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Dominion®

VEP-FRD-41, Rev. 0.1-A

Vepco Reactor System Transient Analyses Using the RETRAN Computer Code [Volume 1]

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VEPCO REACTOR SYSTEM TRANSIENT ANALYSES USING THE RETRAN COMPUTER CODE

By

NUCLEAR ANALYSIS & FUEL

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June 2004

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VEP-FRD-41, Rev 0.1-A

RETRAN TOPICAL REPORT

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CLASSIFICATION/DISCLAIMER

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ABSTRACT

This Topical Report describes Virginia Electric and Power Company's (Dominion's) Reactor System Transient Analysis models for use with the RETRAN-02 Computer Code. These models have been qualified for UFSAR Chapter 14 and 15 Non-LOCA transient analyses to support Surry and North Anna Units 1 and 2. The various reactor system component models are described and qualified for their intended applications. Comparisons to plant data and alternate calculations are provided. Restrictions and limitations and conditions of use imposed by the USNRC's generic Safety Evaluation Reports for the RETRAN-02 computer code are addressed.

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1. INTRODUCTION

This report is an update of VEP-FRD-41-A, VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code (Appendix 1). This report has been designated VEP-FRD-41-A, Rev. 0.1 (Rev. 0, Mod 1). The purpose of Mod 1 of the report is to consolidate all changes to the Dominion RETRAN models and to the RETRAN Code which have been reviewed and approved since the initial issuance of Rev. 0-A of the topical report in April 1985 (Reference 1.0-1). Responses to NRC requests for additional information (RAIs) pertaining to review of Rev. 0 of the report are provided for reference in **Appendixes 2-4.**

RETRAN-02 is a general purpose thermal hydraulics code for transient analyses of complex fluid flow systems. It contains an input structure that allows for the development of models to represent all types of light water reactors. RETRAN-02 has been used extensively by the U.S. and international safety analysis community for two decades. Details of the RETRAN-02 theory, numerics, qualification and application guidelines may be found in Reference 1.0-2.

Dominion's models are currently being used with RETRAN-02 Mod 5.2 (Reference 1.0-3). A history of generic RETRAN code approvals is provided in Section 3.

For a general description of Dominion's Surry and North Anna plants, see Section 3 of Appendix 1. Section 4 of Appendix 1 provides an overview of Dominion's system transient analysis methodology.

Section 5 of Appendix 1 provides qualification comparisons to vendor (UFSAR) accident analyses and plant transient data for the original 1-loop and 2-loop RETRAN models using RETRAN-01. Following NRC approval of Rev. 0 of this report, Dominion changed the models over to RETRAN-02 based models. Qualification of these new models was established by performing comparisons with the old RETRAN-01 based model. This was submitted to NRC for information and is provided for reference in **Appendix 5.**

In 1993, Dominion documented a major upgrade to the North Anna RETRAN model, including discrete modeling of all three reactor coolant loops. This was submitted to the NRC for information in Reference 1.0-4. This submittal provided qualification comparsions between the old and new models for several UFSAR transients as well as new comparisons to plant data for the 1987 steam generator tube rupture event.

The NRC has reviewed all of the essential elements the Dominion RETRAN models, including the 1993 upgrade as well as subsequent model refinements as part of the approval process for Revision 2 of Dominion's Reload Nuclear Design Methodology Topical Report VEP-FRD-42A (Reference 1.0-5). Relevant correspondence documenting this review process is shown in Table 1.1. The NRC documented their review and approval of these model upgrades in Reference 1.0-6, which is included for reference as **Appendix 6.** This is discussed further in Section 2.

Section 3 provides a brief review of the evolution of the RETRAN code.

Section 4 presents an overview of Dominion's RETRAN-02 three-loop model for Surry and North Anna. Separate models are maintained for the two plants to reflect specific design differences, but the modeling approach, philosophy and plant noding are the same. Modeling nomenclature, configuration, noding and RETRAN code option selections are discussed.

Section 5 discusses specific component models. Model features described (with corresponding section numbers) are:

- Generic problem definitions (5.1)
- Reactor protection system (5.2)
- Reactor vessel and core (5.3)
- Primary piping (5.4)
- Reactor coolant pumps (5.5)
- Pressurizer (5.6)
  - Pressurizer Sprays
  - Pressurizer PORVs
  - Pressurizer Safety Valves
- Steam Generators (5.7)
- Main Steam System (5.8)
  - Main Steam Safety Valves
  - Atmospheric Steam Relief Valves (PORVs)
  - Steam line non-return valves
  - MS Isolation Valves
  - Condenser Steam Dump System
- Main feedwater system (5.9)
- Auxiliary feedwater system (5.10)
- Turbine EHC system and automatic runback (5.11)
- Safety Injection System (5.12)
- Reactor Kinetics (5.13)

Details of the qualification bases of the various component models are discussed in Section 5. Overall model qualification is addressed in Section 6.

REFERENCES FOR SECTION 1.0

1.0-1 Letter from C. O. Thomas (NRC) to W. L. Stewart (Virginia Power), Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, "Virginia Power Reactor System Transient Analyses Using the RETRAN Computer Code," April 11, 1985.

1.0-2 EPRI NP-1850-CCM-A, RETRAN-02, A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Rev. 6, December 1996.

1.0-3 Letter from G. L. Swindlehurst (RETRAN Maintenance Group) to T. E. Collins (NRC/RSB), RETRAN-02 MOD005.2 Code Version, Notification of Code Release, November 24, 1997.

1.0-4 Letter from Virginia Electric and Power Company to USNRC, "Supplemental Information on the RETRAN NSSS Model," Serial No. 93-505, August 10, 1993.

1.0-5 VEP-FRD-42, Revision 2-A, "Reload Nuclear Design Methodology", August 2003.

1.0-6 Letter from Scott Moore, USNRC, to D. A. Christian, Dominion, Virginia Electric and Power Company, Acceptance of Topical Report VEP-FRD-42, Revision 2, "Reload Nuclear Design Methodology", North Anna and Surry Power Stations, Units 1 and 2, Serial No. 03-381, June 11, 2003.

2. DOMINION'S RETRAN MODEL HISTORY

Since the initial issuance of this report, significant extension and refinement of Dominion's RETRAN models have occurred. These changes were implemented under the provisions of 10 CFR 50.59 and USNRC Generic Letter 83-11 Supplement 1 (Reference 2-1). An overview of these model upgrades is provided in Section 2.4 of this report.

The NRC reviewed these model upgrades as part of the review of a separate topical report, VEP-FRD-42, Revision 2, Reload Nuclear Design Methodology. Details of that review and the subsequent evaluation are discussed below. Table 2.1 summarizes the relevant NRC licensing correspondence regarding RETRAN models, and code use and application at Dominion.

REFERENCE FOR SECTION 2

2-1 USNRC, Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses," June 24, 1999.

2.1 Review and Approval of Code and Model Updates through June 2003

The model updates and qualifications summarized in Section 2.4, **Overview of RETRAN Model Upgrades** and described in Section 5, **System Component Model Descriptions**, have been reviewed by the USNRC as part of the approval of Topical Report VEP-FRD-42, Rev. 2.0-A (Reference 2.1-1). Review of RETRAN as part of the Reload Nuclear Design Methodology stems from the use of RETRAN in generating certain parameter limits in the Core Operating Limits Report (COLR) in accordance with the provisions of USNRC Generic Letter 88-16 (Reference 2.1-2).

During the review process for VEP-FRD-42A (Ref. 2.1-1), Dominion submitted information related to the qualification and use of the model upgrades described in Sections 2.4 and 5. The NRC's review and approval of these changes is summarized in the Reference 2.1-1 SER and is provided below for completeness. References excerpted from the SER have been renumbered and are also provided below.

> *In a letter dated August 10, 1993 (Reference 2.1-3), VEPCO informed the NRC staff of various modifications and updates to its RETRAN model, and that these changes were to be implemented under the provisions of 10 CFR 50.59. This letter described several changes to the VEPCO RETRAN models, including expansion to a three-loop Reactor Coolant System and multi-node steam generator secondary side. Although this letter was submitted for the North Anna Power Station, VEPCO provided additional information on December 2, 2002 (Reference 2.1-4), and March 21, 2003 (Reference 2.1-5), justifying the applicability of the RETRAN model to both the Surry and North Anna Power Stations. By letter dated December 2, 2002, VEPCO provided additional information regarding its capability to make modifications to the RETRAN model. The NRC staff's SE dated April 11, 1985 (Reference 2.1-6) for the VEPCO RETRAN model recognized that model maintenance activities would be performed under the utility's 10 CFR 50 Appendix B Quality Assurance program and stated, "The staff requires that all future modifications of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures." The NRC staff has determined that VEPCO has followed the requirements specified in the NRC staff's SE in updating the RETRAN models. Additionally, the NRC staff has also determined the qualification, documentation, and implementation of the new models was performed in a manner that meets the programmatic elements of NRC GL 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," dated June 24, 1999 (Reference 2.1-7).*

VEPCO is currently using RETRAN-02/MOD005.2. As such the NRC staff requested additional information describing how each of the limitations, restrictions, and items identified as requiring additional user justification in the generic NRC staff's Ses, through the currently used version, are satisfied. This includes RETRAN02/M0D002 (Reference 2.1-8), RETRAN02/MOD003 and MOD004 (Reference 2.1-9) and RETRAN02/MOD005 (Reference 2.1-10). By letter dated March 21, 2003 (Reference 2.1-5), VEPCO provided detailed information describing how each limitation (approximately 48 total) is treated in the North Anna and Surry RETRAN models. The NRC staff has reviewed VEPCO's responses and finds the limitations, restrictions and items identified as requiring additional user justification are satisfactorily addressed.

Based on the above discussions, the NRC staff finds that the VEPCO RETRAN models and the use of RETRAN continue to be acceptable for use in licensing calculations for the North Anna and Surry Power Stations.

Table 2.1 provides a summary of relevant licensing correspondence with the USNRC regarding Dominion's RETRAN models.

REFERENCES FOR SECTION 2.1

2.1-1 Letter from Scott Moore, USNRC, to D. A. Christian, Dominion, "Virginia Electric and Power Company, Acceptance of Topical Report VEP-FRD-42, Revision 2, 'Reload Nuclear Design Methodology', North Anna and Surry Power Stations, Units 1 and 2," Serial No. 03-381, June 11, 2003.

2.1-2 USNRC, Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 13, 1988.

2.1-3 Letter from Virginia Electric and Power Company to USNRC, "Supplemental Information on the RETRAN NSSS Model," Serial No. 93-505, August 10, 1993.

2.1-4 Letter from E. S. Grecheck (Dominion) to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," December 2, 2002 (Serial No. 02-662).

2.1-5 Letter from L. N. Hartz (Dominion) to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Request for Additional Information on VEP-FRD-42, Reload Nuclear Design Methodology," March 21, 2003 (Serial No. 03-183).

2.1-6 Letter from C. O. Thomas (NRC) to W. L. Stewart (Virginia Power), "Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, 'Virginia Power Reactor System Transient Analyses Using the RETRAN Computer Code," April 11, 1985.

2.1-7 USNRC, Generic Letter 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," June 24, 1999.

2.1-8 Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

REFERENCES FOR SECTION 2.1 (CONT)

2.1-9 Letter from A. C. Thadani (USNRC) to R. Furia (GPU), "Acceptance for Referencing Topical Report EPRI-NP-1850-CCM-A Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004," October 19, 1988.

2.1-10 Letter from A. C. Thadani (NRC) to W. J. Boatwright (RETRAN02 Maintenance Group), Acceptance for Use of RETRAN02 MOD005.0, November 1, 1991.

Table 2.1
Vepco / Dominion RETRAN Model Correspondence

Date	Document Title	Contents
4/14/81	Letter from W. N. Thomas to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses," SN 215.	Requests review of Topical Report VEP-FRD-41.
6/8/82	USNRC Inspection Reports 50-338/82-16, 50-339/82-16, 50-280/82-16 and 50-281/82-16	Reports no findings for inspection of Vepco RETRAN V&V activities.
10/6/83	Letter from W. L. Stewart to H. R. Denton (USNRC), Amendment to Operating Licenses DPR-32 and DPR-37, Surry Power Station Units 1 and 2, Proposed Technical Specifications Change, SN 521A.	Resubmits info copy of VEP-FRD-41 to support a licensing application of the code & model (Surry BIT Removal)
2/27/84	Letter from W. L. Stewart to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses, Supplemental Information", SN 060.	Responds to a NRC Request for Additional Information (RAI)- provides more detailed noding description of 1 and 2-loop models described in the topical report. **[See Appendix 2]**
7/12/84	Letter from W. L. Stewart to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses", SN 376.	Responds to NRC RAI- provides description of system component models, input options, and provides the results of sensitivity studies for several transients. **[See Appendix 3]**
8/24/84	Letter from W. L. Stewart to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses", SN 376A.	Responds to NRC RAI- provides description of control system models, and proprietary and nonproprietary versions of comparisons between Vepco RETRAN model and LOFTRAN predictions. **[See Appendix 4]**
4/11/85	Letter from C. O. Thomas (USNRC) to W. L. Stewart, " Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, 'Vepco Reactor System Transient Analysis Using RETRAN Computer Code.'"	Provides VEP-FRD-41 Rev. 0 SER. **[Incorporated in VEP-FRD-41A Rev. 0- see Appendix 1]**

Table 2.1 (CONTINUED)
Vepco / Dominion RETRAN Model Correspondence

Date	Document Title	Contents
7/3/85	Letter from W. L. Stewart to H. L. Thompson (USNRC), "Virginia Power, Issuance of the RETRAN Code Report", SN 85-277.	Issues VEP-FRD-41A. **[See Appendix 1]**
8/21/85	Letter from W. L. Stewart to H. R. Denton (USNRC, "Virginia Power, Surry and North Anna Power Stations, Reactor System Transient Analyses," SN 85-570.	Confirms that an input deck listing of the Surry 1-Loop model was provided to Standardization and Special Projects Branch in fulfillment of a condition in the VEP-FRD-41 SER.
11/19/85	Letter from W. L. Stewart to H. R. Denton (NRC), "Virginia Electric and Power Company, Surry and North Anna Power Stations, Reactor System Transient Analyses, (Serial No. 85-753).	Submits (for information) comparisons between RETRAN-01 and RETRAN-02 and documents Vepco's intention to transition to RETRAN-02. **[See Appendix 5]**
8/10/93	Letter from M. L. Bowling to USNRC, North Anna Power Station Units 1 and 2, Supplemental Information on the RETRAN NSSS Model. (Serial No. 93-505)	Forwards description and qualification of the North Anna 3-Loop model for information, and affirms that model upgrades have been performed under the provisions of 10 CFR 50.59.
3/27/02	NRC RAI, "Dominion's Reload Nuclear Design Methodology Topical Report, VEP-FRD-42 Revision 2, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Docket Nos. 50-338/339 and 50-280/281 Dated October 8, 2001, " March 27, 2002.	NRC RAI forwards observation that the August 1993 submittal, previous entry, was not reviewed and requests clarification on the acceptability of the upgraded models to support COLR limits under the provisions of Generic Letter 88-16. Also asks for information relating to the applicability of the RETRAN models to Framatome Fuel.

Table 2.1 (CONTINUED)
Vepco / Dominion RETRAN Model Correspondence

Date	Document Title	Contents
5/13/02	Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," May 13, 2002 (Serial No. 02-280).	Responds to NRC RAI of 3/27/02. Describes the development of RETRAN model overlays for Framatome fuel. Presents elements of Dominion's model maintenance philosophy.
10/25/02	Letter from S. R. Monarque and G. E. Edison, USNRC, to D. A. Christian, "North Anna Power Station Units 1 and 2 and Surry Power Station Units 1 and 2, Request for Additional Information on Virginia Electric and Power Company's Reload Nuclear Design Methodology Topical Report VEP-FRD-42."	States NRC's intention to formally review the 8/10/93 submittal as part of the VEP-FRD-42 Rev. 2 review and makes the observation that the submittal was packaged as a North Anna only model upgrade and asks for Surry information.
12/2/02	Letter from E. S. Grecheck to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," SN 02-662.	Responds to NRC 10/25/02: • Affirms that information in the 8/10/93 submittal is equally applicable to Surry and North Anna. • Affirms that model upgrades are done in accordance with provisions of Appendix B, 10 CFR 50.59 and the VEP-FRD-41 SER. • Describes major maintenance upgrades to the models SINCE the 8/10/93 submittal. • Provides a description of the topical report maintenance (i.e. Topical Mods and Revisions) program and relates it to 10 CFR 50.59, NEI 96-07, Generic Letter 83-11 Supplement 1.

Table 2.1 (CONTINUED)
Vepco / Dominion RETRAN Model Correspondence

Date	Document Title	Contents
2/26/2003	Letter from S. R. Monarque, USNRC, to D. A. Christian, "North Anna Power Station Units 1 and 2 and Surry Power Station Units 1 and 2, Request for Additional Information on Topical Report VEP-FRD-42, Revision 2, 'Reload Design Methodology."	RAI requests the following: • Information on how the Dominion models and applications meet the restrictions and limitations of the generic RETRAN code SERs. • RETRAN input decks • A technical description and qualification of the Doppler reactivity models. • A discussion of the philosophy for using 1-, 2- or 3-loop models and identification of which models are used for each UFSAR Chapter 14/15 transient.
3/21/03	Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Request for Additional Information on Topical Report VEP-FRD-42, Reload Nuclear Design Methodology," SN 03-183.	Responds to 2/26/03 RAI on VEP-FRD-42 Rev. 2. Provides: • Information on the restrictions and qualifications of the generic RETRAN SERs • A statement that affirmed that the original model decks were provided to the NRC as discussed in SN 85-570. • A technical description of the Doppler reactivity feedback model. • A discussion of the philosophy behind choosing 1-, 2- or 3-loop models for safety analysis and tables identifying which models were applied in currently applicable analyses of record.

Table 2.1 (CONTINUED)
Vepco / Dominion RETRAN Model Correspondence

Date	Document Title	Contents
6/11/03	Letter from Scott Moore, USNRC, to D. A. Christian, "Virginia Electric and Power Company-Acceptance of Topical Report VEP-FRD-42, Revision 2, 'Reload Nuclear Design Methodology', North Anna and Surry Power Stations, Units 1 and 2."	Approves VEP-FRD-42 Rev. 2. In the context of this approval: • Staff concluded that Dominion has qualified, implemented and maintained the new models in accordance with the provisions of Generic Letter 83-11, Supplement 1. • Staff concluded that the limitations, qualifications and items requiring additional user justification in the generic RETRAN SER's are satisfactorily addressed. • Vepco RETRAN models continue to be acceptable for use in licensing calculations for Surry and North Anna. • Staff provided concurrence with Dominion's "Analytical Model and Method Approval Process" for implementing certain methodology changes without prior NRC review and approval. [See Appendix 6]

2.2 NRC Review and Approval of Dominion Analytical Model Maintenance Process

Dominion's analytical model maintenance process was described in Reference 2.2-1 and evaluated by the USNRC Staff in Reference 2.2-2. Key elements of the process, as set forth in Reference 2.2-1 are summarized here.

2.2.1 Published NRC Guidance

The determination of the requirement to submit methodology changes to NRC for approval prior to application is based on published NRC guidance, i.e.:

- Generic Letter 88-16, "Removal Of Cycle-Specific Parameter Limits From Technical Specifications"
- 10 CFR 50.59, and in particular 10 CFR 50.59c(2)(viii): "*(2) A licensee shall obtain a license amendment pursuant to Sec. 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would (viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.*"
- NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Evaluations"
- Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments" (endorses NEI 96-07 Rev. 1).
- Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses"

2.2.2 Key Document Discussions

Relevant sections of these documents upon which we base our determination process are as follows:

1. Generic Letter 88-16 establishes the concept of reload cycle dependent operating limits in the Technical Specifications.

   "Generally, the methodology for determining cycle-specific parameter limits is documented in an NRC-approved Topical Report or in a plant-specific submittal. As a consequence, the NRC review of proposed changes to TS for these limits is primarily limited to confirmation that the updated limits are calculated using an NRC-approved methodology and consistent with all applicable limits of the safety analysis. These changes also allow the NRC staff to trend the values of these limits relative to past experience. This alternative allows continued trending of these limits without the necessity of prior NRC review and approval."

   Since changes to the cycle specific parameter limits must be based on "NRC-approved" methods, it is important to establish a clearly defined process and criteria for making upgrades to methodologies without NRC review while maintaining the NRC-approved status.

2. NEI 96-07, Rev. 1, as endorsed by Reg. Guide 1.187, provides guidance for evaluating changes to methods under the provisions of 10CFR50.59. For example, Paragraph 4.3.8.1, states:

3. Guidance for Changing One or More Elements of a Method of Evaluation

"The definition of "departure ..." provides licensees with the flexibility to make changes under 10 CFR 50.59 to methods of evaluation whose results are "conservative" or that are not important with respect to the demonstrations of performance that the analyses provide. Changes to elements of analysis methods that yield conservative results, or results that are essentially the same, would not be departures from approved methods."

4. USNRC Generic Letter 83-11 Supplement 1 provides a method for utility qualification of analysis methodologies, including those used to establish core operating limits, without formal NRC review and approval:

"The U.S. Nuclear Regulatory Commission (NRC) is issuing this supplement to Generic Letter (GL) 83-11 to notify licensees and applicants of modifications to the Office of Nuclear Reactor Regulation (NRR) practice regarding licensee qualification for performing their own safety analyses. This includes the analytical areas of reload physics design, core thermal-hydraulic analysis, fuel mechanical analysis, transient analysis (non-LOCA), dose analysis, setpoint analysis, containment response analysis, criticality analysis, statistical analysis, and Core Operating Limit Report (COLR) parameter generation. It is expected that recipients will review the information for applicability to their facilities. However, suggestions contained in this supplement to the generic letter are not NRC requirements; therefore, no specific action or written response is required."

"To help shorten the lengthy review and approval process, the NRC has adopted a generic set of guidelines which, if met, would eliminate the need to submit detailed topical reports for NRC review before a licensee could use approved codes and methods. These guidelines are presented in the Attachment to this Generic Letter. Using this approach, which is consistent with the regulatory basis provided by Criteria II and III of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50), the licensee would institute a program (such as training, procedures, and benchmarking) that follows the guidelines, and would notify NRC by letter that it has done this and that the documentation is available for NRC audit."

2.2.3 Conclusion and Application

Based on the excerpts above:

- Dominion concludes that utilities can change, under the provisions of 10 CFR 50.59©(2)(viii), NRC approved codes and methodologies used to establish core operating limits, via the processes outlined in NEI 96-07, Rev. 1, without additional NRC review and approval of these changes.

- Dominion concludes that utilities can implement or substitute, under 10 CFR 50.59©(2)(viii), NRC approved codes and methodologies for use in establishing core operating limits via the processes outlined in Generic Letter 83-11 Supplement 1 (Reference 2.2-3), without additional NRC review and approval of these methods.

- Dominion concludes that, in updating the list of approved methodologies for establishing core operating limits in the Technical Specifications, utility affirmation that the changes to the methodologies have been done as described by either of the above is adequate to retain the "approved" status for these methods.

2.2.4 Dominion's Generic Letter 83-11 Program

Dominion has established a formal program for modification of methods and the associated documentation under the provisions of Generic Letter 83-11 Supplement 1 (Reference 2.2-1). This program ensures that the generic guidelines of GL 83-11 Supplement 1, i.e.

1. The analytical method is "generically approved" or approved on a plant's docket.
2. In-house application procedures are in place.
3. An in-house program for training/qualification of analytical method users is implemented.
4. The analytical method has been qualified/benchmarked & documented.
5. The analytical method implementation is performed under a 10 CFR 50 Appendix B Quality Assurance program.

Are followed when the provisions of 10 CFR 50.59©(2)(viii) and GL 83-11 are invoked in maintaining codes and methodologies.

In Reference 2.2-2, the NRC Staff evaluated Dominion's Analytical Model and Method Approval Process as described in Section 2.3 of VEP-FRD-42 Rev. 2 (Reference 2.2-4). The Staff found the process outlined there and described above to be acceptable.

REFERENCES FOR SECTION 2.2

2.2-1 Letter from E. S. Grecheck to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," December 2, 2002 (Serial No. 02-662).

2.2-2 Letter from Scott Moore (USNRC), to D. A. Christian, "Virginia Electric and Power Company, Acceptance of Topical Report VEP-FRD-42, Revision 2, 'Reload Nuclear Design Methodology', North Anna and Surry Power Stations, Units 1 and 2," June 11, 2003.

2.2-3 USNRC, Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses," June 24, 1999.

2.2-4 VEP-FRD-42 Rev. 2-A, "Reload Nuclear Design Methodology".

2.3 Conformance of Dominion's RETRAN Models to Restrictions, Limitations and Conditions of Use in the Generic RETRAN SER's

As discussed in Section 2.1 (see the excerpt from the VEP-FRD-42 SER), Dominion documented the conformance of our RETRAN model to the restrictions, limitations and conditions of use set forth in the generic RETRAN code SER's in Reference 2.3-1. These discussions are presented for reference in **Appendix 7**. Additionally, to facilitate use of this report, portions of these discussions that are relevant to specific component models are also reproduced in Section 5, **System Component Model Descriptions**.

REFERENCE FOR SECTION 2.3

2.3-1 Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Request for Additional Information on VEP-FRD-42, Reload Nuclear Design Methodology," March 21, 2003 (Serial No. 03-183)

2.4 Overview of RETRAN Model Upgrades

Since the original issuance of this report, Dominion's RETRAN models have undergone the following changes:

1. The original models included single and two-loop models. Significant advances in computing power over the last decade have eliminated the need for the approximation of "collapsing" the reactor coolant loop representations. The current models explicitly represent all three reactor coolant loops with discrete noding.

2. The base models use a single node secondary side for the steam generator, consistent with the 1-loop model in VEP-FRD-41A. There is a multi-node SG secondary overlay available for transients where understanding of the detailed steam generator level response is needed.

3. The current models use the 1979 ANS Decay Heat model option.

4. More detailed main steam safety valve (MSSV) modeling was added to ensure that the concerns raised in NRC Information Notice 97-09, "Inadequate Main Steam Safety Valve (MSSV) Setpoints and Performance Issues Associated with Long MSSV Inlet Piping" (Reference 2.4-1) are adequately addressed.

5. The reactor protection and engineered safety features actuation system setpoints are maintained consistent with current Technical Specifications setpoints and I&C calculations of instrument channel uncertainties.

6. Minor reactor vessel noding model changes were made. The core nodes now include only the active fuel region. Volumes between the active fuel and core plates are assigned to the inlet and outlet plena.

7. Hydraulic characteristics in the core regions have been adjusted to reflect current fuel assembly designs.

8. More detailed, mechanistic models for the pressurizer and steam generator level instrumentation were added.

9. The local conditions heat transfer model has been qualified for use with the single node SG secondary model option for loss of heat sink events.

10. A more detailed feedwater control system model was added (not typically used in licensing analyses).

11. An electrohydraulic turbine control (EHC) and runback model was added (not typically used in licensing analyses).

12. A detailed rod control system model was added.

13. A separate main steam line break (MSLB) add-on module was developed that retains the basic modeling features of the two-loop model presented in VEP-FRD-41A, i.e.,

    - A split core with two azimuthal zones
    - Imperfect temperature mixing between loops
    - Asymmetric reactivity weighting to model a stuck rod.

14. The MSLB module uses a more mechanistic (but still conservative) heat transfer model than the constant UA model of VEP-FRD-41A. The overlay model applies a separate heat transfer coefficient to the secondary side of each steam generator conductor based on the MAXIMUM of:

    - Rohsenow pool boiling
    - Schrock-Grossman forced convection vaporization
    - Thom nucleate boiling
    - Chen combined nucleate boiling and forced convection vaporization
    - Single phase conduction to steam (Dittus-Boelter)

15. A detailed set of RETRAN card overlays was developed to model Framatome ANP Fuel (FANP) cores in North Anna Units 1 and 2. The development of the FANP overlays was described in Reference 2.4-2 and is summarized in Section 5.3.

Table 2.1 provides a summary of relevant licensing correspondence with the USNRC regarding Dominion's RETRAN models.

A detailed model description is provided in Section 5.0, **System Component Models**.

REFERENCES FOR SECTION 2.4

2.4-1 USNRC, Information Notice 97-09, "Inadequate Main Steam Safety Valve (MSSV) Setpoints and Performance Issues Associated with Long MSSV Inlet Piping," March 12, 1997.

2.4-2 Letter from L. N. Hartz (Vepco) to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," May 13, 2002 (Serial No. 02-280).

3. EVOLUTION OF THE RETRAN CODE

3.1 Dominion's RETRAN Code History

The RETRAN computer code is the result of an extensive code development effort by EPRI beginning in 1975. The resulting code is a variable node code with many diverse modeling features that can be used to represent light water reactor systems.

3.1.1 RETRAN-01

The RETRAN-01 code was released in December 1978 (Reference 3.1-1). RETRAN-01 featured:

- A one-dimensional, homogeneous equilibrium mixture (HEM) thermal-hydraulic representation of the reactor coolant system (RCS)
- A point neutron kinetics model for the reactor core
- Auxiliary component models, including a nonequilibrium pressurizer model and a temperature transport delay model for pipe like regions of the RCS
- A versatile control system model that allowed construction of customized control and protection system representations using "control blocks", or numerical representations of various analogue modules such as summers, amplifiers and filters
- A steady state initialization technique

Dominion (Virginia Power) participated actively in the EPRI System Analysis Working Group, a group of utilities that developed plant models as well as separate effects test models to exercise various features of the code and provide feedback to the RETRAN-01 code developers. Many of the benchmark comparisons to vendor calculations and plant transient data presented in the original version of this topical report were initially performed with RETRAN-01.

Generic NRC approval of RETRAN-01 was provided in Reference 3.1-2.

3.1.2 RETRAN-02

At the time of the RETRAN-01 code release, a number of theoretical limitations to the code were known and documented. The RETRAN-02 code development effort was initiated to remove some of these limitations and to extend the capabilities of the code, particularly in the areas of modeling Boiling Water Reactor (BWR) transients, small break loss of coolant accidents, anticipated transients without scram (ATWS) and certain balance of plant features, such as turbines.

To address these needs, a number of the RETRAN-01 models were revised and/or extended. Revisions included:

- An improved solution technique for the nonequilibrium pressurizer model
- A modified critical flow solution
- An equation of state for water valid over the range 0.1 psia to 6000 psia.
- A revised momentum mixing calculation (primary for modeling BWR jet pumps

In addition, RETRAN-02 (Reference 3.1-3) includes the following additional models:

- Dynamic and algebraic slip models for two-phase flow
- A one dimensional space-time neutron kinetics model
- A set of two-phase natural convection heat transfer correlations
- An iterative solution scheme for the fluid field equations
- A turbine model and a condensing heat transfer model for balance of plant analyses
- A local conditions heat transfer model (important for ATWS and other severe loss of inventory conditions).
- A vector momentum representation of the fluid
- An auxiliary model (profile fit) to compute void fraction for void reactivity feedback (primarily for BWRs)
- Thermophysical properties and a forced convection heat transfer correlation for supercritical water
- A steam separator efficiency model (primarily for BWRs)

As with RETRAN-01, VEPCO was an active participant in the RETRAN-02 code development and testing. A number of our studies were published by EPRI and elsewhere (References 3.1-4 through 3.1-12)

The NRC Staff's approval of RETRAN-02 was subject to a number of conditions and limitations described in the safety evaluations (SE) for the various RETRAN-02 versions and in the accompanying Technical Evaluation Reports (TERs) prepared by the NRC staff's contractors (References 3.1-13 through 3.1-15). These conditions and limitations are addressed in detail in Appendix 7 of this report. Additionally, for user reference, conditions and limitations that are pertinent to particular system component models are also addressed in the System Component and Model Descriptions in Section 5.

Dominion transitioned from RETRAN-01 to RETRAN-02 by performing comparisons for representative calculations for several transients and showing that the results were either essentially the same or could be understood in the context of the RETRAN-02 code improvements. These studies were provided to the USNRC in Reference 3.1-16.

SECTION 3.1 REFERENCES

3.1-1 Moore, K.V., et. Al., "RETRAN - A program for One Dimensional Transient Thermal-Hydraulic Analyses of Complex Fluid Flow Systems," EPRI-CCM-5, 1978.

3.1-2 Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

3.1-3 Computer Simulation and Analysis, Inc., "RETRAN-02 - A program for One Dimensional Transient Thermal-Hydraulic Analyses of Complex Fluid Flow Systems," NP-1850-CCM-A, December 1995 (Rev. 6).

3.1-4 EPRI Report NP-2175, "RETRAN-01 - A program for One Dimensional Transient Thermal-Hydraulic Analyses of Complex Fluid Flow Systems," Revision of EPRI-CCM-5, Volume 4, "Applications", presents Dominion studies of Uncontrolled Rod Withdrawal at Power, Loss of Flow Accident, Loss of External Electrical Load, Isothermal Pump Coastdown (Comparison to Plant Data), and Main Steam Line Break (model development studies).

3.1-5 EPRI Report NP-2494-SR, Conference Proceedings, Second Annual RETRAN Conference, July, 1982, "Auxiliary Feedwater System Performance Calculations Using RETRAN" (Loss of Normal Feedwater Studies); "Investigation of the 1-D Reactor Kinetics Model in RETRAN-02 MOD002", July 1982.

3.1-6 Miller, J. G. and Erb, J. O., "Vepco Evaluation of the Control Rod Ejection Transient", VEP-NFE-2-A, Vepco, Richmond, VA, December, 1984.

3.1-7 Erb, J. O. and Miller, J. G., "RETRAN Modeling of The PWR Control Rod Ejection Transient," EPRI, Fifth International RETRAN Meeting, November, 1987.

3.1-8 R. W. Cross, F. W. Sliz and N. A. Smith, "Non-LOCA Transient Safety Analysis Using the RETRAN Computer Code", Transactions of the American Nuclear Society (ANS), 32, June 1979.

3.1-9 N. A. Smith, "Best Estimate Operational Transient Analysis Using the RETRAN Computer Code", Trans. Of the ANS, 32, Supp. 1, August 1979.

3.1-10 N. A. Smith, "Analysis of a PWR Cooldown Using RETRAN", Trans. Of the ANS, 39, November 1981.

3.1-11 . R. W. Cross and N. A. Smith, "Development of an In-house Safety Analysis Capability for Plant Operational Support", ANS International Meeting on Thermal Reactor Safety, August/September 1982.

3.1-12 J. O. Erb and N. A. Smith, "Thermal Hydraulic Analyses of Steamline Break Accidents with Reduced Boric Acid Concentration In the Safety Injection System", Proc. Of Third International Topical Meeting on Reactor Thermal Hydraulics, ANS, October 1985.

SECTION 3.1 REFERENCES (CONT)

3.1-13 Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

3.1-14 Letter from A. C. Thadani (USNRC) to R. Furia (GPU), "Acceptance for Referencing Topical Report EPRI-NP-1850-CCM-A Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004," October 19, 1988.

3.1-15 Letter from A. C. Thadani (NRC) to W. J. Boatwright (RETRAN02 Maintenance Group), Acceptance for Use of RETRAN02 MOD005.0, November 1, 1991.

3.1-16 Letter from W. L. Stewart, (VEPCO) to H. R. Denton (NRC), "Virginia Electric and Power Company, Surry and North Anna Power Stations, Reactor System Transient Analyses," November 19, 1985, (Serial No. 85-753).

3.2 RETRAN Code Approvals

Dominion is currently using RETRAN02 MOD005.2. NRC review and approval of RETRAN02 through Mod 5.0 was discussed in the USNRC SER for VEP-FRD-42 excerpted and discussed in Section 2.1, **Review and Approval of Code and Model Updates through June 2003**. Discussions between the utilities and the NRC led to the conclusion that MODs 5.1 and 5.2, which were essentially maintenance upgrades, did not require additional NRC review for utility implementation (References 3.2-1 and 3.2-2).

REFERENCES FOR SECTION 3.2

3.2-1. Letter from M. J. Virgilio (NRC) to C. R. Lehmann (RETRAN Maintenance Group), Acceptance for Referencing of the RETRAN-02 MOD005.1 Code, April 12, 1994.

3.2-2. Letter from G. L. Swindlehurst (RETRAN Maintenance Group) to T. E. Collins (NRC/RSB), RETRAN-02 MOD005.2 Code Version, Notification of Code Release, November 24, 1997.

4 RETRAN-02 THREE LOOP MODEL, OVERVIEW

4.1 NOMENCLATURE

Throughout the document, reference to control volume or junction numbers may include an "X" where X denotes the loop number of the region, i.e., 1, 2 or 3. (All loop related control volumes and junctions begin with one of these digits.) For example, volume X15 refers to the cold leg. For each of the loops in the 3-loop geometry, this refers to volumes 115, 215 and/or 315.

A description of abbreviations used throughout the report is provided in **Table 4.1.**

4.2 MODEL CONFIGURATION AND ORGANIZATION

The model is available in two geometric configurations for each site (Surry and North Anna):

1. 3-loop, multi-node SG
2. 3-loop, single node SG

Unit-specific models for each site are not maintained, i.e. the model is applicable to either unit.

The technical basis and detailed input development for each model is maintained in a configuration controlled document. Cards for a specific system are organized within the decks in the same order as documented in the model documentation and are preceded by an identification number equivalent to the controlled document section number for that system. For example, I.D.s beginning with the number "3" (e.g., 3.2.5.1) describe either the 3-loop, multi-node SG model configuration or are generic to all configurations. Those beginning with the number "5" are specific to the 3-loop, single node SG configuration. The numbering system used to identify the various component and system models is shown in **Table 4.2.**

The primary side noding of both the single node and multi-node steam generator configurations are identical; i.e., both have ten steam generator tube volumes and ten heat conductors per steam generator. The secondary side of the single node steam generator configuration has a single RETRAN volume per steam generator.

It should be noted that the multi-node secondary model is used for sensitivity studies and benchmarks as an aid to understanding. Its use in licensing calculations requires additional qualfication of the RETRAN drift flux and dynamic slip correlations for PWR applications –see Section 5.7.2.

Figures 4.1 through 4.3 represent nodalization diagrams of the three-loop, multi-node steam generator secondary configuration. Control volume, junction and heat conductor region numbers starting with an X refer to three-loop geometry regions where X can have the value of 1, 2 or 3. Control volume region numbers are underlined whereas junction and heat conductor region numbers are not. Junctions are denoted by arrows pointed in the direction associated with positive flow. The

region number for an unlabeled junction is equal to the region number of the downstream control volume for the junction.

Unlike the earlier one-loop geometry, the reactor vessel region above the core is more realistically divided into two volumes, an upper plenum region and an upper head region.

The earlier NSSS RETRAN model included a separate steam generator inlet volume. In the current model, the hot leg volume runs from the reactor vessel outlet nozzle to the top of the steam generator hot side tubesheet. The RCP (reactor coolant pump) suction leg runs from the top of the steam generator cold side tubesheet to the RCP intake. The steam generator portions of the hot leg and RCP suction volumes reflect the dimensions of the Surry and North Anna replacement steam generators.

Figures 4.1 and 4.3 in conjuction with a single node representation of the secondary side of each steam generator (rather than the multinode configuration of Figure 4.2) represent the most frequently used noding configuration in current applications. Dominion's analytical experience has shown this noding to be quite robust for a wide range of transients. However, the analyst may opt to provide more noding detail (i.e. additional volumes and junctions) as dictated by specific analysis requirements. The bases for deviations from the reference configuration are documented in individual application calculations. An example of this is the use of additional core and reactor vessel plenum noding in steamline break calculations, as discussed in Sections 5.3 and 5.13.

Control systems modeled include the Reactor Protection (RPS) and Engineered Safety Features (ESF) systems, pressurizer level instrumentation, steam generator level control, main feedwater and auxiliary feedwater systems, the turbine EHC system and automatic turbine runback, and high-head safety injection.

The following reactivity components are modeled:

a. Doppler feedback
b. Moderator feedback
c. Control rod withdrawal
d. Automatic rod control
e. Reactor trip

In addition the model is designed to allow for changes in soluble boron reactivity to be incorporated when required for a particular transient analysis.

The Doppler reactivity feedback is calculated by a correlation of Doppler reactivity as a function of core average fuel temperature and core burnup. For a reanalysis of a FSAR transient, the Doppler feedback algorithm is capable of being adjusted to a target Doppler temperature coefficient or Doppler power defect by the application of a suitable weighting factor.

Moderator reactivity feedback can be computed either using a moderator temperature coefficient, or a reactivity function based on moderator density for a transient involving significant core voiding.

The decay heat is modeled with sufficient conservatism to ensure bounding the decay heat predicted by the 1979 ANS Decay Heat Standard with a two standard deviation uncertainty applied to the latter.

4.3 Noding and Options

Tables 4.3 and 4.4 provide summaries of the control volume and momentum junction nodalization. The tables reflect a multi-node steam generator secondary geometry. Abbreviations are "SG" for steam generator, "MS" for main steam, and "PORV" for pressure operated relief valve. All control volumes are standard HEM (homogeneous equilibrium mixture) volumes except volume 17, which is the nonequilibrium pressurizer. All junctions use the Baroczy two-phase multiplier with Fanning wall friction and have specified single-stream compressible flow except junction 21 (surge line to hot leg). Except where mandated by the differences in nodalization, the control volume and momentum junction options specified are identical to those of the earlier two-loop and single-loop models.

Table 4.1 – LIST OF ABBREVIATIONS

AFW	Auxiliary feedwater
ATWS	Anticipated Transient Without SCRAM
βeff	Effective delayed neutron fraction
CB	Control block
CSA	Channel Statistical Allowance (i.e. instrument uncertainty)
DNB	Departure from nucleate boiling
DTC	Doppler temperature coefficient
EHC	Electrohydraulic turbine control
ESF	Engineered safety features
FW	Feedwater
HEM	Homogeneous equilibrium
HFP	Hot full power
HTC	Heat transfer coefficient
HZP	Hot zero power
I.C.	Initial condition(s)
LOCA	Loss-of-coolant accident
lp	Prompt neutron lifetime
MSLB	Main steamline break
NAPS	North Anna Power Station
NR	Narrow range
OPΔT	Overpower delta-T
OTΔT	Overtemperature delta-T
pcm	percent milli-rho (1 pcm = 1.0×10^{-5} δk/k reactivity)
PORV	Power operated relief valve
PZR	Pressurizer
RCP	Reactor coolant pump
RCS	Reactor coolant system
RPS	Reactor protection system
Rx	Reactor
SBLOCA	Small break loss-of-coolant accident
SG	Steam generator
SI	Safety injection
Tavg	RCS loop average or vessel average coolant temperature
Tin	Core inlet coolant temperature
Tref	Programmed reference temperature

TABLE 4.2
MODEL ORGANIZATION

3.1	General Code Input
	Title Card and Model Description Notes
	Problem Control and Dimensions
	Problem Data Card
	Minor Edit Variable Data Cards
	Time Step Data Cards
	General Trip Control Data Cards
	Steady State Initialization
	Control System Problem Dimensions
3.2	Component and System Models
3.2.1	Reactor Protection System
3.2.2	Reactor Vessel and Core
3.2.2.1	Reactor Vessel and Core Volumes
3.2.2.2	Reactor Vessel and Core Junctions
3.2.2.3	Reactor Vessel and Core Heat Conductors
3.2.2.4	Reactor Vessel and Core Material Properties
3.2.3	Primary Piping
3.2.4	Reactor Coolant Pumps
3.2.5	Pressurizer
3.2.5.1	Surge Line
3.2.5.2	Pressurizer Level
3.2.5.3	Pressurizer Pressure Control
3.2.5.3.1	Pressurizer Heaters
3.2.5.3.2	Spray
3.2.5.3.3	Power Operated Relief Valves
3.2.5.4	Safety Valves

Table 4.2 (cont.)
MODEL ORGANIZATION

3.2.6	Steam Generators
3..2.6.1	Primary Side
3.2.6.2	Secondary Side
3.2.6.3	Tubes
3.2.6.4	Steam Generator Water Level Instrumentation
3.2.6.5	Steam Generator Mass Summation
3.2.7	Main Steam System
3.2.7.1	Steam Line Isolation
3.2.7.2	Steam Lines
3.2.7.3	Condenser Steam Dump System
3.2.7.4	Main Steam Relief Valves
3.2.7.5	Main Steam Safety Valves
3.2.8	Main Feedwater System
3.2.9	Auxiliary Feedwater System
3.2.10	Turbine EHC System and Automatic Runback
3.2.11	Sink Volume
3.2.12	Safety Injection System
3.2.13	Reactor Kinetics
3.2.13.1	Reactivity Models
3.2.13.2	Rod Control System
4.0	Initialization Parameters
5.3.x	3-Loop, Single Node Steam Generator Configuration
8.0	Steam Line Break Module

Table 4.3

CONTROL VOLUME SUMMARY

Volume Description	Volume #	Bubble Index	Tmp Trp Delay
Rx vessel upper plenum	1	0	No
Rx vessel upper head	10	x	No
Rx vessel downcomer	11	0	No
Rx vessel lower plenum	12	0	No
Core bypass	13	0	Yes
Lower core section	14	0	No
Mid core section	15	0	No
Upper core section	16	0	No
Hot leg piping	X01	0	Yes
Pump suction piping	X13	0	Yes
Reactor coolant pump	X14	0	No
Cold leg piping	X15	0	Yes
Pressurizer	17	x	No
Surge line	18	0	No
SG tubes	X03-X12	0	No
SG downcomer *	X39	0	Yes
SG tube bundles *	X40-X48	0	No
SG separator *	X49	0	No
SG steam dome	X50	x	No
Steam lines	X60, X61	0	No
Main steam header	400	0	No

Notes

Bubble index = 0 indicates volume is treated as homogeneous,
= x indicates a bubble index applied.

Tmp Trp Delay = temperature transport delay option.

X = Loop No = 1, 2 or 3

* Present only in multi-node steam generator secondary geometry configurations.

Table 4.4

MOMENTUM JUNCTION SUMMARY

Junction Description	Jct #	Type	Valve	Chok	Trp
Upper head — upper plenum	10	Norm	0	No	No
Downcomer — lower plenum	11	Norm	0	No	No
Lower plenum — bypass	12	Norm	0	No	No
Lower plenum — core 1	13	Norm	0	No	Yes
Core 1 – core 2	14	Norm	0	No	Yes
Core 2 – core 3	15	Norm	0	No	Yes
Core 3 – upper plenum	16	Norm	0	No	Yes
Bypass – upper plenum	17	Norm	0	No	No
Rx vessel outlet nozzle	X01	Norm	0	No	No
Hot leg – SG tubes	X03	Norm	0	No	Yes
SG – RCP suction	X13	Norm	0	No	Yes
Rx vessel inlet nozzle	X16	Norm	0	No	No
Pump suction	X14	Norm	0	No	No
Pump discharge	X15	Norm	0	No	No
Pressurizer — surge line	20	Norm	0	No	No
Surge line — "C" hot leg	21	Norm	0	No	No
Pressurizer spray intake	18	Fill	0	No	No
Pressurizer spray	19	Fill	0	No	No
Pressurizer PORV No. 1	24	Norm	x	Yes	No
Pressurizer PORV No. 2	25	Norm	x	Yes	No
Pressurizer safety valves	27	Norm	x	Yes	No

Notes

Type = junction type, i.e., normal or fill.

Valve = 0 indicates no valve model is specified.

Chok = choking option.

Trp = enthalpy transport option.

X = Loop No = 1, 2 or 3

Table 4.4 (cont.)

Junction Description	Jct #	Type	Valve	Chok	Trp
SG tubes	X04-X12	Norm	0	No	Yes
SG dome – downcomer	X39	Norm	0	No	No
SG downcomer – hot riser	X40	Norm	0	No	Yes
SG hot side tube bundles	X41-X44	Norm	0	No	Yes
SG downcomer – cold riser	45	Norm	0	No	Yes
SG cold side tube bundles	X46-X49	Norm	0	No	Yes
SG tube bundle – separator	X50	Norm	0	No	Yes
SG separator – dome	X51	Norm	0	No	No
SG outlet	X60	Norm	0	No	No
MS isolation valves	X61	Norm	x	No	No
MS header inlet	X62	Norm	0	No	No
Condenser steam dump	401	Norm	x	Yes	No
MS PORV (relief valve)	X26	Norm	x	Yes	No
SG safety valves	X27-X31	Norm	x	Yes	No
Feedwater inlet	X35	Fill	0	No	No
Turbine steam flow	400	Fill	0	No	No

Notes

Type = junction type, i.e., normal or fill.

Valve = 0 indicates no valve model is specified.

Chok = choking option.

Trp = enthalpy transport option.

X = Loop No = 1, 2 or 3

FIGURE 4.1

NAPS/Surry RETRAN Model –Primary Nodalization

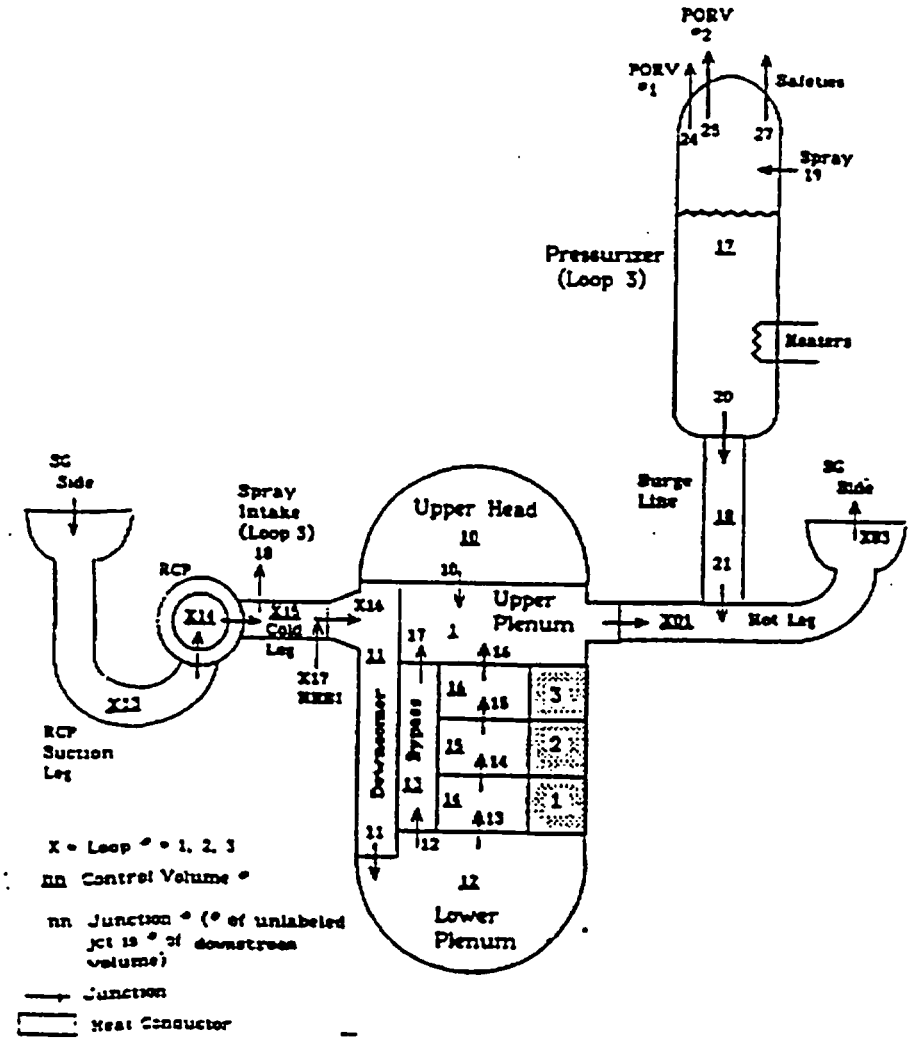


FIGURE 4.2

NAPS/Surry RETRAN Model –Multi-Node
Steam Generator Nodalization

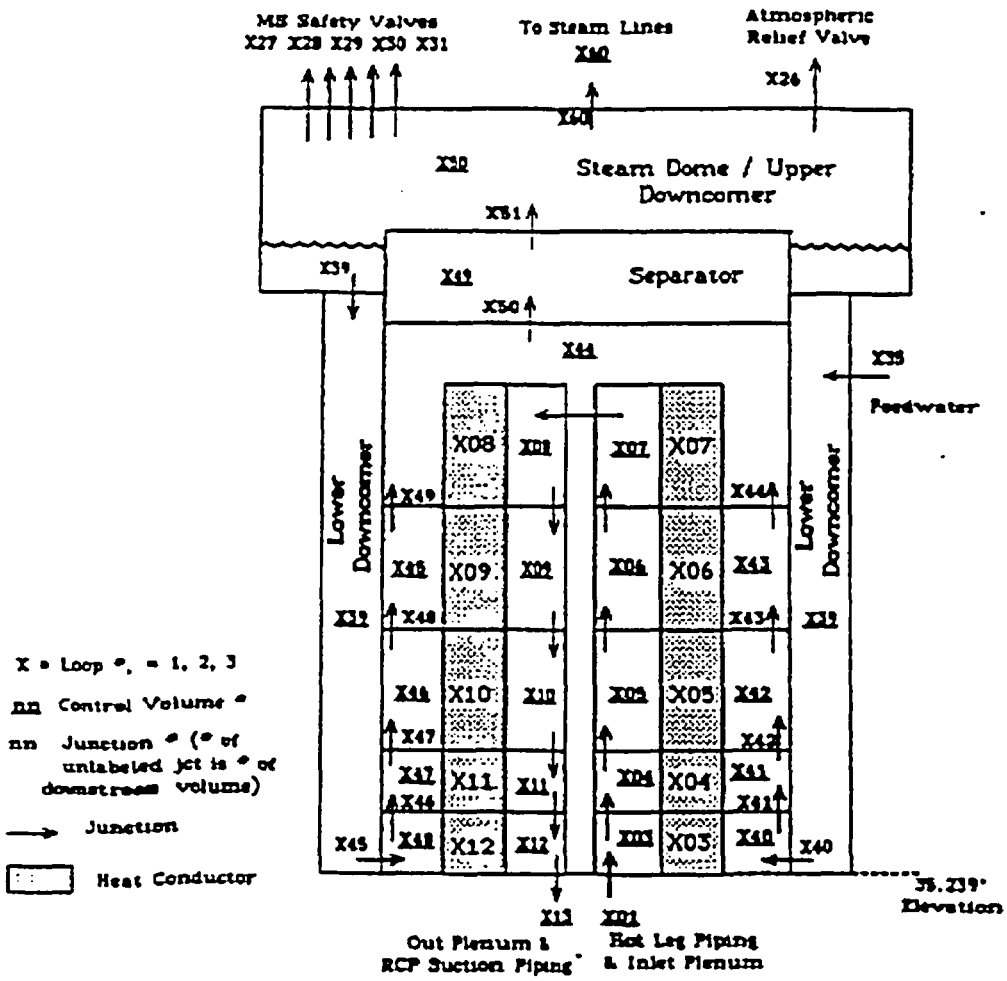
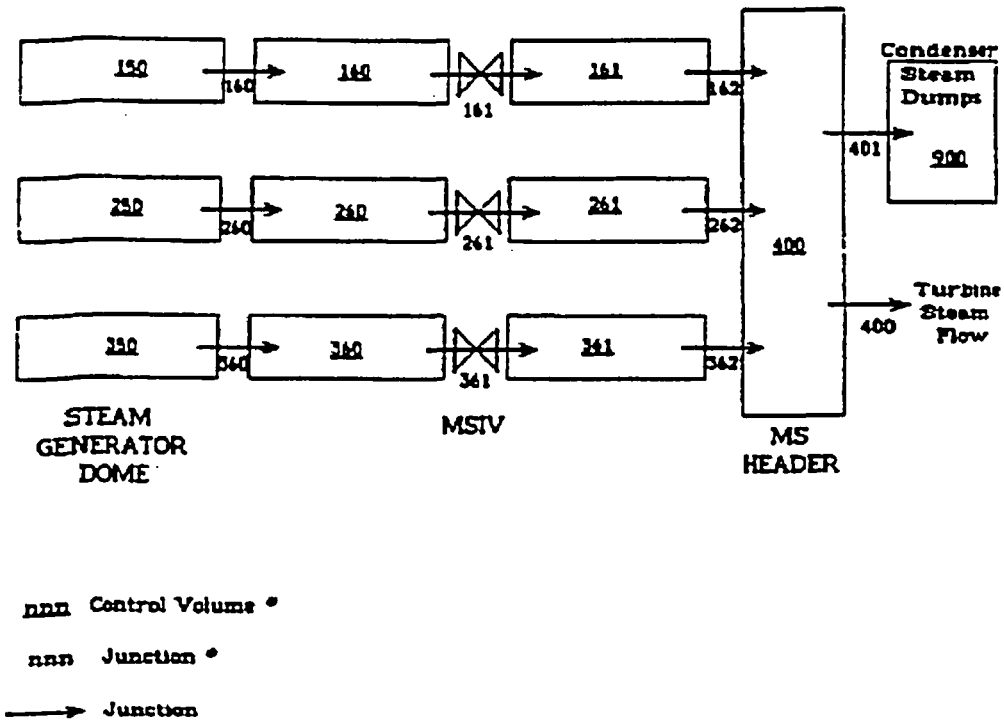


FIGURE 4.3

NAPS/Surry RETRAN Model Steam Line Nodalization



5.0 SYSTEM COMPONENT MODEL DESCRIPTIONS

Note: Material in Section 5 referring to "the model" refers to the base model configuration, i.e. three discrete reactor coolant loops and a single steam generator secondary side volume, unless otherwise noted.

5.1 Generic Problem Definitions

Table 5.1 provides a summary of the generic problem control assumptions for the Dominion model (i.e., the assumptions specified on the 01000X problem control cards).

Table 5.1-1
PROBLEM CONTROL ASSUMPTION SUMMARY

1. The two stream momentum mixing (jet pump) option is not used.
2. The core kinetics is based on one prompt group, six delayed groups, decay heat represented by eleven pseudo isotopes, and U-239 and Np-239.
3. No metal water reaction is calculated.
4. No equivalent level calculation is performed.
5. The dynamic slip model is used.
6. The steady state initialization option is used.
7. The non-equilibrium pressurizer option is used.
8. The transport delay option is used.
9. The auxiliary DNB model is not used.
10. The RETRAN01 heat transfer map is used.
11. The iterative numerics solution is used.
12. The local conditions heat transfer model is not used*.
13. The turbine model is not used.
14. The equation of state is used for core voids.
15. An arithmetic average volume flow is used for the momentum flux.
16. No non-equilibrium separators are used.

* The local conditions heat transfer model has been qualified for use with the single node SG secondary model option for loss of heat sink events (see Section 5.7.2). The base models do not include this option.

5.2 Reactor Protection System

5.2.1 General

Each trip function is represented independently by its own RETRAN trip ID. This allows the actuation time of each trip function to be edited separately even when disabled. Representative reactor trip functions, setpoints and delay times are summarized in Table 5.2-1.

Several of the RPS functions require further description.

5.2.2 Neutron Flux Signals

RPS functions using neutron flux signals use the neutron power as a percent of rated power. The decay heat model and the input decay heat multiplier indirectly define the amount of core power from neutrons. For example, if the reference model decay heat input results in 7.2375% decay heat, the remaining 92.7625% core power is due to neutrons. Therefore, a gain of 1/0.927625 is applied to the total core power to convert it to neutron power.

5.2.3 Disabled Functions

The low power range and intermediate range high flux trip functions are disabled except in the HZP I.C. modules decks. The low RCP bus frequency and voltage functions are set to trip on time and disabled since the RCP power source is not modeled.

5.2.4 Pressurizer Pressure Functions

The high pressurizer pressure function is triggered directly on the pressurizer control volume pressure. The low pressurizer pressure function and OTΔT functions use a compensated (lead-lag) pressurizer pressure.

5.2.5 Overtemperature Delta-T

The OTΔT setpoint is given by:

$$\Delta T = \Delta tref * \{K1 - K2*[\frac{(1 + t1*S)}{(1 + t2*S)}] * (T\text{-}Tref) + K3*(P\text{-}Pref) - f(\Delta I)\}$$

where,

ΔT	= OTΔT setpoint
Δtref	= Rated (HFP) hot leg T minus cold leg T
Tref	= Reference RCS Loop Tavg
Pref	= Reference pressurizer pressure
T	= RCS instrument loop average temperature
P	= Compensated pressurizer pressure
f(ΔI)	= Core axial power offset modifier
t1	= Lead time constant
t2	= Lag time constant
K1	= Constant
K2	= Constant
K3	= Constant

The values for the constants are found in the Technical Specifications and/or the Technical Requirements Manual. The RETRAN model K1 value has been increased from the nominal value by an amount that envelops the Channel Statistical Allowance (CSA).

The value of Δtref for the OTΔT and OPΔT trips is dependent on the power and RCS flow rate. Therefore, Δtref is established as the initial ΔT for HFP at the specified RCS flow rate. The user must change Δtref for cases based on a different power level or RCS flow rate.

The measured temperature, T, is the RTD instrument loop average temperature. A small time delay is applied to model thermal and hydraulic mixing in the RTD scoops. A lag time is applied to the temperatures to model the RTD thermal time constant. An additional delay is included to account for the remaining electrical and mechanical equipment.

5.2.6 Overpower Delta-T

The OPΔT setpoint is given by:

$$\Delta T' = \Delta tref * \{K4 - K5 * [\frac{t3*S}{(1 + t3*S)}] * T + K6 * (T - Tref) - f(\Delta I)\}$$

where, with the following exceptions, terms are defined as in the OTΔT function above.

ΔT'	= OPΔT setpoint
t3	= Time constant
K4	= Constant
K5	= Constant, %/°F if T is increasing
	= 0 if T is decreasing
K6	= Constant

The time delays (scoop mixing and electronics) and lag (RTD thermal response) described for the OTDT apply to this trip as well.

The values for the constants are found in the Technical Specifications and/or the Technical Requirements Manual. The RETRAN model K4 value has been increased from the nominal value by an amount that envelops the CSA.

5.2.7 Qualification and Restrictions

Dominion has established a configuration control program to document the relationship between the following quantities and to calculate associated analytical margins:

- the reactor protection system and engineered safety features actuation system setpoints assumed in the safety analyses (the safety analysis limits or SALs),
- the nominal and allowable setpoints specified in the Technical Specifications,
- the actual setpoints implemented in plant procedures, and
- the calculated instrument uncertainty allowances (referred to as channel statistical allowances or CSAs).

Table 5.2-1
REACTOR PROTECTION SYSTEM MODELS
TYPICAL SAFETY ANALYSIS LIMITS
(MODEL SETPOINTS)

Trip Signal	Setpoint	Delay (sec)
Low and intermediate range, high flux	≥ 35% rated power	0.5
High range, high level flux	≥ 118% rated power	0.5
Positive flux rate (North Anna)	≥ 5%/sec	0.5
Negative flux rate (North Anna)	≤ -5%/sec	0.5
High pressurizer pressure	≥ 2396 psia	2
High pressurizer level	≥ 100% pzr span	2
Low pressurizer pressure	≤ 1845 psia	2
Overtemperature delta-T	**	**
Overpower delta-T	**	**
Low reactor coolant flow	≤ 87% nominal flow	1

Notes

** See description in Section 5.2

Table 5.2-1 (CONT)
REACTOR PROTECTION SYSTEM MODELS
TYPICAL SAFETY ANALYSIS LIMITS
(MODEL SETPOINTS)

Trip Signal	Setpoint	Delay (sec)
Low feedwater flow***	≤ 25% SG NR level	2
	≥ 50% steam flow minus FW flow	0.1
SG low-low NR level	0% SG NR level	2
Rx trip on turbine trip	turbine trip	0.1
Rx trip on safety injection	safety injection	0.1

Notes

*** Not credited in limiting cases which establish AFW requirements.

5.3 Reactor Vessel and Core

5.3.1 Description

5.3.1.1 General

Because of its similarity to a plenum, the temperature transport delay option was not selected for the downcomer.

The downcomer volume includes the fluid region between the reactor vessel barrel and core baffle although the transit time through this region is considerably longer than that through the region between the vessel and barrel and is, therefore, effectively dead space.

The core sections consist only of the active core (i.e., from the bottom to the top of the fuel pellet cold dimension) and do not include the volume between the lower and upper core plates.

The upper head is represented as a stagnant volume. The upper head bubble model is set to provide a maximum gradient and complete separation such that only liquid will be delivered to the upper plenum during transients that result in upper head flashing (until the head empties).

The fraction of power generated in each core section is based on a cosine-shaped axial power distribution throughout the core. This results in 50% of the power being generated in the mid core section and 25% in each of the remaining core sections.

All conductor power is assumed to be generated in the fuel pellet and none in the cladding.

A bounding fuel melt temperature is reflected in the fuel materials properties tables. However, fuel melt is not a phenomenon which typically experienced for the average core in non-LOCA transient analyses.

The gap conductivity is adjusted to predict a steady state core average fuel temperature that matches the vendor fuel performance model at nominal full power conditions. The RETRAN gap expansion model is not in the base model and its use has not been qualified to date.

The initial core average fuel temperature is changed as dictated by analysis considerations by changing the gap material thermal conductivity. For example, to minimize the core average fuel temperature, a very large gap conductivity is input.

5.3.1.2 Development of Fuel Design and/or Vendor-Specific Core Models

In preparation for application of the Dominion RETRAN model to Framatome ANP (FANP) fuel, a FANP-specific fuel and core model was developed (Reference 5.3-1). The development process is described here for reference, as it represents the general approach that Dominion uses for qualfying the RETRAN system model for fuel vendor and/or other fuel design changes.

5.3.1.2.1 Fuel properties

The Framatome ANP model developed from fuel and clad properties data supplied by Framatome ANP which are consistent with those used in the approved Framatome ANP safety analysis models. Fuel properties covered included:

Material properties of the three conductor materials (the fuel pellet, the pellet-cladding helium gap, and the M5 cladding)

- Thermal conductivity
- Volumetric heat capacity
- Thermal linear expansion coefficient

Plots of the data, the analytical equations used to develop the data, and graphical and numerical comparisons were presented of the Framatome ANP data to the corresponding data in

- the existing Westinghouse fuel based model
- The International Nuclear Safety Center (INSC) Material Database, Argonne National Laboratory for the US Department of Energy
- NUREG/CR-6150 (MATPRO) Reference 5.3-2

Generally, only minor differences in the data were observed. The most significant property differences are those associated with the M5 versus ZIRLO cladding.

5.3.1.2.2 Core geometry input

The Framatome ANP model was developed from Framatome ANP supplied dimensional data for the Framatome ANP fuel assemblies. Input changes were developed in the following areas:

- Core bypass geometry
  - Volume
  - Flow area
  - Flow diameter
- Active core geometry
  - Volume
  - Flow area
  - Flow diameter
- Reactor vessel flow path length and area
- Reactor vessel form loss coefficients
- Reactor core target pressure drops
- Active core inlet mass flow rate
- Geometry of the active core heat conductors

The parameter changes represented minor adjustments with respect to the existing inputs.

Steady-state initializations were run with and without the Framatome ANP models to ensure adequate convergence of the new models. Detailed comparisons of the steady-state initialization results were presented in the engineering calculation in tabular format. Review of these results showed that there are only minor differences in the Westinghouse Fuel based and Framatome ANP Fuel based models.

5.3.2 Qualifications and Restrictions

The RETRAN02/MOD002 Safety Evaluation Report (Reference 5.3-3), Enclosure 2 (Technical Evaluation Report-TER) Section II.C discussed general limitations of application of RETRAN02/MOD002. Those qualifications and restrictions that are applicable to the vessel and core model are discussed and evaluated in this section. The letter designations for the qualifications and restrictions are those of the TER.

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

Discussion
The rod ejection accident is the only non-LOCA transient analyzed with RETRAN where the metal-water reaction is applied. Dominion's RETRAN hot pin model was benchmarked against a similar vendor model and produced consistent temperature transients for consistent transient pin powers. These results are discussed in Reference 5.3-4, which documents Dominion's rod ejection methodology in its entirety.

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

Discussion
Dominion RETRAN models do not currently use the vector momentum option. As discussed further in section 5.13, Reactor Kinetics, incomplete fluid mixing between loops is modeled for steam line break based on the Indian Point 1/7 scale model mixing tests performed by Westinghouse. This is done by dividing the downcomer into two azimuthal sectors and specifying cross-flow junctions between the cold legs and downcomer sectors with form-loss coefficients to give the proper steady state mixing flows.

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

Discussion

The core conductor model in Dominion RETRAN system models does not use the gap expansion model. Dominion's hot spot model for calculating the hot pin thermal transient in rod ejection analyses models rapid gap closure following the ejection with an essentially infinite gap thermal conductivity, as described in Reference 5.3-4. Qualification comparisons of the hot spot model to vendor calculations are presented in Section 4.3.2 of Reference 5.3-4.

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

Discussion

Dominion PWR RETRAN models use bubble rise in the pressurizer, reactor vessel upper head, and steam generator dome regions [Table 4.3].

The upper head applies the bubble rise model to provide complete phase separation to account conservatively for upper head flashing during a main steam line break (MSLB). Complete separation ensures that only liquid will be delivered to the upper plenum during transients that exhibit upper head flashing. The effect of upper head flashing is seen in the abrupt change in slope in the reactor coolant system pressure following a MSLB. Dominion's RETRAN model predicts results that are similar to the vendor FSAR MSLB analysis in VEP-FRD-41-A Rev. 0 (Figure 5.47) (Reference 5.3-5).

References for Section 5.3

5.3-1 Letter from L. N. Hartz (Vepco) to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," May 13, 2002 (Serial No. 02-280).

5.3-2 NUREG/CR-6150, "SCDAP/RELAP4/MOD3.3 Code Manual, Volume 4: MATPRO: A Library of Materials Properties for Light Water Reactor Accident Analysis," Revision 2, September 2000.

5.3-3 Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

5.3-4 Virginia Power Topical Report VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient", Rev. 0, NRC SER dated September 26, 1984.

5.3-5 Virginia Power Topical Report VEP-FRD-41-A, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code," May 1985.

5.4 PRIMARY PIPING

5.4.1 Description

Unlike the model presented in Rev. 0 of this report, there is no separate SG inlet volume. The hot leg volume now runs from the reactor vessel outlet nozzle to the top of the SG hot side tubesheet. The RCP suction leg runs from the top of the SG cold side tubesheet to the RCP intake.

For scenarios that result in two phase natural circulation (not normally encountered in non-LOCA analyses), a more accurate representation of the thermal driving head can be achieved by breaking the RCP suction leg into at least two volumes; e.g., one from the SG to the lowest point in the leg and the second back up to the pump suction.

The temperature transport delay option is applied to all RCS piping volumes.

5.4.2 Qualifications and Restrictions

The three loop model predicts natural circulation conditions consistent with those measured during the North Anna Unit 2 Natural Circulation Tests conducted July 3 through 9, 1980 (Reference 5.4-1). Test 2-ST-8, conducted at 3% RTP measured a stable vessel ΔT of 36-40 ° F. (see Figure 5.4-1). Figure 5.4-2, below, taken from Section 15.2 of the North Anna UFSAR, shows the response to a loss of offsite power predicted by the RETRAN model. Note during the period of stable natural circulation and boiloff of available steam generator inventory, the vessel ΔT is of the same magnitude as measured in 2-ST-8.

The RETRAN02/MOD002 Safety Evaluation Report (Reference 5.4-2), Enclosure 2 (Technical Evaluation Report-TER) Section II.C discussed general limitations of application of RETRAN02/MOD002. Those qualifications and restrictions that are applicable to the primary piping model are discussed and evaluated in this section. The letter designations for the qualifications and restrictions are those of the TER.

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

Discussion

Dominion RETRAN models use the enthalpy transport delay model in the reactor coolant system piping and core bypass volume, where a dominant flow direction is expected. Flow reversal is not normally encountered in these volumes during non-LOCA accident analyses. For accidents that produce a flow reversal or flow stoppage, the analyst may use the transport delay model if it adds conservatism to the results (e.g., if RCS pressure is higher during a locked rotor event with the model activated).

REFERENCES FOR SECTION 5.4

5.4-1 Letter from W. L. Stewart (VEPCO) to H. R. Denton (USNRC), "Virginia Electric Power Company, North Anna Power Station Units No. 2, Response to the Additional Request for Information Concerning Low Power Natural Circulation Testing," Serial No. 427A, August 25, 1983.

5.4-2 Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

FIGURE 5.4-1

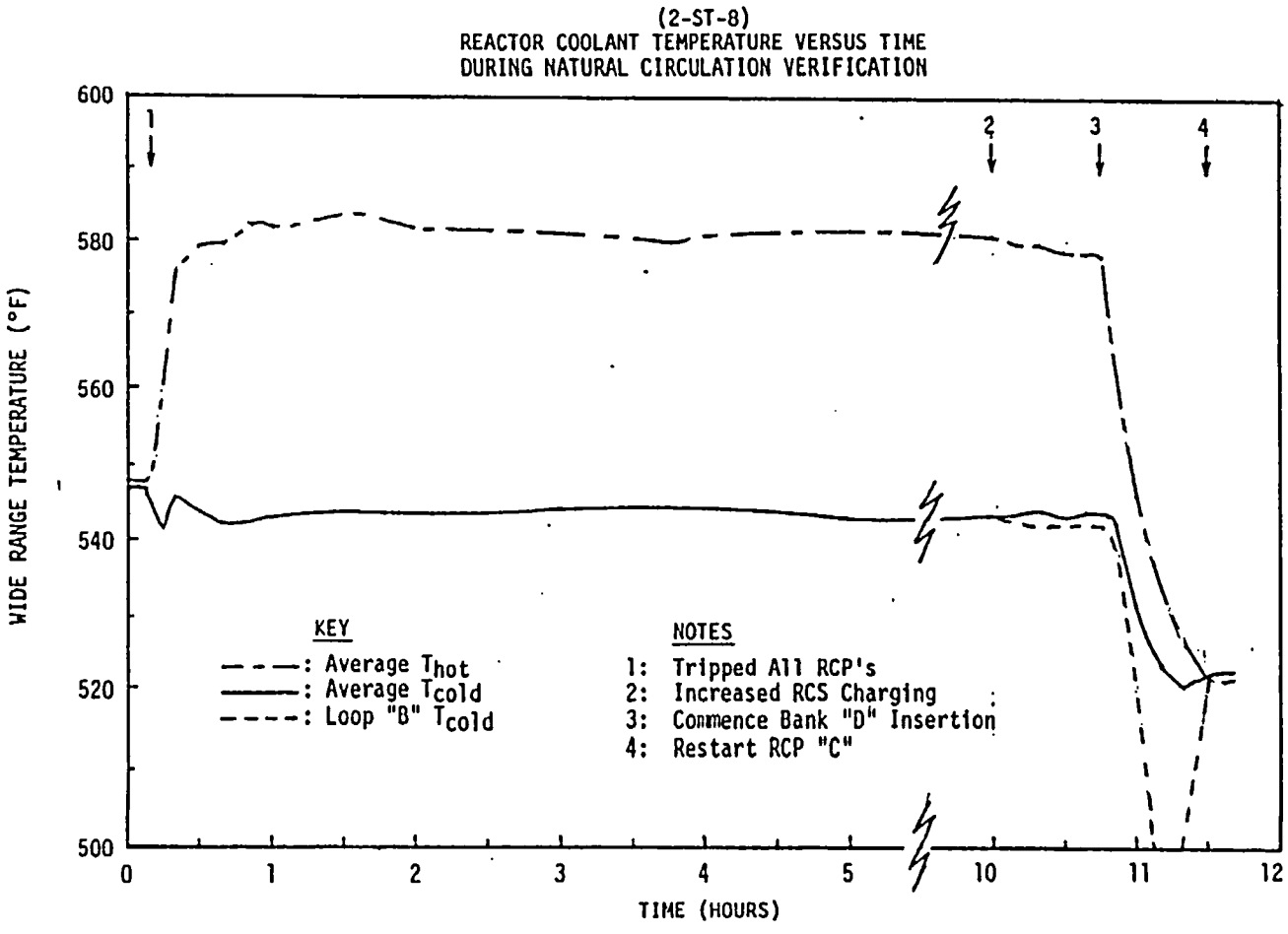
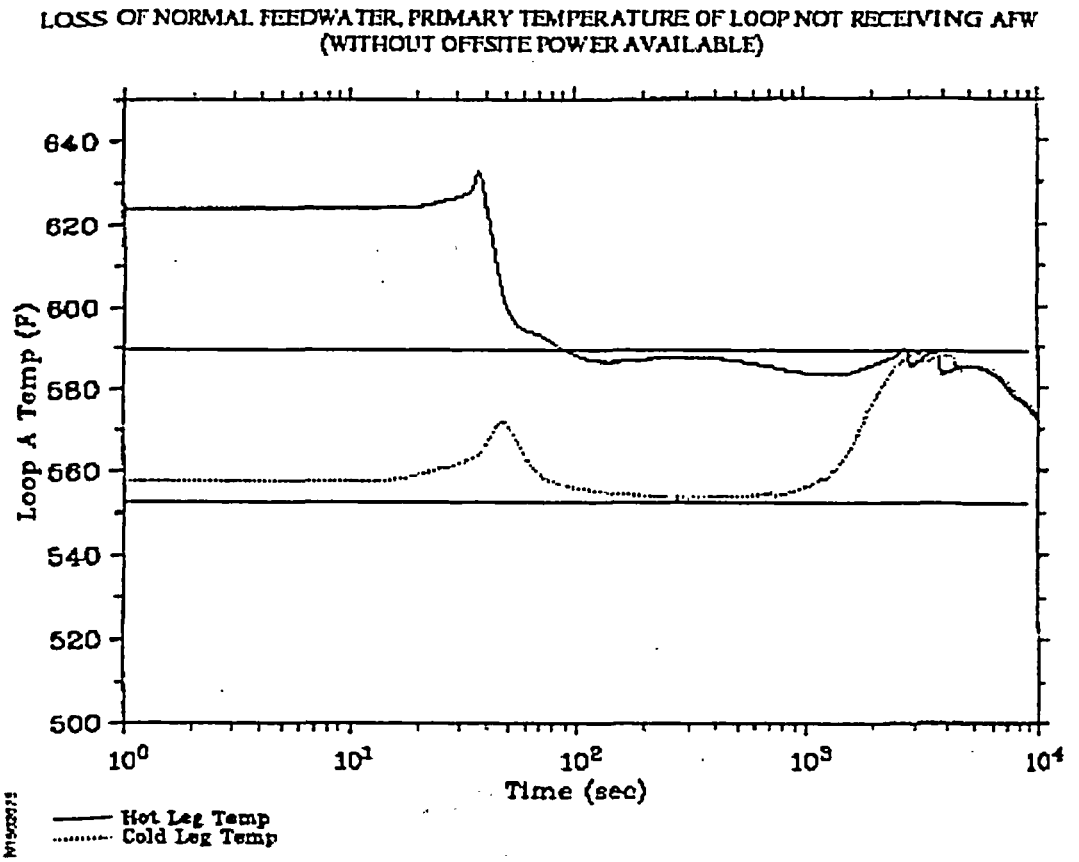


FIGURE 2: AVERAGE HOT AND COLD LEG WIDE RANGE VS. TIME

FIGURE 5.4-2

Natural Circulation Response Following

A Loss of Offsite Power



5.5 REACTOR COOLANT PUMPS

5.5.1 Description

The built-in pump curves for the Westinghouse 5200 specific speed pump are selected with additional override cards for the first quadrant which includes plant- specific pump data and the latest Westinghouse locked rotor/sheared shaft coefficients.

5.5.2 Qualification and Restrictions

The rated pump parameters (hydraulic torque, moment of inertia and frictional torque) are adjusted to provide a conservative prediction of the RCP flow coastdown curve relative to startup test plant data. (Figure 5.5-1).

The RETRAN02/MOD002 Safety Evaluation Report (Reference 5.5-1), Enclosure 2 (Technical Evaluation Report-TER) Section II.C discussed general limitations of application of RETRAN02/MOD002. Those qualifications and restrictions that are applicable to the reactor coolant pump model are discussed and evaluated in this section. The letter designations for the qualifications and restrictions are those of the TER.

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

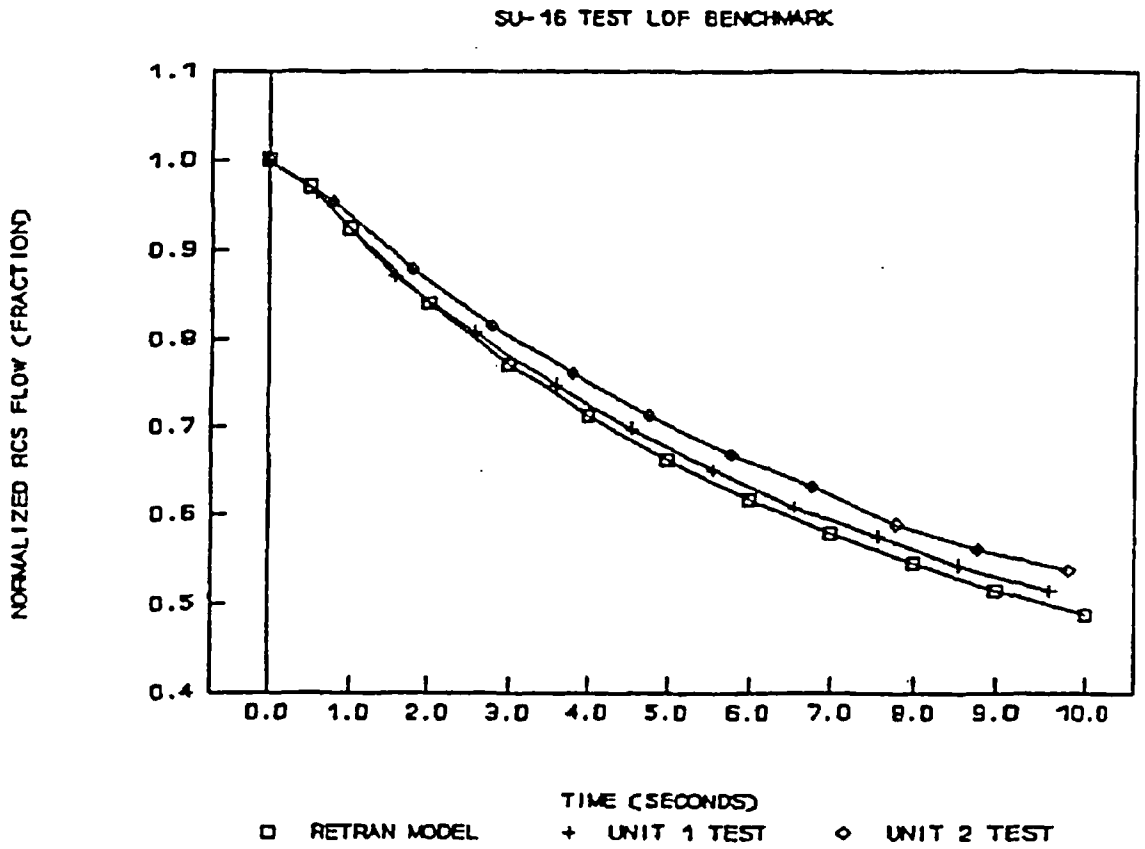
Discussion
As discussed in VEP-FRD-41-A (Appendix 1), the plant-specific pump head vs. flow response for first quadrant operation is used in the Dominion RETRAN models. The homologous curves in the model represent single-phase conditions. The RETRAN default curves are not used. The pump coastdown verifications in Section 5.3 of VEP-FRD-41-A demonstrate the adequacy of the centrifugal reactor coolant pump model versus plant-specific operational test data. Changes to the RCP coastdown model, as described in Reference 5.5-2, provide conservative coastdown flow predictions for loss of flow events relative to the actual coastdown measured at the plant. The latest Westinghouse locked rotor/sheared shaft coefficients have also been implemented.

REFERENCES FOR SECTION 5.5

5.5-1 Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

5.5-2 Letter, M.L. Bowling (VEPCO) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Supplemental Information on the RETRAN NSSS Model," Serial 93-505, August 10, 1993.

FIGURE 5.5-1
NORTH ANNA RETRAN FLOW COASTDOWN
VS STARTUP TESTS
[3-PUMP COASTDWN]



5.6 PRESSURIZER

5.6.1 Description

5.6.1.1 General

Several of the pressurizer inputs must be considered by the analyst in the context of the analysis to be performed. These include: pressurizer spray option (ISP), rainout velocity, inter region heat transfer, and the bubble gradient. The assumptions used in the base model of the pressurizer are considered to be best estimate for most transients.

The pressurizer level instrument model does not include the impact of changes in reference leg temperature that may occur as a result of changes in containment temperature. The effect of reference leg heatup is accounted for in instrument uncertainty calculations (see Section 5.2.8).

Since the physical model of the level instrument is represented, the instrument output depends on the calibration conditions for the instrument. At off-nominal conditions the same physical level will produce different instrument output. These effects are captured in the instrument model.

The base pressurizer model uses industry standard values for the bubble gradient and velocity.

The flow area through the pressurizer diffuser may have impact in those rare occasions when choking occurs at this junction (e.g., ATWS may have insurge choking and LOCA may have outsurge choking). This is not expected to be a limitation for most UFSAR transients.

The loss coefficients for the pressurizer/surge line junctions were derived from Idel'chik (Reference 5.6-1). The Idel'chik correlations for the situation of the surge line/hot leg junction, where one flow stream enters another flow stream, predicts that when the flows are sufficiently different, a jet pump phenomena occurs. For most outsurges when the RCPs are running, the loss coefficient of the hot leg-surge line junction will be negative (i.e., the hot leg flow will tend to suck the surge flow out like a jet pump). For natural circulation situations, the loss coefficient will move into the positive range. The analyst must consider these effects in specific applications.

5.6.1 Description (CONT)

5.6.1.2 Pressurizer Spray

The basic governing equation for the spray flow rate can be written

$$Q_{SPRAY} = [\Delta P / \Delta P_N]^{1/2} * Q_{RATED} * X_D$$

where

Q_{SPRAY} = volumetric spray flow rate, gpm
ΔP = dynamic pressure difference from cold leg to pressurizer, psi
ΔP_N = normal pressure difference from cold leg to pressurizer, psi
Q_{RATED} = rated volumetric spray flow rate, gpm
X_D = normalized spray flow demand (0 to 1)

This basic characteristic is modeled using control blocks.

The use of the "Normal" ΔP in the equation above ties the model to the RCS flow rate and loop pressure drop characteristics. Therefore, if the spray is important and if RCS hydraulic changes are made (e.g. a steam generator replacement), the spray model inputs are updated.

Pressurizer spray option ISP = 1 causes the spray flow to remove mass and energy from the vapor region and deposit the spray and condensed flow directly in the mixture region. This will tend to cause the vapor region to superheat during insurges and will result in slightly higher pressures. The ISP = 0 option will retain the spray in the vapor region and let it rainout according to the rainout velocity input on the pressurizer volume cards. ISP = 0 will maintain the vapor region in a saturated condition. The two options represent the ends of the behavior spectrum. The ISP = 0 option is probably the most appropriate from a best estimate standpoint, especially with the normal minimum flow through the spray nozzles. Furthermore, the desire to have the hardest response or the softest response depends on the transient being analyzed.

The base model uses the ISP = 0 option. It should be noted that, for analyses that assess transients against the RCS overpressure criterion, the pressurizer spray is typically disabled, so this option has no effect on the results. For cases where the full pressure control system is assumed active (e.g. cases assessed against a core DNBR criterion where lower pressure is conservative), the default (ISP=0) is more conservative.

5.6.1 Description (CONT)

5.6.1.3 Pressurizer PORVs

The PORV model assumes that no valve movement occurs during the early portions the PORV open and close stroke times to account for process lags.

The PORVs use the isoenthalpic critical flow model to provide an appropriate transition from steam to liquid relief. The normalized junction area can be varied for specific transients (e.g. ATWS) for conservatism. No special features are added for liquid relief.

5.6.1.4 Pressurizer Safety Valves

The three valves on the pressurizer are represented by a single valve in the model.

For the case of undrained pressurizer safety valve loop seals (currently applicable to North Anna), the safety valve model has been updated to represent the model described in WCAP-12910, "Pressurizer Safety Valve Set Pressure Shift." **[Reference 5.6-2]** This model is generally referred to as the *pop-and-blow* model. The valve begins opening at a pressure which is above the nominal setpoint by the Technical Specifications setpoint tolerance plus an additional 1% "medium shift" as defined in Ref. 5.6-2 and then "pops" completely open. A time delay is applied to the opening to model clearing of water from the loop seal as described in Ref. 5.6-2. On decreasing pressure, an open valve begins closing at the reference setpoint and fully closes at a pressure below the setpoint to account for blowdown.

Note that Surry has drained the pressurizer loops seals, so the WCAP-12910 model is no longer applicable to Surry. However, it is retained for reference as a modeling option. This option provides more limiting overpressure case results. The default Surry model is therefore similar to the North Anna model. A more realistic drained loop seal model may be specified by eliminating the 1% "medium shift" and the loop seal clearing time delay from the valve opening characteristic.

The contraction coefficient for each of the safety valves (pressurizer and main steam) is calculated based on the assumption that the rated flow is achieved at a pressure corresponding to the setpoint plus tolerance plus accumulation. This provides a low flow rate which is conservative for transients that are assessed against the RCS overpressure criterion and support the design basis for these valves.

5.6.2 Qualifications and Restrictions

5.6.2.1 RETRAN02/MOD002 SER

The RETRAN02/MOD002 Safety Evaluation Report (Reference 5.6-3), Enclosure 2 (Technical Evaluation Report-TER) Section II.C discussed general limitations of application of RETRAN02/MOD002. Those qualifications and restrictions that are applicable to the pressurizer model are discussed and evaluated in this section. The letter designations for the qualifications and restrictions are those of the TER.

f) Equilibrium thermodynamics is assumed for the thermal hydraulics field equations although there are nonequilibrium models for the pressurizer and the subcooled boiling region.

Discussion

The current version of RETRAN-02 in use at Dominion (MOD005.2) allows for multiple nonequilibrium volumes. In Dominion RETRAN models, the nonequilibrium region option is generally only used for the pressurizer, except when applied to the reactor vessel upper head in main steamline break analyses. Toward the end of the transient, the upper head, which has experienced drainage, flashing and phase separation during the cooldown, will begin to refill due to continued operation of safety injection. An equilibrium model in the head can produce nonphysical pressure oscillations. While this phenomenon generally occurs beyond the time of interest for evaluating core performance, the nonphysical behavior is avoided by using a nonequilibrium model in the upper head. This is physically reasonable for the head geometry and the limited hydraulic communication between the head and the upper plenum.

Section 5.3.3 of VEP-FRD-41-A Rev. 0 **[Reference 5.6-4]** presented comparisons of RETRAN pressure predictions to plant data for a cooldown and safety injection transient at North Anna. The nonequilibrium pressurizer model response was in good agreement with the observed plant response. Those figures are presented here for reference (Figures 5.6-1 through 5.6-5).

FIGURE 5.6-1

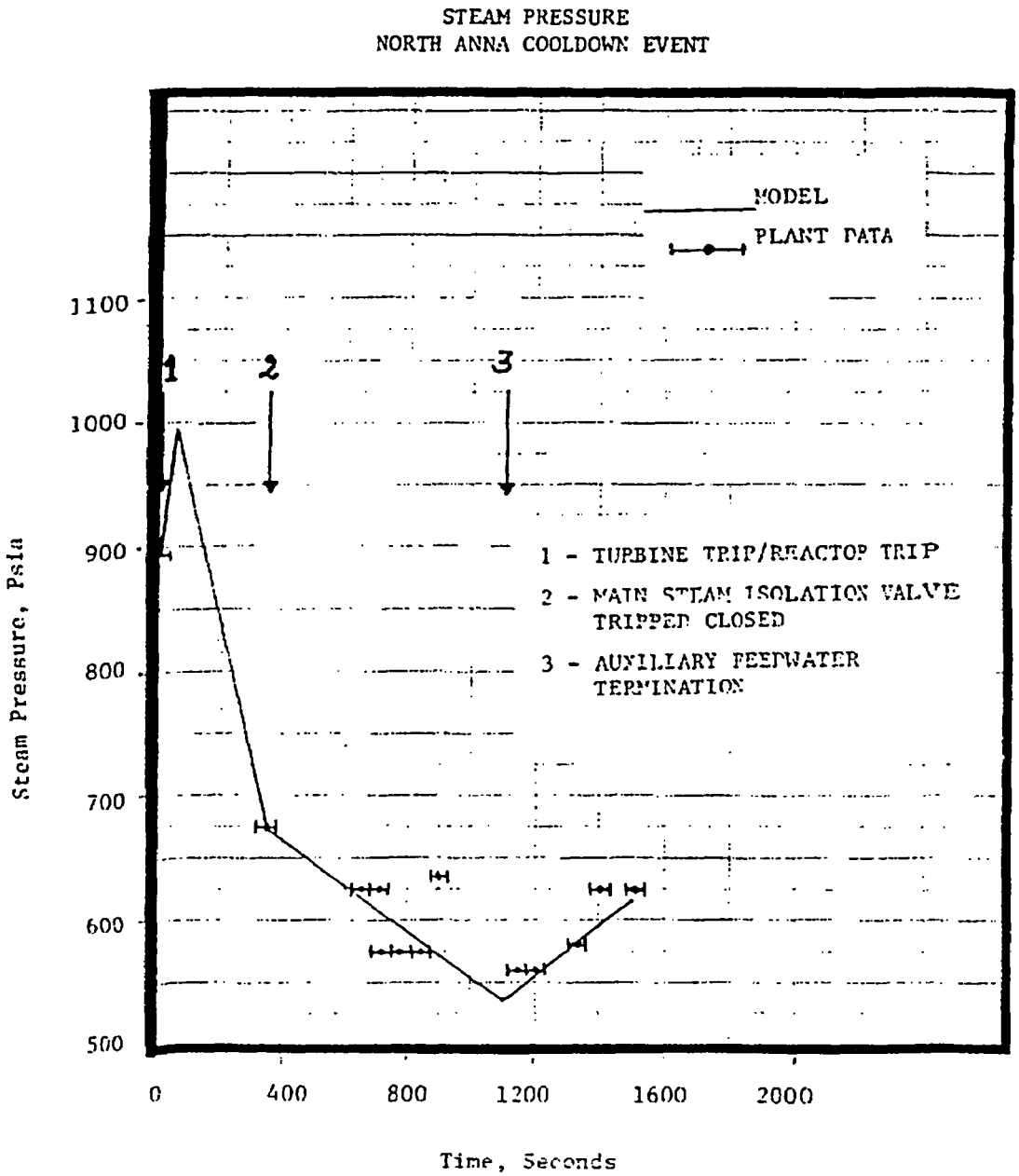

STEAM PRESSURE
NORTH ANNA COOLDOWN EVENT
MODEL
PLANT DATA
1 - TURBINE TRIP/REACTOR TRIP
2 - MAIN STEAM ISOLATION VALVE TRIPPED CLOSED
3 - AUXILIARY FEEDWATER TERMINATION
1
2
3
1100
1000
900
800
700
600
500
Steam Pressure, Psia
0
400
800
1200
1600
2000
Time, Seconds

FIGURE 5.6-2

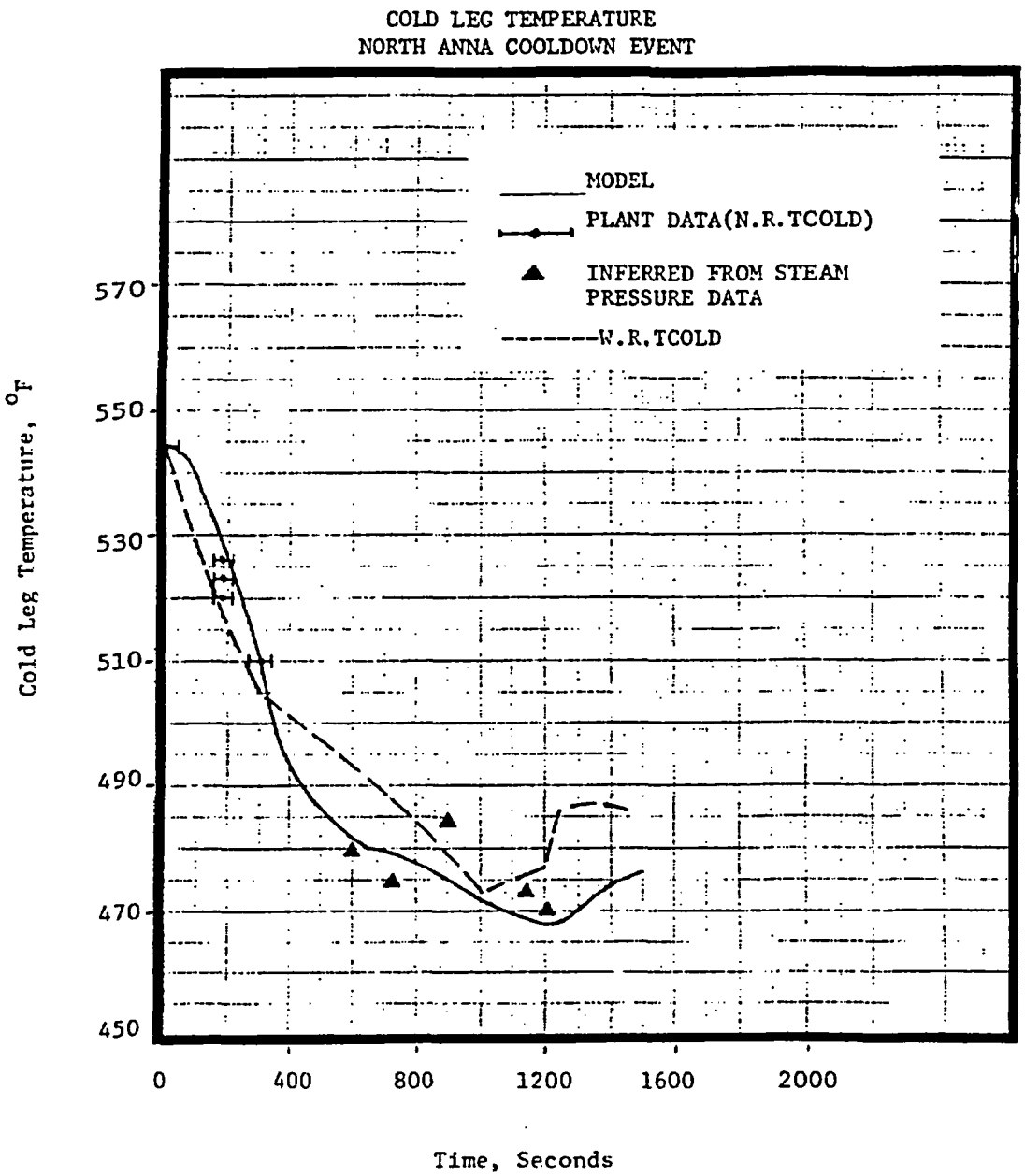

COLD LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT
MODEL
PLANT DATA(N.R.TCOLD)
INFERRED FROM STEAM PRESSURE DATA
W.R.TCOLD
570
550
530
510
490
470
450
Cold Leg Temperature, °F
0
400
800
1200
1600
2000
Time, Seconds

FIGURE 5.6-3

HOT LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT

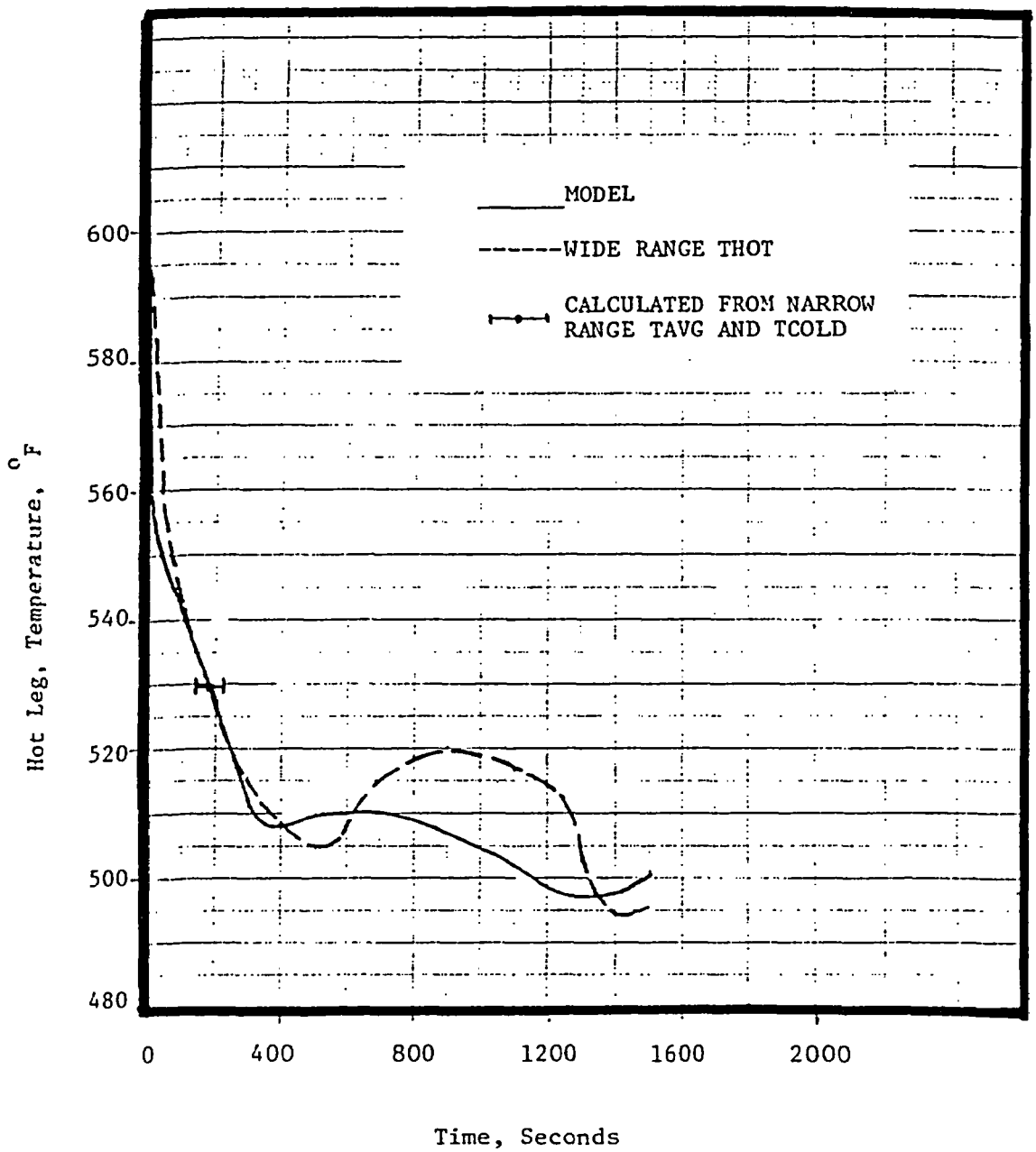


FIGURE 5.6-4

PRESSURIZER PRESSURE
NORTH ANNA COOLDOWN EVENT

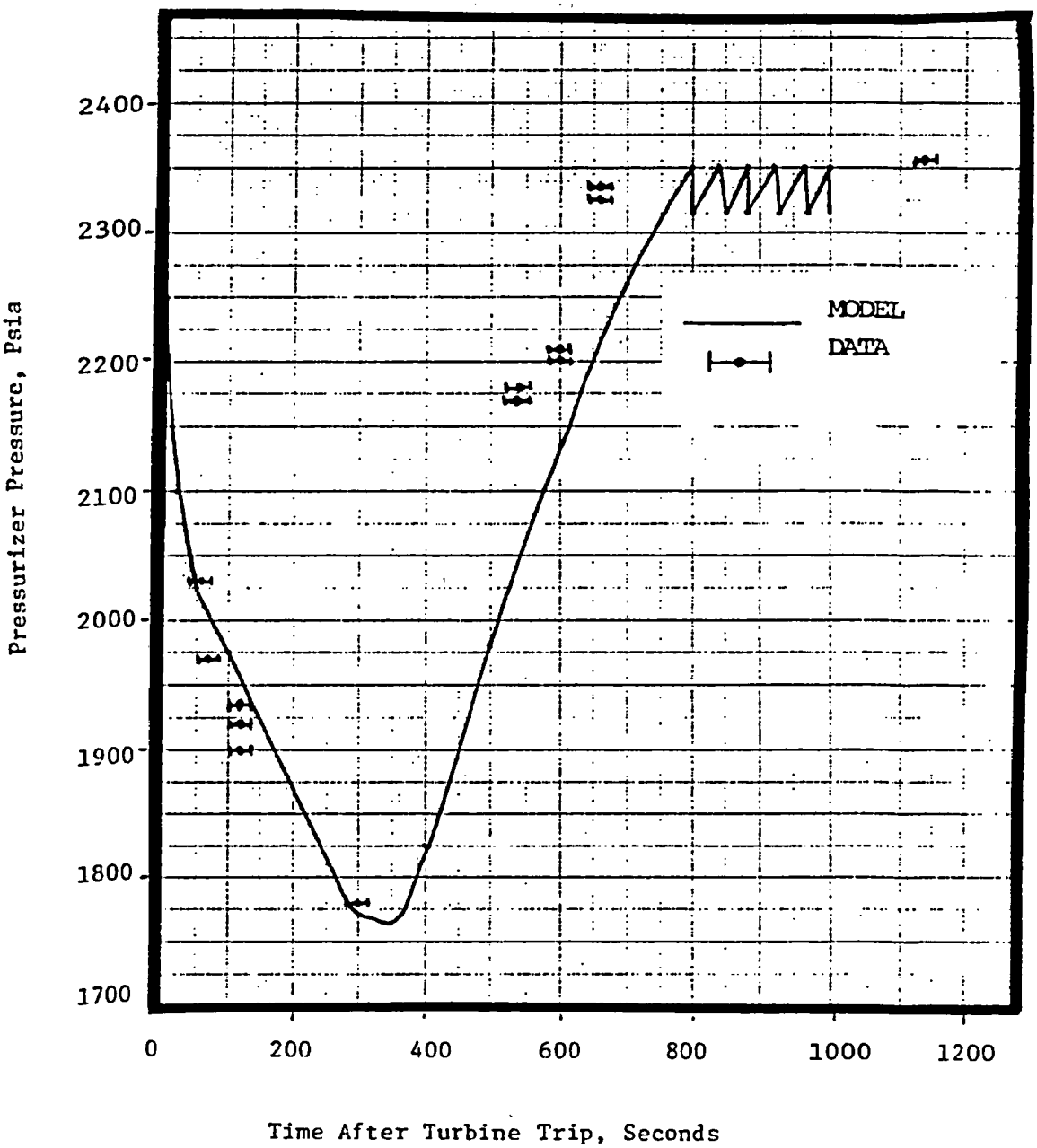
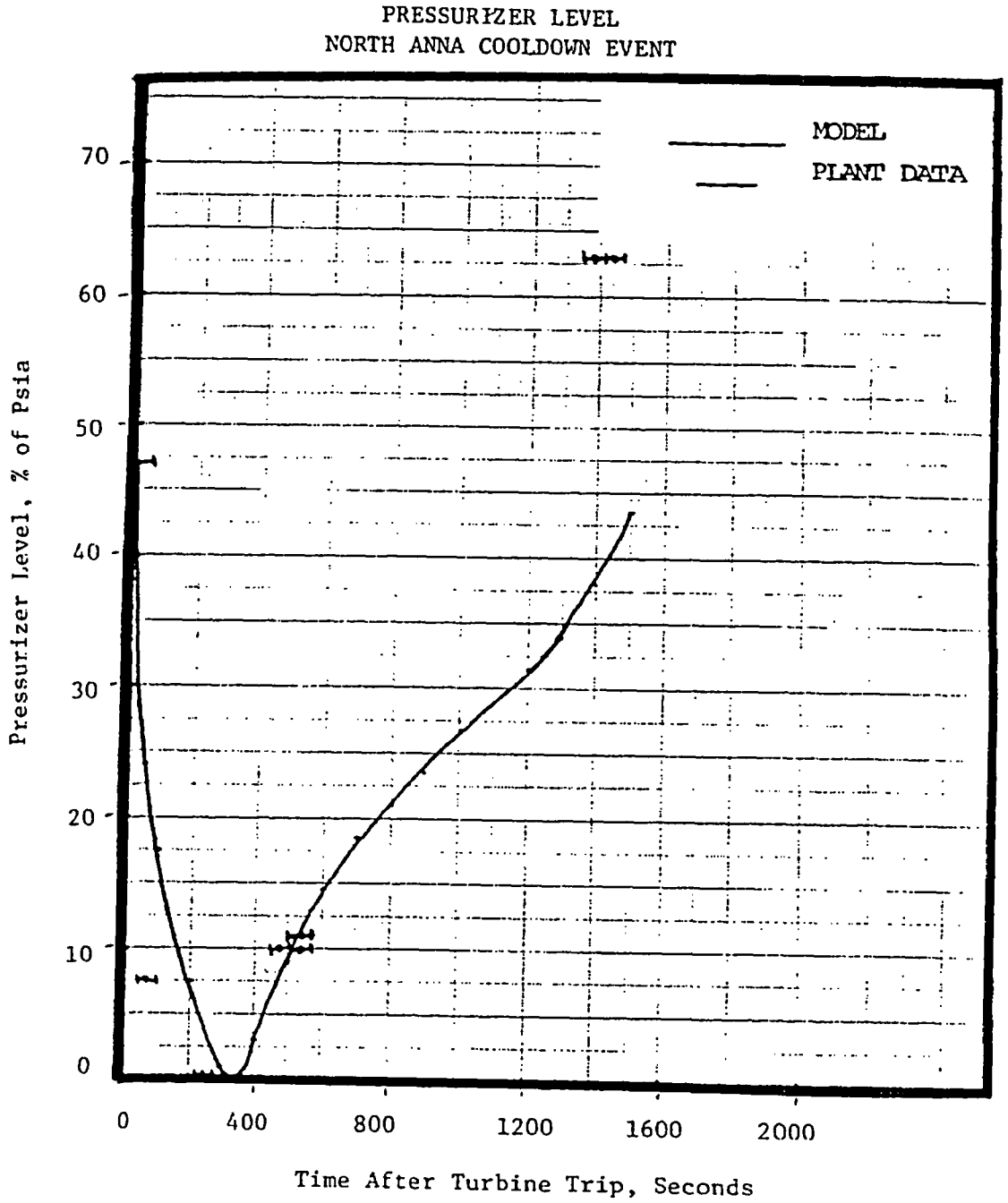


FIGURE 5.6-5


PRESSURIZER LEVEL
NORTH ANNA COOLDOWN EVENT
MODEL
PLANT DATA
Pressurizer Level, % of Psia
70
60
50
40
30
20
10
0
0
400
800
1200
1600
2000
Time After Turbine Trip, Seconds

5.6.2 Qualifications and Restrictions (CONT)

5.6.2.1 RETRAN02/MOD002 SER (CONT)

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

Discussion

VEP-FRD-41-A **[Appendix 1; Reference 5.6-4]** describes that the Dominion RETRAN pressurizer model uses the non-equilibrium model to ensure accurate modeling of transient conditions that may involve a surge of subcooled liquid into the pressurizer or to ensure appropriate treatment of pressurizer spray and heaters. While a wall heat transfer model, including vapor condensation, was added in version MOD003 **[Reference 5.6-5]**, Dominion continues to model the non-equilibrium volume walls as an adiabatic surface. However, a pressurizer heat loss term is modeled using the RETRAN pressurizer heater model. The heat loss term is set to match the design output of the proportional heater bank during steady state operation at the nominal pressure control setpoint.

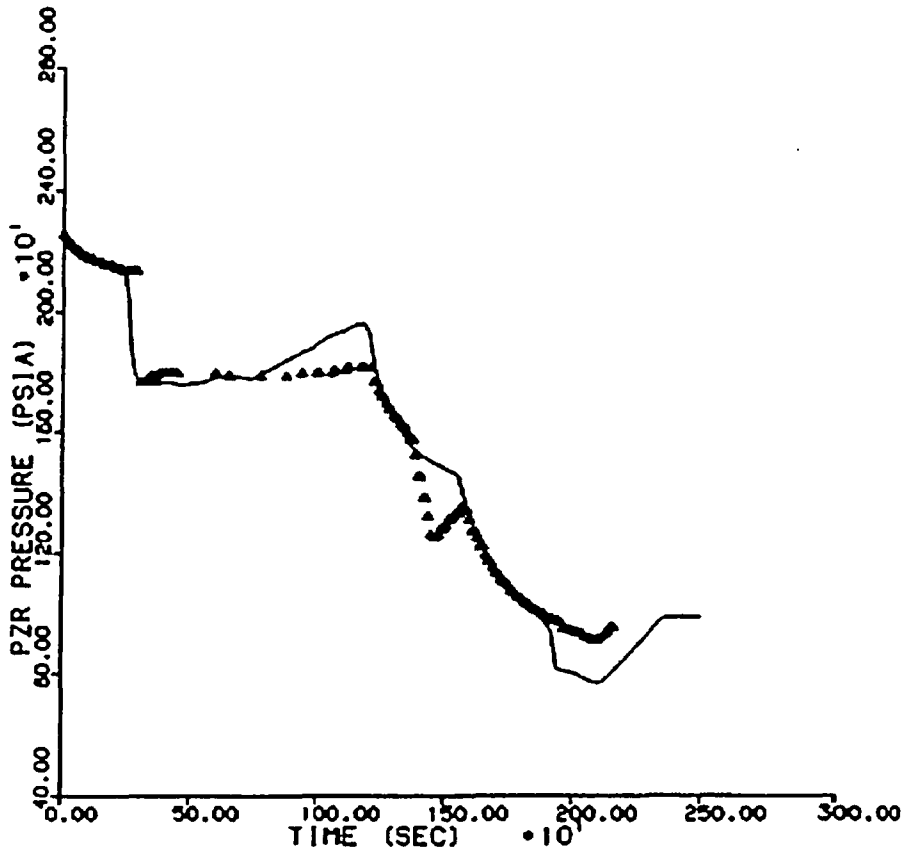
The North Anna Unit 2 Natural Circulation Tests conducted in July 1980 measured the effect of convective heat losses from the pressurizer with all heaters secured. The observed effect was about 5 F/hr liquid temperature cooldown and about 38 psi/hr pressure loss **[Reference 5.6-6]**- see Figure 5.6-12. The significant plant response for UFSAR non-LOCA transients occurs within the first 30 minutes of the event initiator. Therefore, pressurizer wall heat transfer is a phenomenon that is not significant over the time frame of interest for UFSAR non-LOCA analyses.

Section 5.3.3 of VEP-FRD-41-A includes a RETRAN simulation of a North Anna cooldown event, demonstrating the adequacy of the RETRAN pressurizer modeling assumptions compared to actual plant response. Both the observed data and the model indicated that level indication was lost for a brief portion of the transient. Overall, the RETRAN prediction of pressurizer pressure and level indicate that the non-equilibrium pressurizer model adequately describes the behavior for large swings in pressure and level. In addition, the model predicted the time when level indication was lost close to the observed data. Therefore, the RETRAN non-equilibrium pressurizer model is able to perform accurate predictions of a draining pressurizer.

Reference 5.6-7 included a RETRAN simulation comparison to the 1987 North Anna steam generator tube rupture event. Figures 5.6-6 and 5.6-7, taken from Reference 5.6-7, demonstrate that the RETRAN non-equilibrium pressurizer model provides good predictions of pressure and level behavior over a wide range of actual accident conditions. The model closely predicted the pressurizer level recovery near 1700 seconds.

FIGURE 5.6-6

NAPS 1987 STEAM GENERATOR TUBE RUPTURE EVENT
PRESSURIZER PRESSURE



SOLID LINE • RETRAN PREDICTION
SYMBOLS • PLANT DATA

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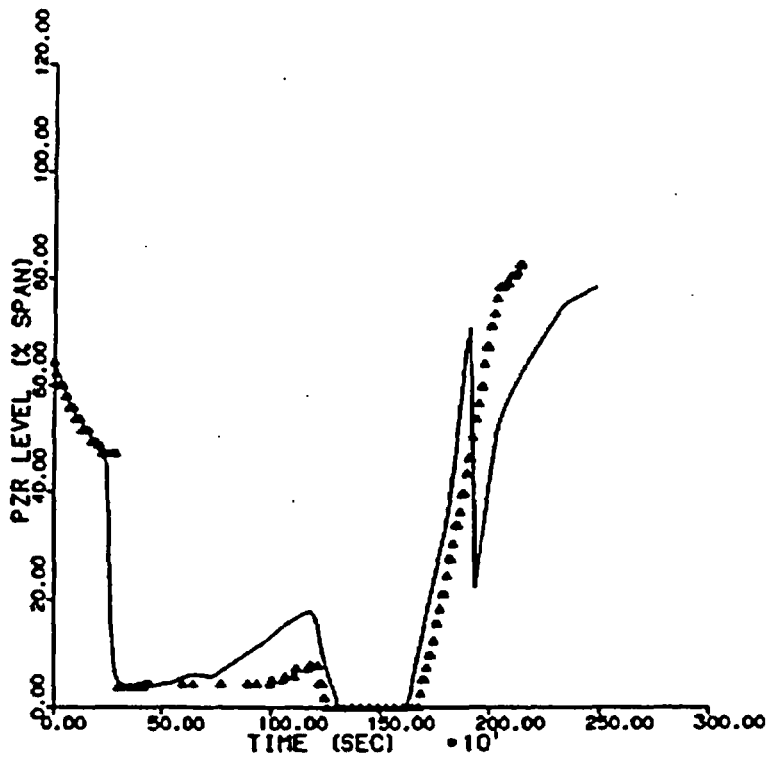
FIGURE 5.6-7

NAPS 1987 STEAM GENERATOR TUBE RUPTURE EVENT
PRESSURIZER LIQUID LEVEL



SOLID LINE = RETRAN PREDICTION
SYMBOLS = PLANT DATA

1/22/93 14.00.00

5.6.2 Qualifications and Restrictions (CONT)

5.6.2.1 RETRAN02/MOD002 SER (CONT)

RETRAN has been used to analyze the North Anna main feedwater line break (MFLB) UFSAR event, which reaches a pressurizer fill condition. The RETRAN analysis was benchmarked to a Westinghouse LOFTRAN analysis and showed good agreement for pressurizer pressure and water volume. The codes predicted similar times for the pressurizer to reach a fill condition and similar RCS conditions long-term after the pressurizer is filled. Dominion RETRAN simulations for the MFLB event do not exhibit any unusual pressurizer behavior or numerical discontinuities when the pressurizer fills and remains filled. See the Table and Figures below.

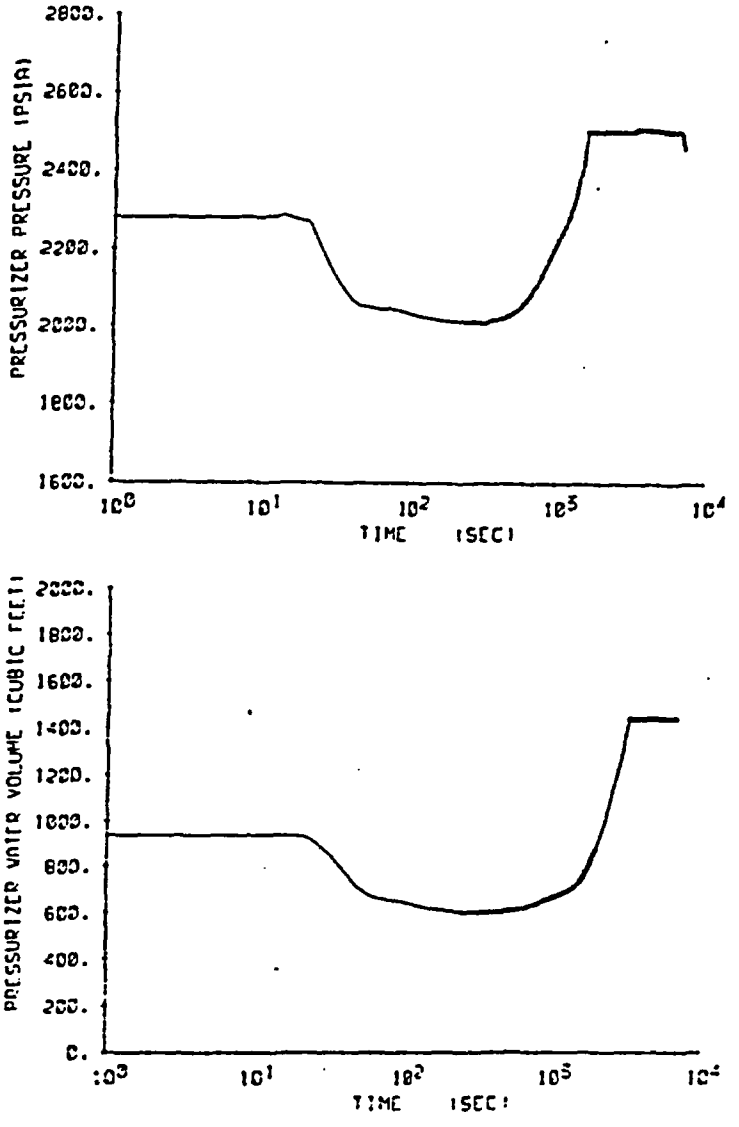
TABLE 5.6-1

TIME SEQUENCE OF EVENTS FOR MFLB COMPARISON

0.717 ft**2 break with offsite power and 340 gpm AFW flow. Time in seconds.

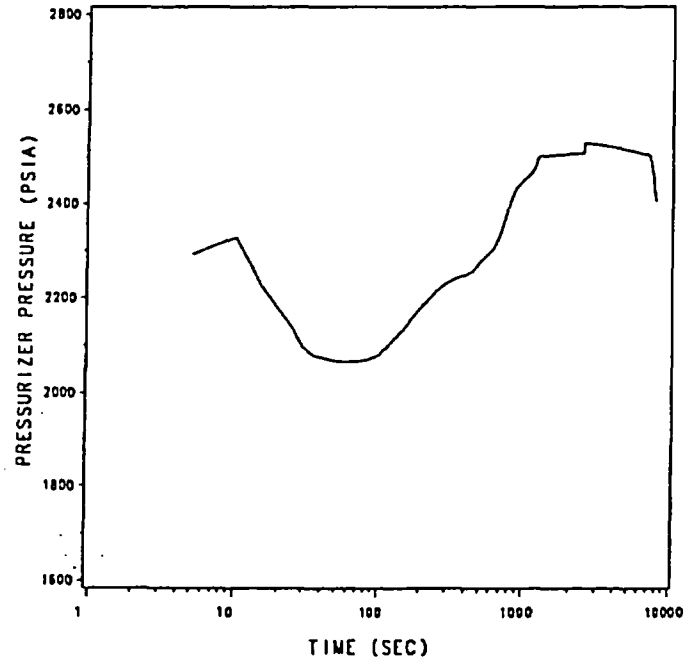
RETRAN Time	West. Time	Event
0.0	0.0	Main feedline rupture occurs
6.8	6.1	Low-low SG water level trip setpoint reached
8.8	8.1	Rods begin to drop
31.7	10.5	High steamline differential pressure S.I. setpoint reached
16.4	15.5	SG safety valve setpoint reached in intact SGs
66.8	66.1	One motor-driven AFW pump starts
350	335	Cold auxiliary feedwater is delivered to intact SG
1300	1580	Pressurizer safety valve setpoint reached
7000	6600	Core decay heat plus pump heat decreases to AFW heat removal capacity

FIGURE 5.6-8
FEEDLINE BREAK
.717 Ft^2 Break/ 340 GPM Auxiliary Feedwater
LOFTRAN Response



Main Feedline Rupture with Offsite Power Available - 0.717 ft^2 Break - Pressurizer Pressure and Water Volume as a Function of Time.

FIGURE 5.6-9
FEEDLINE BREAK
.717 Ft^2 Break/ 340 GPM Auxiliary Feedwater
RETRAN Response



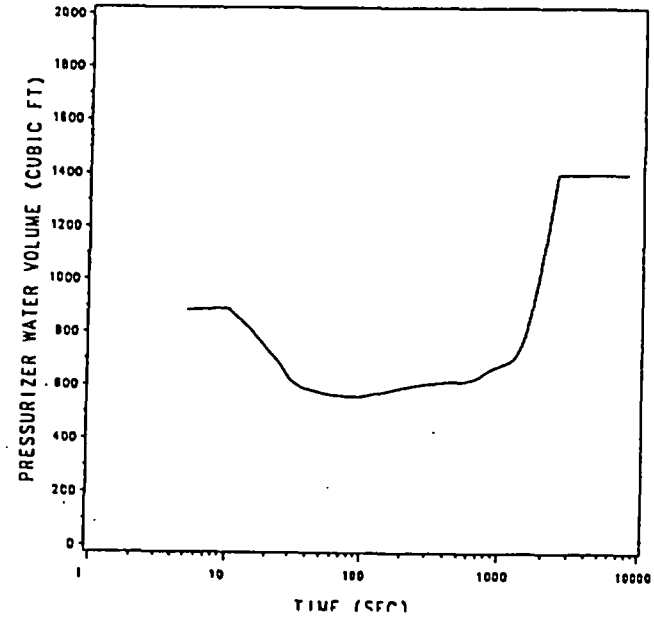


FIGURE 5.6-10
FEEDLINE BREAK
.717 Ft^2 Break/ 340 GPM Auxiliary Feedwater
LOFTRAN Response

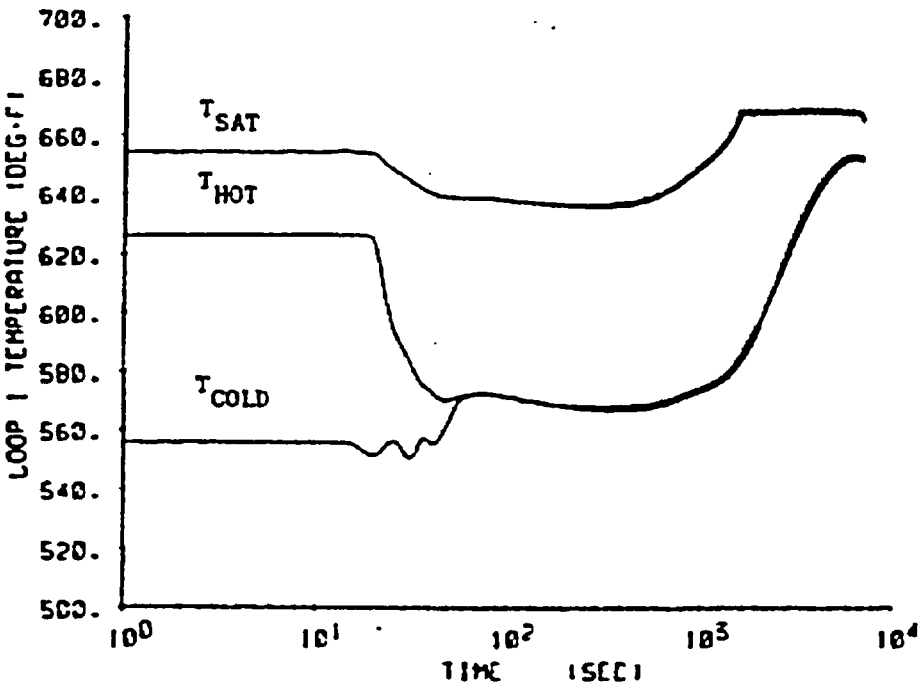


FIGURE 5.6-11
FEEDLINE BREAK
.717 Ft^2 Break/ 340 GPM Auxiliary Feedwater
RETRAN Response

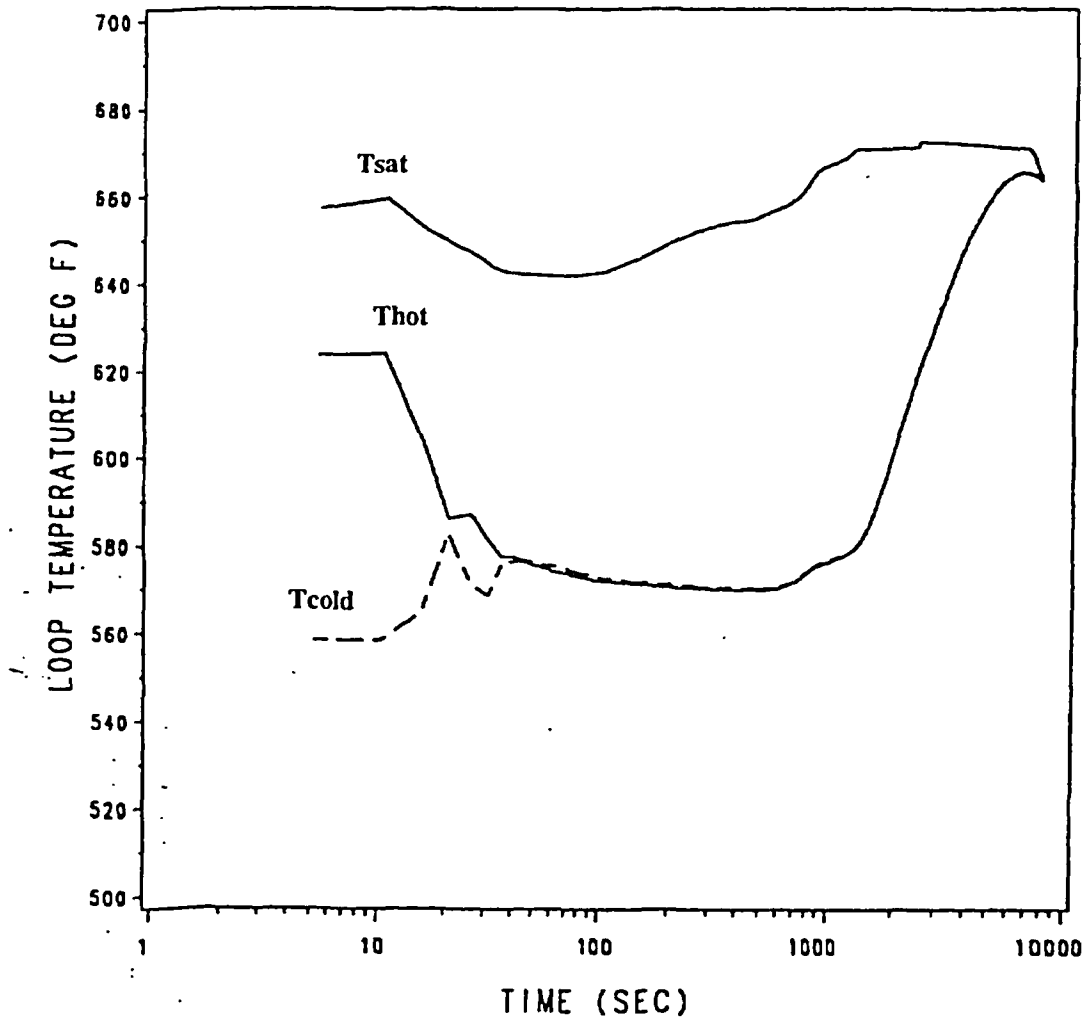


FIGURE 5.6-12

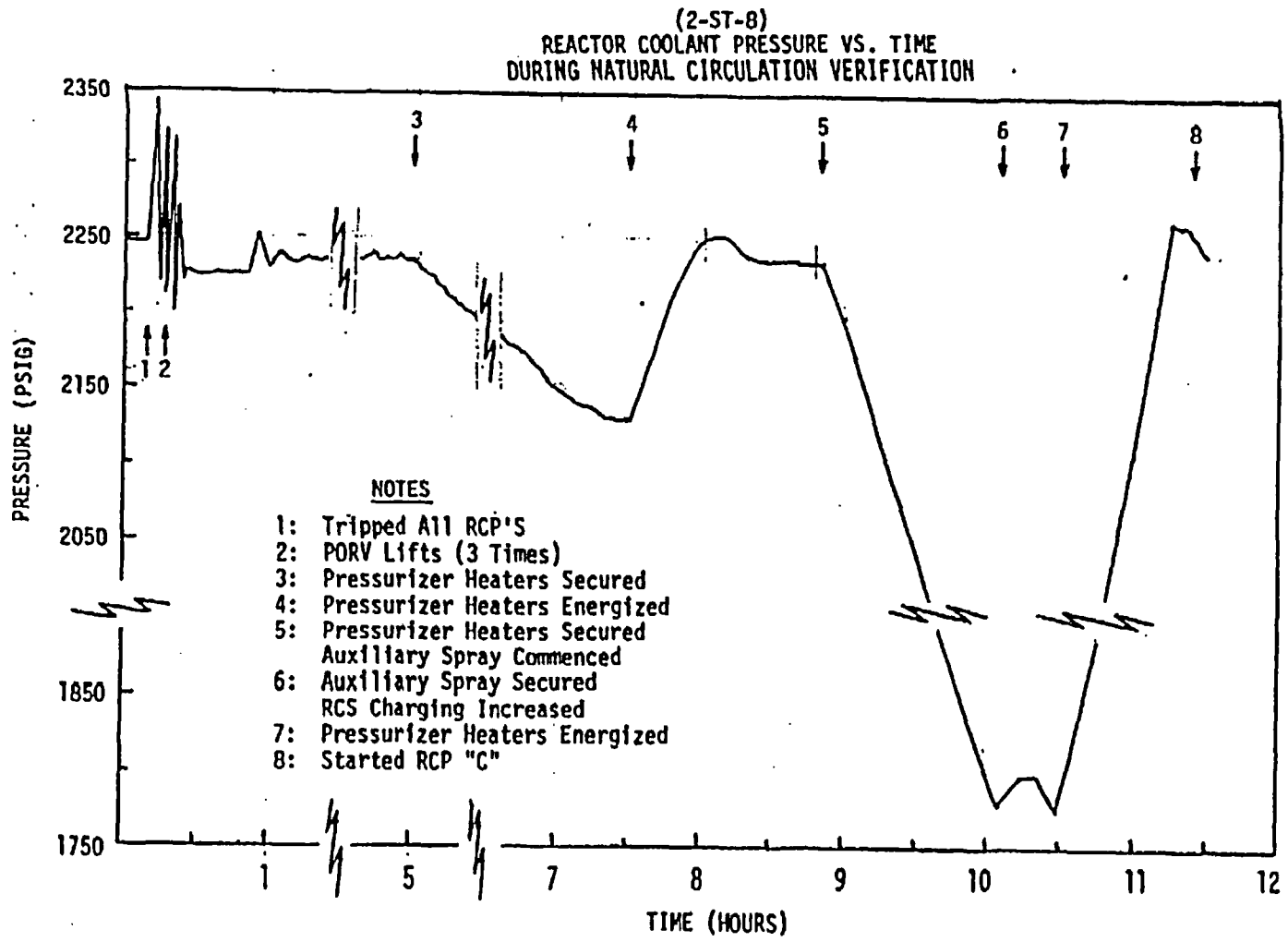


FIGURE 3: REACTOR COOLANT PRESSURE VS. TIME

5.6.2 Qualifications and Restrictions (CONT)

5.6.2.1 RETRAN02/MOD002 SER (CONT)

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

Discussion
Dominion PWR RETRAN models use bubble rise in the pressurizer, reactor vessel upper head, and steam generator dome regions [Table 4.3].

The pressurizer model applies the maximum bubble density at the interface between the mixture and vapor region. The use of the bubble rise model in the pressurizer has been qualified against licensed transient analysis codes and plant operational data as follows:

- VEP-FRD-41-A Rev. 0 (Ref. 5.6-4) RETRAN analyses show pressurizer conditions similar to the licensed FSAR analyses for several accidents: uncontrolled rod withdrawal at power, loss of load event, main steamline break, and excessive heat removal due to feedwater system malfunction.
- VEP-FRD-41-A, Section 5.3.3, RETRAN simulations show good agreement with pressurizer response operational data from the 1978 North Anna cooldown transient (see plots above) .
- RETRAN simulations show good agreement of transient pressurizer conditions compared to the 1987 North Anna Unit 1 steam generator tube rupture event (see plots above).

Implicit in the agreement between plant operational data and RETRAN is that the bubble rise model accurately predicts conditions in the pressurizer over a wide range of temperature, pressure, and level transient conditions. Therefore, Dominion has justified appropriate use of the bubble rise model through adequate benchmarking against physical data and other licensed transient analysis codes.

5.6.2 Qualifications and Restrictions (CONT)

5.6.2.1 RETRAN02/MOD002 SER (CONT)

Technical Evaluation Report (TER) "Items Requiring Further Justification"

The RETRAN-02/MOD002 TER, page E2-54, (Reference 5.6-3) includes two items that require further justification for PWR systems analysis, including one relevant to the pressurizer model.

ii) The pressurizer model requires qualification work for the situations where the pressurizer either goes solid or completely empties.

Discussion
Refer to the response to Limitation o. Dominion has shown that the non-equilibrium pressurizer model is adequate over the expected range of pressurizer conditions that occur in North Anna and Surry UFSAR non-LOCA events analyzed with RETRAN. Specifically,

- The UFSAR main steam line break events analyzed with RETRAN show a response for a drained pressurizer that is consistent with vendor methods [Reference 5.6-4, Figure 5.47].
- The North Anna UFSAR main feedline break event (case with offsite power available), which results in a filled pressurizer, shows a response that is consistent with vendor results
- Comparisons to the North Anna Cooldown Transient [Reference 5.6-4, Section 5.3.3] and Steam Generator Tube Rupture [Reference 5.6-7, Section 3.2] show reasonable agreement with plant data for the case of pressurizer drain and subsequent refill.

The RETRAN-02/MOD002 TER includes "implications of these limitations" on page E2-55. One implication is relevant to the pressurizer model:

vi) For PWR transients where the pressurizer goes solid or completely drains the pressurizer behavior will require comparison against real plant or appropriate experimental behavior.

Discussion
See the response to Limitation O and "Item For Additional Justification Item ii". Dominion notes that the RETRAN 3-D pressurizer model has been explicitly approved for filling and draining events.

