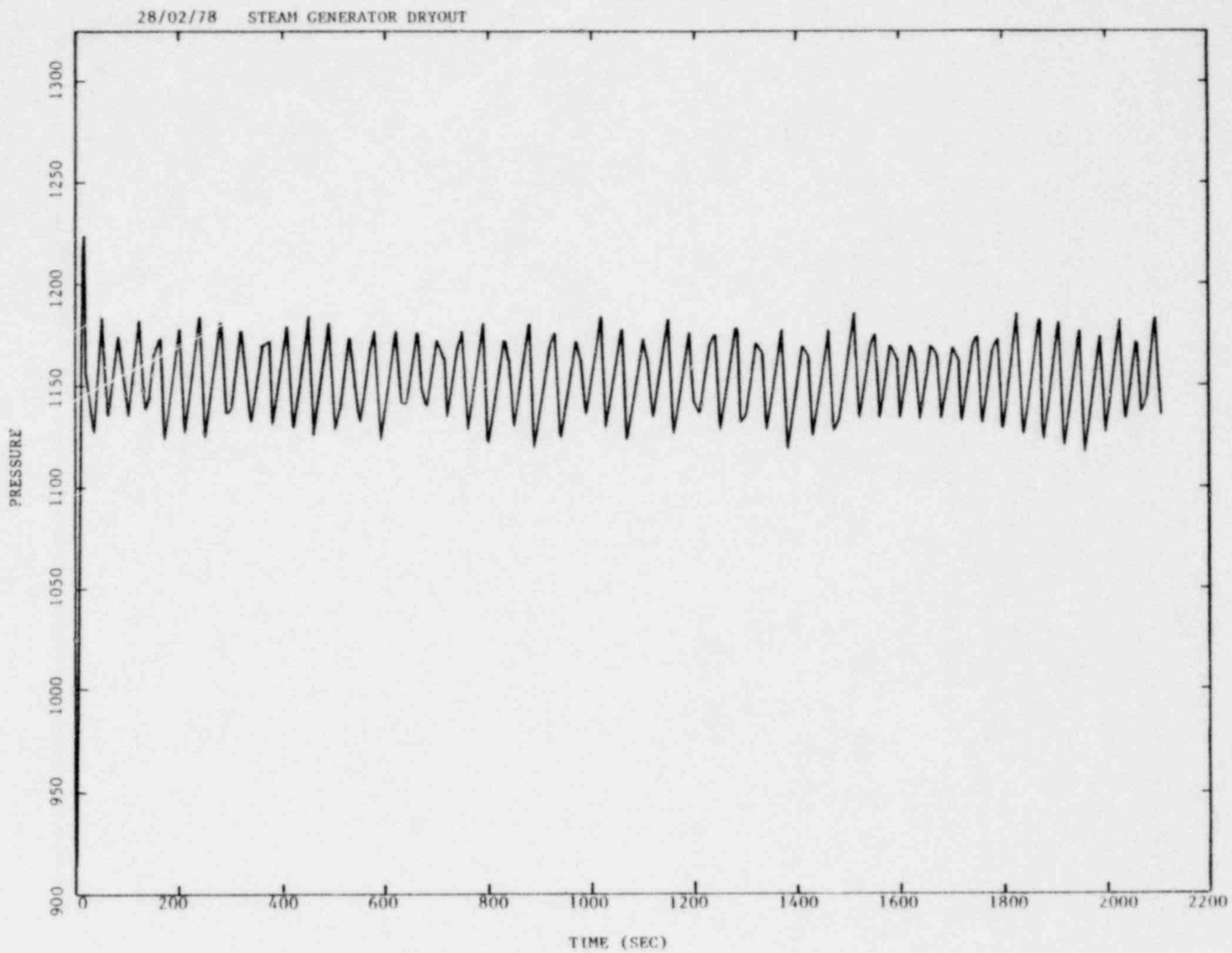

FIGURE IV.1-15 RETRAN Time-Dependent Steam Generator Secondary Side Main Relief (Bounding Analysis).

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1.4 Reactor Cavity Subcompartment Pressurization Analysis

During normal operations, a reactor vessel cavity annulus seal ring is suspended above the annular gap between the reactor vessel and the biological shield by a set of support jacks. The seal ring is a steel ring, 11-in. wide by 3/4-in. thick. If a LOCA should occur that pressurizes the reactor vessel cavity, a portion of the resultant blowdown will vent through the annular gap and exert a jet impingement load on the seal ring. A subcompartment pressurization analysis performed in conjunction with an evaluation of the asymmetric loads on the reactor vessel supports indicated that the seal ring could become a potentially damaging missile. The pressurization analysis was performed using a standard architect-engineer, multiple-compartment computer code, with a mass and energy release generated by a vendor blowdown code specified as input data. A rather coarse nodalization of the reactor vessel cavity volume was used. A RETRAN analysis was performed to determine the effect of increased nodalization in the vicinity of the pipe break and seal ring on the uplift forces. A discussion of the multicompartment code and RETRAN analyses is given in Reference IV.1-5.

1.4.1 Description of Model

An isometric view of the region studied is shown in Figure IV.1-16. A 5-volume, 5-junction RETRAN model was developed to reproduce the original subcompartment analysis (Figure IV.1-17a). The K-factors specified were 1.47 and 2.08 for critical and subcritical flow conditions. One-half of the mass and energy specified in the 288-in.^{2} hot-leg break blowdown data was released to Volume 1 by use of a fill junction. Volumes 2, 12, 14 and 20 were treated as time-dependent volumes, with their transient pressure behavior specified from the results of the original calculation. The time-dependent temperatures and qualities were derived from the blowdown data. All of the volumes were assumed to be initially filled with two-phase water (no air present) at the same conditions as existed 0.001 sec following the pipe break. For the increased nodalization analysis, a 9-volume, 15-junction RETRAN model was used to reflect the volume subdivisions as shown in Figure IV.1-17b.

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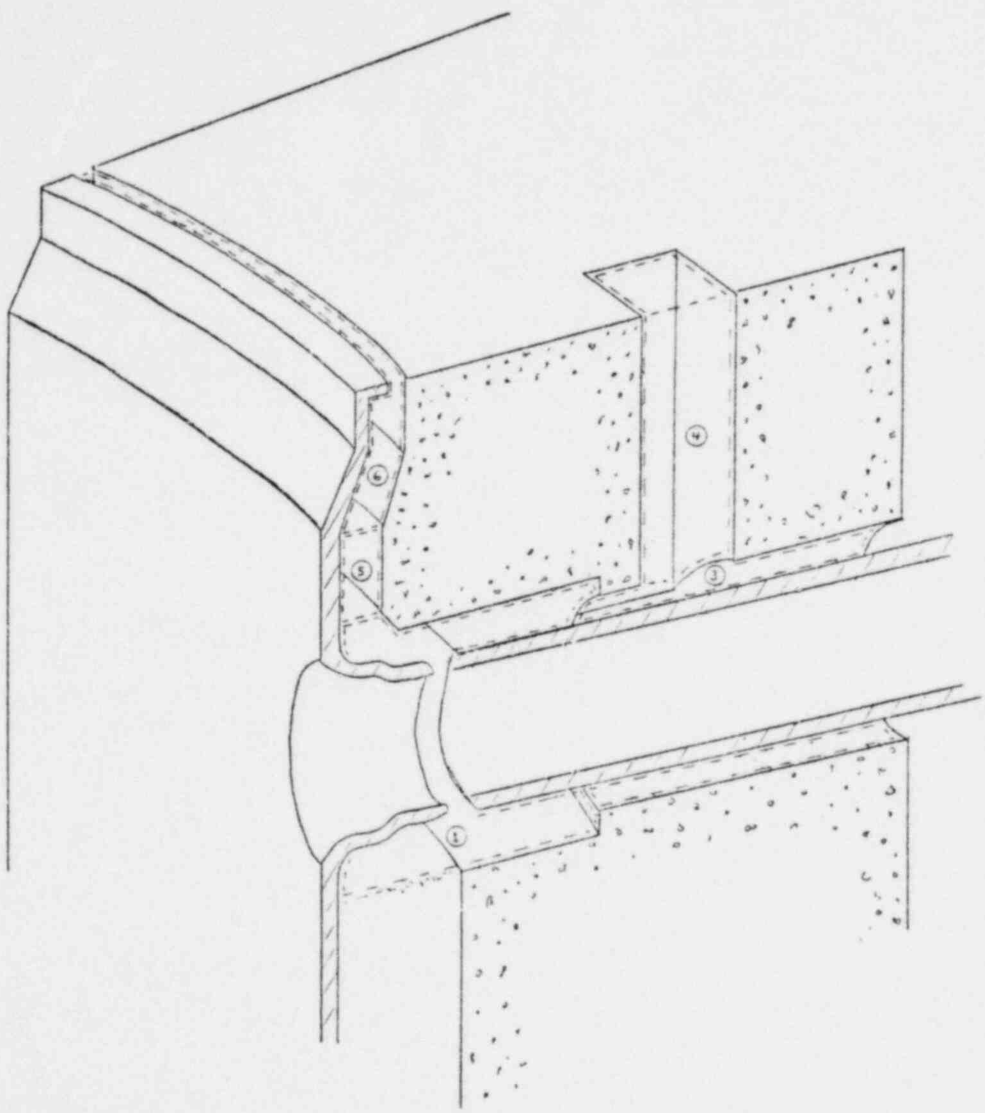

FIGURE IV.1-16 Isometric Schematic for Increased Nodalization Analysis

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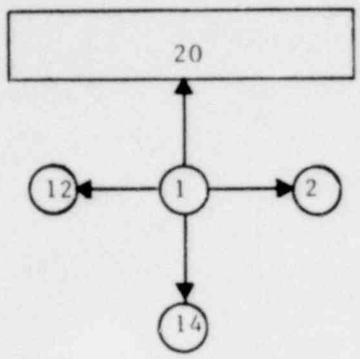

FIGURE IV.1-17a RETRAN Nodalization Flow Diagram for Original Subcompartment Analysis.

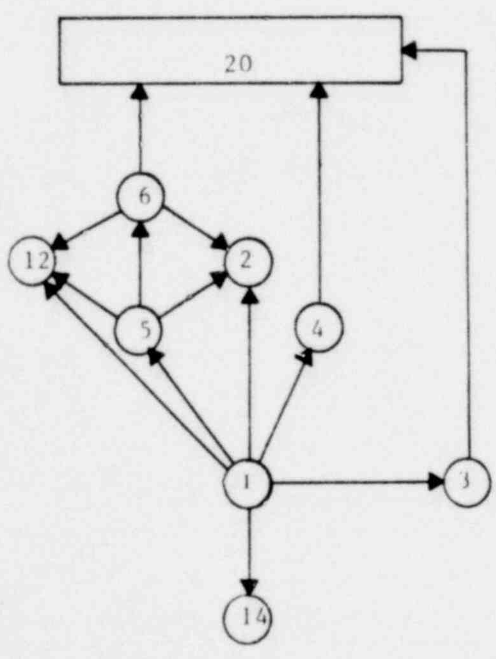

FIGURE IV.1-17b RETRAN Flow Diagram for Increased Nodalization Analysis.

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1.4.2 Results of Analyses

Before the RETRAN code and modeling techniques for subcompartment analysis could be generally applied, it was necessary to verify the appropriateness of the method. This was accomplished by comparing RETRAN calculated results to those from a standard multiple-compartment code for the same problem. Such a comparison is made in Figure IV.1-18 for the Compartment 1 pressures during the first 0.2 sec following the original hot-leg break. The excellent agreement gives confidence in the RETRAN model.

The results from the RETRAN increased nodalization analysis are presented in Figure IV.1-19. As expected, if the volume encompassing the pipe break (Volume 1) is made smaller, the pressure response is of a greater magnitude. Similarly, the pressure response is reduced for Volume 6 as compared to the original, undivided volume since resistances to flow have been introduced.

1.4.3 Summary of Results

The RETRAN code proved useful to perform nodalization studies of containment subcompartment pressurization analyses. A model that considers only two-phase mass and energy addition (no air) to a nodalized volume surrounded by time-dependent volumes representing boundary conditions was shown to be valid. This computational model was also relatively inexpensive to run.

1.5 Summary of Results

Two items are worthy of noting for the analyses discussed in this section. The first item, relative to RETRAN qualification, is that RETRAN results were in good agreement with test data and/or other code calculations for these problems. It also is noted that, although none of the analyses were for complete reactor systems, the analyses performed are of interest and value for utilities. The analyses also demonstrate that RETRAN is a valuable and useful computational tool for evaluating simple problems.

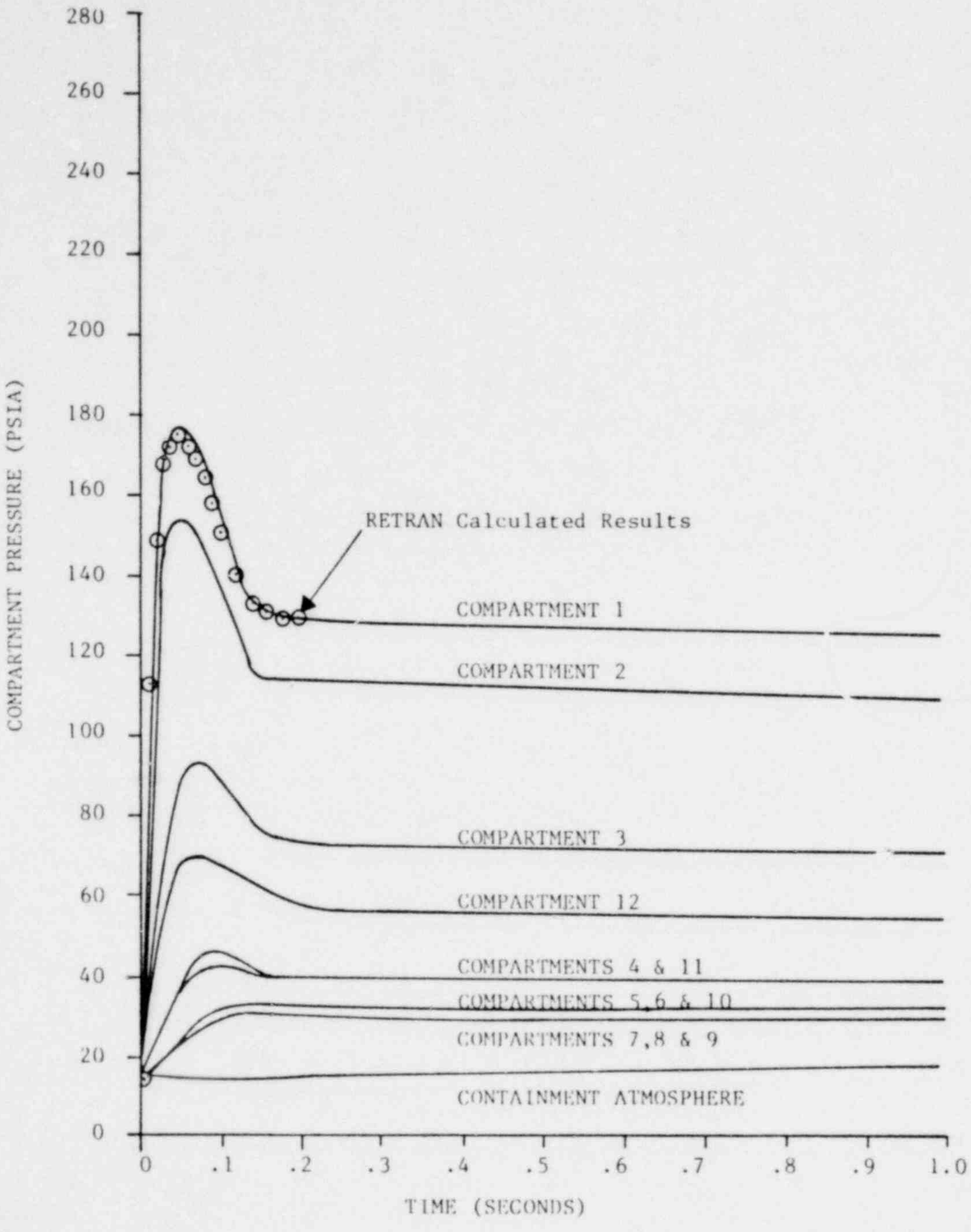

FIGURE IV.1-18 288-Inch^2 Hot Leg Break for Node Elevations 57 Feet 6 Inches to 62 Feet 6 Inches MSL.

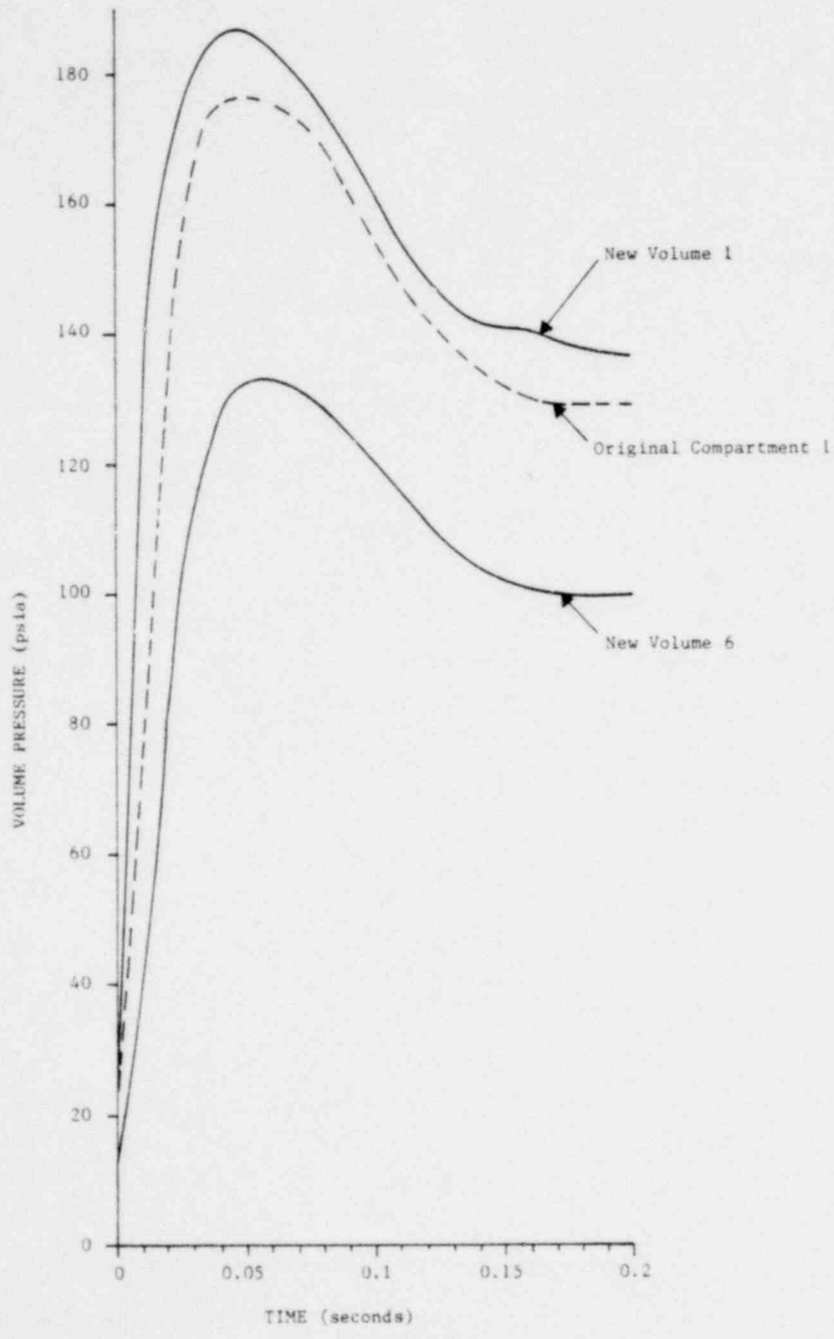

FIGURE IV.1-19 Results of RETRAN Increased Nodalization Analysis (288-Inch^2 Hot Leg Break).

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2.0 YANKEE ANALYSIS

Yankee Atomic Electric Company performed two analyses similar to those discussed in Section 1.0. These analyses were for small systems which are part of a complete reactor system. One RETRAN calculation was designed to evaluate the temperature in a BWR suppression pool. The report[IV.2-1] presents the RETRAN calculations and includes comparisons with results from another code, TEMPOOL[IV.2-2]. The second study[IV.2-3] involved an analysis of a spent fuel pit, and the results were compared to RELAP4 and hand calculations.

2.1 Suppression Pool Temperature Response

A suppression pool is basically the primary containment for a BWR. In addition to this function for a LOCA, the suppression pool is also used to condense steam released from safety relief valves during reactor pressurization transients. The analysis discussed in this section is based on such a transient.

The RETRAN calculations were performed to investigate a nonconducting heat exchanger option which has subsequently been incorporated in the code. The RETRAN results were studied for time step convergence and were also compared to TEMPOOL calculations. TEMPOOL[IV.2-2] is a computer code designed to evaluate these kinds of condensing heat transfer problems.

2.1.1 Description of Model

The RETRAN model for this analysis comprises one volume, one fill junction and a nonconducting heat exchanger (Figure IV.2-1). The heat exchanger represented the heat exchange between the pool water and the service water, and was dependent on both time and the primary side temperature. The fill flow rate and enthalpy were determined from a RETRAN analysis of the reactor system with a stuck open relief valve.

2.1.2 Results of Analysis

Five runs of the basic model were made. As shown in Figures IV.2-2 and IV.2-3, the RETRAN and TEMPOOL results were in good agreement. Three time step sizes were used for RETRAN to determine convergence. The results of these runs, shown in Table IV.2-1, indicate a converged solution for a maximum time step size of 10 sec.

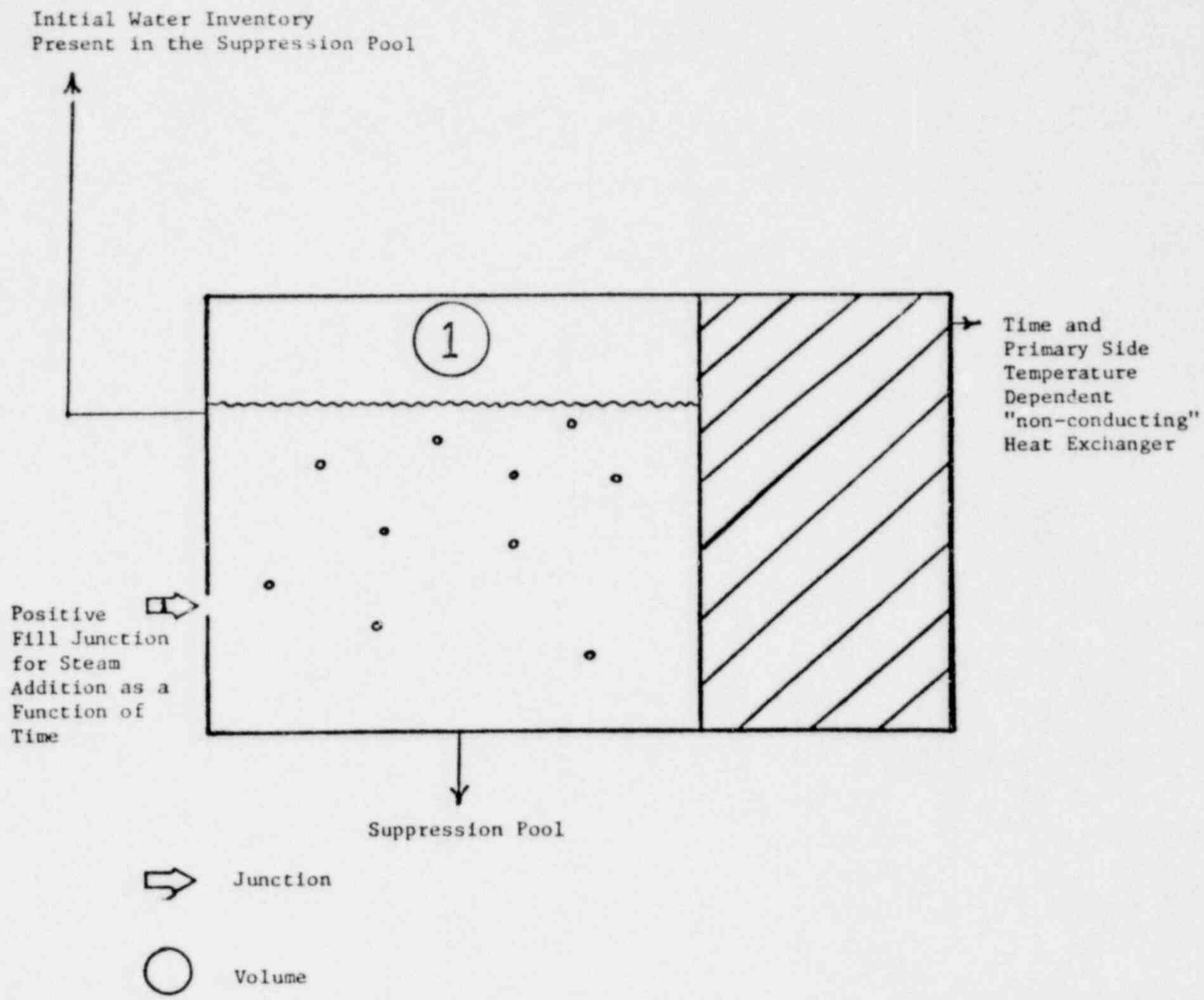

FIGURE IV.2-1 Vermont Yankee Suppression Pool RETRAN Model.

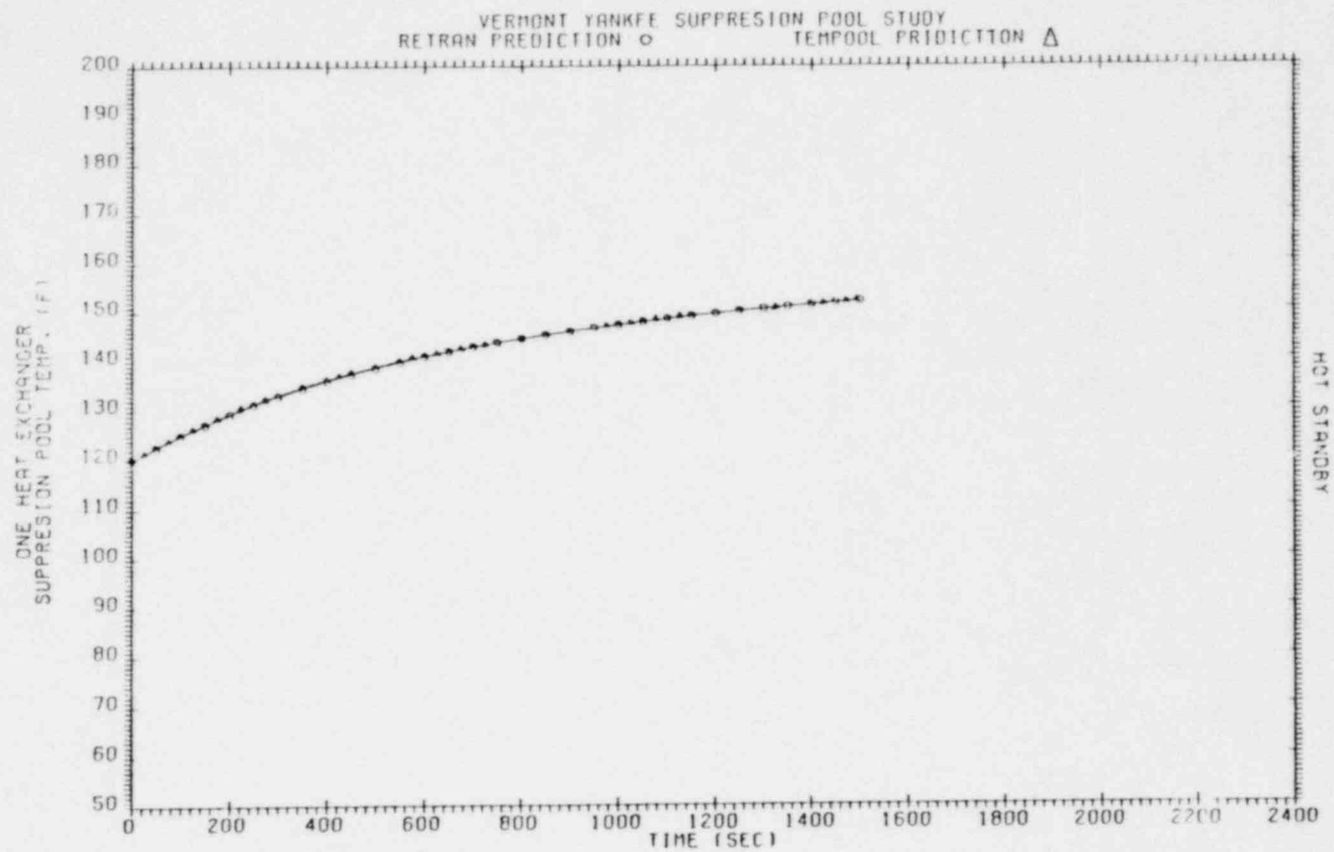

FIGURE IV.2-2 Temperature Prediction With One Heat Exchanger, RETRAN Versus TEMPOOL.

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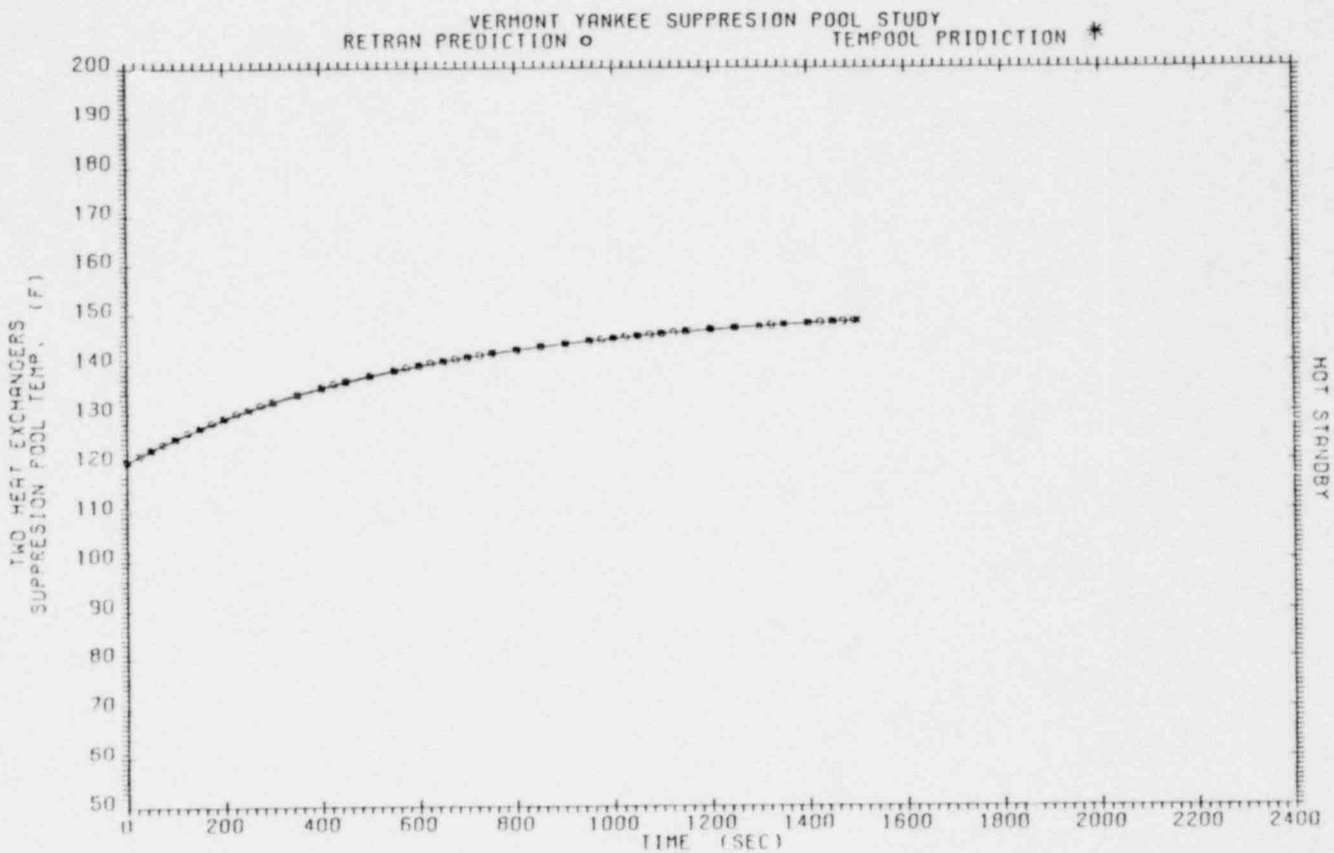

FIGURE IV.2-3 Temperature Prediction with Two Heat Exchangers, RETRAN Versus TEMPOOL.

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TABLE IV.2-1

SUMMARY OF RETRAN VS. TEMPOOL RESULTS

Description RETRAN Runs Hot Standby	Temperature Prediction By TEMPOOL (°F)	RETRAN Model Temperature Prediction (°F)	T Difference Between RETRAN&TEMPOOL (°F)	Max Time Step Size RETRAN (Secs)
Run #1 One Heat Exchanger on @ 0.0 sec	151.838	151.849	.011	10.0
Run #2 Two Heat Exchangers on @ 0.0 sec	148.223	148.234	.011	10.0
Run #3 One Heat Exchanger on @ 0.0 sec	--	151.943	--	5.0
Run #4 Two Heat Exchangers on @ 0.0 sec	--	148.321	--	5.0
Run #5 One Heat Exchanger on @ 0.0 sec	--	152.044	--	1.0

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2.2 Spent Fuel Pit Thermal-Hydraulic Analysis Model

The second system effects analysis conducted by Yankee Atomic was based on a RELAP4 analysis of the Yankee Rowe spent fuel pit. The original study with RELAP4 was designed to determine the adequacy of natural circulation flow to cool the fuel bundle furthest from the down flow region of the pool. Hand calculations performed subsequent to the RELAP4 investigation showed an error in the way the code used the enthalpy transport model. The RETRAN analysis was performed to evaluate the improved enthalpy transport model in RETRAN. The study is discussed in further detail in Reference IV.2-3.

2.2.1 Description of Model

The geometric representation of the spent fuel pit is shown in Figure IV.2-4. The model has 32 volumes and 46 junctions. Fifteen core conductors were used to represent the fuel bundles, which were assumed to have experienced peak equilibrium burnup prior to removal from the core. The 15 bundle model was used because it represented the greatest distance which could be achieved between a bundle and a down flow area in the pool.

The analysis was initiated from a zero power, zero flow condition. The power was increased to the specified value over several time steps. This calculation is actually a pseudo-transient, i.e., a steady-state problem is begun from a non-steady-state condition. The solution is achieved by using a transient calculation until steady-state is reached.

2.2.2 Results of Analysis

Typical values of volume and junction enthalpies computed by RETRAN and RELAP4 are given in Table IV.2-2 along with values for a hand calculation. These results demonstrate that the enthalpy transport model in RETRAN is correct, while the RELAP4 model is in error. Typical values of junction flow and junction enthalpy for the two codes are shown in Figures IV.2-5 and IV.2-6. It is noted that different steady-state solutions are achieved for the two codes. Since RELAP4 computes a higher flow than actually occurs, the peak fuel temperature is lower than computed with RETRAN. This is a non-conservative result.

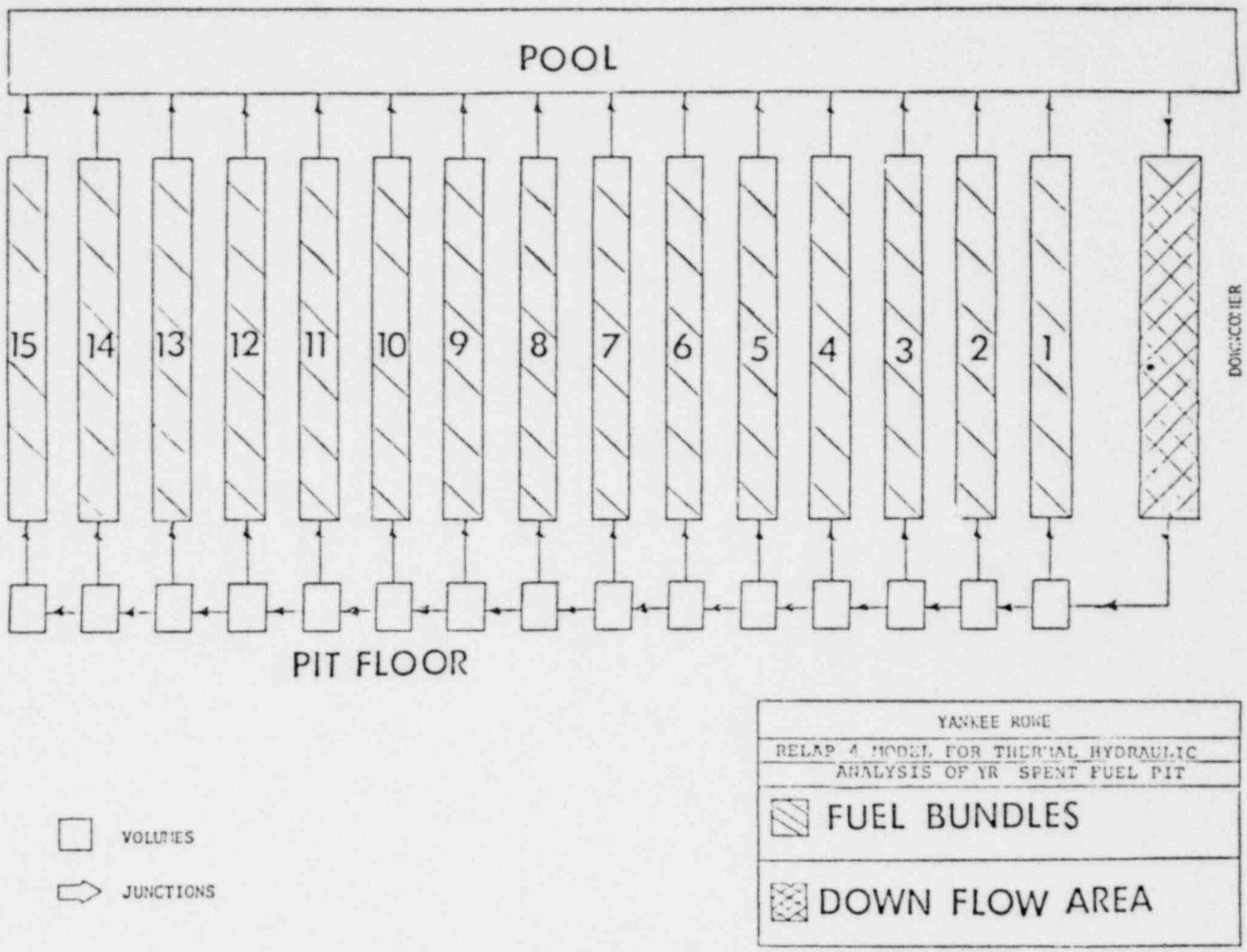

FIGURE IV.2-4 RETRAN Model for Thermal-Hydraulic Analysis of Yankee Rowe Spent Fuel Pit.

9 184

TABLE IV.2-2

COMPARISONS OF VOLUME AVERAGE ENTHALPY CALCULATIONS*
FOR CONTROL VOLUMES

	Inlet Junction Enthalpy (Btu/lb.)	Outlet Junction Enthalpy (Btu/lb.)	Volume Average Enthalpy (Btu/lb.)	Hand Calculation Volume Average Based on Junction Enthalpies (Btu/lb.)	Code Prediction Vs. Hand Calculation % Δ
RETRAN 12B	98.0257	137.571	118.334	118.334	0.45
RELAP4 - MOD3	98.0261	128.277	128.271	113.15	13.36

In steady state conditions the volume average enthalpy can be approximated by:

$$\bar{h}_{av} = \frac{\text{Inlet Junction Enthalpy + Outlet Junction Enthalpy}}{2} \quad \frac{\text{Btu}}{\text{lb.}}$$

The hand calculations were based on this equation.

* These calculations are based on the computer output for "Yankee Rowe Spent Fuel Pit Thermal-Hydraulic Analysis Model" at the time 200.2 seconds when steady-state conditions are achieved.

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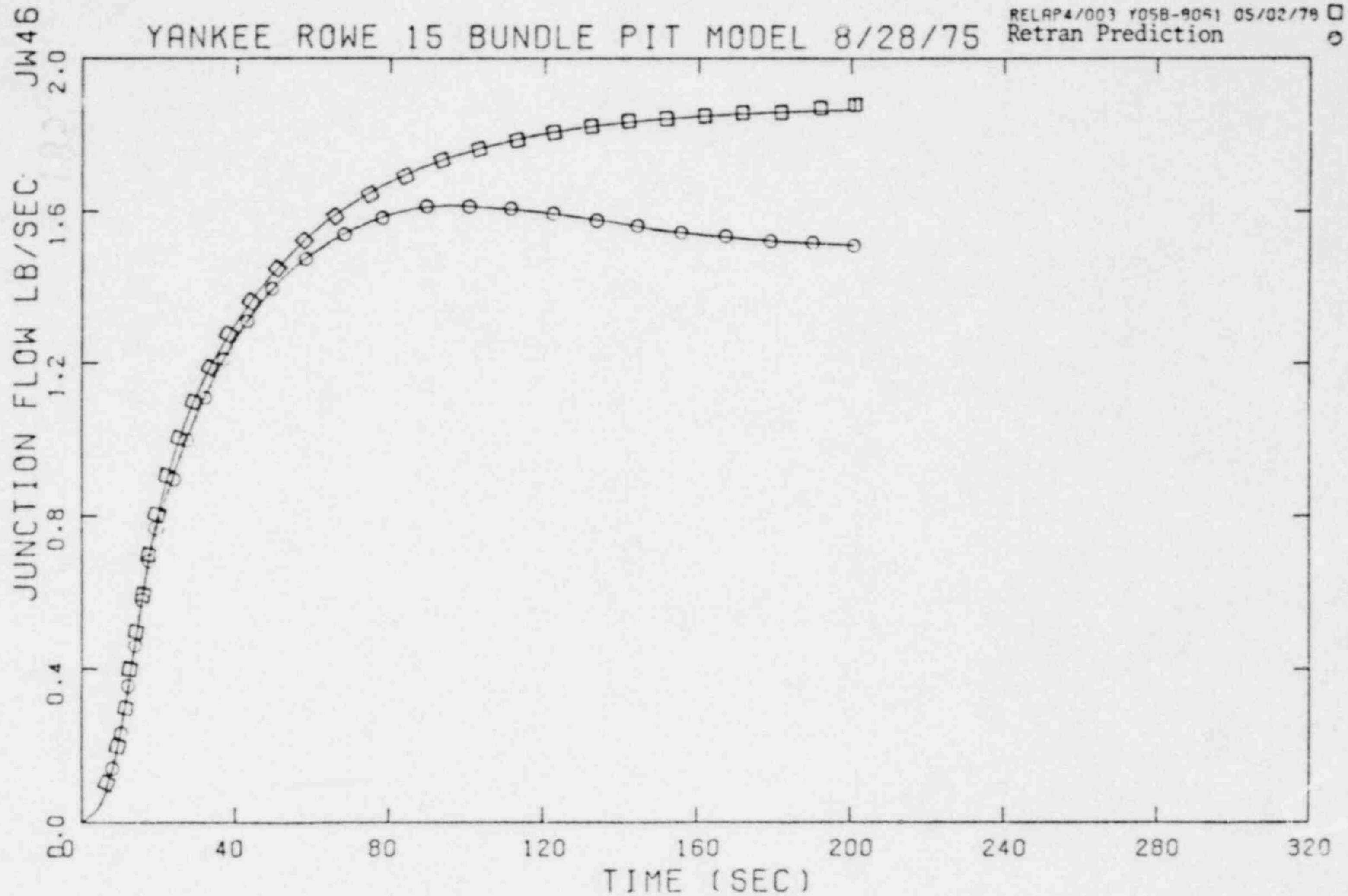

FIGURE IV.2-5 RELAP4 and RETRAN Comparison of Junction Flow from the Furthermost Bundle.

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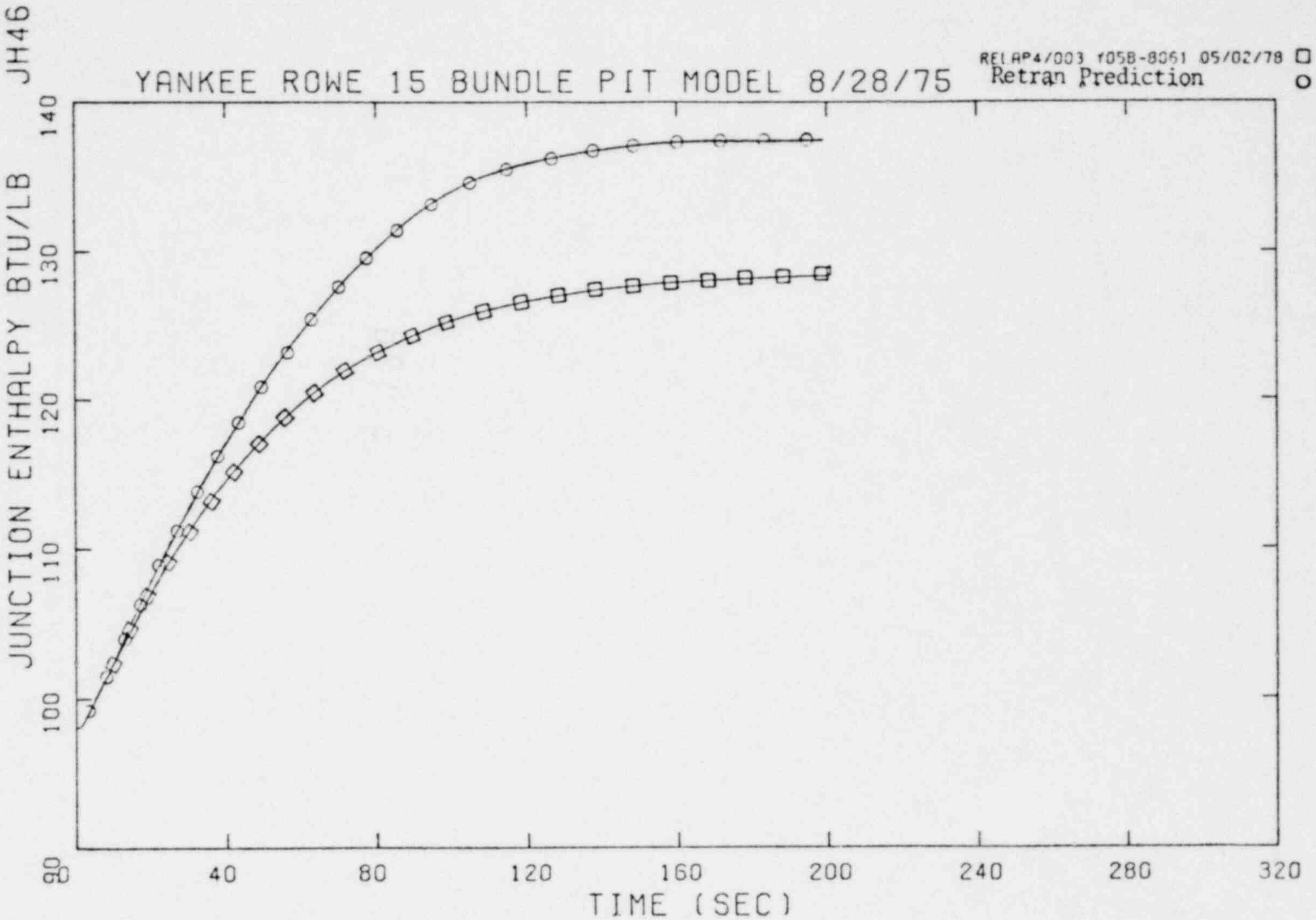

FIGURE IV.2-6 RELAP4 and RETRAN Prediction of Junction Enthalpy from the Furthermost Fuel Bundle. (Note: This Enthalpy Determines the Maximum Pool Temperature.)

2.3 Summary of Results

These analyses provide additional qualification for the use of RETRAN on small system problems. An additional observation of the suppression pool study is related to the boundary condition at the fill. This boundary condition was determined from a separate RETRAN analysis. Ansari described the value of this approach with the following words, "The decoupling between the two models allowed a number of parametric studies to be performed on the smaller model of the Suppression Pool, which is relatively inexpensive to run."[IV.2-1]

Two conclusions can be drawn from the spent fuel pit analysis. Of immediate interest is the verification that the enthalpy transport model in RETRAN is correct. The second conclusion is really a practical lesson for all code users. Although RELAP4 produced results which were not unreasonable, they were in error as determined by a simple hand calculation. In large codes, errors can occur. Experienced code users recognize this and review the results carefully, especially for new or unconventional applications.

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3.0 TWO-LOOP TEST APPARATUS

The Two-Loop Test Apparatus (TLTA) is operated by General Electric to produce experimental data for use in analyzing BWR blowdown transients. The TLTA is a scaled simulation of a two-loop BWR with an electrically heated 7x7 core bundle. The scaling and design of the TLTA was not intended as a direct one-to-one performance of a BWR but only to reflect a system blowdown response similar to a BWR on a real time basis. The 1:560 scaling of the TLTA was based on the fact that the reference BWR has 560 fuel rod bundles with each bundle containing a 7x7 array of nuclear fuel rods. The TLTA core consists of one heater rod bundle containing a 7x7 array of electrical heater rods for simulation of the nuclear fuel rods. Volume, mass, energy and flow rates are scaled on a 1:560 basis in the TLTA.

Test 4906 was designated as Standard Problem Four by the NRC. Energy Incorporated has evaluated this test and 3 others for EPRI using a RELAP4 code. A detailed discussion of these analyses is given in Reference IV.3-1. Yankee Atomic used the basic RELAP4 model for Standard Problem Four and evaluated the test with RETRAN. These results[IV.3-2] are discussed in the following section.

3.1 Yankee Atomic Analysis

The input model used by Yankee Atomic in this analysis was developed as part of the study in Reference IV.3-1. Yankee Atomic used this deck for a RETRAN analysis and compared the results with the RELAP4 calculations and the experimental data.

3.1.1 Description of Model

The geometric representation of the TLTA as modeled for RETRAN is shown in Figure IV.3-1. The model has 29 volumes, 36 junctions and 38 heat conductors. The reactor core was modeled with 8 volumes which corresponded to thermocouple locations. There were three volumes for the intact recirculation loop while the broken loop was modeled with 4 volumes. Additional detail of the system model is given in References IV.3-1 and IV.3-2.

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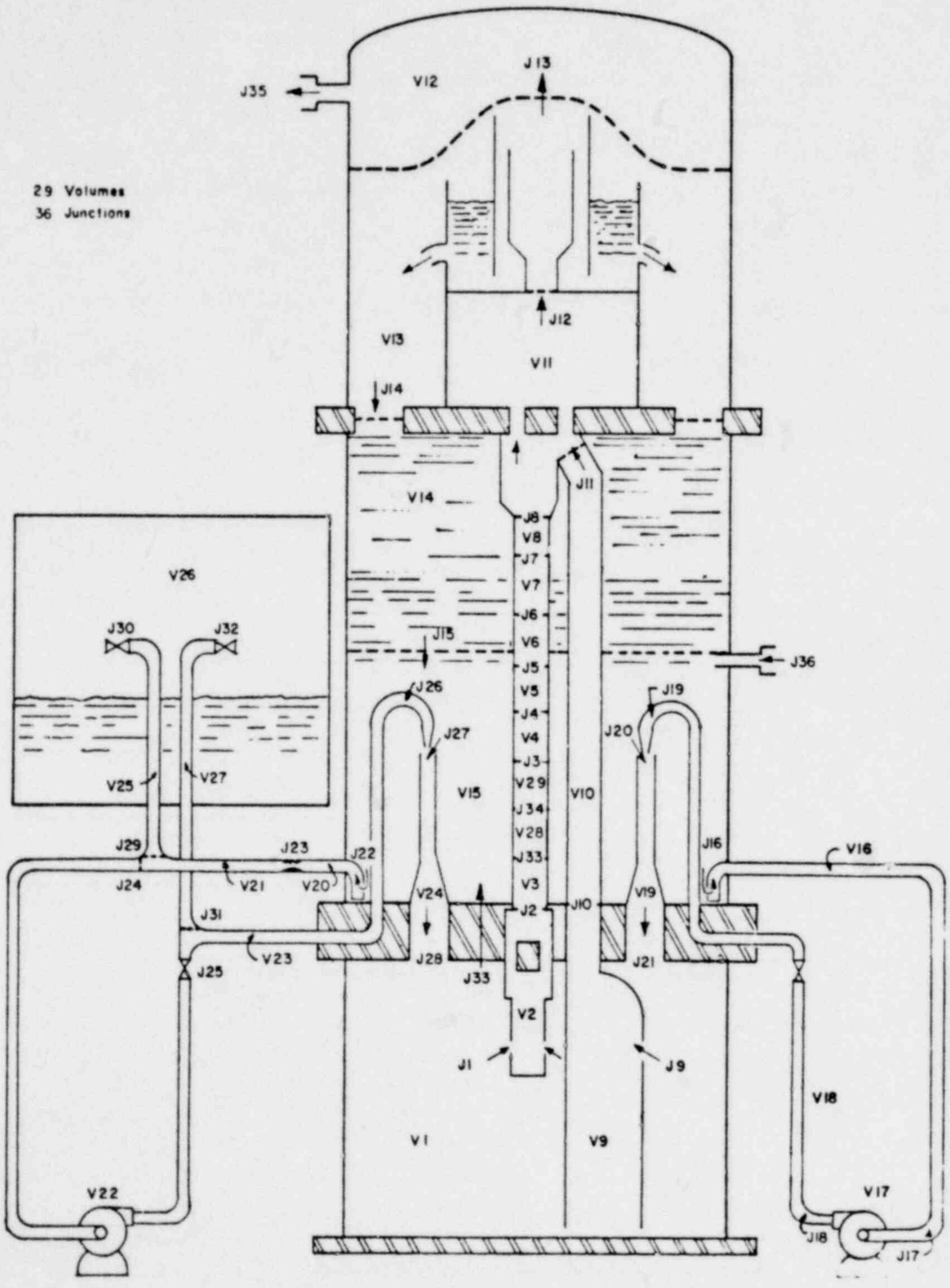

FIGURE IV.3-1 Model for Standard Problem Four.

3.1.2 Results of Analysis

Selected results from the Yankee Atomic analysis were compared with RELAP4 calculations and experimental data. The steam dome pressure during the transient is shown in Figure IV.3-2. The calculated pressure values agree quite well with the data through the first 17 seconds of the transient, after which they fall below the data. This trend may be due to the break flow model and/or the jet pump response in the broken loop.

Uncovery times for the jet pump suction and recirculation line suction are given in Table IV.3-1. These comparisons also indicate that the calculated break flow early in the transient is greater than was observed experimentally. The core inlet flow (Figure IV.3-3) computed with RETRAN agrees quite well with the data except for the first part of the transient when the computed values are greater than the experimental values. This probably delays the time for CHF to occur (Figures IV.3-4 and IV.3-5). After CHF occurs, RETRAN and RELAP4 both show a more rapid rate of increase in the clad temperature than was observed experimentally.

3.3 Summary of Results

The RETRAN calculations agree quite well with the RELAP4 results for this problem. Both codes give reasonably good agreement with the data. However, the critical flow models and jet pump models should be studied further based on the results given here and in References IV.3-1 and IV.3-2.

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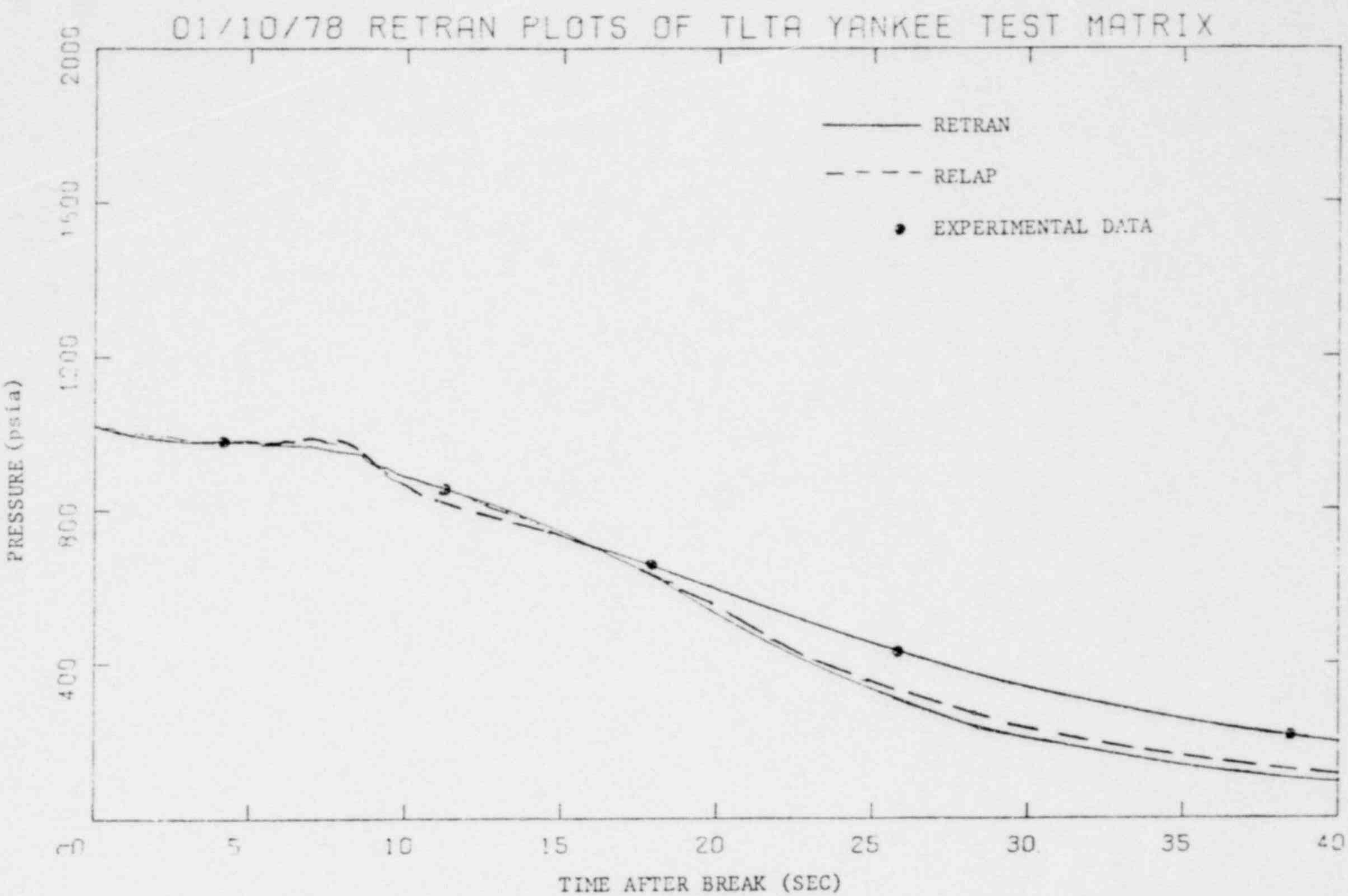

FIGURE IV.3-2 Steam Dome Pressure - Test 4906.

TABLE IV.3-1

JET PUMP UNCOVERY TIME COMPARISONS

TEST 4906

120% BREAK SIZE

Event	RETRAN	RELAP4	G. E. Experimental
Jet Pump Suction (sec)	6.91	7.18	7.24
Recirculation Line Suction (sec)	8.46	8.70	9.71
Jet Pump to Recirculation Line Time Difference (sec)	1.55	1.52	2.47

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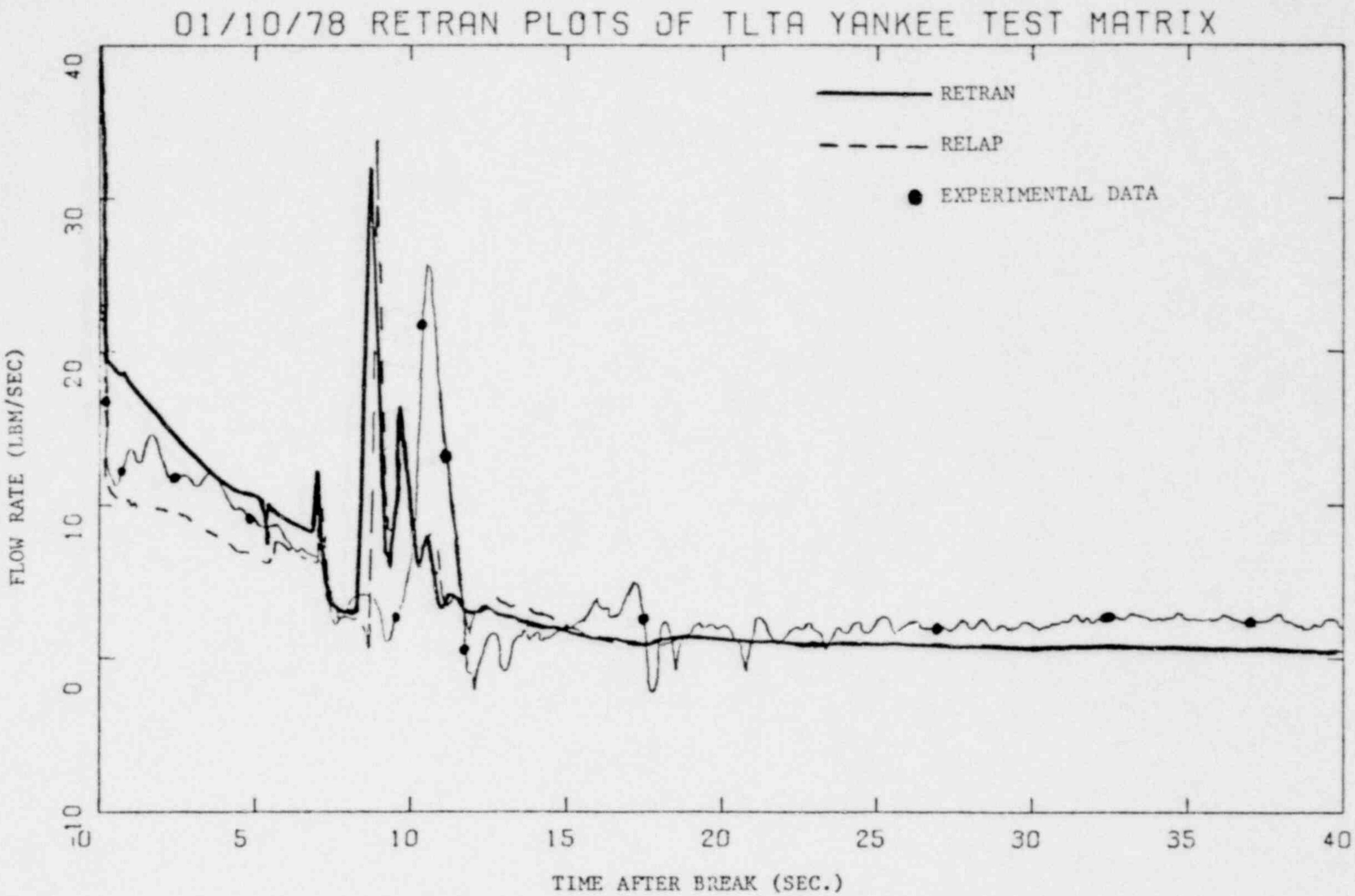

FIGURE IV.3-3 Core Inlet Flow - Test 4906.

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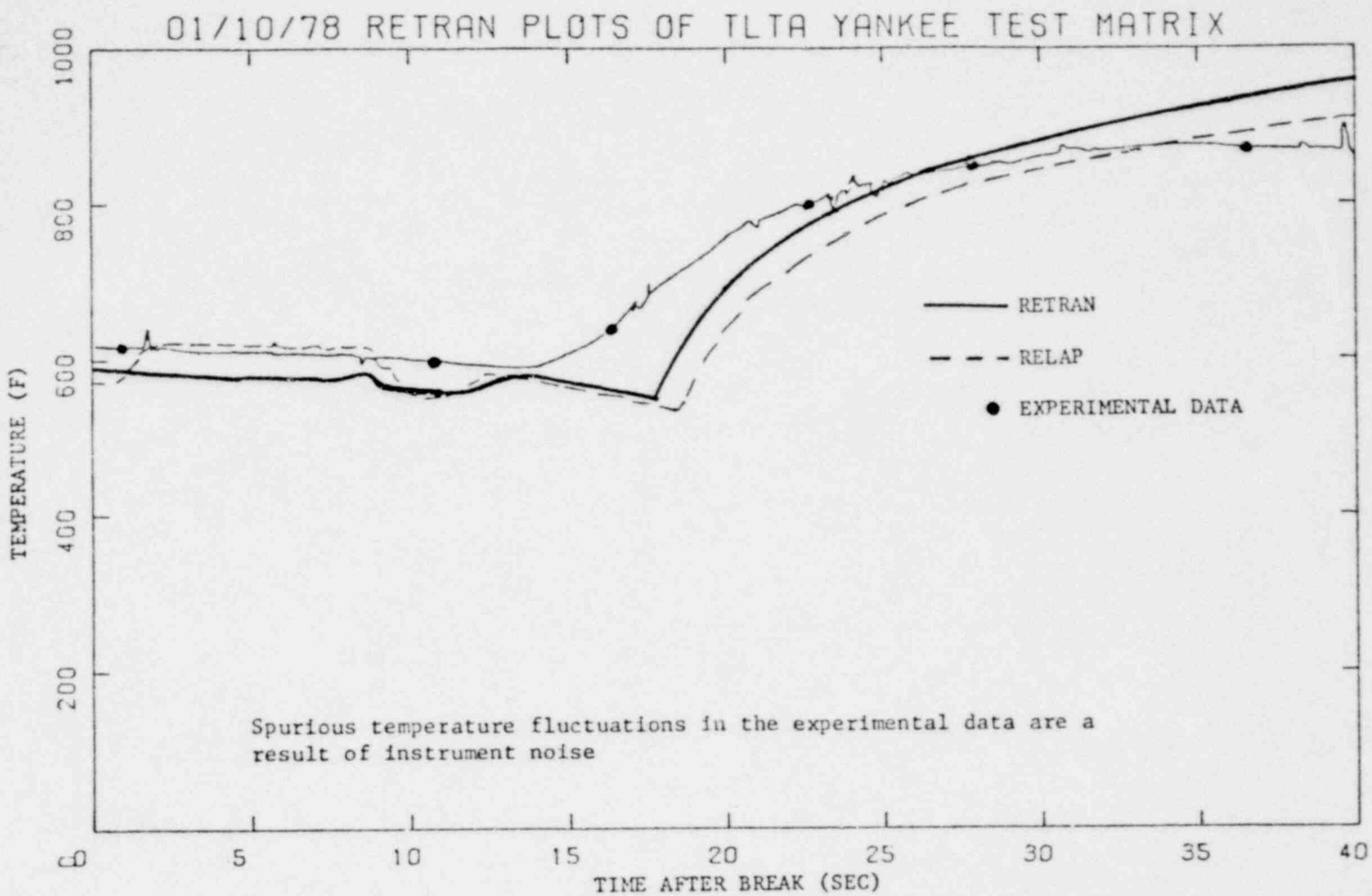

FIGURE IV.3-4 Hot Rod Temperature , Elevation 78 Inches - Test 4906.

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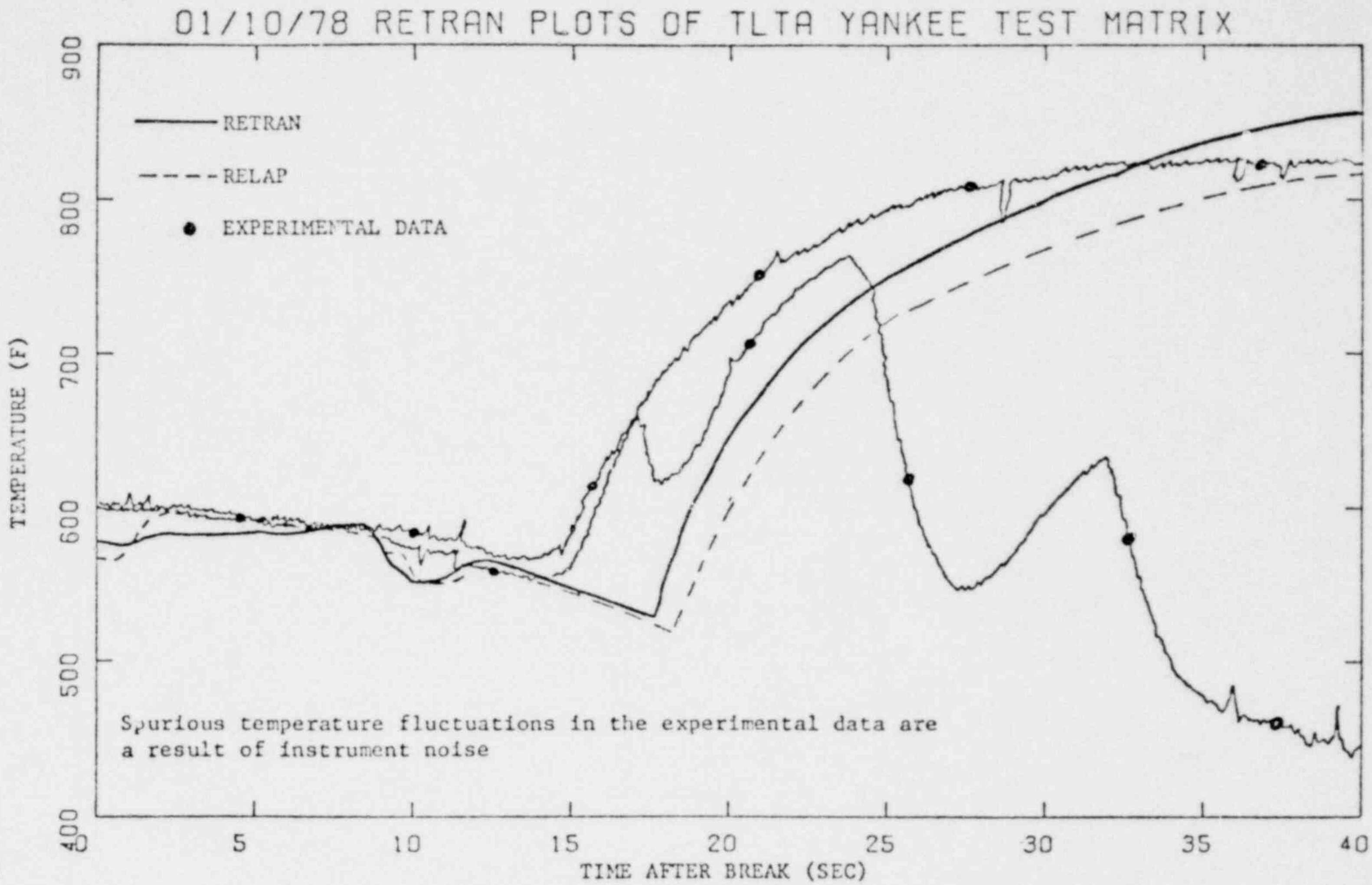

FIGURE IV.3-5 Average Rod Temperature, Elevation 98 Inches - Test 4906.

4.0 SEMISCALE 1000 SERIES TESTS

The Semiscale test system is operated as part of the NRC sponsored nuclear safety program. The 1000 Series of tests were performed with a 1-1/2 loop system, and can be generally categorized as isothermal blowdowns. Two of these tests were designated as standard problems for the NRC CASP program. Standard Problem Two was based on Test 1011, and is discussed in Reference IV.4-1. Test 1009 was identical to Test 1011 except that an emergency core cooling system was added. This test was Standard Problem Three and is discussed in Reference IV.4-2.

Energy Incorporated performed "best estimate" calculations for these standard problems for EPRI. The analyses [IV.4-3] were performed with a RELAP4 code. Some members of the Utility Working Group have planned additional analyses for these tests with RETRAN. The results of these calculations will be documented in this section.

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5.0 SEMISCALE S-02 TEST SERIES

The Semiscale MOD-1 test facility is designed to provide data for evaluating reactor safety models applied to hypothetical LOCA analysis. The tests performed during the Series 2 phase of the project were blowdown heat transfer tests with power supplied from an electrically heated core. These tests were performed with an active steam generator in the intact loop. The broken loop included a simulated steam generator and pump.

5.1 Semiscale Test S-02-8

The NRC selected Test S-02-8 to be analyzed as Standard Problem Five in the CASP program. This test was a 200 percent double-ended break from full power. Prior to the break, the coolant was at 2200 psi and 600°F. There was no ECC injection for this test.

An analysis of Standard Problem Five was performed by Energy Incorporated for EPRI. This analysis was done with a RELAP4 code and is reported in Reference IV.5-1. Mirsky[IV.5-2] used the model developed for the RELAP4 analysis and used RETRAN to evaluate this problem. The model and the results of this analysis are discussed in the following sections.

5.1.1 Description of Model

The RETRAN model for Standard Problem Five has 45 volumes and 47 junctions (Figure IV.5-1). There also are 34 conductors, one heat exchanger, one pump and two valves. Two of the junctions are fill junctions, and are used to model the flow into and out of the steam generator secondary side. Detailed information on the model is given in References IV.5-1 and IV.5-2.

This analysis employed a valve in the intact loop which was used to modify the critical flow multiplier. The critical flow model used was a combination of Henry (subcooled) and Moody (saturated). To achieve a transition at the time of saturation, a trip was set to reduce the vessel side valve flow area to 60 percent of its fully open area. This has the effect of using a value of 0.6 for the multipier on the Moody tables.

Mod-1 Semiscale Model For Test S-02-8 (Standard Problem 5)

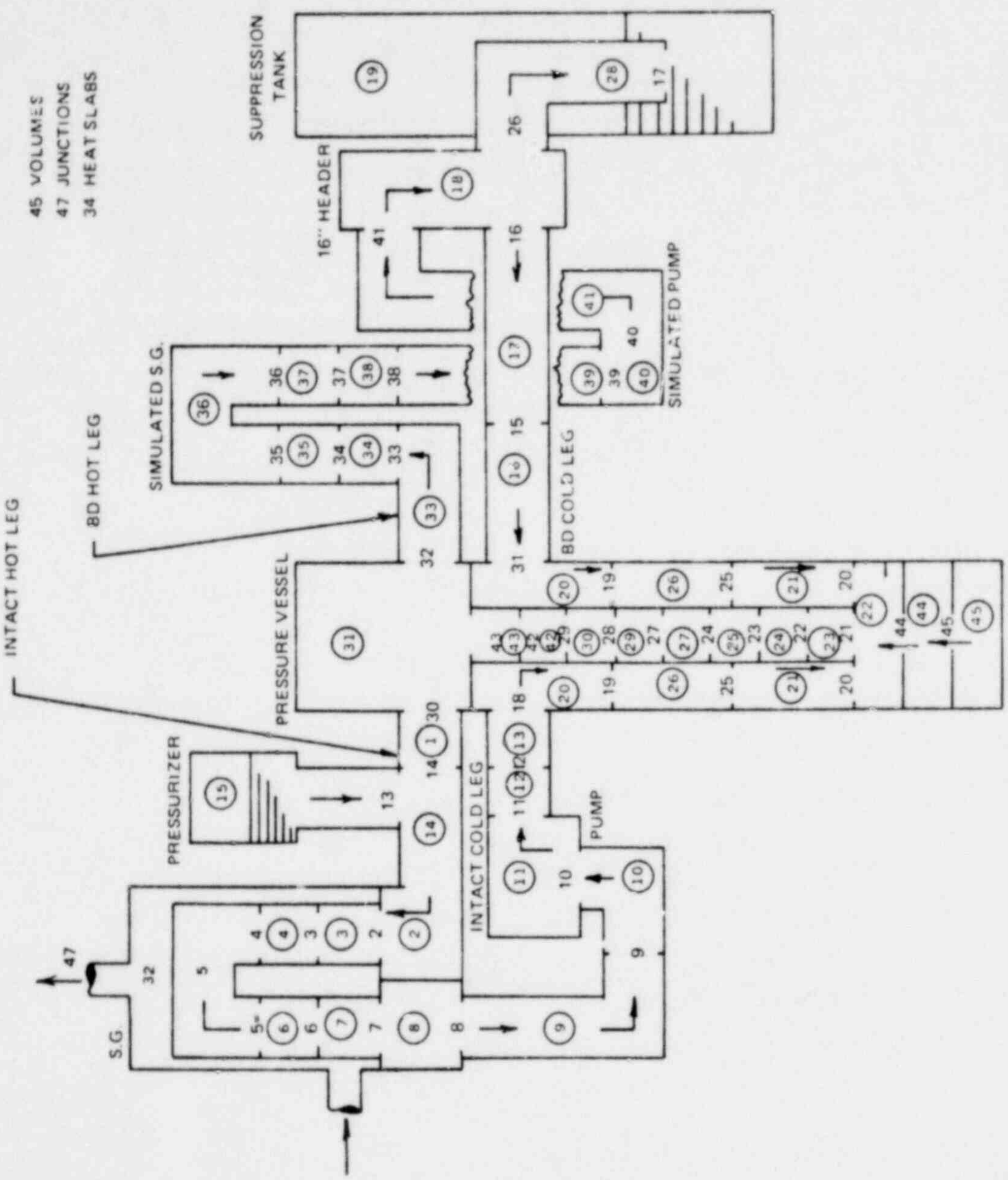

FIGURE IV.5-1 RETRAN Model for Semiscale Test S-02-8 (Standard Problem Five).

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5.1.2 Results of Analysis

The RETRAN analysis of Standard Problem Five was performed for 25 seconds of transient time. The values of pressure in the upper plenum are shown in Figure IV.5-2. The RETRAN results agree quite well with those computed by RELAP4 and the data. Flow rates in the broken loop are given in Figures IV.5-3 and IV.5-4, with the RETRAN values in good agreement with the experimental data. The fluid temperature in the upper plenum is given in Figure IV.5-5. The RETRAN results give better agreement with the data than does RELAP4. This is thought to be due to an improved enthalpy transport model in RETRAN.

The measured clad surface temperature 59 inches above the bottom of the core is compared with RETRAN (Version RET09B) and RELAP4 results in Figure IV.5-6, which indicates the RETRAN and RELAP4 values are in poor agreement. This run was performed with the RET09B version of the code. An error in the CHF routine was discovered after this run was made. The analysis was subsequently repeated with the RET12B code version, which included corrections in the CHF routine. As shown in Figure IV.5-6, the temperature values computed by RETRAN and RELAP4 are in good agreement. The released version of RETRAN includes the CHF corrections of version RET12B.

5.1.3 Summary of Results

The RETRAN analysis of Standard Problem Five produced results which were in generally good agreement with the experimental data. An exception to this were the pin temperatures computed with version RET09B, which disagree with both the data and RELAP4 results. This discrepancy was resolved after an error was identified and corrected in the CHF calculation. This is an example of the value of the pre-release activity relative to identification of coding errors.

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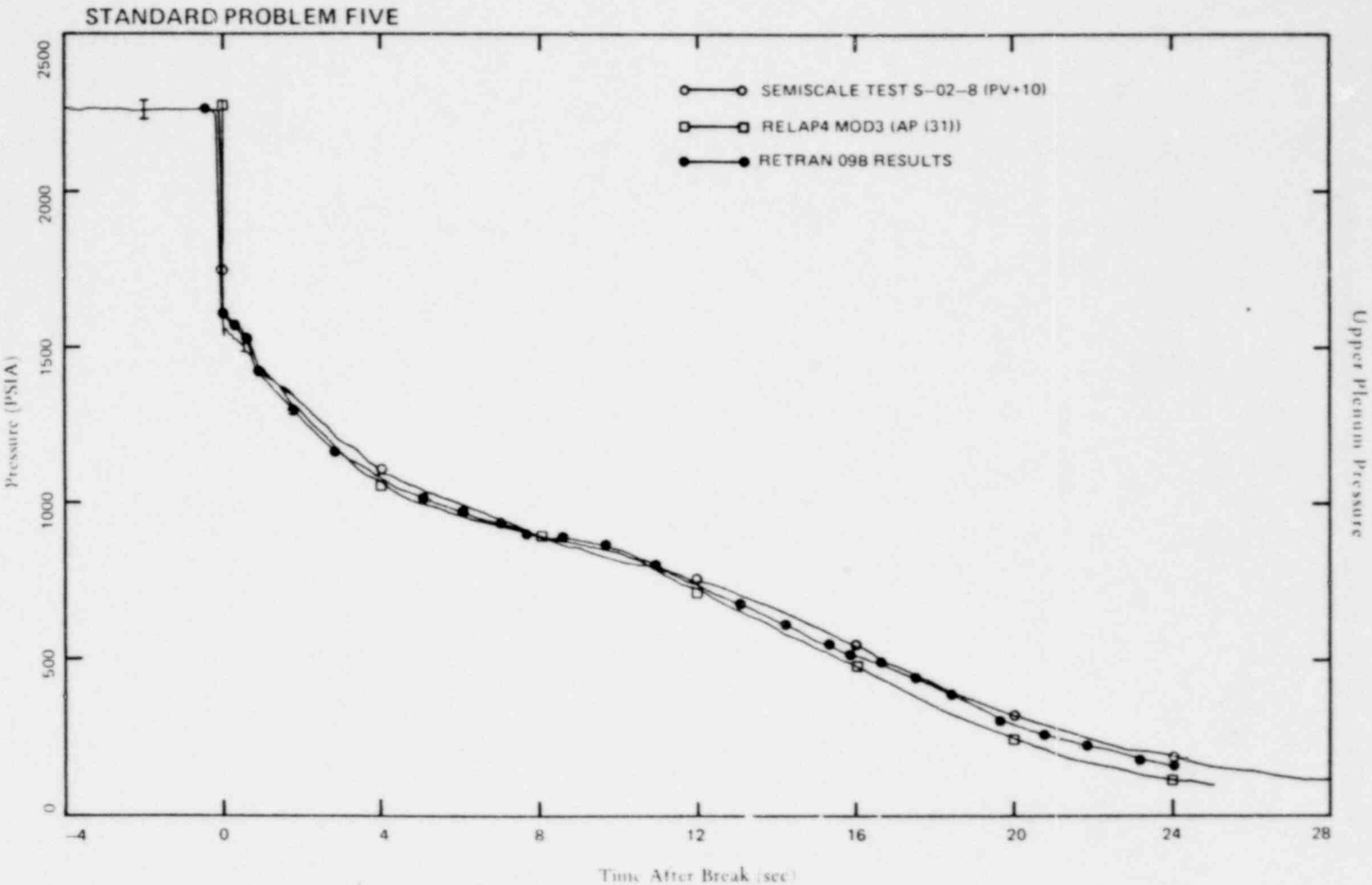

FIGURE IV.5-2 Upper Plenum Pressure.

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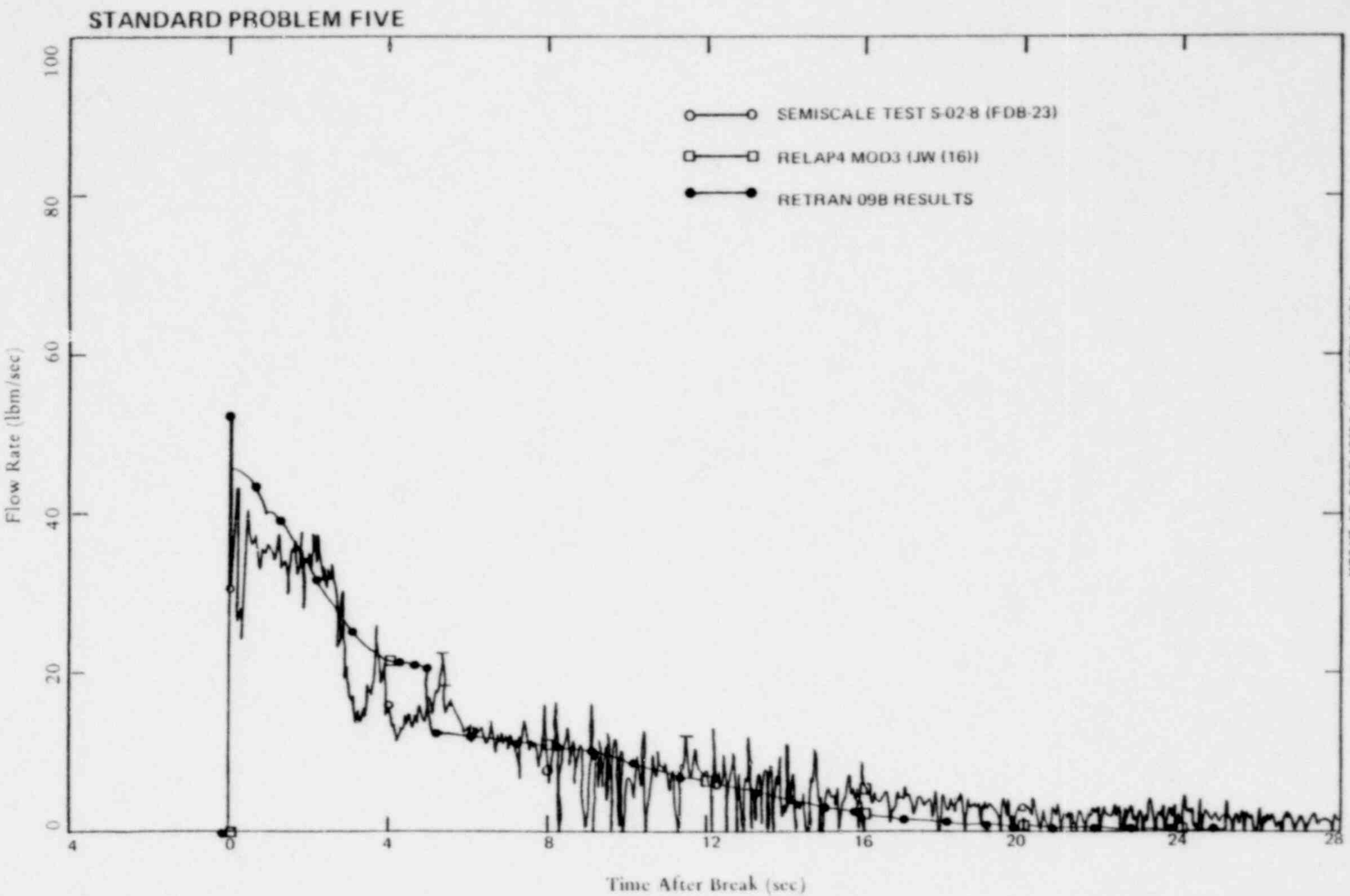

FIGURE IV.5-3 Flow Rate - Vessel Side of Break.

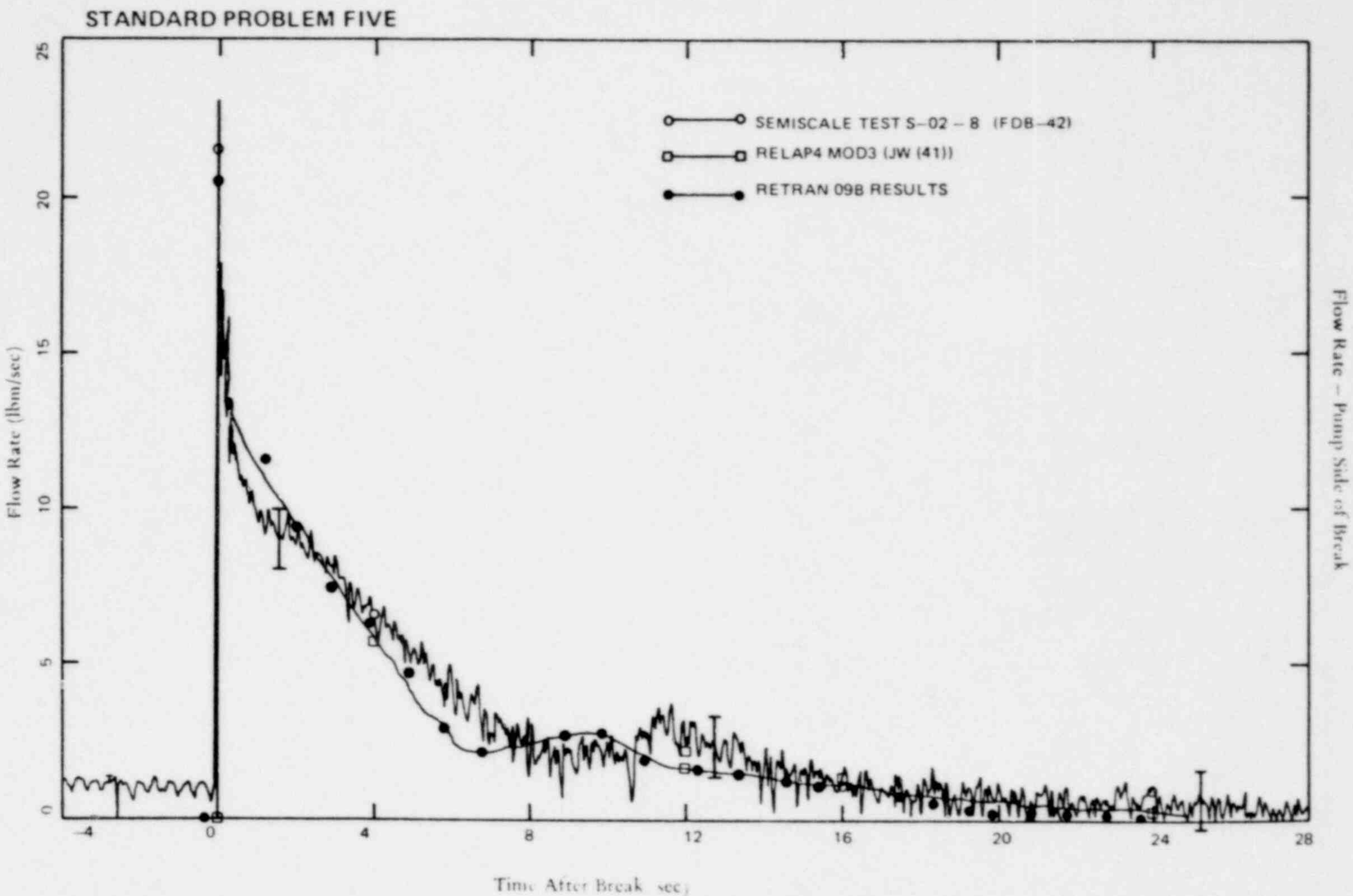

FIGURE IV.5-4 Flow Rate - Pump Side of Break.

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Fluid Temperature in Upper Plenum

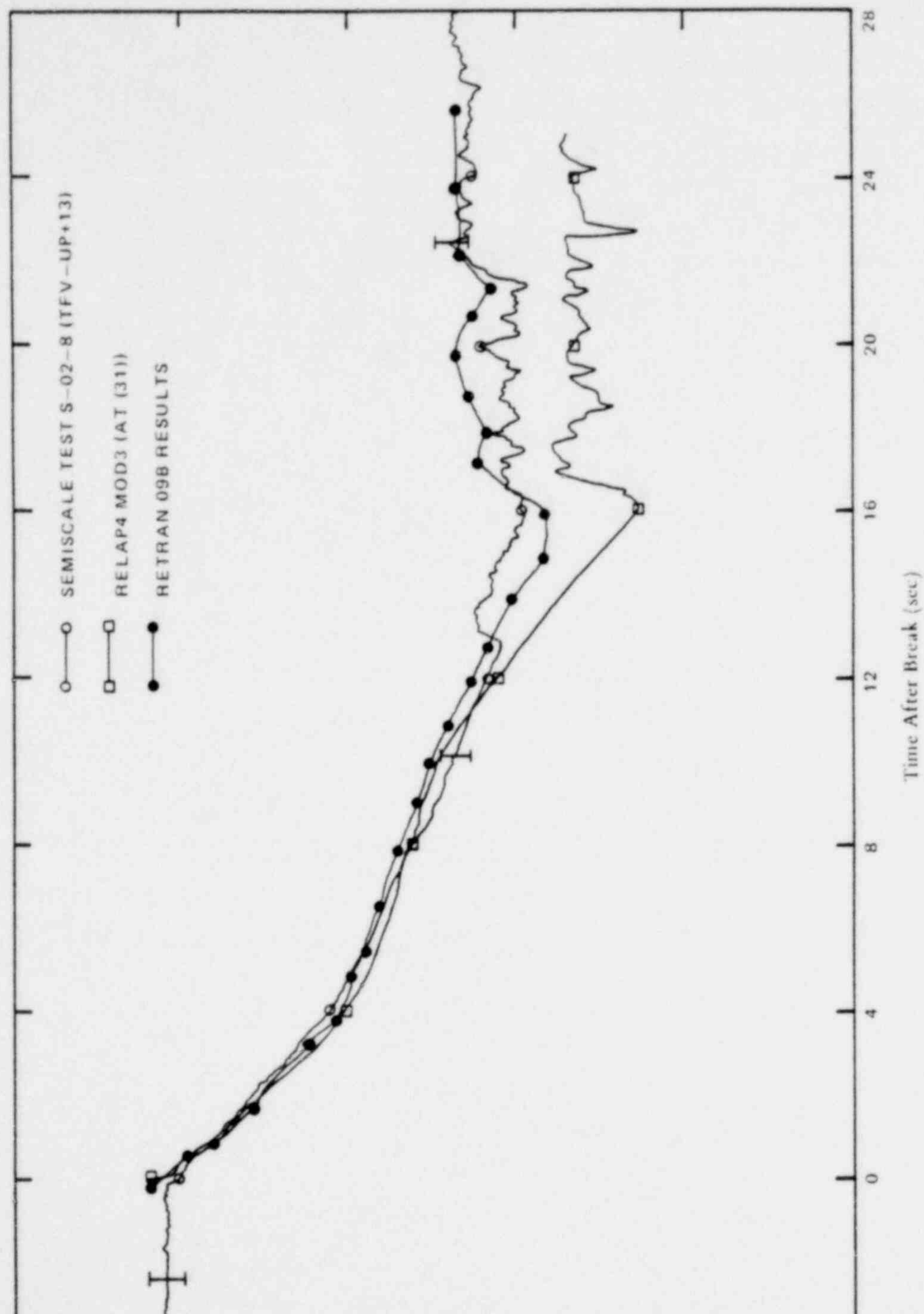

FIGURE IV.5-5 Fluid Temperature in Upper Plenum.

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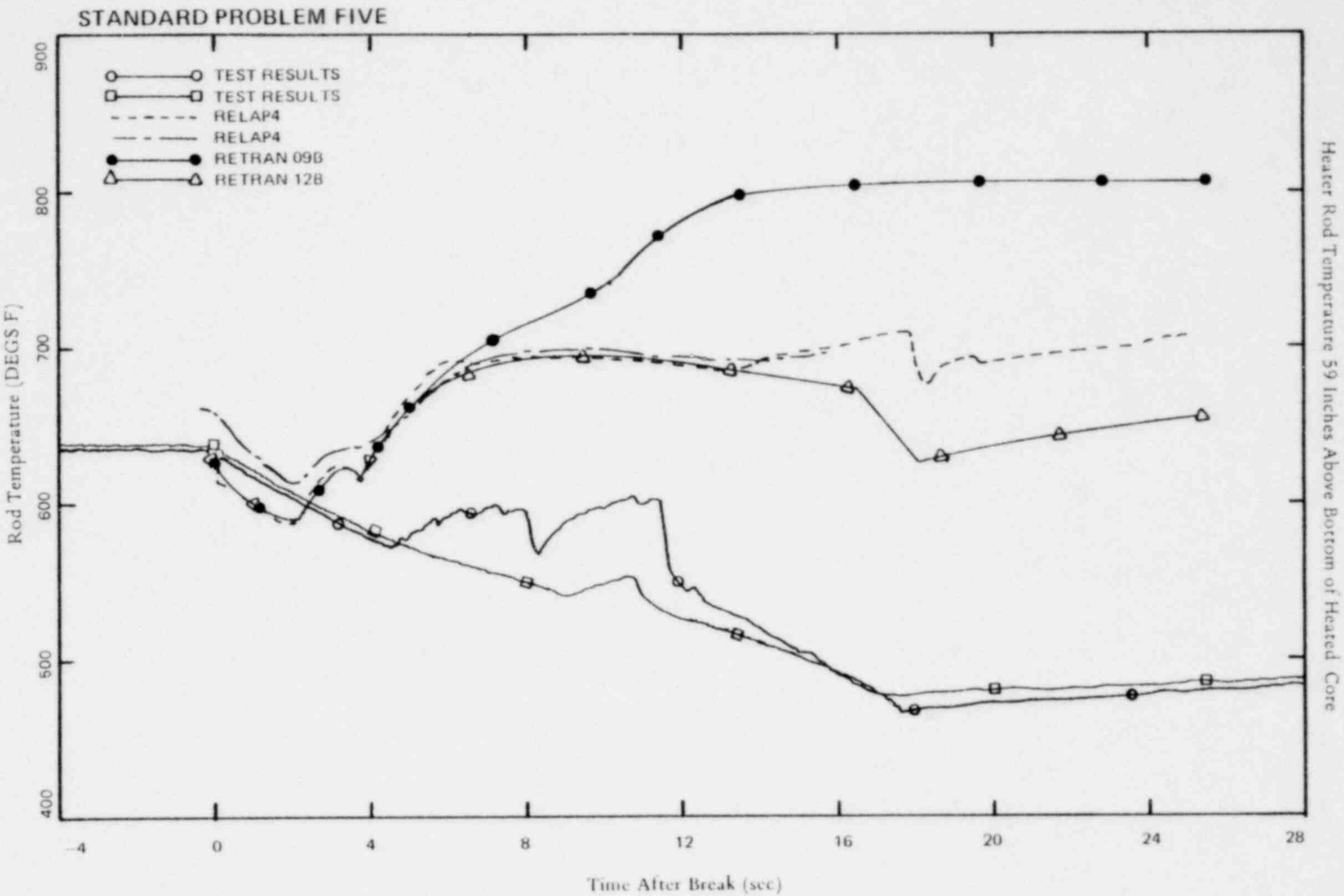

FIGURE IV.5-6 Heater Rod Temperature 59 Inches Above Bottom of Heated Core.

