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Reactor System Transient Analyses Using The Retran Computer Code
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VEPCO REACTOR SYSTEM TRANSIENT ANALYSES

USING THE

RETRAN COMPUTER CODE

BY
N. A. SMITH

NUCLEAR FUEL ENGINEERING GROUP
FUEL RESOURCES DEPARTMENT

VIRGINIA ELECTRIC AND POWER COMPANY
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RECOMMENDED FOR APPROVAL:

Roger W. Cross

R. W. Cross
Supervisor, Nuclear Fuel Engineering

APPROVED:

R. M. Berryman

R. M. Berryman
Director, Nuclear Fuel Engineering

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 11, 1985

Mr. W. L. Stewart
Vice President
Nuclear Operations
Virginia Electric and Power Company
P. O. Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT VEP-FRD-41, "VEPCO REACTOR SYSTEM TRANSIENT ANALYSIS USING RETRAN COMPUTER CODE"

We have completed our review of the subject topical report submitted by Virginia Electric and Power Company (VEPCO) by letters dated April 14, 1981, February 27, 1984, July 12, 1984 and August 24, 1984. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that VEPCO publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, VEPCO and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization and Special
Projects Branch
Division of Licensing

Enclosure:
As stated

ENCLOSURE

SAFETY EVALUATION REPORT ON THE VEPCO
TOPICAL REPORT VEP-FRD-41, "REACTOR SYSTEMS
TRANSIENT ANALYSIS USING THE RETRAN COMPUTER CODE"

1. Introduction

The VEPCO topical report VEP-FRD-41, "Reactor System Transient Analysis Using the RETRAN Computer Code" was submitted to demonstrate the capability which VEPCO has developed for performing transient analysis using the RETRAN 01/MOD03 Computer Code. This submittal is consistent with our Generic Letter 83-11. This analysis capability is to be utilized by VEPCO to support plant operation and provide future reload safety analyses for both Surry and North Anna Nuclear Power Stations. The report provides some overview of the RETRAN Computer Code, but refers to EPRI documentation for further material on the RETRAN models and for qualification support of these models. The staff evaluation of the RETRAN Computer Code, has been completed. A staff safety evaluation report has been issued on the acceptability of that RETRAN computer code for analyzing reactor transients for licensing applications. The acceptance was subject to restrictions as specified in the staff SER for the generic RETRAN Computer Code. The VEPCO topical report VEP-FRD-41 was submitted by VEPCO in a letter dated April 14, 1981. In response to the staff requests for additional information, additional supporting materials were submitted in VEPCO letters dated February 27, 1984, July 12, 1984 and August 24, 1984. The staff evaluation is addressed below.

2. VEPCO NSSS Models

Discussion of the RETRAN plant models developed for the three-loop Westinghouse designed Surry and North Anna Units is provided in the topical report VEP-FRD-41. The transient analysis to be performed determines the level of detail required by the model. A single-loop and a two-loop RETRAN nodalization were submitted for staff review. The single-loop model has been formulated by representing the three reactor coolant loops as a single loop. This model was developed for use on transients which produce symmetric plant response in all unaffected reactor coolant loops. Examples of such transients would include a complete loss of a.c. power to all of the reactor coolant pumps (a loss of flow transient), a core reactivity insertion resulting from the uncontrolled withdrawal of a Rod Cluster Control Assembly, or a loss of external electrical load transient. The two-loop model was developed with one loop representing a single primary coolant loop and the other representing the remaining two primary coolant loops. The two-loop model was designed for use on transients which produce asymmetric thermal-hydraulic conditions among one of the three loops. Examples of such transients would include a postulated main steam line break resulting in the rapid cooldown of one reactor cooling loop, or a loss of power supply to a single reactor coolant pump, which results in a rapid flow coastdown of one reactor cooling pump.

In response to the staff request for additional information, VEPCO in letters dated July 12, 1984 and August 24, 1984, provided detail descriptions in the following areas: 1) Volume and flow path network including heat slabs, 2) Component models used and user modifications to default models, 3) Control system models, and 4) RETRAN input option selections.

The staff has reviewed the above VEPCO model descriptions and finds them acceptable for demonstrating understanding of the RETRAN code.

3. <u>Analysis Methodology</u>

VEPCO intends to reference VEP-FRD-41 as their basic model for reload applications. Following determination of the key reload parameters, the safety analyst will apply the appropriate boundary conditions required for the specific application. The evaluation is to ensure that those key parameters which may influence the transient response are consistent with the bounds or limits established by the technical specifications and parameters used in the reference analysis. For cases where a parameter falls outside these previously defined limits an evaluation of the impact of the change on the results for the appropriate transients must be made. For cases where significant variations occur, or for parameters which have a strong influence on accident results, reanalysis of the affected transient is required. The results of a reanalysis are compared to the appropriate analysis acceptance criteria. If the results of a reanalysis meet the acceptance criteria, the reload evaluation process is complete. If the analysis acceptance criteria are not met, more detailed analysis methods or Technical Specification changes may be required to meet the acceptance criteria. The NRC will be informed of the results of the evaluations in accordance with the requirements of 10 CFR 50.59. VEPCO will use analysis methodology and acceptance criteria identified in the following documents: 1) Surry Power Station Units 1 and 2, Final Safety Analysis Report, 2) North Anna Power Station Units 1 and 2, Final Safety Analysis Report, and 3) WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," which has been reviewed and approved by NRC in 1980. We

require that the licensee fully document all assumptions and boundary conditions used in each application. This review does not constitute a transient specific methodology approval.

4. Qualification Comparisons

The VEPCO has developed a system transient analysis capability using the RETRAN Computer Code for non-LOCA initiating events. In order to demonstrate VEPCO's ability to correctly use the RETRAN Computer Code, verification work has been performed by benchmarking both actual plant transient data and independent safety analyses previously performed by the NSSS vendor and documented in the FSAR.

For plant transient data benchmarking, the VEPCO RETRAN Computer Code was developed to model both Surry and North Anna power stations in a best estimate mode. This permits direct comparisons to the actual measured plant data. Comparisons were made with flow coastdown tests performed at both the Surry and North Anna plants and a plant cooldown transient which occurred at North Anna Unit 1. In the comparison of RETRAN analyses to the data obtained from the flow coastdown tests, both single-loop and two-loop RETRAN models were used to simulate pump coastdown tests of various configurations (i.e. one pump coastdown, three pump coastdown). The results of the comparison as documented in the topical report indicate that the VEPCO RETRAN predictions are in close agreement with the data obtained from Surry and North Anna. A RETRAN analysis was performed to simulate the plant cooldown transient which occurred at North Anna Unit 1 on September 25, 1979. The transient was initiated by a turbine trip and

subsequent reactor trip. Safety injection was actuated on a low pressurizer pressure during the transient due to RCS depressurization in response to a fully stuck open steam dump valve. The VEPCO RETRAN model used to simulate the cooldown scenario was a single-loop representation of the North Anna Unit. The calculated transient parameters including steam pressure, RCS temperatures, pressurizer pressure, and pressurizer level, were compared to the actual data taken during the event. The results of the comparison show agreement between the best estimate calculation and the actual transient data.

VEPCO provided comparisons of FSAR licensing safety analysis with analyses performed using the RETRAN Computer Code. The basis for the event selection were: 1) Consideration of those events which have previously been determined limiting and have been most frequently subjected to reanalyses during each reload (e.g. Rod Withdrawal from Power and Complete loss of flow); 2) Selecting analyses in each of the major categories of initiating events which include changes in reactivity (e.g. rod withdrawal transients), variations in primary coolant flow rate (e.g. loss of flow transient), and variations in primary to secondary system heat transfer rates (e.g. main steam line break); and 3) Transients which are both symmetric (e.g. loss of load transient) and asymmetric (e.g. single pump flow coastdown) with respect to the thermal hydraulic response of the reactor coolant loops.

The results of analyses performed by VEPCO (using the RETRAN Computer Code) for the above stated events compared favorably to those obtained by

its NSSS vendor. The similarities in system response hold for a broad variety of transients and result in identical conclusions regarding core and system conditions.

In response to the staff request, VEPCO, in a letter dated July 12, 1984, provided results of RETRAN sensitivity studies for the following transients: 1) Rod withdrawal at power, 2) Rod withdrawal from sub-critical, 3) loss of load, 4) excessive load increase, and 5) Complete loss of flow.

The staff has evaluated the results of the VEPCO's sensitivity studies and finds them consistent with the NSSS Vendor's analyses, as documented in the Surry and North Anna FSARs.

To further verify the comparability of the VEPCO RETRAN model to the NSSS Vendor's analysis model, VEPCO, in a letter dated August 24, 1984, submitted a supplement to VEP-FDR-41 which compared parallel calculations of RETRAN and LOFTRAN performed by VEPCO. The LOFTRAN code is an NRC approved analytical program developed and maintained by the Westinghouse Electric Corporation for use in performing general non-LOCA transient and accident analyses. VEPCO has obtained access to LOFTRAN via a special licensing agreement with Westinghouse. The comparisons were performed with a LOFTRAN model of the Surry plant assembled by VEPCO applying the same data base used for developing the VEPCO RETRAN models. Thus the basic plant geometric and thermal parameters are consistent for the two models. The following transients were calculated and compared using both computer models: 1) Reactor trip from hot full power followed by a turbine trip, 2) Turbine trip from hot full power. No credit taken for

direct reactor trip on the turbine trip, and 3) Simulataneous trip of all three reactor coolant pumps at hot full power. No credit taken for reactor trip on pump under voltage or under frequency. The results of these analyses confirmed that the VEPCO RETRAN models could produce compatible analysis results with that from the LOFTRAN models.

5. Conclusions

Based on the VEPCO RETRAN model and the qualification comparisons discussed above, the staff concludes that VEPCO has demonstrated their capability to analyze non-LOCA initiated transients and accidents using the RETRAN Computer Code. VEPCO intends to perform future reload analyses and supporting plant operations for Surry and North Anna plants. We find VEPCO qualified to perform the non-LOCA initiated transients and accident analyses using the RETRAN models and methodology. This topic report does not include the Rod Ejection Accident analysis which has been addressed in a separate VEPCO Topic Report VEP-NFE-2 and a separate staff safety evaluation report. VEPCO has not provide information to address the restrictions stated in the staff SER for the generic RETRAN Computer Code. The acceptance of the VEPCO RETRAN models is subject to the restrictions to the general RETRAN computer code specified in the staff safety evaluation report issued in July 1984 on RETRAN. VEPCO has not provided an input deck to the NRC staff as was required by the staff SER for the generic RETRAN code. We continue to require that this input deck be provided to us as a condition of this approval.

With respect to the quality assurance requirement of the VEPCO RETRAN Computer Code, the staff has performed an audit at VEPCO with satisfactory results. The staff requires that all future modification of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures.

CLASSIFICATION/DISCLAIMER

The data, information, analytical techniques, and conclusions in this report have been prepared solely for use by the Virginia Electric and Power Company (the Company), and they may not be appropriate for use in situations other than those for which they were specifically prepared. The Company therefore makes no claim or warranty whatsoever, express or implied, as to their accuracy, usefulness, or applicability. In particular, THE COMPANY MAKES NO WARRANTY OF MERCHANTABILITY OR FITNESS FOR A PARTICULAR PURPOSE, NOR SHALL ANY WARRANTY BE DEEMED TO ARISE FROM COURSE OF DEALING OR USAGE OF TRADE, with respect to this report or any of the data, information, analytical techniques, or conclusions in it. By making this report available, the Company does not authorize its use by others, and any such use is expressly forbidden except with the prior written approval of the Company. Any such written approval shall itself be deemed to incorporate the disclaimers of liability and disclaimers of warranties provided herein. In no event shall the Company be liable, under any legal theory whatsoever (whether contract, tort, warranty, or strict or absolute liability), for any property damage, mental or physical injury or death, loss of use of property, or other damage resulting from or arising out of the use, authorized or unauthorized, of this report or the data, information, analytical techniques, or conclusions in it.

ACKNOWLEDGEMENTS

This report is the culmination of several years of development effort in which numerous individuals have participated. The analytical work was performed by Messrs. R. W. Cross, S. M. Mirsky, G. M. Suwal and the author. Valuable technical and editorial comments were provided by Messrs. W. C. Beck, M. L. Bowling, E. R. Smith, Jr. and T. L. Wheeler. Ms. Miranda Cooper and Ms. Sharon Kulp provided patient typing support. The efforts of these and numerous other individuals are gratefully acknowledged.

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SECTION 1 - INTRODUCTION

The Virginia Electric and Power Company (Vepco) has developed the capability to perform system transient analyses of the North Anna and Surry Nuclear Power Stations. This capability, coupled with the core thermal/hydraulic analysis capability discussed in Reference 1, encompasses the conservative non-LOCA licensing analyses required for the Conditions I, II and III transients addressed in the Final Safety Analysis Report (limited application to Condition IV transients is also included). In addition, the capability for performing best estimate analyses for plant operational support applications has also been developed.

The purpose of this effort is to 1) develop expertise in the system transient analysis area, 2) support reactor operation and 3) provide a basis for the reload core safety analysis and licensing process. The principal analysis tool is the RETRAN computer code[2] which determines the time dependent or transient thermal-hydraulic response of a Nuclear Steam Supply System (NSSS). The RETRAN computer code calculates 1) general system parameters as a function of time and 2) boundary conditions for input into more detailed calculations of Departure from Nucleate Boiling or other thermal and fuel performance margins. The theory and numerical algorithms, the programming details, and the user's input information for the RETRAN computer code have been documented by its developers, Energy Incorporated (EI) and the Electric Power Research Institute (EPRI), in Volumes I through IV of Reference 2. Volume IV of Reference 2 provides the results of the extensive verification and qualification of the code which was performed by a group consisting of EI, EPRI, and 15 utilities including Vepco. The verification activity consisted of qualification of the code by comparison of code results with separate effects experiments, with systems effects tests, and with integrated system responses based on actual plant data or FSAR results.

Performance of system transient analysis requires both single and multiloop

modeling of the NSSS in order to analyze the required range of FSAR and operational support transients. Those transients for which the system thermal-hydraulic response of all reactor coolant loops is essentially identical require only a single loop representation. However, some transients are expected to have different responses in one or more of the reactor coolant loops, and these transients require multiloop representation of the NSSS. The RETRAN computer code, which is a variable geometry code, has the high degree of flexibility necessary for various system representations. Consequently, several models, including both single and multiloop representations, have been developed for the Vepco nuclear power stations.

In conjunction with both an analysis tool and system models, the development of a non-LOCA licensing analysis capability requires conservative analysis assumptions and input data. For licensing calculations, the Vepco analysis assumptions are consistent with those documented in the units' FSAR's (References 3 and 4). However, the specific analysis input may change as a result of plant modifications such as core reloads. Consequently, the appropriate licensing analysis input consists of the current limiting values for the important safety parameters. For best-estimate analyses, nominal input values and actual operating histories of the Vepco nuclear power stations are used.

The remainder of the report is organized in the following manner. Section 2 provides an overview of the RETRAN computer code, and Section 3 describes the Vepco models appropriate for the Surry and North Anna Nuclear Steam Supply Systems, as illustrated by a discussion of models developed for the Surry units. Section 4 provides a discussion of the Vepco transient analysis techniques and their relationships to other aspects of the licensing analysis process. Section 5 provides the results of a range of comparative analyses using the RETRAN code and the models of the NSSS discussed in Section 3 with calculations performed for the 1) design and licensing of the Surry Nuclear Power Station and 2) actual Surry and North Anna transient data. The report conclusions and references are provided in Sections 6 and 7, respectively.

SECTION 2 - OVERVIEW OF THE RETRAN COMPUTER CODE

The RETRAN computer code was developed by Energy Incorporated under the auspices of the Electric Power Research Institute [2]. As such, the RETRAN package is based upon the computer code RELAP4/003 Update 85 which was released by the United States Nuclear Regulatory Commission (NRC) as part of the Water Reactor Evaluation Model (WREM) [5]. A detailed description of the RETRAN computer code can be found in Volume I of Reference 2. The following paragraphs summarize the important features of the code.

RETRAN contains the same fluid differential and state equations as RELAP4 for describing homogeneous equilibrium flow in one dimension. The representations used in previous RELAP codes for control volumes and junctions are also used in RETRAN and allow the analyst to model a system in as much detail as desired. The modeling flexibility of the code is important and will be discussed in more detail in Section 3. The equation systems, which describe the flow conditions within the channels, are obtained from the local fluid conservation equations of mass, momentum and energy by use of mathematical integral-averaging techniques. Forms of the momentum equation are available for both compressible and incompressible flow.

The heat conduction representation capabilities of RETRAN have been increased over previous RELAP versions. The principal augmentation to RETRAN is the capability to more accurately calculate two-sided heat transfer. The appropriate heat transfer correlation is selected based on thermodynamic conditions in each of two flow streams, on either side of a heat conducting solid. Consequently, representations of the heat transfer processes occurring in the steam generator, for example, are more accurate than previously possible.

Reactor kinetics are represented in RETRAN using a point kinetics model with reactivity feedback. The reactivity feedback can be represented by constant

coefficients or in tabular form and accounts for explicit control actions (e.g., rod scram) and changes in fuel temperature, moderator temperature and density, and soluble boron concentration.

The system component models utilized in RETRAN include a pump model that describes the interaction between the centrifugal pump and the primary system fluid, and valve models that represent either simple valves, check valves or inertial valves. The flexibility of the valve representation and their configuration is important in allowing a wide variety of options to the user for the modeling of system dynamics. Several representations for heat exchangers can be modeled by the code. These include the previously discussed two-sided heat transfer and several representations of one-sided heat transfer in conjunction with user specified boundary conditions. A non-equilibrium pressurizer can be modeled in which the thermodynamic state solutions of the liquid and vapor regions of the pressurizer are determined from a distinct mass and energy balance for each region.

As in RELAP, a variety of trip functions can be modeled in the RETRAN code to represent various reactor protection system actions. A refinement of the RETRAN code over the RELAP code is the additon of a reactor control system modeling capability. Consequently, the dynamics of linear and non-linear control systems are represented with RETRAN models of the more common analog computer elements. This additional capability is necessary for both best-estimate and licensing analysis, since the responses of various control and protection systems may have a significant effect on the overall system response.

SECTION 3 - REPRESENTATIVE VEPCO NSSS MODELS

3.1 Introduction

The RETRAN computer code is a variable-geometry code which allows the analyst to model a system in as much detail as required for a particular analysis. To illustrate this concept, two models developed for the Surry Nuclear Power Station will be discussed in detail in this section. (The modeling methodology is also applicable to the North Anna Nuclear Power Station).

The Surry Nuclear Power Station consists of two units, Surry Units No. 1 and 2, which are identical Westinghouse designed three coolant loop pressurized water reactors with core thermal ratings of 2441 Mwt. The three similar heat transfer loops are connected in parallel to the reactor vessel with each loop containing a centrifugal pump, loop stop valves and a steam generator. The system includes a presssurizer and the associated control system and instrumentation necessary for operational control and protection.

The reactor vessel encloses the reactor core consisting of 157 fuel assemblies with each assembly having 204 fuel rods and 21 thimble tubes arranged in a 15 x 15 array. The fuel used in the Surry cores consists of slightly enriched uranium dioxide fuel pellets contained within a Zircaloy-4 cladding. General thermal and hydraulic design parameters for the reactor system are listed in Table 3.1.

The RETRAN thermal hydraulic model is formulated by representing individual portions of the hydraulic system as nodes or control volumes. Control volumes are specified by the thermodynamic state of the fluid within the volume and basic geometric data such as volume, flow area, equivalent diameter and elevation. The flow paths connecting volumes or boundary conditions associated with a volume are designated as junctions. Junctions are described by specifying the flow, flow area, elevation, effective geometric inertia, form loss coefficient and flow equation specification for that particular flow path. Thermal interactions with system metal in the

NSSS are modeled with heat conductors. Heat conductors may represent heat transfer from passive sources such as the metal of the reactor coolant system piping or the steam generator tubes. In addition, the internal generation of heat in the core may be represented by active heat conductors designated as powered conductors. Heat conductors are primarily specified by providing the heat transfer area, volume, hydraulic diameter, heated equivalent diameter and channel length of the particular part of the system being modeled. Temperature - dependent materials properties (specific heat, thermal conductivity and linear thermal expansion coefficient) are also input. In general, the basic NSSS model is formulated with the code capabilities discussed above. An extensive research effort was conducted to determine the appropriate input required for the models of the Surry and North Anna units. Information was obtained from plant drawings, the Final Safety Analysis Reports[3, 4], Vepco internal operating documents, equipment technical manuals and specific information requested from the NSSS vendor. Specific control capabilities and constitutive models of system components will be discussed in the following paragraphs.

3.2 Single Loop Model

The analysis to be performed and level of detail required dictates the general form of the models which are required. Many transients are expected to produce similar responses simultaneously in all reactor coolant loops. Examples of such transients would include a complete loss of power simultaneously to all reactor coolant pumps resulting in a pump coastdown, a core reactivity insertion resulting from the uncontrolled withdrawal of a Rod Cluster Control Assembly (RCCA), or a loss of external electrical load resulting in a large, rapid steam load reduction.

To perform these transients, a single loop model of a Surry unit has been formulated by representing the three actual reactor coolant loops as one loop. This approach is consistent with currently used safety analysis methodology[6]. The resulting representation is provided in Figure 3.1 and consists of 19 volumes, 28 junctions and 7 heat conductors. While the specific model input for the Surry and North Anna plants is

different, the basic model description is the same for the single loop models of both plants. The reactor vessel includes representation of the downcomer, upper and lower plenums, core bypass, and reactor core. The steam generator is represented by four volumes on the primary side, one volume on the secondary side and four heat conductors representing the tubes. Single volumes represent the hot leg piping, steam generator inlet plenum, pump suction piping, reactor coolant pump, cold leg piping, pressurizer, and pressurizer surge line. Primary system boundary conditions are specified with junctions representing the pressurizer relief and safety valves. Junctions representing the feedwater inlet, steam outlet, atmospheric steam relief and steam line safety valves provide secondary system boundary conditions. Specific aspects of the basic model will be discussed below.

The RETRAN code contains several system component models which are used in the Surry Single Loop Model. These include pump models which describe the interaction between the centrifugal pump and the primary system fluid. These models calculate pump behavior through the use of empirically developed pump characteristic curves which uniquely define the head and torque response of the pump as functions of volumetric flow and pump speed. RETRAN includes "built-in" pump characteristics which are representative of pumps supplied by the major reactor coolant pump manufacturers. These curves may be modified, as appropriate, by the user to more realistically represent a specific pump design. Although the built-in data are not appreciably different from Vepco's plant-specific curves, Vepco's Single Loop Models incorporate the specific head vs. flow response for first quadrant operation found in the Units' FSAR's[3, 4].

The Single Loop Model incorporates the RETRAN pressurizer model which defines two separate thermodynamic regions that are not required to be in thermal equilibrium. A non-equilibrium capability is particularly necessary when the transient involves a surge of subcooled liquid into the pressurizer. In addition, the Single Loop

Model represents the effects of subcooled spray, electrical immersion heaters, liquid droplet rainout and vapor rise in the pressurizer.

The reactor systems trip logic is modeled to the detail required for a specific analysis. RETRAN trip functions are used to model 1) protective functions, such as the overtemperature ΔT trip, which result in reactor scram, 2) control system bistable element logic, such as coincidence trips which model "majority" logic and 3) general problem control (e.g., problem termination, etc.).

The protective function trips necessary for the analyses documented in Section 5 and modeled in the Single Loop Model include:

1. High flux
2. Overtemperature ΔT
3. Overpower ΔT
4. Low/high pressurizer pressure
5. High pressurizer level
6. Low coolant flow
7. Loss of power to reactor coolant pumps.

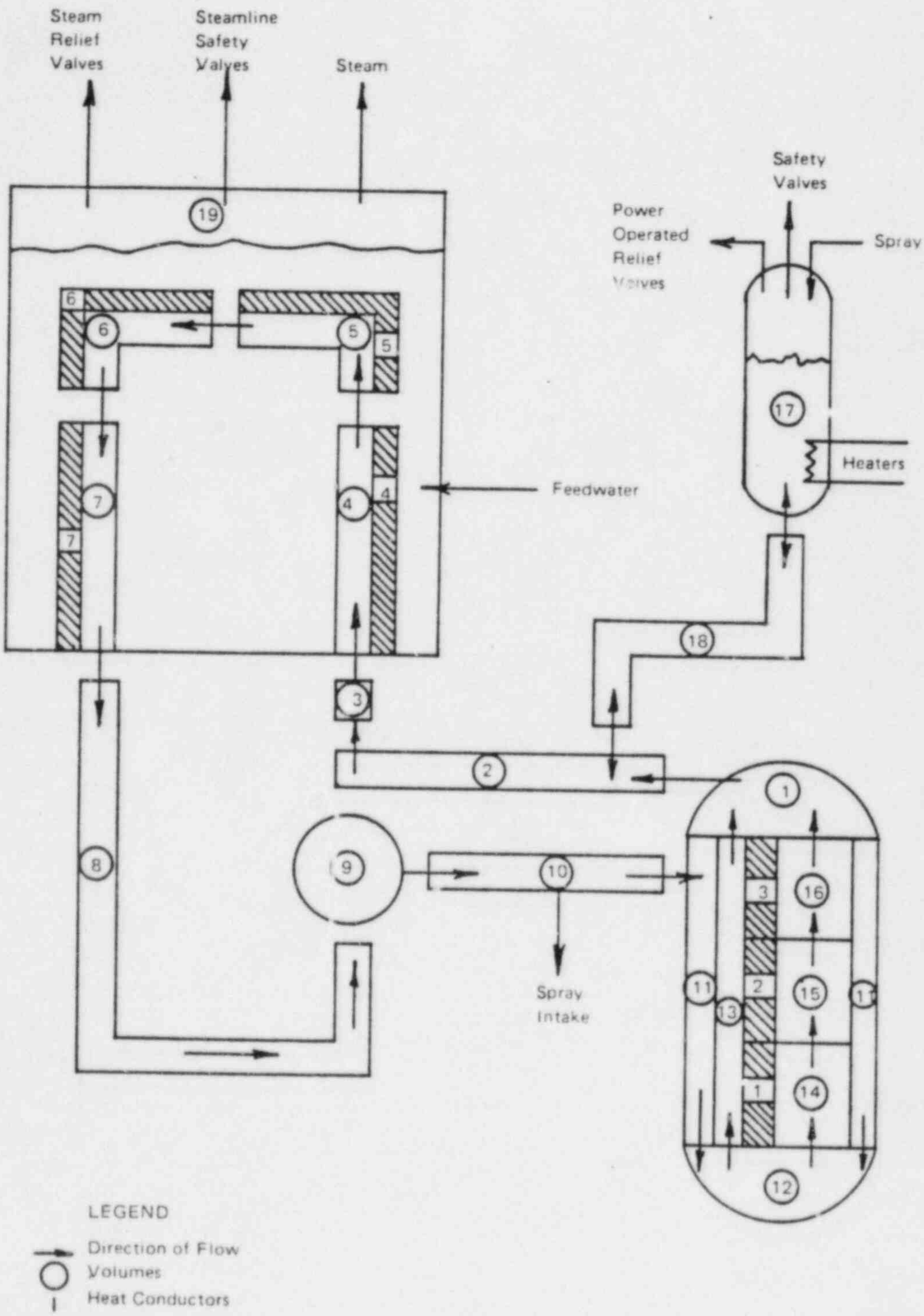
The Single Loop Model also incorporates the RETRAN control system capability to model the following NSSS control and protection features:

1. Overtemperature ΔT setpoint
2. Overpower ΔT setpoint
3. Pressure controller
4. Lead/lag compensation of the low pressure trip signal.

The core power response is determined by the point kinetics model in conjunction with explicit reactivity forcing functions and thermal feedback effects from moderator and fuel in the three core regions. The point kinetics model specified for the Single Loop Model incorporates one prompt neutron group and six delayed neutron groups with decay heat represented by 11 delayed gamma emitters and the important radioactive actinides, U-239 and Np-239. Explicit reactivity forcing functions

Figure 3.1

ONE LOOP SURRY RETRAN MODEL



represent reactor scram an. reactivity insertion due to control rod withdrawal in the Single Loop Model as the particular analysis requires. Constant temperature coefficients or reactivity tables as a function of temperature (fuel), density (moderator) or power represent feedback effects. Core power is distributed axially among the three core conductors approximating a symmetric cosine shape. Three core materials regions are used to represent the UO_2 fuel pellets, the helium filled gap and the Zircaloy cladding. Several radial nodes are specified in the pellet region, in the gap and in the cladding. Direct moderator heating is appropriately accounted for in the model. The transient fuel and clad temperatures are calculated based on temperature-dependent thermal properties, which are input in tabular form.

The preceding paragraphs have discussed the Surry Single Loop Model in some detail. Some of the input is transient specific and the important assumptions and parameter values will be discussed for each analysis presented in Chapter 5.

3.3 Multi-loop Model

Some transients are expected to have different responses in one or more of the reactor coolant loops. These transients require multi-loop representation of the NSSS. Several examples include the rupture of a main steam line resulting in the rapid cooldown of only one reactor coolant loop or the loss of power to a single reactor coolant pump resulting in a flow coastdown in only one coolant loop.

Consequently, a two loop model has been developed which represents the Surry units. One loop of the model represents a single primary coolant loop while the other loop is structured to represent two primary coolant loops. This approach is consistent with current system transient analysis methodology[6]. The model is designed with a geometrical noding which is detailed enough to analyze transients where flow and temperature asymmetries within the reactor vessel are significant.

The Surry Two Loop RETRAN Model, with a reactor vessel configuration appropriate for analyzing a Main Steam Line Break (MSLB) transient is shown in Figure

3.2. (The input structure of RETRAN allows rapid alterations in noding and flow path representations, as may be appropriate for analyzing multiloop transients requiring less reactor vessel detail.)

This particular configuration consists of 42 volumes, 56 junctions and 16 heat conductor nodes. Single volumes in each loop represent the hot leg piping, steam generator inlet plenum, pump suction piping, reactor coolant pump and cold leg piping. Each steam generator is represented by four primary side volumes and four heat conductor nodes for the tube region.

The reactor vessel representation includes a two volume, "split" downcomer, and similarly divided inlet and outlet plena. Junctions representing interloop flow mixing in the inlet and outlet plena allow for a range of mixing assumptions to be specified, such as "perfect" or complete mixing or an incomplete mixing assumption based on actual test data (see, for example, Reference 7). The latter assumption, combined with appropriate azimuthal weighting factors applied to the temperature coefficients, may be used to conservatively model the core kinetics response to a MSLB transient. This is facilitated by a split core model in which the reactor core is represented by two azimuthal sectors, with each sector being divided axially into four coolant volumes. Thus, for an analysis in which an imperfect interloop flow mixing assumption is conservative, each azimuthal core sector receives more of its flow from the nearest loop than would be dictated by complete mixing.

Eight powered heat conductors represent the core and four passive heat conductors represent the tube region in each steam generator. Junctions representing the feedwater inlet and steam outlet in each steam generator provide secondary side boundary conditions. A junction representing safety injection of borated water via the cold leg injection path models a primary side boundary condition. Specific model aspects will be discussed in more detail below.

As in the Single Loop Model, the Two Loop Model incorporates a Surry specific first-quadrant pump head curve and the non-equilibrium pressurizer option.

The Two Loop Model also makes use of the RETRAN valve system component model. The simple valve option models the main steam valves and the break opening simulation associated with the severance of a main steam line.

Trip functions are modeled in a manner similar to that discussed for the Single Loop Model. Specific protective function trips currently in the Two Loop Model include:

A. Steam Break Protection

 1. Safety injection initiated by any of the following:

  a. Low Pressurizer pressure

  b. High header/steam line pressure differential

  c. High steam flow coincident with either 1) low steam pressure or 2) low primary system average temperature

 2. Main steam line isolation

B. Other-Reactor trip on low coolant loop flow.

The core power response is calculated via point kinetics in the Two Loop Model as previously discussed for the One Loop Model. A specific reactivity forcing function represents the effects of increased soluble boron levels in the core following safety injection for transients, such as the Main Steam Line Break, where safety injection is important. The time-varying core boron concentration is generated by a submodel using the RETRAN control system capability which performs a detailed calculation of the dilution and transport of safety injection fluid. Moderator and Doppler feedback effects are represented using reactivity functions in a manner consistent with that reported in References 3, 4 and 7. The feedback effects are weighted axially based on perturbation theory approximations; azimuthal weighting may be by volume, or for situations where skewed inlet temperature distributions are important, a conservative non-uniform weighting scheme such as discussed in Reference 7 is used. Noding in the fuel, gap and cladding regions is the same as that discussed for the One Loop Model.

Figure 3.2

TWO LOOP SURRY RETRAN MODEL

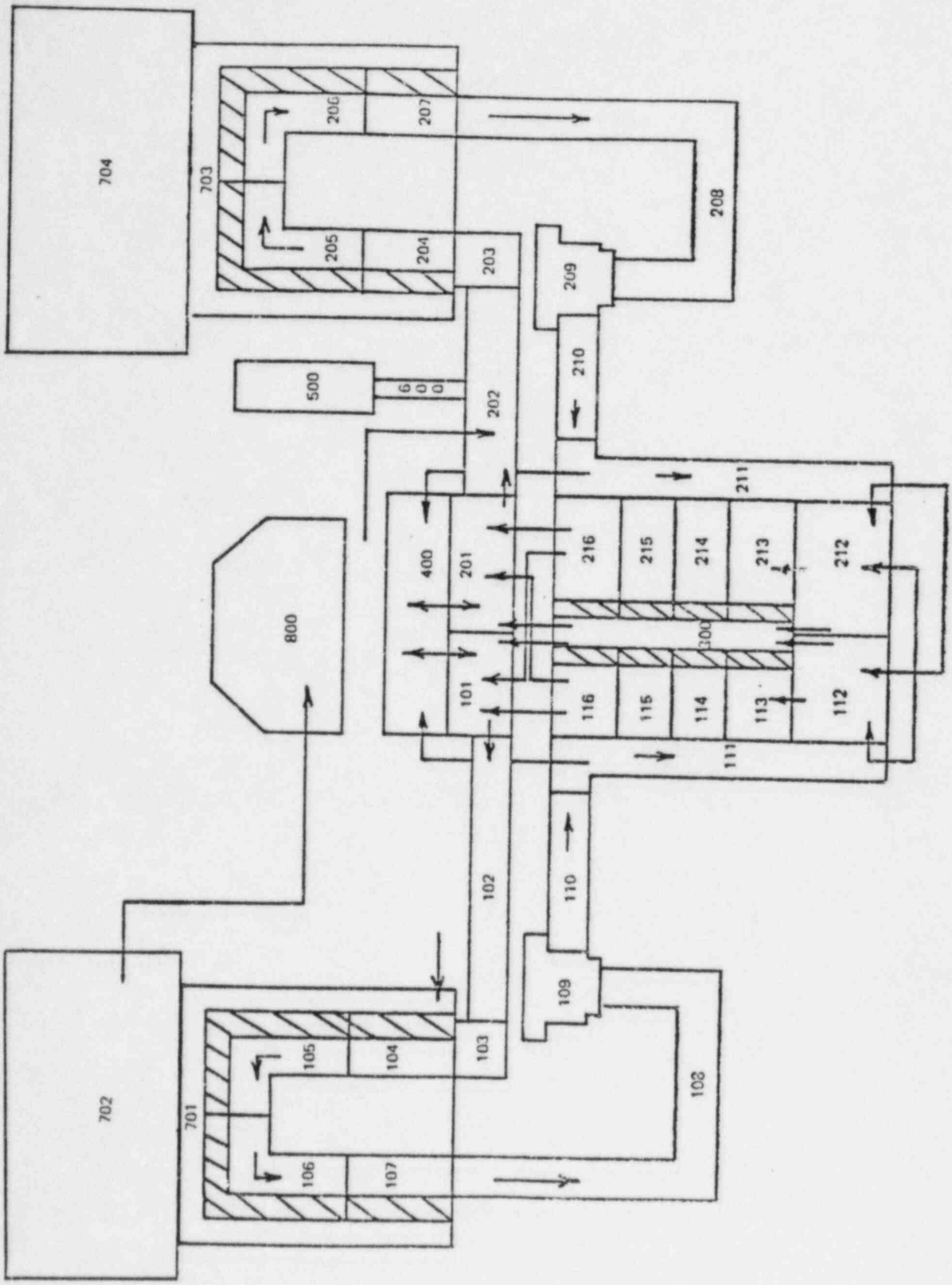


Table 3.1

Thermal - Hydraulic Design Parameters - Surry Plant

Total core heat output, Mwt	2441
Heat generated in fuel, %	97.4
System operating pressure, psi	2250
Total coolant flow rate, lb./hr.(gpm)	100.7×10^6 (265,500)
Coolant Temperatures, ^oF (@100% power)	
Nominal inlet	543
Average rise in the core	65.5
Average rise in vessel	62.6
Average in the core	577.0
Average in vessel	574.
Nominal core outlet	608.5
Nominal vessel outlet	605.6
Average linear power density, Kw/ft.	6.2

SECTION 4 - SYSTEM TRANSIENT ANALYSIS METHODOLOGY

4.1 Introduction

As discussed in the introduction, Vepco system transient analysis is intended for both best estimate and licensing applications. Since core reloads are the most common and expected reason for accident reanalysis, Vepco's system transient methodology will be discussed in that context.

In general, Vepco intends to continue the reference analysis approach which has been employed by our nuclear fuel vendor in support of our nuclear plants. This approach is fully explained in Reference 8 and requires reanalysis of an accident, which is part of the licensing basis for our plants, only under certain conditions. These conditions and the licensing evaluation process are summarized in Section 4.2. Section 4.3 discusses the system transient analysis methodology and its relation to the licensing process.

4.2 Licensing Evaluation Process

The actual execution of transient analyses forms part of an integrated system of evaluations performed to verify the acceptability of a reload core design from the standpoints of safety, economics and operational flexibility. The purpose of this section is, therefore, to provide a brief overview of the relationship of transient analyses to the integrated reload design and licensing process. The reload design process will be described in detail in a future Vepco topical report. However, the process has been generally described in Reference 8 and consists of a design initialization, design of the core loading pattern, and detailed characterization of the core loading pattern by the nuclear designer. The latter process determines the values of key reload parameters. These key reload parameters are provided to the safety analyst who uses them in conjunction with current plant operating configurations and limits to evaluate the impact of the core reload on plant safety.

In performing this evaluation, it is necessary to ensure that those key parameters which influence accident response are maintained within the bounds or "limits" established by the parameter values used in the reference analysis (i.e. the currently applicable licensing calculation). The reference analysis (and the associated parameter limits) may be updated from time to time in support of a core reload or to evaluate the impact of some other plant parameter change.

For cases where a parameter falls outside these previously defined limits, an evaluation of the impact of the change on the results for the appropriate transients must be made. This evaluation may be based on known sensitivities to changes in the various parameters in cases where a parameter change is small or the influence on the accident results is weak. For cases where larger parameter variations occur, or for parameters which have a strong influence on accident results, explicit reanalysis of the affected transients is required and performed as discussed in Section 4.3. Past analytical experience has allowed the correlation of the various accidents with those parameters which have a significant impact on them.

The results of such a correlation are summarized in References 3, 4 and 8. If required, a reanalysis is performed and the results are compared to the appropriate analysis acceptance criteria identified in References 3, 4 and 8. The reload evaluation process is complete if the acceptance criteria are met, and internal documentation of the reload evaluation is provided for the appropriate Vepco safety review. If the analysis acceptance criteria are not met, more detailed analysis methods and/or Technical Specifications changes may be required to meet the acceptance criteria. The NRC will be informed of the results of the evaluation process in accordance with the requirements of 10CFR 50.59.

4.3 System Transient Analysis

The production of a conservative, reliable safety analysis of a given anticipated or postulated transient is accomplished by combining a system transient model with

appropriate transient specific input. A system transient model, such as those discussed in Section III, is designed to provide an accurate representation of the reactor plant and those associated systems and components which significantly affect the course of the transient. Transient specific input ensures that the dynamic response of the system to the postulated abnormality is predicted in a conservative manner, and includes a) initial conditions, b) core reactivity parameters such as Doppler and moderator temperature coefficients, and control rod insertion and reactivity characteristics, and c) assumptions concerning overall systems performance. Important systems performance assumptions include the availability of certain system components (such as pressurizer spray or relief valves) and control and protective characteristics (setpoints, instrument errors, delay times).

A summary of key analysis assumptions for those transients discussed in Chapter 5 is included in the Appendix. A general discussion of this transient specific input is provided in the paragraphs which follow.