5.6.2 Qualifications and Restrictions (CONT)

5.6.2.2 RETRAN 02/MOD003-004 Restrictions

Section 3.0, Conclusions, of the RETRAN02/MOD003 and MOD004 SER (Reference 5.6-5) presents six restrictions for RETRAN02/MOD003 and MOD004 code versions, one of which (Number 5) was relevant to pressurizer modeling:

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

Discussion
Dominion RETRAN models do not use the wall heat transfer model for non-equilibrium volumes. Dominion RETRAN comparisons to plant transients show that adiabatic modeling of the pressurizer walls is adequate (see response to RETRAN02/MOD002 SER Limitation o).

5.6.3 Conclusion

The results of RETRAN comparisons to plant operational data and to other licensed transient analysis codes demonstrate that the non-equilibrium pressurizer model is adequate over the expected range of pressurizer conditions that occur in North Anna and Surry UFSAR non-LOCA events analyzed with RETRAN.

REFERENCES FOR SECTION 5.6

5.6-1 I. E. Idel-chik, "Handbook of Hydraulic Resistance, Coefficients of Local Resistance and of Friction," AEC-TR-6630, 1960

5.6-2 WCAP-12910, Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift", Westinghouse, May 1993.

5.6-3 Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

5.6-4 VEP-FRD-41-A, "VEPCO Reactor System Transient Analysis Using the RETRAN Computer Code", May 1985.

5.6-5 Letter from A. C. Thadani (USNRC) to R. Furia (GPU), "Acceptance for Referencing Topical Report EPRI-NP-1850-CCM-A Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004," October 19, 1988.

5.6-6 Letter from W. L. Stewart (VEPCO) to H. R. Denton (USNRC), "Virginia Electric Power Company, North Anna Power Station Units No. 2, Response to the Additional Request for Information Concerning Low Power Natural Circulation Testing," Serial No. 427A, August 25, 1983.

5.6-7 Letter, M.L. Bowling (VEPCO) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Supplemental Information on the RETRAN NSSS Model," Serial 93-505, August 10, 1993.

5.7 STEAM GENERATORS

5.7.1 Description

RETRAN case-specific input is used to adjust the steam generator tube heat transfer areas, metal volume and primary side water volume to reflect the desired amount of SG tube plugging.

The multi-node SG model was developed for studies where a detailed understanding of level response is needed. However, most UFSAR accident analyses can be adequately represented with a single node SG secondary model. Use of the multinode model in licensing applications is predicated on performing additional qualification of the RETRAN slip models as discussed in the Qualifications and Restrictions section below.

Like the pressurizer level instrument model, the SG level instrument model does not include the impact of changes in reference leg temperature that may occur as a result of changes in containment temperature. This phenomenon is accounted for in the instrument uncertainty calculations (see Section 5.2.8).

The SG low level trip setpoint (0 %) may not be reached in the multinode secondary model if the pressure increases during the course of the transient prior to trip. This behavior is real. This process measurement bias is accounted for in the CSA (Section 5.2.8).

In the single node SG model, the SG secondary side is represented by a single volume. The multi-node SG configuration calculates the SG NR level based on the actual differential pressure algorithm. The single node SG configuration uses a derived mass versus level correlation based on the steady state hot full power mass distribution. Basically, the design total mass at the level setpoint was used to estimate the mass at each end of the NR instrument range.

5.7.2 Qualification and Restrictions

As described above, the most significant approximation associated with the single node secondary side model is the loss of detail in the downcomer level response, which is significant for transients where low or high steam generator level protection is of importance. Dominion addresses this loss of detail with the following conservatisms:

- For loss of steam generator inventory events, no credit is taken for protective action on the low steam generator level coincident with steam flow / feed flow mismatch signal.
- A bounding value of the low-low steam generator level setpoint is assumed (typically 0% narrow range SG level span) which accommodates level channel measurement uncertainties with margin.

The RETRAN02/MOD002 Safety Evaluation Report (Reference 5.7-1), Enclosure 2 (Technical Evaluation Report-TER) Section II.C discussed general limitations of application of RETRAN02/MOD002. Those qualifications and restrictions that are applicable to the pressurizer model are discussed and evaluated in this section. The letter designations for the qualifications and restrictions are those of the TER.

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

Discussion

Dominion RETRAN models specify the use of the dynamic slip option on the primary side and zero slip on the secondary side of the steam generator (SG) tubes. However, two-phase flow is not normally encountered on the primary side during non-LOCA PWR transients. The exception is for steam line break, where the pressurizer may drain during the cooldown, and the upper head may flash, resulting in some carryunder to the upper plenum region as the head drains. The RCS pressure response obtained in Dominion steam line break analyses, including the effects of pressurizer and upper head flashing and drainage, is consistent with that obtained by vendor models as discussed in VEP-FRD-41-A (Appendix 1; Ref. 5.7-2).

Dominion does have a multi-node steam generator secondary model overlay that uses dynamic slip modeling. This model is not used in licensing calculations, but it is occasionally used in studies to confirm that the standard steam generator models are providing conservative results. The standard model features involve a single-node secondary side model and the associated heat transfer response and level-versus inventory correlations that are used to model low and low-low SG level reactor protection. The multi-node model treats the horizontal flow between the lower downcomer and tube bundle as bubbly flow.

Reference 5.7-3 presented comparisons between the multi-node and single-node SG versions of the model for a complete loss of load and for a 200%/minute turbine runback transient at full power. The response comparisons for pressurizer pressure and liquid volumes, RCS temperature, and steam pressure showed essentially identical responses for the two models. The most pronounced differences were in predicted changes in steam generator level and inventory, as expected. These results are reproduced below (Figures 5.7-1 through 5.7-13).

FIGURE 5.7-1

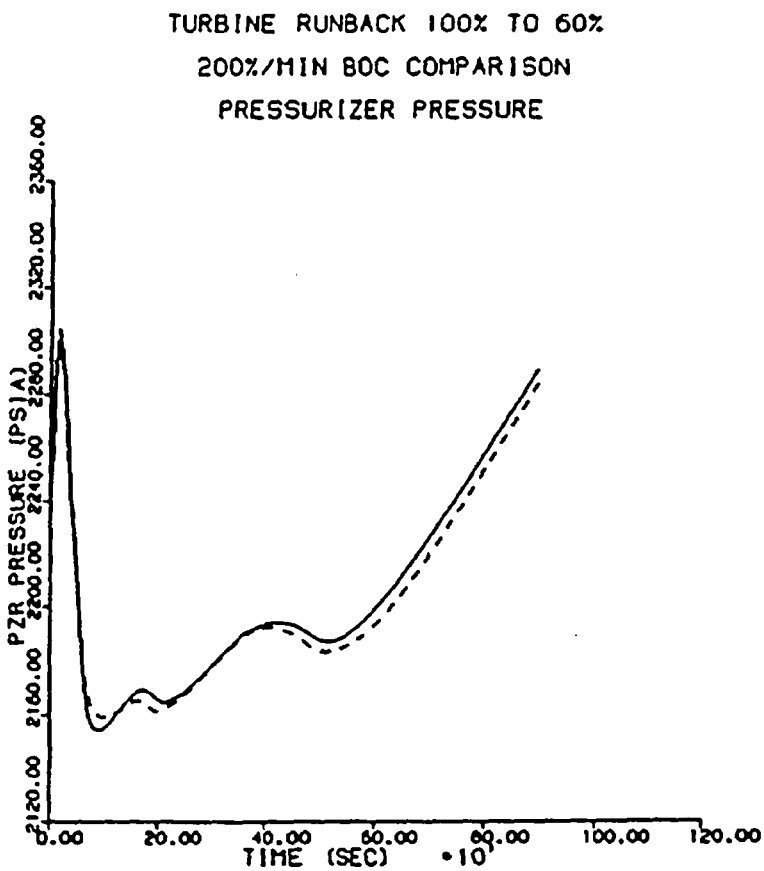


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FIGURE 5.7-2

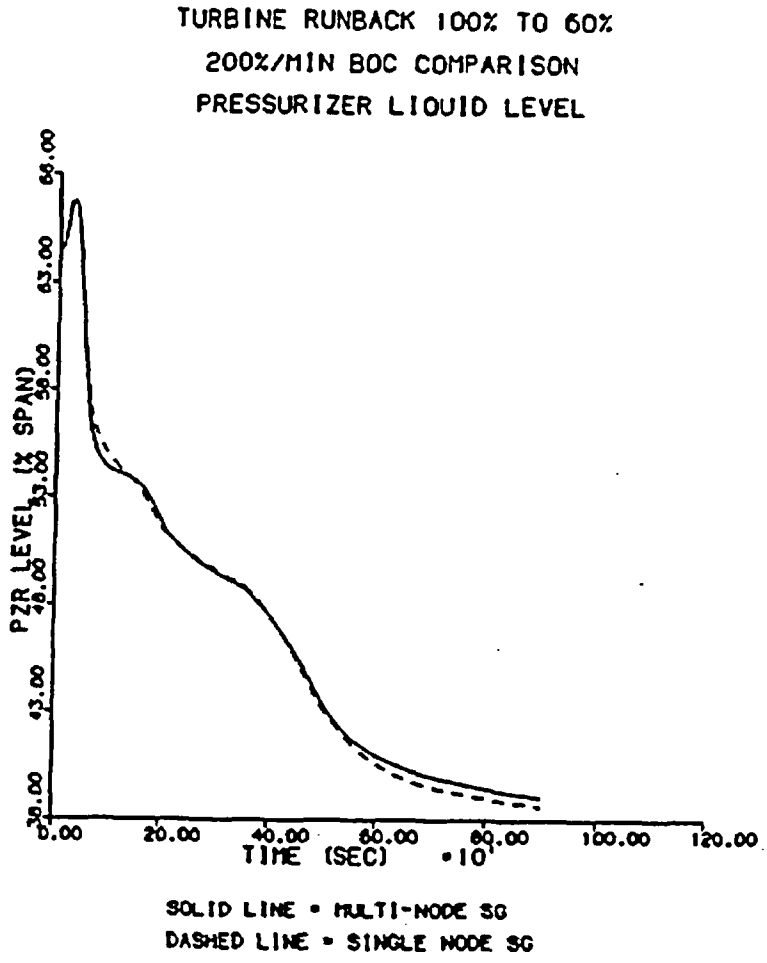

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FIGURE 5.7-3

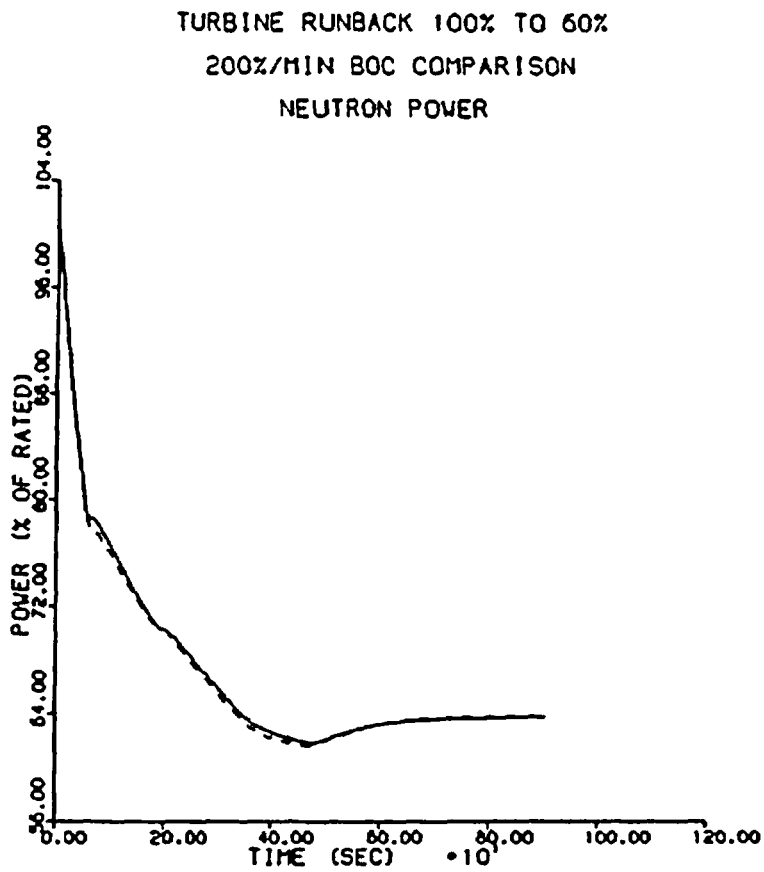
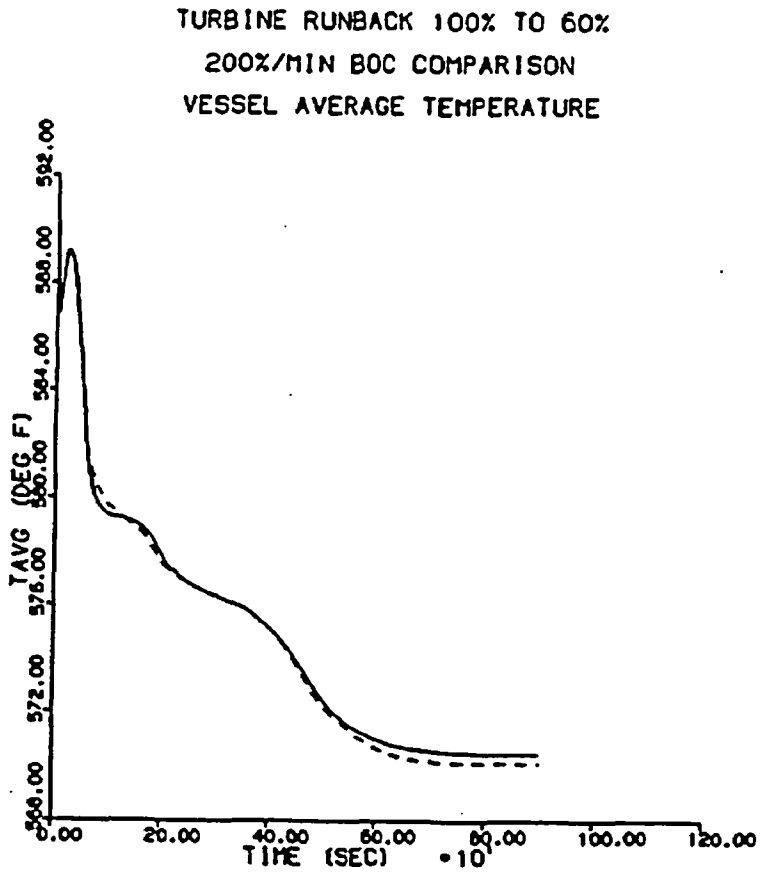

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
NEUTRON POWER
POWER (% OF RATED)
104.00
96.00
88.00
80.00
72.00
64.00
56.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG

FIGURE 5.7-4



1/22/93 13.44.40

FIGIURE 5.7-5

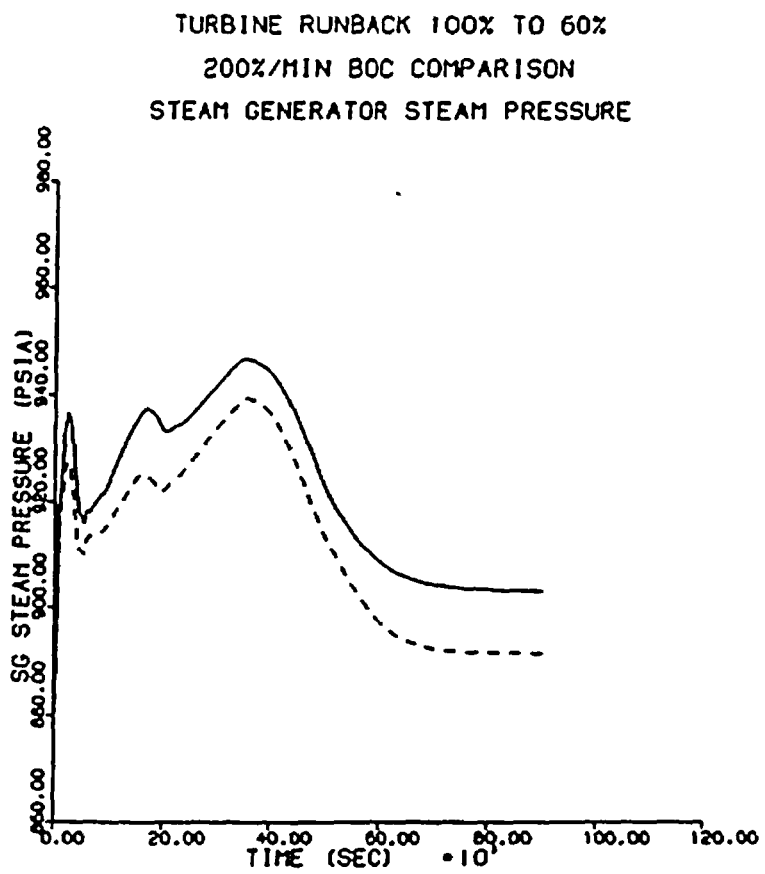


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FIGURE 5.7-6

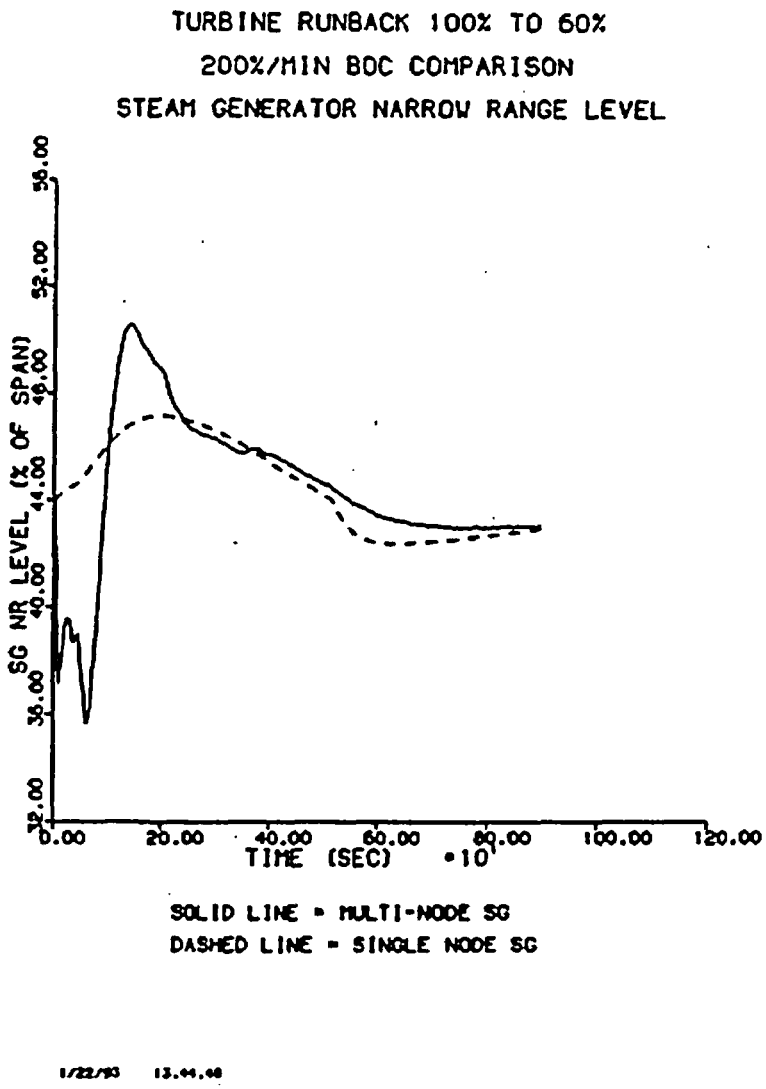

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FIGURE 5.7-7

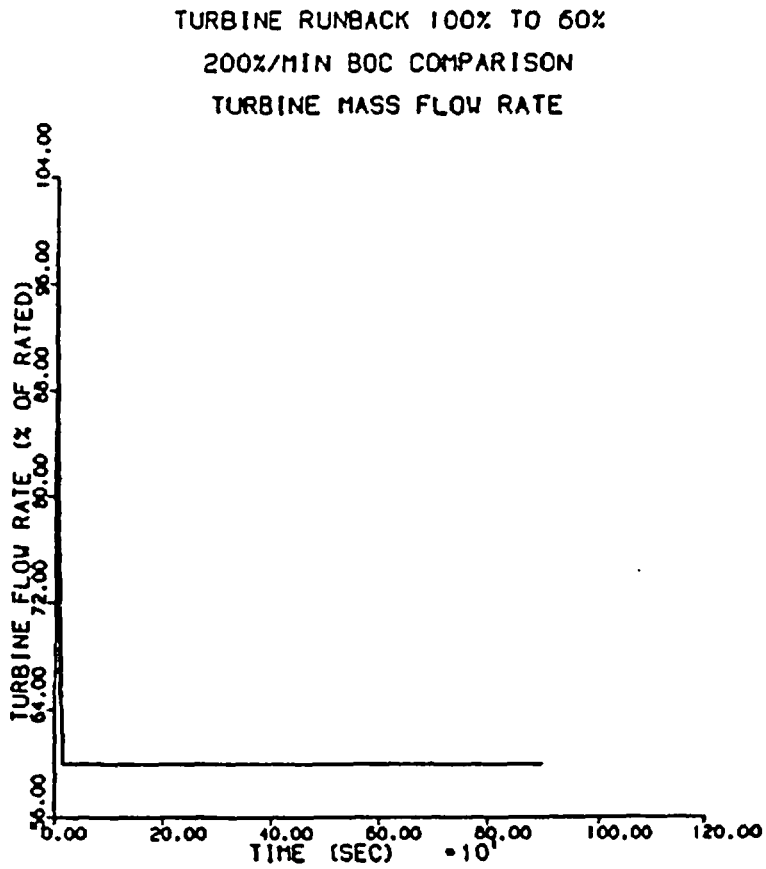

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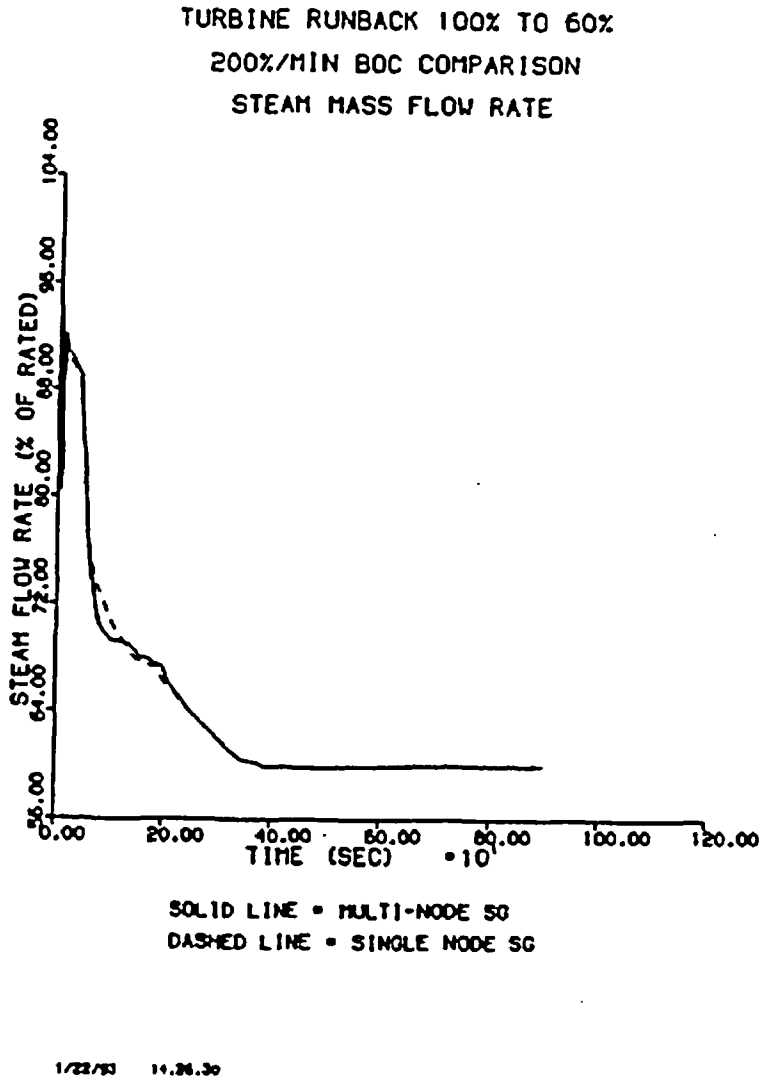
FIGURE 5.7-8



1/22/93 14.36.30

FIGURE 5.7-9

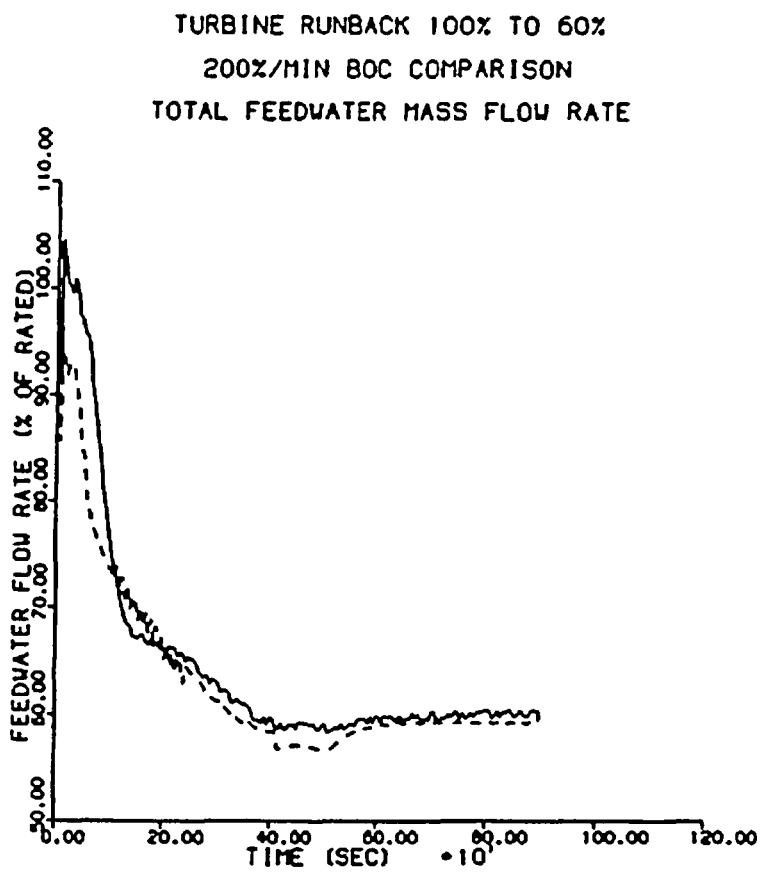


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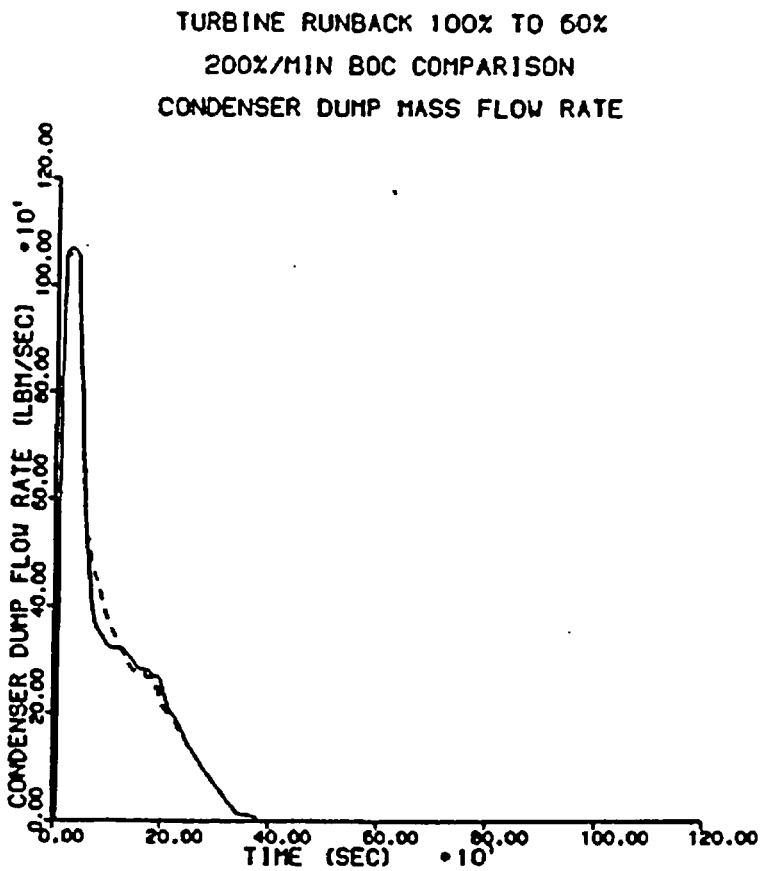
FIGURE 5.7-10



1/22/93 14.31.03

FIGURE 5.7-11

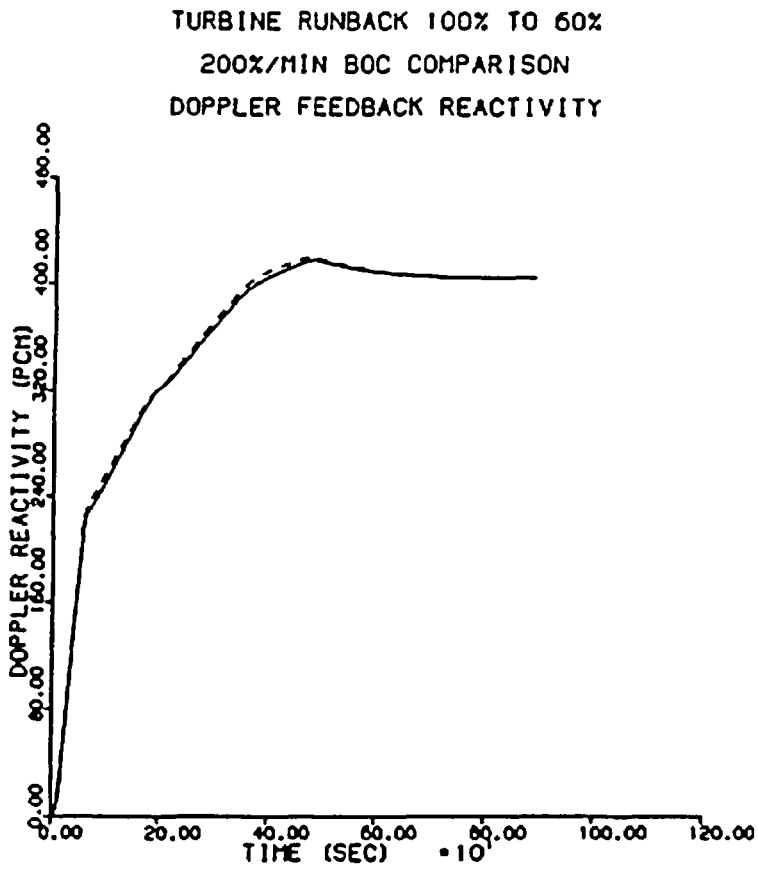


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FIGURE 5.7-12

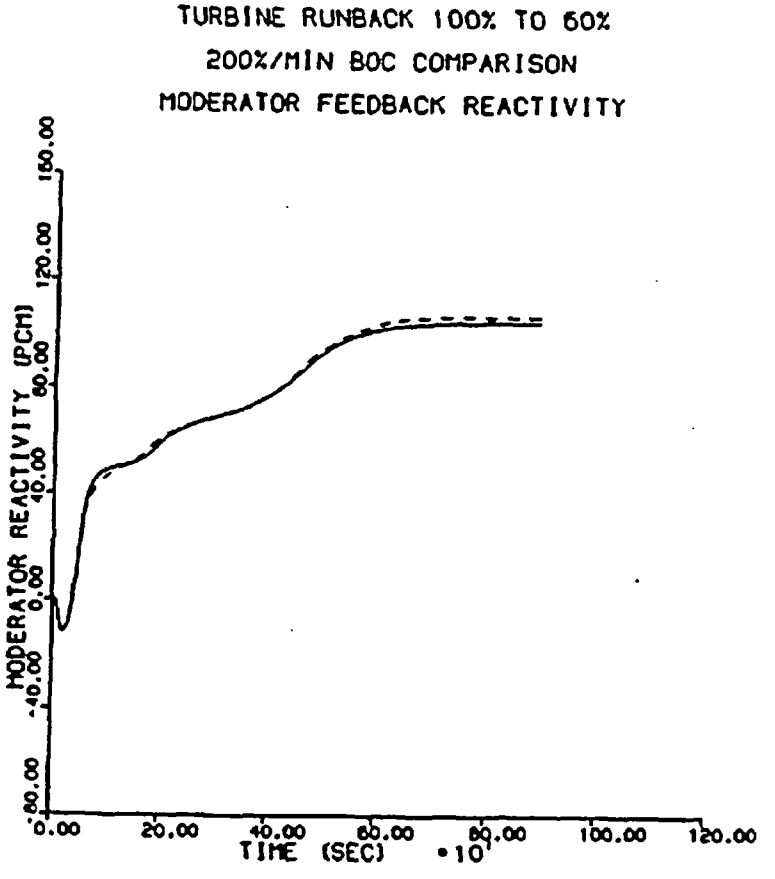
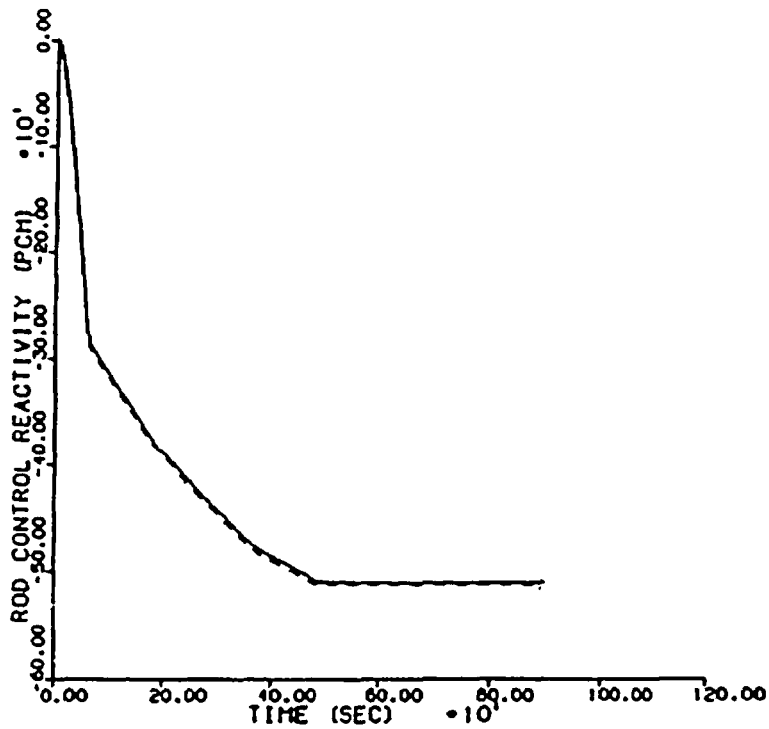

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
MODERATOR FEEDBACK REACTIVITY
MODERATOR REACTIVITY (PCM)
150.00
120.00
80.00
40.00
0.00
-40.00
-80.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG

FIGURE 5.7-13

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
ROD CONTROL REACTIVITY



SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG

1/22/93 13.44.40

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

Discussion
Refer to the response to Limitation h2, above.

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

Discussion
Dominion PWR RETRAN system models use heat transfer correlations in three areas:

- Reactor core conductors
- Primary (RCS) side of the steam generator tubes
- Secondary (steam) side of the steam generator tubes

For all non-LOCA accident analyses, the core heat transfer remains in the single-phase convection and subcooled nucleate boiling regions. The event that presents the most severe challenge to subcooled nucleate boiling on a corewide basis is the locked reactor coolant pump rotor event presented in Sections 15.4.4 and 14.2.9.2 of the North Anna and Surry UFSARs, respectively. For the locked rotor event, the heat transfer mode remains subcooled forced convection at the core inlet node and nucleate boiling at the mid core and top core node throughout the event.

Similarly, subcooled forced convection is the dominant heat transfer mode on the inside of the steam generator tubes for all non-LOCA events.

On the secondary (steam) side of the steam generator tubes, the heat transfer mode is typically saturated nucleate boiling (Mode 2) for non-LOCA transients. Exceptions occur when:

- a steam generator approaches dryout, such as for a large feedline break accident
- a steam generator blows down, as in the main steam line break event.
- there is no flow through the single-node secondary side of the steam generator, such as during a loss of load (turbine trip) with feedline isolation.

These cases will be addressed in turn.

For cases where significant steam generator dryout is anticipated, Dominion uses the RETRAN local conditions heat transfer option in conjunction with the single-node steam generator secondary side model. Dominion has performed analyses to evaluate the physical realism of the modeling results, including a steam generator tube noding sensitivity study. The behavior of the model is such that nucleate boiling heat transfer (RETRAN Mode 2) is predicted for nodes below the collapsed liquid level. For nodes above the collapsed level, the model predicts a rapid transition from single-phase convection to steam (RETRAN Mode 8).

For the steam line break calculation, Dominion uses a set of overlay cards to predict a conservatively large heat transfer coefficient on the secondary side, in order to maximize the RCS cooldown. This is done using control blocks.

For nodes below the collapsed liquid level, the overlay model applies a separate heat transfer coefficient to the secondary side of each steam generator conductor based on the maximum of the following, independent of which regime the RETRAN logic would pick:

- Rohsenow pool boiling
- Schrock-Grossman forced convection vaporization
- Thom nucleate boiling
- Chen combined nucleate boiling and forced convection vaporization
- Single phase conduction to steam (Dittus-Boelter)

This maximum coefficient represents the heat transfer for the "wet" heat transfer surface in the steam generator.

To better represent the variation of the film coefficient for the conductors at different elevations, a model was developed to calculate a collapsed liquid level and apply the maximum "wet" coefficient below this level and the forced convection to steam above this level. This provides a realistic and smooth transition in heat transfer capability as the steam generator inventory is depleted.

For cases with no flow calculated through the single-node secondary side (e.g., turbine trip with no condenser dumps and assumed feedwater line isolation at the time of turbine trip), the heat transfer on the entire secondary surface of the tubes will rapidly transition to forced convection vaporization with a very small heat transfer coefficient. This behavior is non-physical, because a significant portion of the tube bundle remains covered with two-phase mixture and would remain in the nucleate boiling regime. However, the results are conservative and Dominion's experience has been that this calculational anomaly only occurs for brief periods of time such that the key results (e.g., peak RCS pressure) are not significantly impacted.

In summary, the limitations of RETRAN's regime-dependent heat transfer models are considered in Dominion licensing analyses. Appropriate assumptions and approximations are made to ensure that the accident analyses are conservative.

N3) While FRIGG tests comparisons have been presented for the dynamic slip option the issues concerning the Schrock-Grossman round tube data comparisons should be resolved before the dynamic slip option is approved. Plant comparisons using the option should also be required.

Discussion
Refer to the response to Limitation h2, above.

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

Discussion
Dominion PWR RETRAN models use bubble rise in the pressurizer, reactor vessel upper head, and steam generator dome regions [Table 4.3].

The single-node steam generator secondary model is initialized with a low mixture quality so that the steady-state initialization scheme selects a large bubble rise velocity. The initialization models complete phase separation as a surrogate for the operation of the mechanical steam separators and dryers in the steam generators.

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

Discussion
As discussed in the response to Limitation m, Dominion restricts use of the local conditions heat transfer model to loss of secondary heat sink events. The model predicts a rapid transition from nucleate boiling to single-phase convection to steam on the secondary side as the tube bundle dries out.

Nodal sensitivity studies were performed to show that the default tube bundle noding provides an adequate representation of the primary to secondary heat transfer. The single-node secondary side is initialized with a low mixture quality. As a result, a high bubble rise velocity is calculated by the steady state initialization routine. This drives the RETRAN calculated mixture level to the collapsed liquid level and conservatively maximizes the rate of tube bundle uncovery as the inventory is depleted. The fluid condition on the inside of the tubes remains single phase, and thus the restriction is met.

A loss of normal feedwater with delayed (600 seconds) auxiliary feedwater initiation was modeled for the noding sensitivity study The results using 10 conductor nodes vs 20 conductor nodes per steam generator were compared. These are presented in Table 5.7-1.

TABLE 5.7-1
Time Sequence of Events
Loss of Normal Feedwater
[Local Conditions Heat Transfer Noding Study]

lchtbase = 10 node model
lchtqual = 20 node model

Key Parameter Predictions

lchtbase	lchtqual	Parameter Description
2349.6	2349.5	Maximum pressurizer pressure, psia
1490	1230	Time of maximum, sec
-568	-571	Minimum net core reactivity, pcm
30	30	Time of minimum
1060.8	1060.8	Maximum SG "A" steam pressure, psia
50	50	Time of maximum, sec
550.1	550.0	Minimum loop "A" actual T_{avg}, °F
2010	2020	Time of minimum, sec
850.5	851.3	Maximum pressurizer liquid volume, ft^3
30	30	Time of maximum, sec
25.58	25.57	Time of reactor trip on lo-lo SG NR level, sec
1232	1231	Time of first activation of pressurizer PORV no. 1, sec

REFERENCES FOR SECTION 5.7

5.7-1 Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

5.7-2 VEP-FRD-41A, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code," May 1985 (Appendix 1).

5.7-3 Letter, M.L. Bowling (VEPCO) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Supplemental Information on the RETRAN NSSS Model," Serial 93-505, August 10, 1993.

5.8 MAIN STEAM SYSTEM

5.8.1 General

The main steam system model includes the following components:

- Main steam piping
- Main steam safety valves
- Atmospheric steam relief valves (PORVs)
- Steam line non-return valves
- Main steam isolation valves
- Condenser steam dump system

5.8.1.1 Main Steam Safety Valves

The main steam safety valve models assume a "pop and blow" characteristic, i.e. the valve opens rapidly after steam pressure reaches the nominal setpoint + Technical Specifications setpoint tolerance. Upon depressurization, the valve begins closing at the nominal setpoint and is completely closed at a pressure at the nominal setpoint minus blowdown.

The contraction coefficient for each of the safety valves (pressurizer and main steam) is calculated based on the assumption that the rated flow is achieved at a pressure corresponding to the setpoint plus tolerance plus accumulation. This results in a conservatively low calculated relief rate.

The model includes the effects of dynamic pressure loss terms in the MSSV inlet piping to address the concerns raised in NRC Information Notice 97-09 (Reference 5.8-1).

If specific analyses should require water relief from the main steam safety valves, the entire steam line and valve model would need further review.

5.8.1.2 Atmospheric Steam Relief Valves (PORVs)

Although the valves are actually attached to the steam line, they are modeled as connected to the steam generator. This provides a more stable execution, especially when the MSIVs are closed. The atmospheric relief valve is modeled as a critical flow junction with a valve. The junction area was calculated by determining the saturated steam critical mass flux for the isoenthalpic model at the nominal set pressure and dividing this mass flux into the design relief capacity.

5.8.1.3 Steam Line Non Return Valves

The steam line non return valves are modeled implicitly by large reverse loss coefficients in the steam lines.

5.8.1.4 Main Steam Isolation Valves

The main steam isolation valve cannot be reopened after closing. Low steam line pressure or low-low Tavg coincident with high steam flow will initiate main steam line isolation. The logic requires 2/3 coincidence from each of these signals; i.e., two of the three loops must exceed their threshold.

5.8.1.5 Condenser Steam Dump System

Two signals provide permissives to allow the steam dump control system to operate. Either a sudden load loss or a turbine trip will provide a signal to open the arming solenoids to allow air to be admitted to the actuator. A low-low Tavg signal will cause the air to be dumped from the actuator, thus causing the steam dumps to close, or not to open. The typical encounter with the low-low Tavg blocking signal is after the dumps are open and the RCS has been overcooled for some reason.

If the temperature error signal exceeds a certain threshold the steam dump valves are provided signals to open fully, i.e., trip open. The threshold depends on whether or not the turbine has tripped.

Modulation of the steam dump demand signal is also provided via the load rejection and turbine trip controllers depending on whether or not a turbine trip is present. If the turbine trip signal is present, the valves are modulated such that the dump capacity is a roughly linear function of the difference between the dynamic reactor coolant system average temperature and the programmed no-load temperature. If the turbine trip signal is absent, the valves are modulated to vary the relief capacity as a roughly linear function of the difference between the reactor coolant system average temperature and the programmed load-dependent reference temperature. This capability is achieved via use of the RETRAN control system models.

The condenser steam dumps are tandem trim valves that contain a pilot valve and a main plug. Upon receiving a signal to open, only the pilot valve moves so as to vent a balancing chamber in order to equalize the chamber's pressure with the downstream pressure. During this period there is no flow through the valve. The main plug then opens to allow flow.

The condenser dump valve model is a best-estimate model and is normally disabled for UFSAR transient analyses.

5.8.2 Qualification and Restrictions

See the discussions in Section 5.7 for the Steam Generator models. Additionally, as part of the original (Rev. 0) review process, VEPCO performed comparisons of the 1-loop RETRAN model results to results obtained with the Westinghouse LOFTRAN code (Ref. 5.8-2). Three transients were examined: a spurious reactor trip, a spurious turbine trip and a flow coastdown event. Several parameters characteristic of secondary side performance were compared. The results showed very similar behavior. The Ref. 5.8-2 analysis (CONTAINS WESTINGHOUSE PROPRIETARY INFORMATION) is shown in Appendix 4.

REFERENCES FOR SECTION 5.8

5.8-1 USNRC, Information Notice 97-09, "Inadequate Main Steam Safety Valve (MSSV) Setpoints and Performance Issues Associated with Long MSSV Inlet Piping," March 12, 1997.

5.8-2 Letter from W. L. Stewart, (VEPCO) to H. R. Denton, USNRC, "VEPCO Reactor System Transient Analyses," August 24, 1984, Serial No. 376A.

5.9 MAIN FEEDWATER SYSTEM

The main FW flow and temperature are pure boundary conditions. Hence, the normal response of the controller and system components must be approximated by the user. For licensing transients, conservative assumptions are made.

Main FW isolation is modeled by a linear reduction in main FW flow.

Two alternatives are provided for the control of the fill flux: 1) a simple FW controller based on the level and steam-feedwater mismatch errors and 2) a FW matches steam assumption. The base model is set up to use the FW controller function.

Note that the FW matches steam assumption is appropriate for the pre-trip portion of transients and is not normally used post-trip.

The default FW controller function model is considered to be adequate for slow to moderate secondary transients where the FW control valve operates in its normal control range.

The feedwater enthalpy model is included to provide the proper steady state endpoints for changes in power level as well as approximate dynamics of main FW enthalpy changes. For many transients, the "FW enthalpy follows load" function is often turned off and the FW enthalpy held at the full load value.

In addition, the purge time characteristic of the volume from the AFW connection to the SG is included in this model. This will allow the FW junction enthalpy to adjust automatically from the main FW value to that of the AFW once the main FW inventory in each line is purged from the main feedwater piping by the AFW flow.

5.10 AUXILIARY FEEDWATER SYSTEM

The auxiliary feedwater flow and temperature are pure boundary conditions. Hence, the normal response of the system components must be approximated by the user. For UFSAR transients, appropriately conservative assumptions are made.

Both the main feedwater and auxiliary feedwater enter through the same fill junction(s). RETRAN will typically add a bias to the main FW enthalpy in order to calculate the required thermodynamic enthalpy for the feedwater junction at steady state. The value of this "fill enthalpy bias" is found in the 'JUNCTION DATA ACTUALLY BEING USED' RETRAN output edit. This same bias will also be added to the AFW enthalpy. Therefore if the user does not wish the "fill enthalpy bias" to be added to the AFW enthalpy, he must adjust the specified AFW enthalpy accordingly.

The AFW flow tables are not applicable to any particular transient.

5.11 TURBINE EHC SYSTEM AND AUTOMATIC RUNBACK

The EHC model assumes that total steam flow and first stage pressure are linearly related, i.e., steam flow can be used as a surrogate for first stage pressure. The model also assumes that the response of the EHC is essentially instantaneous. In other words, there is an imperceptible delay between changes in demand and changes in steam flow. Experience with other plants has shown this to be a reasonable assumption.

The model represents the EHC in automatic. Straightforward modifications can be made if manual operation is to be represented. Turbine runback is actuated on approach to the OTΔT and OPΔT reactor trips. The reduction of turbine load causes a decrease in reactor power and increases the margin of operation to unsafe OTΔT and OPΔT conditions that would require a reactor trip.

Automatic turbine runback is activated on OTΔT and OPΔT at a setpoint which is less than the corresponding reactor trip setpoint by design. The runback is actuated if the setpoint is exceeded in 2 out of 3 loops.

When automatic runback trips are activated, the load rejection operates on a cycle, running back at a constant rate for time interval t_1, stopping for time interval t_2 sec, then continuing in a similar fashion until the OTΔT or OPΔT trip condition is cleared or zero turbine load is reached.

The turbine flow is limited by the pressure at the stop valve inlet. As pressure increases, the maximum flow increases.

5.12 Safety Injection System

The initiating functions for safety injection which are modeled are shown in Table 5.12-1. Only the high head safety injection pumps are modeled, as these are of primary interest for non-LOCA transient analyses. Typical analyses values for the setpoints and delay times are shown.

The base models contain a pressure dependent table of injection flow vs RCS pressure for a single train of high head safety injection (1 pump). The flows are based on a conservative model of pump head degradation, injection line hydraulic resistance and emergency power frequency degradation which minimizes the injection flow rates, since this is the assumption of interest for most safety analyses. For cases where maximum flow is of interest (e.g. a steam generator overfill study following tube rupture), the base model tables may be overridden with more appropriate input.

For modeling steam line break transients, the user has access to an overlay containing a RETRAN control block representation of the transport and mixing of soluble boron from the injection stream throughout the reactor coolant system (see Section 5.13). This feature is important for transients where reactivity phenomena are significant, such as main steam line break and may be added to the base deck when needed.

Table 5.12-1
SI Initiation Functions:
[Typical Safety Analysis Setpoints]

Delay Trip Signal	Setpoint	sec
Low-low pressurizer pressure, psia	≤1610	13
High steamline Δ pressure, psi *	<150	13
High steam flow coincident with low-low Tavg		
- steam flow, % of rated	**	15
- low-low Tavg, °F	≤539	
High steam flow coincident with low steamline pressure		
- steam flow, % of rated	**	13
- steamline pressure, psia	459	

* Pressure in any one steamline more than 150 psi less than that in the other two steamlines (North Anna). The Surry system compares steamline pressure to the main steam header pressure.

** Steam flow setpoint is 40% of nominal for turbine load less than or equal to 20% nominal and increasing linearly with turbine flow to 110% of nominal steam flow at full nominal turbine flow

5.13 REACTOR KINETICS

5.13.1 Reactivity Feedback

Reactivity feedback in the RETRAN models is computed entirely by control systems instead of using the RETRAN point kinetics model input and tables. Each reactivity component is computed separately in units of pcm, summed into a timestep's net reactivity, also in pcm, and converted to dollars ($) before input to the point kinetics algorithm.

The following reactivity components are modeled:

1. Doppler feedback
2. Moderator feedback
3. Soluble boron
4. Control rod withdrawal
5. Normal rod control
6. SCRAM

The user can edit each of the reactivity components, as well as the total, in pcm and can bias the initial output of each component to zero.

5.13.2 Reactivity Model Inputs

The following control inputs and control blocks are available to the user :

- Delayed neutron fraction, βeff
- Time for SCRAM rods to reach dashpot
- Total SCRAM worth
- Zero bias for boron reactivity
- Transient bottom core boron concentration
- Transient mid core boron concentration
- Transient top core boron concentration
- Doppler weighting factor
- Moderator weighting factor
- DTCREF
- Zero bias for Doppler
- Zero bias for moderator
- Zero bias for control rods

5.13.3 Reactivity Model Outputs

The reactivity model control blocks provide outputs of the following:

- Core average fuel temperature, °F
- Doppler reactivity, pcm
- Core average moderator density, lbm/ft^3
- Core average moderator temperature, °F
- Moderator reactivity, pcm
- Rod withdrawal reactivity, pcm
- Rod control plus SCRAM reactivity, pcm
- Net reactivity, pcm
- Net reactivity, $

Specification of a reactivity bias is for editing purposes only and will not impact the RETRAN predictions since the point kinetics algorithm adjusts the initial net reactivity to zero.

5.13.4 Decay Heat

The decay heat calculation is based on the 1979 ANS Standard. The decay heat calculated by RETRAN will deviate slightly from that calculated using the ANS Standard mainly due to the way RETRAN models the termination of fission due to a reactor trip. Whereas the ANS Standard assumes the fission process ceases precisely at some time t, RETRAN more correctly simulates a reactor trip through the response of the fission process to a negative reactivity insertion over time. Prompt neutrons will decay during the trip effectively simulating a power rampdown instead of a guillotine cessation of the fission reaction.

In RETRAN, decreasing the prompt neutron lifetime, lp, or the rate and/or magnitude of negative reactivity insertion will decrease the predicted RETRAN decay heat energy. However, the RETRAN input format does not allow for lp alone to be modified. Rather, the user provides the quotient of the βeff divided by lp. Decreasing the value of lp input effectively increases βeff and therefore decreases the amount of reactivity inserted.

Another difference between the ANS standard and the RETRAN decay heat calculation is that, for the 1979 ANS Standard, RETRAN ignores the effects of neutron capture. This effect can only be accommodated in RETRAN by use of the decay heat multiplier option (either through supplying a constant multiplier or defining a time-dependent multiplier via control blocks).

Finally, RETRAN uses a different correlation than that provided in the 1979 ANS Standard to compute the actinide contribution to decay heat (i.e., the contributions due to U-239 and Np-239). The RETRAN actinide correlation is that of Branch Technical Position APCSB9-2 (see Volume 1 of the EPRI RETRAN-02 Code Report, EPRI NP-1850-CCM-A, Rev. 4). The RETRAN input of the breeding ratio UDUF (i.e., the number of Pu-239 atoms produced per U-235 atoms fissioned) only impacts the calculation of the actinide contribution. The greater the value of UDUF, the higher the predicted decay heat fraction. In the 1979 ANS Standard, the actinide correlation parameter that

corresponds to UDUF is the parameter R, the number of U-239 atoms produced per second per fission at the time of shutdown.

The RETRAN model uses the following assumptions in the calculation of decay heat:

Operating period, days:	1,500
Load factor:	100%
Q, MeV/fission:	190
Decay heat fissioning nuclides:	U-235 only
Actinide component:	APCSB correlation with UDUF = 0.77
No neutron capture component	

5.13.5 Direct Moderator Heating

A direct moderator heating fraction of 0.026 assumed for all three core sections.

5.13.6 Doppler Feedback

Doppler feedback is based on a correlation developed from detailed studies with Dominion's approved PDQ models. The RETRAN model correlation has a core average fuel temperature, T_f, component, DTCTF, and a burnup component, BURNMP.

The DTC correlation is qualified over the range of core design DTC limits for North Anna and Surry and is described by the following equation:

$$DTC(pcm/°F) = DTC_{Tf} * BURNMP * WF$$

where

DTC_{Tf}, the fuel temperature dependence, equals $A*T_f^{0.5} + B*T_f + C$

T_f is the effective core average fuel temperature in °F and A, B, and C are correlation coefficients

BURNMP, which models burnup changes, equals DTC_{ref}/DTC_{Tf547}

DTC_{ref} is the reference DTC at the burnup of interest at hot-zero-power with 2000 ppm boron (pcm/°F)

DTC_{Tf547} is the solution to the above DTC_{Tf} equation at 547 °F.

WF is the user supplied weighting factor term that allows the user to adjust the design information to bound specific Doppler defects.

The Doppler feedback can be adjusted to a target DTC at a given fuel temperature by changing the weighting factor.

5.13.7 Reactivity Parameter Selection

The selection of specific reactivity parameter values for accident analysis is based on ensuring the predicted response is bounding for the range of valuess realized over the entire burnup range for currently operating reload cores. The "bounding parameter" approach, originally documented by Westinghouse in WCAP-9272 (Reference 5.13-1), has been adopted and applied by Dominion as described in detail in Reference 5.13-2.

5.13.8 Qualifications and Restrictions

The RETRAN02/MOD002 Safety Evaluation Report (Reference 5.13-16), Enclosure 2 (Technical Evaluation Report-TER) Section II.C discussed general limitations of application of RETRAN02/MOD002. Those qualifications and restrictions that are applicable to the core kinetics model are discussed and evaluated in this section. The letter designations for the qualifications and restrictions are those of the TER.

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

Discussion
The point kinetics approximation is used in the Dominion RETRAN model, consistent with standard industry safety analysis practice. Reactivity effects are modeled using standard fuel and moderator temperature coefficients and control bank worths which are shown to be bounding for Dominion cores using static core physics models which account for full 3-D effects.

Most non-LOCA transients do not involve significant temporal variations in the core power distributions, and industry experience over many years has shown the point kinetics approximation to be valid for this type of accident. Two notable exceptions are the control rod ejection and main steam line break events.

For the control rod ejection event, Dominion uses a point kinetics model to calculate the core average power response. The Doppler feedback is calculated using a spatial power weighting factor that is a function of the radial power peaking factor in the vicinity of the ejected rod, which is calculated using static neutronics calculations. Local power peaking is also calculated via static methods. The power peaking and core average time dependent power response are then used in conjunction with a conservative hot spot fuel pin model to calculate the limiting local fuel thermal response. Dominion's rod ejection methods have been benchmarked against full 3-D space-time kinetics calculations and shown to be conservative in VEP-NFE-2-A [Reference 5.13-3].

Dominion's methodology for steam line break is described in Sections 5.2.3.4 and 5.2.3.5 of VEP-FRD-41-A Rev. 0 [Appendix 1; Reference 5.13-4]. Asymmetric reactivity effects associated with the cold leg temperature imbalance and the assumption of a stuck control rod are modeled by breaking the core into two azimuthal sectors and providing an empirical weighting factor to the moderator temperature coefficients in the two sectors. Fluid mixing between the two regions is modeled based on scale model mixing tests performed by Westinghouse.

Power reactivity feedback is also modeled with an empirical curve of reactivity feedback versus heat flux. The validity of these curves is checked for every reload by static neutronics methods that show that the magnitude of the post-trip return to power predicted by RETRAN is conservatively high. Local power peaking is also calculated using static neutronics methods. Core DNB performance is calculated in a separate code (e.g. COBRA or VIPRE).

This approach for using a combination of point kinetics and static 3-D neutronics calculations for analyzing the steam line break event is similar to that used by fuel vendors (see for example References 5.13-5 through 5.13-7).

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

Discussion
Dominion meets this restriction. Dominion initiates low power events, such as rod withdrawal from subcritical, and the hot zero power rod ejection event from a critical condition with a low initial power level representative of operation within the range of operability for the source range nuclear instrumentation channels. For the "zero power" steam line break, the models are initialized in the same way, and then the design shutdown margin is simulated by a rapid negative reactivity insertion coincident with the break opening.

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

Discussion
A generalized boron transport model was added to RETRAN02/MOD005 [Reference 5.13-8]. However, Dominion uses the RETRAN control system to model boron transport in the reactor coolant system for steam line break analyses.

During initial steamline break model development, RETRAN's general transport model was considered but not selected. The primary reason this option was not chosen was that the general transport model uses the default assumption of perfect mixing. Non-mixing regions like pipes cannot be conveniently modeled with a delay-type of behavior. The user may adjust mixing by changing the junction efficiency with a control system. However, this results in just as many control system cards devoted to mixing efficiency calculation as a control block based, full-transport model. Therefore, boron transport is modeled with a control system as in previous analyses. The general modeling philosophy is consistent with that described in Figure III-12 of Reference 5.13-9, which was submitted to support the original VEP-FRD-41 review. However, the model in Reference 5.13-9 assumed a constant reactor coolant system flow rate. The model was made more robust by incorporating variable transport delays and a dynamic plenum mixing model as described below, so that variable RCS flows are now handled accurately.

The boron transport model is broken into four major parts: 1) Refueling Water Storage Tank (RWST) to Boron Injection Tank (BIT); 2) the BIT; 3) BIT to the Reactor Coolant System (RCS); and 4) the RCS.

BIT Mixing Model

The BIT mixing model begins with the same basic equations as the RCS mixing model. The model makes the approximation that the density of the BIT is constant and is also equal to the density of the incoming fluid.

Following are the mixing region equations:

$$\frac{dC}{dt} = w_i c_i - w_o c_o$$

$$\frac{dC}{dt} = \frac{Mdc}{dt} + \frac{cdM}{dt}$$

$$\frac{dc}{dt} = \frac{w}{M}(c_i - c_o)$$

$$c(t) = \int \frac{dc}{dt} + c_o$$

C = Volume Mass M * Concentration c (lbm-ppm)
c_i = concentration of fluid entering volume, ppm
c_o = concentration of fluid leaving volume, ppm
w_i = mass flow into volume, lbm/sec
w_o = mass flow out of volume, lbm/sec

The first equation states that the rate of change of the mass times the concentration is equal to the mass flow rates in and out times their respective concentrations. The second equation expands the large C derivative into its constituents. The dM/dt term in the second equation is assumed to be zero and w_i is assumed to be equal to w_o. The third equation is formed by combining the first two with dM/dt = 0. The integral of dc/dt provides the dynamic concentration out of the BIT.

By assuming that the density of the BIT and the incoming fluid are equal, the w/M term is equal to the volumetric flow divided by the volume. The equations above are represented with the appropriate control blocks.

BIT to RCS Transport

The transport time through the BIT to RCS piping is calculated in several pieces: the common BIT to SI header delay, and the individual delays from the header to each cold leg. A DIV control block divides the BIT to HDR volume by the total flow rate. The transport time is then used as input to a DLY control block. The same function is performed for each of the header-to-loop segments. The fluid is assumed to be at an initial boron concentration of zero ppm.

RCS Boron Transport

The RCS is broken into several regions for boron transport:

1) the cold leg between the SI point and the vessel (DELAY)
2) the downcomer and lower plenum (MIXING)
3) each core section (DELAY)
4) core bypass (DELAY)
5) the outlet plenum (MIXING)
6) the hot leg, SG tubes, loop seal, RCP, and cold leg between the RCP and SI point. (DELAY)

The model used to represent the transport through each region is noted in parentheses above. The upper head concentration is assumed to be zero for the duration of the transient.

The technique used in each "DELAY" region is as follows:

1) Total "boron flowrate" entering the region is computed by summing the inlet fluid flows times their respective boron concentrations.
2) Total fluid flow entering the region is computed by summing the inlet fluid flows.
3) The total "boron flowrate" is divided by the total fluid flowrate to get a mixed boron concentration.
4) The masses of the volumes in the transport region are summed.
5) The total mass is divided by the total fluid flow to get the transport delay for the region.
6) The mixed boron concentration is propagated to the next region using the transport delay.

The technique used in each "MIXING" region is as follows:

1) The net "boron flowrate" in a region is computed by summing the inlet and outlet fluid flows times their respective boron concentrations.
2) This represents the rate of change of region mass times concentration (dC/dt) which is then integrated to determine C(t).
3) The concentration (c(t)) is then calculated by dividing (C(t)) by the region mass (M).

For the steamline break event, the peak core heat flux is sensitive to the timing of the initial boron increase in the core (i.e., the transport delay from the safety injection system to the core inlet) and is not sensitive to the exact shape of the boron buildup curve. Core inlet boron is only a few ppm at the time of peak heat flux. Dominion's model and vendor models predict comparable times for the introduction of boron to the core as shown in benchmark calculations.

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

Discussion

Control rod motion in the Dominion RETRAN point kinetics models is simulated by a reactivity input calculated from a time-dependent control bank position and a function generator containing integral bank worth versus position. For cases with automatic rod control simulated, the bank worth model is typically associated with the D-control bank only, which is the only bank in the core at or near full power.

For cases with reactor trip, the integral worth assumed is that associated with all control and shutdown banks at the power dependent insertion limit, less the most reactive control assembly in the core, which is assumed not to insert. The shape of the integral worth curve is based on a conservative bottom-skewed power distribution which delays the reactivity effects. This integral worth curve is checked for every reload core.

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

Discussion

The Dominion PWR RETRAN models do not use the subcooled void model to calculate the neutronic feedback from subcooled boiling region voids. Dominion models use a moderator temperature coefficient except for the steamline break event, which applies an empirical curve of reactivity feedback versus core average power. This curve is validated as conservative on a reload basis using static, 3-D, full-core neutronics calculations with Dominion's physics models. Dominion experience has indicated that the calculated DNBR's for the limiting steamline break statepoints show a weak sensitivity to the effects of void reactivity. The profile blending algorithm approved for RETRAN-02 MOD003 resolved this limitation [Reference 5.13-11, page 29].

RETRAN 02/MOD005.0 Restrictions

The RETRAN02/MOD005.0 Generic SER (Reference 5.13-21), Section 4.0, Staff Conclusions, contained the following conditions of use:

1. *The user must justify, for each use of the ANS 1979 standard decay heat model, the associated parameter inputs, as discussed in Section 2.1* of this SER.*

*Typo in the SER. Should have referenced Section 2.2.

Section 2.2 of the RETRAN-02 MOD005.0 SER specifies the following parameter inputs:

a. *power history*
b. *fission fraction*
c. *energy per fission of each isotope*
d. *neutron capture in fission products by use of a multiplier*
e. *production rate of 239 isotopes*
f. *activation decay heat other than 239*
g. *delayed fission kinetic modeling*
h. *uncertainty parameters*

Discussion
The Dominion RETRAN models use the following assumptions in the calculation of decay heat:

- An operating period of 1,500 days with a load factor of 100% is input to the Dominion RETRAN models.
- The model assumes 190 MeV/fission. The reduction of the Q value to 190 MeV/fission from the default RETRAN value of 200 MeV/fission is conservative since, in the 1979 ANS Standard, decay heat power is inversely proportional to Q.
- There is no neutron capture component.
- Decay heat fissioning is solely from U-235. The assumption that all decay heat is produced from U-235 fissioning nuclides is conservative.
- The RETRAN actinide correlation is that of Branch Technical Position APCSB9-2 [References 5.13-17 and 5.13-18]. The RETRAN input of the breeding ratio UDUF (i.e., the number of Pu-239 atoms produced per U-235 atoms fissioned) is 0.77 and only impacts the calculation of the actinide contribution. The greater the value of UDUF, the higher the predicted decay heat fraction.
- A value of 1.0 is input for the RETRAN model for the decay heat multiplier.

The results of a RETRAN calculation with the 1979 decay heat model and the assumptions listed above were compared to a vendor calculated decay heat curve based on the 1979 ANS standard with 2-sigma uncertainty added. The results indicated that the decay heat fraction calculated with RETRAN is higher than the vendor calculated decay heat. Therefore, the Dominion application of the ANS 1979 standard decay heat model is conservative.

3. *Because of the inexactness of the new reactivity edit feature, use of values in the edit either directly or as constituent factors in calculations of parameters for comparisons to formal performance criteria must be justified.*

The editing feature provided in RETRAN 02/MOD005.0 is not used as a quantitative indicator of reactivity feedback and is not used to report analysis results.

Additional Discussion of Doppler Model

In Reference 5.13-19, the NRC asked for additional information regarding the Doppler Reactivity Feedback model described in Reference 5.13-12. The response, provided in Reference 5.13-20 is included here for completeness.

2. *Doppler Reactivity Feedback (page 8 of the submittal dated August 10, 1993)*

a. *The Doppler reactivity feedback is calculated by VEPCO's correlation of Doppler reactivity as a function of core average fuel temperature and core burnup. Please provide a technical description of how this correlation is derived, including the codes and methods used. Discuss any limitations or restrictions regarding the use of this correlation.*

b. *Discuss the method of calculation and application of suitable weighting factors used to acquire a target Doppler temperature coefficient or Doppler power defect. Indicate the Updated Final Safety Analysis (UFSAR) transients that use this method.*

Discussion

The North Anna and Surry RETRAN models use a Doppler feedback correlation that is derived from data that models the dependence of Doppler Temperature Coefficient (DTC) on changes in fuel temperature, boron concentration, moderator density and fuel burnup. Through sensitivity studies using the XSDRNPM computer code [Reference 5.13-13], the DTC at various conditions was determined. XSDRNPM is a member of the SCALE code package.

The data gathered for North Anna and Surry was used to develop models to predict DTCs. A procedure to calculate a least squares fit to non-linear data with the Gauss-Newton iterative method was used to determine fit coefficients for the collected data. The model values and the percentage difference between the model and XSDRNPM values were determined. The model was also compared to 2D PDQ and 3D PDQ quarter core predictions. The PDQ code is described in Reference 5.13-10. The largest percentage difference between the model and the XSDRNPM and PDQ cases is within the nuclear reliability factor for DTC in Reference 5.13-15 over the range of conditions of interest to non-LOCA accident analysis.

It was shown that the effect of burnup, boron, and moderator specific volume could be represented as multipliers to the base DTC versus fuel temperature curve. The Doppler correlation has a core average fuel temperature component, DTC_{Tf}, and a burnup component, BURNMP. Since during a transient the burnup may be assumed to be constant, the burnup multiplier of the Doppler correlation is also assumed to be constant. To separate the reactivity feedbacks into a prompt and slower component, the impact of boron concentration and moderator density changes on the Doppler are assumed to be accounted for in the moderator feedback modeling, as these are slower feedback phenomena. Hence, the Doppler reactivity feedback is dependent only on changes in fuel temperature, which provides the prompt feedback component. The boron concentration and moderator density (specific volume) multipliers in the DTC correlation are thereby set to 1.

The DTC correlation is qualified over the range of core design DTC limits for North Anna and Surry and is described by the following equation:

$$DTC(pcm/°F) = DTC_{Tf} * BURNMP * WF$$

where

DTC_{Tf}, the fuel temperature dependence, equals $A*T_f^{0.5} + B*T_f + C$

T_f is the effective core average fuel temperature in °F and A, B, and C are correlation coefficients

BURNMP, which models burnup changes, equals DTC_{ref}/DTC_{Tf547}

DTC_{ref} is the reference DTC at the burnup of interest at hot-zero-power with 2000 ppm boron (pcm/°F)

DTC_{Tf547} is the solution to the above DTC_{Tf} equation at 547 °F.

WF is the user supplied weighting factor term that allows the user to adjust the design information to bound specific Doppler defects.

The Doppler feedback can be adjusted to a target DTC at a given fuel temperature by changing the weighting factor. For FSAR analyses in which the Doppler reactivity feedback is a key parameter, the target DTC used in RETRAN is either a least negative or most negative DTC. The RETRAN Doppler weighting factor is set so that RETRAN will initialize to the Reload Safety Analysis Checklist (RSAC) DTC limit at a core average fuel temperature that corresponds to the conditions at which the RSAC DTC limit was set.

To set the weighting factor to provide a least negative DTC, the DTC correlation is solved for the Doppler weighting factor, WF, for the appropriate core average fuel temperature and least negative DTC values. This value of the weighting factor is then entered in RETRAN control input. Likewise, to set the weighting factor to provide a most negative DTC, the weighting factor is solved using the DTC correlation with the appropriate core average fuel temperature and most negative DTC value.

All non-LOCA UFSAR transient RETRAN analyses, with the exception of the rod ejection event, apply an appropriate weighting factor to acquire a target Doppler temperature coefficient.

The rod ejection event requires additional Doppler reactivity feedback. This additional feedback is calculated as a PWF (power weighting factor), and the Doppler weighting factor calculated as described herein needs to be multiplied by the PWF before being input to the RETRAN model. The application of the power weighting factor rod ejection analyses is described in Section 2.2.3 of Reference 5.13-3.

REFERENCES FOR SECTION 5.13

5.13-1 WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology", March 1978.

5.13-2 VEP-FRD-42 Rev 2.0-A "Reload Nuclear Design Methodology", September 2003.

5.13-3 VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient", NRC SER dated September 26, 1984.

5.13-4 Virginia Power Topical Report VEP-FRD-41-A, "VEPCO Reactor System Transient Analysis using the RETRAN Computer Code," May 1985.

5.13-5 Westinghouse report WCAP-9227, "Reactor Core Response to Excessive Secondary Steam Releases," January 1978.

5.13-6 Westinghouse report WCAP-8844, "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," November 1977.

5.13-7 Westinghouse report WCAP-7907-A, "LOFTRAN Code Description," April 1984.

5.13-8 Letter from A. C. Thadani (USNRC) to W. J. Boatwright (RETRAN02 Maintenance Group), "Acceptance for Use of RETRAN02 MOD005.0," November 1, 1991.

5.13-9 Letter from W. L. Stewart (VEPCO) to Harold R. Denton (USNRC), "VEPCO Reactor System Transient Analyses", Serial No. 376, July 12, 1984.

5.13-10 Virginia Power Topical Report VEP-NAF-1, "The PDQ Two Zone Model," July 1990.

5.13-11 Letter, Stuart A. Richards (USNRC) to Gary Vine (EPRI), "Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," January 25, 2001.

5.13-12 Letter, M.L. Bowling (VEPCO) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Supplemental Information on the RETRAN NSSS Model," Serial 93-505, August 10, 1993.

5.13-13 ORNL-NUREG-CSD-2-Vol 2, Rev. 1, "XSDRNPM-S: A One-Dimensional Discrete-Ordinates Code for Transport Analysis," June 1983.

5.13-14 [Deleted]

5.13-15 Virginia Power Topical Report VEP-FRD-45A, "VEPCO Nuclear Design Reliability Factors," October 1982.

5.13-16 Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

5.13-17 Branch Technical Position APCSB9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," 1975.

5.13-18 EPRI Report, EPRI-NP-1850-CCM-A, Volume 1, Rev. 4, "RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems."

REFERENCES FOR SECTION 5.13, CONTINUED

5.13-19 Letter from Stephen R. Monarque (USNRC) to David A Christian, " North Anna Power Station, Units 1 and 2, and Surry Power Station, Units 1 and 2, Request for Additional Information on Topical Report VEP-FRD-42, Revision 2, 'Reload Design Methodology'," February 26, 2003.

5.13-20 Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Request for Additional Information on Topical Report VEP-FRD-42, Reload Nuclear Design Methodology," SN 03-183, March 21, 2003.

5.13-21 Letter from A. C. Thadani (NRC) to W. J. Boatwright (RETRAN02 Maintenance Group), Acceptance for Use of RETRAN02 MOD005.0, November 1, 1991.

6. Integrated Model Qualification

Qualification of the various component models for the Dominion RETRAN model has been described in the previous section. As discussed there, the integrated model has been benchmarked and tested against plant data and alternate code calculations. For completeness, a summary of these benchmarks is discussed here. Additional details are provided in Section 5, and cross references to sections of interest are provided here.

6.1 Benchmarks to Plant Data

1. The model capability to predict natural circulation flow was assessed against the Unit 2 natural circulation special tests. See Section 5.4.

2. The model was assessed against the 1979 North Anna 1 Stuck Open Steam Dump Valve event. See Section 5.6.

3. The model was assessed against the 1987 Steam Generator Tube Rupture event. See Section 5.6.

4. Flow coastdown benchmarks were performed against test data. See Section 5.5.

6.2 Benchmarks to Alternate Code Calculations

1. The original VEP-FRD-41A (Rev. 0) qualification set included benchmarks against various vendor calculations published in the UFSAR. See Section 5.2 of Appendix 1.

2. Comparisons to Westinghouse LOFTRAN calculations for several loop-symmetric transients were performed and reviewed by the USNRC as part of the original topical report approval process. See Appendix 4.

3. Comparison to Westinghouse LOFTRAN calculations for main feedline break are presented in Section 5.6 of this report.

This validation set has demonstrated that Dominion's RETRAN models are producing reasonable transient analysis results which are consistent with measured plant data and vendor code calculations.

6.3 Uncertainty Quantification/ Accuracy Assessment

No specific code uncertainty has been quantified for the various RETRAN transient output parameters. Consistent with current industry practice, the overall conservatism of the RETRAN output results for specific licensing applications is assured by selection of bounding inputs, which includes but is not limited to:

- Limiting initial condition selection (conservative end of control and instrument uncertainty band)
- Limiting single failure of the protection system
- No credit for control system operation when such operation produces less limiting results.
- Conservative protection system setpoints (Inclusion of instrument uncertainties)
- Conservative (bounding) trip delay times
- Selection of core physics (i.e. reactivity) characteristics which conservatively bound the range of core burnup and other related conditions such as xenon distributions. These reactivity inputs are reassessed for every reload core to ensure they remain bounding.

6.4 Restrictions, Cautions and Limitations

Application of Dominion's RETRAN models to licensing applications is subject to the following general limitations:

1. The generic RETRAN code restrictions, limitations and conditions of used imposed by the USNRC's generic Safety Evaluation Reports (SERs), as discussed in Section 5 and Appendix 7 must continue to be addressed.
2. The licensing basis assumptions set forth in the UFSAR for the various analyzed accidents must be addressed for each new analysis.
3. Model Application Procedures exist. Certain precautions and limitations of the range of applicability of various component models are highlighted in these application procedures, and Dominion safety analysts must remain cognizant of these precautions.

7 CONCLUSIONS

The Virginia Electric and Power Company (Dominion) has developed the capability to perform system transient analyses with the RETRAN-02 computer code. The general code features have been discussed and a description of the Surry and North Anna input description (i.e. the "models") has been provided. The adequacy of these models has been demonstrated via a series of benchmark calculations to alternate codes, UFSAR vendor results and plant data.

The generic RETRAN code restrictions, cautions and limitations set forth in the USNRC's various code Safety Evaluation Reports (SERs) have been discussed and addressed.

This report and the references cited herein form the basis for the ongoing applicability of these models to licensing and plant operational support of the Surry and North Anna Power Stations.

This version of the Dominion RETRAN topical report has been designated VEP-FRD-41 Rev. 0.1-A. The basis for retention of the -A designation is the NRC review and approval of the technical basis for this report documented in Reference 7-1 and summarized in Section 1.

REFERENCE FOR SECTION 7

7-1 Letter from Scott Moore, USNRC, to D. A. Christian, Dominion, Virginia Electric and Power Company, Acceptance of Topical Report VEP-FRD-42, Revision 2, "Reload Nuclear Design Methodology", North Anna and Surry Power Stations, Units 1 and 2, Serial No. 03-381, June 11, 2003.

APPENDIX 1
VEP-FRD-41A Rev. 0

VEP-FRD-41-A, Rev. 0

Vepco Reactor System Transient Analyses Using the RETRAN Computer Code

Nuclear Analysis & Fuel
Nuclear Engineering & Services

May, 1985

VEP-FRD-41

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
RICHMOND, VIRGINIA

MARCH, 1981

RECOMMENDED FOR APPROVAL:

R. W. Cross
Supervisor, Nuclear Fuel Engineering

APPROVED:

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. Analysis Methodology

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, 1986, 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.

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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.

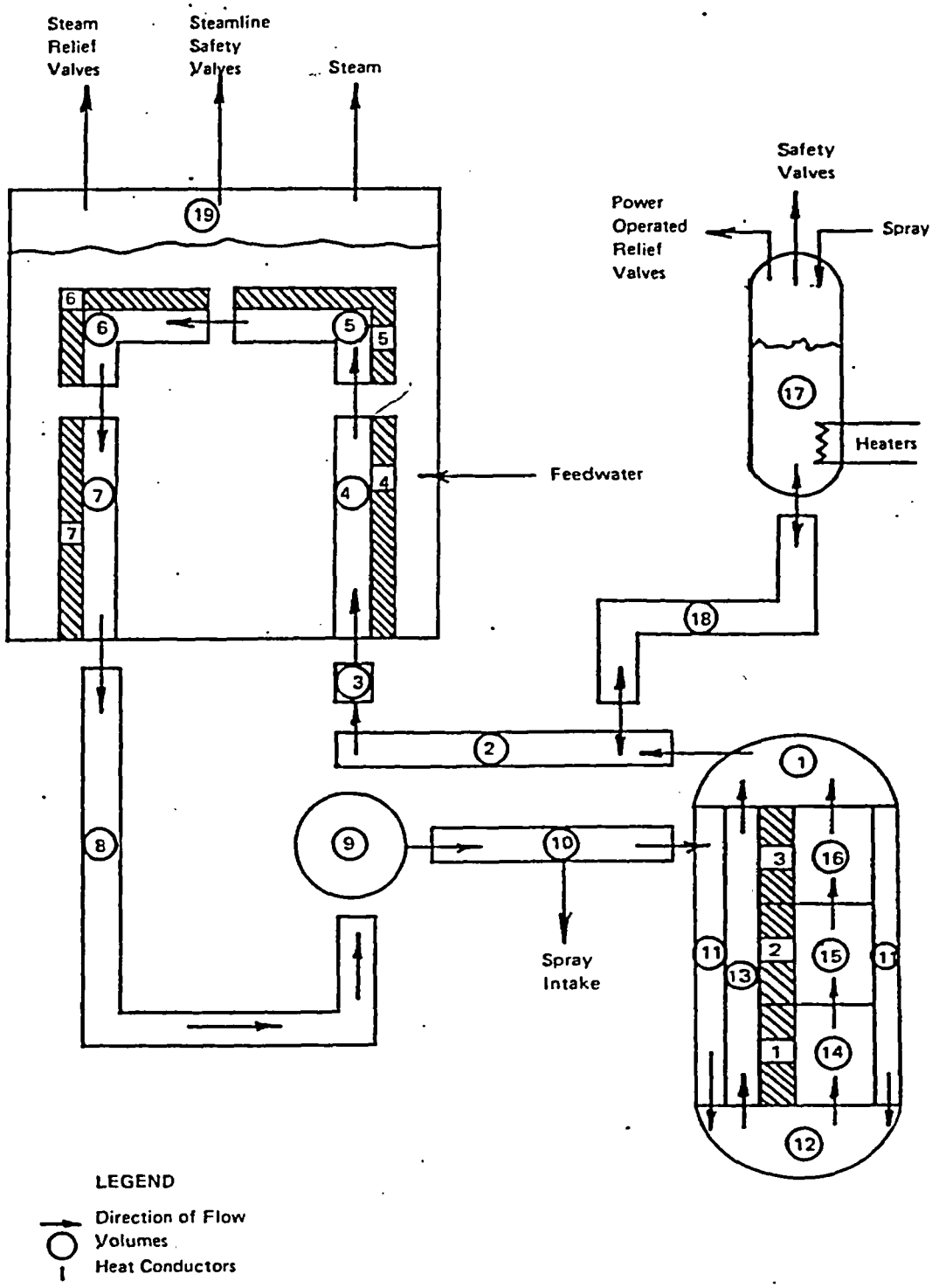
The Single Loop Model also incorporates the RETRAN control system capability to model the following NSSS control and protection features:

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 and 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

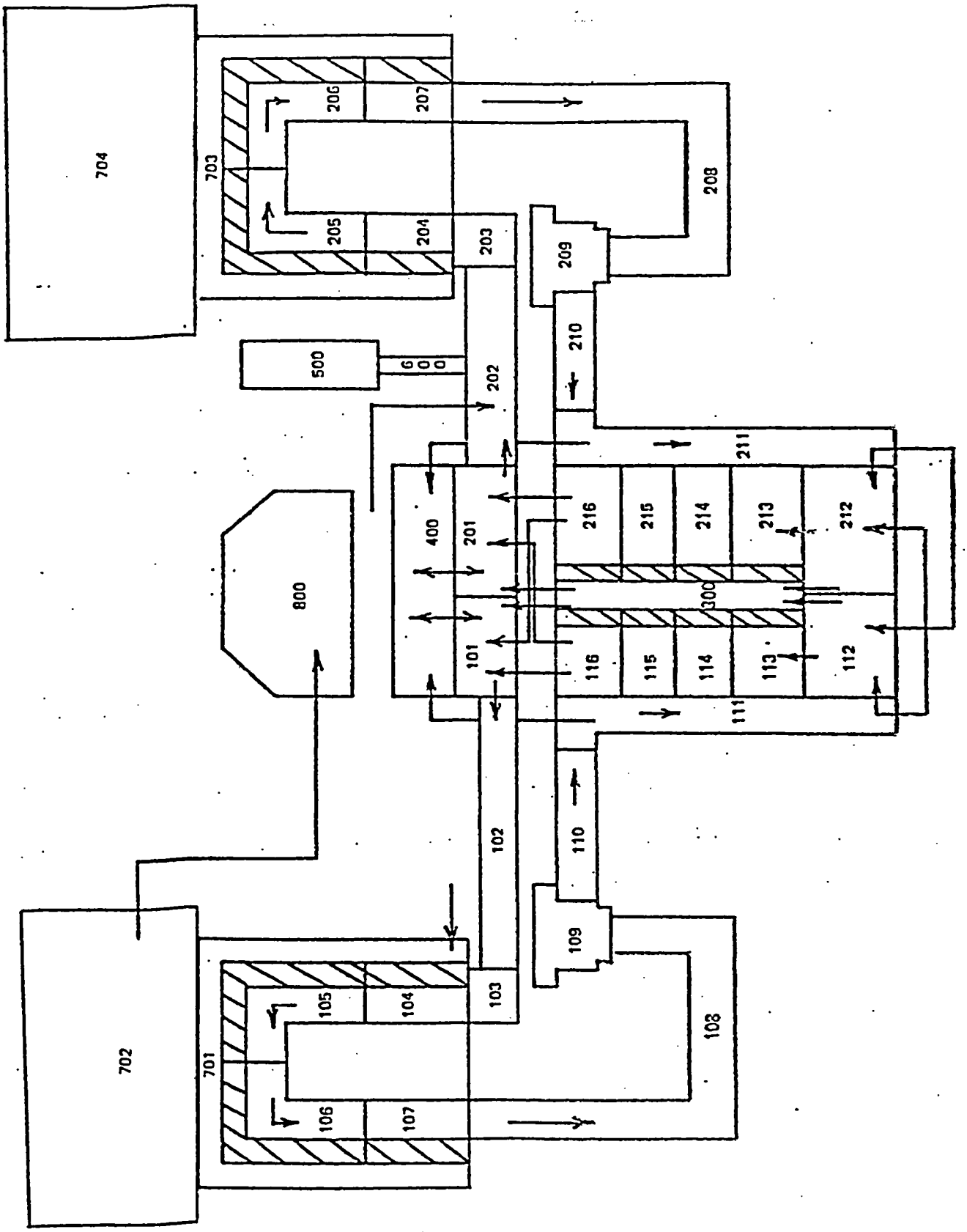


Table 3.1

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, °F (@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 Uncontrolled Control Rod Assembly Withdrawal at Power Transient - Current Analysis

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

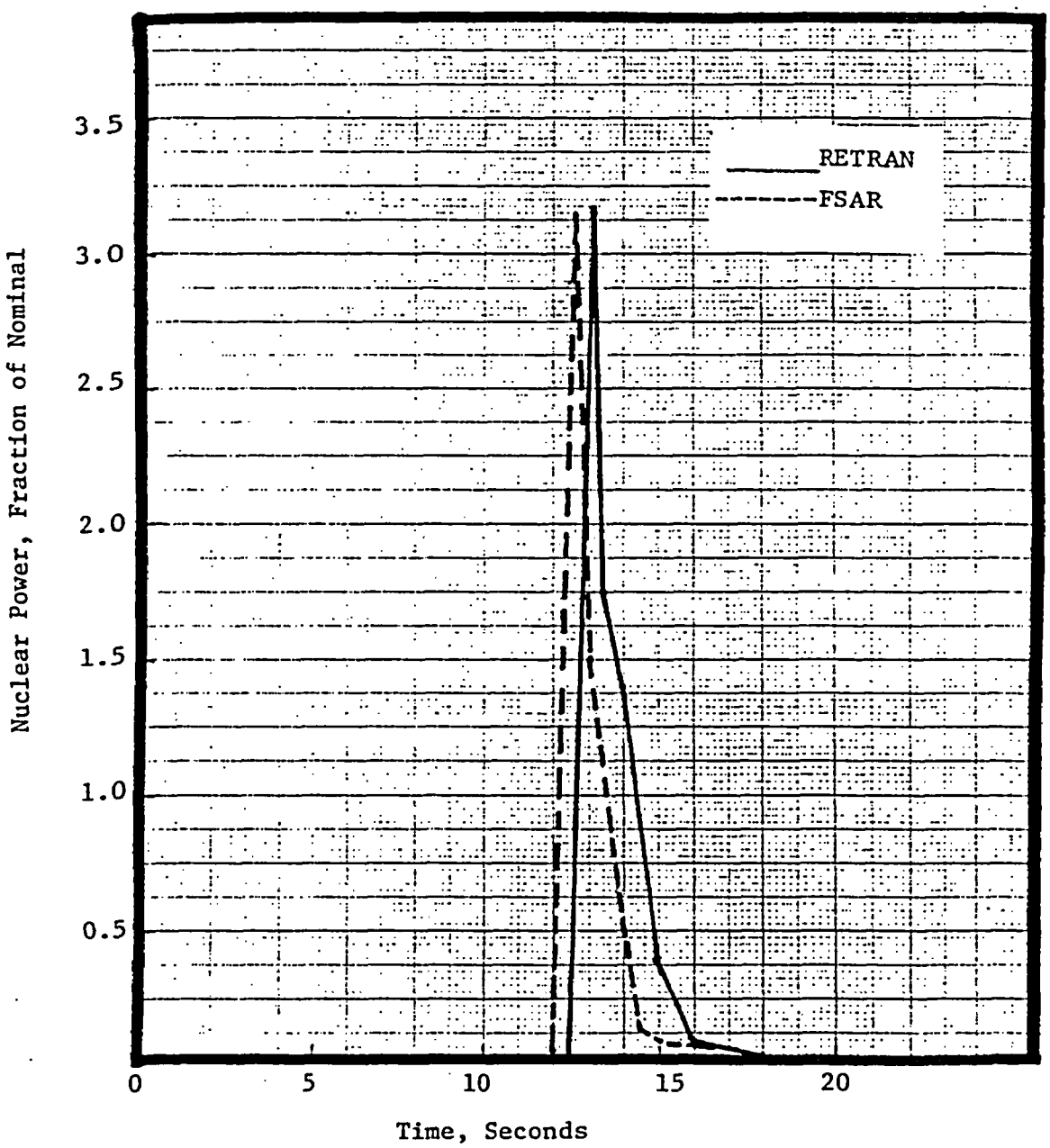


Figure 5.2

AVERAGE FUEL TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

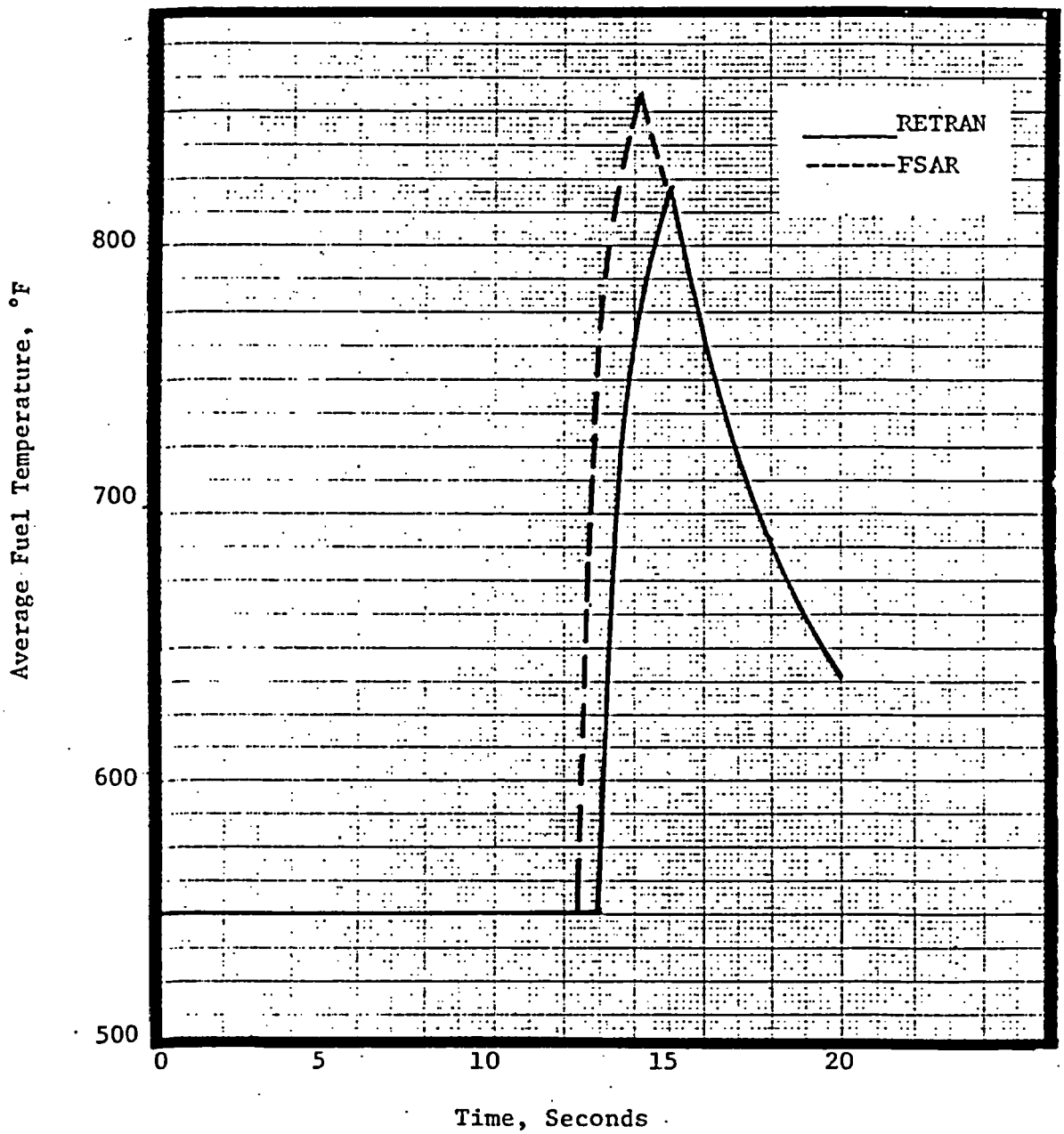


Figure 5.3

AVERAGE CLAD TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

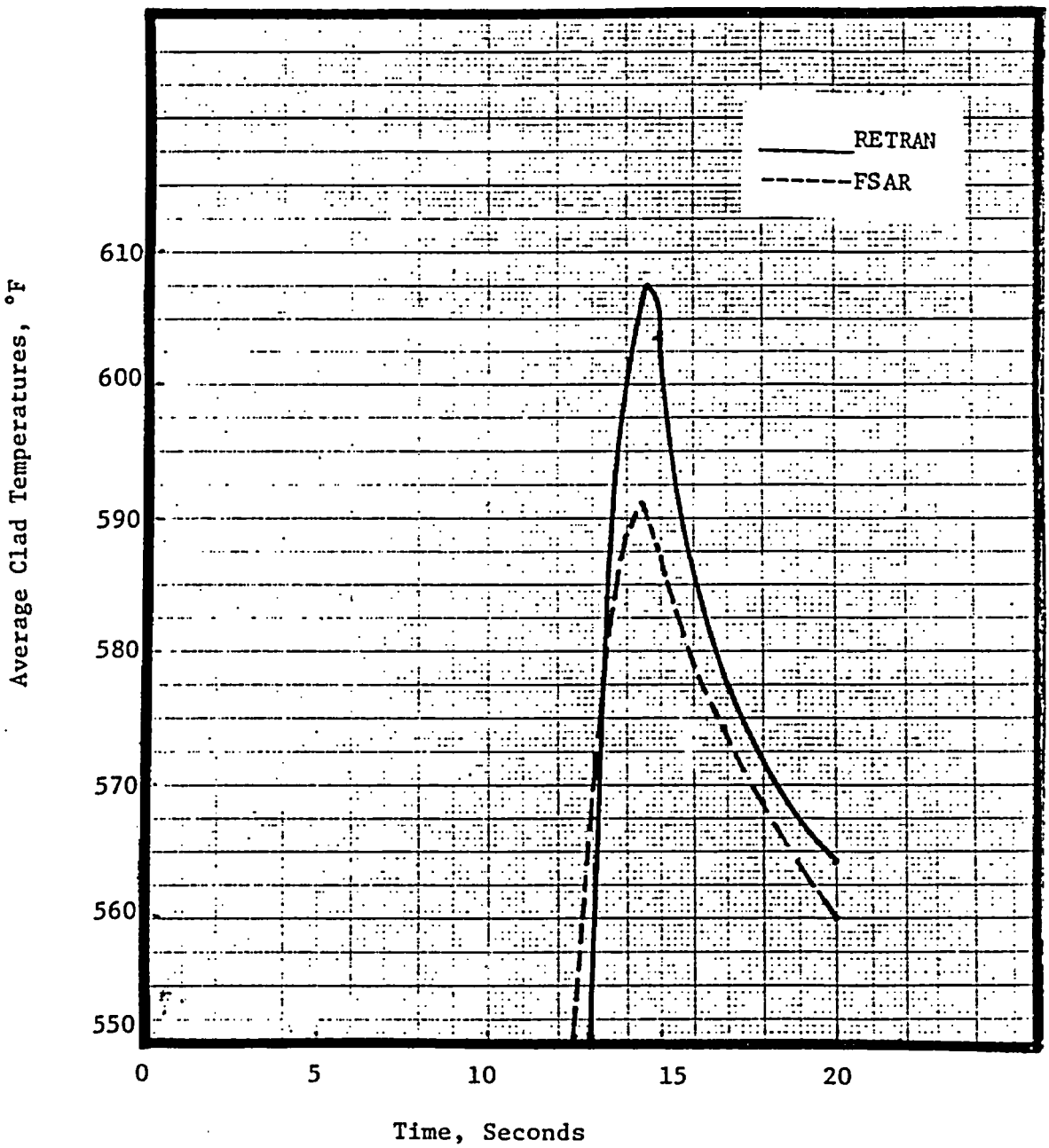


Figure 5.4

CORE HEAT FLUX
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

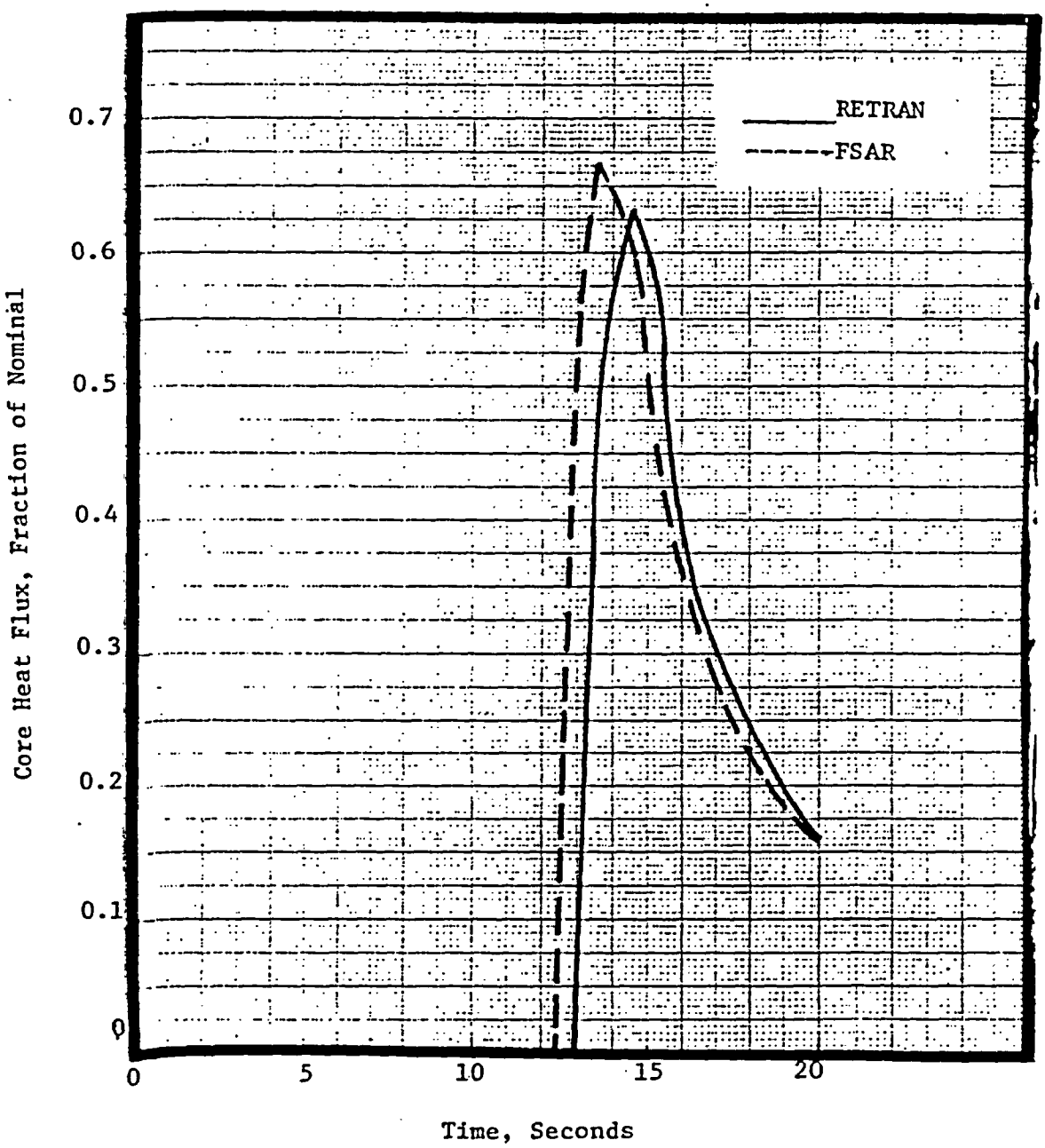


Figure 5.5

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

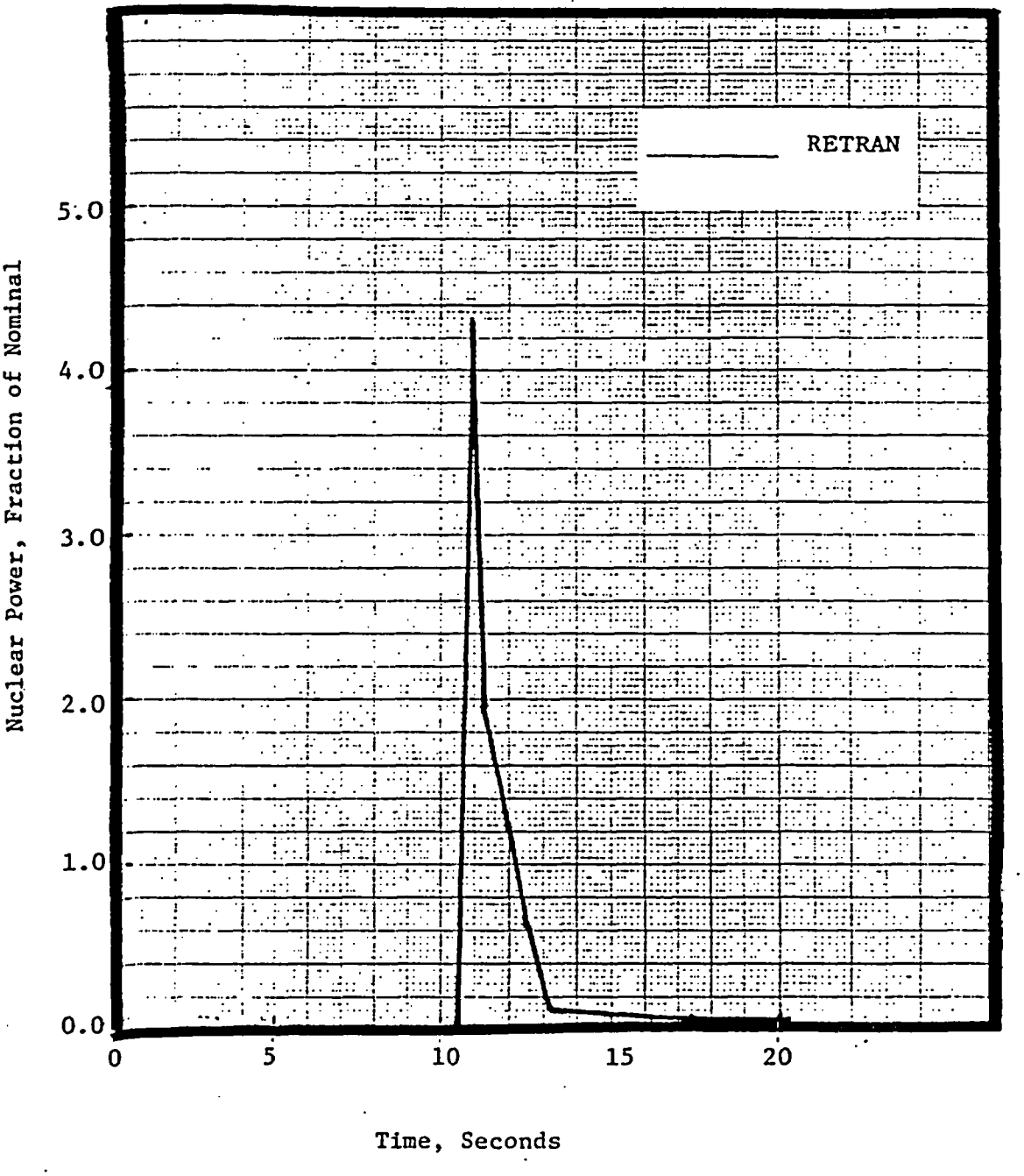


Figure 5.6

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

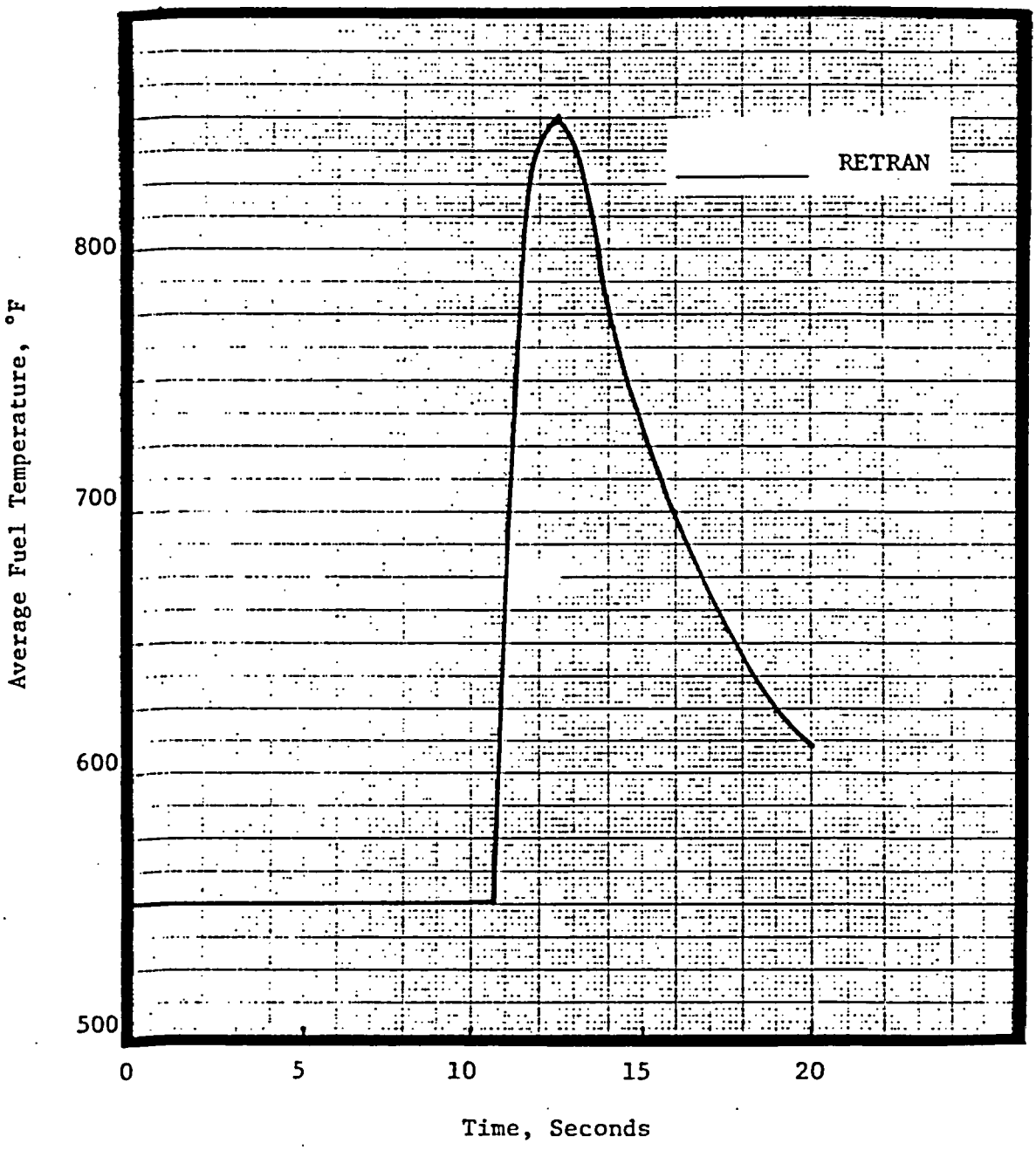


Figure 5.7

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

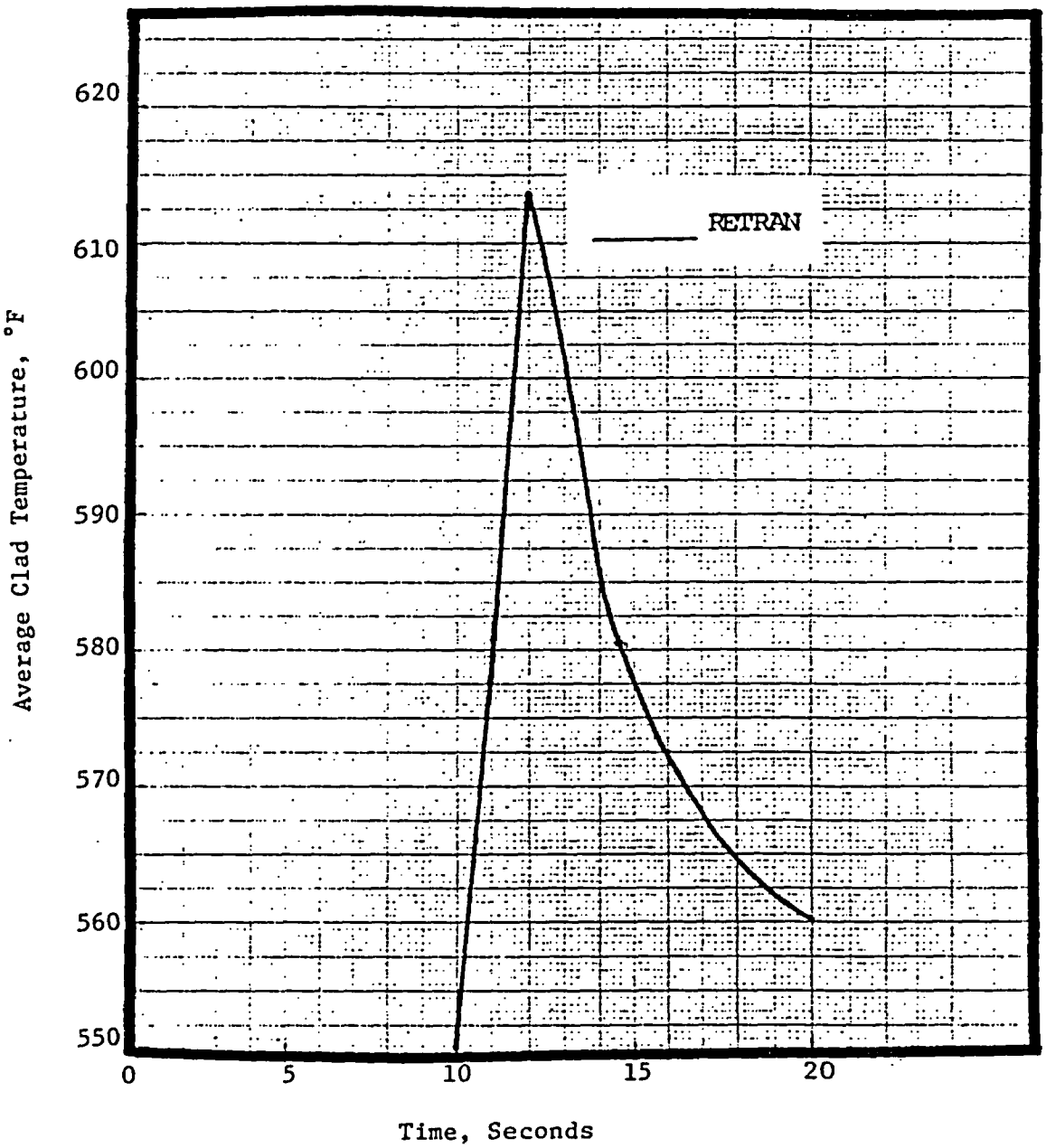


Figure 5.8

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

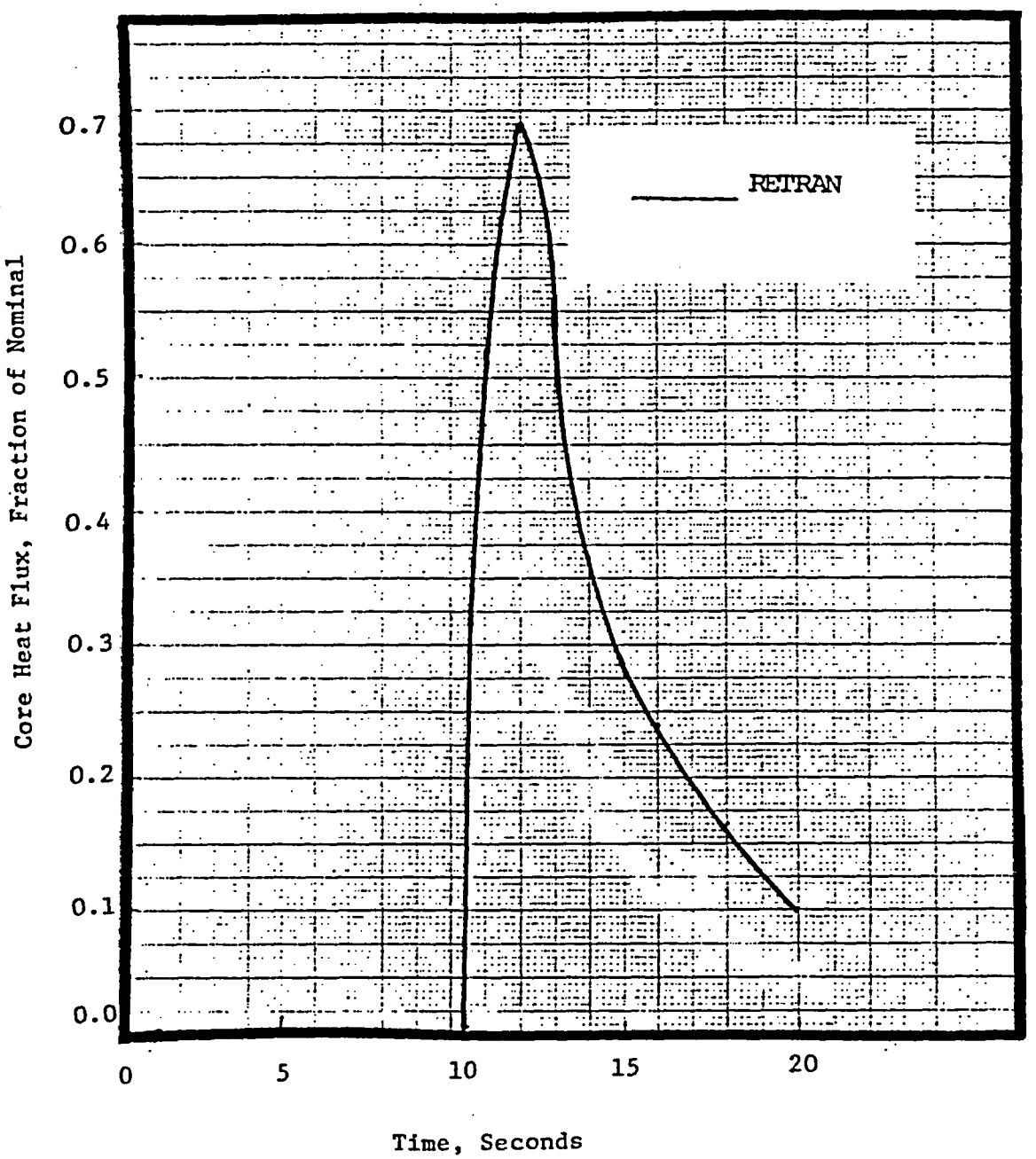


Figure 5.9
NUCLEAR POWER
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

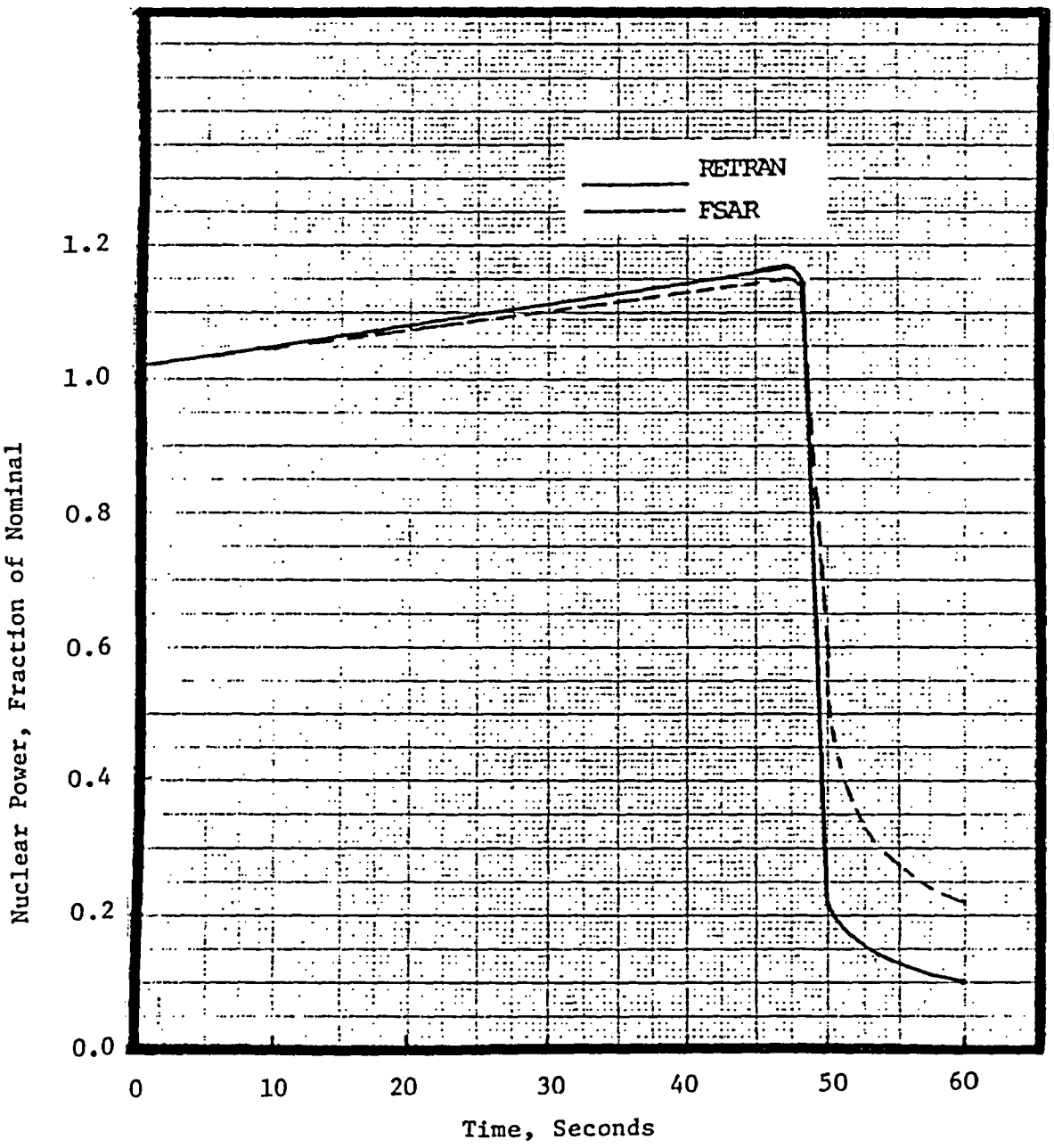


Figure 5.10
PRESSURIZER PRESSURE
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

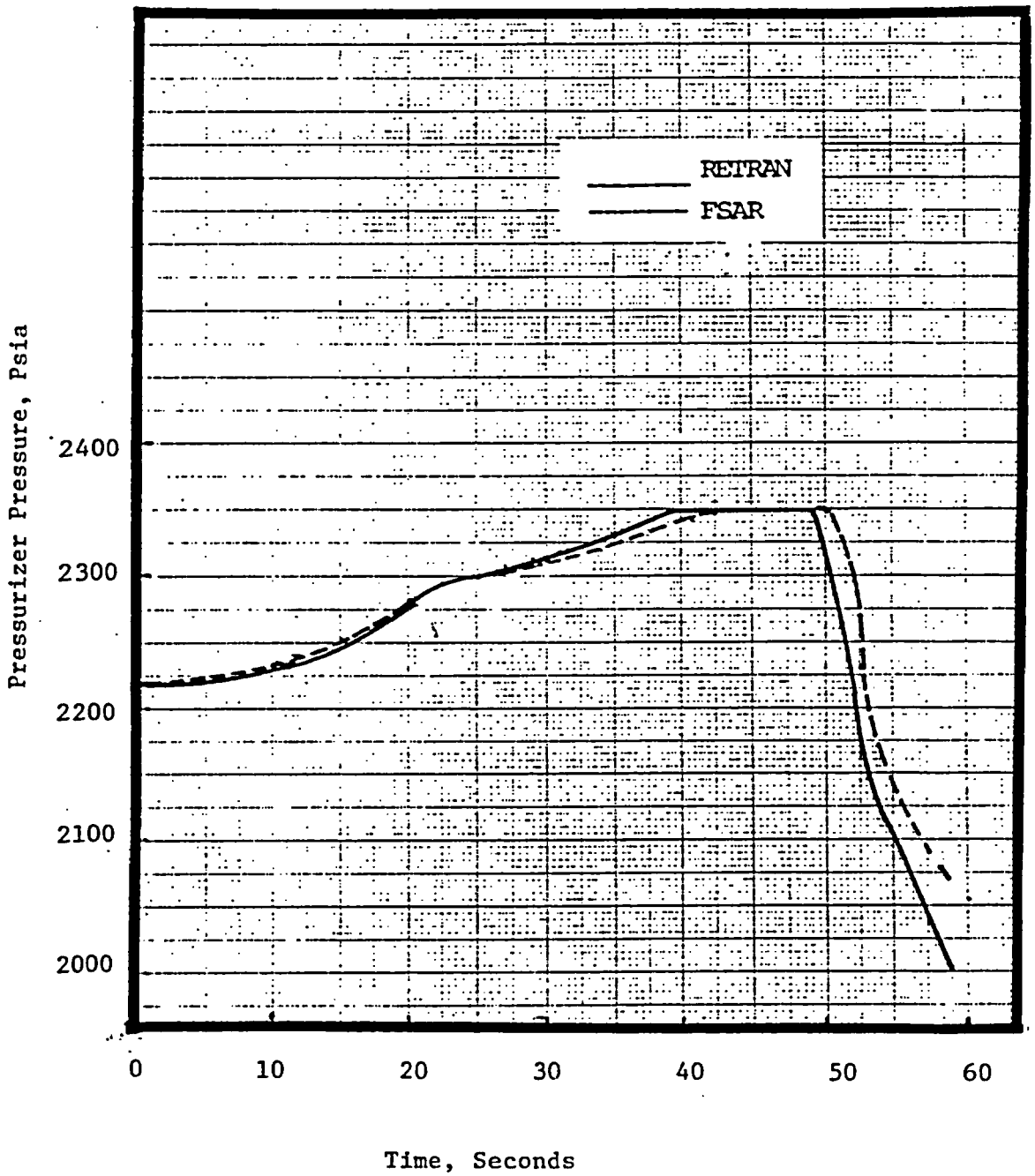


Figure 5.11

AVERAGE COOLANT-TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

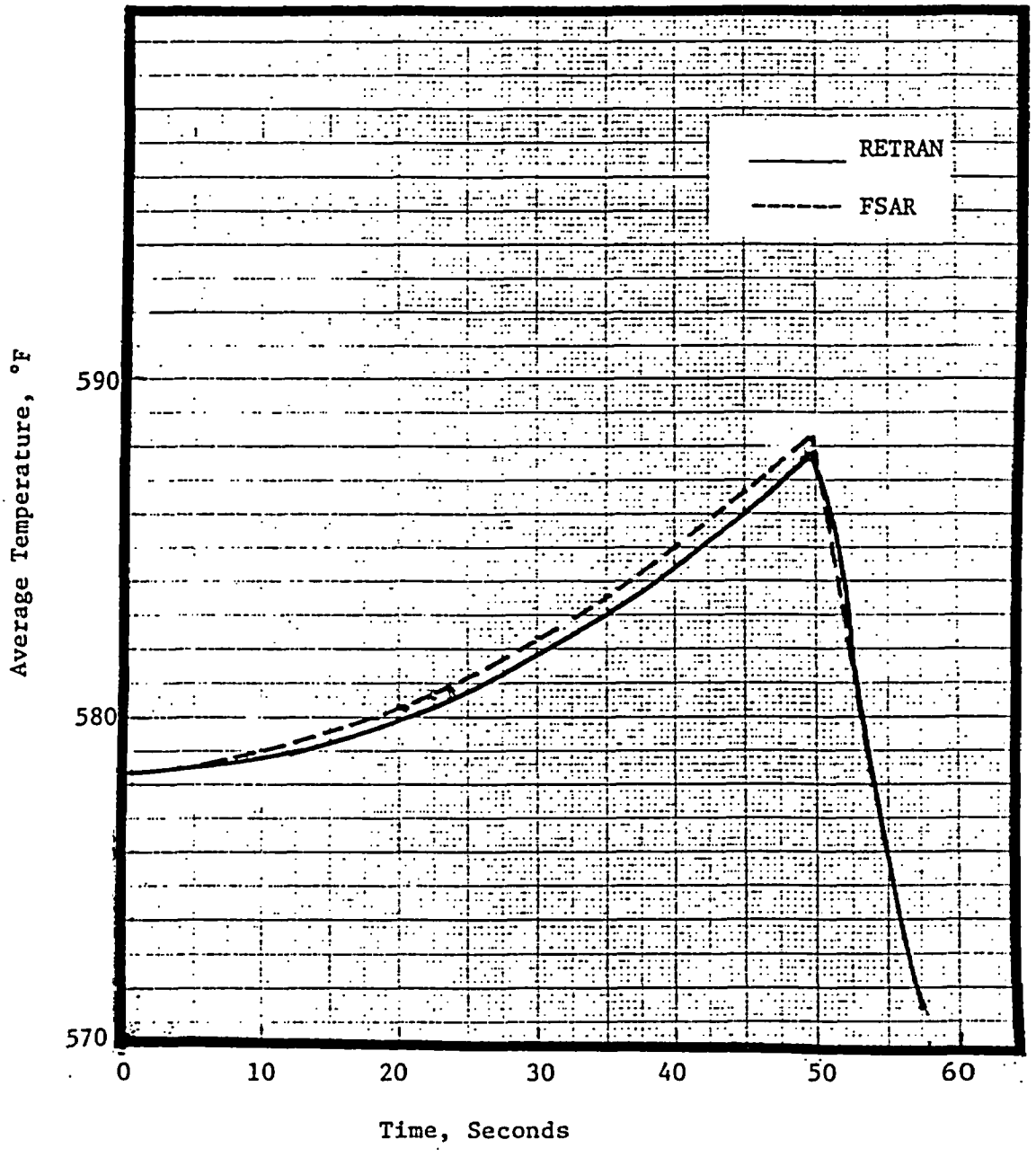


Figure 5.12

DNB RATIO
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

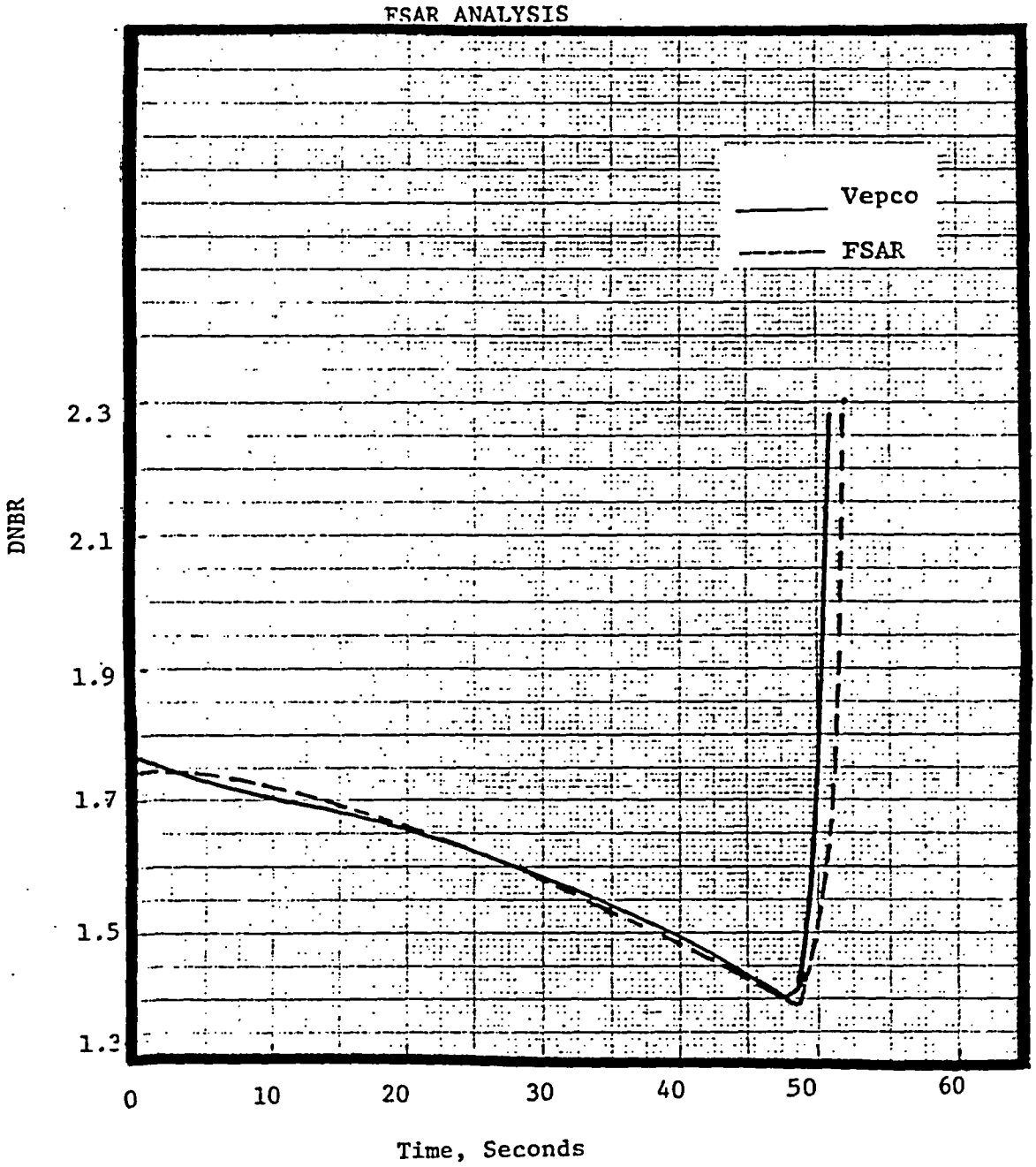


Figure 5.13

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

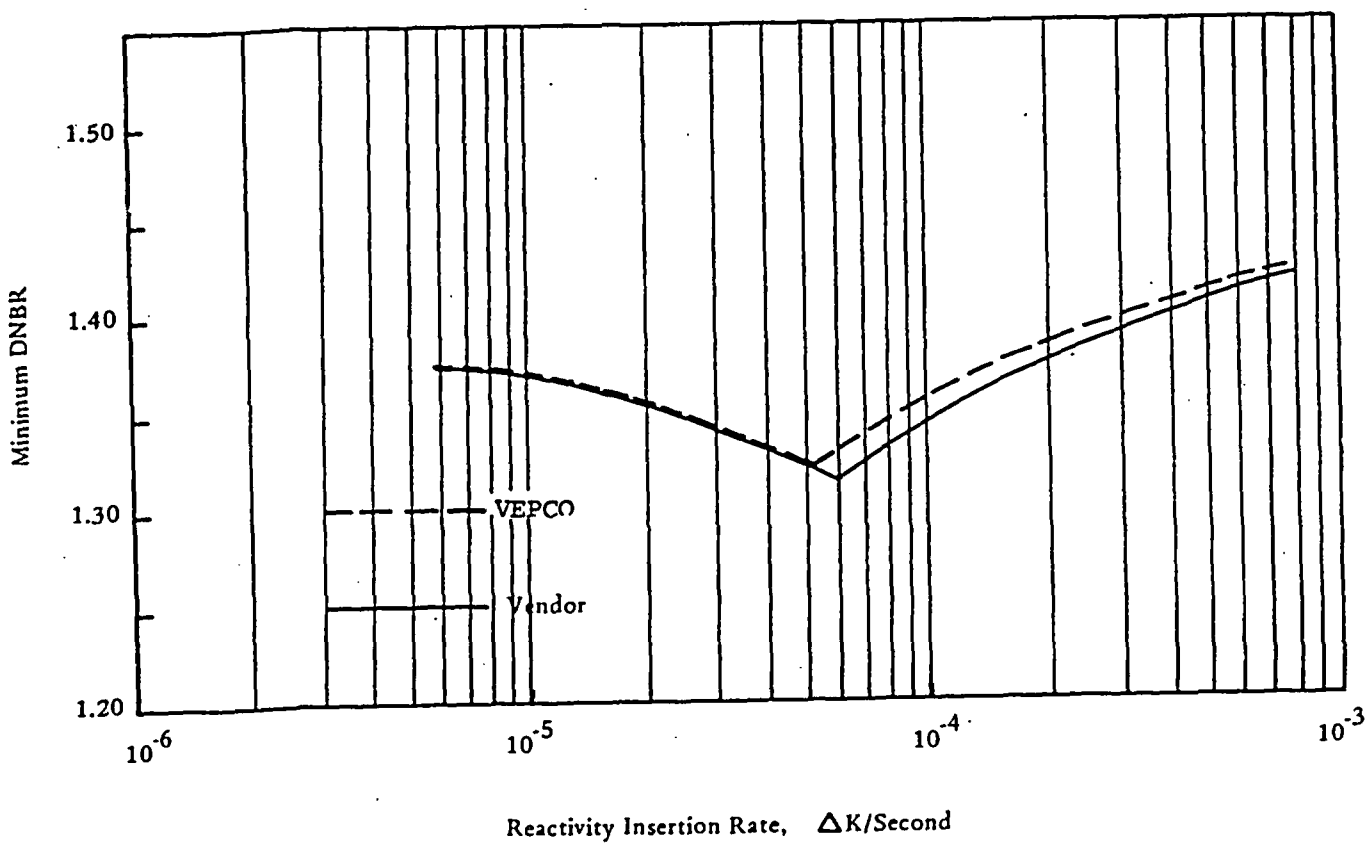
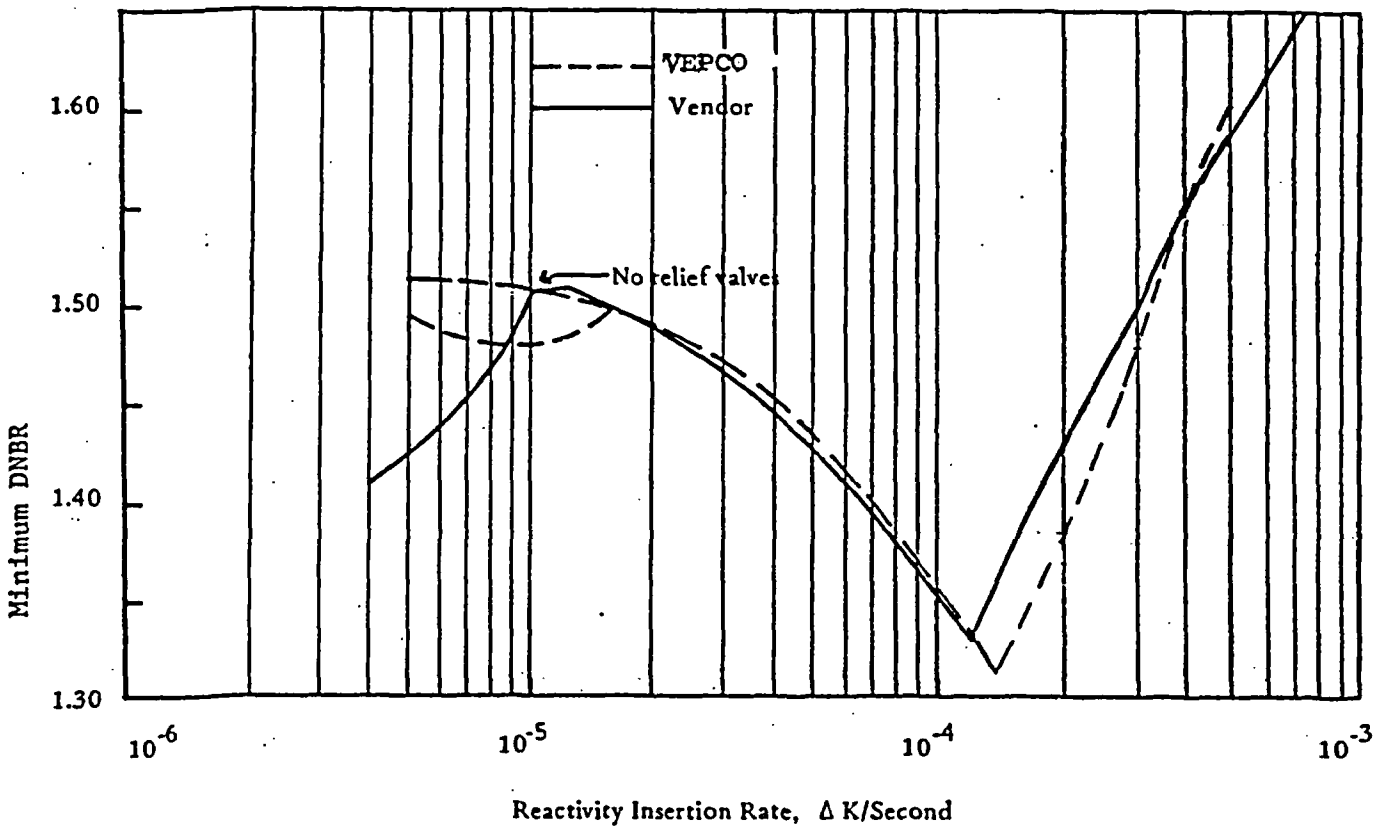