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V. BOILING WATER REACTOR TRANSIENT ANALYSES

V. BOILING WATER REACTOR TRANSIENT ANALYSES

The analyses presented in this section are the results of modeling transients which can be anticipated to occur for boiling water reactors (BWR). The approach used by General Electric to classify and model these transients is discussed in Reference V-1. A summary of the BWR transient classification is given in Table V-1. When applicable, the analyses presented in this section are discussed in terms of the categories in Table V-1.

1.0 TURBINE TRIP

The turbine trip transient is classified as a pressure increase transient and can occur with and without bypass flow. As discussed in Reference V.1-1, the turbine is tripped and the bypass valves either open to relieve the pressure increase or fail to operate, depending on the assumptions made. The scram signal is received from a position switch on the turbine stop valve. The trip occurs when this valve is 10 percent closed.

The transient is initiated by closure of the turbine stop valves. Once these valves are closed, the steam flow leaving the vessel decreases. Since the core is continuing to generate power with a reduced steam flow, the reactor vessel pressure increases. The pressure continues to increase until the safety relief valves open. This rise in pressure causes a reduction in the core voids which results in a core power increase. The power continues to rise until the new voids generated by the higher power, the Doppler reactivity feedback, and the scram reactivity feedback override this positive effect and begin to reduce the power. The rise in core power is followed by a rise in fuel center temperature and fuel rod surface heat flux. This results in a decrease in the minimum departure from nucleate boiling ratio (MDNBR).

The turbine trip is one of the BWR transients scheduled for evaluation by all members of the Working Group modeling BWR's. This transient, without bypass flow, has also been analyzed with a preliminary version of the RETRAN code [V.1-1], but those results are not discussed in this report. Detroit Edison has analyzed this transient for the Fermi-2 reactor [V.1-2] and EPRI [V.1-3 to V.1-5] has modeled the turbine trip transient tests conducted at the Peach Bottom-2 facility. These analyses are discussed in the following sections.

TABLE V-1

SOME BWR SAR-CHAPTER 15 TRANSIENTS

Transient	Condition			System Analysis Required
PRESSURE INCREASE TRANSIENTS				
° Generator load rejection	II			
° Turbine trip with/without bypass	II			Yes
° Steam line isolation valve(s) closure	II			Yes
° Pressure regulator failure (close)	II			Yes
° Loss of condenser vacuum	II			Yes
° Turbine control valve fast closure		III		Yes
° Above incidents with delayed scram		III		Yes
REACTOR VESSEL WATER TEMPERATURE DECREASE				
° Loss of feedwater heater	II			Yes
° Inadvertent injection pump start	II			Yes
REACTOR VESSEL COOLANT INVENTORY DECREASE				
° Loss of feedwater flow	II			Yes
° Pressure regulator failure (open)	II			Yes
° Relief or safety valve failure (open)	II	III		Yes
° Loss of auxiliary power	II			Yes
CORE COOLANT FLOW DECREASE				
° Recirculation pump seizure		III		Yes
° Recirculation pump(s) trip	II			Yes
° Recirculation flow control failure	II			Yes
CORE COOLANT FLOW INCREASE				
° Recirculation flow control failure	II			Yes
° Startup of idle recirculation pump		III		Yes
POSITIVE REACTIVITY INSERTION				
° Continuous rod withdrawal				
- From subcritical	II			Yes
- From power	II			Yes
° Rod ejection			IV	Yes

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TABLE V-1 (cont'd)

SOME BWR SAR-CHAPTER 15 TRANSIENTS

Transient	Condition	System Analysis Required
ANTICIPATED TRANSIENT WITHOUT SCRAM	III	Yes
STEAM LINE BREAK	IV	Yes
RECIRCULATION LINE RUPTURE		
° Large break	IV	Yes
° Small break	II III	Yes
IMPROPER CORE ASSEMBLY	III	No
ROD REMOVAL ERROR AT REFUELING	IV	No
FUEL HANDLING ACCIDENT	IV	No

9 209

1.1 Fermi-2 Analyses

The Detroit Edison analyses of a TTWOB for Fermi-2 were compared to the FSAR calculations performed by the vendor. In addition to the base case, sensitivity studies of separator inertia, moderator heating, steam bypass flow and scram rates were performed. A detailed discussion of the analyses and plant model is given in Reference V.1-2.

1.1.1 Description of Model

The RETRAN model of the Fermi-2 plant is shown in Figure V.1-1. The model has 23 volumes and 32 junctions. Each of the two recirculation loops includes a lumped representation of 10 jet pumps. The four steam lines are lumped into a single line with four volumes based on an earlier study [V.1-1]. The steam separator was modeled as a single volume with an inlet junction from the upper plenum and two outlet junctions, one to the downcomer and one to the steam dome. The reactor core was modeled with five volumes and the core bypass flow was represented by one volume. Five conductors were used to model the heat source in the core.

Three sets of safety relief valves were included in the model, each with different set points for pressure. Flows through these valves were modeled by negative fills, as was the steam flow to the turbine. The feedwater flow was assumed to be constant during the transient.

1.1.2 Results of Analyses

Eight studies are reported in Reference V.1-2. Since no data are available, comparisons of RETRAN results were made only with vendor calculations. With respect to the sensitivity studies, "...qualitative comparisons were made only to assure that no major errors exist in the RETRAN code and/or input models"[V.1-2]. The self-initialization option in RETRAN was used to compute the steady-state conditions for an initial power level of 104 percent of nominal. The steam flow was 105 percent of normal, and all other input values were for normal operating conditions.

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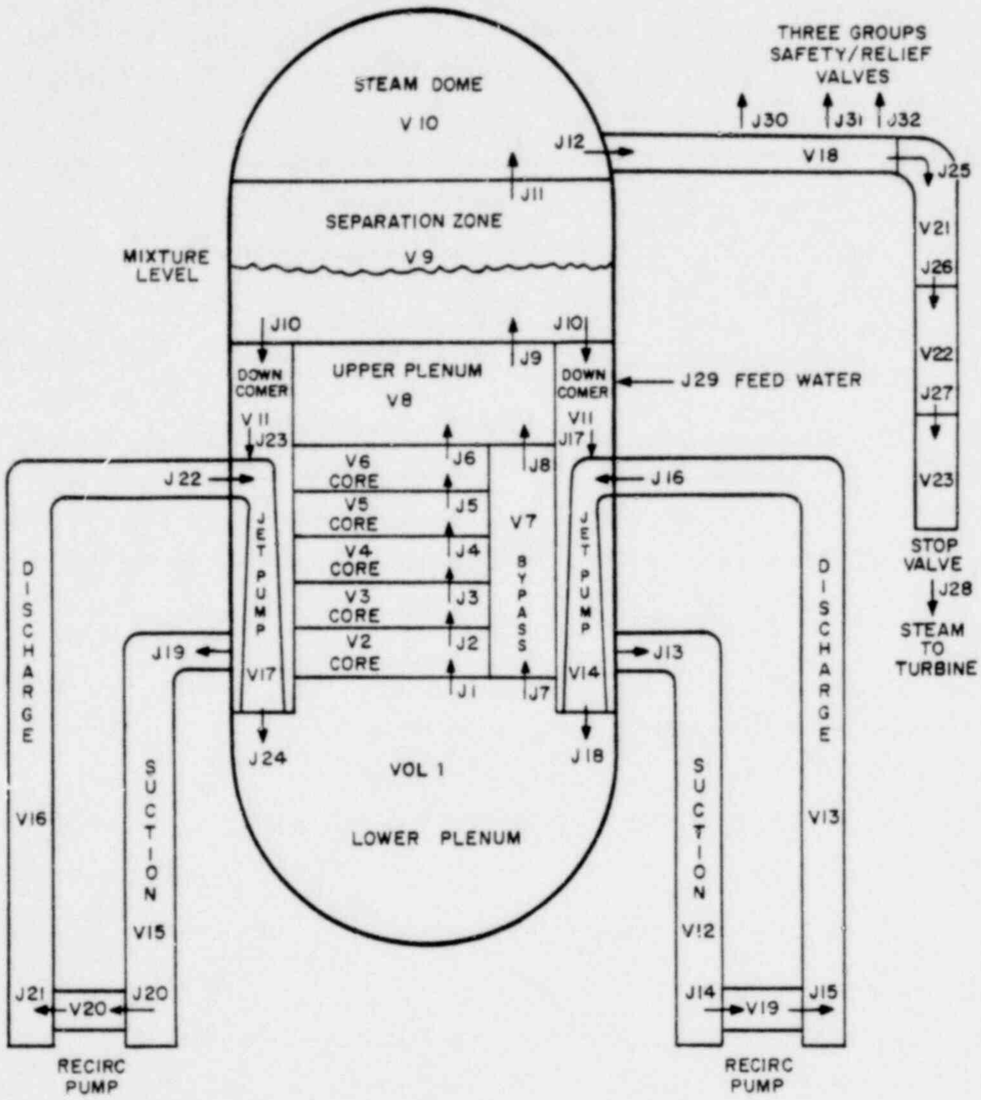

RETRAN DYNAMIC MODEL FOR FERMI-2
TURBINE TRIP PRESSURE TRANSIENT STUDIES

FIGURE V.1-1 RETRAN Model for Fermi-2 Turbine Trip Pressure Transient Studies.

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1.1.2.1 Base Case Analyses

For this case, the input parameters were kept as close as possible to the values used in the vendor analyses. The system power and steam dome pressure are shown in Figures V.1-2 and V.1-3. Both the peak values and the time these values are achieved agree reasonably well with the vendor results. The input for the second base case was the same as for the FSAR comparison with the exception that a nominal scram curve and void reactivity coefficient were used. The system power and steam dome pressure for this run are shown in Figures V.1-4 and V.1-5. The main difference between these two runs is seen in the peak power value, which is significantly lower in the nominal base case than in the FSAR comparison case.

1.1.2.2 Sensitivity Studies

Results of six sensitivity studies were reported and are summarized in Table V.1-1. Input parameters for these runs are given in Table V.1-2. The inertia was varied in an attempt to determine an appropriate value to use for modeling the separator in transient analyses. The results tabulated for case 4 reflect the effect of a 10 percent steam bypass flow which is specific for the Fermi-2 reactor.

Two runs (Runs 6 and 7) were made to study the effect of prompt moderator heating. For the first case, no moderator heating was accounted for, while in the second case, it was weighted by the initial density for each core volume. The results of these runs were quite close to the nominal base case (Run 2).

The final run reported was performed in an attempt to incorporate features unique to Fermi-2 into the model. The most significant changes [Table V.1-2] are the 10 percent steam bypass flow and a turbine stop valve closure time of 0.22 seconds. With these assumptions, the peak power is significantly lower than the base case analyses.

1.2 EPRI Analyses

Three turbine trip tests were conducted in April 1977 at the Peach Bottom-2 reactor. These tests were sponsored by Philadelphia Electric Company, EPRI and General Electric. The tests were conducted with initial power levels of

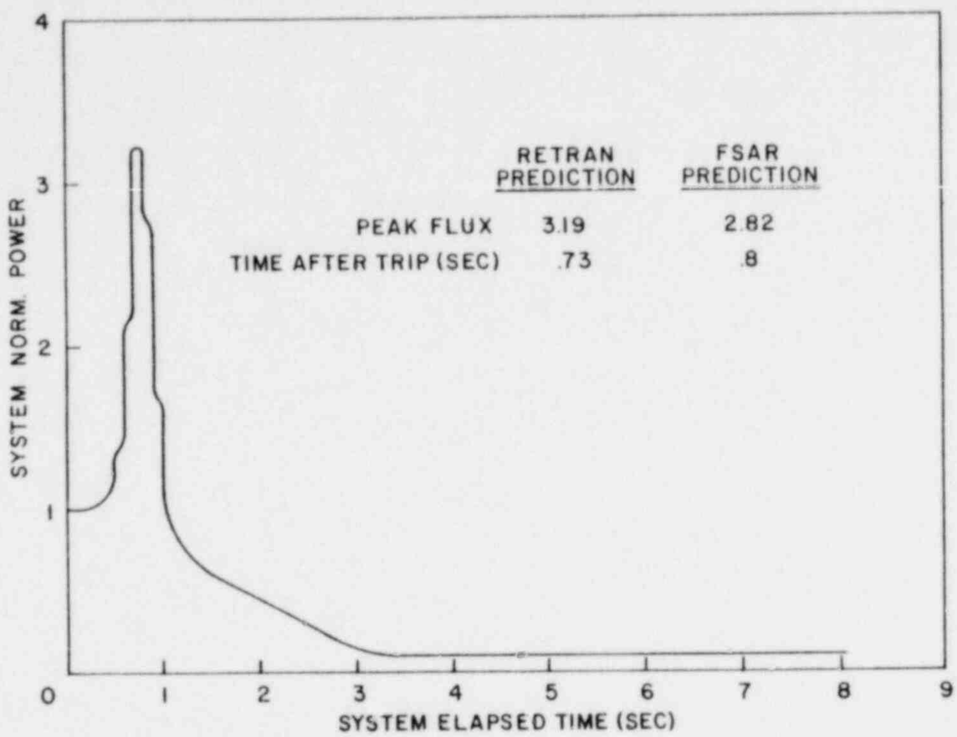

FIGURE V.1-2 Normalized Power - FSAR Base Case.

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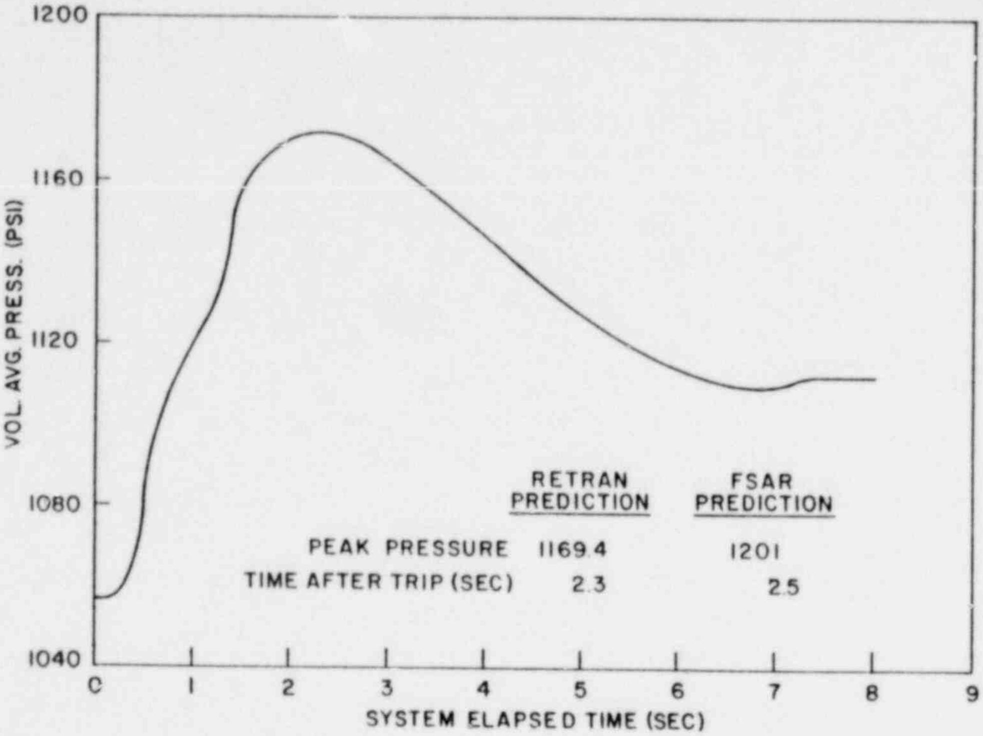

FIGURE V.1-3 Dome Pressure - FSAR Base Case.

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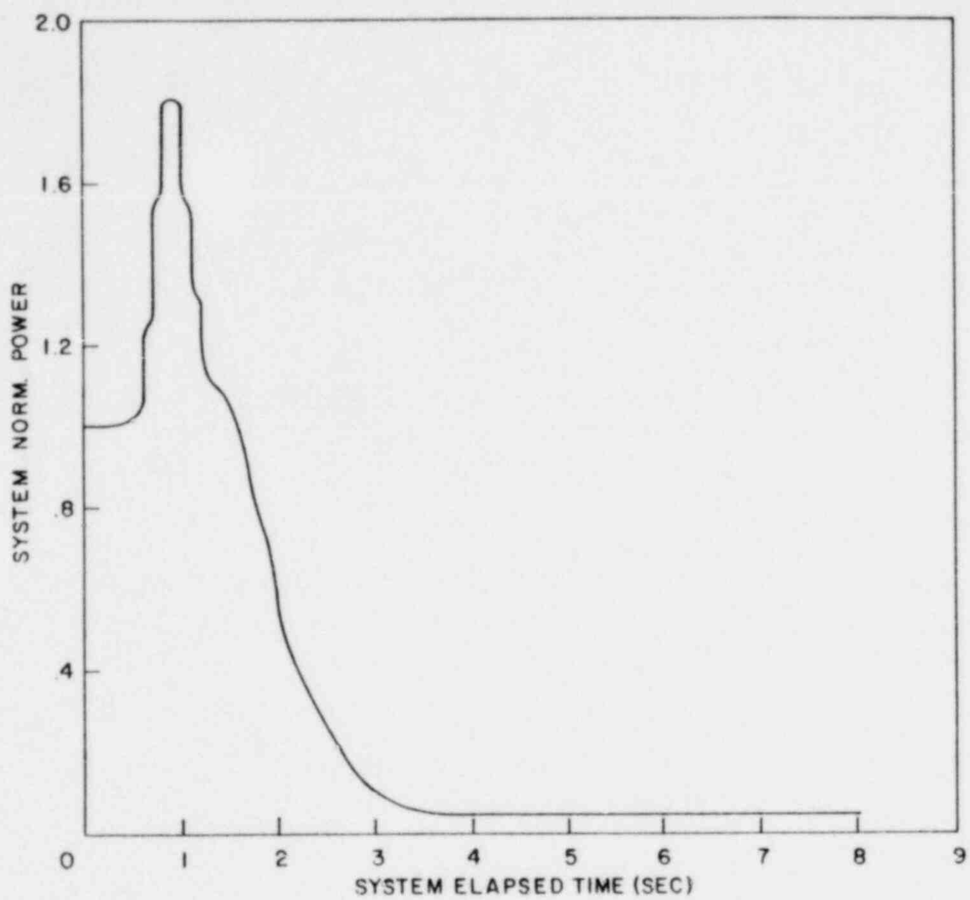

FIGURE V.1-4 Normalized Power - Nominal Base Case.

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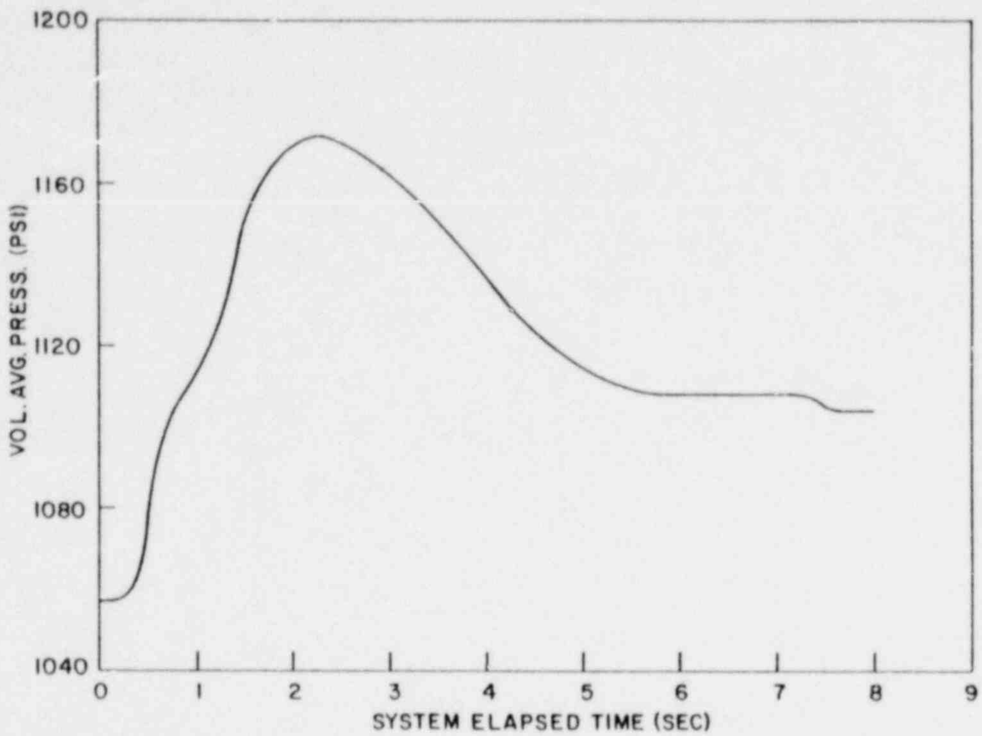

FIGURE V.1-5 Dome Pressure - Nominal Base Case.

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TABLE V.1-1

SUMMARY OF RETRAN TTWOB RUNS

	Description	Peak Power	Peak Pressure
1.	FSAR Base Case	3.190 @ 0.73 second	1169.4 @ 2.3 second
2.	Nominal Base Case	1.781 @ 0.85 second	1168.4 @ 2.4 second
3.	Low Separator Inertia	1.673 @ 0.90 second	1166.2 @ 2.2 second
4.	10% Steam Bypass	1.511 @ .085 second	1162.4 @ 2.3 second
5.	Conservative Scram	2.688 @ 0.82 second	1172.2 @ 2.6 second
6.	No Moderator Heating	1.905 @ 0.87 second	1166.8 @ 2.3 second
7.	Moderator Heating Based on Density	1.784 @ 0.84 second	1168.1 @ 2.4 second
8.	Fermi-2 Plant Unique	1.450 @ 0.92 second	1163.4 @ 2.4 second

TABLE V.1-2

SUMMARY OF KEY INPUT PARAMETERS

	Run Description	(¢/% Void Change)	Bypass (%)	Scram	Mod. Heating	Sep. Inertia (ft.$^{-1}$)
1.	FSAR Base Case	11.5	0.	Conserv.	Straight 2%	.664
2.	Nominal Base Case	7.48	0.	Nominal	Straight 2%	.664
3.	Low Separator Inertia	7.48	0.	Nominal	Straight 2%	.212
4.	10% Steam Bypass	7.48	10.	Nominal	Straight 2%	.664
5.	Conservative Scram	7.48	0.	Conserv.	Straight 2%	.664
6.	No Moderator Heating	7.48	0.	Nominal	None	.664
7.	Moderator Heating Based on Density	7.48	0.	Nominal	Weighted 2%	.664
8.	Fermi-2 Plant Unique	7.48	10.	Nominal	Straight 2%	.664

NOTE 1: FSAR Base Case includes recirculation pump trip.

47.4 percent, 61.6 percent and 69.1 percent of full power. Test 1 (45.4% rated power) had an initial flow of 98.8% normal, and the scram level was set at 85% of full power. The initial flow for Test 2 (61.6% rated power) was reduced to 80.9% of the normal flow and the scram level was set at 95% of full power. For the third test, the initial flow was 99.4% of the rated value and the reactor was set to scram at 77% of full power. Results of these tests are given in References V.1-6 and V.1-7. EPRI has performed a number of analyses of these tests. The results discussed in the following sections are taken from preliminary reports[V.1-3 to V.1-5], and from discussions with EPRI personnel. The analyses will be published as an EPRI report.

1.2.1 Description of Model

A geometric description of the RETRAN model is shown in Figure V.1-6. The model has 35 control volumes, 23 junctions and 13 heat conductors. The base model has 12 core volumes, although sensitivity studies have been performed with 3, 4 and 6 volumes. The additional noding allowed a more detailed representation of the initial flux shape as well the reactivity feedback during the transient. Many other sensitivity studies (e.g., moderator heating variations, steam line noding, void reactivity changes) were also performed, and the models for these studies as well as the results are presently being documented.

1.2.2 Results of Analyses and Comparisons with Data

In addition to the base analyses, a large number of sensitivity studies were performed, including one for the separator inertia. Based on test data for the separator pressure drop, Hornyik and Naser [V.1-4] concluded that the separator inertia for a RETRAN-type code should be lower than computed from geometric considerations alone. Figure V.1-7 shows the test pressure at the core exit and values of this same parameter computed by RETRAN for two values of separator inertia. These results tend to support the conclusion on separator inertia.

Experimental data are compared with the RETRAN calculations for Test 1 in Figures V.1-8 to V.1-12. Comparisons are made for the normalized power as well as pressure values for the steam dome, upper plenum, steam line and at the turbine inlet. The RETRAN results for these system parameters agree quite favorably with the experimental data, although the calculated values fall below

PEACH BOTTOM RETRAN MODEL

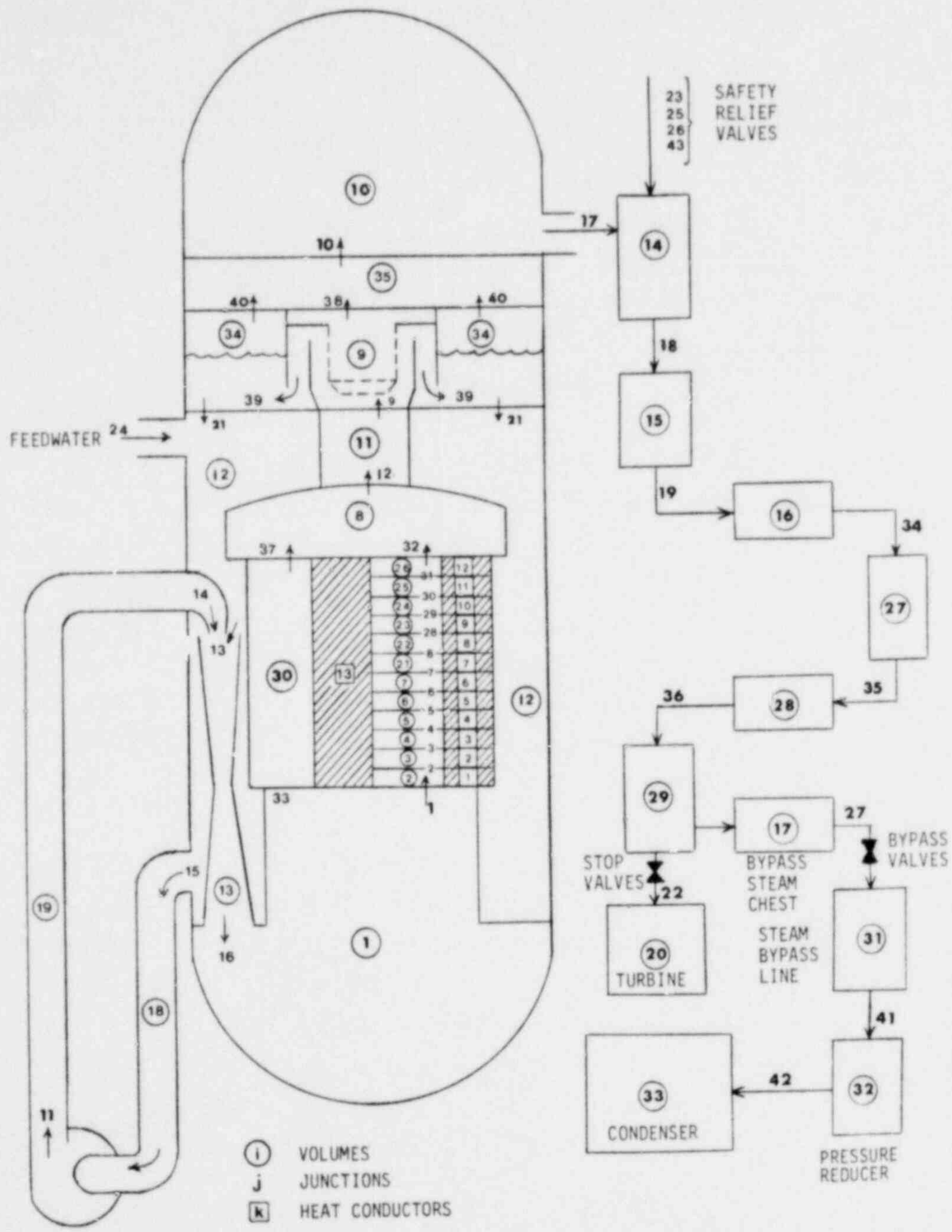

FIGURE V.1-6 RETRAN Model for Peach Bottom 2 Turbine Trip.

1759 220

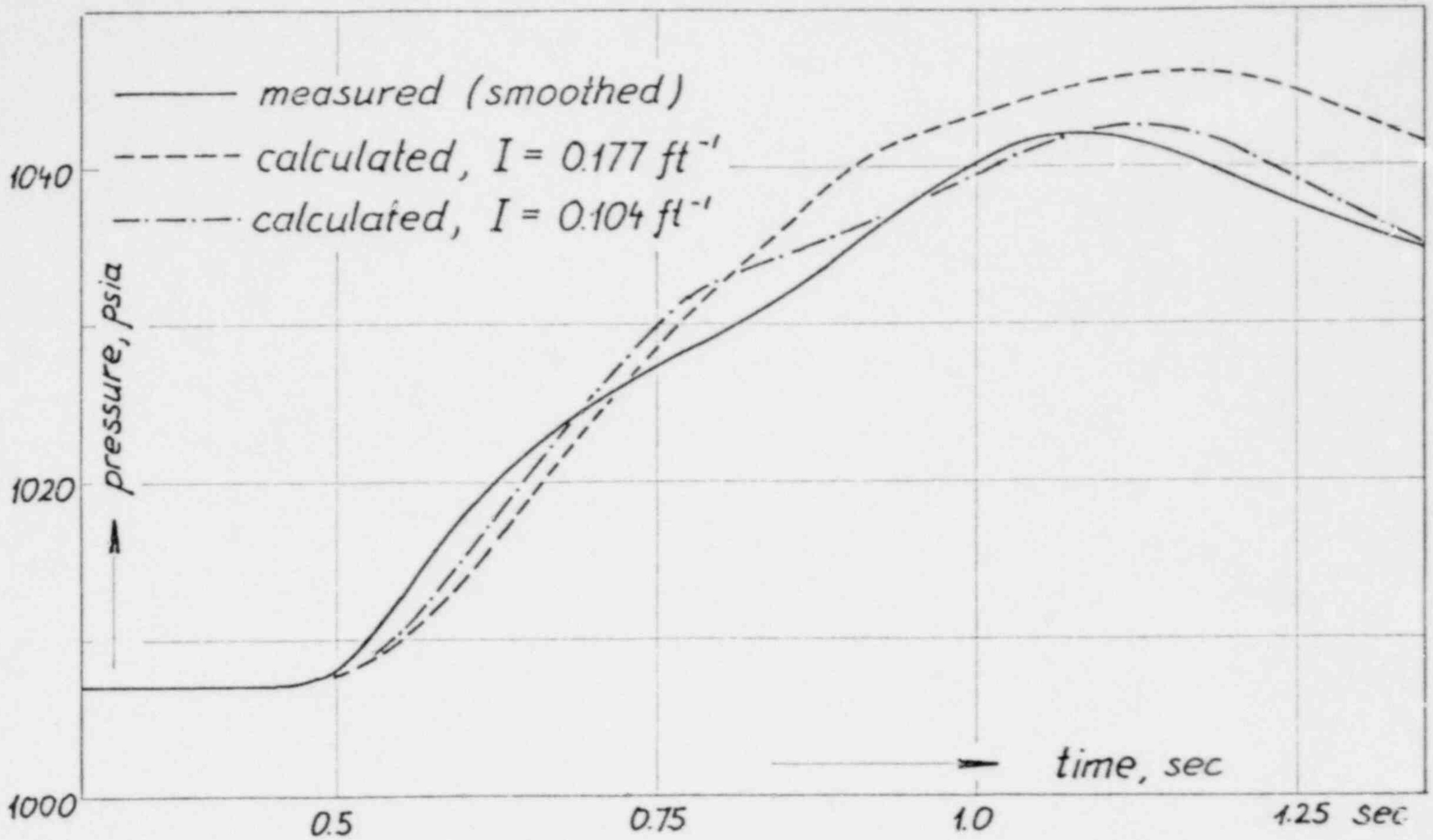

FIGURE V.1-7 Core Exit Pressure - Test 1.

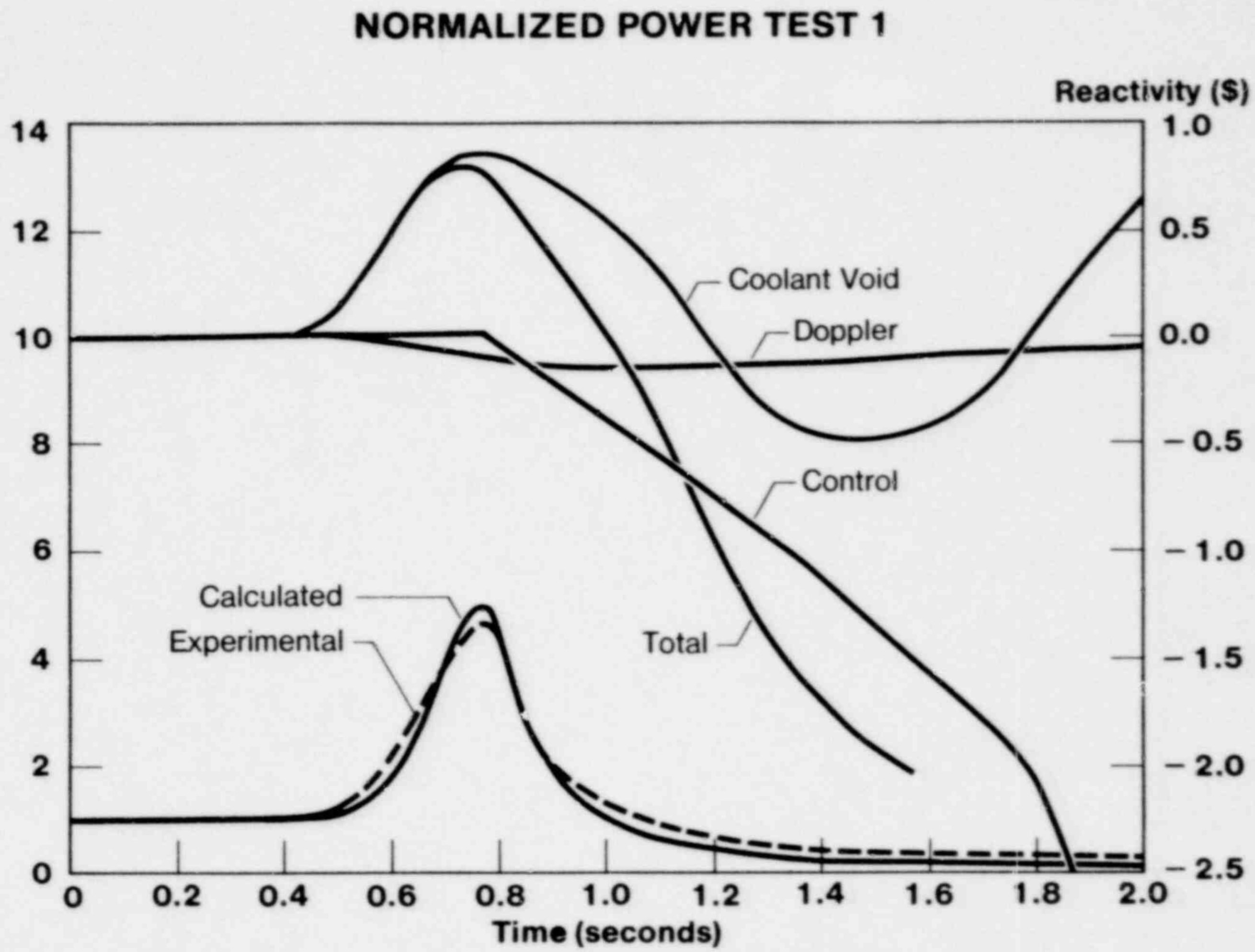

FIGURE V.1-8 Normalized Power - Test 1.

222

STEAM DOME PRESSURE TEST 1

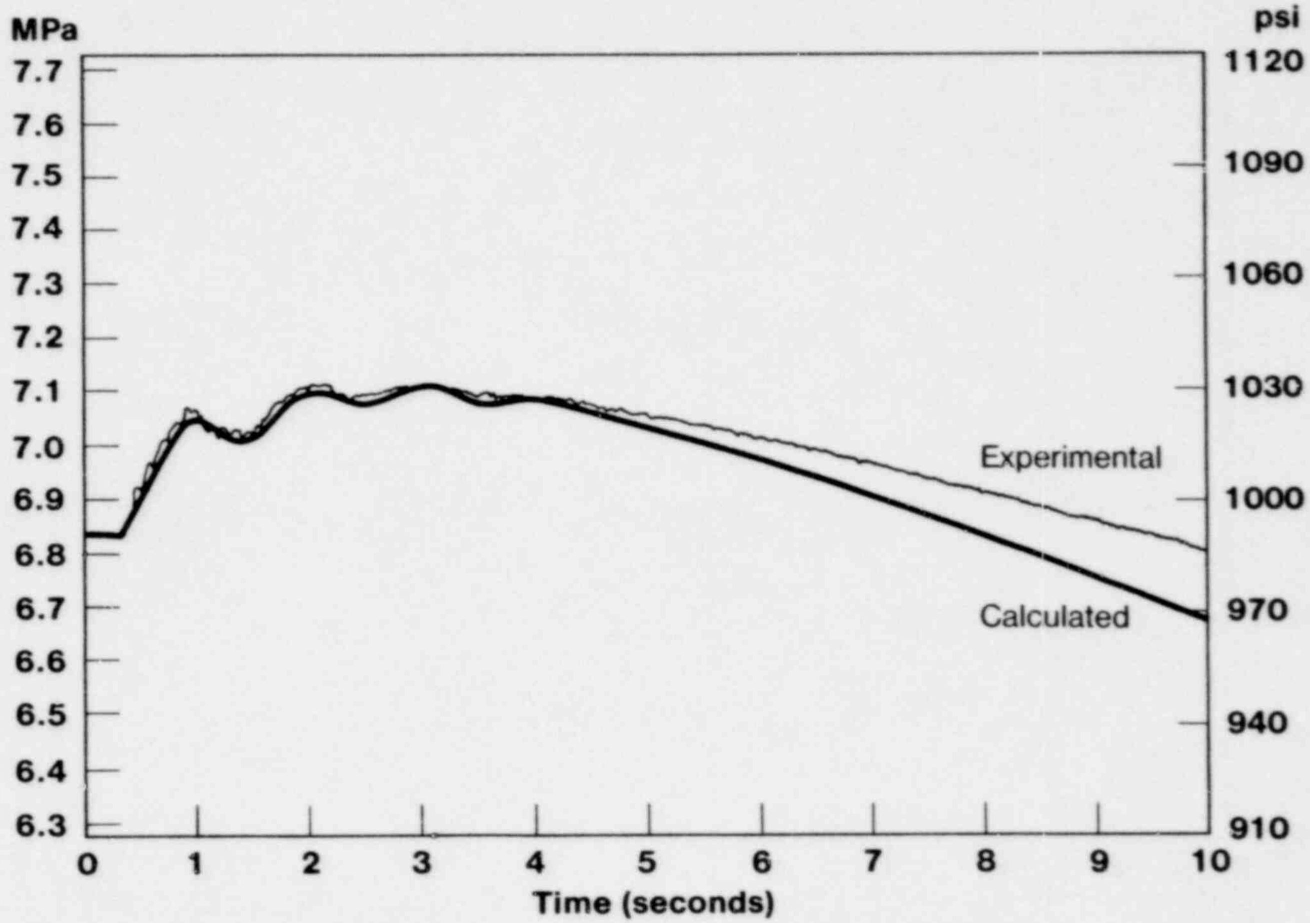

FIGURE V.1-9 Steam Dome Pressure - Test 1.

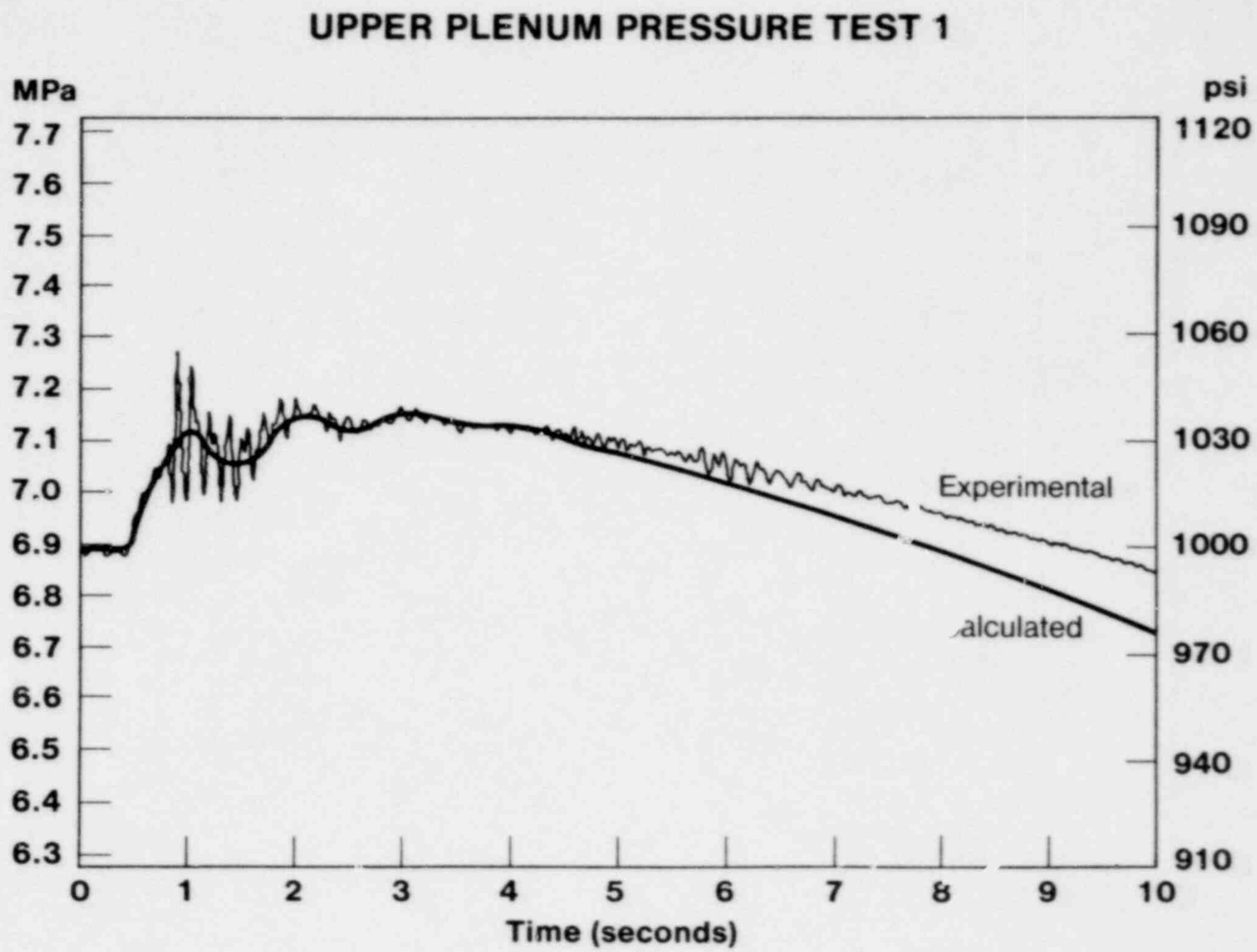

FIGURE '.1-10 Upper Plenum Pressure - Test 1.

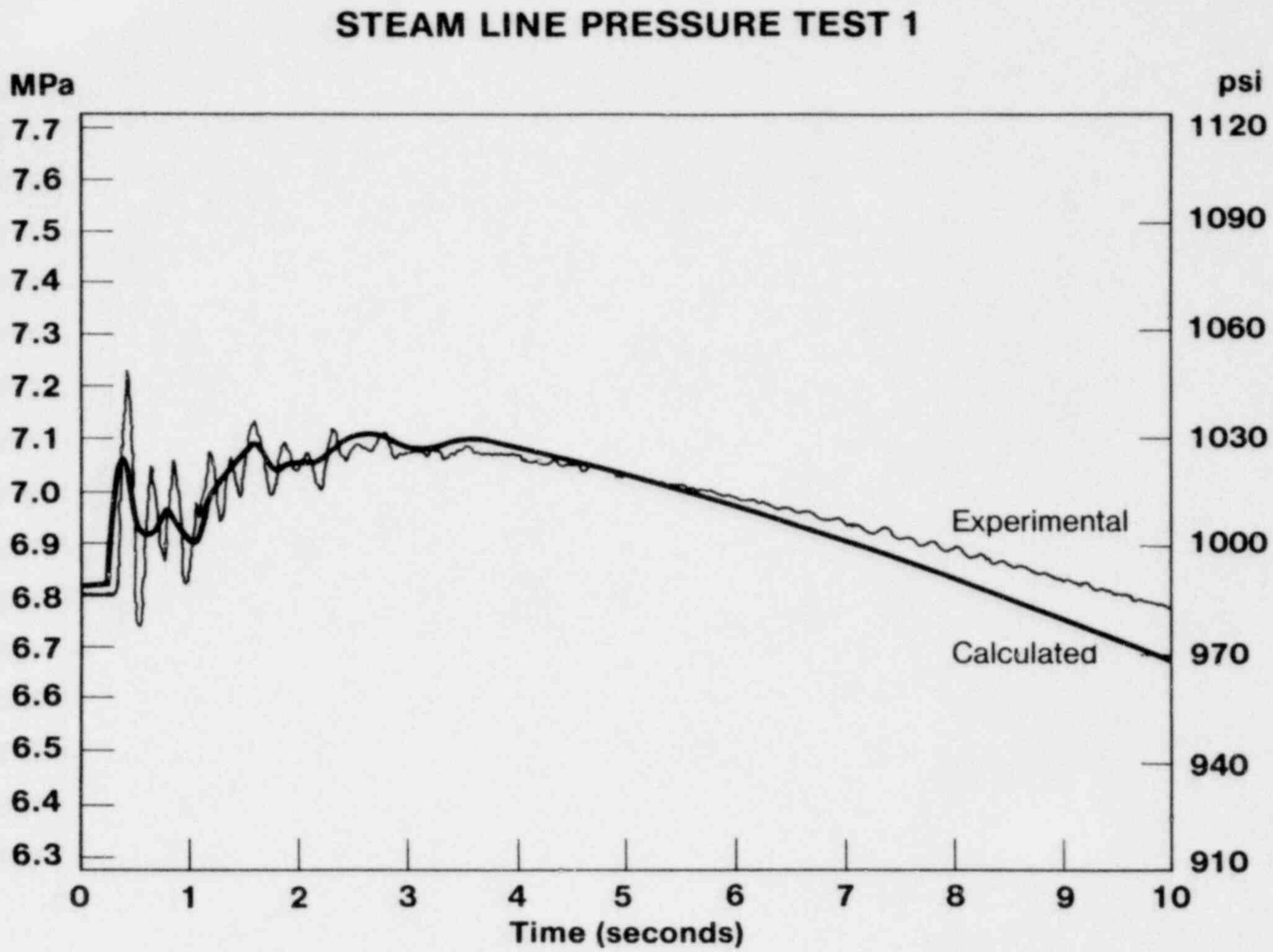

FIGURE V.1-11 Steam Line Pressure - Test 1.

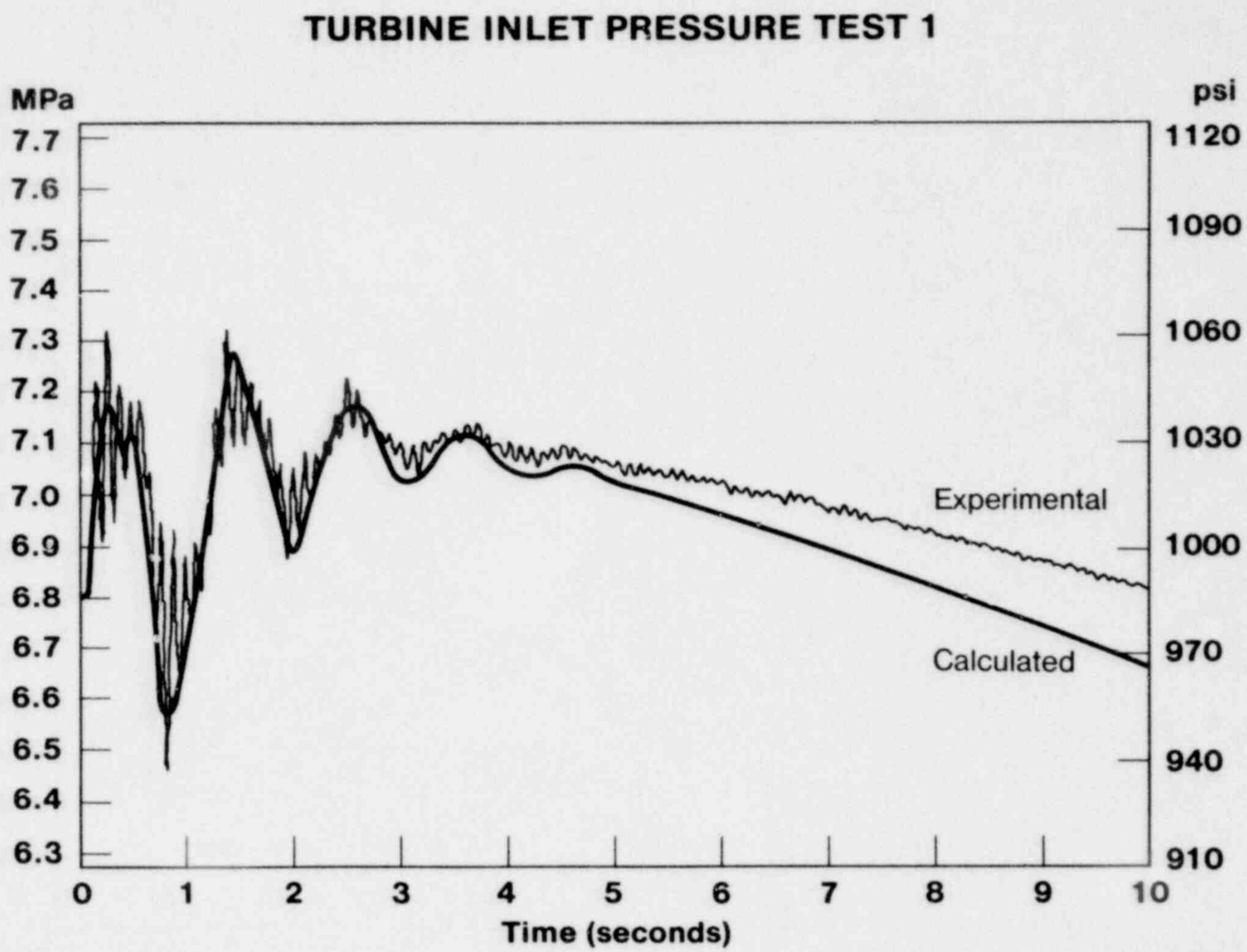

FIGURE V.1-12 Turbine Inlet Pressure - Test 1.

the data late in the transient. Similar comparisons are given in Figures V.1-13 to V.1-17 for Test 2 and Figures V.1-18 to V.1-22 for Test 3. From an overall viewpoint, the RETRAN calculations give good agreement with the experimental data. This is noteworthy when one considers that these analyses were performed with equilibrium assumptions and point kinetics. As such, the Peach Bottom data comparisons provide a very good preliminary qualification for some BWR pressure increase transients.

1.3 Summary of Results

The Detroit Edison analysis of the turbine trip transient indicates the basic model and the RETRAN code are probably adequate for analyzing these types of transients. Preliminary results from the EPRI analysis show that the present version of RETRAN can predict the pressure history and other parameters quite well. However the value of the peak power is sensitive to a variety of neutronic and hydraulic parameters which require additional study.

9 227

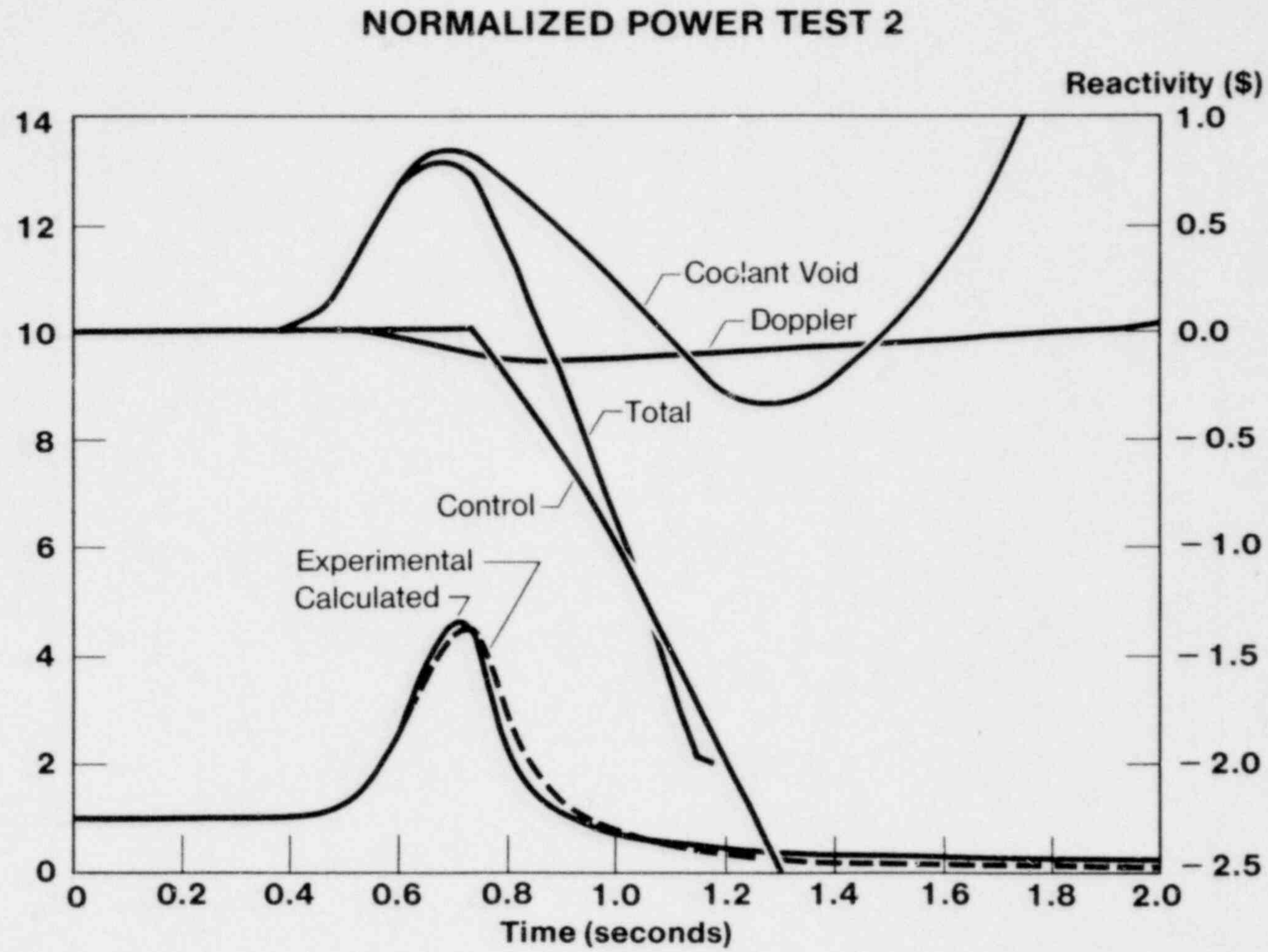

FIGURE V.1-13 Normalized Power - Test 2.

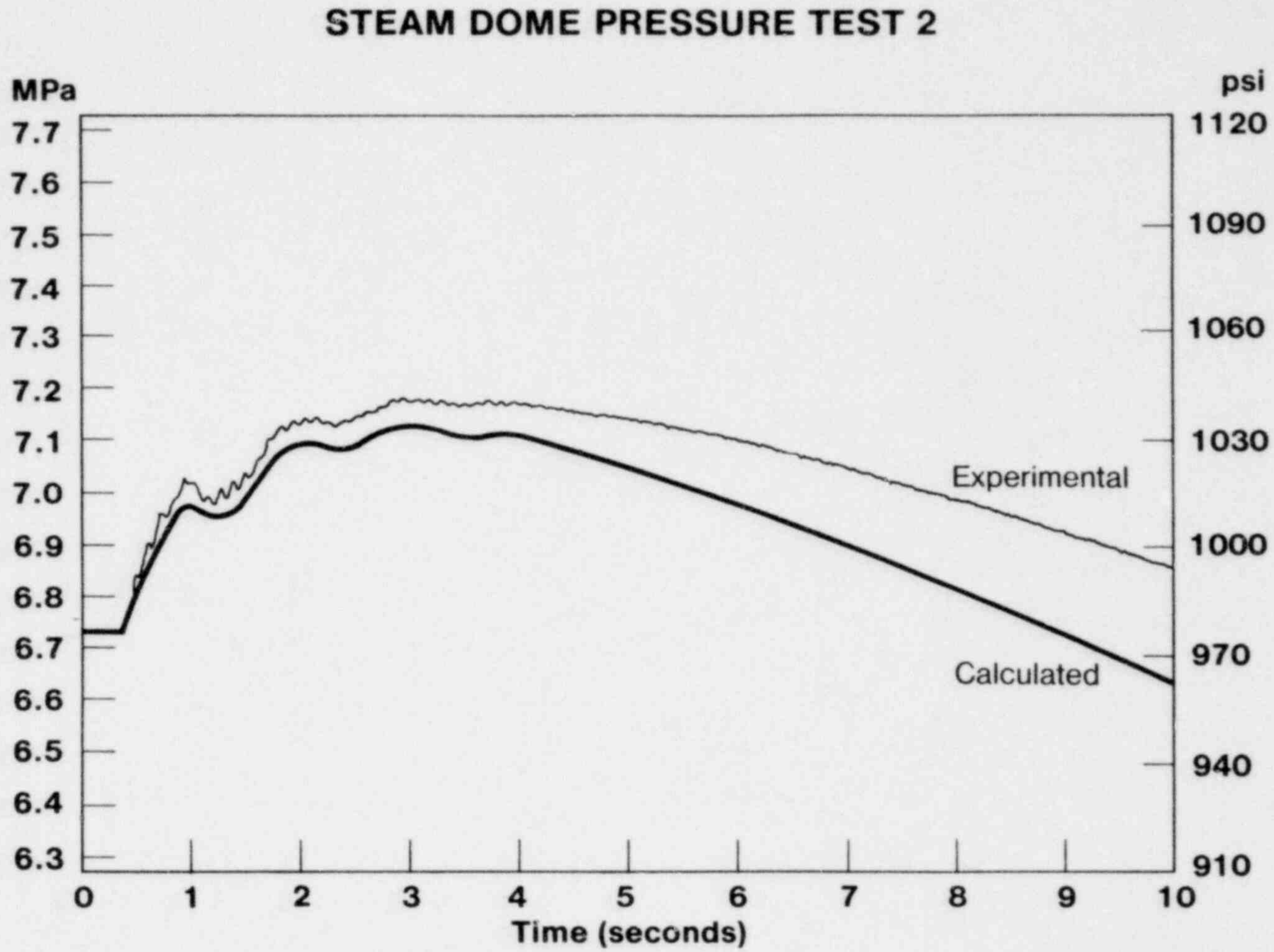

FIGURE V.1-14 Steam Dome Pressure - Test 2.

9 229

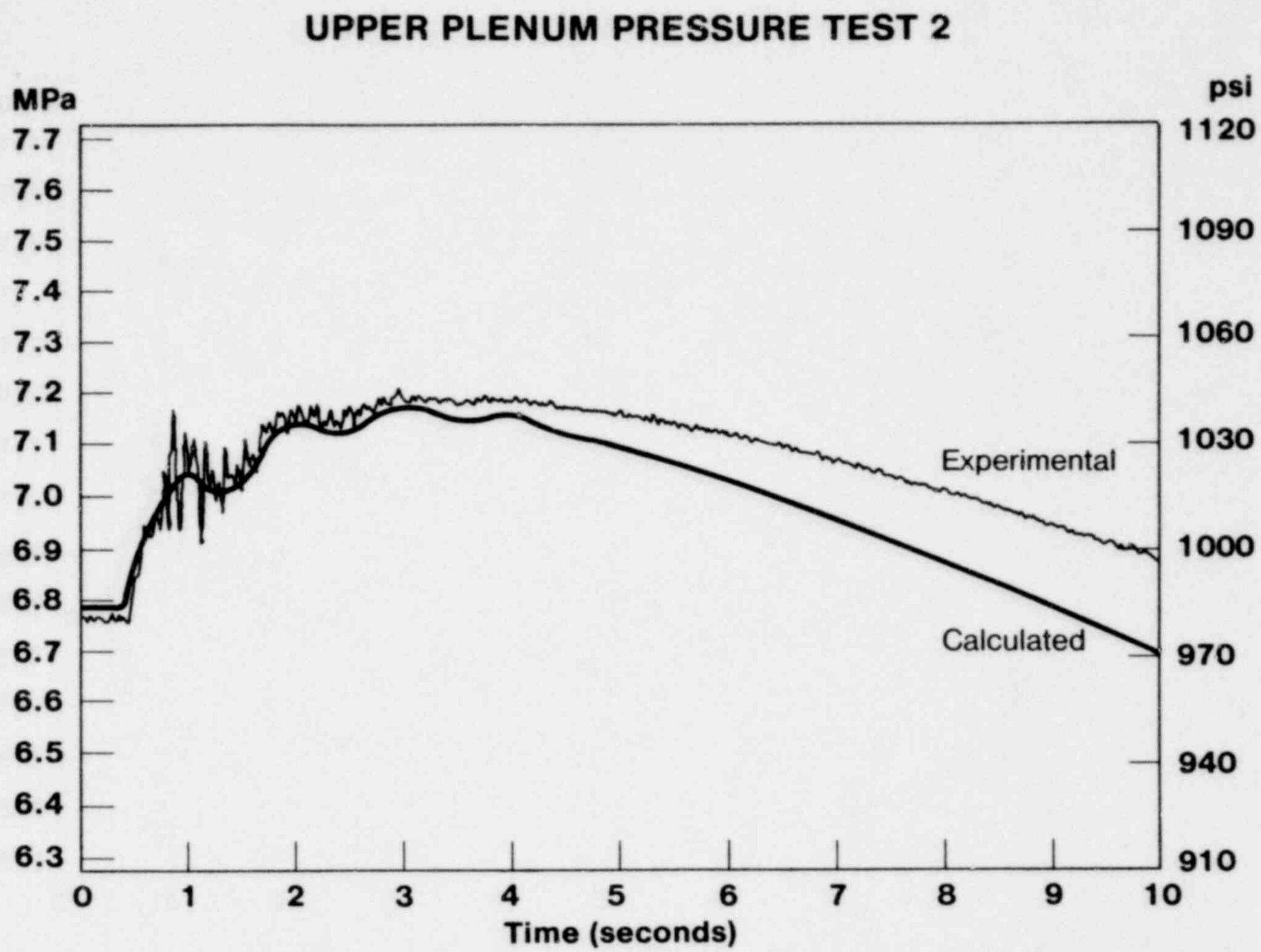

FIGURE V.1-15 Upper Plenum Pressure - Test 2.

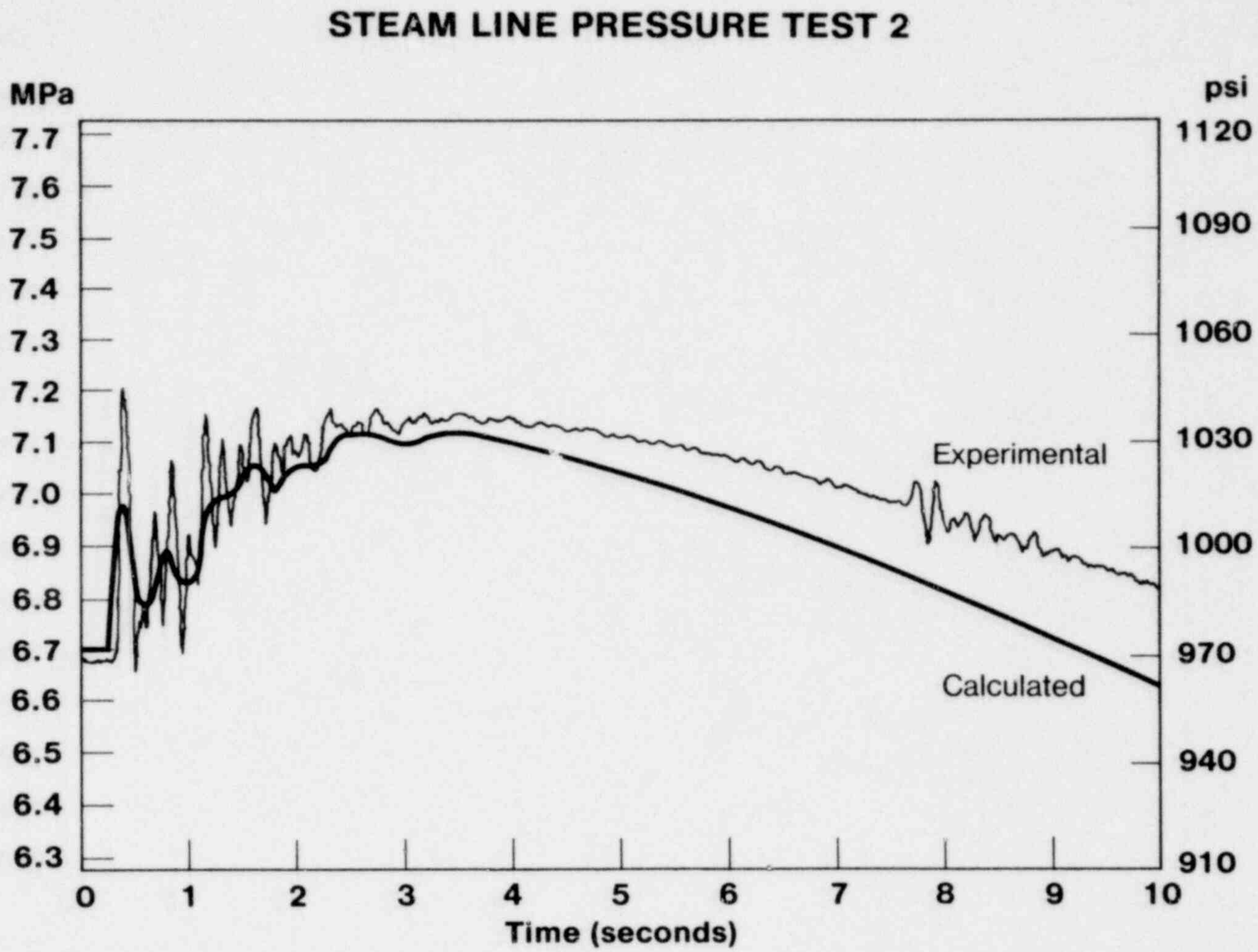

FIGURE V.1-16 Steam Line Pressure - Test 2.

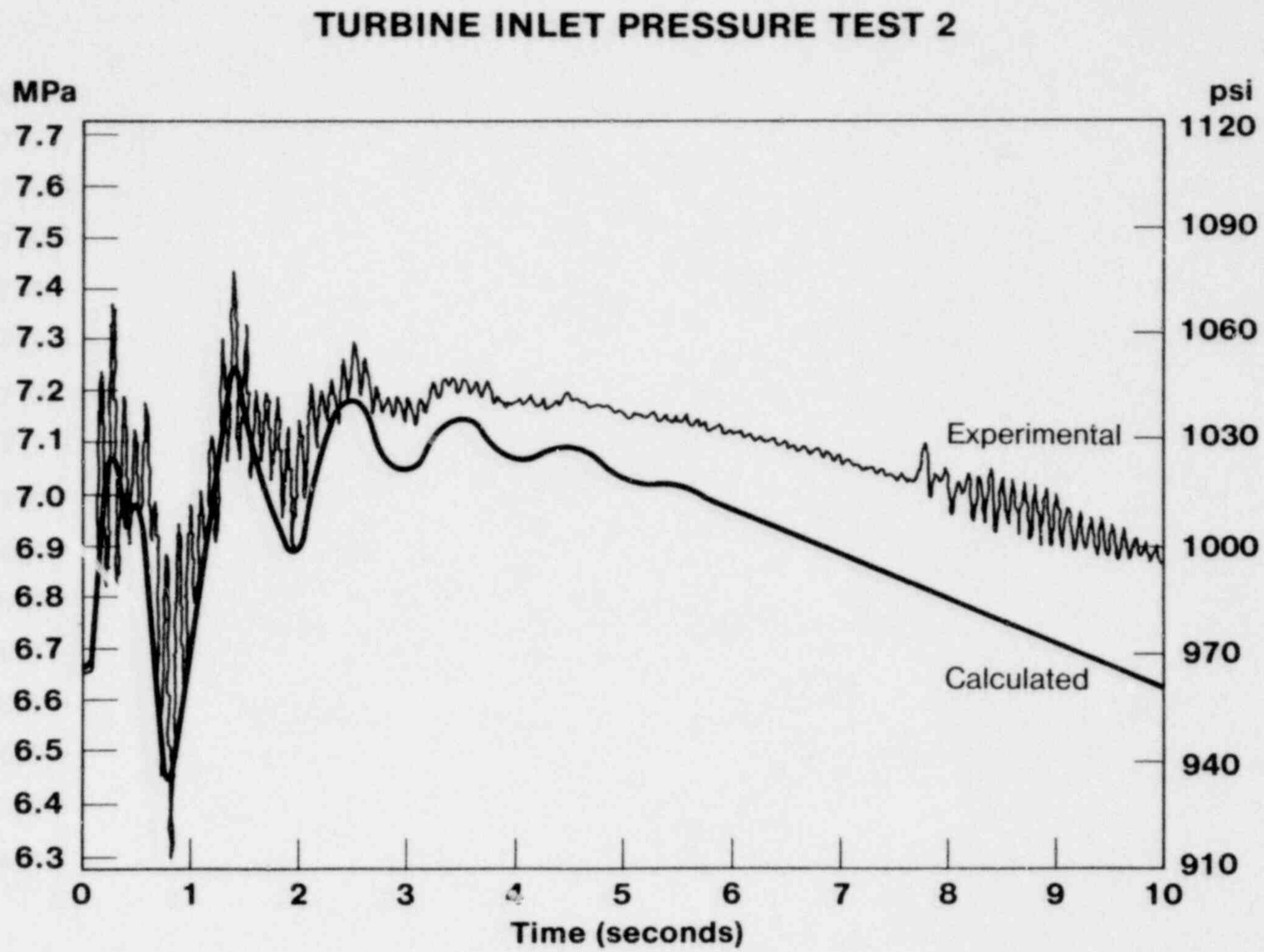

FIGURE V.1-17 Turbine Inlet Pressure - Test 2.

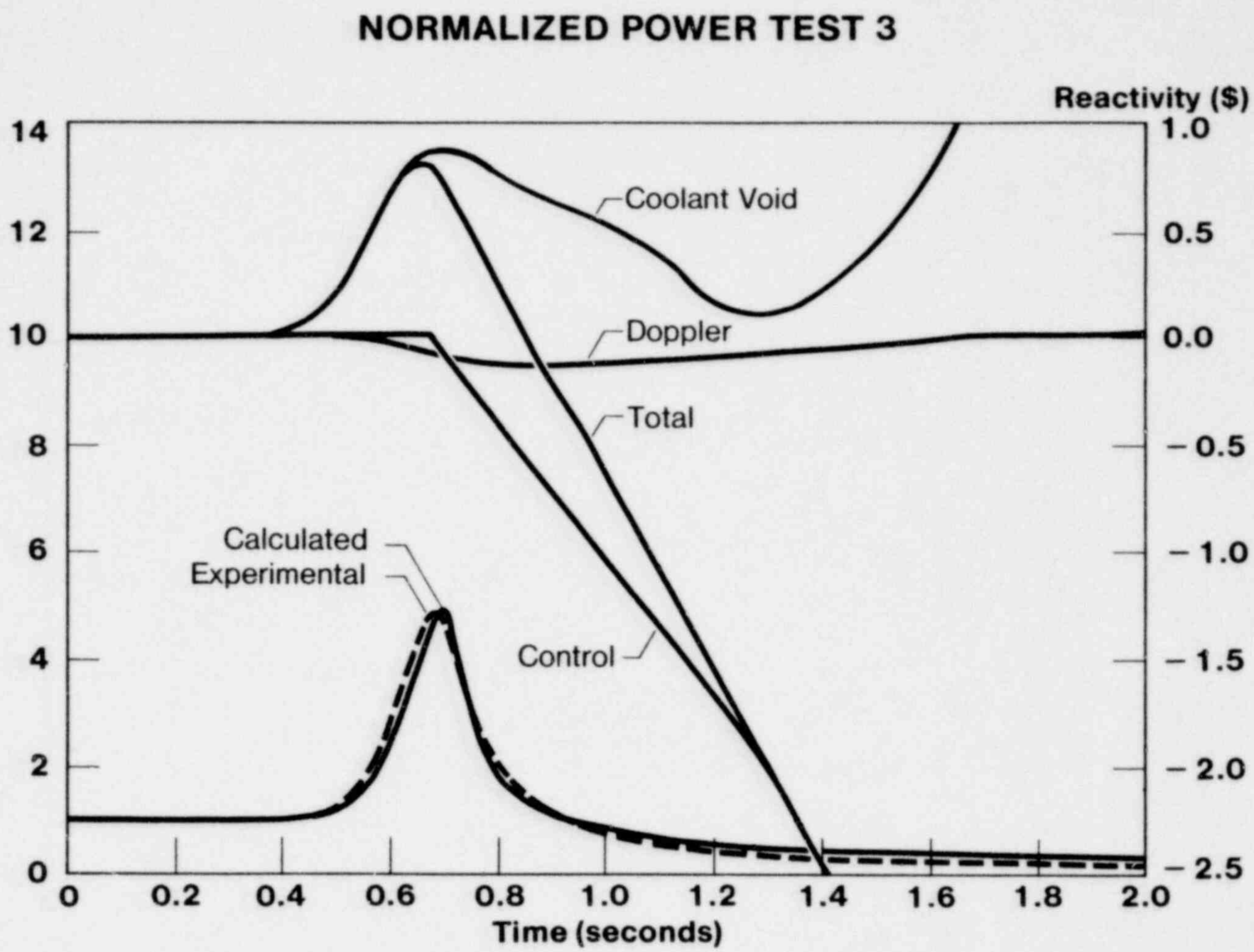

FIGURE V.1-18 Normalized Power - Test 3.

9 233

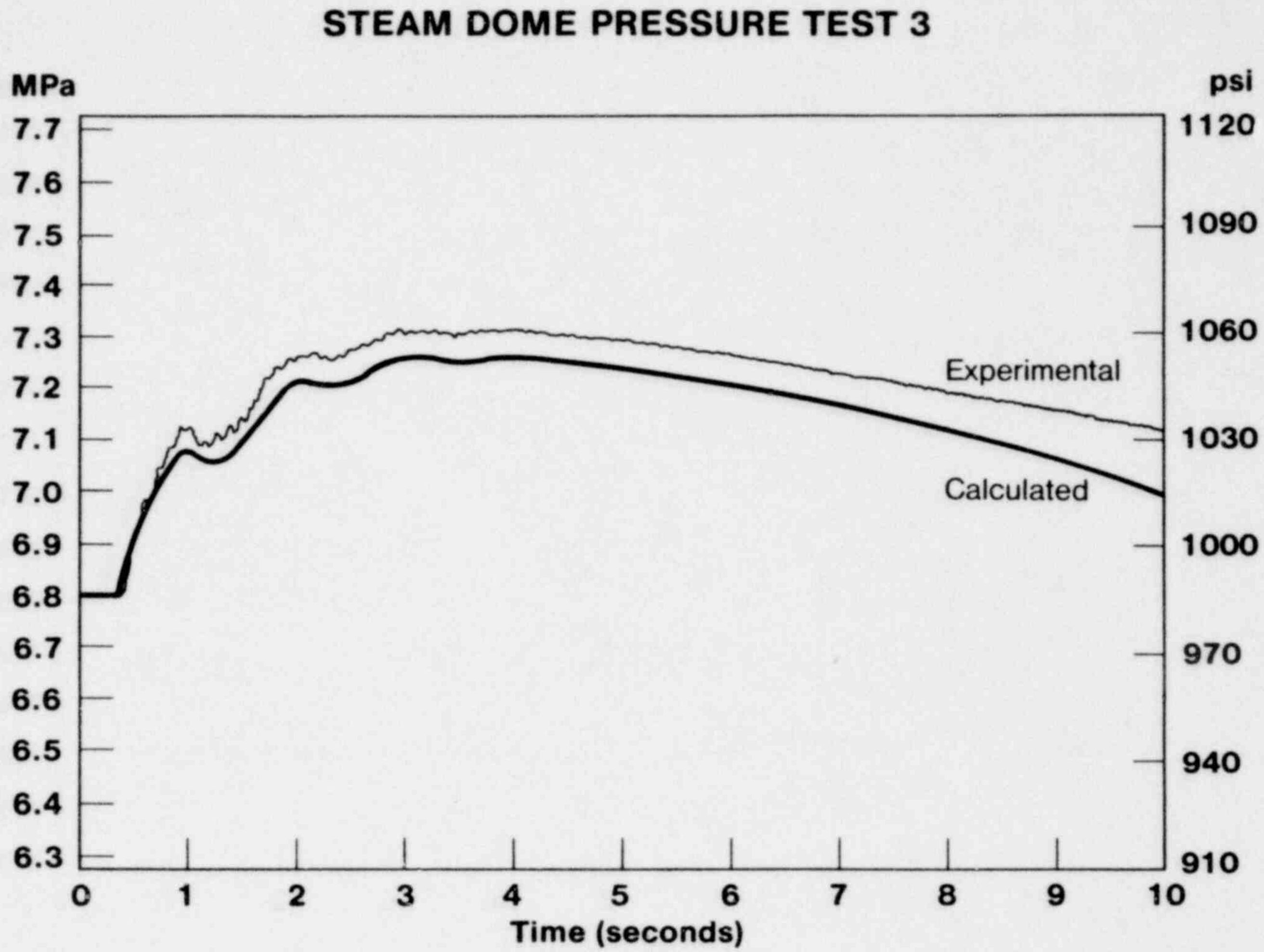

FIGURE V.1-19 Steam Dome Pressure - Test 3.

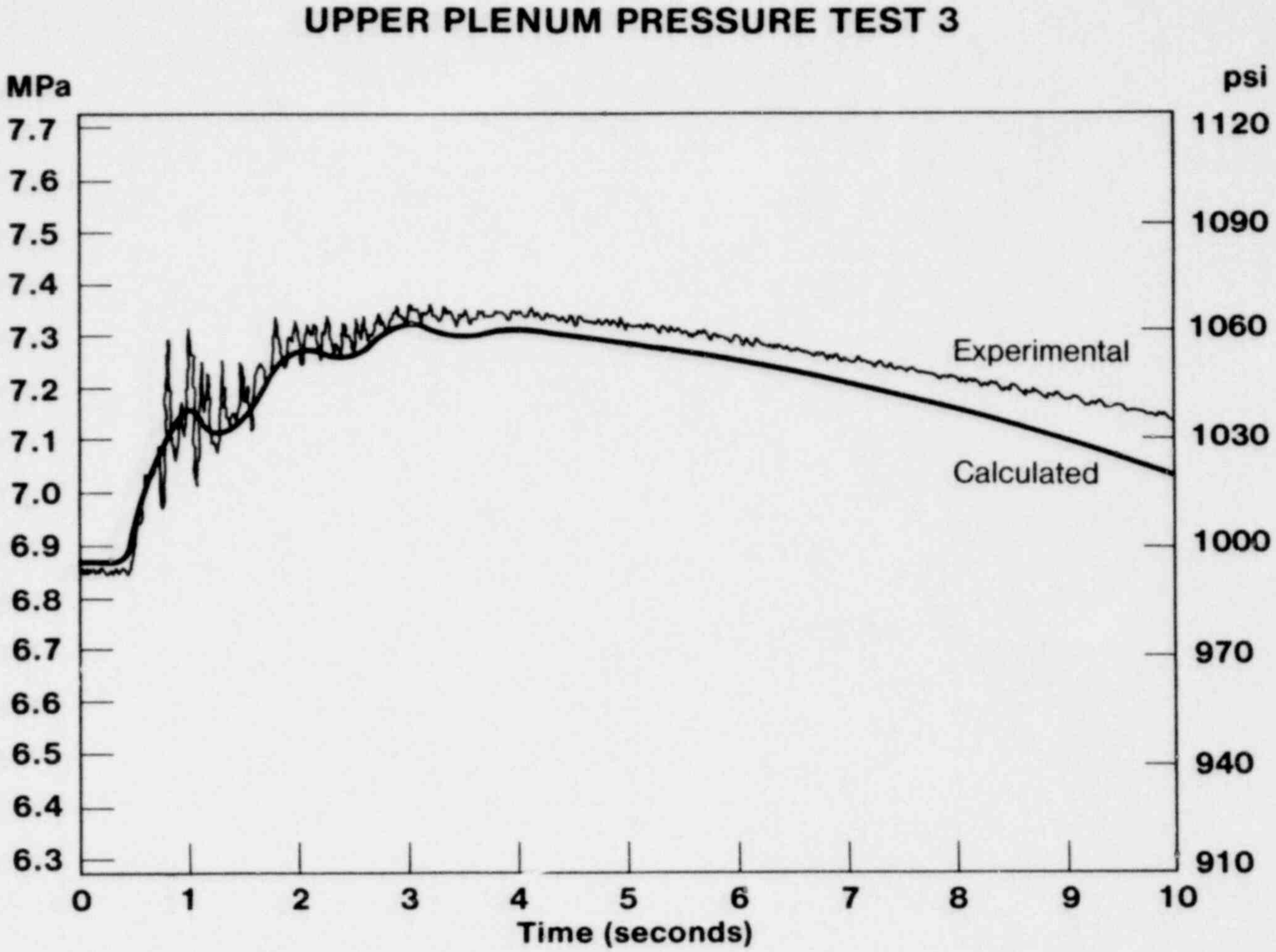

FIGURE V.1-20 Upper Plenum Pressure - Test 3.

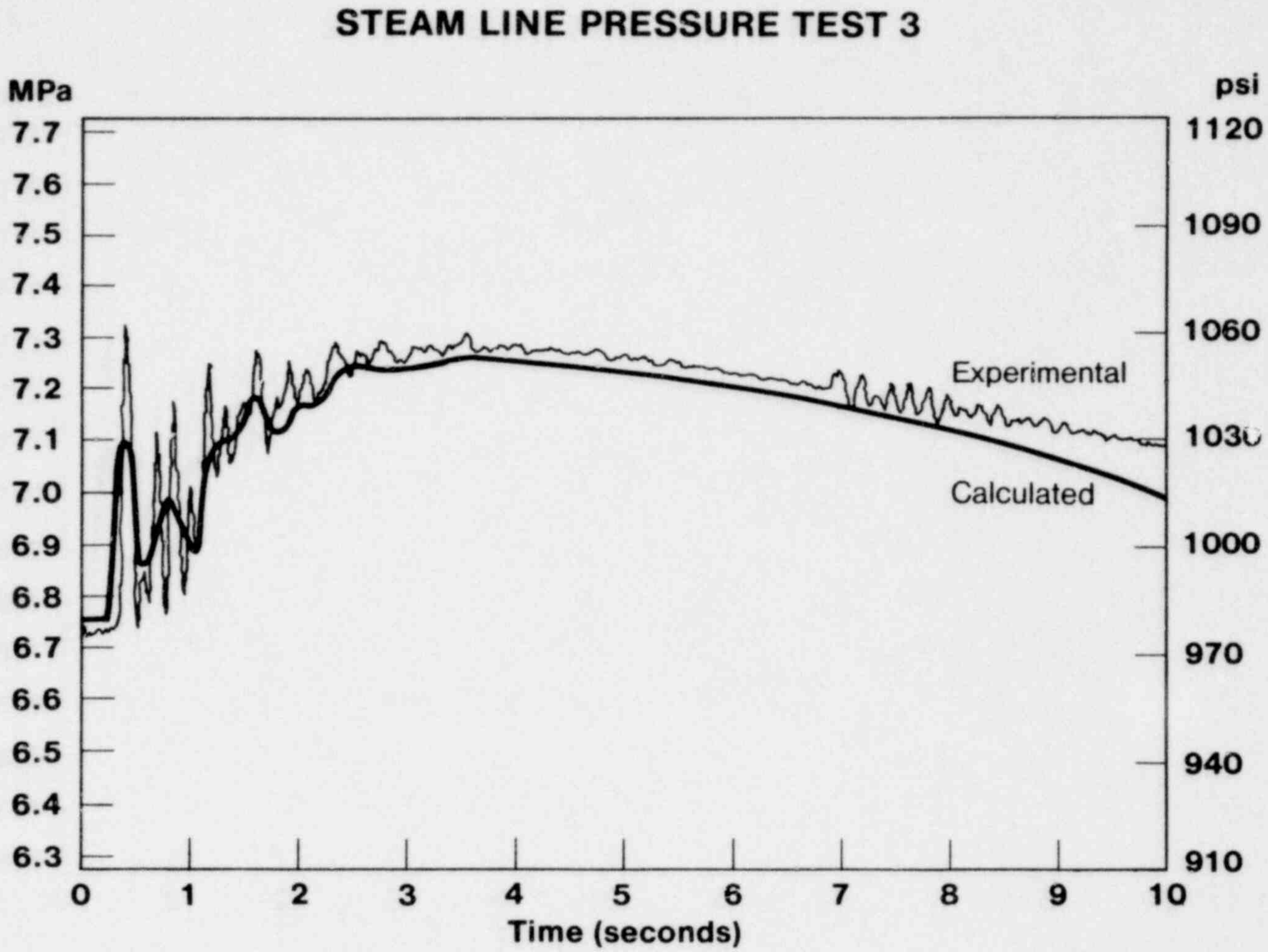

FIGURE V.1-21 Steam Line Pressure - Test 3.

1709 236

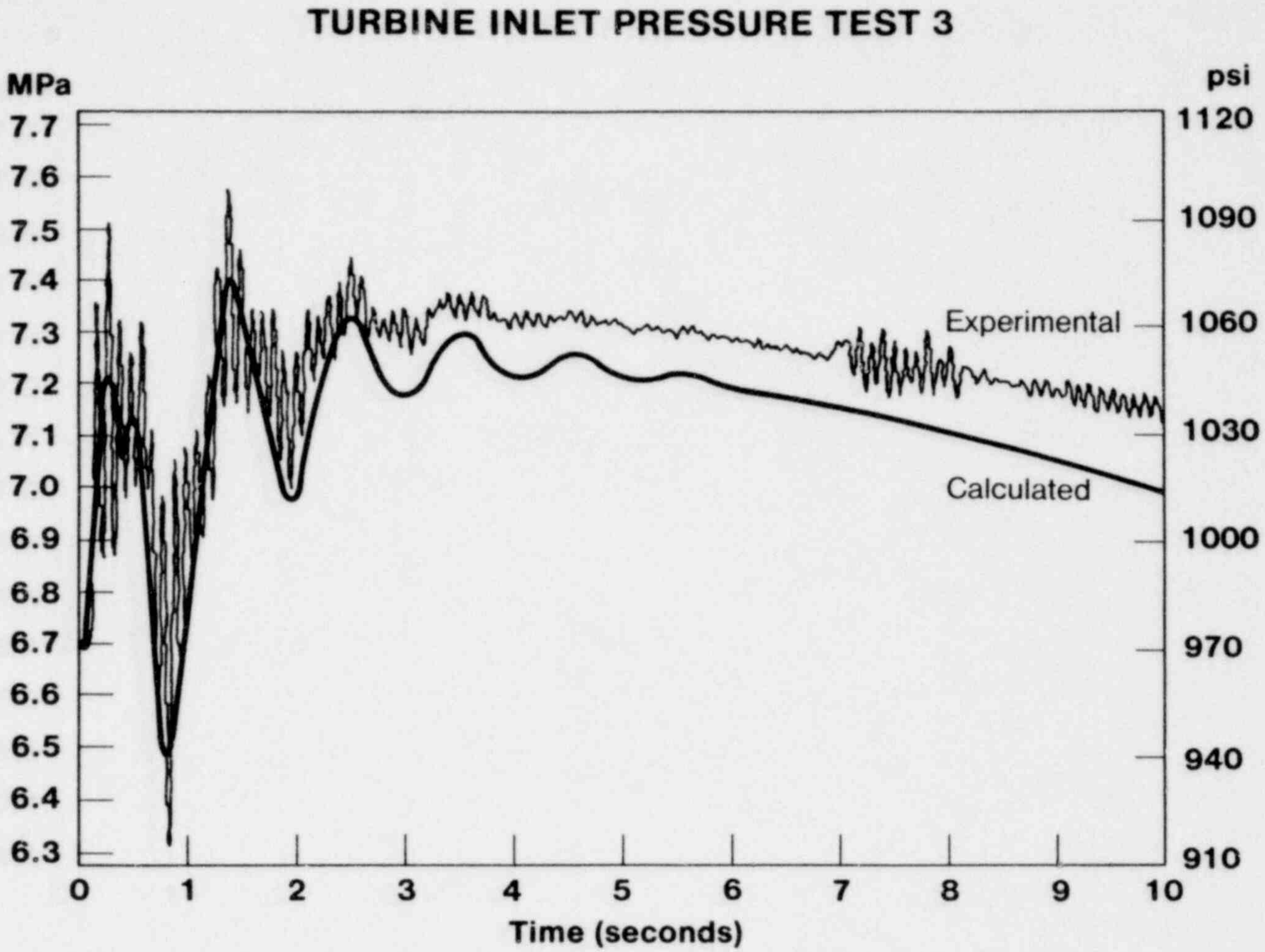

FIGURE V.1-22 Turbine Inlet Pressure - Test 3.

2.0 GENERATOR LOAD REJECTION

The generator load rejection transient is pressure increase type transient. It is a specific transient for the general type of transient which occurs when the turbine control valve closes rapidly. The general sequence of events begins with an action which results in closing the valve controlling the flow to the turbine. The bypass valves then start to open which causes a decrease in the system pressure. The reactor is then scrammed and closure of the turbine stop valves results in an increasing pressure. This pressure rise is terminated by the opening of relief valves. In the analysis discussed in this section, the actions during the transient followed a different sequence than would normally occur.

The Tennessee Valley Authority (TVA) performed this transient during testing of a 3293 MWt BWR/4. Data from this test are compared to RETRAN results [V.2-1] in the following section.

2.1 TVA Analysis

The sequence of events (Table V.2-1) which occurred during this test are slightly different than those discussed above. The two main exceptions are (1) the power/load unbalance determination failed to produce a fast control valve closure and (2) the low water level set point was set too high resulting in a second pressure rise and the initiation of the high pressure coolant injection and the reactor core isolation cooling systems. Since these two systems were not included in the RETRAN model, the analysis of the transient was terminated earlier than the test. The failure of the power/load unbalance trip was included in the RETRAN analysis. A detailed discussion of the analysis is given in Reference V.2-1. The summary which follows is taken from this reference.

2.1.1 Description of Model

A description of the RETRAN geometric model for this analysis is shown in Figures V.2-1 to V.2-3. The model has 36 volumes and 58 junctions, of which 12 are fill junctions. The model shown was developed for use in a number of analyses (e.g., Sections 3.0 and 4.0) and thus has more detail than is required for this transient.

TABLE V.2-1

SEQUENCE OF EVENTS FOR TVA GENERATOR LOAD REJECTION TRANSIENT

Time (sec)	Events
0.0	Main transformer breakers open.
0.02	Power/Load Unbalance (PLU) Circuit generates control valve fast closure signal but failed relay prevents initiation.
0.12	Control valves begin to close due to turbine overspeed. Bypass valves begin to open and pressure begins to decrease.
1.63	Turbine stop valves trip. Reactor scram initiated.
1.81	Stop valves fully closed. Pressure rising.
4.4	Initial reactor pressure increase at maximum. Two relief valves are open.
6.4	Reactor water level has decreased ~60 inches. Low water level causes isolation and recirc M-G drive motor breakers open. (HPCI and RCIC auto initiated but not simulated with RETRAN.)
10.0	MSIV's fully closed. Pressure rising.
16.0	End transient simulation.

9 239

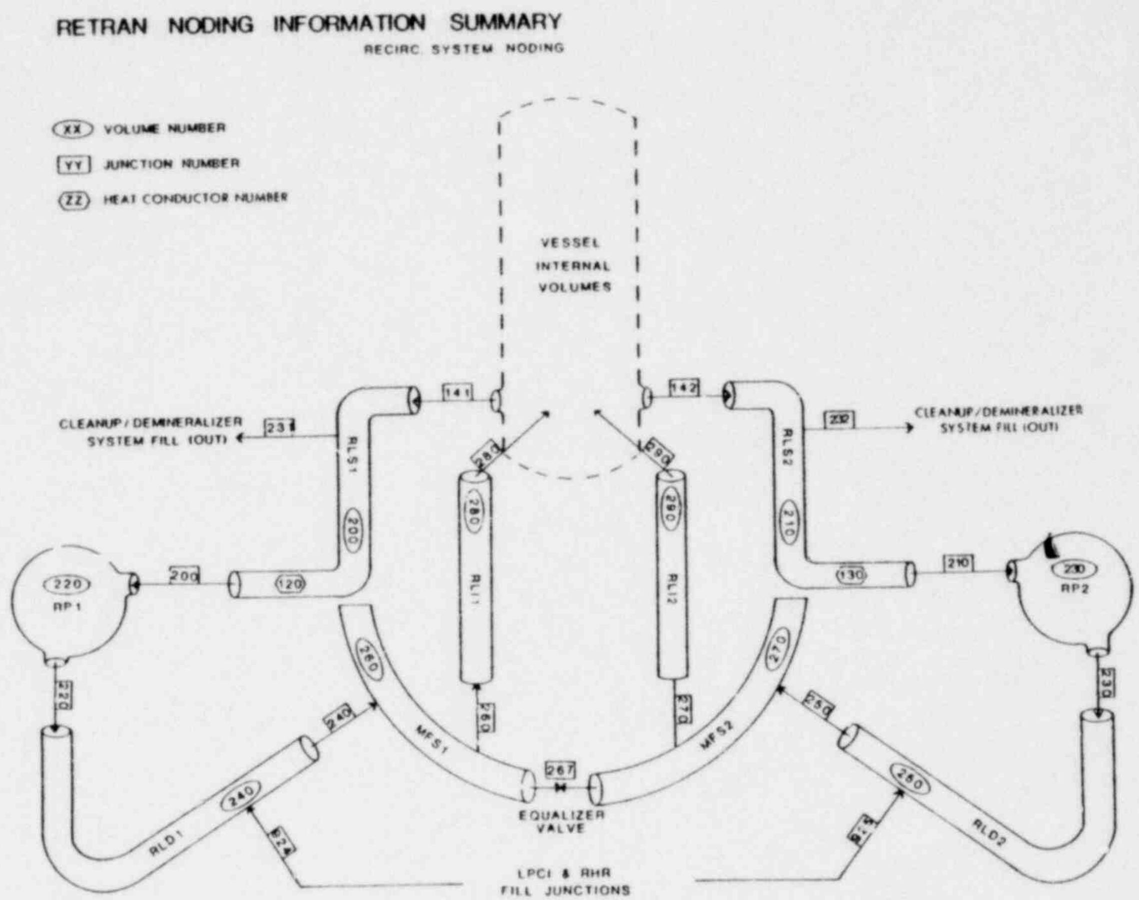

FIGURE V.2-1 TVA RETRAN Noding Summary for Recirculation System

9 240

RETRAN NODING INFORMATION SUMMARY

FEEDWATER, MAIN STEAM & CONTAINMENT NODING

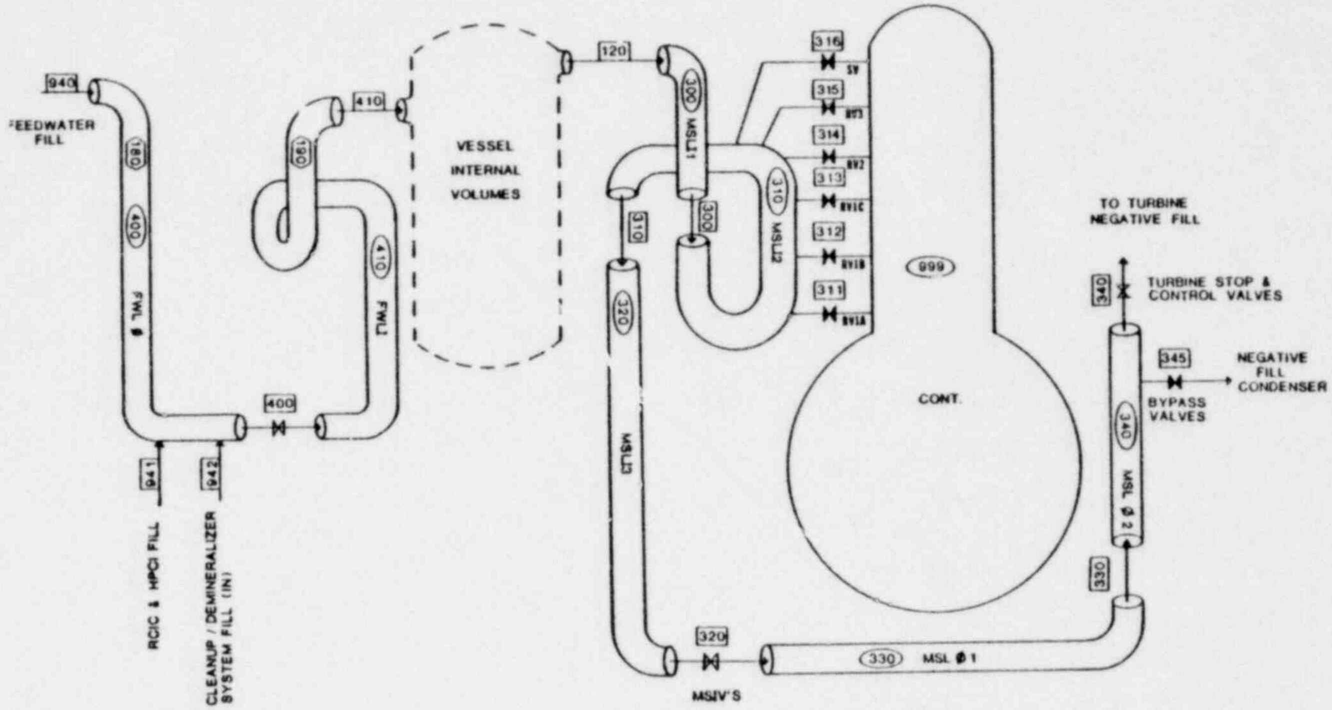

FIGURE V.2-2 TVA RETRAN Noding Summary for Feedwater, Main Steam Line and Containment

1 7 9 241

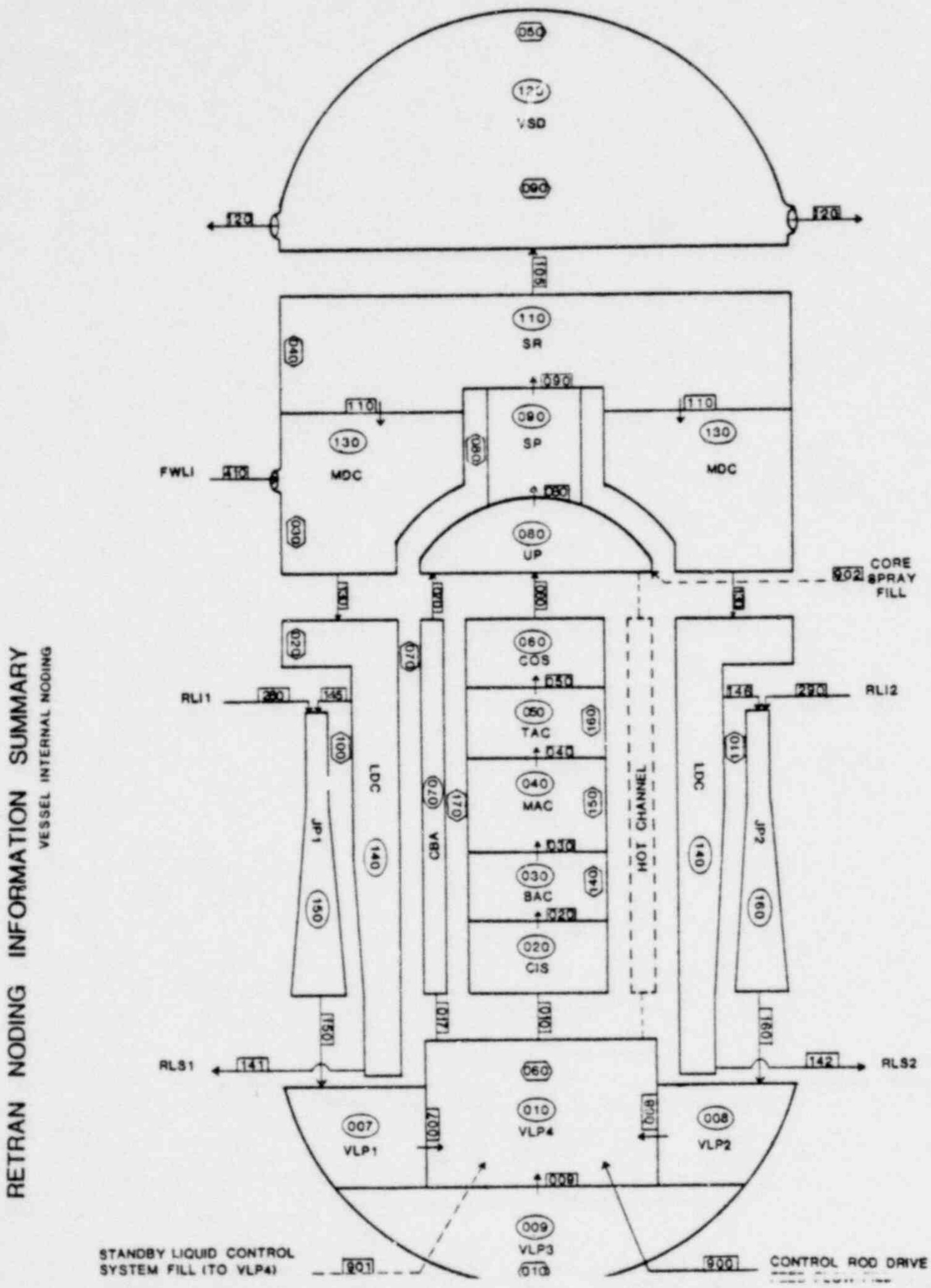

FIGURE V.2-3 TVA RETRAN Noding Summary for Reactor Vessel.