4.3.1 System Model Application

While RETRAN affords the modeling flexibility to develop an infinite number of representations for a given nuclear plant, practical considerations dictate that a small number of standard plant models be assembled and maintained for performance of the entire spectrum of system transient analyses. Section 3 provides examples of the types of models that are required for system transient analysis. RETRAN makes use of an input structure which allows modification of the base deck input for specific cases by use of override cards. Thus, specific transient cases may be executed without altering the base plant models.

The base models are designed to provide a basic system description comprised of those parameters which would not ordinarily change from cycle to cycle. Thus such parameters as system volumes and flow areas, characteristics of various relief and safety valves, primary coolant pump characteristics, etc. form part of the base models.

Since occasional changes to such "fixed" parameters do occur as a result of equipment modifications or replacement or upgrades to various safety-related systems, the base models are reviewed periodically to ensure that the latest system-related changes have been adequately reflected. Generally this review is performed during the initial core design stages of each reload cycle.

4.3.2 Transient Specific Input

As discussed earlier, input parameters which may be varied for a specific analysis to ensure a conservative representation of the system response include initial conditions, core reactivity parameters and assumptions concerning systems performance. For a given type of accident, not all parameters have a significant influence on the accident response. Those parameters which are significant, and their limiting directing (i.e., maximum or minimum) are determined from:

a) the unit's FSAR

b) sensitivity studies such as those summarized in Reference 8.

The most important of these safety-related parameters are examined in more detail in the following discussions.

4.3.2.1 Initial Conditions

Most accidents exhibit some sensitivity to the initial conditions assumed. For accident evaluation, the initial conditions are obtained by adding or subtracting, as appropriate, maximum steady-state errors to or from rated values. Steady-state errors which are applied are:

a)	Core Power	+ 2 percent allowance for calorimetric error
b)	Average reactor coolant system temperature	$\pm$ 4 ^{o}F (Surry) allowance for deadband and measurement error.
c)	Pressurizer pressure	$\pm$ 30 psi allowance for operational fluctuations and measurement error.

In general, errors are chosen in the directions which minimize core thermal margin or margin to other plant design criteria and are therefore dictated by the type of analysis being performed.

4.3.2.2 Reactivity Parameters

Reactivity parameters, which may have a significant impact on the transient response to an abnormal condition, include the Doppler and moderator temperature coefficients of reactivity, delayed neutron fractions, the trip reactivity and insertion characterics, and the differential control bank worth. The reactivity parameters are normally chosen in a manner which tends to maximize the nuclear power during the transient. The limiting value of a given parameter is dictated by the type of transient involved as indicated by the examples in Chapter 5. For example, for transients where large decreases in moderator temperature are a concern (such as a steamline break), large negative moderator temperature coefficients tend to be limiting. On the other hand, for transients where increases in moderator temperature are the major concern (for example, a loss of external electric load or turbine trip) the most positive value of moderator temperature coefficient tends to produce a more severe transient. The choice of the limiting reactivity parameter value, as discussed earlier, is made to ensure that the accident analyses are bounding with respect to the range of parameter values realized over the life of the reload core.

4.3.2.3 System Performance Assumptions

The predicted transient performance is influenced by assumptions concerning the availability of various system components and the characteristics of the reactor protection and control system.

In many instances the mitigating effect of various system design features on postulated transients are ignored. This provides additional conservatism and confidence that the calculation conservatively "bounds" the actual expected system performance. For example, the analysis of the Uncontrolled Rod Withdrawal from Subcritical transient conservatively takes no credit for the source range or intermediate range flux level trips or for the intermediate range control rod stop function. For certain control system components (e.g., relief and spray valves), it is conservative to assume

availability for some transients and unavailability for others. The choice of whether or not to include the effect of a particular system component is based on prior experience and sensitivity studies. These assumptions normally remain constant from analysis to analysis of a given transient.

In order to adequately account for the impact of instrumentation errors and signal delays, conservative protection system characteristics are assumed when performing accident analyses. Thus, expected instrument errors and system response times are conservatively bounded by the analysis assumptions, thereby adding to the previously discussed conservatisms employed in a transient analysis. Examples of protection system setpoints and delays used in performing Surry safety analyses are shown in Table 4.1. Periodic review of protection system setpoints as defined in the plant Precautions, Limitations and Setpoints is performed to ensure that the safety analysis models continue to conservatively reflect current safety system settings.

4.4 Use of System Transient Results

The results of a system thermal hydraulics analysis are used either for direct comparison to accident analysis acceptance criteria (e.g. system pressure limits) or as a boundary condition for more detailed core thermal hydraulic analyses, using the Vepco capability documented in Reference 1, or for more detailed fuel rod analyses, as required for some condition IV transients.

TABLE 4.1 - PROTECTION SYSTEM CHARACTERISTICS ASSUMED IN SAFETY ANALYSIS

Mode of Protection	Surry Setpoint (Delay time, sec.)
High neutron flux,	
Fraction of Rated	
Low Power Range	0.35(0.5)
High Power Range	1.18(0.5)
Overtemperature ΔT	Variable(6.0*)
Loss of Pump Power	**(1.2)
Low Reactor Coolant Loop Flow,	
Fraction of Full Flow	0.87(0.6)
High Pressurizer Pressure, psia	2425(1.0)
Initiation of Safety Injection flow	
on high Steamline ΔP, psi	150.0(Variable)
on low pressurizer pressure, Psia	1715(Variable)

* This value includes loop and RTD bypass line transport delays, the RTD thermal time constant and electronic signal processing delays.

** Undervoltage trip setpoint not used in analysis.

SECTION 5.0 - QUALIFICATION COMPARISONS

5.1 Introduction

As discussed in earlier sections, the primary Vepco objectives in developing a system transient analysis capability are to provide a basis for the reload core safety analysis and licensing process and to support reactor operations. As verification of this capability, appropriate results and comparisons are provided for a representative series of analyses of licensing and best estimate plant transients. The selection of licensing analyses for presentation was based on 1) consideration of those transients which are thermally limiting and have been most frequently subject to reanalysis during the reload licensing process (e.g. Rod Withdrawal from Power and Complete Loss of Flow); 2) providing a selection of analyses for each of the major categories of initiating events which include changes in reactivity (such as rod withdrawal transients), variations in primary coolant flow rate (such as loss of flow transients) and variations in primary to secondary system heat transfer rates (e.g. Main Steam Line Break); and 3) examination of transients which are both symmetric (such as a Loss of Load) and asymmetric (such as a single pump flow coastdown) with respect to the thermal hydraulic response of the reactor coolant loops.

Comparisons to plant startup flow coastdown test data and the data taken during a reactor cooldown transient experienced at North Anna in 1979 are also provided to illustrate typical best estimate modeling applications.

Comparisons for small and large break Loss of Coolant Accidents (LOCA) and Rod Ejection are beyond the current intended scope of application of Vepco's models and are not presented.

5.2 Verification Against Licensing Analyses

5.2.1 Transients Resulting from Changes in Reactivity

Several transients result primarily from a postulated reactivity change. These transients include an Uncontrolled Control Rod Assembly Withdrawal From a

Subcritical Condition (UCRW from Subcritical), an Uncontrolled Control Rod Assembly Withdrawal at Power (UCRW at Power), Control Rod Assembly Drop, Chemical and Volume Control System Malfunction, Startup of an Inactive Loop, Single Control Rod Assembly Withdrawal at Power and Control Rod Assembly Ejection. The first two accidents were chosen for analysis because they are subject to reanalysis for reload cores based on past Vepco experience. In addition, these two accidents represent a limiting condition for reactivity change rate (UCRW from Subcritical) and DNBR (UCRW from Power) with respect to the other Condition II accidents.

5.2.1.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition Transient - FSAR Analysis

A control rod assembly withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of control rod assemblies resulting in a power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the Reactor Control or Control Rod Drive Systems. Section 14.2.1 of the Surry FSAR (Reference 3) discusses the mitigating automatic safety systems appropriate for this transient in more detail.

The nuclear power response to a continuous reactivity insertion from a subcritical condition is characterized by a very fast rise terminated by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the initial power excursion is of prime importance during a startup incident, since it limits the power to a tolerable level prior to external control action. After the initial power excursion, the nuclear power is momentarily reduced, and then, if the incident is not terminated by a reactor trip, the nuclear power increases again but at a much slower rate.

This is a Condition II event, and the analysis is performed to demonstrate that the DNB criterion for Condition II events is met.

In order to give comparable results, the analysis assumptions used in this investigation are the same as those indicated in Reference 3. The limiting input values and analysis assumptions assumed for this investigation are provided in the Appendix (Item 1a). The Single Loop Model, discussed in Section 3, was used for the analysis.

Figures 5.1 through 5.4 present the results of the analysis using the RETRAN computer code as compared to the FSAR results for nuclear power, average fuel and clad temperature and core heat flux, respectively.

The RETRAN results are based on a single integrated kinetics and thermal-hydraulic calculation. The FSAR results, in contrast, reflect separate core kinetics (power) and heat transfer calculations, performed with two computer codes, with distinct sets of input assumptions designed to conservatively maximize core heat flux. This distinction in analytical approach most likely accounts for the differences in results for the average fuel and clad temperatures.

Note that both calculations result in predicted heat fluxes, and fuel and clad temperatures which are well below steady-state full power values. Therefore large margins to the Condition II DNB limits are maintained throughout the transient.

5.2.1.2 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition Transient - Current Analysis

Due to changes in the calculated limit for the reactivity insertion rate parameters, this transient was reanalyzed for several reload cores. The latest reanalysis was for Cycle 4 of Surry Unit 2.[9] The assumptions used for this analysis are the same as those discussed in Section 5.2.1.1, with the exception of the limiting reactivity insertion rate which was increased to a value of 75 pcm/sec*, and a modification in the trip reactivity (see the Appendix, Item 1b).

The comparison of the vendor reload analysis and RETRAN results is indicated by the excellent agreement for the core heat flux, the limiting analysis result, as reported in the licensing submittal. The RETRAN and vendor reload analyses both

* 1 pcm = 1.0×10^{-5} $\Delta K/K$

yielded peak values of 69% of nominal full power core heat flux. Figures 5.5 through 5.8 provide the complete RETRAN transient response for the appropriate parameters. The vendor transient results are proprietary and are omitted. The transient response is similar to and consistent with the comparisons indicated in Figures 5.1 through 5.4.

5.2.1.3 Uncontrolled Control Rod Assembly Withdrawal at Power Transient - FSAR Analysis

This postulated transient, which is a Condition II event, was analyzed because it is a limiting reactivity perturbation transient with respect to the minimum DNBR criterion and because it is subject to reload reanalysis. This transient is defined as an uncontrolled addition of reactivity to the reactor core while in an at-power condition resulting in a power excursion and an increase in core heat flux. Since the heat extraction from the steam generator remains relatively constant until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, to prevent the possibility of damage to the cladding, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below its limit. The automatic features of the Reactor Protection System, which would prevent core damage in a control rod assembly withdrawal incident at power, are discussed in detail in Reference 3.

In order to obtain conservative results (i.e., minimum DNBRs) for this transient and to provide a consistent comparison, the analysis assumptions are the same as those indicated in the FSAR.[3] These assumptions, and the limiting values assumed for this analysis are provided in the Appendix (Item 2a). The Single Loop Model, discussed in Section 3, was used for this analysis. It should be noted that the Overtemperature Delta T Trip setpoint equation, which is important for this transient, is explicitly modeled in the Single Loop Model using the control system capability in RETRAN.

The FSAR presents the results of this transient for several initial power levels and for various reactivity insertion rates. However, a full range of system parameter transient results is presented only for two analyses from an initial power level of 100%. The two 100% analyses are for differing reactivity insertions rates to demonstrate the protective action of both the High Flux and the Overtemperature Delta T Trip functions. Of the two transients, the more limiting results are for the slow reactivity insertion (2 pcm/sec) which is terminated by the Overtemperature Delta T Trip. Consequently, the analysis used for comparison of the RETRAN and FSAR results assumed a slow reactivity insertion rate of 2 pcm/sec starting from 102% of nominal full power. Analysis results for a range of reactivity insertion rates are discussed in the next section.

Figures 5.9 through 5.12 present the RETRAN results, compared to the FSAR for nuclear power, pressurizer pressure, average coolant temperature and transient DNBR, respectively. The DNBR's were calculated with COBRA IIIC/MIT[1] using input forcing functions of core heat flux, coolant inlet temperature, coolant inlet mass velocity and RCS pressure, all from the RETRAN analysis. Note the similarities in time of trip (Figure 5.9). The decay heat level shown in the FSAR result apparently reflects the conservatism used by the vendor prior to the development of the ANS standard decay heat curves. Note also the similarity in predicted pressure responses in Figure 5.10, including the effects of automatic spray and Power Operated Relief Valve (PORV) actuation. The RETRAN analysis shows, as does the FSAR, that the Condition II DNB criterion is met for this transient.

5.2.1.4 <u>Uncontrolled Control Rod Assembly Withdrawal at Power Transient - Current Analysis</u>

The most recent reanalysis of this accident was required as a consequence of the plugging of steam generator tubes at the Surry Nuclear Power Station.[10] It was determined that steam generator tube plugging would result in lower initial flows with consequently less initial margin to DNB and the need for revision of the constants

associated with the Overpower and Overtemperature Delta Temperature setpoint equation. Consequently, the UCRW at Power transient was reanalyzed to verify that the new setpoint equation constants did in fact result in minimum DNBRs above the appropriate criterion of 1.3. The only information available for comparison purposes from the licensing reanalysis was the minimum DNBR as a function of reactivity insertion rate. An analysis of the transient was performed using the Single Loop Surry RETRAN Model with those assumptions specified in the Appendix (Item 2b), including several modeling changes to reflect the impact of the low flow assumption (i.e. lower flows, lower steam generator heat transfer areas, etc.). Key input parameter values assumed for this analysis are also provided in the Appendix (Item 2b).

The RETRAN results were then used as boundary conditions in the Vepco version of the COBRA IIIC/MIT[1] code. The results of this transient reanalysis are presented in Figure 5.13.

Another analysis of the transient was performed at an initial power level of 62% of nominal full power. The results of this analysis and a comparison to licensing reanalysis results are provided in Figure 5.14. RETRAN results were generated with and without the assumption of operable steam generator relief valves, as shown. These results show that the RETRAN/COBRA analysis supports the conclusion provided by the licensing reanalysis, i.e., that the updated setpoint equation constants are sufficient to provide margin to the Condition II DNBR limit for reactor operation with 90% or greater of thermal design flow.

5.2.2 Transients Resulting from Changes in Primary System Flowrate

Several FSAR transients result primarily from the loss of Reactor Coolant System (RCS) flow and the corresponding decreased transfer of heat from the reactor core. Transients in this category include the Loss of Reactor Coolant Flow (partial and complete) and the Locked Rotor transients. The Complete Loss of Reactor Coolant Flow Transient was chosen for comparative analysis because it has been subject to reanalysis for reload cores based on past Vepco experience. In addition, it is the most

Figure 5.1
NUCLEAR POWER
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

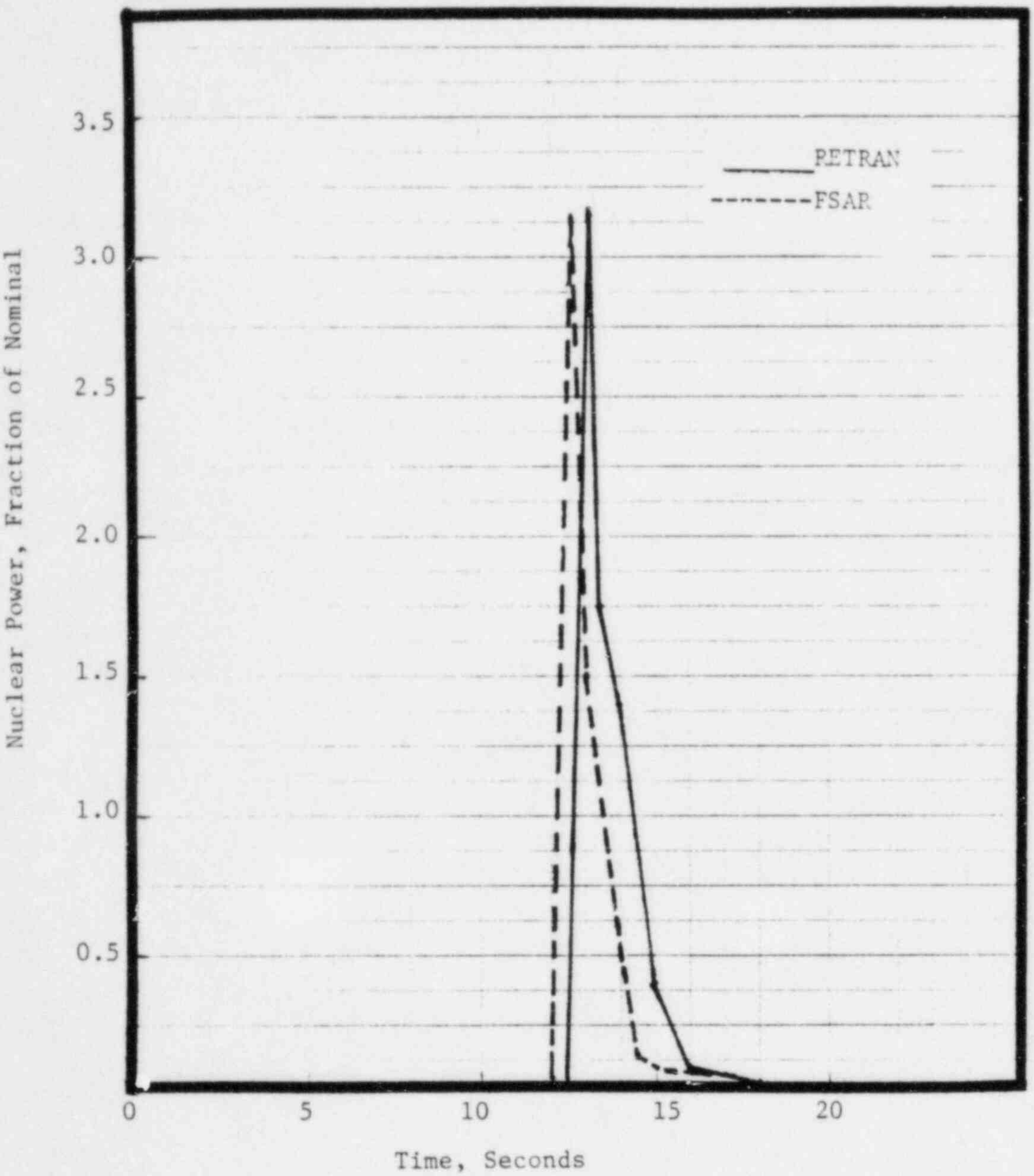


Figure 5.2

AVERAGE FUEL TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

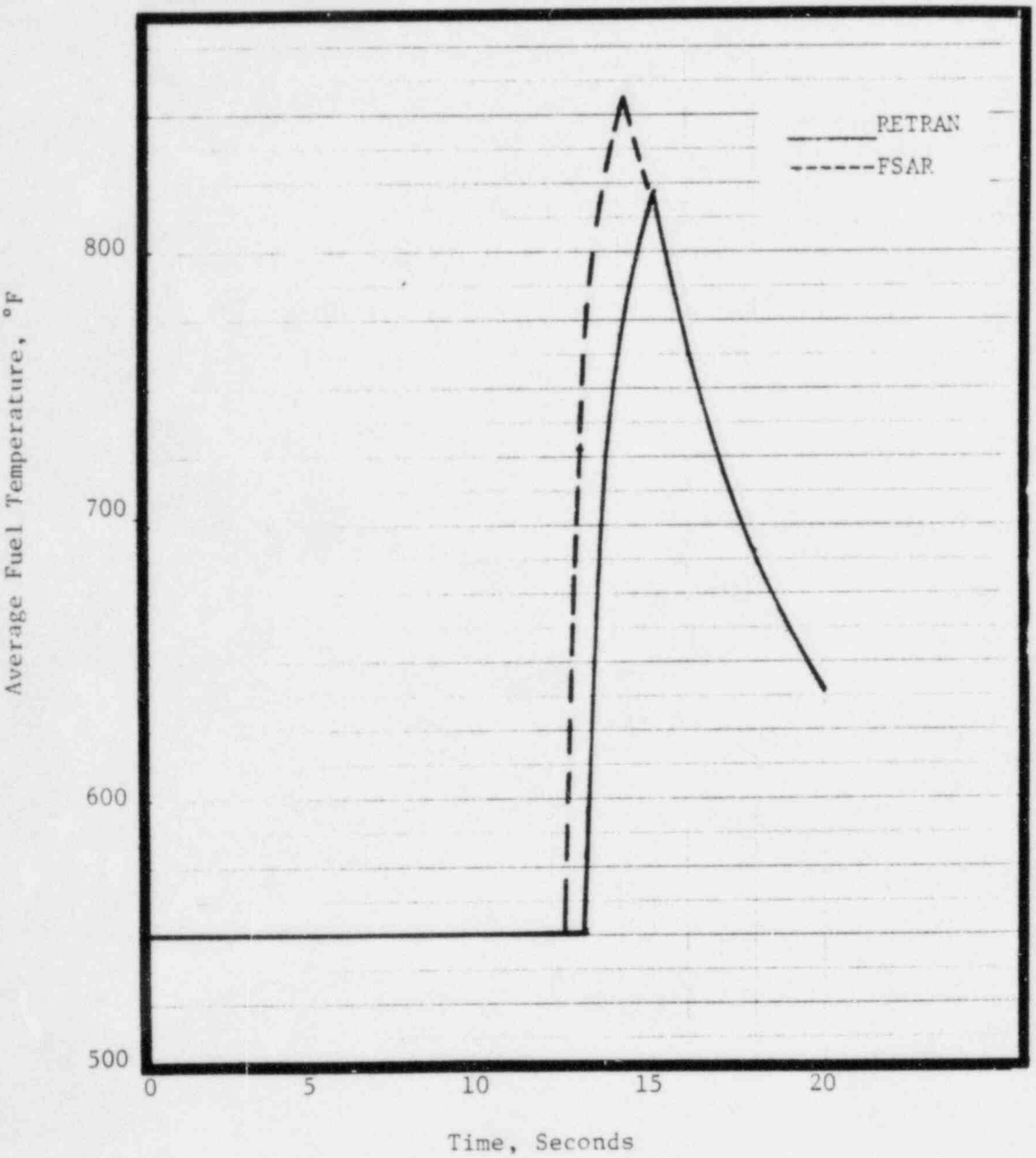


Figure 5.3
AVERAGE CLAD TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

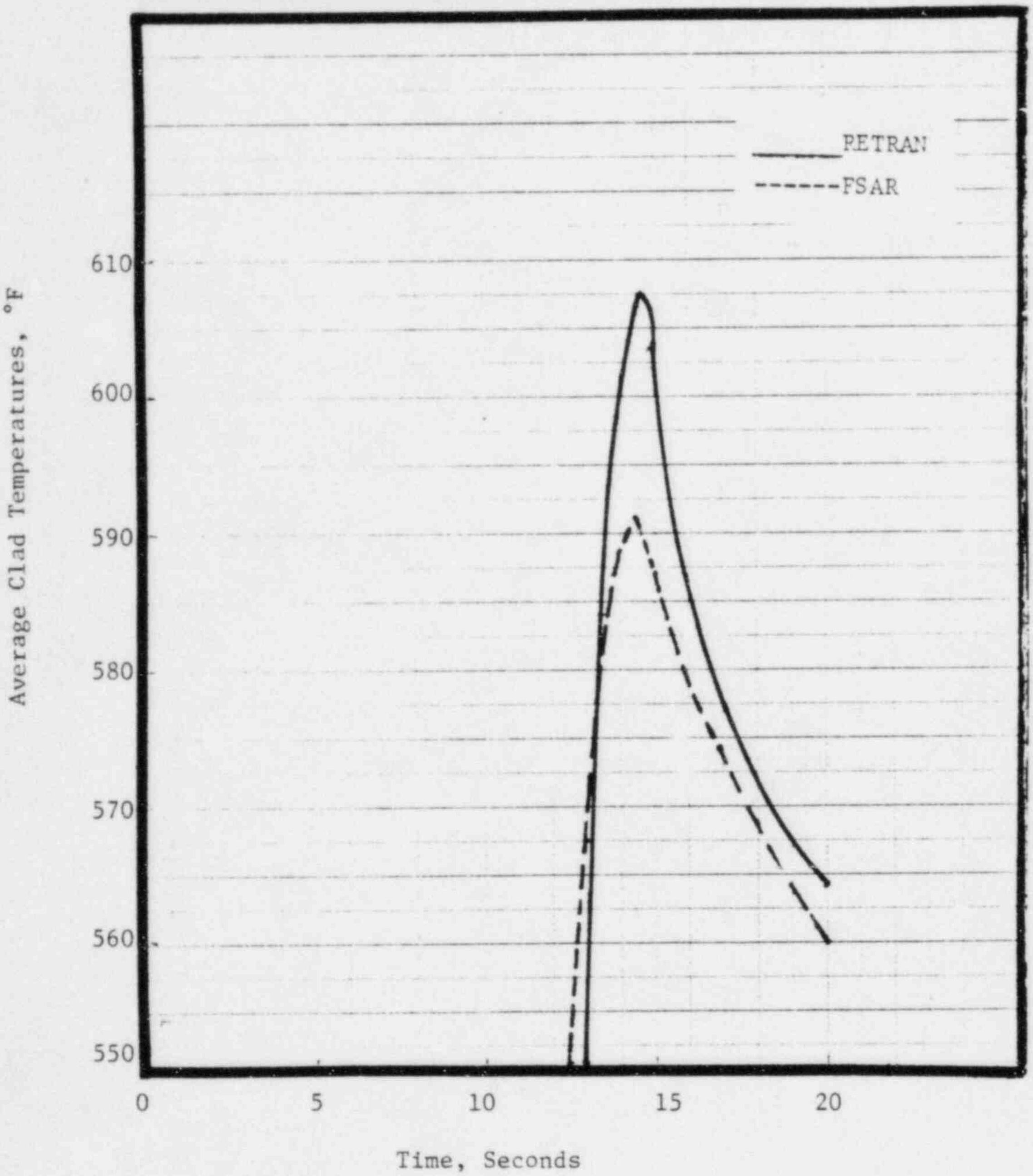


Figure 5.4

CORE HEAT FLUX
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

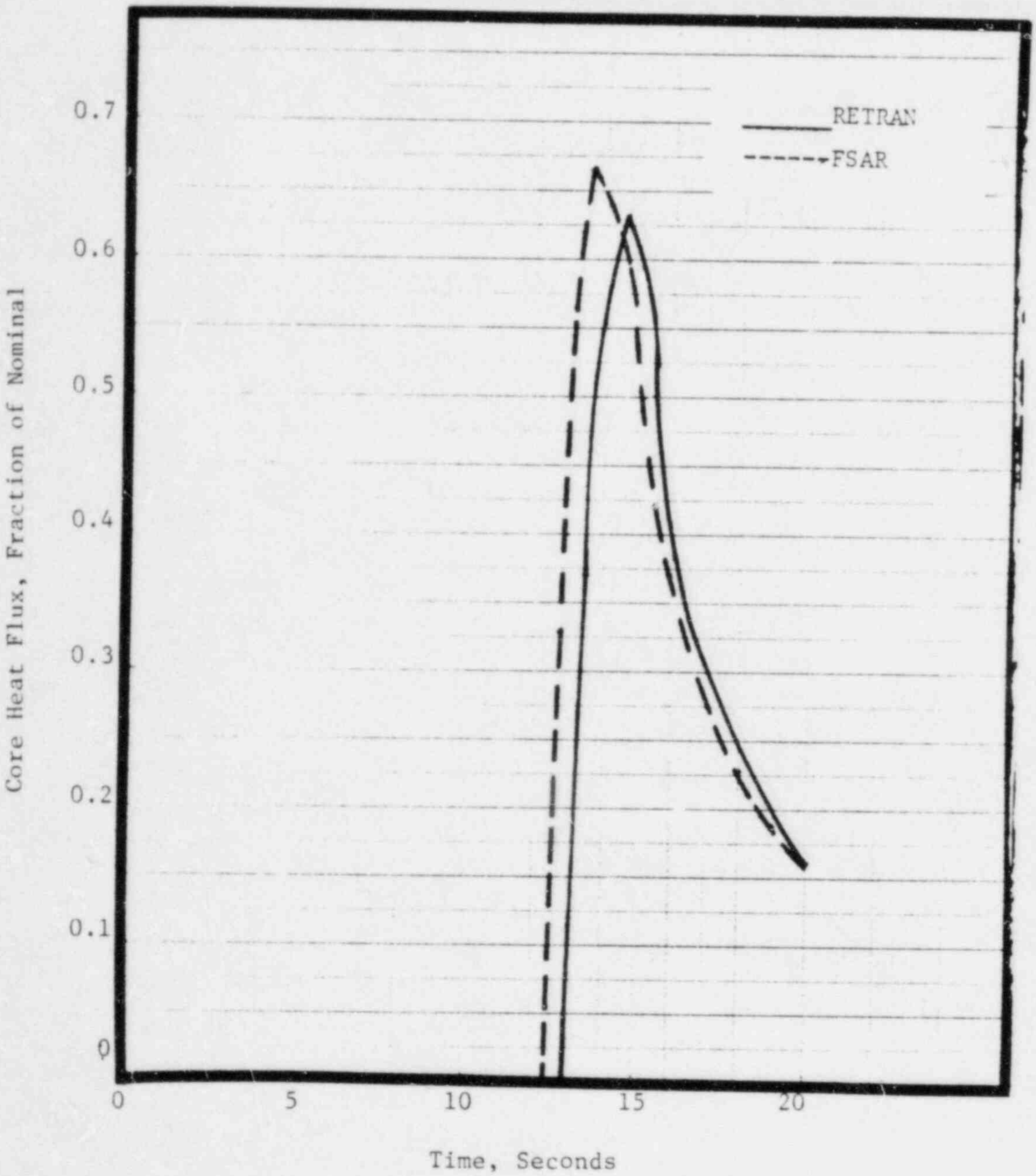


Figure 5.5

NUCLEAR POWER
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
SURRY 2 CYCLE 4 REANALYSIS

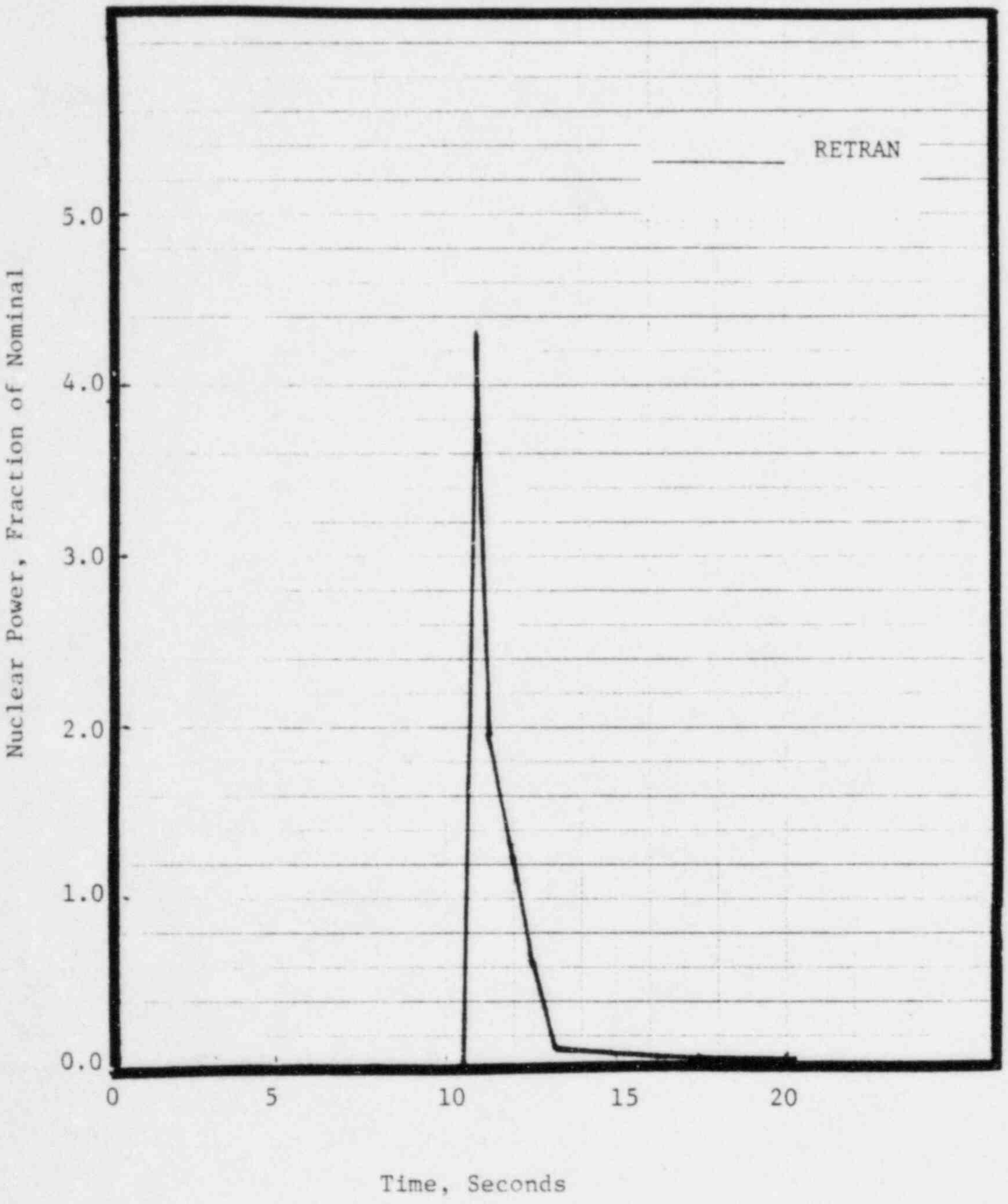


Figure 5.6

AVERAGE FUEL TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
SURRY 2 CYCLE 4 REANALYSIS

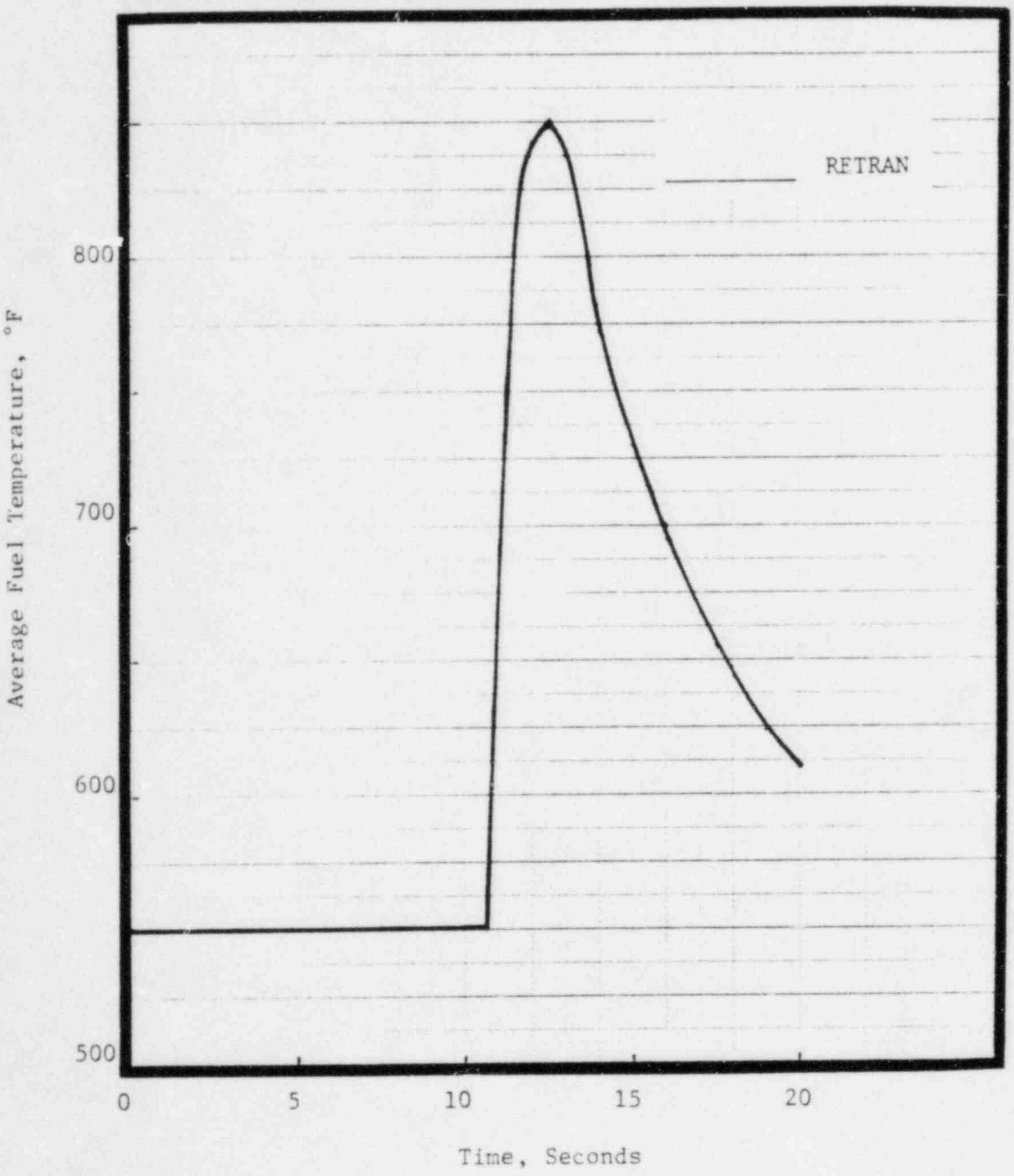


Figure 5.7

AVERAGE CLAD TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
SURRY 2 CYCLE 4 REANALYSIS

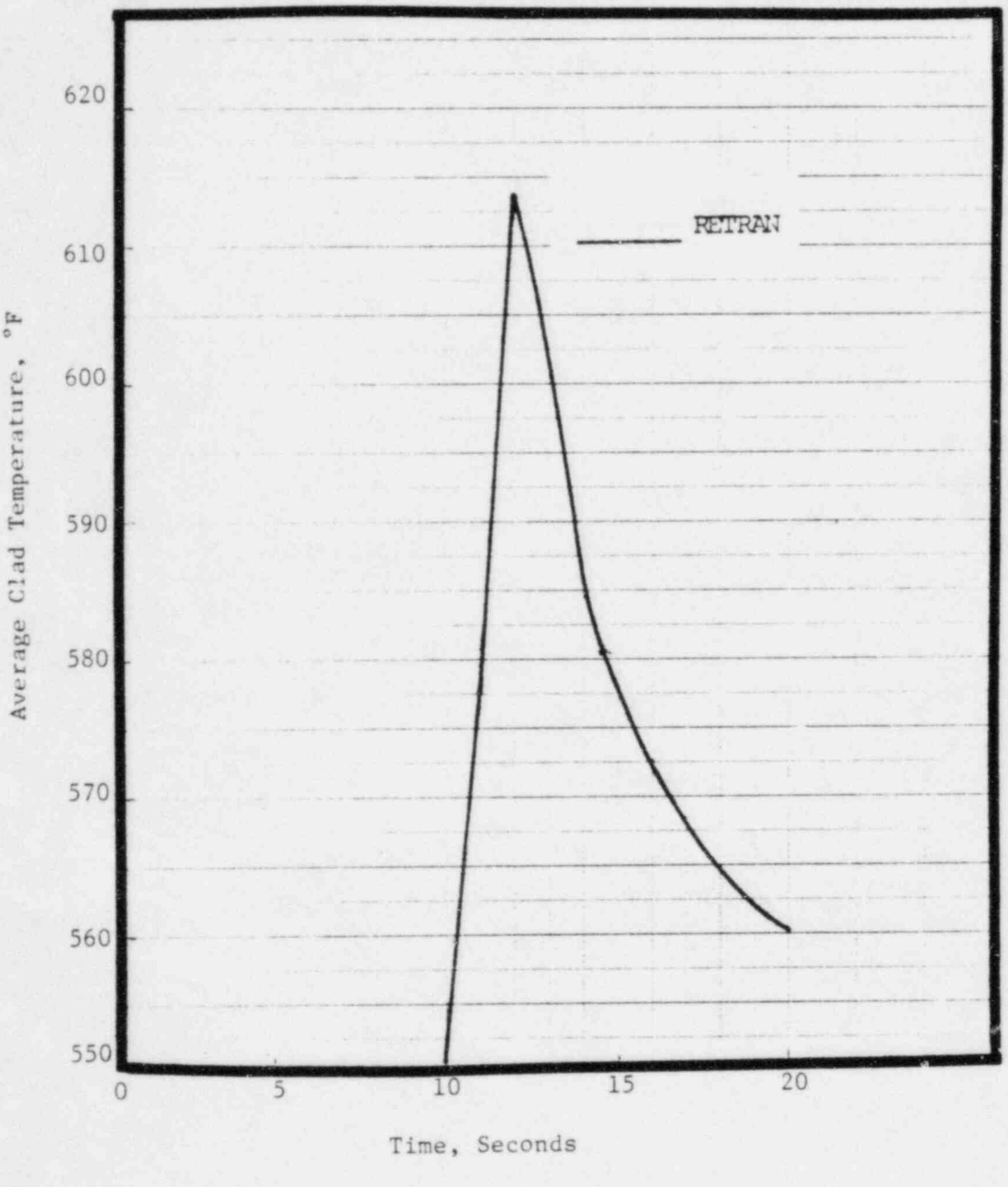


Figure 5.8

CORE HEAT FLUX
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
SURRY 2 CYCLE 4 REANALYSIS