Figure 5.14

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

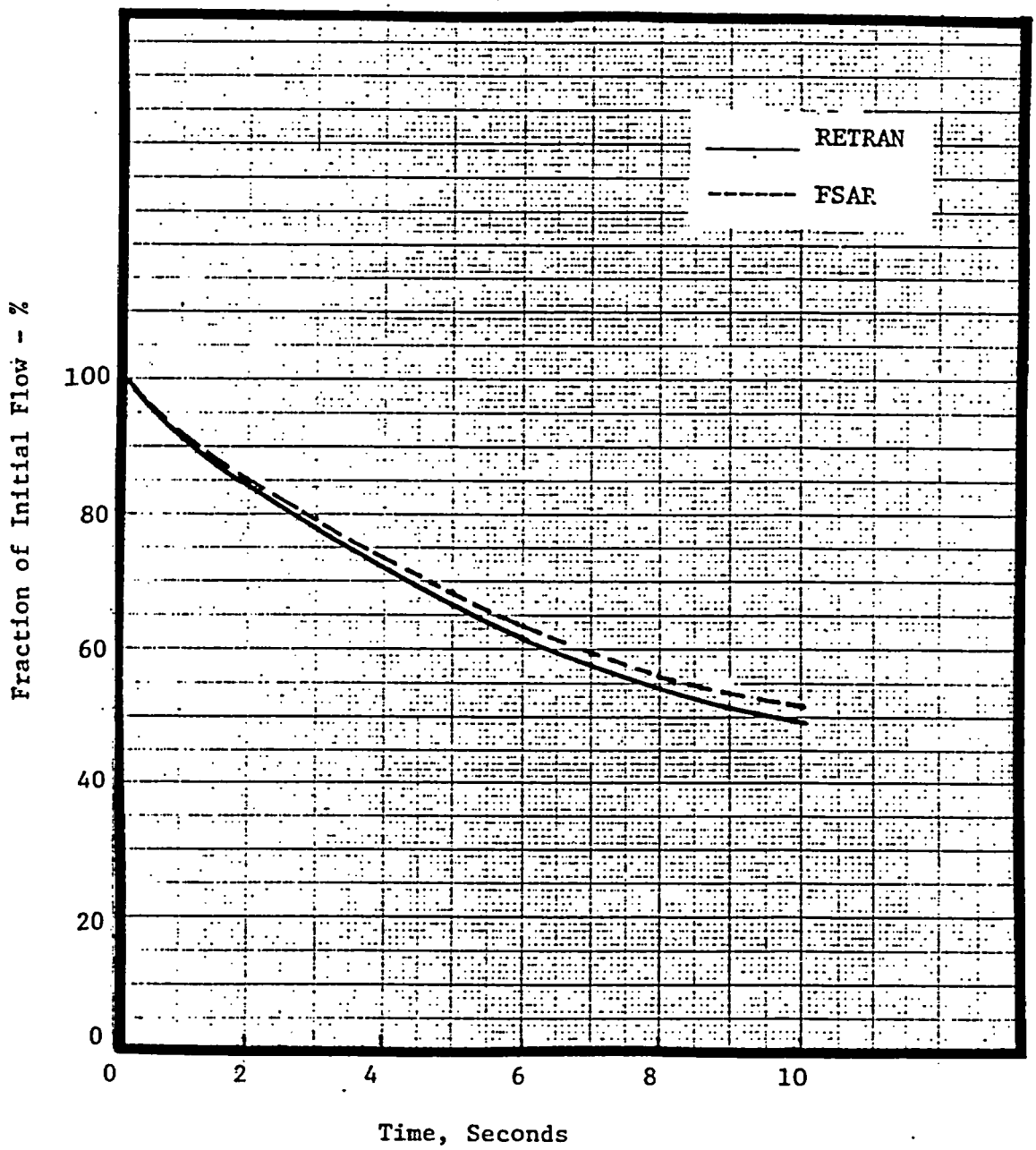


Figure 5.16

NUCLEAR-POWER
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

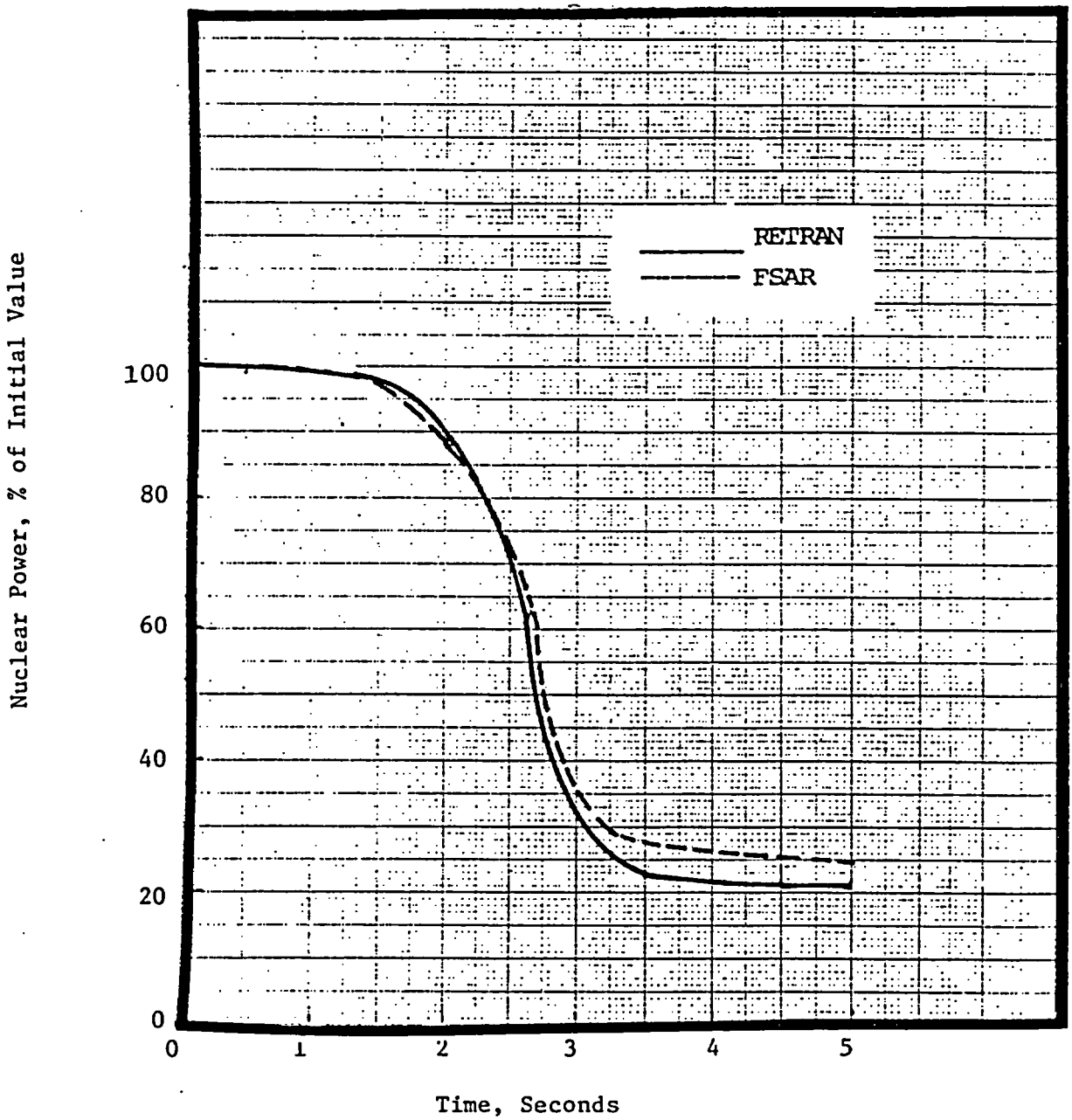


Figure 5.17

AVERAGE HEAT FLUX
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

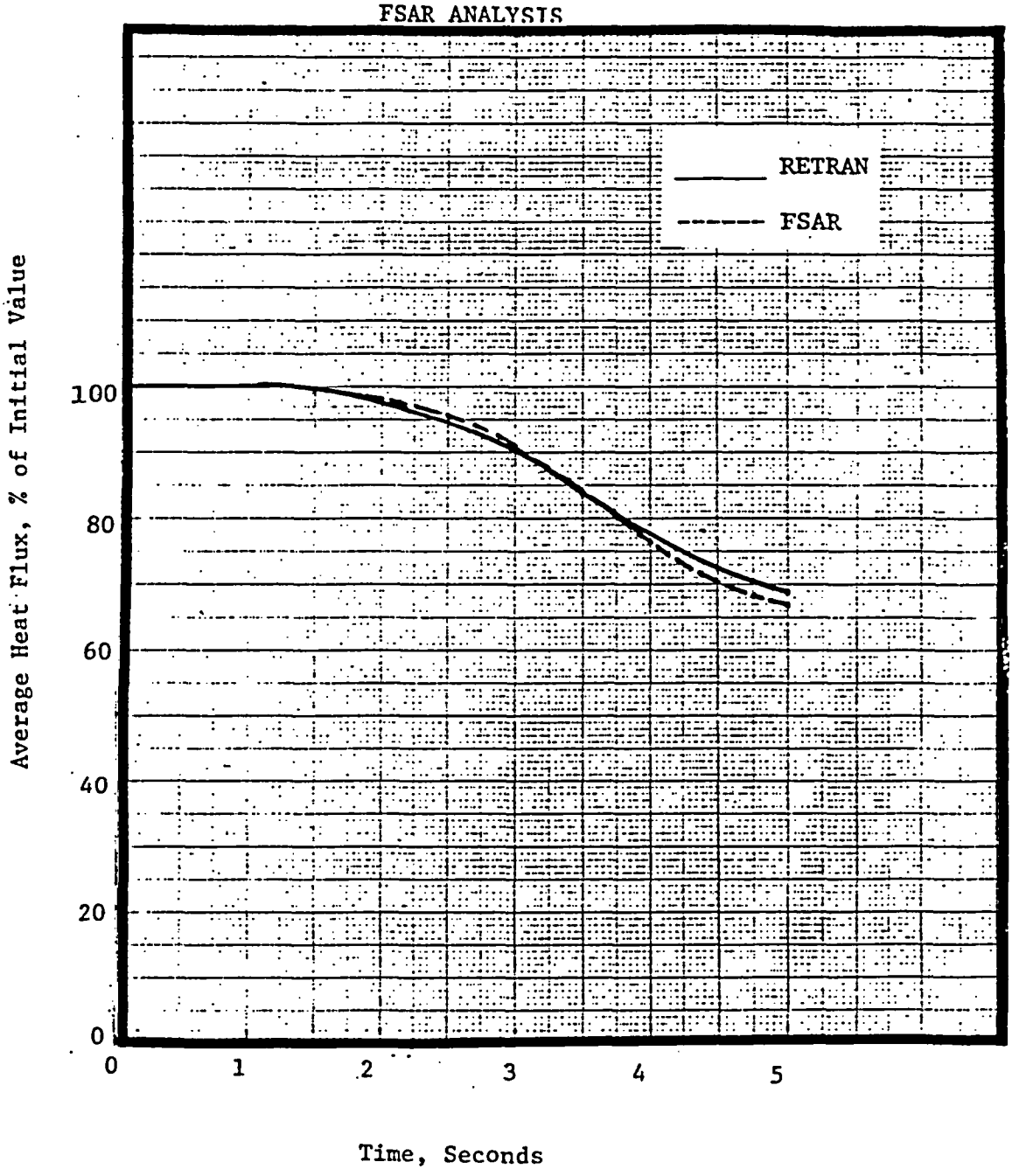


Figure 5.18

DNB RATIO
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

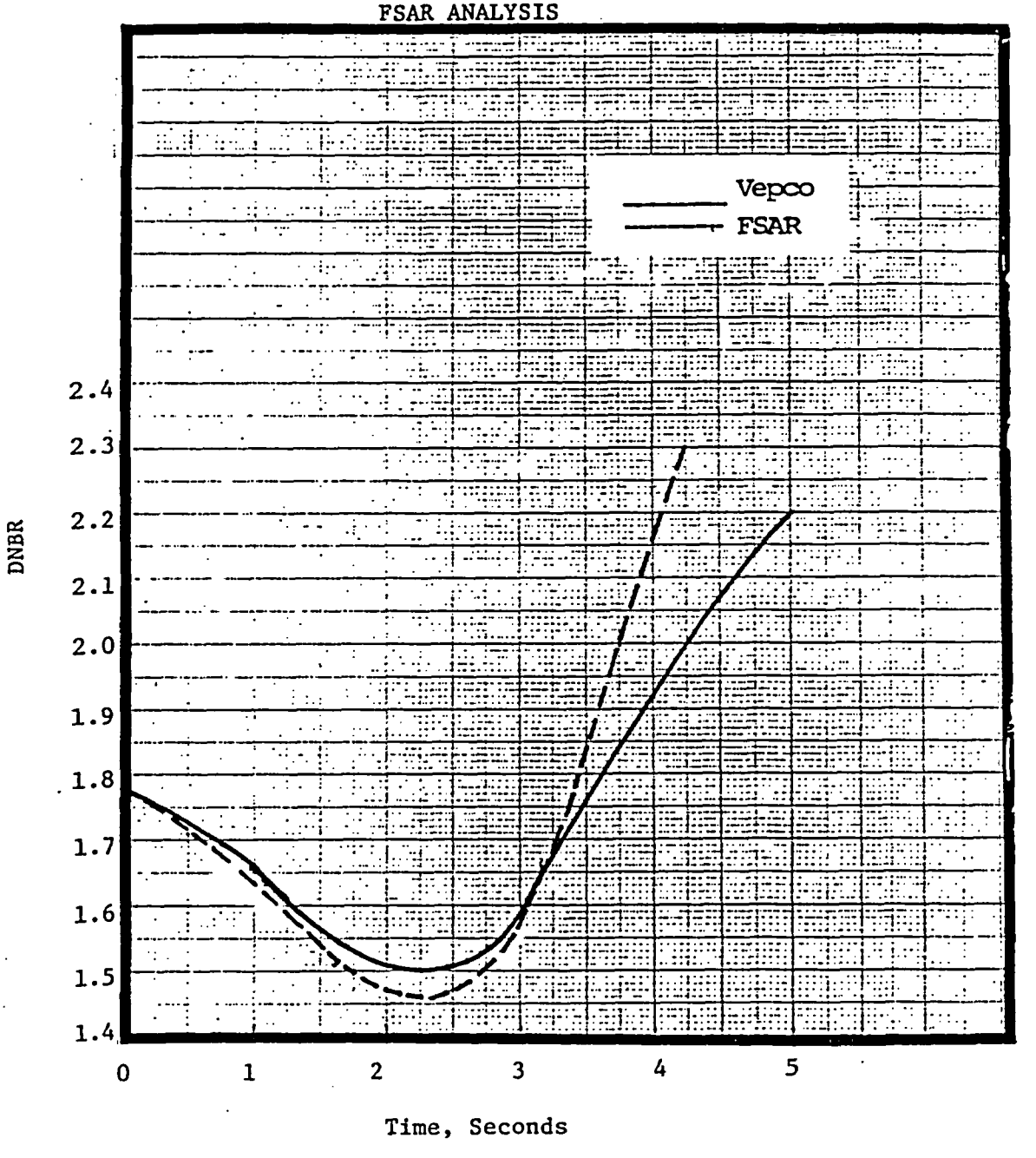


Figure 5.19

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

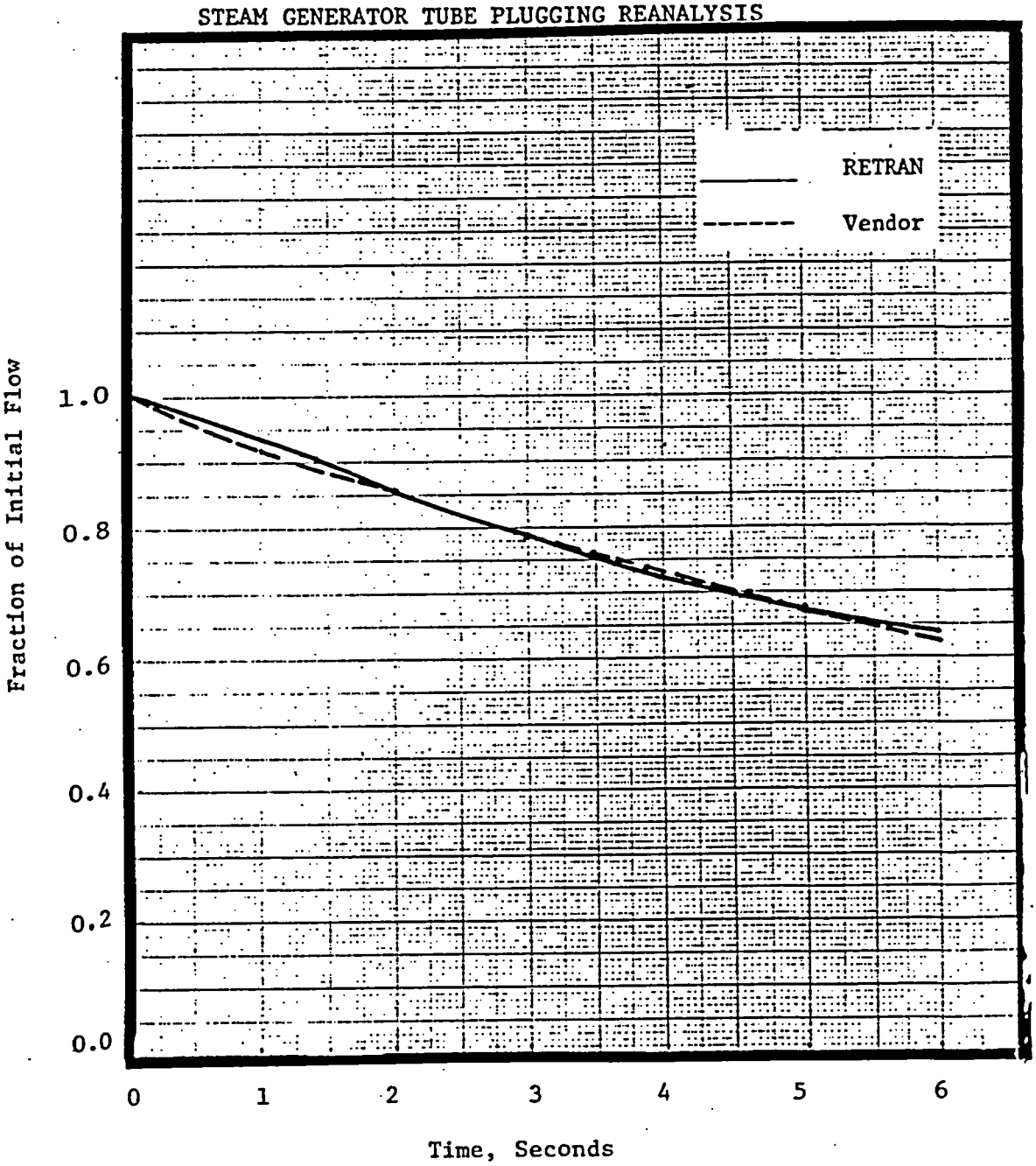


Figure 5.20

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

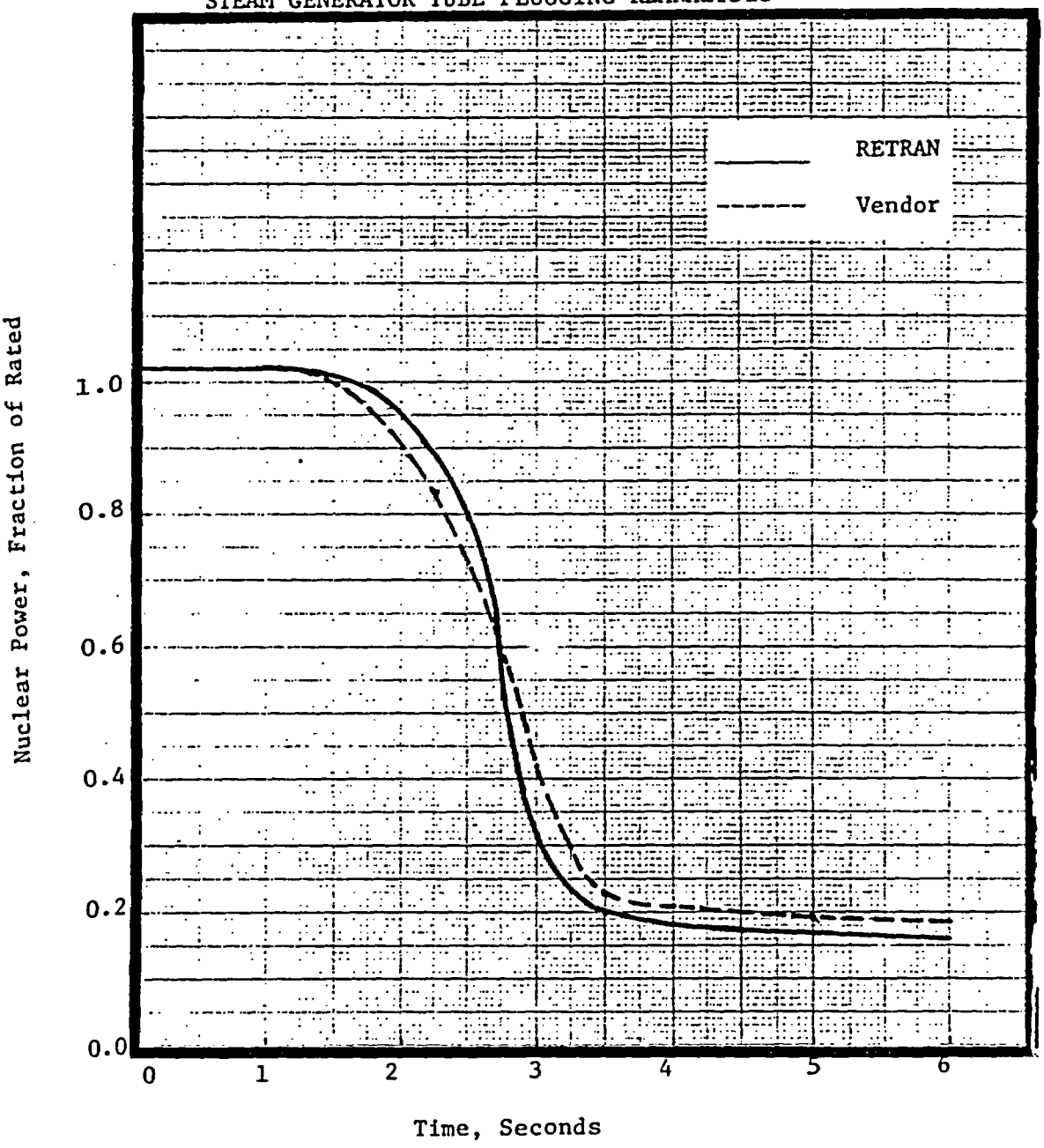


Figure 5.21

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

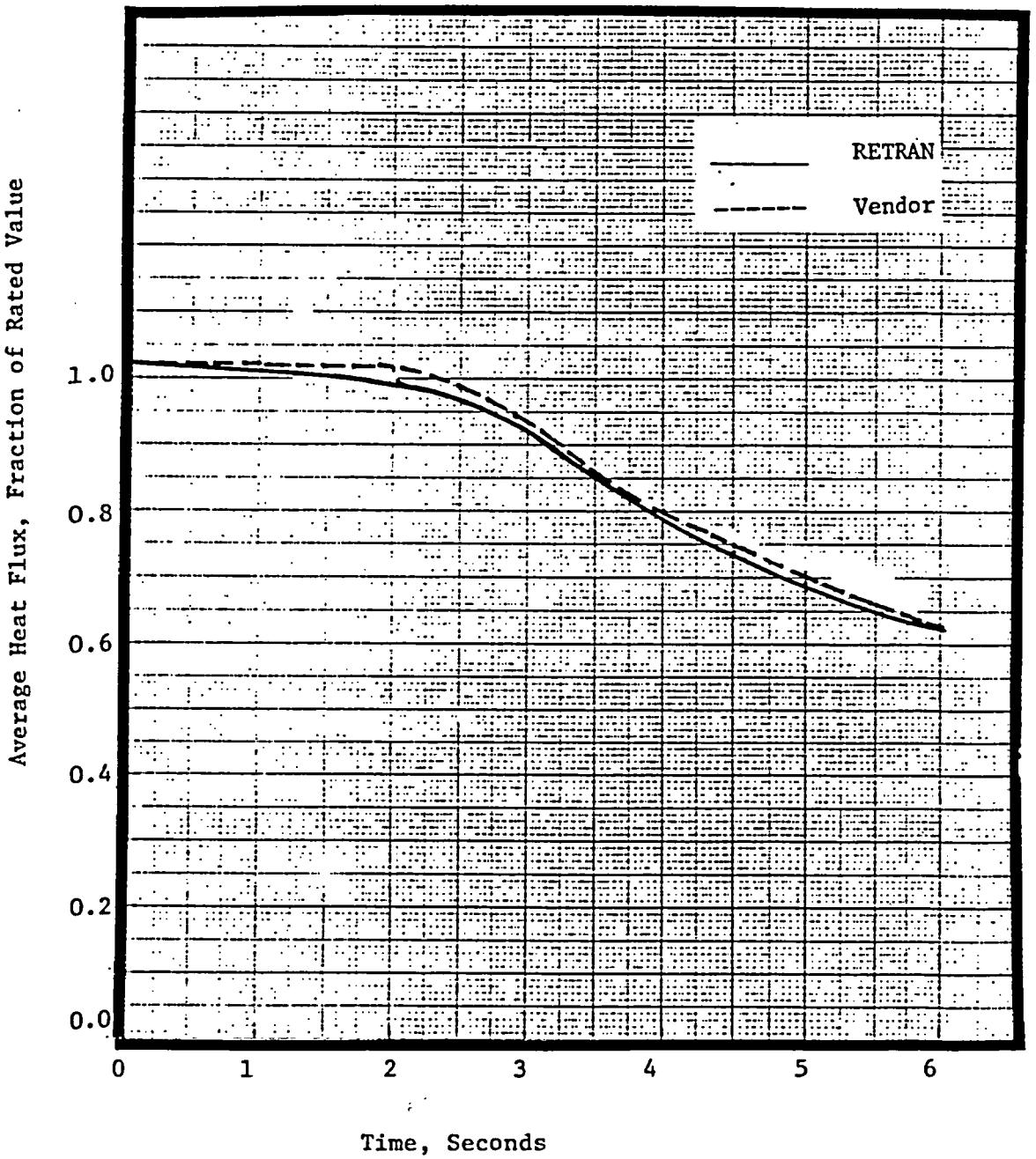


Figure 5.22

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

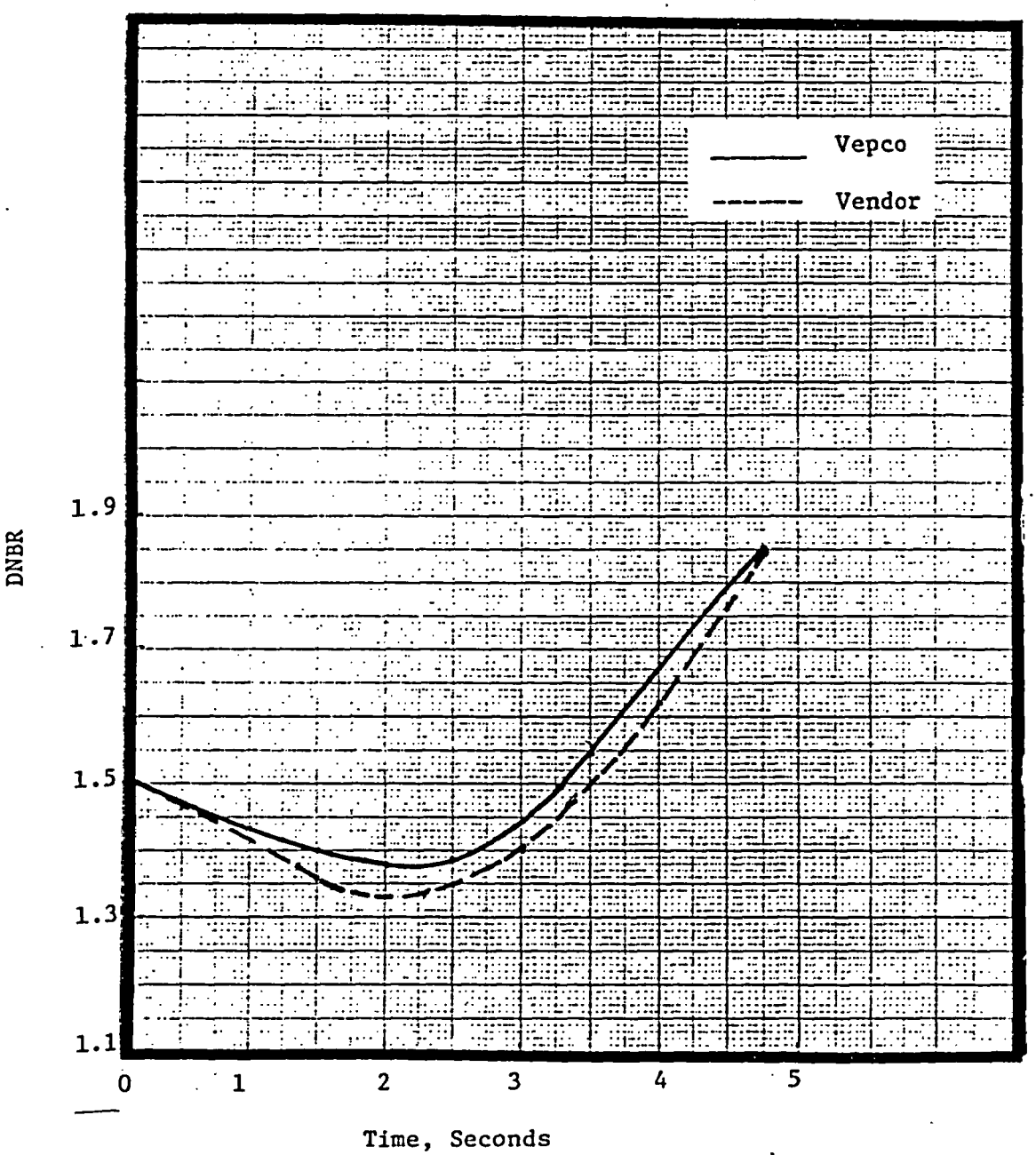


Figure 5.23

CORE FLOW COASTDOWN
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

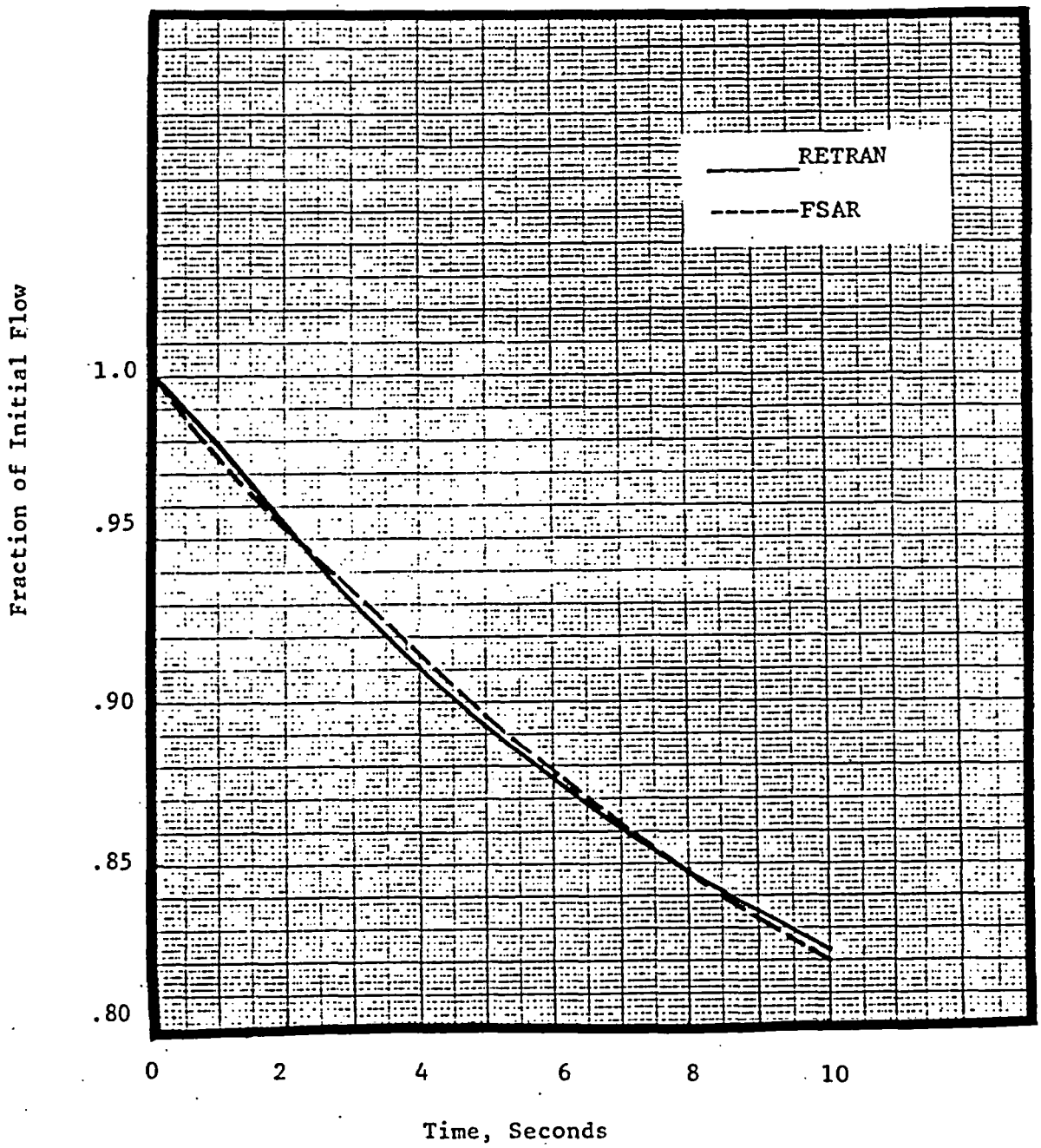


Figure 5.24

NUCLEAR POWER
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

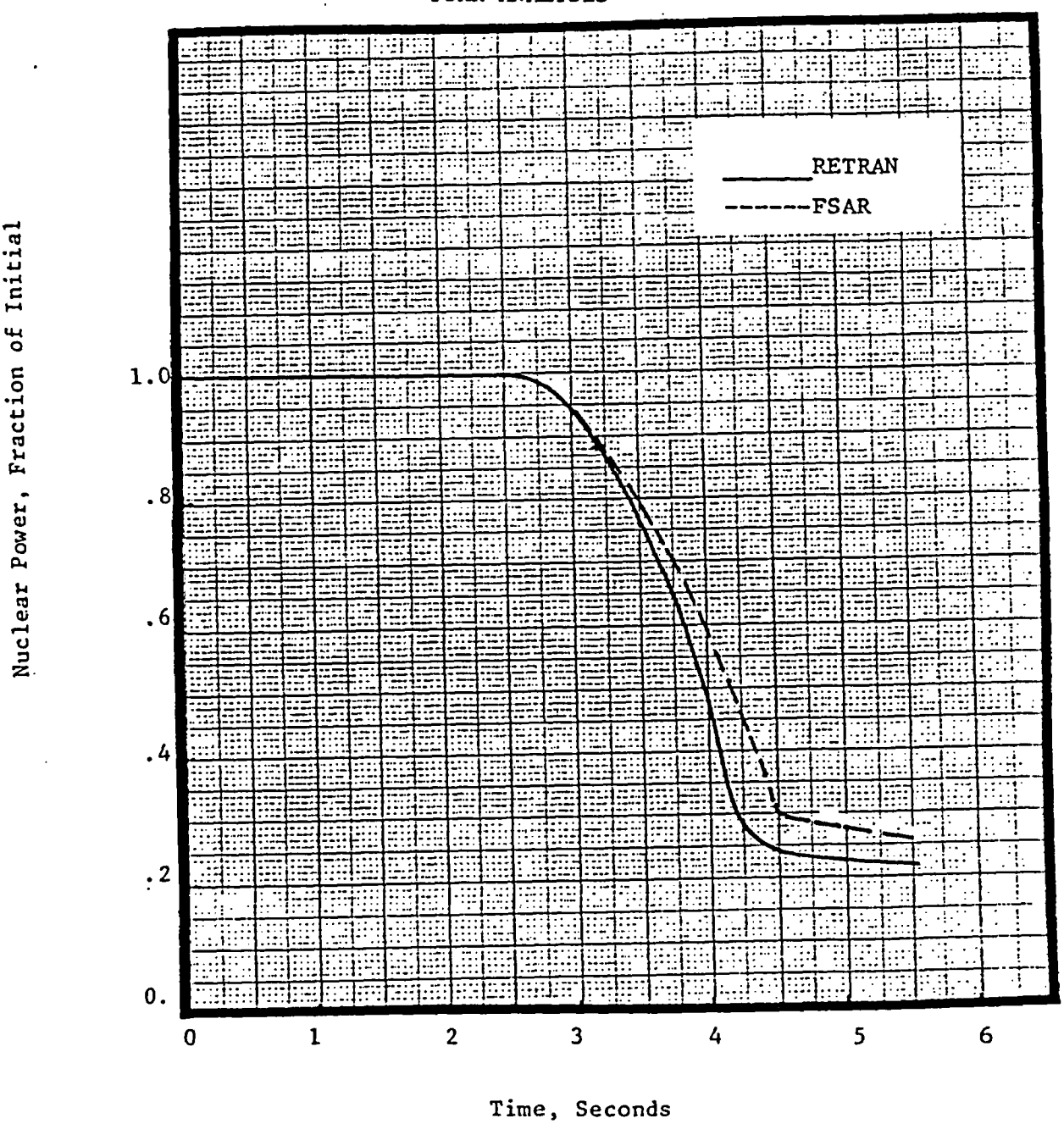


Figure 5.25

CORE AVERAGE HEAT FLUX
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

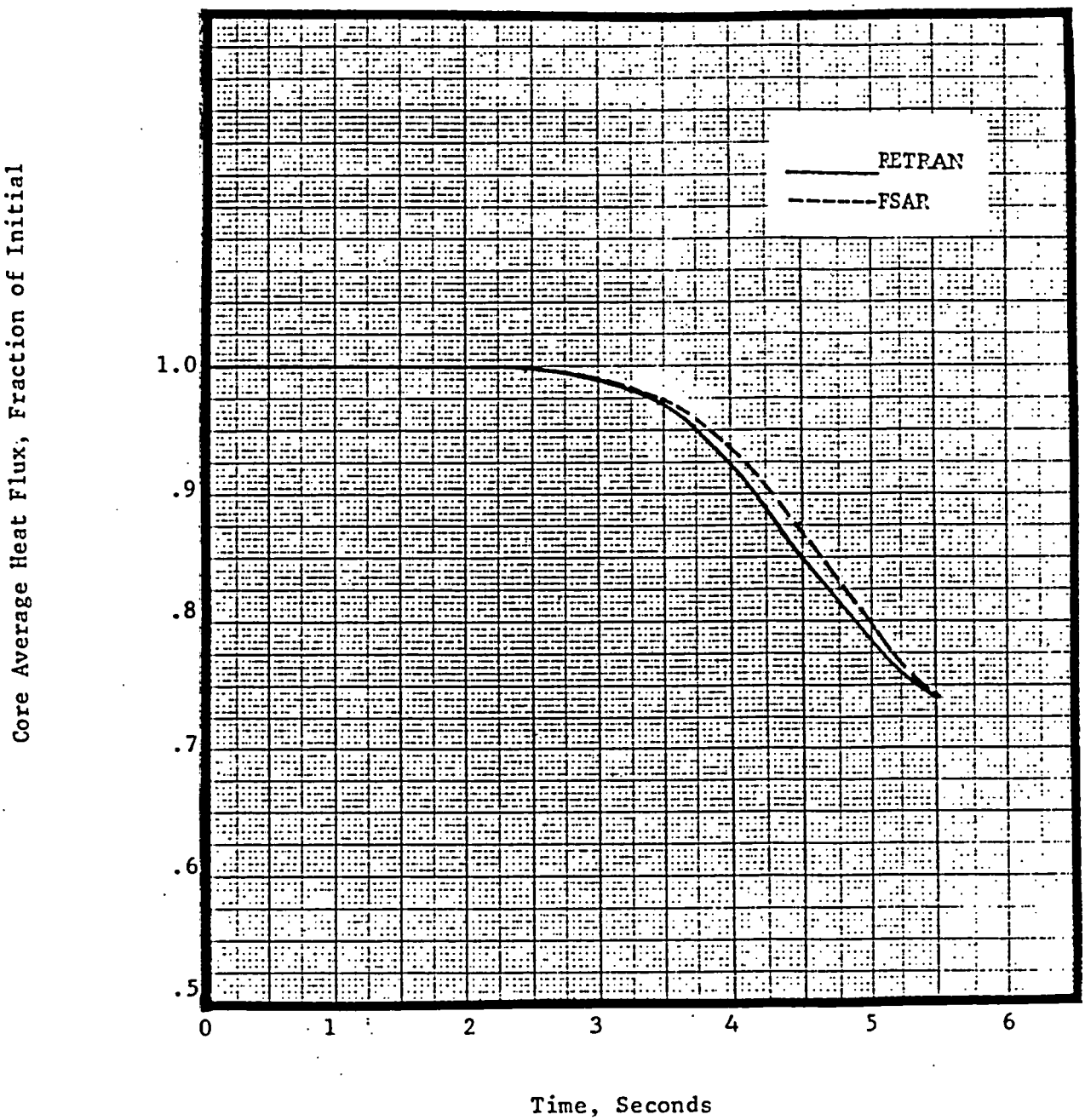
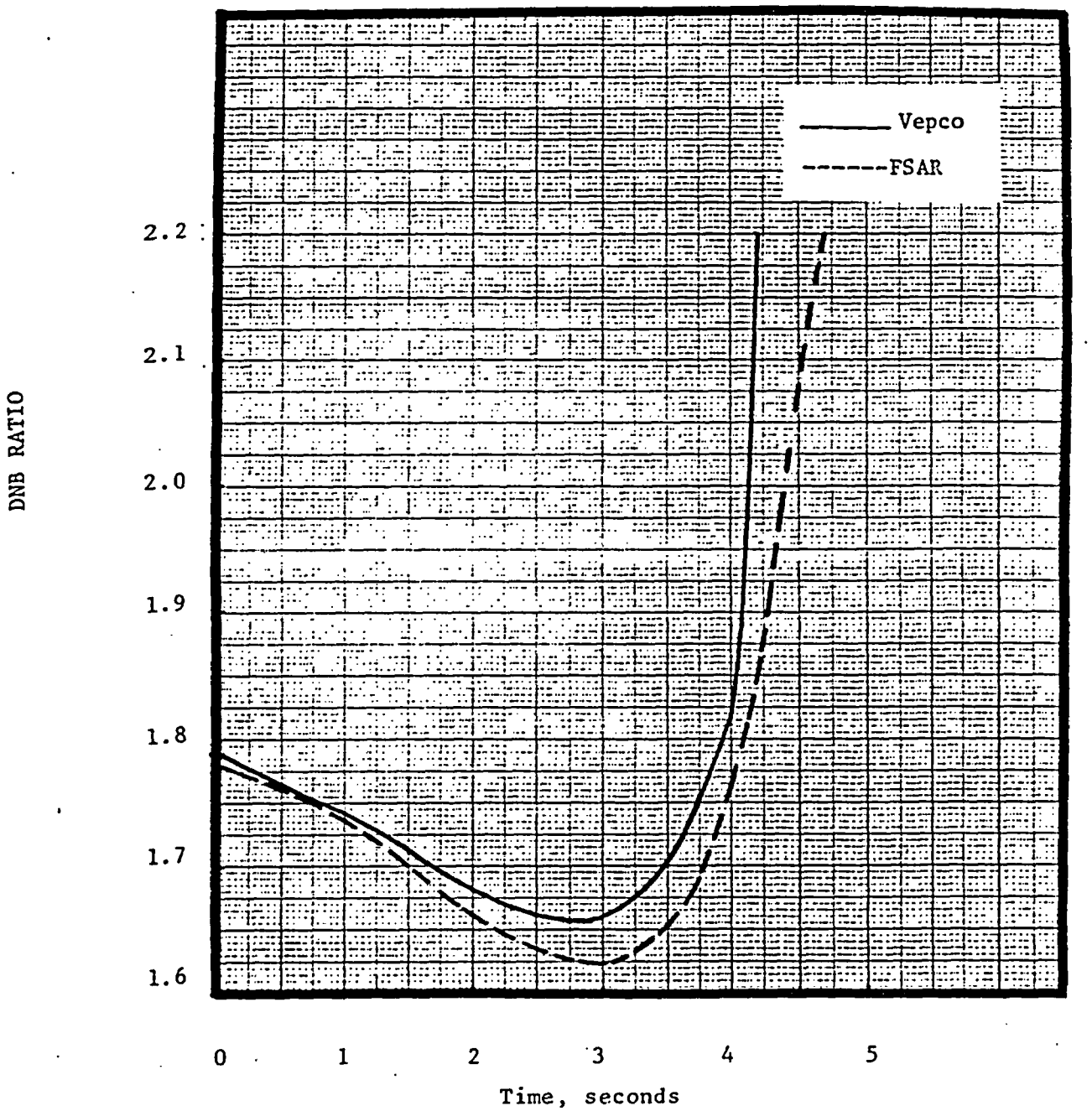


Figure 5.26

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	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

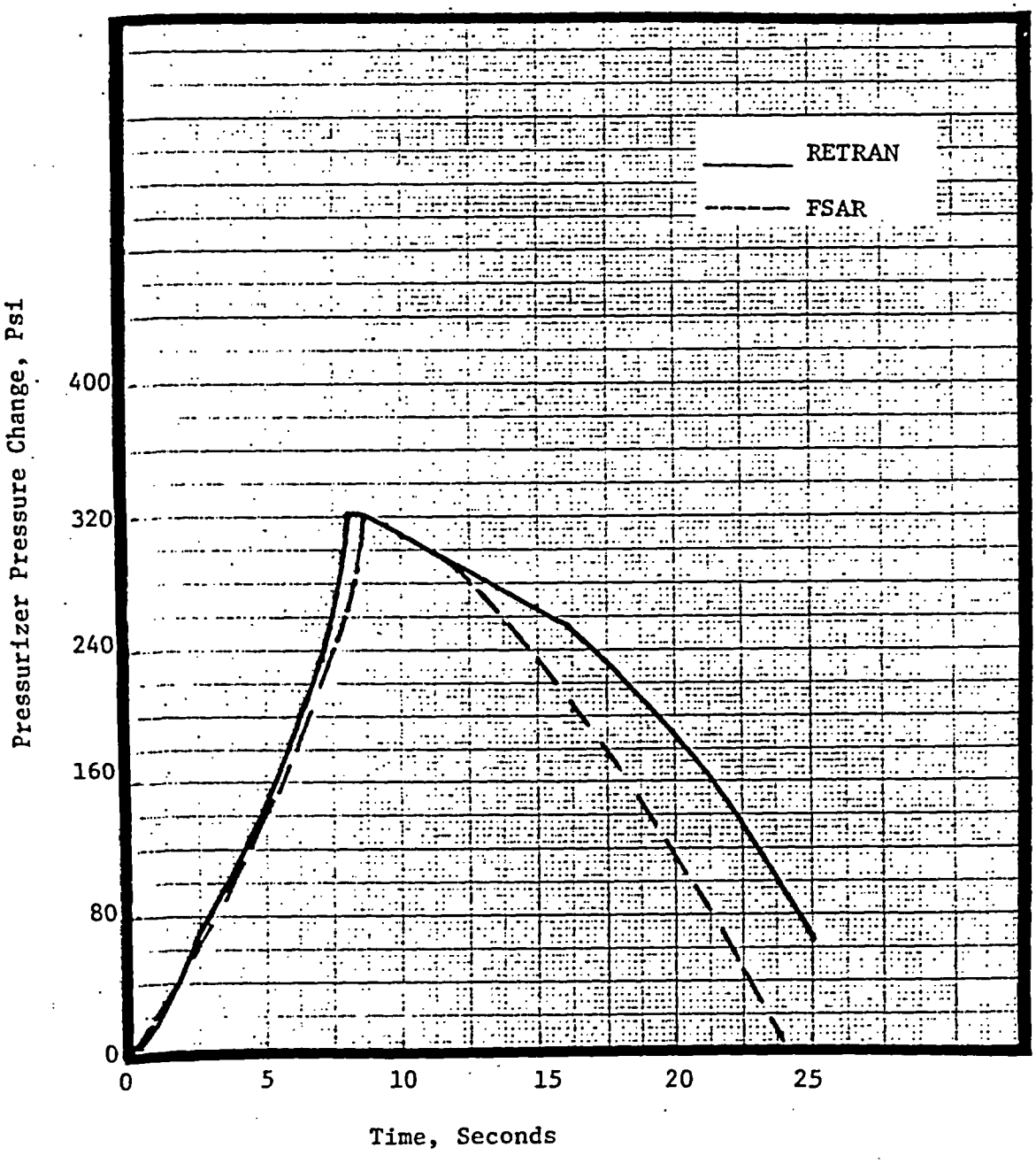


Figure 5.28

NUCLEAR POWER
LOSS OF LOAD TRANSIENT
BOL - FSAR ANALYSIS

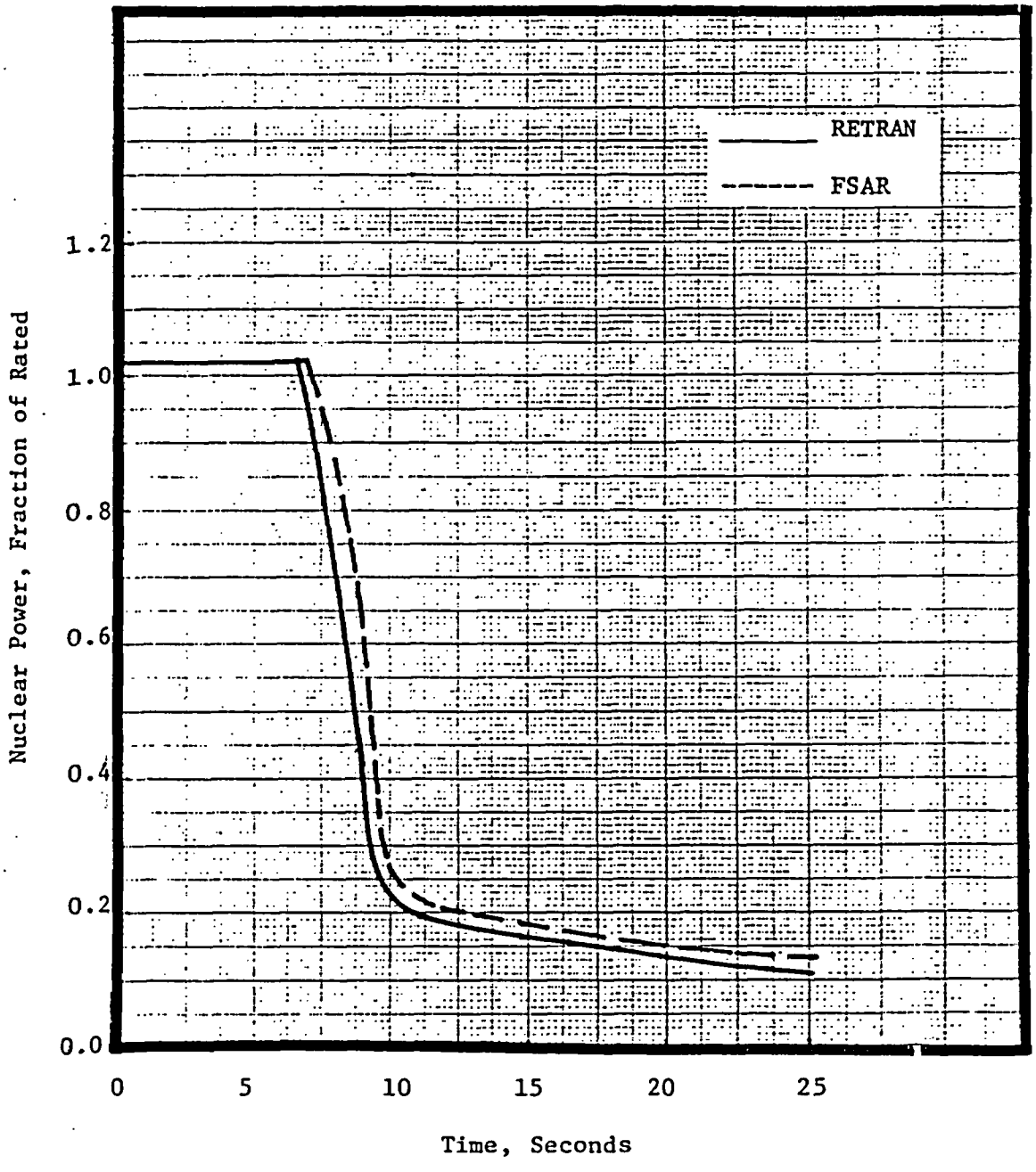


Figure 5.29

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

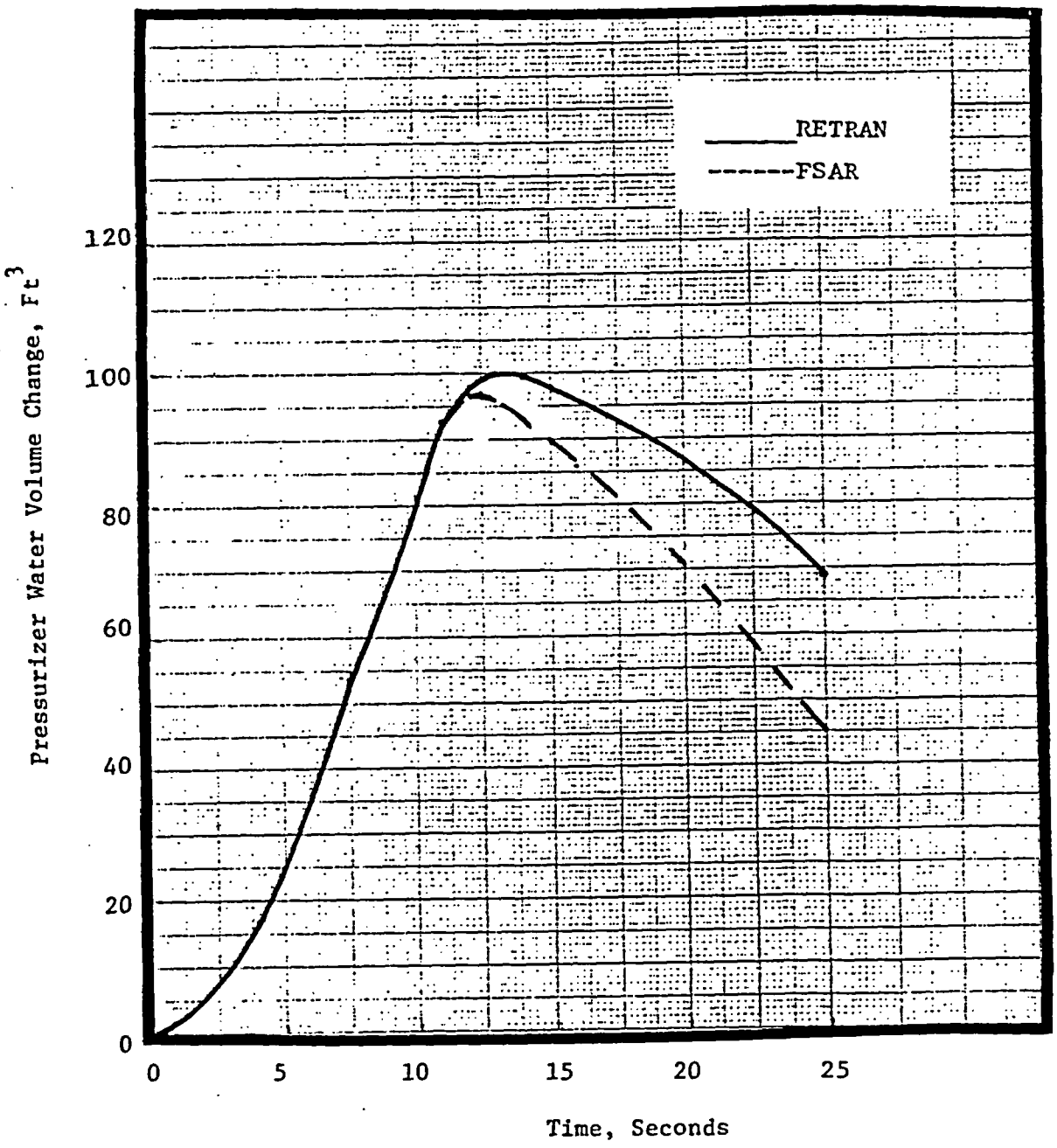


Figure 5.30

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

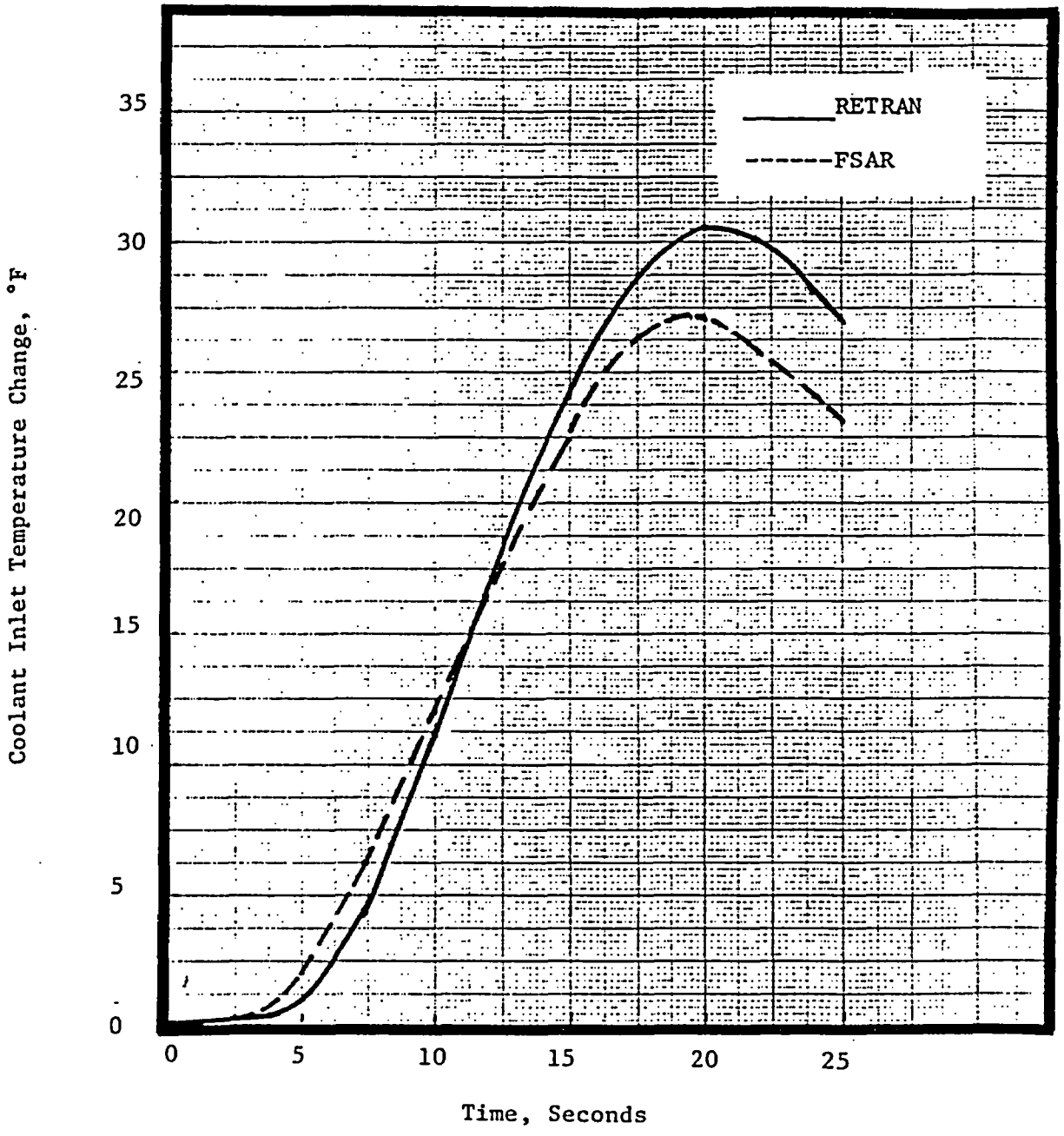


Figure 5.31

DNB RATIO
LOSS OF LOAD TRANSIENT
BOL - FSAR ANALYSIS

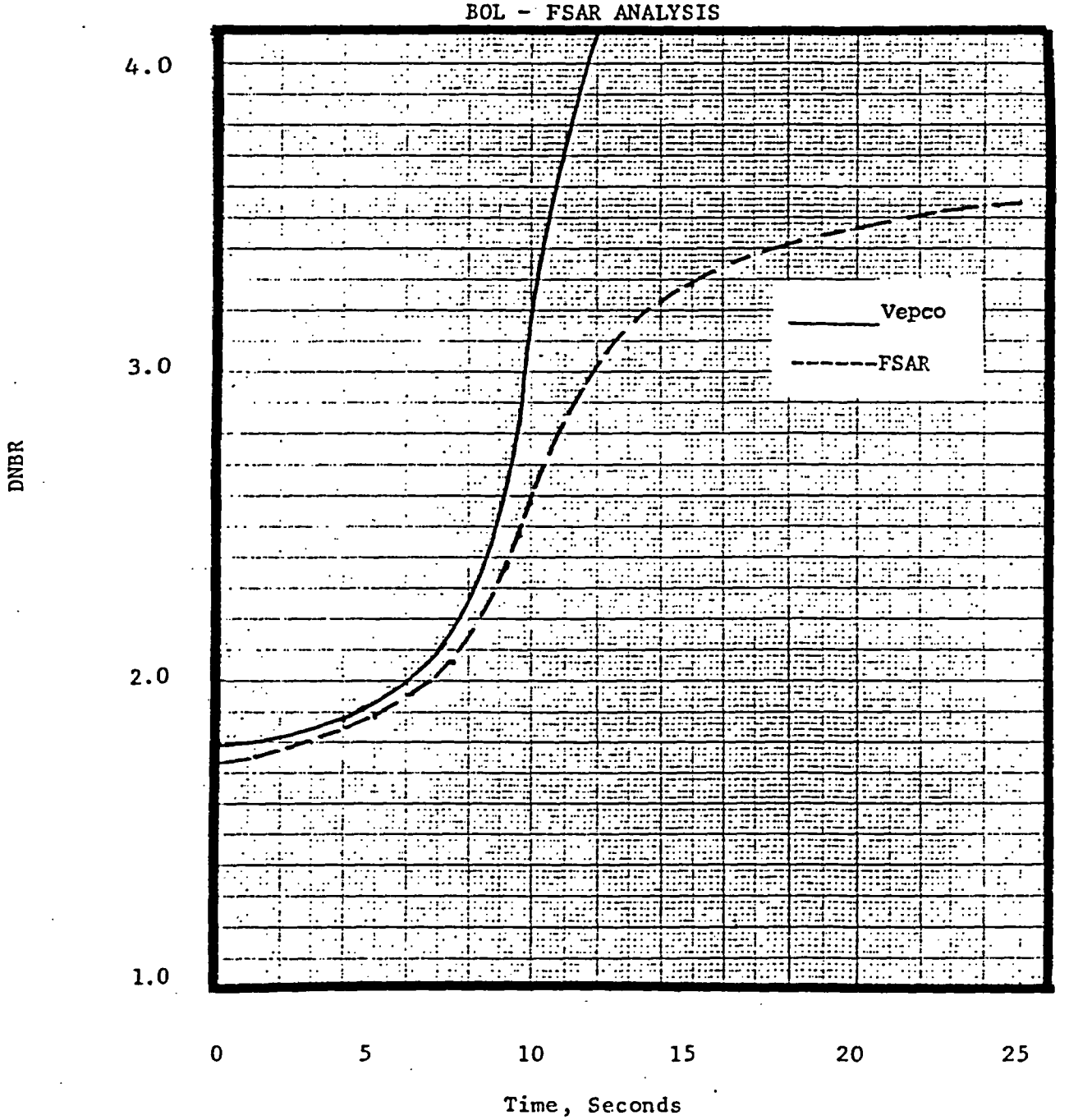


Figure 5.32

PRESSURIZER PRESSURE CHANGE
LOSS OF LOAD TRANSIENT
EOL-FSAR ANALYSIS

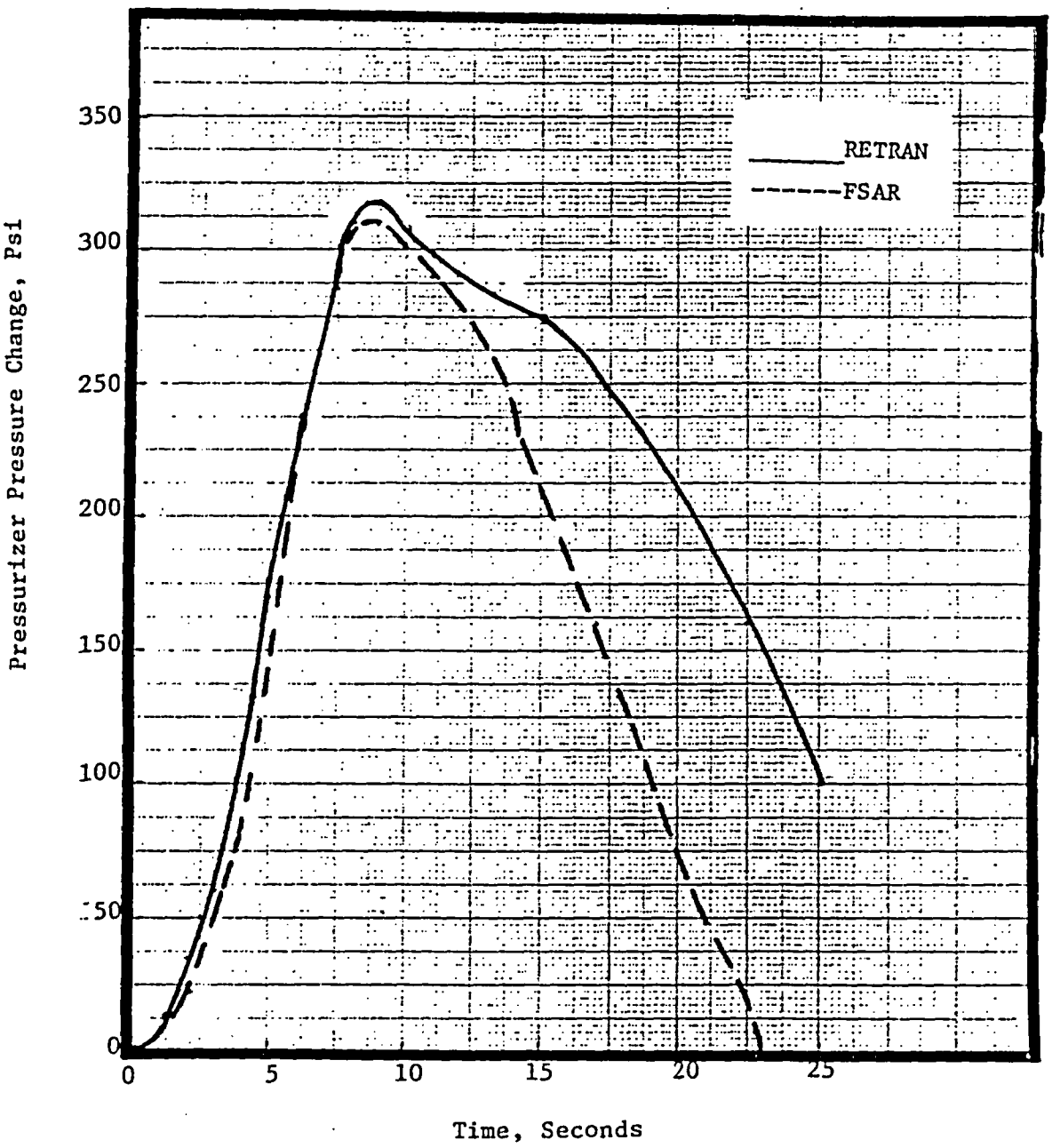


Figure 5.33

NUCLEAR POWER
LOSS OF LOAD TRANSIENT
EOL - FSAR ANALYSIS

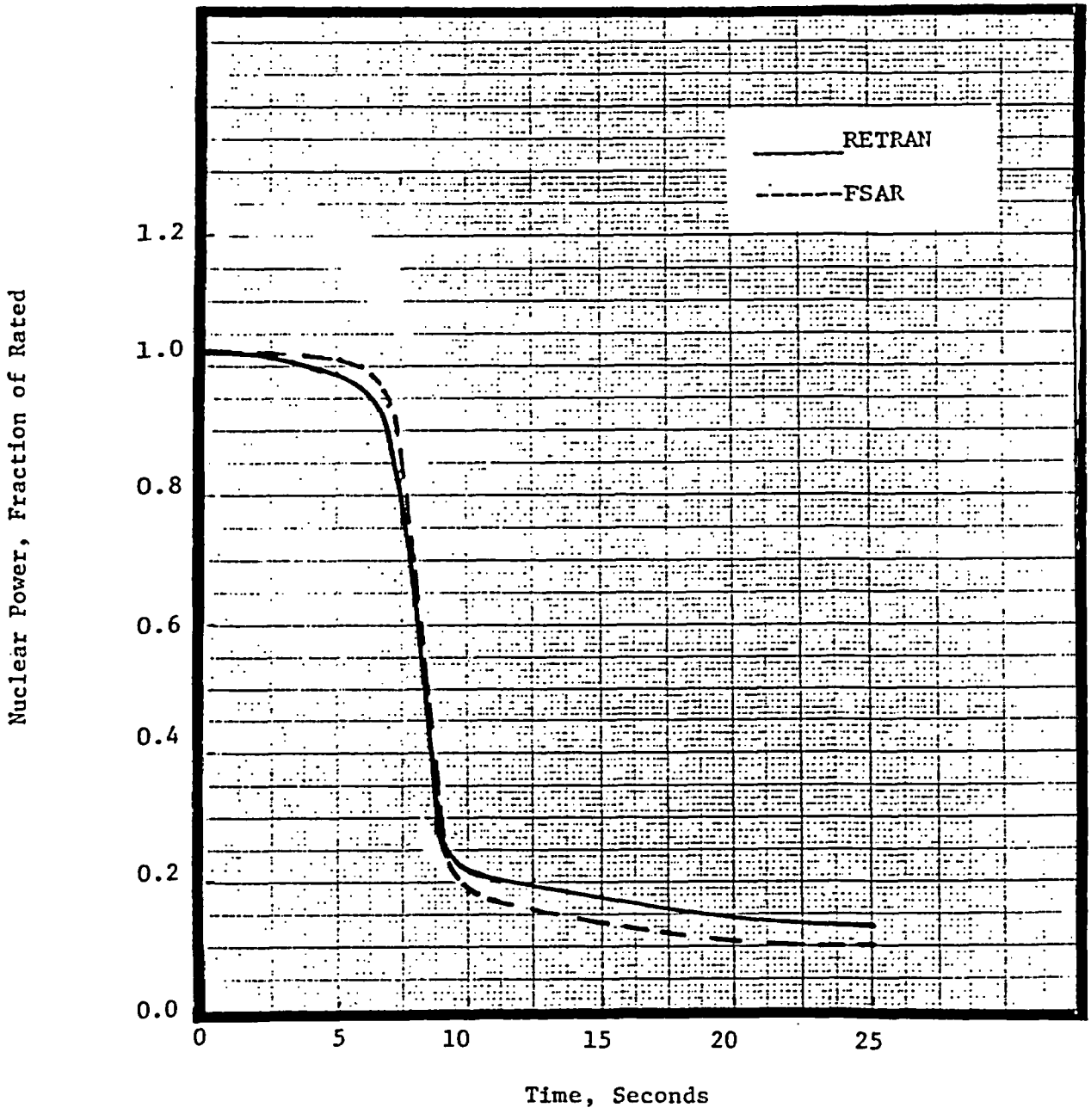


Figure 5.34

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

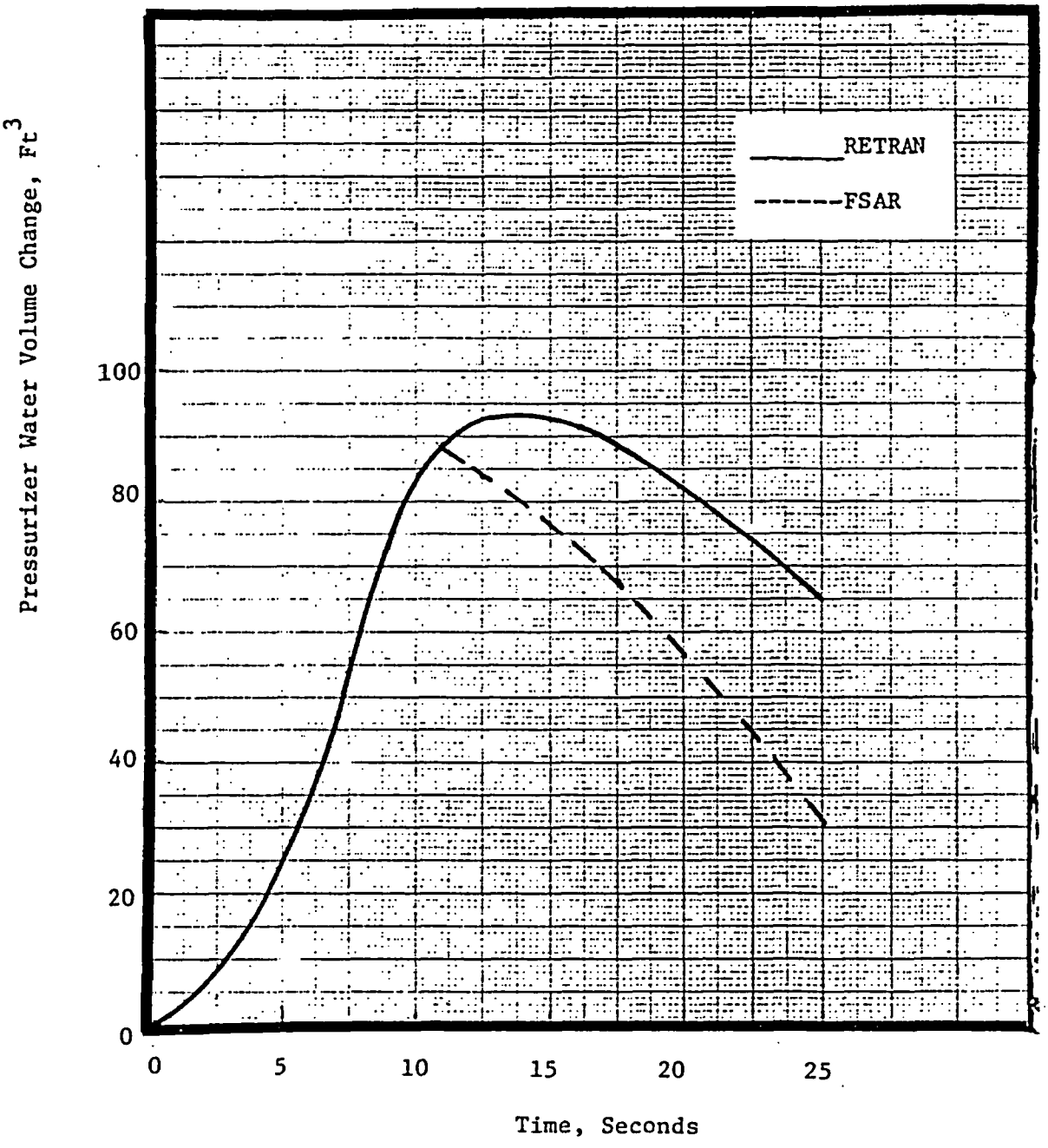


Figure 5.35

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

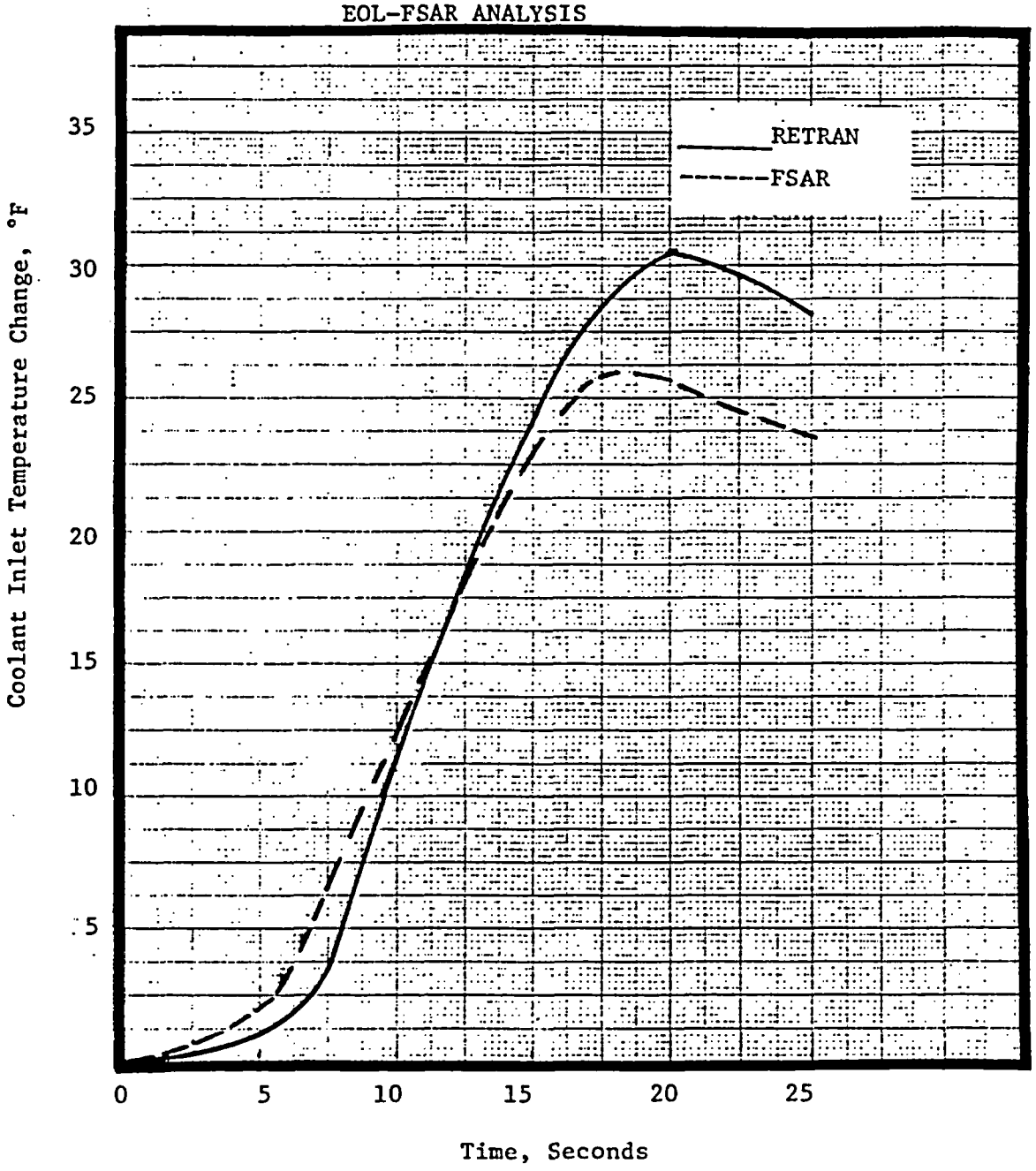


Figure 5.36

DNB RATIO
LOSS OF LOAD TRANSIENT
EOL FSAR ANALYSIS

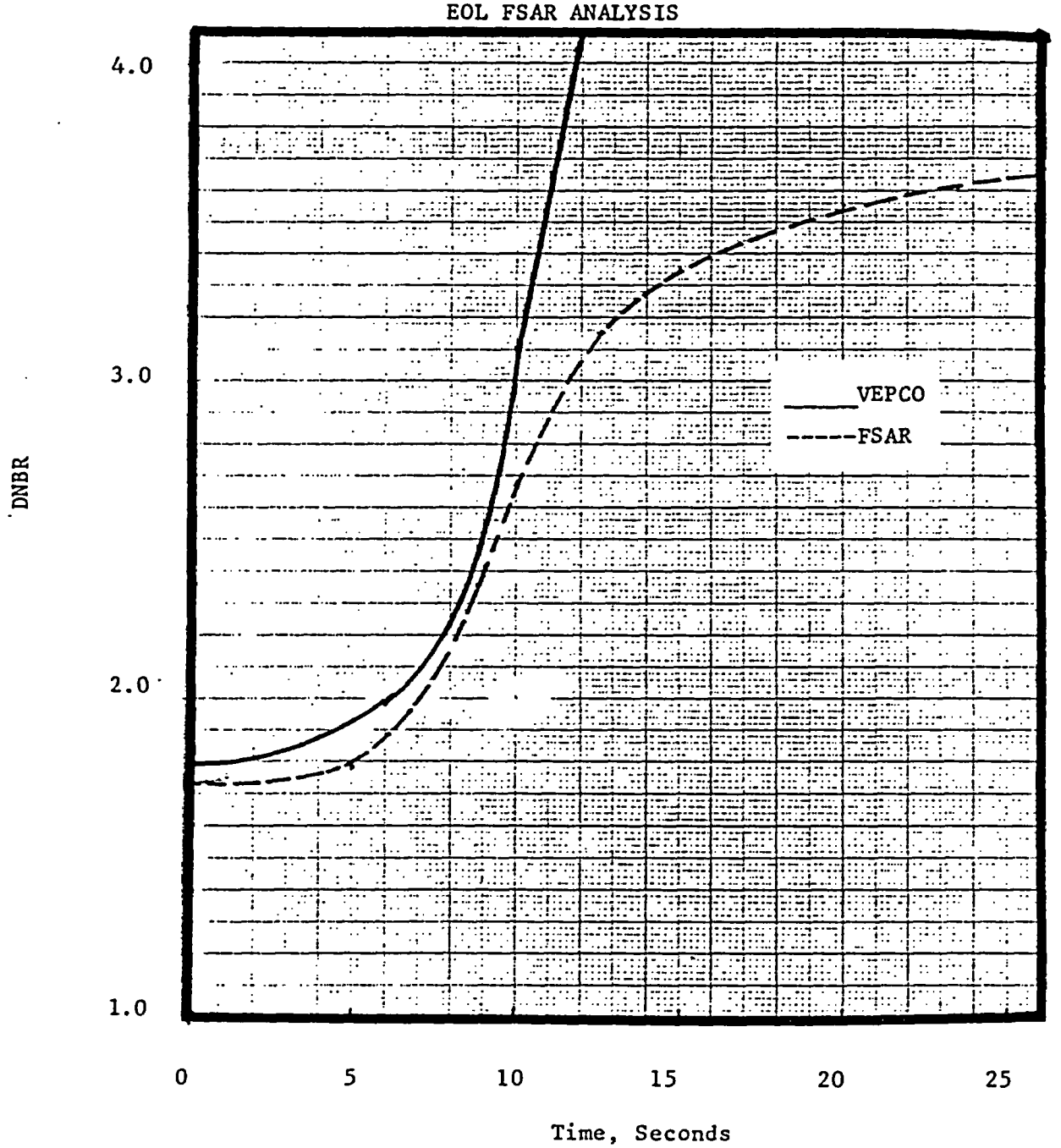


Figure 5.37

PRESSURIZER PRESSURE
LOSS OF LOAD TRANSIENT
POSITIVE MODERATOR COEFFICIENT REANALYSIS

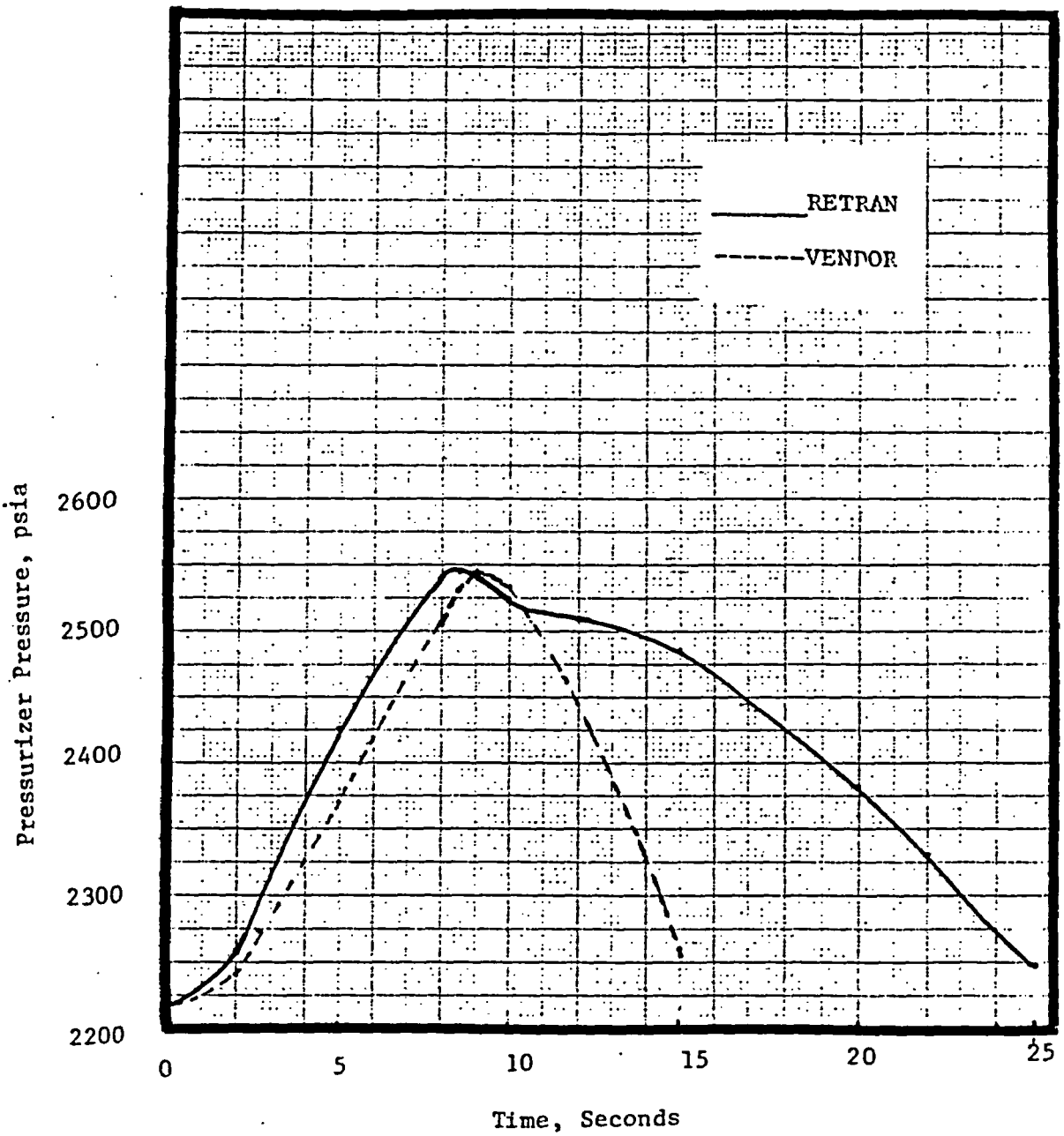


Figure 5.38

NUCLEAR POWER
LOSS OF LOAD TRANSIENT

POSITIVE MODERATOR COEFFICIENT REANALYSIS

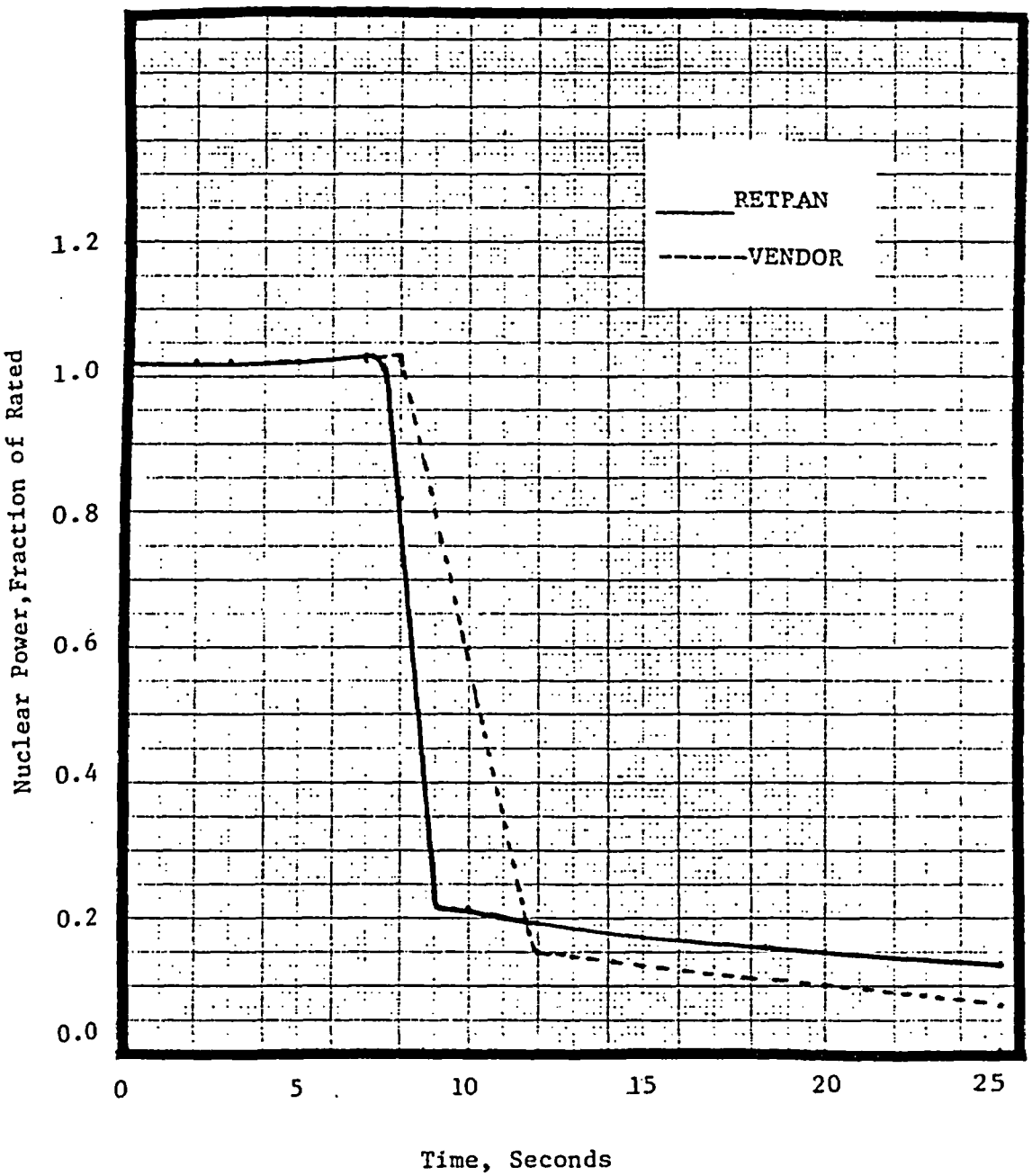


Figure 5.39

AVERAGE COOLANT TEMPERATURE
LOSS OF LOAD TRANSIENT
POSITIVE MODERATOR COEFFICIENT REANALYSIS

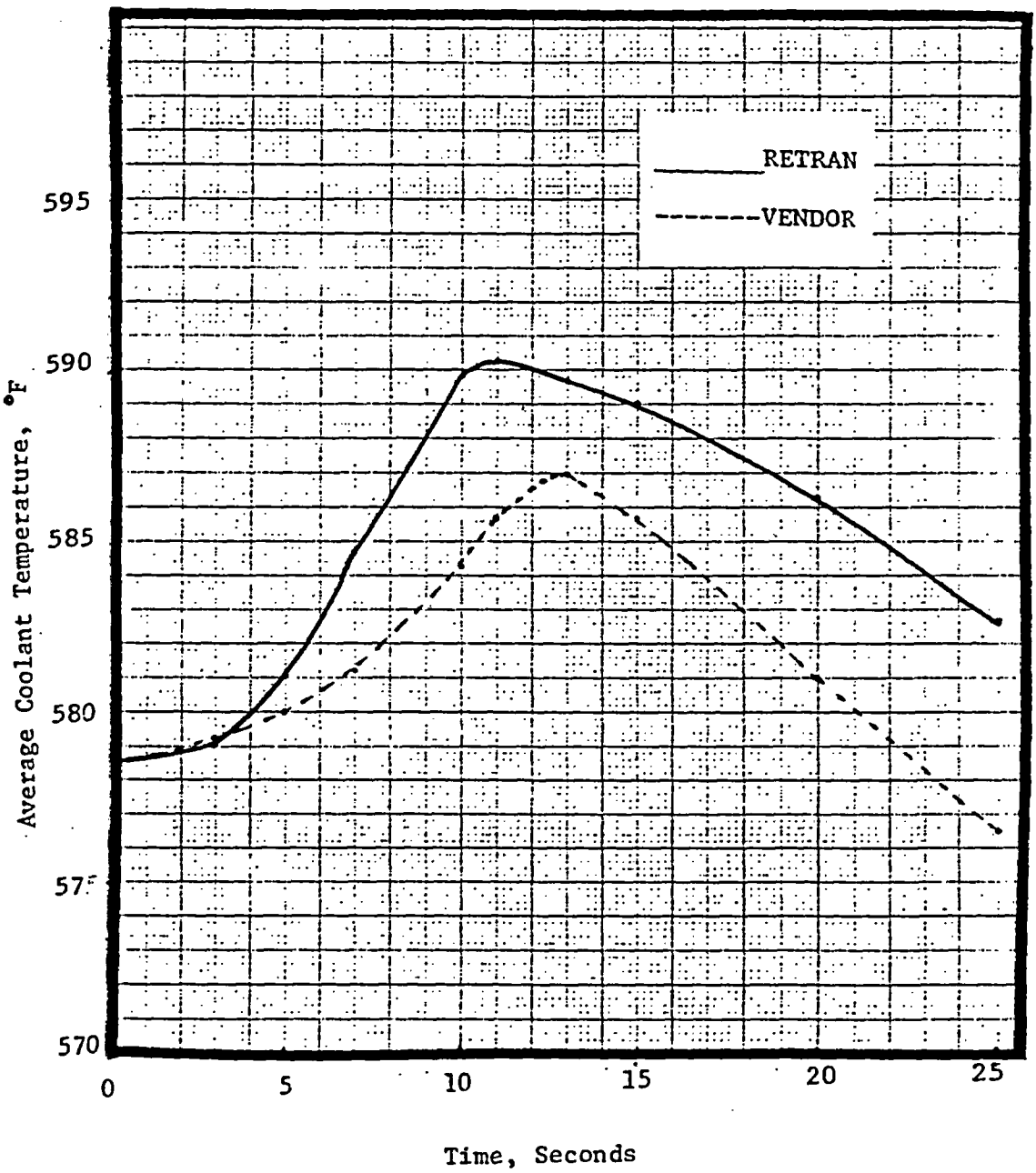


Figure 5.40

DNB RATIO
LOSS OF LOAD TRANSIENT
POSITIVE MODERATOR COEFFICIENT ASSUMPTION REANALYSIS

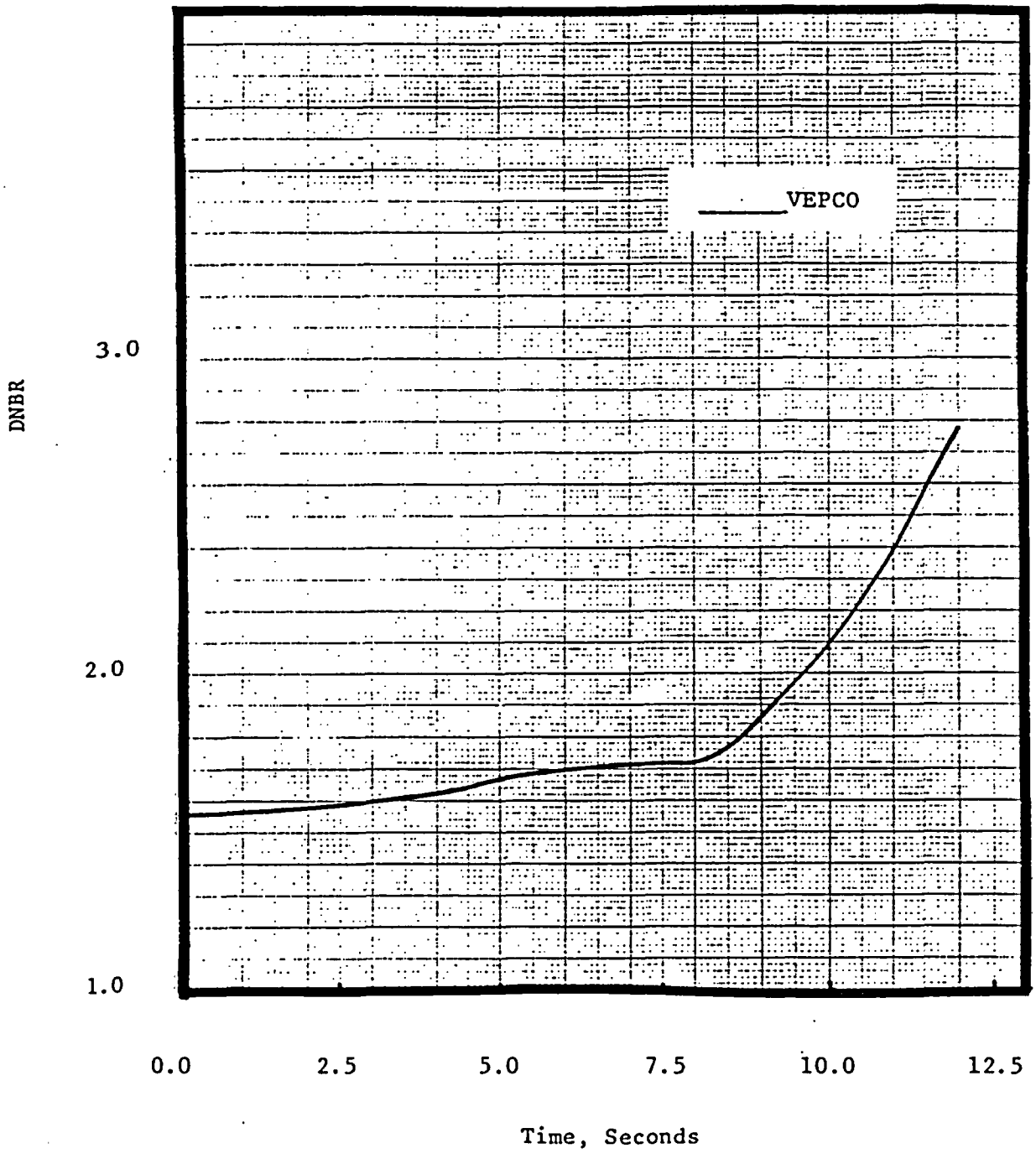


Figure 5.41

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

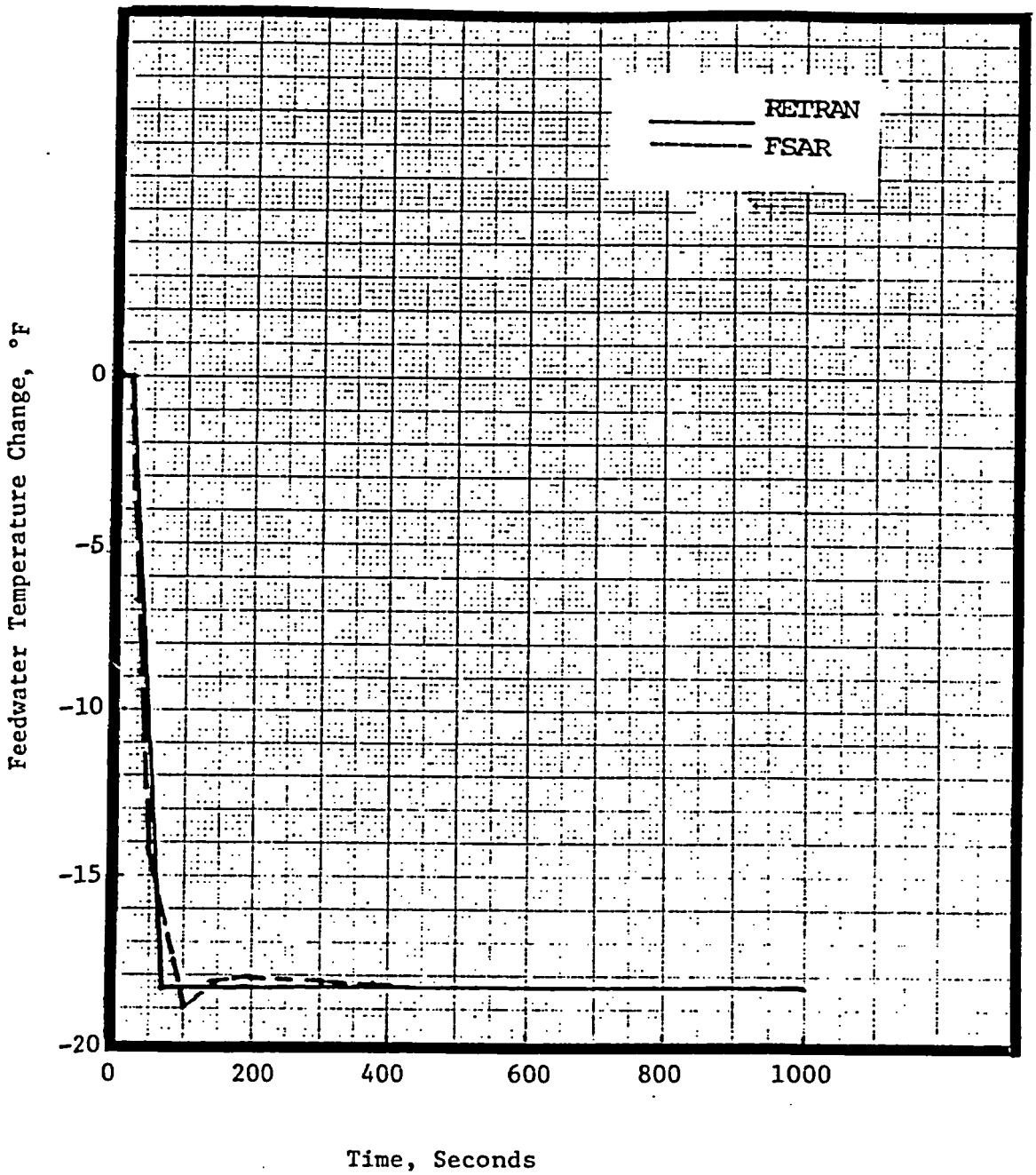


Figure 5.42

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

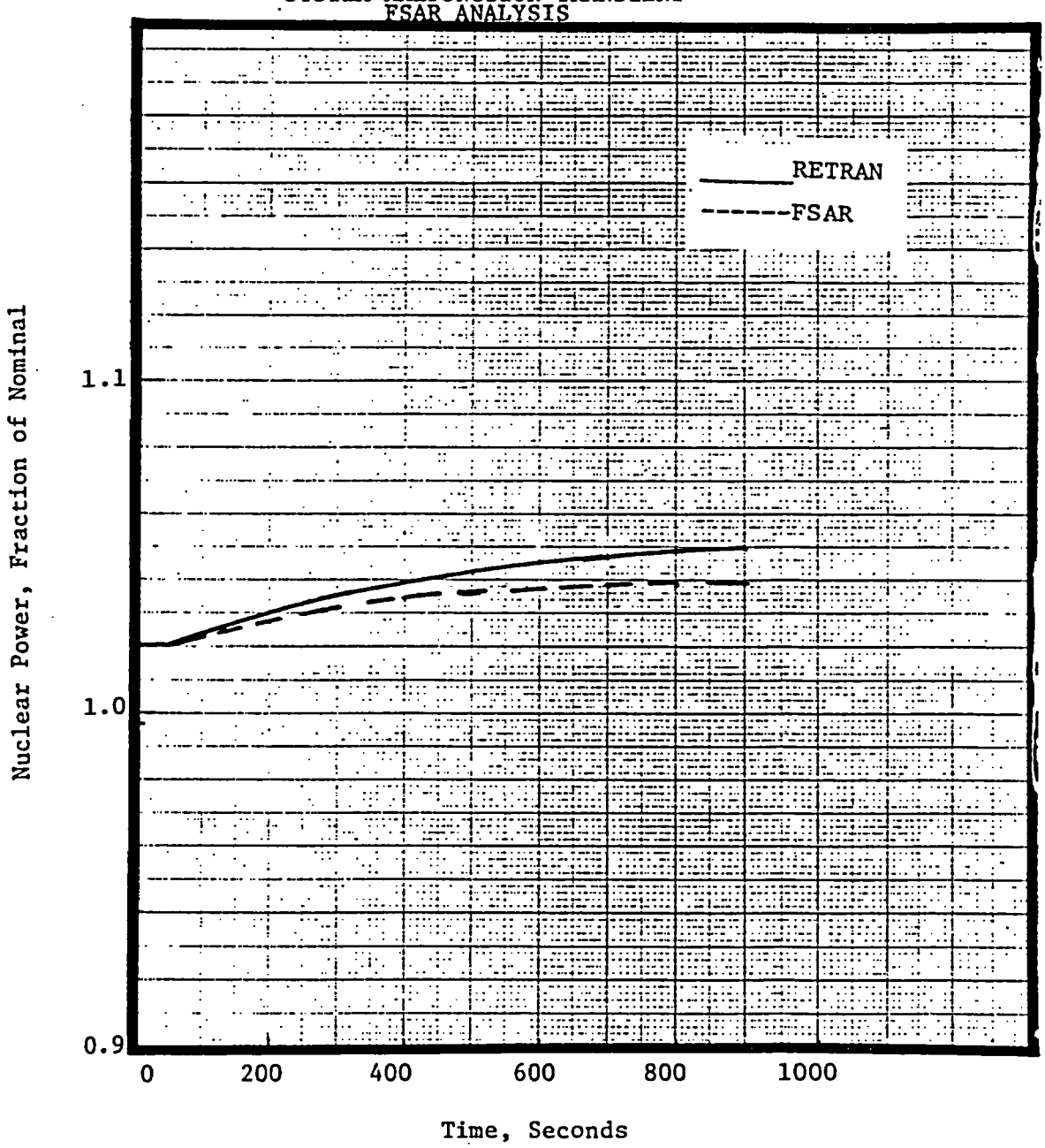


Figure 5.43

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

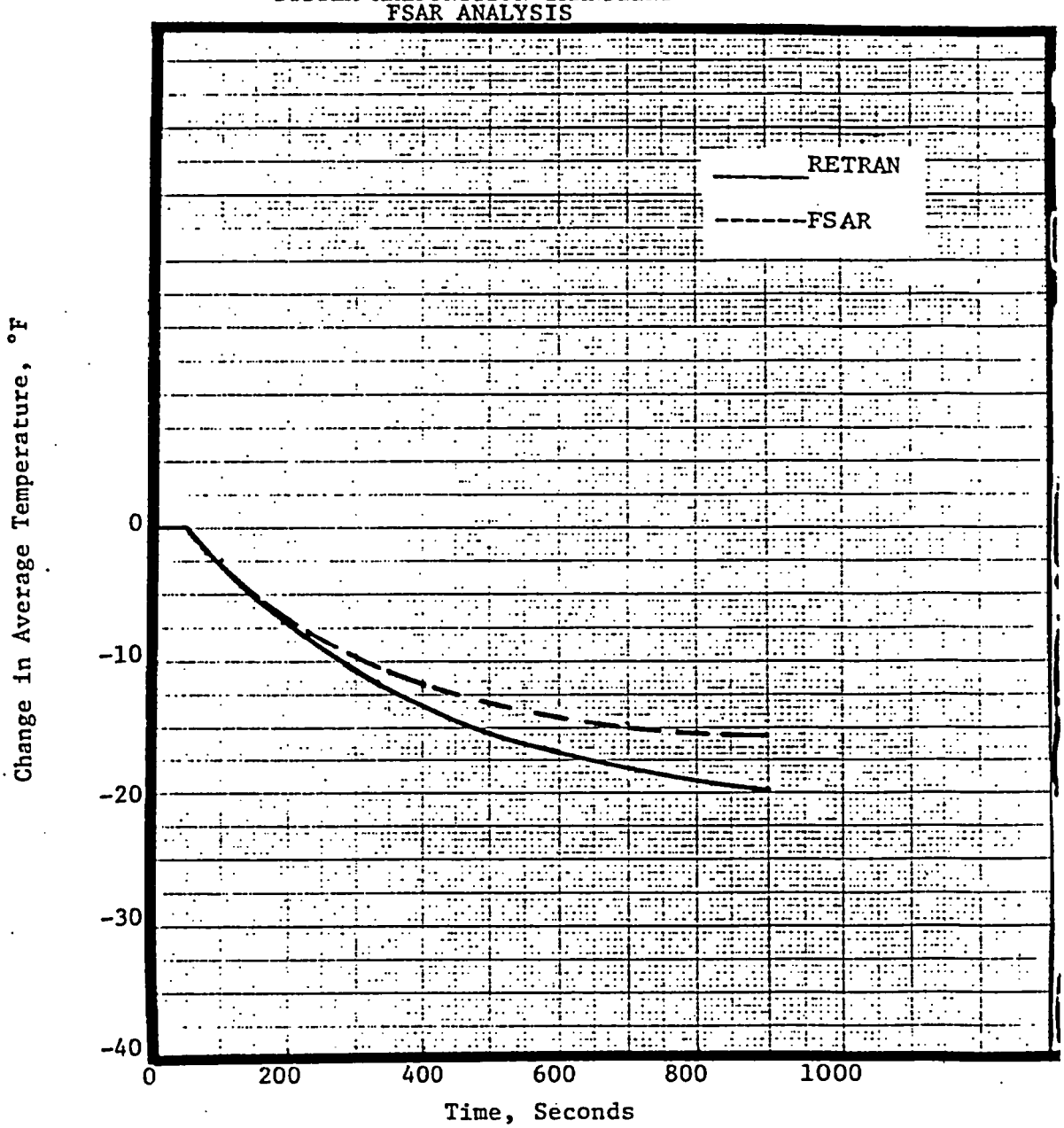


Figure 5.44

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

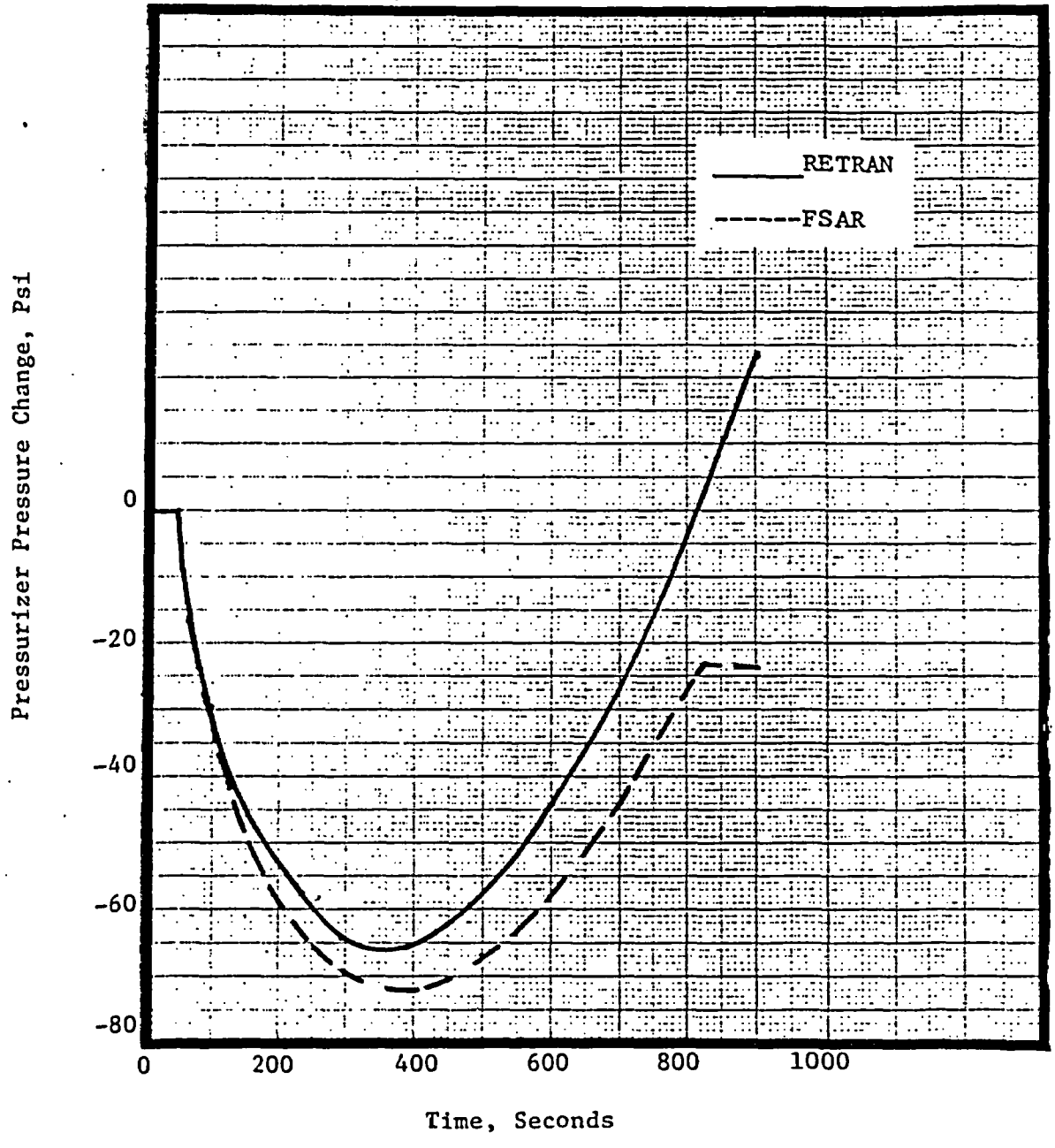


Figure 5.45

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

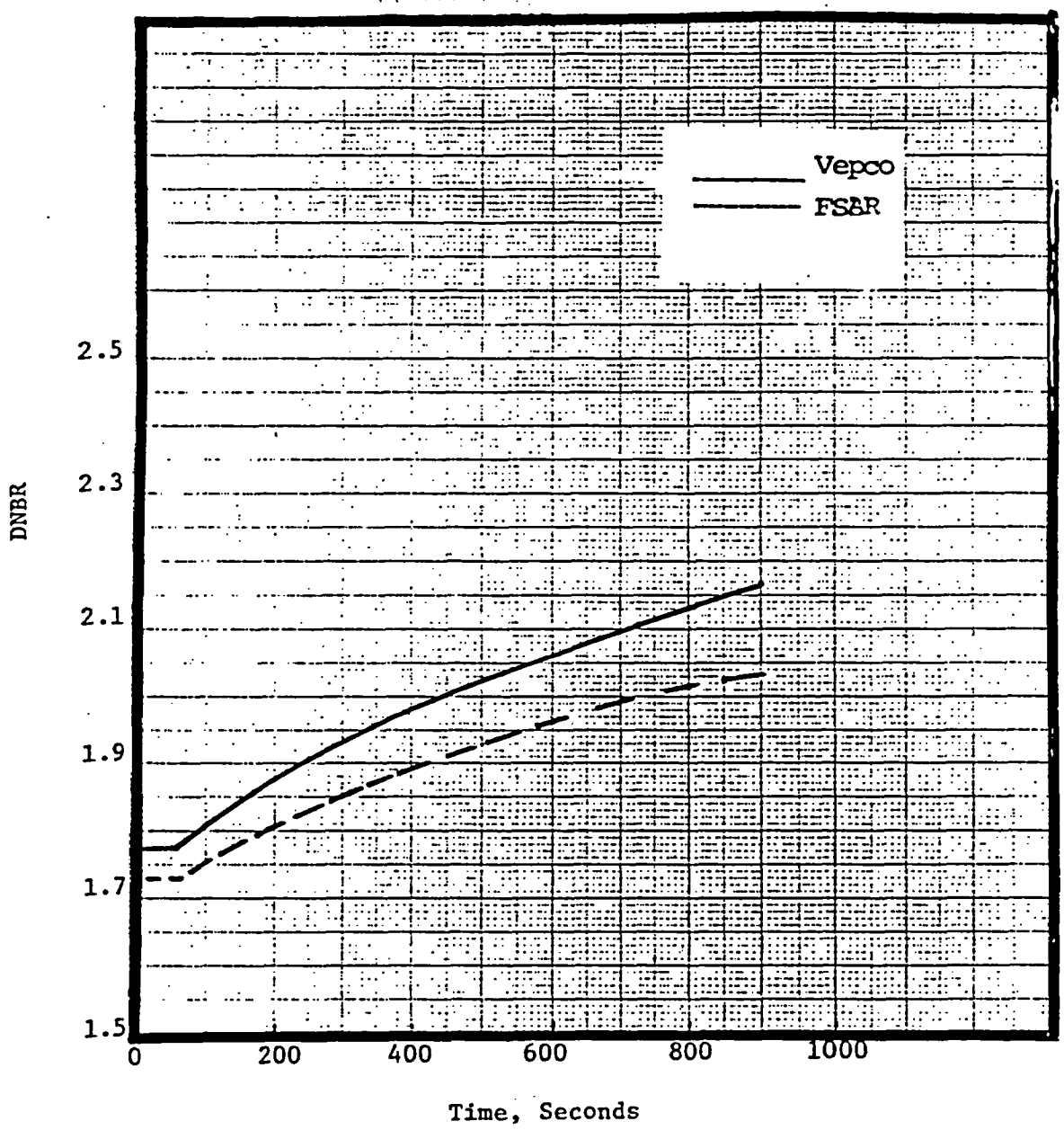


Figure 5.46

BREAK FLOW RATE
MAIN STEAM LINE BREAK
FSAR ANALYSIS

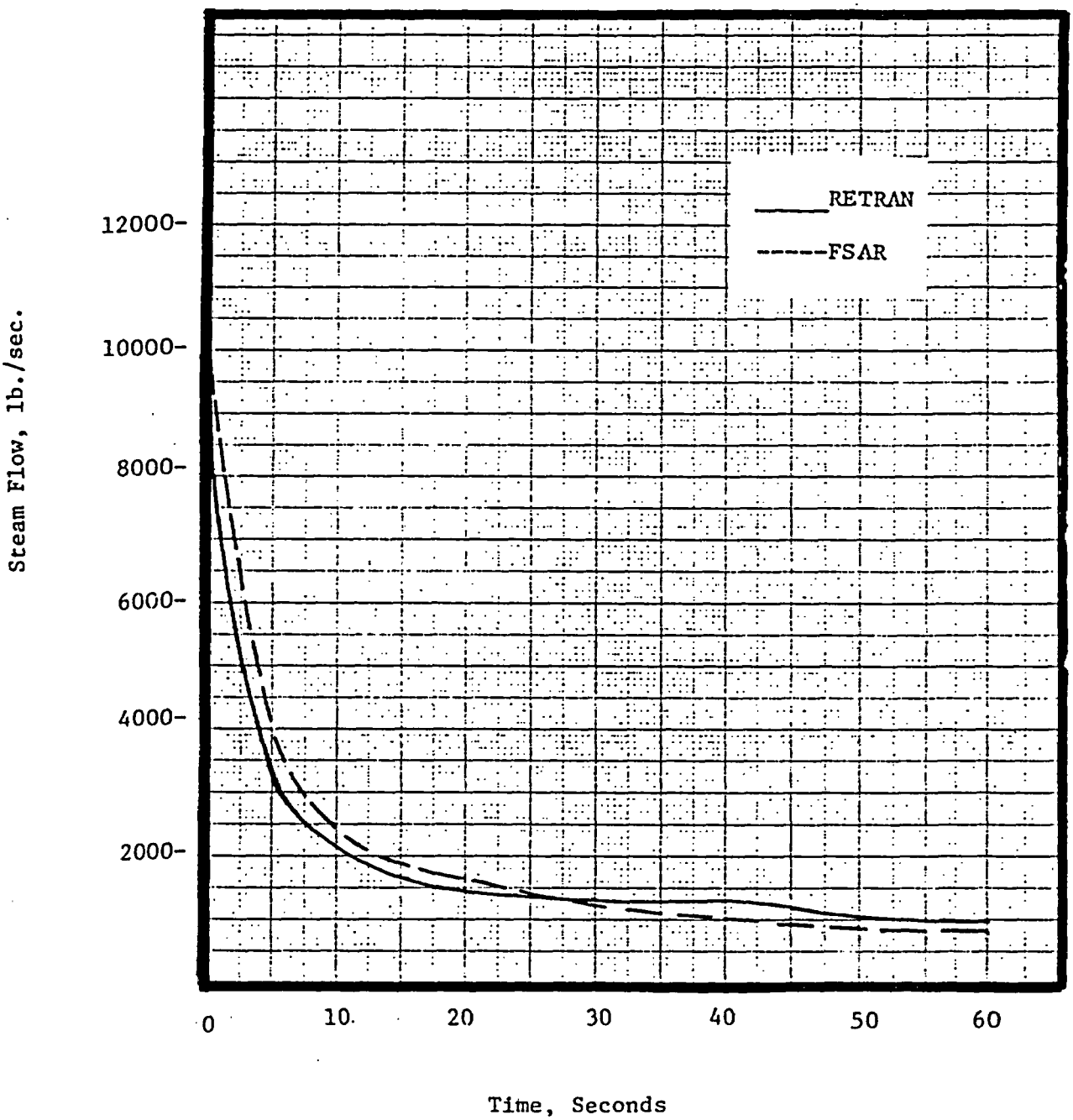


Figure 5.47

PRESSURIZER PRESSURE
MAIN STEAM LINE BREAK
FSAR ANALYSIS

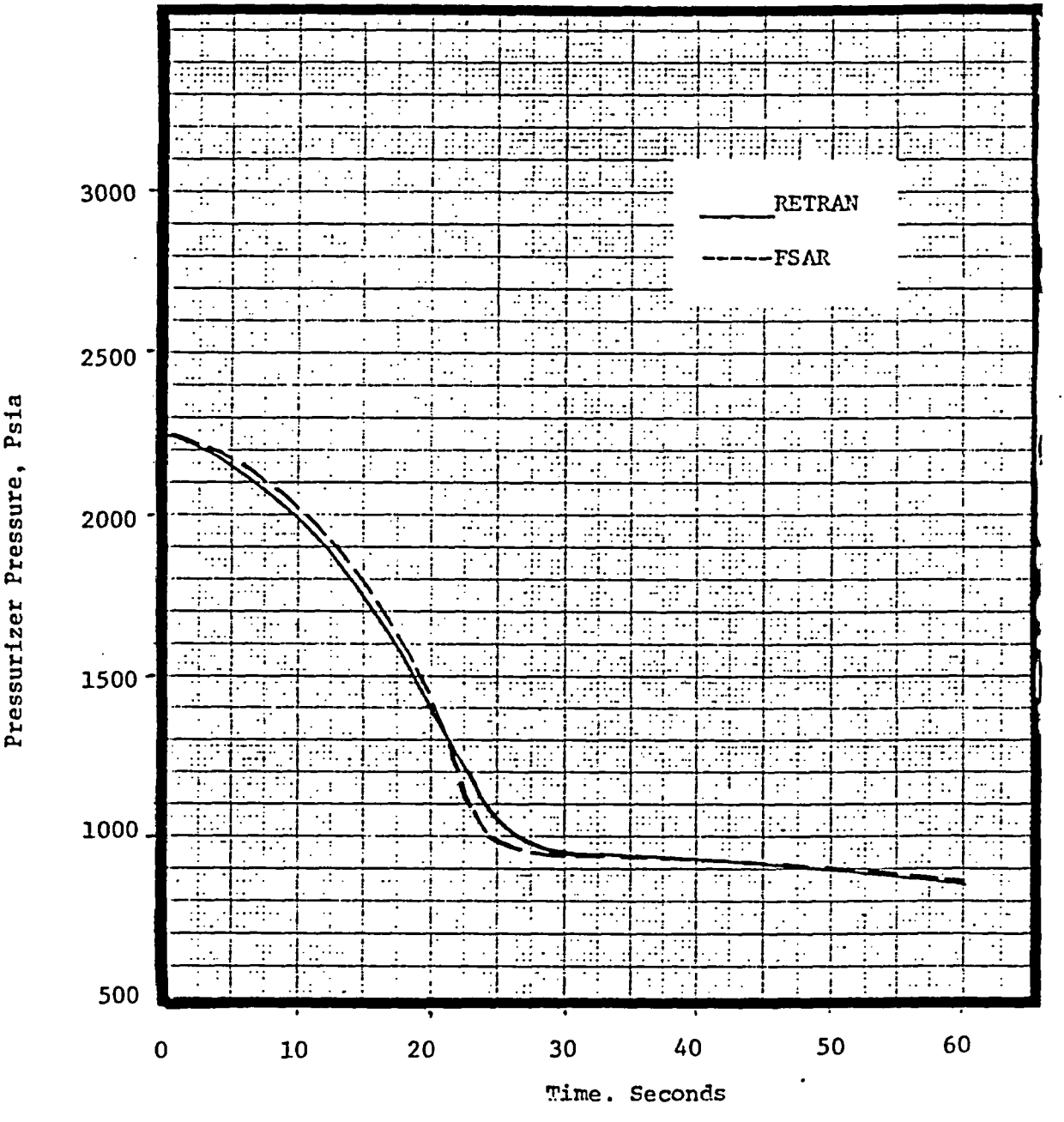


Figure 5.48

TOTAL REACTIVITY
MAIN STEAM LINE BREAK
FSAR ANALYSIS

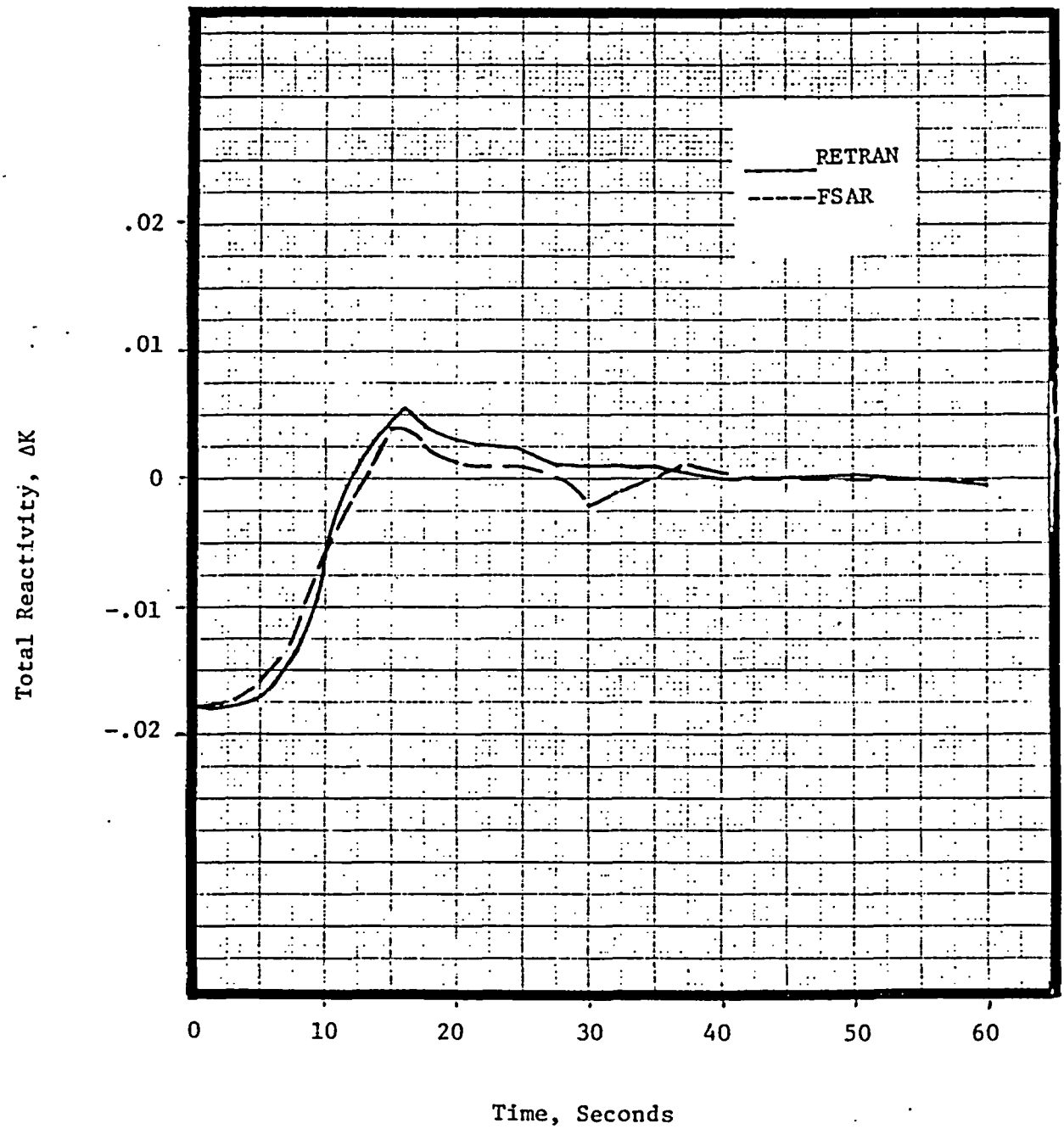


Figure 5.49

CORE HEAT FLUX
MAIN STEAM LINE BREAK
FSAR ANALYSIS

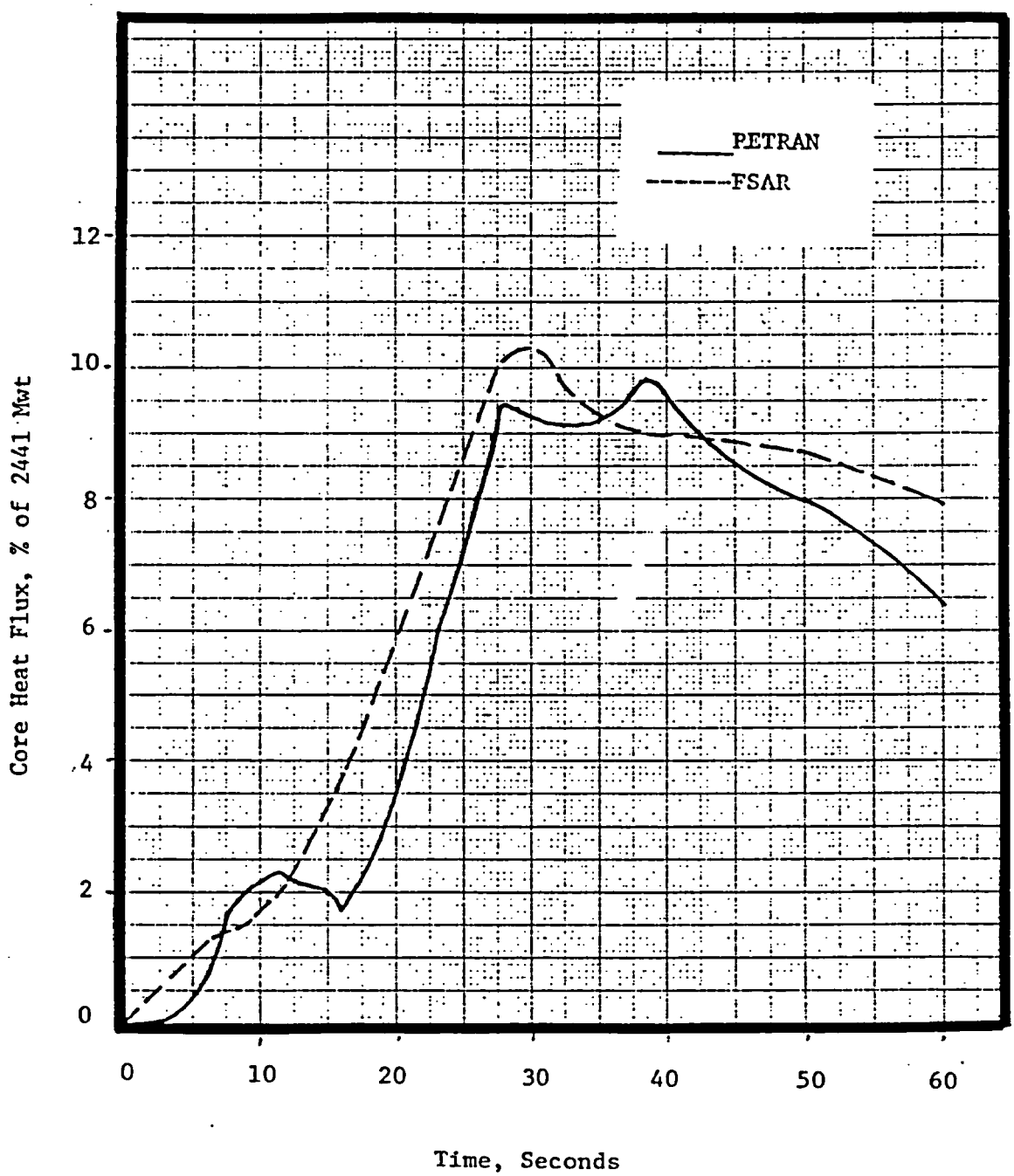


Figure 5.50

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

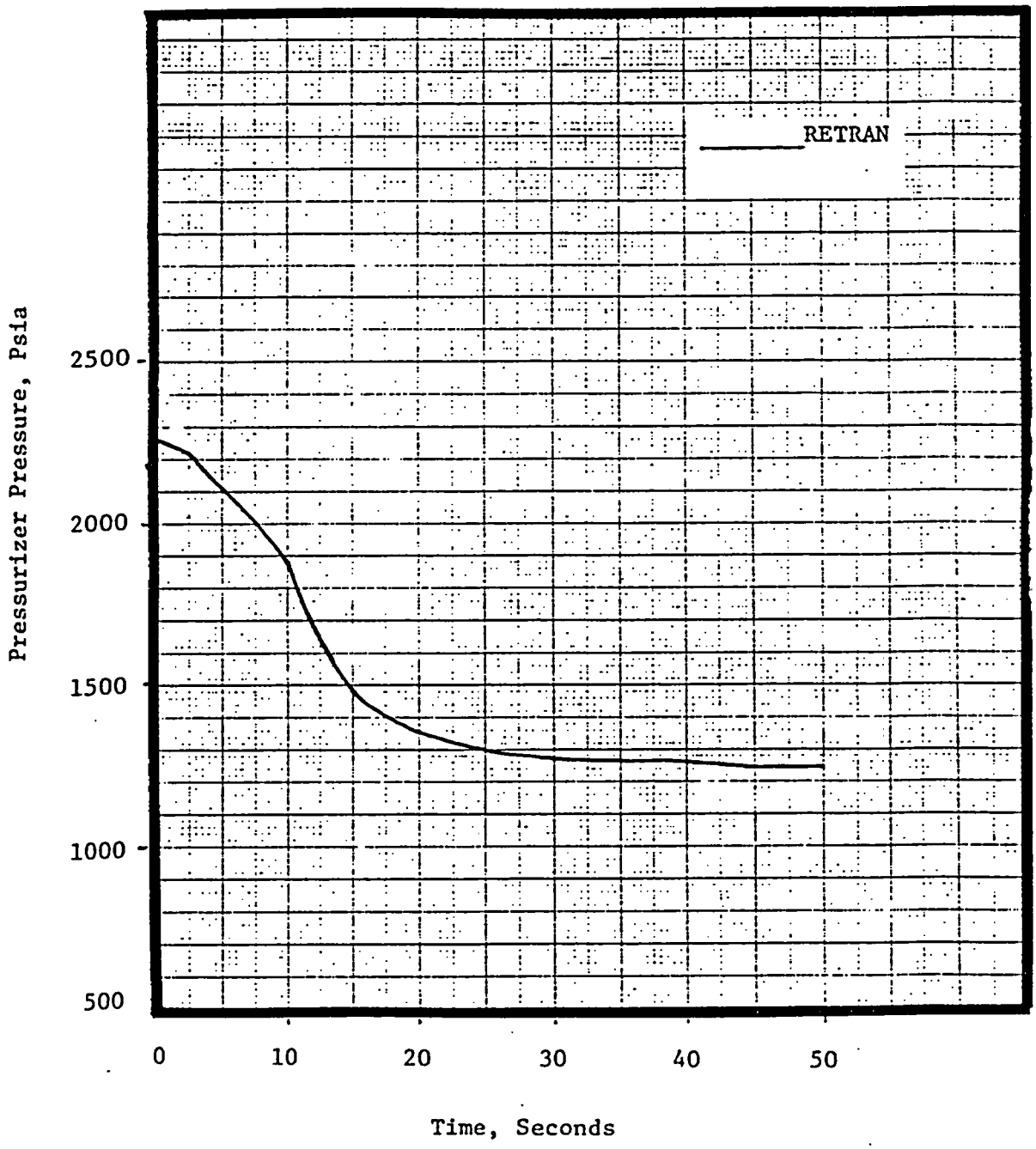
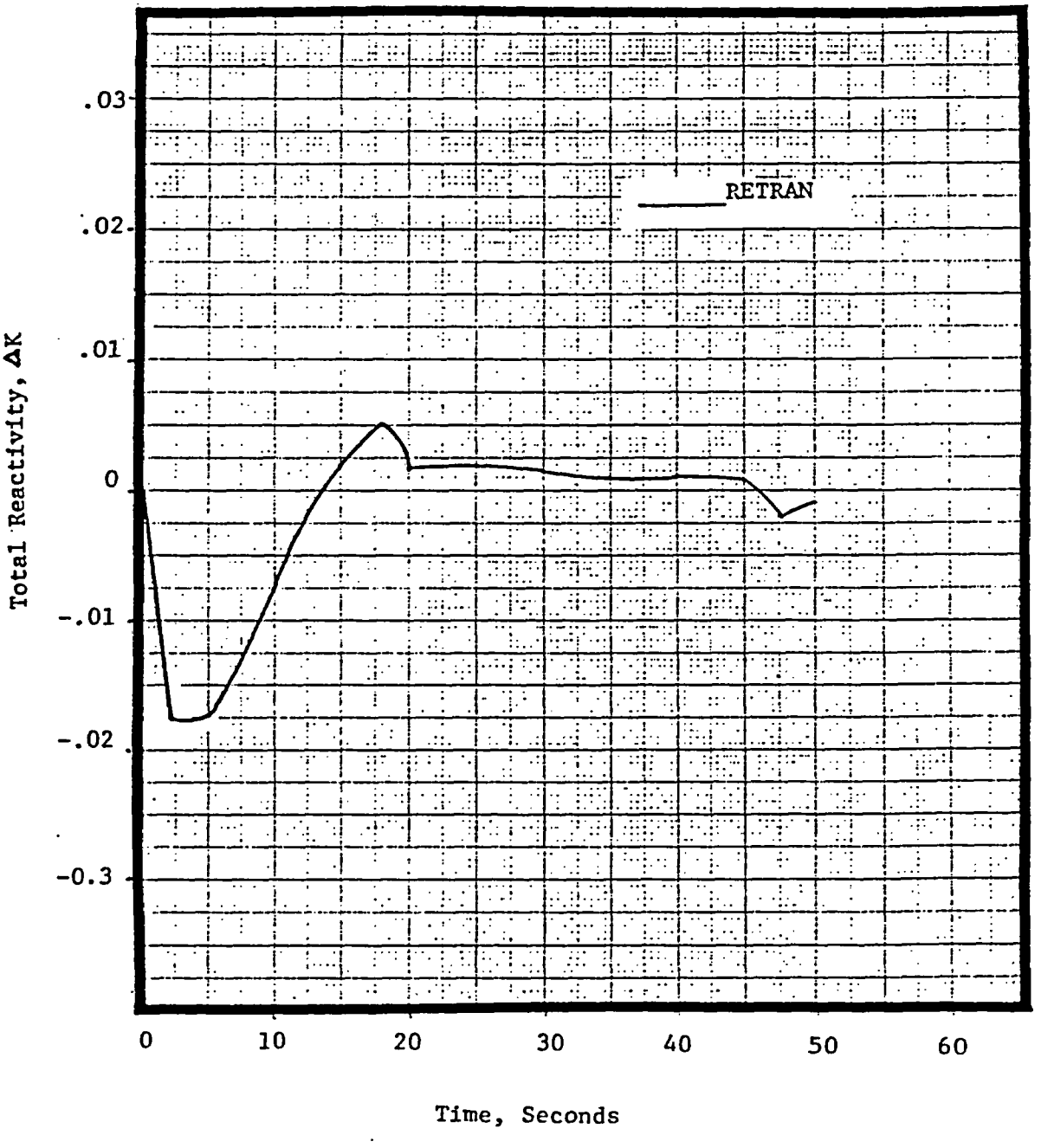


