ATTACHMENT 3

VEP-FRD-41-NP-A, REVISION 0, MINOR REVISION 3, VEPCO REACTOR SYSTEM TRANSIENT ANALYSIS USING THE RETRAN COMPUTUER CODE (NON-PROPRIETARY)

MILLSTONE POWER STATION UNIT 3
DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

Dominion Energy®

VEP-FRD-41-NP-A, Revision 0, Minor Revision 3

VEPCO Reactor System Transient Analysis Using the RETRAN Computer Code

January 2019

VEP-FRD-41-NP-A, Revision 0, Minor Revision 3

VEPCO REACTOR SYSTEM TRANSIENT ANALYSIS USING THE RETRAN COMPUTER CODE

By

NUCLEAR ENGINEERING & FUEL STAFF
DOMINION ENERGY
RICHMOND, VIRGINIA

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Prepared By:	Emily J. Tomlinson E. J. Tomlinson	Nuclear Safety Analysis Design
Reviewed By:	T. N. Ho	Nuclear Safety Analysis Design
Recommended For Approval:	C. D. Clemens	Supervisor – Nuclear Safety Analysis Design
Approved:	W. M. Adams	Director – Nuclear Engineering and Fuel

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APPENDICES

CLASSIFICATION/DISCLAIMER

ABSTRACT

This Topical Report describes Virginia Electric and Power Company's (Dominion Energy's) Reactor System Transient Analysis models for use with the RETRAN Computer Code. These models have been qualified for UFSAR Chapter 14 and 15 non-LOCA transient analyses to support Surry Units 1 and 2, North Anna Units 1 and 2, and Millstone Unit 3. 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 computer code are addressed.

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1 INTRODUCTION

Dominion Energy was formerly known as Dominion, Virginia Power, or (prior to January 15, 1985) as Virginia Electric and Power Company (VEPCO) and the topicals referenced were submitted using the former names in their titles. The current report introduces the Dominion Energy designation but retains the prior nomenclature for citation of historical references.

This report is an update of VEP-FRD-41, VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code (**Appendix 1**). This report has been designated VEP-FRD-41-NP-A, Revision 0, Minor Revision 3. The purpose of this update is to reflect application of the topical report to Millstone Power Station Unit 3 (MPS3). The NRC granted approval of the License Amendment Request (LAR, Reference 1.0-1) which justified this application via the Safety Evaluation Report (SER) in Reference 1.0-2. The SER is included in **Appendix 11**. Minor Revision 2 of the report supplemented the information provided in Minor Revision 1 with discussion concerning the transition from the RETRAN-02 to the RETRAN-3D version of the RETRAN code. Minor Revision 1 consolidated 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 1.0-3). 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-3D 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 has been used extensively by the U.S. and international safety analysis community for decades. Details of the RETRAN-3D theory, numerics, qualification and application guidelines may be found in Reference 1.0-4. 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 1.0-6. This submittal provided qualification comparisons 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 of 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-42-A (Reference 1.0-7). Relevant correspondence documenting this review process is shown in Table 2.1. The NRC documented their review and approval of these model upgrades in Reference 1.0-8, which is included for reference as **Appendix 6.** As part of the NRC review of VEP-FRD-42-A, the NRC generated several RAIs. Responses to these RAIs are included for reference as **Appendix 7.** This is discussed further in Section 2.

In 2001, the NRC approved Topical Report NP-7450 Rev. 4, RETRAN-3D – A program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems (Reference 1.0-4). As part of the Safety Evaluation Report (SER) (Reference 1.0-5) included as **Appendix 8**, the NRC stated that organizations with NRC-approved RETRAN-02 methodologies can use the RETRAN-3D code in the RETRAN-02 mode without NRC approval, provided that none of the new RETRAN-3D models restricted by the SER are used. Dominion transitioned to RETRAN-3D based models in accordance with the conditions and limitations of the generic RETRAN-3D SER. **Appendix 9** presents a qualification of the RETRAN-3D models by comparison with the RETRAN-02 models for a representative set of transients.

In 2015, Dominion Energy submitted a LAR requesting application of Dominion Energy Safety Analysis and Core Design Methodologies at MPS3 which included the transient analysis methodology using RETRAN. Validation of the Dominion Energy RETRAN methodology and MPS3 model involved comparing Dominion Energy calculations to the MPS3 analysis of record for selected transients. The benchmark analysis submitted to the NRC for review is presented in **Appendix 10**. Relevant correspondence documenting the NRC review process is shown in Table 2.1. The analysis demonstrates acceptable agreement with the Westinghouse FSAR analysis performed for the MPS3 Stretch Power Uprate. The MPS3 Methods Transition was approved by the NRC in 2016 in Reference 1.0-2 and is included as **Appendix 11**.

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

Section 4 presents an overview of Dominion's RETRAN-3D three-loop model for Surry and North Anna and four-loop model for MPS3. Separate models are maintained for the three plants to reflect differences in plant design and nodalization, but the modeling approach and philosophy are the same. Modeling nomenclature, configuration, nodalization, and RETRAN code option selections are also 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
  - Main steam 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.

A list of abbreviations used throughout the report is provided in **Section 8**.

REFERENCES FOR SECTION 1.0

1.0-1 Letter from Mark D. Sartain (Dominion) to USNRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," May 8, 2015, (Serial No. 15-159).

1.0-2 Letter from Richard V. Guzman (USNRC) to D. A. Heacock (Dominion), "Millstone Power Station, Unit No. 3 – Issuance of Amendment Adopting Dominion Core Design and Safety Analysis Methods and Addressing the Issues Identified in Three Westinghouse Communication Documents (CAC No. MF6251)," July 28, 2016 (Serial No. 16-317).

1.0-3 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.

1.0-4 EPRI NP-7450-CCM-A, RETRAN-3D, - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Rev. 9, March 2014.

1.0-5 Letter from S. A. Richards (USNRC) to G. L. 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.

1.0-6 Letter from Virginia Electric and Power Company to USNRC, "Supplemental Information on the RETRAN NSSS Model," Serial No. 93-505, August 10, 1993.

1.0-7 VEP-FRD-42-A, Revision 2, Minor Revision 2, "Reload Nuclear Design Methodology", October 2017.

1.0-8 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 (TAC NOS. MB3141, MB3142, MB3151, AND MB3152)," 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 all model upgrades through June 2003 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.

The transition to RETRAN-3D requires upgrades to the formatting and input structure of some parts of Dominion's models. No changes to the modeling approach, nodalization, and philosophy previously approved by the NRC have been conducted as part of this transition. Changes to the calculational modeling techniques intrinsic to the RETRAN code as part of RETRAN-3D is documented in Section 3.1.3. The transition to RETRAN-3D is implemented via the provisions of 10 CFR 50.59 and USNRC Generic Letter 83-11 Supplement 1 (Reference 2-1).

The Dominion Energy transient analysis methodology using RETRAN was extended to MPS3 in 2016. The Dominion Energy MPS3 RETRAN base model contains alterations in noding with respect to the modeling that is documented in VEP-FRD-41-P-A for the North Anna and Surry plants. These changes are outlined in Section 2.4.3. The nodalization changes were communicated to the NRC in the MPS3 Methods Transition LAR (Reference 2.1-15). Additional changes are made to reflect the MPS3 plant design and configuration. The application of VEP-FRD-41 to MPS3 was approved by the NRC in Reference 2.1-12.

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

2.1.1 Updates through June 2003

The model updates and qualifications summarized in Section 2.4.1, **Model Upgrades Through June 2003** 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 2.4.1 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.

2.1.2 Updates from June 2003 Through Issuance of Minor Revision 2

The model updates and qualifications summarized in Section 2.4.2, **Model Upgrades from June 2003 Through Issuance of Minor Revision 2** and described in Section 5, **System Component Model Descriptions**, have been approved through the processes described in Section 2.2.1. Specifically, the transition from RETRAN-02 to RETRAN-3D has been performed in accordance with limitation and condition of use 40 provided in the SER (Reference 2.1-11). The statement is provided below for completeness.

Organizations with NRC-approved RETRAN-02 methodologies can use the RETRAN3D code in the RETRAN-02 mode without additional NRC approval, provided that none of the new RETRAN-3D models listed in the definition are used. Organizations with NRC-approved RETRAN-02 methodologies must obtain NRC approval prior to applying any of the new RETRAN-3D models listed above for UFSAR Chapter 15 licensing basis applications. Organizations without NRC-approved RETRAN-02 methodologies must obtain NRC approval for such methodologies or a specific application before applying the RETRAN-02 code or the RETRAN-3D code for UFSAR Chapter 15 licensing basis applications. Generic Letter 83-11 provides additional guidance in this area. Licensees who specifically reference RETRAN-02 in their technical specifications will have to request a Technical Specification change to use RETRAN-3D.

2.1.3 Updates Supporting Topical Report Application to Millstone Power Station Unit 3 (Minor Revision 3)

The model updates and qualifications summarized in Section 2.4.3, **Model Upgrades Supporting Application to Millstone Power Station Unit 3 (Minor Revision 3)** and Section 6.2, Item 4 **Benchmarks to Alternate Code Calculations**, have been reviewed by the USNRC as part of the approval to adopt Dominion Energy Core Design and Safety Analysis Methods at MPS3 (Reference 2.1-12). The NRC's statement granting approval of VEP-FRD-41 application to MPS3 is provided below for completeness. References excerpted from the SER have been renumbered and are also provided below.

> *Based on the discussion of the benchmark analysis in Subsection 3.1.4.2.1 through Subsection 3.1.4.2.7 [of Reference 2.1-12], the NRC finds that: (1) the Dominion MPS3 RETRAN benchmarking analysis has included appropriate non-LOCA cases discussed in MPS3 FSAR; (2) the Dominion MPS3 RETRAN model compares reasonably well with the vendor RETRAN model in predicting the trend of the RCS response for the selected non-LOCA cases; (3) the differences in the magnitude of the RCS response can be explainable based on differences in nodal schemes, inputs, or modeling assumptions, and; (4) the use of the Dominion RETRAN method is within the NRC-accepted conditions. Therefore, the NRC staff determines that the RETRAN methodology, as discussed in VEP-FRD-41, Rev. 02, References 2.1-13, 2.1-14, [Attachment 5 of] 2.1-15 and Section 3.4 of [Attachment 4,] Reference 2.1-15, is applicable to MPS3.*

Table 2.1 includes a summary of relevant licensing correspondence with the USNRC regarding application of VEP-FRD-41 to MPS3.

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.

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.

2.1-11 Letter from S. A. Richards (USNRC) to G. L. 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.

2.1-12 Letter from Richard V. Guzman (USNRC) to D. A. Heacock (Dominion), "Millstone Power Station, Unit No. 3 – Issuance of Amendment Adopting Dominion Core Design and Safety Analysis Methods and Addressing the Issues Identified in Three Westinghouse Communication Documents (CAC No. MF6251)," July 28, 2016 (Serial No. 16-317).

2.1-13 Letter from Mark D. Sartain (Dominion) to USNRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," February 25, 2016, (Serial No 16-011A).

2.1-14 Letter from, Daniel G. Stoddard (Dominion) to USNRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," March 29, 2016, (Serial No. 16-011B).

2.1-15 Letter from Mark D. Sartain (Dominion) to USNRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," May 8, 2015, (Serial No. 15-159).

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]**
5/8/2015	Letter from Mark D. Sartain (Dominion) to USNRC, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03" (Serial No. 15-159).	Submits RETRAN benchmark information for application of VEP-FRD-41-P-A to MPS3, consisting of: • RETRAN model description • Comparison of Dominion model key characteristics to FSAR model • Analysis results and comparison for five (5) FSAR transients (MSLB, LOL/TT, LONF, LOCROT, and RWAP)

TABLE 2.1 (CONTINUED)
VEPCO / Dominion RETRAN Model Correspondence

Date	Document Title	Contents
1/8/2016	Letter from Richard V. Guzman (USNRC) to David A. Heacock (Dominion), "Millstone Power Station, Unit 3 – Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods (CAC No. MF62514)".	RAI request on the 5/8/2015 MPS3 Methods Transition submittal.
1/28/2016	Letter from Mark D. Sartain (Dominion) to USNRC, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," (Serial No. 16-011).	Responds to 1/8/2016 RAI. Contains the following regarding VEP-FRD-41: • Details the MSLB split vessel nodal scheme
2/25/2016	Letter from Mark D. Sartain (Dominion) to USNRC, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," (Serial No. 16-011A).	Responds to 1/8/2016 RAI and contains the following regarding VEP-FRD-41: • Updates the 5/8/2015 RETRAN benchmarking to address a discrepancy in the pressurizer heat shell conductor • Provides additional benchmarking for the FLB event • Answers questions on the LOL, LOCROT, LONF, and RWAP benchmarking
3/29/2016	Letter from Daniel G. Stoddard (Dominion) to USNRC, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," (Serial No. 16-011B).	Responds to 1/8/2016 RAI and. Contains the following regarding VEP-FRD-41: • Provides additional benchmarking for the SGTR event [**See Appendix 10** containing revised RETRAN benchmarking for 7 events]

TABLE 2.1 (CONTINUED)
VEPCO / Dominion RETRAN Model Correspondence

Date	Document Title	Contents
7/28/2016	Letter from Richard V. Guzman (USNRC) to D. A. Heacock (Dominion), "Millstone Power Station, Unit No. 3 – Issuance of Amendment Adopting Dominion Core Design and Safety Analysis Methods and Addressing the Issues Identified in Three Westinghouse Communication Documents (CAC No. MF6251)," (Serial No. 16-317).	Approves usage of RETRAN methodology for transient analysis of MPS3. **[See Appendix 11]**

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:

   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."

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(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 concludes that utilities 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 (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(c)(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-A, Revision 2, Minor Revision 2, "Reload Nuclear Design Methodology," October 2017.

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 conformance of its RETRAN models developed in accordance with VEP-FRD-41 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**. Note, the Appendix 7 discussions were written at a time when VEP-FRD-41 was applicable to only North Anna and Surry. To facilitate use of this report, portions of the discussions that are relevant to specific component models are also reproduced in Section 5, **System Component Model Descriptions**. These discussions have been updated to address the conditions and limitations discussed in the SER for RETRAN-3D (Reference 2.3-2) and the MPS3 Methods Transition LAR (Reference 2.3-3).

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).

2.3-2 Letter from S. A. Richards (USNRC) to G. L. 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.

2.3-3 Letter from Mark D. Sartain (Dominion) to USNRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," May 8, 2015, (Serial No. 15-159).

2.4 Overview of RETRAN Model Upgrades

Dominion's models have undergone several changes since the original issue of this report. Changes through June 2003 were reviewed and approved by the NRC as part of the submittal of VEP-FRD-42 Rev. 2. Changes through the issuance of this report have been reviewed and approved through the methods outlined in Section 2.2.

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 Model Descriptions**.

2.4.1 Model Upgrades Through June 2003

From the original issuance of this report through June 2003, 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.

2.4.2 Model Upgrades from June 2003 Through Issuance of Minor Revision 2

From June 2003 to the issuance of this report, Dominion's RETRAN models have undergone the following changes:

1. A detailed set of RETRAN card overlays was developed to model the Measurement Uncertainty Recapture power uprate at North Anna Units 1 and 2.

2. A detailed set of RETRAN card overlays was developed to model the Westinghouse Robust Fuel Assembly 2 (RFA-2) fuel core in North Anna Units 1 and 2.

3. The current RETRAN-02 models for North Anna and Surry were converted to RETRAN-3D format as part of the transition to that version of the code.

2.4.3 Model Supporting Topical Report Application to Millstone Power Station Unit 3 (Minor Revision 3)

A Dominion Energy RETRAN-3D model was developed for performing MPS3 non-LOCA transient analyses using the VEP-FRD-41-P-A methodology. The base MPS3 RETRAN model noding is virtually identical to the Surry and North Anna models with the exception of some minor noding differences listed as follows:

1. The MPS3 model explicitly models the SI accumulators.

2. The MPS3 model has separate volumes for the SG inlet and outlet plenums.

3. The MPS3 model includes cooling paths between downcomer and upper head.

These modeling differences were outlined in the MPS3 Methods Transition LAR (Reference 2.4-3). Other changes as compared to the North Anna and Surry base modes were made to reflect MPS3 plant design and configuration. Section 4.3.1, **4-Loop Model Configuration and Organization** details the base model and MSLB overlays developed for MPS3.

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).

2.4-3 Letter from Mark D. Sartain (Dominion) to USNRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," May 8, 2015, (Serial No. 15-159).

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 VEPCO's 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.

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.

3.1.3 RETRAN-3D

In July of 1998, the RETRAN Maintenance Group proposed review of the RETRAN-3D code to the NRC. The code documentation (Reference 3.1-17) was submitted in September of that year for review and approval by the NRC. The RETRAN-3D code development was aimed at easing many of the limitations in the RETRAN-02 version of the code. The main objectives established for its development were:

- to extend the analyses capabilities of RETRAN by revising some existing models in RETRAN-02 and adding new models as necessary,
- to improve the performance by making the code more dependable, easier to use, and faster running, and
- to have a more transportable code.

This version of RETRAN-3D includes the following adaptations from RETRAN-02:

- an implicit numerical solution method used for the solution of the steady-state equations and the transient equations,
- a generalized algebraic slip option applicable for concurrent and countercurrent flow conditions,
- improved constitutive relations for terms in the dynamic slip equations,
- the 1979 ANS standard for decay heat,
- a generalized transport model to transport impurities (e.g., boron) with the fluid in the system,
- significant modifications to some RETRAN-02 models including the countercurrent flow logic and the one-dimensional neutron kinetics solution algorithms,
- an option to model nonequilibirium phenomena (five-equation model),
- an option to include noncondensable gas flow,
- an option to analyze multidimenstional neutron kinetics conditions, and
- an improved model to calculate steady-state initial conditions for low power situations.

The NRC issued an SER for RETRAN-3D in January 2001 (Reference 3.1-18). This SE is found in **Appendix 8** of this report. The NRC staff's approval of RETRAN-02 was subject to a number of limitations described in the SEs for the various RETRAN-02 versions and in the Technical Evaluation Reports (TERs). The NRC reviewed and commented on these same limitations in the SE for RETRAN-3D. In MOD02 of this report, Section 5 has been updated to include how Dominion has addressed the limitations with regard to Dominion's application of RETRAN-3D.

The use of RETRAN-3D as a substitute for RETRAN-02 is predicated on the selection of models and options that constitute a near version of RETRAN-02 in the RETRAN-3D code. The NRC approved the use of RETRAN-3D in a "02 mode" given the licensee follows the guidelines outlined in the SE (Reference 3-18). Several changes were made to the code to make RETRAN-3D more robust than RETRAN-02. These changes are always active in RETRAN-3D and include the following:

- Improved transient numerical solution (fully implicit solution of the balance equations, component models and source term are linearized)
- Improvements to the time-step selection logic
- Improved water property curve fits

Other model options were improved with the improvements being active when the particular option is selected. For these options, the RETRAN-02 model was replaced by the improved model and there is no backward compatibility option. Consequently, the following improvements, if selected by the user , may be used in "02 mode" for analyses:

- Fully implicit stead-state solution
- Implicit pressurizer solution
- Wall friction model revised to use the Colebrook equation, allowing consideration of wall roughness rather than smooth pipe
- Control system solution revised to solve a coupled system of equations using a Gauss-Seidel method rather than the single pass marching scheme
- Enthalpy transport model revised by eliminating several simplifying assumptions
- Improved dynamic slip formulation adding form losses
- Improved countercurrent flow junction properties
- Implicit solution of the heat conduction equation
- Combined heat transfer map updated with an improved set of heat transfer correlations and smoothed transitions
- Wall friction and hydrostatic head loses included in critical flow pressure

The SE also stipulated a set of features available in RETRAN-3D but restricted from use in “02-mode” without prior approval from the NRC. These are the following:

- Generalized laminar friction model
- Dynamic gap conductance model
- Accumulator model
- Dynamic flow regime model
- New control blocks added to improve functionality
- Govier horizontal flow regime map and stratified flow friction model
- Chexal-Lellouche drift flux model
- Method of characteristics enthalpy option
- Noncondensable gas flow model
- 3D kinetics
- 5-equation nonequilibrium model

The Dominion transition from RETRAN-02 to RETRAN-3D in “02 mode” was validated 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-3D code improvements. These studies are documented in **Appendix 9**.

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.

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).

3.1-17 EPRI NP-7450-CCM-A, RETRAN-3D, A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Rev. 9, March 2014.

3.1-18 Letter from S. A. Richards (USNRC) to G. L. 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.

3.2 RETRAN Code Approvals

NRC review and approval of RETRAN-3D through MOD003 was discussed generically in the USNRC SER for RETRAN-3D (Reference 3.2-1). During the review process of MOD003, a number of code modifications were suggested by the NRC staff. These modifications along with error corrections that were made during the review period were incorporated into the code by adding these revisions to MOD003.0. The new code version was identified as MOD003.1. Since the issuance of MOD003.1, the code has been modified several times.

REFERENCES FOR SECTION 3.2

3.2-1 Letter from S. A. Richards (USNRC) to G. L. 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.

3.2-2 EPRI NP-7450-CCM-A, RETRAN-3D, A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Rev. 6, August 2005.

4 RETRAN-3D 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, 3 or 4. (All loop related control volumes and junctions begin with one of these digits.) For example, volume X01 refers to the hot leg. For each of the loops in the 3-loop geometry, this refers to volumes 101, 201 and/or 301. For each of the loops in the 4-loop geometry, this refers to volumes 101, 201, 301 and/or 401.

A description of abbreviations used throughout the report is provided in **Section 8**.

4.2 North Anna and Surry 3-Loop Model

4.2.1 Model Configuration and Organization

The North Anna and Surry 3-loop models are available in two geometric configurations for each site:

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. An example of the numbering system used to identify the various component and system models is shown in Table 4.1.

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 qualification 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 conjunction 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.

4.2.2 Noding and Options

Tables 4.2 and 4.3 provide summaries of the control volume and momentum junction nodalization for the Surry and North Anna 3-loop models. The tables reflect a multi-node steam generator secondary geometry. 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) which omits the momentum flux term. 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
EXAMPLE 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 4.1 (CONT.)
EXAMPLE 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 4.2
North Anna/Surry 3-Loop Model 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 secondary segments *	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 4.3
North Anna/Surry 3-Loop Model 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 4.3 (CONT.)
North Anna/Surry 3-Loop Model Momentum Junction Summary

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
North Anna/Surry 3-Loop Model – Primary Nodalization

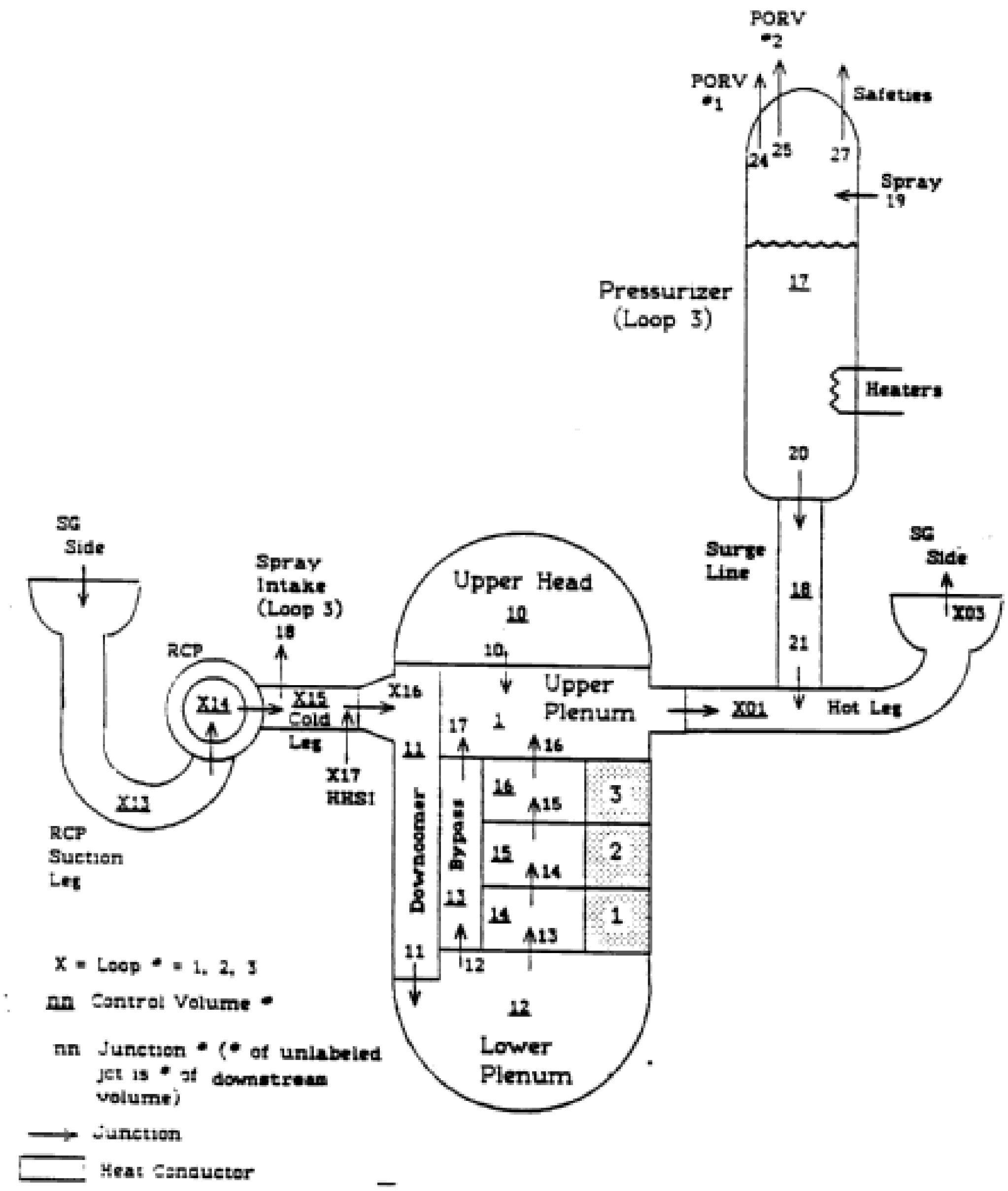


FIGURE 4.2
North Anna/Surry 3-Loop Model – Multi-Node Steam Generator Nodalization

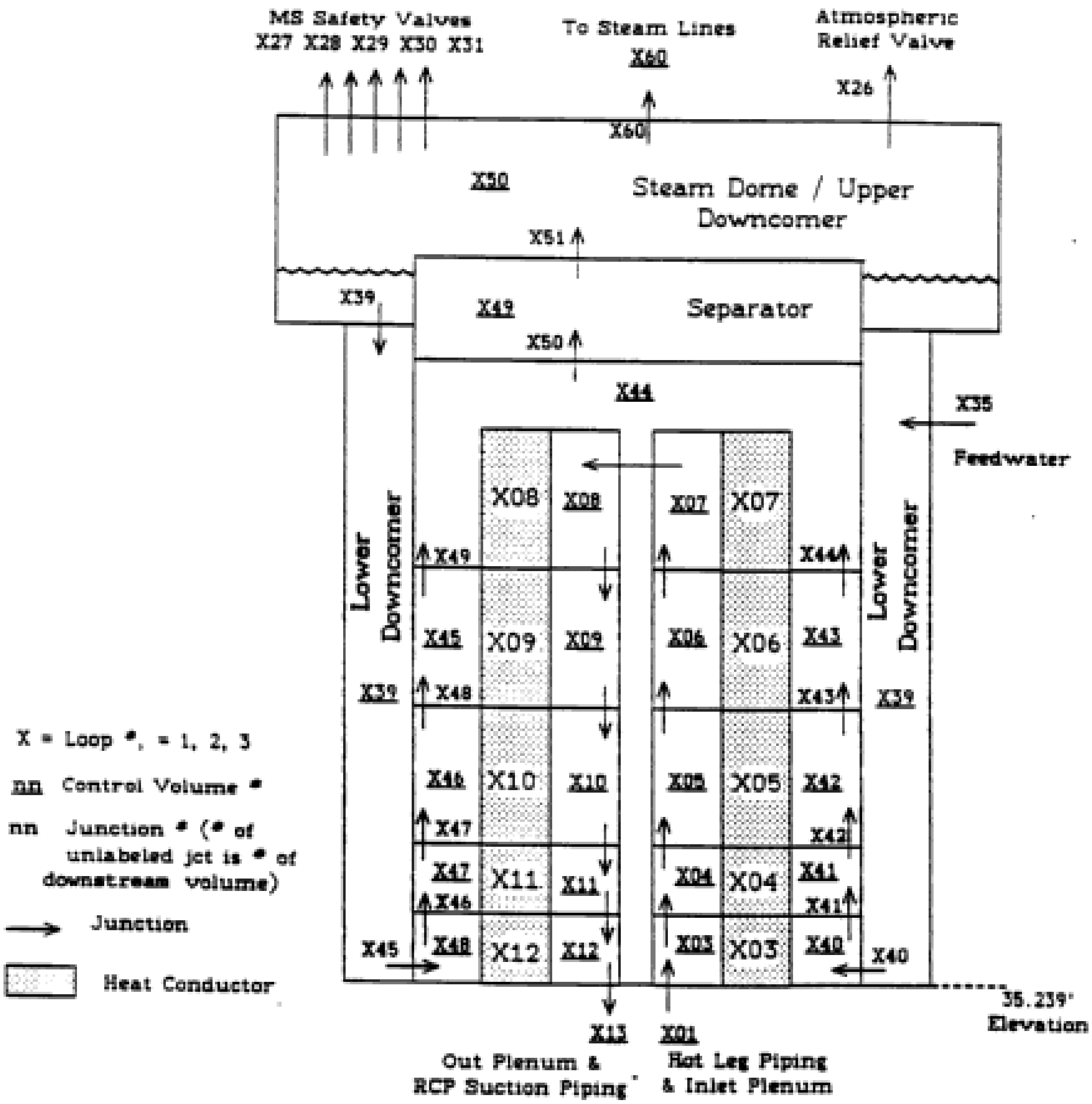
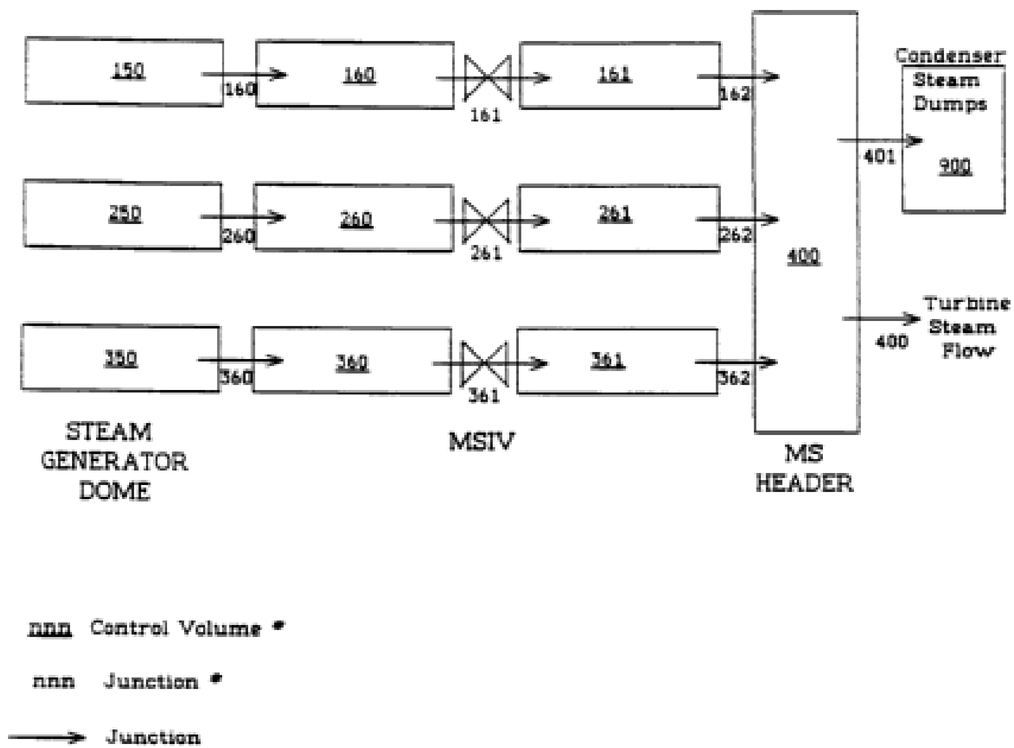


FIGURE 4.3
North Anna/Surry 3-Loop Model – Steam Line Nodalization



4.3 Millstone Unit 3, 4-Loop Model

4.3.1 Configuration and Organization

The base MPS3 model noding is virtually identical to the Surry and North Anna models with the exception of some minor noding differences listed as follows:

a) The MPS3 model explicitly models the SI accumulators.
b) The MPS3 model has separate volumes for the SG inlet and outlet plenums.
c) The MPS3 model includes cooling paths between downcomer and upper head.

The MPS3 base model noding diagram for a representative loop is shown on Figure 4.4. This model simulates all four reactor coolant system (RCS) loops and has a single-node steam generator (SG) secondary side. Volume numbers are circled, junctions are represented by arrows, and the heat conductors are shaded. Control volume, junction and heat conductor region numbers starting with an X refer to four-loop geometry regions where X can have the value of 1, 2, 3 or 4. The SG primary nodalization includes ten steam generator tube volumes and conductors and a single volume for the secondary side.

Figure 4.4 represents 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. A noding diagram of the split MPS3 reactor vessel is shown in Figure 4.5.

There is a multi-node SG secondary overlay that can be added to the base model for sensitivity studies and benchmarks as an aid to understanding. It should be noted that the multi-node SG's use in licensing calculations requires additional qualification of the RETRAN drift flux and dynamic slip correlations for PWR applications – see Section 5.7.2.

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, and high-head/intermediate-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.

The technical basis and detailed input development for each model is maintained in a configuration controlled document. Cards are organized within the decks in numerical order.

4.3.2 Noding and Options

Tables 4.4 and 4.5 provide summaries of the control volume and momentum junction nodalization for the MPS3 4-loop model with single-node secondary SG. 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) which omits the momentum flux term.

TABLE 4.4
Millstone Unit 3, 4-Loop Model 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
SG inlet plenum	X02	0	No
SG outlet plenum	X13	0	No
Pump suction piping	X14	0	Yes
Reactor coolant pump	X15	0	No
Cold leg piping	X16	0	Yes
Accumulator	X20	x	No
Pressurizer	17	x	No
Surge line	18	0	No
SG tubes	X03-X12	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, 3 or 4

TABLE 4.5
Millstone Unit 3, 4-Loop Model Momentum Junction Summary

Junction Description	Jct #	Type	Valve	Chok	Trp
Downcomer – upper head	9	Norm	0	No	No
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 inlet plenum	X02	Norm	0	No	No
SG inlet plentum – SG tubes	X03	Norm	0	No	Yes
SG tubes	X04-X12	Norm	0	No	Yes
SG tubes– SG outlet plentum	X13	Norm	0	No	Yes
SG outlet plenum – RCP suction	X14	Norm	0	No	No
Pump suction	X15	Norm	0	No	No
Pump discharge	X16	Norm	0	No	No
Rx vessel inlet nozzle	X17	Norm	0	No	No
Safety injection	X18	Fill	0	No	No
Accumulator	X20	Norm	x	Yes	No
Charging	121	Fill	0	No	No
Letdown	321	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, 3 or 4

TABLE 4.5 (CONT.)
Millstone Unit 3, 4-Loop Model Momentum Junction Summary

Junction Description	Jct #	Type	Valve	Chok	Trp
Pressurizer spray intake	18	Fill	0	No	No
Pressurizer spray	19	Fill	0	No	No
Pressurizer – surge line	20	Norm	0	No	No
Surge line – “B” hot leg	21	Norm	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
Feedwater inlet	X35	Fill	0	No	No
Auxillary feedwater inlet	X36	Fill	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
SG safety valves	X27-X31	Norm	x	Yes	No
MS PORV (relief valve)	X32	Norm	x	Yes	No
Condenser steam dump	995	Norm	x	Yes	No
Turbine steam flow	999	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, 3 or 4

FIGURE 4.4
Millstone Unit 3, 4-Loop Model – Primary and Single Node Steam Generator Secondary Nodalization

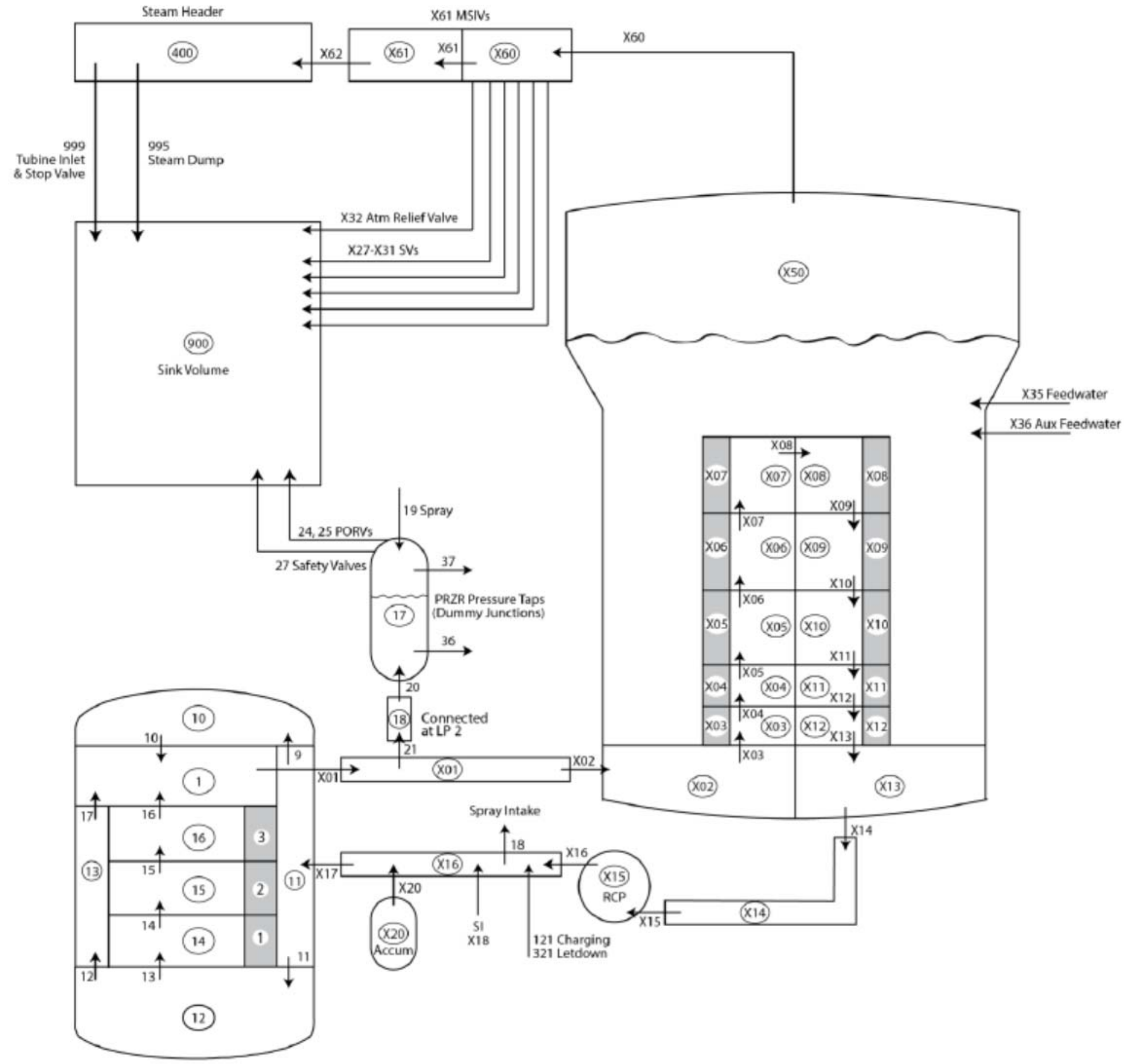
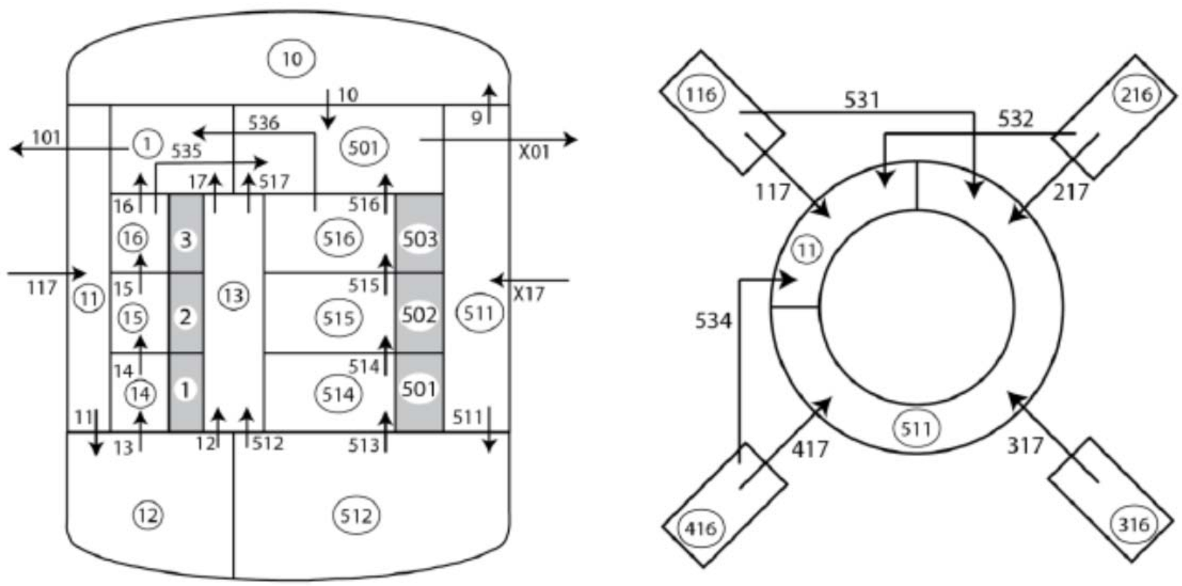


FIGURE 4.5
Millstone Unit 3, 4-Loop Model – Main Steam Line Break Split Core Nodalization



5 SYSTEM COMPONENT MODEL DESCRIPTIONS

Section 5 describes Dominion Energy's conservative modeling of nuclear power plant systems and components for transient evaluations using RETRAN. These discussions may contain plant-specific design information (e.g. setpoints, logic). The conservative modeling features discussed are incorporated into each Dominion Energy RETRAN model developed in accordance with VEP-FRD-41 but in a manner that reflects the design of the plant of interest.

Further, the safety analysist is responsible for determining the input and modeling that predicts a transient's dynamic response in a conservative manner. The analysist considers initial conditions, core reactivity parameters, and assumptions concerning overall systems performance such as component availability and protection system characteristics. The bases for any deviations from the descriptions presented in Section 5 are documented in the individual application calculation.

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 (with minor differences depending on plant design) 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

As determined by the plant design, the high pressurizer pressure, low pressurizer pressure, and OTΔT functions use a compensated (lead-lag) or uncompensated (control volume) pressurizer pressure signal as input to the protection function.

5.2.5 Overtemperature Delta-T

A representative OTΔT setpoint is given by:

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

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	= Pressurizer pressure (compensated or uncompensated, per plant design)
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

A representative OPΔT setpoint is given by:

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

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
REPERSENTATIVE 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
REPRESENTATIVE 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 in the North Anna and Surry 3-loop models. The MPS3 model includes an additional cooling path between downcomer and upper head. The cooling path is included to appropriately model upper head T-cold conditions. In all three models, 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 qualifying the RETRAN system model for fuel vendor and/or other fuel design changes. Other fuel specific models have been developed using the same method.

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. These limitations were evaluated for RETRAN-3D/MOD003 in Reference 5.3-6. Those qualifications and restrictions that are applicable to the vessel and core model are discussed and evaluated in this section. The number designations for the qualifications and restrictions are those of the Safety Evaluation Report for RETRAN-3D/MOD003.

5) 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

Dominion's RETRAN hot pin model is used in rod ejection, rod withdrawal from subcritical, and locked rotor analyses with the metal-water reaction option selected. 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.

7) 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.

11) 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.

24) 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 [Tables 4.2 and 4.4].

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.

5.3-6 Letter from S. A. Richards (USNRC) to G. L. 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.

5.4 Primary Piping

5.4.1 Description

Unlike the model presented in Rev. 0 of this report, there is no separate SG inlet volume for the North Anna and Surry models. 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. The MPS3 base model includes separate volumes for the SG inlet and outlet plenums.

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. These limitations were evaluated for RETRAN-3D/MOD003 in Reference 5.4-3 Section V. Those qualifications and restrictions that are applicable to the primary piping section are discussed and evaluated in this section. The number designations for the qualifications and restrictions are those of the Safety Evaluation Report for RETRAN-3D/MOD003.

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

<u>Discussion</u>
Dominion RETRAN models use the temperature 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.

5.4-3 Letter from S. A. Richards (USNRC) to G. L. 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.

FIGURE 5.4-1

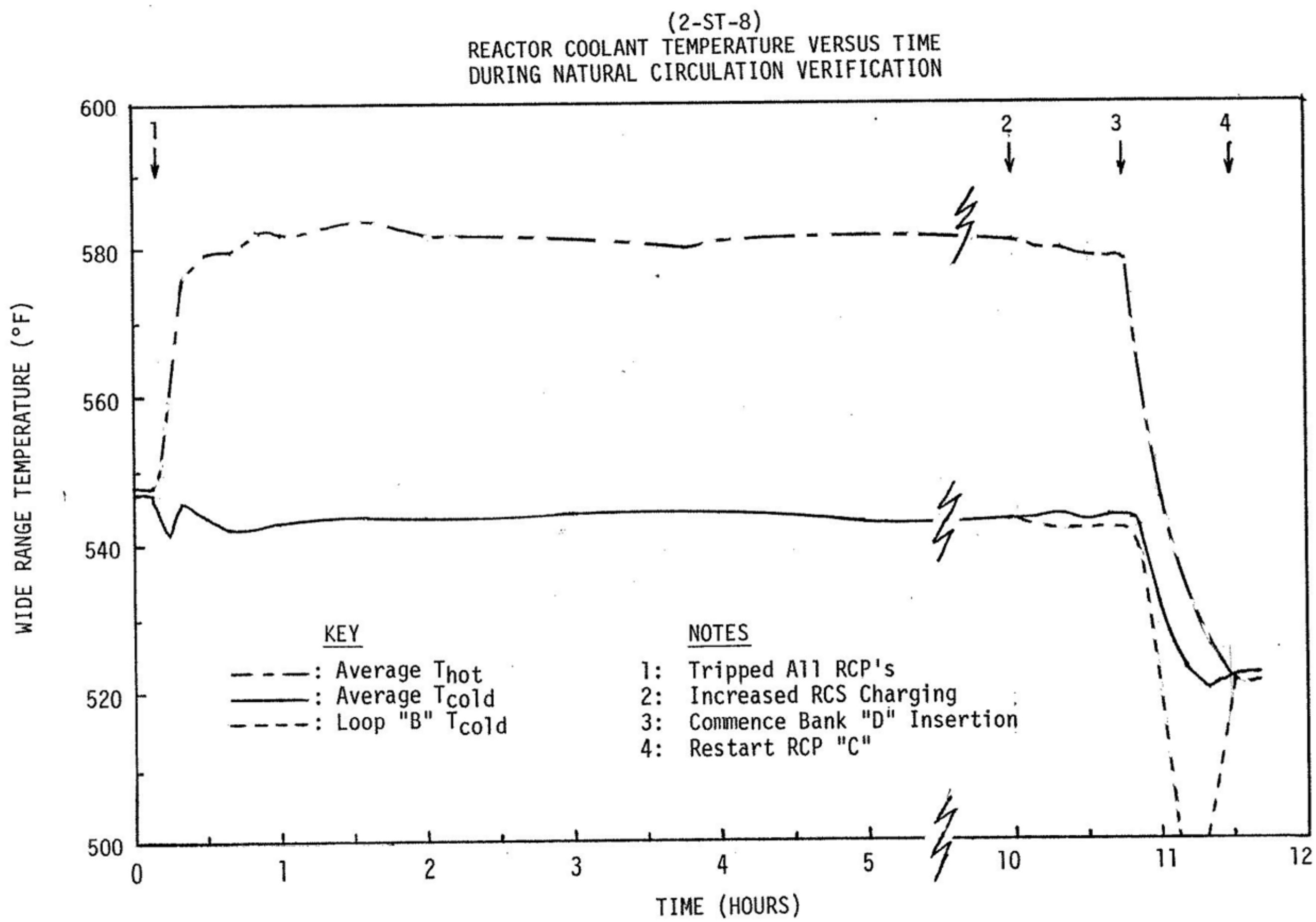
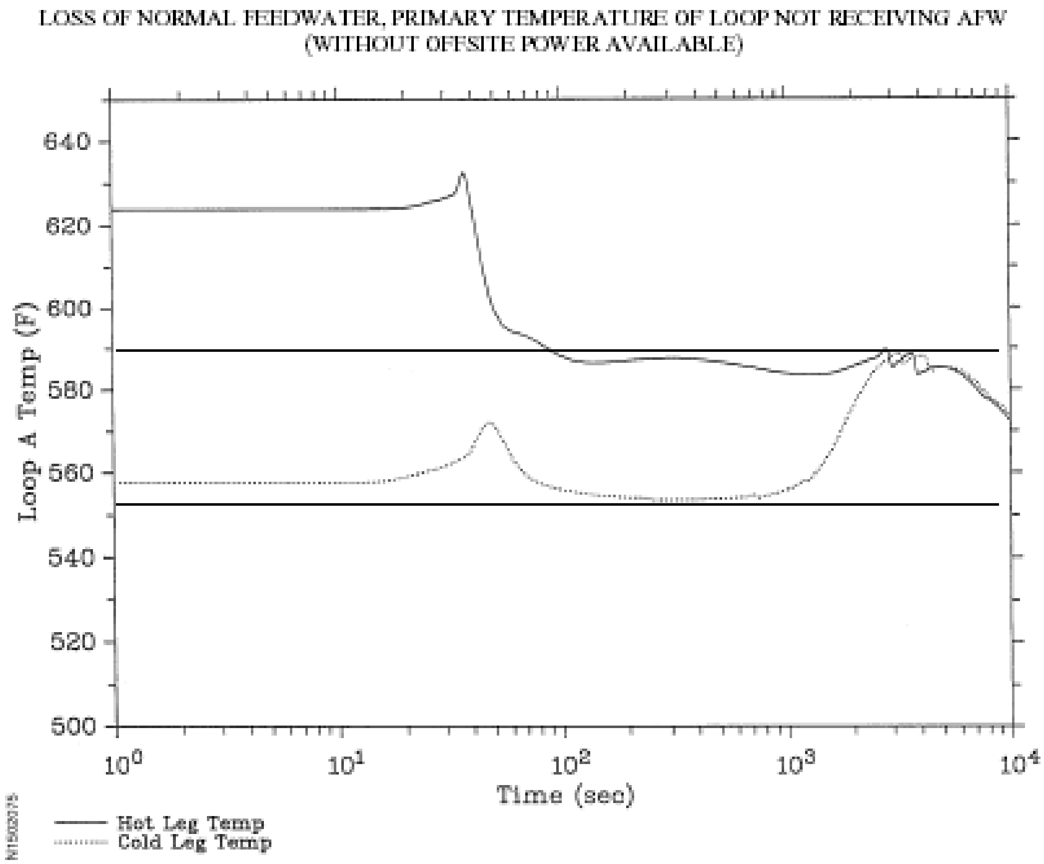


FIGURE 2: AVERAGE HOT AND COLD LEG WIDE RANGE VS. TIME

FIGURE 5.4-2

Natural Circulation Response Following

a Loss of Offsite Power



5.5 Reactor Coolant Pumps

5.5.1 Description

Plant specific pump curves are used at least for the first quadrant. Westinghouse 5200 pump curves are used for other segments where plant specific curves are not applied.

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. These limitations were evaluated for RETRAN-3D/MOD003 in Reference 5.5-3 Section V. Those qualifications and restrictions that are applicable to the reactor coolant pump model are discussed and evaluated in this section. The number designations for the qualifications and restrictions are those of the Safety Evaluation Report for RETRAN-3D/MOD003.

20) 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 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.

MPS3 plant specific RCP pump curves are used in the MPS3 base model.

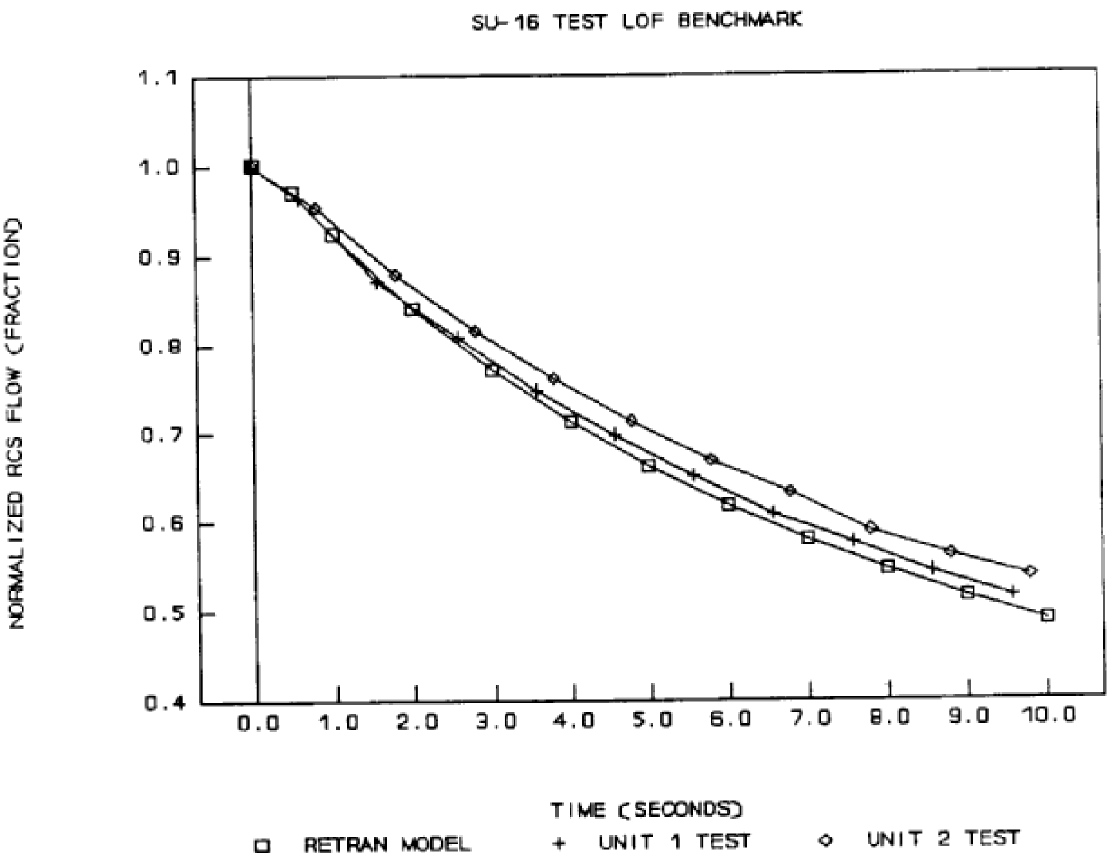
REFERENCES FOR SECTION 5.5

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.

5.5-3 Letter from S. A. Richards (USNRC) to G. L. 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.

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.

5.6.1.2 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.

5.6.1.3 Pressurizer PORVs

The PORV model may assume no valve movement occurs during the early portions the PORV open and close stroke times to account for process lags, consistent with the plant design.

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 and MPS3 have drained the pressurizer loops seals, so the WCAP-12910 model is no longer applicable to these plants. However, it is retained for reference as a modeling option. This option provides more limiting overpressure case results. A more realistic drained loop seal model may be specified by eliminating the 1% "medium shift" and/or 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 RETRAN-3D/MOD3 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. These limitations were evaluated for RETRAN-3D/MOD003 in Reference 5.6-8 Section V. Those qualifications and restrictions that are applicable to the pressurizer model are discussed and evaluated in this section. The number designations for the qualifications and restrictions are those of the Safety Evaluation Report for RETRAN-3D/MOD003.

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

<u>Discussion</u>
RETRAN-3D/MOD003 includes a five equation option for modeling nonequilbirum conditions between liquid and vapor phases. However to be compliant with the stipulations of "02 mode", this option is not used. Therefore the RETRAN-02 modeling option which allows certain volumes to be modeled as nonequilibrium regions is used.

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

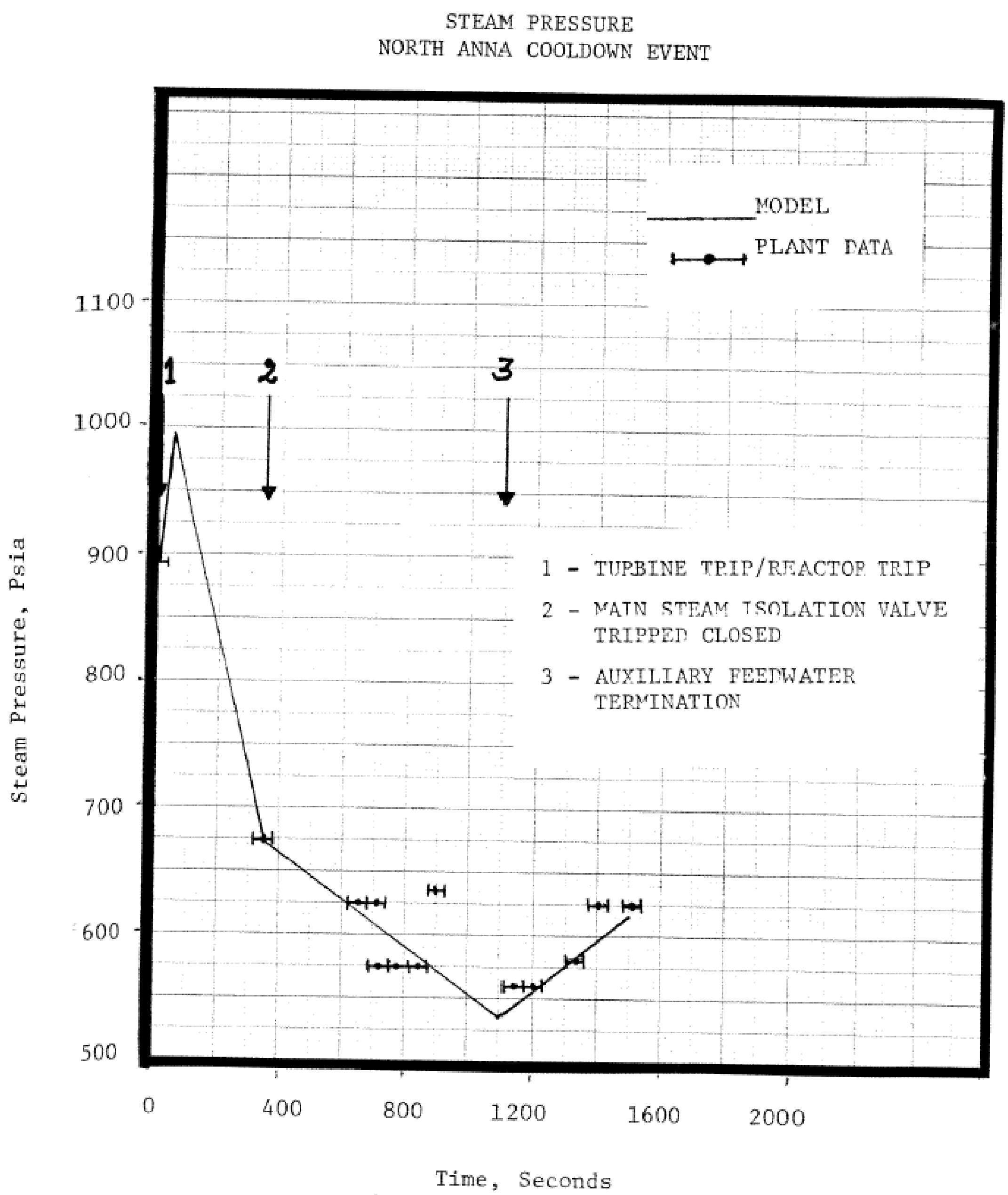

STEAM PRESSURE
NORTH ANNA COOLDOWN EVENT
MODEL
PLANT DATA
1
2
3
1 - TURBINE TRIP/REACTOR TRIP
2 - MAIN STEAM ISOLATION VALVE TRIPPED CLOSED
3 - AUXILIARY FEEDWATER TERMINATION
Steam Pressure, Psia
1100
1000
900
800
700
600
500
0
400
800
1200
1600
2000
Time, Seconds

FIGURE 5.6-2

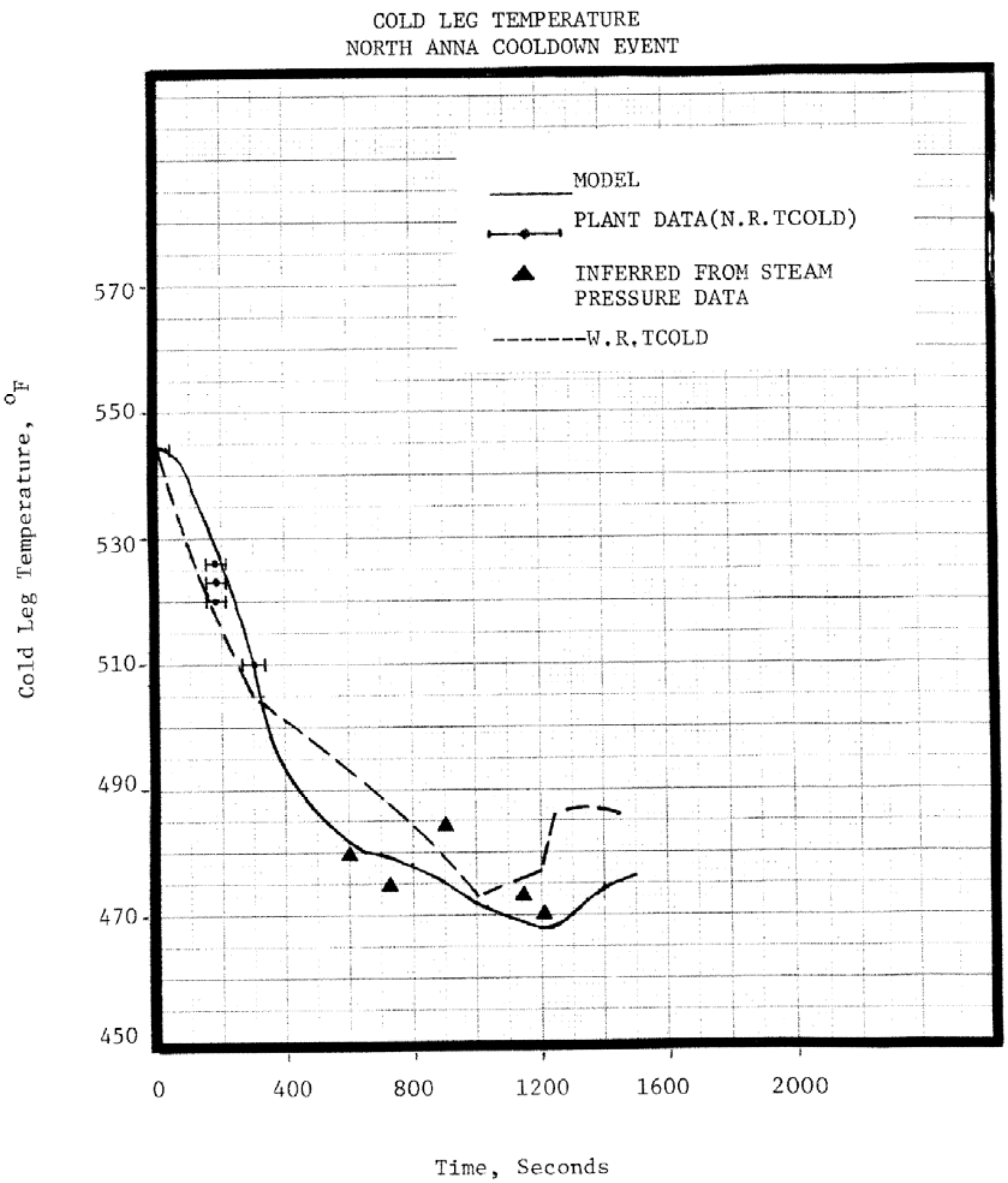

COLD LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT
MODEL
PLANT DATA(N.R.TCOLD)
INFERRED FROM STEAM PRESSURE DATA
W.R.TCOLD
Cold Leg Temperature, °F
570
550
530
510
490
470
450
0
400
800
1200
1600
2000
Time, Seconds

FIGURE 5.6-3

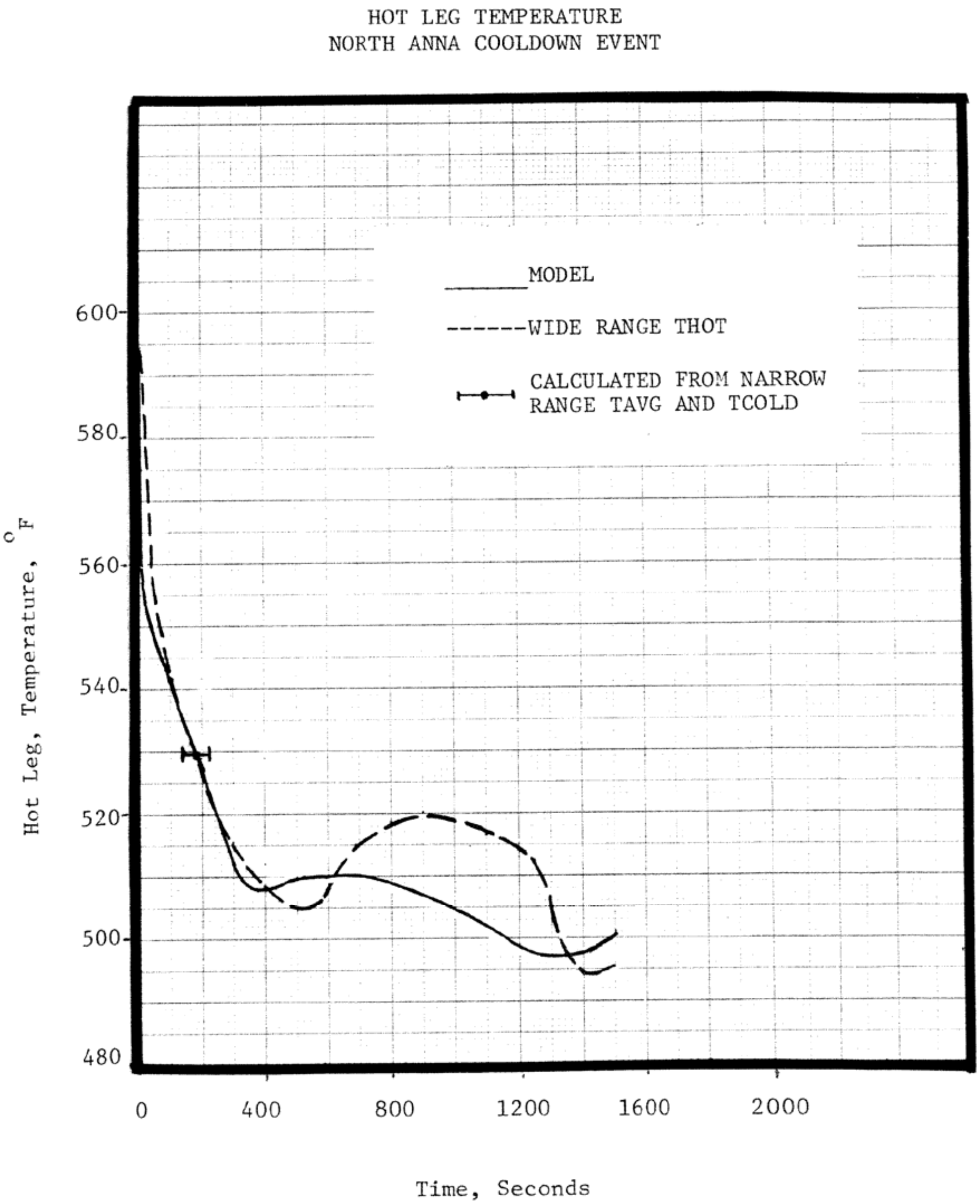

HOT LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT
MODEL
------WIDE RANGE THOT
CALCULATED FROM NARROW RANGE TAVG AND TCOLD
600
580
560
540
520
500
480
Hot Leg, Temperature, °F
0
400
800
1200
1600
2000
Time, Seconds

FIGURE 5.6-4

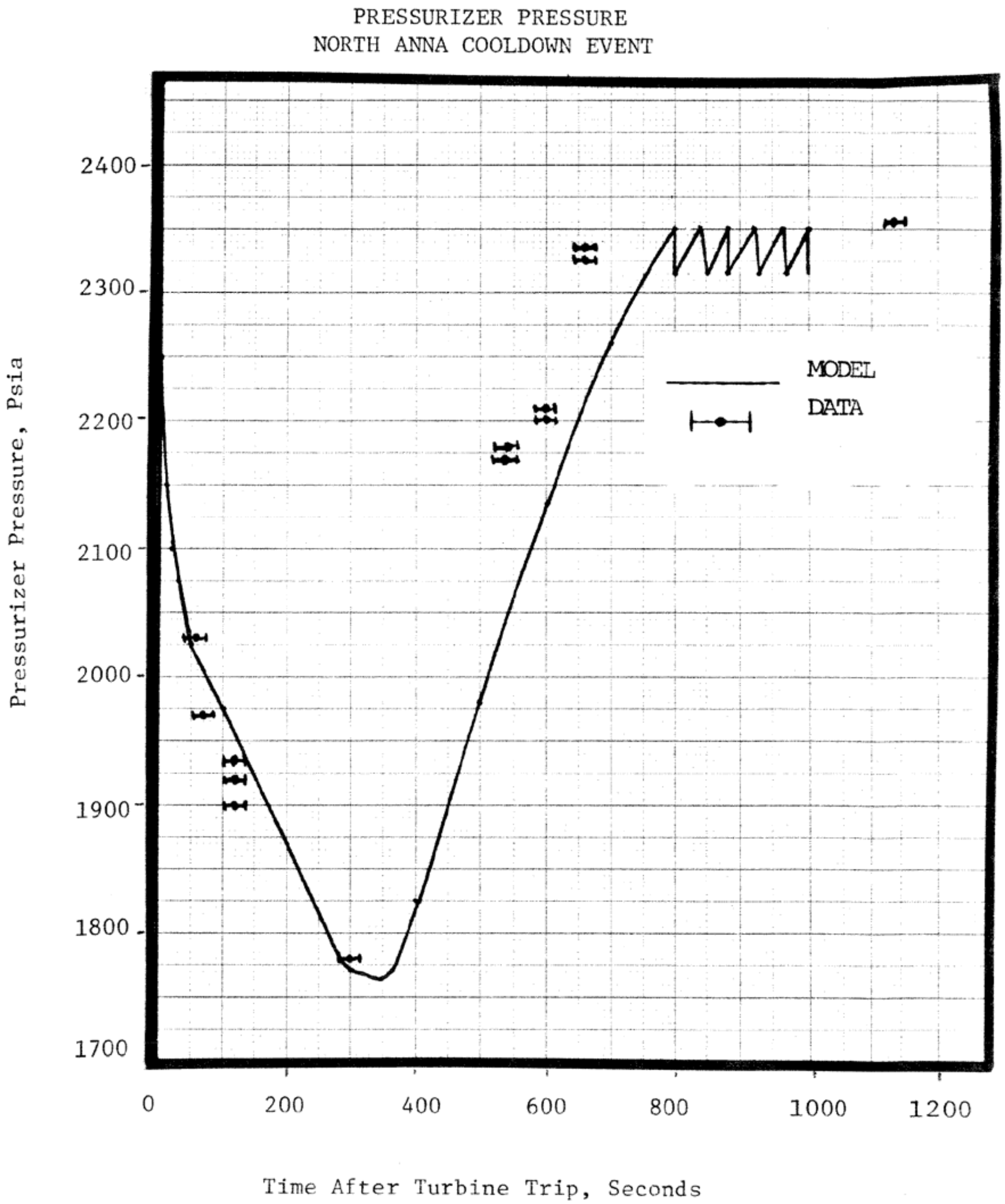
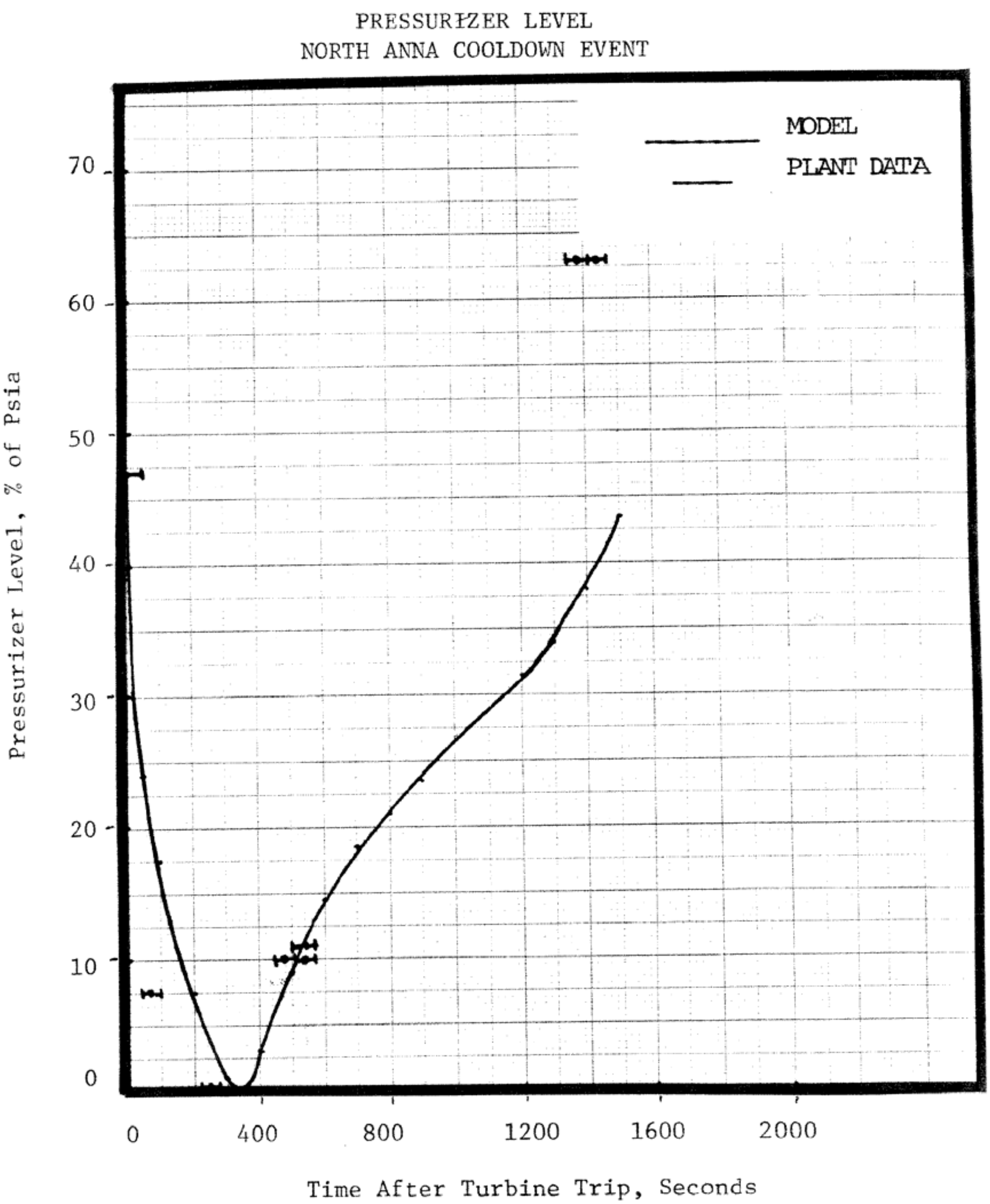

PRESSURIZER PRESSURE
NORTH ANNA COOLDOWN EVENT
Pressurizer Pressure, Psia
2400
2300
2200
2100
2000
1900
1800
1700
0
200
400
600
800
1000
1200
MODEL
DATA
Time After Turbine Trip, Seconds

FIGURE 5.6-5


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

18) 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 of RETRAN-02 [Reference 5.6-5] and is subsequently approved for use in RETRAN-3D per the SE [Reference 5.6-8], 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.

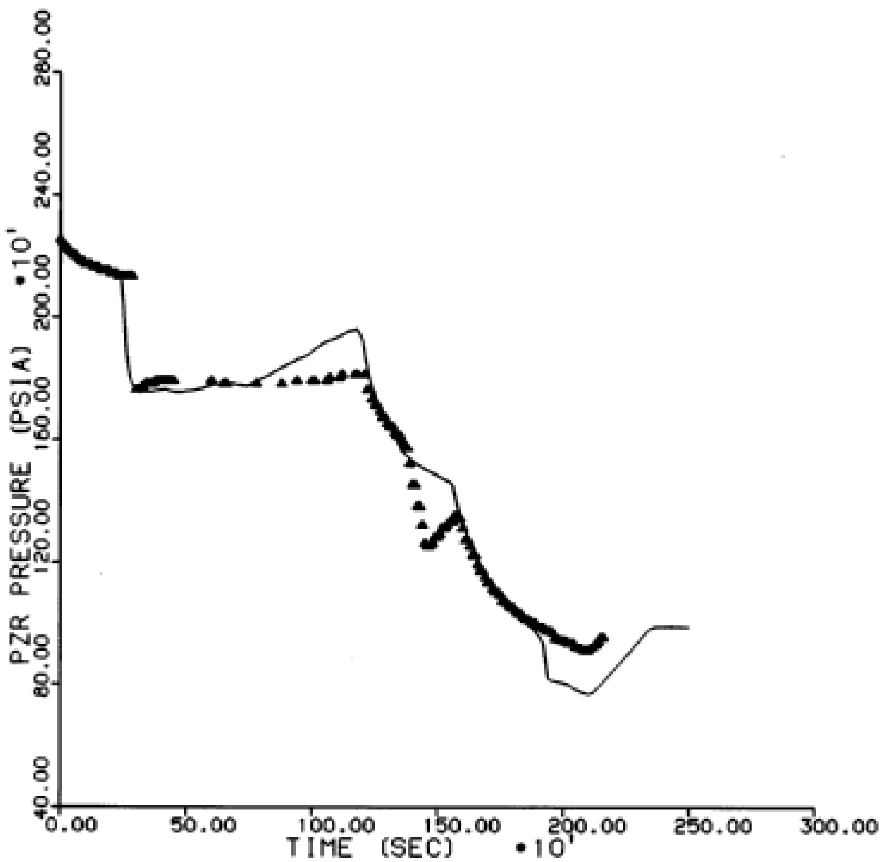
The North Anna Unit 2 Natural Circulation Tests conducted in July 1980 measured the effect of convective heat losses from the pressurizer with all heaters secured. The observed effect was about 5 F/hr liquid temperature cooldown and about 38 psi/hr pressure loss [Reference 5.6-6]-see Figure 5.6-12**.** 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

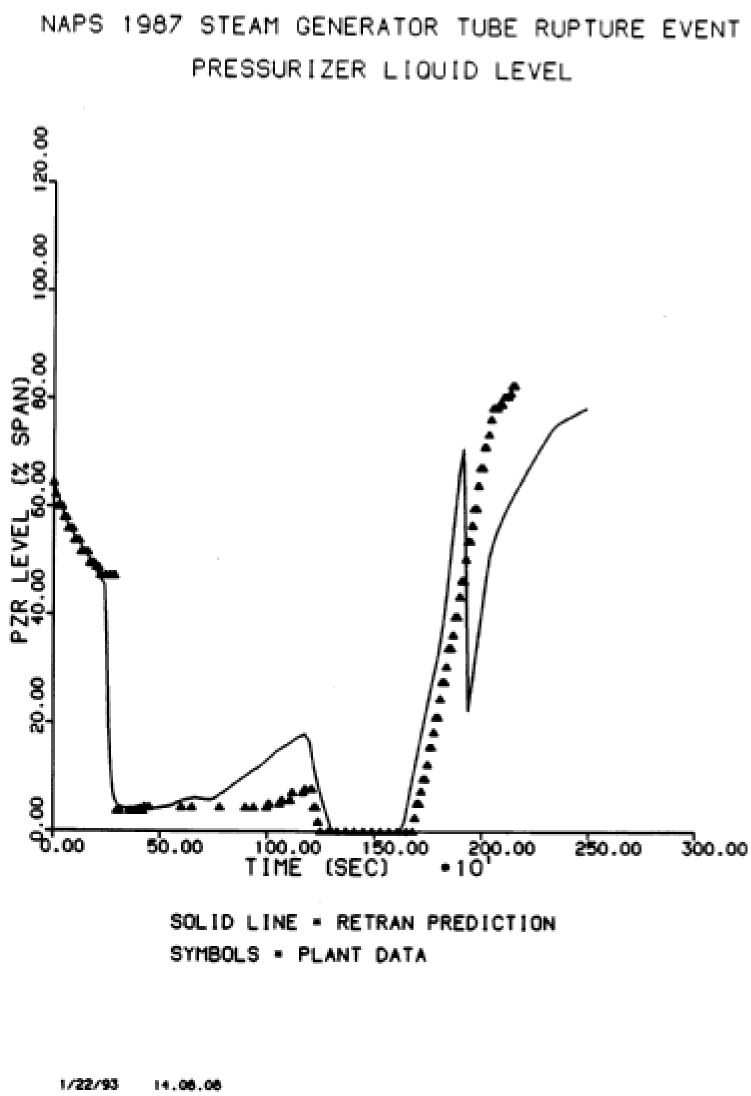
NAPS 1987 STEAM GENERATOR TUBE RUPTURE EVENT
PRESSURIZER PRESSURE



SOLID LINE = RETRAN PREDICTION
SYMBOLS = PLANT DATA

1/22/93 14.07.56

FIGURE 5.6-7


NAPS 1987 STEAM GENERATOR TUBE RUPTURE EVENT
PRESSURIZER LIQUID LEVEL
PZR LEVEL (% SPAN)
0.00
20.00
40.00
60.00
80.00
100.00
120.00
0.00
50.00
100.00
150.00
200.00
250.00
300.00
TIME (SEC) *10¹
SOLID LINE = RETRAN PREDICTION
SYMBOLS = PLANT DATA
1/22/93 14.06.06

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.

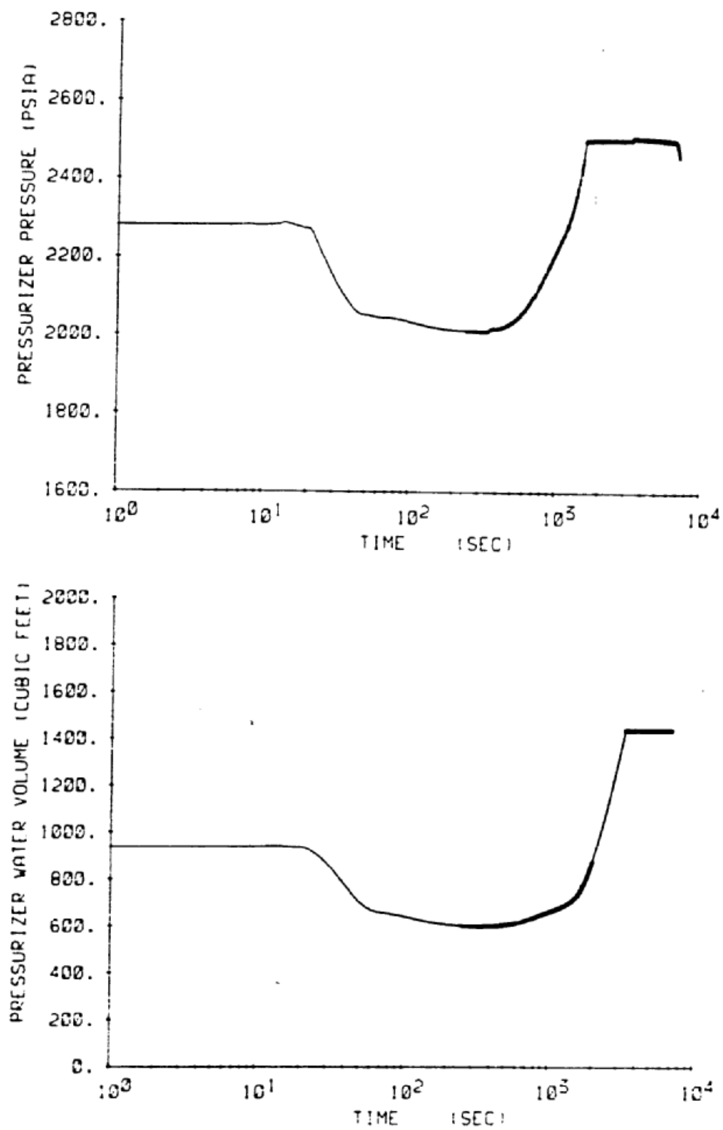
TABLE 5.6-1

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

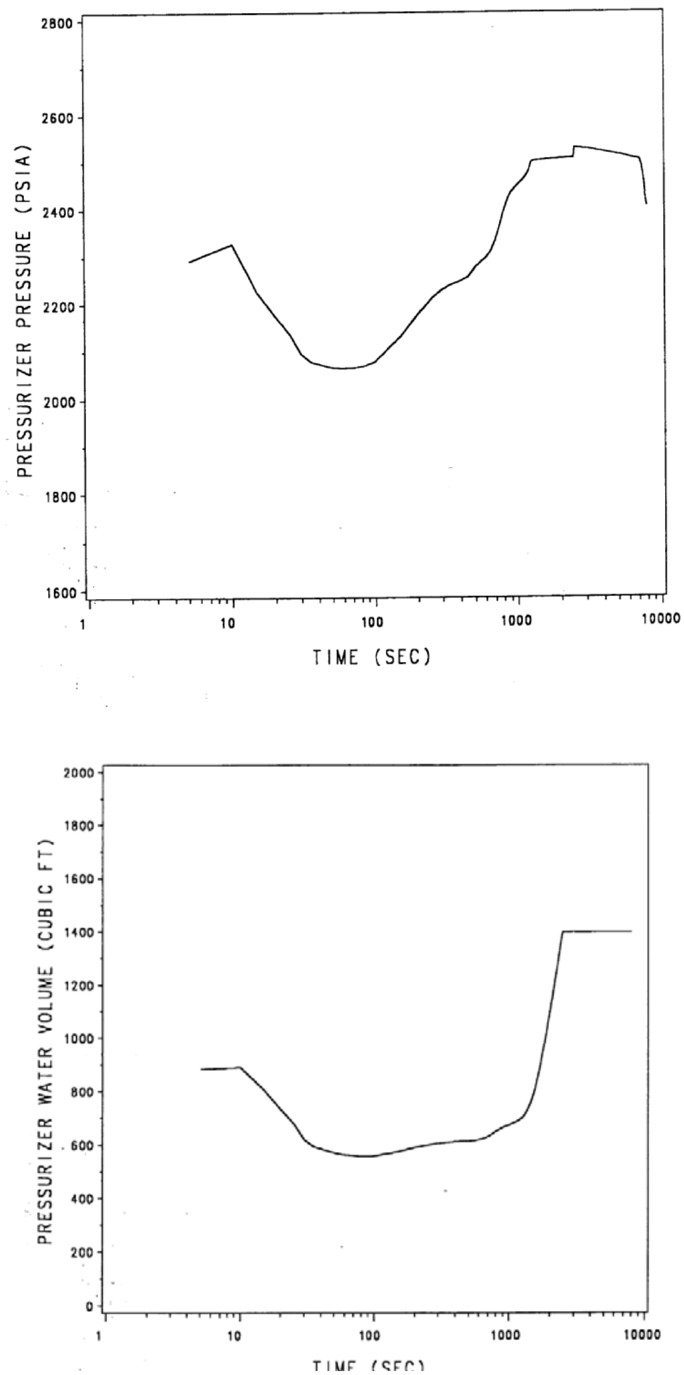


FIGURE 5.6-10
FEEDLINE BREAK
.717 Ft^2 Break/ 340 GPM Auxiliary Feedwater
LOFTRAN Response

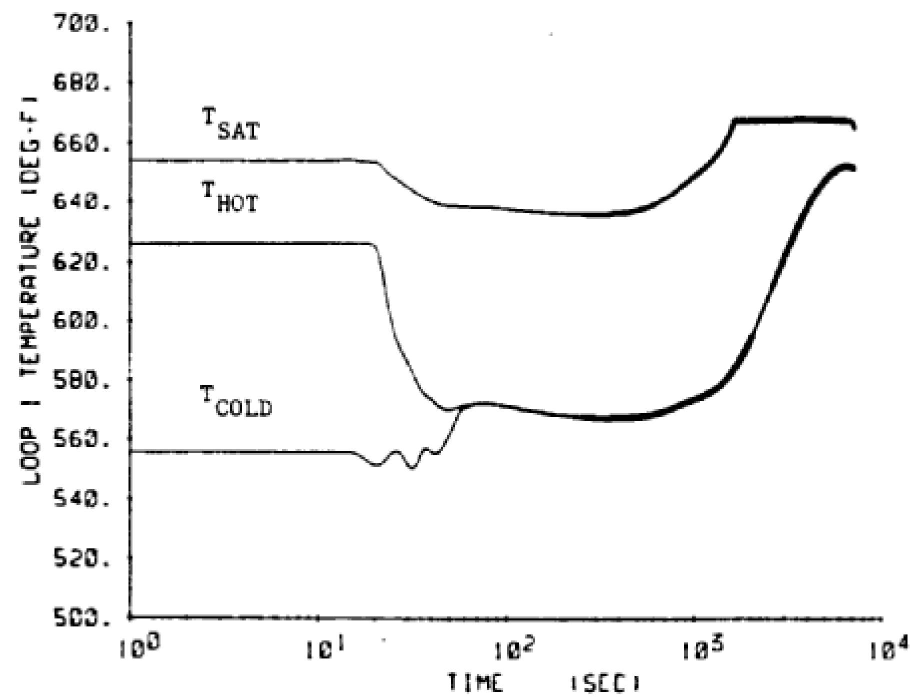


FIGURE 5.6-11
FEEDLINE BREAK
.717 Ft^2 Break/ 340 GPM Auxiliary Feedwater
RETRAN Response

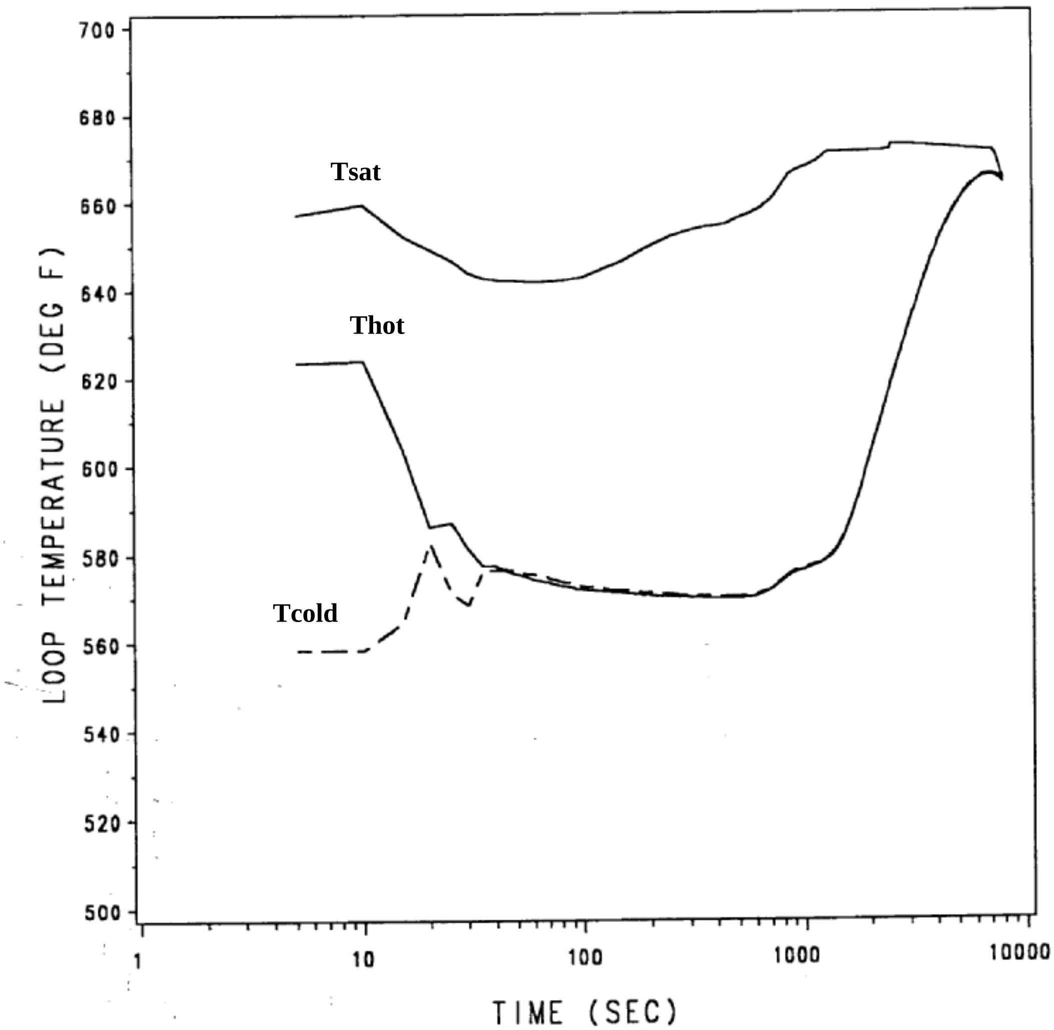


FIGURE 5.6-12

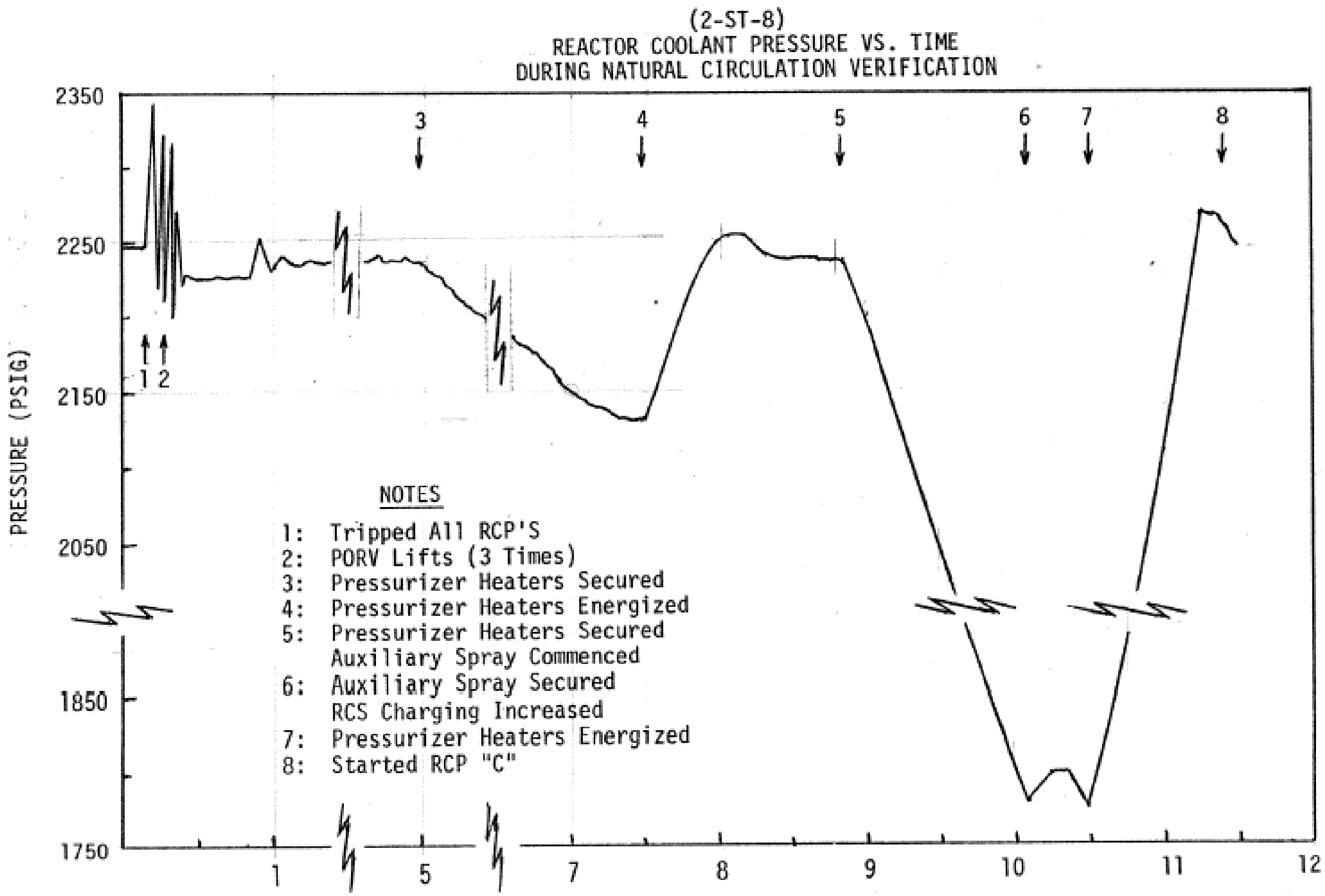


FIGURE 3: REACTOR COOLANT PRESSURE VS. TIME

24) 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.

<u>Discussion</u>

Dominion PWR RETRAN models use bubble rise in the pressurizer, reactor vessel upper head, and steam generator dome regions [Tables 4.2 and 4.4].

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.

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

<u>Discussion</u>

Refer to the response to Limitation 18. 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.

Furthermore, the SE for RETRAN-3D approved the pressurizer model for use with filling and draining events as outline in condition 18.

37) 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 Limitations 18 and 31.

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. Since the MPS3 non-equilibrium pressurizer is modeled as described in Section 5.6.1 and its subsections, the conclusions of the North Anna and Surry RETRAN comparisons are applicable to the MPS3 plant.

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.

5.6-8 Letter from S. A. Richards (USNRC) to G. L. 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.

5.7 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. These limitations were evaluated for RETRAN-3D/MOD003 in Reference 5.7-4 Section V. Those qualifications and restrictions that are applicable to the steam generator model are discussed and evaluated in this section. The number designations for the qualifications and restrictions are those of the RETRAN-3D Safety Evaluation Report.

9) 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 (16) are met.

<u>Discussion</u>
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

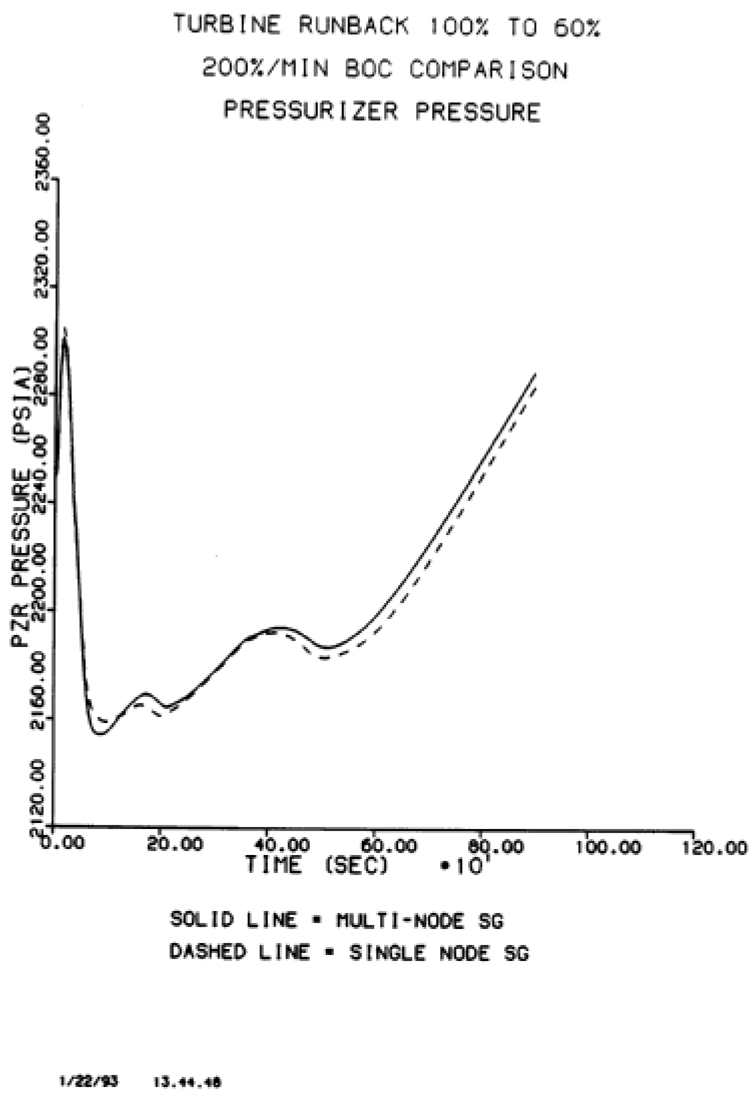

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
PRESSURIZER PRESSURE
PZR PRESSURE (PSIA)
2120.00
2160.00
2200.00
2240.00
2280.00
2320.00
2360.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10[1]
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 13.44.46

FIGURE 5.7-2

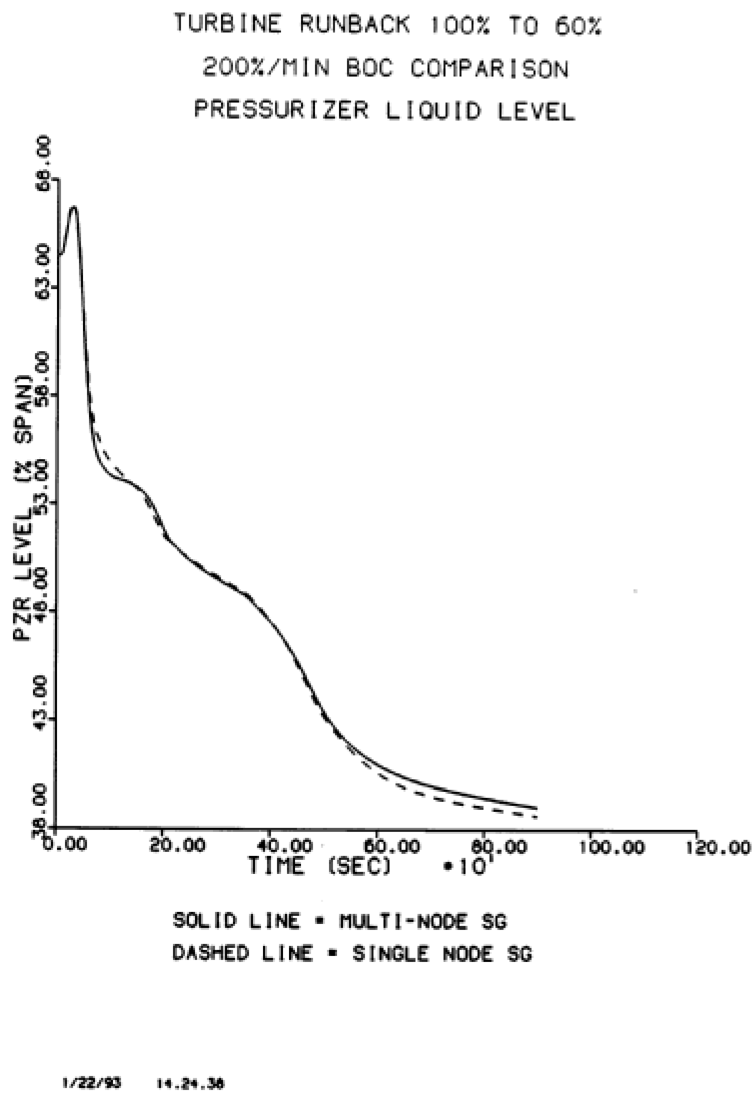

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
PRESSURIZER LIQUID LEVEL
PZR LEVEL (% SPAN)
38.00
43.00
48.00
53.00
58.00
63.00
68.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10¹
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 14.24.38

FIGURE 5.7-3

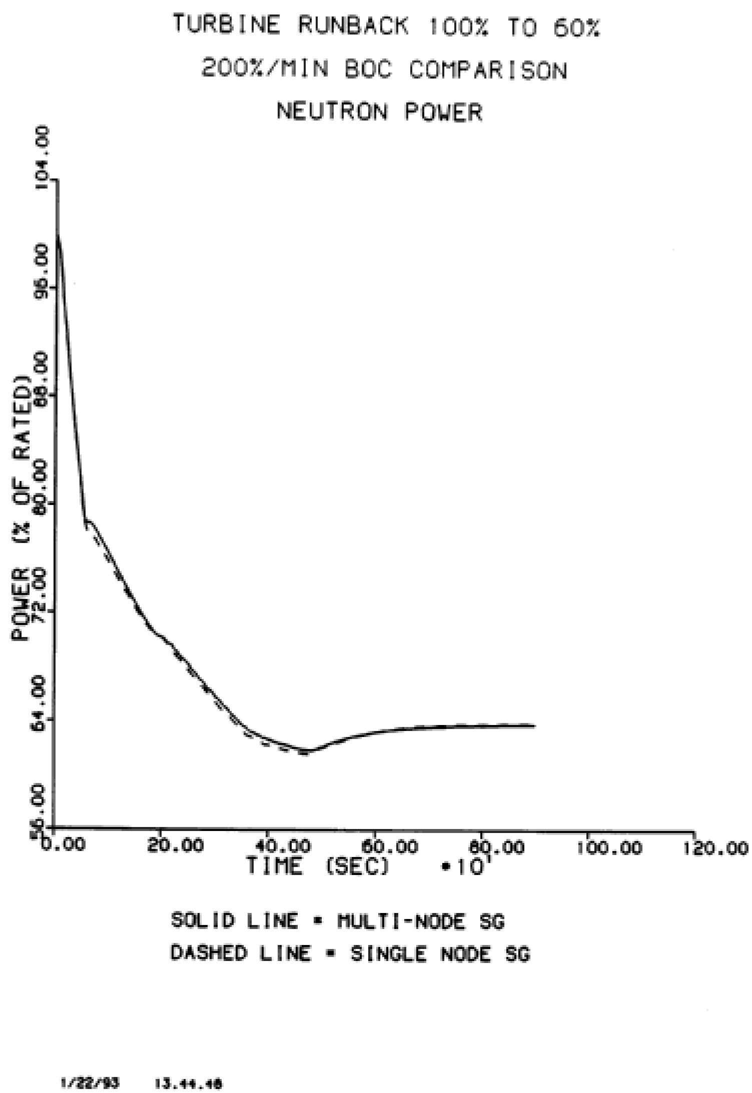

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
NEUTRON POWER
POWER (% OF RATED)
104.00
96.00
88.00
80.00
72.00
64.00
56.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10¹
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 13.44.46

FIGURE 5.7-4

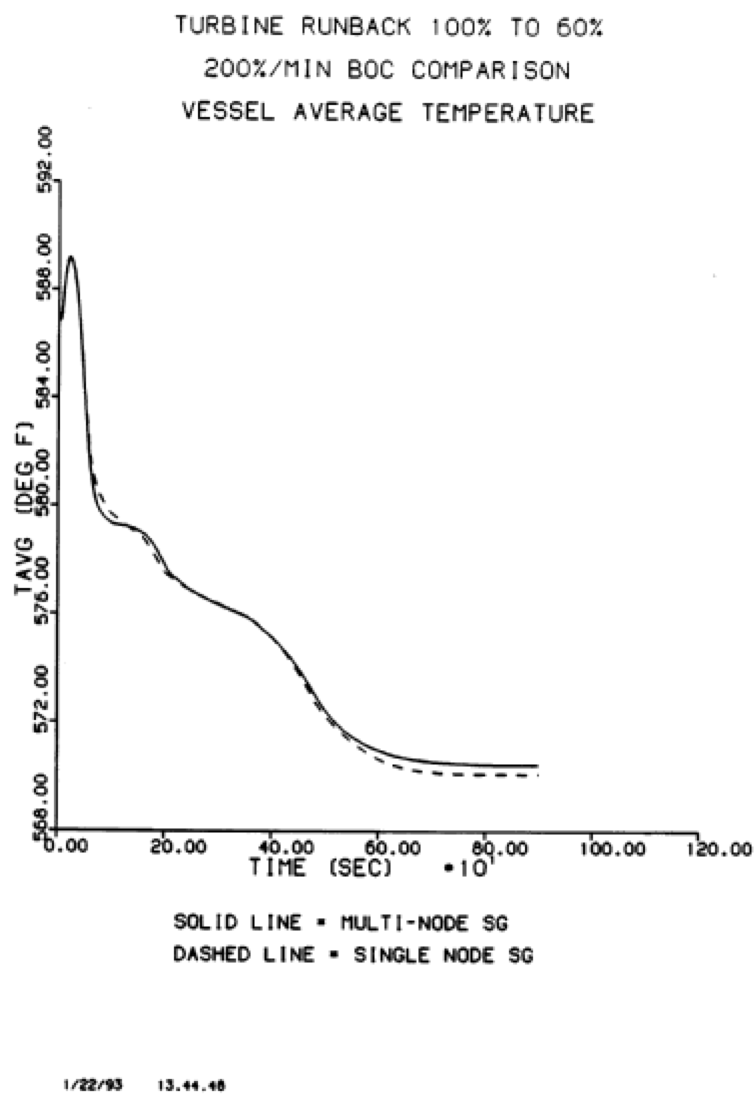

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
VESSEL AVERAGE TEMPERATURE
TAVG (DEG F)
568.00
572.00
576.00
580.00
584.00
588.00
592.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 13.44.48

FIGURE 5.7-5

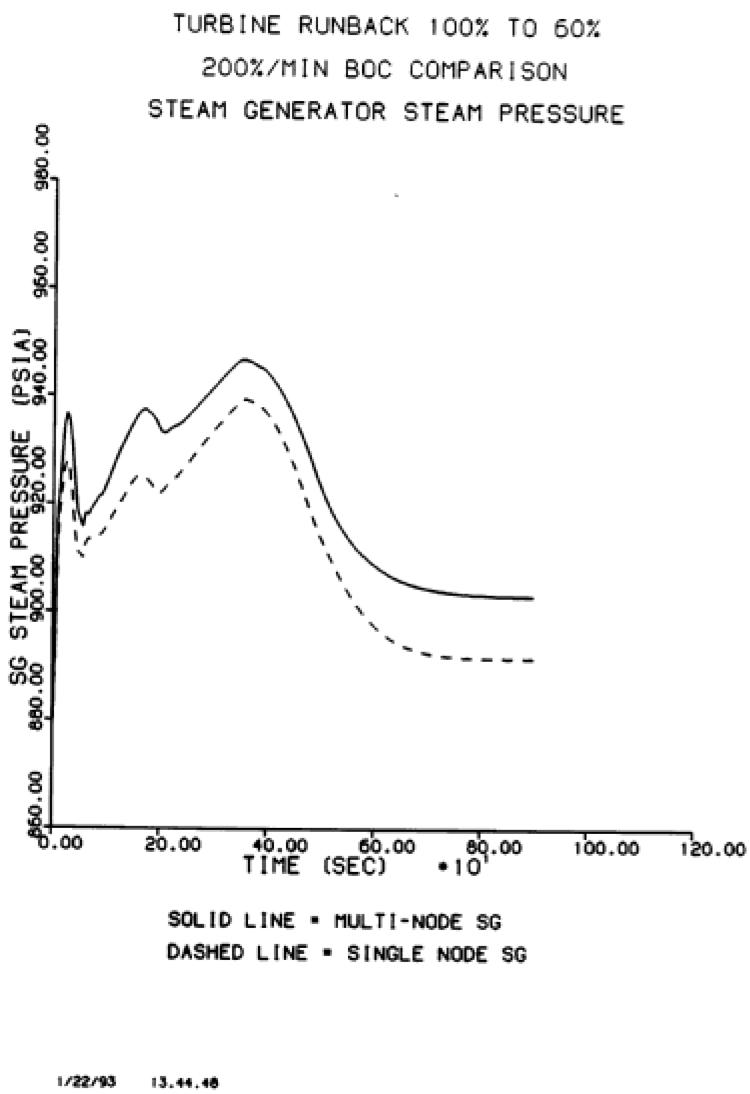

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
STEAM GENERATOR STEAM PRESSURE
SG STEAM PRESSURE (PSIA)
860.00
880.00
900.00
920.00
940.00
960.00
980.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 13.44.48

FIGURE 5.7-6

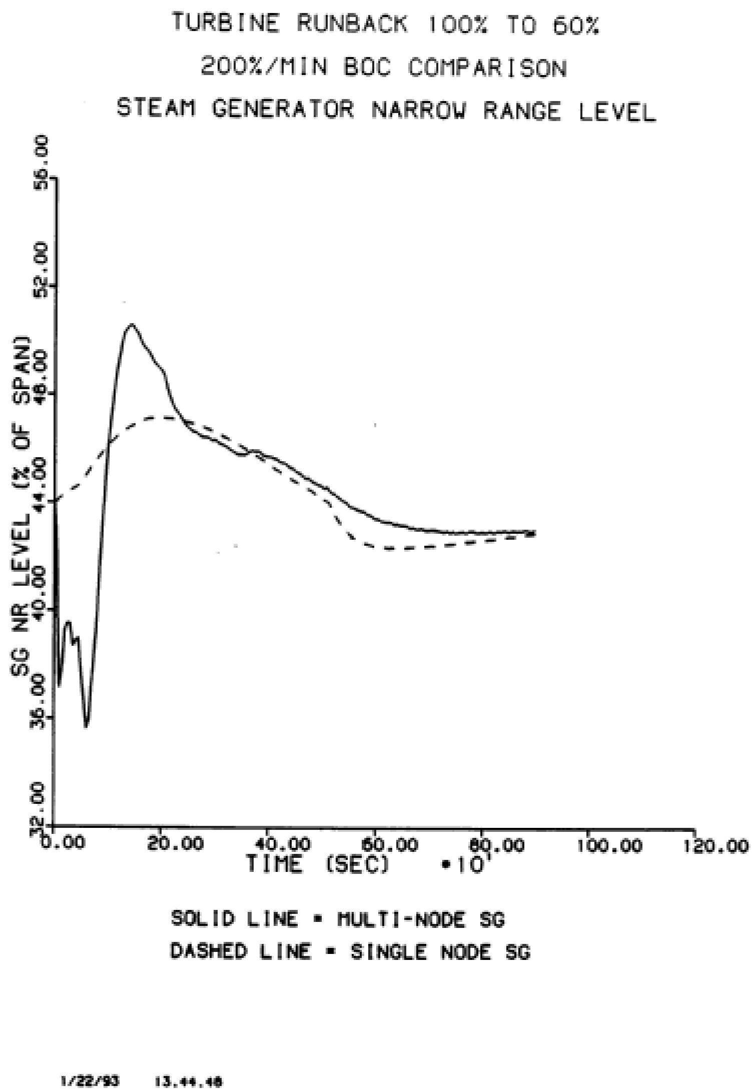

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
STEAM GENERATOR NARROW RANGE LEVEL
SG NR LEVEL (% OF SPAN)
56.00
52.00
48.00
44.00
40.00
36.00
32.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10¹
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 13.44.48

FIGURE 5.7-7

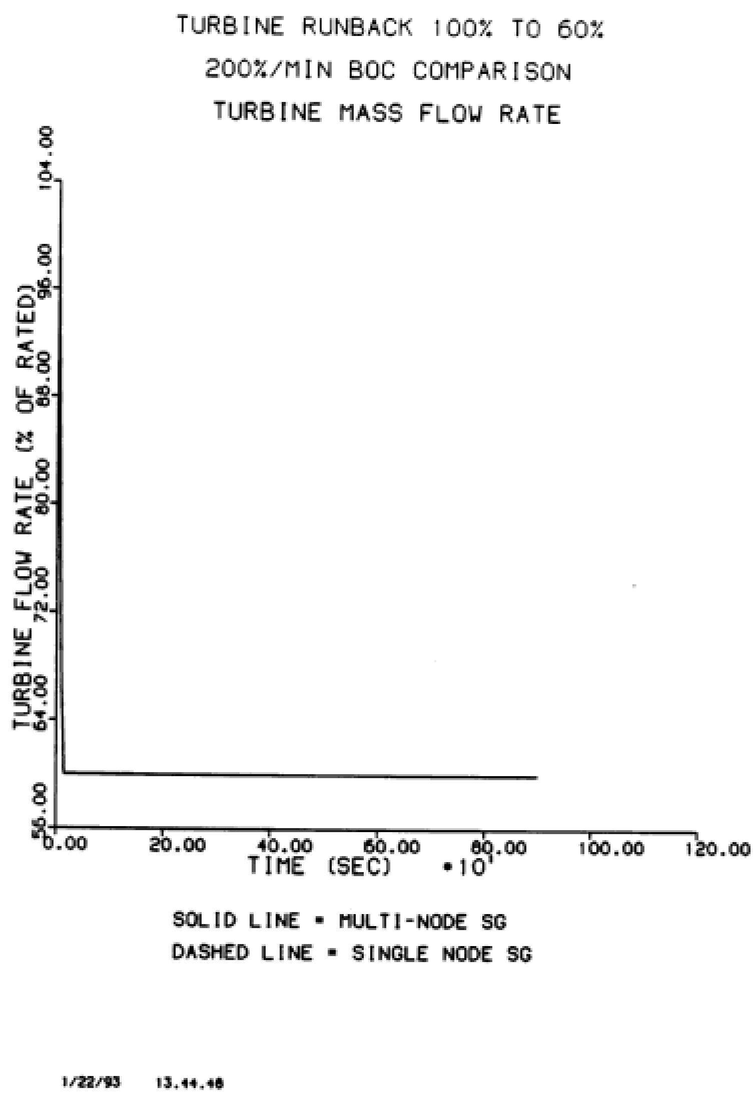

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
TURBINE MASS FLOW RATE
TURBINE FLOW RATE (% OF RATED)
56.00
64.00
72.00
80.00
88.00
96.00
104.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10[1]
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 13.44.48

FIGURE 5.7-8

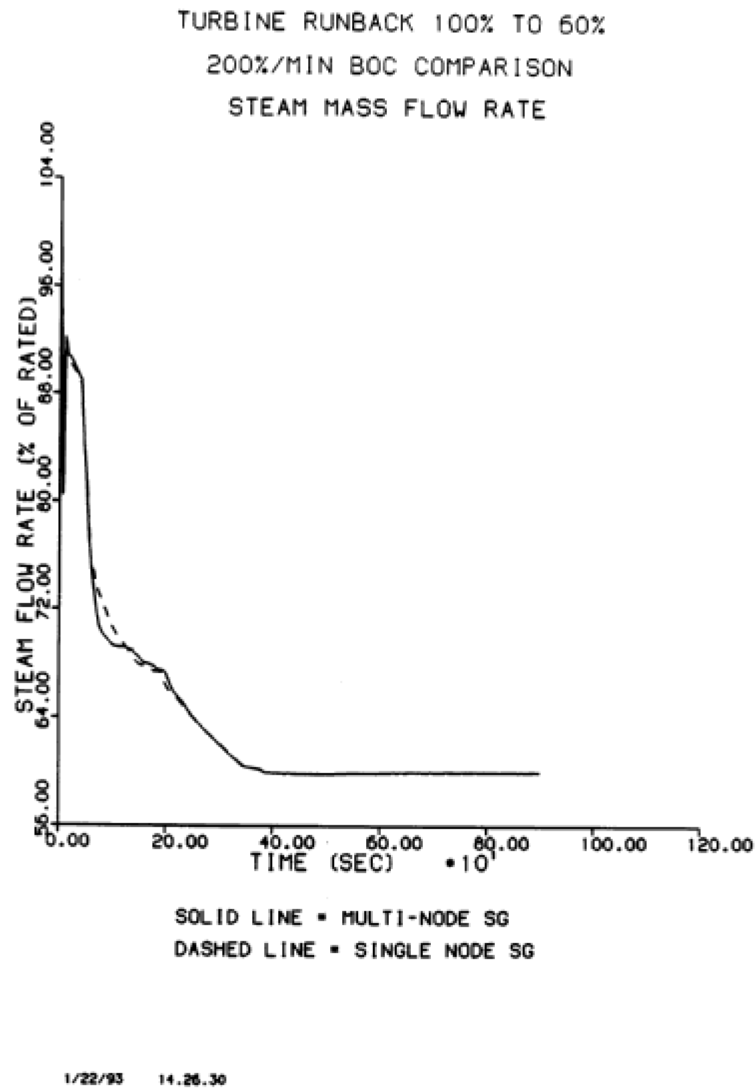

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
STEAM MASS FLOW RATE
STEAM FLOW RATE (% OF RATED)
56.00
64.00
72.00
80.00
88.00
96.00
104.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10[1]
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 14.26.30

FIGURE 5.7-9

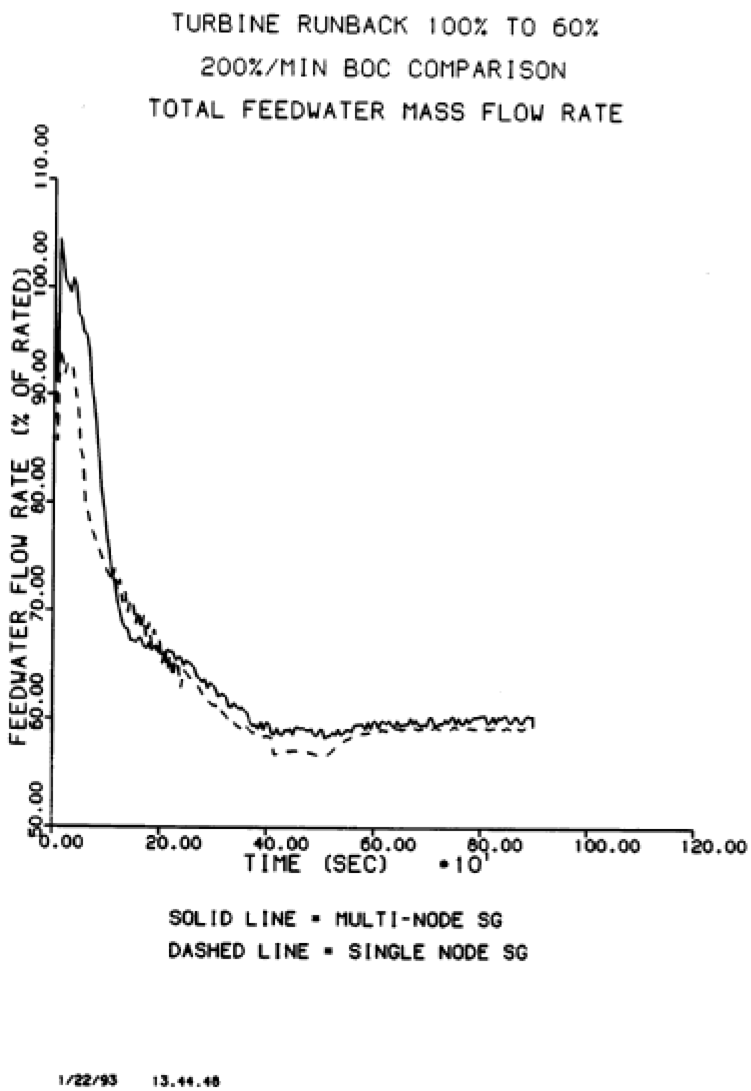

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
TOTAL FEEDWATER MASS FLOW RATE
FEEDWATER FLOW RATE (% OF RATED)
50.00
60.00
70.00
80.00
90.00
100.00
110.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10^1
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 13.44.46

FIGURE 5.7-10

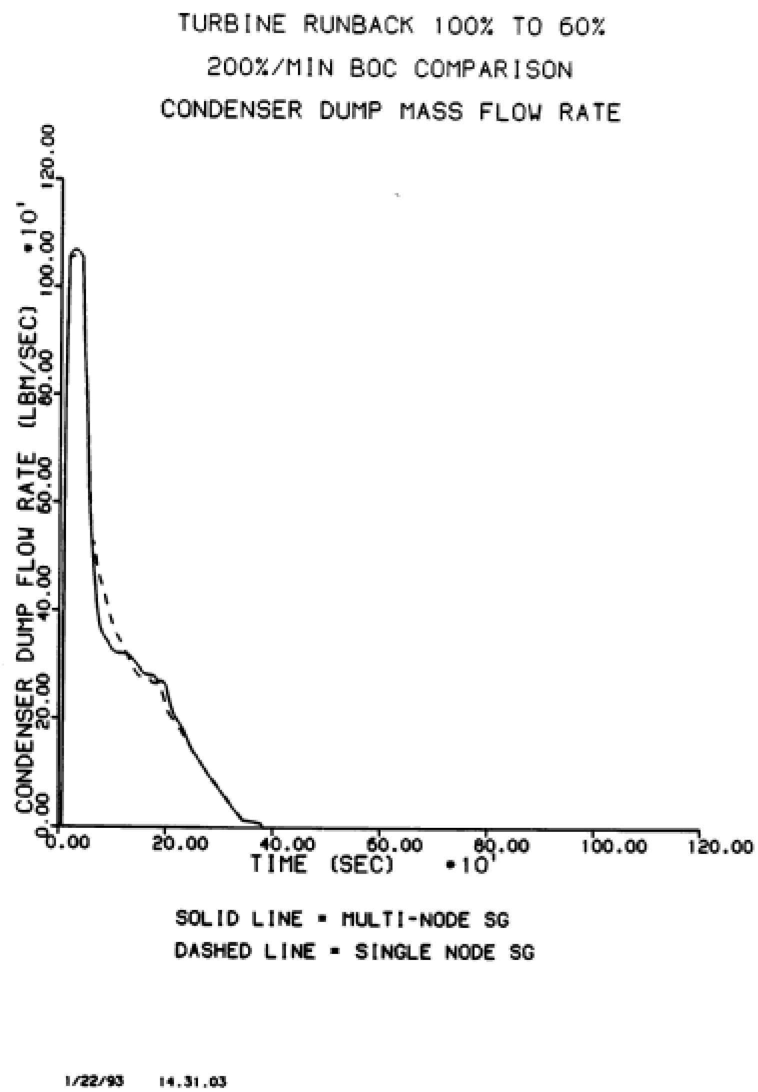

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
CONDENSER DUMP MASS FLOW RATE
CONDENSER DUMP FLOW RATE (LBM/SEC) *10¹
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10¹
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 14.31.03

FIGURE 5.7-11

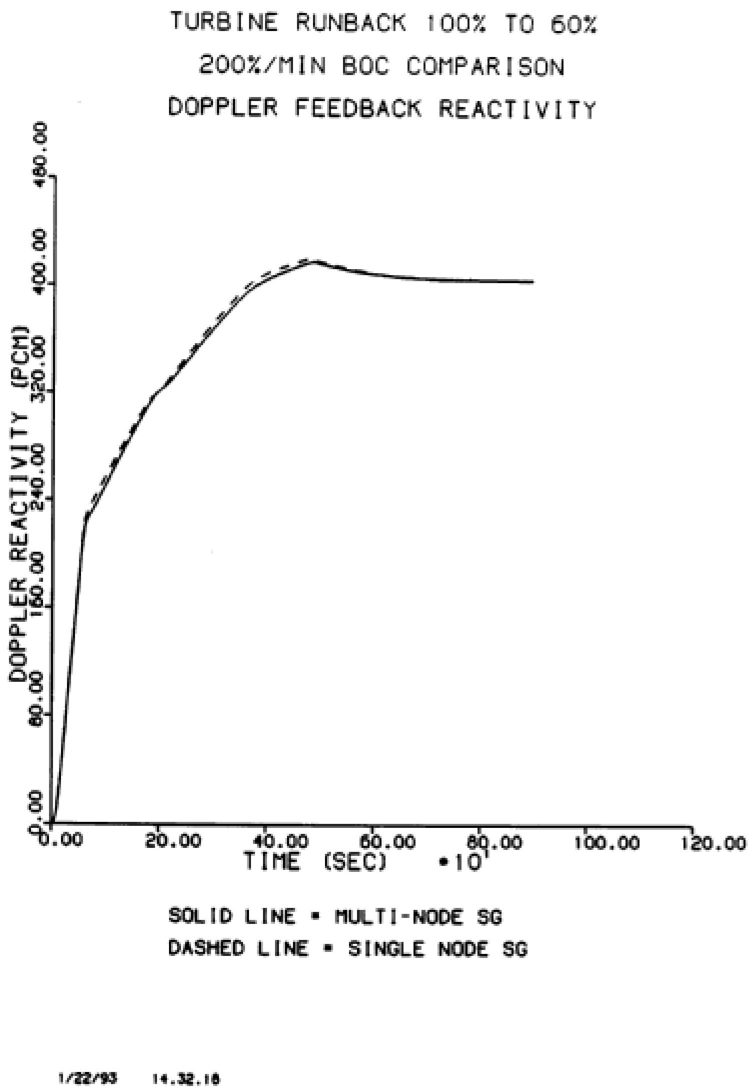

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
DOPPLER FEEDBACK REACTIVITY
DOPPLER REACTIVITY (PCM)
0.00
80.00
160.00
240.00
320.00
400.00
480.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10¹
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 14.32.18

FIGURE 5.7-12

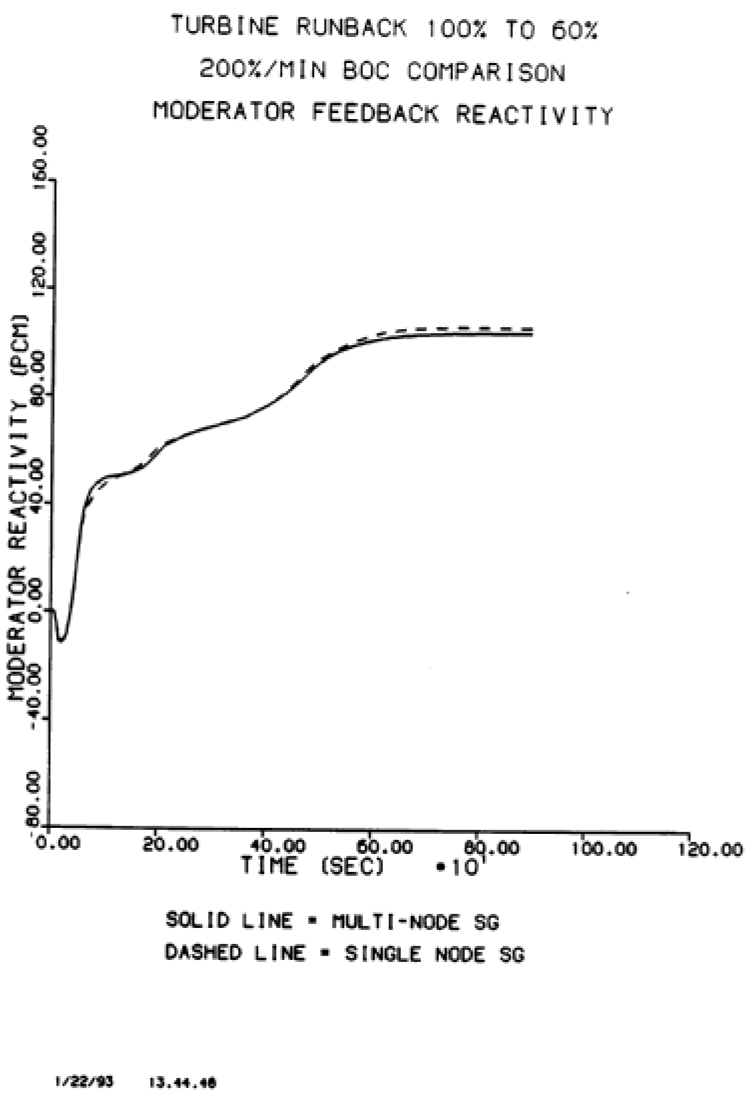
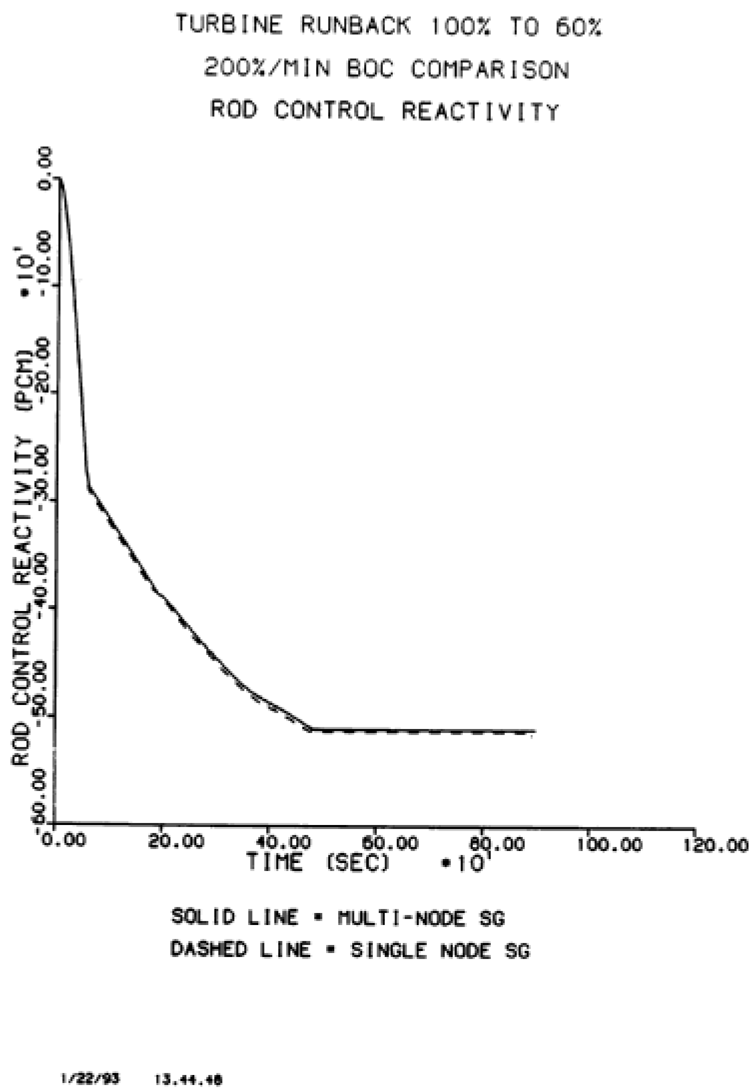

TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
MODERATOR FEEDBACK REACTIVITY
MODERATOR REACTIVITY (PCM)
150.00
120.00
80.00
40.00
0.00
-40.00
-80.00
0.00
20.00
40.00
60.00
80.00
100.00
120.00
TIME (SEC) *10
SOLID LINE = MULTI-NODE SG
DASHED LINE = SINGLE NODE SG
1/22/93 13.44.48

FIGURE 5.7-13


TURBINE RUNBACK 100% TO 60%
200%/MIN BOC COMPARISON
ROD CONTROL REACTIVITY
ROD CONTROL REACTIVITY (PCM) *10'
0.00
-10.00
-20.00
-30.00
-40.00
-50.00
-60.00
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
1/22/93 13.44.48

10) 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 (17).

Discussion

Refer to the response to Limitation 9, above.

14) 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.

17) 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.

<u>Discussion</u>
Refer to the response to Limitation 9, above.

24) 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.

<u>Discussion</u>
Dominion PWR RETRAN models use bubble rise in the pressurizer, reactor vessel upper head, and steam generator dome regions [Tables 4.2 and 4.4].

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.

28) 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 14, 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.

5.7-4 Letter from S. A. Richards (USNRC) to G. L. 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.

5.8 Main Steam System

5.8.1 Description

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 atmospheric steam relief valves are actually attached to the steam line, they may be modeled as connected to the steam generator. This generally 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

For plants containing steam line non return valves, the 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, low-low Tavg coincident with high steam flow, or other applicable ESF logic will initiate main steam line isolation. The logic requires the applicable coincidence from each of these signals; for example, two of the three loops must exceed their threshold.

5.8.1.5 Condenser Steam Dump System

Although small differences in condenser steam dump operation exist between the various plants modeled by Dominion Energy, in general 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. The level of detail and features modeled for the condenser steam dump system will vary according to the needs of and intended application for each plant.

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 may be modeled to 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, when modeled, 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 logic requires satisfaction of the applicable coincidence; for example, two of the three loops must exceed the threshold.

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

5.12.1 Description

Representative initiating functions for safety injection which are modeled are shown in Table 5.12-1. Only the high head or intermediate head safety injection pumps are modeled, consistent with plant design, as these are of primary interest for non-LOCA transient analyses. Representative 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 or intermediate 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.

The MPS3 base model explicitly models the SI accumulators. Standard RETRAN volumes are used to represent the accumulators.

5.12.2 Qualifications and Restrictions

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. These limitations were evaluated for RETRAN-3D/MOD003 in Reference 5.6-8 Section V. Those qualifications and restrictions that are applicable to the modeling of safety injection are discussed and evaluated in this section. The number designations for the qualifications and restrictions are those of the Safety Evaluation Report for RETRAN-3D/MOD003.

40. Organizations with NRC-approved RETRAN-02 methodologies can use the RETRAN3D code in the RETRAN-02 mode without additional NRC approval, provided that none of the new RETRAN-3D models listed in the definition are used. Organizations with NRC-approved RETRAN-02 methodologies must obtain NRC approval prior to applying any of the new RETRAN-3D models listed above for UFSAR Chapter 15 licensing basis applications.

Discussion:
The RETRAN-3D SER states *"A RETRAN-02 mode model must not use any of the new RETRAN-3D features such as:… accumulator model."* The NRC restriction refers to a specific accumulator model component within RETRAN-3D. Standard RETRAN volumes are used to represent the accumulators in the MPS3 base model, but the RETRAN-3D accumulator model is not used. MSLB is the only non-LOCA transient that could potentially actuate the accumulators.

TABLE 5.12-1
SI Initiation Functions:
[Representative Safety Analysis Setpoints]

Trip Signal	Setpoint	Delay 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-3D Code Report, EPRI NP-7450-CCM-A, Rev. 9). 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 the plant of interest and is described by the following equation:

$$DTC(pcm/{}^{\circ}F) = DTC_{Tf} * BURNMP * WF$$

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 values 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 RETRAN3D/MOD003 Safety Evaluation Report (Reference 5.13-11) Section 5 discussed general limitations of application of RETRAN-3D/MOD003. Those qualifications and restrictions that are applicable to the core kinetics model are discussed and evaluated in this section. The number designations for the qualifications and restrictions are those of the RETRAN-3D Safety Evaluation Report.

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

Discussion

RETRAN-3D/MOD3 includes a 3-dimensional nodal kinetics model. However, this model option is restricted from use in “02 mode”. 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).

2) There is no source term in the neutronics models and the maximum number of energy groups is two. The space-time options 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.

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

Discussion
A generalized boron transport model is present in RETRAN-3D/MOD3 [Reference 5.13-22]. 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.

<u>BIT to RCS Transport</u>

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.

<u>RCS Boron Transport</u>

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.

4) 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.

23) The subcooled void model is a nonmechanistic profile fit using a modification of EPRI recommendations 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 (MOD003) 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. These condtions of use were not explicitly discussed in the RETRAN-3D/MOD003 SER. As these conditions are based upon Dominion's current modeling techniques, justification for these condtions is provided based upon application with RETRAN-3D/MOD003. The numbering used for each condition is based on the RETRAN02/MOD005.0 Generic SER (Referecne 5.13-21).

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.

The ANS 1979 standard decay heat model was added as part of RETRAN-3D/MOD003. However, the RETRAN-3D SER does not explicitly state acceptance for use. Therefore, justification based on the condtion of use stated in the RETRAN-02/MOD005.0 Generic SER (Referne 5.13-21) is provided.

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 and subquently RETRAN-3D/MOD003 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. The discussion is specific to North Anna and Surry, but applies to MPS3 as MPS3 models Doppler reactivity feedback using the same approach.

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:

$DTC(pcm/^{\circ}F) = DTC_{Tf} * BURNMP * WF$

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-A, Revision 2, Minor Revision 2 “Reload Nuclear Design Methodology”, October 2017.

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.”

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.

5.13-22 EPRI NP-7450-CCM-A, “RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems,” Rev. 9, March 2014.

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 North Anna 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 North Anna 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.

4. The qualification set developed for application of VEP-FRD-41-P-A (Rev. 0.2) to MPS3 consisted of benchmarks to various vendor calculations published in the MPS3 FSAR. The analysis evaluated a wide range of transient phenomena and covered the spectrum of FSAR event types. See **Appendix 10.**

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 use 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 Energy) has developed the capability to perform system transient analyses with the RETRAN-3D computer code. The general code features have been discussed and a description of the North Anna, Surry, and Millstone Unit 3 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 includes the restrictions for the use of RETRAN-3D in an “02 mode”.

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 North Anna, Surry, and Millstone Unit 3 Power Stations.

This version of the Dominion RETRAN topical report has been designated VEP-FRD-41-NP-A, Revision 0, Minor Revision 3. The basis for retention of the -A designation is the approval of this report through the 10 CFR 50.59 process.

8 LIST OF ABBREVATIONS

AFW	Auxiliary feedwater
ANS	American Nuclear Society
ATWS	Anticipated transient without SCRAM
βeff	Effective delayed neutron fraction
BWR	Boiling water reactor
CB	Control block
COLR	Core Operating Limits Report
CSA	Channel statistical allowance (i.e. instrument uncertainty)
DNB	Departure from nucleate boiling
DTC	Doppler temperature coefficient
EHC	Electrohydraulic turbine control
EPRI	Electric Power Research Institute
ESF	Engineered safety features
FANP	Framatome ANP (fuel type)
FLB	Feedline Break
FSAR	Final Safety Analysis Report
FW	Feedwater
GL	Generic letter
HEM	Homogeneous equilibrium
HFP	Hot full power
HTC	Heat transfer coefficient
HZP	Hot zero power
I.C.	Initial condition(s)
I&C	Instrumentation and controls
LAR	License amendment request
LOCA	Loss-of-coolant accident
LOCROT	Locked rotor
LOL/TT	Loss of load / turbine trip
LONF	Loss of normal feedwater
lp	Prompt neutron lifetime
MS	Main steam
MSSV	Main steam safety valves
MSLB	Main steam line break
MPS3	Millstone Power Station Unit 3
NAPS	North Anna Power Station
NR	Narrow range
NRC	Nuclear Regulatory Commission
NSAL	Nuclear Safety Advisory Letter
OPΔT	Overpower delta-T
OTΔT	Overtemperature delta-T
pcm	percent milli-rho (1 pcm = 1.0x 10^{-5} $\delta k/k$ reactivity)
PORV	Power operated relief valve
PSV	Pressurizer safety valve
PWR	Pressurized water reactor
PZR	Pressurizer

RAI	Request for additional information
RCP	Reactor coolant pump
RCS	Reactor coolant system
RFA-2	Robust Fuel Assembly – 2 (fuel type)
RPS	Reactor protection system
RWAP	Rod withdrawal at power
Rx	Reactor
SBLOCA	Small break loss-of-coolant accident
SE	Safety evaluation
SER	Safety Evaluation Report
SG	Steam generator
SGTR	Steam generator tube rupture
SI	Safety injection
SPS	Surry Power Station
Tavg	RCS loop average or vessel average coolant temperature
TER	Technical Evaluation Report
Tin	Core inlet coolant temperature
TR	Technical Report
TRM	Technical Requirements Manual
Tref	Programmed reference temperature
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VEPCO	Virginia Electric and Power Company
V&V	Validation and verification

APPENDIX 1
VEP-FRD-41-A 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

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:

Roger W. Cross

R. W. Cross
Supervisor, Nuclear Fuel Engineering

APPROVED:

R. M. Berryman

R. M. Berryman
Director, Nuclear Fuel Engineering

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

April 11, 1985

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

Dear Mr. Stewart:

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

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

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

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

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

Sincerely,

Cecil O. Thomas

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

Enclosure:
As stated

ENCLOSURE

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

1. Introduction

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

2. VEPCO NSSS Models

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

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

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

3. 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. <u>Qualification Comparisons</u>

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, 1985, submitted a supplement to VEP-FDR-41 which compared parallel calculations of RETRAN and LOFTRAN performed by VEPCO. The LOFTRAN code is an NRC approved analytical program developed and maintained by the Westinghouse Electric Corporation for use in performing general non-LOCA transient and accident analyses. VEPCO has obtained access to LOFTRAN via a special licensing agreement with Westinghouse. The comparisons were performed with a LOFTRAN model of the Surry plant assembled by VEPCO applying the same data base used for developing the VEPCO RETRAN models. Thus the basic plant geometric and thermal parameters are consistent for the two models. The following transients were calculated and compared using both computer models: 1) Reactor trip from hot full power followed by a turbine trip, 2) Turbine trip from hot full power. No credit taken for direct reactor trip on the turbine trip, and 3) Simulataneous trip of all three reactor coolant pumps at hot full power. No credit taken for reactor trip on pump under voltage or under frequency. The results of these analyses confirmed that the VEPCO RETRAN models could produce compatible analysis results with that from the LOFTRAN models.

5. Conclusions

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

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

CLASSIFICATION/DISCLAIMER

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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.