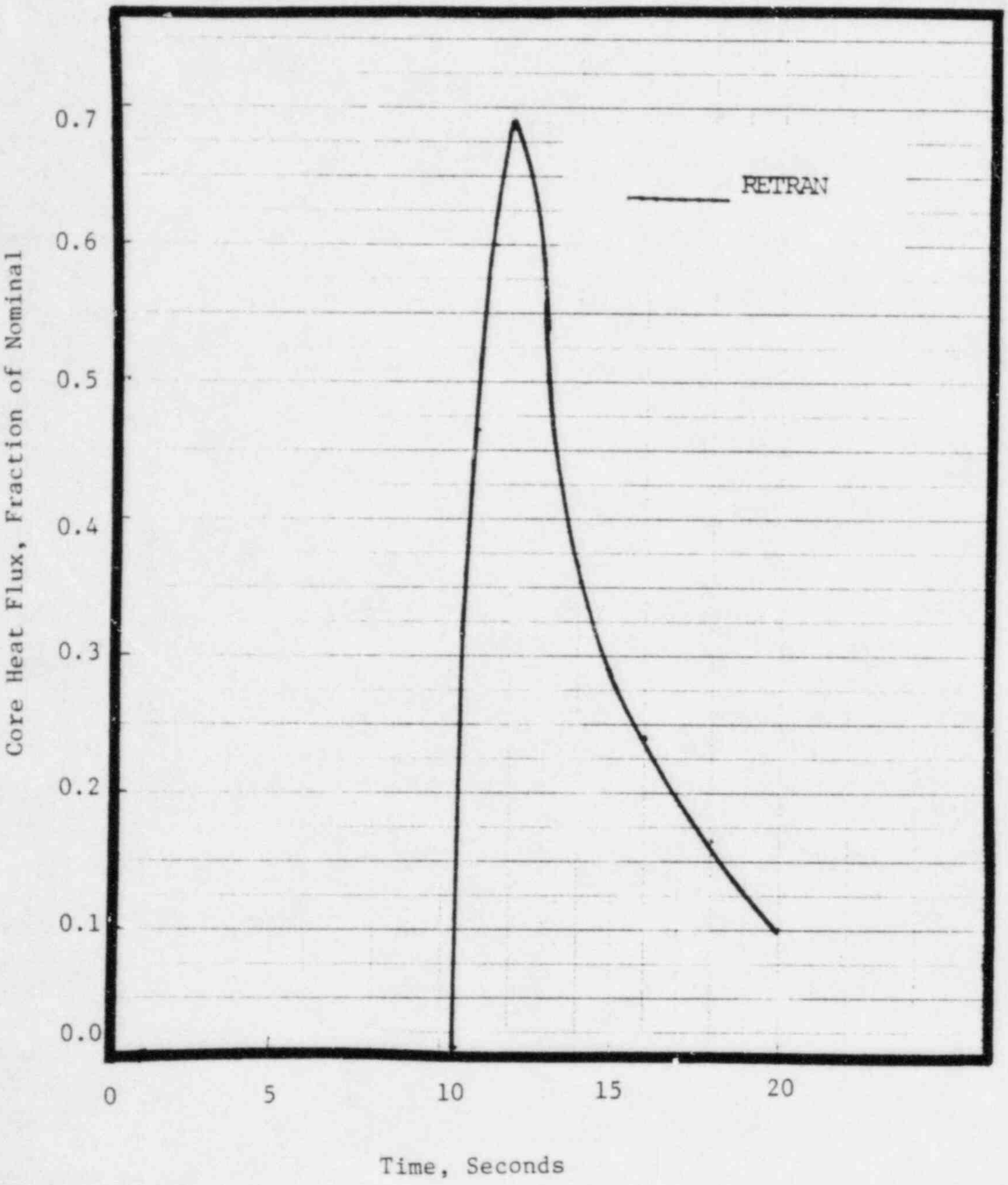


Figure 5.9
NUCLEAR POWER
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

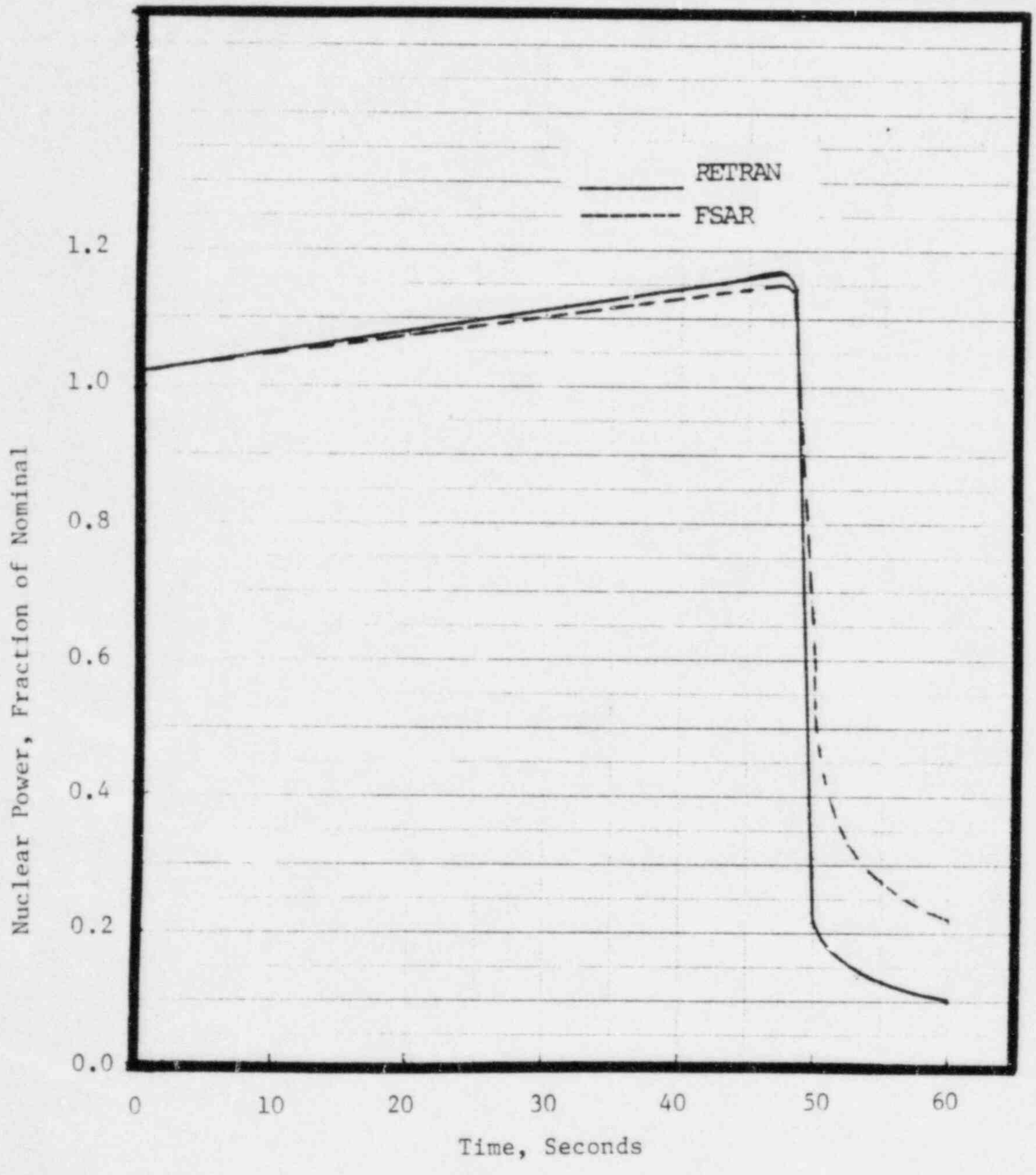


Figure 5.10
PRESSURIZER PRESSURE
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

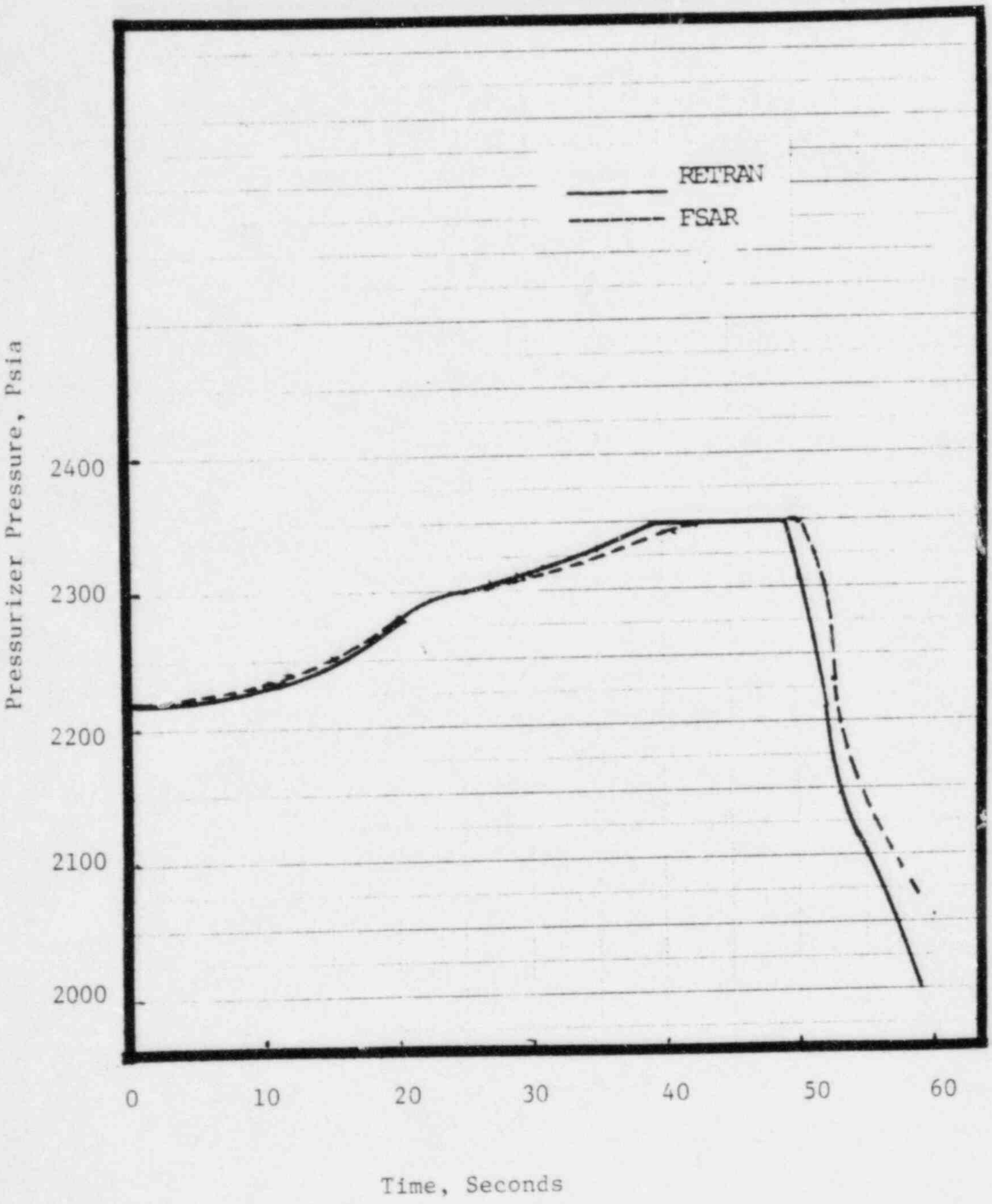


Figure 5.11

AVERAGE COOLANT TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

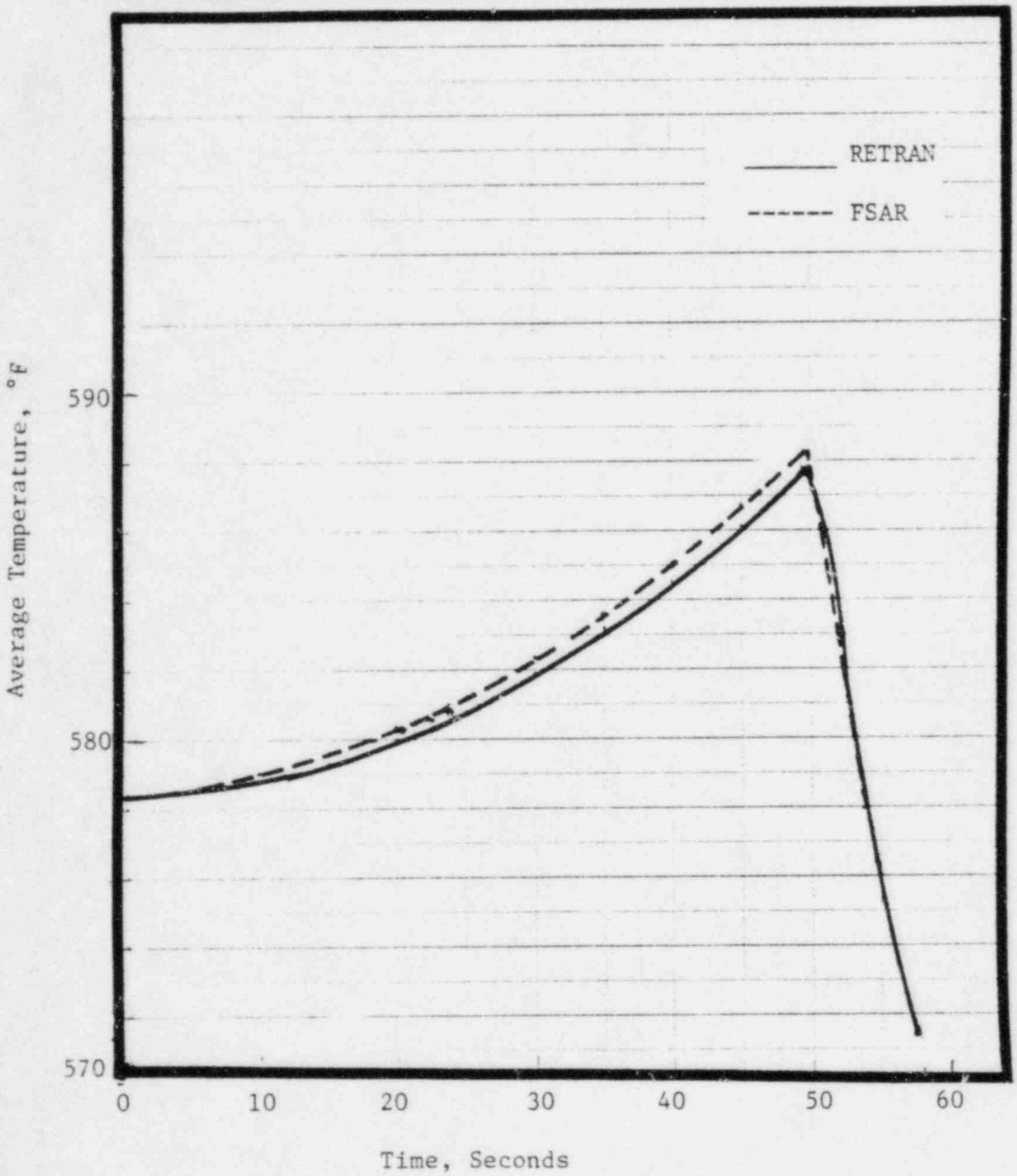


Figure 5.12

DNB RATIO
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

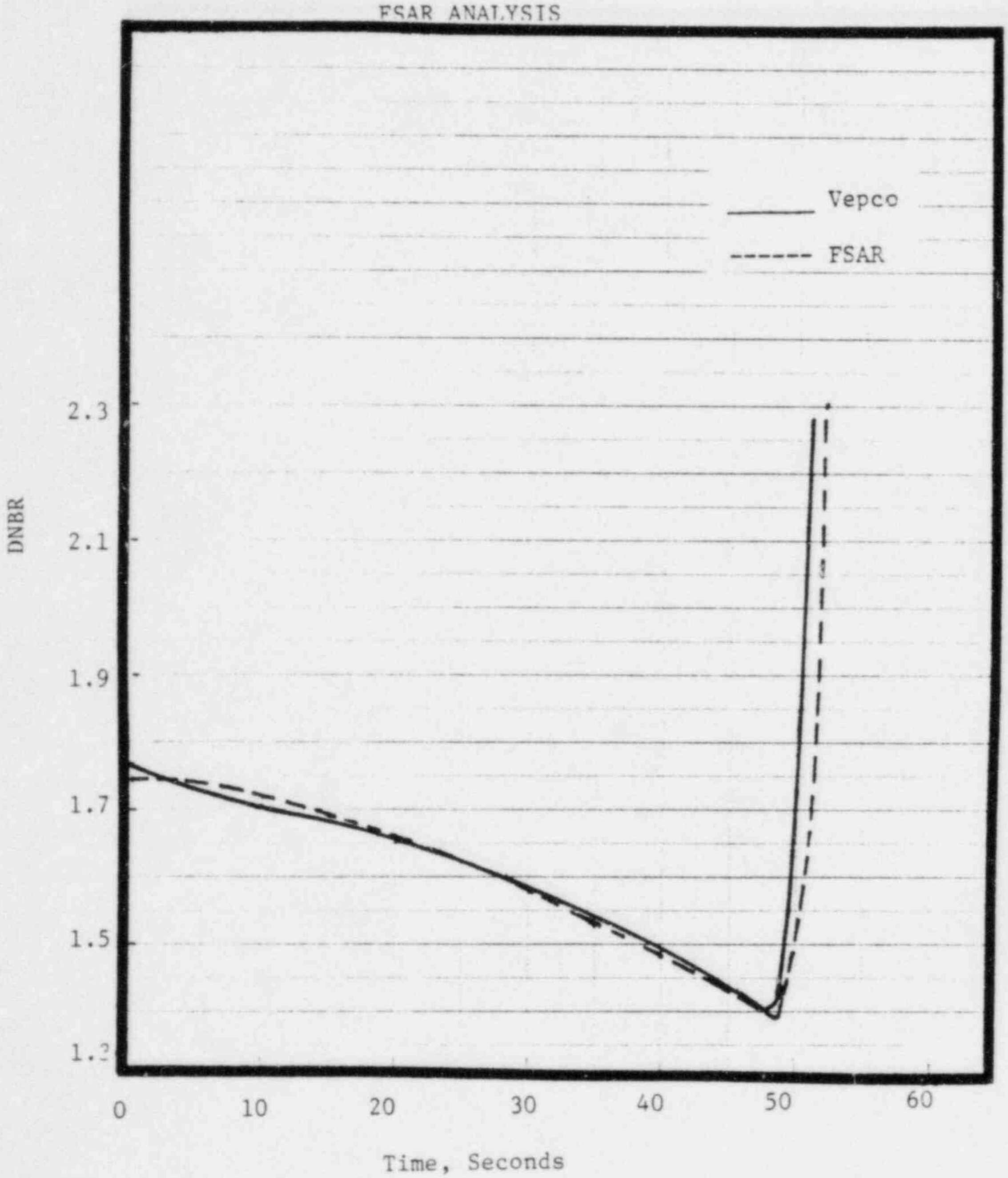


Figure 5.13

VARIATION OF MINIMUM DNBR WITH
REACTIVITY INSERTION RATE
ROD WITHDRAWAL FROM 102% POWER
STEAM GENERATOR TUBE PLUGGING REANALYSIS

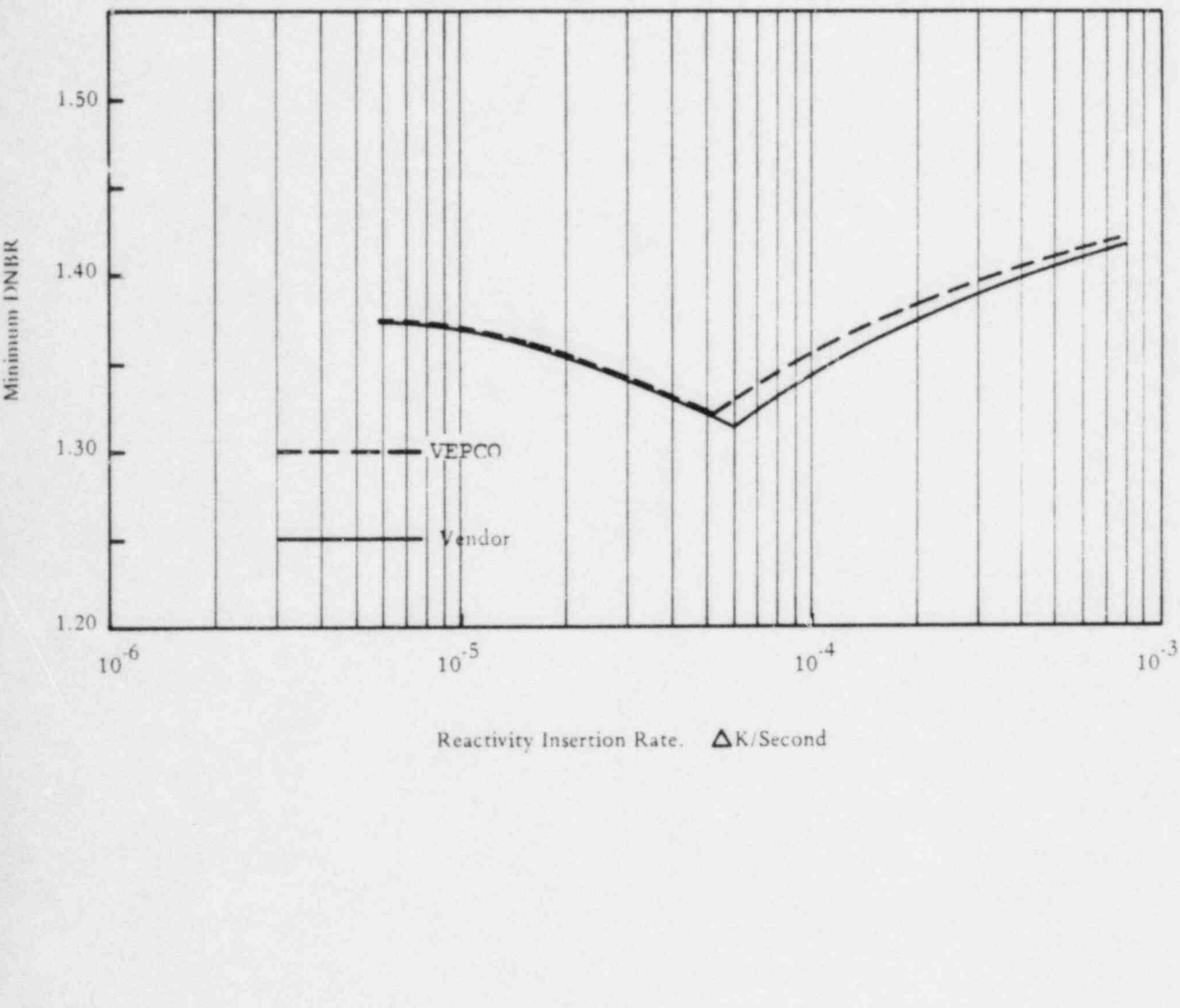
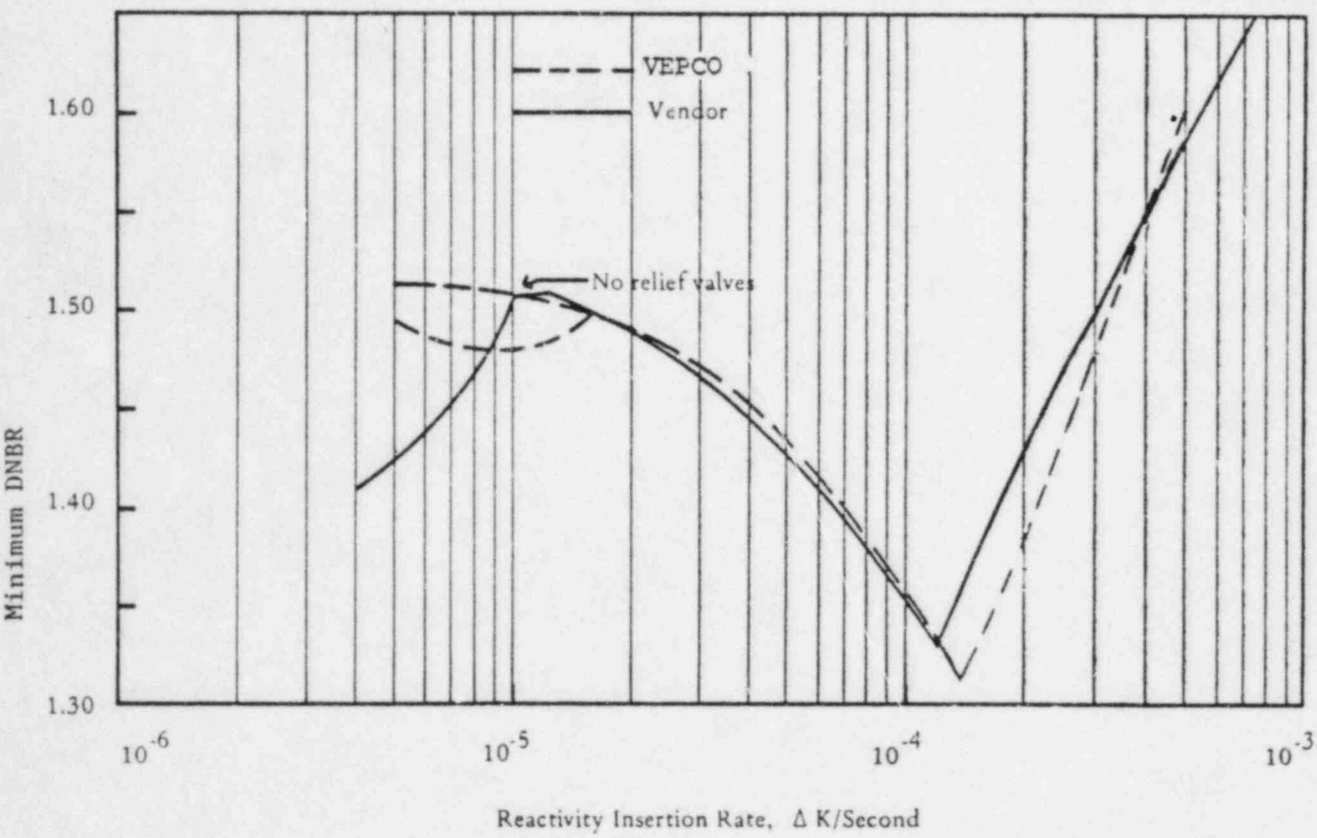


Figure 5.14

VARIATION OF MINIMUM DNBR WITH
REACTIVITY INSERTION RATE
ROD WITHDRAWAL FROM 62% POWER
STEAM GENERATOR TUBE PLUGGING REANALYSIS



severe credible loss of flow condition. The Partial (one-pump) Loss of Flow was analyzed to provide qualification of the Two Loop Model.

5.2.2.1 Complete Loss of Flow Transient - FSAR Analysis

This postulated transient, which is a Condition III event, is defined as the simultaneous loss of electrical power to all reactor coolant pumps at full power resulting in a rapid RCS flow reduction and consequent coolant temperature increase with the possibility of Departure from Nucleate Boiling (DNB) if the reactor is not tripped promptly. The necessary protection action to preclude DNB is discussed in more detail in Reference 3.

The conservative assumptions used in the RETRAN analysis, which are delineated in the Appendix, (Item 3a) are the same as those presented in Reference 3. Specific limiting parameter values assumed are also provided in the Appendix. The RETRAN analysis was performed with the Single Loop Model discussed in Section 3.

Figures 5.15 through 5.18 present the results of the comparisons for this transient for flow coastdown, nuclear power, core heat flux and DNBR, respectively.

As discussed previously, the DNBRs were calculated with the Vepco version of the COBRA IIIC/MIT computer code using boundary conditions obtained from the RETRAN analysis. The minimum DNBR predicted by the Vepco analysis was 1.50 and compares very favorably with the value of 1.46 reported in the FSAR analysis. Time of occurrence of minimum DNBR also compared well and was approximately 2.3 seconds for both analyses. Thus the RETRAN/COBRA results support the FSAR conclusion that, while Complete Loss of Flow is a Condition III transient, the Condition II DNB criterion is met for this event.

5.2.2.2 Complete Loss of Flow Transient - Current Analysis

The Complete Loss of Flow transient has had to be reanalyzed in the past for the Surry plants. The most recent analysis was required as a consequence of the plugging of steam generator tubes.[10] The tube plugging resulted in reduced primary coolant flow and less initial margin in DNB. Since the Loss of Flow transient was

potentially affected, the transient was reanalyzed to verify the continued acceptability of the results.

An analysis of the transient was performed with RETRAN using the assumptions specified in the Appendix (Item 3b). The specific parameter values assumed for this analysis are also provided in the Appendix. The Single Loop Model, as modified to reflect the effects of steam generator tube plugging (lower flows, steam generator heat transfer areas, etc.), was used for the analysis. A conservatively low value of initial flow was assumed in the analysis.

The comparative results of this reanalysis are provided in Figures 5.19 through 5.22. Figure 5.19 shows the comparison of pump coastdown for the respective analyses, and Figure 5.20 compares the nuclear power response. Figure 5.21 presents the results for core average heat flux, and the DNBR response using the RETRAN/COBRA methodology is compared in Figure 5.22 to the prediction reported in the licensing reanalysis. The Vepco predicted minimum DNBR again agrees well in both magnitude and time of occurrence to the licensing reanalysis results and confirms that the Condition II DNB criterion is met for this event.

5.2.2.3 Partial Loss of Flow Transient - FSAR Analysis

In addition to the Complete Loss of Flow transient, discussed in the two previous sections, various Partial Loss of Flow Accidents may be postulated, in which power is lost to one or more reactor coolant pumps, with the remaining pumps continuing to operate at full speed. Such a transient would result from failure of a single pump bus. Since this does not constitute loss of line voltage or frequency, no credit is taken for the direct reactor trip on low voltage. Instead, protection of the core is provided by a reactor trip actuated by low measured reactor coolant flow in any primary coolant loop.

Since this transient involves unbalanced reactor coolant loop flow rates, the Surry Two Loop Model is used for the RETRAN analysis. The case analyzed assumes initial operation of all reactor coolant loops, with a subsequent loss of pump power in a

single loop. Specific parameter values and initial conditions assumed for this analysis are shown in the Appendix (Item 4). The low coolant flow trip setpoint and delay time assumed are consistent with Table 4.1.

The results of the RETRAN analysis are compared to the corresponding FSAR[3] results in Figures 5.23 to 5.26 for core flow, nuclear power, core average heat flux and DNBR, respectively.

As in previous DNB analyses presented in this section, the Vepco curve was generated with the Vepco version of COBRA IIIC/MIT, using input forcing functions from the two loop RETRAN analysis. Again, the Vepco results confirm the conclusion that the Condition II DNB criterion is met for this transient.

5.2.3 Change in Primary to Secondary Heat Transfer

The remaining types of non-LOCA perturbations analyzed for a nuclear plant in a FSAR are characterized by changes in primary system pressure and temperature resulting from changes in primary to secondary heat transfer. Accidents in this category would include Excessive Heat Removal Due to Feedwater System Malfunction, Loss of External Electrical Load, Excessive Load Increase Incident, Loss of Normal Feedwater, Loss of all AC Power to the Station Auxiliaries, Turbine Generator Unit Overspeed and Main Steam Line Break. The majority of these transients are nonlimiting and have not been reanalyzed since the FSAR. However, the Main Steam Line Break and Loss of Load transients have required reanalysis as a result of core reloads and for that reason were chosen for comparative analysis. In addition, the Main Steam Line Break transient reanalysis required a multiloop capability and served to qualify the Two Loop Model discussed in Section III. Finally, the Feedwater System Malfunction transient was analyzed to further demonstrate the capability of the Single Loop Model to represent a secondary side initiated transient.

5.2.3.1 Loss of External Electrical Load Transient - FSAR Analysis

The Loss of Load transient is defined as the loss of external electrical load which may result from an abnormal variation in network frequency, or other adverse