Figure 5.51

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



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 5.52

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

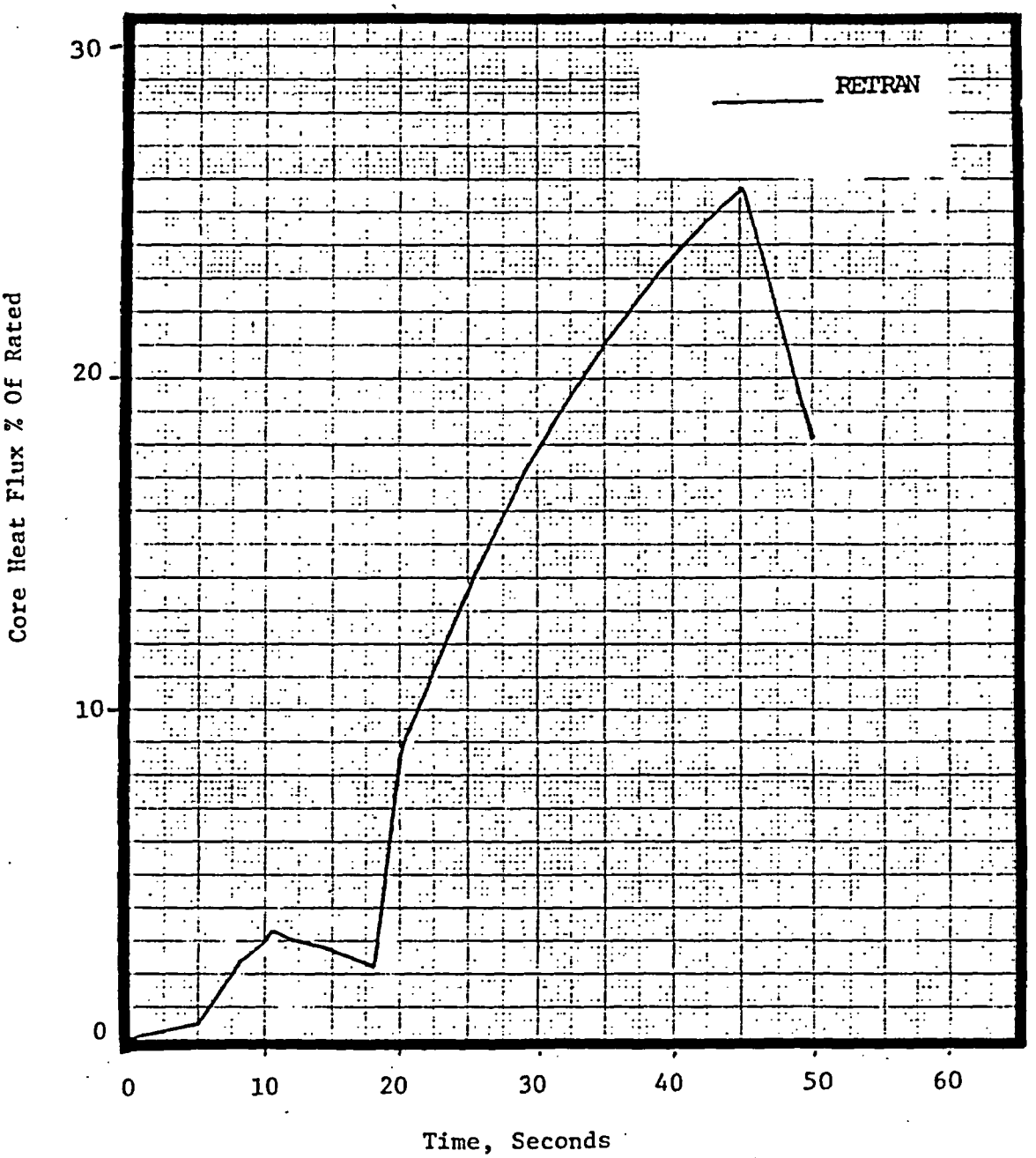


Figure 5.54 compares analytical predictions made with the Surry Two Loop Model with test data for a single pump (two pumps remaining at full speed) coastdown. The data are presented in terms of loop flows. As may be seen, the RETRAN predictions are in close agreement with the data. The same data are 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 ~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 General Conclusions-Best Estimate Transient Analyses

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

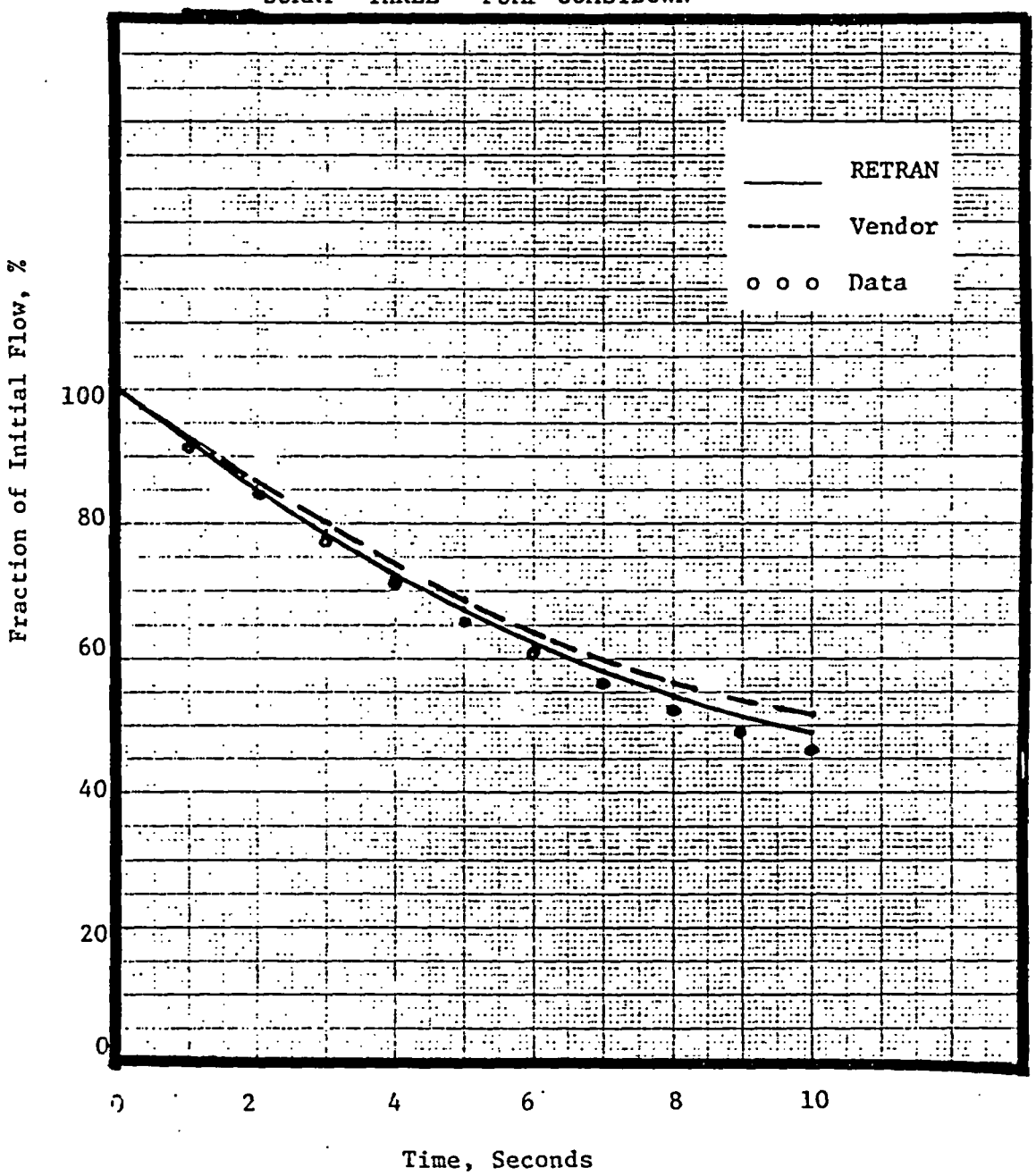


Figure 5.54

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

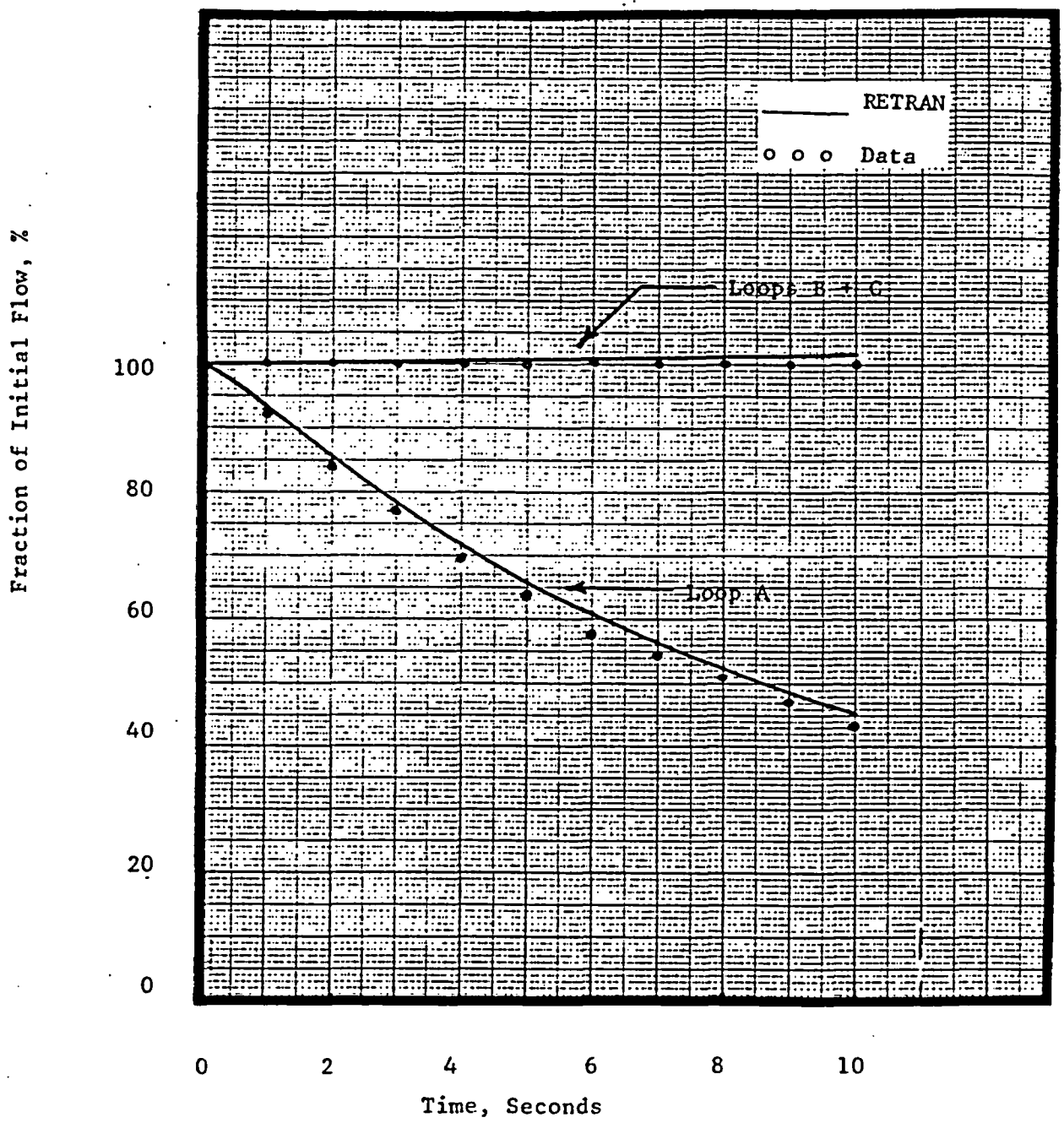


Figure 5.55

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

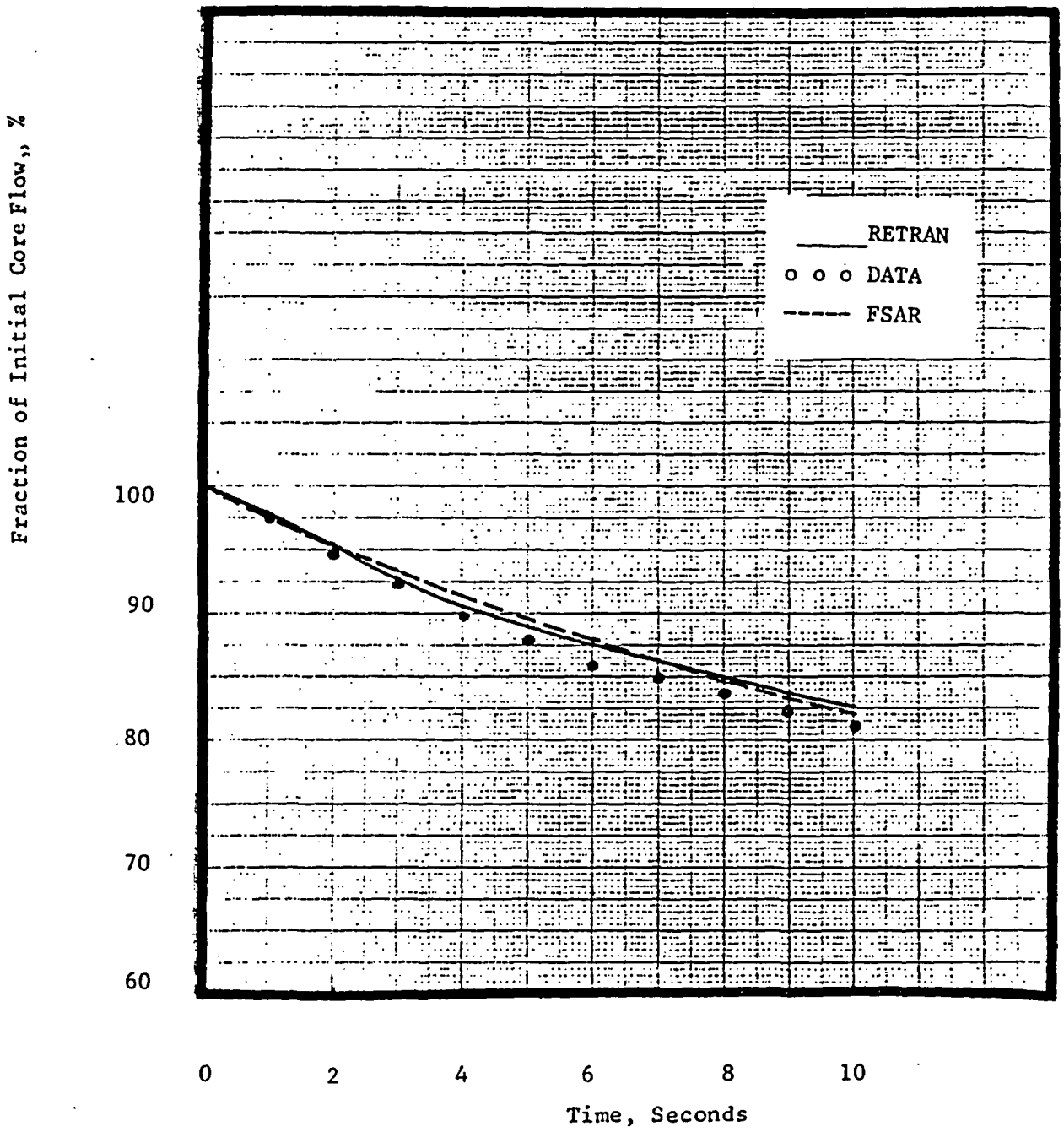


Figure 5.57

STEAM PRESSURE
NORTH ANNA COOLDOWN EVENT

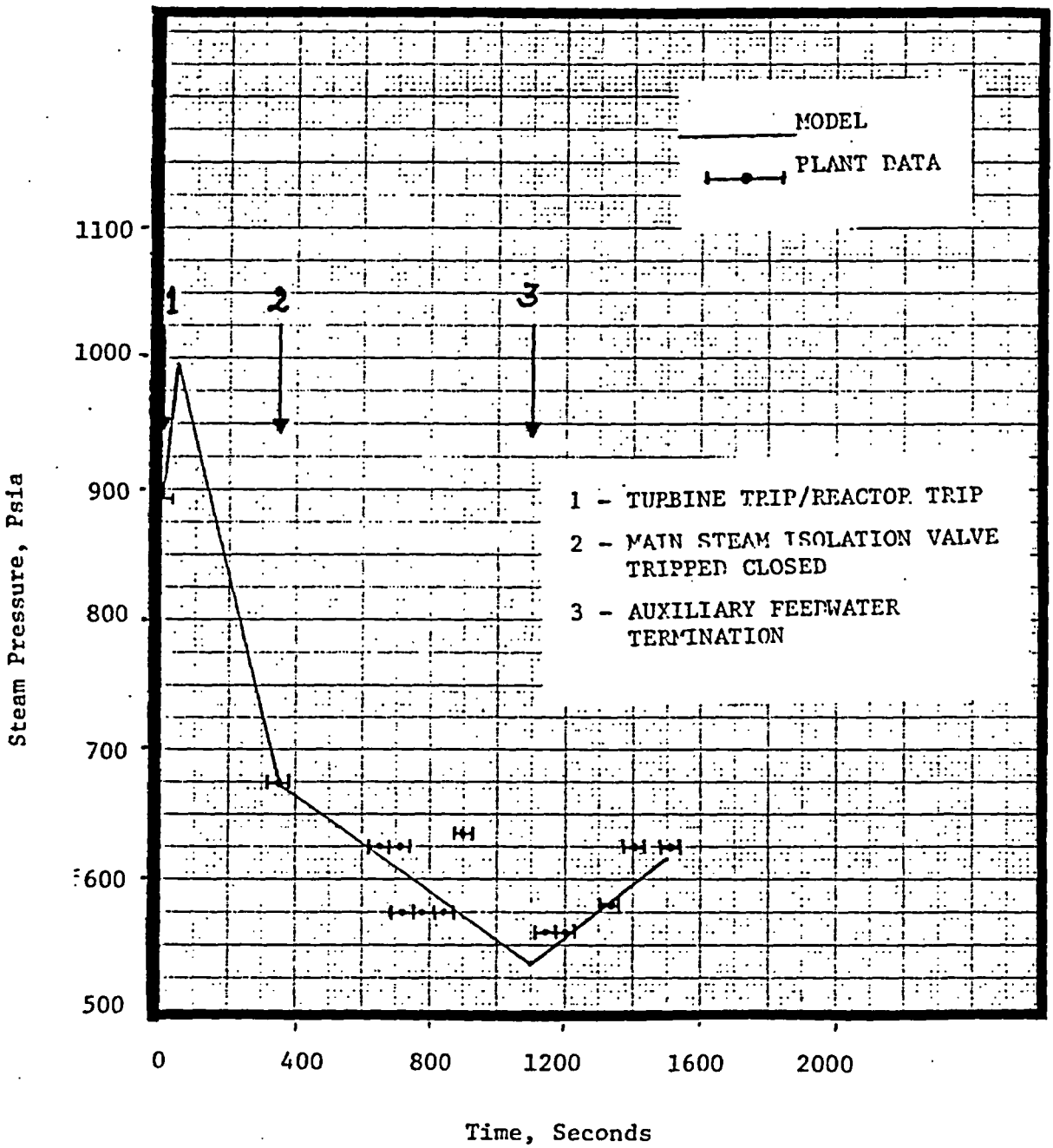


Figure 5.56

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

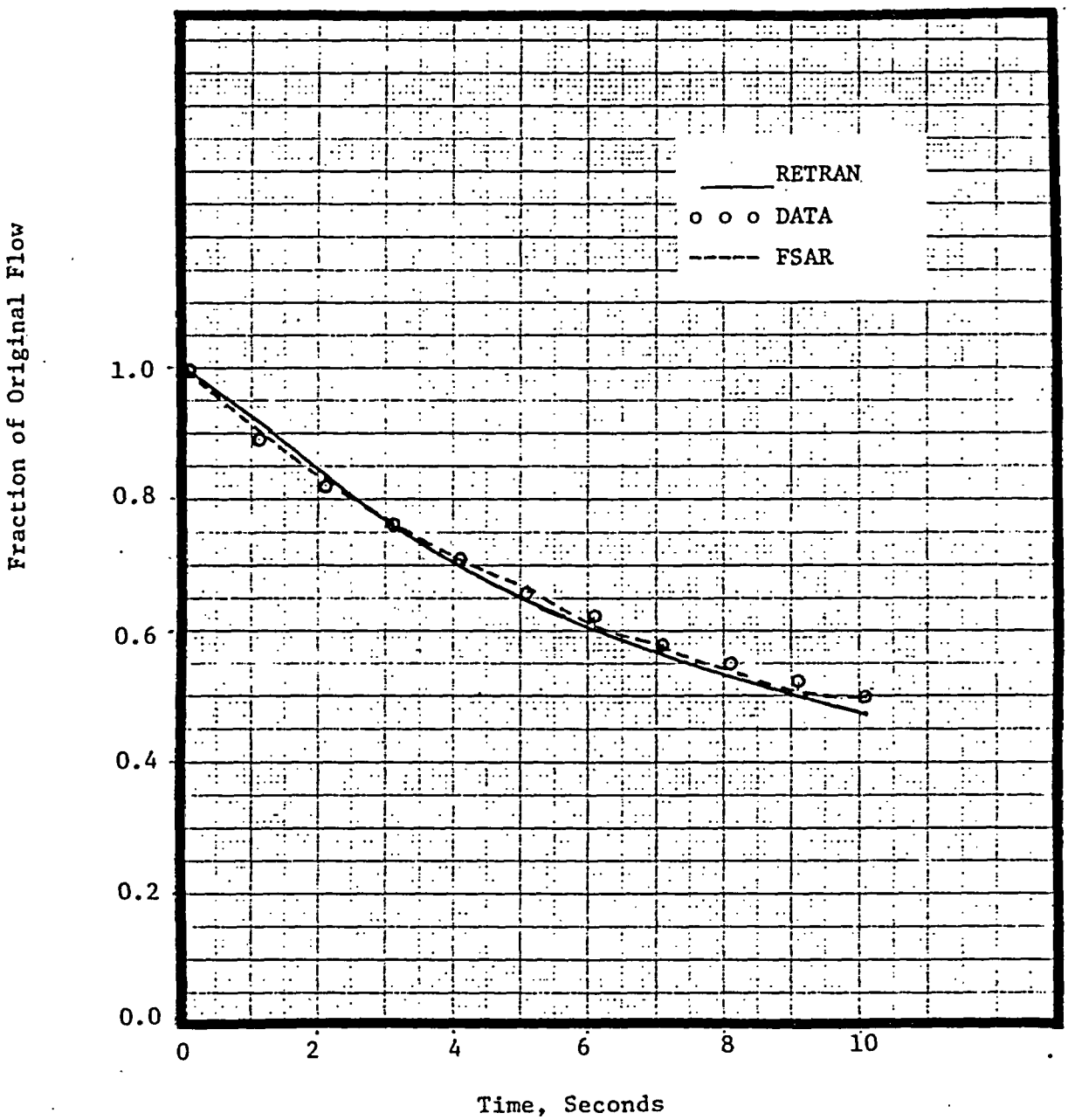


Figure 5.58

COLD LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT

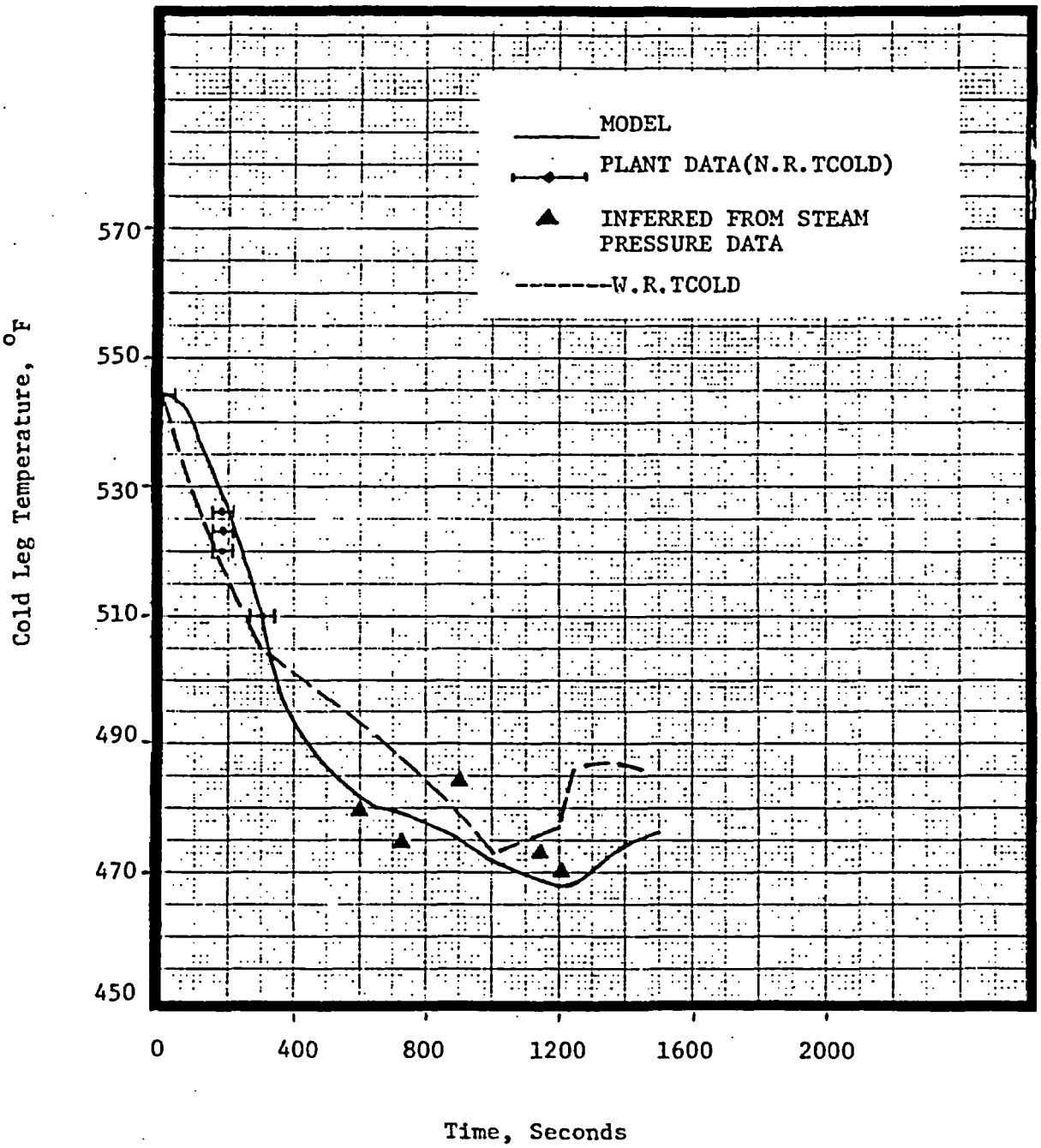


Figure 5.59

HOT LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT

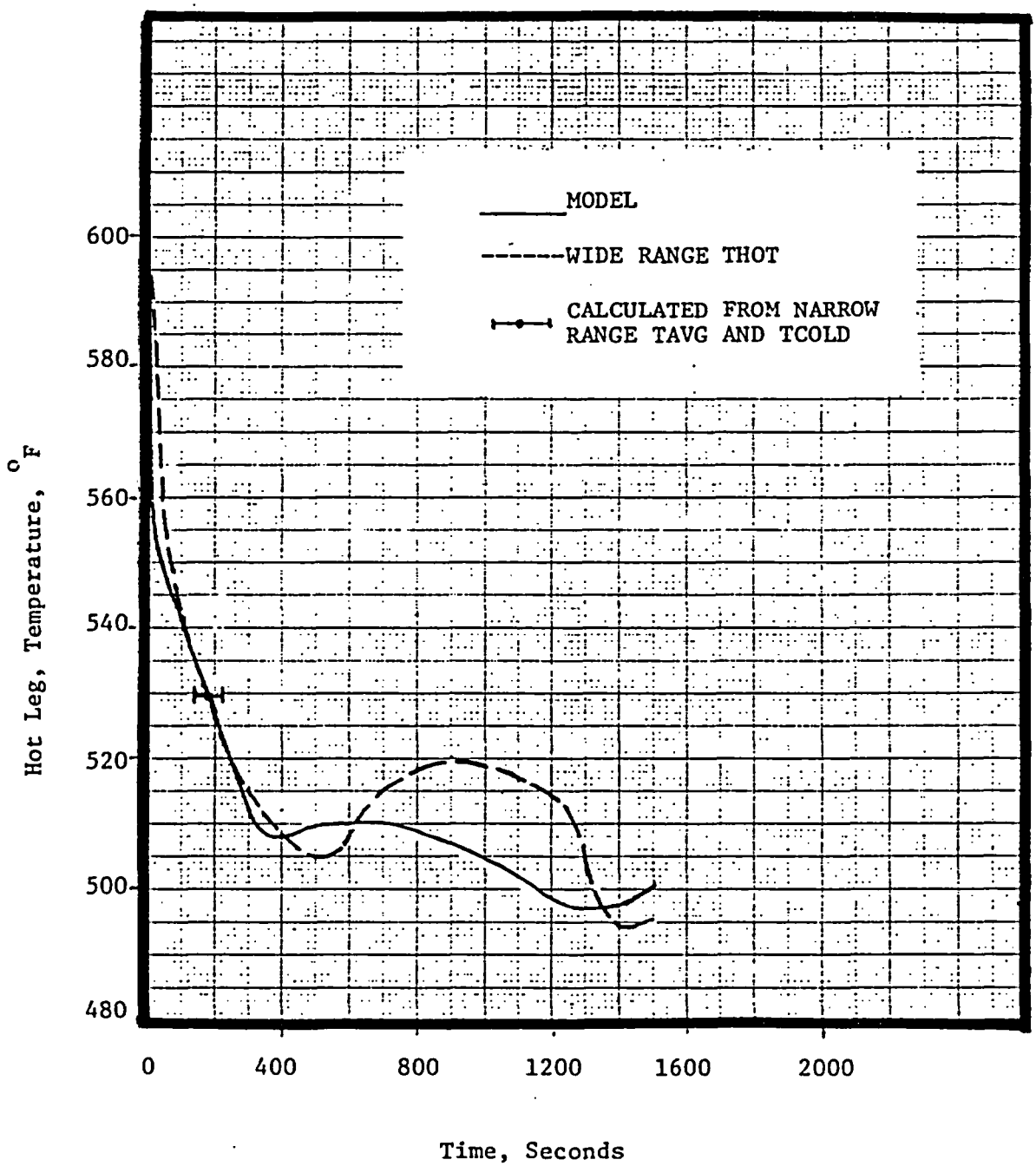


Figure 5.61

PRESSURIZER LEVEL
NORTH ANNA COOLDOWN EVENT

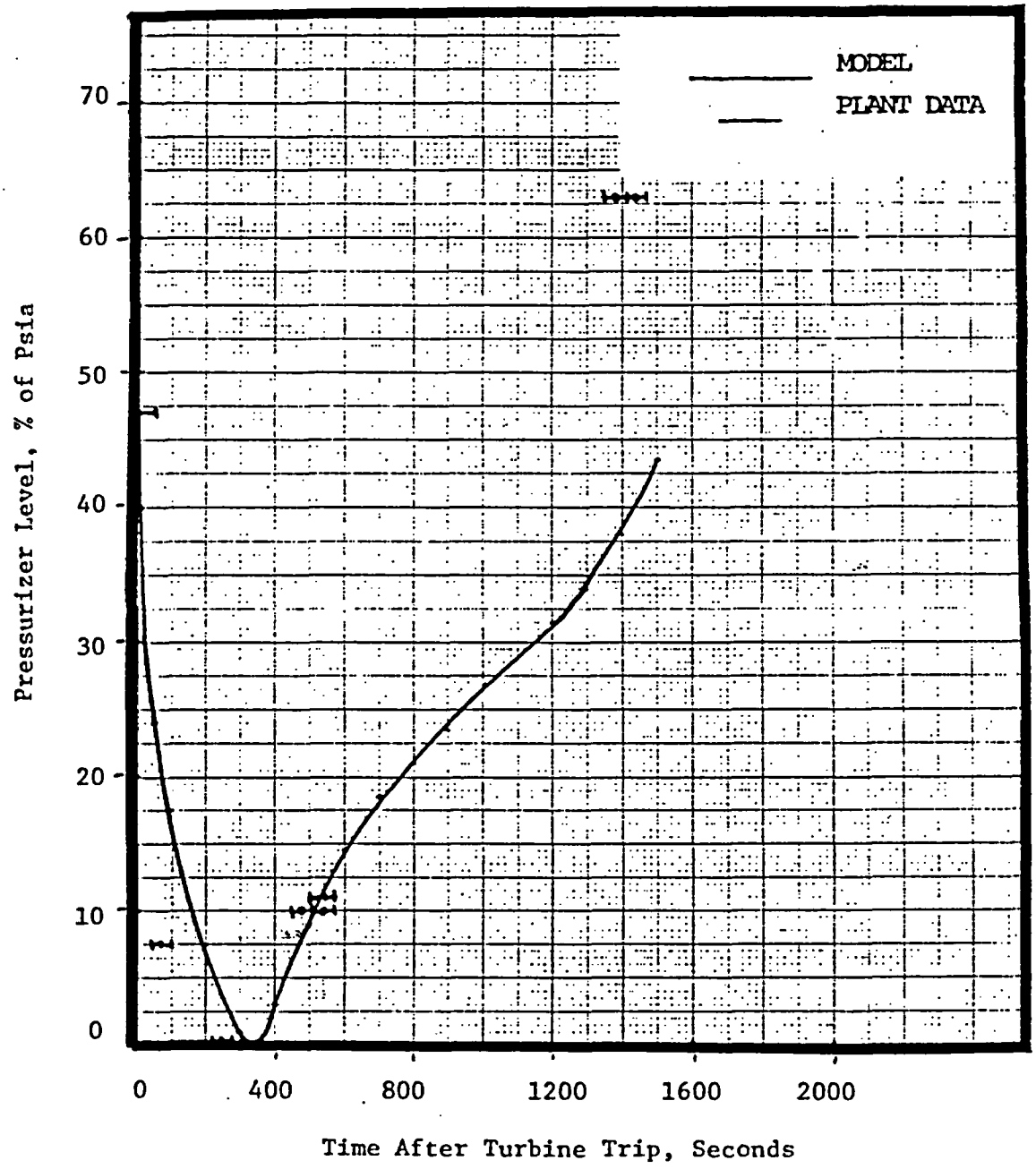
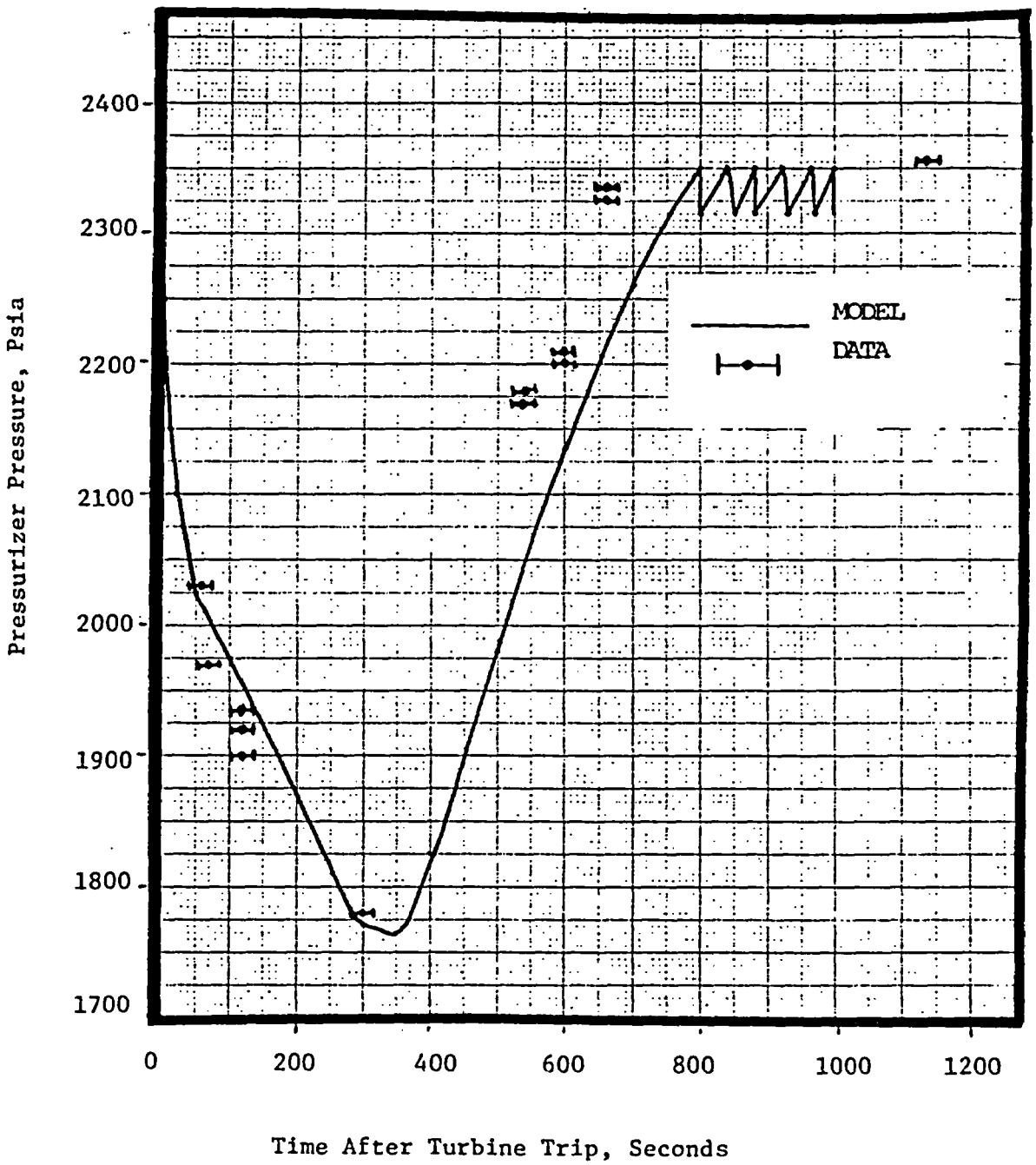


Figure 5.60

PRESSURIZER PRESSURE
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/O1T-O, 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 x 10^{-5} ΔK/K

APPENDIX (Continued)

SUMMARY OF IMPORTANT ASSUMPTIONS USED IN SAFETY 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
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

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/°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/°F αDOPPLER=-1.6 pcm/°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/°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
2. Uncontrolled Rod Withdrawal from Power				
b) Current Analysis	Surry One Loop (Modified to reflect steam generator tube plugging)	Core power = 1.02 x rated	αMOD=+1.0 pcm/°F	No credit taken for: 1) High neutron flux rod stop 2) High Overtemperature ΔT rod stop 3) High overpower ΔT rod stop or trip
1) From 102% power		Pressure = 2220 psia	Doppler power coefficient =-6.0 pcm/% at 100% power	
		T-inlet = 543.4°F		
		RCS Flow = 90% of full power thermal design	Reactivity insertion rate varied	
			Trip Reactivity: Figure A.1 curve (a), total = 2.8% ΔK/K	Source of protection: High power range high neutron flux trip or High overtemperature ΔT trip
2) From 62% power		Core power = 0.62xrated	Trip Reactivity: Fig. 4.1 curve (a), total=2.8% ΔK/K	
		Pressure = 2220 psia	αMOD=+1.0pcm/°F	Assumptions same as high power case
		T-inlet = 550.3°F	αDOP =-7.3 pcm/% at 62% power	
		RCS Flow = 90% of full power thermal design	Reactivity insertion rate varied	

ΔT trip equation used (includes errors):

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

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
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

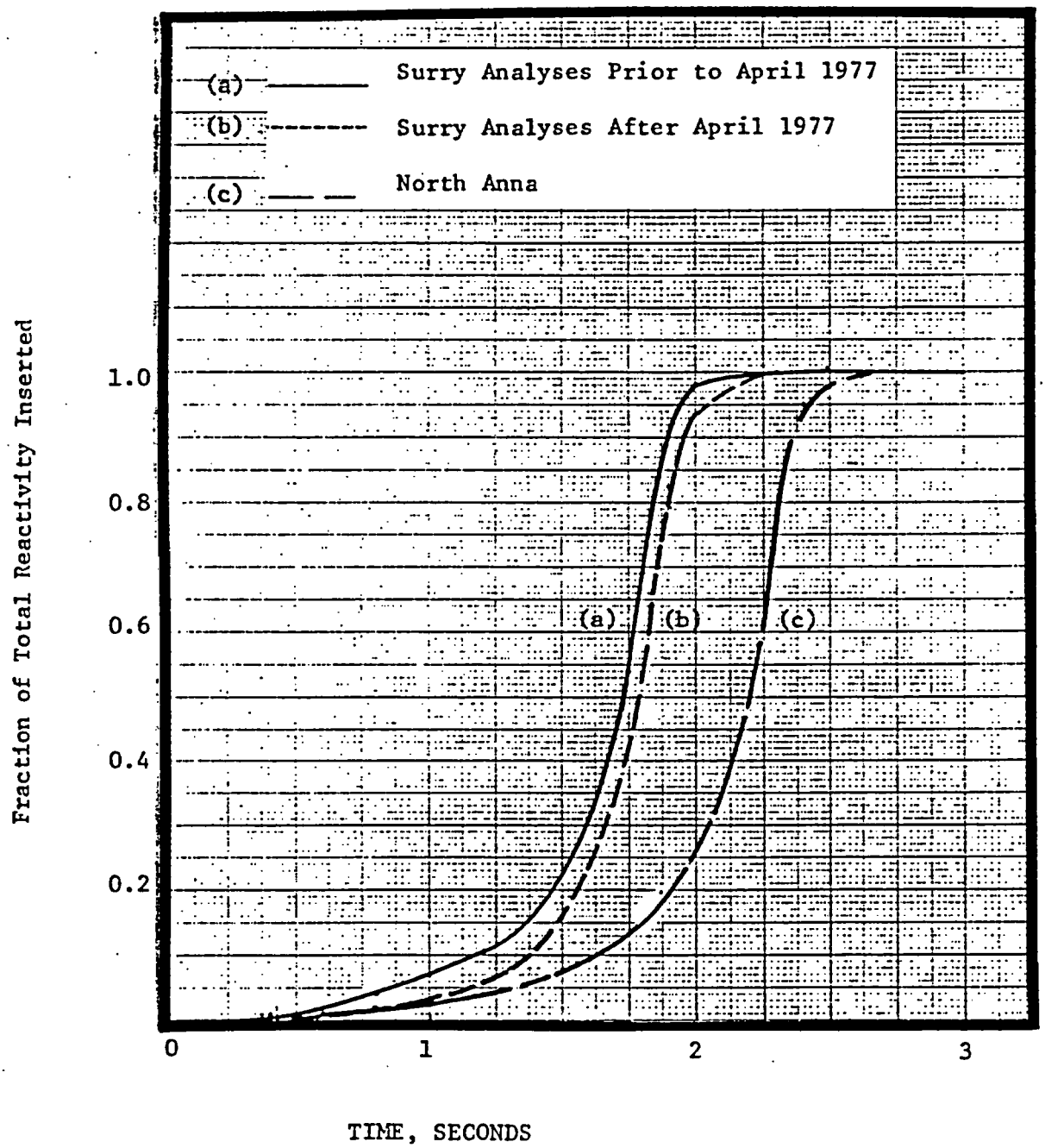
APPENDIX (Continued)

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=4×10^{-8} xrated Pressure = 2251 psia T-inlet = 549.7°F	αMOD=-25.4pcm/°F @)550°F, -13.8pcm/°F @)300°F αDoppler(Zero power) =-1.6pcm/°F 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



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 2
Response to VEP-FRD-41 Rev. 0 RAI #1

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

February 27, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. D. G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 060
PSE/NAS/cdk/0022N
Docket Nos. 50-280
50-281
50-338
50-339
License Nos. DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

VEPCO REACTOR SYSTEM TRANSIENT ANALYSES
SUPPLEMENTAL INFORMATION

In our letter to you of April 14, 1981, Serial No. 215, we transmitted our Topical Report VEP-FRD-41, "Vepco Reactor System Transient Analyses Using The RETRAN Computer Code". The report, which was provided for review by your staff, describes the system transient analysis capability developed by Vepco for analysis of certain transients which are determined to require reanalysis as a result of core reloads or other operational or design changes at our nuclear units.

In November of 1982 Mr. James L. Carter of the Division of Systems Integration informally provided us with a request for additional information which would be required to complete the review. The information requested fell into five general categories outlined in Attachment 1.

Attachment 2 provides a portion of the requested information. Specifically, the information is intended to address the request of item (1) on Attachment 1. We are currently assembling the additional information requested. Our intent is to submit this additional data by mid-1984.

If you have any questions on this material or on our topical report, please contact us.

Very truly yours,

W. L. Stewart

W. L. Stewart

8403020195 840227
PDR ADOCK 05000280
P PDR

cc: Mr. J. L. Carter
Division of Systems Integration

A001
1/1

ATTACHMENT 1

ADDITIONAL INFORMATION REQUESTED TO COMPLETE VEPCO RETRAN TOPICAL REVIEW

Plant Models

1. Volume and flow path network description, including heat slabs.
2. Component models used; description of user modifications to default models.
3. Discussion, description, and qualification of control system models.
4. Discussion of RETRAN input options selected.

Model Qualification

5. Provide additional comparison to actual plant data and/or other similar code calculations and supporting discussions.

cdk/0022N/3

ATTACHMENT 2

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2-2	Two-Loop Model Junction Description	9
2-3	Two-Loop Model Heat Conductor Description	12
2-4	Two-Loop Model Trip Description	13

TABLE 1-1

SINGLE LOOP MODEL CONTROL VOLUME DESCRIPTION

Volume ID	Description	Mixture Type	Temperature Transport Delay
1	Vessel upper plenum	H	No
2	Reactor hot leg	H	Yes
3	S/G inlet plenum	H	No
4	S/G tube volume 1	H	No
5	S/G tube volume 2	H	No
6	S/G tube volume 3	H	No
7	S/G tube volume 4	H	No
8	Pump suction piping*	H	Yes
9	Reactor coolant pump	H	No
10	Reactor cold leg	H	Yes
11	Downcomer	H	Yes
12	Vessel lower plenum	H	No
13	Core bypass	H	Yes
14	Core section 1	H	No
15	Core section 2	H	No
16	Core section 3	H	No
17	Pressurizer	N	No
18	Pressurizer surge line	H	Yes
19	S/G secondary side	T	No

Abbreviations:
S/G - steam generator
H - homogeneous equilibrium
N - two-phase non-equilibrium
T - two-phase equilibrium

*Includes S/G outlet plenum

TABLE 1-2

SINGLE LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Two-Phase Fanning Friction Multiplier	Valve Index	H/V
1	Vessel outlet nozzle	Normal	Baroczy	No	V
2	Hot leg outlet	Normal	Baroczy	Yes	H
3	S/G inlet plenum	Normal	Baroczy	No	H
4	S/G tubes	Normal	Baroczy	No	H
5	S/G tubes	Normal	Baroczy	No	V
6	S/G tubes	Normal	Baroczy	No	H
7	S/G-pump suction	Normal	Baroczy	No	H
8	Pump intake	Normal	Baroczy	No	H
9	Pump discharge	Normal	Baroczy	No	V
10	Vessel inlet nozzle	Normal	Baroczy	No	V
11	Downcomer outlet	Normal	Baroczy	No	H
12	Bypass inlet	Normal	Baroczy	No	H
13	Lower plenum - core	Normal	Baroczy	No	H
14	Core internal	Normal	Baroczy	No	H
15	Core internal	Normal	Baroczy	No	H
16	Core - upper plenum	Normal	Baroczy	No	H
17	Bypass outlet	Normal	Baroczy	No	H
18	Cold leg spray intake	Fill	Baroczy	No	V
19	Przr. spray	Spray	Baroczy	No	H
20	Przr. - surge line	Normal	Baroczy	No	H
21	Surge line - hot leg	Normal	Baroczy	No	H
22	Feedwater fill	Fill	Baroczy	No	V
23	S/G outlet	Fill	Homog.	Yes	H
24	PORV 1	Fill	Baroczy	No	H
25	PORV 2	Fill	Baroczy	No	H

TABLE 1-2 (cont.)

SINGLE LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Two-Phase Fanning Friction Multiplier	Valve Index	H/V
26	S/G atm. steam relief	Fill	Homog.	No	H
27	Pzzr. safety valve	Fill	Baroczy	No	H
28	Steamline safety valve 1	Fill	Homog.	No	H
29	Steamline safety valve 2	Fill	Homog.	No	V

Notes:
All junctions have single-stream compressible flow except junction 21 which is incompressible flow.

Abbreviations:
PORV - power operated relief valve
atm. - atmospheric
S/G - steam generator
Pzzr. - pressurizer
Homog. - homogeneous
V - vertically distributed junction area
H - horizontally distributed junction area

TABLE 1-3
SINGLE LOOP MODEL HEAT CONDUCTOR DESCRIPTION

Conductor ID	Description	Left Volume	Right Volume	Geometry	Heat Exchg. No.
1	Bottom core	0	14	Cylind.	-
2	Middle core	0	15	Cylind.	-
3	Top core	0	16	Cylind.	-
4	S/G tubes 1(inlet)	4	19	Cylind.	1
5	S/G tubes 2	5	19	Cylind.	1
6	S/G tubes 3	6	19	Cylind.	1
7	S/G tubes 4(outlet)	7	19	Cylind.	1

TABLE 1-4

SINGLE LOOP MODEL TRIP DESCRIPTION

Trip ID	Cause of Trip Activation	Trip Action
1	End of transient time	End calculation
2	High flux (normalized power)	Scram
3	Overtemperature delta-T	Scram
4	Overpower delta-T	Scram
5	High pressurizer pressure	Scram
6	Low pressurizer pressure	Scram
7	High pressurizer level	Scram
8	Low coolant flow	Scram
9	User specified time *	Close loop isolation valves
10	Low backup heater setpoint	Turn pressurizer heaters on
11	High backup heater setpoint	Turn pressurizer heaters off
12	User specified time *	Shut off reactor coolant pumps
13	Transient time = 0 sec	Trip initialization
14	User specified time *	Uncontrolled rod withdrawal
15	User specified time *	Scram
16	High pressurizer pressure	Open PORV # 1
17	Low pressurizer pressure	Close PORV # 1
18	High spray setpoint	Open PORV # 2
19	Low spray setpoint	Close PORV # 2
20	High S/G pressure	Open atm. steam relief valve
21	Low S/G pressure	Close atm. steam relief valve
22	High S/G pressure	Open S/G safety valves
23	Low S/G pressure	Close S/G safety valves
24	High pressurizer pressure	Open pressurizer safety valves
25	Low pressurizer pressure	Close pressurizer safety valves

TABLE 1-4 (cont.)

SINGLE LOOP MODEL TRIP DESCRIPTION

Trip ID	Cause of Trip Activation	Trip Action
26	User specified time *	Turbine trip
27	Low power	End calculation
28	Low-low steam generator mass	Scram
29	Low-low steam generator mass	Auxiliary feedwater on
30	Scram	Turbine trip

Notes:
* Not applicable for most transients.

Abbreviations:
PORV - power operated relief valve
atm. - atmospheric
S/G - steam generator

TABLE 2-1

TWO LOOP MODEL CONTROL VOLUME DESCRIPTION

Volume ID	Description	Mixture Type	Temperature Transport Delay	Two-phase Fanning Friction Multiplier
ONE LOOP SIDE				
101	Vessel upper plenum	H	No	Baroczy
102	Reactor hot leg	H	Yes	Baroczy
103	S/G inlet plenum	H	No	Baroczy
104	S/G tube volume 1	H	No	Baroczy
105	S/G tube volume 2	H	No	Baroczy
106	S/G tube volume 3	H	No	Baroczy
107	S/G tube volume 4	H	No	Baroczy
108	Pump suction piping*	H	Yes	Baroczy
109	Reactor coolant pump	H	No	Baroczy
110	Reactor cold leg	H	Yes	
111	Downcomer	H	Yes	Baroczy
112	Vessel lower plenum	H	No	Baroczy
113	Core section 1	H	No	Baroczy
114	Core section 2	H	No	Baroczy
115	Core section 3	H	No	Baroczy
116	Core section 4	H	No	Baroczy
701	S/G Secondary side riser	H	No	Baroczy
702	S/G Secondary side dome	H	No	Homog.

*Includes S/G outlet plenum

TABLE 2-1 (cont.)

TWO LOOP MODEL CONTROL VOLUME DESCRIPTION

Volume ID	Description	Mixture Type	Temperature Transport Delay	Two-phase Fanning Friction Multiplier
TWO LOOP SIDE				
201	Vessel upper plenum	H	No	Baroczy
202	Reactor hot leg	H	Yes	Baroczy
203	S/G inlet plenum	H	No	Baroczy
204	S/G tube volume 1	H	No	Baroczy
205	S/G tube volume 2	H	No	Baroczy
206	S/G tube volume 3	H	No	Baroczy
207	S/G tube volume 4	H	No	Baroczy
208	Pump suction piping*	H	Yes	Baroczy
209	Reactor coolant pump	H	No	Baroczy
210	Reactor cold leg	H	Yes	Baroczy
211	Downcomer	H	Yes	Baroczy
212	Vessel lower plenum	H	No	Baroczy
213	Core section 1	H	No	Baroczy
214	Core section 2	H	No	Baroczy
215	Core section 3	H	No	Baroczy
216	Core section 4	H	No	Baroczy
703	S/G Secondary side riser	N	No	Baroczy
704	S/G Secondary side dome	H	No	Homog.
300	Core bypass	H	Yes	Baroczy
400	Upper head region	H	No	Baroczy
500	Pressurizer + Surge line	N	No	Baroczy
800	Containment Sink	H	No	Baroczy

Abbreviations:
- S/G - steam generator
- H - homogeneous equilibrium
- N - two-phase non-equilibrium
- T - two-phase equilibrium
- HOMOG - homogeneous

*Includes S/G outlet plenum

TABLE 2-2

TWO LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Valve Index	H/V
101	Vessel outlet nozzle	Normal	No	V
102	Hot leg outlet	Normal	No	H
103	S/G inlet plenum	Normal	No	H
104	S/G tubes	Normal	No	H
105	S/G tubes	Normal	No	H
106	S/G tubes	Normal	No	H
107	S/G-pump suction	Normal	No	H
108	Pump intake	Normal	No	H
109	Pump discharge	Normal	No	V
110	Vessel inlet nozzle	Normal	No	V
111	Downcomer outlet	Normal	No	H
112	Bottom plenum - core	Normal	No	H
113	V113 - V114 -core internal	Normal	No	H
114	V114 - V115 -core internal	Normal	No	H
115	V115 - V116 -core internal	Normal	No	H
116	Core - upper plenum	Normal	No	H
117	Core - upper plenum	Normal	No	H
**********				
701	Riser - drum	Normal	No	H
801	Drum - containment	Normal	Yes	H
802	Drum - containment	Normal	Yes	H
901	Feedwater fill	Fill	No	H
**********				

TABLE 2-2 (cont.)

TWO LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Valve Index	H/V
201	Vessel outlet nozzle	Normal	No	V
202	Hot leg outlet	Normal	No	H
203	S/G inlet plenum	Normal	No	H
204	S/G tubes	Normal	No	H
205	S/G tubes	Normal	No	V
206	S/G tubes	Normal	No	H
207	S/G-pump suction	Normal	No	H
208	Pump intake	Normal	No	H
209	Pump discharge	Normal	No	V
210	Vessel inlet nozzle	Normal	No	V
211	Downcomer outlet	Normal	No	H
212	Bottom plenum - core	Normal	No	H
213	V213 - V214 -core internal	Normal	No	H
214	V214 - V215 -core internal	Normal	No	H
215	V215 - V216 -core internal	Normal	No	H
216	Core - upper plenum	Normal	No	H
217	Core - upper plenum	Normal	No	H
**********				
702	Riser - drum	Normal	No	H
803	Drum - containment	Normal	No	H
804	Drum - containment	Normal	No	H
902	Feedwater fill	Fill	Yes	V
903	Safety Injection fill	Fill	Yes	H
**********				

TABLE 2-2 (cont.)

TWO LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Valve Index	H/V
301	Bypass - upper plenum(11)	Normal	No	H
302	Bypass - upper plenum(21)	Normal	No	H
303	Bottom plenum - bypass(11)	Normal	No	H
304	Bottom plenum - bypass(21)	Normal	No	H
402	Upper plenum - Head	Normal	Yes	H
403	V110-V211	Normal	No	H
404	V210-V111	Normal	No	H
500	Pressurizer - Hot Leg	Normal	No	H

Abbreviations:
- PORV - power operated relief valve
- Atm. - atmospheric
- S/G - steam generator
- Pzr. - pressurizer
- Homog. - homogeneous
- V - vertically distributed junction area
- H - horizontally distributed junction area
- 11 - one loop
- 21 - two loop

TABLE 2-3
TWO LOOP MODEL HEAT CONDUCTOR DESCRIPTION

Conductor ID	Description	Left Volume	Right Volume	Geometry	Heat Exchg. No.
		Single Loop Side			
101	Bottom core	0	113	Cylind.	-
102	Middle core 1	0	114	Cylind.	-
103	Middle core 2	0	115	Cylind.	-
104	Top core	0	116	Cylind.	-
105	S/G tubes 1(inlet)	104	701	Cylind.	1
106	S/G tubes 2	105	701	Cylind.	1
107	S/G tubes 3	106	701	Cylind.	1
108	S/G tubes 4(outlet)	108	701	Cylind.	1
		Double Loop Side			
201	Bottom core	0	213	Cylind.	-
202	Middle core 1	0	214	Cylind.	-
203	Middle core 2	0	215	Cylind.	-
204	Top core	0	216	Cylind.	-
205	S/G tubes 1(inlet)	204	703	Cylind.	2
206	S/G tubes 2	205	703	Cylind.	2
207	S/G tubes 3	206	703	Cylind.	2
208	S/G tubes 4(outlet)	208	703	Cylind.	2

TABLE 2-4

TWO LOOP MODEL TRIP DESCRIPTION

Trip ID	Cause of Trip Activation	Trip Action
1	End of transient time	End calculation
2	Transient time = 0 sec	Trip Initialization
3	Low pressurizer pressure	Safety Injection actuation
4	Blank	For future use
5	Steamline high delta - P	No credit taken
6	High steam flow	------
7	Low Tavg	------
8	Low steam pressure	------
9	Coincidence trips 6 and 7	Safety Injection actuation
10	Coincidence trips 6 and 8	Safety Injection actuation
11	Coincidence trips 6 and 7	Isolate steamlines
12	Coincidence trips 6 and 8	Isolate steamlines
13	Low pressurizer pressure	Heaters on
14	High pressurizer pressure	Heaters off
15	User specified time	Pumps off
16	User specified time	Isolate feedline
17	Transient time = 0 second	Steamline break initiation

NON-PROPRIETARY

Dominion®

VEP-FRD-41, Rev. 0.1-A

Vepco Reactor System Transient Analyses Using the RETRAN Computer Code [Volume 2]

Nuclear Analysis & Fuel
Nuclear Engineering & Services

June, 2004

NON-PROPRIETARY

APPENDIX 3
Response to VEP-FRD-41 Rev. 0 RAI #2

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

July 12, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. D. G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 376
PSE/NAS:acm
Docket Nos. 50-280
50-281
50-338
50-339
License Nos. DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

VEPCO REACTOR SYSTEM TRANSIENT ANALYSES

In our letter to you of April 14, 1981, Serial No. 215, we transmitted our Topical Report VEP-FRD-41, "Vepco Reactor System Transient Analyses using the RETRAN Computer Code". The Report, which was provided for review by your staff, describes the system transient analysis capability which Vepco is using in support of core reloads or other operational or design changes at our nuclear units.

In November of 1982, Mr. James L. Carter of the Division of Systems Integration informally provided us with a request for additional information which would be required to complete the review. The information requested fell into five general categories, as outlined in Attachment 1. The information which addressed item (1) of Attachment 1 was transmitted to you by our letter of February 27, 1984, Serial No. 060. For your convenience, this information is reproduced as Section 1 of Attachment 2.

Sections 2 and 4 of Attachment 2 provides the requested information for two of the four remaining categories outlined in Attachment 1 (i.e. a description of Vepco's system component models and a discussion of RETRAN input options selected).

Section 3, which addresses the description and qualification of our RETRAN control system models, is given in outline form only, as this material has not been completed at this time. As discussed in our May 22, 1984 meeting with Mr. Carter and Mr. David Moran of the Standardization and Special Projects Branch of the Division of Licensing, we intend to provide this material in an additional submittal on or about August 15, 1984.

Section 5 will address the remaining "model qualification" information requested in Attachment 1, by providing results of comparisons of RETRAN calculations to calculations performed with LOFTRAN, a code developed by Westinghouse Electric Corporation. This section is undergoing final review and we will transmit this additional information shortly.

Section 6 of Attachment 2 provides the results of certain sensitivity studies performed with our RETRAN models which may help your staff in completing their review.

8407250336

Vepco is currently engaged in analytical work with RETRAN which will form the basis of a submittal justifying an amendment request to the North Anna Technical Specifications. This amendment, which would allow operation with a slightly positive moderator temperature coefficient, is required to support operation with our North Anna Unit 1 Cycle 6 Reload Core. The amendment submittal, which is scheduled for September 1984, will contain reanalyses of approximately 6 FSAR transients. These reanalyses will be based on the models and methods described in Topical Report VEP-FRD-41 and in VEP-NFE-2, "Vepco Evaluation of the Control Rod Ejection Transient", submitted by our letter of November 23, 1983, Serial No. 657. In order to incorporate a positive moderator temperature coefficient into the design of North Anna Unit 1, Cycle 6 an initial review and NRC comments on the acceptability of the amendment submittal would be required by November 1, 1984, and approval of the amendment request and the supporting Topical Reports (VEP-FRD-41 and VEP-NFE-2) would be required by January 15, 1985.

As we discussed with Mr. Moran, we will be meeting with the appropriate NRC staff on July 19, 1984, to discuss this material and to provide any amplification or clarification which may be required for completion of a review consistent with the schedule outlined above.

Very truly yours,

W. L. Stewart

Attachments

cc: Mr. D. H. Moran
Standardization and Special Projects Branch

Mr. J. L. Carter
Reactors Systems Branch

Mr. James P. O'Reilly
Regional Administrator
Region II

Mr. James R. Miller, Chief
Operating Reactors Branch No. 3
Division of Licensing

Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Mr. D. J. Burke
NRC Resident Inspector
Surry Power Station

Mr. M. W. Branch
NRC Resident Inspector
North Anna Power Station

ATTACHMENT 1

ADDITIONAL INFORMATION REQUESTED TO COMPLETE VEPCO RETRAN TOPICAL REVIEW

Plant Models

1. Volume and flow path network description, including heat slabs.
2. Component models used; description of user modifications to default models.
3. Discussion, description, and qualification of control system models.
4. Discussion of RETRAN input options selected.

Model Qualification

5. Provide additional comparison to actual plant data and/or other similar code calculations and supporting discussions.

Attachment 2

SUPPLEMENTAL INFORMATION

RETRAN TOPICAL VEP-FRD-41

VIRGINIA ELECTRIC AND POWER COMPANY

CONTENTS

SECTION	TITLE
I	RETRAN VOLUME AND FLOW PATH INFORMATION
II	DESCRIPTION OF VEPCO COMPONENT MODELS
III	VEPCO RETRAN CONTROL SYSTEM MODELS DESCRIPTION/QUALIFICATION
IV	VEPCO RETRAN MODELS-INPUT OPTIONS
V	COMPARISON TO ALTERNATE CODE CALCULATIONS
VI	RETRAN SENSITIVITY STUDIES

I. RETRAN VOLUME AND FLOW PATH NETWORK DESCRIPTION

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TABLE 1-1

SINGLE LOOP MODEL CONTROL VOLUME DESCRIPTION

Volume ID	Description	Mixture Type	Temperature Transport Delay
1	Vessel upper plenum	H	No
2	Reactor hot leg	H	Yes
3	S/G inlet plenum	H	No
4	S/G tube volume 1	H	No
5	S/G tube volume 2	H	No
6	S/G tube volume 3	H	No
7	S/G tube volume 4	H	No
8	Pump suction piping*	H	Yes
9	Reactor coolant pump	H	No
10	Reactor cold leg	H	Yes
11	Downcomer	H	Yes
12	Vessel lower plenum	H	No
13	Core bypass	H	Yes
14	Core section 1	H	No
15	Core section 2	H	No
16	Core section 3	H	No
17	Pressurizer	N	No
18	Pressurizer surge line	H	Yes
19	S/G secondary side	T	No

Abbreviations:
S/G - steam generator
H - homogeneous equilibrium
N - two-phase non-equilibrium
T - two-phase equilibrium

*Includes S/G outlet plenum

TABLE 1-2

SINGLE LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Two-Phase Fanning Friction Multiplier	Valve Index	H/V
1	Vessel outlet nozzle	Normal	Baroczy	No	V
2	Hot leg outlet	Normal	Baroczy	Yes	H
3	S/G inlet plenum	Normal	Baroczy	No	H
4	S/G tubes	Normal	Baroczy	No	H
5	S/G tubes	Normal	Baroczy	No	V
6	S/G tubes	Normal	Baroczy	No	H
7	S/G-pump suction	Normal	Baroczy	No	H
8	Pump intake	Normal	Baroczy	No	H
9	Pump discharge	Normal	Baroczy	No	V
10	Vessel inlet nozzle	Normal	Baroczy	No	V
11	Downcomer outlet	Normal	Baroczy	No	H
12	Bypass inlet	Normal	Baroczy	No	H
13	Lower plenum - core	Normal	Baroczy	No	H
14	Core internal	Normal	Baroczy	No	H
15	Core internal	Normal	Baroczy	No	H
16	Core - upper plenum	Normal	Baroczy	No	H
17	Bypass outlet	Normal	Baroczy	No	H
18	Cold leg spray intake	Fill	Baroczy	No	V
19	Przr. spray	Spray	Baroczy	No	H
20	Przr. - surge line	Normal	Baroczy	No	H
21	Surge line - hot leg	Normal	Baroczy	No	H
22	Feedwater fill	Fill	Baroczy	No	V
23	S/G outlet	Fill	Homog.	Yes	H
24	PORV 1	Fill	Baroczy	No	H
25	PORV 2	Fill	Baroczy	No	H

TABLE 1-2 (cont.)

SINGLE LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Two-Phase Fanning Friction Multiplier	Valve Index	H/V
26	S/G atm. steam relief	Fill	Homog.	No	H
27	Pzzr. safety valve	Fill	Baroczy	No	H
28	Steamline safety valve 1	Fill	Homog.	No	H
29	Steamline safety valve 2	Fill	Homog.	No	V

Notes:
All junctions have single-stream compressible flow except junction 21 which is incompressible flow.

Abbreviations:
PORV - power operated relief valve
atm. - atmospheric
S/G - steam generator
Pzzr. - pressurizer
Homog. - homogeneous
V - vertically distributed junction area
H - horizontally distributed junction area

TABLE 1-3
SINGLE LOOP MODEL HEAT CONDUCTOR DESCRIPTION

Conductor ID	Description	Left Volume	Right Volume	Geometry	Heat Exchg. No.
1	Bottom core	0	14	Cylind.	-
2	Middle core	0	15	Cylind.	-
3	Top core	0	16	Cylind.	-
4	S/G tubes 1(inlet)	4	19	Cylind.	1
5	S/G tubes 2	5	19	Cylind.	1
6	S/G tubes 3	6	19	Cylind.	1
7	S/G tubes 4(outlet)	7	19	Cylind.	1

TABLE 1-4

SINGLE LOOP MODEL TRIP DESCRIPTION

Trip ID	Cause of Trip Activation	Trip Action
1	End of transient time.	End calculation
2	High flux (normalized power)	Scram
3	Overtemperature delta-T	Scram
4	Overpower delta-T	Scram
5	High pressurizer pressure	Scram
6	Low pressurizer pressure	Scram
7	High pressurizer level	Scram
8	Low coolant flow	Scram
9	User specified time *	Close loop isolation valves
10	Low backup heater setpoint	Turn pressurizer heaters on
11	High backup heater setpoint	Turn pressurizer heaters off
12	User specified time *	Shut off reactor coolant pumps
13	Transient time = 0 sec	Trip initialization
14	User specified time *	Uncontrolled rod withdrawal
15	User specified time *	Scram
16	High pressurizer pressure	Open PORV # 1
17	Low pressurizer pressure	Close PORV # 1
18	High spray setpoint	Open PORV # 2
19	Low spray setpoint	Close PORV # 2
20	High S/G pressure	Open atm. steam relief valve
21	Low S/G pressure	Close atm. steam relief valve
22	High S/G pressure	Open S/G safety valves
23	Low S/G pressure	Close S/G safety valves
24	High pressurizer pressure	Open pressurizer safety valves
25	Low pressurizer pressure	Close pressurizer safety valves

TABLE 1-4 (cont.)

SINGLE LOOP MODEL TRIP DESCRIPTION

Trip ID	Cause of Trip Activation	Trip Action
26	User specified time *	Turbine trip
27	Low power	End calculation
28	Low-low steam generator mass	Scram
29	Low-low steam generator mass	Auxiliary feedwater on
30	Scram	Turbine trip

Notes:
* Not applicable for most transients.

Abbreviations:
PORV - power operated relief valve
atm. - atmospheric
S/G - steam generator

TABLE 2-1

TWO LOOP MODEL CONTROL VOLUME DESCRIPTION

Volume ID	Description	Mixture Type	Temperature Transport Delay	Two-phase Fanning Friction Multiplier
ONE LOOP SIDE				
101	Vessel upper plenum	H	No	Baroczy
102	Reactor hot leg	H	Yes	Baroczy
103	S/G inlet plenum	H	No	Baroczy
104	S/G tube volume 1	H	No	Baroczy
105	S/G tube volume 2	H	No	Baroczy
106	S/G tube volume 3	H	No	Baroczy
107	S/G tube volume 4	H	No	Baroczy
108	Pump suction piping*	H	Yes	Baroczy
109	Reactor coolant pump	H	No	Baroczy
110	Reactor cold leg	H	Yes	
111	Downcomer	H	Yes	Baroczy
112	Vessel lower plenum	H	No	Baroczy
113	Core section 1	H	No	Baroczy
114	Core section 2	H	No	Baroczy
115	Core section 3	H	No	Baroczy
116	Core section 4	H	No	Baroczy
701	S/G Secondary side riser	H	No	Baroczy
702	S/G Secondary side dome	H	No	Homog.

*Includes S/G outlet plenum

TABLE 2-1 (cont.)

TWO LOOP MODEL CONTROL VOLUME DESCRIPTION

Volume ID	Description	Mixture Type	Temperature Transport Delay	Two-phase Fanning Friction Multiplier
TWO LOOP SIDE				
201	Vessel upper plenum	H	No	Baroczy
202	Reactor hot leg	H	Yes	Baroczy
203	S/G inlet plenum	H	No	Baroczy
204	S/G tube volume 1	H	No	Baroczy
205	S/G tube volume 2	H	No	Baroczy
206	S/G tube volume 3	H	No	Baroczy
207	S/G tube volume 4	H	No	Baroczy
208	Pump suction piping*	H	Yes	Baroczy
209	Reactor coolant pump	H	No	Baroczy
210	Reactor cold leg	H	Yes	Baroczy
211	Downcomer	H	Yes	Baroczy
212	Vessel lower plenum	H	No	Baroczy
213	Core section 1	H	No	Baroczy
214	Core section 2	H	No	Baroczy
215	Core section 3	H	No	Baroczy
216	Core section 4	H	No	Baroczy
703	S/G Secondary side riser	N	No	Baroczy
704	S/G Secondary side dome	H	No	Homog.
300	Core bypass	H	Yes	Baroczy
400	Upper head region	H	No	Baroczy
500	Pressurizer + Surge line	N	No	Baroczy
800	Containment Sink	H	No	Baroczy

Abbreviations:
- S/G - steam generator
- H - homogeneous equilibrium
- N - two-phase non-equilibrium
- T - two-phase equilibrium
- HOMOG - homogeneous

*Includes S/G outlet plenum

TABLE 2-2

TWO LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Valve Index	H/V
101	Vessel outlet nozzle	Normal	No	V
102	Hot leg outlet	Normal	No	H
103	S/G inlet plenum	Normal	No	H
104	S/G tubes	Normal	No	H
105	S/G tubes	Normal	No	H
106	S/G tubes	Normal	No	H
107	S/G-pump suction	Normal	No	H
108	Pump intake	Normal	No	H
109	Pump discharge	Normal	No	V
110	Vessel inlet nozzle	Normal	No	V
111	Downcomer outlet	Normal	No	H
112	Bottom plenum - core	Normal	No	H
113	V113 - V114 -core internal	Normal	No	H
114	V114 - V115 -core internal	Normal	No	H
115	V115 - V116 -core internal	Normal	No	H
116	Core - upper plenum	Normal	No	H
117	Core - upper plenum	Normal	No	H
**********				
701	Riser - drum	Normal	No	H
801	Drum - containment	Normal	Yes	H
802	Drum - containment	Normal	Yes	H
901	Feedwater fill	Fill	~~No~~ Yes	H
**********				

TABLE 2-2 (cont.)

TWO LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Valve Index	H/V
201	Vessel outlet nozzle	Normal	No	V
202	Hot leg outlet	Normal	No	H
203	S/G inlet plenum	Normal	No	H
204	S/G tubes	Normal	No	H
205	S/G tubes	Normal	No	V
206	S/G tubes	Normal	No	H
207	S/G-pump suction	Normal	No	H
208	Pump intake	Normal	No	H
209	Pump discharge	Normal	No	V
210	Vessel inlet nozzle	Normal	No	V
211	Downcomer outlet	Normal	No	H
212	Bottom plenum - core	Normal	No	H
213	V213 - V214 -core internal	Normal	No	H
214	V214 - V215 -core internal	Normal	No	H
215	V215 - V216 -core internal	Normal	No	H
216	Core - upper plenum	Normal	No	H
217	Core - upper plenum	Normal	No	H
**********				
702	Riser - drum	Normal	No	H
803	Drum - containment	Normal	No	H
804	Drum - containment	Normal	No	H
902	Feedwater fill	Fill	Yes	V
903	Safety Injection fill	Fill	Yes	H
**********				

TABLE 2-2 (cont.)

TWO LOOP MODEL JUNCTION DESCRIPTION

Junction ID	Description	Type	Valve Index	H/V
301	Bypass - upper plenum(11)	Normal	No	H
302	Bypass - upper plenum(21)	Normal	No	H
303	Bottom plenum - bypass(11)	Normal	No	H
304	Bottom plenum - bypass(21)	Normal	No	H
402	Upper plenum - Head	Normal	Yes	H
403	V110-V211	Normal	No	H
404	V210-V111	Normal	No	H
500	Pressurizer - Hot Leg	Normal	No	H

Abbreviations:
- PORV - power operated relief valve
- Atm. - atmospheric
- S/G - steam generator
- Przr. - pressurizer
- Homog. - homogeneous
- V - vertically distributed junction area
- H - horizontally distributed junction area
- 11 - one loop
- 21 - two loop

TABLE 2-3
TWO LOOP MODEL HEAT CONDUCTOR DESCRIPTION

Conductor ID	Description	Left Volume	Right Volume	Geometry	Heat Exchg. No.
	Single Loop Side				
101	Bottom core	0	113	Cylind.	-
102	Middle core 1	0	114	Cylind.	-
103	Middle core 2	0	115	Cylind.	-
104	Top core	0	116	Cylind.	-
105	S/G tubes 1(inlet)	104	701	Cylind.	1
106	S/G tubes 2	105	701	Cylind.	1
107	S/G tubes 3	106	701	Cylind.	1
108	S/G tubes 4(outlet)	108	701	Cylind.	1
	Double Loop Side				
201	Bottom core	0	213	Cylind.	-
202	Middle core 1	0	214	Cylind.	-
203	Middle core 2	0	215	Cylind.	-
204	Top core	0	216	Cylind.	-
205	S/G tubes 1(inlet)	204	703	Cylind.	2
206	S/G tubes 2	205	703	Cylind.	2
207	S/G tubes 3	206	703	Cylind.	2
208	S/G tubes 4(outlet)	208	703	Cylind.	2

TABLE 2-4

TWO LOOP MODEL TRIP DESCRIPTION

Trip ID	Cause of Trip Activation	Trip Action
1	End of transient time	End calculation
2	Transient time = 0 sec	Trip Initialization
3	Low pressurizer pressure	Safety Injection actuation
4	Blank	For future use
5	Steamline high delta - P	No credit taken
6	High steam flow	------
7	Low Tavg	------
8	Low steam pressure	------
9	Coincidence trips 6 and 7	Safety Injection actuation
10	Coincidence trips 6 and 8	Safety Injection actuation
11	Coincidence trips 6 and 7	Isolate steamlines
12	Coincidence trips 6 and 8	Isolate steamlines
13	Low pressurizer pressure	Heaters on
14	High pressurizer pressure	Heaters off
15	User specified time	Pumps off
16	User specified time	Isolate feedline
17	Transient time = 0 second	Steamline break initiation

II. DESCRIPTION OF VEPCO COMPONENT MODELS

VEPCO RETRAN MODELS
RCS PUMP MODELS
(SINGLE LOOP AND TWO LOOP MODELS)

PARAMETER	OPTION/VALUE
PUMP CURVE SET	WESTINGHOUSE NS=5200 (BUILT-IN)
USER-MODIFIED CURVES	- FIRST QUADRANT HEAD VS FLOW (FSAR VALUES)
REVERSAL OPTION	REVERSAL NOT ALLOWED
TWO-PHASE MULTIPLIERS	NOT USED
MOTOR TORQUE OPTION	NOT USED
RATED HEAD/FLOW*	FROM FSAR/VENDOR PUMP TECH MANUAL
RATED PUMP TORQUE*	CALCULATED FROM RATED HEAD, FLOW AND HYDRAULIC EFFICIENCY
RATED MOTOR TORQUE*	CALCULATED FROM RATED HORSEPOWER AND SPEED
FULL SPEED FRICTION TORQUE*	- ESTIMATED FROM DIFFERENCE OF MOTOR TORQUE AND PUMP TORQUE
FRICTION TORQUE VARIATION WITH SHAFT SPEED	- PROPORTIONAL TO SQUARE OF SPEED

: IN SINGLE LOOP MODEL :
RATED HEAD = 1 X SINGLE PUMP VALUE
RATED FLOW = 3 X SINGLE PUMP VALUE
RATED TORQUE = 3 X SINGLE PUMP VALUE
INERTIA= 3 X SINGLE PUMP VALUE

* IN TWO LOOP MODEL :
SINGLE LOOP SIDE
RATED HEAD = 1 X SINGLE PUMP VALUE
RATED FLOW = 1 X SINGLE PUMP VALUE
RATED TORQUE = 1 X SINGLE PUMP VALUE
INERTIA= 1 X SINGLE PUMP VALUE
DOUBLE LOOP SIDE
RATED HEAD = 1 X SINGLE PUMP VALUE
RATED FLOW = 2 X SINGLE PUMP VALUE
RATED TORQUE = 2 X SINGLE PUMP VALUE
INERTIA= 2 X SINGLE PUMP VALUE

-PUMP MODEL QUALIFICATION: COMPARISON TO 1-PUMP AND 3-PUMP COASTDOWN DATA FROM SURRY AND NORTH ANNA STARTUP TESTING (REFERENCE VEP-FRD-41 SECTION 5.3)

VEPCO RETRAN MODELS
VALVES

VALVE DESCRIPTION	USED IN:
MAIN STEAM ISOLATION VALVES	TWO-LOOP MODEL

PARAMETER	OPTION/VALUE
LOCATION	JUNCTION BETWEEN STEAM DRUM AND CONTAINMENT-(SEE SECT. I.) IN DOUBLE LOOP
VALVE TYPE	TIME DEPENDENT AREA DEFINED BY CONTROL SYSTEM
OTHER DESCRIPTIVE INFORMATION	CONTROL SYSTEM MODEL OPENS THE VALVE FROM CLOSED TO FULL OPEN IN 0.01 SEC TO SIMULATE A STEAM LINE BREAK. FOLLOWING RECEIPT OF A MAIN STEAM LINE ISOLATION SIGNAL (SEE TWO LOOP MODEL TRIP DESCRIPTION), THE VALVE IS RAMPED CLOSED OVER THE MAXIMUM ALLOWED MSIV CLOSURE TIME (SEE CONTROL SYSTEM MODEL DESCRIPTIONS)

VEPCO RETRAN MODELS
VALVES

VALVE DESCRIPTION	USED IN:
STEAM LINE BREAK	TWO-LOOP MODEL

PARAMETER	OPTION/VALUE
LOCATION	JUNCTION 801 (STEAM DRUM-VOL 702 TO CONTAINMENT-VOL 800) IN SINGLE LOOP
VALVE TYPE	TIME DEPENDENT AREA DEFINED BY GENERAL DATA TABLE
OTHER DESCRIPTIVE INFORMATION	TABLE OPENS THE VALVE FROM CLOSED TO FULL OPEN IN 0.01 SEC TO SIMULATE A STEAM LINE BREAK IN THE "FAULTED" LOOP. THIS BREAK IS MODELED AS NON-ISOLABLE.

VEPCO RETRAN MODELS
VALVES

VALVE DESCRIPTION	USED IN:
ISOLATE NORMAL STEAM FLOW	TWO-LOOP MODEL

PARAMETER	OPTION/VALUE
LOCATION	JUNCTION 802 (STEAM DRUM-VOL 702 TO CONTAINMENT-VOL 800) IN SINGLE LOOP JUNCTION 804 (STEAM DRUM-VOL 702 TO CONTAINMENT-VOL 800) IN SINGLE LOOP
VALVE TYPE	TIME DEPENDENT AREA DEFINED BY GENERAL DATA TABLE
OTHER DESCRIPTIVE INFORMATION	DURING STEADY STATE HOT ZERO POWER OPERATION, A SMALL AMOUNT OF STEAM FLOW TO THE ATMOSPHERIC DUMPS IS SIMULATED TO REMOVE RCS PUMP HEAT. THIS VALVE CLOSES THIS STEAM FLOW PATH UPON INITIATION OF A STEAM LINE BREAK.

VEPCO RETRAN MODELS
VALVES

VALVE DESCRIPTION	USED IN:
FEEDLINE ISOLATION	TWO-LOOP MODEL

PARAMETER	OPTION/VALUE
LOCATION	JUNCTION 901 (FILL JUNCTION TO RISER SECTION OF STEAM GENERATOR IN SINGLE LOOP) JUNCTION 902 (FILL JUNCTION TO RISER SECTION OF STEAM GENERATOR IN DOUBLE LOOP)
VALVE TYPE	TIME DEPENDENT AREA DEFINED BY GENERAL DATA TABLE
OTHER DESCRIPTIVE INFORMATION	IN TWO-LOOP SIDE, TABLE CLOSES THE VALVE IN 0.1 SEC FOLLOWING RECEIPT OF A FEEDLINE ISOLATION SIGNAL (VALVE CLOSURE TIME IS ACCOUNTED FOR IN TRIP DELAY) - SEE TWO LOOP MODEL TRIP DESCRIPTION. IN ONE-LOOP SIDE, FOLLOWING RECEIPT OF A FEEDLINE ISOLATION SIGNAL, VALVE RAMPS TO A FRACTIONAL AREA VALUE CALCULATED TO DELIVER FULL AUXILIARY FEED FLOW RATE TO FAULTED GENERATOR (STEAM LINE BREAK ONLY)

VEPCO RETRAN MODELS
VALVES

VALVE DESCRIPTION	USED IN:
HIGH HEAD SAFETY INJECTION PUMP DISCHARGE VALVES	TWO-LOOP MODEL

PARAMETER	OPTION/VALUE
LOCATION	JUNCTION 903 (FILL JUNCTION TO SINGLE LOOP COLD LEG) JUNCTION 904 (FILL JUNCTION TO DOUBLE LOOP COLD LEG)
VALVE TYPE	TIME DEPENDENT AREA DEFINED BY GENERAL DATA TABLE
OTHER DESCRIPTIVE INFORMATION	VALVE RAMPS OPEN FOLLOWING RECEIPT OF A SAFETY INJECTION SIGNAL (SEE TWO LOOP MODEL TRIP DESCRIPTIONS). THE RAMP-OPEN TIME SIMULATES THE COMBINED EFFECTS OF VALVE OPENING AND ACCELERATION OF THE HIGH HEAD SAFETY INJECTION PUMPS

VEPCO RETRAN MODELS
VALVES

VALVE DESCRIPTION	USED IN:
TURBINE STOP VALVES	TWO-LOOP MODEL ONE-LOOP MODEL

PARAMETER	OPTION/VALUE
LOCATION (TWO LOOP MODEL)	JUNCTION 702 (FILL JUNCTION TO SINGLE LOOP STEAM DRUM) JUNCTION 704 (FILL JUNCTION TO DOUBLE LOOP STEAM DRUM)
(ONE LOOP MODEL)	JUNCTION 23 (FILL JUNCTION TO STEAM GENERATOR SECONDARY)
VALVE TYPE	TIME DEPENDENT AREA DEFINED BY GENERAL DATA TABLE
OTHER DESCRIPTIVE INFORMATION	STEAM FLOW TO THE TURBINES DURING POWER OPERATION IS REPRESENTED BY A NEGATIVE FILL. OPERATION OF THE TURBINE STOP VALVES FOLLOWING A TRIP IS SIMULATED BY THIS VALVE. A TURBINE TRIP SIGNAL FOLLOWS A REACTOR TRIP SIGNAL BY A SPECIFIED DELAY TIME (SEE THE TRIP DESCRIPTIONS). THE VALVE IS THEN RAMPED CLOSED OVER A 0.01 SEC. INTERVAL

VEPCO RETRAN MODELS
STEAM GENERATOR MODEL
(ONE LOOP MODEL)

PARAMETER	OPTION/VALUE
NO. PRIMARY VOLUMES	5 (INCLUDING INLET PLENUM-SEE SECTION I OF THIS SUPPLEMENT FOR NODING DESCRIPTION
NO. SECONDARY VOLUMES	1 (SEE SECTION I)
SECONDARY SIDE PHASE SEPARATION MODEL	SECONDARY SIDE IS TREATED AS A SEPARATED VOLUME. THE RETRAN BUBBLE RISE MODEL IS USED. A VERY LOW MIXTURE QUALITY IS SPECIFIED FOR STEADY STATE INITIALIZATION. THIS RESULTS IN A LARGE VALUE FOR THE BUBBLE RISE VELOCITY, SIMULATING THE EFFECTS OF THE MOISTURE SEPARATION EQUIPMENT WITH ESSENTIALLY PERFECT PHASE SEPARATION.
EAT CONDUCTORS	
NO.	4 (SEE SECTION I FOR GEOMETRIC DESCRIPTION)
MATERIALS PROPERTIES	INCONEL ALLOY 600
POST-CHF HEAT TRANSFER	
INSIDE	DOUGALL-ROHSENOW
OUTSIDE	DOUGALL-ROHSENOW
FLUID BOUNDARY CONDITIONS	
MAIN FEEDWATER	TIME-DEPENDENT FILL JUNCTION WITH SPECIFIED MASS FLOW RATE AND ENTHALPY. FLUID ENTHALPY IS ADJUSTED DURING STEADY STATE INITIALIZATION FOR PRIMARY/SECONDARY ENERGY BALANCE

VEPCO RETRAN MODELS
STEAM GENERATOR MODEL
(ONE LOOP MODEL) - CONT.

FLUID BOUNDARY CONDITIONS	
AUXILIARY FEEDWATER	TIME-DEPENDENT FILL JUNCTION WITH SPECIFIED VOLUMETRIC FLOW RATE AND ENTHALPY. FILL IS INITIATED ON LOW MASS IN SECONDARY SIDE. EFFECT OF THE TIME DELAY TO PURGE HOTTER MAIN FEEDWATER FROM FEED LINES AND FEED RING IS ACCOUNTED FOR IN THE FILL TABLE INPUT.
MAIN STEAM FLOW	TIME-DEPENDENT FILL JUNCTION WITH SPECIFIED (NEGATIVE) MASS FLOW RATE. THIS IS THE POWER REMOVAL JUNCTION ON THE STEADY-STATE POWER REMOVAL SYSTEM DATA CARD
ATMOSPHERIC STEAM RELIEF VALVES	TIME-DEPENDENT FILL JUNCTION WITH SPECIFIED (NEGATIVE) MASS FLOW RATE. FILL TABLE IS TRIPPED ON/OFF ON STEAM PRESSURE. ACCUMULATION AND DEADBAND ARE NEGLECTED. IN SOME SAFETY ANALYSES (E.G. LOSS OF LOAD), THESE VALVES ARE MADE INACTIVE VIA A LONG DELAY TIME ON THE TRIP-OPEN SIGNAL.

VEPCO RETRAN MODELS
STEAM GENERATOR MODEL
(ONE LOOP MODEL) - CONT.

FLUID BOUNDARY CONDITIONS

MAIN STEAM SAFETY VALVES	PRESSURE-DEPENDENT FILL WITH WITH SPECIFIED (NEGATIVE) MASS FLOW RATE AS A FUNCTION OF STEAM PRESSURE. SETPOINT IS SET TO CORRESPOND TO THE HIGHEST PLANT VALUE (ACTUAL SETPOINTS VARY WITH EACH OF FIVE VALVES PER STEAM LINE). THIS IS CONSERVATIVE SINCE PLANT HEATUP RATES WILL BE MAXIMIZED, AND THE SAFETY VALVES ONLY OPEN ON HEATUP TRANSIENTS. THE TABLE ASSUMES 3% ACCUMULATION FROM SETPOINT TO FULL RATED FLOW CONDITIONS.

SPECIAL CONSIDERATIONS

* THE HEAT CAPACITY OF THE STEAM GENERATOR SHELLS, TUBE SHEETS AND INTERNALS (EXCLUDING THE TUBES) IS NEGLECTED. THIS IS CONSERVATIVE FOR SAFETY ANALYSES SINCE IT AMPLIFIES THE EFFECTS OF TEMPERATURE CHANGES AND RESULTS IN MORE SEVERE RESPONSES TO INITIATING EVENTS IN GENERAL.

QUALIFICATION INFORMATION

* COMPARISON TO FSAR AND OTHER LICENSING CALCULATIONS FOR INCREASE/DECREASE IN HEAT REMOVAL BY SECONDARY SYSTEM IN SECTION 5.2.3 OF TOPICAL REPORT.

* COMPARISON TO LOFTRAN HEAT REMOVAL AND SECONDARY RESPONSE DURING REACTOR TRIP AND TURBINE TRIP (SEE SECTION V OF THIS SUPPLEMENT).

* COMPARISON TO MEASURED PLANT RESPONSE TO ACCIDENTAL DEPRESSURIZATION OF MAIN STEAM SYSTEM AT NORTH ANNA (SEE SECTION 5.3.3 OF THE TOPICAL REPORT).

VEPCO RETRAN MODELS
STEAM GENERATOR MODEL
(TWO LOOP MODEL)

PARAMETER	OPTION/VALUE	
	SINGLE LOOP SIDE	DOUBLE LOOP SIDE
NO. PRIMARY VOLUMES	5 (SEE SECTION I OF THIS SUPPLEMENT)	5
NO. SECONDARY VOLUMES	2 (SEE SECTION I OF THIS SUPPLEMENT)	2
SECONDARY SIDE PHASE SEPARATION MODEL	RISERS ARE TREATED AS SEPARATED VOLUMES. THE RETRAN BUBBLE RISE MODEL IS USED. A VERY LOW MIXTURE QUALITY IS SPECIFIED FOR STEADY STATE INITIALIZATION. THIS RESULTS IN A LARGE VALUE FOR THE BUBBLE RISE VELOCITY, SIMULATING THE EFFECTS OF THE MOISTURE SEPARATION EQUIPMENT WITH ESSENTIALLY PERFECT PHASE SEPARATION.	SAME
HEAT CONDUCTORS		
NO.	4(SEE SECTION I FOR GEOMETRIC DESCRIPTION)	
MATERIALS PROPERTIES	INCONEL ALLOY 600	SAME
POST-CHF HEAT TRANSFER		
INSIDE	DOUGALL-ROHSENOW	SAME
OUTSIDE	DOUGALL-ROHSENOW**	SAME**

** EXCEPT FOR STEAMLINE BREAK CALCULATIONS. STEAMLINE BREAK USES A CONSERVATIVE HIGH CONSTANT VALUE FOR THE SECONDARY SIDE HEAT TRANSFER COEFFICIENT THROUGHOUT THE BLOWDOWN.

VEPCO RETRAN MODELS
STEAM GENERATOR MODEL
(TWO LOOP MODEL) - CONT.

FLUID BOUNDARY CONDITIONS		
MAIN FEEDWATER	TIME-DEPENDENT FILL JUNCTION WITH SPECIFIED MASS FLOW RATE AND ENTHALPY. FLUID ENTHALPY IS ADJUSTED DURING STEADY STATE INITIALIZATION FOR PRIMARY/SECONDARY ENERGY BALANCE	SAME
AUXILIARY FEEDWATER	FOR STEAM LINE BREAK, AUXILIARY FEEDWATER IS SIMULATED BY A VALVE IN THE MAIN FEEDWATER JUNCTION. UPON RECEIPT OF A FEEDLINE ISOLATION SIGNAL, THIS VALVE RAMPS TO A FRACTIONAL AREA VALUE CALCULATED TO DELIVER FULL AUXILIARY FEED FLOW TO FAULTED GENERATOR .	FOR STEAM LINE BREAK, NO AFW IS DELIVERED TO THE TWO LOOP SIDE.
	FOR OTHER ACCIDENTS, AFW IS MODELED SAME AS ONE LOOP MODEL, BUT WITH 1/3 TIMES THE FLOW	SAME AS ONE LOOP MODEL BUT WITH 2/3 TIMES THE FLOW
MAIN STEAM FLOW	TIME-DEPENDENT FILL JUNCTION WITH SPECIFIED (NEGATIVE) MASS FLOW RATE. THIS IS THE POWER REMOVAL JUNCTION ON THE STEADY-STATE POWER REMOVAL SYSTEM DATA CARD	SAME

VEPCO RETRAN MODELS
STEAM GENERATOR MODEL
(TWO LOOP MODEL) - CONT.

FLUID BOUNDARY CONDITIONS

	ONE LOOP SIDE	TWO LOOP SIDE
ATMOSPHERIC STEAM RELIEF VALVES	SAME AS ONE LOOP MODEL, BUT WITH 1/3 TIMES THE FLOW	SAME AS ONE LOOP MODEL, BUT WITH 2/3 TIMES THE FLOW
MAIN STEAM SAFETY VALVES	SAME AS ONE LOOP MODEL, BUT WITH 1/3 TIMES THE FLOW	SAME AS TWO LOOP MODEL, BUT WITH 2/3 TIMES THE FLOW

SPECIAL CONSIDERATIONS

* THE HEAT CAPACITY OF THE STEAM GENERATOR SHELLS, TUBE SHEETS AND INTERNALS (EXCLUDING THE TUBES) IS NEGLECTED. THIS IS CONSERVATIVE FOR SAFETY ANALYSES SINCE IT AMPLIFIES THE EFFECTS OF TEMPERATURE CHANGES AND RESULTS IN MORE SEVERE RESPONSES TO INITIATING EVENTS IN GENERAL.

QUALIFICATION INFORMATION

* SEE ONE LOOP MODEL DESCRIPTION

VEPCO RETRAN MODELS
PRESSURIZER MODEL

PARAMETER	OPTION/VALUE
PHASE SEPARATION MODEL	RETRAN BUBBLE RISE MODEL
EQUATION OF STATE	RETRAN NON-EQUILIBRIUM PRESSURIZER MODEL
NORMAL INITIAL CONDITION	SATURATED STEAM OVER SATURATED LIQUID (ZERO MIXTURE QUALITY)
HEATER MODEL	NONCONDUCTING HEAT EXCHANGER WITH FIXED STEADY STATE POWER PLUS INPUT TIME CONSTANT. PROPORTIONAL AND BACKUP HEATERS ARE LUMPED TOGETHER. THE HEATERS ARE CONTROLLED BY A PROPORTIONAL PLUS INTEGRAL ON/OFF PRESSURE CONTROLLER MODELED WITH THE CONTROL SYSTEM. THE SETPOINTS ARE THOSE ASSOCIATED WITH THE BACKUP HEATERS. THE CONTROL SETPOINT MAY BE RAISED OR LOWERED TO ACCOUNT FOR PRESSURE MEAUREMENT ERRORS, DEPENDING ON THE APPLICATION.

VEPCO RETRAN MODELS
PRESSURIZER MODEL (CONT.)

PARAMETER	OPTION/VALUE
SPRAY MODEL	POSITIVE FILL JUNCTION WITH FILL FLUX AND ENTHALPY CONTROLLED BY CONTROL SYSTEM SPRAY FRACTION PROPORTIONAL TO OUTPUT OF THE SAME PROPORTIONAL PLUS INTEGRAL CONTROLLER WHICH GOVERNS THE HEATERS. SPRAY IS ALSO ASSUMED TO BE DIRECTLY PROPORTIONAL TO COLD LEG FLOW RATE (SPRAY IN PLANT IS DRIVEN BY COLD LEG DYNAMIC HEAD). SPRAY ENTHALPY IS SET EQUAL TO COLD LEG ENTHALPY A NEGATIVE FILL JUNCTION IS USED TO REMOVE MASS FROM THE COLD LEG EQUIVALENT TO THAT BEING ADDED TO THE PRESSURIZER VIA THE SPRAY JUNCTION.
POWER OPERATED RELIEF (PORV) MODEL	TWO PORV'S ARE MODELED, EACH AS A TIME DEPENDENT (NEGATIVE) FILL JUNCTION. THE FILL TABLE FOR PORV #1 IS TRIPPED ON/OFF ON PRESSURIZER PRESSURE. DEADBAND AND ACCUMULATION ARE NOT MODELED. WHEN THE PORV IS "OPEN", THE MASS REMOVAL RATE IS CONSTANT. PORV #2 IS MODELLED IN THE SAME MANNER, EXCEPT IT IS TRIPPED ON/OFF BASED ON THE OUTPUT OF THE SAME PROPORTIONAL PLUS INTEGRAL CONTROLLER WHICH CONTROLS THE HEATERS AND SPRAYS.

VEPCO RETRAN MODELS
PRESSURIZER MODEL (CONT.)

PARAMETER	OPTION/VALUE
PRESSURIZER SAFETY VALVE MODEL	THE THREE SAFETY VALVES ARE MODELED BY A SINGLE PRESSURE-DEPENDENT NEGATIVE FILL JUNCTION. THE ASSOCIATED FILL TABLE IS ACTIVATED BY HIGH PRESSURIZER PRESSURE. THE MASS FLOW VS PRESSURE TABLE WAS CONSTRUCTED BY ASSUMING 3% ACCUMULATION. BLOWDOWN IS NOT MODELED.

PRESSURIZER MODEL QUALIFICATION DATA:

- PRESSURIZER RESPONSE DURING INSURGE AND OUTSURGE COMPARED TO MEASURED PLANT DATA IN NORTH ANNA COOLDOWN ANALYSIS PRESENTED IN VEP-FRD-41 SECTION 5.3.3

- PRESSURE RESPONSE COMPARED TO VENDOR RESULTS FOR NUMEROUS SAFETY ANALYSES IN VEP-FRD-41 SECTION 5.2

- PRESSURE RESPONSE COMPARED TO VEPCO GENERATED LOFTRAN RESULTS IN SECTION V OF THIS SUBMITTAL

III. VEPCO RETRAN CONTROL SYSTEM MODELS DESCRIPTION/QUALIFICATION

III. VEPCO RETRAN CONTROL SYSTEM MODELS DESCRIPTION/QUALIFICATION

SCHEDULE FOR COMPLETION: AUGUST 15, 1984

WHERE USED:

- OVERTEMPERATURE/OVERPOWER DELTA-T TRIPS
- SIGNAL CONDITIONING FOR LOW PRESSURE TRIP
- PRESSURIZER HEATER & SPRAY CONTROL SYSTEM
- GENERATION OF POWER FEEDBACK REACTIVITY FUNCTION
- CALCULATION OF BORON TRANSPORT AND MIXING FOLLOWING SAFETY INJECTION
- GENERATION OF MAIN STEAM ISOLATION VALVE FLOW AREA VS TIME FOLLOWING RECEIPT OF MAIN STEAM ISOLATION SIGNAL
- CALCULATION OF NORMALIZED STEAM GENERATOR ENERGY REMOVAL RATES, INTEGRATED BREAK MASS AND ENERGY RELEASES, ETC. FOR EDITING PURPOSES

IV. INPUT OPTIONS

VEPCO RETRAN MODELS
INPUT OPTIONS

OPTION	WHERE USED	DESCRIPTION
BUBBLE RISE MODEL	STEAM GENERATORS	SEE STEAM GENERATOR DESCRIPTION UNDER COMPONENT MODELS
	PRESSURIZER	SEE PRESSURIZER DESCRIPTION UNDER COMPONENT MODELS
CENTRIFUGAL PUMP MODELS	REACTOR COOLANT PUMP	SEE RCP DESCRIPTION UNDER COMPONENT MODELS
VALVE MODELS	MAIN STEAMLINE ISOLATION	SEE COMPONENT MODELS
	MAIN FEEDLINE ISOLATION	"
	TURBINE STOP VALVES	"
	HIGH HEAD SI PUMP ACCELERATION MODEL	"

VEPCO RETRAN MODELS
INPUT OPTIONS

OPTION	WHERE USED	DESCRIPTION
GENERALIZED DATA TABLES	VALVE AREA TABLES	SEE COMPONENT MODEL DESCRIPTIONS
	DOPPLER POWER COEFFICIENT	FOR 'SLOW' TRANSIENTS WHERE THE POWER COEFFICIENT CONCEPT IS APPROPRIATE, A FUNCTION GENERATOR CONTROL BLOCK IS USED TO GENERATE (NEGATIVE) REACTIVITY FEEDBACK AS A FUNCTION OF NORMALIZED CORE POWER. A TABLE IS GENERATED BY INTEGRATING THE DOPPLER POWER COEFFICIENT AS A FUNCTION OF POWER AND CONVERTING TO DOLLAR REACTIVITY VALUES BY DIVIDING BY THE DELAYED NEUTRON FRACTION. THE CONTROL BLOCK NUMBER OF THE FUNCTION GENERATOR IS REFERENCED ON THE SCRAM TABLE (141XYY) DATA CARDS.

VEPCO RETRAN MODELS
INPUT OPTIONS (CONT)

OPTION	WHERE USED	DESCRIPTION
GENERALIZED DATA TABLES	MODERATOR TEMPERATURE DEFECT	FOR CASES WHERE THE VARIATION OF MODERATOR TEMPERATURE COEFFICIENT WITH TEMPERATURE IS A SIGNIFICANT EFFECT (STEAM LINE BREAK ONLY) A FUNCTION GENERATOR IS USED IN CONJUNCTION WITH A GENERALIZED DATA TABLE THE DATA TABLE CONTAINS ENTRIES OF REACTIVITY (IN DOLLARS) VS MODERATOR TEMPERATURE. THE INPUT (FORCING) FUNCTION FOR THE GENERATOR IS A WEIGHTED AVERAGE OF THE FLUID TEMPERATURES OF ALL THE CORE VOLUMES, WHICH IS ALSO GENERATED WITH THE CONTROL SYSTEM MODELS. FOR STEAM LINE BREAK, THE CORE VOLUMES ASSOCIATED WITH THE 'COLD', OR FAULTED, LOOP RECEIVE A HIGHER WEIGHTING THAN THE 'HOT', OR INTACT LOOPS. FOR STEAM LINE BREAK, REACTIVITY FEEDBACK SO GENERATED IS CHECKED FOR CONSERVATISM AGAINST DETAILED 3-D NEUTRONICS CALCULATIONS.

VEPCO RETRAN MODELS
INPUT OPTIONS (CONT)

OPTION	WHERE USED	DESCRIPTION
FILL TABLES	MAIN FEEDWATER	SEE STEAM GENERATOR COMPONENT DESCRIPTIONS
	AUXILIARY FEEDWATER	"
	ATMOSPHERIC STEAM RELIEF VALVES	"
	MAIN STEAM SAFETY VALVES	"
	MAIN STEAM FLOW	CONSTANT (NEGATIVE) FLOW USED FOR ALL CASES WHERE THE STEAM PRESSURE EFFECT ON LOAD IS IGNORED. FOR CASES WHERE THE TURBINE GOVERNOR VALVES ARE SIMULATED, A CONTROL SYSTEM MODEL IS USED. THE MODEL GENERATES A FLOW RATE WHICH IS THE MINIMUM OF THE DEMAND FLOW RATE OR A CONSTANT OF PROPORTIONALITY TIMES THE STEAM PRESSURE. IN THIS WAY, OPENING OF THE GOVERNOR VALVES TO MAINTAIN A FIXED LOAD UNDER REDUCED PRESSURE IS SIMULATED. THE CONSTANT OF PROPORTIONALITY IS SELECTED SUCH THAT STEAM FLOW WILL BEGIN TO VARY WITH STEAM PRESSURE ONCE THE GOVERNOR VALVES ARE WIDE OPEN.

VEPCO RETRAN MODELS
INPUT OPTIONS (CONT)

OPTION	WHERE USED	DESCRIPTION
FILL TABLES	PRESSURIZER POWER OPERATED RELIEF VALVES	SEE PRESSURIZER MODEL DESCRIPTION
	PRESSURIZER SAFETY VALVES	"
	PRESSURIZER SPRAY	"
	SAFETY INJECTION	SAFETY INJECTION IS MODELED AS A PRESSURE-DEPENDENT FILL JUNCTION CONNECTED TO THE COLD LEG VOLUME. THE FLOW RATES ARE CONSERVATIVELY LOW WITH RESPECT TO BEST ESTIMATE CALCULATIONS. THE EFFECTS OF PUMP ACCELERATION ARE MODELED WITH A TIME-DEPENDENT VALVE, AS DISCUSSED IN IN THE COMPONENT DESCRIPTIONS.

VEPCO RETRAN MODELS
INPUT OPTIONS (CONT)

OPTION	WHERE USED	DESCRIPTION
HEAT CONDUCTOR MODELS	STEAM GENERATORS	SEE VOLUME AND FLOW PATH NETWORK AND STEAM GENERATOR COMPONENT DESCRIPTIONS.
	CORE	SEE ATTACHED DESCRIPTION

VEPCO RETRAN MODELS
INPUT OPTIONS (CONT)
CORE HEAT CONDUCTOR MODELS

PARAMETER	OPTION/VALUE	
	ONE LOOP MODEL	TWO LOOP MODEL
NO. CONDUCTORS	3 (VEP-FRD-41 FIG. 3.1)	8 (VEP-FRD-41 FIG. 3.2)
GEOMETRY	CYLINDRICAL	CYLINDRICAL
NO. MATERIAL REGIONS	3 - UO2 GAP CLAD(ZIRCALLOY)	3 - UO2 GAP CLAD(ZIRCALLOY)
PELLET POWER DISTRIBUTION	UNIFORM	UNIFORM
GAP EXPANSION MODEL	YES	YES
POST-CHF HEAT TRANSFER	DOUGALL-ROHSENOW	DOUGALL-ROHSENOW

VEPCO RETRAN MODELS
INPUT OPTIONS (CONT)
MATERIALS PROPERTIES TABLES

MATERIAL	PROPERTY	TEMP RANGE °F	NO.PTS	SOURCE
UO2	THERM. COND.	100-5072	12	ANCR-1263
	HEAT CAPACITY	0-5072	14	"
	LINEAR EXP. COEFF.	0-4892	12	"
ZIRCALLOY CLAD	THERM. COND.	100-2012	11	ANCR-1263
	HEAT CAPACITY	200-2000	15	"
	LINEAR EXP. COEFF.	200-1430	8	"
INCONEL .. G. TUBES	THERM. COND.	200-1800	9	HUNTINGTON ALLOYS CORP. TECHNICAL DATA
	HEAT CAPACITY	200-1652	10	"
	LINEAR EXP. COEFF.	70-1000	2	"

VEPCO RETRAN MODELS
INPUT OPTIONS (CONT)
MATERIALS PROPERTIES TABLES

MATERIAL	PROPERTY	TEMP RANGE °F	NO.PTS	SOURCE
FUEL/CLAD GAP	THERM. COND.	-	1	VALUE ADJUSTED TO MATCH FUEL AVG TEMP TO STEADY STATE DESIGN CODES.
	HEAT CAPACITY	-	1	KREITH, "HEAT TRANSFER", 2ND ED.
	LINEAR EXP. COEFF.	-	1	USED 0.0
S. G. TUBE "CRUD"	THERM. COND.	-	1	VALUE ADJUSTED TO YIELD DESIGN HEAT TRANSFER AREA AT DESIGN HFP STEAM PRESSURE DURING STEADY STATE INITIALIZATION.
	HEAT CAPACITY	-	1	USED ARBITRARILY SMALL VALUE (E-4)
	LINEAR EXP. COEFF.	-	1	USED INCONEL VALUE

VEPCO RETRAN MODELS
INPUT OPTIONS

OPTION	WHERE USED	DESCRIPTION
NON-CONDUCTING HEAT EXCHANGERS	PRESSURIZER HEATERS	SEE PRESSURIZER MODEL DESCRIPTION. THERMAL TIME CONSTANT IS ESTIMATED ASSUMING FREE CONVECTION AT THE ROD SURFACE.
POWER CALCULATION OPTION	-	MODEL = 3 - ONE PROMPT NEUTRON GROUP - SIX DELAYED NEUTRON GROUPS (RETRAN DEFAULT PRECURSOR DECAY CONSTANTS AND YIELD FRACTIONS ARE USED) - ELEVEN DELAYED GAMMA EMITTERS - HEAVY ISOTOPE (U239/NP239) DECAY (EXCEPT STEAMBREAK, WHERE OMMITTED).

VEPCO RETRAN MODELS
INPUT OPTIONS

OPTION	WHERE USED	DESCRIPTION
SPECIFIED HEAT TRANSFER COEFFICIENT	SECONDARY SIDE OF STEAM GENERATOR	THIS OPTION IS USED ONLY WITH STEAMLINE BREAK CALCULATIONS. A CONSERVATIVELY HIGH CONSTANT VALUE WHICH EXCEEDS THE NUCLEATE BOILING VALUE THROUGHOUT THE TRANSIENT IS USED. THUS NO CREDIT IS TAKEN FOR DNB OR LOCAL TUBE DRYOUT DURING THE TRANSIENT.
STEADY STATE INITIALIZATION OPTION	ONE-LOOP AND TWO-LOOP MODELS	USE FOR ALL NON-RESTART CALCULATIONS
NON-EQUILIBRIUM PRESSURIZER	ONE-LOOP AND TWO-LOOP MODELS	USED FOR ALL CALCULATIONS (SEE COMPONENT MODELS FOR FURTHER DETAILS ON PRESSURIZER MODEL).

VEPCO RETRAN MODELS
INPUT OPTIONS

OPTION	WHERE USED	DESCRIPTION
TEMPERATURE TRANSPORT DELAY MODEL	SEE CONTROL VOLUME SECTION OF FLOW PATH NETWORK DESCRIPTION	20 MESH INTERVALS IS STANDARD INPUT FOR VOLUMES WHERE THIS OPTION IS USED.

V. COMPARISON TO ALTERNATE CODE CALCULATIONS

(TO BE SUPPLIED LATER)

VI. RETRAN SENSITIVITY STUDIES

VI. RETRAN SENSITIVITY STUDIES

This section presents the results of a series of studies performed to demonstrate the sensitivity of the calculated RETRAN response to key safety parameters for several accidents.

As discussed in Section 4.2 of VEP-FRD-41 (the Report), one of the principal applications of RETRAN at Vepco is in the integrated reload design and safety analysis process. This process has been described in detail in Reference 8 of the Report. A brief review of this process and its relationship to the sensitivity studies presented here is in order.

Following design of a core reload, a detailed characterization of the core is performed. This involves determination of the values of various "key reload parameters" (kinetics characteristics, trip reactivities, temperature coefficients, peaking factors, etc.). These parameters are then used by the safety analyst in conjunction with the current plant operating configuration and a compilation of parameter values used in previous safety analyses to evaluate the impact of the reload on plant safety. If the value of one or more key safety parameters falls outside the range defined by the input to the existing safety analyses, an evaluation of the impact on the analyses must be made. In some cases (i.e. where large parameter variations occur, or for parameters which have a strong influence on the results of the accident analyses), explicit reanalysis of the transient may be performed.

'ast analytical experience has allowed the correlation of the various accidents with those parameters which have a significant impact on them. This experience forms the basis for the selection of the specific transient cases presented in licensing correspondence such as the FSAR, and is summarized for Westinghouse plants in Reference 8 of the Report. The sensitivity studies presented here supplement Reference 8 by establishing the significance (or insignificance) of the various parameters and the limiting direction (e.g. high or low, positive or negative) for analyses performed with Vepco's RETRAN models.

In performing the sensitivity studies, a set of transients was selected which envelopes the types of non-LOCA transients which will potentially be analyzed with RETRAN. The transients selected are shown in Table VI-1. Note hat the transients cover each of the initiating event types discussed in Section 5 of the Report, i.e. changes in reactivity (both at low power and high power), changes in primary system flowrate and changes in primary to secondary heat transfer (both increases and decreases). The results of the studies are presented in the following sections.

TABLE VI-1

RETRAN SENSITIVITY STUDIES

TRANSIENTS EXAMINED

Transient Category	Initiating Event
Reactivity addition	Rod Withdrawal at Power
	Rod Withdrawal from Subcritical
Change in Primary to Secondary Heat Transfer	Loss of Load
	Excessive Load Increase
Decrease in RCS Flow Rate	Complete Loss of Flow

I.1 Rod Withdrawal at Power Studies

The rod withdrawal at power study examined variations in six parameters. Table VI-2 shows the parameters and the variations assumed for each study.

The base case consisted of a slow (4X10**-5 delta k/k per second) rod withdrawal initiated from full power. The initial conditions included the steady state errors on power, reactor coolant pressure and reactor coolant average temperature discussed in Section 4.3.2.1 of the Report. The Doppler power coefficient used was the least negative value presented in the Surry UFSAR. A moderator temperator coefficient of +3.0x10**-5 deltak/k-°F, which is the most positive value allowed by the Surry Technical Specifications, as used. Thus the base case represents minimum reactivity feedback.

Figure VI-1 shows the effect of increasing the Doppler power coefficient (in absolute value) by 25%. Only the power trace is presented, since this is the key parameter in determining thermal performance for this event. Vepco's nuclear design reliability factor for Doppler power coefficient, as documented in Reference VI.1, is 10%.

As shown, the effect of increasing the feedback is to retard the rate of power increase slightly, resulting in a slightly delayed trip. The sensitivity case trips at a slightly (about 1%) lower power due to a slight increase in average temperature, which lowers the overtemperature delta-t trip setpoint. Note that the same variation in response could have been obtained by a slight variation in the control bank reactivity insertion

rate. This is significant because the standard analysis of this accident covers a range of reactivity insertion rates, as shown in Section 5.2.1.2 of the Report. Thus the effects of reactivity feedback variations are effectively covered.

Figure VI-2 shows the effect of varying the moderator temperature coefficient from +3.0 pcm/°F (pcm=.00001 delta k/k) to -3.0 pcm/°F. (Vepco's nuclear design reliability factor for moderator coefficient is 3.0 pcm/°F - see Reference VI.1). Note that the effect of the selected variation in MTC is virtually identical to that produced by the Doppler power coefficient variation discussed above. Again the effects of variations in MTC are effectively covered in the standard analysis by examining a spectrum of insertion rates.

Figure VI-3 shows the effect of an increase in the trip reactivity worth from 4.0% delta k/k to 5.0%, or an absolute variation of 25%. (Vepco's nuclear design reliability factor for cumulative integral bank worth is 10%). As expected, the only impact is a slight increase in the rate of power decrease following the trip. The peak power reached is insignificantly impacted. Thus, trip reactivity is not a key analysis parameter for the rod withdrawal at power.

Figure VI-4 illustrates the effects of instrument uncertainties on the process parameters feeding the overtemperature delta-T trip circuitry. The base case reflects the safety analysis approach of adding an error term to the "K1" constant term in the setpoint equation. The sensitivity case

reflects best estimate assumptions (no error term). It is interesting to note that the reactor trip is generated at time zero (plus the appropriate delay times) for the sensitivity case. This is because the initial conditions still reflect the steady state control errors (power 2% above nominal, average temperature 4°F above nominal and pressure 30% below nominal). As a result, the overtemperature delta-T trip setpoint is lowered to below 102% power on a best estimate basis. Hand calculations have been performed to verify this condition.

The results in Figure VI-4 show that the effect of uncertainties on the OT delta-T trip setpoint is equivalent to about 12% in peak power for this case. The actual error term added to K1 is less than 12%. The reason the peak power increases by more than k1 is that pressurizer pressure increases in response to the power increase, which acts to raise the trip setpoint above its initial value.

The effect of 10 percent variation in prompt neutron lifetime and delayed neutron fraction on the rod withdrawal at power results was also investigated in this study. The nuclear design uncertainty factor for these parameters is 5%. The impact on the analysis results was negligible, and therefore results are not presented.

In summary, the studies have shown that the moderator and doppler coefficients can have a significant effect on the results for rod withdrawal at power. However, the variation in trip reactivity which is normally included in analyses of this event will provide a range of

transient responses which will envelope the effects of variations in these parameters. The results are not sensitive to delayed neutron fraction or prompt neutron lifetime. The uncertainty added to the overtemperature delta-T trip setpoint to account for calorimetric and process measurement errors represents a significant conservatism in the analysis.

TABLE VI-2

ROD WITHDRAWAL AT POWER SENSITIVITY STUDIES

PARAMETER STUDIED	BASE CASE VALUE	PERTURBED VALUE
Doppler Power Coefficient	Least Negative	Least Neg x1.25
Moderator Temperature Coefficient	+3.0 pcm/°F	-3.0 pcm/°F
Trip Reactivity	4.0% dk/k	5.0% dk/k
OT delta T Trip Setpoint	Nominal + Errors	Nominal
Prompt Neutron Lifetime, l*	Maximum	Maximum-10%
Delayed Neutron Coefficient	Maximum	Maximum-10%

VI.2 Rod Withdrawal from Subcritical Studies

The rod withdrawal from subcritical study examined variations in six parameters. Table VI-3 shows the parameters and the variations assumed for each study.

The base case for the study consisted of a 75 pcm/sec ramp reactivity insertion from an initial power of 10**-13 times full power. A positive moderator temperature coefficient of +10 pcm/°F was assumed. A conservatively low Doppler temperature coefficient which varies with fuel temperature was used.

Figures VI-5 to VI-7 present nuclear power, core heat flux and fuel temperature results which show the effects of reducing the moderator temperature coefficient from +10 pcm/°F to 0.0 pcm/°F, which is more realistic, but still conservative for BOL conditions. As can be seen from the results, the assumption of +10 pcm/°F, which reflects the current safety analysis assumption, is a major analysis conservatism. Use of this assumption results in increases in peak heat flux and fuel average temperature of about 25% of rated full power and 100°F, respectively, relative to the more realistic assumption. Note that even with the conservative assumption the values remain well below nominal full power temperatures and heat fluxes.

Figures VI-8 to VI-10 show the effects of increasing (in absolute value)

che Doppler temperature coefficient. The sensitivity case reflects an increase in Doppler coefficient of 25%. Again, the effect of the variation is marked, with the safety analysis assumption resulting in peak heat flux and fuel temperature which are about 20% of rated full power and 100°F higher, respectively, than the more realistic assumption.

Figures VI-11 to VI-13 present the results of a study of the effects of varying the delayed neutron fraction. The base case used a bounding high value which envelopes the maximum expected BOL delay fraction. The sensitivity case reflects a reduction of 25%, which envelopes the minimum expected BOL delay fraction. The shift in the timing of the prompt power burst reflects the fact that a prompt critical condition is reached earlier ith the reduced beta. The initial power increase is steeper due to the shorter prompt period, which is reflected in a higher peak power for the sensitivity case. This is offset by the increase in effective worth (in dollars) of the doppler feedback and trip reactivity. As a result, the reduced beta case reaches a slightly lower peak heat flux (by about 8% of rated) and slightly lower (about 40 °F) peak fuel temperatures.

The sensitivity of the rod withdrawal from subcritical to trip reactivity worth is shown in Figures VI-14 to VI-16. The sensitivity case reflects a 25% increase in the trip worth relative to the base case (5.0% delta k/k vs 4.0%). As shown , the results show a small senstivity to this parameter. The increased trip reacivity reduced fuel temperature by less than 20 °F and peak heat flux by about 2%.

The effects of the high flux trip setpoint assumption were also studied. The nominal low power range trip setpoint for Surry and North Anna is 25% of Rated Thermal Power (RTP). The base case assumption is 35% RTP, consistent with the FSAR analysis assumption. The sensitivity study examined the effects of raising this flux trip to 118% RTP, which is the UFSAR assumption for the high power range trip setpoint. Even with this large variation, the impact on peak power , heat flux and average fuel temperature was negligible. Therefore the results are not presented.

In summary, the sensitivity studies for rod withdrawal from subcritical show that key analysis parameters for this event are the Doppler temperature coefficient, moderator temperature coefficient and delayed neutron fraction. The FSAR's also indicate that the reactivity insertion rate can significantly influence the results, with high insertion rates giving more severe results. The insertion rate has not been studied here. However, as the Vepco RETRAN model gives results which are comparable to the vendor codes, as demonstrated in Sections 5.2.1.1 and 5.2.1.2 of the Report, the conclusions of the FSAR's regarding reactivity insertion rate will be valid for the Vepco RETRAN models.

TABLE VI-3

ROD WITHDRAWAL FROM SUBCRITICAL SENSITIVITY STUDIES

PARAMETER STUDIED	BASE CASE VALUE	PERTURBED VALUE
Doppler Temperature Coefficient	Least Negative	Least Neg x1.25
Moderator Temperature Coefficient	+10.0 pcm/°F	0.0 pcm/°F
Trip Reactivity	4.0% dk/k	5.0% dk/k
Delayed Neutron Coefficient	Maximum	Maximum-25%
High Flux Trip Setpoint	35% RTP	118% RTP

VI.3 Complete Loss of Load Sensitivity Studies

The loss of load study examined the effects of five parameters as outlined in Table VI-4.

The base case for the study simulated a turbine trip without direct reactor trip from Hot Full Power at beginning of life. The moderator temperature coefficient was assumed constant at +3.0 pcm/°F. No credit was taken for the operation of pressurizer sprays, power operated relief valves or steam generator relief valves.

Figures VI-17 to VI-20 show the effects of varying the Doppler power coefficient on nuclear power, pressurizer pressure, pressurizer water volume and average RCS temperature, respectively. The base case analysis assumed a high (most negative) value, while the sensitivity analysis assumed a 25% reduction (less negative) in this value. As can be seen from the results, the base case yielded only very slightly higher post-trip powers, temperatures and pressures. Thus the loss of load event is relatively insensitive to this parameter.

Figures VI-21 to VI-24 present results for a study of the effects of varying the moderator temperature coefficient from the base case assumption of +3.0 pcm/°f to a more realistic beginning-of-cycle value of -3.0 pcm/°F. The effects of varying this parameter are slightly more pronounced than for the Doppler power coefficient, but again the overall effects are not significant. Use of a more negative EOL value would result in a more

pronounced reduction in power, peak pressure and inlet temperature. For this reason, beginning of life is the limiting condition for this event, both from a system overpressure and DNB standpoint.

The effect of varying trip reactivity worth on the loss of load results is illustrated by Figures VI-25 to VI-28. The base case analysis assumed the standard safety analysis value of -4.0% delta k/k while the sensitivity case assumed -5.0% delta k.k. Again, the effects, while observable, are relatively small (less than 1 psi difference in peak pressure).

The effects of varying assumptions concerning system component availability on the loss of load transient were also examined. Figures VI-29 through VI-32 illustrate the effect of the pressurizer power operated relief valves and sprays on the response. As expected, these systems act to retard the rate of pressure increase and to delay the time of trip on high pressurizer pressure (Figure VI-29). Note that the peak pressurizer pressure is reduced by about 25 psi, although the capacity of the relief valves is not large enough to hold the system at their setpoint (2350 psia). Note also from Figure VI-32 that the delay in time to trip results in a larger system temperature increase (by about 11 °F). This study illustrates why analyses of the loss of load normally consider both the case with PORV's and spray (which is bounding from a DNB standpoint due to lower pressures and higher temperatures) and without them (which is bounding from a system overpressure standpoint).

The effects of the steam generator relief valves (SGRV's) were studied, and

the results are presented in Figures VI-33 to VI-36. As can be seen, this system has an insignificant effect on the transient results.

In summary, the studies show that the loss of load results are insensitive to variations in Doppler power coefficient, trip reactivity and the operation of the steam generator relief valves. Variations in moderator temperature coefficient have slightly more influence, and the most signficant factor in influencing the results is the assumption regarding the availability of PORV's and pressurizer sprays. The limiting directions for the physics parameters are: most positive moderator coefficient, most negative Doppler power coefficient and minimum trip reactivity.

TABLE VI-4

LOSS OF LOAD SENSITIVITY STUDIES

PARAMETER STUDIED	BASE CASE VALUE	PERTURBED VALUE
Doppler Power Coefficient	Most Negative	Most Neg x0.75
Moderator Temperature Coefficient	+3.0 pcm/°F	-3.0 pcm/°F
Trip Reactivity	4.0% dk/k	5.0% dk/k
Steam Generator Relief Valves	Not available	Available
Pressurizer PORV's/Sprays	Not Available	Available

VI-4. Loss of Flow Sensitivity Studies

The effects of three parameters were examined for the loss of flow studies: Doppler power coefficient, moderator temperature coefficient and trip reactivity worth. The parameter variations considered were the same as for the loss of load sensitivity studies, as described in Table VI-4.

The base case analysis consisted of a complete loss of reactor forced coolant flow from hot full power. The steady state errors on power, pressurizer pressure and RCS temperature discussed in Section 4.3.2.1 were reflected in the initial conditions. A moderator temperature coefficient of +3 pcm/°F and a bounding, most negative Doppler power coefficient were assumed.

The study results are presented in terms of nuclear power, core heat flux and pressurizer pressure. Figures VI-37 to VI-39 show the results for the Doppler power coefficient study. The base case is slightly more conservative from a DNB standpoint since the decay in core heat flux is retarded slightly. This effect is also reflected in slightly higher pressurizer pressure.

Figures VI-40 to VI-42 show the results of the trip reactivity study. Again, slightly higher post-trip heat fluxes (about 1% of rated) occur in the base case, confirming the conservatism of the safety analysis assumption. Again, the variation considered (25%) is higher than the nuclear design reliability factor associated with trip reactivity (10%)

The effects of varying moderator temperature coefficient are shown in Figures VI-43 to VI-45. The effects of the variation examined are very slight, as shown, with the more positive value giving very slightly higher powers, temperatures and heat fluxes.

These results are all consistent with the FSARs for Vepco's units regarding limiting directions for the key parameters.

VI-5. Excessive Load Increase Sensitivity Studies

The final transient examined for this study was the excessive load increase event. Three parameters were examined: Doppler power coefficient, moderator temperature coefficient and the effects of pressurizer heaters. The parameter variations considered are summarized in Table VI-5.

The base case consisted of a 10% step load increase from full power. A large negative moderator temperature coefficient which bounds low soluble boron, end of life conditions was assumed. The reactor was assumed in manual control. The effects of steam pressure and automatic operation of the turbine governor valves are included in the analysis.

Figures VI-46 to VI-50 present the results of the moderator coefficient study. The base case assumes a bounding EOL value. The sensitivity case assumed a value which was reduced (in absolute value) by a factor which is grater than the design reliability factor. As reactor power increases to match the increased load, there is a drop in coolant temperature. The magnitude of this drop provides enough positive reactivity to offset the negative reactivity resulting from the increased power. As a result, the drop in temperature is greater for the lower (in absolute value) moderator coefficient. Since the higher inlet temperatures yield lower DNBR's, the more negative MTC's are bounding for this event.

Figures VI-51 to VI-55 illustrate the effects of the Doppler power coefficient on the excessive load increase. The base case assumed a low

absolute (least negative) value for the power coefficient. The sensitivity case assumed a 25% increase in the coefficient. As in the base case, the nuclear power increases to match the increased load demand. However, with the increased power feedback, the inlet temperature undergoes a greater drop in order to offset this increased negative reactivity insertion and return the system to a steady state condition. Again, since higher inlet temperatures are limiting from a DNB standpoint, the base case (least negative Doppler power coefficient reflects the bounding assumption for this event.

Figure VI-56 compares the base case pressurizer pressure response to a sensitivity case which includes the effects of the pressurizer heaters. Plots for the other parameters are omitted sinse there is essentially no difference in the results. Since the heaters act to increase pressure which is a DNBR benefit, Vepco analyses conservatively neglect their effects.

These studies show that the key analysis parameters for the excessive load increase event are the Doppler power and moderator temperature coefficients, and that least negative values for the power coefficient and most negative values for the temperature coefficient will yield limiting results for this event. These conclusions are consistent with the FSAR's.

TABLE VI-5

EXCESSIVE LOAD INCREASE INCIDENT SENSITIVITY STUDIES

Parameter	Base Case	Sensitivity Case
Moderator Temperature Coefficient	Most Negative	0.6 * Most Negative
Doppler Power Coefficient	Least Negative	1.25 * Least Negative
Pressurizer Heaters	Inactive	Active

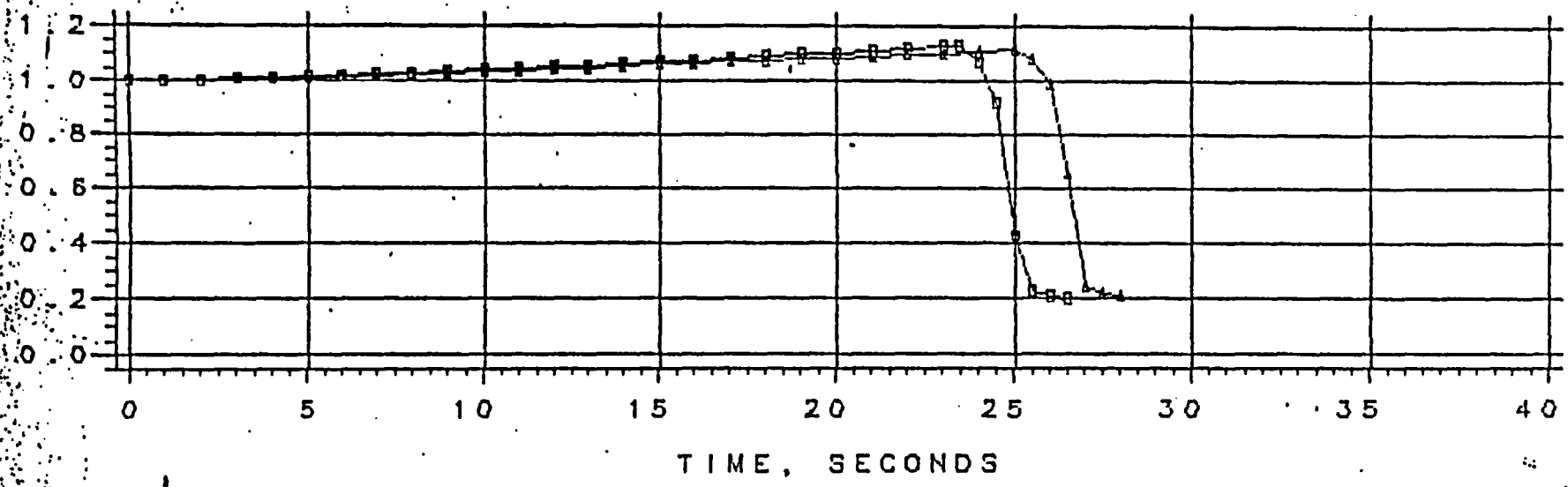
Summary

The sensitivity studies for these five accidents illustrate the impact of variations in various key analysis parameters on the accident reponse. These variations were larger than the associated nuclear design reliability factors, as documented in Reference VI-1, in each case. The studies show that Vepco's RETRAN models show the same general sensitivities as discussed in the Surry and North Anna FSAR's,

REFERENCES (SECTION VI) 1. Vepco Topical Report VEP-FRD-45A, "Nuclear Design Reliability Factors", J. G. Miller, October 1982.