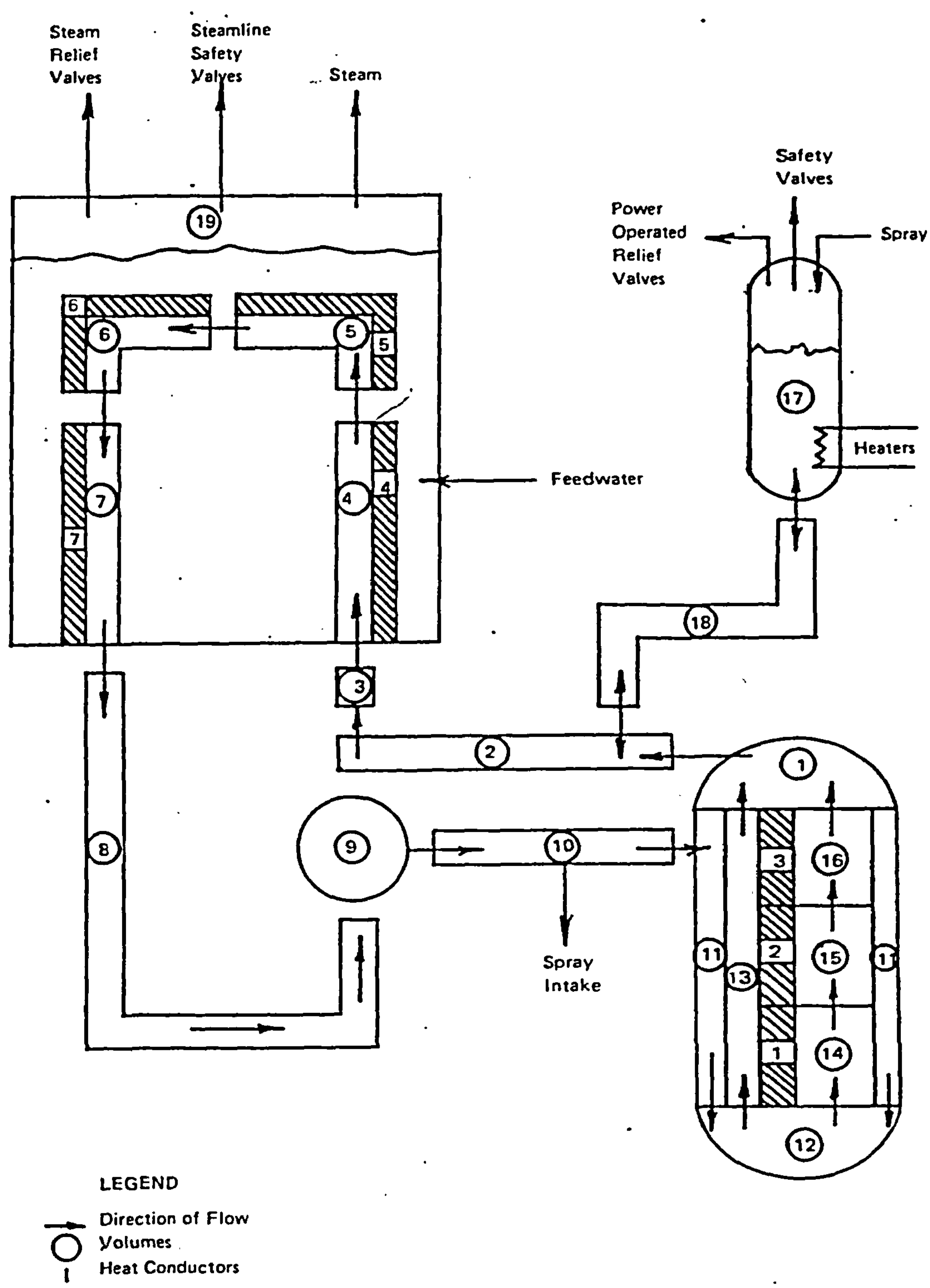
The Single Loop Model also incorporates the RETRAN control system capability to model the following NSSS control and protection features:

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

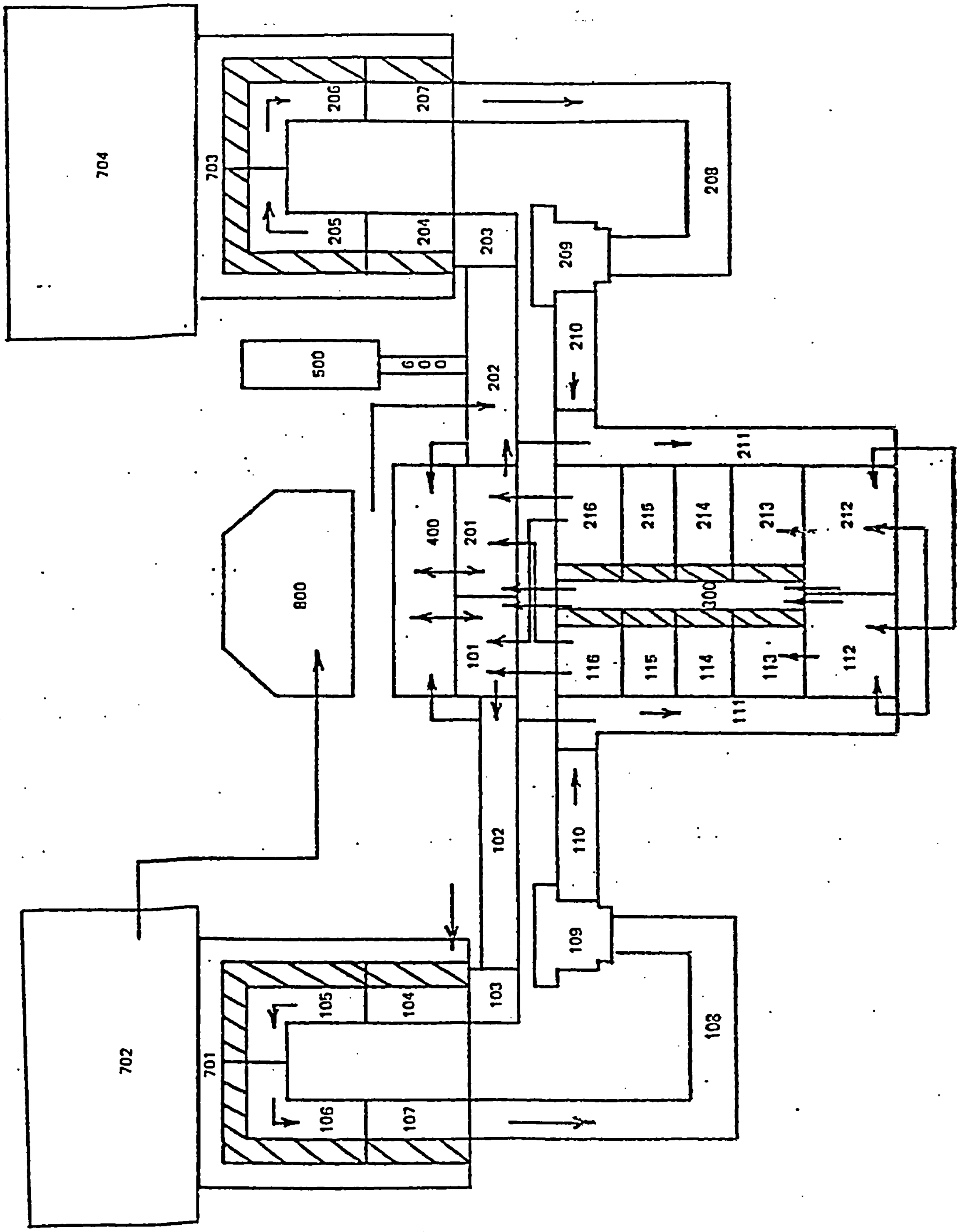


Table 3.1

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	± 4 $^{\circ}$F (Surry) allowance for deadband and measurement error.
c)	Pressurizer pressure	± 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

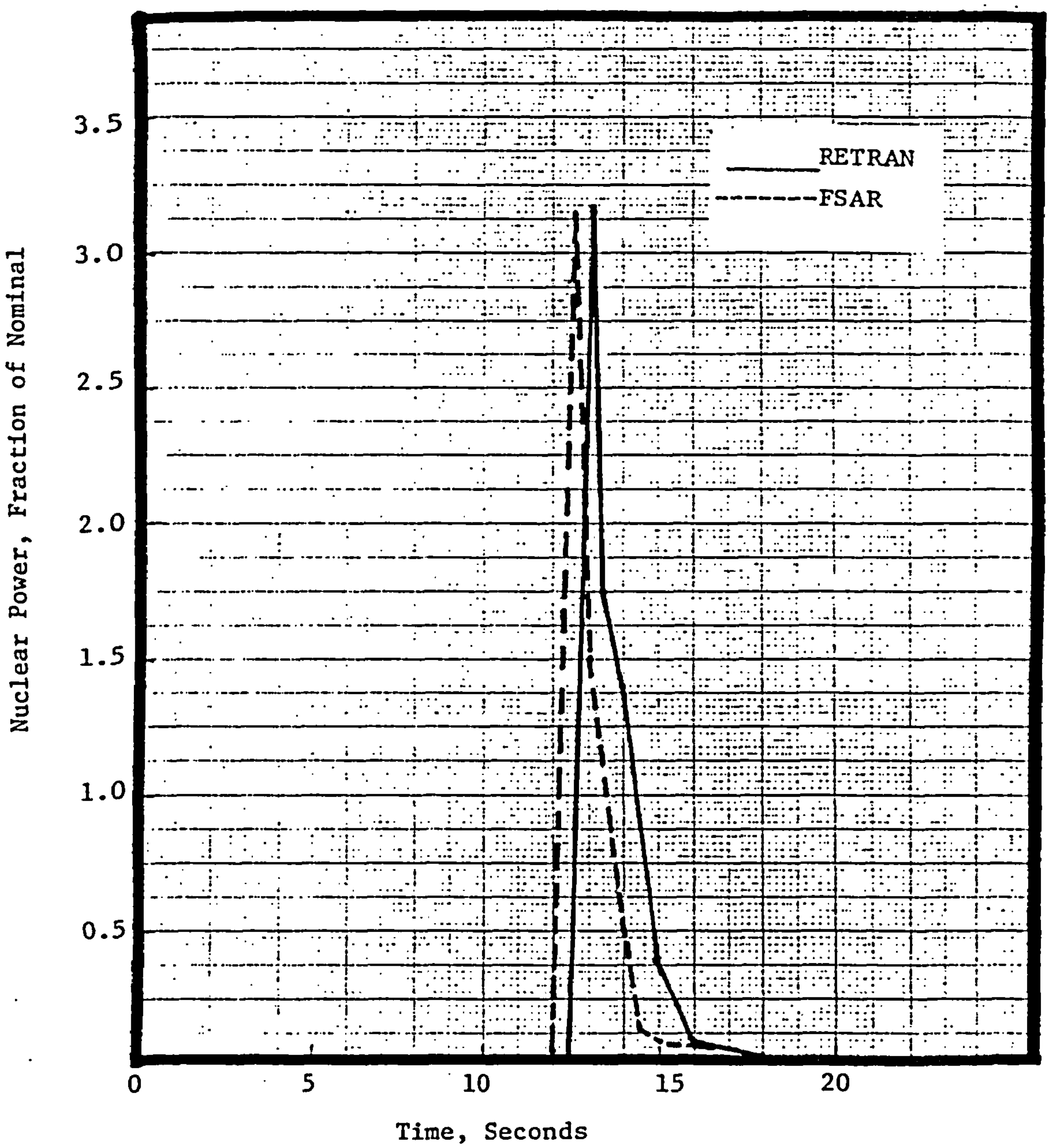


Figure 5.2

AVERAGE FUEL TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

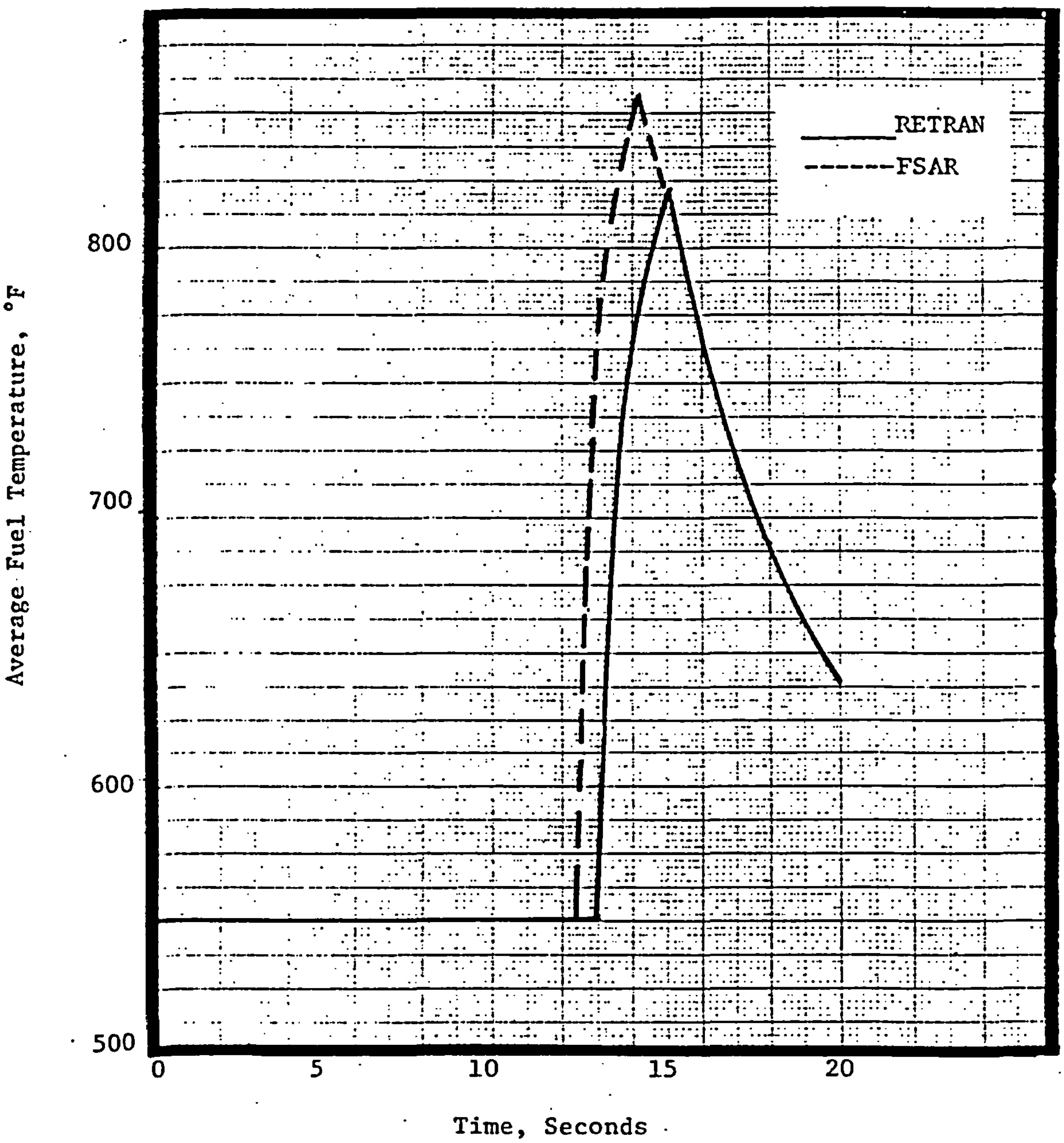


Figure 5.3

AVERAGE CLAD TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

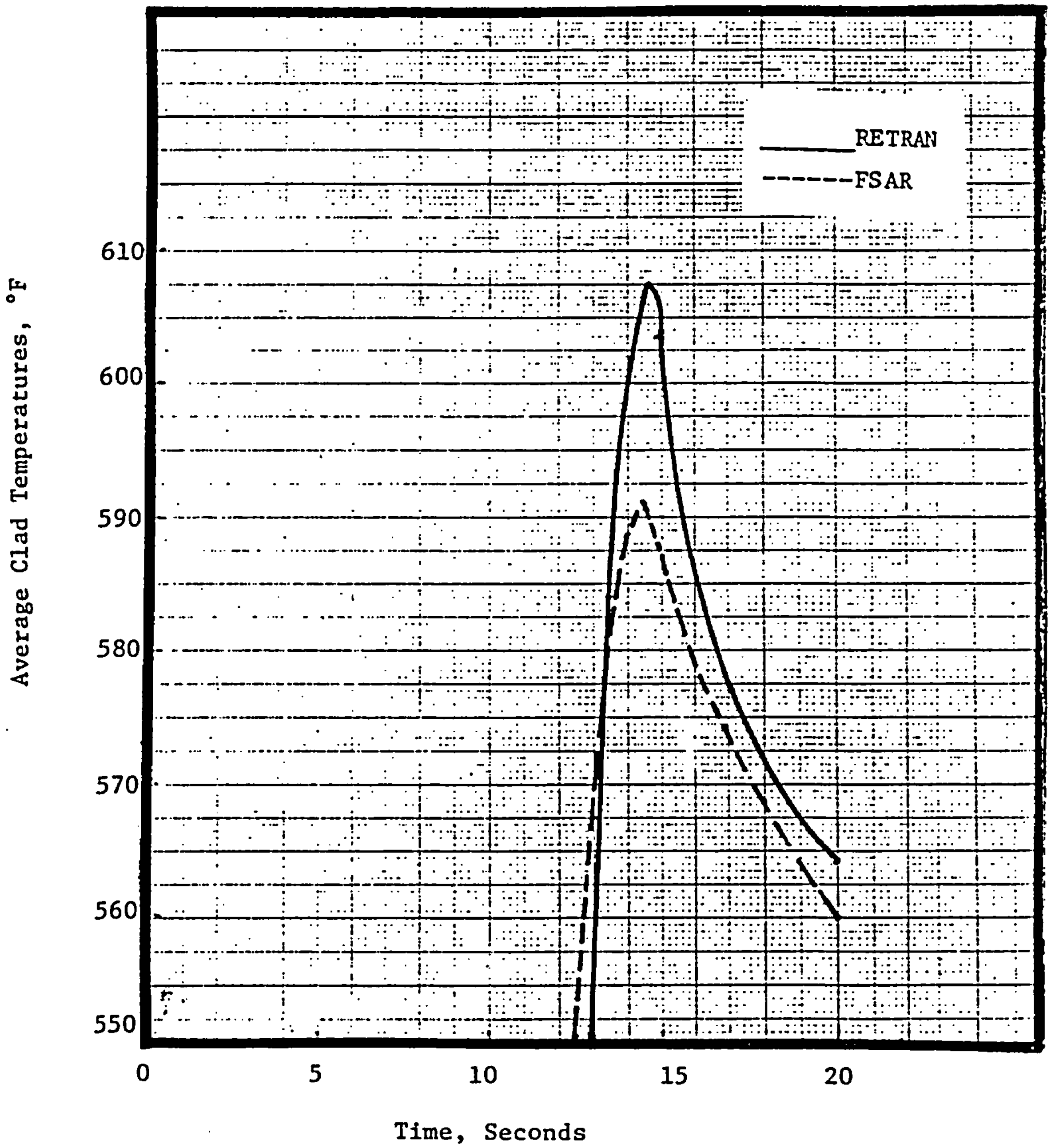


Figure 5.4

CORE HEAT FLUX
UNCONTROLLED ROD WITHDRAWAL FROM SUBCRITICAL TRANSIENT
FSAR ANALYSIS

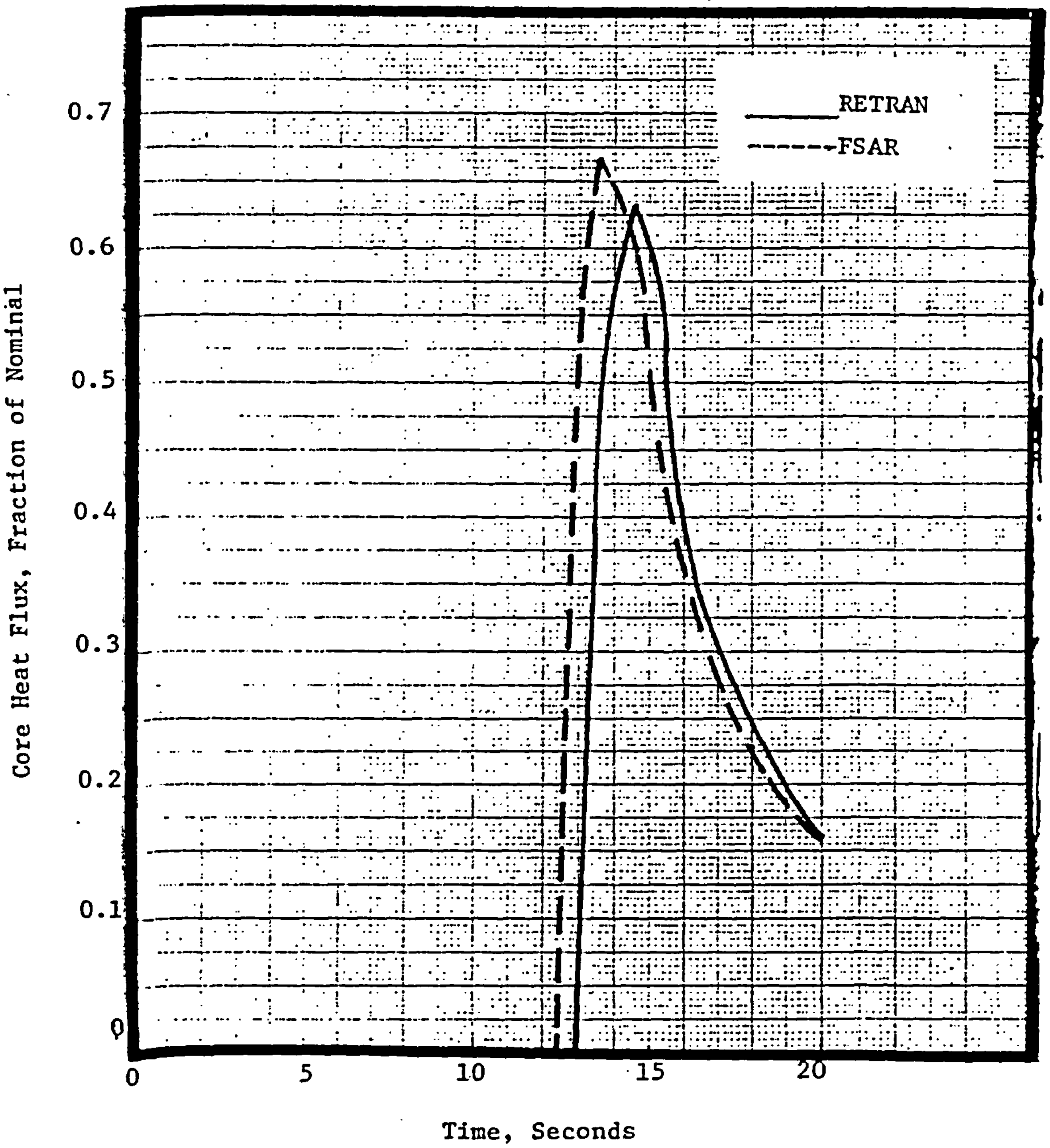


Figure 5.5

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

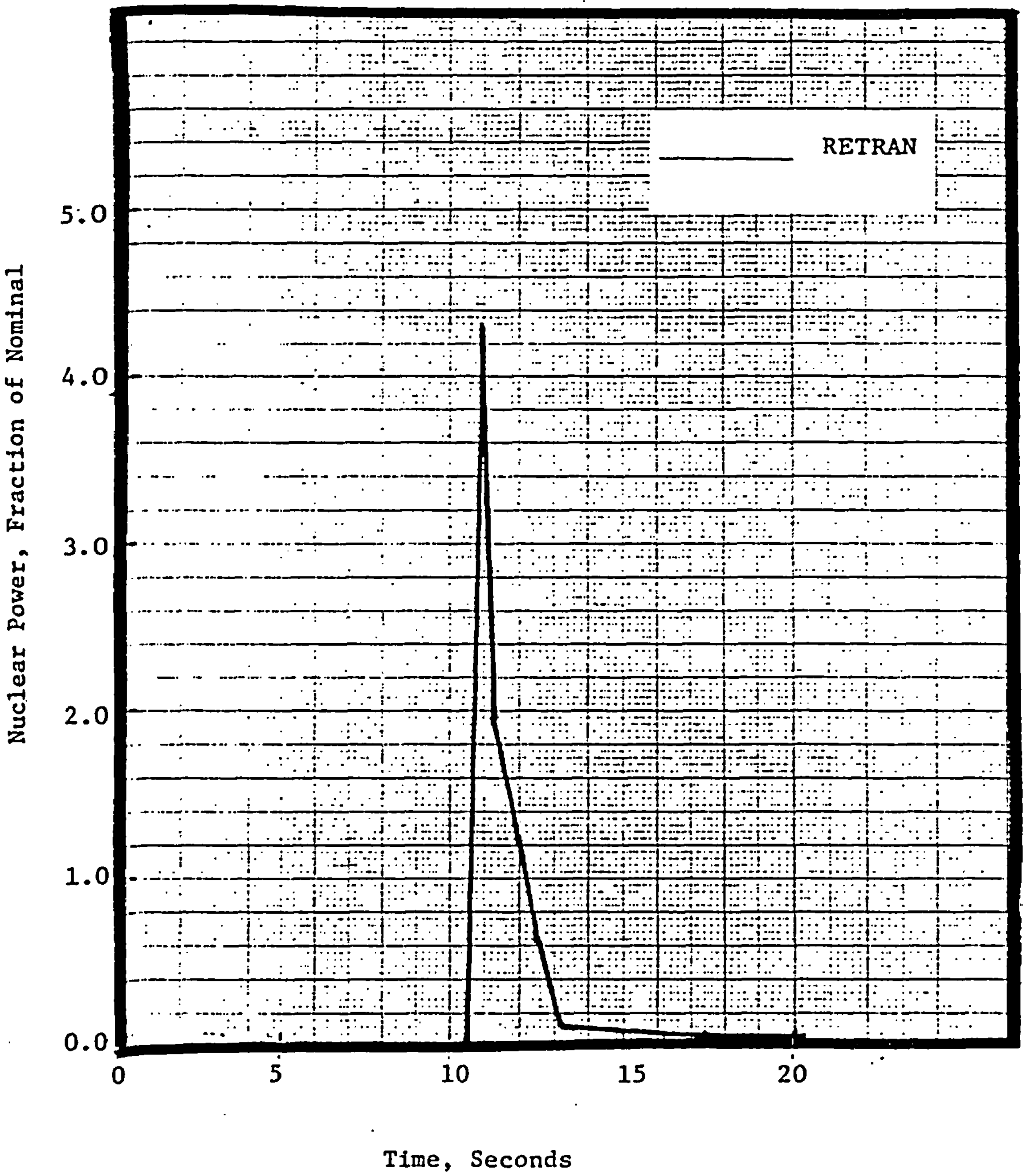


Figure 5.6

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

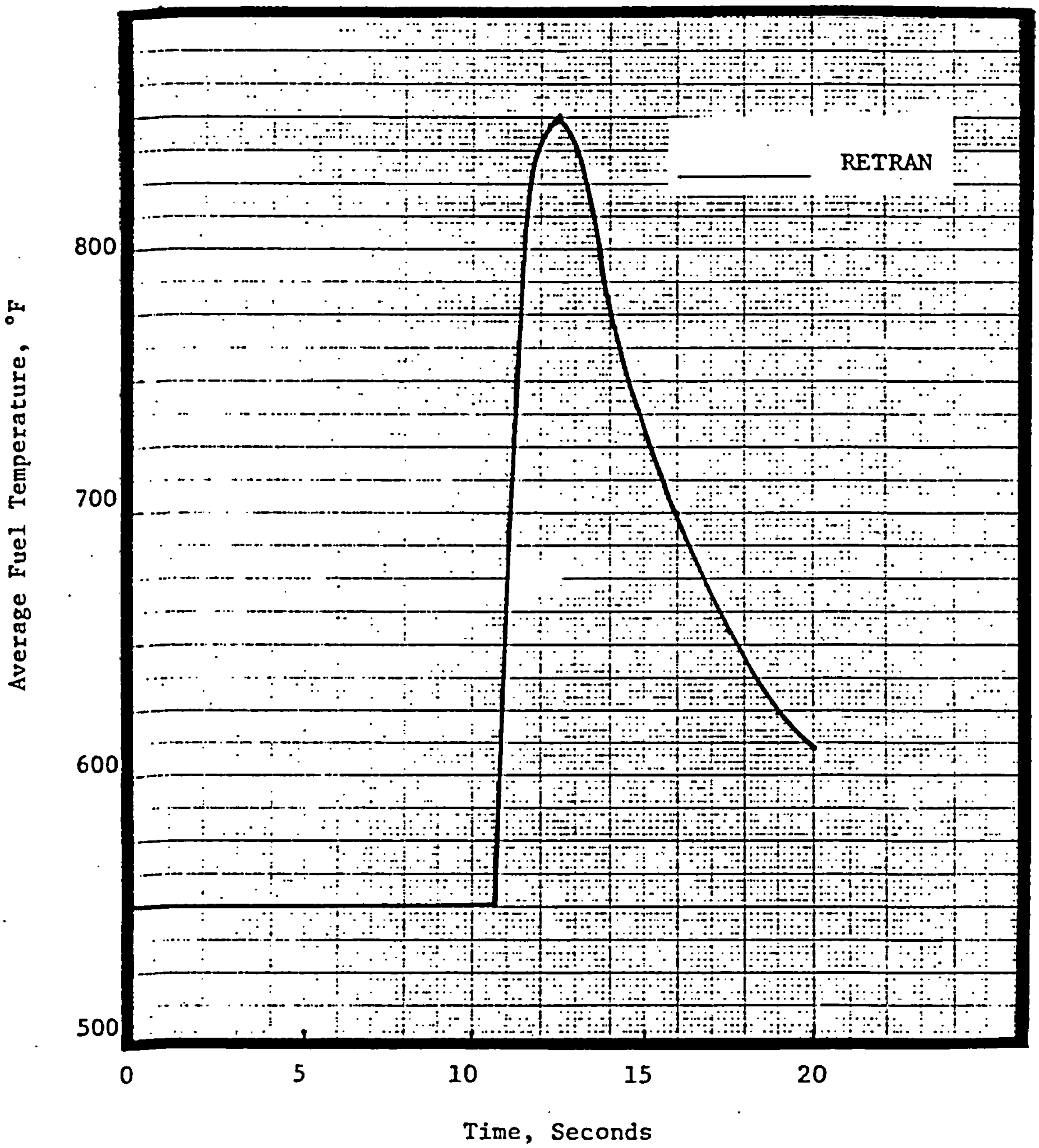


Figure 5.7

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

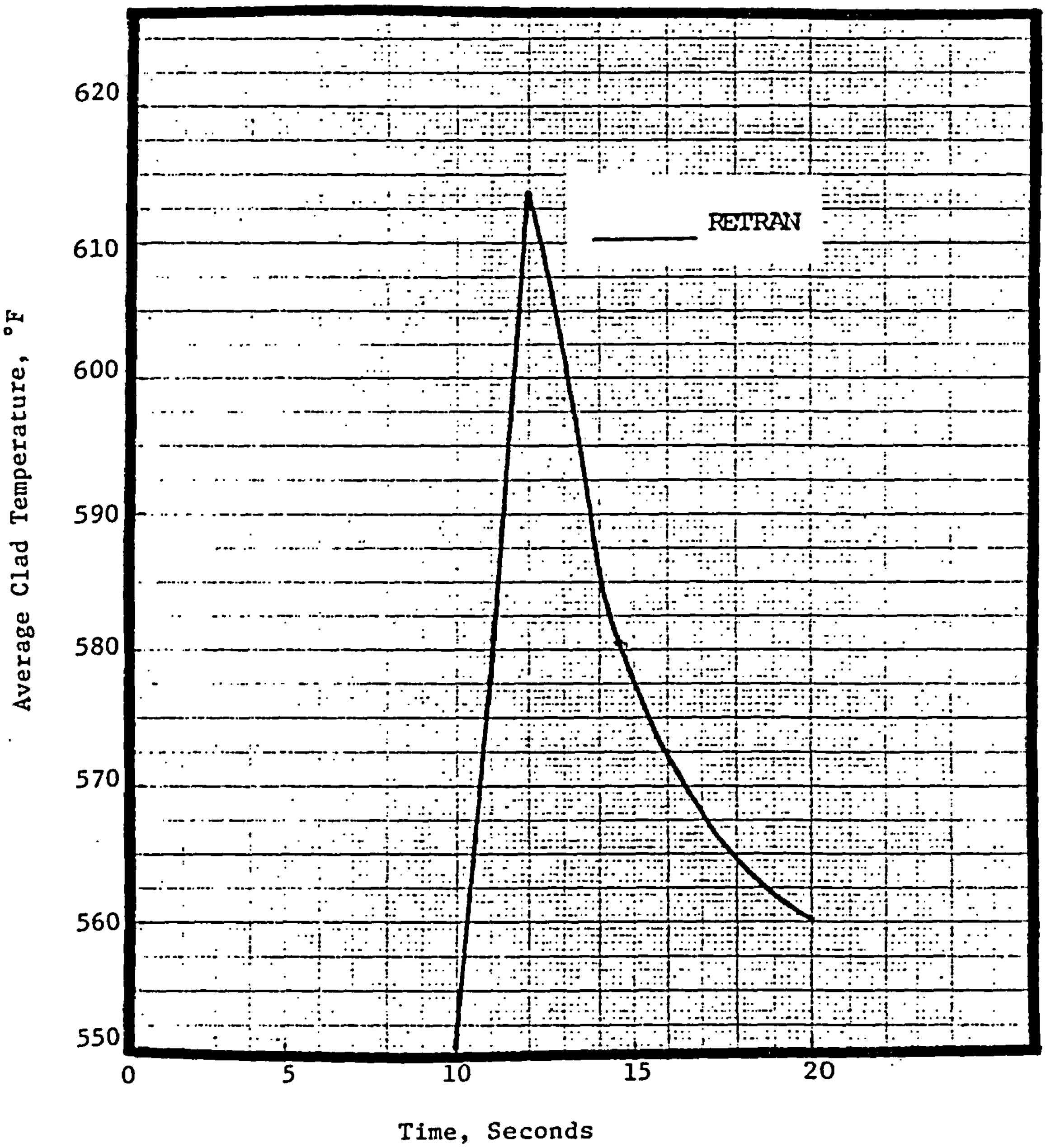


Figure 5.8

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

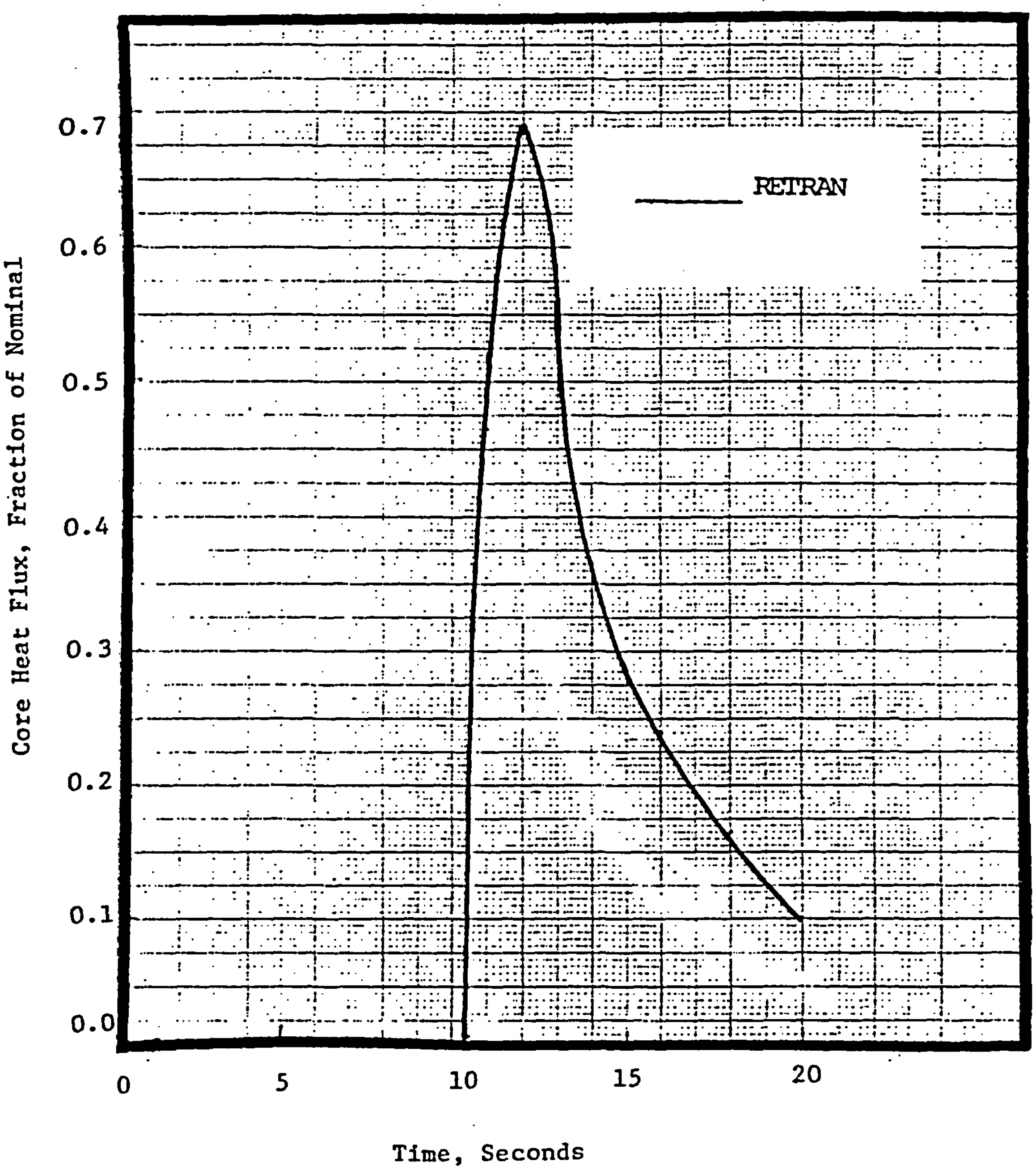


Figure 5.9
NUCLEAR POWER
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

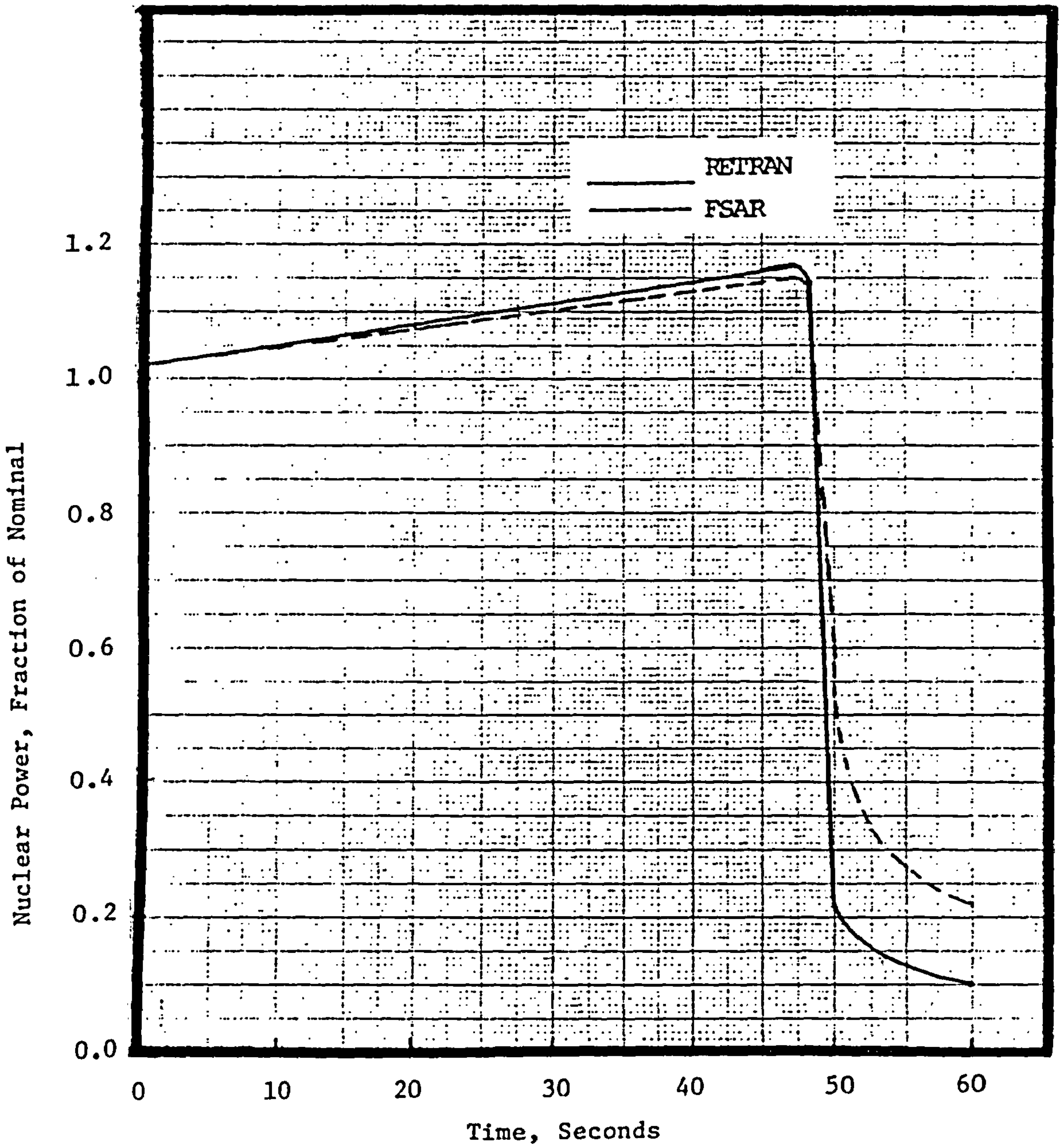


Figure 5.10
PRESSURIZER PRESSURE
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

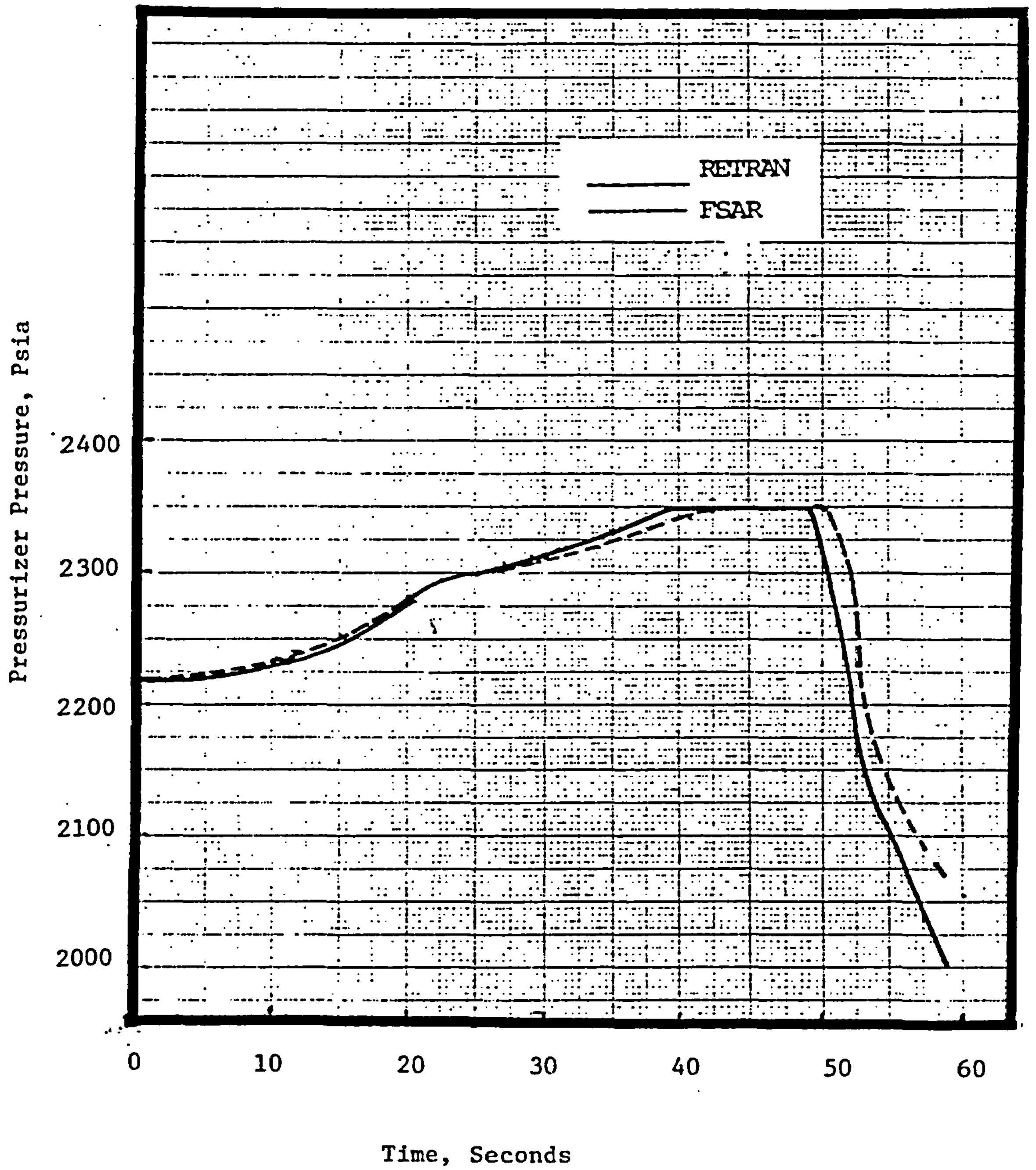


Figure 5.11

AVERAGE COOLANT TEMPERATURE
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

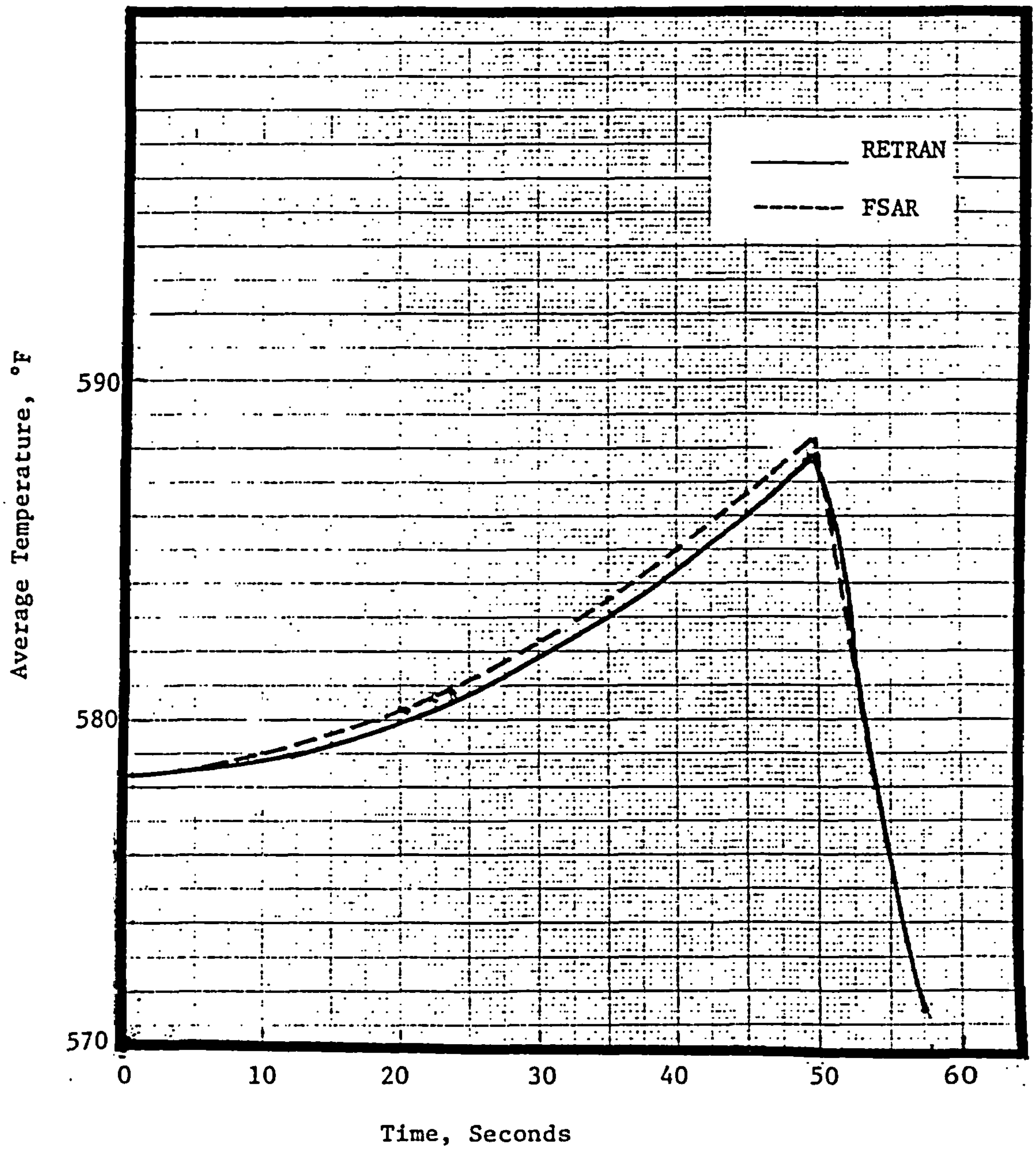


Figure 5.12

DNB RATIO
UNCONTROLLED ROD WITHDRAWAL FROM POWER TRANSIENT
FSAR ANALYSIS

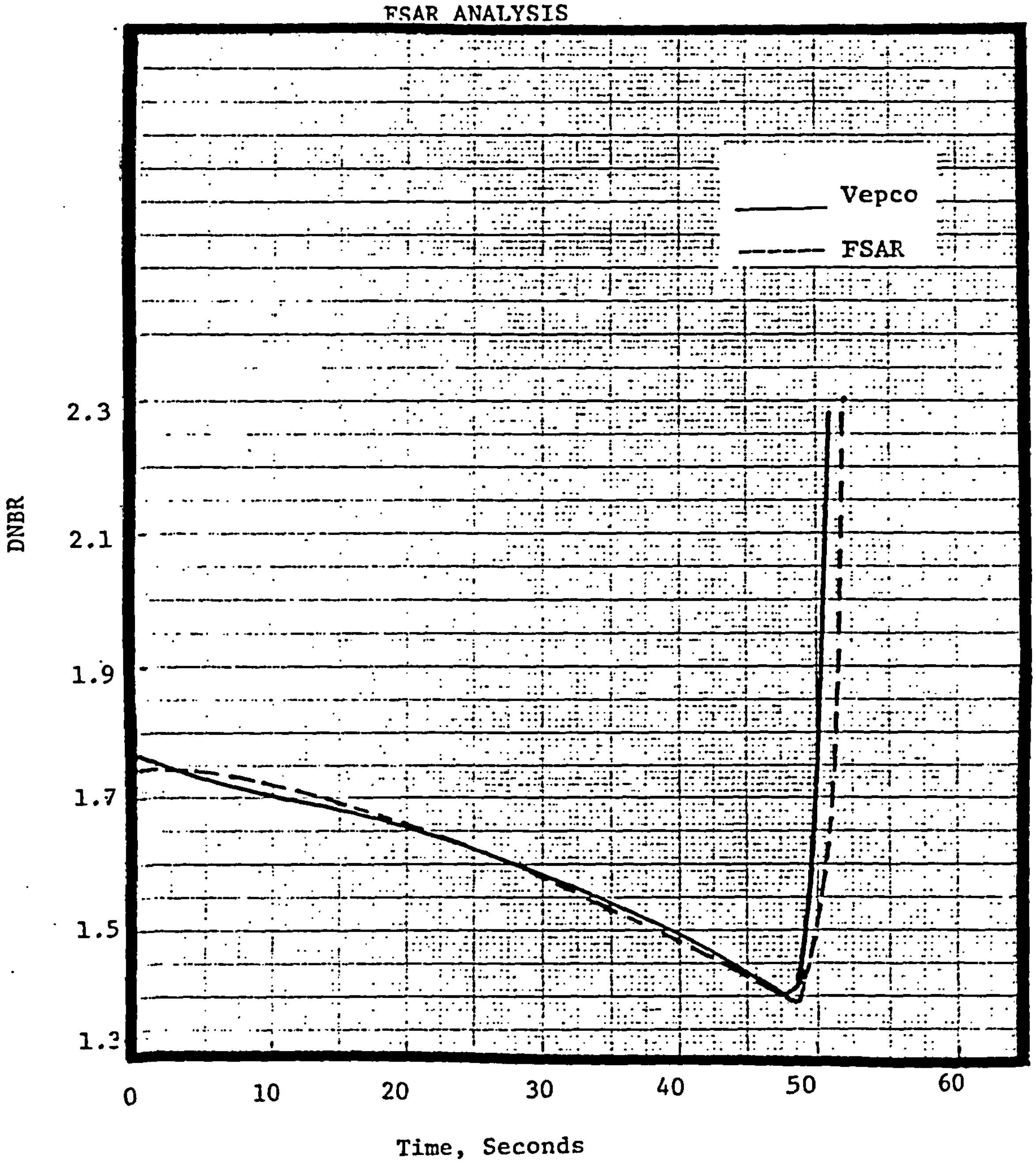


Figure 5.13

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

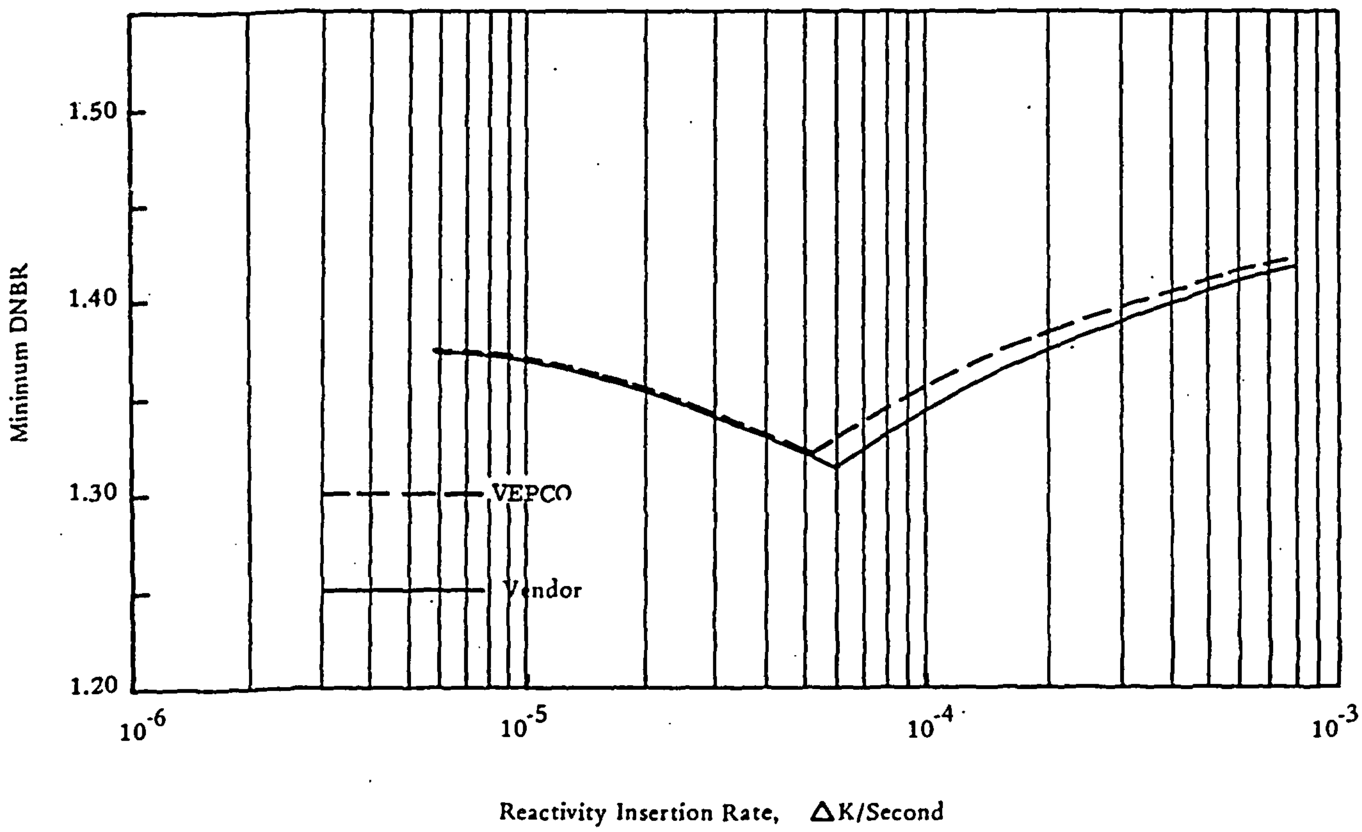
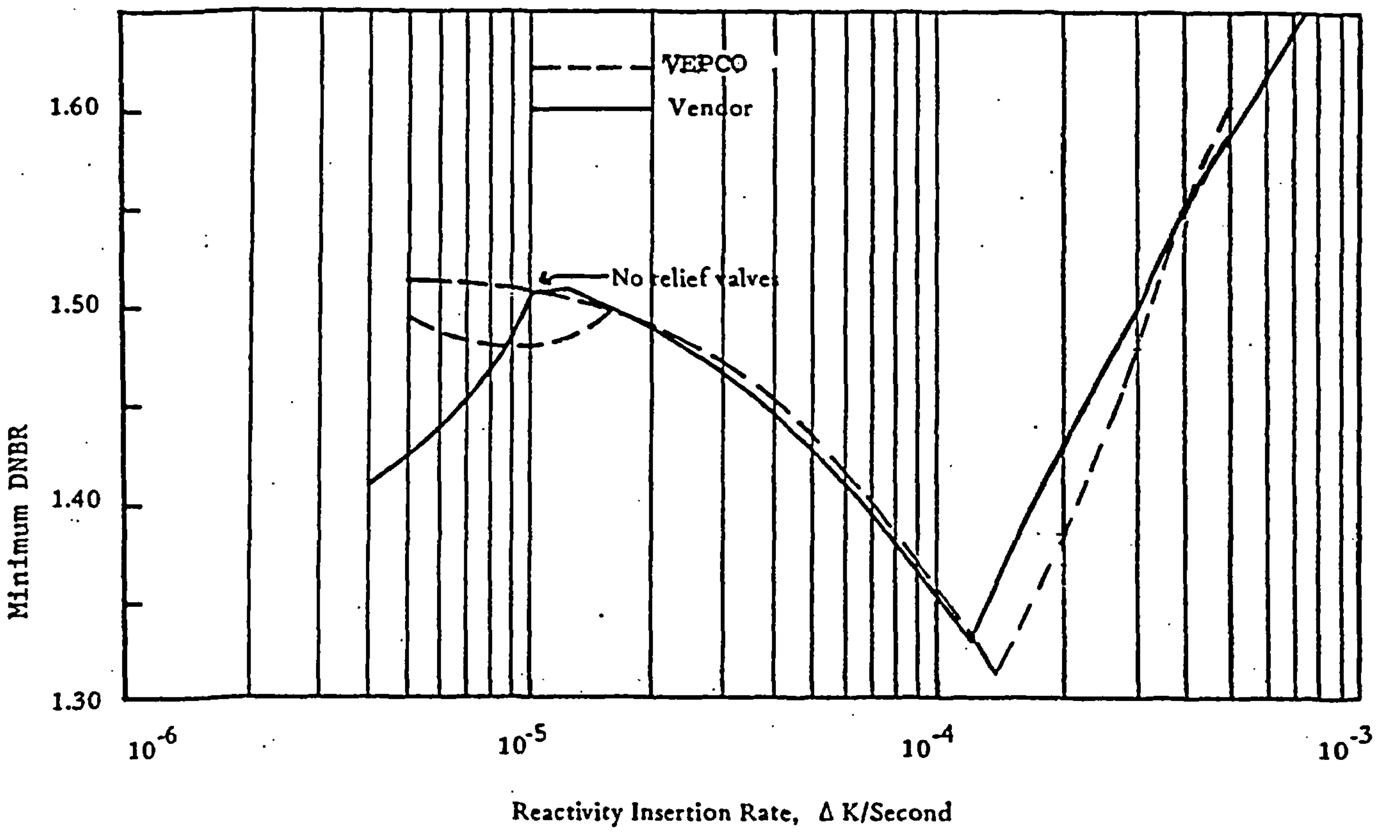


Figure 5.14

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

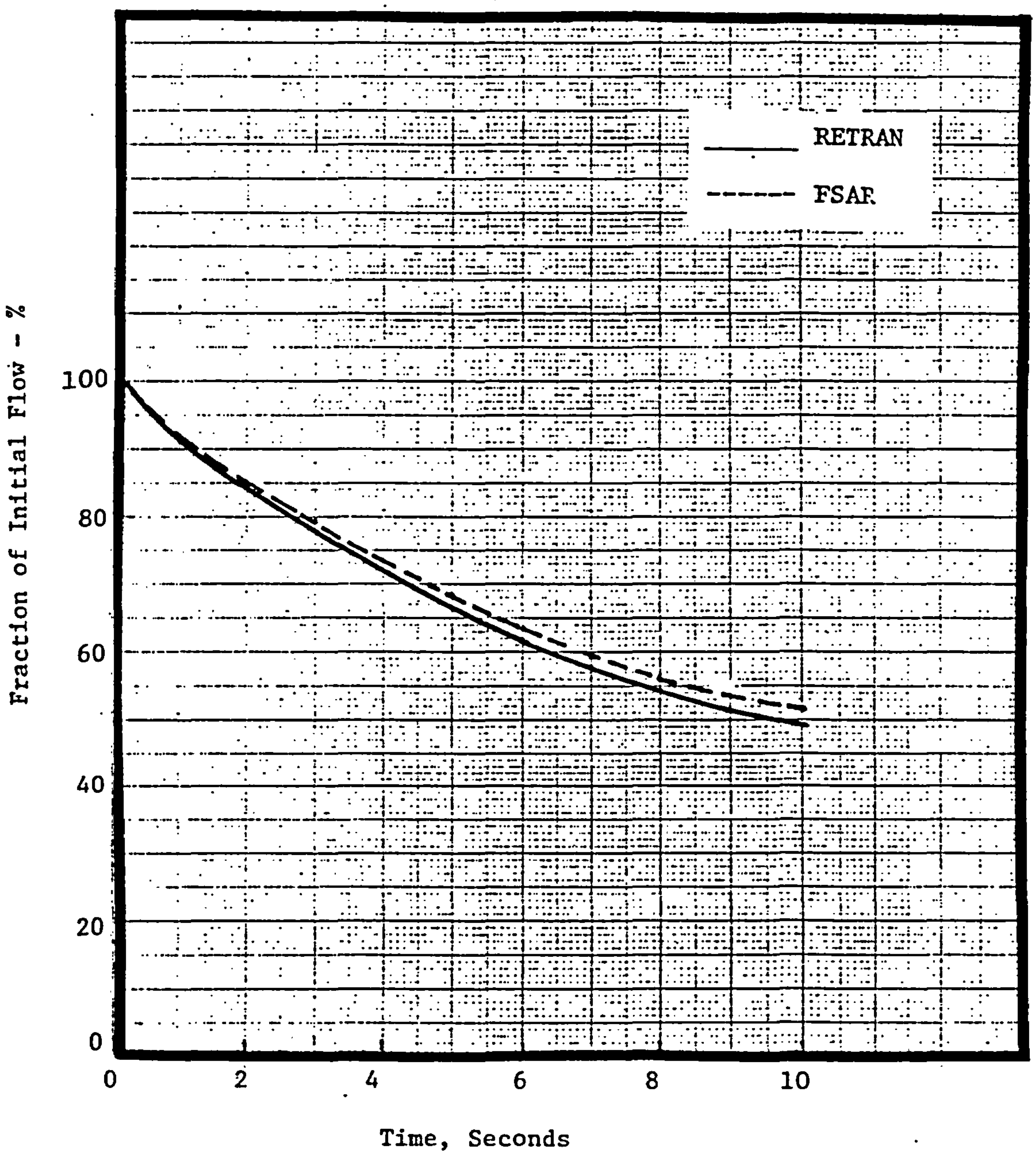


Figure 5.16

NUCLEAR-POWER
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

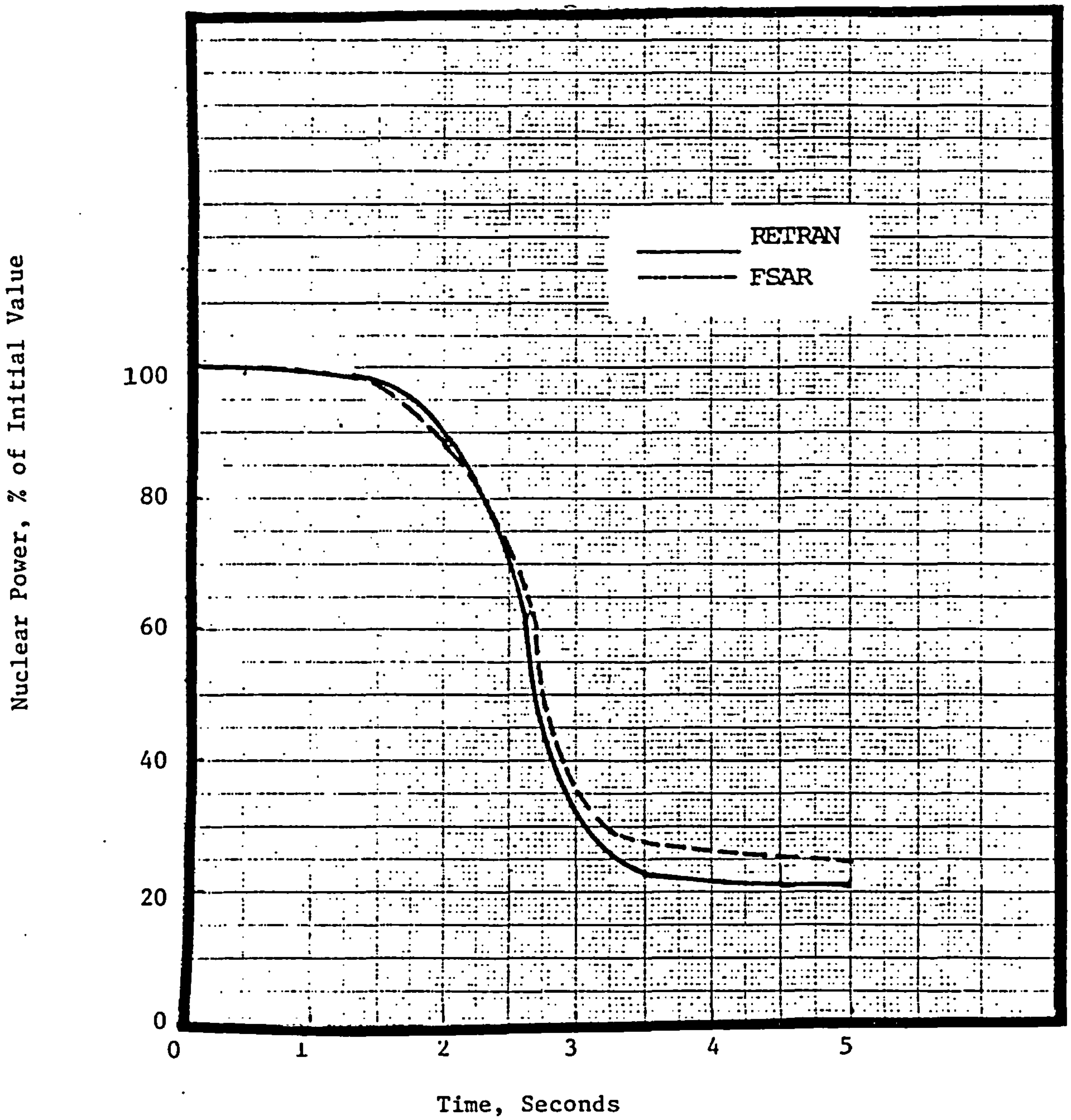


Figure 5.17

AVERAGE HEAT FLUX
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

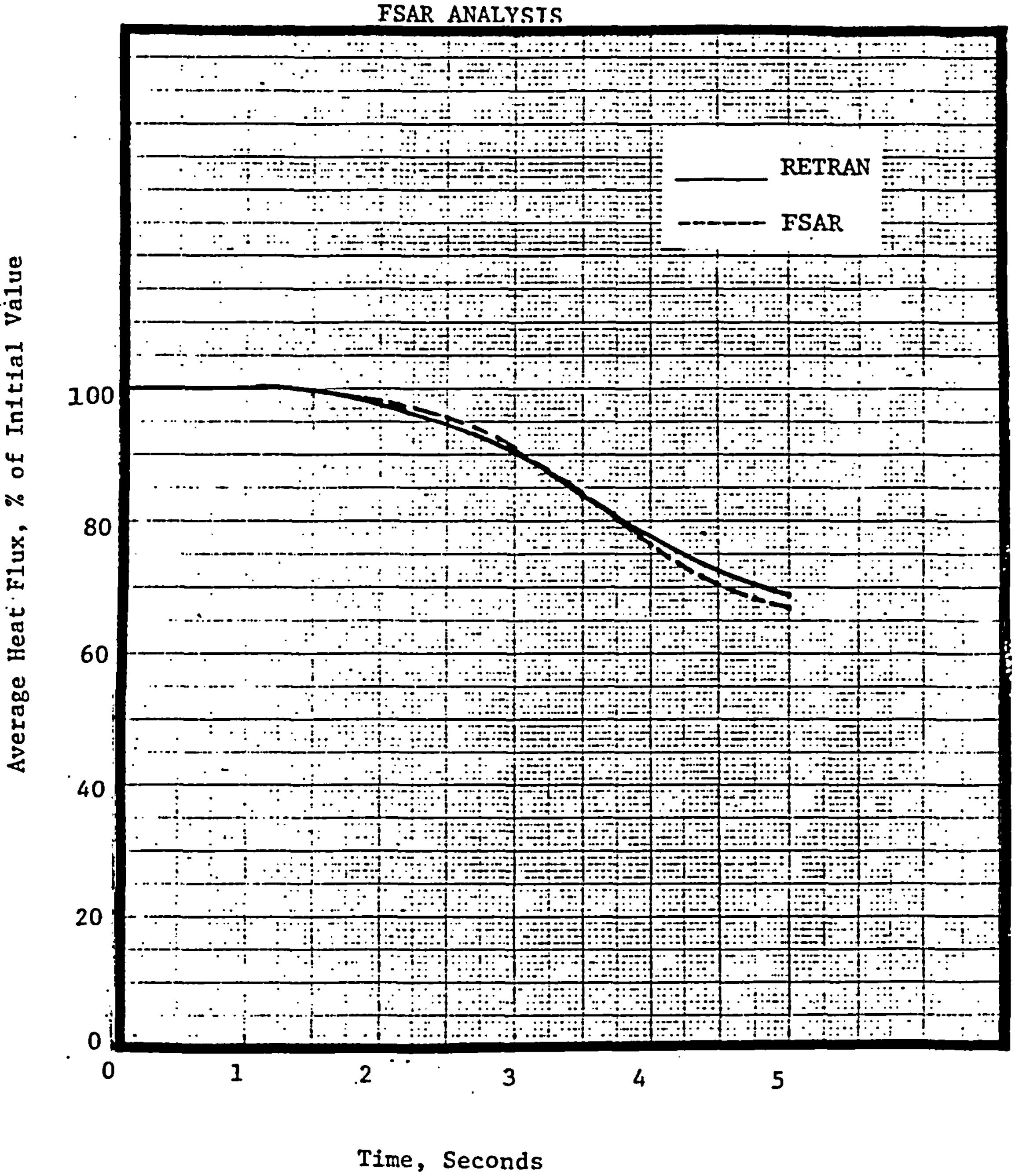


Figure 5.18

DNB RATIO
COMPLETE LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

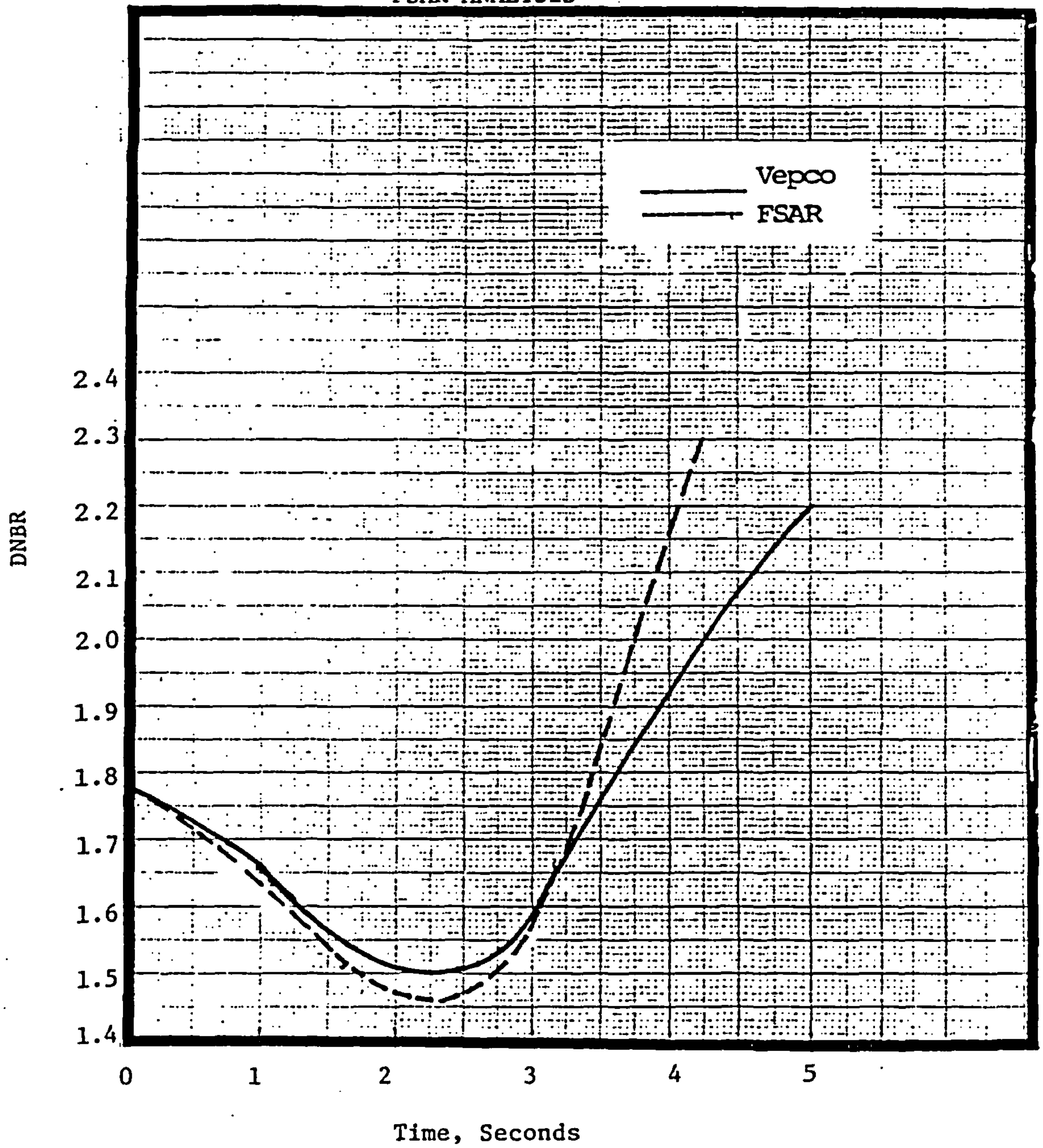


Figure 5.19

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

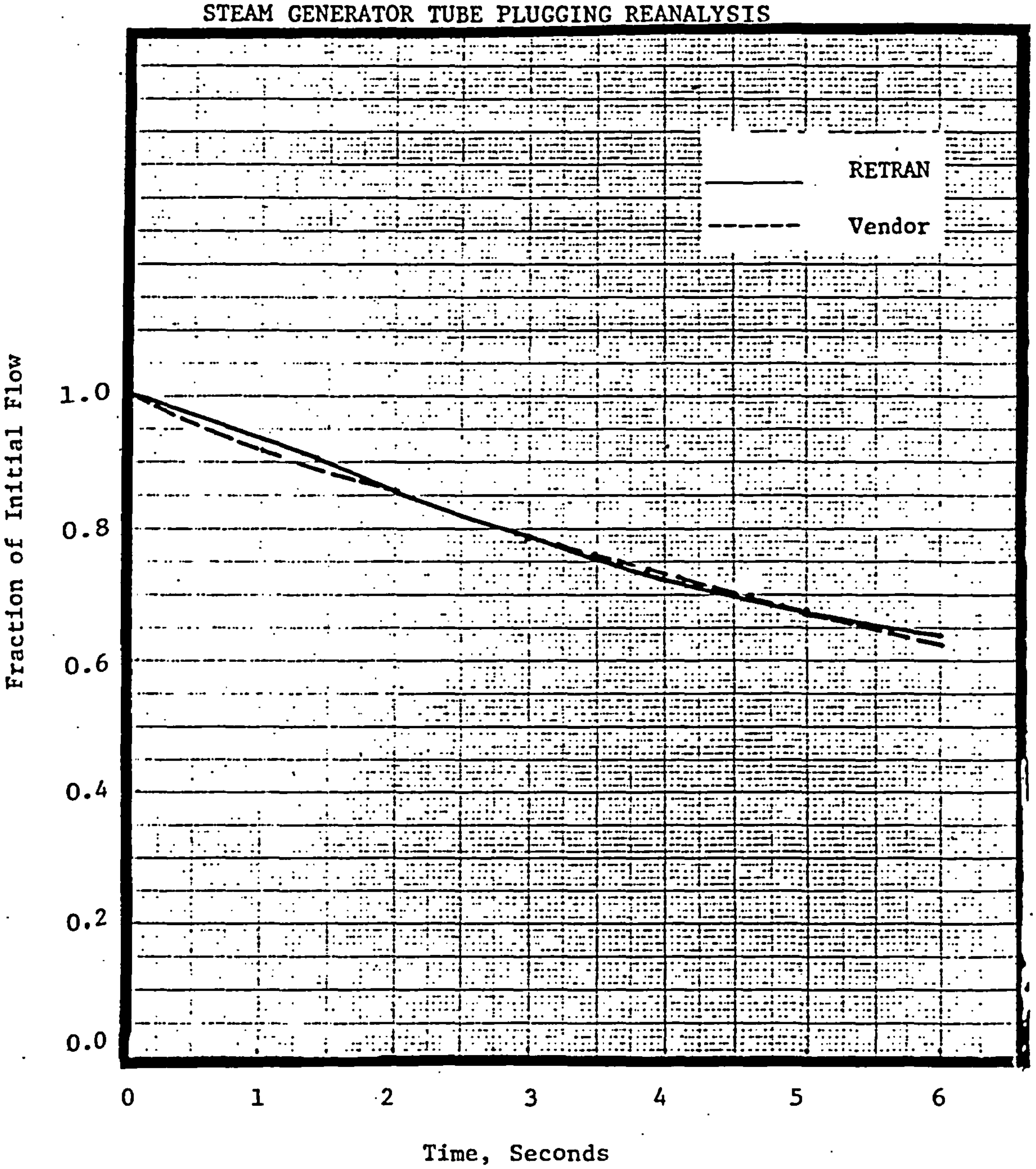


Figure 5.20

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

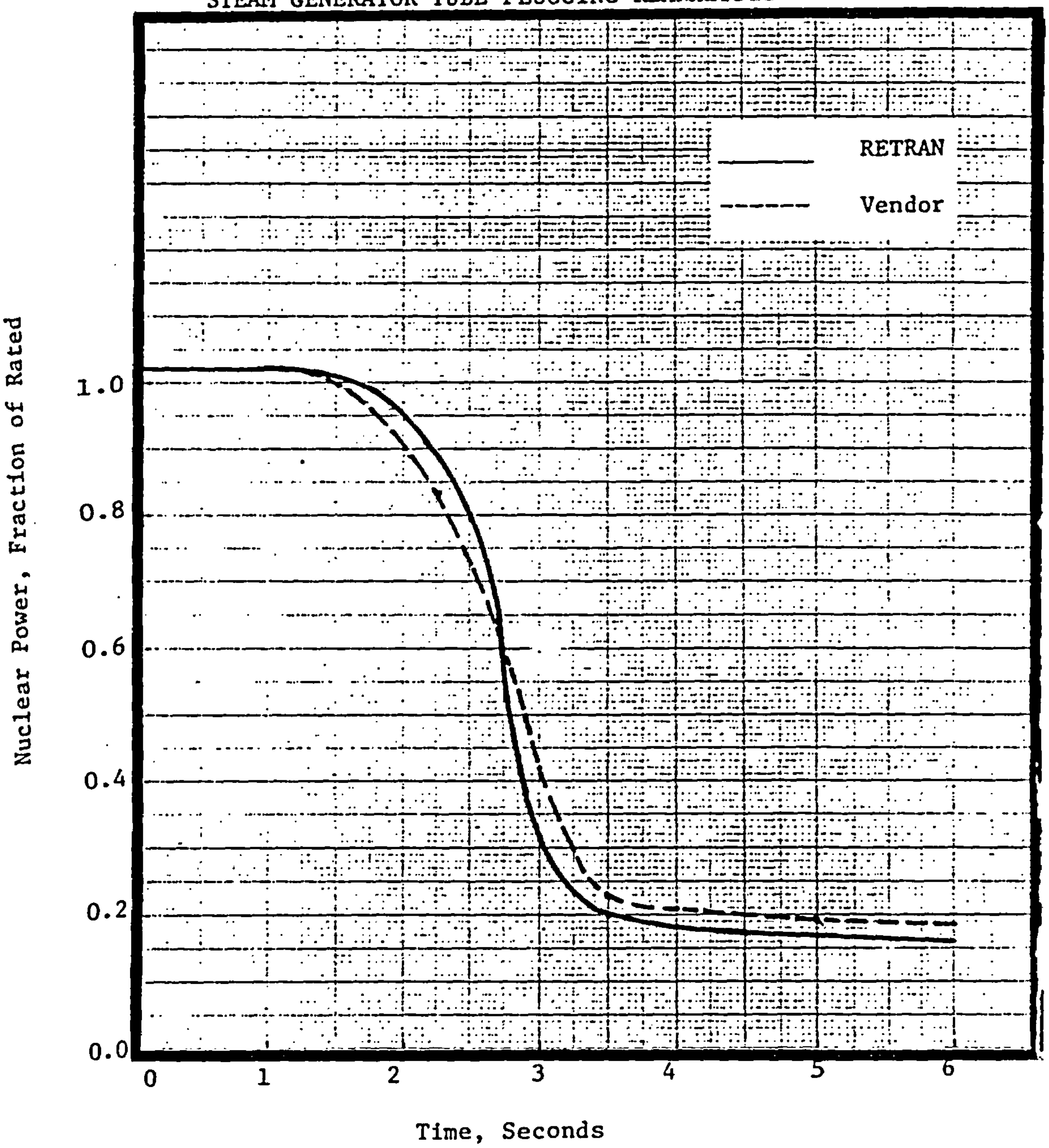


Figure 5.21

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

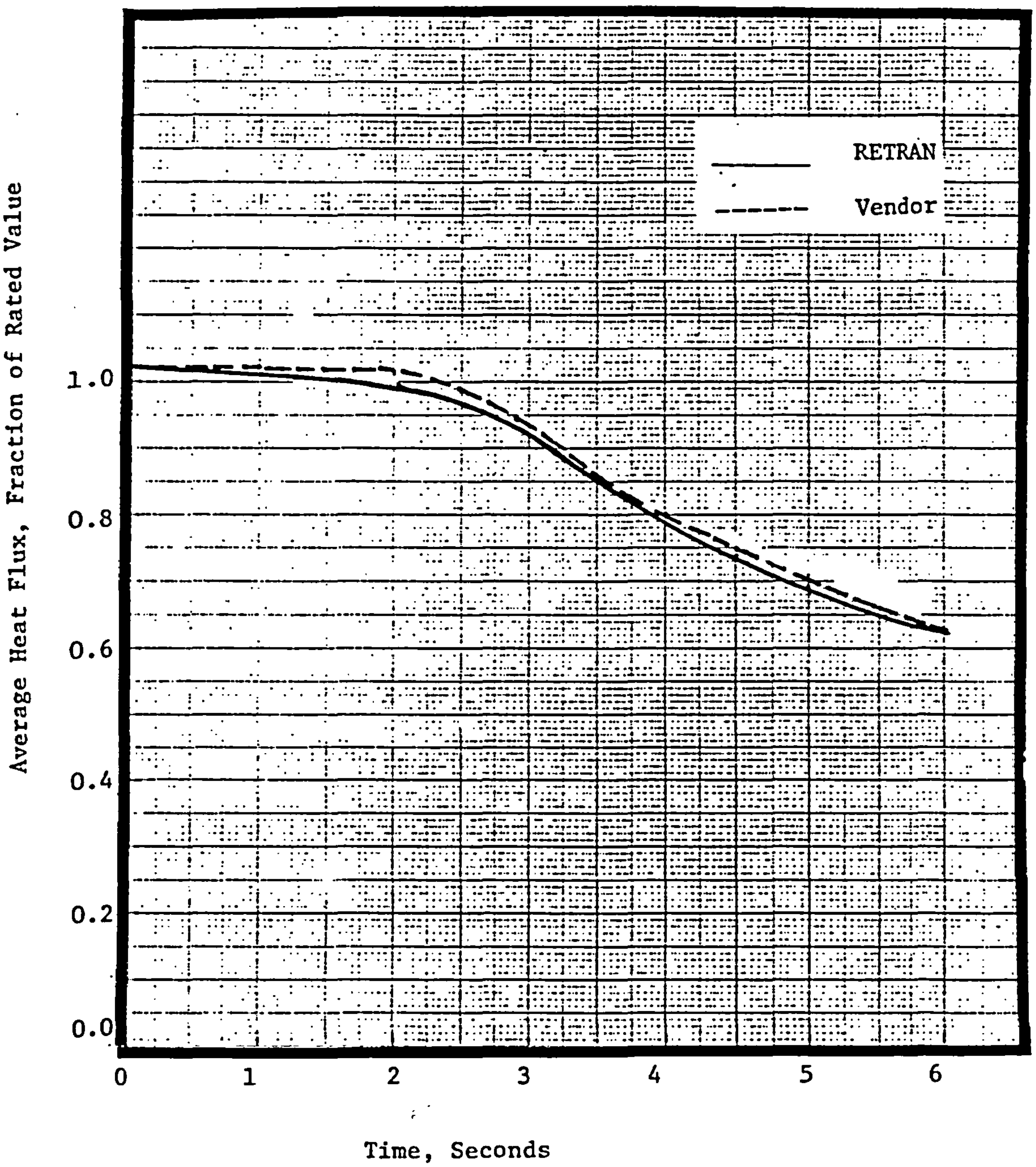


Figure 5.22

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

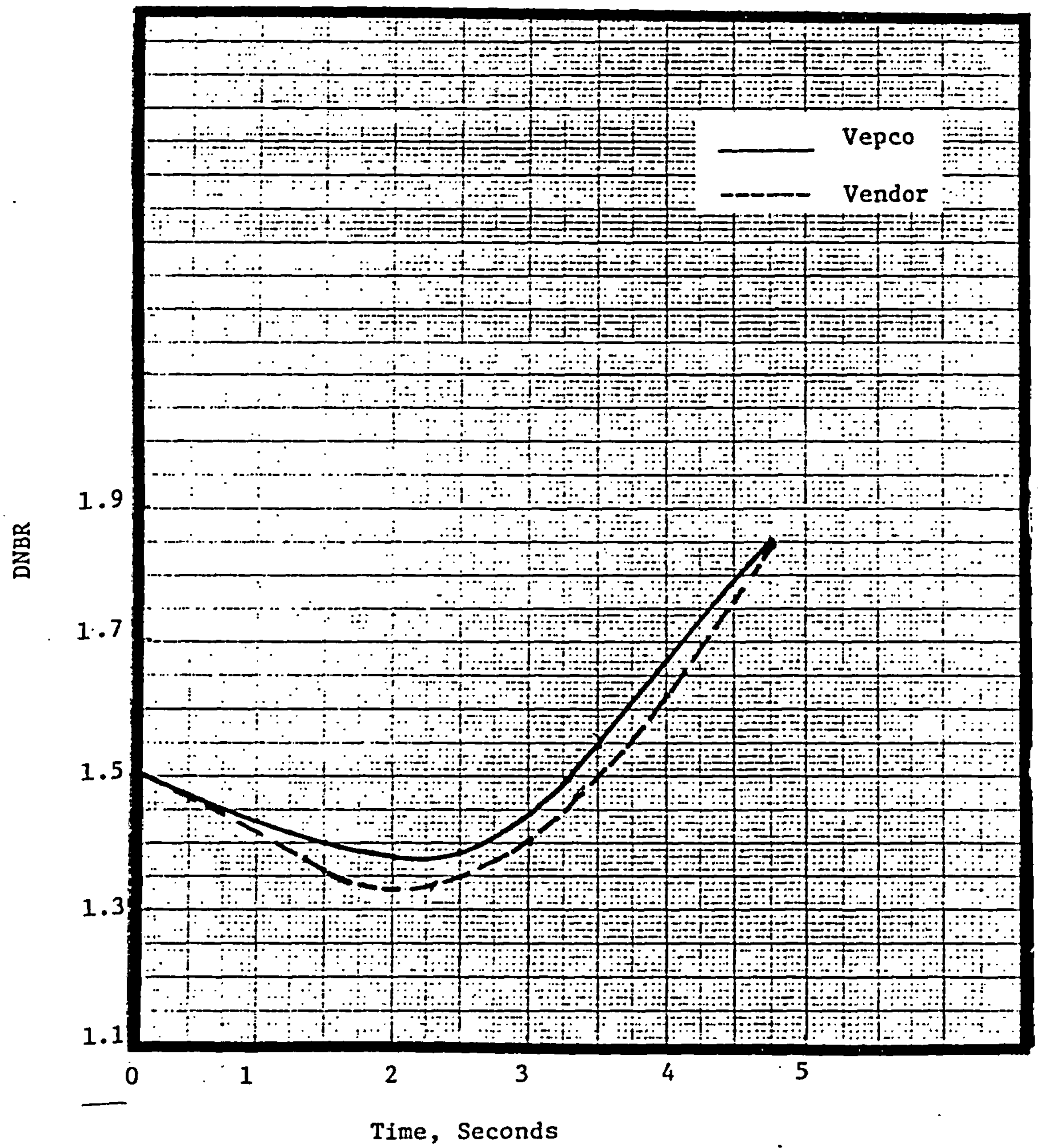


Figure 5.23

CORE FLOW COASTDOWN
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

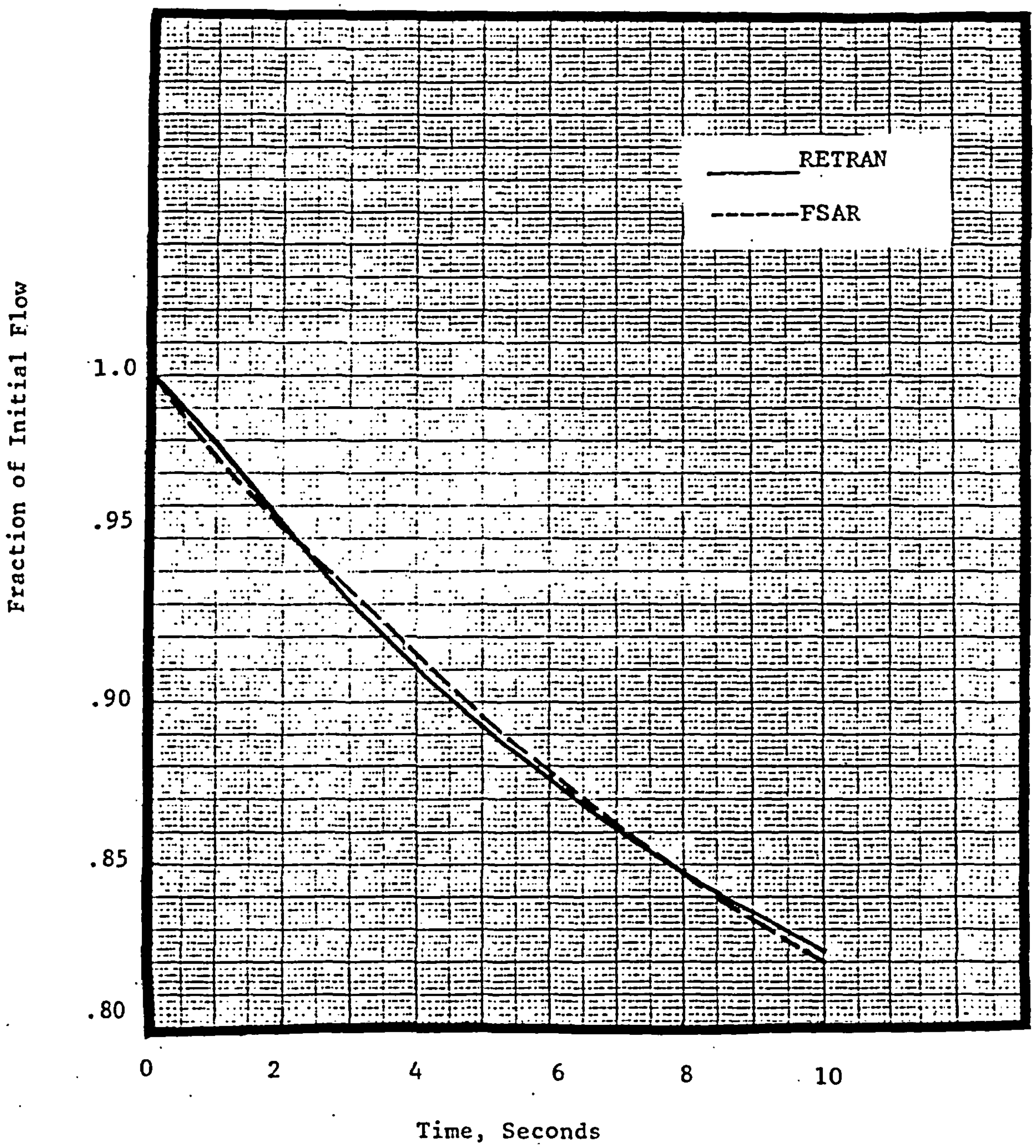


Figure 5.24

NUCLEAR POWER
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

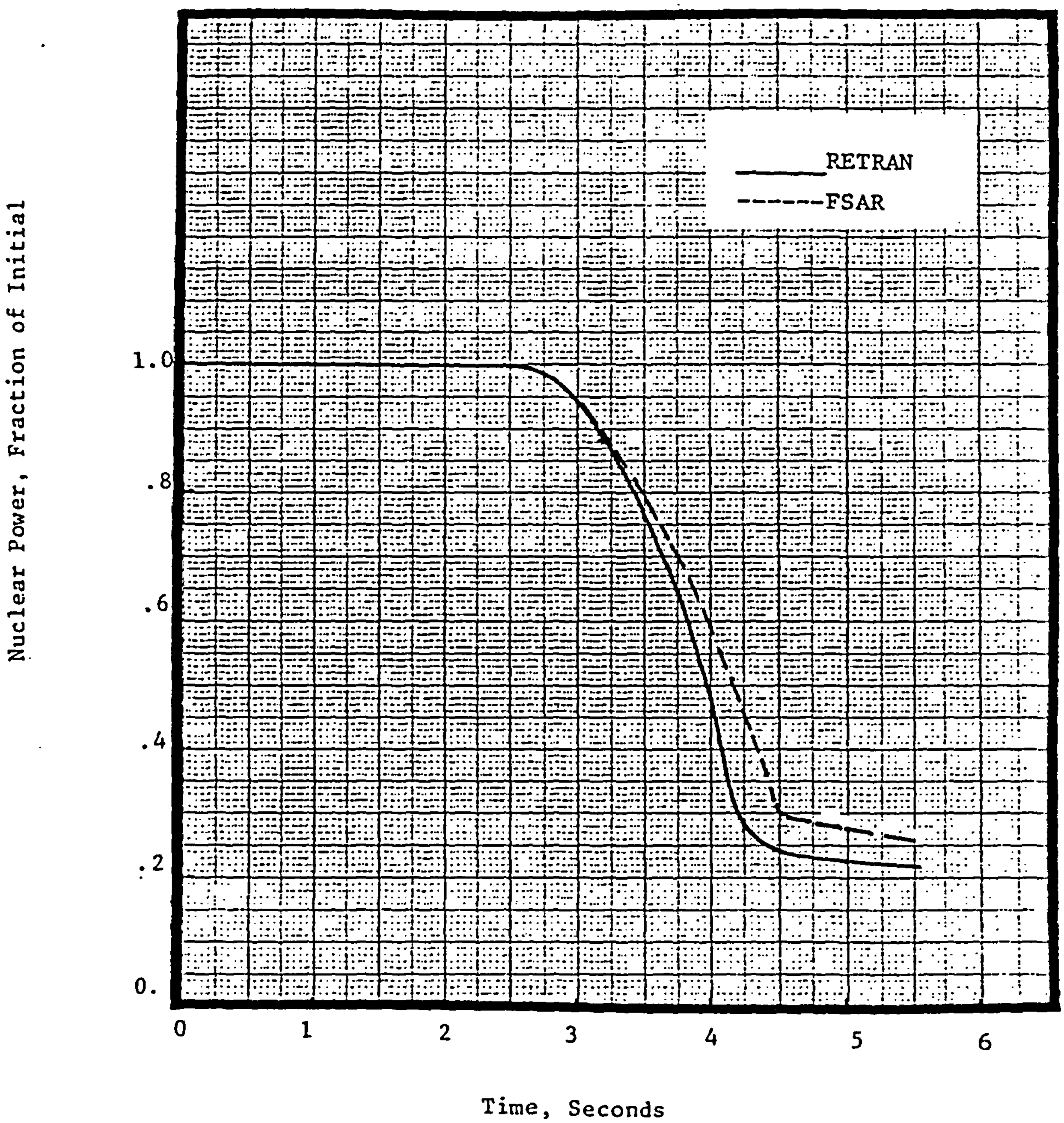


Figure 5.25

CORE AVERAGE HEAT FLUX
PARTIAL LOSS OF FLOW TRANSIENT
FSAR ANALYSIS

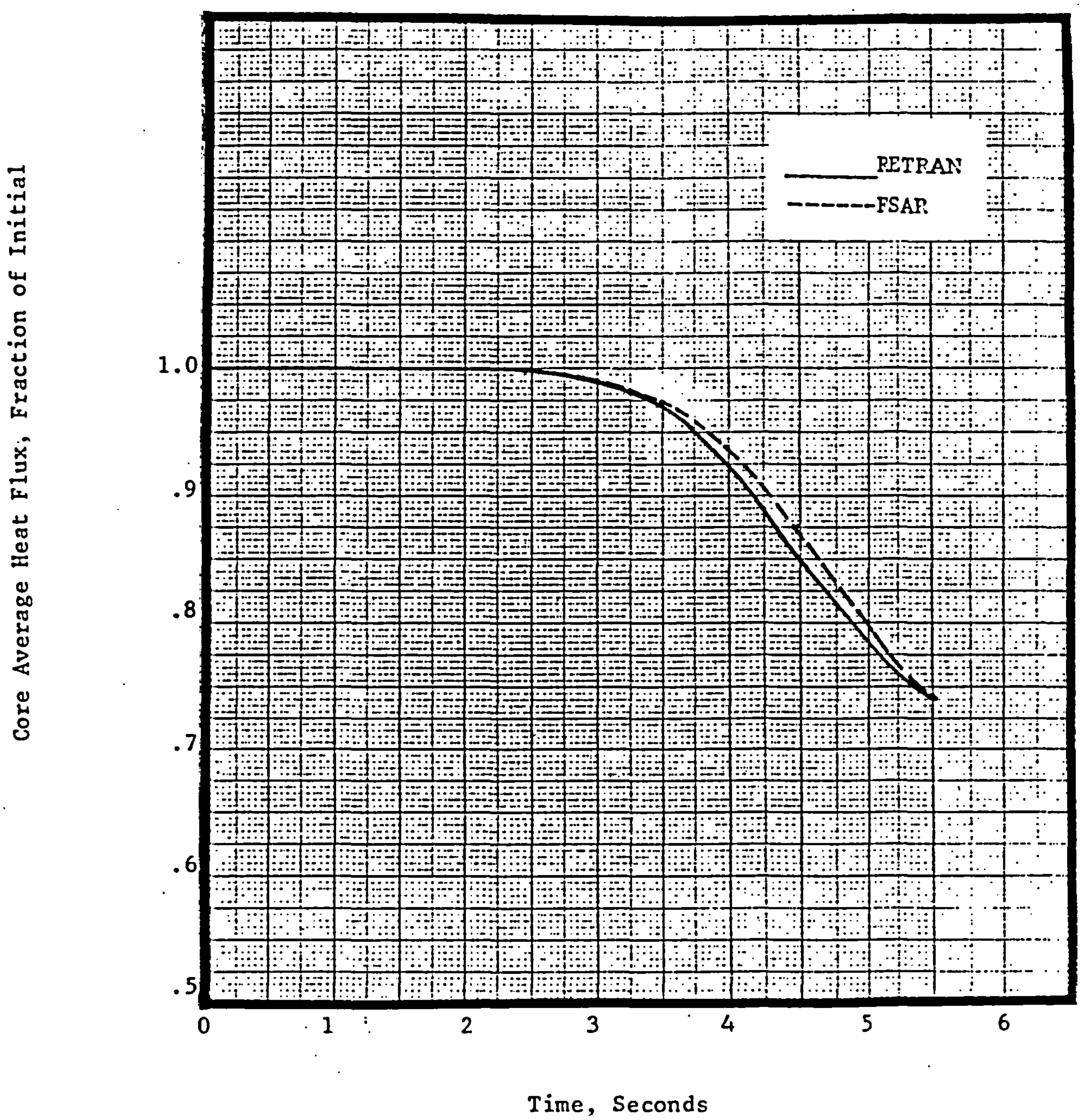
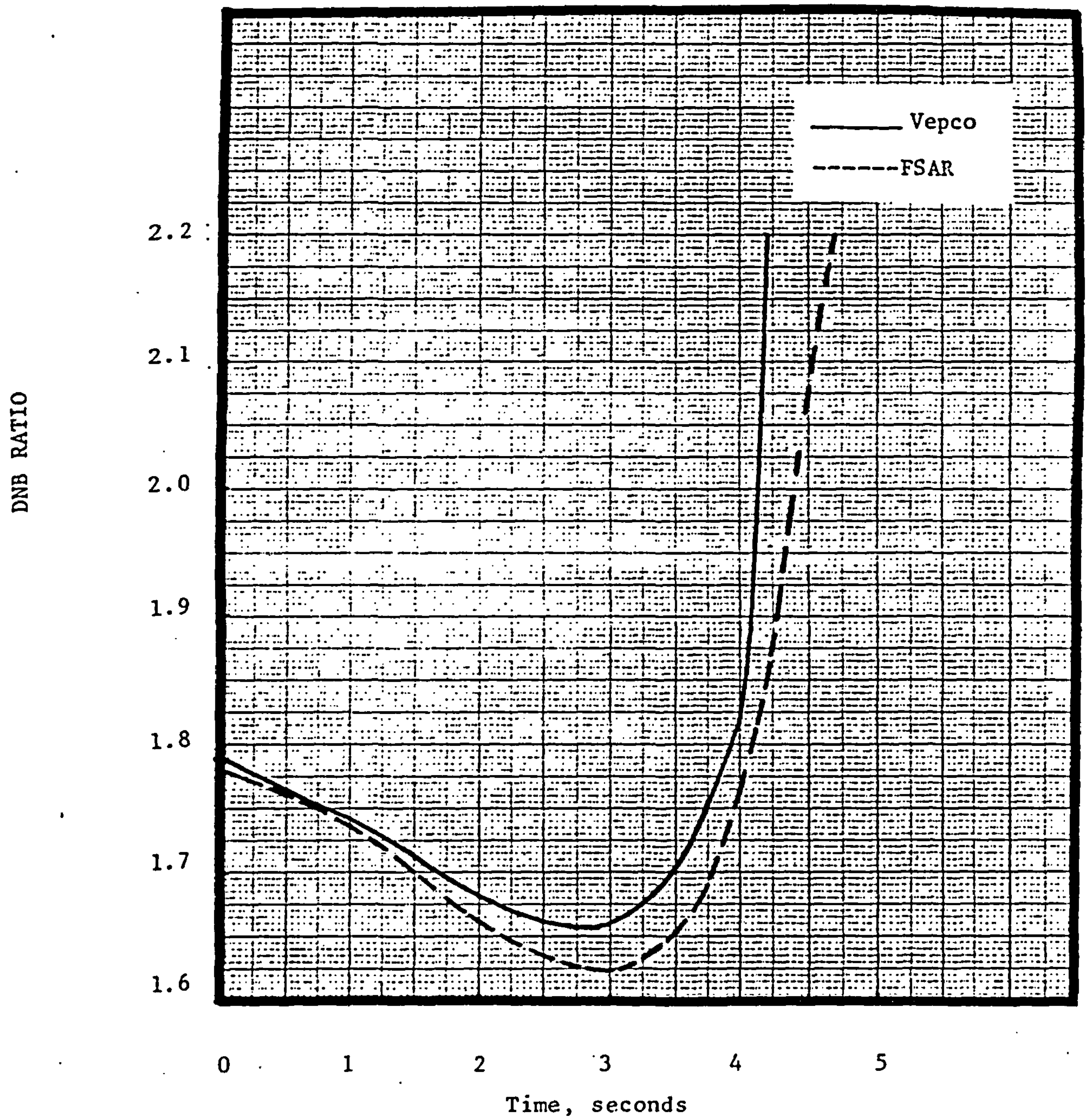