Figure 5.15

FLOW COASTDOWN
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

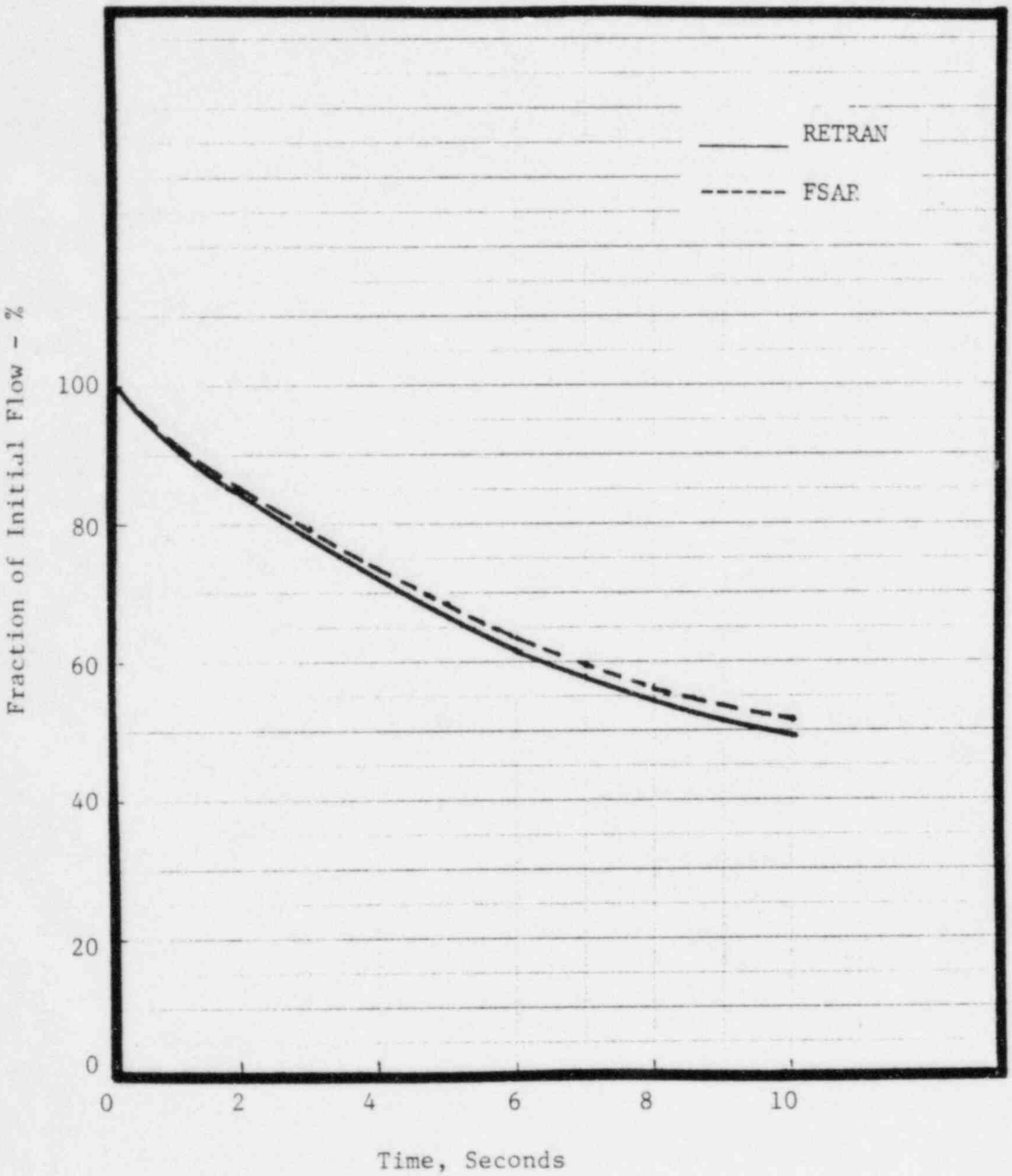


Figure 5.16

NUCLEAR POWER
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

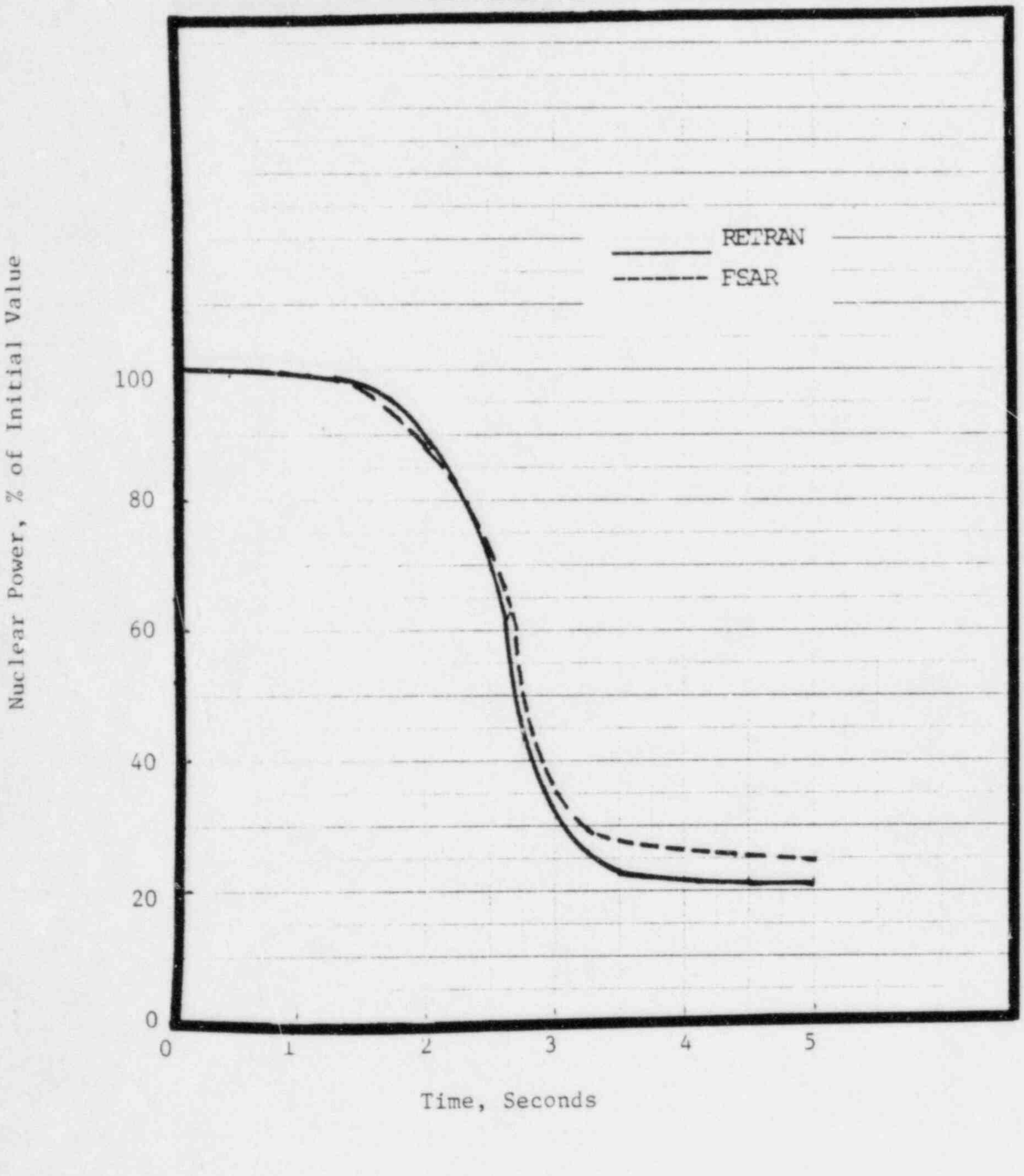


Figure 5.17

AVERAGE HEAT FLUX
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

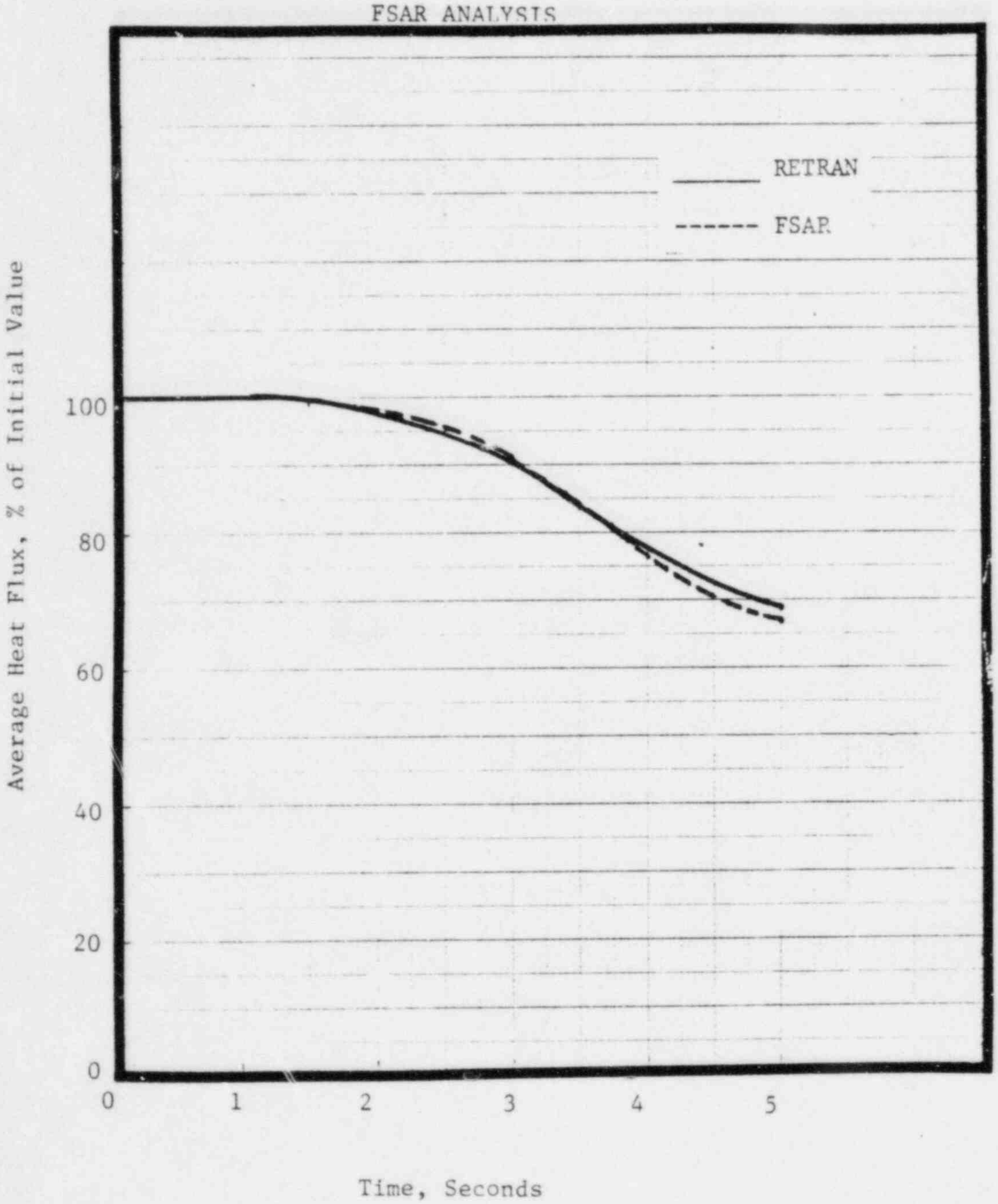


Figure 5.18

DNB RATIO
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

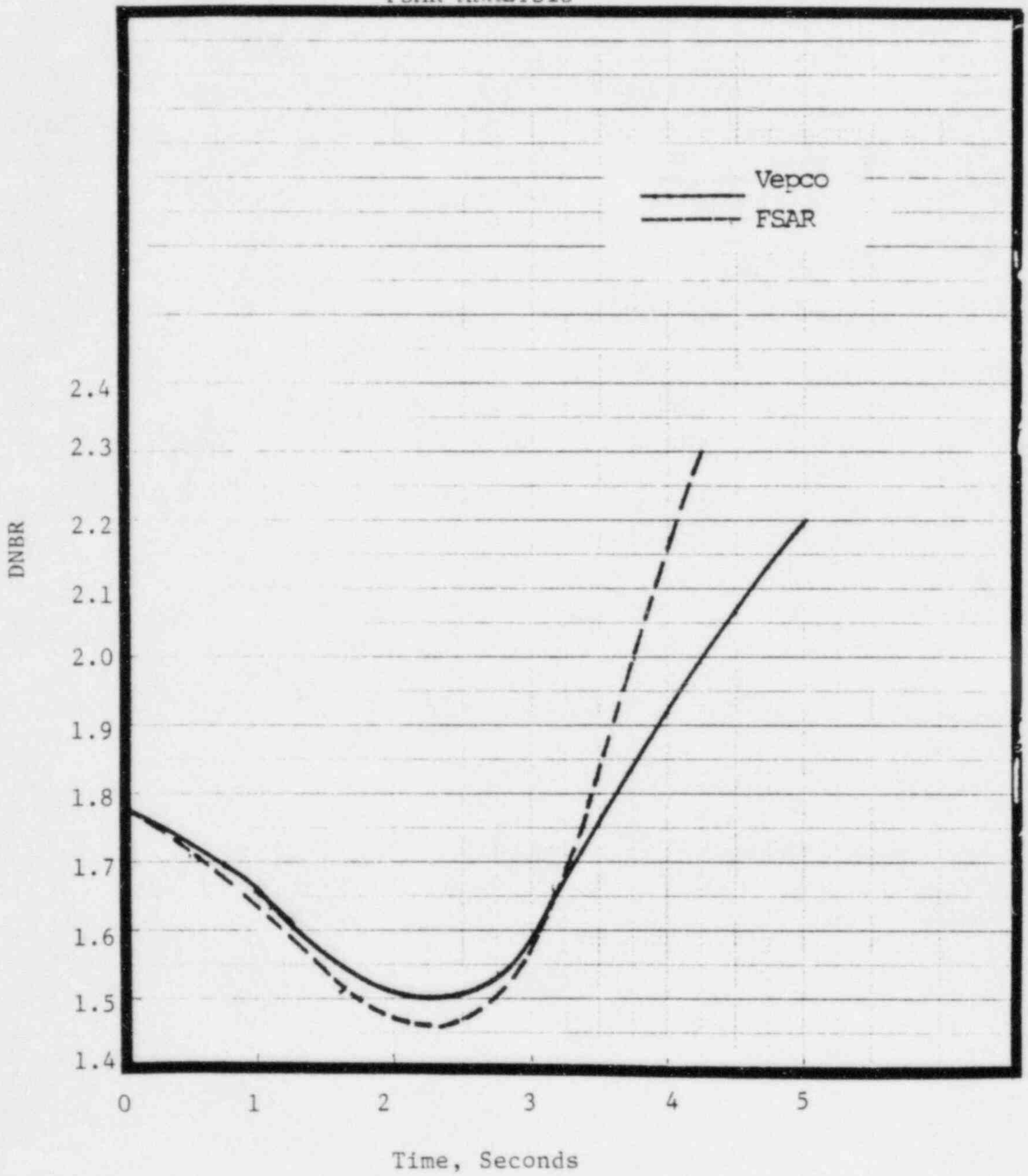


Figure 5.19

FLOW COASTDOWN
COMPLETE LOSS OF FLOW TRANSIENT
STEAM GENERATOR TUBE PLUGGING REANALYSIS

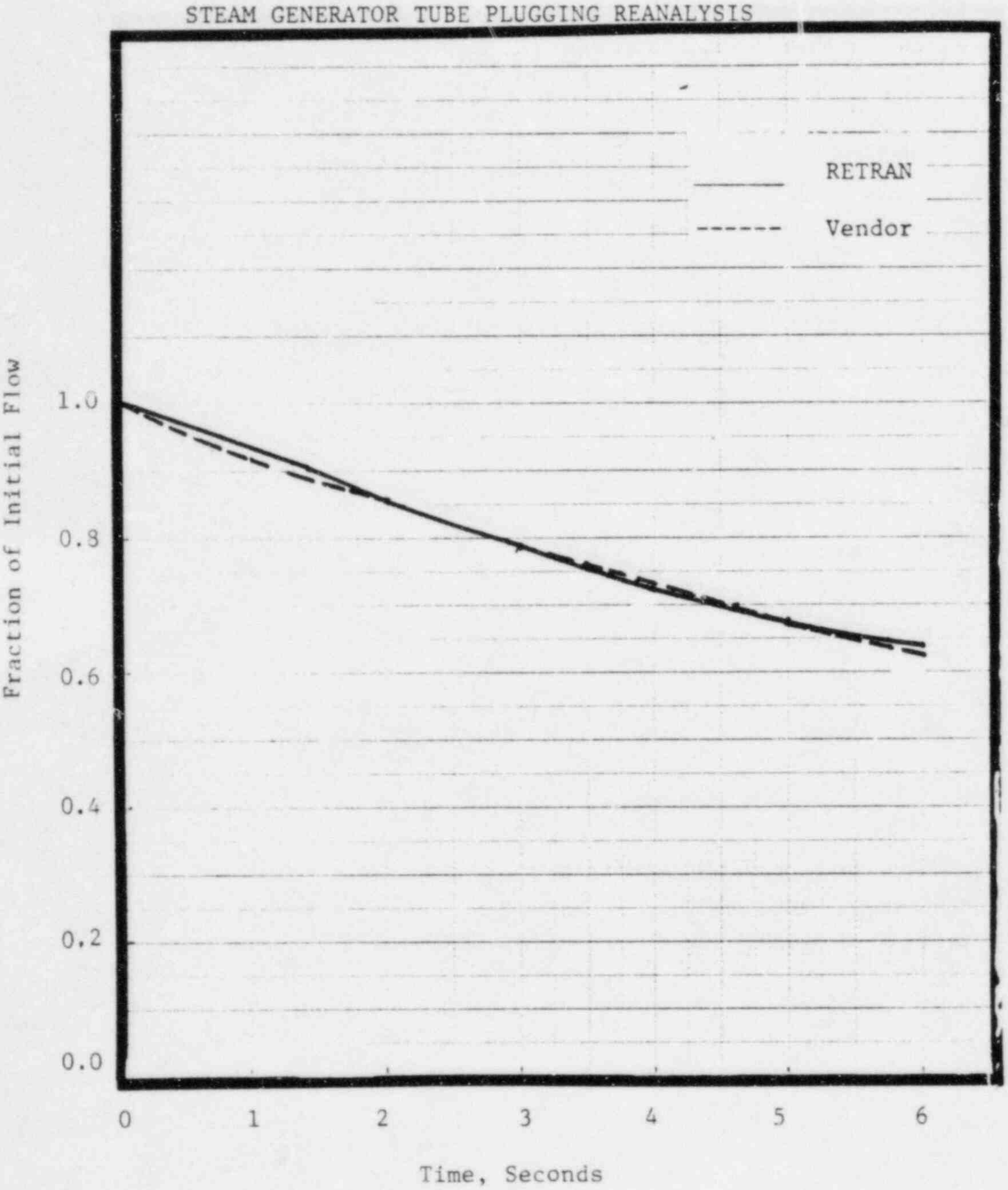


Figure 5.20

NUCLEAR POWER
COMPLETE LOSS OF FLOW TRANSIENT
STEAM GENERATOR TUBE PLUGGING REANALYSIS

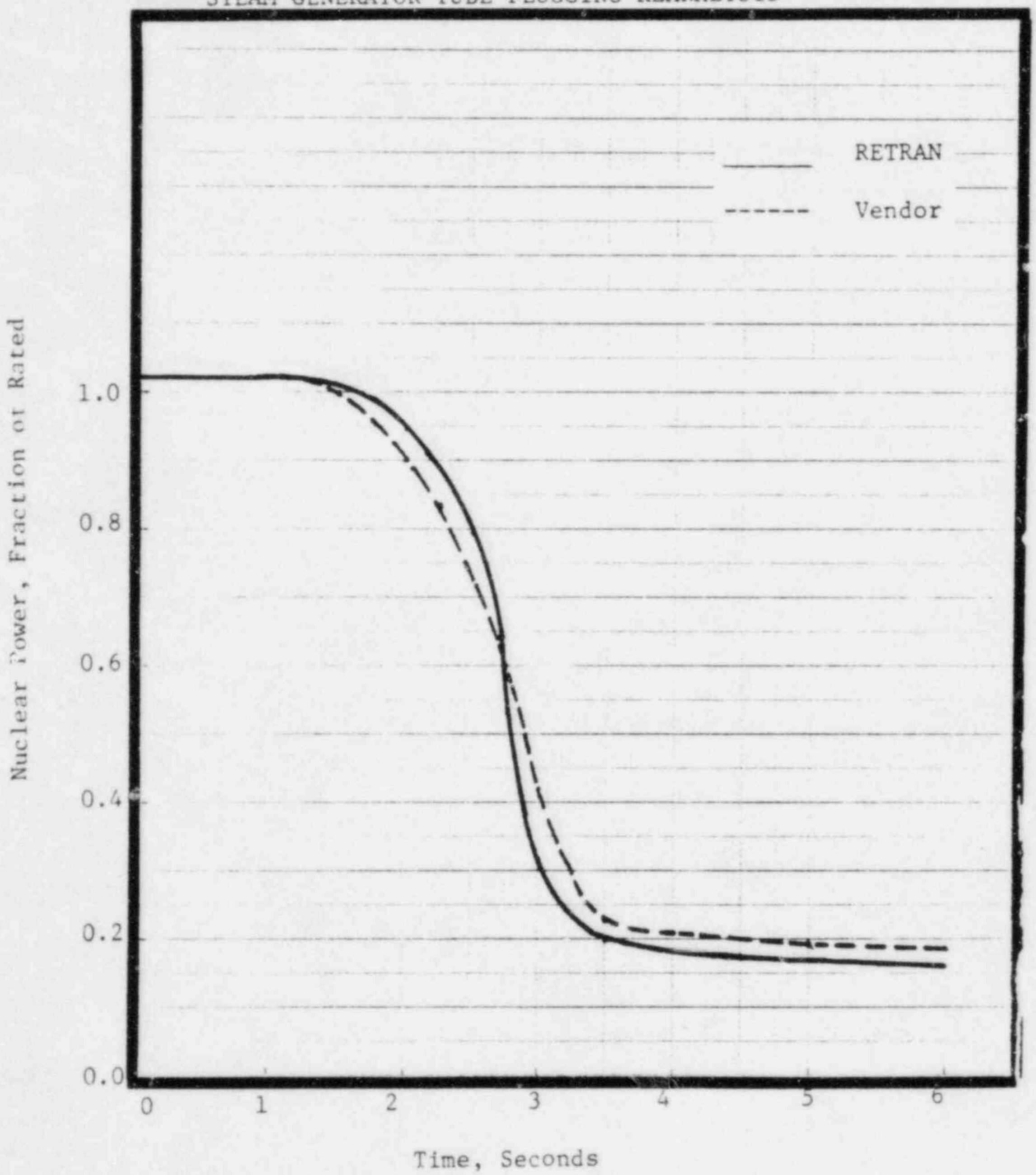


Figure 5.21

AVERAGE HEAT FLUX
COMPLETE LOSS OF FLOW TRANSIENT
STEAM GENERATOR TUBE PLUGGING REANALYSIS

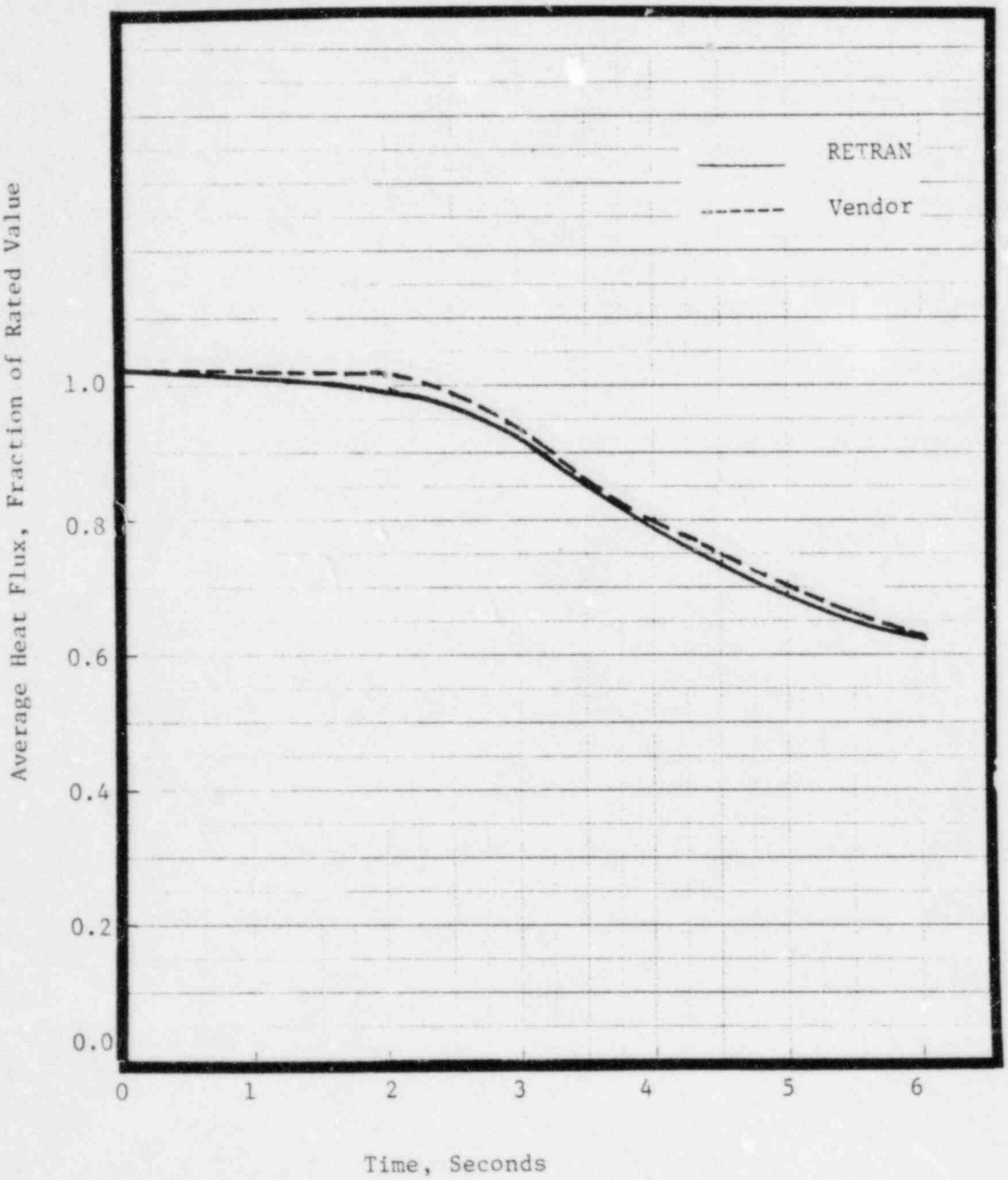


Figure 5.22

DNB RATIO
COMPLETE LOSS OF FLOW TRANSIENT
STEAM GENERATOR TUBF PLUGGING REANALYSIS

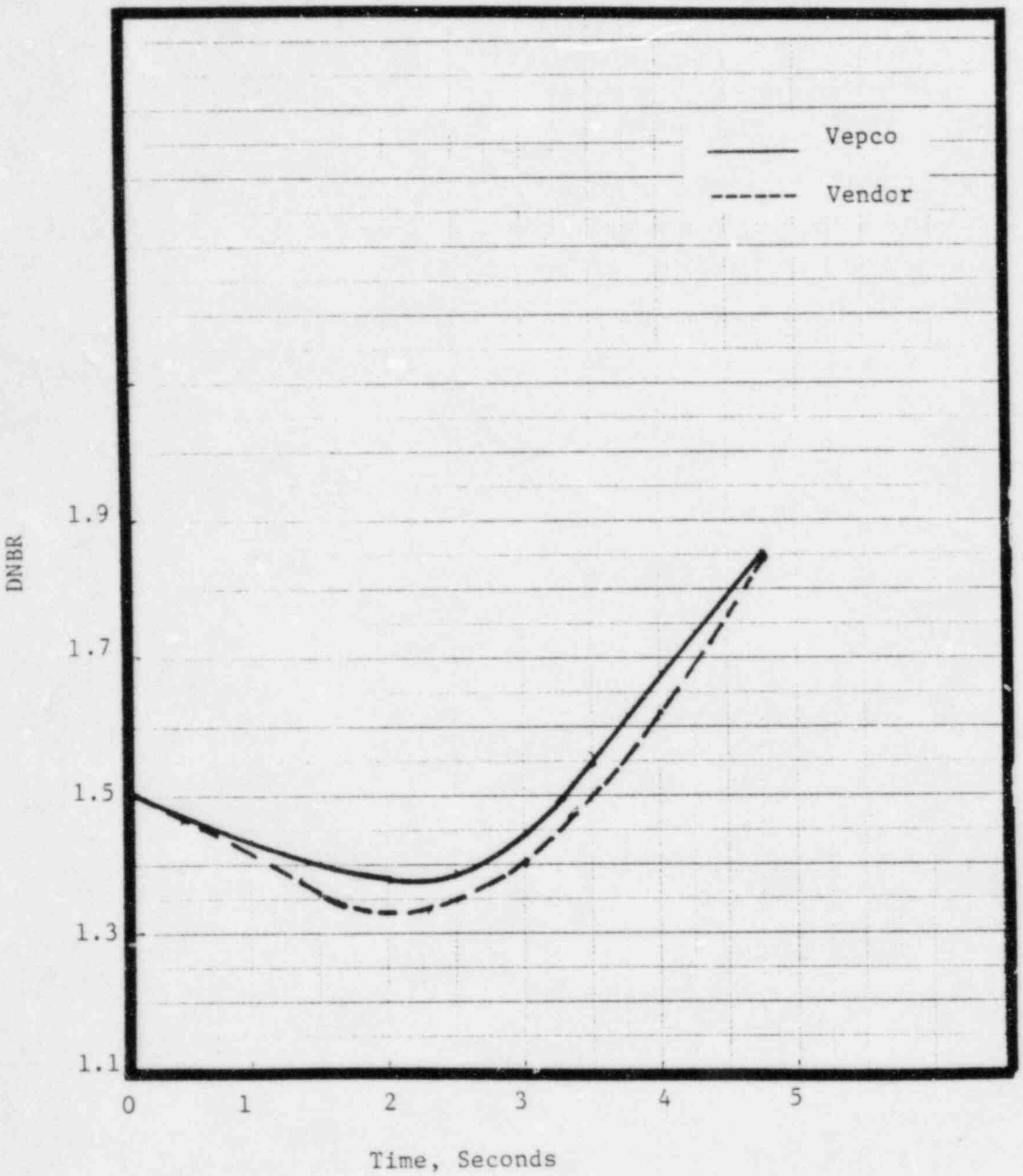


Figure 5.23

CORE FLOW COASTDOWN
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

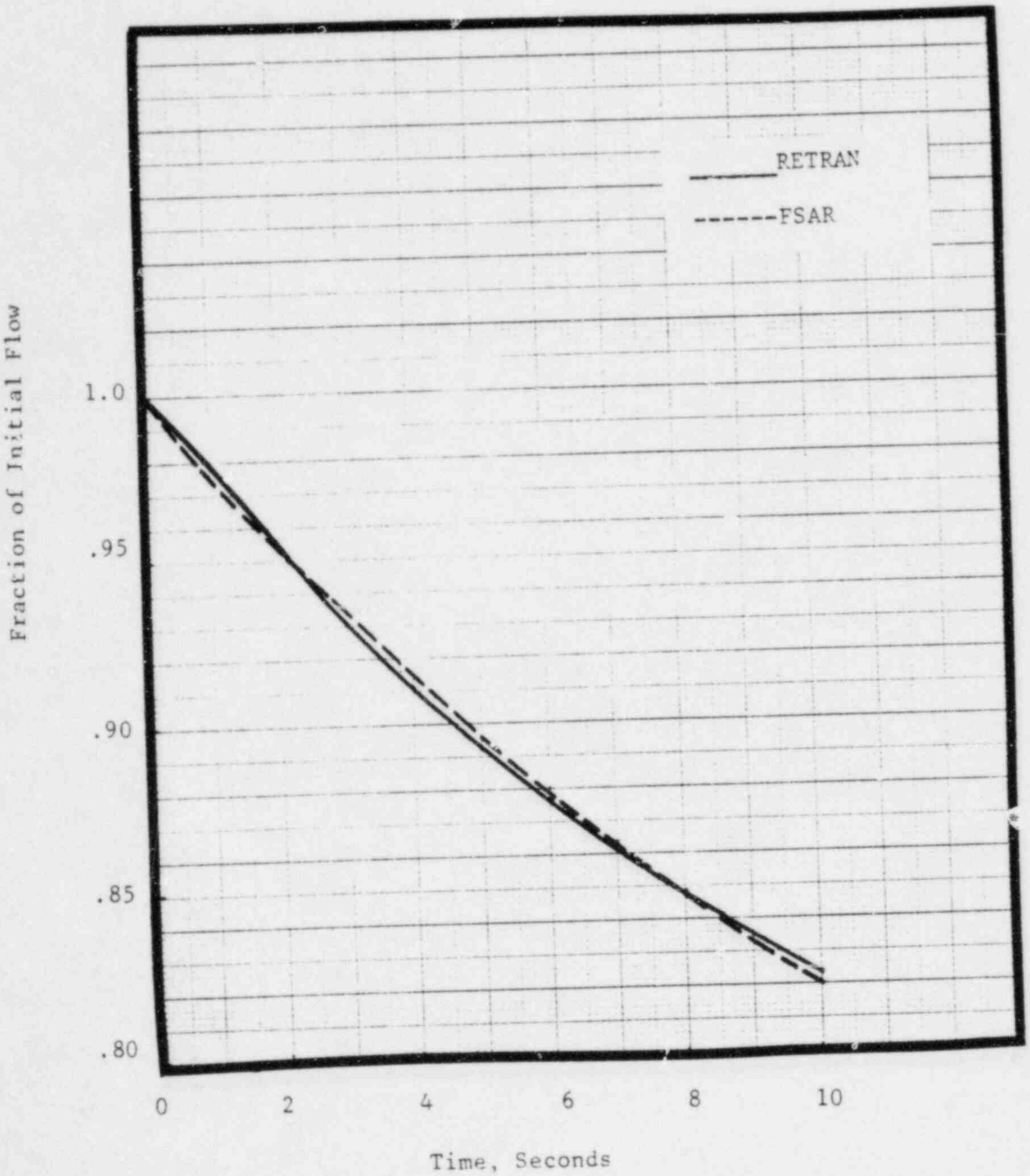


Figure 5.24

NUCLEAR POWER
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

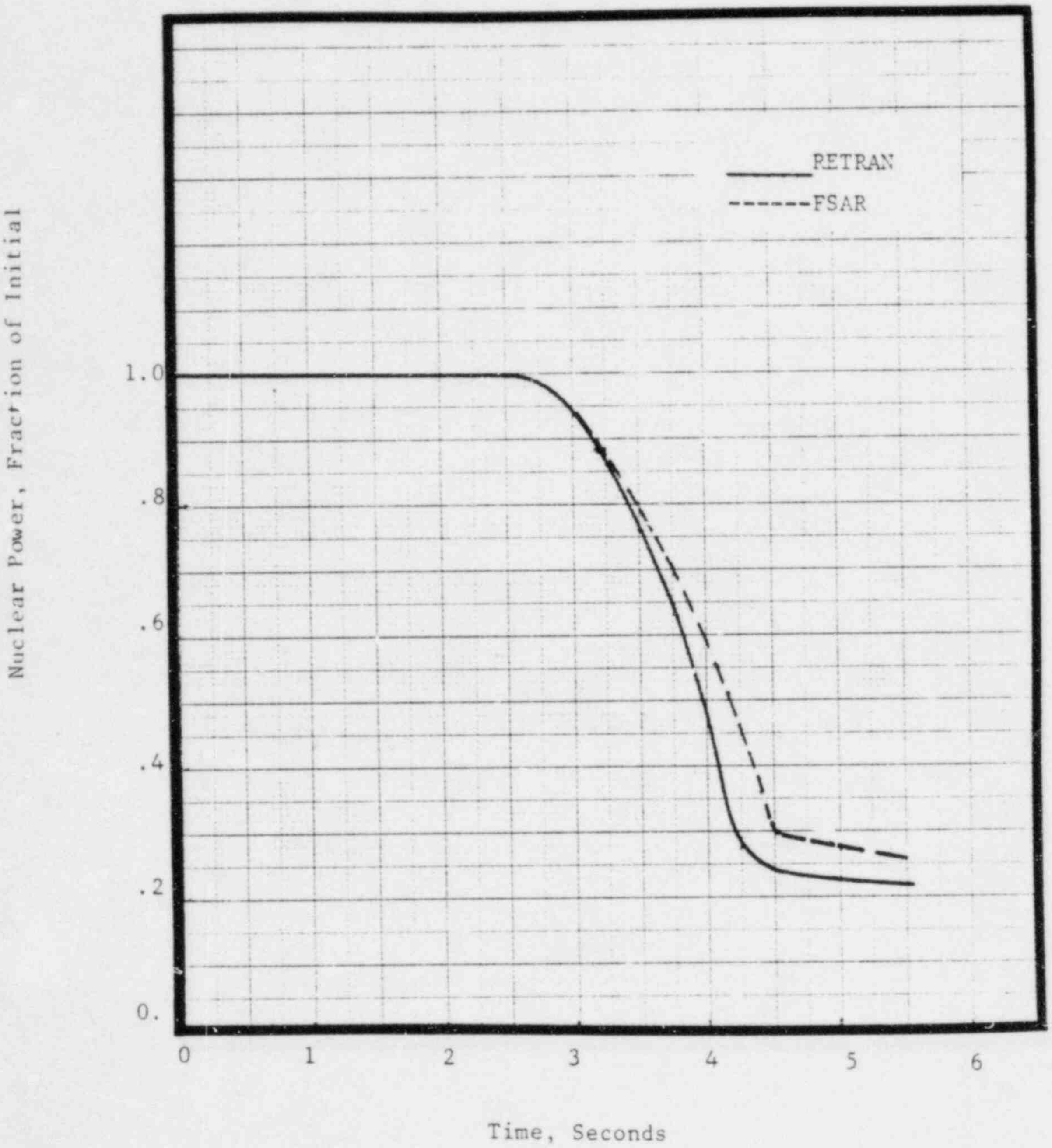


Figure 5.25

CORE AVERAGE HEAT FLUX
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

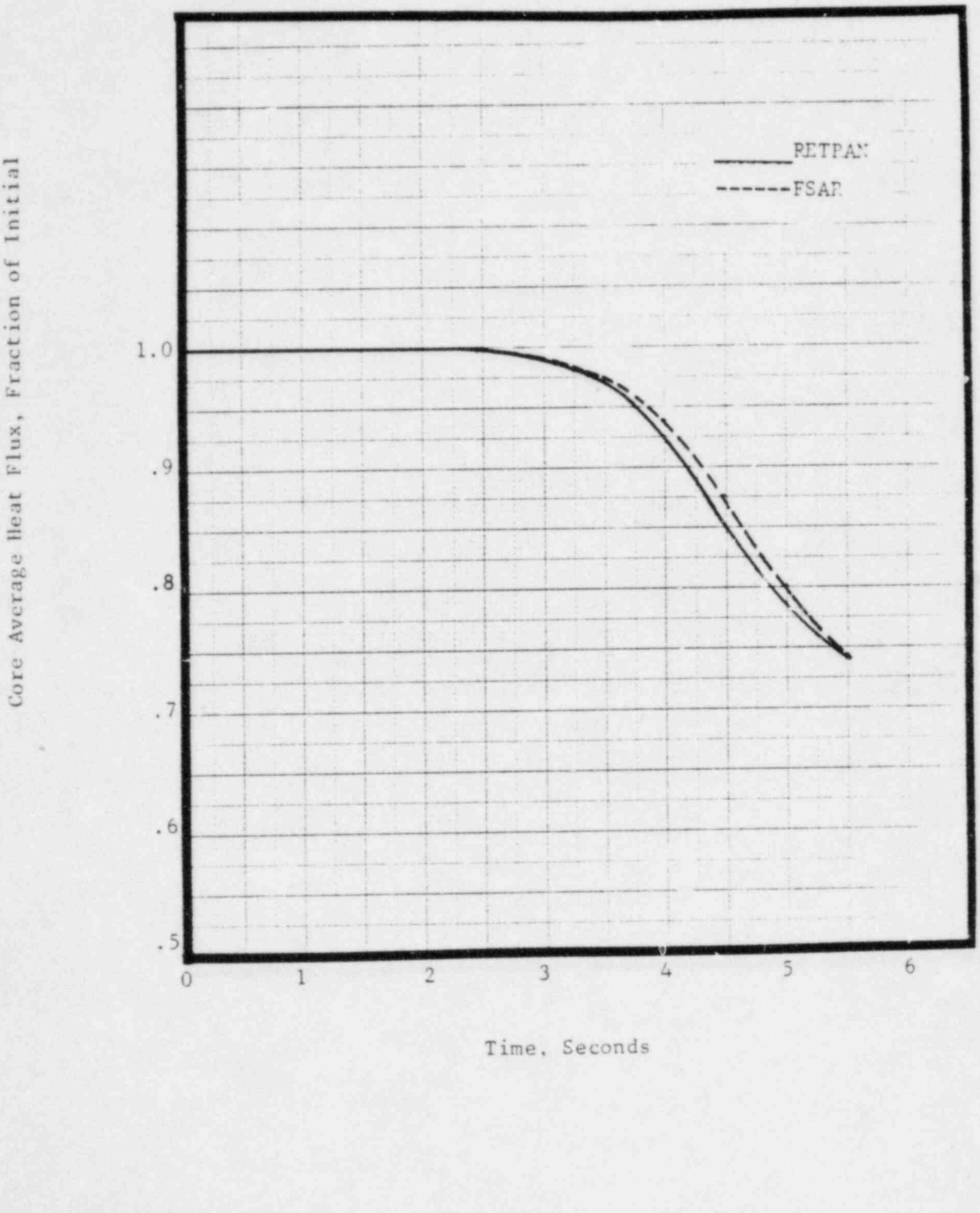
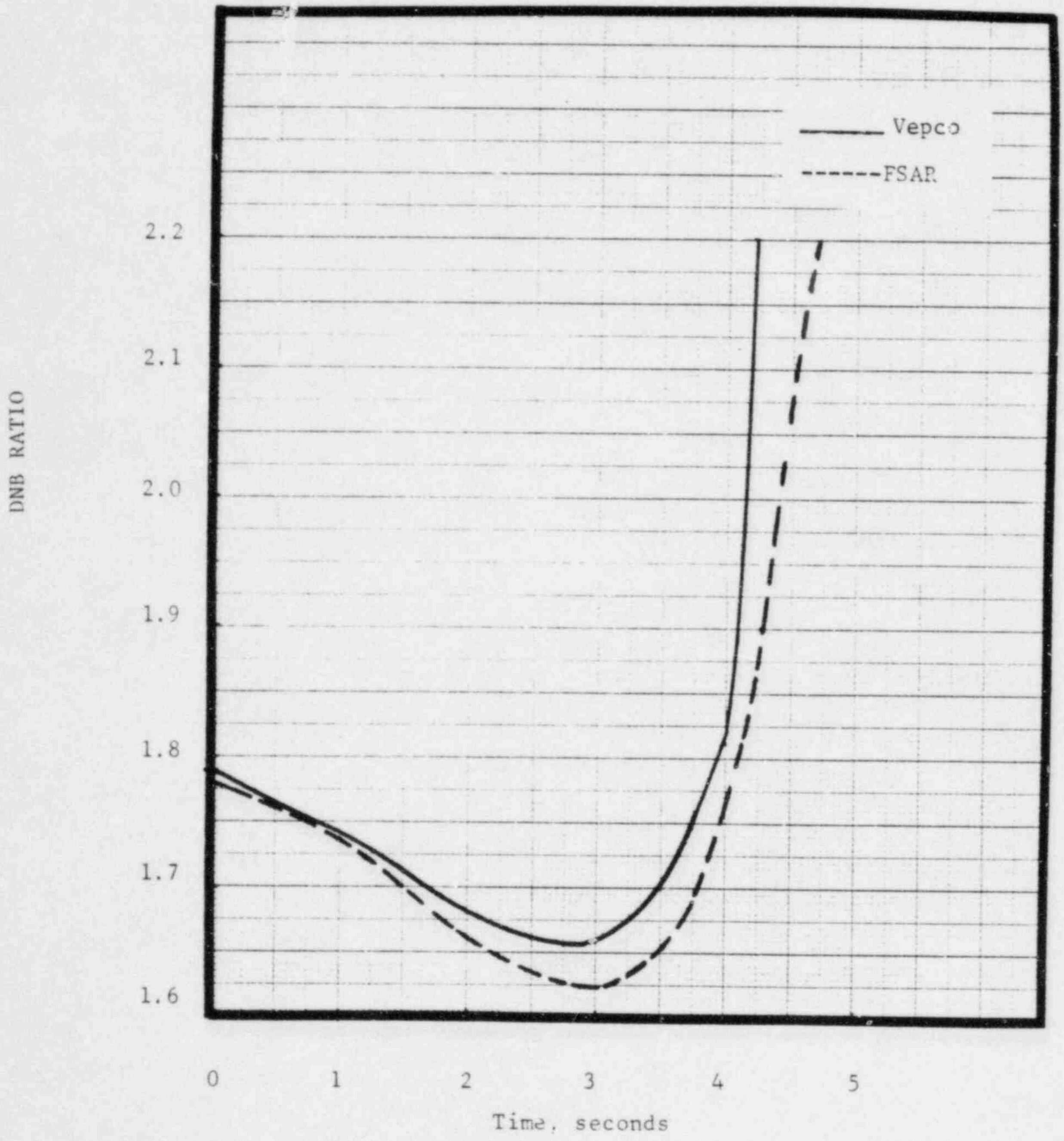


Figure 5.26

DNB RATIO
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS



network operating conditions and is considered a Condition II event. The interaction of the mitigating systems for the various credible initiating actions for this transient are discussed in further detail in References 3 and 11. For analysis purposes, the limiting condition of a complete loss of load from 102% of nominal full power without a direct reactor trip is assumed to demonstrate 1) the adequacy of the pressure relieving devices to maintain the RCS within the Condition II pressure boundary criterion (i.e. 110% of design pressure) and 2) that the Condition II DNB limits are not violated for both beginning of life (BOL) and end of life (EOL) core conditions.

The conservative assumptions used in the Reference 3 analysis were assumed for the RETRAN comparative analysis (note that the limiting FSAR analysis condition for the reactor in manual control was assumed). These asumptions and the specific analysis parameter values are indicated in the Appendix (Item 5a). Note, in particular, that many of the system pressure relieving devices are assumed to be inoperative in order to produce conservative results. The RETRAN analysis was performed with the Single Loop Model.

The comparative results for this analysis are provided in Figures 5.27 through 5.31 for the BOL parameters and Figures 5.32 through 5.36 for the EOL case. The constraining result for this analysis is the pressurizer pressure and the change in this parameter is provided in Figure 5.27. Note that the rate of pressure change, the time of peak pressure and the magnitude of the peak pressure calculated for the respective analyses are in close agreement for the pressurization period of the transient. However, some deviation exists during the depressurization phase of the transient. This deviation most likely results from steam generator secondary side modeling differences used in RETRAN and the FSAR analyses. Both analyses demonstrate that the RCS pressure criterion for Condition II events is met.

Figures 5.28-5.31 provide the RETRAN and FSAR responses for nuclear power, pressurizer water volume, coolant inlet temperature and DNBR, respectively.

The DNBR's were generated with COBRA. DNB is not limiting for this event, as can be seen from Figure 5.31..