FIGURE VI-1
ROD WITHDRAWAL AT POWER

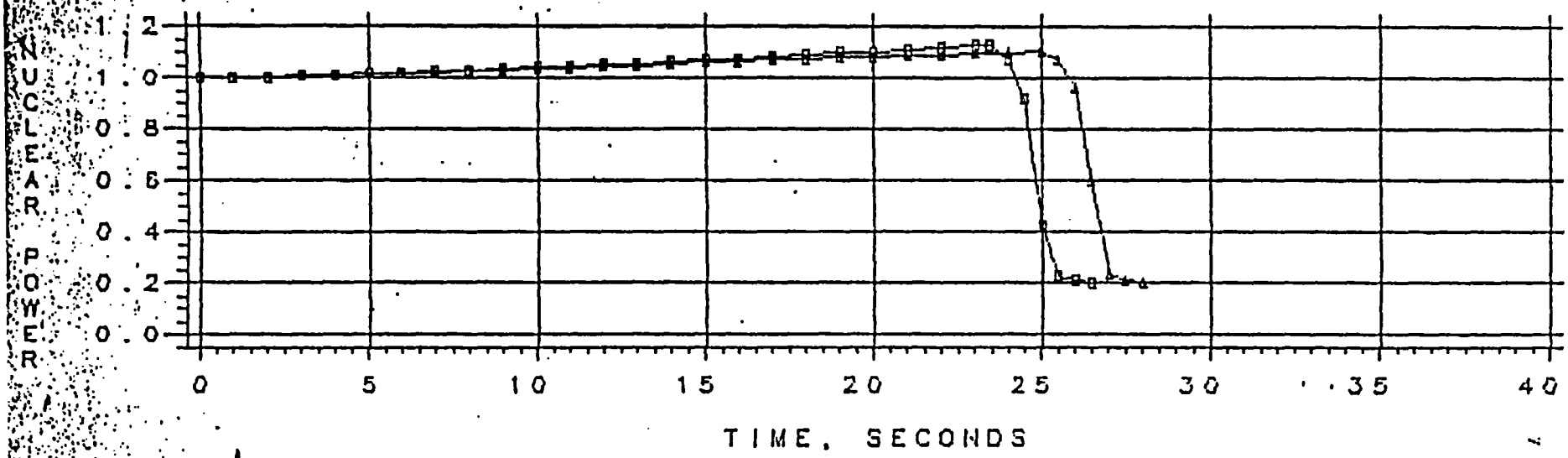
DPC STUDY



SQUARE - LEAST NEGATIVE DPC
TRIANGLE- LEAST NEGATIVE DPCX1.25

FIGURE VI-2
ROD WITHDRAWAL AT POWER

MTC STUDY

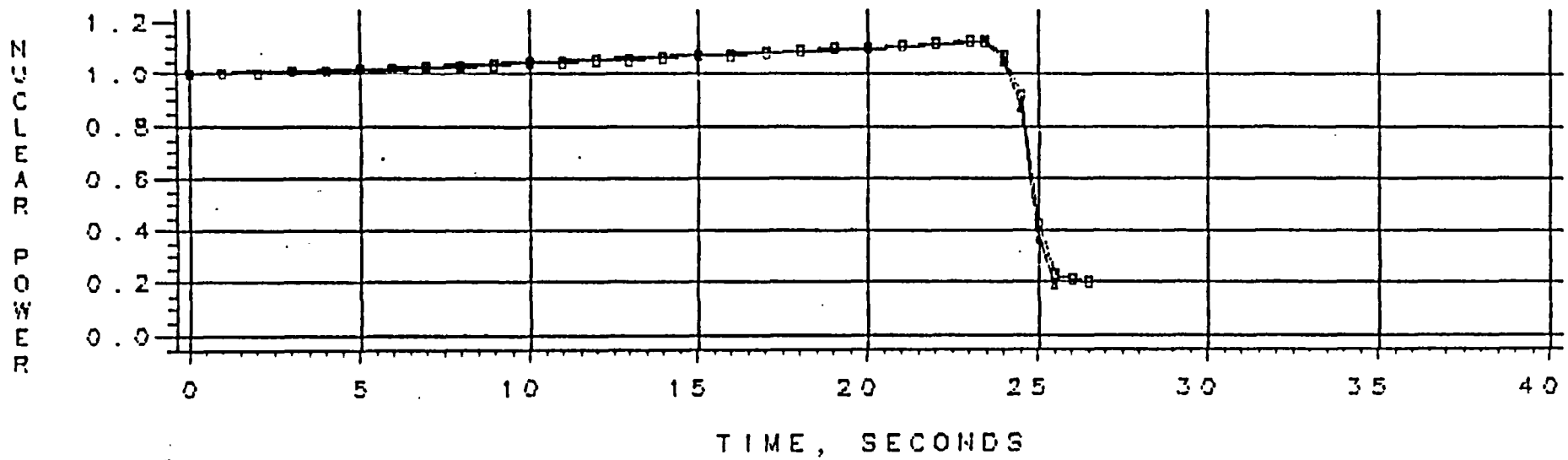


SQUARE - +3.0 PCM/F
TRIANGLE- -3.0 PCM/F

FIGURE VI-3

ROD WITHDRAWAL AT POWER

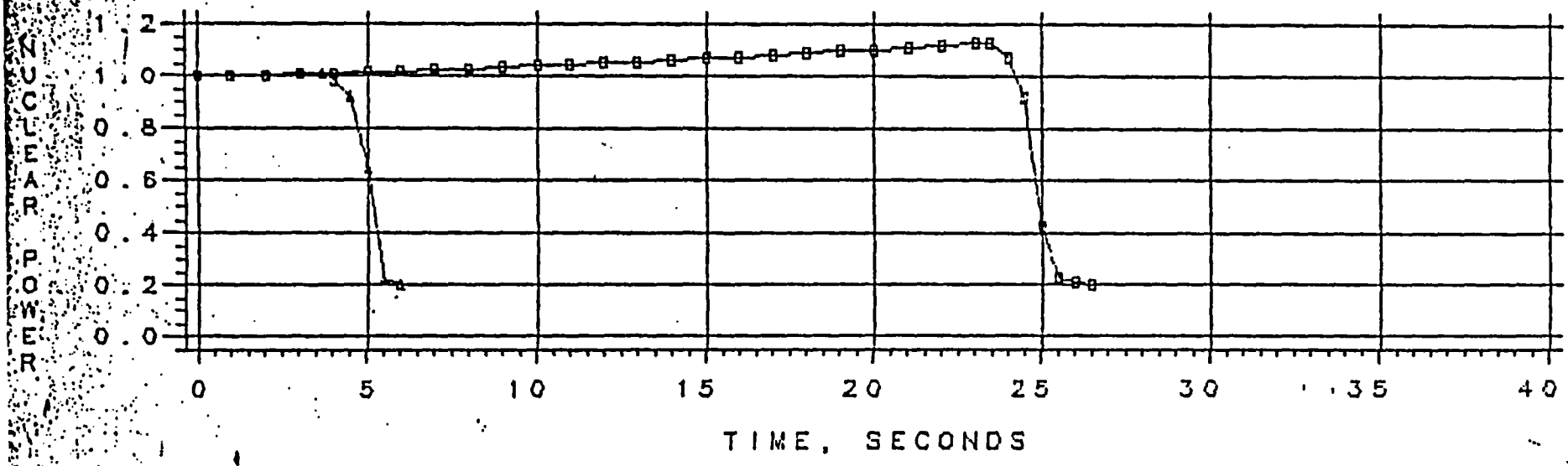
TRIP REACT STUDY



SQUARE - BASE CASE
TRIANGLE- SENS CASE

FIGURE VI-4
ROD WITHDRAWAL AT POWER

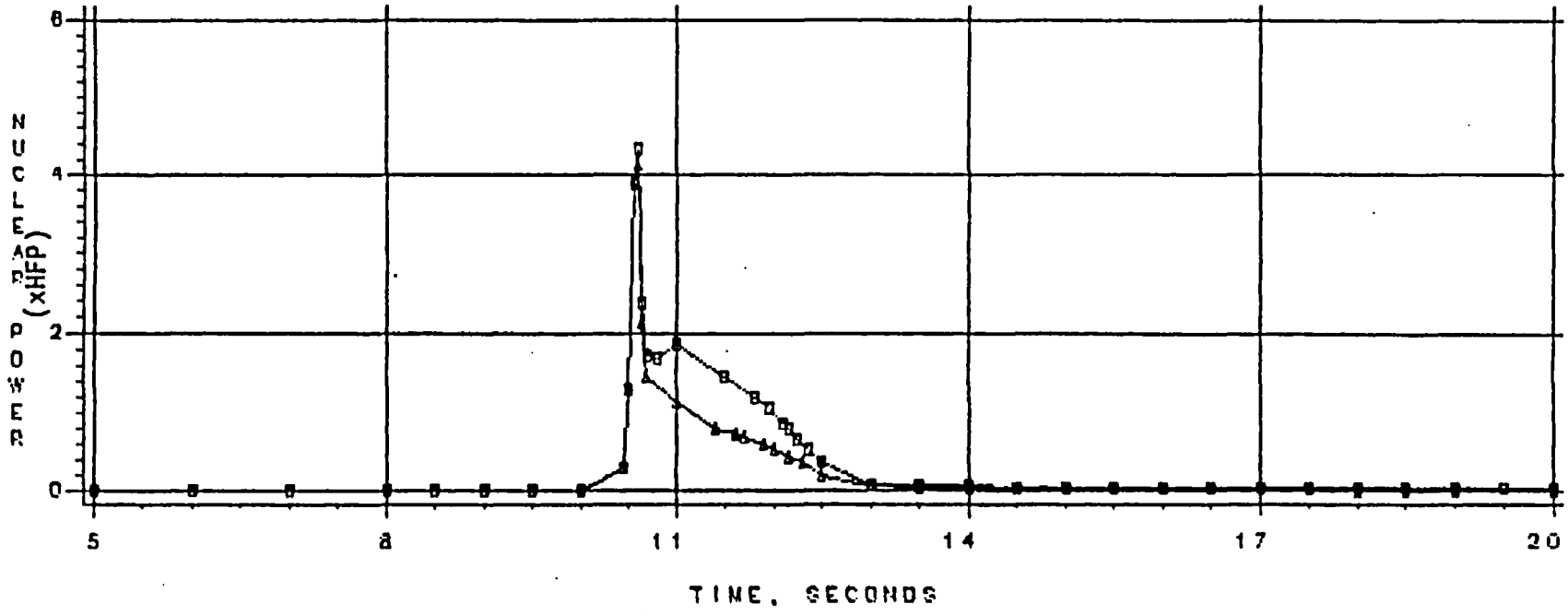
OTDT STUDY



SQUARE - DESIGN K1
TRIANGLE- NOMINAL K1

FIGURE VI-5
ROD WITHDRAWAL FROM SUBCRITICAL

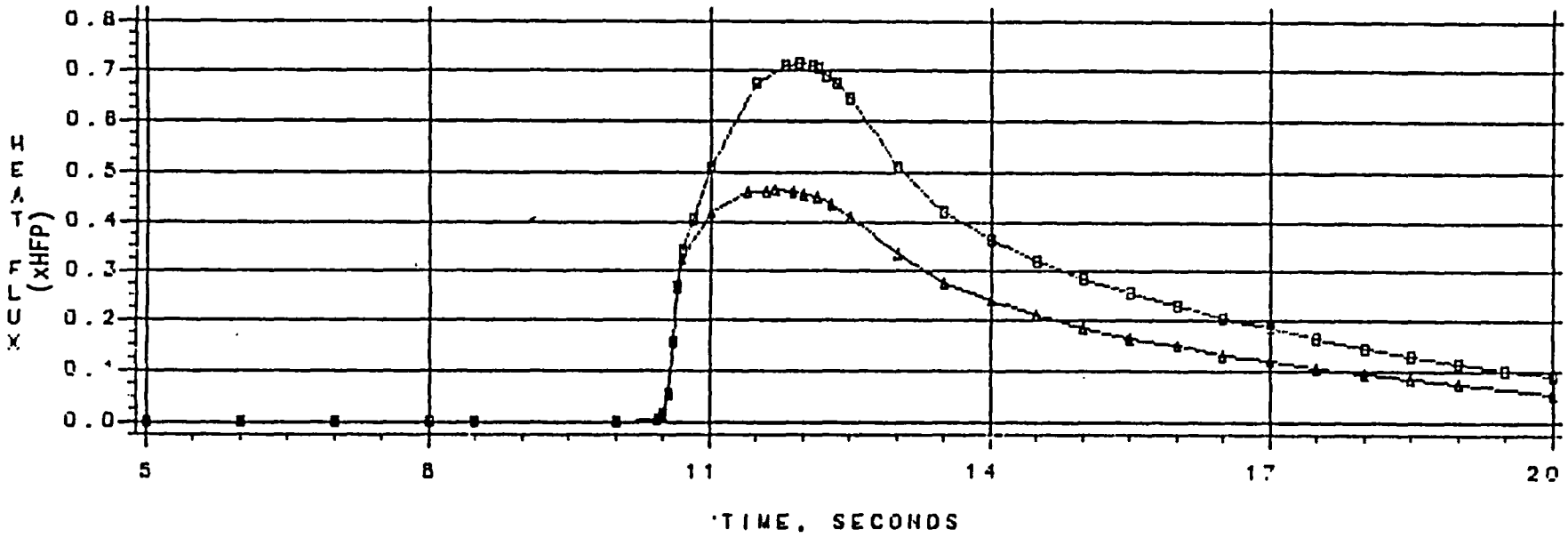
MTC STUDY



SQUARE = MTC BASE CASE
TRIANGLE= MTC 0

FIGURE VI-6
ROD WITHDRAWAL FROM SUBCRITICAL

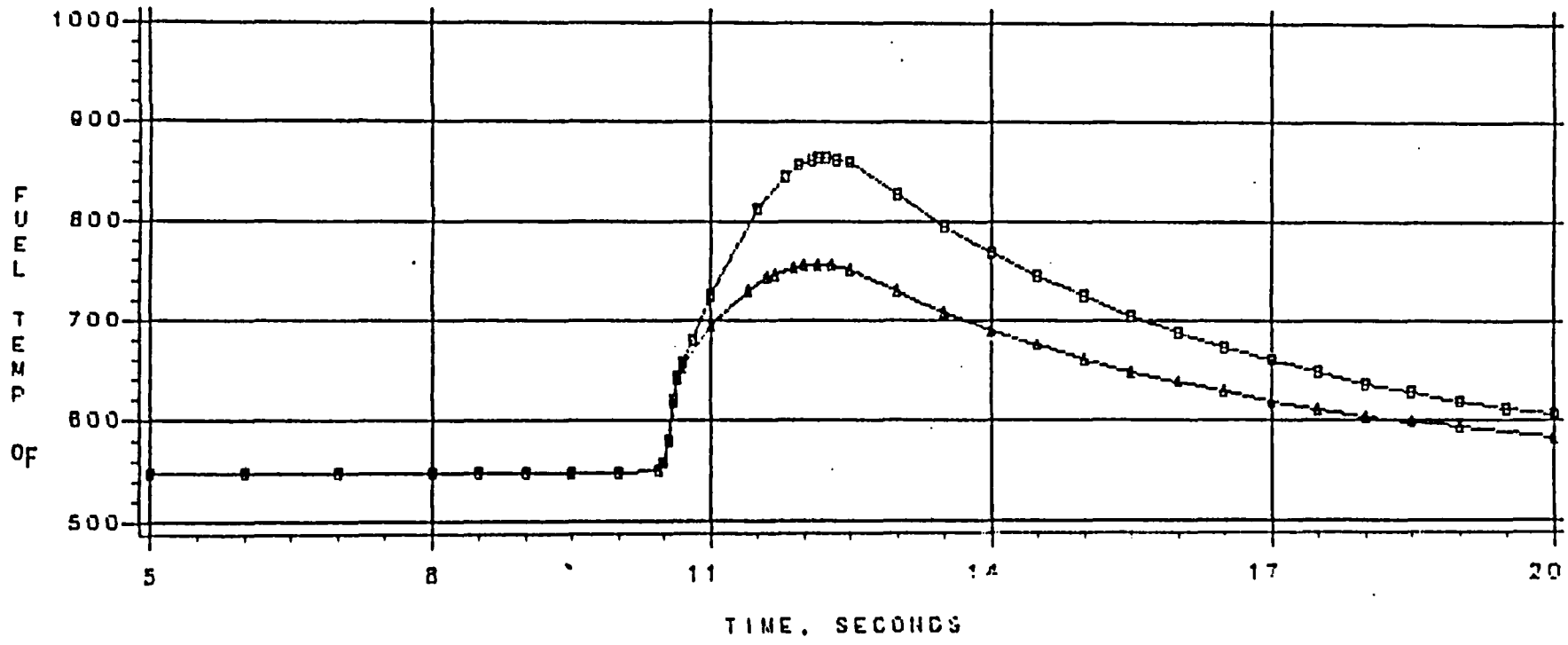
MTC STUDY



SQUARE = MTC BASE CASE
TRIANGLE= MTC 0

FIGURE VI-7
ROD WITHDRAWAL FROM SUBCRITICAL

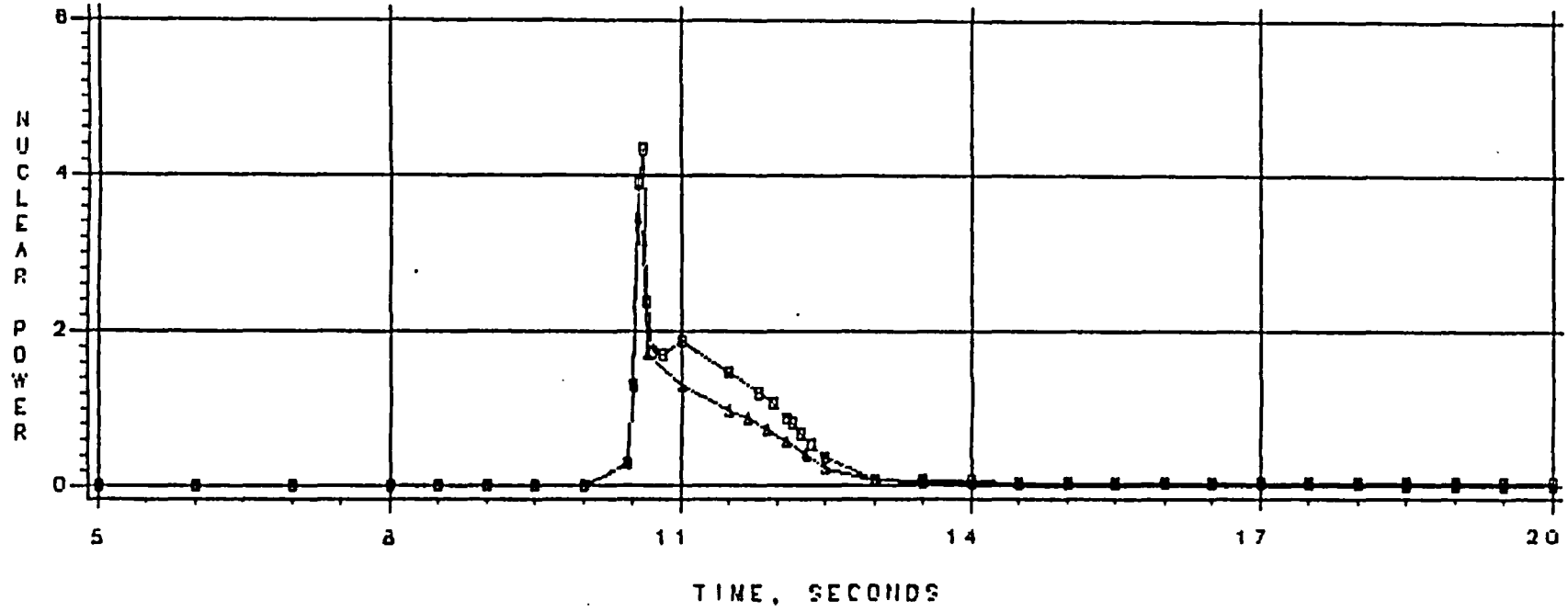
MTC STUDY



SQUARE = MTC BASE CASE
TRIANGLE= MTC 0

FIGURE VI-8
ROD WITHDRAWAL FROM SUBCRITICAL

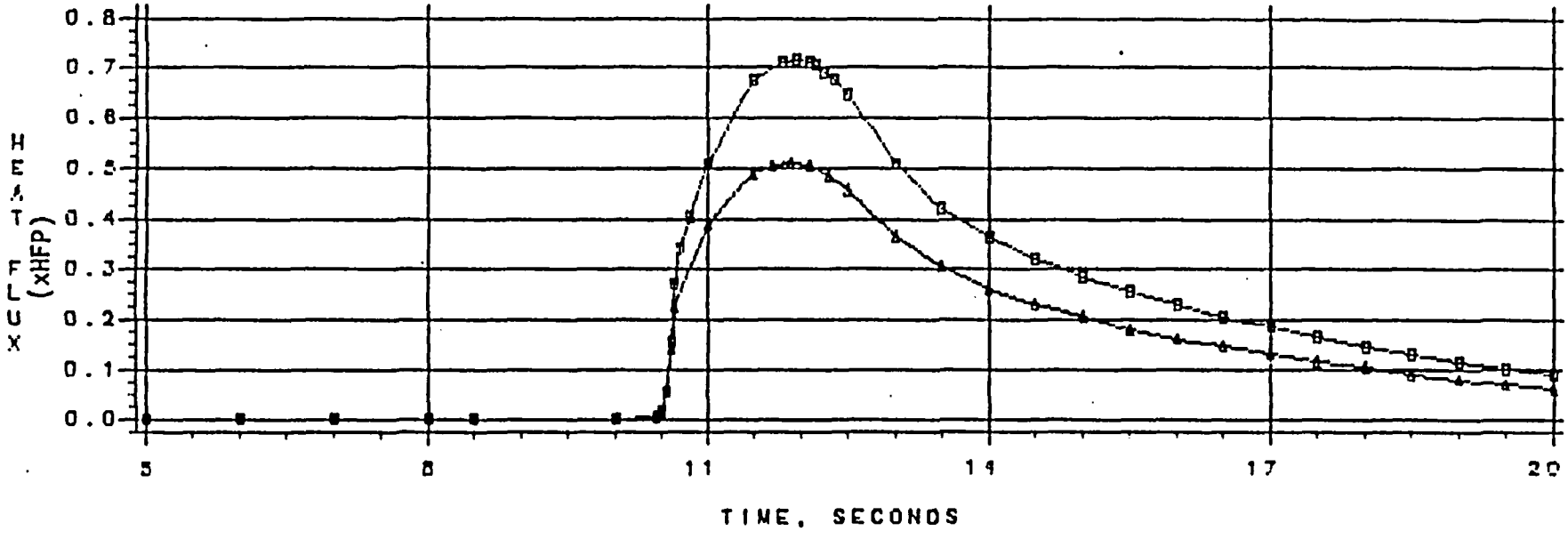
DTC STUDY



SQUARE = DTC BASE CASE
TRIANGLE= DTC +25PCT

FIGURE VI-9
ROD WITHDRAWAL FROM SUBCRITICAL

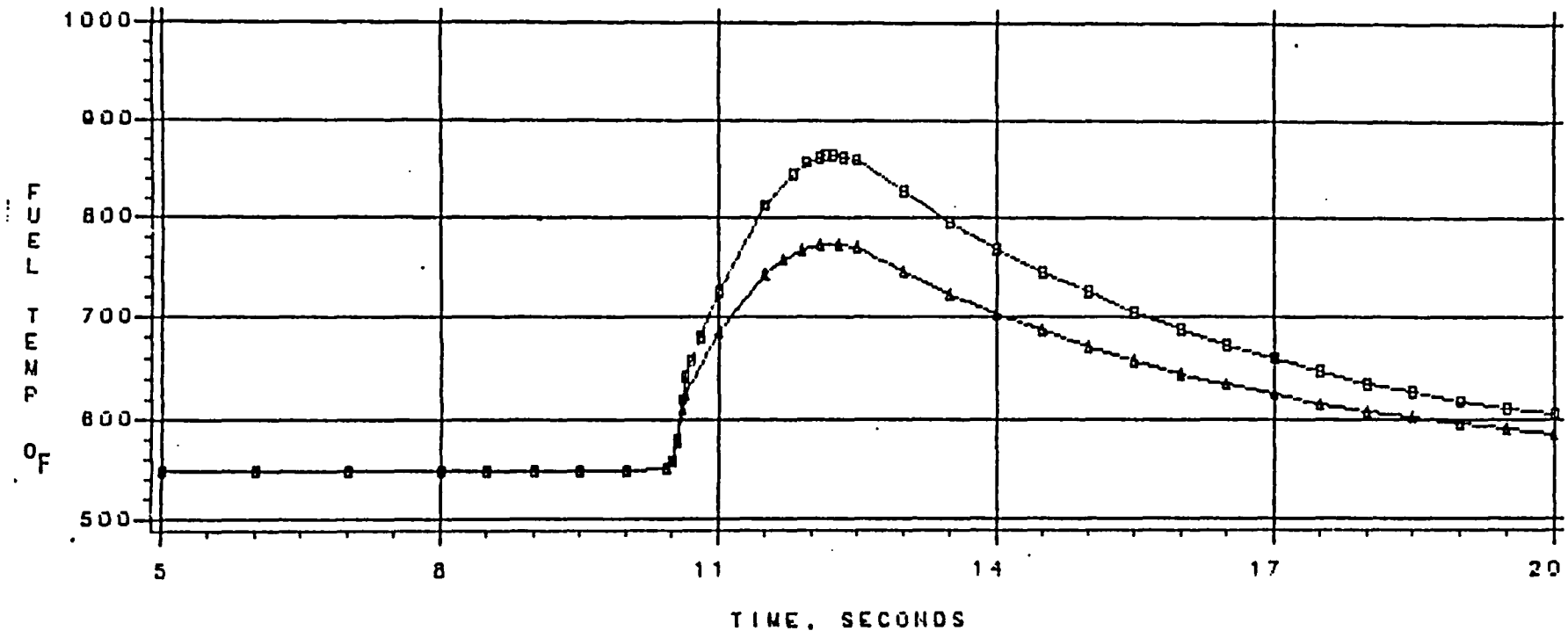
DTC STUDY



SQUARE = DTC BASE CASE
TRIANGLE= DTC +25PCT

FIGURE VI-10
ROD WITHDRAWAL FROM SUBCRITICAL

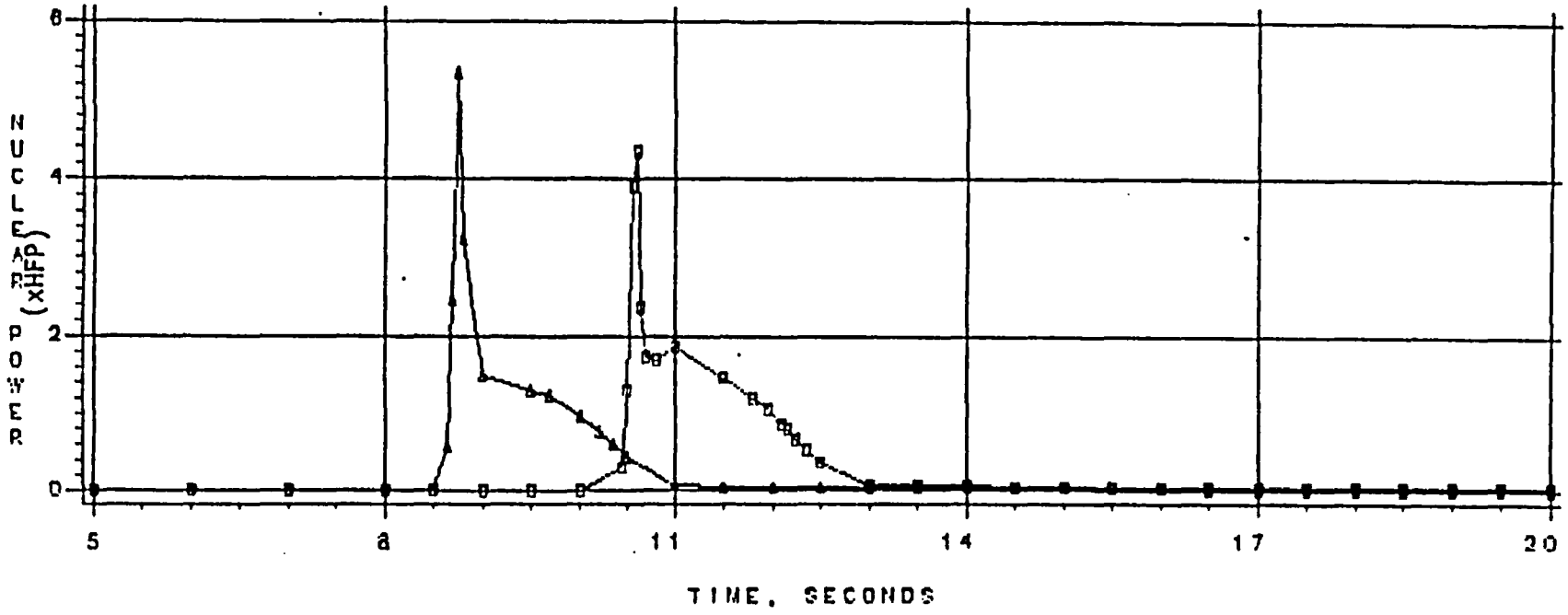
DTC STUDY



SQUARE = DTC BASE CASE
TRIANGLE= DTC +25PCT

FIGURE VI-11
ROD WITHDRAWAL FROM SUBCRITICAL

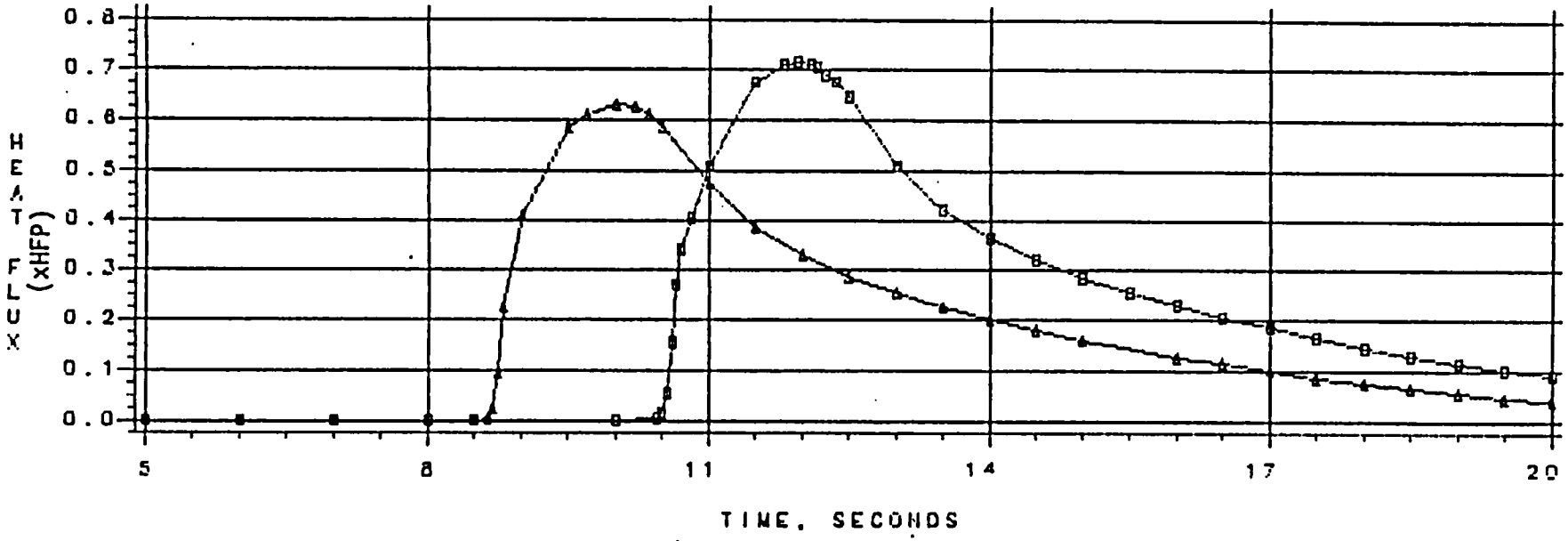
BETA STUDY



SQUARE = BETA BASE CASE
TRIANGLE= BETA -25PCT

FIGURE VI-12
ROD WITHDRAWAL FROM SUBCRITICAL

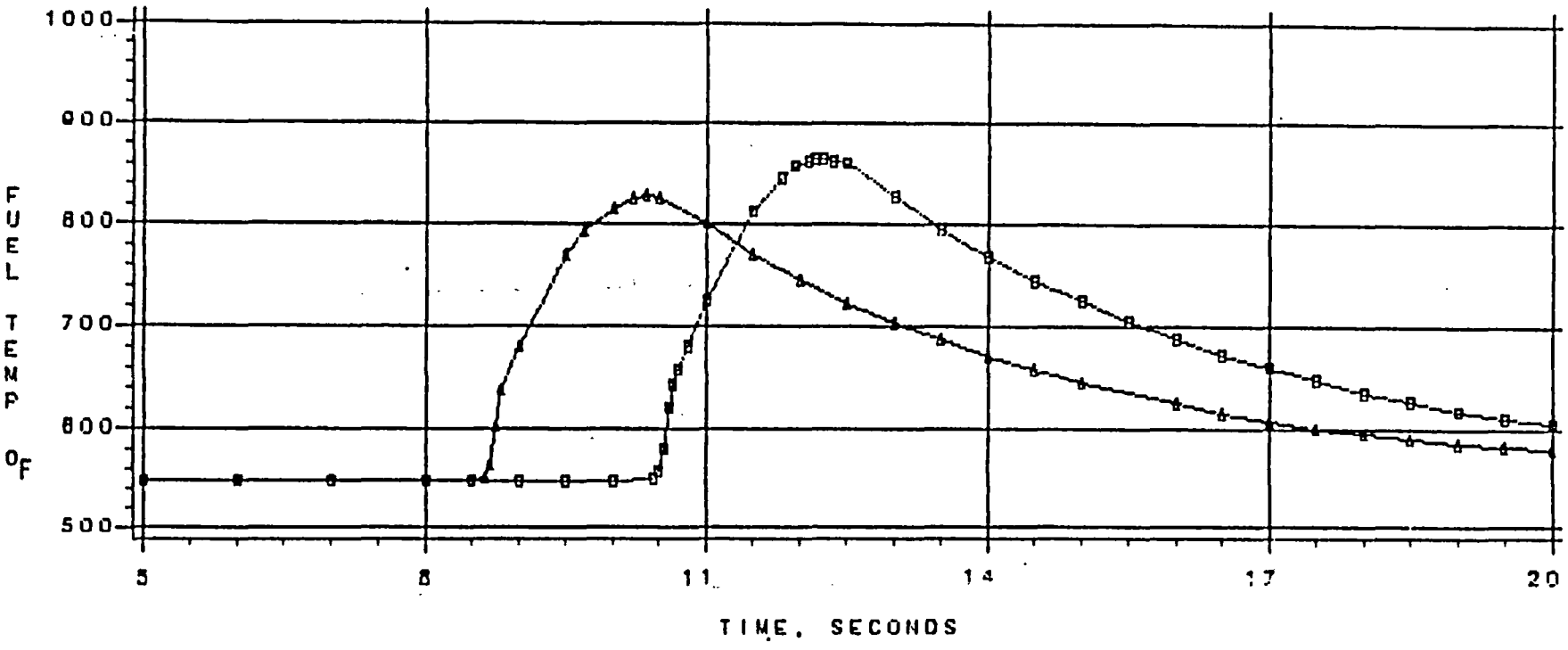
BETA STUDY



SQUARE = BETA BASE CASE
TRIANGLE= BETA -25PCT

FIGURE VI-13
ROD WITHDRAWAL FROM SUBCRITICAL

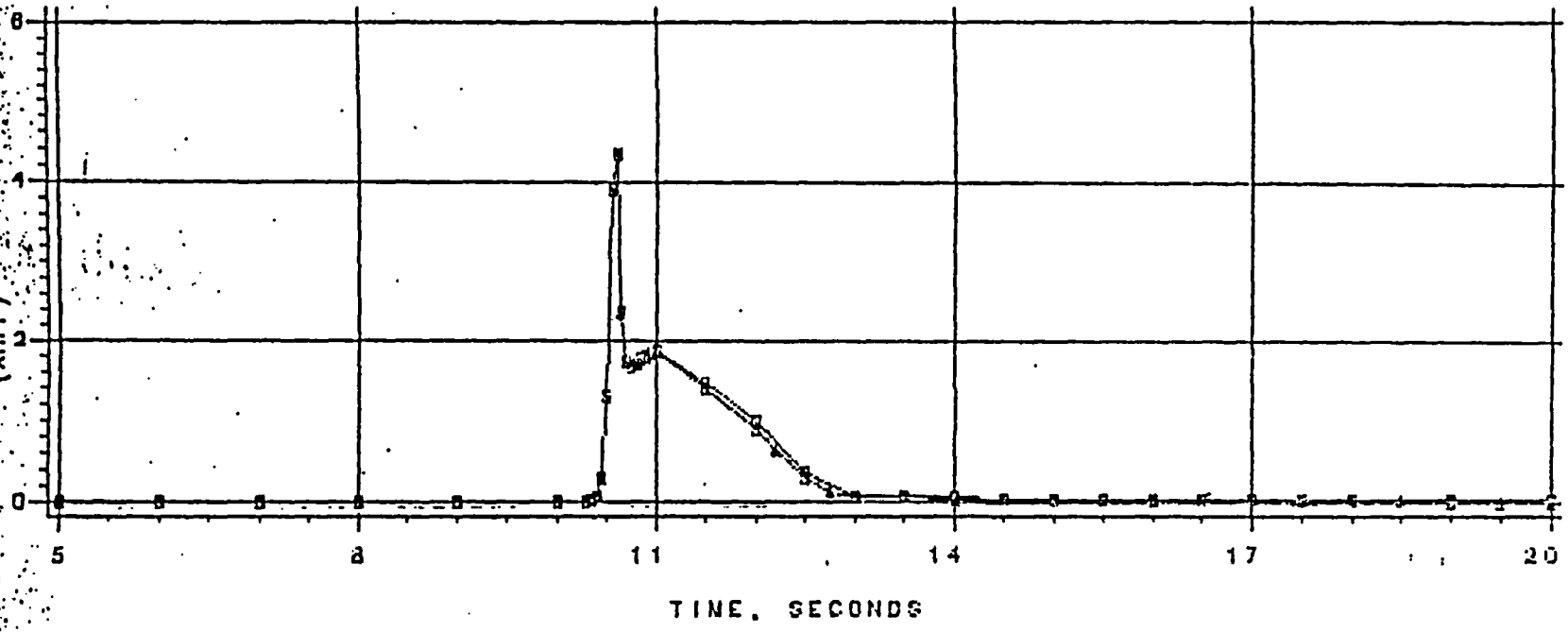
BETA STUDY



SQUARE = BETA BASE CASE
TRIANGLE= BETA -25PCT

FIGURE VI-14
ROD WITHDRAWAL FROM SUBCRITICAL

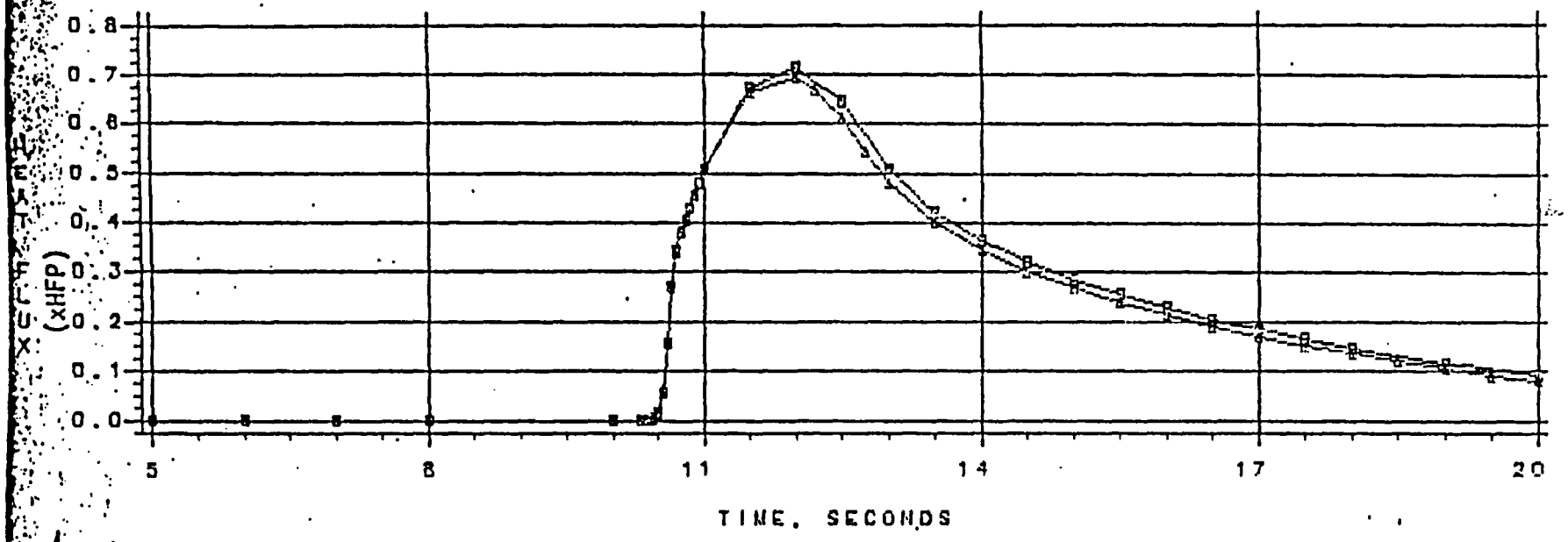
TRIP REACTIVITY STUDY



SQUARE = 4.0 TRIP
TRIANGLE= 5.0 TRIP

FIGURE VI-15
ROD WITHDRAWAL FROM SUBCRITICAL

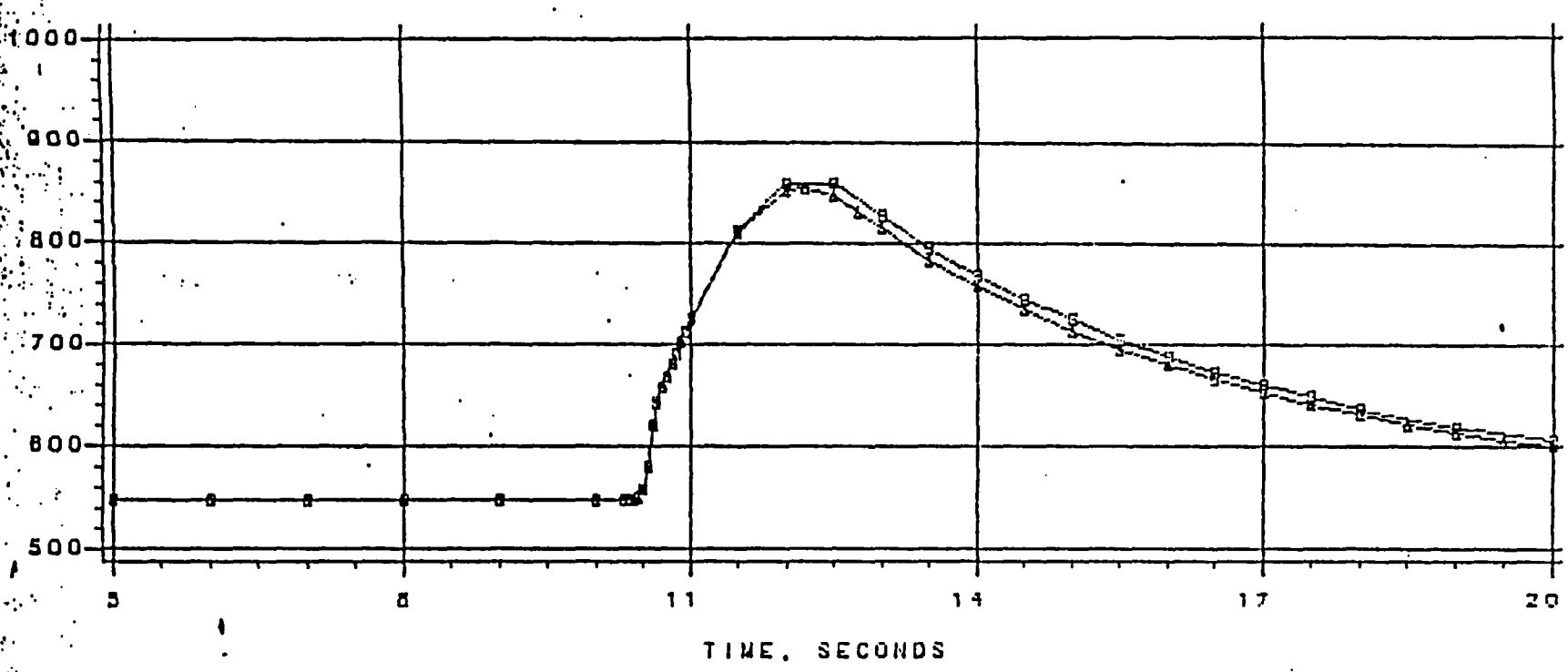
TRIP REACTIVITY STUDY



SQUARE = 4.0 TRIP
TRIANGLE= 5.0 TRIP

FIGURE VI-16
ROD WITHDRAWAL FROM SUBCRITICAL

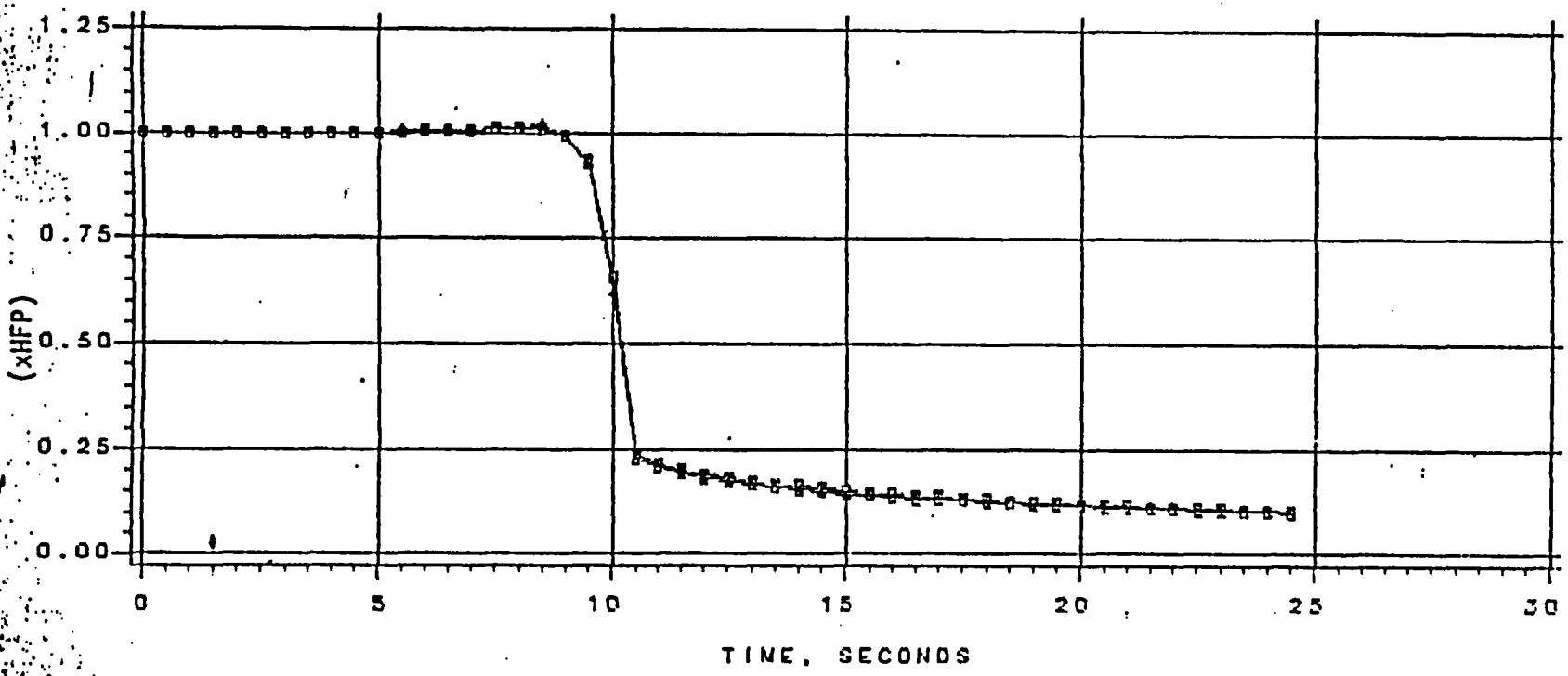
TRIP REACTIVITY STUDY



SQUARE = 4.0 TRIP
TRIANGLE= 5.0 TRIP

FIGURE VI-17
LOSS OF LOAD

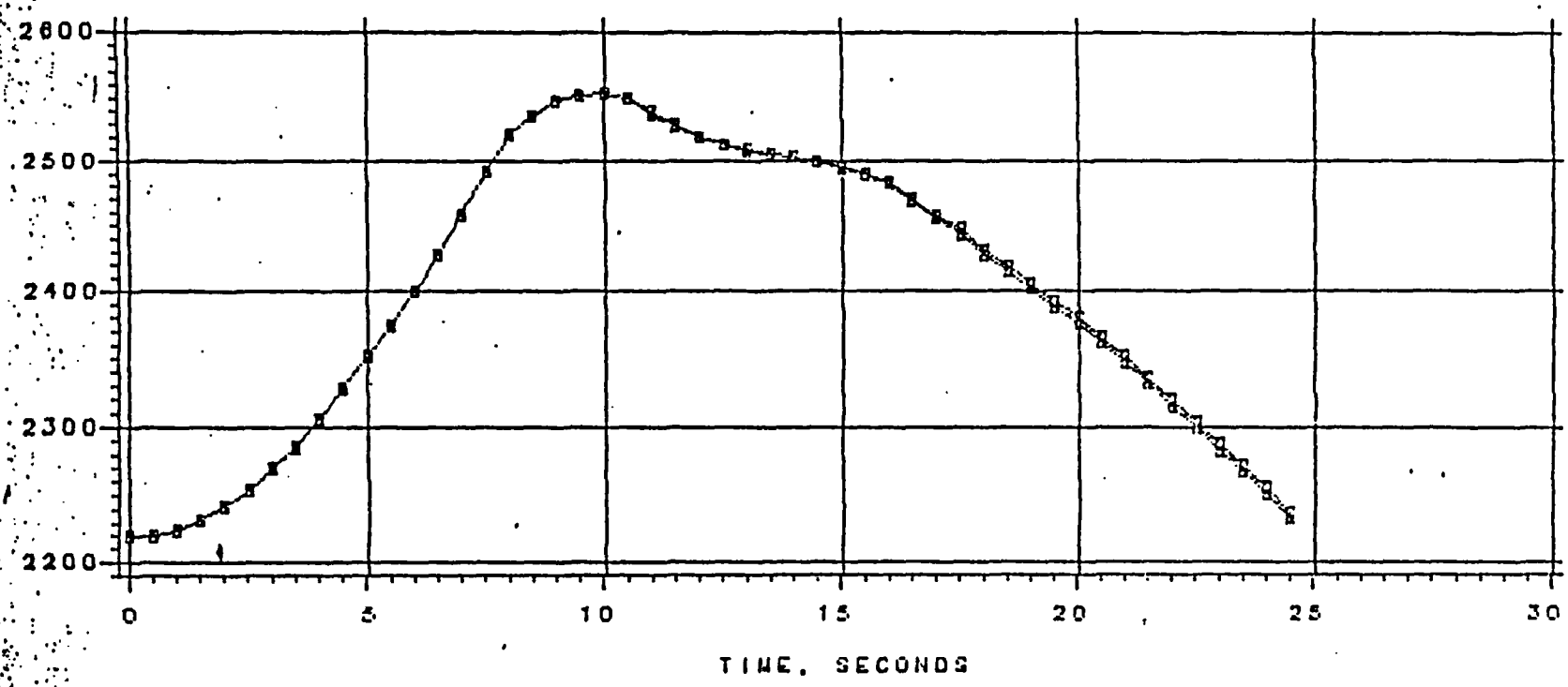
DPC STUDY



SQUARE = MOST NEGATIVE DPC
TRIANGLE= MOST NEGATIVE DPCX0.75

FIGURE VI-18
LOSS OF LOAD

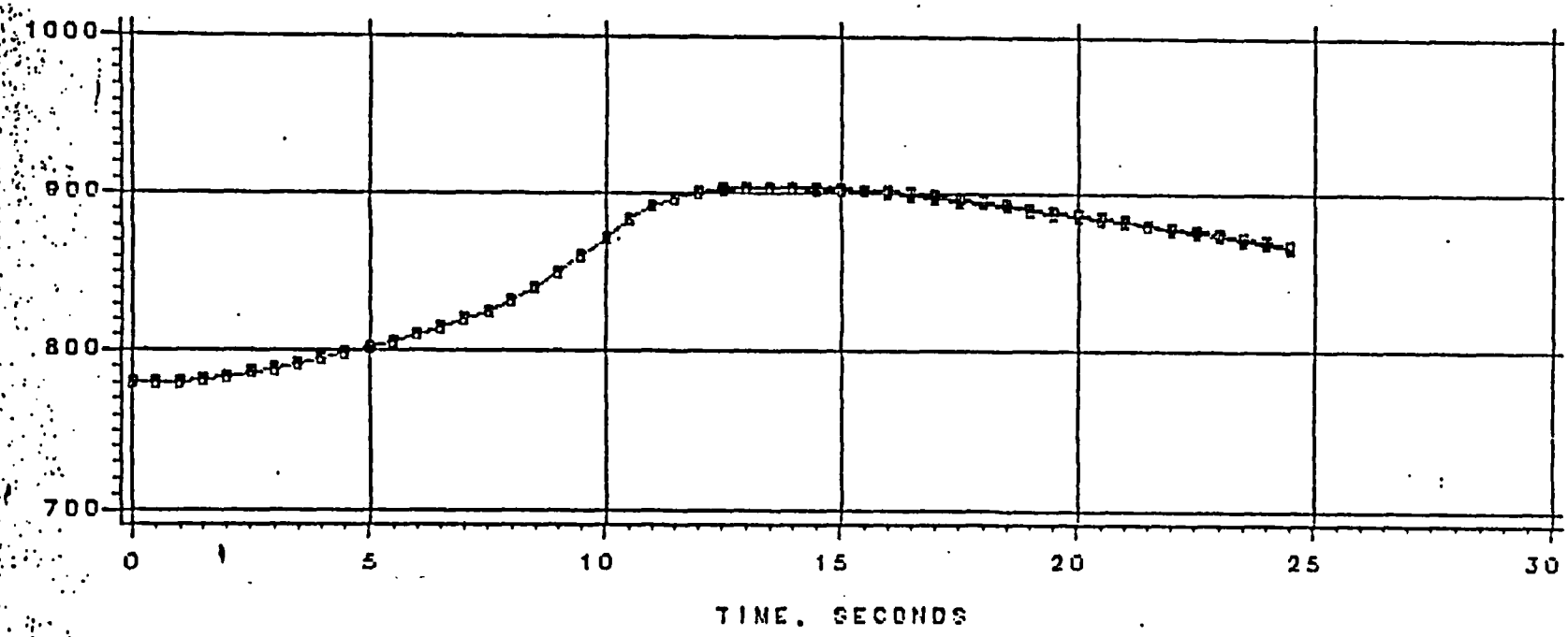
DPC STUDY



SQUARE = MOST NEGATIVE DPC
TRIANGLE= MOST NEGATIVE DPCX0.75

FIGURE VI-19
LOSS OF LOAD

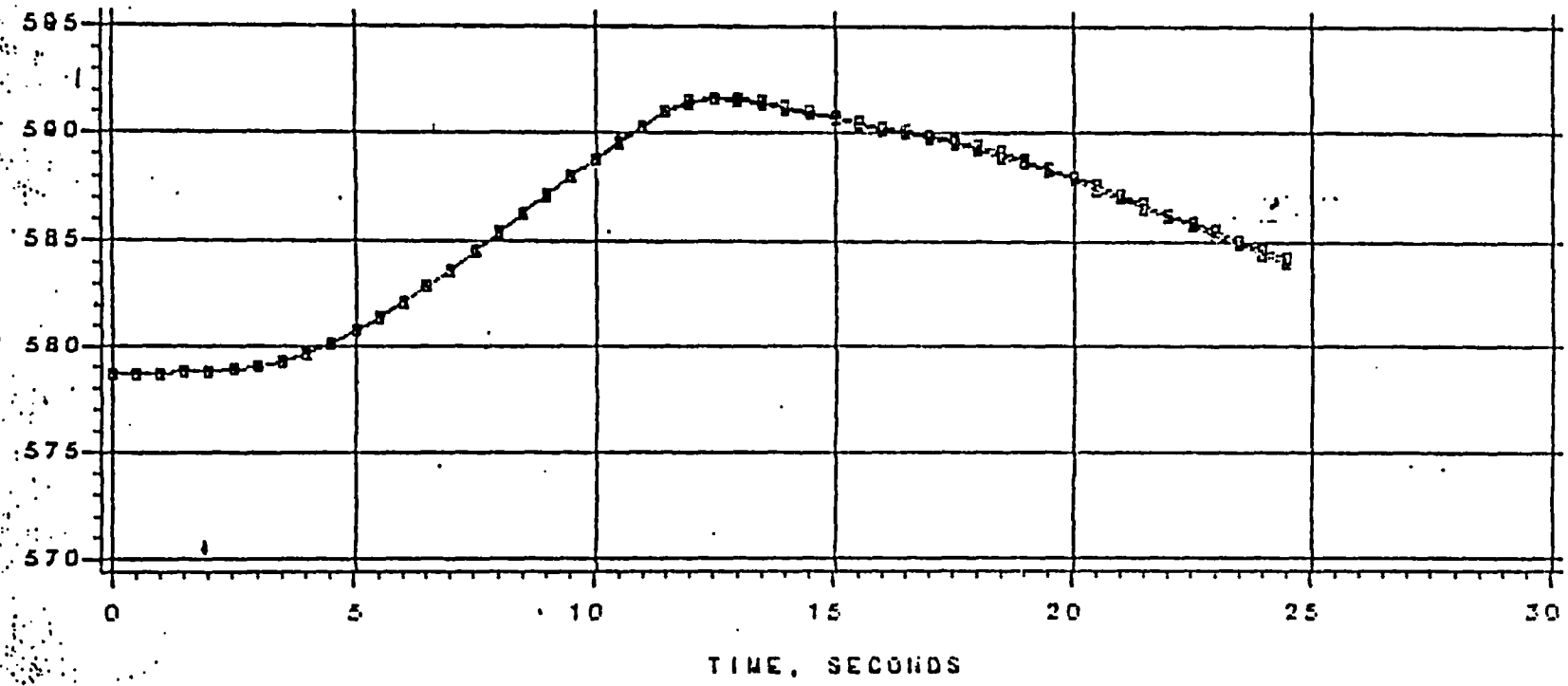
DPC STUDY



SQUARE = MOST NEGATIVE DPC
TRIANGLE= MOST NEGATIVE DPCX0.75

FIGURE VI-20
LOSS OF LOAD

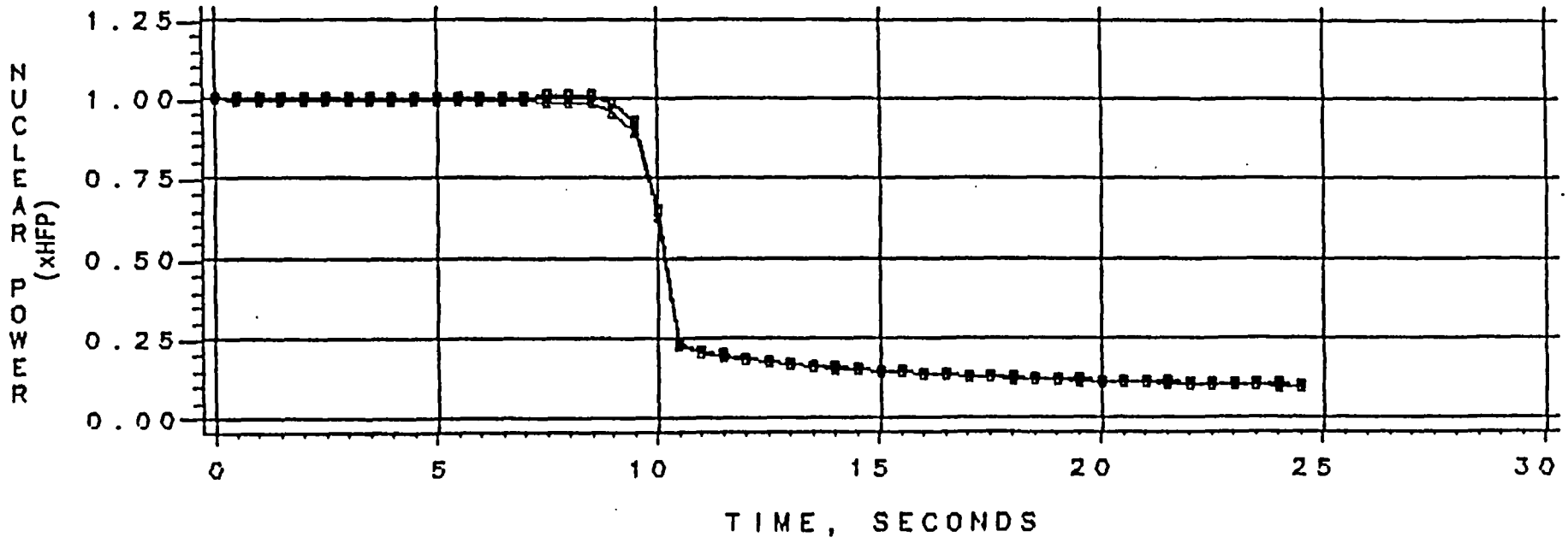
DPC STUDY



SQUARE = MOST NEGATIVE DPC
TRIANGLE= MOST NEGATIVE DPCX0.75

FIGURE VI-21
LOSS OF LOAD

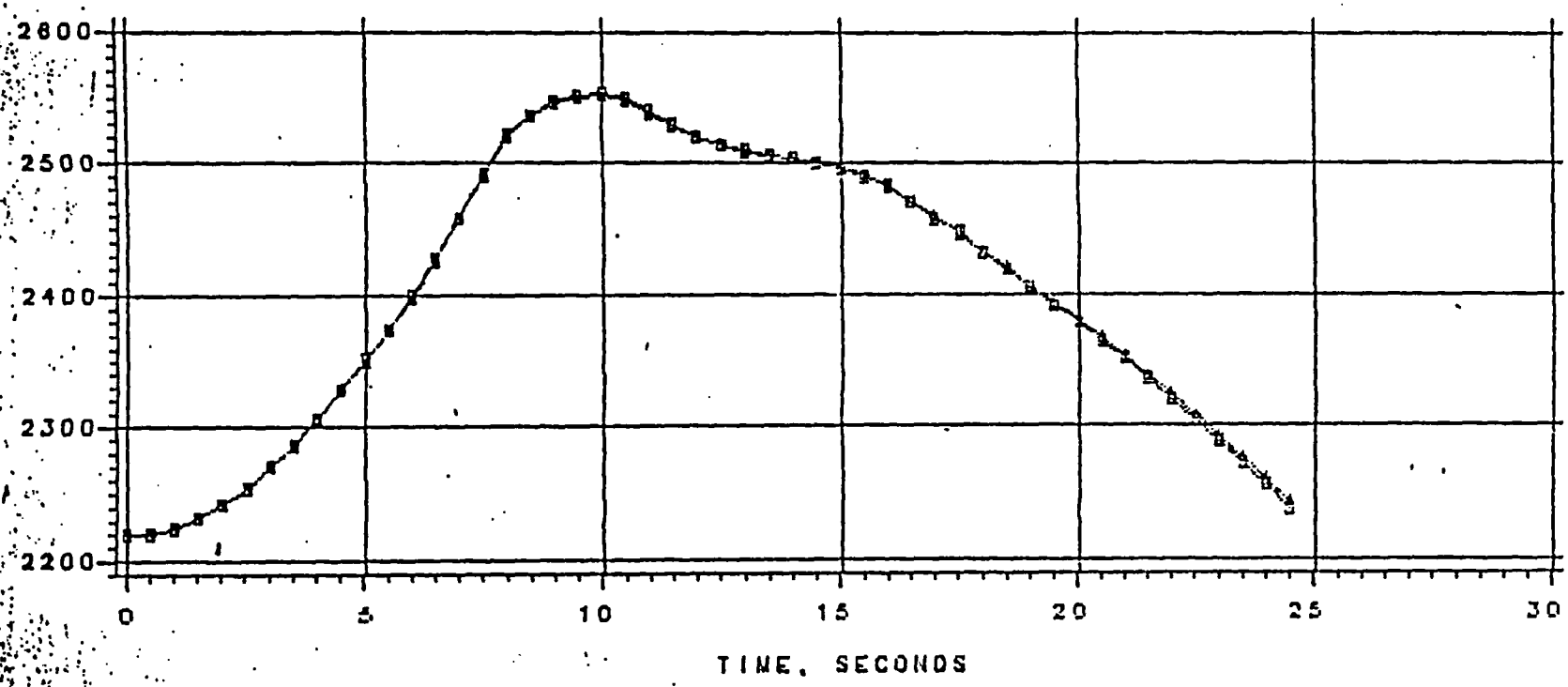
MTC STUDY



SQUARE - MTC-+3PCM/F
TRIANGLE- MTC--3PCM/F

FIGURE VI-22
LOSS OF LOAD

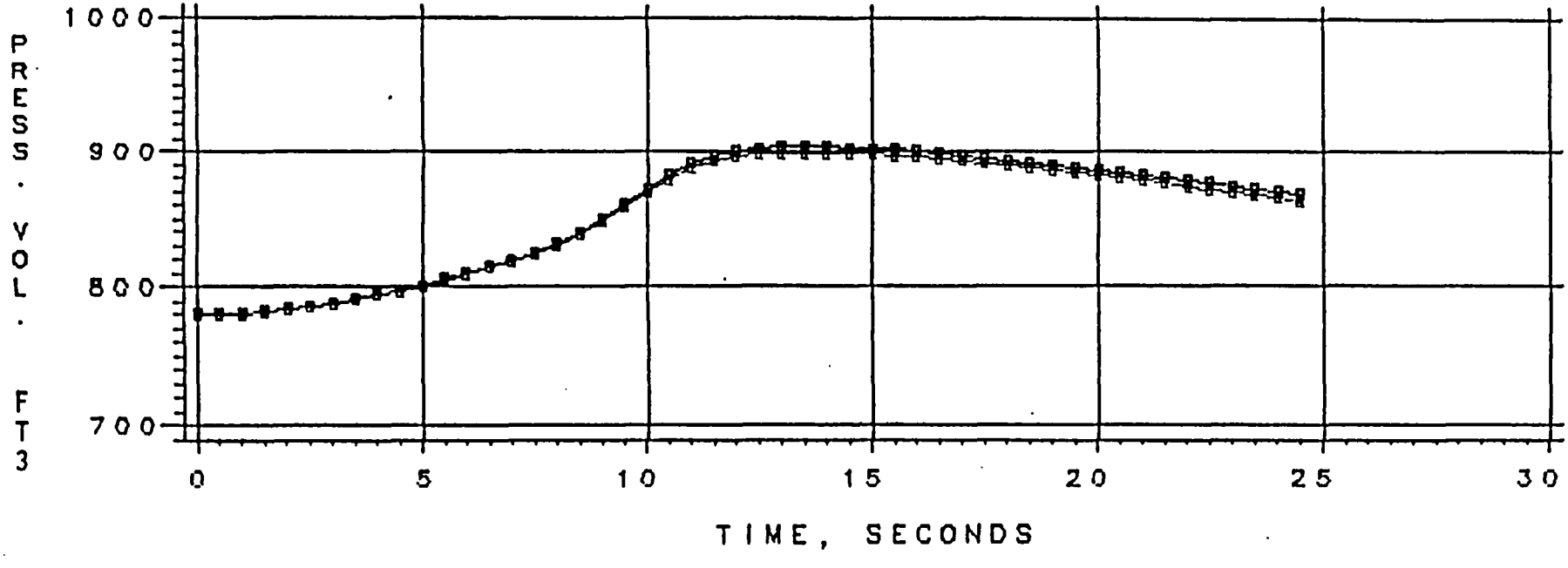
MTC STUDY



SQUARE = MTC=+3PCM/F
TRIANGLE= MTC=-3PCM/F

FIGURE VI-23
LOSS OF LOAD

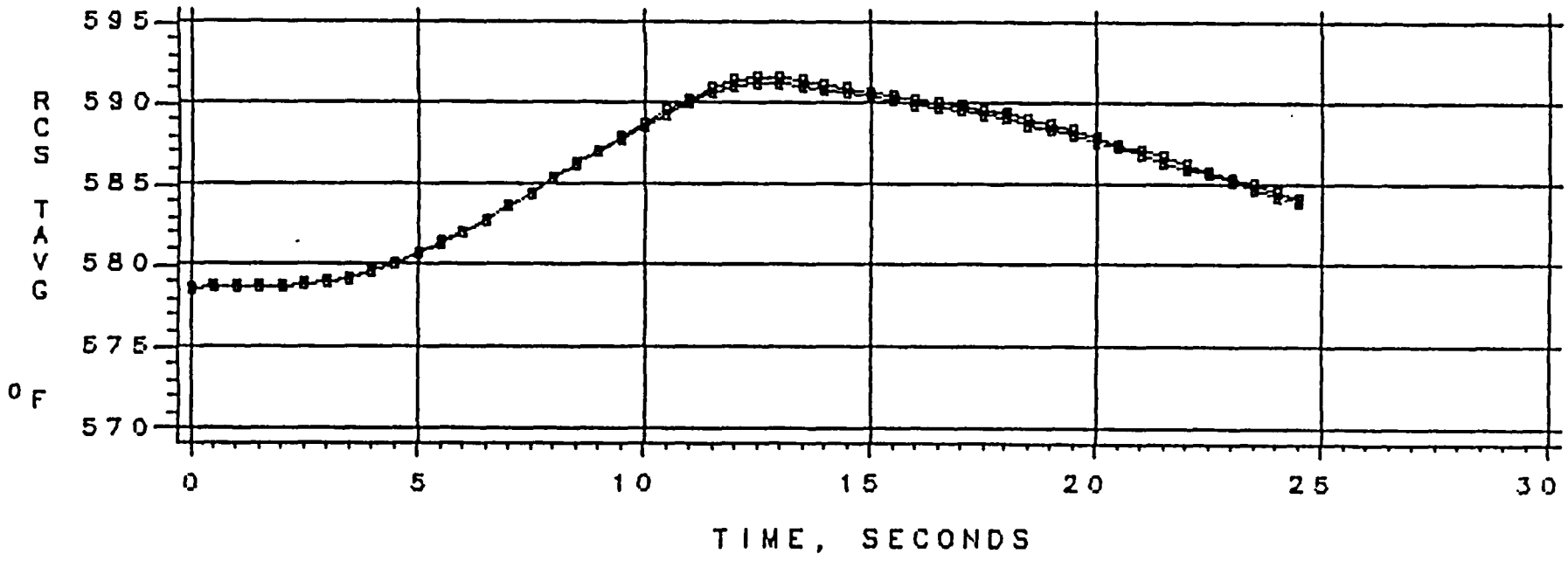
MTC STUDY



SQUARE - MTC-+3PCM/F
TRIANGLE- MTC--3PCM/F

FIGURE VI-24
LOSS OF LOAD

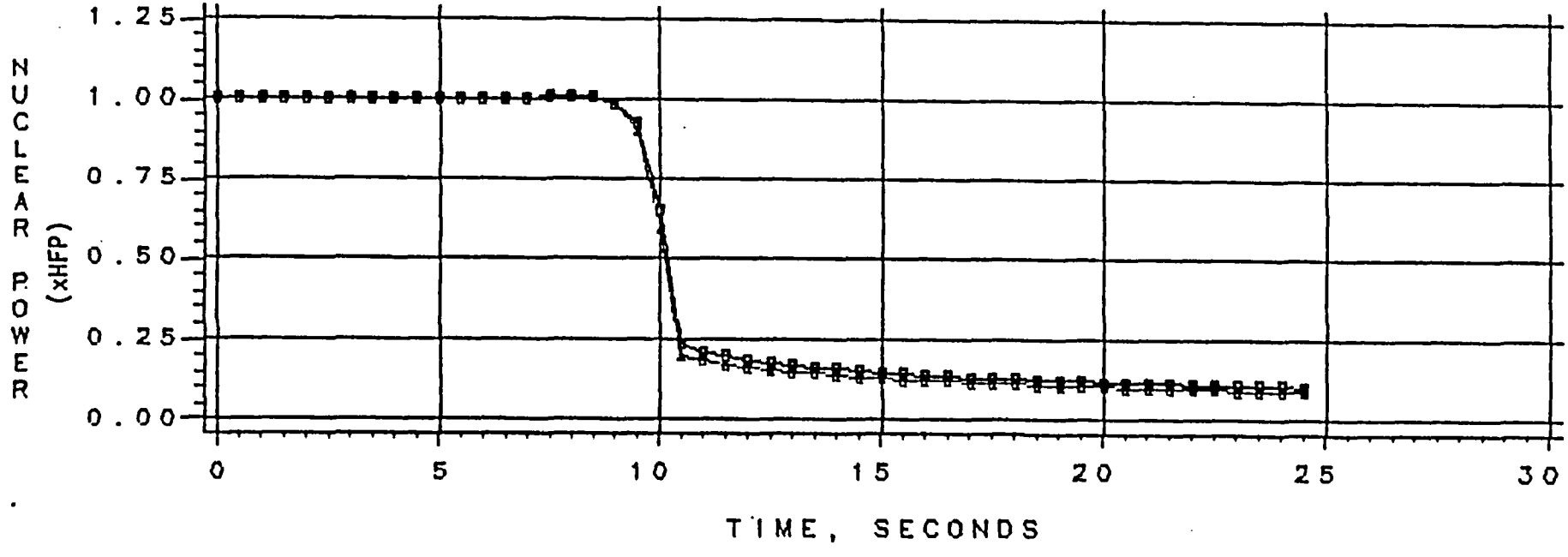
MTC STUDY



SQUARE - MTC-+3PCM/F
TRIANGLE- MTC--3PCM/F

FIGURE VI-25
LOSS OF LOAD

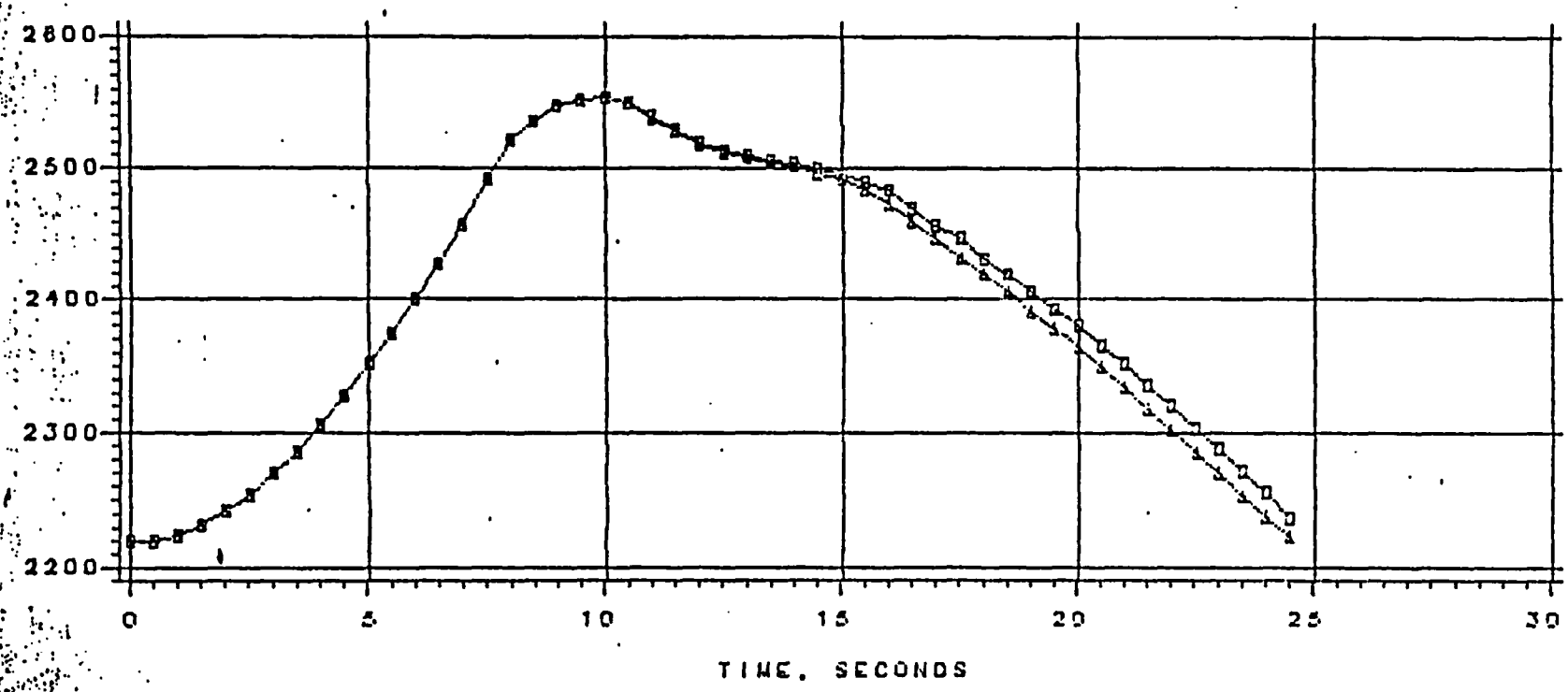
TRIP STUDY



SQUARE - TRIP- 4.0%
TRIANGLE- TRIP- 5.0%

FIGURE VI-26
LOSS OF LOAD

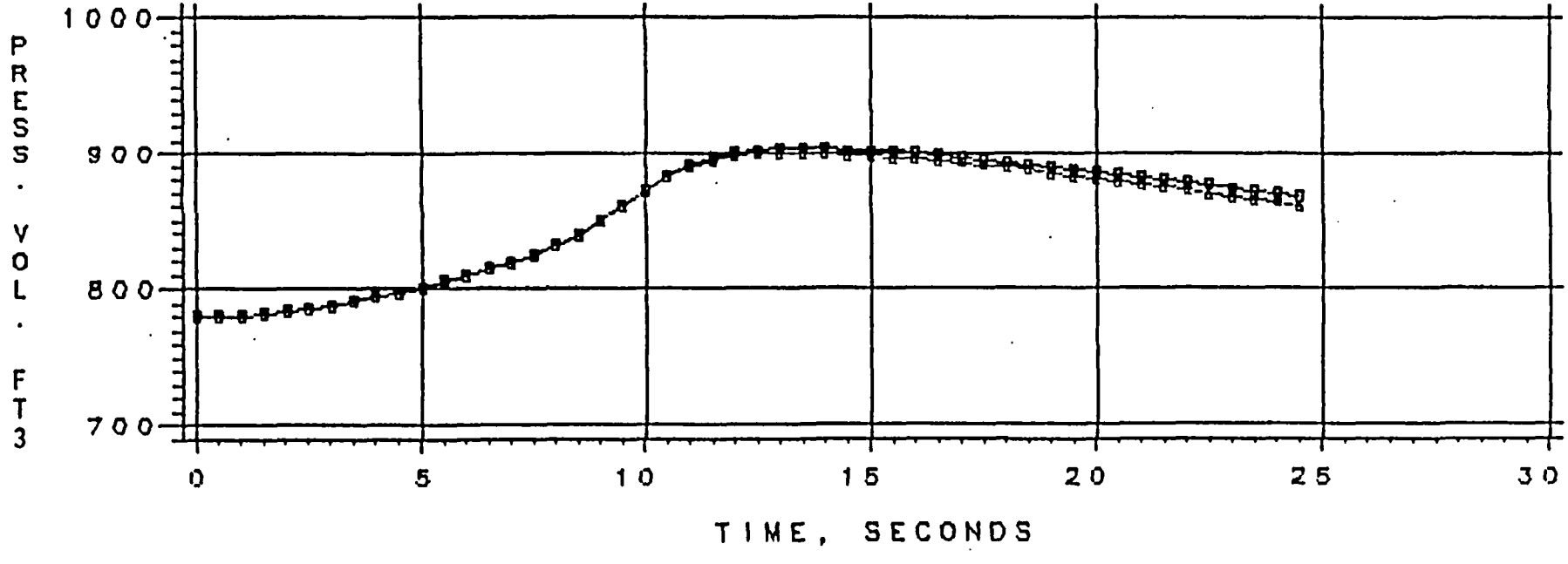
TRIP STUDY



SQUARE = 4.0 % TRIP
TRIANGLE= 5.0% TRIP

FIGURE VI-27
LOSS OF LOAD

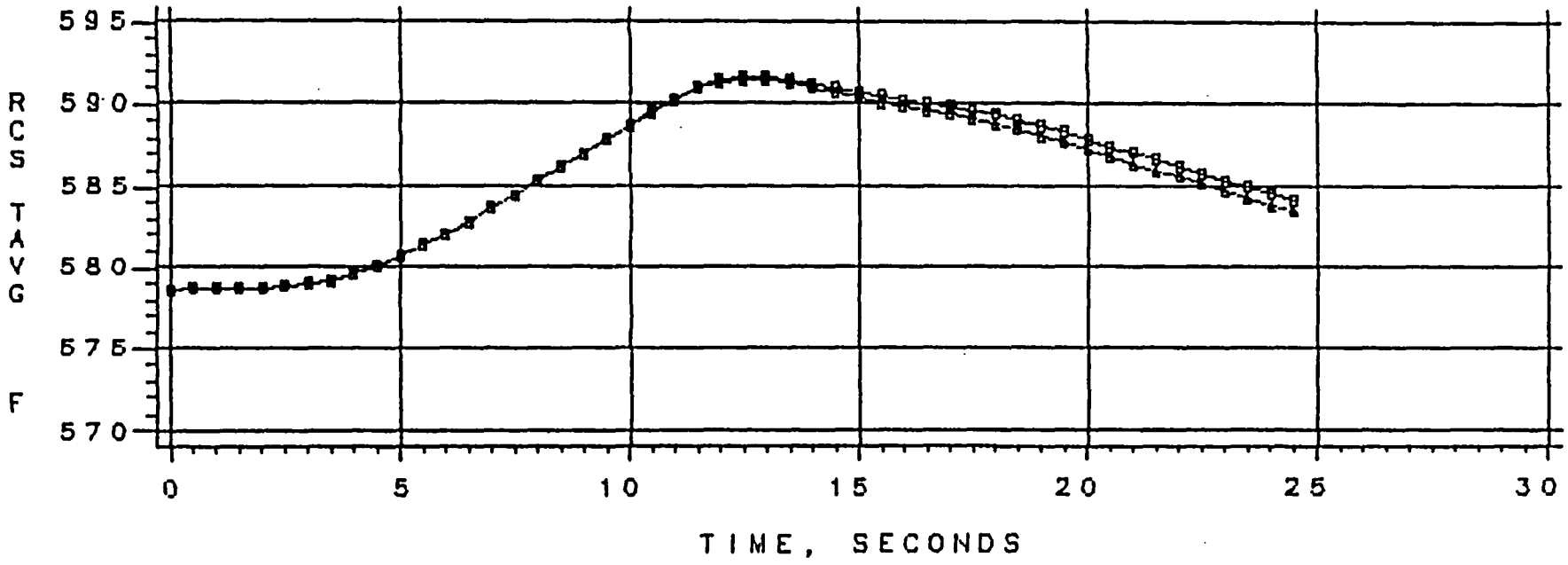
TRIP STUDY



SQUARE - TRIP- 4.0%
TRIANGLE- TRIP- 5.0%

FIGURE VI-28
LOSS OF LOAD

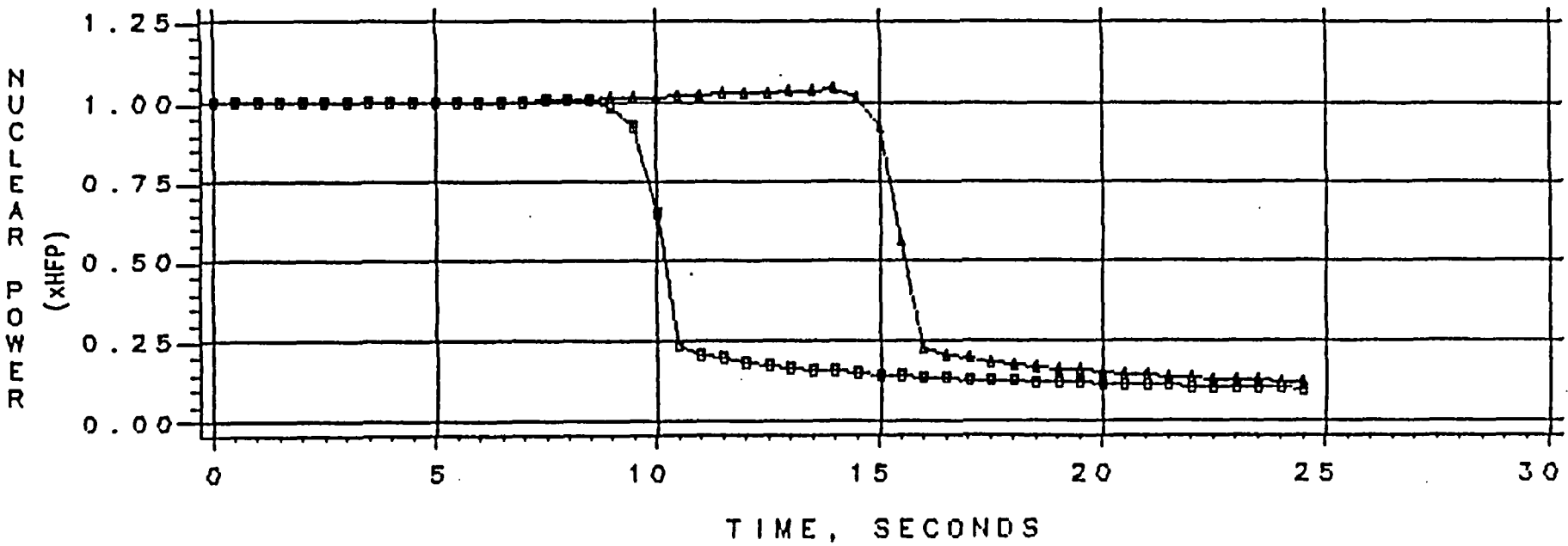
TRIP STUDY



SQUARE - TRIP- 4.0 %
TRIANGLE- TRIP- 5.0 %

FIGURE VI-29 ..
LOSS OF LOAD

PORV/SPRAY STUDY

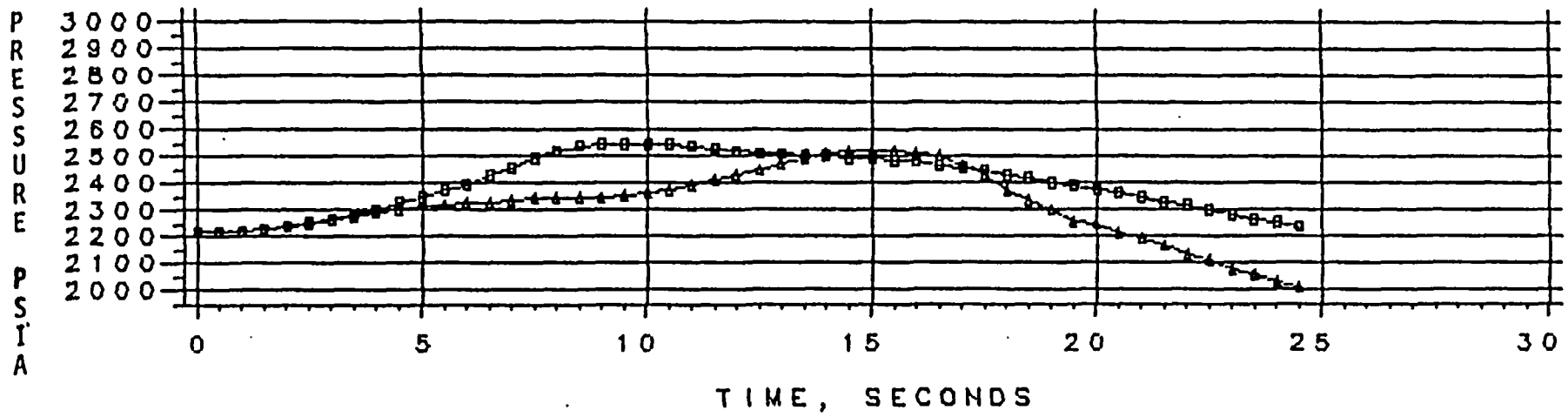


SQUARE - NO PORV/SPRAY
TRIANGLE- PORV/SPRAY

FIGURE VI-30

LOSS OF LOAD

PORV/SPRAY STUDY

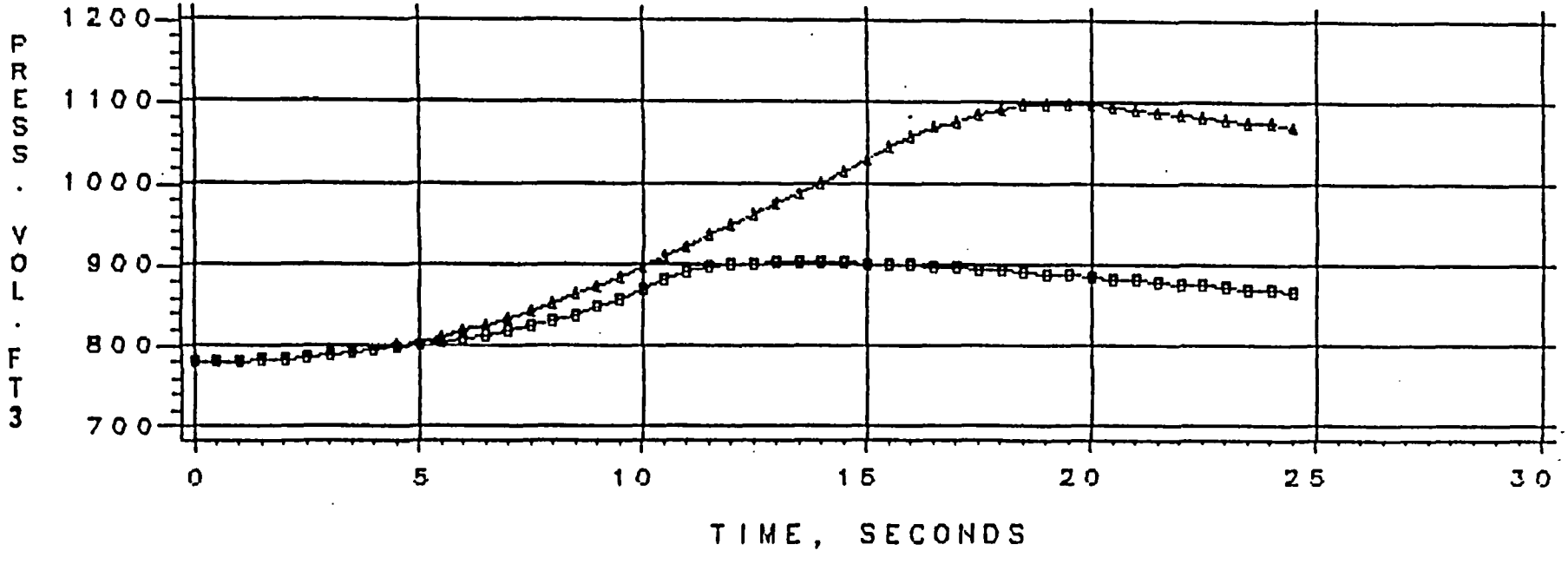


SQUARE - NO PORV/SPRAY
TRIANGLE- PORV/SPRAY

FIGURE VI-31

LOSS OF LOAD

PORV/SPRAY STUDY

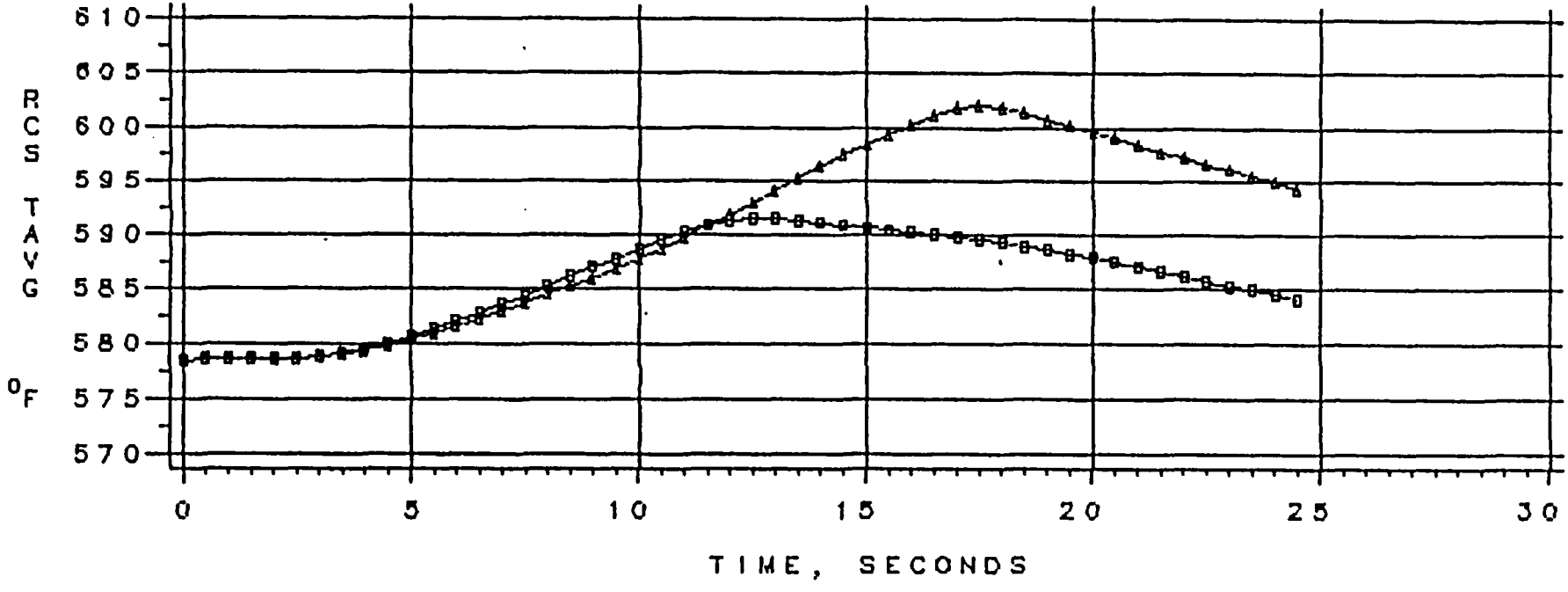


SQUARE - NO PORV/SPRAY
TRIANGLE- PORV/SPRAY

FIGURE VI-32

LOSS OF LOAD

PORV/SPRAY STUDY

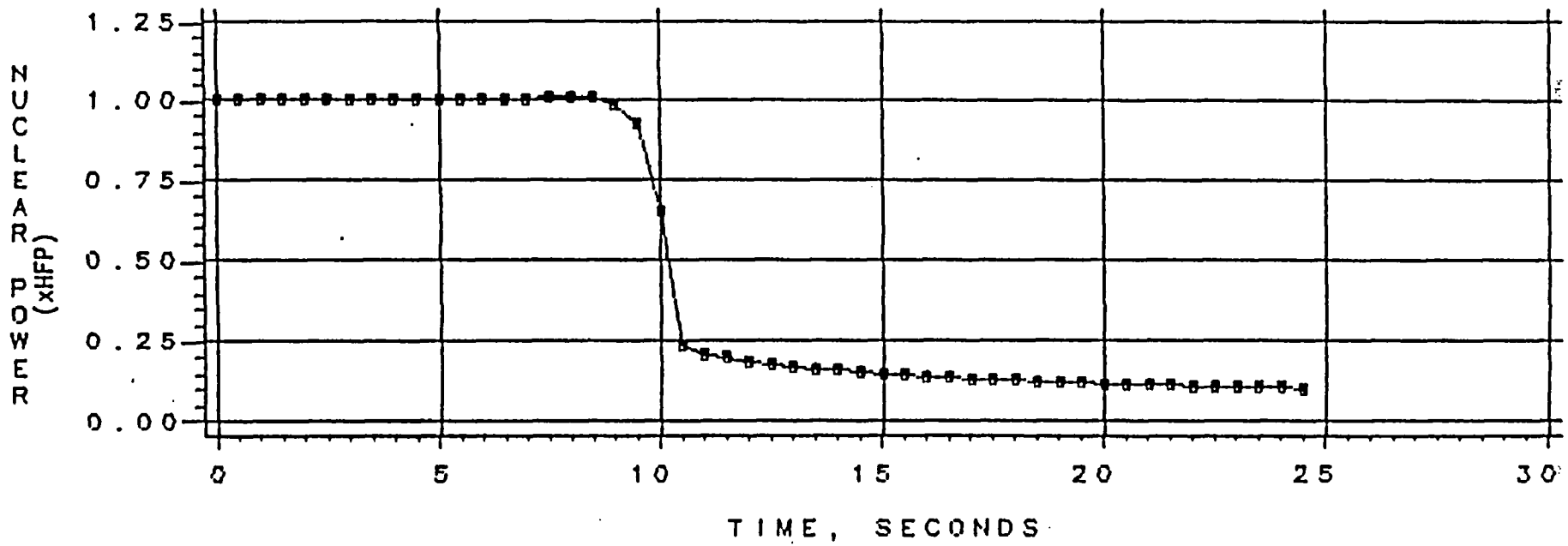


SQUARE - NO PORV/SPRAY
TRIANGLE- PORV/SPRAY

FIGURE VI-33

LOSS OF LOAD

S.G.R.V STUDY

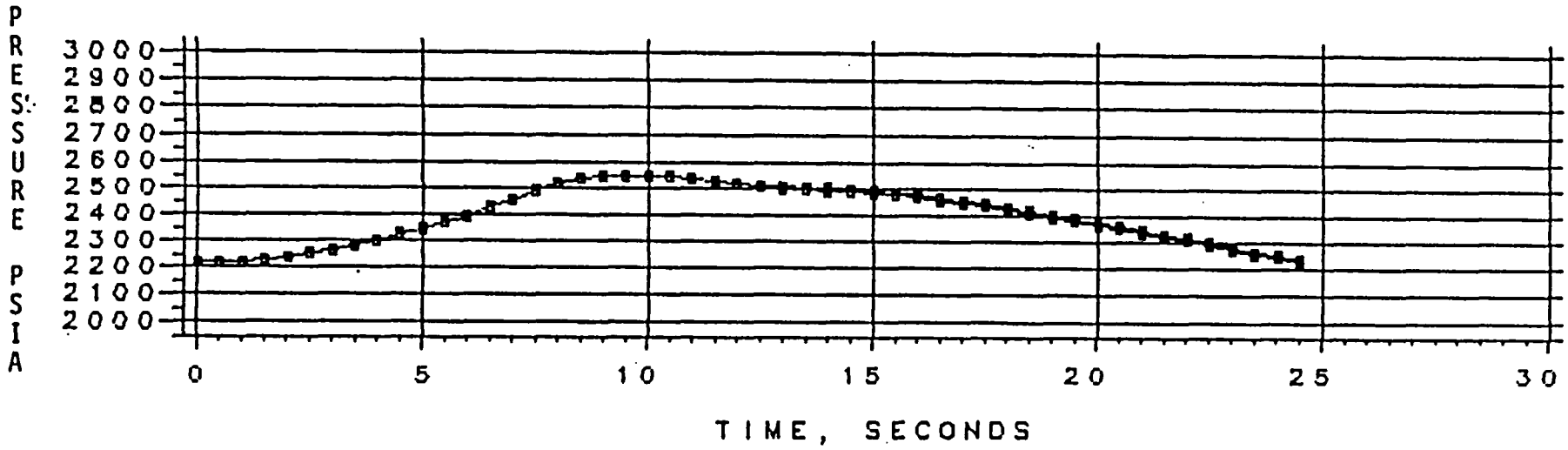


SQUARE – NO S.G.R.V
TRIANGLE– S.G.R.V

FIGURE VI-34

LOSS OF LOAD

S.G.R.V STUDY

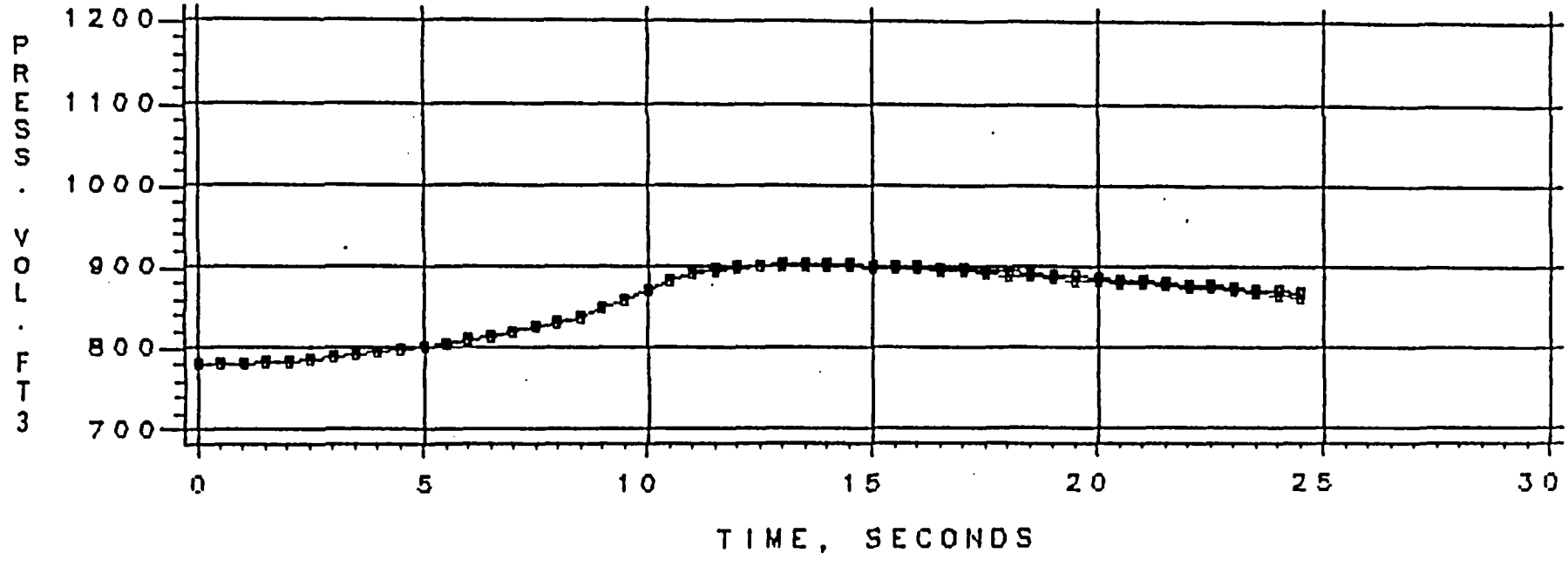


SQUARE – NO S.G.R.V
TRIANGLE– S.G.R.V

FIGURE VI-35

LOSS OF LOAD

S.G.R.V STUDY

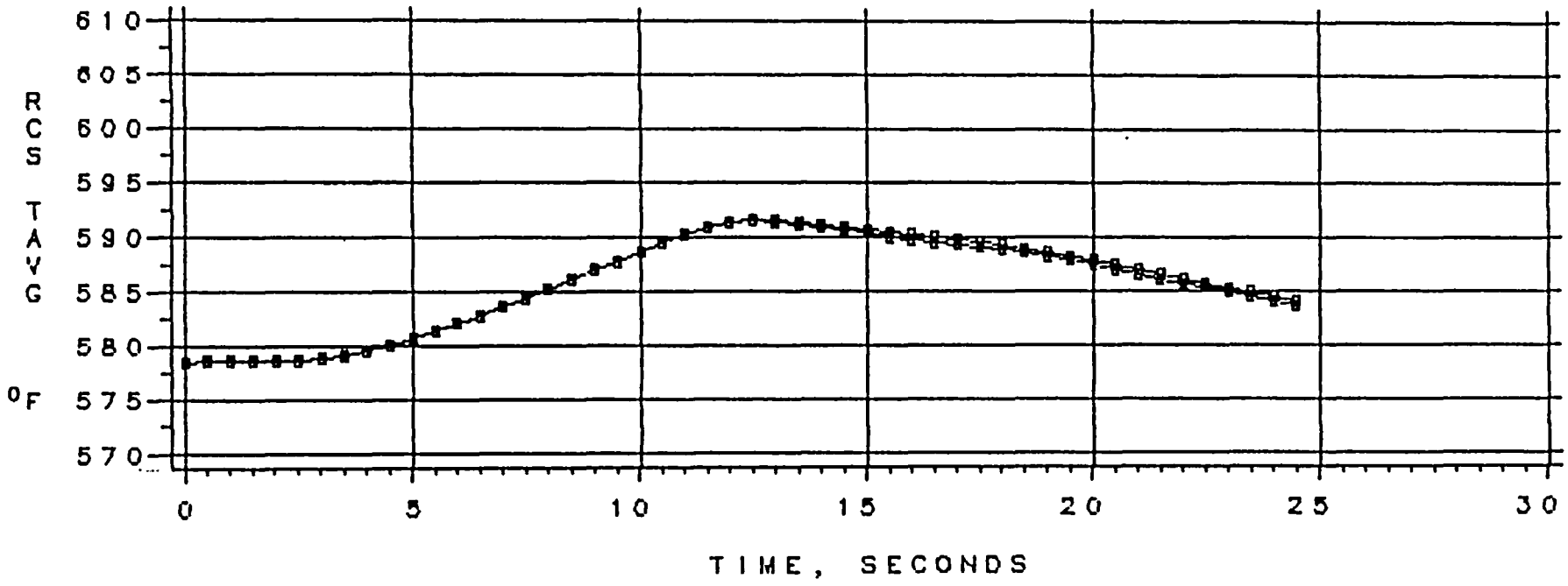


SQUARE – NO S.G.R.V
TRIANGLE– S.G.R.V

FIGURE VI-36

LOSS OF LOAD

S.G.R.V STUDY

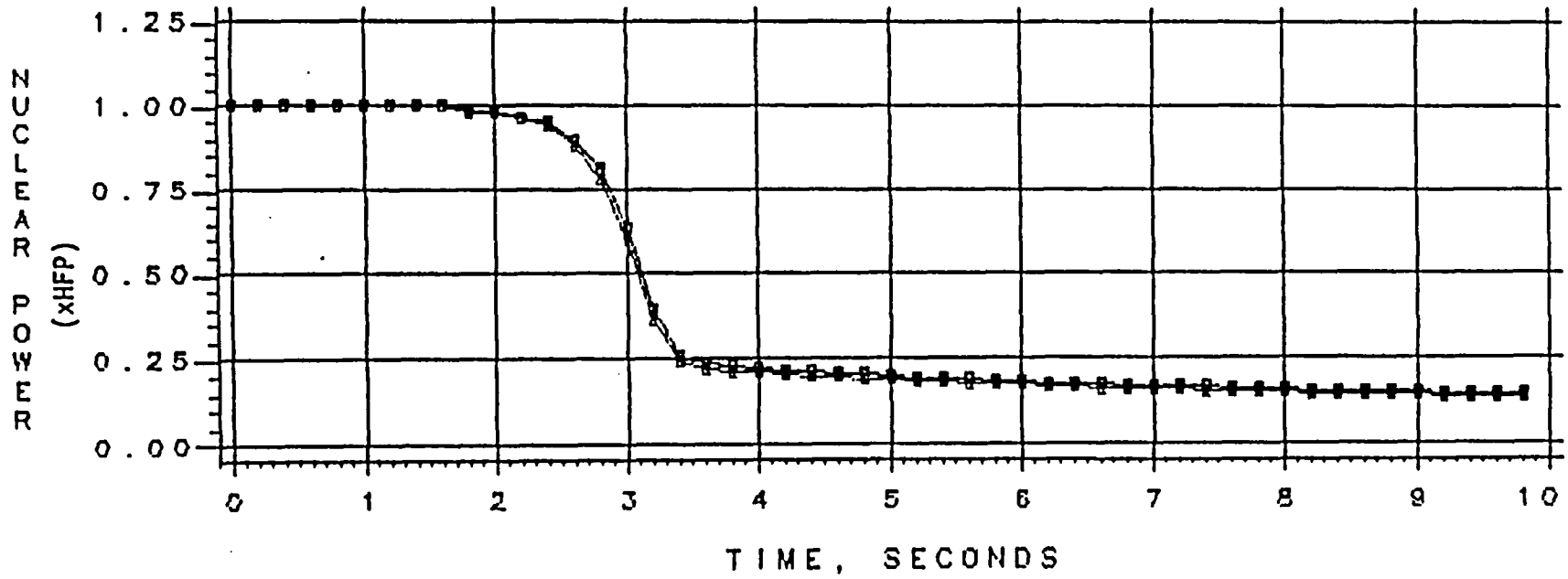


SQUARE – NO S.G.R.V
TRIANGLE– S.G.R.V

FIGURE VI-37

LOSS OF FLOW

DPC STUDY

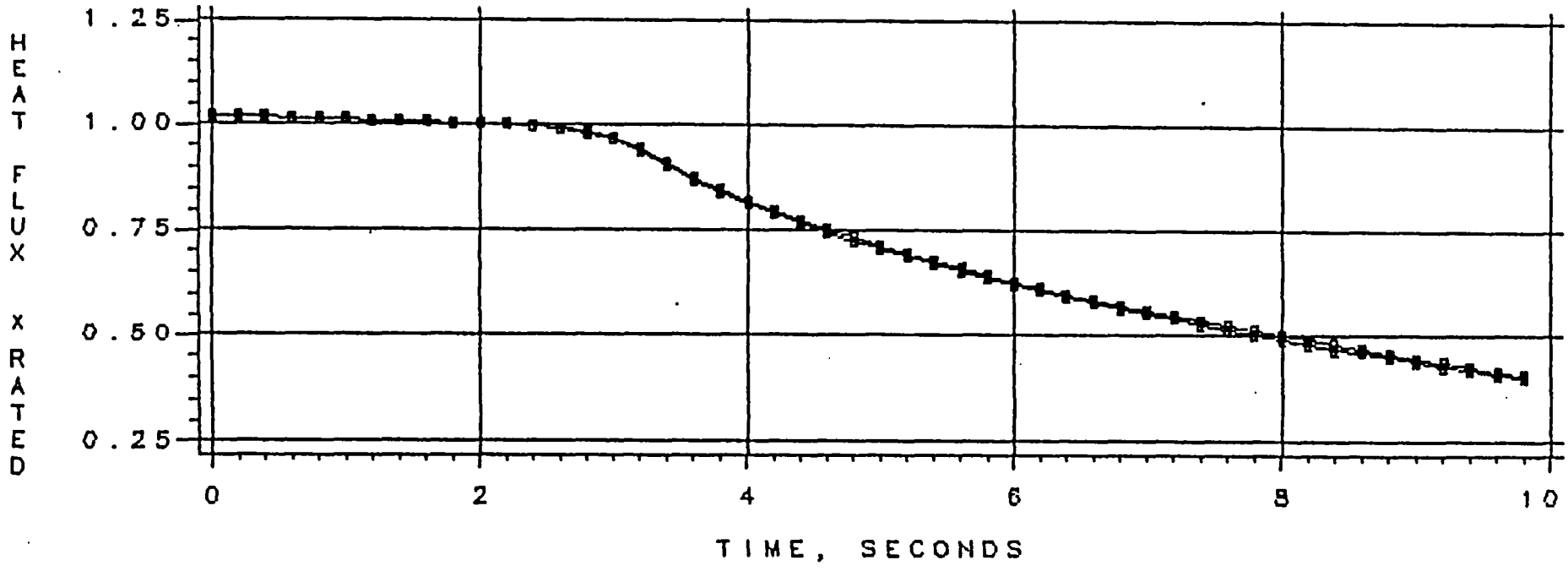


SQUARE - MOST NEG DPC
TRIANGLE- MOST NEG DPC+.75

FIGUR VI-38

LOSS OF FLOW

DPC STUDY

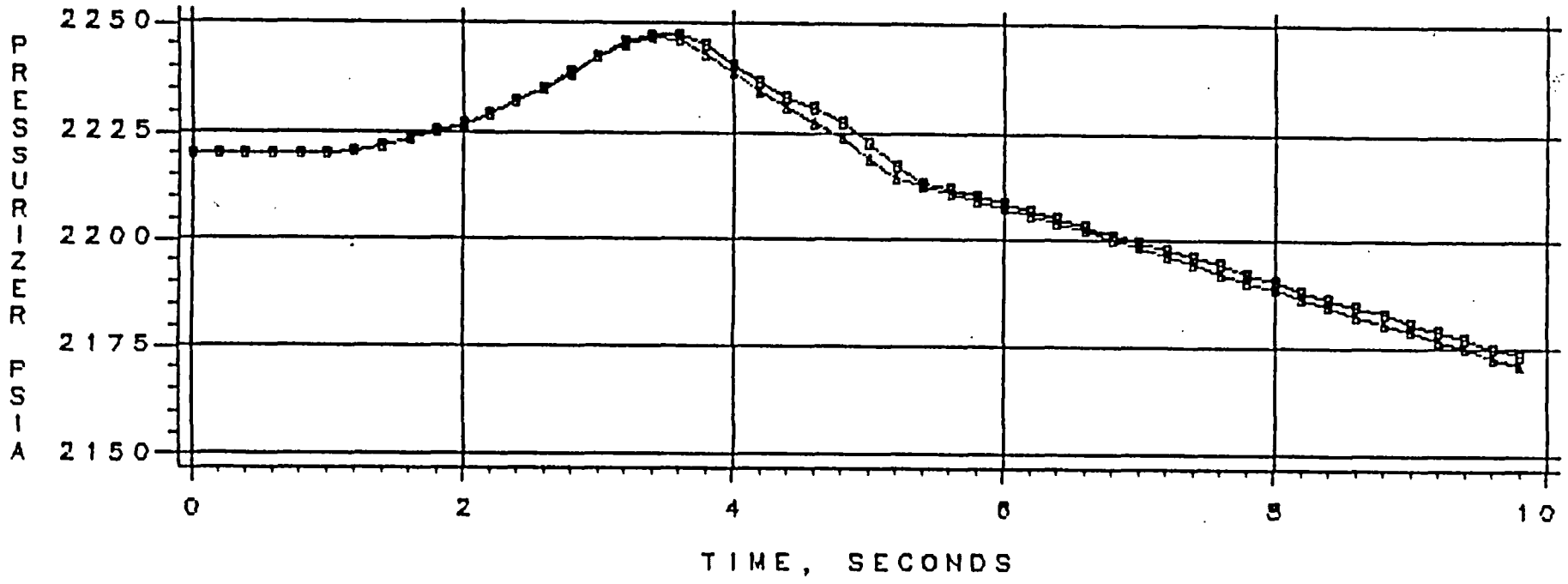


SQUARE - MOST NEG DPC
TRIANGLE- MOST NEG DPC*.75

FIGURE VI-39

LOSS OF FLOW

DPC STUDY

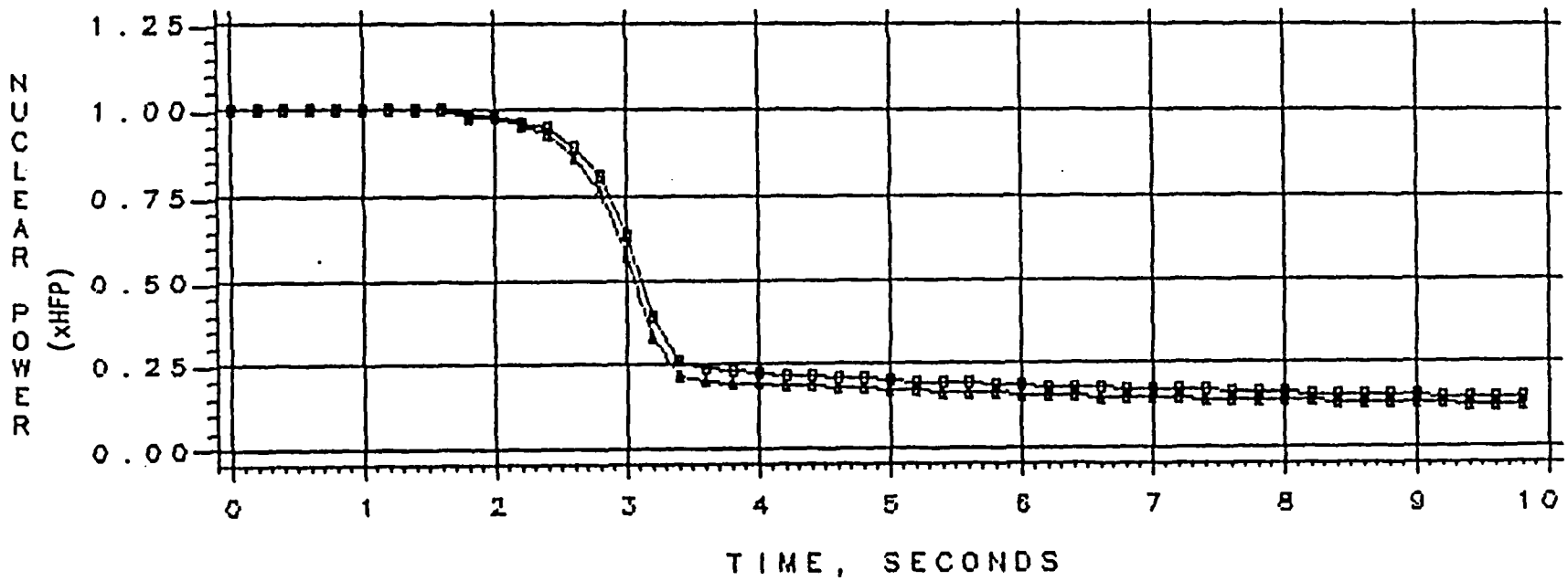


SQUARE - MOST NEG DPC
TRIANGLE- MOST NEG DPC+.75

FIGURE VI-40

LOSS OF FLOW

TRIP STUDY

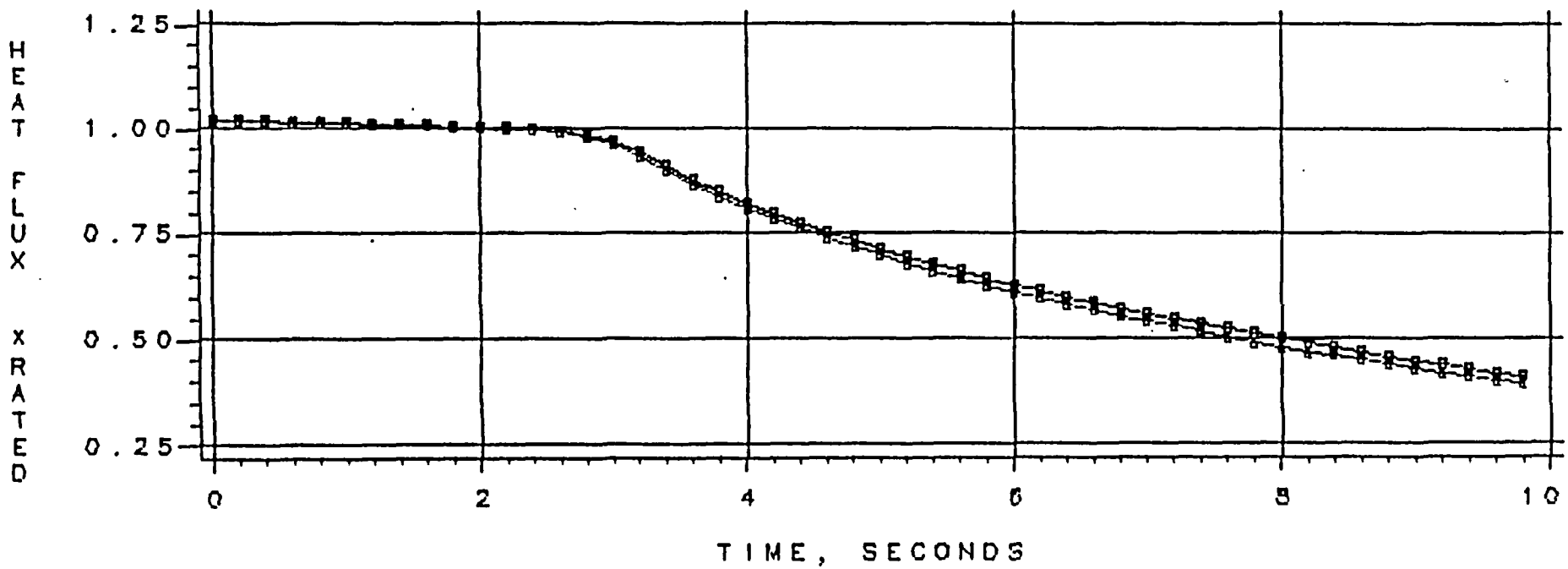


SQUARE - .04 DK/K TRIP
TRIANGLE- .05 DK/K TRIP

FIGURE VI-41

LOSS OF FLOW

TRIP STUDY

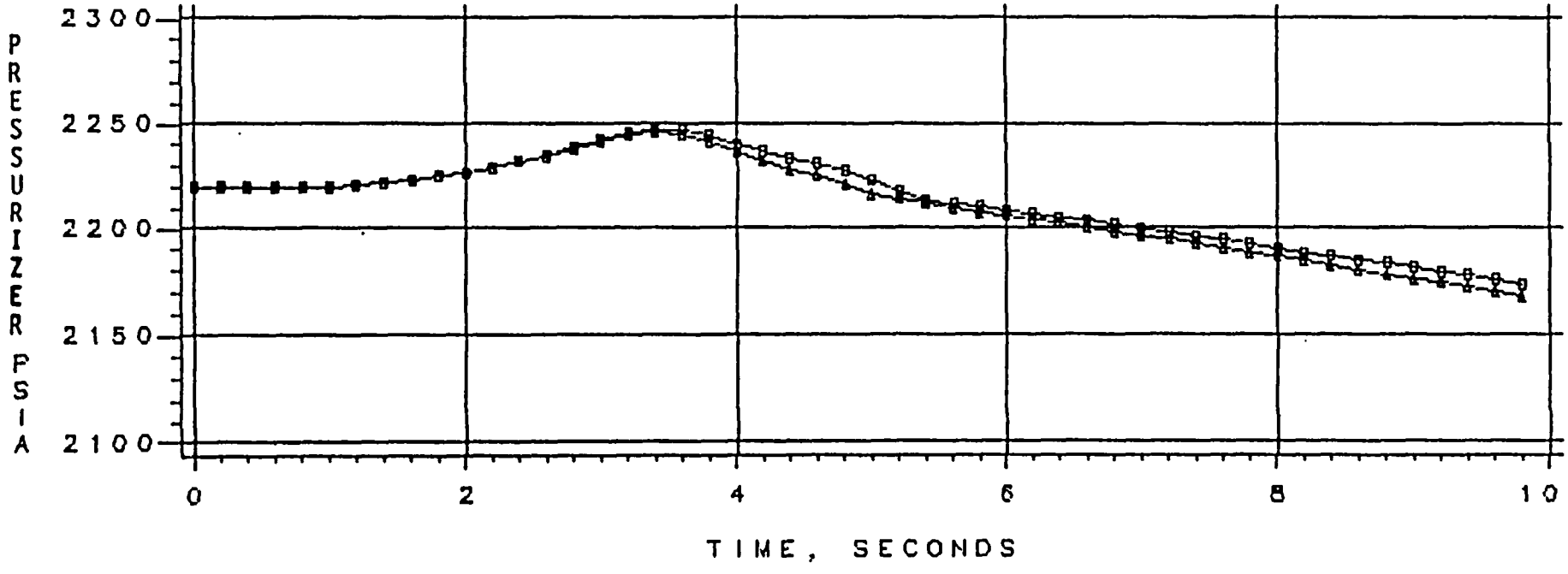


SQUARE - .04 DK/K TRIP
TRIANGLE- .05 DK/K TRIP

FIGURE VI-42

LOSS OF FLOW

TRIP STUDY

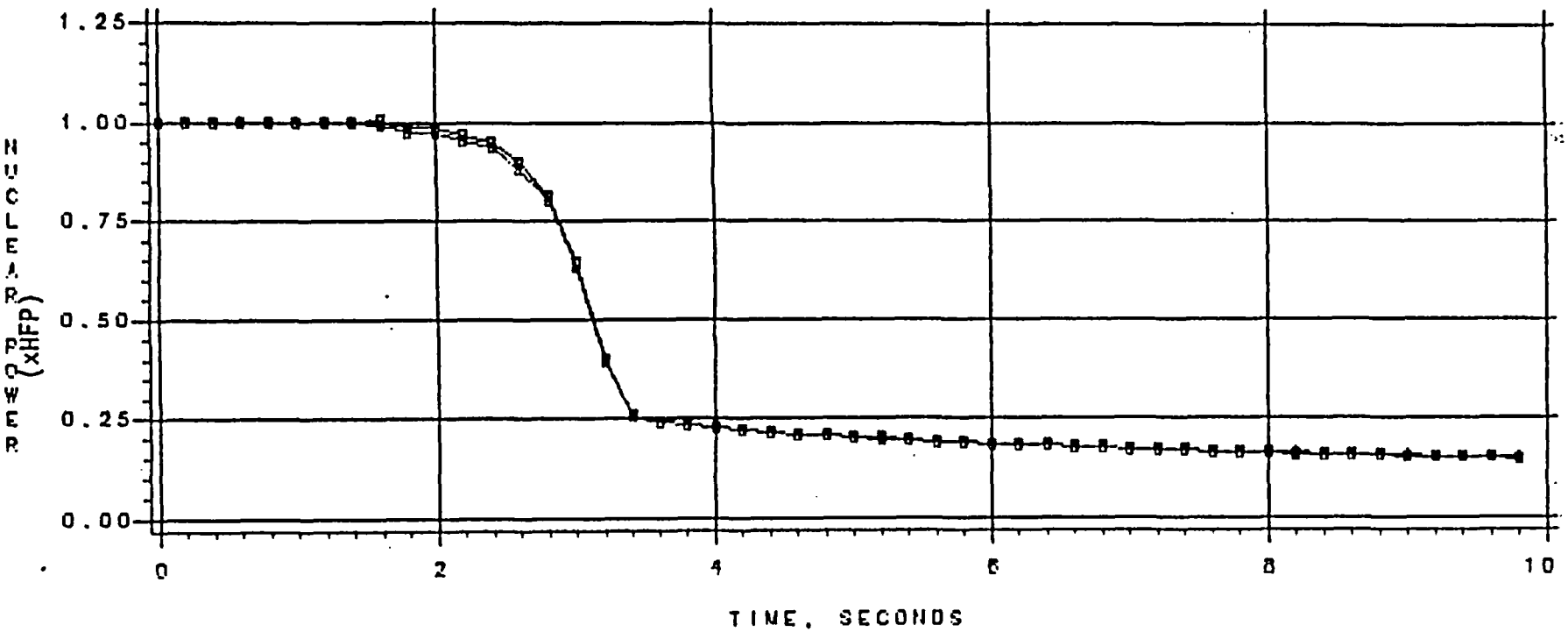


SQUARE - .04 DK/K TRIP
TRIANGLE- .05 DK/K TRIP

FIGURE VI-43

LOSS OF FLOW

MTC STUDY

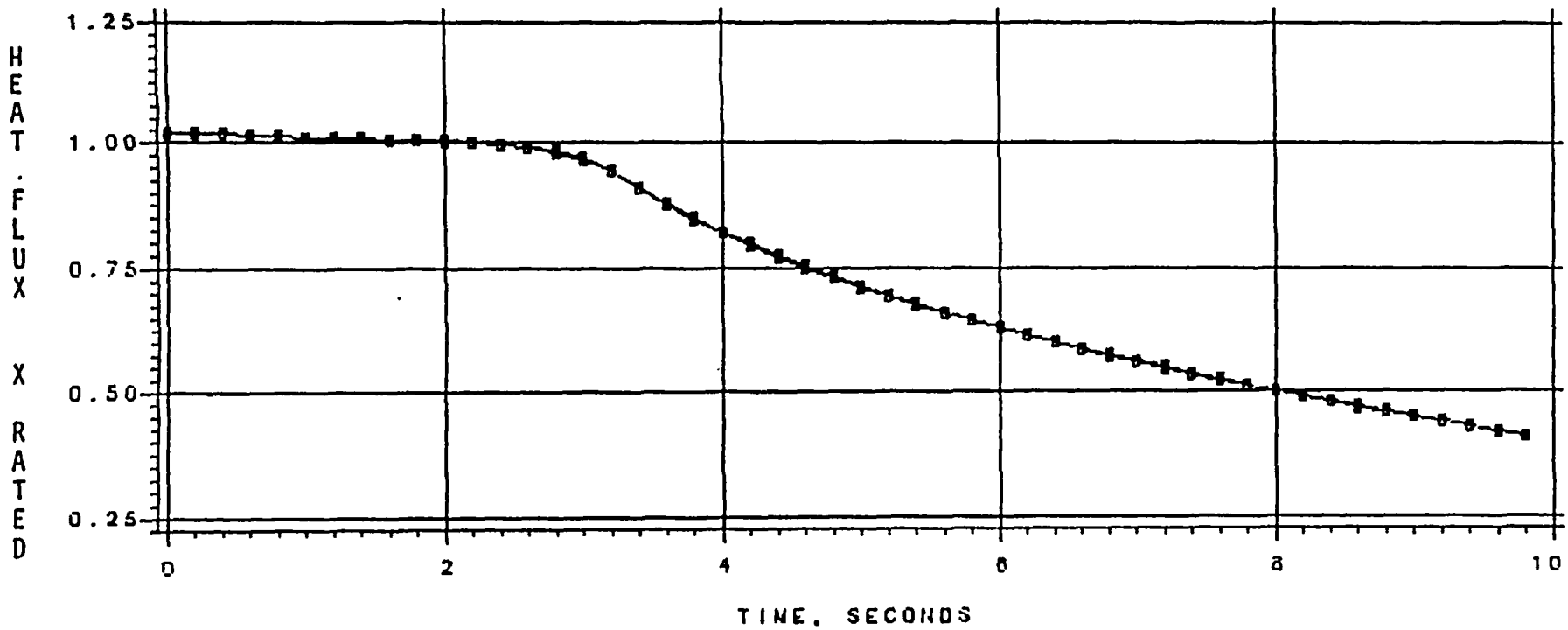


SQUARE = MTC=+3.0 PCM/F
TRIANGLE= MTC=-3.0 PCM/F

FIGURE VI-44

LOSS OF FLOW

MTC STUDY

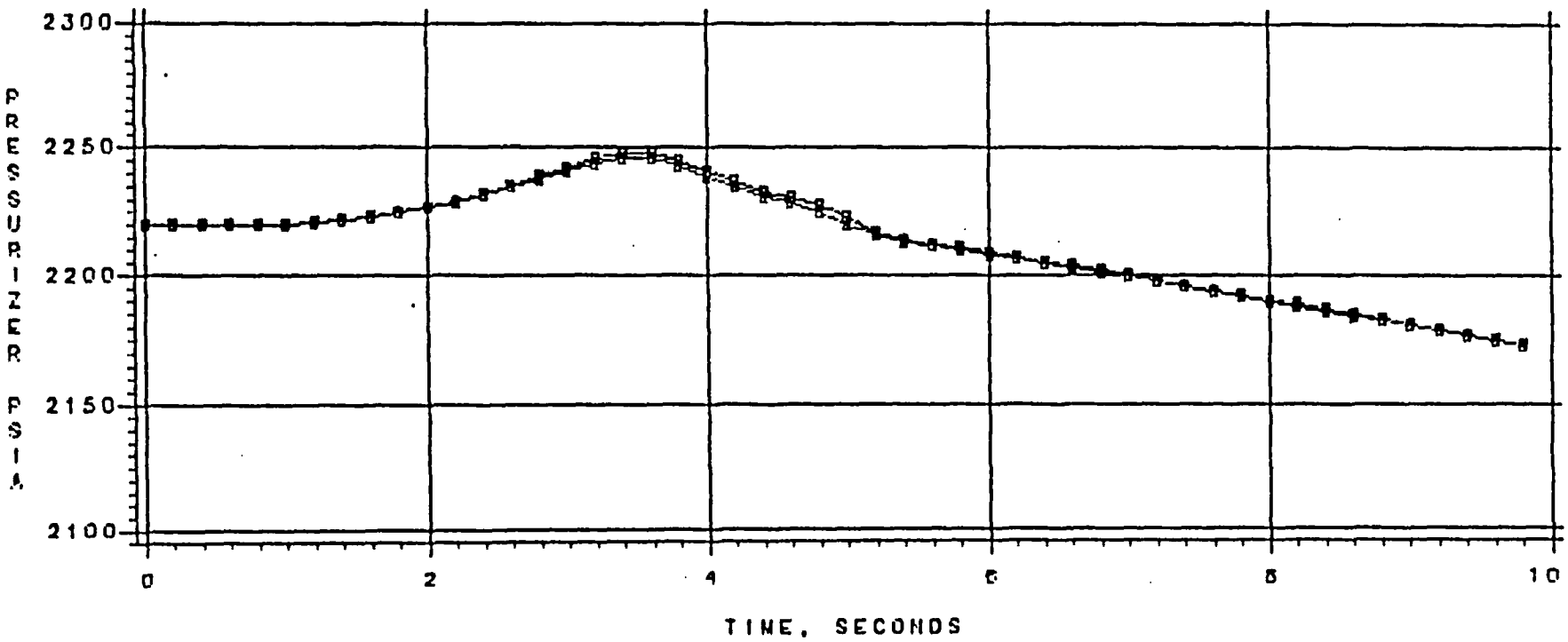


SQUARE = MTC=+3.0 PCM/F
TRIANGLE= MTC=-3.0 PCM/F

FIGURE VI-45

LOSS OF FLOW

MTC STUDY

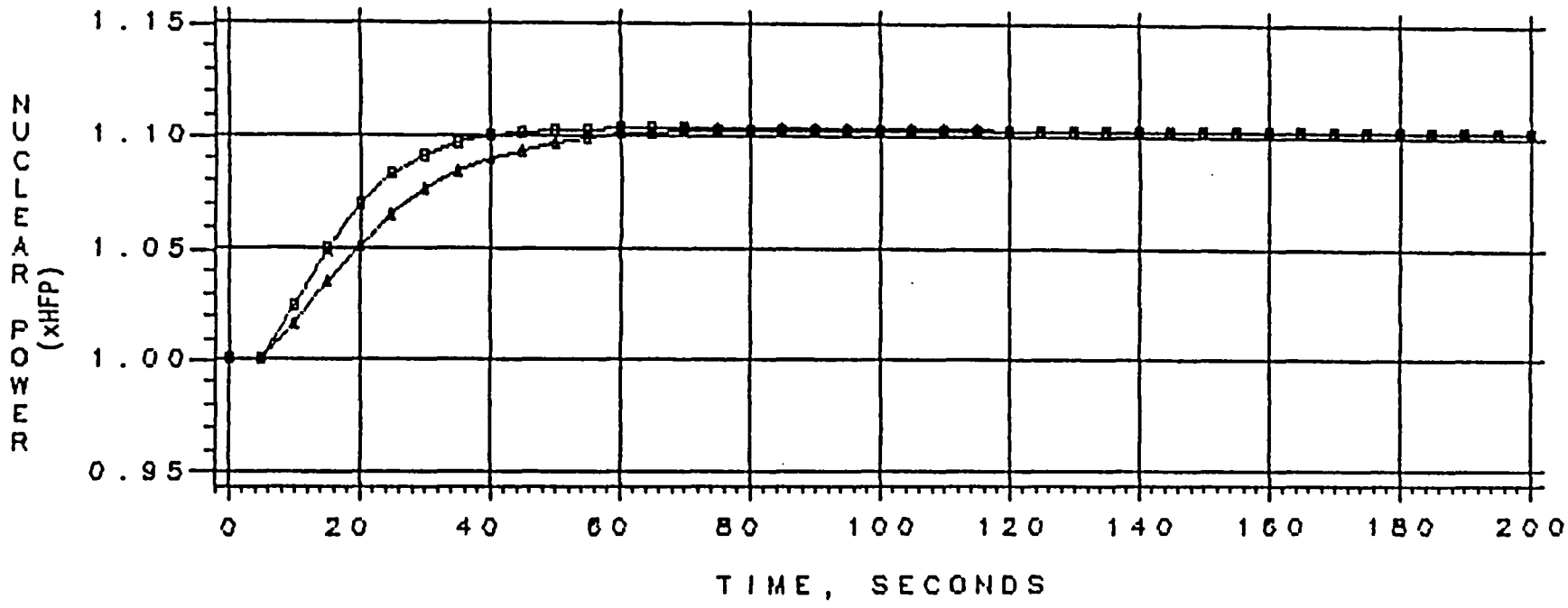


SQUARE = MTC=+3.0 PCM/F
TRIANGLE= MTC=-3.0 PCM/F

FIGURE VI-46

EXCESS LOAD INCREASE

MTC STUDY

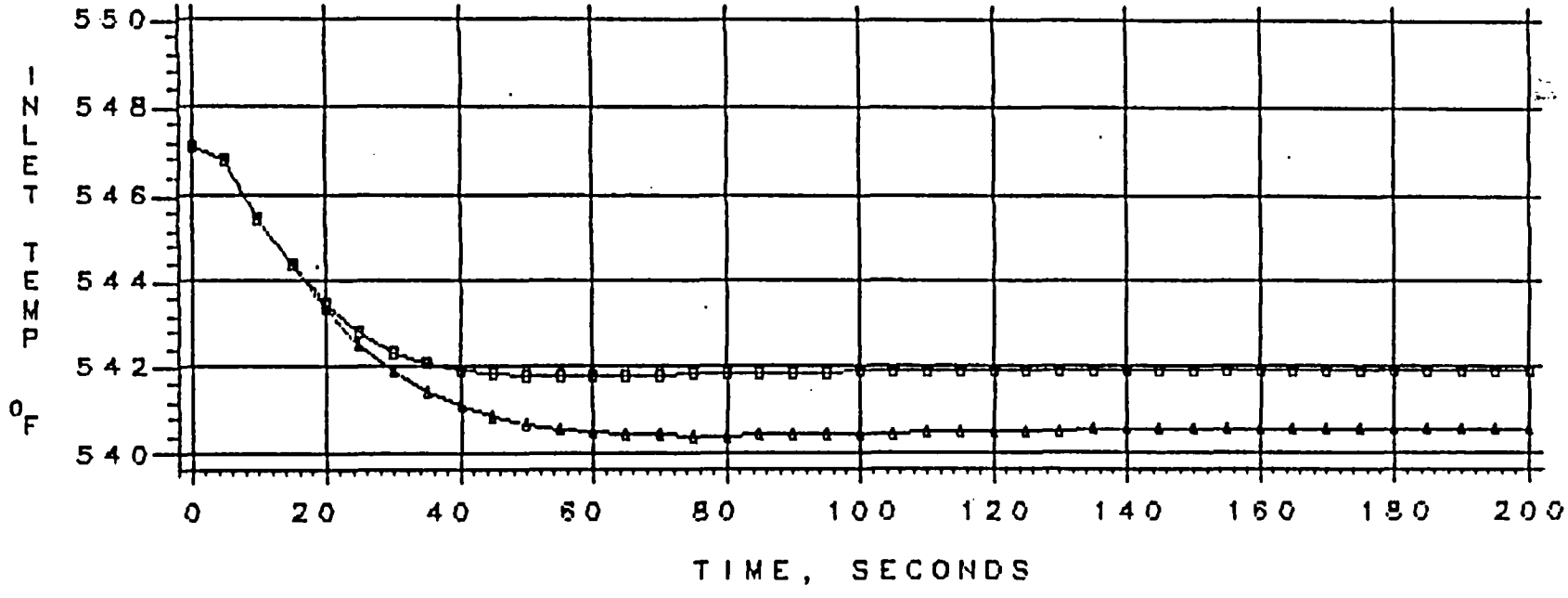


SQUARE - MOST NEG MTC
TRIANGLE- MTC--20PCM/F

FIGURE VI-47

EXCESS LOAD INCREASE

MTC STUDY

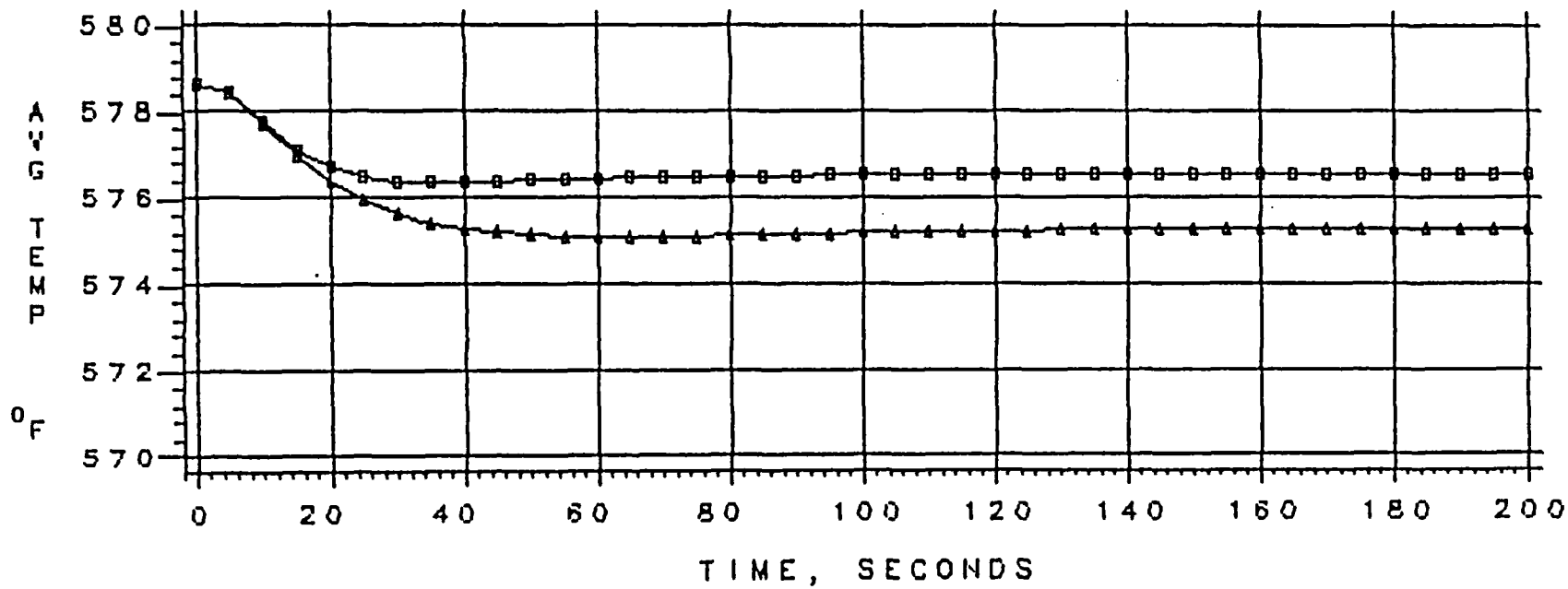


SQUARE - MOST NEG MTC
TRIANGLE- MTC--20PCM/F

FIGURE VI-48

EXCESS LOAD INCREASE

MTC STUDY

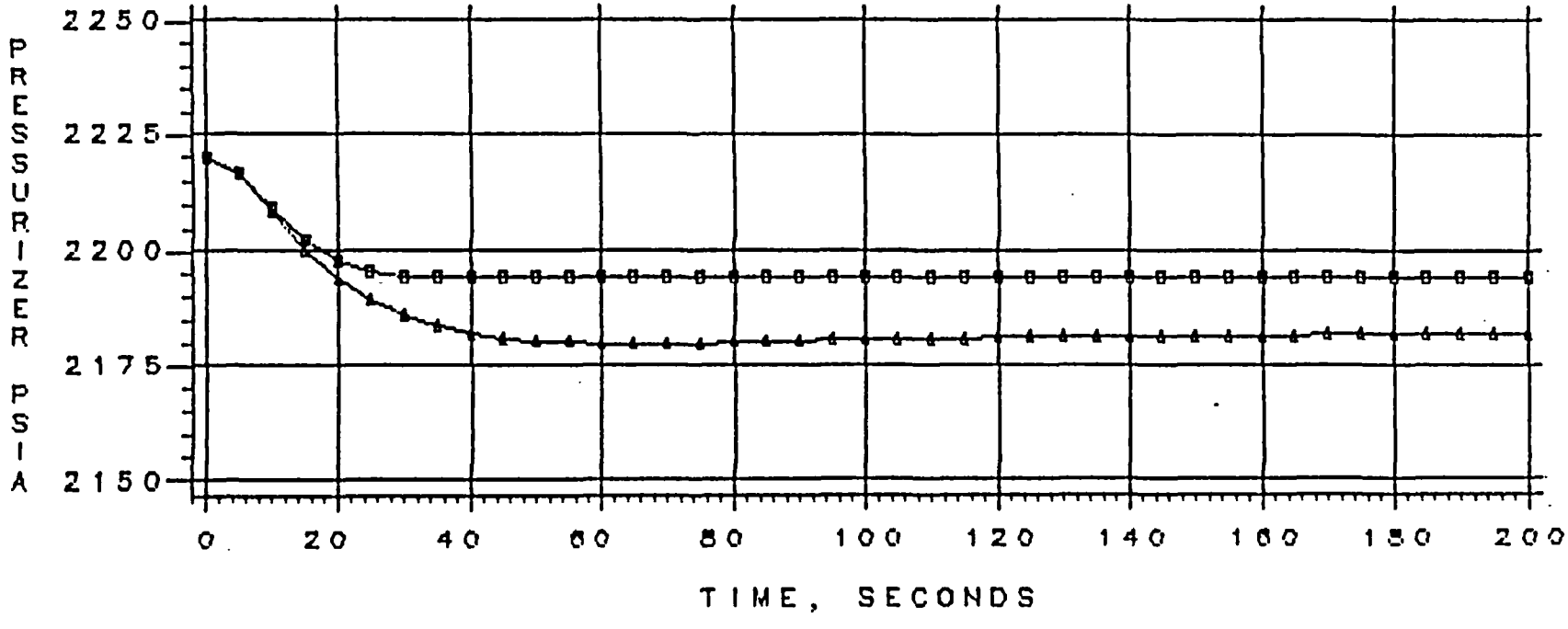


SQUARE - MOST NEG MTC
TRIANGLE- MTC--20PCM/F

FIGURE VI-49

EXCESS LOAD INCREASE

MTC STUDY

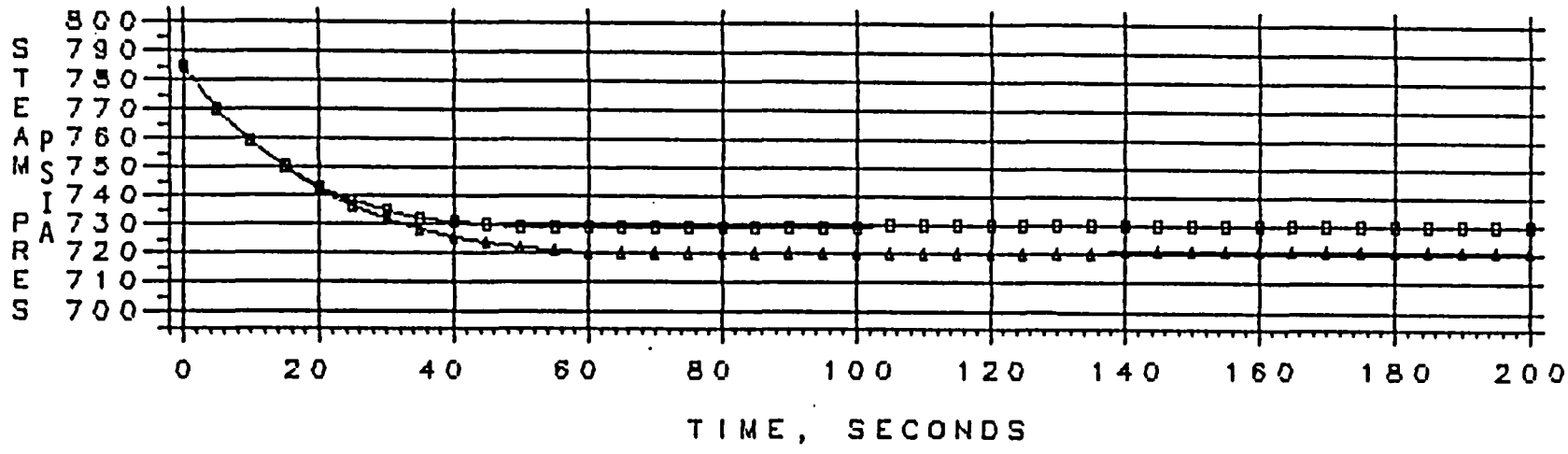


SQUARE - MOST NEG MTC
TRIANGLE- MTC--20PCM/F

FIGURE VI-50

EXCESS LOAD INCREASE

MTC STUDY

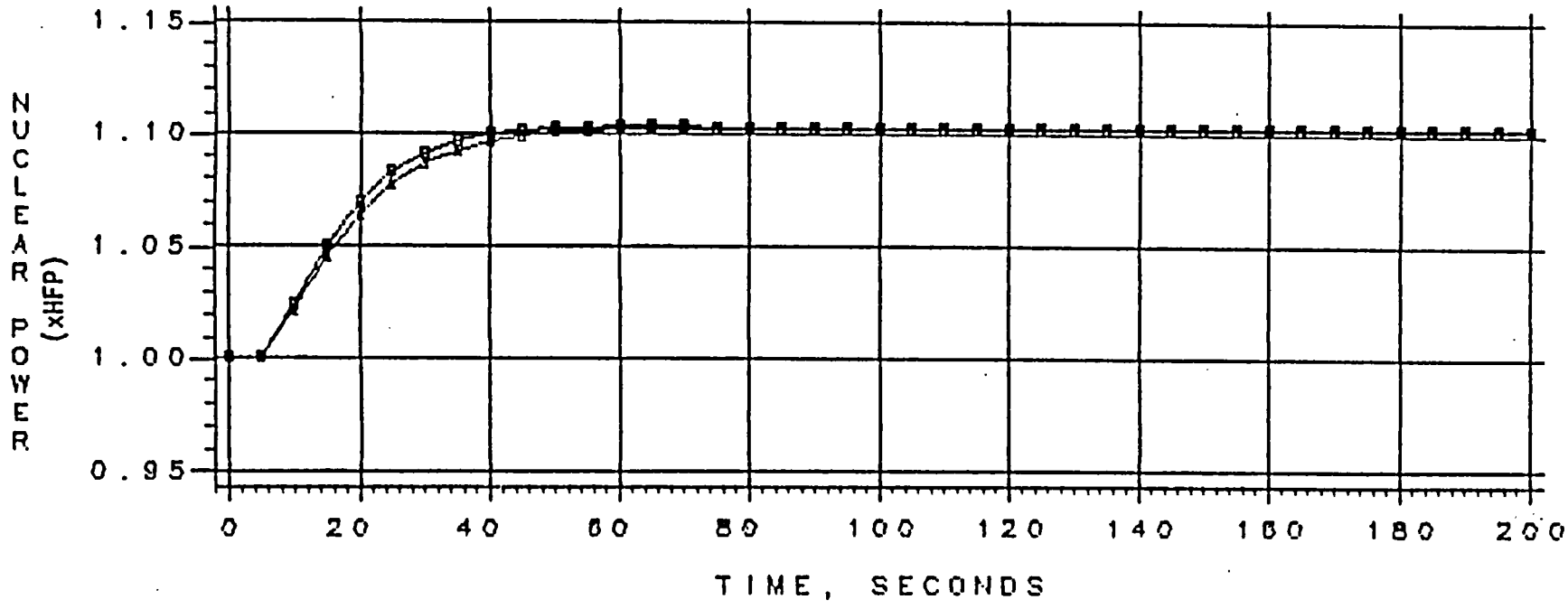


SQUARE - MOST NEG MTC
TRIANGLE- MTC--20PCM/F

FIGURE VI-51

EXCESS LOAD INCREASE

DPC STUDY

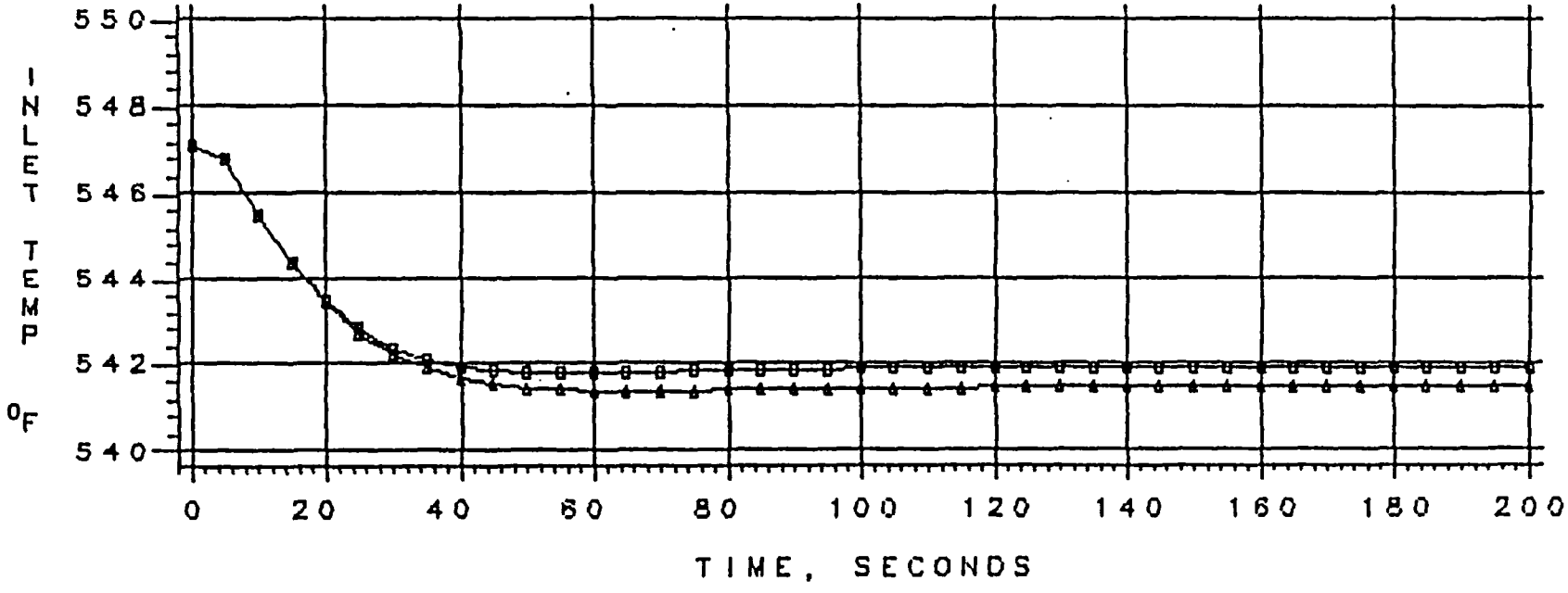


SQUARE - LEAST NEG DPC
TRIANGLE- LEAST NEG DPC*1.25

FIGURE VI-52

EXCESS LOAD INCREASE

DPC STUDY

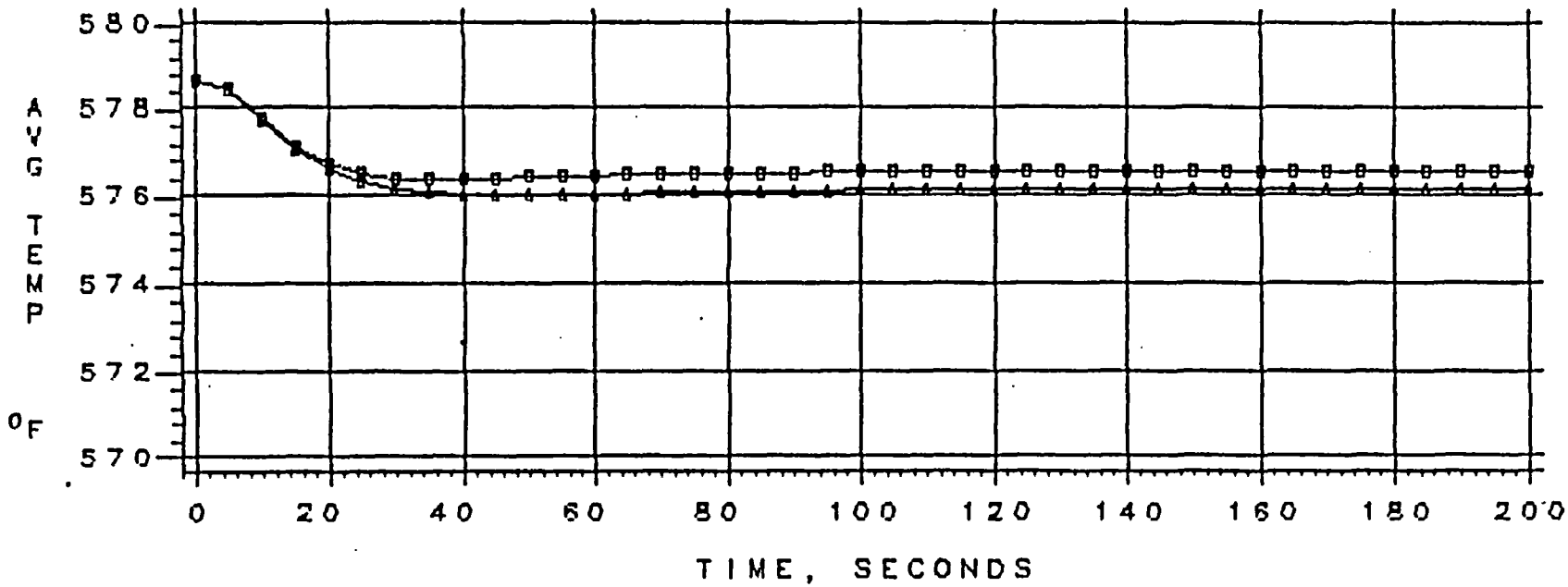


SQUARE - LEAST NEG DPC
TRIANGLE- LEAST NEG DPC*1.25

FIGURE VI-53

EXCESS LOAD INCREASE

DPC STUDY

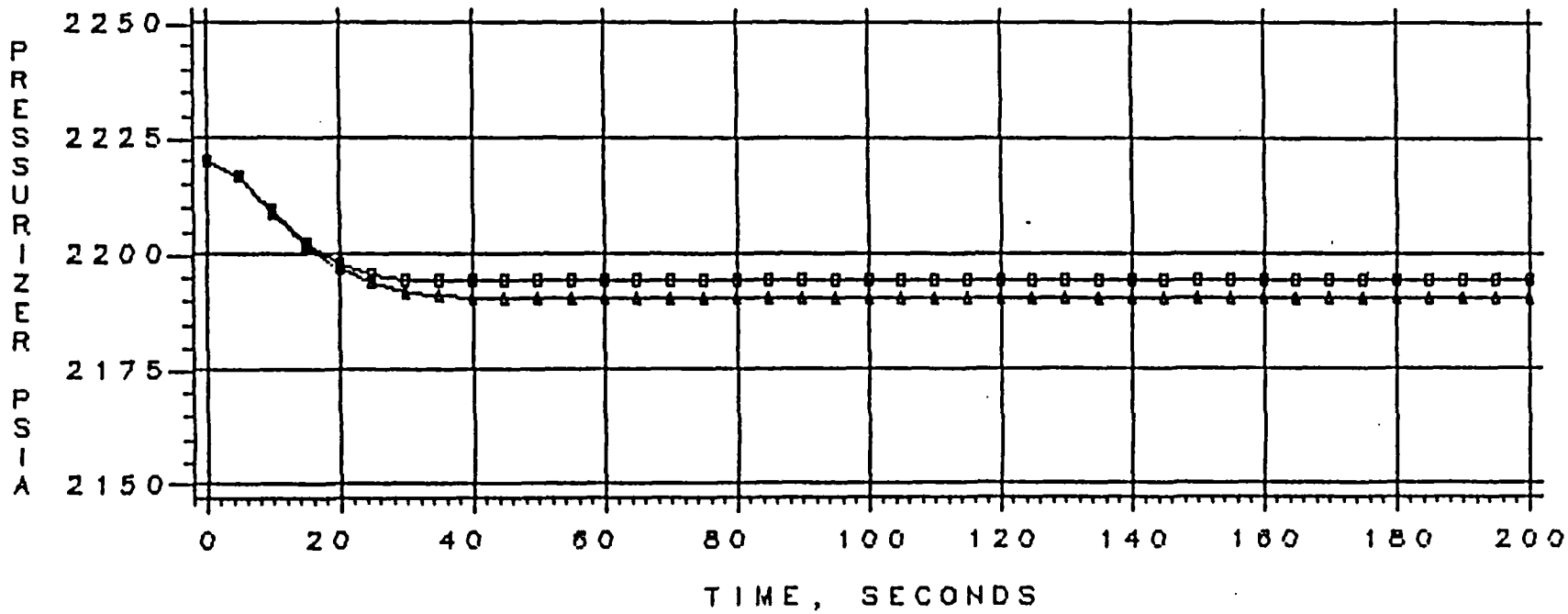


SQUARE - LEAST NEG DPC
TRIANGLE- LEAST NEG DPC*1.25

FIGURE VI-54

EXCESS LOAD INCREASE

DPC STUDY

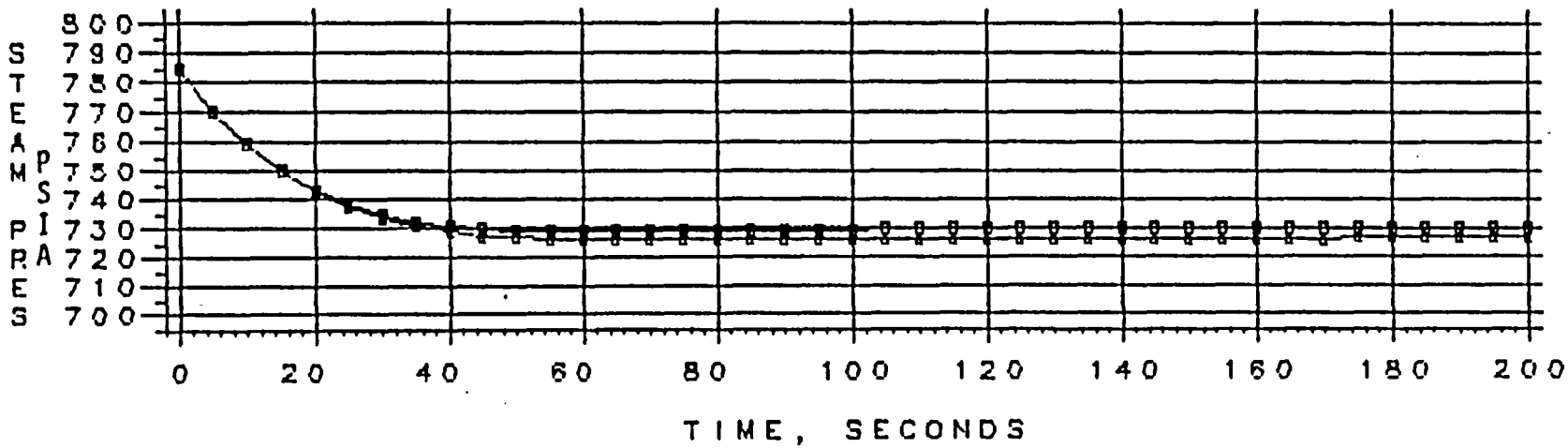


SQUARE - LEAST NEG DPC
TRIANGLE- LEAST NEG DPC*1.25

FIGURE VI-55

EXCESS LOAD INCREASE

DPC STUDY

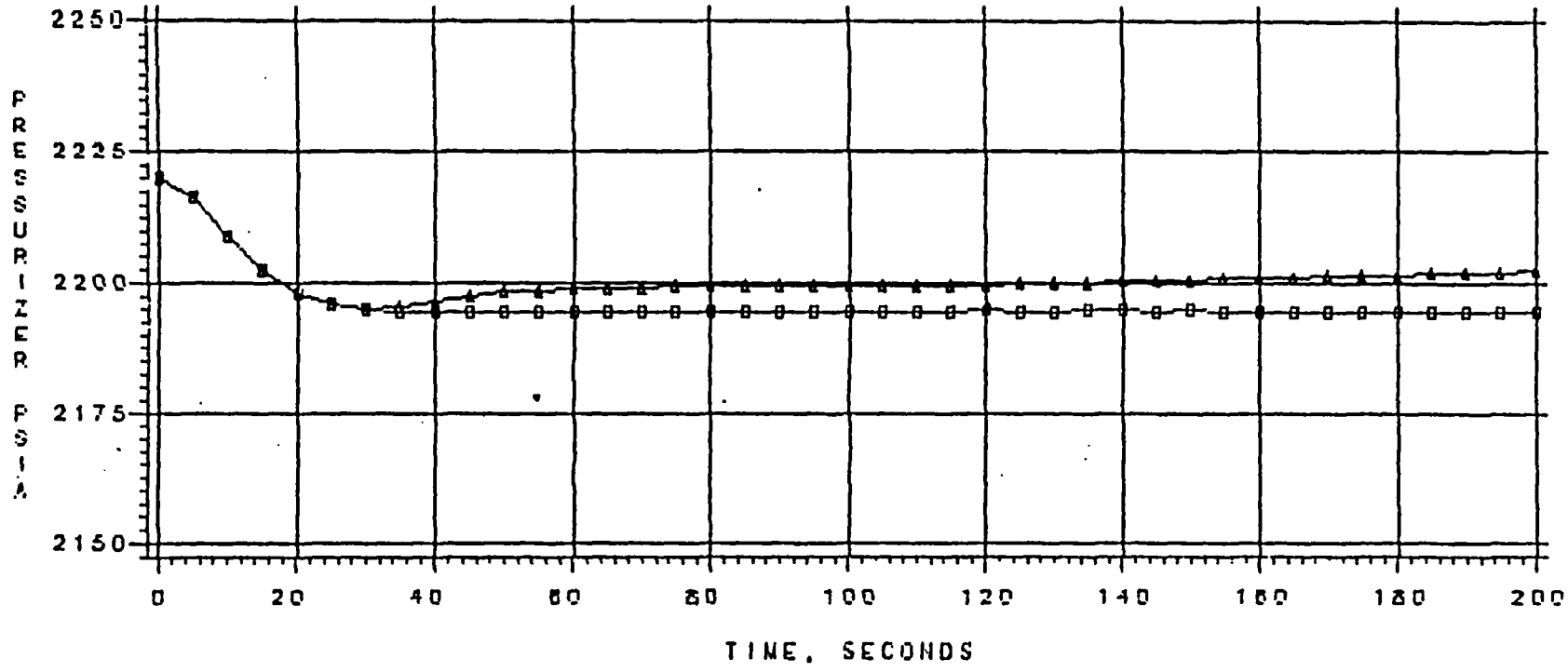


SQUARE - LEAST NEG DPC
TRIANGLE- LEAST NEG DPC*1.25

FIGURE VI-56

EXCESS LOAD INCREASE

HEATER STUDY



SQUARE = NO HEATERS
TRIANGLE= HEATERS

APPENDIX 4
Response to VEP-FRD-41 Rev. 0 RAI #3

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

August 24, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. D. G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 376A
PSE/NAS:vdu
Docket Nos.: 50-280
50-281
50-338
50-339
License Nos.: DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

VEPCO REACTOR SYSTEM TRANSIENT ANALYSES

Attachments 1 through 3 provide supplemental information related to our Topical Report VEP-FRD-41, "Vepco Reactor System Transient Analysis Using the RETRAN Computer Code", transmitted by our letter to you of April 14, 1981, Serial No. 215. Mr. James L. Carter of the Division of Systems Integration informally provided us with a request for additional information which would be required to complete a review of VEP-FRD-41 by the NRC staff.

Portions of this requested information were provided in earlier submittals (Serial No. 060, dated February 27, 1984 and Serial Number 376, dated July 12, 1984).

Attachments 1 through 3 provide the balance of the requested information. Specifically, Attachment 1 provides a description and qualification information for our RETRAN control system models. The results of comparisons of RETRAN calculations to calculations performed with LOFTRAN, a code developed by Westinghouse Electric Corporation, are provided in Attachments 2 (Proprietary) and 3 (Non-Proprietary).

As Attachment 2 contains information proprietary to Westinghouse Electric Corporation, it is supported by an affidavit signed by Westinghouse, the owner of the information (see Attachment 4). The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

A001 1/40 Prog 40 DP Change: POR LPOR NTIS NSIC } Non Prop

8409060272 840824
PDR ADOCK 05000280
P PDR

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations. Correspondence with respect to the proprietary aspects of the Application for Withholding or the supporting Westinghouse affidavit should reference CAW-84-58 and should be addressed to R. A. Wiesemann, Manager, Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, PA 15230.

As we have discussed previously with your staff, we will be happy to meet at any time to discuss the Topical Report and our use of RETRAN in order to assist you in completing your review by your target date of January 15, 1985.

Very truly yours,

W. L. Stewart

Attachments

cc: Mr. D. H. Moran
Standardization and Special Projects Branch

Mr. J. L. Carter
Reactors Systems Branch

Mr. James P. O'Reilly
Regional Administrator
Region II

Mr. James R. Miller, Chief
Operating Reactors Branch No. 3
Division of Licensing

Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Mr. D. J. Burke
NRC Resident Inspector
Surry Power Station

Mr. M. W. Branch
NRC Resident Inspector
North Anna Power Station

ATTACHMENT 1

III. VEPCO RETRAN CONTROL SYSTEM MODELS DESCRIPTION/QUALIFICATION

III. VEPCO RETRAN CONTROL SYSTEM MODELS DESCRIPTION/QUALIFICATION

Vepco's RETRAN models make extensive use of the RETRAN control system modeling capability. The control system feature is used in the following areas:

1. modeling certain features of the reactor protection system. These use signals which are generated by the operation of analog computer elements on various process signals (e.g., the temperature and overpower delta-T trips).

2. modeling certain aspects of the reactor plant control systems which may significantly influence the course of a transient (examples are the pressurizer pressure control system, the turbine governer valve (electrohydraulic) control system and the secondary steam dumps).

3. special submodels which calculate time-dependent boundary conditions or forcing functions which involve several sequential arithmetic operations. The only application of this type which Vepco currrently makes is to a model to describe the transport and mixing of boron in the RCS following a safety injection.

The paragraphs below describe the various models, their development, use

and qualification, where appropriate. Each model is also presented in terms of a block diagram which shows the interrelationships between variables and operations and also describes the interface between the control model and the rest of the system model.

Figures III-1 and III-2 show the overtemperature delta-T reactor trip and the overpower delta-T reactor trip, respectively. Normally, no credit is taken for the overpower delta-T trip feature, and the trip is disabled with a long delay on the corresponding trip card. The overtemperature delta-T logic calculates a delta-T setpoint based on measured average temperature and pressure. The final control block in the sequence differences the actual delta-T with the calculated setpoint. When the difference becomes positive, a reactor trip signal is generated (after an appropriate time delay to account for signal processing delays, etc.). The calculated setpoint conservatively reflects the various processing and setpoint errors. The model has been qualified by comparison of calculated steady-state trip setpoints to hand calculations, and by comparing the calculated time to trip during rod withdrawal transients with FSAR results and with alternate calculations.

Figure III-3 presents the pressurizer pressure control model used by Vepco. The model represents a proportional-plus-integral controller, the output of which drives the pressurizer heaters and spray. The linear variation of spray valve position with controller output is modelled by a weighted summer. Spray flow rate is calculated from the valve position and the loop flow fraction, since the driving force for the spray is the dynamic head of

reactor coolant in the cold leg. The controller output is also used to trip the pressurizer heaters on and off, and to open and close one of the two pressurizer power operated relief valves (the other valve is controlled directly from pressurizer pressure). The controller gain and time constant are taken from plant operating documents. The reference pressure is adjusted up or down during safety analyses as appropriate, to reflect steady state pressure measurement errors.

An example of a comparison of a RETRAN calculated pressure response with the pressure control system assumed to be functional to FSAR results is shown in Figure 5.10 of the topical report. Comparisons with Vepco-generated results using an alternate method are presented and discussed in Section V of this supplement.

Figures III-4 and III-5 illustrate how the pressurizer pressure and steam pressure, respectively, are filtered before passing the signals to the reactor trip and engineered safeguards (safety injection) systems. The lead and lag time constants are best estimate values, taken from plant setpoint documentation.

Figure III-6 illustrates how the control system function generator feature is used to generate power feedback reactivity. This method of representing the reactivity feedback is used in situations where power is varying slowly enough that a defined relationship between power and fuel temperature exists. In most cases the independent variable is taken as the neutron power. For steam line break calculations, where the system returns to power

from a subcritical condition, using neutron power as the independent variable could lead to calculational instabilities in the viginity of the initial power 'jump' following a return to power. For this reason, the heat flux is used as the input variable for steambreak calculations. For transients where the neutron power is varying rapidly (e.g., rod withdrawal from subcritical) the power reactivity concept is not applicable, and doppler feedback is represented as a function of fuel temperature.

Figure III-7 shows how main steam line isolation valve closure following a steam line break is modelled. This model allows the initial opening of the break and the closure of the isolation valve to be modelled at the same junction. The upper integrator simulates the opening of the break in 0.01 seconds. The lower integrator recloses the break path upon reciept of a signal from the trips which model the engineered safety features. The closure time is the maximum allowable value from the technical specifications.

Figure III-8 shows how control blocks are arranged to calculate a region-weighted moderator temperature for use in steam line break calculations. Since point kinetics is used, consistent with vendor methodology, a radial moderator temperature weighting factor is used to approximate the effects of the coldest water entering the core region containing a stuck rod. The function generator allows representation of a nonlinear variation of reactivity with moderator temperature.

Figure III-9 represents the core average heat flux calculation performed in

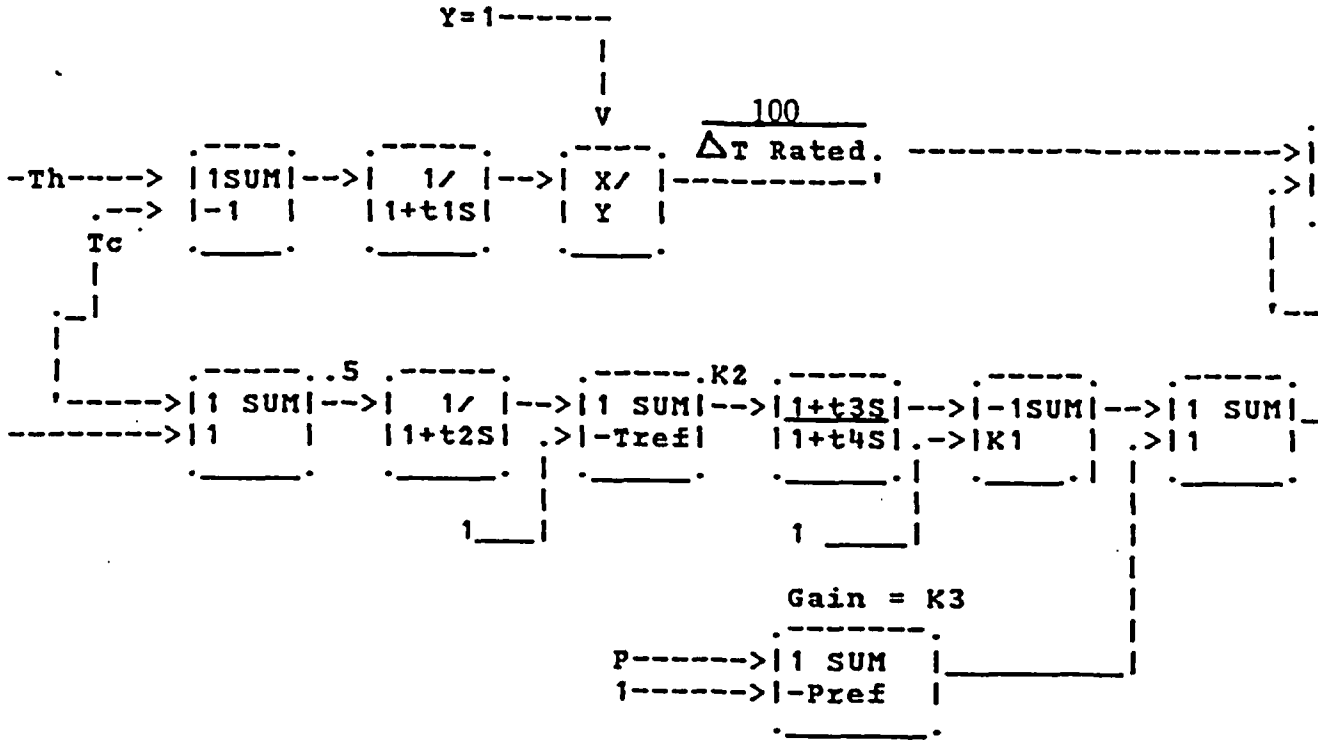
the two loop model. This heat flux is expressed in terms of fraction of the rated full power value, and is used for editing purposes, and to drive the power reactivity feedback calculation described in Figure III-6 during steamline break calculations.

A few of the accidents which may require RETRAN analysis are affected by the turbine governor valve (or electrohydraulic control-EHC) system. A simple control system model is used to describe the effects of this system on steam flow to the turbine; this model is shown in Figure III-10. The model assumes that steam flow is constant with decreasing pressure until the governor valves reach a full open position. Thereafter, steam flow is assumed to decrease linearly with pressure.

Certain best estimate calculations (e.g. the analysis of the North Anna cooldown event discussed in Section 5.3.3 of the topical report), require a representation of the secondary steam dump system. Figure III-11 shows the arrangement of control blocks used to calculate steam dump flow area as a function of average temperature. Following a turbine trip, the steam dumps rapidly trip open to provide load rejection capability for the system. The valves then modulate closed as the measured average coolant temperature decreases and approaches the no-load value. Values for the no-load reference temperature, Tref, the filter time constants T1 and T2 and the program for dump capacity vs (Tavg - Tref) are all taken from current plant setpoint documents. For the North Anna cooldown event, initial post-trip cooldown rates calculated with this model agreed well with observed trends.

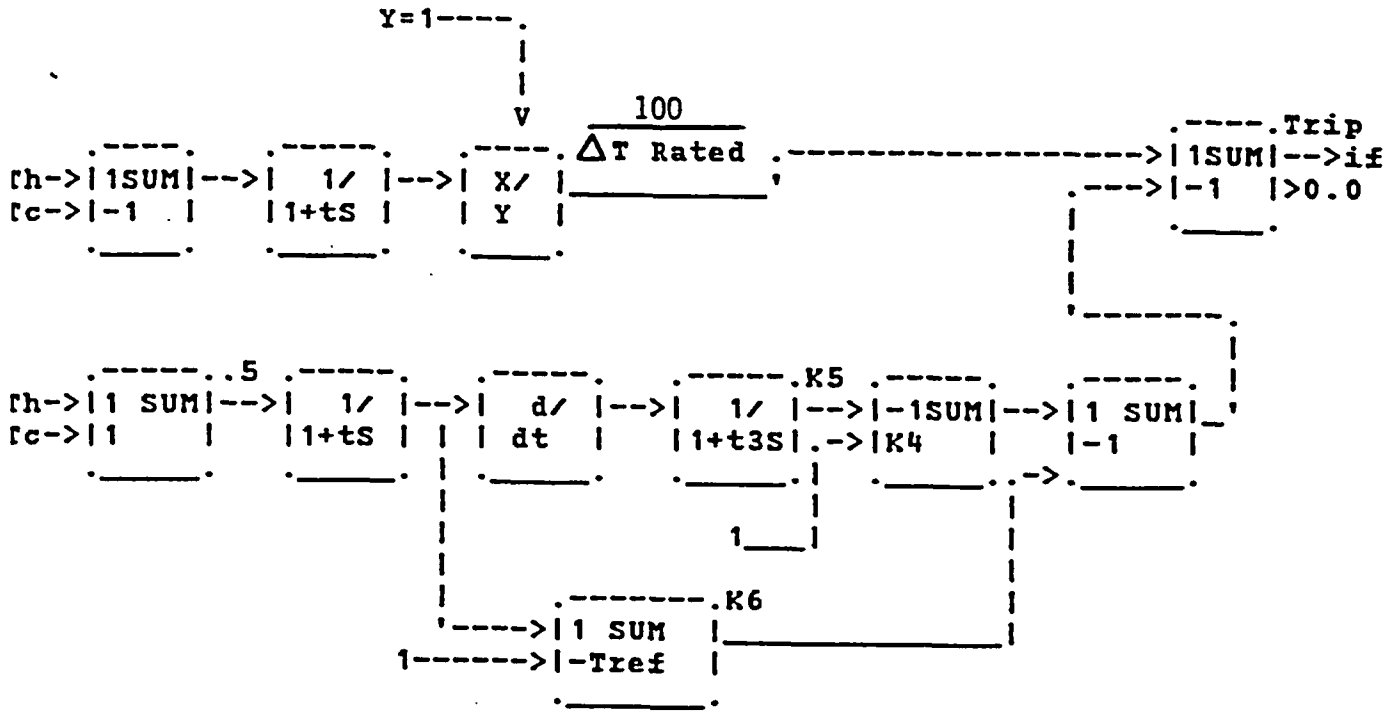
The RETRAN submodel for calculating the mixing and transport of high boron concentration water from safety injection into and around the primary coolant loops is shown in Figure III-12. The model shown is appropriate for full flow conditions in all loops. Pipe-like regions of the system are treated with delay control blocks. Plena are treated with a first order lag. The delay times and time constants are calculated from the nodal fill times for the various regions. Time dependent core boron concentrations obtained with this model agree reasonably well with results obtained with hand calculations and simpler, RCS-average mixing assumptions.

FIGURE III-1
OVERTEMPERATURE DELTA-T TRIP



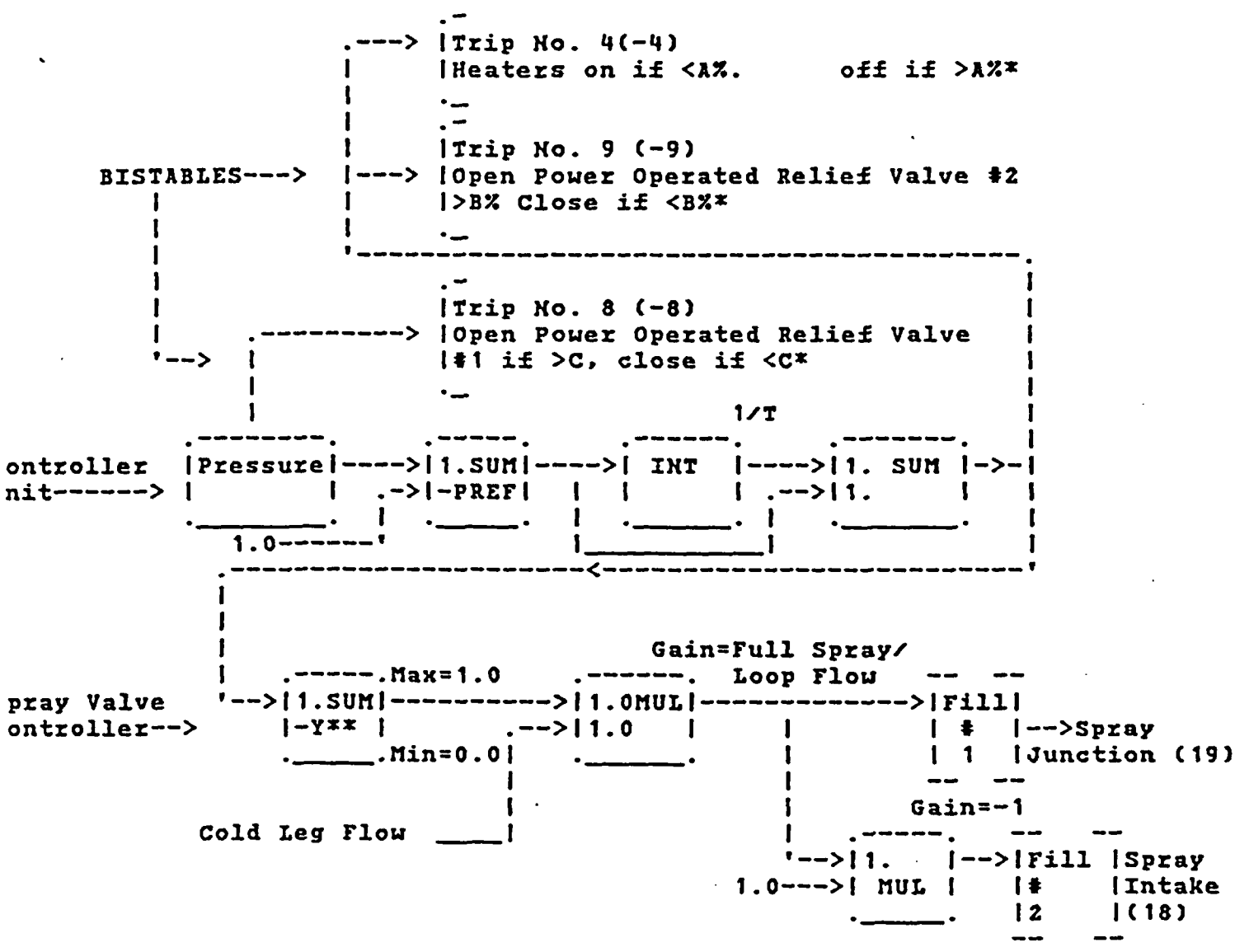
$$\text{rip if} \quad \frac{\Delta T \times 100}{\Delta T \text{ Rated}} > K1 - K2 \frac{(1+t3S)}{(1+t4S)}(Tavg - Tref) + K3\,(P-Pref)$$

FIGURE III-2
OVERPOWER DELTA-T TRIP



$$\text{Trip if } \frac{\Delta T \times 100}{\Delta T \text{ Rated}} > K4 - K5 \frac{t3S}{1+t3S} Tavg - K6 (Tavg - Tref)$$

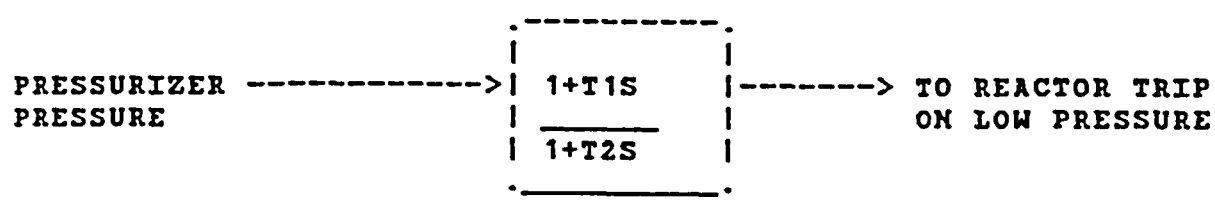
FIGURE III-3
LOGIC FOR PRESSURE CONTROL SYSTEM



* Setpoints A, B and C are best estimate values taken from plant setpoint documents.

** The parameter "Y" in the summer block for the spray valve controller is a measure of the difference between the pressure at which the spray valves begin to open and the reference pressure. "T" is the pressure controller reset time constant.

Figure III-4
LOW PRESSURE TRIP SIGNAL

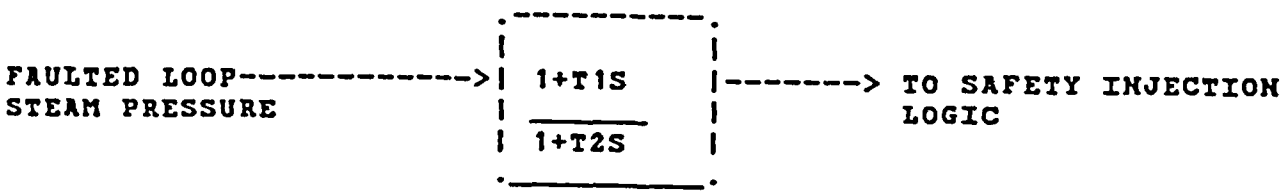


T1 = LEAD TIME CONSTANT
T2 = LAG TIME CONSTANT
S = LAPLACE TRANSFORM VARIABLE

TIME CONSTANTS ARE TAKEN FROM PLANT SETPOINT DOCUMENT
LOW PRESSURE TRIP SETPOINT IS THE SAFETY ANALYSIS VALUE
(INCLUDES UNCERTAINTIES)

SEE ALSO SINGLE LOOP MODEL TRIP DESCRIPTION IN SECTION I.

FIGURE III-5
LOW STEAM PRESSURE SIGNAL



T1 = LEAD TIME CONSTANT
T2 = LAG TIME CONSTANT
S = LAPLACE TRANSFORM VARIABLE

TIME CONSTANTS ARE TAKEN FROM PLANT SETPOINT DOCUMENT
LOW PRESSURE SETPOINT IS THE SAFETY ANALYSIS VALUE
(INCLUDES UNCERTAINTIES)

SEE ALSO TWO LOOP MODEL TRIP DESCRIPTION IN SECTION 1.

FIGURE III-6
POWER REACTIVITY FEEDBACK FUNCTION

```
                                        .---------.
CORE AVERAGE HEAT FLUX----------->|         |
       OR                         |   FNG   |
NORMALIZED-----> X INITIAL------->|         |---->POWER --->TO
POWER              % POWER        ._________.   REACTIVITY  KINETICS
                 (GAIN FACTOR)         |          ($)        TABLES
                                       |
                                       |
      TABLE OF POWER REACTIVITY---------
        VS. POWER IN $
```

SEE ALSO "DOPPLER POWER COEFFICIENT" DESCRIPTION IN SECTION IV-
"INPUT OPTIONS"

FIGURE III-7
SIMULATION OF MAIN STEAM ISOLATION VALVE CLOSURE
FOR STEAMLINE BREAK CALCULATIONS

```
                          .------.GAIN=100
                          |      |MAX=1.0              .-----.MAX=1.0
IP TO INITIATE-------->| INT  |------------------->|+1   |        VALVE
EAK (0 OR 1)              |      |                     | SUM | ----> AREA
                          .______.              .---->|-1   |        TABLE
                                                |     ._____.MIN=0.0
                                                |
GH STEAM FLOW/                                  |
W TAVE                                        .------. MAX = 1.0
  OR          -->TIME--->MAIN STEAM------>|      | GAIN = 1/T1
GH STEAM FLOW/ DELAY    ISOLATION          | INT  | T1=CLOSURE TIME
W PRESSURE              SIGNAL (0 OR 1)    .______.
```

E ALSO "MAIN STEAM ISOLATION VALVES" IN SECTION II - COMPONENT
DELS AND TWO LOOP MODEL TRIP DESCRIPTION IN SECTION I - VOLUME
D FLOW PATH NETWORK DESCRIPTION.

IS LOGIC APPLIES ONLY TO THE "INTACT" LOOP DURING A MAIN STEAM LINE
EAK.

FIGURE III-8
MODERATOR TEMPERATURE DEFECT CALCULATION (TWO LOOP MODEL)

```
          .-----.                .------.GAIN1                  .------.
T113--->|      |              |      |                    |      |<--T115
T114--->| SUM |------------>| SUM  |<------------------ | SUM  |<--T116
          ._____.                ._____.                    ._____.
                                    |     .----------.
                                    |     |REACTIVITY|
                                    |     |    VS    |
                                    |     |MOD TEMP  |
                                    |     .__________.
                                 .--V---.      |       .------.
                                 |      |      '------>|      |        TO
                                 | SUM  |------------>| FNG  |---->REACTIVITY
                                 ._____.               ._____.        TABLES
                                    /|\
          .-----.                .--|---. GAIN2                  .------.
T213--->|      |              |      |                    |      |<--T215
T214--->| SUM |------------>| SUM  |<------------------ | SUM  |<--T216
          ._____.                ._____.                    ._____.
```

TXXX = MODERATOR TEMPERATURE IN VOLUME XXX
GAIN1 = RMWF/4
GAIN2 = (1-RMWF)/4
RMWF = RADIAL MODERATOR TEMPERATURE WEIGHTING FACTOR

SEE ALSO THE GENERALIZED DATA TABLE DESCRIPTION FOR MODERATOR TEMPERATURE DEFECT IN SECTION IV - INPUT OPTIONS, AND THE TWO LOOP MODEL CONTROL VOLUME DESCRIPTION IN SECTION I - VOLUME AND FLOW PATH NETWORK DESCRIPTION.

FIGURE III-9
CORE HEAT FLUX CALCULATION (TWO LOOP MODEL)

```
        .-----.                 .------.                          .------.
Q101--->|     |                 |      |                          |      |<--Q103
Q102--->| SUM |---------------->| SUM  |<------------------------| SUM  |<--Q104
        .-----.                 .------.                          .------.
                                   |
                                   |
                                   |
                                   |
                                   |
                                .--V---. GAIN
                                |      |
                                | SUM  |-----> CORE AVERAGE HEAT FLUX
                                .------.            FOR MINOR EDITS
                                   /\
        .-----.                 .------.                          .------.
Q201--->|     |                 |      |                          |      |<--Q203
Q202--->| SUM |---------------->| SUM  |<------------------------| SUM  |<--Q204
        .-----.                 .------.                          .------.
```

QXXX = POWER TO WATER FROM CONDUCTOR XXX, BTU/HR
GAIN = CONVERSION FACTOR, BTU/HR TO FRACTION OF RATED POWER

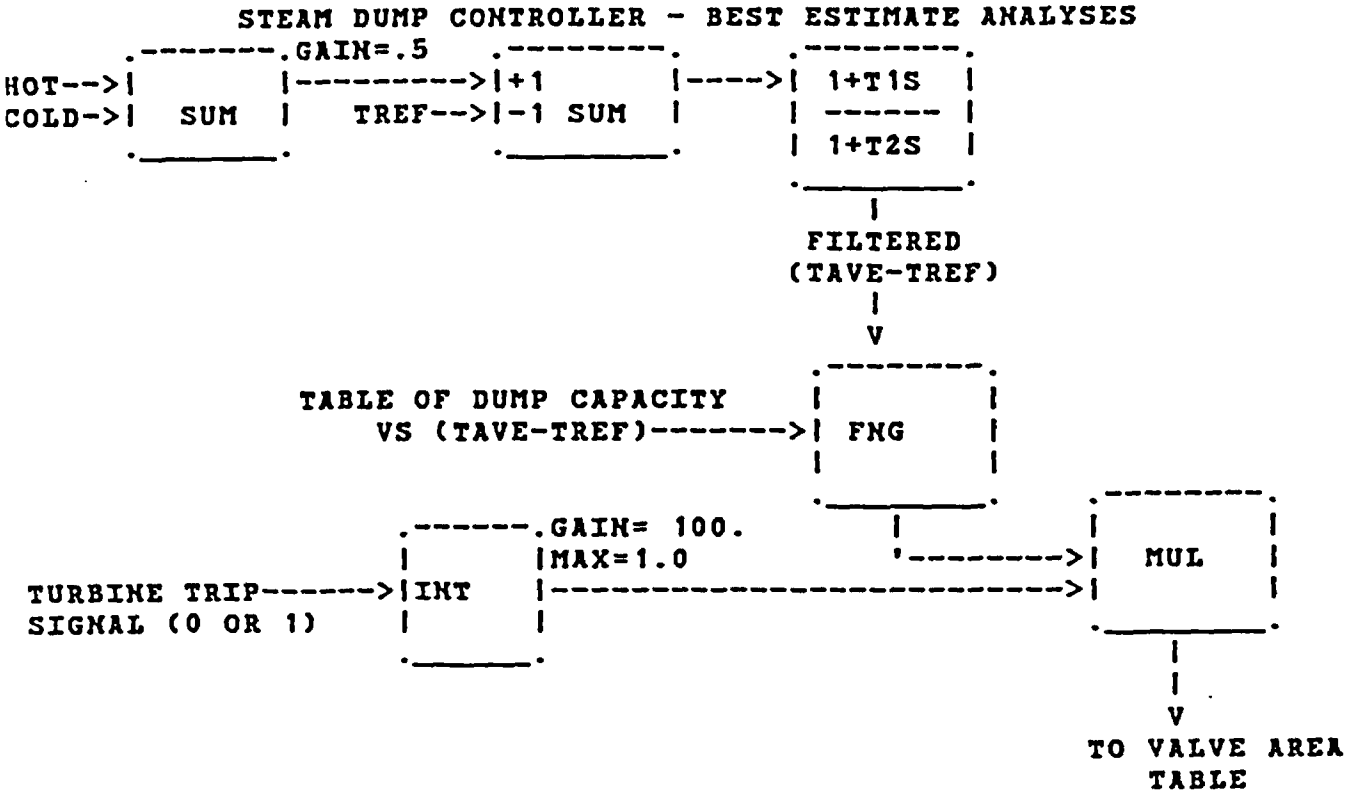
SEE ALSO TWO LOOP MODEL HEAT CONDUCTOR DESCRIPTION IN SECTION 1

FIGURE III-10
SIMULATION OF ELECTROHYDRAULIC TURBINE CONTROL SYSTEM

```
.----------------------.
|GENERAL DATA TABLE    |
|STEAM FLOW VS STEAM   |
|PRESSURE FOR FULL-OPEN|        .--------.
|THROTTLE CONDITIONS   |------>|        |
.______________________.    .-->|  FNG   |--.    .-------.
                            |   .________.  '-->|       |----->STEAM FLOW
 STEAM PRESSURE------------                   .-->|  MIN  |      FILL TABLE
                            .--------.   |    ._______.
 TIME-------------------------->|        |--'
                            .-->|  FNG   |
.----------------------.    |   .________.
|GENERAL DATA TABLE    |----
|STEAM DEMAND VS TIME  |
.______________________.
```

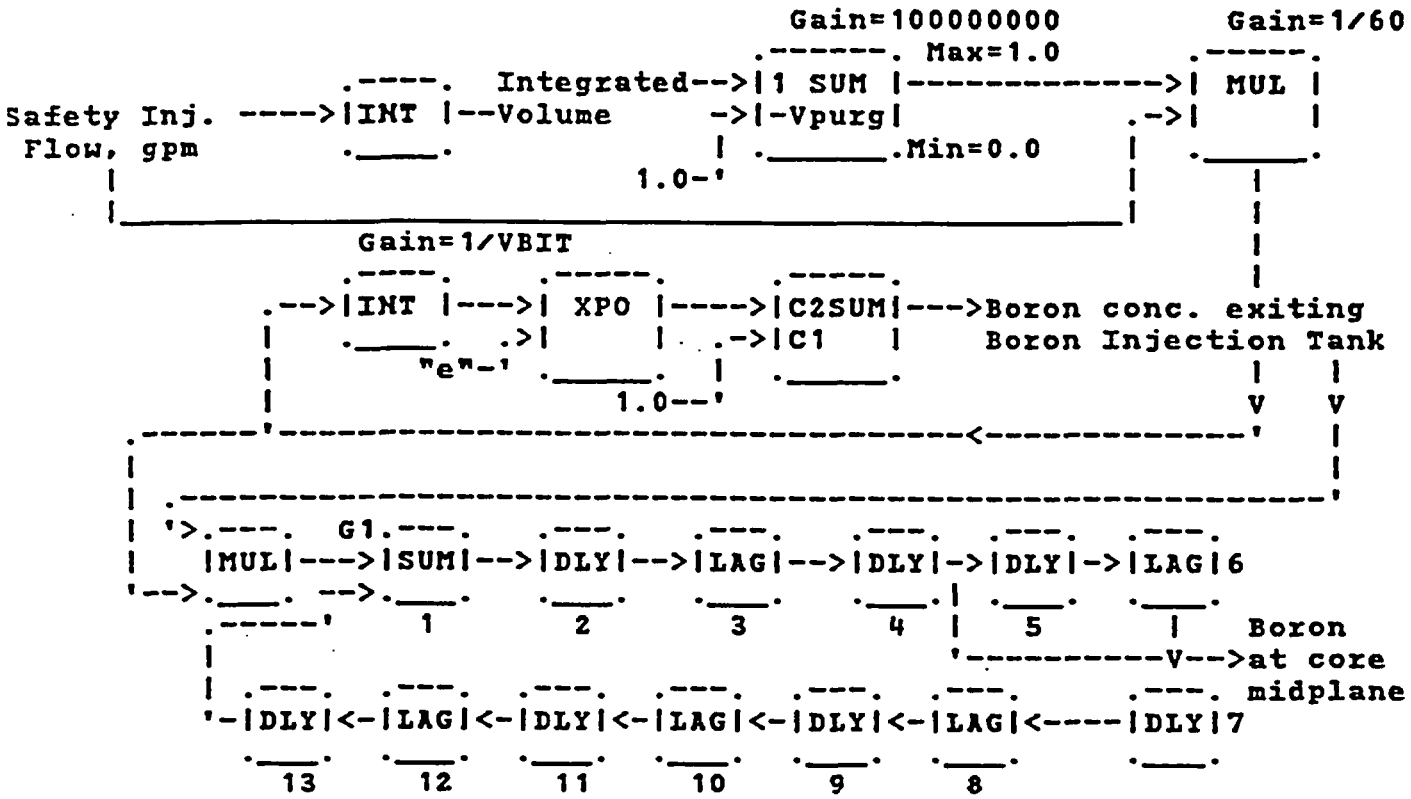
SEE ALSO THE MAIN STEAM FLOW FILL TABLE DESCRIPTION IN SECTION IV.

FIGURE III-11
STEAM DUMP CONTROLLER - BEST ESTIMATE ANALYSES



HIS FEATURE IS NOT USED IN SAFETY ANALYSES, WHICH TAKE NO CREDIT
OR THE LOAD REJECTION CAPABILITY ASSOCIATED WITH STEAM DUMP. THE
EATURE IS USED IN SOME BEST ESTIMATE ANALYSES, SUCH AS THE ANALYSIS
F THE NORTH ANNA COOLDOWN EVENT DISCUSSED IN SECTION 5.3.3 OF THE
OPICAL REPORT.

FIGURE III-12
BORON TRANSPORT MODEL



egion Numbers:
1- cold leg mixing zone
2- cold leg/downcomer
3- bottom plenum
4- bottom core
5- top core
6- outlet plenum
7- hot leg
8- steam generator inlet plenum
9- steam generator tubes
10- steam gen. outlet plenum
11- cold leg 1
12- pump
13- cold leg 2 (pump outlet to mixing zone)

ATTACHMENT 2

V. COMPARISON TO ALTERNATE CODE CALCULATIONS

(PROPRIETARY)

ATTACHMENT 3

V. COMPARISON TO ALTERNATE CODE CALCULATIONS

(NON-PROPRIETARY)

V. COMPARISON TO ALTERNATE CODE CALCULATIONS

In the topical report (VEP-FRD-41), Vepco provided numerous comparisons of transient results obtained with our RETRAN models to licensing results obtained by the NSSS/fuel vendor for Vepco's units. The latter were performed primarily to support the FSAR's and subsequent reload safety evaluations. This section provides a supplement to those comparisons in the form of parallel calculations performed by Vepco using both a standard Vepco RETRAN model and a corresponding LOFTRAN model. The LOFTRAN code is a proprietary code developed and maintained by the Westinghouse Electric Corporation for use in performing general non-LOCA accident analyses. Vepco has had access to LOFTRAN for four years via a special licensing agreement with Westinghouse. A detailed description of the LOFTRAN code is given in Reference V-1.

Vepco safety analysis engineers have undergone extensive training in the use of Westinghouse core design and safety analysis codes, including formal classroom instruction by Westinghouse (see Table V-1) and on-the job-training at Westinghouse and/or Vepco. Part of this training included a formal forty-hour non-LOCA safety analysis course which covered theory, input preparation and applications of LOFTRAN. Surry and North Anna specific models have been assembled in-house and have been reviewed and commented on informally by Westinghouse.