Figure 5.26

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

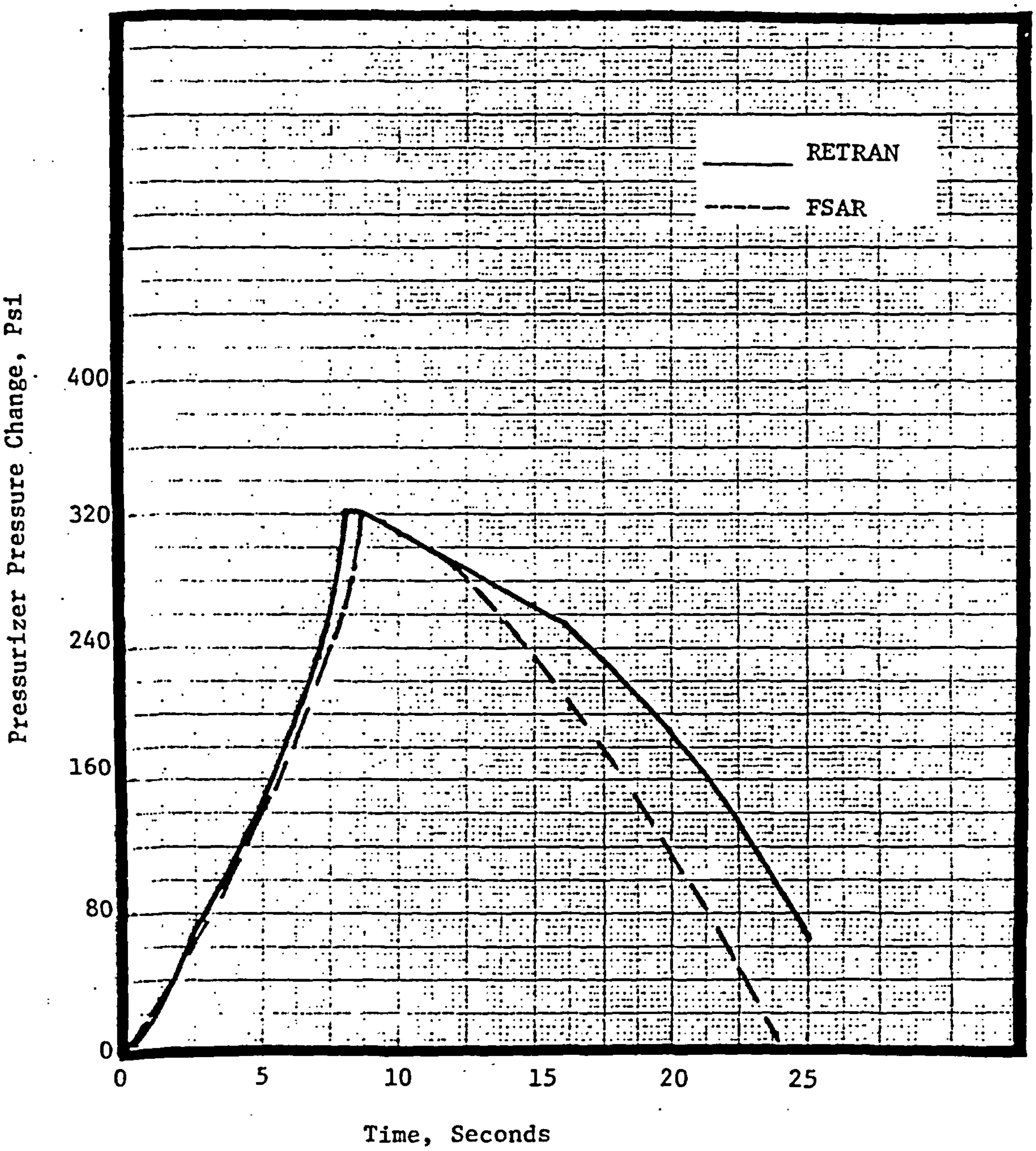


Figure 5.28

NUCLEAR POWER
LOSS OF LOAD TRANSIENT
BOL - FSAR ANALYSIS

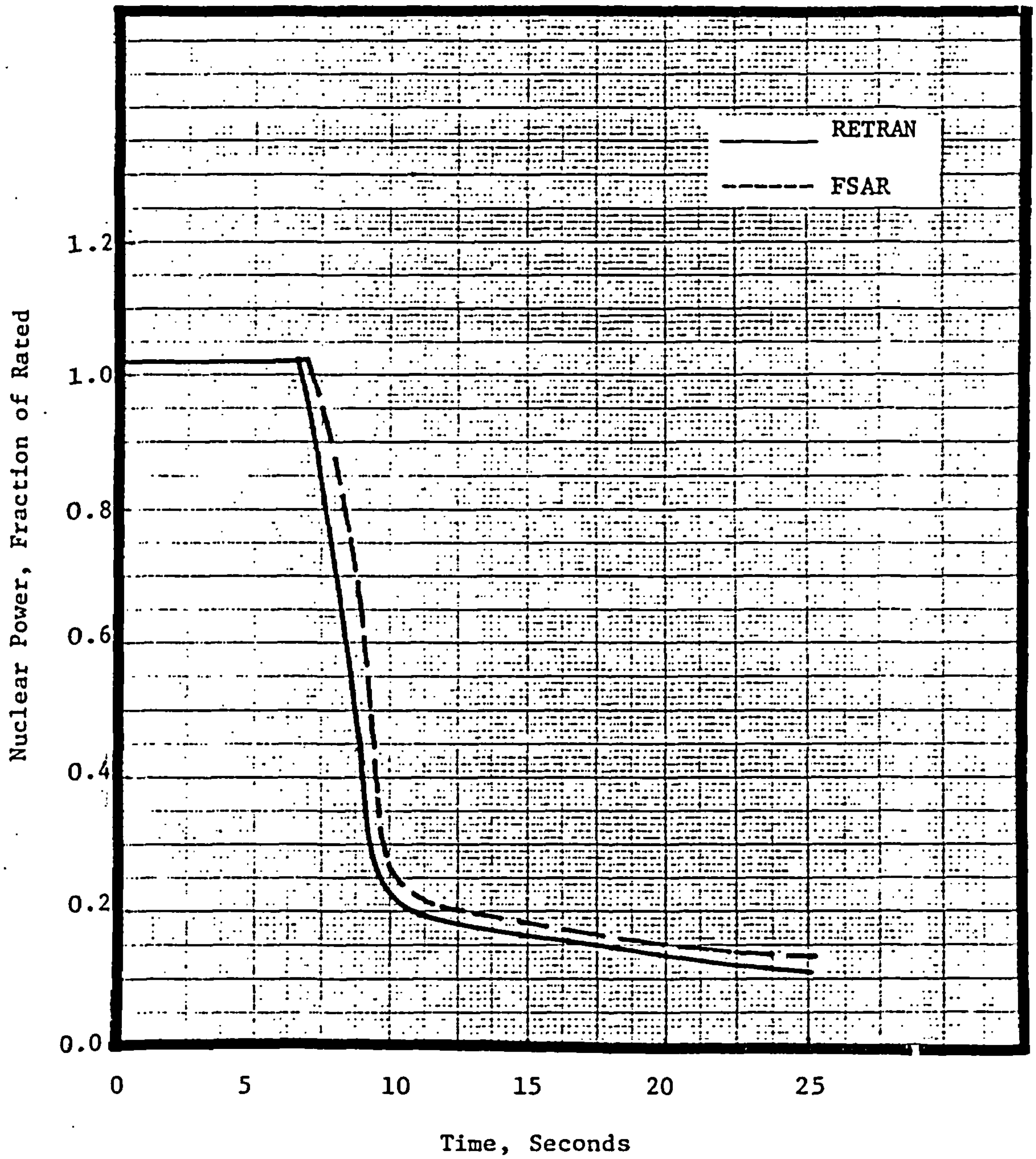


Figure 5.29

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

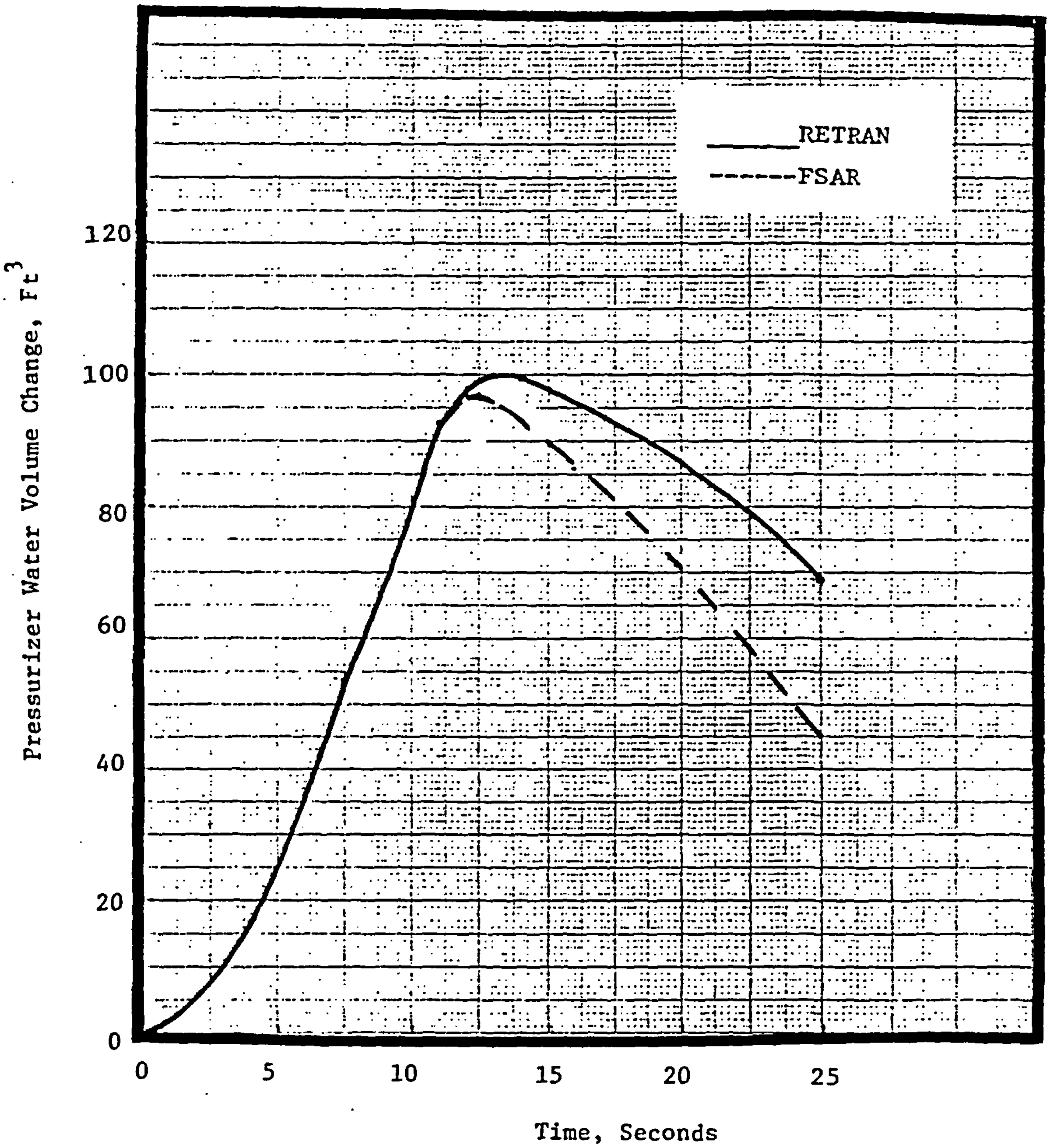


Figure 5.30

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

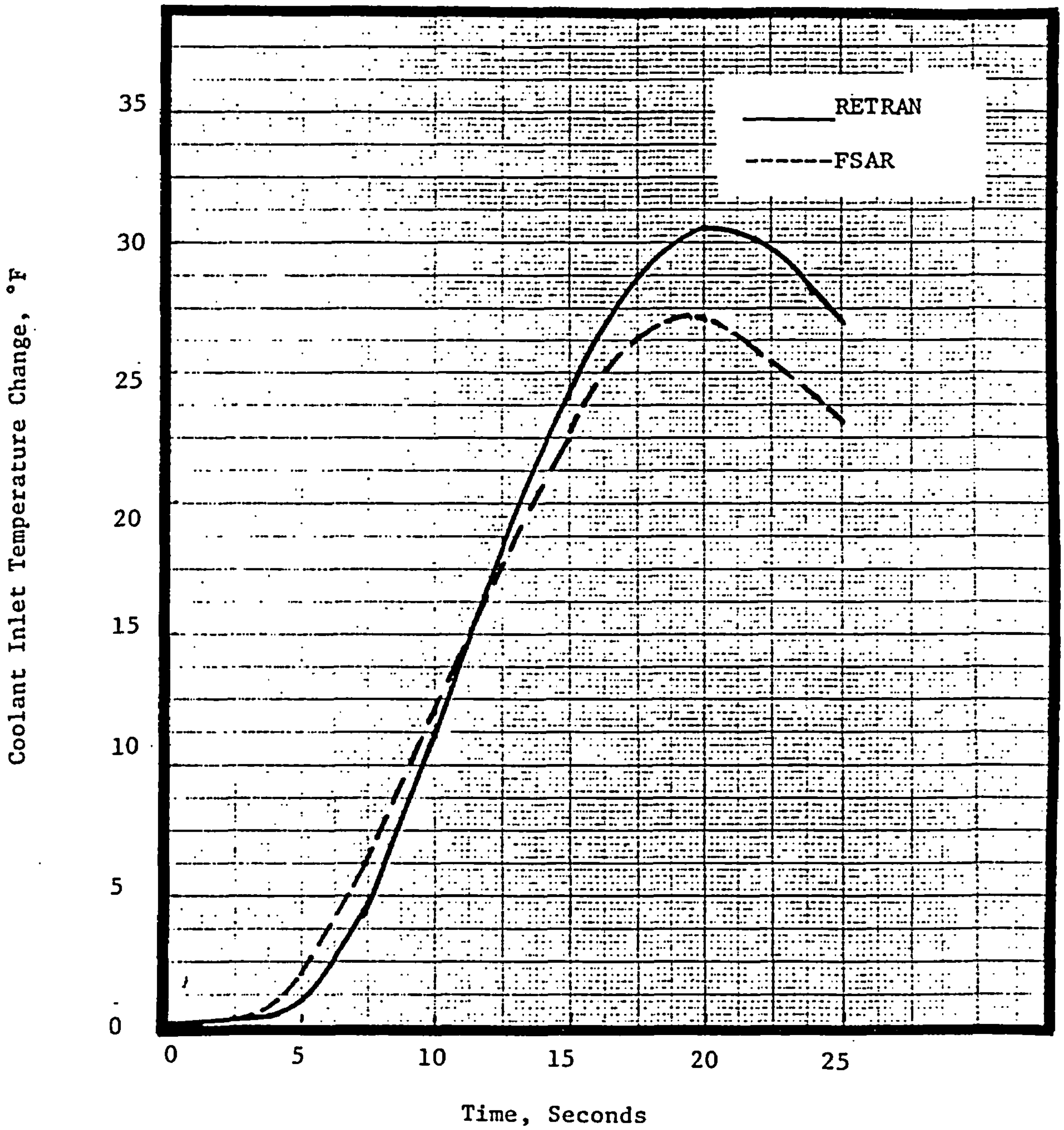


Figure 5.31

DNB RATIO
LOSS OF LOAD TRANSIENT
BOL - FSAR ANALYSIS

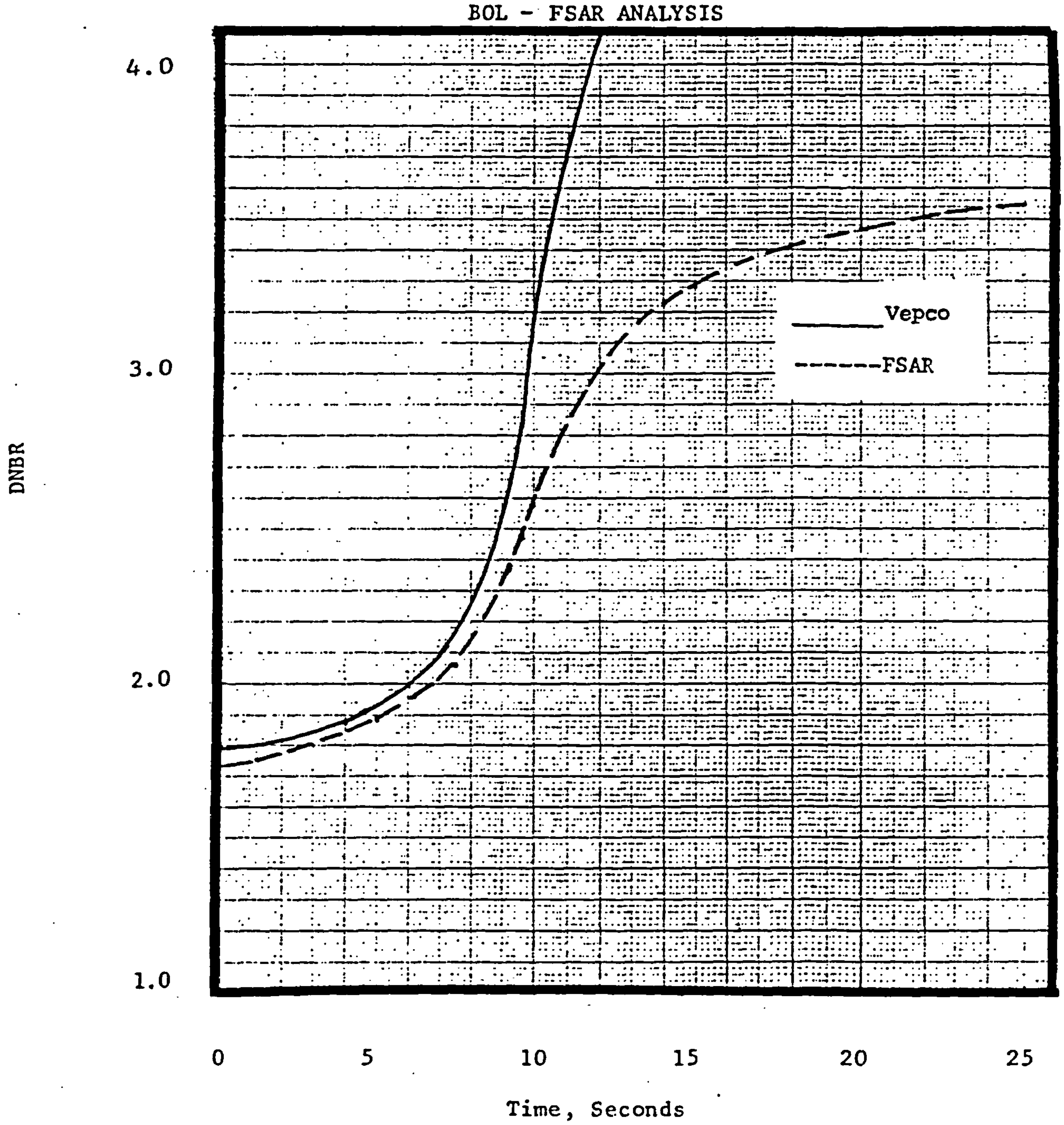


Figure 5.32

PRESSURIZER PRESSURE CHANGE
LOSS OF LOAD TRANSIENT
EOL-FSAR ANALYSIS

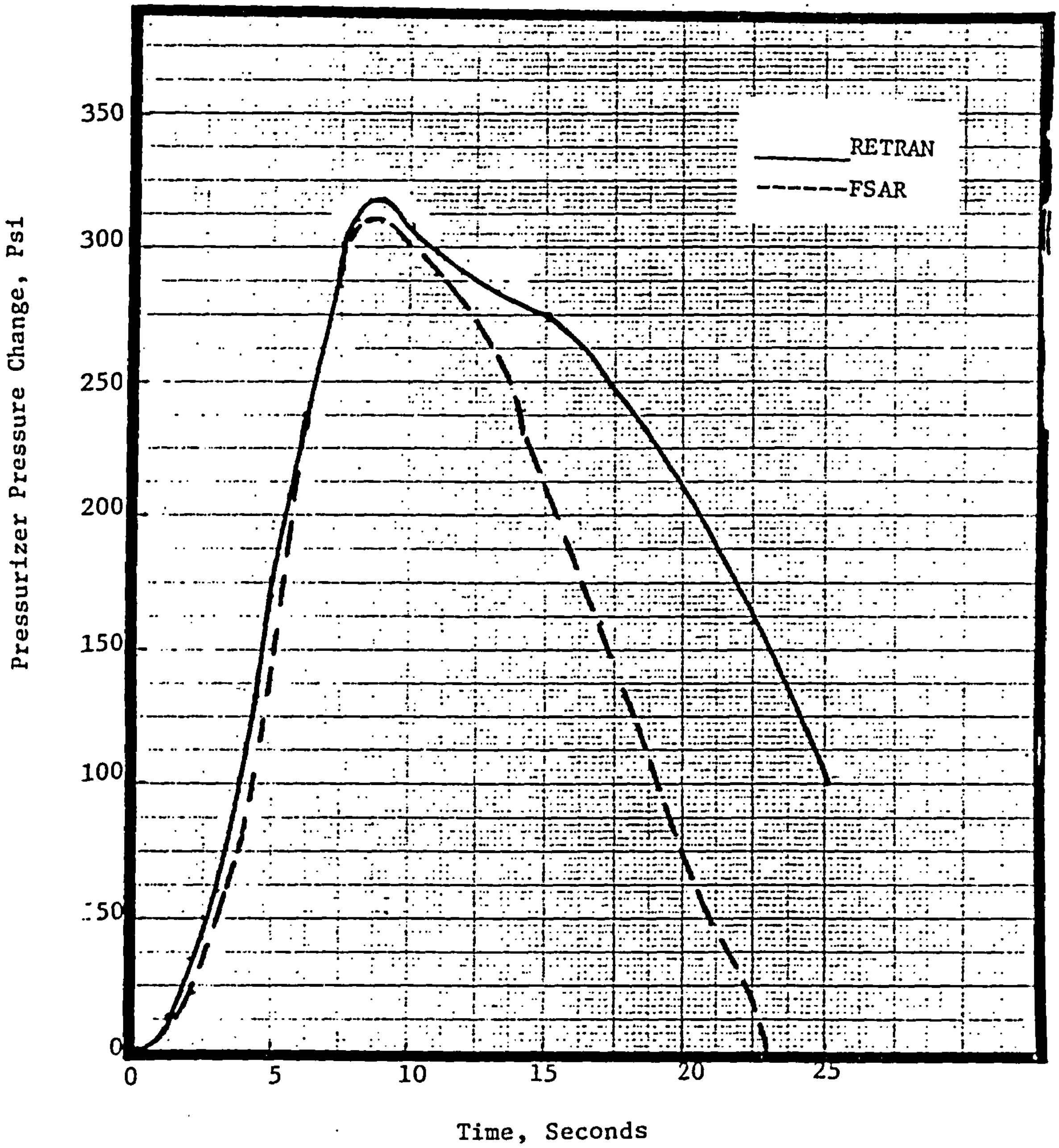


Figure 5.33

NUCLEAR POWER
LOSS OF LOAD TRANSIENT
EOL - FSAR ANALYSIS

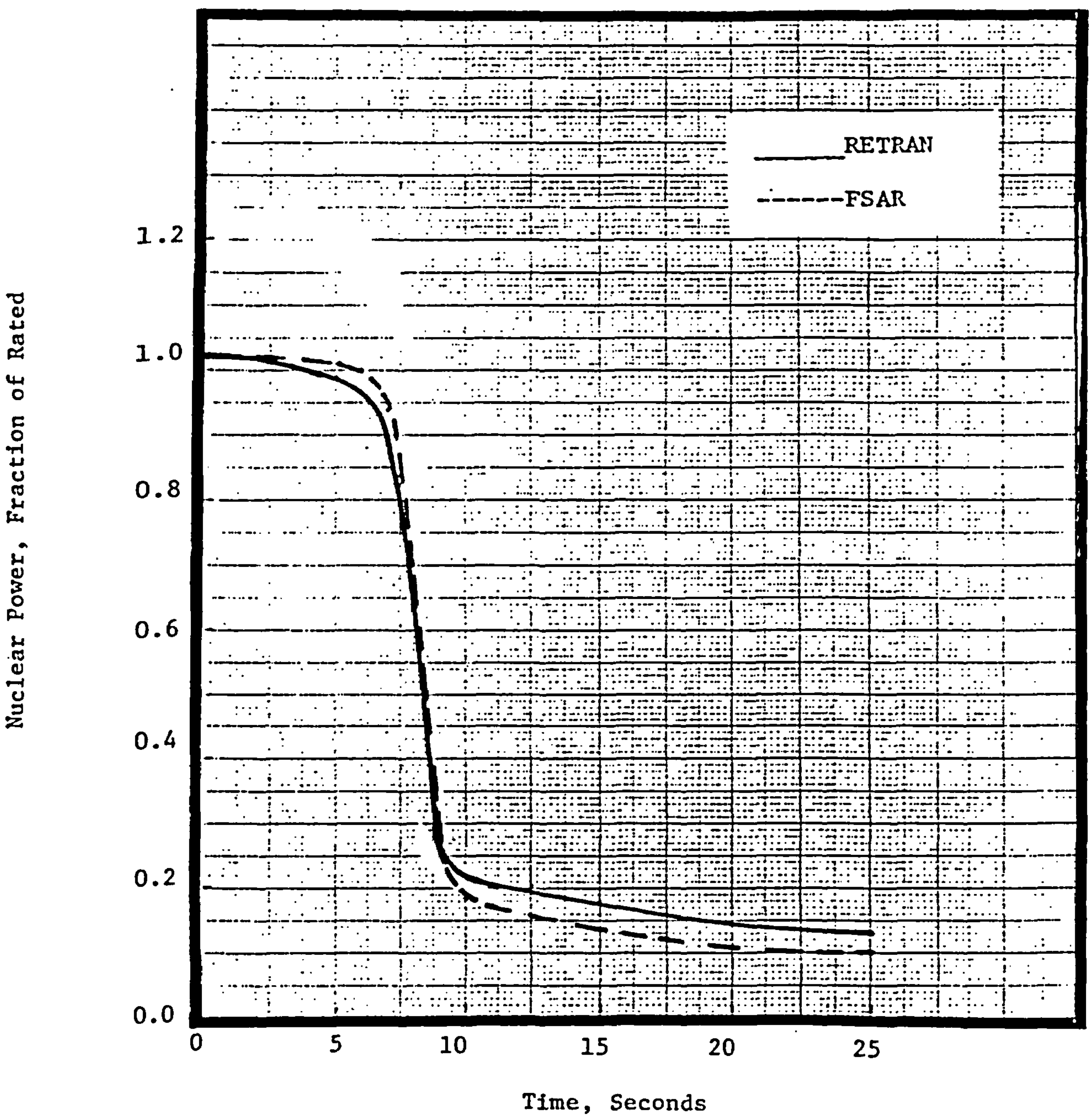


Figure 5.34

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

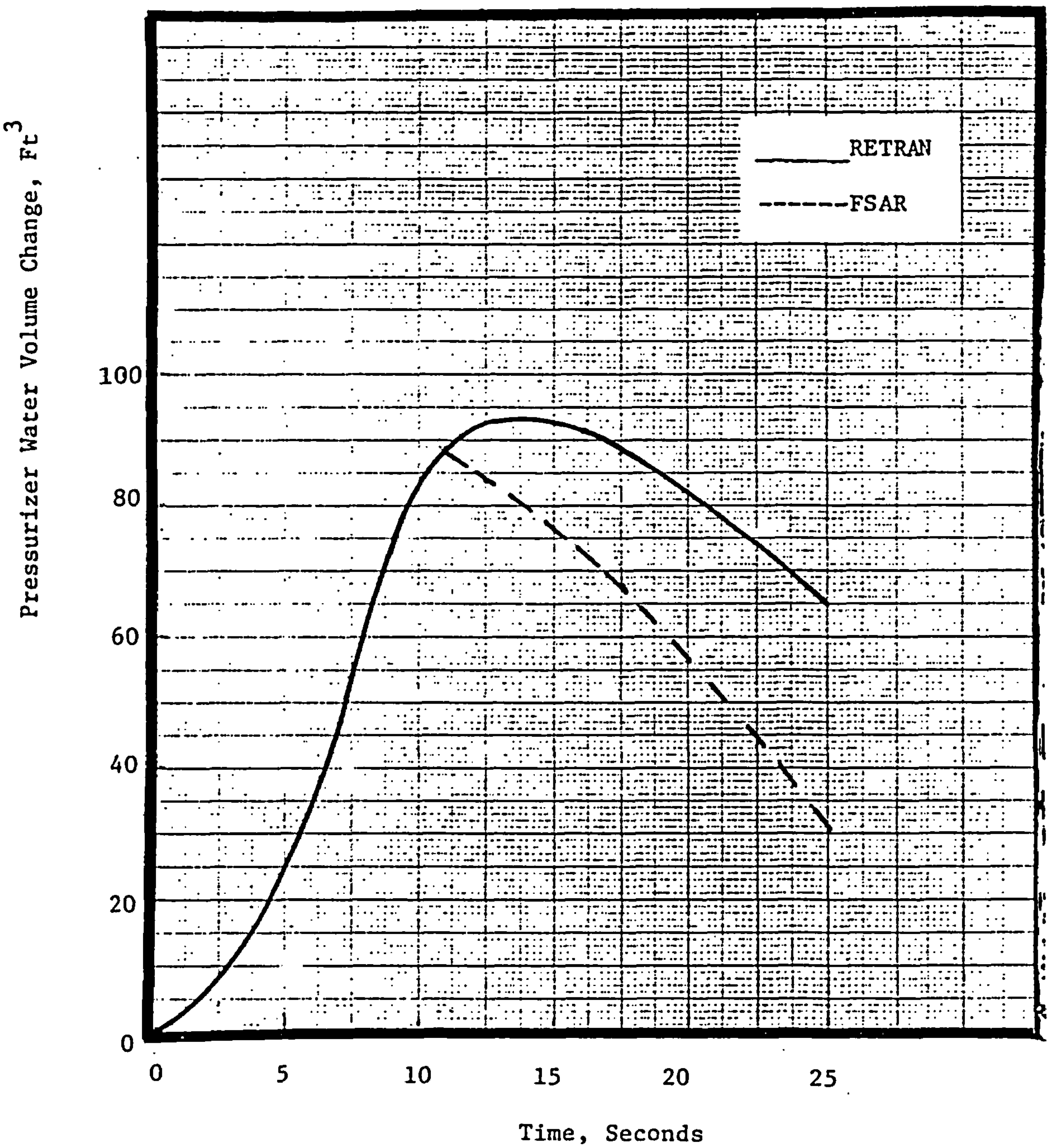


Figure 5.35

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

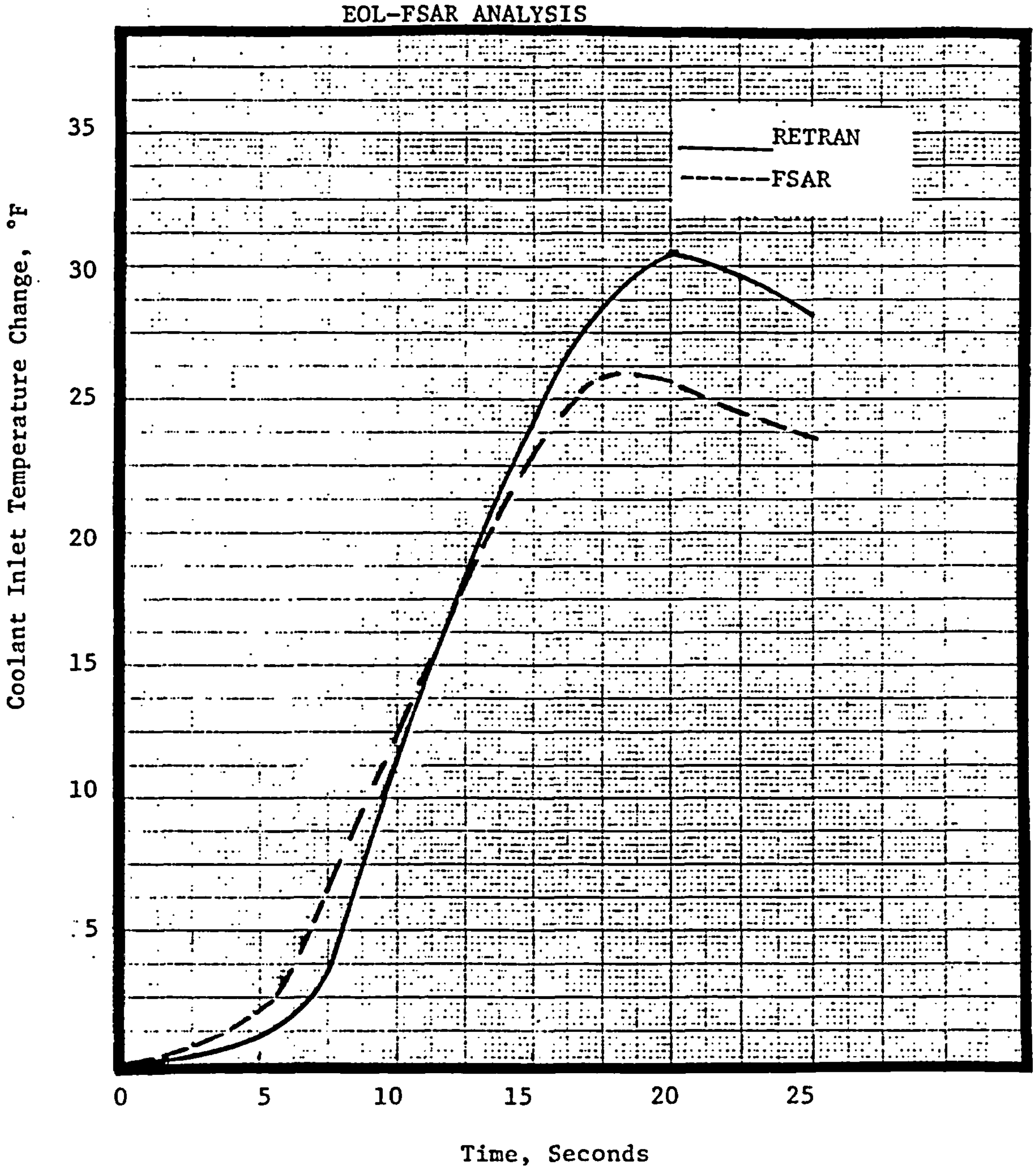


Figure 5.36

DNB RATIO
LOSS OF LOAD TRANSIENT
EOL FSAR ANALYSIS

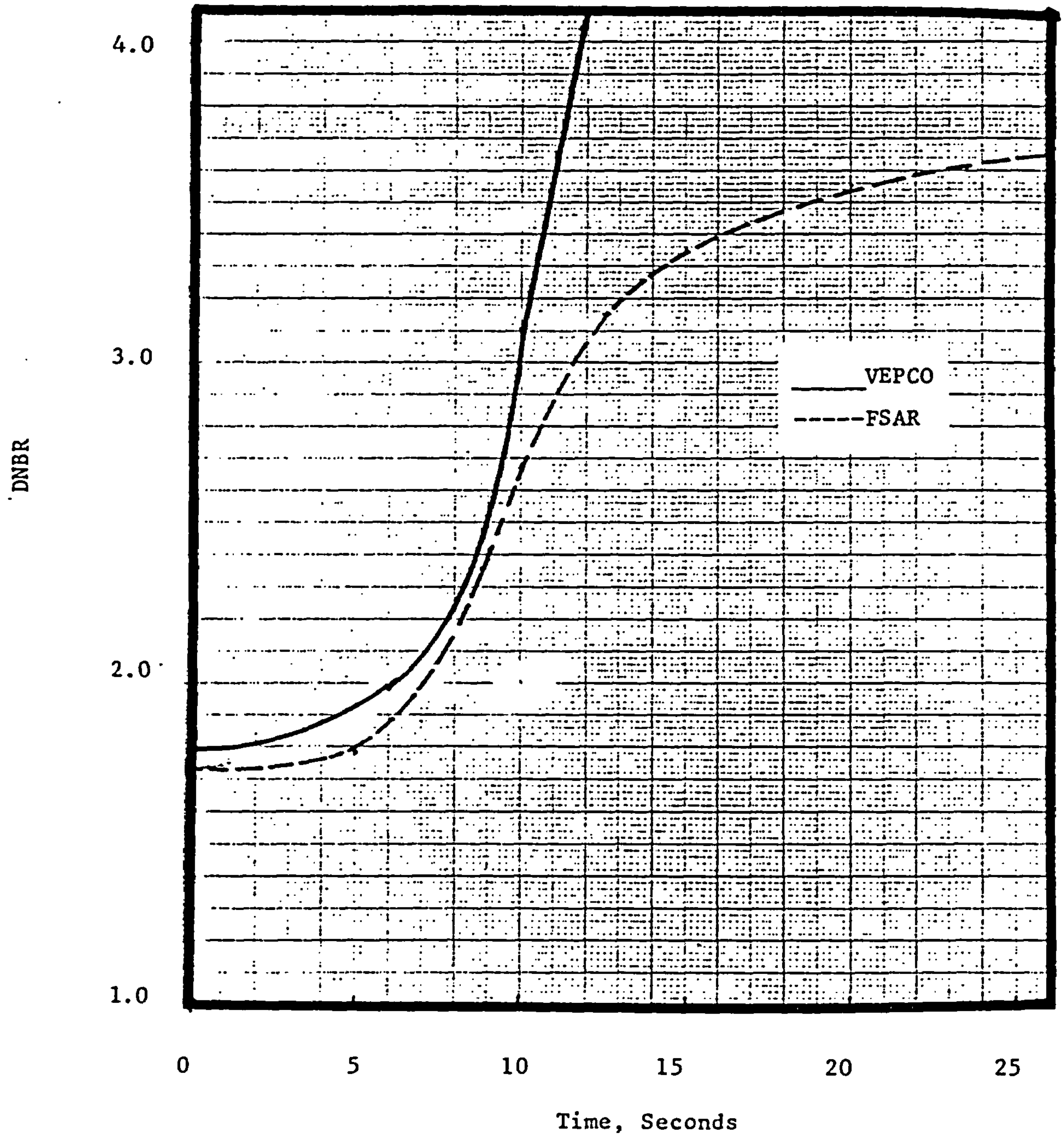


Figure 5.37

PRESSURIZER PRESSURE
LOSS OF LOAD TRANSIENT
POSITIVE MODERATOR COEFFICIENT REANALYSIS

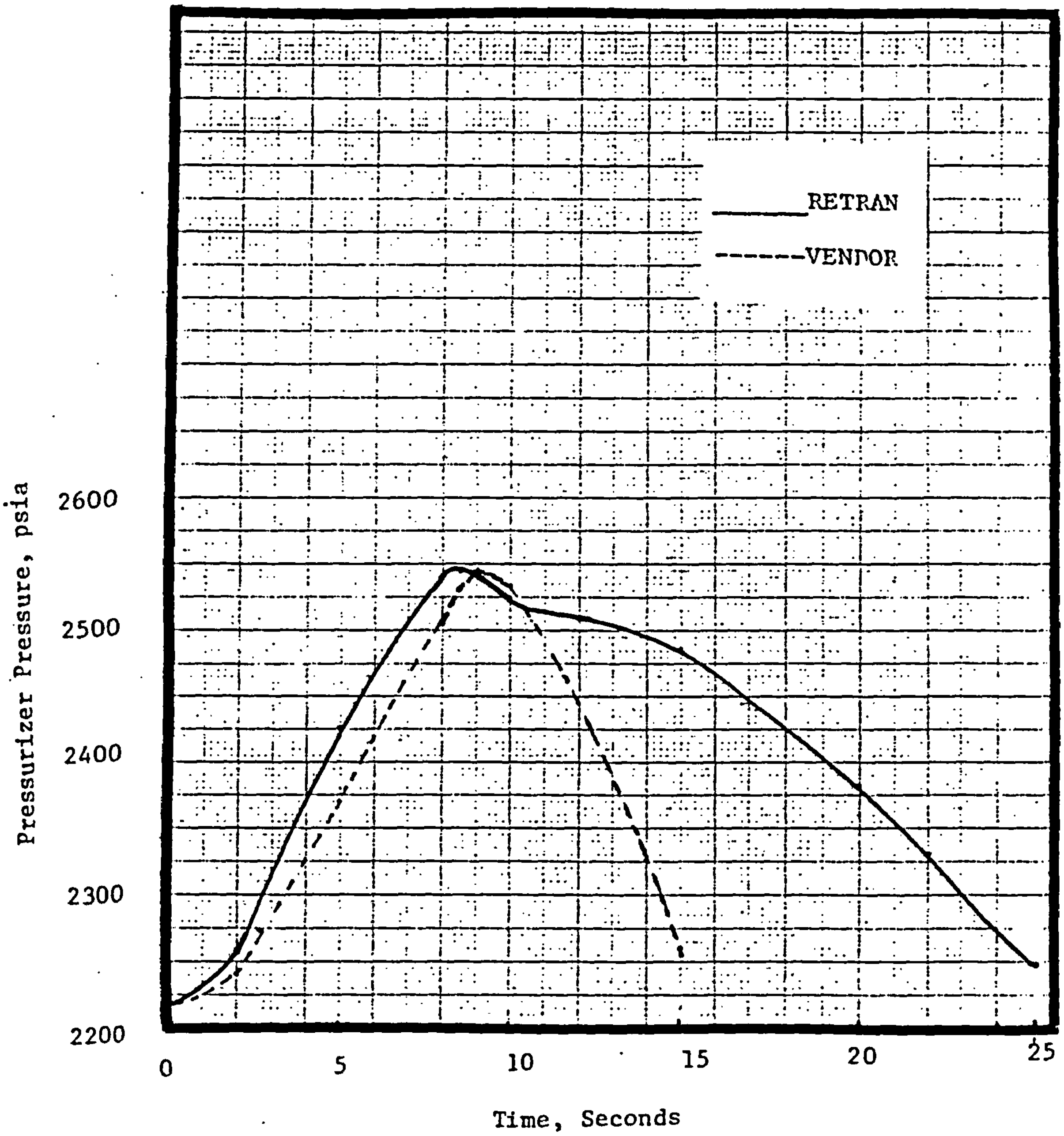


Figure 5.38

NUCLEAR POWER
LOSS OF LOAD TRANSIENT

POSITIVE MODERATOR COEFFICIENT REANALYSIS

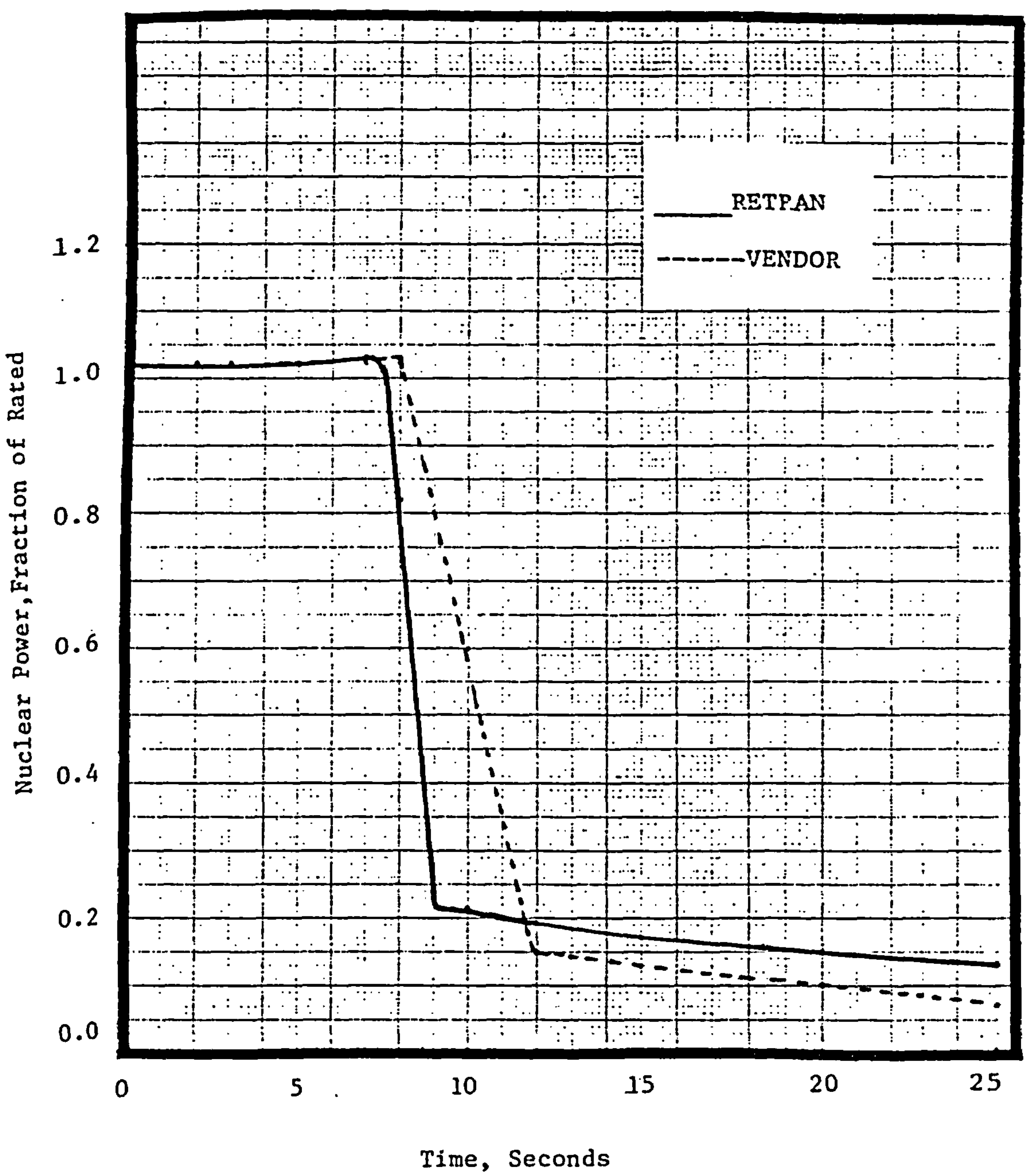


Figure 5.39

AVERAGE COOLANT TEMPERATURE
LOSS OF LOAD TRANSIENT
POSITIVE MODERATOR COEFFICIENT REANALYSIS

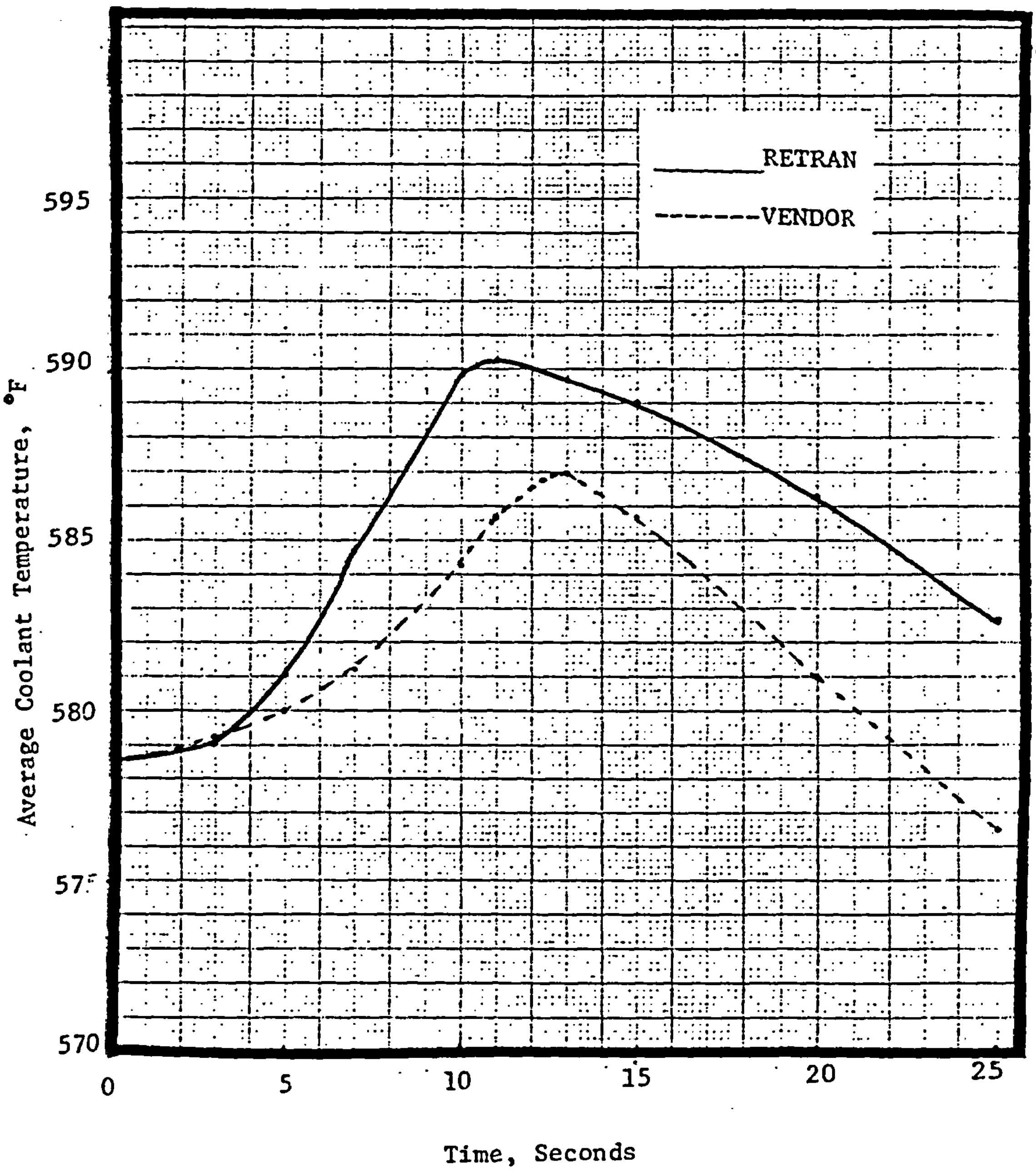


Figure 5.40

DNB RATIO
LOSS OF LOAD TRANSIENT
POSITIVE MODERATOR COEFFICIENT ASSUMPTION REANALYSIS

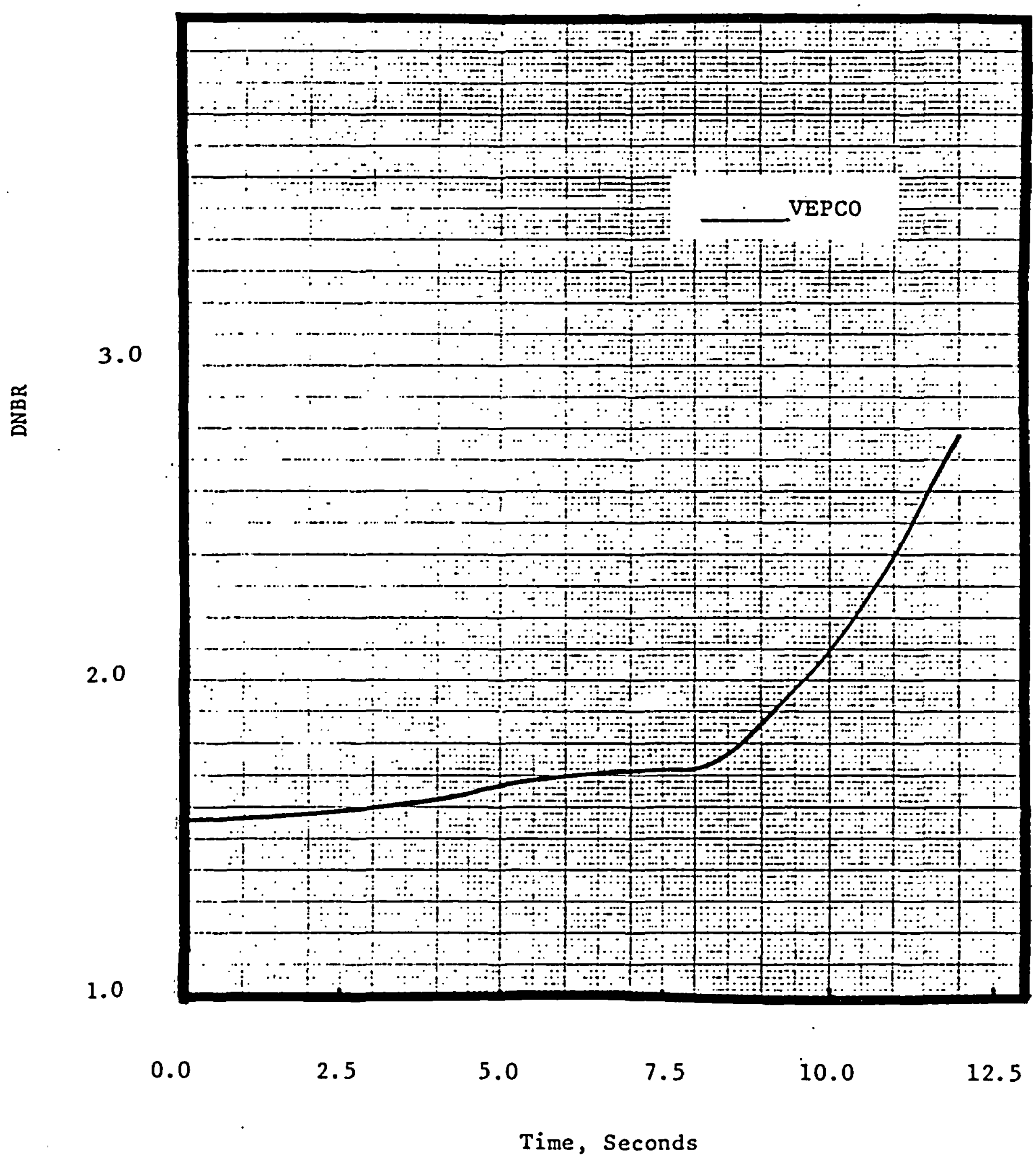


Figure 5.41

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

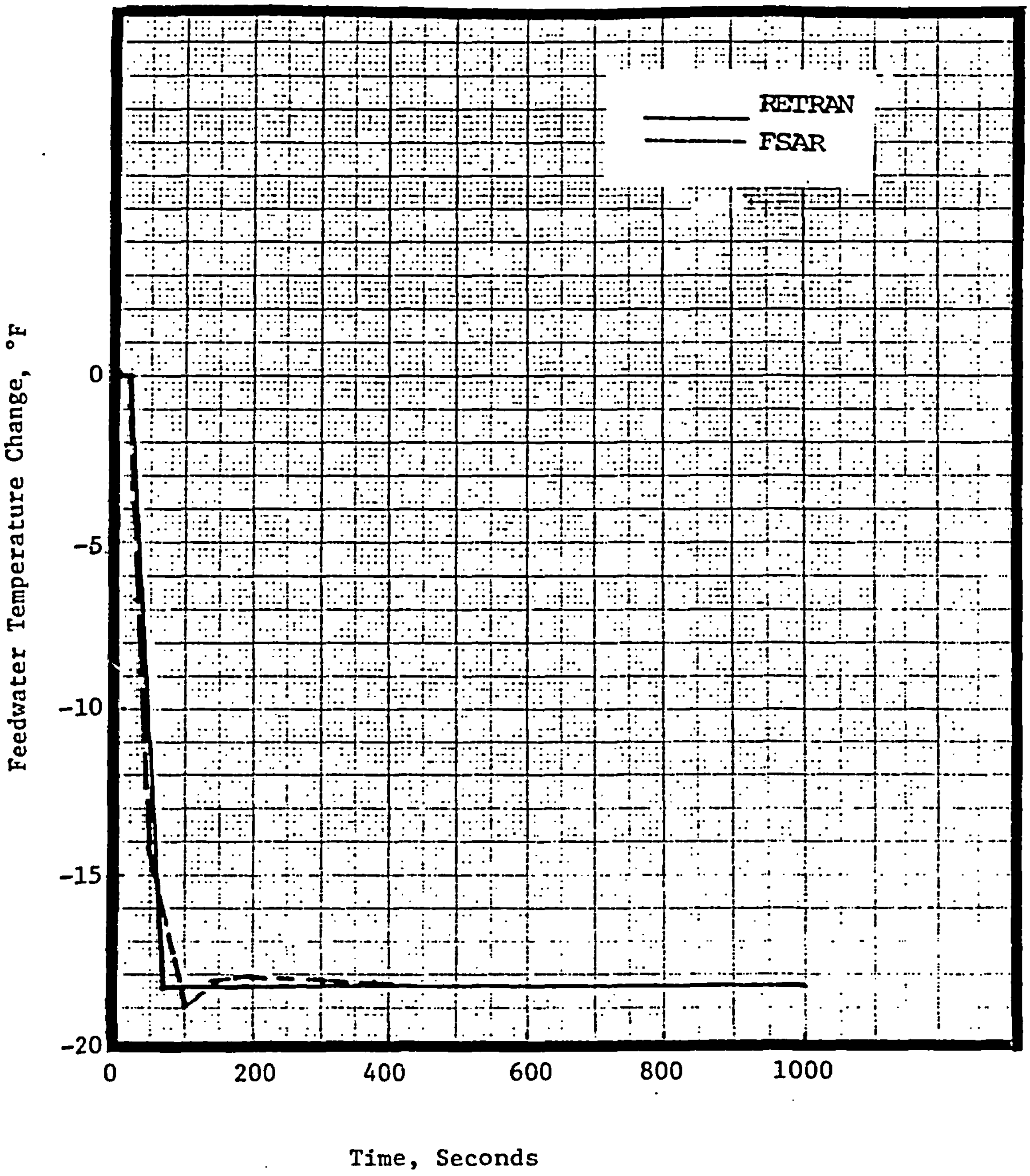


Figure 5.42

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

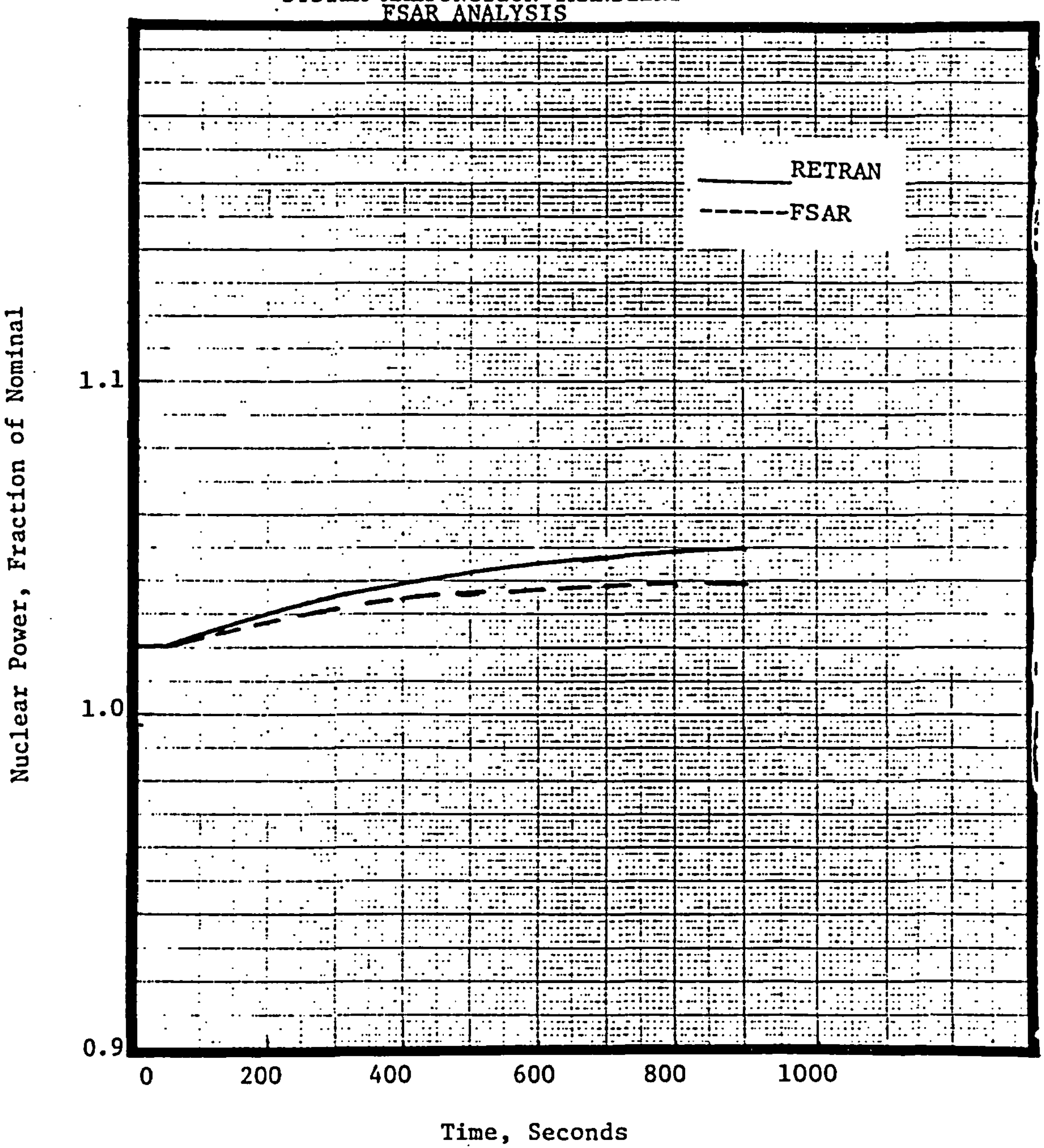


Figure 5.43

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

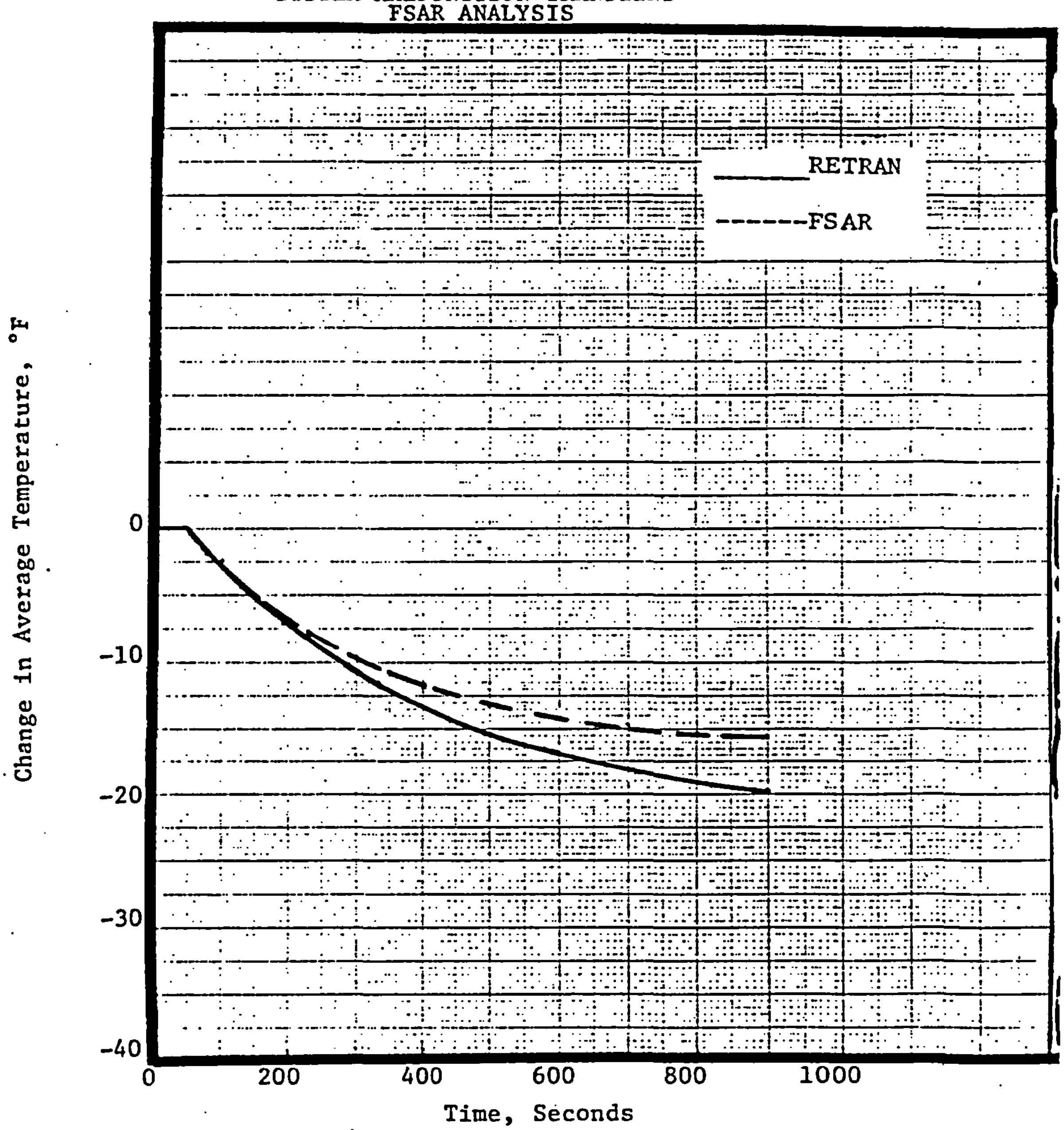


Figure 5.44

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

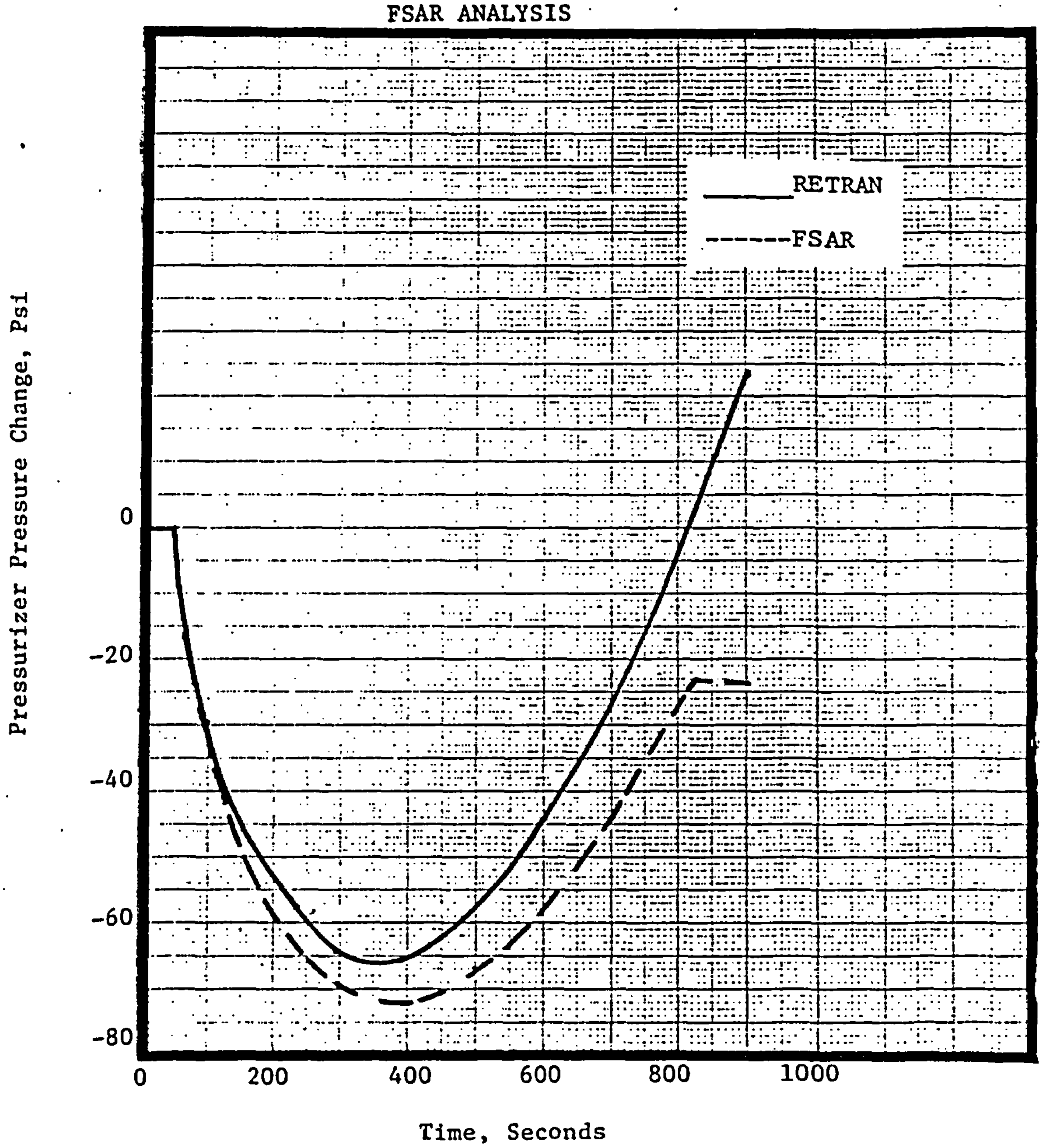


Figure 5.45

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

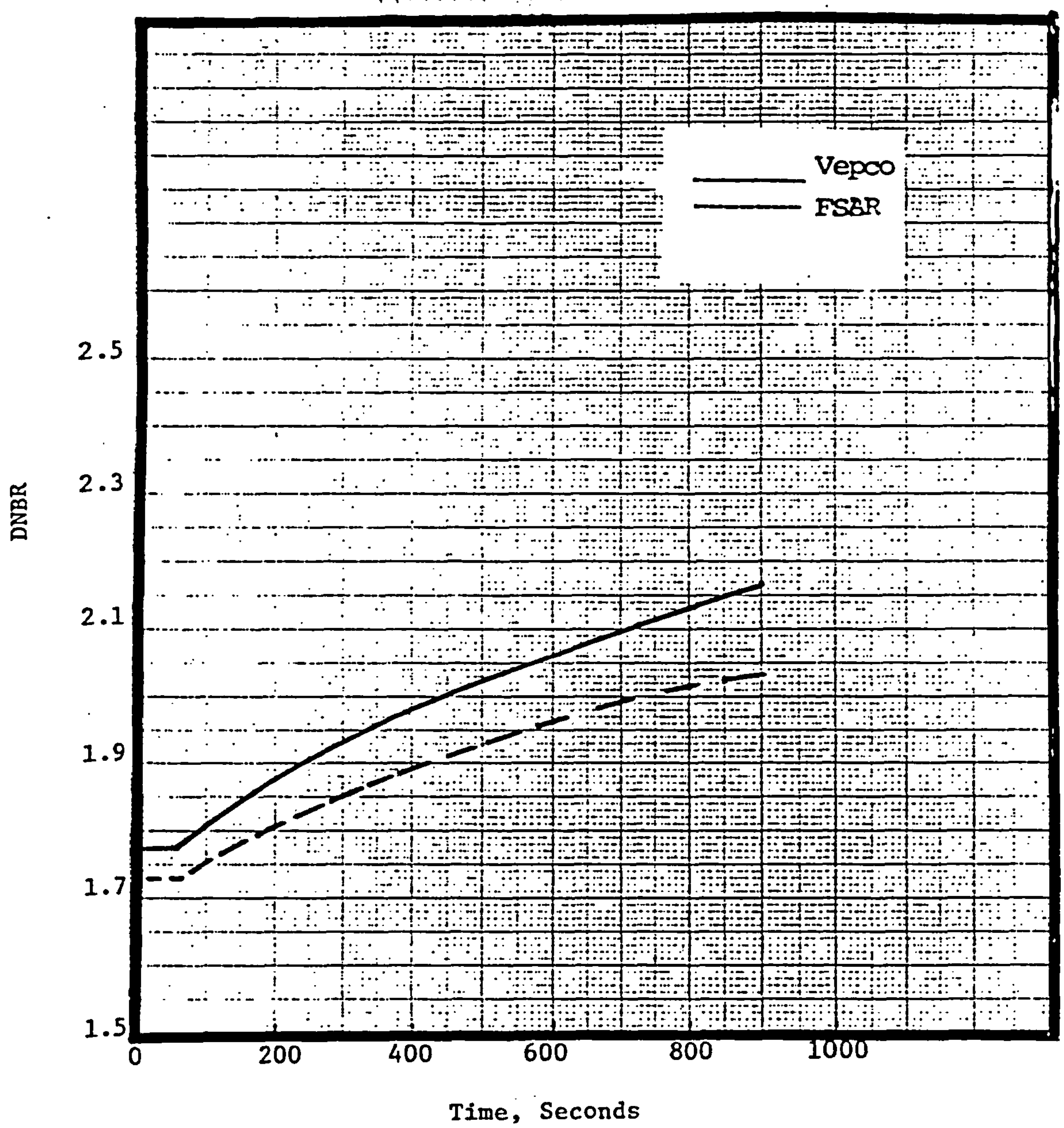


Figure 5.46

BREAK FLOW RATE
MAIN STEAM LINE BREAK
FSAR ANALYSIS

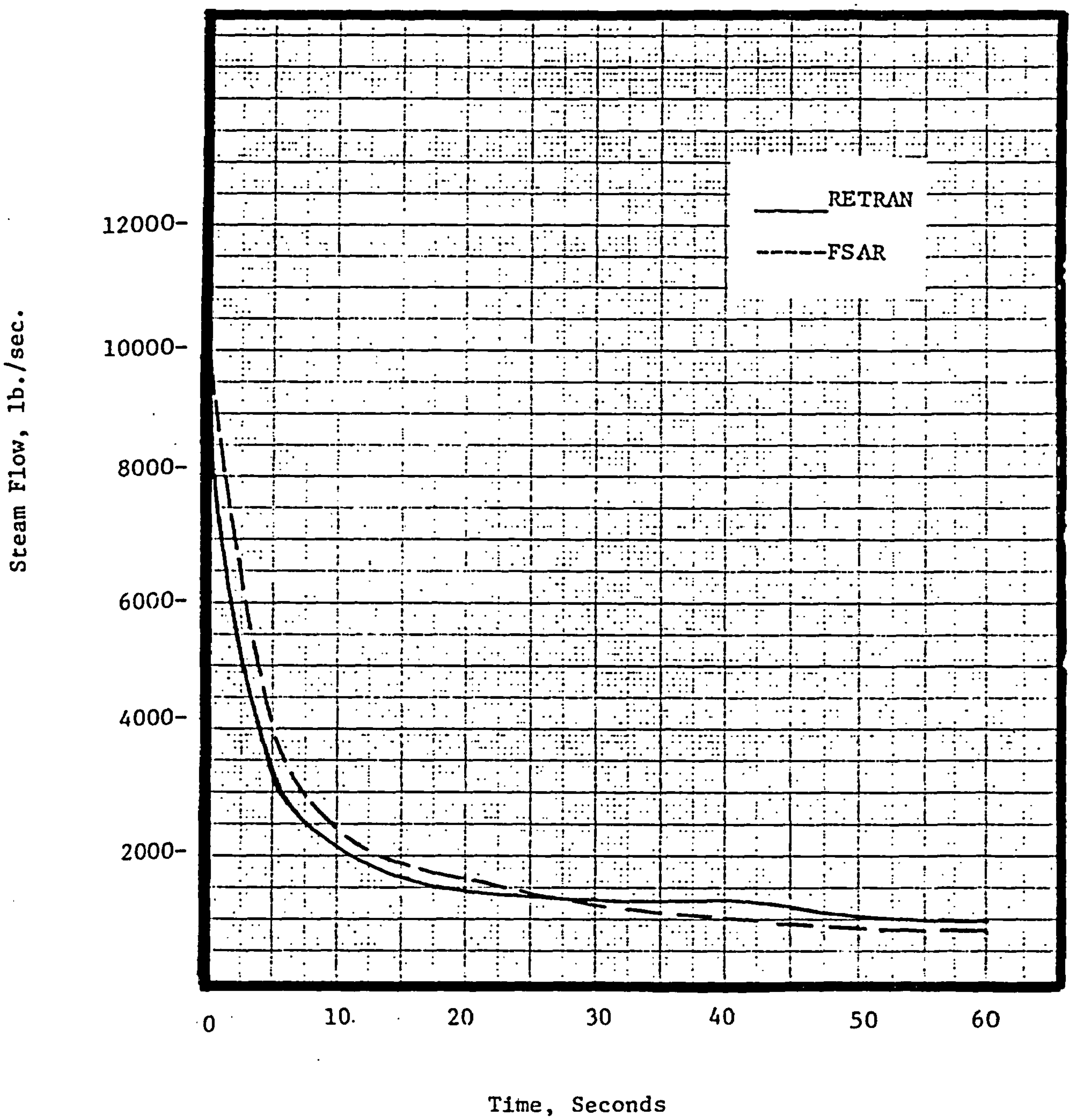


Figure 5.47

PRESSURIZER PRESSURE
MAIN STEAM LINE BREAK
FSAR ANALYSIS

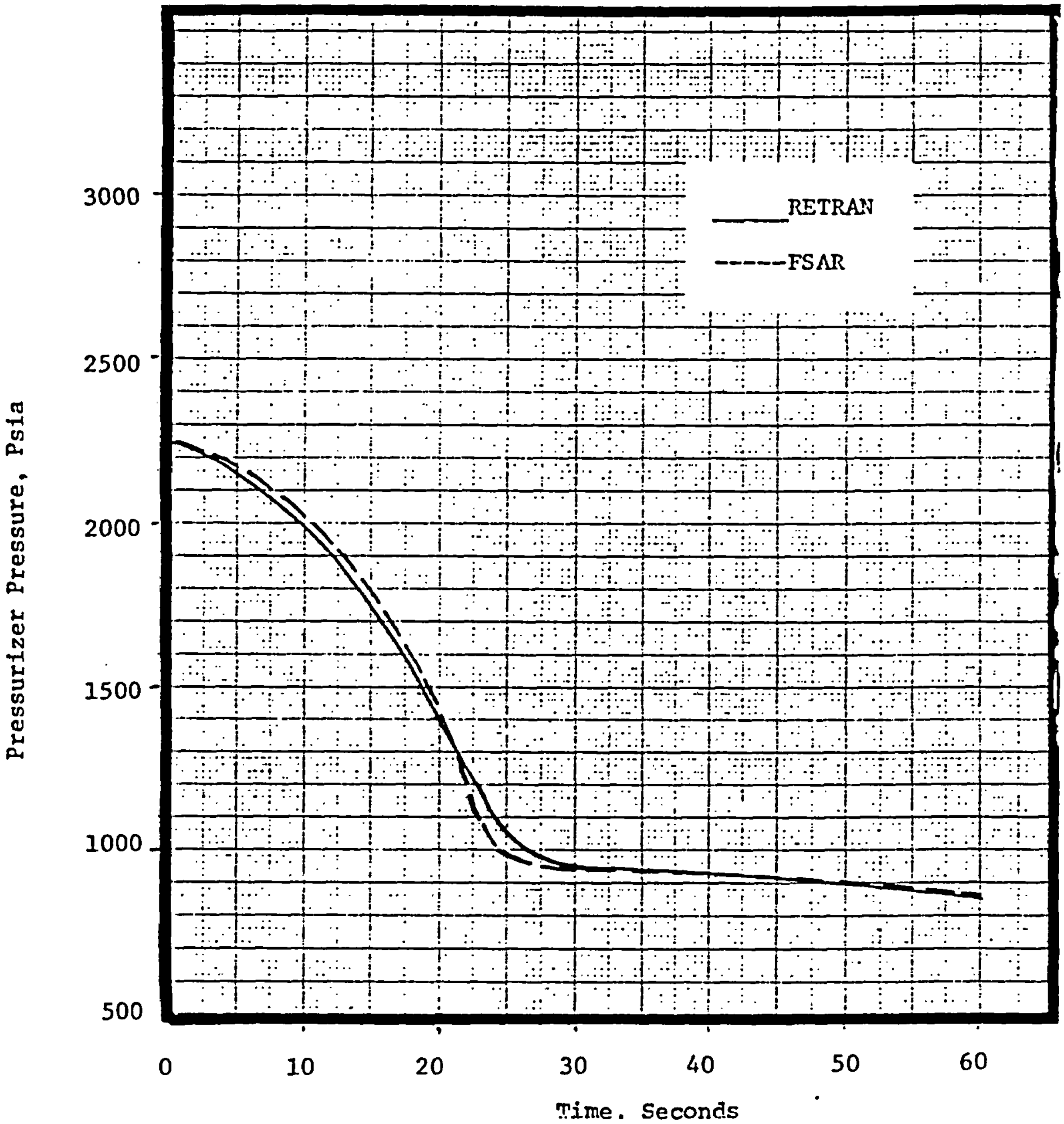


Figure 5.48

TOTAL REACTIVITY
MAIN STEAM LINE BREAK
FSAR ANALYSIS

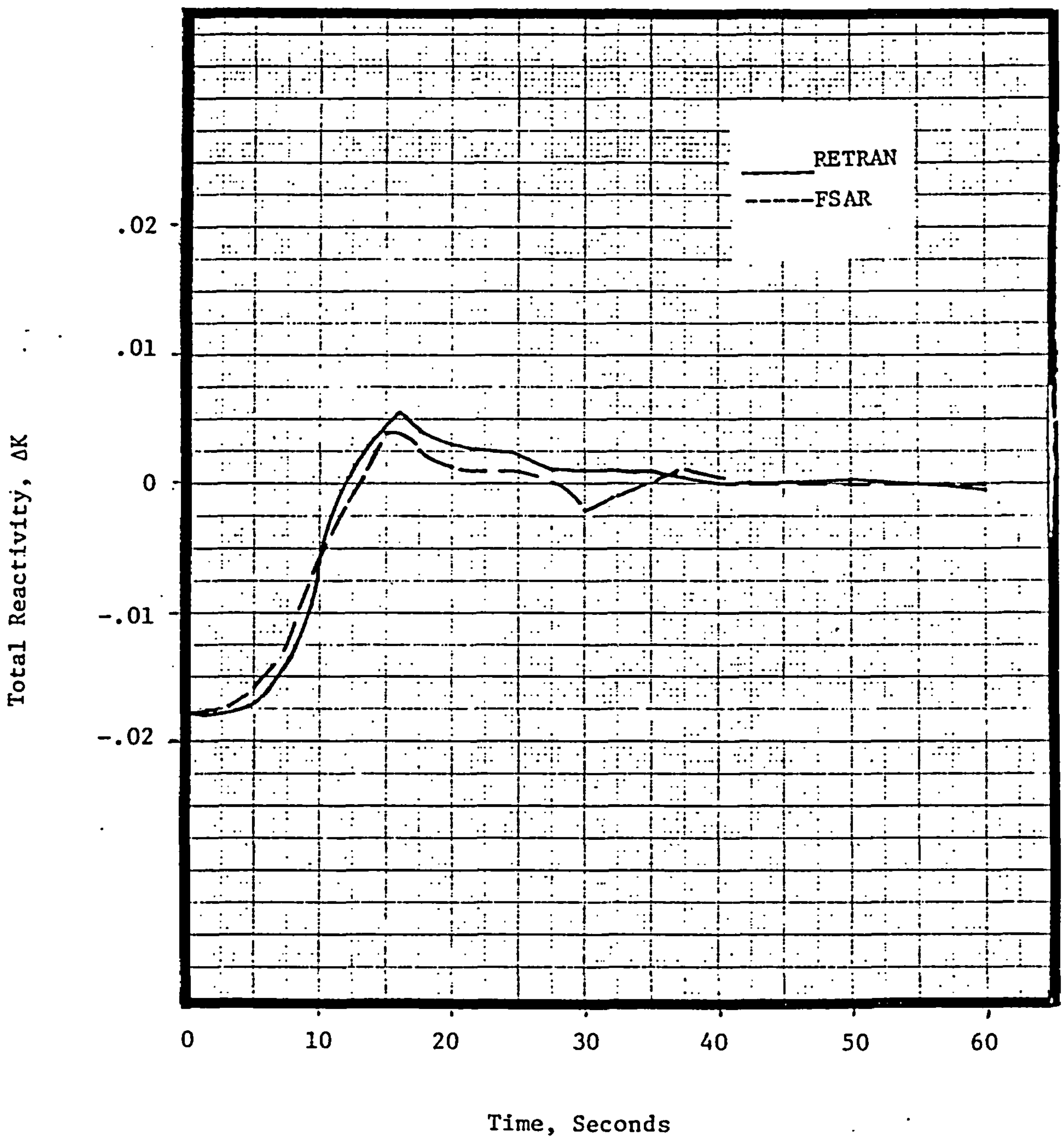


Figure 5.49

CORE HEAT FLUX
MAIN STEAM LINE BREAK
FSAR ANALYSIS

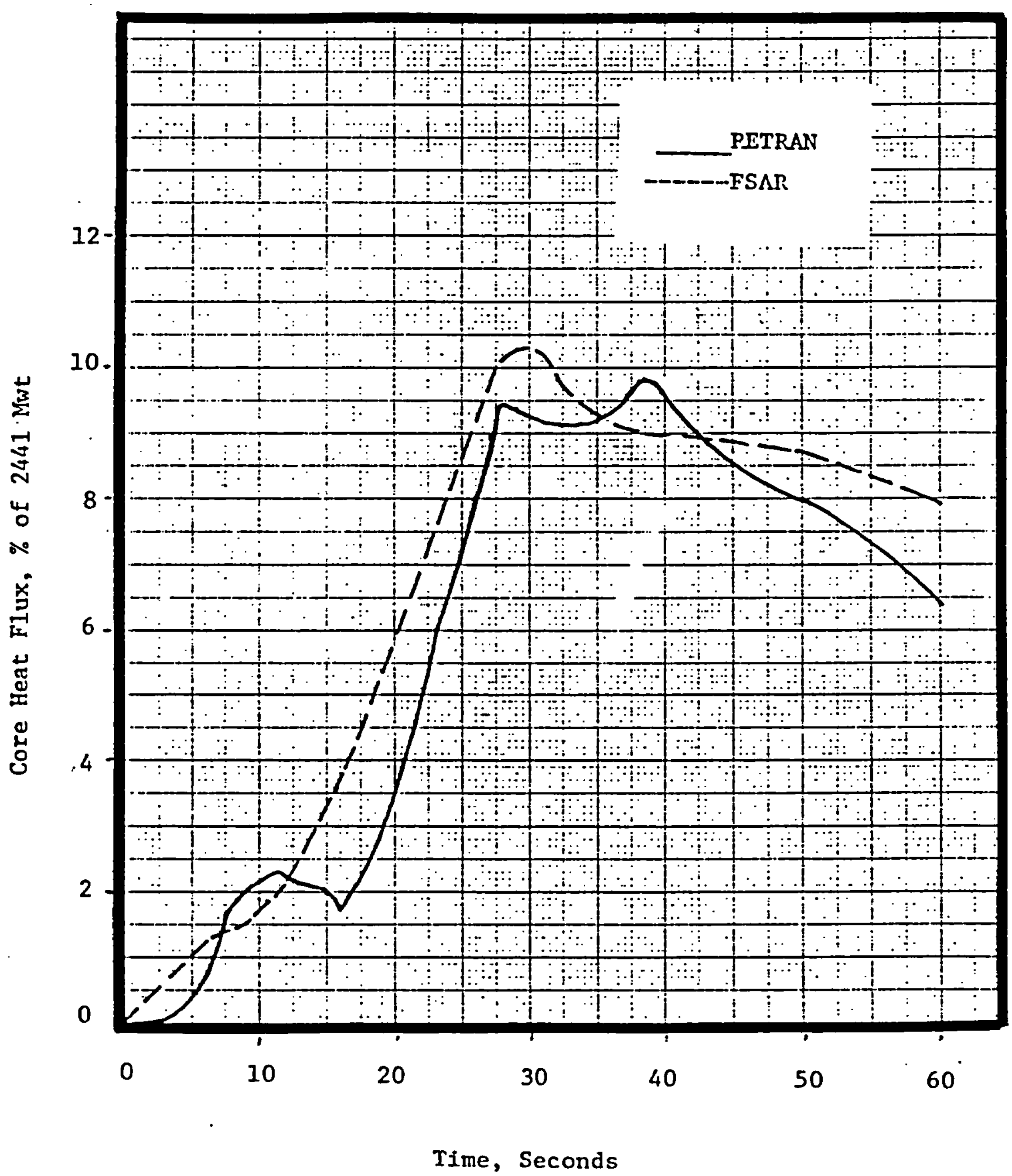


Figure 5.50

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

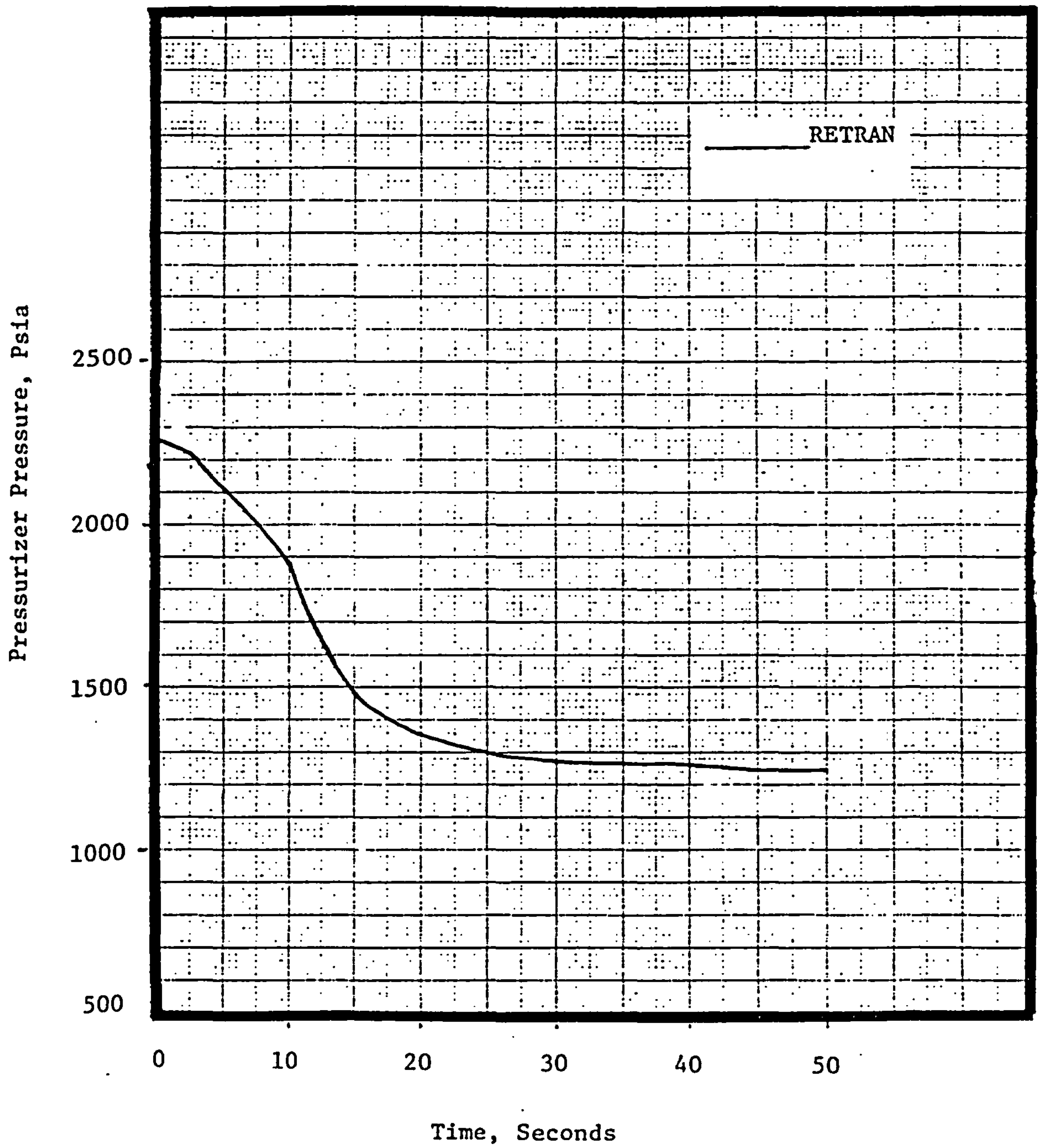
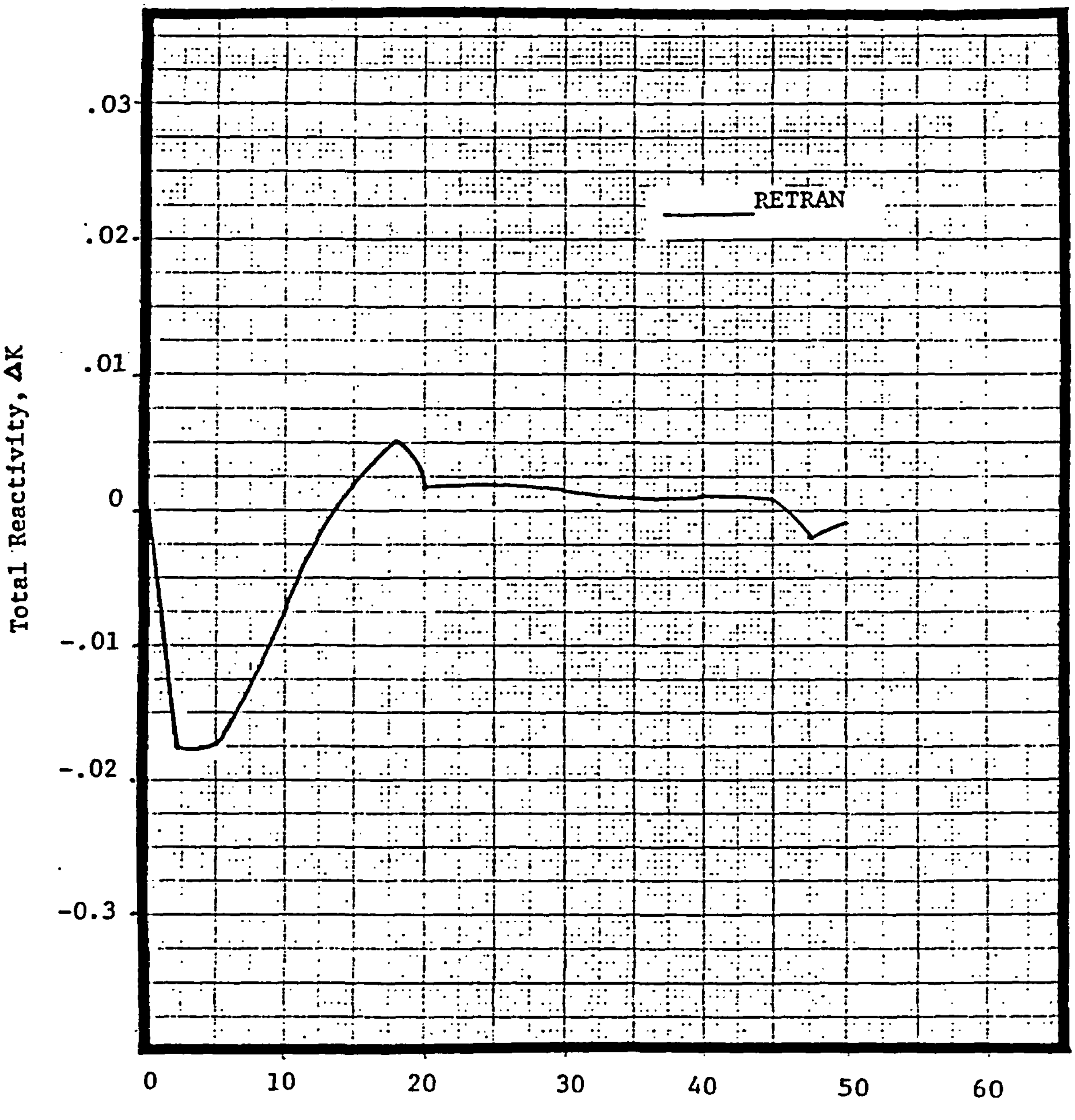


Figure 5.51

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

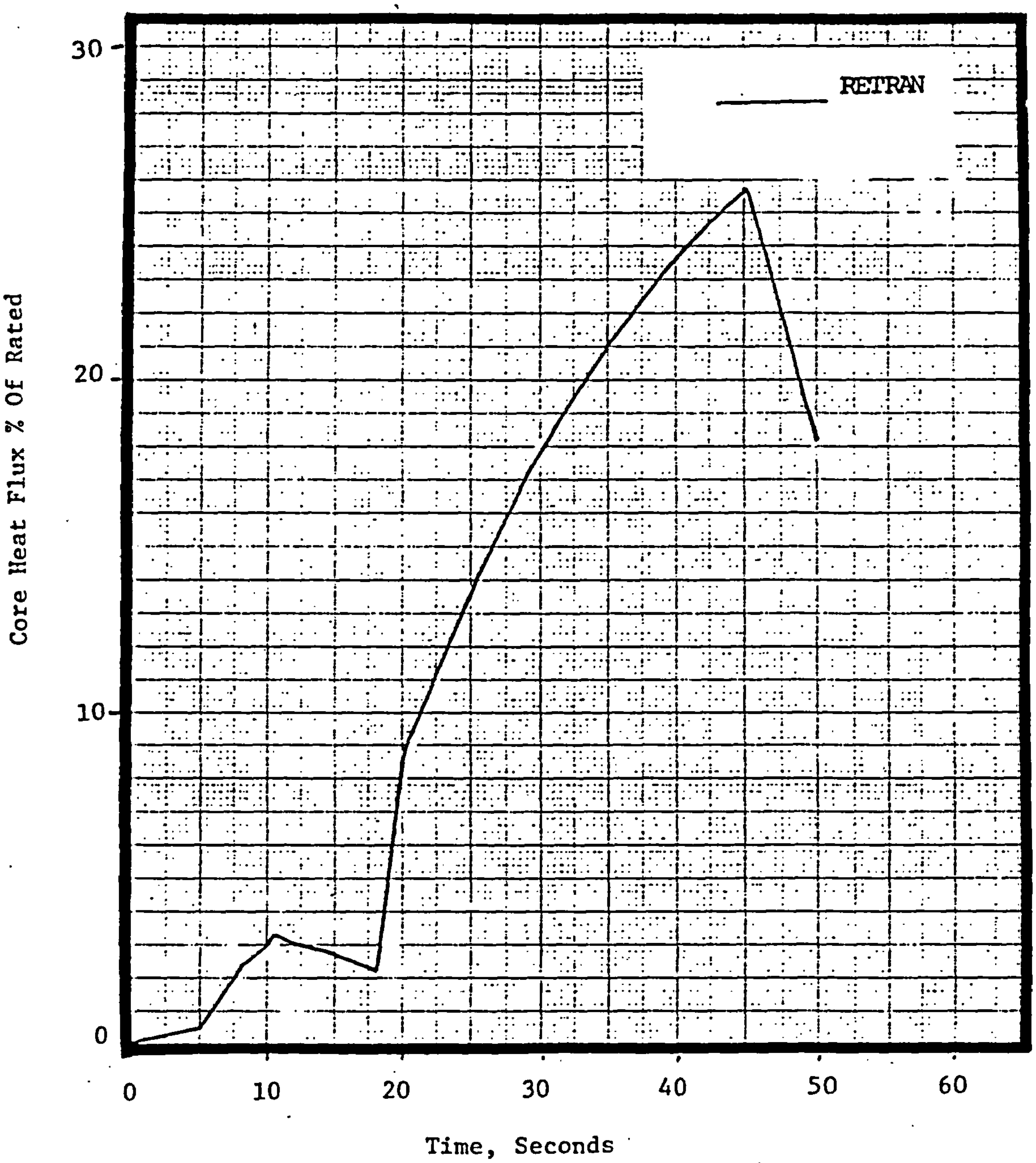


Figure 5.54 compares analytical predictions made with the Surry Two Loop Model with test data for a single pump (two pumps remaining at full speed) coastdown. The data are presented in terms of loop flows. As may be seen, the RETRAN predictions are in close agreement with the data. The same data are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

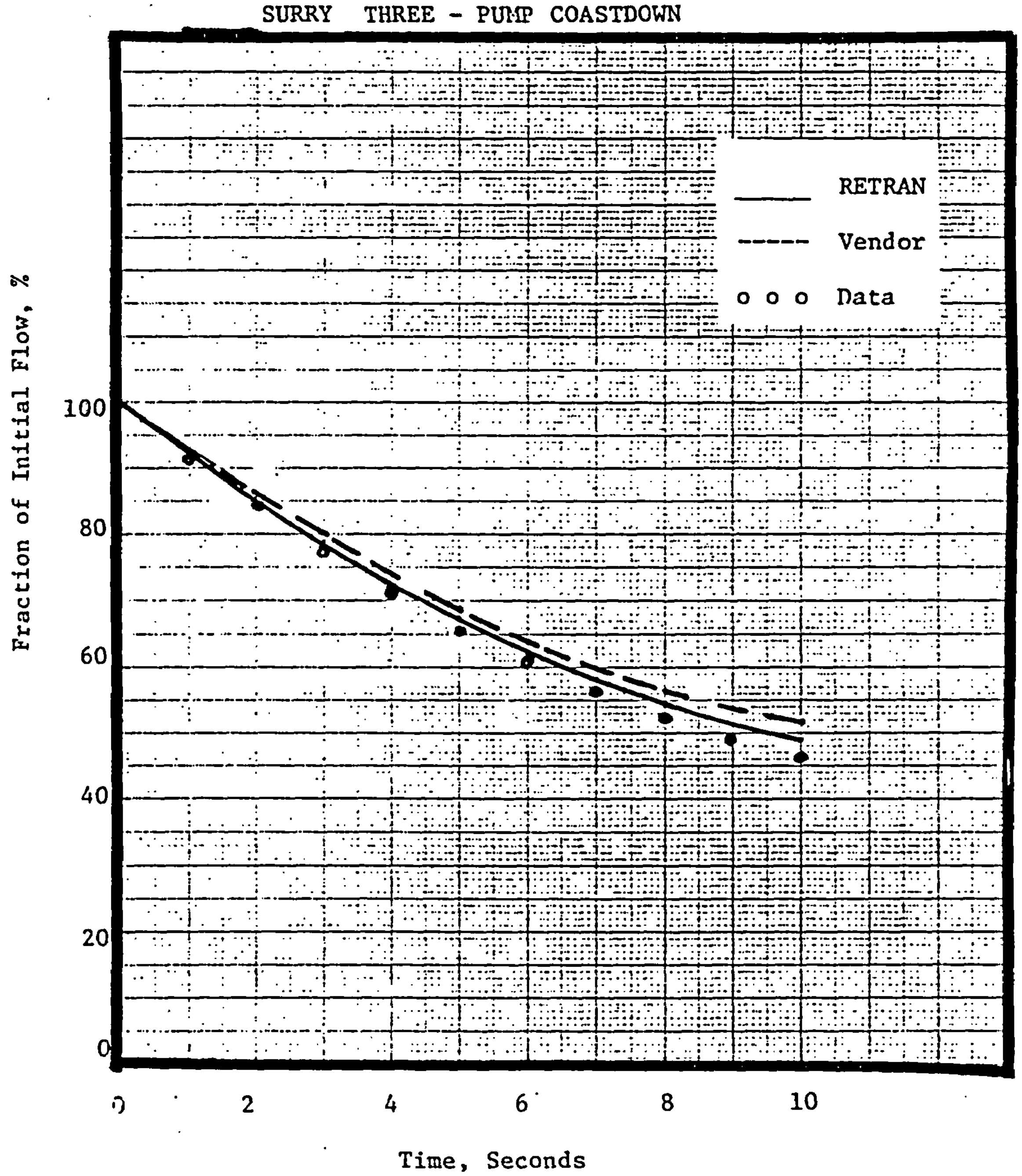


Figure 5.54

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

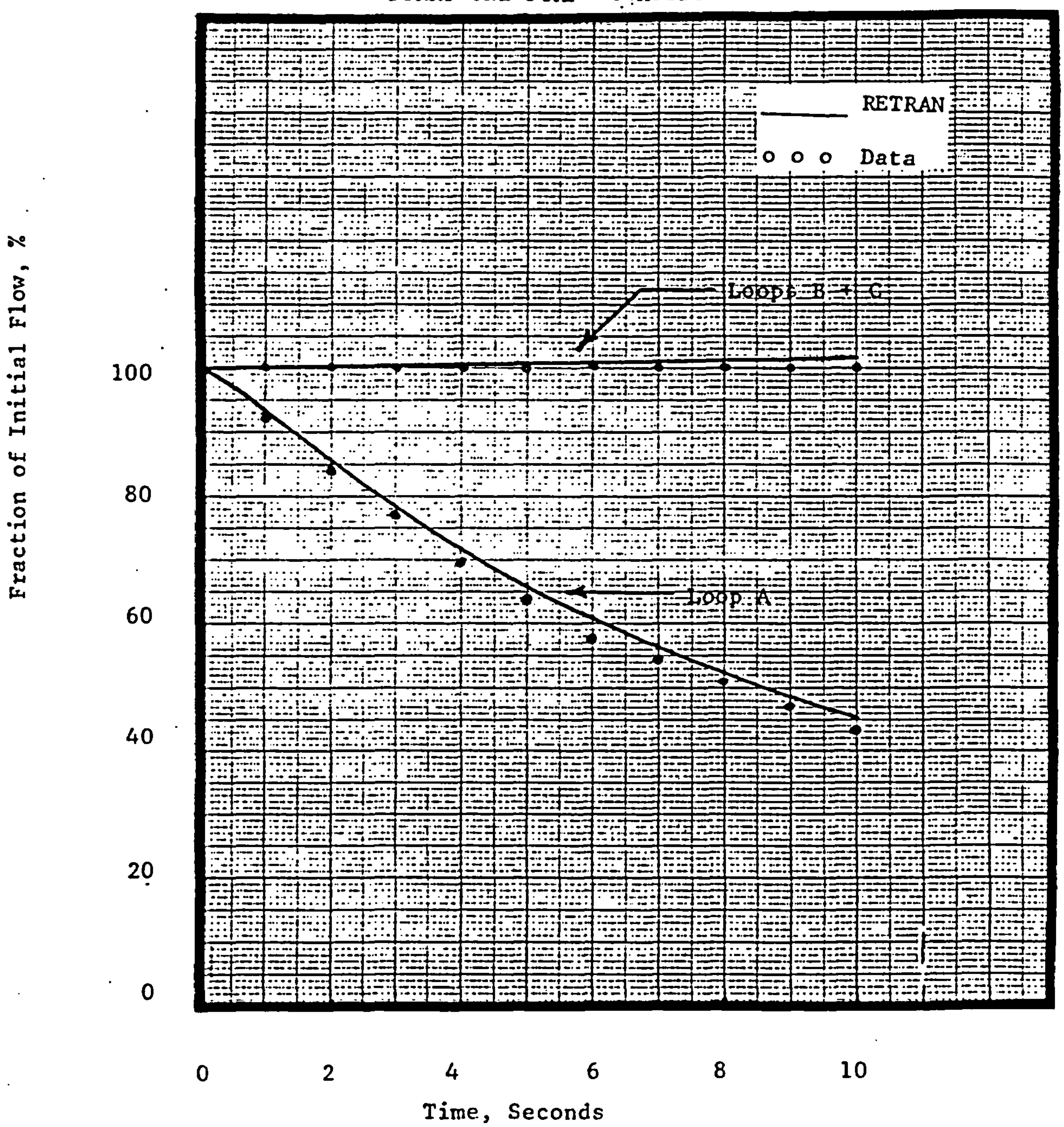


Figure 5.55

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

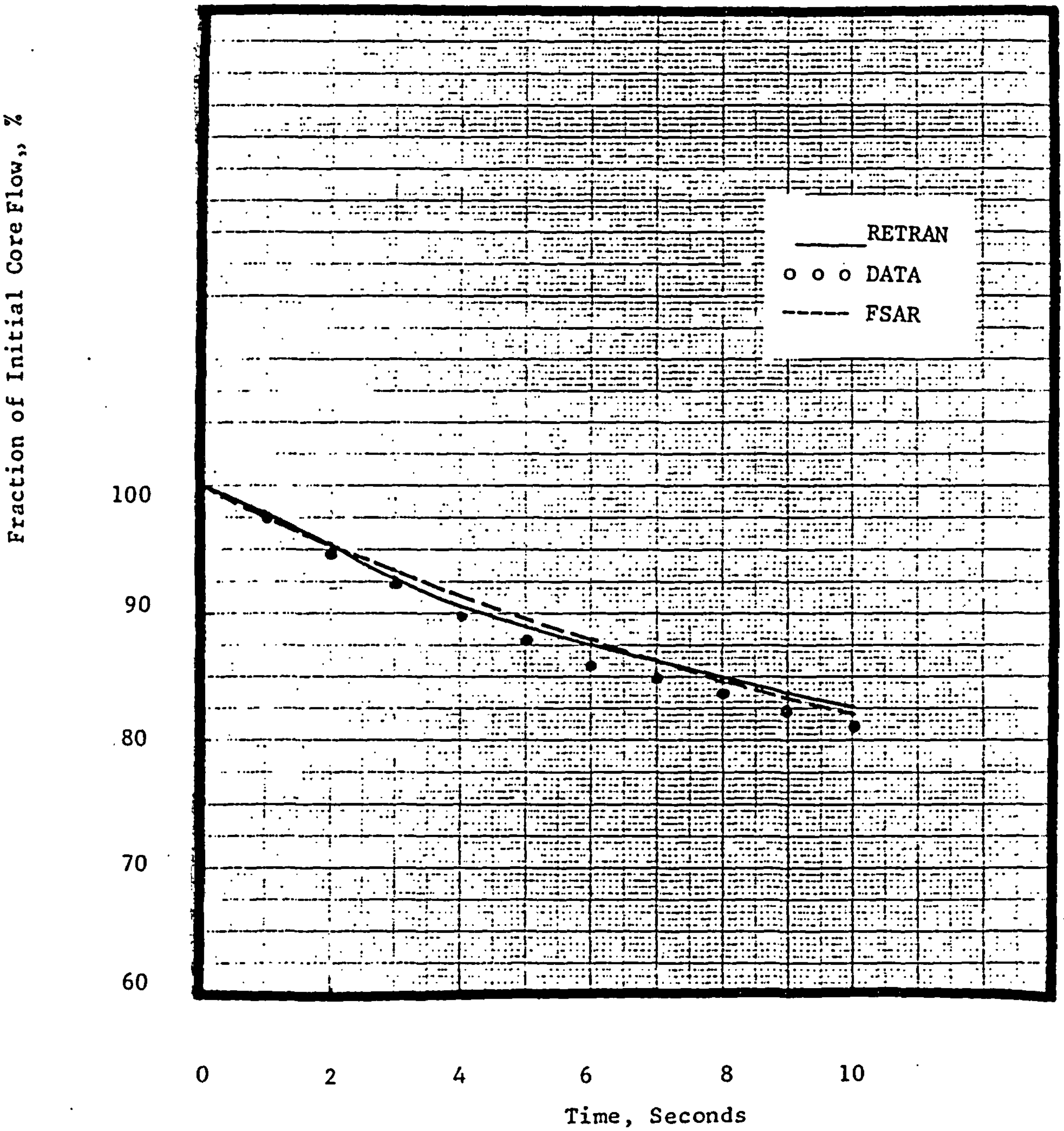


Figure 5.57

STEAM PRESSURE
NORTH ANNA COOLDOWN EVENT

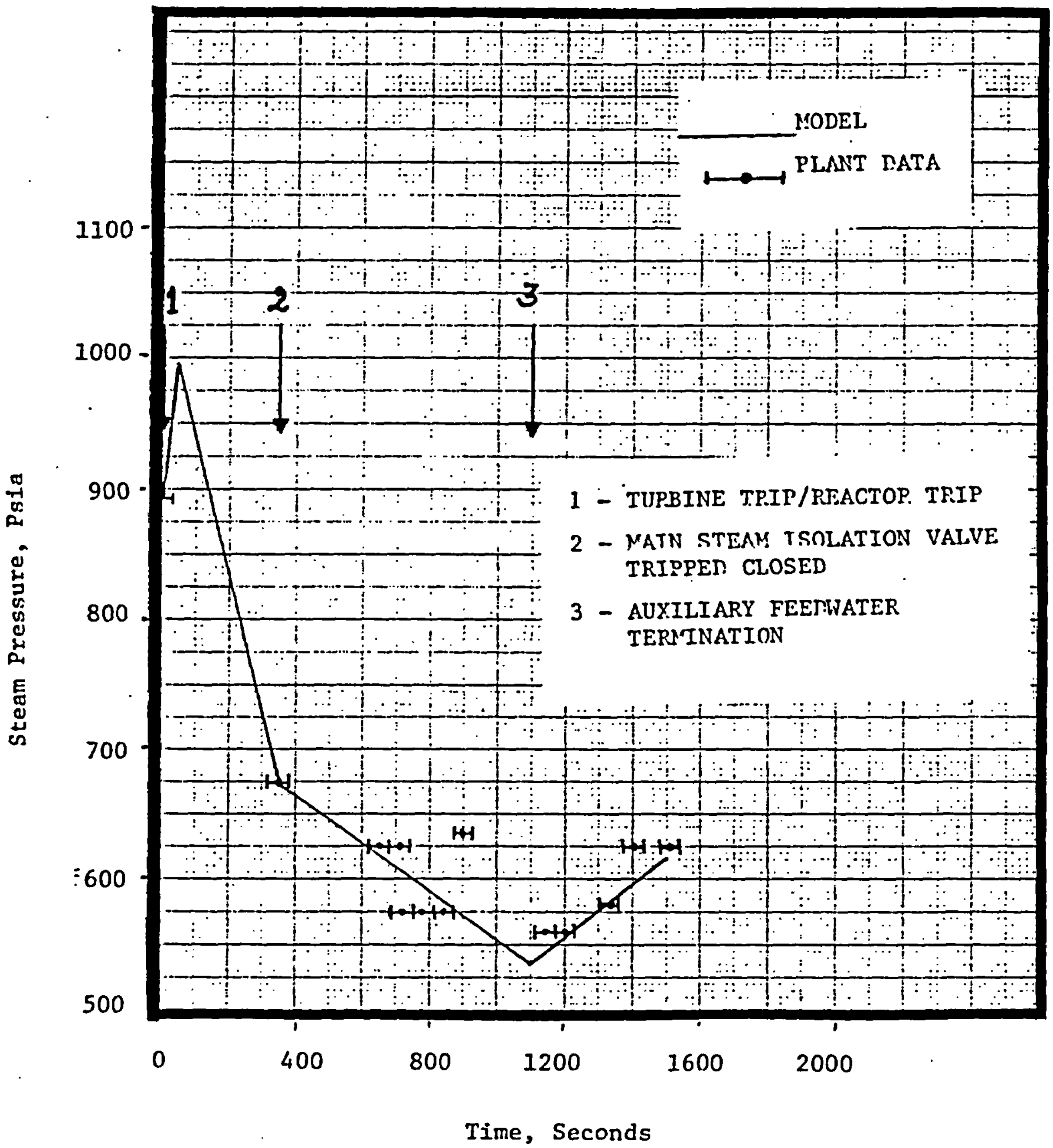
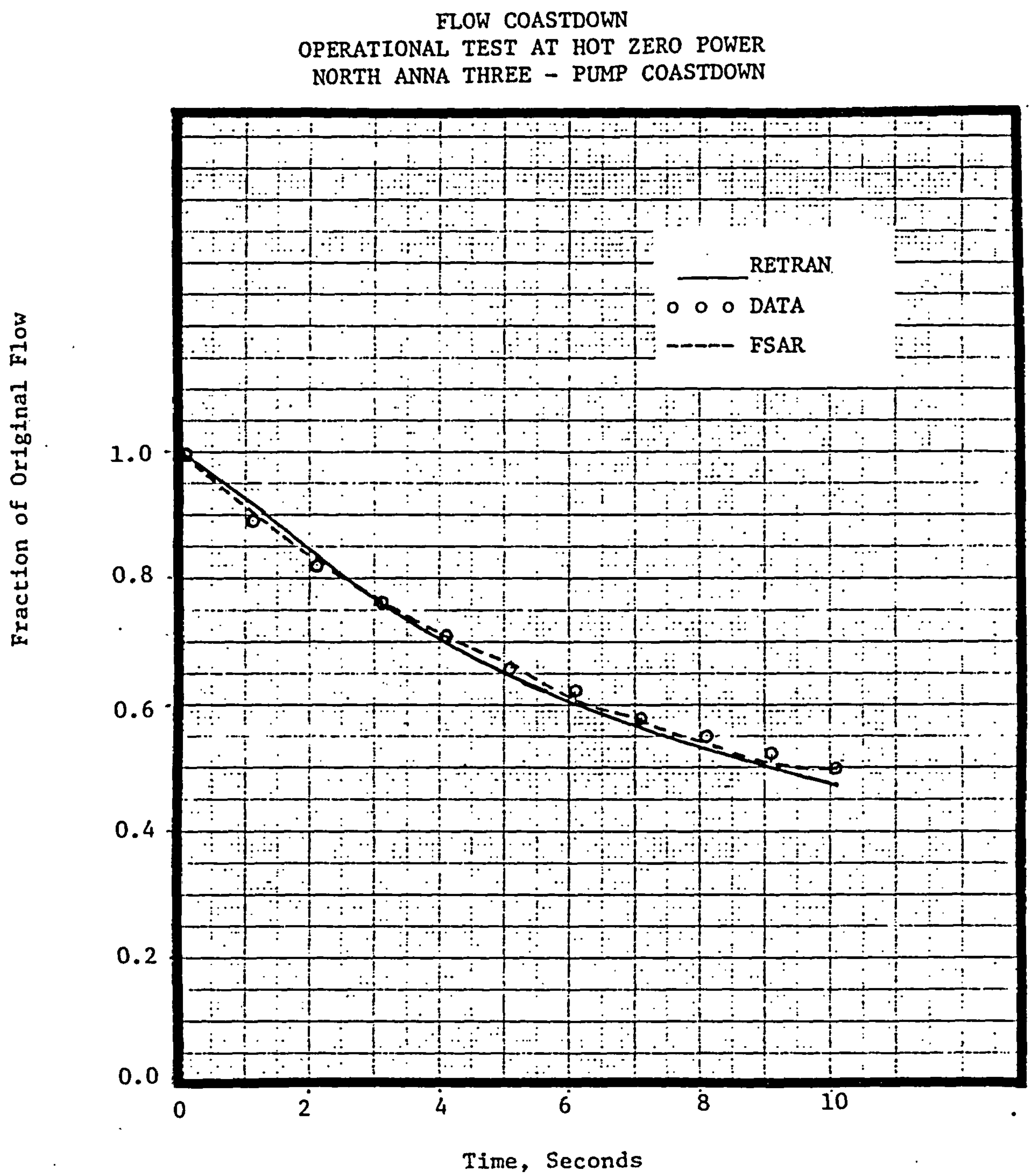


Figure 5.56



FLOW COASTDOWN
OPERATIONAL TEST AT HOT ZERO POWER
NORTH ANNA THREE - PUMP COASTDOWN
RETRAN
o o o DATA
----- FSAR
Fraction of Original Flow
1.0
0.8
0.6
0.4
0.2
0.0
0
2
4
6
8
10
Time, Seconds

Figure 5.58

COLD LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT

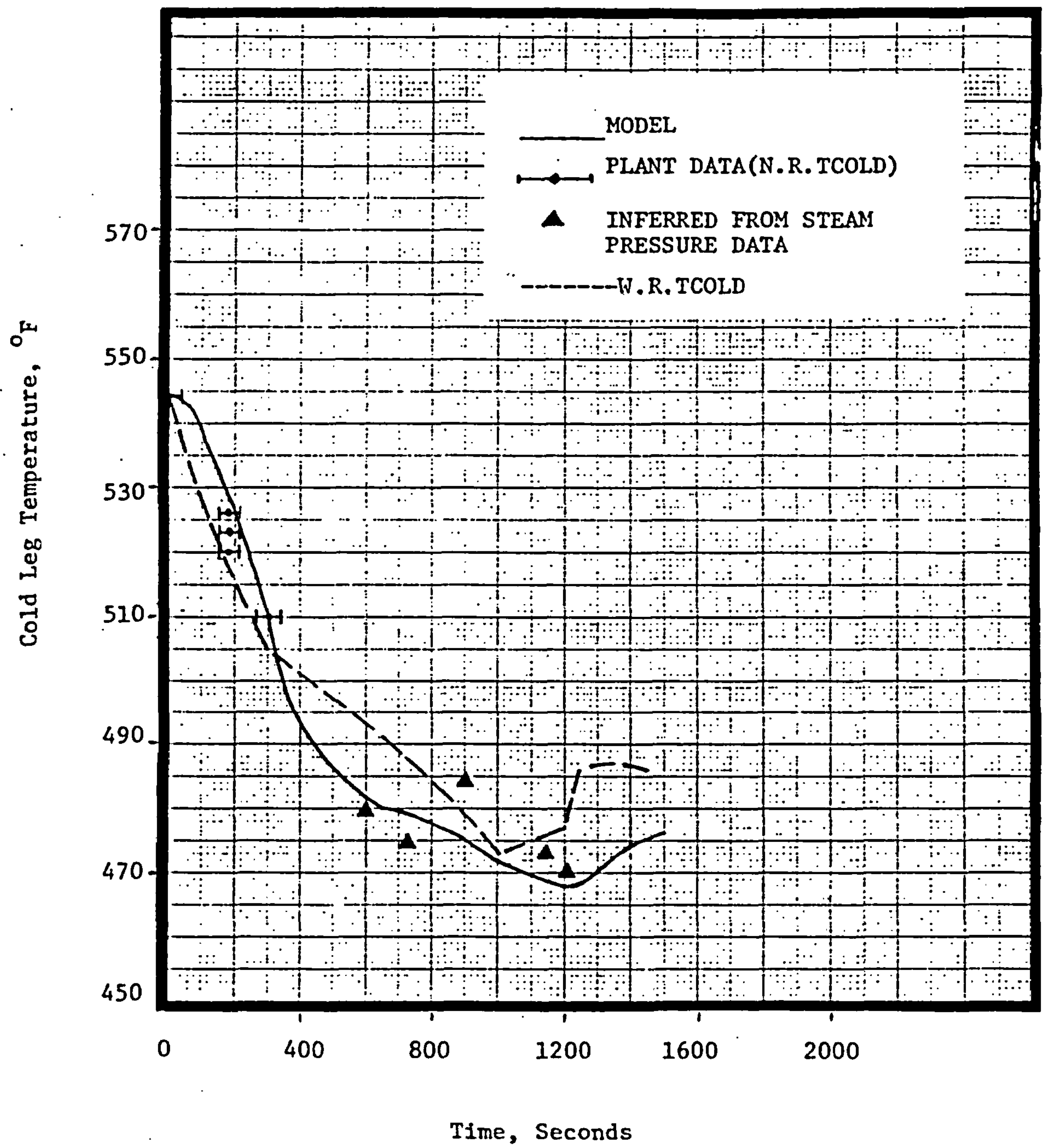


Figure 5.59

HOT LEG TEMPERATURE
NORTH ANNA COOLDOWN EVENT

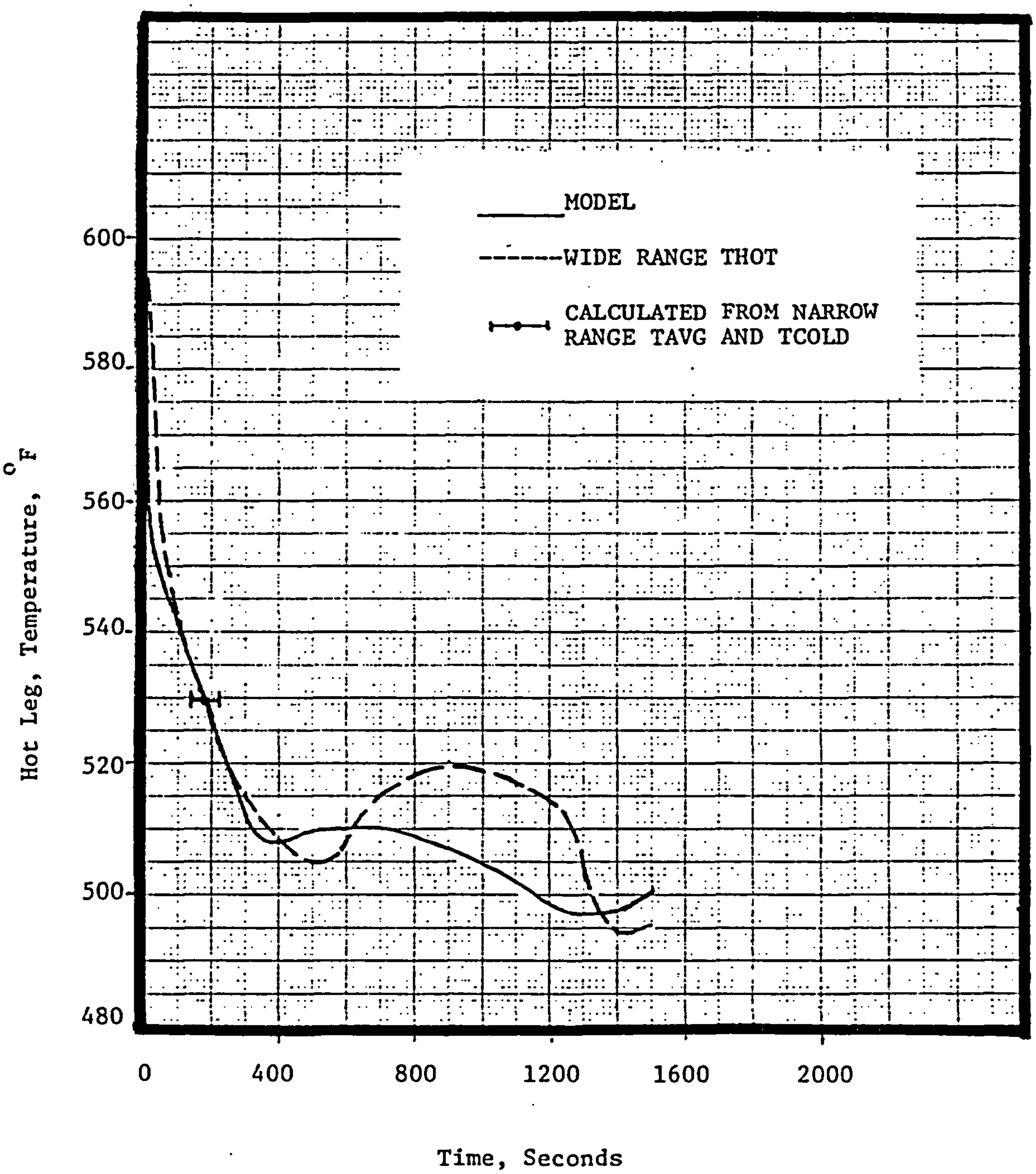


Figure 5.61

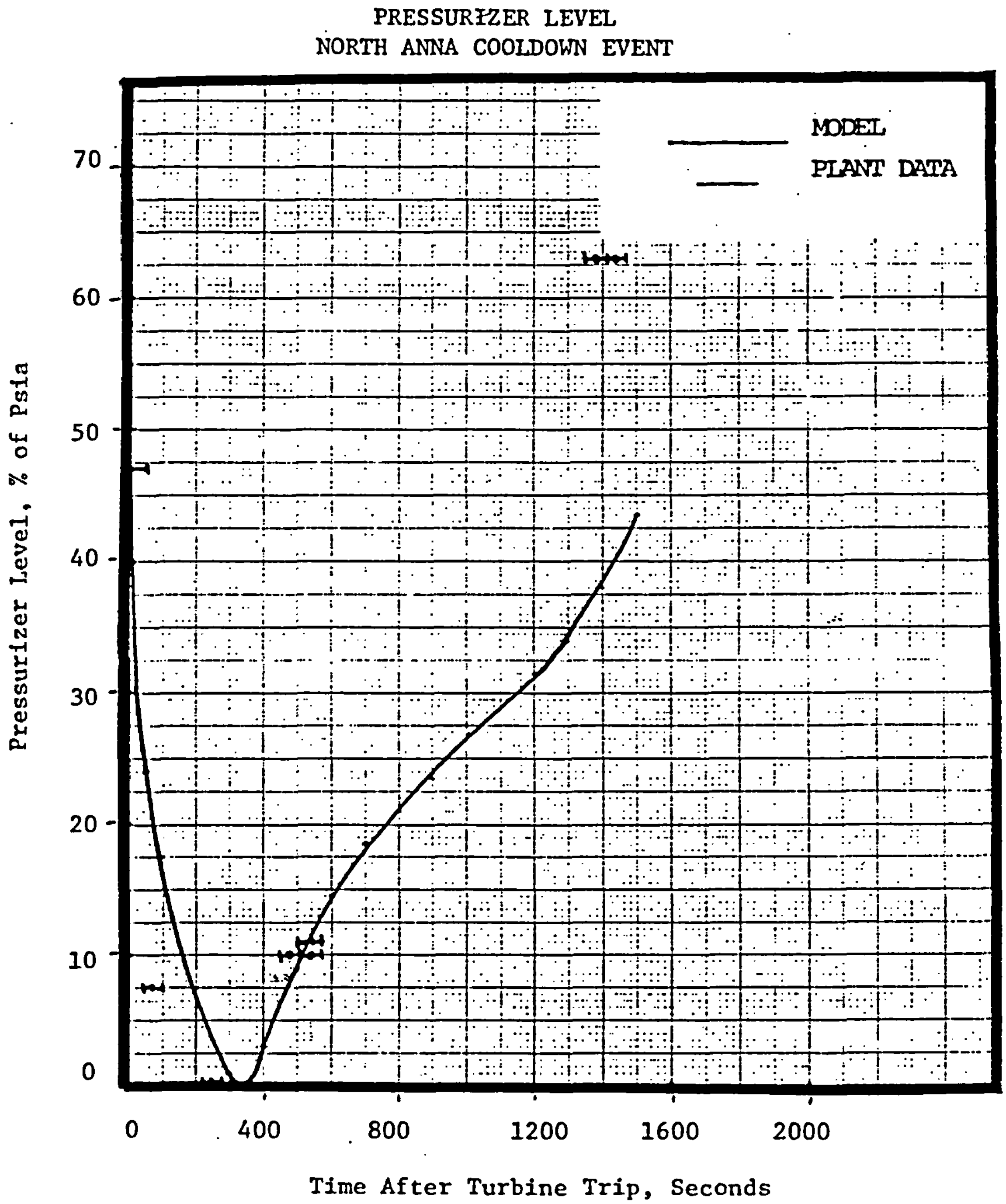
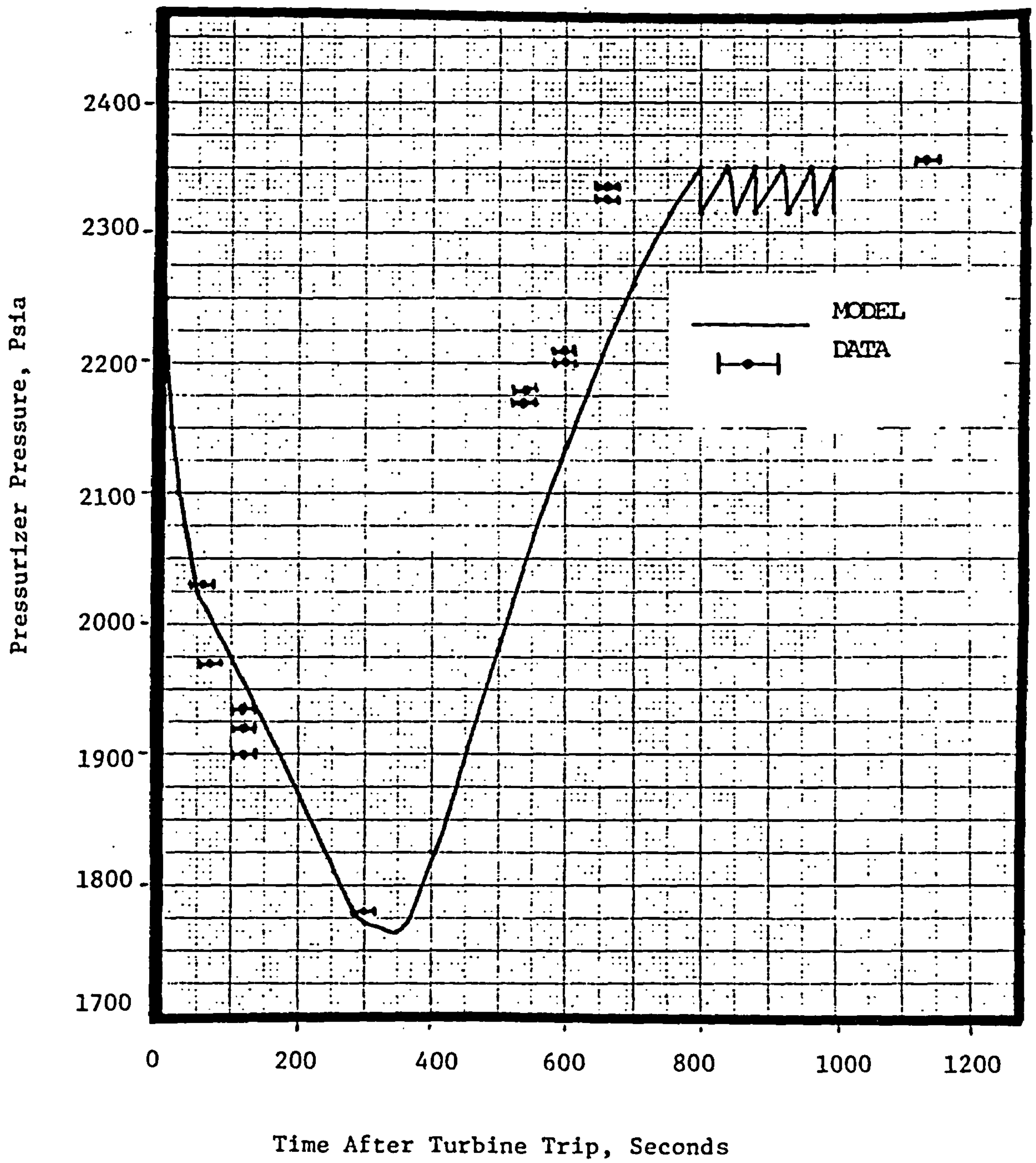

PRESSURIZER LEVEL
NORTH ANNA COOLDOWN EVENT
MODEL
PLANT DATA
70
60
50
40
30
20
10
0
Pressurizer Level, % of Psia
0
400
800
1200
1600
2000
Time After Turbine Trip, Seconds

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

1. VEP-FRD-33, "Vepco Reactor Core Thermal-Hydraulic Analysis Using the COBRA IIC/MIT Compute Code," Virginia Electric and Power Company, August 1979.

2. Electric Power Research Institute Report EPRI CCM-5, "RETRAN - A Program for One-Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," Energy Incorporated, December 1978.

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.

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

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

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

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

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

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

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

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

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

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

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

*1 pcm=1.0 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^{o}F	αMOD = 0 αDOPPLER =-1.6 pcm/^{o}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^{o}F	αMOD=+3.0 pcm/^{o}F αDOPPLER=-1.6 pcm/^{o}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^{o}F	αMOD = 0.0 αDOPPLER=-1.6 pcm/^{o}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 1) From 102% power	Surry One Loop (Modified to reflect steam generator tube plugging)	Core power = 1.02 x rated Pressure = 2220 psia T-inlet = 543.4°F RCS Flow = 90% of full power thermal design	αMOD=+1.0 pcm/°F Doppler power coefficient =-6.0 pcm/% at 100% power Reactivity insertion rate varied Trip Reactivity: Figure A.1 curve (a), total = 2.8% ΔK/K	No credit taken for: 1) High neutron flux rod stop 2) High Overtemperature ΔT rod stop 3) High overpower ΔT rod stop or trip Source of protection: High power range high neutron flux trip or High overtemperature ΔT trip
2) From 62% power		Core power = 0.62xrated Pressure = 2220 psia T-inlet = 550.3°F RCS Flow = 90% of full power thermal design	Trip Reactivity: Fig. 4.1 curve (a), total=2.8%ΔK/K αMOD=+1.0pcm/°F αDOP =-7.3 pcm/% at 62% power Reactivity insertion rate varied	Assumptions same as high power case

ΔT trip equation used (includes errors):

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

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

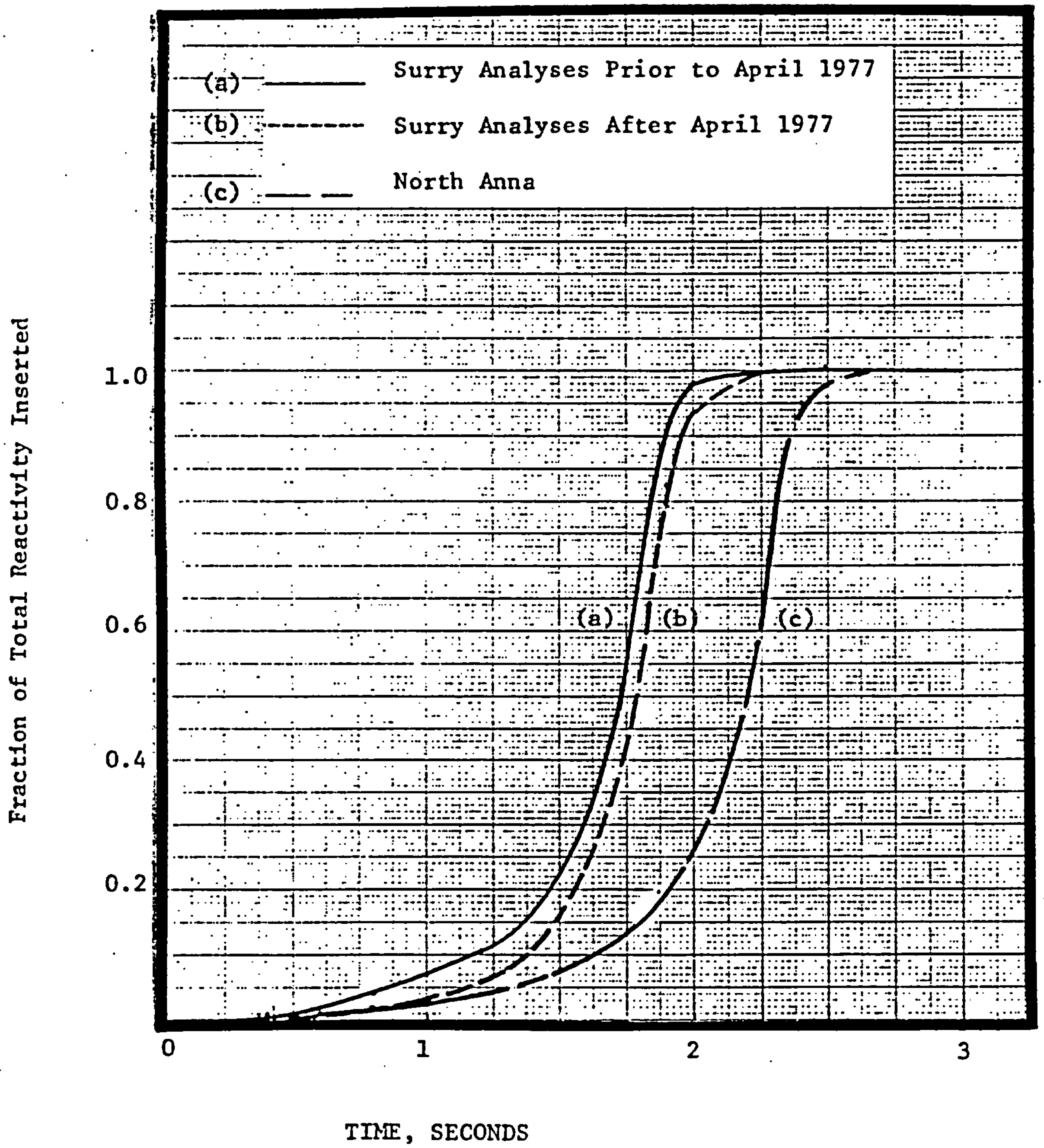
			Transient Specific Input	
Type of Analysis	System Model Description	Initial Conditions	Reactivity Parameters	Key System Performance Assumptions
6. Excessive Heat Removal Due to Feedwater System Malfunction	Surry One Loop	Core power = 1.02xrated Pressure = 2220 psia T-inlet = 547.2^{o}F	αMOD = 0.0 αDoppler=-1.0pcm/^{o}F 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^{o}F	αMOD=-25.4 pcm/^{o}F @)550^{o}F, -13.8 pcm/^{o}F @)300^{o}F αDOPPLER(Zero power) =-1.6pcm/^{o}F Total power defect at 30% power=-.0135ΔK Differential boron worth =- 10pcm/ppm	Technical Specifications value for initial shutdown reactivity margin assumes the highest worth control rod assembly stuck in its fully withdrawn position Safety injection capability based on failure of one high-head safety injection pump No credit is taken for the effect of the main steam line check valves in precluding discharge of secondary fluid from the intact steam generators prior to main steam isolation valve closure

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

Type of Analysis	System Model Description	Initial Conditions	Transient Specific Input: Reactivity Parameters	Transient Specific Input: Key System Performance Assumptions
b) Current Analysis	Surry Two Loop	Core power=$4x10^{-8}$xrated Pressure = 2251 psia T-inlet = 549.7°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

			Transient Specific Input	
Type of Analysis	System Model Description	Initial Conditions	Reactivity Parameters	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
Responses to VEP-FRD-41 Rev. 0 RAI #1

VIRGINIA ELECTRIC AND POWER COMPANY

RECEIVED RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

'84 MAR -3 :02

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

A001
1/1

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

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

TABLE OF CONTENTS

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	Przr. 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
Przr. - 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	N	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
- 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

APPENDIX 3
Responses 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.

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

I. RETRAN VOLUME AND FLOW PATH NETWORK DESCRIPTION

TABLE OF CONTENTS

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	Przr. 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
- Przr. - 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.
HEAT 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 INT-EGRAL 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 PRES-SURE MEAUREMENT ERRORS, DEPEN-DING 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 S. 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	-	NODEL = 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 COEF- FICIENT	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 CALCULAT- IONS (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.

Past 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 that 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	Loss of Load
Heat Transfer	Excessive Load Increase
Decrease in RCS Flow Rate	Complete Loss of Flow

VI.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, was 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, 1*	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) the 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 with 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

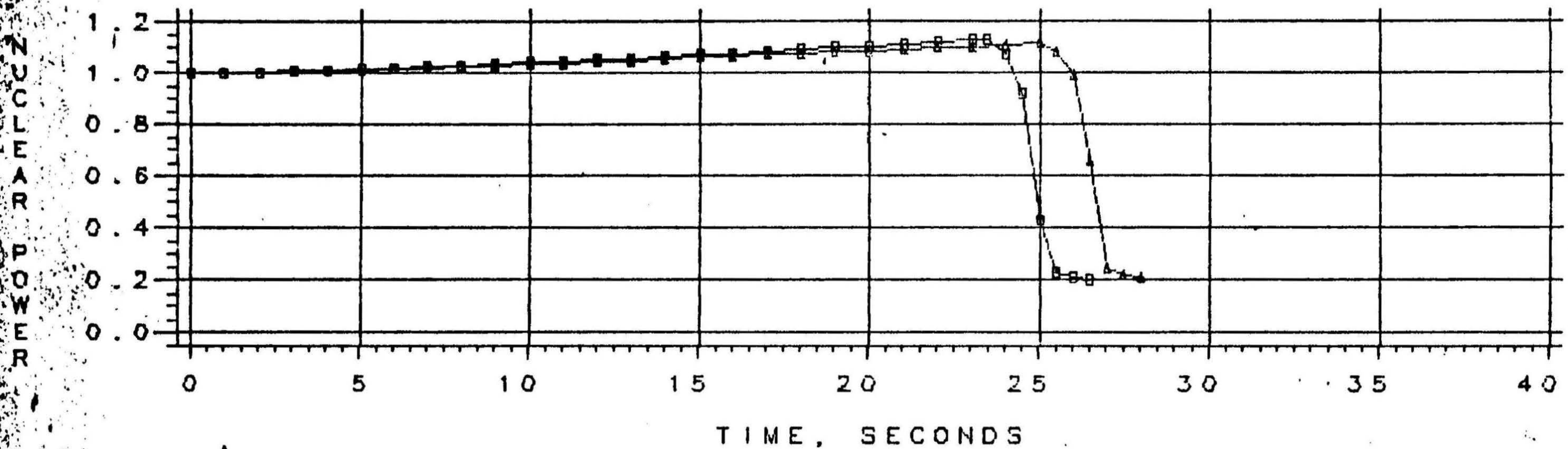
Parameter	Base Case	Sensitivity Case
Moderator Temperature Coefficient	Most Negative	0.6 * Most Negative
Doppler Power Coefficient	Least Negative	1.25 * Least Negative
Pressurizer Heaters	Inactive	Active

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.

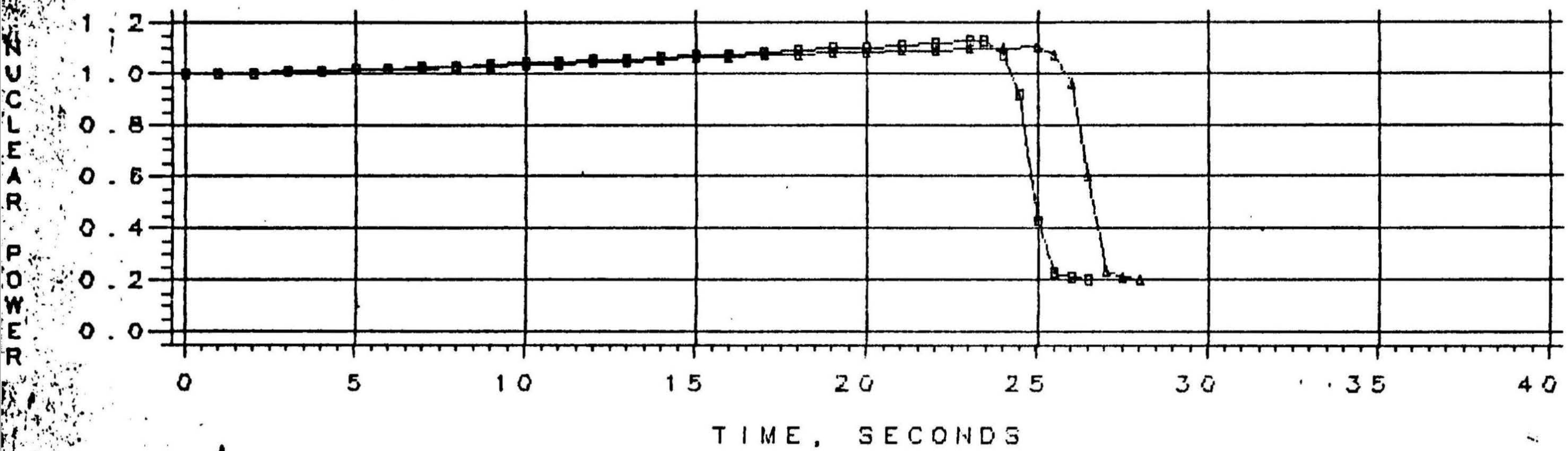
FIGURE VI-1
ROD WITHDRAWAL AT POWER
DPC STUDY



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

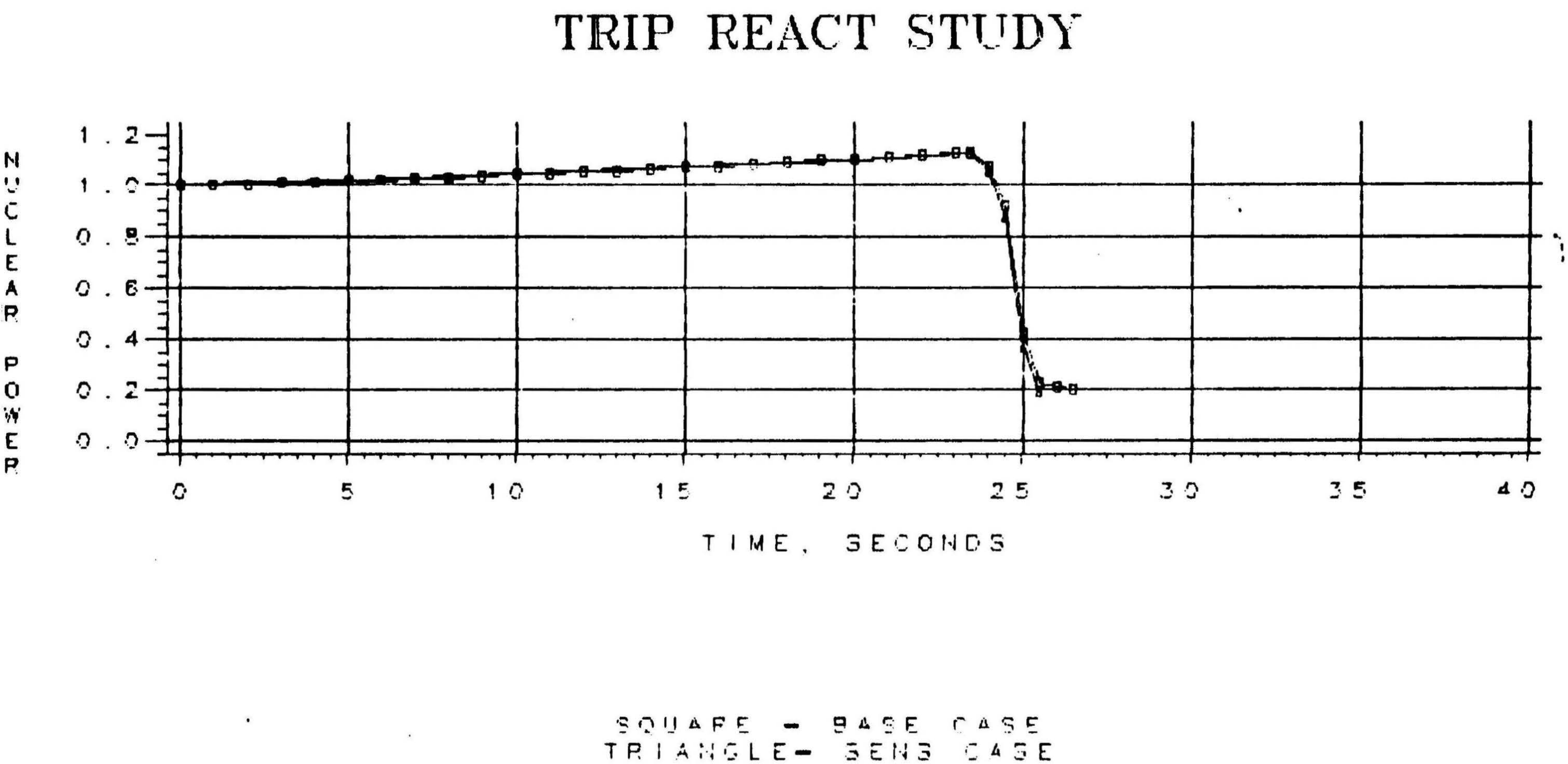


FIGURE VI-4
ROD WITHDRAWAL AT POWER

OTDT STUDY

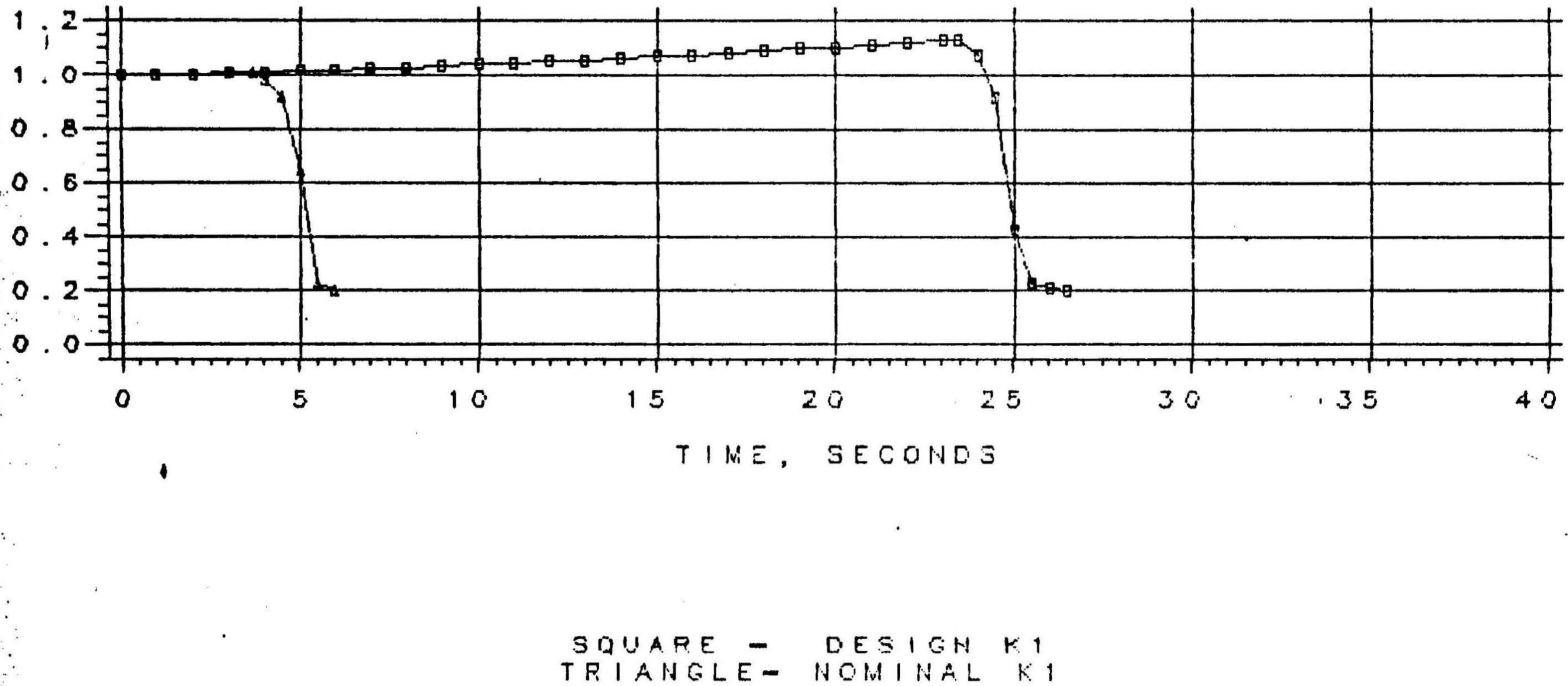
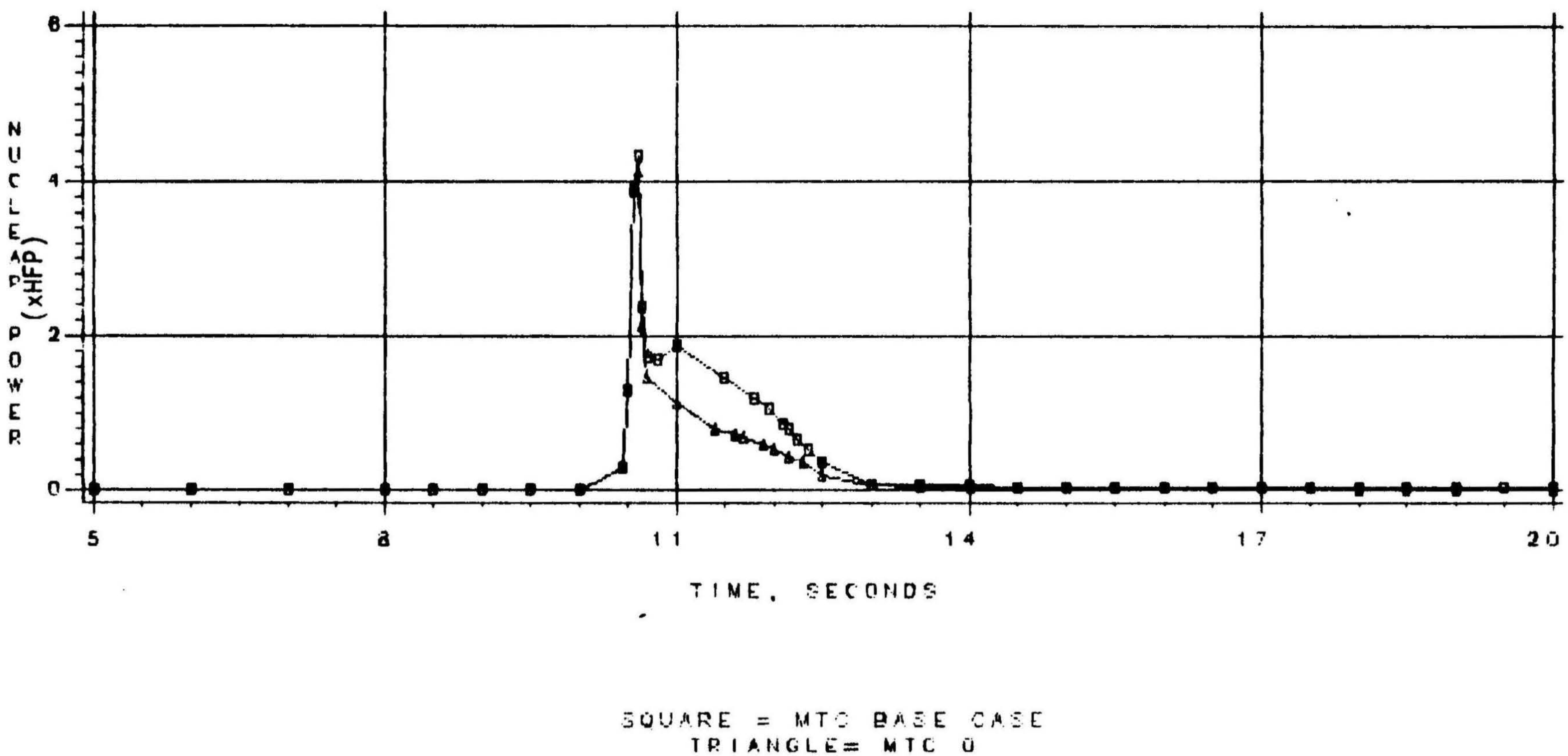


FIGURE VI-5
ROD WITHDRAWAL FROM SUBCRITICAL
MTC STUDY



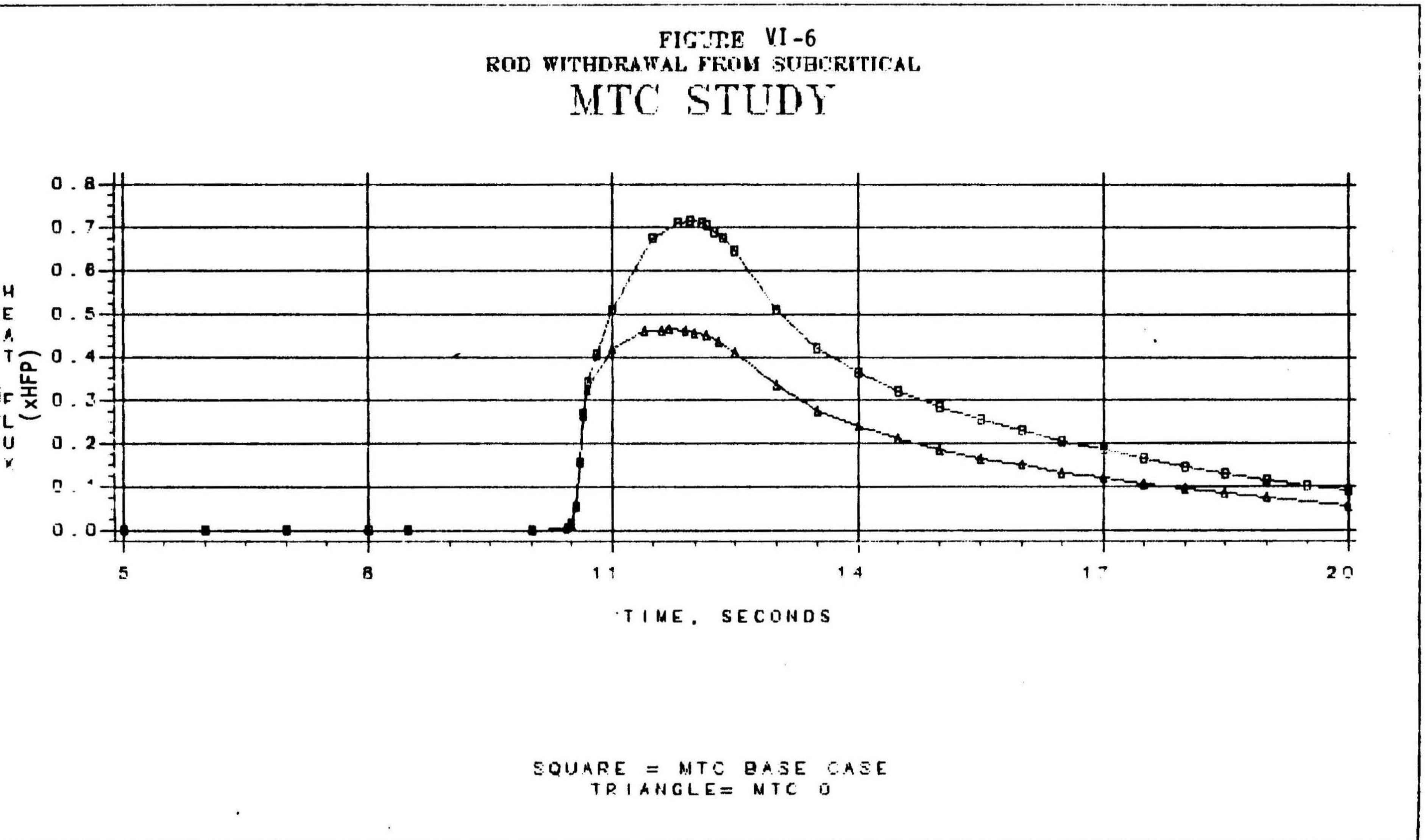


FIGURE VI-6
ROD WITHDRAWAL FROM SUBCRITICAL
MTC STUDY
HEAT FLUX (xHFP)
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
5
8
11
14
17
20
TIME, SECONDS
SQUARE = MTC BASE CASE
TRIANGLE= MTC 0

FIGURE VI-7
ROD WITHDRAWAL FROM SUBCRITICAL
MTC STUDY

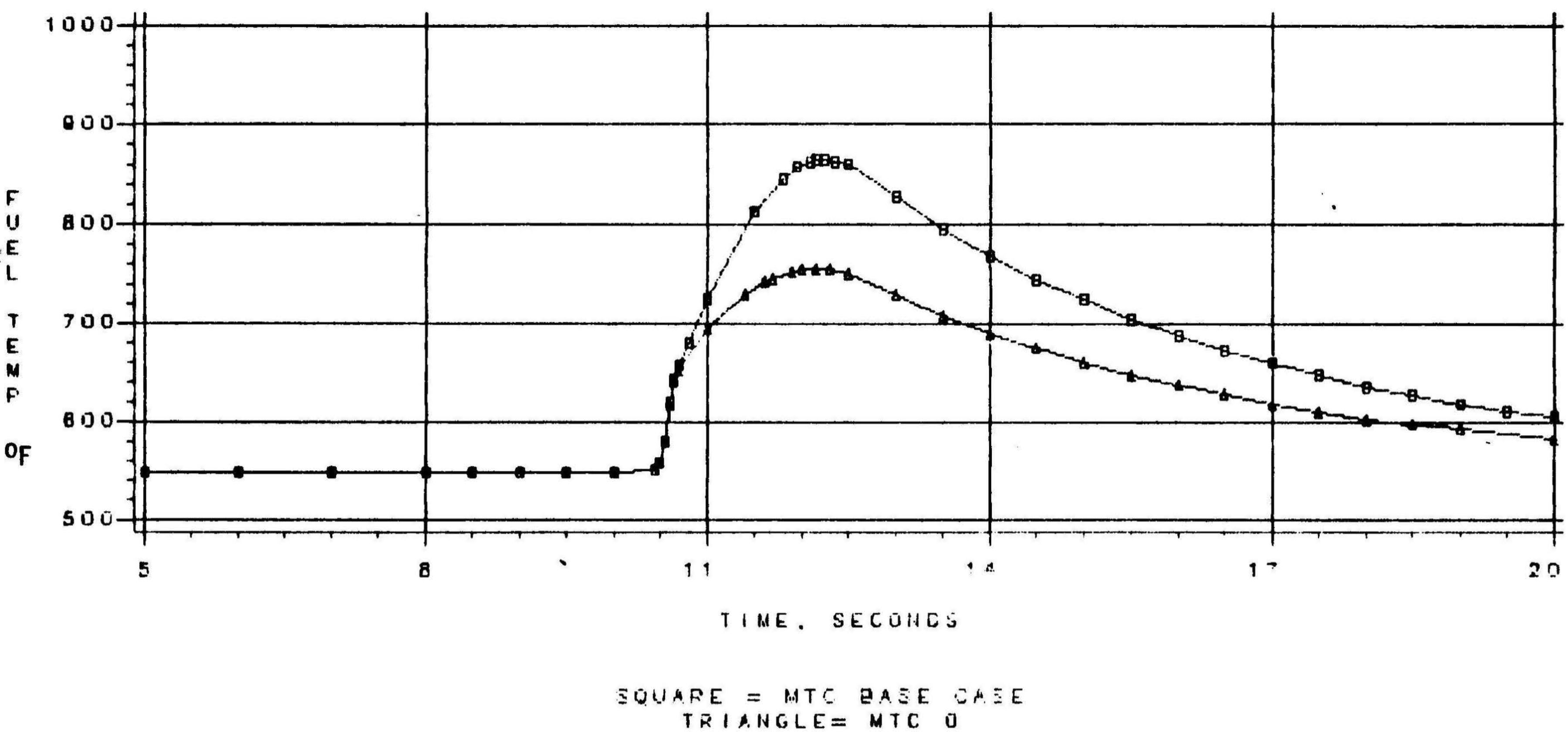
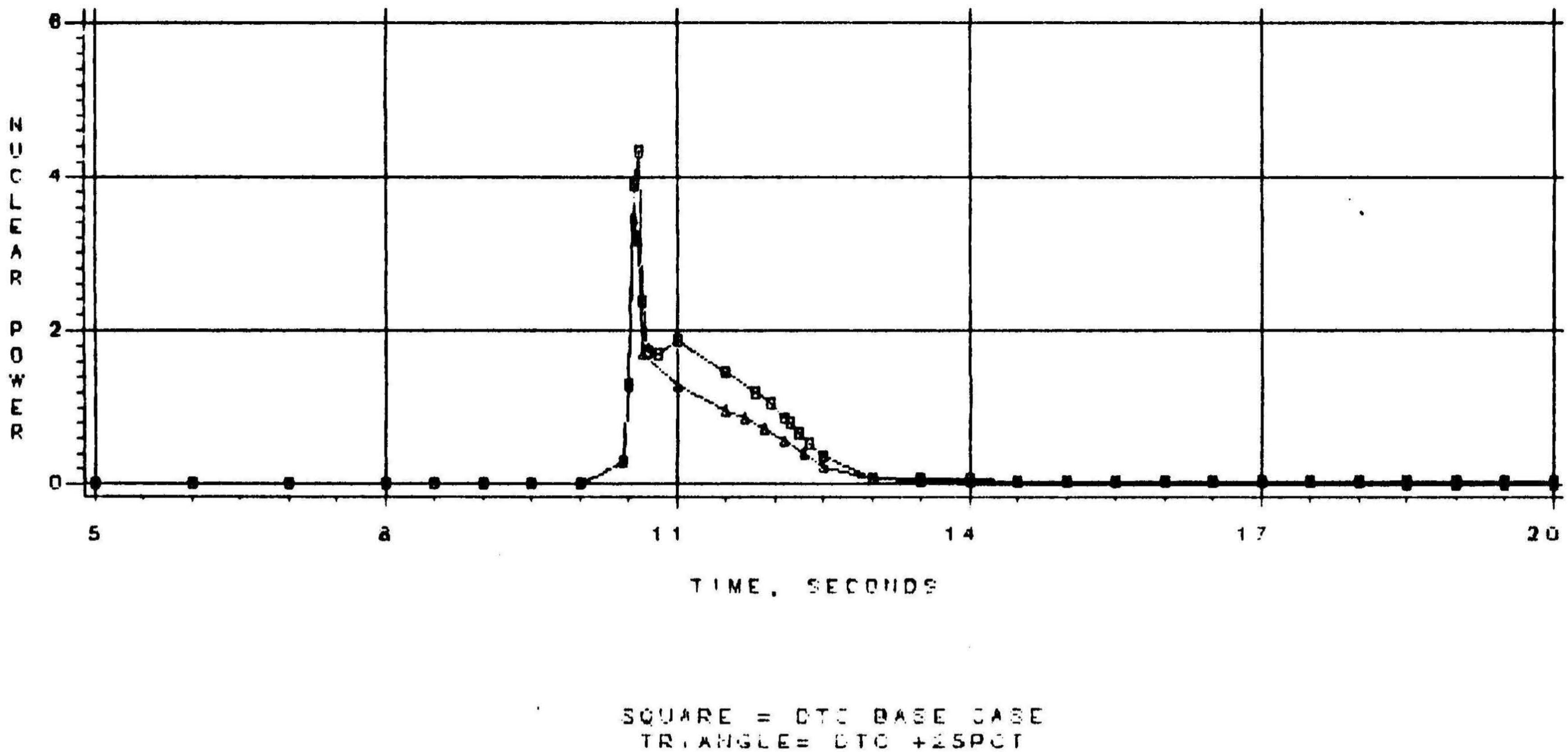


FIGURE VI-8
ROD WITHDRAWAL FROM SUBCRITICAL
DTC STUDY



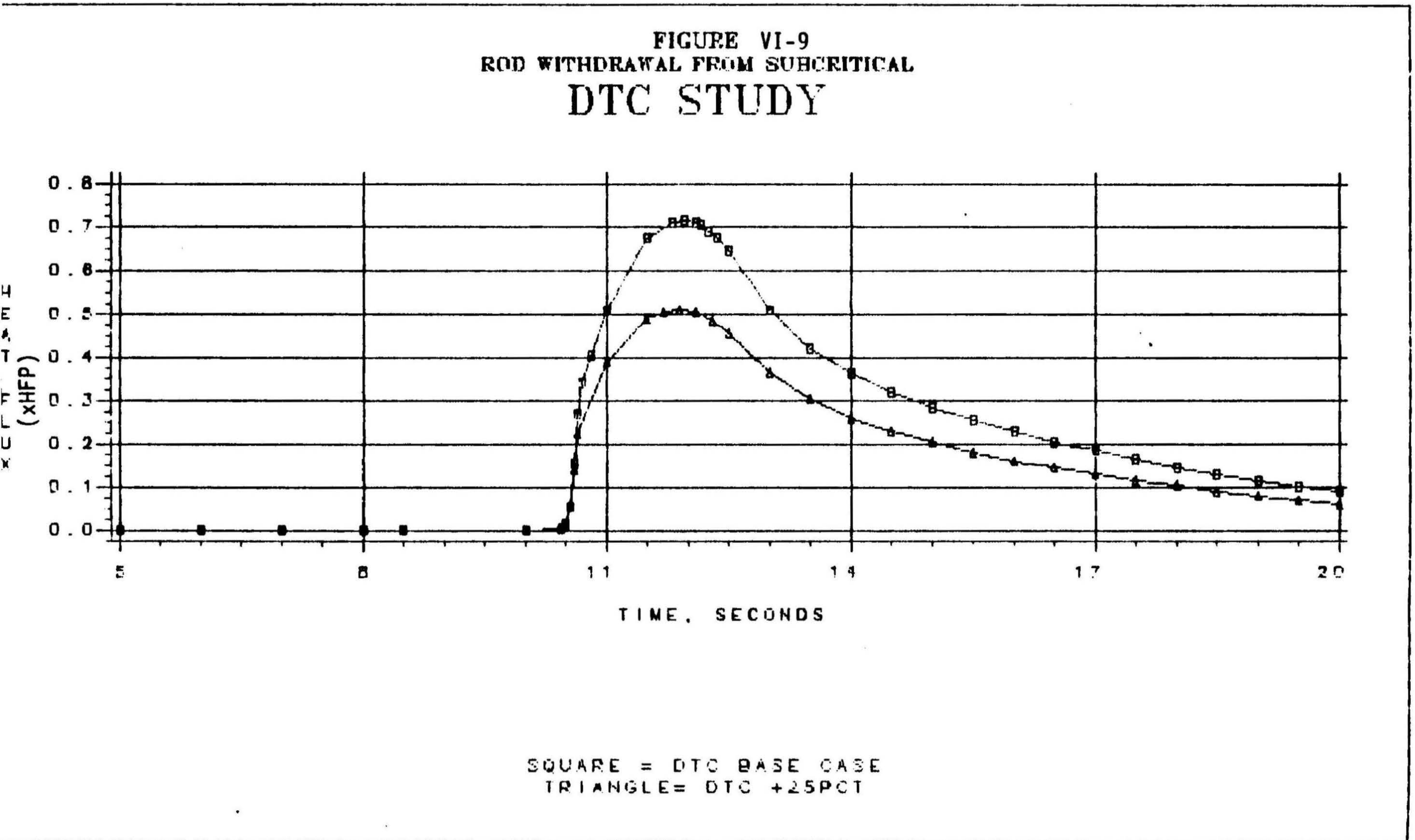


FIGURE VI-9
ROD WITHDRAWAL FROM SUBCRITICAL
DTC STUDY
(xHFP)
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
5
8
11
14
17
20
TIME, SECONDS
SQUARE = DTC BASE CASE
TRIANGLE= DTC +25PCT

FIGURE VI-10
ROD WITHDRAWAL FROM SUBCRITICAL
DTC STUDY

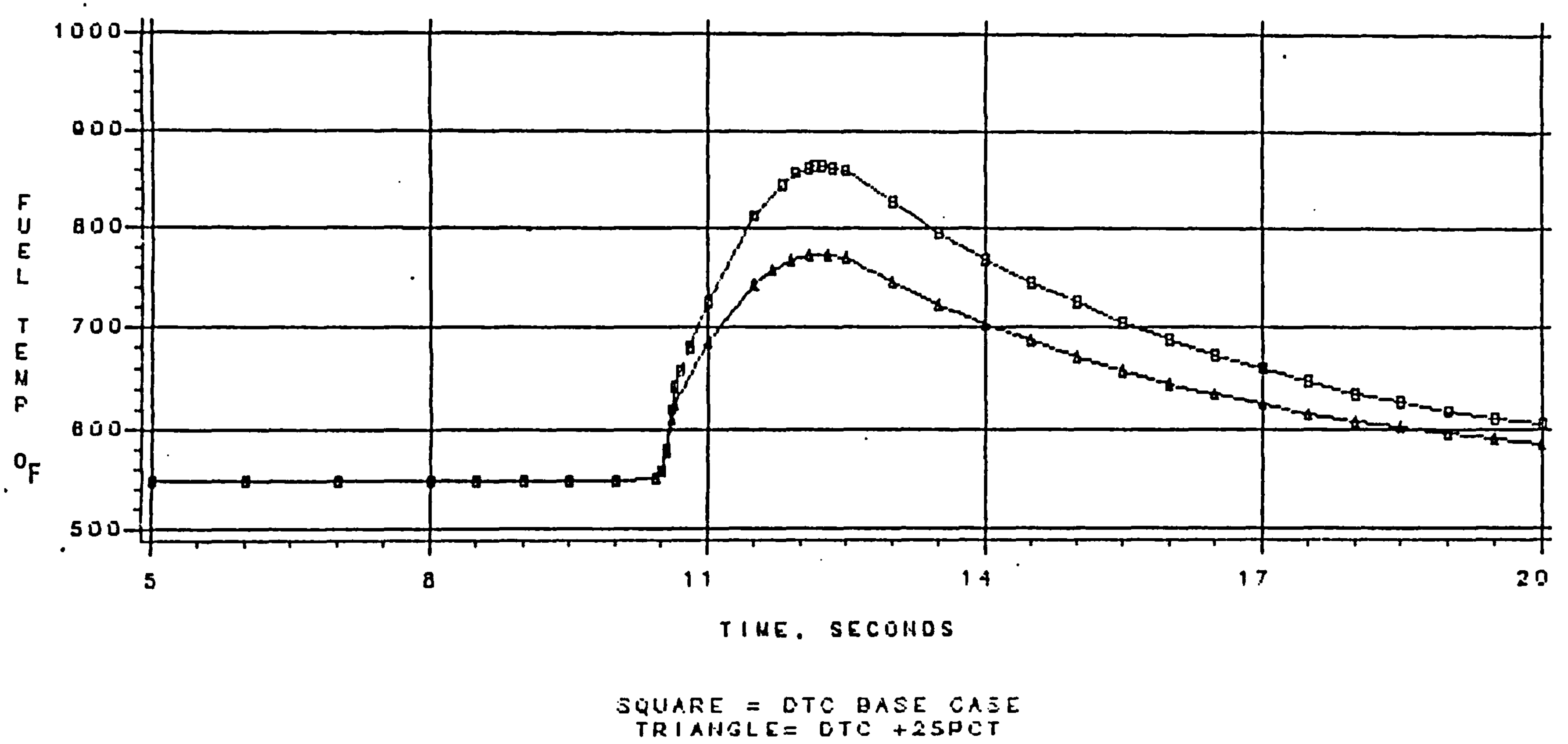


FIGURE VI-11
ROD WITHDRAWAL FROM SUBCRITICAL
BETA STUDY

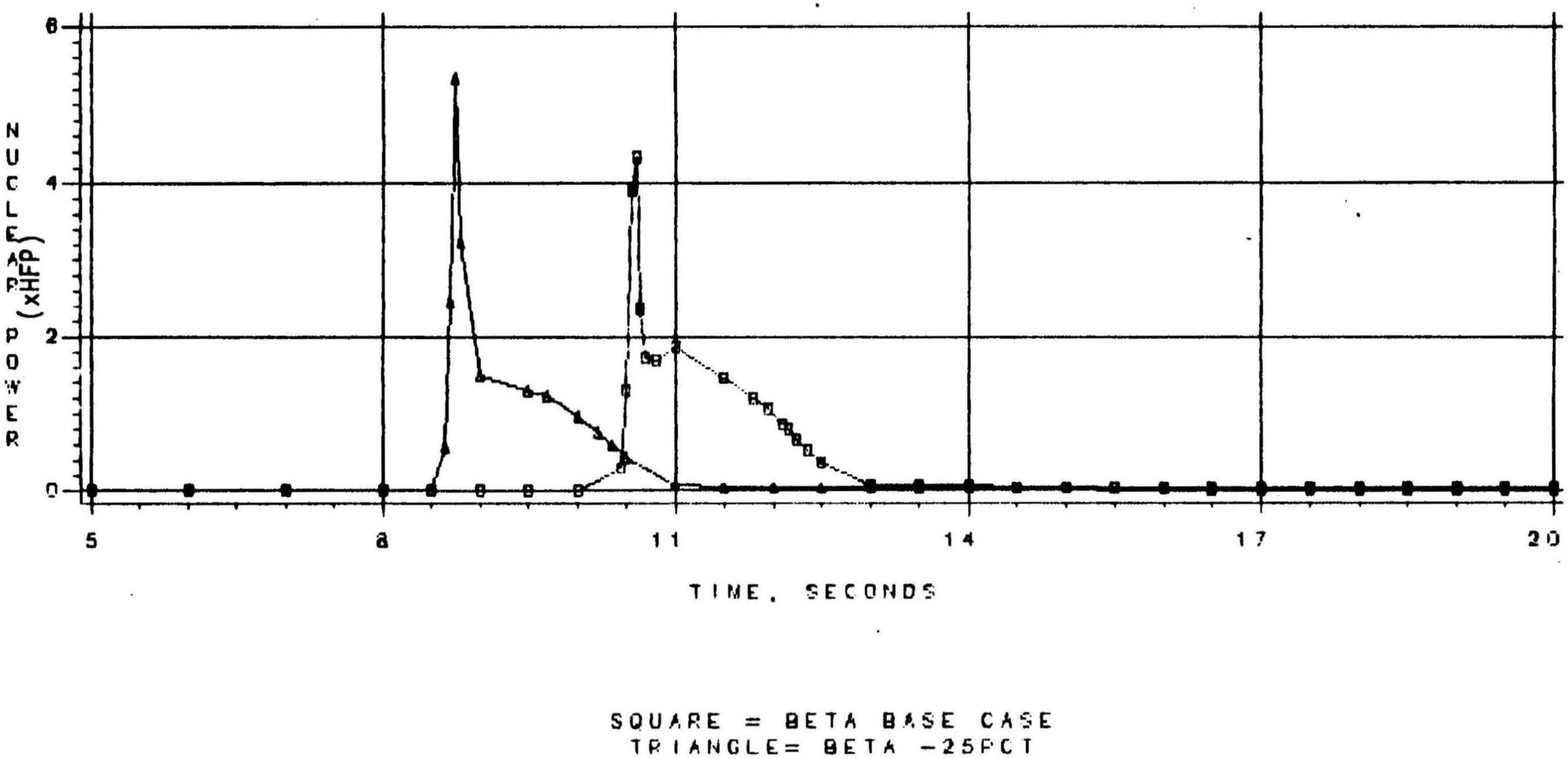
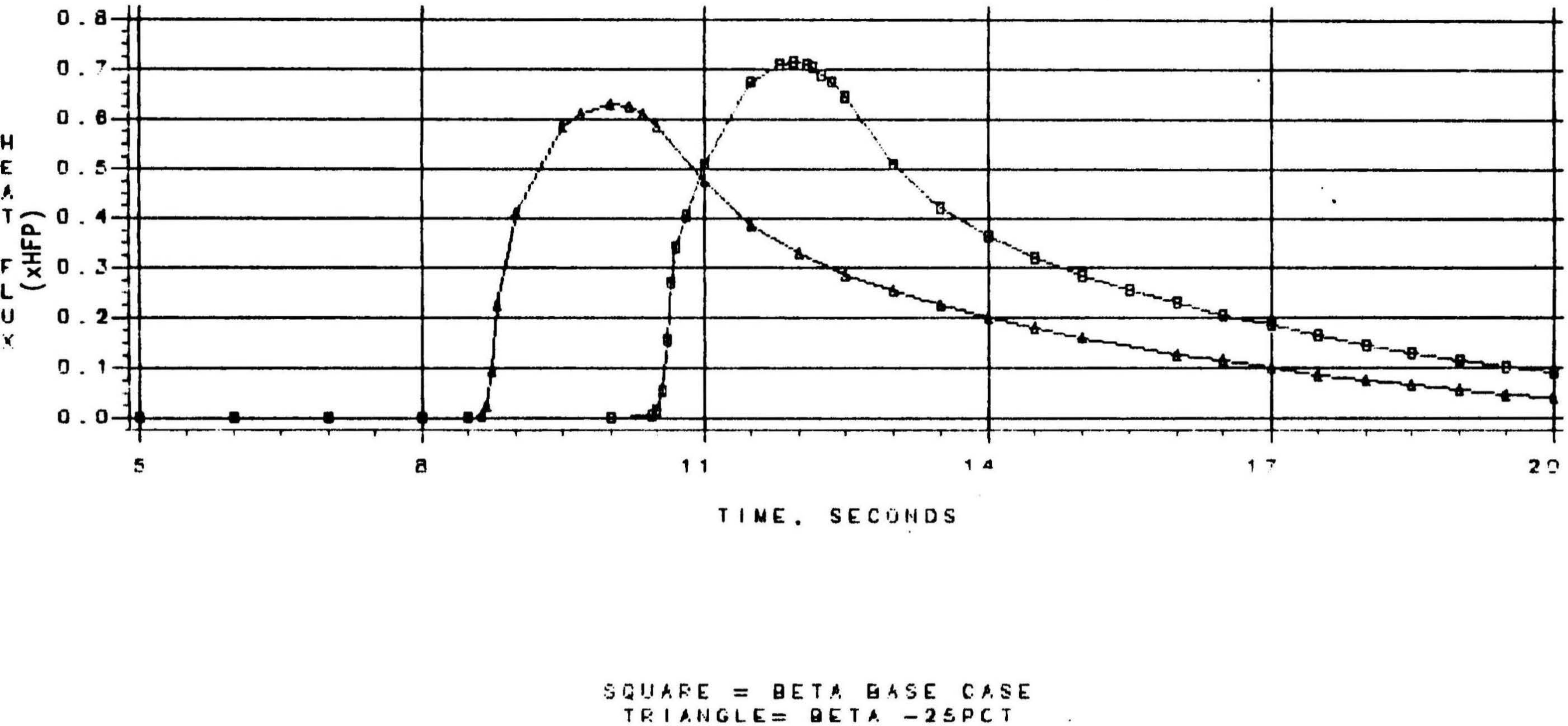


FIGURE VI-12
ROD WITHDRAWAL FROM SUBCRITICAL

BETA STUDY



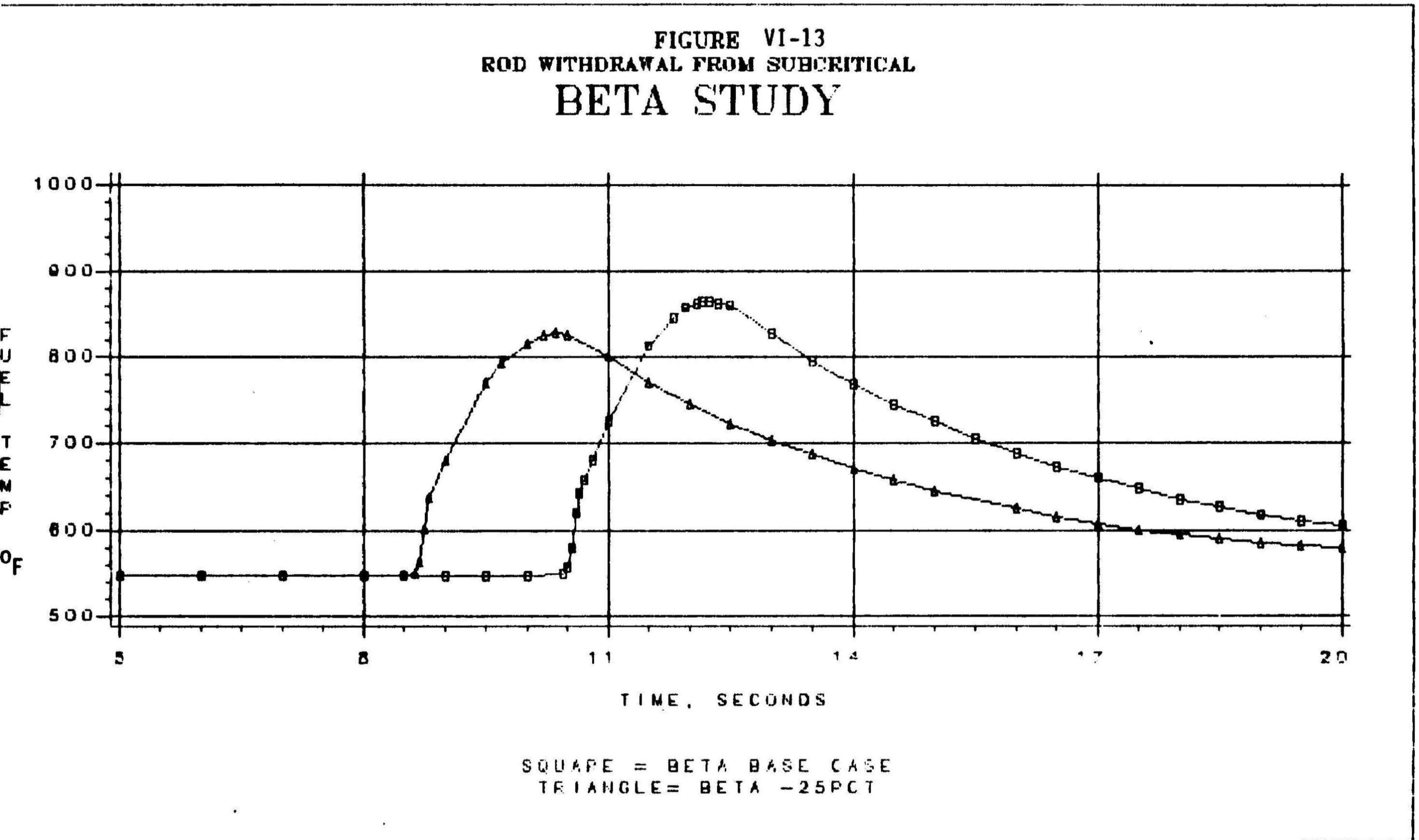


FIGURE VI-13
ROD WITHDRAWAL FROM SUBCRITICAL
BETA STUDY
FUEL TEMP °F
1000
900
800
700
600
500
5
8
11
14
17
20
TIME, SECONDS
SQUARE = BETA BASE CASE
TRIANGLE= BETA -25PCT

FIGURE VI-14
ROD WITHDRAWAL FROM SUBCRITICAL

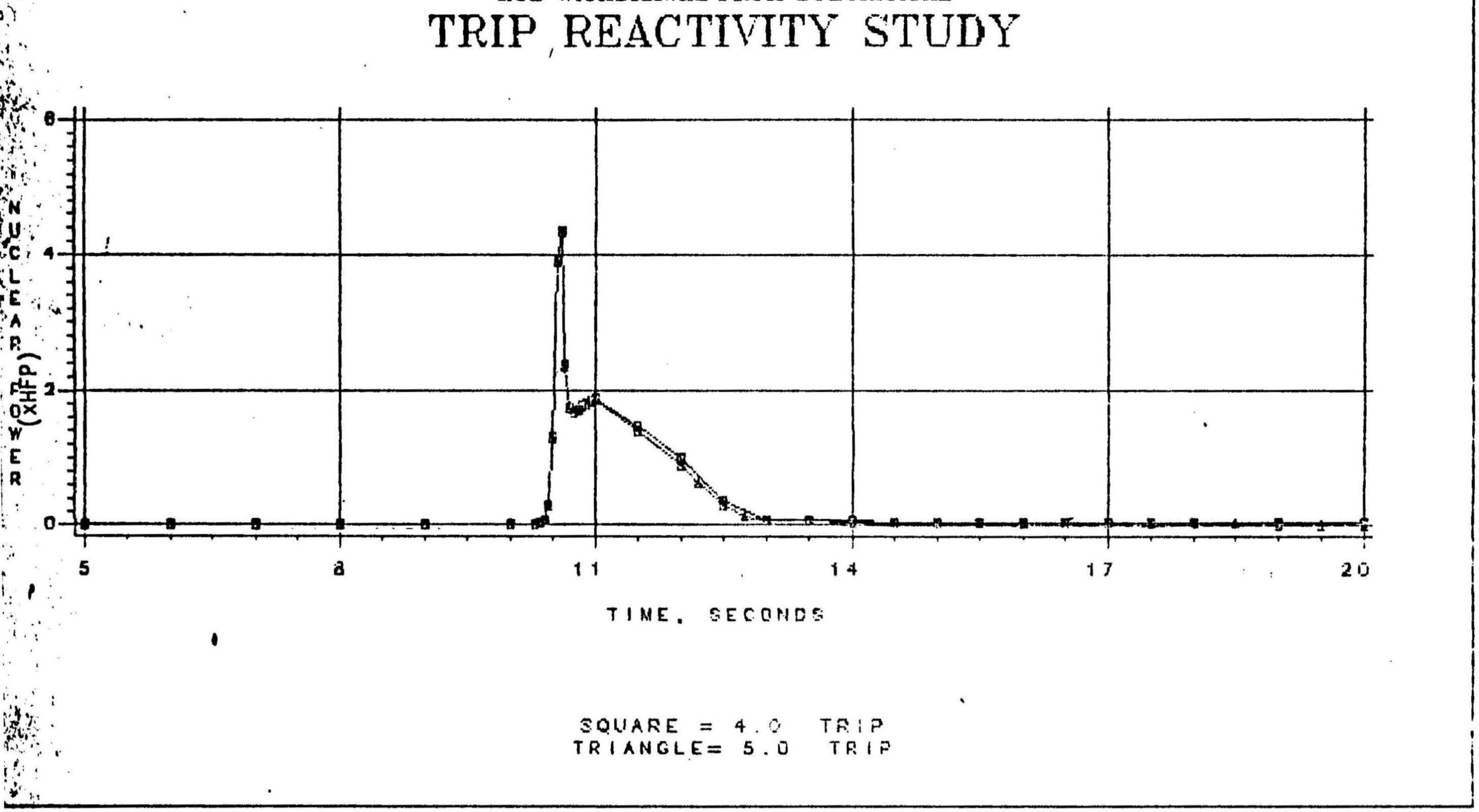


FIGURE VI-15

ROD WITHDRAWAL FROM SUBCRITICAL

TRIP REACTIVITY STUDY

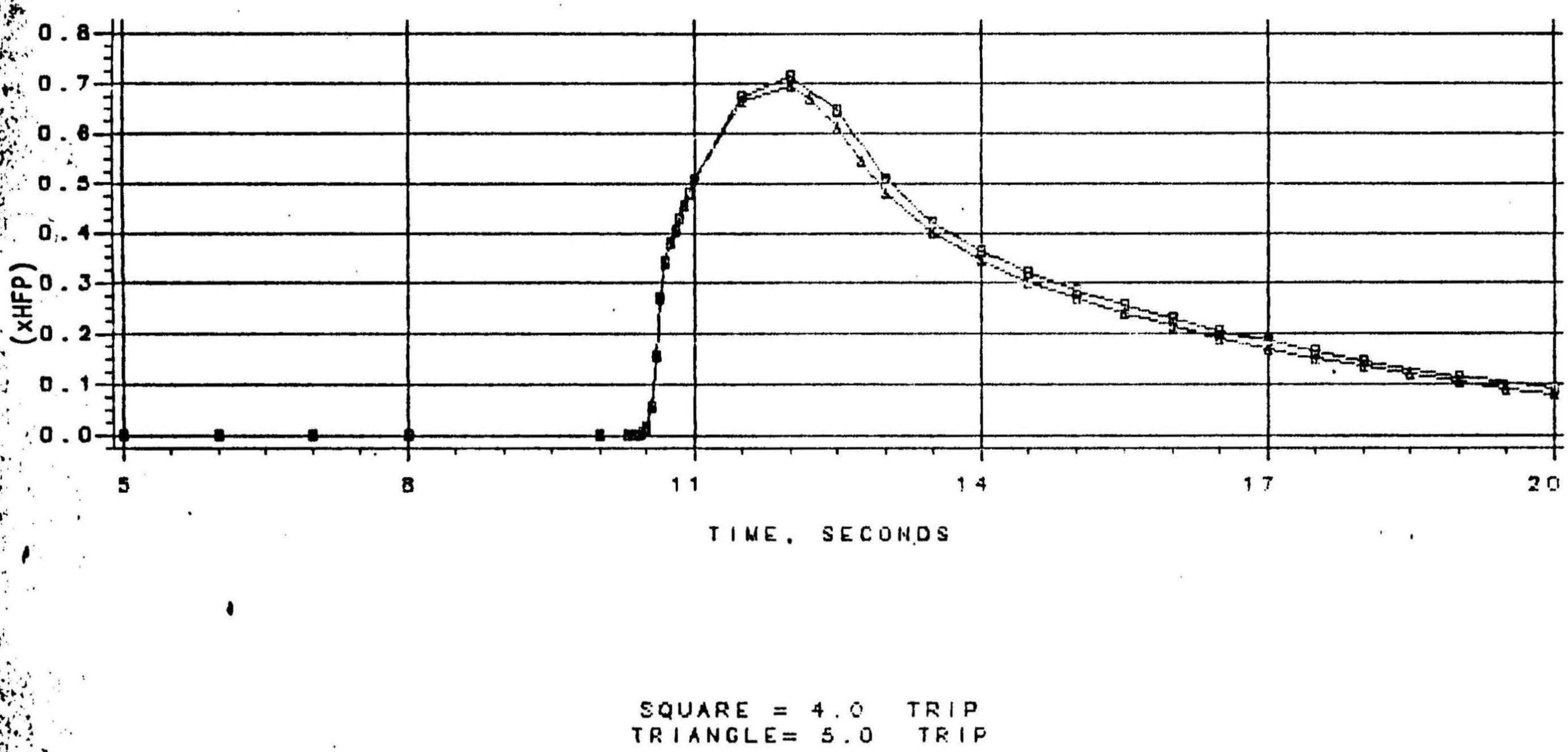


FIGURE VI-16

ROD WITHDRAWAL FROM SUBCRITICAL

TRIP REACTIVITY STUDY

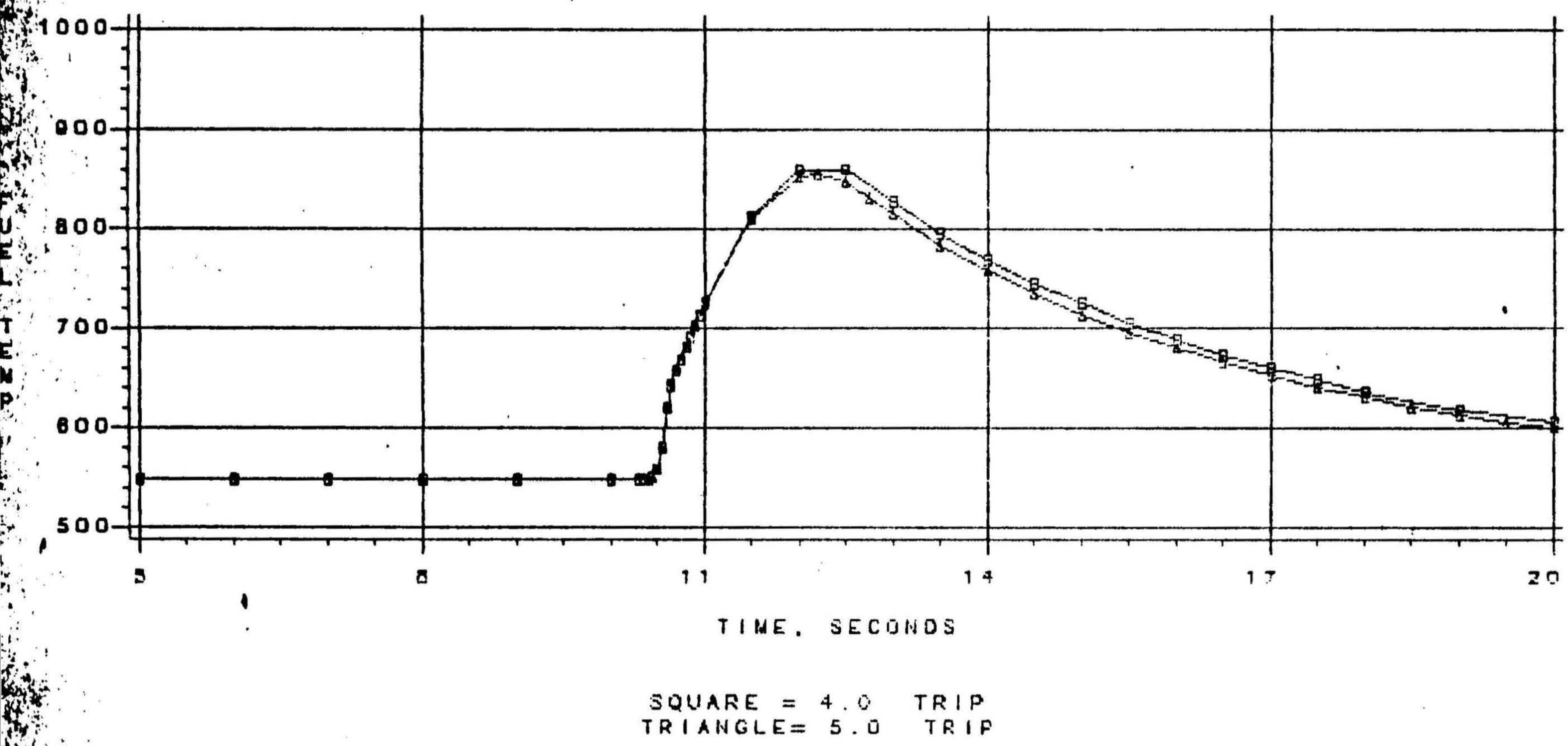
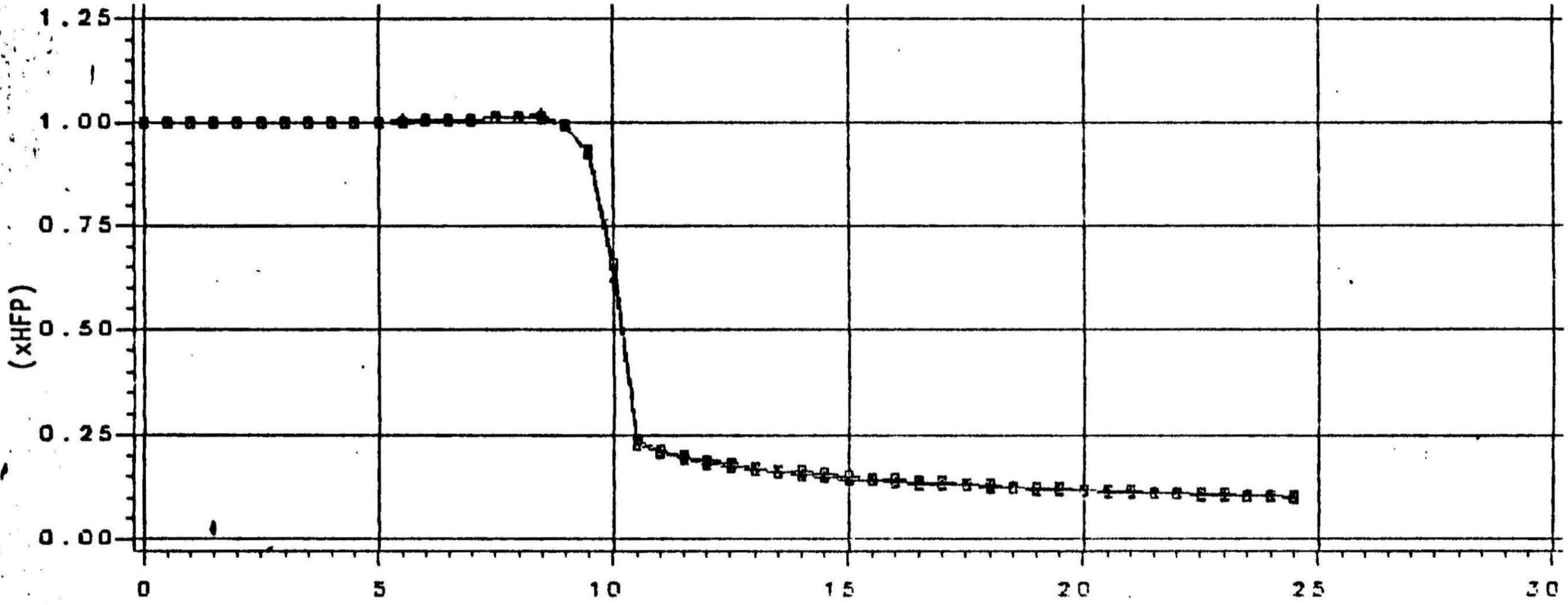


FIGURE VI-17

LOSS OF LOAD

DPC STUDY



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

FIGURE VI-18
LOSS OF LOAD

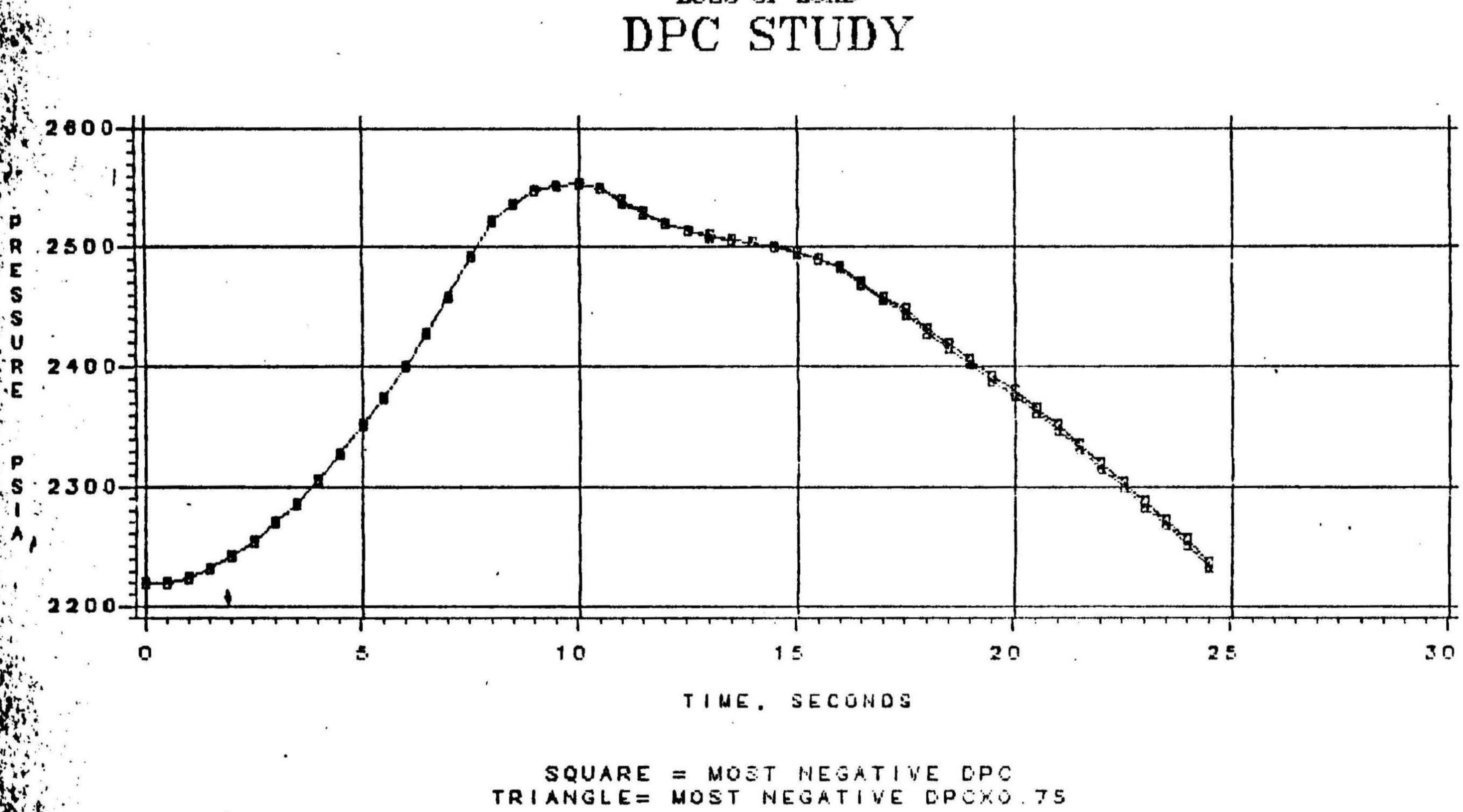


FIGURE VI-19
LOSS OF LOAD
DPC STUDY

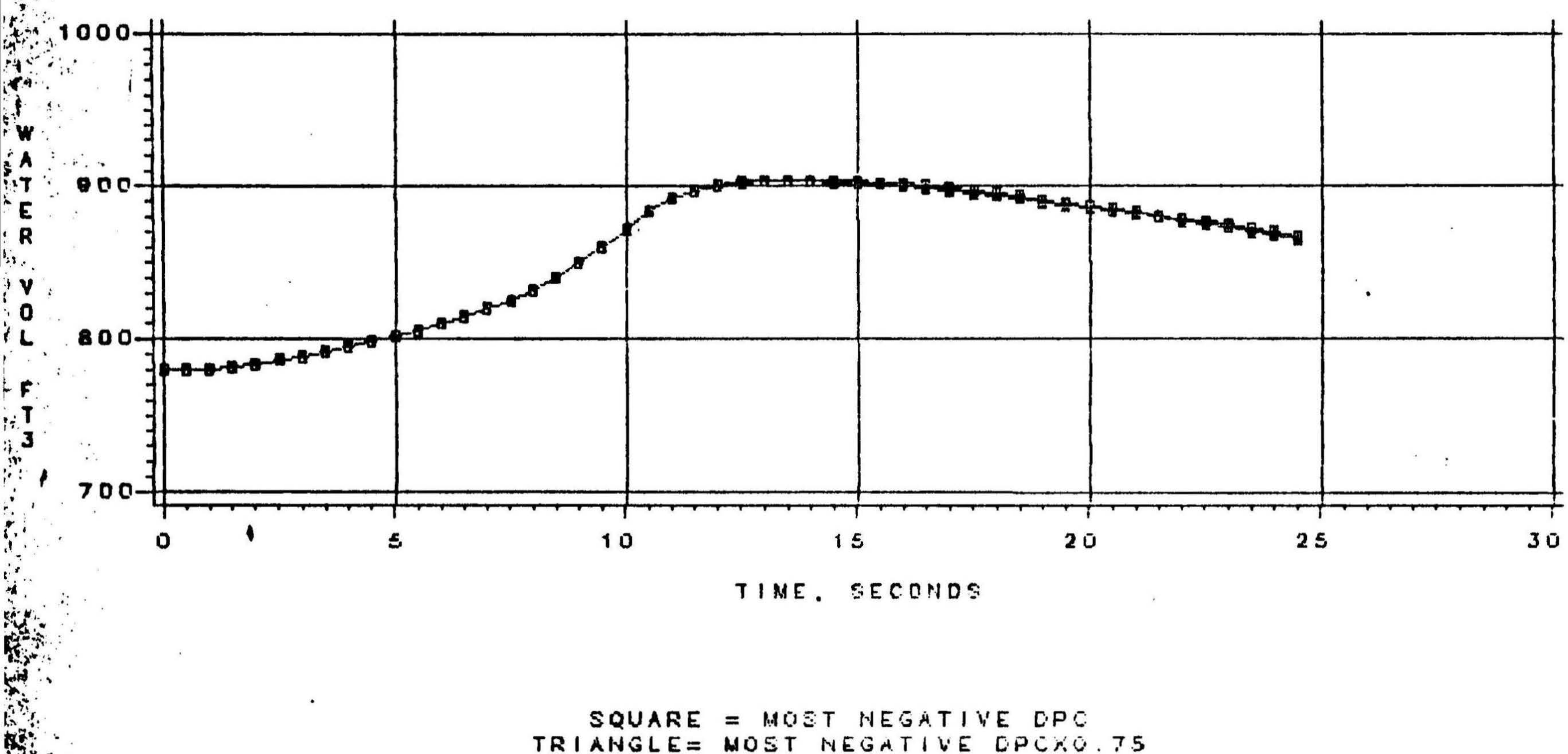
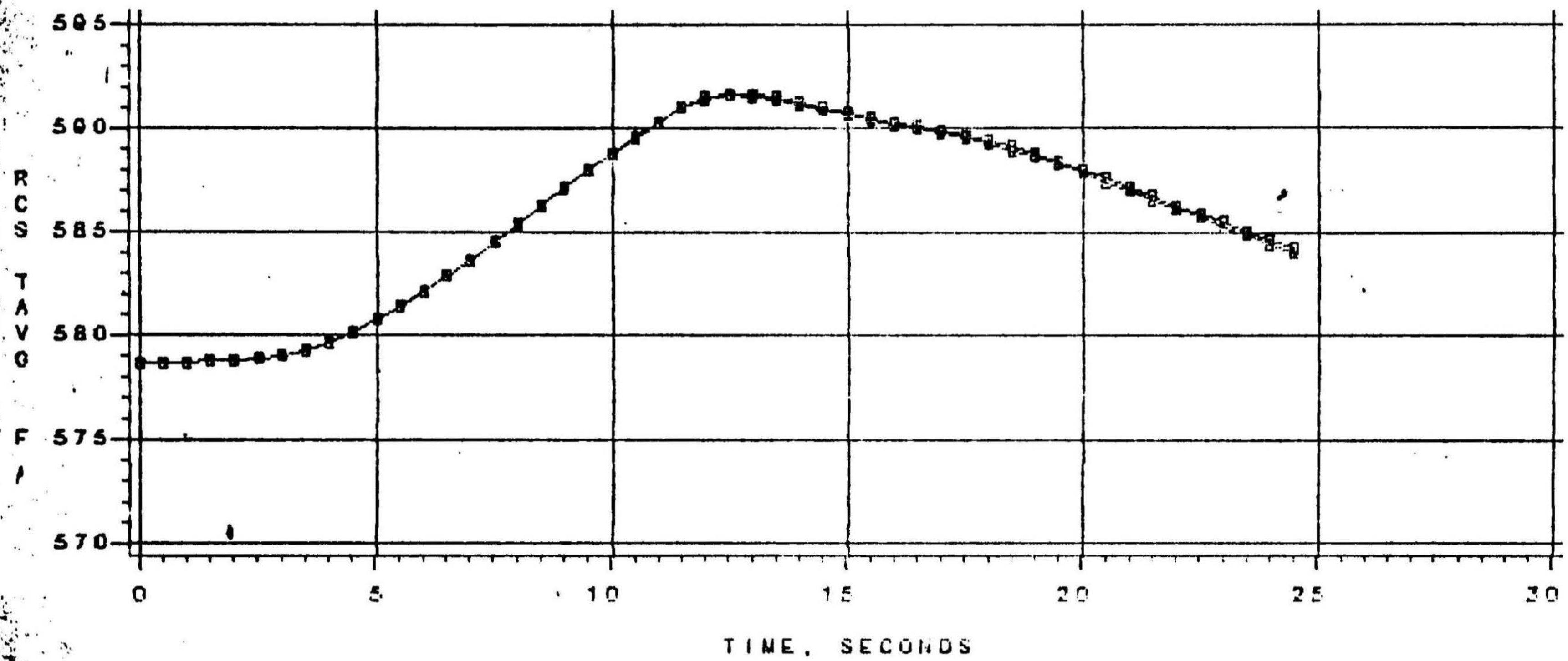


FIGURE VI-20
LOSS OF LOAD
DPC STUDY



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

FIGURE VI-21
LOSS OF LOAD

MTC STUDY

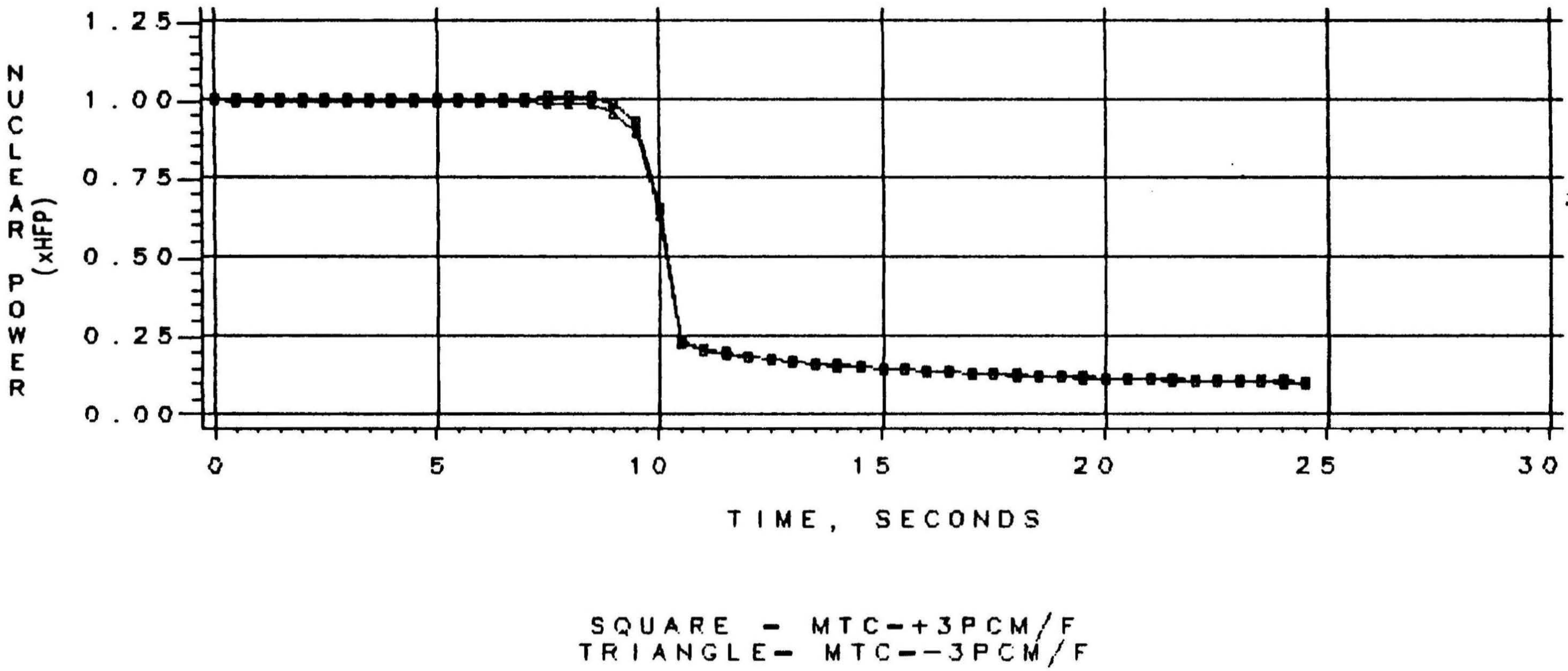
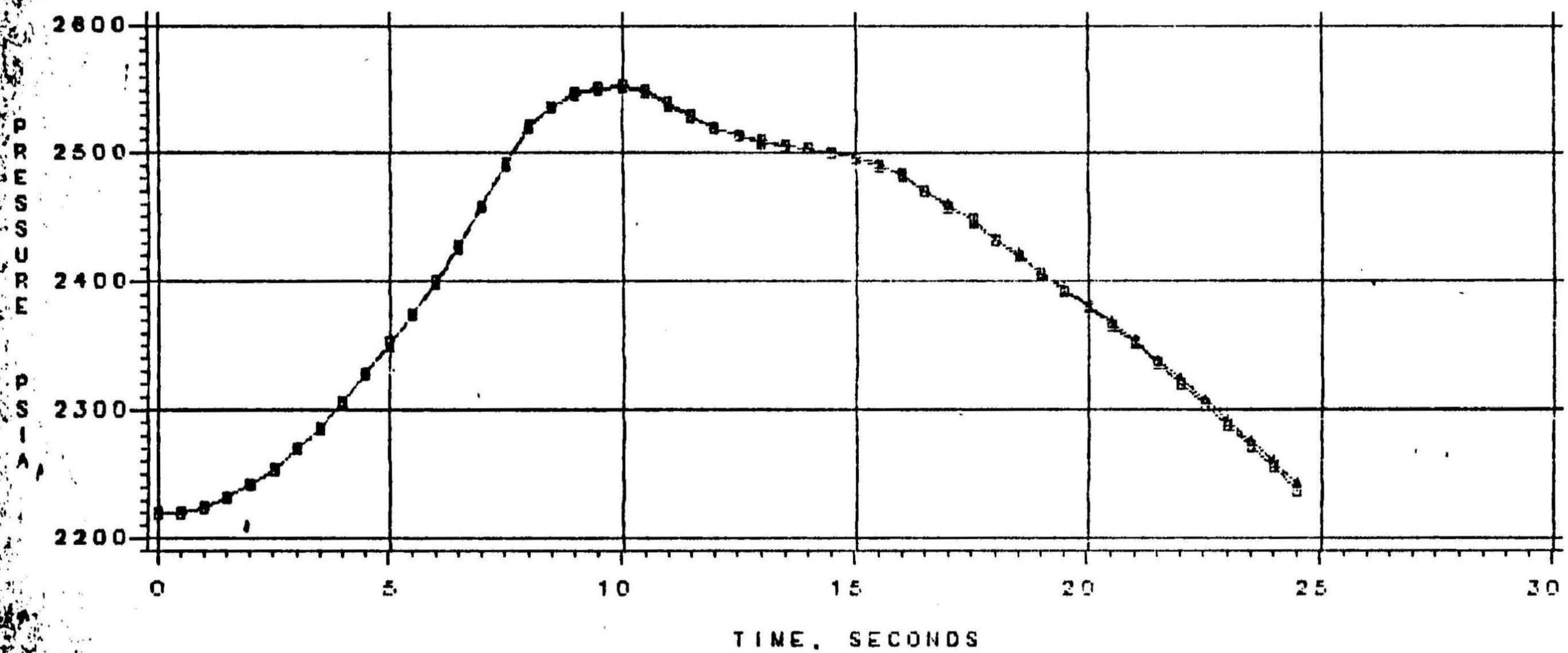


FIGURE VI-22
LOSS OF LOAD
MTC STUDY



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

FIGURE VI-23
LOSS OF LOAD

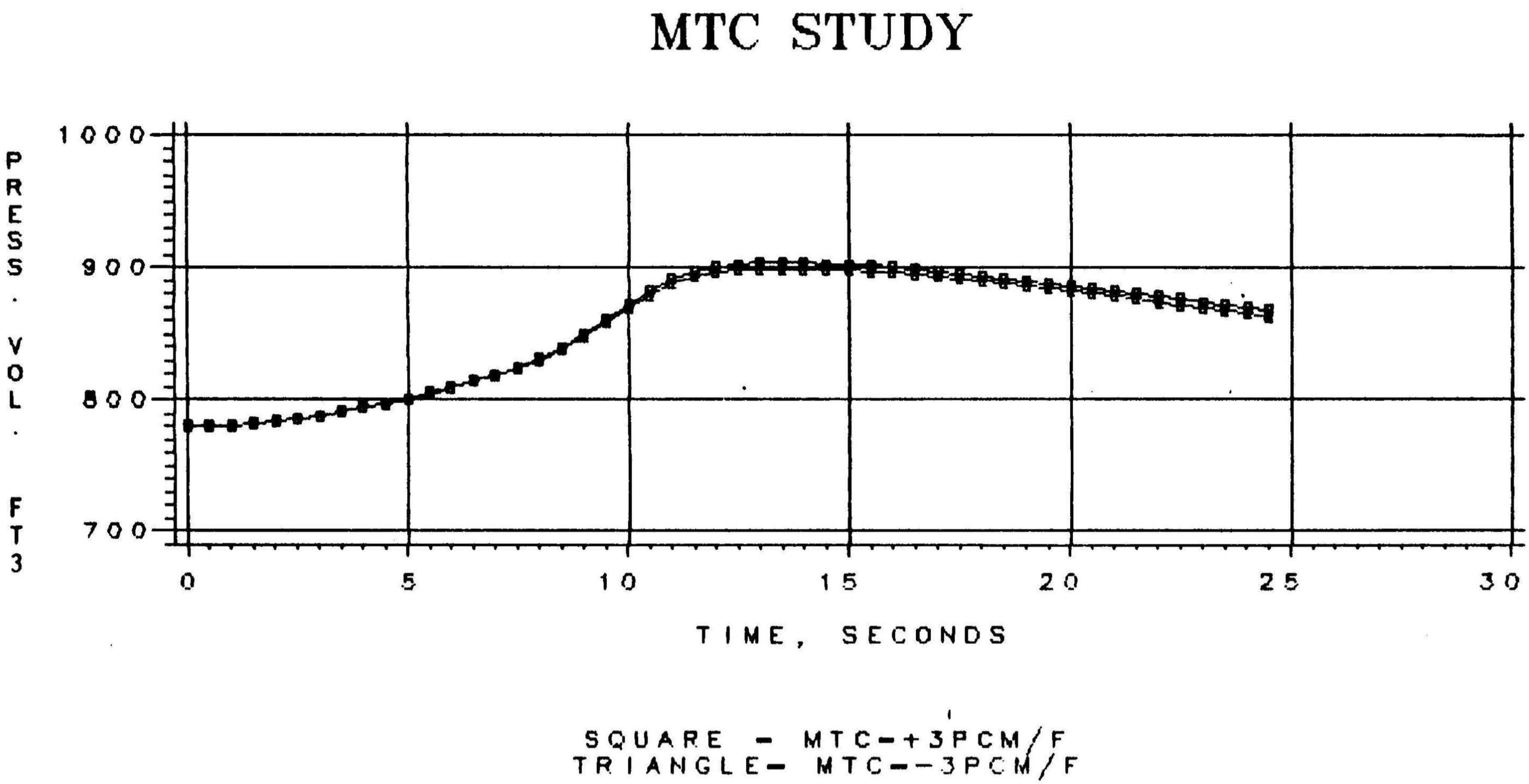
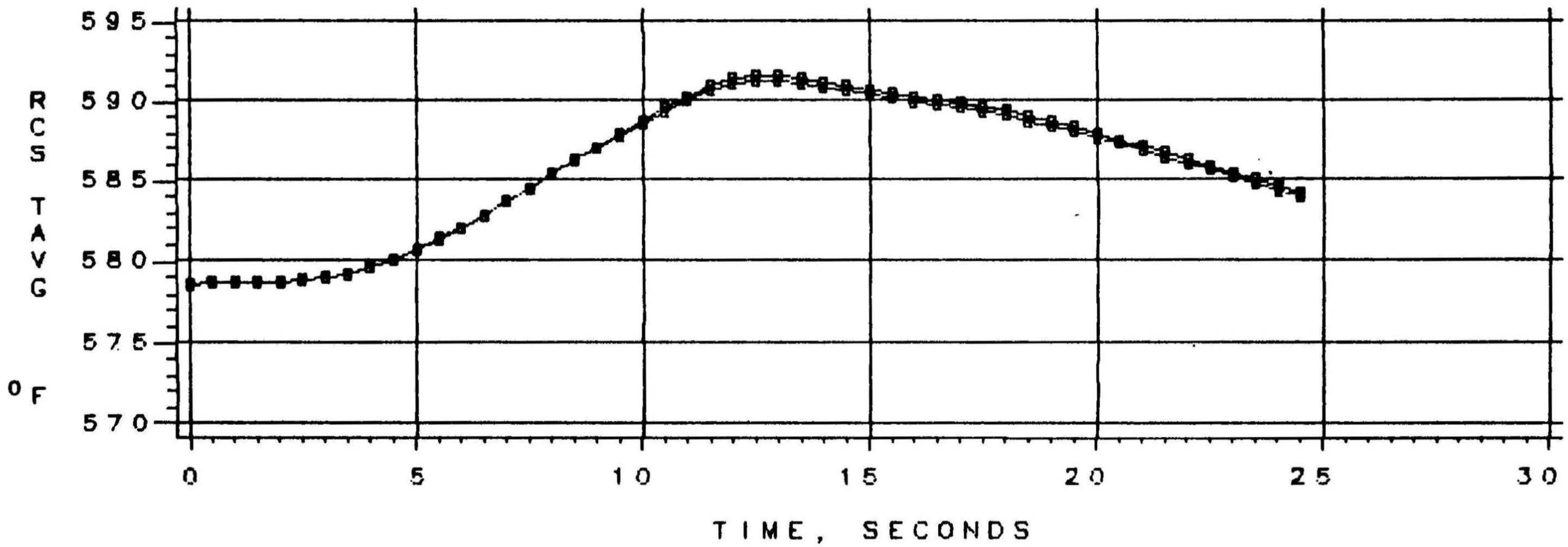