9 242

The recirculation loops were modeled separately, and the noding was determined from flow area changes. Each loop includes a lumped representation of ten jet pumps. The feedwater lines were modeled in a way such that the delay time for feedwater enthalpy changes to reach the reactor vessel is accounted for, for all expected feedwater flow rates. The main steam lines are represented by a single line with five volumes. The steam lines are connected to the containment volume with safety relief valves at junctions. The reactor core has five volumes with a single volume to represent the core bypass. The lower reactor vessel head has four volumes, including a stagnant volume for LOCA analysis.

A total of 19 heat conductors are included in the TVA model, including three in the reactor core. Heat conductors were also included for the reactor vessel barrel and heads, feedwater piping and recirculation loops. These conductors were modeled with a constant temperature for a right side boundary condition.

The control system for this plant representation is given in Figure V.2-4. In additi, to modeling the control system, TVA also used control blocks to represent portions of the system (e.g., feedwater heaters and pumps and the turbine).

Separate analyses were performed to evaluate the core model, to provide kinetics data, and the jet pump model. The core flow rate as a function of core power was evaluated both for RETRAN and a separate code designed to compute core pressure drop. This evaluation produced the input data to model the reactor core. The scram reactivity curve, doppler reactivity and void reactivity input data were determined from three-dimensional analysis. The jet pump analysis was performed to evaluate the momentum equation representation for forward flow. These analyses are described in the TVA report [V.2-1].

2.1.2 Results of Analysis and Comparison with Data

The self-initialization option was used to achieve a steady-state solution for this model. The lower plenum pressure was input and the jet pump exit junction form losses were calculated by RETRAN as was the form loss for the separators. The separator enthalpy was specified and the code was allowed to bias the feed-water enthalpy to achieve steady-state.

9 243

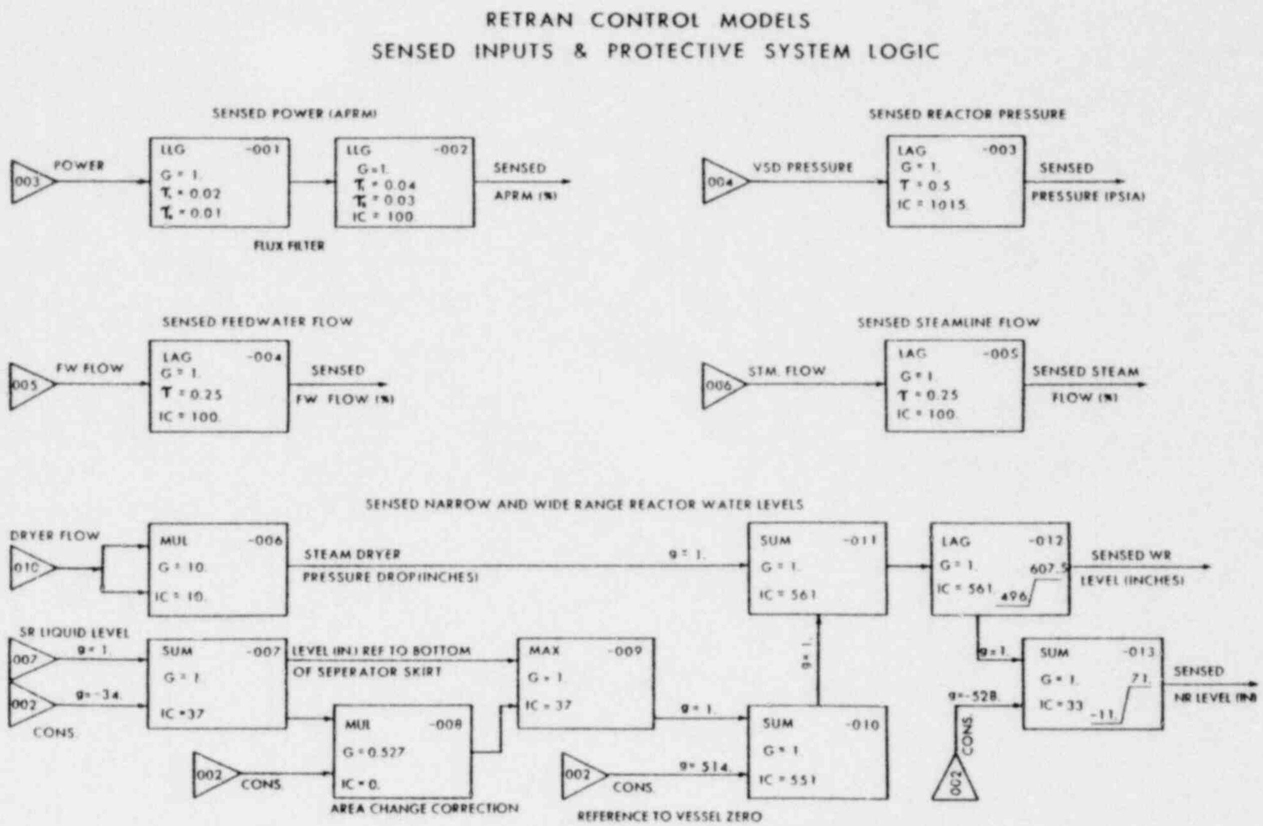

FIGURE V.2-4 RETRAN Control Model Representation for TVA System.

1379 244

RETRAN CONTROL MODELS
SENSED INPUTS & PROTECTIVE SYSTEM LOGIC

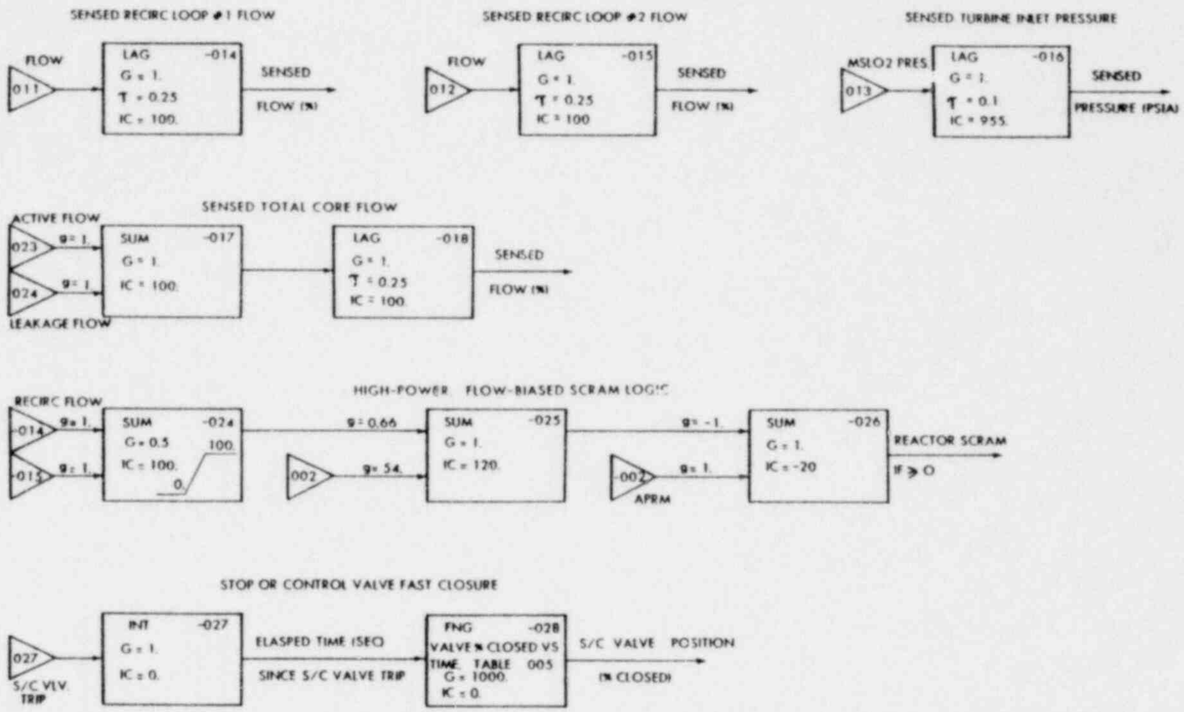

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

779 245

RETRAN CONTROL MODELS
FEEDWATER & LEVEL CONTROLLER

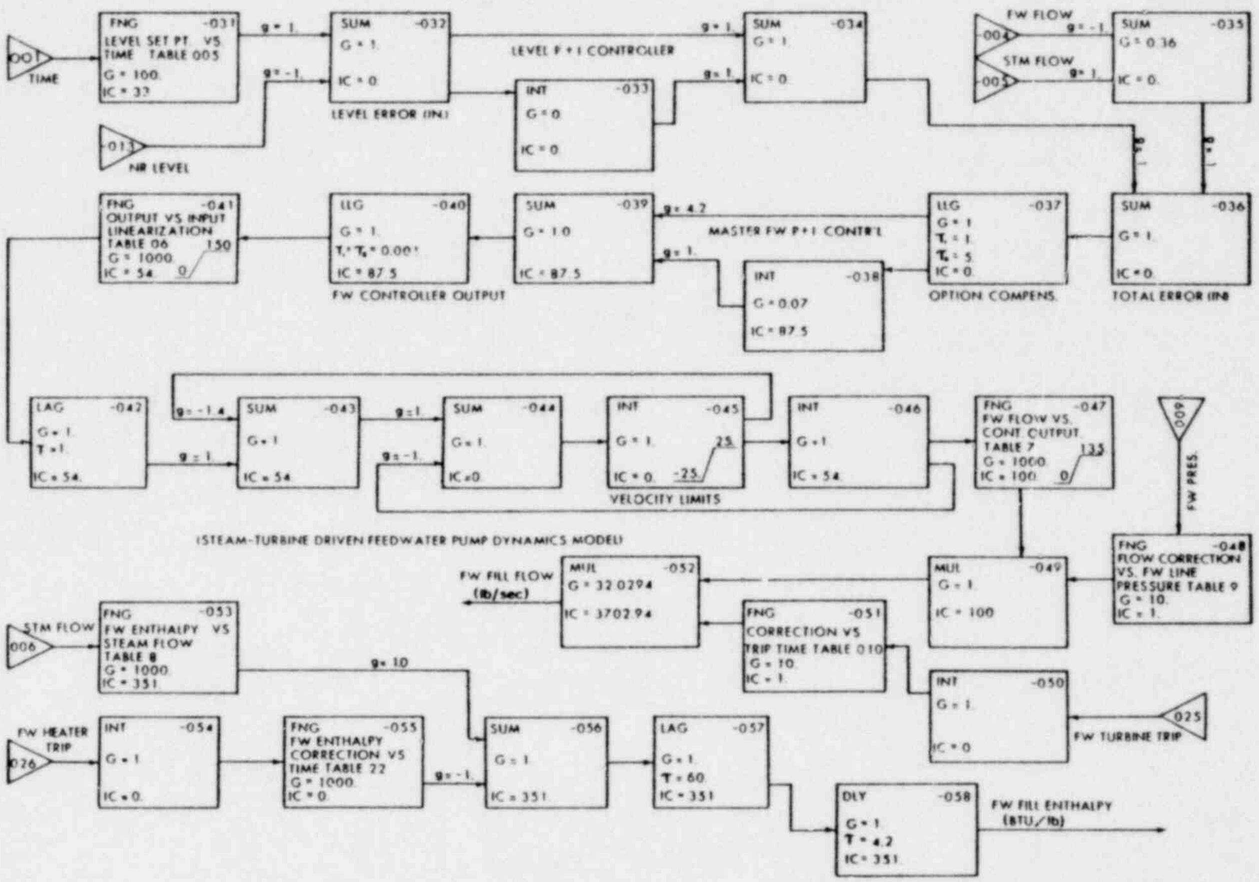

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

1779 246

RETRAN CONTROL MODELS

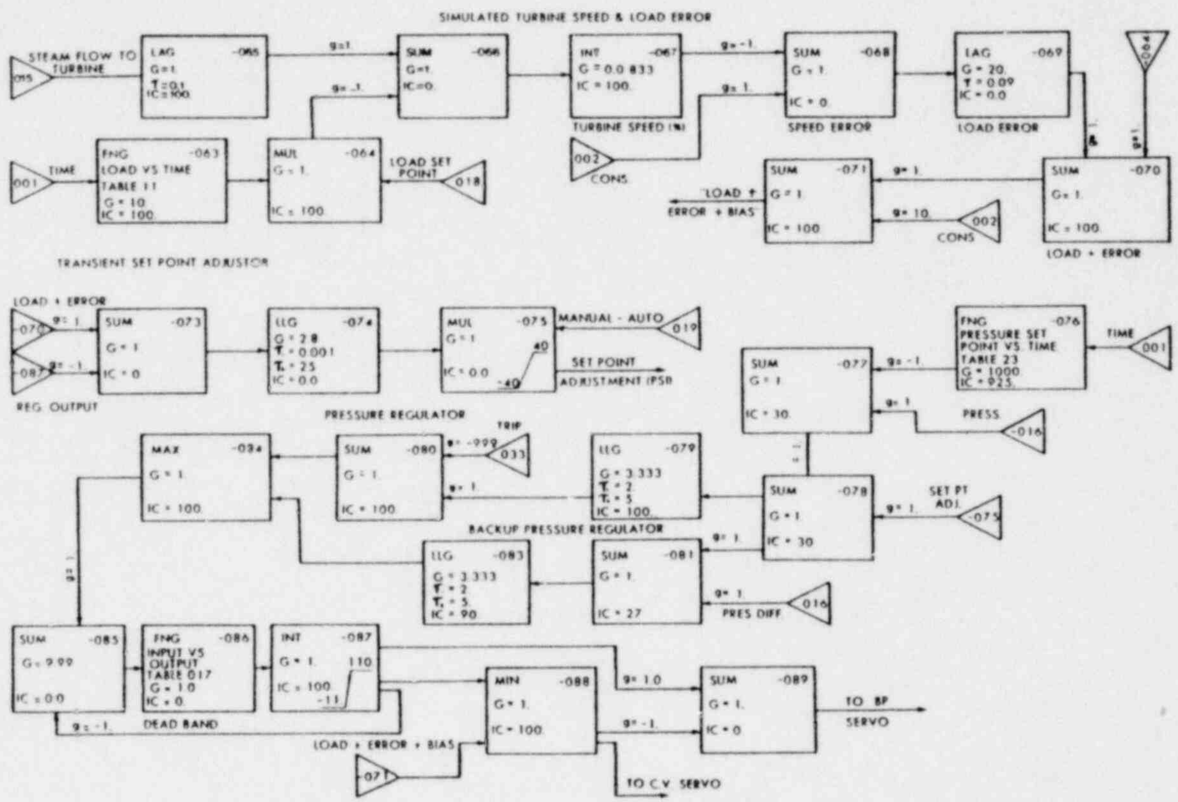

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

9 247

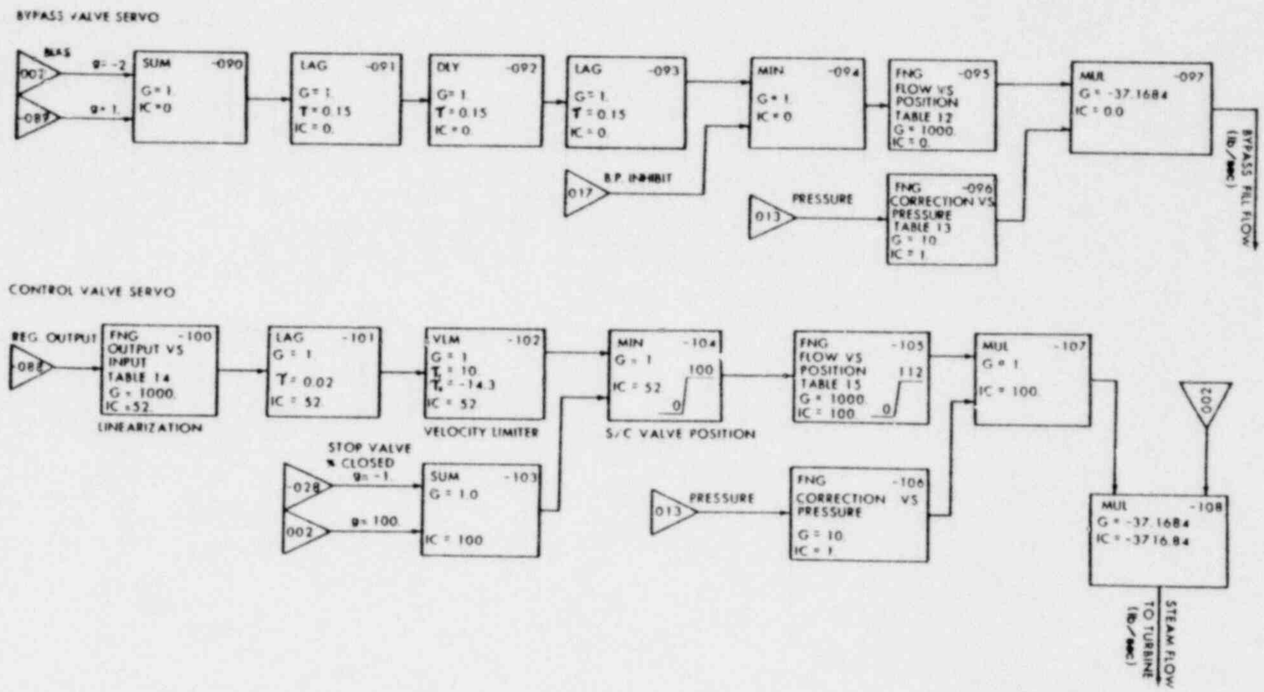

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

1259 248

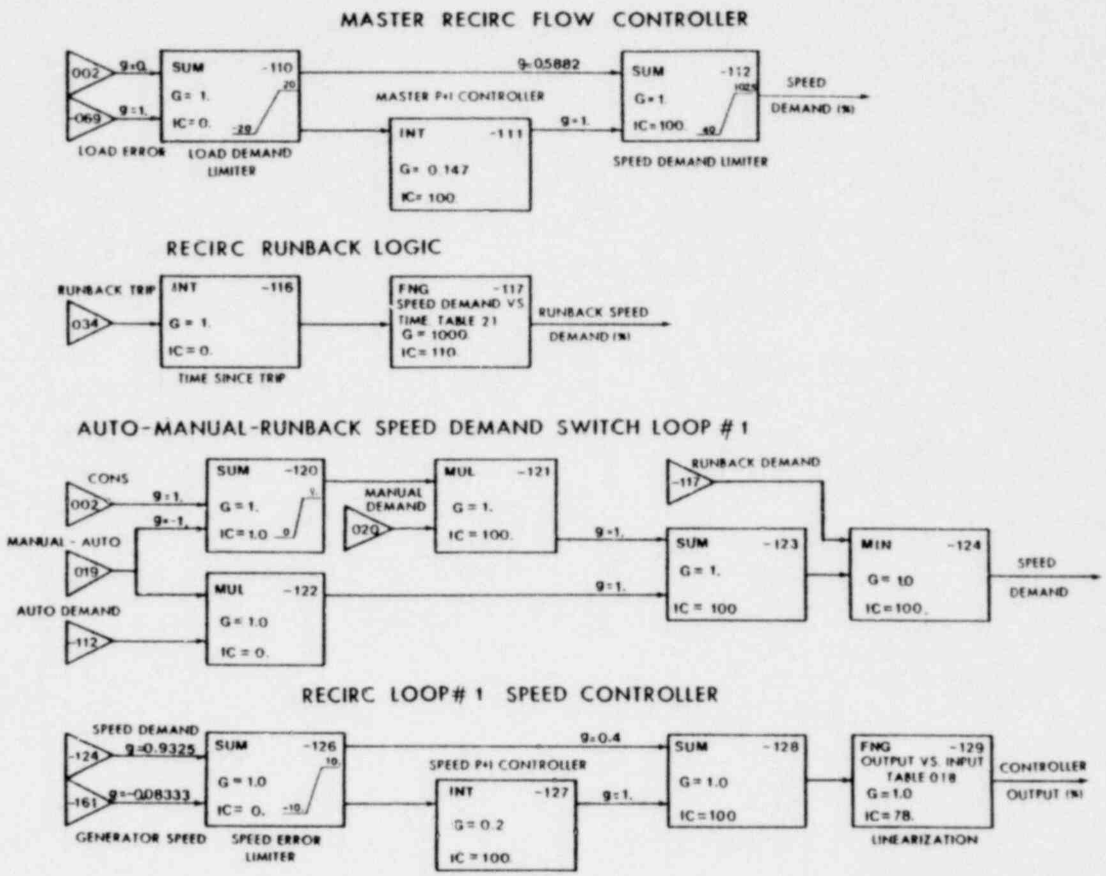

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

9 249

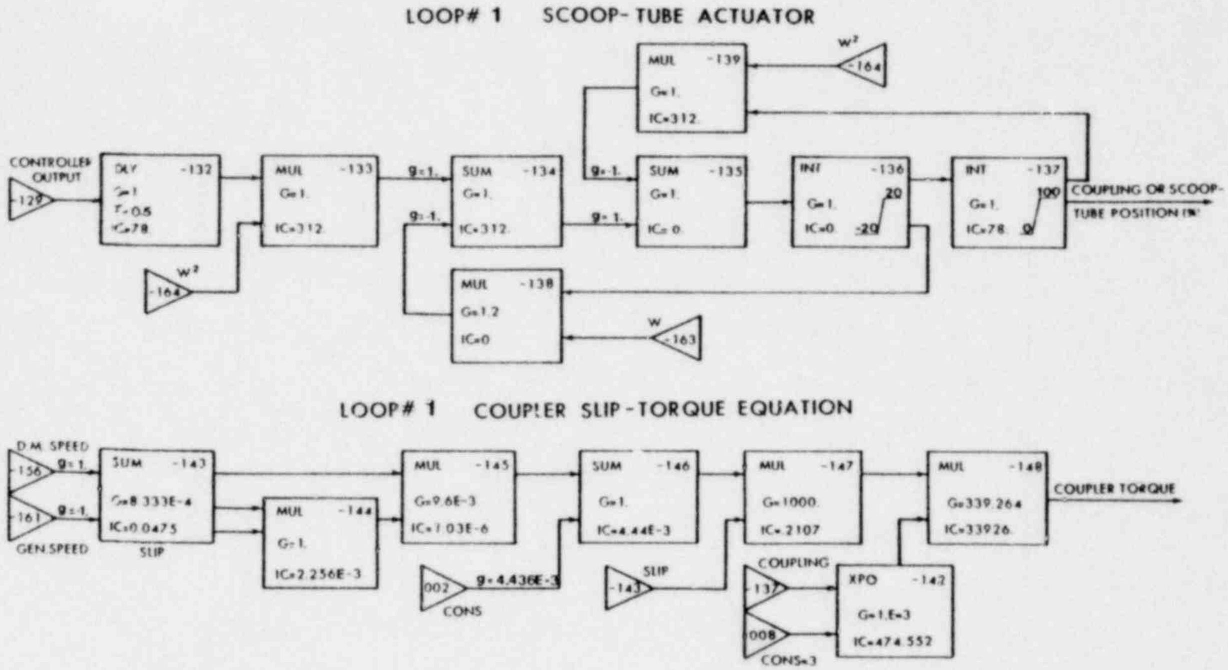

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

779 250

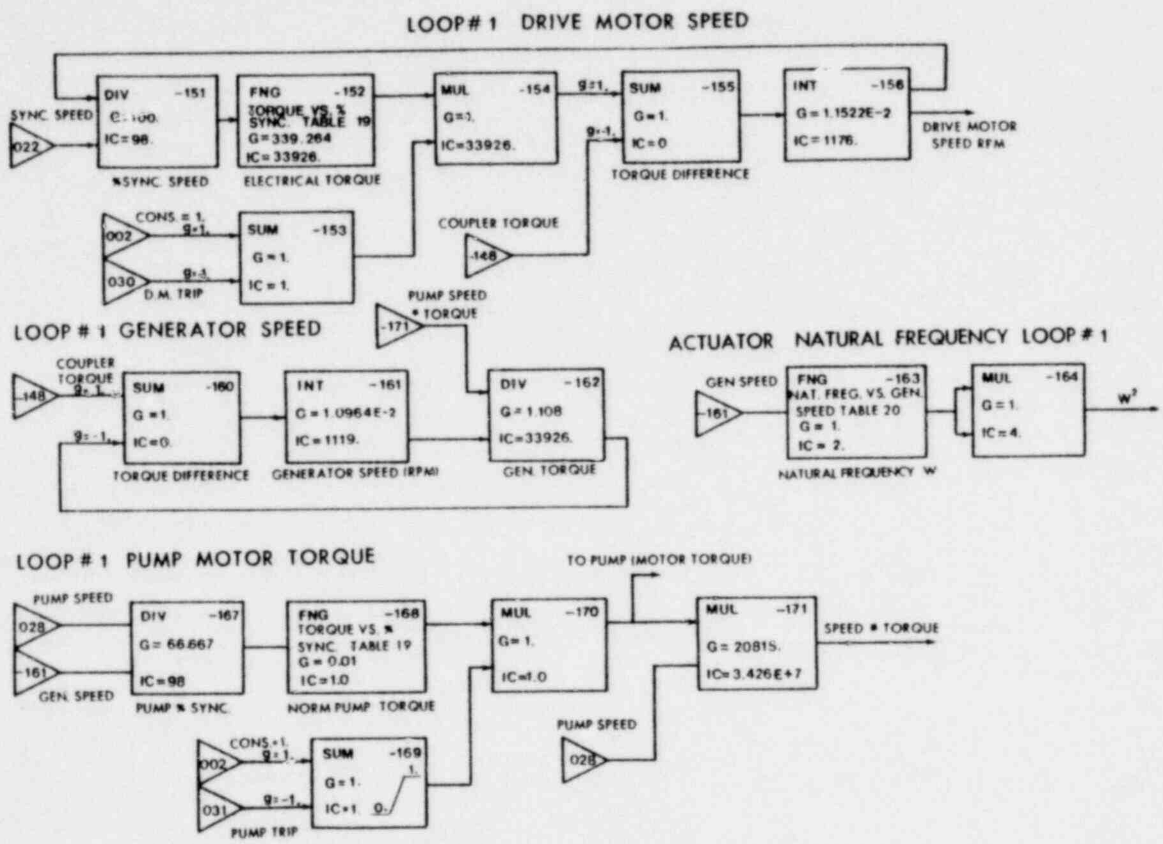

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

1759 258

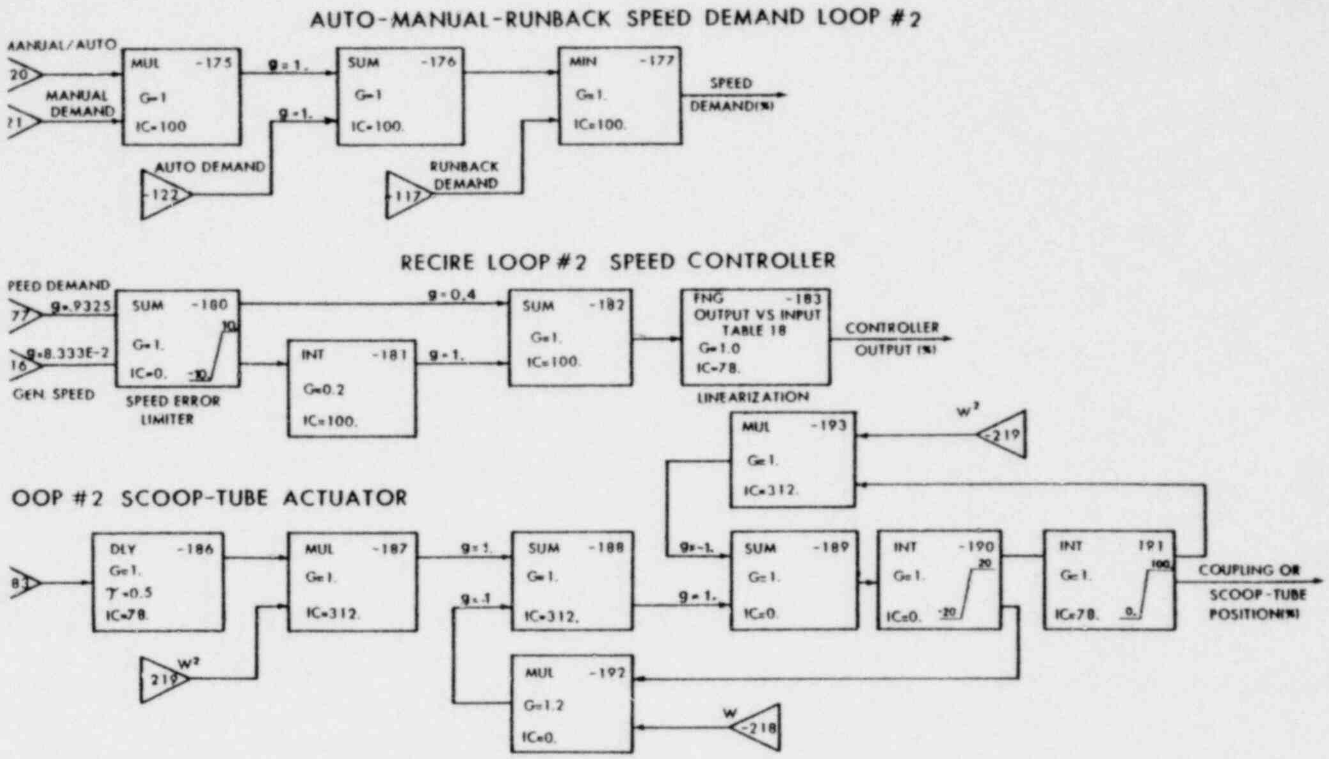

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

252

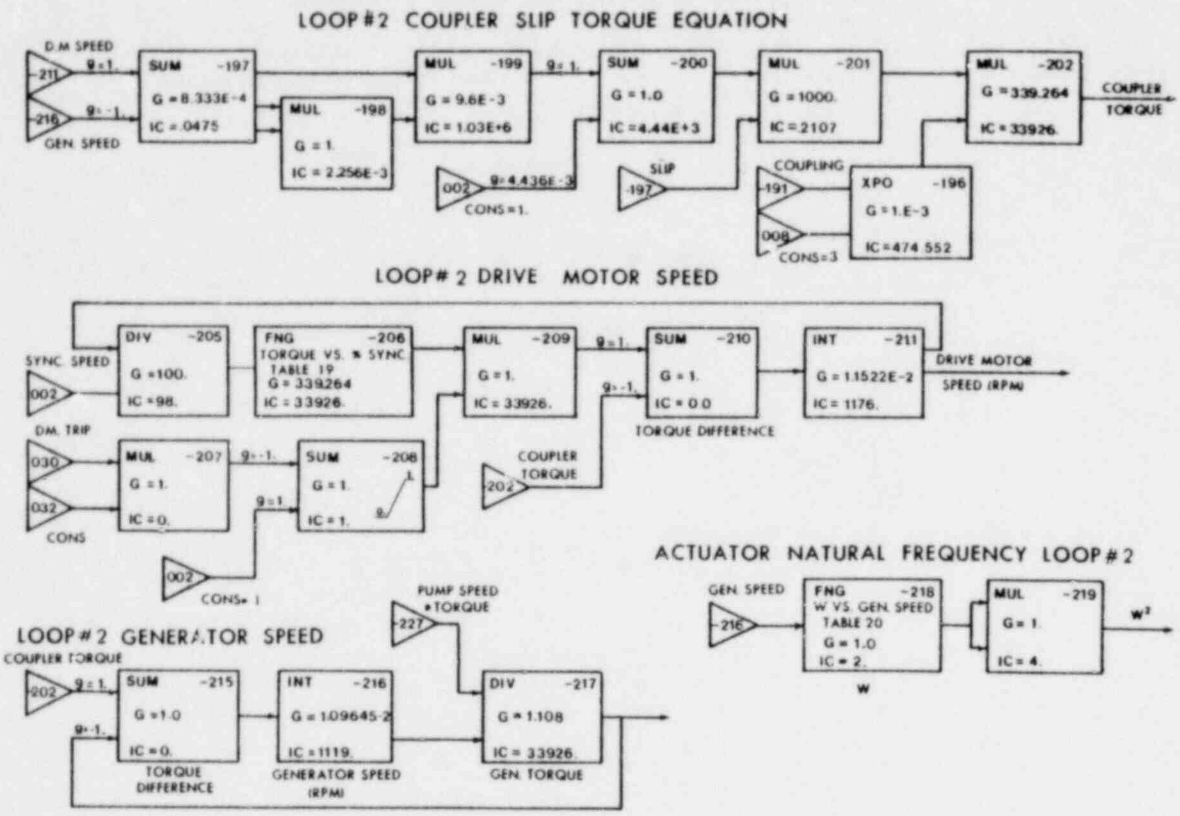

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

9 253

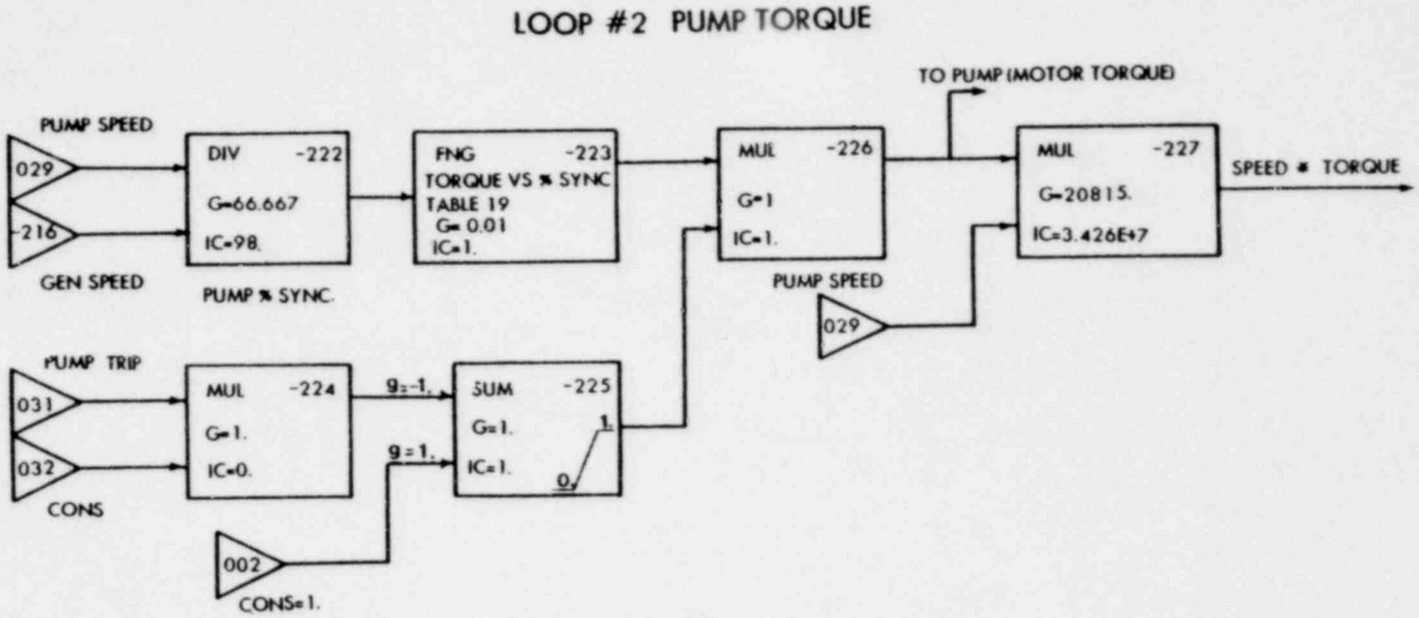

FIGURE V.2-4 RETRAN Control Model Representation for TVA System (cont'd)

7909 254

Comparisons between the test data and RETRAN are shown in Figures V.2-5 to V.2-10. In general, the agreement is quite good. The power response predicted by RETRAN, above the data, may be due to values of void reactivity and scram reactivity which are too low. The observed and calculated values of steam dome pressure (Figure V.2-6) are in good agreement for the first 7 seconds. Between 7 and 12 seconds, the underprediction by RETRAN was attributed to a late closing of the relief valves and a slow reduction in steam flow during closure of the main steam isolation valves. The overprediction of pressure after 12 seconds is due to the infinite operation time assumption in RETRAN for decay heat. The burnup at test time was only 900 MWD/T, and when an analysis was performed without decay heat, " . . . the calculated dome pressure rise at 16 seconds was slightly less than the measured value" [V.2-1].

Values for steamline flow are shown in Figure V.2-7. It is noted that the MSIV's closed at 10 seconds (Table V.2-1), thus the measured value as reported here is in error. This may have resulted from the manual reduction of the strip chart data. The water level response predicted by RETRAN (Figure V.2-8) is generally above the test values. The reason for this discrepancy is not known. The feedwater flow and core flow predictions from RETRAN show good agreement with the experimental data.

2.2 Summary of Results

The comparisons between RETRAN calculations and the test data show that the approach used by TVA to model the generator load rejection transient is valid. The major differences in the comparisons are attributable to different conditions (e.g., operating time for decay heat calculation) rather than code deficiencies.

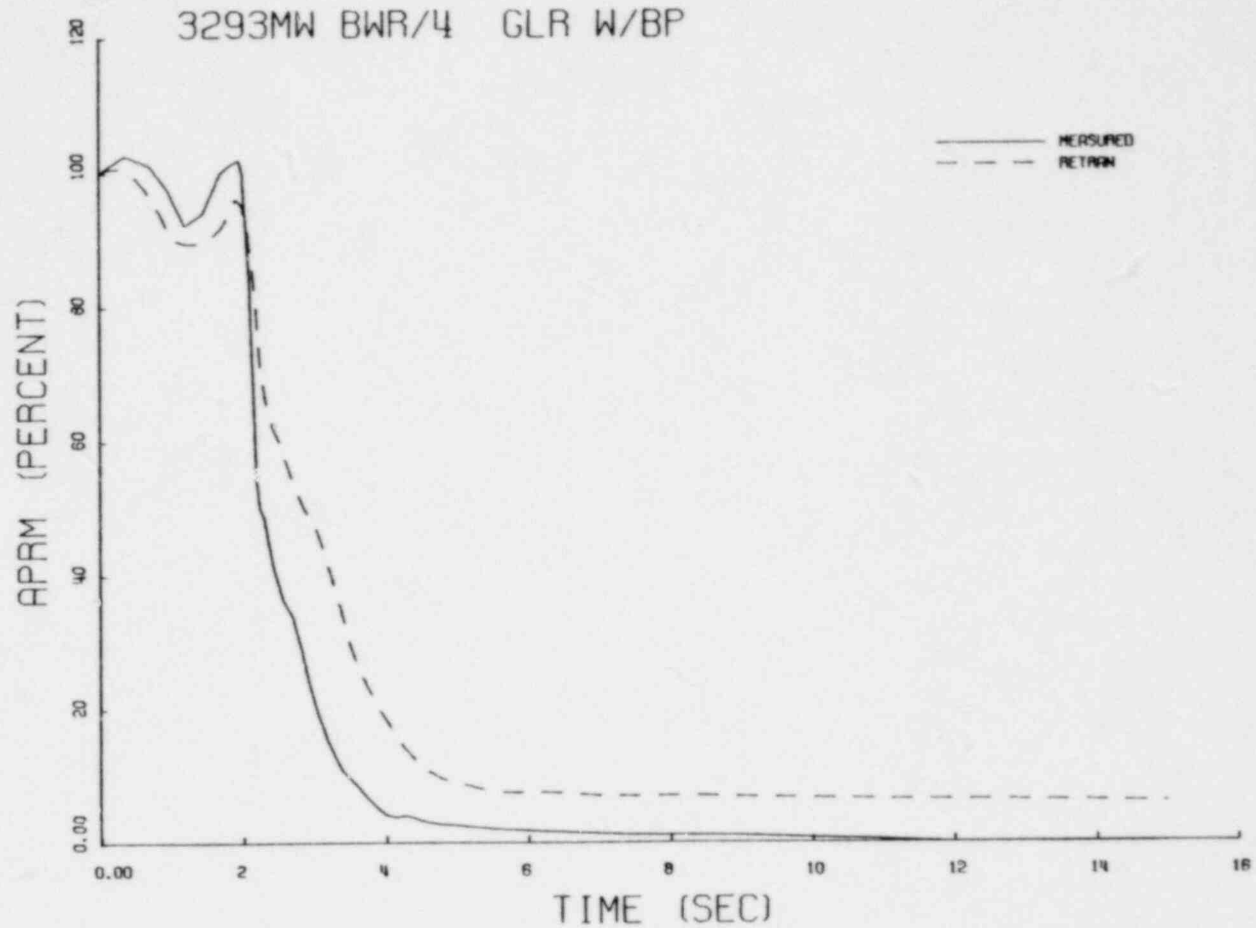

FIGURE V.2-5 System Power Response - Generator Load Rejection

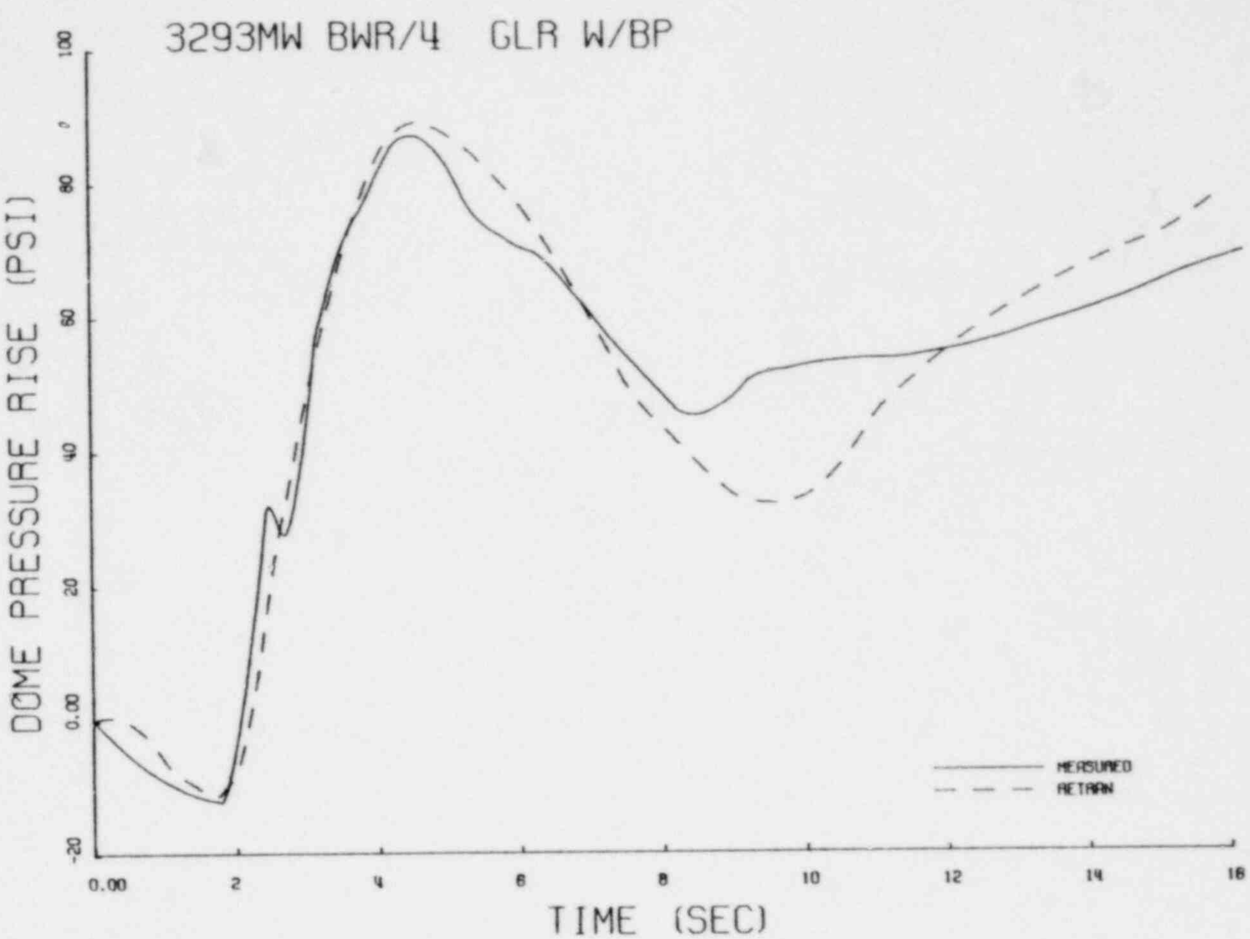

FIGURE V.2-6 Steam Dome Pressure Rise - Generator Load Rejection

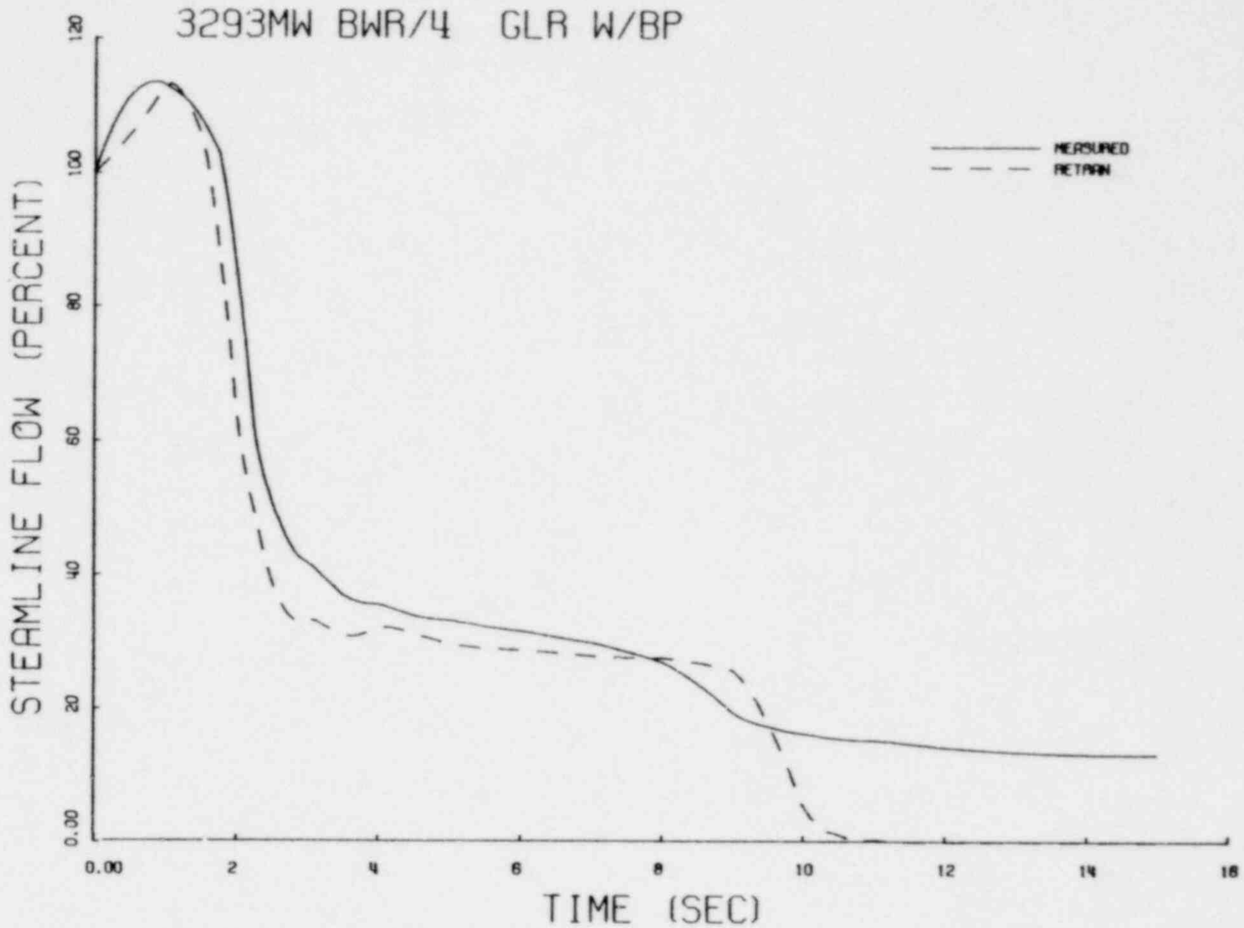

FIGURE V.2-7 Steamline Flow - Generator Load Rejection

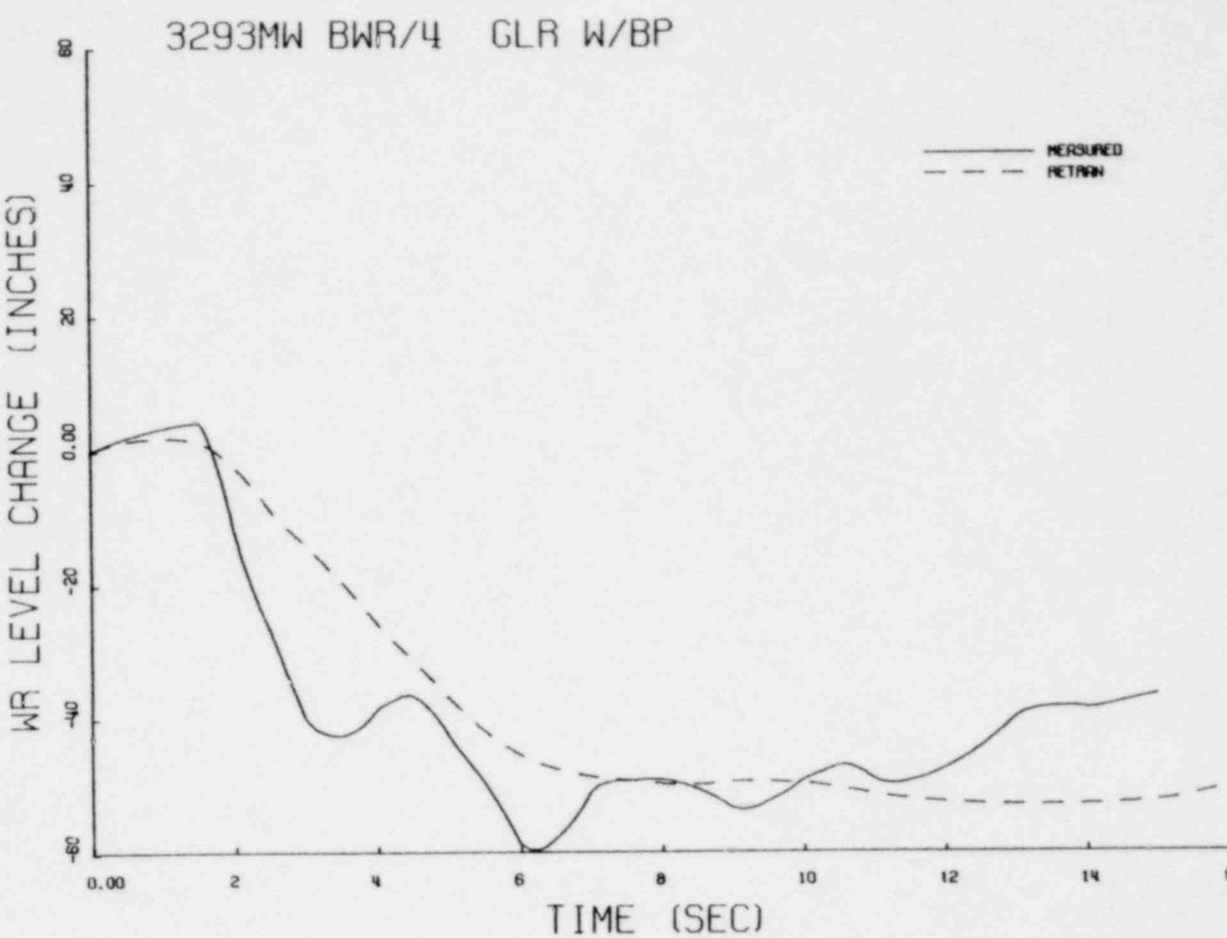

FIGURE V.2-8 Water Level Change - Generator Load Rejection

1759 259

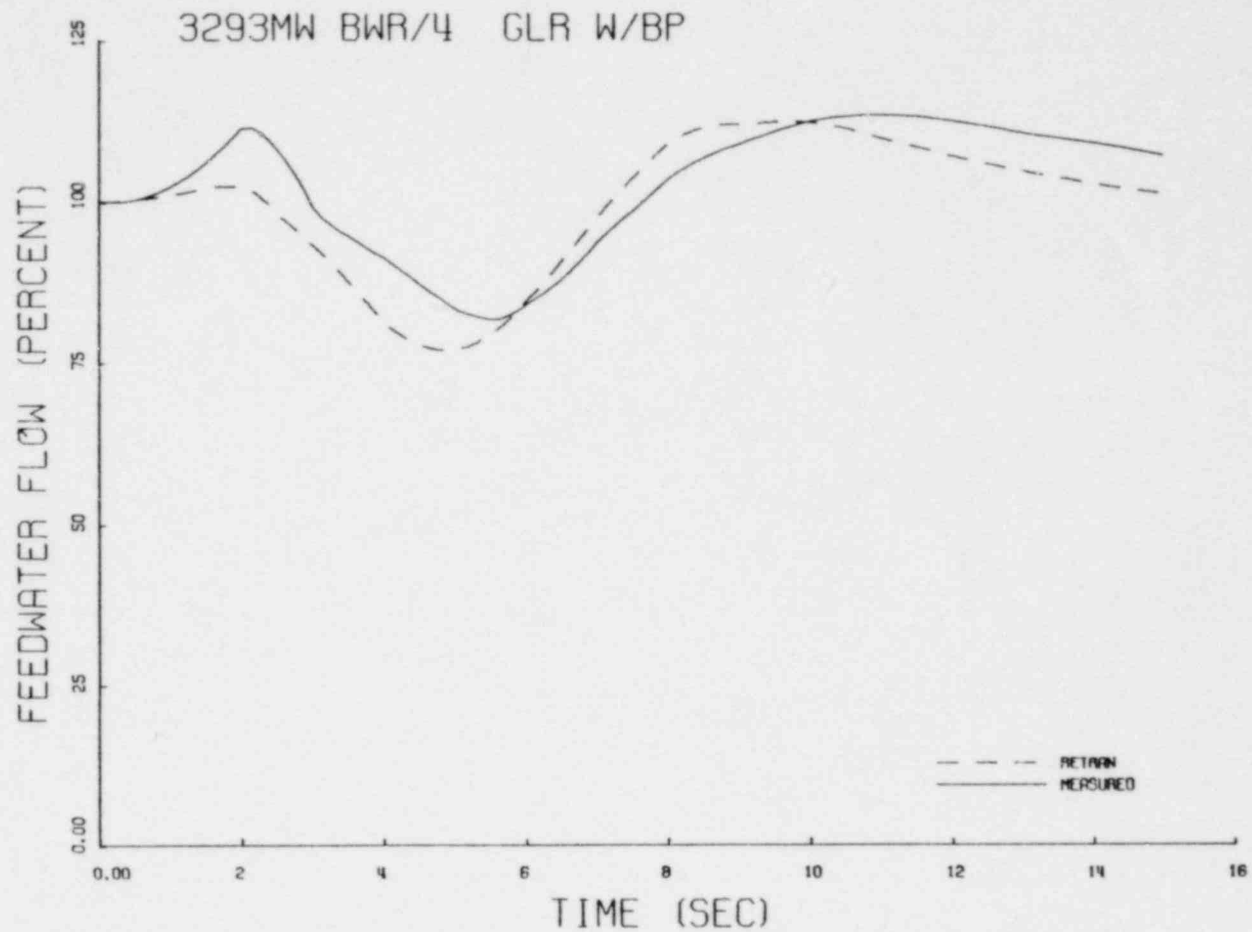

FIGURE V.2-9 Feedwater Flow - Generator Load Rejection

1779 240

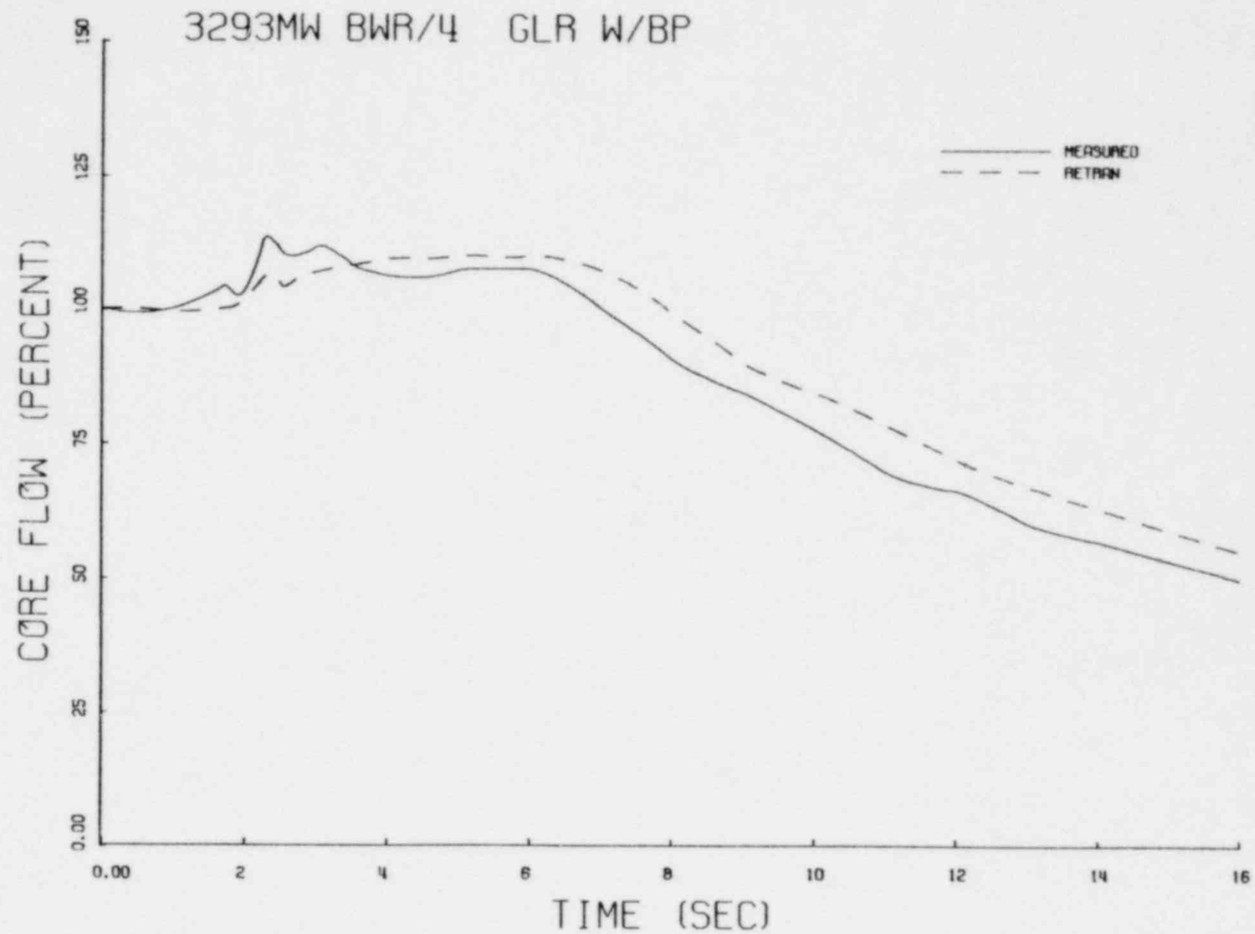

FIGURE V.2-10 Core Flow - Generator Load Rejection

9 261

3.0 FEEDWATER TURBINE TRIP

The feedwater pumps on BWR/4 systems are driven by steam turbines. The transient analyzed in this section results from tripping one of the three turbine-driven pumps. This transient produces a decrease in the reactor vessel coolant inventory.

In this transient, an action occurs which results in a sharp decrease in the feedwater flow. The turbines which are not tripped increase in speed to continue the feedwater flow, although the flow rate is reduced. After the water level drops, runback of the recirculation pumps begins and the core flow decreases. As the steam and core flows decrease, the power level also decreases. These flows eventually level off near the end of the transient.

The Tennessee Valley Authority (TVA) performed this transient during startup testing for one of the Brown's Ferry plants. Data from this test were compared to a RETRAN analysis of the transient [V.3-1]. The analysis, along with an analysis of a water level set point change transient which was performed to check the feedwater controller model, is discussed in the following section.

3.1 TVA Analysis

Prior to analyzing the feedwater turbine trip test, a startup test (3-inch water level setpoint change) of the feedwater controller system was analyzed. One purpose of this test was to optimize the settings for the feedwater controller prior to analyzing the turbine trip. In the turbine trip test, the steam flow to the feedwater turbine was stopped and the turbine and pump coast down. The sequence of events which occurred during the test are given in Table V.3-1.

3.1.1 Description of Model

The geometric model of the reactor system for this transient and the RETRAN input are the same as discussed in Section V.2 and Reference V.3-2 with the exception of those items (e.g. trips) which are unique to these analysis. The same self-initialization was also used for these analyses.

9 262

TABLE V.3-1

SEQUENCE OF EVENTS FOR TVA FEEDWATER TURBINE TRIP TRANSIENT

Time (sec)	Events
0.0	One of the feedwater turbines is tripped. Feedwater flow drops sharply.
4.0	Feedwater flow begins to increase due to the increase in speed from the two on-line turbines.
17.5	Water level has dropped approximately 10 inches. Runback of the recirculation pumps to 70 percent of rated speed begins.
25.5	Core flow begins to drop due to the decrease in the recirculation flow.
30.5	Water level has dropped 13 inches, well above the scram setpoint and begins to increase.
30.5-65.0	Steam and core flow gradually decreasing due to the runback of the recirculation pumps. Power level decreases due to the decrease in core flow. Water level gradually increases.
65.0	Core flow steam flow level off.
70.0	End transient simulation.

9 243

The water level setpoint change transient is initiated by a very rapid drop in the level setpoint. This is modeled by inputing the water level setpoint for the controller as a function of time. During the transient, the water is brought down to the new setpoint level with the level controller and the feedwater system.

In modeling the separator, TVA used a very low (0.0001) value for mixture quality. This assumes very little steam carryunder from the separator, with the result that the input parameters for the bubble rise model are used directly. The elevation of the junction between the downcomer and the separation region was chosen such that the water level would not fall below this junction elevation.

The TVA control model does not presently allow an explicit representation of a feedwater turbine coastdown. Thus, the feedwater flow was modified as a function of time. In addition, the recirculation pump runback was assumed to occur if the water level dropped a fixed amount (10 inches) after the time estimated for the turbine to coast down to 20 percent of the rated speed.

3.1.2 Results of Analyses and Comparisons With Data

3.1.2.1 Level Setpoint Change Transient

The results of the level setpoint transient are compared with test data in Figures V.3-1 and V.3-2. The water level calculated by RETRAN agrees very well with the test data (Figure V.3-1). The feedwater flow calculated by RETRAN (Figure V.3-2) is in good agreement with the test data except in the early part of the transient (3 to 8 second), where the maximum deviation at 5 seconds is approximately 3 percent.

3.1.2.2 Feedwater Turbine Trip Transient

The calculated and measured values of feedwater flow for the feedwater turbine trip transient are shown in Figure V.3-3. The agreement is very good until approximately 50 seconds. This discrepancy is thought to be due to the RETRAN model by Winkler et al.[V.3-1]. After 50 seconds, the tripped turbine has coasted down and the remaining two turbines are operating at maximum capacity. Under this condition, a large change in the feedwater controller output is

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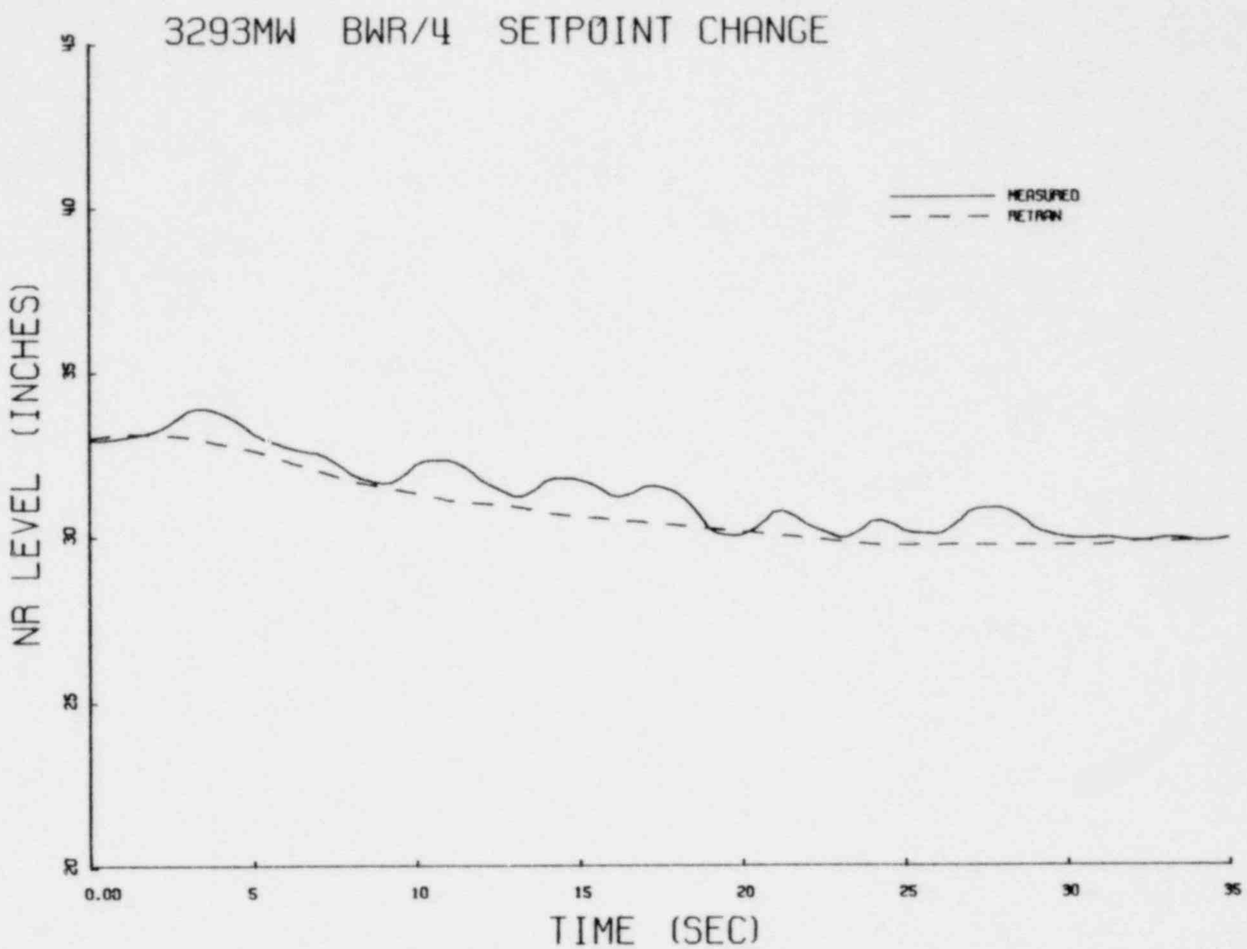

FIGURE V.3-1 Water Level - Setpoint Change

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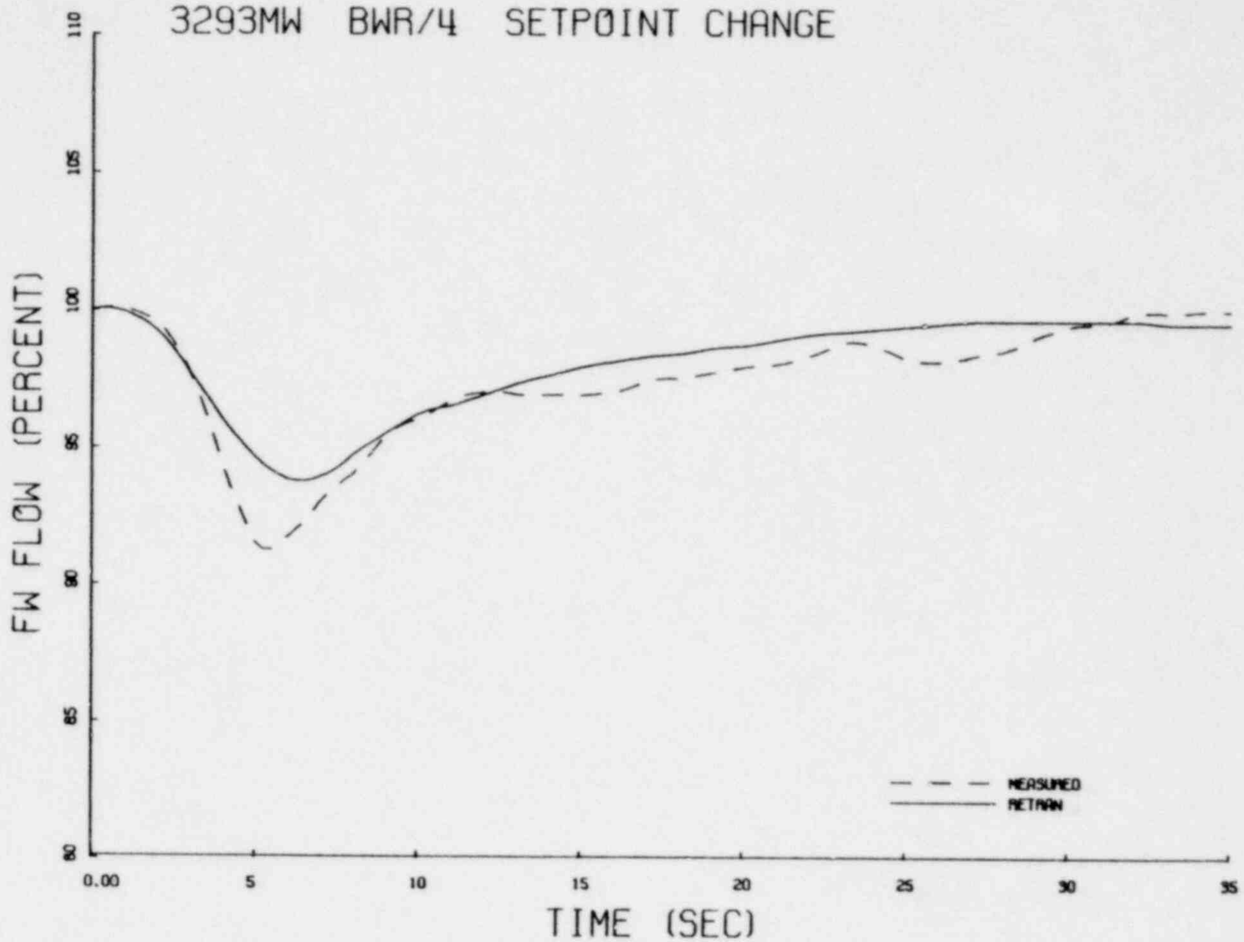

FIGURE V.3-2 Feedwater Flow - Setpoint Change

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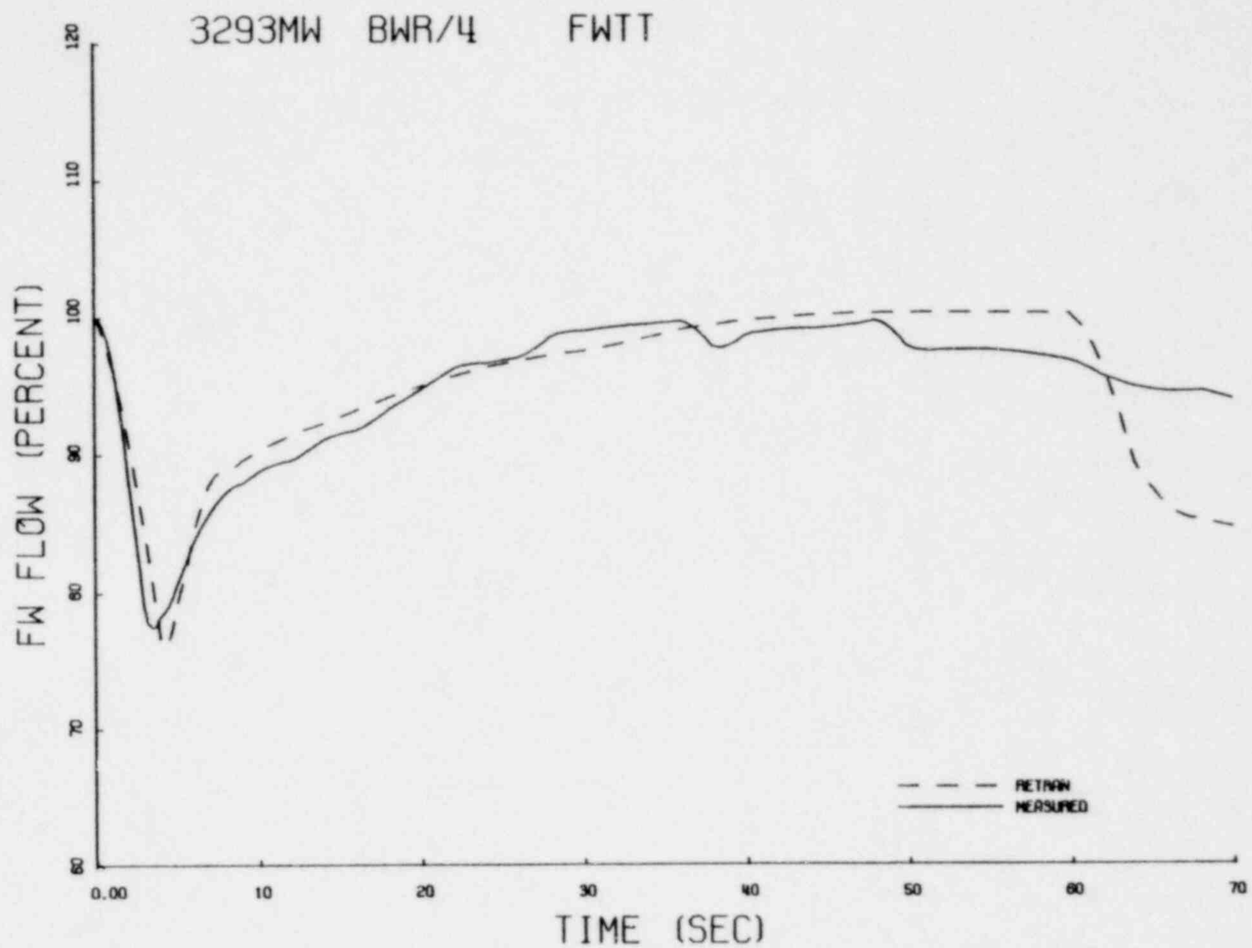

FIGURE V.3-3 Feedwater Flow - Feedwater Turbine Trip

9 267

required to produce a small change in flow rate. It is not clear if the result is due to a deficiency in the RETRAN control models or in the modeling of the a.tual control system.

The core power, as measured by the average power range monitor and calculated in RETRAN, is shown in Figure V.3-4. There is good agreement until recirculation pump runback occurs. This could be due to void reactivity values which are too negative and/or due to the low core flow rate predicted by RETRAN (Figure V.3-5) after runback begins. Another run also was performed using the enthalpy transport delay model in the feedwater and recirculation lines and the lower downcomer. However that analysis gave a peak power which was greater than the measured value and which occurred later in the transient.

Figure V.3-6 shows the calculated and measured steam flow values, with good agreement until about 30 seconds. The trend after this time could be due to the lower power predicted by RETRAN as well as the lower pressure (Figure V.3-7). The change in steam dome pressure calculated by RETRAN was approximately 10 psi greater than was observed experimentally.

The narrow range water level data are shown with RETRAN results in Figure V.3-8. RETRAN slightly underpredicted the actual water level for 45 seconds. After that time (beginning really at about 35 seconds) RETRAN produced a faster rate of increase in level than was measured. This probably is due to the same difficulties discussed for the feedwater flow comparisons.

3.2 Summary of Results

The RETRAN results for the feedwater turbine trip are in good agreement with the experimental data for about the first 30-35 seconds of the transient. Some discrepancies can be attributed to input data (void reactivity) while the reason for others, especially the feedwater flow rate, is not definitely known. Additional studies of the feedwater turbine and controller models are required to make definitive statements on this parameter.

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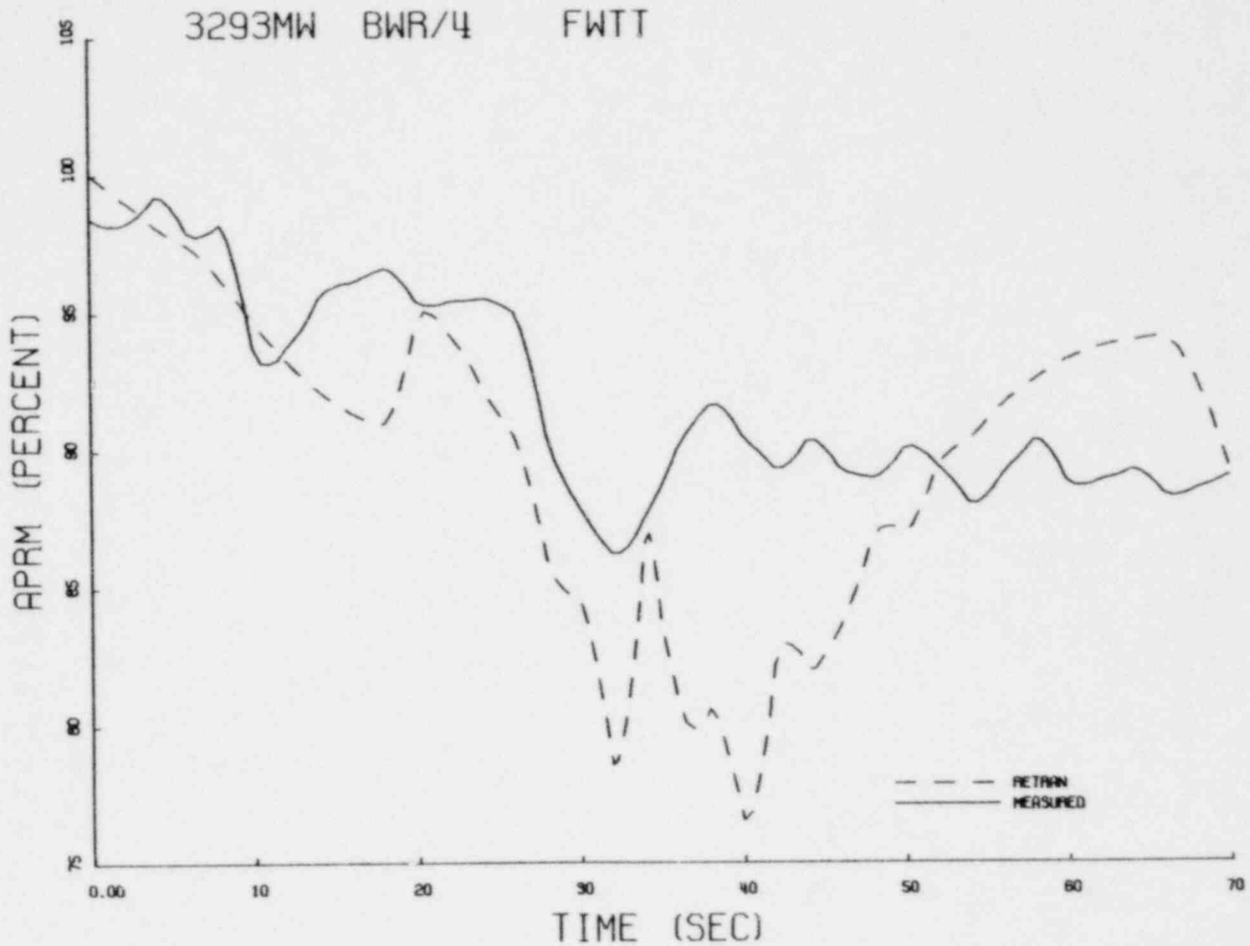

FIGURE V.3-4 Average Power - Feedwater Turbine Trip

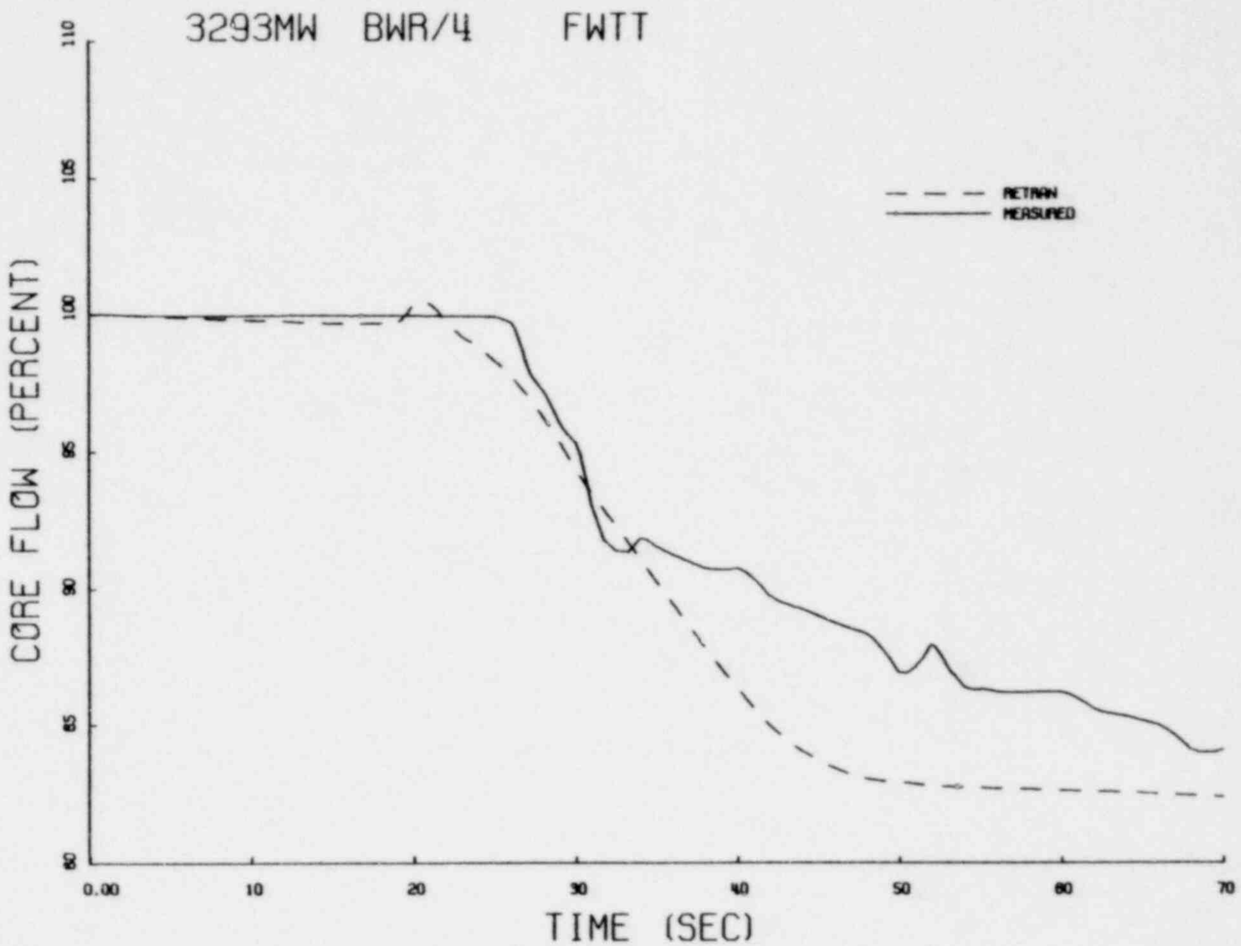

FIGURE V.3-5 Core Flow - Feedwater Turbine Trip

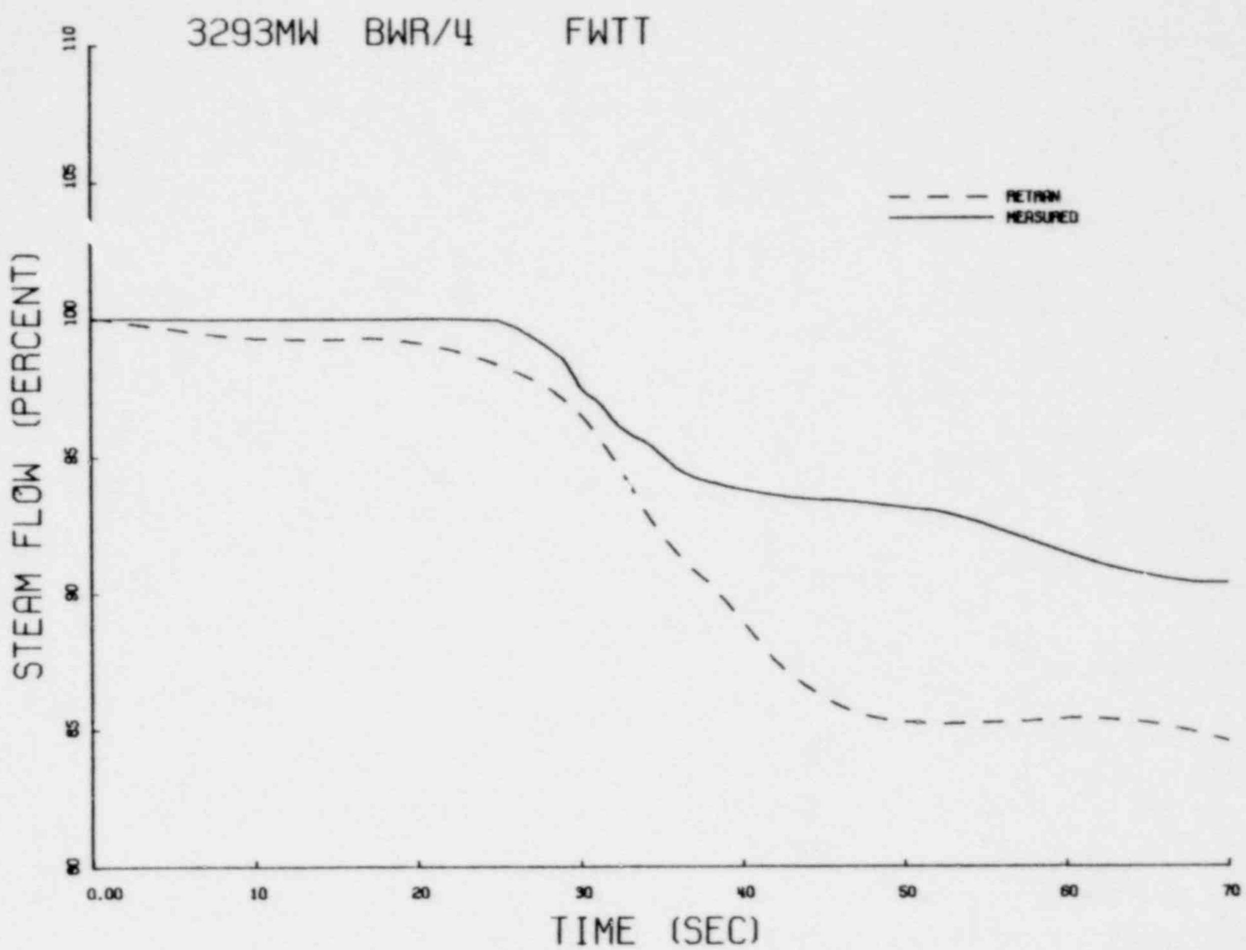

FIGURE V.3-6 Steam Flow - Feedwater Turbine Trip

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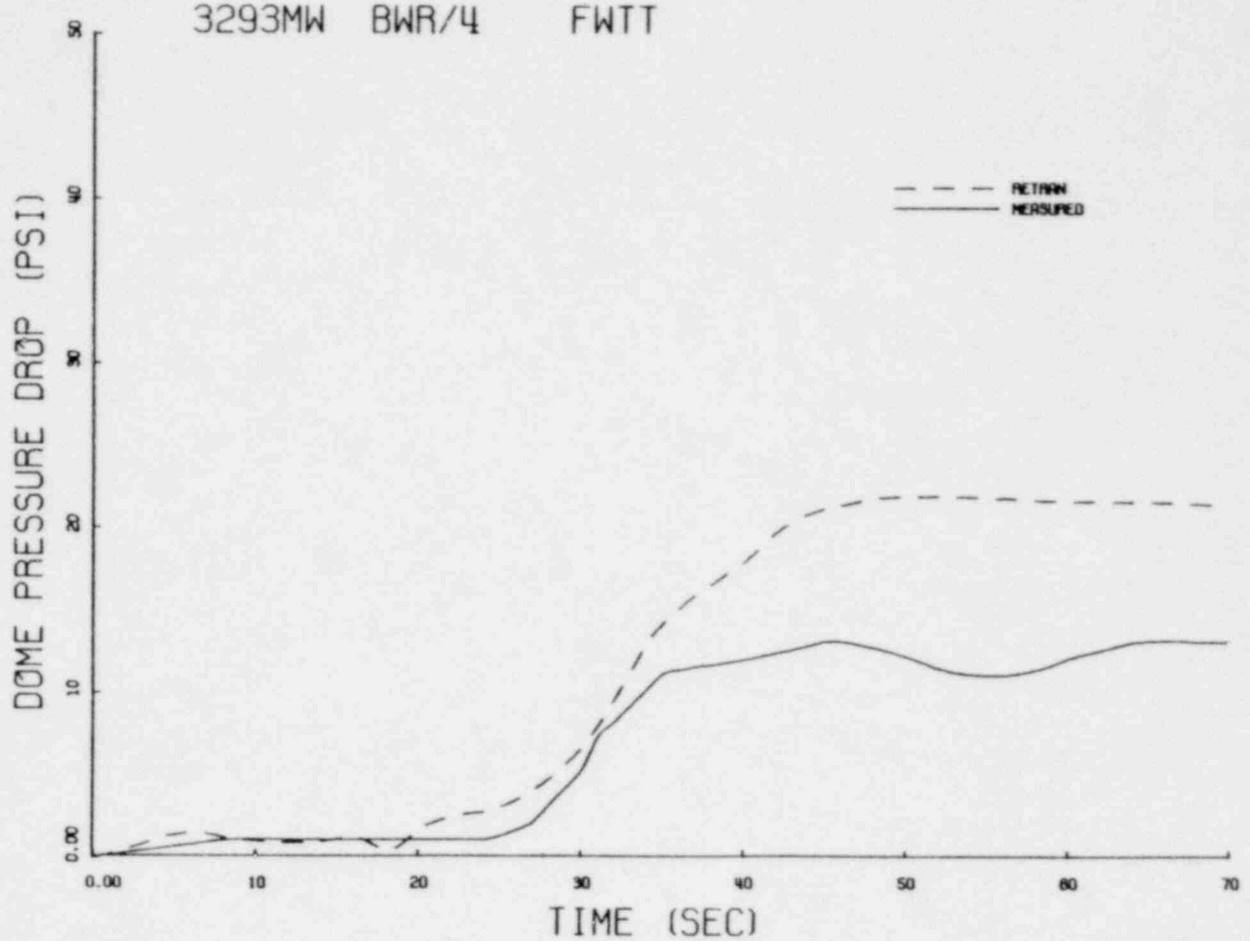

FIGURE V.3-7 Dome Pressure Drop - Feedwater Turbine Trip

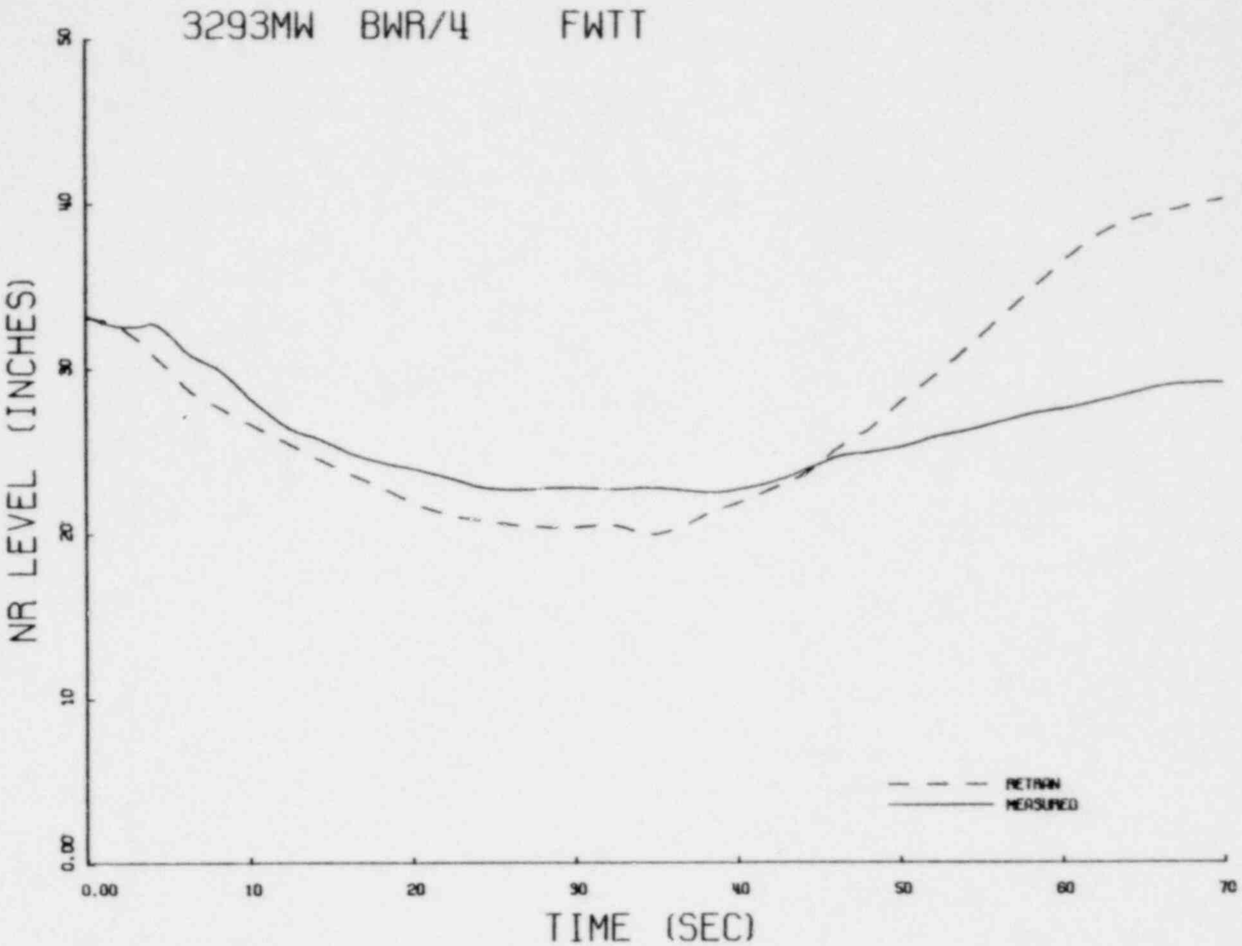

FIGURE V.3-8 Water Level - Feedwater Turbine Trip

4.0 RECIRCULATION PUMP TRIP

The dominant result of a recirculation pump trip transient is a decrease in the core coolant flow. Two transients are included in this group; (1) a single-pump trip and (2) a two-pump trip. When the power supply to pump motor(s) is disrupted, the pump(s) coasts down at a rate dependent on the component and fluid inertia and the hydraulic resistance of the system. The recirculation loop(s) and core flows decrease resulting in an increase in core voids. The negative void reactivity produces a reduction in core power terminating the transient.