The comparisons shown here were performed with a LOFTRAN model of the Surry reactors assembled by Vepco using the same data base used for development

of the RETRAN models. Thus system water volumes, conductor heat transfer areas, initial loop and core flow rates, rated pump parameters, etc. are identical for the two models. Initial conditions such as [ ]a,c were [ ]a,c resulted from the use of [ ]a,c in the two codes to represent the equations of state for the coolant, etc.). Comparison of steady state conditions for the two codes are provided in Table V-2. Table V-3 provides a description of the three transients used in the comparisons. Discussions of the comparisons are given in the paragraphs below.

TRAINING
WESTINGHOUSE DESIGN AND SAFETY ANALYSIS CODES AND METHODOLOGY

COURSE DESCRIPTION	COURSE LENGTH	NUMBER OF PRESENTATIONS	TOTAL TRAINING
INTRODUCTION TO THE W COMPUTER SYSTEM	3 DAYS	2	39 MAN-DAY
BASIC PWR CORE PHYSICS AND THERMAL HYDRAULICS	5 DAYS	1	35 MAN-DAYS
METHDOLOGY AND COMPUTER MODELS FOR W DESIGN CODES	5 DAYS	3	80 MAN-DAYS
WESTINGHOUSE APOLLO AND MIXIN DESIGN CODES	5 DAYS	2	60 MAN-DAY
NUCLEAR DESIGN DEVELOPMENT OF THE RELOAD SAFETY ANALYSIS CHECKLIST	5 DAYS	3	115 MAN-DA
INTRODUCTION TO NON-LOCA SAFETY ANALYSIS	5 DAYS	3	80 MAN-DAY
ROD EJECTION, MAIN STEAMLINE BREAK, DROPPED ROD ANALYSIS	5 DAYS	2	40 MAN-DAY
WESTINGHOUSE THERMAL HYDRAULIC METHODS	5 DAYS	1	20 MAN-DAY
WESTINGHOUSE LARGE BREAK LOCA CODES (THEORY)	5 DAYS	2	30 MAN-DAY
WESTINGHOUSE LARGE BREAK LOCA CODES (ON-THE-JOB TRAINING)	20 DAYS	3	60 MAN-DA
		TOTAL	559 MAN-DA

TABLE V-2

COMPARISON OF RETRAN/LOFTRAN CALCULATED STEADY STATE CONDITIONS

a,c

Parameter	RETRAN Value
Core power, mwt	2489.82 -S
Pump heat, mwt	12.15 -C
Tcold, °F	547.11 (after pump) -S* 546.68 (before pump) -C*
Thot, °F	610.15 -C
Tavg, °F	578.63 -C
Steam Flow, lb/sec	3017.5 -S
Steam Pressure, psia	785.0 -S
Steam generator inventory, lbm	313200 -S
Feedwater enthalpy, btu/lbm	413.69 -C
Steam Enthalpy, btu/lbm	1199.7 -C
Average fuel temperature, °F	1405.7 -C

'C' denotes a code calculated parameter
'S' denotes a parameter specified as input

TABLE V-3
RETRAN/LOFTRAN Transient Comparisons

Case	Description
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. Pressurizer sprays, PORV's and steam generator relief valves are assumed available.
3	Simultaneous trip of all three reactor coolant pumps at hot full power. No credit taken for reactor trip on pump undervoltage or underfrequency. Pressurizer sprays, PORV's and steam generator relief valves are assumed available.

REACTOR TRIP

Figures V-1 to V-4 show the results for the reactor trip. Figure V-1 presents the core response in terms of nuclear power, fuel temperature and core heat flux. As the results show, the core neutron and thermal kinetics models for the two codes give results which are [ ]a,c

Figure V-2 compares the steam generator response in terms of steam pressure and primary to secondary heat transfer, or heat extraction, rate. The response of the reactor coolant system is shown in Figures V-3 (RCS average temperature) and V-4 (pressurizer water volume and pressure). The RCS average temperature response [ ]a,c In RETRAN, the temperature at a specific location is input (in this case the cold leg) and the average temperature is then calculated based on the steady state initialization results. In Figure V-4, about [

]a,c

TURBINE TRIP

Figures V-5 to V-8 show the comparisons for the turbine trip without direct reactor trip. The results are shown out to the time of reactor trip, and present steam pressure, reactor inlet temperature, reactor power and pressurizer pressure, respectively. Figure V-7 is of interest in that it shows a slight difference in the nuclear power response. This difference stems from a different treatment of power reactivity feedback in the two models. The LOFTRAN model generates power feedback as a function of core heat flux. The RETRAN model, on the other hand, uses a tabular representation for the power feedback which relates the feedback directly to neutron power. Since the reactivity feedback is more accurately a function of the fuel temperature, [

]a,c

The [

]a,c in the two models. Vepco's RETRAN models treat the steady state pressure error as a bias in the signal going into the proportional plus integral contoller which controls pressurizer spray and one of the two pressurizer power operated relief valves. Thus spray and one PORV are assumed to open about 30 psi below their nominal setpoints. [

]a,c Since the spray and one PORV are actuated [ ]a,c in the RETRAN model, a [ ]a,c pressure [ ]a,c results. For

safety analyses related to system overpressure and vessel integrity concerns, pressurizer PORV's and spray are assumed not to function, and this modeling [ ]a,c on the results.

FLOW COASTDOWN

Figures V-9 to V-11 show comparisons for the flow coastdown event. Total core flow (this is a three-pump coastdown) is shown in Figure V-9. LOFTRAN uses a lumped parameter approach in solving for loop flow (the rate of change of flow is a characteristic of the entire coolant loop), whereas RETRAN solves a momentum equation at every flow junction in the loop. For incompressible flow, the two models give [ ]a,c results, as shown. Figure V-10 presents the nuclear power and core heat flux repsonse, and pressurizer presssure response for the two codes is presented in Figure V-11. The [ ]a,c following the trip is related to [ ]a,c Spray is driven by the dynamic head of the reactor coolant flowing through the loops. In the RETRAN model, under flow coastdown conditions, spray flow is assumed to be proportional to loop flow. In the LOFTRAN model, the spray flow is assumed proportional to the square of the loop flow. Thus under loss of flow conditions LOFTRAN [ ]a,c in the transient.

CONCLUSIONS

Transient results from the Vepco RETRAN models have been compared to Vepco-generated results using the LOFTRAN code. The [ ]a,c in results is [ ]a,c in the codes.

REFERENCE

1. Burnett, T. W. T., et al, "LOFTRAN CODE DESCRIPTION," WCAP-7907-P-A (Westinghouse Proprietary Class 2), WCAP-7907-A (Westinghouse Non-Proprietary), April 1984 .

POWER OF HEAT FLUX, FRACTION OF INITIAL

TIME, SEC.

FUEL, TEMP °F

a,c

S.G. HEAT EXTRACTION RATE
FRACTION OF INITIAL

TIME, SEC.

STEAM PRESSURE, PSIA

a,c

a,c

TAVG

TIME ,SEC.

PRESSURIZER PRESSURE, PSIA.

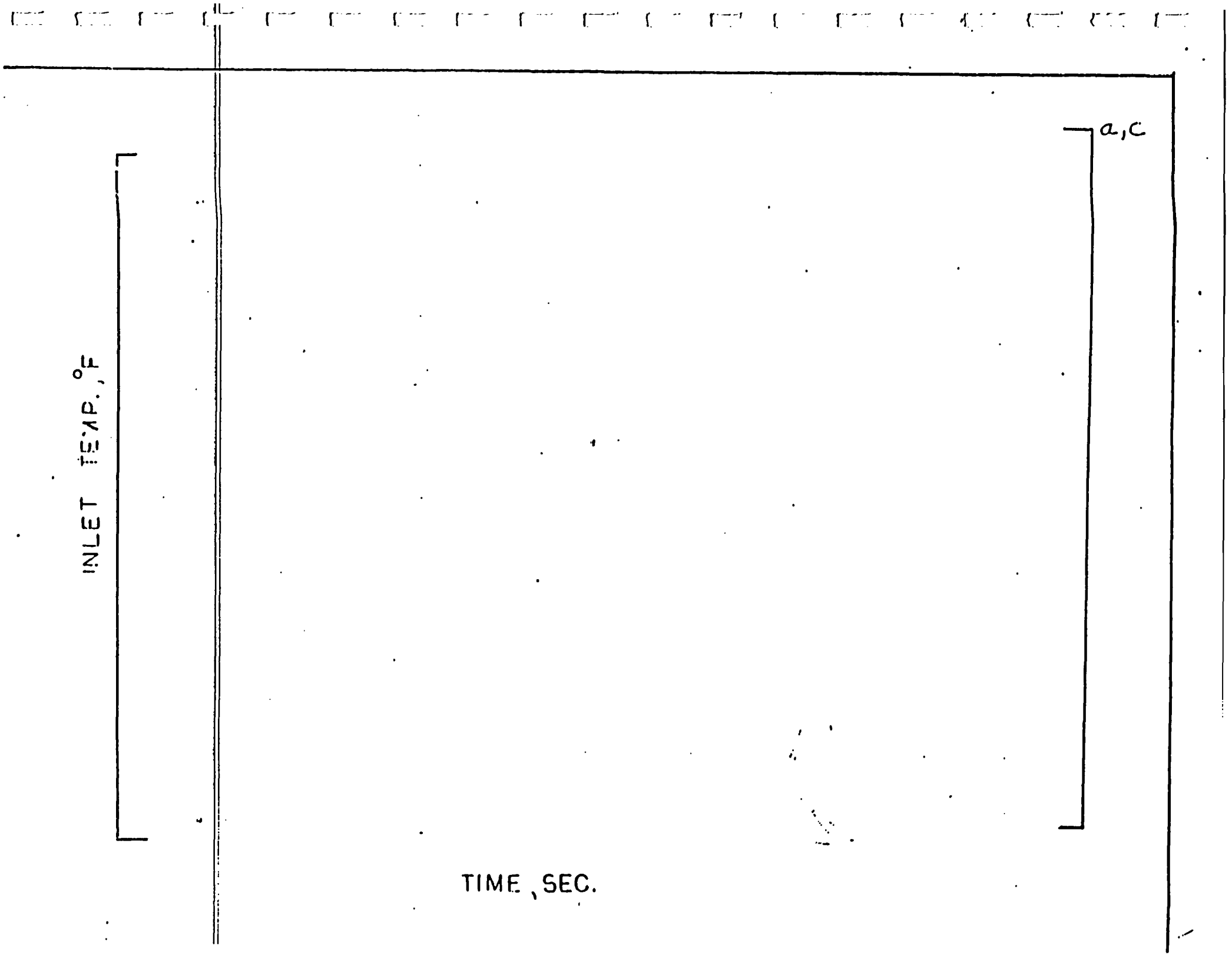
TIME, SEC.

PRESSURIZER WATER VOLUME, $FT.^3$

STEAM PRESSURE, PSIA.

TIME, SEC.

a,c


a,c
INLET TEMP., °F
TIME, SEC.

NUCLEAR POWER ,FRACTION OF INITIAL

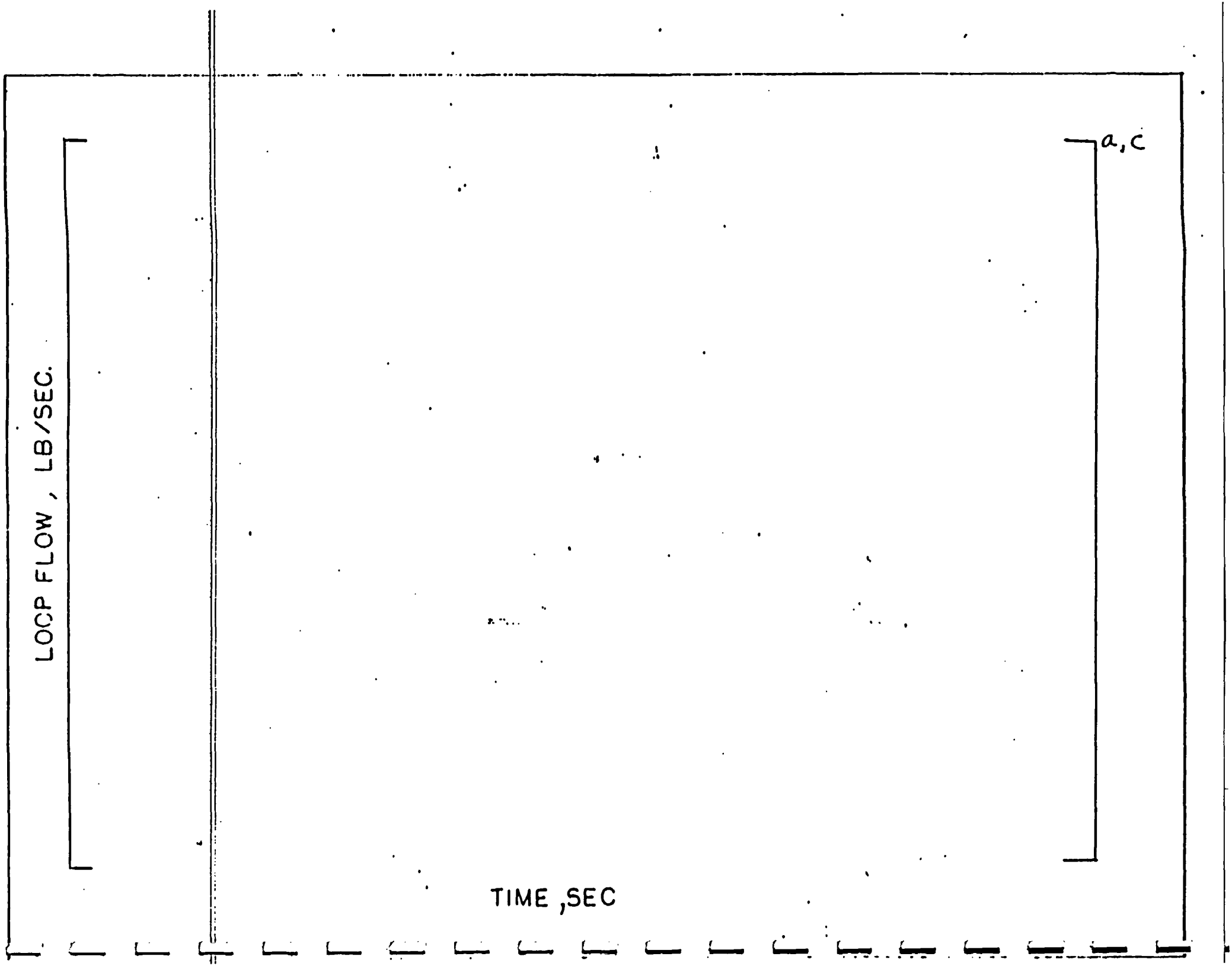
TIME ,SEC.

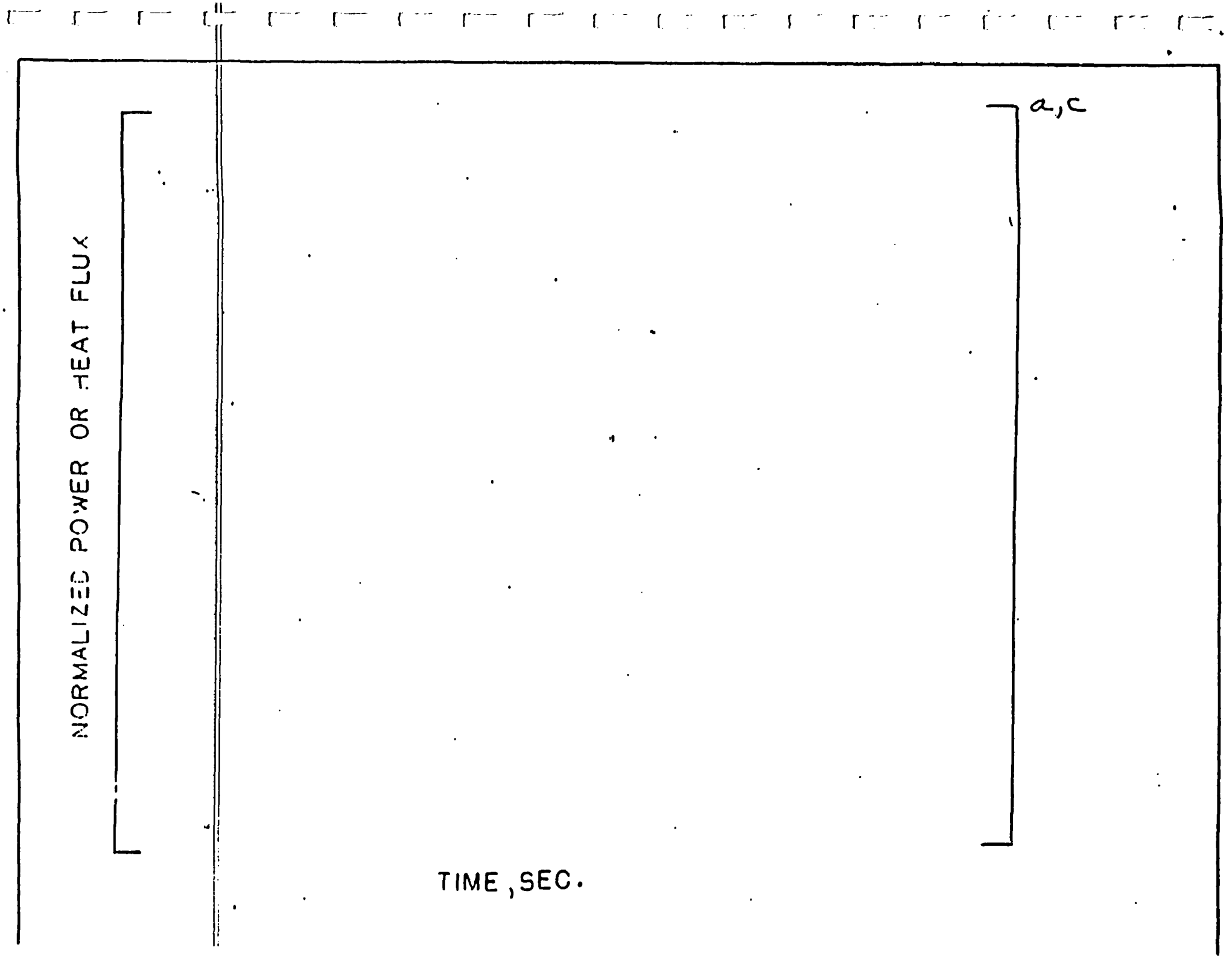
a,c

PRESSURIZOR , PSIA

TIME, SEC

a,c


a,c
LOOP FLOW , LB/SEC.
TIME ,SEC


a,c
NORMALIZED POWER OR HEAT FLUX
TIME, SEC.

a,c

PRESSURIZER PRESSURE, PSIA.

TIME, SEC.

ATTACHMENT 4

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

August 7, 1984
CAW-84-58

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20055

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

REFERENCE: Duke Power Company letter to NRC dated March 1984

Dear Mr. Denton:

The proprietary material for which withholding is being requested in the reference letter by Virginia Electric and Power Company is further identified in an affidavit signed by the owner of the proprietary information, Westinghouse Electric Corporation. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10CFR Section 2.790 of the Commission's regulations.

The proprietary material for which withholding is being requested is of the same technical type as that proprietary material previously submitted with application for withholding AW-76-31.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Virginia Electric and Power Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-84-58, and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

/pj
cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

AW-76-31

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed
before me this 5 day
of July 1976.

Notary Public

RC :IC

... COUNTY
MY COMMISSION EXPIRES APR. 15, 1978

(1) I am Manager, Licensing Programs, in the Pressurized Water Reactor Systems Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition in those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the attachment to Westinghouse letter No. NS-CE-1142, Eicheldinger to Eisenhut dated July 27, 1976 concerning reproductions of viewgraphs used in the Westinghouse presentation to the NRC during the meeting on July 27, 1976 on the subject of Westinghouse Reload Safety Evaluation Methodology.

This information enables Westinghouse to:

(a) Justify the design for the reload core

(b) Assist its customers to obtain licenses

(c) Meet contractual requirements

(d) Provide greater flexibility to customers assuring them of safe and reliable operation.

Further, this information has substantial commercial value as follows:

(a) Westinghouse sells the use of the information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse uses the information to perform and justify analyses which are sold to customers.

(c) Westinghouse uses the information to sell nuclear fuel and related services to its customers.

Public disclosure of this information is likely to cause substantial harm to the competitive position of Westinghouse in selling nuclear fuel and related services.

Westinghouse retains a marketing advantage by virtue of the knowledge, experience and competence it has gained through long involvement and considerable investment in all aspects of the nuclear power generation industry. In particular Westinghouse has developed a unique understanding of the factors and parameters which are variable in the process of design of nuclear fuel and which do affect the in service performance of the fuel and its suitability for the purpose for which it was provided.

In all cases that purpose is to generate energy in a safe and efficient manner while enabling the operating nuclear generating station to meet all regulatory requirements affected by the core loading of nuclear fuel. Confidence in being able to accomplish this comes from the exercise of judgement based on experience.

Thus, the essence of the competitive advantage in this field lies in an understanding of which analyses should be performed and in the methods and models used to perform these analyses. A substantial part of this competitive advantage will be lost if the competitors of Westinghouse are able to use the results of the Westinghouse experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design and licensing of a similar product.

This information is a product of Westinghouse design technology. As such, it is broadly applicable to the sale and licensing of fuel in pressurized water reactors. The development of this information is the result of many years of Westinghouse effort and the expenditure of a considerable sum of money. In order for competitors of Westinghouse to duplicate this process

would require the investment of substantially the same amount of effort and expertise that Westinghouse possesses and which was acquired over a period of more than fifteen years and by the investment of millions of dollars.

Further the deponent sayeth not.

APPENDIX 5
RETRAN-01 to RETRAN-02 Transition Submittal

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

November 19, 1985

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Cecil O. Thomas, Chief
Standardization and Special
Projects Branch
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 85-753
E&C/NAS:asp
Docket Nos.: 50-280
50-281
50-338
50-339
License Nos.: DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY AND NORTH ANNA POWER STATIONS
REACTOR SYSTEM TRANSIENT ANALYSES

In our letter to you of April 14, 1981, Serial No. 215, we transmitted our Topical Report VEP-FRD-41, "Vepco Reactor System Transient Analysis Using the RETRAN Computer Code." The report, which was provided for review by your staff, describes the system transient analysis capability which Vepco is using in support of core reloads, and other operational or design changes at our nuclear units. Following a request for supplemental information, to which Vepco responded with letters dated February 27, 1984, July 12, 1984 and August 24, 1984, the staff issued a letter approving the report for referencing in license applications on April 11, 1985.

In the Safety Evaluation Report (SER) accompanying this approval, the staff referred to Vepco's RETRAN capability "for performing transient analyses using the RETRAN01/MOD03 Computer Code." Since no RETRAN02 analyses were presented in the topical report or the supplemental submittals, no reference to Vepco's use of RETRAN02 was made in the SER. Vepco has informally discussed its desire to have the SER for VEP-FRD-41 extended to RETRAN02/MOD03 applications informally with your staff (Mr. J. Guttmann, USNRC Reactor Systems Branch and Mr. D. Moran, USNRC Standardizations and Special Projects Branch) on April 2, 1985. Based on that discussion, we are submitting for your review an additional set of analyses performed by Vepco with the models documented in VEP-FRD-41 and the supplements discussed above. These analyses provide comparisons of results obtained for identical transients using RETRAN01 (the code version used to perform the analyses presented in VEP-FRD-41) and RETRAN02. As discussed in the attachment, the results are very nearly identical except in the area of nonequilibrium pressurizer behavior, where substantial improvements were made in the solution scheme in RETRAN02.

8511250319 851119
PDR ADOCK 05000280
P PDR

A001
1/20

We are requesting, based on these results, approval to reference VEP-FRD-41A and the associated SER in future licensing applications involving Surry and North Anna Power Stations where analyses have been performed using the RETRAN02 Computer Code. In order to support upcoming licensing submittals, we request your approval by February 1986.

Very truly yours,

W. L. Stewart

Attachment

cc: Dr. J. Nelson Grace
Regional Administrator
Region II

Mr. Harold Bernard
Standardization and Special Projects Branch

Mr. J. L. Carter
Reactor Systems Branch

Mr. J. Guttmann
Reactor Systems Branch

Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Mr. Edward J. Butcher, Acting Chief
Operating Reactors Branch No. 3
Division of Licensing

Mr. D. J. Burke
NRC Resident Inspector
Surry Power Station

Mr. M. W. Branch
NRC Resident Inspector
North Anna Power Station

ATTACHMENT 1

COMPARISON OF RETRAN01 AND RETRAN02 COMPUTER CODE RESULTS

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LIST OF FIGURES	4
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3.0 RESULTS OF ANALYSIS	8
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3a.	Midcore Fuel Temp	16
4a.	Heat Extraction	17
5a.	Steam Pressure	18
6a.	Inlet Temperature	19
7a.	Pressurizer Pressure	20
8a.	Loop Flow	21

Turbine Trip No Rx Trip

No	Title	Page
1b.	Midcore Heat Flux	22
2b.	Nuclear Power	23
3b.	Midcore Fuel Temp	24
4b.	Heat Extraction	25
5b.	Steam Pressure	26
6b.	Inlet Temperature	27
7b.	Pressurizer Pressure	28
8b.	Loop Flow	29

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2c.	Nuclear Power	31
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4c.	Heat Extraction	33
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6c.	Inlet Temperature	35
7c.	Pressurizer Pressure	36
8c.	Loop Flow	37

1.0 INTRODUCTION

Virginia Electric and Power Company (the Company) has performed analyses to compare the results calculated by RETRAN01 and RETRAN02, two versions of the RETRAN computer code which have been released by the Electric Power Research Institute (EPRI). Topical reports related to RETRAN have been submitted by the Utility Group for Regulatory Application and have been accepted by the NRC (Reference 1). The NRC approved the Company's licensing topical report VEP-FRD-41A,"Reactor System Transient Analyses using the RETRAN Computer Code", on April 11, 1985 (Reference 2). The analyses presented in VEP-FRD-41A were performed using RETRAN01. Since the Company intends to use RETRAN02 for its licensing analysis, the NRC requested comparative analyses using RETRAN01 and RETRAN02 to support extension to their review and approval to RETRAN02 (Reference 3). The three transients that were selected for this comparative study were:

1. Reactor Trip
2. Turbine Trip without Reactor Trip
3. Complete Loss of Flow

These transients demonstrate the significant features of the models (nonequilibrium pressurizer behaviour, point kinetics response, response to large flow variations, etc.) but are straightforward enough that differences in paramenter trends are readily identified and assessed. Section 2 describes the models used for the analysis. Sections 3 and 4 contain the results and conclusions.

2. DESCRIPTION OF MODEL

Two RETRAN input decks, a North Anna Single Loop Model compatible with RETRAN01 and a similar model compatible with RETRAN02, with nineteen control volumes and twenty-nine flow junctions were used for this analysis. A general description of these models was provided in References 4-7. RETRAN02 is an extension of RETRAN01 containing additional user conveniences, the ability to optionally model additional phenomena and upgrading of some of the RETRAN01 models. RETRAN02 can be used with the same options available in RETRAN01 with the exception of the following changes which represent an upgrade of the RETRAN01 models:

1. A revised solution technique for the nonequilibrium pressurizer model.

2. Analytical expressions for water properties (as opposed to a table).

3. The use of junction flow and fluid properties for the wall friction calculation.

Only the minimum changes required to convert the RETRAN01 data deck to RETRAN02 were made. Initial conditions for all transients are shown in Table 1.

3. RESULTS OF ANALYSIS

Comparison of the time zero edits shows that the two versions of RETRAN calculate steady state initialization parameters which match to within less than 1%. The transient results are described below.

a. Reactor Trip

The reactor was tripped at 0.0 second and the transient was executed for 10 seconds. Figures 1a through 8a show that the results of the two calculations are essentially identical except for pressurizer pressure. The difference in pressurizer pressure is due to the revised nonequilibrium pressurizer model solution technique. The significant events during the reactor trip transient are listed in Table 2.

b. Turbine Trip

The turbine was tripped at 0.0 second and the transient was executed for 10 seconds. Again figures 1b through 8b show that the results of the two calculations are essentially identical except for pressurizer pressure. The pressurizer pressure increases more rapidly during the transient in RETRAN02 than in RETRAN01, due to the revised nonequilibrium pressurizer model solution technique. The significant events during the turbine trip are listed in Table 3.

c. Complete Loss of Flow

The pumps were tripped at 0.0 second and the transient was executed for 10 seconds. Figures 1c through 8c show that the results of the two calculations are identical except pressurizer pressure. The primary coastdown flow rates calculated by the two versions of the code are essentially identical. The significant events occuring during the loss of flow transient are listed in Table 4.

4.0 CONCLUSIONS

The results of the three transients analyzed above using RETRAN01 and RETRAN02 show that the two codes produce essentially identical results except the primary side pressure calculation. The secondary side pressures predicted by the two codes are essentially identical. The following conclusions can be reached:

1. Steady state calculations show less than 1% difference in such parameters as temperatures, pressures and enthalpies.

2. RETRAN01 and RETRAN02 predicted essentially identical flow coastdown for the loss of flow transient using the same model and initial conditions.

3. RETRAN02 predicts larger and faster changes in the primary side pressure than RETRAN01. This is primarily due to the revised solution technique for the nonequilibrium pressurizer model.

5.0 REFERENCES

1. Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, 'RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems,' and EPRI NP-1850-CCM, 'RETRAN02-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems,'" September 4, 1984.

2. Letter from C. O. Thomas (NRC) to W. L.Stewart (Vepco), Acceptance for Referring of Licensing Topical Report VEP-FRD-41, "Vepco Reactor System Transient Analysis Using the RETRAN Computer Code", April 11, 1985.

3. Conference between R. M. Berryman, K. L. Basehore and N. A. Smith (Vepco) and Messrs J. Guttmann and Chiu Liang, USNRC Reactor Systems Branch and D. Moran, USNRC Standardization and Special Projects Branch, Bethesda, Md., April 11, 1985.

4. Vepco Topical Report, VEP-FRD-41A, "Reactor System Transient Analyses Using the RETRAN Computer Code", submitted by letter from W. L. Stewart (Vepco) to H. L. Thompson, Jr. (NRC), Serial No. 85-77, July 3, 1985.

5. Letter from W. L. Stewart (Vepco) to H. R. Denton (NRC), "Vepco Reactor System Transient Analyses", Serial No. 060, February 27, 1984.

6. Letter from W. L. Stewart (Vepco) to H. R. Denton (NRC), "Vepco Reactor System Transient Analyses", Serial No. 376, July 12, 1984.

7. Letter from W. L. Stewart (Vepco) to H. R. Denton (NRC), "Vepco Reactor System Transient Analyses", Serial No. 376A, August 24,1984.

Table 1. Initial Conditions for Steady-State Operation

Parameters	Value	Units
Core Power	2830.50	Mwt
Total Loop Flow	104.25E+6	lb/hr
Pressurizer Pressure	2220	psia
Enthalpy at Lower Plenum	551.20	btu/lb
Steam/Feed Flow	12.464E+6	lb/hr
Steam Pressure	856.0	psia
Feedwater Enthalpy	418.50	btu/lb

Table 2. Significant Events During Reactor Trip Transient

Event	Setpoint Value	Time(seconds) RETRAN01	RETRAN02
Steady State Initialization	N/A	0.0	0.0
Steady State Operation	N/A	0.0	0.0
Reactor Trip	N/A	0.0	0.0
Turbine Trip	N/A	2.005	2.005
Pressurizer Heaters on	+10% Controller Span	3.445	3.429
Atmospheric Relief Valves Open	1050 psia	9.440	9.379

John Harrell

05/04/2004 11:29 AM

To: Delbert Horn/NUC/VANCPOWER@VANCPOWER
cc: Cary Laroe/NUC/VANCPOWER@VANCPOWER, Dave Bucheit/NUC/VANCPOWER@VANCPOWER, Hink Barker/NUC/VANCPOWER@VANCPOWER, Patricia Guinan/NUC/VANCPOWER@VANCPOWER, Tom Brookmire/NUC/VANCPOWER@VANCPOWER, Tom Hook/NUC/VANCPOWER@VANCPOWER, John R Guerci/MILLSTONE/VANCPOWER@VANCPOWER, Larry D Brooks/MILLSTONE/VANCPOWER@VANCPOWER
Subject: Re: NAF Department Directory Rights Simplification

Delbert,

Global access to all NAF folders does not present any problems, provided the individual personnel directories under NSA\TaskTrak keep their same permissions. (I need to have access to all folders, and each NSA staff member needs to have access only to his/her own.) If other Innsbrook supervisors have similarly protected "personnel" folders, they should let you know the specific protections that need to remain. My guess is that some of the groups identified in your "step 3" (below) may need to be kept.

Let's get together ASAP to talk about the details. Others, please chime in if you have protections that need to be kept.

Thanks

jrh

P.S. Since Millstone is not yet "linked and synced", I have postponed my trip previously scheduled for May 10.

Delbert Horn

Delbert Horn

05/03/04 01:46 PM

To: Cary Laroe/NUC/VANCPOWER@VANCPOWER, John Harrell/NUC/VANCPOWER@VANCPOWER, Dave Bucheit/NUC/VANCPOWER@VANCPOWER, Tom Brookmire/NUC/VANCPOWER@VANCPOWER, Hink Barker/NUC/VANCPOWER@VANCPOWER, Tom Hook/NUC/VANCPOWER@VANCPOWER
cc: Patricia Guinan/NUC/VANCPOWER@VANCPOWER
Subject: NAF Department Directory Rights Simplification

Gentlemen,

The upcoming NAF realignment has emphasized the need to re-work the network group rights to the NAF department directory.

Currently, there are many overlapping network groups that provide segmented access to different directories (e.g, an FAI group only provides access to the FAI directory and the NSF directory.)

MPS personnel are coming in at different levels of access, too (some get everything, others only get one directory.)

I'm sure several people have wanted to move/reorganize directories under NAF before, but the tangled group rights have made that risky impractical.

I would like to propose a change sequence and would like your thoughts:

1. Make sure everyone in NAF is a member of the "I 32 NAF" group.
2. Apply this group to the highest level directory on down to the lowest, e.g., \\INNCNW03\SYS2\DEPARTMENT\NAF and all subdirectories (with the exception of the personnel directories under the NSA directory)
3. Delete all existing lower-level network groups: I 32 Anomaly, I 32 FABDATA, I 32 FAI, I 32 FPA, I 32 PFA MGR, I 32 MCNP, I 32 NCD Approval, I 32 NSF, and I 32 PSA.
4. Formulate and implement a new subdirectory structure to reflect your new organization, while everyone still has rights to everything.
5. If desired, implement new network groups and apply them to the newly organized subdirectories.

With many engineers needing access to many other directories, I feel opening up access would help more than it would hurt. If you feel strongly about limiting access to certain areas, please let me know.

With your approval, steps 1-3 could happen quickly, we can take our time with steps 4 and 5. This effort would greatly simplify NAF user directory access as well as IT maintenance for existing and new NAF personnel.

Delbert L. Horn
Business Systems Specialist - IT Generation
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Innsbrook Technical Center, 1SW
5000 Dominion Boulevard
Glen Allen, VA 23060
Phone: (804) 273-2779
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Table 3. Significant Events During Turbine Trip Transient

Event	Setpoint Value	Time(seconds) RETRAN01	RETRAN02
Steady State Initialization	N/A	0.0	0.0
Steady State Operation	N/A	0.0	0.0
Turbine Trip	N/A	0.0	0.0
PORV Open #2	50% of Controller Span	4.765	4.563
Atmospheric Relief Valves Open	1050 psia	6.610	6.552
PORV Open #1	2350 psia	6.695	6.393

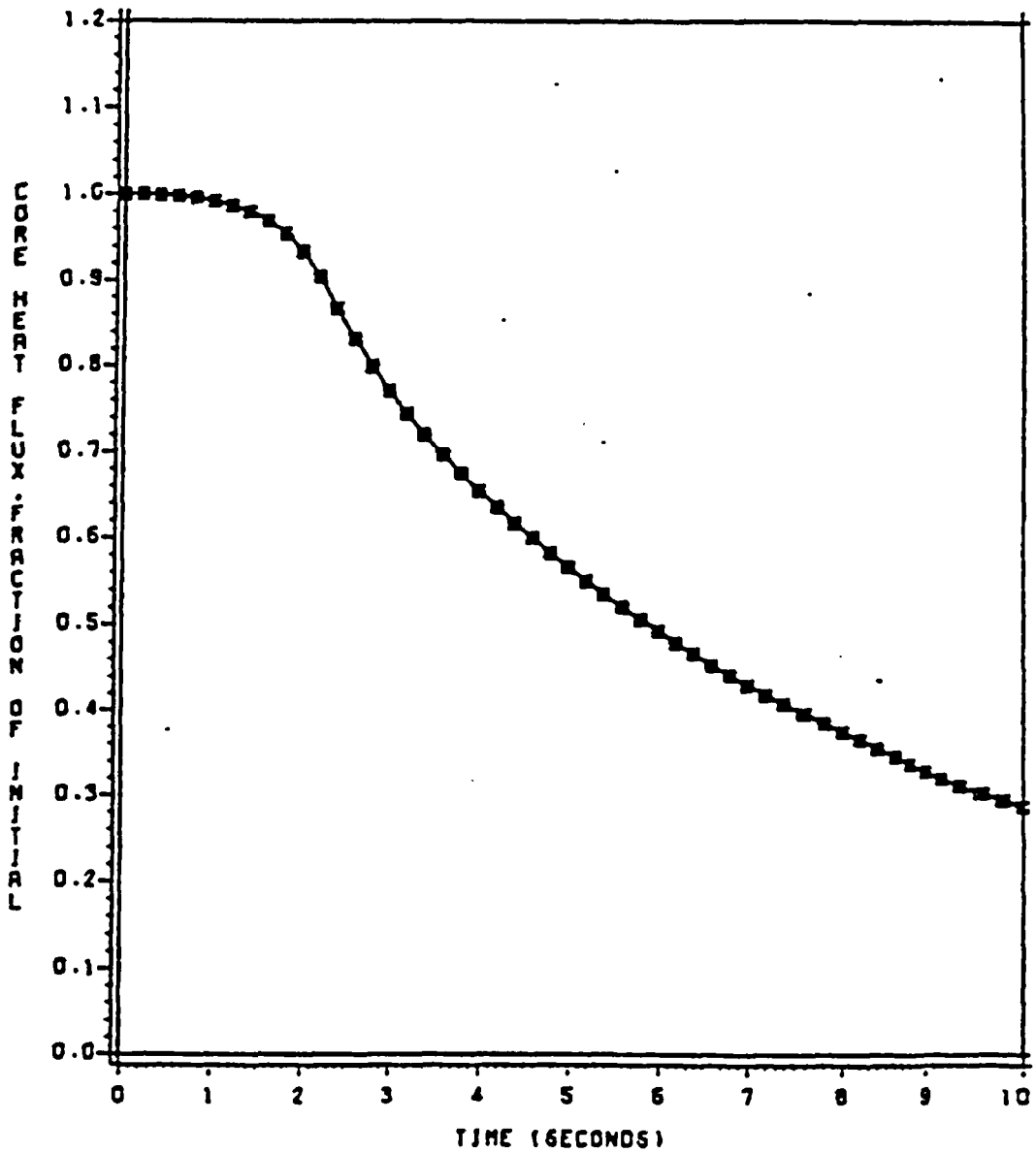
Table 4. Significant Events During Loss of Flow Transient

Event	Setpoint Value	Time(seconds) RETRAN01	RETRAN02
Steady State Initialization	N/A	0.0	0.0
Steady State Operation	N/A	0.0	0.0
Pump Trip	N/A	0.0	0.0
Low Flow Trip	25194.0 lb/sec	2.140	2.138
Reactor Trip	N/A	3.140	3.138
PORV Open #2	50% of Controller Span	******	4.882
Turbine Trip	N/A	5.145	5.138
Pressurizer Heaters on	-10% of Controller Span	8.770	8.941

FIGURE 1a
REACTOR TRIP

MIDCORE HEAT FLUX

RETRAN01 VS RETRAN02

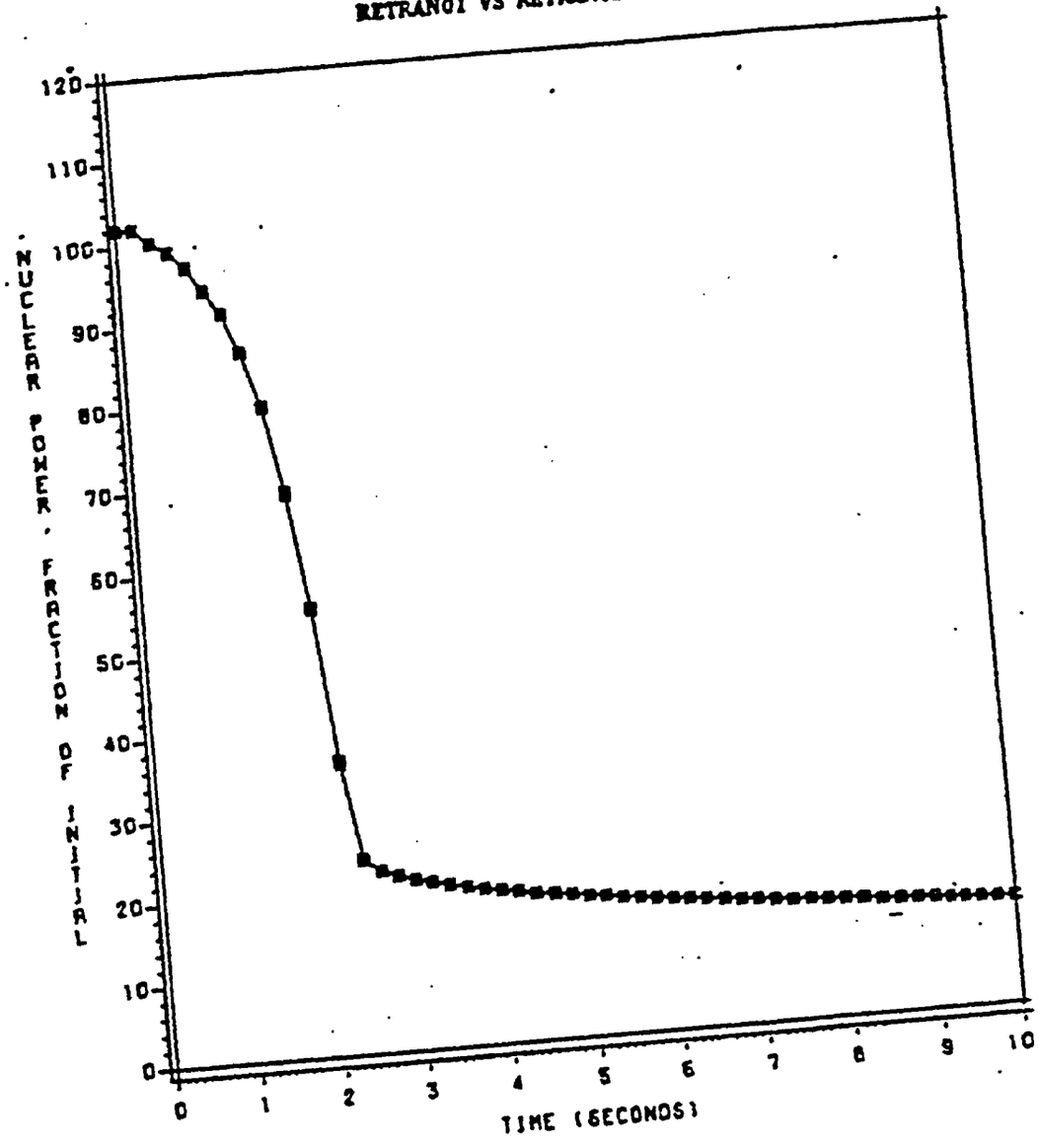


RETRAN01=X
RETRAN02=SQUARE

FIGURE 2a
REACTOR TRIP

NUCLEAR POWER

RETRAN01 VS RETRAN02

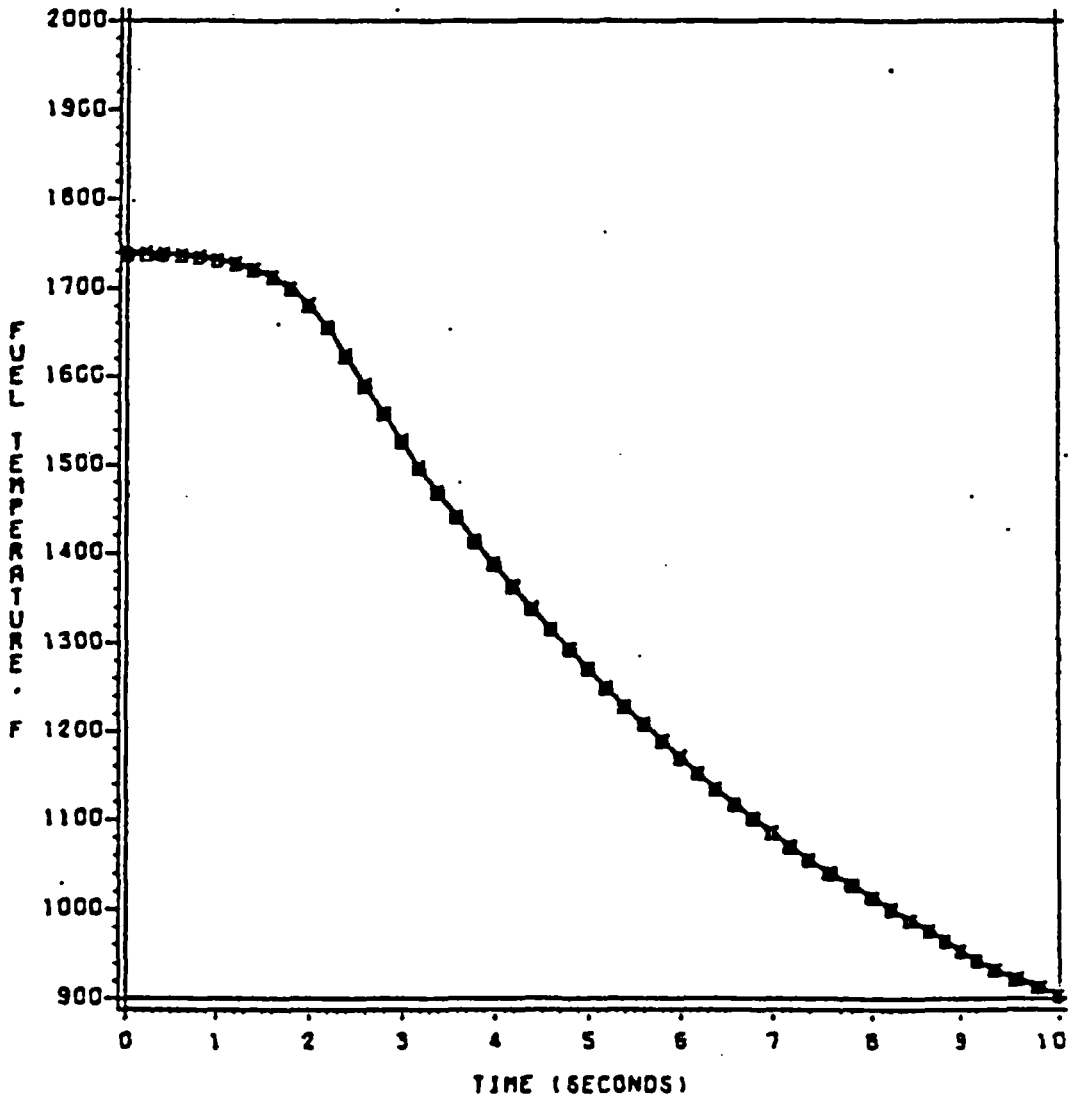


RETRAN01=X
RETRAN02=SQUARE

FIGURE 3a
REACTOR TRIP

MIDCORE FUEL TEMP

RETRAN01 VS RETRAN02

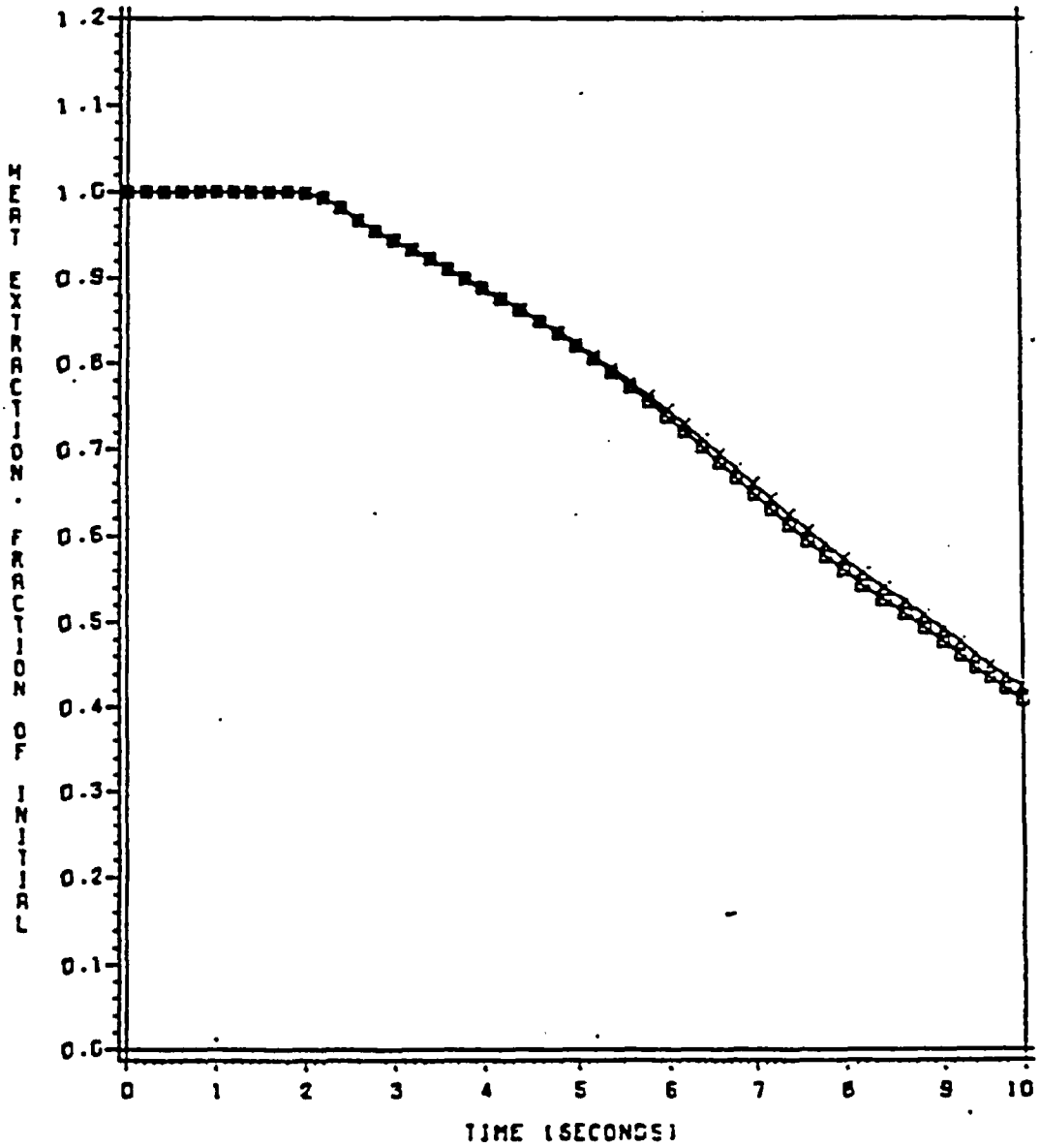


RETRAN01=X
RETRAN02=SQUARE

FIGURE 4a
REACTOR TRIP

SG HEAT EXTRACTION RATE

RETRAN01 VS RETRAN02

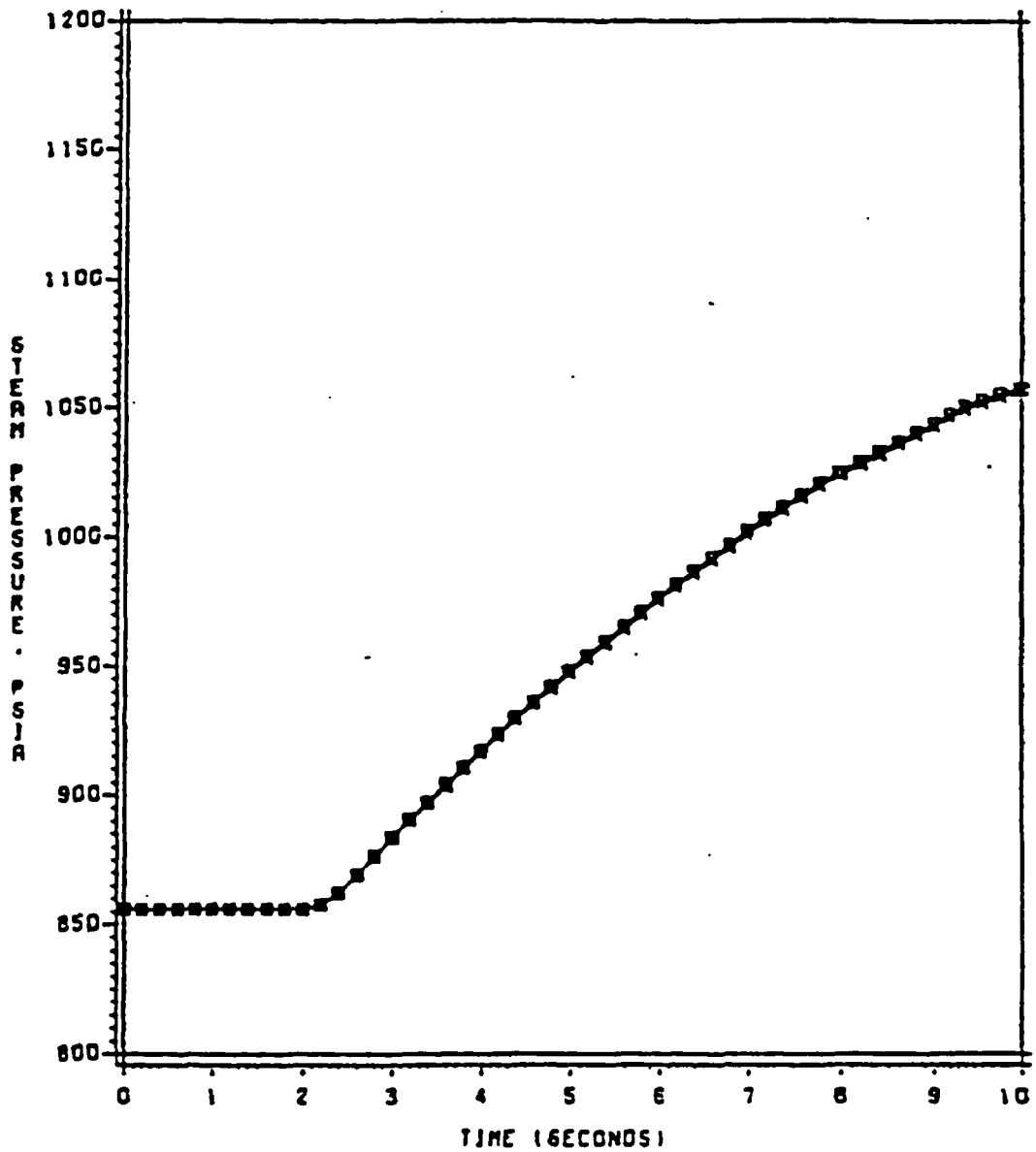


RETRAN01=X
RETRAN02=SQUARE

FIGURE 5a
REACTOR TRIP

STEAM PRESSURE

RETRAN01 VS RETRAN02



RETRAN01=X
RETRAN02=SQUARE

FIGURE 6a
REACTOR TRIP

INLET TEMPERATURE

RETRAN01 VS RETRAN02

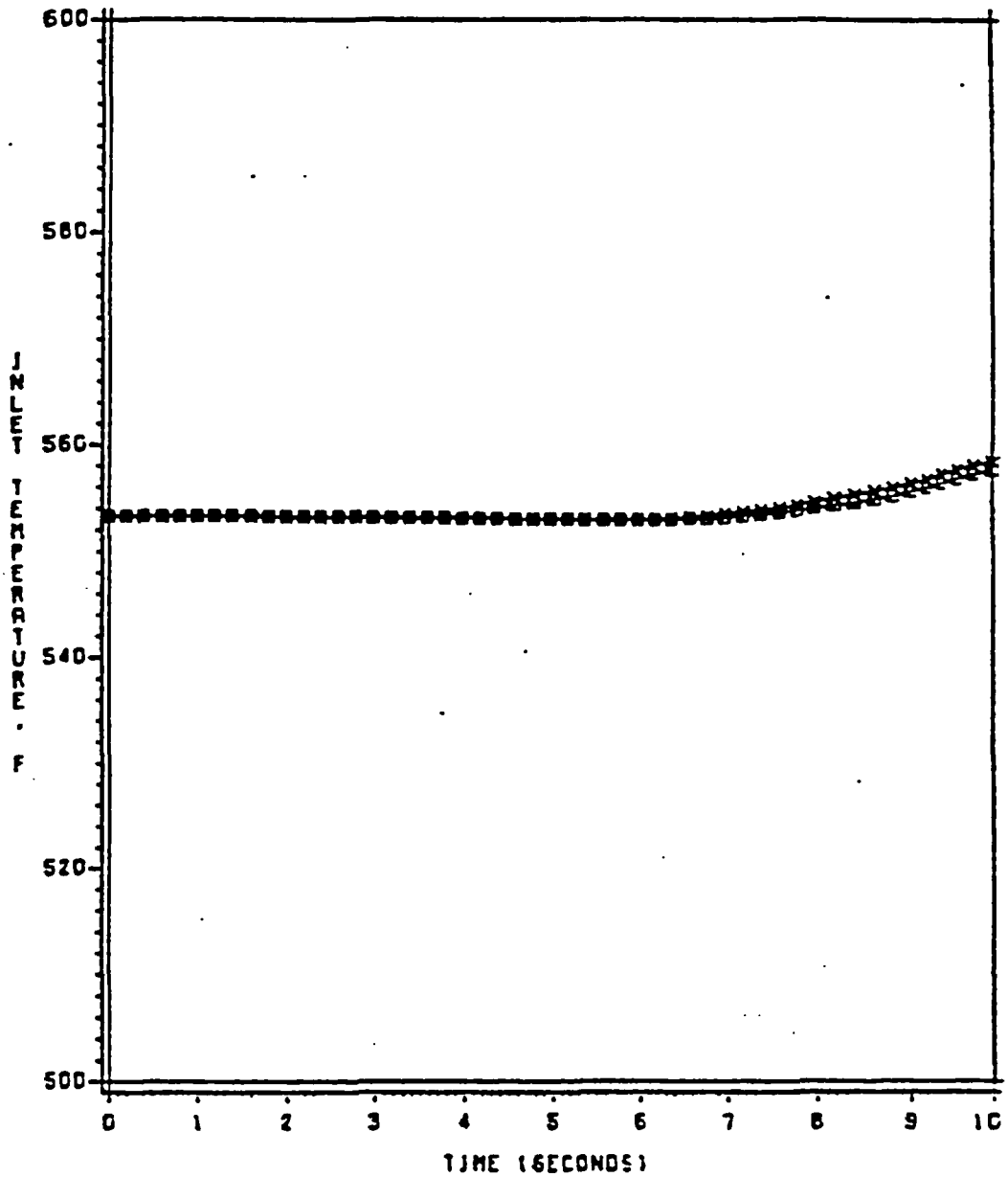
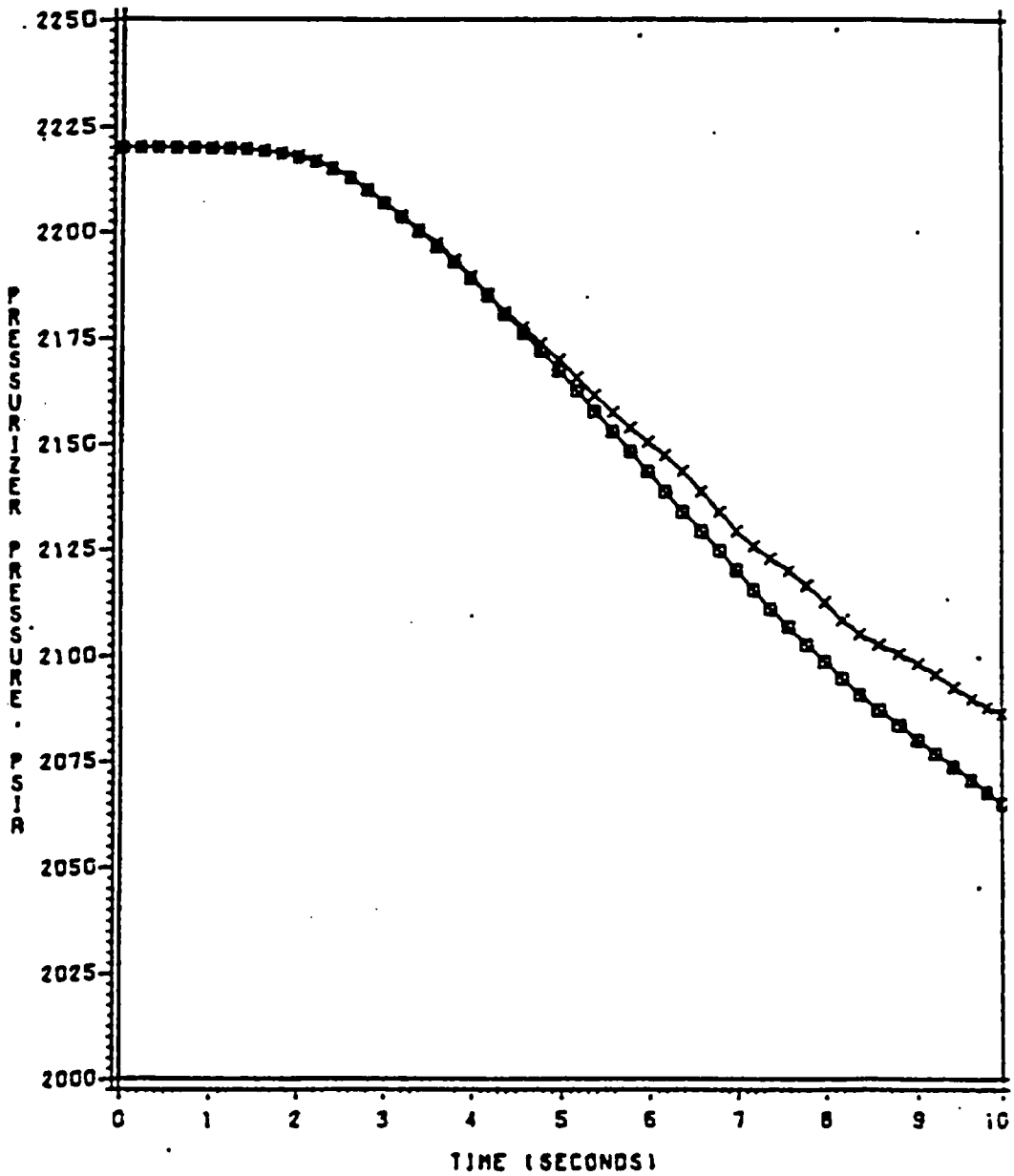