FIGURE VI-24
LOSS OF LOAD

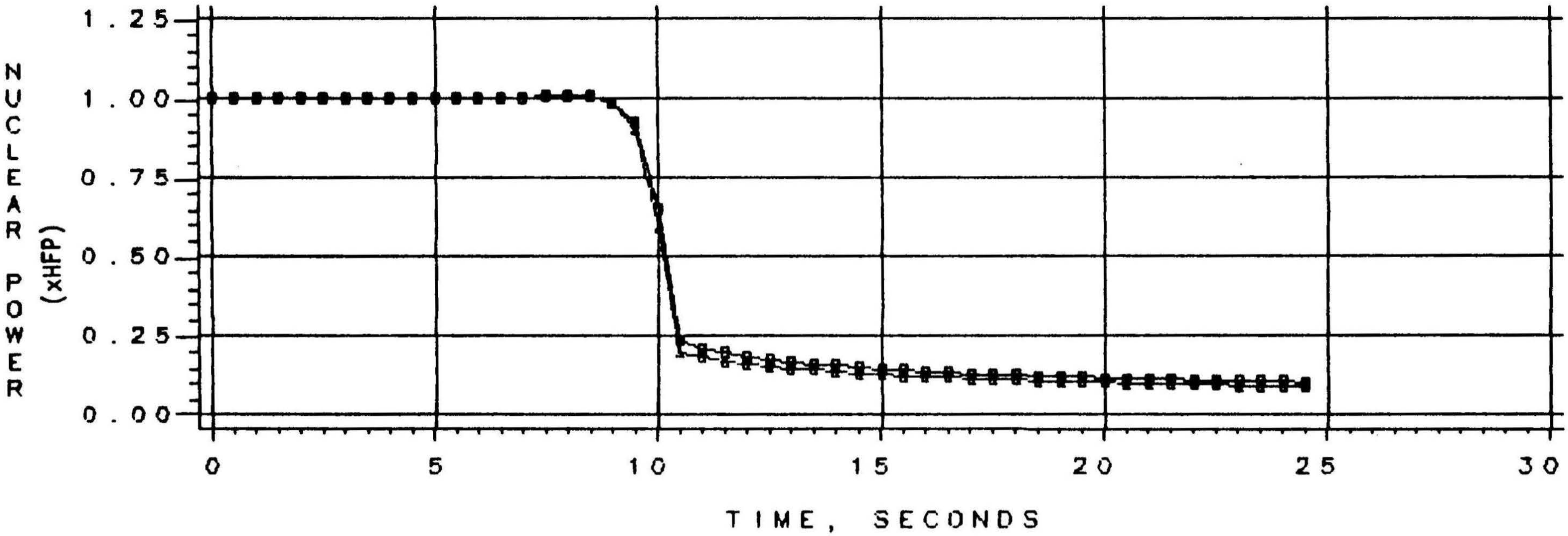
MTC STUDY



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

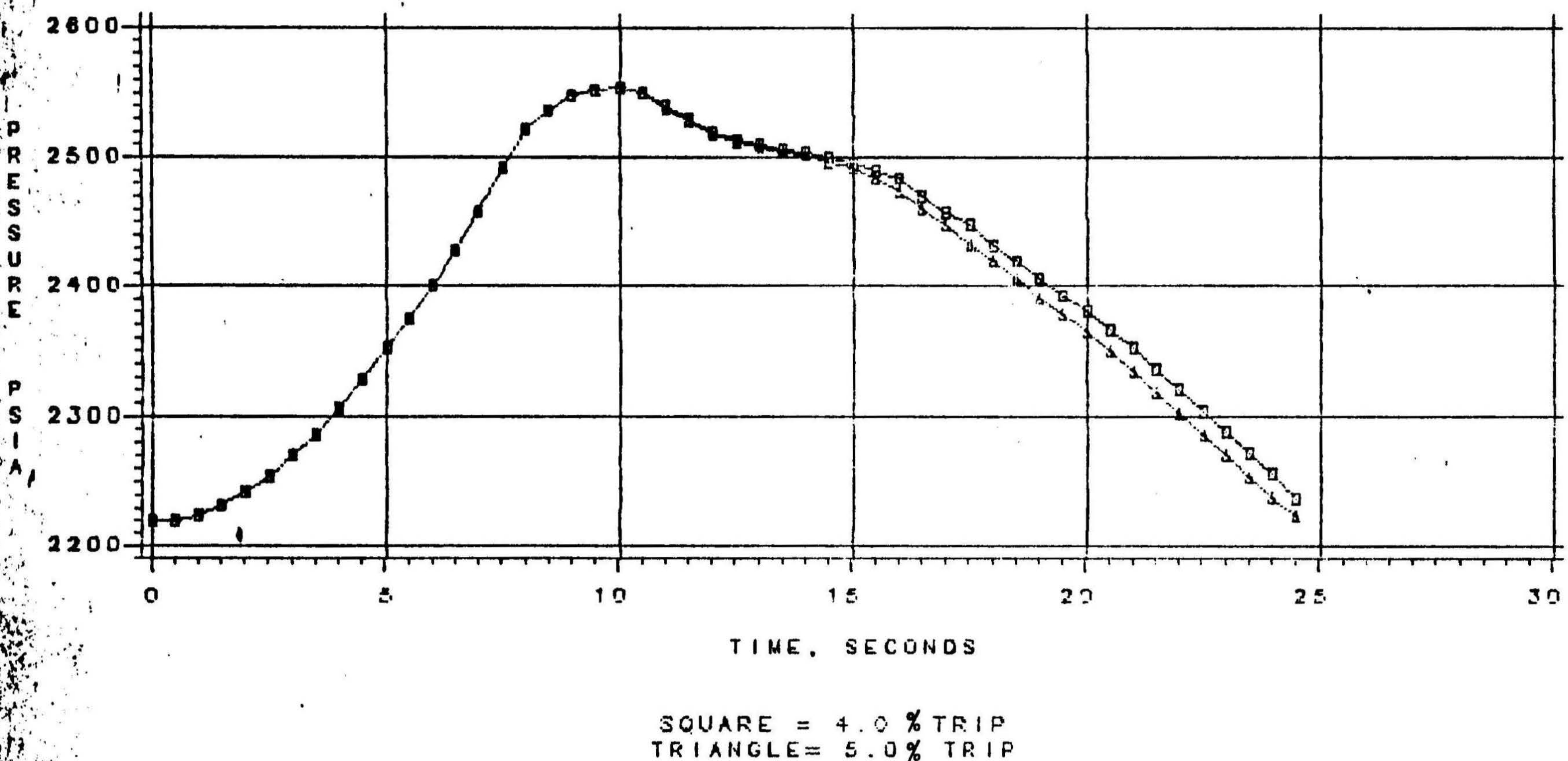


FIGURE VI-27
LOSS OF LOAD

TRIP STUDY

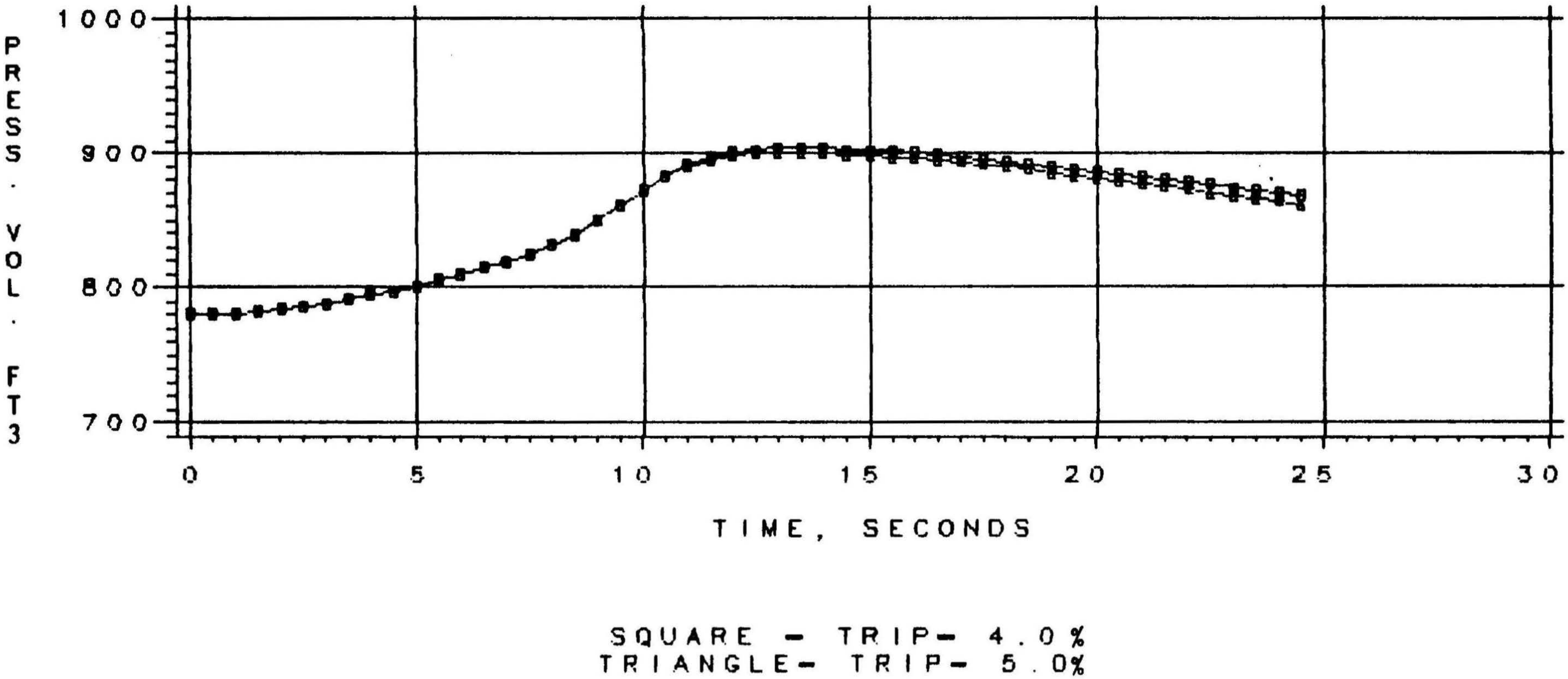
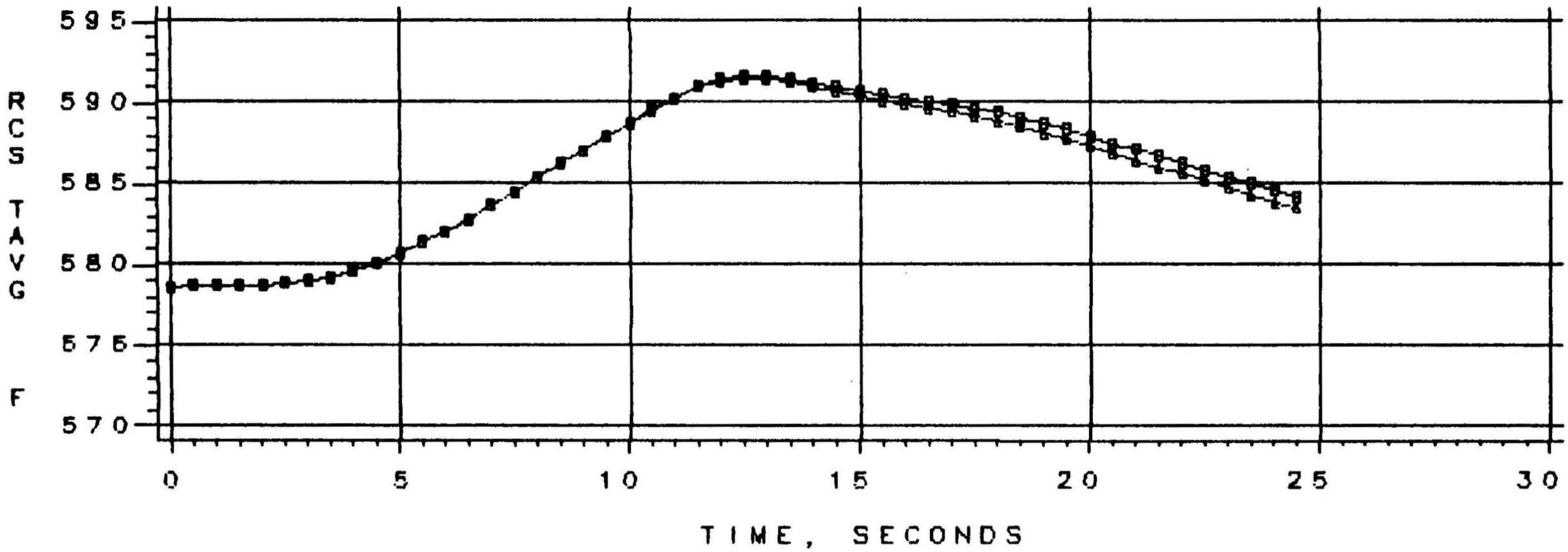


FIGURE VI-28
LOSS OF LOAD

TRIP STUDY

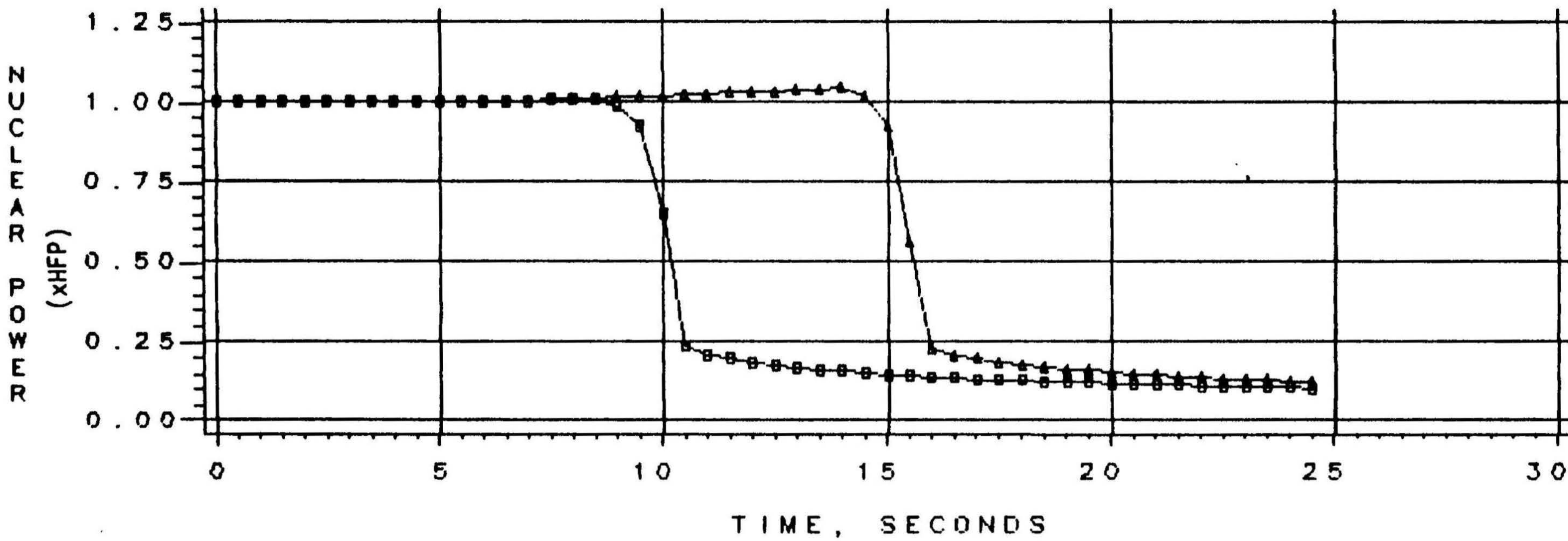


SQUARE - TRIP- 4.0%
TRIANGLE- TRIP- 5.0%

FIGURE VI-29

LOSS OF LOAD

PORV/SPRAY STUDY

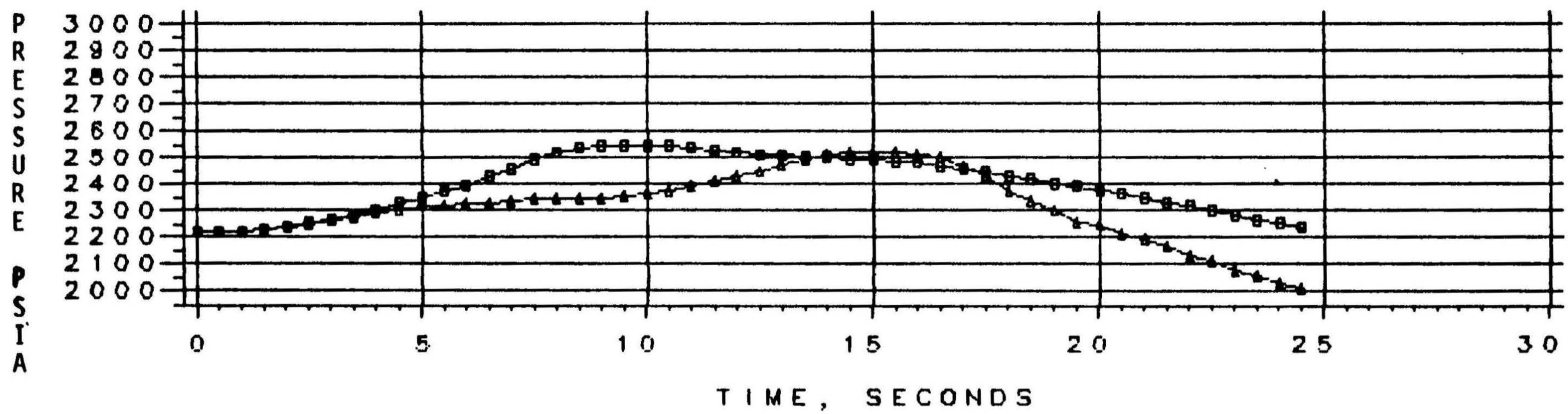


SQUARE - NO PORV/SPRAY
TRIANGLE- PORV/SPRAY

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

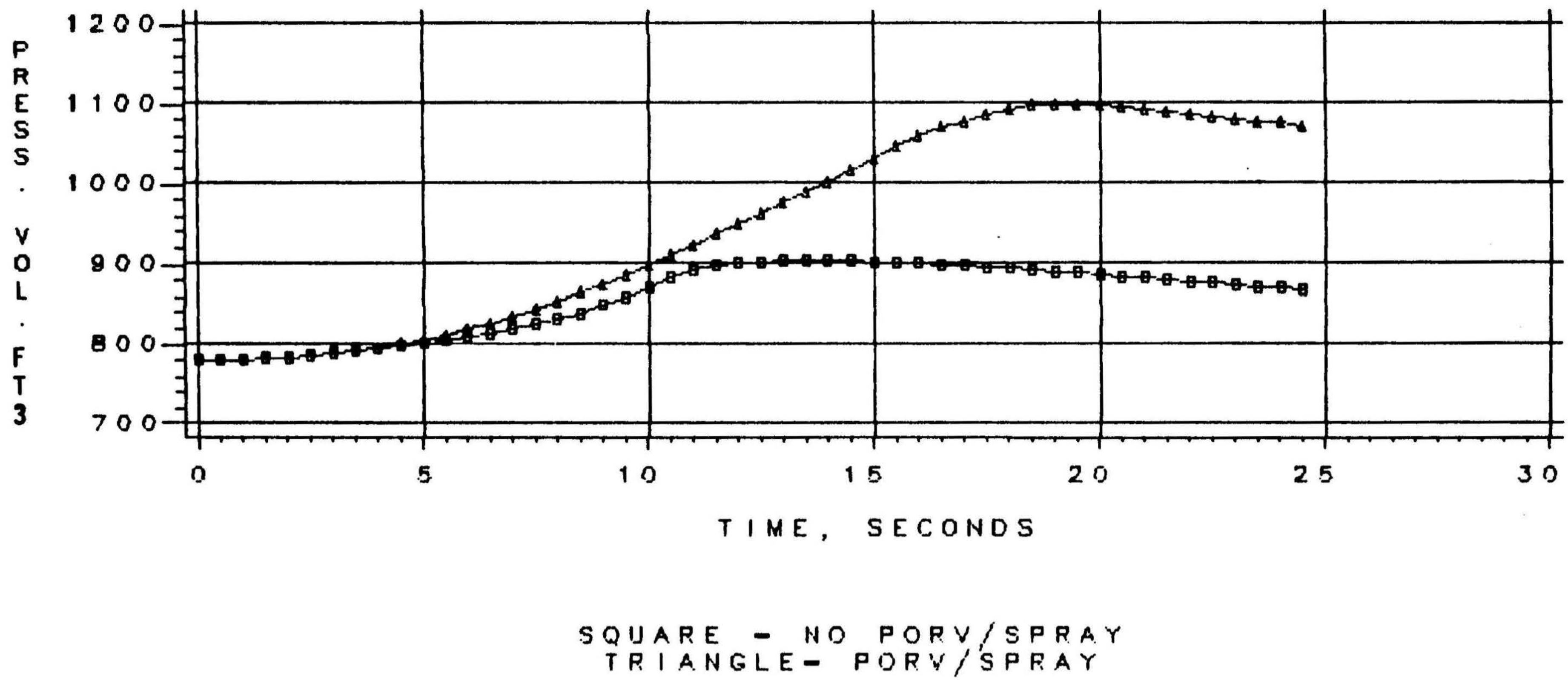
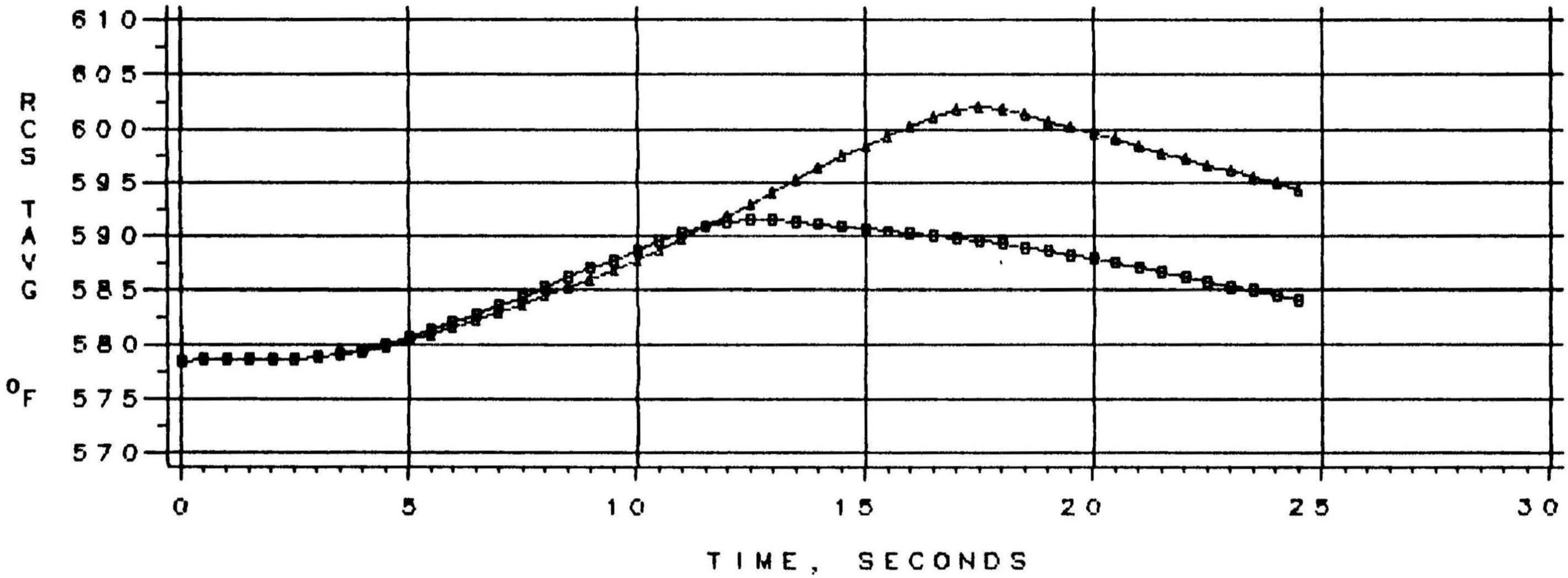


FIGURE VI-32

LOSS OF LOAD

PORV/SPRAY STUDY

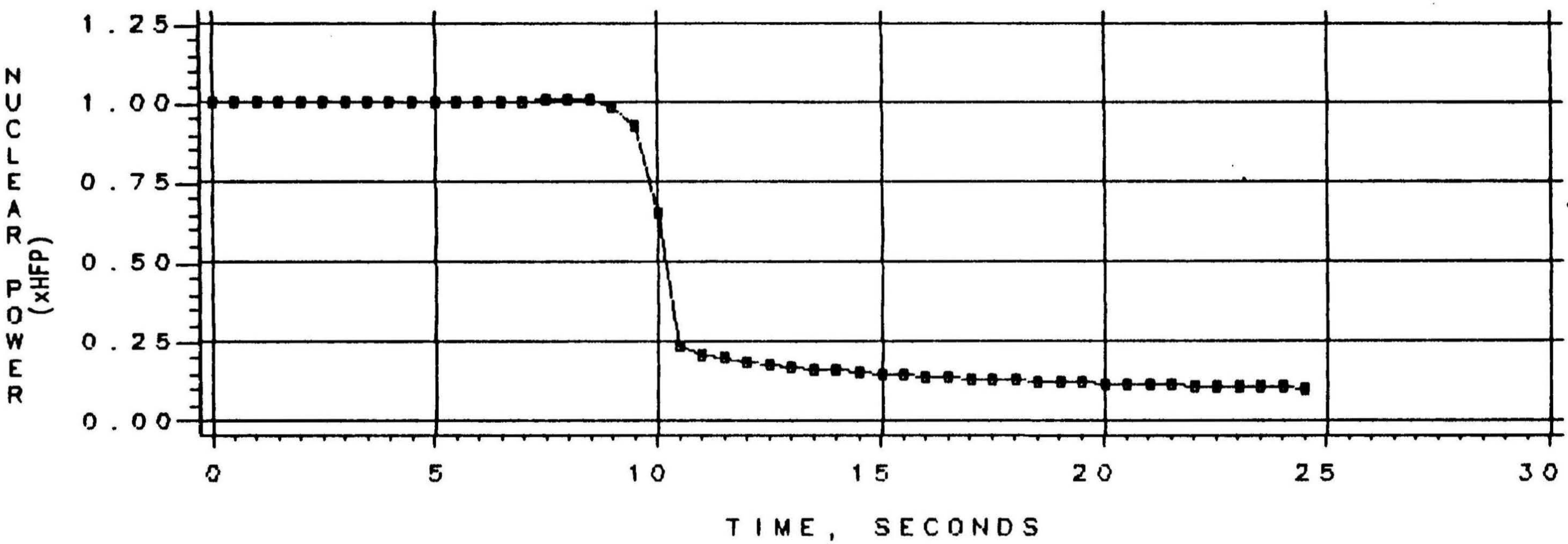


SQUARE - NO PORV/SPRAY
TRIANGLE- PORV/SPRAY

FIGURE VI-33

LOSS OF LOAD

S.G.R.V STUDY

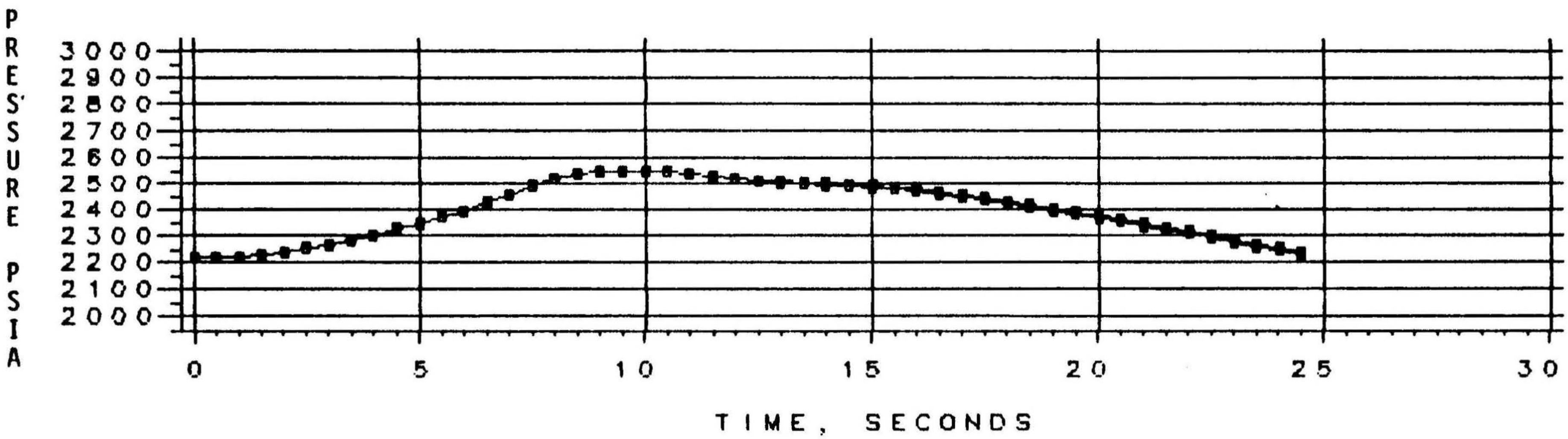


SQUARE - NO S.G.R.V
TRIANGLE- S.G.R.V

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

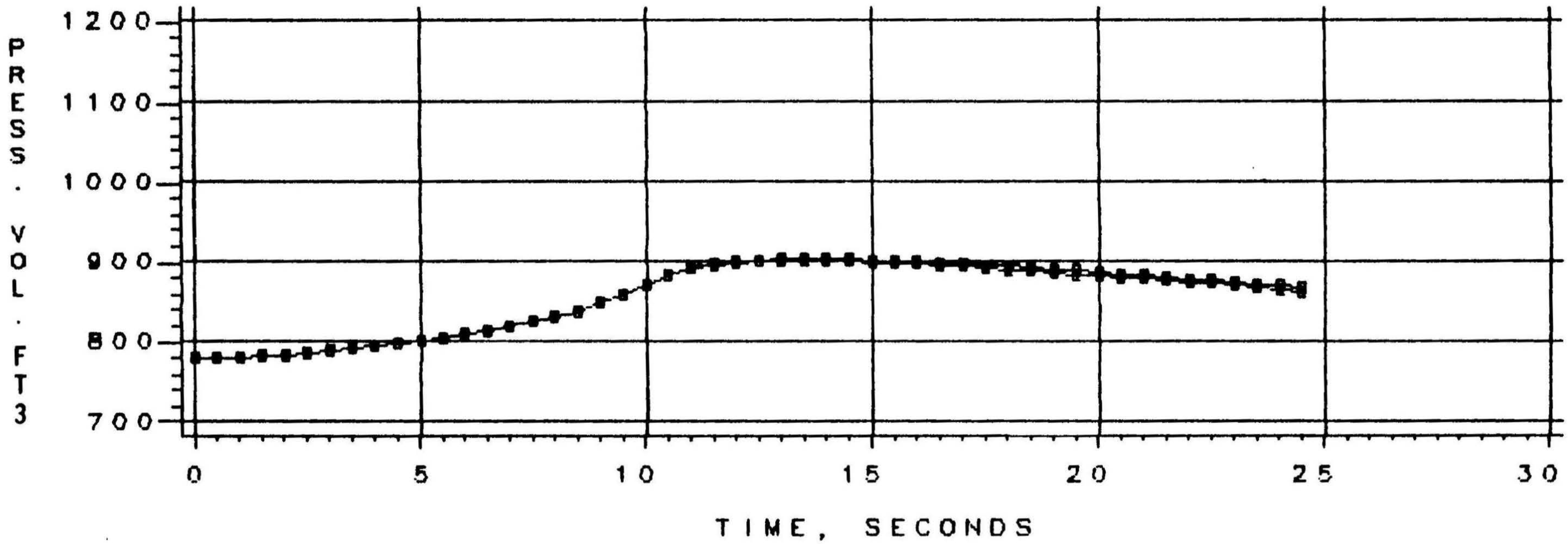


FIGURE VI-36

LOSS OF LOAD

S.G.R.V STUDY

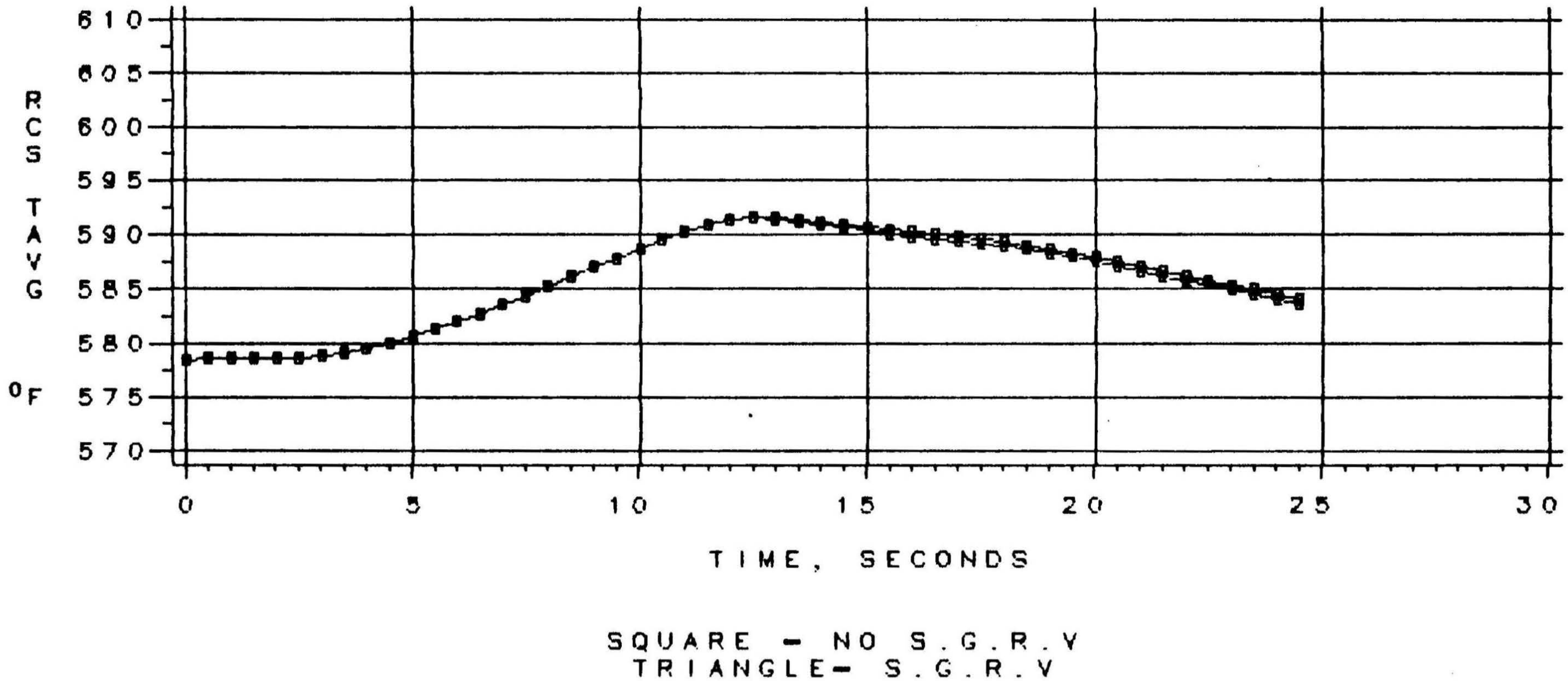
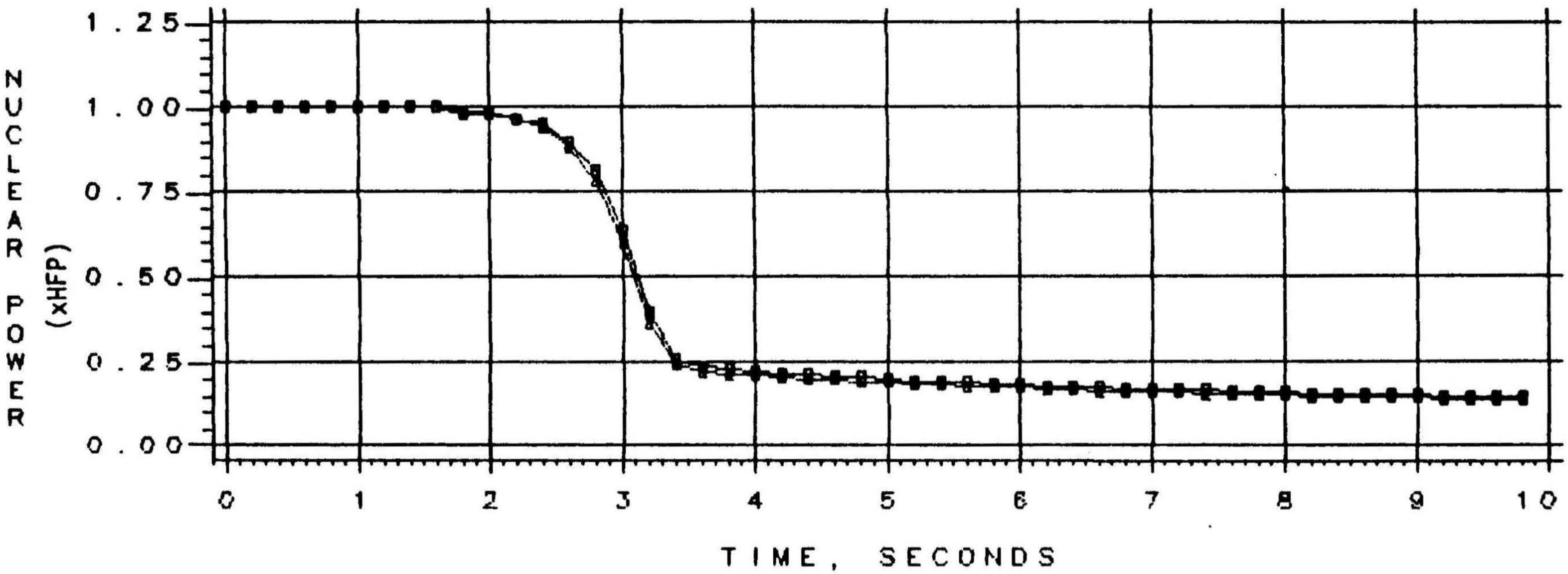


FIGURE VI-37

LOSS OF FLOW

DPC STUDY

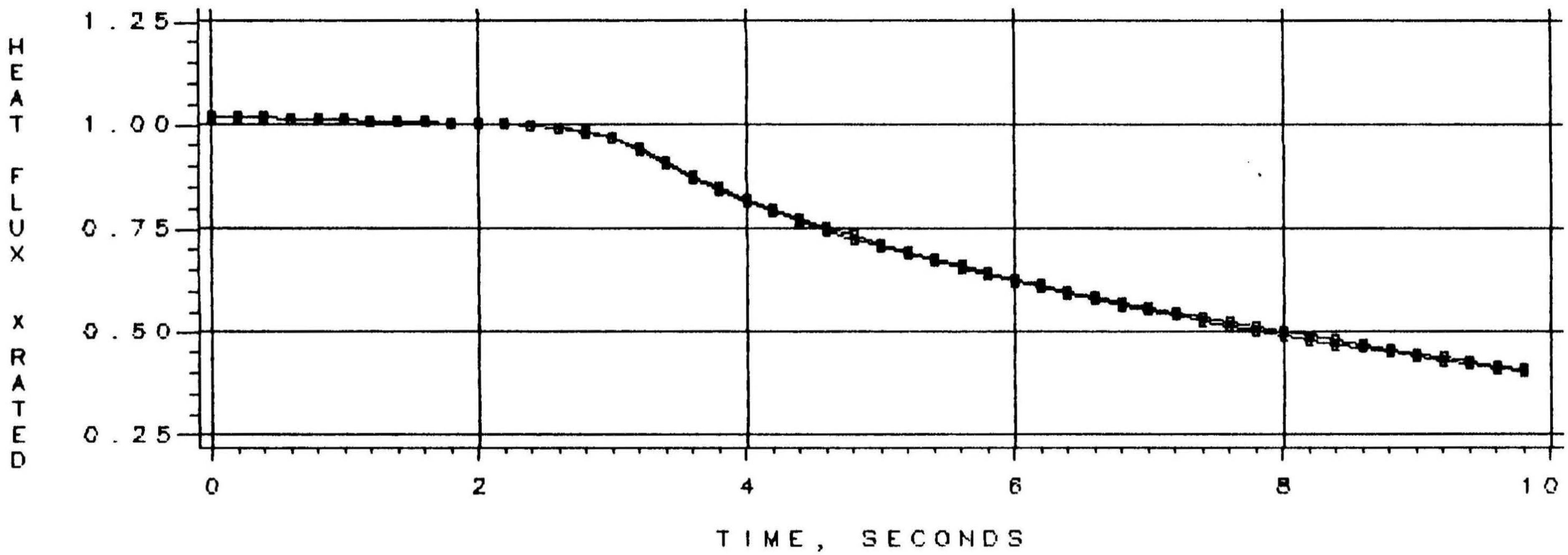


SQUARE - MOST NEG DPC
TRIANGLE- MOST NEG DPC*.75

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

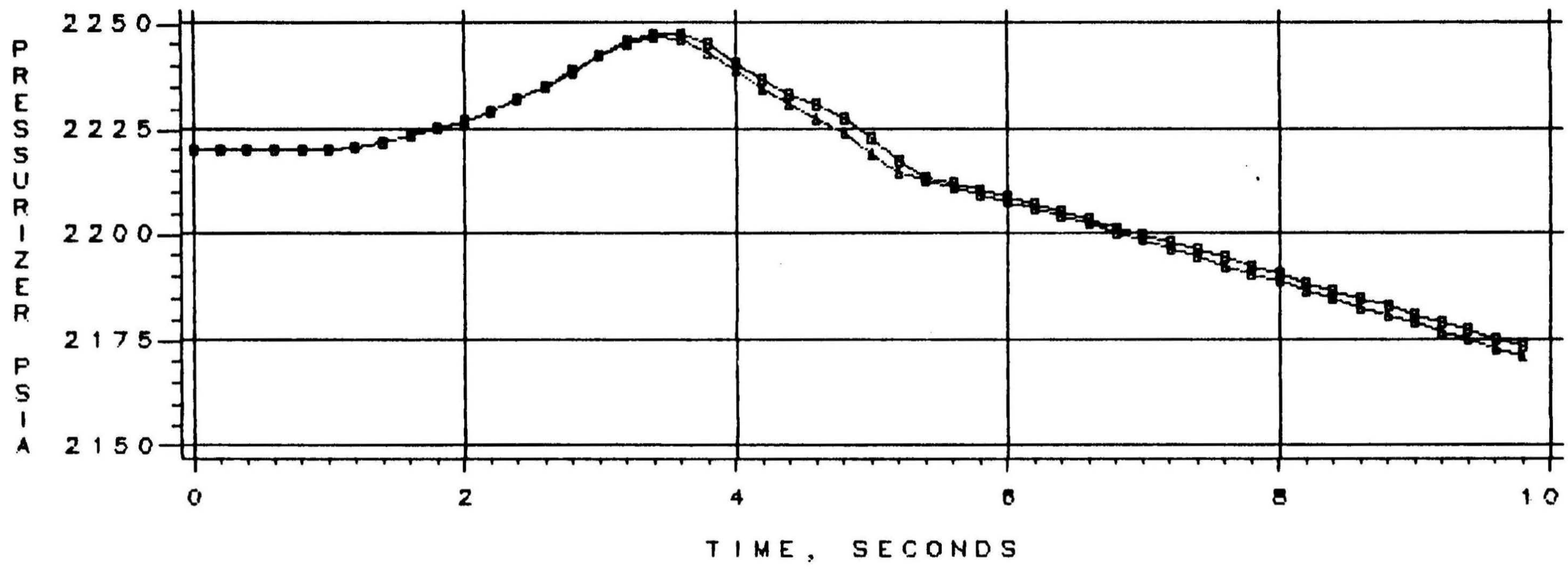
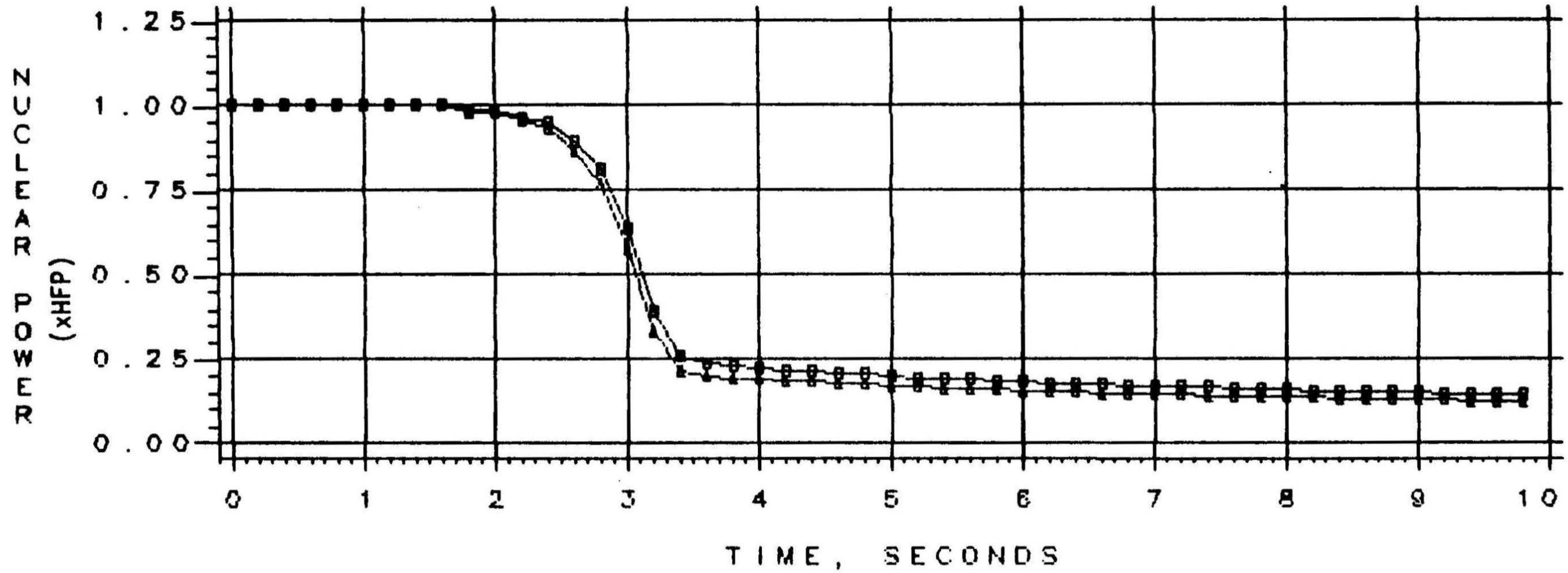


FIGURE VI-40

LOSS OF FLOW

TRIP STUDY



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

FIGURE VI-41

LOSS OF FLOW

TRIP STUDY

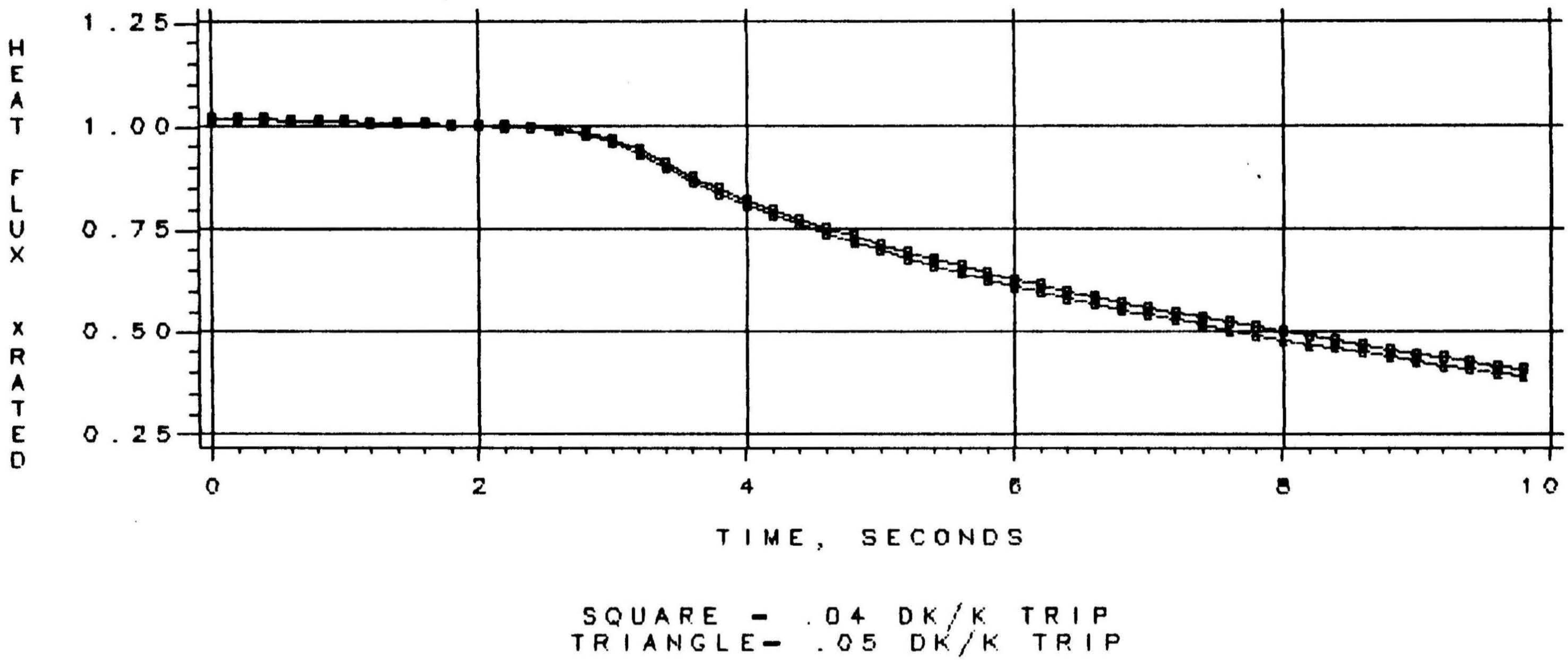
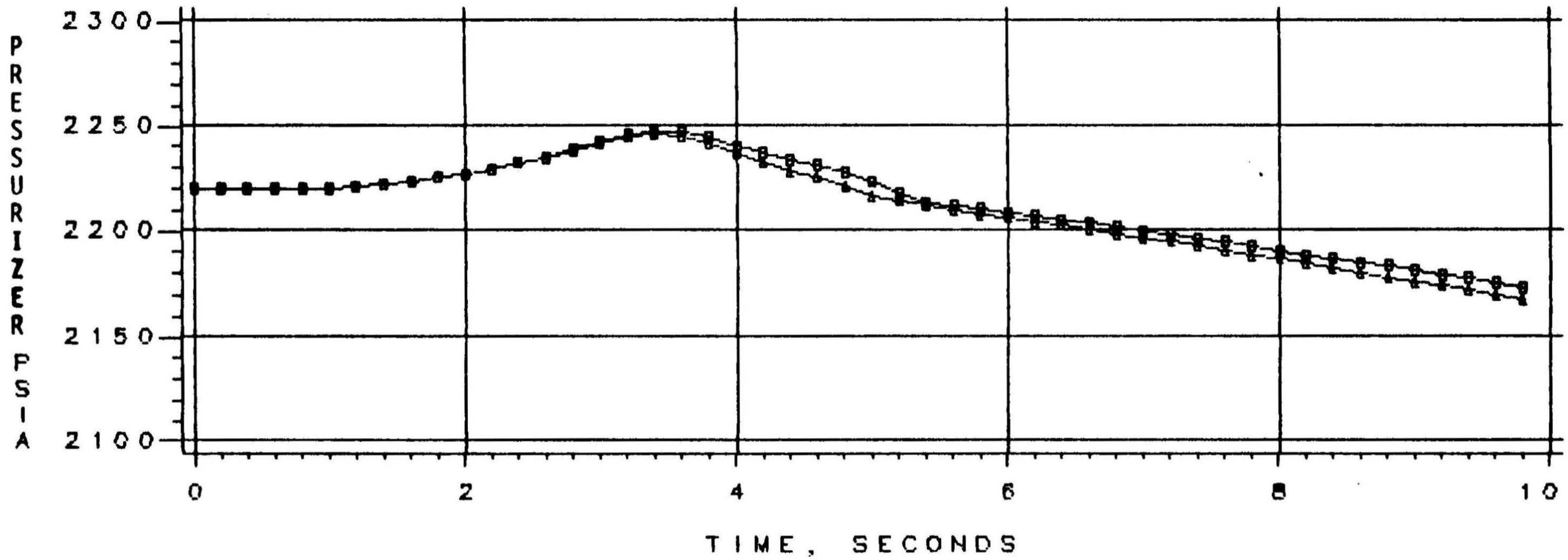


FIGURE VI-42

LOSS OF FLOW

TRIP STUDY

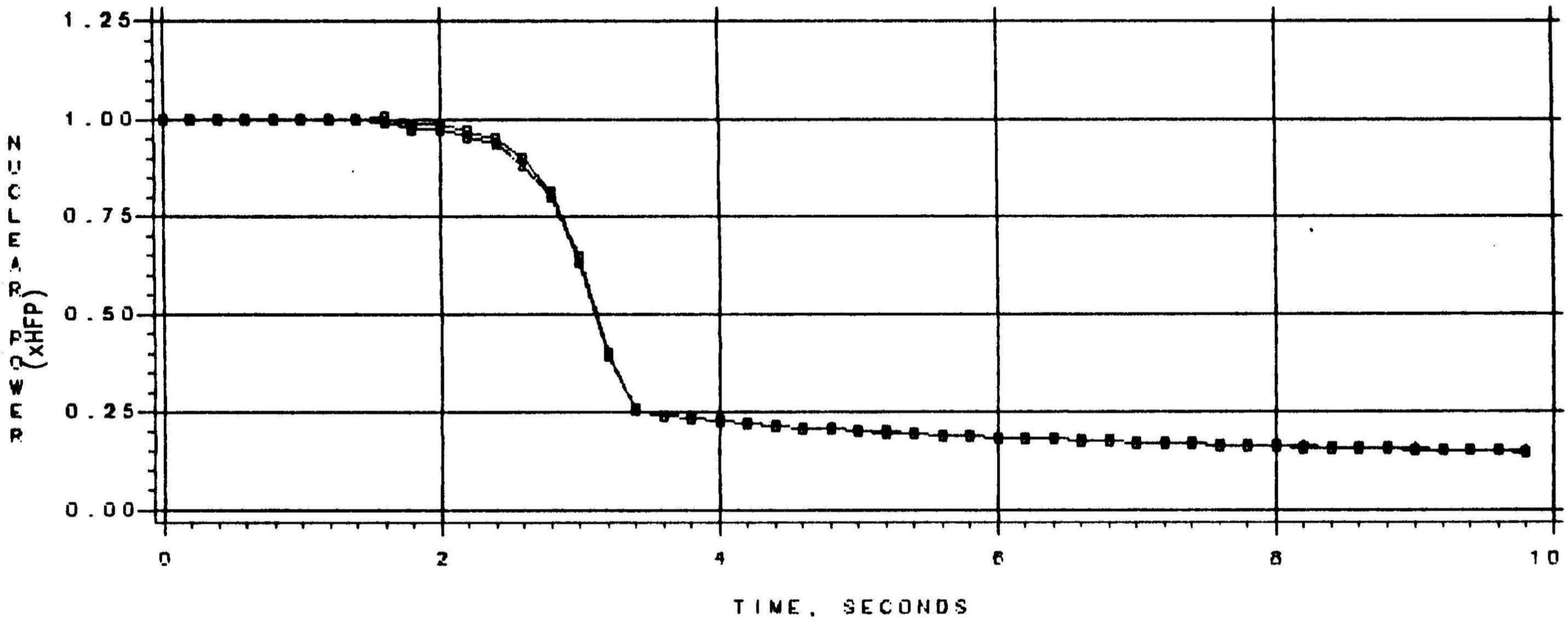


SQUARE - .04 DK/K TRIP
TRIANGLE- .05 DK/K TRIP

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

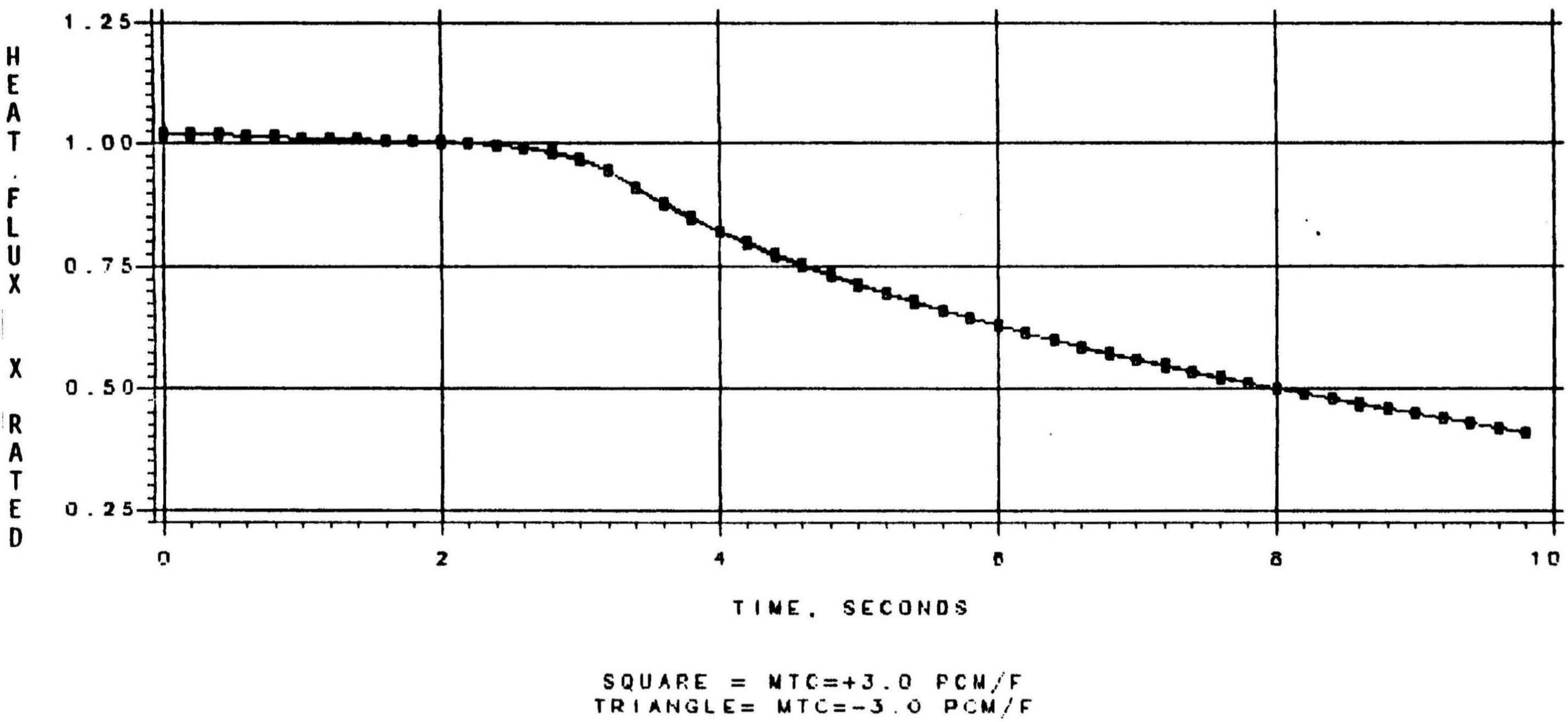
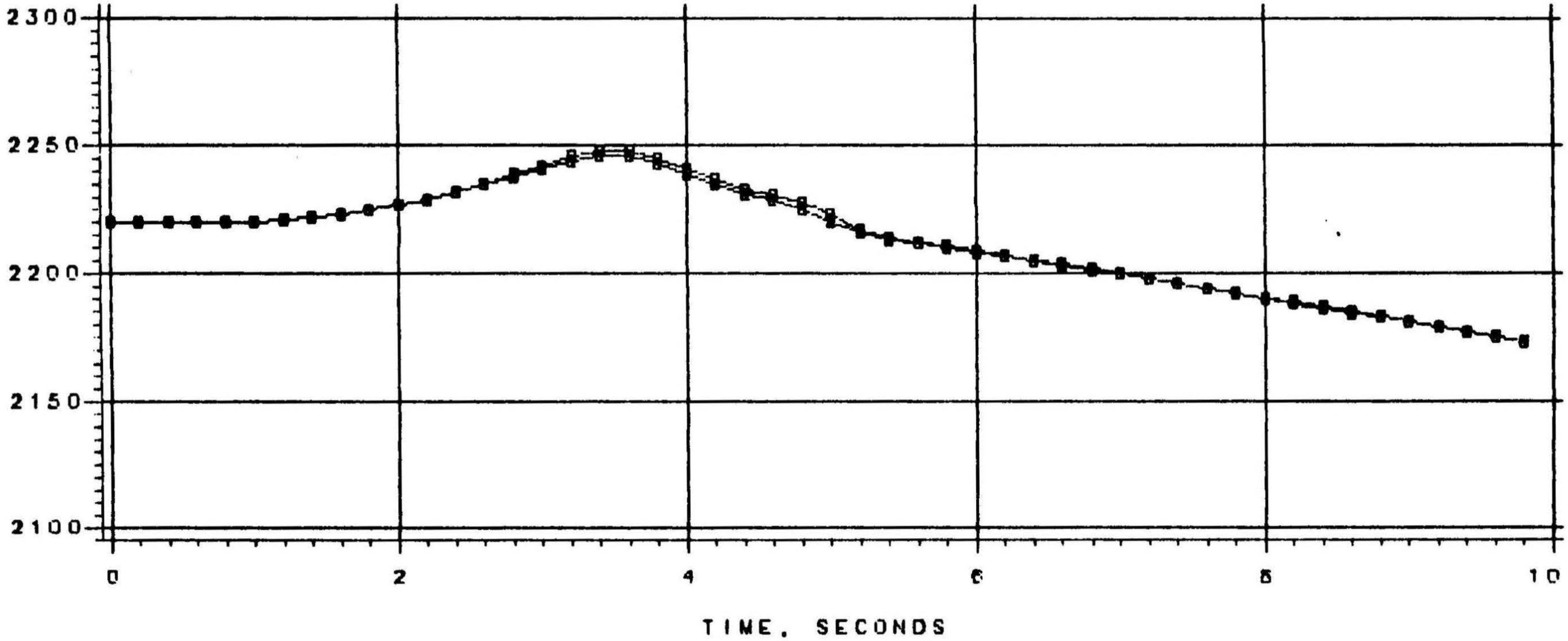


FIGURE VI-45

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

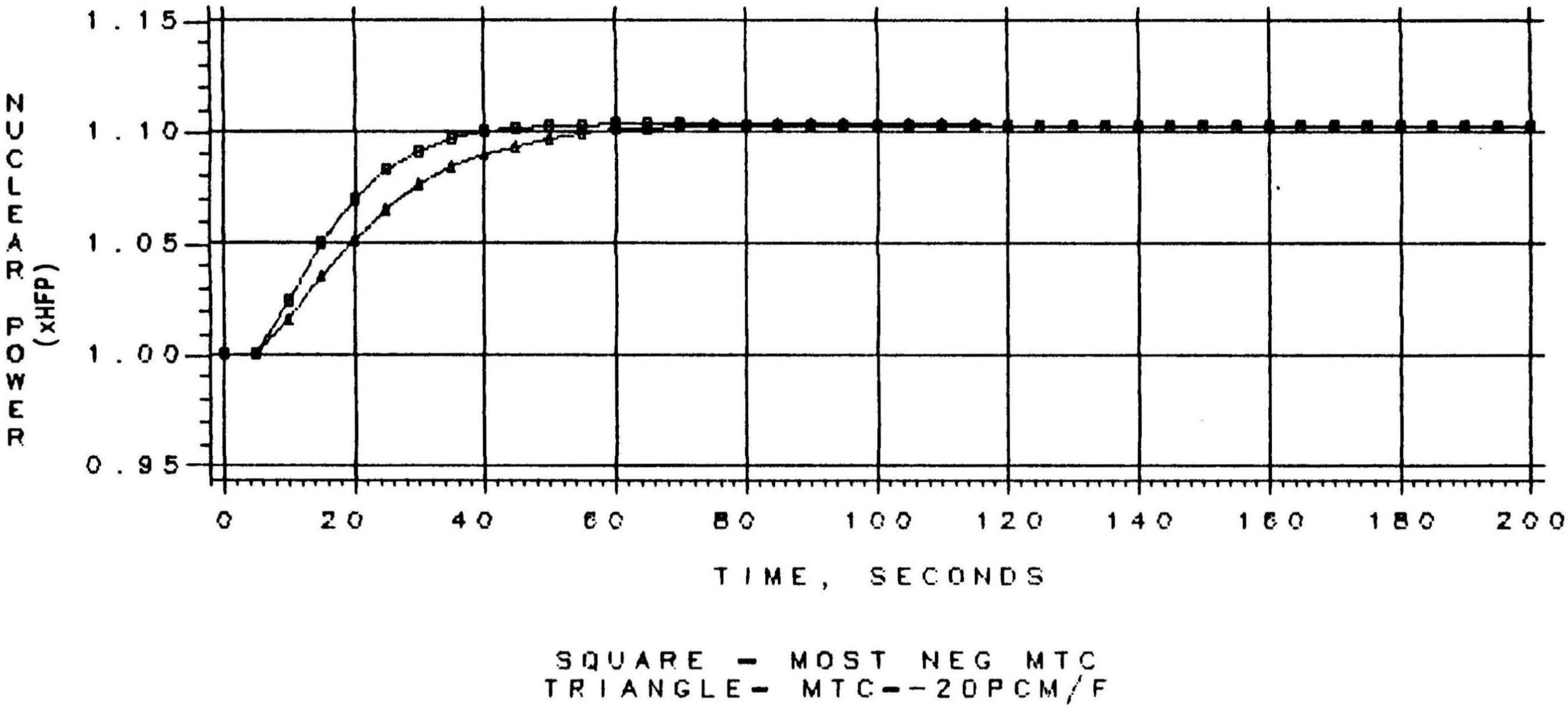


FIGURE VI-47

EXCESS LOAD INCREASE

MTC STUDY

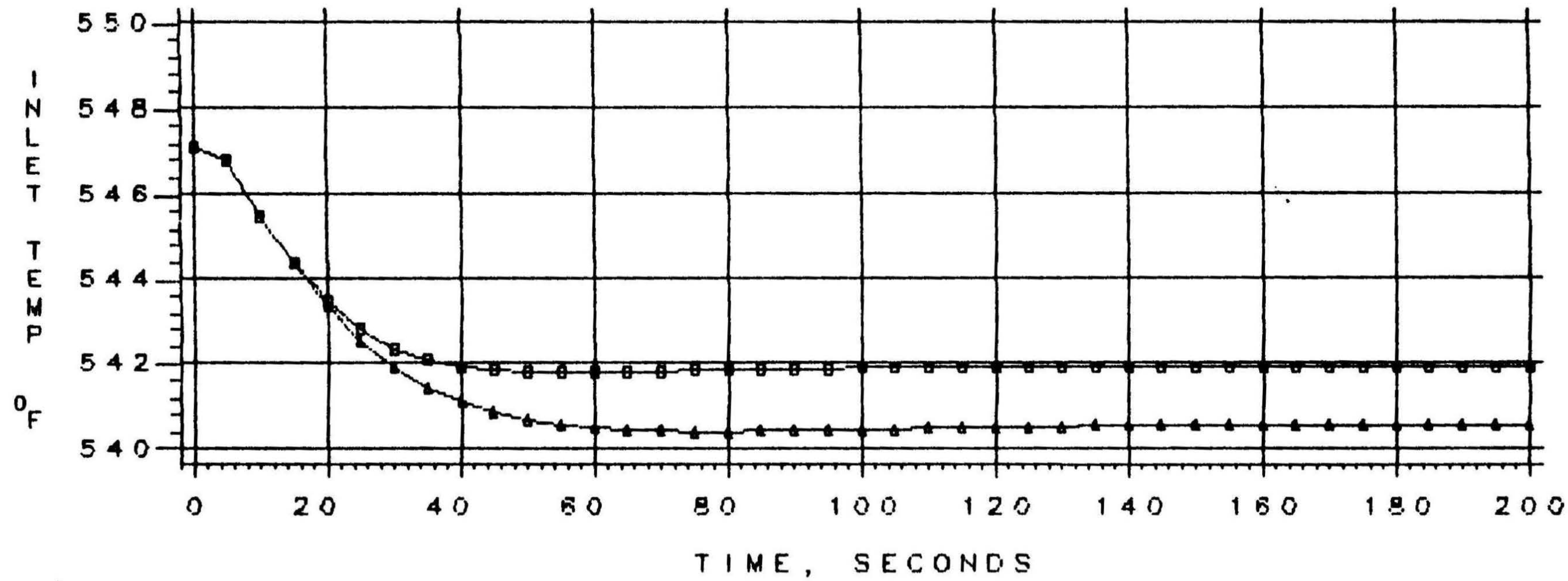
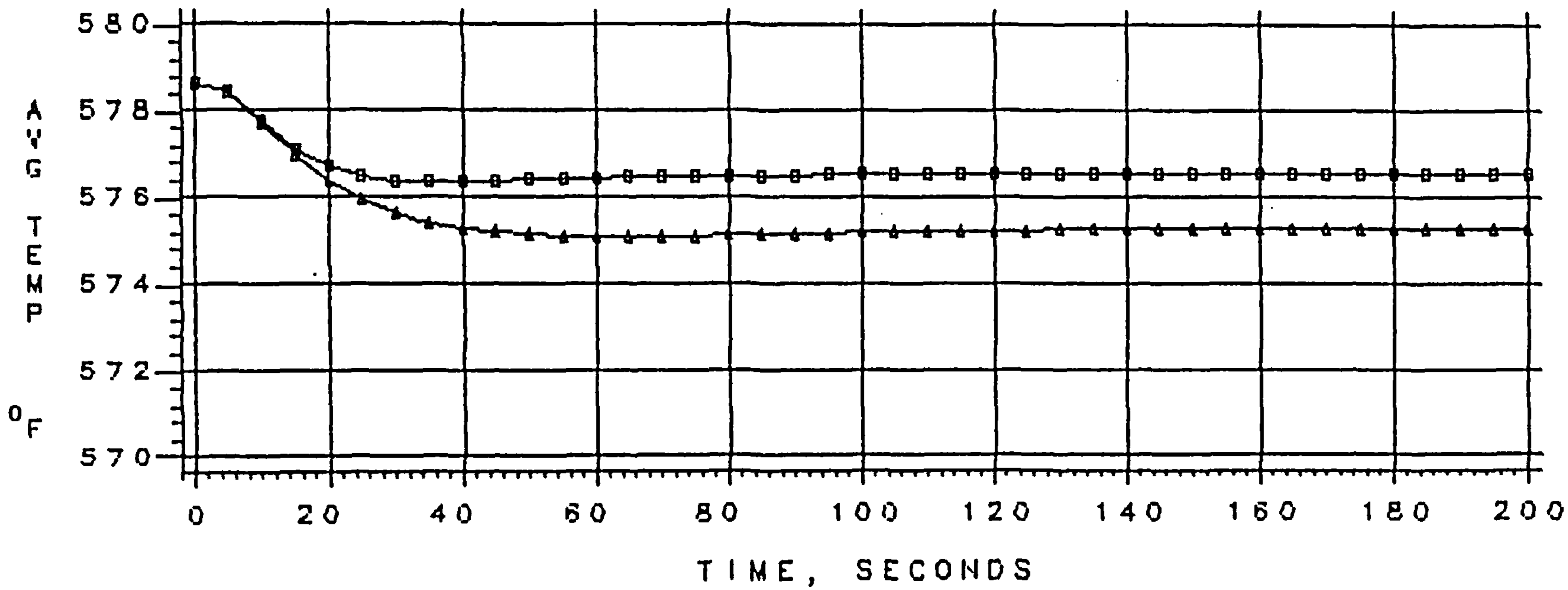


FIGURE VI-48

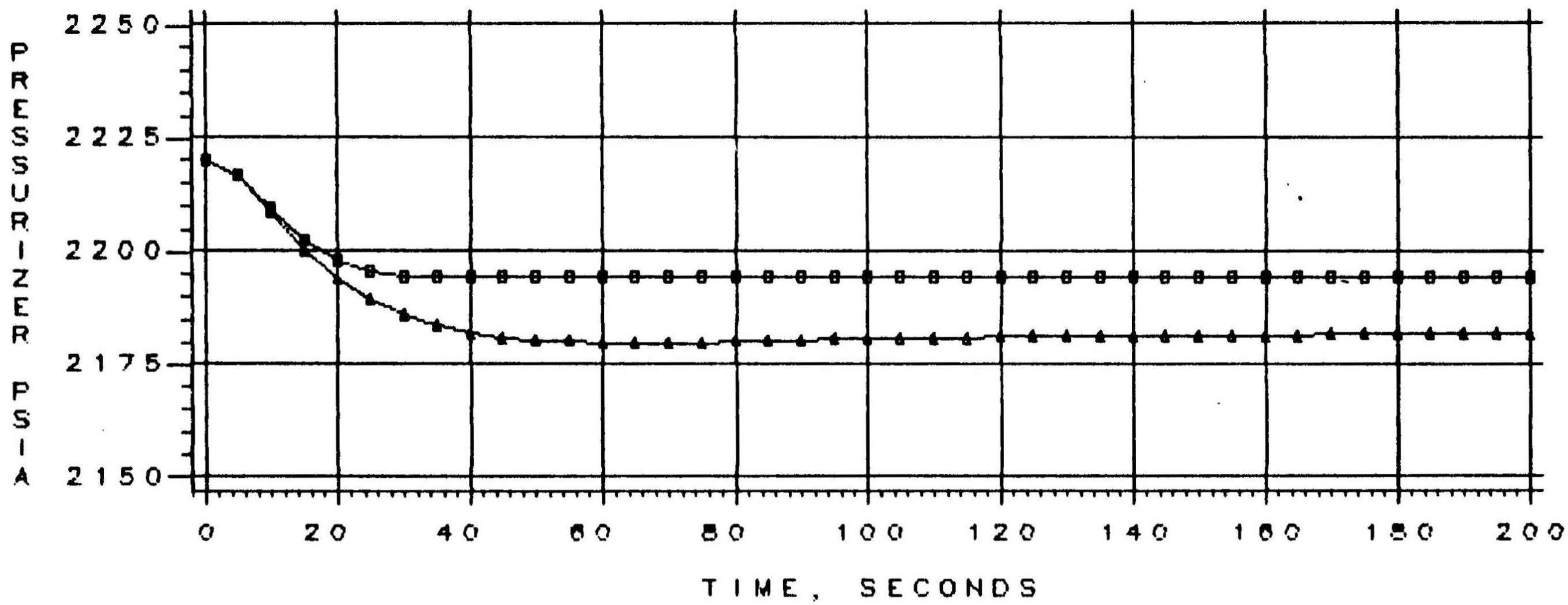
EXCESS LOAD INCREASE

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

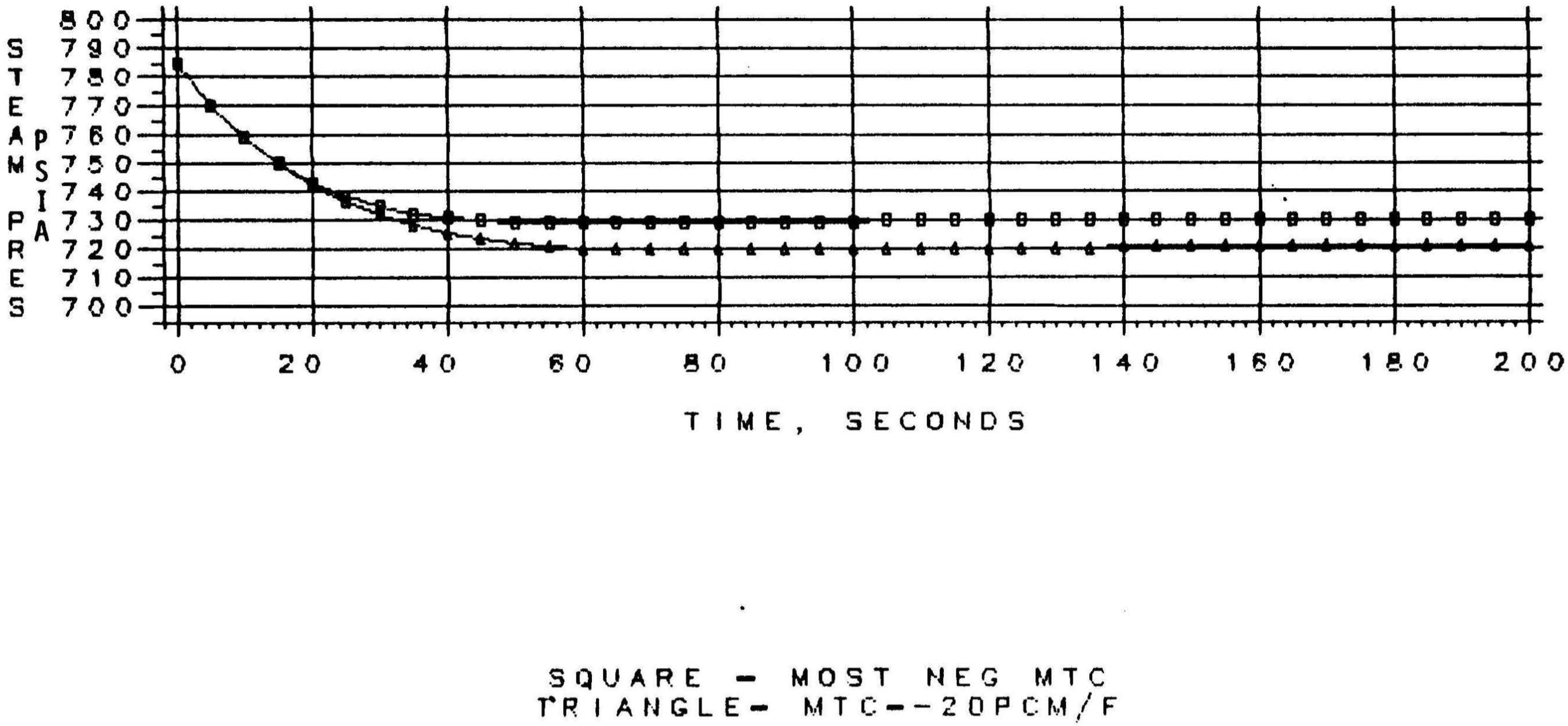
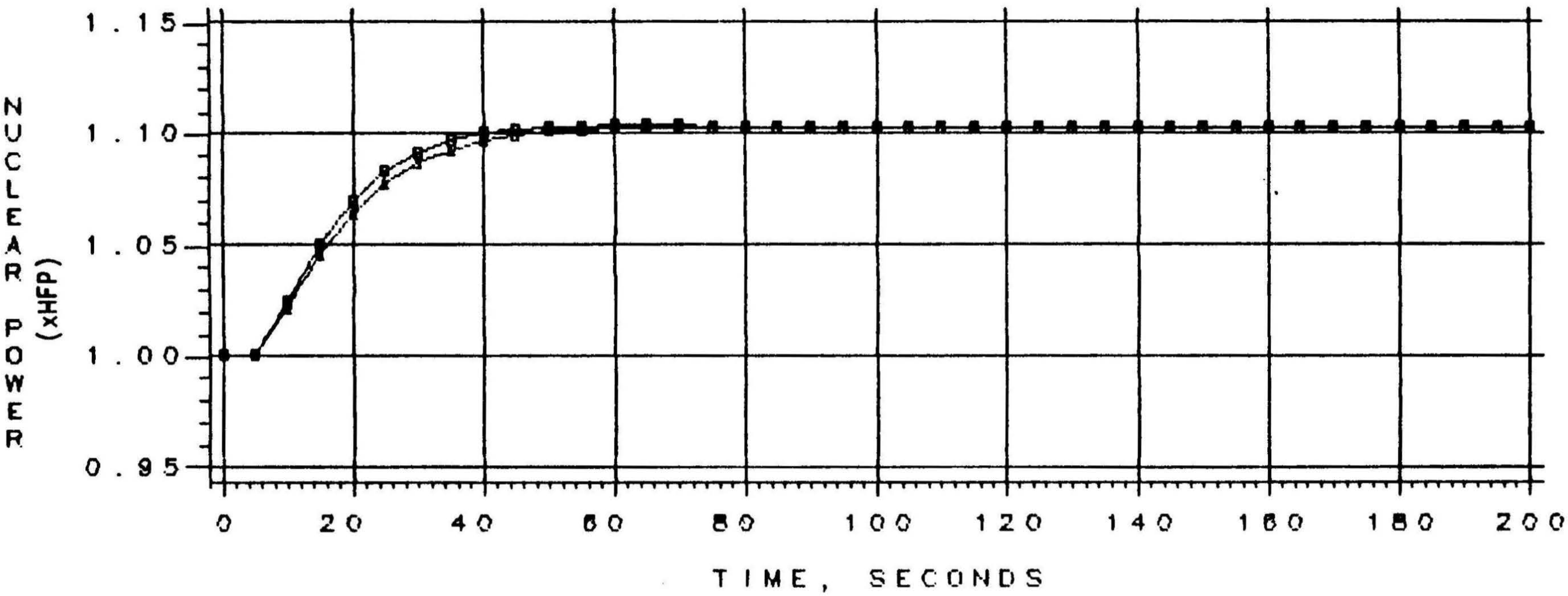


FIGURE VI-51

EXCESS LOAD INCREASE

DPC STUDY

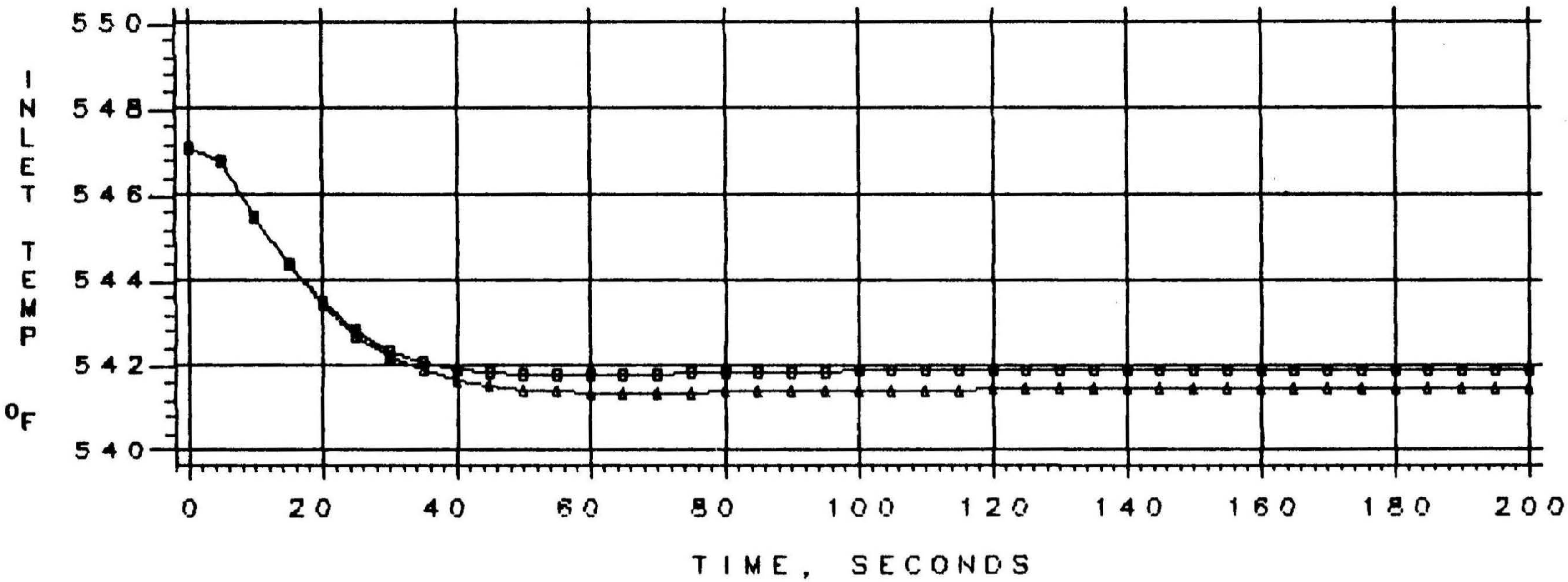


SQUARE - LEAST NEG DPC
TRIANGLE- LEAST NEG DPC*1.25

FIGURE VI-52

EXCESS LOAD INCREASE

DPC STUDY

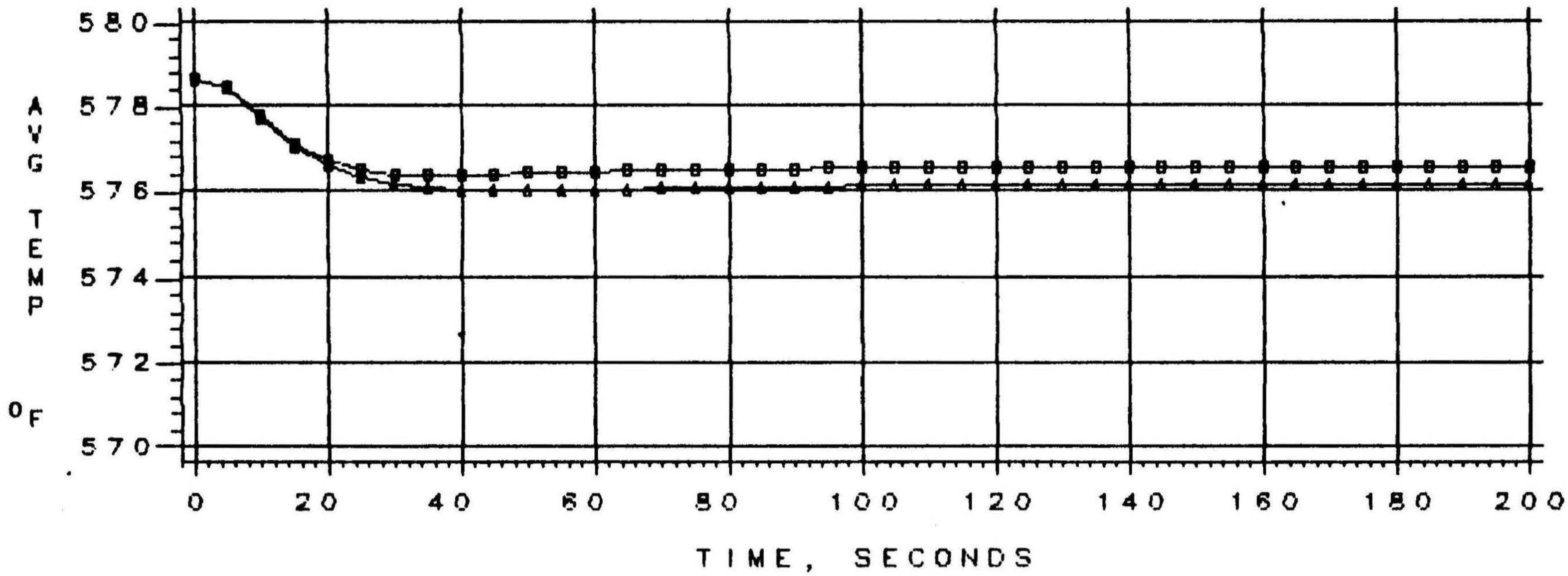


SQUARE - LEAST NEG DPC
TRIANGLE- LEAST NEG DPC+1.25

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

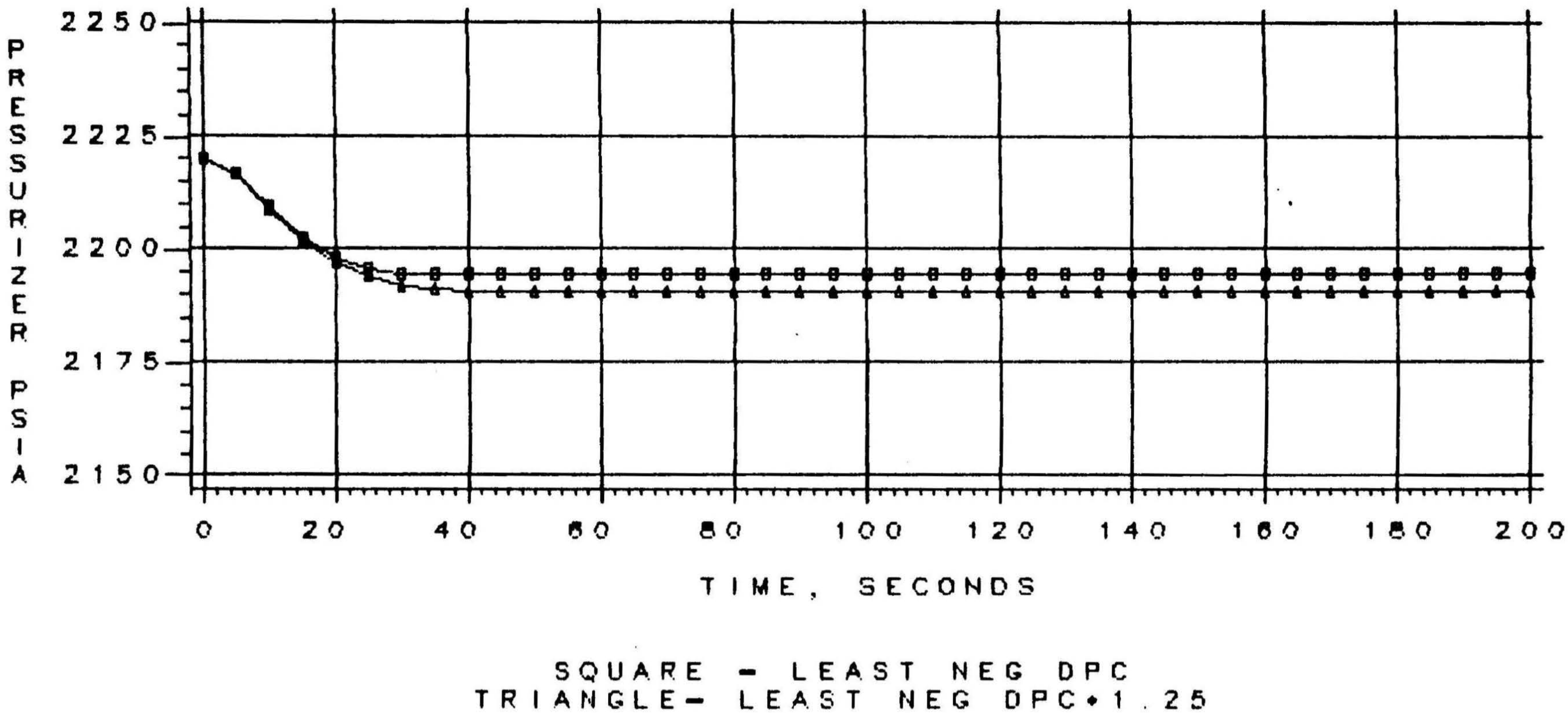


FIGURE VI-55

EXCESS LOAD INCREASE

DPC STUDY

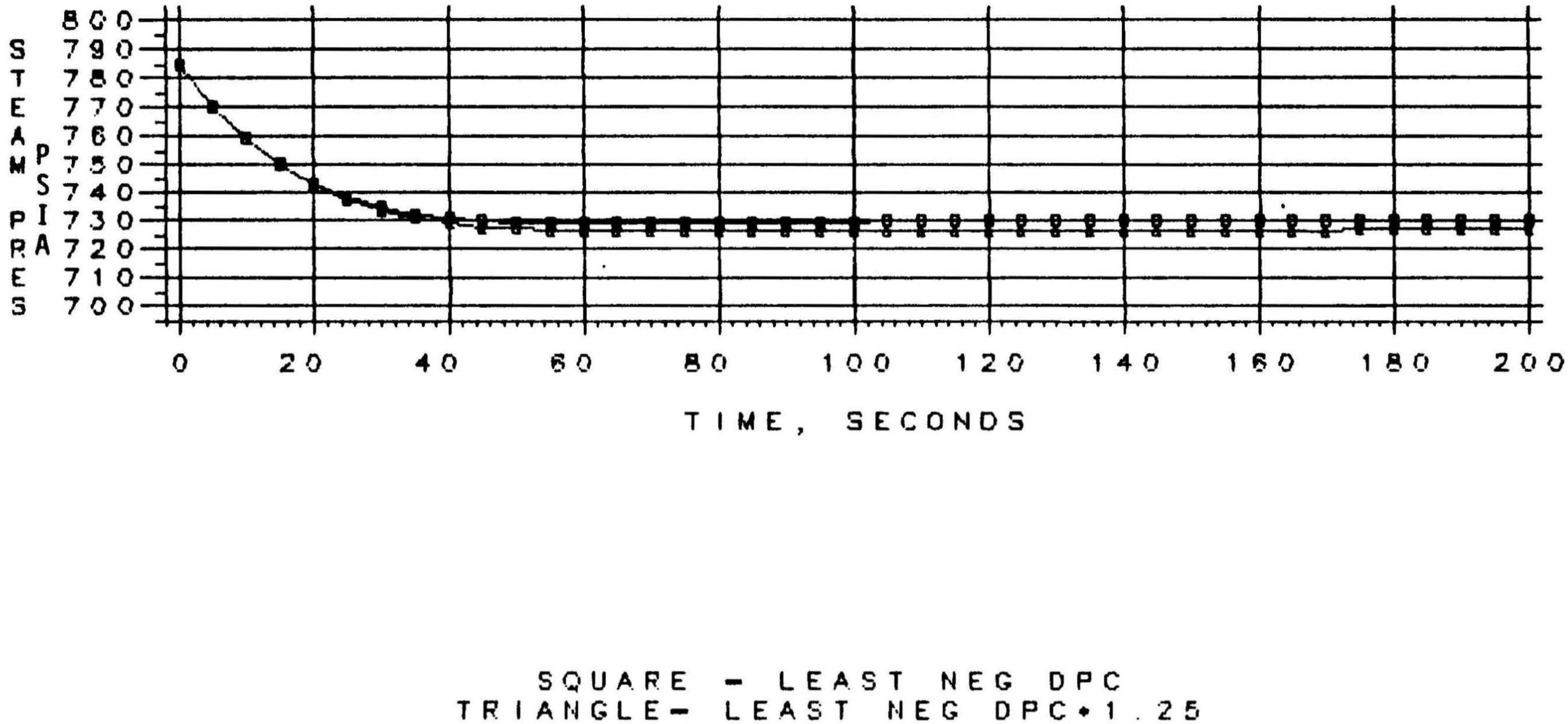
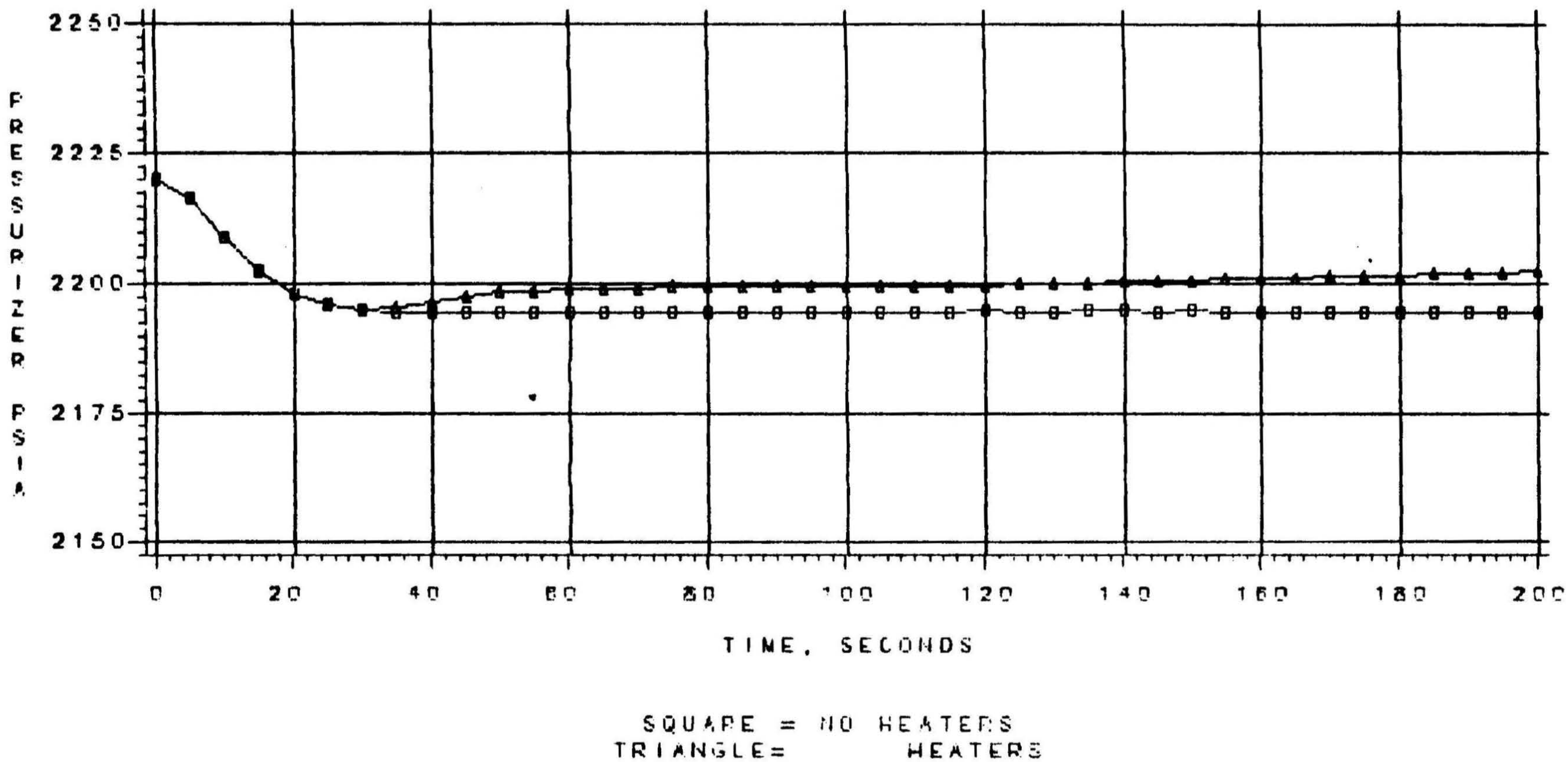


FIGURE VI-56

EXCESS LOAD INCREASE

HEATER STUDY



APPENDIX 4
Responses 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 Prop 1/40 NP Change: PDR LPDR NTIS NSIC } Non Prop

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 vicinity 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.

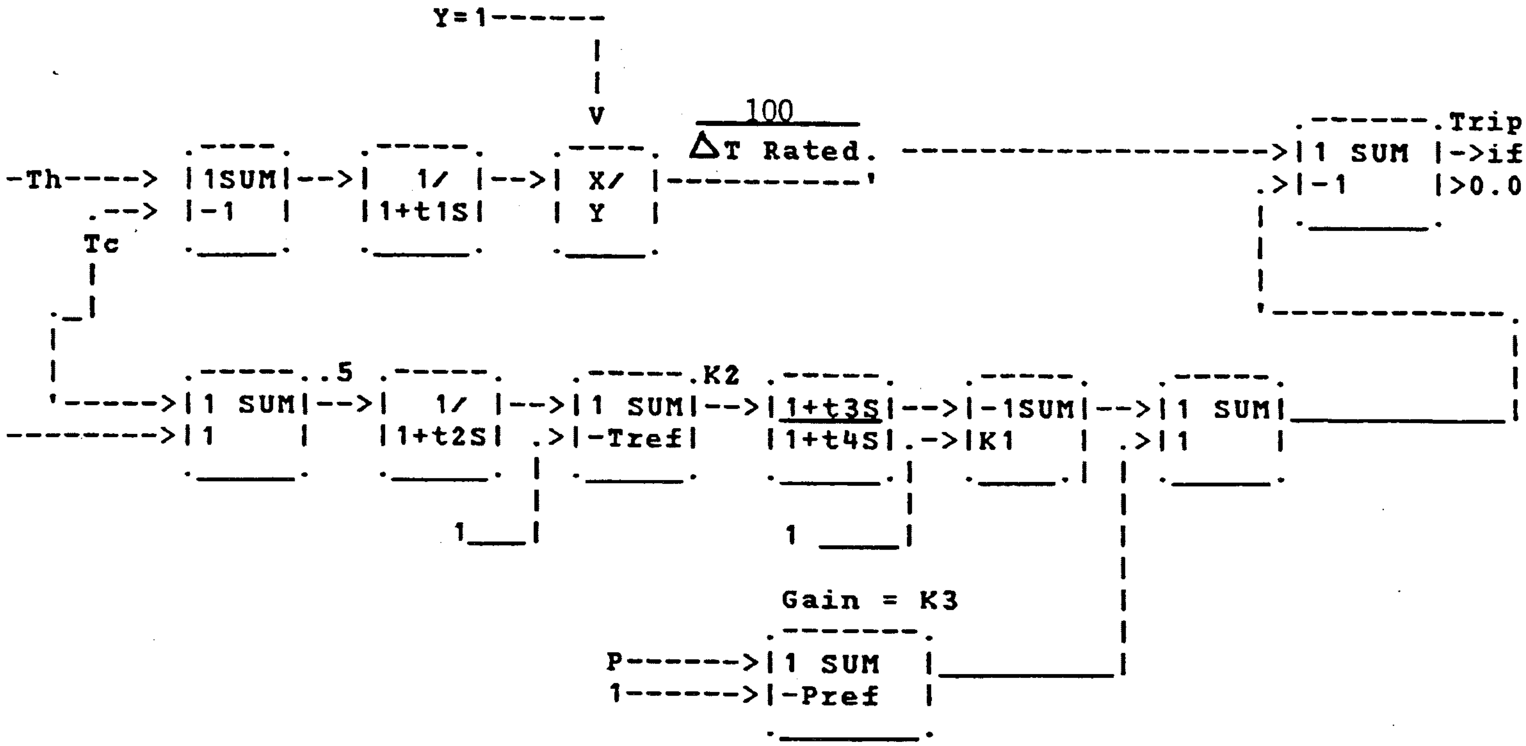
Figure III-9 represents the core average heat flux calculation performed in the two loop model. This heat flux is expressed in terms of fraction of the rated full power value, and is used for editing purposes, and to drive the power reactivity feedback calculation described in Figure III-6 during steamline 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.

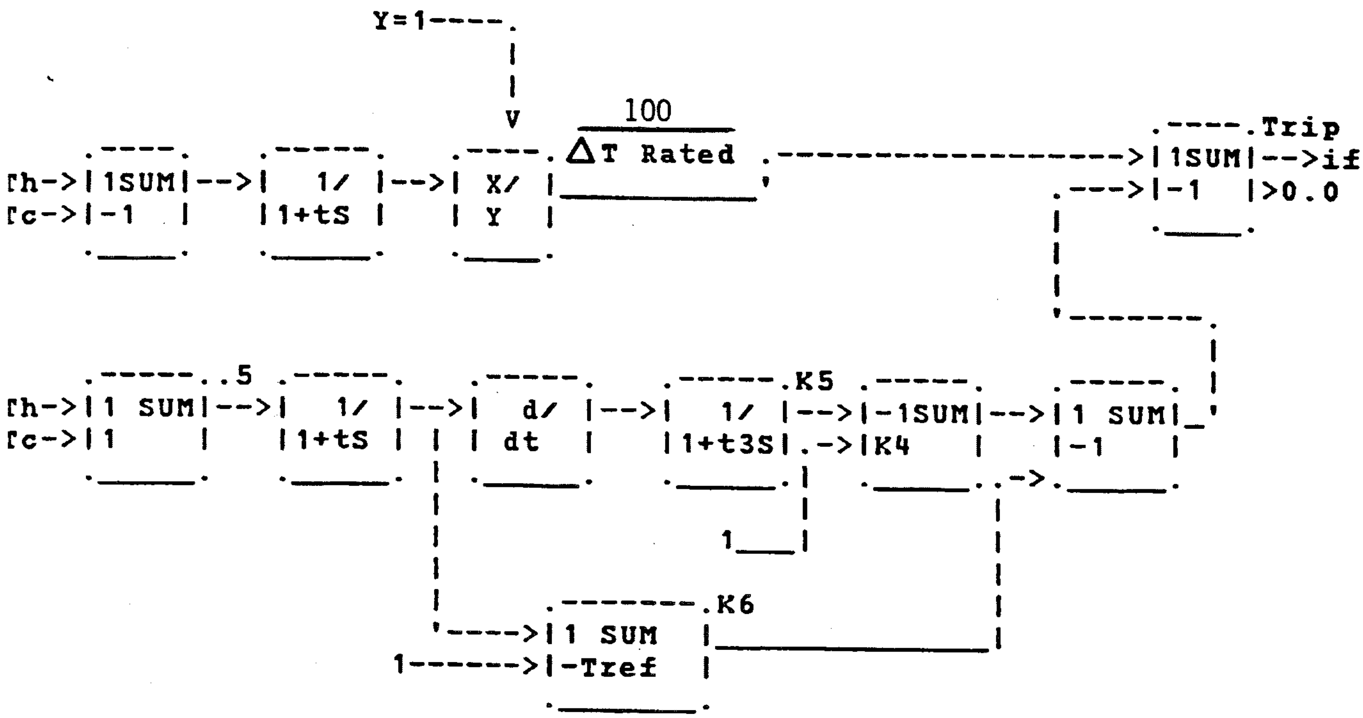
The RETRAN submodel for calculating the mixing and transport of high boron concentration water from safety injection into and around the primary coolant loops is shown in Figure III-12. The model shown is appropriate for full flow conditions in all loops. Pipe-like regions of the system are treated with 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



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

FIGURE III-2
OVERPOWER DELTA-T TRIP



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

FIGURE III-3
LOGIC FOR PRESSURE CONTROL SYSTEM

```
                              .-
                     .---> |Trip No. 4(-4)
                     |     |Heaters on if <A%.      off if >A%*
                     |     .-
                     |     .-
                     |     |Trip No. 9 (-9)
 BISTABLES--->       |---> |Open Power Operated Relief Valve #2
     |               |     |>B% Close if <B%*
     |               |     .-
     |               '-------------------------------------------.
     |                     .-                                    |
     |                     |Trip No. 8 (-8)                      |
     |          .--------> |Open Power Operated Relief Valve     |
     '-->       |          |#1 if >C, close if <C*               |
                |          .-                                    |
                |                             1/T                |
          .---------.    .-----.      .------.     .-------.     |
ontroller |Pressure|---->|1.SUM|---->| INT   |---->|1. SUM  |->-|
nit-----> |         |  .->|-PREF|  |  |       | .-->|1.      |   |
          .________.  |  .____.   |  .______. |   .______.      |
             1.0------'           |___________|                  |
        .--------------------------<-----------------------------'
        |
        |
        |                                 Gain=Full Spray/
        |      .-----.Max=1.0    .------.  Loop Flow    --  --
pray Valve '-->|1.SUM|---------->|1.0MUL|------------->|Fill|
ontroller-->   |-Y** |        .->|1.0   |      |       | #  |-->Spray
               .____.Min=0.0|    .______.      |       | 1  |Junction (19)
                             |                  |       --  --
                             |                  |         Gain=-1
      Cold Leg Flow     _____|                  |    .-----.    --    --
                                                '-->|1.    |-->|Fill |Spray
                                           1.0--->| MUL    |   |#    |Intake
                                                  .______.     |2    |(18)
                                                               --    --
```

* 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

```
                                   .----------.
                                   |          |
PRESSURIZER ------------->|  1+T1S     |-------> TO REACTOR TRIP
PRESSURE                           |  _____   |        ON LOW PRESSURE
                                   |  1+T2S   |
                                   .__________.
```

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

```
                          .-----------.
                          |           |
FAULTED LOOP------------->|  1+T1S    |-------> TO SAFETY INJECTION
STEAM PRESSURE            |  ------   |         LOGIC
                          |  1+T2S    |
                          .-----------.
```

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.

IIS 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
                                 .__A___.                .______.        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

```
           .-------.GAIN=.5    .---------.     .---------.
HOT-->|          |---------->|+1        |---->| 1+T1S   |
COLD->|  SUM     |    TREF-->|-1 SUM    |     | ------  |
           .________.          .__________.     | 1+T2S   |
                                                .__________.
                                                     |
                                                  FILTERED
                                                (TAVE-TREF)
                                                     |
                                                     V
                                                .---------.
                   TABLE OF DUMP CAPACITY       |         |
                       VS (TAVE-TREF)-------->| FNG     |
                                                |         |
                                                .__________.      .---------.
                         .------.GAIN= 100.          |            |         |
                         |       |MAX=1.0            '--------->|  MUL    |
  TURBINE TRIP------>|INT    |-------------------------->|         |
  SIGNAL (0 OR 1)       |       |                                .__________.
                         .______.                                    |
                                                                     |
                                                                     V
                                                              TO VALVE AREA
                                                                 TABLE
```

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

```
                                              Gain=100000000        Gain=1/60
                                             .-------. Max=1.0     .------.
                     .----.  Integrated-->|1 SUM |------------->| MUL  |
Safety Inj. ---->|INT  |--Volume       ->|-Vpurg|             .->|      |
 Flow, gpm       .____.                 | .______.Min=0.0      |  .______.
     |                            1.0-'                        |     |
     |_________________________________________________________|     |
                     Gain=1/VBIT                                      |
                     .----.    .-----.     .-----.                    |
                 .-->|INT  |--->| XPO  |---->|C2SUM|--->Boron conc. exiting
                 |   .____.  .>|      |  .->|C1   |    Boron Injection Tank
                 |     "e"-'  .______.  |   .______.                |   |
                 |                1.0--'                           V   V
          .------'------------------------------------<------------'   |
          |                                                             |
          | .-----------------------------------------------------------'
          | '>.---.  G1.---.    .---.    .---.    .---.  .---.   .---.
          |   |MUL|--->|SUM|-->|DLY|-->|LAG|-->|DLY|->|DLY|->|LAG|6
          '-->.___. -->.___.   .___.   .___.   .___.| .___.  .___.
              .-----'    1       2       3       4  |   5      |   Boron
              |                                     '----------V-->at core
              | .---.  .---.  .---.  .---.  .---.  .---.      .---. midplane
              '-|DLY|<-|LAG|<-|DLY|<-|LAG|<-|DLY|<-|LAG|<----|DLY|7
                .___.  .___.  .___.  .___.  .___.  .___.      .___.
                 13     12     11     10      9      8
```

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 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.

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

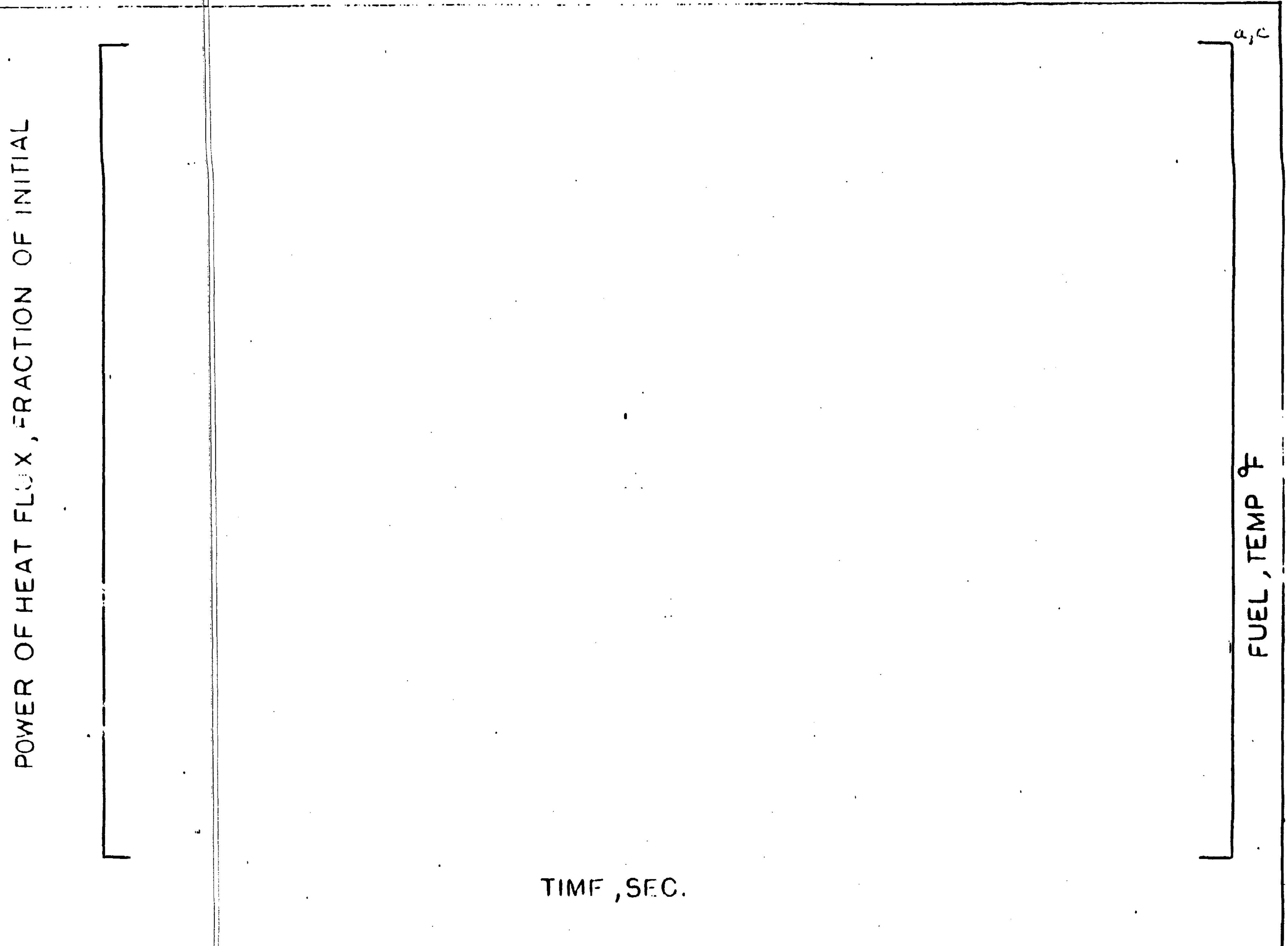
Figures V-9 to V-11 show comparisons for the flow coastdown event. Total core flow (this is a three-pump coastdown) is shown in Figure V-9. LOFTRAN uses a lumped parameter approach in solving for loop flow (the rate of change of flow is a characteristic of the entire coolant loop), whereas RETRAN solves 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 .


a,c
POWER OF HEAT FLUX, FRACTION OF INITIAL
FUEL, TEMP °F
TIMF ,SEC.

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

a,c

STEAM PRESSURE , PSIA.

TIME ,SEC.

a,c

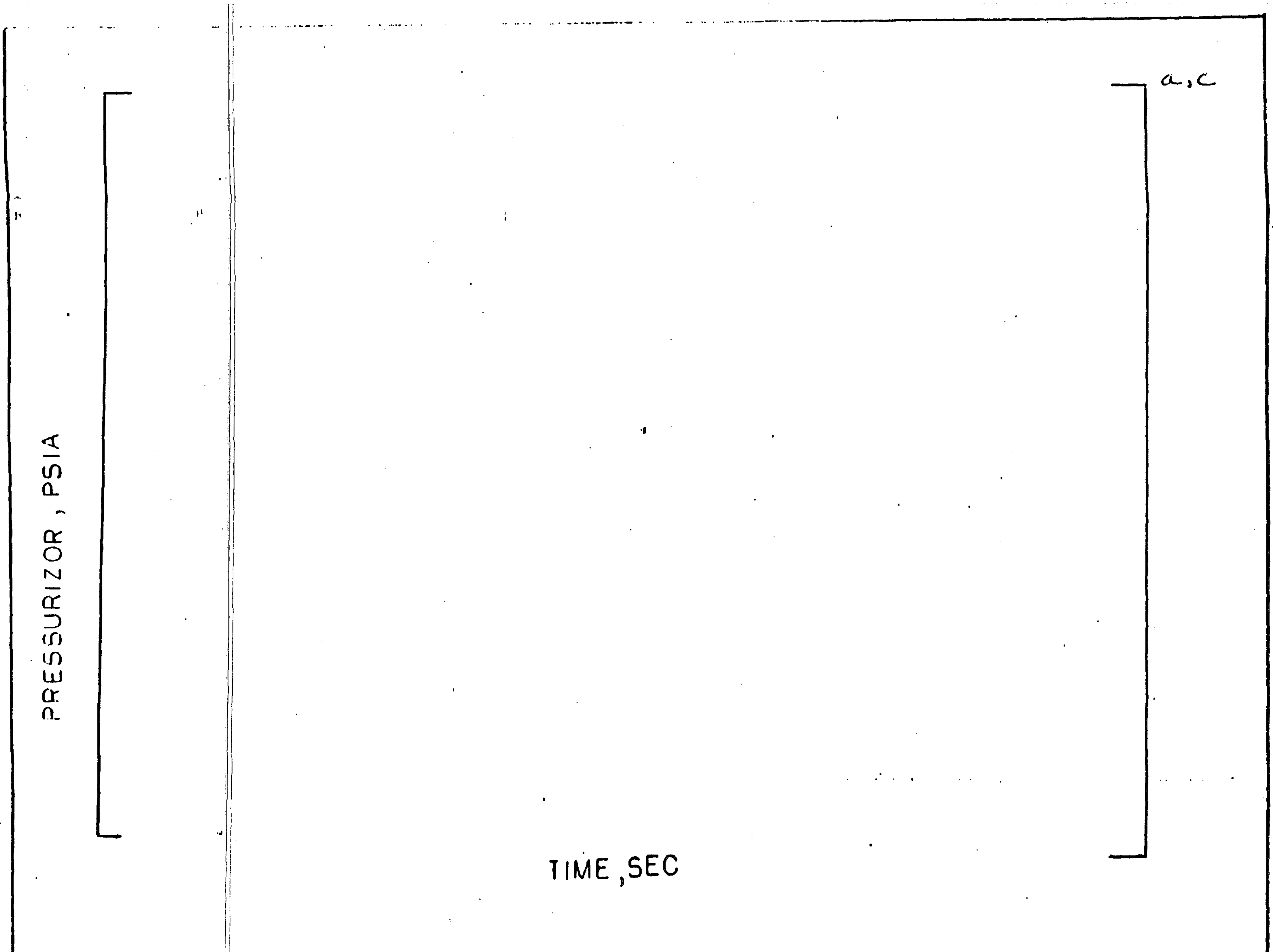
INLET TEMP., °F

TIME, SEC.

NUCLEAR POWER, FRACTION OF INITIAL

a,c

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

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed
before me this 5 day
of July 1976.

Notary Public

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.

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

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Reactor Trip

Turbine Trip No Rx Trip

LIST OF FIGURES (CONT.)

Loss of Flow

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	Time(seconds) 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

Table 3. Significant Events During Turbine Trip Transient

Event	Setpoint Value	Time(seconds) RETRAN01	Time(seconds) 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

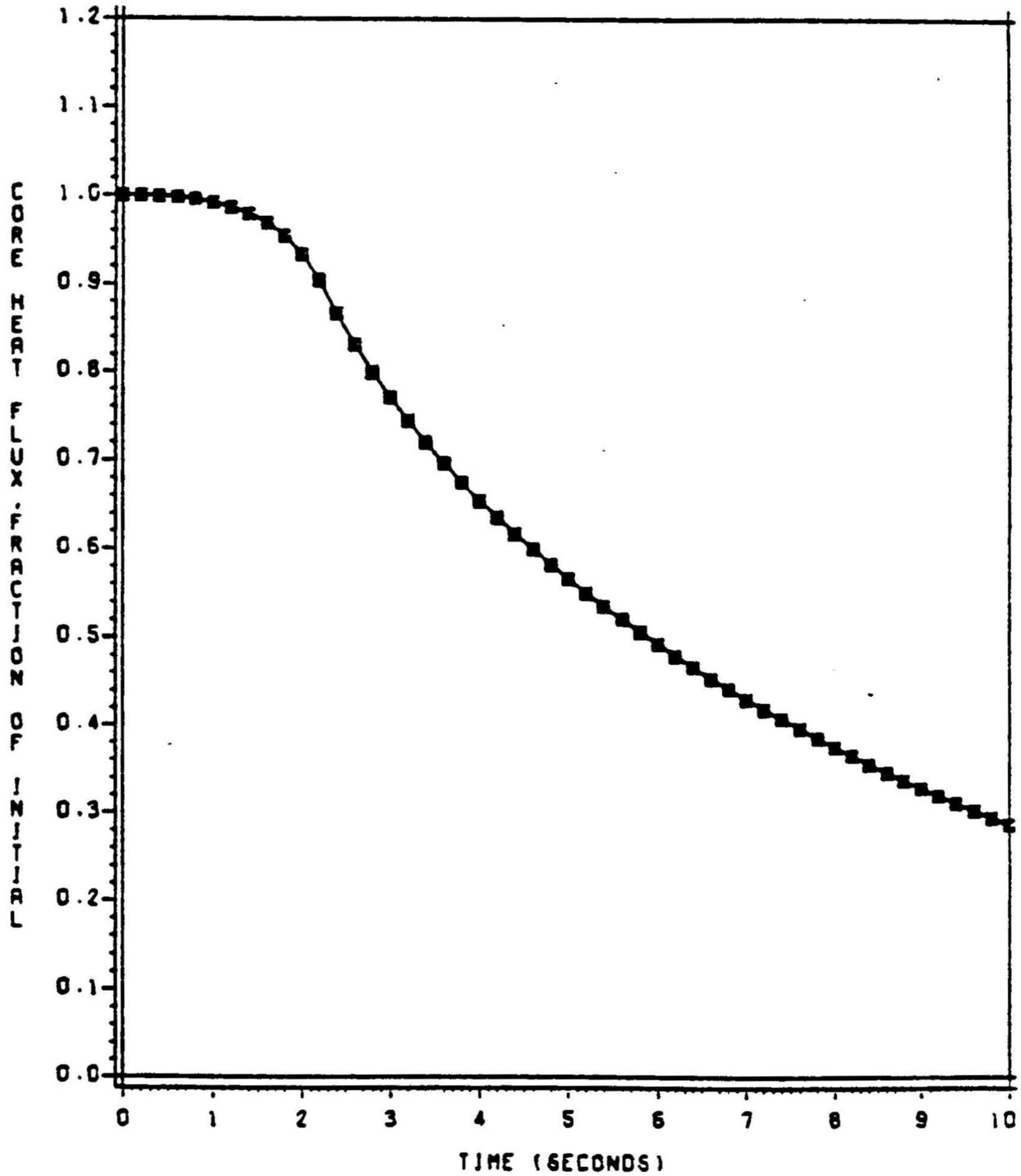
Table 4. Significant Events During Loss of Flow Transient

Event	Setpoint Value	Time(seconds) RETRAN01	Time(seconds) 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

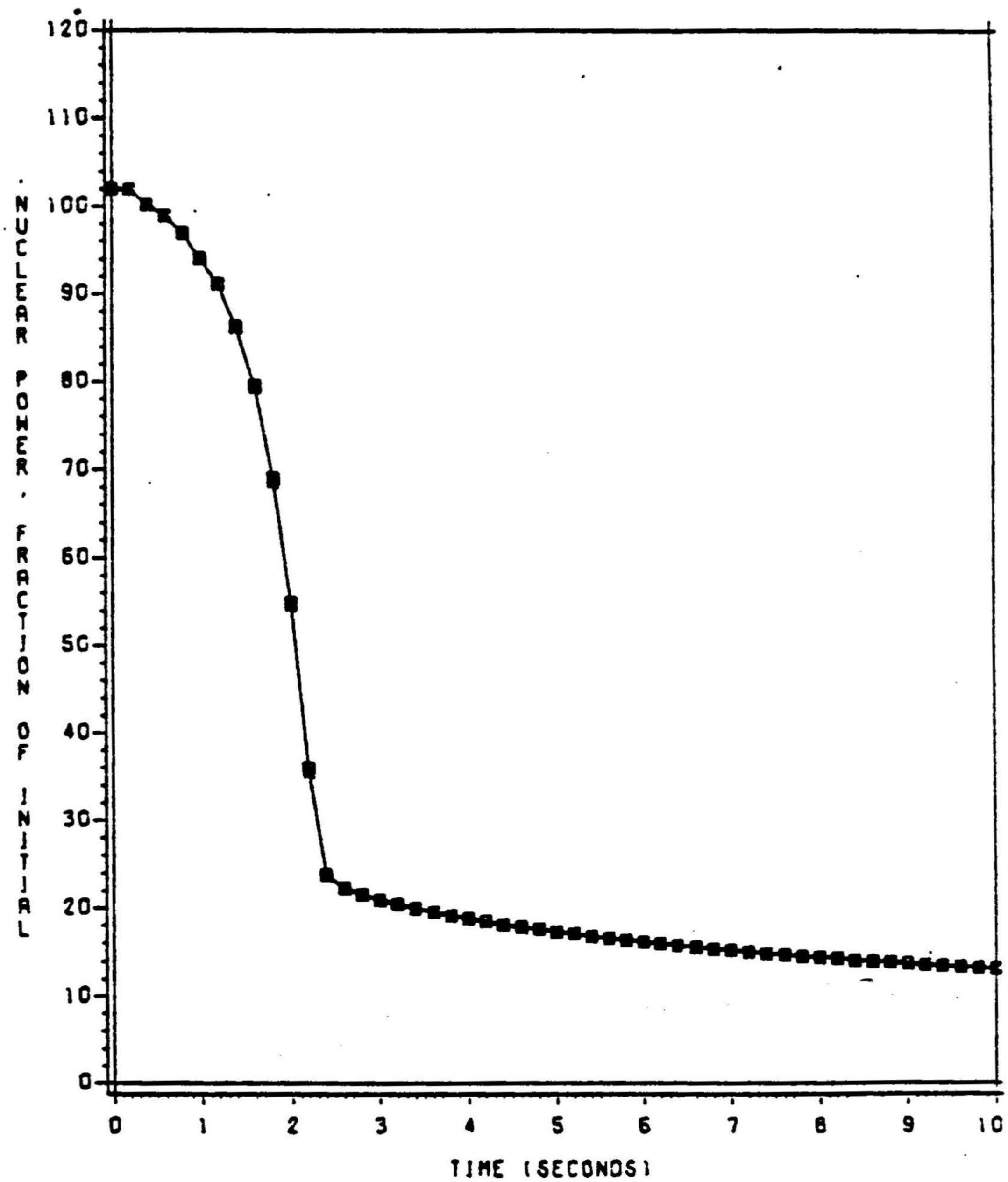


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FIGURE 3a
REACTOR TRIP
MIDCORE FUEL TEMP
RETRAN01 VS RETRAN02

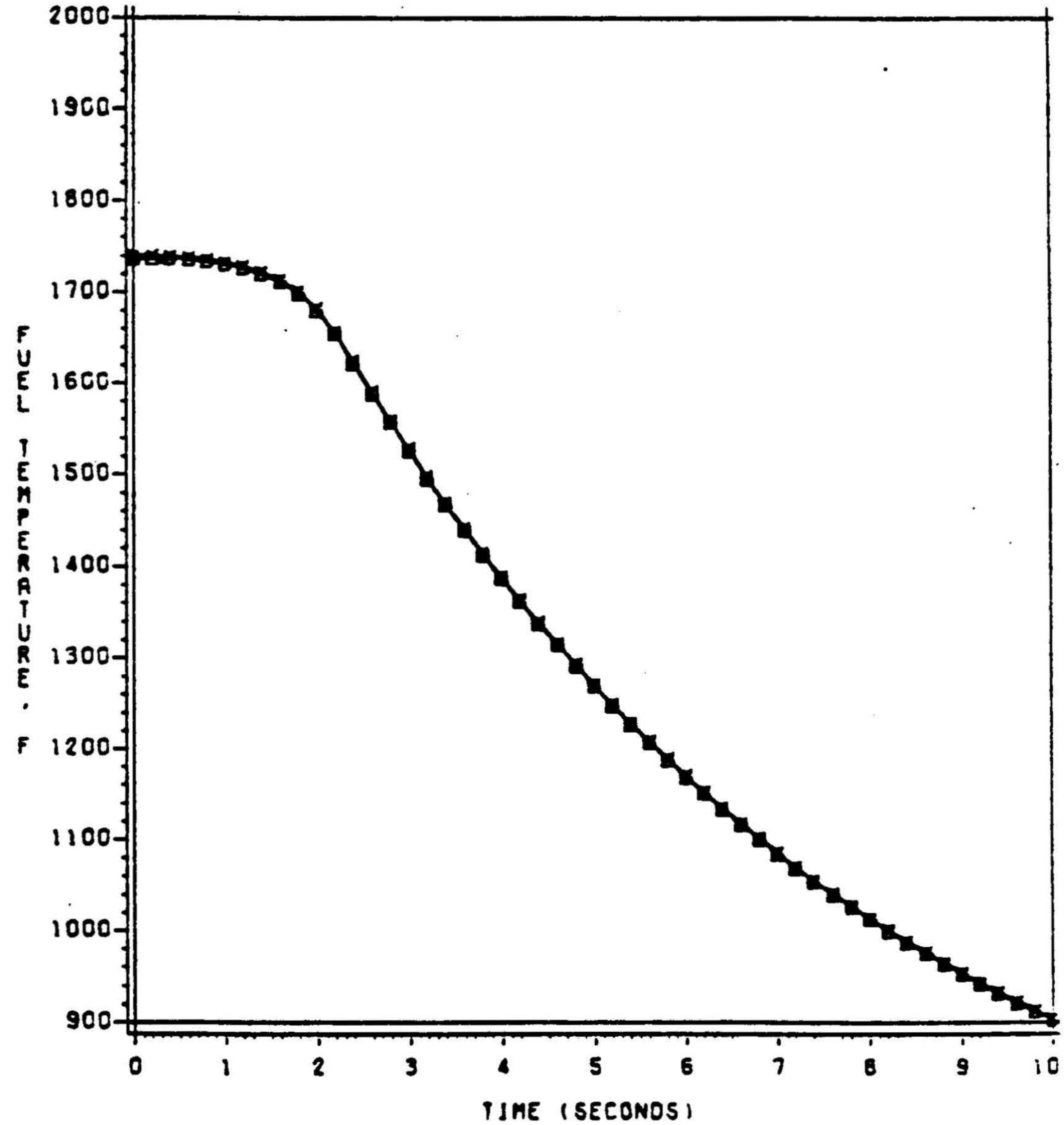

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FIGURE 4a
REACTOR TRIP
SG HEAT EXTRACTION RATE
RETRAN01 VS RETRAN02

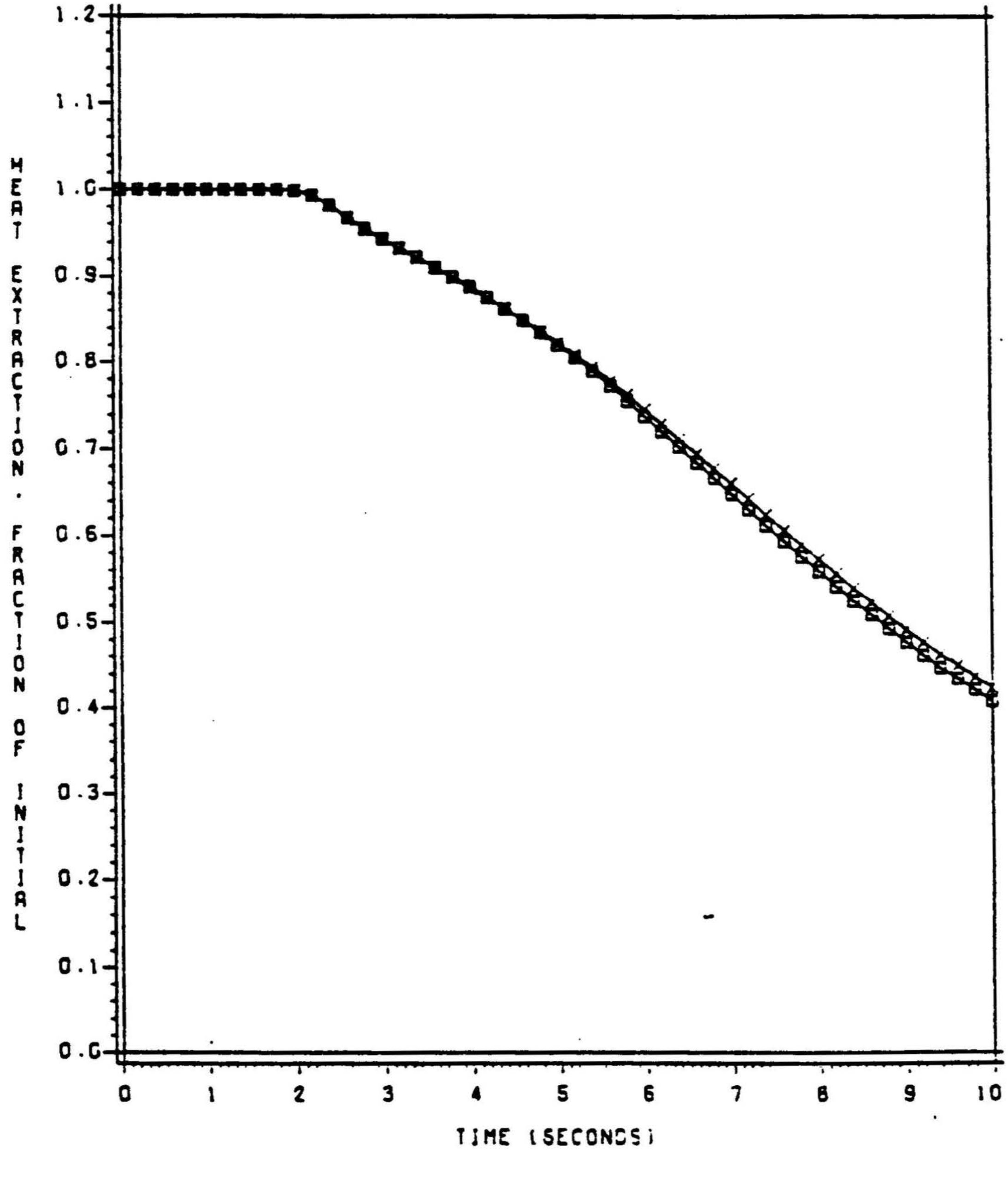

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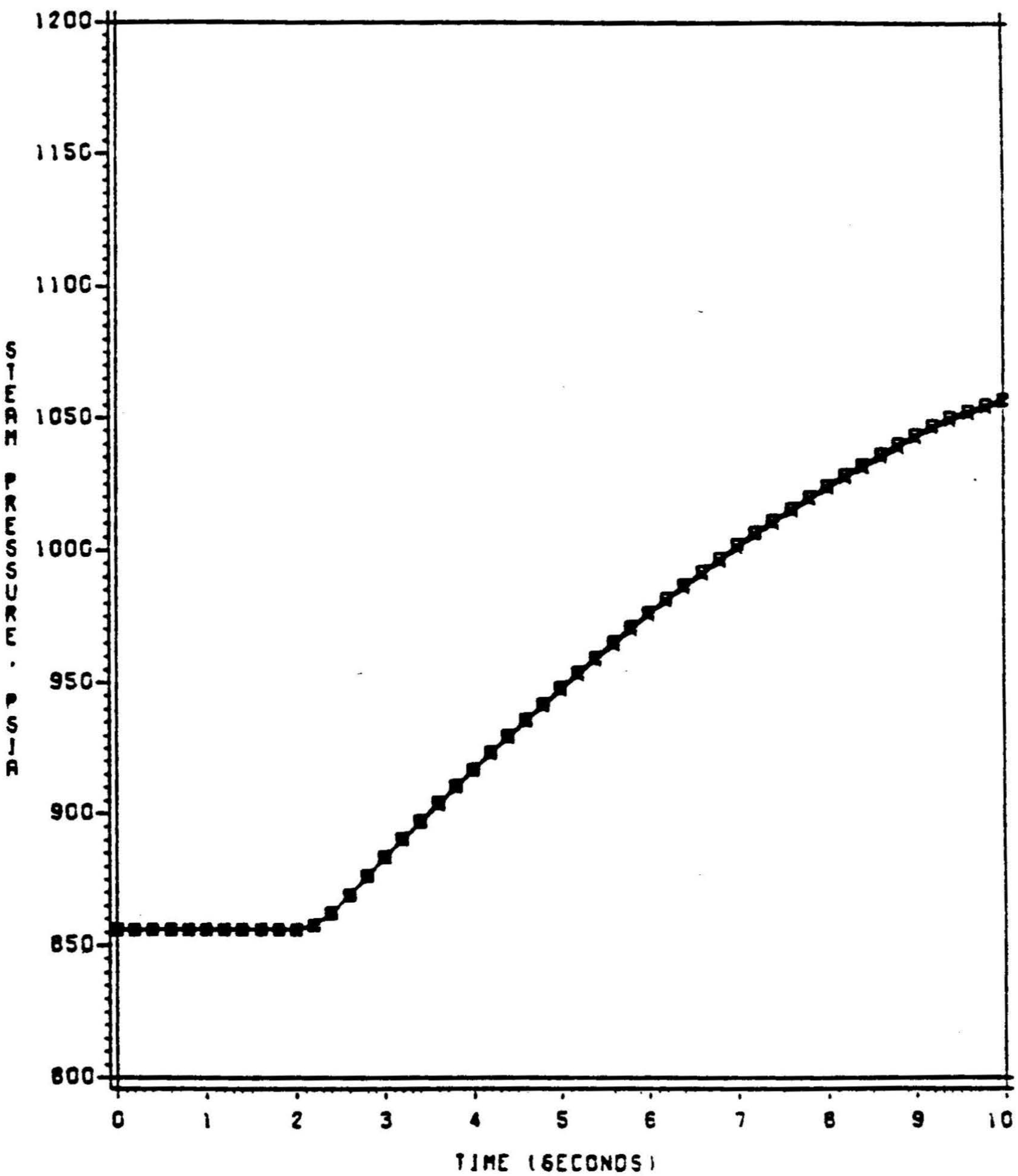
FIGURE 5a
REACTOR TRIP
STEAM PRESSURE
RETRAN01 VS RETRAN02



RETRAN01=X
RETRAN02=SQUARE

FIGURE 6a
REACTOR TRIP
INLET TEMPERATURE
RETRAN01 VS RETRAN02

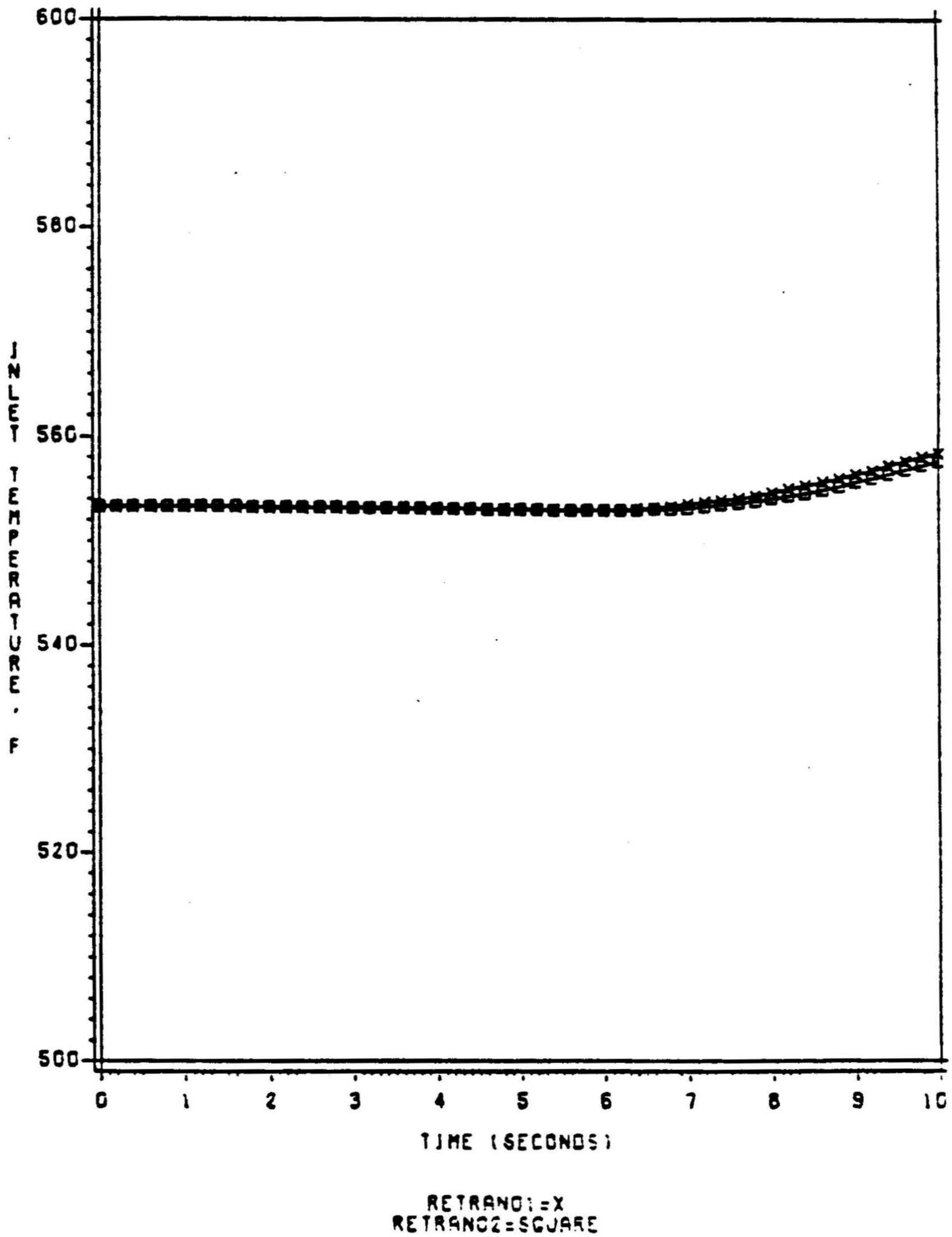


FIGURE 7a
REACTOR TRIP
PRESSURIZER PRESSURE
RETRAN01 VS RETRAN02

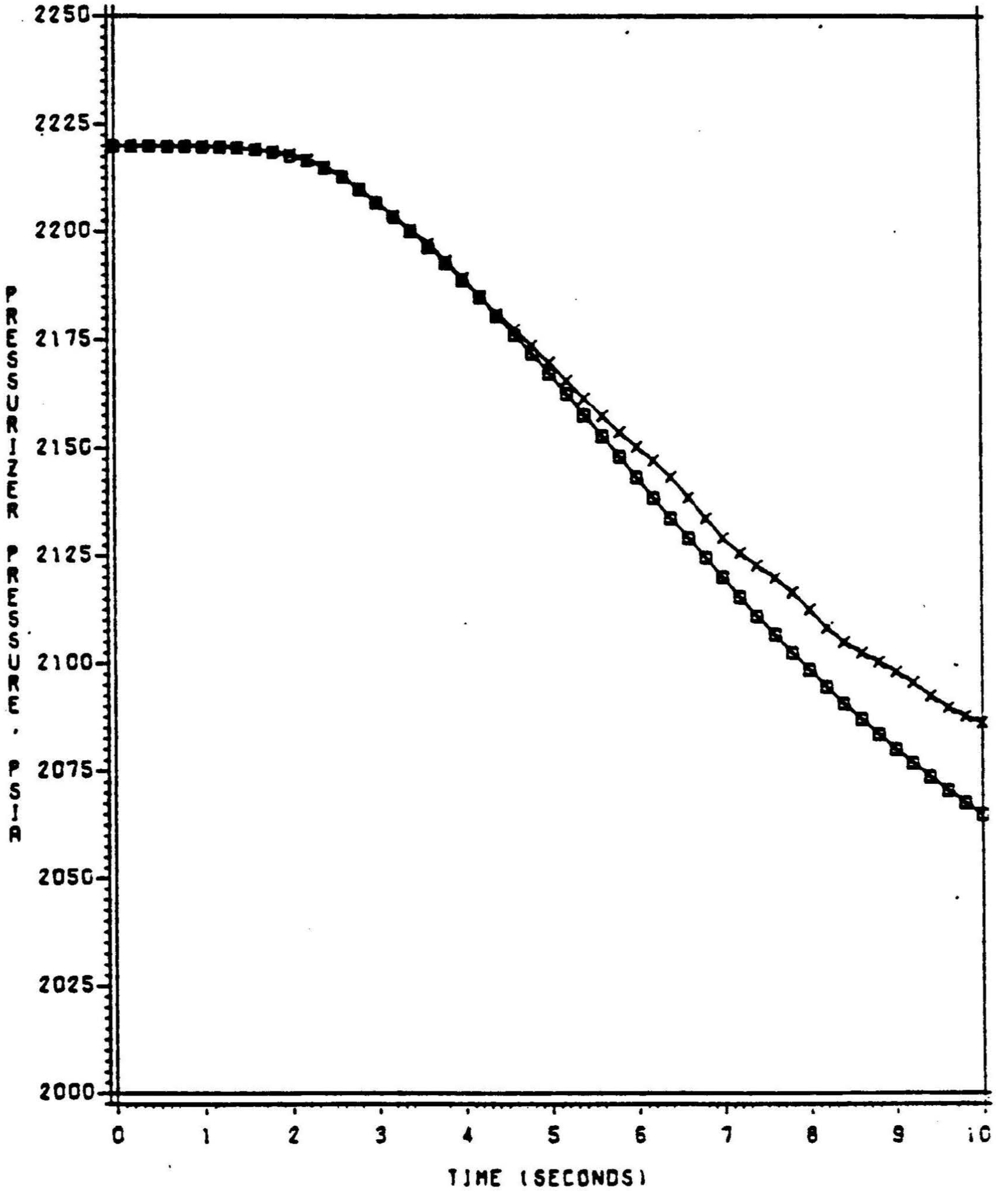
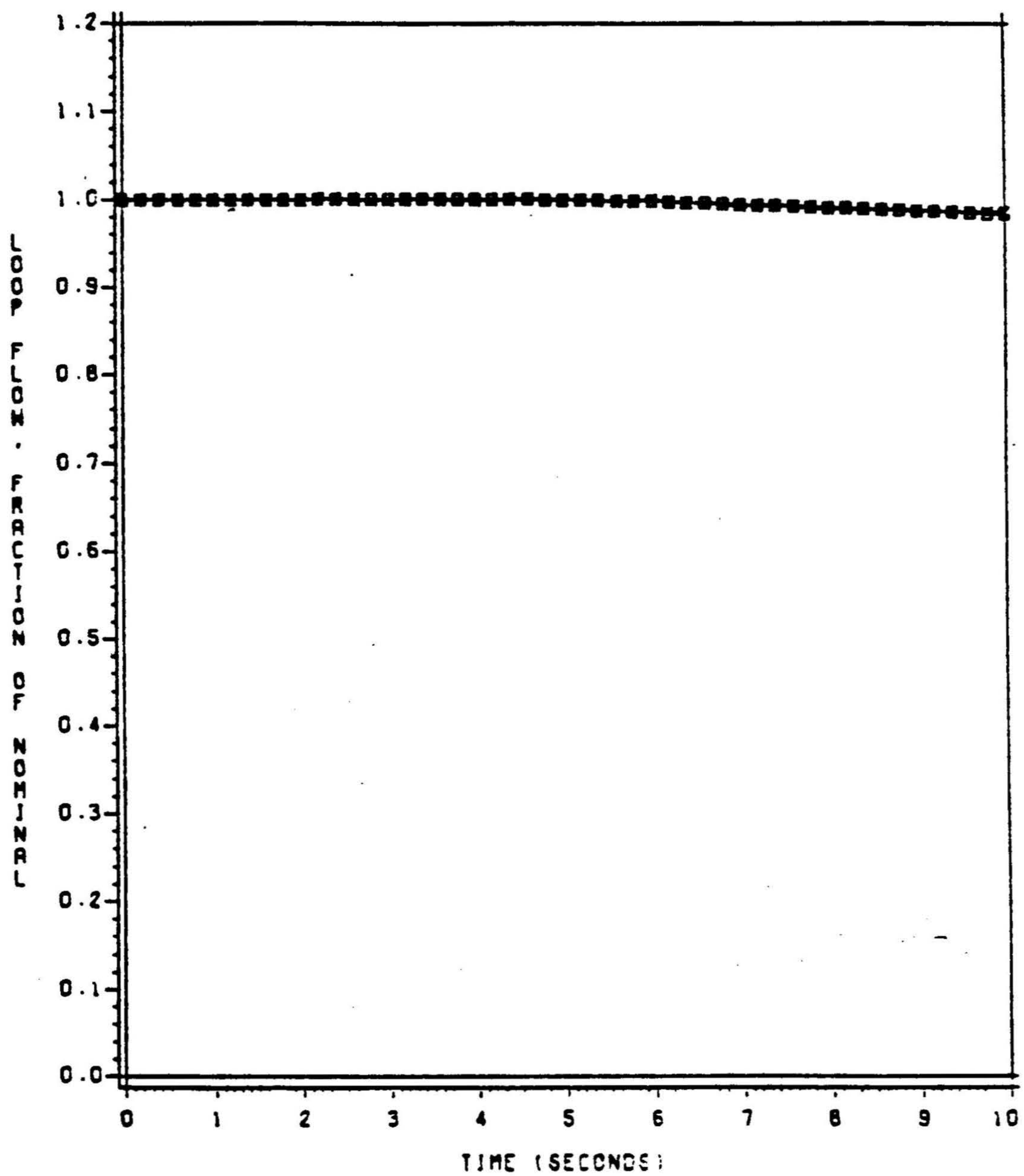


FIGURE 8a
REACTOR TRIP
LOOP FLOW
RETRAN01 VS RETRAN02



RETRAN01=X
RETRAN02=SQUARE

FIGURE 1b
TURBINE TRIP NO RX TRIP
MIDCORE HEAT FLUX
RETRAN01 VS RETRAN02

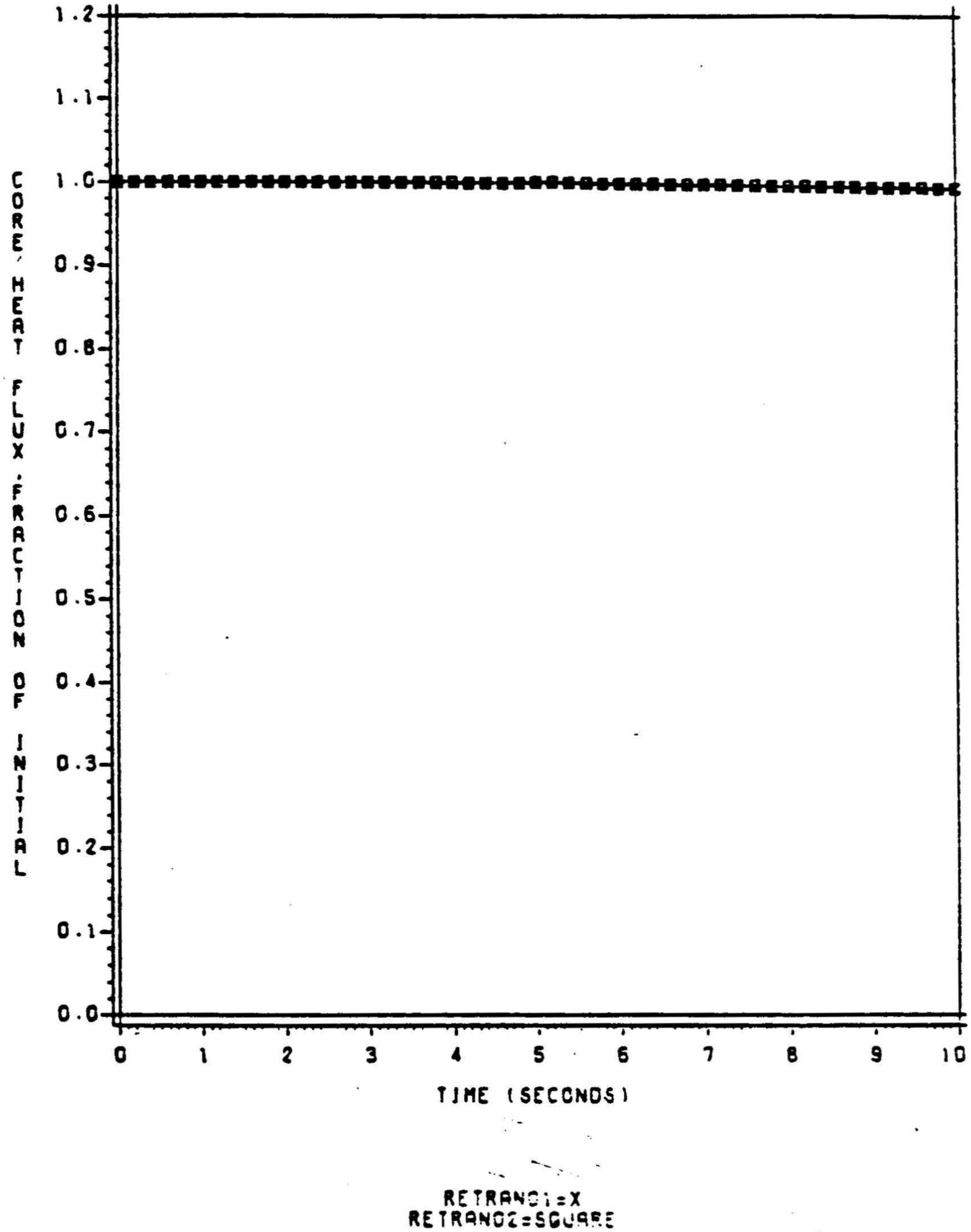


FIGURE 2b
TURBINE TRIP NO RX TRIP
NUCLEAR POWER
RETRAN01 VS RETRAN02

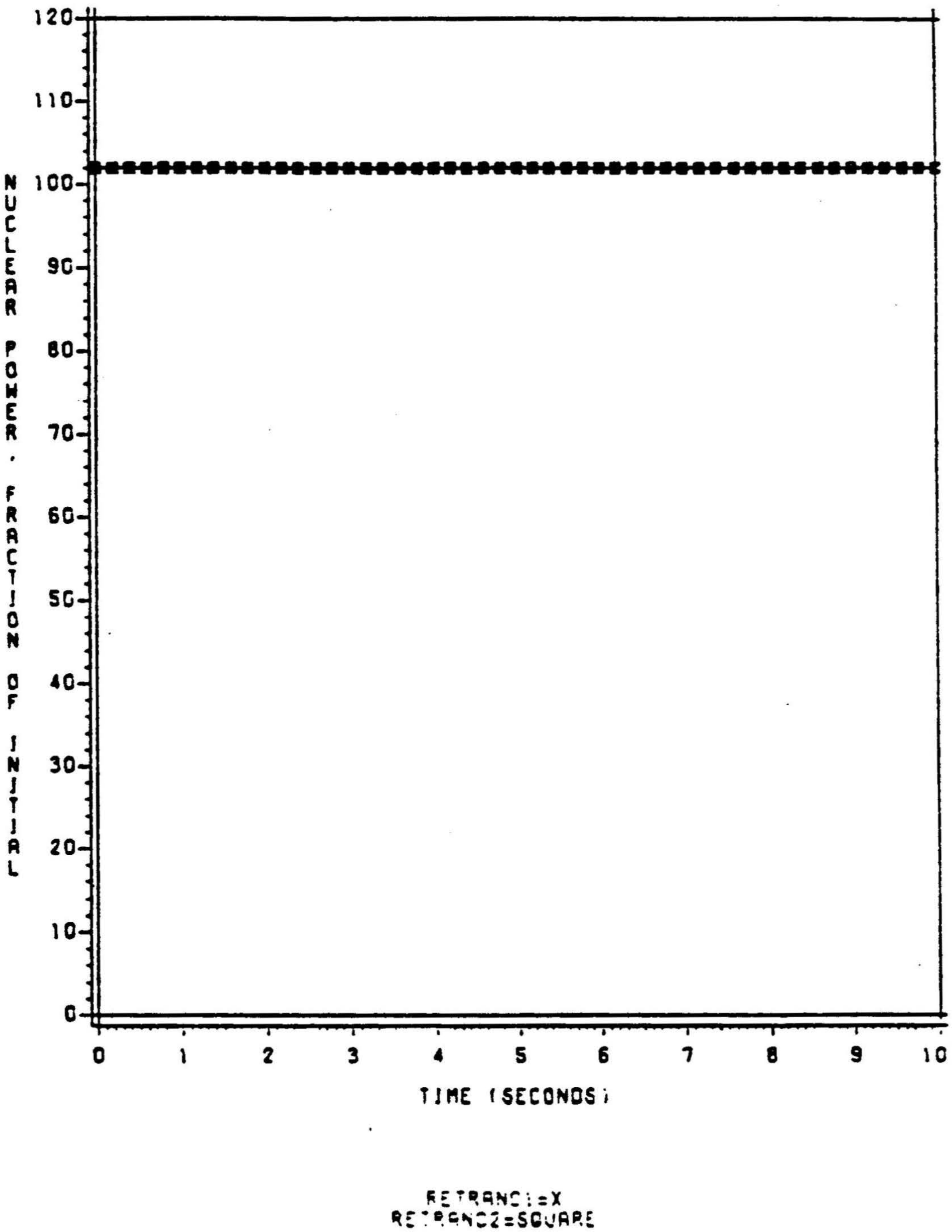


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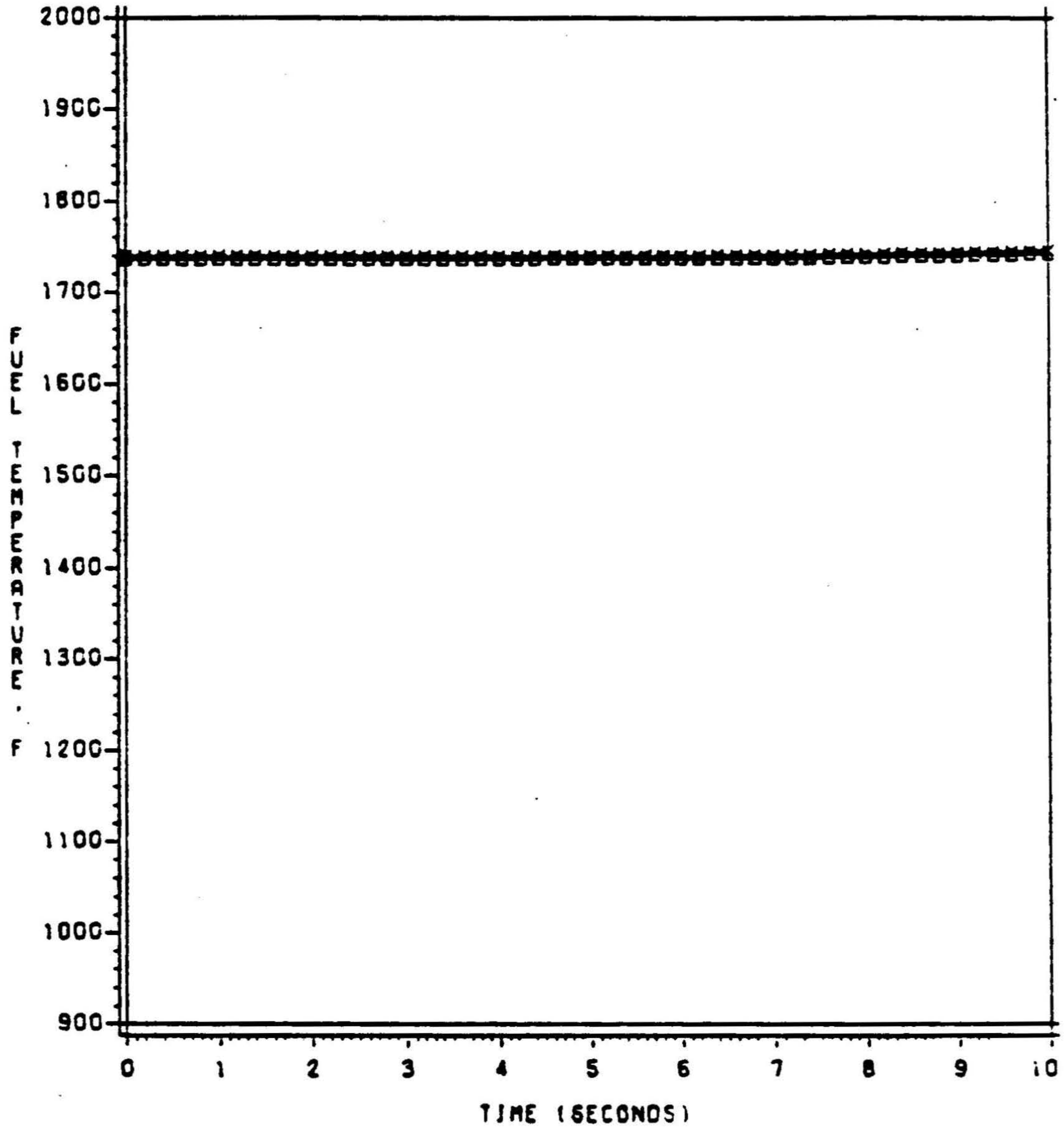
FIGURE 3b
TURBINE TRIP NO RX TRIP
MIDCORE FUEL TEMP
RETRAN01 VS RETRAN02



RETRAN01=X
RETRAN02=SQUARE

2-

FIGURE 4b
TURBINE TRIP NO RX TRIP
SG HEAT EXTRACTION RATE
RETRAN01 VS RETRAN02

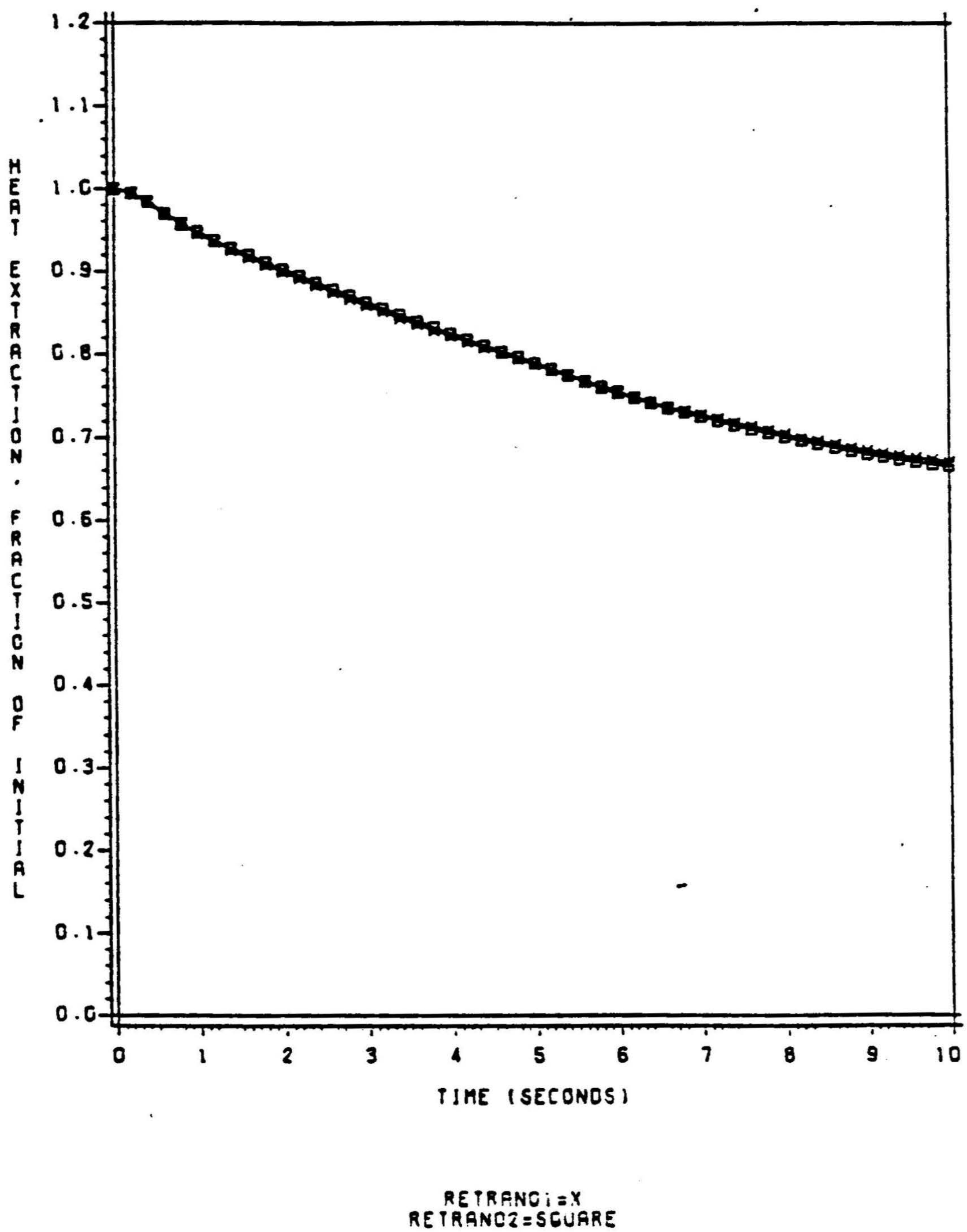


FIGURE 5b

TURBINE TRIP NO RX TRIP

STEAM PRESSURE

RETRAN01 VS RETRAN02

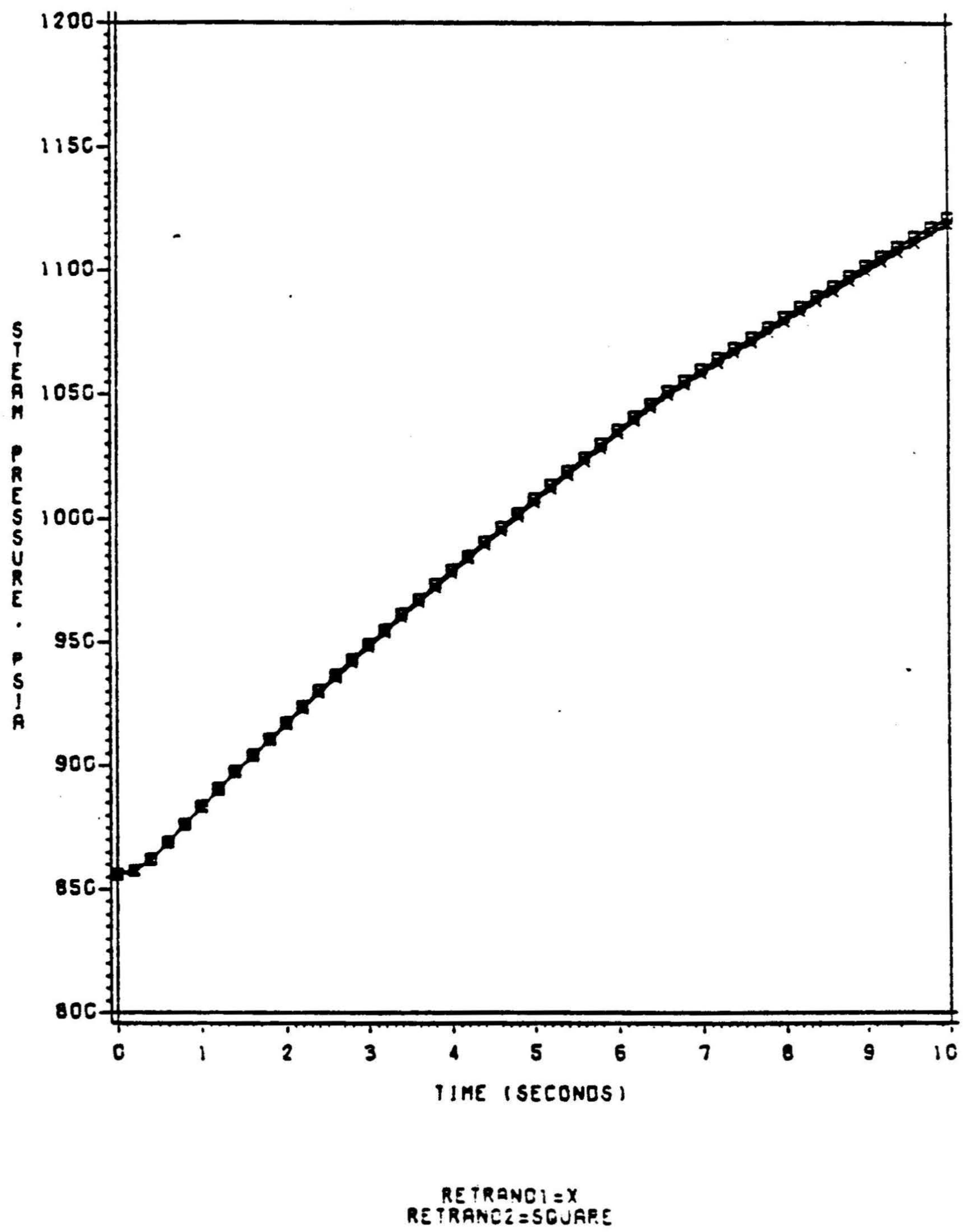


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FIGURE 6b
TURBINE TRIP NO RX TRIP
INLET TEMPERATURE
RETRAN01 VS RETRAN02