The Tennessee Valley Authority performed two analyses of this transient [V.4-1]. In the first case, both recirculation pumps were tripped while in the second case, only one pump was tripped and the second continued to operate at rated conditions. These analyses and comparisons with test data are summarized in the following section.

4.1 TVA Analysis

The transients analyzed by TVA are similar to the ones discussed above.

4.1.1 Description of Model

The RETRAN model for this analysis is the same as described in Section V.2 and Reference V.4-2. The only changes are the trip cards unique to this transient.

4.1.2 Results of Analyses and Comparisons with Data

4.1.2.1 Two-Pump Trip Transient

Test data were taken during startup testing for both the single- and two-pump trip transients. The data were manually scaled from strip chart recordings for use in the report [V.4-1]. The pump speed and recirculation loop drive flow for the two-pump trip are shown in Figures V.4-1 and V.4-2. The calculated values agree very well with the test data.

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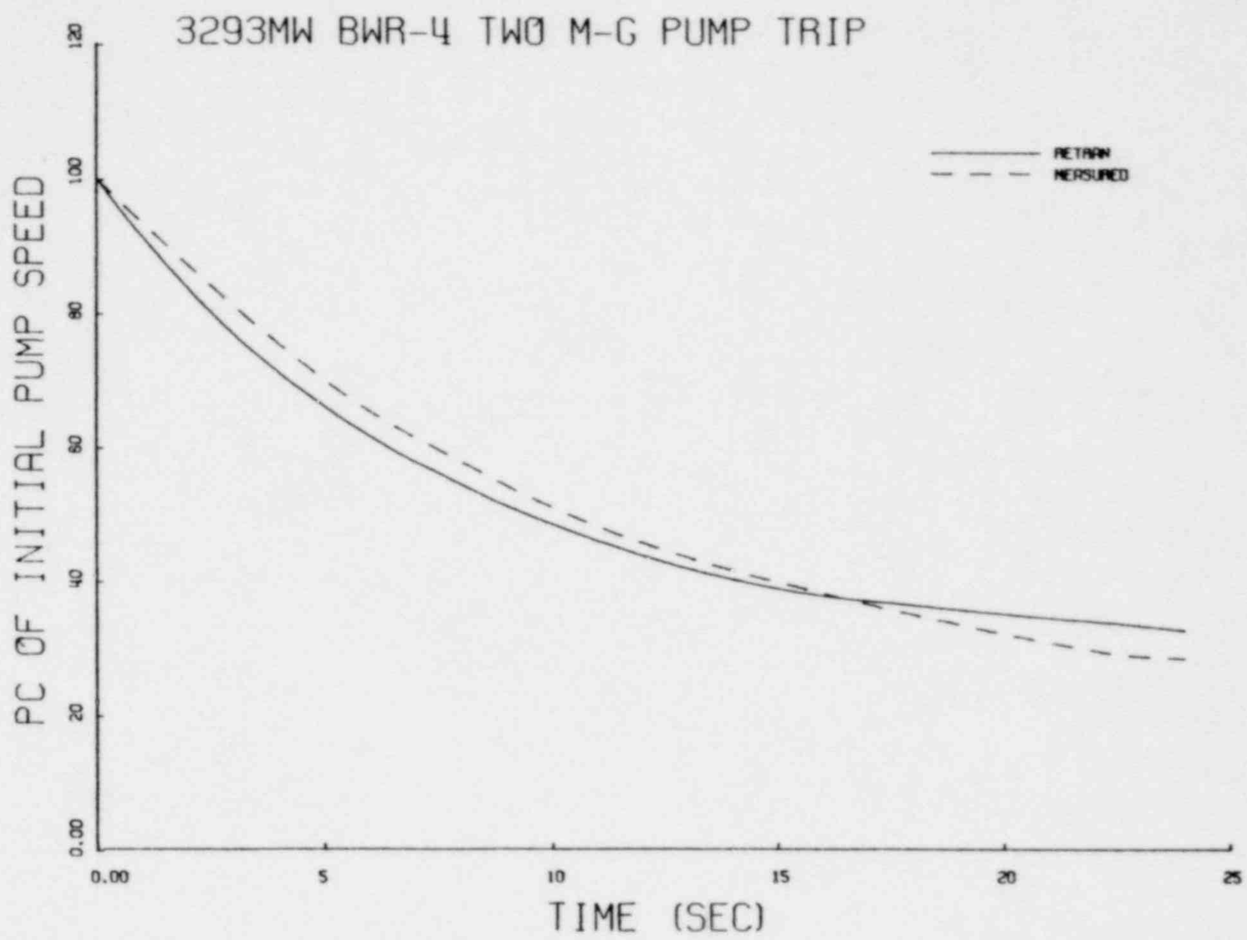

FIGURE V.4-1 Change in Pump Speed for Two Pump Trip

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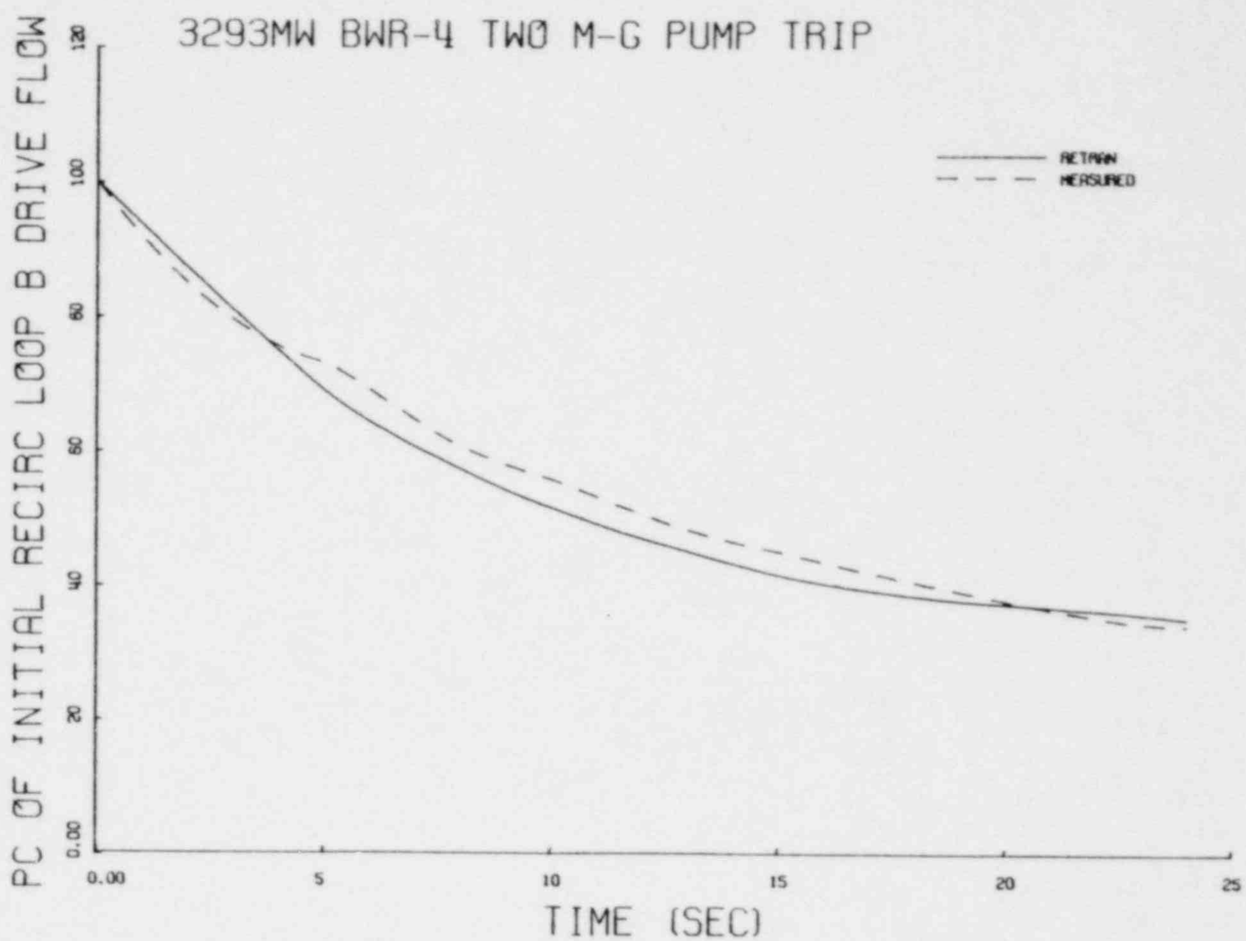

FIGURE V.4-2 Drive Flow for Recirculation Loop B

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Comparison between the RETRAN and test values of core inlet flow is shown in Figure V.4-3, and the agreement is excellent. The authors noted that this is an indication that the jet pumps are modeled correctly. The core power response is shown in Figure V.4-4, with the RETRAN calculation giving good comparisons with the data until about 10 seconds. After this time, RETRAN underpredicts the power. It is not certain if this is due to the void reactivity data or the sytem pressure response (Figure V.4-5). Bell et al. [V.4-1] noted that data for the pressure control system function generator used in the analyses are from a different plant than the test data.

The steam flow rate (Figure V.4-6) also is affected by the function generator which may be the cause of the underprediction. The feedwater flow predicted by RETRAN agrees reasonably well with the test data (Figure V.4-7) until about 15 seconds. At this time, RETRAN shows a higher value of sensed water level increase (Figure V.4-8) and thus will have a lower feedwater flow than is observed from the test. In general, the RETRAN calculation gives good agreement with the two-pump trip data.

4.1.2.2 Single-Pump Trip Transient

Results of the single-pump trip test and analysis are shown in Figures V.4-9 to V.4-18. The single-pump trip results are similar to the two-pump trip with one exception. At about 12 seconds, RETRAN predicted a sudden increase in the recirculation loop flow. Also at this time, RETRAN predicted that the jet pump flow in the tripped loop had reversed. This flow reversal affected the core inlet flow, core power and water level rather significantly. The modeling of this reverse flow condition requires further investigation, especially since the flow through jet pumps is actually two-dimensional, and the RETRAN equations are for one-dimensional flow conditions.

4.2 Summary of Results

RETRAN calculations gave good agreement with the experimental data for the two-pump trip test. An anomalous prediction of reverse flow through the jet pump for the single-pump trip analysis resulted in rapid changes of core inlet flow and power. The modeling of jet pumps for both forward and reverse flow requires further study based on the analysis of this transient.

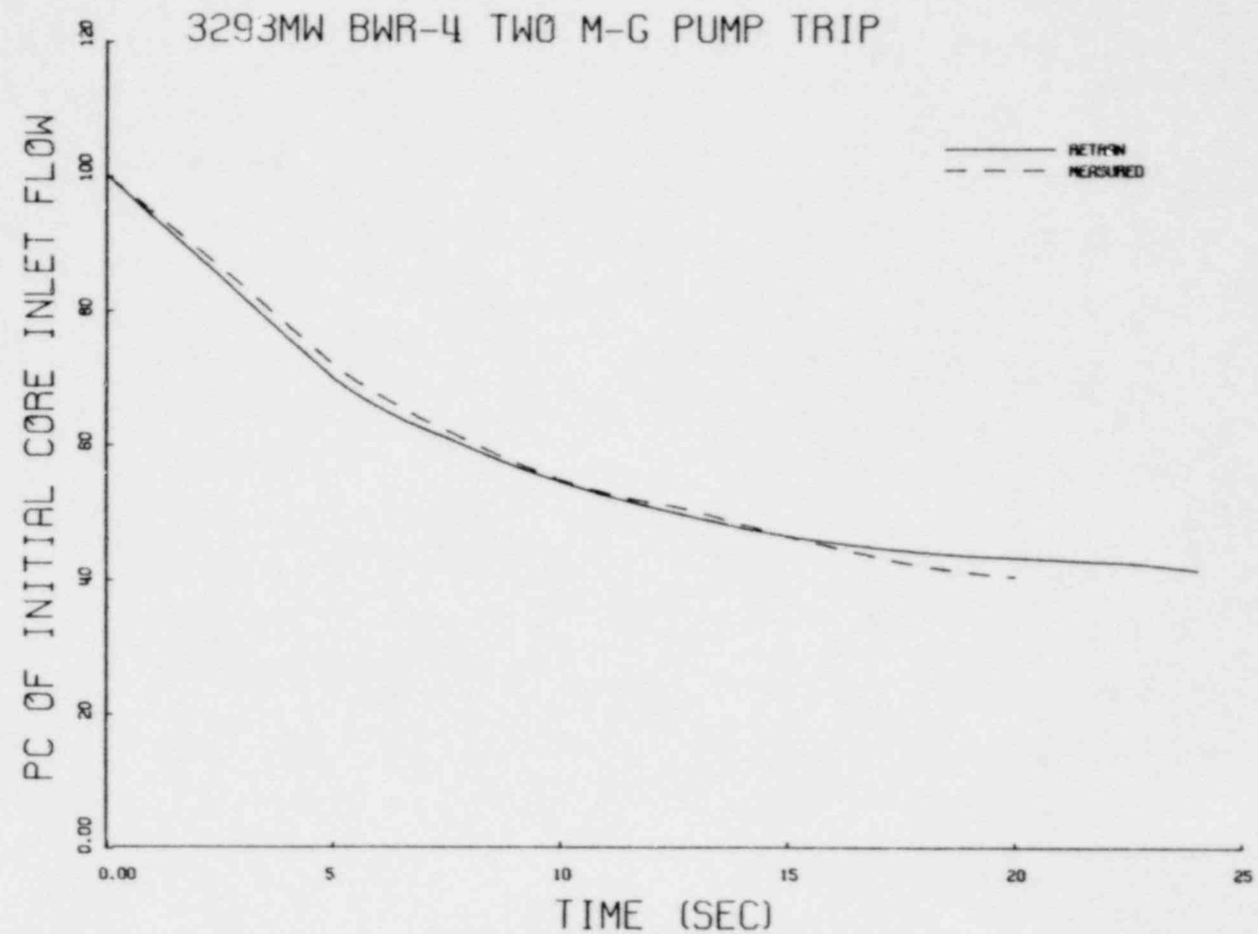

FIGURE V.4-3 Core Inlet Flow - Two Pump Trip

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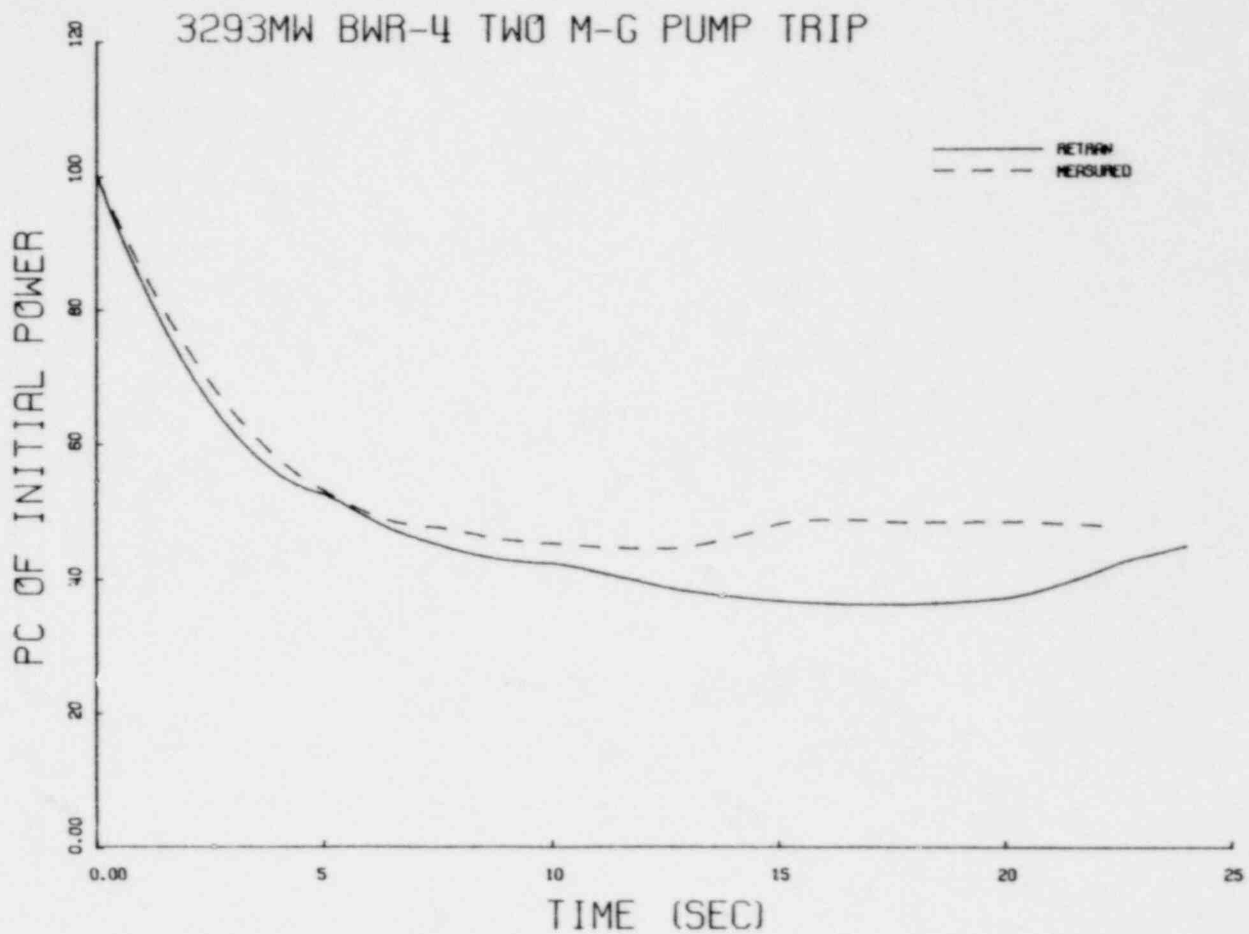

FIGURE V.4-4 Power Response - Two Pump Trip

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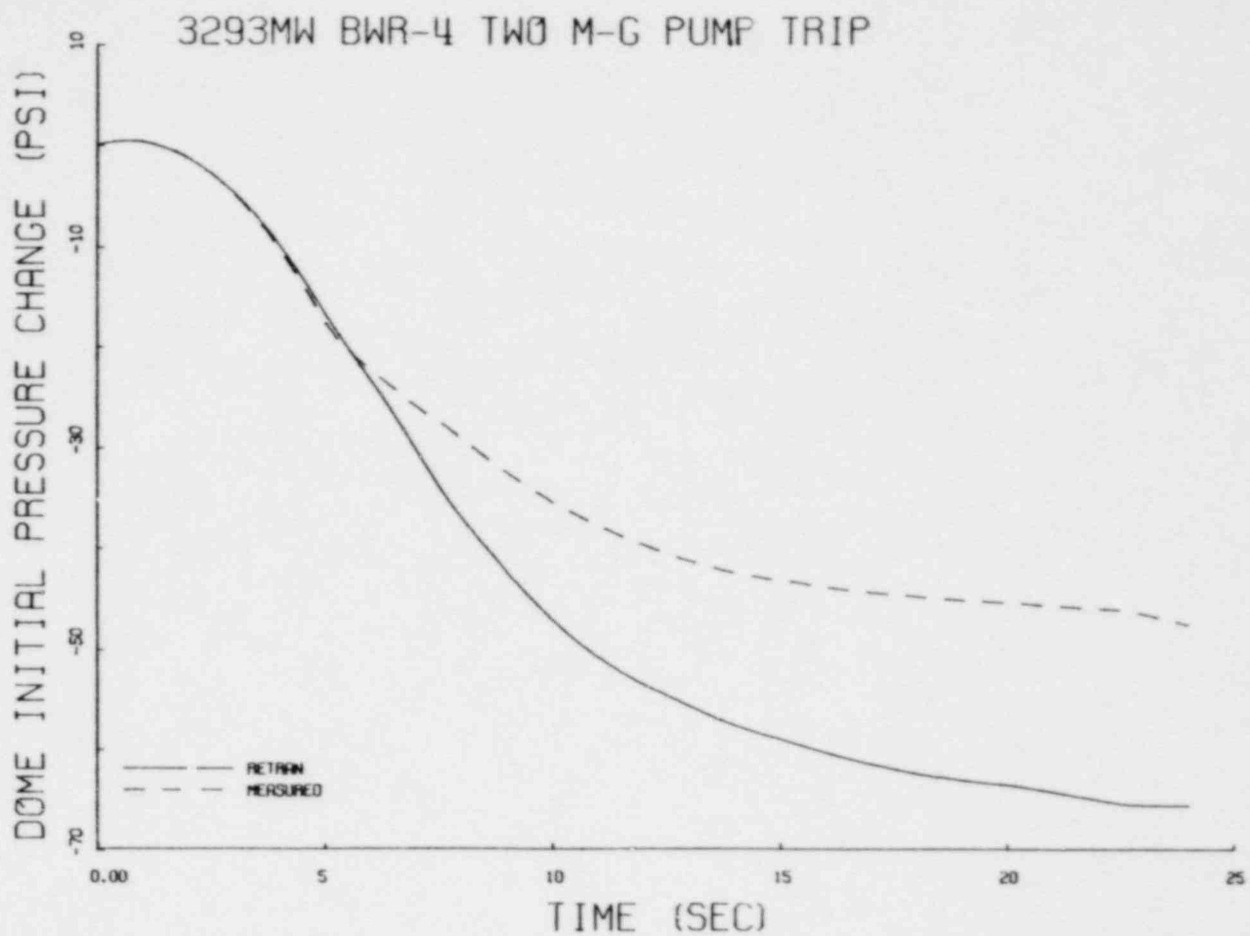

FIGURE V.4-5 Dome Pressure Change - Two Pump Trip

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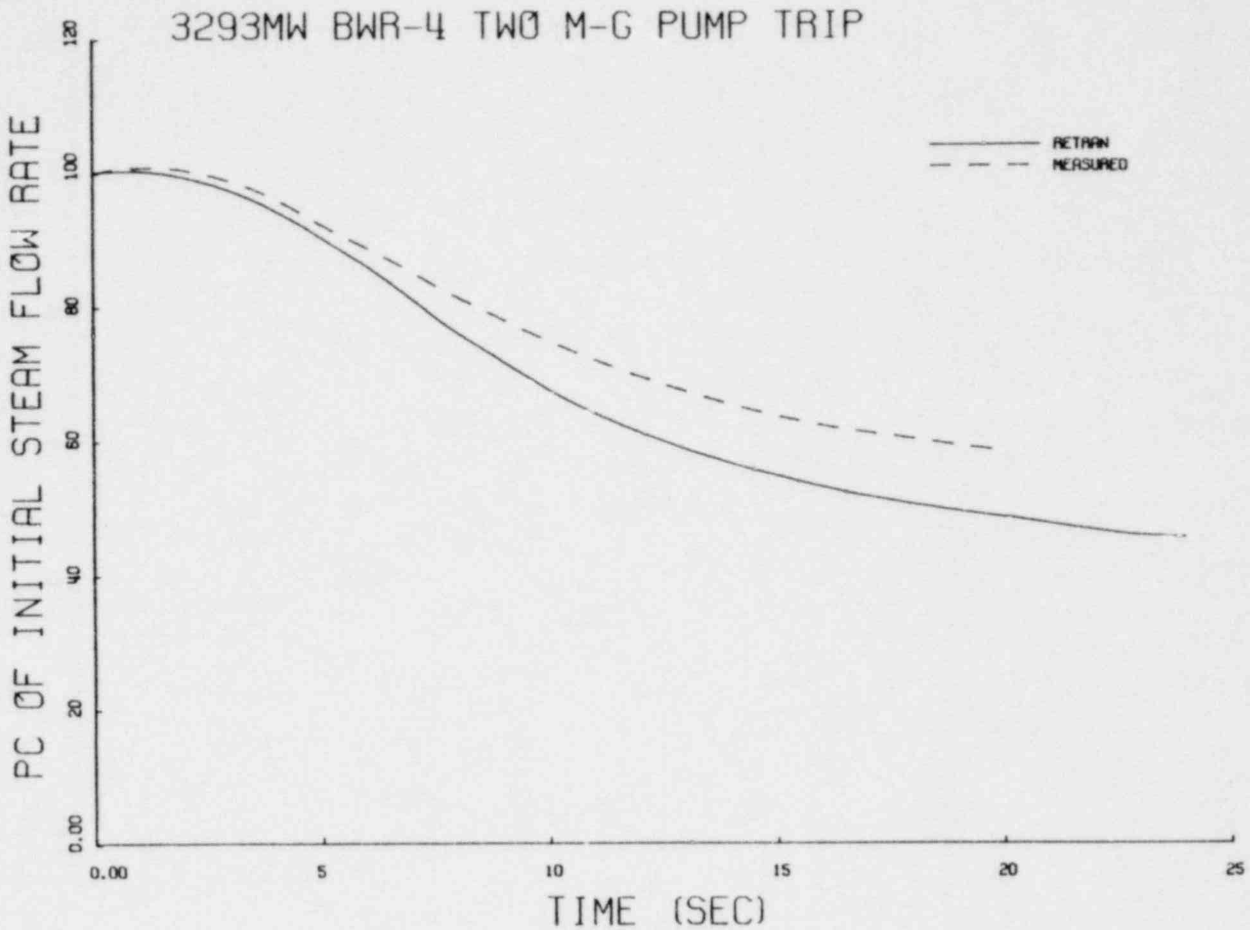

FIGURE V.4-6 Steam Flow Rate - Two Pump Trip

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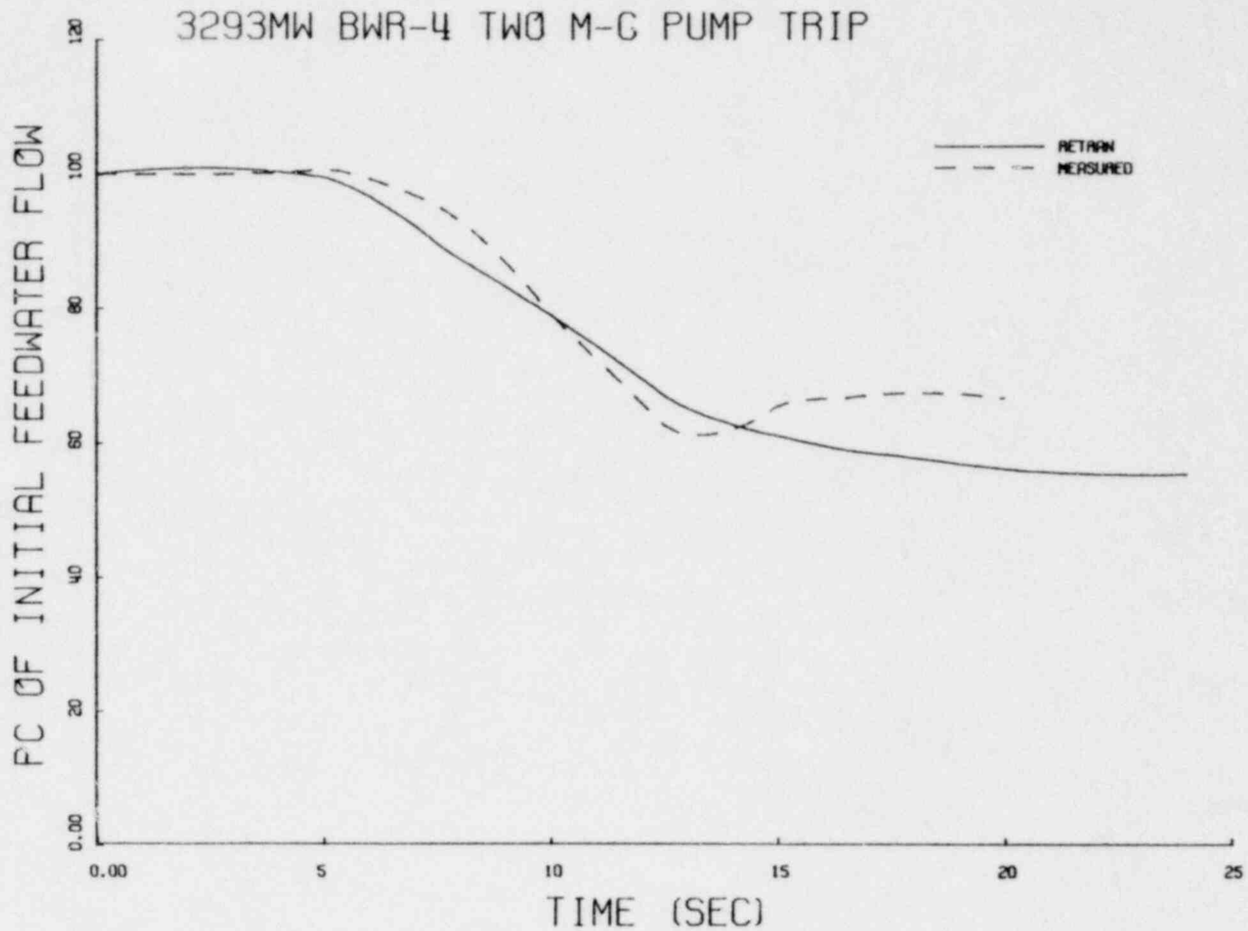

FIGURE V.4-7 Feedwater Flow - Two Pump Trip

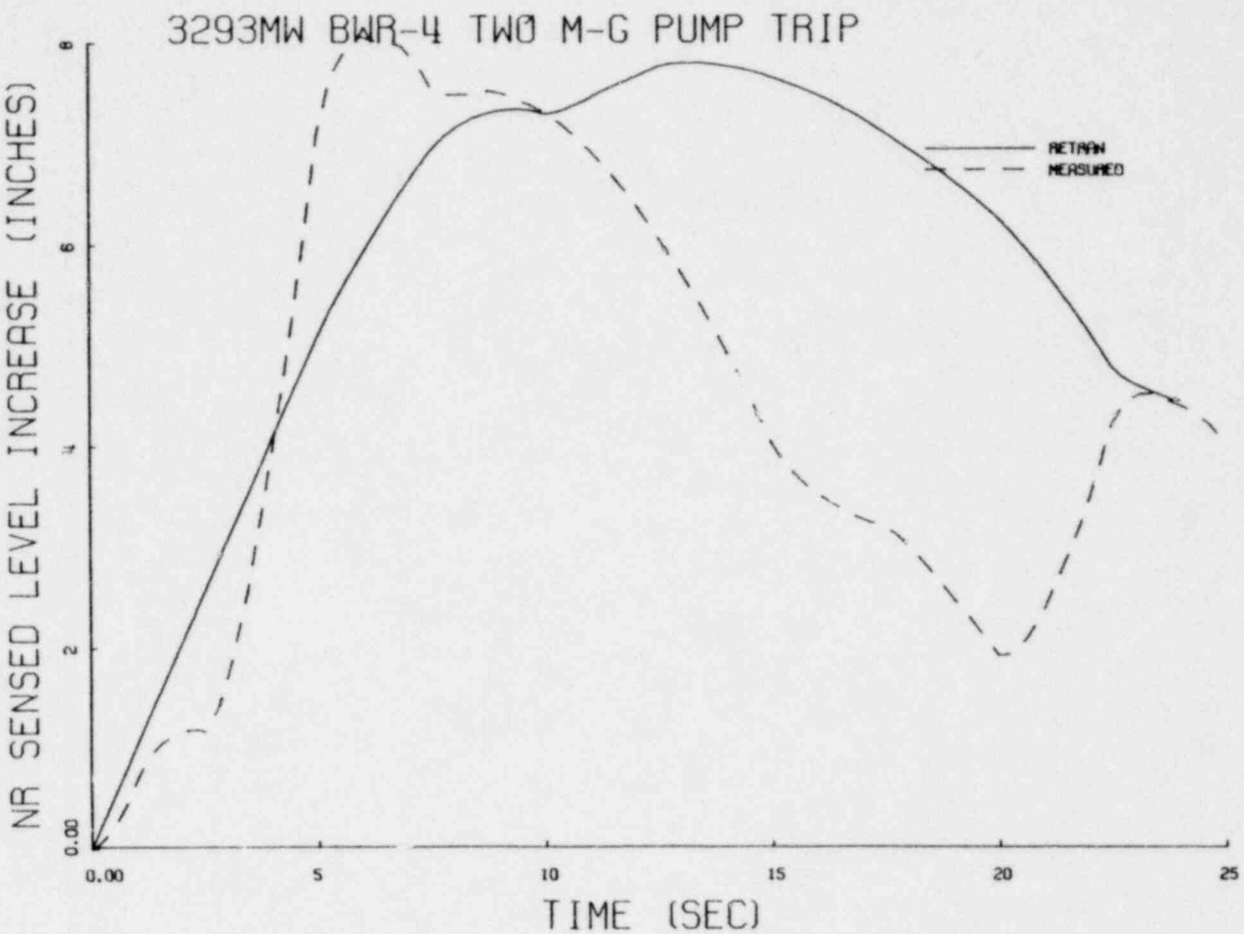

FIGURE V.4-8 Sensed Water Level Change - Two Pump Trip

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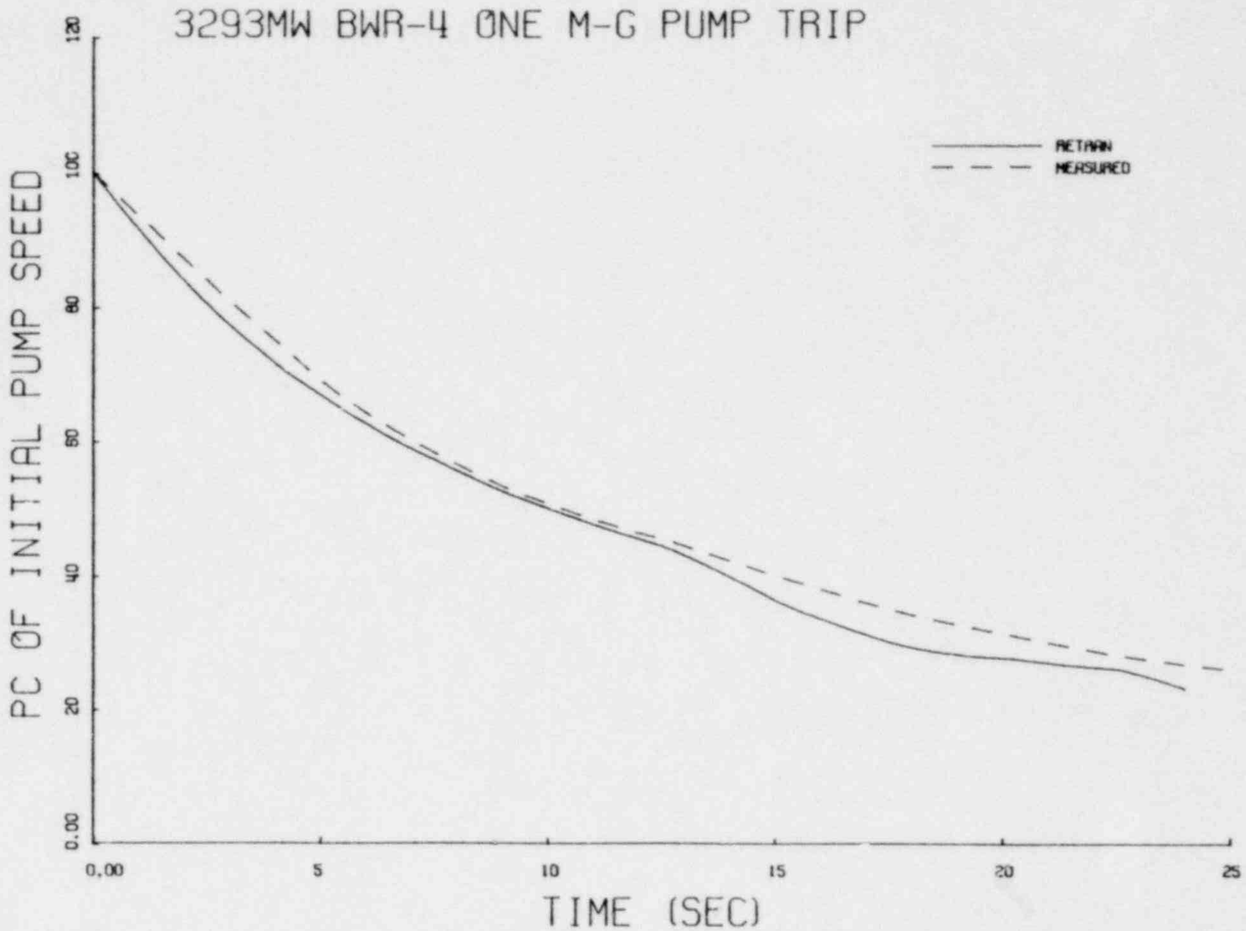

FIGURE V.4-9 Change in Speed of Tripped Pump

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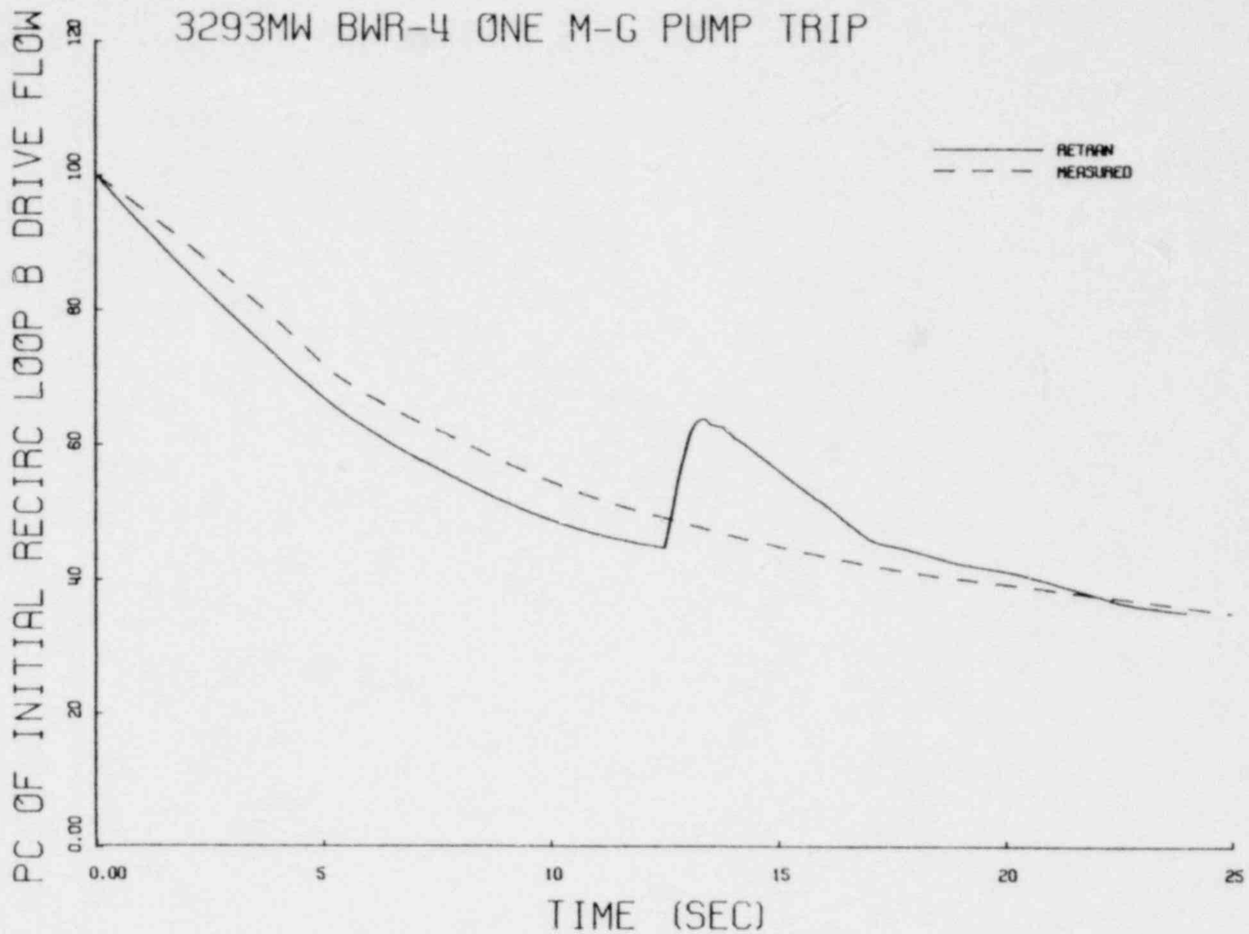

FIGURE V.4-10 Drive Flow in Recirculation Loop - One Pump Trip

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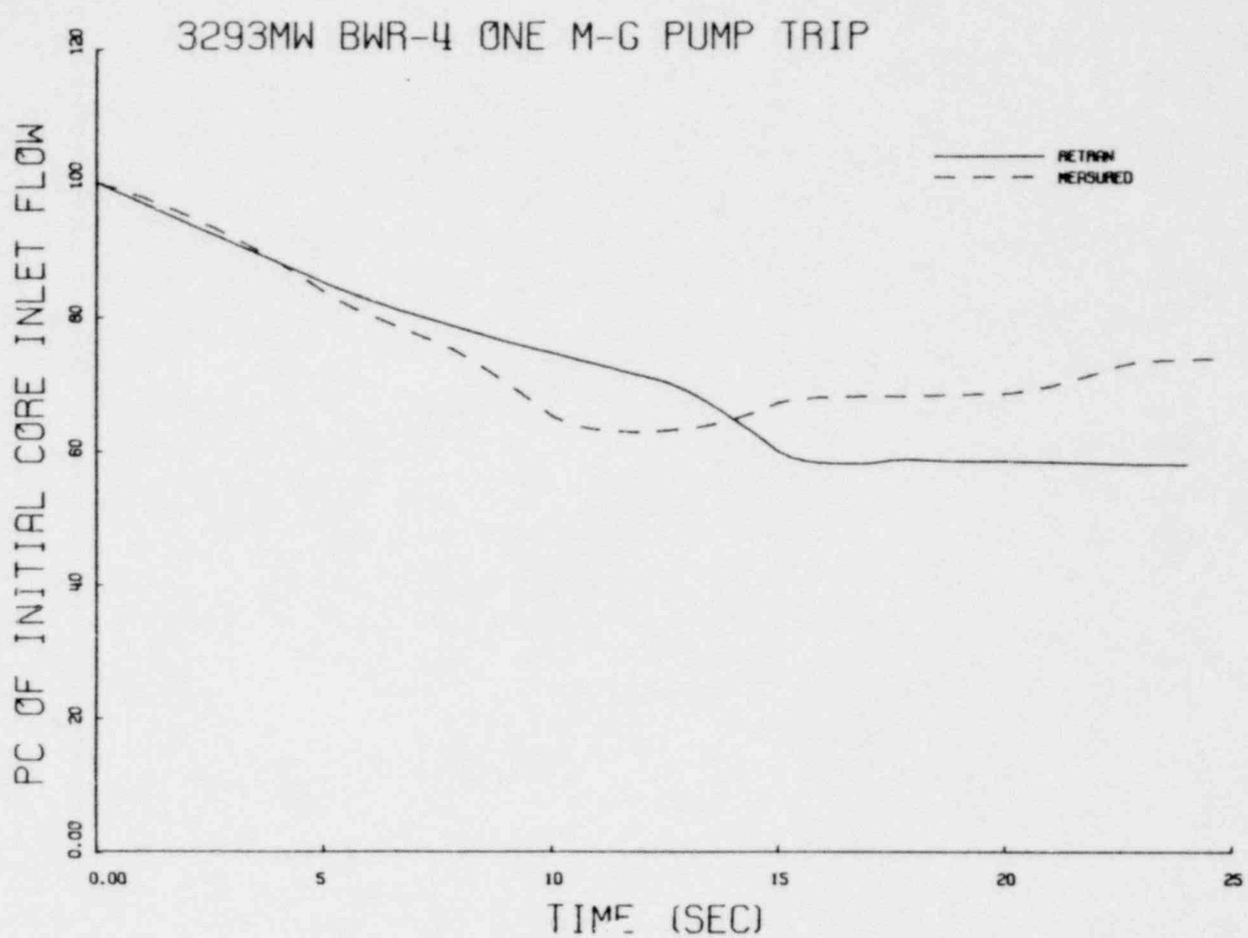

FIGURE V.4-11 Core Inlet Flow - One Pump Trip

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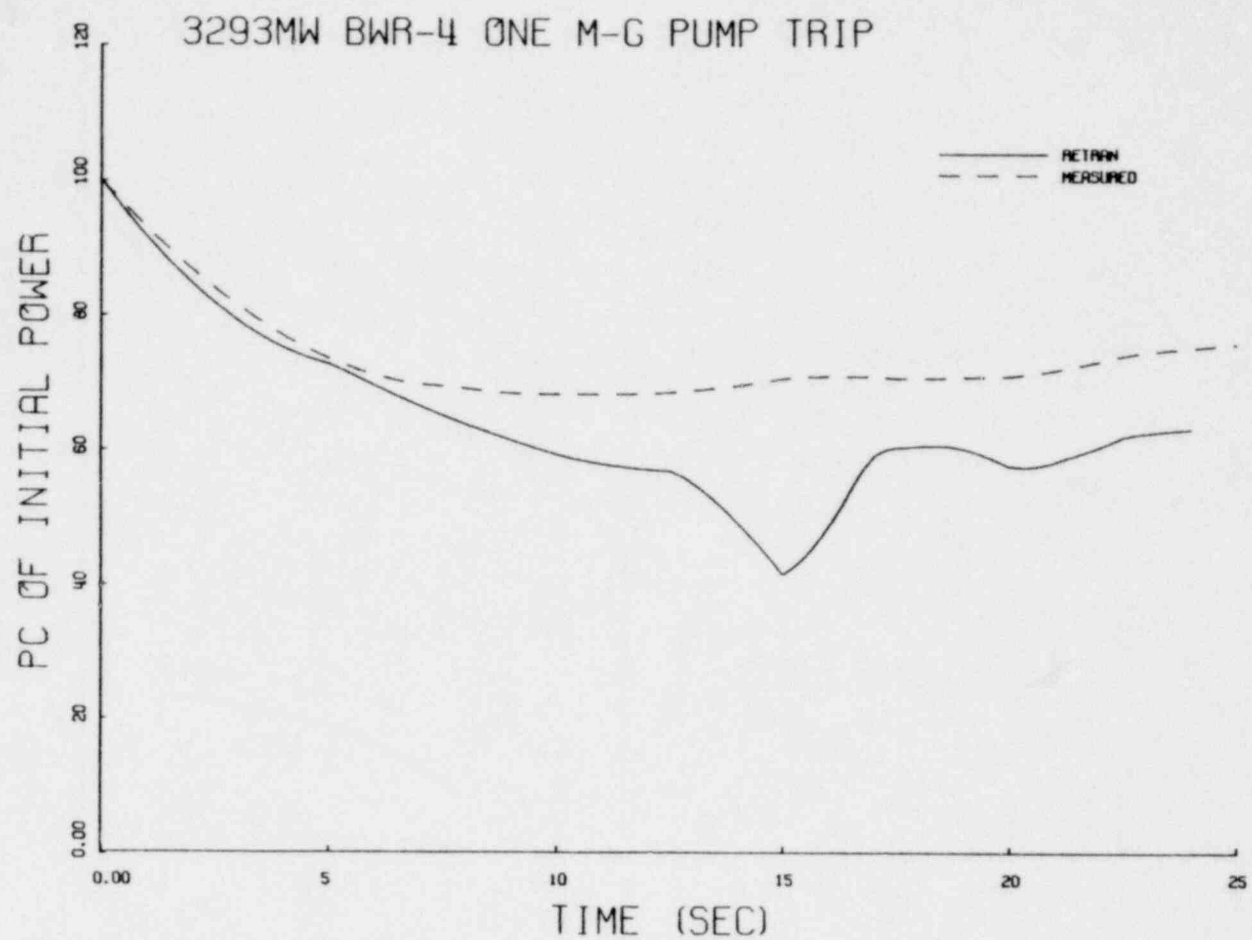

FIGURE V.4-12 Change in Core Power - One Pump Trip

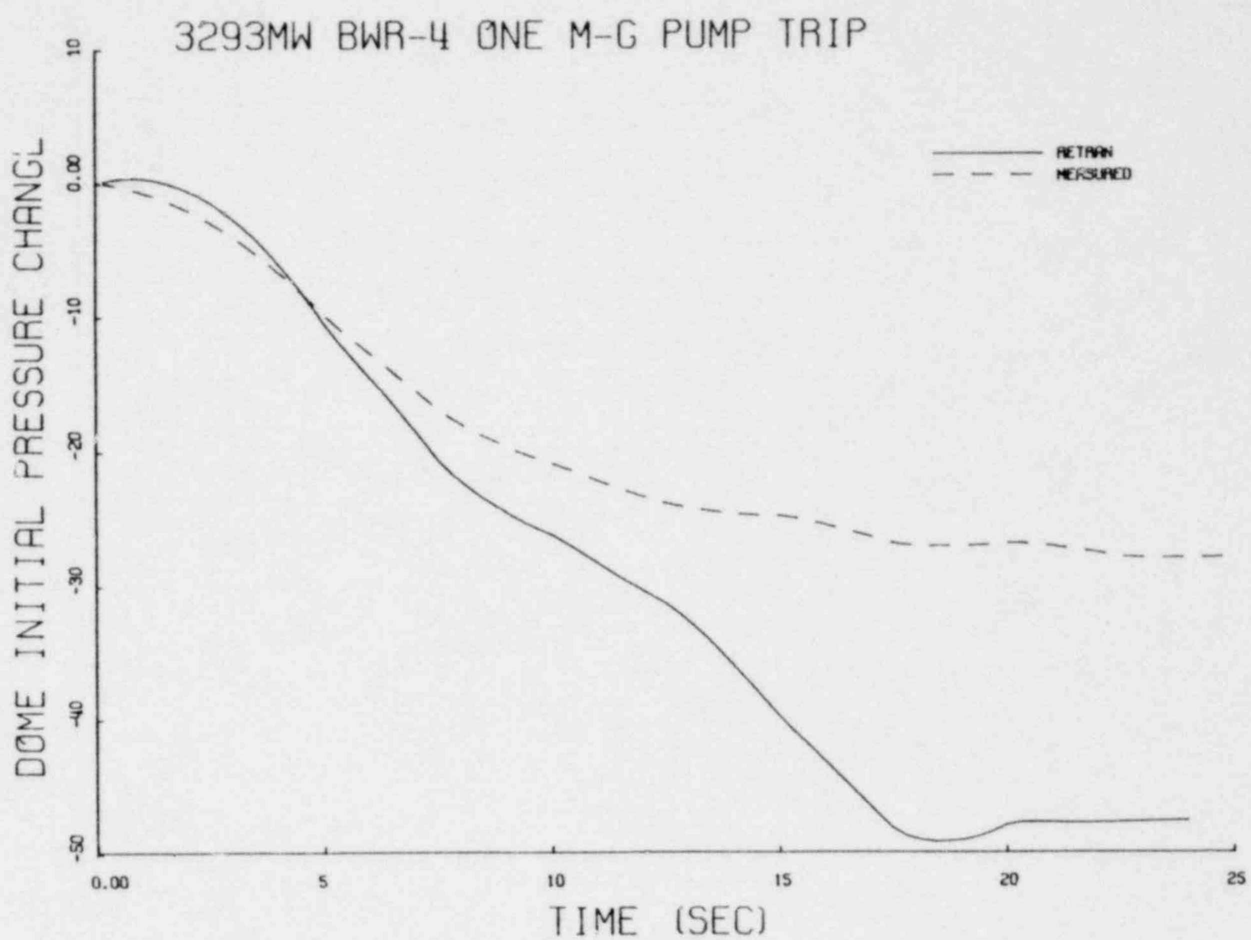

FIGURE V.4-13 Dome Pressure Change - One Pump Trip

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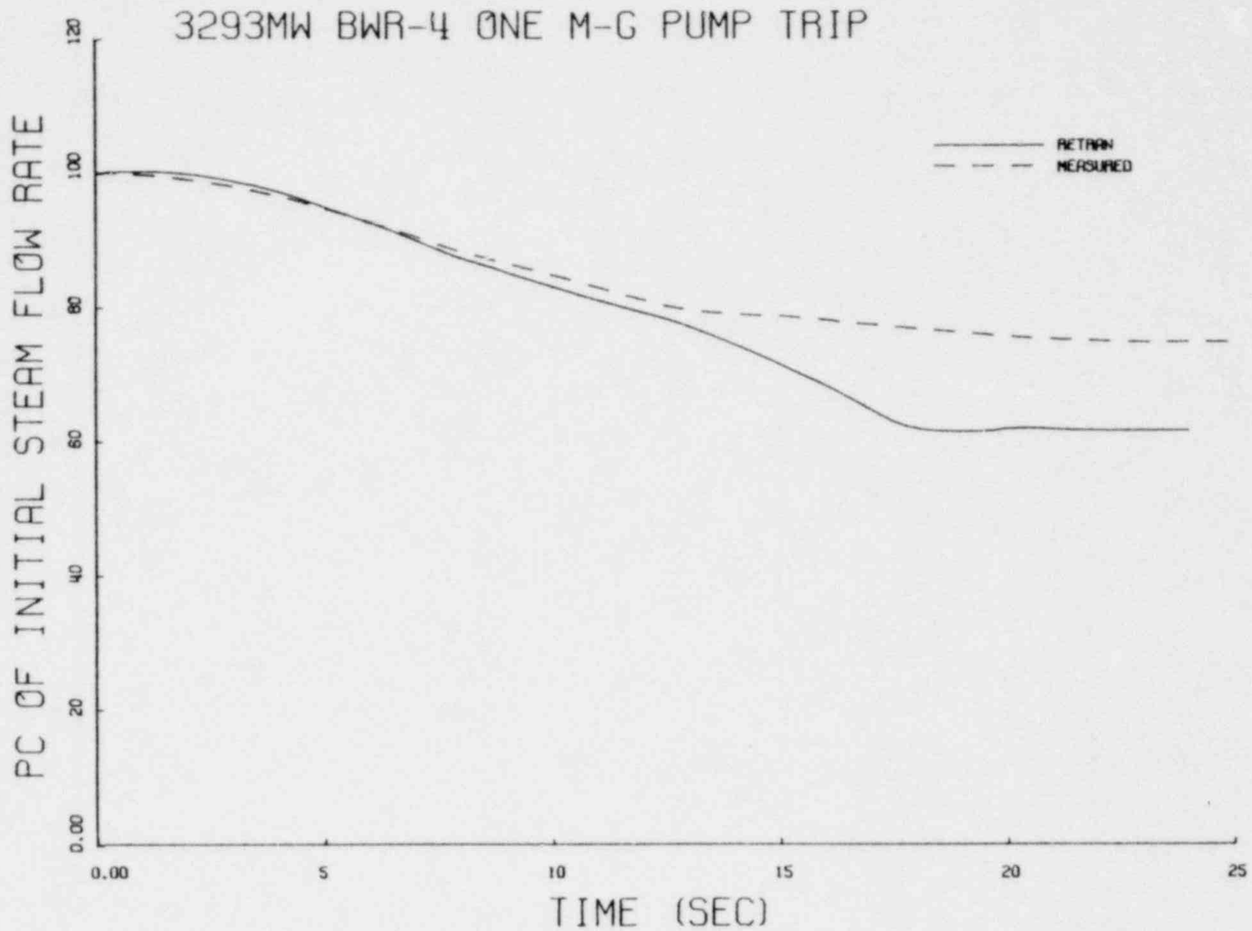

FIGURE V.4-14 Steam Flow Rate - One Pump Trip

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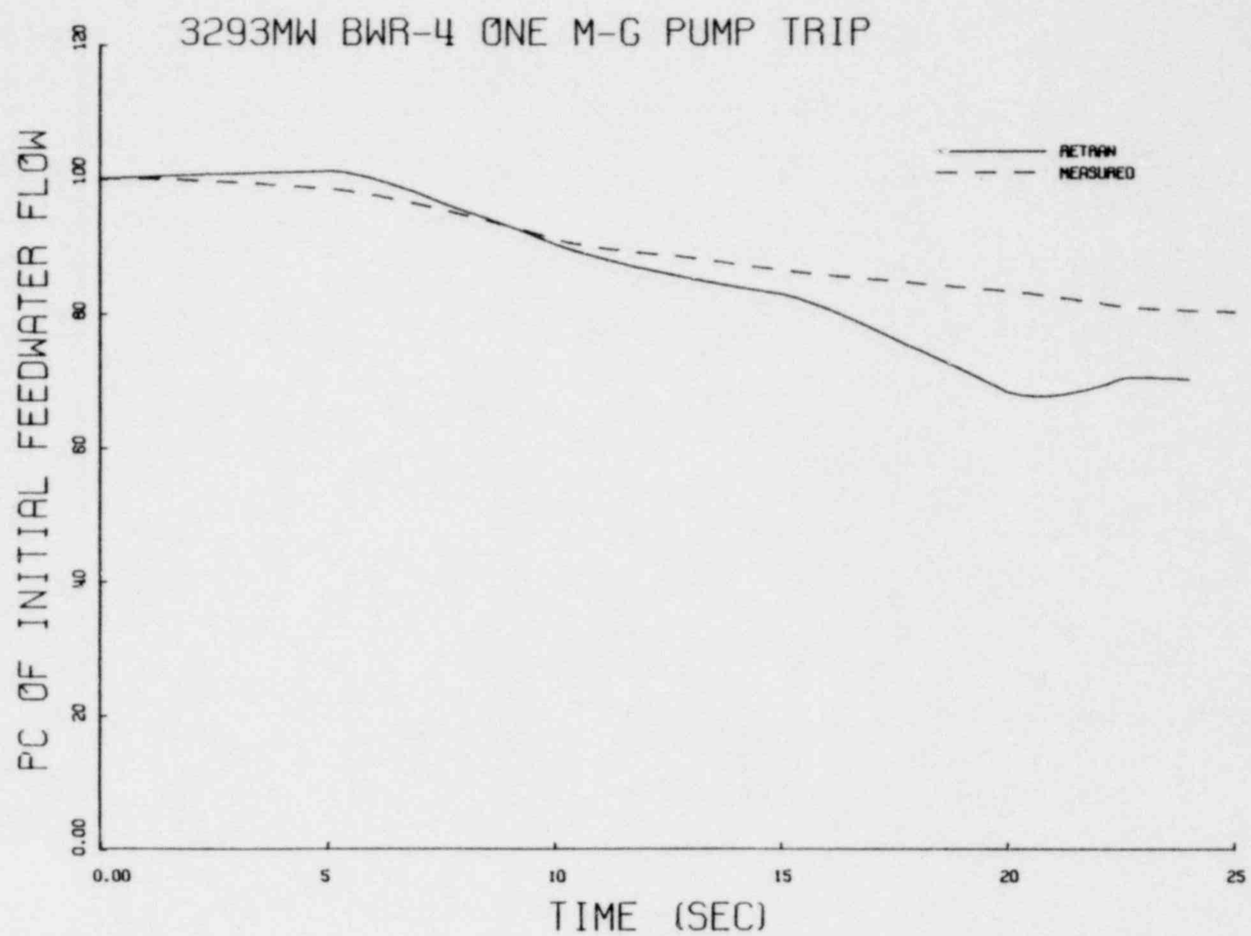

FIGURE V.4-15 Feedwater Flow - One Pump Trip

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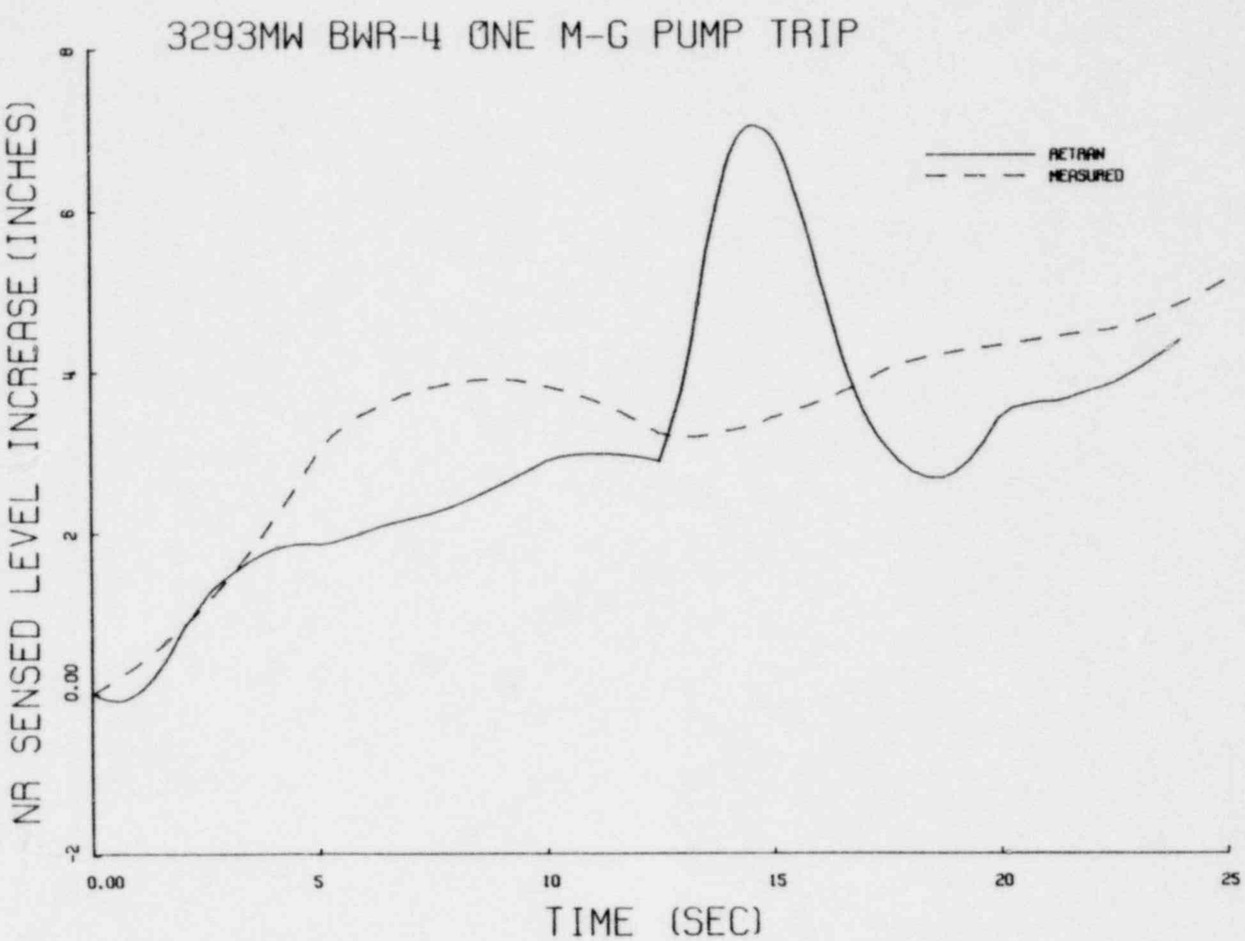

FIGURE V.4-16 Sensed Water Level - One Pump Trip

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VI. PRESSURIZED WATER REACTOR TRANSIENT ANALYSES

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VI. PRESSURIZED WATER REACTOR TRANSIENT ANALYSES

Transient analyses for pressurized water reactors (PWR) are classified according to Condition I, II, III or IV type of transients. Condition I events are those transients which are expected to occur during normal operation. Condition II transients are incidents of moderate frequency, with one incident expected per plant each year. Condition III events are less frequent, with one transient expected per plant lifetime. Incidents which are not expected to occur are classified as Condition IV incidents. A summary of the types of events resulting in Condition II, III and IV transients is summarized in Table VI-1.

The RETRAN analyses of those operational transients, as performed by members of the Utility Working Group, are summarized in this section. The studies performed by Burrell et al.[VI-1] with a preliminary version of RETRAN are not included in the following discussion.

1.0 UNCONTROLLED ROD WITHDRAWAL

The uncontrolled rod withdrawal is a reactivity insertion, Condition II transient. This transient is initiated by a reactivity insertion resulting from the removal of control rods. This reactivity insertion produces a power rise which is accompanied by an increase in the system pressure and temperature. The transient is terminated by a scram resulting from a high neutron flux or an overpower or overtemperature delta T trip.

Three analyses have been submitted for this transient, which is one of three transients designated for evaluation by utilities modeling a PWR. The Florida Power and Light analysis[VI.1-1] was performed for a low reactivity insertion rate. NUSCO analyzed four different insertion rates[VI.1-2] while the VEPCO analysis[VI.1-3] was for a slow reactivity insertion. The Florida Power and Light and VEPCO calculations were compared to vendor FSAR results.

1.1 Florida Power and Light Analysis

The analysis of the uncontrolled rod withdrawal (UCRW) performed by Florida Power and Light[VI.1-1] was compared to vendor calculations for a slow rate of reactivity insertion. This particular transient required the pressurizer pressure

TABLE VI-1

SOME PWR SAR-CHAPTER 15 TRANSIENTS

Transient	Condition	System Analysis Required
UNPLANNED DECREASE IN SECONDARY HEAT REMOVAL		
° Loss of external load	II	Yes
° Turbine trip	II	Yes
° Loss of condenser vacuum	II	Yes
° Steam pressure regulator failure	II	Yes
° Loss of normal feedwater flow	II	Yes
° Loss if A-C power to auxiliaries	II	Yes
UNPLANNED INCREASE IN SECONDARY HEAT REMOVAL		
° Excessive load increase	II	Yes
° Idle loop startup	II	Yes
° Decrease in feedwater temperature	II	Yes
° Increase in feedwater flow rate	II	Yes
° Increase in steam flow rate	II	Yes
° Inadvertent opening of steam generator relief or safety valve	II	Yes
CHANGES IN REACTOR COOLANT SYSTEM INVENTORY (PRIMARY SIDE INITIATED)		
° Inadvertent operation of ECCS	II	Yes
° Accidental depressurization	II	Yes
LOSS OF REACTOR COOLANT FLOW		
° Partial loss of flow	II	Yes
° Complete loss of flow	III	Yes
° Locked rotor	IV	Yes
REACTIVITY INSERTION (PRIMARY SIDE INITIATED)		
° Uncontrolled rod withdrawal		Yes
- from subcritical	II	Yes
- from power	II	Yes
° Control rod misoperation	II	Yes
° Chemical system malfunction	II	Yes
ANTICIPATED TRANSIENTS WITHOUT SCRAM		Yes
STEAM LINE BREAK	IV	Yes

TABLE VI-1 (cont'd)

SOME PWR SAR-CHAPTER 15 TRANSIENTS

Transient	Condition			System Analysis Required
RECIRCULATION LINE BREAK				
° Steam generator tube rupture	II	III		Yes
° Small break		III		Yes
° Loss of coolant accident			IV	Yes
FUEL ASSEMBLY INSERTION ERROR		III		No
CONTROL ROD EJECTION			IV	No
FUEL HANDLING ACCIDENT			IV	No
WASTE GAS DECAY TANK RUPTURE			IV	No

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control system and the overtemperature ΔT trip to be modeled. The model and analysis are discussed in more detail in Reference VI.1-1.

1.1.1 Description of Model

A two loop representation of the Turkey Point Units 3 and 4 was modeled (Figure VI.1-1). The model has 37 volumes and 46 junctions, with each loop composed of 12 volumes. There are 15 conductors to represent the core (3 conductors) and U-tubes (6 conductors) in each steam generator. It is noted that the plant model was developed to analyze a spectrum of transients, and thus has more detail than is required for the UCRW.

The pressure control system for the pressurizer is represented schematically in Figure VI.1-2. A proportional plus integral plus derivative controller is used to control the proportional and backup heater, the spray valves and one relief valve. Another relief valve and three safety valves are controlled by a direct pressure signal. The relief and safety valves were modeled as negative fill junctions based on the Moody critical flow model and the valve performance data.

Four trip signals required for an UCRW were included in the model. These are; (1) nuclear power, (2) high pressurizer pressure, (3) overtemperatu[illegible] and (4) overpower ΔT. The first two were modeled with the trips in RETRAN [illegible] the latter two were based on control system models.

The RETRAN steady-state initialization option was used for the reactor at hot, full power conditions. A null transient was run to verify steady conditions. Table VI.1-1 presents a summary of the input conditions used for the base case analysis. Sensitivity studies of vessel nodalization and power level were also performed.

1.1.2 Results of Analysis

Comparisons between the RETRAN results and FSAR values of pressurizer pressure and core power are given in Figure VI.1-3. It is noted that the initial power was 102 percent of nominal for this analysis. There is very good agreement between the two analyses until approximately 43 seconds. The RETRAN analysis resulted in an overtemperature ΔT trip approximately 8 seconds earlier than for

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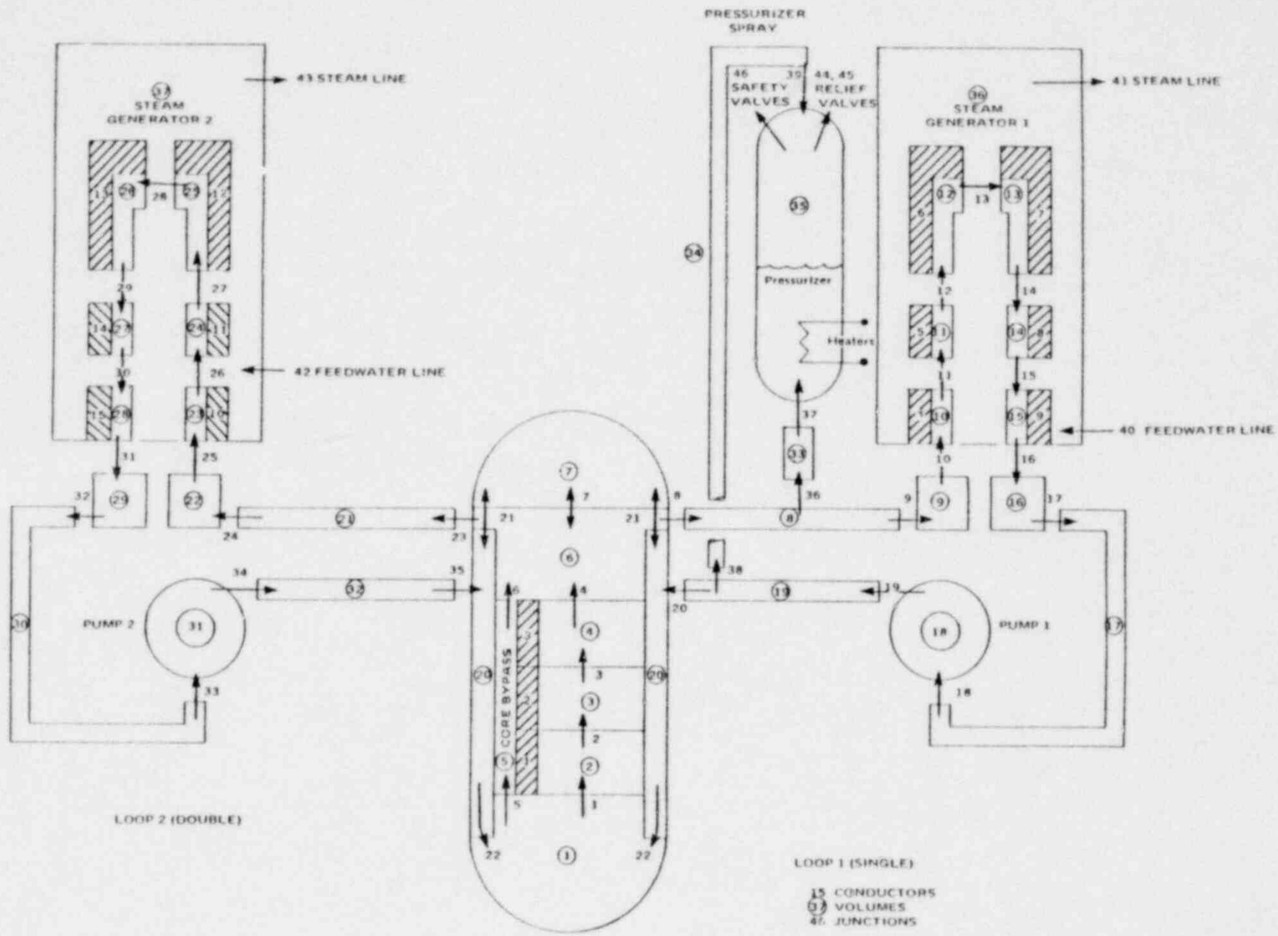

FIGURE VI.1-1 RETRAN Base Model for Turkey Point Units 3 and 4.

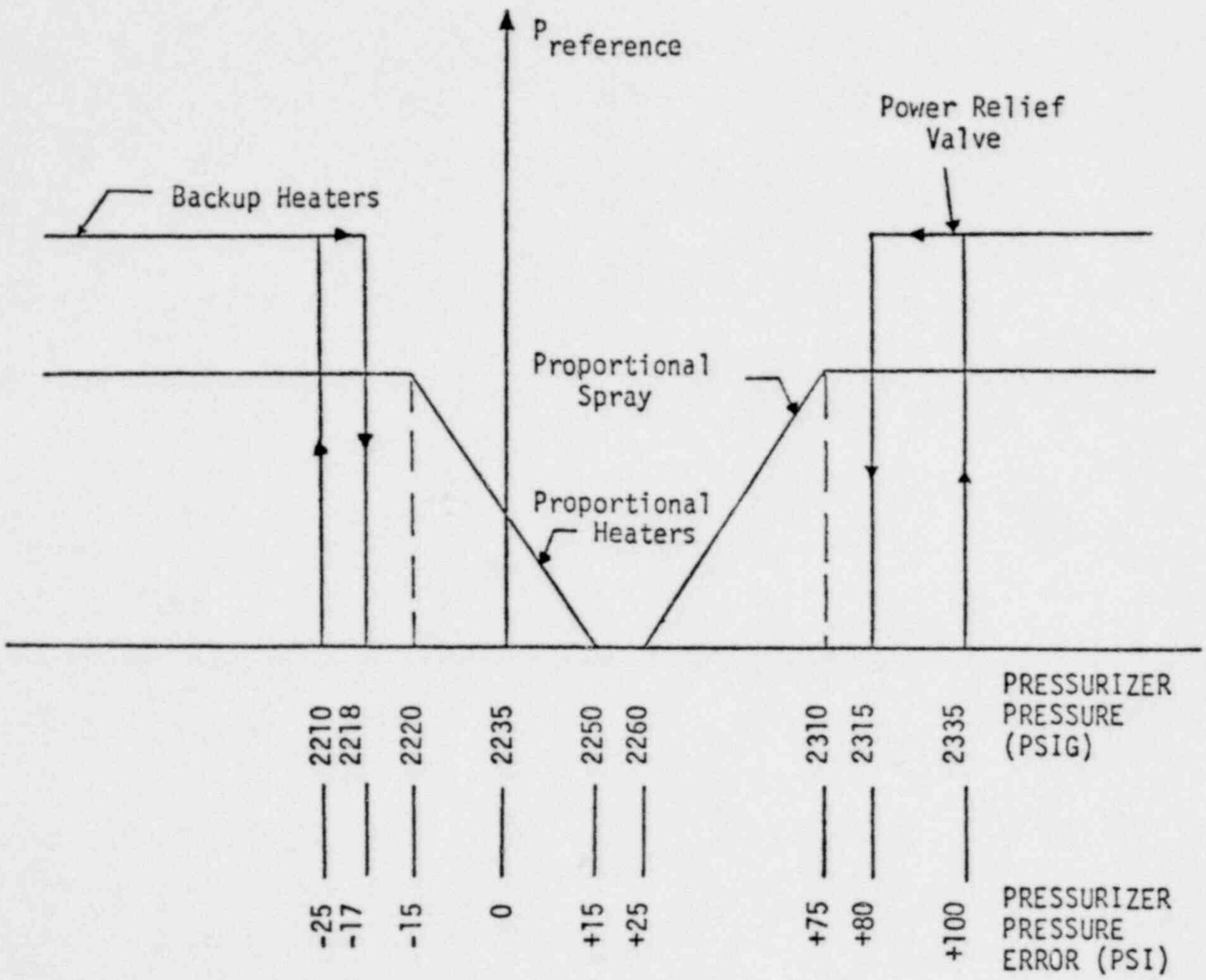

FIGURE VI.1-2 Pressurizer Pressure Control System.

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TABLE VI.1-1

SUMMARY OF INITIAL CONDITIONS AND REACTOR PARAMETERS
UNCONTROLLED RCCA WITHDRAWAL AT POWER

Parameter, Units	RETRAN Input
Initial Conditions	
Reactor Power, MW_t	2244
Pressurizer Pressure, psia	2220
Core Inlet Temperature, °F	550.2
Reactor Parameters	
Reactivity Insertion Rate, ΔK/sec	$2.5 * 10^{-5}$
Moderator Coefficient, ΔK/°F	$-0.40 * 10^{-4}$
Doppler Coefficient, ΔK/°F	$-1.00 * 10^{-5}$
Scram Worth, %ΔK	1.77
βeff	.00721
Generation Time, μsec	27.0
Power Distribution	BOL, HFP

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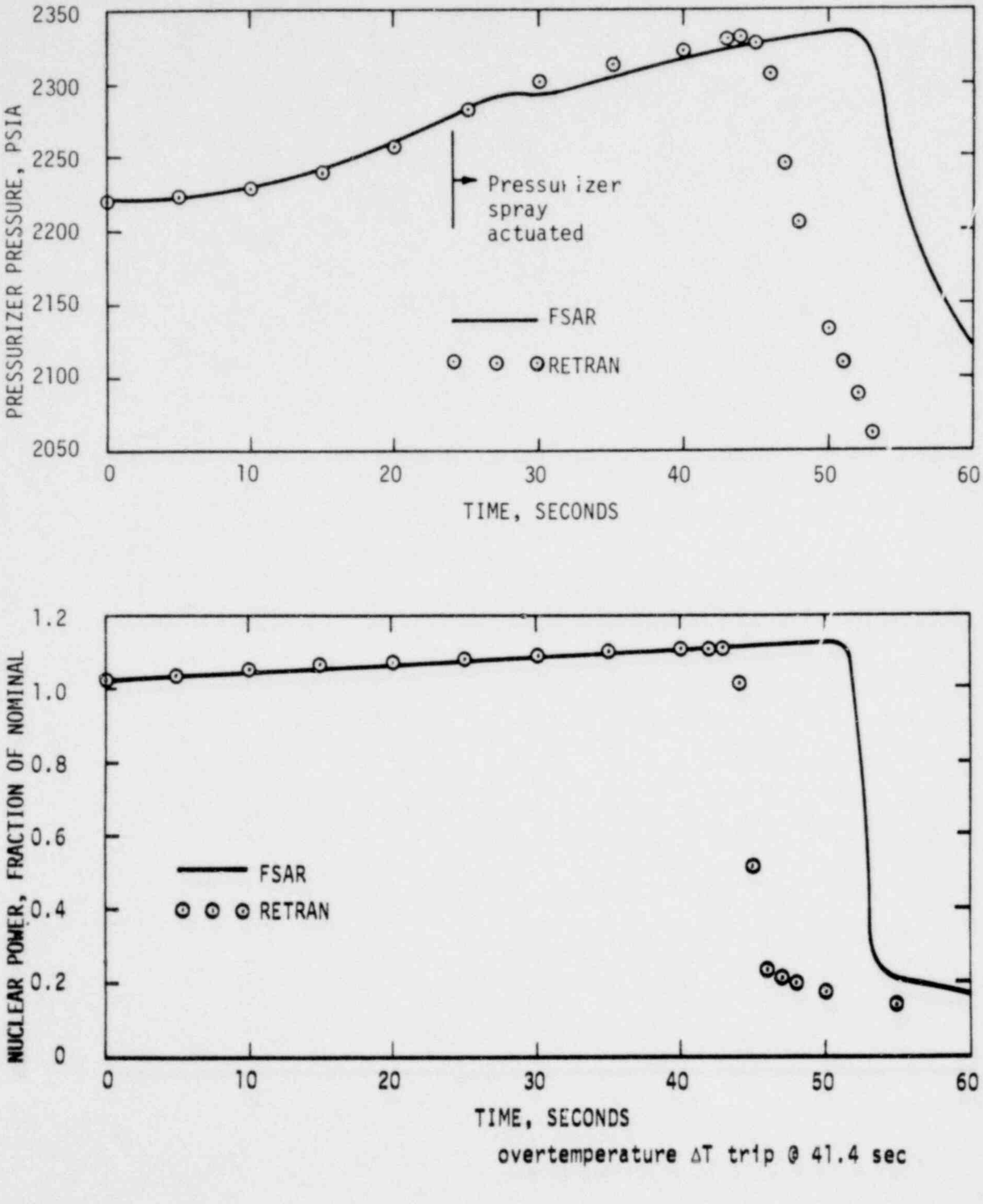

FIGURE VI.1-3 Pressurizer Pressure and Nuclear Power for UCRW Reference Calculation.

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the FSAR case. As seen in Figure VI.1-4, the average fluid temperature computed by RETRAN is increasing at a slightly faster rate than predicted in the FSAR, thus causing an earlier overtemperature ΔT trip.

In an attempt to approximate the vendor model as closely as possible, the upper plenum and the dead volume in the top of the vessel (Figure VI.1-1) were combined to form a single volume. The effect of this change is to extend the time of the overtemperature ΔT trip by 1.8 seconds. The results shown in Figure VI.1-5 indicate this modification produced a significant improvement for both the core power and average temperature values.

The final sensitivity study performed was designed to evaluate the effect of reducing the initial power. The total power input in the previous analyses included the pump heat (~ 10 MWt). Reducing the total power by this amount resulted in the values of core power and average temperature shown in Figure VI.1-6. The overtemperature ΔT trip for this analysis occurred 3.7 seconds later than in the base case run.

1.2 NUSCO Analyses

Like many utilities in the Working Group, NUSCO developed a RETRAN plant model which could be used for a variety of plant analyses. The details of this model are given in Reference VI.1-2. Carlson performed UCRW analyses for 4 values of reactivity insertion rates. The model and results of the analyses are summarized in the following section.

1.2.1 Description of Model

NUSCO developed a two loop representation of Connecticut Yankee (Figure VI.1-7) for operational transient analyses. The model has 25 volumes, 40 junctions and 7 heat conductors. Each steam-generator is modeled with 2 primary side volumes, 1 secondary side volume and 2 conductors. Flow into and out of the secondary side is controlled by fills, and there are 4 lumped representations of relief valves on the secondary.

The pressurizer model includes 4 heaters with a pressure signal dependence which is variable, and 4 heaters which are dependent only on the value of the pressure

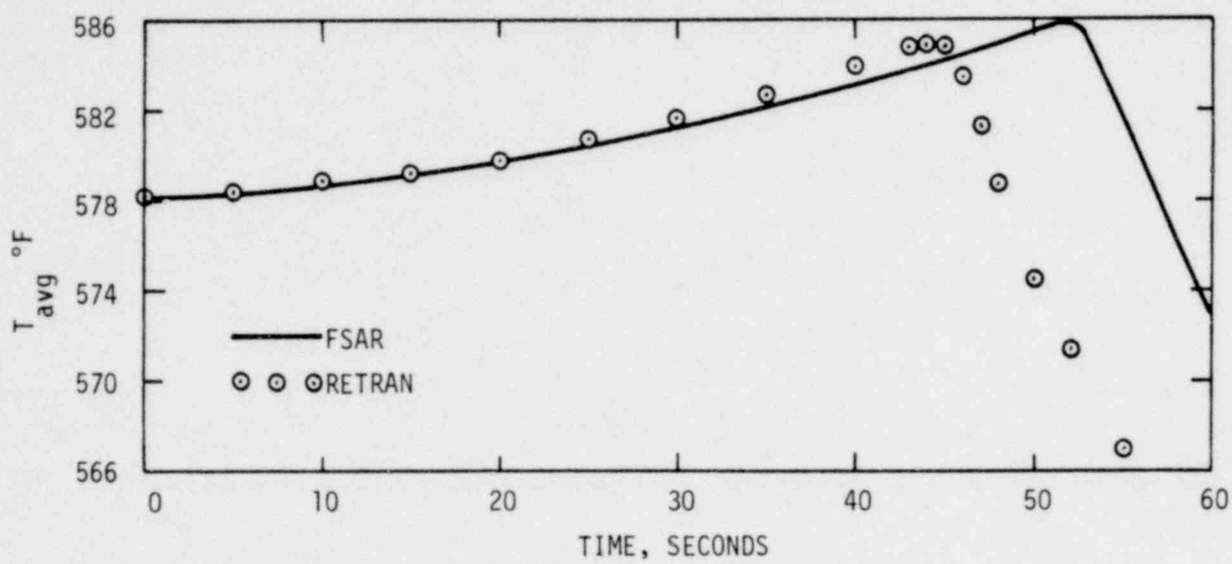

FIGURE VI.1-4 Average Coolant Temperature for UCRW Reference Calculation.

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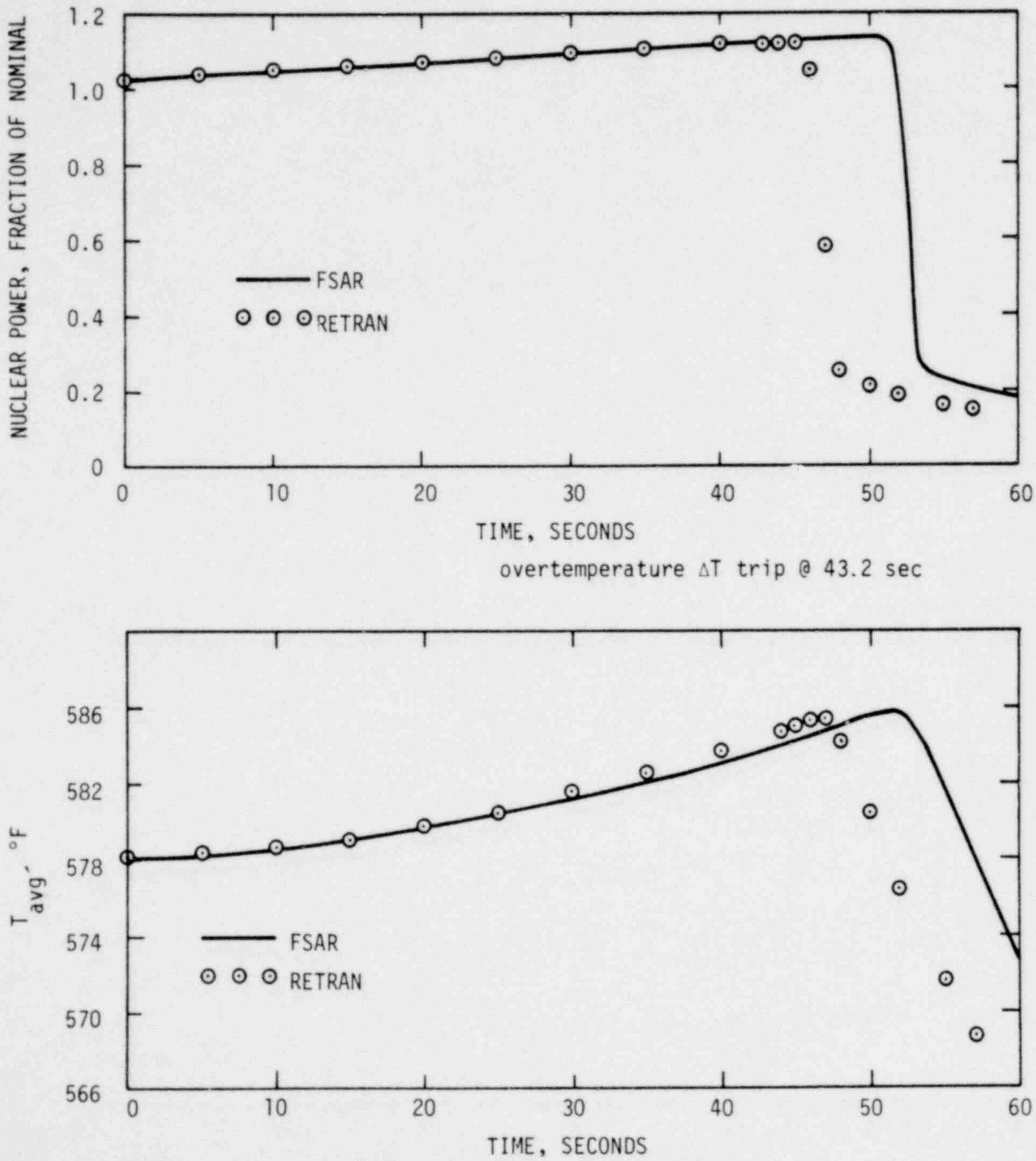

FIGURE VI.1-5 UCRW Reactor Vessel Nodalization Sensitivity Pesults.

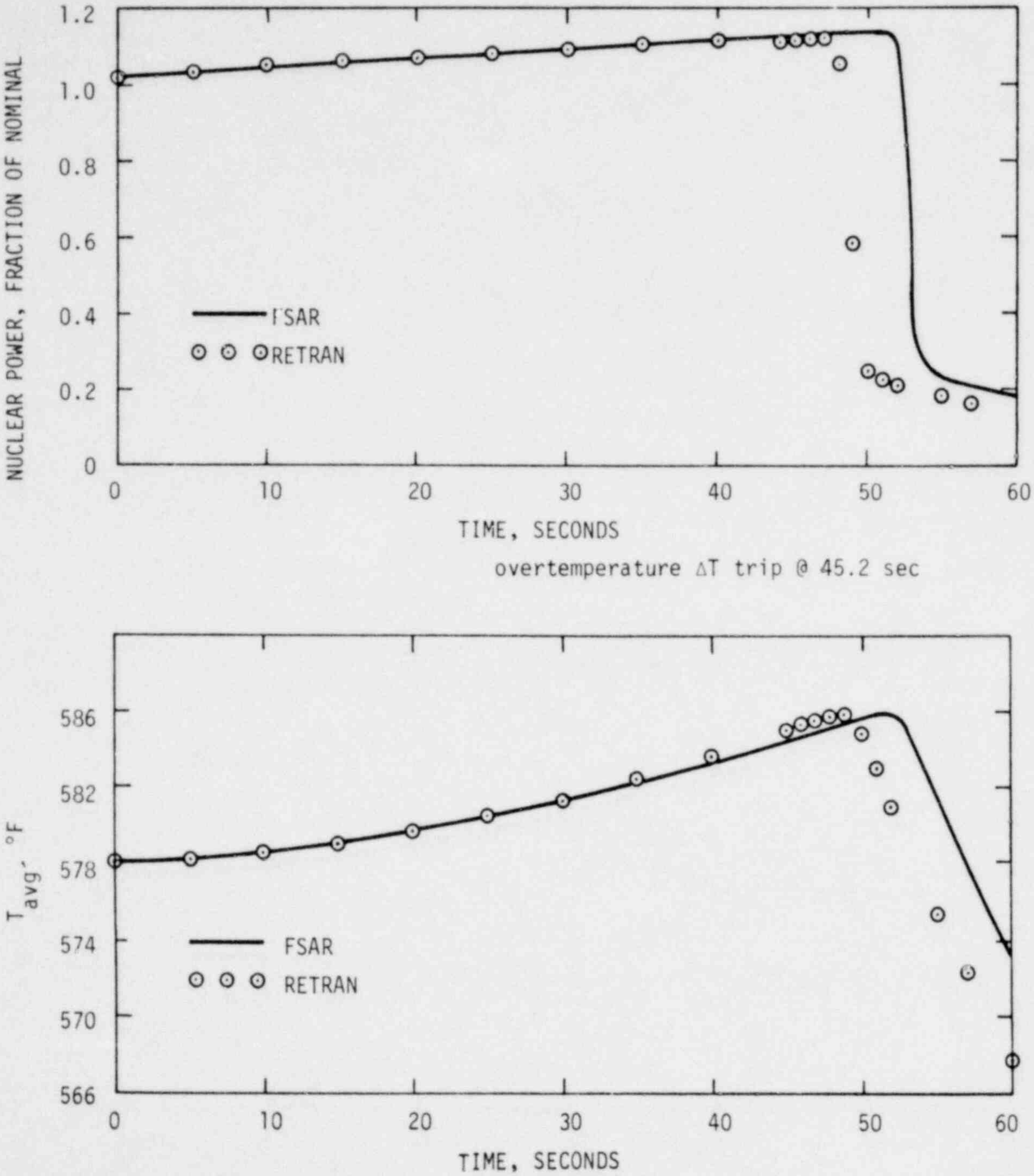

FIGURE VI.1-6 UCRW Reduced Power Sensitivity Results.

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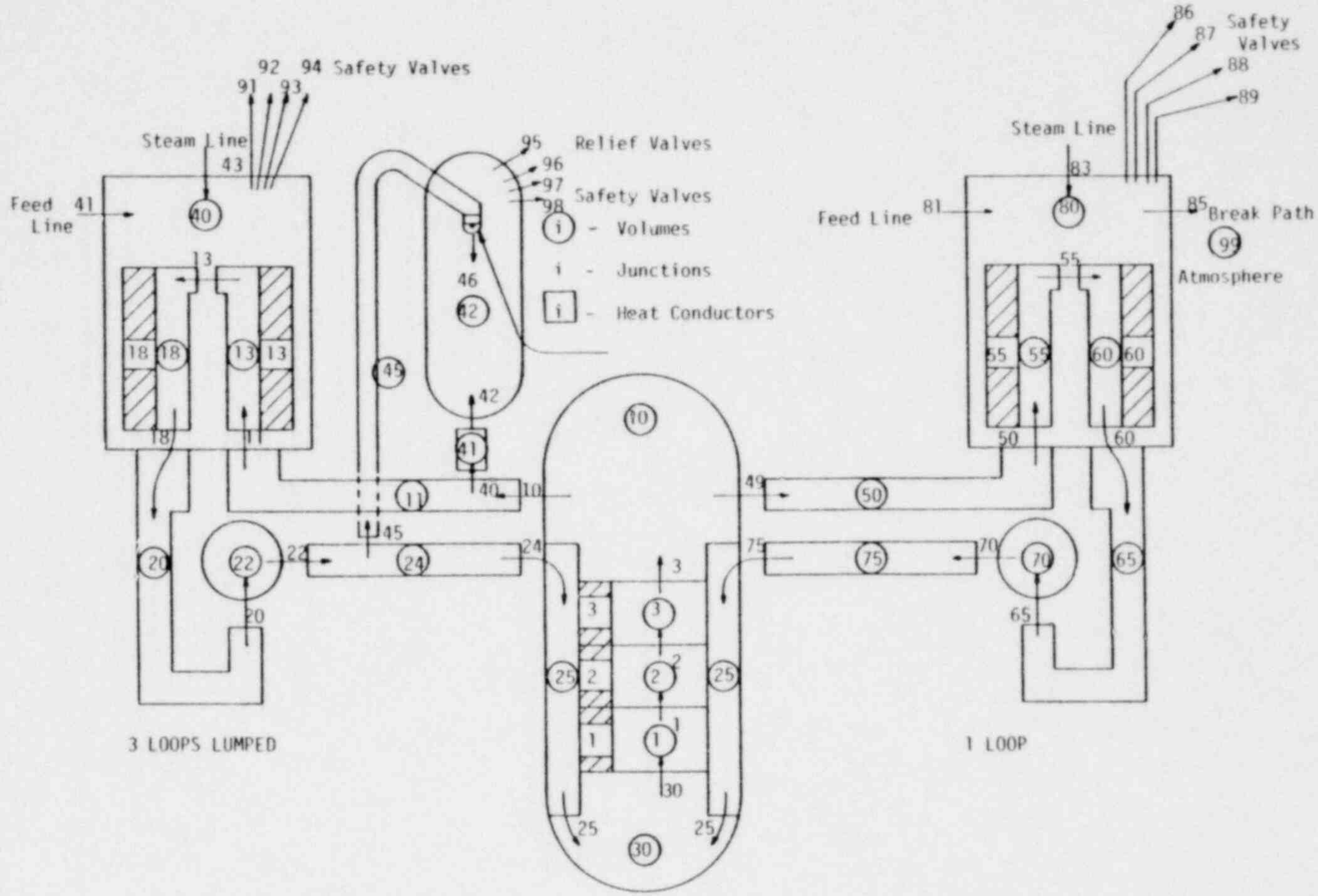

FIGURE VI.1-7 RETRAN Model for Connecticutt Yankee Reactor.

signal to control their operation. The pressurizer spray has a linear control which regulates the flow rate between maximum and minimum values. The flow is regulated by varying the valve flow area. The spray control system is shown in Figure VI.1-8.