Figures 5.32 through 5.36 present the results for the Loss of Load EOL analyses and again confirm that the Condition II pressure and DNB criteria are not violated.

5.2.3.2 Loss of External Electrical Load Transient - Current Analysis

The Loss of Load has been reanalyzed since the FSAR to support a Technical Specification change allowing core operation with a slightly positive moderator temperature coefficient at powers less than hot full power at BOL.[12]

The licensing reanalysis, to be used for comparison purposes, was only performed for the BOL case, since the moderator temperature coefficient would be highly negative at EOL and, therefore, not impacted by the proposed Technical Specifications change. The RETRAN analysis assumptions and parameter values are provided in the Appendix (Item 5b); note that for the moderator temperature coefficient, a value of +3.0 pcm/^{o}F was assumed. The Single Loop Model was used for the RETRAN analysis.

A comparison of the RETRAN and licensing reanalysis results is shown in Figures 5.37 through 5.40. Comparisons are provided, for nuclear power, pressurizer pressure, coolant average temperature, and DNBR. As discussed previously, the secondary side heat transfer modeling differences resulted in some differences in the predictions during the depressurization phase. The RETRAN analysis results confirm the conclusion drawn in the licensing reanalysis, i.e., that the pressure relieving devices are adequate to limit the peak pressure to a value below the Condition II Criterion and that the Condition II DNBR Criterion is also met.

5.2.3.3 Excessive Heat Removal Due to Feedwater System Malfunction Transient - FSAR Analysis

Excessive heat removal incidents resulting from feedwater system malfunctions result from either 1) excessive feedwater flow, such as might result from a failure of the feedwater flow control valve or 2) reductions in feedwater temperature. An

example of the second type of transient, which consists of the accidental opening of the feedwater bypass valve resulting in diversion of flow around the low pressure feedwater heaters, was chosen for analysis.

The case examined, which was analyzed in Reference 3, assumed no reactor control and a zero moderator temperature coefficient. The resulting transient is a very gradual increase in core power in response to the primary coolant and fuel temperature reduction resulting from the decreased temperature of the feedwater to the steam generators. After the core power increases to a level which essentially matches the secondary side heat removal rate, the temperature begins to stabilize and the system pressure increases in response to the pressurizer heaters.

The Appendix (Item 6) summarizes the important analysis assumptions made for both the FSAR[3] and RETRAN analyses, including specific analytical parameter values assumed. The RETRAN analysis was performed with the Single Loop Model discussed in Section 3, and conservatively assumes constant steam flow throughout the transient.

The RETRAN analytical results are compared to the results reported in the FSAR, in Figures 5.41 through 5.45. It should be noted that this transient is calculated over a long time period, approximately 900 seconds.

Figure 5.41, which represents the variation in feedwater temperature with time, depicts the forcing function assumed in the two analyses.

Figures 5.42 - 5.45 present the results for core power, average coolant temperature, pressurizer pressure and DNBR.

The primary FSAR conclusion, that DNBR increases monotonically as the transient proceeds, is supported by both analyses.

5.2.3.4 Accidental Depressurization of the Secondary System/Main Steam Line Break Transient - FSAR Analysis

This class of accidents includes any uncontrolled steam release from a steam generator, such as might be caused by failure of a safety or relief valve or rupture of a main steam pipe. A Main Steam Line Break (MSLB) Transient, which is a Class IV event

and the limiting transient in this category, was chosen for analysis.

The increased steam flow resulting from this accident causes a reduction in primary coolant system temperatures and pressures. The reduced temperature causes a positive reactivity insertion (assuming a negative moderator temperature coefficient). This insertion, coupled with the assumption that the most reactive rod cluster control assembly (RCCA) sticks in its fully withdrawn position, increases the possibility that the reactor will return to a critical condition and resume power generation following reactor trip. This is a potential problem because of the high power peaking factors associated with the stuck RCCA assumption. The core power is limited by the negative Doppler and moderator reactivity effects for which conservative values are assumed in the analysis. The core is ultimately returned to a subcritical condition by boric acid delivered by the safety injection system. A more detailed discussion of the transient and the various mitigating systems is provided in the units' FSAR's.

Several different MSLB transients are discussed in the FSAR [3,4]. The limiting MSLB case, which was analyzed with RETRAN for comparison to the Surry FSAR analysis, consisted of a break adjacent to a steam generator outlet nozzle with continued availability of offsite power. The MSLB was analyzed with the Two Loop Model (See Section 3) which calculates both the primary and secondary system responses, the reactivity effects of safety injection and the core power response following return to criticality.

A summary of important analysis assumptions, which correspond to the assumptions made in the FSAR, is given in the Appendix (Item 7a). Specific analytical values used for the analysis are also shown in the Appendix. Representative results from the FSAR analysis are presented and compared to vendor results in Figures 5.46 to 5.49, for steam flow, pressurizer pressure, core reactivity and core average heat flux, respectively. The slight differences in the shapes of the core heat flux response are believed to be related to differences in the treatment of core boron concentration buildup following safety injection.

The comparisons indicate that Vepco's RETRAN Models provide an appropriate basis for calculating the system transient portion of the Main Steam Line Break analysis.

5.2.3.5 Accidental Depressurization of the Secondary System/Main Steam Line Break Transient - Current Analysis

The Main Steam Line Break Transient has been reanalyzed for several Vepco reload cores. The reanalyses have been necessary to confirm the continued acceptability of the MSLB transient results for variations in the reload core designs. For example, a recent licensing update reanalysis of the system response was performed for the Surry Unit 1, Cycle 4 reload core (see Reference 13). The basic analytical assumptions and parameter values for this reanalysis are shown in the Appendix, (Item 7b.) The comparative results of the two analyses are summarized in Table 5.1. As can be seen the results for temperature, pressure and core heat flux for the two analyses are quite similar.

The dynamic response to the MSLB reload reanalysis is shown in Figures 5.50 - 5.52. Comparison to the FSAR results (Figures 5.46 - 5.49) shows that the general characteristics of the transient responses are the same for the two cases. The vendor results for the analysis are considered proprietary and are omitted.

5.2.4 General Conclusions - Licensing Transient Analyses

The analysis results shown in Figures 5.1 - 5.52 show that Vepco's analysis approach yields results which are comparable to those obtained by our NSSS vendor for previous licensing submittals. The similarities hold for a broad variety of transients of varying levels of severity and result in identical conclusions regarding core and system safety. These comparisons illustrate Vepco's general capability to perform analyses of Condition I-III transients, and the system transient aspects of certain Condition IV transients.

TABLE 5.1

LIMITING PREDICTED RESULTS
MAIN STEAM LINE BREAK TRANSIENT
SURRY 1, CYCLE 4, REANALYSIS

Parameter	Peak Value: Licensing Analysis	Peak Value: RETRAN
Core heat flux, % of rated	28.6	25.8
Reactor inlet temperature (failed loop), ^{o}F	373	373
Reactor inlet temperature (intact loop), ^{o}F	502	504
Pressurizer Pressure, Psia	1167	1255