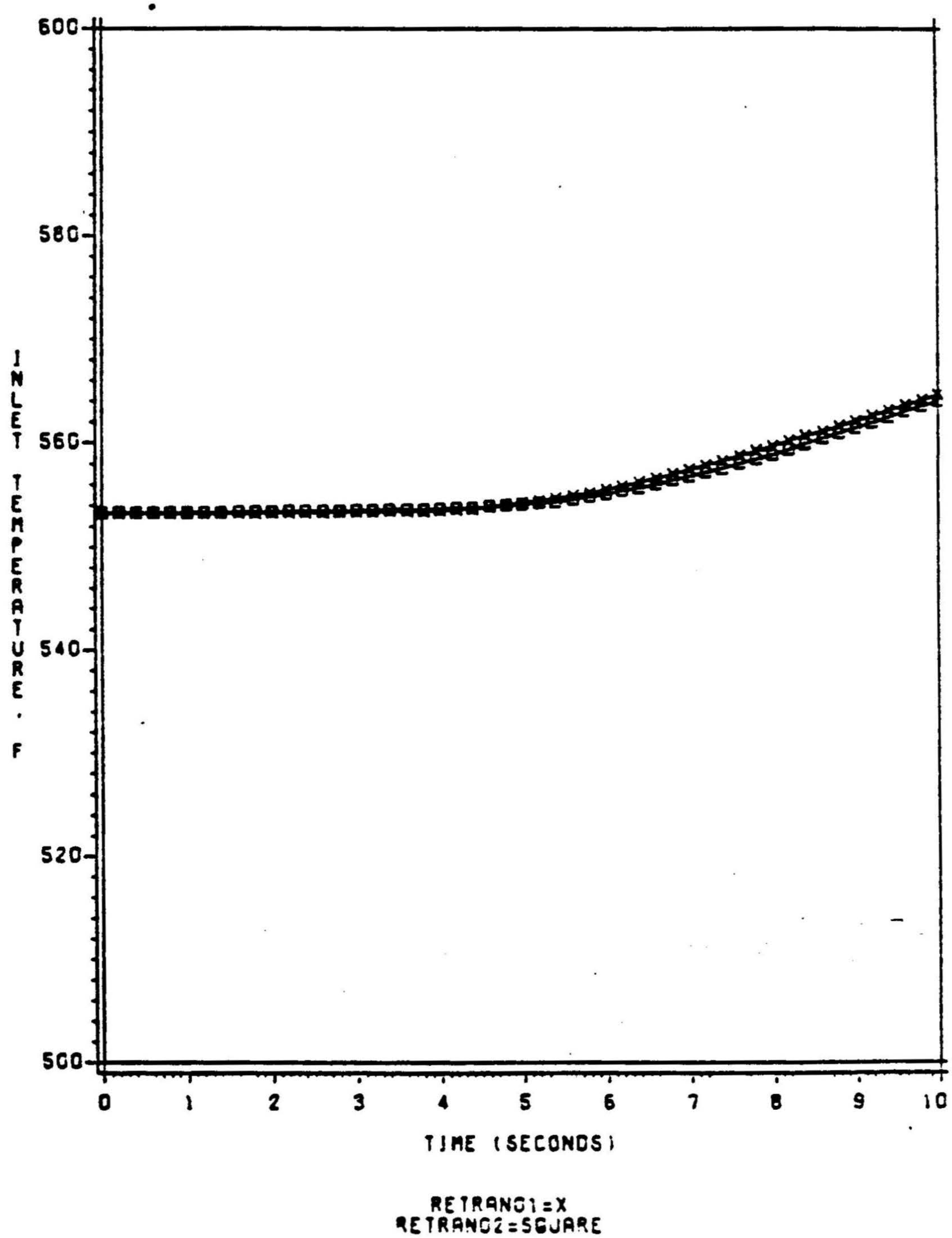

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FIGURE 7b

TURBINE TRIP NO RX TRIP

PRESSURIZER PRESSURE

RETRAN01 VS RETRAN02

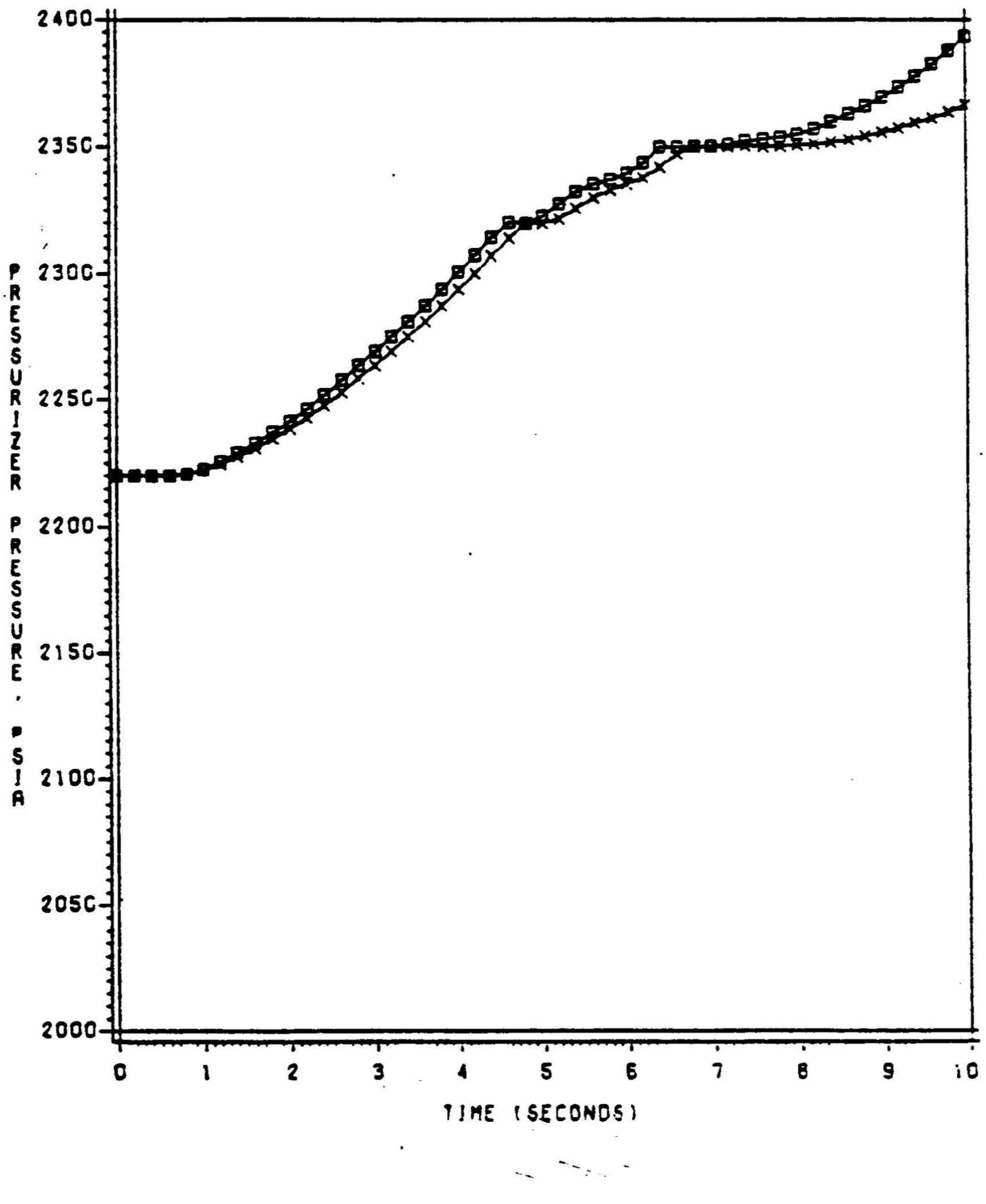
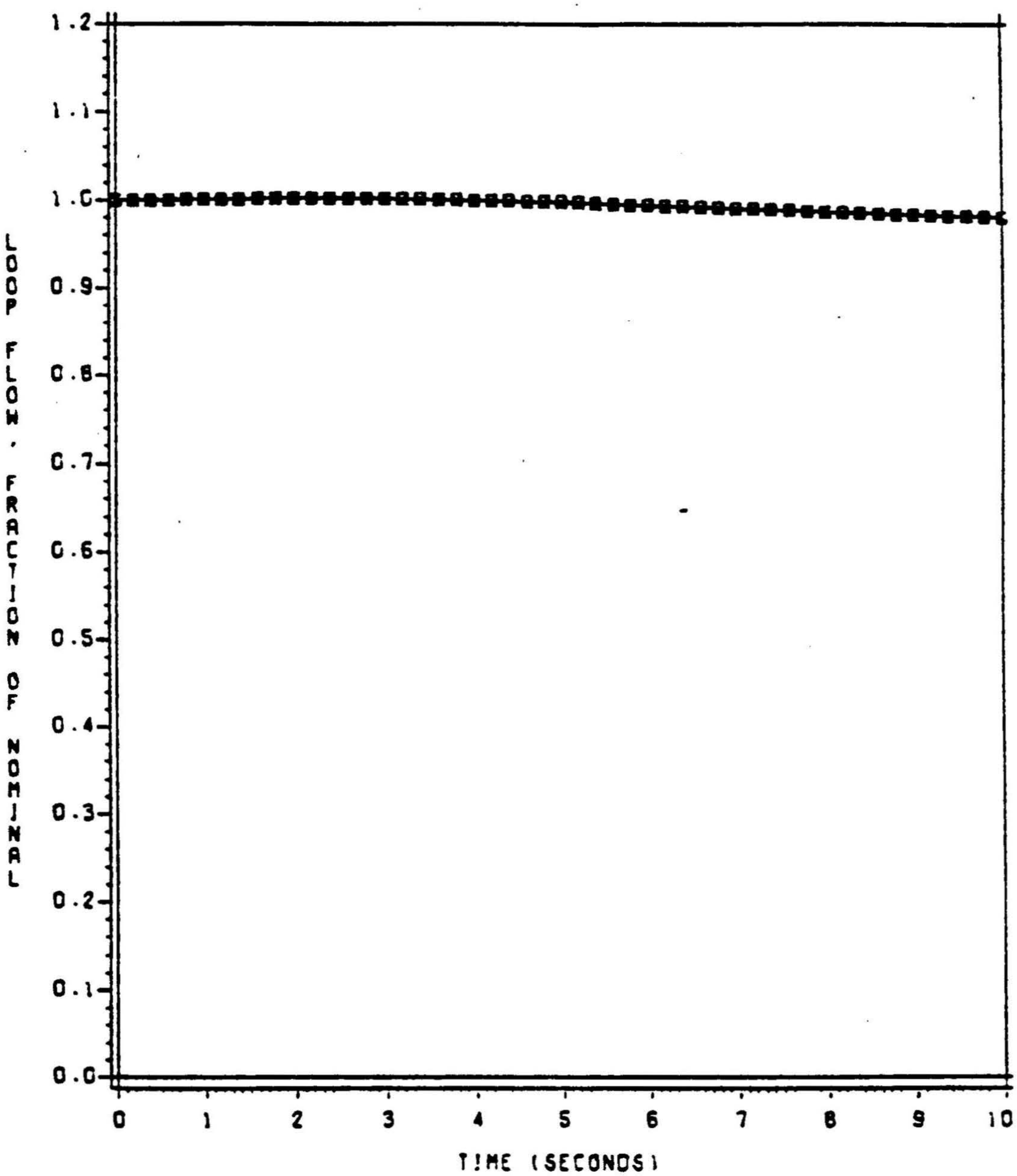


FIGURE 8b

TURBINE TRIP NO RX TRIP

LOOP FLOW

RETRAN01 VS RETRAN02



RETRAN01=X
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FIGURE 1C
LOSS OF FLOW

MIDCORE HEAT FLUX

RETRAN01 VS RETRAN02

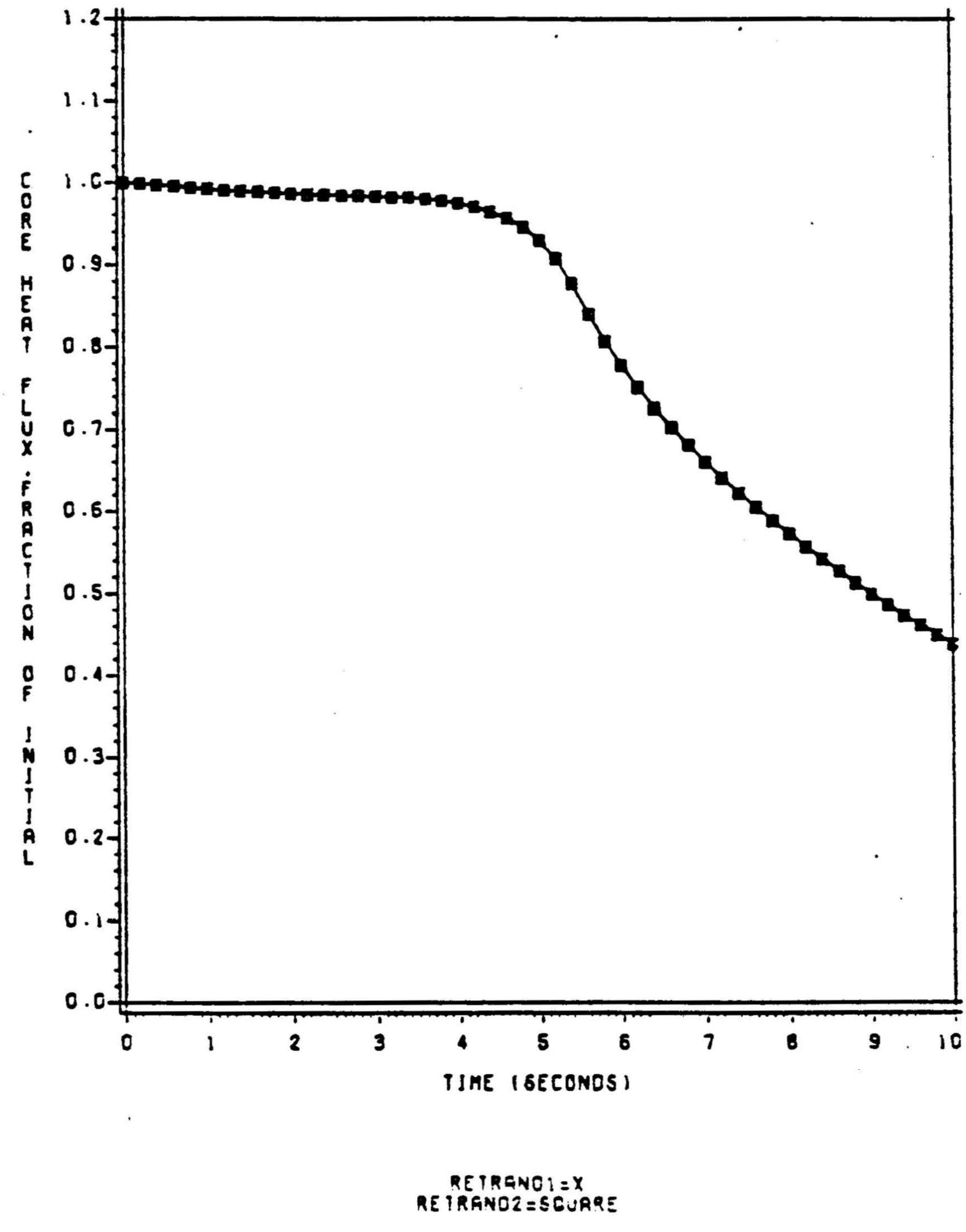
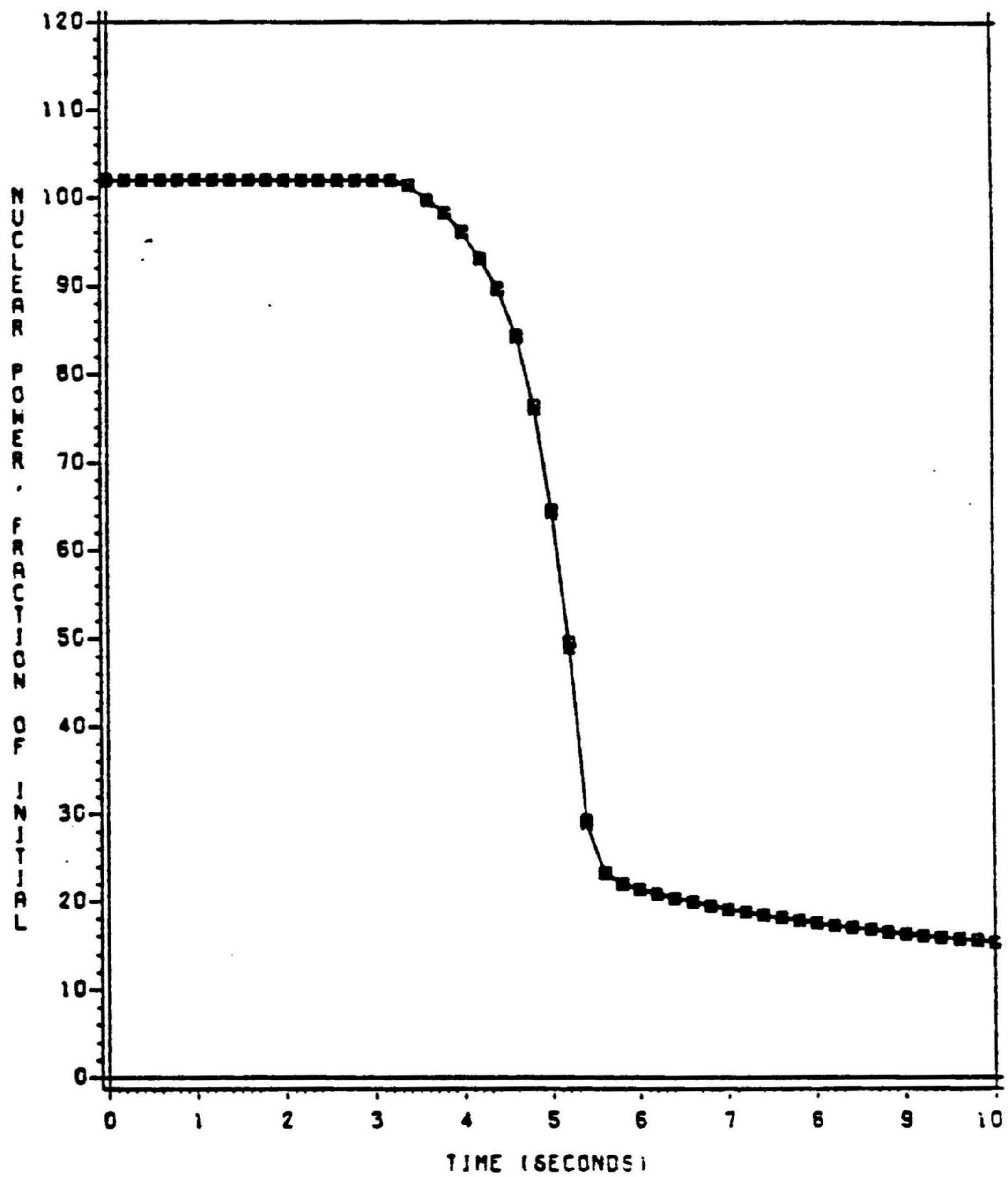


FIGURE 2C
LOSS OF FLOW
NUCLEAR POWER
RETRAN01 VS RETRAN02



RETRAN01=X
RETRAN02=SQUARE

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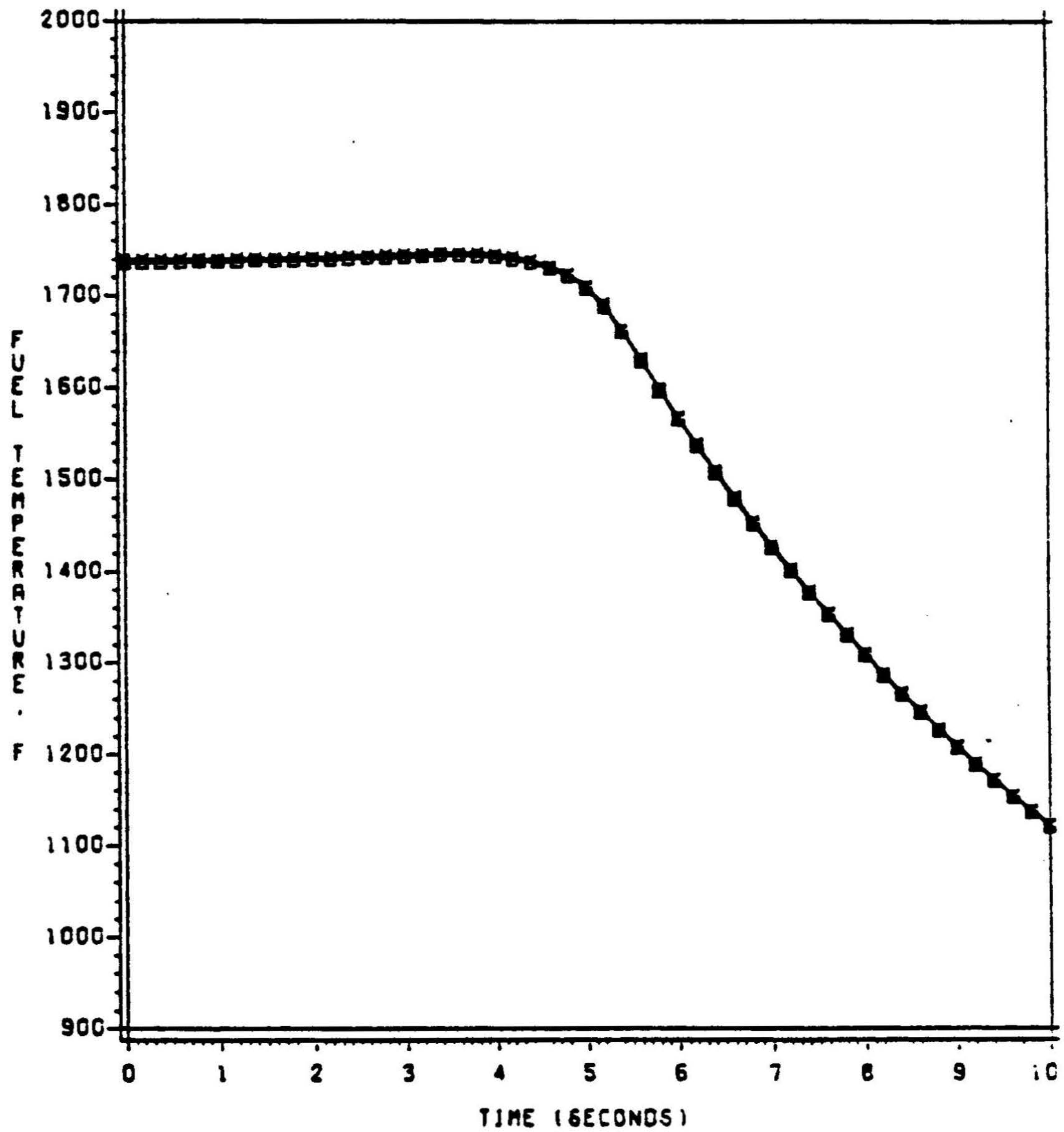
FIGURE 3 C
LOSS OF FLOW
MIDCORE FUEL TEMP
RETRAN01 VS RETRAN02



RETRAN01=X
RETRAN02=SQUARE

FIGURE 4C
LOSS OF FLOW

SG HEAT EXTRACTION RATE

RETRAN01 VS RETRAN02

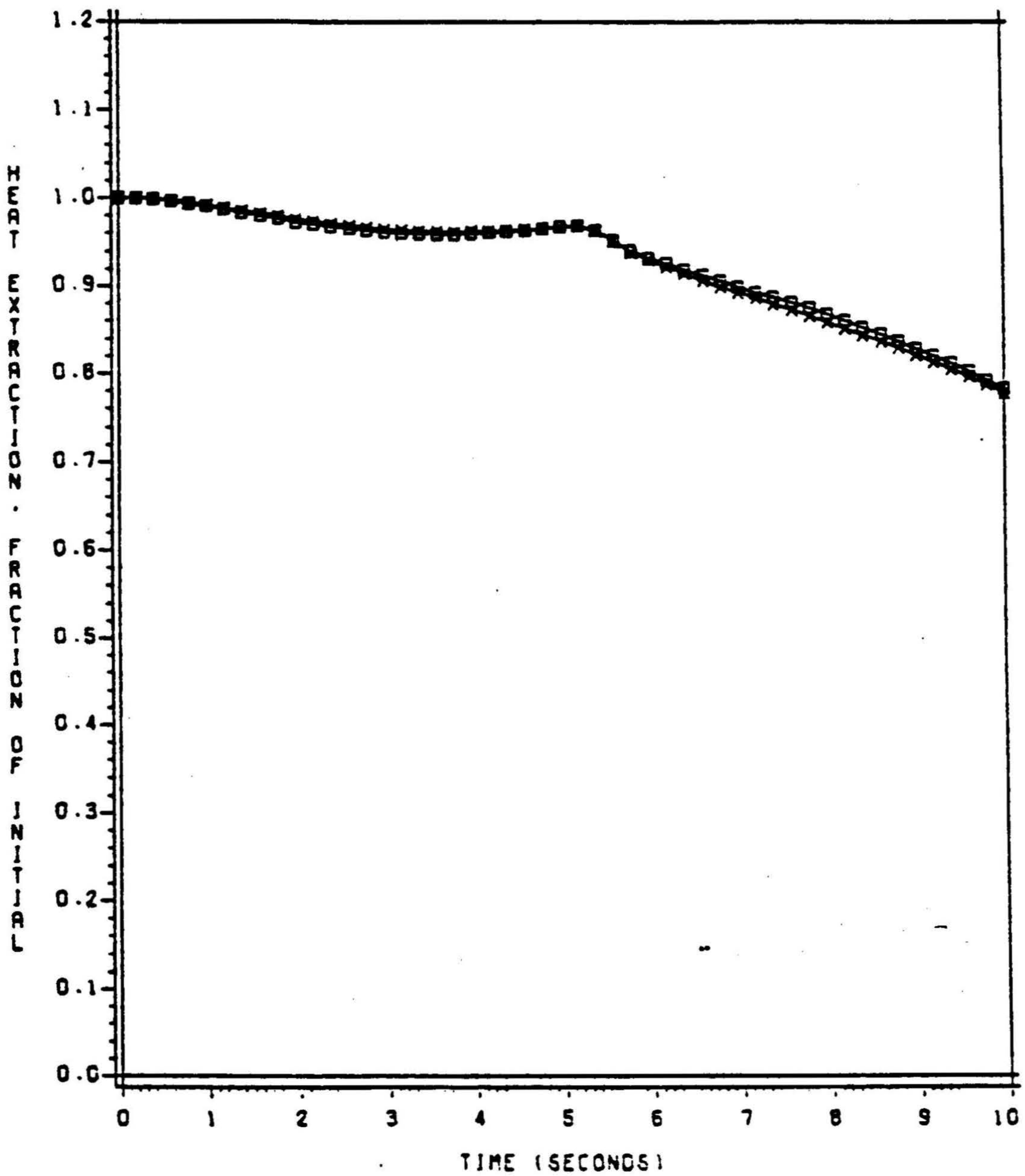


FIGURE 5C
LOSS OF FLOW
STEAM PRESSURE
RETRANO1 VS RETRANO2

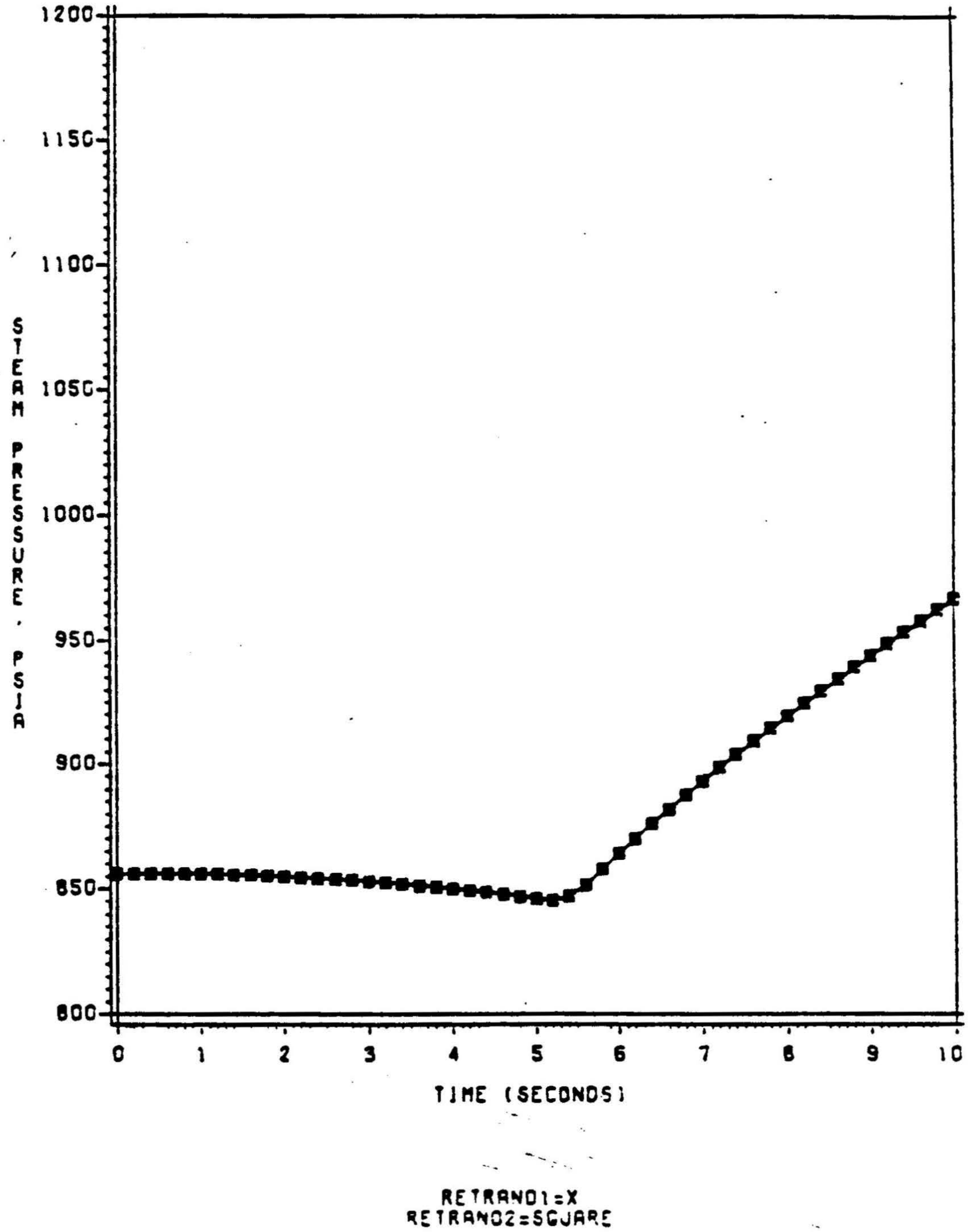


FIGURE 6C
LOSS OF FLOW
INLET TEMPERATURE
RETRAN01 VS RETRAN02

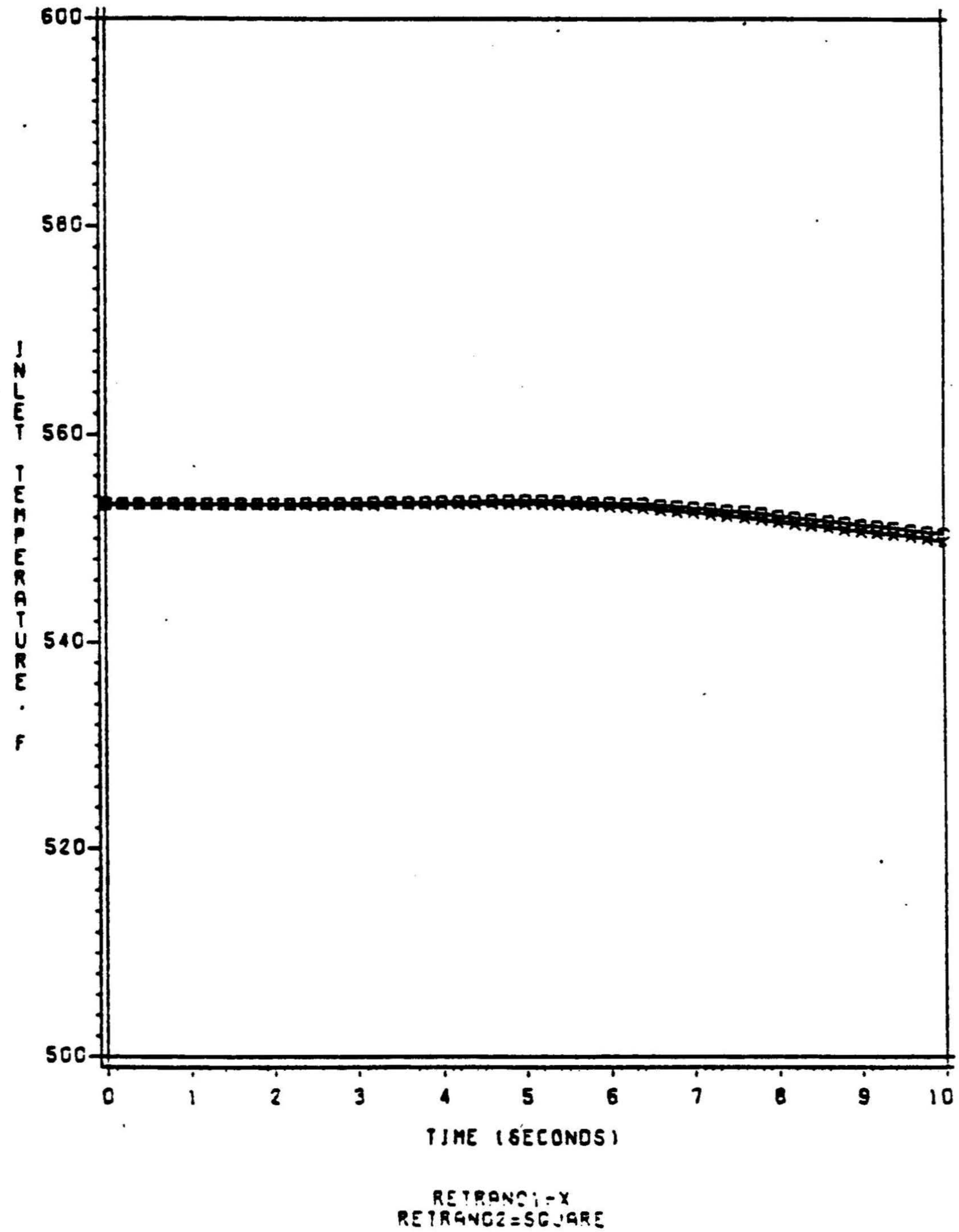


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FIGURE 7C
LOSS OF FLOW
PRESSURIZER PRESSURE
RETRAN01 VS RETRAN02

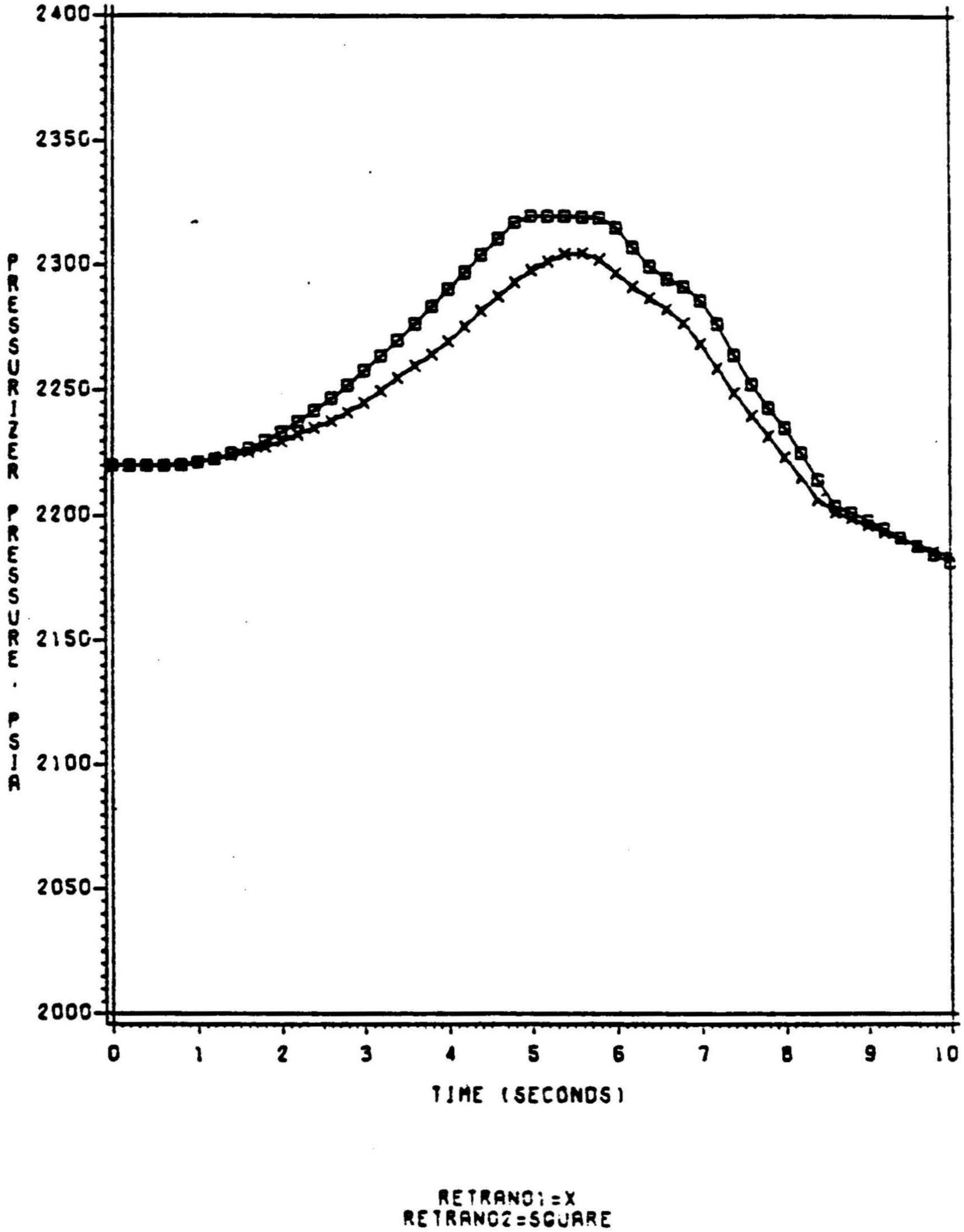
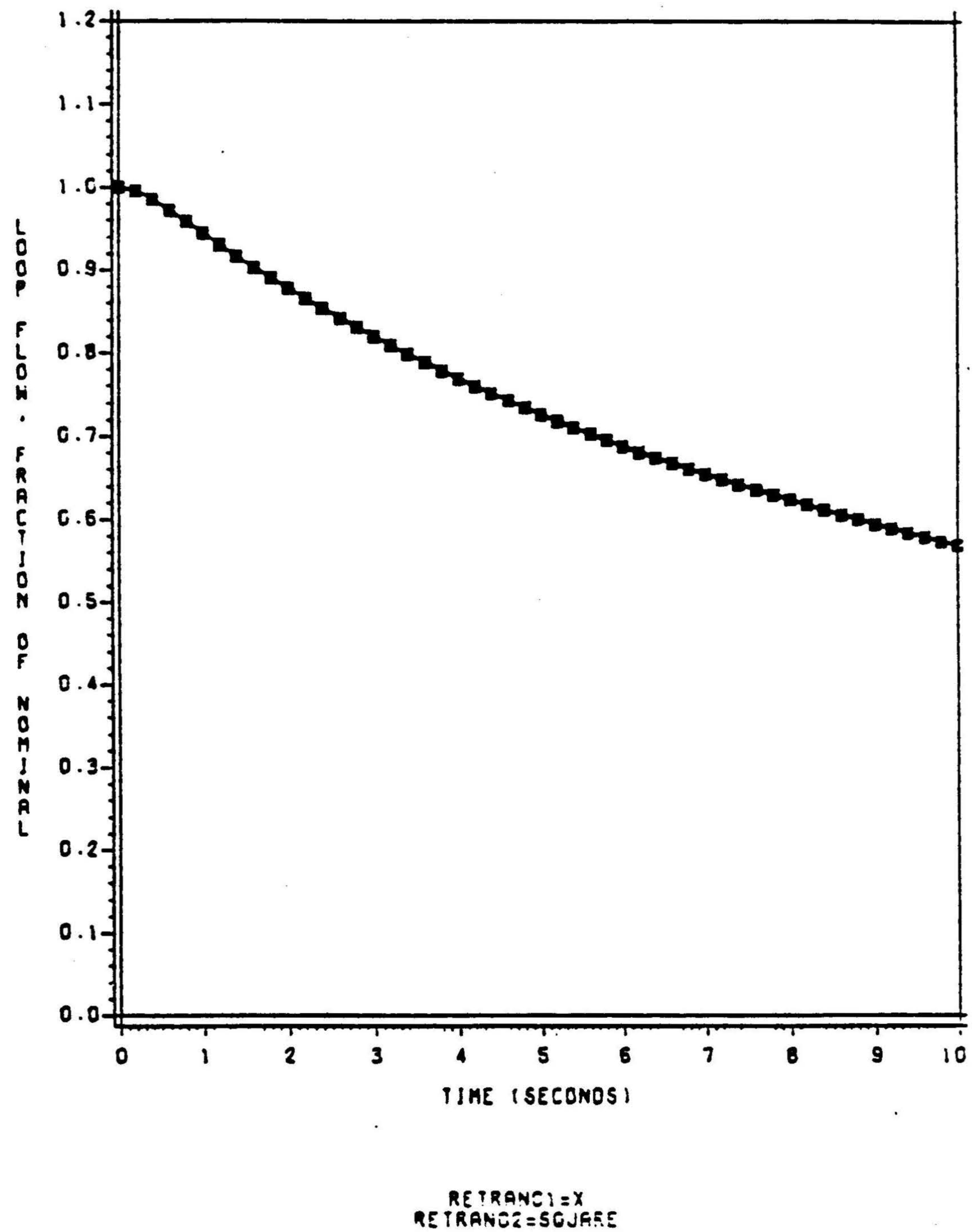


FIGURE 8C
LOSS OF FLOW
LOOP FLOW
RETRAN01 VS RETRAN02



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 M. Moore

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

Mr. David A. Christian
Virginia Electric and Power Company

cc:
Ms. Lillian M. Cuoco, Esq.
Senior Counsel
Dominion Resources Services, Inc.
Millstone Power Station
Building 475, 5th Floor
Rope Ferry Road
Rt. 156
Waterford, Connecticut 06385

Mr. Richard H. Blount, II
Site Vice President
Surry Power Station
Virginia Electric and Power Company
5570 Hog Island Road
Surry, Virginia 23883-0315

Senior Resident Inspector
Surry Power Station
U. S. Nuclear Regulatory Commission
5850 Hog Island Road
Surry, Virginia 23883

Chairman
Board of Supervisors of Surry County
Surry County Courthouse
Surry, Virginia 23683

Dr. W. T. Lough
Virginia State Corporation
Commission
Division of Energy Regulation
P. O. Box 1197
Richmond, Virginia 23209

Robert B. Strobe, M.D., M.P.H.
State Health Commissioner
Office of the Commissioner
Virginia Department of Health
P.O. Box 2448
Richmond, Virginia 23218

Mr. William R. Matthews
Vice President - Nuclear Operations
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, Virginia 23060-6711

Office of the Attorney General
Commonwealth of Virginia
900 East Main Street
Richmond, Virginia 23219

Mr. Chris L. Funderburk, Director
Nuclear Licensing & Operations
Support
Innsbrook Technical Center
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

Mr. David A. Heacock
Site Vice President
North Anna Power Station
Virginia Electric and Power Company
P. O. Box 402
Mineral, Virginia 23117-0402

Mr. C. Lee Lintecum
County Administrator
Louisa County
P.O. Box 160
Louisa, Virginia 23093

Old Dominion Electric Cooperative
4201 Dominion Blvd.
Glen Allen, Virginia 23060

Senior Resident Inspector
North Anna Power Station
U.S. Nuclear Regulatory Commission
1024 Haley Drive
Mineral, Virginia 23117

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.

<u>Core Thermal-Hydraulics and Nuclear Design Models</u>

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

TOTAL P.16

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
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
NRC Senior Resident Inspector
North Anna Power Station

Mr. J. E. Reasor, Jr.
Old Dominion Electric Cooperative
Innsbrook Corporate Center, Suite 300
4201 Dominion Blvd.
Glen Allen, Virginia 23060

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:

PDQ Two-Zone Model

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.

<u>Conservative vs. Nonconservative Results</u>

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.

<u>Essentially the Same</u>

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/MOD03 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 $F\Delta H$) 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
NRC Senior Resident Inspector
North Anna Power Station

Mr. J. E. Reasor, Jr.
Old Dominion Electric Cooperative
Innsbrook Corporate Center, Suite 300
4201 Dominion Blvd.
Glen Allen, Virginia 23060

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.

<u>BIT to RCS Transport</u>
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.

<u>RCS Boron Transport</u>
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.

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

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-O2/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.

DOMINION RESPONSE TO QUESTION 1b

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 Viné (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 $F_{\Delta H}$ (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

PREDICTED / MEASURED / PCT DIFFERENCE

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.246	0.275	0.242						
							0.249	0.280	0.246						
							1.0	1.8	1.6						
2					0.331	0.649	1.053	0.836	1.045	0.647	0.331				
					0.336	0.656	1.061	0.844	1.062	0.674	0.338				
					1.5	1.0	0.8	1.0	1.6	4.3	2.2				
3				0.365	1.179	1.299	1.170	1.301	1.168	1.299	1.183	0.368			
				0.381	1.200	1.313	1.177	1.301	1.175	1.312	1.196	0.379			
				4.3	1.8	1.1	0.6	0.0	0.6	1.0	1.1	3.0			
4			0.368	0.898	1.333	1.323	1.305	1.243	1.304	1.324	1.338	0.897	0.365		
			0.381	0.919	1.358	1.335	1.309	1.244	1.303	1.318	1.343	0.908	0.369		
			3.6	2.3	1.9	0.9	0.3	0.0	-0.1	-0.4	0.4	1.2	1.1		
5		0.356	1.204	1.346	1.221	1.262	1.168	1.236	1.168	1.261	1.220	1.339	1.197	0.355	
		0.367	1.259	1.370	1.228	1.265	1.168	1.233	1.164	1.258	1.222	1.358	1.212	0.361	
		3.1	4.6	1.8	0.6	0.2	0.0	-0.2	-0.3	-0.3	0.1	1.4	1.2	1.9	
6		0.665	1.321	1.334	1.266	1.017	1.145	1.071	1.143	1.017	1.265	1.332	1.319	0.665	
		0.668	1.334	1.338	1.252	1.012	1.139	1.065	1.133	1.012	1.261	1.333	1.330	0.677	
		0.5	0.9	0.3	-1.1	-0.5	-0.6	-0.5	-0.8	-0.4	-0.3	0.1	0.8	1.9	
7	0.258	1.070	1.186	1.317	1.174	1.145	1.038	1.005	1.038	1.148	1.174	1.316	1.187	1.074	0.259
	0.256	1.060	1.166	1.306	1.164	1.135	1.027	0.993	1.018	1.135	1.163	1.294	1.195	1.108	0.267
	-1.0	-0.9	-1.7	-0.8	-0.9	-0.9	-1.1	-1.2	-1.9	-1.1	-0.9	-1.7	0.7	3.2	3.1
8	0.284	0.853	1.320	1.255	1.243	1.074	1.005	0.996	1.005	1.074	1.243	1.255	1.320	0.853	0.284
	0.282	0.845	1.298	1.244	1.232	1.065	0.994	0.984	0.991	1.063	1.233	1.251	1.333	0.879	0.293
	-1.0	-1.0	-1.7	-0.9	-0.9	-0.9	-1.1	-1.2	-1.4	-1.0	-0.7	-0.3	1.0	3.0	2.9
9	0.259	1.074	1.187	1.316	1.174	1.148	1.038	1.005	1.038	1.145	1.174	1.317	1.186	1.070	0.258
	0.258	1.069	1.181	1.310	1.172	1.139	1.025	0.991	1.024	1.133	1.167	1.319	1.201	1.097	0.264
	-0.6	-0.5	-0.5	-0.5	-0.2	-0.8	-1.2	-1.4	-1.3	-1.1	-0.6	0.2	1.3	2.5	2.4
10		0.665	1.319	1.332	1.265	1.017	1.143	1.071	1.145	1.017	1.266	1.334	1.321	0.665	
		0.665	1.326	1.330	1.259	1.007	1.123	1.053	1.127	1.003	1.255	1.339	1.344	0.691	
		0.1	0.6	-0.1	-0.5	-0.9	-1.7	-1.7	-1.6	-1.4	-0.9	0.4	1.8	3.8	
11		0.355	1.197	1.339	1.220	1.261	1.168	1.236	1.168	1.262	1.221	1.346	1.204	0.356	
		0.356	1.203	1.337	1.211	1.246	1.147	1.211	1.146	1.229	1.198	1.361	1.235	0.366	
		0.3	0.5	-0.1	-0.7	-1.2	-1.8	-2.0	-1.9	-2.6	-1.9	1.1	2.6	3.0	
12			0.365	0.897	1.338	1.324	1.304	1.243	1.305	1.323	1.333	0.898	0.368		
			0.373	0.897	1.329	1.307	1.276	1.218	1.286	1.309	1.334	0.939	0.407		
			2.4	-0.1	-0.7	-1.3	-2.2	-2.0	-1.5	-1.1	0.1	4.5	10.6		
13				0.368	1.183	1.299	1.168	1.301	1.170	1.299	1.179	0.365			
				0.367	1.179	1.286	1.149	1.270	1.163	1.292	1.180	0.372			
				-0.3	-0.3	-1.0	-1.6	-2.4	-0.7	-0.5	0.1	1.9			
14					0.331	0.647	1.045	0.836	1.053	0.649	0.331				
					0.342	0.644	1.037	0.834	1.077	0.653	0.332				
					3.4	-0.4	-0.7	-0.1	2.3	0.6	0.2				
15							0.242	0.275	0.246						
							0.243	0.276	0.251						
							0.3	0.2	1.8						

STANDARD DEVIATION =1.235

AVERAGE PCT DIFFERENCE = 1.3

MAP NO: N1-16-01

DATE: 10/10/01

CONTROL ROD POSITIONS:

D BANK AT 150 STEPS

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

PDQ 2 of 13

NORTH ANNA UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
74% POWER

PREDICTED / MEASURED / PCT DIFFERENCE

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1							0.271 0.272 0.4	0.311 0.311 0.1	0.267 0.269 0.9							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							0.267 0.263 -1.4	0.311 0.310 -0.4	0.271 0.273 0.9							15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

STANDARD DEVIATION =0.694

AVERAGE PCT DIFFERENCE = 0.8

SUMMARY

MAP NO: N1-16-02 DATE: 10/11/01 POWER: 74%

CONTROL ROD POSITIONS: F-Q(Z) = 1.848 CORE TILT:

D BANK AT 192 STEPS F-DH(N) = 1.451

F(Z) = 1.184

BURNUP = 24.0 MWD/MTU A.O. = 0.088

NW 1.0014	NE 1.0039
SW 0.9933	SE 1.0014

PDQ 3 of 13

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	

SUMMARY

MAP NO: N1-16-03 DATE: 10/23/01 POWER: 100%

CONTROL ROD POSITIONS: F-Q(Z) = 1.786 CORE TILT:

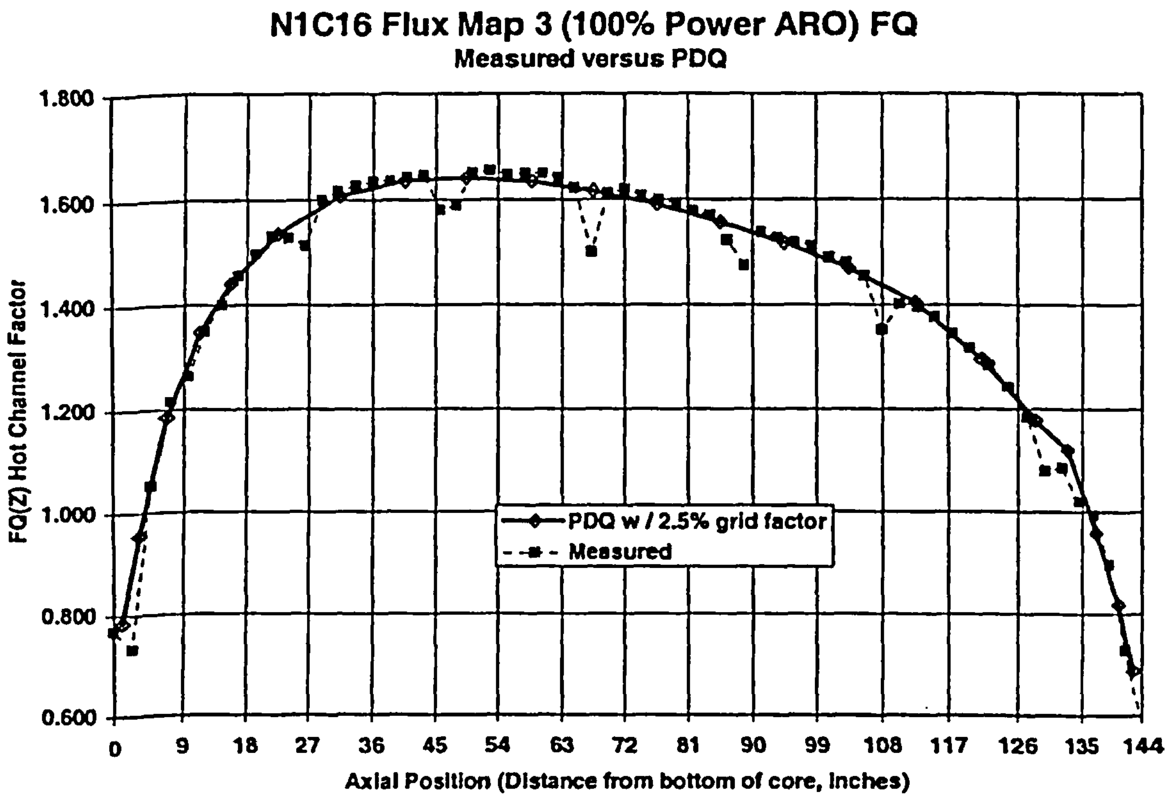
D BANK AT 226 STEPS F-DH(N) = 1.405

F(Z) = 1.139

NW 0.9982	NE 1.0035
SW 0.9954	SE 1.0029

BURNUP = 436.4 MWD/MTU A.O. = -2.537

PDQ 4 of 13



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 n^{th} 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.
$F\Delta H$	Statistical	Statistics use comparisons to measured $F\Delta H$ 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
$F\Delta H$ (North Anna)	1479	0.1%	1.9%	No	N/A	60
$F\Delta H$ (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

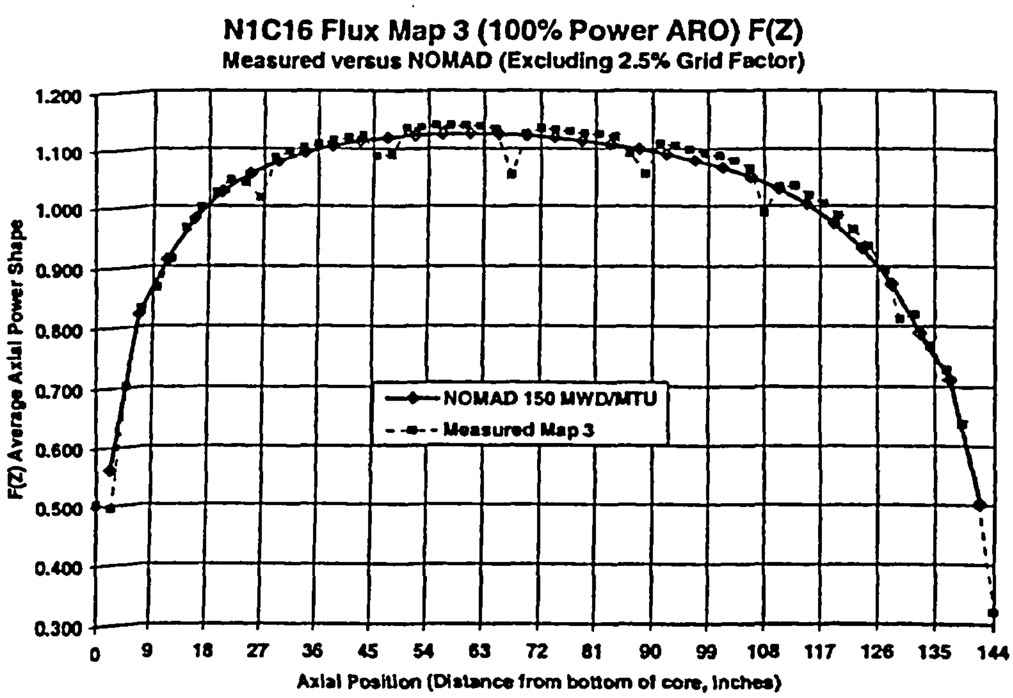
By letter dated December 2, 2002, VEPCO stated that the accuracy of the NOMAD model is verified each cycle during startup physics testing and during routine core follow. Please provide representative results from a recent refueling outage (comparisons between the startup physics test data and the NOMAD 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.

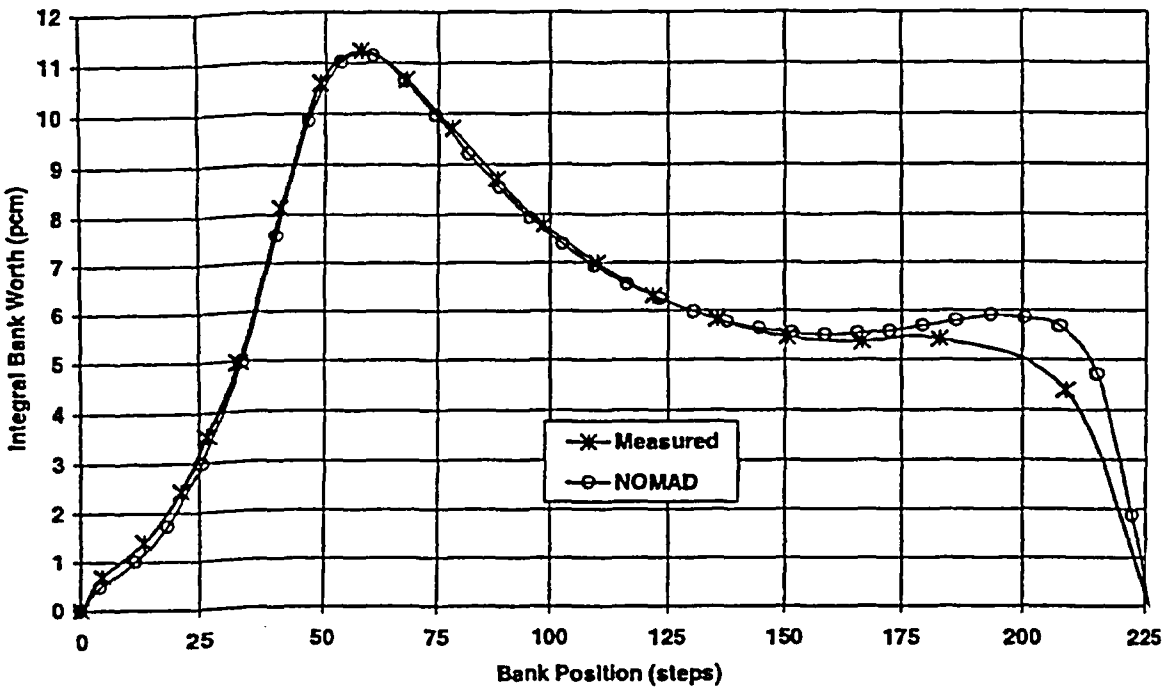
N1C16 STARTUP PHYSICS TESTING RESULTS (October, 2001)

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

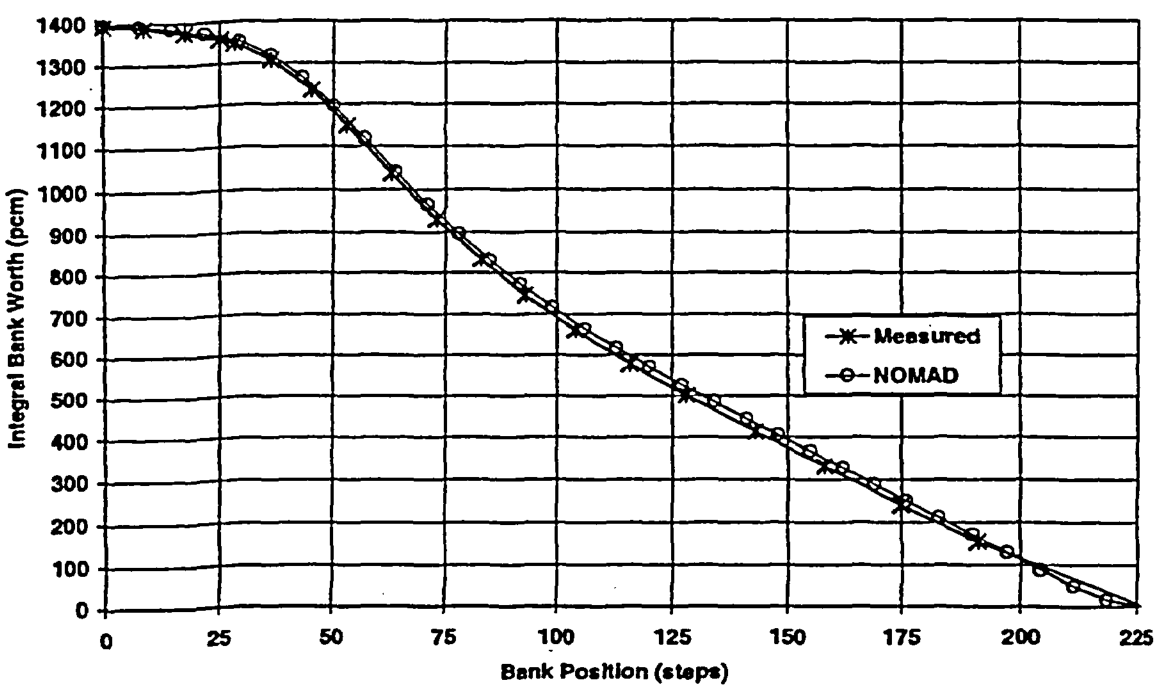


NOMAD 1 of 12

N1C16 HZP BOC B-Bank Differential Worth



N1C16 HZP BOC B-Bank Integral Worth



NOMAD 2 of 12

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.

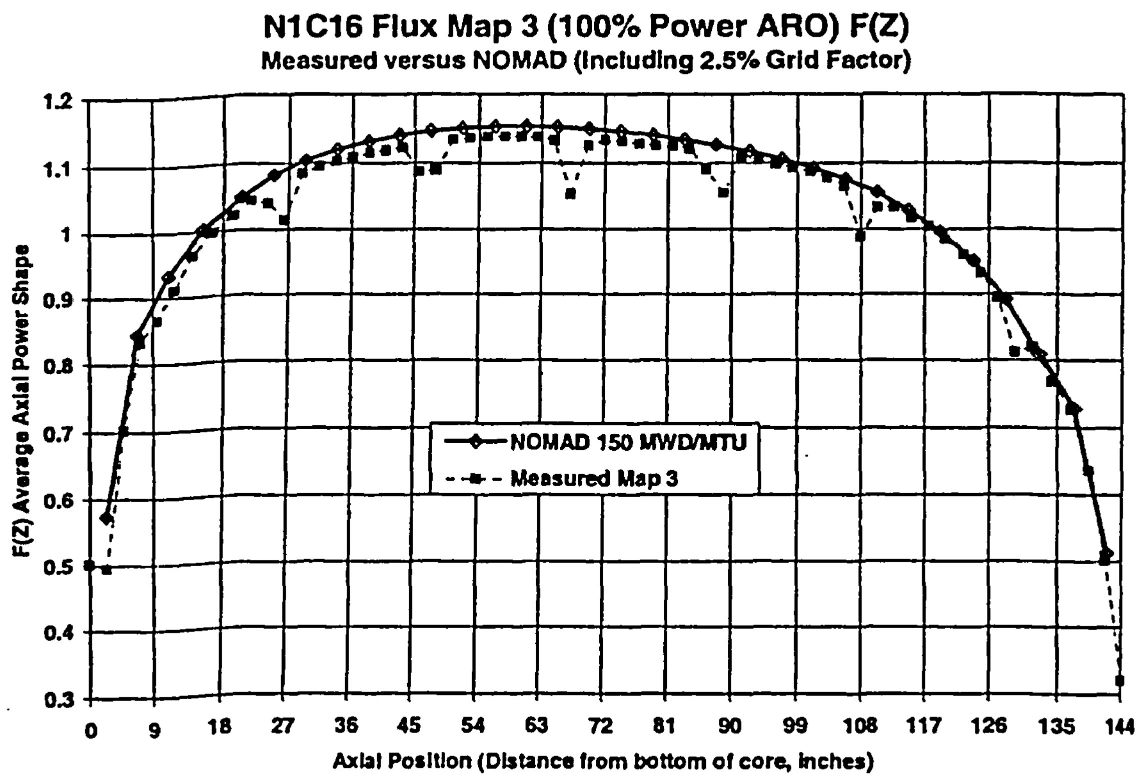
The final element of the control rod model is the ability to normalize bank worth to the PDQ Two Zone value. Although NOMAD was designed to produce acceptable control rod worth results without normalizing to PDQ, normalization is performed routinely for many design calculations to eliminate any difference 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 (F_z 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

NOMAD 11 of 12

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.

APPENDIX 8
RETRAN-3D Generic Safety Evaluation Report

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 25, 2001

Mr. Gary L. Vine
Senior Washington Representative
Electric Power Research Institute
2000 L Street, NW., Suite 205
Washington, DC 20036

SUBJECT: 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" (TAC NO. MA4311)

Dear Mr. Vine:

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the Electric Power Research Institute (EPRI) Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," for analysis of Standard Format Chapter 15 accidents and transients. The report describes modifications to the approved RETRAN-02 analysis code which include the addition of three-dimensional kinetics capability and other changes to the thermal-hydraulic modeling capability.

The staff previously reviewed and accepted earlier versions of this analysis methodology, subject to several conditions and limitations on their use. The review of the new version has found the proposed changes to be acceptable, subject to the conditions and limitations on its use described in the enclosed safety evaluation, that you accepted in your December 13, 2000, letter. Please note that even with this generic approval of the new version, the responsibility for assessment of the code and the new modeling changes continues to rest with the individual user, and approval of all future applications of this code will require the formal submittal of detailed assessment documentation by the code user.

The staff finds that the subject topical report is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation. The safety evaluation, which is enclosed, defines the basis for acceptance of the topical report.

The staff will not repeat its review of the matters described in the subject report, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. In accordance with the procedures established in NUREG-0390, the NRC requests that EPRI publish an accepted version of the report within 3 months of receipt of this letter. The accepted version shall incorporate (1) this letter and the enclosed safety evaluation between the title page and the abstract, (2) all requests for additional information from the staff and all associated responses, and (3) an "-A" (designating "accepted") following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions about the acceptability of the report are invalidated, EPRI or the applicant referencing the report, or both, will be expected to revise and resubmit its respective documentation, or submit justification for the continued effective applicability of the report without revision of the respective documentation.

Pursuant to 10 CFR 2.790, we have determined that the enclosed safety evaluation does not contain proprietary information. However, we will delay placing the safety evaluation in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

If you have any further questions regarding this review, please contact Leonard Olshan at (301) 415-1419.

Sincerely, /RA by Stephen Dembek for/

Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 669

Enclosure: Safety Evaluation

cc w/encl: See next page

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Sincerely,

FOR

Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 669

Enclosure: Safety Evaluation

cc w/encl: See next page

cc:
Mr. James Lang
Director
EPRI
1300 W.T. Harris Boulevard
Charlotte, NC 28262

Dr. Theodore U. Marston
Vice President and Chief Nuclear Officer
EPRI
3412 Hillsview Avenue
Palo Alto, CA 94304

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO EPRI NP-7450(P), "RETRAN-3D - A PROGRAM FOR TRANSIENT THERMAL-HYDRAULIC ANALYSIS OF COMPLEX FLUID FLOW SYSTEMS"

ELECTRIC POWER RESEARCH INSTITUTE

PROJECT NO. 669

1.0 INTRODUCTION

RETRAN-3D is a flexible general purpose, thermal/hydraulic computer code that is used to evaluate the effects of various upset reactor conditions in the reactor coolant system (RCS). The code models the reactor coolant as a single phase or as two equilibrium phases with the exception that a non-equilibrium pressurizer component can be included. Conductive heat structures can be described, including the fuel elements in the reactor core. Changes in reactor power from neutron kinetics and decay heat considerations are calculated to occur with time. The name, RETRAN-3D, refers to the ability of the code to perform three-dimensional neutronic calculations in the core, but not three-dimensional fluid dynamic capability.

RETRAN-3D was developed by CSA, Inc, for the Electric Power Research Institute (EPRI) based on the RETRAN-02 computer code. RETRAN-02 was based on the RELAP4 thermal/hydraulic computer code developed by the NRC. The original code version, RETRAN-01, was released by EPRI in December 1978. The code was subsequently improved to account for slip between the phases, two-phase natural convection heat transfer, improved numerics, and other improvements. The NRC staff completed its review of RETRAN-02 MOD002 and RETRAN-02 MOD003 as described in Reference 1. The countercurrent flow logic and the slip flow modeling were modified and a new heat slab model was added to the non-equilibrium pressurizer in RETRAN-02 MOD003. A new control rod model was added as an option to produce RETRAN-02 MOD004. These modifications were also approved by the NRC staff in Reference 2. The 1979 ANS5.1 decay heat model was added to the code as RETRAN-02 MOD005. This version was approved by the NRC staff in Reference 3. The last version of the revised code as described in Reference 4, was submitted to the NRC for review as RETRAN-3D MOD3.

The staff's approval of RETRAN-02 was subject to a number of limitations described in the safety evaluations (SE) for the various RETRAN-02 versions and in the technical evaluation reports (TERs) prepared by the NRC staff's contractors. Those limitations have been reviewed in the process of preparing this SE. Because of the large amount of flexibility in the user-supplied input selection and choice of nodalization schemes, the NRC staff has required, and will continue to require, that proposed applications of RETRAN-3D be accompanied by a detailed review of the suitability of the code for each specific application.

2.0 STAFF APPROACH TO THE REVIEW

The proposal to review RETRAN-3D was made by the RETRAN Maintenance Group to the staff on July 8, 1998 (Reference 5). The code documentation was then submitted in September of that year. The staff performed the review by assembling a group of four staff members with expertise in thermal-hydraulics, kinetics, and RETRAN use. The review was originally planned to concentrate on those portions of the code which were different from the RETRAN-02 code previously reviewed.

During the course of the review, requests for additional information (RAIs) were developed and transmitted to the applicant (Reference 6). Several meetings were also held with the Advisory Committee on Reactor Safeguards (ACRS) Thermal-Hydraulic and Severe Accident Phenomena Subcommittee. Those meetings and reviews conducted by ACRS members and their consultants along with subsequent staff review resulted in additional RAIs (Reference 7). In addition, as will be discussed later in this report, problems and errors with the "momentum equation" were identified.

The RETRAN-3D review departed from previous computer code reviews in that the code itself was requested from the applicant, was installed on the NRC's computer system, and was exercised extensively. The experience gained in doing so led the staff to several insights regarding needed code user training and guidance on acceptable model, correlation, and option specification.

As a part of the review of RETRAN-3D, the limitations and conditions on use of the previous code versions were reviewed to determine which were no longer applicable and could, thus, be removed on the current code version. In addition, the staff identified the limitations that would be necessary to determine whether or not RETRAN-3D could be used as a direct substitute for RETRAN-02, in other words, use of RETRAN-3D in a RETRAN-02 mode.

Milestones in the Review

- Request for review of RETRAN-3D: July 8, 1998. Receipt of code and documentation: September 1998. Acceptance of code for review by NRC: December 4, 1998.
- Requests for additional information by the staff: April 27, 1999 and August 25, 1999. Responses were submitted by the RETRAN Maintenance Group in References 8 and 9.
- Advisory Committee on Reactor Safeguards, Thermal-Hydraulic Phenomena Subcommittee meetings: December 1998, March, May, July 1999, March 2000
- Staff participation in RETRAN-3D training: August 1999.
- Staff/EPRI meetings: September, December 1998; March, April, May, June, July, August, December 1999; March, April 2000.

3.0 RETRAN-3D MODIFICATIONS AND ADDITIONS

The following RETRAN-02 models were modified or revised in developing RETRAN-3D:

- Mixture Momentum Equation
  - Added EPRI two-phase multiplier
  - Added Colebrook wall friction with pipe roughness

- Dynamic-Slip Equation
  - Added local momentum sources and sinks
  - Added continuous wall-to-phase and interphase friction model
  - Added Govier horizontal flow regime map
  - Added correlation for interphase friction for stratifies flow

- Algebraic Slip
  - Added model option based on EPRI-NP-3989-SR for countercurrent flow

- Point Neutron Kinetics
  - Added 1979 ANS decay heat standard

- Neutron Kinetics
  - Added multidimensional kinetics option
  - Added boron feedback capability
  - Added 1979 ANS decay heat

- Critical Heat Flux (CHF)
  - Added EPRI CHF correlation
  - Updated GE correlation

- Fuel Cladding Interaction
  - Added model from VIPRE-01

- Junction Enthalpy
  - Modified countercurrent flow model
  - Added option using level tracking model

- Iterative Time-Step Controls
  - User controls for dependent variables

- Boundary Conditions
  - Extended for noncondensable gases

- Compilers and Operating Systems
  - Adapted source for FORTRAN 77
  - Converted environmental library to FORTRAN 77

The following new models were added in developing RETRAN-3D (General Applications):

- Fluid Field Equations
  - Added continuity equation for noncondensable gases
- Dynamic-Slip Equation
  - Added dynamic area model and other models for complex geometry
- Neutron Kinetics
  - Added option for multidimensional kinetics
- Accumulator
  - Polytropic expansion model for the cover gas
- Generalized Transport
  - Transport soluble chemical impurities with either the liquid phase or the vapor phase
- Flow Structure Models
  - Added Duckler model for stratified flow
  - Added Zuber model for stratified critical flow
- Transient Solution of Field Equations
  - Implicit solution method for basic equations, components and auxiliary models
- Steady-State Initialization
  - Added implicit solution method option
  - Added option for initialization of steam generators at low power
- Flow Field, Component, and Auxiliary Model Solutions
  - Made implicit with coupling to basic fluid solution
- Method of Characteristics
  - Minimizes numerical dispersion within the energy equation
- Heat Transfer
  - Added correlations for condensation
- Dynamic Gap Conductance
  - New model to account for effects of clad deformation

The following new models were added in developing RETRAN-3D (Nonequilibrium Applications):

- Fluid Field Equations
  - Added vapor mass equations (5-equation model)
  - Added equation for noncondensable gas
- Neutron Kinetics
  - Added option for multidimensional kinetics

- Wall Heat Transfer
    - Added wall-to-phase for 5-equation model

- Interphase Heat and Mass Exchange
    - Added subcooled boiling, condensation and flashing models for 5-eqn. model

- Implicit Solution Method
    - Modified basic linearization for thermal nonequilibrium equation of state

- Iterative Time-Step Controls
    - Extended to nonequilibrium models

- Heat Conduction Solution
    - Extended boundary conditions to nonequilibrium models

- Thermal-Hydraulic Boundary Conditions
    - Extended as necessary for nonequilibrium models

- Solution for Flow Field, Component, and Auxiliary Models
    - Extended for nonequilibrium models as necessary

4.0 EVALUATION OF RETRAN-3D

The review of the RETRAN-3D code and documentation was broken down into four main sections: the thermal-hydraulic models and associated numerics, the neutron kinetics and associated numerics, the code assessment, and code use. Code use included a review of the user guidance, training, and experience in running the code on NRC computers. In performing its review, the staff took into consideration views and concerns raised by members of the ACRS, Thermal-Hydraulic Phenomena Subcommittee and the subcommittee's consultants. As a result, the original plan to review only material that had not previously been reviewed as part of the RETRAN-02 review was expanded to include several aspects of the code that were previously found to present difficulties. This especially included the formulation of the momentum equation. Discussions among the reviewers identified problems that had been raised with the formulation of the momentum equation dating back to 1974. These problems still exist in the older generation of codes, such as RETRAN, that are based on the RELAP3 and RELAP4 codes. Care must be taken in the use of these codes to ensure that situations do not arise in which violation of basic principles of physics occur.

The documentation for RETRAN-3D, as will be mentioned later in the discussions, was found to be misleading in part, and erroneous in part. The known errors will be corrected when the approved version of the documentation is prepared. The user must exercise caution in the use of the documentation when, as noted below, the text and nomenclature are inconsistent and do not follow standard usage. In addition, one significant section of the documentation is incomplete because well-defined user guidelines do not exist at this time. The high degree of complexity in the use of the code coupled with the large number of available options and code flexibility make high quality user guidelines critical to reliable code use.

This review of RETRAN-3D departs from previous computer code reviews in that the staff has installed a copy of the code on NRC computers. This has permitted the staff to exercise the code to assess its ability to perform as intended, evaluate its degree of user-friendliness, and roughly determine its level of robustness. The staff's experience in use of the code is described along with an evaluation of the basic level training being offered for the code user.

4.1 Thermal/Hydraulic Models

Vector Momentum Equation

During the course of the review, it became evident that problems were present in the derivation of the vector momentum equation. Several of the points of concern raised during the staff review of the documentation follow.

- Rigor: in an effort to show a rigorous approach, much effort has been expended on forms of the equation containing (for example) three-dimensional fluid shear stress terms and terms accounting for moving solid surfaces, neither of which are present in RETRAN-3D. This only confuses the exposition being attempted.

- Notation: the momentum equation derivation begins using the indicial notation common to computational fluid dynamics (CFD) texts and then transitions to a non-standard RETRAN-specific notation. The notation changes considerably as one goes through the description so that an equation that was understood on one page is almost unrecognizable on another.

- Typographic Errors: the numerous sections dealing with the momentum equation contain typographical errors, so that one is often not sure whether there has been a typographic error or whether a new notation has been introduced.

- Distributed Description: the description of the momentum equation is strewn over a large number of sections making it very difficult to comprehend.

- Terms Missing from Nomenclature: a number of terms used in the momentum equation are not to be found in the nomenclature causing a lot of time to be wasted trying to find their definition in the text.

- Missing Steps: despite the incredible detail lavished on the initial steps of the derivation; later on, large gaps appear as the interaction of terms are defined.

The code documentation should include a clear and concise statement of the partial differential equations being solved and the implicit assumptions involved, the process used to volume average the equations, how the resulting volume averaged conservation equations are differenced, and the specification of the interaction terms. To lessen confusion on the part of the code user, EPRI will clarify the documentation to address these deficiencies.

Specific Errors in the Momentum Equation

The "RETRAN-3D vector momentum equation" is actually a scalar equation of motion that is the projection of the vector momentum along a control volume dependent direction. The neglect of the momentum in the two directions perpendicular to this direction are never discussed in the documentation. The derivation of the momentum equation needs to be corrected. The "RETRAN-3D vector momentum equation" contains two errors that are manifested in simple demonstration problems such as tees and bends. The errors are discussed in greater detail in the following paragraphs. The equations and figures referred to are found in Volume 1 of Reference 4.

The first error occurs in going from Equation II.3-6 to Equation II.3-7. The last term on the right hand side of Equation II.3-6 will contain a cosine term from the vector dot product. The cosine term is missing in Equation II.3-7. The cosine term is required for a vector momentum equation since the pressure force is normal to the surface in question and the orientation of the surface changes with geometry such as for an elbow. This error will affect pipe bends of any angle. The answer (Reference 10) to Question 15 of the August 25, 1999 RAI (Reference 7) indicates that the effect of the constant pressure approximation stated on page II-74 was not considered. The pressure integral in Equation II.3-6 is not equal to the pressure difference term in Equation II.3-7 if a two region constant pressure approximation is used. If the pressures are not assumed to be constant, then in general, the p's in the pressure difference term of Equation II.3-7 are not the pressures at the end faces of the momentum control volume as is shown in the Porsching paper (Reference 11). Even if the equation was correct, the flow behavior in an actual elbow is far more complex than could be predicted by a one dimensional flow model. In an actual elbow there will be a pressure rise from the entrance to the outside radius of the elbow and a pressure drop to the inside radius (Reference 12). In addition, a multidimensional recirculating pattern will be established and flow separation can occur on the exit side of the elbow. The best that can be achieved with a one dimensional model is the prediction of the pressure drop through the elbow as a function of flow conditions and geometry. If the details of the flow in the elbow make a difference in the solution, a one dimensional flow model is not adequate.

The RETRAN-3D documentation also gives conflicting accounts of the assumed functional dependencies. For example, in the answer to Question 5 of the August 25, 1999, RAI, the pressure is assumed to vary linearly across the volume but page II-74 discusses using a piecewise constant pressure profile.

In a meeting with the staff on November 3, 2000, EPRI provided additional steps for the derivation but the additional information did not resolve the problem with the equation of motion. It is mathematically possible to obtain an equation of motion without a cosine multiplier on the pressure difference term by assuming that the pressure is decomposed into a volume constant pressure and a pressure that has variation over the volume. It is easy to show mathematically that the volume constant pressure integrated over the surface of the control volume results in a pressure difference term with no cosine multiplier. This approach just moves the difficulties into the φ projection of the F_{loc} term which is the projection on the nonuniform normal wall forces in the φ direction. The RETRAN-3D documentation states that this quantity will be computed using empirical models. The anticipated source of information for this specific quantity is unknown to the staff. An additional problem is that an equation for mechanical energy conservation cannot be derived from the resulting equation of motion and therefore it cannot be shown that mechanical energy is conserved by RETRAN-3D.

The second error appears in pipe configurations that contain flow splits such as a tee or an injection mixer like a jet pump. An example that applies the RETRAN-3D vector momentum equation to a flow split begins on page II-87 of Reference 4. To illustrate the error, consider the trivial case where junctions 2 and 4 are both horizontal and $A_2 + A_4 = A_1$. Also let the velocities at all junctions be equal. (This is the inverse of the configuration used for the RETRAN-3D jet pump which is based on the liquid-liquid ejector model from Bird, Stewart and Lightfoot (Reference 13). The jet pump model avoids the error that is contained in the tee example by adding a source term to the RETRAN-3D momentum equation, Δp_{mix}.) In the absence of wall friction, $p_1 - p_2$ should clearly be 0. Applying Equation II.3-35a the calculated pressure is:

$$p_1 - p_2 = \frac{W_2^2}{2\rho}\left[\frac{1}{A_2^2} - \frac{1}{A_1^2}\right].$$

Not only is the pressure difference non-zero, it depends on the area of the exit path and will predict different pressures for p_2 and p_4 if A_2 and A_4 are not equal. The error in Equation II.3-35a is contained in the term:

$$\frac{W_2^2}{2\rho}\left[\frac{1}{A_2^2} - \frac{1}{A_1^2}\right]$$

The term should actually be:

$$\frac{W_2^2}{2\rho}\left[\frac{1}{A_2^2} - \frac{1}{\frac{W_2^2}{W_1^2} A_1^2}\right]$$

to properly account for the pressure difference that is analogous to the pressure difference resulting from the contraction or expansion of a Bernoulli stream tube. The error in the tee EPRI has agreed to fix in order to avoid artificial pressure drops that result from this effective numerical loss. The best that can be done is to insure that the Bernoulli head is conserved in RETRAN-3D and use appropriate experimentally derived loss coefficients that apply to the specific geometry being modeled. In general, the true loss coefficients in the branches of a tee depend on the geometry, the absolute flow rates, and the flow rate ratios between the branches.

Subsequent discussions with the applicant have resulted in agreement to correct this error in the RETRAN-3D code. Therefore, with this correction, the staff accepts the formulation of the momentum equation.

Applicability of Porsching Paper to RETRAN-3D Momentum Equation

The Porsching Paper was submitted on March 6, 2000, in support of the RETRAN-3D "vector momentum equation." The paper does not appear to have any mathematical errors. Unfortunately, the definitions and restrictions on control volumes that are required to be consistent with the mean value theorem make the paper irrelevant to the RETRAN-3D code. The pressures and flows in RETRAN-3D are defined in a control volume with specified functional dependencies. The integrals should be evaluated with the RETRAN-3D assumed function dependence for pressure and flow.

Momentum Transfer Due to Phase Change

The RETRAN-3D four- and five-equation models neglect momentum transfer due to phase change. Neglecting this term can lead to unphysical results. An example of an unphysical result that can occur is that droplets will accelerate as they evaporate in mist flow. The neglect of this momentum transfer may also cause numerical problems and instabilities for the code. This approximation will be fixed so that unphysical results and numerical problems do not occur.

Therefore, with the above corrections, the staff accepts the models.

Constitutive Equations

RETRAN-3D provides many options for heat, mass and momentum transfer that can be selected by the user. Unfortunately, the range of applicability for the correlations is not given and there is a lack assessment for these models. A licensee wishing to use the correlations will have to provide both separate effects and integral effects assessment over the full range of conditions encountered during the application of interest. An assessment of the uncertainties must also be provided. The assessment must address the consistency between the RETRAN-3D calculations and any auxiliary calculations that are part of the overall methodology. Examples of auxiliary calculations are departure from nucleate boiling, critical power ratio or reactor physics calculations.

Generalized Laminar Friction Model

Generalizing the laminar wall friction model is an improvement over using a pipe laminar friction coefficient. Unfortunately there is no proper assessment of the capability of the model. The Purdue thermosyphon test uses this capability to apply a curve fit to both the constant and the exponential dependence of the Reynolds number. Proper modeling of the test facility was not performed using geometry dependent friction coefficients and the theoretically correct inverse Reynolds number dependence.

EPRI will perform the above assessment correctly. When this assessment is complete, the staff will review it for acceptance.

Pressurizer Model

The pressurizer model is not well assessed and is highly dependent upon the user. In the limited assessment provided, there appears to be a large discrepancy between the implicit solution method and the standard solution method. No discussion or assessment is given of this discrepancy. The discrepancy between the two predictions needs to be explained. Assessment and justification of all input parameters must be provided by the user.

While the model does not directly account for thermal stratification, its effects can be included by use of normal nodes below the pressurizer volume. The user will have to justify the lack of thermal stratification or the use of normal nodes below the pressurizer should there be an indication that it would be important in the analysis.

The mixture and two-region energy equations are consistent for the implicit solution method where the mixture energy equation is used with the vapor-region energy equation. This eliminates inconsistency between the two-region and mixture energy equations and the concern regarding a potential drift in the region energies.

The staff notes that when a pressurizer fills or drains, a single region exists for which the normal pressure equation of state is used. Lack of numerical discontinuities in validation analyses of filling and draining pressurizers indicates that the model is functioning properly. It is the responsibility of the code user to justify any numerical discontinuity in the pressurizer during a filling or draining event.

The pressurizer model has options that require user-supplied parameters. Users must provide justification for these model parameters.

5 Equation Non-Equilibrium Model

The 5-equation non-equilibrium model has not been assessed and therefore is not approved for use. Licensees who wish to use the model will have to provide both separate effects and integral effects assessment over the full range of conditions encountered during the application of interest. An assessment of the uncertainties must also be provided. Demonstration problems provided in Volume 4 of Reference 4 show that the peak power in BWR pressurization problems is significantly changed when going from the four equation model to the five equation model. The peak power is reduced in the five equation model apparently due to a less severe void collapse caused by the interfacial heat transfer resistance of the five equation model. In a four equation equilibrium model the interfacial heat transfer rate is effectively infinite. Due to this sensitivity of BWR pressurization applications to the interfacial heat transfer, licensees who decide to use this option will need to specifically address the uncertainty in peak power due to interfacial heat transfer.

Critical Flow

Three critical flow models are included in RETRAN-3D:

- Extended Henry/Fauske
- Moody
- Isoenthalpic Expansion/Homogenous Equilibrium

The three models are stated as acting to put an upper bound on junction flow. However, the staff notes that the code does not have the ability to calculate critical flow in Fanno or Rayleigh like situations such as would occur from broken pipes or steam generator tubes where acceleration is driven by friction or heat addition instead of by area change. While Moody and Henry/Fauske are standard, accepted models in the nuclear industry, no explanation is given to justify what is meant by "extended" Henry/Fauske. The data used in assessing the model was limited to the Fauske straight tube and Marviken data rather than critical flow through nozzles. With only three data sets used for assessment, there is great uncertainty in the results. The Isoenthalpic Expansion Model is really the Isentropic Homogeneous Equilibrium Model. The model can readily give the critical pressure and mass flux (Reference 14).

None of the critical flow models noted above are appropriate when noncondensable gases are present. In the presence of noncondensables, the critical flow model is automatically bypassed by the code. This should exclude the code from analyzing shutdown transients where air can be present in the system. Also, it is not considered good practice to have a code bypass model on its own without warning the code user that this is being done. Doing so places an added burden on the analyst who needs to know when the code is invoking limits and restrictions.

Drift Flux Model - Chexal-Lellouche Model

Although the Chexal-Lellouche model is based on a curve fit rather than being mechanistic, the data base upon which the model is based is large and fairly comprehensive. On the other hand, the model uses a "fluid parameter" that directly affects the value of the distribution coefficient, C_0. If the model were mechanistic in nature, or even based on the appropriate property groups, fluid scaling would be implicit in the model. However, the fluid parameter is a set of empirical relations which have a dependence as a function of void fraction. There is a significant difference in the behavior for steam-water and air-water mixture. This raised three potential concerns:

- Range of Applicability: for steam-water, the fluid parameter is an explicit function of pressure, not a function of fluid properties. Due to its empirical nature, it cannot be extrapolated beyond its database. However, this is not a significant problem as the diabatic steam-water database extends to a pressure of 150 bar, and the adiabatic extends to 180 bar.
- Applicability of Air-Water Validation: a fair amount of the validation work was performed for air-water mixtures. Because of the fluid parameter, this is the validation of a

separate and distinct model. It is not relevant to any steam-water used in RETRAN-3D calculations.

- Slip with Noncondensable: if different fluid parameters are used for steam-water and air-water cases, for the case of a gas phase that is a mixture of steam and noncondensable gas the user must be aware that there is no guidance or provision for determining the appropriate fluid parameters. Justification on a case-by-case basis is needed if the steam-water parameter is used.

Normally the drift flux model is used for vertical flow where the two phases are tightly coupled, as would be expected since a fundamental principle of drift flux is that buoyancy and interfacial forces balance each other. Other codes are careful to use drift flux for regimes such as bubbly-slug flow, but not for the annular flow regime. In annular flow, the buoyancy force becomes progressively less important as the wall shear begins to offset the interfacial friction. Drift flux models are used in which the drift velocity is a function of flow regime.

The Chexal-Lellouche model retains the bubble rise velocity as the cornerstone of the drift velocity. Multipliers are added to adjust the drift velocity for the various flow regimes, annular, horizontal, etc. The multipliers are based, once again, on elaborate curve fits and do not clearly represent the governing physical phenomena. Retaining the bubble rise velocity as a principle component of the drift velocity for flow regimes where it is clearly not relevant raises questions about the applicability of the model itself. The very large database behind the Chexal-Lellouche model leads to the conclusion that there are likely one or more multiplying factors that must be compensating for this error. It is not possible to discern how or where these compensating errors exist from the code documentation. This is seen in the code application manual (Volume 4 of Reference 4), Figures IV.4-9 through IV.4-11, wherein the model does well in bubbly-slug flow but at high values of the void fraction consistently underpredicts the void fraction.

The application of the Chexal-Lellouche model to the annular and annular/mist flow regimes must therefore be used with caution and the effect of underprediction of the void fraction must be explained.

Besides the concerns noted with application of the Chexal-Lellouche model to annular flow, its applicability to horizontal flow must be avoided. In horizontal flow, the balancing forces are now wall drag and interfacial friction. Using the bubble terminal velocity as the foundation of the drift velocity is clearly incorrect. In the steam-water database given in Reference 6, three tube diameters are included ranging from 22 to 75 mm. Less than 3 inches are too small to represent the large diameter pipes found in reactor coolant systems. If the air-water database is included, ignoring the disqualifying effect of the fluid parameter, then the largest pipe diameter is 127 mm which is large enough. However, the mass flux range is so large (3,600-4,700 kg/s-m^2) that the regime would be dispersed. Therefore, the database is insufficient for an empirical model for horizontal flow in reactor scale piping.

Regarding application of the Chexal-Lellouche model to the counter-current flooding limit (CCFL), the staff identified the following concerns:

- Geometry Effects: CCFL in complicated geometries such as tie plates requires a highly empirical correlation specific to that particular geometry. The Chexal-Lellouche model makes no distinction between a tube and a tie plate.

- Air-Water Data Applicability: due to the fluid parameter the air-water data must be excluded from the model's validation database.

- RETRAN-3D Validation: there are no data comparisons given for CCFL in the RETRAN-3D applications manual (Volume 4 of Reference 4). It is unclear if or how CCFL is implemented in RETRAN-3D.

- Pressure Dependency: based on the data comparisons given in Reference 15, Figures 5-6 and 5-10, the pressure dependency of the predicted CCFL seems to be incorrect. In Figure 5-6, which is plotted using the Kutateladze number, the pressure effect is greatly overstated in the calculation, while in Figure 5-10, plotted using the Wallis scaling, the predicted pressure trend is opposite that of the data.

The staff therefore concludes that Chexal-Lellouche cannot be used in situations where CCFL is important unless validation for the precise geometry and expected flow conditions has been perfomed.

The Chexal-Lellouche drift flux model appears to be an improvement over the previous RETRAN drift flux models based on the limited assessment provided. Licensees who wish to use the correlation will have to provide assessment over the full range of conditions encountered during the application of interest. Since the correlation is purely empirical in nature the assessment must be provided for full scale in all variables of interest. An assessment of the uncertainties must also be provided.

In summary, overall the Chexal-Lellouche model is accurate for most applications. However, due to its empirical nature, care must be taken to avoid extrapolation. Also, for the cases noted, such as annular flow in large pipes, horizontal flow, and CCFL, the model should not be used or an explanation should be provided for the effect its use has on the calculation. The user is referred to Condition 16 below for further guidance on use of the Chexal-Lellouche model.

Boron Transport

There are several models in RETRAN-3D to minimize numerical diffusion or provide front tracking for fluid temperature fronts: the method-of-characteristics, the transport delay model, and the enthalpy transport model. Each of these models is used in a particular circumstance as a user option. Boron transport is handled as a passive contaminant by the "general transport model" (Volume 1, Section VII-5.0 of Reference 4). This model uses a first order accurate upwind difference scheme with an implicit temporal differencing. This approach is highly diffusive, especially if the Courant limit is exceeded. This scheme can result in a front arrival that can be spread out over a long period and its amplitude reduced to about half that of the peak. Since RETRAN-3D has the same model as RETRAN-02 MOD003 and subsequent versions that have been approved for use, the RETRAN-3D model is also approved with the caveat that the potential to produce misleading results with this scheme necessitates careful review of the results for any case where boron transport/dilution is important.

4.2 Neutron Kinetics Models

Existing approved versions of RETRAN have a one-dimensional kinetics capability. EPRI has introduced a three dimensional kinetics capability to eliminate some of the limitations in previous versions caused by the use of a one-dimensional model by introducing a solver based on the analytical nodal method (Reference 16). The method used was originally implemented by EPRI in the ARROTTA code and was adapted for use in RETRAN-3D. The current review is limited to the kinetics models that have been introduced into RETRAN since the last approved code version. All of the kinetics models discussed are related to the implementation of the three dimensional solver. Therefore, the review considered the following:

1. Development and implementation of the Analytic Nodal Method (ANM) solver.
2. The performance (validation) of the ANM solver.
3. The cross section model.
4. Coupling to the thermal-hydraulics model.

The staff position that the documentation for a code under review, and the code itself, must be submitted allowed a direct evaluation of the capabilities of the ANM solver relative to the staff's own kinetics methods.

Theoretical Development

The theoretical development of the 3-D kinetics models is described in the RETRAN Theory and Numerics Manual (Volume 1 of Reference 4). This information and the availability of the source code formed the basis of the review of the development and implementation of the ANM solver. The model was developed in a manner similar to other equivalent methods, such as Reference 17, and no apparent deficiencies were identified. The major differences between the methods is in the solution of the 1-D nodal coupling equations. The review began with equations V.2-44 through V.2-46 (the two-group diffusion equations and the precursor equation), but the staff did not review the information presented in Chapter 2 on the derivation of the diffusion equation from the transport equation because there are several different equally acceptable techniques available to derive the diffusion equations and the form of the diffusion equation solved in RETRAN is correct. The global diffusion theory equations and the 1-dimensional nodal balance equations are solved with a technique referred to as the non-linear method which has been successfully implemented in other methods (Reference 20). The nodal leakage source terms can be determined by one of three methods: an explicit method or one of two implicit algorithms. The explicit method is a Gauss-Seidel iterative method which "explicitly" calculates the leakage terms. The basis of the implicit methods is that the leakage equations can be evaluated by using a truncated Neumann Series that "implicitly" calculates the leakage terms.

RETRAN-3D includes a model to calculate the individual contributions to the system reactivity balance from relevant variables such as moderator temperature and fuel temperature. This model is based upon the assumption of space-time separability and the use of the steady state adjoint flux as the weighting function. The equations are separated and the time dependent amplitude function is recast into the point kinetics equation. An equation for reactivity is then extracted from this formulation and separately solved for the reactivity contribution from fuel temperature, moderator temperature, moderator density and control rod insertion. A parameter called residual reactivity is calculated which is the difference between the total reactivity and all of the components of reactivity. The residual reactivity is used to assess the error from the assumption of space-time separability.

Validation of the ANM Solver

The RETRAN-3D three dimensional kinetics solver has been assessed by code-to-code comparisons and comparison to experimental data. RETRAN-3D was originally assessed by EPRI against international standard problems and comparison to other codes. Both of these types of assessment are basically code-to-code comparisons. They are good for evaluating a code's capabilities relative to other solvers, but they do not answer the fundamental question of assessment: do the equations really calculate the physical phenomena? To answer this question, the staff developed a benchmark problem based upon the SPERT series of tests. The cross sections and problem definition were supplied to EPRI and they used RETRAN-3D to predict the problem. This section discusses the results of the assessment of the three dimensional kinetics solver in RETRAN-3D.

Code-to-code comparisons using RETRAN-3D were peformed by both EPRI and the staff. These problems can be further subdivided into steady state assessments and transient assessments. For steady state assessment, EPRI compared its results with two NEACRP problems (NEA sponsored international standard problems) and HERMITE calculations. The staff used RETRAN-3D to compare its results with NESTLE calculations and TORT calculations. All of these comparisons consisted of power distribution and eigenvalue calculations. All of the comparisons demonstrated that RETRAN-3D is capable of predicting power distributions and eigenvalues with accuracy comparable to other codes. The TORT comparison is unique because TORT is a three-dimensional transport theory code capable of calculating higher modes of the flux. The staff performed what is known as an S_8/P_5 calculation, that is, an eighth-order quadrature is used to expand the angular flux with a fifth-order expansion of the scattering kernel. Both rodded and unrodded cases were studied. For more information, refer to Appendix A. Although these types of methods are not necessarily any more accurate for reactor calculations than diffusion theory methods, the staff performed this analysis to confirm the calculation of RETRAN-3D. For transient assessment, EPRI compared RETRAN-3D results to two NEACRP problems and HERMITE calculations. Once again, these comparisons demonstrated that RETRAN-3D is as accurate as other similar methods.

Due to the limitations of code-to-code comparisons, the staff defined a problem using experimental data from the SPERT test series (Reference 19). The staff developed cross sections for the SPERT E-core using a pre-release version of sas2d (a module of SCALE 5 under development) and used these cross sections in a NESTLE, Reference 17, model to predict two rod ejection tests referred to as Tests 81 and 86. Test 81 was initiated from hot zero power conditions and Test 86 was a hot full power case. Figures 1 and 2 indicate that the NESTLE predictions of the experiment are very accurate. The cross sections and geometry information were provided to EPRI to assist their prediction of the SPERT tests with RETRAN-3D. Figures 3 through 6 show that RETRAN-3D also accurately predicted the test results.

The SPERT benchmark is an excellent source of data for prompt critical excursions. However, the SPERT E-core was a very small, tightly coupled reactor and when one examines the results it becomes obvious that the flux does not significantly deviate from the fundamental mode during the rod ejections. This is important because it limits the usefulness of the benchmark. Accurate prediction of these experiments only shows that the balance equations are accurately predicting the neutron population; not that they can accurately predict the neutron population when the flux deviates from the fundamental mode. This discussion is not meant to minimize the importance of the SPERT validation, but, rather, to clarify its value. The SPERT benchmark is important because it demonstrates that the neutron diffusion equation is valid during super-prompt critical excursions and that it accurately predicts the neutron balance which is directly proportional to the power. There is no known experimental data for super-prompt critical excursions involving larger reactors which would exhibit higher modes of the flux. One must, therefore, defer to the types of code-to-code comparisons previously discussed to assess a code's ability to predict super-prompt critical excursions with asymmetric power distributions.

In summary, the validation of the three dimensional kinetics solver in RETRAN-3D which was reviewed allows the staff to conclude that the neutron diffusion equations as solved in RETRAN-3D accurately predict the neutron population and that the code's ability to predict spatial asymmetries is as accurate as higher order methods.

Cross Section Model

The cross section model is a polynomial fit of pre-calculated static cross sections over a range of thermal and hydraulic conditions which will bound the problem of interest. This type of model has been used with considerable success in many other applications (Reference 17). The cross sections are a function of fuel temperature, moderator temperature, moderator density, control fraction, and soluble boron concentration. Assembly discontinuity factors are similarly defined to be functions of these variables. These coefficients are calculated offline and provided to RETRAN-3D through one of several interface files. The use of static cross sections to predict transient conditions is justified by the SPERT validation discussed in the previous section.

Coupling with Thermal-Hydraulics

RETRAN-3D, like many other similar codes, allows for a coarser thermal-hydraulic mesh than what is used to resolve the flux. Furthermore, RETRAN-3D has only a one-dimensional flow and heat transfer capability. The "3D" in the name refers only to the neutron kinetics capability. The applicability of these types of assumptions can only be assessed on a case-by-case basis. For example, the SPERT validation demonstrates that for that reactor, cross flow (radial flow between assemblies) is not important because the predicted power was very accurate. The SPERT validation cannot, however, be extended to the general case for which one does not have experimental data to assess the applicability of a given modeling scheme. The staff concludes that the three-dimensional neutron kinetics model in RETRAN-3D can adequately predict the neutronic response of a neutronics dominated event. However, caution needs to be exercised when applying a model such as RETRAN-3D to analyses where multi-dimensional flow may significantly effect the results, such as the main steam line break. Without adequate data to assess three-dimensional thermal-hydraulics the staff can only conclude that for tightly coupled thermal-hydraulic and neutronic events RETRAN-3D produces results comparable to those of other accident analysis codes. Application of the code to these types of events requires specific assessment and justification by the user.

4.3 Code Assessment

Computer code assessment generally consists of three phases: phenomenological assessment, separate effects assessment, and integral systems tests and full scale plant data (when they exist) assessment. There have been numerous attempts at defining what constitutes adequate assessment and two of the best examples are the work of the Nuclear Energy Agency in Paris, France (Reference 20), and the development of the Code Scaling, Applicability, and Uncertainty (CSAU) effort of the NRC (Reference 21). These efforts have shown that a simple list of data against which a computer code is to be assessed is not sufficient. It is also necessary to determine what the use of the code will be, which models are important, what phenomena are important, and how they rank relative to one another during the application of the code. The clearest way this is done is through use of a Phenomena Identification and Ranking Table (PIRT) as described in the CSAU documents. From the PIRT results, the range of parameters over which a given highly ranked phenomenon is considered to be important will be determined along with a test matrix to assess the model over this range of parameters. Without a PIRT it is more difficult to determine that the model is performing acceptably for the specific application.

The code documentation for RETRAN-3D presents assessment against a brief list of phenomenological, separate effects, and integral systems tests. The assessments have not been performed with the forethought and planning that would be done as part of a PIRT development scheme. No PIRT has been developed or presented. The bulk of the assessment consists of actual plant calculations performed by various participating utilities. Many of the figures provided do not indicate what code version was used for the calculation. As would be expected, actual plant data are very limited in scope and qualification. This makes the evaluation of the applicability and validity of the assessment very difficult.

Additionally, the applicant states in Volume 4, Assessment Manual, of Reference 4,

> Qualification (of the code) is an additional step that lies beyond both verification and validation. Qualification is the process of demonstrating that the code and a specific plant model are adequate for a given application, e.g., analysis of a boiling water reactor response to a turbine trip event for support of reload fuel licensing. Although the code developer can perform generic demonstration analyses to support qualification, completing the qualification is ultimately the responsibility of each individual code user. This statement is particularly appropriate for RETRAN because of the flexibility of the code and because much of the modeling is established by user input.

Assessment of the RETRAN-3D code for the models not explicitly approved in this safety evaluation will be the responsibility of the licensee or applicant. In addition, application of the RETRAN-02 or RETRAN-3D codes for best estimate analysis of UFSAR Chapter 15 licensing basis events may require additional code and model assessment, and an evaluation of uncertainties to assure accurate prediction of best estimate response. This condition is based on the absence, in the best estimate analysis approach, of the conservative assumptions in traditional UFSAR Chapter 15 licensing basis analyses. For each use of RETRAN-3D in a licensing calculation, it will be necessary for a valid approach to assessment to be submitted, which is expected to include a PIRT for each use of the code and the appropriate assessment cases and their results. The scope of the PIRT and validation/assessment will be commensurate with the complexity of the application.

4.4 Code Use

User Options Available in RETRAN

RETRAN is a generalized thermal-hydraulic computer program which can be used to model a variety of thermal-hydraulic configurations. Noding detail and layout are options left to the user. As RETRAN has evolved, numerous options became available to the user. The options include surface heat transfer correlations, critical heat transfer correlations, two phase friction and drift flux correlations. The various correlations are applicable for different fluid conditions. The correct application is strongly influenced by the experience of the user.

The developers of other analysis codes have generally observed the policy of replacing old mathematical models and correlations as new models and correlations were developed, thus keeping the number of options available to the user to a minimum. The developers of RETRAN have used the policy of preserving the ability of the code to "look back" and utilize all previously developed models as options. This policy greatly increases the number of options available to users. During the training course, only the most frequently used options were presented to the students; thus there is a high potential that students could incorrectly apply the other options that were not presented.

The numerics of RETRAN-3D are limited to a maximum of 5 conservation equations, one of which is the relative velocity "slip" between the steam and water phases. Steam in RETRAN-3D is always assumed to be saturated. For this reason the code cannot accurately model emergency feedwater injection into the steam space of a B&W once-through steam generator or ECCS injection into a steam filled pipe during a LOCA. The liquid phase in RETRAN-3D can be either subcooled or superheated. Although code limitations were described by the course instructors, the code limitations are a potential source of error for inexperienced users. The RETRAN-3D retains the older 3 equation and 4 equation formulations from RETRAN-02 giving users the option of assuming complete thermal equilibrium.

In addition to slip models to calculate separation between the steam and liquid phases RETRAN contains bubble rise models from RELAP4. Bubble rise models were useful when reactor systems were described using a few large control volumes. With the more detailed noding, bubble rise models can produce unphysical alternating layers of steam and water in vertical components. Location of slip components adjacent to bubble rise components can result in code errors. It was mentioned that different bubble rise models are used for different steam generator components; however, the theory and calculation needed to obtain the correct bubble rise coefficients were not described.

Some models in RETRAN require particular knowledge by the user. For example, the code can calculate decay heat using either the 1973 or the 1979 ANS5.1 standards. The code does not provide a direct method of inputting the additional energy contribution from neutron capture in stable fission products or the variation in standard deviation with time. These can be input as control functions by experienced users but could be easily left out by a novice.

An accumulator model has been added to RETRAN-3D. Previously users modeled accumulators as equilibrium volumes. Since the accumulator gas may become very cold during discharge from adiabatic expansion, this was a potential source of error. Some users tried to account for the non-equilibrium effect by using increased loss coefficients in the accumulator discharge line. Additionally, the new accumulator model eliminates the use of a valve to inhibit nitrogen from entering the system since RETRAN-3D is capable of handling non-condensables.

Control system (blocks) are now evaluated implicitly with the fluid conservation equations in RETRAN-3D. Previously control systems were solved consecutively. This caused the results to be dependant on the order that the control systems were entered. This section was briefly covered during the training; however, for such a complex input modeling, it was insufficient to give the student an appreciation of the differences in results that are created through the use of the different types of control systems.

One potential source of error in RETRAN-3D is that the spacial power distribution for decay heat is assumed to be the same as that for the neutron flux. The 3D kinetics can calculate instantaneous changes in the spacial neutron flux resulting from control rod movement or local voiding. The decay heat power distribution should be a function of the previous power history which could result in a larger value for the decay heat.

User Experience

During the course of the RETRAN-3D review, the staff built several models to test the user options and exercise the code. Using the code to develop models enabled the staff to evaluate the user's manual and understand the ease of using the code by experienced analysts faced with the new version of RETRAN for the first time. It also gave the staff the understanding of the impact of attempting to upgrade RETRAN-02 MOD5 decks to RETRAN-3D. In addition, building new models enabled the staff to understand the use of new models and options implemented in RETRAN-3D that were not available in previous versions of the code. Exercising the code to assess the user friendliness of the code is especially important for RETRAN because the code is targeted at an audience of multiple users in diverse locations who have varying levels of code experience. These users are not exclusively those working within the company who developed the code; therefore, the users do not have direct access to the code developers when questions need to be answered and detailed understanding of models and how they were intended to be used are needed.

One of the models the staff developed is a RETRAN-3D model of the HERMITE test case. This model was developed based on the BWR rod drop sample problem. The sample problem was extended to include multiple channels. The channels created were developed from the original method of using volumes and heat components to create channels. The user manual did not clearly explain how to create individual channels and the staff found it to be a confusing process. Additionally, the input deck was very difficult to bring to steady state with the steady state initialization scheme because of multiple flow paths through the core and difficulty in dividing the flow into the channels which did not divide into an even number. The small error in flow this created made the code difficult to achieve steady state initialization. While this demonstrates that the code very accurately accounts for mass, it will be more problematic for users, especially those with little experience, building detailed models. The staff also found, when trying to make full core models with multiple channels, that the models exceeded a limit on the number of volumes allowed. This forced a compromise to the modeling and created a less detailed model than was sought. The staff also found that the channel rod mapping was very confusing. It is not well defined in the user manual, and using the sample problem as a guide, required trial and error to obtain the correct solution.

A number of small errors in the user's manual created confusion for the staff while attempting to use new options in the code. These errors ranged from options used in sample problems neither being supported in the user's manual nor there being a discussion of what the option does, to typographical errors that made the user stumble until they figured out the error, or being sent to a section of the documentation that was not relevant.

A concern raised during the use of the code, is that the manual does not explain how to lump and unlump components. This is an area where many inexperienced users will have difficulty without guidance. User guidelines are expected to deal with this concern when they are prepared.

During the experience with the code, it was noted that many of the newer options do not have corresponding minor edits as an option. The lack of minor edits makes it difficult to check the results of the code output to ensure that the code is predicting what is expected and that the input used in the code is what was desired. The lack of user guidance was a hindrance when using the code, especially when trying the new options. The combination of a lack of minor edits and guidance results in even an experienced user introducing errors in the input deck that should not have been there.

Conclusion Regarding User Experience

Both inexperienced and experienced users must use a great deal of caution with RETRAN-3D. The confusion in developing input models, due to the lack of user guidance combined with the lack of minor edits to verify output, could cause excessive undetected errors in input decks. The staff believes that it would be beneficial if the discussion of the new models and their applicability was expanded and part of the discussion that is currently in the theory section where the user is referred was reiterated in the user manual to assist the user with refreshing the memory of applicability and ranges of options.

Training

The NRC staff attended a RETRAN-3D training course in Idaho Falls, Idaho from August 16 to 20, 1999. The purpose was to evaluate the effectiveness of the training program and to obtain more knowledge of the RETRAN-3D computer code. The basic RETRAN-3D training course is intended for users that have no previous experience with the RETRAN-3D code but have a basic understanding of physics and engineering concepts.

The RETRAN Training Course

The principal instructors had worked with the RETRAN code for many years. They had taught RETRAN training classes for years. They appeared to be very familiar with the details of the code, were very comfortable answering a wide range of questions, and reacted graciously when we found an error in the code during the training session. The instructional material was well organized. Over the years of teaching RETRAN the instructors had responded to comments by students and improved the material. The course is a blend of theory and application. The actual fluid and heat transfer equations solved by the code were presented and explained. The strengths and weaknesses of many of the user options were discussed. Although the course was excellent, RETRAN is sufficiently complex that considerably more than one week of training will be required to produce qualified users.

The instructors taught positive modeling techniques for incorporating enhancements to the basic model. These modeling techniques include using additional cards to override previous modeling input instead of changing the original cards in the model and breaking control systems into pieces and adding them into the model in pieces which makes it easier to troubleshoot if errors are introduced into the model. Students were encouraged to draw schematics of their system to visually place problem specifications. Using this modeling technique reduces overspecification problems that arise during the steady state initialization routine which is unique to RETRAN. They also encouraged the exchange of information on modeling techniques through the RETRAN newsletter, which is distributed to the RETRAN users group members.

Since the intended target students are those that have a basic understanding of physics and engineering, some basic engineering concepts were not fully explained. These concepts were those that are used in computer modeling of thermal-hydraulic phenomena such as the Courant limit. When questioned, the instructors were unable to provide a visual representation so that the student understood what the limitation actually represents and how to use it. This lack of understanding might result in development of an inadequate model that could miss key phenomena or result in unreliable results due to numerical instability introduced by the model.

The training course utilized a computer interface to ease model development and preclude introduction of computer dependent behavior. This interface is currently not included in the code release package, so when students return to their site the interface between user and code will be significantly different. Training this way can introduce confusion in users who are unfamiliar with the code. Computer dependencies can occur and ultimately interfere with the proper RETRAN application and analysis of the results.

The students, other than the NRC staff members, had little experience with the code. Achieving code proficiency and analytical judgement requires concentrated and dedicated involvement with a large analysis tool such as RETRAN-3D.

Conclusions Regarding the RETRAN-3D Training

The one week RETRAN training course was excellent in content and the course material was well organized. Considerably more than one week of training will be needed to produce experienced users. The utility personnel at the course appeared to be new RETRAN users with little experienced backup at the home office. The combination of inexperienced users and lack of readily available support will make progress in acquiring the skill necessary to develop adequate code input models and interpretation of analysis results in a long and difficult process.

Users need to read the NRC safety evaluations on RETRAN so as to be appraised of the applications for which the code has been approved. Users also need to review the EPRI code qualification documents showing comparisons to experimental data so as not to use the code for applications and conditions for which it has not been qualified. This was not addressed in the training but needs to be added for the sake of the inexperienced new user.

The training course was a good beginning in the process of development of a competent analyst. The utilities involved in the use of the code should understand that the new user attending the first training course is not sufficiently trained to provide reliable analytic results and insights. Additional training and experience are necessary and should be sufficient to satisfy the position stated in Generic Letter 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," June 24, 1999, Section 2.3, Training and Qualification of Licensee Personnel.

> A training program should be established and implemented to ensure that each qualified user of an approved methodology has a good working knowledge of the codes and methods, and will be able to set up the input, to understand and interpret the output results, to understand the applications and limitations of the code, and to perform analyses in compliance with the application procedure. Training should be provided by either the developer of the code or method, or someone who has been previously qualified in the use of the code or method.

User Guidelines

The development of advanced thermal-hydraulic analysis codes has prompted close examination of the effect of the code user on analysis results. For the last two decades this has been an increasing concern to the international community. Much has been written about the "user-effect," especially following experience in performing International Standard Problems under the auspices of the Nuclear Energy Agency in Paris, France. Multiple users of the same computer code have been given the assignment of modeling one well-documented experiment in an integral test facility but their results have often diverged.

The large number of options available in RETRAN-3D make the affect of the code user on the results significant. For example, there are at least three different ways to model a volume:

thermodynamic equilibrium, partial non-equilibrium, or two-region non-equilibrium. In addition there are multiple options for determining the temperature profile within a volume: the temperature transport delay model, the enthalpy transport model, and the method-of-characteristics, and the options for phase separation within a volume and it is easy to see that there are an almost infinite number of combinations of models from which the user can select.

The situation becomes even more complicated for junctions where there are seven slip options, in at least two of which the user can adjust the model's coefficients. In other places the modeling guidelines acknowledge a modeling deficiency in the code and suggest that the user overcome it by using the control system to adjust a model. Thus, such a large user effect coupled with the opportunities to misuse the code to get a desired answer necessitates well defined code user guidelines and code user training.

Code-specific user guidelines do not exist for RETRAN-3D. EPRI has stated in response to staff RAIs that user guidelines will not be developed for two to five years in the future and not until additional experience with the code has been gained. The staff concludes that the lack of a detailed RETRAN-3D specific user guideline document mandates a statement on the user's experience and qualification with the code when analyses are submitted in support of licensing actions. This statement is expected to be consistent with the guidance of Generic Letter 83-11.

The RETRAN-3D Maintenance Group informed the staff in a meeting in November 2000, that a peer review process was being established by which applications of the RETRAN-3D code would be reviewed for consistency with accepted nodalization and option selection practices. The staff is encouraged by this move on the part of the RETRAN-3D user community. The staff believes that this peer review will be responsive to many of our concerns about application of the code and confidence in the user.

5.0 EVALUATION OF RETRAN-02 CONDITIONS OF USE

Staff reviews of previous versions of RETRAN have resulted in a number of limitations and conditions on use of the code. As a part of the review of RETRAN-3D, the staff has examined the limitations and conditions on the use of the earlier version to determine which are still applicable to RETRAN-3D and which have been responded to through the new models and additions in RETRAN-3D. The staff's evaluation of the limitations and conditions on use follows. Each condition is stated followed by the staff's position on that condition.

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

Staff position: RETRAN-3D has been modified to include a 3-dimensional nodal kinetics model based on the analytic nodalization method similar to accepted codes. The code has been assessed by calculation of the response of the SPERT prompt-critical tests and has been confirmed by the staff by comparisons with calculations performed with the NESTLE and TORT codes. The staff concludes that the code can adequately predict the response to prompt-critical events such as the PWR rod ejection accident and the BWR rod drop accident. If void generation occurs from an initially un-voided case, the user will have to justify crediting this negative feedback in the analysis.

The code was used by a participant in the Nuclear Energy Agency's International Standard Problem calculation of a hypothetical main steam line break (MSLB) at the Three Mile Island Unit 1 plant. The results of the calculation comparison indicates that RETRAN-3D is comparable to any of the other participating codes.

RETRAN-3D is approved for main steam line break analyses subject to the following conditions. Thermal-hydraulic effects can have a large impact on the cross section evaluation and thus on the resulting power distribution and magnitude. Therefore, the licensee must justify the primary side nodalization for mixing in the vessel and core. The licensee must also evaluate the uncertainties in the modeling.

2. *There is no source term in the neutronics and the maximum number of energy groups is two. The space-time options assume 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.*

Staff position: The basic models in RETRAN-3D are unchanged and, therefore, this condition of use applies.

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

Staff position: As noted previously in this report, boron transport is handled as a "contaminant" by the "general transport model." This model uses first order accurate upwind difference scheme with an implicit temporal differencing. This approach is well known for being highly diffusive, especially if the Courant limit is exceeded. Since RETRAN-3D has the same model as RETRAN-02 MOD003 and subsequent versions that have been approved for use, the RETRAN-3D model is also approved with the caveat that the potential to produce misleading results with this scheme necessitates careful review of the results for any case where boron transport/dilution is important.

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

Staff position: The control bank limitation is applied only to the one-dimensional kinetics model. The staff agrees that the 3-dimensional kinetics model need not be restricted in this way.

5. *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.*

Staff position: The basic models in RETRAN-3D are unchanged and, therefore, this condition of use applies. However, since RETRAN-3D is not being reviewed for loss-of-coolant accident analysis, where core uncovery and heatup are significant, this condition does not occur in the transients for which application of RETRAN-3D has been reviewed.

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

Staff position: The RETRAN-3D five equation model permits thermal-hydraulic nonequilibrium between the liquid and vapor phases. While it allows subcooled liquid and saturated steam to be concurrently present, it does not account for subcooled liquid and superheated vapor being concurrently present. Use of the code is not approved for LOCA. Also, the user must be aware of this limitation and avoid conditions which will place subcooled liquid and superheated vapor in contact.

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

Staff position: The basic model in RETRAN-3D is unchanged and, therefore, this comment still applies.

8. *Further justification is required for the use of the homogeneous slip options with BWRs.*

Staff position: RETRAN-3D has five slip equation options for the user to choose from, three of which are retained from RETRAN-02 for compatibility. The recommended model options are based on the Chexal-Lellouche drift flux correlation. The first is the algebraic slip model, which is approved for use with BWR bundle geometry as given in condition (9). The second is a form of the dynamic slip model that uses the Chexal-Lellouche drift flux correlation to evaluate the interfacial friction approved in condition (10). The user must justify the use of any other slip options.

9. *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 (16) are met.*

Staff position: The Chexal-Lellouche drift flux model has been used in comparisons with FRIGG-2 and FRIGG-4 void fraction data and is acceptable for use in BWR bundle geometry.

10. *The profile effect on the interphase drag (among all the profile effects) is neglected in the dynamic slip option. Form loss is also neglected for the slip velocity. For the acceptability of these approximations refer to (17).*

Staff position: Form loss terms have been included in the RETRAN-3D dynamic slip model. The Taugl form of the dynamic slip equation also includes profile effects in the interphase drag model. These RETRAN-3D model improvements adequately address the concerns and the model is approved for use when the Chexal-Lellouche model is used to compute the interphase friction. Approval is subject to the conditions given in (16) for the Chexal-Lellouche drift flux correlation. Users must justify use of any other dynamic slip option.

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

<u>Staff position</u>: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use still applies.

12. *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 noncondensables.*

<u>Staff position</u>: RETRAN-3D has been extended to include a general noncondensable gas capability which resolves the original concern. However, the noncondensable gas flow model is approved for use subject to the following restriction.

As noted in Section III.3.0 of the RETRAN-3D Theory Manual (Reference 4), none of the models available for calculating critical flow are appropriate when noncondensable gases are present. Consequently, the code automatically bypasses the critical flow model when noncondensable gases are present in a junction. Users must confirm that noncondensable flows do not exceed appropriate critical flow values or justify use of values that may exceed critical flow values.

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

<u>Staff position</u>: For enthalpies less than approximately 820 Btu/lbm, the difference between the ASME and RETRAN-3D curve fit values of the specific volume range from less than 0.2 percent to approximately 1.3 percent for pressures ranging from 0.1 to 6,000 psia. Further, for enthalpies greater than 820 Btu/lbm and pressures greater than 4200 psia, the differences in specific volume are also less than 1.0 percent. RETRAN-3D is approved for use with PWR ATWS analyses where the peak pressure resides in the regions described above.

For enthalpies greater than 820 Btu/lbm and pressures between 3200 and 4200 psia, the differences in specific volume increase as the enthalpy increases and the pressure decreases. The maximum error of approximately 3.8 percent occurs at the critical point. PWR ATWS analysis using RETRAN-3D in this region will require additional justification that the difference in specific volume does not adversely affect the calculation of the peak pressure.

14. *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.*

<u>Staff position</u>: RETRAN-3D contains both the "forced convection option" contained in RETRAN-02 which is the basis for this restriction, and a second option referred to as the "combination heat transfer map." If the first option is chosen, the "forced convection option," approval is granted only for use in pre-CHF and single-phase heat transfer regimes. If the second option is chosen, the "combination heat transfer map," then there are no discontinuities between successive heat transfer regimes and the appropriate heat transfer value should result. Therefore, the combination heat transfer option is approved for use.

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

Staff position: The Beattie two-phase multiplier has been removed from RETRAN-3D. The Govier horizontal flow map has been added to supplement the Bennett map for vertical flow and is acceptable.

16. *No separate effects comparison have been presented for the algebraic slip option and it would be prudent to request comparisons with the FRIGG tests before the approval of the algebraic slip option.*

Staff position: The algebraic slip option has been modified to include the Chexal-Lellouche drift flux model. Use of the Chexal-Lellouche drift flux model for BWR and PWR applications within the range of conditions covered by the steam-water database used to develop and validate the model is approved. The model has been qualified with data from a number of steady-state and two-component tests. While the small dimensions of the fuel assembly are covered, as noted previously in this safety evaluation, the data for large pipe diameters, such as reactor coolant system pipes, are not extensive and use of the Chexal-Lellouche model will need justification. Assessment work indicates that the model tends to underpredict the void profile in the range of 12 to 17 MPa. In addition, the accuracy of the model in the range of 7.5 to 10 Mpa, which covers BWR ATWS conditions, has not been fully demonstrated. Results of analyses using the model in these ranges must be carefully reviewed.

The Chexal-Lellouche correlation cannot be used in situations where CCFL is important unless validation for appropriate geometry and expected flow conditions is provided.

17. *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.*

Staff position: Assessment analyses (Reference 4), have shown that "the issues concerning the Schrock-Grossman round tube data comparisons" (actually the Bennett round tube data) are due to early prediction of CHF, which is nearly independent of the slip model used. Since the issue raised in the limitation is not related to the dynamic slip model, the limitation is considered to be resolved. The dynamic slip model is approved for use as given in condition (10).

18. *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.*

Staff position: The concern raised in this limitation of use is partially resolved in RETRAN-3D. Wall heat transfer can be included in the RETRAN-3D pressurizer model. Including wall heat transfer resolves this concern.

While the model does not directly account for thermal stratification, its effects can be included by use of normal nodes below the pressurizer volume. The user will have to justify the lack of thermal stratification or the use of normal nodes below the pressurizer should there be an indication that it would be important in the analysis.

The mixture and two-region energy equations are consistent for the implicit solution method where the mixture energy equation is used with the vapor-region energy equation. This eliminates inconsistency between the two-region and mixture energy equations and the concern regarding a potential drift in the region energies.

The staff notes that when a pressurizer fills or drains, a single region exists for which the normal pressure equation of state is used. Lack of numerical discontinuities in validation analyses of filling and draining pressurizers indicates that the model is functioning properly. It is the responsibility of the code user to justify any numerical discontinuity in the pressurizer during a filling or draining event.

The pressurizer model has options that require user-supplied parameters. Users must provide justification for these model parameters.

19. *The nonmechanistic separator model assumes quasistatics (time constant approximately few tenths of seconds) and uses GE BWR6 carryover/carryunder curves for default values. Use of 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 quadrant is limited and those quadrants should be avoided. Attenuation of pressure waves at low flow/low quality conditions are not simulated well. Specific applications to BWR pressurization transients under those conditions should be justified.*

Staff position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

20. *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 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.*

Staff position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

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

Staff position: Subsequent revisions of RETRAN-02 addressed this limitation. Since RETRAN-3D has the same model as RETRAN-02 MOD003, and subsequent versions, their acceptance applies to RETRAN-3D.

22. *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.*

Staff position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

23. *The subcooled void model is a nonmechanistic profile fit using a modification of EPRI recommendations 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 (MOD003) modifications.*

Staff position: The profile blending algorithm approved for RETRAN-02 MOD003 is used in RETRAN-3D, therefore this condition has been satisfied.

24. *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.*

Staff position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. However, the layering effects encountered in RETRAN-02 can be eliminated using the RETRAN-3D stack model. This partially resolves the concern by resolving the layering limitation through use of the stack model.

25. *The transport delay model should be restricted to situations with a dominant flow direction.*

Staff position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. The appropriate application of the model is for one-dimensional flow. The user will have to justify use of this option in the absence of a dominant flow direction.

26. *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.*

Staff position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

27. *Phase separation and heat addition cannot be treated simultaneously in the enthalpy transport model. For heat addition with multidirectional, multifunction volumes the enthalpy transport model should not be used without further justification. Approval of this model will require submittal of the new manual (MOD003) modifications.*

Staff position: A number of the simplifying assumptions in the RETRAN-02 enthalpy transport model have been eliminated in RETRAN-3D which now allows multiple inlet and outlet flows and eliminates the simplifying assumptions related to mass distribution and pressure change effects. This condition has been adequately addressed.

28. *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 condition 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.*

Staff position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

29. *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 justify steady-state heat balances. These adjustments should be reviewed on a specific application basis.*

Staff position: The over specified condition is identified by the RETRAN-3D steady-state input checking, resolving the concern regarding ill-posed data. The user must still run null transients to ensure that unwanted control or trip actions are not affecting the transient solution.

RETRAN-3D has available a low power steady-state steam generator initialization option that eliminates the heat conductor area change used in the RETRAN-02 initialization scheme. When this option is used, no adjustments are made to the heat transfer area and this specific concern is resolved. However, either the pressure or temperature is adjusted on the secondary side. These adjustments should be reviewed by the user on a specific application basis. The low power steady-state initialization option is approved for use.

30. *Justification of the extrapolation of FRIGG data or other data to secondary-side conditions for PWRs should be provided. Transient analysis of the secondary side must be substantiated. For any transients in which two-phase flow is encountered in the primary all the two-phase flow models must be justified.*

Staff position: The Chexal-Lellouche correlation is approved for use with PWR applications as stated in conditions (10) and (16). The user must justify choosing any other two-phase flow correlation.

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

Staff position: The pressurizer model is approved for use with filling and draining events as given in condition (18).

32. *Transients which involve three-dimensional space-time effects such as rod ejection transients would have to be justified on a conservative basis.*

<u>Staff position</u>: The 3-dimensional kinetics model, as noted in limitation 1 above, satisfies this limitation.

33. *Transients from subcritical, such as those associated with reactivity anomalies should not be run.*

<u>Staff position</u>: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

34. *Transients where boron injection is important, such as steamline break will require separate justification for the user-specified boron transport model.*

<u>Staff position</u>: The generalized transport model was added to RETRAN-3D to provide the capability to track materials such as boron. Specific application of the model to steam line break transients must be justified by the user. The model is approved for use as given in condition (3).

35. *For transients where mixing and cross flow are important, the use of various cross flow loss coefficients has to be justified on a conservative basis.*

<u>Staff position</u>: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

36. *ATWS events will require additional submittals.*

<u>Staff position</u>: RETRAN-3D is approved for PWR ATWS analyses as given in condition (13).

37. *For PWR transients where the pressurizer goes solid or completely drains, the pressurizer behavior will require comparison against real plant or appropriate experimental data.*

<u>Staff position</u>: The pressurizer model is approved for use with filling and draining events as noted in the discussion of conditions (18) and (31).

38. *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.*

<u>Staff position</u>: The use of slip models for PWR applications is approved for use as given in conditions (16) and (30).

39. *BWR transients were asymmetry leads to reverse jet pump flow, such as the one recirculation pump trip, should be avoided.*

<u>Staff position</u>: As noted in the discussion of condition (21), this is resolved.

6.0 RETRAN-3D USE IN A RETRAN-02 MODE

During the RETRAN-3D review, the applicant suggested an approval of use of RETRAN-3D as a substitute for RETRAN-02 when operated in that mode. The staff has determined that it is not possible to use RETRAN-3D in a pure RETRAN-02 mode. The code's numerical solution scheme and various models have been changed so that there is no exact RETRAN-02 substitution that can be performed. However, the code can be used in a near RETRAN-02 mode provided that the user carefully selects models and options that reduce the divergence from those not available to the RETRAN-02 user.

While functionally equivalent to RETRAN-02, RETRAN-3D is more robust. The following models are always active when using RETRAN-3D:

- Improved transient numerical solution (fully implicit solution of the balance equations, component models and source terms are linearized).
- Improvements to the time-step selection logic.
- Improved water property curve fits.

Other model options have been improved with the improvements being active when the particular option is selected in an input model. For these options, the RETRAN-02 model was replaced by the improved model and there is no backward compatibility option. Consequently, the following improvements, if selected by the user, may be used for RETRAN-02 mode analyses:

- Fully implicit steady-state solution,
- Implicit pressurizer solution,
- Wall friction model revised to use the Colebrook equation, allowing consideration of wall roughness rather than assuming smooth pipe,
- Control system solution revised to solve a coupled system of equations using a Gauss-Seidel method rather than the single pass marching scheme,
- Enthalpy transport model revised by eliminating several simplifying assumptions,
- Improved dynamic slip formulation adding form losses,
- Improved countercurrent flow junction properties,
- Implicit solution of the heat conduction equation,
- Combined heat transfer map updated with an improved set of heat transfer correlations and smoothed transitions, and
- Wall friction and hydrostatic head losses included in critical flow pressure.

The new steady-state option available for initializing models with steam generators makes some problems easier to initialize. The low power steam generator steady-state option can be used with RETRAN-02 mode analyses.

A RETRAN-02 mode model must not use any of the new RETRAN-3D features such as:

- Generalized laminar friction model,
- Dynamic gap conductance model,
- Accumulator model,
- Dynamic flow regime model,
- New control blocks added to improve functionality,
- Govier horizontal flow regime map and stratified flow friction model,
- Chexal-Lellouche drift flux model,
- Method of characteristics enthalpy option,
- Noncondensable gas flow model,
- 3D kinetics, and
- 5-equation nonequilibrium model,

40. *Organizations with NRC-approved RETRAN-02 methodologies can use the RETRAN-3D code in the RETRAN-02 mode without additional NRC approval, provided that none of the new RETRAN-3D models listed in the definition are used. Organizations with NRC-approved RETRAN-02 methodologies must obtain NRC approval prior to applying any of the new RETRAN-3D models listed above for UFSAR Chapter 15 licensing basis applications. Organizations without NRC-approved RETRAN-02 methologies must obtain NRC approval for such methodologies or a specific application before applying the RETRAN-02 code or the RETRAN-3D code for UFSAR Chapter 15 licensing basis applications. Generic Letter 83-11 provides additional guidance in this area. Licensees who specifically reference RETRAN-02 in their technical specifications will have to request a Technical Specification change to use RETRAN-3D.*

7.0 <u>ADDITIONAL CONDITIONS OF USE</u>

BWR ATWS

RETRAN may be used for BWR ATWS subject to the following restrictions:

41. *The licensee must validate the chosen void model over the range of pressure, channel inlet flow, and inlet subcooling encountered during the transient that are outside the range of conditions for which assessment is available. Furthermore, the licensee should validate the choice of steam separator model and evaluate its use relative to steam separator performance data relevant to the conditions present during the ATWS simulation. The licensee must also evaluate the uncertainties in the modeling. See Condition (16) and the Staff Position for related information.*

Heat, Mass, and Momentum Transfer

42. *The RETRAN-3D five-equation, or nonequilibrium, model uses flow regime maps and flow pattern dependent heat transfer and interfacial area models to simulate the heat and mass transfer processes between phases. A licensee wishing to apply the five-equation model will have to justify its use outside areas of operation where assessment has been documented. This may include either separate effects or integral systems assessment that cover the range of conditions encountered by the application of interest. An assessment of the uncertainties must also be provided. The model is approved subject to these conditions.*

43. *Assessment performed in support of use of RETRAN-3D must also address consistency between the RETRAN-3D calculations and any auxiliary calculations that are a part of the overall methodology, such as, departure from nucleate boiling or critical power ratio.*

User Guidelines and User Qualification

44. *The staff concludes that the lack of a detailed RETRAN-3D specific user guideline document mandates a statement on the user's experience and qualification with the code when analyses are submitted in support of licensing actions. This statement is expected to be consistent with the guidance of Generic Letter 83-11.*

Code Assessment

45. *Assessment of the RETRAN-3D code for the models not explicitly approved in this safety evaluation will be the responsibility of the licensee or applicant. In addition, application of the RETRAN-02 or RETRAN-3D codes for best estimate analysis of UFSAR Chapter 15 licensing basis events may require additional code and model assessment, and an evaluation of uncertainties to assure accurate prediction of best estimate response. This condition is based on the absence, in the best estimate analysis approach, of the conservative assumptions in traditional UFSAR Chapter 15 licensing basis analyses. For each use of RETRAN-3D in a licensing calculation, it will be necessary for a valid approach to assessment to be submitted, which is expected to include a PIRT for each use of the code and the appropriate assessment cases and their results. The scope of the PIRT and validation/assessment will be commensurate with the complexity of the application.*

8.0 CONCLUSIONS

Development of RETRAN-3D is a significant advancement in analysis tools versus RETRAN-02. The RETRAN-3D code, however, due to its flexibility is a very complex tool to use. The degree to which the user can affect calculational results necessitates stringent controls over the training of the user and close examination of the modeling, assumptions and options used.

RETRAN-3D was submitted for staff review to be a code applicable to all Standard Review and Format Chapter 15 events except the loss-of-coolant accidents. As such it would be expected that broad and extensive assessment of the code would be provided addressing all models and correlations, a broad spectrum of separate effects tests, and a wide range of integral systems tests and actual plant data. This would also be expected to include a wide range of plant types and configurations. The lack of sufficient code assessment makes it incumbent upon the individual licensee or applicant to provide appropriate assessment for each use and application of the code. In addition, the user will have to provide verification that the code is used within the proper range of each and every correlation and model selected.

As a condition on the code used in a "RETRAN-02 mode," it will be necessary to provide adequate demonstration that the code is actually used in that mode where possible and that the only divergences are due to mandatory use of modified numerics and models. It will be essential that demonstration be provided that margins similar to those that would be obtained with RETRAN-02 have been obtained.

The addition of 3-dimensional neutron kinetics is a significant advancement in the code's capability. The performance of the kinetics models has been demonstrated to be consistent with that of other similar methodologies. The models have been compared with other methodologies by the staff and assessed by comparison with existing prompt critical experimental data. The staff concludes that use of the RETRAN-3D kinetics models is acceptable for transients such as the PWR rod ejection and BWR rod drop. In the case of the main steam line break in the PWR, the results are comparable to those obtained with lower order kinetics models since the transient is driven by the thermal-hydraulic conditions. Approval is not given for use of the code for the BWR instability calculation.

The staff believes that establishment of a RETRAN-3D peer review process by the RETRAN-3D Maintenance Group is a positive step in alleviation of staff concerns about user experience and consistency and uniformity in application of the code.

The staff review of RETRAN-02 resulted in a list of 39 limitations and conditions of use. The review of RETRAN-3D results in a reduction of that list, but does not eliminate all of the conditions. Many of the conditions still apply to RETRAN-3D and are, therefore, still in force. The forty-five conditions and limitations discussed above have been agreed to by EPRI and the RETRAN-3D Maintenance Group in a letter dated December 13, 2000.

The Chexal-Lellouche drift flux model appears to be an improvement over the previous RETRAN drift flux models based on the limited assessment provided. A licensee wishing to use the correlation will have to assure its use is in conformance with the conditions noted in Condition 16 above. Use outside the noted range of acceptance, or where CCFL is important, will necessitate that an applicant provide assessment over the full range of conditions encountered during the application of interest. Since the correlation is purely empirical in nature the assessment must be provided for full scale in all variables of interest. An assessment of the uncertainties must also be provided.

Final acceptance of RETRAN-3D for licensing basis calculations depends upon successful adherence to the conditions and limitations on use discussed in this report. The RETRAN-3D documentation is expected to be republished with noted errors corrected and this safety evaluation included. The staff will audit the use of the RETRAN-3D code to verify that the conditions and limitations on use are followed.

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Attachment: Appendix A

Principal Contributor: R. Landry

Date: January 25, 2001

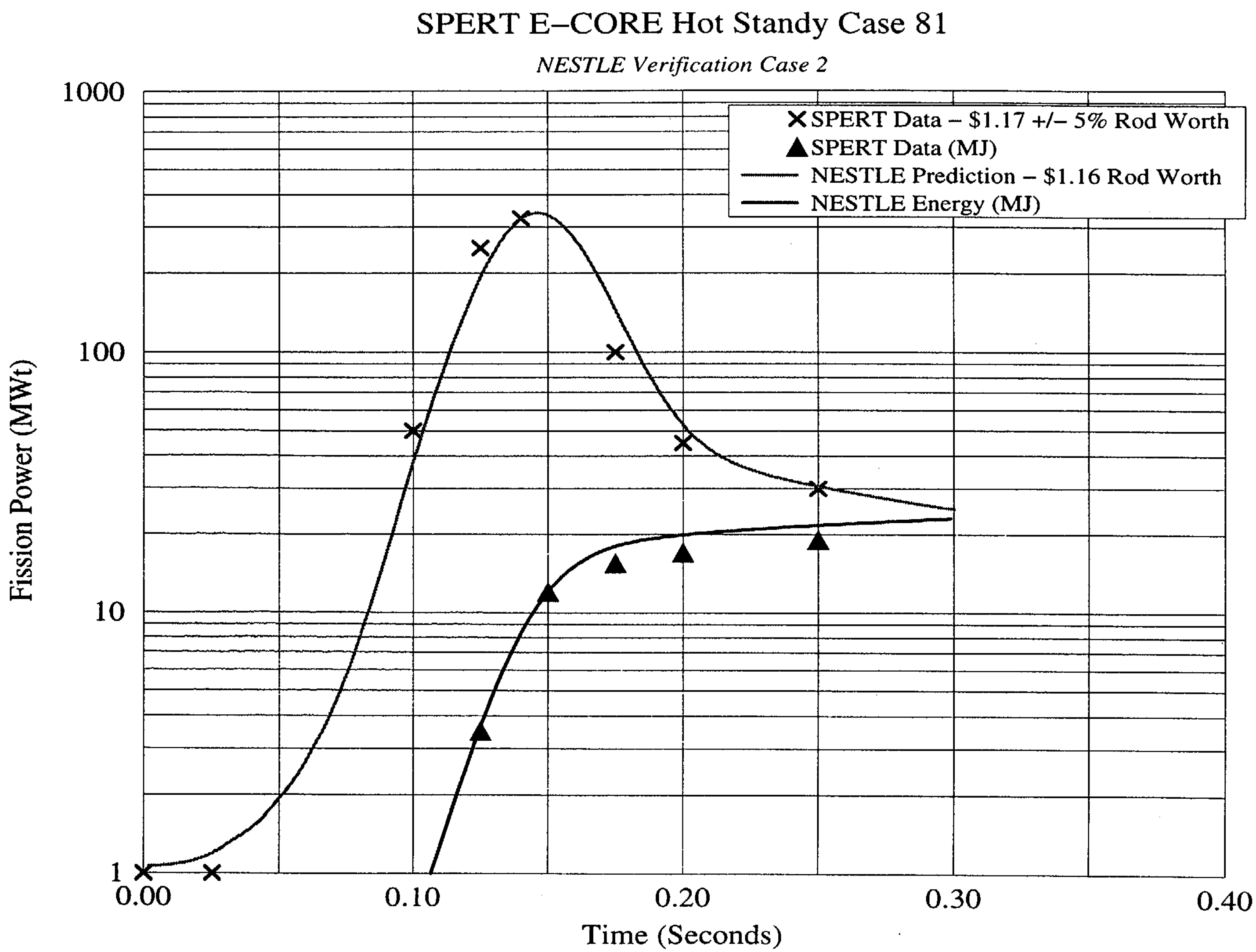

Figure 1 NESTLE Prediction of Spert III E-Core Test 81 - Hot Zero Power Super Prompt Critical Reactivity Excursion

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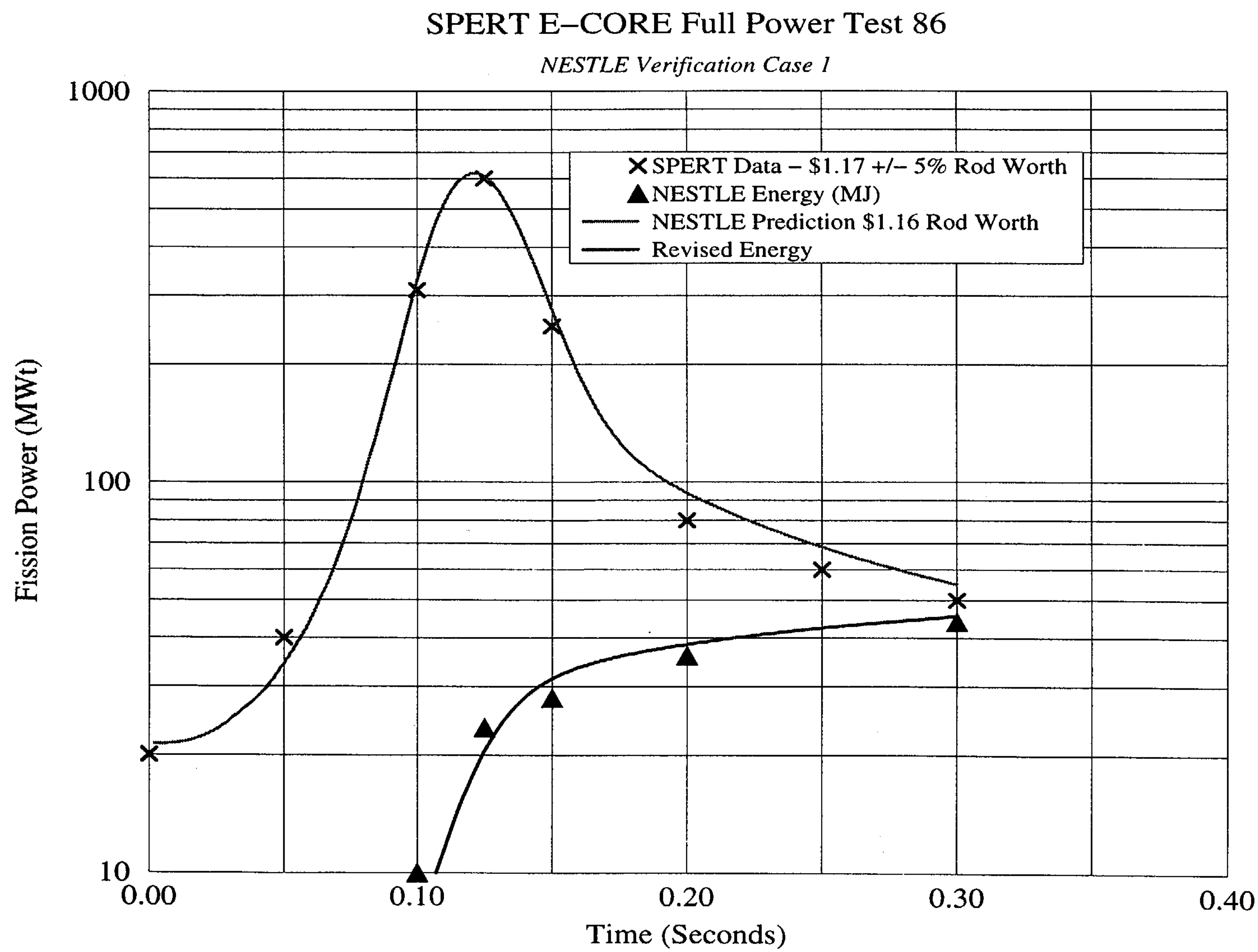

Figure 2 NESTLE Prediction of Spert III E-Core Test 86 - Full Power Super Prompt Critical Reactivity Excursion

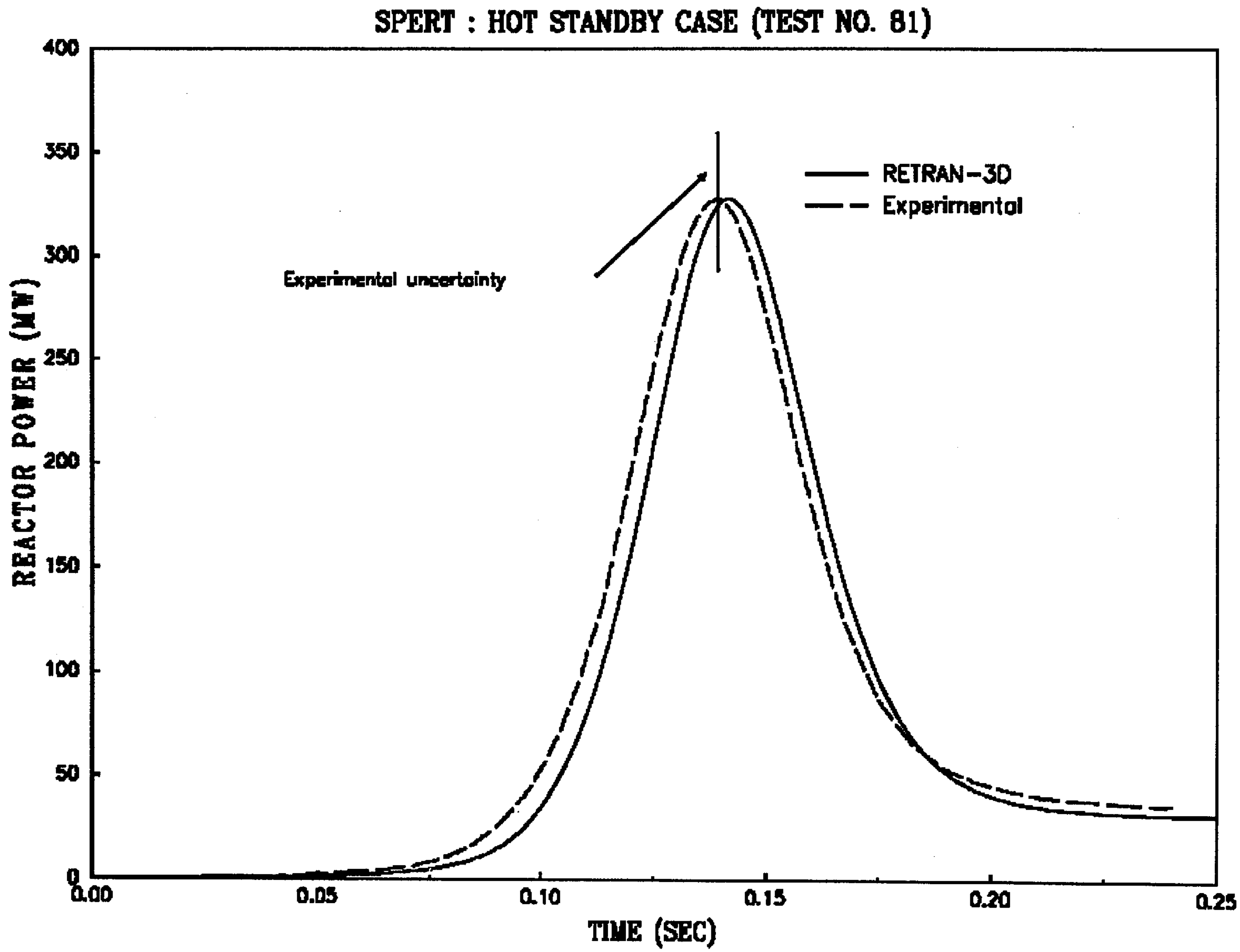

Figure 3 RETRAN-3D Predicted Power for SPERT III E-Core Test 81

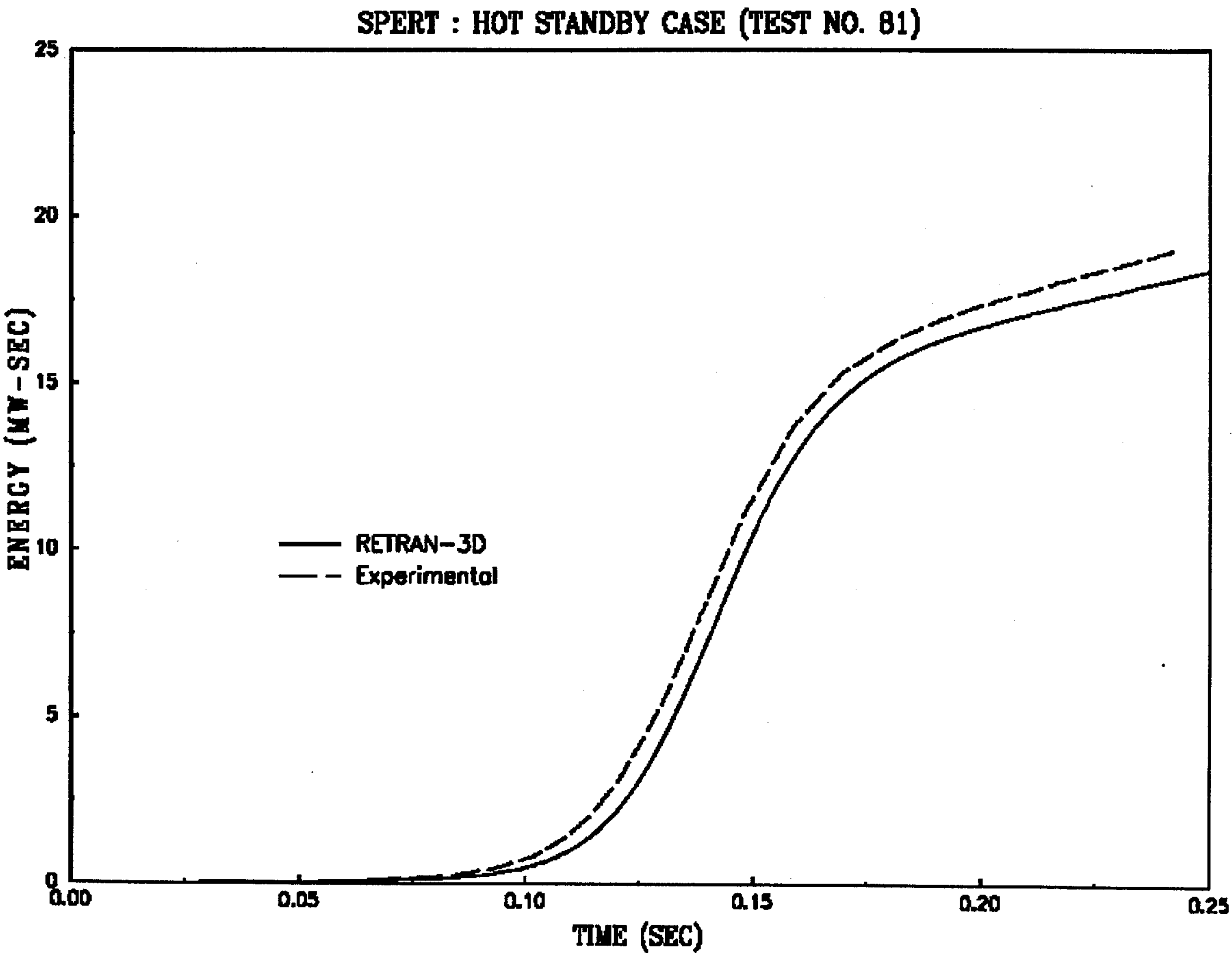

Figure 4 RETRAN-3D Predicted Energy Deposition for SPERT III E-Core Test 81

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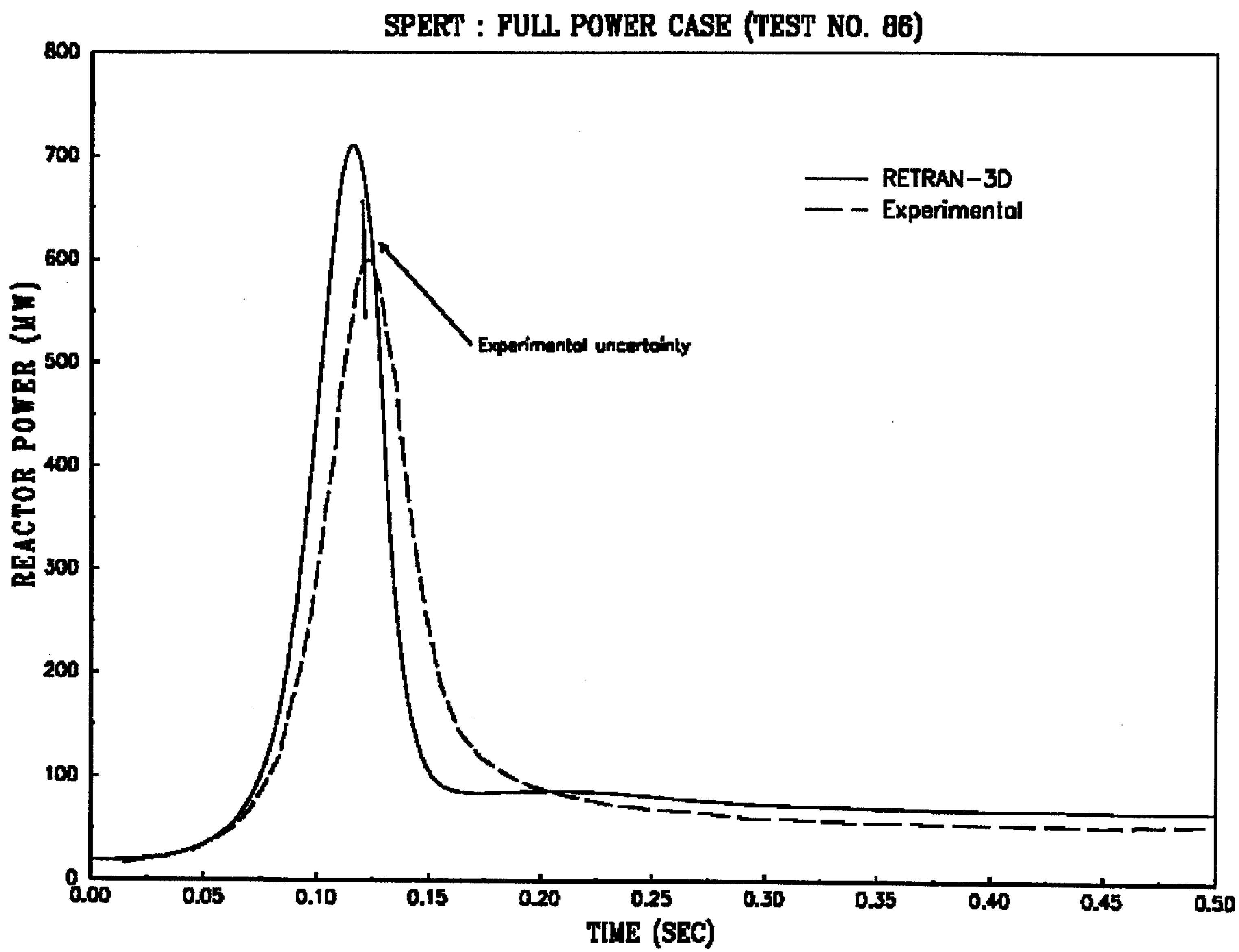

Figure 5 RETRAN-3D Predicted Power for SPERT II E-Core Test 86

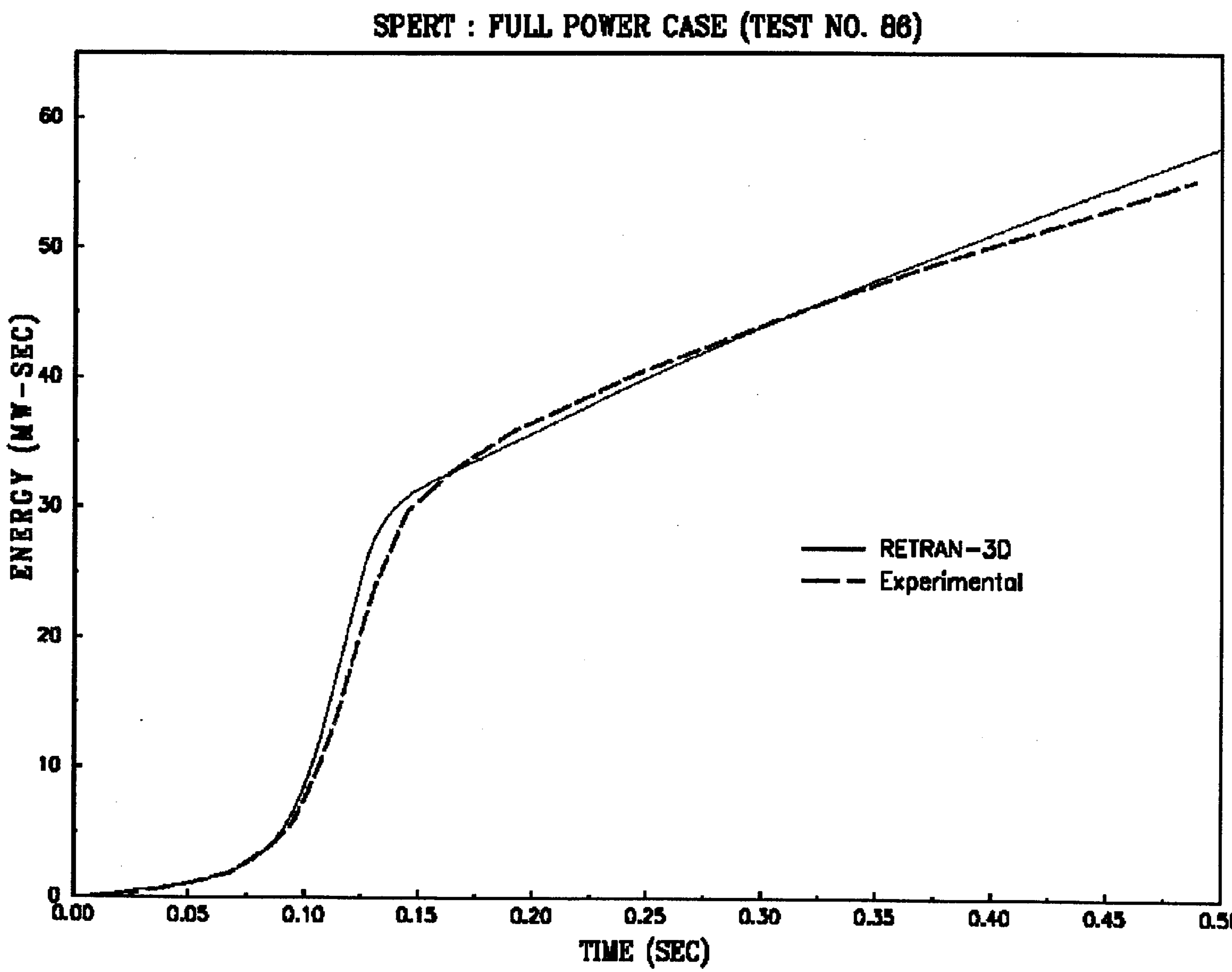

Figure 6 RETRAN-3D Predicted Energy Deposition for SPERT III E-Core Test 86

Appendix A

AP-600 CORE EVALUATED WITH BOTH DIFFUSION THEORY AND TRANSPORT THEORY

INTRODUCTION

The reference AP-600 core was used as a sample problem with which to compare diffusion theory methods and transport theory base methods. The comparison was based on the predicted flux and eigenvalue. The staff used the NESTLE code as the diffusion theory solver and TORT as the transport theory code. This evaluation is by extension by an evaluation of RETRAN-3D because the staff has demonstrated that RETRAN-3D and NESTLE yield essentially identical answers. The intent of this exercise is to assess the capabilities of diffusion theory methods relative to higher order transport methods. For the purposes of this problem, the transport theory results are taken to be the reference (or the "correct") answers.