1.2.2 Results of Analyses

The self-initialization option was used to achieve a steady-state condition, and a null transient was run to verify the solution. The UCRW transient was run with four different reactivity insertion rates. The core power and average coolant temperature values for each of these cases are shown in Figures VI.1-9 and VI.1-10. It is noted that the analysis for the lowest rod withdrawal rate (0.001 $/sec insertion) did not result in a scram. The increase in reactivity from the rod withdrawal was countered by a decrease in reactivity due to the negative moderator temperature coefficient.

1.3 VEPCO Analysis

Whereas the previous two analyses were performed with two loop representations of the reactor system, the VEPCO analysis[VI.1-3] of the UCRW was made for a single loop model. The results of the analysis and a comparison of the results with vendor values are summarized in the following sections. A detailed description of the model and the analysis is given by Smith[VI.1-3].

1.3.1 Description of Model

The single loop model of the Surry reactors is shown in Figure VI.1-11. The model has 19 volumes and 26 junctions. There are 4 volumes in the primary side of the steam generator, one secondary side volume and 4 heat conductors. Each of the major vessel regions (e.g., downcomer, core bypass and lower plenum) are modeled as single volumes with the exception of the core, which has 3 volumes and 3 conductors. Fill junctions are used to represent boundary conditions for the secondary side of the steam generator. Fill junctions are also used for boundary conditions on the primary side, and represent the pressurizer spray flow, the pressurizer cold leg intake and the pressurizer relief valves.

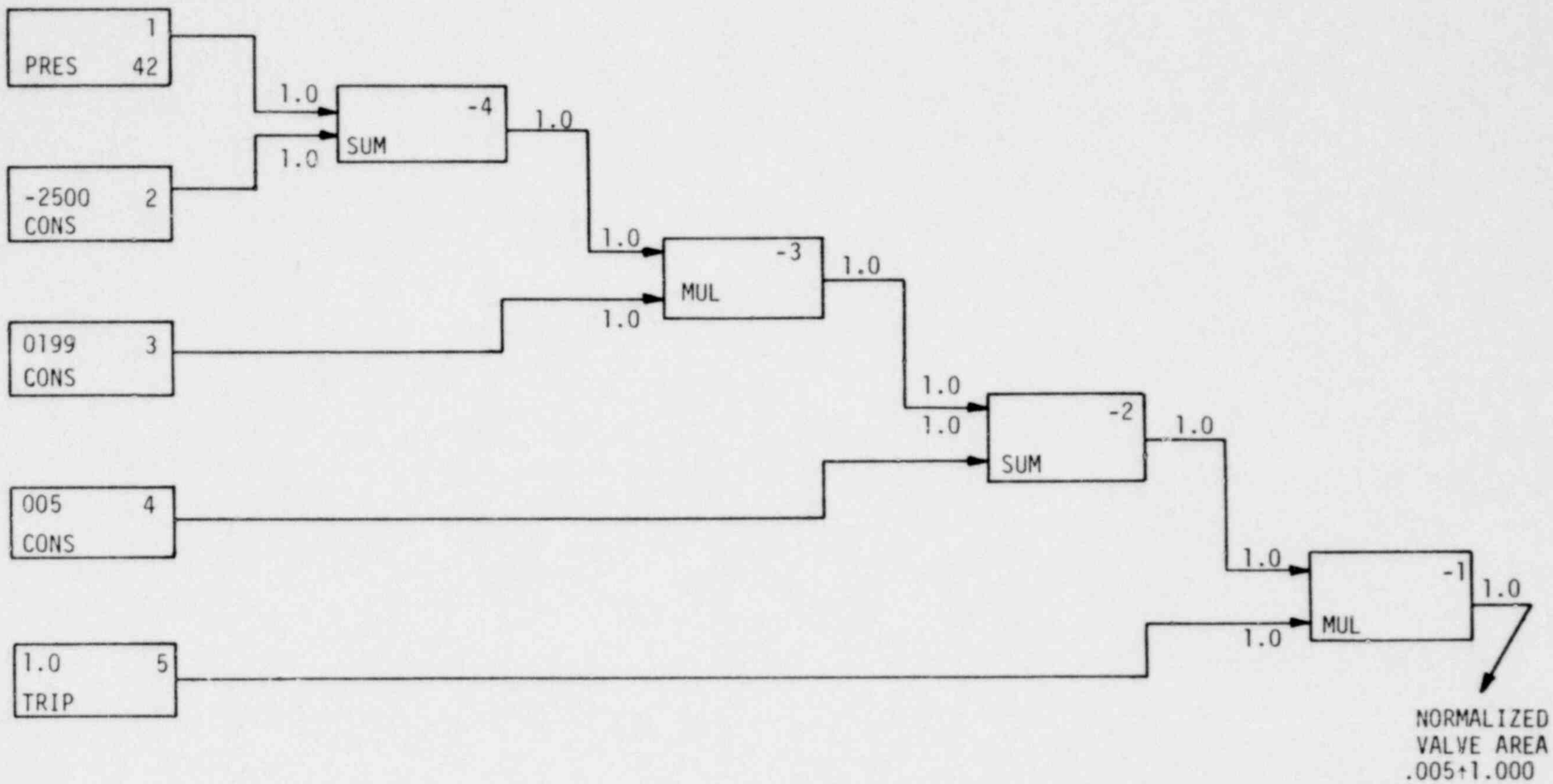

FIGURE VI.1-8 Pressurizer Spray Control System.

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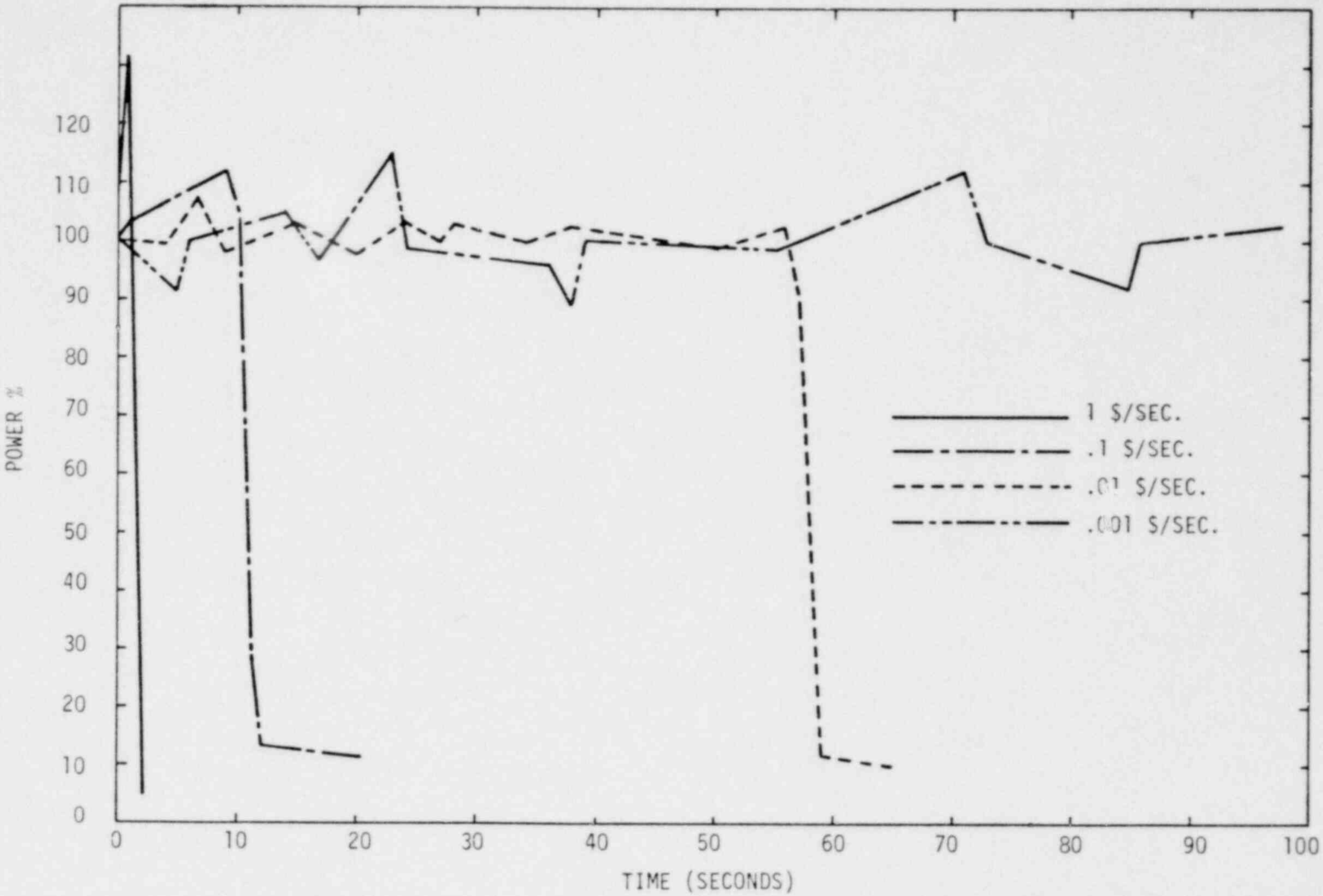

FIGURE VI.1-9 Power Response for Various Negative Reactivity Insertion Rates.

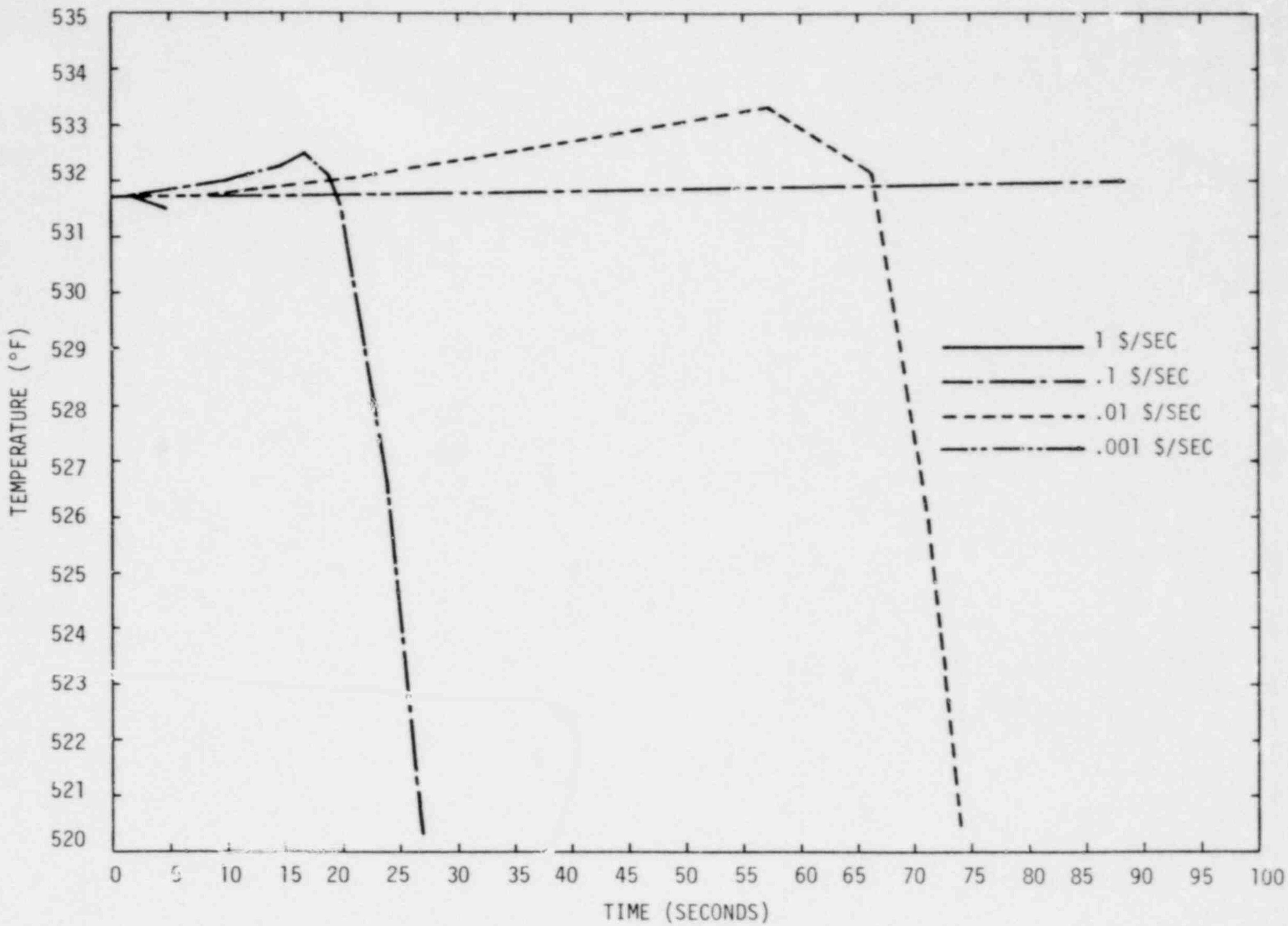

FIGURE VI.1-10 Average Coolant Temperature for Various Negative Reactivity Insertion Rates.

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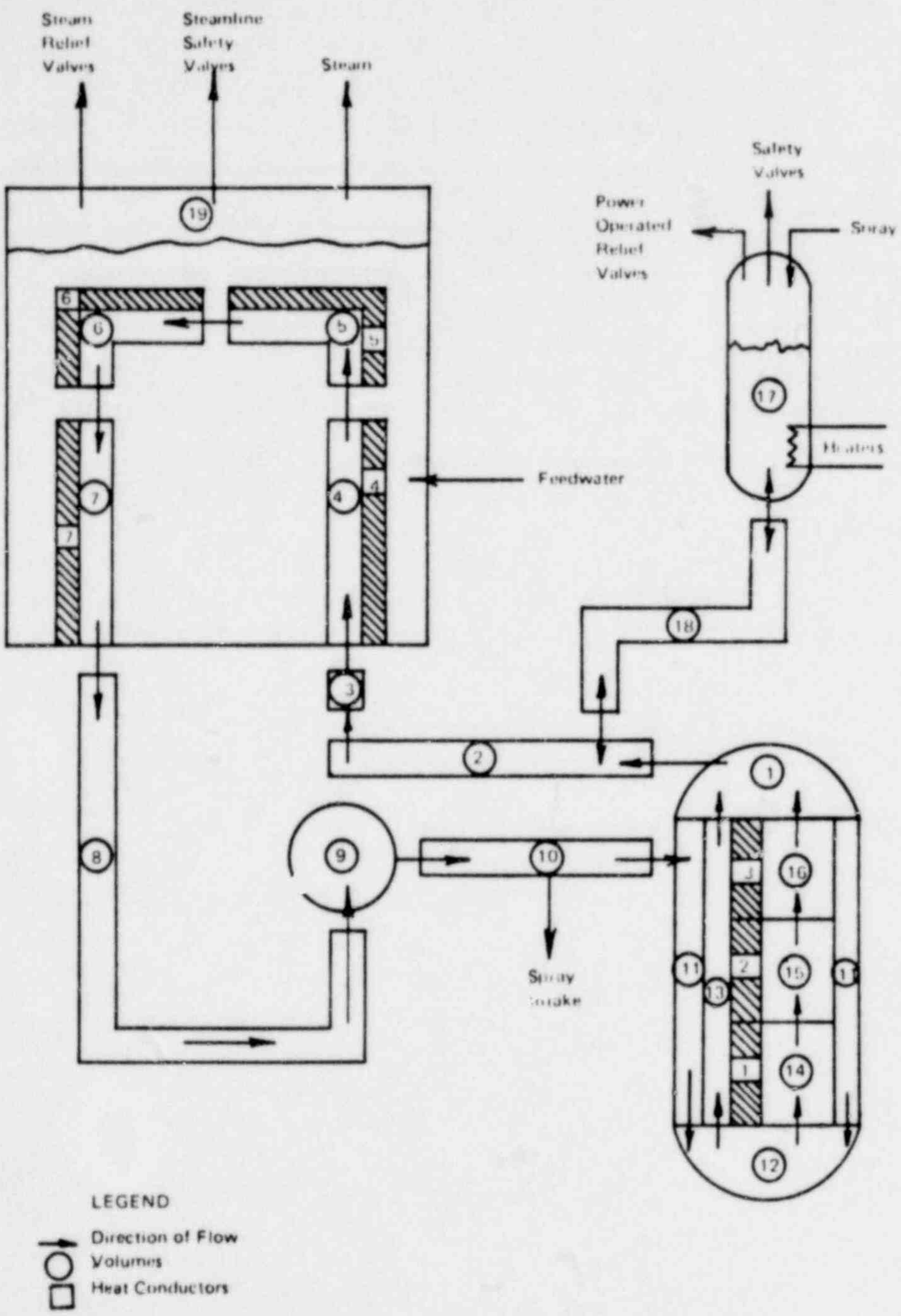

FIGURE VI.1-11 One-Loop Surry RETRAN Model.

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The fills for the pressurizer spray and cold leg intake are controlled by control system models. The control system logic for the pressurizer spray is shown in Figure VI.1-12. The spray flow, pressurizer heater and one relief valve are controlled by the output signal of a proportional-plus-integral controller. The backup heater is modeled either "on" or "off" based on a trip controlled by the controller output. The controllers for the overtemperature and overpower ΔT trips are shown in Figures VI.1-13 and VI.1-14. The temperature signals for these trips were taken at the steam generator inlet plenum and the downcomer.

The initial conditions for the analysis were computed using the self-initialization option in the code. All the junction loss coefficients were input with the exception of values for the vessel inlet nozzle junction and the junction at the exit of the core bypass.

1.3.2 Results of Analysis

The UCRW transient analyzed by Smith was for a reactivity insertion rate of 2.0×10^{-5} ΔK/sec. Results of the analysis are given in Figures VI.1-15 to VI.1-18 along with the FSAR predictions. There is very good agreement between the values for reactor power (Figure VI.1-15), when the unknown differences between the RETRAN and vendor input is considered. The pressurizer pressure response for the two cases is shown in Figure VI.1-16. There is very good agreement between the calculations for pressurizer pressure and the average coolant temperature (Figure VI.1-17). The DNB ratio as computed by the two codes is given in Figure VI.1-18. Although the shape of the two curves is similar, RETRAN predicts a higher minimum DNB ratio than given by the FSAR analysis. This difference is in large part due to the different initial values between the two codes and possibly due to different CHF correlations and grid spacer correction factors. It is also noted that the RETRAN DNB model is to be used only for scoping calculations, as discussed in Section IX.3.6.

1.4 Summary of Results

The comparisons of RETRAN calculations and the results of vendor analyses presented by Poteralski[VI.1-1] and Smith[VI.1-3] show that RETRAN compares very favorably with vendor codes when modeling uncontrolled rod withdrawal transient. Areas where additional analyses or sensitivity studies are required include

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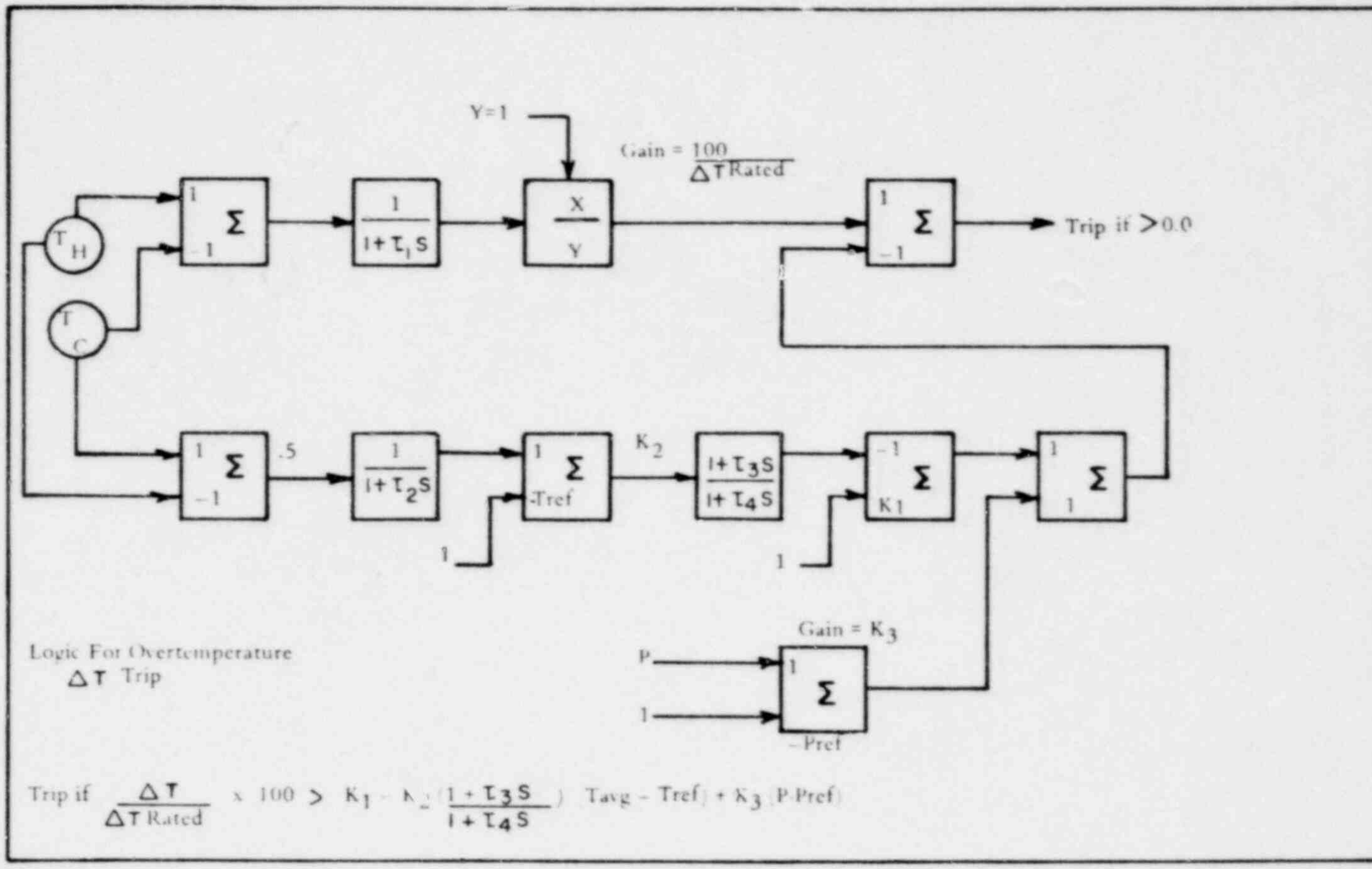

Overtemperature ΔT Trip Logic

FIGURE VI.1-12 Logic for Overtemperature ΔT Trip.

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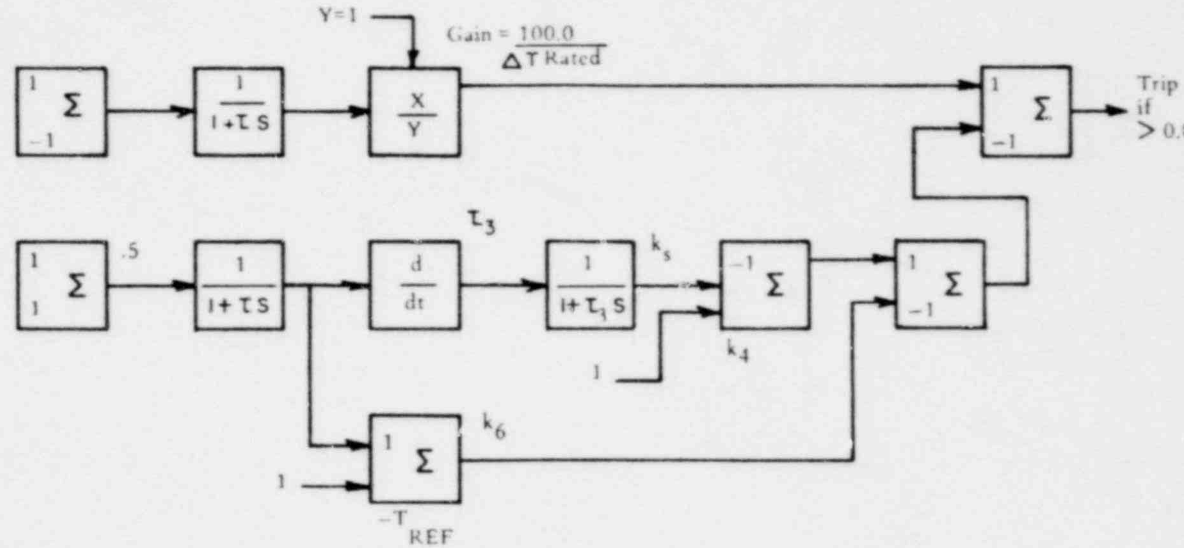
Overpower ΔT Trip Logic



Logic for Overpow r ΔT Trip

$$\text{Trip if } \frac{\Delta T}{\Delta T \text{ Rated}} \times 100 > K_4 - K_5 \frac{\tau_3 S}{1+\tau_3 S} \quad T_{avg} - K_6 (T_{avg} - T_{ref})$$

FIGURE VI.1-13 Logic for Overpower ΔT Trip.

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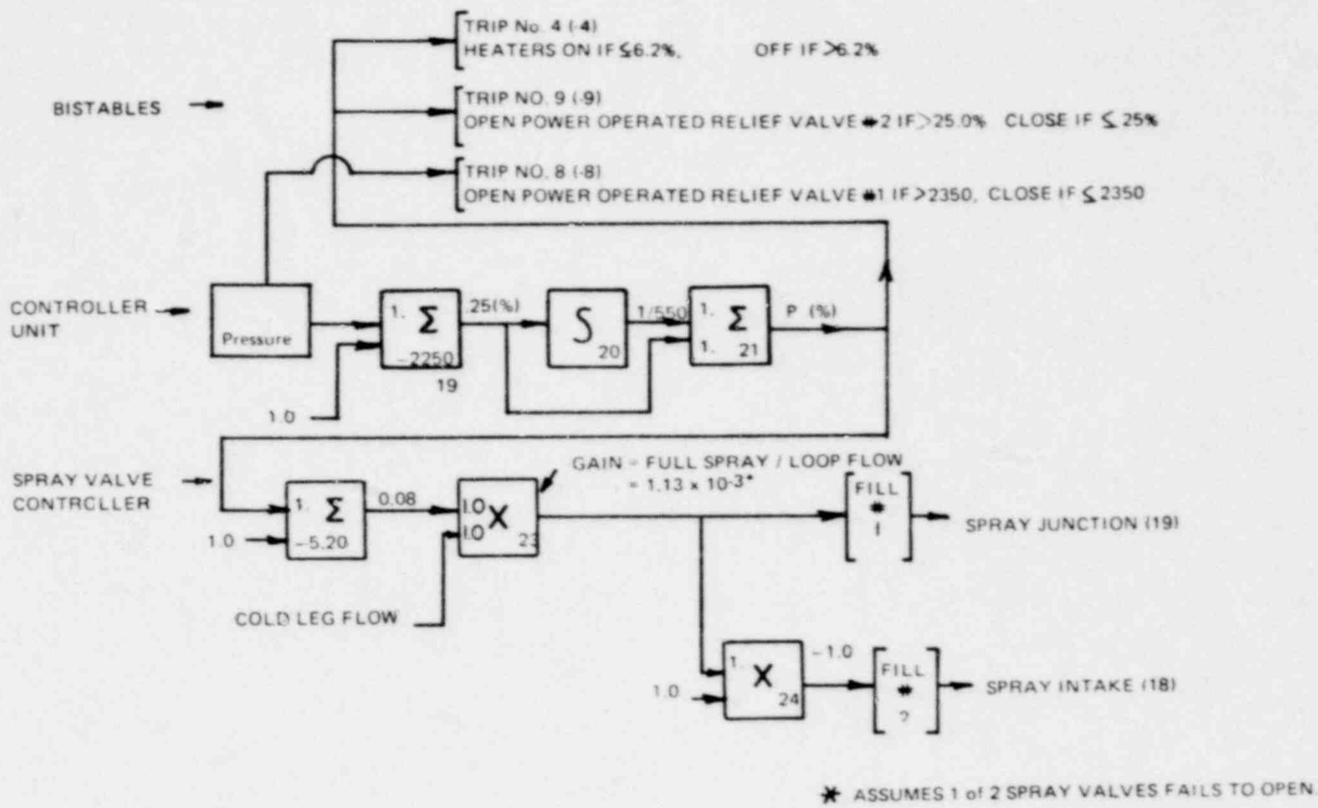

FIGURE VI.1-14 Logic for Pressure Control System.

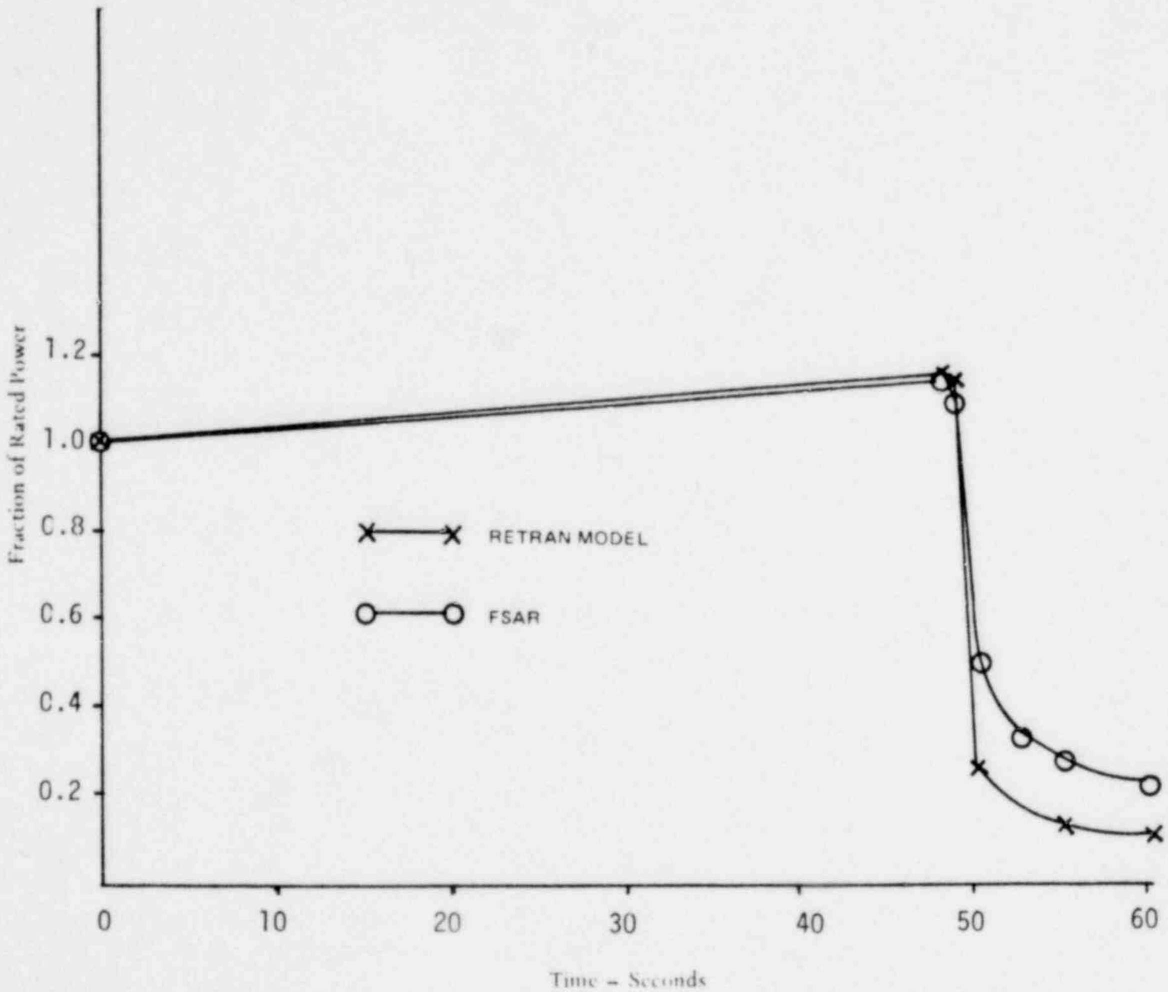

FIGURE VI.1-15 Power Response for UCRW.

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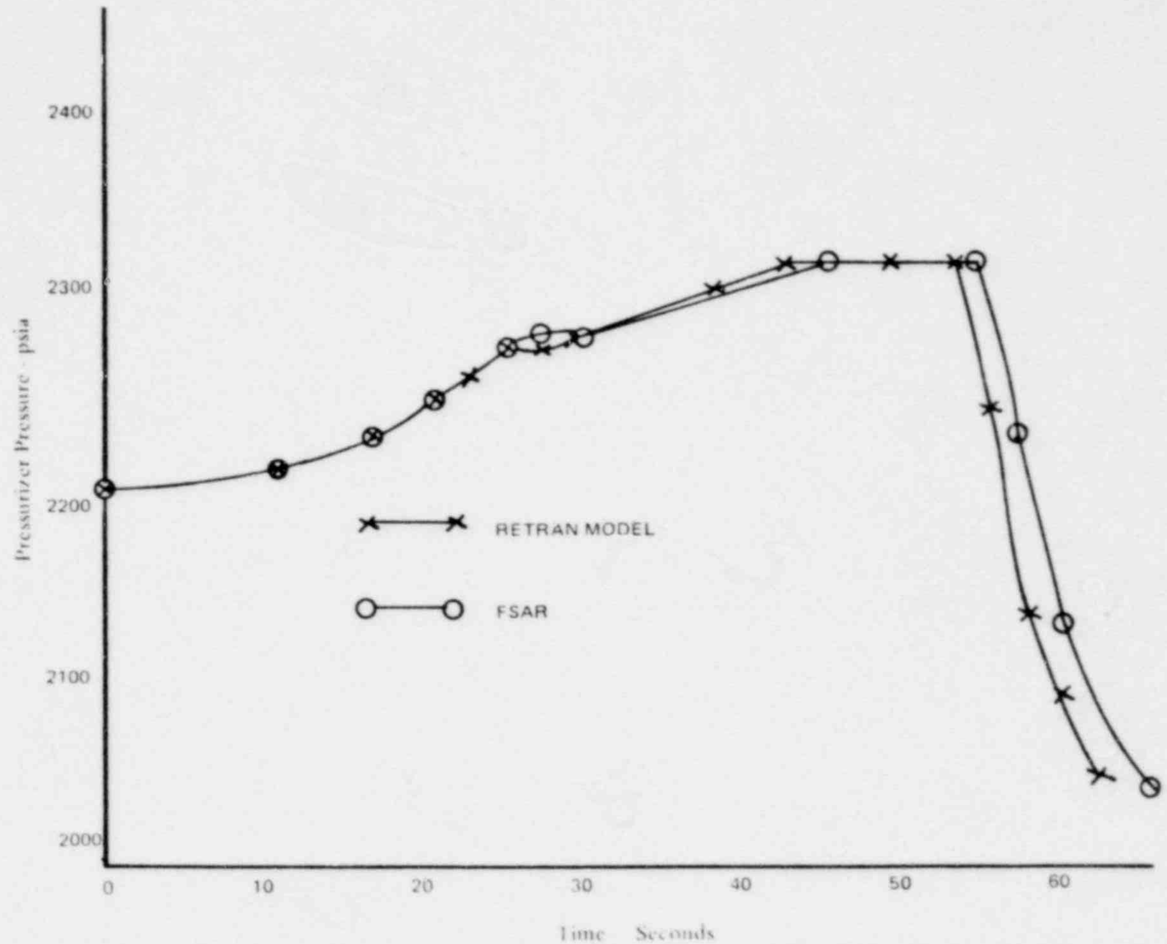

FIGURE VI.1-16 UCRW Pressurizer Pressure.

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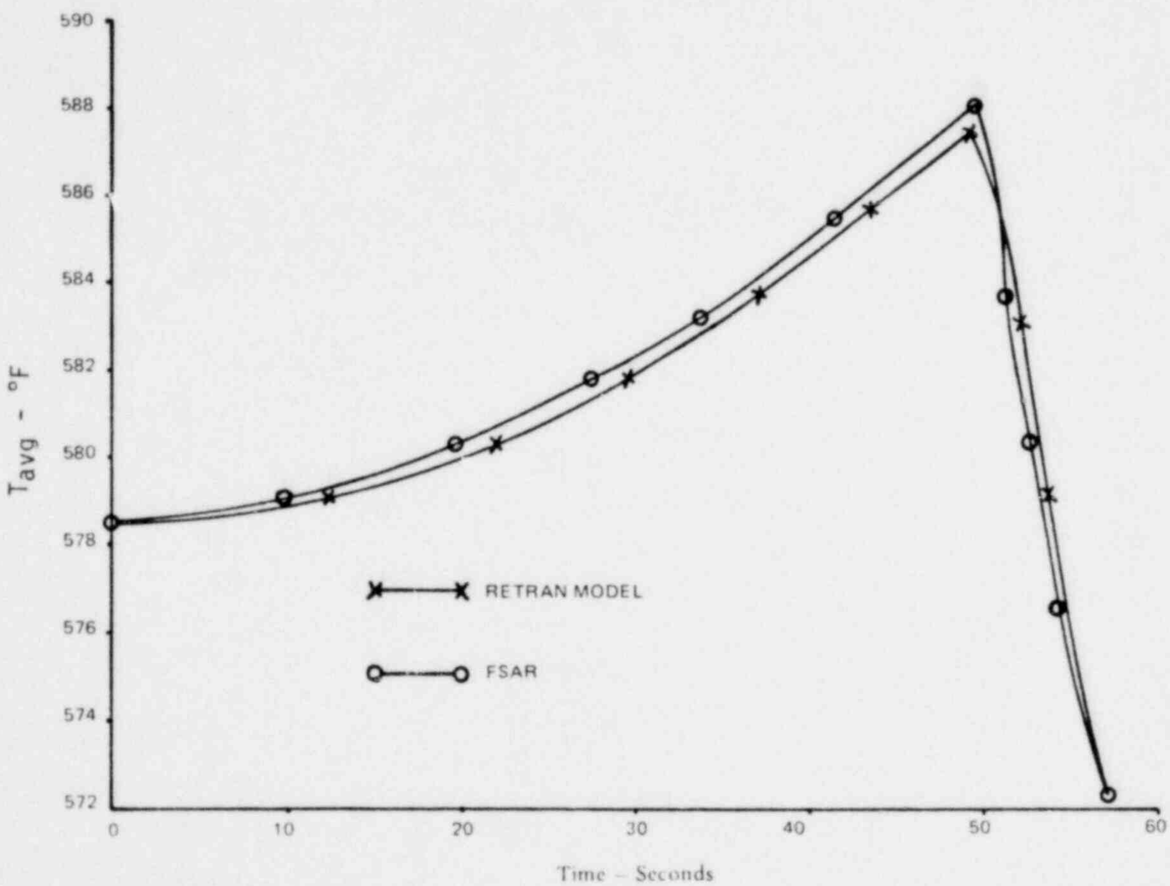

FIGURE VI.1-17 UCRW Average Coolant Temperature.

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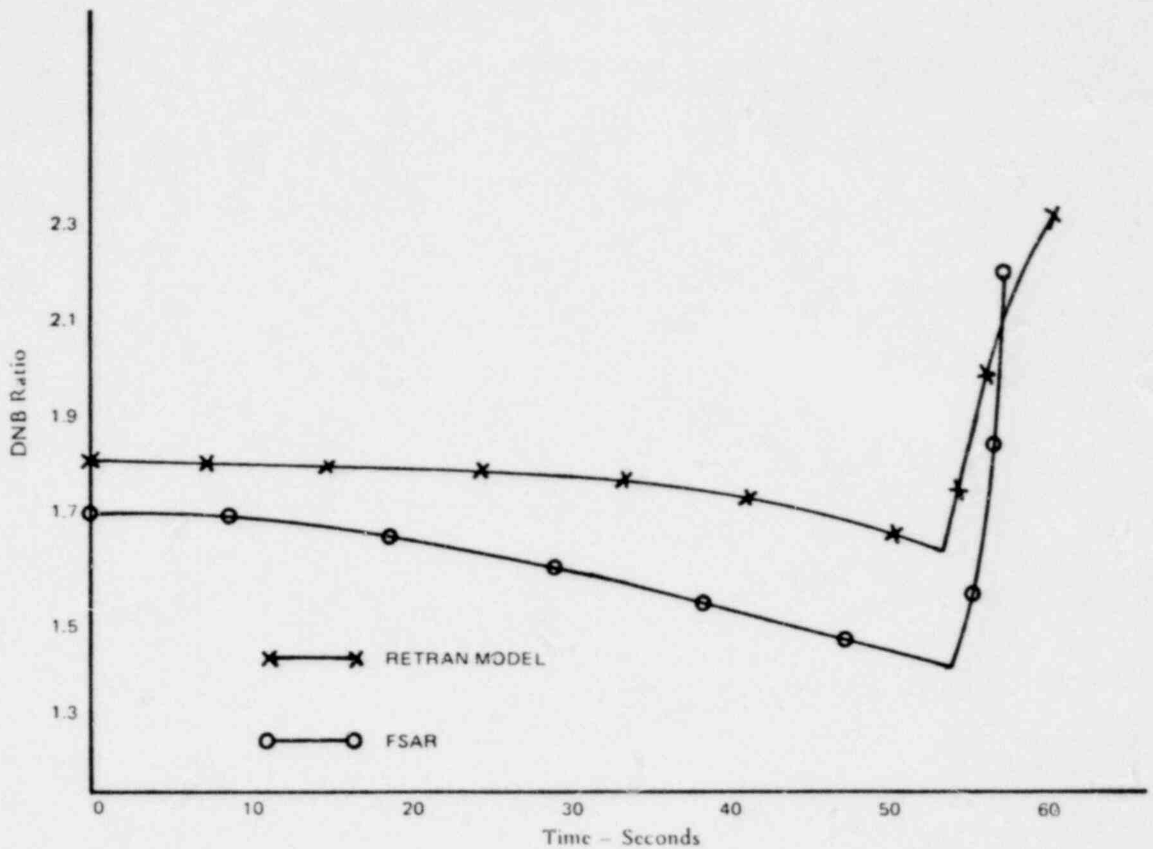

FIGURE VI.1-18 DNBR Response for UCRW Base Case.

9 318

modeling of reactivity feedback and the overtemperature and overpower ΔT trips as well as further investigation of the DNB calculation. Detailed information for the vendor analysis as well as additional RETRAN calculations will help to further evaluate this transient.

9 319

2.0 LOSS OF FLOW

The loss of reactor coolant flow transient is initiated by a loss of power to all the reactor coolant pumps. After the loss of power, a scram signal is activated. The pressurizer pressure and average coolant temperature initially increase before decreasing later in the transient. The loss of flow transient produces a low DNBR value, and thus the fuel characteristics and pump response are important items for this transient.

Three analyses were completed for the loss of flow transient. Poteralski[VI.2-1] evaluated this transient for beginning of core life conditions and compared the results to a vendor analysis. Carlson[VI.2-2] performed sensitivity studies of the pump rotating inertia, and Cross[VI.2-3] analyzed this transient for both beginning and end of core life conditions. Additional analyses by Poteralski and Cross on pump coastdown tests are presented in Section 3.0.

2.1 Florida Power and Light Analysis

The loss of flow transient analyzed by Poteralski[VI.2-1] was for a two-loop model of the Turkey Point units 3 and 4. The model was initialized at hot, full power for beginning of core life conditions, and the results were compared with vendor calculations.

2.1.1 Description of Model

The RETRAN model used for this transient is the same as described for the uncontrolled rod withdrawal analysis[VI.1-1] described in Section VI.1.1. Only the trips were modified for this analysis, with a scram trip included and set to occur 1.6 seconds into the transient. The pumps were tripped at 0.001 seconds.

2.1.2 Results of Analysis

Poteralski[VI.2-1] compared the RETRAN values of core flow, nuclear power and core average heat flux with vendor calculations from the FSAR. These comparisons are shown in Figures VI.2-1 to VI.2-3. The results agree very well with the exception of core flow values, in which case the RETRAN results are slightly lower than predicted by the vendor.

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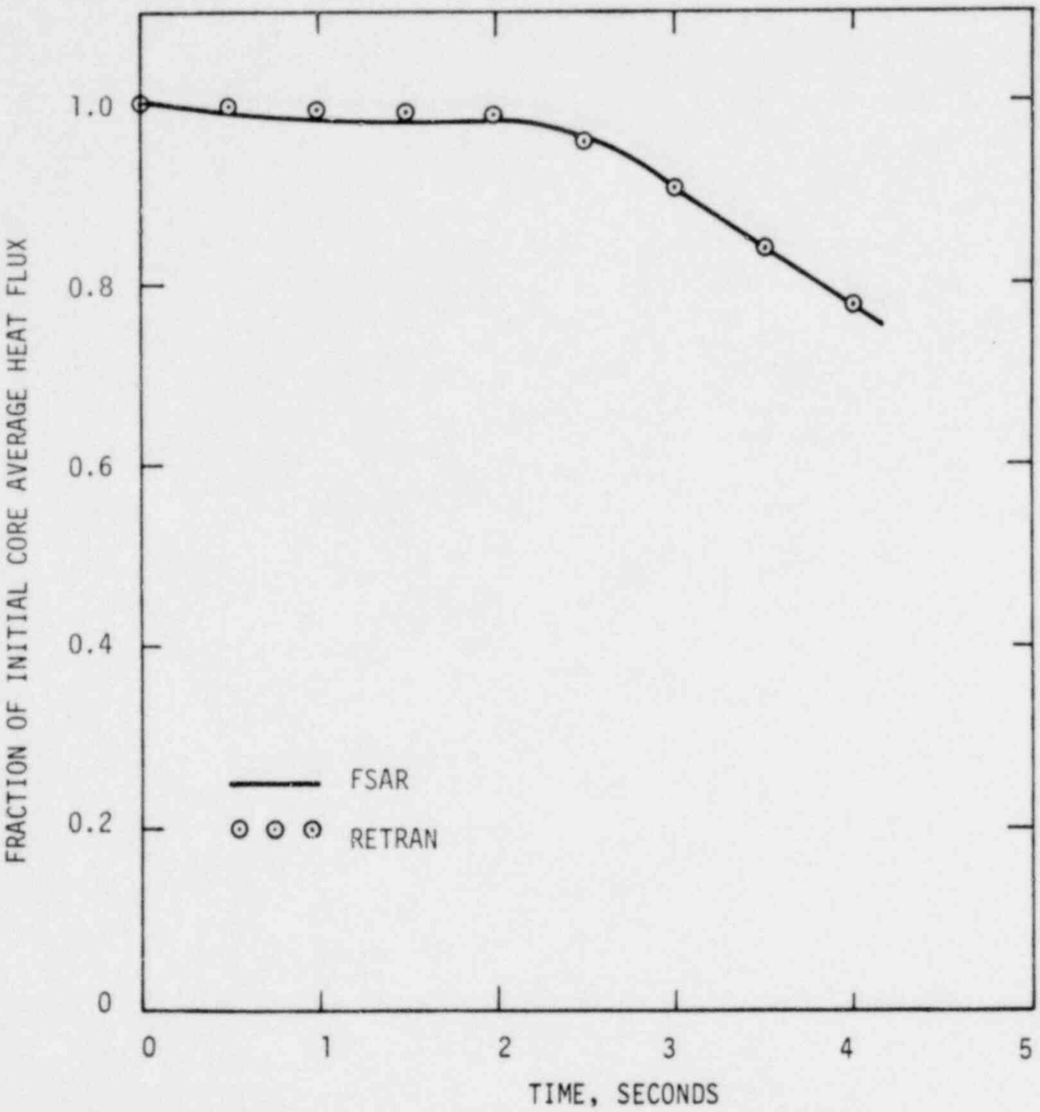

FIGURE VI.2-1 Core Average Heat Flux - Three Pump Flow Coastdown.

9 321

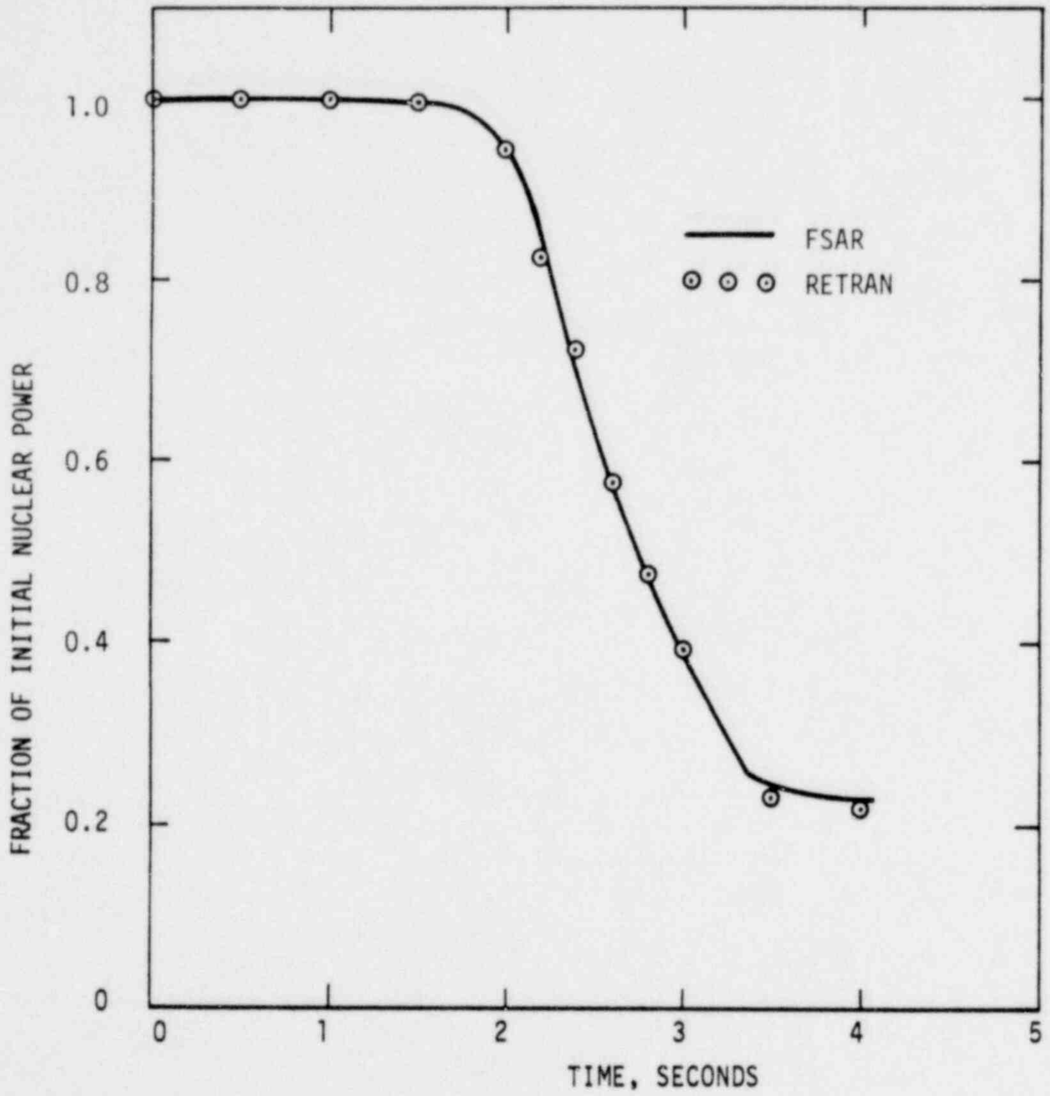

FIGURE VI.2-2 Normalized Core Power - Three Pump Flow Coastdown.

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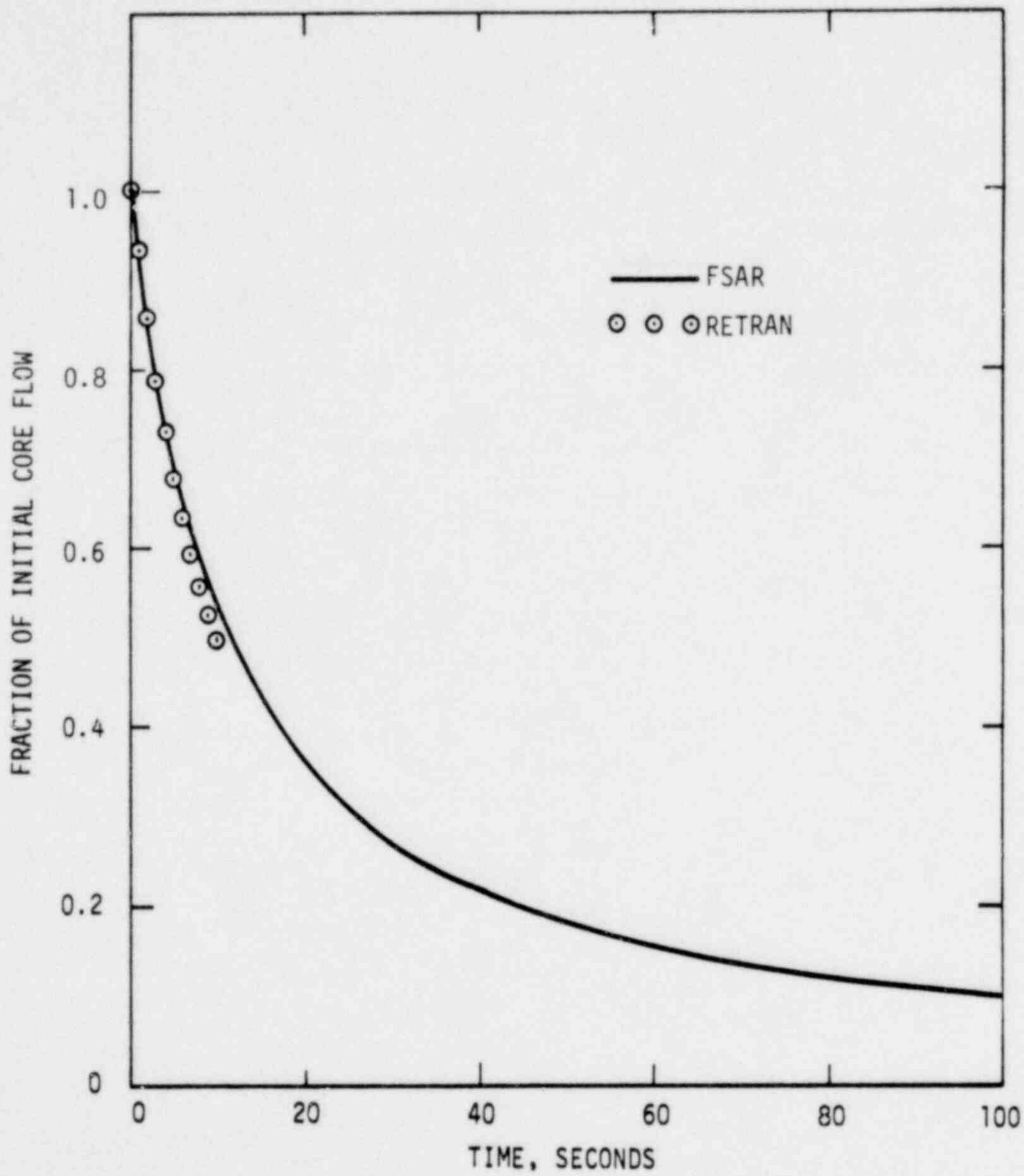

FIGURE VI.2-3 Normalized Core Flow - Three Pump Flow Coastdown.

2.2 <u>NUSCO Analysis</u>

The base RETRAN deck for the Connecticut Yankee plant was used to analyze the loss of flow transient. The core flow rate was compared to FSAR values, and Carlson[VI.2-2] also performed sensitivity studies to evaluate various values of pump rotating inertia.

2.2.1 <u>Description of Model</u>

The model used for these analyses is the same as was described in Section VI.1.2.

2.2.2 <u>Results of Analyses</u>

The core flow for the base case is shown in Figure VI.2-4 and it can be seen that the RETRAN results agree quite well with the vendor calculated values. Three sensitivity runs were also made for this transient by varying the rotating inertia of the pump. The first run was for an inertia value corresponding to the flywheel inertia, while the other two were for different values of the assembly inertia. These results are shown in Figure VI.2-5.

2.3 <u>VEPCO Analyses</u>

Cross[VI.2-3] analyzed the loss of flow transient with the Surry one-loop RETRAN model[VI.1-3] for both beginning and end of core life conditions. A detailed discussion of the results is given in Reference VI.2.3.

2.3.1 <u>Description of Model</u>

The one-loop model of the Surry reactors, described in Reference VI.1-3 and Section VI.1.3, was used for this analysis. The self-initialization option was used with pressure and enthalpy values specified at different locations than for the uncontrolled rod withdrawal case. This was done to verify the self-initialization option of the code, and the same steady-state solution was obtained for both cases.

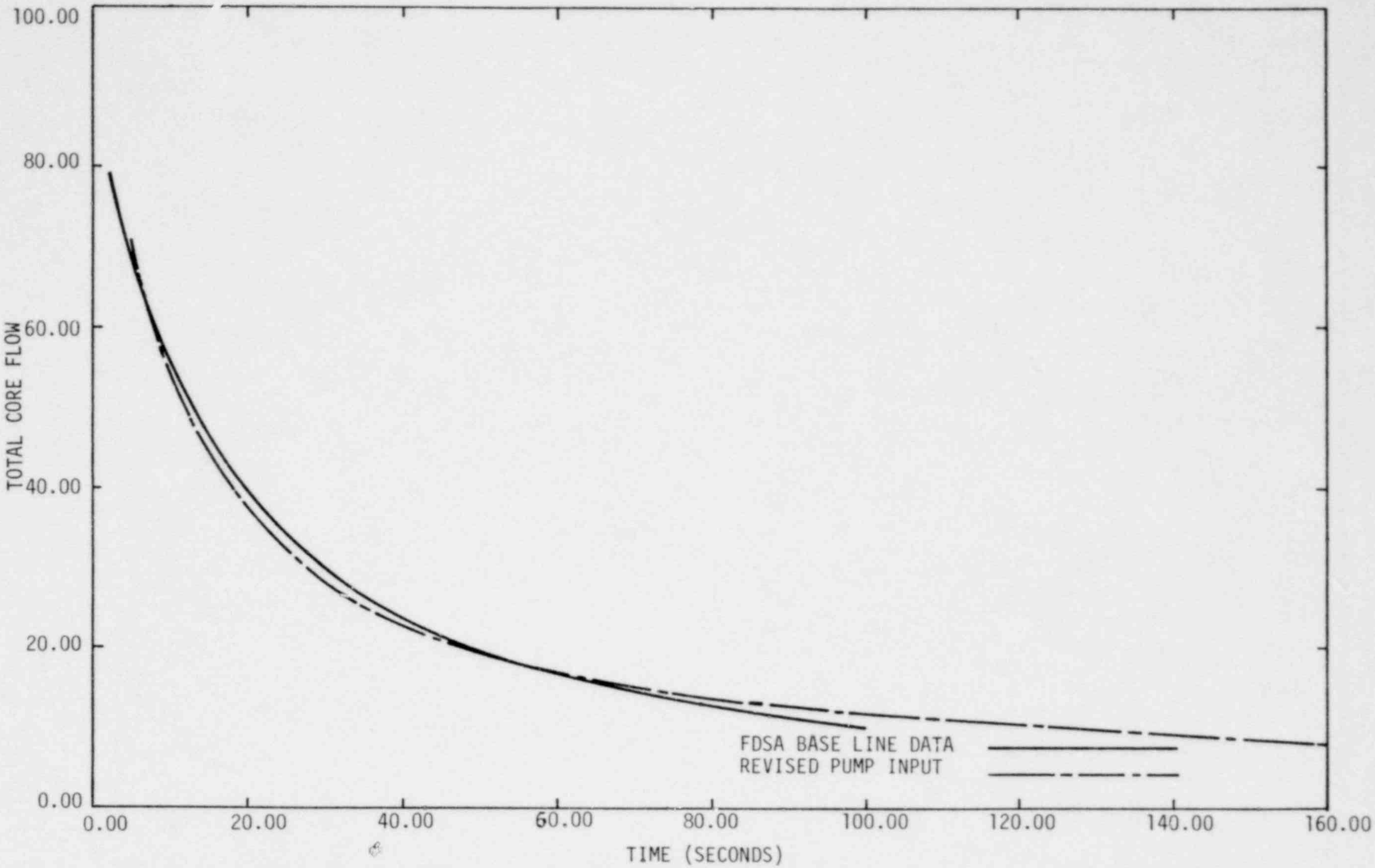

FIGURE VI.2-4 Core Flow - Four Pump Loss of Flow.

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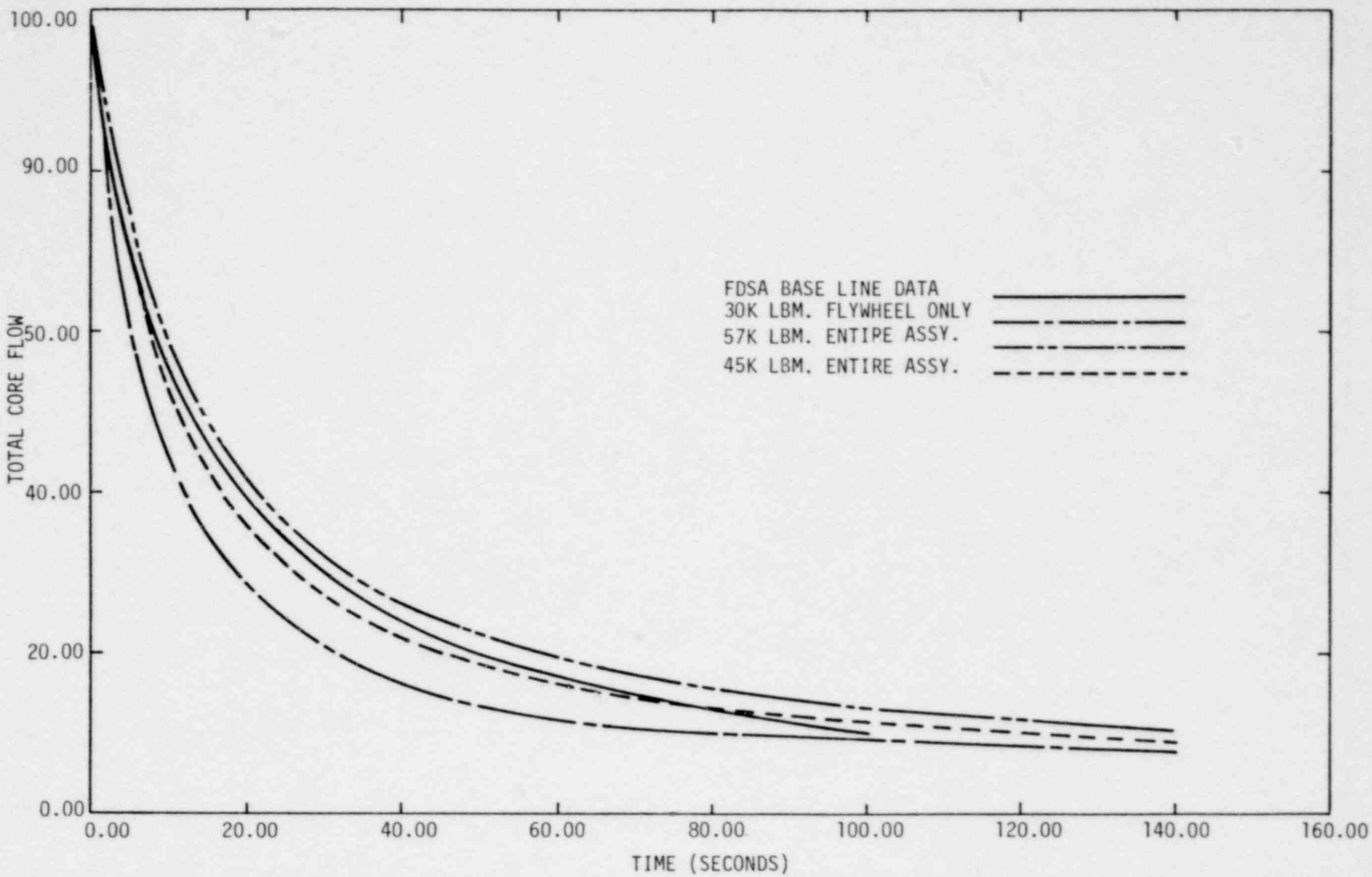

FIGURE VI.2-5 Pump Inertia Sensitivity Study - Four Pump Loss of Flow.

9 326

2.3.2 Results of Analyses

Two analyses of the loss of flow transient were performed, one which used fuel conditions corresponding to the beginning of core life (Case A) and the second for end of life conditions (Case B). The core power and average heat flux values calculated by RETRAN are compared with vendor FSAR results in Figures VI.2-6 and VI.2-7. The agreement between the computed values early in the transient is very good. The differences later in the transient are attributed to different assumptions for decay heat. The results also show that the total reactivity is about the same for both cases at any point in the transient, even though the Doppler coefficient is different for both cases. Values of the DNB ratio are shown in Figure VI.2-8. The trend of the values for the two cases is similar, although the minimum value is lower and achieved sooner for the FSAR than is computed by RETRAN. These differences are due in part to different DNB models in the two codes. As a result of this analysis and other studies conducted by VEPCO, an additional option has been added to RETRAN to provide DNB models similar to those being used by Westinghouse. The use of the auxiliary DNB model is discussed in Section IX.3.6.

2.4 Summary of Results

The comparisons of RETRAN and vendor calculations for the loss of flow transient presented in the previous sections show the codes give very similar values for the parameters of interest. The analyses performed by Cross [VI.2-3] indicate that the transient is relatively insensitive to Doppler reactivity effects for coefficient values corresponding to beginning and end of life conditions. This analysis also indicated deficiencies in the RETRAN DNB model when applied to some Westinghouse reactors. Revisions have been made in the model and are included in the RETRAN code.

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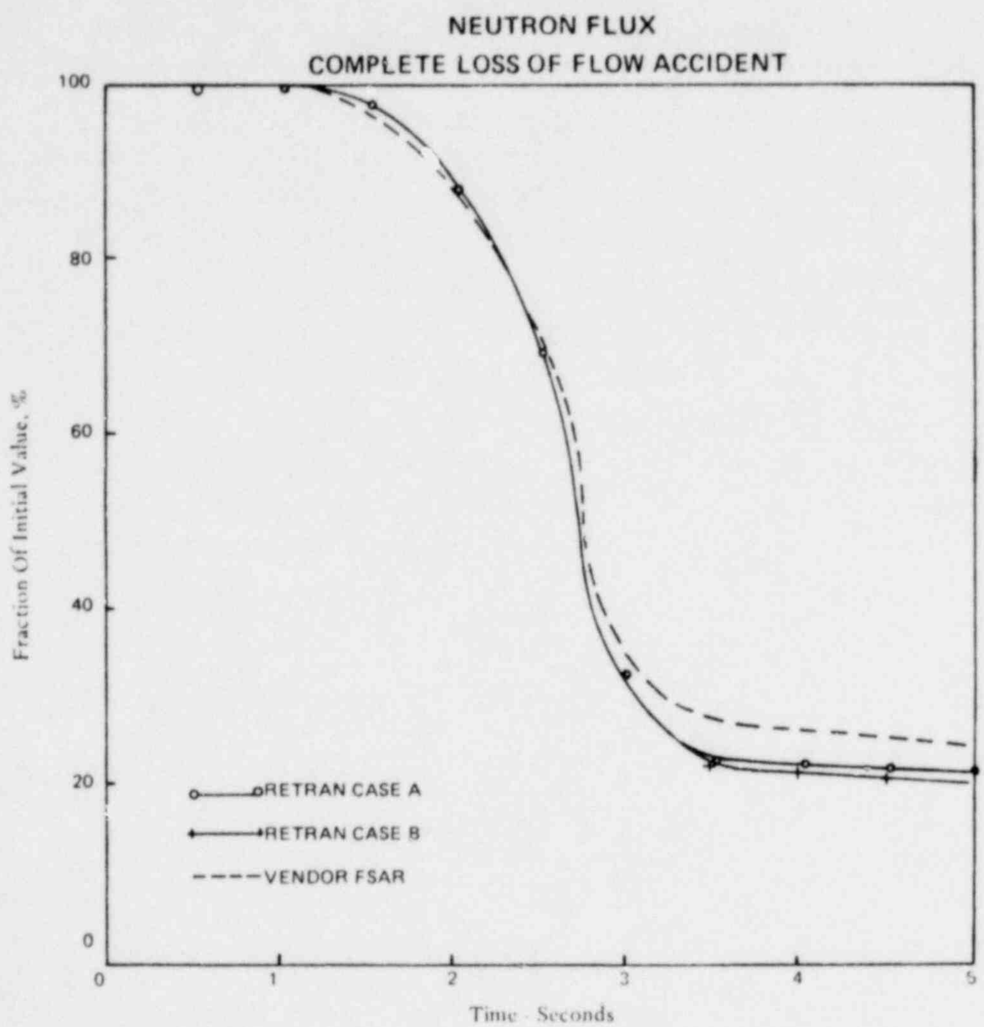

FIGURE VI.2-6 Neutron Flux - Loss of Flow Accident.

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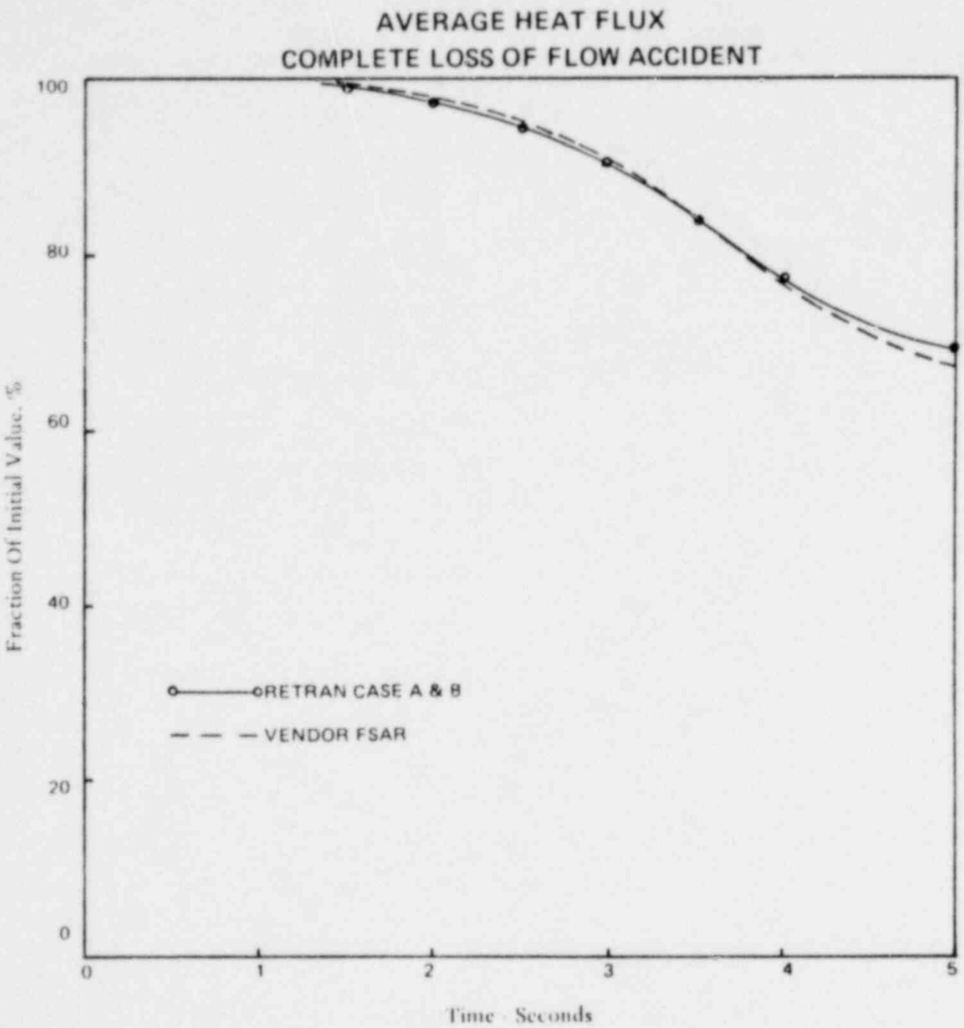

FIGURE VI.2-7 Average Heat Flux - Loss of Flow Accident.

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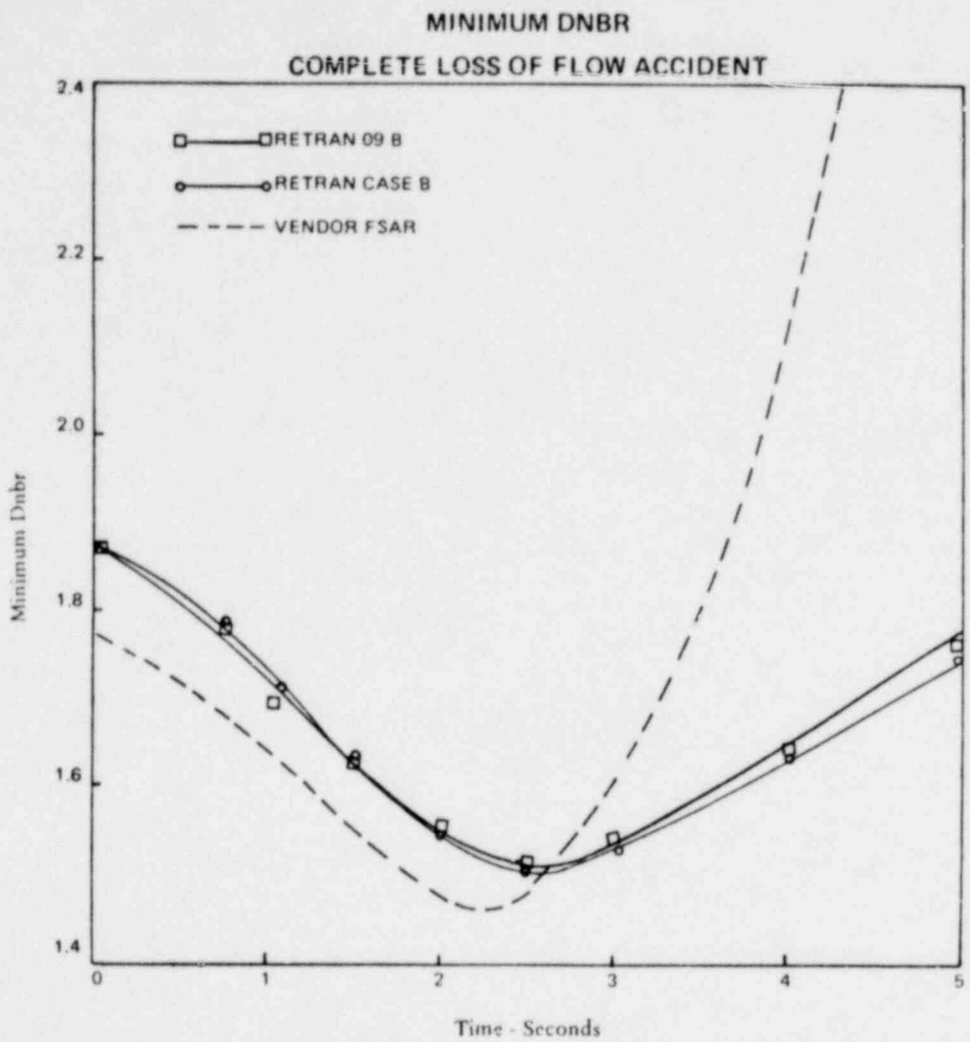

FIGURE VI.2-8 Minimum DNBR - Loss of Flow Accident.

9 330

3.0 PUMP COASTDOWN

During preoperational testing for nuclear power plants, a number of tests are conducted in which one or more pumps are tripped. These pump coastdown tests are used to evaluate the response of the system to a loss of reactor coolant flow. The tests are initiated by tripping the pumps in various combinations, and measurements of coolant flow are made.

Three utilities performed RETRAN analyses of these tests and compared the calculations with the test flow data and vendor calculations. Poteralski[VI.3-1] analyzed one, two and three pump trips for a three loop plant. Piluso and Herborn[VI.3-2] evaluated three and four pump trip transients for a four loop PWR, while Cross [VI.3-3] modeled a test in which all three pumps of the Surry plant were tripped. These analyses and comparisons with test data are summarized in the following sections.

3.1 Florida Power and Light Analyses

Poteralski[VI.3-1] compared RETRAN pump coastdown calculations with test data for the Turkey Point units 3 and 4. He also performed a study to evaluate the sensitivity of pump torque to this analysis and compared these results with the FSAR value of flow.

3.1.1 Description of Model

The two-loop Turkey Point model described in Reference VI.1.1 and Section VI.1.1 was used for these calculations. The test was conducted with control rods inserted (hot, zero power conditions). This change, along with the appropriate trips, were the only input modifications made for the calculations.

3.1.2 Results of Analyses

The measured and calculated values of flow response when the pump in loop one is tripped are shown in Figure VI.3-1. There is good agreement between the RETRAN results and the test data for flow in both loops and the core. Figure VI.3-2 shows similar curves for the case in which the two pumps in loop 2 are tripped while the pump in loop 1 is allowed to run. These results are also in good agreement with the data.

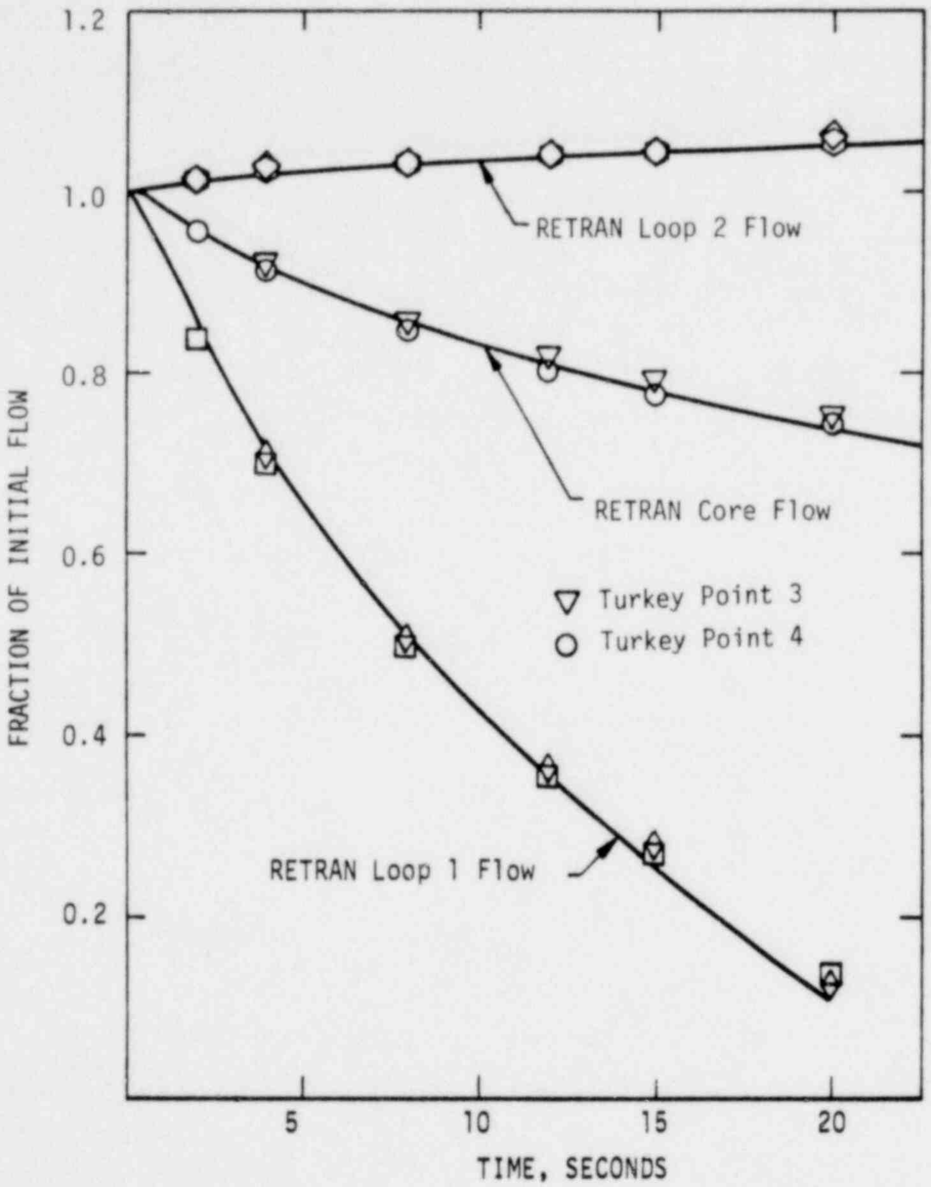

FIGURE VI.3-1 RETRAN and Experimental Flows for One Pump of Three Flow Coastdown.

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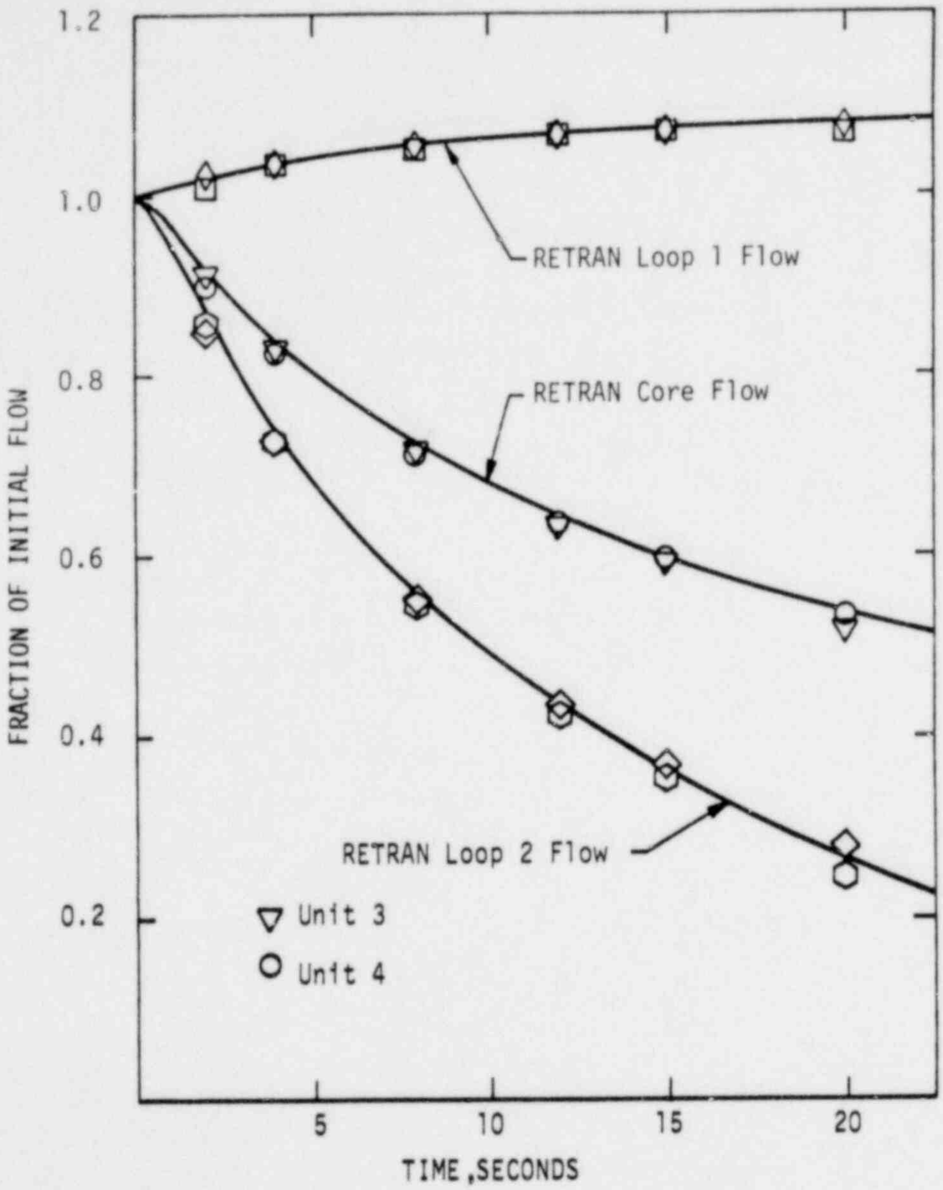

FIGURE VI.3-2 RETRAN and Experimental Flows for Two Pumps of Three Flow Coastdown.

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Results of tripping all three pumps simultaneously are shown in Figure VI.3-3, with the RETRAN values comparing very well with the test data. To investigate the effect of pump hydraulic torque on the calculation, the value was increased from the 22,300 ft-lb value used in the previous analysis to a value of 26,500 ft-lb. Values of core flow for the three pump trip are compared to FSAR calculations in Figure VI.3-4.

3.2 Portland General Electric Analyses

The Trojan plant is a four loop PWR, and Piluso and Herborn[VI.3-2] analyzed three and four loop pump trip transients. The results of these calculations were compared with data taken during preoperational tests and with vendor calculations.

3.2.1 Description of Model

The Trojan reactor was modeled as a two loop plant for the three pump coastdown analysis and as a 1 loop plant for the four pump coastdown calculations. The RETRAN representations are shown in Figures VI.3-5 and VI.3-6.

The 1 loop model has 20 volumes, 21 junctions and 7 conductors. The primary side of the steam generator has 4 volumes, while the secondary is modeled as a single volume with fill junctions used to represent the boundary conditions. The two loop model is similar to the one loop model and has 31 volumes, 35 junctions and 11 conductors. Further details of both representations are given in Reference VI.3-2.

Since the three pump trip test is not a symmetric test for this reactor, the two loop model was used for the RETRAN analysis. If an idle loop is modeled during steady-state conditions (t=0.0 sec), the flow through the remaining loops will be greater than if modeled under normal conditions. The method described by Herborn[VI.6-2], based on a determination of the value of the idle loop loss coefficient, was used to estimate the "correct" idle loop flow. This procedure required three runs with the self-initialization option, and is described in Reference VI.6-2.

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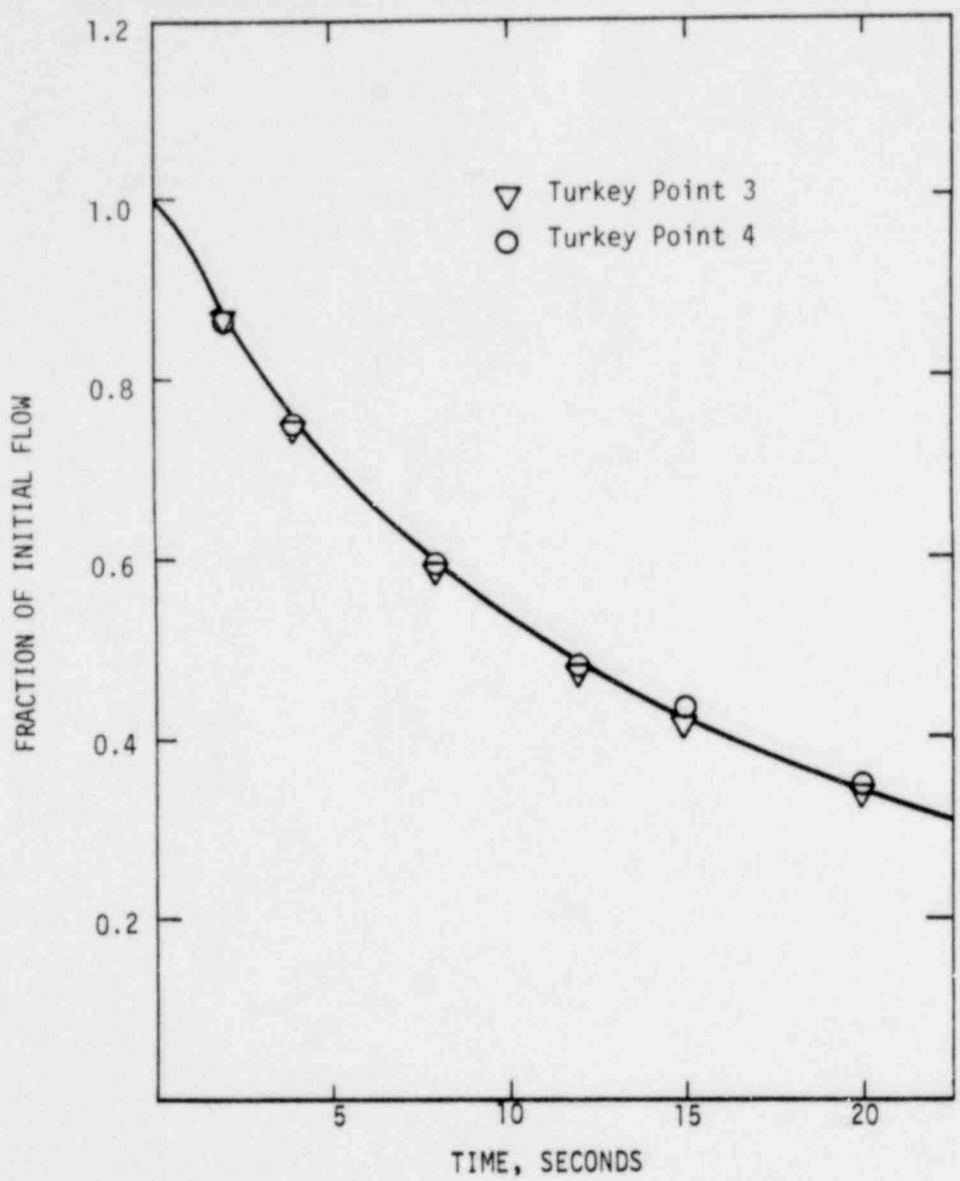

FIGURE VI.3-3 RETRAN and Experimental Flows for Three Pumps of Three Flow Coastdown.

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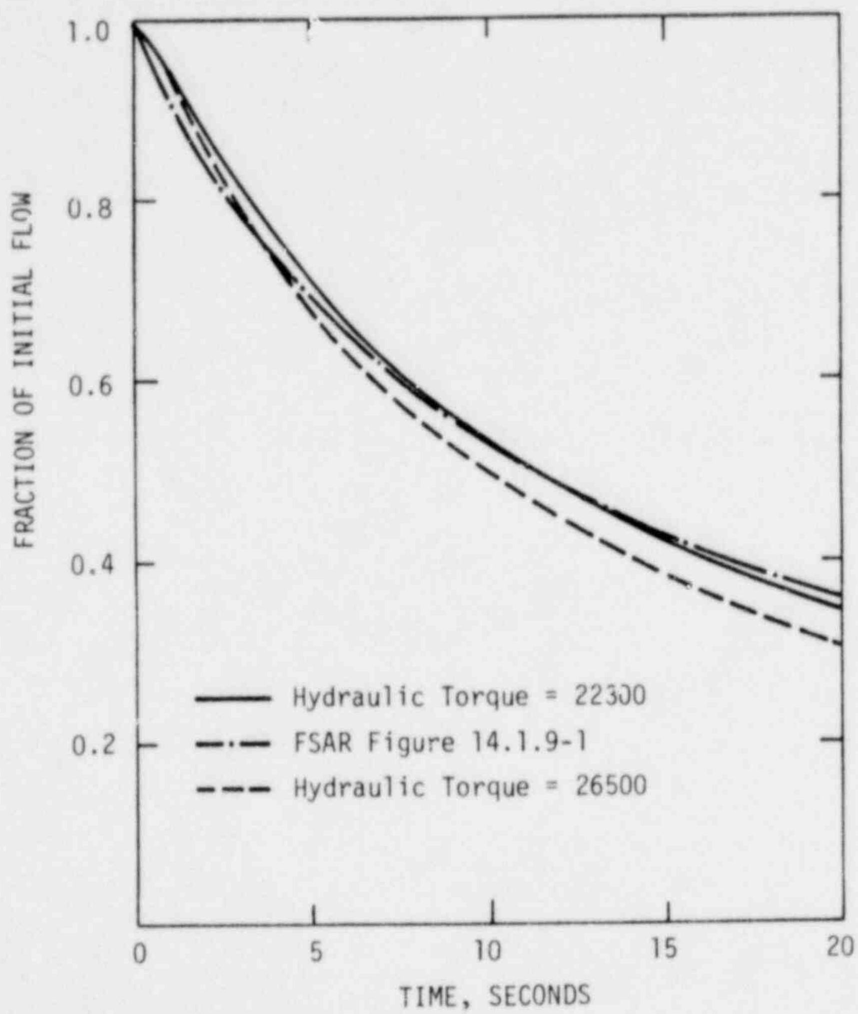

FIGURE VI.3-4 Comparison of RETRAN Coastdown Results With FSAR Three Pump Coastdown Curve.

779 336

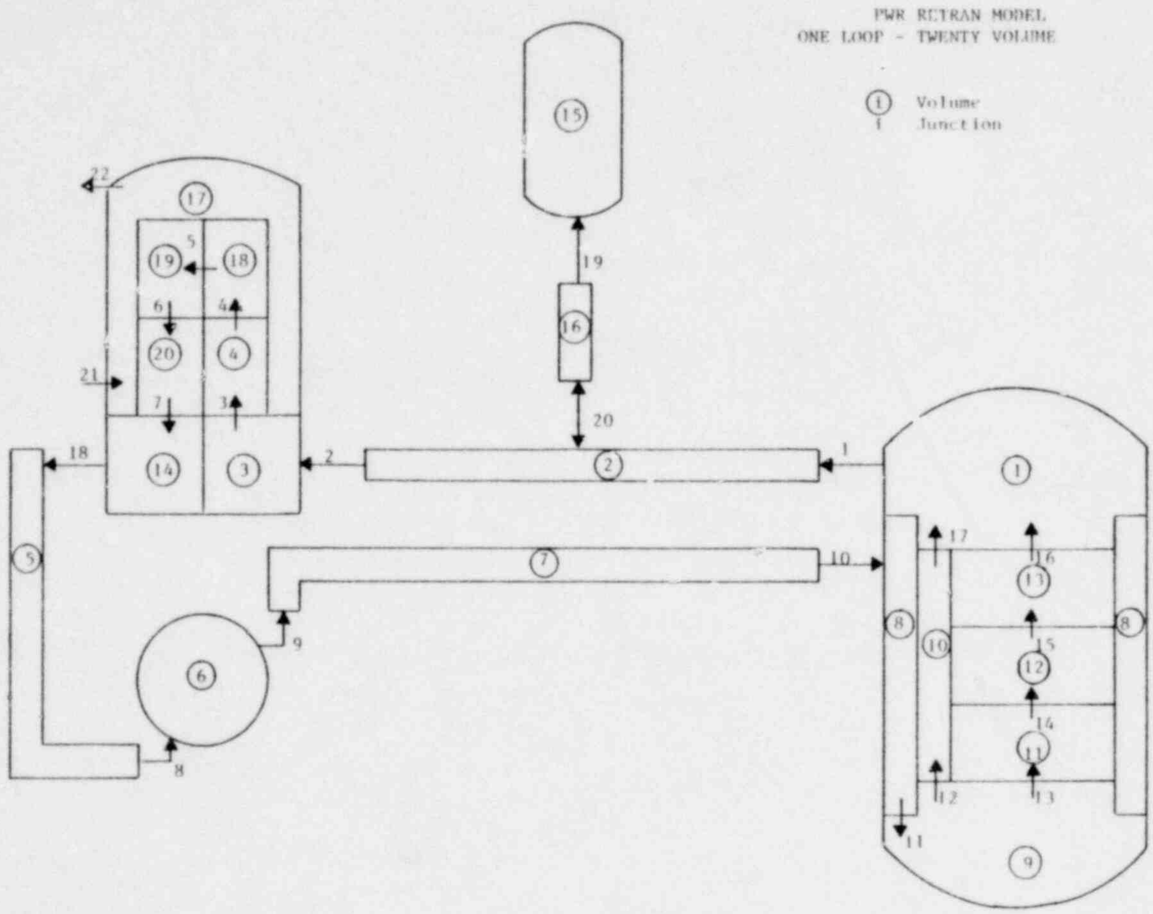

FIGURE VI.3-5 One-Loop Model of Trojan Reactor.