FIGURE 7a
REACTOR TRIP

PRESSURIZER PRESSURE

RETRAN01 VS RETRAN02

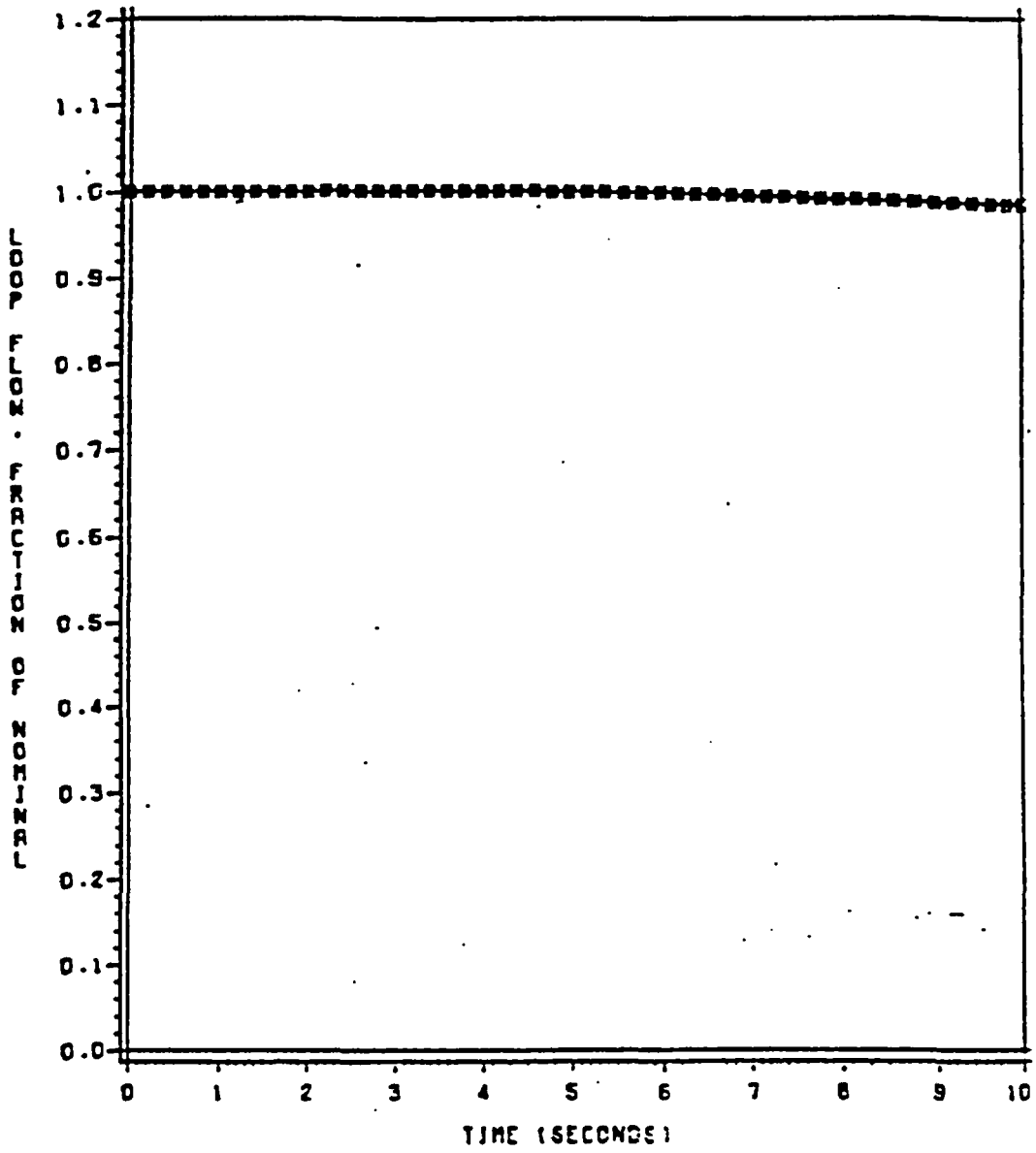


RETRAN01=X
RETRAN02=SQUARE

FIGURE 88
REACTOR TRIP

LOOP FLOW

RETRAN01 VS RETRAN02

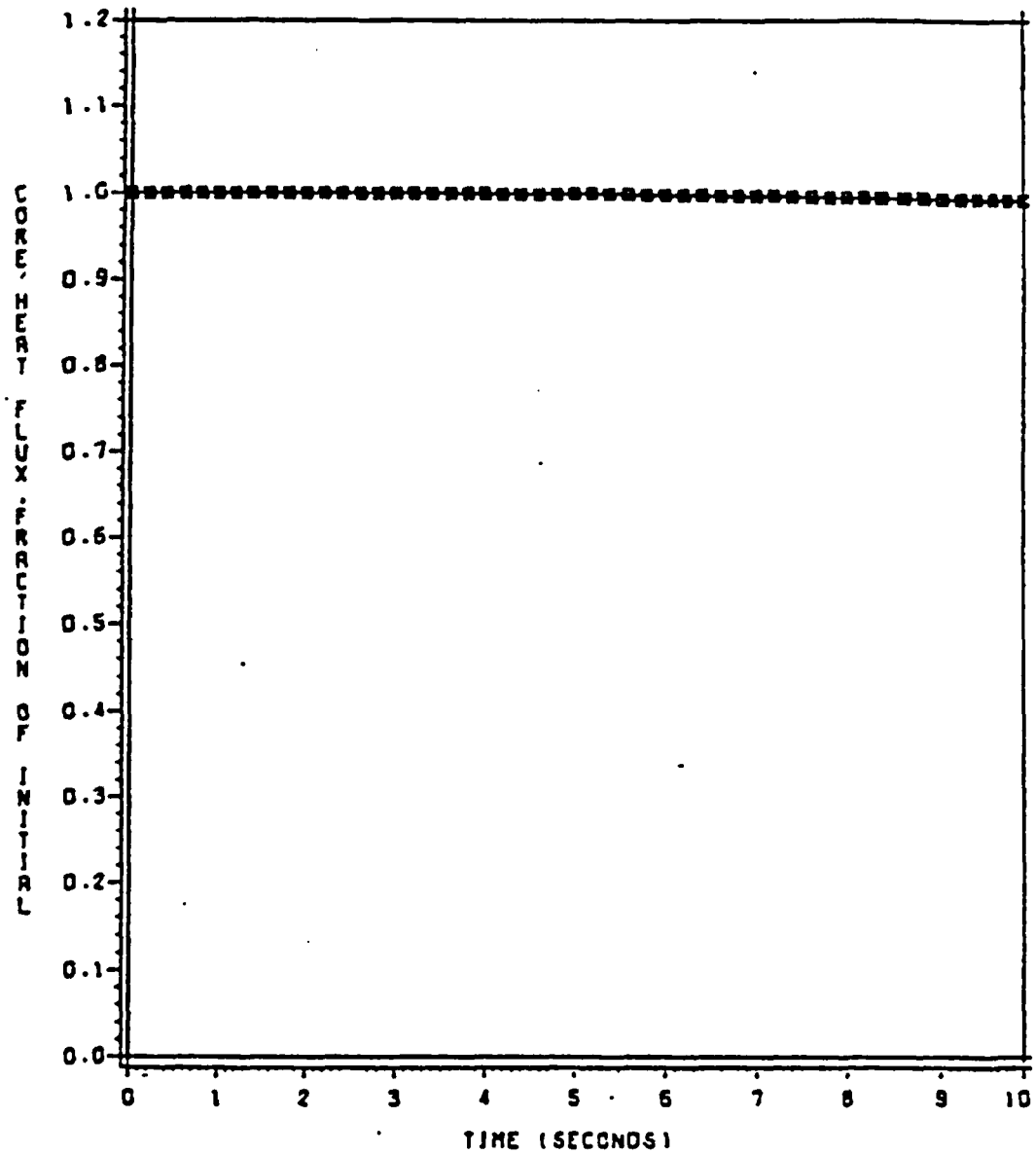


RETRAN01=X
RETRAN02=SQUARE

FIGURE 1b
TURBINE TRIP NO RX TRIP

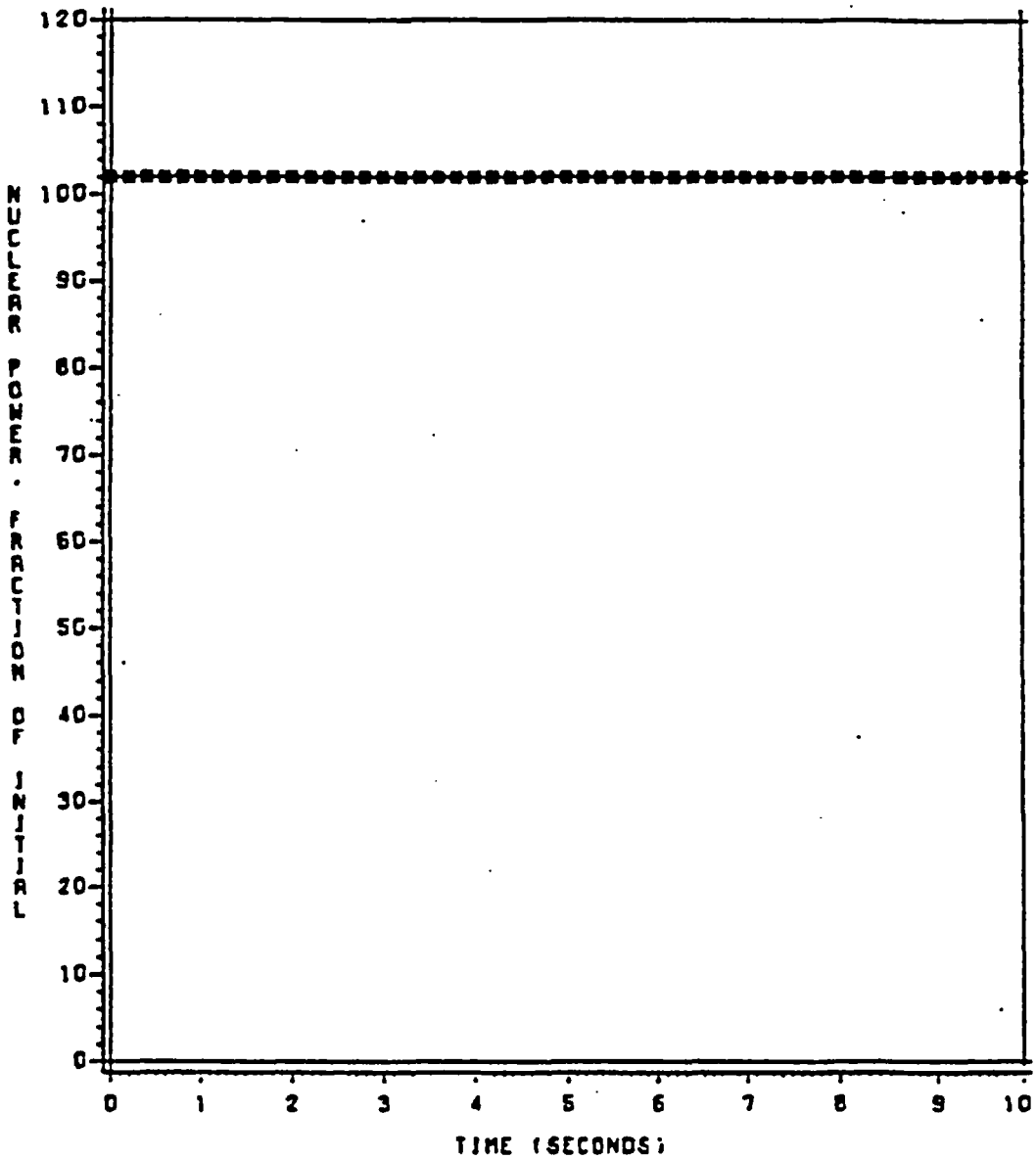
MIDCORE HEAT FLUX

RETRAN01 VS RETRAN02



RETRAN01=X
RETRAN02=SQUARE

FIGURE 2b
TURBINE TRIP NO RX TRIP
NUCLEAR POWER
RETRAN01 VS RETRAN02

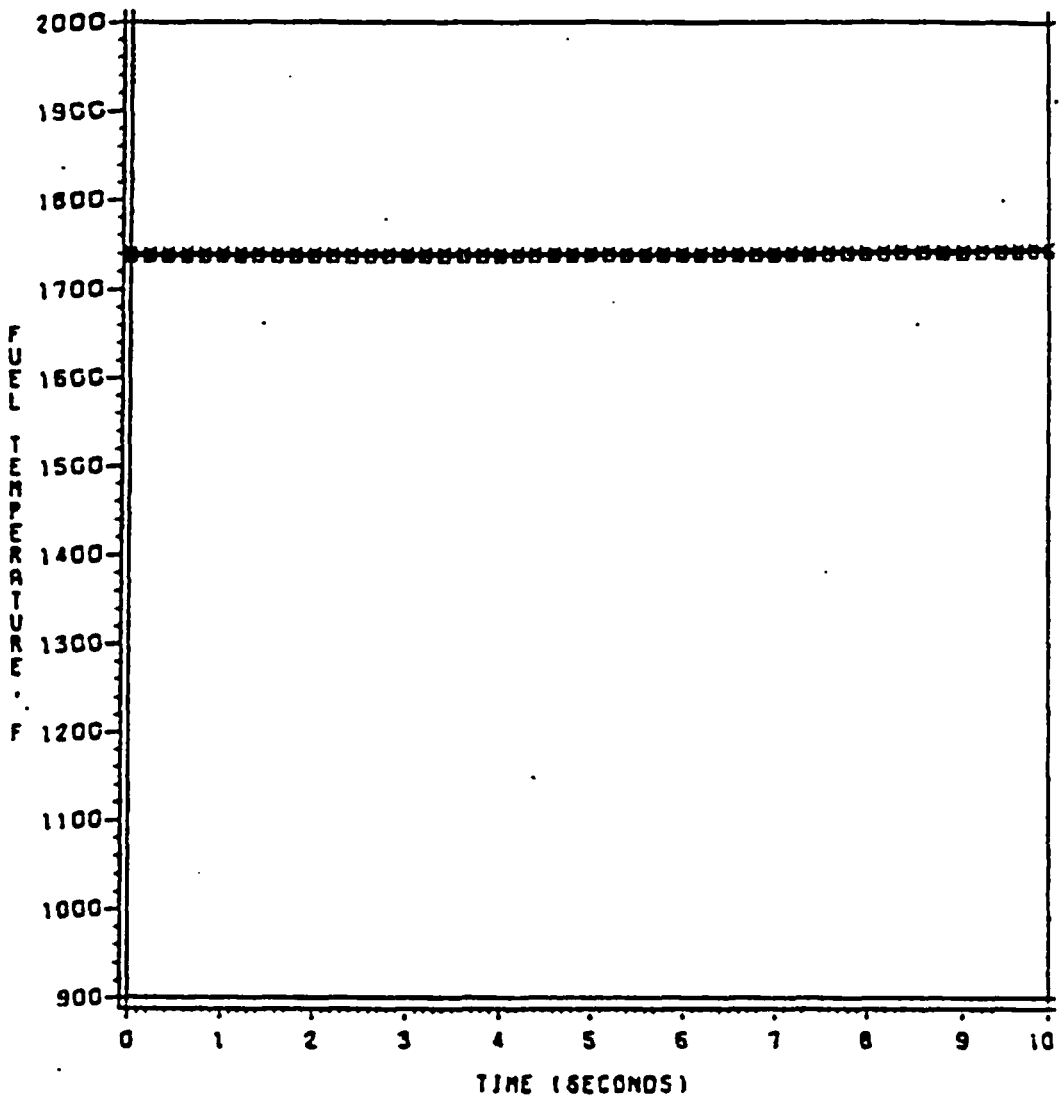


RETRAN01=X
RETRAN02=SQUARE

FIGURE 3b
TURBINE TRIP NO RX TRIP

MIDCORE FUEL TEMP

RETRAN01 VS RETRAN02

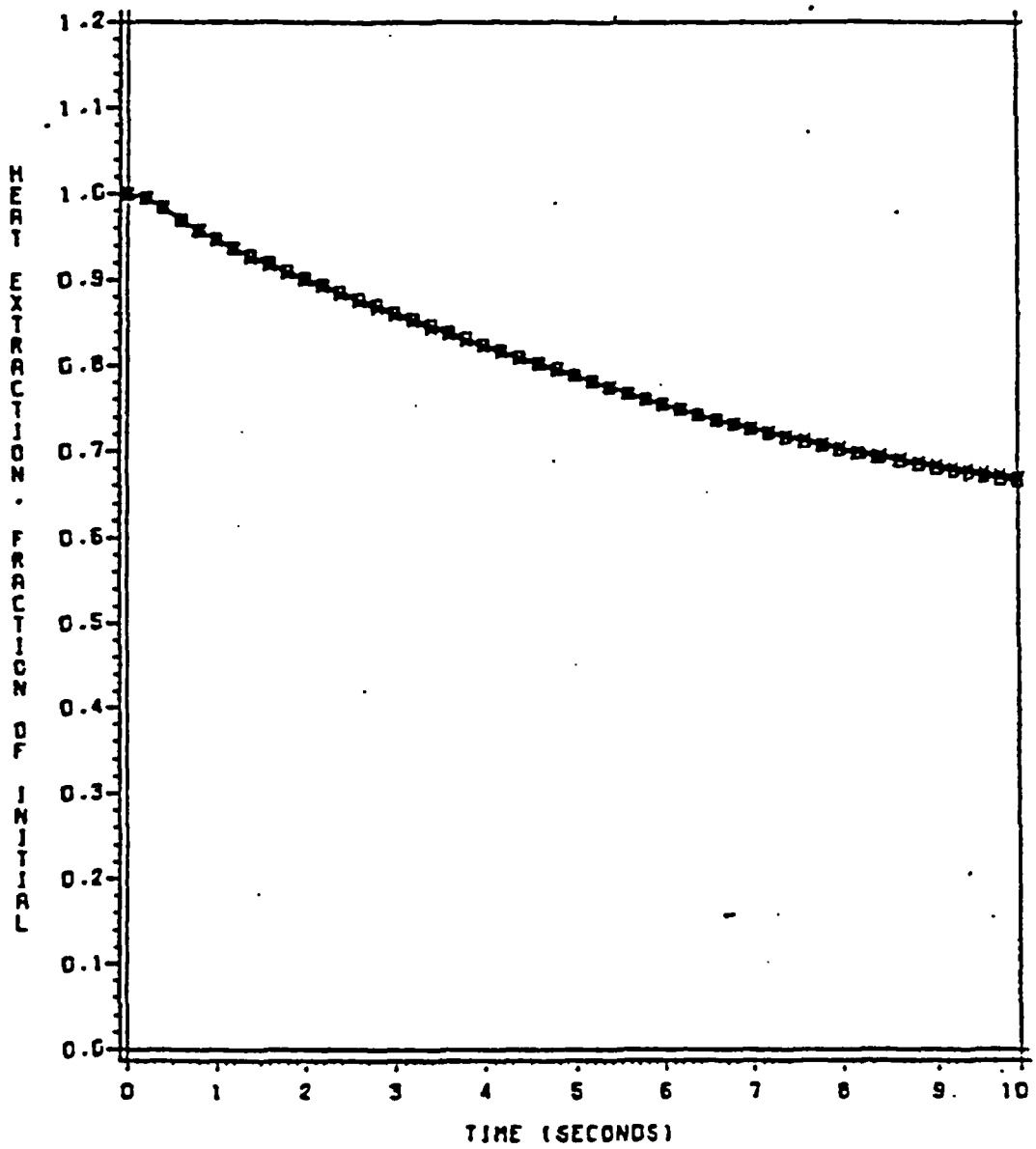


RETRAN01=X
RETRAN02=SQUARE

FIGURE 4b
TURBINE TRIP NO RX TRIP

SG HEAT EXTRACTION RATE

RETRAN01 VS RETRAN02



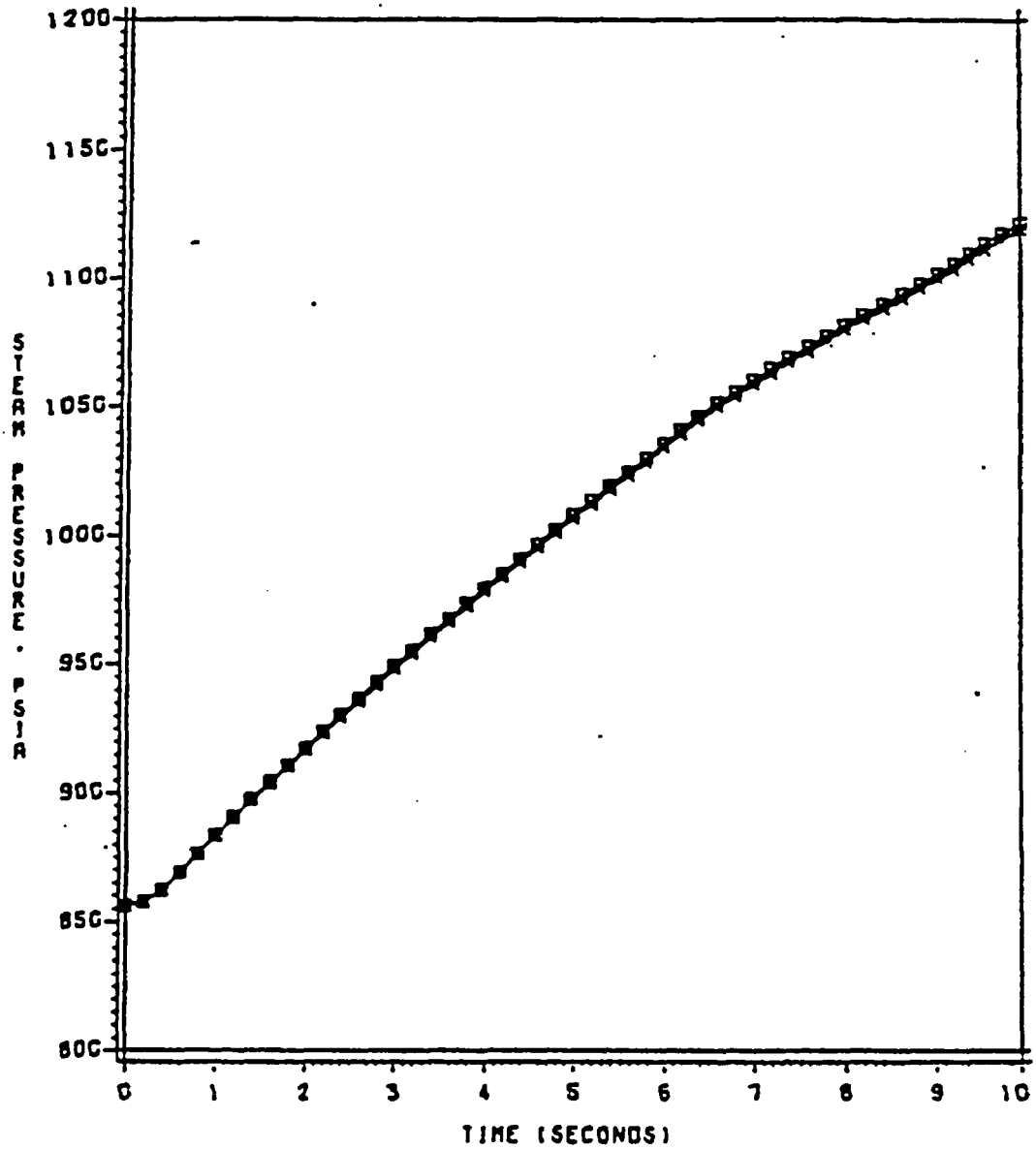
RETRAN01=X
RETRAN02=SQUARE

FIGURE 5b

TURBINE TRIP NO RX TRIP

STEAM PRESSURE

RETRAN01 VS RETRAN02



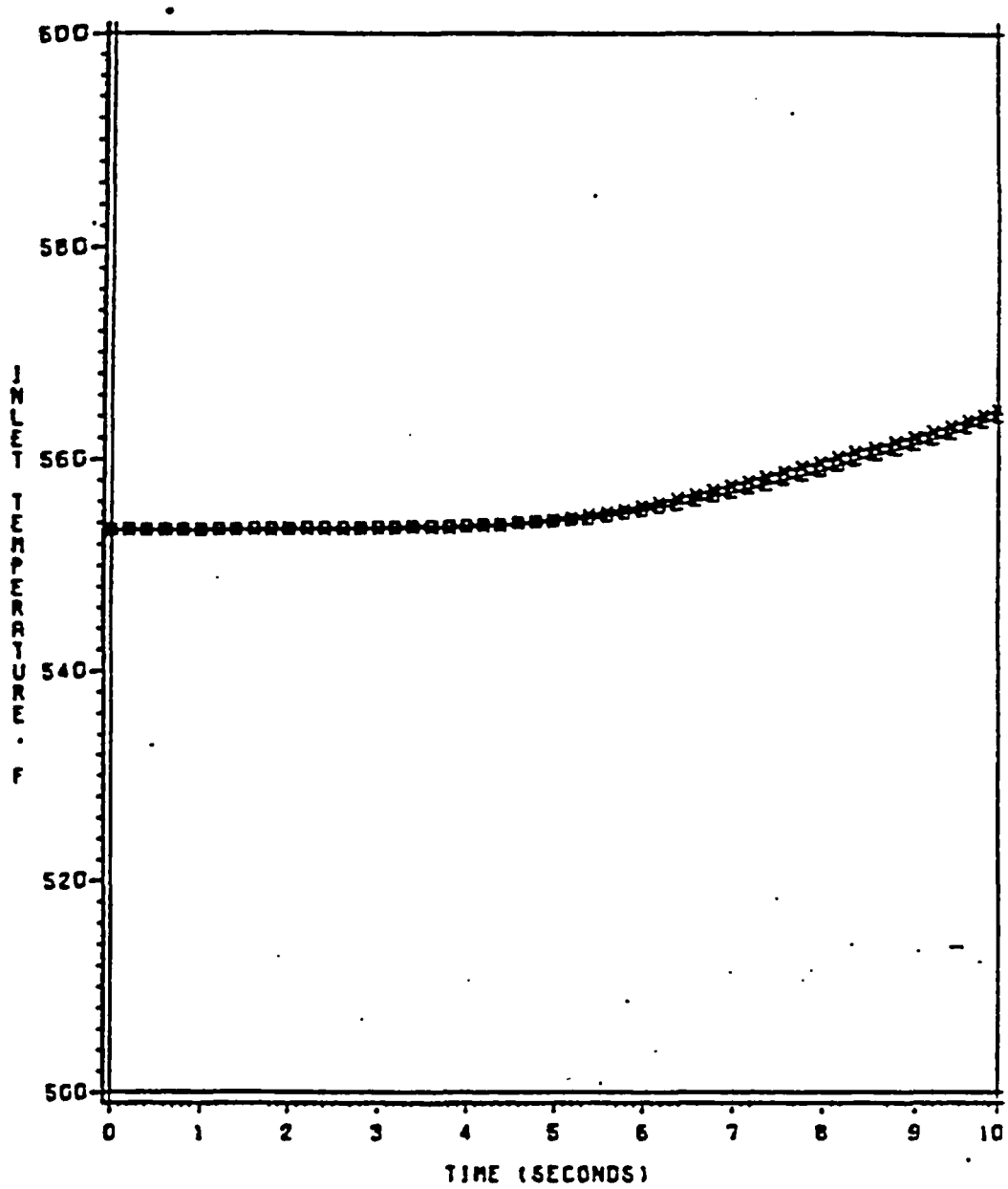
RETRAN01=X
RETRAN02=SQUARE

FIGURE 6b

TURBINE TRIP NO RX TRIP

INLET TEMPERATURE

RETRAN01 VS RETRAN02



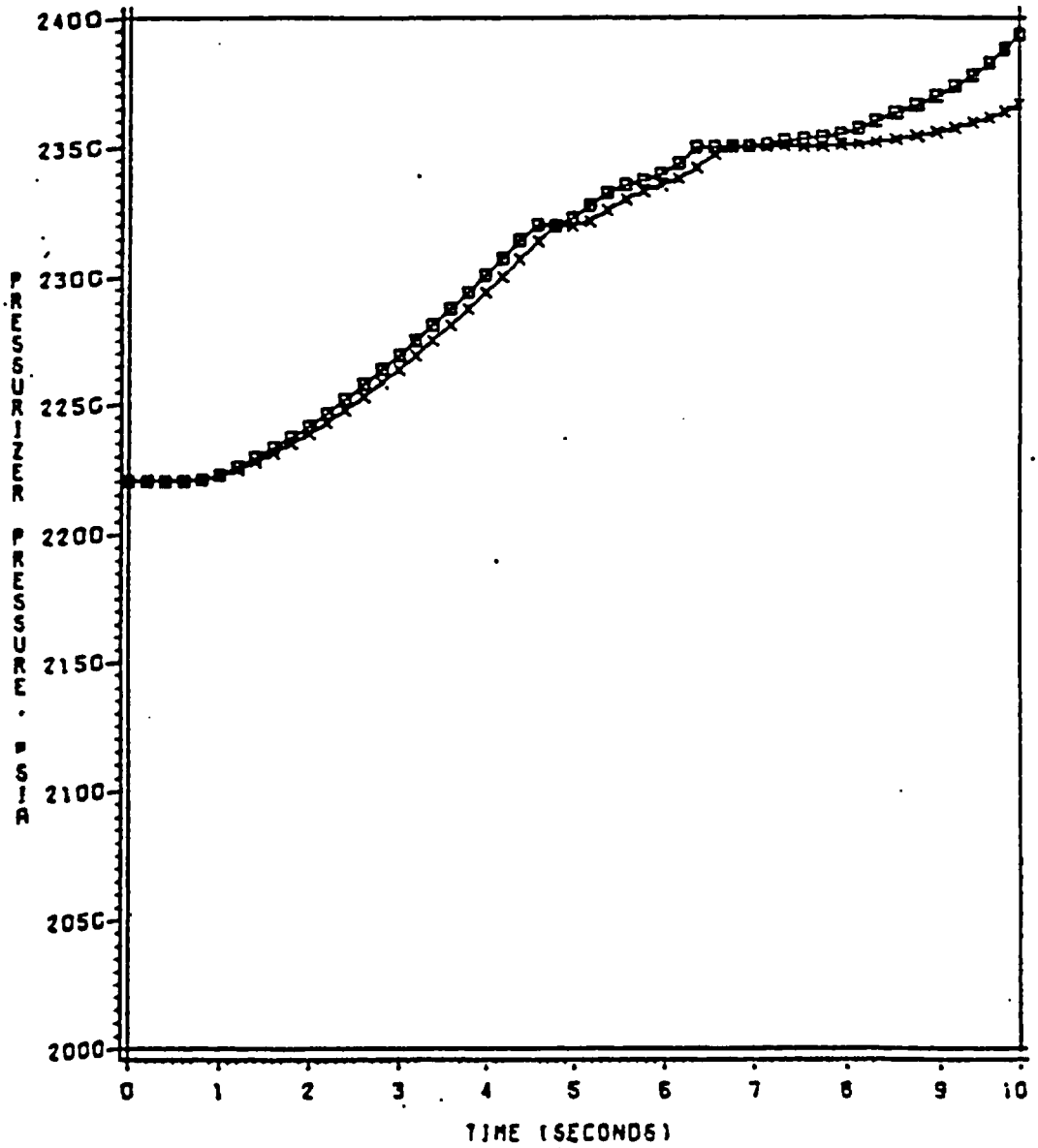
RETRAN01=X
RETRAN02=SQUARE

FIGURE 7b

TURBINE TRIP NO RX TRIP

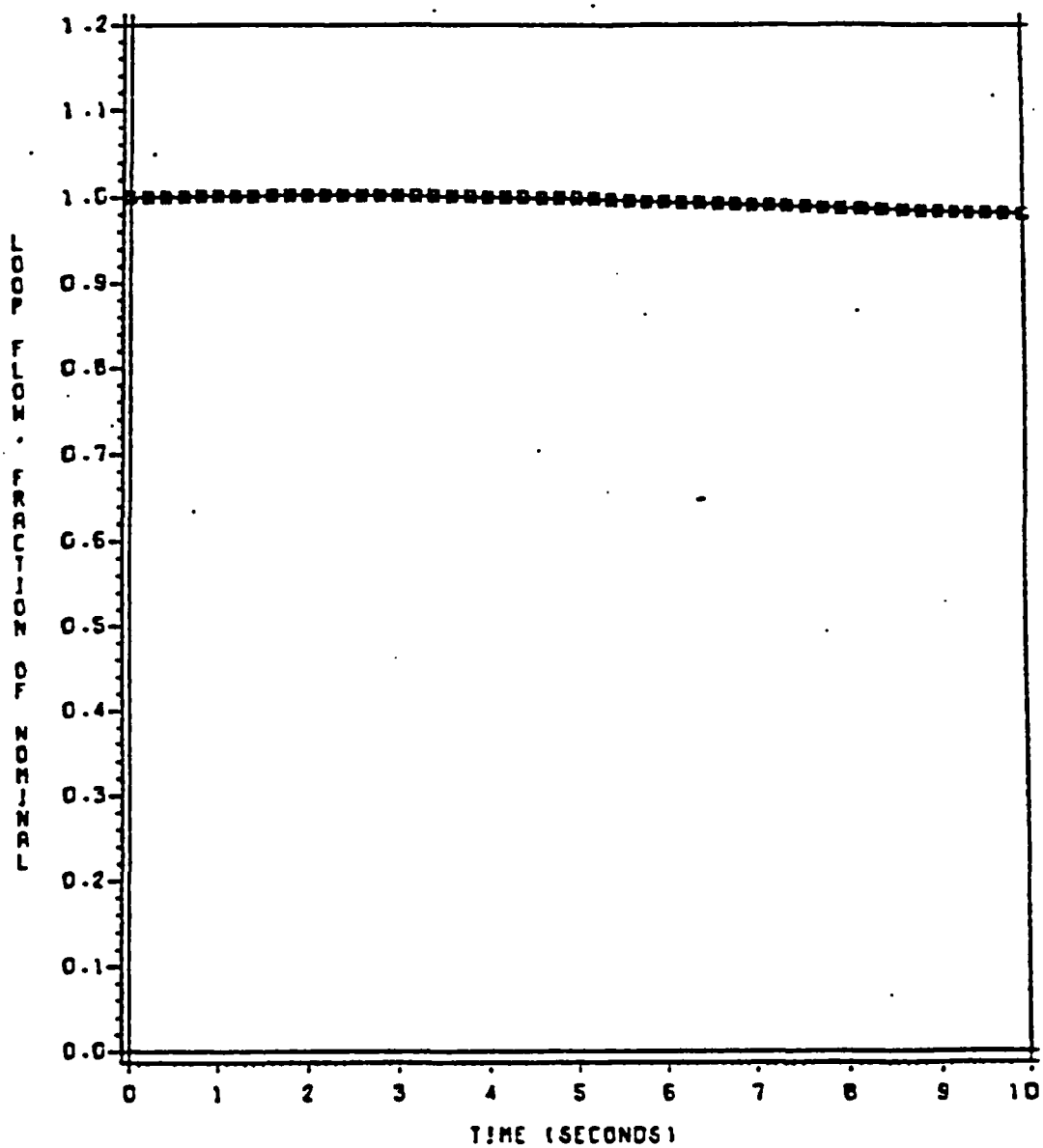
PRESSURIZER PRESSURE

RETRAN01 VS RETRAN02



RETRAN01=X
RETRAN02=SQUARE

FIGURE 8b
TURBINE TRIP NO RX TRIP
LOOP FLOW
RETRAN01 VS RETRAN02

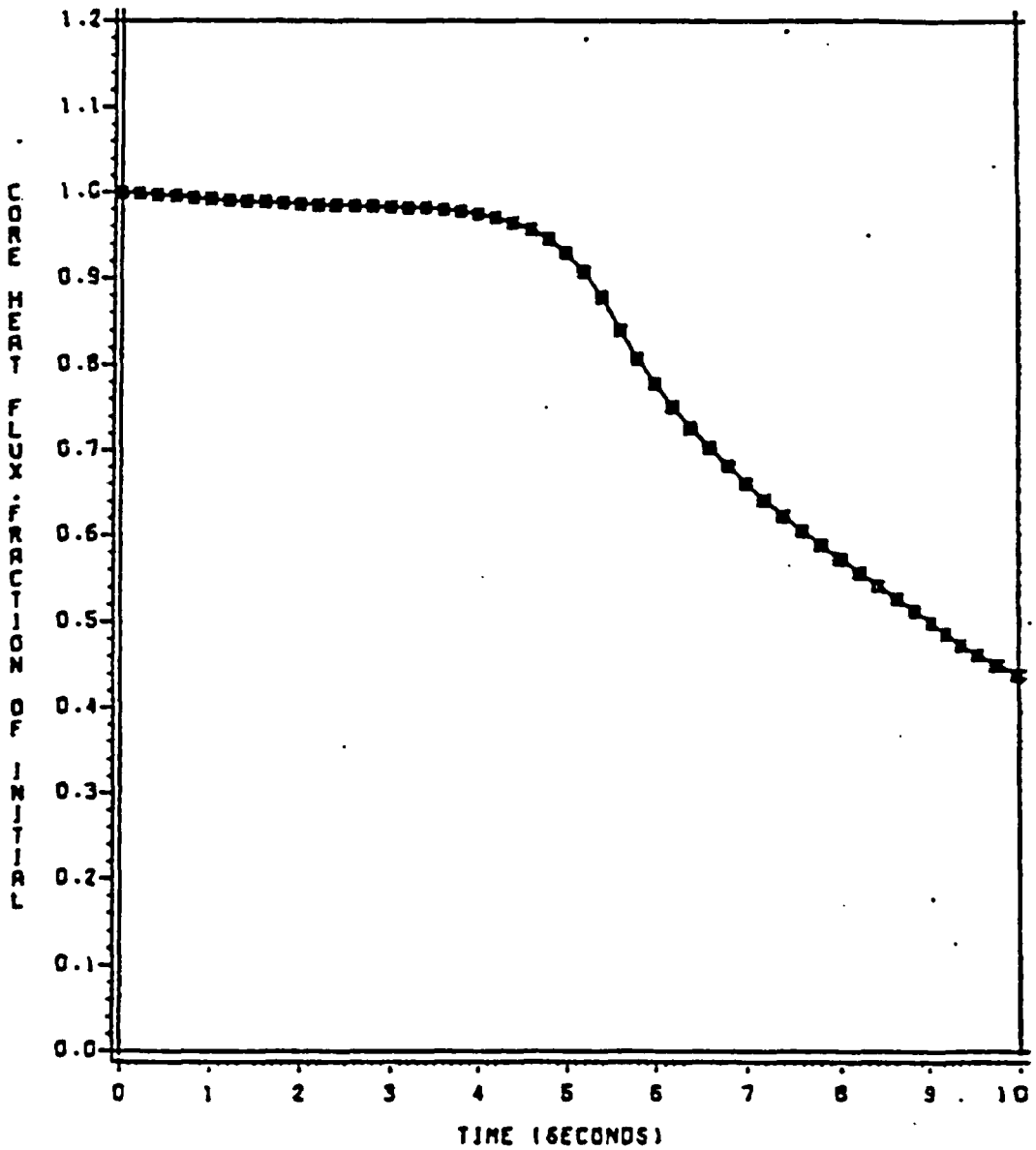


RETRAN01=X
RETRAN02=SOURCE

FIGURE 1C
LOSS OF FLOW

MIDCORE HEAT FLUX

RETRAN01 VS RETRAN02

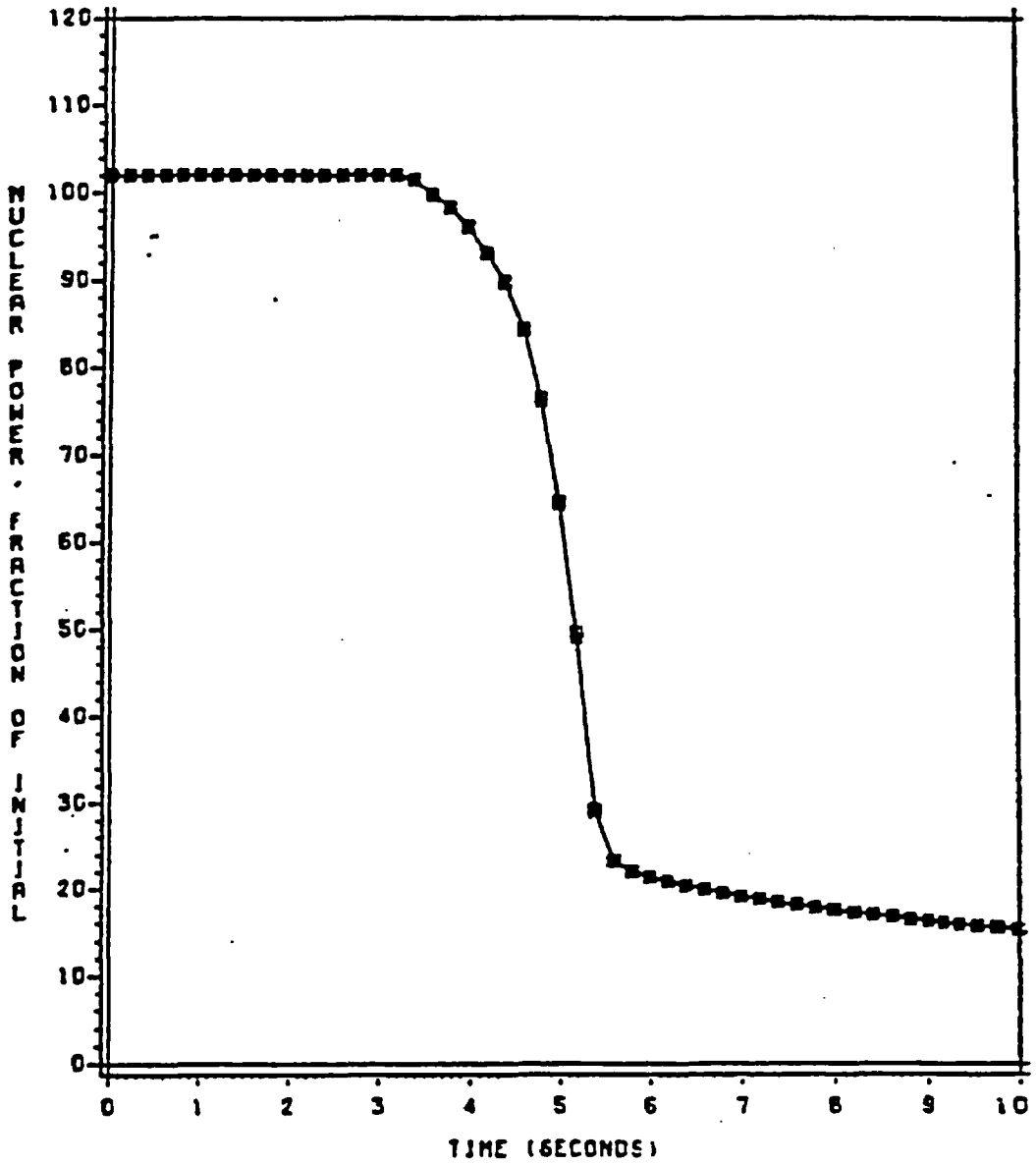


RETRAN01=X
RETRAN02=SQUARE

FIGURE 2C
LOSS OF FLOW

NUCLEAR POWER

RETRAN01 VS RETRAN02

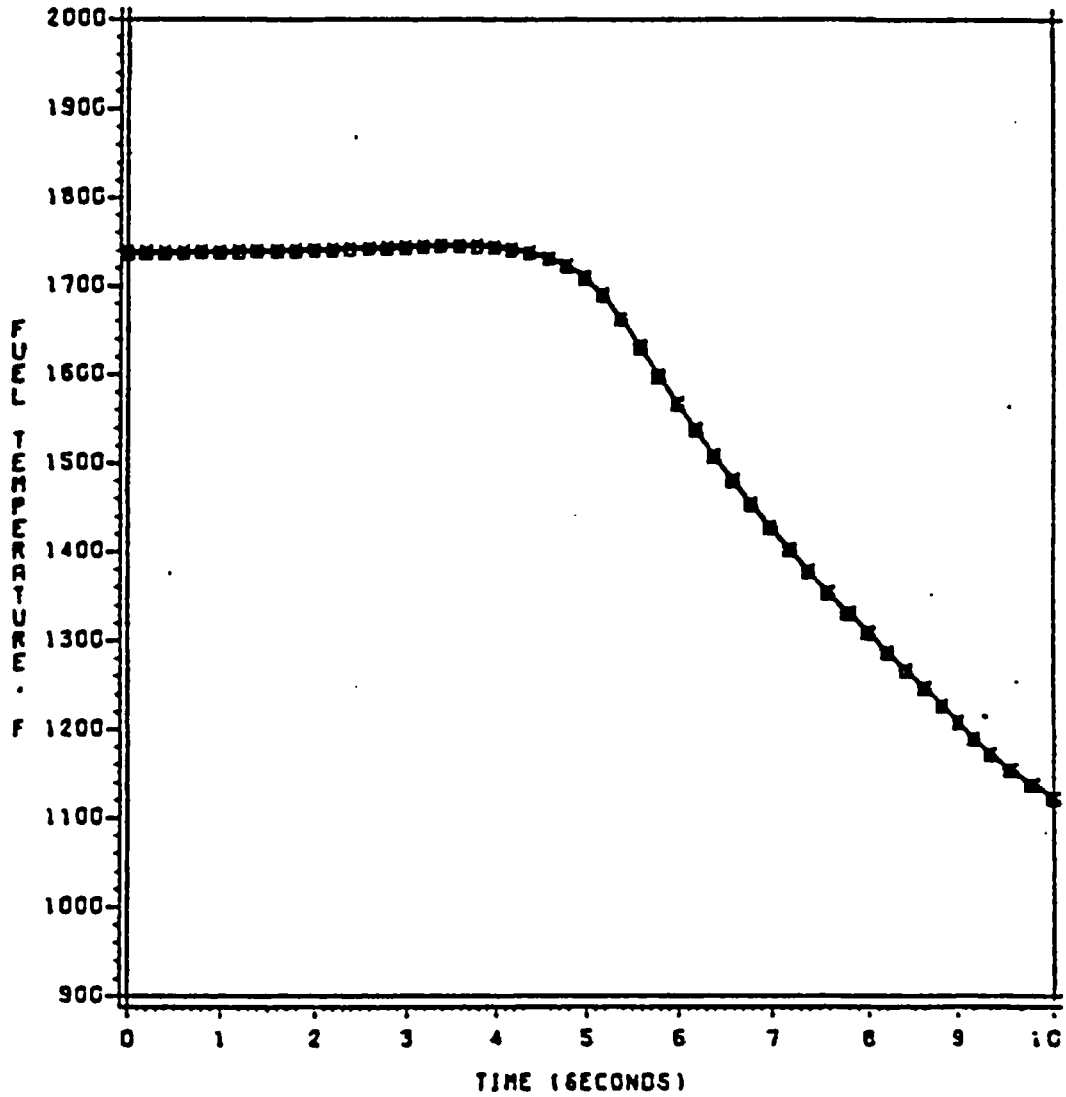


RETRAN01=X
RETRAN02=SQUARE

FIGURE 3 C
LOSS OF FLOW

MIDCORE FUEL TEMP

RETRAN01 VS RETRAN02

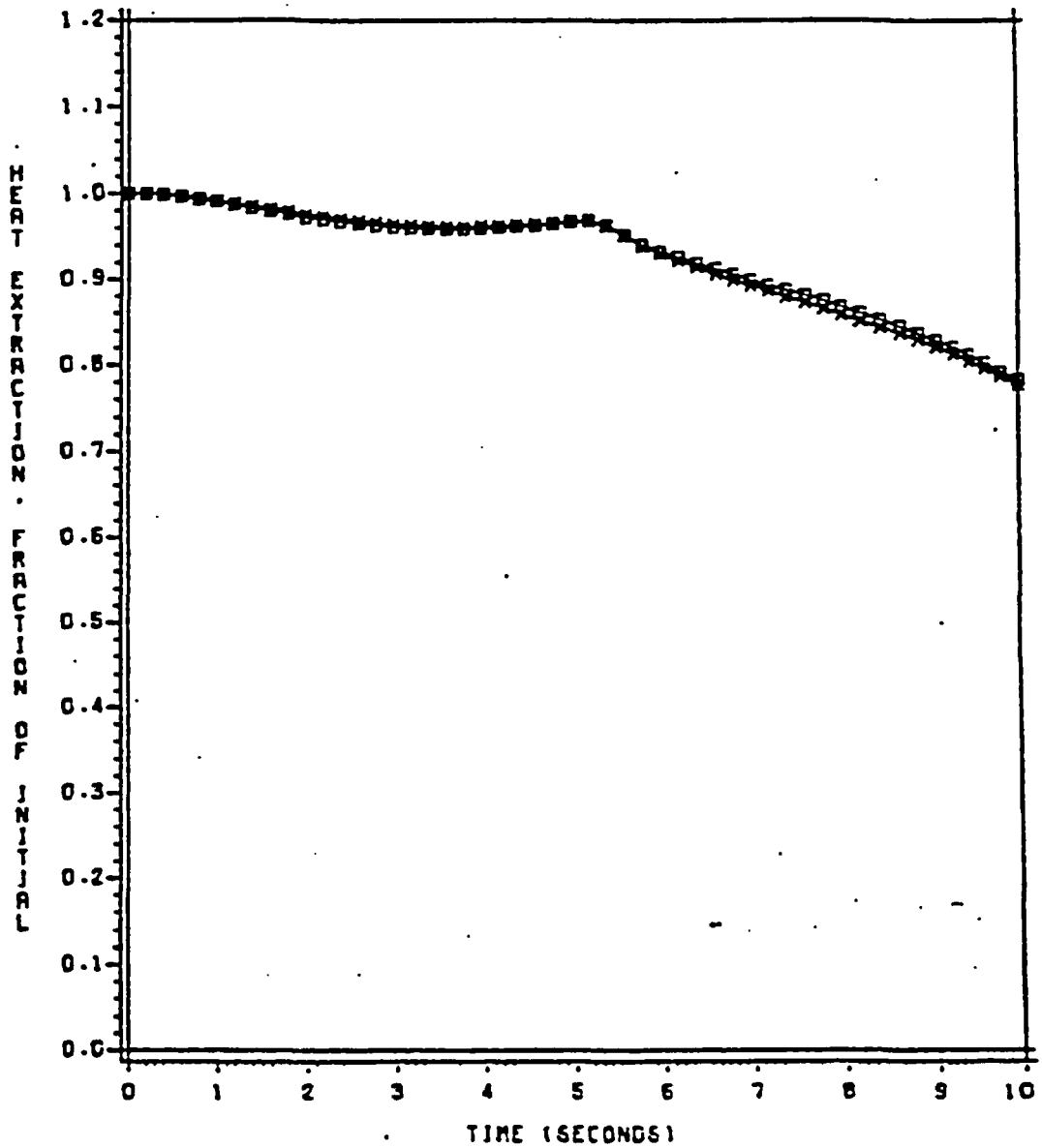


RETRAN01=X
RETRAN02=SQUARE

FIGURE 4C
LOSS OF FLOW

SG HEAT EXTRACTION RATE

RETRAN01 VS RETRAN02

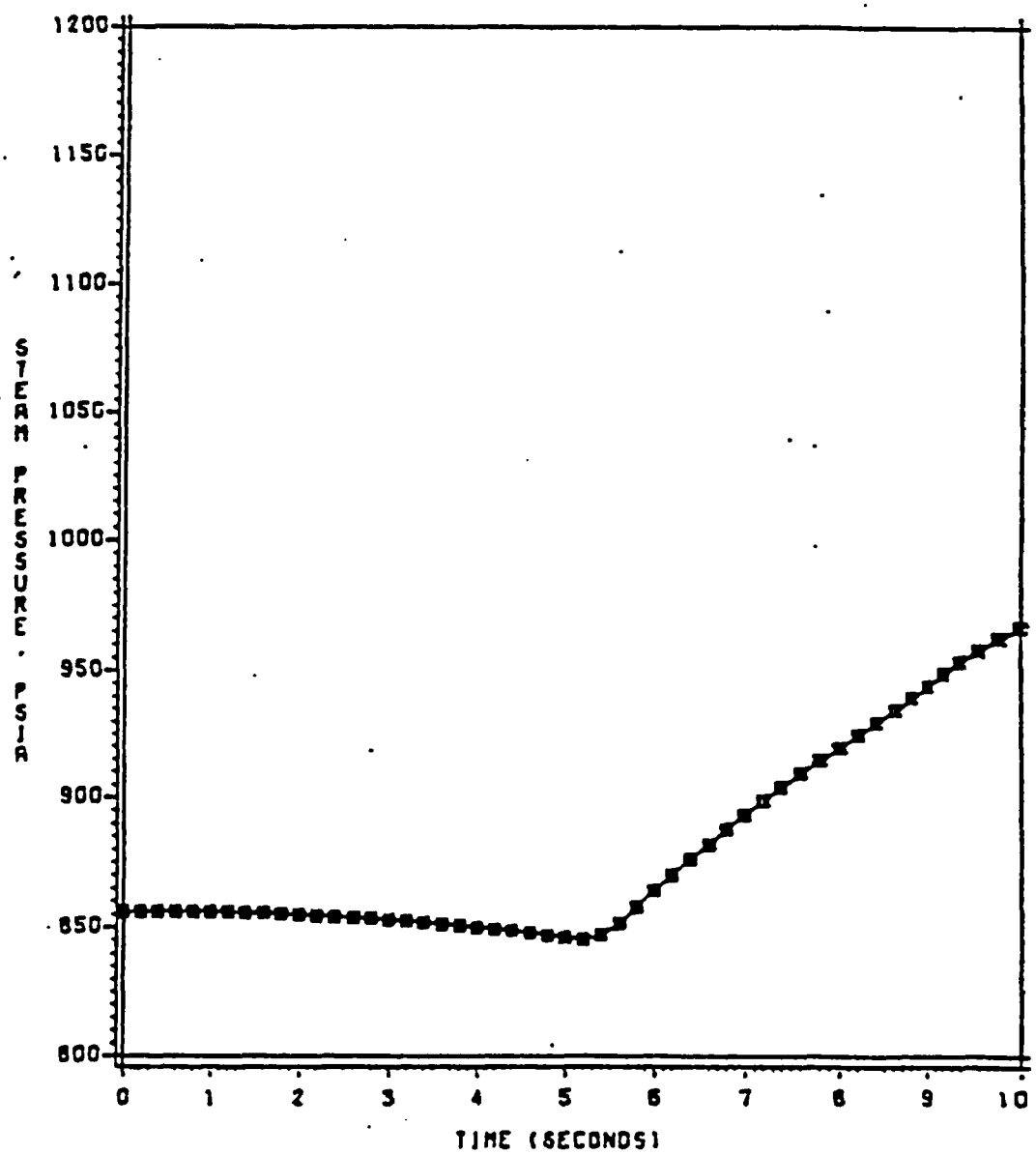


RETRAN01=X
RETRAN02=SQUARE

FIGURE 5C
LOSS OF FLOW

STEAM PRESSURE

RETRAN01 VS RETRAN02

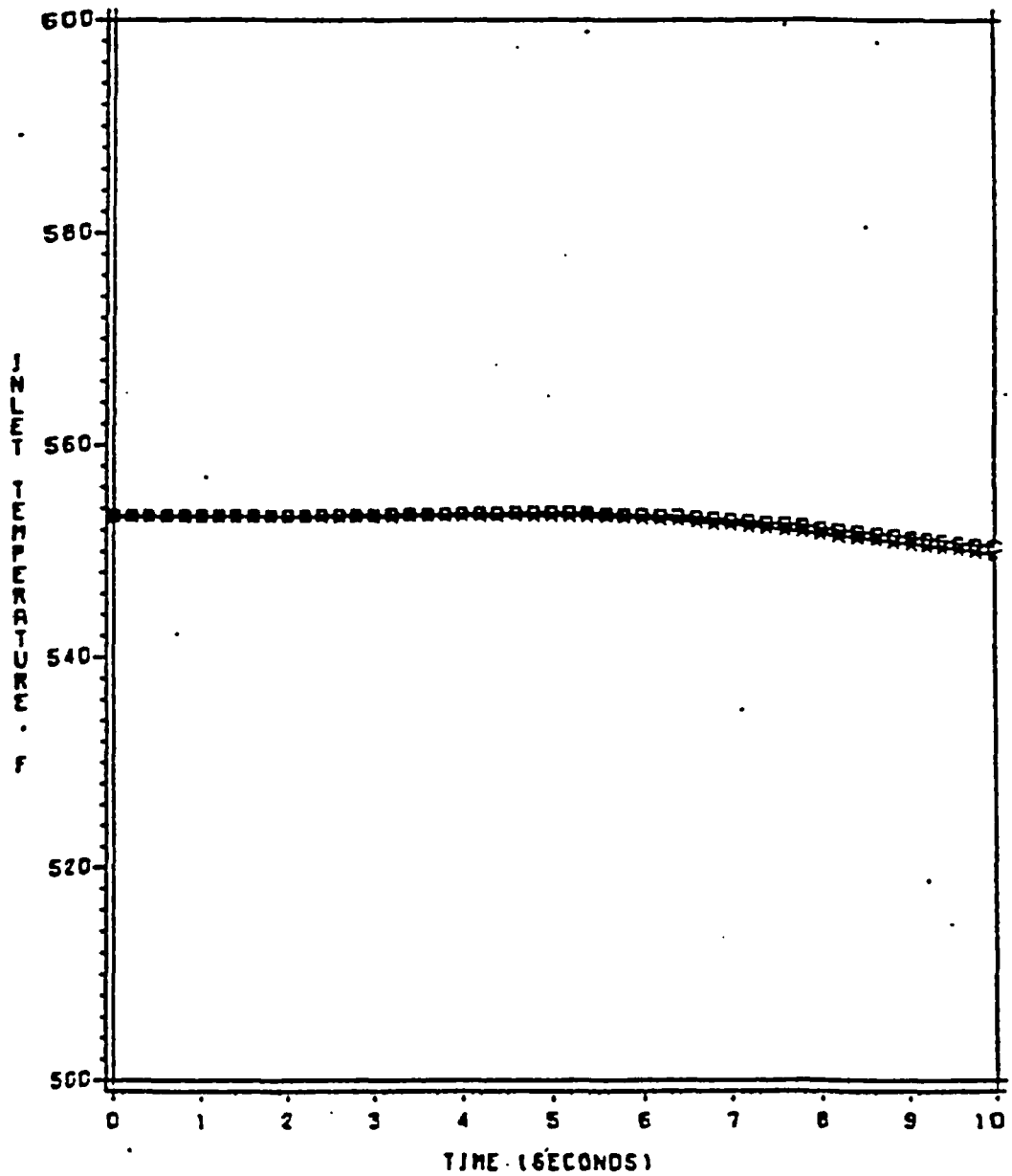


RETRAN01=X
RETRAN02=SQUARE

FIGURE 6C
LOSS OF FLOW

INLET TEMPERATURE

RETRAN01 VS RETRAN02

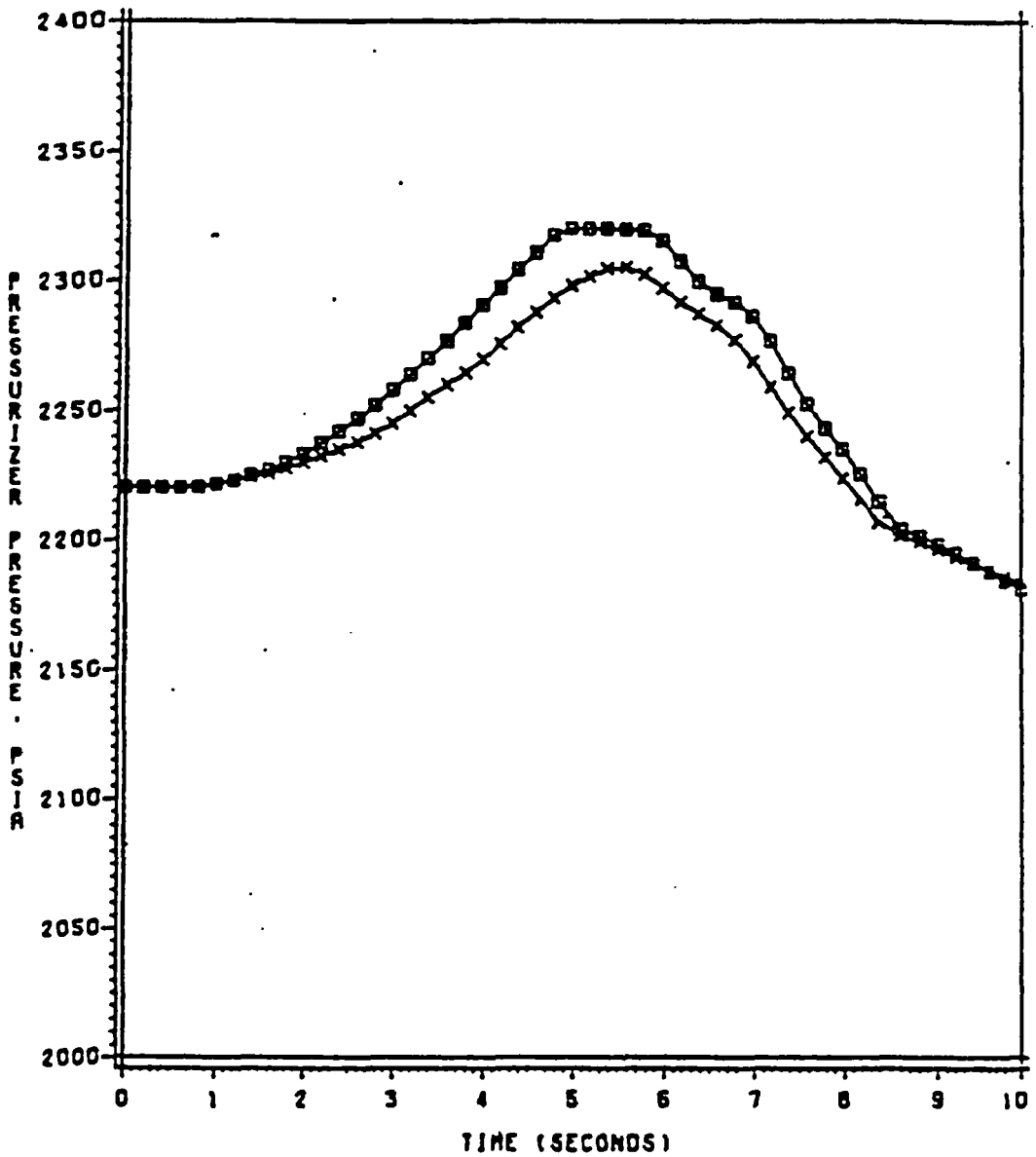


RETRAN01=X
RETRAN02=SQUARE

FIGURE 7C
LOSS OF FLOW

PRESSURIZER PRESSURE

RETRAN01 VS RETRAN02

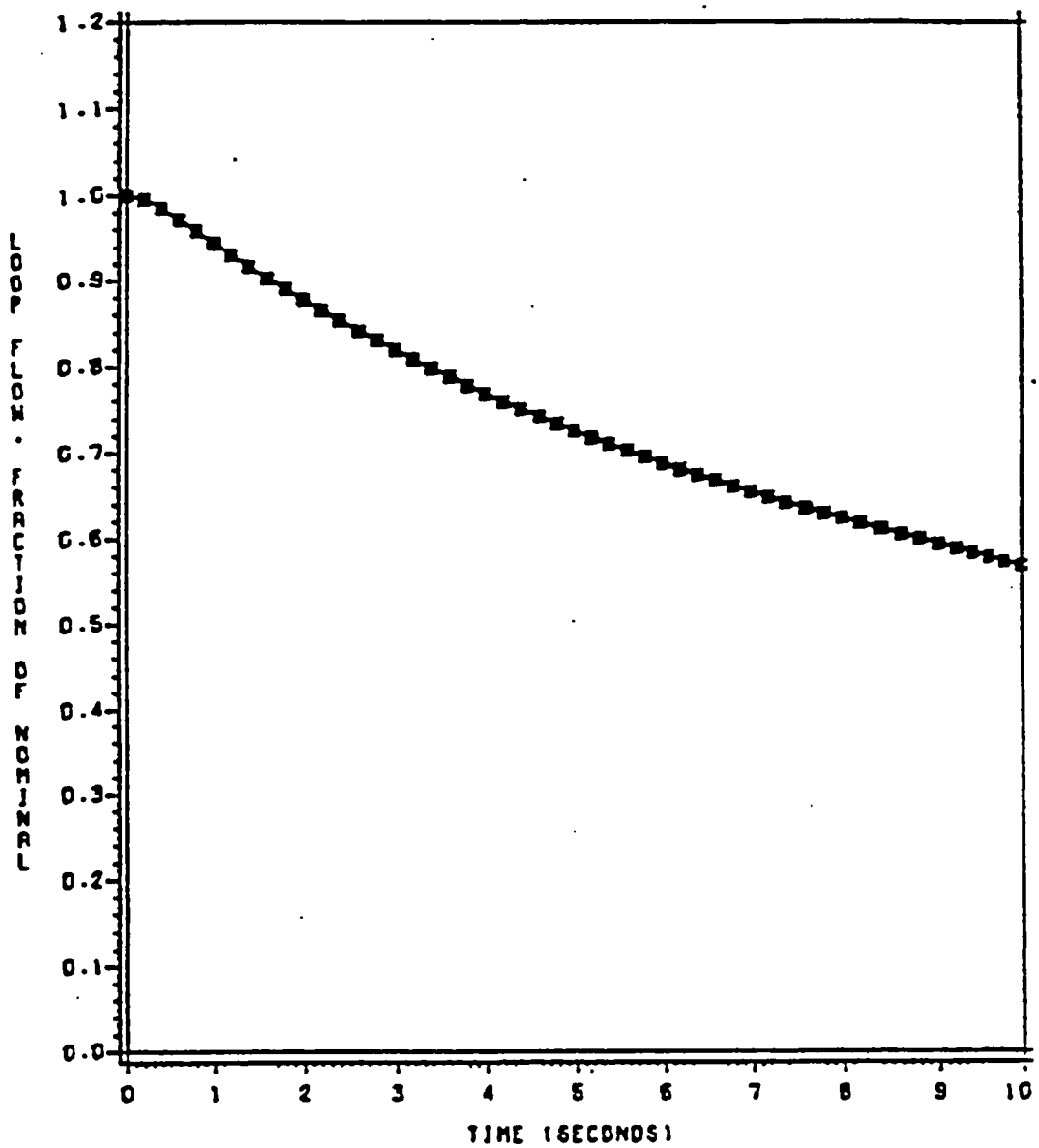


RETRAN01=X
RETRAN02=SQUARE

FIGURE 8C
LOSS OF FLOW

LOOP FLOW

RETRAN01 VS RETRAN02



RETRAN01=X
RETRAN02=SQUARE

APPENDIX 6
SER for VEP-FRD-42 Rev. 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 11, 2003

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, Virginia 23060-6711

SUBJECT: VIRGINIA ELECTRIC AND POWER COMPANY - ACCEPTANCE OF TOPICAL REPORT VEP-FRD-42, REVISION 2, "RELOAD NUCLEAR DESIGN METHODOLOGY," NORTH ANNA AND SURRY POWER STATIONS, UNITS 1 AND 2 (TAC NOS. MB3141, MB3142, MB3151, AND MB3152)

Dear Mr. Christian:

By letter dated October 8, 2001, as supplemented by letters dated May 13, and December 2, 2002, and March 21, 2003, Virginia Electric and Power Company (VEPCO) requested approval of Topical Report VEP-FRD-42, Revision 2, entitled "Reload Nuclear Design Methodology," for North Anna and Surry Power Stations, Units 1 and 2.

The Nuclear Regulatory Commission (NRC) staff has found that Topical Report VEP-FRD-42, Revision 2, is acceptable for referencing in licensing applications for the North Anna and Surry Power Stations, Units 1 and 2, to the extent specified and under the limitations delineated in the report and in the associated NRC Safety Evaluation (SE). The SE defines the basis for acceptance of the report.

Our acceptance applies only to matters approved in the subject report. We do not intend to repeat our review of the acceptable matters described in the report. When the report appears as a reference in licensing applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this topical report will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that VEPCO publish an accepted version of this topical report within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in appendices historical review information, such as questions and accepted responses, and original report pages that were replaced. The accepted version shall include an "-A" (designated accepted) following the report identification symbol.

If the NRC's criteria or regulations change such that its conclusions as to the acceptability of the topical report are invalidated, then VEPCO will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

Scott Moore, Acting Director
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-280, 50-281,
50-338, and 50-339

Enclosure: Safety Evaluation

cc w/encl: See next page

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Virginia Electric and Power Company

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT VEP-FRD-42, REVISION 2

RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT

NORTH ANNA AND SURRY POWER STATIONS, UNITS 1 AND 2

DOCKET NOS. 50-280, 50-281, 50-338, AND 50-339

1.0 INTRODUCTION

By letter dated October 8, 2001 (Reference 1), as supplemented by letters dated May 13, (Reference 2) and December 2, 2002, (Reference 3) and March 21, 2003, (Reference 4) Virginia Electric and Power Company (VEPCO) requested approval of Topical Report VEP-FRD-42, Revision 2, entitled "Reload Nuclear Design Methodology Topical Report," for North Anna and Surry Power Stations, Units 1 and 2. Topical Report VEP-FRD-42 describes the core reload design methodology for performing a nuclear reload design analysis at North Anna and Surry Power Stations. This includes analytical models and methods, reload design and reload safety analysis, and an overview of analyzed accidents. The Nuclear Regulatory Commission (NRC) staff had previously limited the approval of Topical Report VEP-FRD-42, Revision 1-A, "Reload Nuclear Design Methodology," (Reference 5) to licensing applications involving Westinghouse-supplied fuel reloads. Revision 2 of this topical report extends the VEPCO methodology to Framatome ANP Advanced Mark-BW fuel.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34, "Contents of applications; technical information," requires that safety analysis reports be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload process, licensees perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses remain bounding, the licensees confirm that key inputs to the safety analyses are conservative with respect to the current design cycle. If key safety analysis parameters are not bounded, a reanalysis or reevaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

In an effort to limit cycle-specific Technical Specification (TS) changes, the NRC issued Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," (Reference 6) on October 3, 1988, to provide guidance for relocating cycle-specific parameter limits from the TS to a Core Operating Limits Report (COLR). Specifically, this GL allows a licensee to implement a COLR to include cycle-specific parameter limits that are established using NRC-approved methodology. The NRC staff-approved analytical methods used to

Enclosure

determine the COLR cycle-specific parameters are to be identified in the Administrative Controls section of the TS.

Topical Report VEP-FRD-42 is listed in the COLR Administrative Controls section of the North Anna and Surry TS and describes VEPCO's methodology for designing reload cores and performing reload safety analyses. Because the NRC staff previously approved Topical Report VEP-FRD-42, Revision 1-A, the NRC staff's review of Topical Report VEP-FRD-42, Revision 2, focused on the changes made to the approved version. Specifically, the NRC staff review focused on the extension of the methodology to Framatome ANP Advanced Mark-BW fuel types.

3.0 TECHNICAL EVALUATION

Topical Report VEP-FRD-42, Revision 2, describes the methodology applied in the design of reload cores at both the North Anna and Surry Power Stations. This topical report includes descriptions of analytical models and methods, reload nuclear design, reload safety analyses, and an overview of analyzed accidents and key parameter derivations. The NRC staff reviewed and approved Topical Report VEP-FRD-42, Revision 1-A, on July 29, 1986. VEPCO has submitted Revision 2 of this Topical Report to support the transition to Framatome ANP Advanced Mark-BW fuel at the North Anna and Surry Power Stations. In its Safety Evaluation (SE) for VEP-FRD-42, Revision 1-A, the NRC staff stated, "it is clear that the methodology presented is closely related to the Westinghouse methodology, and is applicable in its present form only to Westinghouse supplied reloads of Westinghouse nuclear plants." To support the transition to Framatome ANP Advanced Mark-BW fuel, VEPCO has revised VEP-FRD-42, Revision 1-A, to address this restriction and to present a revised discussion of the reload core design methodology. The Revision 2 changes address the following types of items:

- Applicability of methodology for analysis of incremental fuel design differences
- Generic methodology items impacted by transition to Framatome-ANP fuel
- Consolidation of prior VEPCO submittals regarding code and model updates
- Responses to original NRC staff review questions
- Miscellaneous editorial changes

By letter dated October 8, 2001, VEPCO proposed to apply the methodology described in Topical Report VEP-FRD-42, Revision 2, to both Framatome ANP Advanced Mark-BW and Westinghouse fuel types. In its submittal dated May 13, 2002, VEPCO stated that although the intended extension of this methodology is for the analysis of Framatome ANP Advanced Mark-BW fuel, the methodology is sufficiently robust for use on any fuel product with similar features. However, prior to the use of the Topical Report VEP-FRD-42, Revision 2, methodology for other fuel types, VEPCO must confirm that the impact of the fuel design and its specific features can be completely and accurately modeled with the VEPCO nuclear design and safety analysis codes and methods, that there is no significant effect upon calculated values of key reload safety parameters, and that the safety analysis codes and methods are applicable for analysis of the alternate fuel product. Should the changes necessary to accommodate another fuel product require changes to the reload methodology of Topical Report VEP-FRD-42, Revision 2, these proposed changes would be submitted to the NRC staff for review and approval.

3.1 Analytical Models and Methods

The major analytical models described in Topical Report VEP-FRD-42, Revision 2, and currently used by VEPCO for reload design and safety analysis include:

- Virginia Power PDQ Two-Zone model
- Virginia Power NOMAD model
- VEPCO RETRAN model
- Core Thermal-Hydraulics models

Topical Report VEP-FRD-42, Revision 1-A, listed the applicable computer codes, correlations, and methods used for thermal-hydraulic analyses of reload cores at the North Anna and Surry Power Stations. Topical Report VEP-FRD-42, Revision 2, no longer identifies the specific core thermal-hydraulic methods used; instead it states that the applicable codes and correlations for thermal-hydraulic analyses are listed in the COLR section of the North Anna and Surry TS, respectively. NRC GL 88-16 requires prior NRC staff review and approval of all methodologies used to calculate cycle-specific parameters that are in the COLR, and referenced in the COLR TS section. Thermal-hydraulic methodologies used in designing reload cores are typically fuel specific. The thermal-hydraulic methodologies VEPCO currently applies for the North Anna and Surry Power Stations, for example, the WRB-1 DNB correlation, and the VEPCO COBRA code and a statistical design methodology, are approved for use with the current Westinghouse fuel loaded in the North Anna and Surry cores. As such, in accordance with VEP-FRD-42, Revision 2, methodology, when transitioning to Framatome ANP Advanced Mark-BW fuel, VEPCO must submit a license amendment request to add the applicable and approved thermal-hydraulic methodology references to the COLR TS section. Since NRC GL 88-16 requires prior NRC staff review and approval of the thermal-hydraulic codes, correlations, and methods listed in the COLR section of the TS, the NRC staff finds that generic reference to the thermal-hydraulic methodology listed in the COLR TS section is acceptable.

The NRC staff reviewed and approved all codes used by VEPCO in the physics and thermal-hydraulics analyses of the reload core and described in Topical Report VEP-FRD-42, Revision 1-A. Topical Report VEP-FRD-42, Revision 2, describes the code changes and modifications that have been implemented by VEPCO since the NRC staff approved Topical Report VEP-FRD-42, Revision 1-A, on July 29, 1986. By letters dated October 1, 1990, August 10, 1993, and November 13, 1996, VEPCO formally requested NRC staff approval of these code modifications (References 7 - 9). VEPCO eventually implemented these changes under the provisions of 10 CFR 50.59. Because Topical Report VEP-FRD-42 is listed in the TS COLR section and requires NRC approval, the NRC staff informed VEPCO that the NRC staff must review and approve the analytical methods described within this topical report (Reference 10). Therefore, as part of this review, the NRC staff reviewed the PDQ Two-Zone, NOMAD and RETRAN code modifications described in Topical Report VEP-FRD-42, Revision 2, that were previously implemented under the provisions of 10 CFR 50.59.

PDQ Two-Zone Model

By letter dated October 1, 1990, VEPCO initially requested approval of the PDQ Two-Zone model in order to support the use of axially zoned flux suppression inserts in Surry, Units 1 and 2. The PDQ Two-Zone model is a three-dimensional, coarse mesh model that was developed to replace the PDQ Discrete model described in Topical Report VEP-FRD-42,

Revision 1-A. The PDQ Two-Zone model is used to calculate three-dimensional power distributions, delayed neutron data, radial and axial peaking factors, assembly-wise burnup and isotopic concentrations, differential and integral rod worths, differential boron worth and boron endpoints, xenon and samarium worth, and core average reactivity coefficients such as temperature and power coefficients. In addition, PDQ is used to generate predicted power and flux distributions in order to translate thimble flux measurements into measured power distributions.

As part of the review of Topical Report VEP-FRD-42, Revision 2, the NRC staff reviewed the PDQ Two-Zone model as described in Topical Report VEP-NAF-1, "PDQ Two Zone Model," that VEPCO submitted on October 1, 1990. By letter dated December 2, 2002, VEPCO verified that this topical report was the latest revision that has not received NRC staff approval and that this report contains an accurate representation of current codes and models with regard to methodology. That is, the theory, sources of input data, solution schemes, geometric mesh structure, energy group structure, and use of the models in the core modeling process have not changed since the October 1, 1990, submittal. Because VEPCO has been using the PDQ Two-Zone model in core designs for some time, the NRC staff review focused on model predictions relative to actual plant data.

VEPCO informed the NRC staff of its intent to implement the PDQ Two-Zone model under the provisions of 10 CFR 50.59 in a letter dated November 25, 1992 (Reference 11). Since that time, the PDQ Two-Zone model has been used in numerous core designs for both the North Anna and Surry Power Stations. The accuracy of the PDQ Two-Zone model has been verified each cycle during startup physics testing and during routine core follow. For each cycle, a Startup Physics Test Report and a Core Performance Report is issued to document the behavior of the core relative to the model predictions. By letter dated March 21, 2003, VEPCO provided additional information that demonstrated the accuracy of the PDQ model. This information includes measured and predicted data for key reactor physics parameters and confirmation that the nuclear reliability factors for these parameters are within the NRC-approved acceptance limits. Based on the accuracy demonstrated by these comparisons to actual plant data, the NRC staff finds the PDQ Two-Zone model to be acceptable for continued use in licensing calculations for the North Anna and Surry Power Stations. VEPCO's use of the PDQ Two-Zone model for the North Anna and Surry core designs shall be in accordance with the restrictions and limitations listed in VEPCO's submittal dated March 21, 2003, and with Section 5.0 of this SE.

NOMAD

The VEPCO NOMAD model is a one-dimensional (axial), two energy group, diffusion theory computer code with thermal-hydraulic feedback. The NRC staff approved Topical Report VEP-NFE-1-A, "The VEPCO NOMAD Code and Model," for use of the NOMAD code and model on March 4, 1985. This version of the model is referenced in VEP-FRD-42, Revisions 1 and 2. VEPCO subsequently requested approval of an enhanced version of the NOMAD model on November 13, 1996. The most significant enhancement to the NOMAD model is the use of multi-plane data from the three-dimensional (3-D) VEPCO PDQ Two-Zone model as the primary source of input. All model inputs to NOMAD come either directly or indirectly from the PDQ 3-D model calculations. Other enhancements to the model include improvements to the xenon model, the control rod model, the cross-section fit model, and the buckling model. The NOMAD model is used in the calculation of core average axial power distributions, axial offset,

axial power peaking factors, differential control rod bank worth, integral control rod worth as a function of bank position, fission product poison worth, and reactivity defects.

As part of the review of Topical Report VEP-FRD-42, Revision 2, the NRC staff reviewed the NOMAD model as described in VEPCO's Topical Report VEP-NFE-1-A, Supplement 1, dated November 13, 1996. By letter dated December 2, 2002, VEPCO verified that this was the latest revision of the topical report that has not received NRC staff approval and that this report contains an accurate representation of current codes and models with regard to methodology. That is, the theory, sources of input data, solution schemes, geometric mesh structure, energy group structure, and use of the models in the core modeling process have not changed since the November 13, 1996, submittal. Because VEPCO has been using this enhanced NOMAD model in core designs for some time, the NRC staff review focused on model predictions relative to actual plant data.

VEPCO informed the NRC staff of its intent to implement the enhanced NOMAD model under the provisions of 10 CFR 50.59 in a letter dated November 13, 1996. Since that time, the NOMAD model has been used in numerous core designs for both the North Anna and Surry Power Stations. The accuracy of the NOMAD model has been verified each cycle during startup physics testing and during routine core follow. For each cycle, a Startup Physics Test Report and a Core Performance Report is issued to document the behavior of the core relative to the model predictions. VEPCO provided additional information on March 21, 2003, that demonstrates the accuracy of the NOMAD model. This information includes measured and predicted data for key reactor physics parameters and confirmation that the nuclear reliability factors for these parameters are within the NRC-approved acceptance limits. The NRC staff reviewed the measured data against the predicted data, and based on the accuracy demonstrated by these comparisons to actual plant data, the NRC staff finds the NOMAD model to be acceptable for continued use in licensing calculations for the North Anna and Surry Power Stations. VEPCO's use of the NOMAD model for the North Anna and Surry core designs shall be in accordance with the restrictions and limitations listed in VEPCO's submittal dated March 21, 2003, and with Section 5.0 of this SE.

RETRAN

In the generic RETRAN SE dated September 4, 1984 (Reference 13), the NRC staff generically approved the use of RETRAN-01/MOD003 and RETRAN-02/MOD002 subject to the limitations and restrictions outlined in the SE and its enclosed Technical Evaluation Reports (TERs). The NRC staff reviewed VEPCO's RETRAN models and capabilities and approved the use of RETRAN-01/MOD003 for VEPCO in a letter dated April 11, 1985 (Reference 12). The NRC staff's SE stated that VEPCO had not provided information to address the restrictions stated in the NRC staff's SE for the generic RETRAN computer code and that VEPCO had not provided an input deck to the NRC staff as was required by the NRC staff's SE for the generic RETRAN code. The input deck submittal was required from VEPCO as a condition of the approval to use RETRAN. The NRC staff has verified VEPCO submission of the RETRAN input decks on August 21, 1985 (Reference 16), but could not verify that VEPCO submitted the RETRAN code limitations and restrictions.

In a letter dated August 10, 1993, VEPCO informed the NRC staff of various modifications and updates to its RETRAN model, and that these changes were to be implemented under the provisions of 10 CFR 50.59. This letter described several changes to the VEPCO RETRAN

models, including expansion to a three-loop Reactor Coolant System and multi-node steam generator secondary side. Although this letter was submitted for the North Anna Power Station, VEPCO provided additional information on December 2, 2002, and March 21, 2003, justifying the applicability of the RETRAN model to both the Surry and North Anna Power Stations. By letter dated December 2, 2002, VEPCO provided additional information regarding its capability to make modifications to the RETRAN model. The NRC staff's SE dated April 11, 1985, for the VEPCO RETRAN model recognized that model maintenance activities would be performed under the utility's 10 CFR 50, Appendix B, Quality Assurance program, and stated, "The staff requires that all future modifications of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures." The NRC staff has determined that VEPCO has followed the requirements specified in the NRC staff's SE in updating the RETRAN models. Additionally, the NRC staff has also determined the qualification, documentation and implementation of the new models was performed in a manner that meets the programmatic elements of NRC GL 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," dated June 24, 1999 (Reference 17).

VEPCO is currently using RETRAN 02/MOD005.2. As such, the NRC staff requested additional information describing how each of the limitations, restrictions, and items identified as requiring additional user justification in the generic NRC staff's SEs, through the currently used version, are satisfied. This includes RETRAN02/MOD002 (Reference 13), RETRAN02/MOD003 and MOD004 (Reference 14) and RETRAN02/MOD005 (Reference 15). By letter dated March 21, 2003, VEPCO provided detailed information describing how each limitation (approximately 48 total) is treated in the North Anna and Surry RETRAN models. The NRC staff has reviewed VEPCO's responses and finds that the limitations, restrictions, and items identified as requiring additional user justification are satisfactorily addressed.

Based on the above discussions, the NRC staff finds that the VEPCO RETRAN models and the use of RETRAN continue to be acceptable for use in licensing calculations for the North Anna and Surry Power Stations.

Core Thermal-Hydraulics and Nuclear Design Models

In its submittal dated May 13, 2002, VEPCO provided information to demonstrate that the Framatome ANP Advanced Mark-BW fuel features affecting the safety analysis design inputs were within the modeling capability of the analytical models used as part of the reload design process and were identified in Topical Report VEP-FRD-42, Revision 2. From a core design perspective, the differences in modeling Framatome ANP Advanced Mark-BW fuel relative to Westinghouse fuel are small and are accommodated using model input parameters. These differences between the fuel types are similar in magnitude to incremental changes in Westinghouse fuel over time, which VEPCO has successfully modeled. Some of these minor changes include spacer grid differences, a slight increase in fuel density, a slight difference in the position of the fuel stack, and use of the advanced M5 alloy cladding. VEPCO has performed comparisons of measured and predicted Framatome ANP Advanced Mark-BW lead test assembly axial and integral power distributions over three cycles of operation in North Anna, Unit 1. The results of these comparisons provide direct confirmation of the accuracy with which VEPCO's reload analytical models can model Framatome ANP Advanced Mark-BW fuel. VEPCO has also performed several benchmark calculations to support use of these analytical models. In addition, in its submittal dated May 13, 2002, VEPCO also stated that the modeling

changes associated with the Framatome ANP Advanced Mark-BW fuel are within the restrictions and limitations of the VEPCO core design and safety analysis codes. The NRC staff has reviewed this information provided by VEPCO and agrees that the Framatome ANP Advanced Mark-BW fuel features are within the modeling capability of the VEPCO core design analytical models. As such, the NRC staff finds that this modeling capability is applicable to both Westinghouse and Framatome ANP Advanced Mark-BW fuel types.

Analytical Methods

Topical Report VEP-FRD-42, Revision 2, Section 2.2, "Analytical Methods," provides a description of the various analytical methods used in the cycle design and evaluation. These methods are classified into three types of calculations: core depletions, core reactivity parameters and coefficients, and core reactivity control. Topical Report VEP-FRD-42, Revision 2, provides a very general description of the methods used to calculate these types of core physics parameters. These methods are consistent with those approved by the NRC staff in Topical Report VEP-FRD-42, Revision 1-A. VEPCO has incorporated some very minor changes. For example, the temperature increment and decrement range used in calculating reactivity coefficients can now be ±5°F or ±10°F about the nominal temperature, rather than only ±5°F as in Topical Report VEP-FRD-42, Revision 1-A. VEPCO added the range of ±10°F to minimize 3-D model convergence tolerance on the coefficients. The NRC staff does not consider these types of minor input changes as changes to the reload methodology. Additionally, the NRC staff agrees with VEPCO and finds that the analytical methods discussed in this section of Topical Report VEP-FRD-42, Revision 2, are not inherently dependent upon a specific fuel design or manufacturer. As such, the NRC staff finds that these methods are applicable to both Westinghouse and Framatome ANP Advanced Mark-BW fuel types because the analytical models used to implement these methods have been shown to be applicable for both Westinghouse and Framatome ANP Advanced Mark-BW fuel.

Analytical Model and Method Approval Process

Topical Report VEP-FRD-42, Revision 2, Section 2.3, "Analytical Model and Method Approval Process," is a new section in the topical report that describes acceptable means by which analytical models and methods can achieve approved status for use in the reload methodology. These acceptable means include: implementation in accordance with the provisions of 10 CFR 50.59, independent review and approval by NRC, incorporation as a reference in the COLR section of the plant TS, and incorporation as a reference tool under VEPCO's GL 83-11, Supplement 1, Program. In its submittal dated May 13, 2002, VEPCO provided clarification regarding the types of changes that would be allowed under the provisions of 10 CFR 50.59, and the NRC staff has determined that VEPCO's interpretation is consistent with the intent of 10 CFR 50.59. Each of these means of achieving approved status either requires prior NRC approval or is a mechanism already acceptable to the NRC staff. Therefore, the NRC staff finds the addition of this new section to be acceptable. Additionally, these methods of achieving approved status are not fuel-specific and apply to both Westinghouse and Framatome ANP Advanced Mark-BW fuel types.

3.2 Reload Design

The overall objective of core reload design is to determine fuel enrichment, feed batch size, and a core loading pattern that fulfills cycle energy requirements while satisfying the constraints of

the plant design basis and safety analysis limits. Topical Report VEP-FRD-42, Revision 2, provides a general description of the reload design methodology used for the North Anna and Surry Power Stations, and is largely consistent with the NRC-approved methodology of Topical Report VEP-FRD-42, Revision 1-A. This VEPCO methodology divides the reload design process into three phases: 1) core loading pattern design and optimization, 2) determination of core physics related key analysis parameters for reload safety analysis, and 3) design report, operator curve, and core follow predictions.

In the reload safety analysis process, VEPCO uses a bounding analysis concept. This approach employs a list of key analysis parameters and limiting directions of the key analysis parameters for various transients and accidents. For a proposed core reload design, if all key analysis parameters are conservatively bounded, then the reference safety analysis is assumed to apply, and no further analysis is necessary. If one or more key analysis parameters is not bounded, then further analysis or evaluation of the transient or accident in question is performed. Topical Report VEP-FRD-42, Revision 2, Table 2 lists the key analysis parameters considered in reload design. To account for Framatome ANP Advanced Mark-BW fuel types, VEPCO determined that one additional key analysis parameter is required. This parameter, maximum linear heat generation rate versus burnup, is used in the NRC-approved Framatome ANP methodology for cladding stress evaluations. By letter dated May 13, 2002, VEPCO stated it calculates this key analysis parameter using the existing nuclear design codes PDQ Two-Zone and NOMAD.

The methods VEPCO used to determine the key parameters were consistent with the methods documented in Topical Report VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated F_Q Surveillance Technical Specifications," dated March 1986 (Reference 18), Topical Report WCAP-9272, "Westinghouse Reload Safety Evaluation," dated March 1978 (Reference 19), and Topical Report WCAP-8385, "Topical Report Power Distribution Control and Load Following Procedures," dated September 1974 (Reference 20). Topical Reports WCAP-9272 and WCAP-8385 are Westinghouse WCAP methodologies used for reload safety evaluations, and power distribution control and load following procedures. Topical Report VEP-NE-1-A documents VEPCO's NRC-approved Relaxed Power Distribution Control methodology. As part of the Topical Report VEP-FRD-42, Revision 2, review, the NRC staff questioned the applicability of these methodologies to Framatome ANP Advanced Mark-BW fuel types. By letter dated May 13, 2002, VEPCO provided additional information to the NRC staff, including the justification for the application of these methods for analyzing Framatome ANP Advanced Mark-BW fuel. Topical Reports VEP-NE-1-A and WCAP-8385 describe methodologies involving the simulation of a number of perturbed core states and power distributions using detailed nuclear core design codes and models. These analyses depend upon defining proper design inputs that characterize the reactor core. As discussed in Section 3.1, "Analytical Models and Methods," of this SE, VEPCO has demonstrated that the Framatome ANP Advanced Mark-BW fuel features are within the existing capability and range of applicability of the nuclear core design and safety analysis tools. Topical Report WCAP-9272 describes the Westinghouse reload methodology and forms the basis for VEPCO's reload methodology as described in Topical Report VEP-FRD-42, Revision 2. This Westinghouse methodology defines the specific key parameters for use in accident analyses and provides limiting directions for consideration in reload evaluations. VEPCO evaluated the use of an alternative fuel type and concluded that none of the physical design features invalidate the key parameter definitions or usage as cited in Topical Reports WCAP-9272 or VEP-FRD-42, Revision 1-A.

Topical Report VEP-FRD-42, Revision 2, incorporated Westinghouse's methodology for the analysis of the dropped rod event described in Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990 (Reference 21). This Westinghouse methodology requires that analyses be performed to determine: 1) statepoints (reactor power, temperature and pressure), 2) radial power peaking factors, and 3) DNB analysis at the conditions determined by items 1 and 2. This methodology incorporated data that is both plant-specific and cycle-specific. As part of the Topical Report VEP-FRD-42, Revision 2, review, the NRC staff questioned the applicability of this methodology to Framatome ANP Advanced Mark-BW fuel types. In its submittal dated May 13, 2002, VEPCO provided additional information to the NRC staff justifying the application of this methodology. VEPCO stated that the core physics characteristics of the Framatome ANP Advanced Mark-BW fuel are nearly identical to the Westinghouse fuel it will replace. There is no change in loading pattern strategy associated with the Framatome ANP Advanced Mark-BW fuel that would cause a change in the range of dropped rod worth or in the relationship between dropped rod worth and peaking factor increase. Reload cores, therefore, will not respond in a fundamentally different way to the dropped rod event due to the use of Framatome ANP Advanced Mark-BW fuel. Based on VEPCO's response and a review of the Westinghouse methodology, the NRC staff finds that this methodology would be applicable to both Westinghouse and Framatome ANP Advanced Mark-BW fuel types.

The NRC staff has reviewed the information provided by VEPCO and finds that the reload nuclear design methodology described in Topical Report VEP-FRD-42, Revision 2, is applicable to Framatome ANP Advanced Mark-BW fuel in addition to Westinghouse fuel types. This methodology incorporates several key elements, none of which is inherently dependent upon a specific fuel design or manufacturer. These key attributes of the methodology include:

- analysis framework in which safety analyses establish the acceptable values for reload core key parameters, while nuclear and fuel design codes confirm each core's margin to the limits,
- use of bounding key parameter values in reference safety analyses,
- recurrent validation of nuclear design analytical predictions through comparison with reload core measurement data,
- representation of key fuel features via detailed inputs in core design and safety analysis models, and
- fuel is modeled using approved critical heat flux correlations demonstrated to be applicable and within the range of qualification and identified in the plant COLR section of the TS.

4.0 CONCLUSIONS

The NRC staff has reviewed VEPCO's submittals and supporting documentation. Based on the considerations above, the NRC staff has concluded that the proposed Topical Report VEP-FRD-42, Revision 2, is acceptable for use in licensing applications at the North Anna and Surry Power Stations involving Westinghouse and Framatome ANP Advanced Mark-BW fuel types. Additionally, the NRC staff finds the continued use of PDQ Two-Zone, NOMAD, and RETRAN acceptable for licensing applications at the North Anna and Surry Power Stations involving Westinghouse and Framatome ANP Advanced Mark-BW fuel types.

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) use of this topical report will not be inimical to the common defense and security nor to the health and safety of the public.

5.0 CONDITIONS AND LIMITATIONS

Prior to the use of the Topical Report VEP-FRD-42, Revision 2, methodology for fuel types other than Westinghouse and Framatome ANP Advanced Mark-BW fuel, VEPCO must confirm that the impact of the fuel design and its specific features can be accurately modeled with the VEPCO nuclear design and safety analysis codes and methods as discussed in its submittal dated May 13, 2002. Should the changes necessary to accommodate another fuel product require changes to the reload methodology of Topical Report VEP-FRD-42, Revision 2, these proposed changes are required to be submitted for prior NRC review and approval.

In accordance with the Topical Report VEP-FRD-42, Revision 2, methodology, when transitioning to Framatome ANP Advanced Mark-BW fuel, VEPCO must submit a license amendment request to add the applicable and approved thermal-hydraulic methodology references to the COLR TS section. In addition, NRC GL 88-16 requires prior NRC staff review and approval of the thermal-hydraulic codes, correlations, and methods listed in the COLR section of the TS.

VEPCO's use of the PDQ Two-Zone model for the North Anna and Surry core designs shall be in accordance with the restrictions and limitations listed in Attachment 2 of VEPCO's submittal dated March 21, 2003.

VEPCO's use of the NOMAD model for the North Anna and Surry core designs shall be in accordance with the restrictions and limitations listed in Attachment 3 of VEPCO's submittal dated March 21, 2003.

6.0 REFERENCES

1. Letter from L. N. Hartz, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Dominion's Reload Nuclear Design Methodology Topical Report," Docket Nos. 50-338/339 and 50-280/281, dated October 8, 2001.

2. Letter from L. N. Hartz, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," Docket Nos. 50-338/339 and 50-280/281, dated May 13, 2002.

3. Letter from E. S. Grecheck, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," Docket Nos. 50-338/339 and 50-280/281, dated December 2, 2002.

4. Letter from L. N. Hartz, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Request for Additional Information on Topical Report VEP-FRD-42, Reload Nuclear Design Methodology," Docket Nos. 50-338/339 and 50-280/281, dated March 21, 2003.

5. Letter from C. E. Rossi, USNRC, to W. L. Stewart, VEPCO, "Acceptance for Referencing of Licensing Topical Report VEP-FRD-42, Revision 1-A, Reload Nuclear Design Methodology," dated July 29, 1986.

6. USNRC GL 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 3, 1988.

7. Letter from W. L. Stewart, VEPCO, to USNRC, "Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Topical Report - PDQ Two Zone Model," Docket Nos. 50-280/281 and 50-338/339, dated October 1, 1990.

8. Letter from S. P. Sarver, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Supplemental Information for the NOMAD Code and Model, Reload Nuclear Design Methodology, and Relaxed Power Distribution Control Methodology Topical Reports," Docket Nos. 50-338/339 and 50-280/281, dated November 13, 1996.

9. Letter from M. L. Bowling, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Supplemental Information on the RETRAN NSSS Model," Docket Nos. 50-338/339, dated August 10, 1993.

10. Letter from S. R. Monarque and G. E. Edison, USNRC, to D. A. Christian,VEPCO, "North Anna Power Station Units 1 and 2, and Surry Power Station Units 1 and 2 - Request for Additional Information on Virginia Electric and Power Company's Reload Nuclear Design Methodology Topical Report VEP-FRD-42 (TAC NOS. MB3141, MB3142, MB3151, and MB3152)," dated October 25, 2002.

11. Letter from W. L. Stewart, VEPCO, to USNRC, "Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Topical Report Use Pursuant to 10CFR50.59," Docket Nos. 50-280/281 and 50-338/339, dated November 25, 1992.

12. Letter from C. O. Martin, USNRC, to W. L. Stewart, VEPCO, "Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, Virginia Power Reactor System Transient Analyses Using the RETRAN Computer Code," dated April 11, 1985.

13. Letter from C. O. Thomas (USNRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, RETRAN - A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems, and EPRI NP-1850-CCM, RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," dated September 4, 1984.

14. Letter from A. C. Thadani (USNRC) to R. Furia (GPU), "Acceptance for Referencing Topical Report EPRI-NP-1850 CCM-A, Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004," dated October 19, 1988.

15. Letter from A. C. Thadani (USNRC) to W. J. Boatwright (RETRAN02 Maintenance Group), "Acceptance for Use of RETRAN02/MOD005.0," dated November 1, 1991.

16. Letter from W. L. Stewart, VEPCO, to H. R. Denton, USNRC, "Virginia Power, Surry and North Anna Power Stations, Reactor System Transient Analyses," Docket Nos. 50-280/281 and 50-338/339, dated August 21, 1985.

17. USNRC GL 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," dated June 24, 1999.

18. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated F_Q Surveillance Technical Specifications," dated March 1986.

19. WCAP-9272, "Westinghouse Reload Safety Evaluation," dated March 1978.

20. WCAP-8385, "Topical Report Power Distribution Control and Load Following Procedures," dated September 1974.

21. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990.

Principal Contributor: M. Kowal

Date: June 11, 2003

APPENDIX 7

RAI Responses for VEP-FRD-42 Pertaining to Dominion's RETRAN Capability

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 13, 2002

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	02-280
NL&OS/ETS	R0
Docket Nos.	50-338/339
	50-280/281
License Nos.	NPF-4/7
	DPR-32/37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
SURRY POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
DOMINION'S RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT

Dominion's Reload Nuclear Design Methodology Topical Report has been revised to support the transition to Framatome ANP Advanced Mark-BW fuel at North Anna. In a letter dated October 8, 2001 (Serial No. 01-628) Dominion submitted Revision 2 of VEP-FRD-42, "Reload Nuclear Design Methodology Topical Report," for NRC review and approval. During review of the topical report, the NRC staff identified additional information that is needed to complete their review. This additional information is provided in the attachment to this letter.

If you have any further questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachment

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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Attachment

REQUEST FOR ADDITIONAL INFORMATION
DOMINION'S RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT
VEP-FRD-42, Revision 2

North Anna Power Station Units 1 and 2
Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

In April 15 and 16, 2002 discussions with the NRC staff, regarding Dominion's Topical Report, VEP-FRD-42, Revision 2, "Reload Nuclear Design Methodology," the following additional information was requested.

Question 1:

Is the Dominion reload methodology discussed in Topical Report VEP-FRD-42, Revision 2, intended to be applicable only for Westinghouse and Framatome ANP fuel types? If the intent is for other fuel types, please provide a discussion regarding how applicability determinations will be made and the process for determining the need for prior NRC approval.

Response:

The methodology discussed in VEP-FRD-42, Revision 2 is supported by extensive nuclear design predictions that encompass various evolutionary changes in fuel design features for Westinghouse fuel. Such predictions have been made for more than 40 reload cores, loaded in both North Anna and Surry reactors. Although the intended extension of this methodology is for the analysis of Framatome ANP fuel, the methodology is sufficiently robust for use on any fuel product with similar features. The methodology has several key elements, none of which are inherently dependent upon a specific fuel design or manufacturer. These key attributes of the methodology are:

- Analysis framework in which safety analyses establish the acceptable values for reload core key parameters, while nuclear and fuel design codes confirm each core's margin to the limits
- Use of bounding key parameter values in reference safety analyses
- Recurrent validation of nuclear design analytical predictions through comparison with reload core measurement data
- Representation of key fuel features via detailed inputs in core design and safety analysis models
- Fuel is modeled using approved critical heat flux (CHF) correlations demonstrated to be applicable and within the range of qualification

The Dominion reload design methodology focuses upon determining appropriately conservative values for two types of parameters: 1) the bounding value for key parameters assumed in the safety analyses and 2) the values for these same key parameters calculated for each reload core. The first parameter set constitutes the allowable limits for which the existing safety analyses remain valid. The reload values are determined for each specific core with the objective of confirming that they remain within the limit values. Application of this methodology to alternate fuel types would be accomplished in a fashion that preserves this fundamental approach. Prior to the use of

the Dominion nuclear reload methodology for other fuel types, it is necessary to confirm that the impact of the fuel design and its specific features can be adequately modeled with the Dominion nuclear design and safety analysis codes. This includes comparison with appropriate benchmark data to confirm the capability to model the specific fuel features and to determine the inherent accuracy of such predictions. Results of these comparisons would also be used to determine whether any changes are needed in uncertainties that are applied to the nuclear calculations. If the features of an alternate fuel design can be modeled with comparable accuracy to the existing models and fuel design and require no change in the applied uncertainty factors, the applicability of the nuclear design portion of the methodology is established. This approach confirms that there should be no significant effect upon calculated values of reload key parameters. To determine applicability of safety analysis codes for analysis of alternate fuel products, a similar modeling capability assessment would be performed. This assessment would involve incorporating the appropriate detailed fuel design inputs into safety analysis code calculations and verifying that existing codes and models conservatively model the fuel behavior. This would be accomplished either by direct evaluation of the key phenomena or comparison to available vendor calculation results. The need to obtain prior NRC approval for these changes is governed by the requirements of 10 CFR 50.59, which in Sections (a)(2) and (c)(2)(viii) includes provisions that are relevant to methodology changes. If the changes necessary to accommodate another fuel product required changes to the reload methodology of VEP-FRD-42, Revision 2, these would be submitted for prior NRC review and approval.

Question 2:

The licensee states that the minor changes in Framatome ANP fuel features that could affect safety analysis design inputs are within the modeling capability of Dominion safety and core design analysis codes. Please verify that Framatome ANP fuel features are within all restrictions and limitations of Dominion safety and core design analysis codes.

Response:

Core Design Models

From a core design perspective, the differences in modeling Framatome ANP fuel relative to Westinghouse fuel are small and are accommodated using model input parameters. These differences are similar in magnitude to incremental changes in Westinghouse fuel over time, which have been successfully modeled. Minor changes include spacer grid differences, a slight increase in fuel density, and a slight difference in the position of the fuel stack. The grid differences are primarily due to the presence of intermediate flow mixer grids. In the PDQ and NOMAD models, grids are not explicitly modeled, but are homogenized over the entire length of the fuel stack. The effect of more grid material (primarily zirconium) is directly modeled in PDQ via input parameters (treated as nuclides) representing grid material and moderator

displacement. The macroscopic cross section effect is transferred to the NOMAD model from PDQ. Similarly, cross sections in the PDQ model are a function of fresh fuel isotopic content; therefore, the density effects are also directly modeled.

Minor changes in fuel alignment have occurred in the past due to evolutionary changes in Westinghouse fuel products, such as the incorporation of protective lower grids. If there is a significant shift in the relative alignment of the burnable poison (BP) and the fuel, the burnable poison position is directly modeled by axially volume weighting the BP input in the axial nodes where the BP/fuel boundary changes. Comparison of measured and predicted Framatome ANP lead test assembly (LTA) axial and integral power distributions over three cycles of operation provides direct confirmation of the accuracy of the axial weighting, grid modeling, and fuel density modeling techniques.

RETRAN Models

In preparation for application of the Dominion RETRAN model to Framatome ANP fuel, specific card (record) overlays to the RETRAN input cards were developed. These overlays were developed such that appending them to the end of the current, Westinghouse fuel based model creates a Framatome ANP-specific RETRAN model.

Fuel properties

The Framatome ANP overlays were developed from fuel and clad properties data supplied by Framatome ANP which are consistent with those used in the approved Framatome ANP safety analysis models. Formal documents developed under the Framatome QA program were developed to transmit this data. Fuel properties covered included:

- Material properties of the three conductor materials (the fuel pellet, the pellet-cladding helium gap, and the M5 cladding)
  - Thermal conductivity
  - Volumetric heat capacity
  - Thermal linear expansion coefficient

These data were converted into the RETRAN input structure. Plots of the data, the analytical equations used to develop the data, and graphical and numerical comparisons were presented of the Framatome ANP data to the corresponding data in:

- the existing W fuel based model
- The International Nuclear Safety Center (INSC) Material Database, Argonne National Laboratory for the US Department of Energy
- NUREG/CR-6150 (MATPRO)

Generally, only minor differences in the data were observed. The most significant property differences are those associated with the M5 versus ZIRLO cladding.

Core Geometry Input

The Framatome ANP overlays were developed from Framatome ANP supplied dimensional data for the Framatome ANP fuel assemblies. All dimensional data were transmitted via documentation that was formally prepared and reviewed under Framatome ANP's 10 CFR 50 Appendix B QA program. Input changes were developed in the following areas:

- Core bypass geometry
  - Volume
  - Flow area
  - Flow diameter
- Active core geometry
  - Volume
  - Flow area
  - Flow diameter
- Reactor vessel flow path length and area
- Reactor vessel form loss coefficients
- Reactor core target pressure drops
- Active core inlet mass flow rate
- Geometry of the active core heat conductors

The calculation of each RETRAN input was documented in a reviewed engineering calculation and prepared in accordance with Dominion's 10 CFR 50 Appendix B Quality Assurance Program. The engineering calculation presents detailed comparisons of the Framatome ANP overlay parameters to the base model parameters in tabular format. The parameter changes represented minor adjustments with respect to the existing inputs.

Steady-state initializations were run with and without the Framatome ANP overlays to ensure adequate convergence of the new models. Detailed comparisons of the steady-state initialization results were presented in the engineering calculation in tabular format. Review of these results showed that there are only minor differences in the Westinghouse Fuel Based and Framatome ANP Fuel based models.

The modeling changes associated with Framatome ANP fuel fall within the restrictions and limitations of the Dominion core design and safety analysis codes.

Question 3:

Use of Framatome ANP fuel will require changes to various computer model inputs. Please discuss how the practices of NRC Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses", are applied in making these model changes.

Response:

General comment

The scope and applicability of GL 83-11 Supplement 1 is discussed in Attachment 1 to GL 83-11. An excerpt relevant to this discussion is as follows:

> "This attachment presents a simplified approach for qualifying licensees to use NRC-approved analysis methods. Typically, these methods are developed by fuel vendors, utilities, national laboratories, or organizations such as the Electric Power Research Institute, Incorporated (EPRI). To use these approved methods, the licensee would institute a program (e.g., training, procedures) that follows the guidelines below and notify the NRC that it has done so.
>
> The words 'code' and 'method' are used interchangeably within this document, i.e., a computer program. In many cases, however, an approved method may refer not only to a set of codes, an algorithm within a code, a means of analysis, a measurement technique, a statistical technique, etc., but also to selected input parameters which were specified in the methodology to ensure conservative results. In some cases, due to limitations or lack of appropriate data in the model, the code or method may be limited to certain applications. In these cases, the NRC safety evaluation report (SER) specifies the applicability of the methodology."

Dominion is proposing to apply the existing methodology of VEP-FRD-42 to the analysis of Framatome ANP fuel. Therefore GL 83-11, which involves code and methodology changes, is not directly applicable. However, the principles outlined in Attachment 1 to the GL have been followed in the development of Framatome ANP specific models (input changes) for use with existing, approved codes and methods. The process of Framatome ANP specific model development will be discussed in that context.

Dominion has established and uses a formal GL 83-11 program. Dominion notified the NRC of the establishment of this program in Reference 3.1. This program addresses all of the elements of GL 83-11, Supplement 1, Attachment 1 identified below:

- Application Procedures
- Training and Qualification of Licensee Personnel
- Comparison Calculations
- Quality Assurance and Change Control
- Error/Problem Reporting

Dominion's reload analysis methodology as set forth in VEP-FRD-42 has been developed and qualified in accordance with these principles. For example:

Application Procedures

Specific analytical steps for performing a reload analysis are outlined in the Nuclear Core Design (NCD) Manual and the Safety Analysis Manual (SAM). The NCD Manual is structured such that the calculational process is transparent to fuel type. Specific NCD code input varies according to fuel type as necessary (i.e., grid size differences, grid material difference, etc.). Detailed techniques for determining model input are provided in the NCD Manual and are supplemented by model setup calculations for previous fuel types, and by evaluation of proposed fuel changes in an operational impact assessment. The operational impact assessment is mandated by a departmental Implementing Procedure, which requires evaluations of proposed core changes in light of SOER 96-02.

The Safety Analysis Manual provides detailed calculational instructions for providing reload-specific thermal hydraulic evaluations as well as a chapter of guidance for the performance of analyses of the specific accidents presented in Chapters 14 and 15 of the Surry and North Anna UFSARs, respectively. Typically, accident reanalyses are not performed for core reloads, in that the key analysis parameters are found to be bounded by the assumptions in the accident analyses.

Quality Assurance/Change Control

Core Physics Models – The answer to Question 2 deals with the Framatome ANP changes of importance to the core design models. The changes were identified and evaluated in an operational impact assessment, and specific input changes were determined for Framatome ANP Lead Test Assembly (LTA) modeling using the same techniques used for other fuel types.

RETRAN Models - In preparation for application of the Dominion RETRAN model to Framatome ANP fuel, specific card (record) overlays to the RETRAN input cards were developed. These overlays were developed such that appending them to the end of the current, Westinghouse fuel based model creates a Framatome ANP-specific RETRAN model.

Specific changes modeled were discussed in detail in the Response to Question 2.

The Framatome ANP overlays were developed from the following data:

- Framatome ANP supplied fuel and clad properties data that are consistent with those used in the approved Framatome ANP safety analysis models. Formal documents developed under the Framatome QA program were developed to transmit this data.

- Framatome ANP supplied dimensional data for the Framatome ANP fuel assemblies. All dimensional data was transmitted via documentation that was formally prepared and reviewed under Framatome ANP's 10 CFR 50 Appendix B QA program.

Comparison Calculations

Previously submitted topical reports for PDQ Two Zone Models, NOMAD, and TIP/CECOR contain extensive model benchmarking information. In addition, the accuracy of power distribution predictions for Framatome ANP LTA fuel has been documented for three cycles of operation.

Dominion's RETRAN model has been benchmarked against the following items:

- Westinghouse analyses of record as published in the Surry and North Anna FSAR's in the 1970's and 1980's - see Section 5.2 of VEP-FRD-41A.
- Plant transient data, including:
  - Surry and North Anna pump coastdown tests - see Section 5.3 of VEP-FRD-41A
  - North Anna Unit 1's cooldown and safety injection transient September 25, 1979- See Section 5.3.3 of VEP-FRD-41A.
  - North Anna Unit 1's July 1987 Steam Generator Tube Rupture-see Section 3.2 of Attachment 1 to Letter 93-505, Supplemental Information on the RETRAN NSSS Model, August 10, 1993.
  - Westinghouse LOFTRAN calculations for the following:
    - Reactor trip with turbine trip
    - Turbine trip without direct reactor trip
    - Simultaneous loss of 3 reactor coolant pumps
    - See VEPCO Letter No. 376A, August 24, 1984.

These benchmark calculations have been studied and understood and support the conclusion that the Dominion RETRAN model provides a realistic representation of the Surry and North Anna reactor plants. Conservative results are ensured when the RETRAN model is used for licensing basis analyses through the use of appropriate input assumptions governing availability and performance of systems and components, core reactivity coefficients, and uncertainties in initial conditions.

Reference:

3.1 Virginia Power Letter to the NRC (Serial No. 00-087), dated March 15, 2000, Qualifications for Performing Safety Analyses, Generic Letter 83-11, Supplement 1.

Question 4:

The Dominion Topical Report on Reload Methodology (VEP-FRD-42, Revision 2) includes four computer codes or code modifications which have been implemented for use under the provisions of 10 CFR 50.59:

- PDQ Two Zone - replaced PDQ Discrete Model and the FLAME Model (Transmitted via Ref. 2 and 3 in VEP-FRD-42)
- NOMAD - was significantly modified (transmitted in Ref. 5 in VEP-FRD-42)
- TIP/CECOR - (Transmitted via Ref. 3 in VEP-FRD-42)
- RETRAN - code modifications (Transmitted via Ref. 7 in VEP-FRD-42)

References 2, 3 and 5 in VEP-FRD-42, Revision 2, and an additional letter not referenced in this topical (dated March 1, 1993) requested NRC review and approval of the associated topical reports for the first three codes listed. Dominion (VEPCO at the time) also recognized that these would need NRC approval because North Anna and Surry are COLR plants. For RETRAN, no review was requested, and the transmittal letter was for NRC information only. As such,

a. Have those topical reports/codes and code modifications been reviewed and approved for use by the NRC staff? If so, please provide a reference to the staff SERs. If not, then codes and models will need to be reviewed and approved to permit use in the COLR.

b. Have they been used by Dominion as part of the Reload Design Methodology? If so, why is their use acceptable and not a violation of the requirements for implementing a COLR? Generic Letter 88-16 requires that NRC approved methodology be referenced in the COLR, and VEP-FRD-42, Revision 1 is referenced in the COLR. VEP-FRD-42, Revision 1, and therefore the COLR does not reflect what Dominion is currently using as part of its Reload Methodology.

c. Please submit Technical Specification changes to incorporate references to actual methodology being used.

d. What procedures and controls do you use on the application of computer codes and models for core design and safety analysis? In other words, how does the core designer or safety analyst know he or she is using the right tools?

Response to 4a:

<u>PDQ Two-Zone Model</u>

The PDQ Two-Zone Model was transmitted via References 4.1 and 4.2:

Reference 4.1 requested approval of the 3-D coarse mesh PDQ model (the two-zone model) by the end of the 1st Quarter, 1991 to support the use of axially zoned flux

suppression inserts (FSI's) in Surry Unit 1 Cycle 12.

Reference 4.2 reiterated the need for the 3D capability, to support FSI's, although first use had shifted to Cycle 13. We noted that to support the planned use of FSI's in Cycle 13 would require approval of the topical by the end of the 1st Quarter, 1993. Since the NRC review schedule would not support this, we proposed implementation of the methodology via 10 CFR 50.59 in advance of formal NRC approval of the reports. As noted in Reference 4.2, telephone conversations were held with the Staff on October 7 and 14, 1992 to discuss the 10 CFR 50.59 approach. Although the NRC could not concur with the specific application without formal review, the staff agreed with the use of 10 CFR 50.59 evaluations where applicable. Reference 4.2 documented these discussions. Dominion's request for formal review of the topicals was not withdrawn, although these changes were implemented via 10 CFR 50 .59.

On March 1, 1993 Dominion submitted Topical Report VEP-NAF-1, Supplement 1, entitled, "The PDQ Two-Zone Model," again for review and approval. The Supplement describes a coarse mesh 2-D model that is closely related to and used in conjunction with the 3-D model. We again stated our intent to implement the code via 10 CFR 50.59 prior to NRC review and approval, but requested concurrent review of the VEP-NAF-1 and Supplement 1.

The 10 CFR 50.59 approach to changing "elements of a methodology" as defined in NEI 96-07, Rev. 1 and endorsed by USNRC Regulatory Guide 1.187 is applicable in the case of the PDQ Two-Zone models. We refer specifically to NEI 96-07 Section 4.3.8, entitled, "Does the Activity Result in a Departure from a Method of Evaluation Described in the UFSAR Used in Establishing the Design Bases or in the Safety Analyses?"

The relevant discussion is as follows:

"... The following changes are not considered departures from a method of evaluation described in the UFSAR:

- Departures from methods of evaluation that are not described, outlined or summarized in the UFSAR (such changes may have been screened out as discussed in Section 4.2.1.3).

- Use of a new NRC-approved methodology (e.g., new or upgraded computer code) to reduce uncertainty, provide more precise results or other reason, provided such use is (a) based on sound engineering practice, (b) appropriate for the intended application and (c) within the limitations of the applicable SER. The basis for this determination should be documented in the licensee evaluation.

- Use of a methodology revision that is documented as providing results that are essentially the same as, or more conservative than, either the previous revision of the same methodology or another methodology previously accepted by NRC through issuance of an SER".

Subsection 4.3.8.1 of NEI 96-07 provides guidance for making changes to one or more elements of an existing method of evaluation used to establish the design bases or in the safety analyses. Specifically,

"4.3.8.1 Guidance for Changing One or More Elements of a Method of Evaluation

The definition of 'departure ...' provides licensees with the flexibility to make changes under 10 CFR 50.59 to methods of evaluation whose results are 'conservative' or that are not important with respect to the demonstrations of performance that the analyses provide. Changes to elements of analysis methods that yield conservative results, or results that are essentially the same, would not be departures from approved methods.

Conservative vs. Nonconservative Results

Gaining margin by changing one or more elements of a method of evaluation is considered to be a nonconservative change and thus a departure from a method of evaluation for purposes of 10 CFR 50.59. Such departures require prior NRC approval of the revised method. Analytical results obtained by changing any element of a method are 'conservative' relative to the previous results, if they are closer to design bases limits or safety analyses limits (e.g., applicable acceptance guidelines). For example, a change from 45 psig to 48 psig in the result of a containment peak pressure analysis (with design basis limit of 50 psig) using a revised method of evaluation would be considered a conservative change when applying this criterion. In other words, the revised method is more conservative if it predicts more severe conditions given the same set of inputs. This is because results closer to limiting values are considered conservative in the sense that the new analysis result provides less margin to applicable limits for making potential physical or procedure changes without a license amendment.

In contrast, if the use of a modified method of evaluation resulted in a change in calculated containment peak pressure from 45 psig to 40 psig, this would be a nonconservative change. That is because the change would result in more margin being available (to the design basis limit of 50 psig) for the licensee to make more significant changes to the physical facility or procedures.

Essentially the Same

Licensees may change one or more elements of a method of evaluation such that results move in the nonconservative direction without prior NRC approval, provided the revised result is 'essentially the same' as the previous result. Results are 'essentially the same' if they are within the margin of error for the type of analysis being performed. Variation in results due to routine analysis sensitivities or calculational differences (e.g., rounding errors and use of different computational platforms) would typically be within the analysis margin of error

and; thus, considered 'essentially the same.' For example, when a method is applied using a different computational platform (mainframe vs. workstation), results of cases run on the two platforms differed by less than 1%, which is the margin of error for this type of calculation. Thus, the results are essentially the same, and do not constitute a departure from a method that requires prior NRC approval.

The determination of whether a new analysis result would be considered 'essentially the same' as the previous result can be made through benchmarking the revised method to the existing one, or may be apparent from the nature of the differences between the methods. When benchmarking a revised method to determine how it compares to the previous one, the analyses that are done must be for the same set of plant conditions to ensure that the results are comparable. Comparison of analysis methods should consider both the peak values and time behavior of results, and engineering judgment should be applied in determining whether two methods yield results that are essentially the same."

In the case of the PDQ Two-Zone models, the governing topical report documents extensive comparisons of these models to measured data and demonstrates that the Nuclear Reliability Factors (NRFs) documented in Topical Report VEP-FRD-45-A, "Nuclear Design Reliability Factors" remain bounding. Therefore, from a reload analysis perspective, the results with these new tools (elements of the VEP-FRD-42 methodology) are "essentially the same" and implementation via 10 CFR 50.59 is permissible.

NOMAD

Dominion uses the NOMAD 1-D core physics code to perform both reload design analyses and core operation evaluations. Use of this code and its associated model was approved by the NRC on March 4, 1985, with its issuance of Acceptance for Referencing of Licensing Topical Report VEP-NFE-1-A, "The VEPCO NOMAD Code and Model." As stated in VEP-NFE-1-A, verification of and improvements to the NOMAD code and model would continue to be made as more experience was gained in the application of the model to the units at the Surry and North Anna Power Stations. The primary reload safety analysis use of NOMAD is as one of the analytical tools (elements) of the Relaxed Power Distribution Control and Constant Axial Offset Control Methodologies. Use of NOMAD within the framework of those methodologies was not altered by the model update.

Letter 96-319 (Reference 4.4) documented the NOMAD code and model update. These changes were necessitated by the transition to 3-D PDQ (see discussion above). The NOMAD flux solution and axial nodalization were not altered. The updated NOMAD model was qualified against plant data and its fidelity to the data was found to be as good as or better than that of the original code and model. The Nuclear Reliability Factors currently applied in reload analyses were shown to remain appropriate and reload results obtained with the updated model are essentially the same as those

obtained with the previous version. As such, the code and model updates do not constitute a change in the approved methodology of VEP-FRD-42 or the Code as described in VEP-NFE-1-A (see the discussion of NEI 96-07, Section 4.3.8, above).

TIP/CECOR

The CECOR code was reviewed and approved generically by the NRC and is documented in CENDP-153-P, Rev. 1-P-A. TIP-CECOR uses the same solution algorithm as CECOR, but is adapted to accept input from movable incore detectors as opposed to fixed detectors. Comparisons with experiments and development of uncertainties for TIP-CECOR are consistent with the CECOR topical report and with VEP-FRD-45-A, the Nuclear Design Reliability Factor topical report.

Additionally, comparisons between TIP/CECOR predictions and those from the previously approved INCORE code revealed that the two codes produce essentially the same results. Therefore, the adoption of TIP/CECOR as a replacement for INCORE represented a change to an element of the reload methodology that can be implemented via 10 CFR 50.59 under the guidance of NEI 96-07. Additionally, qualification of TIP/CECOR for Dominion use met the intent of the programmatic elements of Generic Letter 83-11, Supplement 1, Attachment 1.

RETRAN

Dominion's reload methodology incorporates the RETRAN-02 code. RETRAN-02 was generically approved by the NRC in a letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, "RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

Dominion's RETRAN models and capability were approved by the NRC in a letter from C. O. Thomas (NRC) to W. L. Stewart, Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, "Virginia Power Reactor System Transient Analyses Using the RETRAN Computer Code," April 11, 1985.

The RETRAN Topical SER recognized that model maintenance activities would be performed under the control of the utility 10 CFR 50 Appendix B QA program. The VEP-FRD-41 SER emphasized that the NRC viewed the primary objective of the report was to demonstrate Dominion's general capability for performing non-LOCA accident analyses:

- "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/M0D03 computer code."

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

- "Based on the VEPCO RETRAN model and the qualification comparisons ..., the staff concludes that VEPCO has demonstrated their capability to analyze non-LOCA initiated transients and accidents using the RETRAN computer code."

Dominion has demonstrated that use of our models with RETRAN-02 versus RETRAN01 is an equivalent methodology. In a letter (Serial No. 85-753) dated November 19, 1985, Dominion showed that results with RETRAN-02 versus RETRAN-01 were essentially identical except for nonequilibrium pressurizer pressure behavior, where significant improvements were made in the RETRAN-02 solution scheme. This letter requested approval to use RETRAN-02 by February 1986 to support upcoming licensing applications; however, no formal NRC Staff review has been performed to date.

The VEP-FRD-41 SER further stated:

> "The staff requires that all future modifications of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures."

Dominion followed these requirements in updating our RETRAN models. Updated models and the qualification results were documented consistent with our 10 CFR 50 Appendix B, QA program and provided to the NRC for information in letter (Serial No. 93-505) dated August 10, 1993.

It should be noted that the new model results were very similar to those obtained with the old models. No margins in key analysis results were gained. The new models have improved, more mechanistic Doppler reactivity feedback models and more detailed main steam system modeling. This resulted in some changes which were documented and well understood (see Letter 93-505).

While this model upgrade was not a code change, the qualification, documentation and implementation of these new models was done in a manner that meet the programmatic elements of Generic Letter 83-11, Supplement 1.

RETRAN models are code input, and represent an element of Dominion's RETRAN methodology as discussed in NEI 96-07. Because the results obtained with the new models met the "essentially the same" test, we believe that these model upgrades do not represent a change to a method of analysis as defined in 10 CFR 50.59 (c)(2)(viii).

Therefore, VEP-FRD-41A remains the applicable reference for Dominion's approved RETRAN capability.

Response to 4b:

Dominion has used these codes as part of its reload design methodology. However, with respect to the COLR, Dominion notes that the codes above are not listed in the COLR methods reference list in the Technical Specifications, because they do not represent analytical methods that determine core-operating limits. Dominion considers this treatment to be consistent with the guidance in Generic Letter 88-16, which discusses "methodology for determining cycle-specific parameter limits." PDQ and NOMAD represent tools that predict core performance and core parameter values, which are then compared to core operating limits. Similarly, TIP/CECOR processes core surveillance data to confirm that core parameters are behaving as predicted by PDQ and NOMAD and that the operating limits are continuously met. RETRAN provides transient system thermal hydraulic responses that are used in conjunction with the COBRA and LYNXT codes to perform transient DNB calculations for Chapter 15 accidents. The Nuclear Enthalpy Rise Hot Channel Factor ($F\Delta H$) limit in the COLR is established using COBRA and LYNXT in conjunction with the Reactor Core Safety Limits, and not by RETRAN. Similarly the total peaking factor limit (FQ) in the COLR is established by the referenced, approved LOCA methodology, not by the neutronics codes.

Although VEP-FRD-42, Rev. 1 was not formally revised to reflect changes to these codes and models, it was updated via supplements sent with references 4.3 and 4.4. In neither case was there any NRC request or directive given to revise the topical to incorporate these changes. In particular, Reference 4.3 summarizes several changes relevant to VEP-FRD-42, Rev. 1-A and states:

> "These changes have effectively superseded portions of VEP-FRD-42, Rev. 1-A. Supplement 1 to VEP-FRD-42, Rev. 1-A (enclosed) consolidates and summarizes these changes for your information."

Dominion therefore, considers that these supplements are part of VEP-FRD-42, Rev. 1 and that VEP-FRD-42, Rev. 1 continues to represent Dominion's reload methodology for Westinghouse fuel. It is not Dominion's intention to change our reload methodology as outlined in VEP-FRD-42, Rev. 2 under the provisions of 10 CFR 50.59. However, there are analytical tools, which form elements of the methodology, which can be changed under the provisions of 10 CFR 50.59(c)(2)(viii) as discussed in NEI 96-07 Section 4.3.8.

It is Dominion's intent to apply this guidance of NEI 96-07, Rev. 1, as endorsed by Regulatory Guide 1.187, in determining the applicability of 10 CFR 50.59 to proposed changes to analytical tools which support our reload methodology. The qualification and benchmarking of new elements of the methodology for making this determination will be performed and documented in accordance with the provisions of our quality assurance program.

Response 4c:

The code/model updates discussed in the response to 4a and 4b, above, have been incorporated into VEP-FRD-42, Rev. 2 by referencing the appropriate documentation. Since VEP-FRD-42 is currently referenced in the Technical Specifications no additional changes are necessary.

Response 4d:

A. Production Codes

Core designers and safety analysts have access to a controlled Production Code List.

The Production Code List includes the code version, the effective date, a reference to the applicable code file (which contains the software development, qualification and release documentation), the Code Manager and applicable references documenting the qualification and implementation of the code. This documentation is prepared and peer reviewed in accordance with applicable quality assurance procedures. (The Code Manager is an individual designated by the Department Manager to ensure the required code documentation is completed for new codes and changes to existing codes).

Engineers refer to the List when referencing the name and version of a computer code used to perform design calculations. This procedure ensures that any computer code referenced in a Calculation is available for production work and that the appropriate version of the code is used.

The code version and release date is printed on the output header of all computer calculations. Computer code versions are required to be included as formal references in the engineering calculations which document production applications (e.g., reload calculations).

Dominion software control procedures require that qualified code users be notified when modifications to a code are made.

B. Models

A procedure governs the development and control of Nuclear Analysis and Fuel models. A model is defined as a standardized, controlled set of plant specific input to a computer code. The physical model consists of one or more electronic input files. Models are treated as controlled documents.

Production model input files are write-protected with only authorized personnel given change authority, or monitored in such a way that the Model Manager can determine whether the files have been modified. Model users are responsible for ensuring that the appropriate model is used correctly in an analysis.

Recent changes to applicable production codes and models are discussed as part of the reload design initialization process (see VEP-FRD-42, Rev. 2 Section 3.2.1).

References:

4.1 Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2 Topical Report–PDQ Two Zone Model," Serial No. 90-562, October 1, 1990.

4.2 Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2 Topical Report Use Pursuant to 10 CFR 50.59," Serial No. 92-713, November 25, 1992.

4.3 Letter from M. L. Bowling (Virginia Electric and Power Company) to U. S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2, Supplement 1 to VEP-FRD-42 Revision 1-A, Reload Nuclear Design Methodology Modifications," Serial No. 93-723, December 3, 1993.

4.4 Letter from S. P. Sarver (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, North Anna Power Station Units 1 & 2, Surry Power Station Units 1 and 2, Supplemental Information for the NOMAD Code and Model, Reload Nuclear Design Methodology, and Relaxed Power Distribution Control Methodology Topical Reports," Serial No. 96-319, November 13, 1996.

Question 5:

VEP-FRD-42, Revision 1 included the code or model used to calculate each of the Key Analysis Parameters within the sections of the report, which discussed each parameter. This is not done in Revision 2. Please provide a listing of the code or model used to calculate each Key Analysis Parameter used in the reload analysis methodology. Does the use of Framatome ANP fuel introduce any new Key Analysis Parameters?

Response:

The models currently used to calculate each parameter are provided below, in terms of the key parameter list from Table 2 of VEP-FRD-42, Revision 2. It was determined that the Framatome ANP fuel required the addition of one key parameter (item 28 below). This parameter, maximum linear heat generation rate versus burnup, is used in the NRC-approved Framatome ANP methodology for cladding stress evaluations. The code or model currently used to calculate each parameter is listed in the following table. The name PDQ refers to the PDQ two-zone 3D model.

KEY ANALYSIS PARAMETER	CODE OR MODEL
1) Core Thermal Limits (F)	COBRA/LYNXT
2) Moderator Temperature (Density) Coefficient (NS)	PDQ
3) Doppler Temperature Coefficient (NS)	PDQ
4) Doppler Power Coefficient (NS)	PDQ
5) Delayed Neutron Fraction (NS)	PDQ
6) Prompt Neutron Lifetime (NS)	NULIF
7) Boron Worth (NS)	PDQ
8) Control Bank Worth (NS)	PDQ/NOMAD
9) Rod Worth Available for Withdrawal (S)	PDQ/NOMAD
10) Ejected Rod Worth (S)	PDQ/NOMAD
11) Shutdown Margin (NS)	PDQ/NOMAD
12) Boron Concentration for Required Shutdown Margin (NS)	PDQ
13) Reactivity Insertion Rate due to Rod Withdrawal (S)	PDQ/NOMAD
14) Trip Reactivity Shape and Magnitude (NS)	PDQ/NOMAD
15) Power Peaking Factors (S)	PDQ/NOMAD
16) Maximum F_Q * P (S)	PDQ/NOMAD
17) Radial Peaking Factor (S)	PDQ
18) Ejected Rod Hot Channel Factor (S)	PDQ/NOMAD
19) Initial Fuel Temperature (F)	PAD /TACO3
20) Initial Hot Spot Fuel Temperature (F)	PAD /TACO3
21) Fuel Power Census (NS)	PDQ/NOMAD
22) Densification Power Spike (F)	PAD /TACO3
23) Axial Fuel Rod Shrinkage (F)	PAD /TACO3
24) Fuel Rod Internal Gas Pressure (F)	PAD /TACO3
25) Fuel Stored Energy (F)	PAD /TACO3
26) Decay Heat (F)	ANSI ANS-1979 ANSI ANS-1971
27) Maximum Linear Heat Generation Rate (LHGR) (S)	PDQ/NOMAD
28) Maximum LHGR Vs. Burnup (F)	PDQ/NOMAD

Parameter Designation
S: Specific
NS: Non-specific
F: Fuel Performance and Thermal-Hydraulics Related

Question 6:

Regarding Section 2.2.2.1 - Reactivity Coefficients and Defects:

a. Revision 1 discussed a set of four calculations performed to determine temperature and power coefficients at HZP, and an additional four cases to determine the coefficients at power. The Revision 2 methodology includes two cases at ±5°F or ±10°F about the nominal temperature for the temperature coefficients, and two cases at ±5% or ±10% about the nominal power for the power coefficients. Please provide the technical basis supporting this change in methodology.
b. The cases at ±10°F or ±10% were not included in Revision 1 methodology. Please provide the technical basis for these cases.
c. Please discuss the procedures or processes by which the Dominion analyst determines whether to use ±5 or ±10.

Response:

Parts a and b:

Two cases are used for each coefficient. Four cases are still required to determine all three coefficients (ITC, DTC, and MTC). The discussion of HZP coefficients simply reflects the calculation of individual coefficients because all three coefficients are not required at all conditions.

The choice of ±5°F or ±10°F does not have a significant effect on most coefficients (particularly the DTC) because they behave nearly linearly versus temperature over this small a temperature range. Mathematically, as long as the defect is no more complex than a quadratic function of temperature, there is no effect at all in the choice of temperature difference, provided that a centered difference is used. In general, ±5°F is used for all but the DTC. The DTC is always small in magnitude and, therefore, is more susceptible to K-effective convergence tolerance. A range of ±10°F reduces the influence of convergence tolerance. The defining methodology features in the calculation of coefficients are:

1) changing only the variable(s) of interest (fuel temperature, moderator temperature or both, or core power), and
2) the use of a centered difference about the desired point over a range large enough to get a significant change but small enough that the answer still represents the derivative.

As indicated, valid technical reasons may arise which lead to a change in the exact choice of temperature difference or the specific input used to calculate a coefficient. The above discussion also applies to the at-power ITC, DTC, and MTC cases. As in the case of the temperature coefficients, the use of ±10% power for power coefficients does not represent a significant change due to the nearly linear nature of the power coefficients versus power. The primary reason for using ±10% is to minimize 3D-model

THF convergence tolerance on the coefficients. We do not view these specific input changes as changes to the reload methodology.

Part c:

The analyst uses standard techniques described in the core design procedures. These techniques, including the choice of temperature or power change are not changed unless a valid new technical reason arises. A change to the standard technique requires peer review and management approval.

Question 7:

Section 2.3 - Analytical Model and Method Approval Process was added in Revision 2 and discusses the acceptable means by which either analytical models or methods can achieve approved status for use in reload methodology. The first method listed allows reload methodology changes to be implemented in accordance with the provisions of 10 CFR 50.59. The NRC staff does not accept this option as a means to change reload methodology. Implementation under 10 CFR 50.59 would require that new or different methods have already been reviewed and approved by the NRC for the intended application.

Response:

Dominion did not and does not change the reload methodology as outlined in VEP-FRD-42, Rev. 2 under the provisions of 10 CFR 50.59. However, there are analytical tools, which form elements of the methodology, which can be and have been changed under the provisions of 10 CFR 50.59(c)(2)(viii) as discussed in NEI 96-07, Section 4.3.8 (see our response to Question 4, above for further discussion).

The qualification and benchmarking of new or revised inputs or elements of the methodology are performed and documented in accordance with the provisions of our quality assurance program. Dominion then applies the guidance of NEI 96-07, Rev. 1, as endorsed by Regulatory Guide 1.187, in determining the applicability of 10 CFR 50.59 to the proposed changes.

This practice is analogous to that used for previous model updates prior to the issuance of NEI 96-07. For example, application of the 50.59 process to the PDQ model changes (and later the NOMAD and TIP/CECOR changes) was focused on the key issues of whether the change created an unreviewed safety question (USQ), maintaining the "margin of safety," and whether the change involved a change to a Technical Specification. The SER for prior model approvals were reviewed to ascertain the NRC basis for previous approval. In particular, the PDQ Two Zone model was found to be an equivalent replacement of the previous models used for the same purposes inside the existing reload methodology framework and hence the change was determined not to be a USQ. The validation process was at least as broad as for the earlier models, with

far more available data. Although the data supported reductions in some uncertainty factors, the existing uncertainty factors were maintained (no reduction in margin of safety). The process used is functionally equivalent to changing elements of the method under the current 50.59 process. This was an internal review process using the same criteria as the original review as described in associated NRC SERs and using appropriate screening techniques under 50.59. Finally, since PDQ was not directly referenced in the COLR, implementation of the model upgrades did not require a change to the Technical Specifications. As discussed in the response to Question 4b, PDQ is not listed among the analytical methods supporting the COLR in Technical Specifications since it is not used to determine values for core operating limits.

The process for qualifying the new RETRAN models was analogous. The qualification tests performed included comparisons between the new and old models as well as to plant transient data. The qualification supported the conclusion that the new models were an equivalent replacement of the transient analysis element of Dominion's reload methodology.

Question 8:

Regarding Section 3.3.2 - Safety Analysis Philosophy, please discuss the procedural or process type of guidance available to the Dominion analyst for determining whether to evaluate or reanalyze a particular transient. This would be important if a key reload parameter value exceeds the current limit in the reference safety analysis, or if the parameter impact is difficult to quantify.

Response:

Quantitative evaluation of a small departure from a parameter limit of parameter limits may be made in one of several ways. First, if the interplay between the various key safety parameters in determining accident response is well defined, margin in one parameter may be used to offset a small departure in another parameter. A second method of quantitative evaluation involves using tradeoffs of known sensitivities. This process is best defined by presenting some examples:

- Studies performed by Dominion and others have shown that a key parameter in determining the severity of the core power response to a rod ejection event is the ejected rod worth in units of dollars (delta k/k ejected rod worth/delayed neutron fraction). For the case of a cycle-specific departure from the minimum delayed neutron fraction, the safety analyst can take advantage of available cycle-specific margin in ejected rod worth by showing that the ejected worth in dollars is less than the worth assumed in the safety analysis.

- For some reload cycles where small departures (a few percent) from an accident specific limit occur, these studies can be used to show that margin in another key parameter that influences the same accident offsets the departure. For example, the

end of cycle (EOC) least negative moderator temperature coefficient is a key safety parameter for the rod ejection accident, although its influence is relatively weak. For one recent cycle, a small departure from the limit for this parameter was shown to be offset by large margins in the calculated ejected rod worth, which strongly influences the accident analysis results. These sensitivities are documented in VEP-NFE-2-A.

The general philosophy followed in performing an accident evaluation as opposed to a reanalysis is that the analyst must be able to clearly demonstrate that the results of an analysis performed with cycle-specific input would be less severe than the results of the reference analysis. In other words, in performing the evaluation, no credit is taken for margin between the reference analysis results and the design basis criteria, even though this margin may be substantial. In some cases the analyst and/or reviewer may determine that a cycle specific transient analysis should be performed to verify that the reference analysis remains bounding. No specific quantitative criteria have been established for making this determination, but every instance in which an evaluation (as opposed to a reanalysis) of a key parameter departure is performed must be documented. In the documentation the analyst presents the exact numerical values pertaining to the departure from a limit and a detailed discussion of the reasoning and approach used in reaching a conclusion regarding the parameter in question. This documentation is subject to peer review and approval. The results of these cycle specific evaluations are summarized in the Reload Safety Evaluation (RSE) report.

Question 9:

In Section 3.3.2 - Safety Analysis Philosophy, it is stated that, "The methods that will be employed by Dominion to determine these key parameters will be consistent with the methods documented in References 9, 12, and 14" [of VEP-FRD-42, Revision 2]. References 12 and 14 are Westinghouse WCAP methodologies for reload safety evaluations, and power distribution control and load following procedures. Please discuss the evaluations performed to verify that these methodologies are also applicable for Framatome ANP fuel.

Response:

This section of VEP-FRD-42, Revision 2 defines 3 types of key parameters used to characterize the behavior of reload cores to various postulated accidents. The detailed calculation of specific key parameter values for a reload core is performed using the applicable core design or fuel design tools, dependent upon the parameter involved. The reload safety analysis framework involves evaluating the key parameter values determined for each reload to verify that margin exists between the reload value and the limiting value assumed in the reference safety analysis. This bounding value approach requires the existence of certain predefined relationships that identify the relevant key parameters for a given postulated accident, and their sensitivities (i.e., direction of most limiting effect).

References 9 and 14 of VEP-FRD-42, Revision 2 describe the detailed methodology for defining achievable core power distributions and associated operating limits for two different control schemes employed in Dominion analyses. Reference 9 defines the Dominion-developed Relaxed Power Distribution Control (RPDC) methodology and Reference 14 defines the Westinghouse-developed Constant Axial Offset Control (CAOC) methodology. Each of these methodologies involves the simulation, using detailed nuclear core design codes and models, of a defined number of perturbed core states and the corresponding power distributions. Each of these methodologies is used to determine the limits of normal core operation that will ensure that localized core power distributions remain within the values assumed as initial conditions in the accident analyses. Both methodologies are dependent upon defining proper design input details that characterize the core neutronic behavior. The required design input items involve detailed inputs such as nuclear cross-sections, geometry (fuel pellet, fuel rod and fuel assembly) and enrichment and reactor system inputs such as power, temperature and flowrate. There are several features of the Framatome ANP fuel that differ from the existing fuel design, including: theoretical density, use of Mid-Span Mixing Grids and use of alloy M5. The evaluation of these changes has concluded that each represents alteration of a detailed design input, but not a change that affects the reload methodology. Each of these features of the Advanced Mark-BW fuel was reviewed and found to be within the existing capability and range of applicability of the nuclear core design and safety analysis tools. It was thus concluded that the existing methodologies documented in References 9 and 14 could be used for analysis of the Advanced Mark-BW fuel with its slightly different features.

Reference 12 of VEP-FRD-42, Revision 2 documents the Westinghouse-developed reload evaluation methodology that supports the generic basis for the Dominion reload methodology. The Westinghouse methodology defines specific key parameters for use in accident analyses and their limiting directions for consideration in reload evaluations. Reference 12 is referenced in this sense, in that it defines part of the overall framework that constitutes the Dominion methodology. The changes associated with an alternate fuel design may be of two types: 1) changes that reflect physical fuel design features and 2) changes that reflect licensed analysis approaches or requirements. The Advanced Mark-BW fuel design was assessed for both types of change with respect to applicability of the Reference 12 methodology. It was concluded that none of the physical design features invalidate the key parameter definitions or usage as cited in Reference 12 and VEP-FRD-42, Revision 1. The review associated with potential licensed analysis approaches determined that the Framatome ANP fuel required an additional key parameter, which is reflected in Table 2 of VEP-FRD-42, Revision 2. This parameter, maximum linear heat generation rate versus burnup, is used in the NRC-approved Framatome ANP methodology for cladding stress evaluations. This parameter can be calculated with existing nuclear design codes. This review has demonstrated that the citation of Reference 12 as used within the reload methodology of VEP-FRD-42, Revision 2 is valid for reload evaluation of the Framatome ANP fuel.

Question 10:

Please identify and provide a reference for the fuel lattice physics code used to calculate the prompt neutron lifetime key analysis parameter (Section 3.3.3.5). Include a reference to the NRC staff SER approving this code. Please verify and provide the technical basis for the application of this code to expected fuel designs.

Response:

The lattice code referred to in Section 3.3.3.5 is NULIF, which is the same code used in VEP-FRD-42, Rev. 1. NULIF was originally reviewed as part of VEP-FRD-19A (Ref. 10.1) and the prompt neutron lifetime reliability factor was approved in VEP-FRD-45A (Ref. 10.2). NULIF is a pin cell neutron spectrum / isotopic depletion code. The input to NULIF (i.e., fuel density, fuel enrichment, clad material, fuel pin geometry, soluble boron concentration, depletion power, depletion interval, etc.) for Framatome ANP fuel is not significantly different than for Westinghouse fuel. NULIF is used for both Surry (15x15 lattice) and North Anna (17x17 lattice), and the differences between 15x15 and 17x17 fuel are more significant than the differences between Framatome ANP and Westinghouse fuel.

Reference:

10.1 M. L. Smith, "The PDQ07 Discrete Model," VEP-FRD-19A (July 1981).

10.2 Letter from United States Nuclear Regulatory Commission to Mr. W. N. Thomas, Virginia Electric and Power Company, "Acceptance for Referencing of Topical Report VEP-FRD-45 'Nuclear Design Reliability Factors,' " August 5, 1982.

Question 11:

The dropped RCCA(s) event (dropped rod or dropped bank) is evaluated using the methodology described in Westinghouse WCAP-11394-P-A (Reference 15 of this topical report). Please discuss the evaluation performed to verify that this methodology is also applicable for Framatome ANP fuel.

Response:

The dropped rod methodology of WCAP-11394 requires that three analyses be performed in order to perform an evaluation of the dropped rod event. These analyses, referred to as transient, nuclear, and thermal-hydraulic analyses, provide (1) the statepoints (reactor power, temperature, and pressure), (2) the radial power peaking factor, and (3) the DNB analysis at the conditions determined by items 1 and 2, respectively. These analyses are performed using a parametric approach so that cycle specific conditions may be evaluated using the data generated in the three analyses mentioned above.

Westinghouse, in WCAP-12282 (Reference 11.1), provided generic guidelines that established a common approach for implementation of the revised dropped rod methodology. WCAP-12282 indicated that the core physics correlations and transient statepoints generated for the methodology described in WCAP-11394 apply to all Westinghouse plants with 12 or 14 foot cores. However, due to the plant specific nature of the core physics characteristics and the thermal-hydraulic dropped rod limit lines, a generic safety analysis which bounds all plants is not feasible. Therefore, for every fuel cycle, plant specific data are combined with the appropriate set of correlations and statepoints to verify that the DNB design basis is met for the dropped rod event. The transient statepoints have been generated to be independent of reload considerations. The thermal-hydraulic limit lines are determined on a plant specific basis using currently licensed thermal-hydraulic models. The core physics data required for the analysis are generated during the normal course of the reload design.