DESCRIPTION OF PROBLEM

The reference AP-600 core (Reference A.1) was chosen as the core for analysis. This core is attractive for this purpose because it is unburned. The core consists of three different fuel types and has no burnable absorbers. Cross sections were prepared with a pre-release version of sas2d (under development for SCALE-5) which is a two-dimensional S_N solver using the method of characteristics. It also has the unique capability to develop the 5th Order Legendre Polynomial expansion coefficients which are needed for the TORT calculations. Both models used the same cross sections with the obvious exception that TORT used a 5th order Legendre Polynomial expansion to represent the scattering kernel. All other inputs including nodalization were identical. Both rodded and unrodded cases were studied. The control rod was a B_4C control cluster. The figure of merit for this study was the axial flux distribution. The input files are included at the end of this Appendix.

RESULTS AND DISCUSSION

Two plots were created and presented as Figures A.1 and A.2. Figure A.1 shows the axial power distribution averaged for all of the fuel. As expected, the flux follows the familiar cosine shape and the fast (or group 1) flux has a larger magnitude. The TORT and NESTLE predictions are in excellent agreement. Figure A.2 shows the axial flux profiles for the assembly where the B_4C control rod is inserted. This case is much more challenging due to the presence of the strong absorber, but the diffusion theory code still predicts acceptable results with a maximum error of 6 percent for the fast flux. This is considered excellent given that theoretically diffusion theory breaks down in the presence of strong absorbers such as B_4C.

This study provides the staff with some insight into the accuracy of diffusion theory methods relative to higher order transport theory based codes. The two figures presented show that for the reactor which was studied, diffusion theory can accurately predict flux profiles (and, therefore, power distributions) compared to transport theory methods for both rodded and unrodded cases. Although limited in its scope, this study is further confirmation that diffusion theory is adequate for reactor analyses. As a final note, the eigenvalues compared to within 0.6

%Δk/k (approximately 600 pcm) which is considered acceptable for this study. The nodal method employed in TORT is designed for accurate prediction of surface fluxes and is not rigorously validated for eigenvalue predictions so some difference in the eigenvalue was expected.

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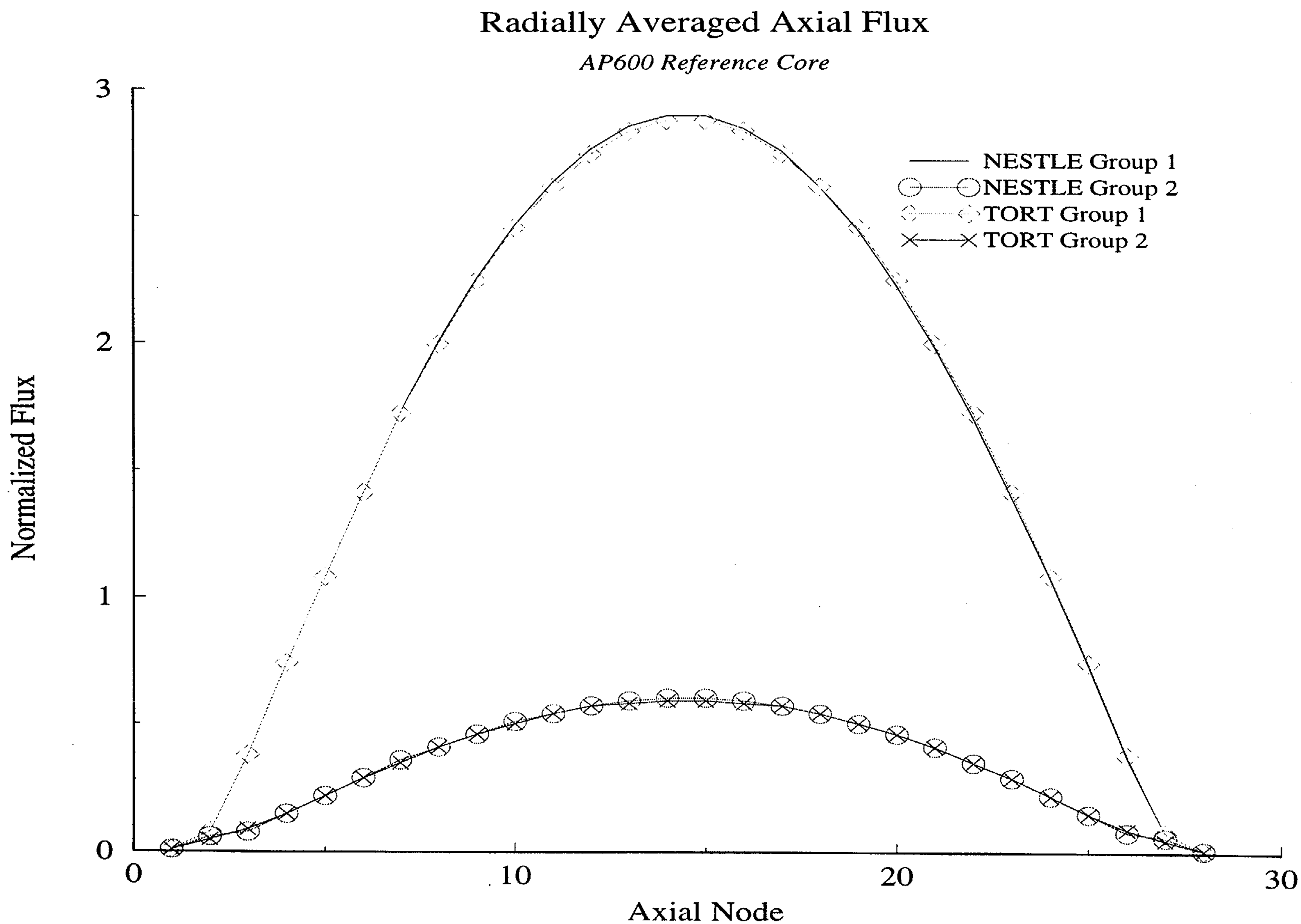


Figure A.1 Radially Averaged Axial Power Distributions

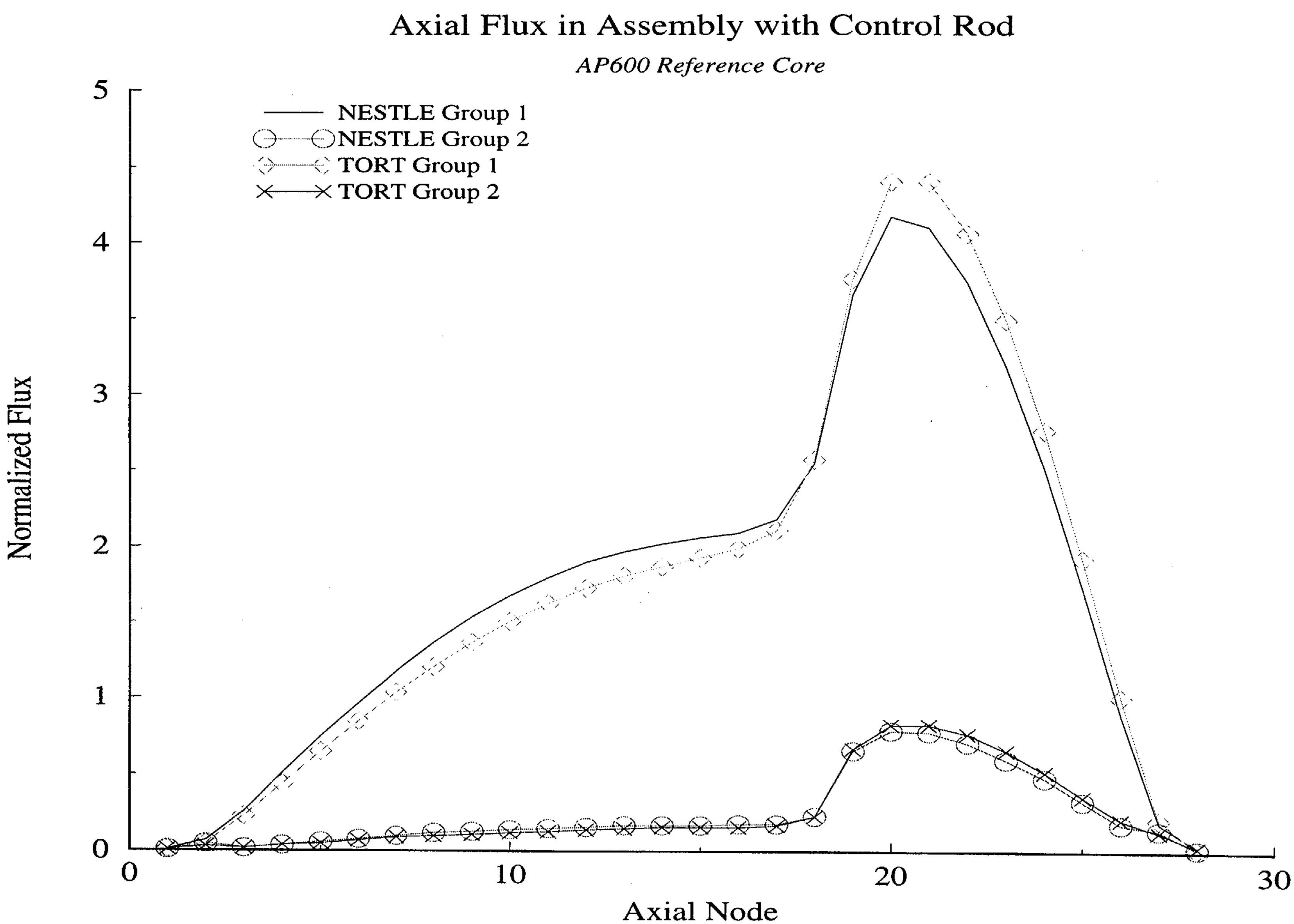


Figure A.2 Axial Power Distribution for Assembly with Control Rod Inserted

APPENDIX 9
RETRAN-02 to RETRAN-3D Transition Qualification

Table of Contents

1.0 Introduction

Dominion has conducted analyses to support the transition from the RETRAN-02 to RETRAN-3D computer code for non-LOCA transient analyses. In the Safety Evaluation Report (SER) [Reference 1] for RETRAN-3D, the Nuclear Regulatory Commission (NRC) provided a condition of use allowing licensees with approved RETRAN-02 methodologies to use RETRAN-3D in the RETRAN-02 mode without seeking NRC approval. RETRAN-02 mode is defined by the NRC as the careful selection of models and options to reduce divergence from those not available to users of RETRAN-02. The RETRAN-3D SER supplied a list of acceptable options that may be used in RETRAN-02 mode and those that are not acceptable. Dominion's RETRAN models have been converted to RETRAN-3D format with careful attention to ensure that no options deemed unacceptable by the NRC for RETRAN-02 mode have been used. Section 2.0 describes the models used for the analyses.

Analyses have been performed by Dominion to compare results calculated by the RETRAN-02 computer code and the RETRAN-3D computer code in RETRAN-02 mode. The analyses presented in this report are a representative sample of transients to ensure that all aspects of the code and models are considered. Section 3.0 contains discussion of the analysis methods and results. Section 4.0 contains conclusions.

2.0 Model Description

The Dominion RETRAN-3D models for Surry Power Station and North Anna Power Station are developed from the existing RETRAN-02 production models. The process by which the existing RETRAN-02 decks are converted to RETRAN-3D format involves changes to the card format. In all cases the decks are converted to conform to the requirements of RETRAN-3D in RETRAN-02 mode. To the extent possible, the same model options were chosen as in RETRAN-02. Due to some enhancements of the RETRAN-02 mode models in RETRAN-3D as discussed in the RETRAN-3D SER, the models may differ in performance slightly. However, no significant differences are expected.

The changes made to the RETRAN-3D input cards, other than formatting, are to ensure that the models initialize to the appropriate plant initial conditions. The energy equation form has changed slightly in RETRAN-3D in that the potential and kinetic energy terms have been dropped. This can create small differences in computed thermal hydraulic initial conditions. In single-phase liquid regions, the changes are generally quite small. A very small adjustment in the enthalpy specified on the primary side may be required to obtain the exact average loop temperature. More significant changes may be noticed on the secondary side where the kinetic energy term for the steam flow paths may be larger. The following changes have been made to the base model and overlay decks to ensure similar initial conditions to RETRAN-02:

- The feedwater enthalpy changes several Btu/lbm since dropping the kinetic energy term (particularly in the steam path) changes the energy balance. Consequently, the initial feedwater enthalpy is changed in the control system to ensure the feedwater enthalpy control system is properly initialized.
- Cold leg enthalpy is changed slightly in the base model to get the desired average loop temperature.
- In the multinode steam generator overlay modules, the steam generator pressure changes slightly to minimize steam generator tube heat transfer area adjustment and the mixture level is also adjusted slightly to get the desired initial steam generator mass. Consequently, the pressure drop versus steam generator narrow-range level is also adjusted slightly to obtain the correct initial steam generator narrow-range level.

Initial conditions of some control blocks are changed to guarantee consistent transition from steady-state initialization to the transient. As described above, there are slight differences in the thermal-hydraulic initialization because of the revised energy equation. Due to these changes in the energy equation, changes are made to the control blocks for feedwater and cold leg enthalpy. This subsequently requires small adjustments to the reactivity feedback control blocks to ensure the model initializes with reactivity close to zero.

The revised numerical solution scheme in RETRAN-3D requires that at least two normal junctions (non-fill junctions) must be present in a model. The original hot spot model in RETRAN-02 has only one non-fill junction. Therefore, an additional dummy volume and junction is added. Also included is a valve in the dummy junction.

The RETRAN hot spot model uses an enthalpy ramp to induce a steam environment in the channel. The speed of the enthalpy ramp causes errors to be encountered in the CONTRL subroutine of RETRAN-3D. This issue is a numerical error due to the calculation scheme used in RETRAN-3D. To alleviate this issue, a small change was made to the time step during the time period the error is encountered.

3.0 Method of Analysis

To ensure that there is no effect on current licensing basis analyses, a series of benchmark cases are run to compare RETRAN-02 results to those obtained with RETRAN-3D in RETRAN-02 mode. Divergences are attributed to one of the various modeling differences between RETRAN-02 and RETRAN-3D.

A set of transient cases were selected as representative of the various types of UFSAR transient that will demonstrate the acceptability of RETRAN-3D. An increase in secondary side heat removal, loss of heat sink, reactivity excursion, and loss of flow accident are selected. Specific events are also chosen based upon their ability to thoroughly exercise the code (i.e. length, severity, limiting parameters). These cases represent a broad variation in behavior and demonstrate the ability to model key phenomena for a range of transient responses. Each selected transient is run for only a single station, as North Anna Power Station (NAPS) and Surry Power Station (SPS) are similar enough to draw appropriate conclusions from. A listing of the selected transients is provided below.

Table 3.0-1 Transient Events for Qualification

Transient	**Station Model Used**
Loss of Normal Feedwater/Loss of Offsite Power to the Station Auxiliaries	NAPS
Loss of Load/Turbine Trip	NAPS
Main Steamline Break	NAPS
Rod Withdrawal from Subcritical	NAPS
Locked Rotor/Sheared Shaft	SPS

For the loss of heat sink event, two cases are selected. The first case is Loss of Normal Feedwater/Loss of Offsite Power to the Station Auxiliaries (LONF/LOAC) event for NAPS. This is the limiting event propagation transient. The analysis must demonstrate that the Reactor Coolant System (RCS) pressurizer does not overfill or relieve liquid water through the safety or relief valves. This may cause valve damage, leakage, and thus propagate from a Condition II transient to a Condition III transient. In addition to being limiting for event propagation, this case is selected due to its runtime and its propensity for exercising the various relief valves in the nuclear steam supply system. The case runs for 10000 seconds and as the main steam system over pressurizes, steam is released through the main steam safety valves. This can happen multiple times during the course of the transient. This presents the opportunity for observation of divergences in the behavior of control systems and valves.

The second loss of heat sink is the loss of load/turbine trip (LOL/TT) event for NAPS. This is the limiting RCS overpressure transient. With the loss of heat removal, the RCS both heats up and pressurizes causing a spike in RCS pressure. The pressure transient is terminated via the high pressurizer pressure trip setpoint with pressurizer steam relieved through the pressurizer safety valves. The analysis must demonstrate that RCS peak pressure does not exceed the maximum pressure limit.

For the increase in secondary side heat removal, the limiting Main Steamline Break (MSLB) event for NAPS was chosen. The MSLB event provides an opportunity to observe the behavior of RETRAN-3D with multiple phenomenons occurring in the RCS and secondary side. The excess steam removal causes both a depressurization and cool down of the RCS. This contributes to reactivity feedback effects and the possibility of pressurizer drain out. A separate MSLB module was developed by Dominion to model the events of the MSLB accident that are extraordinary to the normal non-LOCA transients. This model includes a core containing three sectors to enhance the ability to represent asymmetric core cooling and a boron transport model.

For reactivity excursion event, the Rod Withdrawal from Subcritical (RWSC) event for NAPS was selected. The RWSC event offers an opportunity to observe the neutron flux overshoot the full-power nominal value for a very short time; hence the energy release and the fuel temperature increases are relatively small. Reactor trip is initiated by power range high neutron flux (low setting).

For loss of flow event, the Locked Rotor/Sheared Shaft (LR) RCS Overpressure event for SPS was chosen. The event creates a rapid expansion of the reactor coolant and reduced heat transfer in the steam generators, causing an insurge to the pressurizer and pressure increase throughout the RCS. For this event, the transient analysis is conducted at full power with manual rod control.

4.0 Analysis Results

4.1 Loss of Normal Feedwater

The LONF/LOAC event causes a reduction in heat removal from the primary side to the secondary system. Following a reactor trip, heat transfer to the steam generators continues to degrade resulting in an increase in RCS fluid temperature and a corresponding insurge of fluid into the pressurizer. There is the possibility of RCS pressure exceeding allowable values or the pressurizer becoming filled and discharging water through the relief valves. The event is mitigated when Auxiliary Feedwater (AFW) flow is initiated and adequate primary to secondary side heat removal is restored.

The case considered for this qualification is the limiting RCS overfill case for NAPS. This case assumes that pressurizer heaters, and offsite power are not available while pressurizer sprays and Power Operated Relief Valves (PORVs) are available. The limiting acceptance criteria for this case of LONF/LOAC is that the pressurizer will not go water solid at any point during the transient such that liquid water will be relieved through the pressurizer safety or relief valves.

The results for the LONF/LOAC comparison analysis are presented in Table 4.1-1 and Figures 4.1-1 through 4.1-12. The loss of feedwater flow to the steam generators (SG) results in a reduction in SG level until a reactor trip occurs on Low-Low SG level. Normalized power is shown on Figure 4.1-1. The timing response is shown in Table 4.1-1. The reactor trip signals for both codes are received at essentially the same time.

Table 4.1-1 LONF/LOAC Timing Response

Event	Time (seconds)	
	RETRAN-02	**RETRAN-3D**
Loss of Main Feedwater Flow	10.0	10.0
Pressurizer PORV 1 Opens (first occurrence)	27.14	26.70
Reactor Trip on SG Lo-Lo Level	33.43	33.25
Pressurizer PORV 1 Opens (second occurrence)	34.35	34.00
Turbine Trip	34.43	34.25
Pressurizer PORV 2 Opens	35.75	35.56
Peak Pressurizer Pressure	36.25	36.21
Peak Pressurizer Volume	39.00	38.65
SG Safety Valve Lifts (first occurrence)	39.50	39.25
AFW actuation	91.38	91.20
End of Transient	10000.00	10000.00

The reduction in SG level results in degraded heat transfer from the primary to secondary systems and an increase in RCS temperature. The average temperature for loops A and B are plotted on Figures 4.1-2 and 4.1-3 respectively. The rate of heatup prior to reactor trip is almost identical between the two cases. After the reactor trip occurs, the RCS cools somewhat until the loss of SG level and related heat removal is no longer able to remove decay and residual heat. The temperature then increases until AFW flow is actuated and adequate heat removal is restored.

The effect of the temperature change is reflected in the fluid density and associated pressurizer liquid volume change, as seen on Figure 4.1-4. The initial pressurizer insurge is nearly identical between the two cases. The maximum pressurizer liquid volume for the RETRAN-3D case is 1068.41 cubic feet versus 1066.66 cubic feet for RETRAN-02.

The pressurizer pressure responds to the heatup and pressurizer insurge as shown in Figure 4.1-5. The initial pressure increase causes one of the pressurizer PORV to open. As shown in Table 4.1-1, the timing of the first PORV opening is similar.

The steam generator pressure for loops A and B are shown in Figures 4.1-6 and 4.1-7, respectively. Note that the response differs between the two loops due to the orientation of the auxiliary feedwater system. Since the turbine driven auxiliary feedwater pump is assumed inoperable, no feedwater is delivered to the A steam generator. There is good agreement between the codes during the earlier parts of the transient, especially around the time of peak RCS liquid volume.

Differences in loop average temperature, pressurizer pressure and liquid volume behavior are observed in the later parts of the transient. As there are distinct but small differences in the code, divergences between the codes can propagate over longer transients. This can cause trips and valves to be actuated at varying times. Such variance can be observed in the response of the secondary side specifically with respect to safety valve actuation timing.

Divergence between the two codes is observed around the times of steam safety valve actuation in the later portions of the transient. Figure 4.1-8 shows the pressure in steam generator B on a non-logarithmic scale around the time of safety valve actuation. Due to the 3% blowdown on the safety valve, it remains partially open and steam continues to flow out the valve as shown in Figure 4.1-10. Figure 4.1-9 also shows the normalized valve flow area during this time. Small deviations in the calculated choked flow between the codes are shown while the valve is slightly open. Just after 2400 seconds liquid relief begins through the MSSV resulting in a change in mass flow rate. After the MSSV actuates around 2700 seconds, a small deviation in the calculated choked flow is observed. This small deviation compounds over time leading to the deviation in pressure shown in Figure 4.1-8. The difference in calculated choked flow results from several changes to the choked flow model in RETRAN-3D which is exacerbated by the presence of liquid. To demonstrate this, a case was run with a controller on AFW to limit steam generator overfill. Figure 4.1-11 shows steam generator pressure around the time of the MSSV actuation with the AFW controller in place. As this figure shows, the deviation between the codes in significantly less when liquid relief is not present. This demonstrates that liquid relief in the choked flow models contribute a large portion to the deviations seen in the base case. This is further shown by observing the flow rates through the main steam safety valves in Figure 4.1-12.

Figure 4.1-1 LONF/LOAC - Nuclear Power

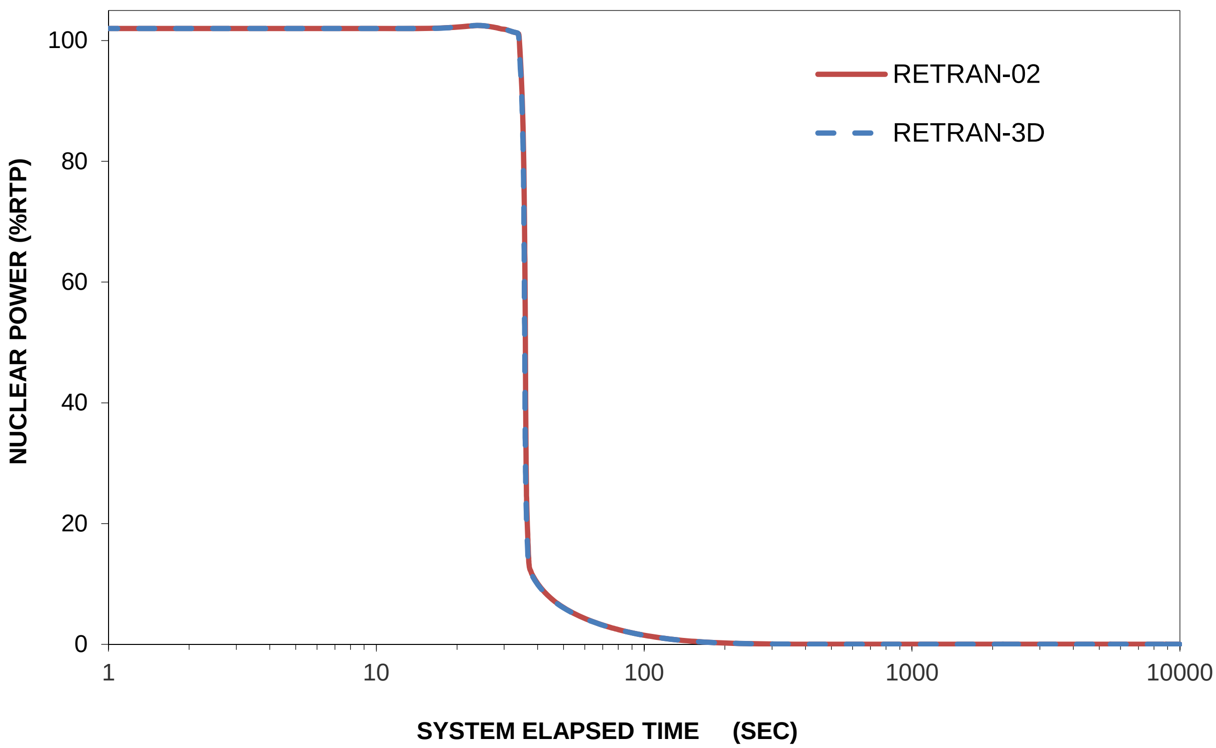


Figure 4.1-2 LONF/LOAC - Loop A Average Temperature

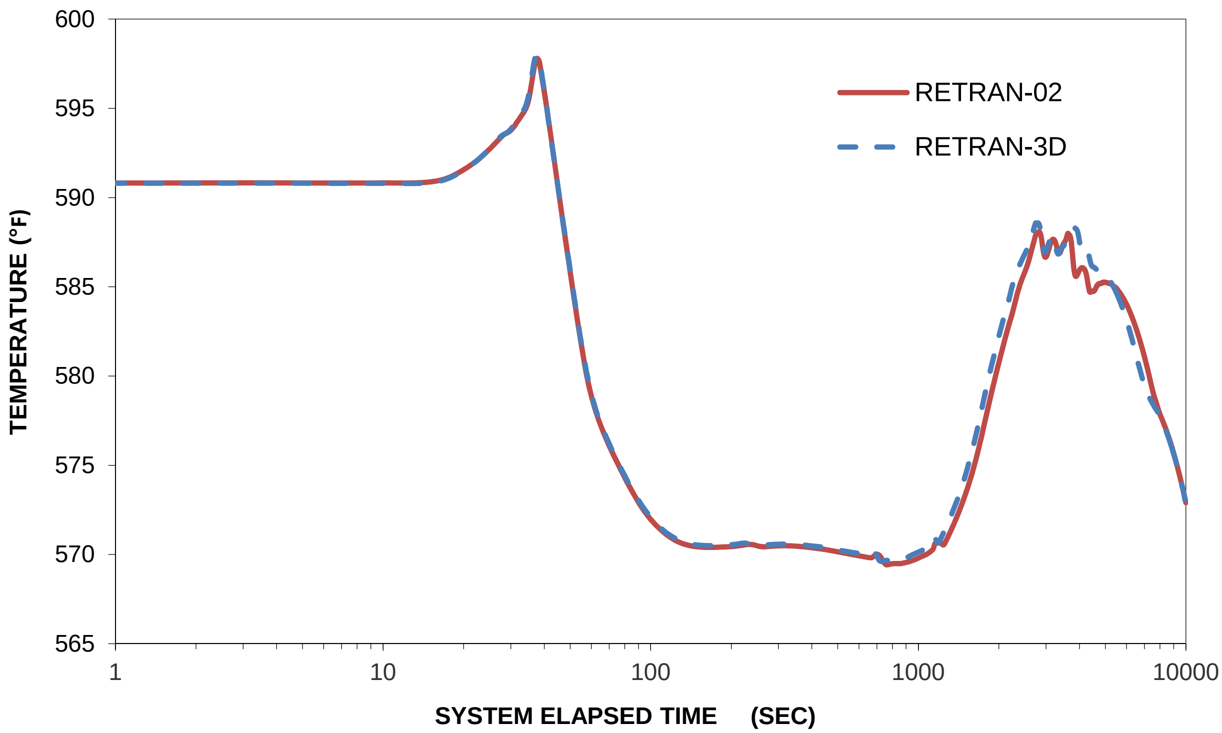


Figure 4.1-3 LONF/LOAC - Loop B Average Temperature

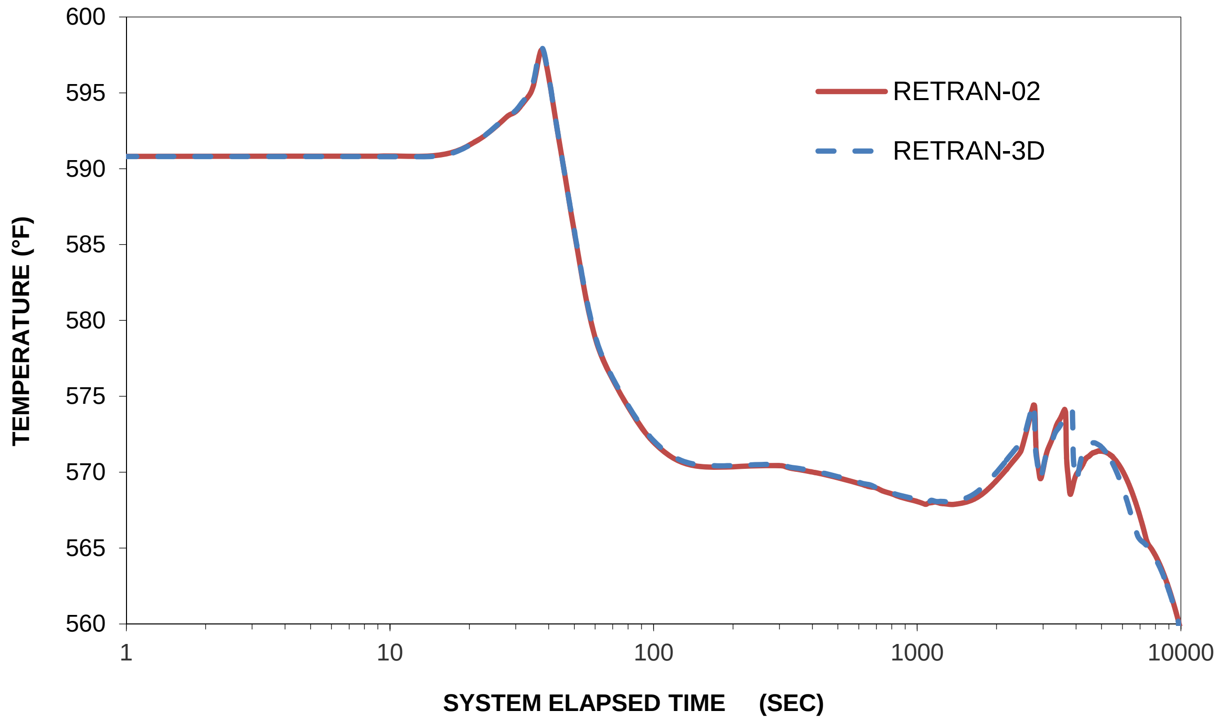


Figure 4.1-4 LONF/LOAC - Pressurizer Liquid Volume

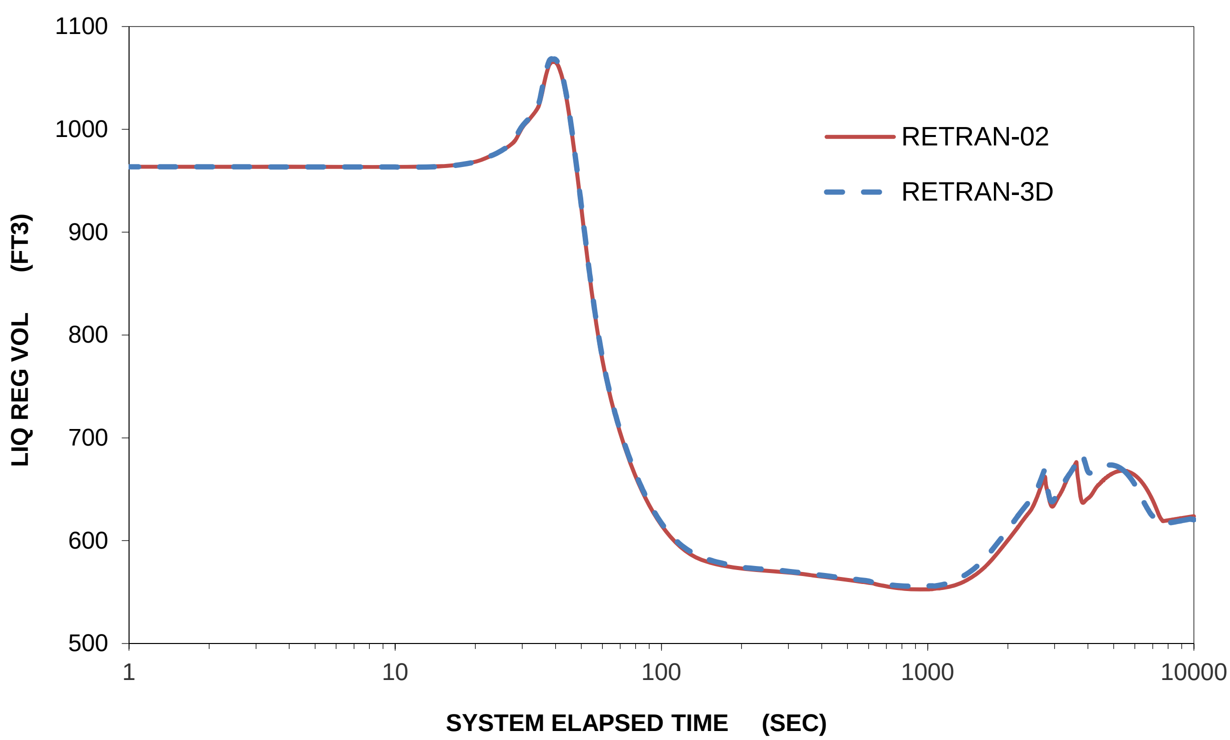


Figure 4.1-5 LONF/LOAC - Pressurizer Pressure

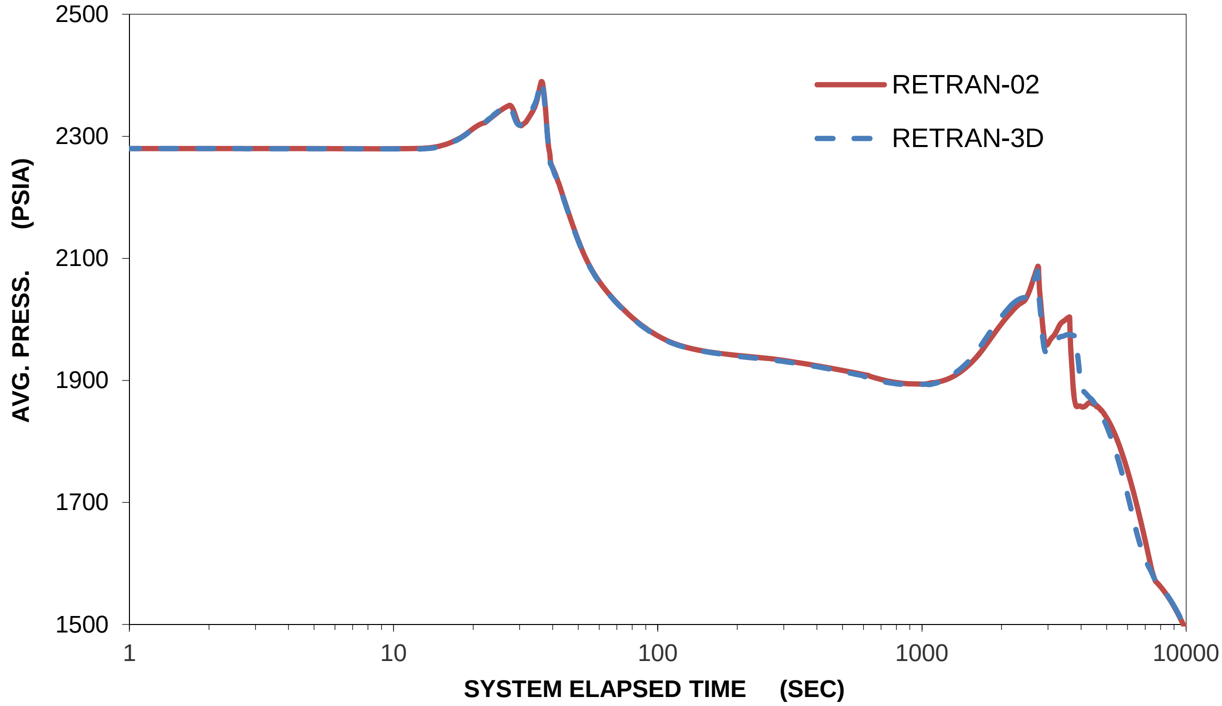


Figure 4.1-6 LONF/LOAC - Steam Generator A Pressure

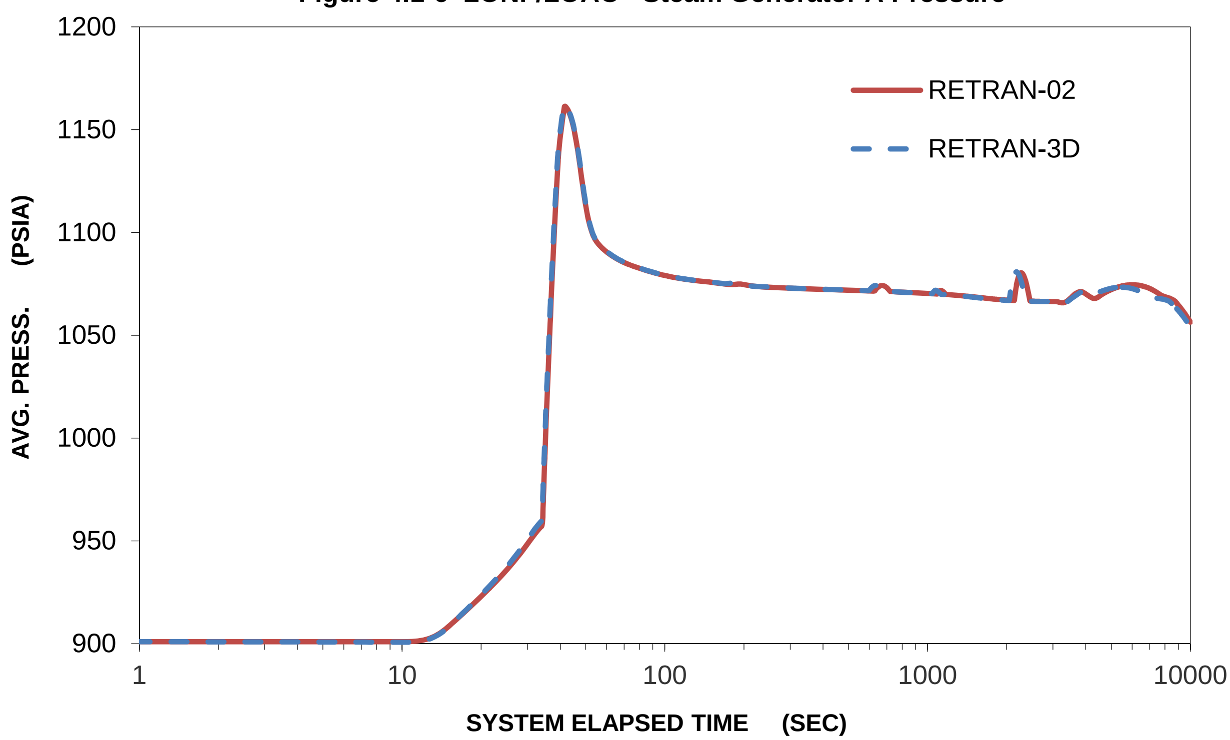


Figure 4.1-7 LONF/LOAC - Steam Generator B Pressure

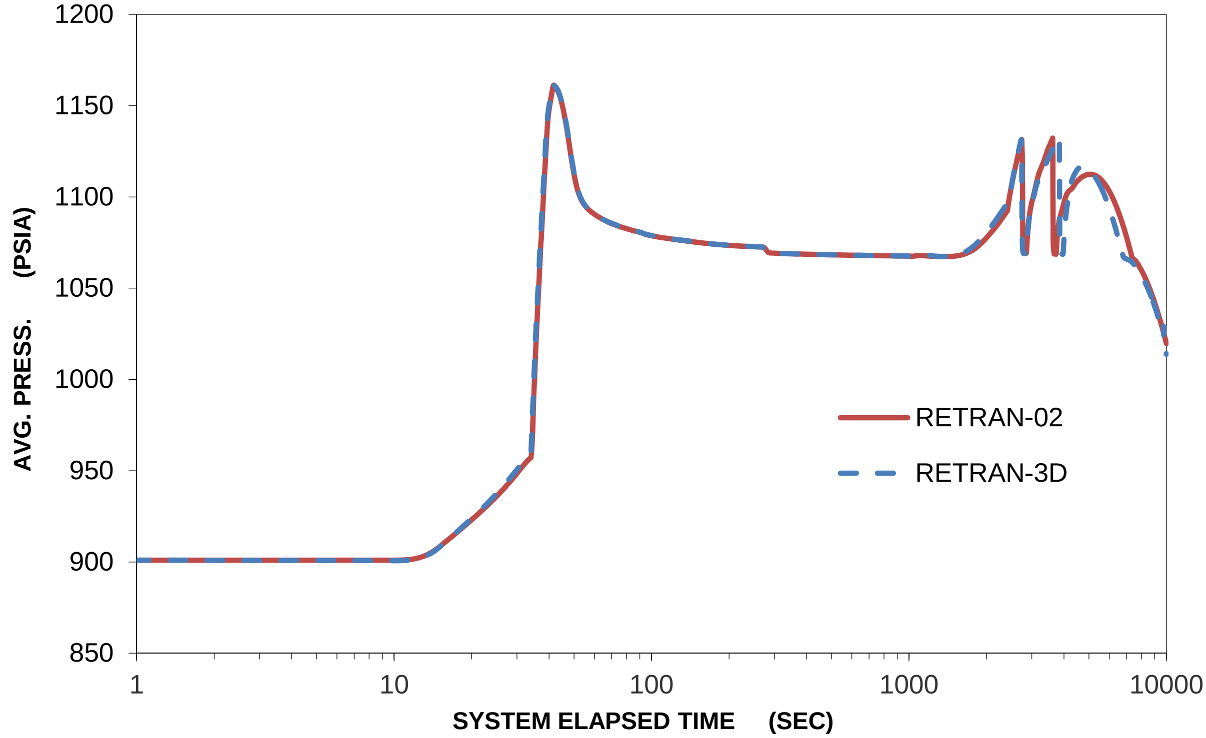


Figure 4.1-8 LONF/LOAC - Steam Generator B Pressure Reponse During MSSV Actuation

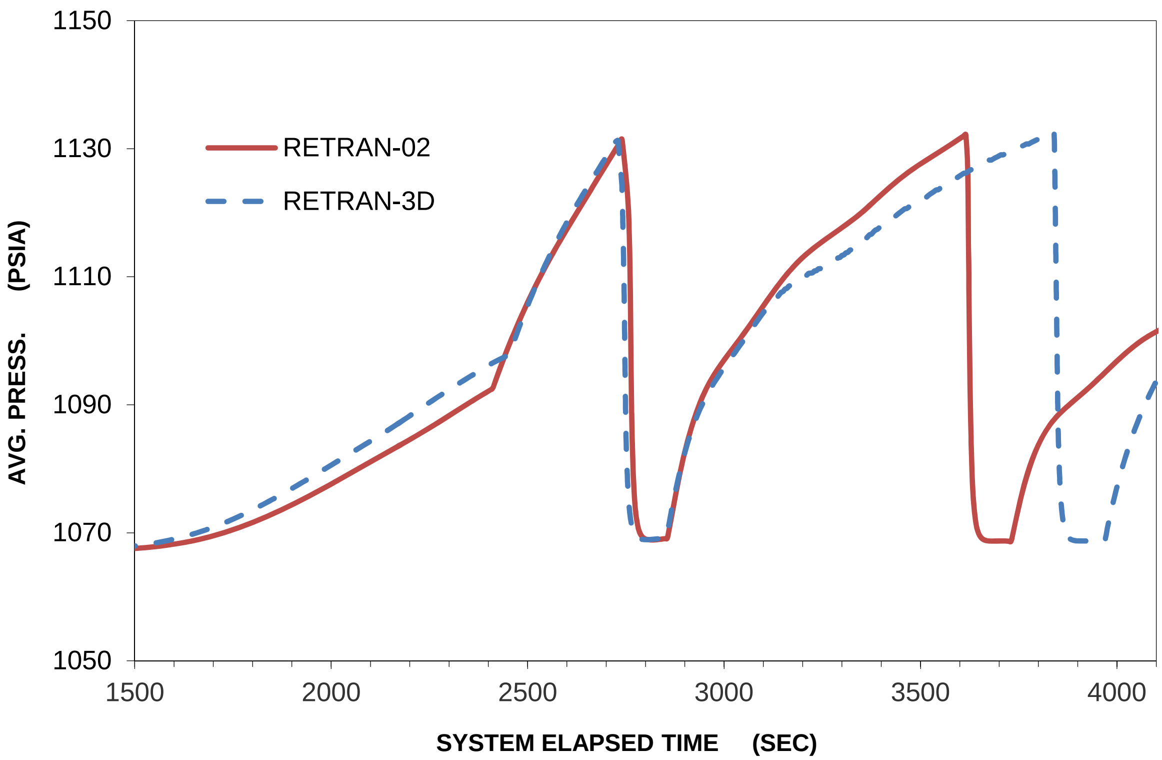


Figure 4.1-9 LONF/LOAC - MSSV 1 Normalized Flow Area

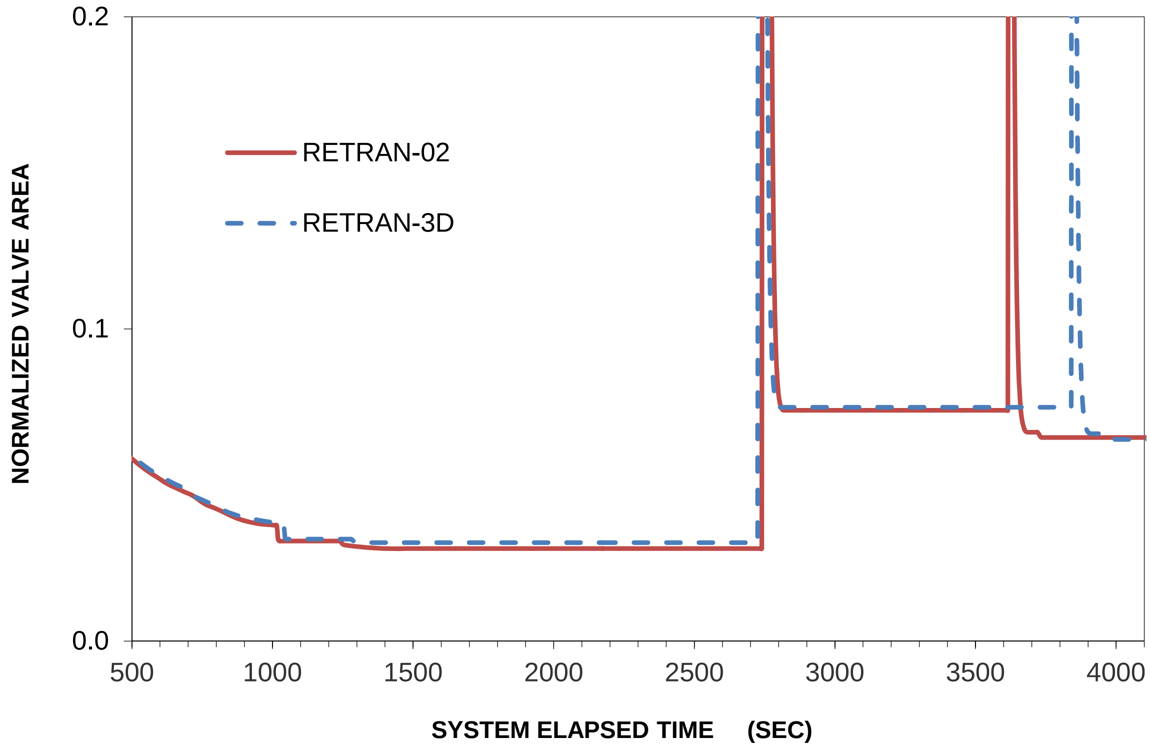


Figure 4.1-10 LONF/LOAC - MSSV 1 Mass Flow

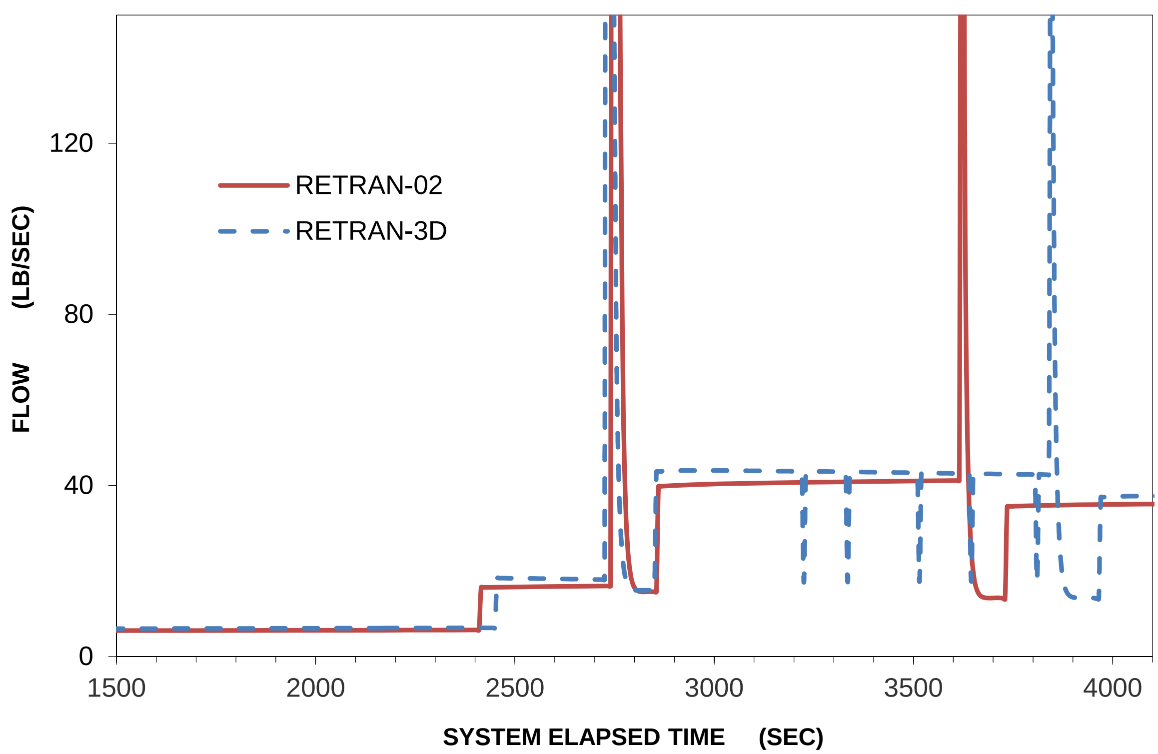


Figure 4.1-11 LONF/LOAC - Steam Generator B Pressure with AFW Controller

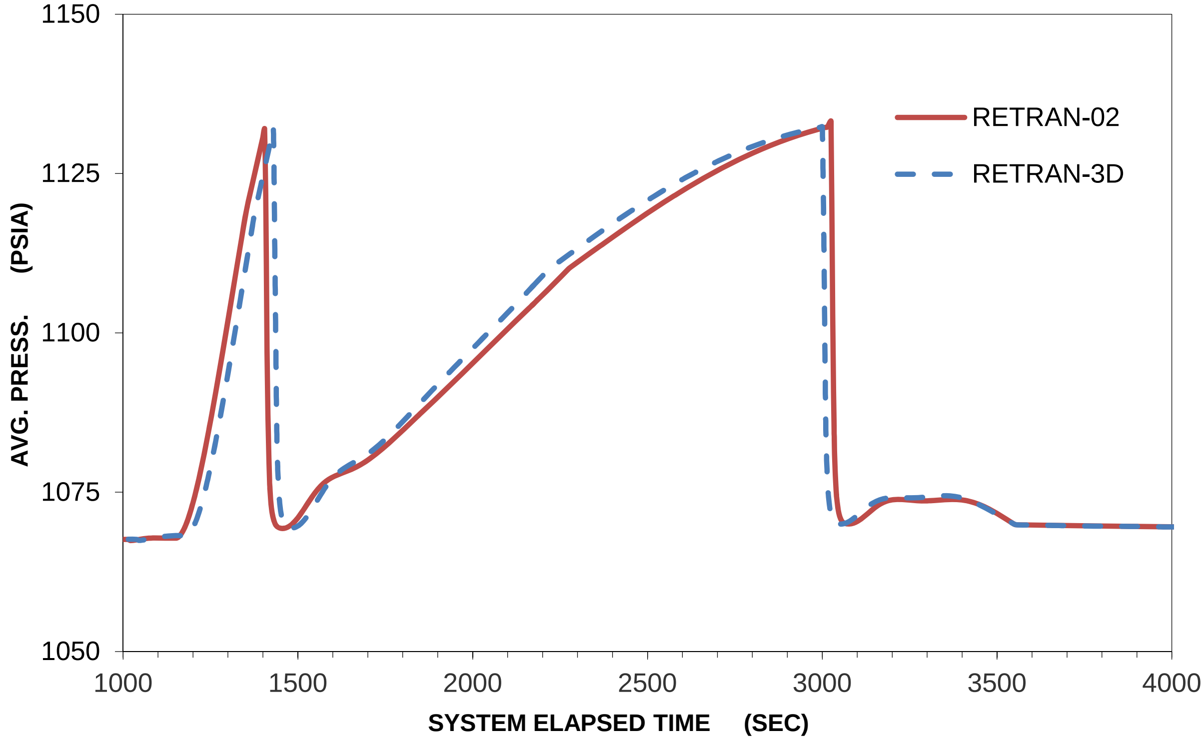
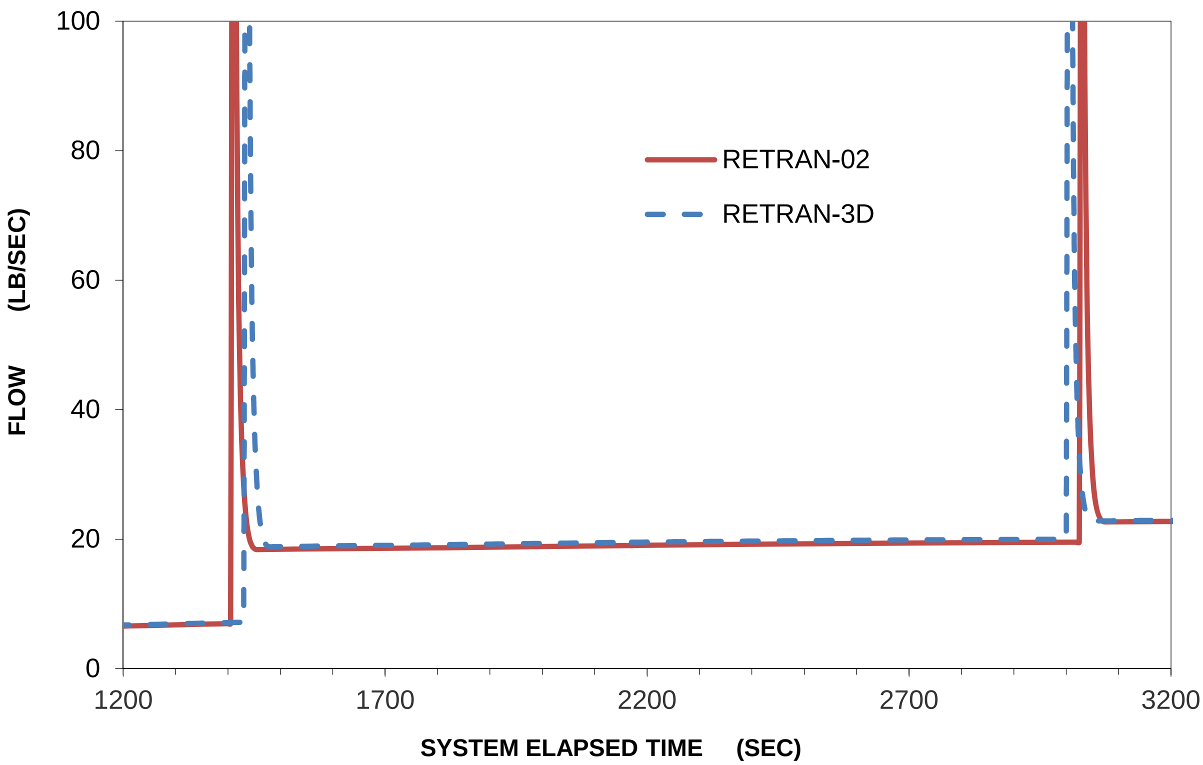


Figure 4.1-12 LONF/LOAC - MSSV 1 Mass Flow with AFW Controller



4.2 Loss of Load / Turbine Trip

The loss of load/turbine trip (LOL/TT) event is defined as a complete loss-of-steam load and turbine trip from full power without a direct reactor trip, resulting in a primary fluid temperature rise and a corresponding pressure increase in the primary system. This transient results in degraded steam generator heat transfer, reactor coolant heat-up, and pressure increase following a manual turbine trip. It is initiated by manually tripping the turbine with no direct reactor trip on turbine trip. The case considered for this benchmark is the analysis of record case for RCS overpressurization at North Anna Power Station.

Pressure in the RCS increases during a loss of load due to degraded heat transfer in the steam generator and is alleviated only when the pressurizer safety valves (PSV) open. The pressurizer pressure response is shown on Figure 4.2-1. The timing response is shown in Table 4.2-1. Maximum values for various parameters are shown in Table 4.2-2.

Table 4.2-1 LOL/TT - Timing Response

Event	Time (seconds)	
	RETRAN-02	RETRAN-3D
Turbine Trip	0.10	0.10
High Pressurizer Pressure Trip Reached	5.55	5.65
Reactor Trip on High Pressure	6.55	6.65
Peak RCS Pressure	9.35	8.98
SG Safety Valve Lifts	9.20	9.15
End of Transient	100.00	100.00

Table 4.2-2 LOL/TT - Maximum Values

Parameter	Value	
	RETRAN-02	RETRAN-3D
Peak Pressurizer Pressure	2664.33 psia	2666.67 psia
Peak RCS Pressure	2754.75 psia	2737.47 psia
Peak SG Pressure	1174.56 psia	1175.44 psia
Peak Pressurizer Level	83.18 %	83.52 %

Figure 4.2-2 shows the cold leg pressure response which is the location of the peak RCS pressure. The peak RCS pressure for RETRAN-02 is 2754.75 psia compared with 2737.47 psia for RETRAN-3D. This is approximately a 17 psi difference in peak pressure. However, these values are within a 1% difference. Due to the rapid nature of the pressure transient experienced during a LOL/TT and given the nature of the changes in RETRAN-3D, including changes to the mixture energy equation, these differences are reasonable. The high pressurizer pressure trip setpoint is reached sooner in RETRAN-02 by only 0.1 seconds.

Core inlet flow is shown in Figure 4.2-4. In addition Figures 4.2-5 through 4.2-8 show the loop average temperature and steam generator pressure for both the A and B loops. The RETRAN-02 and RETRAN-3D loop average temperature and steam generator pressure are virtually identical in loop A. However, in loop B the loop average temperature varies slightly after the reactor trip and a more noticeable difference is observed in steam generator B. In addition there is a deviation in the core inlet flow between the two codes.

Observing the behavior of the main steam safety valves (MSSVs), as shown in Figure 4.2-9 shows that for loop B in the RETRAN-02 case the fourth MSSV does not actuate. However this valve actuates on loops A and C and on all loops in the RETRAN-3D case. A review of the output for the RETRAN-02 steam generator pressures shows less than a psi deviation between loops A and C. In addition, the pressure rises slightly faster in RETRAN-3D as seen through the earlier actuation of the MSSV in Figure 4.2-9. These behaviors result in the deviation in the pressure response seen in Figure 4.2-8 as steam is relieved through the fourth MSSV for the RETRAN-3D case. This is also responsible for the loop average temperature deviation as shown in Figure 4.2-7 due to variations in available heat sink.

To demonstrate that this behavior leads to the steam generator pressure and loop average temperature deviations, two cases are run with the fourth MSSV setpoint lowered slightly (1 psi) to ensure actuation on loop B of the RETRAN-02 case. Figures 4.2-10 through 4.2-12 show good agreement between the RETRAN-02 and RETRAN-3D with the revised set point as all valves now actuate with the same behavior.

Figure 4.2-1 LOL/TT - Pressurizer Pressure

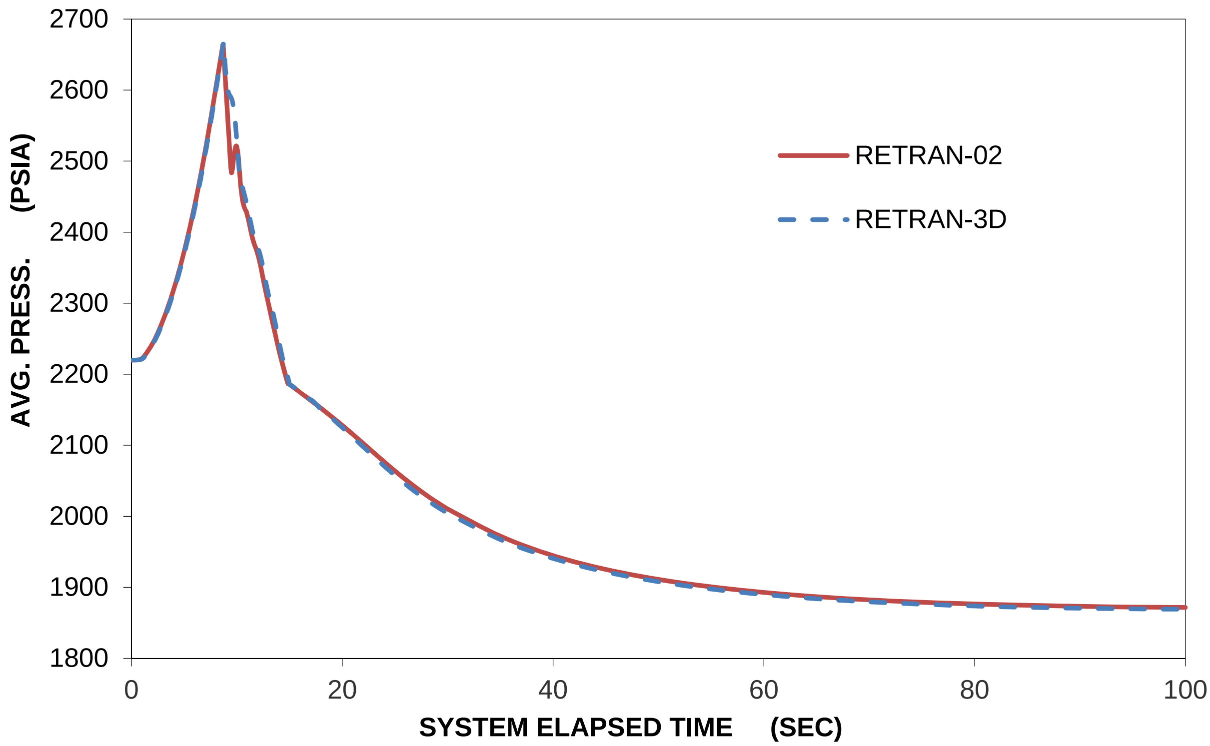


Figure 4.2-2 LOL/TT - Cold Leg Pressure

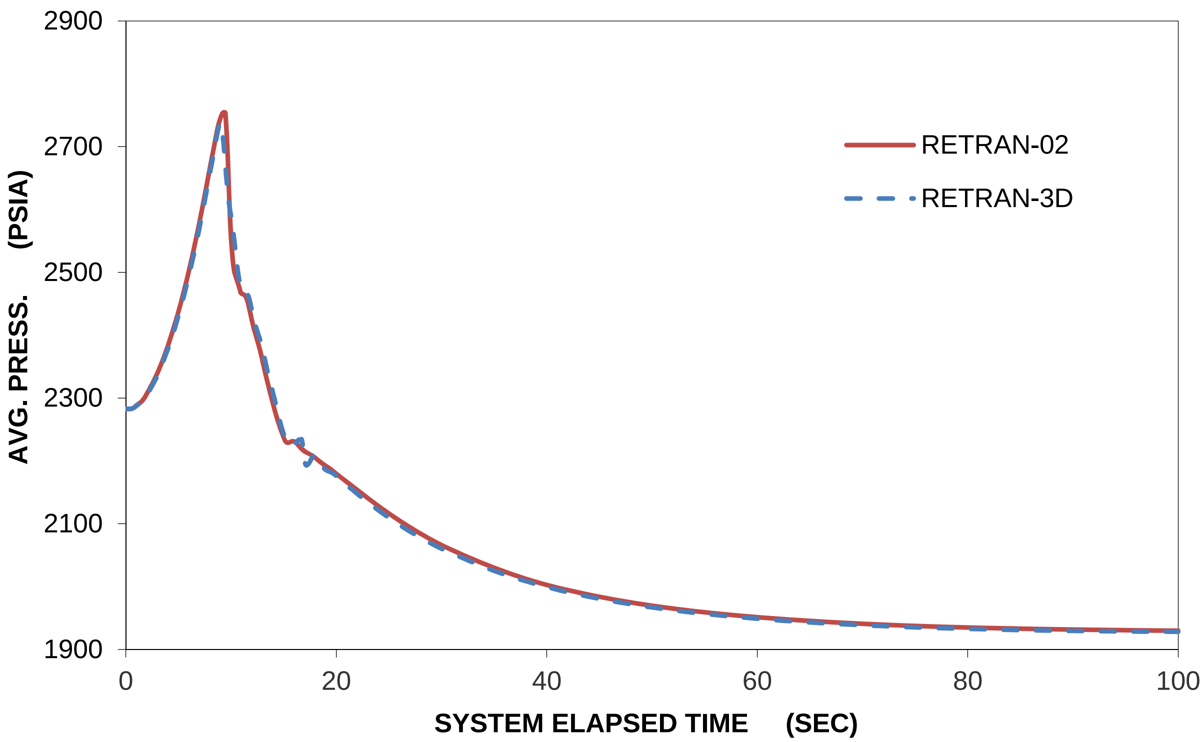


Figure 4.2-3 LOL/TT - Nuclear Power

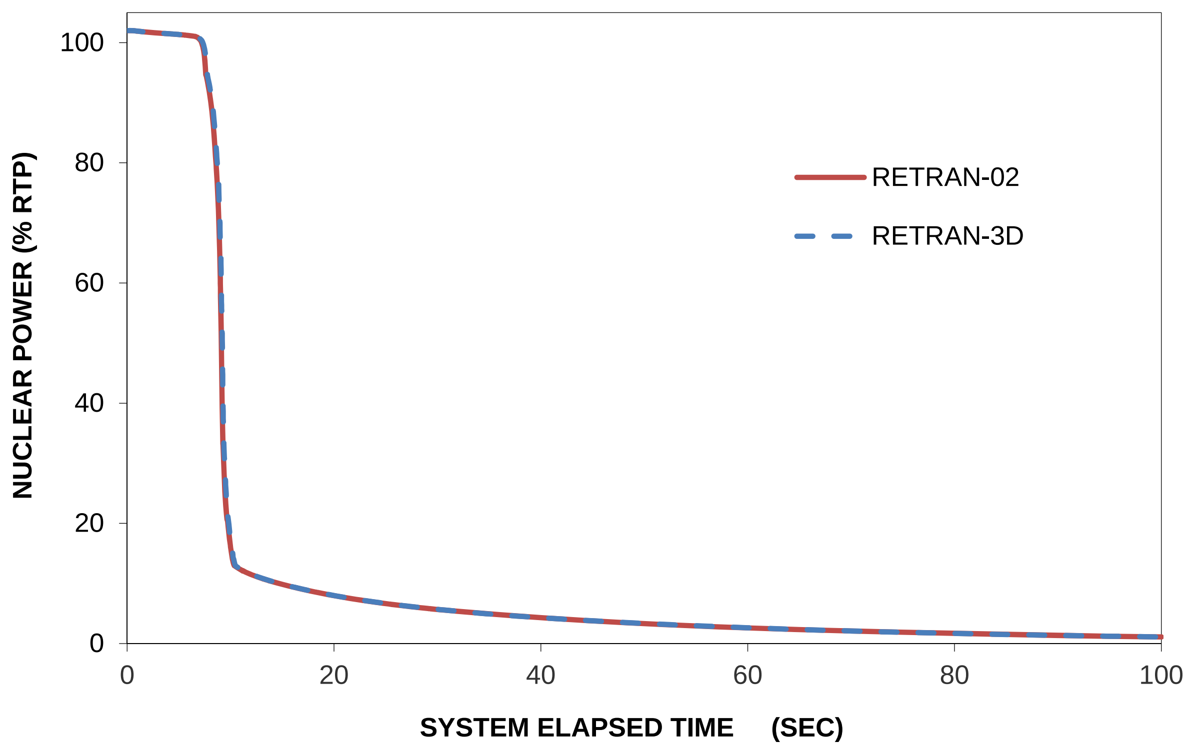


Figure 4.2-4 LOL/TT - Core Inlet Flow

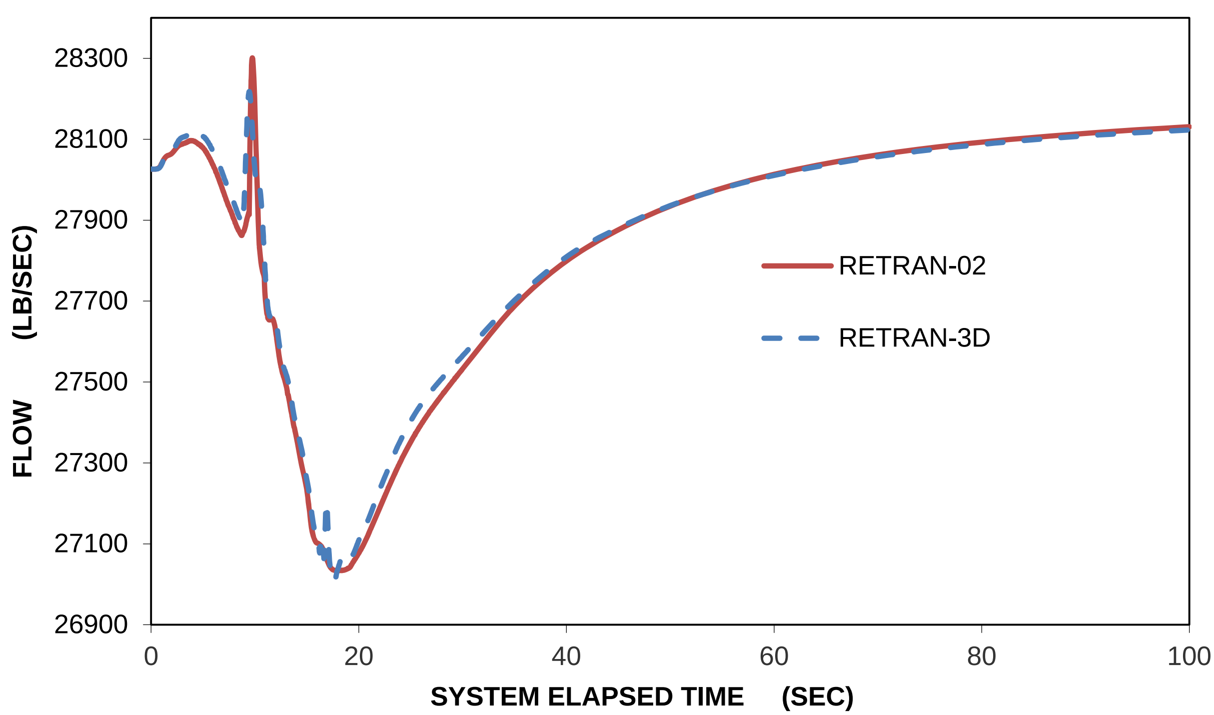


Figure 4.2-5 LOL/TT - Loop A Average Temperature

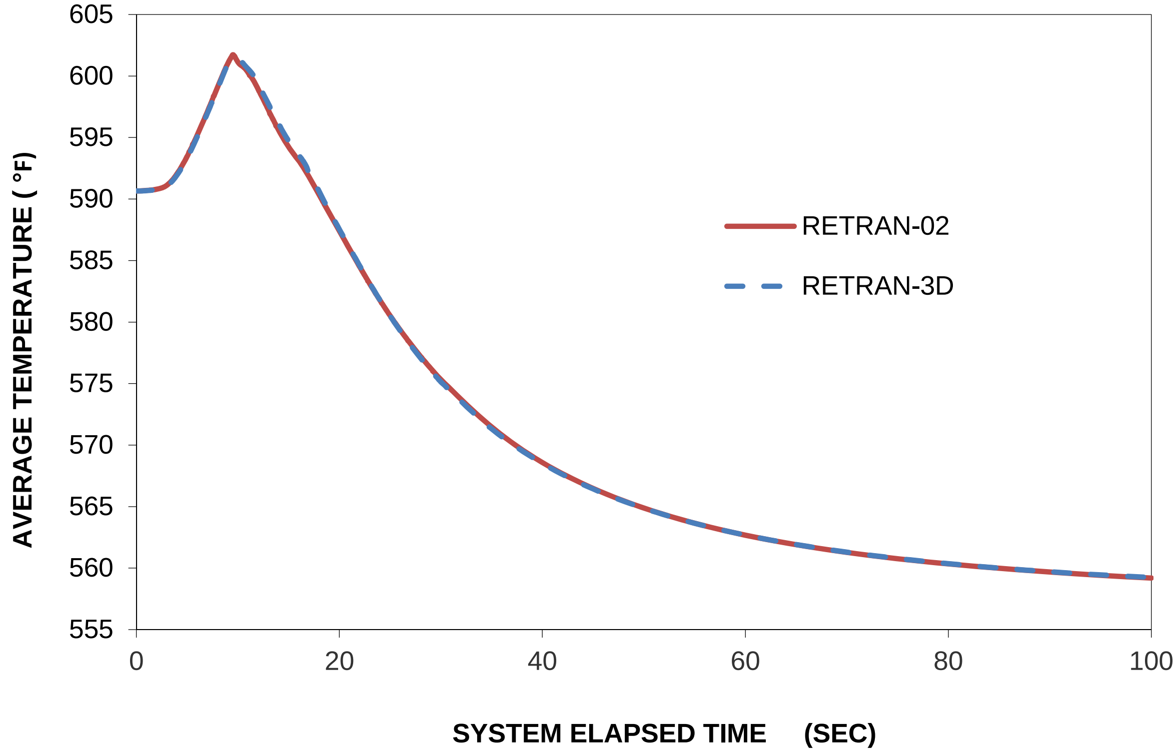


Figure 4.2-6 LOL/TT - Loop A Steam Generator Pressure

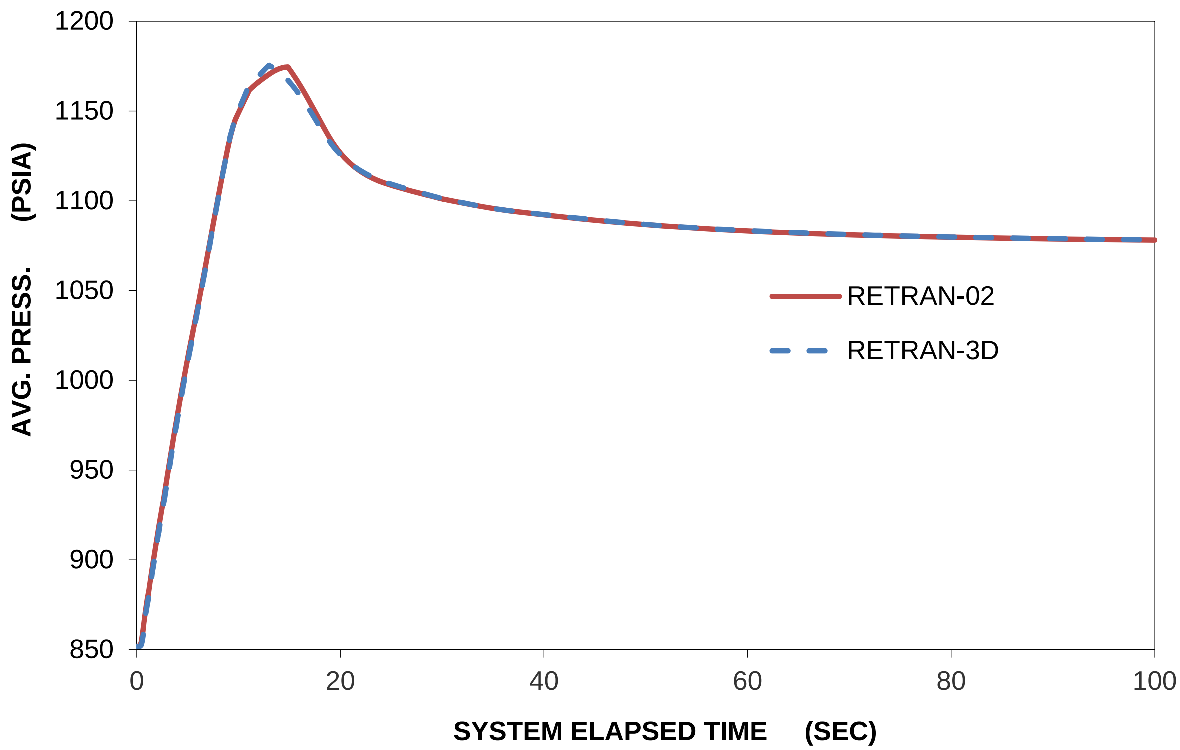


Figure 4.2-7 LOL/TT - Loop B Average Temperature

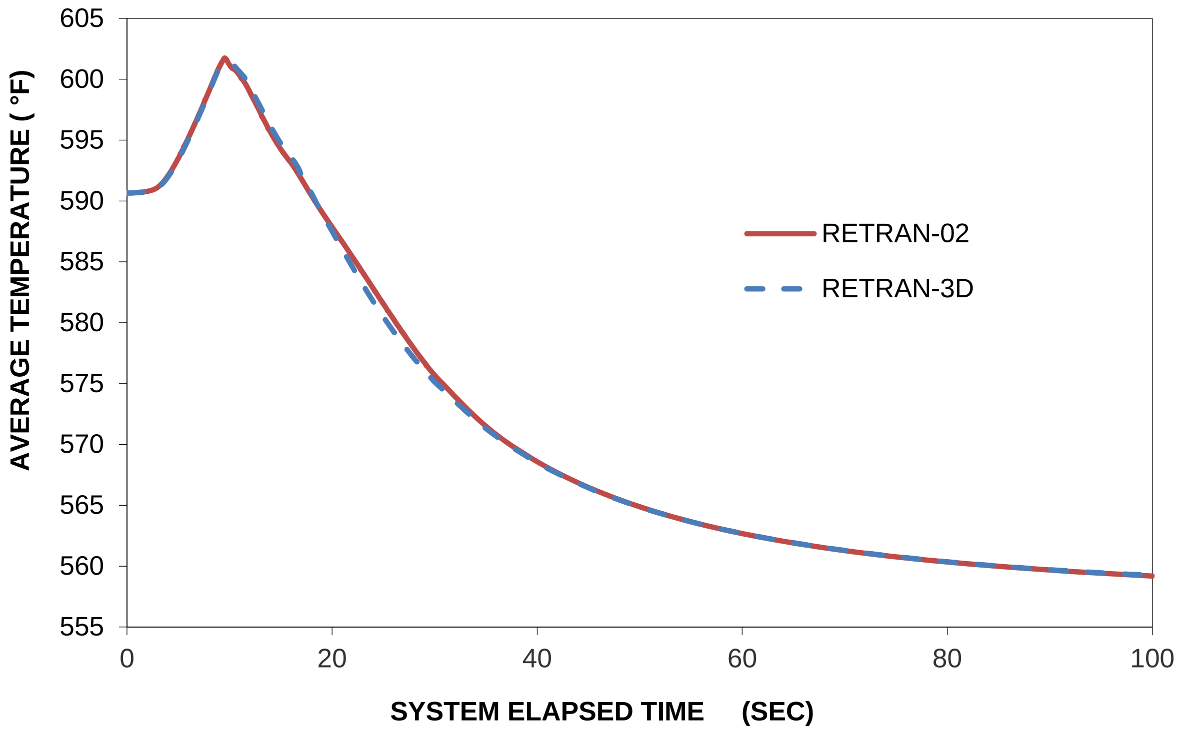


Figure 4.2-8 LOL/TT - Loop B Steam Generator Pressure

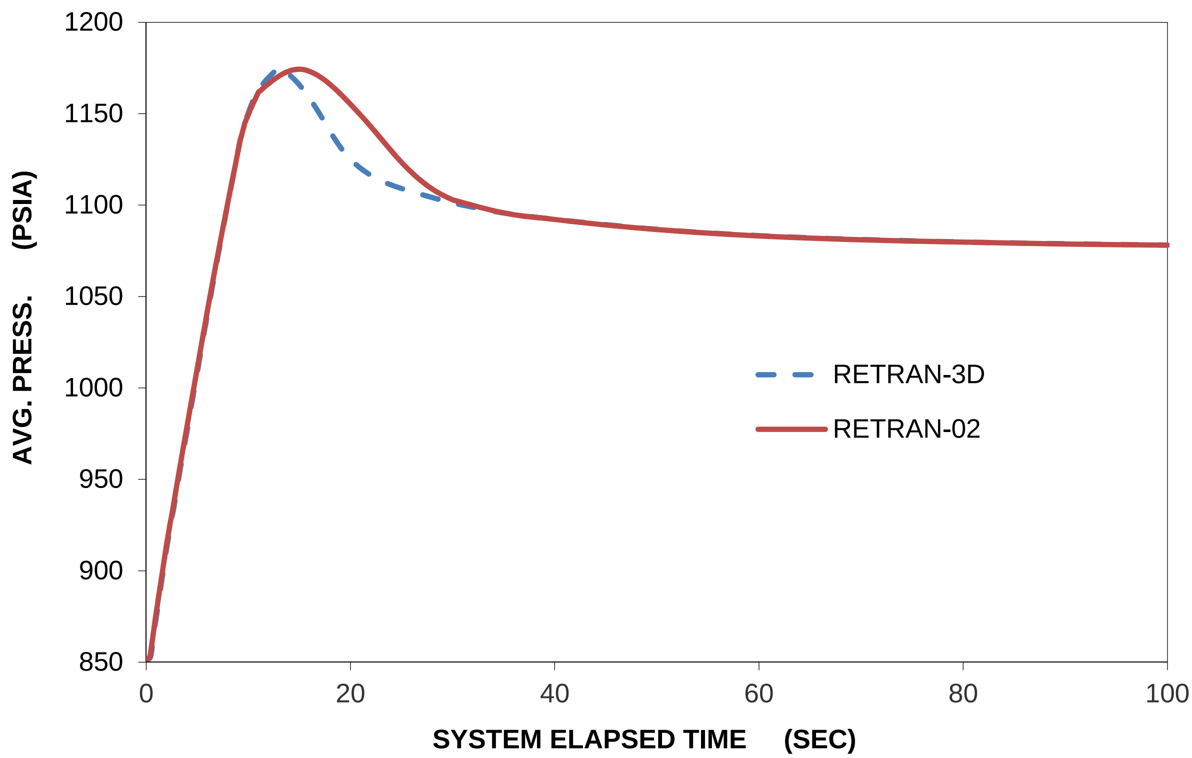


Figure 4.2-9 LOL/TT MSSV #4 Flow Rates

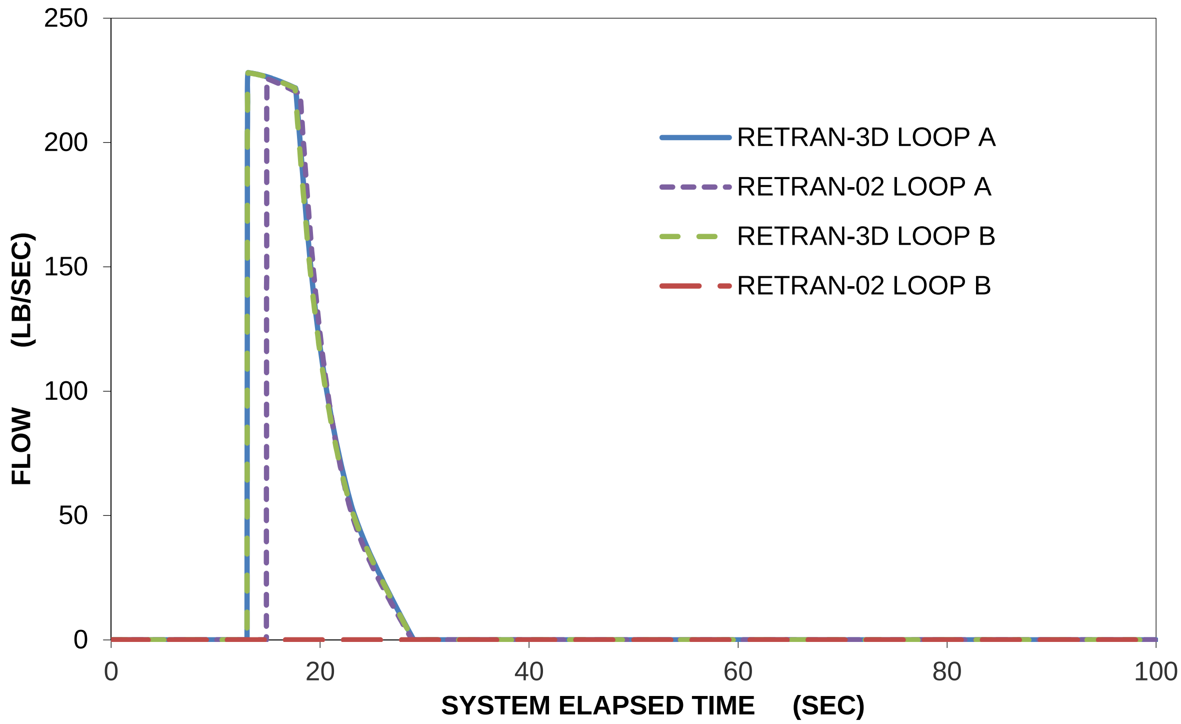


Figure 4.2-10 LOL/TT - Loop B Steam Generator Pressure with MSSV Modification

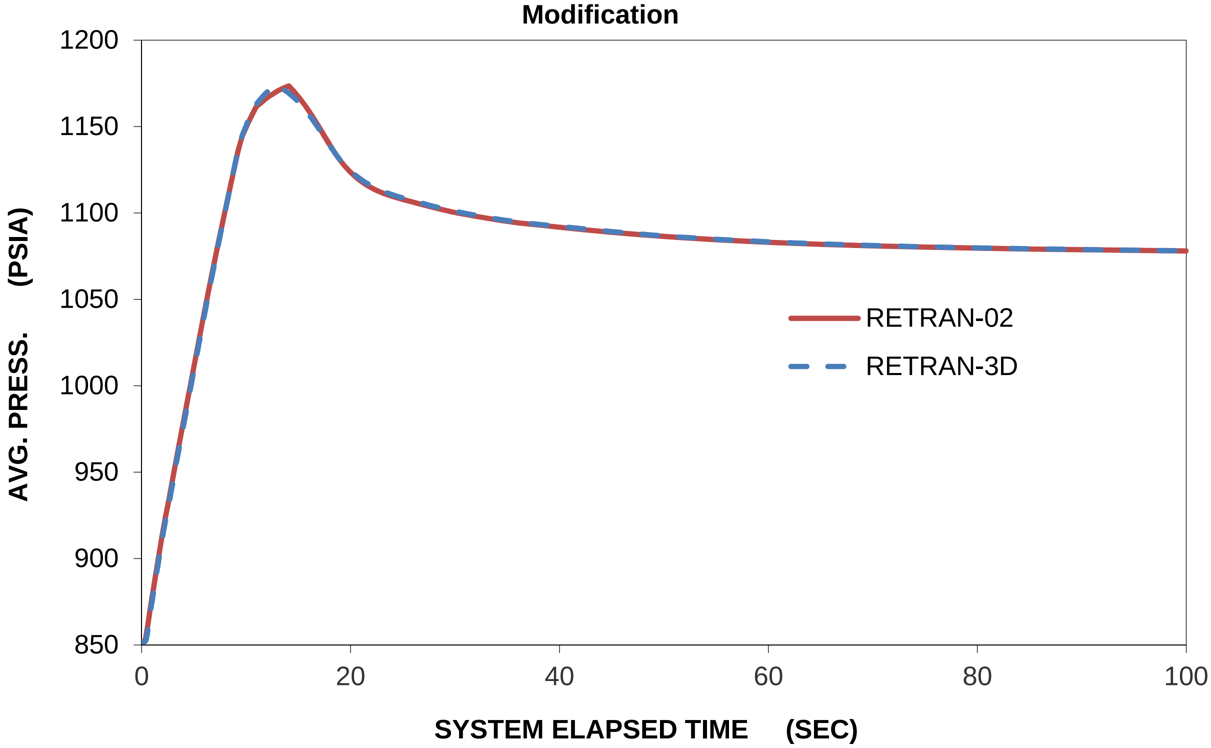


Figure 4.2-11 LOL/TT - Loop B Average Temperature with MSSV Modification

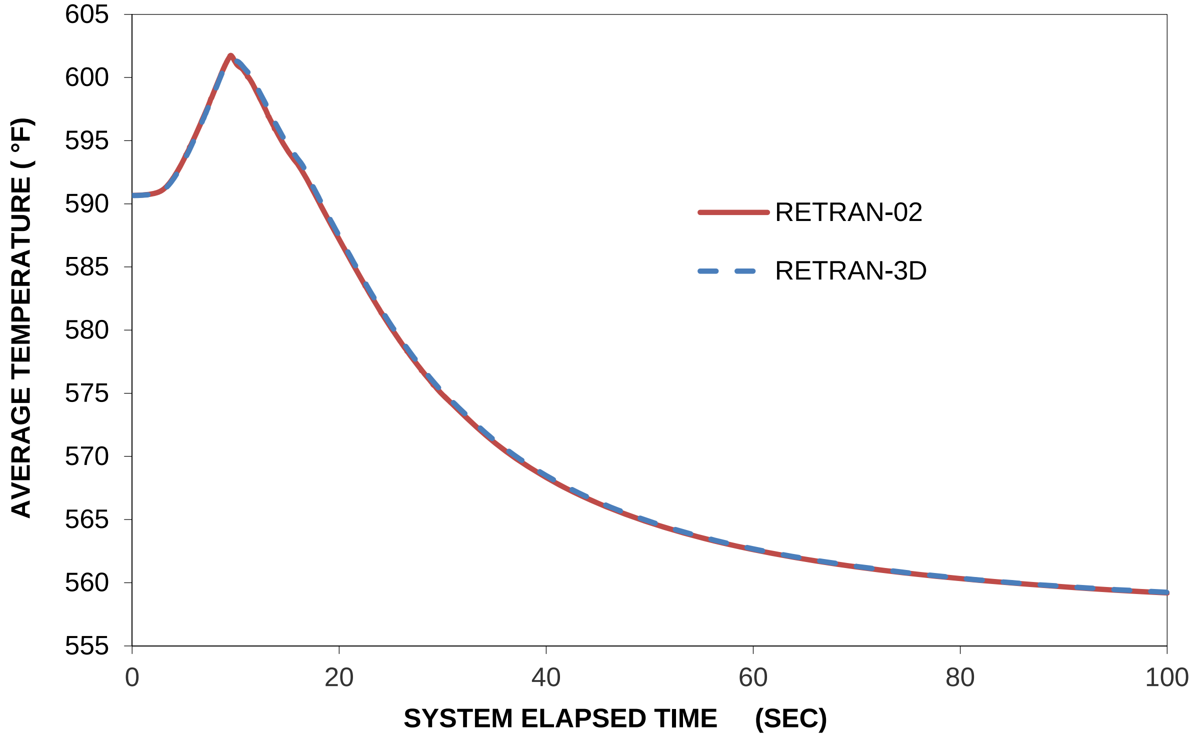
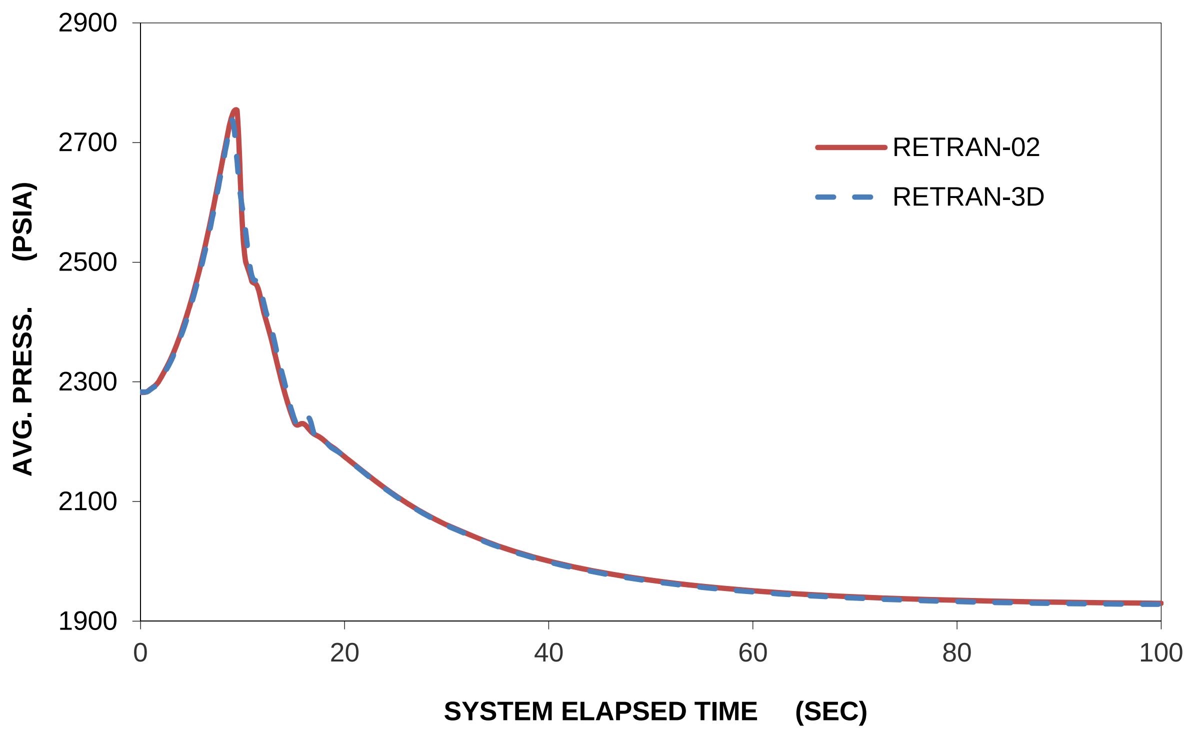


Figure 4.2-12 LOL/TT - Cold Leg Pressure with MSSV Modification



4.3 Main Steamline Break

The MSLB event is a rupture in the main steam piping resulting in a rapid depressurization of the SG secondary and corresponding cooldown of the primary. The temperature reduction results in an insertion of positive reactivity with the potential for core power increase and Departure from Nucleate Boiling Ratio (DNBR) violation. The MSLB transient scenario presented here is modeled as an instantaneous, double-ended break at the nozzle of one steam generator from hot shutdown conditions with offsite power available. The MSLB analysis assumes the most reactive rod cluster control assembly stuck in its fully withdrawn position after reactor trip, which could cause the core to become critical and return to power. The event is mitigated by boric acid injection delivered by the safety injection system and negative Doppler temperature coefficient.

The results for the MSLB comparison analysis are presented in Figures 4.3-1 through 4.3-6. The nuclear power and core heat flux responses are shown in Figures 4.3-1 and 4.3-2. Prompt Criticality occurs around 29 seconds. The small bump in the core heat flux early in the transient is driven by the RCS cooldown rather than nuclear power. The fuel temperature has not increased due to recriticality at this point, rather the difference between the temperature of the fuel and the coolant draws heat from the fuel early in the transient. Core heat flux slightly lags nuclear power with the peak heat flux of 20.3% of the 2942.2 MWt occurring at approximately 245 seconds. At approximately 550 seconds after the initiation of the transient, the core heat flux is less than half of the peak value.

The total core reactivity is shown in Figure 4.3-3. After the prompt criticality spike, Doppler feedback adds negative reactivity which turns the nuclear power around. As the cooldown continues, negative reactivity from Doppler temperature and Doppler power approximately balance the positive reactivity from increasing moderator density. A short time after the power peak occurs, injection of boron from the SI system reaches the core and continues to cause power to decrease. The inlet RCS boron concentration from SI is shown in Figure 4.3-5.

The pressurizer pressure response is shown in Figure 4.3-4. As the upper head starts to flash following the emptying of the pressurizer, the depressurization rate is decreased significantly. The pressurizer pressure bottoms out at 782.67 psia, and slowly increases to 877.6 psia by the end of the 600 second transient. The faulted loop steam generator pressure is plotted in Figure 4.3-6. The pressure in the faulted loop falls off quickly after the transient initiation. After main steam isolation at 18.8 seconds, the unfaulted SG pressure increases for a short period of time and then continues to decrease.

As shown, the responses of various parameters following a MSLB with offsite power event show very good agreement when analyzing with the two codes. Table 4.3-1 shows some representative values for the key parameters of this event.

Table 4.3-1 MSLB – Maximum Analysis Values

	RETRAN-02	RETRAN-3D
Minimum Pressurizer Pressure [psia]	782.67	787.52
Maximum Heat Flux [% of nominal]	20.3	20.2
Maximum Core Reactivity [pcm]	474.225	474.225
Faulted Loop SG Pressure at 600 seconds [psia]	84.3	84.4

Figure 4.3-1 MSLB - Normalized Core Heat Flux

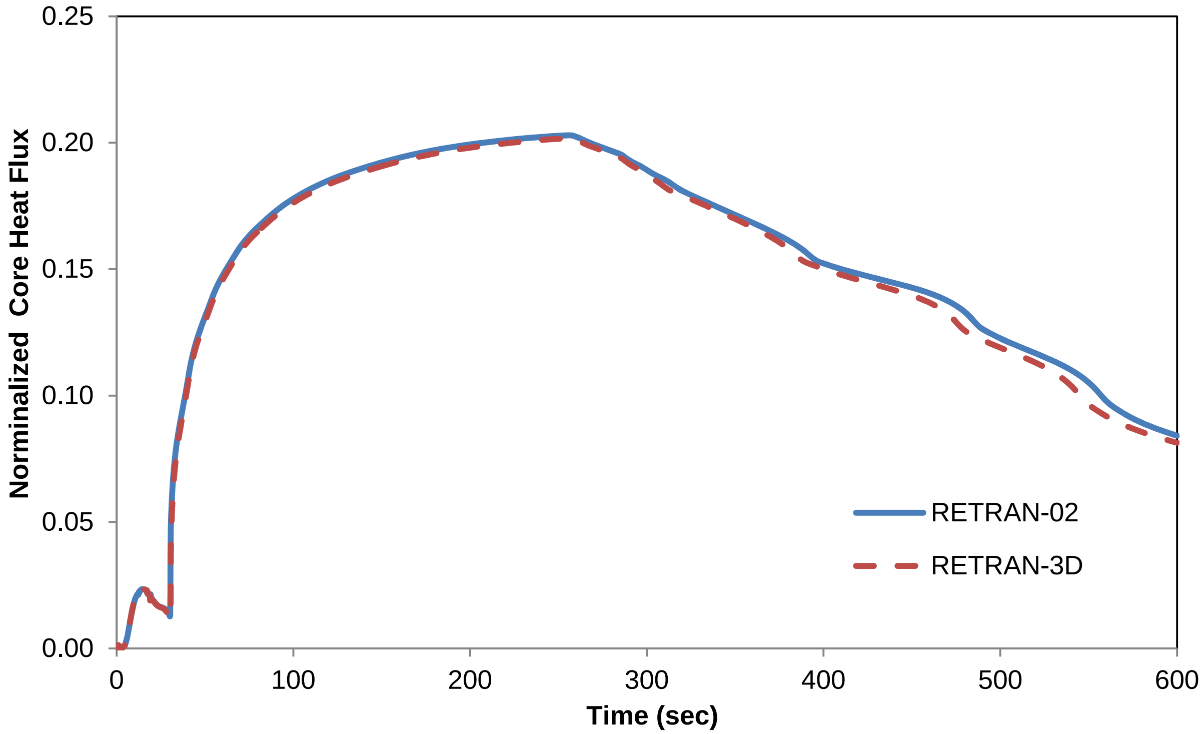


Figure 4.3-2 MSLB - Nuclear Power

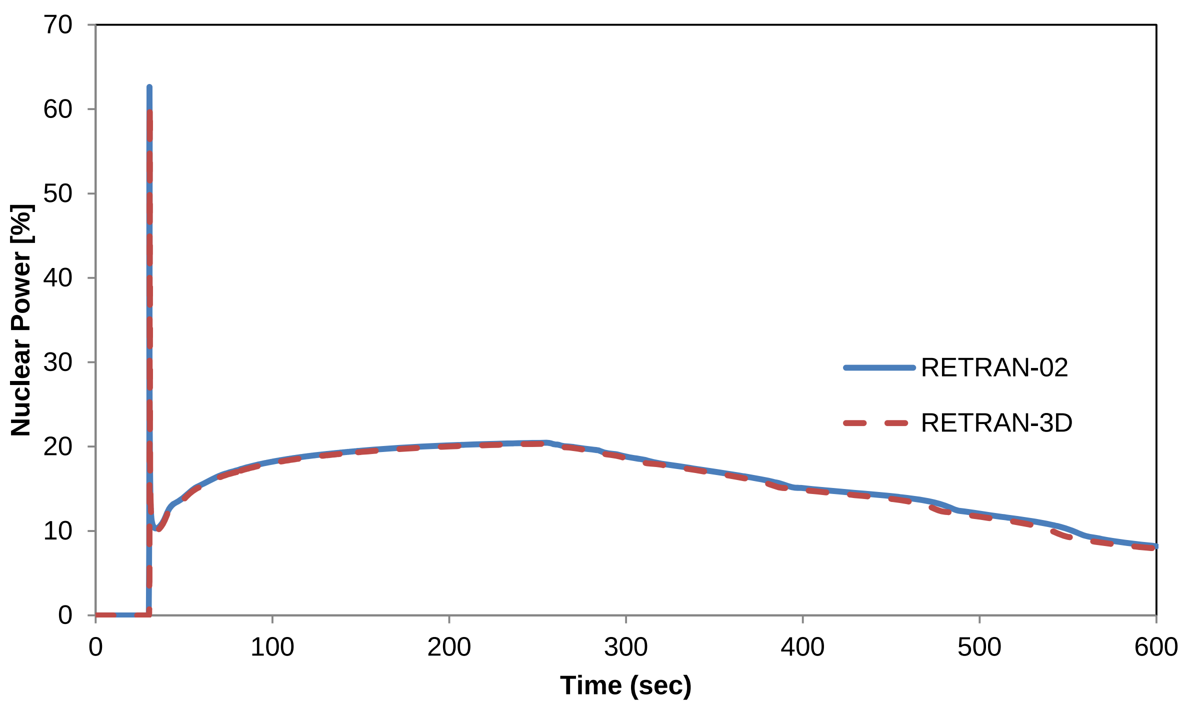


Figure 4.3-3 MSLB - Net Core Reactivity

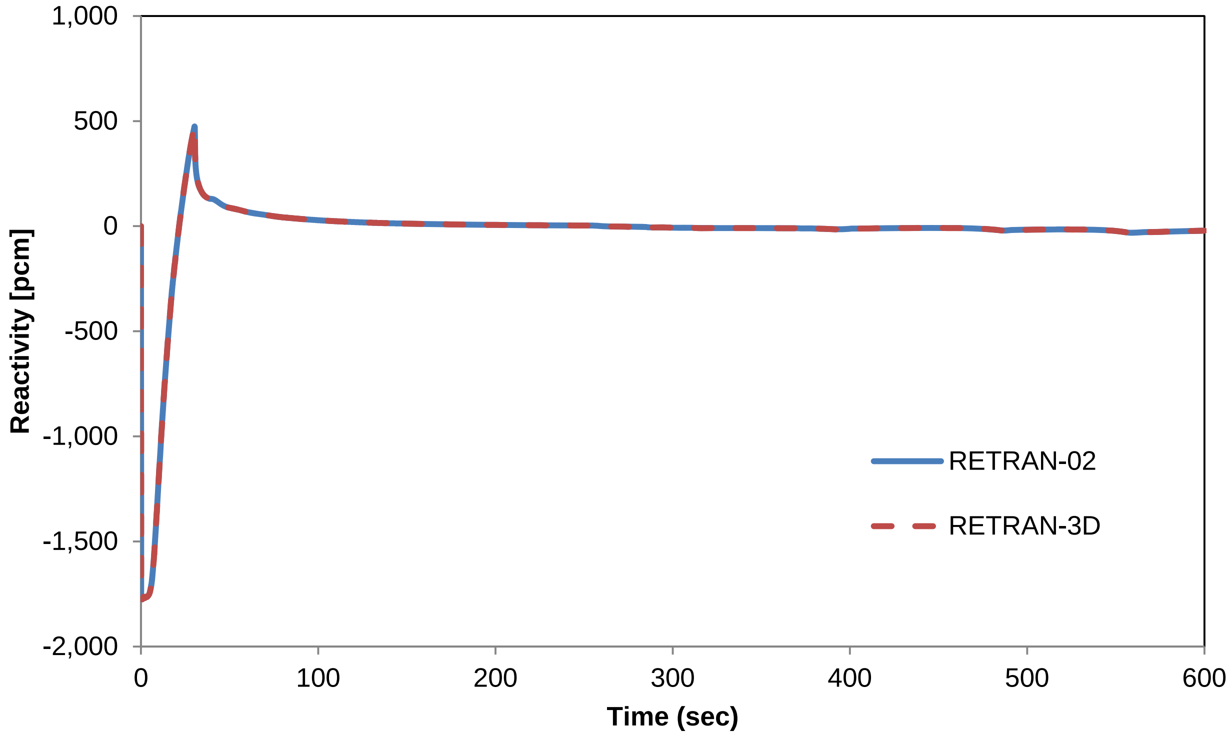


Figure 4.3-4 MSLB - Pressurizer Pressure

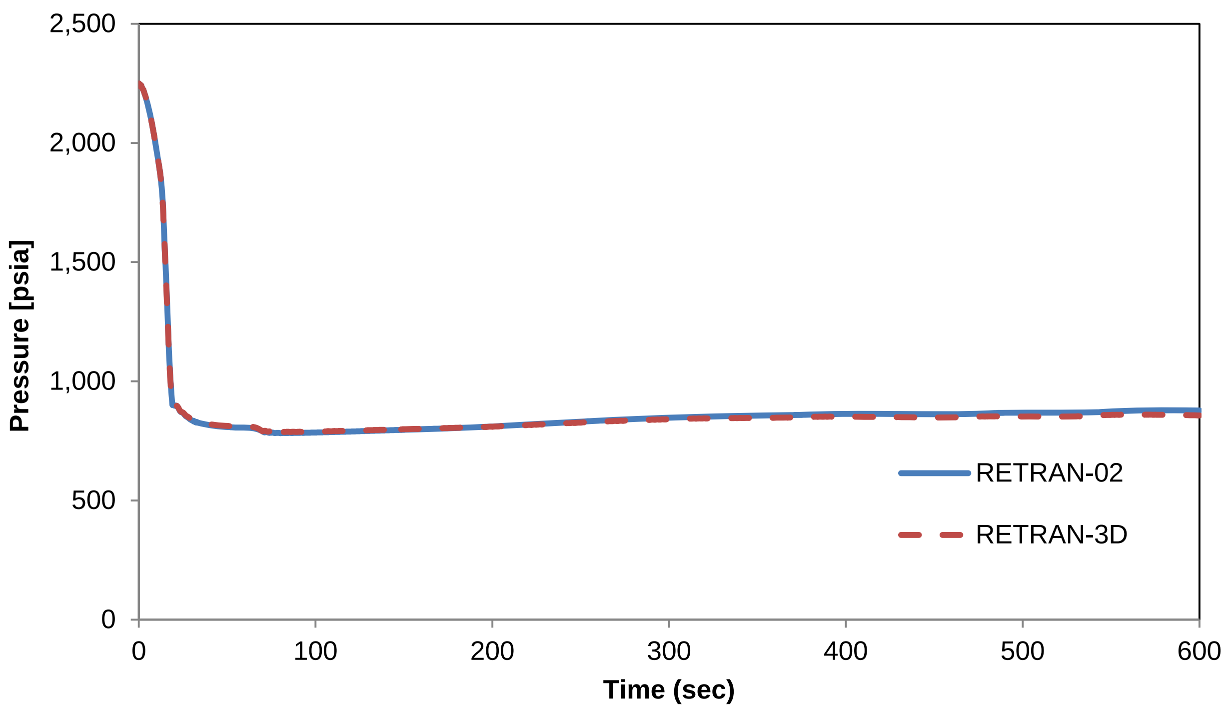


Figure 4.3-5 MSLB - Faulted Loop Core Inlet Boron Concentration

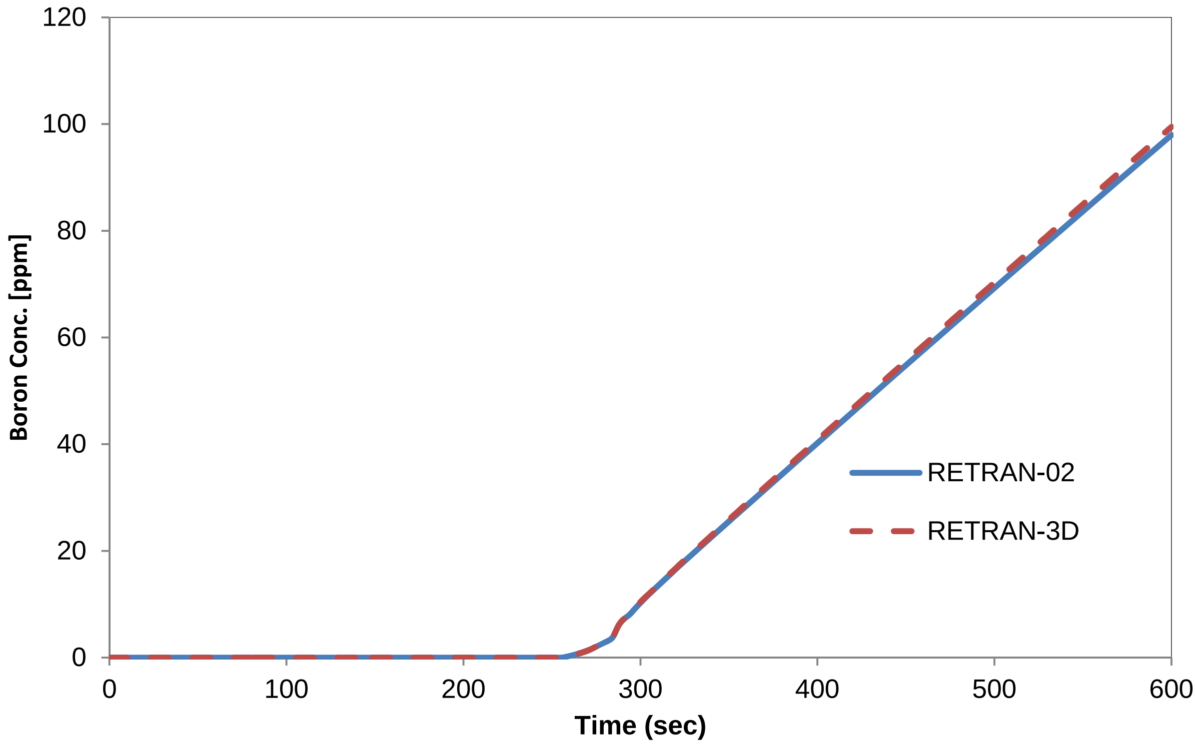
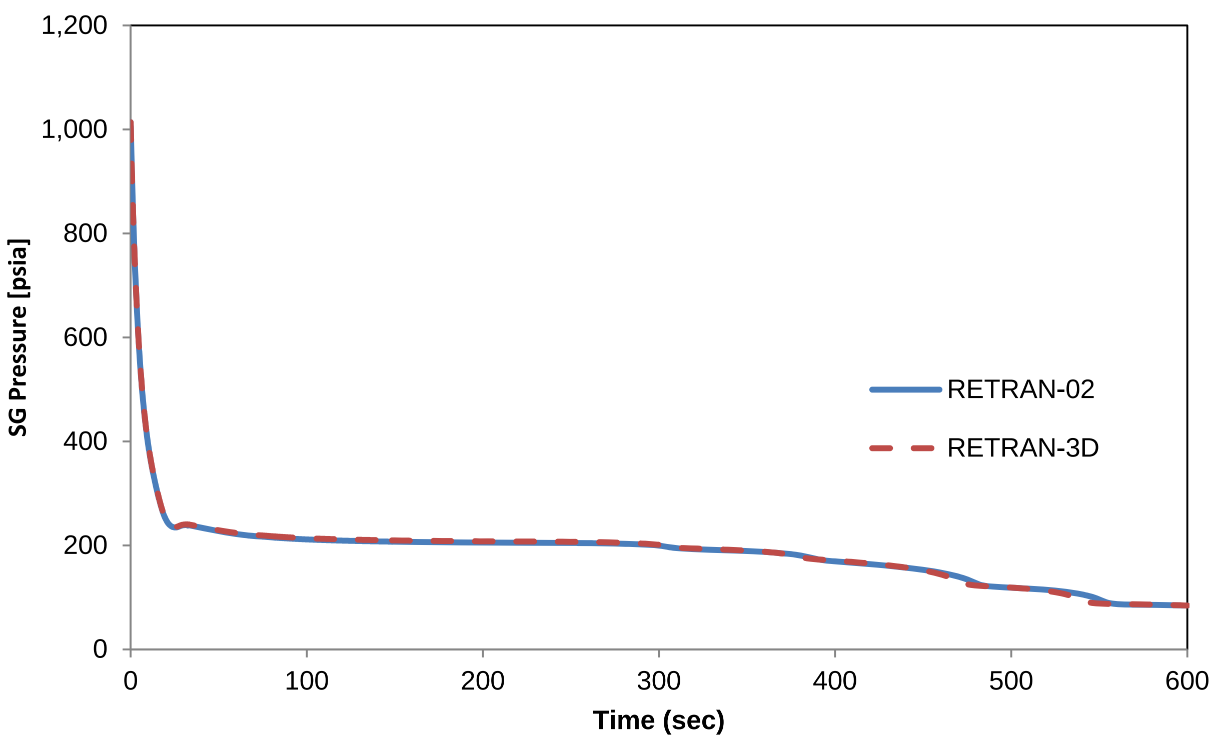


Figure 4.3-6 MSLB - Faulted Loop Steam Generator Pressure



4.4 Locked Rotor

The Locked Rotor event is defined as an instantaneous seizure of a Reactor Coolant Pump (RCP) rotor, rapidly reducing flow in the affected reactor coolant loop leading to a reactor trip on a low-flow signal from the Reactor Protection System. It is initiated by setting one RCP speed to zero as the system is operating at full power. During the Locked Rotor event at full power operation, the reduction in RCS flow results in degradation of the heat transfer between the fuel and the reactor coolant, and between the reactor coolant and the secondary coolant in the steam generator. The reduction in coolant velocity and heat transfer to secondary system cause the RCS average temperature ($T_{avg)}$ to increase, which results in an in-surge of coolant into the pressurizer, this causes an increase in RCS pressure. Fuel temperature also increases due to reduction of the heat transfer from the fuel, which could result in fuel damage if the fuel experiences a Departure from Nucleate Boiling (DNB).

The results for the Locked Rotor comparison analysis are presented in Figures 4.4-1 through 4.4-7. As shown, the responses of various parameters following a RCS Overpressure Locked Rotor event are almost identical when analyzed with the two codes. Reactor trip occurred as expected on low coolant loop flow at 1.0 seconds into the transient. Turbine trip occurred at 1.516 seconds. At about 3.0 seconds, the main feedwater isolation trip was actuated, along with reactor coolant pump trip in the unaffected loops. A peak cold leg pump exit pressure of 2753 psia was reached at 3.5 seconds. The maximum steam generator pressure of 1161 psia was reached at 17.9 seconds.

As shown, the responses of various parameters following a RCS-overpressurization Locked Rotor event show very good agreement when analyzed with the two codes. Table 4.4-1 shows the maximum value for the key parameters of this event.

Table 4.4-1 LR – Maximum Analysis Values

	RETRAN-02	RETRAN-3D
Pressurizer Pressure [psia]	2678	2667
Loop B SG Pressure [psia]	1161	1161
Loop B Cold Leg Pressure [psia]	2753	2734
RCS Average Temperature [ºF]	584.38	584.50

Figure 4.4-1 LR - Pressurizer Pressure

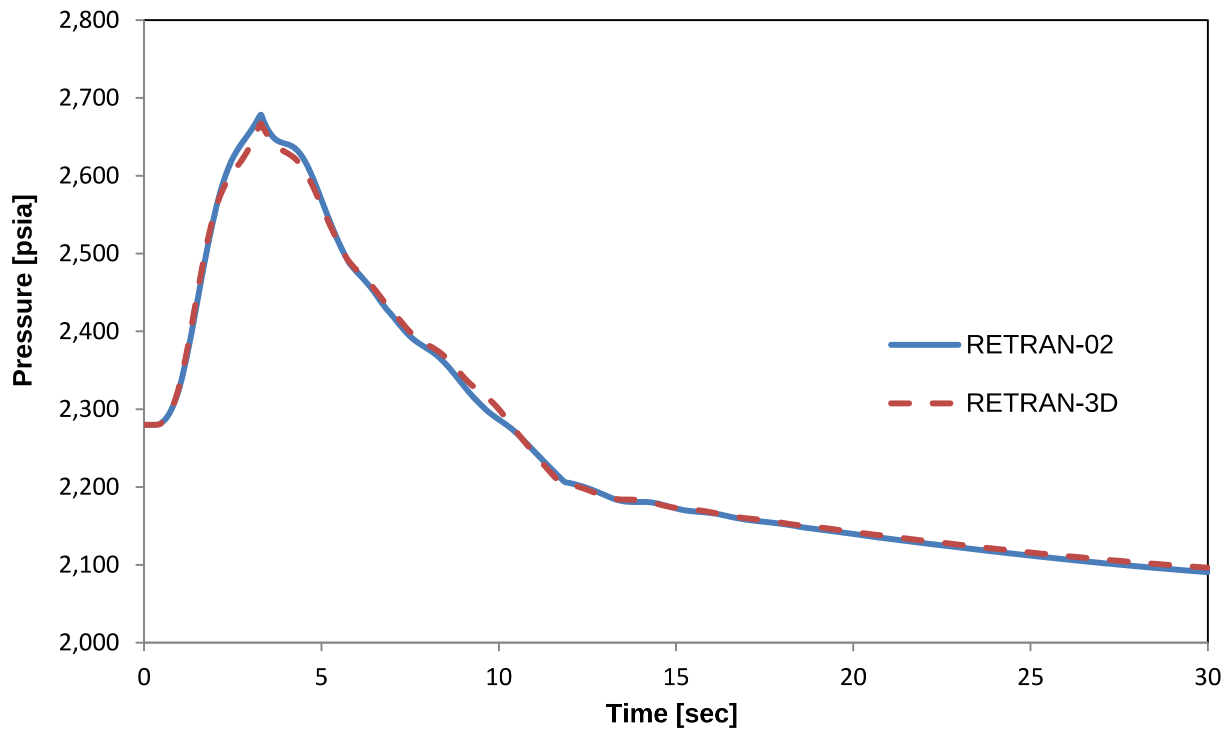


Figure 4.4-2 LR - Core Inlet Temperature

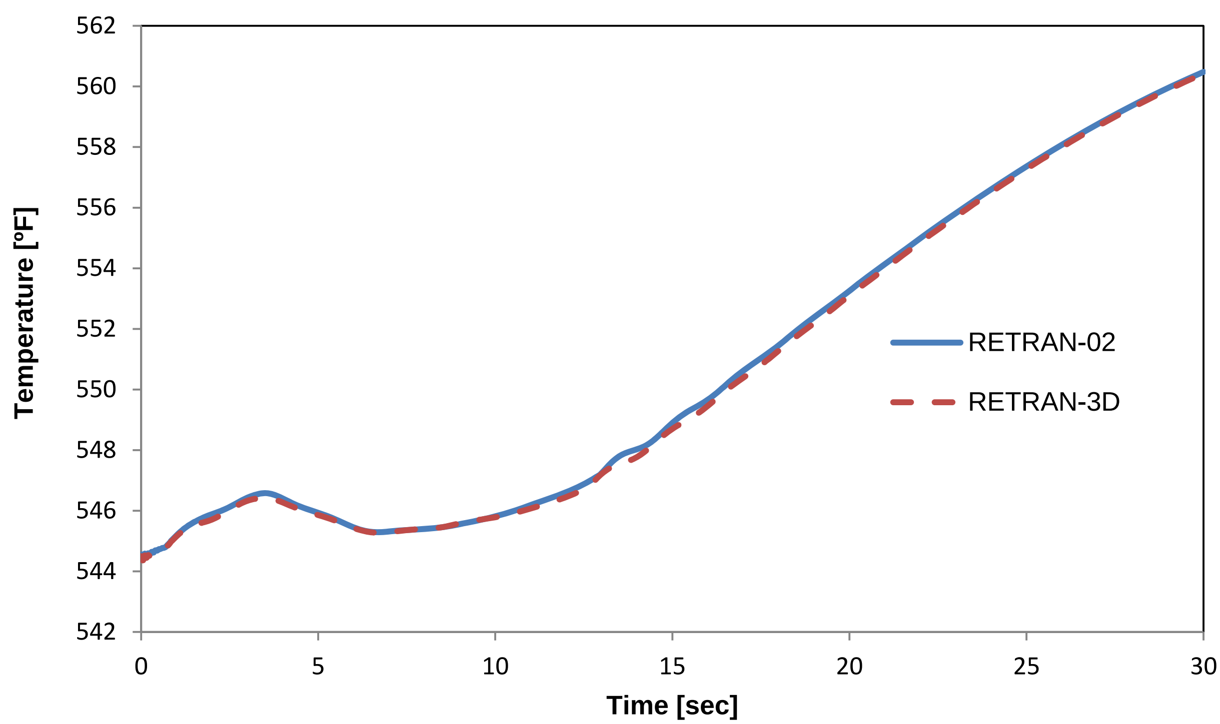


Figure 4.4-3 LR - Core Inlet Flow

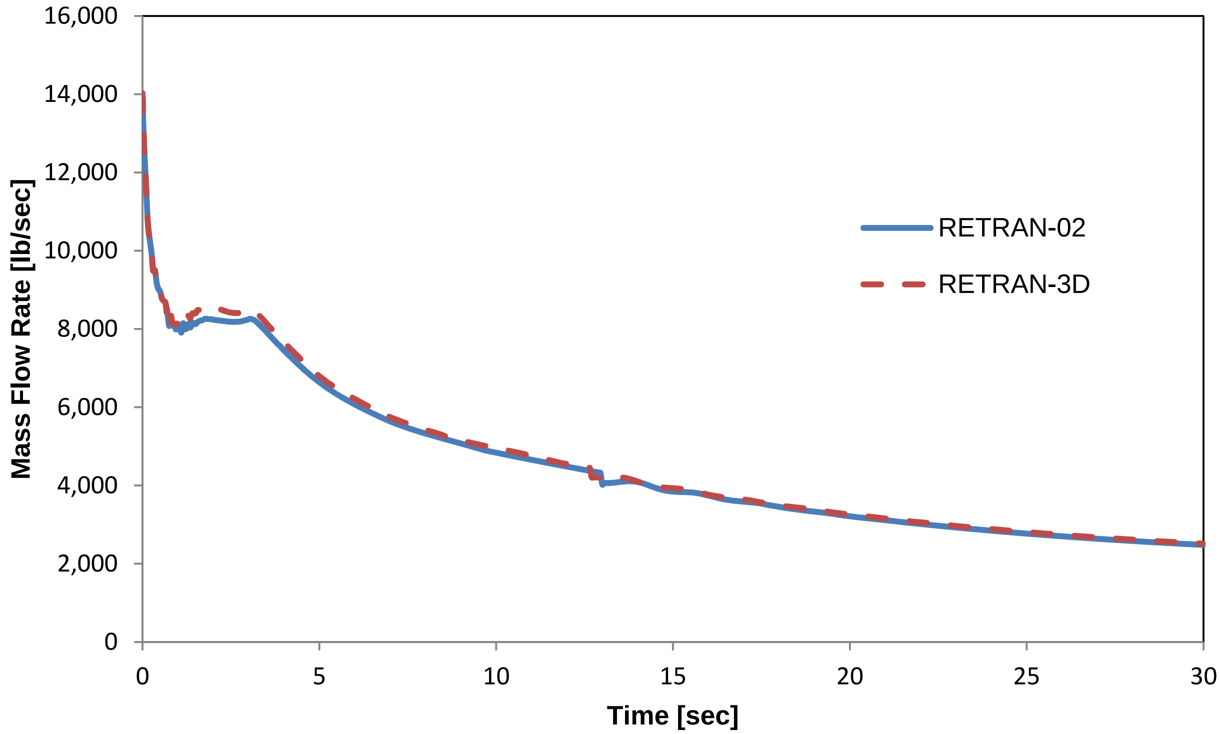


Figure 4.4-4 LR - Nuclear Power

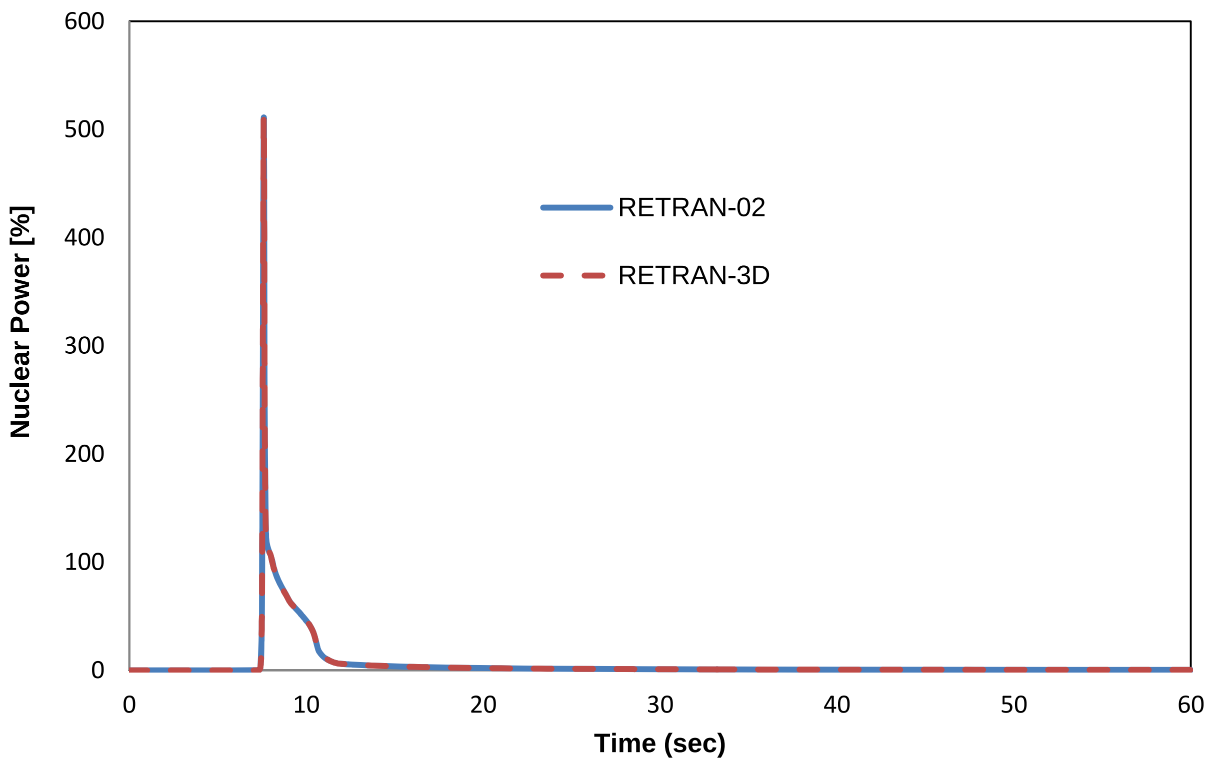


Figure 4.4-5 LR - Steam Gernerator Pressure in Unaffected Loop

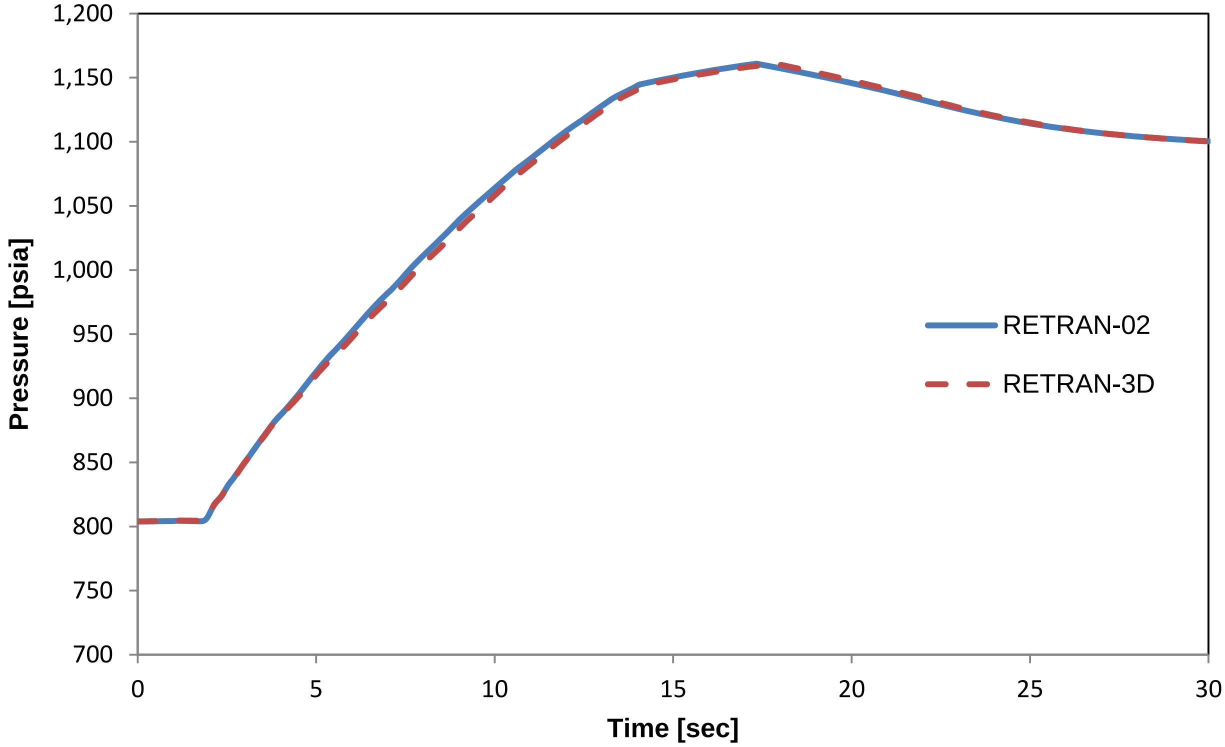


Figure 4.4-6 LR - Pressurizer Safety Valve Mass Flow Rate (Combined Three Valves)

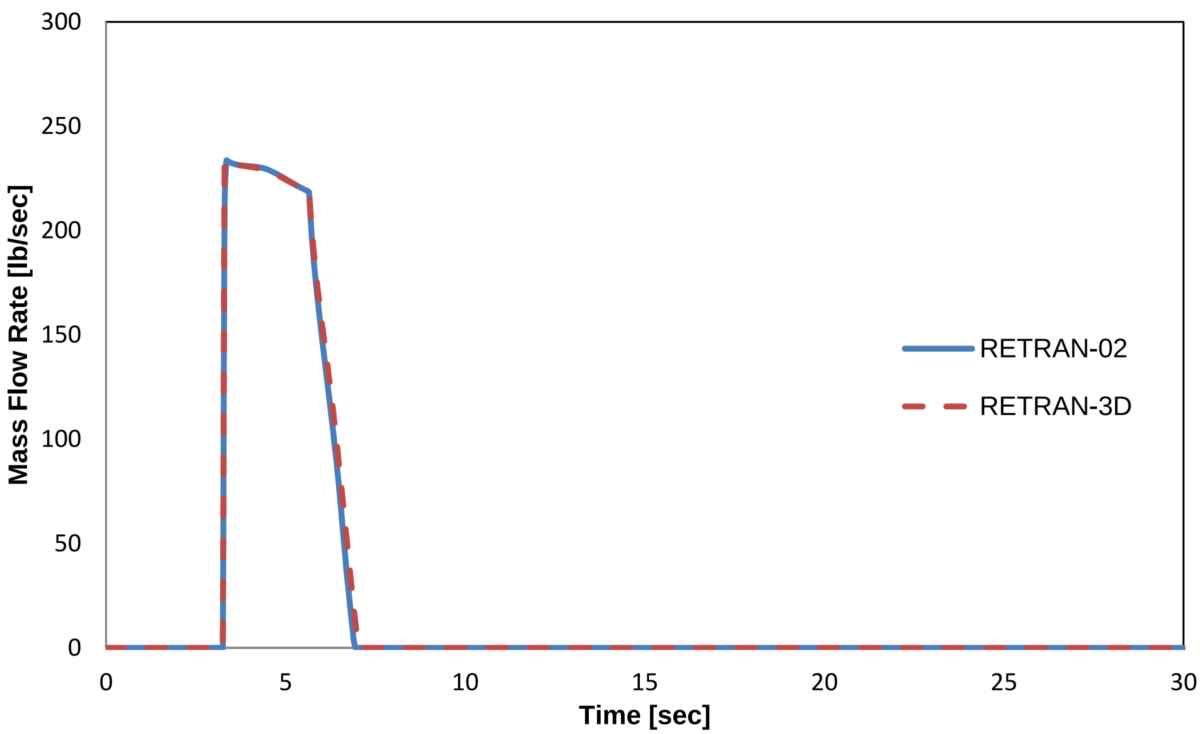
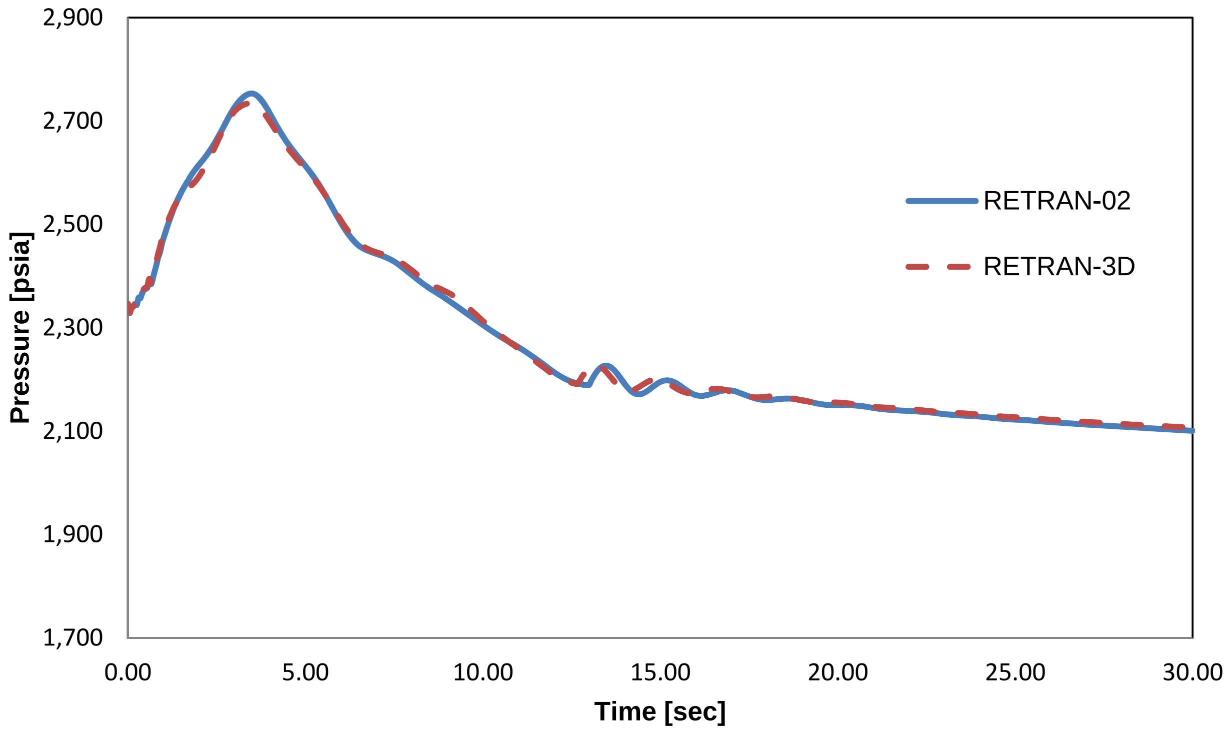


Figure 4.4-7 LR - Cold Leg Pressure



4.5 Rod Withdrawal from Subcritical

A RWSC accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of the rod cluster control assemblies, thereby producing a power excursion. Potential causes for the event include malfunctions of the reactor control and control rod drive systems and operator error.

The results for the RWSC comparison analysis are presented in Figures 4.5-1 through 4.5-6. As the result of the overpressurization RWSC event, the neutron flux overshoots the full-power nominal value, but this occurs for only a very short time. Prompt criticality occurs at 7.15 seconds, with the minor edits showing prompt power spike reaching a maximum value of about 511% power at 7.6 seconds. Power then abruptly decreases as Doppler feedback takes over. Hence, the energy release and the RCS temperature increases are relatively small (about 11ºF), as shown in Figure 4.5-6. The peak core average heat flux of approximately 73000 BTU/hr-ft^2 occurs at about 10.2 seconds, as shown in Figure 4.5-3, due to the beneficial effect of the inherent thermal lag in the fuel.

Figure 4.5-1 shows that the water shifted PSV setpoint of 2550 psia is never reached. A peak pressure of approximately 2510 psia occurs around 13.8 seconds. A peak cold leg pressure of 2567 psia (183 psia margin to the limit) occurs at about 14.35 seconds, as shown in Figure 4.5-5. Figure 4.5-4 shows that this event does not challenge the main steam system overpressure limit of 1210 psia.

As shown, the responses of various parameters following a RWSC overpressurization event show good agreement when analyzing with the two codes. Table 4.5-1 shows the maximum value for the key parameters of the overpressurization RWSC event.

Table 4.5-1 RWSC – Maximum Analysis Values

	RETRAN-02	RETRAN-3D
Nuclear Power [%]	510.609	510.184
Core Average Heat Flux [Btu/hr-ft^2]	72870.7	72715.4
Pressurizer Pressure [psia]	2508.2	2508.2
Loop A SG Pressure [psia]	1036.3	1036.4
Loop A Cold Leg Pressure [psia]	2566.6	2567.6
RCS Average Temperature [ºF]	558.45	558.39

Figure 4.5-1 RWSC - Pressurizer Pressure

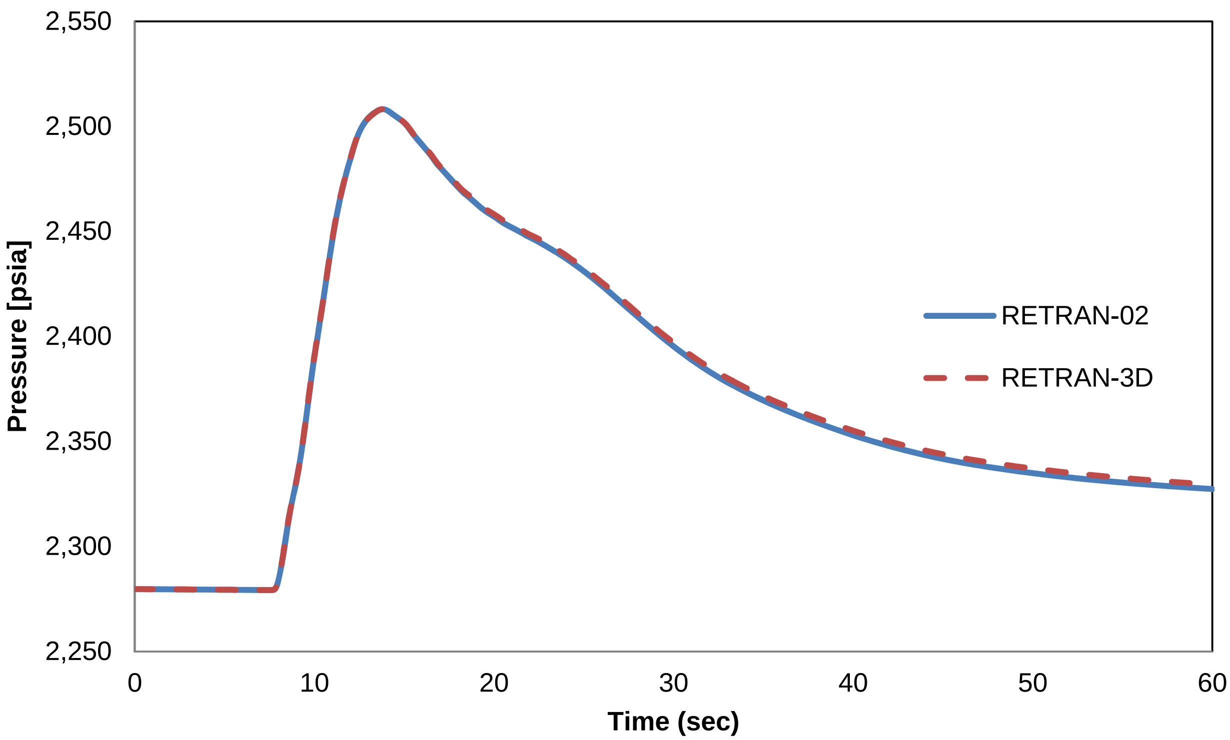


Figure 4.5-2 RWSC - Nuclear Power

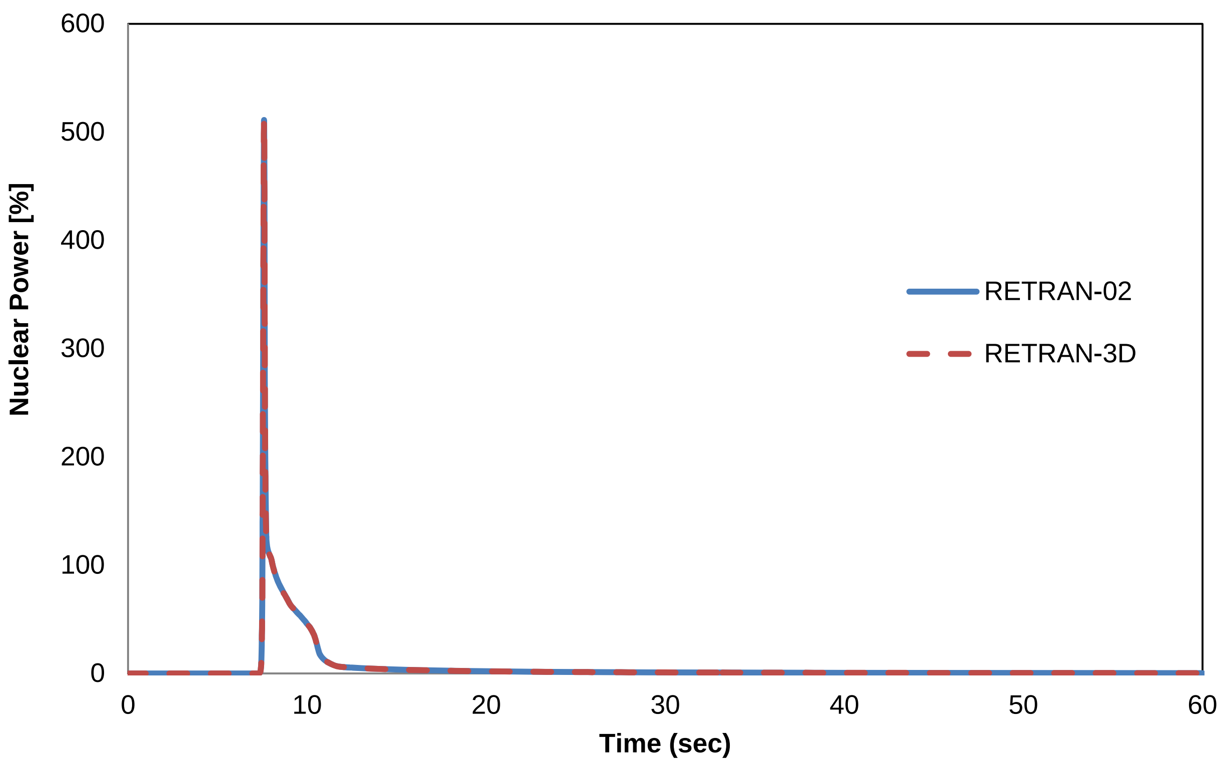


Figure 4.5-3 RWSC - Core Average Heat Flux

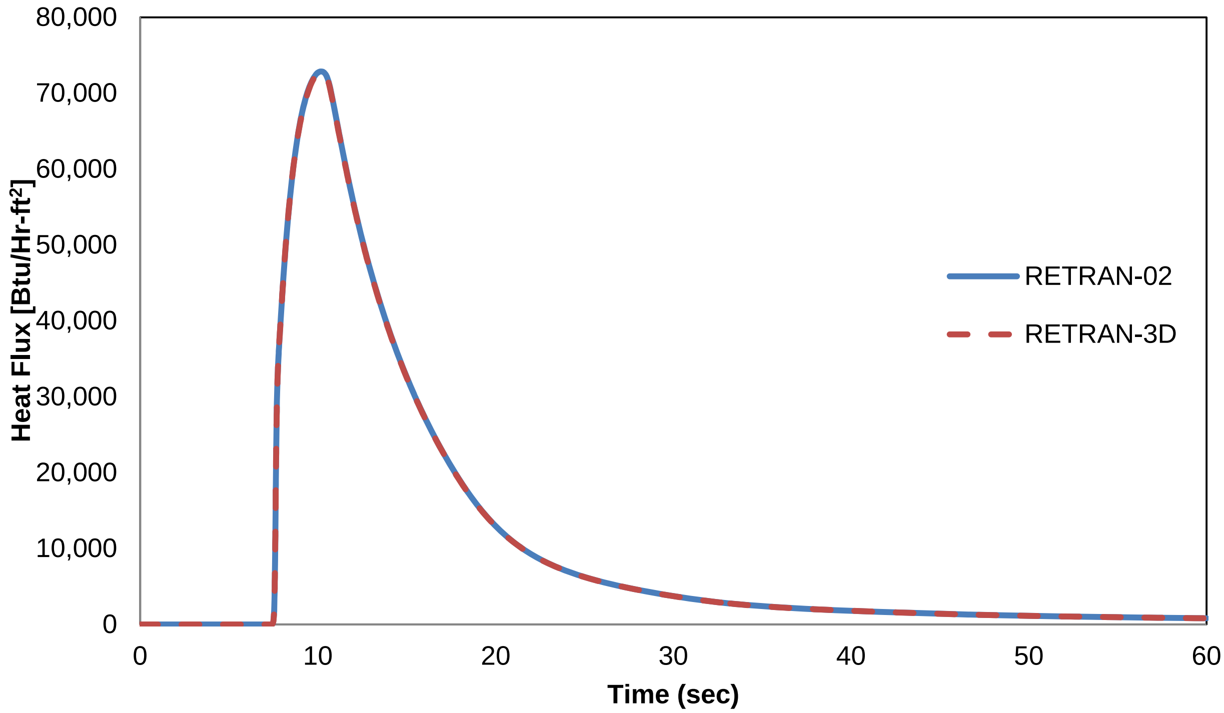


Figure 4.5-4 RWSC - Loop A Steam Generator Pressure

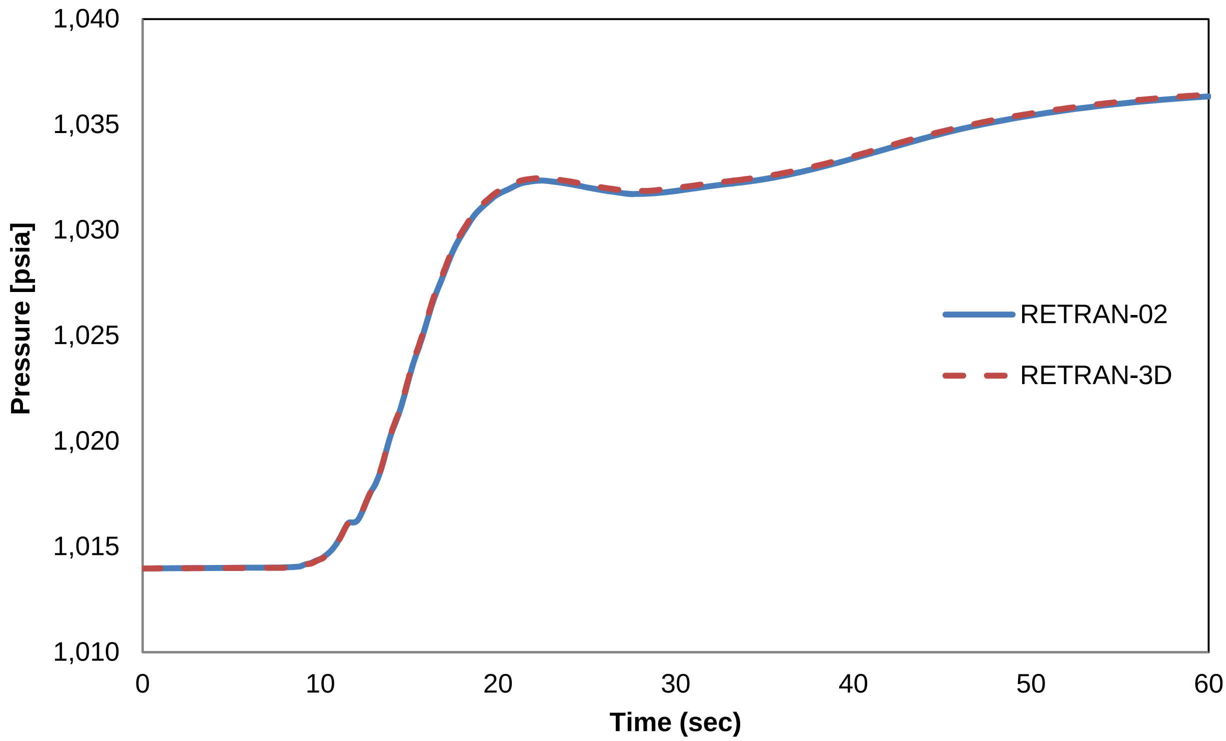


Figure 4.5-5 RWSC - Loop A Cold Leg Pressure

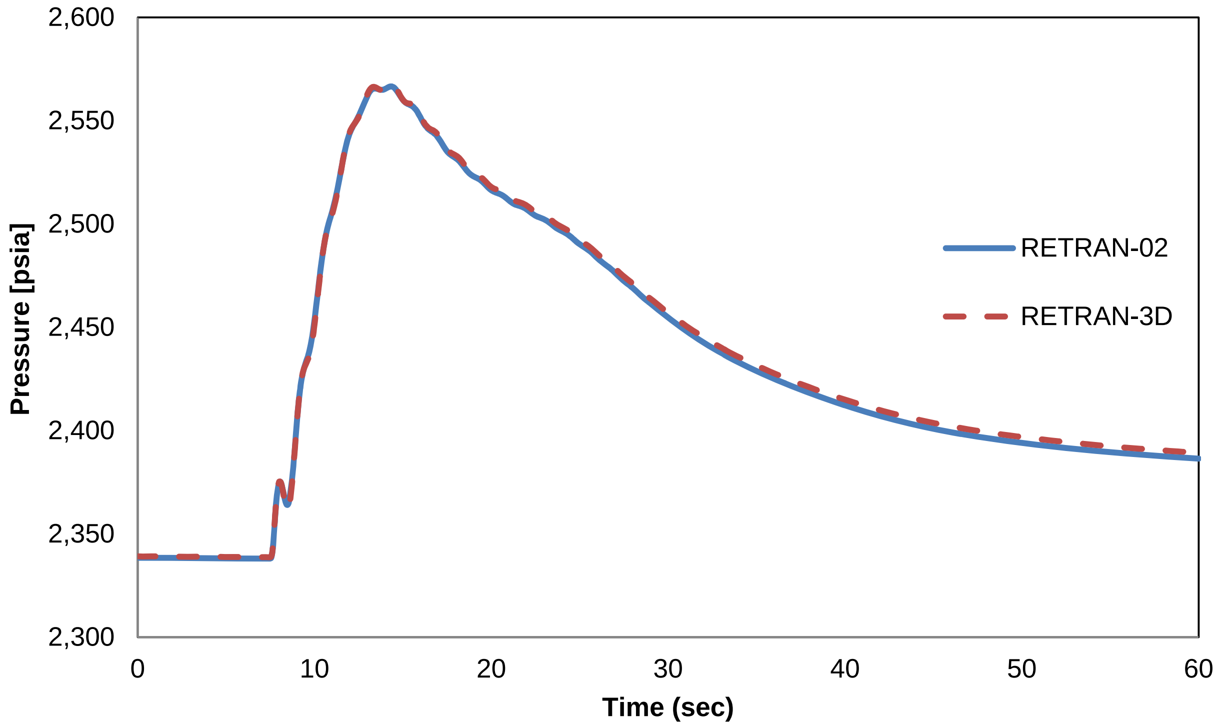
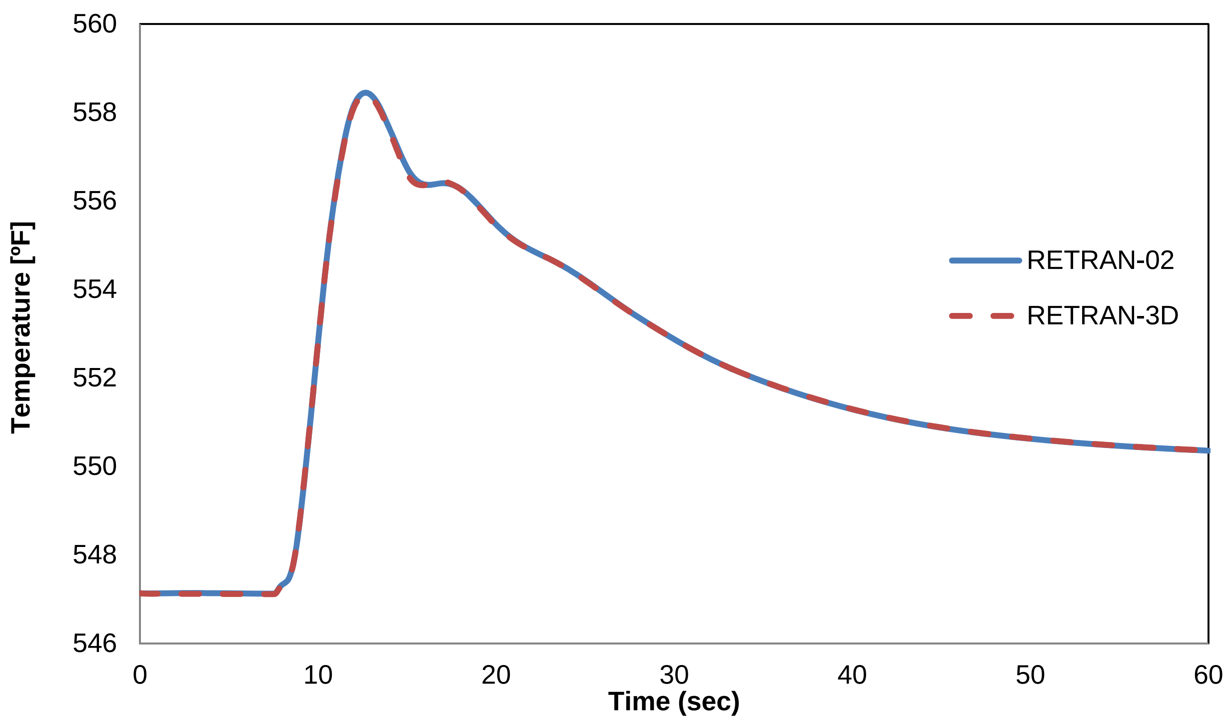


Figure 4.5-6 RWSC - RCS Average Tempearture



5.0 Conclusions

This report presents demonstration transient analysis performed to benchmark the transition from the RETRAN-02 to RETRAN-3D computer code. Analyses have been performed by Dominion to compare results calculated by the RETRAN-02 computer code and the RETRAN-3D computer code in RETRAN-02 mode. The analyses presented in this report are a representative sample of transients to ensure that all aspects of the code and models are considered. Based on the benchmark presented herein, it is concluded that the RETRAN-3D computer code may be used in place of the RETRAN-02 computer code for UFSAR transient analyses.