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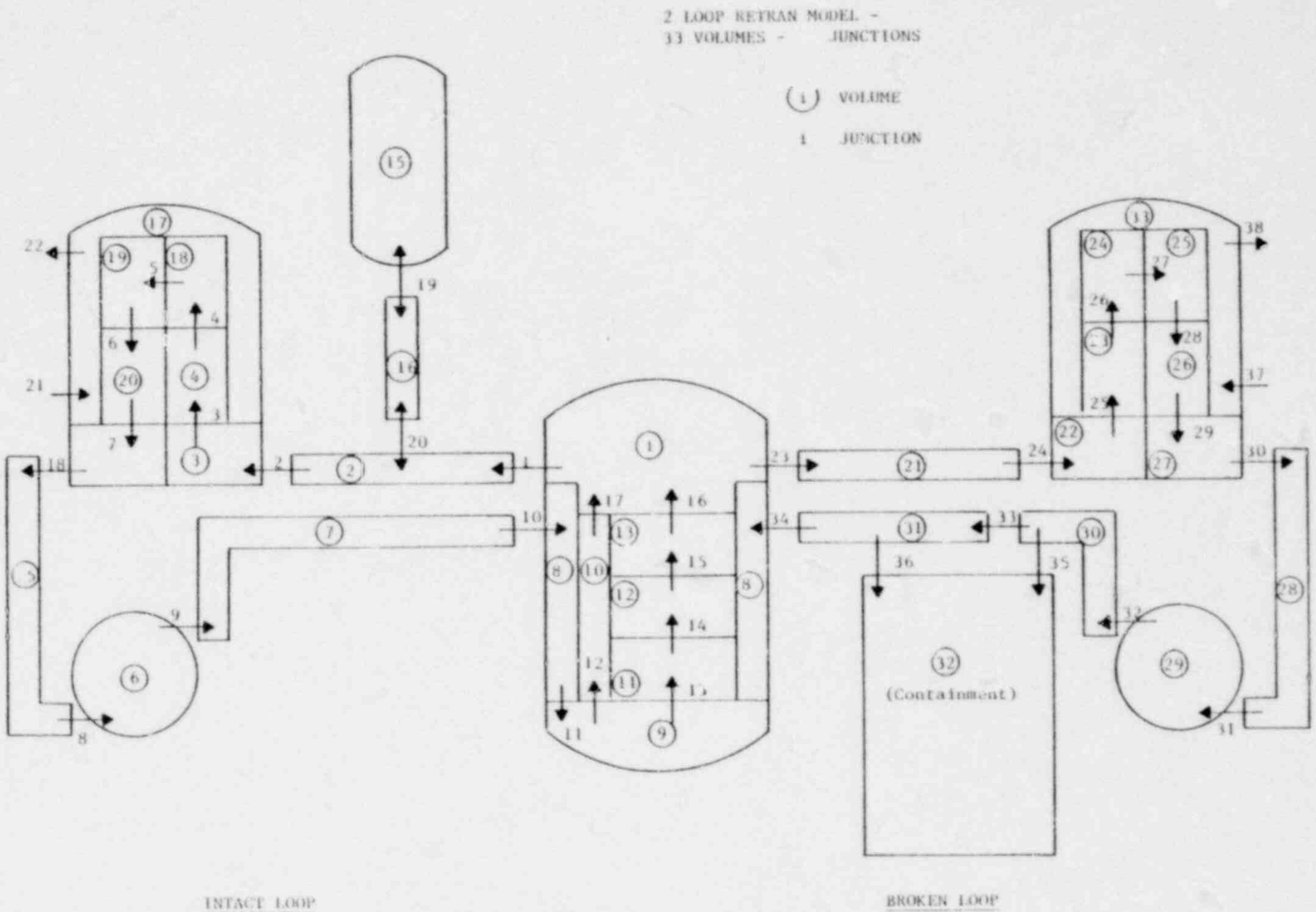

FIGURE VI.3-6 Two-Loop Model of Trojan Reactor.

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3.2.2 Results of Analyses

A summary of specific input parameters for these analyses is given in Table VI.3-1. It is noted that different values of hydraulic torque, pump head and rotating inertia are used for the best estimate (plant data) and vendor comparisons. The RETRAN calculated flow values for the three- and four-pump coastdown analyses are compared to the test data in Figure VI.3-7. The RETRAN values agree favorably with the test data during the initial phase of both analyses, but are above the data after 6 seconds. A comparison of the RETRAN values with a conservative vendor calculation is given in Figure VI.3-8 for the four pump coastdown, with the results in good agreement.

3.3 VEPCO Analysis

An analysis of a three pump coastdown test for the Surry reactor was performed by Cross[VI.3-3]. The results of this calculation were compared with the test data and with vendor FSAR values.

3.3.1 Description of Model

The one loop model of the Surry reactor, described in Section VI.1-3 and Reference VI.1-3, was used for this analysis. The only difference in RETRAN input data was for the trips, with the loss of power assumed to occur at 1.0 seconds.

3.3.2 Results of Analysis

Comparisons of the RETRAN values for flow with vendor calculations and test data are given in Figure VI.3-9. Both codes agree quite well with the data for the first 5 seconds, after which RETRAN underpredicts the coastdown, although by a smaller margin than does the vendor calculation.

3.4 Summary of Results

Comparisons of RETRAN calculated flow values with pump coastdown test data show relatively good agreement. In two analyses, the core flow predicted by RETRAN was above the test data after the early portion of the test. This could be due

TABLE VI.3-1

PUMP COASTDOWN ANALYSES - SUMMARY OF INPUT PARAMETERS

Parameter	Conservative Case	Best-Estimate Cases 4-Loop	Best-Estimate Cases 3-Loop
Power level, MW(t)	0	0	0
Pressure, psia	2250	2250	2250
Temperature, °F	553.3	553.3	553.3
Rated pump flow, gpm	88500	88500	88500
Rated hydraulic torque, ft-lbf	24228.5	23435.5	23435.5
Rated pump head, ft	284.3	276.75	276.75
Rotating inertia, lbm-ft2	73800	82000	82000
Initial loop flowrate, % design value	100	100	104.4

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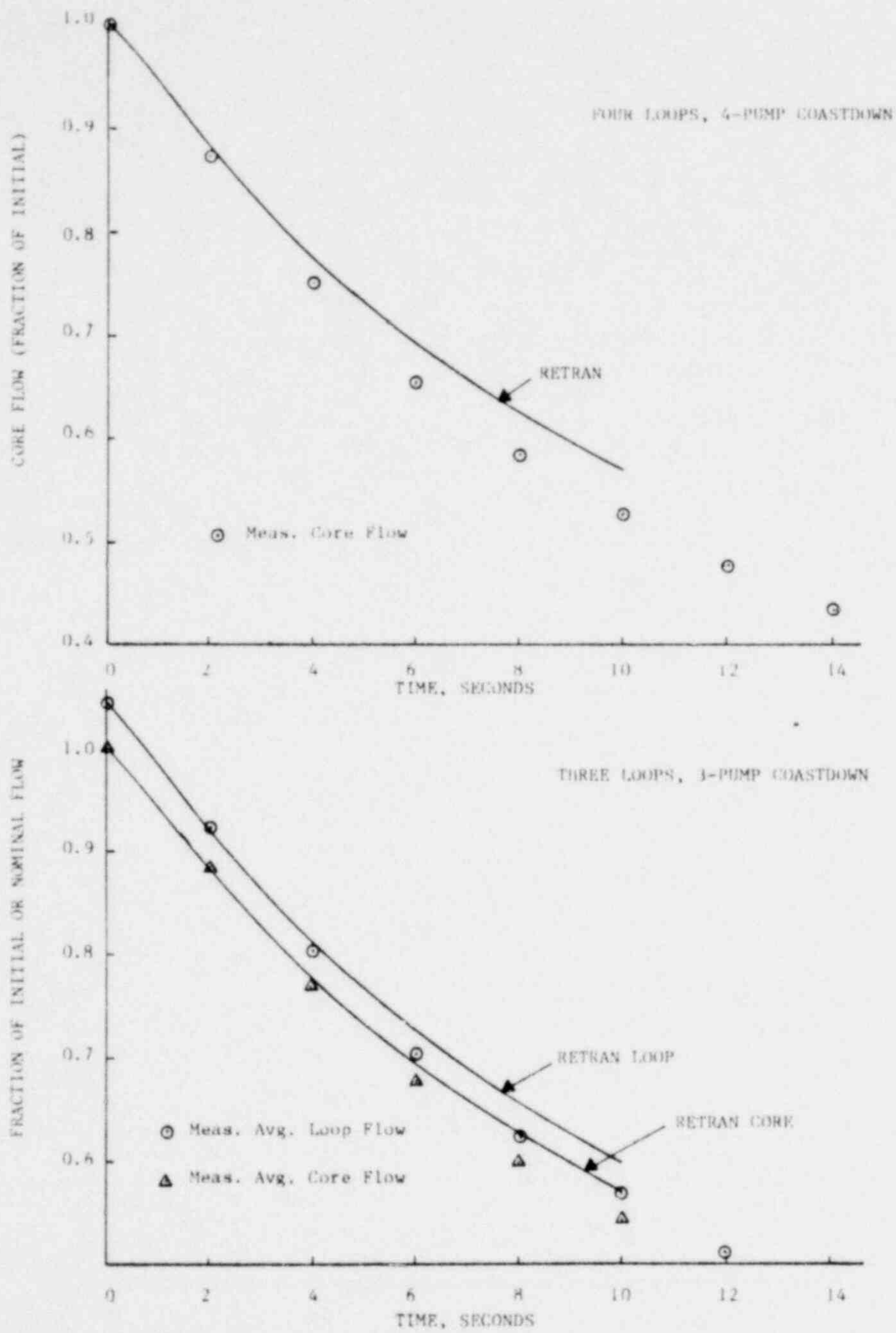

FIGURE VI.3-7 Summary of Coastdown Results.

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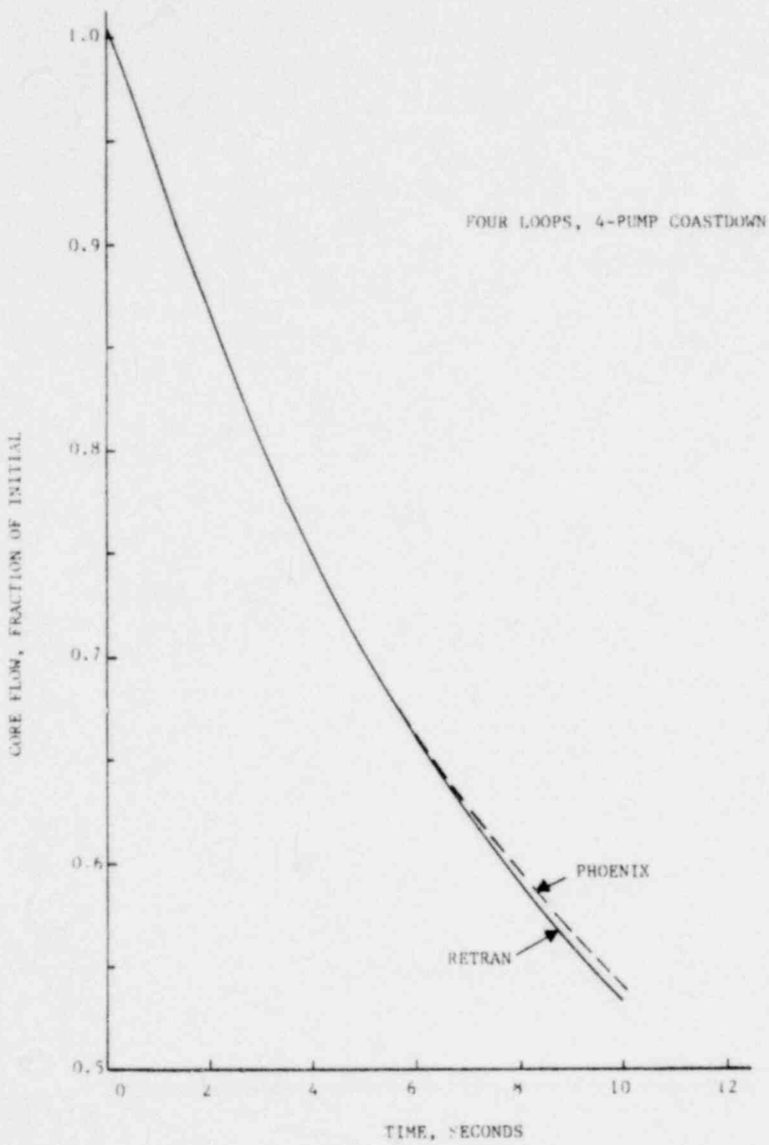

FIGURE VI.3-8 Comparison of RETRAN with PHOENIX.

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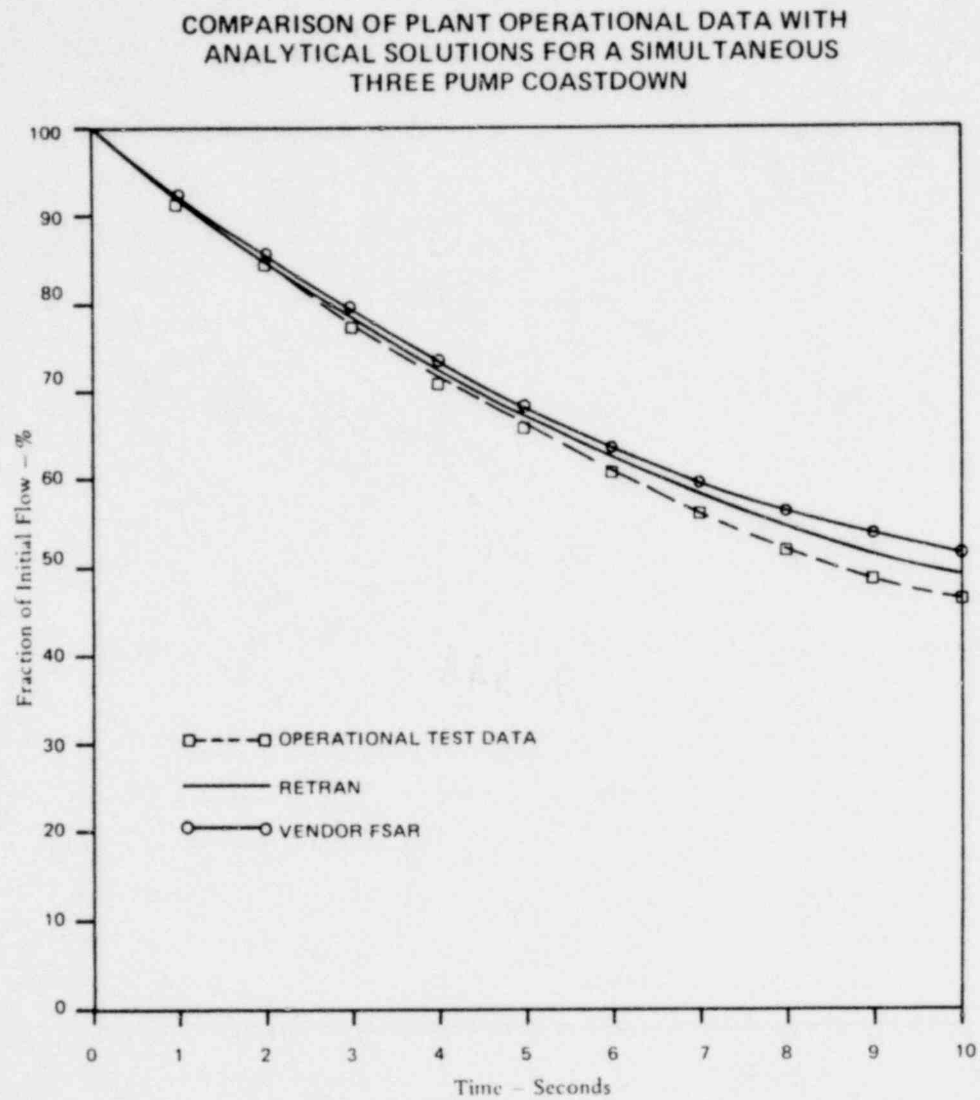

FIGURE VI.3-9 Comparison of RETRAN Flow Results With Test Data and FSAR Values for a Simultaneous Three Pump Coastdown.

9 343

to RETRAN input data (e.g., pump efficiency, friction option) or to the experimental data, the accuracy of which is unknown. The applicability of these particular models depends on the analysis being performed. If a loss of flow transient is considered, the RETRAN models may be adequate since the minimum DNB ratio is reached early in the transient. However, if a steam line break is being evaluated, additional studies most likely would be required since the boron transport and mixing are a function of the loop flows.

9 544

4.0 LOSS OF EXTERNAL LOAD

The loss of external load results in an unplanned decrease in heat removal from the secondary system. This transient is a Condition II event. In the transient modeled by VEPCO[VI.4-1], the pressurizer spray and the steam generator and pressurizer relief valves are assumed to be inactive. The reactor is manually controlled, and there is no control rod movement assumed for this case. As a result, the primary side pressure increase is greater than would be the case if the reactor was automatically controlled during the transient.

4.1 VEPCO Analysis

The loss of external load analysis performed by Mirsky[VI.4-1] is summarized in this section. As discussed above, the reactor is assumed to be under manual control at the time of the incident. This transient was evaluated for both beginning-and end-of-core-life conditions.

4.1.1 Description of Model

The RETRAN representation used for this analysis is the Surry one-loop model discussed in Section VI.1-3 and Reference VI.1-3. Some changes made to model the loss-of-load transient included making the pressurizer spray inactive, and deactivating the pressurizer and steam generator relief valves. The trips were also modified for this particular analysis.

Since the secondary system was not modeled, a constant heat transfer coefficient for the secondary side of the steam generator was desired to duplicate the FSAR assumptions. At the time the work was performed, the constant heat transfer coefficient option in RETRAN was not available. Mirsky was able to obtain a coefficient which was approximately constant by performing sensitivity studies based on the quality of the steam generator secondary side fluid.

4.1.2 Discussion of Results

Comparisons of the change in pressurizer pressure are shown in Figures VI.4-1 and VI.4-2. For both conditions (BOL and EOL), the RETRAN and FSAR values agree quite well up to the point of maximum pressure. The effect of the non-constant

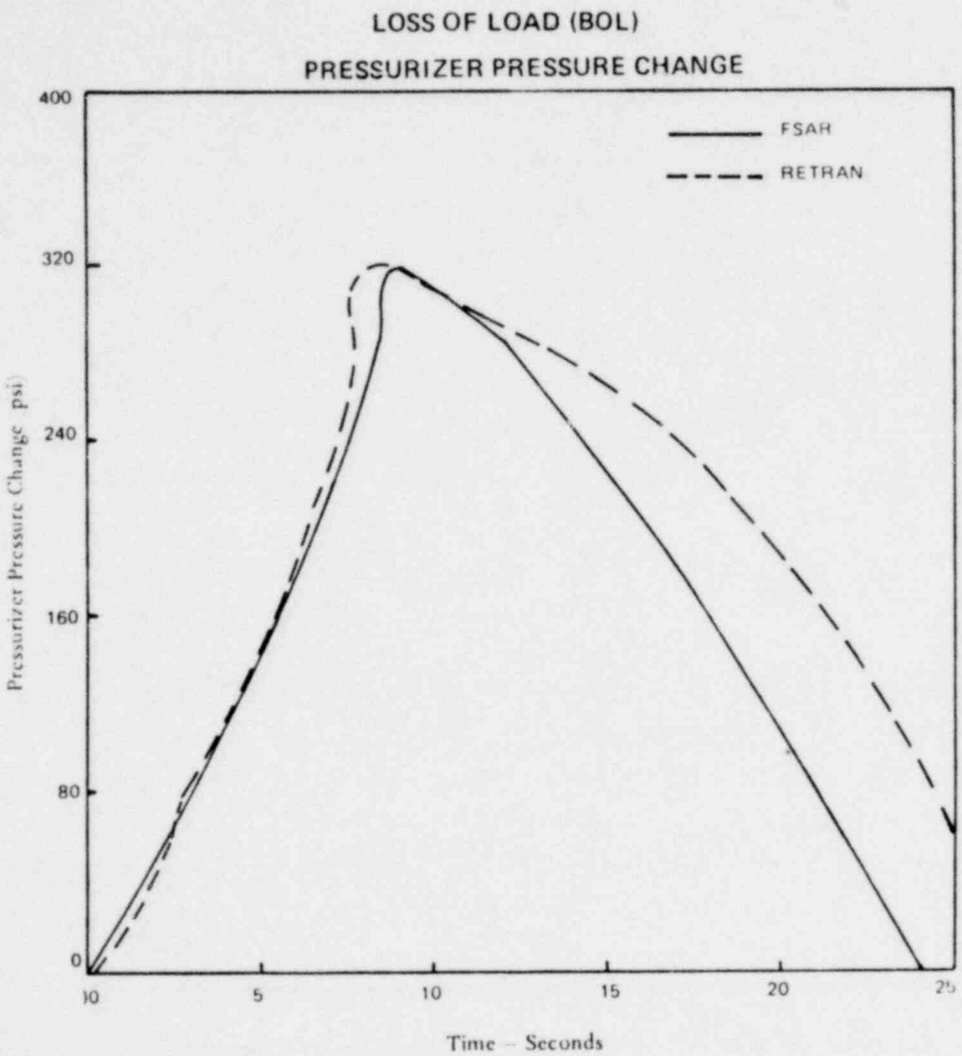

FIGURE VI.4-1 Pressurizer Pressure Change, BOL - Loss of External Load

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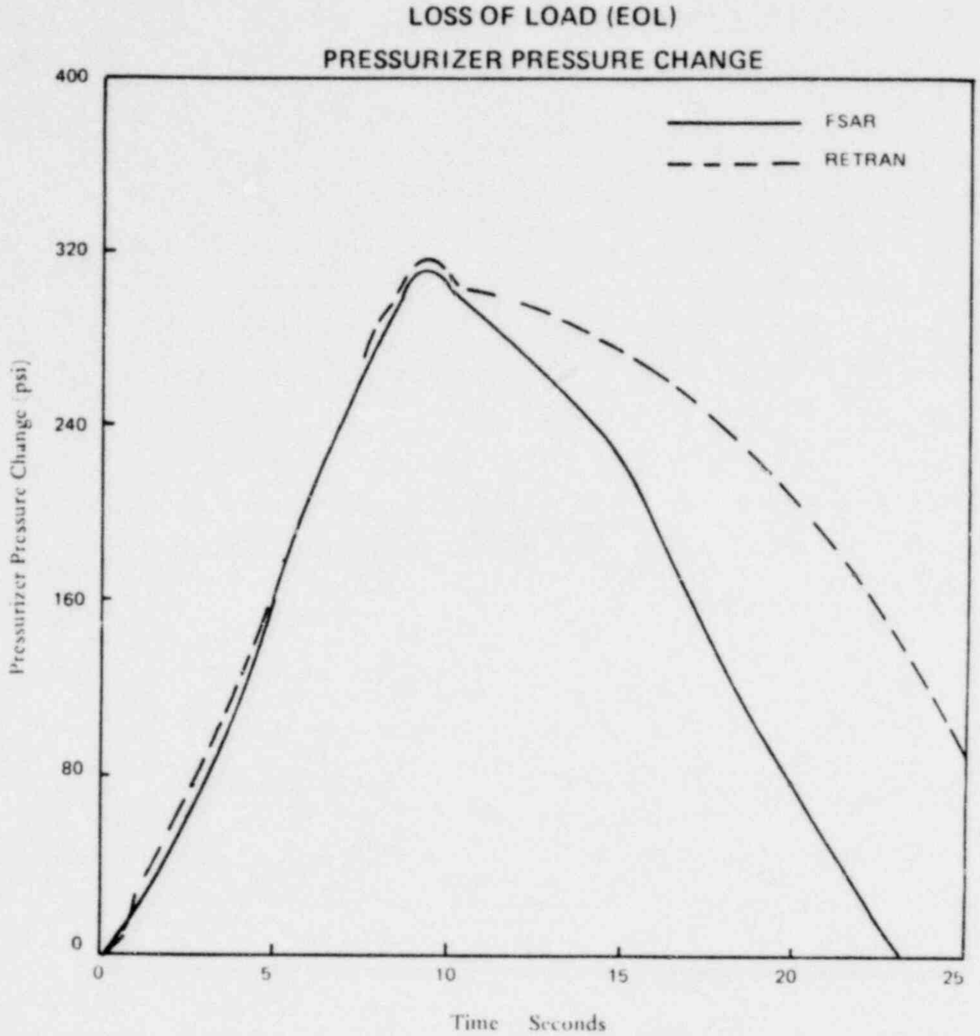


FIGURE VI.4-2 Pressurizer Pressure Change, EOL - Loss of External Load

9 347

heat transfer coefficient in RETRAN is noted after this time, with the pressure computed by RETRAN falling off at a slower rate than for the vendor calculation. The differences in heat transfer coefficient assumptions also produced small discrepancies in the values for coolant inlet temperatures and the water volume in the pressurizer.

The core power response for the two cases is shown in Figures VI.4-3 and VI.4-4. The RETRAN and FSAR results are in good agreement except prior to scram in the EOL case. This difference may be due to different weighting factors for the reactivity coefficients. The DNB ratios calculated by the vendor and RETRAN are shown in Figures VI.4-5 and VI.4-6. Although the trends are similar, the differences between the values are large. Subsequent analysis showed this was due primarily to deficiencies in the RETRAN DNB model and to different DNB models in RETRAN and the vendor calculation. The use of the RETRAN DNB model is discussed in Section IX.3.6.

4.2 Summary of Results

A summary comparison of parameters as calculated by the vendor and with RETRAN is given in Table VI.4-1. The peak pressure values, the main item of interest were in good agreement. Although the constant heat transfer option was not available when this analysis was performed, if further comparisons are made with vendor analysis, this option should be used. In addition, sensitivity studies of some nuclear parameters (delayed neutron fraction, reactivity coefficient values and weighting factors) should be performed.

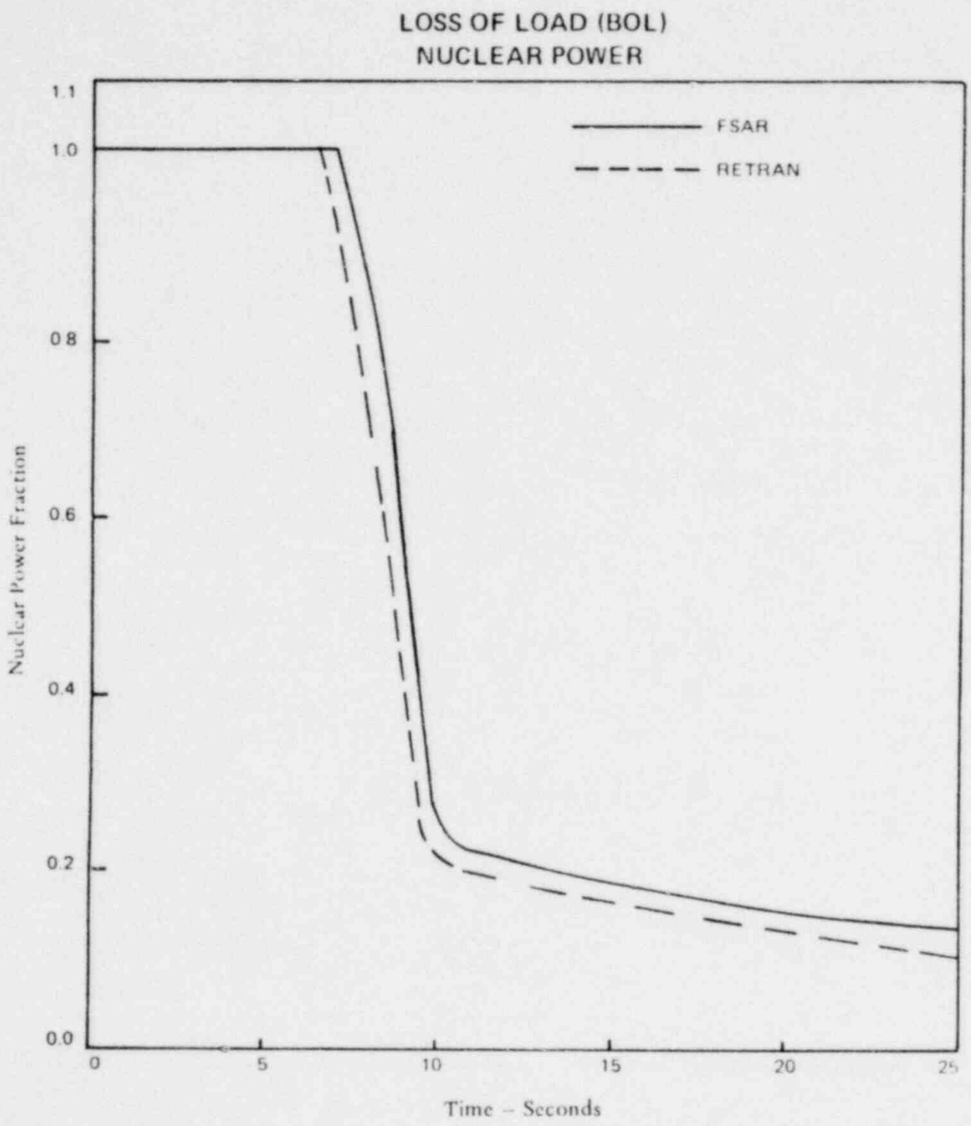


FIGURE VI.4-3 Change in Nuclear Power, BOL - Loss of External Load

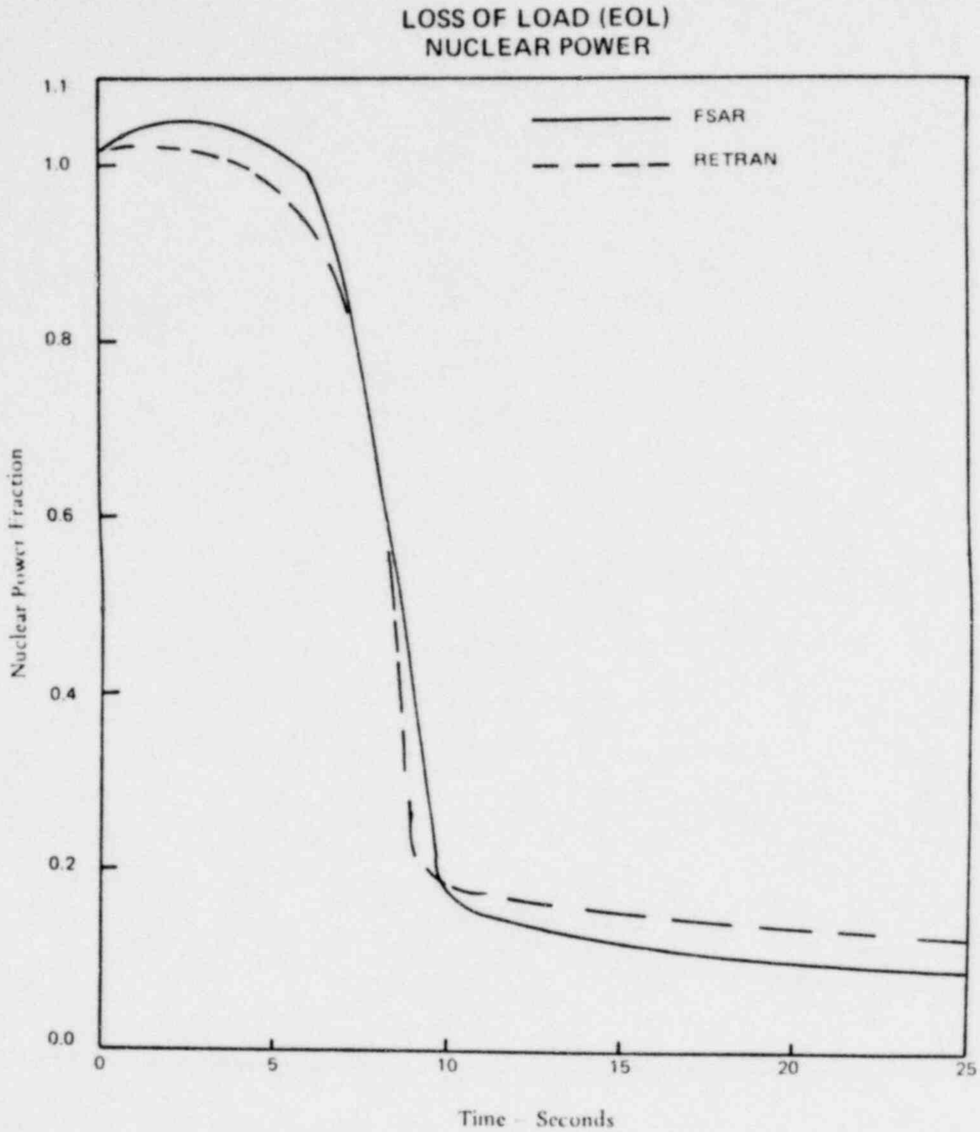


FIGURE VI.4-4 Change in Nuclear Power, EOL - Loss of External Load

1759 350

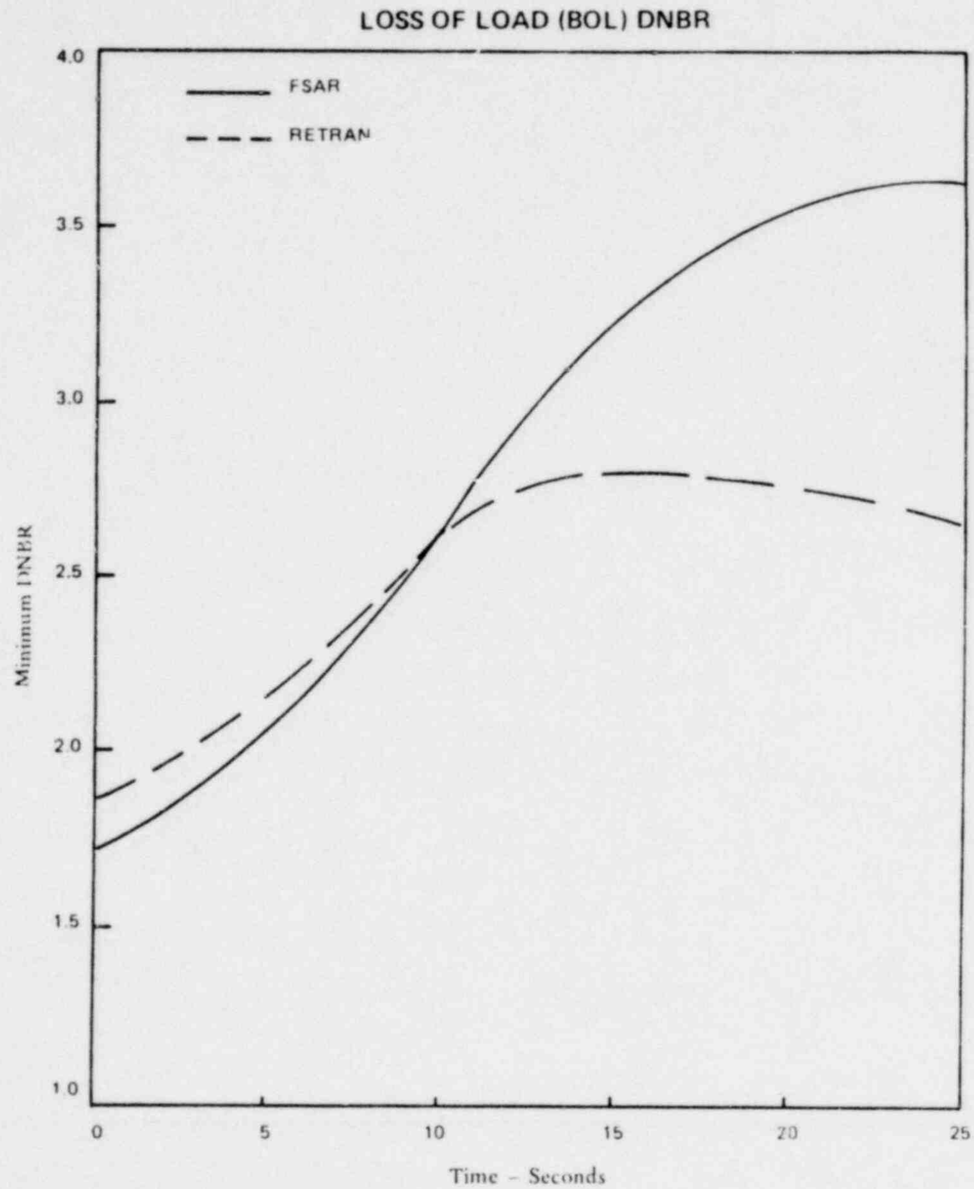


FIGURE VI.4-5 DNB Ratio, BOL - Loss of External Load

1759 351

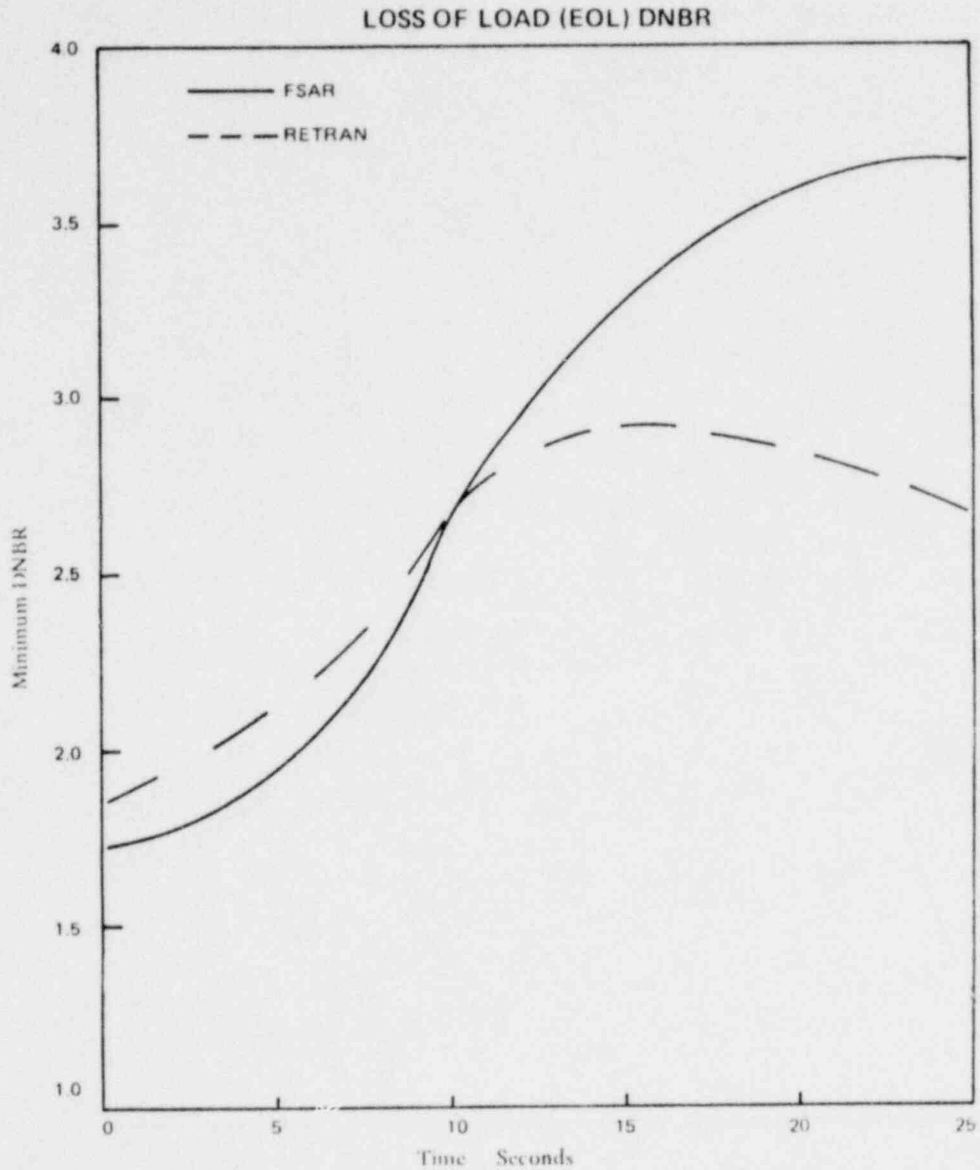


FIGURE VI.4-6 DNB Ratio, EOL - Loss of External Load

79 552

TABLE VI.4-1

LOSS OF LOAD RESULTS COMPARISON

BOL	FSAR	RET11D 5% quality in S/G Secondary Side
Trip Time	6.2 seconds	5.8 seconds
Peak Press. Pressure	2540 psia	2544 psia
Max. Press. Safety Valve Flow Rate, Normalized	56.7%	51.5%
Steamline Safety Valve Opening Time	10 seconds	11.5 seconds
Max. Steamline Safety Valve Flow	67 %	51 %
Time of Max. Normalized Steamline Safety Valve Flow	12 seconds	14.1 seconds
EOL		
Peak Press. Pressure	2534 psia	2538 psia
Max. Press. Safety Valve Flow Rate, Normalized	49.5%	43.8%
Steamline Safety Valve Opening Time	10 seconds	11.5 seconds
Max. Steamline Safety Valve Flow Rate, Normalized	65 %	49 %
Time of Max. Steamline Safety Valve Flow	12 seconds	14.25 seconds

9 353

5.0 STEAM LINE BREAK

A steam line break transient is a Condition IV event, and is typically the most severe transient from a return to power standpoint. The break in the steam line produces an initial increase in steam flow. This results in a reduction of the primary coolant temperature which causes an increase in reactivity. The reactor is eventually shut down via boron injection. This transient has been analyzed by VEPCO[VI.5-1], and is discussed in the following sections.

5.1 VEPCO Analysis

A two-loop model of the Surry reactors was used for this analysis. The transient analyzed is similar to that described above. The transient is initiated with a complete severence of a main steam pipe at the exit of a steam generator. The discharge of secondary system fluid to the containment is limited by:

(1) The presence of a fast-acting check valve in each main steam line which prevents blowdown of the two intact steam generators via the main steam header, and

(2) The critical flow phenomenon, which determines an upper limit for the mass velocity at the break.

The presence of the check valves in the main steam line also serves to create a large pressure differential between the main steam header and the broken steam line, thus generating a signal which activates the safety injection system, after certain time delays. The safety injection system discharges high-concentration borated water to the primary coolant system. The presence of boron with its high thermal neutron absorption cross-section tends to lower the overall reactivity of the system off-setting the reactivity inserted due to cooldown of the primary coolant. A detailed discussion of the plant model and the analysis is given in Reference VI.5-1.

5.1.1 Description of Model

The RETRAN model used for the transient is shown in Figure VI.5-1. It includes two-loop representations, and has 42 volumes, 56 junctions, and 16 conductors.

9 354

RETRAN MAIN STEAM LINE BREAK MODEL

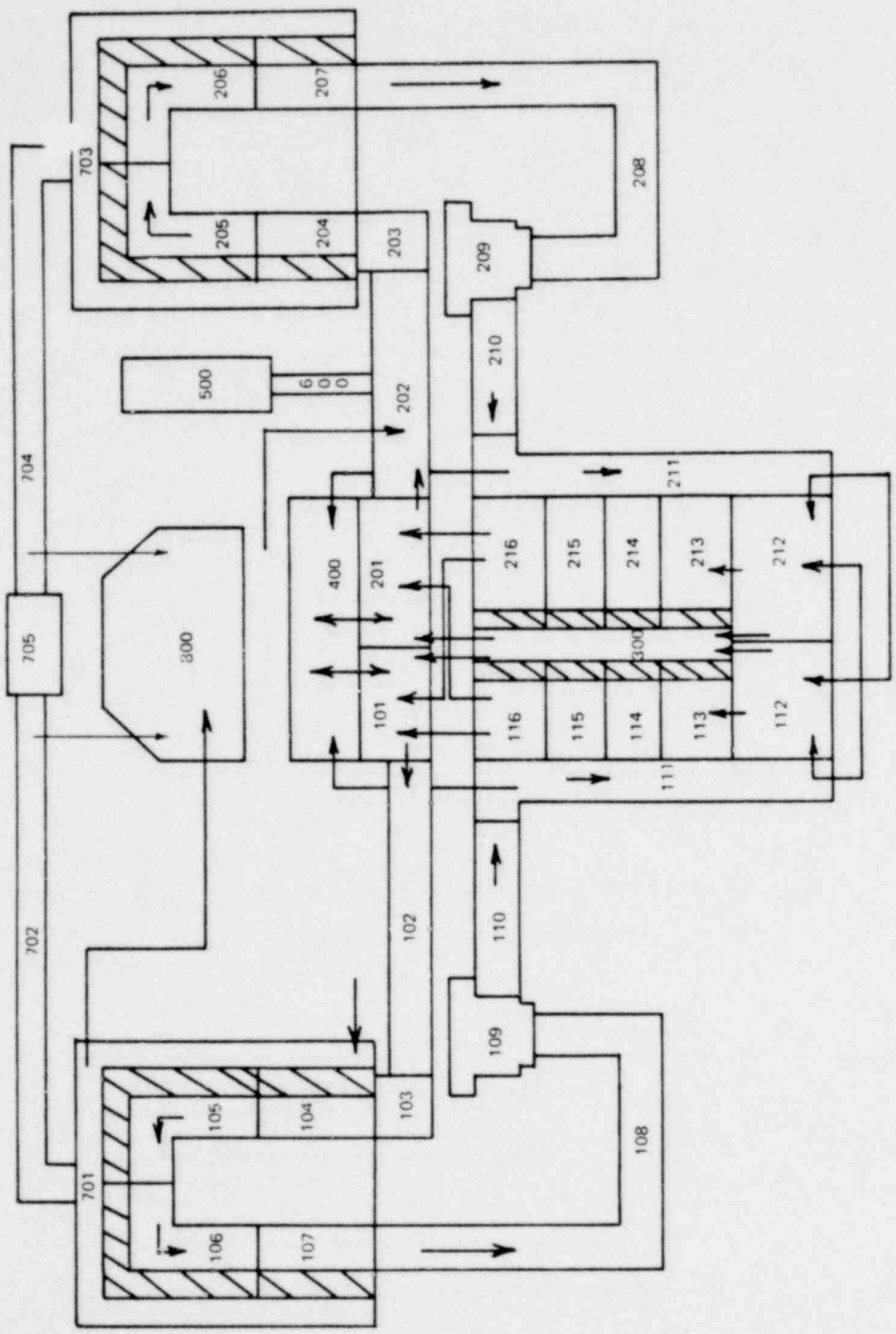


FIGURE VI.5-1 RETRAN Main Steamline Break Model for Surry Reactor

9 355

The core is modeled with two flow paths, with the one nearest the loop where the break occurs representing 1/3 of the core. The vessel upper head region, the pressurizer, surge line and core bypass are each modeled as a single volume. There are two volumes in the lower plenum, with cross flow permitted between those volumes.

Control system models were used to describe the dilution and transport of boron after injection and for controlling the pressurizer heaters. The boric acid is injected into the system via the cold leg piping. Prior to injection, the boron-water mixture from the Refueling Water Storage Tank is injected into the Boron Injection Tank. Since the boron concentrations in these tanks are different, a mixing model is required to determine the concentration entering the reactor. Other items which must be modeled include the boron transport mixing at the point of injection as well as transport through the loop which must be determined to provide boron concentration values. The control model shown in Figure VI.5-2 was used to compute the concentration at the core midplane. Safety injection trips were initiated for the following conditions:

(1) Simultaneous low pressurizer pressure and level

(2) High pressure differential between the main steam header and the single loop steam line

(3) High steam flow at the exit of the double steam generator coincident with either (a) low primary system average temperature, or (b) low steam pressure.

Additional details of the model for this transient are provided in Reference VI.5-1, including a discussion of the steady-state solution.

5.1.2 Results of Analysis

A detailed discussion of the results of this transient calculation and comparisons with FSAR results is given by Smith[VI.5-1]. Values for the break flow rate are shown in Figure VI.5-3, with good agreement until about 90 seconds when RETRAN predictions are above the FSAR values.

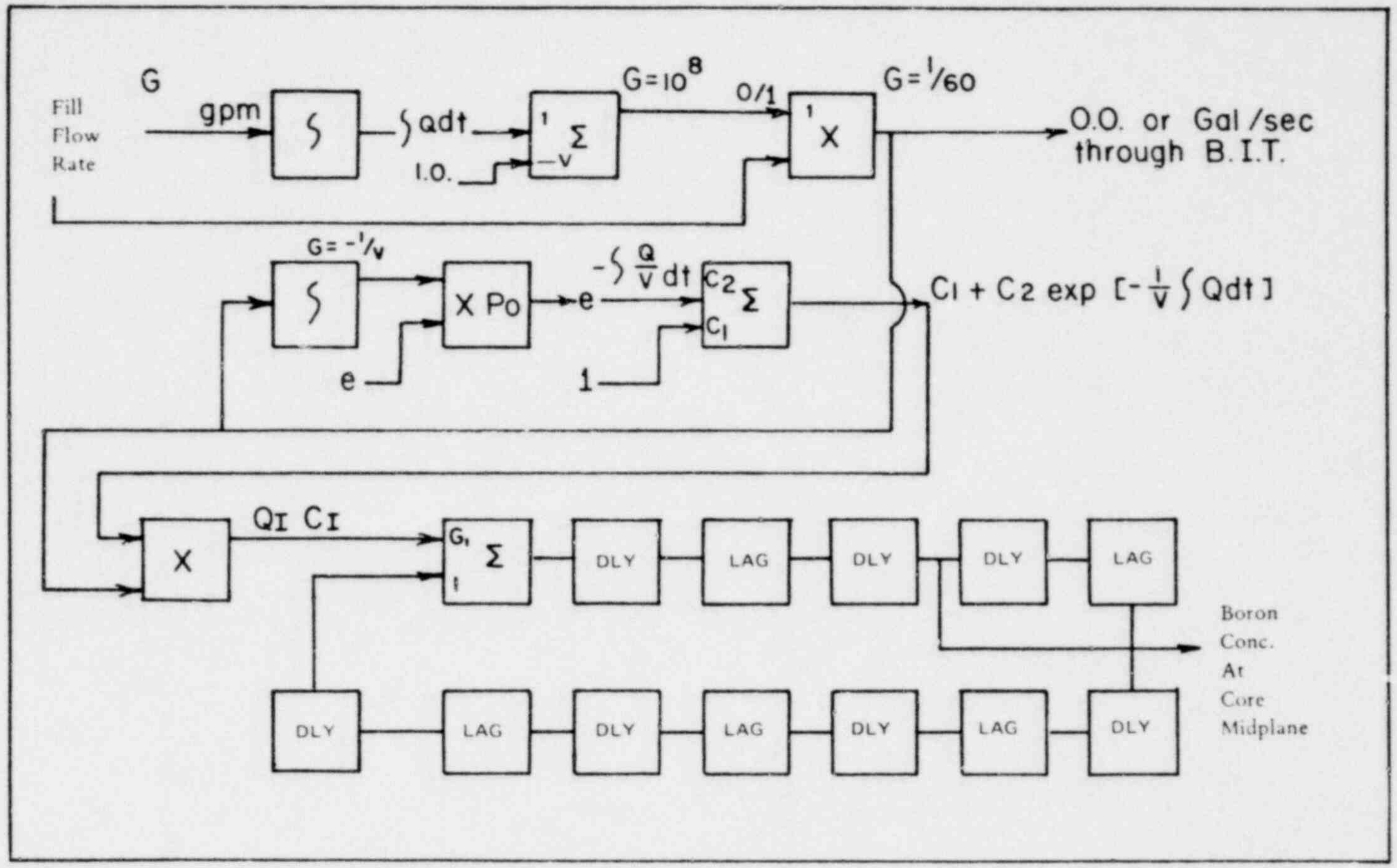


FIGURE VI.5-2 RETRAN Model for Boron Transport

9 357

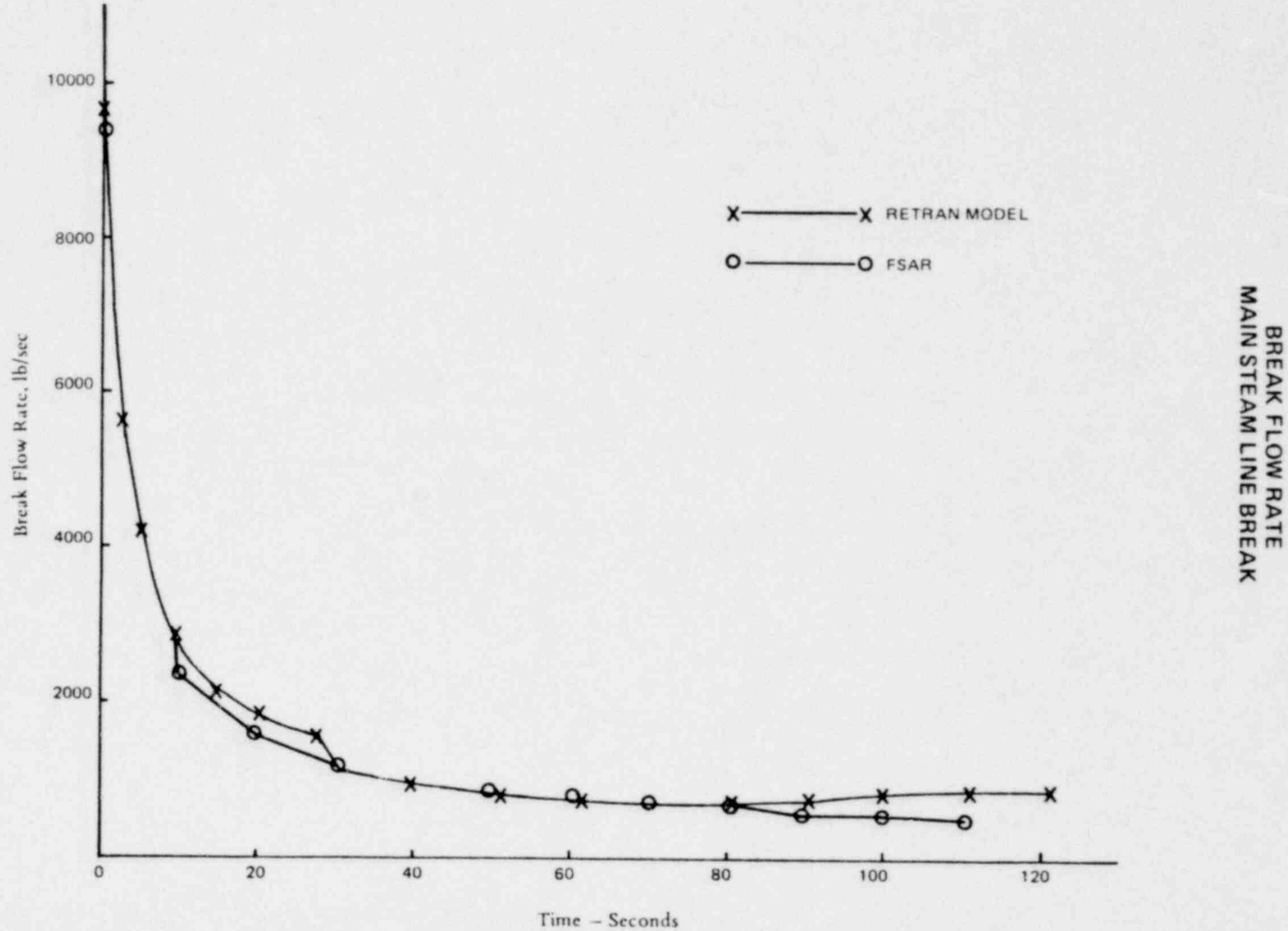


FIGURE VI.5-3 Comparison of RETRAN and FSAR Break Flow Rates - SLB

9 558

The steam-generator heat transfer coefficients are compared in Figure VI.5-4. Whereas the FSAR assumed a constant value, this option was not functioning in RETRAN when the analysis was performed. The single volume steam generator secondary conditions were such that low heat transfer was computed to occur throughout the secondary side very early in the transient. The use of additional nodes or the constant heat transfer coefficient (which is now in the code) will remedy this situation.

The pressurizer pressure and water level are shown in Figure VI.5-5, with RETRAN predicting a faster decline in pressure after 30 seconds than does the FSAR. Values of reactivity from the two calculations are given in Figure VI.5-6, with reasonable overall agreement between the two codes. The oscillations in the RETRAN values between 15 and 50 seconds lead to large power values, resulting from an assumption of a constant value for the moderator temperature coefficient.

Smith[VI.5-1] discussed many other results in addition to those summarized above. He also performed some preliminary calculations with a revised RETRAN model. One example of this analysis is the reactivity shown in Figure VI.5-7. The moderator temperature coefficient in this case was assumed to be temperature dependent, and was computed from a table of moderator feedback versus moderator density. It is noted that a significant change in the RETRAN results occurred due to this modification.

5.2 Summary of Results

The RETRAN and vendor calculations were in reasonable agreement for break flow and pressurizer pressure response for the base case model, however, the reactivity responses were quite different. Subsequent model changes lead to improved agreement for some parameters, but affected others at the same time. The results of this particular analysis demonstrate the need to have additional runs performed with other modeling schemes to better provide qualification of RETRAN for steam line break transients.

9 359

STEAM GENERATOR HEAT TRANSFER COEFFICIENT
MAIN STEAM LINE BREAK

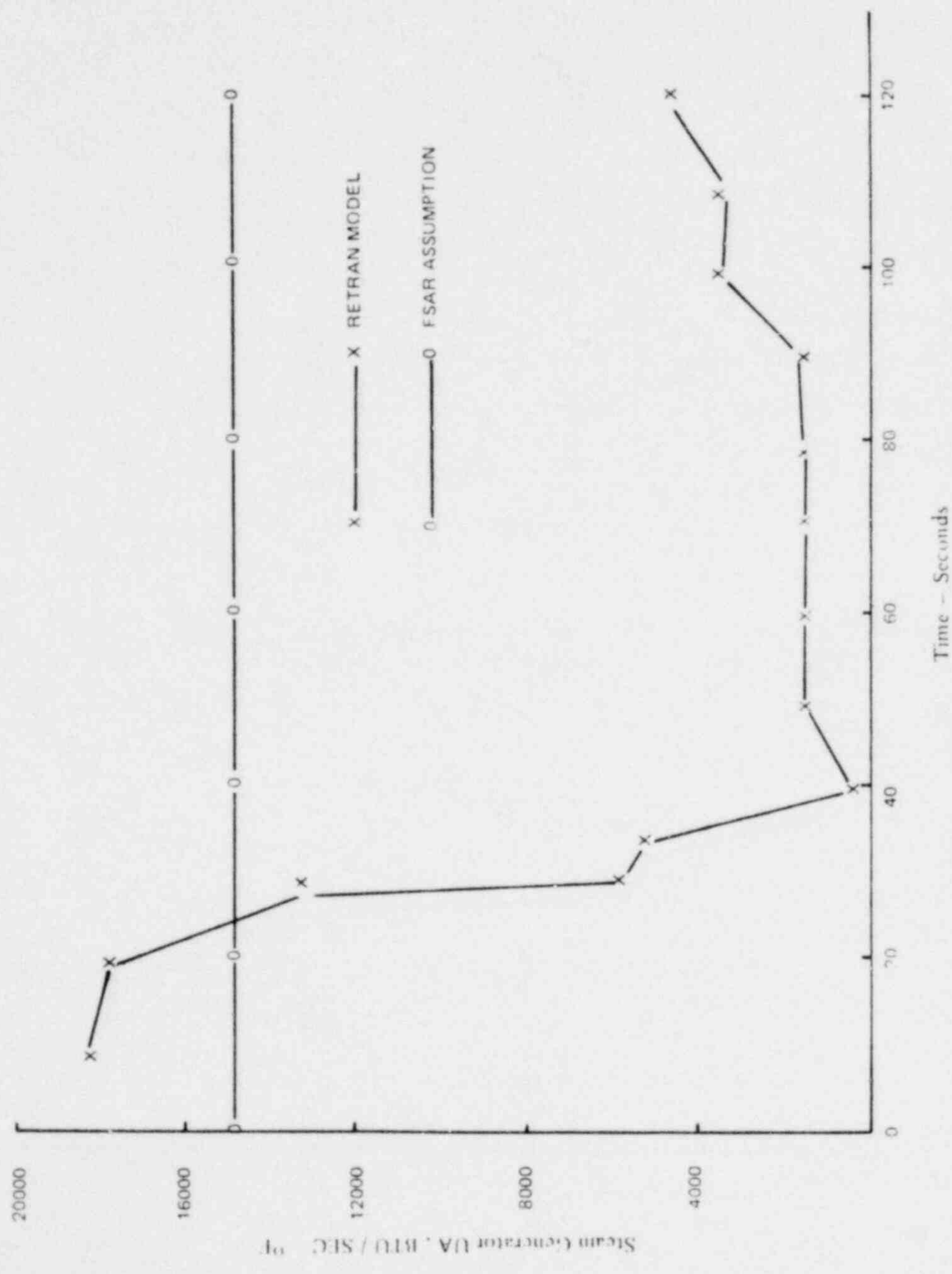
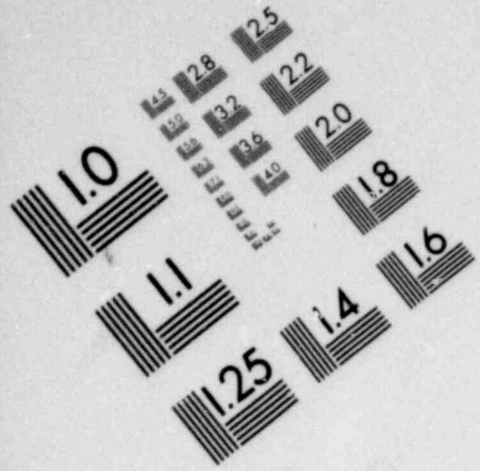


FIGURE VI.5-4 Steam Generator Heat Transfer Coefficients - SLB

9 360



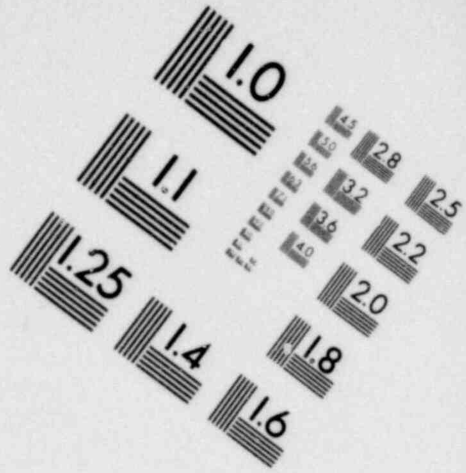
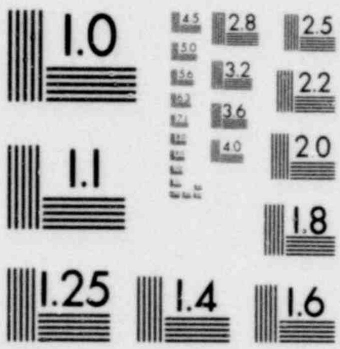


IMAGE EVALUATION TEST TARGET (MT-3)



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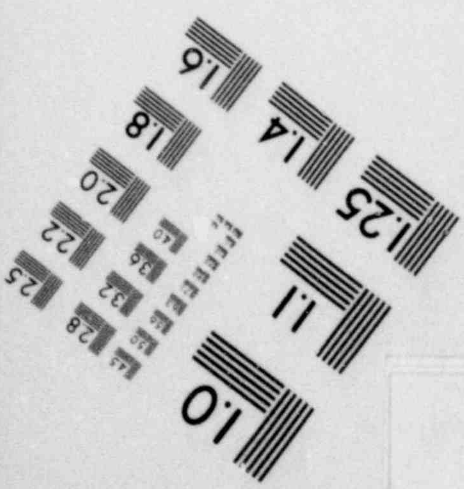
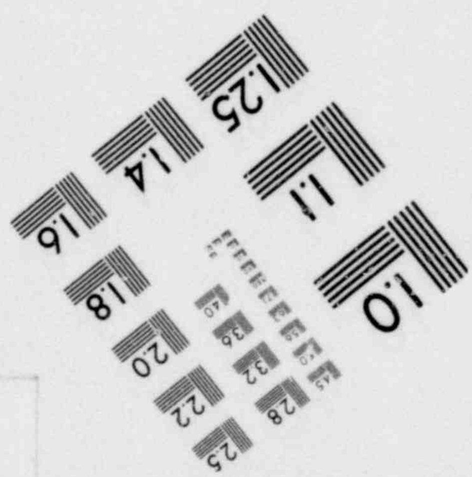
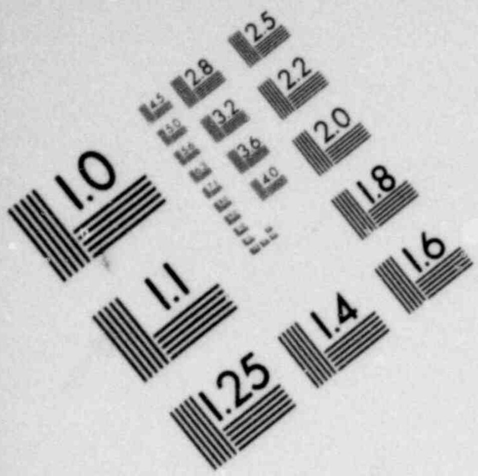






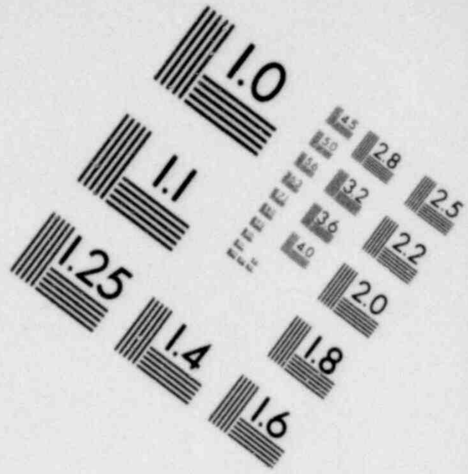
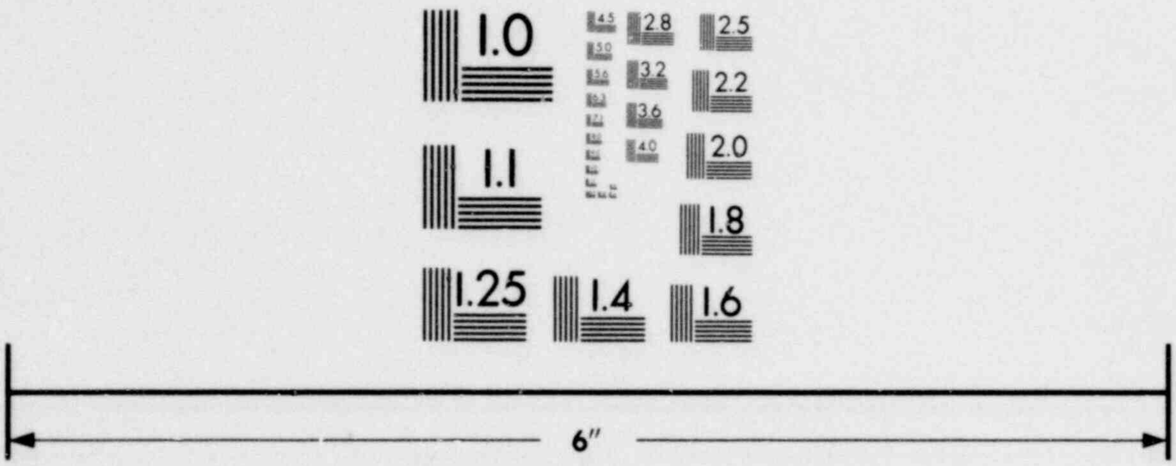
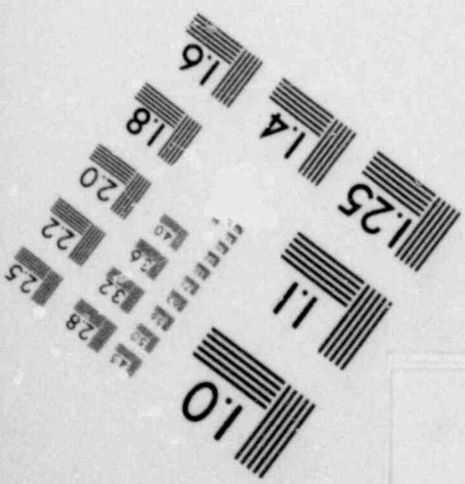


IMAGE EVALUATION TEST TARGET (MT-3)





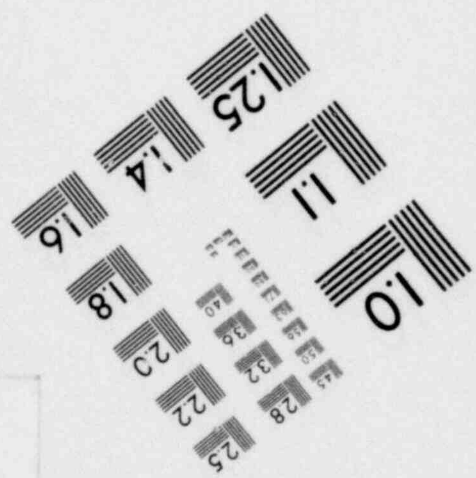


IMAGE EVALUATION TEST TARGET (MT-3)

6"

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

PRESSURIZER PRESSURE RESPONSE
MAIN STEAMLINE BREAK

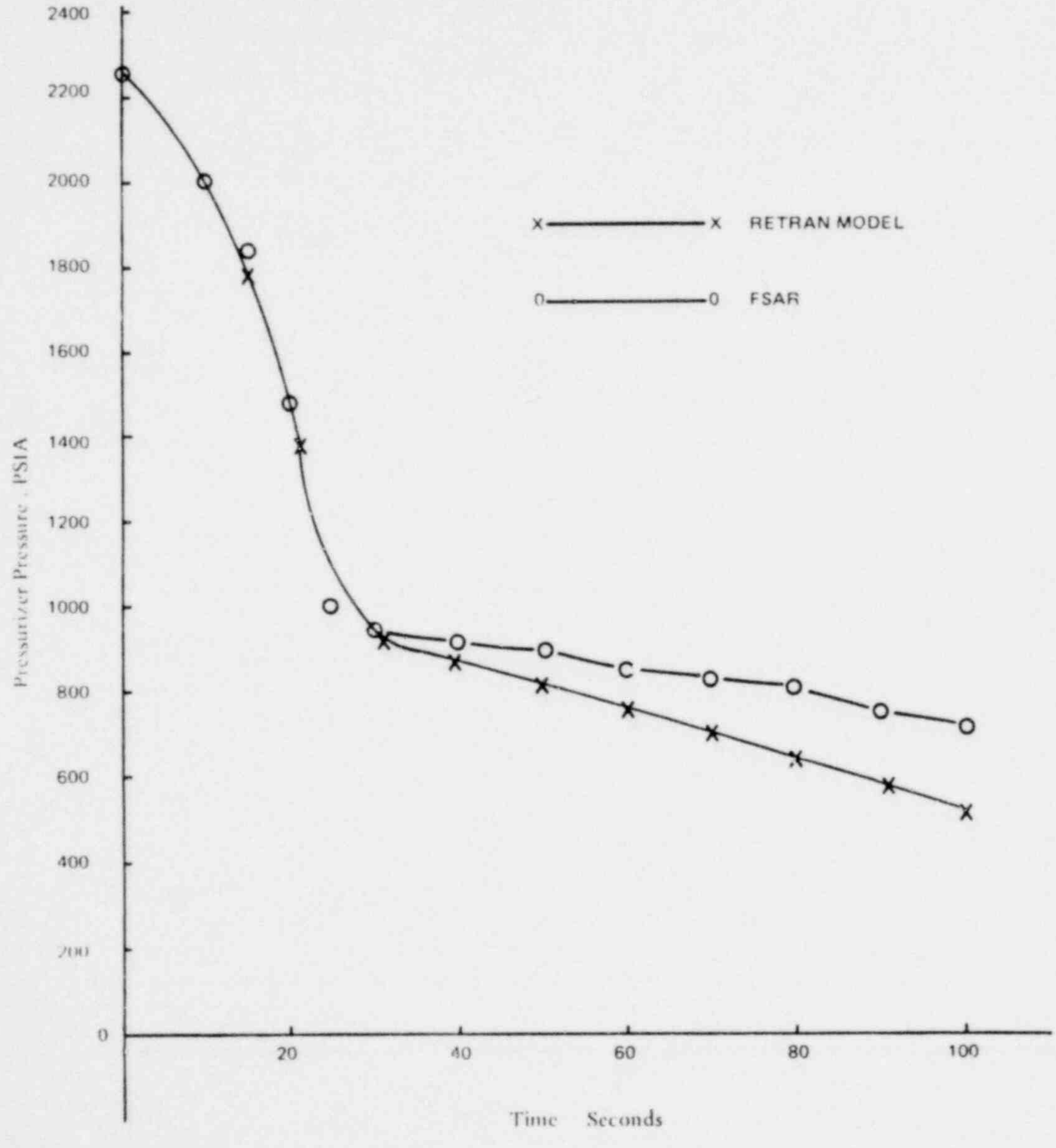
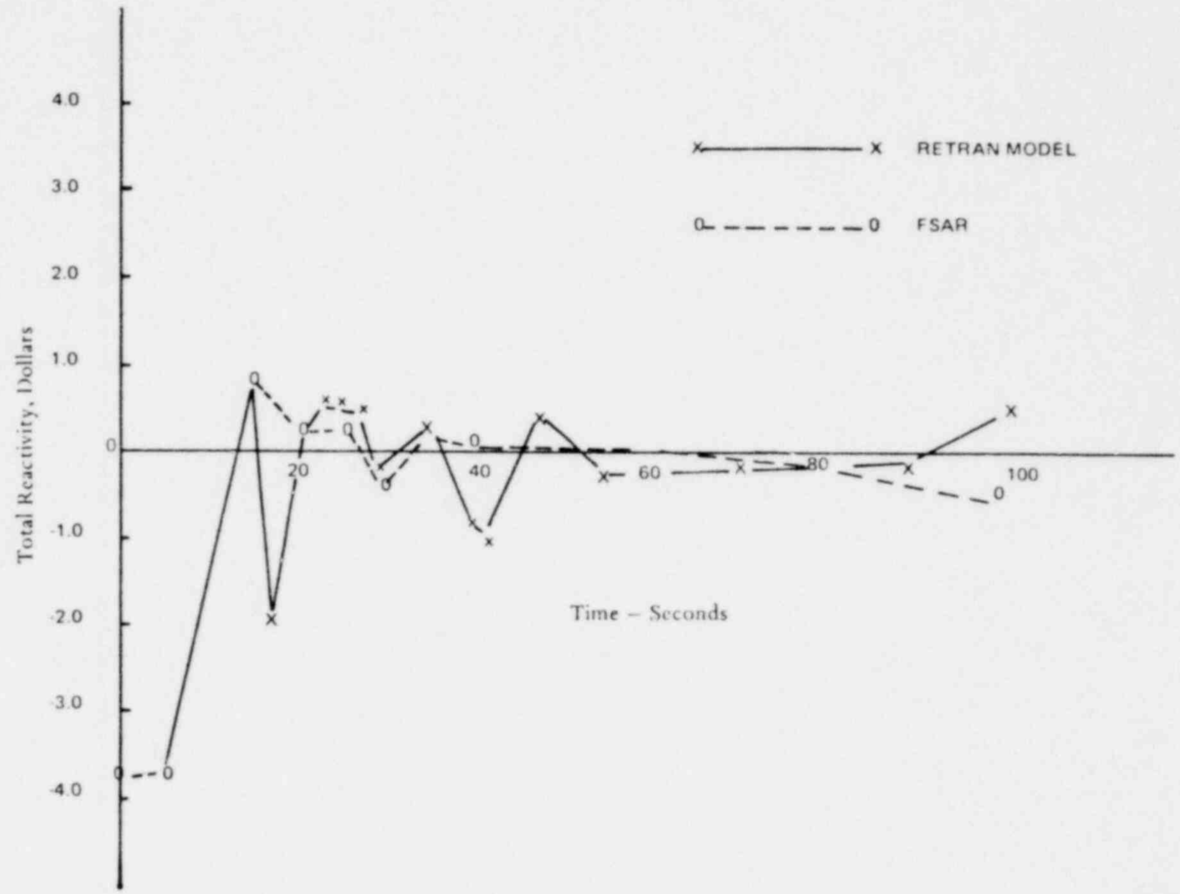


FIGURE VI.5-5 Pressurizer Pressure - SLB

1710 001



TOTAL REACTIVITY RESPONSE
MAIN STEAM LINE BREAK

FIGURE VI.5-6 RETRAN and FSAR Values for Total Reactivity - SLB

TOTAL REACTIVITY RESPONSE
REVISED RETRAN MODEL

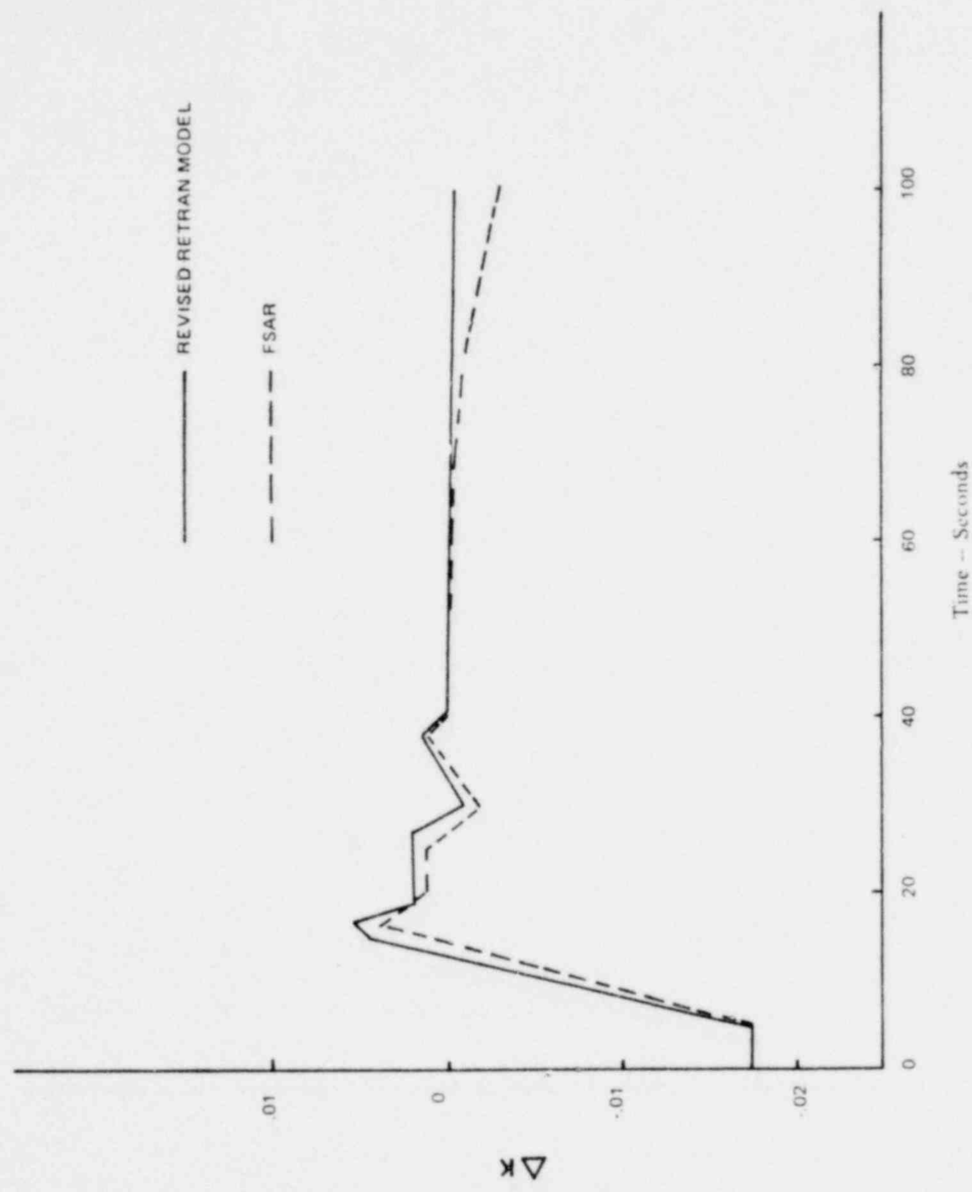


FIGURE VI.5-7 Total Reactivity Response for Revised Model - SLB

1740 003

6.0 OTHER ANALYSIS

In addition to the transients presented in the preceding sections, reports on two other PWR analyses were submitted by Working Group members. One of the analyses, the Three Mile Island Cooldown[VI.6-1] was performed for a transient which occurred during startup testing. As such, data are available and this analysis is a very important part of the RETRAN qualification effort. The second report[VI.6-2] documents the comparisons between a hand initialization and the steady-state solution achieved with the self-initialization option. This exercise was undertaken primarily to gain an understanding of RETRAN and the plant model, and is a valuable exercise for new users of the code.

6.1 Three Mile Island Cooldown

The description of the incident as described in Reference VI.6-1 is given below.

> "During startup testing, TMI Unit 2 experienced a spurious reactor trip caused by a noise spike in the nuclear instrumentation. This resulted in a Turbine Generator trip and subsequent steam line pressurization. One "A" and four "B" steam relief valves opened. Some relief valves failed to reset at their normal reset pressures and this resulted in a rapid pressure decrease in both steam generators.
>
> The rapid depressurization of the steam generators caused a cooldown of the reactor coolant system from 583°F to about 464°F in three minutes. The RCS cooldown caused the pressurizer level to drop below the indicated level range in approximately one minute after trip.
>
> The transient occurred during startup testing, and three of the four reactor coolant pumps and one main feedwater pump were in operation. The plant was stabilizing at about 30% power when the trip occurred.
>
> The primary source of information relating to plant behavior during this transient is the Reactimeter Log. The reactimeter provides three-second data increments for the first three minutes after trip."

6.1.1 Description of Model

A detailed description of the TMI-2 model[Figure VI.6-1] is given in Reference VI.6-1. This plant has once-through steam generators which were modeled with 12 primary volumes, 12 secondary volumes, and 12 conductors. The steam generator feedwater was modeled with a fill since data were available. The non-equilibrium pressurizer model was used; however, the sprays and relief and safety valves were not required for this transient and hence were not included.

The upper region of the reactor vessel was modeled with 4 volumes, one each representing the upper plenum and outlet plenum and two for the upper head. This was found to be necessary configuration during the investigation.

Additional details of the model are given in Reference VI.6-1. Included in the report is a lengthy discussion relative to the evolution of the model. As might be expected, the model was changed during the transient to reflect knowledge gained by performing analyses. During the evaluation of this transient, the RETRAN analyses indicated a possible malfunctioning relief valve which was verified in subsequent testing. The discussion in the referenced document, both of the transient calculations and the self-initialization, provides these details.

6.1.2 Discussion of Results

Comparisons of the RETRAN calculations with the plant data are shown in Figures VI.6-2 to VI.6-10. The apparent disagreement in the temperature values (Figures VI.6-4) after 120 seconds has been attributed to incorrect data. When these calculations were performed, there was a code error in the pressurizer model, which is responsible for the discrepancies shown in Figure VI.6-7 and Figure VI.6-8 during refilling. From an overall viewpoint, the code computed results are in very good agreement with the plant data. The results and knowledge gained while evaluating this incident provide a strong contribution to the RETRAN qualification effort.

6.2 Trojan Hand Initialization

In an effort to gain a better understanding of RETRAN, hand-calculations were performed to obtain a steady-state solution for a one-loop PWR. The values so obtained were compared to the RETRAN steady-state solution by running a null transient.

1720 005

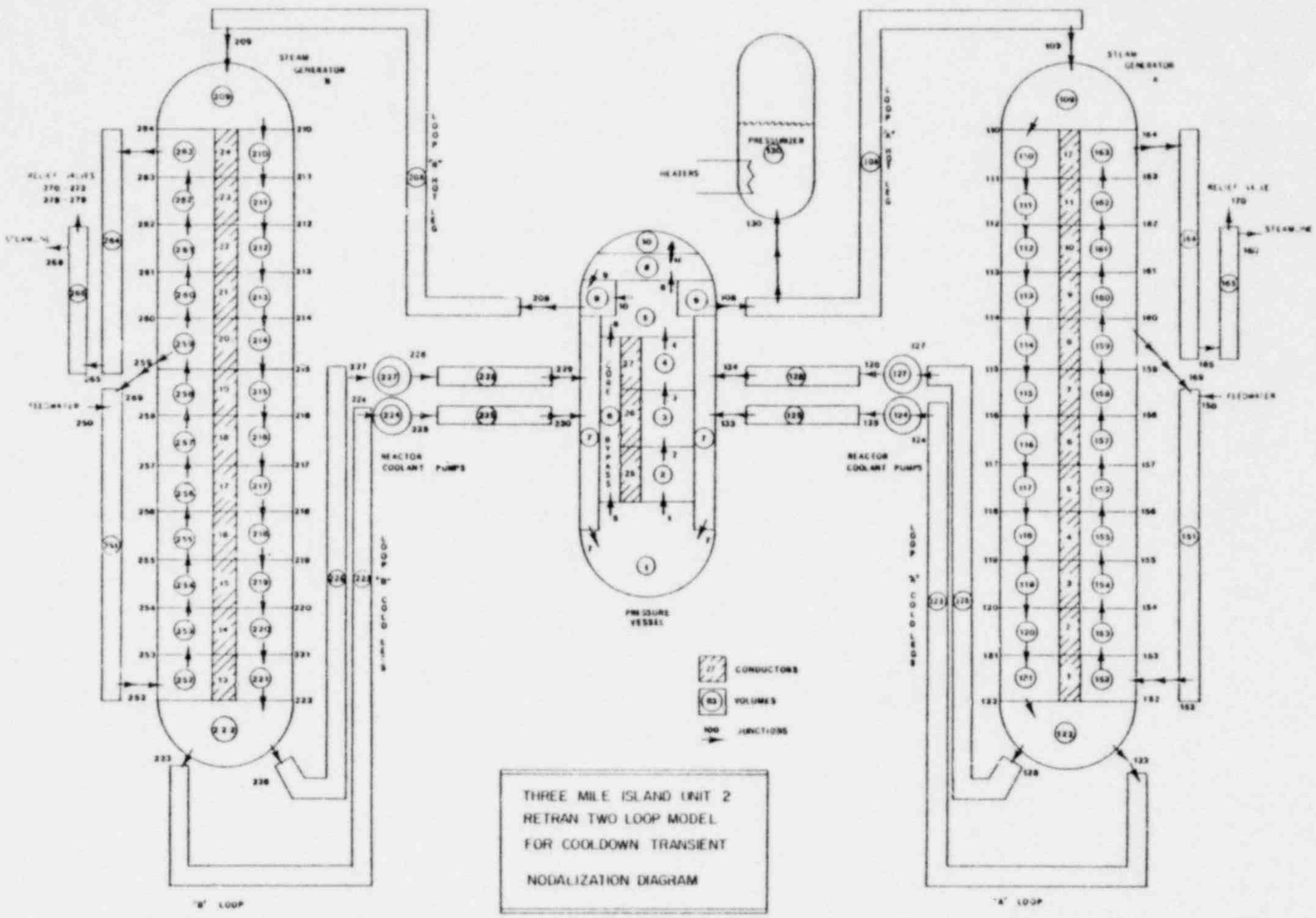


FIGURE VI.6-1 RETRAN Model for TMI-2 Cooldown Transient.

1760 006

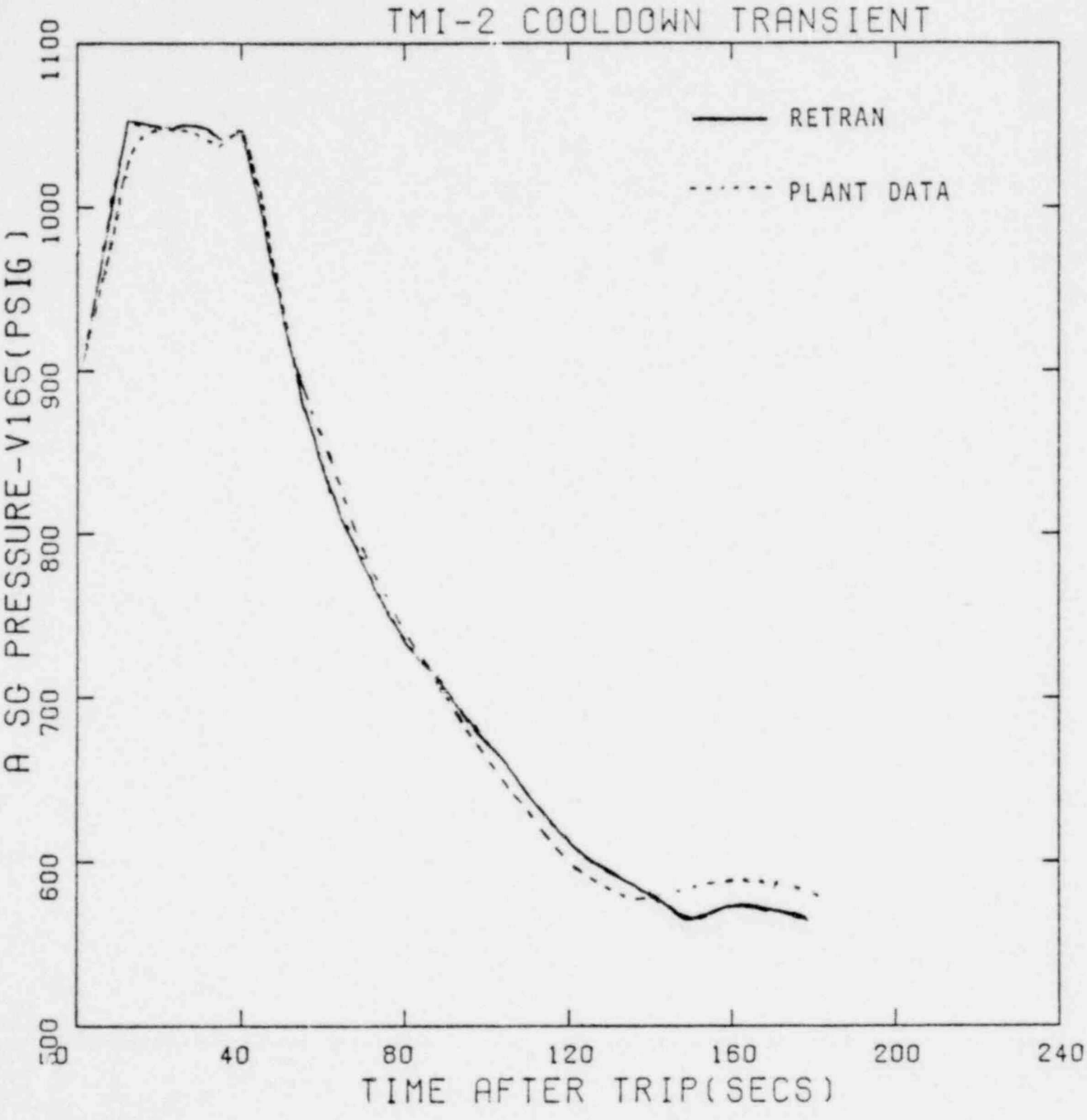


FIGURE VI.6-2 RETRAN Values and Plant Data for Pressure in Steam Generator A

1760 007

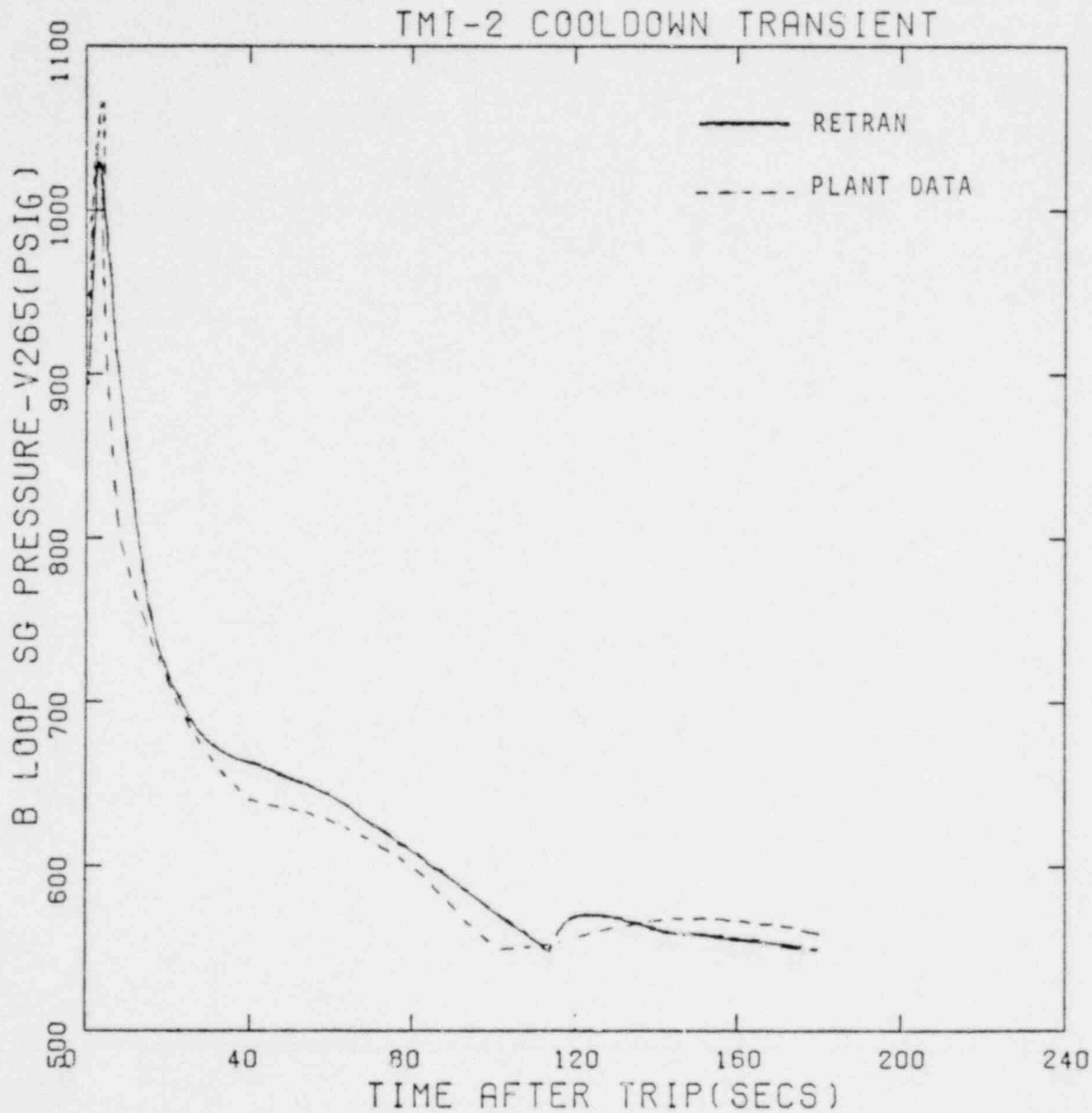


FIGURE VI.6-3 RETRAN Values and Plant Data for Steam Generator B Pressure.

1770 008

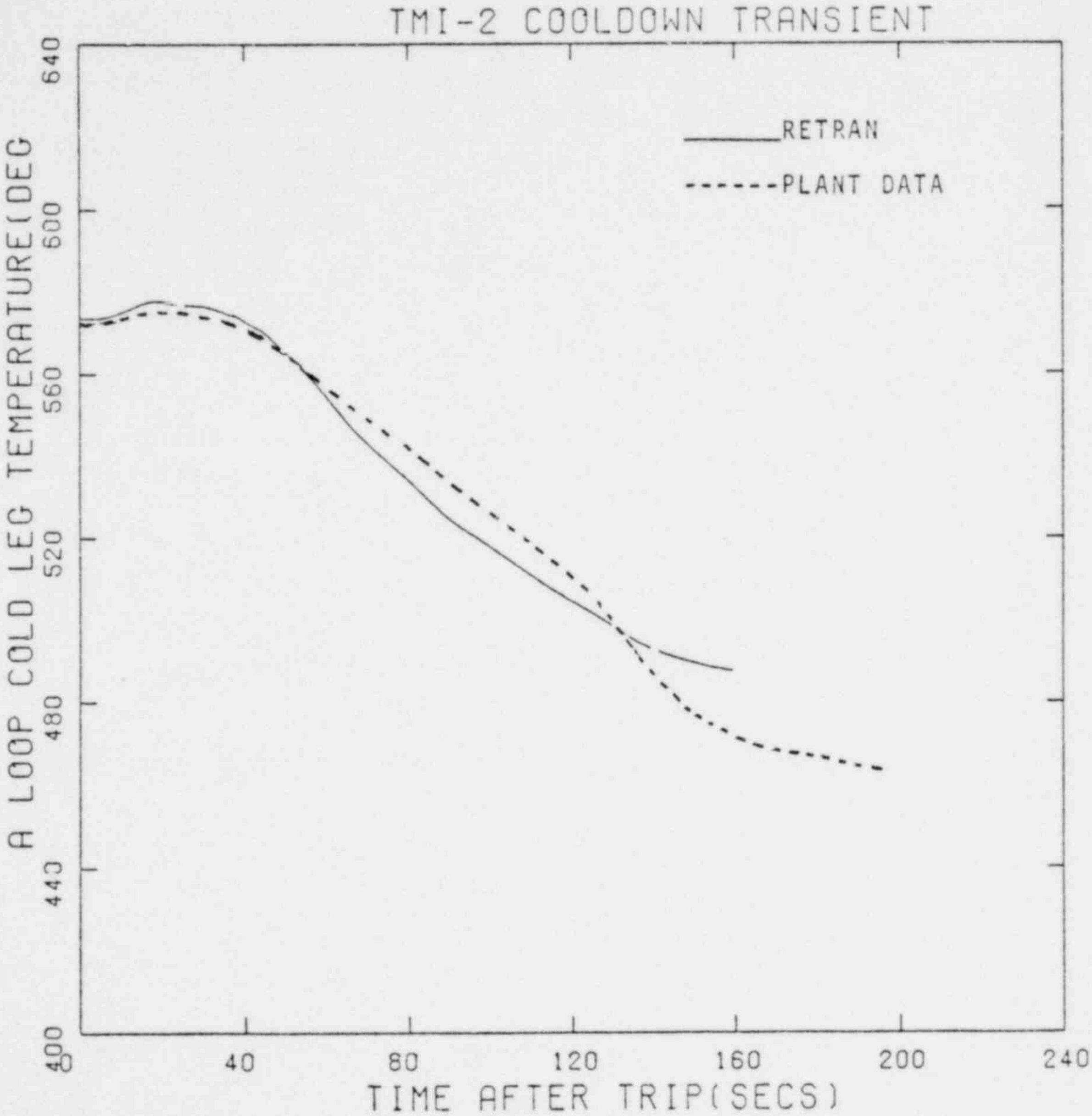


FIGURE VI.6-4 A Loop Cold Leg Temperatures; TMI-2 Cooldown.