The NRC, in Question No. 7 of the request for additional information for WCAP-11394, queried whether the plant/cycle specific calculations are really performed for the items mentioned, or have bounding values been used. The response in WCAP-11394-P-A states that "...the statepoints and R factors are not required to be calculated on a plant or cycle specific basis. Figures IV-1 through IV-8 show the generic applicability of the models used for various fuel types and cycle designs. However, the statepoints and/or R factors would be reassessed for new plants or fuel designs."

As described in WCAP-11394, the transient analysis consists of generating statepoint information (reactor power, temperature, and pressure) for a large number of dropped rod transient events. These statepoints cover a range of reactivity insertion mechanisms for use in the nuclear analysis: the worth of the dropped rod, the moderator temperature coefficient, and the total rod worth available in the control bank which is withdrawn by the Rod Control System when it attempts to restore power to the nominal value. Statepoint data for a large number of transient events, generated by Westinghouse, were used in application of this methodology to North Anna and Surry Power Stations. The statepoint data are influenced by NSSS and protection system features, and were generated to accommodate a wide range of potential core physics conditions. The validity of the statepoint data is, thus, not affected by the transition to Framatome ANP fuel.

The dropped rod methodology employs a bounding empirical correlation between dropped rod worth, $F\Delta H$, and MTC to relate the power change associated with a dropped rod (or rods) to the increase in peaking factor caused by the dropped rod. In order for this correlation to become non-conservative, either the peaking factor change associated with a dropped rod of a particular worth must increase or the power change associated with the dropped rod reactivity insertion must decrease. As indicated in the response to Question 2, the core physics characteristics of the Framatome ANP fuel are nearly identical to the Westinghouse fuel it will replace. There is no change in loading pattern strategy associated with Framatome ANP fuel that would cause a change in the range of dropped rod worth or in the relationship between dropped rod worth and peaking factor increase. Reload cores, therefore, will not respond in a fundamentally

different way to the dropped rod event due to the use of Framatome ANP fuel.

The final portion of the dropped rod methodology is the DNB analysis at the conditions determined from the statepoints (reactor power, temperature, and pressure) and the radial power peaking factor. For the DNB analysis, the methodology employs dropped rod limit lines that are representations of the core conditions (inlet temperature, pressure, core power level, and FΔH) for which the DNBR is equal to the DNBR design limit. The dropped rod limit lines for the resident Westinghouse fuel were shown to be applicable for both fuel types.

Therefore, the methodology described in Westinghouse WCAP-11394-P-A is applicable for Framatome ANP fuel.

Reference:

11.1 R. L. Haessler, "Implementation Guidelines for WCAP-11394 (Methodology for the Analysis of the Dropped Rod Event)," WCAP-12282, June 1989

Question 12:

Section 3.5 - Nuclear Design Report, Operator Curves, and Core Follow Data included the following changes to the list of design report reload parameters:

a. Iodine has replaced Samarium worth, and
b. K-effective at refueling conditions as a function of temperature and rod configuration has been removed from the list.

Please provide the technical basis for these changes.

Response:

Part a:

Iodine has not replaced samarium. Iodine has been added to the xenon information. Samarium has been replaced by "Reactivity due to isotopic decay," which includes the contribution of samarium as well as less significant nuclides which build up or decay after shutdown on a time scale similar to samarium.

Part b:

The K-effective for refueling data is now transmitted to the power station prior to issuance of the design report. This was an administrative change to support outage planning and not a change in methodology.

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 2, 2002

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 02-662
NL&OS/ETS R0
Docket Nos. 50-338/339
50-280/281
License Nos. NPF-4/7
DPR-32/37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
SURRY POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
DOMINION'S RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT

Dominion's Reload Nuclear Design Methodology Topical Report has been revised to support the transition to Framatome ANP Advanced Mark-BW fuel at North Anna. In a letter dated October 8, 2001 (Serial No. 01-628) Virginia Electric And Power Company (Dominion) submitted Revision 2 of VEP-FRD-42, "Reload Nuclear Design Methodology Topical Report," for NRC review and approval. During review of the topical report, the NRC staff identified additional information that is needed to complete their review. The additional information was requested in a letter from the NRC dated October 25, 2002. Attachment 1 to this letter provides the additional information including Dominion's process for the maintenance and modification of "NRC Approved" methodologies.

If you have any further questions or require additional information, please contact us.

Very truly yours,

Eugene S. Grecheck
Vice President – Nuclear Support Services

Attachment

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303-8931

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

Mr. M. J. Morgan
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Attachment

REQUEST FOR ADDITIONAL INFORMATION
DOMINION'S RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT
VEP-FRD-42, Revision 2

North Anna Power Station Units 1 and 2
Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

Background

In a letter dated October 8, 2001 (Serial No. 01-628) Virginia Electric and Power Company (Dominion) submitted Revision 2 of VEP-FRD-42, "Reload Nuclear Design Methodology Topical Report," for NRC review and approval. During review of the topical report, the NRC staff identified additional information that is needed to complete their review. The additional information was requested in a letter from the NRC dated October 25, 2002. The requested information is delineated below.

NRC Request for Additional Information:

"VEPCO is requested to confirm that the submittals listed below are the latest revisions for these codes that have not received NRC staff approval.

1. PDQ - The staff will review Topical Report VEP-NAF-1, July, 1990, submitted in a letter from VEPCO to NRC dated October 1, 1990.

2. NOMAD - The staff will review Topical Report VEP-NFE-1A, Supplement 1, September 1996, submitted in a letter from VEPCO to NRC dated November 11, 1996.

3. TIP/CECOR - The staff will review Topical Report VEP-NAF-2, November 1991, submitted in a letter from VEPCO to NRC dated December 20, 1991.

4. RETRAN - The staff will review the information submitted in a letter from VEPCO to NRC dated August 10, 1993. The information provided in this submittal was only applicable for North Anna, Units 1 and 2."

Dominion Response:

PDQ and NOMAD Codes & Models

For PDQ, the report submitted by letter Serial No. 90-562, dated October 1, 1990 is the latest revision that has not received NRC staff approval. Likewise, the NOMAD report submitted by letter Serial No. 96-319, dated November 13, 1996 (versus November 11, 1996 stated above) is the latest revision that has not received NRC staff approval. For both PDQ and NOMAD, the referenced reports are accurate representations of current codes and models with regard to methodology. That is, the theory, sources of input data, solution schemes, geometric mesh structure, energy group structure, and use of the models in the core modeling process have not changed. There have been subsequent code changes to correct minor errors and to accommodate new code edits and additional computing platforms. There have been changes in input to accommodate the evolution of core design features including increased fuel enrichments, changes in BP design, and use of vessel fluence suppression neutron absorber rods. Throughout this period, accuracy of the PDQ model (and by extension the NOMAD model, since PDQ is the source of data and normalization for NOMAD) has been verified each cycle during startup physics testing and during routine core follow. For each cycle, a Startup Physics Test Report and a Core Performance Report is issued to document the

behavior of the core relative to the model predictions.

TIP/CECOR Code & Model

The topical VEP-NAF-2, submitted by letter Serial No. 91-746, dated December 20, 1991, is the latest revision of TIP/CECOR that has not received NRC staff approval. However, Dominion does not consider review of TIP/CECOR necessary for review of VEP-FRD-42 Rev. 2 (the Reload Topical) for several reasons. First, the focus of the Reload Topical is on core design and safety analysis methodology, not core surveillance. TIP/CECOR is not directly discussed in VEP-FRD-42 Rev. 2 because it is not part of the reload methodology. TIP/CECOR uses data provided by the PDQ model (Reload Topical Section 2.1.1, paragraph 2) to perform core power distribution surveillance. Second, TIP/CECOR is not new methodology for measurement of core power distributions. USNRC review and approval for use of CECOR in the synthesis of core power distributions using fixed in-core detector data is documented in a 1980 Combustion Engineering Topical Report (Reference 5 of VEP-NAF-2). TIP/CECOR, the Dominion version of the model, uses the same solution schemes and techniques but employs data at 61 axial points rather than just a few. Finally, although the current interpretation of "essentially the same" had not yet been applied to 10CFR50.59 evaluations in 1992, the TIP/CECOR Topical Report and the 10CFR50.59 evaluation performed prior to use of the code clearly demonstrate that TIP/CECOR results are essentially the same as those of the previous measurement code (INCORE). The reason for replacing INCORE with CECOR was not to gain analytical margin, but to be able to accept input representing physically different regions of newer, axially non-homogenous cores.

RETRAN Code & Model

Consistent with approaches employed by NSSS vendors, Dominion's RETRAN model is qualified on the basis of the plant class for which it will be used. There is not a separate Surry-specific RETRAN model document that parallels the content of the report submitted in Reference 1. However, as discussed further below, the material in Reference 1 is equally applicable to the Surry and North Anna models. The Surry 3-loop model, which was completed after the submittal of Reference 1, uses the same noding, modeling philosophy and code options as the North Anna model. The following description provides some background discussion relating to the RETRAN models in use for North Anna and Surry.

Dominion's reload methodology incorporates the RETRAN-02 code, which was generically approved by the NRC via Reference 2. Dominion is currently using RETRAN-02, Mod 5.2. The NRC issued a generic approval, transmitted in Reference 3, for RETRAN-02 Mod 5.0. Discussions between the utilities and the NRC led to the conclusion that Mods 5.1 and 5.2, which were essentially maintenance upgrades, did not require additional NRC review for utility implementation (References 4 and 5).

Dominion's RETRAN models and capability were approved in Reference 6. As noted in the SER, the Virginia Electric and Power Company (Dominion) Topical Report was

supplemented in three subsequent submittals (References 7, 8, 9) prepared in response to NRC Requests for Additional Information.

The RETRAN Topical SER (Reference 6) recognized that model maintenance activities would be performed under the utility 10 CFR 50 Appendix B QA program:

> "The staff requires that all future modifications of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures."

Dominion has followed the requirements specified in the SER for VEP-FRD-41 in updating our RETRAN models. Updated models and the qualification results were documented per our 10 CFR 50 Appendix B QA program and provided to the USNRC for information in Reference 1. The qualification, documentation and implementation of these new models was done in a manner that meets the programmatic elements of Generic Letter 83-11, Supplement 1.

Reference 1 presented the 3-loop RETRAN model and qualification results using the North Anna version of the model. The Surry 3-loop model is the same with regard to noding, options and system and component modeling techniques. The Surry and North Anna models differ in order to appropriately reflect plant specific design features such as RCS geometry, system and pump characteristics and setpoint values. Dominion concludes that the model description in Reference 1 accurately describes the key features of the models in use for both Surry and North Anna power stations.

Dominion continues to perform model maintenance activities in accordance with the provisions of the SER and 10 CFR 50 Appendix B. Dominion has made model changes in the past to refine treatment of certain features, to address industry issues or to reflect changes to the plants. These changes were evaluated under the provisions of 10CFR50.59, which will continue to be employed to assess future changes. The following list summarizes several enhancements which are illustrative of the changes that have been made to the models:

- The current models use the 1979 ANS Decay Heat model option.
- More detailed main steam safety valve (MSSV) modeling was added to ensure that the concerns raised in NRC Information Notice 97-09, "Inadequate Main Steam Safety Valve (MSSV) Setpoints and Performance Issues Associated with Long MSSV Inlet Piping" are adequately addressed.
- Hydraulic characteristics in the core regions have been adjusted to reflect current fuel assembly designs.
- More detailed, mechanistic models for the pressurizer and steam generator level instrumentation were added.
- A detailed rod control system model was added.

Dominion's Process for the Maintenance and Modification of "NRC Approved" Methodologies

Section 2.3 of VEP-FRD-42, Rev. 2, entitled "Analytical Model and Method Approval Processes," indicates several acceptable means by which either analytical models or methods can achieve approved status for use in Dominion's reload methodology. The following discussion describes Dominion's approach in performing maintenance and modifications of NRC Approved methodologies. This approach is applied to all models and methodologies that are employed in Dominion's reload design methodology, and which may be cited either by reference within VEP-FRD-42 or in the COLR.

The determination of the requirement to submit methodology changes to NRC for approval prior to application is based on published NRC guidance, i.e.:

- Generic Letter 88-16, "Removal Of Cycle-Specific Parameter Limits From Technical Specifications" .
- 10 CFR 50.59, and in particular 10 CFR 50.59c(2)(viii): "*(2) A licensee shall obtain a license amendment pursuant to Sec. 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would (viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.*"
- NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Evaluations"
- Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments" (endorses NEI 96-07 Rev. 1)
- Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses"

Relevant sections of these documents upon which we base our determination process are as follows:

1. Generic Letter 88-16 establishes the concept of reload cycle dependent operating limits in the Technical Specifications.

   "Generally, the methodology for determining cycle-specific parameter limits is documented in an NRC-approved Topical Report or in a plant-specific submittal. As a consequence, the NRC review of proposed changes to TS for these limits is primarily limited to confirmation that the updated limits are calculated using an NRC-approved methodology and consistent with all applicable limits of the safety analysis. These changes also allow the NRC staff to trend the values of these limits relative to past experience. This alternative allows continued trending of these limits without the necessity of prior NRC review and approval."

2. NEI 96-07, Rev. 1, as endorsed by Reg. Guide 1.187, provides guidance for evaluating changes to methods under the provisions of 10CFR50.59. For example, Paragraph 4.3.8.1, states:

4.3.8.1, Guidance for Changing One or More Elements of a Method of Evaluation

"The definition of "departure ..." provides licensees with the flexibility to make changes under 10 CFR 50.59 to methods of evaluation whose results are "conservative" or that are not important with respect to the demonstrations of performance that the analyses provide. Changes to elements of analysis methods that yield conservative results, or results that are essentially the same, would not be departures from approved methods."

3. USNRC Generic Letter 83-11 Supplement 1 provides a method for utility qualification of analysis methodologies, including those used to establish core operating limits, without formal NRC review and approval:

   "The U.S. Nuclear Regulatory Commission (NRC) is issuing this supplement to Generic Letter (GL) 83-11 to notify licensees and applicants of modifications to the Office of Nuclear Reactor Regulation (NRR) practice regarding licensee qualification for performing their own safety analyses. This includes the analytical areas of reload physics design, core thermal-hydraulic analysis, fuel mechanical analysis, transient analysis (non-LOCA), dose analysis, setpoint analysis, containment response analysis, criticality analysis, statistical analysis, and Core Operating Limit Report (COLR) parameter generation. It is expected that recipients will review the information for applicability to their facilities. However, suggestions contained in this supplement to the generic letter are not NRC requirements; therefore, no specific action or written response is required."

   "To help shorten the lengthy review and approval process, the NRC has adopted a generic set of guidelines which, if met, would eliminate the need to submit detailed topical reports for NRC review before a licensee could use approved codes and methods. These guidelines are presented in the Attachment to this Generic Letter. Using this approach, which is consistent with the regulatory basis provided by Criteria II and III of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50), the licensee would institute a program (such as training, procedures, and benchmarking) that follows the guidelines, and would notify NRC by letter that it has done this and that the documentation is available for NRC audit."

Reflecting this NRC and industry guidance, Dominion's process for maintaining and modifying approved methodologies encompasses these elements:

- Dominion can change, under the provisions of 10 CFR 50.59(c)(2)(viii), NRC approved codes and methodologies used to establish core operating limits, via the processes outlined in NEI 96-07, Rev. 1, without additional NRC review and approval of these changes.

- Dominion can implement or substitute, under 10 CFR 50.59(c)(2)(viii), NRC approved codes and methodologies for use in establishing core operating limits via

the processes outlined in Generic Letter 83-11 Supplement 1, without additional NRC review and approval of these methods.

- Dominion concludes that, in updating the list of approved methodologies for establishing core operating limits in the Technical Specifications, utility affirmation that the changes to the methodologies have been done as described by either of the above is adequate to retain the "approved" status for these methods.

References:

1. Letter from M. L. Bowling (Virginia Electric and Power Company) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1&2, Supplemental Information on the RETRAN NSSS Model," Serial No. 93-505, August 10, 1993.

2. Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, "RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

3. Letter from A. C. Thadani (NRC) to W. J. Boatwright (RETRAN02 Maintenance Group), Acceptance for Use of RETRAN02 MOD005.0, November 1, 1991.

4. Letter from M. J. Virgilio (NRC) to C. R. Lehmann (RETRAN Maintenance Group), Acceptance for Referencing of the RETRAN-02 MOD005.1 Code, April 12, 1994.

5. Letter from G. L. Swindlehurst (RETRAN Maintenance Group) to T. E. Collins (NRC/RSB), RETRAN-02 MOD005.2 Code Version, Notification of Code Release, November 24, 1997.

6. Letter from C. O. Thomas (NRC) to W. L. Stewart (Virginia Power), Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, "Virginia Power Reactor System Transient Analyses Using the RETRAN Computer Code," April 11, 1985.

7. Letter from W. L. Stewart (Vepco) to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses, Supplemental Information," Serial No. 060, February 27, 1984.

8. Letter from W. L. Stewart (Vepco) to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses," Serial No. 376, July 12, 1984.

9. Letter from W. L. Stewart (Vepco) to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses," Serial No. 376A, August 24, 1984.

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 21, 2003

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 03-183
NL&OS/ETS
Docket Nos. 50-338/339
50-280/281
License Nos. NPF-4/7
DPR-32/37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
SURRY POWER STATION UNITS 1 AND 2
REQUEST FOR ADDITIONAL INFORMATION ON
TOPICAL REPORT VEP-FRD-42, RELOAD NUCLEAR DESIGN METHODOLOGY

In an October 10, 2001 letter (Serial No. 01-623) Virginia Electric and Power Company (Dominion) submitted Reload Nuclear Design Methodology Topical Report, VEP-FRD-42 Revision 2, for NRC review. This topical report was revised to support the transition to Framatome-ANP Advanced Mark-BW fuel at North Anna. Revision 2 of VEP-FRD-42 addresses the restriction in the SER for Revision 1 that stated, "it is clear that the methodology presented is closely related to the Westinghouse methodology, and is applicable in its present form only to Westinghouse supplied reloads of Westinghouse nuclear plants." Since the initial submittal of revision 2 to the topical report, additional information has been requested by and provided to the NRC staff in letters dated May 13, 2002 (02-280) and December 2, 2002 (02-662). The NRC Staff has requested additional information in a February 26, 2003 letter. The attachments to this letter provide the additional information to complete the NRC staff review of VEP-FRD-42, Revision 2.

If you have any further questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachments

Commitments made in this letter: None

Attachment 1

Responses to NRC Questions on RETRAN

Virginia Electric and Power Company
(Dominion)
North Anna and Surry Power Stations

RETRAN Code and Model Review –VEPCO Letter dated August 10, 1993

NRC RETRAN QUESTION 1

1. In the generic RETRAN Safety Evaluation Report (SER), dated September 4, 1984 (Reference 1), the NRC staff approved the use of RETRAN-01/MOD003 and RETRAN-02/MOD002 subject to the limitations and restrictions outlined in the SER. By letter dated April 11, 1985, the NRC staff approved the use of RETRAN-01/MOD003 for VEPCO, although the staff stated in this SER that VEPCO had not provided an input deck to the staff nor had it provided the information needed to address the restrictions listed in the staff SER dated September 4, 1984. The NRC staff's SER dated September 4, 1984, had requested this input deck submittal as a condition of approval to use the RETRAN Code.

   a. VEPCO is currently using RETRAN02/MOD005.2. Please provide information describing how each of the limitations, restrictions, and items identified as requiring additional user justification in the generic staff SERs for RETRAN02/MOD002 through RETRAN02/MOD005.0 (References 1-3) are satisfied for the North Anna and Surry RETRAN models.

   b. As required by the staff SERs (References 1-3), please submit RETRAN input decks that represent the current models and code options used for both North Anna and Surry. For each station, please provide input decks initialized to hot full power and hot zero power conditions in electronic format.

DOMINION RESPONSE TO QUESTION 1a

Dominion responses to the limitations in the RETRAN-02 Safety Evaluation Reports (SERs) in References 1-3 are divided into three sections to distinguish between the different SERs: I) RETRAN02/MOD002; II) RETRAN02/MOD003 and MOD004; and III) RETRAN02/MOD005. The responses are applicable to the North Anna and Surry pressurized water reactor RETRAN models. References for responses to Question 1a are included at the end of the attachment.

I. RETRAN 02/MOD002 Restrictions

The Dominion treatment of each RETRAN limitation from Section II.C in Reference 1 is described. The responses address Limitations a through z, two items on page E2-54 that "require further justification", and eight "implications of the limitations" on page E2-55.

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

Dominion Evaluation

The point kinetics approximation is used in the Dominion RETRAN model, consistent with standard industry safety analysis practice. Reactivity effects are modeled using standard fuel and moderator temperature coefficients and control bank worths which are shown to be bounding for Dominion cores using static core physics models which account for full 3-D effects.

Most non-LOCA transients do not involve significant temporal variations in the core power distributions, and industry experience over many years has shown the point kinetics approximation to be valid for this type of accident. Two notable exceptions are the control rod ejection and main steam line break events.

For the control rod ejection event, Dominion uses a point kinetics model to calculate the core average power response. The Doppler feedback is calculated using a spatial power weighting factor that is a function of the radial power peaking factor in the vicinity of the ejected rod, which is calculated using static neutronics calculations. Local power peaking is also calculated via static methods. The power peaking and core average time dependent power responses are then used in conjunction with a conservative hot spot fuel pin model to calculate the limiting local fuel thermal response. Dominion's rod ejection methods have been benchmarked against full 3-D space-time kinetics calculations and shown to be conservative in VEP-NFE-2-A [Reference 4].

Dominion's methodology for steam line break is described in Sections 5.2.3.4 and 5.2.3.5 of VEP-FRD-41-A [Reference 5]. Asymmetric reactivity effects associated with the cold leg temperature imbalance and the assumption of a stuck control rod are modeled by breaking the core into two azimuthal sectors and providing an empirical weighting factor to the moderator temperature coefficients in the two sectors. Fluid mixing between the two regions is modeled based on scale model mixing tests performed by Westinghouse.

Power reactivity feedback is also modeled with an empirical curve of reactivity feedback versus heat flux. The validity of these curves is checked for every reload by static neutronics methods that show that the magnitude of the post-trip return to power predicted by RETRAN is conservatively high. Local power peaking is also calculated using static neutronics methods. Core DNB performance is calculated in a separate code (e.g. COBRA or VIPRE).

This approach for using a combination of point kinetics and static 3-D neutronics calculations for analyzing the steam line break event is similar to that used by fuel vendors (see for example References 6-8).

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

Dominion Evaluation

Dominion meets this restriction. Dominion initiates low power events, such as rod withdrawal from subcritical, and the hot zero power rod ejection event from a critical condition with a low initial power level representative of operation within the range of operability for the source range nuclear instrumentation channels. For the "zero power" steam line break, the models are initialized in the same way, and then the design shutdown margin is simulated by a rapid negative reactivity insertion coincident with the break opening.

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

Dominion Evaluation

A generalized boron transport model was added to RETRAN02/MOD005 [Reference 3]. However, Dominion uses the RETRAN control system to model boron transport in the reactor coolant system for steam line break analyses.

During initial steamline break model development, RETRAN's general transport model was considered but not selected. The primary reason this option was not chosen was that the general transport model uses the default assumption of perfect mixing. Non-mixing regions like pipes cannot be conveniently modeled with a delay-type of behavior. The user may adjust mixing by changing the junction efficiency with a control system. However, this results in just as many control system cards devoted to mixing efficiency calculation as a control block based, full-transport model. Therefore, boron transport is modeled with a control system as in previous analyses. The general modeling philosophy is consistent with that described in Figure III-12 of Reference 19, which was submitted to support the original VEP-FRD-41 review. However, the model in Reference 19 assumed a constant reactor coolant system flow rate. The model was made more robust by incorporating variable transport delays and a dynamic plenum mixing model as described below, so that variable RCS flows are now handled accurately.

The boron transport model is broken into four major parts: 1) Refueling Water Storage Tank (RWST) to Boron Injection Tank (BIT); 2) the BIT; 3) BIT to the Reactor Coolant System (RCS); and 4) the RCS.

<u>BIT Mixing Model</u>
The BIT mixing model begins with the same basic equations as the RCS mixing model. The model makes the approximation that the density of the BIT is constant and is also equal to the density of the incoming fluid.

Following are the mixing region equations:

$$\frac{dC}{dt} = w_i c_i - w_o c_o$$

$$\frac{dC}{dt} = \frac{Mdc}{dt} + \frac{cdM}{dt}$$

$$\frac{dc}{dt} = \frac{w}{M}(c_i - c_o)$$

$$c(t) = \int \frac{dc}{dt} + c_o$$

The first equation states that the rate of change of the mass times the concentration is equal to the mass flow rates in and out times their respective concentrations. The second equation expands the large C derivative into its constituents. The dM/dt term in the second equation is assumed to

be zero and w_i is assumed to be equal to w_o. The third equation is formed by combining the first two with $dM/dt = 0$. The integral of dc/dt provides the dynamic concentration out of the BIT.

By assuming that the density of the BIT and the incoming fluid are equal, the w/M term is equal to the volumetric flow divided by the volume. The equations above are represented with the appropriate control blocks.

BIT to RCS Transport

The transport time through the BIT to RCS piping is calculated in several pieces: the common BIT to SI header delay, and the individual delays from the header to each cold leg. A DIV control block divides the BIT to HDR volume by the total flow rate. The transport time is then used as input to a DLY control block. The same function is performed for each of the header-to-loop segments. The fluid is assumed to be at an initial boron concentration of zero ppm.

RCS Boron Transport

The RCS is broken into several regions for boron transport:

1) the cold leg between the SI point and the vessel (DELAY)
2) the downcomer and lower plenum (MIXING)
3) each core section (DELAY)
4) core bypass (DELAY)
5) the outlet plenum (MIXING)
6) the hot leg, SG tubes, loop seal, RCP, and cold leg between the RCP and SI point. (DELAY)

The model used to represent the transport through each region is noted in parentheses above. The upper head concentration is assumed to be zero for the duration of the transient.

The technique used in each "DELAY" region is as follows:

1) Total "boron flowrate" entering the region is computed by summing the inlet fluid flows times their respective boron concentrations.
2) Total fluid flow entering the region is computed by summing the inlet fluid flows.
3) The total "boron flowrate" is divided by the total fluid flowrate to get a mixed boron concentration.
4) The masses of the volumes in the transport region are summed.
5) The total mass is divided by the total fluid flow to get the transport delay for the region.
6) The mixed boron concentration is propagated to the next region using the transport delay.

The technique used in each "MIXING" region is as follows:

1) The net "boron flowrate" in a region is computed by summing the inlet and outlet fluid flows times their respective boron concentrations.
2) This represents the rate of change of region mass times concentration (dC/dt) which is then integrated to determine C(t).
3) The concentration (c(t)) is then calculated by dividing (C(t)) by the region mass (M).

For the steamline break event, the peak core heat flux is sensitive to the timing of the initial boron increase in the core (i.e., the transport delay from the safety injection system to the core inlet) and is not sensitive to the exact shape of the boron buildup curve. Core inlet boron is only a few ppm at the time of peak heat flux. Dominion's model and vendor models predict comparable times for the introduction of boron to the core as shown in benchmark calculations.

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

Dominion Evaluation

Control rod motion in the Dominion RETRAN point kinetics models is simulated by a reactivity input calculated from a time-dependent control bank position and a function generator containing integral bank worth versus position. For cases with automatic rod control simulated, the bank worth model is typically associated with the D-control bank only, which is the only bank in the core at or near full power.

For cases with reactor trip, the integral worth assumed is that associated with all control and shutdown banks at the power dependent insertion limit, less the most reactive control assembly in the core, which is assumed not to insert. The shape of the integral worth curve is based on a conservative bottom-skewed power distribution which delays the reactivity effects. This integral worth curve is checked for every reload core.

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

Dominion Evaluation

The rod ejection accident is the only non-LOCA transient analyzed with RETRAN where the metal-water reaction is applied. Dominion's RETRAN hot pin model was benchmarked against a similar vendor model and produced consistent temperature transients for consistent transient pin powers. These results are discussed in Reference 4, which documents Dominion's rod ejection methodology in its entirety.

f) Equilibrium thermodynamics is assumed for the thermal hydraulics field equations although there are nonequilibrium models for the pressurizer and the subcooled boiling region.

Dominion Evaluation

The current version of RETRAN-02 in use at Dominion (MOD005.2) allows for multiple nonequilibrium volumes. In Dominion RETRAN models, the nonequilibrium region option is generally only used for the pressurizer, except when applied to the reactor vessel upper head in main steamline break analyses. Toward the end of the transient, the upper head, which has experienced drainage, flashing and phase separation during the cooldown, will begin to refill due to continued operation of safety injection. An equilibrium model in the head can produce nonphysical pressure oscillations. While this phenomenon generally occurs beyond the time of

interest for evaluating core performance, the nonphysical behavior is avoided by using a nonequilibrium model in the upper head. This is physically reasonable for the head geometry and the limited hydraulic communication between the head and the upper plenum.

Section 5.3.3 of VEP-FRD-41-A presented comparisons of RETRAN pressure predictions to plant data for a cooldown and safety injection transient at North Anna. The nonequilibrium pressurizer model response was in good agreement with the observed plant response.

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

Dominion Evaluation

Dominion RETRAN models do not currently use the vector momentum option. As discussed in the response to Limitation A, incomplete fluid mixing between loops is modeled for steam line break based on the Indian Point 1/7 scale model mixing tests performed by Westinghouse. This is done by dividing the downcomer into two azimuthal sectors and specifying cross-flow junctions between the cold legs and downcomer sectors with form-loss coefficients to give the proper steady state mixing flows.

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

Dominion Evaluation

This limitation is not applicable to Dominion PWR RETRAN models.

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

Dominion Evaluation

Dominion RETRAN models specify the use of the dynamic slip option on the primary side and zero slip on the secondary side of the steam generator (SG) tubes. However, two-phase flow is not normally encountered on the primary side during non-LOCA PWR transients. The exception is for steam line break, where the pressurizer may drain during the cooldown, and the upper head may flash, resulting in some carryunder to the upper plenum region as the head drains. The RCS pressure response obtained in Dominion steam line break analyses, including the effects of pressurizer and upper head flashing and drainage, is consistent with that obtained by vendor models as discussed in VEP-FRD-41-A.

Dominion does have a multi-node steam generator secondary model overlay that uses dynamic slip modeling. This model is not used in licensing calculations, but it is occasionally used in studies to confirm that the standard steam generator models are providing conservative results. The standard model features involve a single-node secondary side model and the associated heat transfer response and level-versus inventory correlations that are used to model low and low-low

SG level reactor protection. The multi-node model treats the horizontal flow between the lower downcomer and tube bundle as bubbly flow.

Reference 9 presented comparisons between the multi-node and single-node SG versions of the model for a complete loss of load and for a 200%/minute turbine runback transient at full power. The response comparisons for pressurizer pressure and liquid volumes, RCS temperature, and steam pressure showed essentially identical responses for the two models. The most pronounced differences were in predicted changes in steam generator level and inventory, as expected.

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

Dominion Evaluation

Refer to the response to Limitation h2.

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

Dominion Evaluation

The core conductor model in Dominion RETRAN system models does not use the gap expansion model. Dominion's hot spot model for calculating the hot pin thermal transient in rod ejection analyses models rapid gap closure following the ejection with an essentially infinite gap thermal conductivity, as described in Reference 4. Qualification comparisons of the hot spot model to vendor calculations are presented in Section 4.3.2 of Reference 4.

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

Dominion Evaluation

Dominion PWR RETRAN models do not use air.

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

Dominion Evaluation

Dominion models have not been applied in the supercritical region. Dominion notes that this restriction has been substantially reduced for RETRAN-3D [Reference 10], and the NRC staff has approved RETRAN-3D for ATWS analysis, with a caution for evaluating calculations in the region of enthalpy > 820 Btu/lbm and pressures between 3200 and 4200 psia. Dominion has not yet formally implemented RETRAN-3D nor applied it to ATWS analyses.

Also note that the design basis for the ATWS Mitigation System Actuation Circuitry (AMSAC) for Westinghouse PWRs is to limit the maximum RCS pressure to less than 3200 psig [Reference 11]. Therefore, analytical results which yield supercritical conditions in the RCS are not anticipated for Dominion's nuclear units.

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

Dominion Evaluation

Dominion PWR RETRAN system models use heat transfer correlations in three areas:

- Reactor core conductors
- Primary (RCS) side of the steam generator tubes
- Secondary (steam) side of the steam generator tubes

For all non-LOCA accident analyses, the core heat transfer remains in the single-phase convection and subcooled nucleate boiling regions. The event that presents the most severe challenge to subcooled nucleate boiling on a corewide basis is the locked reactor coolant pump rotor event presented in Sections 15.4.4 and 14.2.9.2 of the North Anna and Surry UFSARs, respectively. For the locked rotor event, the heat transfer mode remains subcooled forced convection at the core inlet node and nucleate boiling at the mid core and top core node throughout the event.

Similarly, subcooled forced convection is the dominant heat transfer mode on the inside of the steam generator tubes for all non-LOCA events.

On the secondary (steam) side of the steam generator tubes, the heat transfer mode is typically saturated nucleate boiling (Mode 2) for non-LOCA transients. Exceptions occur when:

- a steam generator approaches dryout, such as for the North Anna feedline break accident
- a steam generator blows down, as in the main steam line break event.
- there is no flow through the single-node secondary side of the steam generator, such as during a loss of load (turbine trip) with feedline isolation.

These cases will be addressed in turn.

For cases where significant steam generator dryout is anticipated, Dominion uses the RETRAN local conditions heat transfer option in conjunction with the single-node steam generator secondary side model. Dominion has performed analyses to evaluate the physical realism of the modeling results, including a steam generator tube noding sensitivity study. The behavior of the model is such that nucleate boiling heat transfer (RETRAN Mode 2) is predicted for nodes below the collapsed liquid level. For nodes above the collapsed level, the model predicts a rapid transition from single-phase convection to steam (RETRAN Mode 8).

For the steam line break calculation, Dominion uses a set of overlay cards to predict a conservatively large heat transfer coefficient on the secondary side, in order to maximize the RCS cooldown. This is done using control blocks.

For nodes below the collapsed liquid level, the overlay model applies a separate heat transfer coefficient to the secondary side of each steam generator conductor based on the maximum of the following, independent of which regime the RETRAN logic would pick:

- Rohsenow pool boiling
- Schrock-Grossman forced convection vaporization
- Thom nucleate boiling
- Chen combined nucleate boiling and forced convection vaporization
- Single phase conduction to steam (Dittus-Boelter)

This maximum coefficient represents the heat transfer for the "wet" heat transfer surface in the steam generator.

To better represent the variation of the film coefficient for the conductors at different elevations, a model was developed to calculate a collapsed liquid level and apply the maximum "wet" coefficient below this level and the forced convection to steam above this level. This provides a realistic and smooth transition in heat transfer capability as the steam generator inventory is depleted.

For cases with no flow calculated through the single-node secondary side (e.g., turbine trip with no condenser dumps and assumed feedwater line isolation at the time of turbine trip), the heat transfer on the entire secondary surface of the tubes will rapidly transition to forced convection vaporization with a very small heat transfer coefficient. This behavior is non-physical, because a significant portion of the tube bundle remains covered with two-phase mixture and would remain in the nucleate boiling regime. However, the results are conservative and Dominion's experience has been that this calculational anomaly only occurs for brief periods of time such that the key results (e.g., peak RCS pressure) are not significantly impacted.

In summary, the limitations of RETRAN's regime-dependent heat transfer models are considered in Dominion licensing analyses. Appropriate assumptions and approximations are made to ensure that the accident analyses are conservative.

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

Dominion Evaluation

Dominion RETRAN models are not used for conditions involving two-phase horizontal flow. The models use the RETRAN application of Baroczy's correlation for two-phase friction effects, as opposed to Beattie. For steam generator tube rupture calculations, break flow is calculated using a junction loss coefficient computed from Blasius' smooth tube frictional pressure drop assuming single-phase flow. This model overpredicts the actual observed break flow in the 1987 North Anna Unit 1 double-ended rupture.

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

Dominion Evaluation

Dominion RETRAN models specify the use of the dynamic slip option on the primary side and zero slip on the secondary side. Refer to the response to Limitation h2.

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

Dominion Evaluation

Refer to the response to Limitation h2.

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

Dominion Evaluation

VEP-FRD-41-A [Reference 5] describes that the Dominion RETRAN pressurizer model uses the non-equilibrium model to ensure accurate modeling of transient conditions that may involve a surge of subcooled liquid into the pressurizer or to ensure appropriate treatment of pressurizer spray and heaters. While a wall heat transfer model, including vapor condensation, was added in version MOD003 [Reference 2], Dominion continues to model the non-equilibrium volume walls as an adiabatic surface.

The North Anna Unit 2 Natural Circulation Tests conducted in July 1980 measured the effect of convective heat losses from the pressurizer with all heaters secured. The observed effect was about 5 F/hr liquid temperature cooldown and about 38 psi/hr pressure loss [Reference 12]. The significant plant response for UFSAR non-LOCA transients occurs within the first 30 minutes of the event initiator. Therefore, pressurizer wall heat transfer is a phenomenon that is not significant over the time frame of interest for UFSAR non-LOCA analyses.

Section 5.3.3 of VEP-FRD-41-A includes a RETRAN simulation of a North Anna cooldown event, demonstrating the adequacy of the RETRAN pressurizer modeling assumptions compared to actual plant response. Both the observed data and the model indicated that level indication was lost for a brief portion of the transient. Overall, the RETRAN prediction of pressurizer pressure

and level indicate that the non-equilibrium pressurizer model adequately describes the behavior for large swings in pressure and level. In addition, the model predicted the time when level indication was lost close to the observed data. Therefore, the RETRAN non-equilibrium pressurizer model is able to perform accurate predictions of a draining pressurizer.

Reference 9 included a RETRAN simulation comparison to the 1987 North Anna steam generator tube rupture event. Figures 71 and 72 demonstrate that the RETRAN non-equilibrium pressurizer model provides good predictions of pressure and level behavior over a wide range of actual accident conditions. The model closely predicted the pressurizer level recovery near 1700 seconds.

RETRAN has been used to analyze the North Anna main feedwater line break (MFLB) UFSAR event, which reaches a pressurizer fill condition. The RETRAN analysis was benchmarked to the licensed LOFTRAN analysis and showed good agreement for pressurizer pressure and water volume. The codes predicted similar times for the pressurizer to reach a fill condition and similar RCS conditions long-term after the pressurizer is filled. Dominion RETRAN simulations for the MFLB event do not exhibit any unusual pressurizer behavior or numerical discontinuities when the pressurizer fills and remains filled.

The results of RETRAN comparisons to plant operational data in References 5 and 9 and to other licensed transient analysis codes demonstrate that the non-equilibrium pressurizer model is adequate over the expected range of pressurizer conditions that occur in North Anna and Surry UFSAR non-LOCA events analyzed with RETRAN.

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

Dominion Evaluation

The non-mechanistic separator model is not applied in Dominion PWR RETRAN models.

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

Dominion Evaluation

VEP-FRD-41-A describes that the plant-specific pump head vs. flow response for first quadrant operation is used in the Dominion RETRAN models. The homologous curves in the model represent single-phase conditions. The RETRAN default curves are not used. The pump

coastdown verifications in Section 5.3 of VEP-FRD-41-A demonstrate the adequacy of the centrifugal reactor coolant pump model versus plant-specific operational test data. Changes to the RCP coastdown model were made in Reference 9 to provide conservative coastdown flow predictions for loss of flow events relative to the actual coastdown measured at the plant. The latest Westinghouse locked rotor/sheared shaft coefficients have also been implemented.

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

Dominion Evaluation

The jet pump model is not applied in Dominion PWR RETRAN models.

s) **The nonmechanistic turbine model assumes symmetrical reaction staging, maximum stage efficiency at design conditions, a constant L/A, and a pressure behavior dictated by a constant loss coefficient. It should only be used for quasistatic conditions and in the normal operating quadrant.**

Dominion Evaluation

The non-mechanistic turbine model is not applied in Dominion PWR RETRAN models.

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

Dominion Evaluation

The Dominion PWR RETRAN models do not use the subcooled void model to calculate the neutronic feedback from subcooled boiling region voids. Dominion models use a moderator temperature coefficient except for the steamline break event, which applies an empirical curve of reactivity feedback versus core average power. This curve is validated as conservative on a reload basis using static, 3-D, full-core neutronics calculations with Dominion's physics models [Reference 15]. Dominion experience has indicated that the calculated DNBR's for the limiting steamline break statepoints show a weak sensitivity to the effects of void reactivity. The profile blending algorithm approved for RETRAN-02 MOD003 resolved this limitation [Reference 10, page 29].

u) **The bubble rise model assumes a linear void profile; a constant rise velocity (but adjustable through the control system); a constant L/A; thermodynamic equilibrium and makes no attempt to mitigate layering effects. The bubble mass equation assumes**

zero junction slip which is contrary to the dynamic and algebraic slip model. The model has limited application and each application must be separately justified.

Dominion Evaluation

Dominion PWR RETRAN models use bubble rise in the pressurizer, reactor vessel upper head, and steam generator dome regions [Reference 9, Table 1].

The upper head applies the bubble rise model to provide complete phase separation to account conservatively for upper head flashing during a main steam line break (MSLB). Complete separation ensures that only liquid will be delivered to the upper plenum during transients that exhibit upper head flashing. The effect of upper head flashing is seen in the abrupt change in slope in the reactor coolant system pressure following a MSLB. Dominion's RETRAN model predicts results that are similar to the licensed FSAR MSLB analysis in VEP-FRD-41-A (Figure 5.47).

The single-node steam generator secondary model is initialized with a low mixture quality so that the steady-state initialization scheme selects a large bubble rise velocity. The initialization models complete phase separation as a surrogate for the operation of the mechanical steam separators and dryers in the steam generators.

The pressurizer model applies the maximum bubble density at the interface between the mixture and vapor region. The use of the bubble rise model in the pressurizer has been qualified against licensed transient analysis codes and plant operational data as follows:

- VEP-FRD-41-A RETRAN analyses show pressurizer conditions similar to the vendor FSAR analyses for several accidents: uncontrolled rod withdrawal at power, loss of load event, main steamline break, and excessive heat removal due to feedwater system malfunction.
- VEP-FRD-41-A, Section 5.3.3, RETRAN simulations show good agreement with pressurizer response operational data from the 1978 North Anna cooldown transient.
- Reference 9 RETRAN simulations show good agreement of transient pressurizer conditions compared to the 1987 North Anna Unit 1 steam generator tube rupture event.

Implicit in the agreement between plant operational data and RETRAN is that the bubble rise model accurately predicts conditions in the pressurizer over a wide range of temperature, pressure, and level transient conditions. Therefore, Dominion has justified appropriate use of the bubble rise model through adequate benchmarking against physical data and other licensed transient analysis codes.

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

Dominion Evaluation

Dominion RETRAN models use the enthalpy transport delay model in the reactor coolant system piping and core bypass volume, where a dominant flow direction is expected. Flow reversal is not normally encountered in these volumes during non-LOCA accident analyses. For accidents

that produce a flow reversal or flow stoppage, the analyst may use the transport delay model if it adds conservatism to the results (e.g., if RCS pressure is higher during a locked rotor event with the model activated).

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

Dominion Evaluation

Dominion PWR RETRAN models do not employ the auxiliary DNBR model.

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

Dominion Evaluation

Dominion PWR RETRAN models do not use the enthalpy transport model in separated volumes. The enthalpy transport model is used only for the reactor core and the steam generator tubes primary side. The restriction is met.

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

Dominion Evaluation

As discussed in the response to Limitation m, Dominion restricts use of the local conditions heat transfer model to loss of secondary heat sink events. The model predicts a rapid transition from nucleate boiling to single-phase convection to steam on the secondary side as the tube bundle dries out.

Nodal sensitivity studies were performed to show that the default tube bundle noding provides an adequate representation of the primary to secondary heat transfer. The single-node secondary side is initialized with a low mixture quality. As a result, a high bubble rise velocity is calculated by the steady state initialization routine. This drives the RETRAN calculated mixture level to the collapsed liquid level and conservatively maximizes the rate of tube bundle uncovery as the inventory is depleted. The fluid condition on the inside of the tubes remains single phase, and thus the restriction is met.

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

Dominion Evaluation

Dominion's RETRAN user guidelines contain appropriate guidance and cautions about the potential impact of the feedwater enthalpy bias term on transient results. The guidance for initializing the models for other than the default conditions instructs the user to run a null transient and check the results for a stable solution, and to check the calculated heat transfer area on the steam generators to ensure that primary and secondary side conditions are properly matched.

<u>Technical Evaluation Report (TER) "Items Requiring Further Justification"</u>

The RETRAN-02/MOD002 TER, page E2-54, includes two items that require further justification for PWR systems analysis. Dominion responses to these items are provided below.

i) **Justification of the extrapolation of the FRIGG data or other data to secondary side conditions for PWRs should be provided. Transient analyses of the secondary side must be substantiated. For any transient in which two-phase flow is encountered in the primary, all the two-phase flow models must be justified.**

Dominion Evaluation

These restrictions were addressed in the evaluations for Limitations h2, m, n1, u, x, and y.

ii) **The pressurizer model requires qualification work for the situations where the pressurizer either goes solid or completely empties.**

Dominion Evaluation

Refer to the response to Limitation o. Dominion has shown that the non-equilibrium pressurizer model is adequate over the expected range of pressurizer conditions that occur in North Anna and Surry UFSAR non-LOCA events analyzed with RETRAN. Specifically,

- The UFSAR main steam line break events analyzed with RETRAN show a response for a drained pressurizer that is consistent with vendor methods [Reference 5, Figure 5.47].
- The North Anna UFSAR main feedline break event (case with offsite power available), which results in a filled pressurizer, shows a response that is consistent with vendor results.
- Comparisons to the North Anna Cooldown Transient [Reference 5, Section 5.3.3] and Steam Generator Tube Rupture [Reference 9, Section 3.2] shows reasonable agreement with plant data for the case of pressurizer drain and subsequent refill.

Technical Evaluation Report "Implications of these Limitations"

The RETRAN-02/MOD002 TER includes "implications of these limitations" on page E2-55. Dominion responses to the eight implications are provided.

i) Transients which involve 3-D space time effects such as rod ejection transients would have to be justified on a conservative basis.

Dominion Evaluation

See the response to Limitation a and Topical Report VEP-NFE-2-A.

ii) Transients from subcritical, such as those associated with reactivity anomalies, should not be run.

Dominion Evaluation

See the response to Limitation b.

iii) Transients where boron injection is important will require separate justification for the user specified boron transport model.

Dominion Evaluation

See the response to Limitation c.

iv) For transients where mixing and cross flow are important the use of various cross flow loss coefficients have to be justified on a conservative basis.

Dominion Evaluation

See the responses to Limitations a and g.

v) ATWS events will require additional submittals.

Dominion Evaluation

See the response to Limitation l.

vi) For PWR transients where the pressurizer goes solid or completely drains the pressurizer behavior will require comparison against real plant or appropriate experimental behavior.

Dominion Evaluation

See the response to Limitation o and "Item For Additional Justification Item ii". Dominion notes that the RETRAN 3-D pressurizer model has been explicitly approved for filling and draining events [Reference 10].

vii) PWR transients, such as steam generator tube rupture, should not be analyzed for two-phase conditions beyond the point where significant voiding occurs on the primary side.

Dominion Evaluation

Dominion meets this restriction with the exception of the main steam line break event analysis, which produces a limited amount of flashing in the stagnant upper head volume. Refer to Dominion's Evaluation of Limitations F and U for justification of the use of the bubble rise model with complete phase separation for the upper head volume in the reactor coolant system.

viii) BWR transients where asymmetry leads to reverse jet pump flow, such as the one recirculation pump trip, should be avoided.

Dominion Evaluation

This caution does not apply to Dominion PWR RETRAN models.

II. RETRAN 02/MOD003-004 Restrictions

Section 3.0 of Reference 2 presents six restrictions for RETRAN02/MOD003 and MOD004 code versions. The Dominion evaluation for each is provided.

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

Dominion Evaluation

VEP-FRD-41-A documents the NRC-approved RETRAN analysis methodology employed by Dominion. The topical report included 1-loop and 2-loop RETRAN models, their nodalization schemes, and specific comparisons to licensed FSAR analyses and to plant operational events. Reference 9 notified the NRC of modifications to the RETRAN models, including development of a 3-loop model and the primary and secondary systems nodalization schemes. The Dominion 3-loop models include discrete noding for every major geometry feature in the reactor coolant system. The steam generator secondary model is a lumped volume; Dominion experience has confirmed the adequacy and conservative nature of this model.

Analyses from the qualification set were provided in References 5 and 9 to demonstrate the adequacy and conservatism of the model nodalization and selection of model options. Dominion meets the NRC SER restrictions and has justified the model options over the range of conditions expected for non-LOCA transients for North Anna and Surry. The RETRAN user manual and training describe the limitations for the selected optional models to ensure appropriate use within the qualified range of application.

Dominion has qualified its RETRAN models against plant operational data and other licensed transient analysis codes sufficiently to justify the nodalization schemes and the model options that are used for non-LOCA transients analyzed with RETRAN.

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

Dominion Evaluation

Dominion treatment of the RETRAN02/MOD002 SER restrictions is provided earlier in this attachment.

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

Dominion Evaluation

Refer to the response to RETRAN02/MOD002 SER Limitation h2.

4. **Certain changes were made in the momentum mixing for use in the jet pump model. These changes are acceptable. However, those limitations on the use of the jet pump momentum mixing model which are stated in the original SER (Ref. 1) remain in force.**

Dominion Evaluation

Dominion PWR RETRAN models do not use jet pump models.

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

Dominion Evaluation

Dominion RETRAN models do not use the wall heat transfer model for non-equilibrium volumes. Dominion RETRAN comparisons to plant transients show that adiabatic modeling of the pressurizer walls is adequate (see response to RETRAN02/MOD002 SER Limitation o).

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

Dominion Evaluation

Dominion RETRAN models use the iterative solution technique. This technique allows the results of the time advancement to be evaluated before the solution is accepted. If a converged solution is not achieved in a given number of iterations, the time advancement can be reevaluated with a smaller time step. The Courant limit default value of 0.3 is applied in Dominion models.

The default value limits the time step size to less than 1/3 of the time interval required for the fluid to traverse the most limiting (i.e. fastest sweep time) control volume in the system. This is considered a very robust method for ensuring that the Courant limit is not exceeded.

Dominion user guidelines require that time step studies be performed for each new RETRAN analysis to ensure that a converged numerical solution is reached. This practice eliminates the impact of variations in the selected Courant limit input constant.

III. RETRAN 02/MOD005.0 Restrictions

The Dominion treatment of each limitation from Reference 3, Section 4.0, is described.

1. **The user must justify, for each transient in which the general transport model is used, the selected degree of mixing with considerations as discussed in Section 2.1 of this SER.**

Dominion Evaluation

Dominion does not use the general transport model. A description of the Dominion boron transport modeling for steamline break analyses is provided in the response to Limitation c in Section I.

2. **The user must justify, for each use of the ANS 1979 standard decay heat model, the associated parameter inputs, as discussed in Section 2.2 of this SER.**

Dominion Evaluation

Section 2.2 of the RETRAN-02 MOD005.0 SER specifies the following parameter inputs:

a. power history
b. fission fraction
c. energy per fission of each isotope
d. neutron capture in fission products by use of a multiplier
e. production rate of 239 isotopes
f. activation decay heat other than 239
g. delayed fission kinetic modeling
h. uncertainty parameters

The Dominion RETRAN models use the following assumptions in the calculation of decay heat:

- An operating period of 1,500 days with a load factor of 100% is input to the Dominion RETRAN models.
- The model assumes 190 MeV/fission. The reduction of the Q value to 190 MeV/fission from the default RETRAN value of 200 MeV/fission is conservative since, in the 1979 ANS Standard, decay heat power is inversely proportional to Q.
- There is no neutron capture component.
- Decay heat fissioning is solely from U-235. The assumption that all decay heat is produced from U-235 fissioning nuclides is conservative.
- The RETRAN actinide correlation is that of Branch Technical Position APCSB9-2 [References 17 and 18]. The RETRAN input of the breeding ratio UDUF (i.e., the number of Pu-239 atoms produced per U-235 atoms fissioned) is 0.77 and only impacts the calculation of the actinide contribution. The greater the value of UDUF, the higher the predicted decay heat fraction.
- A value of 1.0 is input for the RETRAN model for the decay heat multiplier.

The results of a RETRAN calculation with the 1979 decay heat model and the assumptions listed above were compared to a vendor calculated decay heat curve based on the 1979 ANS standard with 2-sigma uncertainty added. The results indicated that the decay heat fraction calculated with RETRAN is higher than the vendor calculated decay heat. Therefore, the Dominion application of the ANS 1979 standard decay heat model is conservative.

3. **Because of the inexactness of the new reactivity edit feature, use of values in the edit either directly or as constituent factors in calculations of parameters for comparisons to formal performance criteria must be justified.**

Dominion Evaluation

The editing feature provided in RETRAN 02/MOD005.0 is not used as a quantitative indicator of reactivity feedback and is not used to report analysis results.

<u>DOMINION RESPONSE TO QUESTION 1b</u>

As required by the VEP-FRD-41-A SER, Dominion provided RETRAN model decks to NRC in 1985 as described in Reference 13. Therefore, Dominion satisfied the VEP-FRD-41-A SER requirement. The SER Conclusions section for VEP-FRD-41-A states "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." Dominion has complied with this requirement. Dominion does not interpret the original SER restriction to require submission of model decks after changes are made, especially for changes to plant inputs. Reference 13 was provided to NRC staff on February 26, 2003.

NRC RETRAN QUESTION 2

2. Doppler Reactivity Feedback (page 8 of the submittal dated August 10, 1993)

   a. The Doppler reactivity feedback is calculated by VEPCO's correlation of Doppler reactivity as a function of core average fuel temperature and core burnup. Please provide a technical description of how this correlation is derived, including the codes and methods used. Discuss any limitations or restrictions regarding the use of this correlation.

   b. Discuss the method of calculation and application of suitable weighting factors used to acquire a target Doppler temperature coefficient or Doppler power defect. Indicate the Updated Final Safety Analysis (UFSAR) transients that use this method.

DOMINION RESPONSE TO QUESTION 2.a

The North Anna and Surry Version 1 RETRAN models use a Doppler feedback correlation that is derived from data that models the dependence of Doppler Temperature Coefficient (DTC) on changes in fuel temperature, boron concentration, moderator density and fuel burnup. Through sensitivity studies using the XSDRNPM computer code [Reference 14], the DTC at various conditions was determined. XSDRNPM is a member of the SCALE code package.

The data gathered for North Anna and Surry was used to develop models to predict DTCs. A procedure to calculate a least squares fit to non-linear data with the Gauss-Newton iterative method was used to determine fit coefficients for the collected data. The model values and the percentage difference between the model and XSDRNPM values were determined. The model was also compared to 2D PDQ and 3D PDQ quarter core predictions. The PDQ code is described in Reference 15. The largest percentage difference between the model and the XSDRNPM and PDQ cases is within the nuclear reliability factor for DTC in Reference 16 over the range of conditions of interest to non-LOCA accident analysis.

It was shown that the effect of burnup, boron, and moderator specific volume could be represented as multipliers to the base DTC versus fuel temperature curve. The Doppler correlation has a core average fuel temperature component, DTC_{Tf}, and a burnup component, BURNMP. Since during a transient the burnup may be assumed to be constant, the burnup multiplier of the Doppler correlation is also assumed to be constant. To separate the reactivity feedbacks into a prompt and slower component, the impact of boron concentration and moderator density changes on the Doppler are assumed to be accounted for in the moderator feedback modeling, as these are slower feedback phenomena. Hence, the Doppler reactivity feedback is dependent only on changes in fuel temperature, which provides the prompt feedback component. The boron concentration and moderator density (specific volume) multipliers in the DTC correlation are thereby set to 1.

The DTC correlation is qualified over the range of core design DTC limits for North Anna and Surry and is described by the following equation:

$$DTC(pcm/°F) = DTC_{Tf} * BURNMP * WF$$

where

DTC_{Tf}, the fuel temperature dependence, equals $A*T_f^{0.5} + B*T_f + C$

T_f is the effective core average fuel temperature in °F and A, B, and C are correlation coefficients

BURNMP, which models burnup changes, equals DTC_{ref}/DTC_{Tf547}

DTC_{ref} is the reference DTC at the burnup of interest at hot-zero-power with 2000 ppm boron (pcm/°F)

DTC_{Tf547} is the solution to the above DTC_{Tf} equation at 547 °F.

WF is the user supplied weighting factor term that allows the user to adjust the design information to bound specific Doppler defects.

DOMINION RESPONSE TO QUESTION 2.b

The Doppler feedback can be adjusted to a target DTC at a given fuel temperature by changing the weighting factor. For FSAR analyses in which the Doppler reactivity feedback is a key parameter, the target DTC used in RETRAN is either a least negative or most negative DTC. The RETRAN Doppler weighting factor is set so that RETRAN will initialize to the Reload Safety Analysis Checklist (RSAC) DTC limit at a core average fuel temperature that corresponds to the conditions at which the RSAC DTC limit was set.

To set the weighting factor to provide a least negative DTC, the DTC correlation is solved for the Doppler weighting factor, WF, for the appropriate core average fuel temperature and least negative DTC values. This value of the weighting factor is then entered in RETRAN control input. Likewise, to set the weighting factor to provide a most negative DTC, the weighting factor is solved using the DTC correlation with the appropriate core average fuel temperature and most negative DTC value.

All non-LOCA UFSAR transient RETRAN analyses, with the exception of the rod ejection event, apply an appropriate weighting factor to acquire a target Doppler temperature coefficient.

The rod ejection event requires additional Doppler reactivity feedback. This additional feedback is calculated as a PWF (power weighting factor), and the Doppler weighting factor calculated as described herein needs to be multiplied by the PWF before being input to the RETRAN model. The application of the power weighting factor to rod ejection analyses is described in Section 2.2.3 of Reference 4.

NRC RETRAN QUESTION 3

3. By letter dated August 10, 1993, VEPCO discussed the expansion of the North Anna RETRAN model from two geometric configurations to four geometric configurations. The model options increased from a one-loop and two-loop reactor coolant system (RCS) geometry with a single-node steam generator secondary side, to one-loop and three-loop RCS geometry with either single- or multi-node steam generator secondary side. Please discuss the process used for choosing which of the four configurations to use for a particular transient, and identify which model is used for each of the North Anna and Surry UFSAR, Chapter 15, transients that were evaluated using RETRAN.

DOMINION RESPONSE TO QUESTION 3

Historically, choosing between the 1-loop and 2-loop RCS RETRAN models was based on the expected plant response from the transient and on the importance of modeling differences between RCS loops. For example, a steamline break affects the conditions in the faulted steam generator RCS loop different from the other loops. When advances in computer processor speed and memory eliminated the need to collapse symmetric loops, Dominion developed 3-loop RCS models and retired the 1-loop and 2-loop models. Some UFSAR analyses of record reflect 1-loop and 2-loop RETRAN analyses because the events have not been reanalyzed since the implementation of the 3-loop models. RETRAN analyses in the UFSAR use the single-node SG secondary model. Dominion uses the multi-node steam generator secondary model for sensitivity studies to confirm the conservatism in the single-node SG secondary. Subsequent to retirement of the 1-loop and 2-loop models, licensing analyses have used the 3-loop RCS geometry with a single-node steam generator. Dominion anticipates that this will continue to be our RETRAN analysis model going forward.

Tables 3a and 3b below show the selected RCS model type for each UFSAR event analyzed with RETRAN for North Anna and Surry, respectively. All analyses use a single-node steam generator secondary model. Note that some UFSAR non-LOCA events have not been analyzed with RETRAN. Future applications of RETRAN may involve analyzing these events to remove the dependence on the vendor. Those analyses would be performed in accordance with regulatory requirements and limitations in the RETRAN SERs and VEP-FRD-41-A.

Table 3a: North Anna UFSAR Chapter 15 Event and RETRAN Model

Event	UFSAR Section	RETRAN Model
Condition II: Events of Moderate Frequency		
Uncontrolled Rod Cluster Control Assembly from a Subcritical Condition	15.2.1	1-Loop
Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power	15.2.2	3-Loop
Uncontrolled Boron Dilution	15.2.4	1-Loop
Loss of External Electric Load and/or Turbine Trip	15.2.7	3-Loop
Loss of Normal Feedwater	15.2.8	3-Loop
Loss of Offsite Power to the Station Auxiliaries	15.2.9	3-Loop
Excessive Heat Removal Due to Feedwater System Malfunctions	15.2.10	2-Loop
Excessive Load Increase Incident	15.2.11	1-Loop, 3-Loop
Accidental Depressurization of the Reactor Coolant System	15.2.12	1-Loop
Accidental Depressurization of the Main Steam System	15.2.13	3-Loop
Condition III: LOCA and Related Accidents		
Minor Secondary System Pipe Breaks	15.3.2	3-Loop
Complete Loss of Forced Reactor Coolant Flow	15.3.4	1-Loop
Condition IV: Limiting Faults		
Major Secondary System Pipe Rupture	15.4.2	3-Loop
Steam Generator Tube Rupture	15.4.3	2-Loop and 3-Loop
Locked Reactor Coolant Pump Rotor	15.4.4	2-Loop and 3-Loop
Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	15.4.6	1-Loop

Note that the Rupture of a Control Rod Drive Mechanism Housing, Complete Loss of Forced Reactor Coolant Flow, and Locked Reactor Coolant Pump Rotor analyses have been performed with the RETRAN 3 Loop model as part of the transition to Framatome fuel. These evaluations are currently being reviewed by the NRC and are therefore not incorporated in the current North Anna UFSAR.

Table 3b: Surry UFSAR Chapter 14 Event and RETRAN Model

Event	UFSAR Section	RETRAN Model
Condition II: Events of Moderate Frequency		
Uncontrolled Control-Rod Assembly Withdrawal From a Subcritical Condition	14.2.1	1-Loop
Uncontrolled Control-Rod Assembly Withdrawal at Power	14.2.2	1-Loop
Chemical and Volume Control System Malfunction	14.2.5.2.3	1-Loop
Excessive Heat Removal Due to Feedwater System Malfunctions	14.2.7	FW Temp. Reduction - 3-Loop Excess Feedwater Flow - 2-Loop
Excessive Load Increase Incident	14.2.8	3-Loop
Loss of Reactor Coolant Flow Flow Coastdown Incidents	14.2.9.1	1-Loop
Locked Rotor Incident	14.2.9.2	3-Loop
Loss of External Electrical Load	14.2.10	3-Loop
Loss of Normal Feedwater	14.2.11	3-Loop
Loss of all Alternating Current to the Station Auxiliaries	14.2.12	3-Loop
Standby Safeguards Analyses		
Steam Generator Tube Rupture	14.3.1	2-Loop
Rupture of a Main Steam Pipe (DNB)	14.3.2	3-Loop
Rupture of a Control Rod Drive Mechanism Housing (Control Rod Assembly Ejection)	14.3.3	1-Loop
Feedline Break outside Containment	Appendix 14B	3-Loop

References used in Dominion Responses to RETRAN Questions

1) Letter from C.O. Thomas (USNRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, RETRAN – A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems, and EPRI NP-1850-CCM, RETRAN-02 – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 2, 1984.

2) Letter from A. C. Thadani (USNRC) to R. Furia (GPU), "Acceptance for Referencing Topical Report EPRI-NP-1850 CCM-A, Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004," October 19, 1988.

3) Letter from A. C. Thadani (USNRC) to W. J. Boatwright (RETRAN02 Maintenance Group), "Acceptance for Use of RETRAN02 MOD005.0," November 1, 1991.

4) Virginia Power Topical Report VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient", NRC SER dated September 26, 1984.

5) Virginia Power Topical Report VEP-FRD-41-A, "VEPCO Reactor System Transient Analysis using the RETRAN Computer Code," May 1985.

6) Westinghouse report WCAP-9227, "Reactor Core Response to Excessive Secondary Steam Releases," January 1978.

7) Westinghouse report WCAP-8844, "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," November 1977.

8) Westinghouse report WCAP-7907-A, "LOFTRAN Code Description," April 1984.

9) Letter, M.L. Bowling (VEPCO) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Supplemental Information on the RETRAN NSSS Model," Serial 93-505, August 10, 1993.

10) Letter, Stuart A. Richards (USNRC) to Gary Vine (EPRI), "Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," January 25, 2001.

11) Westinghouse report WCAP-10858-P-A, "AMSAC Generic Design Package," October 1986.

12) Letter from W. L. Stewart (VEPCO) to H. R. Denton (USNRC), "Virginia Electric Power Company, North Anna Power Station Unit No. 2, Response to the Additional Request for Information Concerning Low Power Natural Circulation Testing," Serial No. 427A, August 25, 1983.

References used in Dominion Responses to RETRAN Questions (continued)

13) Letter, W. L. Stewart (VEPCO) to Harold R. Denton (USNRC), "Virginia Power, Surry and North Anna Power Stations, Reactor System Transient Analyses," Serial No. 85-570, August 21, 1985.

14) ORNL-NUREG-CSD-2-Vol 2, Rev. 1, "XSDRNPM-S: A One-Dimensional Discrete-Ordinates Code for Transport Analysis," June 1983.

15) Virginia Power Topical Report VEP-NAF-1, "The PDQ Two Zone Model," July 1990.

16) Virginia Power Topical Report VEP-FRD-45A, "VEPCO Nuclear Design Reliability Factors," October 1982.

17) Branch Technical Position APCSB9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," 1975.

18) EPRI Report, EPRI-NP-1850-CCM-A, Volume 1, Rev. 4, "RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems."

19) Letter from W. L. Stewart (VEPCO) to Harold R. Denton (USNRC), "VEPCO Reactor System Transient Analyses", Serial No. 376, July 12, 1984.

Attachment 2

Response to NRC
PDQ Two Zone Model Questions

Virginia Electric and Power Company
(Dominion)
North Anna and Surry Power Stations

PDQ Code and Model Review, Topical Report VEP-NAF-1, "PDQ Two Zone Model," VEPCO submittal dated October 1, 1990

NRC PDQ QUESTION 1

1. By letter dated December 2, 2002, VEPCO stated that the accuracy of the PDQ model is verified each cycle during startup physics testing and during routine core follow. Please provide representative results from a recent refueling outage (comparisons between the startup physics test data and the PDQ predictions) that demonstrate the accuracy of this model.

DOMINION RESPONSE TO QUESTION 1

The following results are from the N1C16 startup physics tests in October, 2001.

N1C16 STARTUP PHYSICS TESTING RESULTS (October, 2001)

Parameter	Measured	Predicted	Difference (P-M) or (P-M)/M*100	Nuclear Reliability Factor
Critical Boron Concentration (HZP, ARO) ppm	2109	2133	24	± 50
Critical Boron Concentration (HZP, reference bank in) ppm	1897	1917	20	± 50
Critical Boron Concentration (HFP, ARO, EQ XE) ppm	1405	1429	24	± 50
Isothermal Temperature Coefficient (HZP, ARO) pcm/°F	-2.87	-3.29	-0.42	± 3.0
Differential Boron Worth (HZP, ARO) pcm/ppm	-6.59	-6.46	-2.0%	1.10
Reference Bank Worth (B-bank, dilution) pcm	1393.2	1396	0.2%	1.10
D-bank Worth (Rod Swap), pcm	944.6	979	3.6%	1.10
C-bank Worth (Rod Swap), pcm	760.4	779.3	2.5%	1.10
A-bank Worth (Rod Swap), pcm	356.6	348.4	-2.3%	1.10
SB-bank Worth (Rod Swap), pcm	930.5	969.8	4.2%	1.10
SA-bank Worth (Rod Swap), pcm	1012.5	1003.4	-0.9%	1.10
Total Bank Worth, pcm	5397.6	5476	1.5%	1.10
HFP ARO EQ XE FΔH (BOC)	1.405	1.378	-1.9%	1.05
HFP ARO EQ XE F_Q (BOC)	1.654	1.601	-3.2%	1.075
HFP ARO EQ XE Axial Offset (BOC)	-2.5	-3.0	-0.5%	N/A

NORTH ANNA UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
29% POWER

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1			PREDICTED MEASURED PCT DIFFERENCE				0.246 0.249 1.0	0.275 0.280 1.8	0.242 0.246 1.6			PREDICTED MEASURED PCT DIFFERENCE				1
2					0.331 0.336 1.5	0.649 0.656 1.0	1.053 1.061 0.8	0.836 0.844 1.0	1.045 1.062 1.6	0.647 0.674 4.3	0.331 0.338 2.2					2
3				0.365 0.381 4.3	1.179 1.200 1.8	1.299 1.313 1.1	1.170 1.177 0.6	1.301 1.301 0.0	1.168 1.175 0.6	1.299 1.312 1.0	1.183 1.196 1.1	0.368 0.379 3.0				3
4			0.368 0.381 3.6	0.898 0.919 2.3	1.333 1.358 1.9	1.323 1.335 0.9	1.305 1.309 0.3	1.243 1.244 0.0	1.304 1.303 -0.1	1.324 1.318 -0.4	1.338 1.343 0.4	0.897 0.908 1.2	0.365 0.369 1.1			4
5		0.356 0.367 3.1	1.204 1.259 4.6	1.346 1.370 1.8	1.221 1.228 0.6	1.262 1.265 0.2	1.168 1.168 0.0	1.236 1.233 -0.2	1.168 1.164 -0.3	1.261 1.258 -0.3	1.220 1.222 0.1	1.339 1.358 1.4	1.197 1.212 1.2	0.355 0.361 1.9		5
6		0.665 0.668 0.5	1.321 1.334 0.9	1.334 1.338 0.3	1.266 1.252 -1.1	1.017 1.012 -0.5	1.145 1.139 -0.6	1.071 1.065 -0.5	1.143 1.133 -0.8	1.017 1.012 -0.4	1.265 1.261 -0.3	1.332 1.333 0.1	1.319 1.330 0.8	0.665 0.677 1.9		6
7	0.258 0.256 -1.0	1.070 1.060 -0.9	1.186 1.166 -1.7	1.317 1.306 -0.8	1.174 1.164 -0.9	1.165 1.135 -0.9	1.038 1.027 -1.1	1.005 0.993 -1.2	1.038 1.018 -1.9	1.148 1.135 -1.1	1.174 1.163 -0.9	1.316 1.294 -1.7	1.187 1.195 0.7	1.074 1.108 3.2	0.259 0.267 3.1	7
8	0.284 0.282 -1.0	0.853 0.845 -1.0	1.320 1.298 -1.7	1.255 1.244 -0.9	1.243 1.232 -0.9	1.074 1.065 -0.9	1.005 0.994 -1.1	0.996 0.984 -1.2	1.005 0.991 -1.4	1.074 1.063 -1.0	1.243 1.233 -0.7	1.255 1.251 -0.3	1.320 1.333 1.0	0.853 0.879 3.0	0.284 0.293 2.9	8
9	0.259 0.258 -0.6	1.074 1.069 -0.5	1.187 1.181 -0.5	1.316 1.310 -0.5	1.174 1.172 -0.2	1.148 1.139 -0.8	1.038 1.025 -1.2	1.005 0.991 -1.4	1.038 1.024 -1.3	1.145 1.133 -1.1	1.174 1.167 -0.6	1.317 1.319 0.2	1.186 1.201 1.3	1.070 1.097 2.5	0.258 0.264 2.4	9
10		0.665 0.665 0.1	1.319 1.326 0.6	1.332 1.330 -0.1	1.265 1.259 -0.5	1.017 1.007 -0.9	1.143 1.123 -1.7	1.071 1.053 -1.7	1.145 1.127 -1.6	1.017 1.003 -1.4	1.266 1.255 -0.9	1.334 1.339 0.4	1.321 1.344 1.8	0.665 0.691 3.8		10
11		0.355 0.356 0.3	1.197 1.203 0.5	1.339 1.337 -0.1	1.220 1.211 -0.7	1.261 1.246 -1.2	1.168 1.147 -1.8	1.236 1.211 -2.0	1.168 1.146 -1.9	1.262 1.229 -2.6	1.221 1.198 -1.9	1.346 1.361 1.1	1.204 1.235 2.6	0.356 0.366 3.0		11
12			0.365 0.373 2.4	0.897 0.897 -0.1	1.338 1.329 -0.7	1.324 1.307 -1.3	1.304 1.276 -2.2	1.243 1.218 -2.0	1.305 1.286 -1.5	1.323 1.309 -1.1	1.333 1.334 0.1	0.898 0.939 4.5	0.368 0.407 10.6			12
13				0.368 0.367 -0.3	1.183 1.179 -0.3	1.299 1.286 -1.0	1.168 1.149 -1.6	1.301 1.270 -2.4	1.170 1.163 -0.7	1.299 1.292 -0.5	1.179 1.180 0.1	0.365 0.372 1.9				13
14					0.331 0.342 3.4	0.647 0.644 -0.4	1.045 1.037 -0.7	0.836 0.834 -0.1	1.053 1.077 2.3	0.649 0.653 0.6	0.331 0.332 0.2					14
15			STANDARD DEVIATION =1.235				0.242 0.243 0.3	0.275 0.276 0.2	0.246 0.251 1.8			AVERAGE PCT DIFFERENCE = 1.3				15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

MAP NO: N1-16-01

CONTROL ROD POSITIONS:

D BANK AT 150 STEPS

DATE: 10/10/01

F-Q(Z) = 2.108

F-DH(N) = 1.546

F(Z) = 1.283

BURNUP = 5.0 MWD/MTU

SUMMARY

POWER: 29%

CORE TILT:

NW 1.0037	NE 1.0031
SW 0.9922	SE 1.0010

A.O. = -6.233

NORTH ANNA UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
74% POWER

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1			PREDICTED MEASURED PCT DIFFERENCE				0.271 0.272 0.4	0.311 0.311 0.1	0.267 0.269 0.9			PREDICTED MEASURED PCT DIFFERENCE				1
2					0.339 0.342 0.8	0.660 0.664 0.6	1.082 1.088 0.6	0.921 0.927 0.7	1.074 1.087 1.2	0.657 0.678 3.1	0.339 0.344 1.6					2
3				0.369 0.376 2.0	1.140 1.152 1.1	1.258 1.267 0.7	1.164 1.169 0.4	1.297 1.300 0.2	1.162 1.168 0.6	1.258 1.268 0.9	1.143 1.152 0.8	0.372 0.375 0.8				3
4			0.371 0.381 2.5	0.880 0.892 1.4	1.278 1.293 1.2	1.285 1.292 0.6	1.275 1.279 0.3	1.227 1.230 0.2	1.274 1.276 0.2	1.285 1.286 0.1	1.282 1.287 0.4	0.879 0.886 0.8	0.368 0.371 0.7			4
5		0.361 0.369 2.1	1.159 1.199 3.5	1.287 1.302 1.2	1.196 1.198 0.2	1.253 1.255 0.1	1.172 1.174 0.2	1.227 1.227 0.0	1.171 1.172 0.1	1.253 1.254 0.1	1.195 1.199 0.3	1.281 1.296 1.1	1.154 1.164 0.9	0.360 0.362 0.5		5
6		0.673 0.672 0.0	1.274 1.279 0.4	1.291 1.291 0.0	1.256 1.244 -0.9	1.103 1.101 -0.2	1.176 1.173 -0.3	1.100 1.098 -0.2	1.173 1.171 -0.2	1.103 1.104 0.1	1.255 1.255 0.0	1.290 1.292 0.2	1.272 1.281 0.7	0.673 0.682 1.3		6
7	0.283 0.279 -1.3	1.094 1.081 -1.3	1.174 1.152 -1.9	1.282 1.270 -0.9	1.175 1.167 -0.7	1.175 1.169 -0.6	1.084 1.077 -0.7	1.053 1.048 -0.5	1.084 1.079 -0.5	1.177 1.173 -0.4	1.175 1.171 -0.4	1.282 1.269 -1.0	1.176 1.184 0.7	1.098 1.125 2.4	0.284 0.290 2.2	7
8	0.319 0.315 -1.3	0.935 0.923 -1.3	1.310 1.282 -2.1	1.235 1.223 -1.0	1.231 1.226 -0.4	1.101 1.097 -0.4	1.053 1.048 -0.6	1.049 1.043 -0.5	1.053 1.048 -0.5	1.101 1.096 -0.5	1.231 1.228 -0.3	1.235 1.235 0.0	1.310 1.321 0.8	0.935 0.951 1.7	0.319 0.324 1.6	8
9	0.284 0.281 -1.0	1.098 1.089 -0.9	1.176 1.166 -0.8	1.282 1.276 -0.5	1.175 1.178 0.2	1.177 1.173 -0.3	1.084 1.078 -0.6	1.053 1.046 -0.7	1.084 1.078 -0.6	1.175 1.169 -0.5	1.175 1.174 -0.1	1.282 1.287 0.4	1.174 1.186 1.0	1.094 1.112 1.6	0.283 0.284 0.5	9
10		0.673 0.671 -0.3	1.272 1.275 0.2	1.290 1.287 -0.2	1.255 1.252 -0.3	1.103 1.098 -0.4	1.173 1.163 -0.9	1.100 1.089 -0.9	1.176 1.166 -0.9	1.103 1.097 -0.6	1.256 1.254 -0.2	1.291 1.303 0.9	1.274 1.291 1.4	0.673 0.692 2.8		10
11		0.360 0.360 -0.1	1.154 1.155 0.1	1.281 1.278 -0.2	1.195 1.190 -0.5	1.253 1.243 -0.8	1.171 1.158 -1.1	1.227 1.211 -1.3	1.172 1.158 -1.2	1.253 1.234 -1.5	1.196 1.186 -0.8	1.287 1.302 1.1	1.159 1.176 1.5	0.361 0.368 1.9		11
12			0.368 0.371 0.7	0.879 0.877 -0.3	1.282 1.274 -0.6	1.285 1.273 -0.9	1.274 1.256 -1.5	1.227 1.209 -1.5	1.275 1.261 -1.1	1.285 1.275 -0.8	1.278 1.282 0.3	0.880 0.914 3.9	0.371 0.380 2.3			12
13				0.372 0.370 -0.5	1.143 1.138 -0.5	1.258 1.247 -0.9	1.162 1.146 -1.4	1.297 1.270 -2.1	1.164 1.155 -0.7	1.258 1.248 -0.8	1.140 1.140 0.0	0.369 0.375 1.6				13
14					0.339 0.343 1.2	0.657 0.653 -0.7	1.074 1.064 -0.9	0.921 0.916 -0.5	1.082 1.096 1.3	0.660 0.661 0.1	0.339 0.339 0.0					14
15			STANDARD DEVIATION =0.694				0.267 0.263 -1.4	0.311 0.310 -0.4	0.271 0.273 0.9			AVERAGE PCT DIFFERENCE = 0.8				15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

MAP NO: N1-16-02

DATE: 10/11/01

CONTROL ROD POSITIONS:

D BANK AT 192 STEPS

F-Q(Z) = 1.848

F-DH(N) = 1.451

F(Z) = 1.184

BURNUP = 24.0 MWD/MTU

SUMMARY

POWER: 74%

CORE TILT:

NW 1.0014	NE 1.0039
SW 0.9933	SE 1.0014

A.O. = 0.088

NORTH ANNA UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
100% POWER

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1		PREDICTED MEASURED PCT DIFFERENCE					0.283 0.283 0.1	0.330 0.334 1.3	0.279 0.281 0.9			PREDICTED MEASURED PCT DIFFERENCE				1
2					0.342 0.343 0.2	0.662 0.662 -0.1	1.087 1.085 -0.2	0.963 0.961 -0.2	1.080 1.089 0.8	0.660 0.682 3.3	0.342 0.349 2.0					2
3				0.371 0.384 3.4	1.118 1.122 0.4	1.239 1.238 -0.1	1.156 1.150 -0.5	1.297 1.279 -1.4	1.154 1.155 0.1	1.239 1.251 1.0	1.121 1.135 1.3	0.373 0.384 2.9				3
4			0.372 0.379 1.6	0.871 0.878 0.8	1.255 1.254 -0.1	1.263 1.262 -0.1	1.265 1.262 -0.3	1.217 1.213 -0.3	1.265 1.266 0.1	1.264 1.269 0.4	1.259 1.268 0.7	0.870 0.876 0.6	0.370 0.371 0.4			4
5		0.362 0.367 1.2	1.133 1.160 2.3	1.263 1.272 0.7	1.185 1.190 0.4	1.258 1.257 -0.1	1.173 1.173 0.0	1.228 1.230 0.2	1.173 1.176 0.3	1.258 1.262 0.3	1.185 1.192 0.6	1.258 1.257 -0.1	1.128 1.133 0.4	0.361 0.371 2.8		5
6		0.673 0.669 -0.6	1.251 1.248 -0.2	1.268 1.261 -0.5	1.260 1.242 -1.5	1.158 1.154 -0.3	1.196 1.199 0.3	1.110 1.119 0.9	1.194 1.198 0.4	1.158 1.158 0.0	1.259 1.259 0.0	1.267 1.262 -0.4	1.250 1.250 0.0	0.673 0.679 0.9		6
7	0.294 0.289 -1.6	1.097 1.079 -1.6	1.164 1.139 -2.1	1.270 1.255 -1.2	1.176 1.165 -0.9	1.195 1.192 -0.2	1.100 1.108 0.7	1.067 1.074 0.7	1.100 1.107 0.6	1.197 1.202 0.4	1.176 1.173 -0.2	1.270 1.253 -1.4	1.165 1.159 -0.5	1.101 1.117 1.4	0.295 0.299 1.5	7
8	0.337 0.332 -1.7	0.975 0.958 -1.8	1.307 1.276 -2.4	1.223 1.208 -1.2	1.231 1.222 -0.7	1.111 1.109 -0.2	1.067 1.069 0.2	1.062 1.065 0.3	1.067 1.073 0.5	1.111 1.126 1.4	1.231 1.231 0.0	1.223 1.213 -0.8	1.307 1.282 -1.9	0.975 0.994 2.0	0.337 0.343 1.9	8
9	0.295 0.291 -1.3	1.101 1.087 -1.2	1.165 1.152 -1.1	1.270 1.264 -0.5	1.176 1.184 0.7	1.197 1.198 0.1	1.100 1.100 -0.1	1.067 1.066 -0.1	1.100 1.098 -0.2	1.195 1.187 -0.7	1.176 1.174 -0.1	1.270 1.272 0.1	1.164 1.167 0.3	1.097 1.116 1.7	0.294 0.300 2.0	9
10		0.673 0.668 -0.7	1.250 1.245 -0.4	1.267 1.265 -0.1	1.259 1.261 0.1	1.158 1.156 -0.1	1.194 1.189 -0.4	1.110 1.103 -0.6	1.196 1.188 -0.7	1.158 1.150 -0.7	1.260 1.260 0.0	1.268 1.281 1.0	1.251 1.270 1.5	0.673 0.696 3.5		10
11		0.361 0.361 -0.1	1.128 1.131 0.3	1.258 1.259 0.1	1.185 1.186 0.1	1.258 1.252 -0.4	1.173 1.162 -0.9	1.228 1.214 -1.1	1.173 1.161 -1.1	1.258 1.239 -1.5	1.185 1.183 -0.2	1.263 1.285 1.7	1.133 1.157 2.1	0.362 0.372 2.6		11
12			0.370 0.381 3.0	0.870 0.874 0.4	1.259 1.258 -0.1	1.264 1.255 -0.7	1.265 1.242 -1.8	1.217 1.198 -1.6	1.265 1.252 -1.1	1.263 1.254 -0.7	1.255 1.264 0.7	0.871 0.913 4.8	0.372 0.389 4.5			12
13				0.373 0.374 0.2	1.121 1.122 0.1	1.239 1.231 -0.6	1.154 1.138 -1.4	1.297 1.267 -2.3	1.156 1.147 -0.8	1.239 1.225 -1.2	1.118 1.120 0.2	0.371 0.379 2.1				13
14					0.342 0.351 2.6	0.660 0.659 -0.2	1.080 1.073 -0.6	0.963 0.959 -0.4	1.087 1.108 1.9	0.662 0.664 0.2	0.342 0.343 0.1					14
15		STANDARD DEVIATION =0.906					0.279 0.279 0.2	0.330 0.330 0.0	0.283 0.287 1.4			AVERAGE PCT DIFFERENCE = 0.9				15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

MAP NO: N1-16-03 DATE: 10/23/01

CONTROL ROD POSITIONS: D BANK AT 226 STEPS

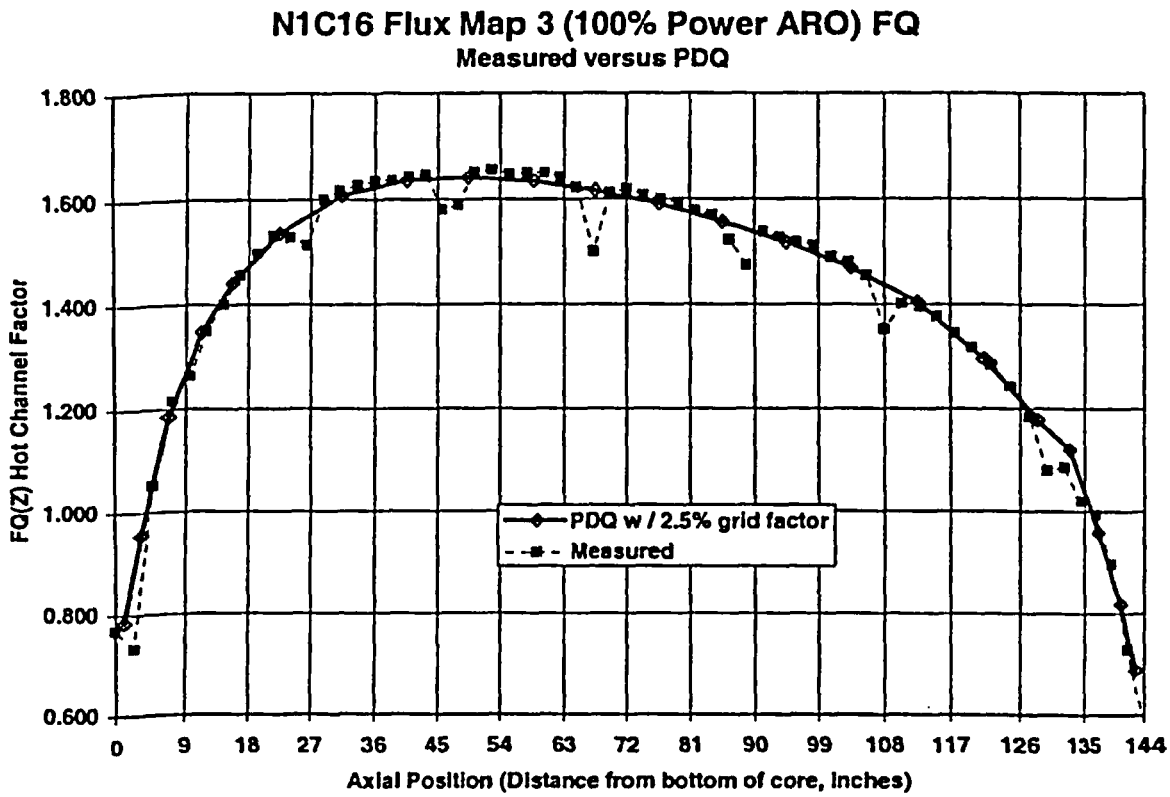
F-Q(Z) = 1.786
F-DH(N) = 1.405
F(Z) = 1.139
BURNUP = 436.4 MWD/MTU

SUMMARY
POWER: 100%

CORE TILT:

NW 0.9982	NE 1.0035
SW 0.9954	SE 1.0029

A.O. = -2.537



NRC PDQ QUESTION 2

There do not appear to be any limitations or restrictions associated with the use of PDQ Two Zone as described in VEP-NAF-1. Please justify that PDQ Two Zone is applicable over all ranges of operation expected for North Anna and Surry.

DOMINION RESPONSE TO QUESTION 2

Use of the PDQ Two Zone Model is limited to North Anna and Surry cores containing fuel that is similar to existing 17x17 and 15x15 designs. The range of applicability is stated in general terms in Section 2.1 of VEP-FRD-42 Rev 2:

"These models have been used to model the entire range of cores at the Surry and North Anna power stations, including evolutionary changes in fuel enrichment, fuel density, loading pattern strategy, spacer grid design and material, fuel clad alloy, and burnable poison material and design. Some of these changes were implemented as part of various Lead Test Assembly programs, and have included fuel assemblies from both Westinghouse and Framatome-ANP. The predictive accuracy of the models throughout these changes demonstrates that incremental design variations in fuel similar to the Westinghouse design are well within the applicable range of the core design models. Each model has sufficient flexibility such that minor fuel assembly design differences similar to those noted can be adequately accounted for using model design input variables."

Limitations associated with the PDQ Two Zone models stem primarily from consideration of the source of collapsed cross section data (primarily CELL2, a pin cell model) and from practical considerations involving the level of complexity that can be accommodated in PDQ. Based on these considerations, the scope of benchmarking that has been performed to date, and the range of core designs successfully modeled in the past, the PDQ Two Zone model should be restricted according to the following characteristics:

1) Geometry
   a) Square pitch fuel (cylindrical fuel pellets and rods)
   b) 15x15 or 17x17 design
   c) 5x5 mesh blocks per assembly (x-y)
   d) 26 axial nodes (22 in the fuel region)
   e) ¼ core or full core representation
2) Fuel Material
   a) Low enriched UO_2 (4.6 w/o U_{235} or less)
      i) Cores with fuel up to 4.45 w/o have been successfully modeled to date
      ii) Cross section behavior (enrichment trends and fidelity to CELL2) has been checked up to 4.6 w/o U_{235} for burnups up to 76 GWD/T.
   b) Fuel pin burnup of approximately 70 GWD/T has been achieved in PDQ Two Zone designed cores as part of a high burnup demonstration program.
3) Burnable poisons
   a) Discrete rods inserted into fuel assembly guide thimbles
      i) Both annular borosilicate glass and solid B4C in alumina designs have been well predicted throughout many cycles of operation
      ii) Both SS304 and zirconium based cladding has been used
   b) Modeling flexibility has been demonstrated for BP configuration (number of fingers, boron enrichment, poison length, and poison stack axial alignment)
4) Control rods
   a) Ag-In-Cd rods with stainless steel clad (extensive validation and experience)
   b) Hf metal rods in zirconium based clad have been used for vessel fluence reduction in Surry Unit 1
5) Fuel assembly
   a) Modeling flexibility has been demonstrated for Inconel and zirconium based grids of various designs and sizes

There are no current plans for fuel design, core design, or operating strategy changes that would exceed the design characteristics outlined above. There are fuel products in use in the industry, which would be technically possible, but impractical to model in the PDQ Two Zone model (such as fuel with integral poisons). No further development is planned for PDQ and NOMAD. Rather, Dominion plans to transition from using PDQ and NOMAD as primary design tools to use of the CMS models (principally CASMO-4 and SIMULATE-3) as soon as practicable. Topical Report DOM-NAF-1 was submitted in June of 2002. The NRC SER for DOM-NAF-1 was received on March 12, 2003.

NRC PDQ QUESTION 3

PDQ Two Zone cross section representation has been improved through the addition of multiple G-factor capability. Please discuss the methodologies used to determine these factors and discuss when and how they are applied. Include a discussion of the "fictitious crod isotope" mentioned on page 2-23 of your dated October 1, 1990.

DOMINION RESPONSE TO QUESTION 3

The addition of multiple G-factor capability was required to meet these goals for the PDQ Two Zone model:

1) A unified set of cross section data to accurately span the entire operating range of the cores (i.e., temperatures, boron concentration, BP combinations, burnup, etc.)
2) A system with the flexibility to model variations such as spacer grid changes, BP enrichment variations, fuel enrichment changes, and clad isotopic changes without requiring the generation of new cross section data.

The process used for G-factor selection can be broken down as follows:

1) Identify known required physical variables (such as moderator temperature, moderator density, fuel temperature, and soluble boron concentration).
2) Identify significant isotopic inter-dependencies (such as the U-235 / Pu-239 interaction in thermal absorption and thermal fission cross sections) using CELL-2.
3) Sort in order of importance and modeling complexity.
4) Develop the primary dependence tables.
5) Develop the G-factor (multiplier) tables.

The importance of a particular factor was judged by estimating the first-order reactivity impact (essentially a partial derivative). The complexity of modeling varies according to the degree of separability from other variables. PDQ uses a table system to represent cross sections. The first table for a particular cross section represents the variation of the cross section using the three most important variables. Additional tables are treated as multipliers (G-factors) on the interpolant from the first table.

Each table has a primary variable (called the diagonal) and up to two secondary variables. The diagonal represents the nominal combination of the three variables. Branch cases are used to perturb each secondary variable. The tables can be considered a dual 2-D representation and not a true 3-D representation since the secondary variables cannot be changed simultaneously.

For example, the U^{235} microscopic thermal absorption cross section is a function of the U^{235} number density, the Pu^{239} number density, and the Pu^{241} number density. The diagonal represents the U^{235} cross section at combinations of the three nuclides found in a CELL-2 depletion of a particular enrichment at nominal conditions. The branch cases vary the quantity of Pu^{239} or Pu^{241} at several of the nominal burnup points. In this way, the second order reactivity impact of depleting a fuel assembly in PDQ at off-nominal conditions (such as more BP, hotter moderator temperatures, or more soluble boron) resulting in more Pu is directly captured without use of a "history" variable. In addition, this type of representation makes the model flexible for modeling different fuel enrichments (typically within ± 0.2 w/o of the CELL-2 enrichment).

Important cross section effects that are not captured in the main cross section table are applied by use of multiplicative G-factors. Each G-factor table is constructed in the same manner as the main cross section table. Using the previous example for U^{235}, one G-factor for the thermal absorption cross section is a function of moderator temperature, moderator density, and fuel burnup. The value of the G-factor at the "reference" moderator temperature (583.4 °F for North Anna) is 1.0. The ratio of the U^{235} thermal absorption cross section at other temperatures to the reference value at 583.4 °F is provided at several diagonal points ranging from HFP to CZP temperatures. The variation in these ratio values caused by changes in moderator density (same moderator temperature but a different pressure) or burnup is provided at the branch points.

An important factor in this method of cross section representation is that PDQ Two Zone features a predominantly microscopic model. That is, most cross sections are represented by means of direct tracking of nuclide number densities via depletion chains coupled with microscopic cross section data. A total of 34 physical nuclides are tracked in addition to several pseudo-nuclides which represent state variables (such as moderator temperature) or lumped macroscopic effects (such as the remaining fission products or control rod insertion). Tracking individual nuclides means that the first order effect on reactivity of a change in nuclide concentration is directly modeled even with a constant microscopic cross section. Complex representation of microscopic cross section dependence serves to provide accuracy at the second and third order level even over an extended range of state variables, and provides modeling flexibility for physical changes in fuel design (such as grid material or grid volume changes).

The cross section modeling process described is complex and was designed to be a one-time event. Sufficient modeling flexibility was designed in to preclude the need for core designers to perform cross section modeling in addition to core design work. Over the 14 years since the G-factor strategy was developed, few changes have been made. These changes have been predominantly to extend capabilities rather than revise strategy. One such change was the addition of cross section data to model use of Hafnium rods for reactor vessel fluence suppression.

An important component of cross section modeling is the verification that the cross section representation is accurate and robust. Part of the G-factor development process involved comparison of PDQ single assembly model eigenvalues to CELL-2 using a wide range of state variables and burnup. A goal of matching reactivity within 100 pcm was usually met for cases using unrodded fuel (the only comparison to a pin cell model that can be made accurately). In addition, comparisons to KENO calculations were made for fresh fuel over a wide range of state variables, with and without control rods and BP rods. The KENO benchmarking / normalization loop is shown in Figure 2-1 of VEP-NAF-1.

The "crod" isotope is one of the pseudo-nuclides mentioned above. Because CELL-2 is a pin cell model and cannot properly represent control rod insertion, control rod macroscopic cross sections were obtained from a KENO model. These cross sections include not only the primary effect of a change in macroscopic absorption, but also the net change in fuel macroscopic cross sections (including removal and fission). In order to overlay these macroscopic changes on the fuel cross sections, the control rod insertion is treated as the addition of a nuclide named "crod" with a number density of 1.0. The macroscopic cross section changes are represented in tables as microscopic cross sections. When multiplied by the crod number density of 1.0, the full macroscopic effect of the rod insertion is obtained. This model also makes possible an approximate modeling of fractional control rod insertion (insertion into only part of a node

axially) by specifying a volume weighted value for the crod nuclide. For insertion into the top half of a node, the crod nuclide number density is set to 0.5 in that node. Because the crod number density and cross sections are non-physical for a microscopic model, the crod nuclide is specified as non-depleting.

NRC PDQ QUESTION 4

Table 3.2 of this submittal lists the existing nuclear reliability factors and the PDQ Two Zone nuclear uncertainty factors (NUF). Please discuss the methodology used to calculate each of the PDQ NUF values, and indicate when NRC approval was obtained.

DOMINION RESPONSE TO QUESTION 4

VEP-FRD-19A (The PDQ 07 Discrete Model, SER dated May 18, 1981) and VEP-FRD-45A (VEPCO Nuclear Design Reliability Factors, SER dated August 5, 1982) are two NRC approved references relevant to a discussion of nuclear reliability factor methodology.

In VEP-FRD-19A, a total of four cycles of data (startup physics measurements, flux map data, and boron letdown curves) were provided for comparison between predictions and measurements. Overall averages of vendor code differences (measured versus predicted) were also presented. No statistical methodology was used. In the conclusion section, results were stated to be *"predicted typically within"* the following percentages:

- Assembly average power, 2% standard deviation
- Peak FΔH, 2.5%
- Assembly average burnup, 2.5%
- Critical soluble boron concentration, 30 ppm
- Boron worth, 3%
- Integral control rod worth, 6%

The SER for VEP-FRD-19A restates these values and provides the following assessment, which indicates the acceptability of using *"sufficient examples"* which support reasonable uncertainties:

"We have reviewed the data presented to support the conclusions regarding the uncertainties in the calculated results. We conclude the sufficient examples of comparisons between calculation and measurement to permit the evaluation of calculational uncertainties. We concur with the particular values of uncertainties given in the topical report and repeated in Section 1 above."

In VEP-FRD-45A, a more statistically rigorous method was used to derive the NUF/NRF for the total peaking factor F_Q. Flux map data processed by the INCORE code was used to compare measured and predicted peak pin power in monitored fuel assemblies. Comparisons were made conservatively at points axially mid-way between spacer grids (PDQ does not model the grid depressions or the between grid power peaking) for assemblies of greater than average power. Flux maps from three cycles were included in the data.

The Kolmogorov-Smirnov test (the D test) was used to assess the assumption of normality for the percent difference data. The assumption of normality was found to be acceptable for the

pooled data for each of the three cycles based on the results of the D test. A one-sided upper tolerance limit was defined as:

$$TL = X + (K \times S)$$

where K is the one sided tolerance factor for 95% probability and a 95% confidence level (95/95). X is the mean and S is the standard deviation of the % difference data. VEP-FRD-45A references USNRC Regulatory Guide 1.126, Rev. 1 (March 1978) as a source for values of K based on sample size. The NUF was defined as:

$$NUF = 1 + (TL/100)$$

For example, if the value of TL is 10%, the NUF is 1.10. The NRF is then set to conservatively bound the NUF. A discussion of this methodology may be found in Sections 3.1, 3.2, and 3.3 of VEP-FRD-45A. The statistical approach was only used for the F_Q NRF. As stated in the SER:

> *"Only the total peaking factor NRF is derived from comparisons of predicted and measured power distributions. The NRFs for the first four parameters are derived from analytical engineering arguments"*
>
> *"We find this reliability factor to be acceptable, based on comparisons with the uncertainties which have been obtained with other currently approved design methods."*
>
> *"Sufficient information is presented in the report to permit a knowledgeable person to conclude that the NRFs established by Vepco for the Doppler coefficient, the delayed neutron parameters, and the total peaking factor are conservative and acceptable."*

The SER therefore considers engineering arguments, statistical data from comparisons of measurements and predictions, and consistency with uncertainty factors approved for other codes to be valid methods of assessing the adequacy of reliability factors. The PDQ Two Zone model NUFs were determined based on a similar combination of comparison to measured data, statistical treatment of the comparisons where appropriate, analytical engineering arguments, and comparisons to reliability factors obtained with other approved models. Because VEP-NAF-1 contains comparisons with 31 operating cycles of measured data, there is greater reliance on statistical treatment of the differences than was possible in the previous reports. Dominion concurs with the use of these methods for determining appropriate reliability factors, and believes that the data presented in VEP-NAF-1 is sufficient to support use of the reliability factors indicated.

One issue that arises in VEP-NAF-1 is the treatment of data for which the hypothesis of normality is rejected (based on the D test). The non-parametric method of Sommerville described and referenced in USNRC Regulatory Guide 1.126, Rev. 1 was used for such samples to construct a 95/95 one-sided upper tolerance limit. This method effectively requires sorting of the data by sign and magnitude and choosing the nth value from the sorted list starting from the most non-conservative value (n=1). The value of n is based on the sample size and is applicable for sample sizes of 60 or greater. The Tables below indicate for each NUF the method used to derive the NUF, associated statistics, and any special considerations used.

NUF Derivation Methods

Parameter	Primary NUF technique(s)	Comments
Control Rod Worth – Integral worth, individual banks	Statistical	Statistics use comparisons to measured rod worth data from 31 cycles of startup physics tests. Assessment of impact of reactivity computer bias included. NRF of 1.10 supported with or without accounting for reactivity computer contribution to uncertainty.
Control Rod Worth – Integral worth, all banks combined	Engineering arguments	The cumulative bank uncertainty is bounded by the individual bank uncertainty.
Differential Bank Worth	Engineering arguments	A qualitative assessment of 14 plots of measured and predicted differential rod worth from 11 cycles (startup physics testing) was performed. All plots are included in the report. This is similar to the treatment used in VEP-FRD-24A for the FLAME model.
Critical Boron Concentration	Statistical	Statistics use comparisons to critical boron measurements from startup physics testing as well as post-outage restarts during each cycle. Conclusions are supported qualitatively by HFP boron letdown curves (measured and predicted) from 30 operating cycles included in the report.
Differential Boron Worth	Statistical and Engineering arguments	Statistics use comparisons to boron worth measurements from startup physics testing. Due to a proportionally large contribution from measurement uncertainty, comparison statistics alone do not lead to a physically reasonable NRF. Engineering arguments were used to assess the level of measurement uncertainty and to support a reasonable NRF via indirect evidence (primarily critical boron concentration).
Moderator Temperature Coefficient	Statistical	Statistics use comparisons to isothermal temperature coefficient measurements from startup physics testing. There is a relatively small Doppler component included, but the range of measured ITCs (-14 to +3 pcm/°F) ensures that the comparison is valid for determining MTC uncertainty. Any uncertainty contribution from the Doppler component is included in the statistics.
FΔH	Statistical	Statistics use comparisons to measured FΔH from incore flux maps for assemblies of greater than average relative power.
F_Q	Statistical	Statistics use comparisons to measured F_Q from incore flux maps for assemblies of greater than average relative power.
Doppler Temperature or Power Coefficient	Engineering Arguments	ECP critical boron predictions (effectively an observation of consistency between HFP and HZP critical boron agreement) are mentioned as indirect evidence supporting the NRF determined for previous models (1.10). Arguments in VEP-FRD-45A remain the primary basis for this NRF. Because it was not explicitly treated for the Two Zone model, this NRF is not listed in the report.

NUF Derivation Methods (Continued)

Parameter	Primary NUF technique(s)	Comments
Effective Delayed Neutron Fraction and Prompt Neutron	None	Arguments in VEP-FRD-45A remain the basis for these NRFs. Because they were not explicitly treated for the Two Zone model, these NRFs are not listed in the report.

Additional Information for Statistically Derived NUF Data

Parameter	Number of observations	Mean	Standard Deviation	Normality assumed?	Standard Deviation Multiplier (K)	N^{th} value (n)
Control Rod Worth – Integral worth, individual banks (raw data)	157	1.0%	4.5%	Yes	1.88	N/A
Critical Boron Concentration	54	6.3 ppm	20.0 ppm	Yes	2.05	N/A
Differential Boron Worth (raw data)	30	-0.3%	4.4%	No	N/A	N/A
Isothermal Temperature Coefficient	57	-0.8 pcm/°F	0.96 pcm/°F	No	N/A	1
FΔH (North Anna)	1479	0.1%	1.9%	No	N/A	60
FΔH (Surry data)	1878	0.0%	1.7%	No	N/A	78
F_Q(North Anna)	9046	-2.2%	2.8%	No	N/A	401
F_Q(Surry data)	9372	-2.6%	3.0%	No	N/A	416

Notes:

1) Difference is defined as Measured – Predicted or as (Measured – Predicted)/Measured.
2) The W test (Shapiro and Wilk) for normality was used for the differential boron worth because the sample size was too small for the D test. A physically realistic uncertainty factor could not be developed based on this non-normal small sample, therefore indirect evidence was presented in the Topical Report in support of the DBW NRF.

NRC PDQ QUESTION 5

Please discuss how the measured data used for statistical comparison to the PDQ Two Zone predicted values were obtained. How were uncertainties in the measured data addressed in the statistical analyses?

DOMINION RESPONSE TO QUESTION 5

Measured data is routinely collected as part of plant operations. Sources of measured data for VEP-NAF-1 include startup physics testing, daily critical boron concentration measurements, criticality condition data, and flux maps (from both startup physics testing and monthly peaking factor surveillance). Much of the data is summarized in a Startup Physics Test Report published following each initial core load or refueling and in a Core Performance Report published following the end of each cycle. The Table below indicates the source of each measured value and an indication of the measurement technique involved.

Measured Parameter	Source	Techniques Involved
Control Rod Worth – Integral bank worth	Startup physics testing (HZP)	Dilution (periodic reactivity computer measurements during a controlled boron dilution) and rod swap (swap of the test bank with a reference bank previously measured by dilution).
Control Rod Worth – Differential bank worth	Startup physics testing (HZP)	Dilution.
Critical boron concentration	Startup physics testing (HZP), daily boron measurements (HFP), ECP procedure (used for mid-cycle return to critical; HZP)	RCS samples are measured by chemical titration. Multiple measurements are used during startup physics testing.
Differential Boron Worth	Startup physics testing (HZP)	Derived from measured reference bank worth and the ARO and reference bank inserted critical boron concentrations. Boron concentrations are measured by chemical titration.
Isothermal Temperature Coefficient	Startup physics testing (HZP)	Reactivity computer measurements during controlled temperature change at HZP.
$F\Delta H$, F_Q	In–core flux maps	Flux maps in this report are taken with movable incore detectors and transformed into measured power distributions using the INCORE code. Maps were taken during startup physics testing (typically <5% power, ~30% power, ~70% power, and ~100% power) and monthly throughout the cycle (typically near HFP).

Measurement uncertainty is inherently and conservatively included in the differences between measured and predicted quantities. NUFs and NRFs derived from such comparisons effectively attribute any measurement uncertainty present to model predictive uncertainty. This type of "raw" comparison data supports all NRFs derived in this report, with the exception of the differential boron worth NRF. Only in the case of the differential boron worth NRF is it necessary to address the effects of measurement uncertainty to support the NRF.

Attachment 3

Responses to NRC Questions on NOMAD

Virginia Electric and Power Company
(Dominion)
North Anna and Surry Power Stations

NOMAD Code Model Review, Topical Report VEP-NFE-1-A. Supplement 1, "VEPCO NOMAD Code and Model," VEPCO Submittal dated Novcemver 13, 1996

NRC NOMAD QUESTION 1

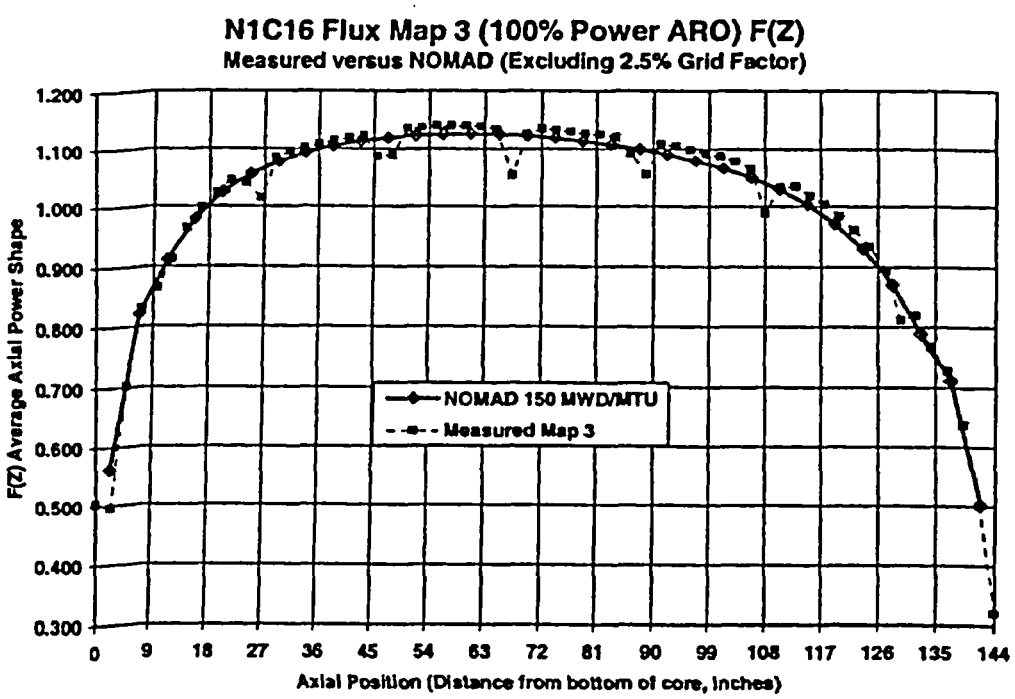
By letter dated December 2, 2002, VEPCO stated that the accuracy of the NOMAD model is verified each cycle during startup physics testing and during routine core follow. Please provide representative results from a recent refueling outage (comparisons between the startup physics test data and the NOMAD predictions) that demonstrate the accuracy of this model.

DOMINION RESPONSE TO QUESTION 1

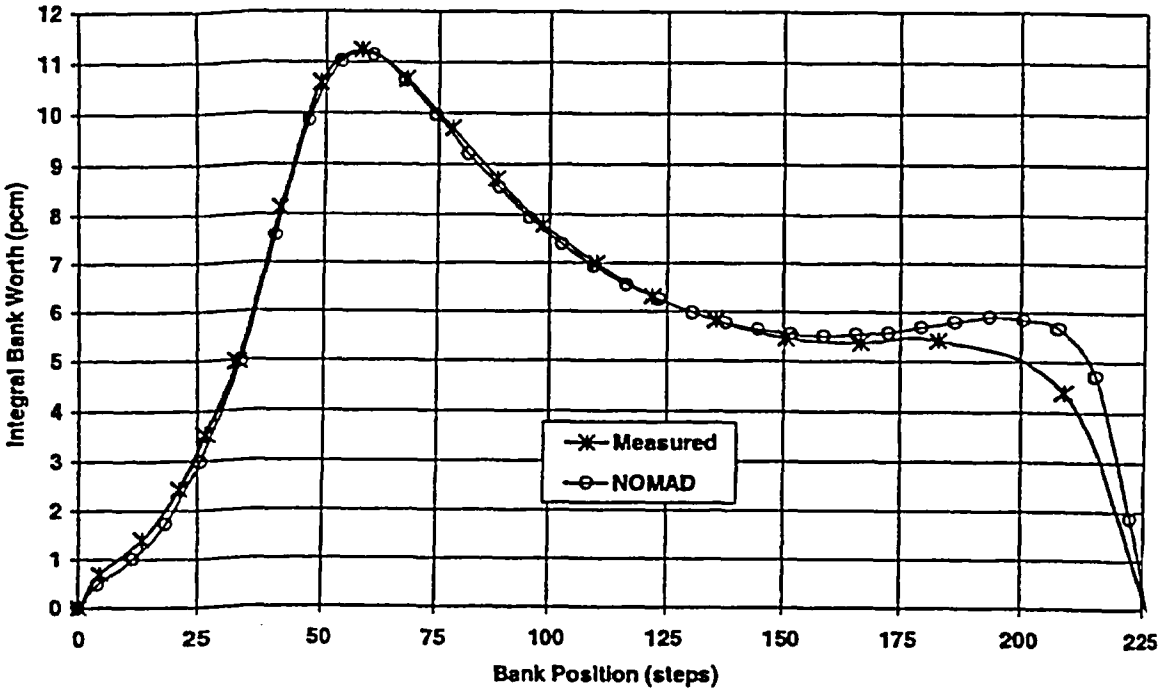
Verification of NOMAD accuracy comes primarily by extension through comparison to PDQ Two Zone model (Topical Report VEP-NAF-1) predictions during the NOMAD model setup process (see also the response to questions 3 and 7). The NOMAD model setup procedure provides specific power distribution and reactivity acceptance criteria for these comparisons that must be met. There are, however, a few direct comparisons to startup physics test data that can be made. The following results are from the N1C16 startup physics tests in October 2001.

N1C16 STARTUP PHYSICS TESTING RESULTS (October, 2001)

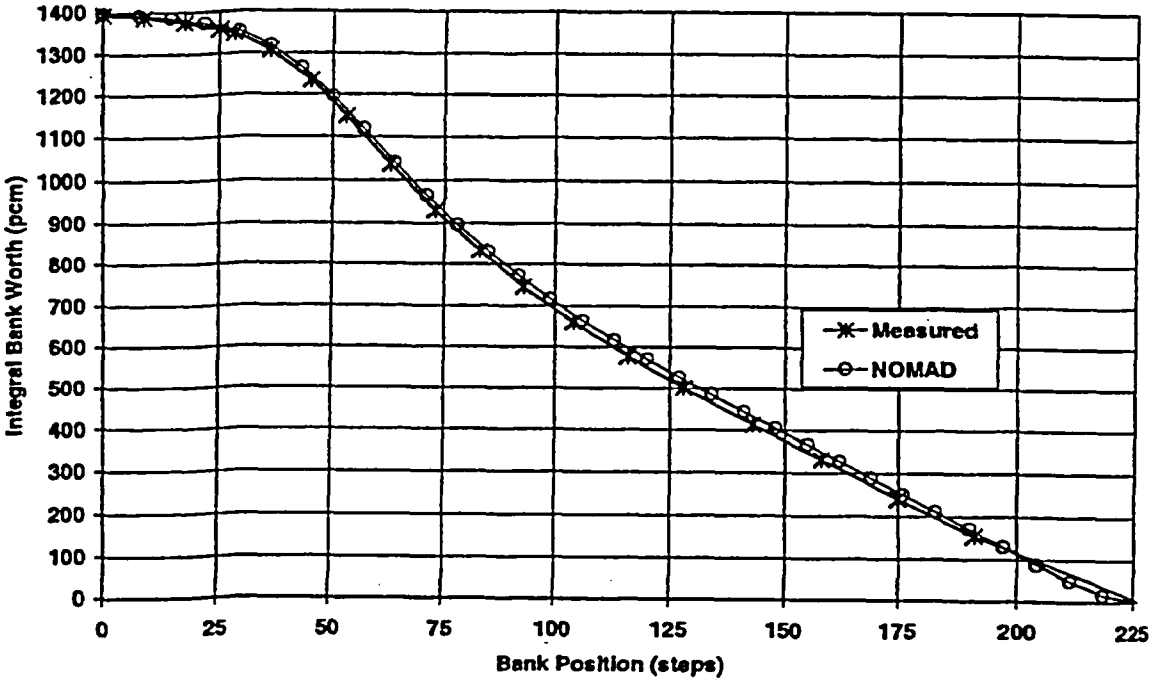
Parameter	Measured	Predicted	Difference	Nuclear Reliability Factor
Critical Boron Concentration (HFP, ARO, EQ XE) ppm	1405	1429	24	±50 ppm
HFP ARO EQ XE Axial Offset	-2.5	-3.0	-0.5%	N/A



N1C16 HZP BOC B-Bank Differential Worth



N1C16 HZP BOC B-Bank Integral Worth



NRC NOMAD QUESTION 2

There do not appear to be any limitations or restrictions associated with the use of NOMAD as described in this submittal. Please justify that NOMAD is applicable over all ranges of operation expected for North Anna and Surry.

DOMINION RESPONSE TO QUESTION 2

NOMAD is by design constrained by the limitations of the PDQ Two Zone Model. All cycle-dependent NOMAD input data comes from the PDQ Two Zone model, and the quality control process used to verify the NOMAD model for each core involves comparison to PDQ Two Zone model predictions. Therefore NOMAD should have the same restrictions and limitations as listed for the PDQ Two Zone model. The PDQ Two Zone model is restricted according to the following characteristics:

1) Geometry
   a) Square pitch fuel (cylindrical fuel pellets and rods)
   b) 15x15 or 17x17 design
   c) 5x5 mesh blocks per assembly (x-y)
   d) 26 axial nodes (22 in the fuel region)
   e) ¼ core or full core representation
2) Fuel Material
   a) Low enriched UO_2 (4.6 w/o U_{235} or less)
      i) Cores with fuel up to 4.45 w/o have been successfully modeled to date
      ii) Cross section behavior (enrichment trends and fidelity to CELL2) has been checked up to 4.6 w/o U_{235} for burnups up to 76 GWD/T.
   b) Fuel pin burnup of approximately 70 GWD/T has been achieved in PDQ Two Zone designed cores as part of a high burnup demonstration program.
3) Burnable poisons
   a) Discrete rods inserted into fuel assembly guide thimbles
      i) Both annular borosilicate glass and solid B4C in alumina designs have been well predicted throughout many cycles of operation
      ii) Both SS304 and zirconium based cladding has been used
   b) Modeling flexibility has been demonstrated for BP configuration (number of fingers, boron enrichment, poison length, and poison stack axial alignment)
4) Control rods
   a) Ag-In-Cd rods with stainless steel clad (extensive validation and experience)
   b) Hf metal rods in zirconium based clad have been used for vessel fluence reduction in Surry Unit 1
5) Fuel assembly
   a) Modeling flexibility has been demonstrated for Inconel and zirconium based grids of various designs and sizes

There are no current plans for fuel design, core design, or operating strategy changes that would exceed the design characteristics outlined above. There are fuel products in use in the industry which would be technically possible but impractical to model in the PDQ Two Zone and NOMAD models (such as fuel with integral poisons). No further development is planned for PDQ and NOMAD. In addition, the simplicity of the NOMAD control rod cross section model requires normalization for low temperature

use (significantly below 547 °F). This precaution is listed in the NOMAD Code Manual. There are no current uses for NOMAD at low temperatures.

NRC NOMAD QUESTION 3

Please discuss the user-defined tolerances used in the Radial Buckling Coefficient model, including how they are calculated and used in the model. Also discuss the process in place that ensures that correct values are calculated and entered into the model by the user.

DOMINION RESPONSE TO QUESTION 3

The great majority of radial buckling effects are automatically captured (without any user intervention) via the data handling routines that collapse the 3-D PDQ Two Zone model data into 1-D NOMAD data. Design procedures indicate that reactivity agreement within 250 pcm of PDQ (HZP and HFP from BOC-EOC) is normally achieved using the "raw" (pre-buckling search) NOMAD model. Axial offset agreement within 2% is also typical. The buckling search can therefore be thought of as the means of capturing second and third order effects.

User defined tolerances control the rate and degree of convergence of the radial buckling search. Convergence is determined automatically in NOMAD by comparison of the NOMAD eigenvalue, peak nodal power, and individual node powers to the corresponding PDQ Two Zone values. Design procedures specify a standard set of convergence tolerances for use in the NOMAD model setup and review. Design procedures also require independent review of each NOMAD model setup prior to use in the core design process.

The values of the standard tolerance set are based on experience with previous NOMAD model setups (in particular the models which produced the benchmark data in Supplement 1 to VEP-NFE-1A) and represent the level of convergence normally achievable for a correctly constructed NOMAD model. These values were set at a level that would assure convergence consistent with Supplement 1 models, that would assure convergence as tight as reasonably achievable, but that could result in occasional minor non-convergence events.

If convergence is not achieved for a particular case, a warning message is printed that prompts a review of the model setup. One option available to the user is to change the rate of convergence (by changing the relaxation parameters) to reduce the chance of overshoot or undershoot. Cases of non-convergence are evaluated according to which parameter failed to converge and the degree of non-convergence involved. A large violation of a convergence tolerance is a good indication of a model error. Based on prior experience, non-convergence incidents are rare and of very small magnitude. Documentation for the most recent NOMAD model setups for North Anna and Surry indicates that convergence was achieved within the standard tolerances using the standard relaxation parameters.

There are other user-adjustable buckling parameters that are provided to accommodate the fact that the automated buckling search is only performed at HFP. Parameters are provided to improve axial offset and reactivity agreement between NOMAD and PDQ for lower power levels. In essence, these factors control the portion of the buckling search adjustments that are retained as power is reduced. Once again, a standard set of values is provided for use in the design procedures based on prior model setup experience. The adequacy of the standard values is verified directly by comparison of NOMAD and PDQ results at low power during the model setup process. A review of the history of NOMAD model

setups revealed only one change to the standard values that has been implemented in order to meet the model acceptance criteria. Guidance for achieving an acceptable NOMAD model, including the user actions described above are incorporated in design procedures.

NRC NOMAD QUESTION 4

The xenon model in NOMAD allows a user-supplied multiplier to be applied to the xenon or iodine production terms. Please discuss the purpose of this multiplier and how the value is determined. Also discuss the process in place that ensures that correct values are calculated and entered into the model by the user.

DOMINION RESPONSE TO QUESTION 4

Iodine and xenon production multipliers were included in the NOMAD model for investigative purposes and possible future applications, but were never incorporated into the normal model design process. There are no current uses for these multipliers. Design procedures specify a value of 1.0 for these values. The xenon model requires very little user intervention and is verified by direct comparison to PDQ xenon concentration and xenon offset. Design procedures require independent review of each NOMAD model setup prior to use in the core design process.

NRC NOMAD QUESTION 5

The Control Rod Model requires several user input constants or multipliers. Please discuss the purpose of these user inputs, and the methods used to determine their values. Also discuss the process in place that ensures that correct values are calculated and entered into the model by the user.

DOMINION RESPONSE TO QUESTION 5

The Control Rod Model is very similar to the Radial Buckling Coefficient model in that a large majority of the NOMAD control rod information is obtained automatically from PDQ via data processing codes without any user-adjustable input. For the remaining effects, user input constants are provided in each of the following four categories:

A) Cusping corrections
B) Second order temperature or density effects
C) Geometry data (physical control rod overlap)
D) Worth normalization

The control rod cusping model accounts for the approximation made for control rod insertions in which the rodded/unrodded axial boundary occurs between nodal boundaries (partial insertions). For partial insertions NOMAD volume weights the control rod effects and applies the weighted values over the entire node. Without cusping corrections, the differential control rod worth shape exhibits a sawtooth behavior as the control rods are inserted in small steps. The cusping model corrects for this effect using two alternate approximations. The first alternative recognizes that the degree of cusping is a function of node size and insertion fraction. The second recognizes that the degree of cusping is a function of the local power gradient and insertion fraction. User input allows for the use and scaling of either alternative. Although cusping is not a significant practical problem due to the relatively small node size

in NOMAD, standard input factors determined during the development of NOMAD were shown to significantly reduce the magnitude of cusping. These factors have not been changed since their development because neither the control rod type nor the NOMAD mesh structure have changed. Design procedures specify use of the recommended values for NOMAD model setup.

In the HZP-HFP operating range, control rod cross sections do not vary significantly. The small variation that exists is approximated by linear coefficients of moderator temperature or density. Based on PDQ Two Zone model control rod cross section data, a standard set of coefficients were developed during NOMAD development. These coefficients have not been changed because the control rod design has not changed. Design procedures specify use of the recommended values for NOMAD model setup. In the event of a control rod design change, detailed calculations are referenced in the design procedure that provide the techniques used to calculate these parameters.

User input is provided for the control rod ARO position and the normal operation control rod overlap. This input is based on actual core operating limits and specifications set each cycle.

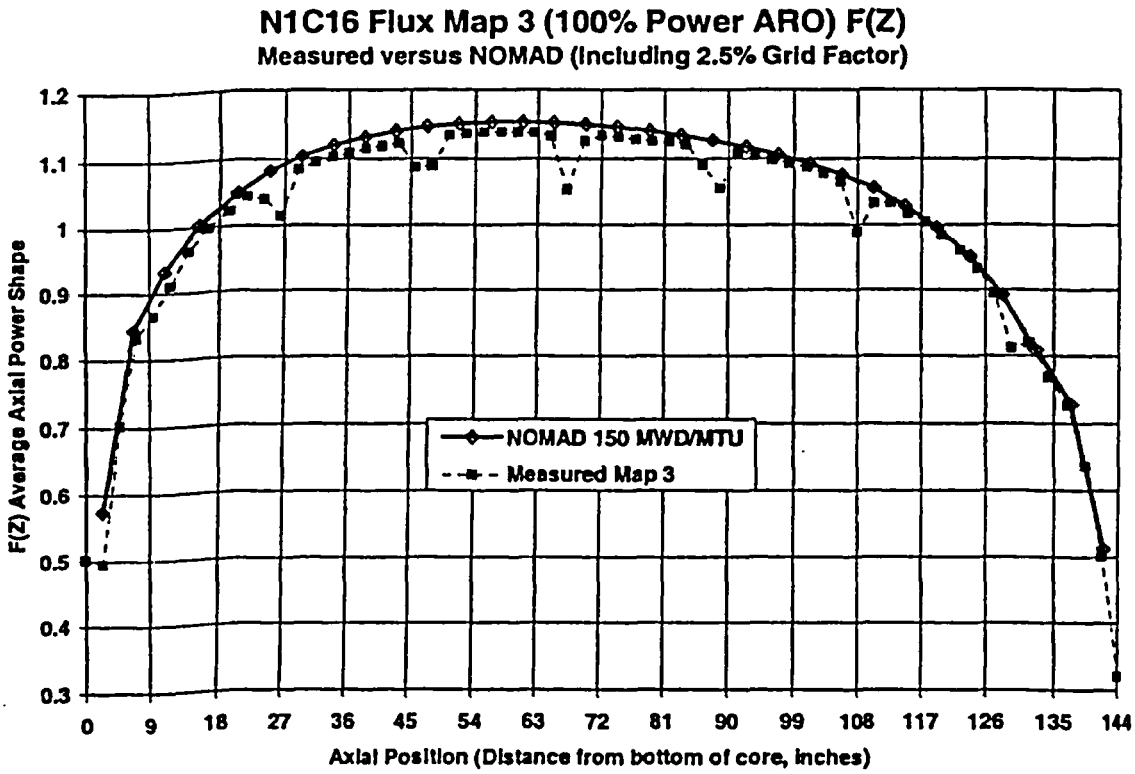
The final element of the control rod model is the ability to normalize bank worth to the PDQ Two Zone value. Although NOMAD was designed to produce acceptable control rod worth results without normalizing to PDQ, normalization is performed routinely for many design calculations to eliminate any difference between PDQ and NOMAD. In this way, calculations involving data from both models is completely consistent. In addition, normalization permits the modeling of non-physical part-length rods that are used to conservatively skew the axial power shape for certain types of calculation. Design procedures provide specific normalization instructions for each type of calculation. Design procedures also require independent review of each NOMAD model setup prior to use in the core design process.

NRC NOMAD QUESTION 6

In the $F_Q(z)$ x relative power calculations, a correction factor for grids is applied. Please discuss the method used to calculate these correction factors. Discuss how the correction factors change as the location of interest moves away from a grid location and provide typical values for these correction factors as a function of axial location.

DOMINION RESPONSE TO QUESTION 6

The grid factor is a constant multiplier of 1.025 that is conservatively applied to all axial locations rather than just between grids. The magnitude was retained from previous models but can be justified both qualitatively and quantitatively. A qualitative example is the power shape plot below. This is the same plot presented in the answer to NOMAD question 1, except that the grid factor has been applied. The predicted power shape effectively bounds the measured shape in this example, demonstrating that for this core and at this time in life, the grid factor is conservative.



Quantitatively, the grid factor can be determined from the mean of the Fz data presented in Table 3.0.3 of VEP-NFE-1A Supplement 1. Both the measured and predicted Fz shapes are normalized to an average value of 1.0 by definition. The Fz mean in Table 3.0.3 is the average difference between NOMAD and measured Fz at positions mid-way between grids for flux map data acquired during five different cycles. These are the axial positions where the NOMAD model exhibits the greatest degree of under-prediction due to the effect of the grids on the measured power shape. The mean difference of –2.4% is consistent with the magnitude of the NOMAD grid factor (1.025 or 2.5%).

NRC NOMAD QUESTION 7

Regarding the method of qualifying the NOMAD model, please address why data from only a few select operating cycles for North Anna, Unit 1, and Surry, Unit 2, were chosen for benchmarking purposes. Are the number of data points used for the various verifications adequate for a statistically significant decision?

DOMINION RESPONSE TO QUESTION 7

Unlike the PDQ Two Zone model, NOMAD is not developed sequentially by building on the depletion from the previous cycle. NOMAD is set up directly from the PDQ Two Zone model. Consequently, there was not a NOMAD model available for each historical cycle as a result of the development process. The primary use of NOMAD is for FAC (Final Acceptance Criteria) or RPDC (Relaxed Power Distribution Control) modeling, which involves the use of load follow transient axial power shapes. With this in mind, the cycles presented were chosen based on three criteria:

1) Availability of measured operational transient data.
2) Representation of the full range of cycle designs for Surry and North Anna.
3) Quantity of data similar to or greater than presented for the approved NOMAD model documented in VEP-NFE-1A.

The following Table summarizes the cycles used to support conclusions in VEP-NFE-1A and in Supplement 1.

Parameter	VEP-NFE-1A Cycles	Supplement 1 Cycles
Startup Physics Measurements	N1C2, N1C3, N1C4, N2C2, S1C6, S1C7	N1C3, N1C6, N1C9, S2C2, S2C11, S2C13
Operational Transients	N1C2, N1C3	N1C3, N1C6, N1C9, S2C2, S2C11, N1C11
Flux Maps (Fz and F_Q comparisons)	N/A*	N1C3, N1C6, N1C11, S2C2, S2C13
Estimated Critical Position (ECP; Mid-cycle HZP criticality measurements)	N/A	N1C9, S2C11, S2C13
FAC Analysis	N2C2, N1C4 (Verbal description of comparison to vendor model results)	S2C13 (Graphical comparison to approved NOMAD model F_Q envelope)
RPDC N(Z)	N/A (Pre-RPDC)	N1C11 (Graphical comparison to approved NOMAD model N(Z) function)

* BOC Fz plots were provided for 5 cycles (N1C2, N1C3, N1C4, N2C2, and S1C6)

As shown in the Table, Supplement 1 provides more NOMAD verification information than did the approved NOMAD Topical Report VEP-NFE-1A. There is no direct development of reliability factors in VEP-NFE-1A and no discussion of specific NOMAD reliability factors in the SER. The NOMAD SER cites comparisons to measurements, comparisons to higher order calculations (FLAME and PDQ), and the NOMAD normalization process as reasons for the approval. In particular, the normalization of NOMAD to FLAME is mentioned as a means of ensuring agreement with higher order calculations. NOMAD therefore was implicitly considered to share reliability factors with the models to which it is normalized.

The enhanced NOMAD model described in Supplement 1 can be supported based on this normalization argument and based on statistical comparisons to measured data. Design procedures specify these acceptance guidelines (comparison to PDQ Two Zone model predictions) to be met to support the conclusion that a NOMAD model has been set up properly:

1) Peak nodal power within 0.5% (HFP depletion)
2) All nodal powers within 2.5% (HFP depletion)
3) Equilibrium Xenon concentration within 0.5% (BOC and EOC)
4) Xenon offset within 0.2%
5) Axial offset within 2% (BOC-EOC, HZP and HFP)
6) Reactivity within 10 pcm (BOC-EOC, HFP)
7) Total power defect within 100 pcm (BOC, MOC, EOC)
8) HFP fuel temperature within 10 °R (BOC and EOC)
9) Calculation specific rod worth normalization

Because of these normalization requirements and the designed-in close connection between NOMAD and the 3D PDQ Two Zone model, the PDQ reliability factors (based on far more data) can be extended to the NOMAD model. This is analogous to the extension of FLAME reliability factors to the approved NOMAD version.

Although the number of observations in the measurement comparison data presented in Supplement 1 is not in all cases sufficient for a statistics-based determination of NOMAD uncertainty factors, the data presented is sufficient to demonstrate consistency with PDQ Two Zone Model comparisons. The conclusion in Supplement 1 that *"comparison of NOMAD uncertainty factors to Nuclear Reliability Factors.....verify.... the applicability of the NRF's for NOMAD calculations"* is not clearly qualified to indicate that the only parameters for which NOMAD uncertainty factors were directly statistically developed in Supplement 1 are Fz and F_Q. For other parameters, a better characterization is that comparison of NOMAD *results* to Nuclear Reliability Factors verify the accuracy of the NOMAD model and the applicability of the NRF's for NOMAD calculations.

For Fz and F_Q, a total of 134 observations were available for both, and the derived F_Q uncertainty factor is nearly identical to that calculated for the PDQ model (6.9% versus PDQ values of 6.7% for North Anna and 7.2% for Surry). The F_Q NRF of 1.075 conservatively bounds all these values.

The Table below compares PDQ Two Zone model and NOMAD statistics (differences between model predictions and measurements) for other parameters. PDQ statistics are contained in Topical Report VEP-NAF-1. Note that for critical boron and ITC, the sign of the NOMAD mean has been changed to reflect different definitions used in the respective reports and allow appropriate comparison to PDQ results. The range of NOMAD differences is bounded by the range of PDQ model differences, and the

NOMAD standard deviations are similar to or smaller than the corresponding PDQ standard deviations. The means show more variation, but are reasonable considering the sample sizes and the relative magnitude of the standard deviations. The comparison supports a conclusion that the PDQ Two Zone model reliability factors are appropriate for use with the closely related NOMAD model. Note that only the un-normalized (raw) rod worth results were presented in Supplement 1. The Table below also includes the normalized rod worth results (see the response to NOMAD question 5).

Comparison of NOMAD and PDQ Statistical Data

Parameter	Model	Number of observations	Mean	Standard Deviation	Maximum	Minimum
Control Rod Worth – Rod Swap	PDQ	95	1.8%	4.2%	11.5%	-11.3%
	NOMAD (raw)	25	2.99%	5.1%	11.4%	-7.8%
	NOMAD (normalized)	25	-0.1%	4.5%	7.6%	-8.1%
Control Rod Worth – Dilution	PDQ	62	-0.2%	4.8%	10.7%	-9.9%
	NOMAD (raw)	7	-0.6%	4.4%	7.1%	-6.7%
	NOMAD (normalized)	7	0.8%	4.1%	7.2%	-3.5%
Boron Worth	PDQ	30	-0.3	4.4%	7.4%	-6.1%
	NOMAD	6	-2.2%	2.3%	1.4%	-4.1%
HZP Critical Boron Concentration	PDQ	54	6 ppm	20 ppm	58 ppm	-30 ppm
	NOMAD	13	21 ppm	17 ppm	36 ppm	-17 ppm
HZP ITC (pcm/°F)	PDQ	57	-0.8	1.0	2.6	-2.9
	NOMAD	9	0.2	0.6	1.5	-0.5

NRC NOMAD QUESTION 8

Please discuss the methodology used to calculate each of the NOMAD NUF and indicate when NRC approval was obtained.

DOMINION RESPONSE TO QUESTION 8

As indicated in the response to NOMAD question 7, the only parameters for which NOMAD uncertainty factors were directly statistically developed in Supplement 1 are Fz and F_Q. The methodology is described briefly in Supplement 1, Section 3.1.4.1. This methodology is ultimately rooted in VEP-FRD-45A (SER date August 5, 1982) and is the same as described for the PDQ Two Zone model F_Q NRF. The only difference is that only the peak F_Q at each axial level can be used for the 1-D NOMAD comparisons rather than individual assembly F_Q's used for the 3-D PDQ model comparisons. A full discussion of the comparison and statistical methodology is provided in the response to PDQ question 4.

For all other parameters, uncertainty factors derived for other models were shown to be reasonable for use with NOMAD. VEP-FRD-45A summarizes the reliability factors derived for the PDQ Discrete model (VEP-FRD-19A, SER date May 18, 1981), the PDQ One Zone model (VEP-FRD-20A, SER date May 20, 1981), and the FLAME model (VEP-FRD-24A, SER date May 13, 1981). These same reliability factors were re-validated for the PDQ Two Zone model in VEP-NAF-1. Most of the approved reliability factors summarized in VEP-FRD-45A were approved not based on statistics, but on a combination of engineering arguments and consistency with uncertainty factors approved for other models (see the response to PDQ question 4). This is the approach taken in Supplement 1, except that more statistical data based on comparisons to measured data have been provided than in the approved NOMAD Topical. Dominion concurs with the use of these methods for determining appropriate reliability factors, and believes that the data presented in Supplement 1 is sufficient to support use of the reliability factors indicated.

NRC NOMAD QUESTION 9

Please discuss how the measured data used for statistical comparison to the NOMAD predicted values were obtained. How were uncertainties in the measured data addressed in the statistical analyses?

DOMINION RESPONSE TO QUESTION 9

Please refer to the response to PDQ question 5. Plant transient data (not used for statistical comparisons) was obtained either from plant computer records (delta-I based on ex-core detectors, calorimetric power based on the plant computer heat balance calculations, and control rod position indications) or from routine periodic measurements (critical boron concentration). No corrections for measurement bias or uncertainty were applied to the plant transient data.