6.0 References

1. Letter from S. A. Richards (USNRC) to G. L. 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, (TAC No. MA4311).” 2001-25-05, ADAMS Accession Number ML010470342.

APPENDIX 10
RETRAN Benchmarking Information
Supporting Application of VEP-FRD-41 to MPS3

TABLE OF CONTENTS

1.0 Introduction and Summary

1.1 Introduction

Topical report VEP-FRD-41-P-A, “VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code,” (Reference 1) details the Dominion methodology for Nuclear Steam Supply System (NSSS) non-LOCA transient analyses. This methodology encompasses the non-LOCA licensing analyses required for the Condition I, II, III, and IV transients and accidents addressed in the Final Safety Analysis Report (FSAR). The VEP-FRD-41-P-A methods are also used in support of reload core analysis. In addition, this capability is used to perform best-estimate analyses for plant operational support applications. The material herein supports the applicability assessment of the VEP-FRD-41-P-A methods to Millstone Power Station Unit 3 (MPS3) for the stated applications.

1.2 Summary

This attachment provides a description of the RETRAN base model for MPS3 and results of benchmarking analyses using this model. The MPS3 model was developed in accordance with the methods in VEP-FRD-41-P-A, with certain noding changes noted below. This assessment confirms the conclusion that the Dominion RETRAN methods, as documented in topical report VEP-FRD-41-P-A, are applicable to MPS3 and can be applied to MPS3 licensing analysis for reload core design and safety analysis. Dominion analyses of MPS3 will employ the modeling in VEP-FRD-41-P-A, as augmented with the noding changes listed below. Thus, VEP-FRD-41-P-A, as augmented, is the Dominion methodology for analyses of non-LOCA NSSS transients for MPS3.

The MPS3 RETRAN base model contains the following alterations in noding with respect to the modeling that is documented in VEP-FRD-41-P-A.

a) The MPS3 model explicitly models the safety injection (SI) accumulators.
b) The MPS3 model has separate volumes for the steam generator inlet and outlet plenums.
c) The MPS3 model includes cooling paths between downcomer and upper head.

2.0 MPS3 RETRAN Model

The MPS3 RETRAN-3D Base Model and associated model overlays are developed using Dominion analysis methods described in the Dominion RETRAN topical report (Reference 1). The Dominion analysis methods are applied consistent with the conditions and limitations described in the Dominion topical report and in the applicable NRC Safety Evaluation Reports (SERs).

The MPS3 Base Model noding diagram for a representative loop is shown on Figure 2-1. Volume numbers are circled, junctions are represented by arrows, and the heat conductors are shaded. This model simulates all four reactor coolant system (RCS) loops and has a single-node steam generator (SG) secondary side, consistent with Dominion methodology. The SG primary nodalization includes 10 steam generator tube volumes and conductors. There is a multi-node SG secondary overlay that can be added to the Base Model for sensitivity studies although none of the analysis results presented herein utilize this overlay.

In addition to the base MPS3 model, an overlay deck is used to create a split reactor vessel model to use when analyzing Main Steam Line Break (MSLB) events, consistent with Dominion methodology. This overlay adds volumes to create a second, parallel flow path through the active core from the lower plenum to the upper plenum such that RCS loop temperature asymmetries can be represented. This noding is consistent with the method described in VEP-FRD-41-P-A. A noding diagram of the split reactor vessel is shown on Figure 2-2.

The base MPS3 model noding is virtually identical to the Surry (SPS) and North Anna (NAPS) models with the exception of some minor noding differences listed as follows.

a) The MPS3 model explicitly models the SI accumulators.
b) The MPS3 model has separate volumes for the SG inlet and outlet plenums.
c) The MPS3 model includes cooling paths between downcomer and upper head.

The SI accumulators are part of the MPS3 model because injection from the accumulators occurs in the current FSAR analysis for MSLB. The use of separate volumes for the inlet and outlet should have little effect on transient response since the fluid temperature in these volumes is generally the same as the connecting RCS piping. The cooling paths are included to appropriately model upper head T-cold conditions.

The Dominion models, including the MPS3 model, have some differences compared to the vendor RETRAN model that was used to perform the current FSAR analyses. Table 2-1 and the subsequent text discussion provide an overview of these differences. Additional details concerning differences between the Dominion MPS3 and FSAR RETRAN models are discussed in the benchmarking analyses in Section 4.

A description of the Dominion RETRAN methodology is provided in Reference 1, where specific model details are discussed in Sections 4 and 5 of that reference.

Table 2-1 RETRAN Model Comparison of Key Characteristics

Parameter	**Dominion**	**FSAR**
Code Version:	RETRAN-3D in “02 mode”	RETRAN-02
Noding:		
Reactor Vessel	Single flow path (special split core overlay for MSLB only)	Multiple parallel flow paths
Steam Generator	Single node secondary. Five axial levels (10 nodes) for SG tubes primary side. Local Conditions Heat Transfer model available for loss of heat sink events.	Multi-node secondary.
Reactivity Model		
Doppler Feedback	Doppler temperature coefficient that is a function of T_{FUEL}.	Doppler-only power coefficient and a Doppler temperature coefficient effect driven by moderator temperature.
Moderator Feedback	Moderator temperature coefficient	Moderator density coefficient
Decay Heat	ANS 1979 Standard U-235 with 1500 day burn. Q = 190 MeV/fission. 1.0 Decay Heat Multiplier Bounds additional 2σ uncertainty	ANS 1979 Standard Bounds additional 2σ uncertainty

Figure 2-1 MPS3 Base Model Nodalization Diagram

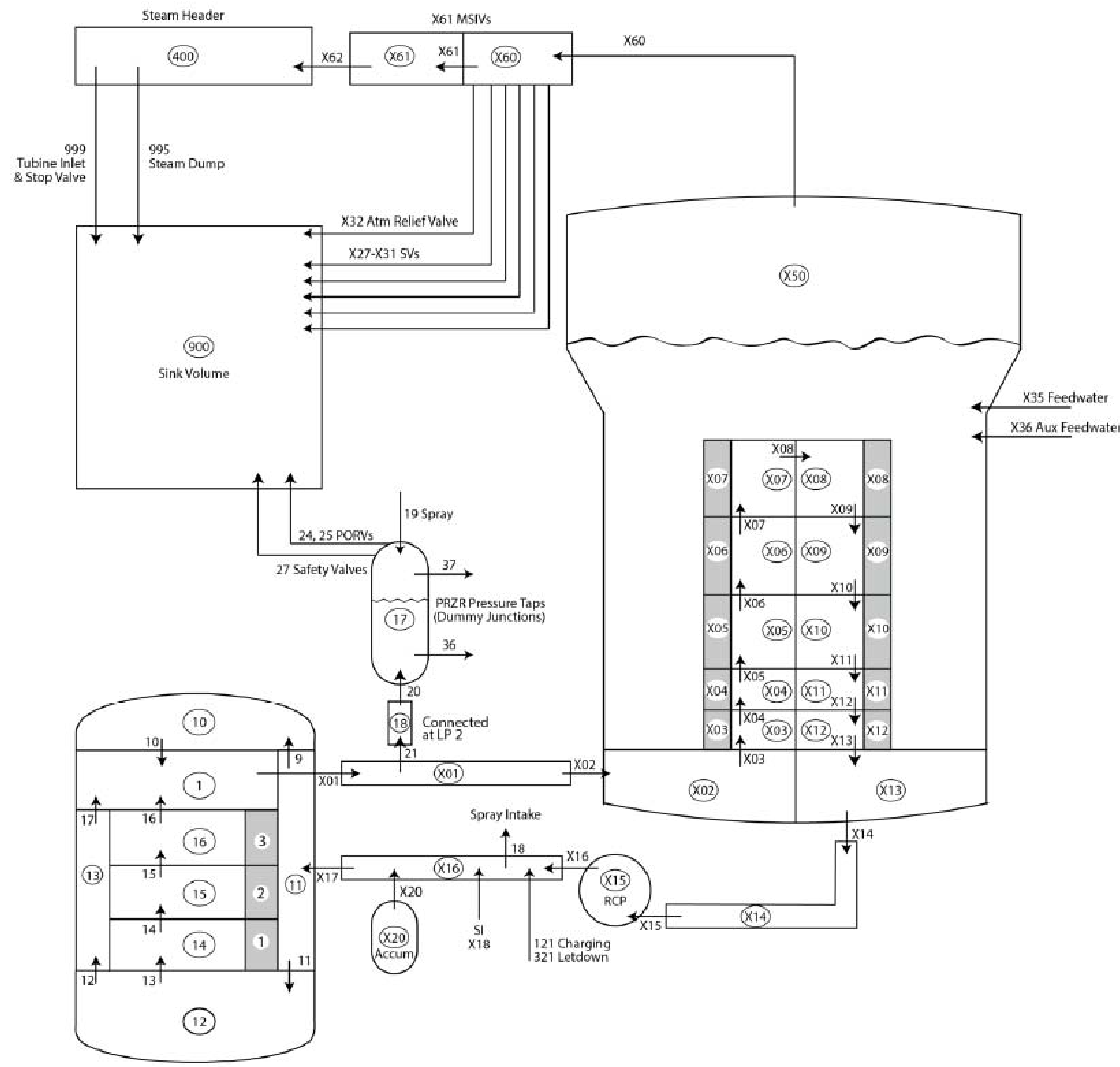
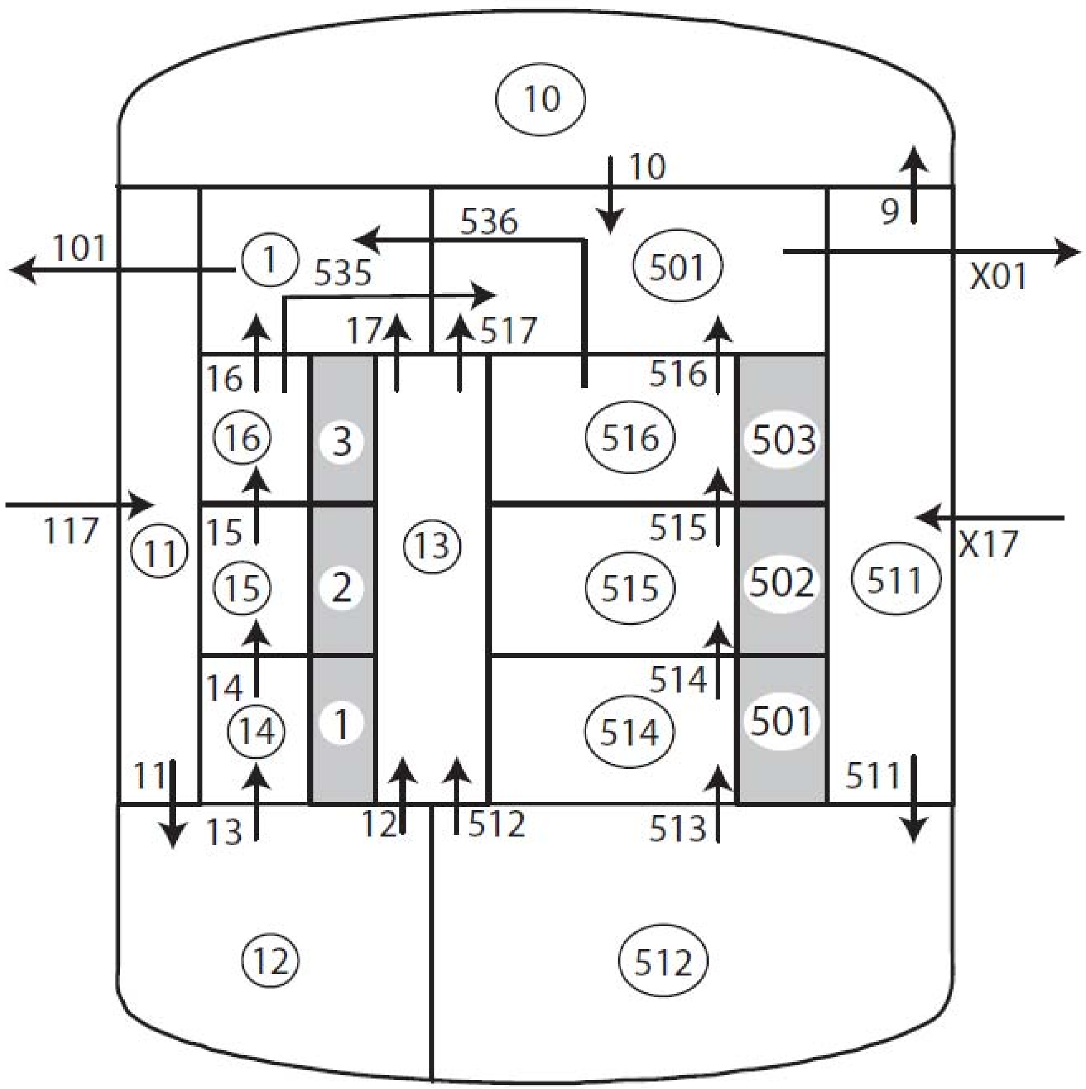


Figure 2-2 MPS3 Split Vessel Nodalization



3.0 Method of Analysis

Validation of the Dominion MPS3 RETRAN method involves comparison of RETRAN analyses to the MPS3 FSAR analysis of record (AOR) for select events. The Dominion analyses presented herein are not replacements for the existing AORs. These events represent a broad variation in behavior (e.g. RCS heatup, RCS cooldown/depressurization, reactivity excursion, loss of heat sink, etc.), and demonstrate the ability to appropriately model key phenomena for a range of transient responses. The transients selected for comparison with their corresponding MPS3 FSAR section are provided in Table 3-1. For each transient, an analysis is performed using the Dominion MPS3 RETRAN model and compared with the current FSAR analysis. Initial conditions and inputs are established for each benchmark to provide an adequate comparison of specific transient behavior.

Table 3-1 Transients Analyzed for FSAR Comparison

Transient	**MPS3 FSAR Section**
Main Steam Line Break	15.1.5
Loss of Load/Turbine Trip	15.2.3
Loss of Normal Feedwater	15.2.7
Locked Rotor	15.3.3
Control Rod Withdrawal at Power	15.4.2
Main Feedwater Line Break	15.2.8

4.0 Benchmarking Analysis Results

A summary for each transient comparison is presented in the following sections. Included in each section is an input summary identifying key inputs and assumptions along with differences from FSAR assumptions. A comparison of the results for key parameters is provided with an explanation of key differences between the Dominion and FSAR cases.

4.1 Loss of Load/Turbine Trip

The Loss of Load/Turbine Trip (LOL) event is defined as a complete loss-of-steam load and turbine trip from full power without a direct reactor trip, resulting in a primary fluid temperature rise and a corresponding pressure increase in the primary system. This transient results in degraded steam generator heat transfer, reactor coolant heatup and pressure increase following a manual turbine trip.

The LOL transient scenario presented here was developed to analyze primary RCS overpressurization. It is initiated by decreasing both the steam flow and feedwater flow to zero immediately after a manual turbine trip. The input summary is provided in Table 4.1-1.

Table 4.1-1 LOL Input Summary

Parameter	Value	Notes
Initial Conditions		
Core Power (MW)	3723	Includes 2% uncertainty
RCS Flow (gpm)	363,200	Thermal Design
Vessel T_{AVG} (F)	576.5	Low Tavg plus uncertainty
Pressurizer Pressure (psia)	2200	Includes -50 psia uncertainty
Pressurizer Level (%)	52.5	Low Tavg Target plus uncertainty
SG Level (%)	50.0	Nominal
SG tube plugging (%)	10	Maximum
Pump Power (MW/Pump)	5.0	Maximum
Assumptions/Configuration		
Reactor trip	-	only Hi Pzr Pressure is active
Automatic rod control	-	Not credited
Pressurizer sprays, PORVs	-	Not credited
Main steam dumps, SG PORV	-	Not credited
AFW flow	-	Not credited
Reactivity Parameters		
Doppler Reactivity Feedback	Least Negative	
Moderator Feedback	Most Positive	

Results - LOL

Pressure in the RCS increases during a LOL due to degraded heat transfer in the steam generator and is alleviated only when the pressurizer safety valves (PSV) open as well as the main steam safety valves (MSSV). The pressurizer pressure response is shown on Figure 4.1-1, RCP outlet pressure in Figure 4.1-2, and the peak RCS pressure values are listed in Table 4.1-2. The Dominion case predicts a pressurizer pressure and RCP outlet pressure response that agrees very well with the FSAR results past the point of peak RCS pressure.

Following the initial decrease in primary system pressure, the FSAR pressure levels out where the Dominion case results continue to decrease. The difference is due to differing secondary safety valve modeling in the vendor model, specifically in that the Dominion model includes the modeling of blowdown in the main steam safety valves and the vendor model does not. Hence, more energy is removed through the secondary system in the Dominion case once the main steam safety valves actuate than is removed from the secondary system in the vendor model.

Figure 4.1-3 shows the power response is nearly identical both before and after the reactor trip on high pressurizer pressure and control rod insertion. The Dominion case trips slightly earlier than the FSAR data because of the higher RCS pressurization rate.

The Dominion model vessel inlet temperature, Figure 4.1-4, and coolant average temperature, Figure 4.1-5, agrees in trend and rate of increase although the response lags the FSAR response before the inlet temperature peaks at a slightly lower value. This indicates that the FSAR steam generator heat transfer degrades sooner than what is predicted by Dominion model and is attributed to the difference expected between the use of a multi-node steam generator (MNSG) in the FSAR model and the single-node steam generator (SNSG) model employed in the Dominion model. Overall, both the Dominion model and FSAR models exhibit similar trends in the temperature responses and the differences have no effect on peak RCS pressure.

Table 4.1-2 LOL RCS Overpressure Results

Parameter	Dominion	FSAR
Sequence of Events:		
High Pressurizer Pressure Setpoint Reached (sec)	5.6	6.2
Peak RCS Pressure (sec)	9.2	9.9
Peak RCS Pressure (psia)	2705	2725

Figure 4.1-1 LOL - Pressurizer Pressure

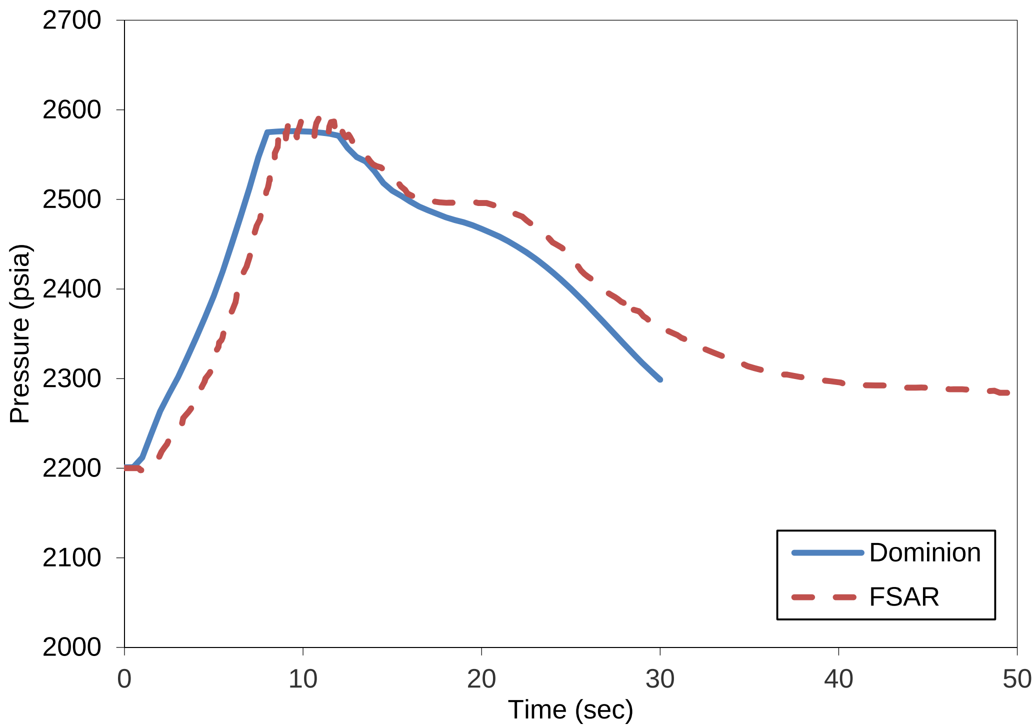


Figure 4.1-2 LOL – RCP Outlet Pressure

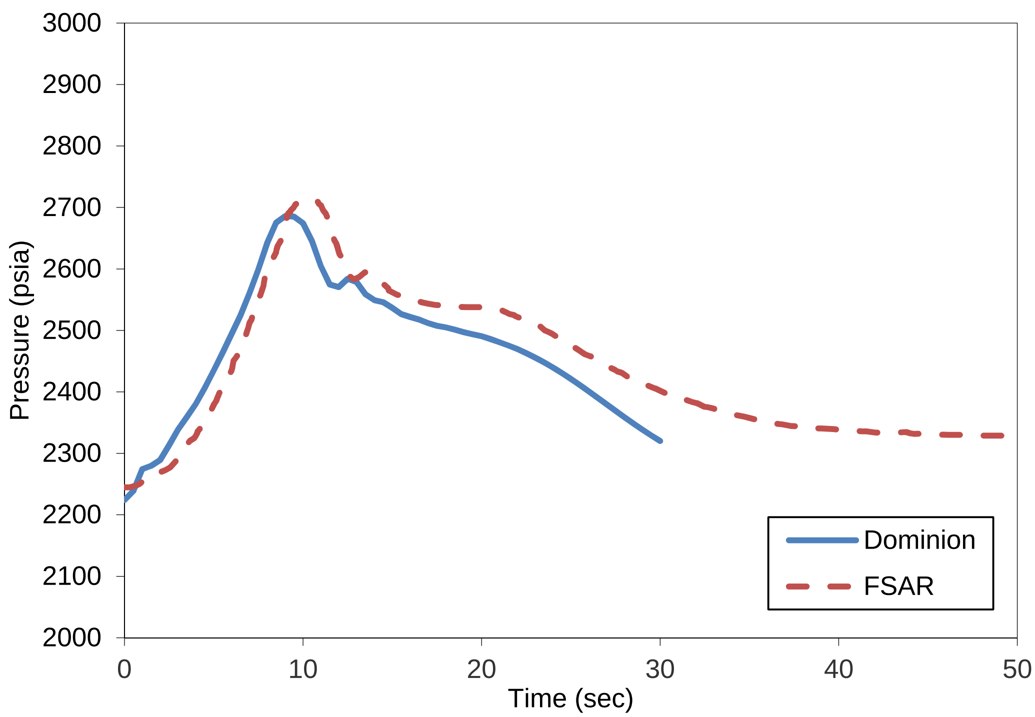


Figure 4.1-3 LOL – Nuclear Power

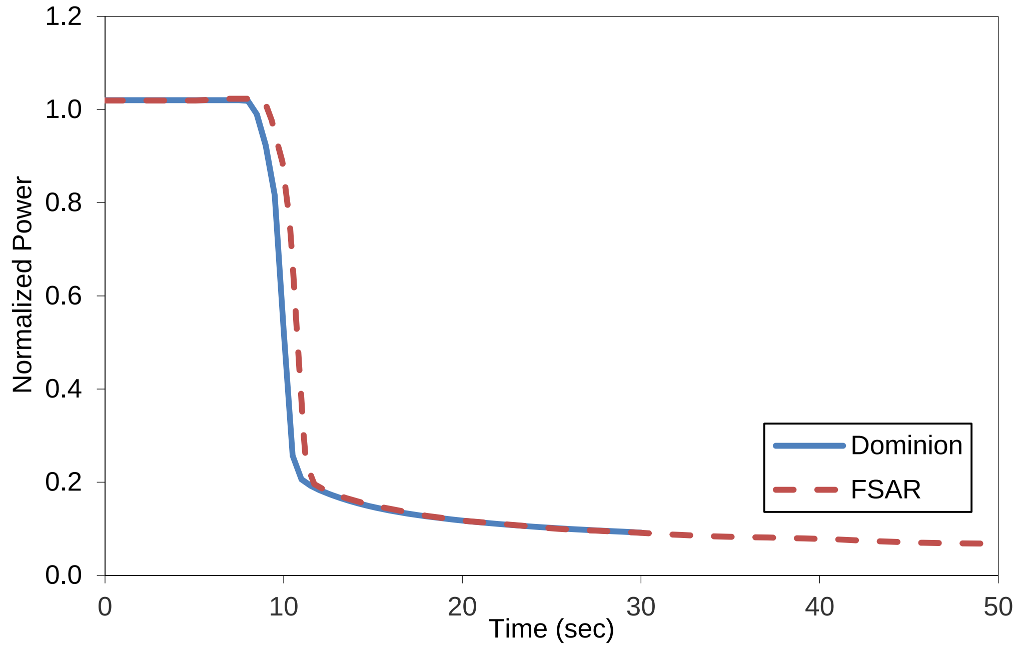


Figure 4.1-4 LOL– Vessel Inlet Temperature

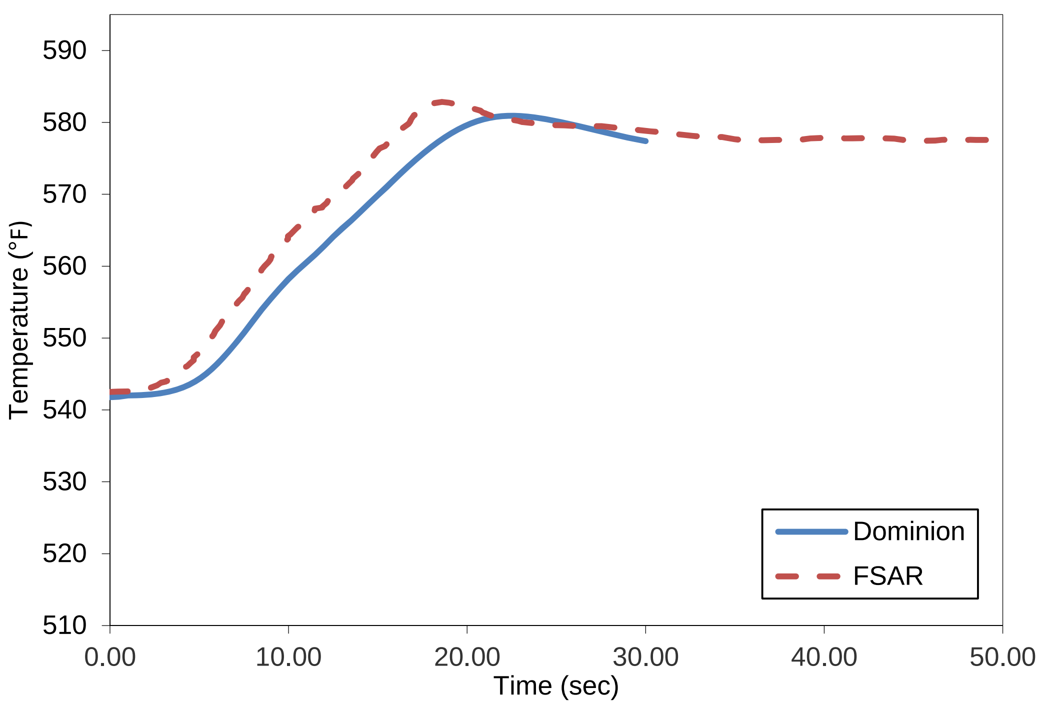
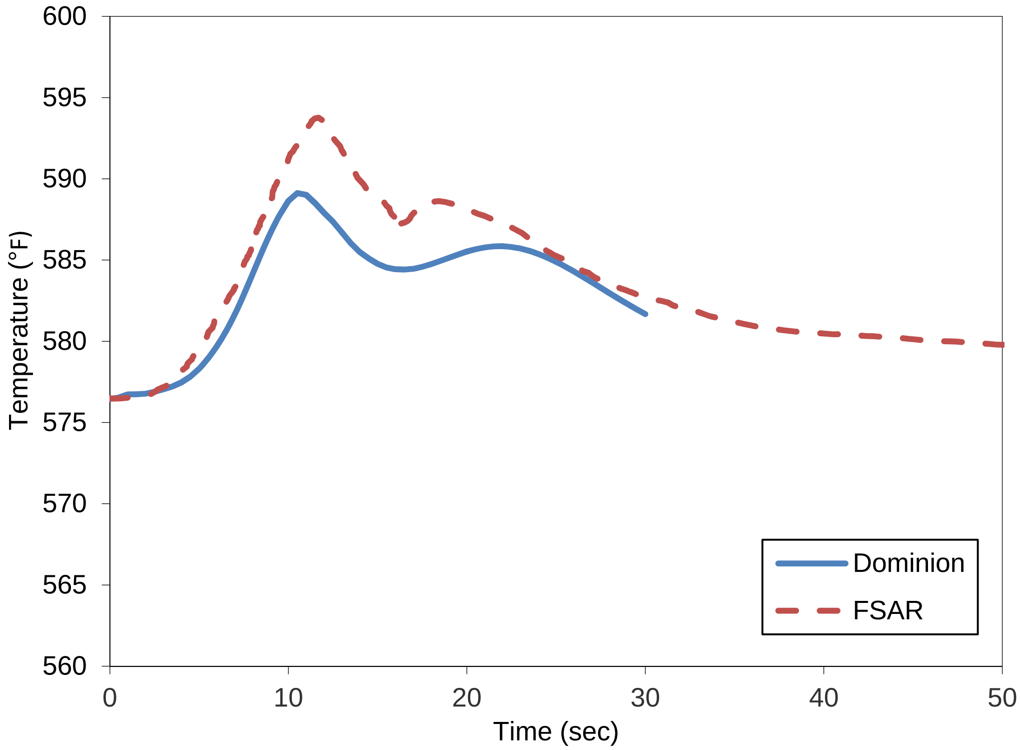


Figure 4.1-5 LOL – Vessel Average Temperature



Summary - LOL

The Dominion MPS3 analysis provides results that are similar to the FSAR analysis for the LOL event. The RCS peak pressures are essentially the same although the pressure diverge somewhat later in the event after pressure relief begins due to differences in MSSV modeling. There are small differences in the RCS temperature response due to differences in the SG models, however, this has no effect on the RCS peak pressure. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

4.2 Locked Rotor

The Locked Rotor / Shaft Break (LR) event is defined as an instantaneous seizure of a Reactor Coolant Pump (RCP) rotor, rapidly reducing flow in the affected reactor coolant loop leading to a reactor trip on a low-flow signal from the Reactor Protection System. The event creates a rapid expansion of the reactor coolant and reduced heat transfer in the steam generators, causing an insurge to the pressurizer and pressure increase throughout the reactor coolant system (RCS).

The LR transient scenario presented here was developed to analyze primary RCS overpressurization. It is initiated by setting one RCP speed to zero as the system is operating at full power. The reactor coolant low loop flow reactor trip is credited, with a setpoint of 85% of the initial flow. The input summary is provided in Table 4.2-1. Most of the input parameters are the same as those used in the FSAR Chapter 15 analyses.

Table 4.2-1 LR Input Summary

Parameter	Value	Notes
Initial Conditions		
Core Power (MW)	3723	Includes 2% uncertainty
RCS Flow (gpm)	363,200	Thermal Design Flow
Vessel T_{AVG} (F)	594.5	Nominal + 5°F
Pressurizer Pressure (psia)	2300	Includes +50 psia uncertainty
Pressurizer Level (%)	64	Nominal
SG Level (%)	50	Nominal
Assumptions/Configuration		
Reactor trip	-	Only Low RCS Loop Flow is credited
Automatic rod control	-	Not credited
Pressurizer sprays, PORVs	-	Not credited
Main steam dumps, SG PORV	-	Not credited
AFW flow	-	Not credited
SG tube plugging (%)	10[1]	Max value
Reactivity Parameters		
Doppler Reactivity Feedback	Most Negative	Dominion model adjusted to use FSAR Doppler Power Coeffcient
Moderator Feedback	Most Positive	

[1] Original benchmark case inadvertently assumed 0% SG tube plugging

Results – LR RCS Overpressure Case

Pressure in the RCS increases during a LR event due to degraded heat transfer in the steam generator and is alleviated only when the pressurizer safety valves (PSV) open. The magnitude of the Dominion model pressure response both in the reactor vessel lower plenum, Figure 4.2-1, and at the RCP exit, Figure 4.2-2, is greater than the FSAR model response, while following the same trends as the FSAR data. At the limiting point in the transient response, the Dominion model conservatively predicts a pressure approximately 63 psi greater than the FSAR model in the reactor vessel lower plenum. The difference between the Dominion model and FSAR model's peak responses is the same at the RCP exit as in the lower plenum.

The Dominion faulted loop flow response (Figure 4.2-3) and unfaulted loop flow response (Figure 4.2-4) are in good agreement with the FSAR model response up to or just beyond the point of rod insertion . Following reactor trip there is some divergence in the unfaulted loop flow trends, which are consistent with the core heat flux predictions and assumed minor differences in the loop friction losses between the Dominion and FSAR models. With respect to the faulted loop flow response, the maximum reverse flow seen in the FSAR model is slightly greater than seen in the Dominion model, which is also attributed to small differences in the loop friction losses between the Dominion and FSAR models.

For the total core inlet flow response (Figure 4.2-5), the Dominion model predicts a lower flow than the FSAR model for approximately the first 4 seconds of the transient. After 4 seconds the FSAR and Dominion model core flow responses cross and the Dominion model predicts a slightly higher core flow rate. The limiting point in the transient occurs prior to 4 seconds such that RETRAN-3D produces a more limiting response than the FSAR model for the Locked Rotor/Shaft Break event.

The nuclear power response, Figure 4.2-6, predicted by the Dominion model agrees well with the FSAR data, with the Dominion model response slightly over predicting power during rod insertion following the reactor trip on low RCS flow. Similarly, the Dominion model core heat flux response, Figure 4.2-7, also slightly over predicts the FSAR model's response in the same time frame during control rod insertion. Additionally, the Dominion model heat flux response shows a slightly larger decrease at the initiation of the event over the decrease seen in the FSAR data. Both the initial under prediction of the heat flux response, followed by an over prediction during the rod insertion is indicative of the fuel rod heat transfer being modeled differently in the FSAR methods than in the Dominion model. However, the over prediction of both nuclear power and heat flux will lead to conservative results at the limiting point in the transient for both RCS overpressurization and DNB during rod insertion. Overall the nuclear power and heat flux predictions are very similar.

A summary of the LR transient analysis comparison is provided in Table 4.2-2.

Table 4.2-2 LR RCS Overpressure Results

Parameter	Dominion	FSAR
Sequence of Events:		
Low RCS Flow Setpoint Reached (sec)	0.1	0.1
Rods Begin to Drop (sec)	1.1	1.1

Peak RCS Pressure (sec)	3.8	4.1
Peak RCS Pressure (psia)	2680	2617

Summary - LR RCS Overpressure Case

The Dominion Millstone analysis provides responses that are similar to the FSAR analysis for the LR event, with the Dominion model predicting higher peak RCS pressures. Differences are attributed to loop friction losses and fuel rod modeling differences. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

Figure 4.2-1 LR – Reactor Vessel Lower Plenum Pressure

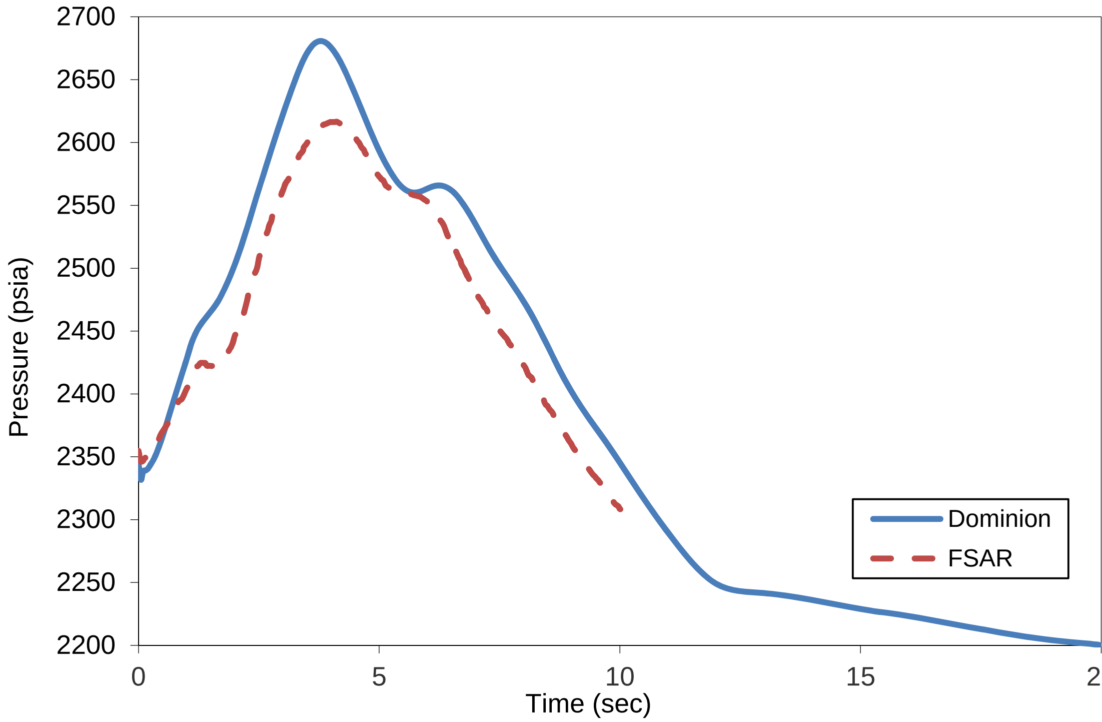


Figure 4.2-2 LR – RCP Outlet Plenum Pressure

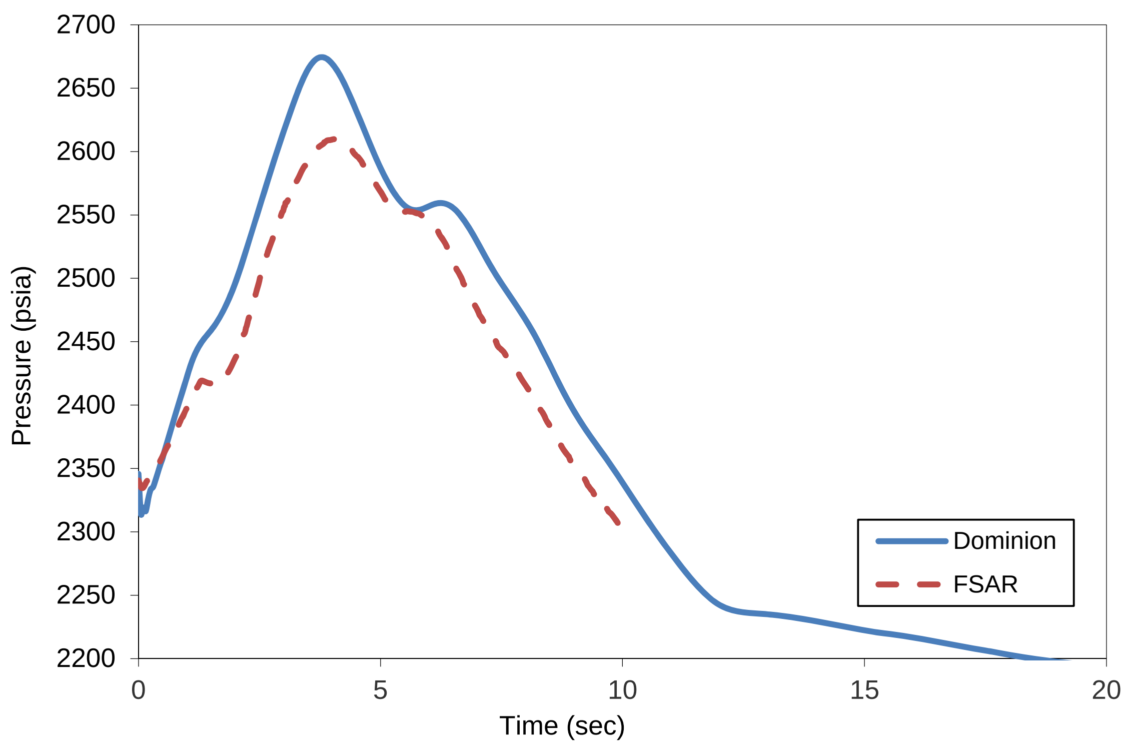


Figure 4.2-3 LR – Faulted Loop Normalized Flow

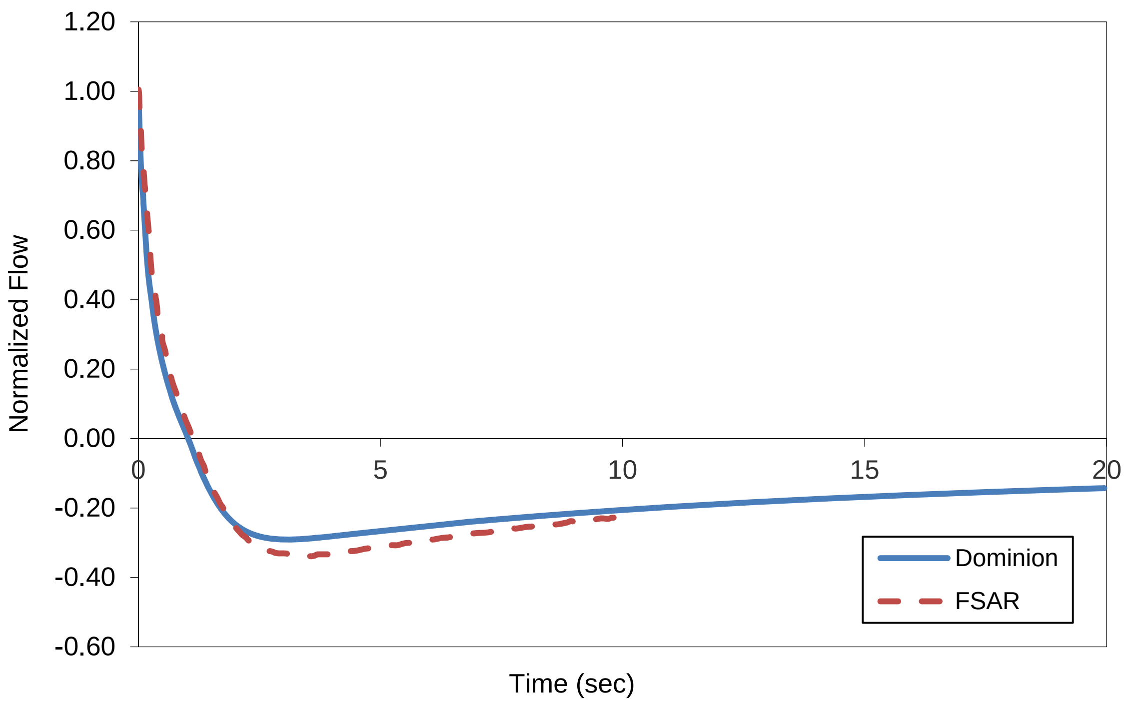


Figure 4.2-4 LR – Unfaulted Loop Normalized Flow

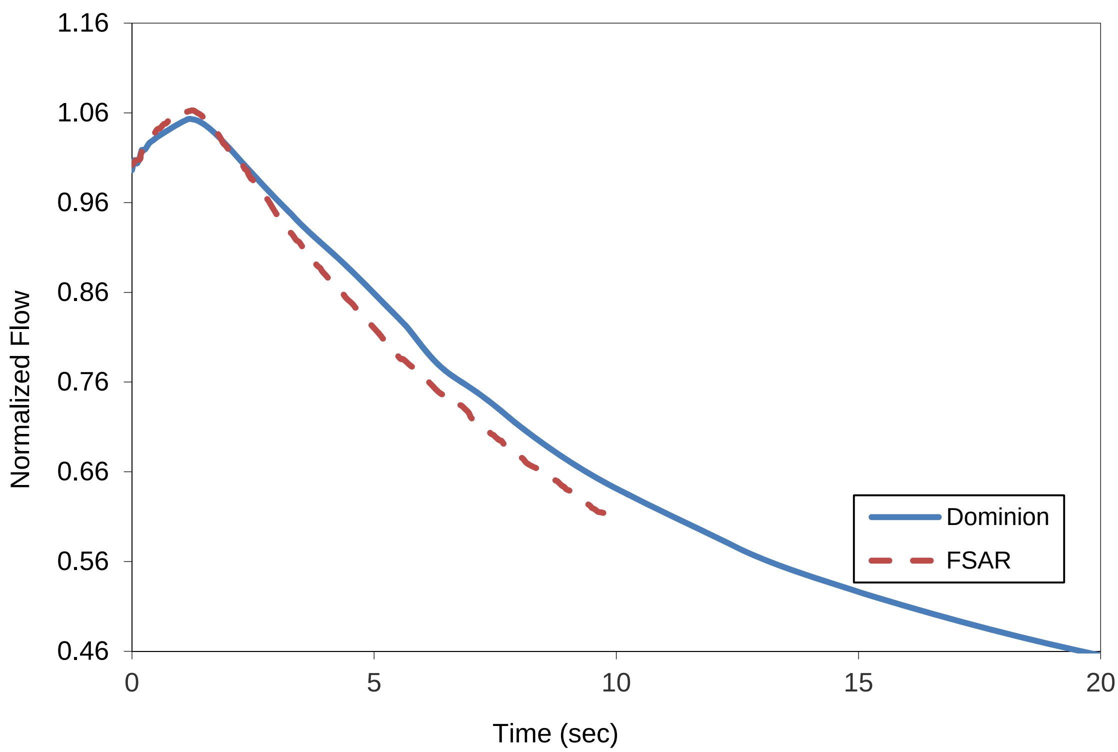


Figure 4.2-5 LR – Core Inlet Normalized Flow

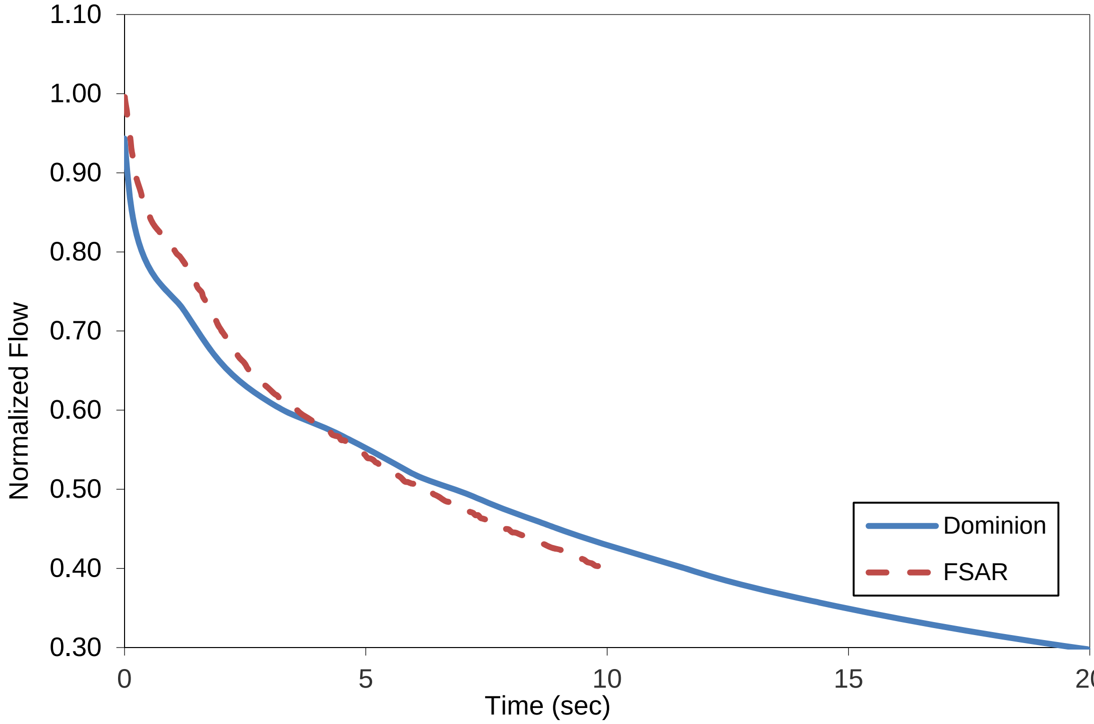


Figure 4.2-6 LR – Nuclear Power

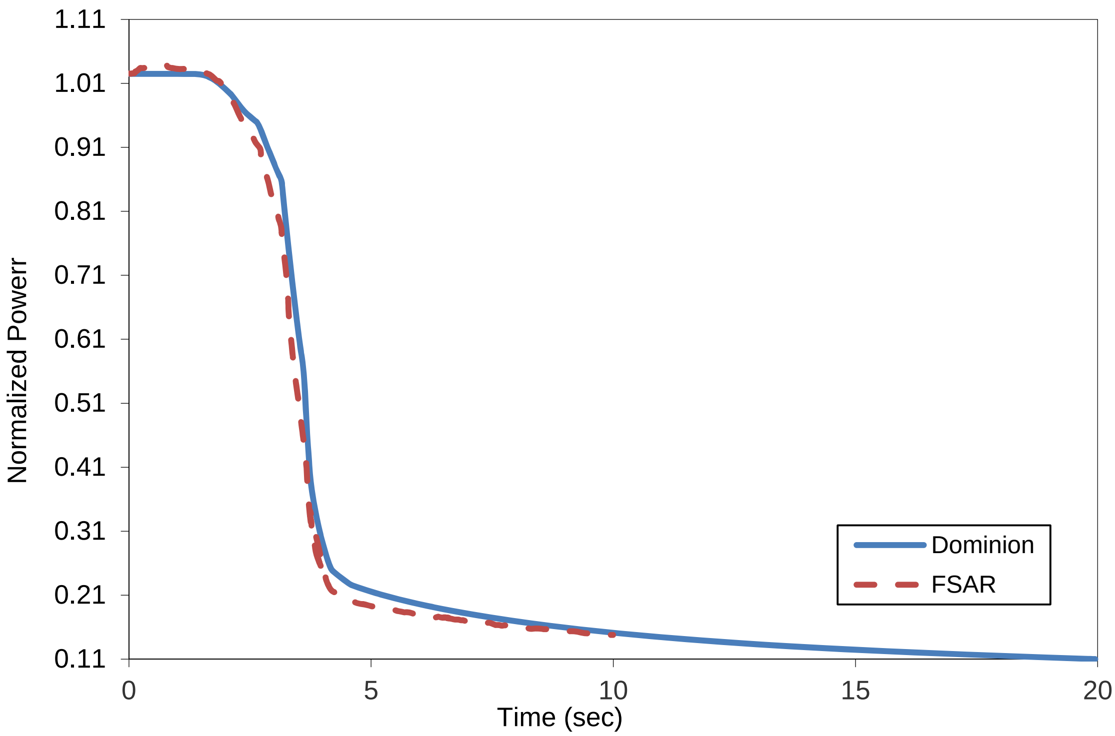
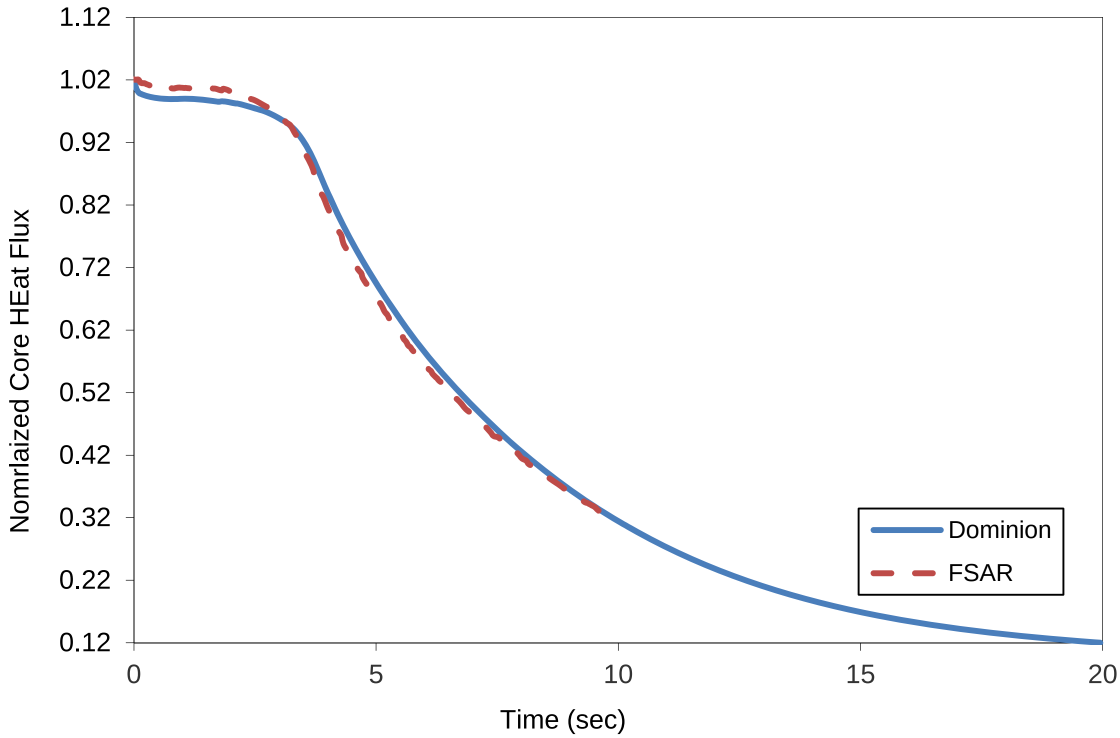


Figure 4.2-7 LR – Core Heat Flux



LR Peak Cladding Temperature

The Locked Rotor event is also analyzed to demonstrate that a coolable core geometry is maintained. A hot spot evaluation is performed to calculate the peak cladding temperature and oxidation level. The Dominion Hot Spot model is described in Topical Report VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient." (Reference 2). The Dominion Hot Spot model was used to evaluate the MPS3 PCT and oxidation level for the LR event.

The Dominion hot spot model is used to predict the thermal-hydraulic response of the fuel for a hypothetical core hot spot during a transient. The hot spot model describes a one-foot segment of a single fuel rod assumed to be at the location of the peak core power location during a transient. The hot spot model uses boundary conditions from the LR system transient analysis to define inlet flow and core average power conditions. The hot spot model uses MPS3-specific values for fuel dimensions, fuel material properties, fluid volume, and junction flow areas.

The hot spot model is run to 0.1 seconds and a restart file is saved. Upon restart, the fuel/cladding gap conductance (thermal conductivity) is modified to simulate gap closure by setting the gap heat transfer coefficient to 10,000 Btu/ft^2-hr-°F for a gap conductance of 2.708 Btu/ft-hr-°F. The hot spot model input summary is provided in Table 4.2-3. Most of the input parameters are the same as those used in the FSAR Chapter 15 analyses. Where differences from the FSAR inputs exist, they are indicated in the Notes column.

Table 4.2-3 Hot Spot Model Input Summary

Parameter	Value	Notes
Computer Code Used	RETRAN-3D	FSAR uses VIPRE
Initial Conditions		
Ratio of Initial to Nominal Power	1.02	
RCS Flow (gpm)	363,200	
Hot Spot Peaking Factor	2.60	
Assumptions/Configuration		
Pre-DNB Film Heat Transfer Coefficient	Thom	
Time of DNB (sec)	0.1	
Post DNB Film Boiling Heat Transfer Coefficient	Bishop-Sandberg-Tong	
Fuel Pin Model		
Post DNB Gap Heat Transfer Coefficient (Btu/hr-ft2-°F)	10,000	
Gap Thermal Expansion Model activated?	Yes	
Zircaloy-Water Reaction activated?	Yes	

LR Peak Cladding Temperature Results

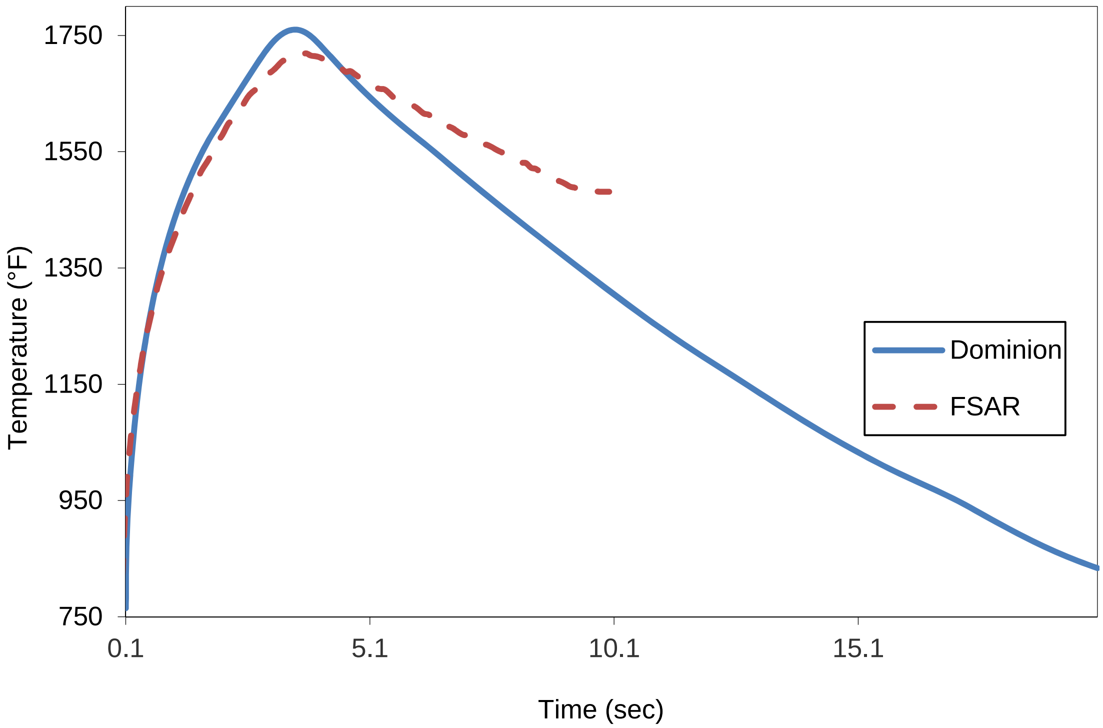
The peak cladding temperature obtained from Dominion's MPS3 hot spot model for the locked rotor event is 1760 °F. The maximum zircaloy-water reaction depth is 3.60875E-06 feet, which corresponds to approximately 0.19% by weight based on the nominal cladding thickness of 1.875E-03 feet. A summary of the LR Peak Cladding Temperature Hot Spot analysis comparison is provided in Table 4.2-4. The cladding inner surface temperature is shown in Figure 4.2-8.

Table 4.2-4 LR Hot Spot Results

Parameter	**Dominion**	**FSAR**
Peak Cladding Temperature	1760 °F	1718 °F
Maximum Zr-water reaction (w/o)	0.19	0.22

The Dominion peak cladding temperature and maximum oxidation values are comparable to the FSAR values. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

Figure 4.2-8 LR Hot Spot – Cladding Inner Surface Temperature



4.3 Loss of Normal Feedwater

The Loss of Normal Feedwater (LONF) event causes a reduction in heat removal from the primary side to the secondary system. Following a reactor trip, heat transfer to the steam generators continues to degrade resulting in an increase in RCS fluid temperature and a corresponding insurge of fluid into the pressurizer. There is the possibility of RCS pressure exceeding allowable values or the pressurizer becoming filled and discharging water through the relief valves. The event is mitigated when Auxiliary Feedwater (AFW) flow is initiated and adequate primary to secondary side heat removal is restored. This analysis shows that the AFW system is able to remove core decay heat, pump heat and stored energy such that there is no loss of water from the RCS and pressure limits are not exceeded. The LONF input summary is provided in Table 4.3-1.

Table 4.3-1 LONF Input Summary

Parameter	Value	Notes
Initial Conditions		
Core Power (MW)	3723	Includes 2% uncertainty
RCS Flow (gpm)	363,200	Thermal Design Flow
Vessel T_{AVG} (F)	583	FSAR value
RCS Pressure (psia)	2300	Nominal + 50 psi
Pressurizer Level (%)	71.6	Nominal + 7.6%
SG Mass	~ 89000	Dominion model adjusted to be consistent with FSAR analysis
Assumptions/Configuration		
Low-Low Level Reactor Trip Setpoint	0%	Percent of narrow range span
Pressurizer: sprays, heaters, PORVs	-	Assumed operable
AFW Temperature (F)	120	Max value
AFW Pump configuration	-	2 motor-driven pumps feed 4 SGs
Auxiliary feedwater flow rate (gpm)	-	Variable as function of SG press.
Local Conditions Heat Transfer model	active	SG secondary side FSAR= multi-node SG
Decay Heat	-	FSAR decay heat constants are applied for this case
Reactivity Parameters		
Doppler Reactivity Feedback	Most negative	Dominion model adjusted to use FSAR Doppler Power Coeffcient
Moderator Feedback	Most Positive	

Results - LONF

The results for the LONF comparison analysis are presented in Table 4.3-2 and Figures 4.3-1 through 4.3-7. The loss of feedwater flow to the steam generators (SG) results in a reduction in SG level until a reactor trip occurs on Low-Low SG level. Normalized power is shown on Figure 4.3-1 and normalized core heat flux in Figure 4.3-2. The nuclear power response and heat flux response predicted by the Dominion model are in excellent agreement with the FSAR data, indicating that the scram on low-low steam generator level occurred at essentially the same time shown for the FSAR data. The results continue to demonstrate good agreement through the end of the event.

Figure 4.3-3 shows the steam generator pressure response. The Dominion steam generator pressure is initialized at a slightly different pressure than the FSAR model because the Dominion model initial condition is adjusted to minimize the steam generator area adjustment. Between 10 and 34 seconds the FSAR pressure increases more rapidly to a pressure ~43 psi greater than the Dominion model prediction when the steam line is isolated. This difference is attributed to differing heat transfer degradation in the MNSG model used in the FSAR analysis versus the SNSG model used in the RETRAN-3D model. Steam line isolation occurs at nearly the same time, causing pressure to increase rapidly. The peak pressure is limited by the main steam safety valves (MSSVs), resulting in an almost identical peak pressure in both the Dominion and FSAR responses. However, the Dominion model pressure decreases following the peak value, where the FSAR model response remains at a constant value near the peak value, due to differences in MSSV modeling. Figure 3.1-4 shows the steam generator liquid mass. The steam generator liquid mass depletes faster in the Dominion cases than in the FSAR cases. This is consistent with the increased relief flow as shown in the steam generator pressure response.

The response in the pressurizer is shown in Figures 4.3-5 and 4.3-6. Between the FSAR and Dominion model, the pressure responses are in good agreement until around 45 – 50 seconds where the Dominion pressure is lower than the FSAR, reflecting less heat transfer degradation during this period. This is followed by a second pressure peak that is higher for Dominion than the FSAR. Based on the sharpness of the Dominion peak compared with the FSAR data, this difference is most likely driven by differences in the pressurizer spray models and primary to secondary heat transfer.

For the pressurizer water volume, shown in Figure 4.3-6, the Dominion model results follow the same trends as the FSAR data, but drops lower in the period from 63 to 900 seconds, then demonstrates a strong insurge during the second heat-up period in the transient while peaking at a somewhat lower value than the FSAR. The difference seen in the pressurizer volume results is primarily due to the previously discussed MSSV modeling differences and the resultant increased steam release from the Dominion model compared to the FSAR model as well as possible differences in the pressurizer spray models.

Table 4.3-2 LONF Results

Parameter	Dominion	FSAR
Peak PZR Liquid Volume (ft3)	1610	1730

Figure 4.3-1 LONF - Nuclear Power

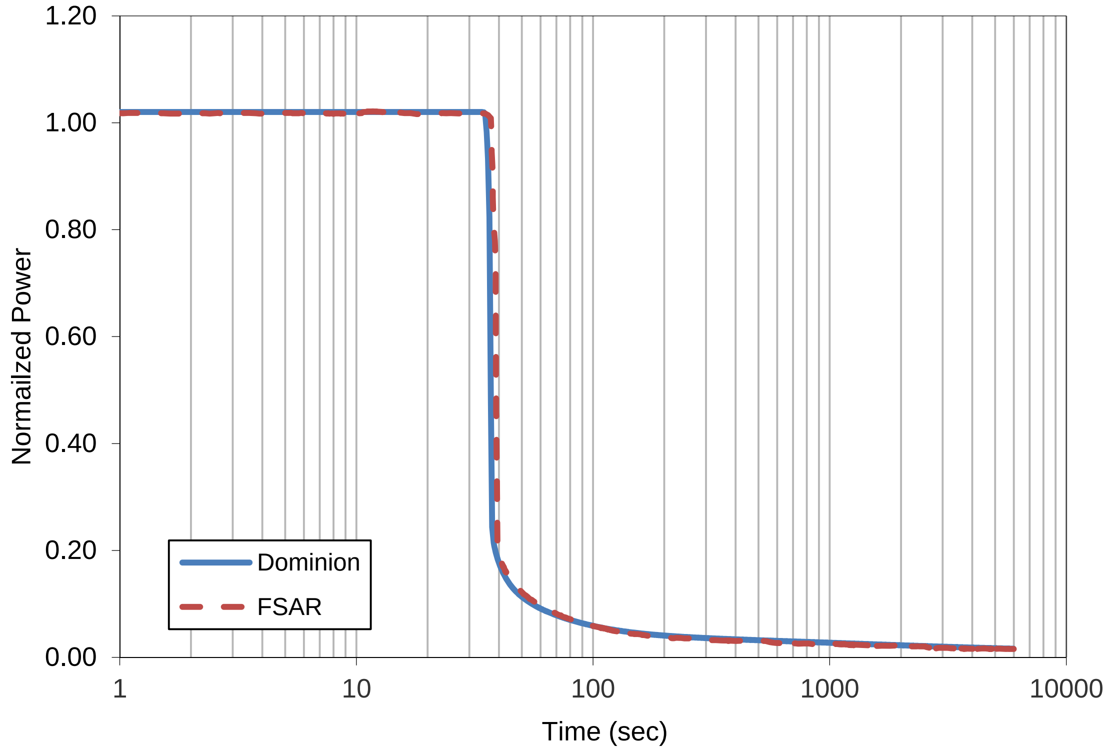


Figure 4.3-2 LONF - Normalized Core Heat Flux

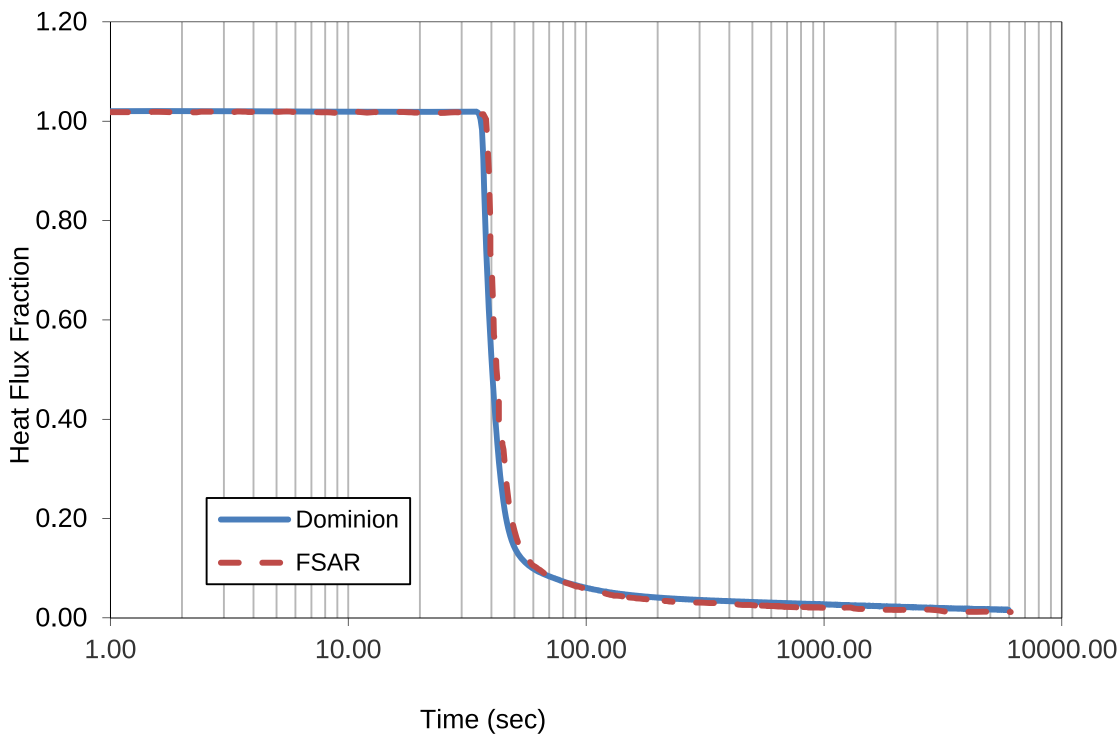


Figure 4.3-3 LONF – Steam Generator Pressure

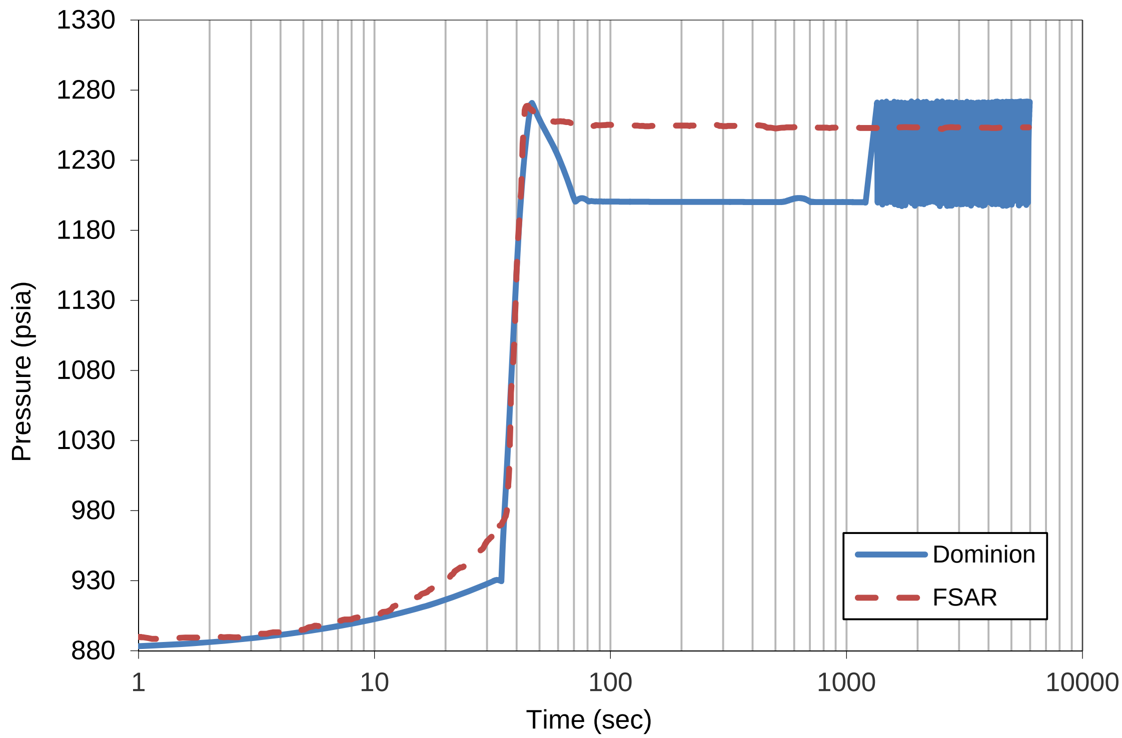


Figure 4.3-4 LONF – Steam Generator Liquid Mass

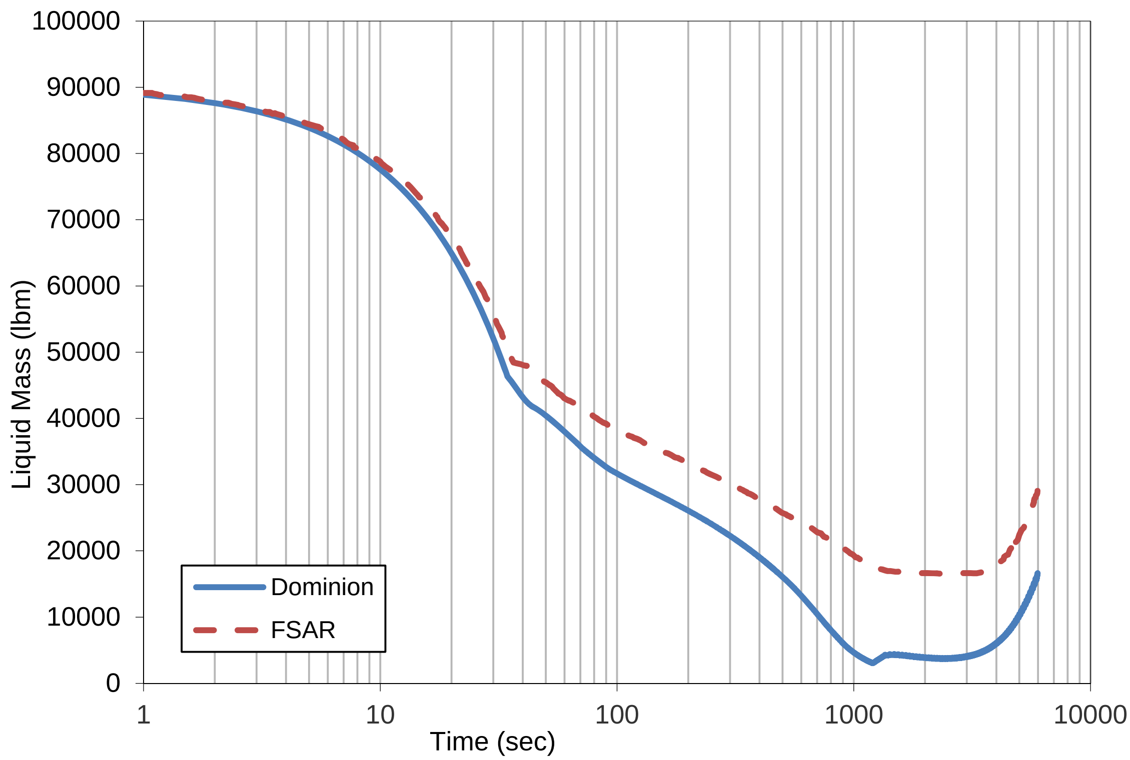


Figure 4.3-5 LONF – Pressurizer Pressure

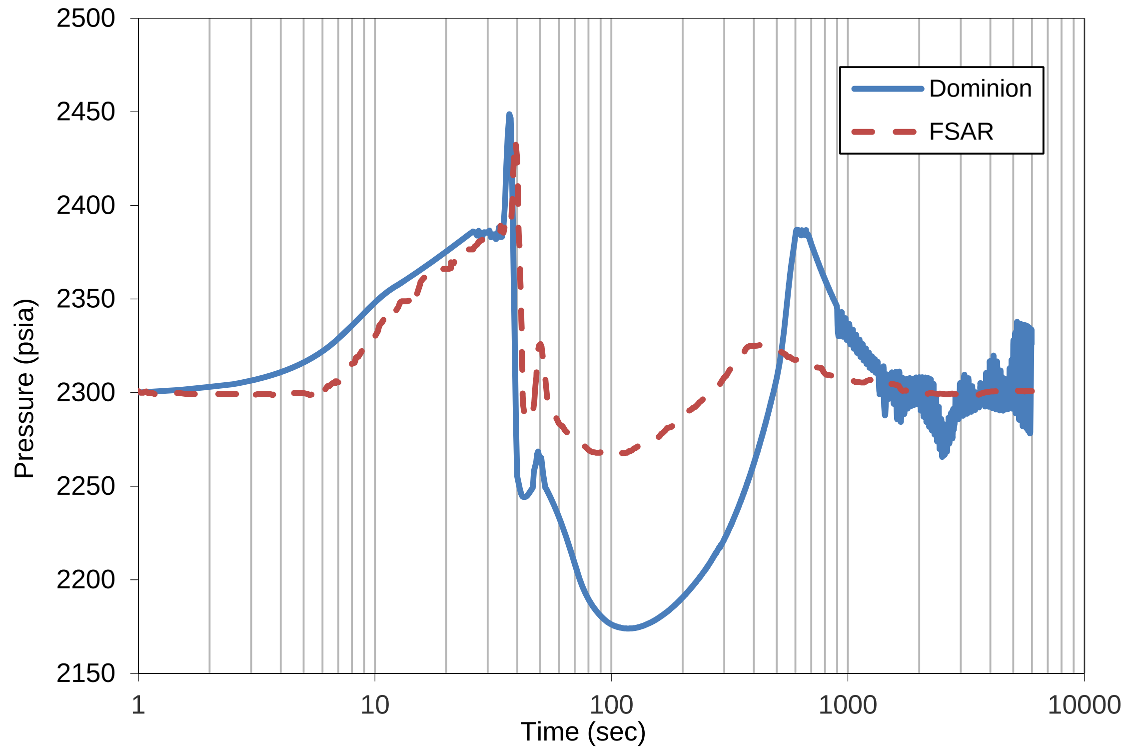


Figure 4.3-6 LONF – Pressurizer Water Volume

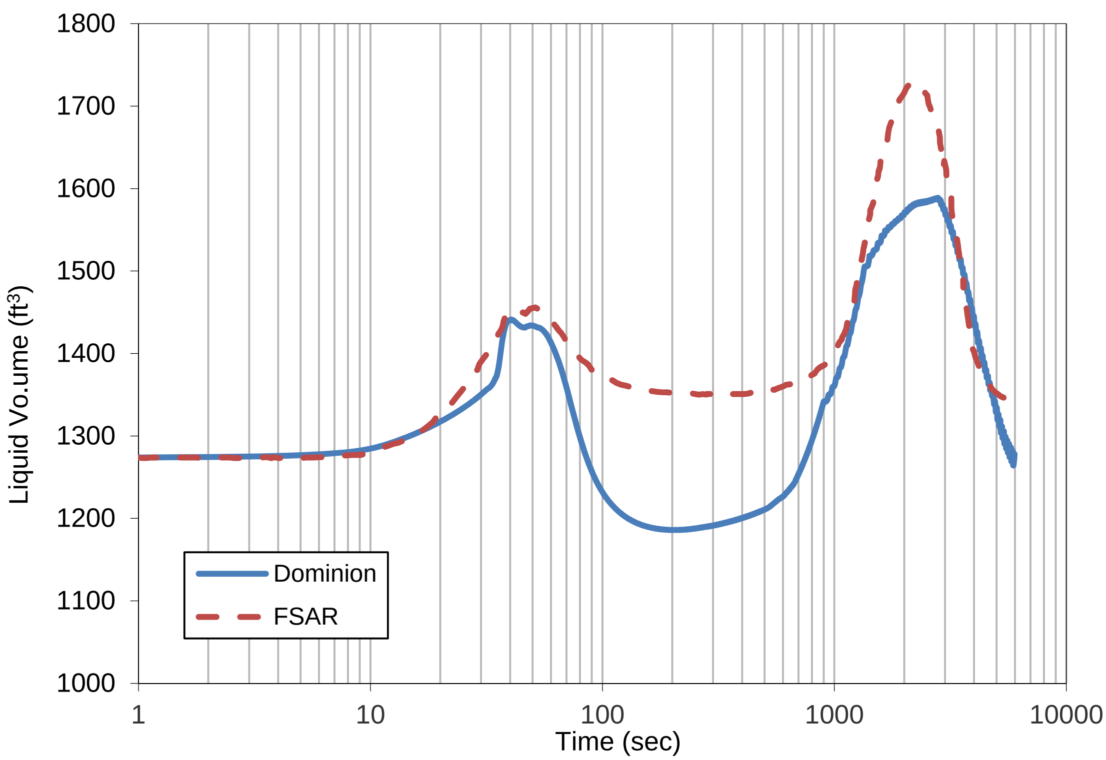
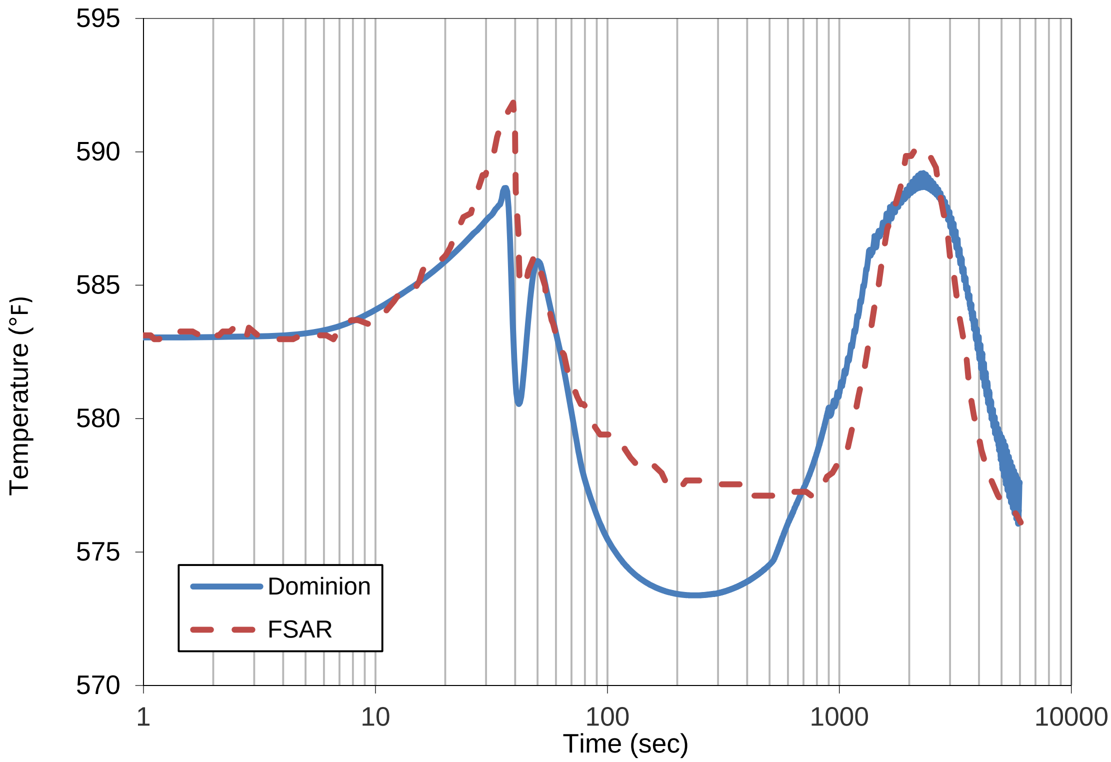


Figure 4.3-7 LONF – Loop Average Temperature



Summary - LONF

The Dominion analysis provides results that are similar to the FSAR analysis for the LONF event. The major differences result from the main steam safety relief valve modeling, which results in higher steam releases and a subsequent increase in heat transfer following the reactor trip. In addition, the steam generator nodalization and related heat transfer along with other modeling differences such as pressurizer spray also affect the transient response. These effects are cumulative resulting in a somewhat smaller long-term pressurizer insurge and higher pressurizer pressure peak compared to the FSAR results. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

4.4 Main Steam Line Break

The Main Steam Line Break (MSLB) event is a rupture in the main steam piping resulting in a rapid depressurization of the SG secondary and corresponding cooldown of the primary. The temperature reduction results in an insertion of positive reactivity with the potential for core power increase and DNBR violation.

The MSLB transient scenario presented here is modeled as an instantaneous, double-ended break at the nozzle of one steam generator from hot shutdown conditions with offsite power available. The input summary is provided in Table 4.4-1.

Table 4.4-1 MSLB Input Summary

Parameter	Value	Notes
Initial Conditions		
Core power (MW)	~1%	HZP
Pump power (MW)	0.0	
RCS Flow (gpm)	363,200	Thermal Design Flow
Vessel T_{AVG} (F)	557	HZP nominal
RCS Pressure (psia)	2250	Nominal
Pressurizer Level (%)	28	HZP nominal
SG Level (%)	50	Nominal
Assumptions/Configuration		
Heat transfer option	Forced HT Map (note 1)	FSAR uses a proprietary heat transfer formulation
Main feedwater flow (% HFP value)	100	initiated at time 0 sec
Auxiliary feedwater flow rate (gpm)	Max	initiated at time 0 sec
SG tube plugging (%)	0	Minimum value
Reactivity Parameters		
RWST Boron	Credited	FSAR does not credit boron from the SI system
Accumulator Boron	Not Credited	
Doppler Reactivity Feedback	Doppler Only Power defect, DTC model disabled	FSAR - Doppler power defect plus DTC included in moderator density feedback
Moderator Feedback	Moderator density feedback	Moderator density feedback

1 - Dominion method maximizes heat transfer coefficients for the faulted SG secondary side.

Results – MSLB with Offsite Power Available

The faulted loop steam flow and steam generator pressure responses shown in Figure 4.4-1 and Figure 4.4-3 match the FSAR data reasonably well with the steam flow and pressure in the Dominion model remaining somewhat higher than the FSAR data. This is partly caused by the slightly larger break junction area and the higher initial steam pressure for the Dominion model. In addition, the Dominion model uses conservatively high heat transfer coefficients in the faulted steam generator, which allow the faulted steam generator to pull heat faster from the primary side.

The Intact loop steam flow (Figure 4.4-2) shows a different response due to differences in the MSIV closure. In the Dominion model, the MSIVs close linearly over 10 seconds, while the FSAR model uses a delay of 10 seconds to conservatively increase RCS overcooling. The initial steam flow is higher for the Dominion case, decreasing below the FSAR value as the MSIVs close. The steam generator mass and pressure responses, shown in Figure 4.4-8 and Figure 4.4-4, reveals the differences in MSIV modeling with the Dominion model releasing somewhat less liquid inventory prior to valve closure.

For both the faulted and intact loops the main feedwater and auxiliary feedwater responses (Figure 4.4-5) give an excellent match to the FSAR data. The steam generator inventory (Figure 4.4-7) for the faulted loop depletes faster in the Dominion model than in the FSAR case due to the higher steaming rate from the faulted steam generator and the quicker and more conservative return to power.

The nuclear power and core heat flux responses (Figure 4.4-9 and Figure 4.4-10) calculated by the Dominion model peak higher and more quickly than the FSAR data. This response is contributed to by the greater cooling effects of the faulted steam generator on the RCS due to its higher steam production. The quicker return to power is also a result of differences in the nodalization and mixing at the core inlet and outlet between the Dominion model and the FSAR model. The return to power also drops off approximately 50 seconds sooner in the Dominion model. This is also caused by the higher steam rate in the Dominion model which causes the faulted steam generator to dry out sooner. The power response for both models is not affected by the delivery of boron to the RCS. This is because the FSAR model does not credit boron and in the Dominion model boron does not reach the RCS from the SI system until after the termination of the transient. Overall, the Dominion model results in a more conservative response for core heat flux and power.

The pressurizer pressure response (Figure 4.4-12) agrees very well with the pressure predicted by the FSAR model for the first 50 seconds of the transient, after which the FSAR data falls approximately 100 psi lower than the pressure calculated by the Dominion model. This difference is a result of using only a single upper head leakage path in the Dominion model. The upper head leakage is taken from the three intact loops and does not credit any flow from the lower temperature, faulted loop. This causes the upper head temperature to remain slightly higher than would actually be the case, which allows a vapor bubble in the upper head to form sooner and become larger. This in turn prevents the RCS pressure from falling lower.

The pressurizer drains at approximately the same rate for the Dominion model and FSAR models (Figure 4.4-13). However, for the Dominion model the pressurizer begins to refill approximately 100 seconds sooner. The quicker refilling is a result of the higher and quicker return to power which causes the RCS temperature to rise sooner in the Dominion model. This causes the RCS fluid inventory to expand which results in the pressurizer refilling sooner in the Dominion model than is seen from the FSAR model.

Table 4.4-2 MSLB with Offsite Power Results

Event	Time (sec) From Start of Transient	
	Dominion	**FSAR**
Steam Line Ruptures	0	0
Manual Reactor Trip	0	0
Increase MFW to 100% of Nominal HFP Value	0	0
Initiate Maximum AFW to Faulted Steam Generator	0	0
Main Feedwater Isolation	7.5	8.2
MSIVs Closed	12.5	13.5
Pressurizer Empty	15.5	20.5
Criticality Attained	33.5	28.0
Safety Injection Flow Initiation	47.9	72.8
Faulted Steam Generator Dries Out	298	~350

Figure 4.4-1 MSLB – Faulted Loop Steam Flow

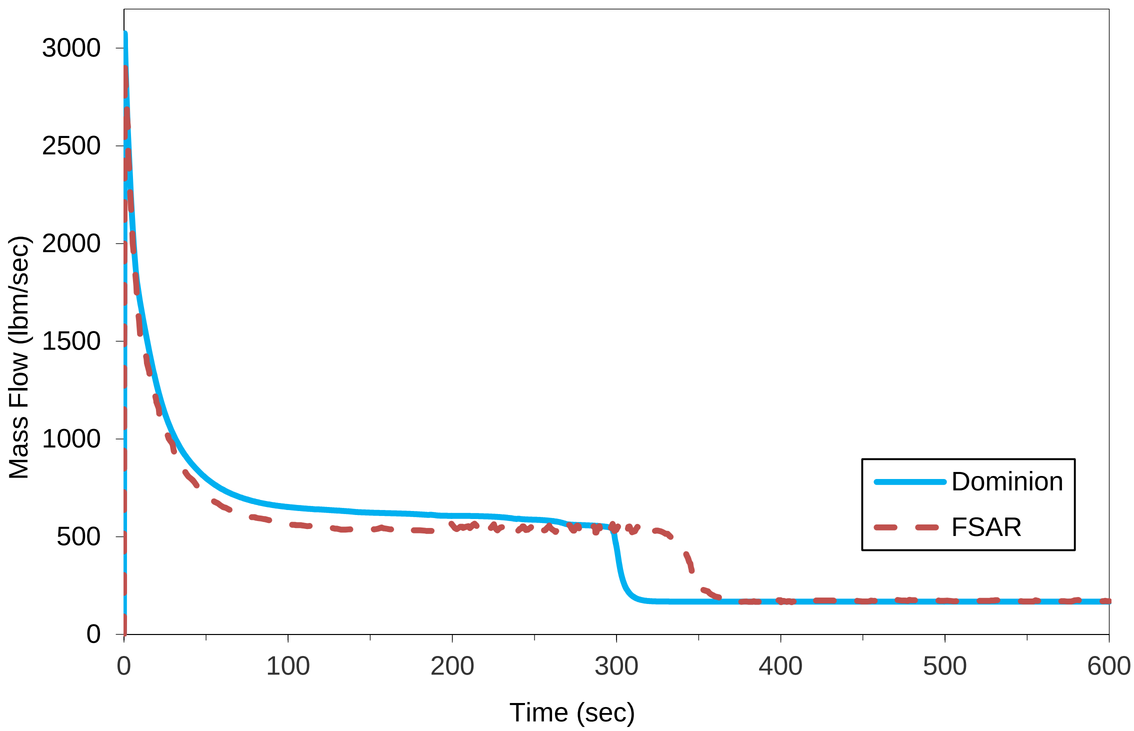


Figure 4.4-2 MSLB – Intact Loop Steam Flow

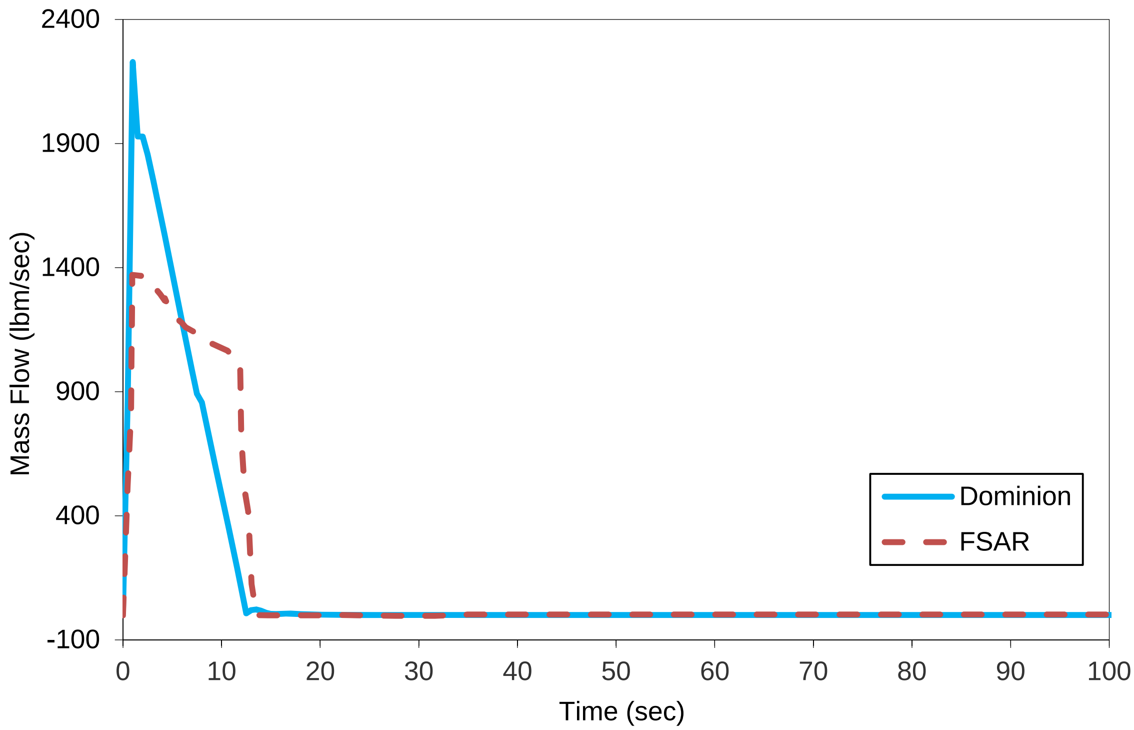


Figure 4.4-3 MSLB – Faulted Loop Steam Generator Pressure

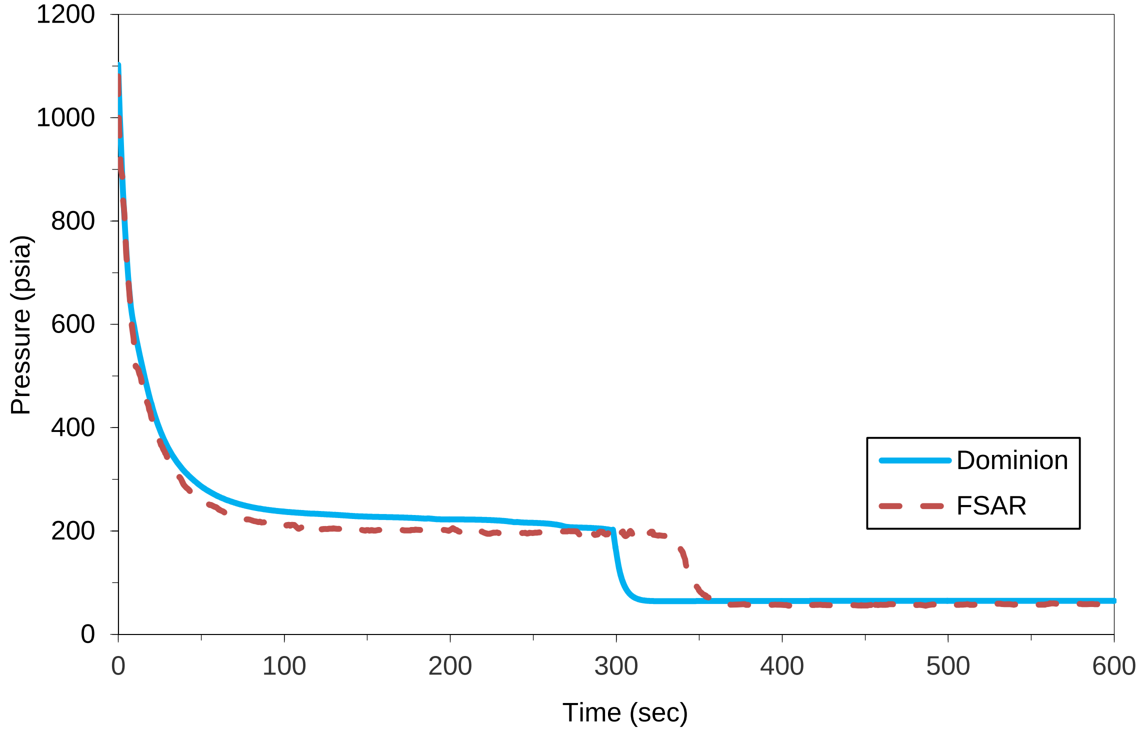


Figure 4.4-4 MSLB –Intact Loop Steam Generator Pressure

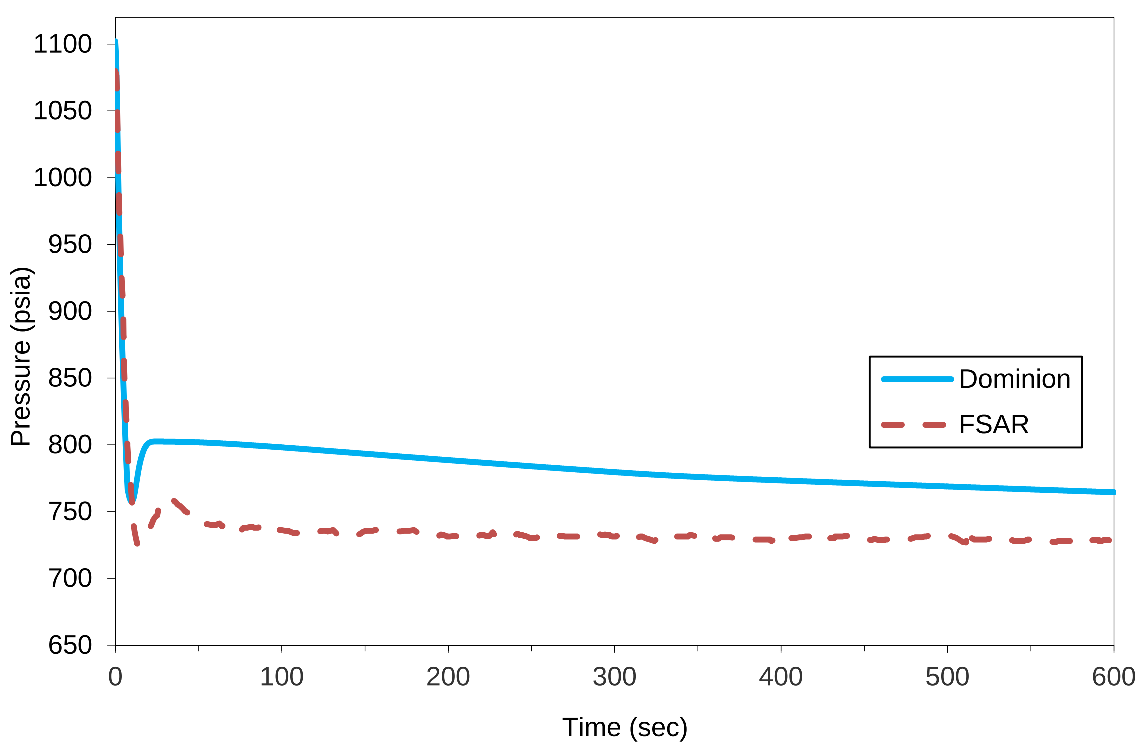


Figure 4.4-5 MSLB – Faulted Loop Total Feedwater Flow

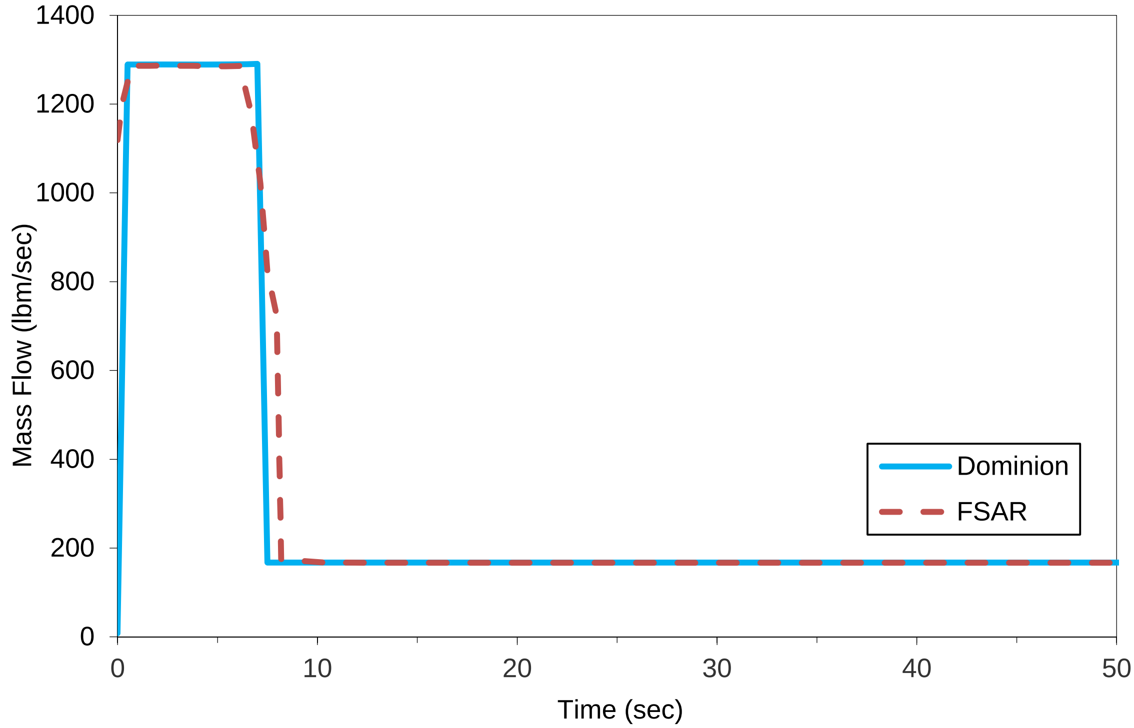


Figure 4.4-6 MSLB – Intact Loop Total Feedwater Flow

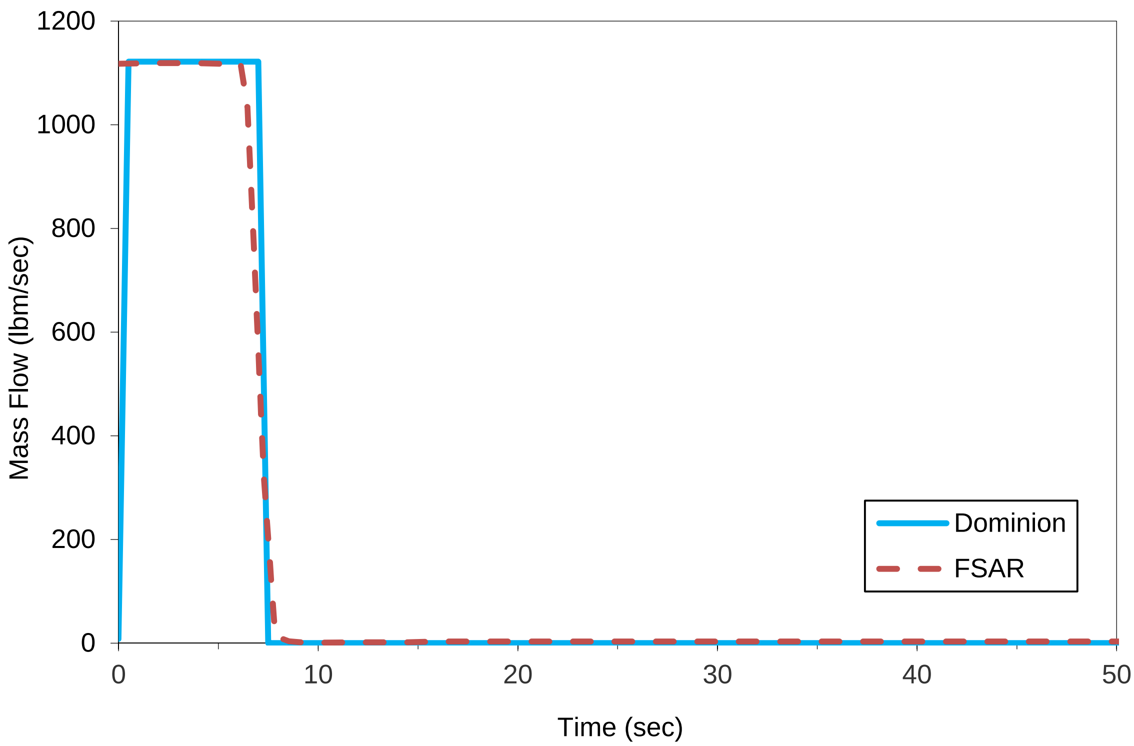


Figure 4.4-7 MSLB – Faulted Loop SG Liquid Mass

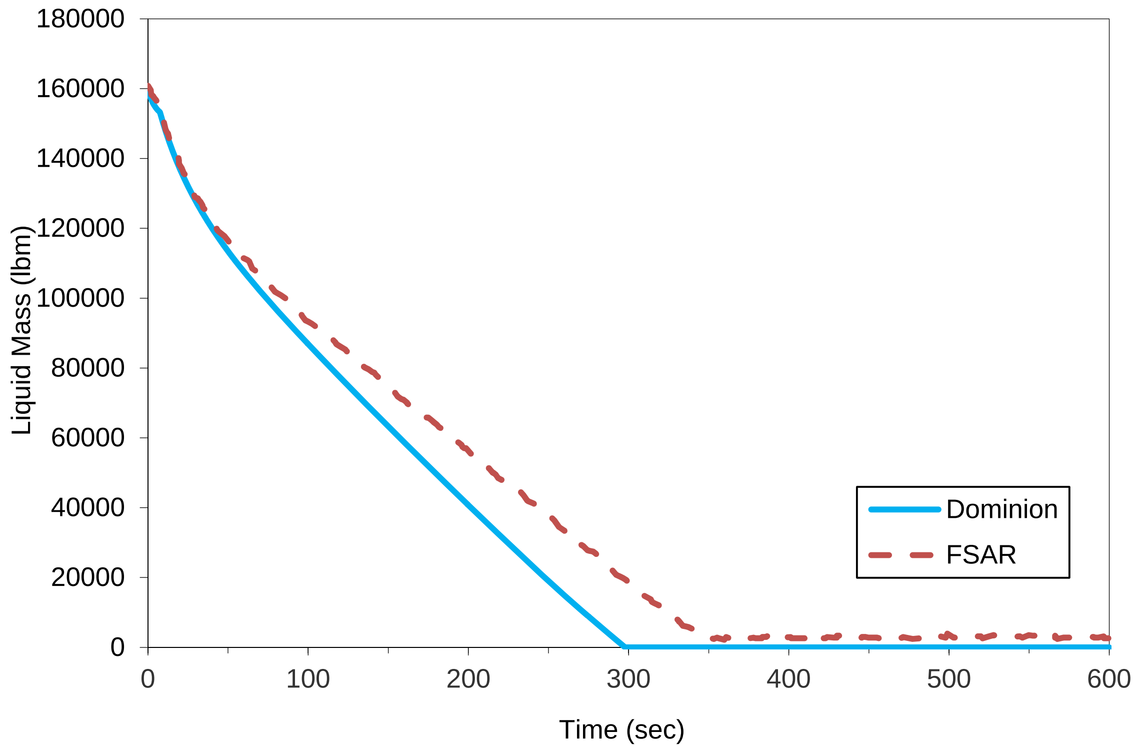


Figure 4.4-8 MSLB – Intact Loop SG Liquid Mass

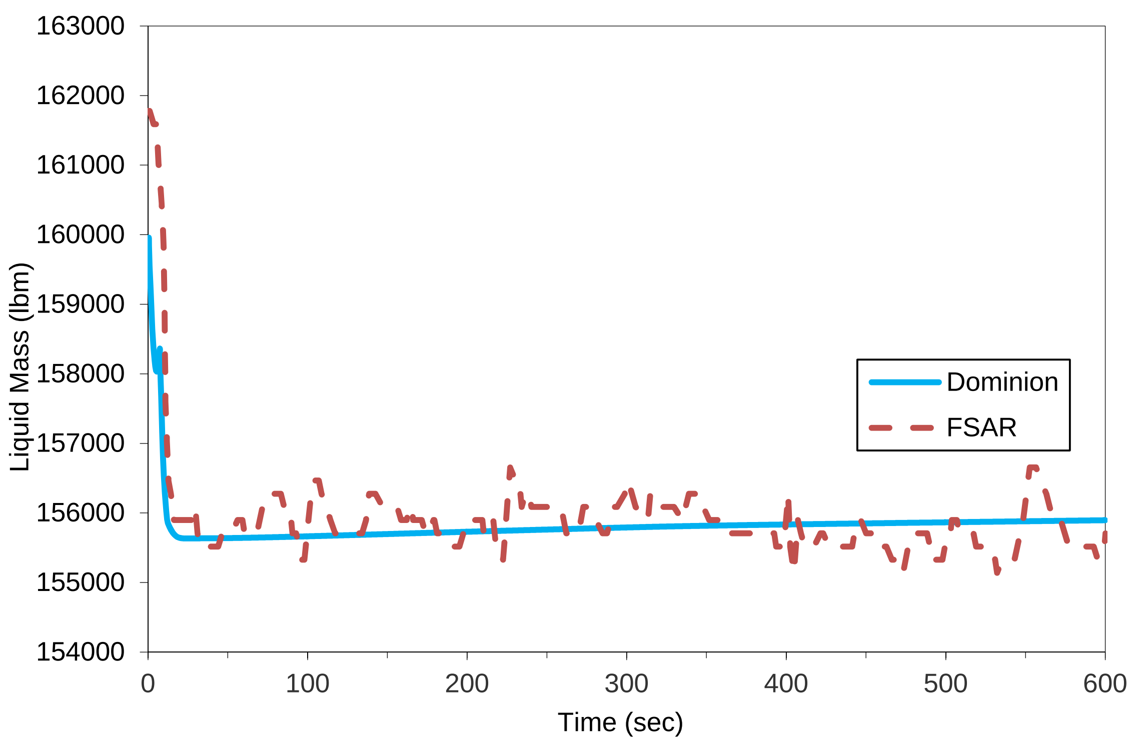


Figure 4.4-9 MSLB – Normalized Core Power

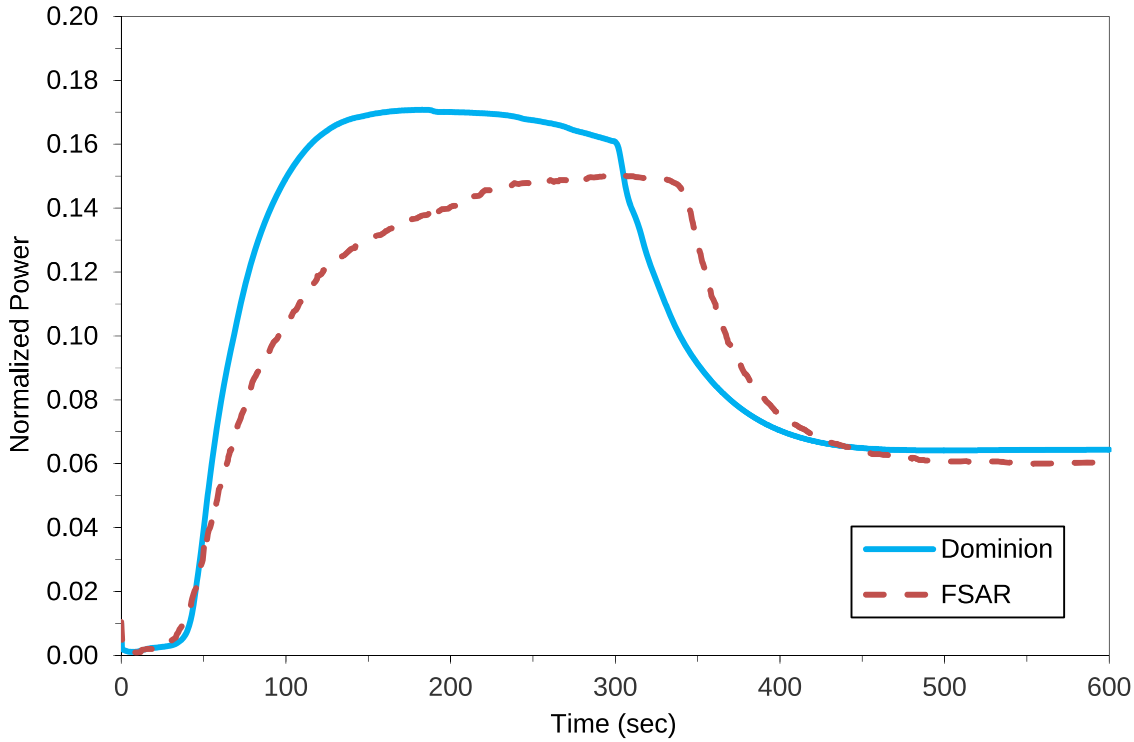


Figure 4.4-10 MSLB – Normalized Core Heat Flux

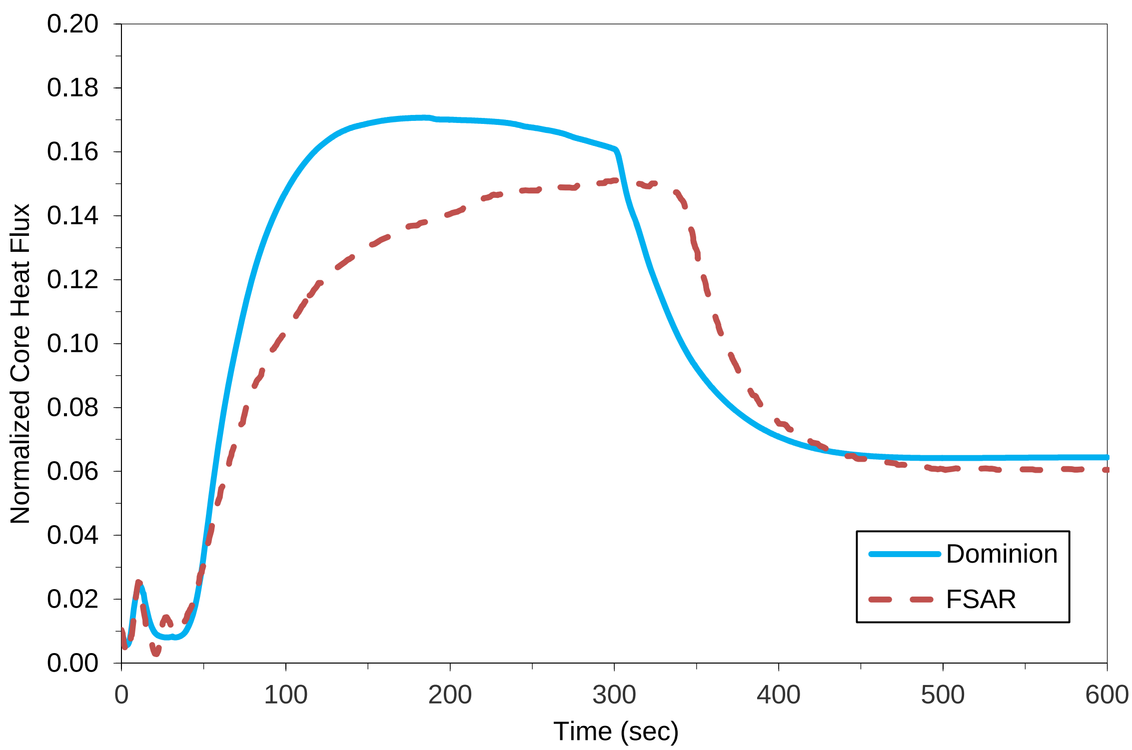


Figure 4.4-11 MSLB – Reactivity Feedback

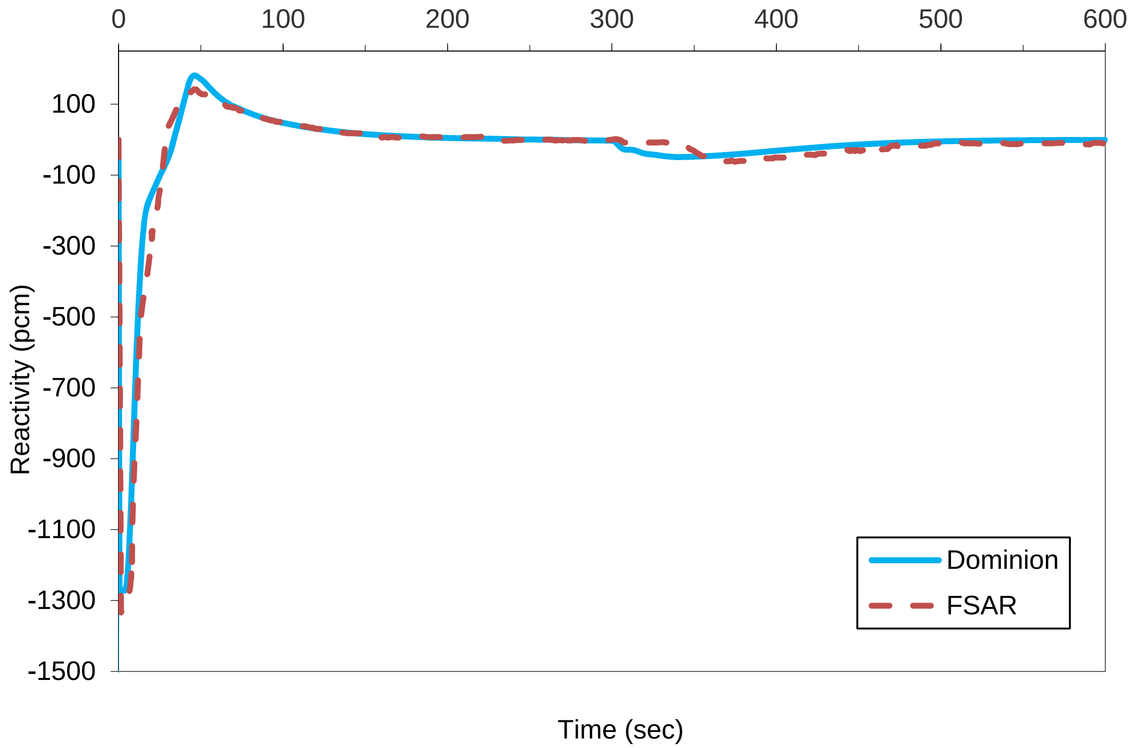


Figure 4.4-12 MSLB – Pressurizer Pressure

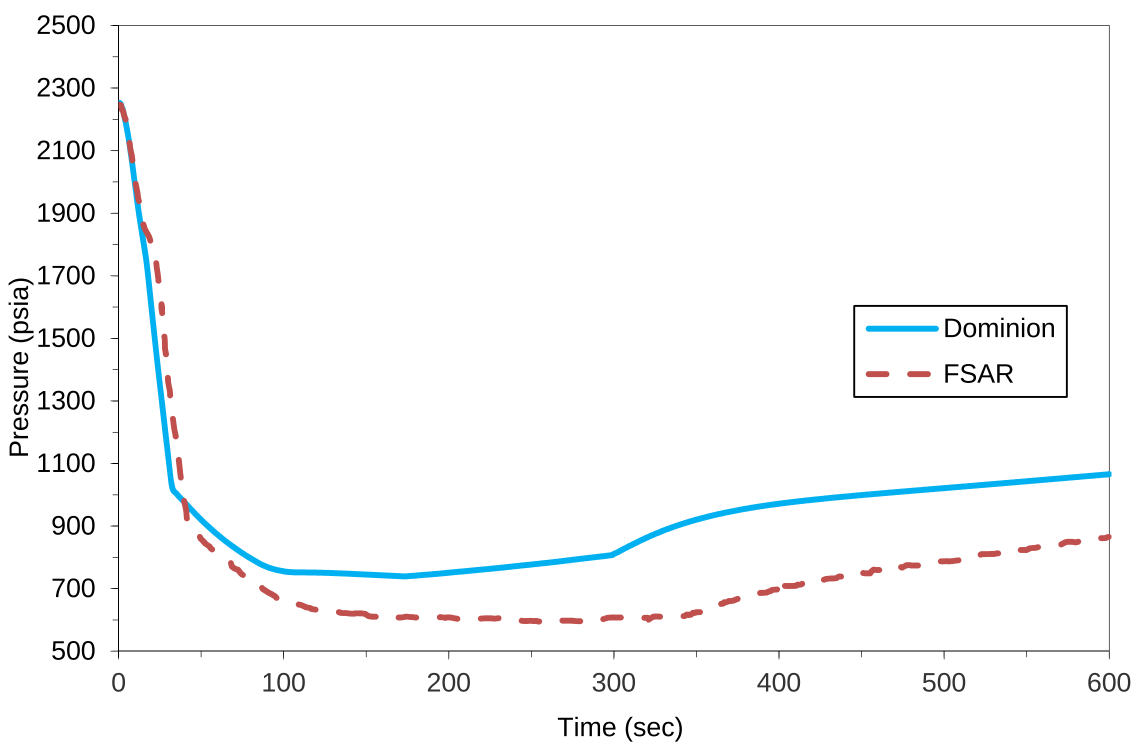


Figure 4.4-13 MSLB – Pressurizer Liquid Volume

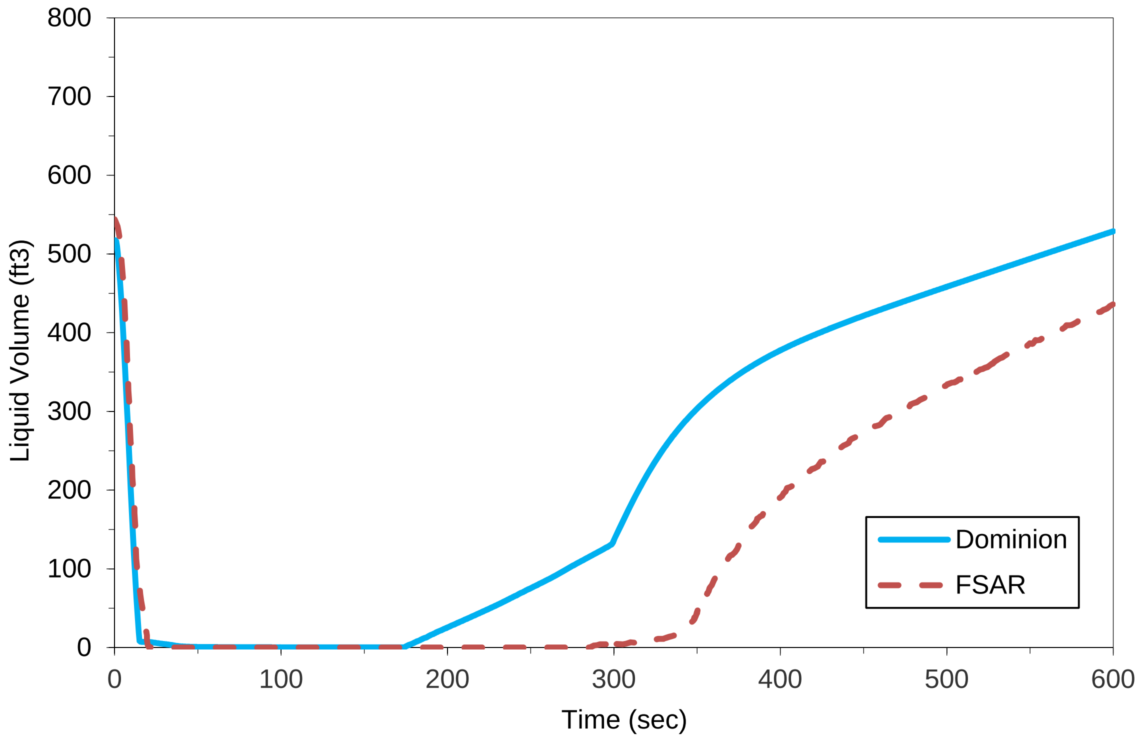


Figure 4.4-14 MSLB – Faulted Loop Vessel Inlet Temperature

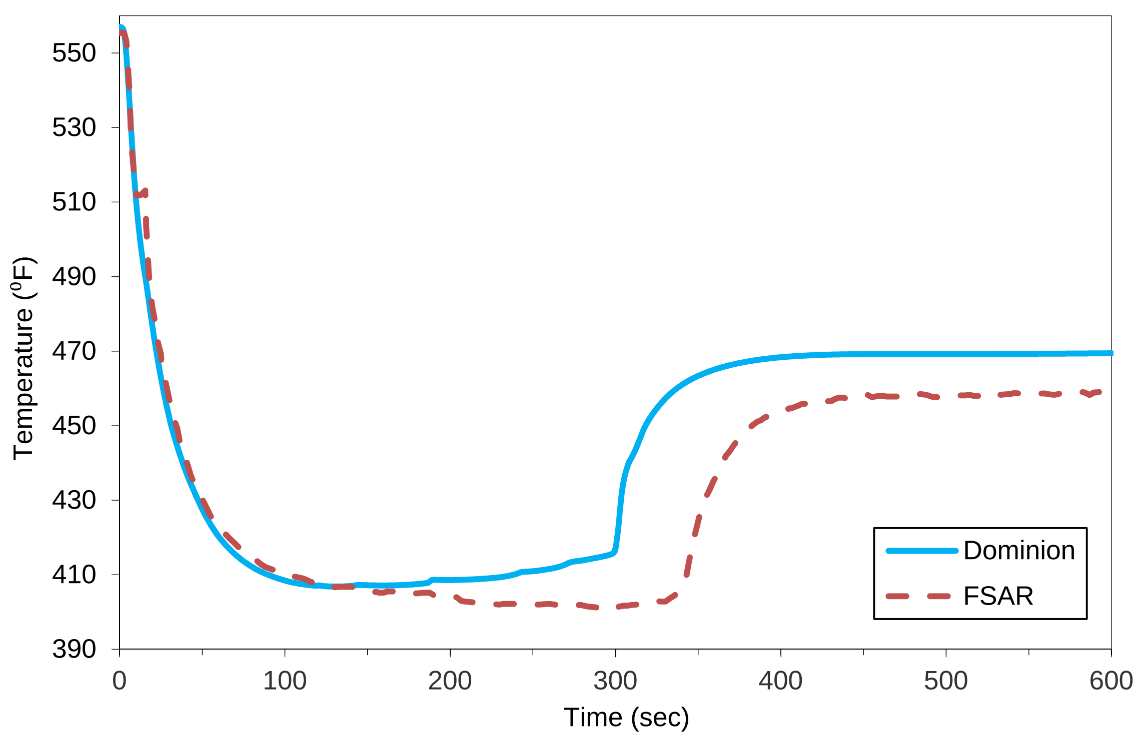


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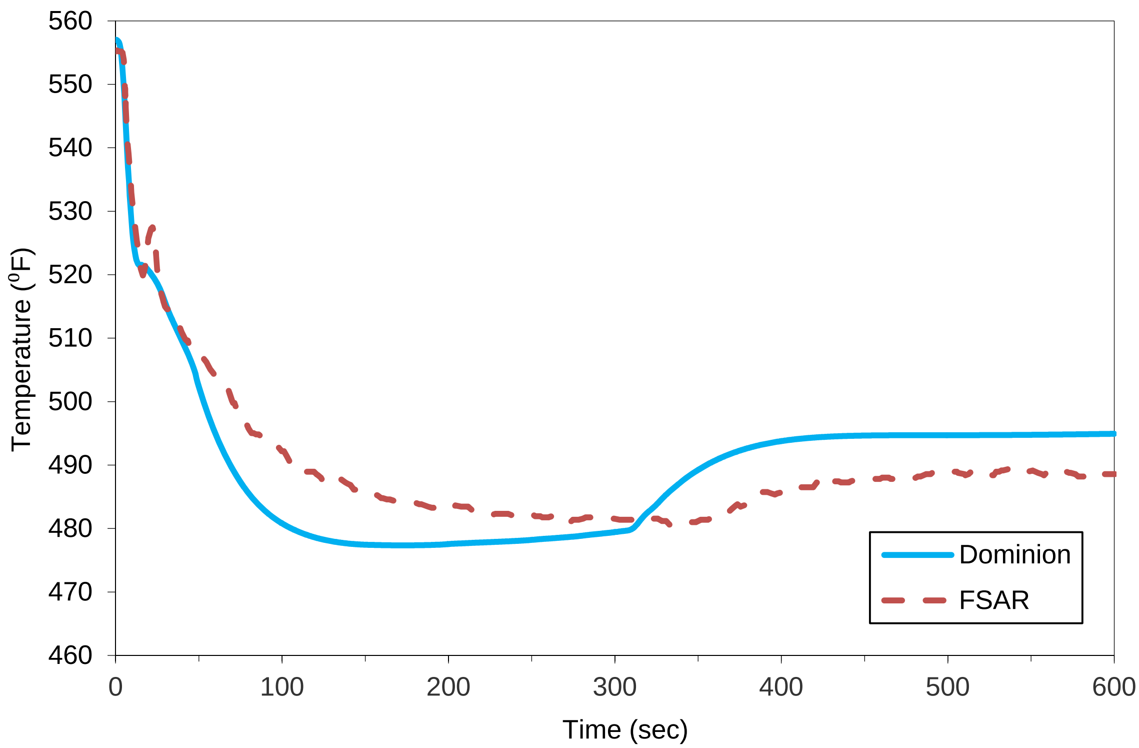
Figure 4.4-15 MSLB – Intact Loop Vessel Inlet Temperature



Summary - MSLB

This section presents a comparison of a RETRAN-3D Main Steam Line Break transient calculation with the Millstone model using the Dominion RETRAN transient analysis methods (Reference 1) compared to the FSAR results. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR. The key observations from these comparisons are that:

1) The peak power and heat flux reached with the Dominion methods is higher than the FSAR result.

2) Core and steam generator nodalization effects asymmetric transients such as a MSLB.

4.5 Control Rod Bank Withdrawal at Power

The Control Rod Bank Withdrawal at Power (RWAP) event is defined as the inadvertent addition of core reactivity caused by the withdrawal of rod control cluster assembly (RCCA) banks when the core is above no load conditions. The RCCA bank withdrawal results in positive reactivity insertion, a subsequent increase in core nuclear power, and a corresponding rise in the core heat flux. The RWAP event described here is terminated by the Reactor Protection System on a high neutron flux trip or the overtemperature ΔT trip (OTΔT), consistent with the FSAR analyses.

The RWAP event is simulated by modeling a constant rate of reactivity insertion starting at time zero and continuing until a reactor trip occurs. The Dominion analysis involves two different reactivity insertion rates, 1 pcm/sec and 100 pcm/sec that match the reactivity insertion rates presented plots in the FSAR. Most of the input parameters are the same as those used in the FSAR Chapter 15 analyses. Where differences from the FSAR inputs exist, they are indicated in the Notes column.

Table 4.5-1 RWAP Input Summary

Parameter	Value	Notes
Initial Conditions		
Core Power (MW)	3650	Nominal
RCS Flow (gpm)	379,200	Minimum Measured Flow
Vessel T_{AVG} (F)	589.5	Nominal
RCS Pressure (psia)	2250	Nominal
Pressurizer Level (%)	64	Nominal
SG Level (%)	50	Nominal
Initial Fuel Temperature	Minimum	Uses current FSAR analysis conductivity adjustments
Assumptions/Configuration		
Reactor trip	-	High neutron flux or OTΔT
Automatic rod control	-	Not credited
Pressurizer level control	-	Not credited
Pressurizer heaters	-	Not credited
Pressurizer sprays, PORVs	-	Active
SG tube plugging (%)	10	Max value
Reactivity Parameters		
Doppler Reactivity Feedback	Least Negative	
Moderator Feedback	Most Positive	Zero MTC for cases from full power

Results – RWAP 1 pcm/sec Case

Figure 4.5-1 shows the core power response. The core power rate of increase for the Dominion model is greater than the FSAR data. This leads to the Dominion modeling tripping on high neutron flux at about 74 seconds. The FSAR case rises in power at a slower rate, which trips on an OTΔT signal at about 93 seconds. The difference in reactor trip mechanisms between the Dominion and FSAR cases is reasonable considering the breakpoint for switching between OTΔT and high flux as shown in FSAR Figure 15.4-10. The pressure response also affects the OTΔT setpoint such that the lower FSAR pressure (see below) will act to reduce the setpoint.

The pressurizer pressure response is shown in Figure 4.5-2. For the Dominion model, the pressure rises faster than the FSAR result. At about 42 seconds, the Dominion model reaches the pressurizer relief valve setpoint and begins to cycle. The FSAR more slowly increases in pressure and reaches the relief valve set point around 10 seconds prior to the reactor trip. The difference in pressure response can be attributed to the difference in core power response as each cases pressure response initially mimics the energy generated by the core as seen in Figure 4.5-1 and the higher spray flow assumed in the FSAR analysis, which acts to suppress pressure. The same can be seen in the vessel average temperature response where the FSAR case lags the Dominion response, yet reaches a temperature approximately 5 degrees higher than the Dominion case due to the FSAR case tripping later in the transient.

Table 4.5-2 RWAP 1 pcm/sec Time Sequence of Events

Event	Time (seconds)	
	Dominion	**FSAR**
Reactivity Insertion at 1 pcm/sec	0.0	0.0
Reactor Trip Signal Initiated	73.7*	93.6**

* *Trip on high neutron flux*

** *Trip on OTΔT*

Results – RWAP 100 pcm/sec Case

Figure 4.5-4 shows the core power response for the current FSAR analysis and the Dominion model. The Dominion model trips on a high neutron flux at about 1.17 seconds, compared to about 1.29 seconds for the current FSAR analysis. The 100 pcm/sec transient is a fast transient and the time period before the reactor trip is so brief that any differences in fuel pin heat transfer modeling assumptions have little impact on Doppler reactivity feedback. Overall, the Dominion model peaks at a higher, thus more conservative power level.

The pressurizer pressure response is shown in Figure 4.2-5. The Dominion model matches very well with the FSAR analysis. The main difference being that the Dominion model peaks at a higher pressure than the FSAR analysis. This correlates with the power response shown in Figure 4.2-4 where the Dominion model peaks at a higher overall nuclear power. Figure 4.2-6 shows the vessel average temperature. For the 100 pcm/sec case the Dominion model matchs very closely with the FSAR analysis

Table 4.5-3 RWAP 100 pcm/sec Time Sequence of Events

Event	Time (seconds)	
	Dominion	**FSAR**
Reactivity Insertion at 100 pcm/sec	0.0	0.0
Reactor Trip Signal Initiated	1.17*	1.29*

* *Trip on high neutron flux*

Figure 4.5-1 RWAP – 1 pcm/sec Nuclear Power

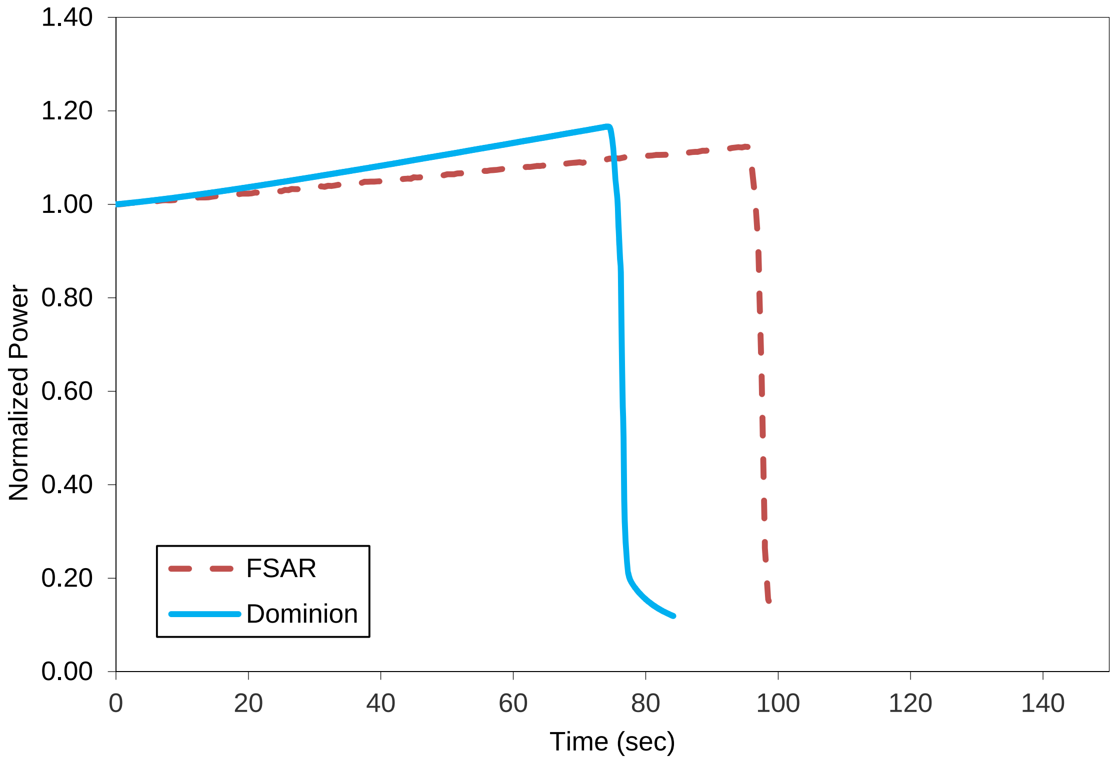


Figure 4.5-2 RWAP – 1 pcm/sec Pressurizer Pressure

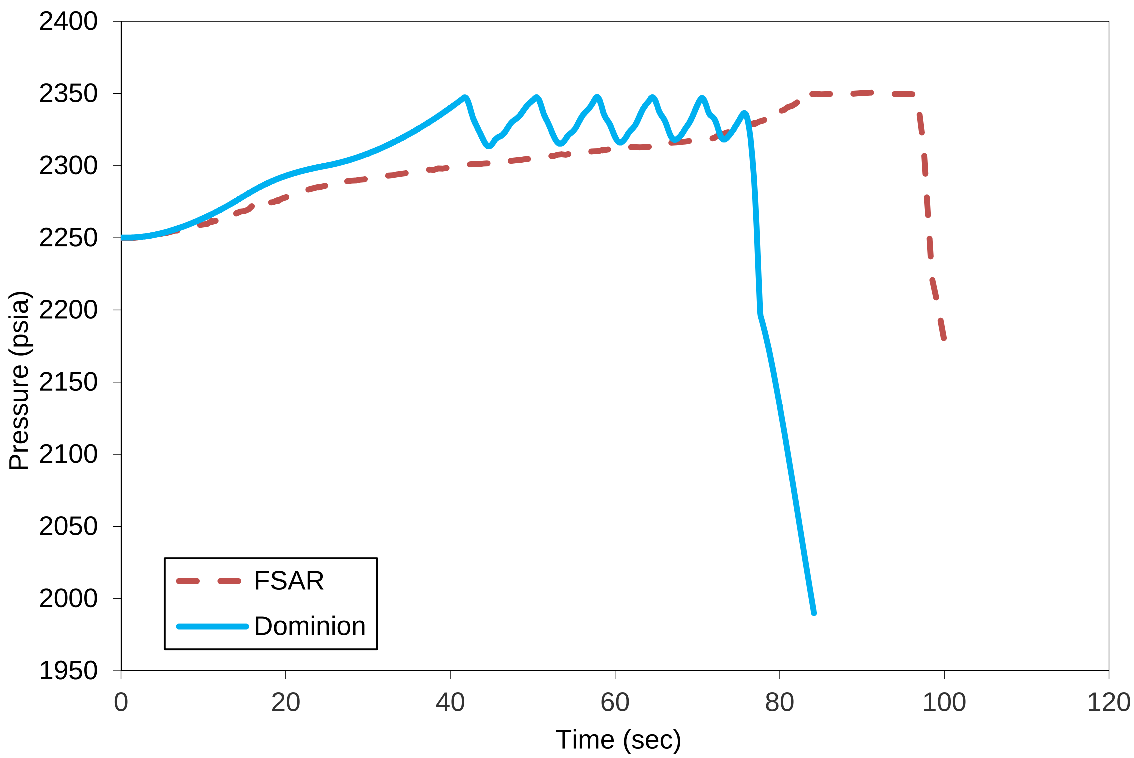


Figure 4.5-3 RWAP – 1 pcm/sec Vessel Average Temperature

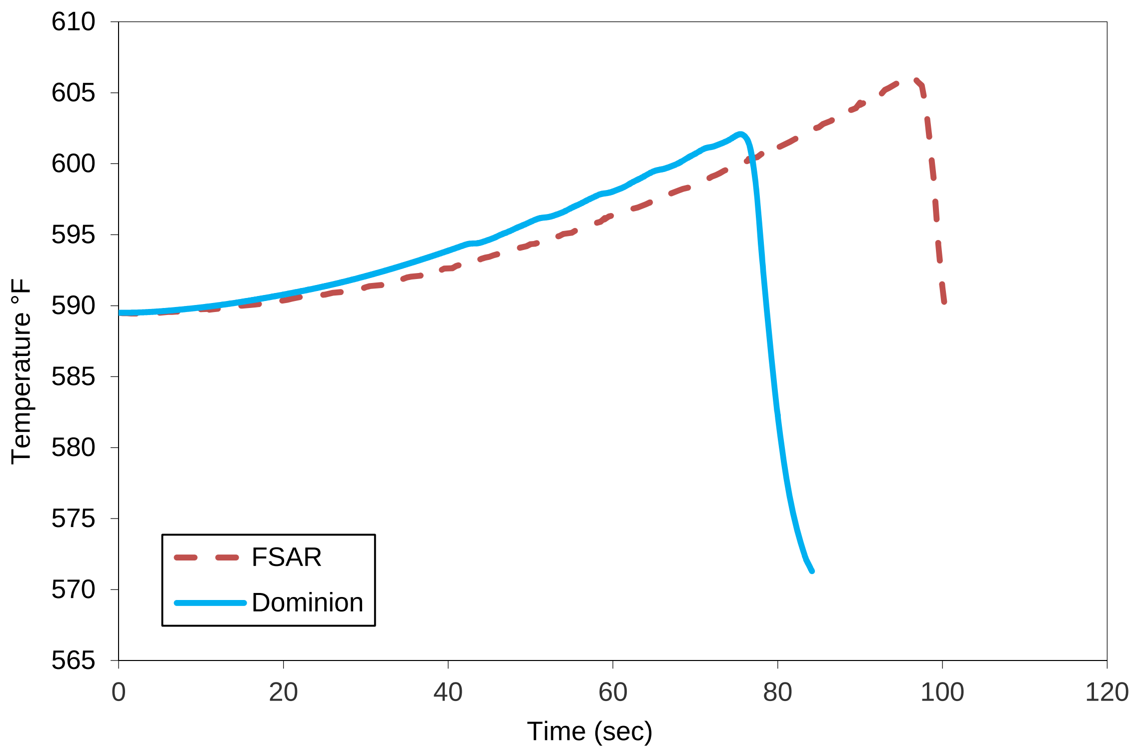


Figure 4.5-4 RWAP – 100 pcm/sec Nuclear Power

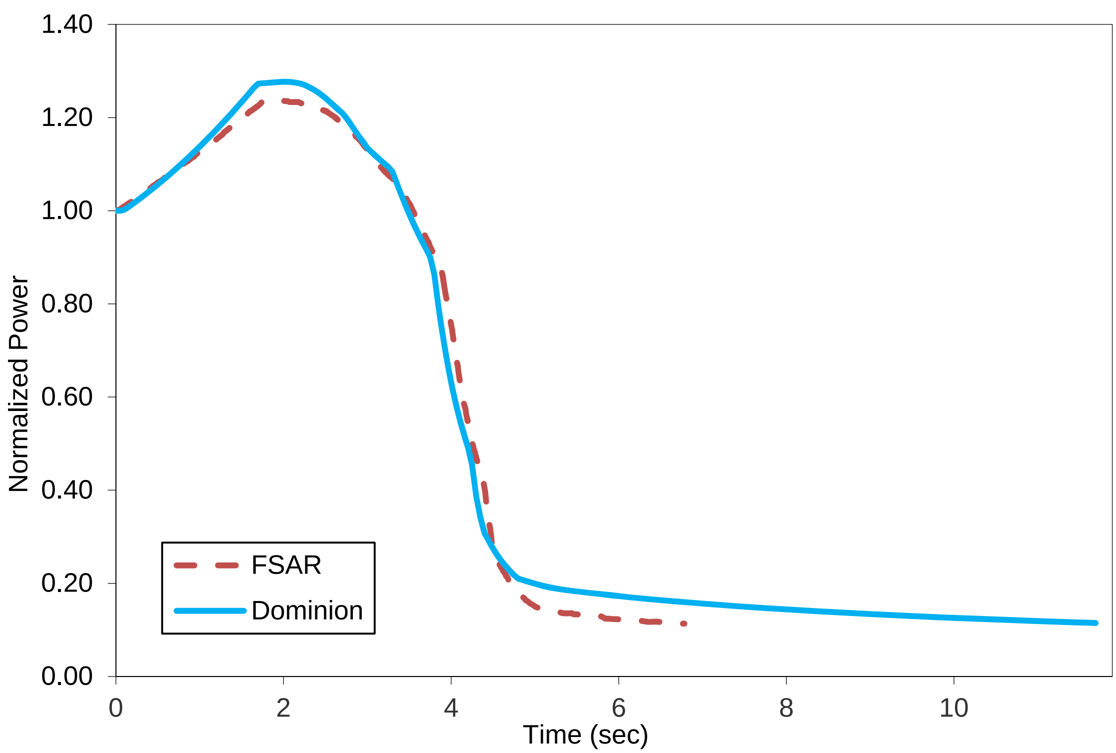


Figure 4.5-5 RWAP – 100 pcm/sec Pressurizer Pressure

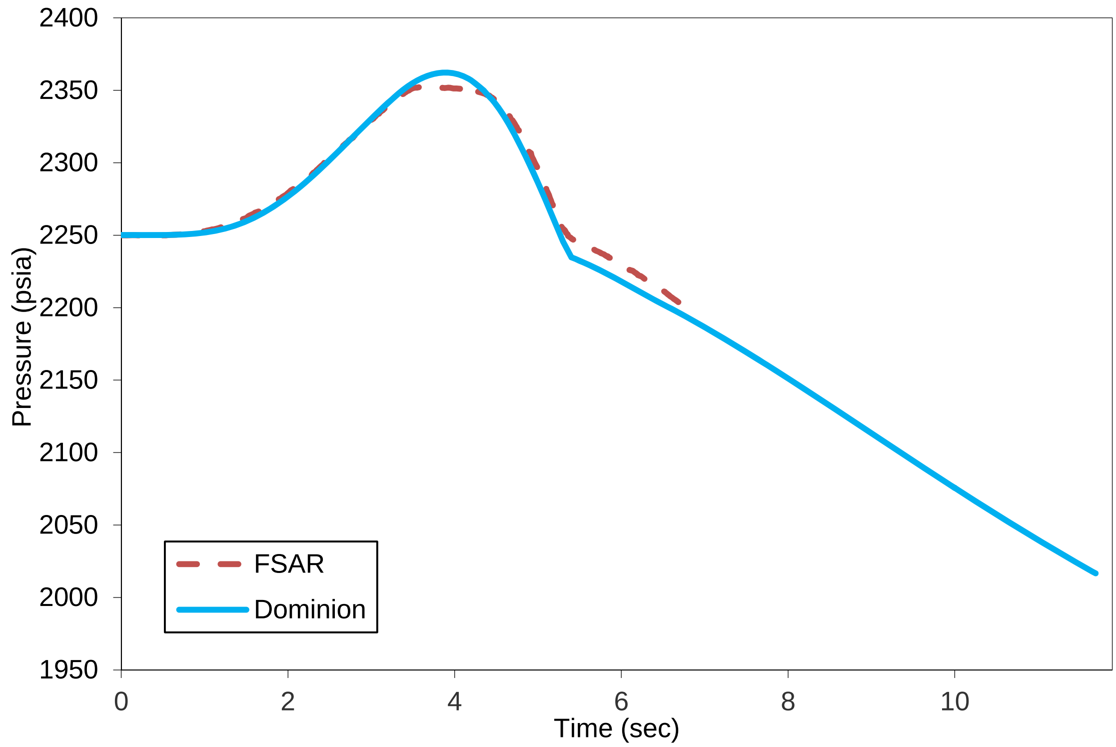
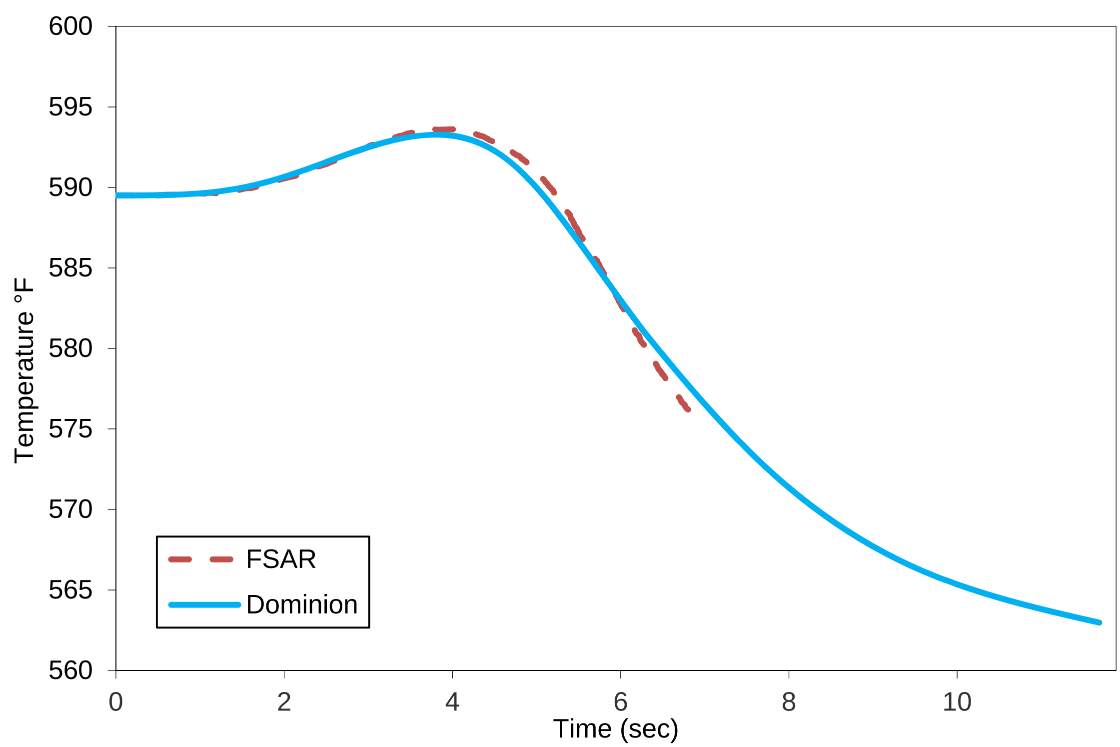


Figure 4.5-6 RWAP – 100 pcm/sec Vessel Average Temperature



Summary - RWAP

The Dominion Millstone model provides results that are similar to the FSAR analysis for the RWAP event. At higher insertion rates, the results match very well. At lower insertion rates, the power increases at a greater rate in the Dominion model than the FSAR model. However, the temperature increases to a higher peak in the FSAR analysis. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

4.6 Main Feedwater Line Break

The Main Feedwater Line Break (MFLB) event is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break) or a RCS heatup. The FSAR analysis presents the RCS heatup scenario.

A major feedwater line rupture is classified as an ANS Condition IV event as discussed in FSAR Section 15.0.1. A main feedwater line rupture is the most limiting event in the decrease in secondary heat removal category. Based on a number of prior analyses, it is concluded in FSAR Section 15.2.8 that the most limiting feedwater line rupture is a double ended rupture of the largest feedwater line, occurring at full power with and without offsite power available. Cases both with and without offsite power available are simulated for the benchmark analysis herein.

The MFLB transient is initiated in the Dominion model by opening the break on steam generator 1 and stopping main feedwater to all four steam generators (SG) as the reactor is operating at full power. Upon transient initation, the break path opens and allows blowdown from the faulted SG secondary side inventory to the atmosphere. The input parameters are the same as those used in the FSAR Chapter 15 analyses as shown in Table 4.6-1 below.

The results for the MFLB transient need to demonstrate that the reactor core remains covered, the RCS does not overpressurize, and the AFW system is able to adequately remove decay heat.

Table 4.6-1 MFLB Input Summary

Parameter	Value	Notes
Initial Conditions		
Core Power (MW)	3723	Includes 2% uncertainty
RCS Flow (gpm)	363,200	Thermal Design Flow
Vessel T_{AVG} (F)	594.5	Nominal + 5 °F
RCS Pressure (psia)	2300	Nominal + 50 psi
Pressurizer Level (%)	71.6	Nominal + 7.6%
SG Level (%)	62 38	Nominal + 12% (Faulted Loop) Nominal – 12% (Intact Loops)
SG tube plugging (%)	10	Maximum
Pump Power (MW/pump)	5.0	Maximum
Assumptions/Configuration		
Low-Low Level Reactor Trip Setpoint	0%	% narrow range span in faulted SG
Pressurizer: sprays, heaters, PORVs	-	Not credited
AFW Temperature (F)	120	Max value
Auxiliary feedwater flow rate (gpm)	-	Variable as function of SG press.
Main Feedwater	0	All MFW assumed lost at time of break
Reactivity Parameters		
Doppler Reactivity Feedback Moderator Feedback	Most Negative	Conservative assumption

Results – MFLB Case With Offsite Power Available

The results for the MFLB case with offsite power available are presented on Figure 4.6-1 through Figure 4.6-8. The nuclear power response (Figure 4.6-1) predicted by the Dominion model is in good agreement with the FSAR data, with the reactor trip occurring on low-low steam generator level. There is a return to power between approximately 100-200 seconds due primarily to moderator reactivity feedback effects during the primary side cooldown prior to steam line isolation (SLI). After that time, the core remains subcritical for the duration of the transient.

The response for pressurizer pressure and pressurizer water volume are shown on Figure 4.6-2 and Figure 4.6-3. The Dominion results trend well with the FSAR results for pressurizer pressure and water volume. One difference is a brief increase in pressurizer pressure and associated insurge into the pressurizer around the point of reactor trip for the Dominion case. This increase occurs due to differences in the primary-to-secondary heat transfer following the reactor and turbine trips between the MNSG FSAR model and the Dominion SNSG. The SNSG responds more quickly to the decrease in secondary side level following the loss of main feedwater compared to the MNSG, which initially experiences less reduction in SG level and associated heat transfer. This effect only occurs for a relatively brief duration. Eventually, steam line isolation (SLI) occurs on low steam line pressure resulting in a primary side heatup as the intact SGs repressurize. Pressurizer pressure increases until the pressurizer safety valve (PSV) setpoint is reached and remains essentially constant at the PSV relief pressure until a downturn in pressure occurs near the end of the transient. This indicates the termination of the event as sufficient cooling is being provided by auxiliary feedwater (AFW) for the removal of primary side energy.

The hot leg and cold leg temperature response is shown on Figure 4.6-4 for the faulted loop and on Figure 4.6-5 for the intact loops. There is good agreement between the Dominion and FSAR cases with temperatures exhibiting the same trends throughout the event and deviating only slightly prior to SLI, which has a negligible effect on the overall results for this comparison due to the long term nature of this event. As noted for the pressure response discussion above, the temperatures are decreasing at the end of the transient indicating adequate long term heat removal.

The Dominion RCS flow fraction results are shown on Figure 4.6-6. Since power to the reactor pumps is not lost for this case, flow is maintained throughout the transient and varies only with coolant conditions. The Dominion case is in good agreement with the FSAR data throughout the transient.

The secondary system pressure response is presented on Figure 4.6-7 where SG pressure increases briefly following the reactor trip then decreases due to the loss of fluid mass through the feed line break. After SLI occurs, the intact SG pressure increases to the MSSV setpoint while the faulted SG pressure continues to decrease to atmospheric pressure as the remaining fluid mass is depleted. The Dominion and FSAR cases show good agreement as both the magnitude and trends of faulted and intact loops are consistent following the point of reactor trip and subsequent SLI.

Figure 4.6-8 shows excellent agreement between the main feedwater break flow rate response in both the Dominion and FSAR case. One difference is seen around the point of reactor trip over a period of approximately 12 seconds that is related to the steam generator modeling differences. As discussed relative to the pressurizer pressure response, the Dominion SNSG model results in a faster reduction in liquid level and more rapid increase in break flow quality such that flow falls off more quickly as the break is uncovering. After this brief transition period the break flow rates continue to agree well and this difference has a negligible effect on the overall transient response.