Figure 5.27

PRESSURIZER PRESSURE CHANGE
LOSS OF LOAD TRANSIENT
BOL - FSAR ANALYSIS

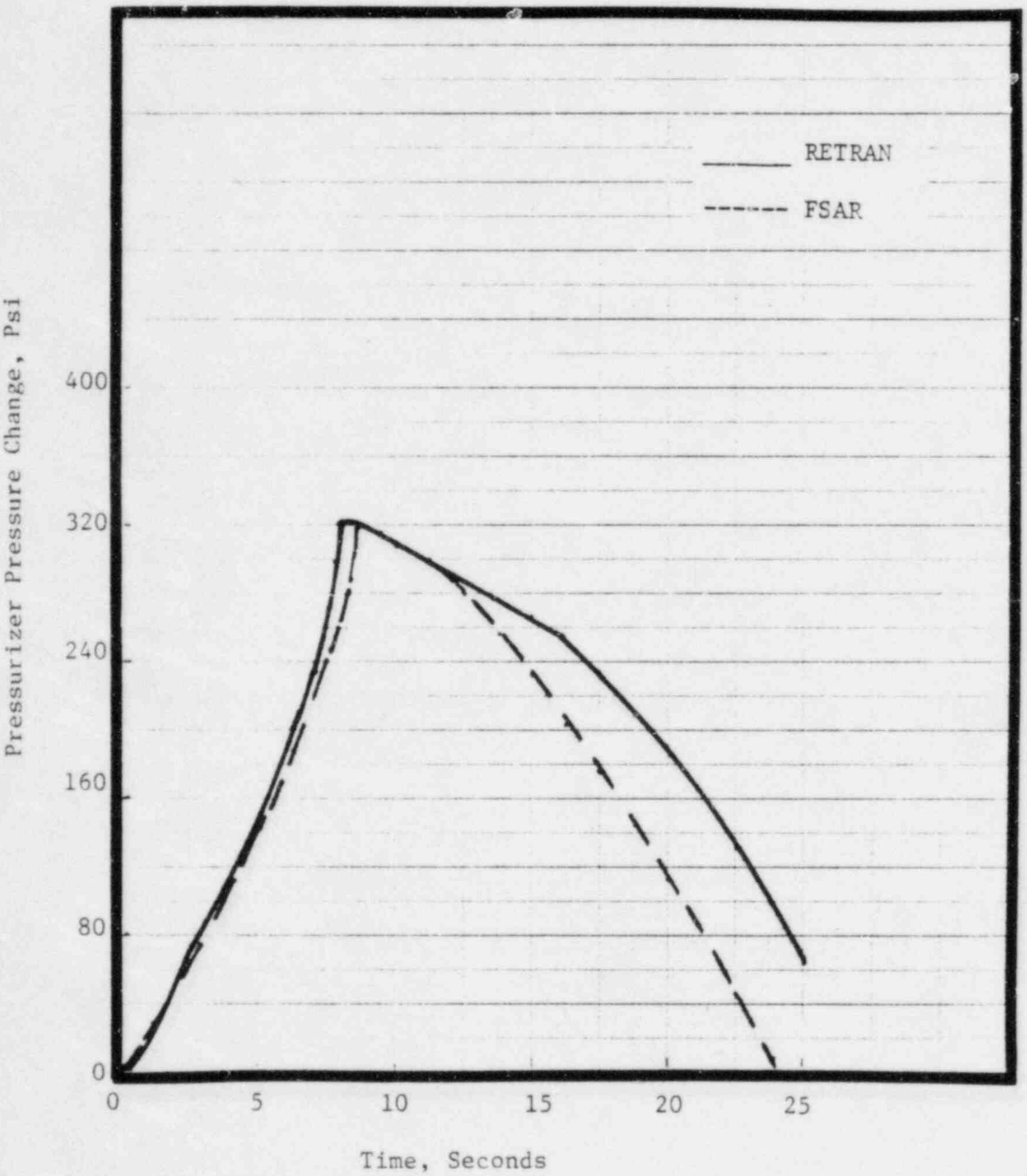


Figure 5.28

NUCLEAR POWER
LOSS OF LOAD TRANSIENT
BOL - FSAR ANALYSIS

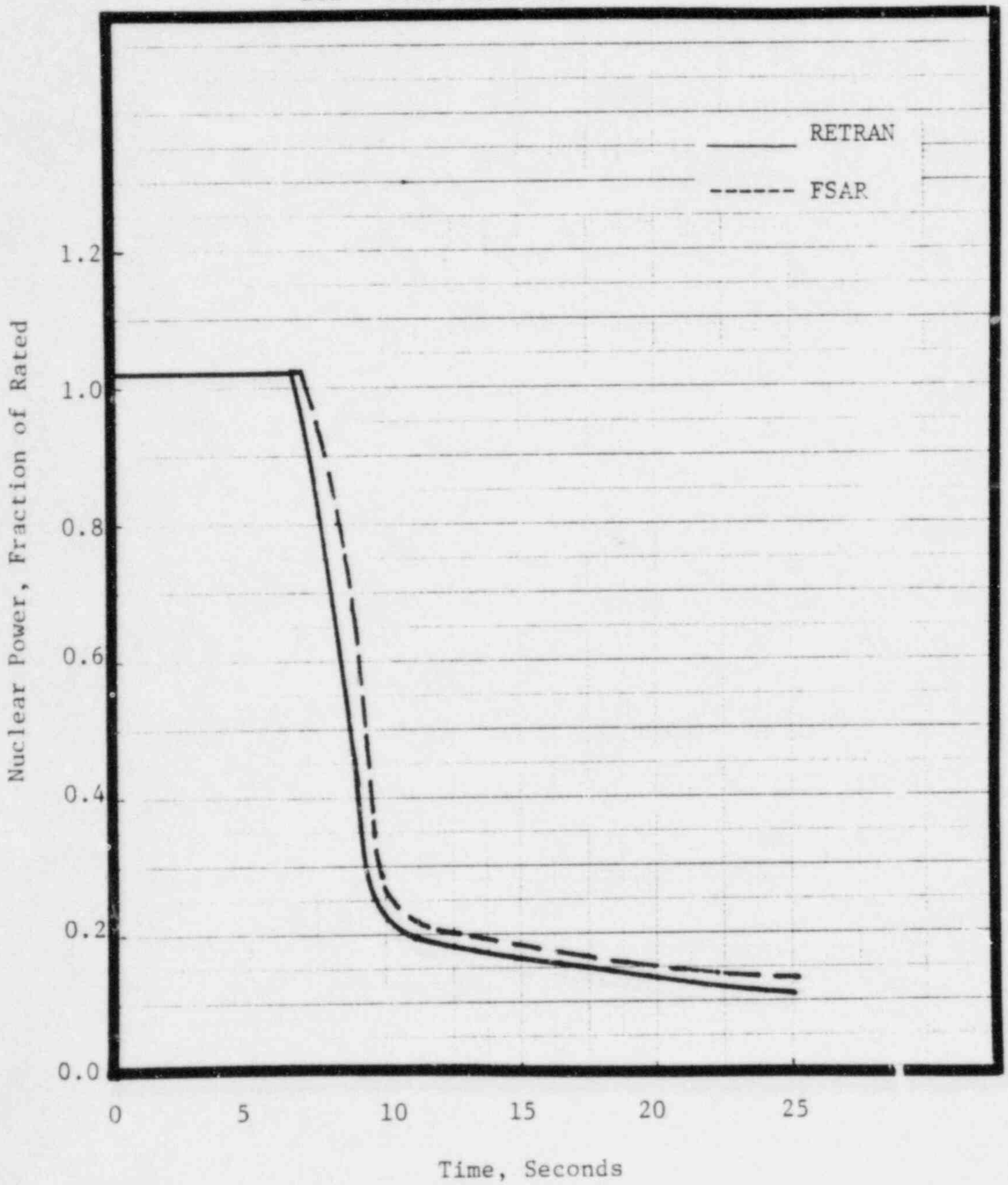


Figure 5.29

PRESSURIZER WATER VOLUME CHANGE
LOSS OF LOAD TRANSIENT
BOL - FSAR ANALYSIS

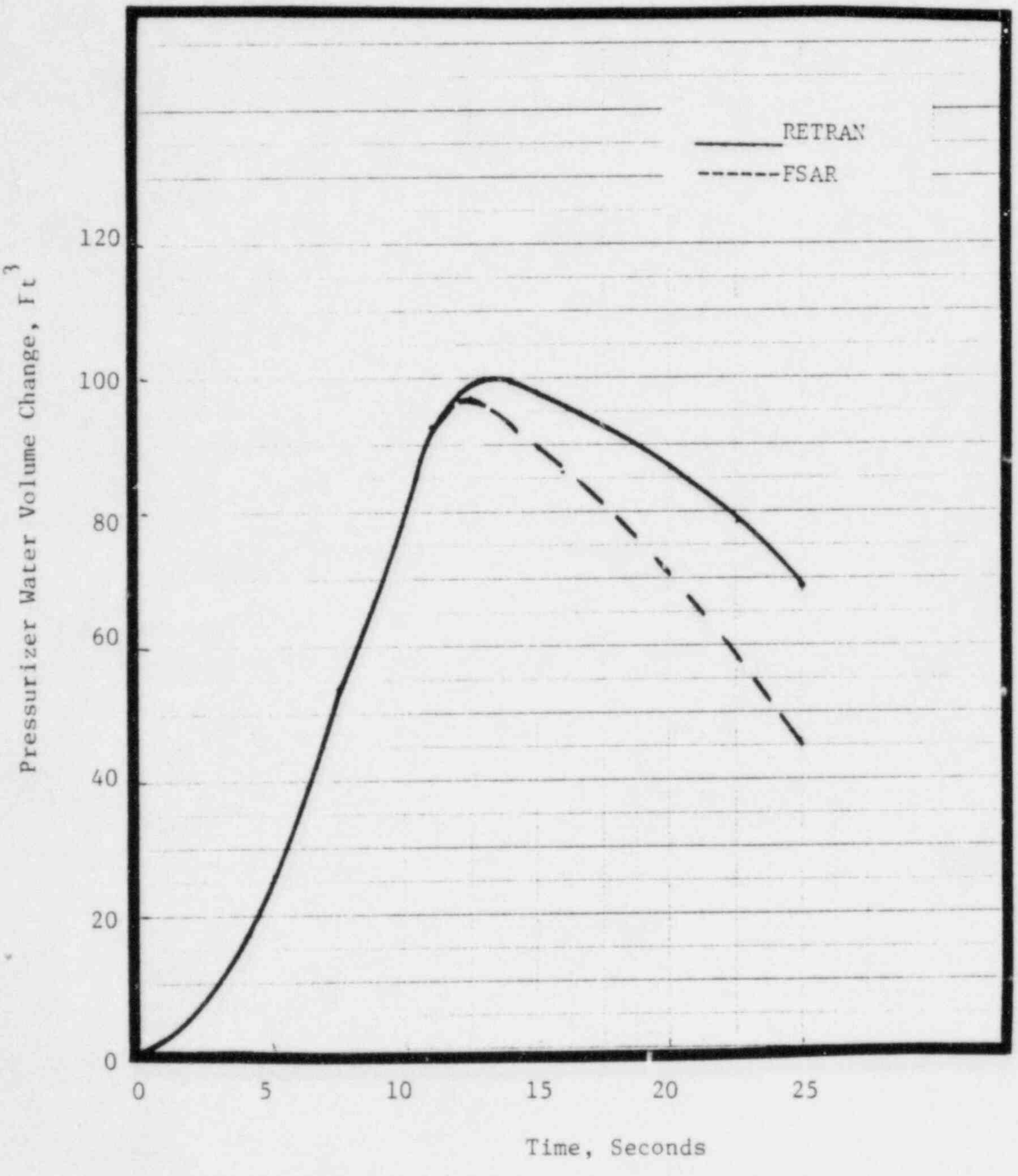


Figure 5.30

COOLANT INLET TEMPERATURE CHANGE
LOSS OF LOAD TRANSIENT
BOL - FSAR ANALYSIS

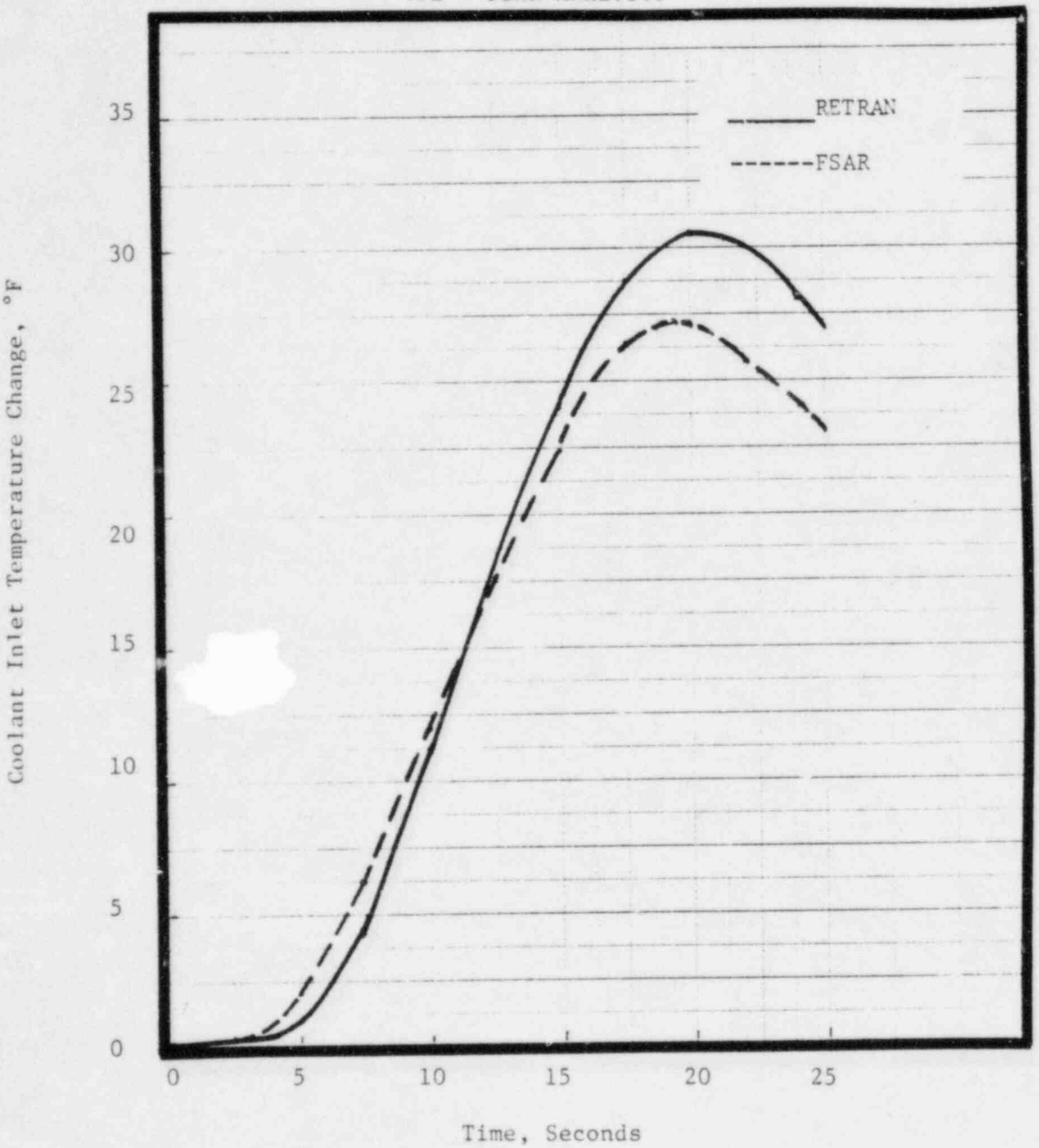


Figure 5.31

DNB RATIO
LOSS OF LOAD TRANSIENT
BOL - FSAR ANALYSIS

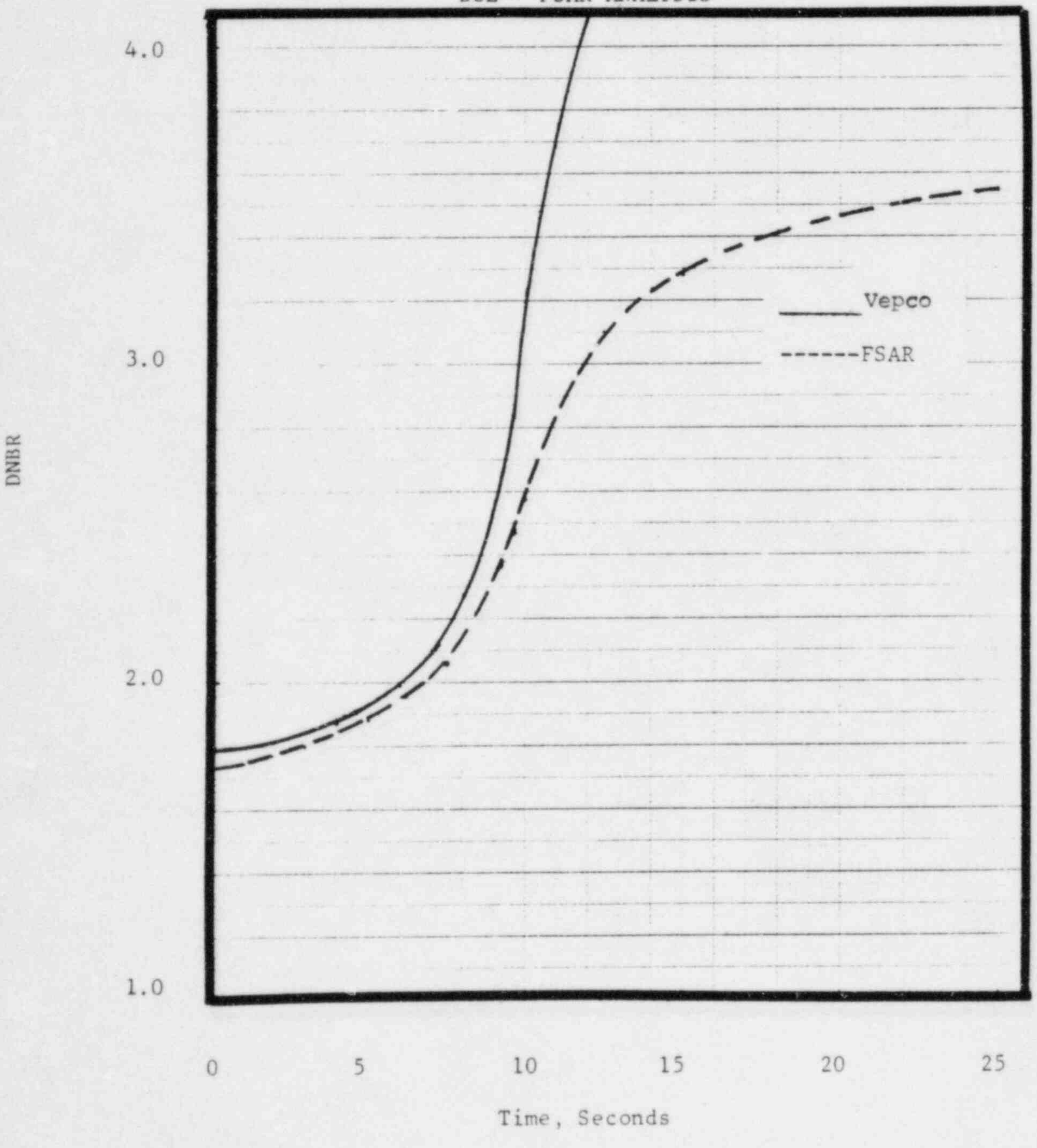


Figure 5.32

PRESSURIZER PRESSURE CHANGE
LOSS OF LOAD TRANSIENT
EOL-FSAR ANALYSIS

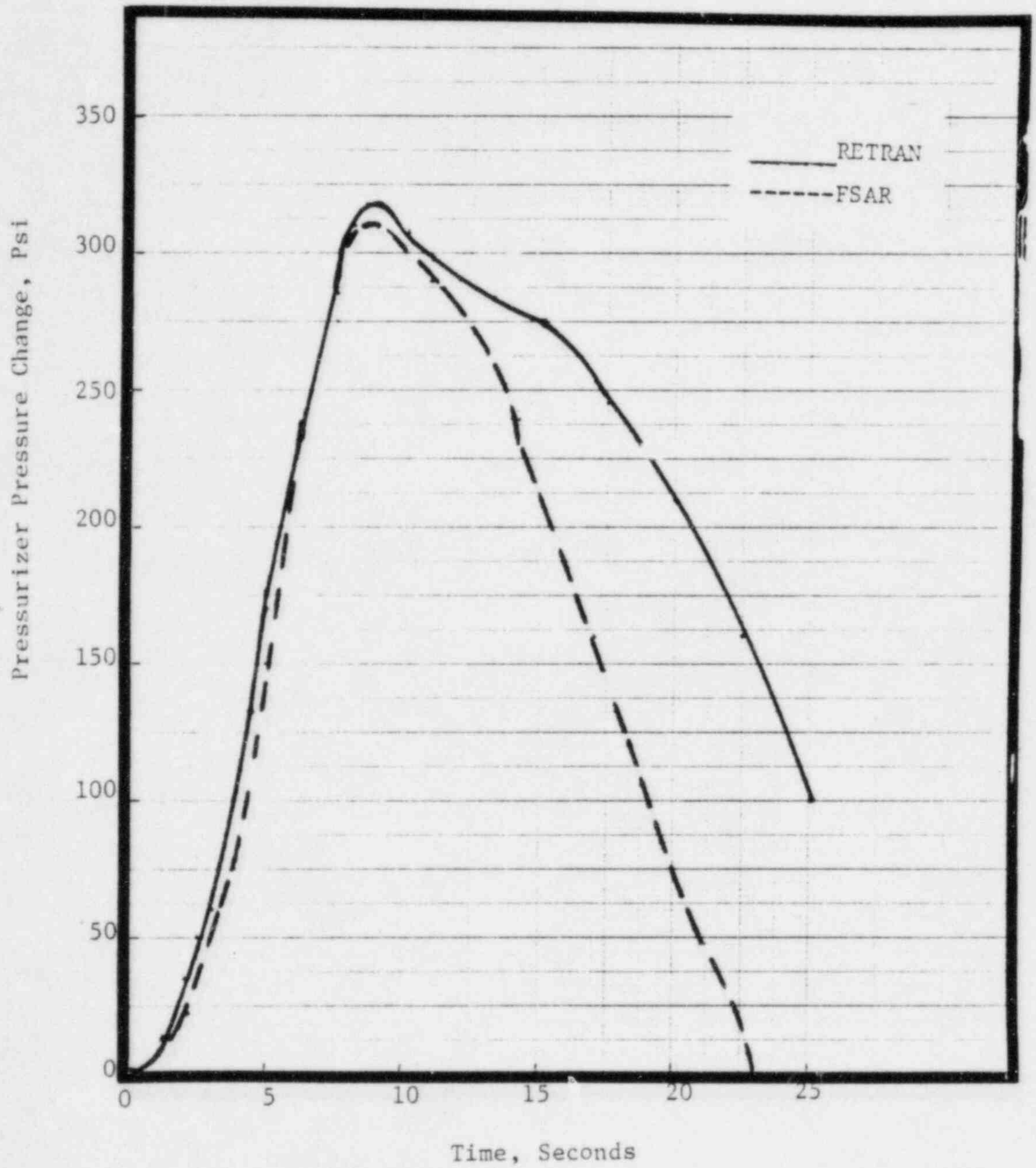


Figure 5.33

NUCLEAR POWER
LOSS OF LOAD TRANSIENT
EOL - FSAR ANALYSIS

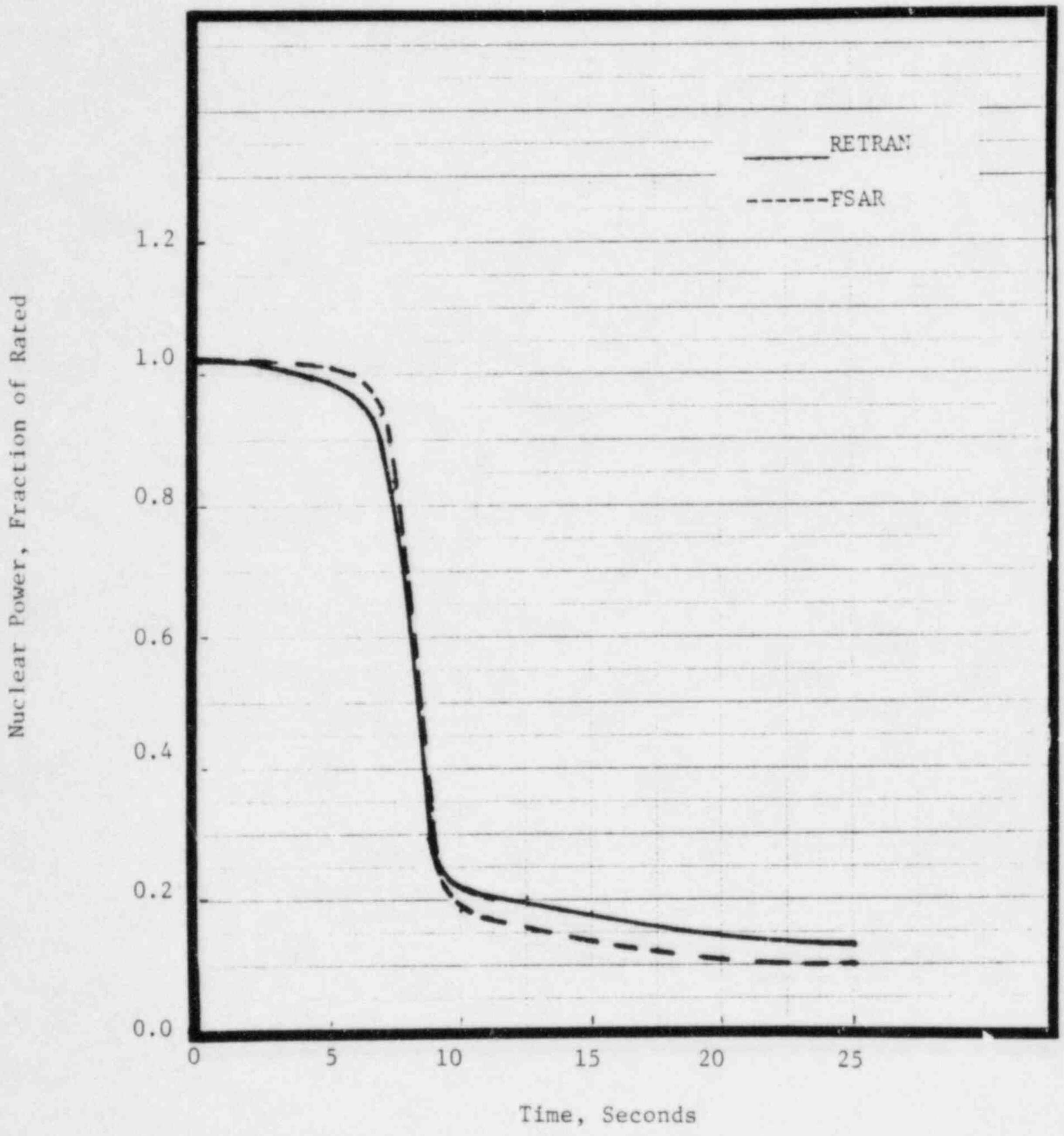


Figure 5.34

PRESSURIZER WATER VOLUME CHANGE
LOSS OF LOAD TRANSIENT
EOL-FSAR ANALYSIS

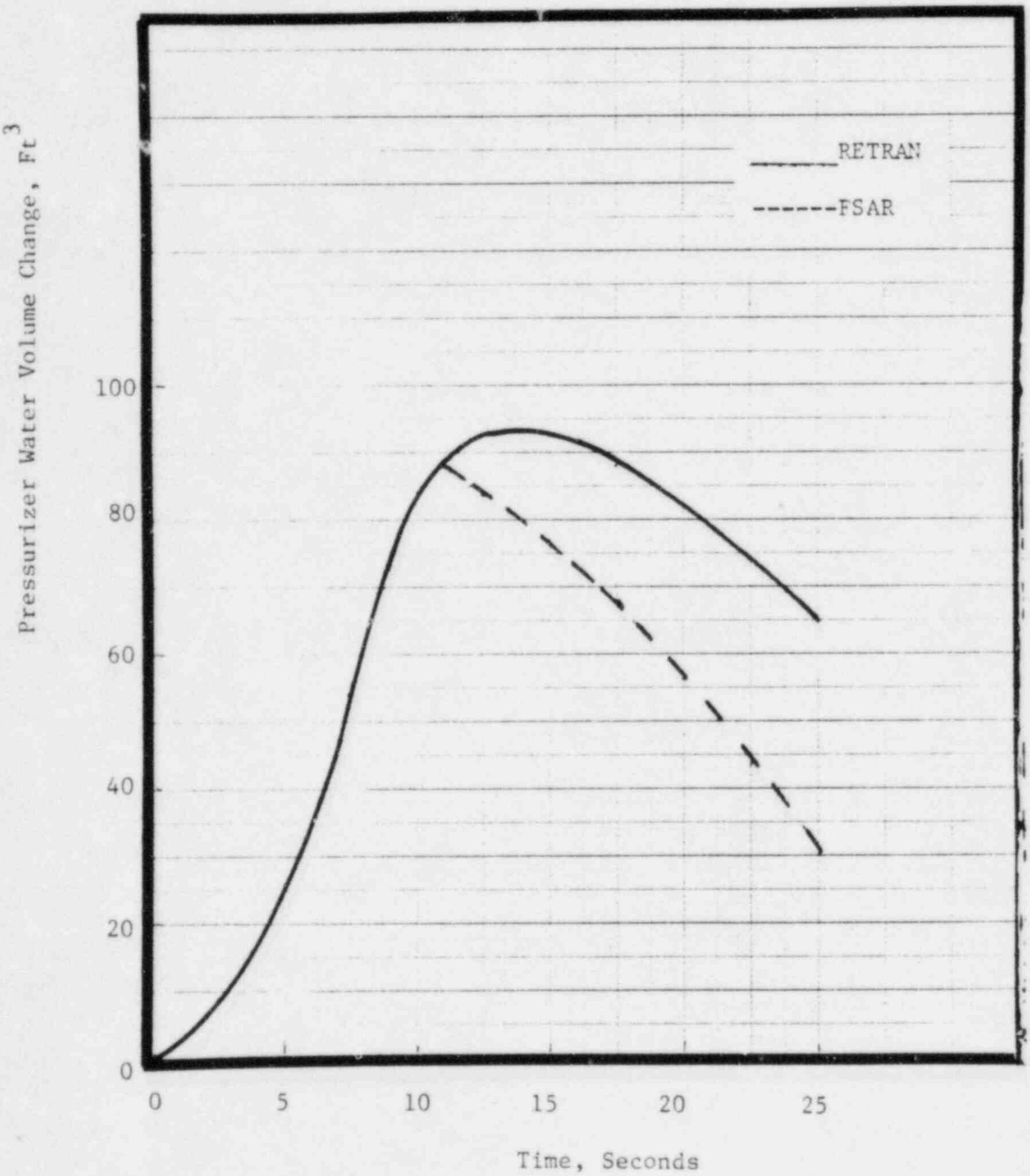


Figure 5.35

COOLANT INLET TEMPERATURE CHANGE
LOSS OF LOAD TRANSIENT
EOL-FSAR ANALYSIS

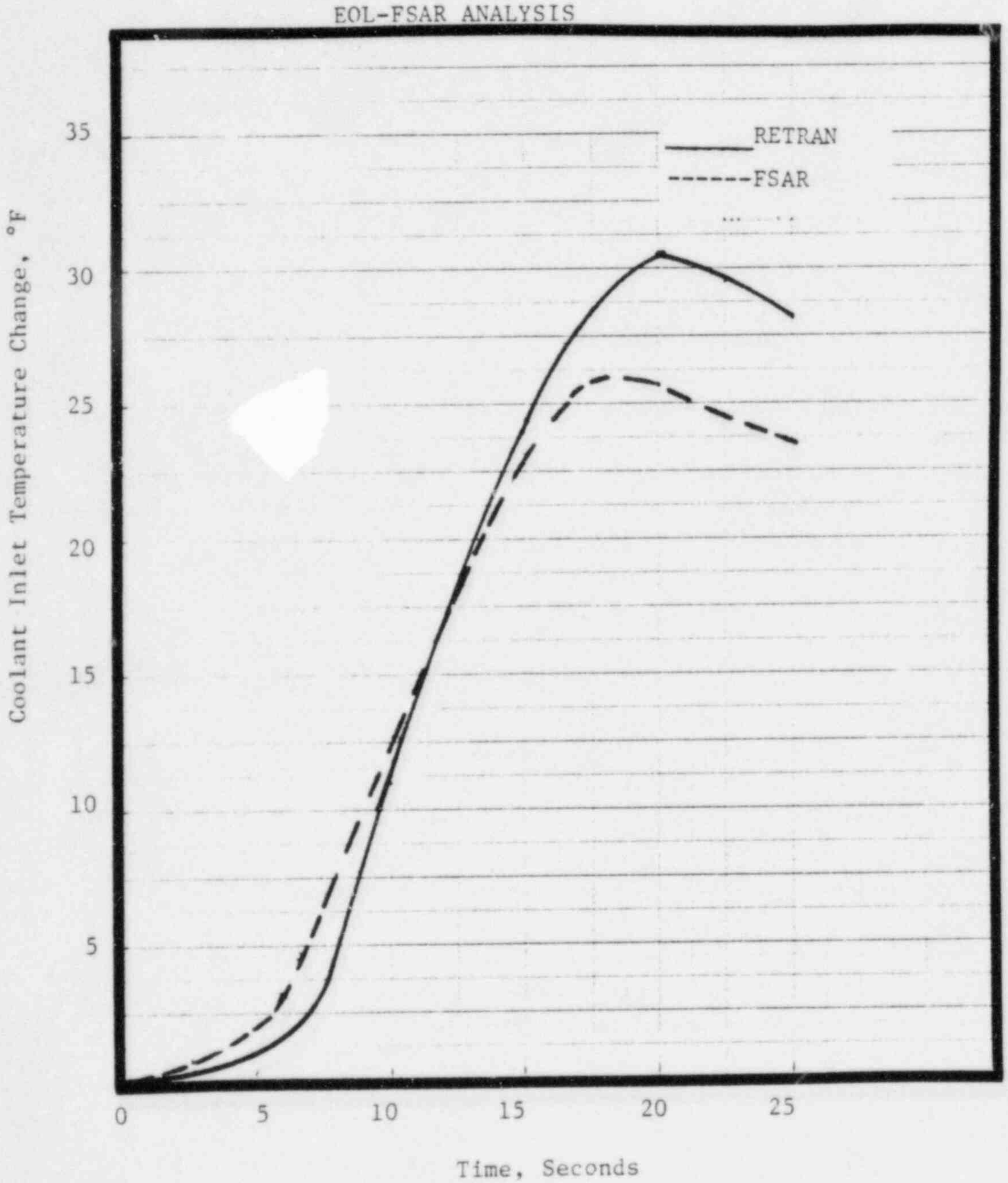


Figure 5.36

DNB RATIO
LOSS OF LOAD TRANSIENT
EOL FSAR ANALYSIS

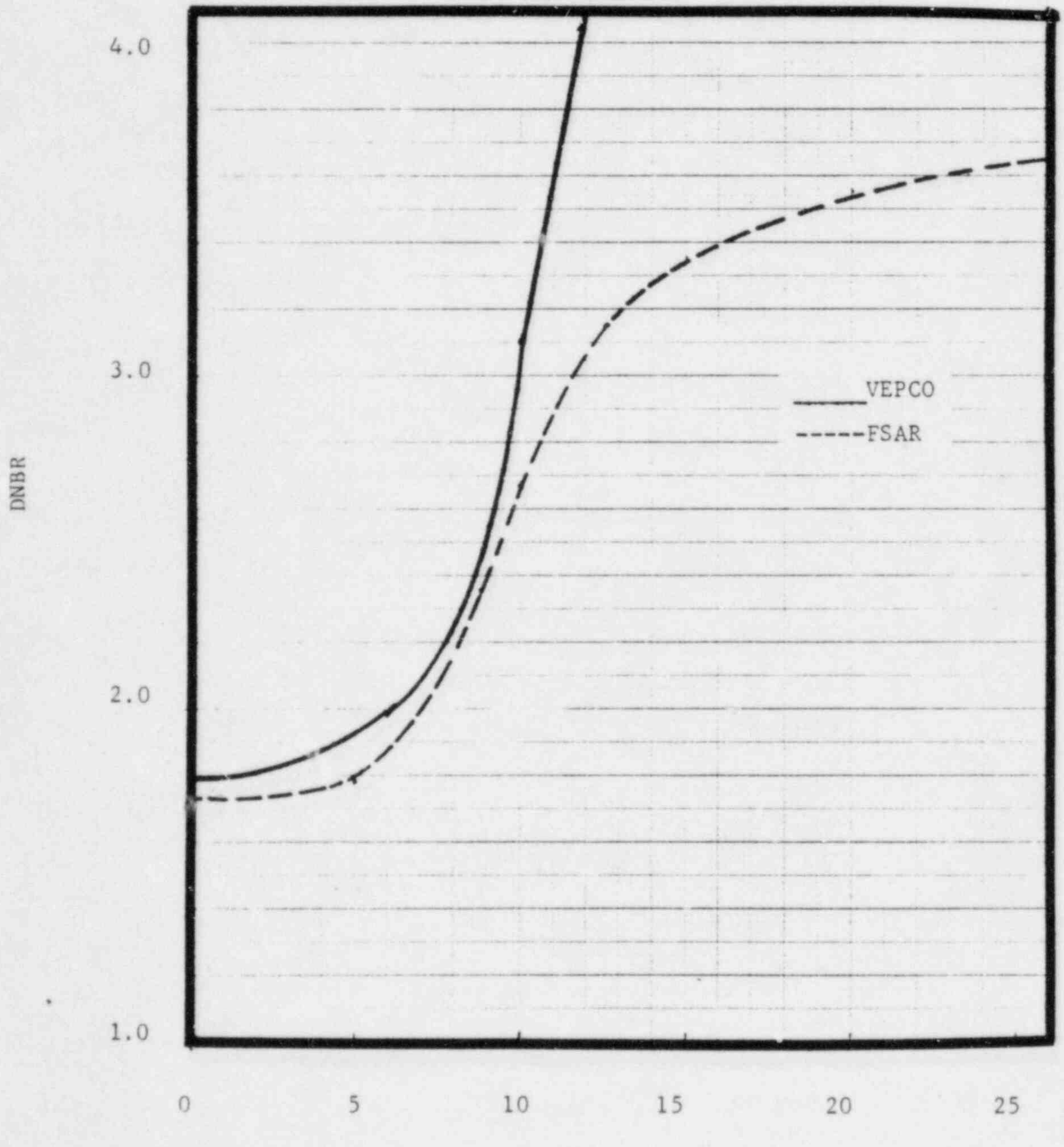


Figure 5.37

PRESSURIZER PRESSURE
LOSS OF LOAD TRANSIENT
POSITIVE MODERATOR COEFFICIENT REANALYSIS

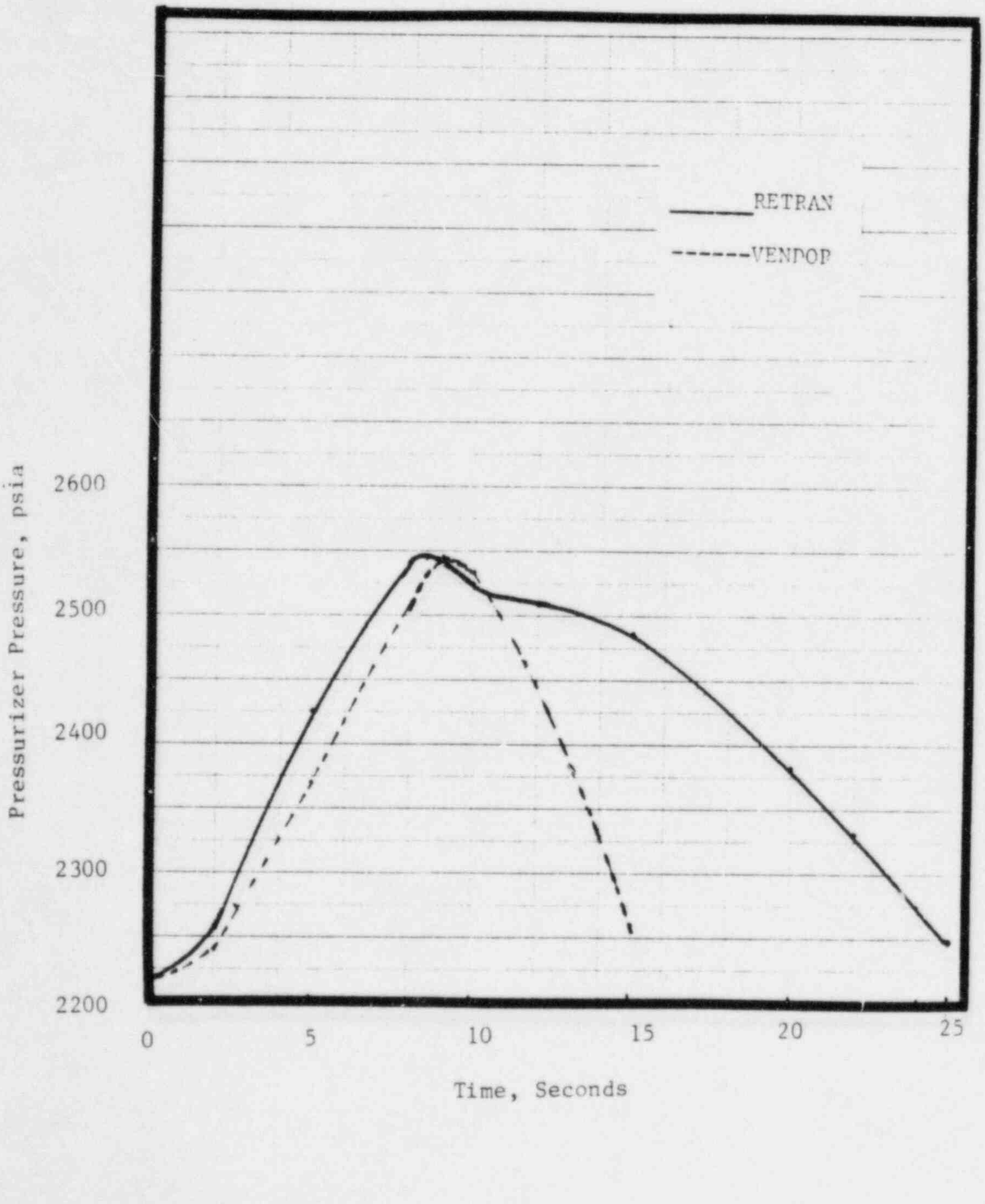


Figure 5.38

NUCLEAR POWER
LOSS OF LOAD TRANSIENT

POSITIVE MODERATOR COEFFICIENT REANALYSIS

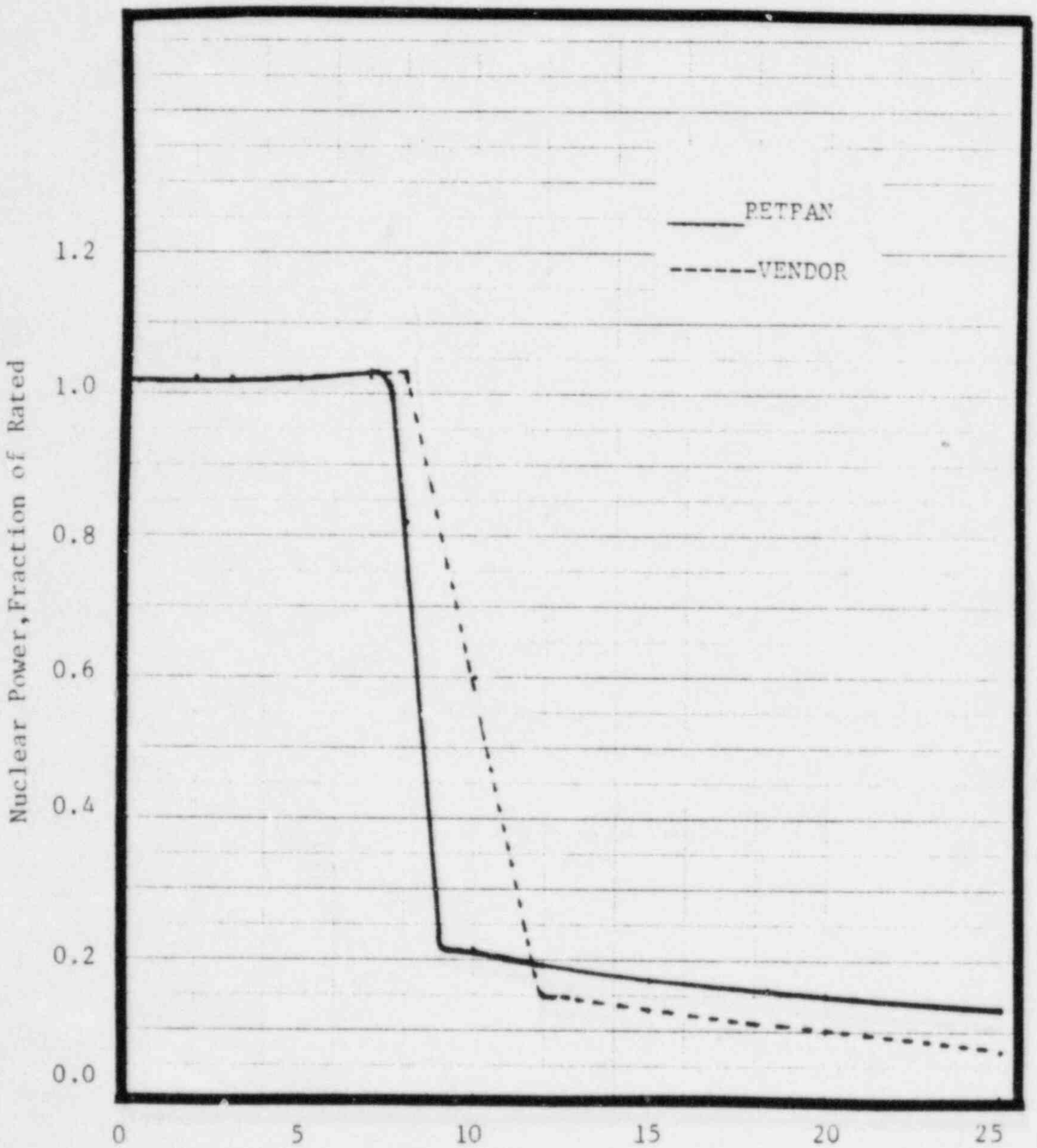


Figure 5.39

AVERAGE COOLANT TEMPERATURE
LOSS OF LOAD TRANSIENT
POSITIVE MODERATOR COEFFICIENT REANALYSIS

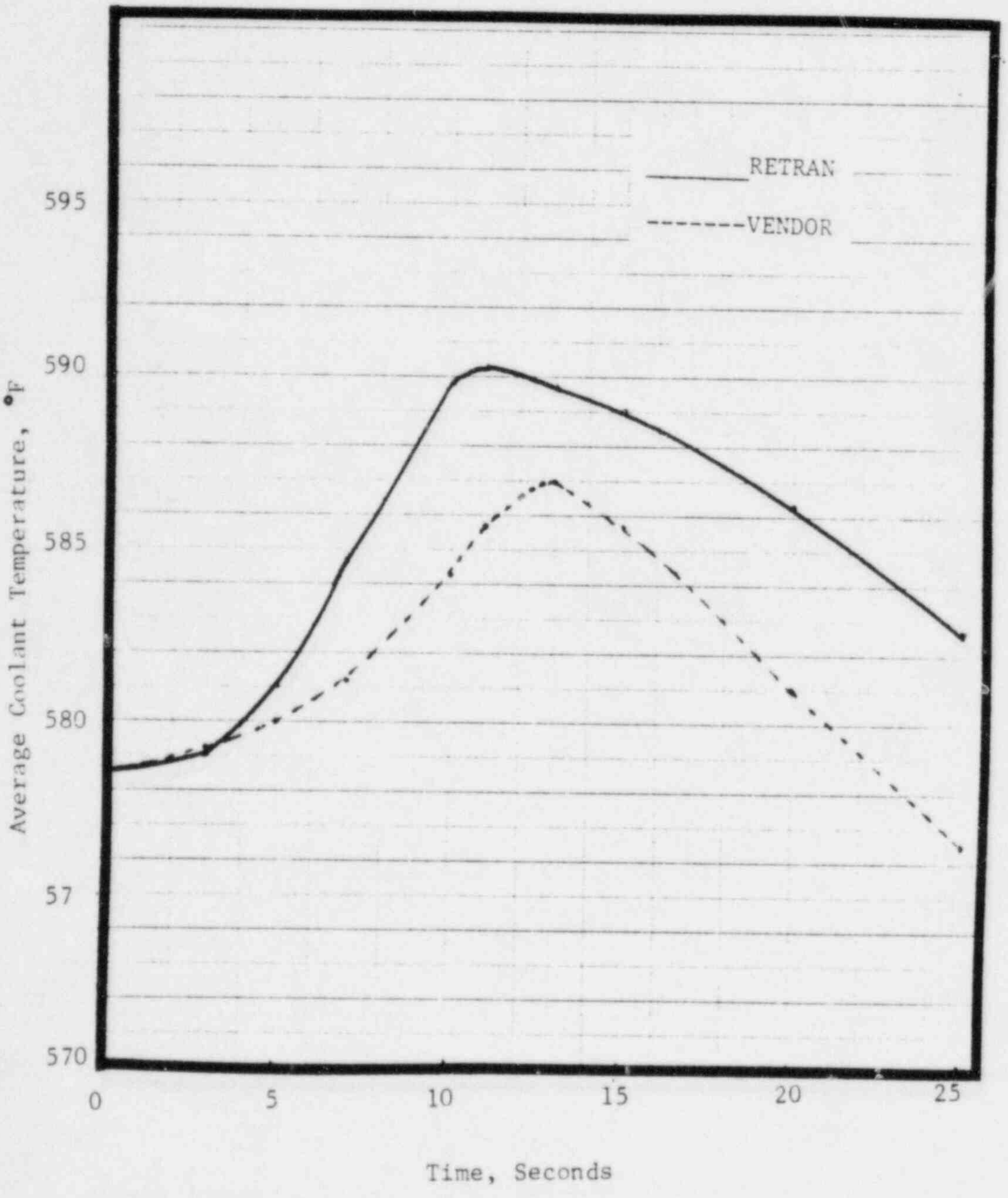


Figure 5.40

DNB RATIO
LOSS OF LOAD TRANSIENT
POSITIVE MODERATOR COEFFICIENT ASSUMPTION REANALYSIS

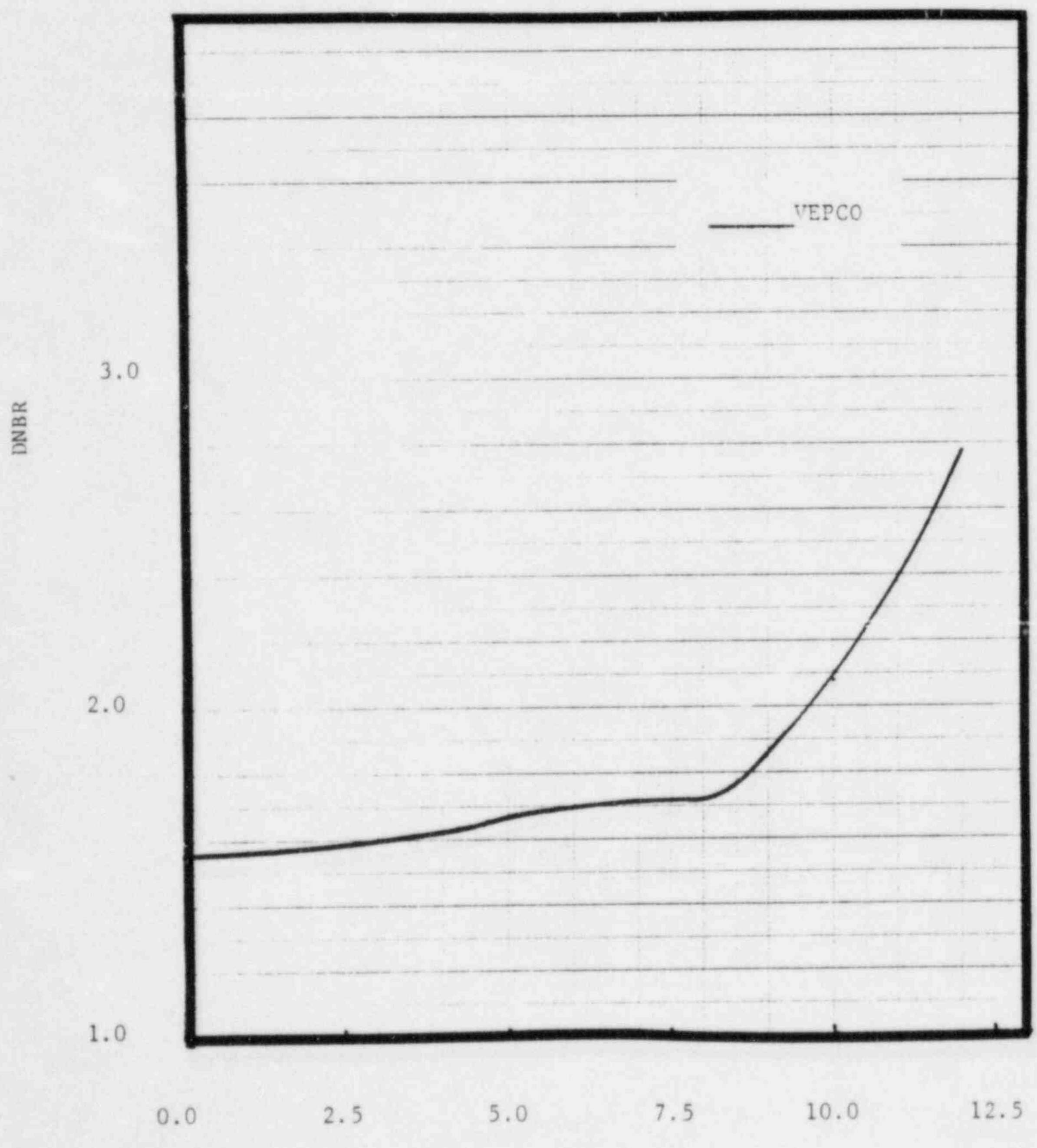


Figure 5.41

FEEDWATER TEMPERATURE CHANGE
EXCESSIVE HEAT REMOVAL DUE TO
FEEDWATER SYSTEM MALFUNCTION TRANSIENT
FSAR ANALYSIS

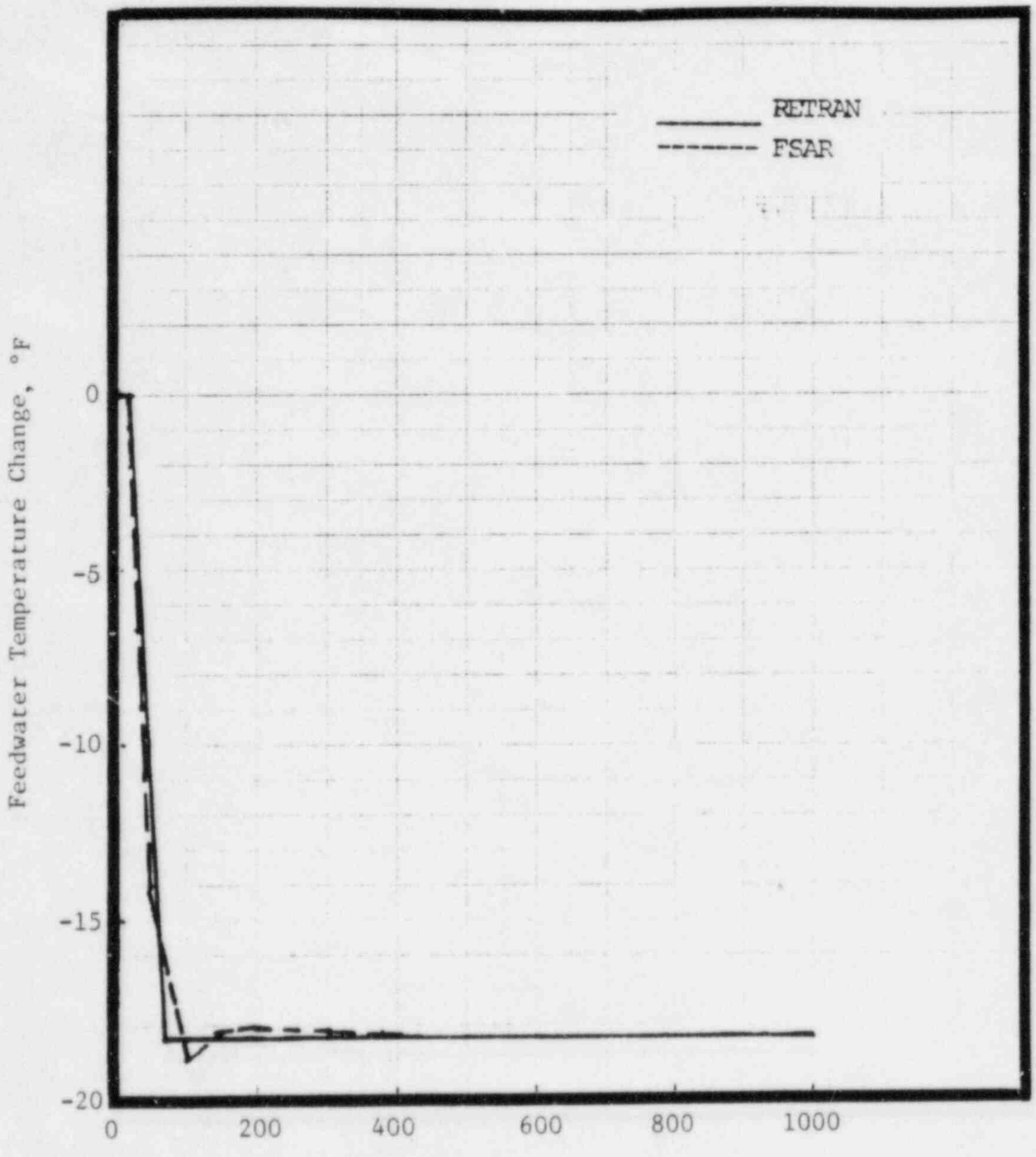


Figure 5.42

NUCLEAR POWER
EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER
SYSTEM MALFUNCTION TRANSIENT
FSAR ANALYSIS

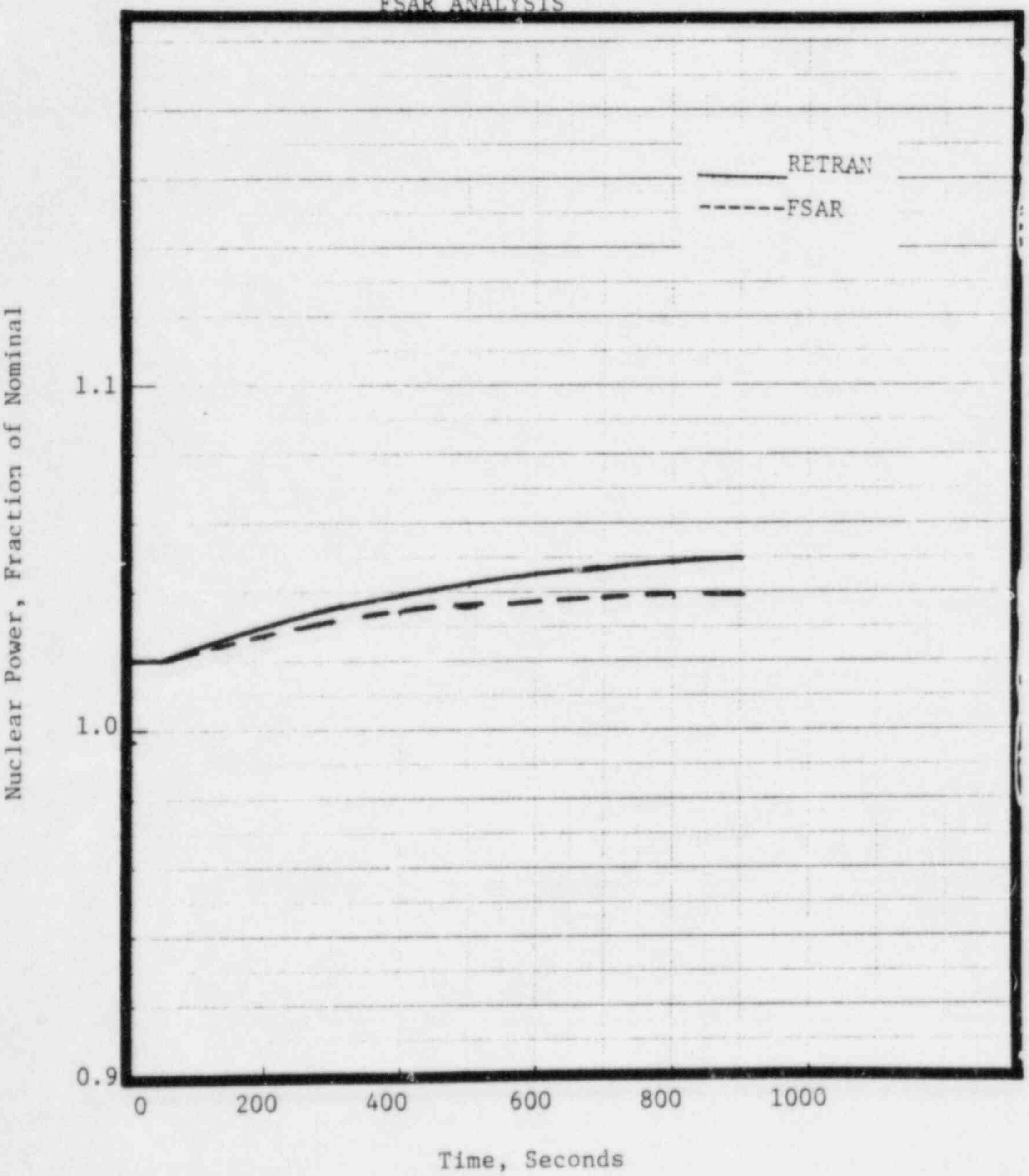


Figure 5.43

CHANGE IN AVERAGE COOLANT TEMPERATURE
EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER
SYSTEM MALFUNCTION TRANSIENT
FSAR ANALYSIS

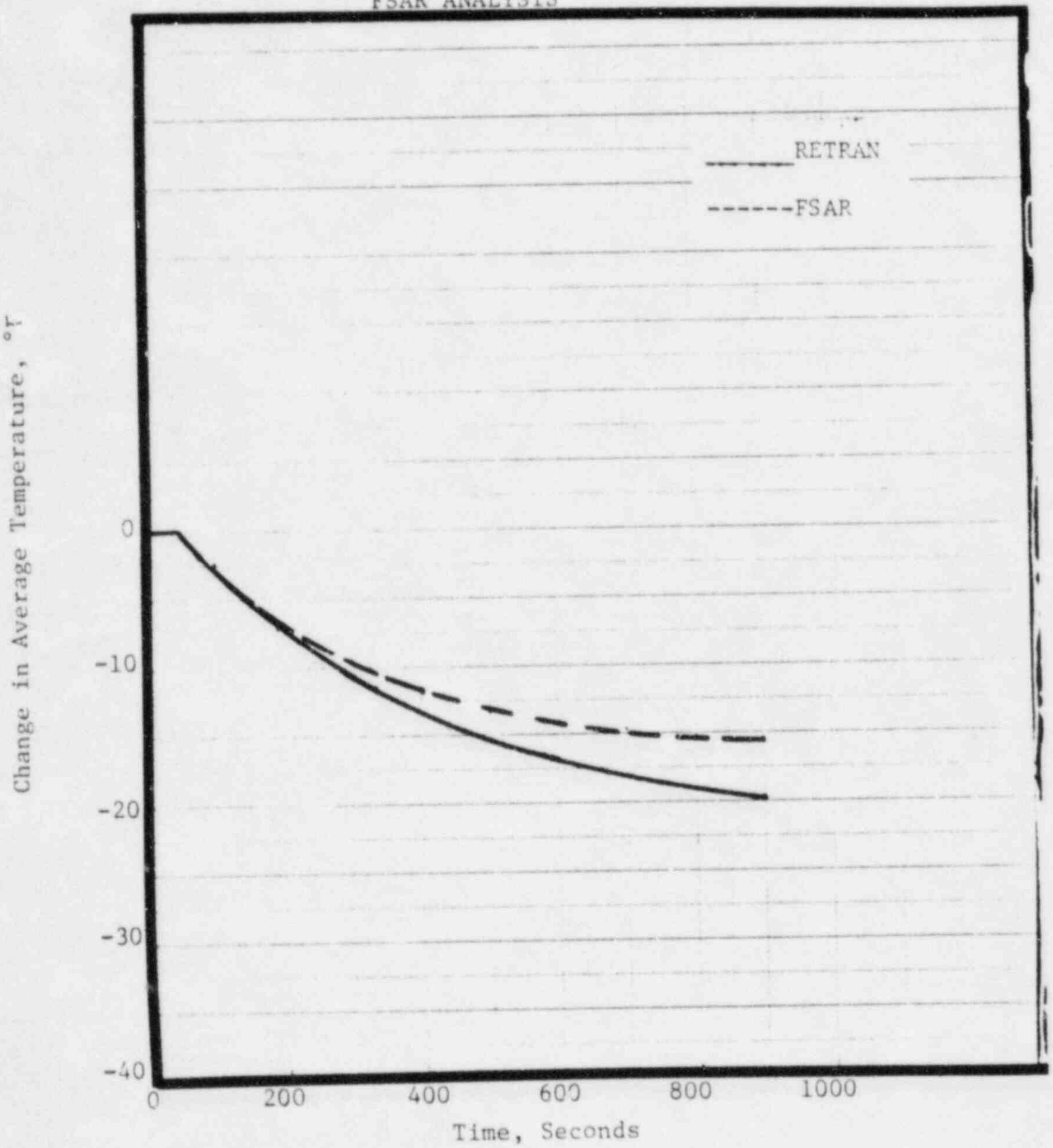


Figure 5.44

PRESSURIZER PRESSURE CHANGE
EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER
SYSTEM MALFUNCTION TRANSIENT
FSAR ANALYSIS