1770 009

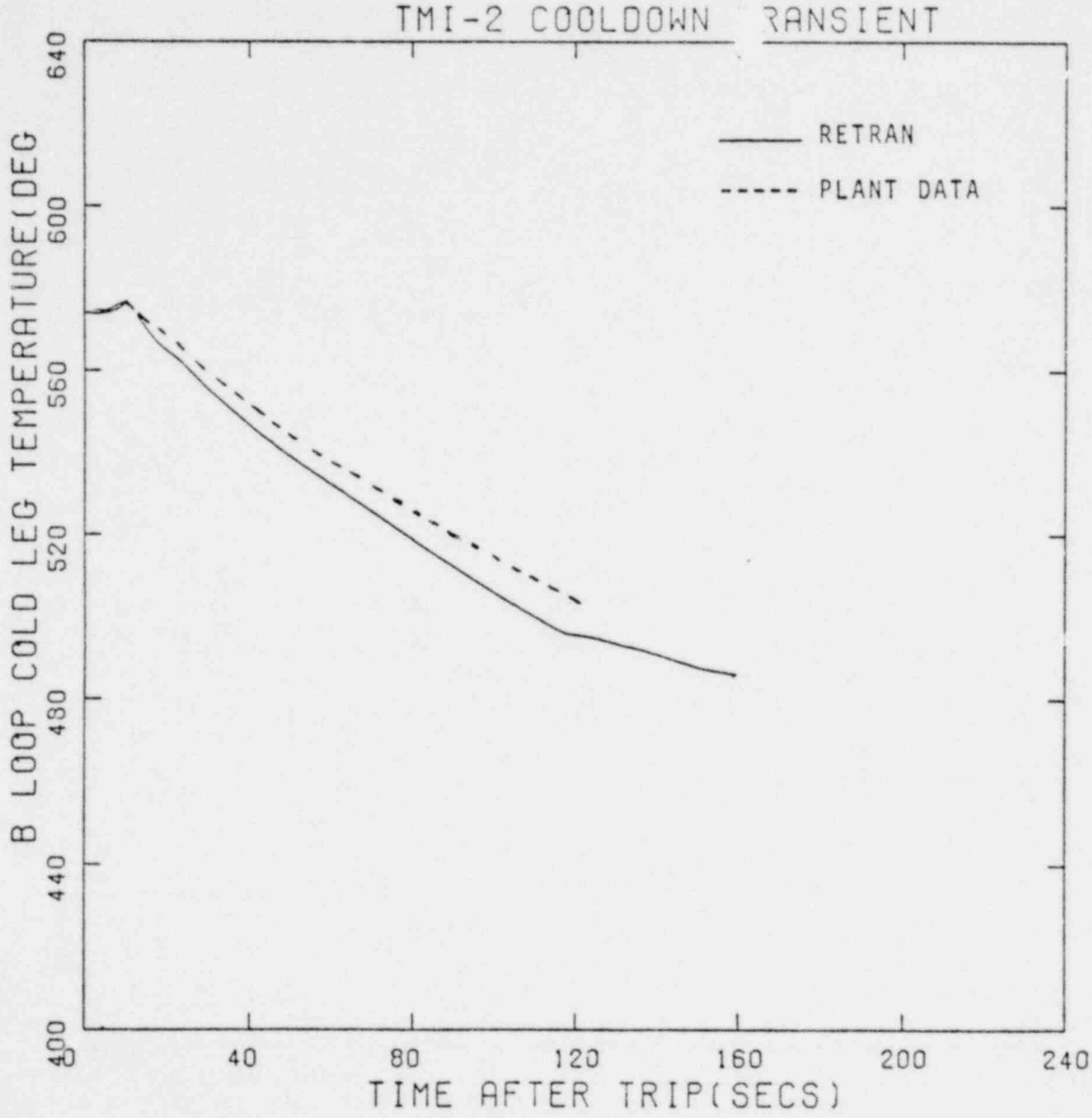


FIGURE VI.6-5 B Loop Cold Leg Temperatures; TMI-2 Cooldown.

1?/0 010

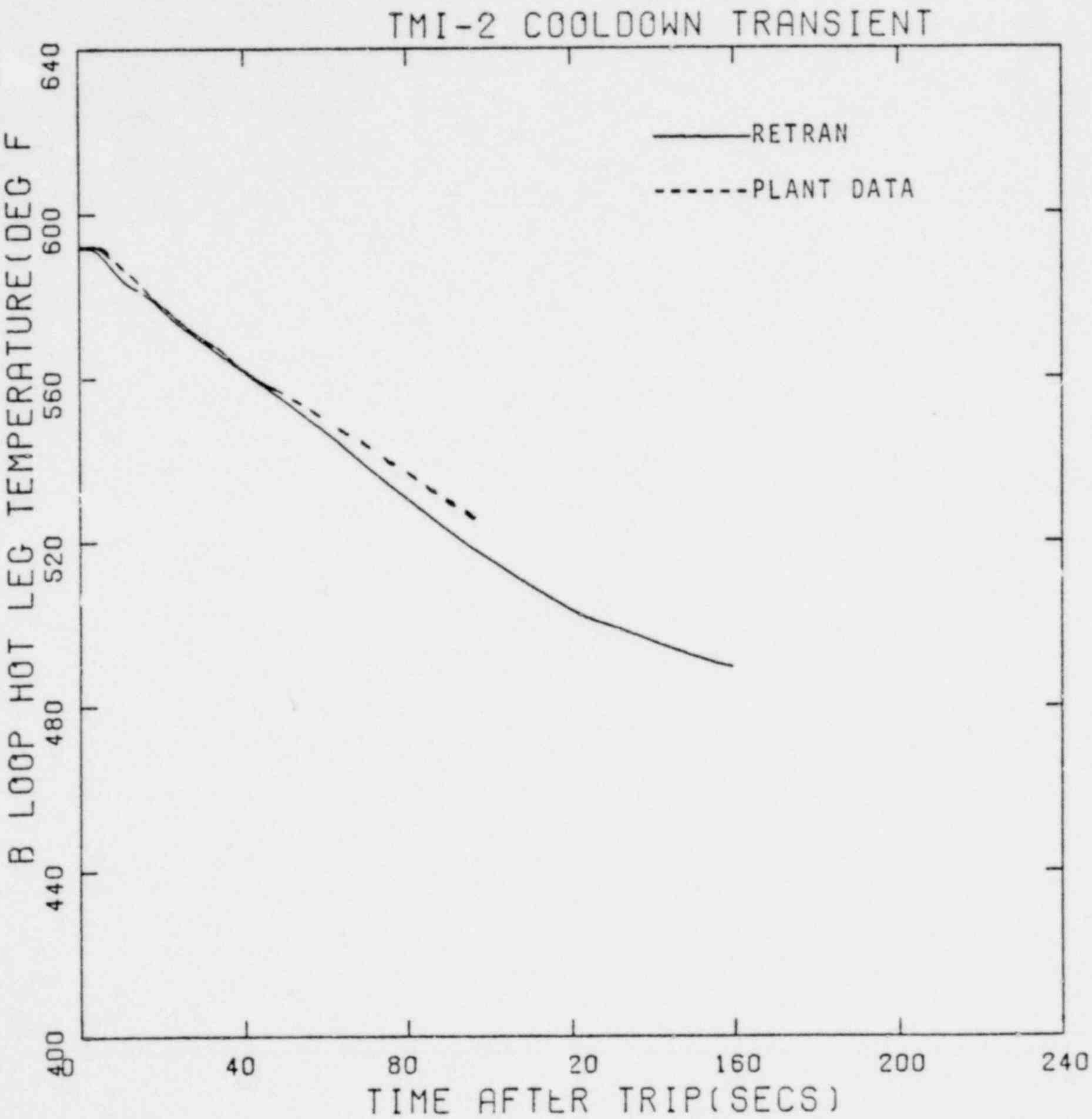


FIGURE VI.6-6 B Loop Hot Leg Temperatures; TMI-2 Cooldown.

1770 011

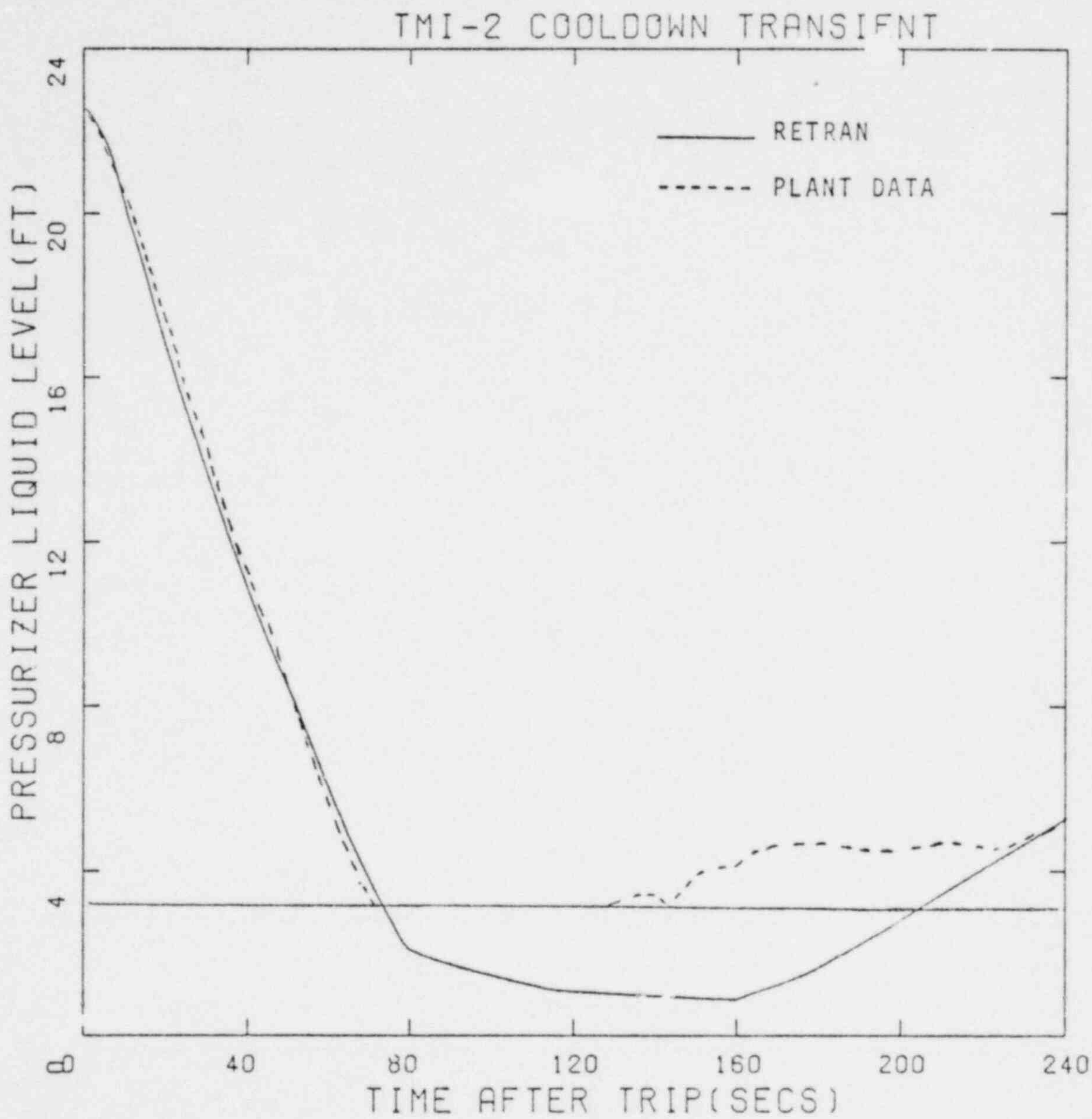


FIGURE VI.6-7 Pressurizer Liquid Level, TMI-2 Cooldown.

1770 012

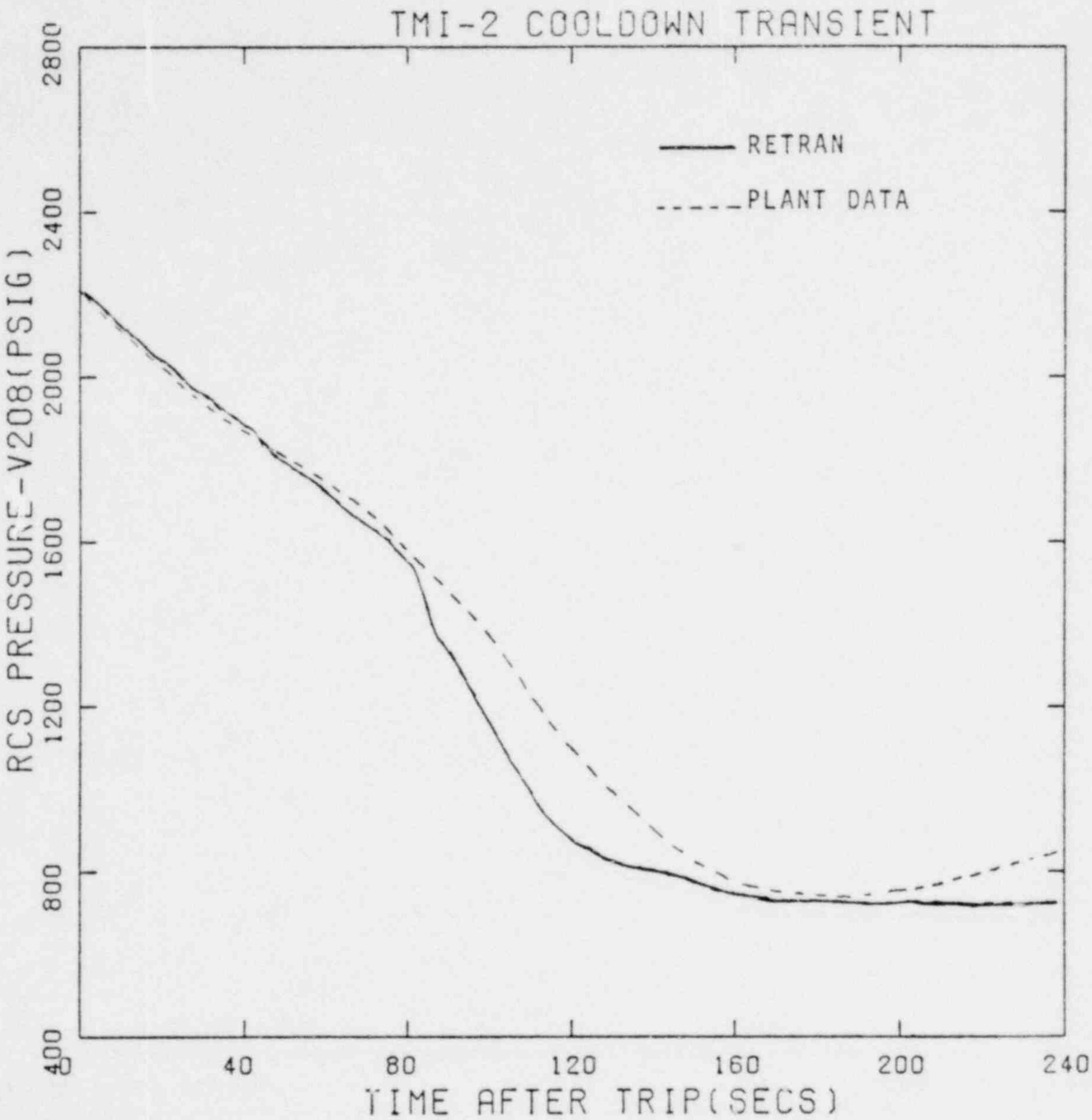


FIGURE VI.6-8 RCS Pressure, TMI-2 Cooldown.

1770 013

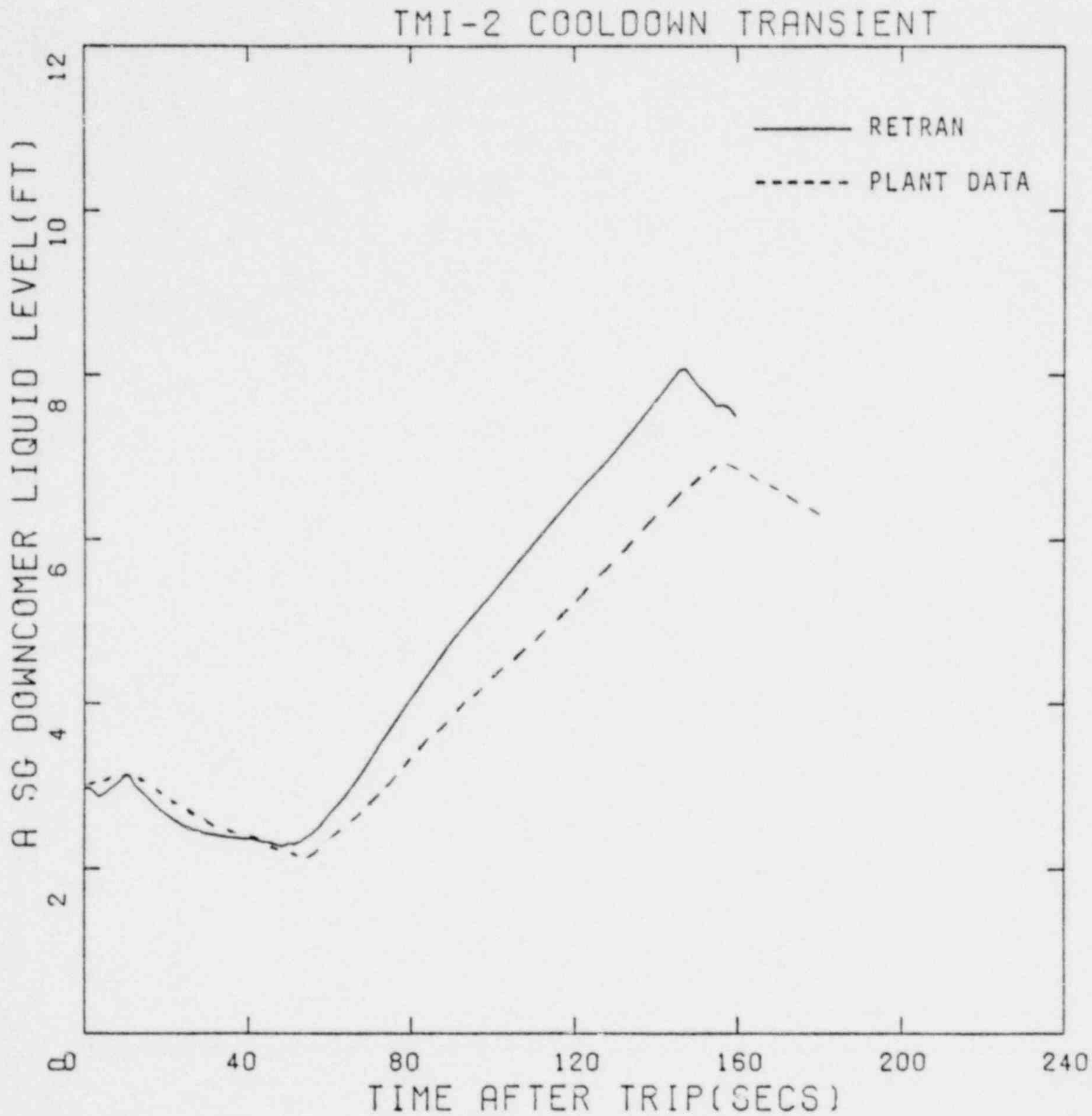


FIGURE VI.6-9 Liquid Level in A Steam Generator Downcomer.

1770 014

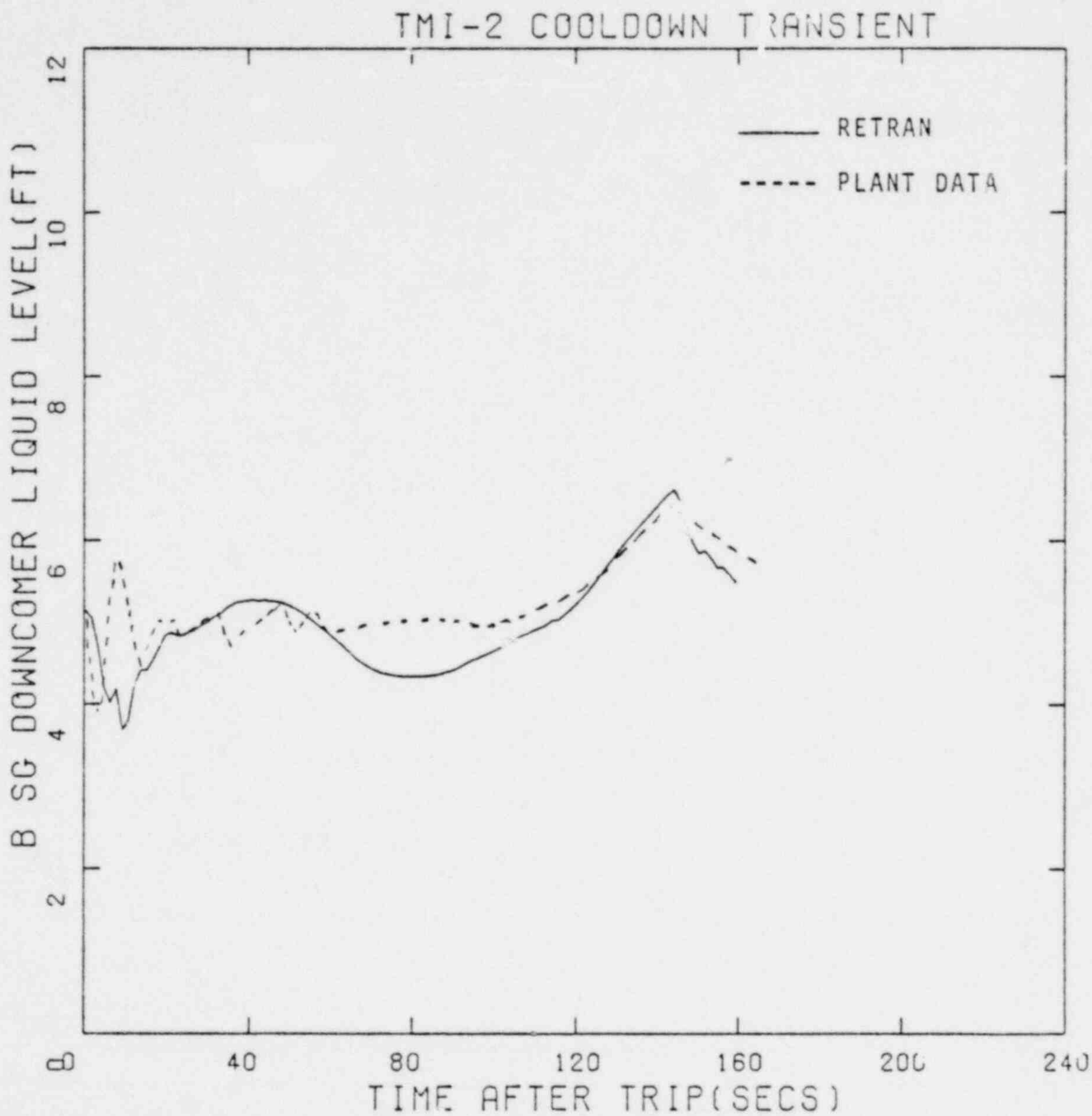


FIGURE VI.6-10 Liquid Level in B Stream Generator Downcomer.

170 015

6.2.1 Description of Model

The one-loop representation of the Trojan reactor is shown in Figure VI.6-11. The momentum equation was solved for the entire loop at each junction to give volume pressures. The steam generator has four heat transfer regions. An energy balance was computed by adjusting the heat transfer areas in a manner similar to that used by RETRAN, and by specifying a temperature distribution in the primary side.

6.2.2 Results of Analysis

Results of the null transient calculations for the hand-calculated problem are shown in Figures VI.6-12 and VI.6-13. It is noted that acceleration pressure drops for the hand calculation were on the order of 10^{-2} at time zero. This null transient was also run with different time step sizes (Figure VI.6-14) to ascertain time step size convergence. As the results show, the calculated transient which resulted was due solely to an inadequate steady-state solution from the hand calculation.

6.3 Summary of Results

The TMI-2 cooldown transient analysis demonstrated that RETRAN can be used for evaluating plant transients. The analysis in some cases identified deficiencies in plant data. Discrepancies which existed between the calculation and the data helped to identify code errors. In general, RETRAN predicted the behavior which occurred during the transient very well.

Although the results of the hand calculated steady-state solution for Trojan are not as dramatic as are those of the TMI-2 cooldown, they serve an important example of the benefits which can be achieved by performing such a calculation. The user acquires a better understanding of the code as well as the plant model.

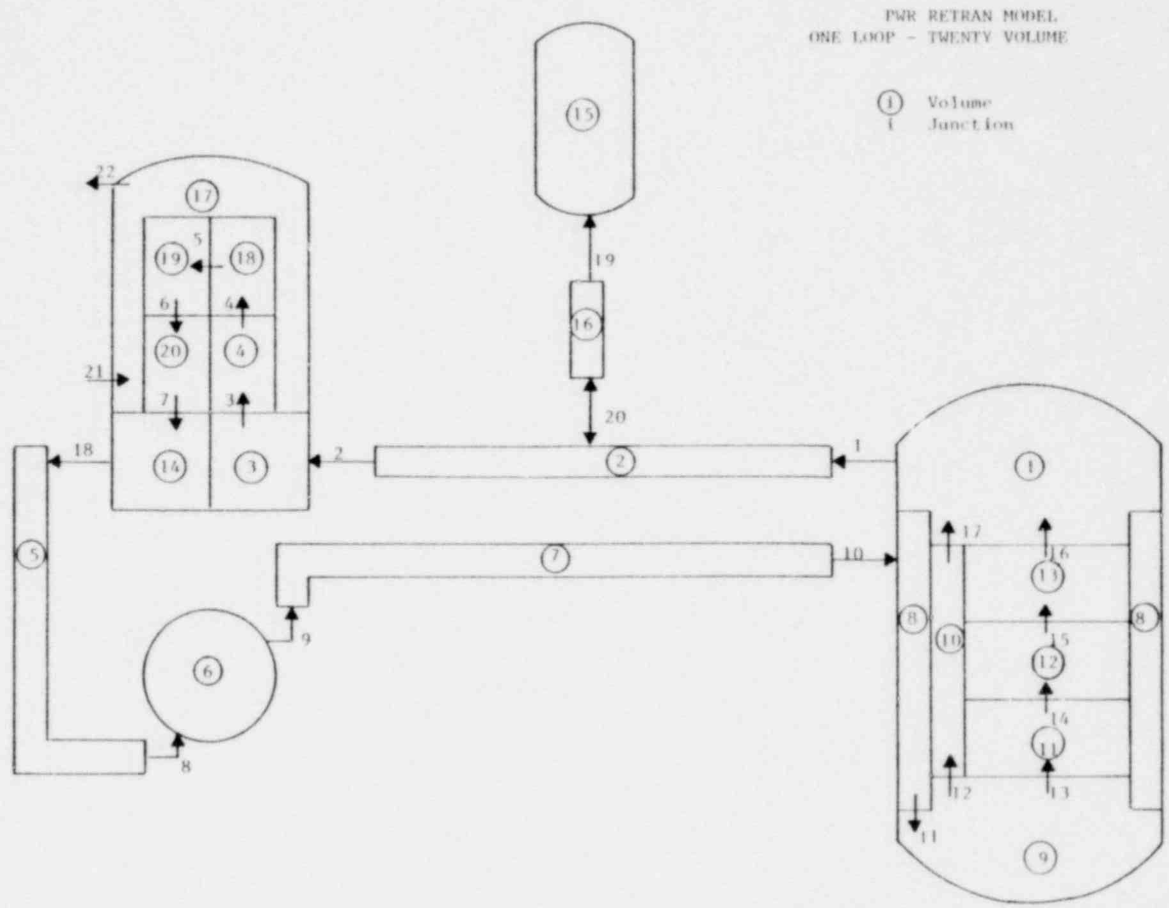


FIGURE VI.6-11 Trojan One Loop Model

0 017

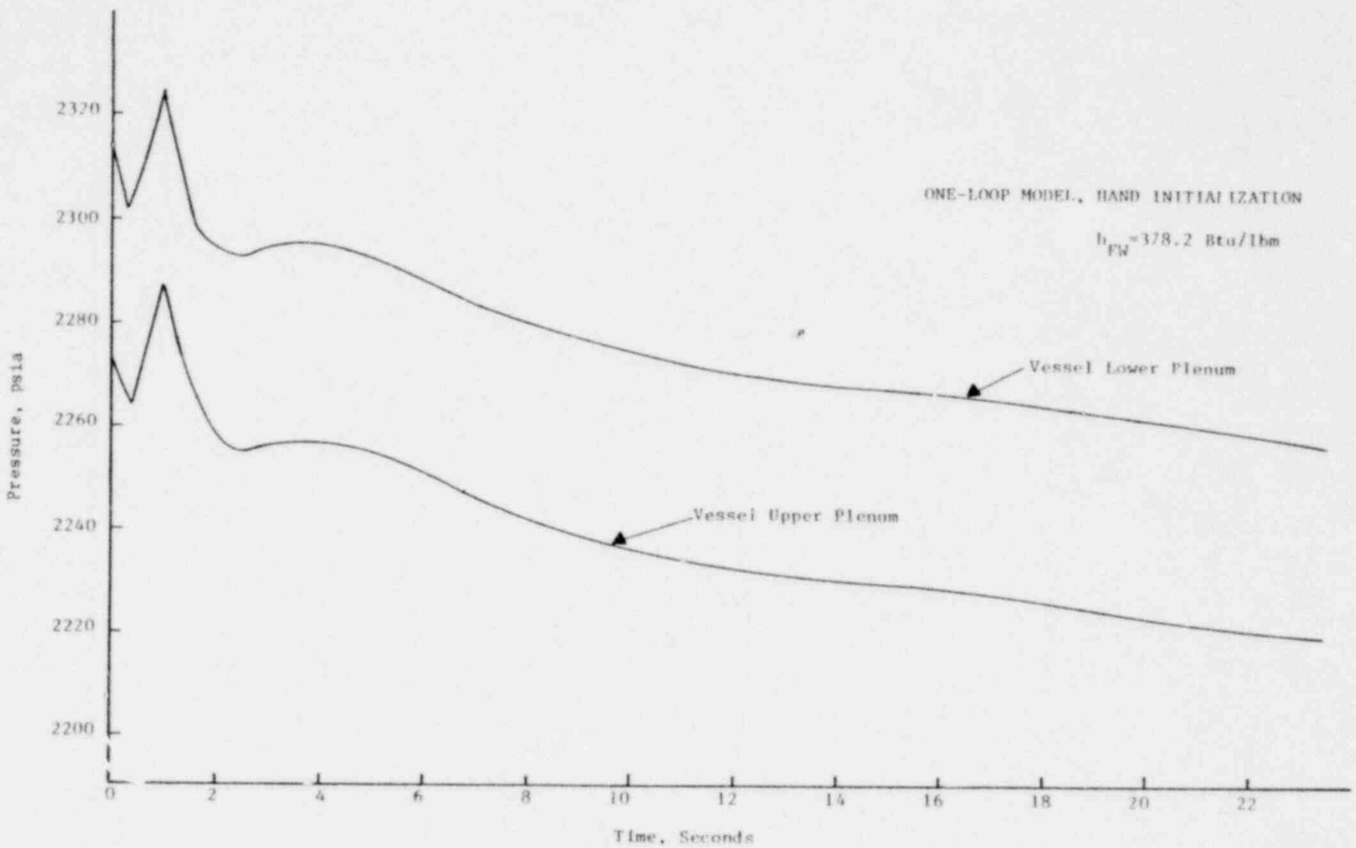


FIGURE VI.6-12 Effect of Low Feedwater Enthalpy on Steady-State Pressure

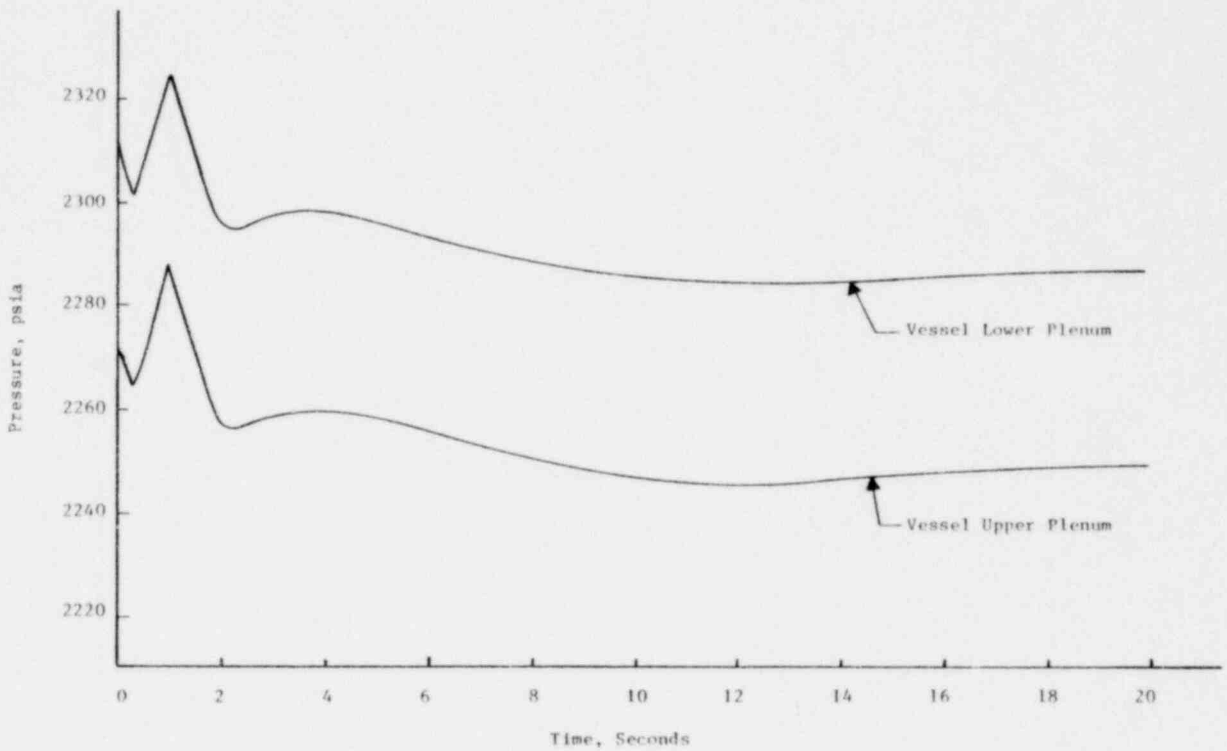


FIGURE VI.6-13 Effect of High Steam Generator Heat Transfer Area

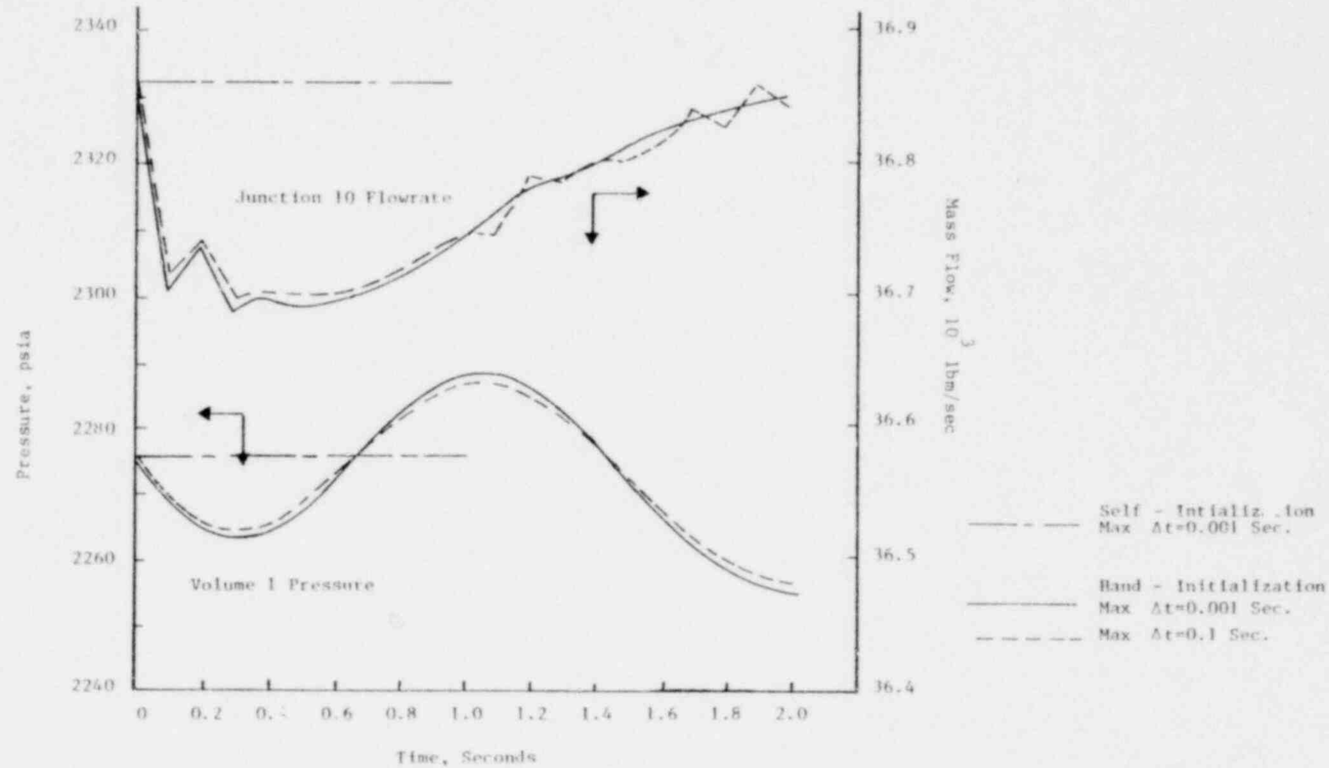


FIGURE VI.6-14 Effect of Maximum Time Step Size

1770 020

POOR ORIGINAL

VII. BWR LOSS-OF-COOLANT-ACCIDENT ANALYSES

1760 021

VII. BWR LOSS-OF-COOLANT-ACCIDENT ANALYSES

Loss-of-Coolant-Accident(LOCA) analyses for boiling water reactors are to be discussed in this section. There have not been any BWR LOCA analyses performed with RETRAN at this time. When such analyses are completed, the results will be included in this section.

[illegible]0 023

VIII. FSAR LOSS-OF-COOLANT-ACCIDENT ANALYSES

VIII. PWR LOSS-OF-COOLANT-ACCIDENT ANALYSES

Verification and qualification of the RETRAN computer code would not be complete without loss-of-coolant accident (LOCA) analysis. The prediction of system behavior following a LOCA with emergency core cooling system (ECCS) injection is a necessary test of the thermal-hydraulic transient capability of RETRAN. In particular, a best estimate RETRAN analysis of LOCA/ECCS phenomena contributes toward improving the analytical understanding of what would take place following a LOCA for a large reactor system.

In contrast to the reactor licensing "conservative" assumptions, the RETRAN code performs a "best estimate" analysis, which provides the opportunity for detailed comparison with experimental data, such as the LOFT or Semiscale test series. Thus, qualification of the code can be logically pursued without the arbitrary "conservative" assumptions. The System Effects Analyses and their comparisons with experimental data were discussed in Section IV. This section discusses LOCA analyses for two typical large PWR's, the Combustion Engineering System 80 and a typical Westinghouse four-loop system.

The results of the PWR LOCA analyses show that RETRAN is capable of modeling the blowdown portion of the LOCA for large PWR's. The results are reasonable, given the assumptions made for the analyses. Although the blowdown portion of the LOCA is a significant test of the calculated capabilities of the RETRAN code, this analysis should be extended into the refill and reflood portions of the transient.

1.0 COMBUSTION ENGINEERING SYSTEM 80

Combustion Engineering, Inc. has developed a new Nuclear Steam Supply System (NSSS) design to standardize the nuclear systems which they produce. The system, referred to as "System 80", is a 3817 MWt pressurized water reactor (PWR). The system incorporates a number of performance and safety improvements over previous C-E NSSS offerings as described in the Combustion Engineering Standard Safety Analysis Report (CESSAR).[VIII.1-1]

EI, on behalf of EPRI, performed a Best Estimate LOCA analysis and several sensitivity studies for the CE System 80 with the RETRAN code. This report summarizes the results of these analyses, which are discussed in more detail in Reference [VIII.2-2].

1.1 RETRAN Geometric Model Description

The geometric model of System 80 used for the RETRAN analysis is shown schematically in Figure VIII.1-1. The actual CE System 80 plant has two hot legs and each of these bifurcates into two cold legs upon exit from the steam generators. The RETRAN model combines two cold legs to form an intact double loop and considers each of the other cold legs individually to form an intact single loop and broken single loop.

The model consists of 50 fluid volumes, 60 junctions and 75 heat conductors. The nodalization was selected so as to identify important hydraulic features of the system and provide sufficient detail to attain accuracy in solution. The System 80 reactor vessel was modelled with 10 control volumes. Two volumes were used to model the downcomer region, two volumes in the lower plenum, 3 axial core regions, and one volume each for the core bypass, the upper plenum and the upper head.

The primary coolant piping and steam generators were modelled in some detail so that area changes and elevation differences would be included in the model. The hot leg was represented by one volume while the pump suction leg was represented by two volumes. The break location selected by CE was the discharge leg elbow. The RETRAN model divides the cold leg into two volumes such that the break junction will simulate this location. Six volumes were used to model the steam generator; one each for the inlet and outlet plena and four in the U-tube region. The pressurizer, pressurizer surge line, and primary coolant pump were each represented by a single control volume.

For this analysis, which was performed through the blowdown phase only, a very large volume was used to model the containment, resulting in a constant pressure sink.

Heat conductors were defined to model heat transfer from all significant metal masses in the System 80. At least one heat conductor was attached to each fluid volume in the System 80 model. The heat conductors were described by both

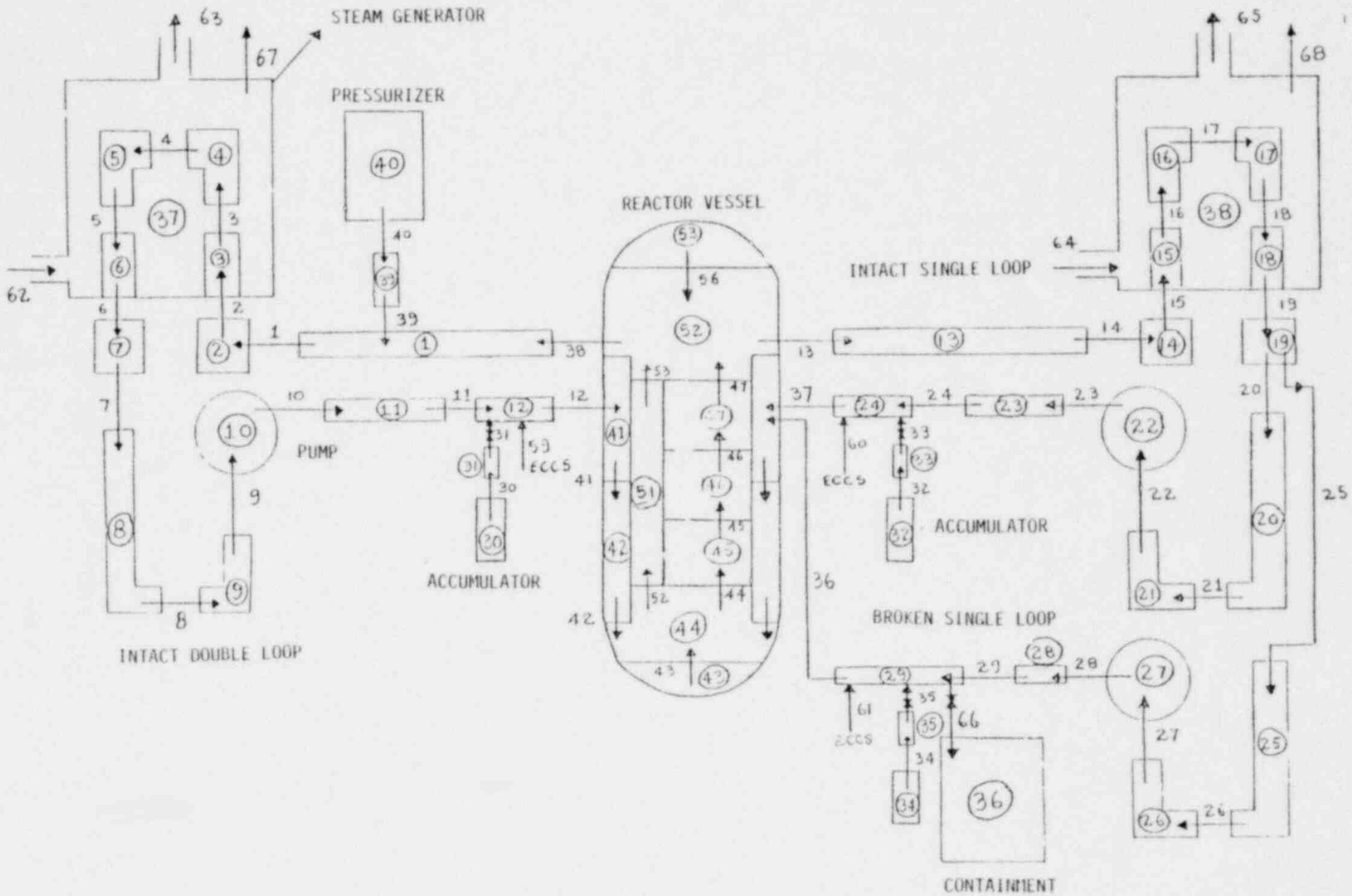


FIGURE VIII.1.1 RETRAN Model for CE System 80 Best Estimate LOCA Analysis cylindrical and rectangular geometry, the former being used mainly for pipes and the latter for slabs. The core conductors were treated in some detail, with three axial conductors (one for each fluid volume) used to describe the average assembly, three axial conductors for the hot assembly and twenty axial conductors to describe the hot rod so that the peak clad temperature could be closely approximated.

1.2 LOCA Assumptions and Modeling Options

The same basic LOCA assumptions as used by CE[VIII.1-3] were used in this study. The analyses in this report are only for the blowdown portion of the transient, and the basic assumptions discussed below are applicable only for this duration.

1.2.1 Pipe Break Size

The CE Best-Judgment analysis defines the largest mechanistic pipe rupture in the most adverse location that could occur at a System 80 plant as a 3.7 sq. ft. partial guillotine modeled as a longitudinal break (75% of the cross-sectional area) in the discharge leg pipe elbow at the reactor vessel end. This limiting size and location was developed using conservative stress criteria.

1.2.2 Critical Flow Model

RETRAN has the option of selecting a combination of choking as calculated by the compressible flow equations and explicit critical flow models based on Moody's or Henry's work. For System 80, input options were selected so that Henry's critical flow model was used in the subcooled region and Moody's critical flow model was used for the saturated region. Together with the choking model, the user has the choice of a choking coefficient or break flow multiplier that modifies the calculated critical flow. In general, this parameter is selected between 0.6 and 1.2 depending upon the fluid conditions present - the larger values usually being used for highly subcooled flow.

The CE Best-Judgment calculation used the homogeneous equilibrium model to determine the critical flow rate for two-phase fluid and the Henry-Fauske model with a discharge coefficient of 0.6 for subcooled fluid. For the case in which EI used approximately the same choking assumptions as CE, multipliers of 0.5 and

0.7 were used with the Moody model to produce flows roughly equivalent to the homogeneous equilibrium model. For the EI Best Estimate prediction, a contraction coefficient of 1.0 with the Henry model during subcooled blowdown and 0.6 with the Moody model during saturated blowdown was used. This selection was based on the comparisons with experimental data obtained from the Standard Problem Program using this combination.

1.2.3 Reactor Coolant Pumps

The CE Best-Judgment calculation assumes that the pumps would remain running through the blowdown portion of the transient. The logic for this assumption is that after pipe rupture, with a simultaneous loss of offsite power, house power would still be available through the turbine-generator until it is tripped when the reactor scrams on high containment pressure (roughly two seconds after break). The turbine-generator coasting-down will continue to supply power to the pumps through a time delay circuit for an additional 20-30 seconds. The turbine-pump link is designed to prevent pump overspeed, but also maintains a positive flow in the core and thus removes stored heat from the fuel.

In the RETRAN model, the primary coolant pumps are represented by fluid control volumes with single-phase pump characteristic curves supplied to describe pump head and torque performance. Two-phase effects on pump performance were simulated by providing difference curves (single-phase minus two-phase) for both pump head and torque in addition to a table of head and torque multipliers versus void fraction, based on the two-phase performance of the Semiscale pump.

1.2.4 Secondary System

The CE Best Judgment assumptions for the secondary system follow the most likely sequence of events which could occur during a LOCA. The assumption of simultaneous loss of offsite power with pipe rupture would cause a loss of electrical load (LOEL) at the turbine generator. This would close the turbine control valve resulting in an increase in secondary pressure which would cause a quick-opening of the main steam bypass valves, relieving 55% of the full power steam flow. The high containment pressure triggers a main steam line isolation signal causing a closure of the main steam isolation valves (MSIV). The feedwater flow is reduced concurrently. The timing of events is

Time After Rupture	Secondary System Reaction	Steam Flow	Feedwater Flow
0 secs	Normal operation	100%	100%
0.2 secs	LOEL, opens bypass valves	55%	100%
2 secs	High containment pressure trips MSIV	55%	100%
7 secs	5 second closure for MSIV and Feedwater valve	0%	0%

The RETRAN model for the System 80 used positive and negative fills on the steam generators such that the steam and feed flow simulate the above sequence of events.

1.2.5 Momentum Flux

In the model for System 80, the compressible single stream flow equation was used for all junctions where momentum flux was desired. This was applicable wherever control volumes represented a one-dimensional stream tube. The junctions where no momentum flux was used were those where water entered the system at right angles to the flow. Examples of this are all tee junctions like the pressurizer surge line entrance to the primary coolant system. For these junctions, a RETRAN combination of JCALCI = -4 and MVMIX = 3 have been used. Also, geometries where predominantly multidimensional flows occur were modelled with large volume flow areas, such as the reactor vessel upper and lower plenums, and reactor vessel downcomer. The large volume flow area effectively eliminates momentum flux terms.

1.2.6 Phase Separation

Another important RETRAN input option is the application of the phase separation or "bubble-rise" model. This option permits steam to become separated from liquid within a control volume, or to have a bubble density gradient in a fluid volume without separation. Hence, one quantity specifies the bubble density gradient within the mixture while another quantity specifies the velocity of the escaping steam bubbles relative to the mixture interface.

In the analysis of System 80, the pressurizer, lower plenum and steam generator secondary side were modelled with a density gradient of 0.8 and a bubble separation velocity of 3.0 ft/sec. The downcomer regions and the lower head were modelled with a gradient of 1.0 and a very large velocity. In all other volumes, homogenous flow conditions were assumed.

1.2.7 Safety Injection System

The RETRAN model of the Safety Injection System includes a safety injection tank volume and a fill junction to simulate the combined HPSI and LPSI flows. However, the safety injection system was not included for any of the analyses.

1.3 Results of Analysis

The RETRAN input for the System 80 model was developed from information provided by CE.[VIII.1-4] Steady state conditions for the System 80 model were obtained by the RETRAN code, with the system pressures propagated from a specified hot leg pressure. The gap conductivity was adjusted to get the initial fuel temperatures defined by CE. Some of the important initial conditions are listed below.

INITIAL CONDITIONS FOR SYSTEM 80 LOCA

Parameter	Value
Thermal Power	3800 Mwt
Primary Pressure	2250 psia
System Flow Rate	164×10^6 lbm/hr
Break Area	3.7 ft^2
Core Average Temperature	593°F
Steam Generator Secondary Pressure	1070 psia

0 030

Several LOCA analyses were performed with the RETRAN model for System 80. These include predictions and sensitivity studies for:

- o A LOCA prediction with CE break flow assumptions.
- o A LOCA prediction with EI break flow assumptions.
- o A sensitivity study with and without reactor scram.
- o A sensitivity study with and without reactor coolant pump trip.
- o A sensitivity study with the pressurizer in the broken loop.

1.3.1 LOCA Prediction with CE Break Flow Assumptions

As mentioned in Section 1.2.2, CE used the Henry-Fauske critical flow model with a 0.6 multiplier for subcooled blowdown and the homogeneous equilibrium model for saturated flow. Since the homogeneous equilibrium model is not available in RETRAN, a comparison of this model with the Moody model was made. It was determined that for this LOCA analysis, a constant multiplier in the order of 0.5 to 0.7 applied to the Moody model would make it approximately equivalent to the homogenous equilibrium model over the pressure and quality range that would be experienced during the blowdown.

Consequently, two predictions were made; one using a 0.5 multiplier and the other a 0.7 multiplier on the Moody model for saturated flow. Both predictions used the 0.6 multiplier for subcooled blowdown. The model with the 0.5 Moody multiplier was established as the base case and all sensitivity studies were made about this base case by changing other parameters.

The LOCA predictions for some important parameters are shown in Figures VIII.1-2 through VIII.1-5. The results are typical of a pressurized water reactor large break blowdown, where the system decompresses rapidly to the saturation point with the pressure continuing to decrease smoothly to an ambient condition. The high containment pressure reactor scram was simulated by a time trip occurring at 2 seconds, however the power is reduced considerably in one second due to the formation of voids in the core.

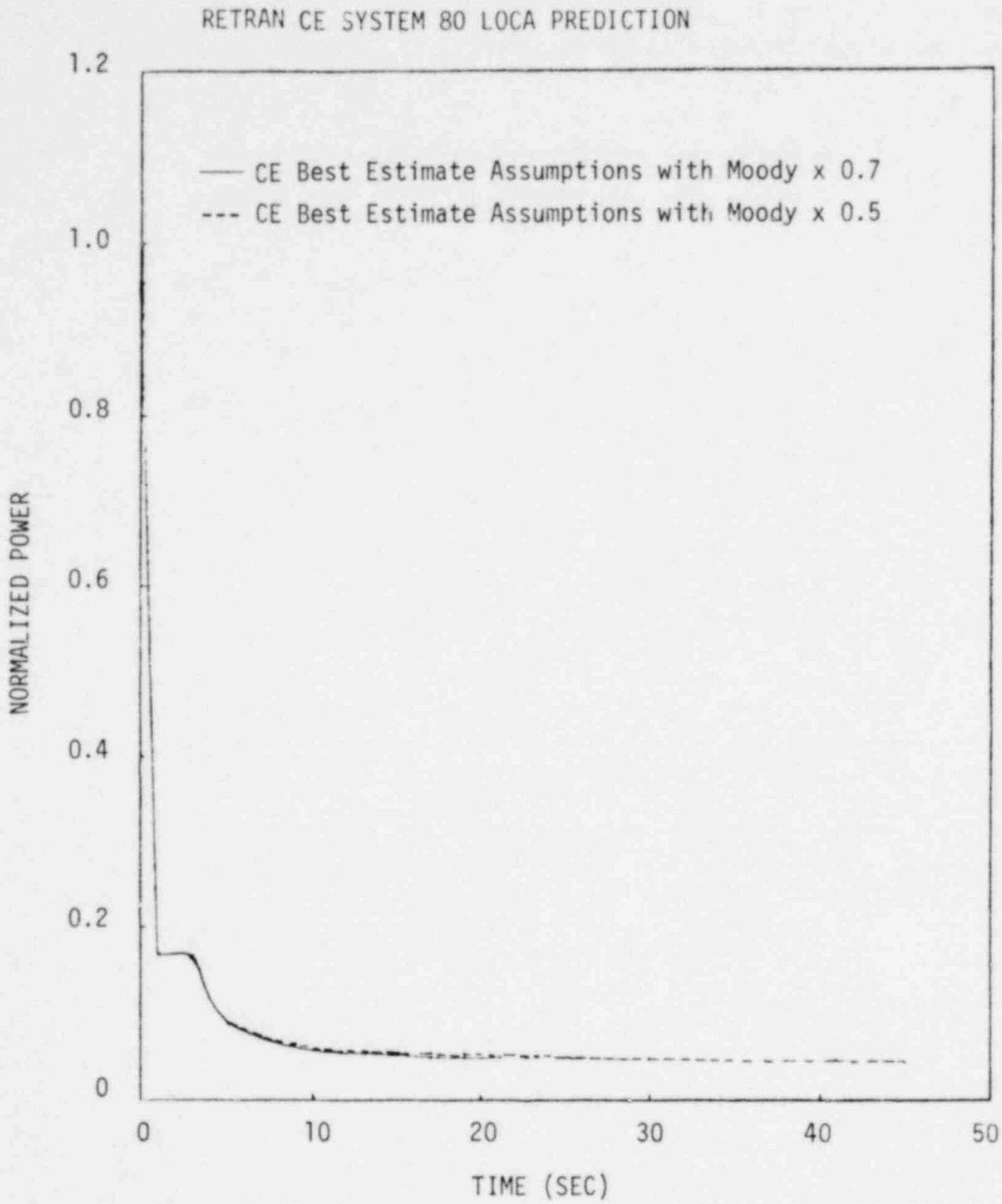


FIGURE VIII.1-2 Normalized Power

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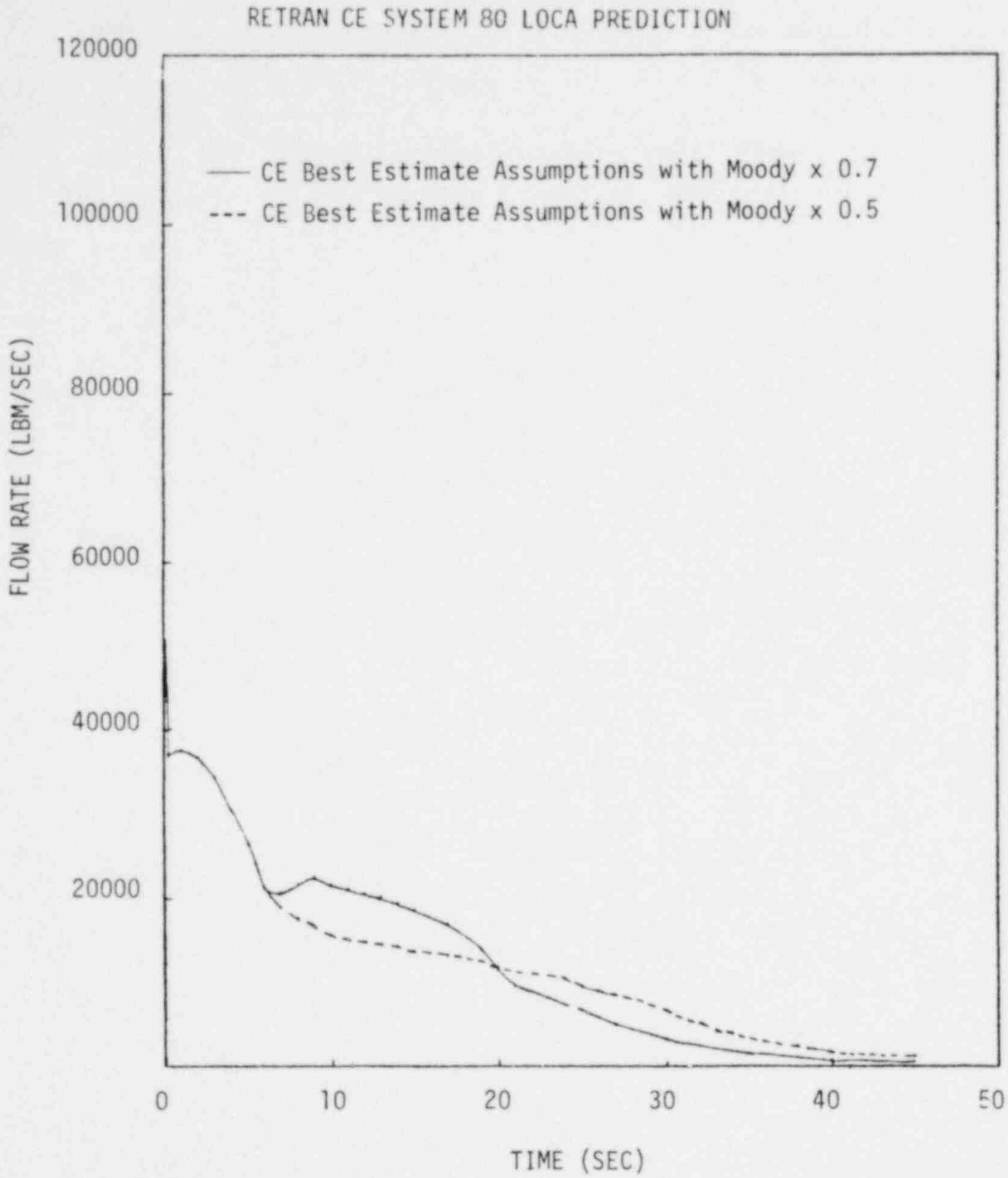

FIGURE VIII.1-3 Break Flow Rate (Junction 66)

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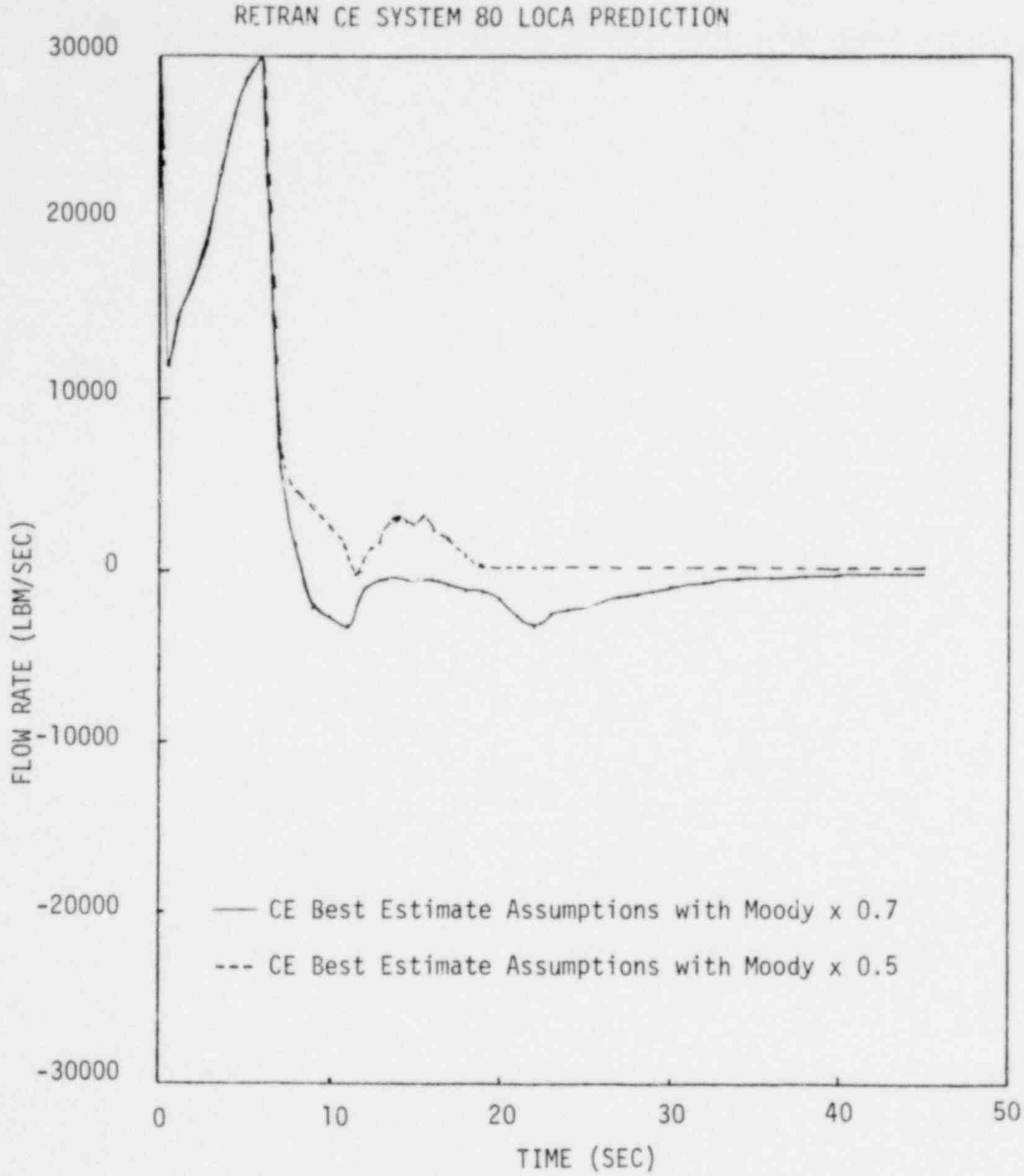

FIGURE VIII.1-4 Core Inlet Flow Rate (Junction 44)

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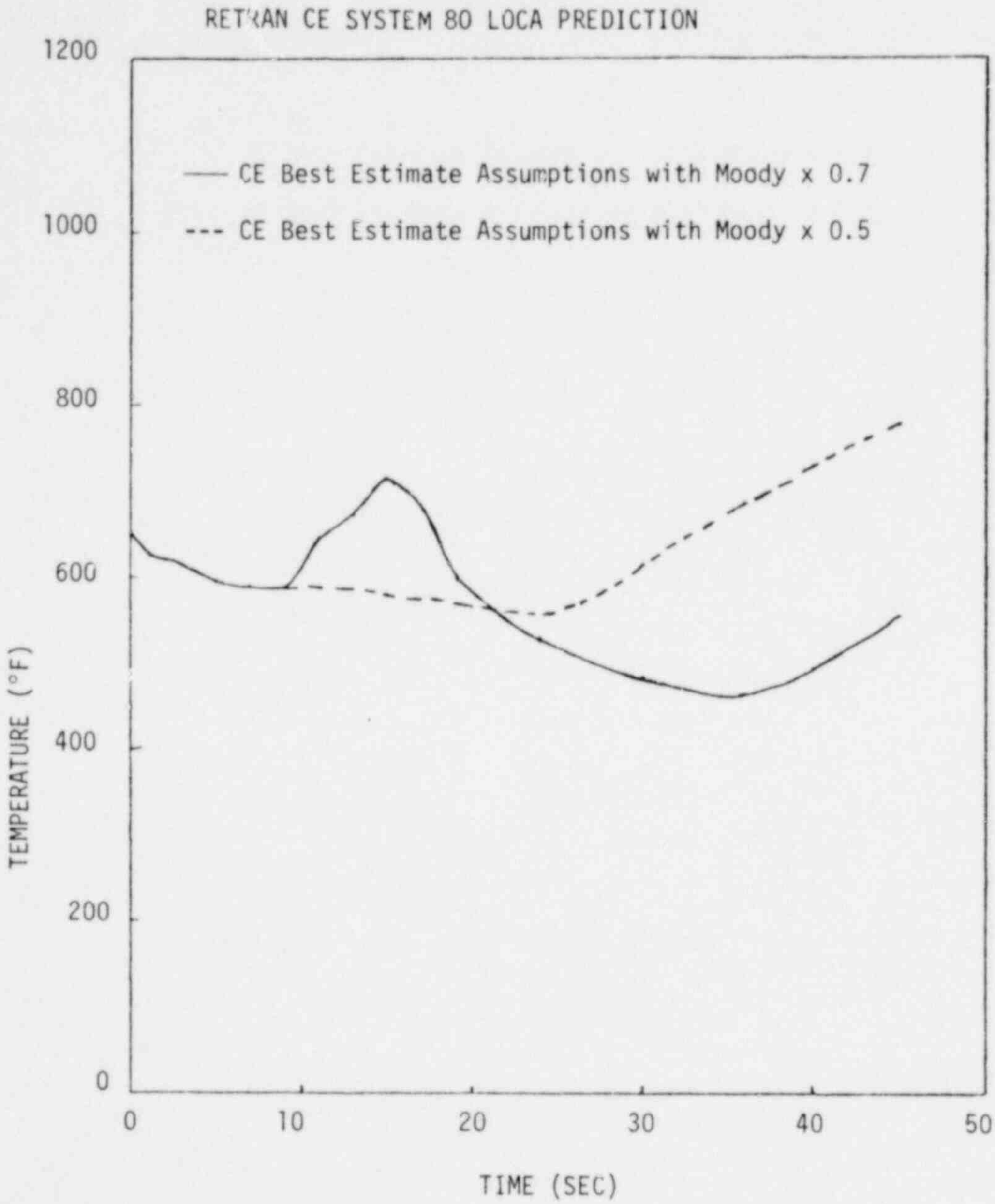

FIGURE VIII.1-5 Hot Assembly Clad Temperature at 70% Elevation (Average of Conductors 5 & 6)

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The break flow rate is higher with the 0.7 Moody multiplier after saturation conditions are reached. This effect is reflected in the core inlet flow, which remains positive throughout the transient for the case with a 0.5 multiplier on the Moody model, but reverses for the case with the larger multiplier, as more flow goes up the downcomer and out the break. These trends are shown in Figures VIII.1-3 and VIII.1-4. The intact loop flows do not deviate much between the two cases.

The effects of the core flow reversal for the 0.7 multiplier case are shown in Figures VIII.1-5 and VIII.1-6, which show clad temperatures that peak after this occurs, due to the departure from nucleate boiling. Later in the transient, the temperatures start increasing as a result of dryout in the core.

1.3.2 LOCA Predictions with EI Break Flow Assumptions

The EI Best Estimate LOCA differs from the CE assumptions in the use of the break flow multipliers. From past experience, especially with the Standard Problem Program, the Henry model for subcooled blowdown and the Moody model with a 0.6 multiplier for saturated blowdown has produced good predictions of the system blowdown.

Some of the predictions with this critical flow model for the System 80 LOCA are shown in Figures VIII.1-7 to VIII.1-11. The results are considerably different from the cases analyzed with the constant break multipliers. The depressurization is much faster due to the higher initial break flow, since a break multiplier of 1.0 is applied to the Henry model. This causes the core flow to go negative very early in the transient and results in an early peaking of the clad temperature following the departure from nucleate boiling. A second peaking occurs again when the core flow goes negative later in the transient.

1.3.3 Sensitivity Study With and Without a Reactor Scram

In the base case large break LOCA analysis, reactor scram was assumed to occur as a result of a high containment pressure signal. This trip was modeled to occur two seconds after rupture, since a detailed containment model was not used in the current studies. One of the sensitivity studies examined the effect of scram failure on the system. As expected, the analysis showed (Figure VIII.1-12)

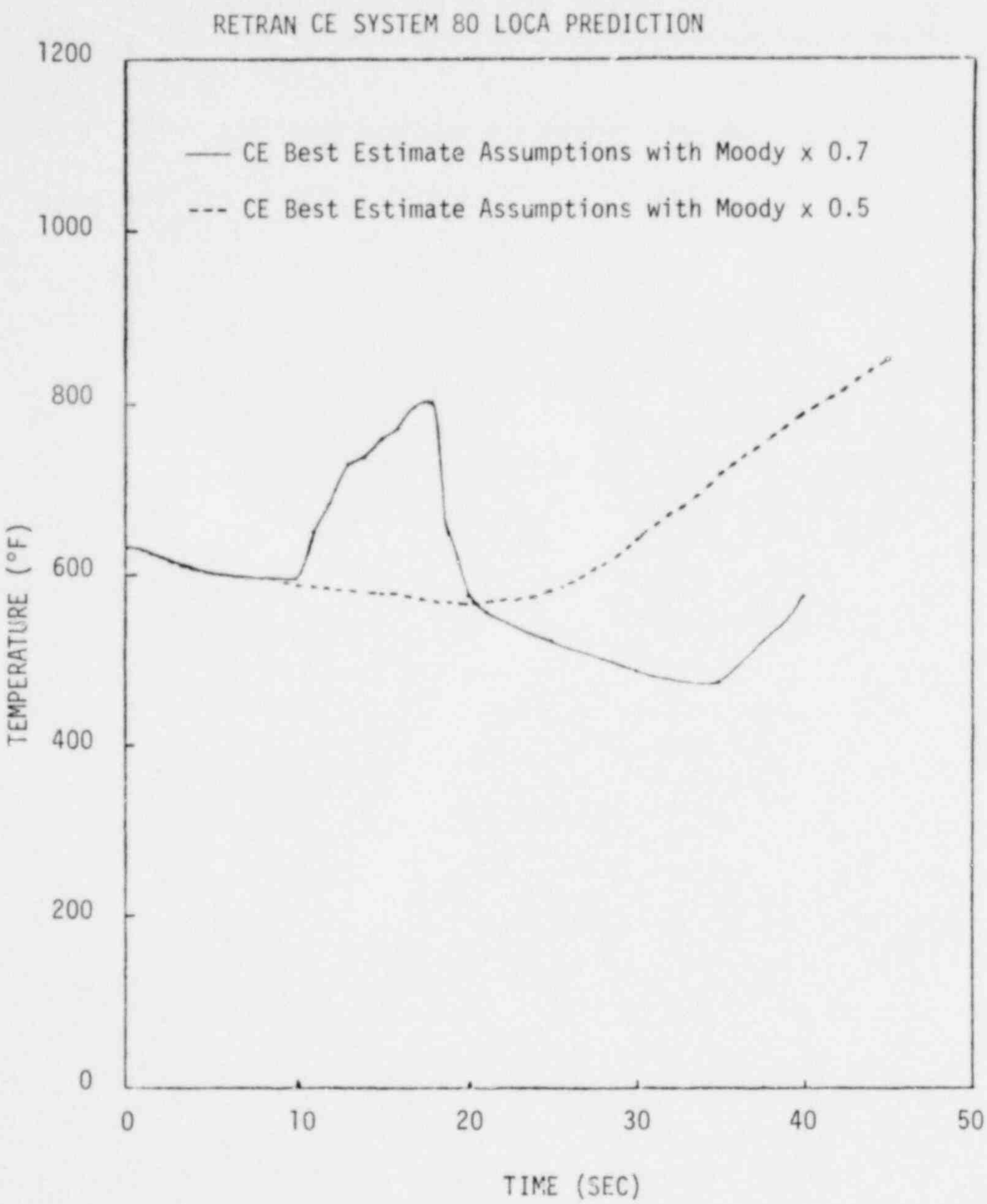

FIGURE VIII.1-6 Peak Clad Temperature (Conductor 19)

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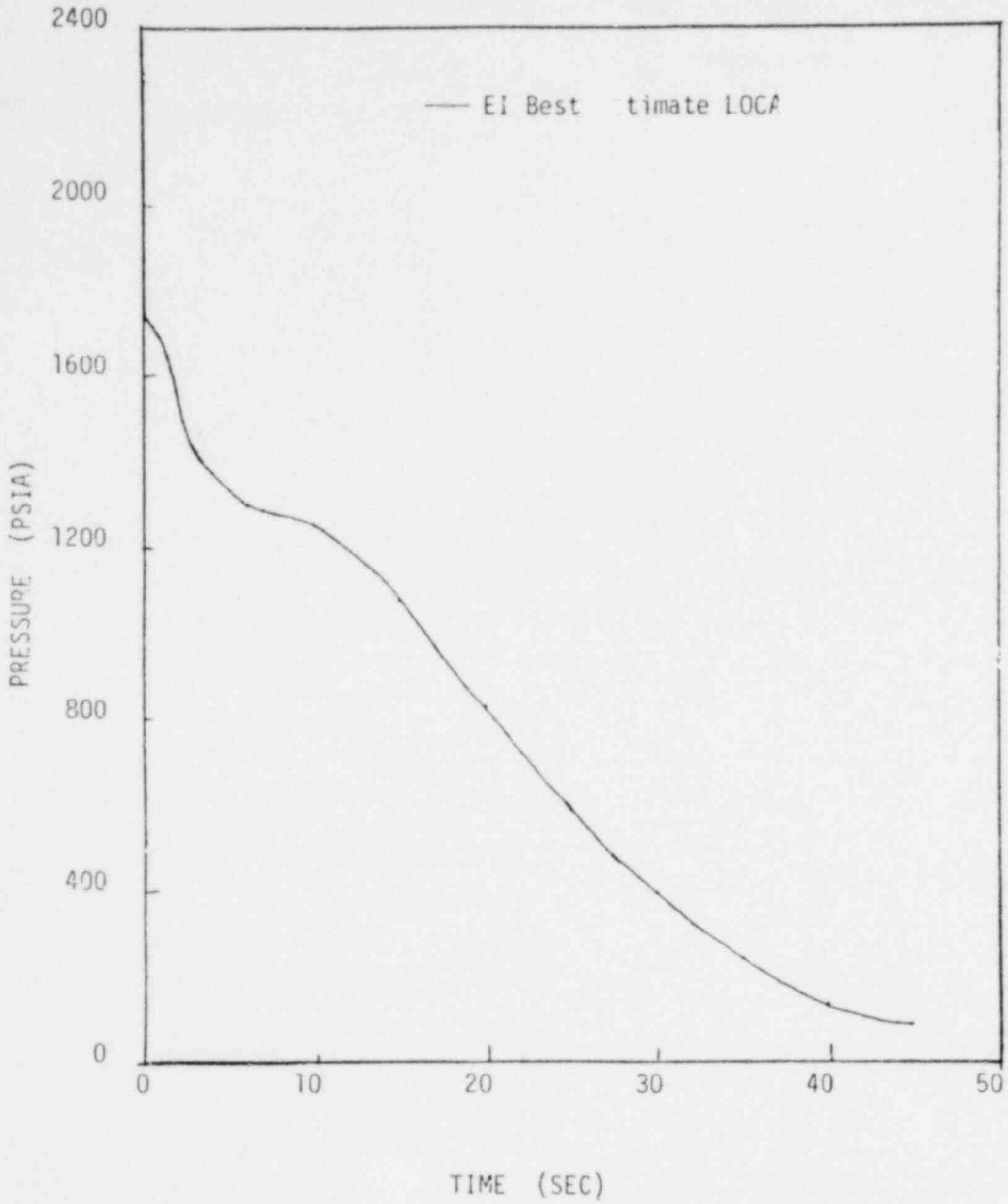

FIGURE VIII.1-7 Upper Plenum Pressure

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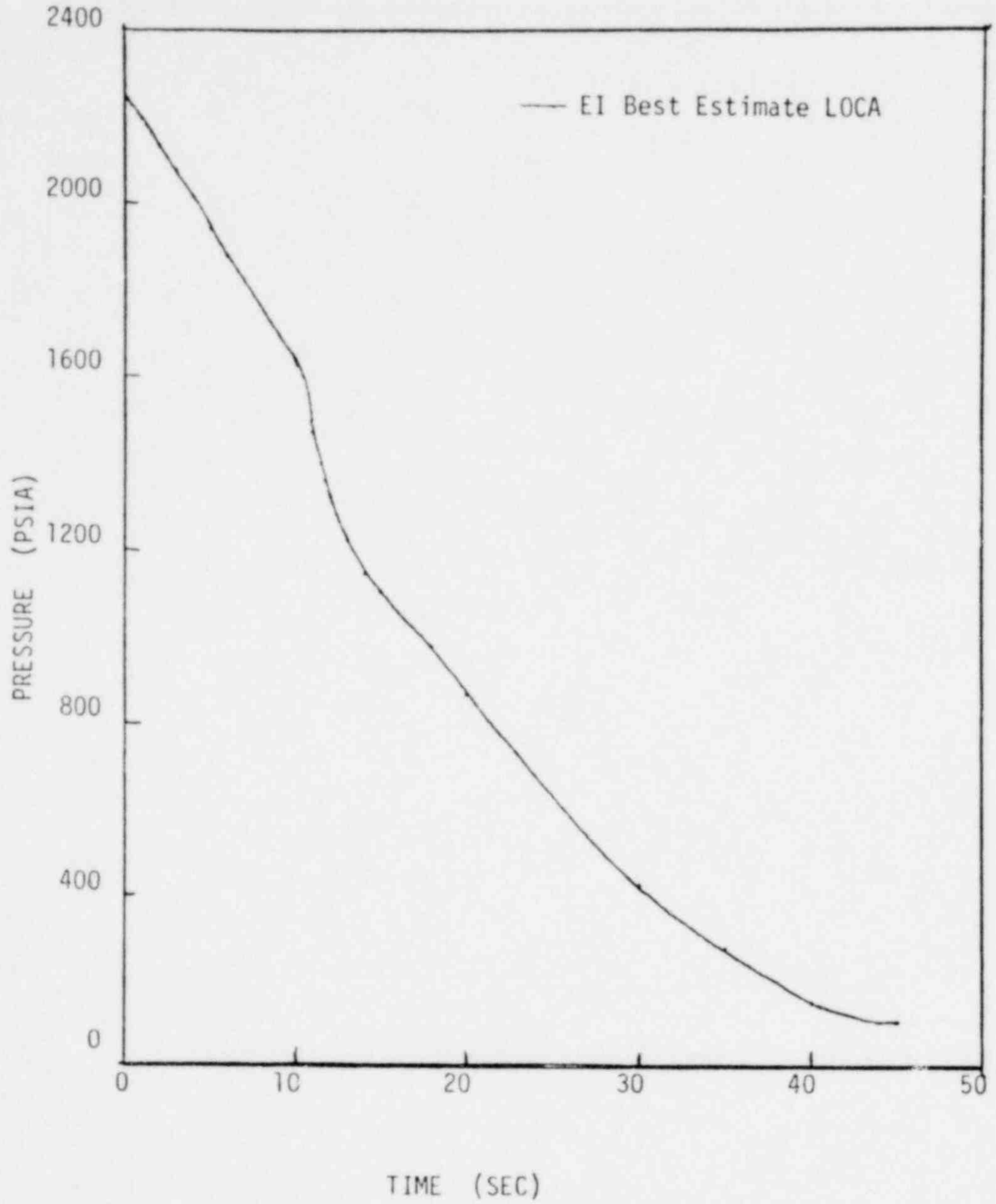

FIGURE VIII.1-8 Pressurizer Pressure

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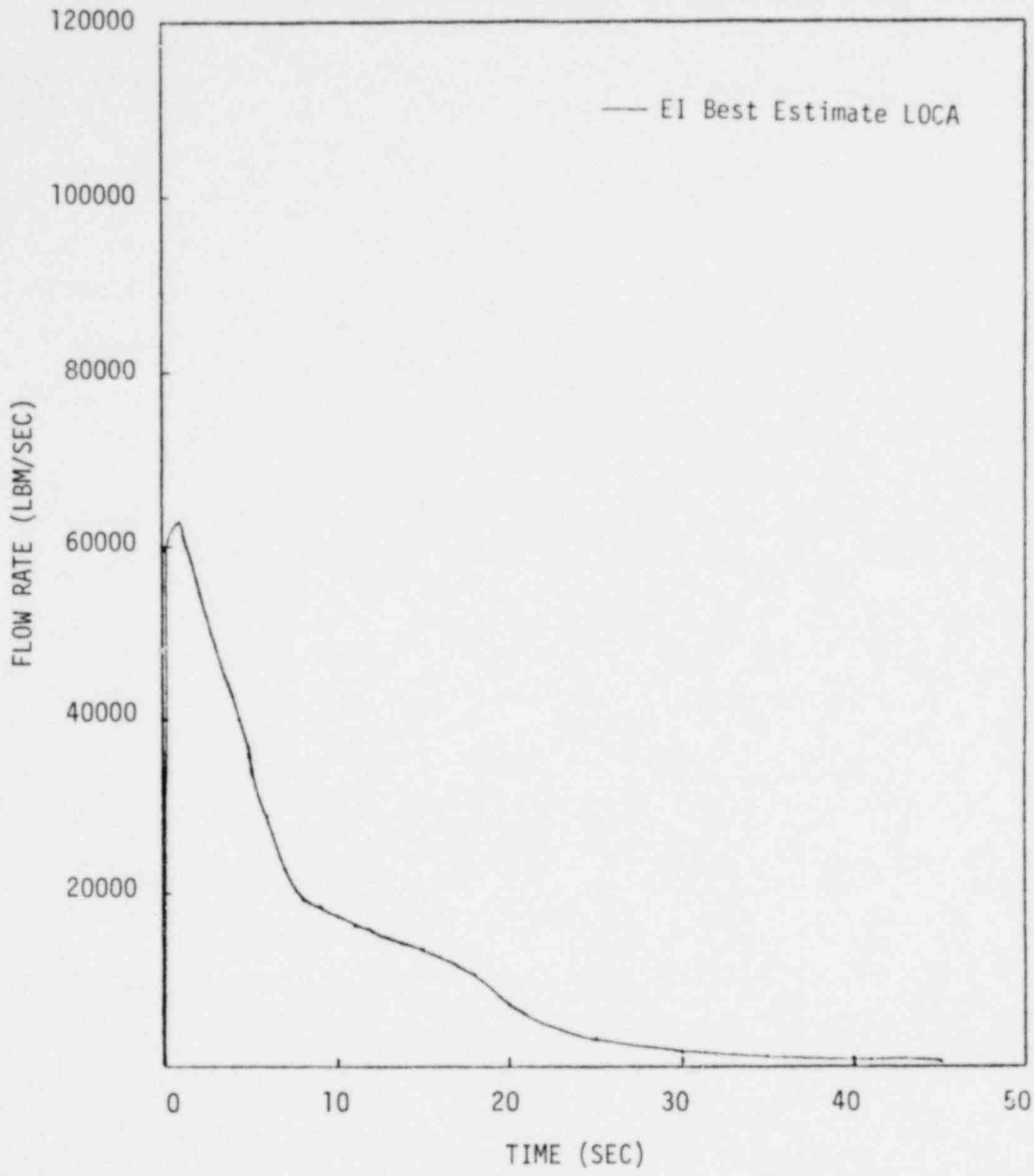

FIGURE VIII.1-9 Break Flow Rate

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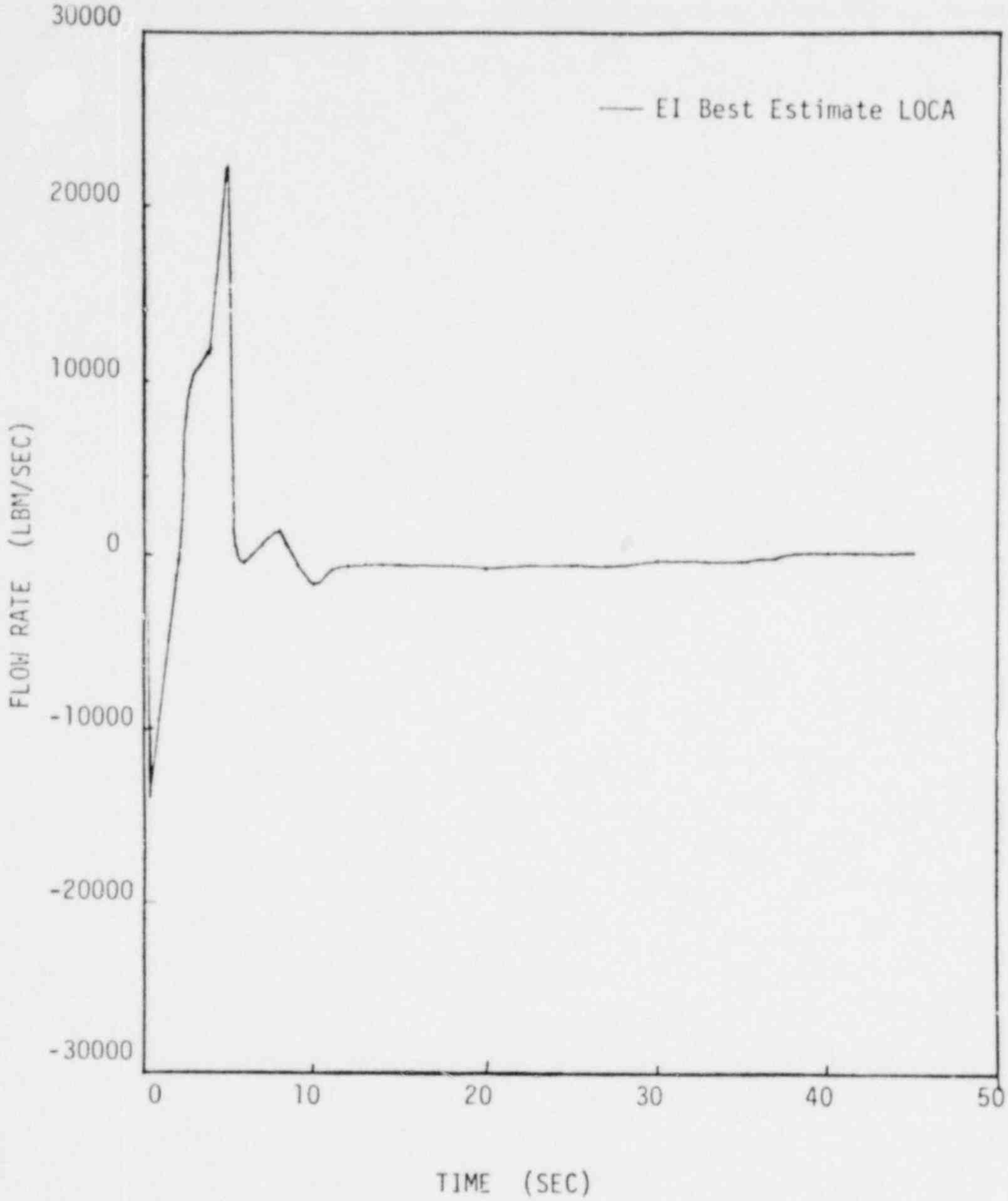

FIGURE VIII.1-10 Core Inlet Flow Rate

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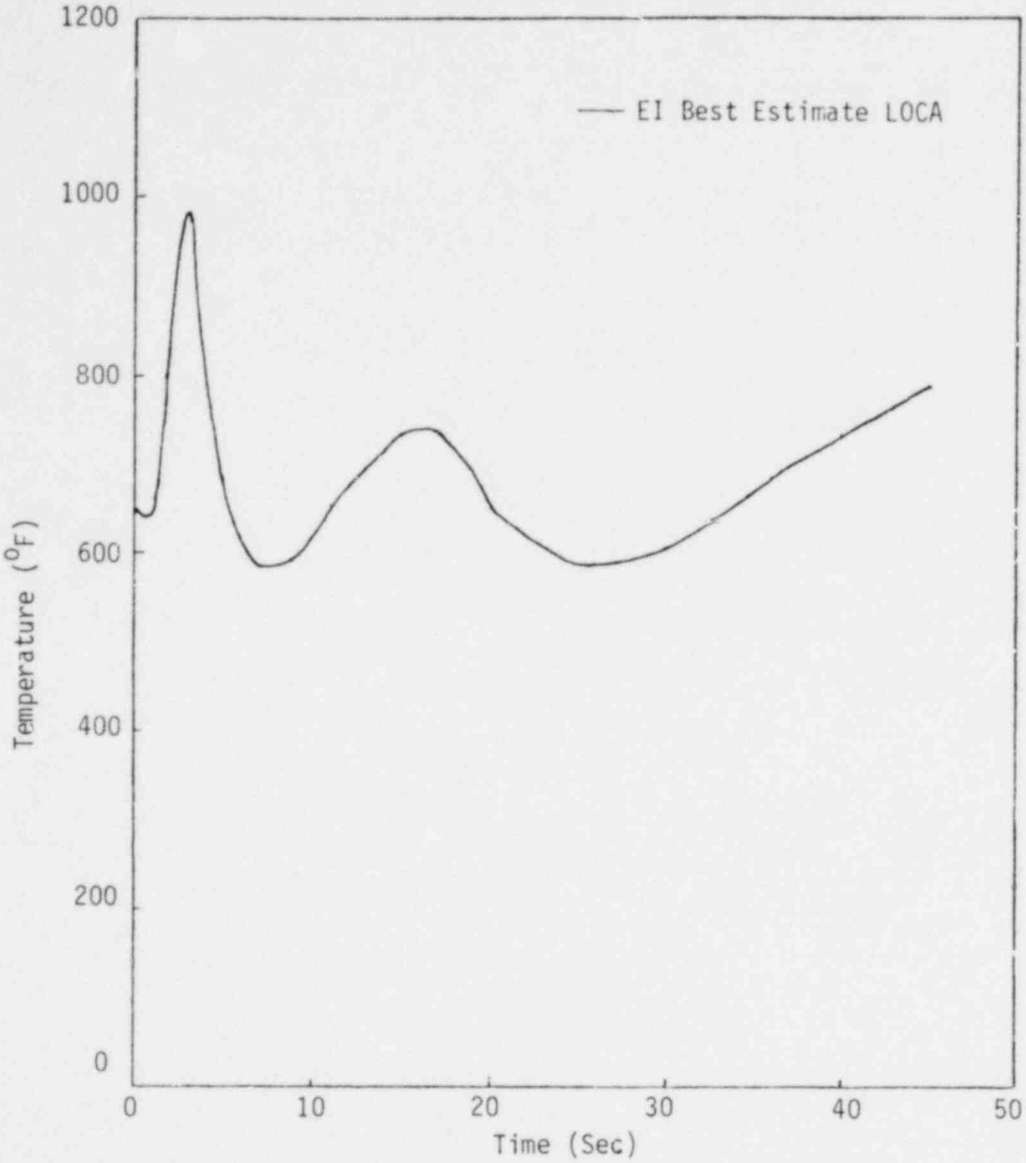

FIGURE VIII.1-11 Hot Assembly Clad Temperature at 70% Elevation

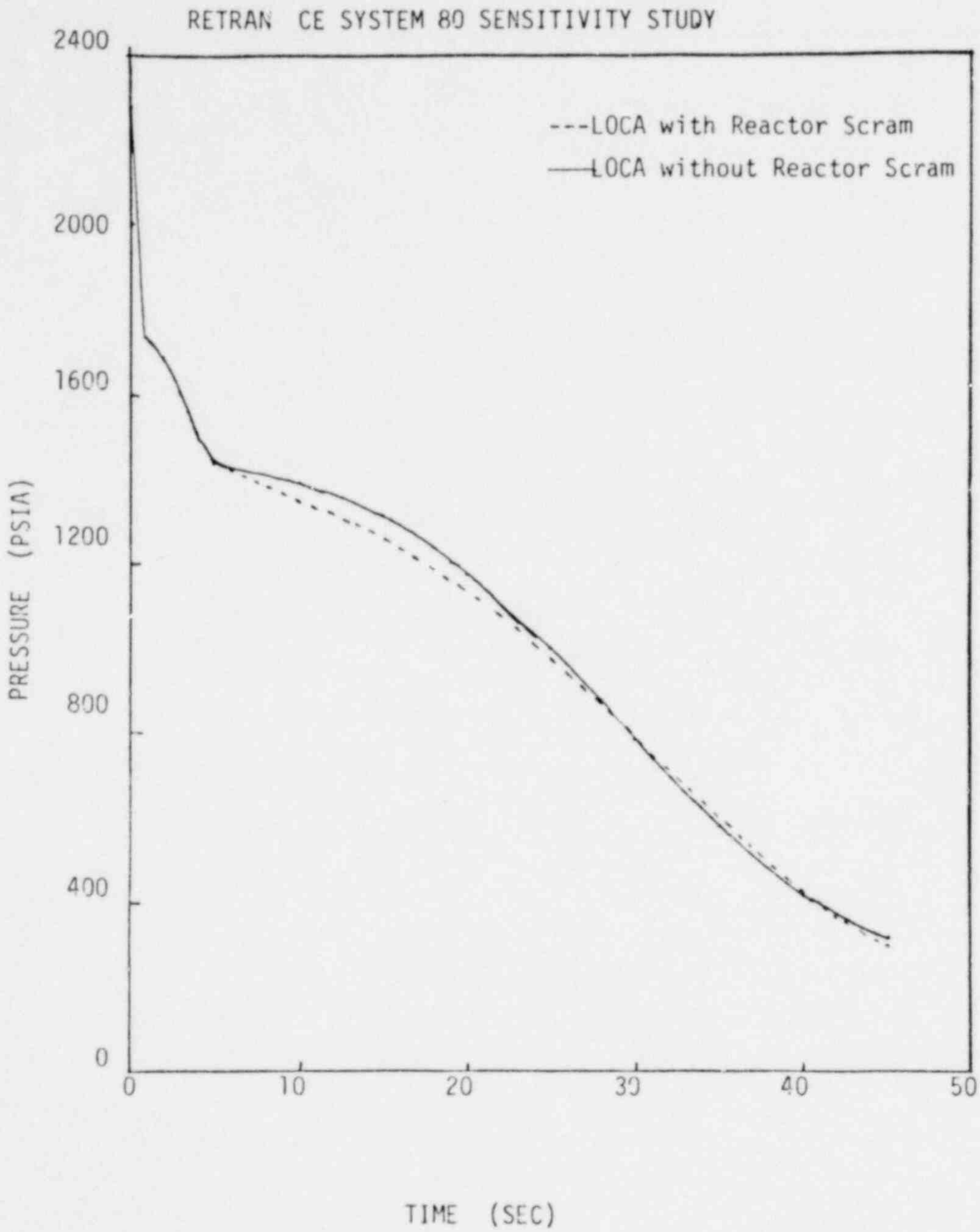

FIGURE VIII.1.12 Upper Plenum Pressure

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that the occurrence of a reactor scram has little or no effect on the important system variables during a large break LOCA. This is a direct consequence of the substantial void formation that takes place in the core during a large break. It was interesting to note, however, that a slight core power surge takes place during the period from 3-5 seconds after rupture. A detailed study showed that, from 1-5 seconds after rupture, there is a partial re-establishment of flow through the core and a resulting reduction of the core void content.

1.3.4 Sensitivity Study With and Without Reactor Coolant Pump Trip

The reactor coolant pumps are assumed to be in operation for the Best Estimate analysis. While the logic is valid for 25-30 seconds of the blowdown, the conservatism of the simultaneous pump coast down with pipe rupture is examined in this sensitivity study. Figures VIII.1-13 through VIII.1-16 emphasize the effects of the pump trip in the core inlet flow rate and the hot assembly clad temperature. Positive flow cannot be maintained in the core without the pumps running and this results in a higher peak clad temperature for the case without pumps running. The increase in temperature after 25 seconds is due to dryout in the core.

1.3.5 Sensitivity Study with Pressurizer in Broken Loop

The base case analysis assumed that the pressurizer was connected to the hot leg of the intact loop. To determine the significance of this assumption, an analysis was performed with the pressurizer connected to the broken loop hot leg. The location of the pressurizer was found to have no significant effect on the transient.

1.4 Conclusions

The LOCA predictions with CE break flow assumptions and with EI break flow assumptions show that critical flow models and the break flow multipliers used for the analyses play an important role in the system response. More work should be done in determining the best critical flow assumptions to be used with different types and sizes of breaks.