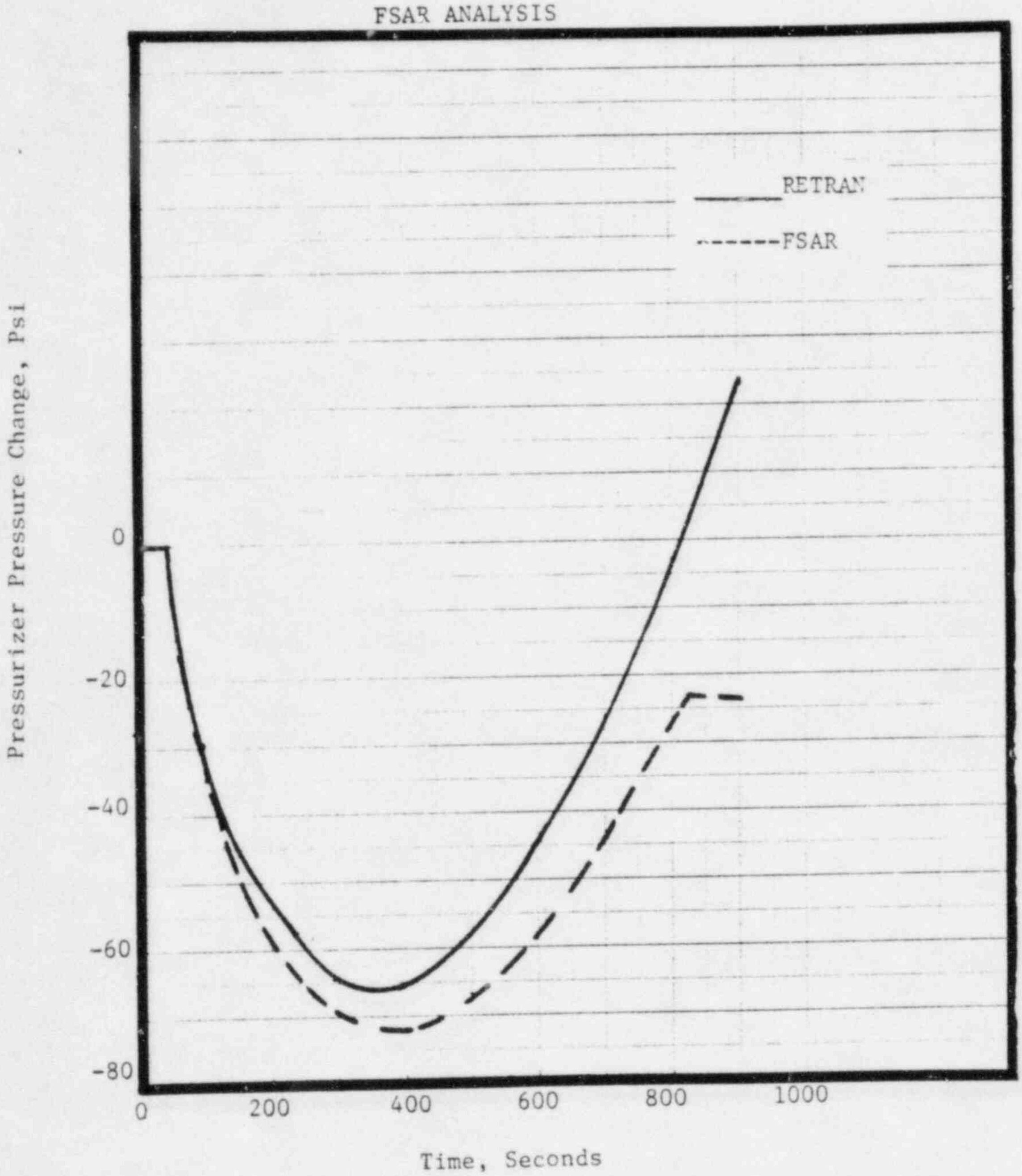


Figure 5.45

DNB RATIO
EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER
SYSTEM MALFUNCTION TRANSIENT
FSAR ANALYSIS

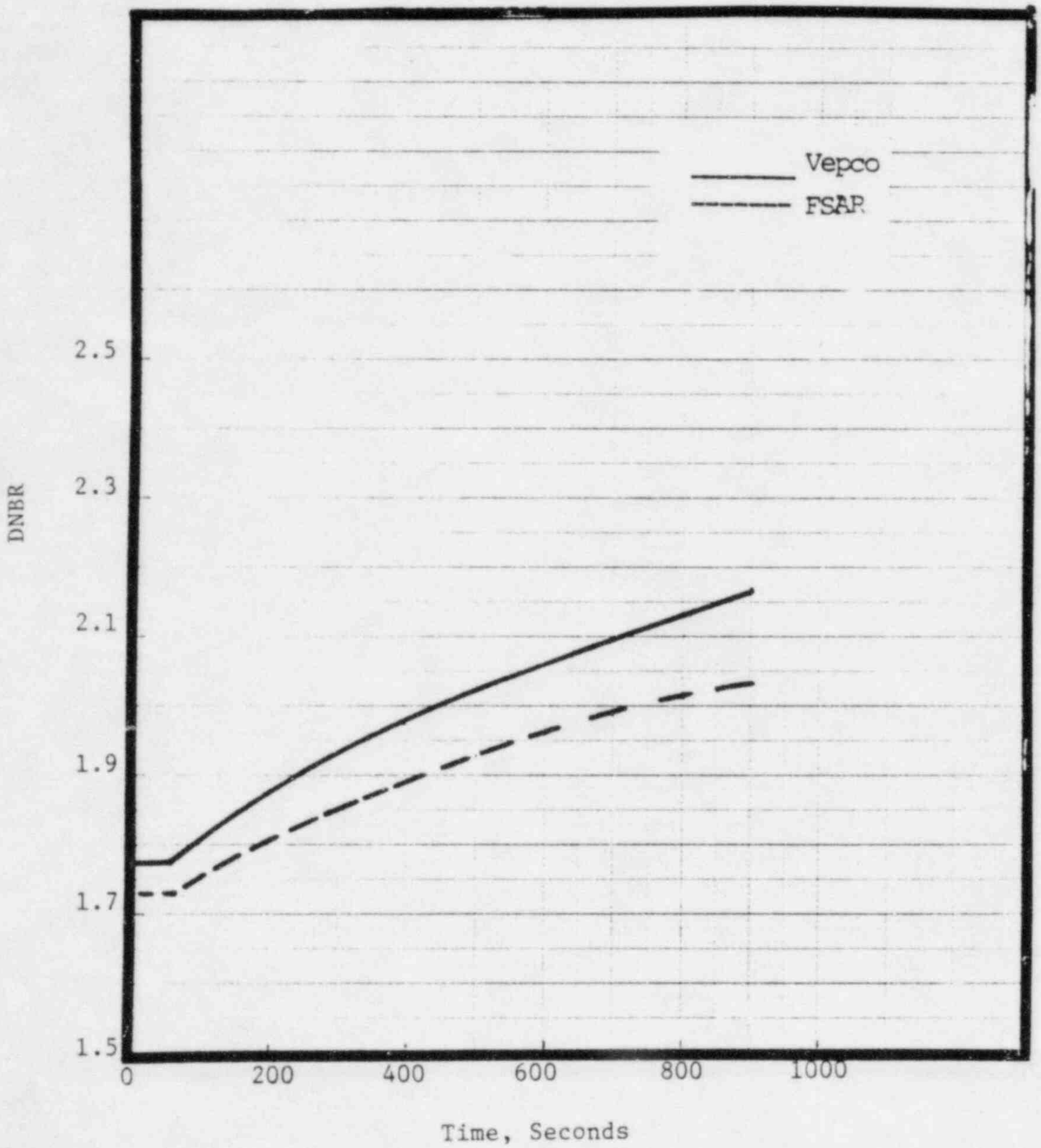


Figure 5.46

BREAK FLOW RATE
MAIN STEAM LINE BREAK
FSAR ANALYSIS

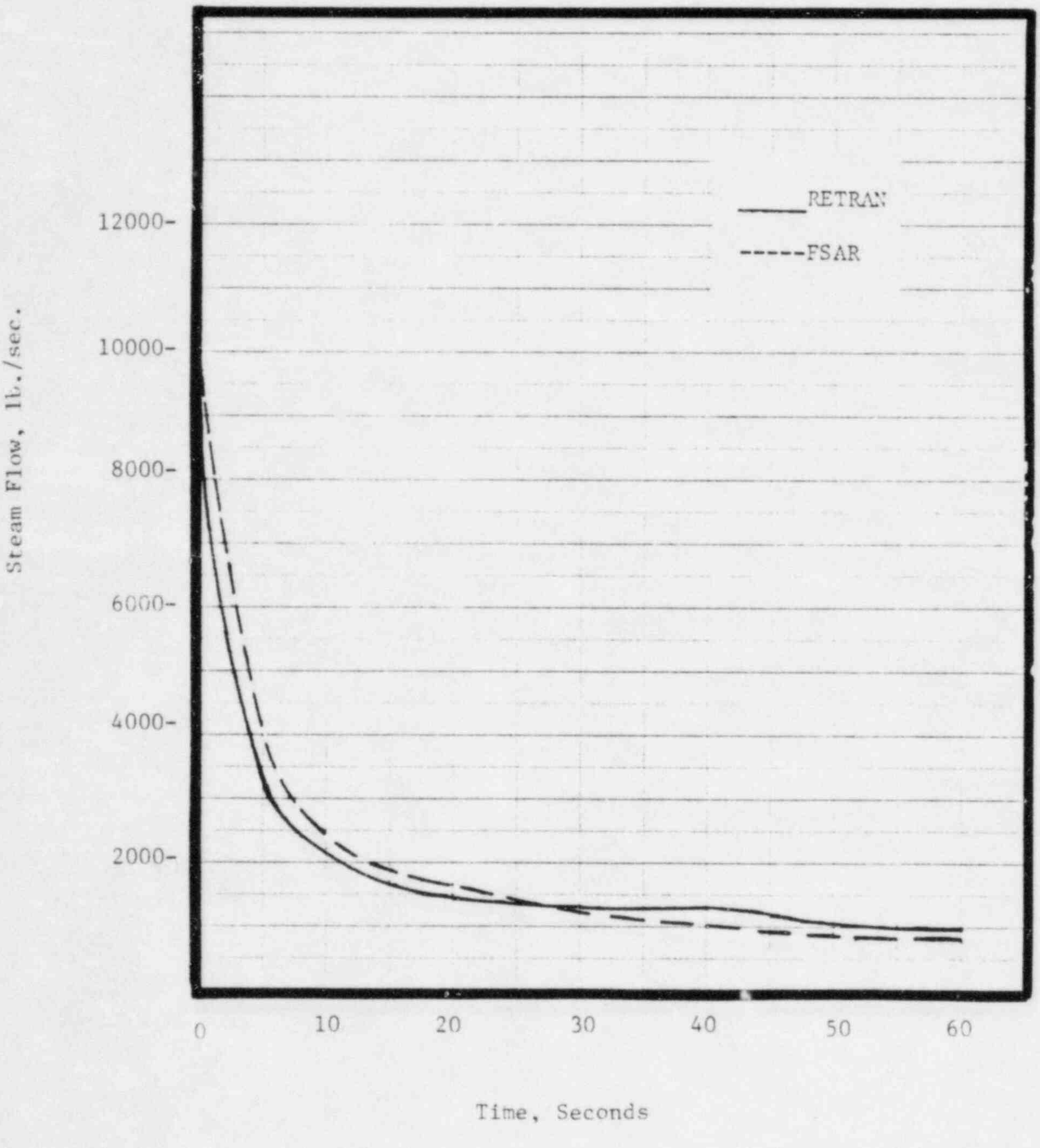


Figure 5.47

PRESSURIZER PRESSURE
MAIN STEAM LINE BREAK
FSAR ANALYSIS

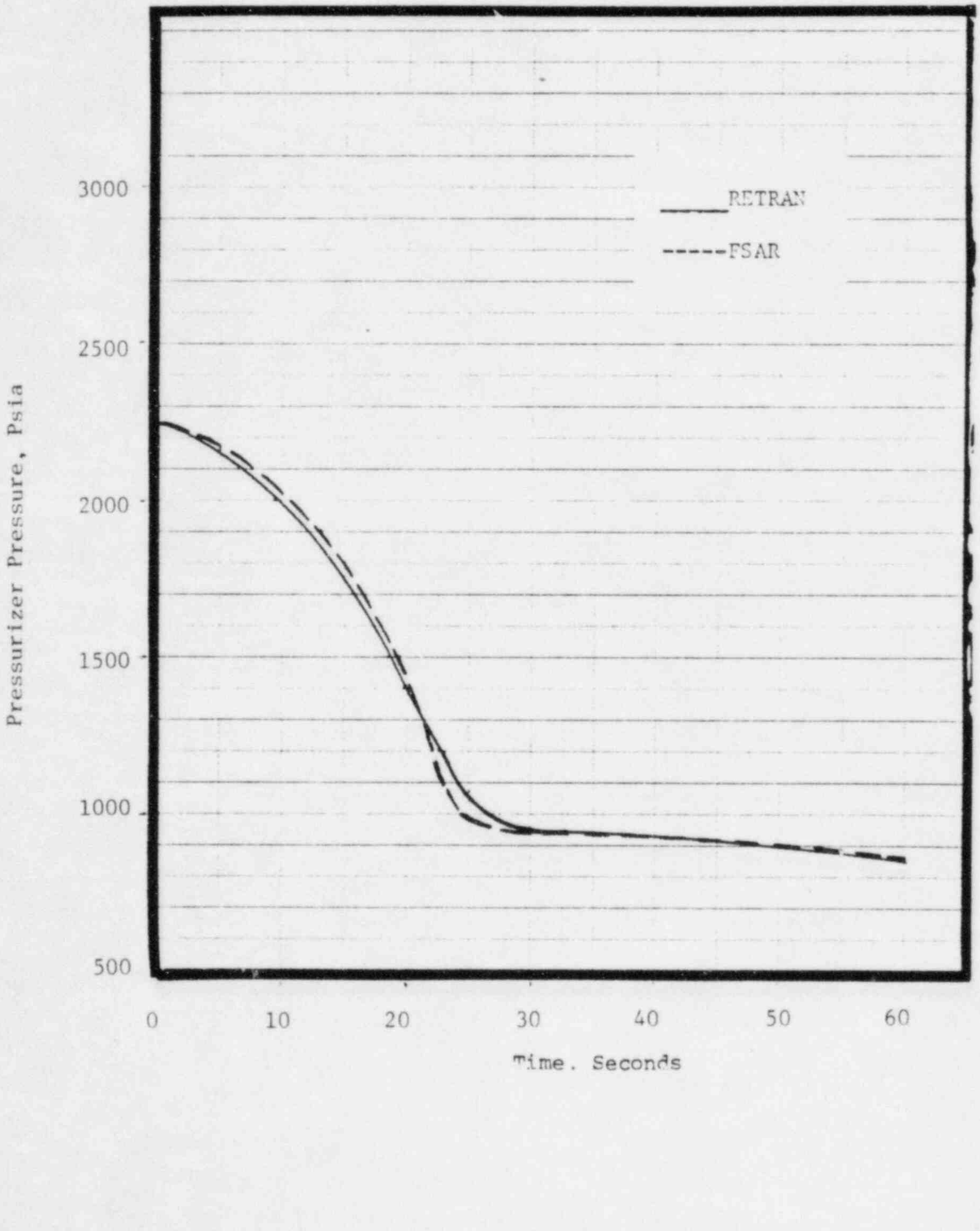


Figure 5.48

TOTAL REACTIVITY
MAIN STEAM LINE BREAK
FSAR ANALYSIS

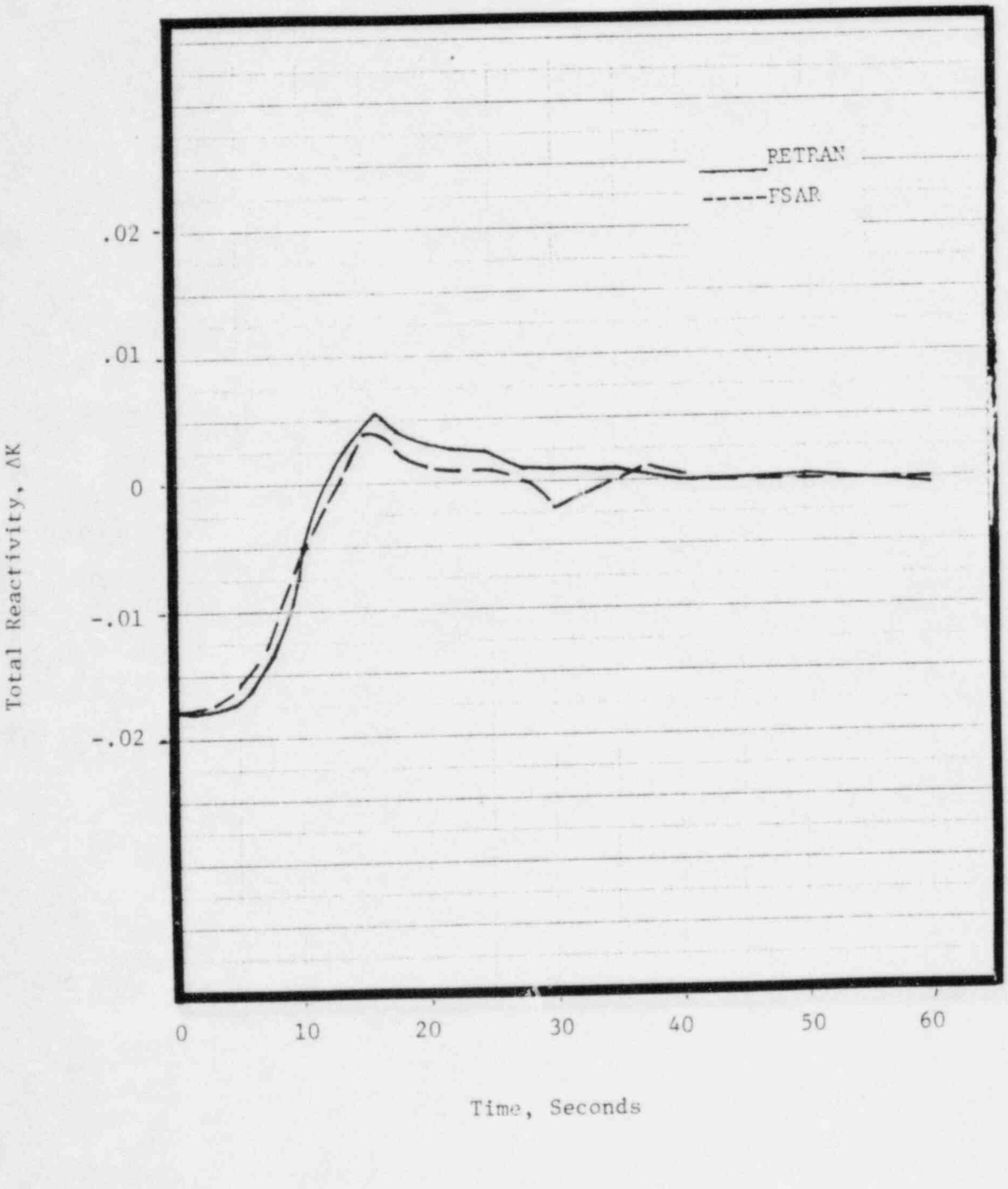


Figure 5.49

CORE HEAT FLUX
MAIN STEAM LINE BREAK
FSAR ANALYSIS

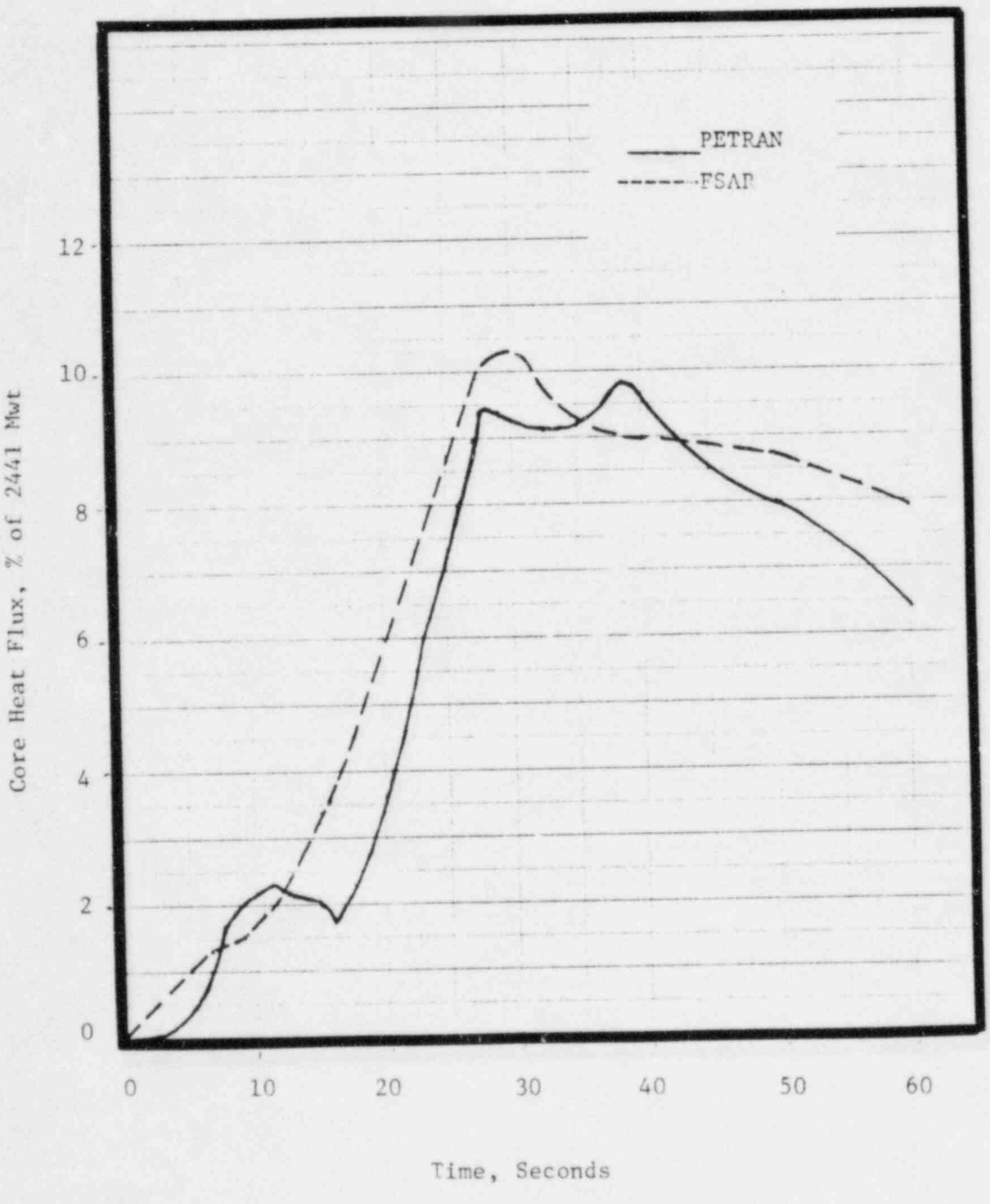


Figure 5.50

PRESSURIZER PRESSURE
MAIN STEAM LINE BREAK TRANSIENT
SURRY 1, CYCLE 4 REANALYSIS

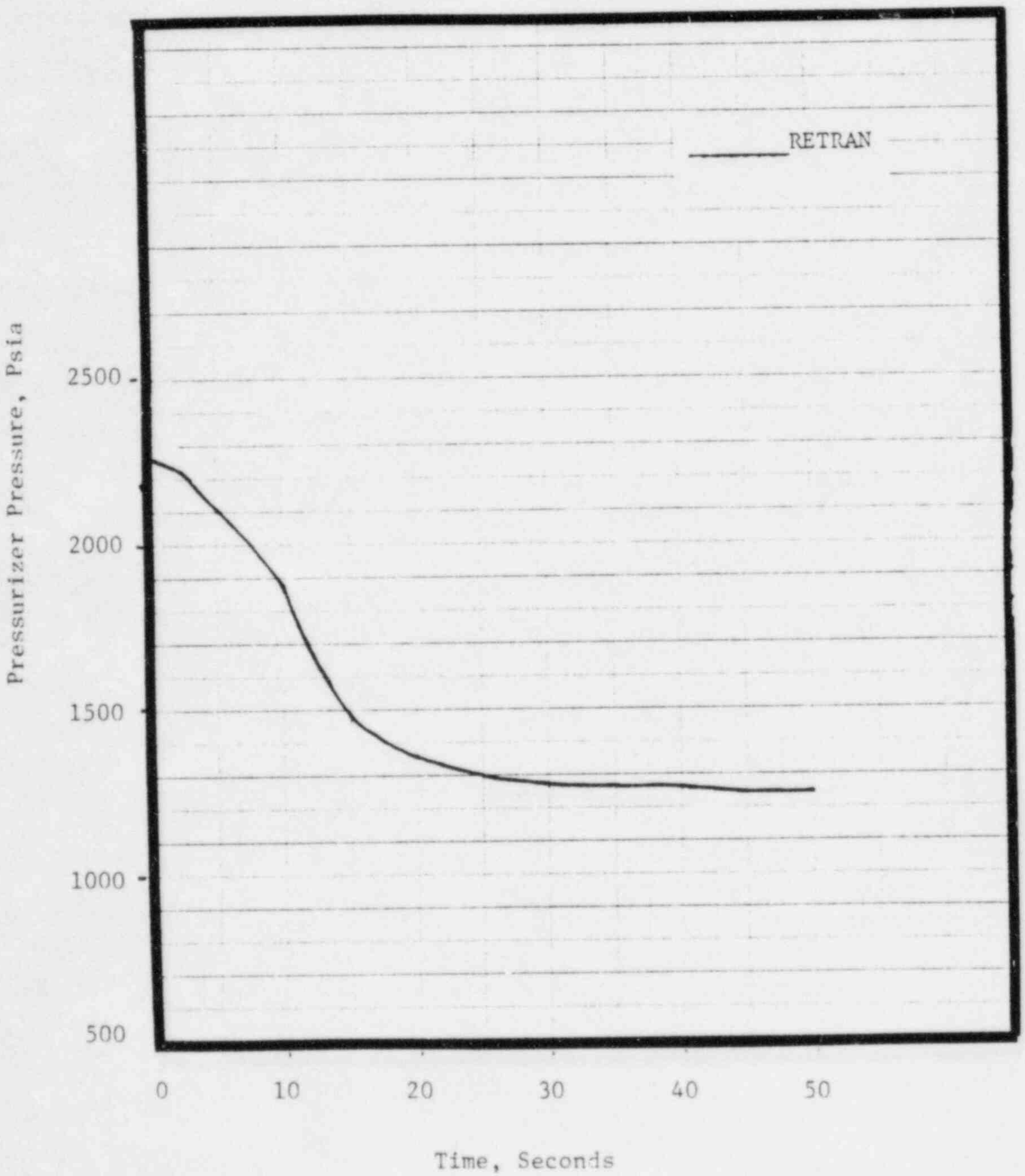


Figure 5.51

TOTAL REACTIVITY
MAIN STEAM LINE BREAK TRANSIENT
SURRY 1, CYCLE 4 REANALYSIS

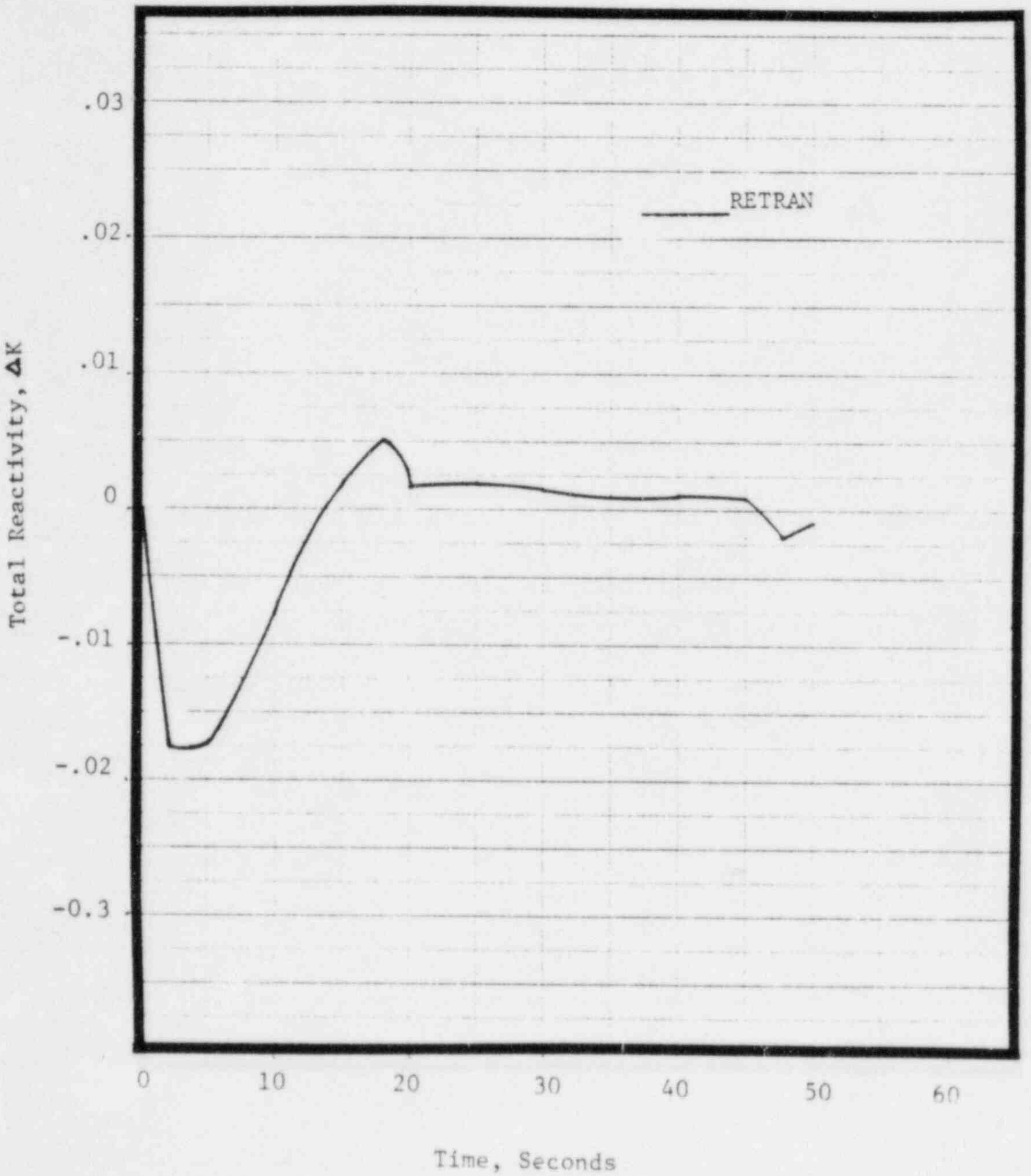
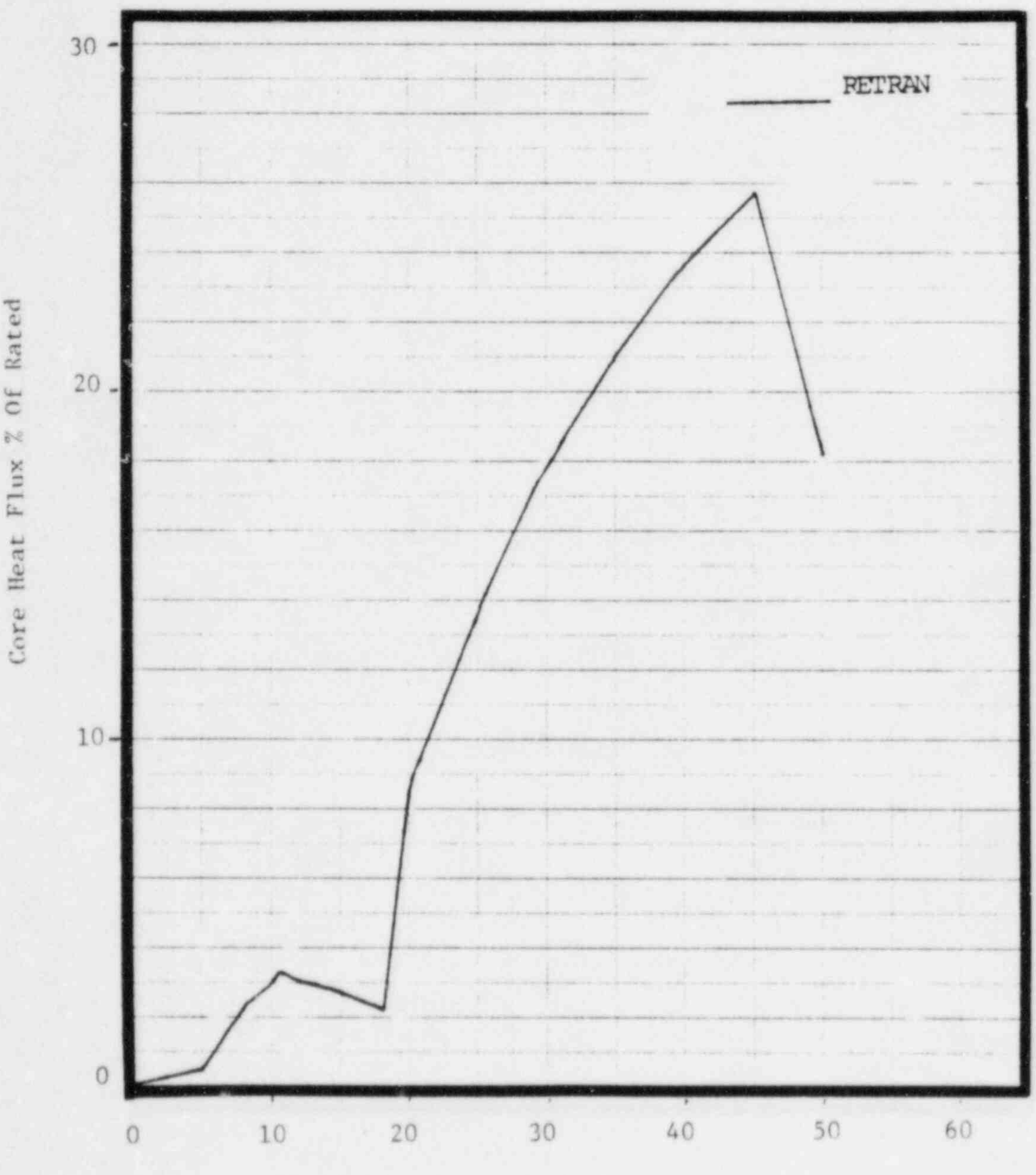


Figure 5.52

CORE HEAT FLUX
MAIN STEAM LINE BREAK TRANSIENT
SURRY 1, CYCLE 4 REANALYSIS



5.3 Verification Against Operational Data

5.3.1 Introduction

The purpose of comparing RETRAN predictions to plant operational data is to demonstrate that the code, coupled with appropriate plant models and best estimate input values, provides physically realistic predictions of integrated system response to various perturbations. Vepco RETRAN comparisons are for the pump coastdown tests performed at both the Surry and North Anna plants and a plant cooldown event which occurred at North Anna Unit 1.

5.3.2 Pump Coastdown Tests

Pump coastdown tests of various configurations (i.e., coastdown of a single pump, two pumps, three pumps, etc.) are performed as part of the initial startup test sequence for new nuclear units. The sections below discuss RETRAN comparisons for a single pump and a simultaneous three pump coastdown for Surry Unit No. 1 and for a simultaneous three pump coastdown performed on North Anna Unit No. 1. Both single loop and two loop RETRAN models were used for the comparisons, as discussed below.

5.3.2.1 Surry Pump Coastdowns

Pump coastdown tests were performed at the Surry Power Station Unit No. 1 in January 1975. The tests were performed with the reactor at hot shutdown conditions with all Rod Cluster Control Assemblies (RCCA) fully inserted. The test results of reactor system flow versus time have been compared with the flow coastdown associated with the Loss of Flow transients reported in the Surry FSAR and with RETRAN analytical predictions using both the Single Loop and Two Loop Surry Models described in Section 3.

The comparison for the simultaneous three pump coastdown is shown in Figure 5.53. The RETRAN code predicts a flow coastdown curve which lies between the FSAR[3] prediction and the test data. Results for this case (3 pump coastdown) were generated with both the Single Loop and Two Loop Surry RETRAN Models. The coastdown curves generated by the two models were essentially identical.

Figure 5.54 compares analytical predictions made with the Surry Two Loop Model with test data for a single pump (two pumps remaining at full speed) coastdown. The data are presented in terms of loop flows. As may be seen, the RETRAN predictions are in close agreement with the data. The same data are presented in terms of core flow fraction in Figure 5.55 to allow an additional comparison to be made, i.e., to the FSAR accident analysis results. As with the three-pump coastdown, the RETRAN curve lies between the FSAR and the data in the region of interest (minimum DNBR for the single pump loss of flow accident occurs at $\sim$3 seconds - see Figure 5.26). It should be noted that although the data indicate a slightly more rapid flow coastdown than either the FSAR or the RETRAN predictions, use of either analytical curve in combination with the conservative FSAR assumptions concerning trip delay times has been shown to provide conservative results for the postulated loss of flow accident.

5.3.2.2 North Anna Pump Coastdown

The three pump coastdown test was performed on North Anna Unit No. 1 in April, 1978. As with the Surry Unit No. 1 test, hot shutdown conditions were maintained. The reactor coolant flow versus time was measured out to 10 seconds following the loss of pump power. The comparison to the FSAR[4] flow coastdown predictions and to the RETRAN analytical predictions is shown in Figure 5.56.

The RETRAN results agree quite well with the FSAR, particulary over the first four seconds of the transient, which in a complete loss of flow accident is the most limiting period from the standpoint of DNB. Note that both the FSAR and RETRAN predict a slightly slower coastdown than the data indicates over this same period. As discussed above, slight deviations are evaluated at the time of the test to ensure the overall conservatism of the FSAR analyses.

In summary, the RETRAN pump coastdown calculations performed with the Surry One and Two Loop Models and the North Anna One Loop model have been shown to give results which agree well with the measured data.

5.3.3 North Anna Cooldown and Safety Injection Transient

An analysis was performed to simulate the unplanned cooldown event which occurred at North Anna Unit 1 on September 25, 1979.[14] The following sections provide 1) a brief description of the event; 2) a description of the RETRAN model used for the analytical simulation; 3) comparisons of RETRAN results with plant data taken at the time of the event; and 4) conclusions regarding the analysis and data comparisons.

The North Anna cooldown event resulted from a turbine trip and subsequent reactor trip on high feedwater heater condensate level. The high level signal was the result of tube leakage inside the heater drain cooler. Following the trip the eight condenser dump valves tripped fully open to supplement the reactor trip in providing load rejection capability. As the plant began to cool down seven of the eight dump valves modulated closed as designed. The remaining valve stuck in its fully open position. This resulted in additional cooldown beyond the no-load temperature, causing a depressurization of the reactor coolant system and initiation of Safety Injection on low pressurizer pressure. Following Safety Injection, the operator tripped the reactor coolant pumps in accordance with procedures and the system rapidly repressurized to the normal pressure range. One of the two high head safety injection pumps was tripped; the RCS continued to repressurize at a slower rate until one of two pressurizer Power Operated Relief Valves (PORV's) opened on a high pressure signal. This valve then cycled to maintain RCS pressure at the relief setpoint. Normal pressure was restored by a combination of operator actions, including initiation of auxiliary spray, realignment of the charging pumps to the normal charging path, throttling the charging flow and reestablishment of letdown flow.

The RETRAN model used to simulate the cooldown event is a 20-volume, single loop representation of the North Anna Reactor Coolant System, steam generators and associated control systems. The general description of Vepco's Single Loop Models, given in Section 3, is also applicable to this model. Additional features included in this

model to provide a best estimate analysis capability include the following:

1) Representation of the automatic steam dump control system.

2) Simplified representation of the feedwater control (steam generator level) system.

3) Representation of the High Head Safety Injection system

4) Automatic charging flow (pressurizer level) control in combination with RCS letdown.

5) Representation of the following operator actions as boundary conditions:

   -Manual tripping of the primary coolant pumps shortly after Safety Injection

   -Manual tripping of one charging pump after Safety Injection had restored pressurizer pressure and level to their normal values

   -Manual tripping of the Main Steam Isolation Valves to terminate the steam release shortly after Safety Injection initiation

   -Manual termination of auxiliary feedwater flow.

The following discussion provides a comparison of analytical results to plant data obtained at the time of the cooldown. Plant data sources include alarm typewriter printout and control room strip chart recordings. The resolution of the alarm printout, which is the source of most of the data, is plus or minus thirty seconds.

Figure 5.57 shows the depressurization of the main steam system. The alarm typewriter data are representative of all three loops. Examination of the data indicated that the depressurization took place in a symmetric manner. Note from the figure the pronounced impact of operator intervention on the pressure response.

Figures 5.58 and 5.59 compare calculated and observed cold and hot leg temperatures, respectively. The cold leg temperature data in Figure 5.58 from 0 to 300 seconds are based on alarm typewriter printout of narrow range Tcold. The data points

represented by triangles are Tcold values inferred from alarm typewriter steam pressure data. These points were derived by table lookup of the saturation temperature of the steam system and correction by the calculated primary to secondary temperature difference.

The dashed line represents control room strip chart data. As can be seen, the general agreement of the model with the data is good. The predicted reactor vessel ΔT under natural circulation conditions is slightly lower than the measured value.

Figure 5.60 shows the pressurizer pressure response. The calculated initial depressurization and repressurization following Safety Injection initiation at 300 seconds show excellent agreement. This good agreement provides further qualification for the RETRAN nonequilibrium pressurizer model.

Figure 5.61 shows the pressurizer level response. Both the observed data and the model indicate that pressurizer level indication was lost for a brief portion of the transient. The model predicted a slightly lower drain rate during cooldown than was observed. This may reflect a difference in the assumed initial pressurizer mixture quality and the actual plant condition. The general agreement is still quite good over the first 10 minutes of the transient. The underprediction at 1400 seconds is possibly related to the integral effects of RETRAN's underprediction of the safety injection flow rate at elevated system pressures.

5.3.4 <u>General Conclusions-Best Estimate Transient Analyses</u>

The comparisons of best estimate RETRAN predictions to plant data presented in sections 5.3.1-5.3.3 (Figures 5.53-5.61) are indicative of Vepco's best estimate analytical capabilities; the favorable results shown here provide a sound basis for applying this capability to general plant operational support.

Figure 5.53

FLOW COASTDOWN
OPERATIONAL TEST AT HOT ZERO POWER
SURRY THREE - PUMP COASTDOWN

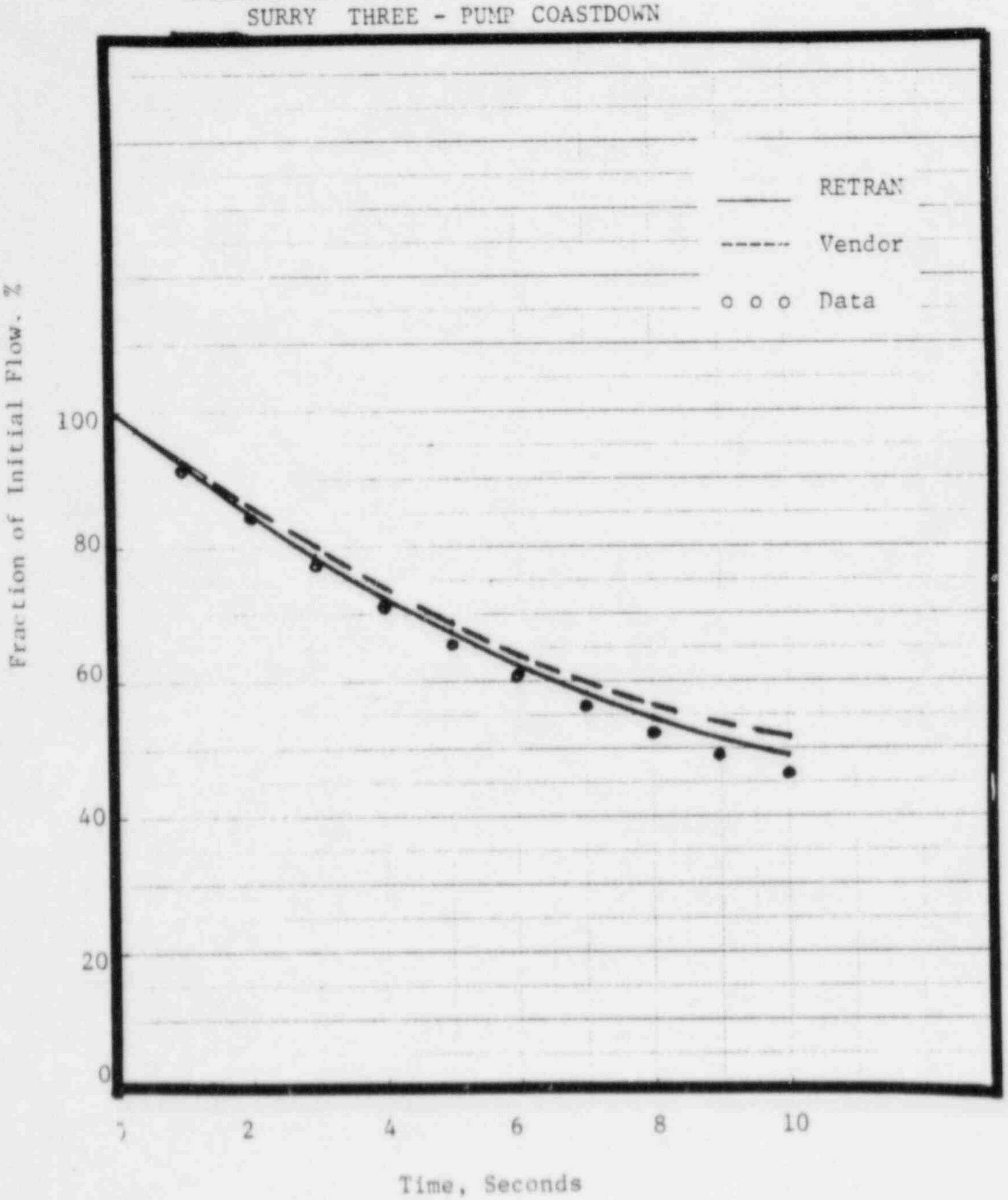


Figure 5.54

FLOW COASTDOWN
OPERATIONAL TEST AT HOT ZERO POWER
SURRY ONE PUMP COASTDOWN

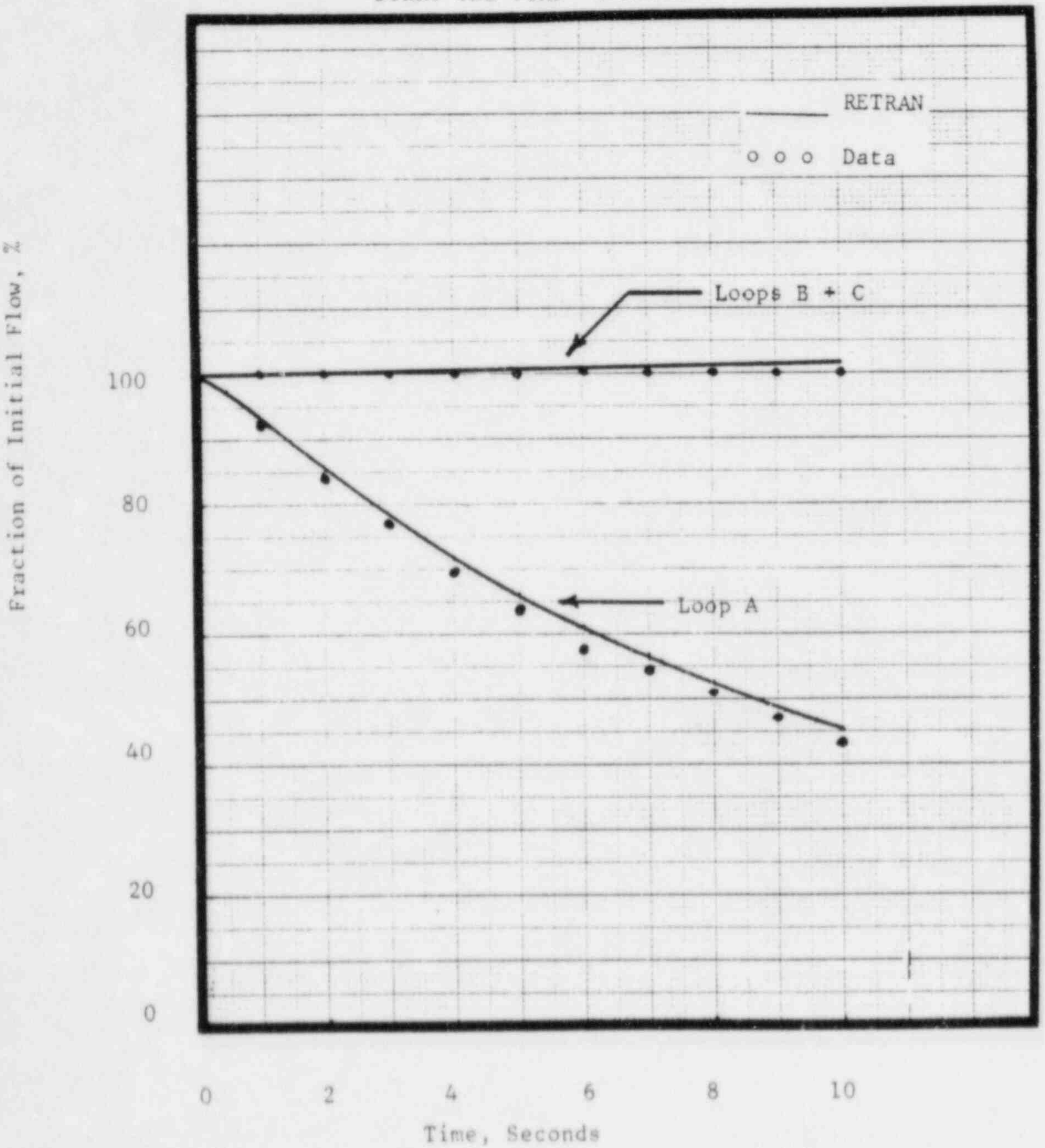


Figure 5.55

CORE FLOW COASTDOWN
OPERATIONAL TEST AT HOT ZERO POWER
SURRY ONE-PUMP COASTDOWN

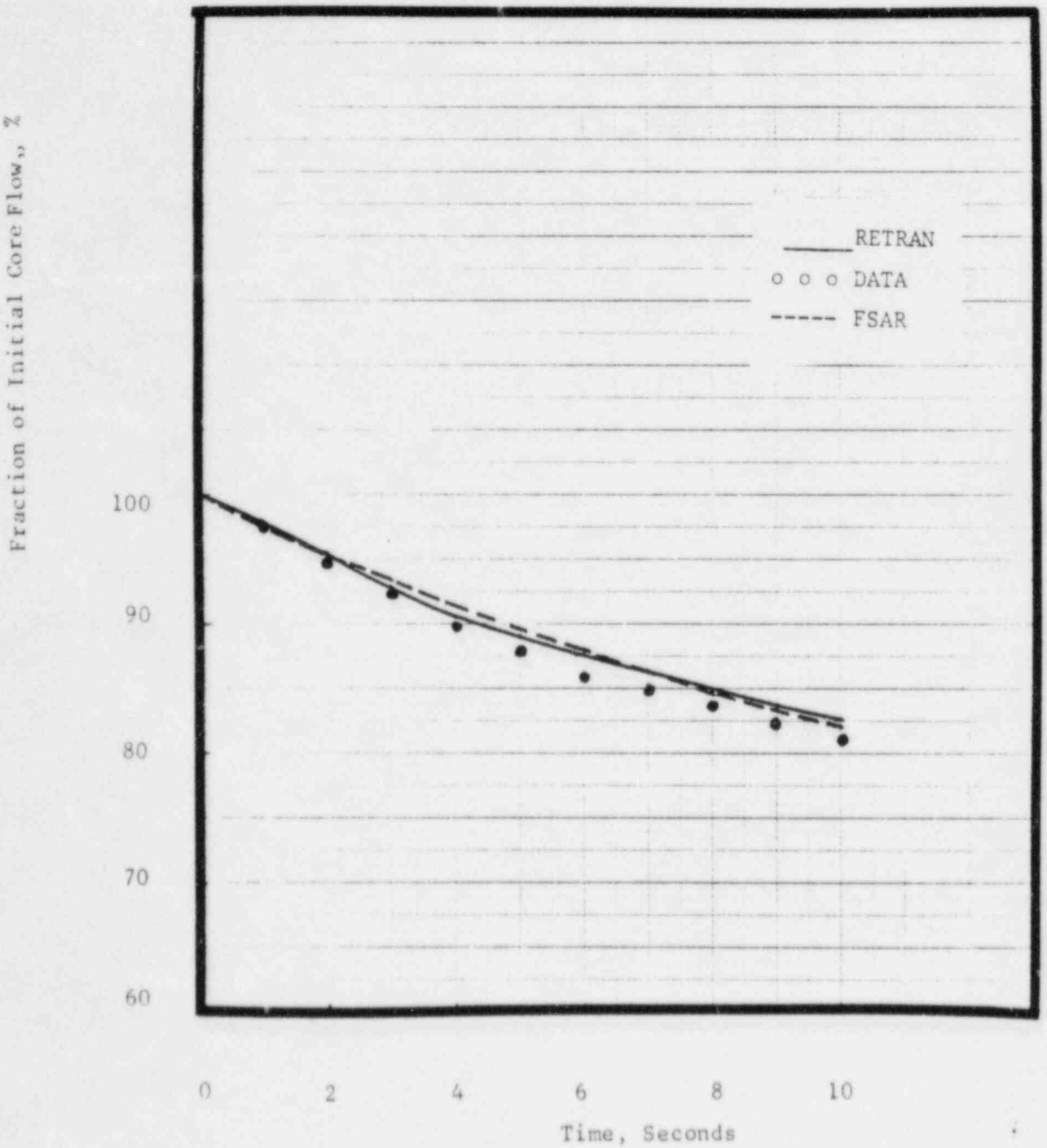


Figure 5.56

FLOW COASTDOWN
OPERATIONAL TEST AT HOT ZERO POWER
NORTH ANNA THREE - PUMP COASTDOWN

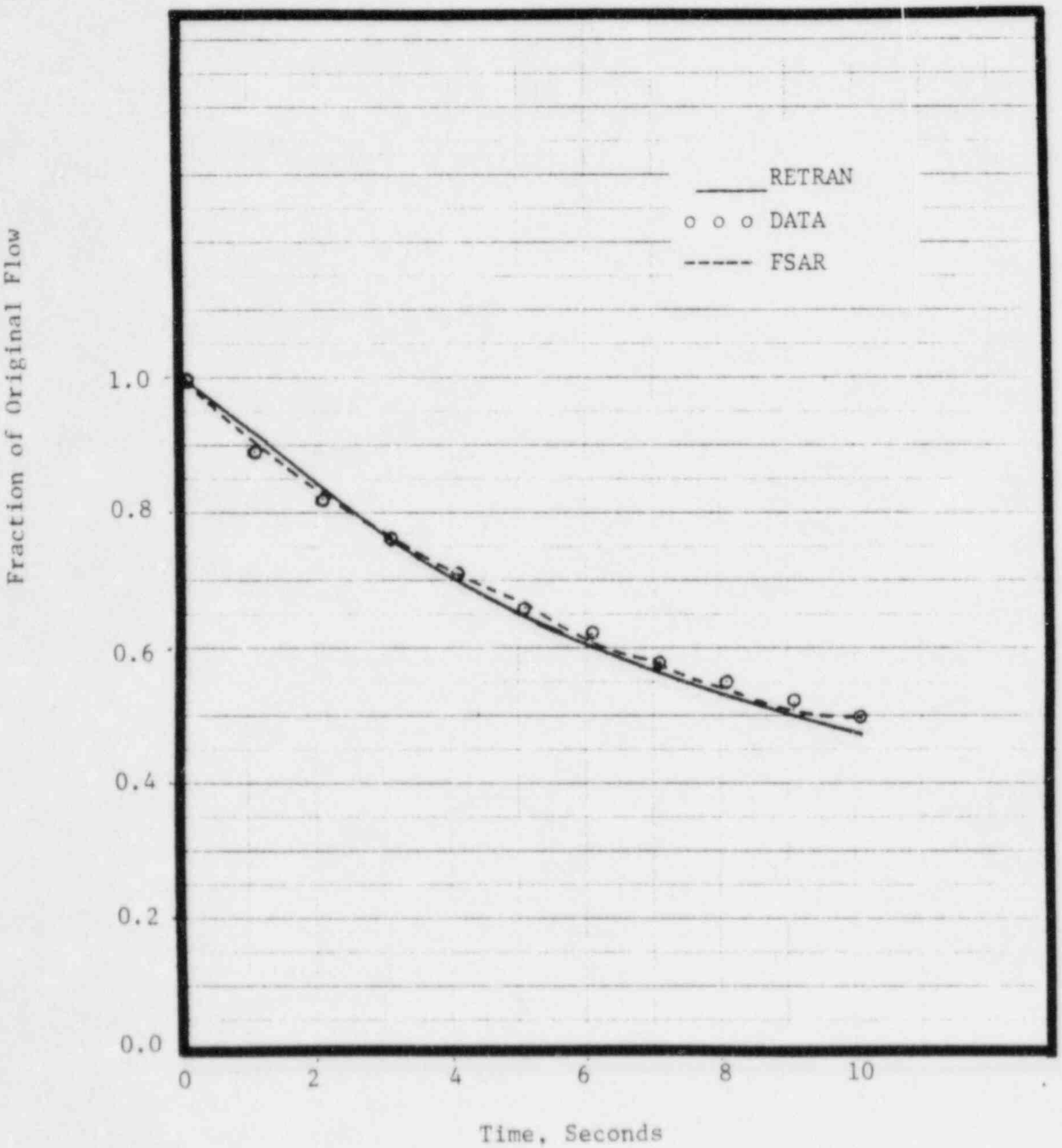


Figure 5.57

STEAM PRESSURE
NORTH ANNA COOLDOWN EVENT

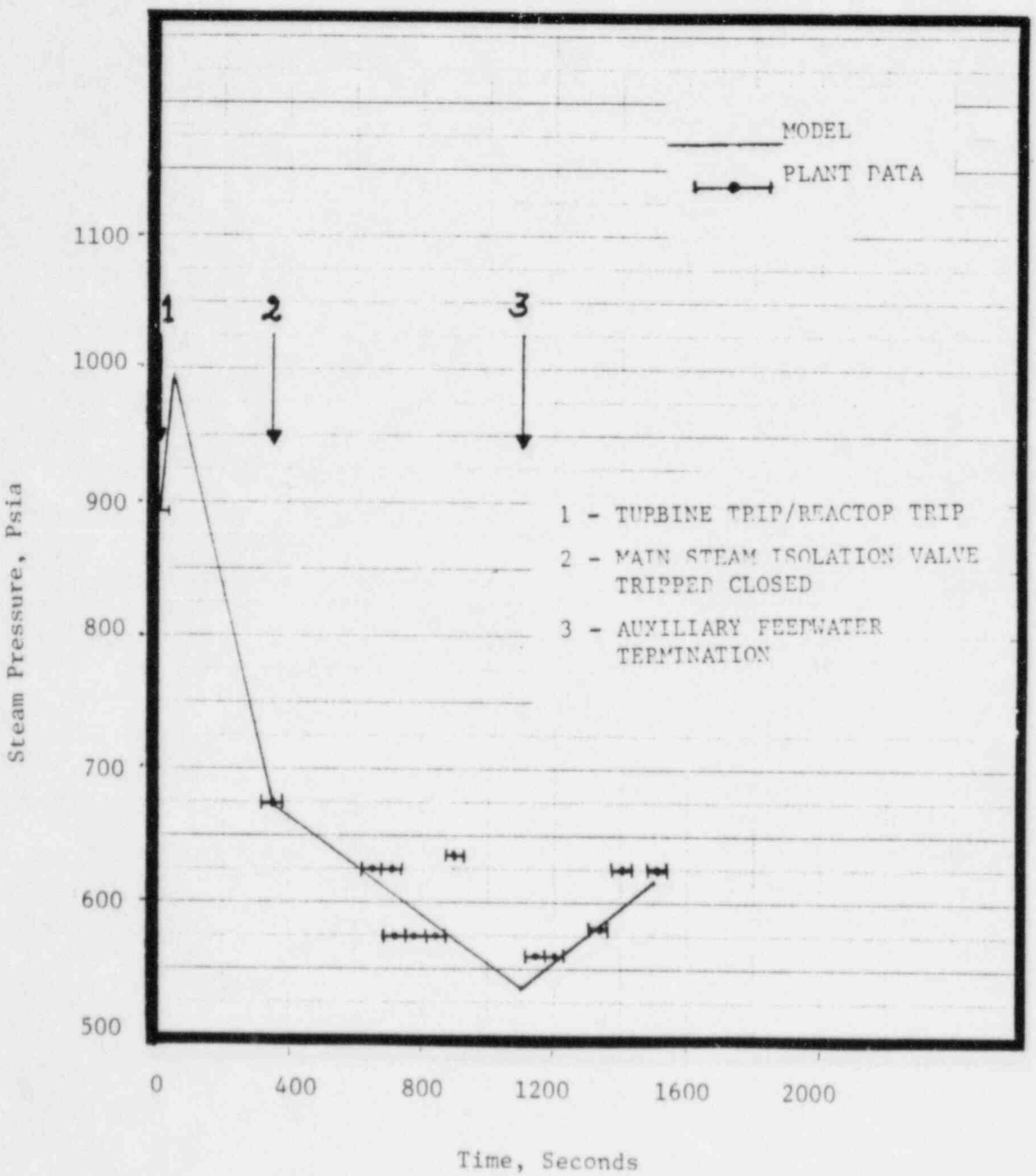


Figure 5.58

COLD LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT

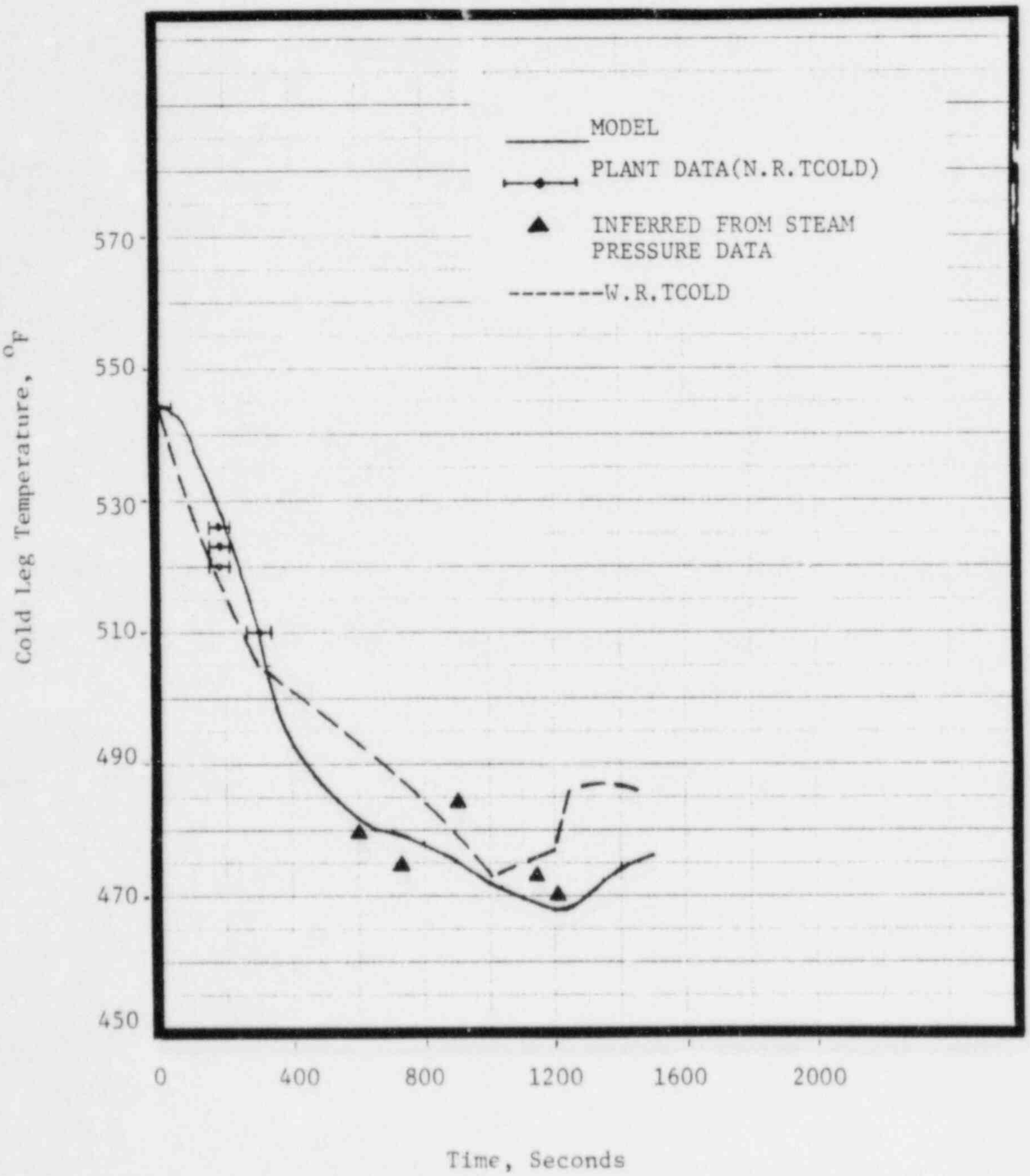


Figure 5.59

HOT LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT

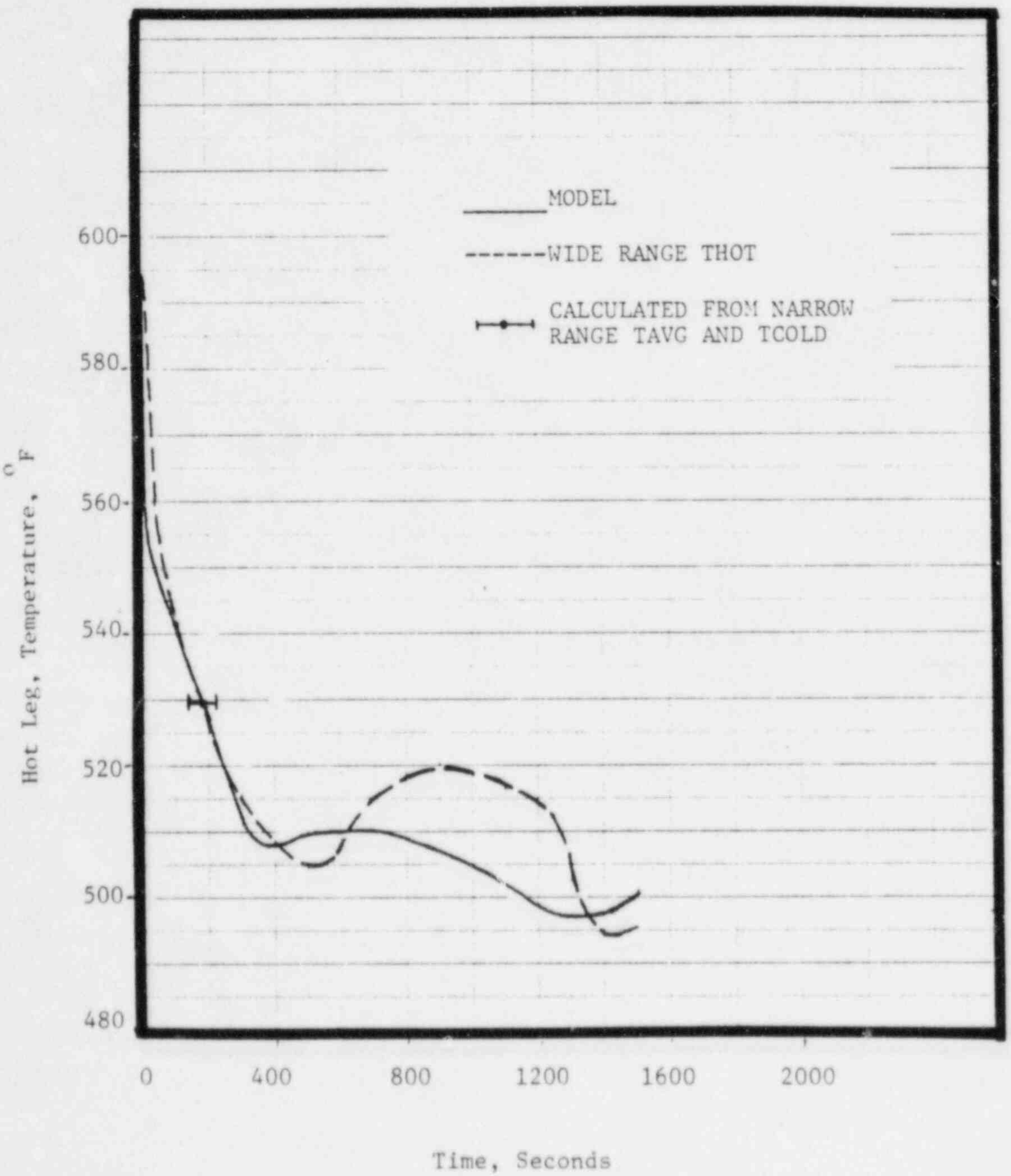


Figure 5.60

PRESSURIZER PRESSURE
NORTH ANNA COOLDOWN EVENT

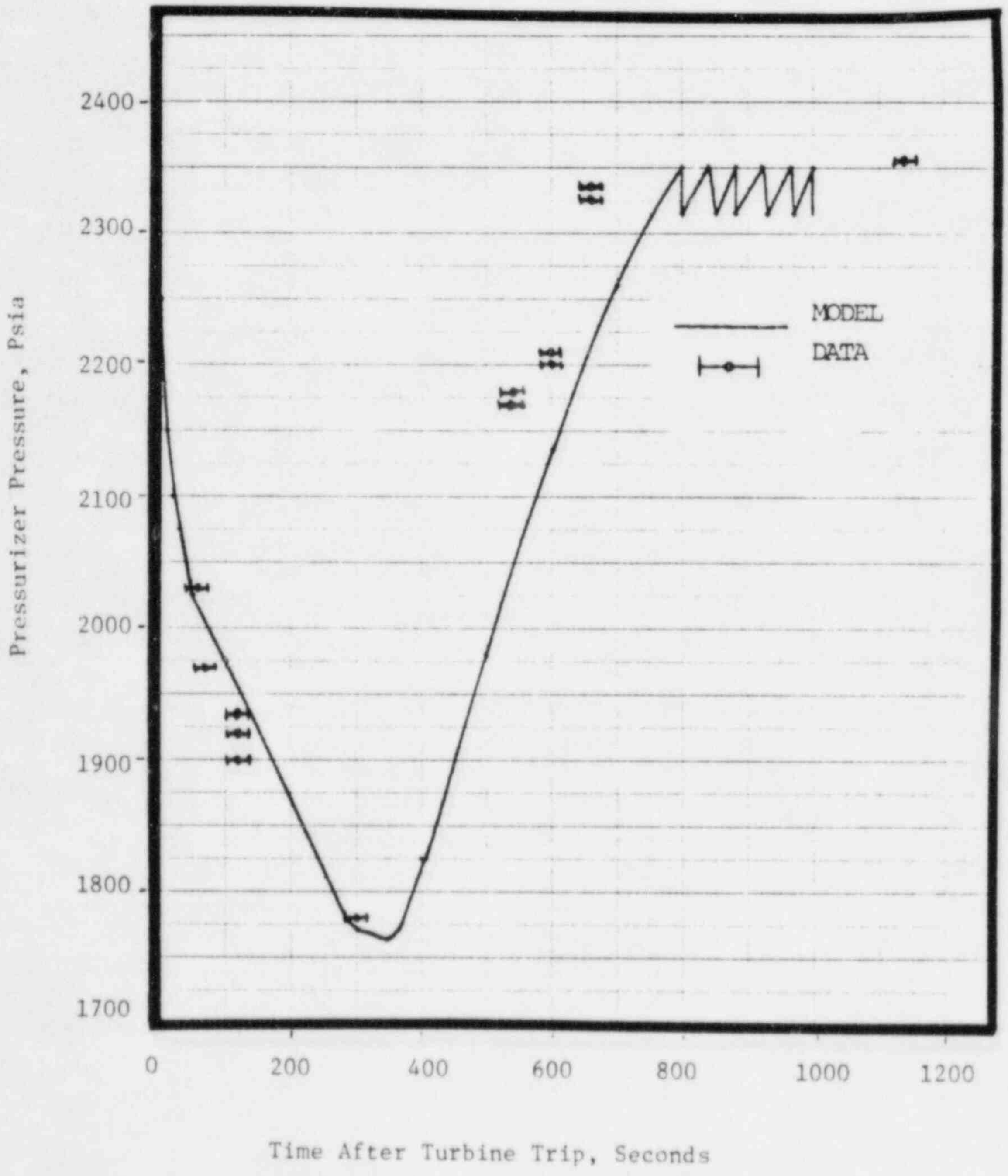
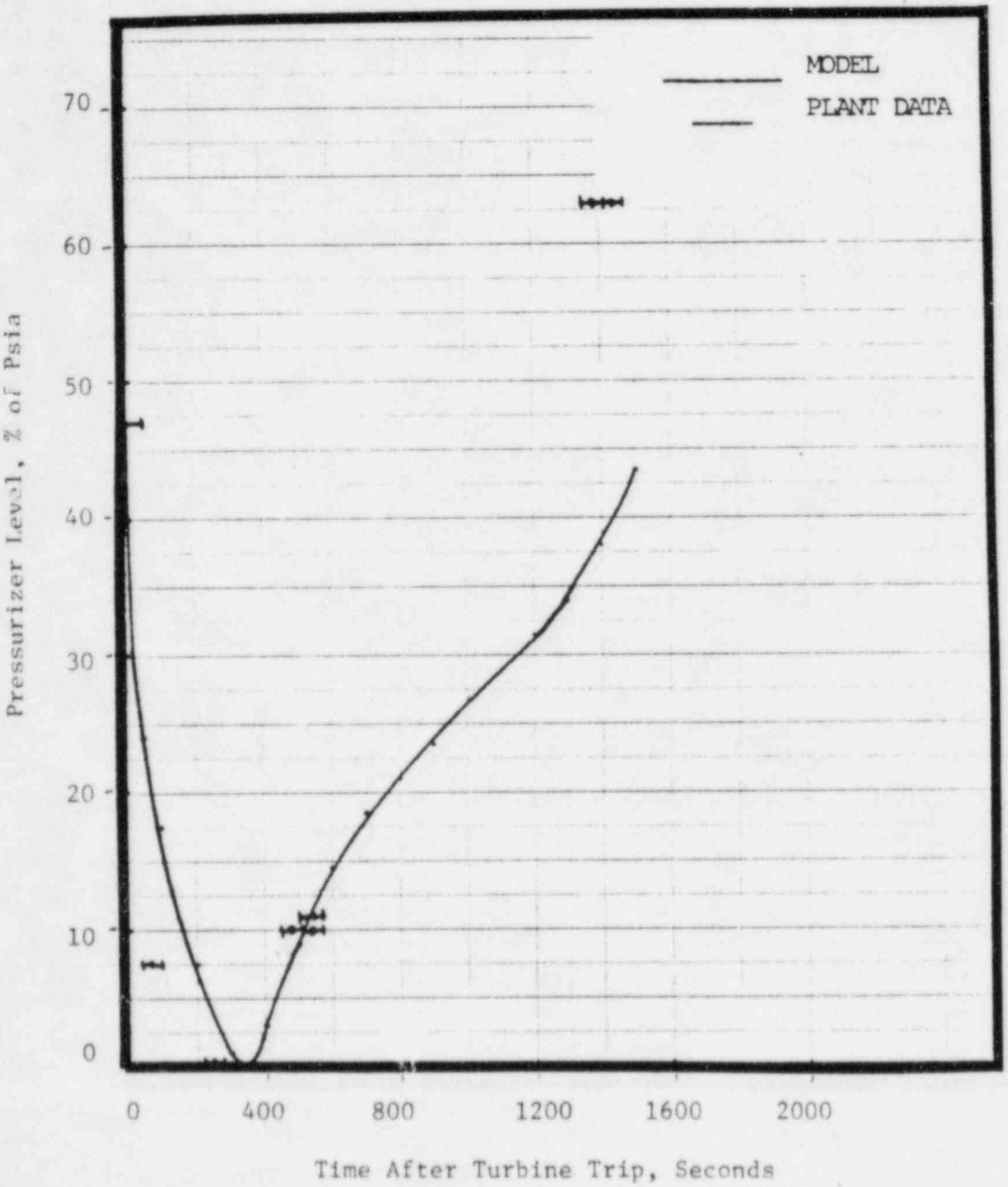


Figure 5.61

PRESSURIZER LEVEL
NORTH ANNA COOLDOWN EVENT



SECTION 6 - CONCLUSIONS

The Virginia Electric and Power Company (Vepco) has developed the capability to perform system transient analyses using the RETRAN computer code. The general code features and the types of models developed for analysis of the Surry and North Anna Units 1 and 2 have been discussed. The adequacy of these models and the associated accident analysis methodology has been demonstrated by comparison of selected analytical results to vendor calculations and to plant data. The overall good agreement realized in these comparisons demonstrates that these models and methods can be used for operational and licensing support of Vepco's nuclear plants.

SECTION 7 - REFERENCES

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3. Surry Power Station Units 1 and 2, Final Safety Analysis Report, Virginia Electric and Power Company, December 1, 1969.

4. North Anna Power Station Units 1 and 2, Final Safety Analysis Report, Virginia Electric and Power Company, January 3, 1973 as amended.

5. NUREG-75/056, "WREM: Water Reactor Evaluation Model," Revision 1, May 1975.

6. WCAP-7978, "LOFTRAN Code Description," Rev. 1, Westinghouse Electric Corporation, January 1977.

7. WCAP-7909, "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," Westinghouse Electric Corporation, October 1972.

8. WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," Westinghouse Electric Corporation, March 1978.

9. Letter from C. M. Stallings (Vepco) to E. G. Case (NRC), Serial No. 403, September 9, 1977.

10. Letter from C. M. Stallings (Vepco) to E. G. Case (NRC), Serial No. 344, August 9, 1977.

11. WCAP-7769, "Overpressure Protection for Westinghouse Pressurized Water Reactors," Westinghouse Electric Corporation, June, 1972.

12. Letter from C. M. Stallings (Vepco) to K. R. Goller (NRC), Serial No. 553, June 5, 1975.

13. Letter from C. M. Stallings (Vepco) to B. C. Rusche (NRC), Serial No. 256, September 27, 1976.

14. Letter from C. M. Stallings (Vepco) to J. P. O'Reilly (NRC), Serial No. 829, transmitting LER 79-128/01T-0, October 9, 1979.

APPENDIX

SUMMARY OF IMPORTANT ASSUMPTIONS USED IN TRANSIENT ANALYSES DISCUSSED IN SECTION 5

Type of Analysis	System Model Description	Initial Conditions	Transient Specific Input: Reactivity Parameters	Transient Specific Input: Key System Performance Assumptions[1]
1. Uncontrolled Rod Withdrawal from Subcritical				
a) FSAR Analysis	Surry One Loop	Core power = 10^{-13} x rated Pressure = 2220 psia T-inlet = 550°F	αMOD=+10 pcm/°F αDOP=-1.75pcm*/°F (@/550°F)	
			Delayed neutron fraction = 0.0072	No credit taken for: 1) Source range high flux trip
			Reactivity Insertion Rate=60 pcm/sec	2) Intermediate range high flux trip 3) Intermediate range control rod stop
			Trip Reactivity: Fig. A.1 curve(a), total = 2.8% ΔK/K	Source of protection: Low power range high neutron flux trip
b) Current Analysis	Surry One Loop	Same as case (a)	Same as case (a), except Reactivity Insertion Rate =75 pcm/sec	Other assumptions same as case(a)
			Trip reactivity: Fig. A.1 curve (b), total=4.0% ΔK/K	

(1) Trip setpoints and delay times assumed are consistent with Table 4.1

*1 pcm=1.0×10^{-5} ΔK/K

APPENDIX (Continued)

SUMMARY OF IMPORTANT ASSUMPTIONS USED IN SAFETY ANALYSES DISCUSSED IN SECTION 5

			Transient Specific Input	
Type of Analysis	System Model Description	Initial Conditions	Reactivity Parameters	Key System Performance Assumptions
2. Uncontrolled Rod Withdrawal from Power				
a) FSAR Analysis	Surry One Loop	Core power = 1.02 x rated Pressure = 2220 psia T-inlet = 547.1 °F	αMOD = 0.0 αDOP= -0.725 pcm/°F Reactivity Insertion Rate = 2.0 pcm/sec Trip Reactivity: Figure A.1 curve (a), total =2.8% ΔK/K	No credit taken for: 1) High neutron flux rod stop 2) High overtemperature ΔT rod stop 3) High overpower ΔT rod stop or trip Source of Protection: High overtemperature ΔT trip*

ΔT trip equation used (includes errors):

$$\Delta T\ (\text{setpoint}) = \left[1.2044 - .0113\frac{1+25s}{1+3s}(T_{ave}-574.4) + .00056\ (P-2250)\right] \times \Delta T\text{-Rated}$$

APPENDIX (Continued)

SUMMARY OF IMPORTANT ASSUMPTIONS USED IN TRANSIENT ANALYSES DISCUSSED IN SECTION 5

			Transient Specific Input	
Type of Analysis	System Model Description	Initial Conditions	Reactivity Parameters	Key System Performance Assumptions
2. Uncontrolled Rod Withdrawal from Power				
b) Current Analysis 1) From 102% power	Surry One Loop (Modified to reflect steam generator tube plugging)	Core power = 1.02 x rated Pressure = 2220 psia T-inlet = 543.4°F RCS Flow = 90% of full power thermal design	αMOD=+1.0 pcm/°F Doppler power coefficient =-6.0 pcm/% at 100% power Reactivity insertion rate varied Trip Reactivity: Figure A.1 curve (a), total = 2.8% ΔK/K	No credit taken for: 1) High neutron flux rod stop 2) High Overtemperature ΔT rod stop 3) High overpower ΔT rod stop or trip Source of protection: High power range high neutron flux trip or High overtemperature ΔT trip
2) From 62% power		Core power = 0.62xrated Pressure = 2220 psia T-inlet = 550.3°F RCS Flow = 90% of full power thermal design	Trip Reactivity: Fig. 4.1 curve (a), total=2.8%ΔK/K αMOD=+1.0pcm/°F αDOP =-7.3 pcm/% at 62% power Reactivity insertion rate varied	Assumptions same as high power case

ΔT trip equation used (includes errors):

$$\Delta T(\text{Setpoint}) = (1.166 - .0095\ \left(\frac{1+30s}{1+4s}\right)(T_{ave} - 574.4) + .0005\ (P-2250))\ X\ \Delta T\ \text{Rated}$$

APPENDIX (Continued)

SUMMARY OF IMPORTANT ASSUMPTIONS USED IN TRANSIENT ANALYSES DISCUSSED IN SECTION 5

Type of Analysis	System Model Description	Initial Conditions	Transient Specific Input: Reactivity Parameters	Transient Specific Input: Key System Performance Assumptions
3. Complete Loss of Forced Reactor Coolant Flow				
a) FSAR Analysis	Surry One Loop	Core power = 1.02x rated Pressure = 2220 psia T-inlet = 547°F	αMOD = 0 αDOPPLER =-1.6 pcm/$^{\circ}$F Trip reactivity: Fig. A.1, Curve (a) Total = 2.8% ΔK/K	Source of protection: Low RC Pump voltage
b) Current Analysis	Surry One Loop (Modified to reflect steam generator tube plugging)	Core Power = 1.02x rated Pressure = 2220 psia T-inlet = 547.1°F	αMOD=+3.0 pcm/$^{\circ}$F αDOPPLER=-1.6 pcm/$^{\circ}$F Trip reactivity: Fig. A.1 Curve (b) Total = 4.0% ΔK/K	Source of protection: Low RC Pump voltage Conservative (low)initial flow was assumed
4. Partial Loss of Forced Reactor Coolant Flow	Surry Two Loop	Core power = 1.02xrated Pressure = 2220 psia T-inlet = 547.1°F	αMOD = 0.0 αDOPPLER=-1.6 pcm/$^{\circ}$F Trip Reactivity: Fig. A.1, curve(a) Total =2.8% ΔK/K	Source of protection: Low RC loop flow rate

APPENDIX (Continued)

SUMMARY OF IMPORTANT ASSUMPTIONS USED IN TRANSIENT ANALYSES DISCUSSED IN SECTION 5

Type of Analysis	System Model Description	Initial Conditions	Transient Specific Input: Reactivity Parameters	Transient Specific Input: Key System Performance Assumptions
5. Loss of External Electrical Load				
a) FSAR Analysis	Surry One Loop	Core power = 1.02xrated Pressure = 2220 psia T-inlet = 547.2°F	Beginning of Life: αMOD - 0.0 αDOPPLER =-1.6 pcm/°F Delayed neutron fraction=.0072 End of Life: αMOD=-35pcm/°F αDOPPLER=-1.6 pcm/°F Delayed neutron fraction=.0048 Trip reactivity: Fig.A.1,curve(a) Total = 2.8% ΔK/K	No credit taken for: 1) Pressurizer spray 2) Pressurizer power operated relief valves 3) Atmospheric steam dump valves 4) Atmospheric steam relief valves 5) Direct reactor trip resulting from a tubine-generator trip Source of protection: High pressurizer pressure trip
b) Current Analysis	Surry One Loop	Core power = 1.02xrated Pressure = 2220 psia T-inlet = 547.2°F	αMOD=+3.0pcm/°F αDOPPLER=-1.6 pcm/°F Delayed neutron fraction = .0072 Trip reactivity: Fig.A.1,curve(a) Total = 2.8% Δ K/K	Key assumptions are the same as for the FSAR

APPENDIX (Continued)

SUMMARY OF IMPORTANT ASSUMPTIONS USED IN TRANSIENT ANALYSES DISCUSSED IN SECTION 5

Type of Analysis	System Model Description	Initial Conditions	Transient Specific Input: Reactivity Parameters	Transient Specific Input: Key System Performance Assumptions
6. Excessive Heat Removal Due to Feedwater System Malfunction	Surry One Loop	Core power = 1.02xrated Pressure = 2220 psia T-inlet = 547.2°F	αMOD = 0.0 αDoppler=-1.0pcm/°F Trip Reactivity: Fig.A.1, curve (a) Total = 2.8% ΔK/K	Reactor assumed to be in manual control (Tave control inactive) Source of protection: none required
7. Main Steam Line Break				
a) FSAR Analysis	Surry Two Loop	Core power = $4x10^{-8}$x rated Pressure = 2251 psia T-inlet = 549.7°F	αMOD=-25.4 pcm/°F @550°F, -13.8 pcm/°F @300°F αDOPPLER(Zero power) =-1.6pcm/°F Total power defect at 30% power=-.0135ΔK Differential boron worth =- 10pcm/ppm	Technical Specifications value for initial shutdown reactivity margin assumes the highest worth control rod assembly stuck in its fully withdrawn position Safety injection capability based on failure of one high-head safety injection pump No credit is taken for the effect of the main steam line check valves in precluding discharge of secondary fluid from the intact steam generators prior to main steam isolation valve closure

APPENDIX (Continued)
SUMMARY OF IMPORTANT ASSUMPTIONS USED IN TRANSIENT ANALYSES DISCUSSED IN SECTION 5

Type of Analysis	System Model Description	Initial Conditions	Transient Specific Input: Reactivity Parameters	Transient Specific Input: Key System Performance Assumptions
b) Current Analysis	Surry Two Loop	Core power=$4x10^{-8}$xrated Pressure = 2251 psia T-inlet = 549.7^oF	αMOD=-25.4pcm/^oF @)550^oF, -13.8pcm/^oF @)300^oF αDoppler(Zero power) =-1.8pcm/^oF Total power defect at 30% power =-.0148ΔK Differential boron worth =-10pcm/ppm	Key performance assumptions are the same as for the FSAR Analysis, above

FIGURE A.1

TRIP REACTIVITY INSERTION CHARACTERISTICS