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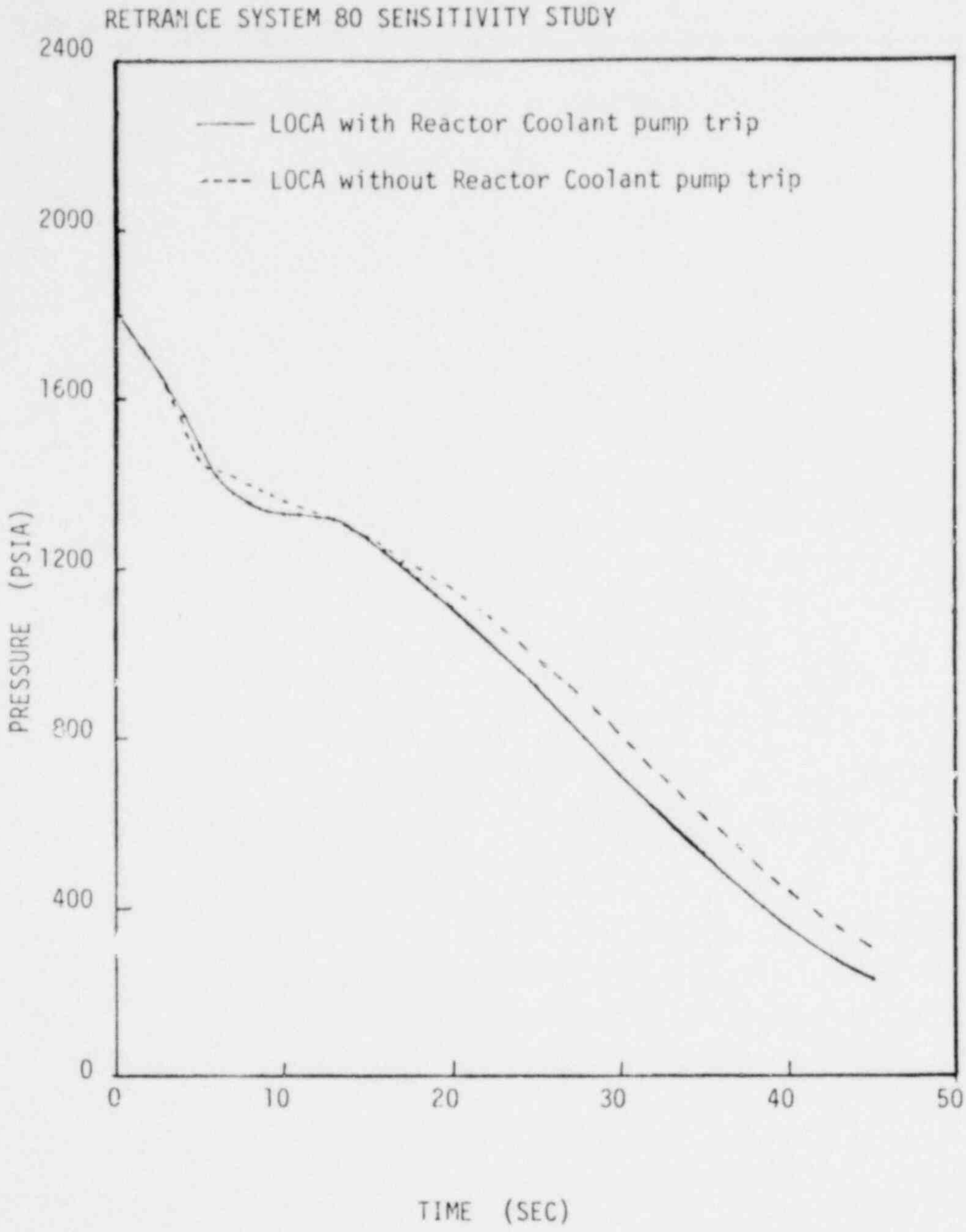

FIGURE VIII.1-13 Upper Plenum Pressure

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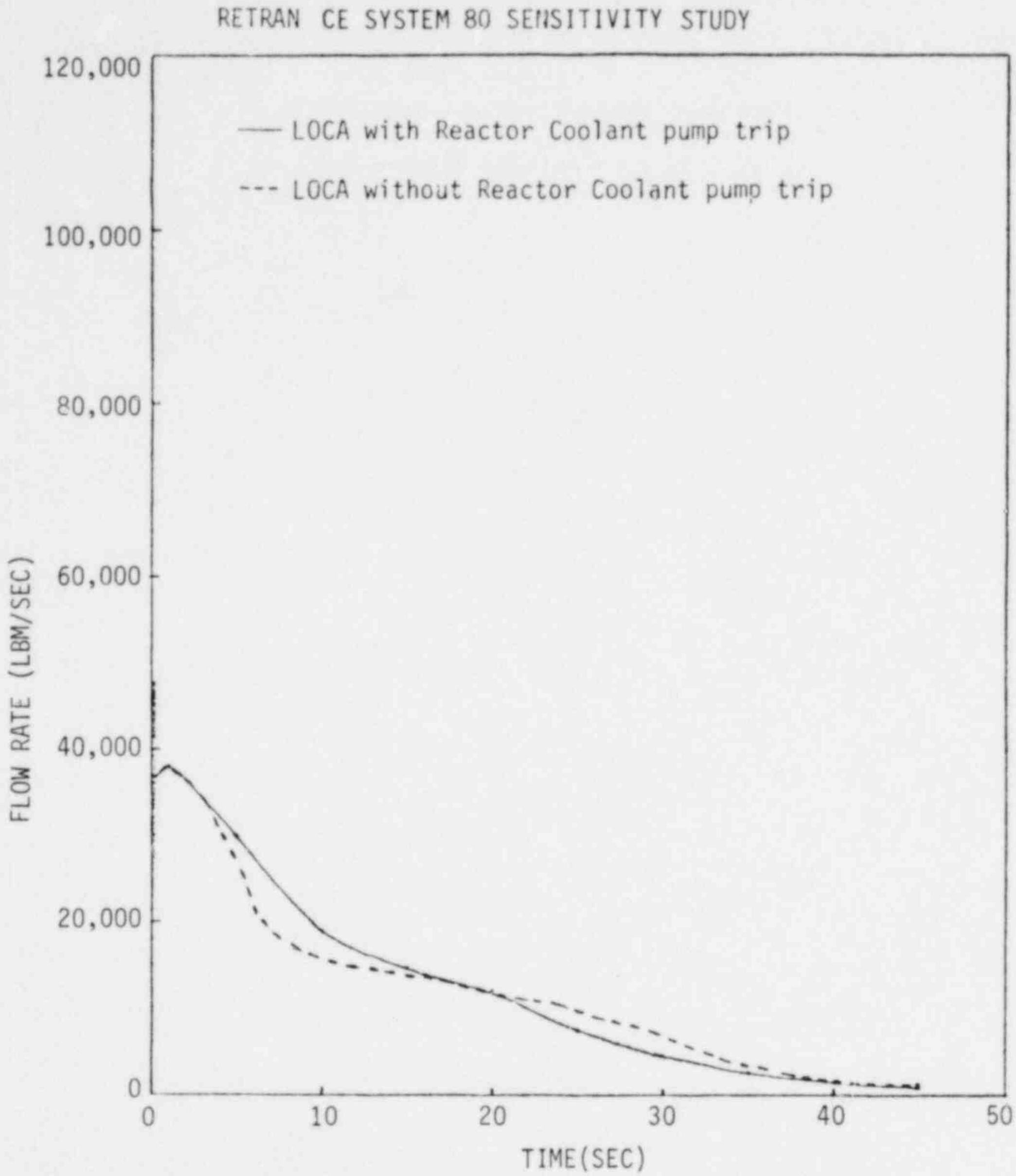

FIGURE VIII.1-14 Break Flow Rate

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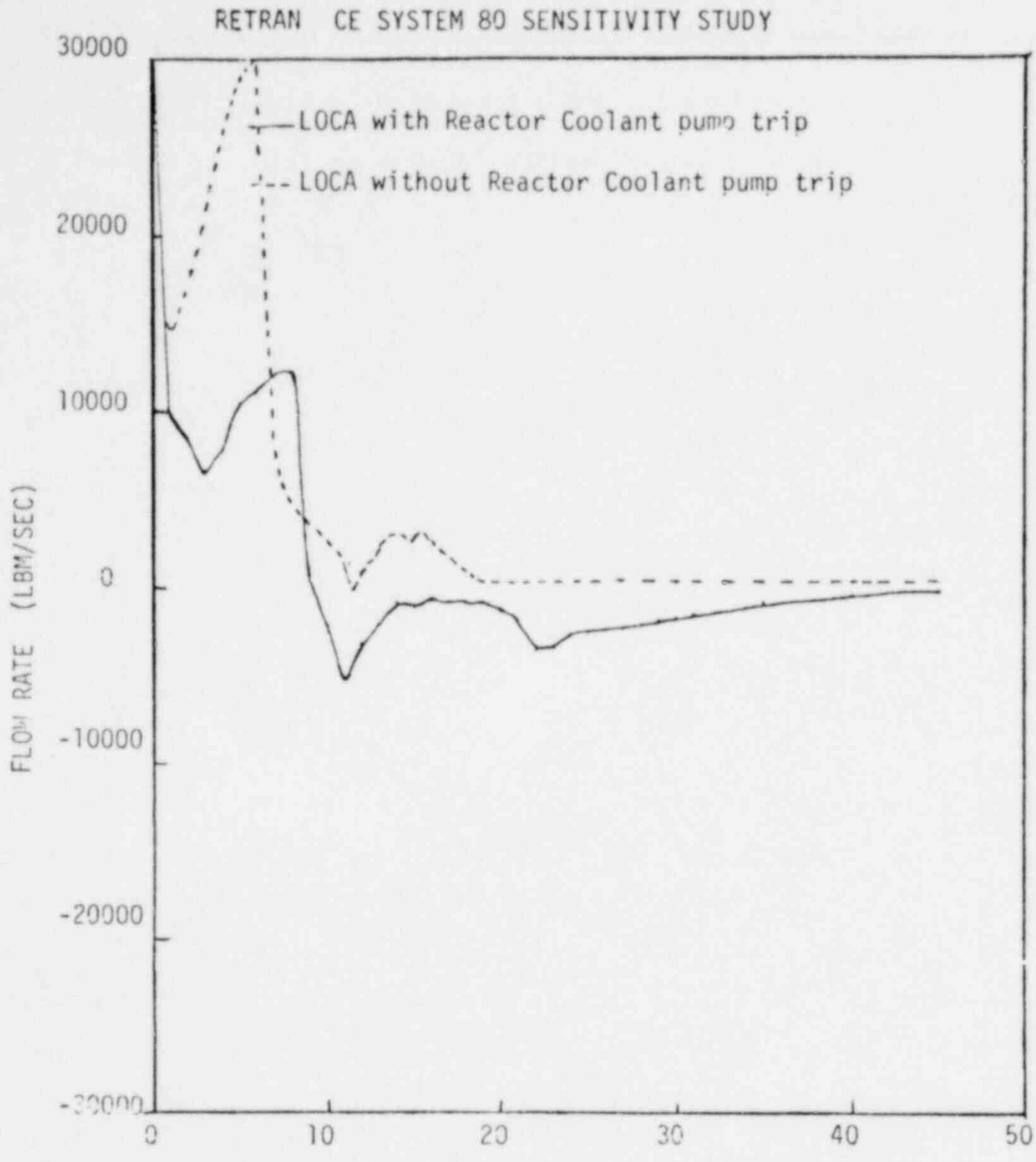

FIGURE VIII.1-15 Core Inlet Flow Rate

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RETRAN CE SYSTEM 80 SENSITIVITY STUDY

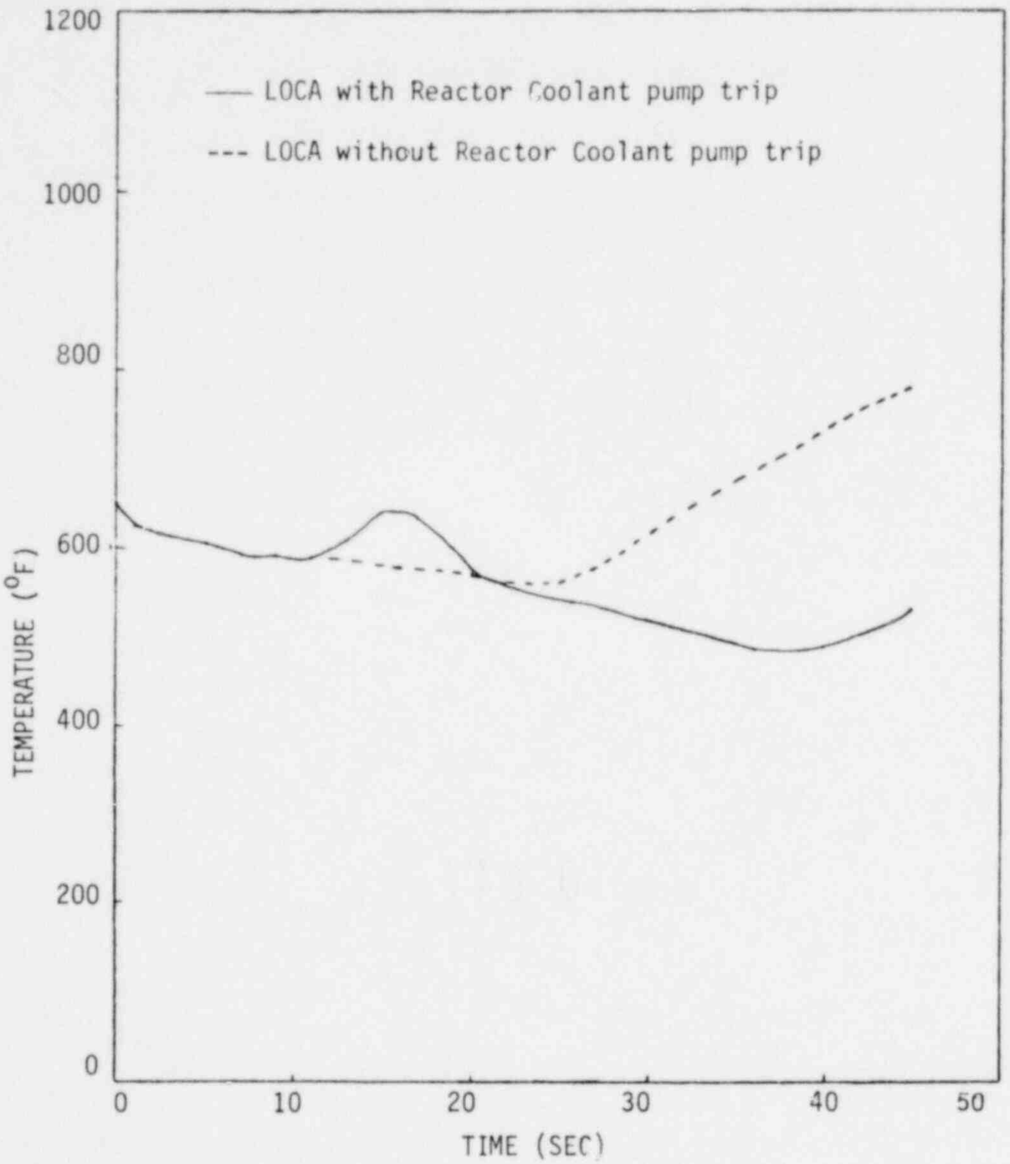

FIGURE VIII.1-16 Hot Assembly Clad Temperature at 70% Elevation

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The sensitivity studies showed that the reactor scram has little or no effect on the system response for this accident. The operation of the pumps through the blowdown has a significant effect in maintaining a positive core flow which removes stored energy from the fuel. Finally, the location of the pressurizer has no significant effect on the system response.

While the blowdown portion of the LOCA is a significant test of the calculational capabilities of the RETRAN code, this analysis should be extended into the REFILL and REFLOOD portions of the transient so that the proper functioning of other aspects and models of the code will be suitably demonstrated.

2.0 WESTINGHOUSE 4-LOOP LOCA ANALYSES

Loss of Coolant Accident (LOCA) analyses of a typical 4-loop Westinghouse PWR were completed[VIII.2-1]. The analysis approach was to obtain a RETRAN best estimate blowdown analysis and perform sensitivity studies using the RETRAN model. Only the blowdown phase of the LOCA was considered.

For licensing analyses, the NRC specifies a number of conservative assumptions. Conversely, a significant effort can be required to determine appropriate best estimate assumptions for a particular reactor system. Since a lot of the documentation available for a particular plant is based on licensing requirements, and not on normal plant response to an accident, detailed knowledge of both control system and operator responses are necessary for best estimate analysis.

2.1 RETRAN Geometric Description

The RETRAN model developed for this LOCA analysis has 42 volumes, 69 junctions and 11 heat conductors. The volume numbers and nodalization scheme employed are depicted in Figure VIII.2-1. This model represents the reactor vessel and internals, primary coolant piping, coolant pumps, steam generators, accumulators and lines, pressurizer and surge line, and the containment.

The nodalization scheme employed was based primarily on the geometric boundary conditions, using engineering judgement. In general, regions within the fluid system in which hydraulic characteristics were similar were divided into distinct volumes. Where abrupt changes in flow area or thermal characteristics occur, junctions were inserted to separate the regions into separate volumes.

The four loops of the Westinghouse prototype are modeled as two asymmetrical loops. Three prototype loops are combined and designated as the intact loop. The fourth is modeled as a single (broken) loop.

Twelve volumes were used to model the reactor vessel and internals. The inlet annulus was modeled as two volumes. By representing the inlet annulus in this fashion, in which three-quarters of the volume is in communication with the intact loop cold leg and one-quarter communicates directly with the broken loop cold leg, the model represents the proper time delay for mass and enthalpy

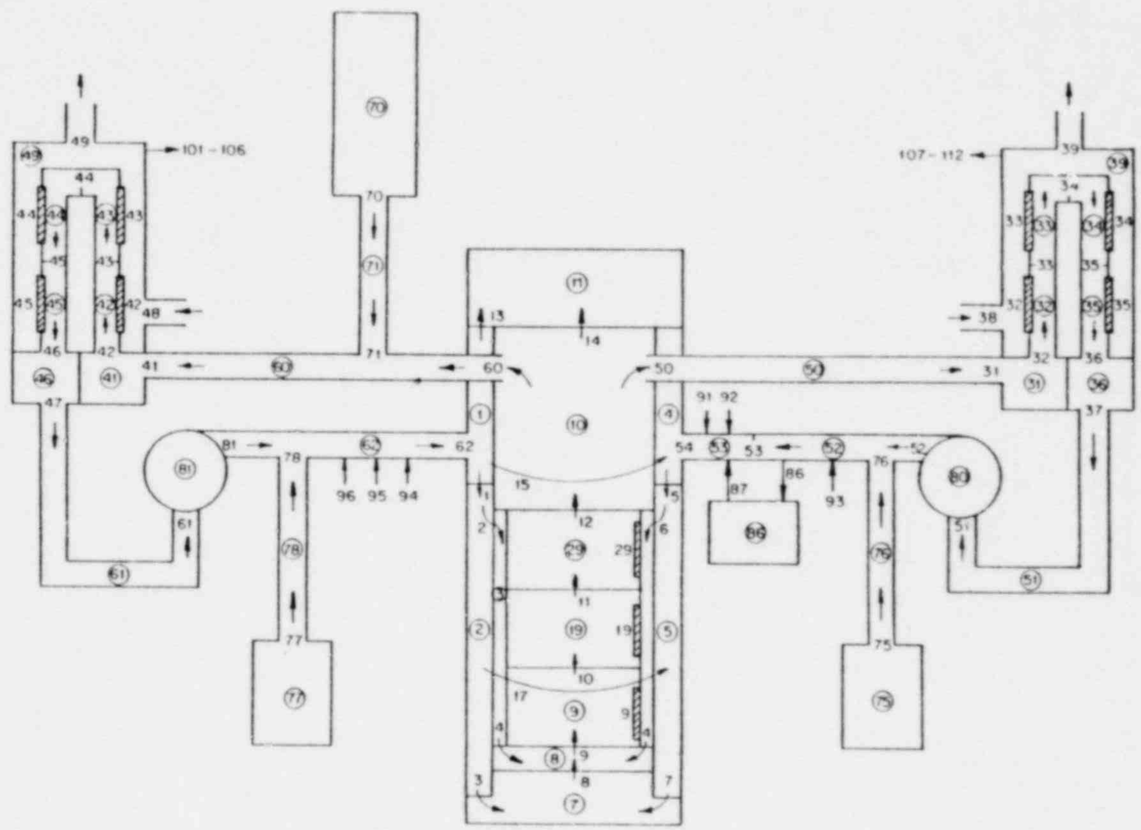

FIGURE VIII.2-1 RETRAN Model for 4-Loop Westinghouse Plant

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transport from the intact loop to the broken loop. Similarly, the lower downcomer is modeled as two volumes. The downcomer bypass is represented as a single volume with both of the volumes of the downcomer having junctions allowing flow into the bypass.

The lower plenum is modeled as a two volume region. Two volumes rather than one are modeled because the hydraulics associated with the downcomer and the downcomer bypass differ significantly. The two volume lower plenum helps to model this hydraulic difference more accurately. The reactor core is modeled as three geometrically identical volumes, although the power distribution differs.

Each of the two primary coolant loops contain three volumes representing piping (hot leg, pump suction leg and cold leg), a six volume steam generator of which four are active heat transfer volumes, and a pump. The volumes, flows, flow areas, etc., have been adjusted on the intact loop to reflect the lumping of three loops into one. Three accumulators have been combined into one volume and connect to the intact loop cold leg through the volume representing the accumulator surge line. The broken loop accumulator line connects to the broken loop cold leg between the pump and the break location. This results in the flow from the broken loop accumulator discharging through the break to the containment. The pressurizer is located on the intact loop.

Each steam generator secondary side was modeled as a single volume. The flows to and from these volumes were described by use of a positive fill junction and a negative fill junction for each steam generator. To account for the recirculation within the steam generator, the flow area was reduced by a factor of four from the prototype. The relief and safety valves of the steam generator secondary side are modeled as negative fill junctions controlled by pressure setpoints.

Because the containment pressure during the blowdown is not significant for primary system response, the containment is modeled as an extremely large volume resulting in an essentially constant back pressure.

Eleven heat conductors are employed in the current model. Of these, three in the vessel represent the fuel rods and four are used in each steam generator. The fraction of core power produced in the three core volumes is 29.79% for the bottom, 44.65% for the middle, and 25.56% for the top. The steam generators are modeled to remove 25% and 75% of the total power for the broken loop and the intact loop respectively.

2.2 LOCA Assumptions and Modeling Options

The design basis accident assumed for this reactor is a full double-ended cold leg break. Information available for this system indicated that the status of plant off-site power would be important to the best estimate analysis. The probability that there would be a coincident loss-of-offsite power was considered to be high for a break of this magnitude. Consequently, the best estimate analysis performed by Energy Incorporated assumes the loss of plant offsite power at the time of the break. This implies that the loss of electrical load would cause the reactor to scram, and the primary coolant pumps and the safety injection system to be tripped at the time of the break.

In addition to the best estimate analysis, sensitivity studies were performed to examine the effects of:

- Loss of offsite power
- ECCS injection
- Time of reactor scram.

2.2.1 Momentum Equation

In general, the single stream compressible flow form of the momentum equation (MVMIX = 0) and the RETRAN modified Baroczy two-phase friction multiplier (JTPMV = 0) are used at each junction. The exception occurs at volumes that contain more than two junctions. The single-stream compressible form of the momentum equation should be used at the junctions that form a one-dimensional primary flow path. At other junctions the incompressible momentum equation with no momentum flux (MVMIX = 3) should be used as for example, tee junctions. For the special case of multi-dimensional volumes, a large volume flow area was used to minimize the three dimensional momentum flux effects.

2.2.2 Break Model

The critical flow models usually limit the break flow and greatly affect the system's blowdown behavior. The best estimate characterization of the critical flow at break junctions is the Extended Henry model for subcooled flow and the Moody model for saturated flow. In this break model the break junction flow area is controlled by the junction quality. This representation of break flow area in effect models a variable contraction coefficient for the critical flow correlations. The normalized break flow area for this model is defined as a function of quality as follows:

$A = 1.0$, for $X \leq 0.001$

$A = 0.6$, for $X > 0.020$

where X is the break junction quality and A is the normalized flow area. Between a quality of 0.001 and 0.020, the break flow area is calculated by linear interpolation between 1.0 and 0.6.

2.2.3 Pump Model

The pump characteristic homologous curves were input as Westinghouse pump NS=5200. Motor torque is supplied as a tabular function of torque vs. speed as referenced from WREM. The pump head multiplier curve, pump torque multiplier curve, two phase head difference curves and the two phase torque difference curves are based on the two-phase performance of the Semiscale pump. The characteristic curves for the combined pump in the intact loop are the same as for the one pump in the broken loop, with appropriate pump rated conditions (torque and flow) multiplied by 3, i.e., the number of pumps being combined.

2.2.4 Phase Separation

The assumption of homogeneous fluid conditions is considered to be inappropriate in such reactor components as the vessel upper dome, accumulators, pressurizer, containment and steam generator secondary side. Therefore, a bubble rise model is available as a non-homogenity correction in RETRAN. In general, the bubble rise model is used with a density gradient of 0.8 and a bubble velocity of

0 054

3.0 ft/sec for the volumes with well-defined phase separation. But in the containment volume, a bubble velocity of 10^6 ft/sec is used for instantaneous phase separation.

2.2.5 Trip Controls

An extensive set of trip controls is necessary when modeling a best estimate blowdown for a PWR. Trips are used to model steady-state operations, break openings, the action of the reactor protective system (such as reactor scram, accumulator actuation, centrifugal pump charging injection, high head safety injection, and low head safety injection), steam generator secondary side feedwater flow, and safety relief valves.

2.2.6 Safety Injection System

The Safety Injection System (SIS) was modeled by using fill tables of flow vs. pressure for the appropriate components. However, the fills were initiated by various combinations of control signals which can delay the actual beginning of fill flow. No credit was taken for the boron content of the injection water.

2.2.7 Steam Generator Secondary Side

The operation of the steam generator relief and safety valves, the main steam isolation valve, the feedwater valve, and the turbine following a break of this size was difficult to ascertain.

A review of available information indicated feedwater flow is terminated 5.0 seconds and the steam flow in 0.5 seconds after the feedwater isolation signal is activated. The relief and safety valves were modeled to relieve excess pressure, although the rapid depressurization of the primary side following a large break reduces the probability of high steam generator secondary side pressure for this plant.

2.3 Results of Analyses

The typical 4-loop Westinghouse plant operating conditions were used to initialize the blowdown analyses. Steady-state initialization was used, based on the initial input conditions.

2.3.1 The EI Best Estimate Blowdown Analysis

The EI Best Estimate Blowdown Analysis assumes a simultaneous loss of plant offsite power with the double-ended cold leg guillotine break. In the event of a loss of offsite power, the loss of electrical load causes the reactor, the primary coolant pumps and the Safety Injection System (SIS has a 25.0 second delay) to be tripped at the beginning of the accident. The Extended Henry model with a 1.0 multiplier for subcooled blowdown and the Moody model with a 0.6 multiplier for saturated blowdown were used to model the critical flow in the break. At the time of the break, the feedwater isolation signal is activated. The feedwater flow is terminated 5.0 seconds after the signal and the steam flow is terminated 0.5 seconds after the signal. The steam generator relief and safety valves are modeled to relieve possible excess pressure. The accumulators are actuated when reactor coolant system pressure drops to 600 psia. No credit is taken in the analysis for the boron content of the ECCS water.

The analysis showed a rapid decompression of the reactor cooling system to the saturation point, after which the pressure decreases more slowly to the ambient conditions (Figure VIII.2-2). Since the pumps are tripped at the time of the break, substantial flow reversal is seen in the reactor core (Figure VIII.2-3). The effects of the flow reversal are reflected in the resulting cladding surface temperature of the center core region (Figure VIII.2-4) and the corresponding core quality (Figure VIII.2-5), which reach peaks when this occurs. The SIS is activated at the time of break, but due to the 25.0 second delay in the system, the SIS has little effect on the blowdown phase of the LOCA. The response of the accumulators is presented in Figures VIII.2-6 and VIII.2-7.

2.3.2 Sensitivity Study With and Without Loss of Offsite Power

In the best estimate analysis without loss of offsite power, the reactor scram, the primary coolant pumps and the SIS are tripped on system parameter set points and not on the loss of load at the time of break. Without power loss, the SIS no longer has the 25.0 second delay and the primary coolant pumps are not shut down immediately. Otherwise, the best estimate model without loss of offsite power is identical to the EI Best Estimate Model. A comparison of the time sequence of major blowdown events with and without loss of offsite power is shown in Table VIII.2-1.

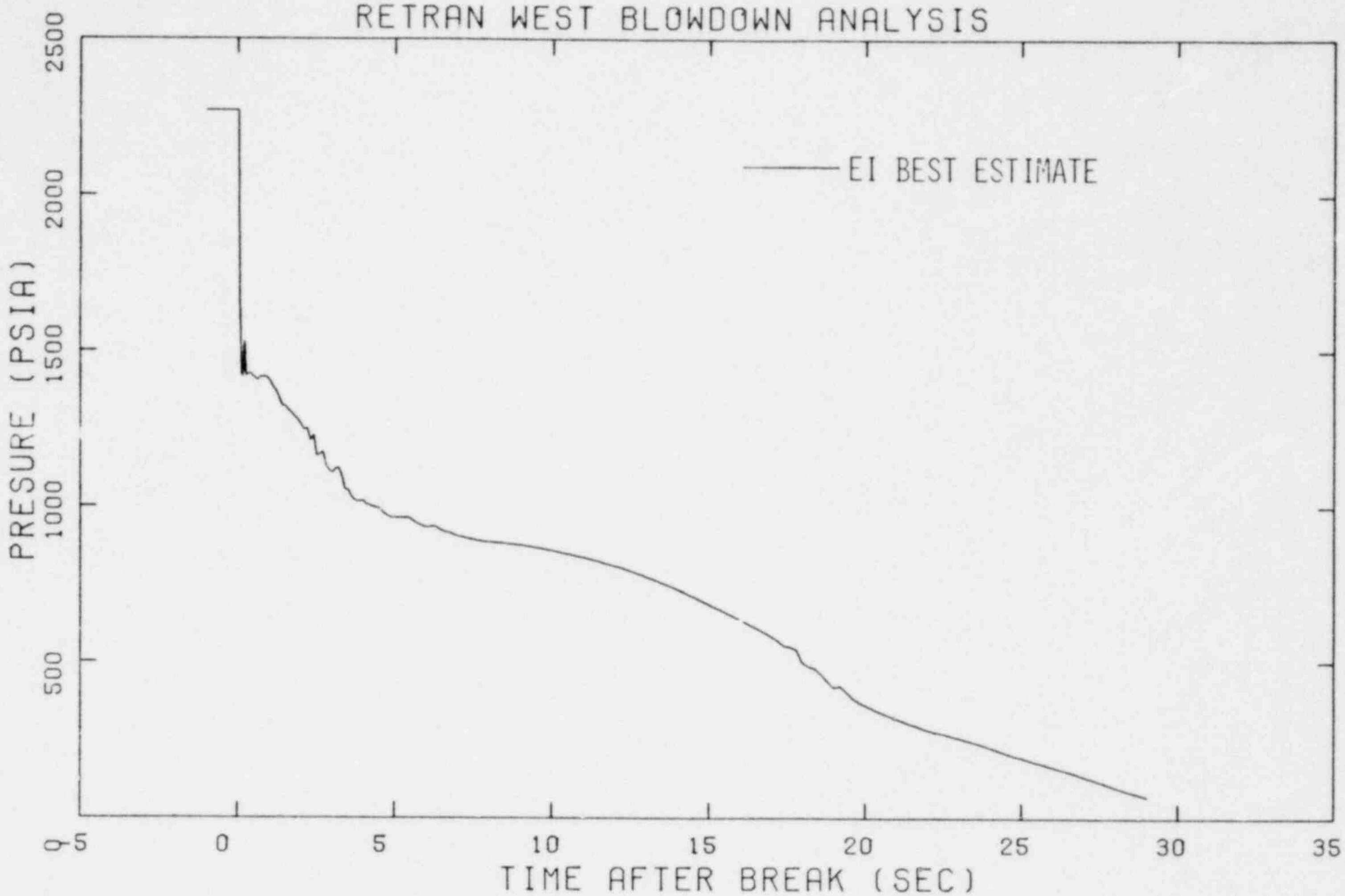

FIGURE VIII.2-2 Upper Plenum Pressure

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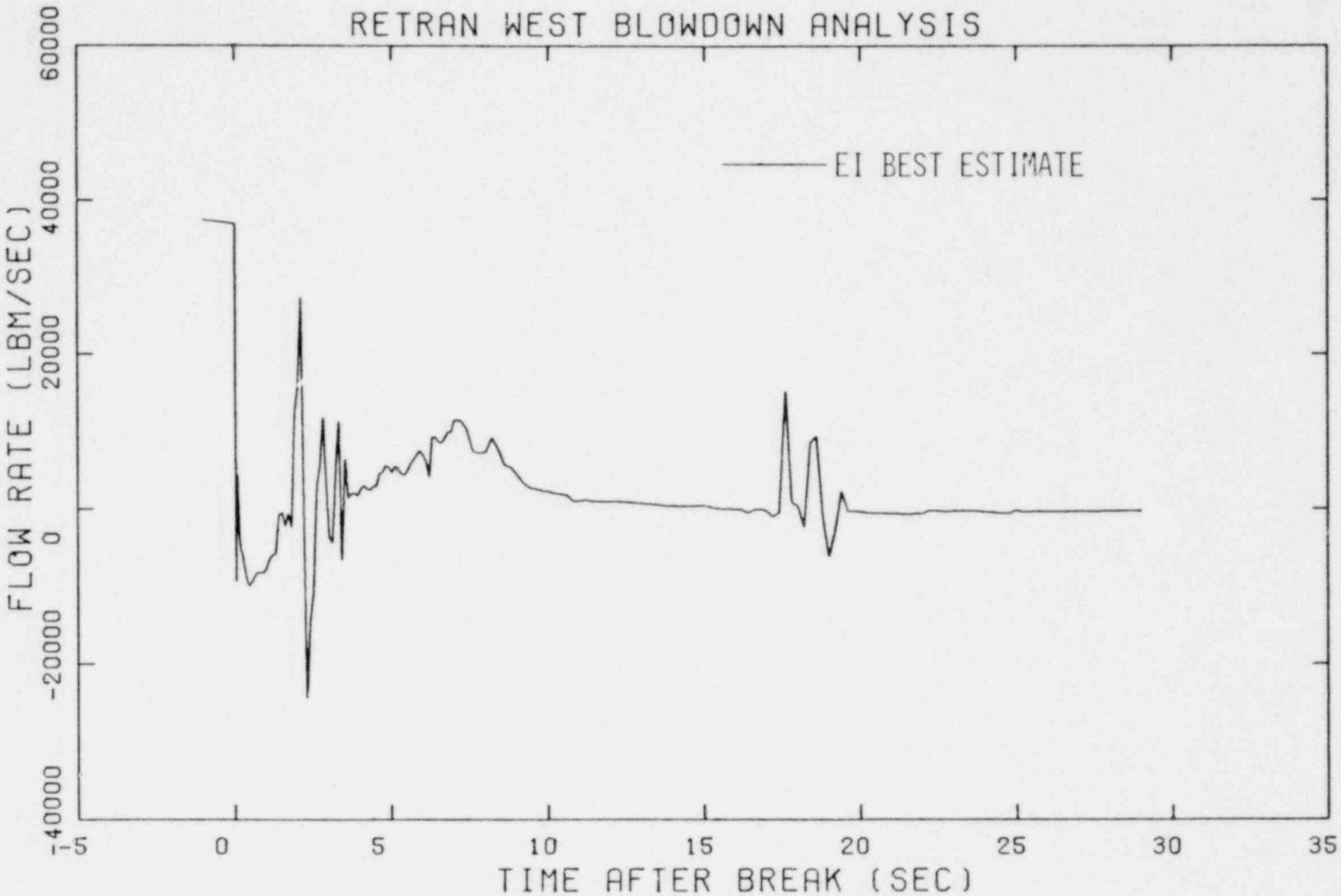

FIGURE VIII.2-3 Core Inlet Flow Rate.

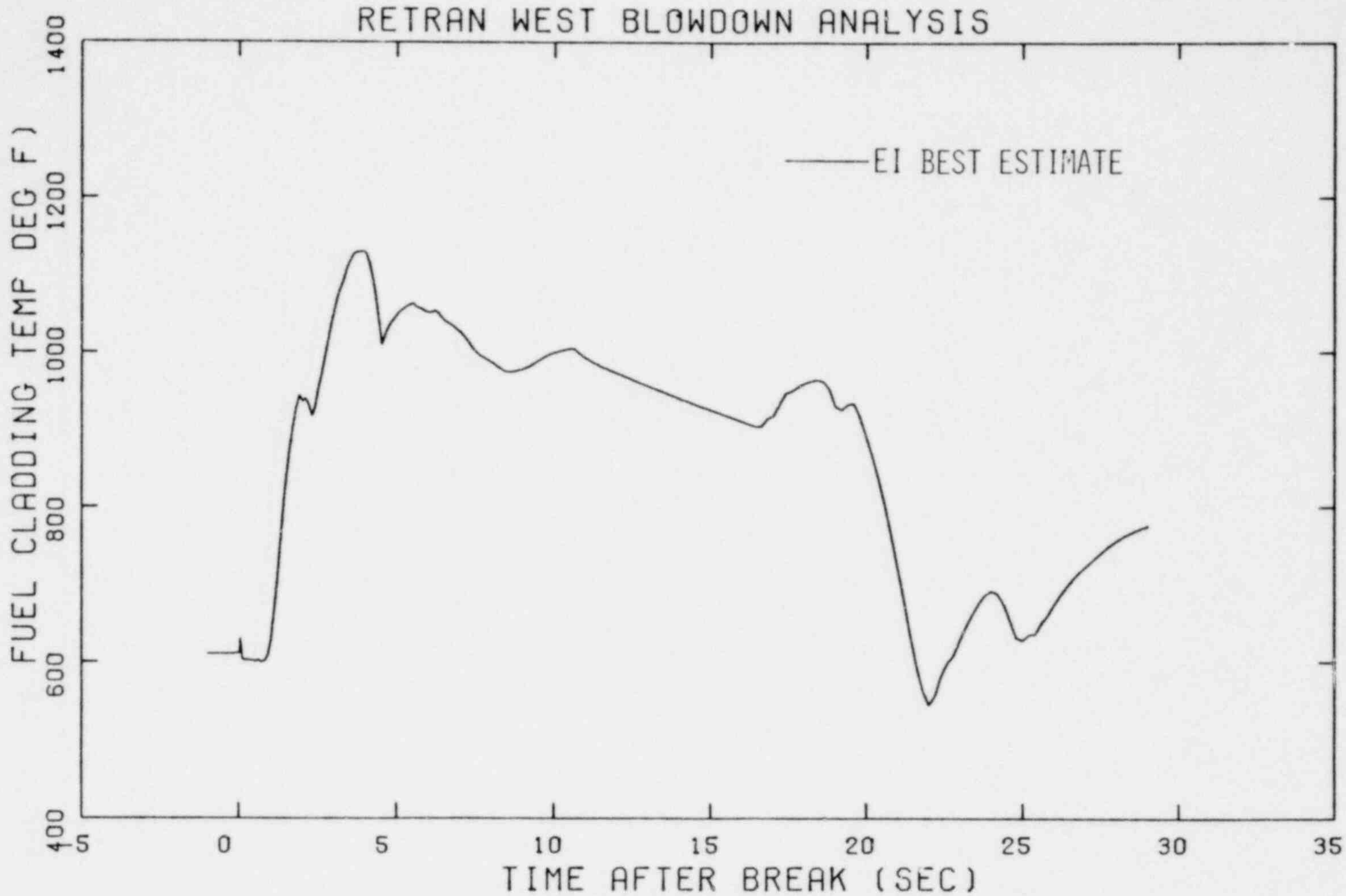

FIGURE VIII.2-4 Fuel Cladding Temperature

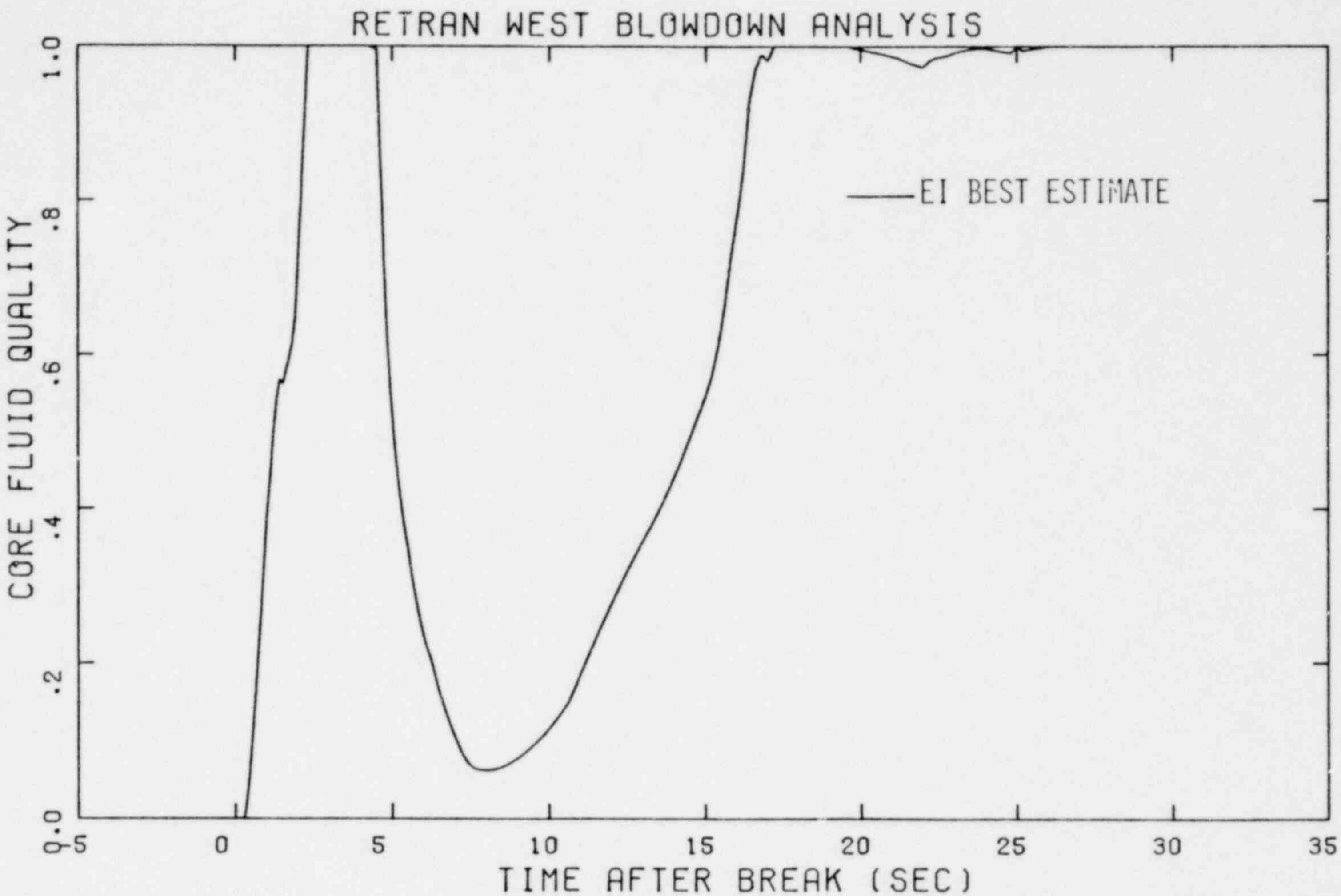

FIGURE VIII.2-5 Core Quality

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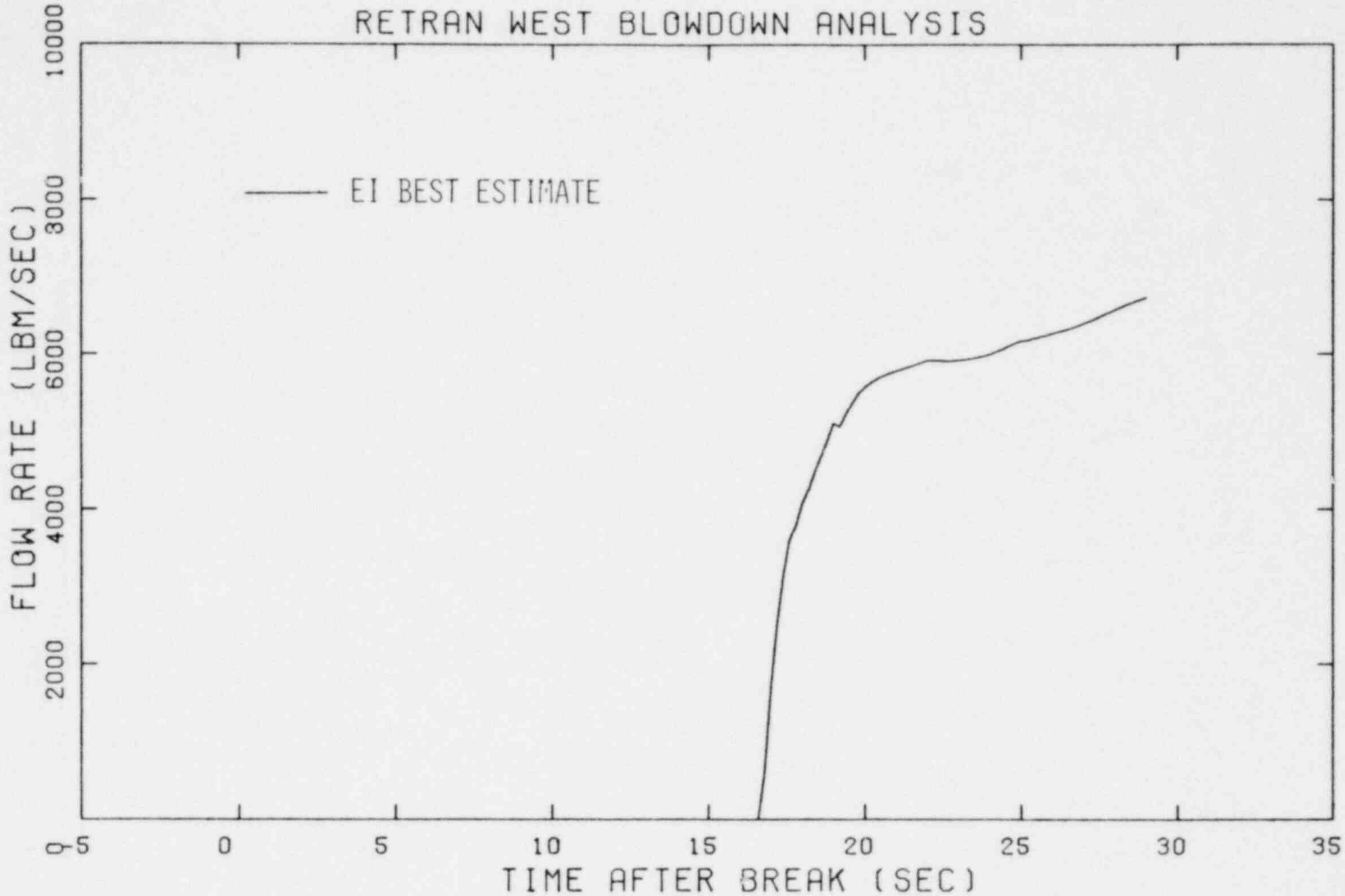

FIGURE VIII.2-6 Intact Loop Accumulator Flow Rate

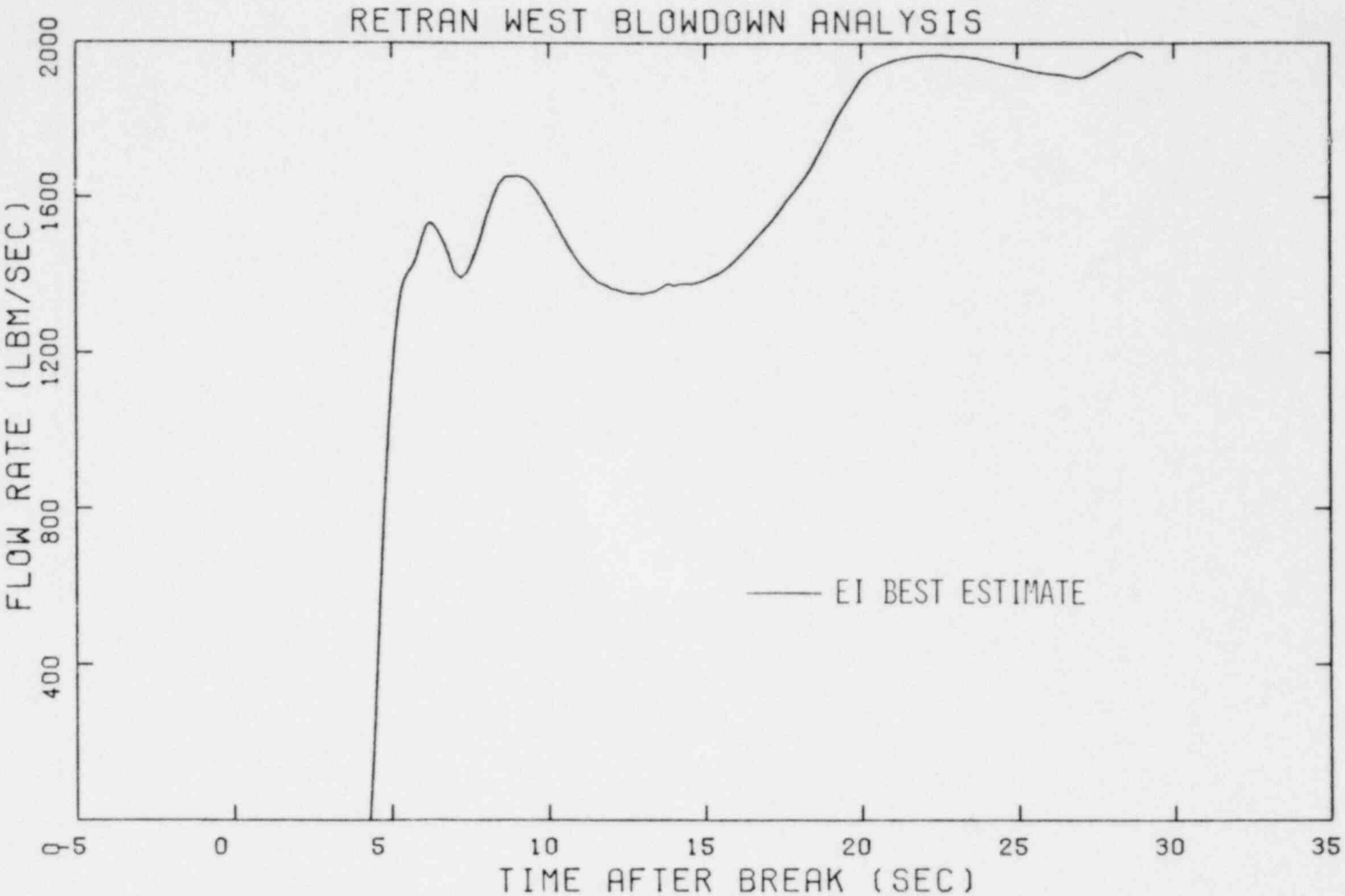

FIGURE VIII.2-7 Broken Loop Accumulator Flow Rate

TABLE VIII.2-1

MAJOR BLOWDOWN EVENTS WITH AND WITHOUT OFFSITE POWER

Event	Time After Break (seconds) Offsite Power	No Offsite Power
Start	0.0	0.0
Reactor Scram	4.2	0.0
Accumulator Injection, BL	4.6	4.4
Accumulator Injection, IL	16.8	16.8
Pressurizer Empty	7.6	7.6
Pump Trip	7.0	0.0
Safety Injection Signal	7.0	0.0
Centrifugal Pumps Charging	19.2	25.0
HPIS Charging	19.2	25.0
LPIS Charging	25.2	*
End of Blowdown	~30	~30

*Occurs beyond the time frame of this analysis.

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The analysis shows that without the loss of offsite power, the primary coolant pumps are in operation up to 7.0 seconds into the transient at which time they trip on low primary coolant flow rate in the broken leg. Figure VIII.2-8 shows the difference in core inlet flow rate due to the difference in the operation of the primary coolant pumps. During the first 7 seconds after the break, the core flow rate for the analysis without loss of offsite power is greater than for the case with loss of offsite power. This results in significantly different responses of the fuel clad temperature and the core quality for the two analyses, as shown in Figures VIII.2-9 and VIII.2-10.

2.3.3 Sensitivity Study of Best Estimate Analysis With and Without ECCS

An ECCS sensitivity analysis was conducted using the EI Best Estimate model. A comparison of the time sequence of major blowdown events with and without ECCS is given in Table VIII.2-2.

The response of the total break flow is compared with the EI Best Estimate prediction in Figure VIII.2-11. The effect of the additional water in the system when the accumulators are available can be readily seen. It is evident that the end of blowdown is delayed by the ECCS. The major contrast between the two analyses is in the prediction of the fuel cladding temperature. Figure VIII.2-12 shows that the intact loop accumulators have a significant effect on minimizing the fuel clad temperature during the RCS blowdown after the system pressure falls below 600 psia.

2.3.4 Sensitivity Study of Best Estimate Analysis With and Without a Delay in the Reactor Scram Trip

An analysis was conducted in which the reactor scram was tripped on system parameter set points, instead of at the time of the break as assumed for the loss of offsite power. Figure VIII.2-13 shows the difference in fuel cladding temperature. The delay in the scram trip has a small effect on the peak cladding temperature.

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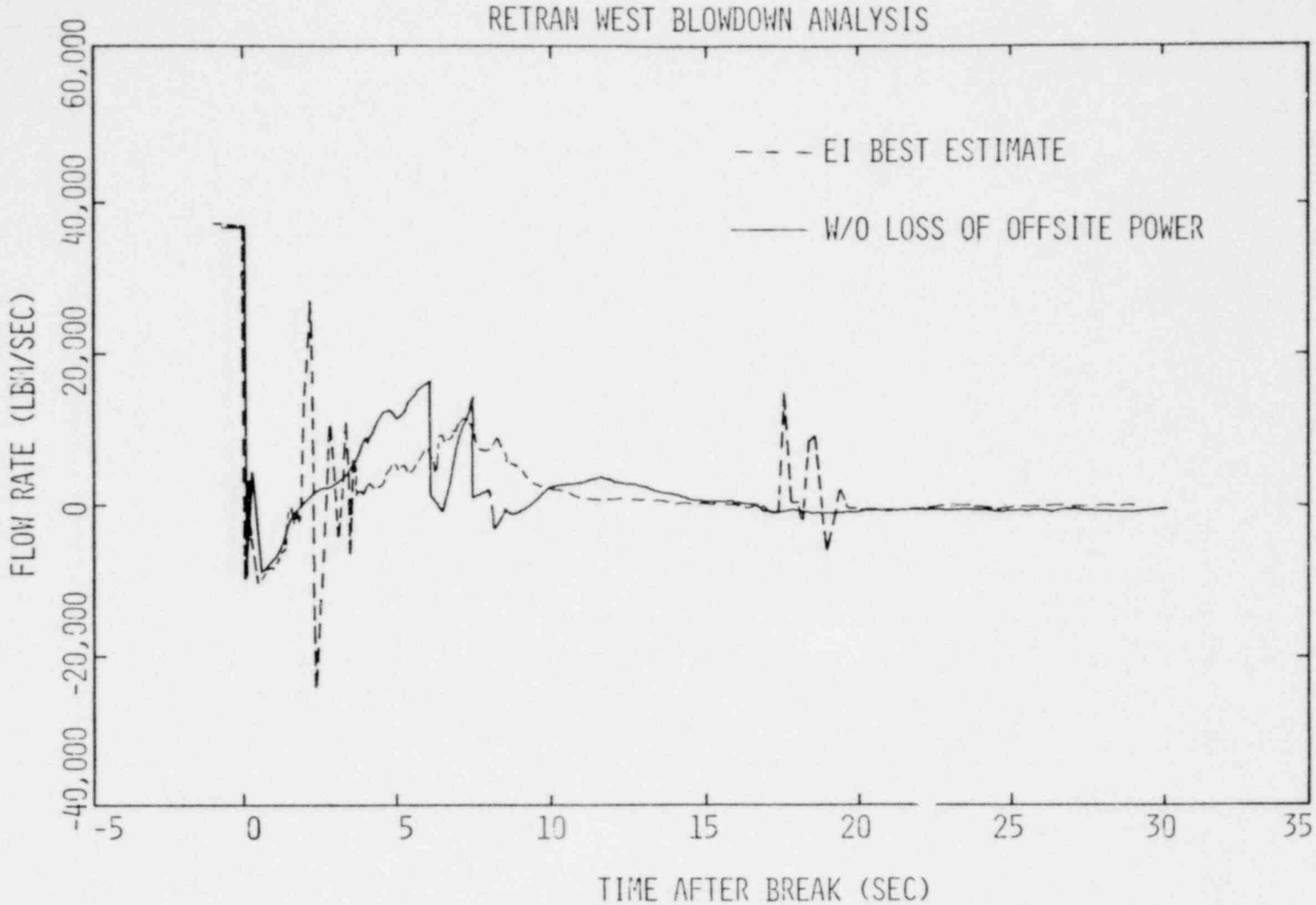

FIGURE VIII.2-8 Core Inlet Flow Rate

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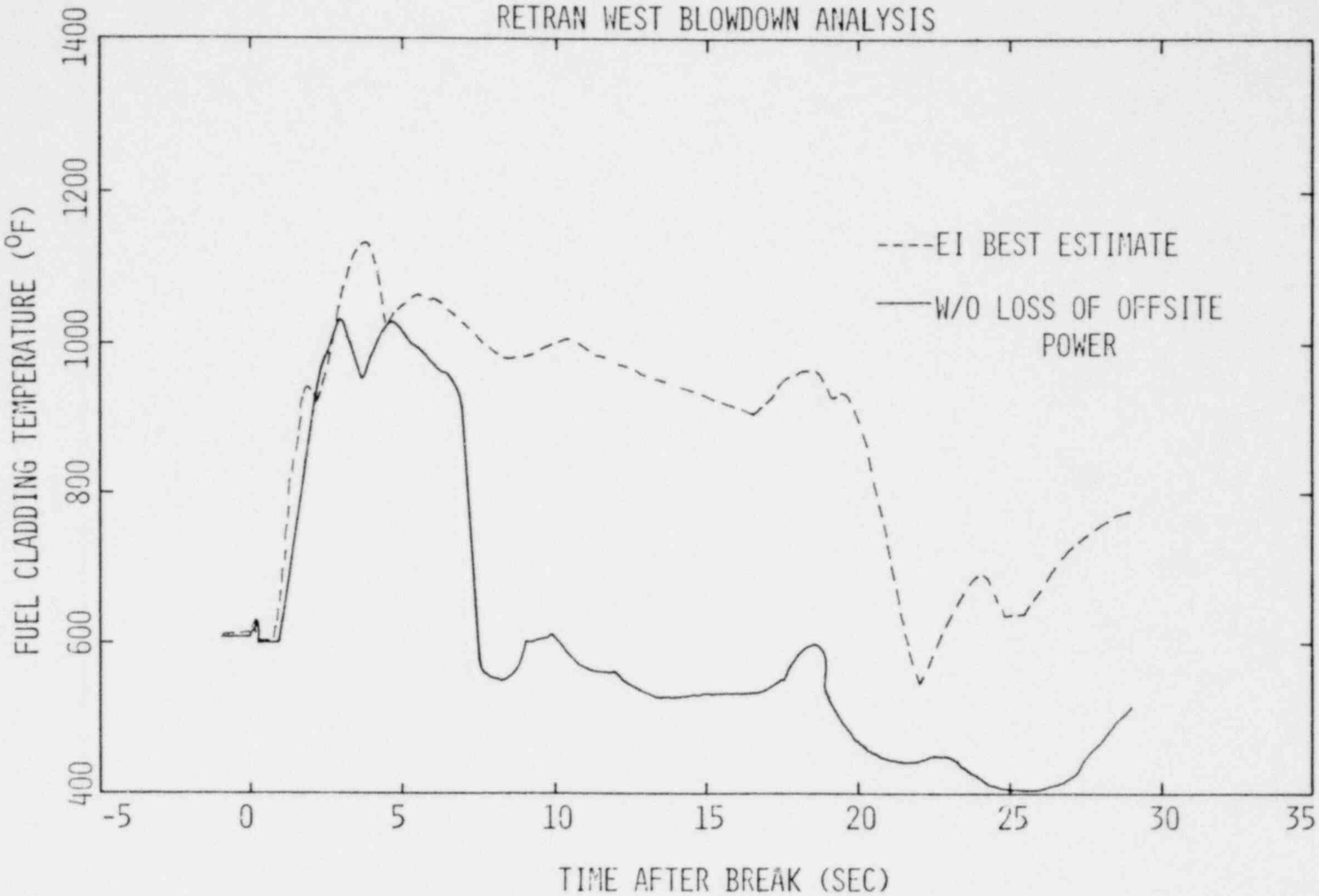

FIGURE VIII.2-9 Fuel Cladding Temperature

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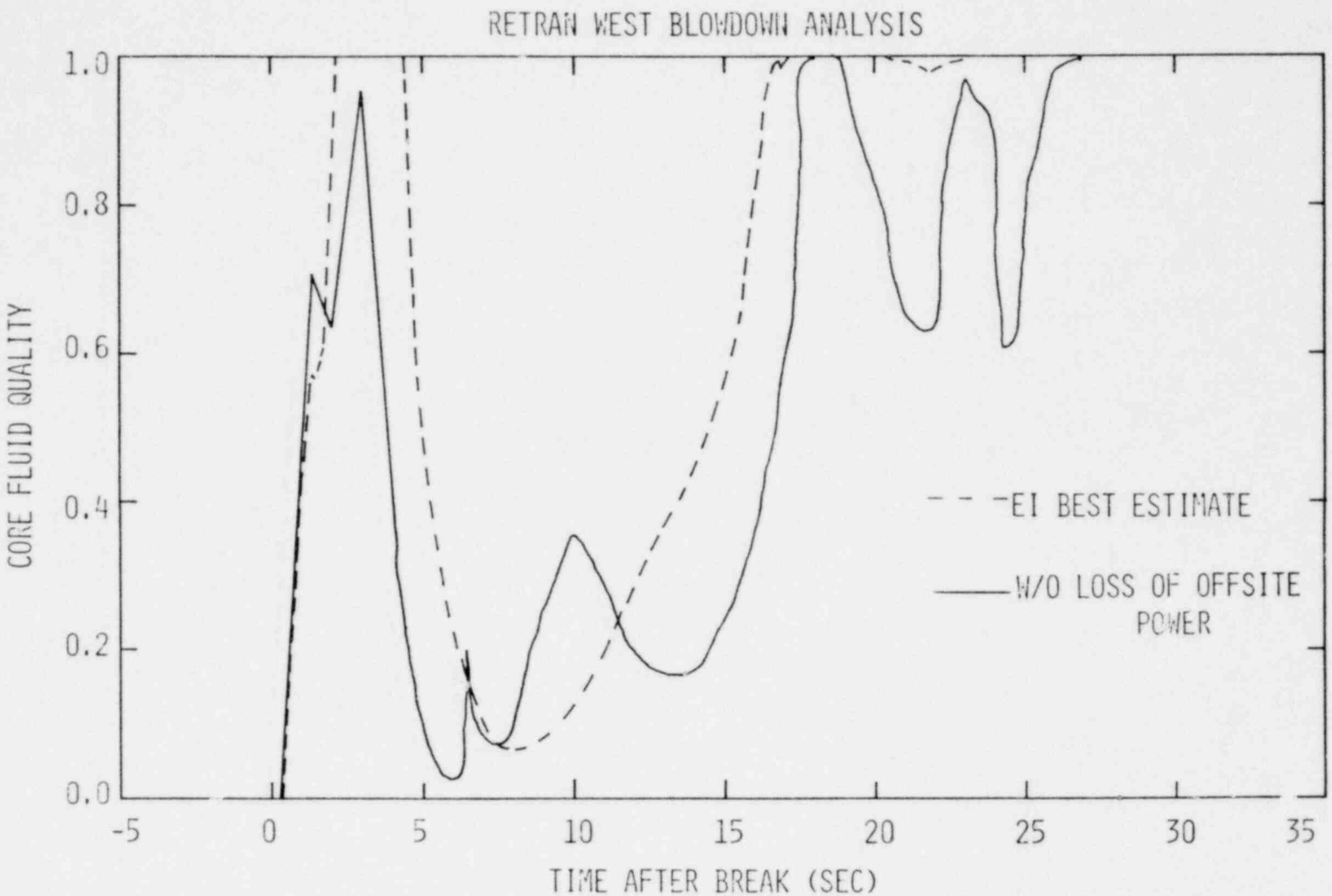

FIGURE VIII.2-10 Core Quality

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TABLE VIII.2-2

MAJOR BLOWDOWN EVENTS WITH AND WITHOUT ECCS

Event	Time After Break (seconds) No ECCS	ECCS
Start	0.0	0.0
Reactor Scram	0.0	0.0
Accumulator Injection, BL	None	4.4
Accumulator Injection, IL	None	16.8
Pressurizer Empty	7.6	7.6
Pump Trip	0.0	0.0
Safety Injection Signal	0.0	0.0
Centrifugal Pumps Charging	None	25.0
HPIS Charging	None	25.0
LPIS Charging	None	*
End of Blowdown	~30	~30

*Occurs out of time frame of this analysis.

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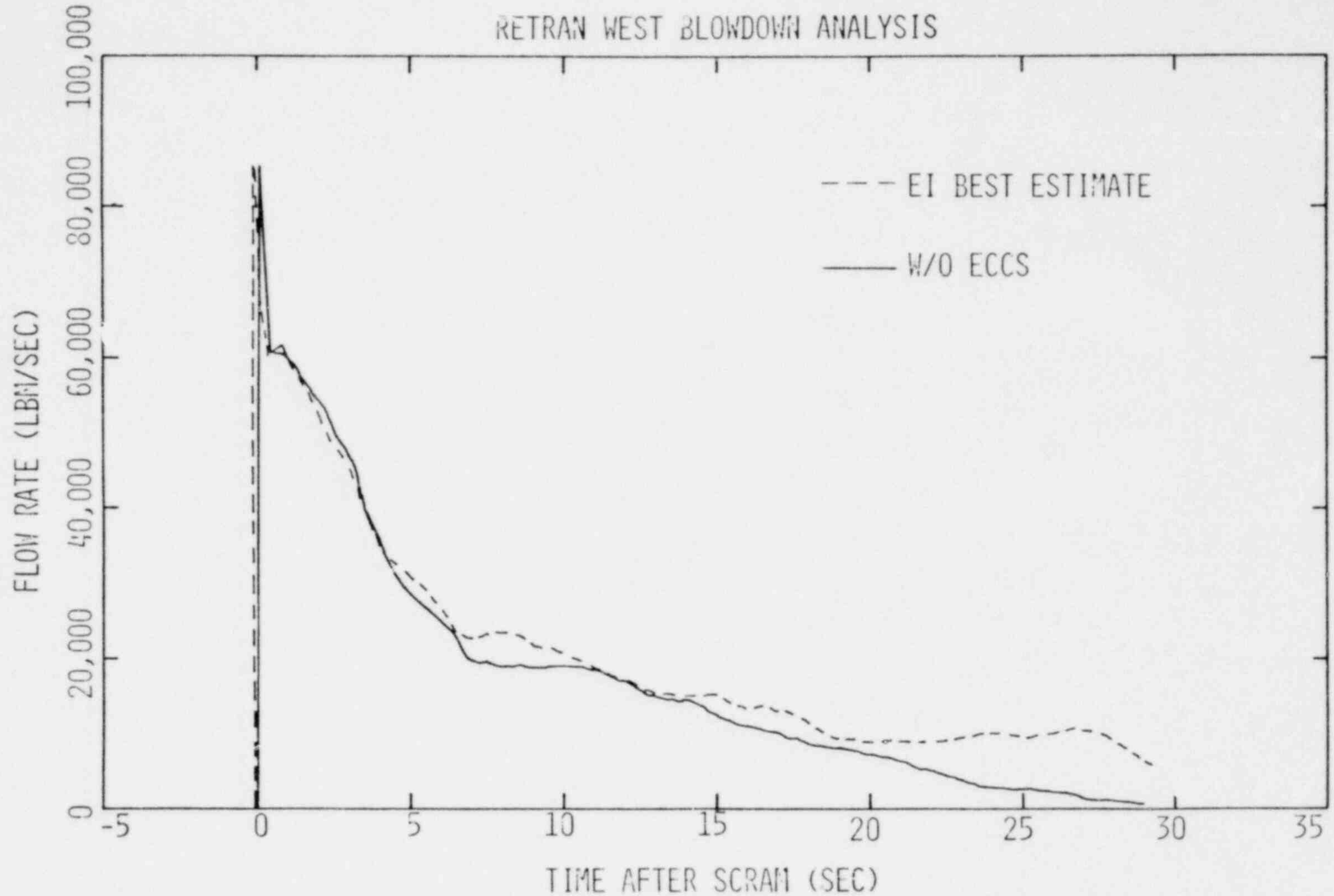

FIGURE VIII.2-11 Total Break Flow

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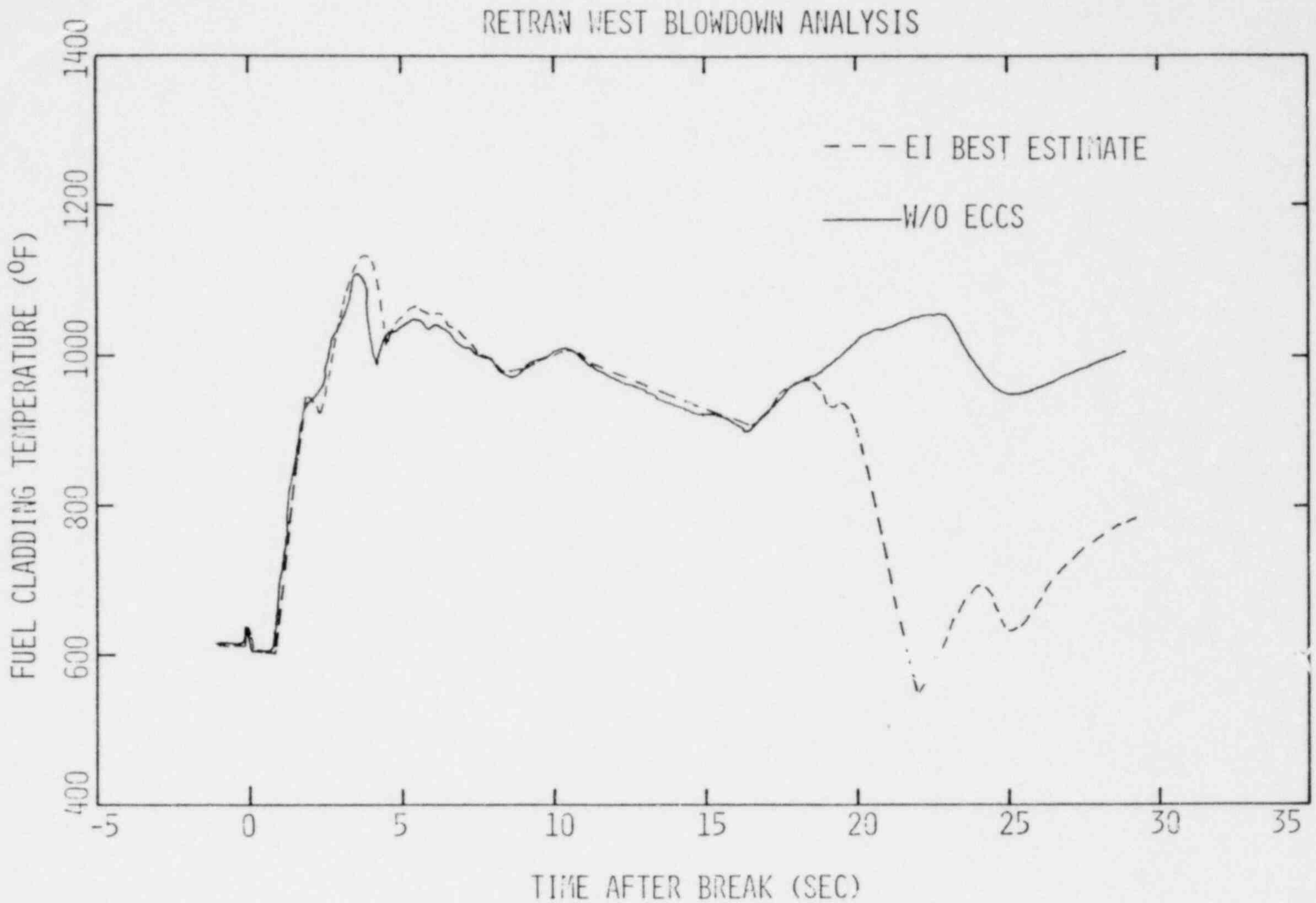

FIGURE VIII.2-12 Fuel Cladding Temperature

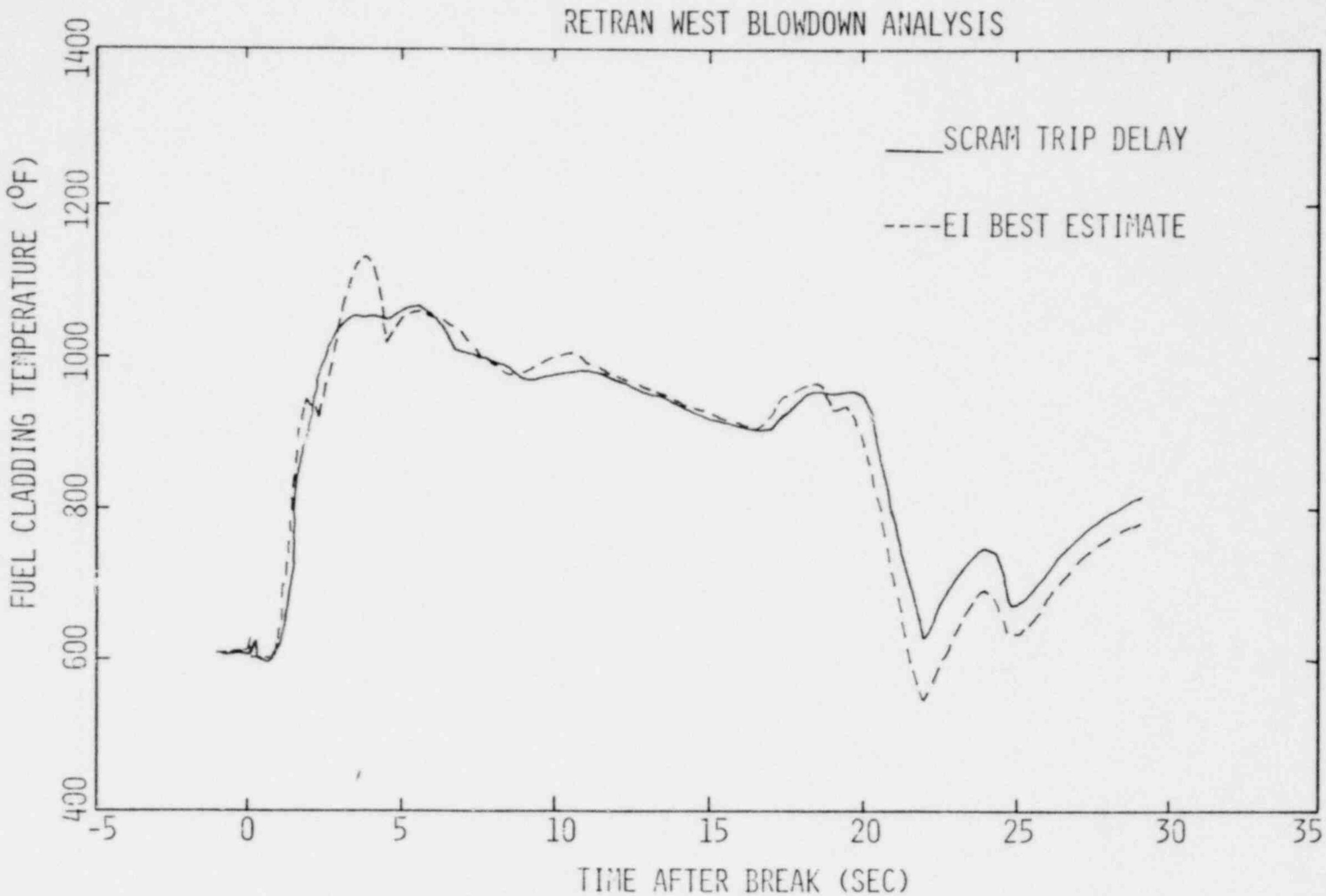

FIGURE VIII.2-13 Fuel Cladding Temperature

2.3.5 Sensitivity Study of the Best Estimate Analysis With and Without SIS

The analysis presented in Section 2.3.3 was conducted with the accumulators and the SIS unavailable during the blowdown. A sensitivity study in which the accumulators are still unavailable, but the SIS is operable, was also done. The differences with and without the SIS operable are negligible during the blowdown portion of the LOCA. The 25.0 second delay on the SIS makes its impact felt only at the end of blowdown.

2.3.6 Sensitivity Study of Best Estimate Analysis Without Loss of Offsite Power, With and Without the SIS 25.0 Second Delay

A sensitivity analysis was conducted of the Best Estimate Analysis without loss of offsite power (Section 2.3.2) in which the response of the SIS was delayed 25.0 seconds instead of allowed to be immediate as in Section 2.3.2. The prediction of selected system parameters during the RCS blowdown with a 25.0 second delay on the SIS showed that the SIS has little effect on the results of the blowdown. This is evident in Figure VIII.2-14.

2.4 Conclusions

The results show that RETRAN is capable of modeling the blowdown portion of the LOCA for a Westinghouse 4-loop plant. The sensitivity studies showed the operation of the pumps has a significant effect on the thermal-hydraulic response of the core. The accumulators also have a significant effect on the thermal response of the core during blowdown. Finally, the reactor scram and the SIS have little effect on the system response during the blowdown portion of the LOCA for this reactor system.

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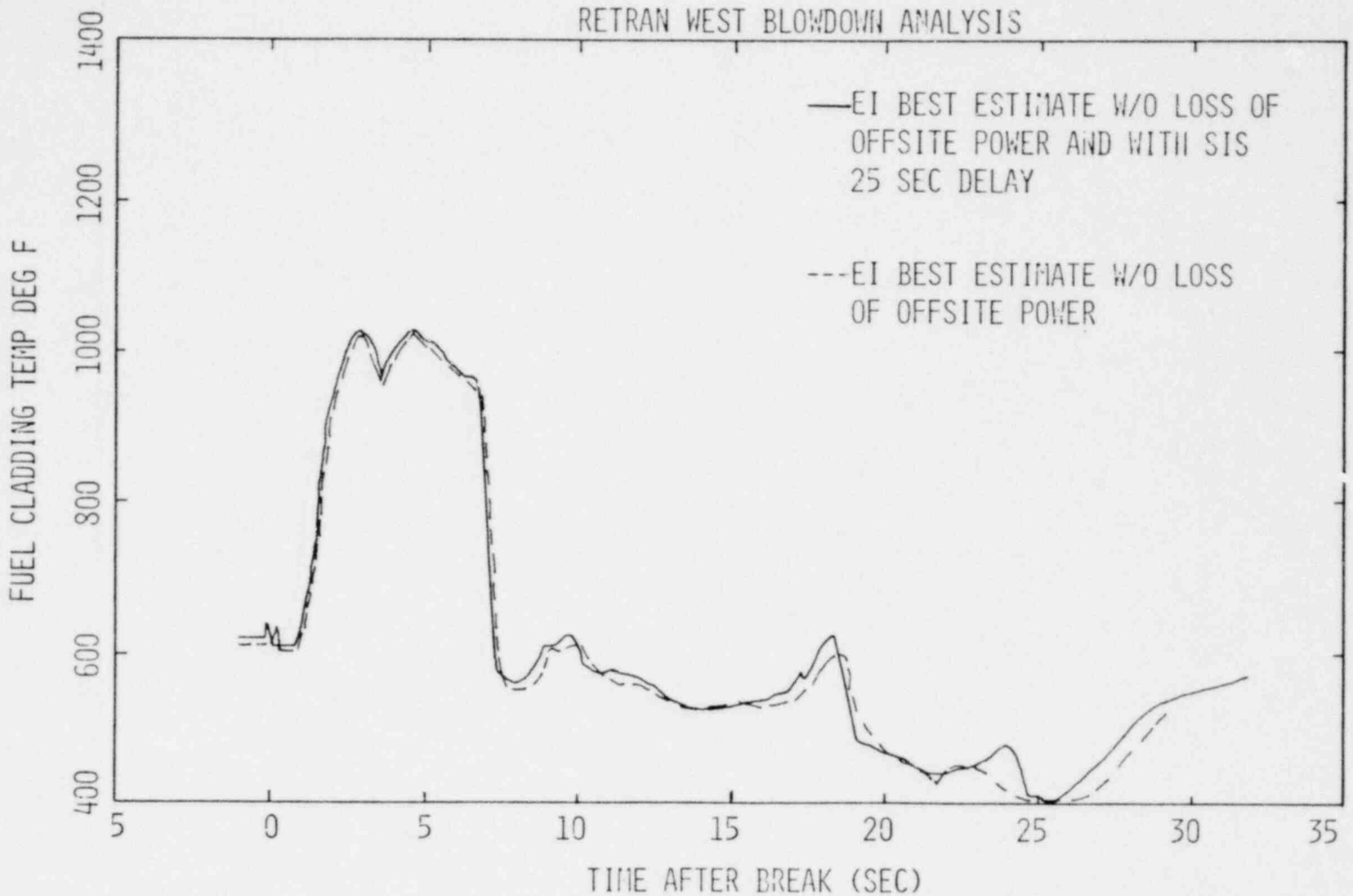

FIGURE VIII.2-14 Fuel Cladding Temperature

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IX. USER EXPERIENCE

IX. USER EXPERIENCE

The purpose of this section is to identify recommended modeling approaches for the analyses discussed in other sections of this document and to provide general comments for future users of RETRAN. These recommendations to a large degree, have been developed by evaluating the results of a specific transient analysis with experimental data and with calculations performed by a number of investigators. General comments are the result of various studies and observations which have been made during this phase of the RETRAN project.

The comments and recommendations are organized into three major subject areas; general, BWR and PWR. In the case of the BWR operational transients, RETRAN calculations, performed by more than one utility, have been completed only for the turbine trip. Three PWR transients were modeled by more than one investigator, with test data comparisons included for the pump coastdown analyses. Other transients modeled by three utilities were the uncontrolled rod withdrawal and the loss of reactor coolant flow. The pump coastdown analyses produced good agreement between the RETRAN calculations and test data, and thus provide a measure of confidence in the loss of flow analyses.

Some sensitivity studies were performed for those analyses which could not be directly compared to data, however additional analyses are desirable before modeling recommendations can be made for specific transients. Thus, modeling guidelines presented in the following sections are made for individual items (e.g., jet pump, separator, steam generator) rather than specific transients. The techniques and approaches discussed identify areas to be considered when modeling certain components or when using certain code options.

1.0 GENERAL

1.1 Errors

A practical lesson for all users was noted by Ansari [IV.2-3] (Section IV.3.2), where answers from a certain analysis appeared to be generally correct and of proper magnitude but were in fact in error. The lesson that should be recognized is that in any large code, errors can exist which are very subtle. This is particularly true when the code is being used in a new or unconventional application. Where models have been tailored for specific purposes, their use outside of the intended range should always be studied prior to accepting the results. Hand calculations are an important technique to quickly check the results and add a large increment of confidence in the work for the individual engineer.

1.2 FSAR Comparisons

During the course of the qualification work, many cases were analyzed and compared with FSAR predictions. In general, the results of the PWR uncontrolled rod withdrawal and steam line break transients predicted by RETRAN were different from those in the FSAR. Comparisons with FSAR predictions are less meaningful than comparisons to experimental data. In general, all the assumptions made in the FSAR calculations are not known and conservative models may have been used. The fact that a RETRAN value does not exactly agree with an FSAR result does not mean that either is in error. These comparisons should be viewed qualitively. More weight should be given to the comparisons with actual data. Those transients for which data are available to evaluate the RETRAN analyses are much more valuable for code qualification.

1.3 Causal Solution Options

The causal volume and causal conductor options for improving run time have been used to a slight degree by several Working Group participants and by EI in various problems. As a result of this work, it can be generally concluded that the causal volume option can be used with a reasonable level of confidence (with the recommended epsilon) for a wide variety of transients. The overall running time improvement is a strong function of the problem size and type. The study of the causal conductor option has not been as conclusive as is the case for the causal volume. The causal conductor option has produced improvements in run time on certain problems, while on others the opposite has been the case. Further, the answers in some cases have been of sufficient accuracy while others have been quite different. This option should be used cautiously and results examined for each type of transient and model on which it is exercised before engaging in general use.

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1.4 Kinetics

The RETRAN decay heat models assume an infinite operating time at the initial power level to establish the decay heat. Transients analyzed at the beginning of core life or from initial conditions not consistent with the stable long operation assumption may be affected by this decay heat assumption.

1.5 Controls

The general acceptability of the control block capabilities to predict responses in feedback situations were demonstrated in both PWR and BWR studies. Experience with the control blocks has shown that there is a tendency to establish algebraic loops of components in setting up certain types of models. An algebraic loop can result in system instabilities and is not rejected by the input survey of the variables. This problem may be alleviated by including a lag at any location within the system. In some cases there may be an analytical basis for sizing the time constant of the lag while in other cases it may be largely based on engineering judgement. In either case, the output of the system should be examined to insure that the lag has not overdamped the system.

Users have also noted that the time step should be smaller than the time constants and delays of the associated control blocks. Automatic time step controls are not based on the characteristics of the control blocks being used.

1.6 Momentum Equation

The RETRAN Working Group members generally used only two forms of the momentum equation, the compressible form with momentum flux (MVMIX=0) and the incompressible form without momentum flux (MVMIX=3). The first option was used for the largest fraction of the junction representations, and was used successfully with only one known exception, that being a steam generator model which contained two volumes, a bundle volume and a steam dome volume. In this case, an anomalous flow response resulted from using an inordinately small bundle area. This was alleviated by changing to the incompressible momentum equation form without momentum flux (MVMIX=3). The MVMIX=3 form was used to represent the flow to or from the primary path to a tee or cross flow path such as the pressurizer surge line. No problems were noted with this approach.

The group generally treated multidimensional regions such as the downcomer, lower and upper plenum, by increasing the flow area to an extremely large value. This effectively eliminates the momentum flux term in this volume while preserving it in the adjacent volumes. This is contrasted with MVMIX=3 which eliminates the term from both sides of the junction. This approach has worked satisfactorily although it is significant only in extreme flow transients such as LOCA.

1.7 Critical Flow

The enthalpy used in the critical flow tables is the volume average thermodynamic enthalpy and is therefore not consistent with the table requirement of stagnation enthalpy. The significance of this difference is problem dependent and while it generally introduces an unimportant error, at least one member of the Working Group found it important in establishing the critical flow rate.

One report on the critical flow options available in RETRAN illustrated the reason caution must be used when specifying critical flow input. The only tabulated model derived for subcooled conditions is the extended Henry-Fauske model. However, the RETRAN input permits using the Moody or Henry-Fauske models in this region, even though they are based on saturation conditions. This is a carry-over from a long standing practice with RELAP4 users, in which these options are sometimes used with a critical flow multiplier to match analysis results with experimental break flow data. RETRAN users must select these options cautiously, and recognize that specifying the Henry-Fauske option does not guarantee that the extended tables will be used in the subcooled region.

1.8 Long Term Transients

In mild, long duration transients, large time steps are generally acceptable for the hydraulic, thermal, and kinetics solutions. However, there may be situations where a single parameter may vary with a much different time constant or in a non-linear manner. The user should be aware of this possibility and view the results accordingly. Further, the transport delay model can specifically cause problems when a large portion of the volume is swept out in each time step. Junction enthalpy variations should be examined critically to ensure proper behavior when using this option.

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1.9 Steady State Initialization

The RETRAN Working Group has found that the default values for steady state convergence are very tight and in many problems may not be attainable. The user should examine those volumes that do not converge to ensure that the non-convergence is not an indication of another problem. The user can then proceed by forcing the code to continue after a given number of iterations or by changing the convergence criteria. The first option results in a better steady state as all but the troublesome volumes achieve the default convergence. It is also recommended that the user run a null transient to test the steady state solution. It has been generally found that the larger the area adjustment in the steam generator the more difficult the convergence.

1.10 Inverse Inertia Effect

Due to the logic of the coding for the flow solution, junctions with critical flow rates sometimes exhibit the strange behavior of flow rate varying inversely with the junction inertia. That is, increasing the inertia causes the junction flow to increase. This problem is noted predominately in analyses which involve saturated steam blowdowns, although it has also been observed in small breaks. The best solution to the problem at this time is to run a parameter study on the inertia for the initial portion of the transient and check the results against the tabular values for critical flow.

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2.0 BWR USER EXPERIENCE

The modeling techniques and options used in developing a RETRAN model for BWR transient analysis are very dependent on the transient being evaluated. Different transients will require different nodalization schemes. However, based on a review of the verification work performed by various users of RETRAN, there are some basic approaches that appear acceptable in a majority of the cases analyzed. These approaches are summarized in the following paragraphs. The user is cautioned that these conclusions are based on a limited number of similar type analyses and that further studies may result in different assumptions.

2.1 Jet Pump Modeling

Combining the various jet pumps supplied by one recirculation loop into a single RETRAN volume will adequately model jet pump performance in the normal operating range. With a single volume, representing both mixing and diffuser sections, the volume flow area and equivalent diameter may need to be increased to achieve the proper M-N characteristics. In those transients where reverse flow is encountered in the jet pump, RETRAN may not correctly predict this flow. Additional analyses, along with further investigation of both the input options and plant model, are required for evaluation of this problem.

2.2 Steam Line Modeling

In pressure increase transients such as a turbine trip, momentum and inertial effects in the steam line need to be considered. Combining the steam lines together does not appear to affect the results. The number of volumes used to represent the steam line does, however, affect the transient. The cases analyzed in the verification effort ranged from four volumes to eight volumes and both appeared to give reasonable results for the turbine trip transient.

2.3 Separator Modeling

In the various transients analyzed, different nodalization schemes were used to model the separator region. Even though different nodalization schemes were used, there are some basic assumptions that are common to each model. In modeling the separator region, the bubble rise model is used. It appears that the assumption of instantaneous separation with no initial carry under, i.e., high bubble rise velocity, is the most representative of the actual separation process. Combining the standpipes, the separators, and the region surrounding the separators gives reasonable results for the turbine trip transient. This assumption does limit the accuracy of the water level prediction which is very important in other transients where the water level actuates various trips and is used as an input signal to controllers such as the feedwater controller. By dividing the region into more detailed areas, the water level response is improved and, in the limited studies performed, agreed reasonably well with experimental water level data.

The studies performed on the pressure increase transients all showed a considerable amount of sensitivity to the separator inertia. The general conclusion of these studies is that the separator inertia should be less than computed from geometric considerations alone. This is reasonable since the flow path is variable and depends on the quality of inlet fluid to the separator and the resultant water layer thickness on the separator wall. Therefore, inertia values used in the analysis should be based on experimental data for the average quality throughout the transient.

2.4 Lower Plenum Modeling

For the pressure increase transients, it appears that a single volume representation for the lower plenum results in acceptable results.

2.5 Core Modeling

All the studies performed indicate a high degree of sensitivity to the reactivity coefficients used. The RETRAN limitation of point kinetics may be inadequate for some transients that result in significant changes in the neutron flux distribution. To compensate for this limitation, the user is cautioned to evaluate the reactivity coefficients, core nodalization and weighting factors in great detail, and to perform sensitivity studies over the various ranges possible from a point kinetics standpoint. The effect of weighting factors based on different flux shapes expected during the transient should be carefully evaluated. The core nodalization needs more evaluation before a recommendation can be made in this area.

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It does appear that a single volume bypass region is adequate, provided the plant is designed such that the bypass is subcooled throughout the entire length. In some plants where vibration fixes have resulted in voiding in upper portions of the bypass, a more detailed nodalization of the bypass region may be required.

3.0 PWR USER EXPERIENCE

The models used in the study of PWR transients are strongly dependent on the transient under consideration. Some transients can be modeled with a single loop while a non-symmetric transient requires more than one loop to be modeled. In a like manner, specific models (or input values for these models) also may vary with the transient under investigation. In the following sections, suggestions on the use of specific models are presented, and where possible recommendations are made for applying these models to individual transients.

3.1 Pump Models

The results of the pump coastdown and loss of flow analyses performed for the various reactors all showed the same trend. Using the built in curves for a Westinghouse pump provide RETRAN results which agree reasonably with FSAR and test data during the initial (0 - 4 second) portion of the transient. After this time period, RETRAN begins to predict slightly higher flows than shown by the test data. The FSAR values show an even higher flow than predicted by RETRAN. Additional studies show that by increasing the torque input values or decreasing the inertia input values slightly, the RETRAN prediction can be tured to match the experimental data. However, this tuning should not be taken as the solution to the problem. The determination of the hydraulic torque for the pump as installed does entail a degree of uncertainty. The discrepancy could also be a result of differences in the homologous curves, particularly at low rotational speeds. Perhaps the form of the friction model is not sufficient to represent the hydrodynamic bearings and seal assembly as it slows down. The time response and accuracy of the data can also be questioned prior to the establishment of any firm conclusions. In any case, a conservative calculation for the coastdown may be obtained by an overestimate of the torque or an underestimate of the inertia. A best estimate coastdown would have to reflect study of previous pump coastdown data.

3.2 Steam Generator Modeling

In most of the studies on a U-tube type steam generator, the secondary system has been represented by a single volume. For transients in which the heat transfer between secondary and primary is an important consideration, and in which the secondary conditions are changing, the single volume representation may not be adequate for a realistic transient representation. This inadequacy is due to the single volume approach which results in an unrealistic heat transfer coefficient. This is not a problem if the constant heat transfer coefficient model is being used.

When using a single volume secondary, a most appropriate heat transfer coefficient can be accomplished by reducing the secondary volume flow area by the recirculation ratio. This reduction in flow area increases the volumetric flow rate to account for recirculation flow. However, this change can cause some problems in predicting break flow for steam line break transients. To solve this problem, the secondary can be divided into two volumes with actual flow area input for the volume above the tubes where the heat transfer is not a concern. Although somewhat preliminary (further work needs to be performed), it appears that for transients requiring evaluation of secondary to primary heat transfer with changing secondary conditions, a more detailed nodalization of the secondary is required.

RETRAN adjusts the heat transfer area when using the steady state initialization to obtain an energy balance. If it is desirable to maintain the actual area, the steam generator pressure can be adjusted until the proper area is obtained. Another alternative used to force the initialized steam generator surface to the design value is to vary the conductivity of the tubing, in effect adding a constant fouling factor. It has not been established whether one of these approaches is better or if one has definite advantages over the other. The effect of the initialization process on the steam generator surface should be reviewed for each transient under consideration. It has also been found that if the differential temperature across the steam generator is very small (zero power transients), the steady state solutions will be reached easier if an artificially low feedwater enthalpy is used as a starting point for the steady state routine.

3.3 Upper Vessel Plenum Modeling

The use of a single volume in modeling the upper vessel is acceptable for conservative calculations of the uncontrolled rod withdrawal transient. The single volume approach results in a lower vessel outlet temperature calculation due to the mixing in the total volume, thereby delaying scram initiation and resulting in a higher power increase. RETRAN comparisons with plant data on a rapid cooldown transient indicate that a more detailed nodalization in the upper vessel head is more realistic than the single volume approach. For transients such as a steam line break, where flashing occurs in the upper vessel region, a multiple volume nodalization is required.

3.4 Pressurizer Modeling

The studies performed to date indicate without question that use of the pressurizer non-equilibrium model results in closer comparisons to experimental data than does the homogeneous approach. The use of this model is recommended for all transients.

In many cases, combining of the surge line volume with the pressurizer volume can result in a significant reduction in run times without appreciable altertation of the results. This modeling approach represents a possible saving in many cases, but it should be used only after a careful evaluation.

3.5 Transport Delay Model

The temperature delay model is necessary for transients which require an accurate prediction of temperature response throughout the loop. This model is recommended in those volumes in which the flow is similar to a pipe, i.e., long transport times with little mixing. In regions such as plenums, where a substantial amount of mixing occurs, this option is not recommended.

3.6 Auxiliary DNBR Model

At least one member of the Working Group compared the auxiliary DNBR model to the reactor vendors predictions. The results were qualitatively good but many quantitative differences were found. These differences are probably the result of model simplifications that have to be made when representing reactor core thermal hydraulics in a model of the complete reactor system. Since the auxiliary DNBR model was not intended as a stand alone model, it should be benchmarked against a standard thermal-hydraulics code for several different problems. Once this comparison is complete, the technical features of the auxiliary DNBR model

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can be assessed as to their importance in the prediction of DNBR's. Reiterating, the model is generally intended for comparison and trend usage, not for the prediction of an absolute DNBR.

3.7 Fuel Rod Thermal Model

At least one user found certain transients to be sensitive to the fuel rod thermal model input. This input consists of pellet, gap and rod parameters including dimensions, conductivities, and specific heats. The importance of these parameters may not be noted in many transients. However, they become quite important if the nuclear kinetics and power response are tied closely to the thermal response.

Often, discrepancies in transient heat flux predictions can also be traced to these parameters. The user should make sure that the parameters used are consistent with the time in life being analyzed and that they are consistent with any reference material that may be used for comparison purposes. Users should generally check the resulting gap coefficient and initial fuel temperatures for magnitude and appropriateness.

3.8 Steamline Break

While there exists many difficulties and input uncertainties in modeling the steamline break transient in PWRs, the work of Smith[VI.5-1] noted an important consideration not previously evident. In this study, the moderator temperature coefficient was handled first by using constant coefficient and secondly with a table of values which varied with moderator temperature. The second approach resulted in a much better agreement with the vendor late in the transient. This conclusion is worth noting whenever the user is making a serious attempt to match the vendor.

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X. REFERENCES

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