Figure 4.6-1 MFLB – Nuclear Power (case with power)

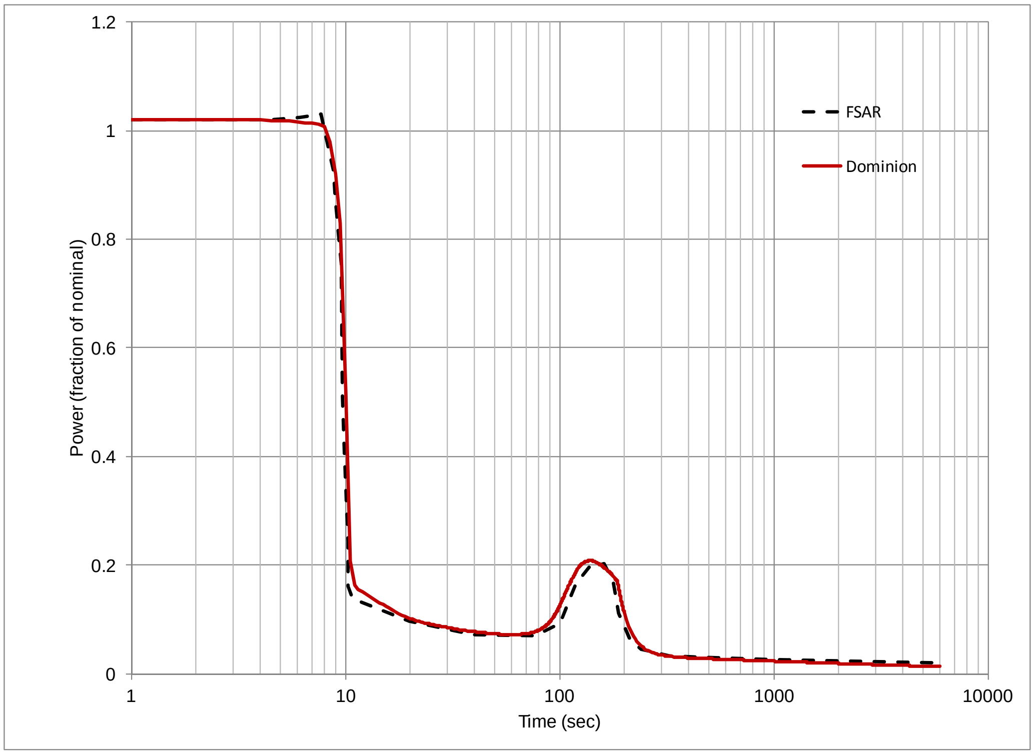


Figure 4.6-2 MFLB – Pressurizer Pressure (case with power)

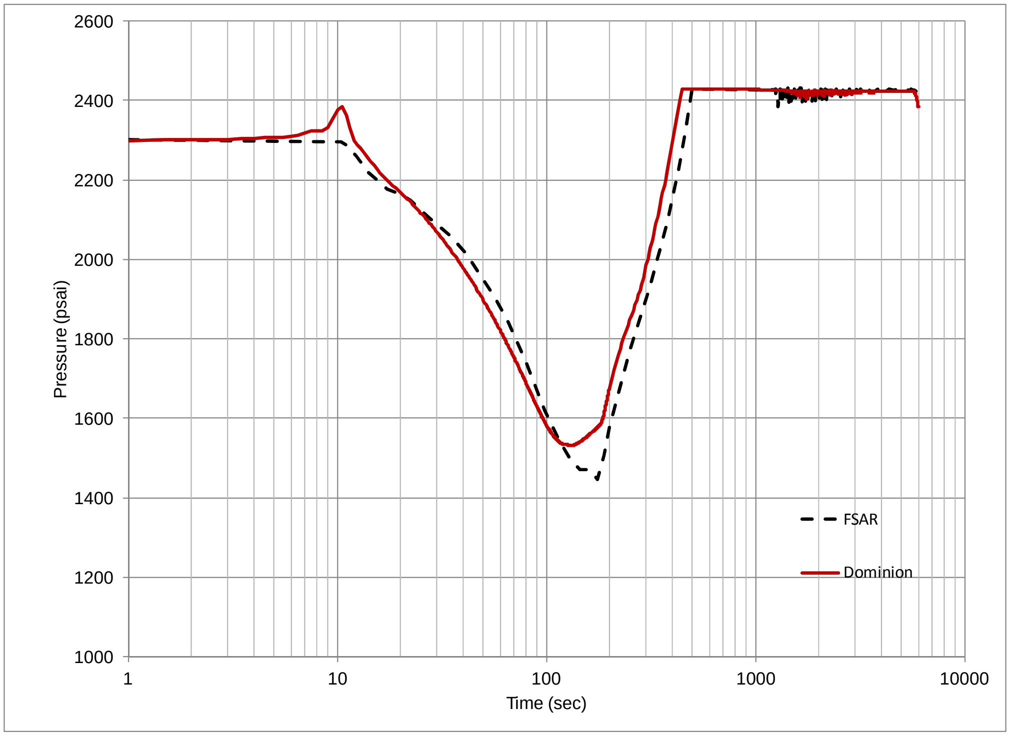


Figure 4.6-3 MFLB – Pressurizer Liquid Volume (case with power)

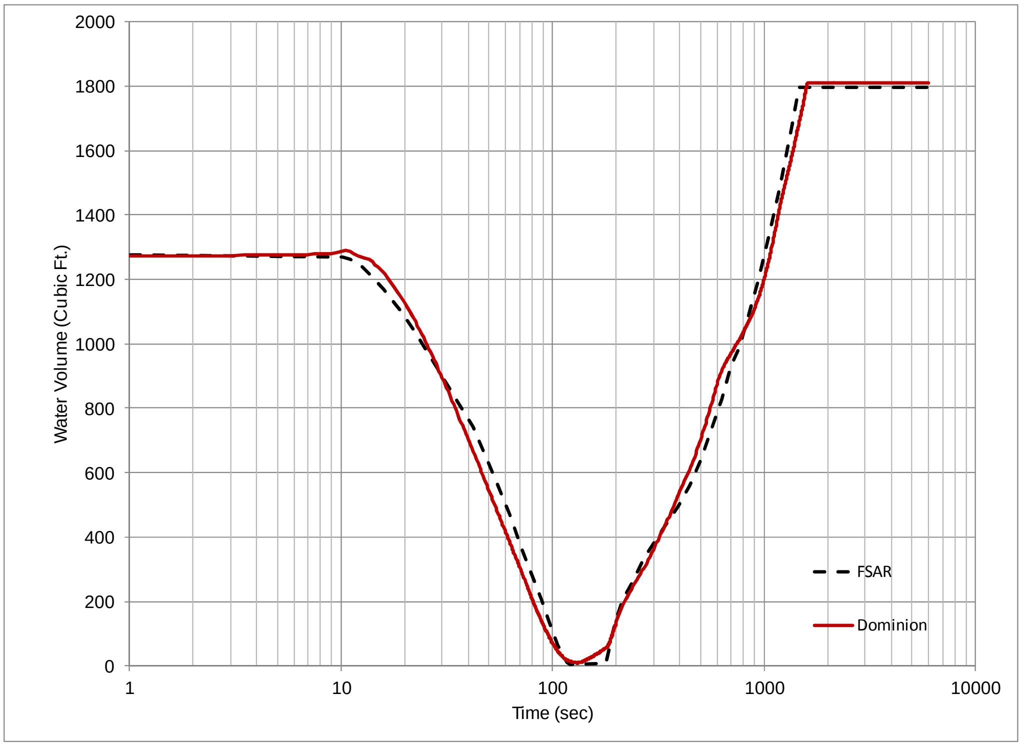


Figure 4.6-4 MFLB – RCS Temperatures – Faulted Loop (case with power)

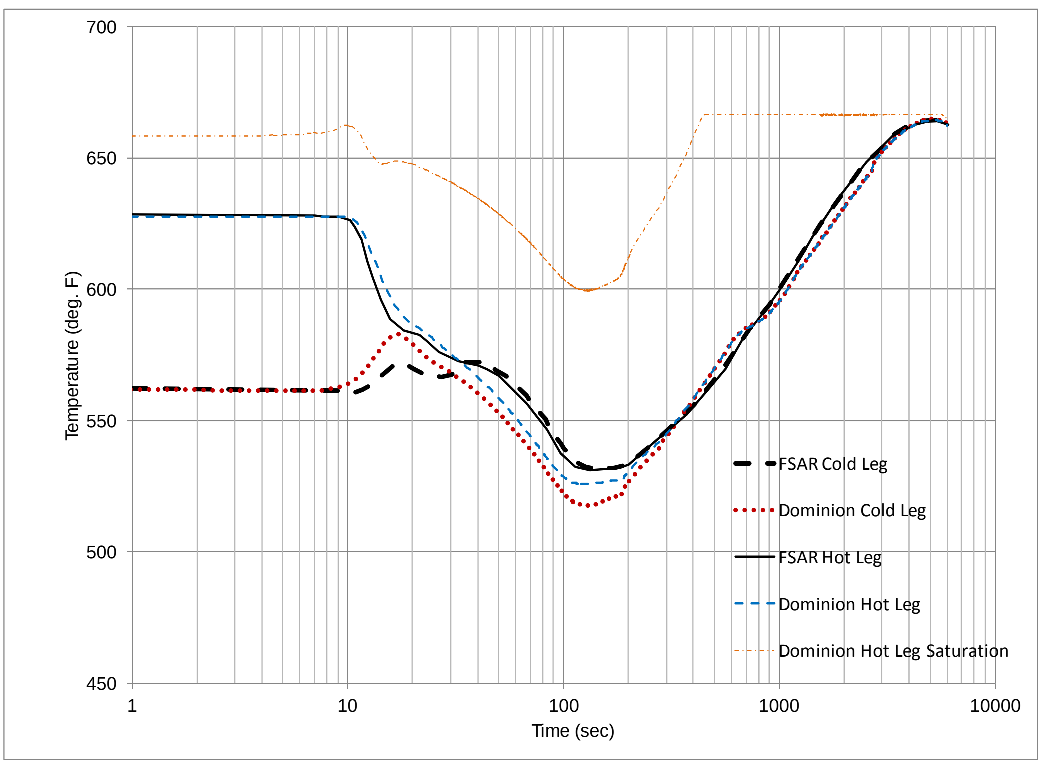


Figure 4.6-5 MFLB – RCS Temperatures – Intact Loops (case with power)

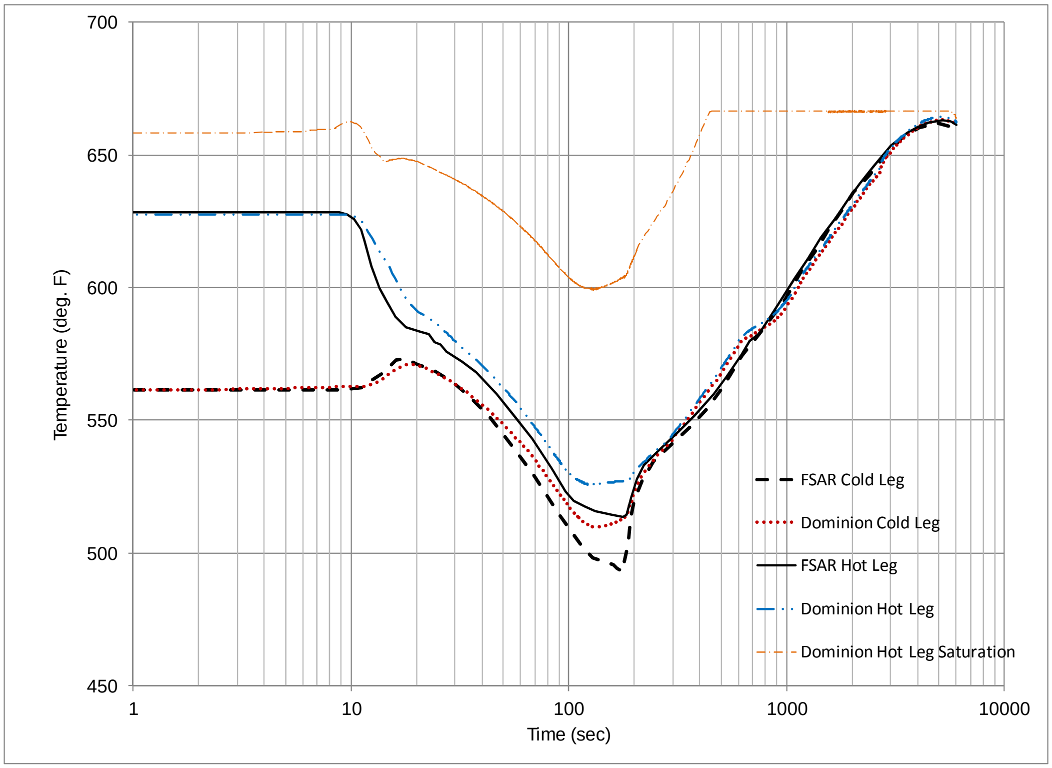


Figure 4.6-6 MFLB – Normalized RCS Flow (case with power)

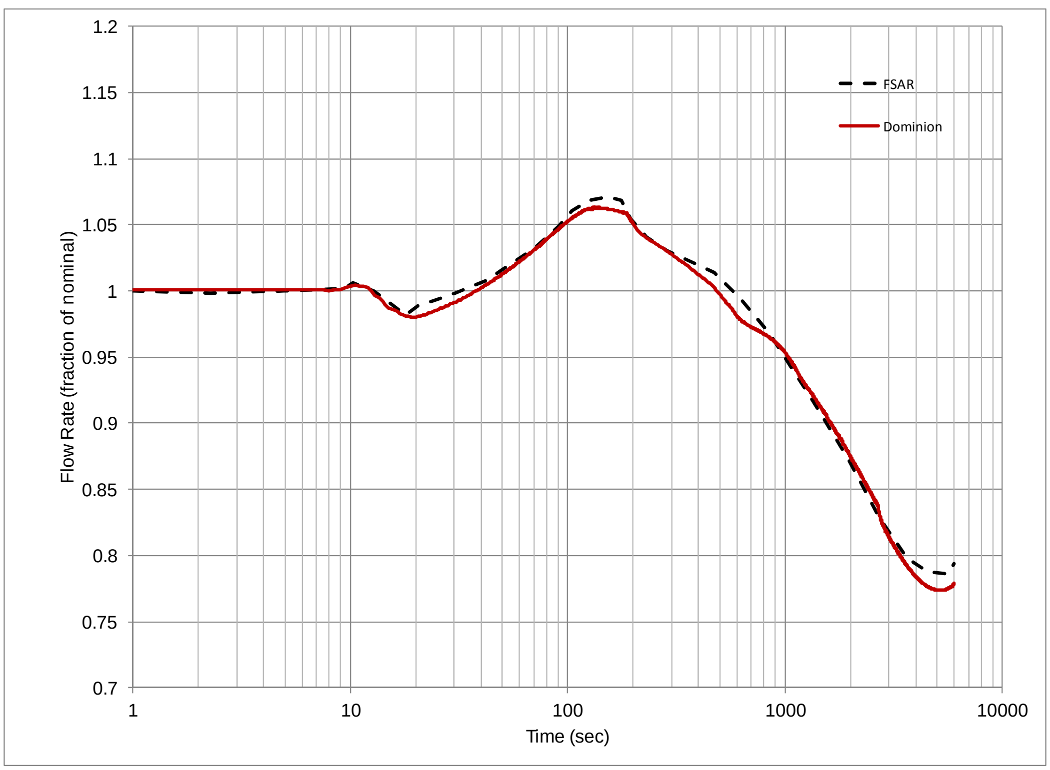


Figure 4.6-7 MFLB – Steam Generator Pressure (case with power)

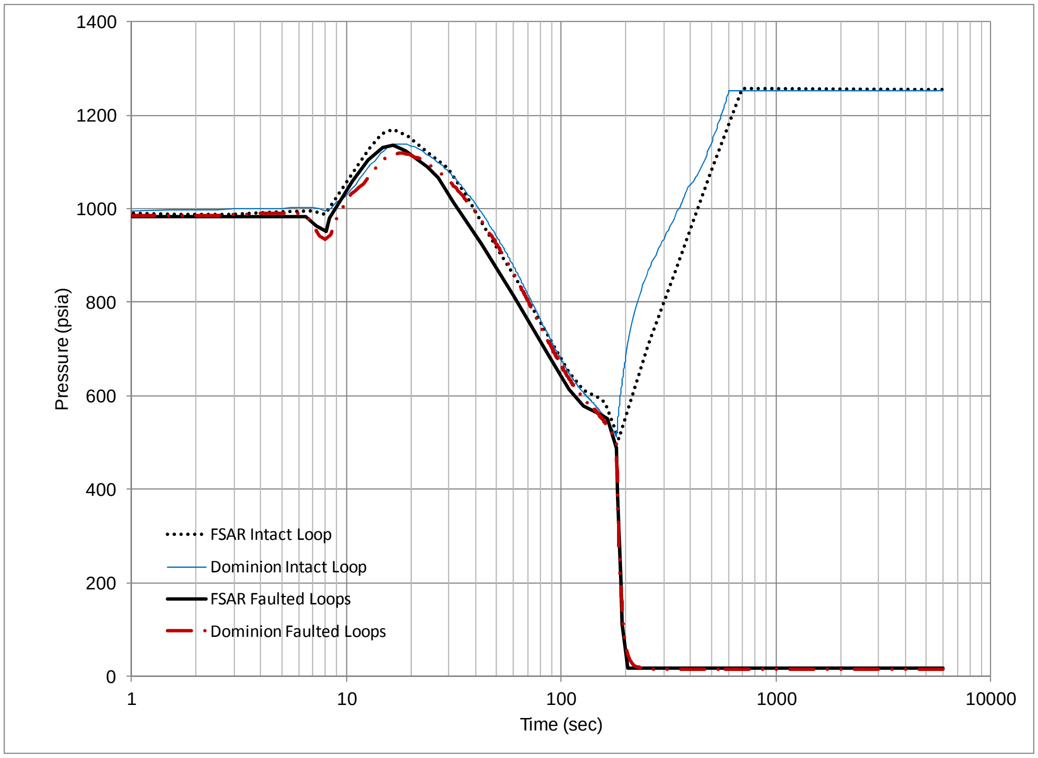
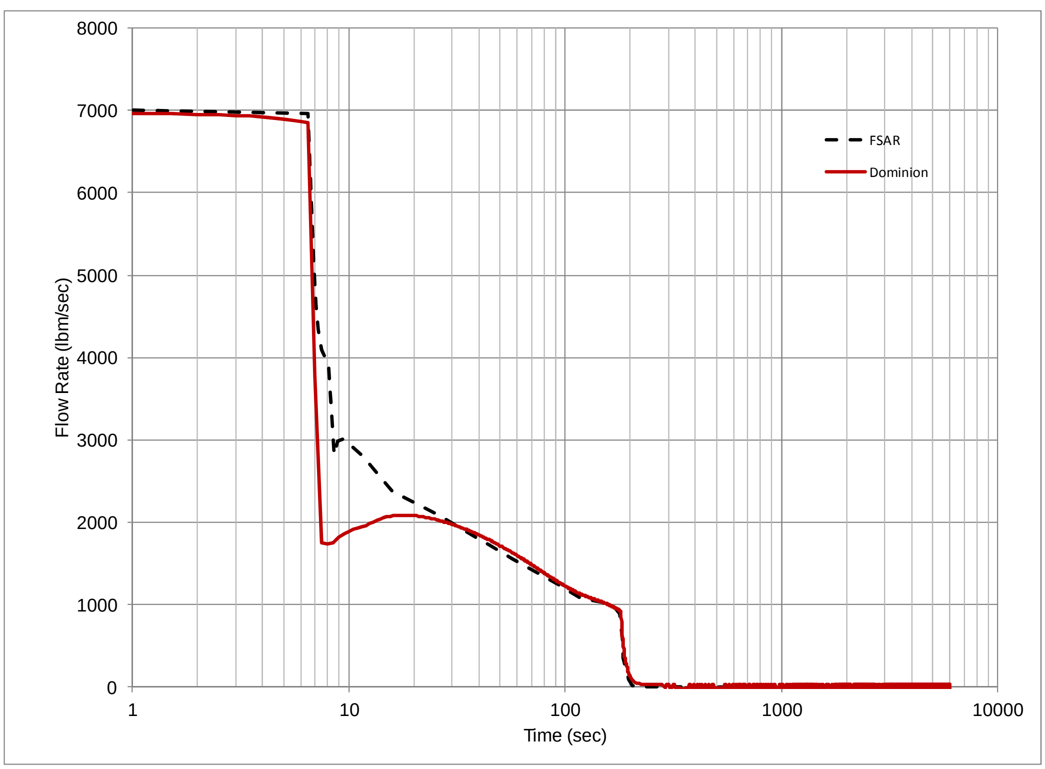


Figure 4.6-8 MFLB – Feed Line Break Flow (case with power)



Results – MFLB Case Without Offsite Power Available

The results for the MFLB case without offsite power are similar to the case with power available but are generally less limiting for long-term primary side heat removal since the RCPs are not running and adding heat to the primary side fluid.

The nuclear power response (Figure 4.6-9) predicted by the Dominion case is in good agreement with the FSAR data. As shown for this case, there is no return to power during the early portion of the cooldown due to less reactivity feedback and the reactor core remains subcritical for the duration of the transient.

The responses for pressurizer pressure and primary side temperatures are shown on Figures 4.6-10 through 4.6-12. As discussed above for the case with offsite power, the Dominion case exhibits a brief increase in pressure around the time of reactor trip but otherwise the response is similar to the FSAR case with long-term pressure maintained at the PSV setpoint. The hot leg and cold leg temperature response shown on Figure 4.6-11 and Figure 4.6-12 also demonstrate similar trends. One difference is that the cooldown that occurs prior to SLI is more pronounced for the Dominion case, which is primarily attributed to higher primary to secondary heat transfer. This is the result of a somewhat slower rate of flow decrease following the RCP trip for the Dominion case, resulting in maintaining better primary side heat removal during that phase. In addition, SLI occurs slightly later in the Dominion case, which also enhances heat removal prior to the time of isolation. Similarly, the delay in break isolation delays the point of steam generator dry-out, such that additional heat is extracted through the break. As shown, these differences have little effect on the long-term temperature response as the Dominion and FSAR temperatures agree very well through the end of the transient. This case results in lower long-term temperatures, as the RCPs trip due to the loss of offsite power and do not contribute any pump heat to the system.

The secondary system pressure response, presented in Figure 4.6-13, is similar to the response for the case with power. Since there is less primary side heat generation and heat removal for this case, the SG depressurizes more quickly and SLI occurs earlier in the transient, compared to the case with offsite power available. Long term trends are similar with heat removal via the MSSVs on the intact SGs. There is good agreement between the Dominion and FSAR cases with the FSAR case depressurizing slightly faster prior to SLI.

The Dominion RCS flow fraction results are in good agreement with the FSAR result as shown on Figure 4.6-14, where the loss of flow associated with the loss of power and associated RCP trip are seen. As noted above, the flow decreases somewhat more quickly for the FSAR case, which appears to affect the intermediate temperatures but does not impact the long term temperature results.

Figure 4.6-15 shows good agreement between the main feedwater break flow rate response in both the Dominion and FSAR data. The small differences seen around the point of reactor trip are due to differences in the Dominion SNSG and the FSAR MNSG as discussed above for the case with power available. That is, the Dominion SNSG model results in a faster reduction in liquid level and more rapid increase in break flow quality such that flow falls off more quickly as the break is uncovering. After this brief transition period the break flow rates continue to agree well and this difference has a negligible effect on the overall transient response

Figure 4.6-9 MFLB – Nuclear Power (case without power)

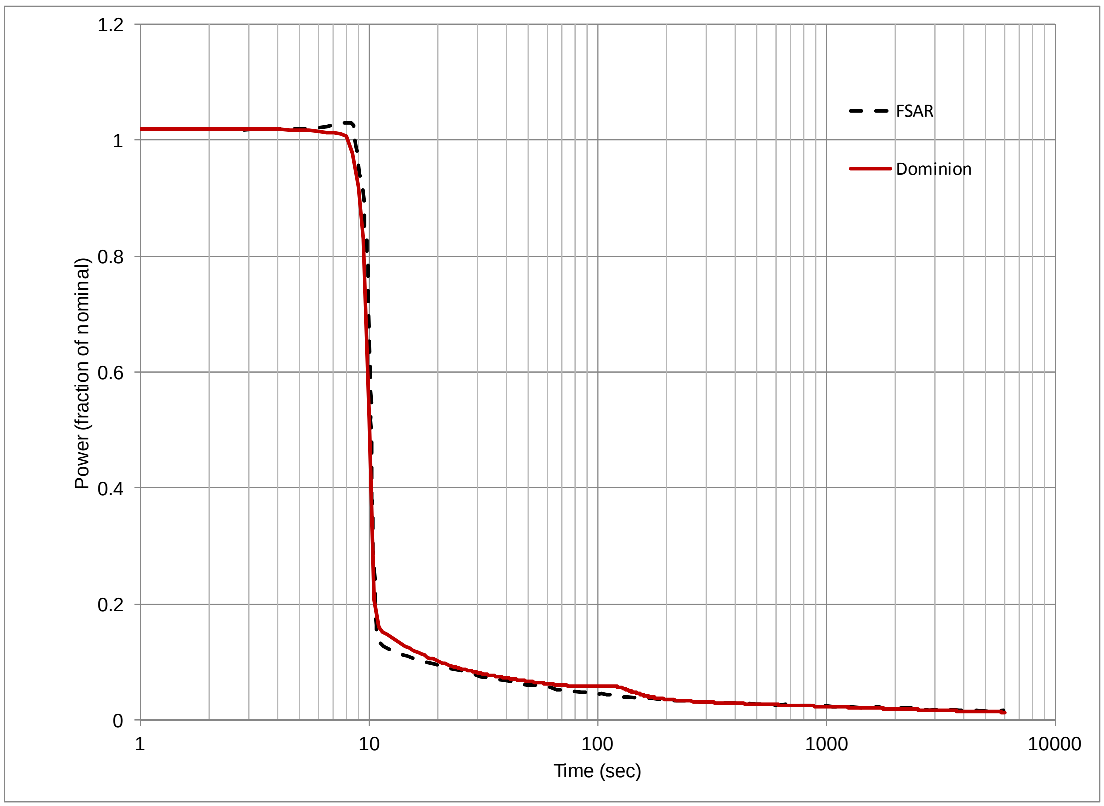


Figure 4.6-10 MFLB – Pressurizer Pressure (case without power)

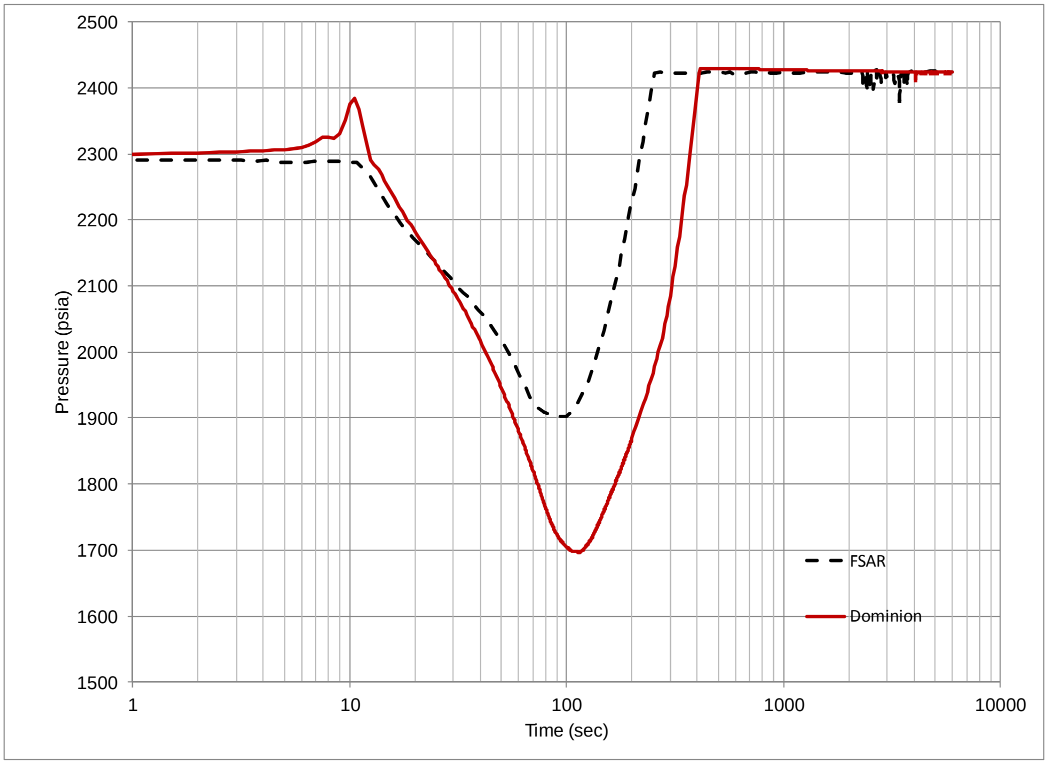


Figure 4.6-11 MFLB – RCS Temperatures – Faulted Loop (case without power)

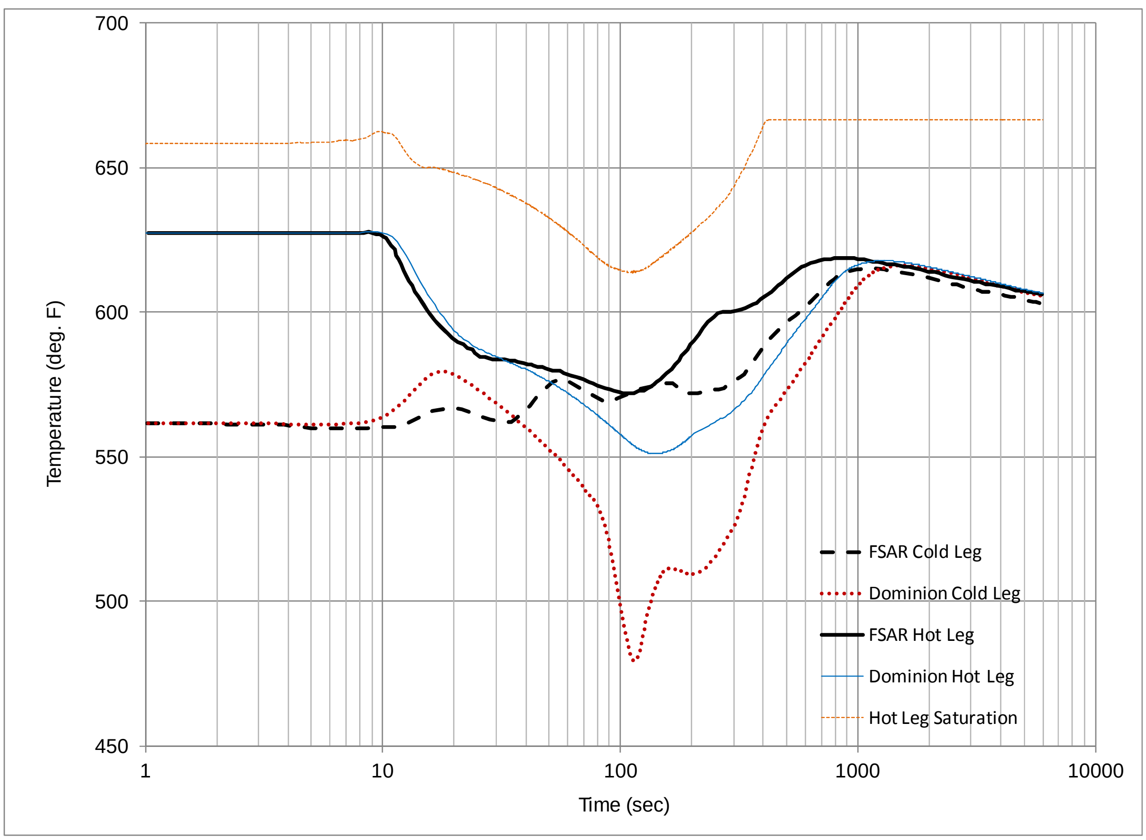


Figure 4.6-12 MFLB – RCS Temperatures – Intact Loops (case without power)

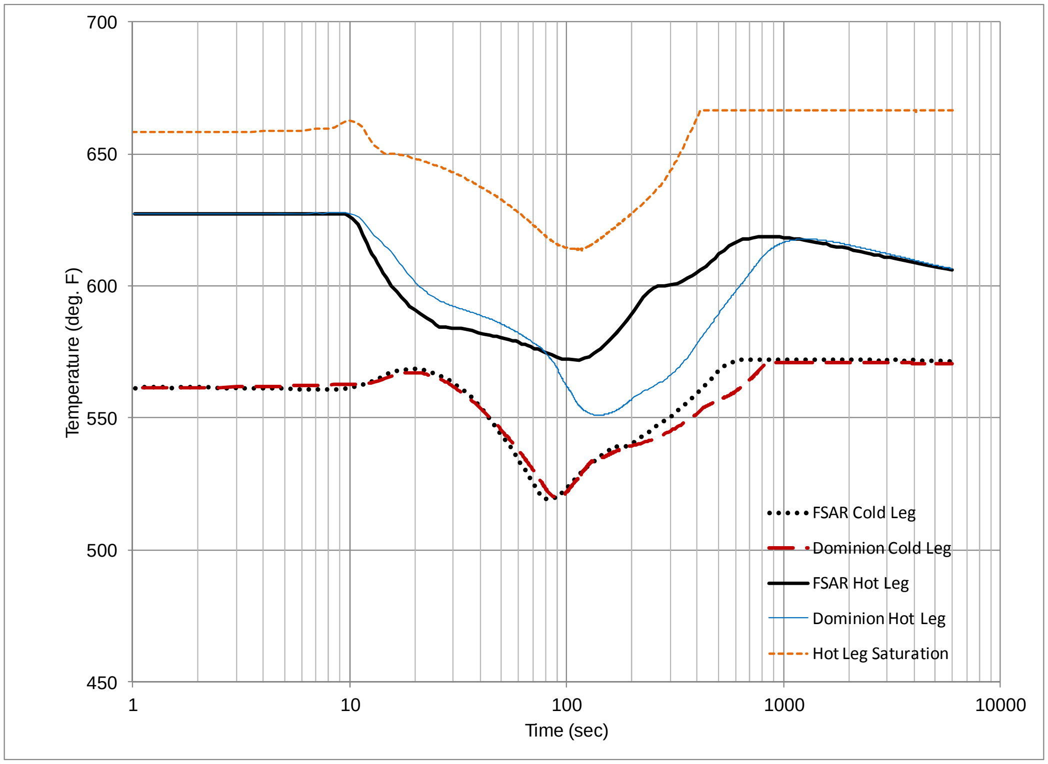


Figure 4.6-13 MFLB – Steam Generator Pressure (case without power)

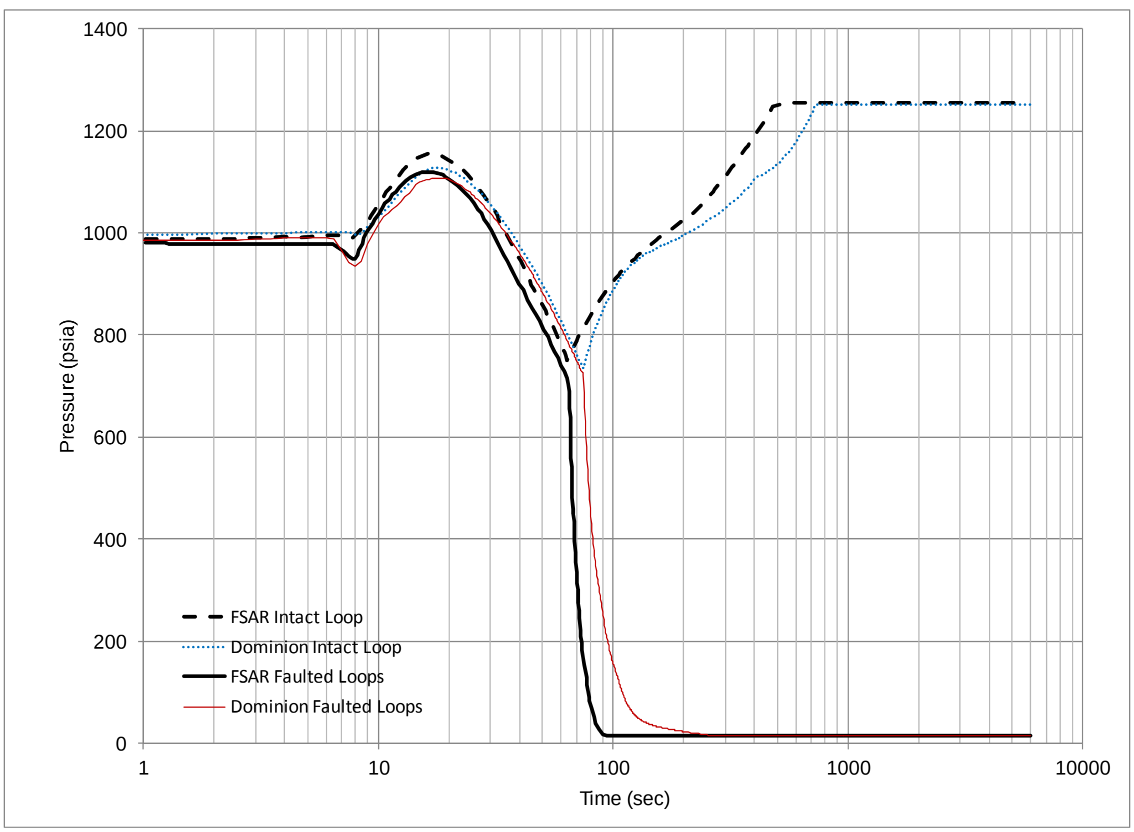


Figure 4.6-14 MFLB – Normalized RCS Flow (case without power)

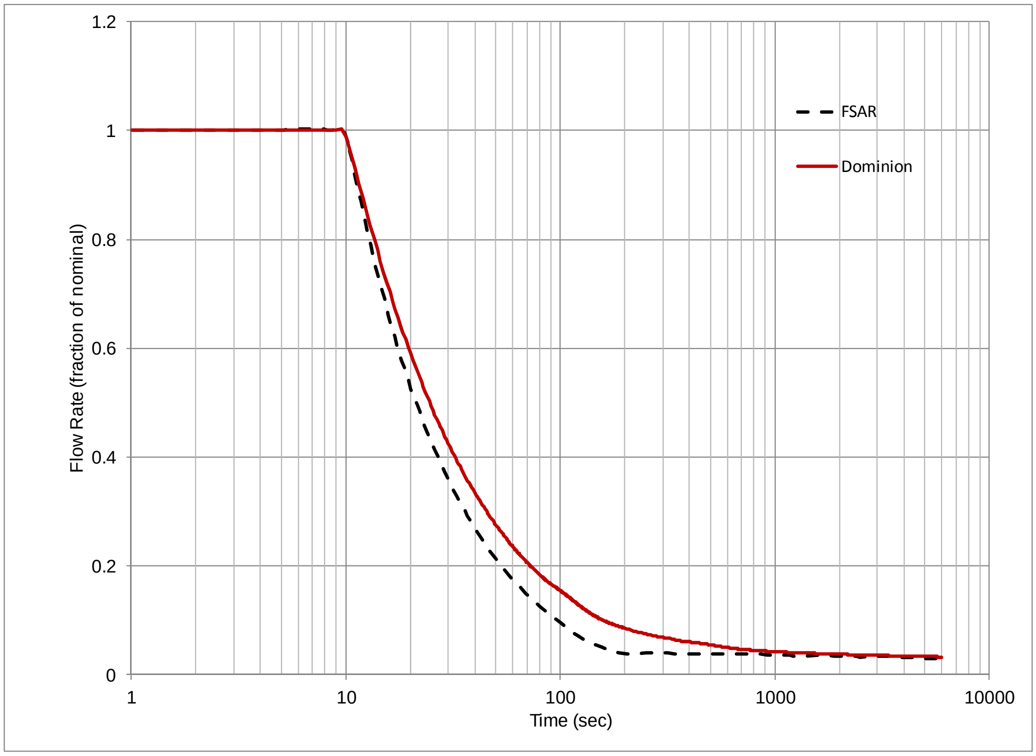
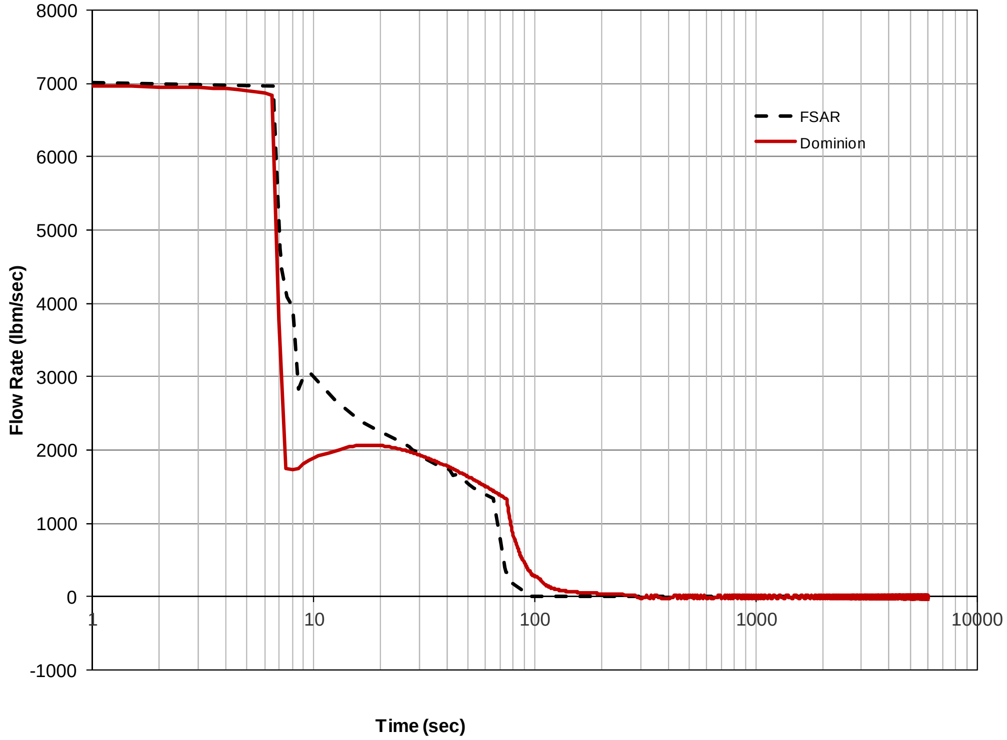


Figure 4.6-15 MFLB – Feed Line Break Flow (case without power)



Summary - MFLB

The Dominion Millstone model provides results that are similar to the FSAR analysis for the MFLB event. Two cases are analyzed, one with offsite power available and another without offsite power. Some small differences are observed early in the transient for RCS temperatures, which are attributable to differences in the Dominion SNSG model and the FSAR MNSG model; however, these differences have a negligible effect on the long-term primary side heat removal and associated temperature response. All acceptance criteria are satisfied for both cases.

4.7 Steam Generator Tube Rupture

The Steam Generator Tube Rupture (SGTR) event is a breach of the Reactor Coolant Pressure (RCP) Boundary via a steam generator (SG) tube. The accident examined is the complete severance of a single steam generator tube. Such a break results in a loss of Reactor Coolant System (RCS) fluid to the secondary side of the affected SG. Two different analyses were performed for the SGTR event including a thermal-hydraulic case to determine the mass releases to atmosphere for radiation dose, and a case for the margin to SG overfill. These analysis cases are described in FSAR Sections 15.6.3.2.2 and 15.6.3.2.1, respectively, where it is noted that the FSAR analyses are performed using the LOFTTR2 computer code. The SGTR is classified as an ANS Condition IV event as discussed in FSAR Section 15.0.1.

The SGTR transient is initiated from full power by modeling the complete severance of a SG tube. Upon transient initiation, the break path opens and allows fluid to flow from the RCS primary into the ruptured SG secondary. Several operator actions are credited in the analysis to mitigate the effect of the transient. These operator actions and other input parameters assumed for this analysis are shown in Table 4.7-1 below.

Table 4.7-1 SGTR Input Summary

Parameter	Value	Notes
NSSS Power (MW)	3739	Includes 2% core power uncertainty; 16 MW reactor coolant pump power
RCS Flow (gpm)	363,200	Thermal Design Flow
Vessel T_{AVG} (F)	571.5	Low Tavg with coastdown
RCS Pressure (psia)	2200	Nominal - 50 psi
Pressurizer Level (%)	45.4	Consistent with Low Tavg
SG tube plugging (%)	0	Mass release case. 10% assumed for overfill case
Auxiliary feedwater flow rate (gpm)	1200	Maximum total
Loss of Offsite Power (LOOP)	Assumed	Occurs at reactor trip
Single failure	Relief valve failure	Mass Release – ADV fails open on ruptured SG at time of steam line isolation. Overfill – ADV bypass valve fails to function on two intact SGs.
Key Operator Actions		
Isolate AFW flow to the ruptured SG	See notes	Based on achieving target SG level
Isolate ruptured steam generator	25 minutes	After initiation of break
Isolate failed opened ADV (mass release case only)	20 minutes	After ADV fails
Initiate RCS cooldown	8 minutes	After ruptured SG is isolated
Initiate RCS Depressurization	3 minutes	After RCS cooldown is complete
Initiate SI flow termination	6 minutes	After RCS depressurization complete (or based on termination criteria)

Results – SGTR Mass Release Case

The results for the Mass Release case are provided on Figure 4.7-1 through Figure 4.7-9 and the Sequence of Events is presented in Table 4.7-2. The pressurizer pressure response is shown on Figure 4.7-1. The Dominion pressurizer pressure tracks closely with the FSAR data through most of the event. After SI is isolated near the end of the event, the pressures diverge as the primary and secondary side pressures equilibrate, with the Dominion pressure decreasing more due to the lower secondary side pressure (Figure 4.7-3). This phase of the event is discussed in additional detail below. Similarly, the pressurizer level response shown on Figure 4.7-2 shows similar trends between the Dominion response and the FSAR data. During the RCS cooldown phase (approximately 3200-3700 seconds), the FSAR level decreases more than the Dominion level. This occurs as the primary to secondary heat transfer is reduced for the Dominion case due to the loss of natural circulation flow on the ruptured SG and during a period when the SI flow is increasing significantly due to the reduction in RCS pressure. These points are discussed in additional detail below. After SI is isolated, the longer duration in break flow for the FSAR case is reflected in lower pressurizer level at the end of the transient. It is noted that these divergences occur late in the transient well after the flow path to atmosphere through the failed ADV has been isolated and do not have a significant effect on the overall results.

The SG pressure response for the ruptured and intact SGs is shown on Figure 4.7-3. As shown, the Dominion and FSAR pressures for the intact SGs (dashed lines) are in good agreement. For the ruptured SGs, there is also good agreement although the pressures diverge near the end of the transient. This is an indication that the primary-to-secondary heat transfer for the Dominion case is significantly reduced, which is due primarily to the effect of the RCS cooldown on natural circulation RCS flows and the associated heat transfer to the ruptured SG. After the failed ADV is isolated (2702 seconds), the pressure in the ruptured SG increases toward the relief valve setpoint for both the Dominion and FSAR cases. During this time period, the RCS cooldown is initiated on the intact SGs (3182 seconds) as indicated by the decreasing intact SG pressures, which ultimately reduces the heat transfer to the ruptured SG and slows the rate of pressure increase. As shown, the FSAR pressure slowly increases toward the relief valve setpoint while the Dominion pressure turns over and slowly begins to decrease, indicating that there is insufficient heat transfer from the RCS primary to sustain secondary side pressure. A better understanding of this is obtained from Figure 4.7-9, where the Dominion RCS flow rate for the ruptured loop decreases to a negligible value at approximately 3600 seconds. This occurs when the RCS temperature difference in the ruptured loop (Figure 4.7-5) has been reduced to a value that is unable to sustain appreciable natural circulation flow and reverse heat transfer is occurring from the SG secondary into the RCS. Even though more energy is being removed by the ruptured SG for the FSAR case, the mass release rates to the atmosphere are very small for the remainder of the transient as shown on Figure 4.7-7.

Natural circulation continues to be maintained in the intact RCS loops following the RCS cooldown and most of the heat removal occurs through the intact SGs as indicated by the mass release rates to the atmosphere shown on Figure 4.7-8.

The primary side temperature response is shown on Figure 4.7-4 for the intact SGs and Figure 4.7-5 for the ruptured SGs. As shown on Figure 4.7-4, the Dominion and FSAR results for the intact SG temperatures are in very good agreement. For the ruptured SGs, there is good agreement between the Dominion and FSAR cases until about 3600 seconds, at which time the Dominion cold leg temperature trends below the FSAR results. This is due to the negligible natural circulation flow rate discussed above that occurs on the ruptured loop as a result of the RCS cooldown. With the small RCS loop flow rate, the SI flow has a more noticeable effect on cold leg fluid temperature. The FSAR cold leg temperature for the ruptured loop also decreases well below the saturation temperature for the SG secondary, but is likely mixing with a higher natural circulation flow since some heat transfer is being sustained. Nevertheless, this has very little effect on the overall results for the transient since most of the heat removal occurs through the intact SGs during this time as discussed above and the ruptured SG has been previously isolated.

The break flow rate through the ruptured SG tube is shown on Figure 4.7-6. There is very good agreement between the Dominion and FSAR cases until the period late in the transient after SI has been isolated and the break flow is trending towards zero. This difference occurs late in the transient and the effect on the overall results is small since the ruptured SG has been isolated by this time. Additional discussion relative to this response is provided with the Overfill case below.

Table 4.7-2 SGTR – Mass Release Case Sequence of Events

Event	Time (seconds)	
	Dominion	**FSAR**
SG Tube Ruptured	0.0	0
Reactor Trip (OTDT)	208	135
SI Actuated	216	143
AFW Flow Initiated	268	195
Ruptured SG Steamline Isolated	1500	1500
Ruptured SG ADV fails open	1502	1502
Ruptured SG ADV isolated	2702	2702
RCS Cooldown Initiated	3182	3182
RCS Cooldown Terminated	3740	3690
RCS Depressurization Initiated	3920	3872
RCS Depressurization Terminated	3991	3952
SI Terminated	4352	4312
Total Break Flow Terminated	5635	6412

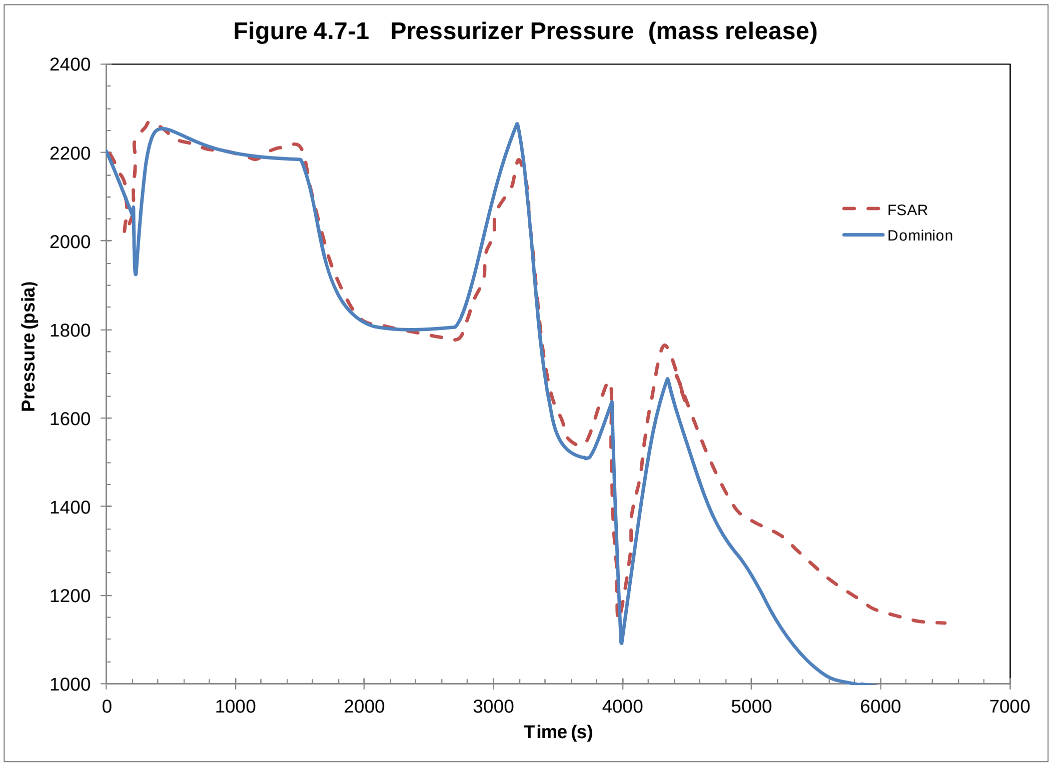


Figure 4.7-1 Pressurizer Pressure (mass release)
Pressure (psia)
2400
2200
2000
1800
1600
1400
1200
1000
0
1000
2000
3000
4000
5000
6000
7000
Time (s)
FSAR
Dominion

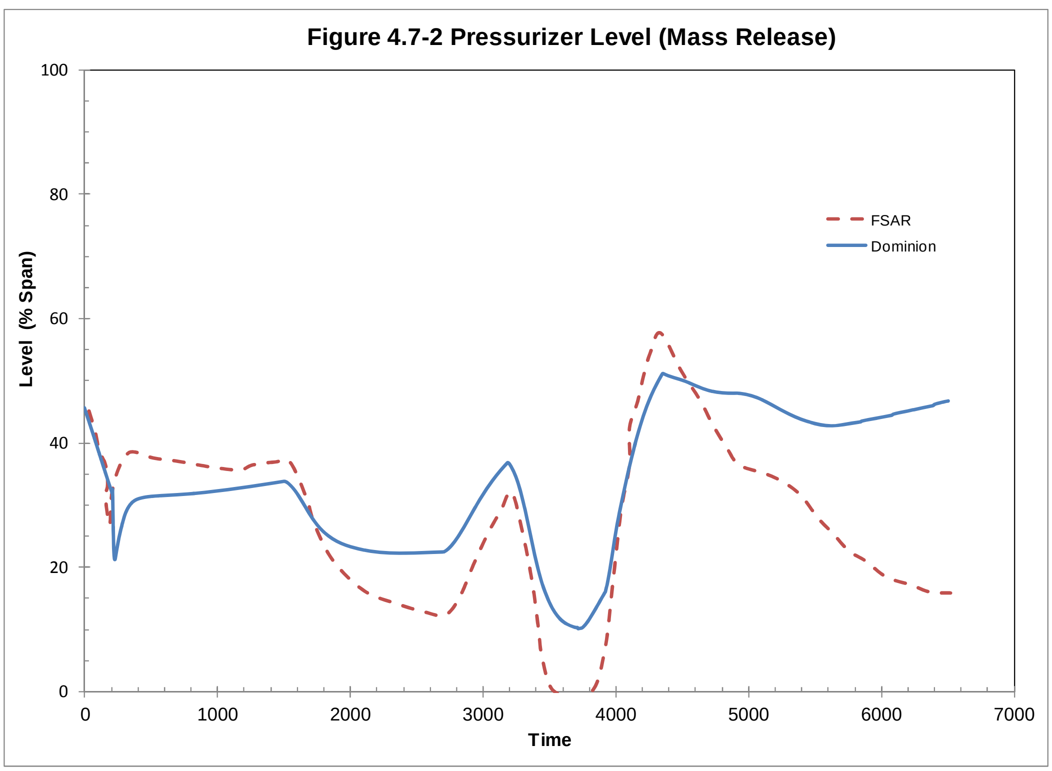


Figure 4.7-2 Pressurizer Level (Mass Release)
Level (% Span)
100
80
60
40
20
0
0
1000
2000
3000
4000
5000
6000
7000
Time
FSAR
Dominion

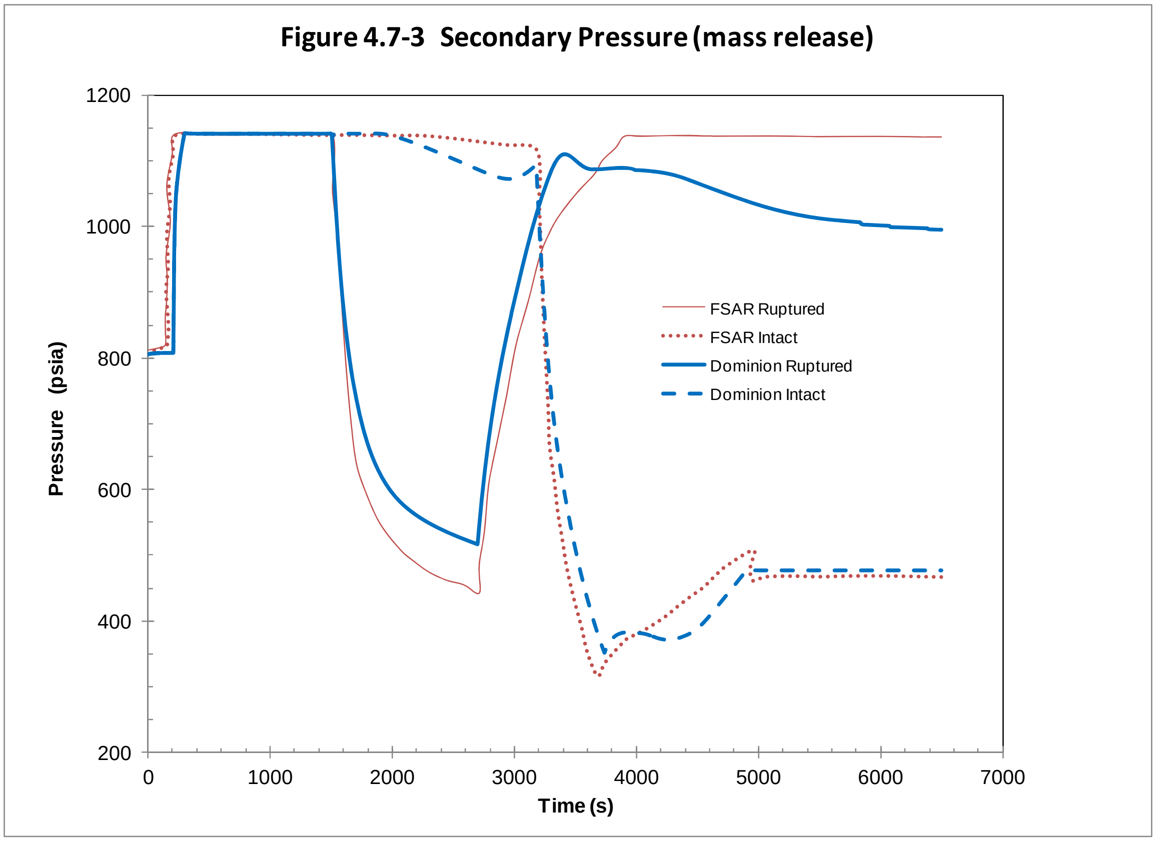


Figure 4.7-3 Secondary Pressure (mass release)
Pressure (psia)
1200
1000
800
600
400
200
0
1000
2000
3000
4000
5000
6000
7000
Time (s)
FSAR Ruptured
FSAR Intact
Dominion Ruptured
Dominion Intact

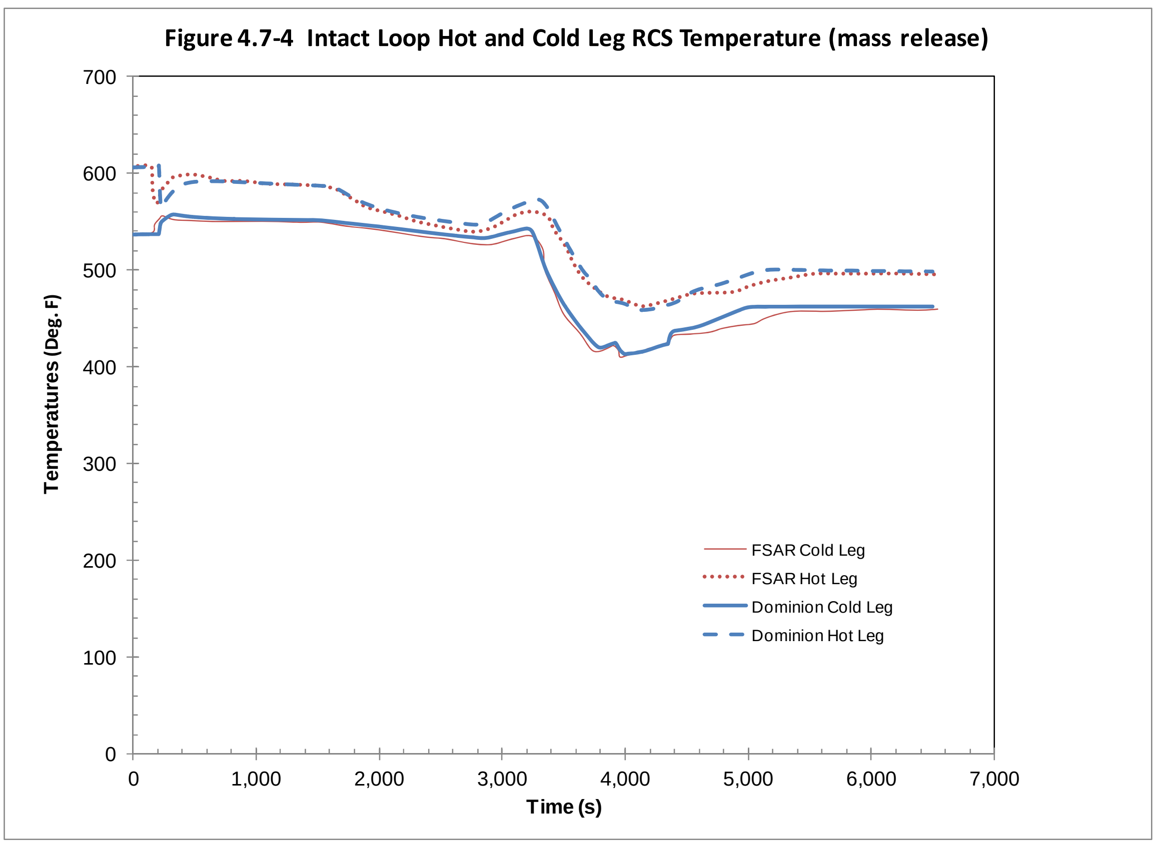


Figure 4.7-4 Intact Loop Hot and Cold Leg RCS Temperature (mass release)
Temperatures (Deg. F)
700
600
500
400
300
200
100
0
1,000
2,000
3,000
4,000
5,000
6,000
7,000
Time (s)
FSAR Cold Leg
FSAR Hot Leg
Dominion Cold Leg
Dominion Hot Leg

Figure 4.7-5 Ruptured Loop Hot and Cold Leg RCS Temperature (mass release)

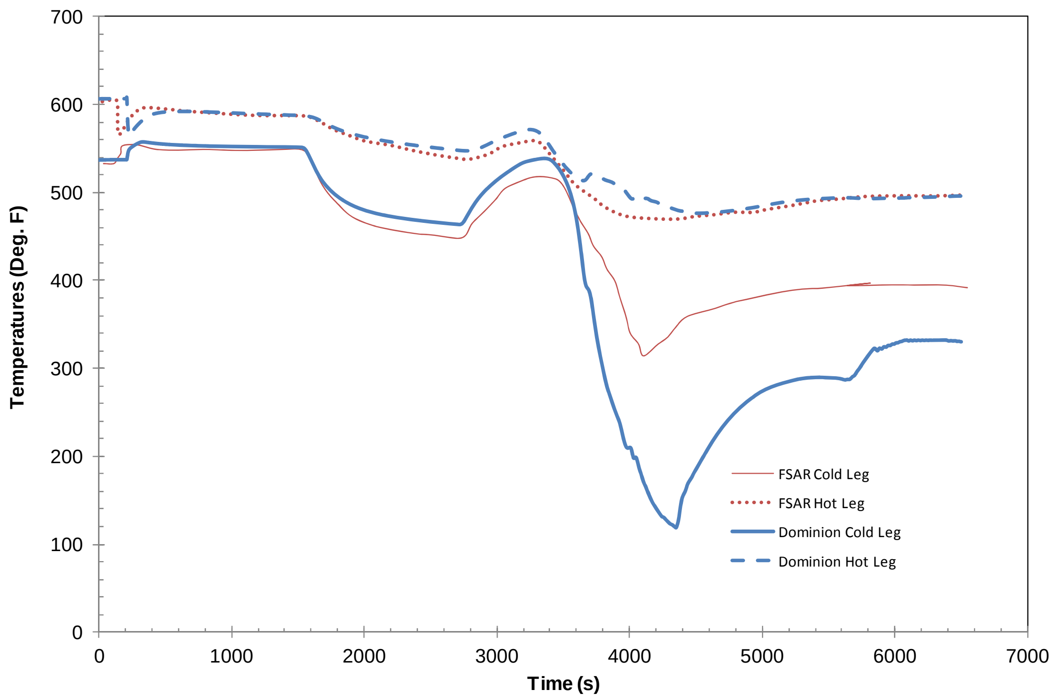
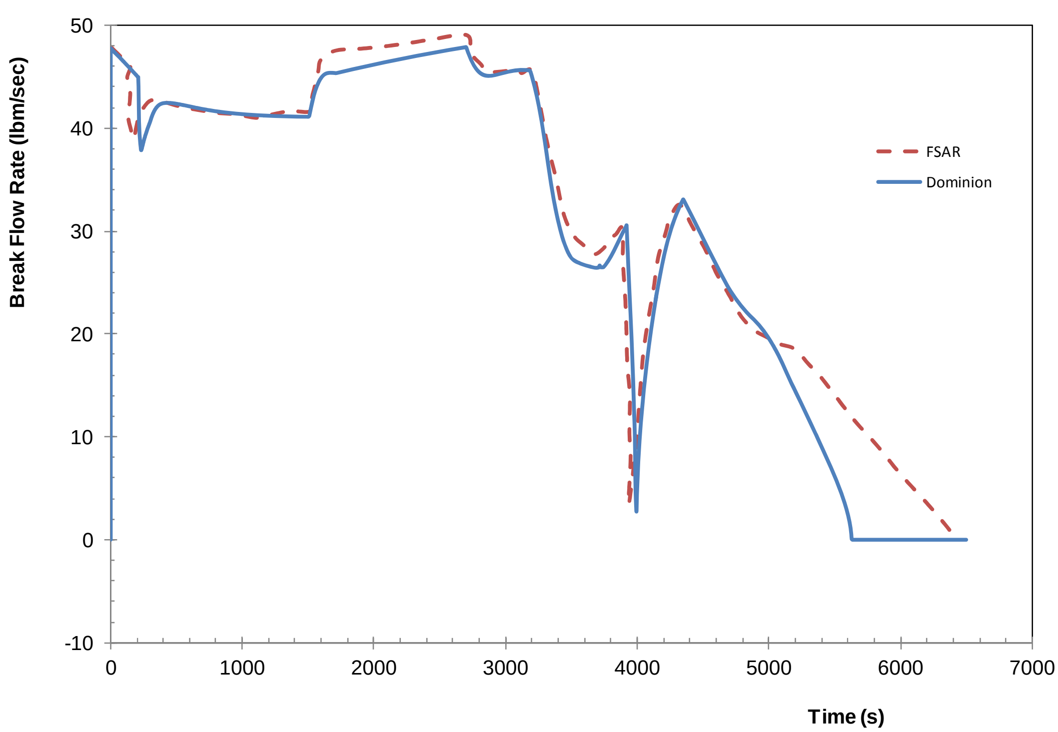


Figure 4.7-6 Primary to Secondary Break Flow Rate (mass release)



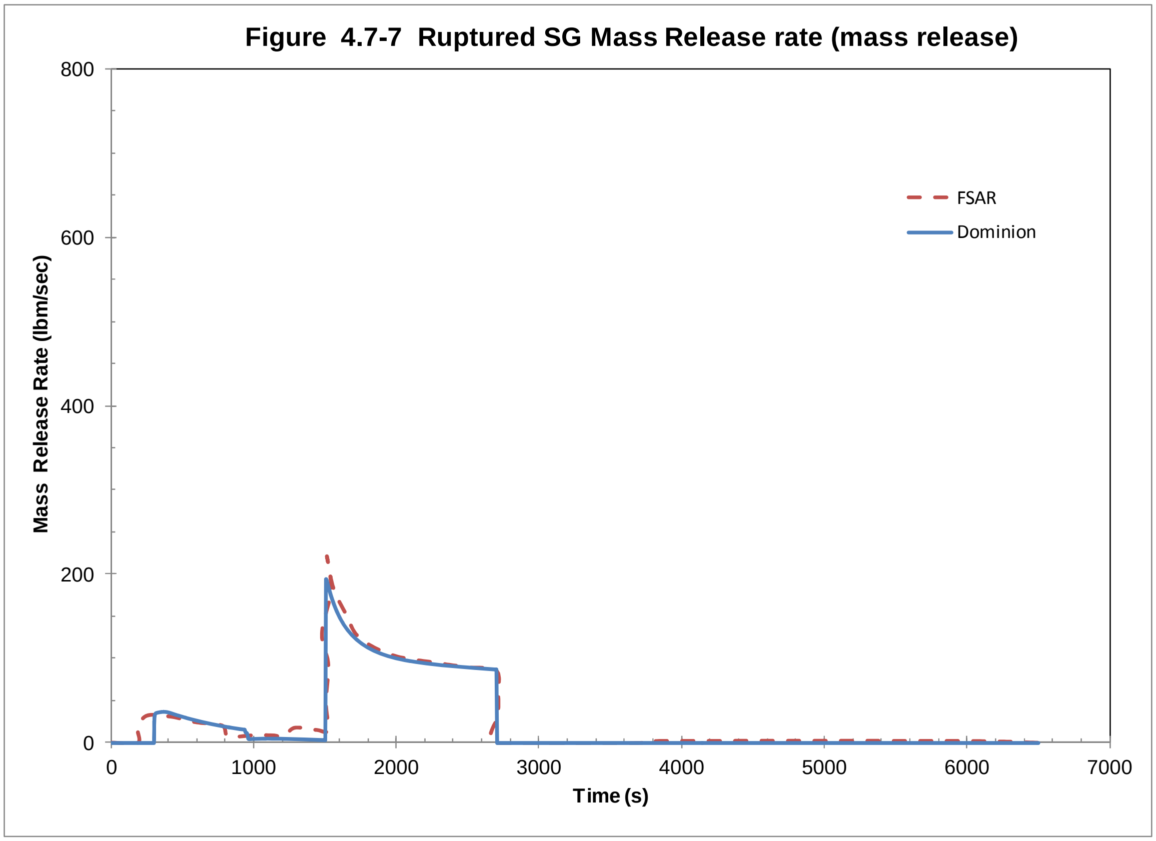


Figure 4.7-7 Ruptured SG Mass Release rate (mass release)
800
600
400
200
0
Mass Release Rate (lbm/sec)
0
1000
2000
3000
4000
5000
6000
7000
Time (s)
FSAR
Dominion

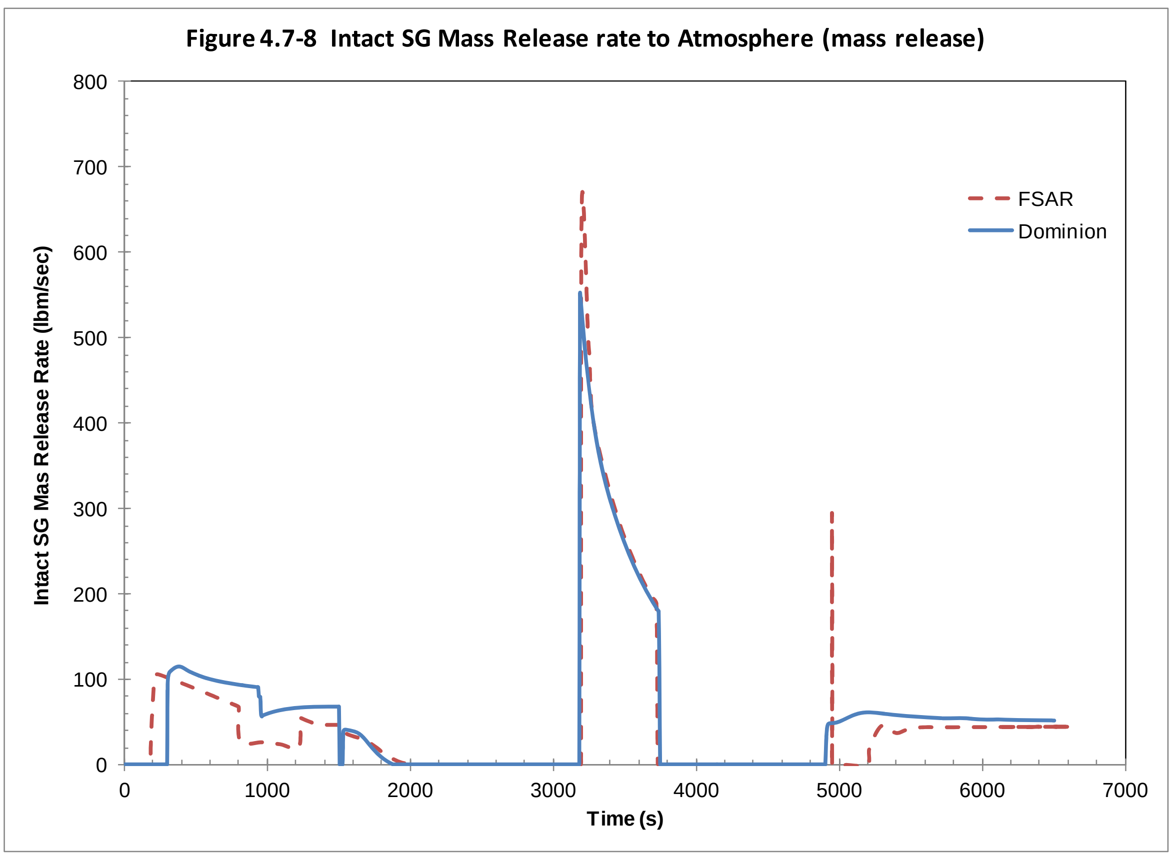


Figure 4.7-8 Intact SG Mass Release rate to Atmosphere (mass release)
800
700
600
500
400
300
200
100
0
Intact SG Mas Release Rate (lbm/sec)
0
1000
2000
3000
4000
5000
6000
7000
Time (s)
FSAR
Dominion

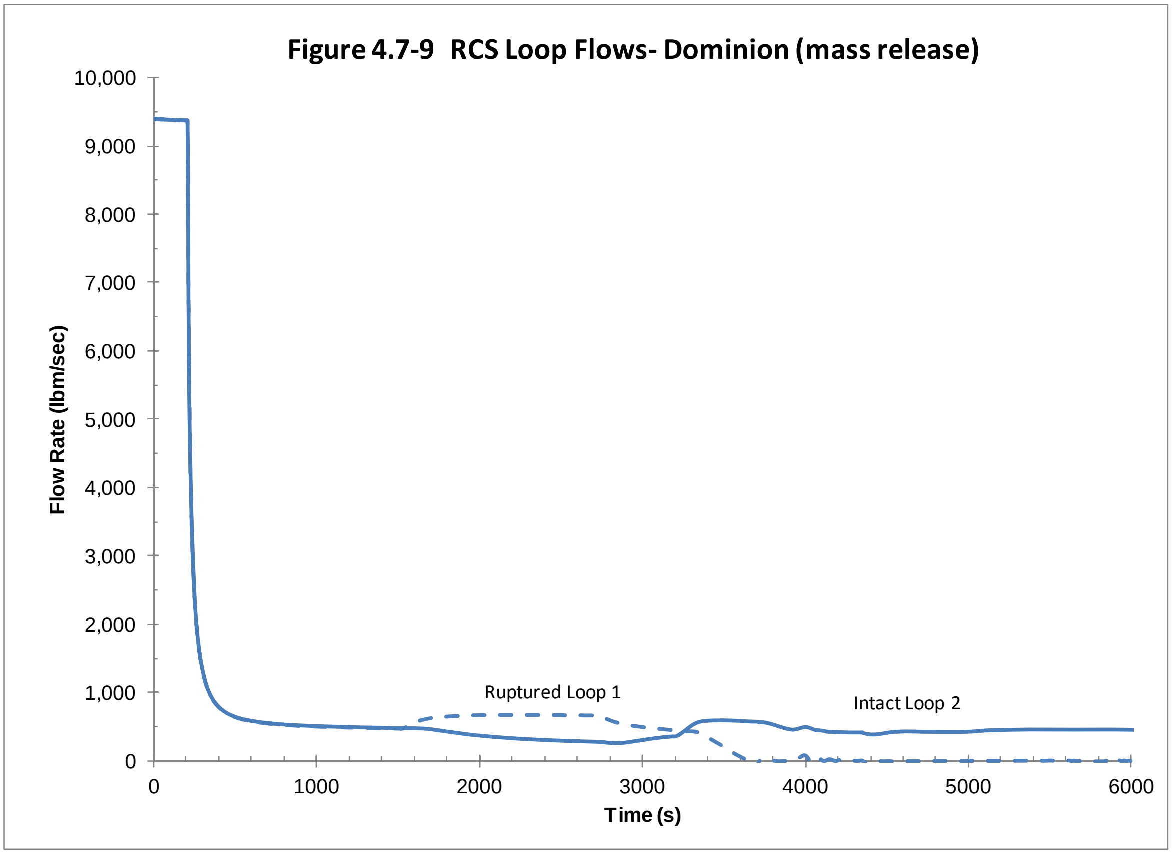


Figure 4.7-9 RCS Loop Flows- Dominion (mass release)
Flow Rate (lbm/sec)
10,000
9,000
8,000
7,000
6,000
5,000
4,000
3,000
2,000
1,000
0
Ruptured Loop 1
Intact Loop 2
0
1000
2000
3000
4000
5000
6000
Time (s)

Results – SGTR Overfill Case

The response for the SG Overfill case is shown on Figure 4.7-10 through Figure 4.7-17 and the sequence of events is provided in Table 4.7-3. In general, the overfill case trends are similar to the Mass Release case except that the ADV on the ruptured SG is not assumed to fail open when the main steam lines are isolated. In addition, the RCS cooldown phase takes longer since only one valve is available to perform that function. The FSAR contains no plots for the SG Overfill case which could be used for comparison. Therefore, for this benchmark, comparisons are based on the SGTR analysis presented in the Stretch Power Uprate (SPU) licensing report (Attachment 5 of Reference 1).

The pressurizer pressure response is shown on Figure 4.7-10. The Dominion pressurizer pressure tracks closely with the SPU data through most of the event. After SI is isolated near the end of the event, the Dominion pressure is less than the SPU pressure and remains below for the duration of the event, which is consistent with lower SG pressure (Figure 4.7-12) and the pressurizer pressure results provided for the Mass Release case above. The higher SPU pressurizer pressure when SI is isolated is also consistent with the higher SPU pressurizer fluid insurge prior to that period shown on Figure 4.7-11 as discussed in more detail below.

The SG pressure response for the ruptured and intact SGs is shown on Figure 4.7-12. As shown, the Dominion and SPU trends (dashed lines) are in good agreement for the intact SGs. For the ruptured SGs, the Dominion and SPU pressures agree well until the heat transfer is reduced due to the loss of appreciable natural circulation flow around 2600 seconds as shown by the RCS flows on Figure 4.7-17. As discussed for the Mass Release case, this is the result of the reduced ruptured loop temperatures following the RCS cooldown that limit natural circulation flow and yield reverse heat transfer from the ruptured SG secondary into the RCS. After this time the Dominion pressure is no longer maintained at the ADV relief valve setpoint and begins to slowly decrease.

The primary side temperature response is shown on Figure 4.7-13 for the intact SGs and Figure 4.7-14 for the ruptured SGs. As shown on Figure 4.7-13, the Dominion and SPU results for the intact SG temperatures are in very good agreement. For the ruptured SGs, there is good agreement between the Dominion and SPU cases until about 2600 seconds when natural circulation flow is lost in the ruptured RCS loop and the cold leg temperatures are more strongly affected by the cooler SI flow as discussed above for the Mass Release case. After SI flow is terminated, the Dominion cold leg temperature trends toward the SPU value.

The break flow rate through the ruptured SG tube is shown on Figure 4.7-15. There is very good agreement between the Dominion and SPU cases until the period late in the transient after SI has been isolated and the break flow is trending towards zero. This is also seen for the ruptured SG liquid volume response shown on Figure 4.7-16 where the Dominion and SPU responses agree well although the Dominion value stabilizes at a somewhat lower value near the end of the transient. Although there is not enough information available to determine the exact cause of this difference, there are several factors that could influence the final SG fluid volume. First, any difference in the assumed decay heat profile results in a different amount of fluid boiled from the SG secondary and associated liquid volume. Second, any differences in the integrated SI fluid injection affect the RCS fluid inventory available for release to the ruptured SG. It is noted that during the RCS depressurization phase which occurs just prior to SI isolation, SI flow rates increase dramatically due to flow from the intermediate head SI pumps and the FSAR case shows a greater increase in pressurizer level during this time. On the secondary side, differences in the integrated AFW flow rates affect the fluid delivered to the ruptured SG fluid volume as well as the energy removed by the intact SGs. Similarly, differences in SG relief valve flow rates affect mass and energy removal from the system. Lastly, it should be noted that any differences in the Dominion and SPU model noding and related assumptions could affect the differential pressure between the respective fluid levels in the RCS and SG secondary, which would also affect the final equilibrium level and associated fluid volume. Nevertheless, there is good overall agreement between the Dominion and SPU results.

Table 4.7-3 SGTR – Overfill Case Sequence of Event

Event	Time (seconds)	
	Dominion	**SPU**
SG Tube Ruptured	0	0
Reactor Trip (OTDT)	206	135
SI Actuated	216	145
AFW Flow Initiated	236	165
Ruptured SG AFW Isolated	855	794
Ruptured SG Steamline Isolated	1500	1500
RCS Cooldown Initiated	1980	1980
RCS Cooldown Terminated	2830	2850
RCS Depressurization Initiated	3010	3030
RCS Depressurization Terminated	3094	3124
SI Terminated	3454	3484
Break Flow Terminated	4535	5082

Figure 4.7-10 Pressurizer Pressure (overfill)

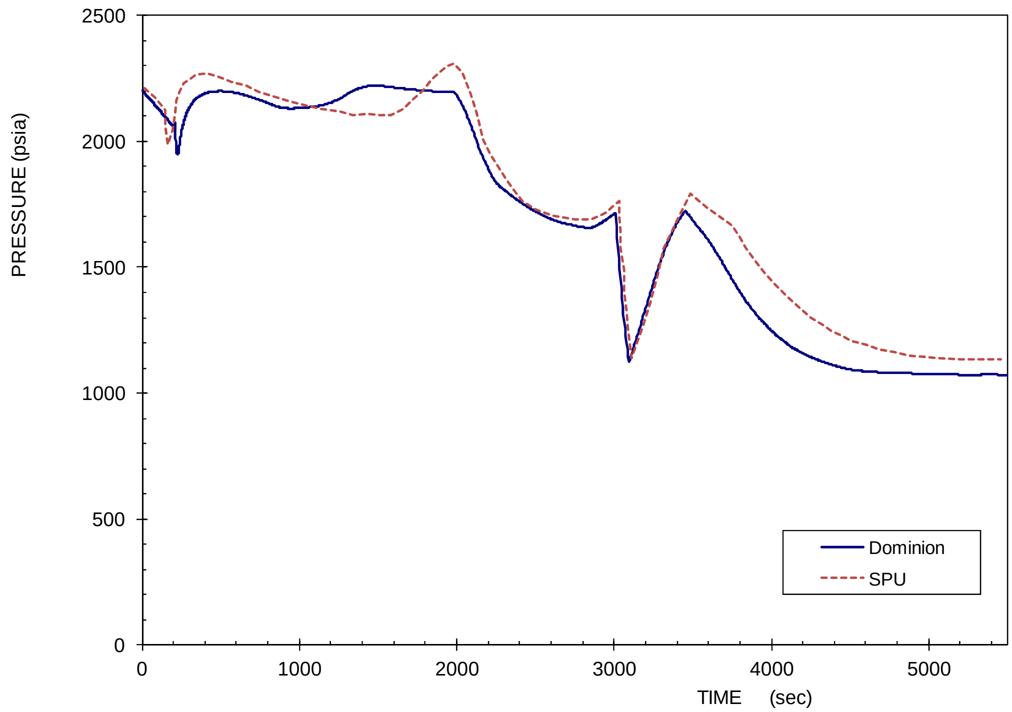
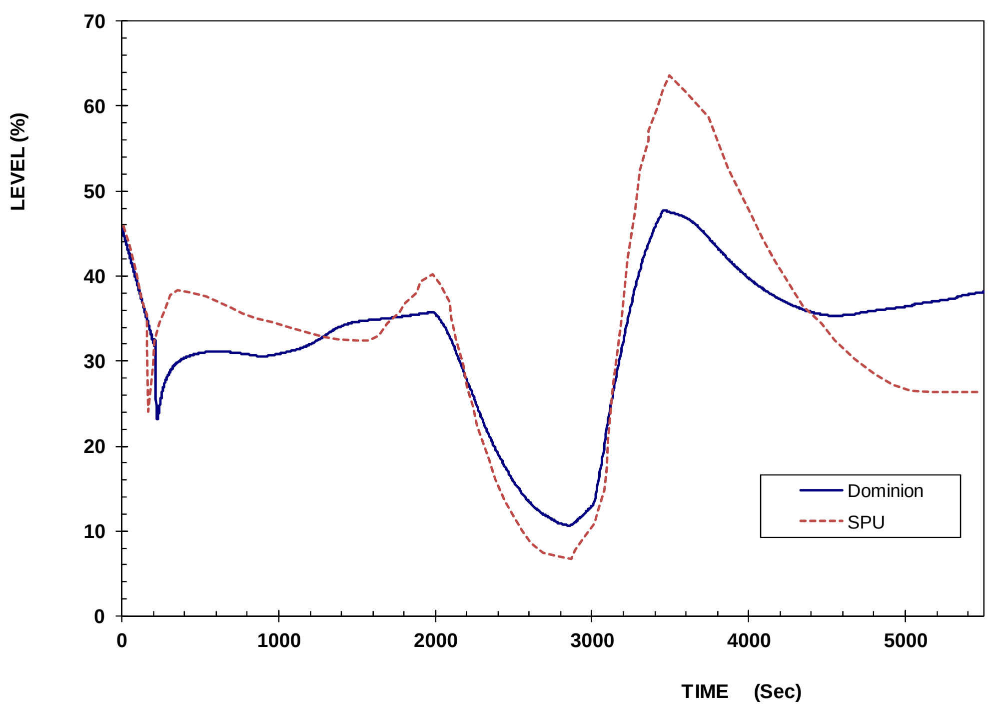


Figure 4.7-11 Pressurizer Level (overfill)



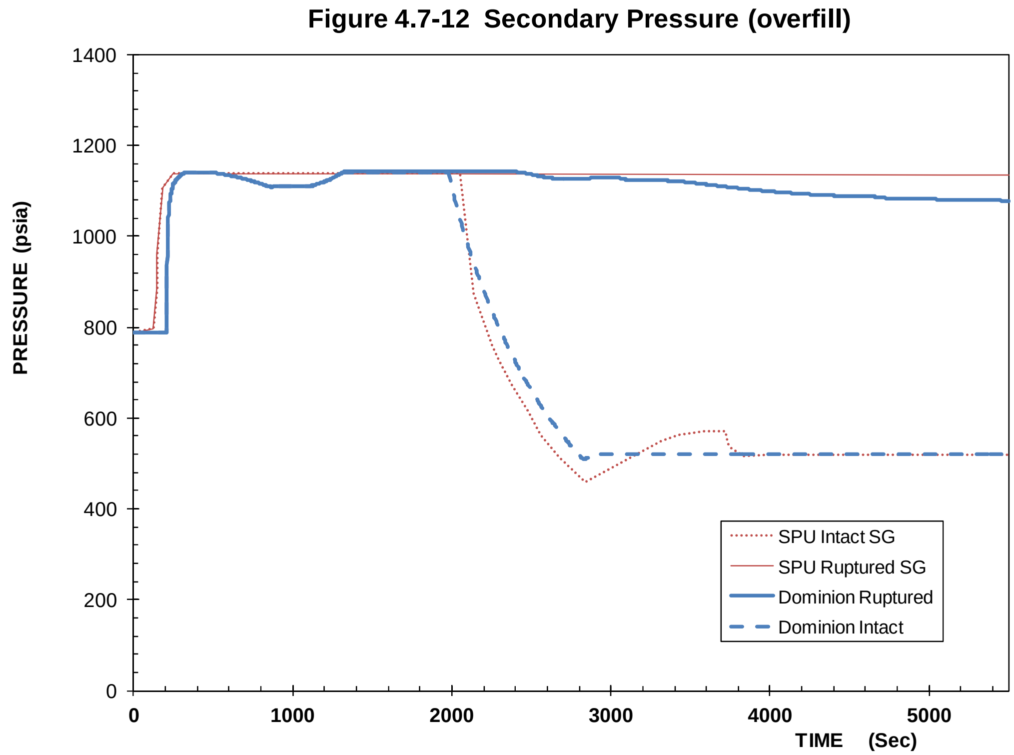


Figure 4.7-12 Secondary Pressure (overfill)
PRESSURE (psia)
1400
1200
1000
800
600
400
200
0
0
1000
2000
3000
4000
5000
TIME (Sec)
SPU Intact SG
SPU Ruptured SG
Dominion Ruptured
Dominion Intact

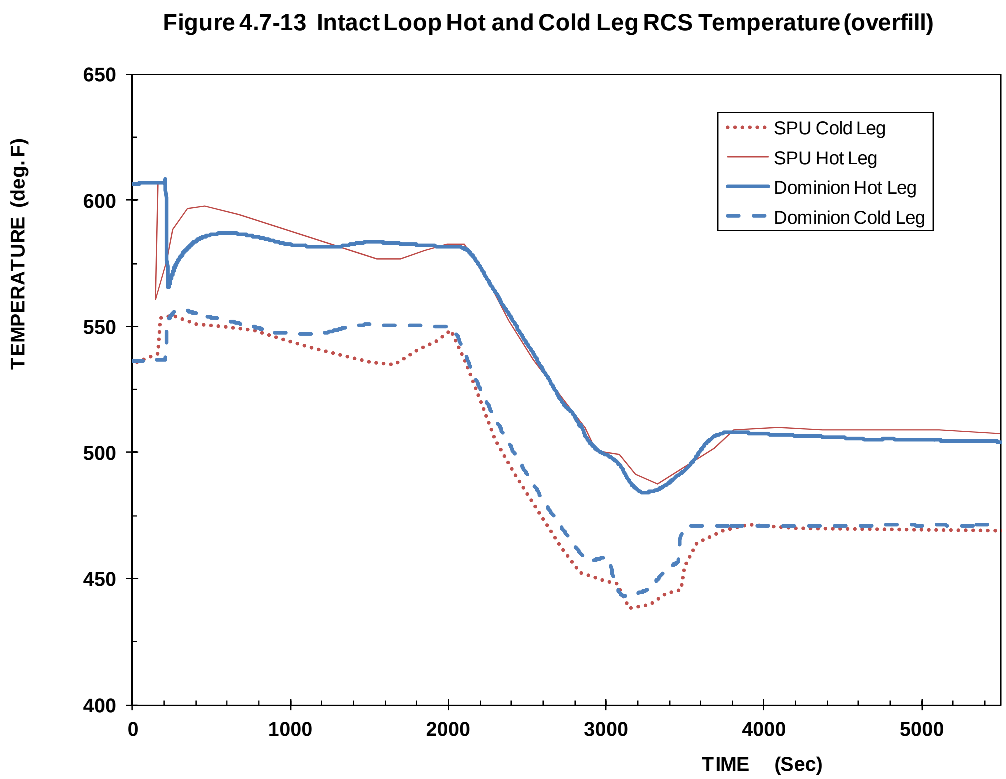


Figure 4.7-13 Intact Loop Hot and Cold Leg RCS Temperature (overfill)
TEMPERATURE (deg.F)
650
600
550
500
450
400
0
1000
2000
3000
4000
5000
TIME (Sec)
SPU Cold Leg
SPU Hot Leg
Dominion Hot Leg
Dominion Cold Leg

Figure 4.7-14 Ruptured Loop Hot and Cold Leg RCS Temperature (overfill)

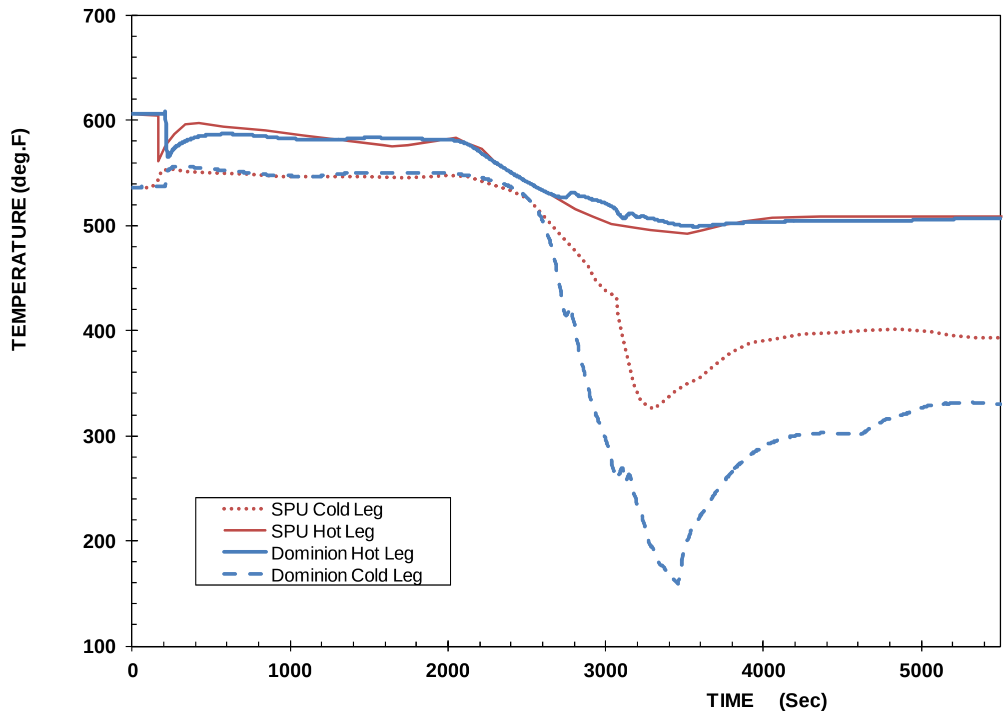


Figure 4.7-15 Primary to Secondary Break Flow Rate (overfill)

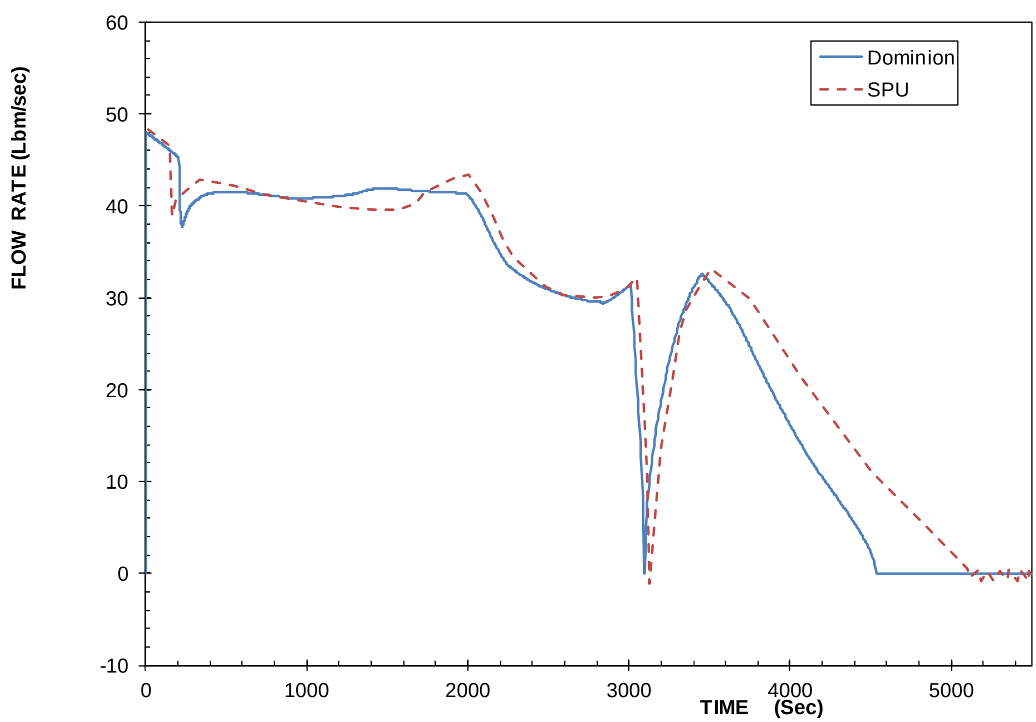


Figure 4.7-16 Ruptured SG Liquid Volume (overfill)

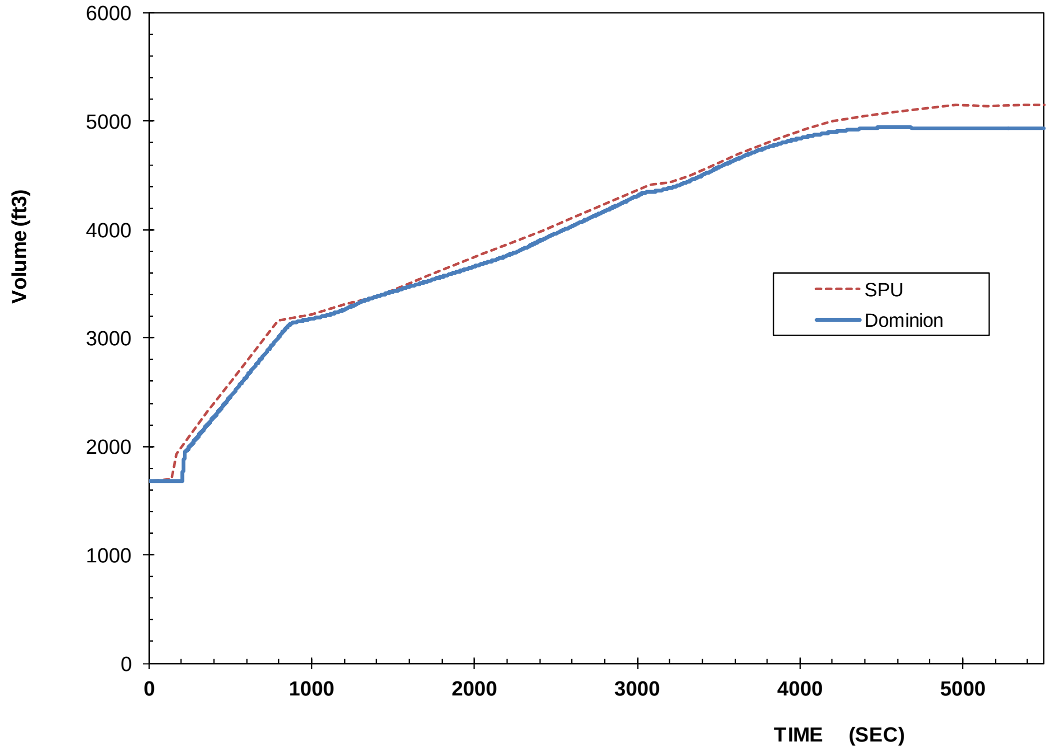
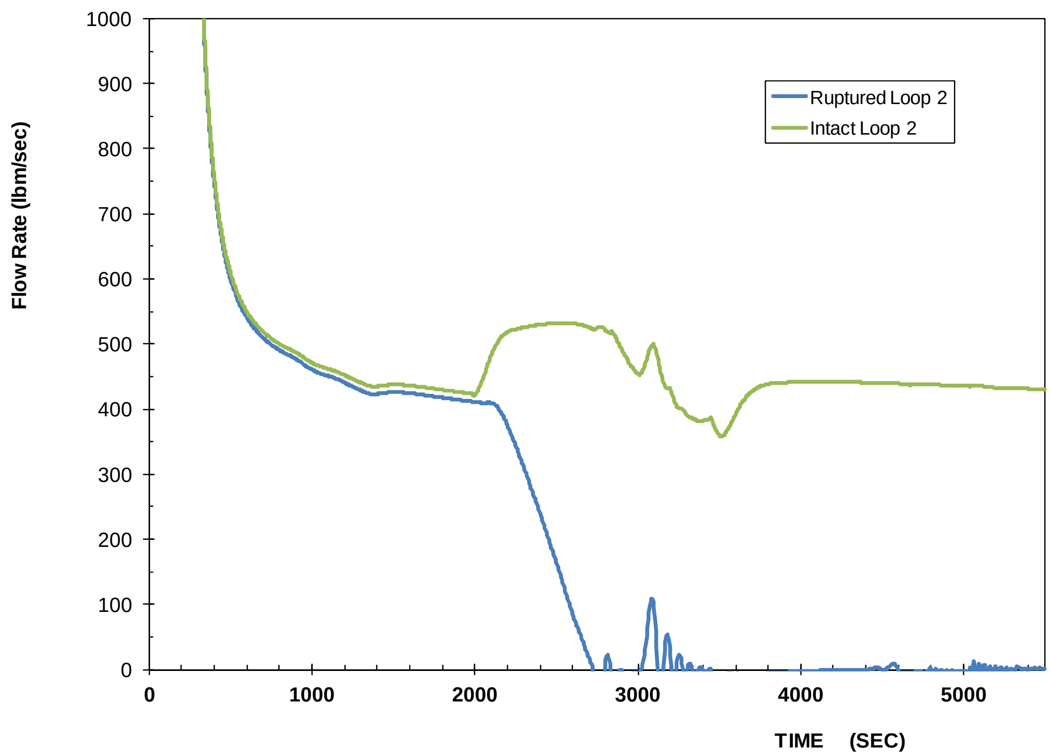


Figure 4.7-17 RCS Loop Flows - Dominion (overfill)



Summary - SGTR

The Dominion Millstone model provides results that are similar to the FSAR and SPU analyses for the SGTR event. Two cases are analyzed, a thermal-hydraulic case to determine mass releases to the atmosphere for radiological dose, and a second case to ensure that SG overfill does not occur. There is overall good agreement in the parameters for both cases although some differences occurring near the end of the events have been noted with an explanation provided.

References

1. DNC Letter 07-0450, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3 License Amendment Request Stretch Power Uprate," July 13, 2007. (ADAMS Accession No. ML072000386)

5.0 Conclusions

This attachment presents benchmarking transient analyses performed with the MPS3 RETRAN model developed in accordance with VEP-FRD-41-P-A. These analysis results are compared with current Millstone FSAR results. The following conclusions are drawn based on these analyses.

1) It is demonstrated that the Dominion RETRAN-3D model and analysis methods can predict the response of transient events with results that compare well to FSAR results.

2) Where there are differences between the Dominion results and the FSAR results, they are understood based on differences in noding, inputs, or other modeling assumptions.

3) The Dominion Millstone RETRAN-3D model is consistent with current Dominion methods (Reference 1). These methods have been applied extensively for Surry and North Anna licensing, engineering and plant support analyses.

4) The RETRAN comparison analyses satisfy the applicability assessment criteria and provide further validation of the conclusion that Dominion's RETRAN analysis methods are applicable to Millstone and can be applied to Millstone licensing analysis for reload core design and safety analysis.

6.0 References

1) Topical Report, VEP-FRD-41-P-A, Rev. 0.2, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code," March 2015.

2) Topical Report, VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient," December 1984.

APPENDIX 11
MPS3 Methods Transition SER

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 28, 2016

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT ADOPTING DOMINION CORE DESIGN AND SAFETY ANALYSIS METHODS AND ADDRESSING THE ISSUES IDENTIFIED IN THREE WESTINGHOUSE COMMUNICATION DOCUMENTS (CAC NO. MF6251)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 268 to Renewed Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3. This amendment is in response to your application dated May 8, 2015, as supplemented by letters dated January 28, February 25, March 23, March 29, and May 2, 2016.

The amendment revises the Technical Specifications (TSs) to (1) allow the use of Dominion nuclear safety and reload core design methods; (2) allow the use of applicable departure from nucleate boiling ratio design limits for VIPRE-D; (3) update the approved reference methodologies cited in TS 6.9.1.6.b; (4) remove the base load mode of operation that is not a feature of the Dominion Relaxed Power Distribution Control power distribution control methodology; and (5) address the issues identified in Westinghouse Nuclear Safety Advisory Letter (NSAL-09-5), Rev. 1, NSAL-15-1, and Westinghouse Communication 06-IC-03. Additionally, the amendment relocates certain equations, supporting descriptions and surveillance requirements from the TSs to licensee-controlled documents.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

RGuzman

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures:
1. Amendment No. 268 to NPF-49
2. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 268
Renewed License No. NPF-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Dominion Nuclear Connecticut, Inc. (DNC) dated May 8, 2015, as supplemented by letters dated January 28, February 25, March 23, March 29, and May 2, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-49 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, revised through Amendment No. 268 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DNC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate

Travis L. Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License
and Technical Specifications

Date of Issuance: July 28, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 268

RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
4	4

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
1-7	1-7
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-11	3/4 2-11
3/4 2-19	3/4 2-19
3/4 2-20	3/4 2-20
6-19a	6-19a
6-20	6-20
6-20a	6-20a
---	6-20b

(2) Technical Specifications

The Technical Specifications contained in Appendix A, revised through Amendment No. 268 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DNC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) DNC shall not take any action that would cause Dominion Resources, Inc. (DRI) or its parent companies to void, cancel, or diminish DNC's Commitment to have sufficient funds available to fund an extended plant shutdown as represented in the application for approval of the transfer of the licenses for MPS Unit No. 3.

(4) Immediately after the transfer of interests in MPS Unit No. 3 to DNC, the amount in the decommissioning trust fund for MPS Unit No. 3 must, with respect to the interest in MPS Unit No. 3, that DNC would then hold, be at a level no less than the formula amount under 10 CFR 50.75.

(5) The decommissioning trust agreement for MPS Unit No. 3 at the time the transfer of the unit to DNC is effected and thereafter is subject to the following:

(a) The decommissioning trust agreement must be in a form acceptable to the NRC.

(b) With respect to the decommissioning trust fund, investments in the securities or other obligations of Dominion Resources, Inc. or its affiliates or subsidiaries, successors, or assigns are prohibited. Except for investments tied to market indexes or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

(c) The decommissioning trust agreement for MPS Unit No. 3 must provide that no disbursements or payments from the trust, other than for ordinary administrative expenses, shall be made by the trustee until the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.

(d) The decommissioning trust agreement must provide that the agreement cannot be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

VENTING

1.39 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

SPENT FUEL POOL STORAGE PATTERNS:

STORAGE PATTERN

1.40 STORAGE PATTERN refers to the blocked location in a Region 1 fuel storage rack and all adjacent and diagonal Region 1 (or Region 2) cell locations surrounding the blocked location. The blocked location is for criticality control.

3-OUT-OF-4 AND 4-OUT-OF-4

1.41 Region 1 spent fuel racks can store fuel in either of 2 ways:

(a) Areas of the Region 1 spent fuel racks with fuel allowed in every storage location are referred to as the 4-OUT-OF-4 Region 1 storage area.

(b) Areas of the Region 1 spent fuel racks which contain a cell blocking device in every 4th location for criticality control, are referred to as the 3-OUT-OF-4 Region 1 storage area. A STORAGE PATTERN is a subset of the 3-OUT-OF-4 Region 1 storage area.

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

1.43 Deleted

1.44 Deleted

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. The limits specified in the CORE OPERATING LIMITS REPORT (COLR)

b. Deleted

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the applicable limits specified in the COLR,

1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux--High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Deleted

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

* See Special Test Exception 3.10.2

SURVEILLANCE REQUIREMENTS

4.2.1.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at the frequency specified in the Surveillance Frequency Control Program when the AFD Monitor Alarm is OPERABLE:

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.1.3 Deleted

4.2.1.1.4 Deleted

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2.1 $F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. With Specification 4.2.2.1.2.b not being satisfied:

(1) Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower DT Trip setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit, and

(2) Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by item (1) above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits.

b. With Specification 4.2.2.1.2.c not being satisfied, all of the following ACTIONS shall be taken:

(1) a. Within 4 hours, control the AFD to within the new reduced AFD limits specified in the COLR that restores $F_Q(Z)$ to within its limits, and

b. Reduce the THERMAL POWER by the amount specified in the COLR that restores $F_Q(Z)$ to within its limits within 4 hours, and

c. Reduce the Power Range Neutron Flux - High Trip Setpoints by $\geq$1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

d. Reduce the Overpower ΔT Trip Setpoints by $\geq$1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

e. Within 8 hours, reset the AFD Alarm Setpoints to the modified limits, and

f. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION b(1)b above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits.

(2) Deleted

c. Deleted

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SURVEILLANCE REQUIREMENTS

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Evaluate the computed heat flux hot channel factor by performing both of the following:

(1) Determine the computed heat flux hot channel Factor, $F_Q^M(Z)$ by increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties, and

(2) Verify that $F_Q^M(Z)$ satisfies the requirements of Specification 3.2.2.1 for all core plane regions, i.e., 0-100% inclusive.

SURVEILLANCE REQUIREMENTS (Continued)

c. Verify $F_Q^M(Z)$ satisfies the non-equilibrium limits specified in the COLR.

d. Measuring $F_Q^M(Z)$ according to the following schedule:

(1) Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined,*** or

(2) At the frequency specified in the Surveillance Frequency Control Program, whichever occurs first.

e. Compliance with the non-equilibrium limits shall be conservatively accounted for during intervals between $F_Q^M(Z)$ measurements by performing either of the following:

(1) Increase $F_Q^M(Z)$ by an appropriate factor specified in the COLR and verify that this value satisfies Specification 4.2.2.1.2.c, or

(2) Verify $F_Q^M(Z)$ satisfies its limits at least once per 7 Effective Full Power Days.

f. The limits specified in Specifications 4.2.2.1.2c and 4.2.2.1.2e above are not applicable in the core plane regions defined in the Bases.

4.2.2.1.3 Deleted

4.2.2.1.4 Deleted

4.2.2.1.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specifications 4.2.2.1.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

***During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and power distribution map outlined.

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3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate ≥ 363,200 gpm and greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR), and

b. $F^N_{\Delta H} \leq F^{RTP}_{\Delta H} \; [1.0 + PF_{\Delta H}(1.0 - P)]$

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement,

3) $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER in the COLR,

4) $PF_{\Delta H}$ = The power factor multiplier for $F^N_{\Delta H}$ provided in the COLR, and

5) The measured value of RCS total flow rate shall be used since uncertainties for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate to within the limits specified above and in the COLR and $F^N_{\Delta H}$ to within the above limit, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that the RCS total flow rate is restored to within the limits specified above and in the COLR and $F_{\Delta H}^{N}$ is restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}^{N}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.1.2 $F_{\Delta H}^{N}$ shall be determined to be within the acceptable range:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At the frequency specified in the Surveillance Frequency Control Program.

4.2.3.1.3 The RCS total flow rate shall be determined to be within the acceptable range by:

a. Verifying by precision heat balance that the RCS total flow rate is ≥ 363,200 gpm and greater than or equal to the limit specified in the COLR within 7 days after reaching 90% of RATED THERMAL POWER after each fuel loading, and

MONTHLY OPERATING REPORTS

6.9.1.5 Deleted

CORE OPERATING LIMITS REPORT

6.9.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Reactor Core Safety Limit for Specification 2.1.1.
2. Overtemperature ΔT and Overpower ΔT setpoint parameters for Specification 2.2.1.
3. SHUTDOWN MARGIN for Specifications 3/4.1.1.1.1, 3/4.1.1.1.2, and 3/4.1.1.2.
4. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.
5. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5.
6. Control Rod Insertion Limits for Specification 3/4.1.3.6.
7. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1.1.
8. Heat Flux Hot Channel Factor Limits for Specification 3/4.2.2.1.
9. RCS Total Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3.1.
10. DNB Parameters for Specification 3/4.2.5.
11. Shutdown Margin Monitor minimum count rate for Specification 3/4.3.5.
12. Boron Concentration for Specification 3/4.9.1.1.

CORE OPERATING LIMITS REPORT (Cont.)

6.9.1.6.b The analytical methods used to determine the core operating limits in Specification 6.9.1.6.a shall be those previously reviewed and approved by the NRC and identified below. The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number, title, revision, date, and any supplements).

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," (W Proprietary). Methodology for Specifications:
   - 2.1.1 Reactor Core Safety Limits
   - 3.1.1.1.1 SHUTDOWN MARGIN – MODE 1 and 2
   - 3.1.1.1.2 SHUTDOWN MARGIN – MODES 3, 4 and 5 Loops Filled
   - 3.1.1.2 SHUTDOWN MARGIN – Cold Shutdown – Loops Not Filled
   - 3.1.1.3 Moderator Temperature Coefficient
   - 3.1.3.5 Shutdown Rod Insertion Limit
   - 3.1.3.6 Control Rod Insertion Limits
   - 3.2.1.1 AXIAL FLUX DIFFERENCE
   - 3.2.2.1 Heat Flux Hot Channel Factor
   - 3.2.3.1 RCS Total Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
   - 3.9.1.1 REFUELING Boron Concentration
   - 3.2.5 DNB Parameters
   - 3.3.5 Shutdown Margin Monitor
2. Deleted
3. Deleted
4. WCAP-10216-P-A-R1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," (W Proprietary). (Methodology for Specifications 3.2.1.1--AXIAL FLUX DIFFERENCE and 3.2.2.1--Heat Flux Hot Channel Factor)
5. WCAP-12945-P-A, "CODE QUALIFICATION DOCUMENT FOR BEST ESTIMATE LOCA ANALYSIS," (W Proprietary). (Methodology for Specification 3.2.2.1--Heat Flux Hot Channel Factor.)
6. WCAP-16009-P-A, "REALISTIC LARGE-BREAK LOCA EVALUATION METHODOLOGY USING THE AUTOMATED STATISTICAL TREATMENT OF UNCERTAINTY METHOD (ASTRUM)," (W Proprietary). (Methodology for Specification 3.2.2.1--Heat Flux Hot Channel Factor.)

CORE OPERATING LIMITS REPORT (Cont.)

7. WCAP-11946, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Millstone Nuclear Power Station Unit 3," (W Proprietary). Methodology for Specification:
   - 3.1.1.3 - Moderator Temperature Coefficient
8. WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)
9. WCAP-10079-P-A, "NOTRUMP - A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)
10. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)
11. Deleted
12. Deleted
13. Deleted
14. Deleted
15. Deleted
16. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis." Methodology for Specification:
   - 3.2.2.1 - Heat Flux Hot Channel Factor
17. WCAP-10054-P-A, Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model." Methodology for Specification:
   - 3.2.2.1 - Heat Flux Hot Channel Factor
18. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature DT Trip Functions," (Westinghouse Proprietary Class 2). (Methodology for Specifications 2.2.1 -- Overtemperature ΔT and Overpower ΔT Setpoints.)

CORE OPERATING LIMITS REPORT (Cont.)

19. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

20. VEP-FRD-42-A, "Reload Nuclear Design Methodology." Methodology for Specifications:

    - 2.1.1 Reactor Core Safety Limits
    - 3.1.1.1.1 SHUTDOWN MARGIN – MODE 1 and 2
    - 3.1.1.1.2 SHUTDOWN MARGIN – MODES 3, 4 and 5 Loops Filled
    - 3.1.1.2 SHUTDOWN MARGIN – Cold Shutdown – Loops Not Filled
    - 3.1.1.3 Moderator Temperature Coefficient
    - 3.1.3.5 Shutdown Rod Insertion Limit
    - 3.1.3.6 Control Rod Insertion Limits
    - 3.2.2.1 Heat Flux Hot Channel Factor
    - 3.2.3.1 Nuclear Enthalpy Rise Hot Channel Factor
    - 3.3.5 Shutdown Margin Monitor
    - 3.9.1.1 REFUELING Boron Concentration

21. VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications." Methodology for Specifications:

    - 3.2.1.1 AXIAL FLUX DIFFERENCE
    - 3.2.2.1 Heat Flux Hot Channel Factor

22. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology." Methodology for Specifications:

    - 3.2.3.1 RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
    - 3.2.5 DNB Parameters

23. DOM-NAF-2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code." Methodology for Specifications:

    - 3.2.3.1 RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
    - 3.2.5 DNB Parameters

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 268

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

1.0 INTRODUCTION

By application dated May 8, 2015 (Reference 1), as supplemented on January 28, 2016 (Reference 2), February 25, 2016 (Reference 3), March 23, 2016 (Reference 4), March 29, 2016 (Reference 5), and May 2, 2016 (Reference 6), Dominion Nuclear Connecticut, Inc. (DNC, the licensee) submitted a license amendment request (LAR) to revise the technical specifications (TSs) for Millstone Power Station, Unit 3 (MPS3) that would allow use of Dominion nuclear safety and reload core design methods and address the issues identified in three Westinghouse communication documents. Specifically, the proposed TS changes would: (1) allow the use of Dominion nuclear safety and reload core design methods; (2) allow the use of applicable departure from nucleate boiling ratio (DNBR) design limits for VIPRE-D; (3) update the approved reference methodologies cited in TS 6.9.1.6.b; (4) remove the base load mode of operation that is not a feature of the Dominion Relaxed Power Distribution Control (RPDC) power distribution control methodology; and (5) address the issues identified in Westinghouse Nuclear Safety Advisory Letter (NSAL-09-5), Rev. 1, NSAL-15-1, and Westinghouse Communication 06-IC-03. Additionally, the proposed changes would involve, in part, the relocation of certain equations, supporting descriptions and surveillance requirements from the TSs to licensee-controlled documents.

The Dominion reload methods documented in the following topical reports (TRs) were previously approved by the U.S. Nuclear Regulatory Commission (NRC) for use in the reload analysis and licensing applications for Dominion nuclear plants including North Anna Power Station (NAPS), Surry Power Station (SPS) and Kewaunee Power Station (KPS).

- VEP-FRD-42-A, Reload Nuclear Design Methodology (Reference 7)
- VEP-NE-1-A, Relaxed Power Distribution Control Methodology (Reference 8)
- DOM-NAF-1-P-A, Core Management System (CMS) Reactor Physics Methods (Reference 9)
- VEP-FRD-41-P-A, RETRAN NSSS Non-LOCA Analysis (Reference 10)

Enclosure 2

- VEP-NE-2-A, Statistical DNBR Evaluation Methodology (Reference 11)
- DOM-NAF-2-P-A, Core Thermal-Hydraulics Using VIPRE-D (Reference 12)

The supplements dated January 28, February 25, March 23, March 29, and May 2, 2016 provided additional information that clarified the application, did not expand the scope of the application, and did not change the NRC staff's original proposed no significant hazards consideration determination as originally noticed in the *Federal Register* (FR), 80 FR 52804 on September 1, 2015. A subsequent notice was published in the FR on June 13, 2016 (81 FR 38226), to include the added clarification that the proposed amendment involves the relocation of TS information either to the TS Bases or the Core Operating Limits Report (COLR) which are both licensee-controlled documents. There were no changes to the no significant hazards consideration determination as originally noticed.

2.0 REGULATORY EVALUATION

2.1 Applicable Regulatory Requirements

The NRC used the following requirements and guidance documents in evaluating the licensee's amendment request:

In Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.34, "Contents of application; technical information," the NRC established its regulatory requirements that safety analysis reports 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, licensees confirm that the inputs to the safety analyses are conservative with respect to the current design cycle. These inputs are checked using analytical models; and if key safety analysis parameters are not bounded, further analysis of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

In 10 CFR 50.36, the NRC established its regulatory requirements related to the content of TSs. The regulation at 10 CFR 50.36(a)(1) states that a summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the TSs. Paragraph 10 CFR 50.36(b) requires that each license authorizing the operation of a facility will include TSs and will be derived from the analyses and evaluation included in the safety analysis report. The categories of items required to be in the TSs are provided 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(2)(i), the TSs will include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Per 10 CFR 50.36(c)(2)(i), when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. Paragraph 10 CFR 50.36(c)(2)(ii)(B) Criterion 2 requires that an LCO be established for: "A process variable design feature or operating restriction that is an initial condition or a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

Paragraph 10 CFR 50.36(c)(3) requires TSs to include items in the category of surveillance requirements, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

Paragraph 10 CFR 50.36(c)(5), "Administrative controls," are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

In addition to the above regulatory requirements, the following guidance documents were considered during this review:

- NUREG-0800, Standard Review Plan, Section 16, Revision 3.0, "Technical Specifications" (Reference 13)
- Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications" (Reference 14)
- NUREG-1431, Rev. 4.0, "Standard Technical Specifications for Westinghouse Plants" (Reference 15)

2.2 Background

MPS3 is a four loop pressurized water reactor (PWR) of Westinghouse design with a subatmospheric reactor containment. As part of the design basis of the plant, thermal and hydraulic characteristics are incorporated in the core design. Therefore, when it is operated with consideration for mechanical and thermal limits, in combination with plant equipment characteristics, instrumentation, and the reactor protection system, no fuel damage will occur during normal operation or abnormal operating transients.

DNC proposed changes to the power distribution limit TSs. The purpose of the power distribution limit TSs (MPS3 TS Section 3/4.2) is explained by the following excerpt from the MPS3 TS bases document:

> The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR [departure from nucleate boiling ratio] in the core greater than or equal to the design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.
>
> The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods; and

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

The use of Axial Flux Difference limits (TS 3/4.2.1) is explained by the following TS bases excerpt:

> The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.
>
> Target flux difference is determined at equilibrium xenon conditions. The full-length [control] rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Dominion methods involve the use of RPDC versus the relaxed axial offset control (RAOC) or constant axial offset control (CAOC) axial power distribution methodologies more frequently used at Westinghouse PWRs for establishment of operating power distribution limits. The Dominion method involves establishment of a variable axial flux difference (delta-I) band power distribution control strategy. As power decreases the allowed delta-I band increases "... maintaining an approximately constant analysis margin to the design bases limits at all power levels."

Currently MPS3 employs a method of operation when at power levels below the nuclear design allowed power level (APL^{ND}) the limits on AFD are defined in the COLR consistent with RAOC. At power levels greater than APL^{ND} 2 modes of operation are allowed: (1) RAOC with the AFD limits defined in the COLR; or (2) base load operation which is defined as the maintenance of the AFD within the COLR specifications band about a target value.

The Dominion power distribution control strategy uses a variable AFD delta-I band. The delta-I band is a calculated analysis output. The objective of the RPDC analysis is to determine acceptable delta-I bands that maintain margin to all the applicable design basis criteria during normal operation, abnormal operating occurrences or analyzed accidents especially a LOCA or loss of flow accident. The calculated delta-I bands will change depending on the specific core loading pattern for the cycle and core burnup; therefore, they will be located in the COLR.

In the LAR, DNC states that these changes accomplish three key objectives:

- Accommodate the implementation of the Dominion RPDC,
- Removal of base load operation, and
- Provide resolution of issues documented in Westinghouse notification documents NSAL-09-05, Rev. 1, 06-IC-03, and NSAL -15-1.

The same power distribution control parameters of AFD, Heat Flux Hot Channel Factor, Reactor Coolant System (RCS) Flowrate and Nuclear Enthalpy Rise Hot Channel Factor are employed in TS for either strategy to protect the fuel. The licensee states that the proposed TS changes are structured in a manner that is independent of specific power distribution control methodology (RAOC or RPDC).

In the LAR, DNC has proposed the following changes to MPS3 TSs:

- Remove TS 1.43 definition of minimum allowable nuclear design power level for base load operation (APL^{ND});
- Remove TS 1.44 definition of maximum allowable power level when transitioning to base load operation, (APL^{BL});
- Change TS 3/4.2.1.1, AFD, to support adoption of Dominion's RPDC methods;
- Change TS 3/4.2.2.1, Heat Flux Hot Channel Factor, to support adoption of Dominion's RPDC methods;
- Change TS 3/4.2.3.1, RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, to remove a specific uncertainty and adjust the flow rate when the precision heat balance is done; and
- Change TS 6.9.1.6, COLR, to support adoption of Dominion's RPDC methods.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the licensee's LAR in combination with the licensee's response to the NRC's requests for additional information (RAIs) and the relevant NRC-approved Dominion TRs. The staff's evaluation of the LAR is discussed in Sections 3.1 and 3.2 below.

3.1 Dominion Core Reload and Safety Analysis Methodologies

MPS3 became part of the Dominion nuclear fleet following DNC's acquisition of Millstone Power Station in 2001. Currently, the fuel supplier (Westinghouse) performs the reload analysis for MPS3, whereas the licensee performs the reload analysis using the Dominion methods for other Dominion nuclear plants including NAPS, SPS, and KPS. In its LAR, the licensee proposed to apply the Dominion reload methods to MPS3 for the analysis of the non-LOCA transients and accidents to support MPS3 reload applications. In support of its proposal, the licensee provided justifications for the application of Dominion methods to MPS3 in Attachment 4 of its LAR, "Application of Dominion Nuclear Core Design and Safety Analysis Methods," (Reference 16), which includes bases for the use of the following methodologies: (1) the reload nuclear design methodology; (2) the RPDC methodology; (3) CMS reactor physics methodologies; (4) the methodology of the reactor system transient analyses using RETRAN; (5) the statistical DNBR evaluation methodology; and (6) the methodology of the reactor core thermal-hydraulics analysis using VIPRE-D computer code. All of the above methods are documented in the NRC-approved TRs for use in other Westinghouse-manufactured plants operated by Dominion, including NAPS, SPS, and KPS. Since MPS3 is also a Westinghouse-manufactured plant, the NRC staff's review discussed in the following Subsections 3.1.1 to 3.1.6 focuses on whether the licensee's proposed use of the Dominion safety analysis methodologies for MPS3 is in a manner complying with the conditions and limitations imposed by NRC safety evaluation reports (SERs) approving the relevant Dominion TRs.

3.1.1 Reload Nuclear Design Methodology (TR VEP-FRD-42, Revision 2.1-A)

The reload nuclear design methodology discussed in Dominion TR VEP-FRD-42, Revision 2.1-A, "Reload Nuclear Design Methodology" and Section 3.1 of Reference 16 consists of the following elements:

- Analytical models including CMS models, VEPCO RETRAN models, and core thermal-hydraulics VIPRE-D models;

- Analytical methods for core depletions, core reactivity parameters and coefficients, core reactivity control, safety analysis, and statistical DNB;

- Reload design process for the core loading pattern design & optimization and key parameter treatment in nuclear design analyses; and

- Reload safety evaluation process and nuclear design report.

The Dominion reload methodology is an iterative process that involves the determination of a core loading pattern that fulfills cycle energy requirements and the demonstration that the plant with the reload core satisfies the constraints of the plant design basis and safety analysis limits. The Dominion reload methodology and the current MPS3 reload methodology use the same method discussed in Westinghouse TR, WCAP-9272-A, "Westinghouse Reload Safety Evaluation" (Reference 17). The reload method uses a bounding analysis concept in which key analysis parameters with limiting directions are identified such that, if all key analysis parameters are conservatively bounded, a reference safety analysis is applicable and no further analysis is necessary. If any values are not bounded, further analysis of the transient or accident in question is performed, the applicable safety analyses are revised, or changes are made in the operating requirements specified in the TSs or COLR to satisfy applicable safety analysis criteria. The safety analysis process typically consists of steady state nuclear calculations used to derive the core physics related key analysis parameters as well as a dynamic accident analysis that utilizes these parameters to determine the accident result.

While MPS3 has differences in the nuclear steam supply system (NSSS), reactor protection system (RPS), and fuel features, these differences can be modeled using the existing methodology and analytical methods, namely VEP-FRD-42, Revision 2.1-A, with the appropriate selection of input variables. As indicated in Section 3.1.2 of Reference 16, VEP-FRD-42 SER limits the use of VEP-FRD-42, prohibiting its application to fuel types other than Westinghouse and Framatome ANP Advanced Mark-BW fuel. The restriction of the SER states that if the changes necessary to accommodate another fuel product require changes to the reload methodology of Dominion TR VEP-FRD-42-A, these proposed changes are required to be submitted for prior NRC review. The NRC staff finds that this SER restriction is met, since the MPS3 uses a Westinghouse fuel (robust fuel assembly with redesigned mid-grids fuel, RFA-2 which is the same as that of NAPS). As part of the implementation of Dominion methods, the licensee will verify the boration requirements for MPS3 on a reload basis using the same constituent equations utilized in TR WCAP-1441, which is currently used for MPS3 (as confirmed by the licensee in its response to NRC RAI-8 (SRXB) of Reference 2). Therefore, the NRC staff determines that the use of the Dominion reload methodology discussed in TR VEP-FRD-42, Revision 2.1-A and Section 3.1 of Reference 16 is acceptable to support licensing applications for MPS3.

3.1.2 Relaxed Power Distribution Control Methodology (TR VEP-NE-1, Rev. 0.1-A)

The RPDC methodology discussed in Dominion TR VEP-NE- 1, Rev. 0.1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications" and Section 3.2 of Reference 16 is a Dominion method for axial power distribution control with a delta-I band. This method provides an increasing delta-I band with decreasing power in order to maintain approximately constant analysis margin to the design bases limits at all power levels. The RPDC analysis process consists of: (1) the generation of power shapes that bound the delta-I range; (2) the selection of delta-I bands such that all bands satisfy the COLR height dependent heat flux hot channel factor, $F_Q(Z)$, limit with verification that the proposed delta-I bands satisfy LOCA FQ [total peaking factor] and loss of flow accident thermal-hydraulic evaluations; (3) the analysis of limiting Condition II events to ensure the power shapes within the final delta-I band are used as initial conditions; (4) the verification to confirm that over-power delta-temperature (OPΔT) and over-temperature delta-T (OTΔT) limits are conservative to ensure that margin to fuel design limits is maintained; and (5) the formulation of N(Z) functions [non-equilibrium power distribution multiplier] to support the implementation of FQ TS surveillance.

A number of similarities between the Dominion RPDC methods and the Westinghouse-RAOC methods currently used for MPS3 are shown in Table 3.2.1 of Reference 16. Section 3.2.3 of Reference 16 also indicates that the cooldown transient assumption of 30°F currently used for the Westinghouse method at MPS3 will be used unless a MPS3-specific analysis demonstrates that a plant trip will occur before reaching 30°F. The NRC SER approving Dominion TR VEP-NE-1, Rev. 0.1-A accepted the Dominion RPDC method for use at NAPS and SPS, and also allowed the RPDC method for use at plants with reload cores similar to those of NAPS and SPS. As previously discussed in this SE, MPS3, SPS, and NAPS are Westinghouse-manufactured plants, their NSSS, RPS, and fuel designs are similar such that its features are capable of being reflected via modeling inputs in the TR analytical methods without any changes to the methodology. Also, since both MPS3 and NAPS use Westinghouse RFA-2 fuel design (Robust Fuel Assembly with redesigned mid-grids), their reload cores are essentially identical. Therefore, the NRC staff finds that MPS3 satisfies the SER restriction, limiting the use of the TR to the reload cores similar to those of SPS and NAPS, and therefore, determines that the use of the RPDC method discussed in Dominion TR, VEP-NE-1, Rev 0.1-A, and Section 3.2 of Reference 16 is acceptable to support MPS3 licensing applications.

3.1.3 Core Management System Reactor Physics Methods (TR DOM-NAF-1-A)

The CMS methods discussed in Dominion TR DOM-NAF-1-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations," and Section 3.3 of Reference 16 involve two major computer codes, CASMO-4 and SIMULATE-3. CASMO-4 is a multi-group, two-dimensional transport theory code used for depletion and branch calculations for a single assembly. SIMULATE-3 is a two-group, three-dimensional diffusion theory code coupled with thermal-hydraulic and Doppler feedback.

The CMS methods model the core physics characteristics of the reload core including depletion/isotopic effects, reactivity, reactivity coefficients, power distribution, and shutdown margin. DNC uses the CMS methods in the analysis for RPDC, and licensing applications, including core reload design, core operation, and key core parameters for reload safety analyses.

The CMS benchmarking data provided in DOM-NAF-1-A is based on the 15x15 and 17x17 fuel designs used at SPS and NAPS, respectively, while MPS3 currently uses 17x17 fuel which is within the range of the CMS benchmarking data. In addition, the NRC SER approving TR DOM-NAF-1-A limits the TR use, prohibiting its application to "significantly different or new fuel designs." This restriction is met, since the current MPS3 fuel design bases on the Westinghouse RAF-2 fuel design, which is the same as that of NAPS. Therefore, the NRC staff determines that the use of the CMS methodology discussed in DOM-NAF-1-A and Section 3.3 of Reference 16 is acceptable to support the licensing applications for MPS3.

3.1.4 Reactor System Transient Analyses Using RETRAN (TR VEP-FRD-41-A, Rev. 0.2)

Dominion uses RETRAN discussed in TR VEP-FRD-41, Rev 0.2-A, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code," and Section 3.4 of Reference 16 to perform the analyses for non-LOCA events presented in the Final Safety Analysis Report (FSAR) for Dominion's plants with Westinghouse-manufactured reactors including NAPS, SPS and KPS. RETRAN calculates general system parameters as a function of time and boundary conditions for input into more detailed calculations of DNB or other thermal and fuel performance margins. The licensee performs analyses for non-LOCA events to confirm the adherence of reload core design limits to the bounds established by the reference analysis of record (AOR) parameter values, as well as to verify that the core is acceptable from a safety operational point of view.

3.1.4.1 MPS3 RETRAN Model

As discussed in Section 2.0 of Reference 19, the proposed nodal scheme of the RETRAN model for MPS3 is essentially identical to the NAPS and SPS models with the following differences:

1. The MPS3 model explicitly models the safety injection (SI) accumulators.
2. The MPS3 model has separate volumes for the steam generator (SG) inlet and outlet plenums.
3. The MPS3 model includes cooling paths between downcomer and upper head.
4. The MPS3 model includes a nodal scheme with a second parallel flow path through the active core from the lower plenum to the upper plenum for the analysis of the steam line break (SLB).

The Dominion RETRAN models also have some differences compared to the vendor RETRAN model that was used to perform the current FSAR Chapter 15 analyses. Table 2-1 of Reference 19 identifies the model differences, including differences in code versions, nodal schemes for the reactor vessel and steam generator, and the reactivity feedback models.

3.1.4.2 RETRAN Benchmarking Analysis

The Dominion MPS3 RETRAN models have been benchmarked by selecting representative non-LOCA design-basis events and comparing the results of the MPS3 RETRAN models to the vendor RETRAN model that was used to perform the current FSAR analyses. This approach is similar to that discussed in TR VEP-FRD-41-A. The results of the MPS3 RETRAN benchmark are provided in Reference 19. Subsequent to the submittal of the results of the bench analysis, the licensee identified a discrepancy between the MPS3 RETRAN base model pressurizer shell heat conductor and the Dominion RETRAN TR. The MPS3 RETRAN base input deck models the pressurizer shell as a heat conductor, which differs from TR VEP-FRD-41-A, which states that "Dominion continues to model the non-equilibrium wall as an adiabatic surface." This model is not used in any AOR, so this discrepancy has no impact on plant licensing or operations. However, the RETRAN base model was used to benchmark/replicate Westinghouse AOR in support of the subject LAR for generic NRC approval of the Dominion application for reload design analysis methods to MPS3. The licensee repeated each of the benchmarking cases supporting the LAR with the needed model correction and discussed the results in Reference 3. In response to the NRC staff's RAI, the licensee also analyzed two additional cases: the feedwater line break (FLB) and steam generator tube rupture (SGTR) events. As part of the RAI response, the licensee provided an update to the RETRAN benchmarking information in Reference 5. The NRC staff has reviewed the information of the RETRAN benchmarking analyses in References 3, 5, and 19, and discusses its evaluation for each benchmark case in Subsections 3.1.4.2.1 through 3.1.4.2.7 for MPS3 as follows:

3.1.4.2.1 Analysis of the Loss of Load/Turbine Trip Event

Section 4.1 of Attachment 2 to Reference 5 discusses the updated benchmark analysis for the loss of load/turbine trip (LOL) event. The event is initiated from a complete loss-of-steam flow and turbine trip from full-power conditions. The loss-of-steam flow results in a rapid increase in secondary system pressure and temperature, as well as a reduction of the heat transfer rate in the SGs, which, in turn, causes the reactor coolant system (RCS) pressure and temperature to rise. The licensee listed in Table 4.1-1 the initial plant conditions and the assumptions used in the LOL analysis and showed no differences in the key input and assumptions used in both the benchmark analysis and FSAR analysis. The results of the analysis is presented in Figure 4.1-1 to Figure 4.1-5. A comparison of the Dominion case with the FSAR case shows a comparable trend with small differences in magnitudes of key parameters during the LOL event.

The results of the pressure predictions in the LOL benchmark analysis show that: (1) the Dominion case trips slightly earlier than the FSAR case; and (2) the calculated peak RCS pressure for the Dominion case is lower than that of the FSAR case. During the review, the NRC staff requested the licensee to explain the causes for differences in the pressure response of the Dominion and FSAR cases. In its response to RAI-10 (SRXB) (Reference 3), the licensee indicated that the slightly earlier pressurization is attributed to differences in the SG primary-to-secondary heat transfer associated with the Dominion single-node SG (SNSG) model compared to the FSAR multi-node SG (MNSG) model. For the SNSG model, the secondary-side temperature corresponds to the saturation temperature for the secondary side pressure, and will increase with an increase in pressure. The MNSG model represents SG tube regions that may be either saturated or subcooled. It would predict higher heat transfer rates during transient conditions due to an increase in the nodal number and modeling of dynamic effects for the liquid/vapor flow through the tube bundle. These effects result in a slightly earlier heat-up for the SNSG model and associated increase in primary-side pressure. For the Dominion case, because the pressure increase starts earlier, the reactor trip on high pressurizer pressure occurs slightly earlier.

In addressing item 2 regarding differences in the peak pressure prediction, the licensee indicated that the peak RCS pressure, which occurs after the reactor trip, is closely related to the response of the pressurizer safety valves (PSV). Since the main steam safety valves (MSSV) actuate after the time of peak RCS pressure, they do not affect the calculated peak RCS pressure. As shown in Figure 4.1-1 in Attachment 2 to Reference 5, the peak pressurizer pressure varies over a small range for the FSAR case, achieving pressures that are slightly higher than the Dominion case which has a relatively flat pressure profile when the PSVs open. Since the LOL event results in a very rapid pressure increase, small differences in PSV response (e.g., delays, opening profiles, etc.) can significantly affect the peak pressure. These differences are more pronounced in the RCS cold-leg and reactor vessel lower plenum where peak pressures exceed 2,700 psia [pounds per square inch absolute] and are affected by differences in loop response (RCS loop, reactor vessel, and surge line loss coefficients, reactor coolant pump head dynamics, etc.).

The difference in the over-all predicted RCS pressure between the Dominion and FSAR cases is attributed to a difference in the secondary safety valve models. Specifically, the Dominion model includes the modeling of blowdown in the main SG safety valves and the vendor model does not. The licensee clarified in an e-mail dated March 3, 2016 (Reference 18) that the valve blowdown applies to the closing phase of the valve and results in the valve not becoming fully closed until the steam pressure is less than the pressure at which the valve opened. As a result, the MSSV continues to provide relief flow at pressure below the opening pressure during the closing cycle for the valve.

The result comparison also shows that for the Dominion and FSAR cases, the vessel inlet temperature and RCS coolant average temperature agree in trend and rate of increase, with the Dominion case lagging the FSAR response before the inlet temperature peaks at a slightly lower value, which indicates that the FSAR SG heat transfer degrades sooner than that predicted by Dominion model. This difference in the temperature response is caused by the difference between the use of a MNSG in the FSAR model and the SNSG model employed in the Dominion model.

3.1.4.2.2 Analysis of the Locked Rotor Event

Section 4.2 of Attachment 2 to Reference 5 discusses the updated benchmark analysis for the locked rotor (LR) event. For the LR event, flow through the affected reactor-loop drops rapidly, leading to a reactor trip on a low-flow signal. After the reactor trip, energy stored in the fuel rods continues to be transferred to the reactor coolant, causing the RCS temperature to increase and the coolant to expand. During the transient, heat transfer to the shell-side of the SGs drops because the reduced flow results in a decreased SG tube film coefficient. The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the SGs, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves. For the over-pressure analysis, the licensee assumed that the event was initiated from full-power conditions with one RCP speed set to zero, and credited the reactor coolant low loop flow reactor trip, with a setpoint of 85 percent of the initial flow. The licensee listed in Table 4.2-1 the initial plant conditions and the assumptions used in the LR analysis which showed no differences in the key input and assumptions used in both benchmark analysis and FSAR analysis. The licensee presented the results of the analysis in Figure 4.2-1 to Figure 4.2-7. A comparison of the Dominion analysis with the FSAR analysis shows that the responses are comparable in trend for the LR event, with the Dominion model predicting higher peak RCS pressures. As discussed in the response to RAI-11 (SRXB) (Reference 3), the differences are caused by RCS loop friction losses and fuel rod heat transfer model differences.

3.1.4.1.3 Analysis of the Loss Normal Feedwater Event

Section 4.3 of Attachment 2 to Reference 5 discusses the benchmark analysis for the loss of normal feedwater (LONF) event. During a LONF event, the SG water inventory decreases as a consequence of continuous steam supply to the turbine. The mismatch between the steam flow to the turbine and the feedwater flow leads to the reactor trip on a low-low SG level signal, which actuates the auxiliary feedwater system. As the SG pressure increases following the trip, the SG safety valves open to remove the decay heat. Consistent with the FSAR approach, the licensee analyzed the event as an overpressure event. The licensee listed the initial plant conditions and the assumptions used in the LONF analysis in Table 4.3-1 and showed no differences in the key input and assumptions used in both benchmark analysis and FSAR analysis. The licensee presented the results of the analysis in Figure 4.3-1 to Figure 4.3-7 in Reference 5.

A comparison of the Dominion analysis with the FSAR analysis shows that the transient responses are similar, with a predicted higher peak pressure for the Dominion case during an

LONF event. The differences result mainly from the SG safety relief valve model, which includes the modeling of blowdown in the Dominion analysis and not in the FSAR analysis. This model difference results in higher steam releases and a subsequent increase in heat transfer following the reactor trip. The SG nodal scheme and related heat transfer models along with other modeling differences such as pressurizer spray also affect the transient response. These effects are cumulative, resulting in a higher pressurizer pressure peak compared to the FSAR results.

3.1.4.2.4 Analysis of the Main Steam Line Break Event

Section 4.4 of Attachment 2 to Reference 5 discusses the updated benchmark analysis for the main steam line break (MSLB) event. During an MSLB event, the steam release causes a decrease in the RCS temperature and SG pressure. In the presence of a negative moderator temperature coefficient, the RCS temperature decrease results in an addition of positive reactivity with the potential of power increase. The licensee analyzed the MSLB analysis for the maximum peak power increase that determines a minimum margin to an acceptable fuel design limit. The MSLB analysis assumed that the event was initiated from an instantaneous, double-ended break at the nozzle of one SG from hot shutdown conditions with offsite power available. The licensee listed the initial plant conditions and the assumptions used in the MSLB analysis in Table 4.4-1 of Attachment 2 to Reference 5 and identified three major differences used in the benchmark analysis and FSAR analysis: (1) the Dominion analysis used a heat transfer model that maximizes heat transfer coefficients for the faulted SG secondary side, while the FSAR analysis used a Westinghouse proprietary heat transfer formulation; (2) the Dominion analysis credited boron from the SI system, while the FSAR case did not; and (3) the Dominion analysis used only the Doppler power coefficient (DPC) while the FSAR cases credited the DPC plus the Doppler temperature coefficient (DTC) in the moderator density feedback. The licensee used the reactor vessel nodal scheme in Figure 2-2 of Reference 19 for the analysis of the MSLB event, which is an asymmetric response transient with lower temperature in the core next to the ruptured SG and higher temperature in the other side of the core. The reactor vessel model with a specification of mixing flow fractions was used to simulate conditions from complete to incomplete mixing for the flow from both the cold-side and hot-side of the core. The mixing flow fractions were based on scale model mixing tests performed by Westinghouse (the licensee's response to RAI-13 (SRXB), Reference 2). The assumption of imperfect mixing used in the MSLB analysis is consistent with the methodology documented in Topical Report, VEP-FRD-41-P-A.

The analysis shows that the peak power and heat flux based on the Dominion methods are higher and occur more quickly than the FSAR data. The differences are caused by the SNSG model employed in the Dominion model that calculates a higher steam rate, resulting in a greater cooling effect of the faulted SG on the RCS. They are also the results of differences in the nodal scheme and mixing at the core inlet and outlet between the Dominion case and the FSAR case. The power response for both models is not affected by the delivery of boron to the RCS. This is because the FSAR model does not credit boron, and in the Dominion model, boron does not reach the RCS from the SI system until after the termination of the transient.

3.1.4.2.5 Analysis of the Control Rod Bank Withdrawal at Power Event

Section 4.5 of Attachment 2 to Reference 5 discusses the updated benchmark analysis for the control rod bank withdrawal at power event. The effect of this event is an increase in fuel and coolant temperature. The licensee listed the initial plant conditions and the assumptions used in the Control Rod Bank Withdrawal at Power (RWAP) analysis in Table 4.5-1 and showed no differences in the key input and assumptions used in both benchmark analysis and FSAR analysis. The licensee presented the results of the analysis in Figure 4.5-1 to Figure 4.5-6. A comparison of the Dominion case with the FSAR case shows a comparable trend, with small differences in magnitudes of key parameters during the RWAP event.

For the RWAP 1 percent millirho per second (pcm/sec) case, the core power response shows that its rate of increase for the Dominion model is greater than the FSAR data. The faster power increase rate leads to the Dominion modeling tripping on high neutron flux at about 74 seconds, and the lower power increase rate for the FSAR case results in a reactor trip on an overtemperature ΔT (OTΔT) signal at about 93 seconds. In the RAI-14 (SRXB) response (Reference 3), the licensee indicated that the differences in the core power predictions are caused by the differences in the models for the moderator and Doppler reactivity feedback effects. The moderator reactivity feedback is assumed to be zero for both cases. For Doppler reactivity, the Dominion case uses a DTC while the FSAR case uses a DPC with minimum reactivity feedback conservatively assumed for both cases. The licensee also indicated that the reactor core model used in the FSAR case incorporates proprietary mechanisms to modify the removal of heat from the core. The difference in reactor trip mechanisms between the Dominion and FSAR cases reflects the breakpoint for switching between OTΔT and high flux as shown in FSAR Figure 15.4-10. The results of MPS3 FSAR Chapter 15 non-LOCA analyses indicates the RWAP event is the most limiting event in terms of the margin to the safety limit DNBR in the category of the anticipated operating occurrences (AOOs). Since the licensee proposed to use the RETRAN and Dominion VIPRE-D method to perform DNBR calculations for assessing the fuel integrity during AOOs and accidents, the NRC staff requested the licensee to include in its benchmark analysis the results of the DNBR calculation by using the Dominion VIPRE-D method. In response to RAI-15 (SRXB) (Reference 3), the licensee provided calculated DNBRs as Figure 4.5-7 for the RWAP 1 pcm/sec case. A comparison of the Dominion case with the FSAR case shows a comparable trend with small differences in magnitudes of the predicted values of DNBR. As shown in Figure 4.5-1 of Attachment 2 to Reference 5, the core power rate of increase in the 1 pcm/sec case for the Dominion RETRAN model is greater than the FSAR data such that the reactor trip occurs approximately 20 seconds earlier. The inverse effect of power on DNB is observed in the transient DNBR plot shown in Figure 4.5-7 of Attachment 1 to Reference 3 and the minimum DNBR values for the Dominion and FSAR cases are comparable. In addition, the licensee confirmed that the thermal-hydraulic conditions of the RWAP transient analyzed are within the acceptable range of the NRC-approved DNBR correlations utilized in the VIPRE-D model (WRB-2M and ABB-NV) consistent with the limitations on the use of DOM-NAF-2-P-A.

3.1.4.2.6 Analysis of the Feedwater Line Break Event

MPS3 FSAR Section 15.2.8 discusses the FLB analysis for both cases with and without offsite power available. FSAR Figures 15.2-13 and 15.2-19 indicates that a post-trip return-to-power will occur for the case with offsite power available, and core will remain subcritical throughout the transient for the case without offsite power available. Also, FSAR page 15.2-16 indicates that the FLB is the most limiting event in the decrease in secondary removal category. The analysis of the FLB needs to use a broad scope of the models in RETRAN, including FLB break flow model, RC [reactor coolant] pumps coastdown model, SG heat transfer model, and reactivity feedback model. Although RETRAN is an NRC-approved code, it has not been applied to MPS3 for the FLB analysis. During the review, the NRC staff requested the licensee to provide an FLB benchmark analysis to demonstrate that the code produces acceptable results when applied to MPS3. In response, the licensee performed the RETRAN benchmark analysis for the FLB event for both FLB cases with and without offsite power available. The licensee listed the initial plant conditions and the assumptions used in the FLB analysis in Table 4.6-1 of Attachment 2 to Reference 5 and showed no differences in the key input and assumptions used in the Dominion analysis and FSAR analysis. It presented the results of the analysis in Figure 4.6-1 through Figure 4.6-8, and Figure 4.6-9 through Figure 4.6-15 of Attachment 2 to Reference 5 for the FLB with offsite power case and the FLB without offsite power available case, respectively. A comparison of the Dominion analysis with the FSAR analysis shows that for the FLB with offsite power case, the transient responses are in good agreement, and for the FLB without offsite power available case, the transient responses are comparable in trend, with small differences observed early in the transient (for a period from 100 seconds to 1,000 seconds into the transient) for RCS temperatures. The RCS temperature differences are caused by the differences in the Dominion SNSG model and the FSAR MNSG model. These differences in the SG models have a negligible effect on the long-term primary side heat transfer and associated temperature response.

3.1.4.2.7 Analysis of the Steam Generator Tube Rupture Event

MPS3 FSAR 15.6.3 discusses the SGTR analysis for two cases: (1) the SG overfill margin analysis that is used to validate the assumption of no water released from the affected SG to atmosphere; and (2) the mass release analysis that is used as input to a computer code for calculating the dose releases. This analysis involves simulation of the mitigating strategies directing operators to identify and isolate the ruptured SG, cooldown the RCS to establish subcooling margin, depressurize to restore RCS inventory, and terminate safety injection to stop primary-to-secondary leakage. Although RETRAN is an NRC-approved code, it has not been applied to MPS3 for the SGTR analysis. During the review, the NRC staff requested the licensee to provide an SGTR benchmark analysis to demonstrate that the code produces acceptable results when applied to MPS3. In response, the licensee performed the RETRAN benchmark analysis for the SGTR event for two cases: a mass releases case and a SG overfill case. The licensee presented the results in Section 4.7 of Attachment 2 to Reference 5. The licensee listed the initial plant conditions and the assumptions used in the SGTR analysis in Table 4.7-1 of Attachment 2 to Reference 5 and showed no differences in the key input, operator actions, and assumptions used in the Dominion analysis and FSAR analysis. Although RETRAN was approved previously by NRC for use in the SGTR analysis, the NRC SER approving RETRAN limited its use in Limitation 38 (Reference 30), which indicates that the SGTR event should not be analyzed for two-phase flow conditions without further justification of two-phase slip models used in the analysis. In the response to NRC RAI-17 (SRXB) (Reference 3) regarding compliance with the SER limitation, the licensee confirmed that the

RCS flow remains single-phase and subcooled throughout the entire STGR benchmark analysis, justifying that it meets the cited SER limitation.

Mass Releases Analysis

The licensee presented the results of the mass releases analysis in Figure 4.7-1 to Figure 4.7-9. A comparison of the mass releases analysis for the Dominion case and FSAR case shows good agreement in transient responses of the pressurizer pressure, SG pressure, intact-loop RCS temperature, primary to secondary break flow rate, ruptured SG mass release rate, intact SG mass release rate, and RCS flow. The differences of transient response of the pressurizer level and ruptured-loop cold-leg temperature are discussed below.

For the pressurizer level response, Figure 4.7-2 shows that the FSAR level decreases more than the Dominion level during the RCS cooldown phase (approximately 3,200-3,700 seconds). The differences occur because the primary to secondary heat transfer is reduced for the Dominion case caused by the loss of natural circulation flow on the ruptured SG and during a period when the SI flow is increasing significantly due to the reduction in RCS pressure. After SI is isolated, the longer duration in break flow for the FSAR case is reflected in lower pressurizer level at the end of the transient. These divergences in pressurizer level occur late in the transient well after the flow path to atmosphere through the failed atmosphere dump valve (ADV) has been isolated and do not have a significant effect on the overall results of mass releases.

For the ruptured SGs, the predicted RCS temperatures (shown in Figure 4.7-5) are in good agreement between the Dominion and FSAR cases until about 3,600 seconds, at which time the Dominion cold-leg temperature trends below the FSAR results. This is caused by a small natural circulation flow rate that occurs on the ruptured loop as a result of the RCS cooldown. With the small RCS loop flow rate, the SI flow with a low temperature has a more noticeable effect on cold-leg fluid temperature. The predicted low cold-leg temperature in the ruptured SG has a small effect on the overall results for the transient since most of the heat removal occurs through the intact SGs during this time and the ruptured SG has been previously isolated.

SG Overfill Analysis

For the SG overfill analysis, the licensee presented the results in Figures 4.7-10 through Figure 4.7-17. A comparison of the SG overfill analysis for the Dominion case and AOR (stretch power uprate (SPU), Reference 29) case shows good agreement in transient responses of the pressurizer pressure, SG pressure, intact-loop RCS temperature, and RCS flow. The differences of transient response of the pressurizer level, ruptured-loop cold-leg temperature, primary-to-secondary break flow rate, and ruptured SG liquid volume are discussed below.

For the pressurizer level response, Figure 4.7-11 shows similar trends between the Dominion case and SPU case, with the SPU level decreasing more than the Dominion level during the RCS cooldown phase (approximately 2200-3200 seconds). The differences occur because the primary to secondary heat transfer is reduced for the Dominion case caused by the loss of natural circulation flow on the ruptured SG and during a period when the SI flow is increasing significantly due to the reduction in RCS pressure. After the SI is isolated, the higher break flow rates shown in Figure 4.7-15 for the SPU case are reflected in lower pressurizer level at the end of the transient.

For the ruptured SGs, the predicted RCS cold-leg temperatures (shown in Figure 4.7-14) are in good agreement between the Dominion and FSAR cases until about 2600 seconds when natural circulation flow is lost in the ruptured RCS loop and the cold-leg temperatures are more strongly affected by the cooler SI flow as discussed above for the mass release analysis after SI flow is terminated, the Dominion cold-leg temperature trends toward the SPU value. For the primary-to-secondary break flow rate, Figure 4.7-15 shows there is good agreement between the Dominion and SPU cases until the period late in the transient after SI has been isolated and the break flow is trending towards zero. This is also observed for the ruptured SG liquid volume response shown on Figure 4.7-16 where the Dominion and SPU responses agree well, with the Dominion value stabilizing at a lower value near the end of the transient. The licensee indicated that the following factors could affect the final SG fluid volume: (1) any difference in the assumed decay heat profile resulting in a different amount of fluid boiled from the SG secondary and associated liquid volume; (2) any differences in the integrated SI fluid injection affecting the RCS fluid inventory available for release to the ruptured SG; (3) on the secondary side, differences in the integrated auxiliary feedwater (AFW) flow rates affecting the fluid delivered to the ruptured SG fluid volume as well as the energy removed by the intact SGs; (4) differences in SG relief valve flow rates affecting mass and energy removal from the system, and (5) any differences in the Dominion and SPU model nodal scheme and related assumptions affecting the differential pressure between the respective fluid levels in the RCS and SG secondary, which would affect the final equilibrium level and associated fluid volume.

Based on the discussion of the benchmark analysis in Subsection 3.1.4.2.1 through Subsection 3.1.4.2.7 above, the NRC finds that: (1) the Dominion MPS3 RETRAN benchmarking analysis has included appropriate non-LOCA cases discussed in MPS3 FSAR; (2) the Dominion MPS3 RETRAN model compares reasonably well with the vendor RETRAN model in predicting the trend of the RCS response for the selected non-LOCA cases; (3) the differences in the magnitude of the RCS response can be explainable based on differences in nodal schemes, inputs, or modeling assumptions, and; (4) the use of the Dominion RETRAN method is within the NRC-accepted conditions. Therefore, the NRC staff determines that the RETRAN methodology, as discussed in VEP-FRD-41, Rev. 02, References 3, 5, 19, and Section 3.4 of Reference 16, is applicable to MPS3.

3.1.5 Statistical DNBR Evaluation Methodology (TR VEP-NE-2-A)

3.1.5.1 Introduction and Background

This section describes plant-specific application of statistical DNBR methodology for MPS3 cores containing the Westinghouse 17x17 Robust Fuel Assembly (RFA-2) fuel product. This section provides technical basis and documentation for the application of NRC-approved Dominion Topical Report (TR), VEP-NE-2-A (Reference 11) to MPS3. This application employs VIPRE-D thermal-hydraulics (T-H) computer code (Reference 12) with the Westinghouse WRB-2M, ABB-NV, and WLOP Critical Heat Flux (CHF) correlations for the T-H analysis of Westinghouse 17x17 RFA-2 fuel products for MPS3. Attachment 6 of the LAR (Reference 20) describes the development and implementation of the statistical DNBR limit evaluation methodology as applied to the MPS3 fuel design.

The licensee is seeking approval for the inclusion of TR VEP-NE-2-A and Fleet Report DOM-NAF-2-P-A, Appendix C and D (References 11 and 12) to the TS 6.9.1.6.b list of NRC-approved methodologies used to determine core operating limits and in the reference list of the COLR.

VIPRE-D is the Dominion version of the VIPRE computer code that was originally developed for Electric Power Research Institute (EPRI) by Pacific Northwest National Laboratory (PNNL) to predict the CHF and DNBR of reactors. The NRC-approved fleet report, DOM-NAF-2-P-A, Appendix C describes the verification and qualification of the WRB-2M CHF correlation and Appendix D describes the verification and qualification of the ABB-NV and WLOP CHF correlations. The WRB-2M CHF correlation is applicable to the DNBR evaluation of the Westinghouse 17x17 RFA-2 fuel design. The ABB-NV and WLOP CHF correlations are applicable to the DNBR evaluation of the Westinghouse 17x17 RFA-2 fuel product. RFA-2 fuel product for transients that leads to low primary system pressure. The statistical design limits (SDLs) obtained by this implementation are for the following applications:

1. Technical Specifications Change Request to add DOM-NAF-2-P-A and relevant Appendixes to the plant's COLR list,

2. SDL(s) for the relevant code/correlation(s),

3. Any TS changes related to thermal over-temperature ΔT (OTΔT), overpower ΔT (OPΔT), axial power distribution (FΔI), enthalpy rise factor (FΔH) or other reactor protection function, as well as revised Reactor Core Safety Limits (RCSLs), and

4. List of FSAR transients for which the code/correlations will be applied.

NRC-approved TR, VEP-NE-2-A describes a methodology for the statistical treatment of key uncertainties in core Thermal-Hydraulics (T-H) DNBR analysis and provides DNBR margin through statistical analysis rather than deterministic uncertainty treatment. This TR was approved by the NRC staff subject to the following conditions for its use:

1. The selection and justification of normal statepoints used for plant specific implementation,

2. Justification of the distribution, mean and standard deviation for all statistically treated parameters should be included in the submittal,

3. Justification of the value of model uncertainty must be included, and

4. For the relevant CHF correlations, justification of the 95/95 DNBR limit and the normality of the M/P distribution, its mean and standard deviation must be included in the submission.

3.1.5.2 Implementation of Statistical Methodology

TR VEP-NE-2-A, "Statistical DNBR Evaluation Methodology" (Reference 11), describes Dominion's methodology for statistically treating several of the important uncertainties in the DNBR analysis. The methodology in TR VEP-NE-2-A is employed to develop SDLs for the VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs for Westinghouse RFA-2 fuel at MPS3. The VIPRE-D/WLOP code/correlation is not used for statistical analyses for RFA-2 fuel at MPS3. With the uncertainties accounted for in the statistical analyses, the new SDL is larger than the deterministic code/correlation pair design limit, however, it is advantageous to use the SDL since statistical methodology permits the use of nominal operating initial conditions instead of requiring the application of evaluated uncertainties to the initial conditions for statepoints and transient analysis.

SDL is a Monte Carlo type analysis where two-thousand (2,000) random statepoints are generated for each statepoint and supplied to the VIPRE-D code which calculated the minimum departure from nucleate boiling ratio (MDNBR) for each statepoint. Each MDNBR is randomized by a code correlation uncertainty described in TR VEP-NE-2-A using the 95 percent confidence limit on the VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pair measured-to-predicted (M/P) CHF ration standard deviation.

In a response to an NRC RAI-4 (SNPB) (Reference 2), the licensee responded that the randomized DNBR is consistent with the methodology described in the LAR and calculated as:

$$Randomized\ DNBR = \frac{Calculated\ DNBR}{[1.0+s\left(\frac{M}{P}\right).K(95).Normalized\ Random\ Number\ \ ]}$$

Where:

s(M/P) is the standard deviation of the code/correlation M/P database for the CHF correlation taken from DOM-NAF-2-P-A, and

K (95) is a sample correction factor that depends on the size of the experimental database supporting the correlation, and is calculated based on the equation given in Statistical DNBR Evaluation Methodology and is equal to:

$$\mathrm{K}(95) = \sqrt{[}\frac{2.(n-a)}{(\sqrt{2n}-3-1.645)\text{^}2} \quad \text{(Reference 11)}$$

The standard deviation of the resultant randomized DNBR distribution is increased by a correction factor to obtain a 95 percent upper confidence limit and then combined root-sum-square with code and model uncertainties to obtain a total DNBR standard deviation (s_{total}) as:

$$SDL = 1 + 1.645 * s_{total}$$

Where, the 1.645 multiplier is the z-values for the one-sided 95 percent probability of a normal distribution. The SDL provides peak fuel rod DNBR protection at greater than the “95/95 level” (i.e., provides a 95% probability at a 95% confidence level (95/95) that the peak rod does not experience DNB).

Consistent with VEP-NE-2-A methodology, inlet temperature, pressurizer pressure, core thermal power, reactor vessel flow rate, core bypass flow, nuclear enthalpy rise factor ($F_{\Delta H}{}^{N}$, and the engineering enthalpy rise factor ($F_{\Delta}{}^{E}$) were selected parameters in the implementation of statistical analysis. These uncertainties are listed in Table 3.1.5-1.

Table 3.1.5-1 MPS3 Parameter Uncertainties

Parameter	Nominal Value	Standard Deviation	Uncertainty	Distribution	Uncertainty Description
Pressure (psia)	2250	30 psi	±58.8 psi at 2σ	Normal	Uncertainty corresponds to two-sided 95% probability
Temperature (°F)	557.06	2.5	±4.9 at 2σ	Normal	Two-sided, 95% probability distribution
Power (MWt)	3712	1.0%	±1.96 at 2σ	Normal	Two-sided, 95% probability distribution
Flow (gpm)	379200	1.5%	±2.94% at 2σ	Normal	Two-sided, 95% probability distribution
$F_{\Delta H}{}^{N}$	1.635	2.0%	±4.0% at 2σ	Normal	Two-sided, 95% probability distribution
$F_{\Delta H}{}^{E}$	1.0	N/A	±	Uniform	VEP-NE-2-A (Reference 11) treats this uncertainty as a uniform probability distribution
Bypass (%)	7.6	N/A/	±1.0%	Uniform	Monte Carlo analysis used a best estimate bypass flow of 7.6% with an uncertainty of 1% and uniformly distributed

3.1.5.2.1 CHF Correlation Uncertainty

Only the WRB-2M/ABB-NV/WLOP CHF correlations that are used for DNBR calculations for Westinghouse 17x17 RFA-2 fuel product and the WRB-2M and ABB-NV CHF correlations are applicable to operating conditions at which the statistical DNBR methodology is applied. The WLOP CHF correlation is used deterministically. The ABB-NV correlation is only used below the first mixing grid and the WLOP correlation is used when the operating conditions are outside of the range of validity of the WRB-2M and ABB-NV CHF correlations, such as the MSLB evaluation, where there is reduced temperature and pressure. Table 3.3-1 of the LAR lists the deterministic DNBR limit deterministic design limit (DDL) correlation data for VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs. Consistent with the methodology in VE-NE-2-A a 95 percent upper confidence limit (K(95) in Section 3.1.5.2) is applied to the calculated correlation statistics.

NRC issued Information Notice (IN) 2014-01 dated February 21, 2014 (Reference 22) which raised a concern that the DNBR safety limit generated from statistical methodologies may not properly account for a conservative bias that may be included in the NRC-approved CHF correlation limit as defined in the SER for VEP-NE-2-A. The IN further discussed the fact that the correction of this inconsistency may increase the statistically-based DNBR safety limit. The magnitude of the increase is dependent on the difference between the CHF correlation's 95/95 statistics and the NRC-approved CHF correlation limit.

The licensee stated in their LAR submittal that their implementation of SDL is consistent with the methodology of TR, VEP-NE-2-A. The acceptability of the use of the calculated standard deviations is based on the use of a 95 percent upper confidence factor that is essentially equivalent to the Owen's tables for ensuring a 95 percent probability at a 95 percent confidence limit. In section 3.6.1 of Attachment 6 of the LAR, the licensee provided supporting information and introduced the correction factor (S_{DNBR}) to ensure that the SDL developed in accordance with the methodology of VEP-NE-2-A and using the calculated correlation statistics provides a 95 percent probability at a 95 percent confidence level (95/95) that the peak rod does not experience DNB. In its response to RAI-5 (SNPB) regarding the IN 2014-01 (Reference 2), the licensee reiterated that the SDL calculation was developed per the VEP-NE-2-A methodology, and there is no need to modify the licensee's calculations. The NRC staff verified the licensee's calculational procedures and methods and determined that their SDL calculations are acceptable.

3.1.5.2.2 Model Uncertainty

Condition 3 of the SER for VEP-NE-2-A states that the licensee must provide justification of the value of model uncertainty (F_M) and be included in the plant specific submittal. The VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pair SDLs for MPS3 were developed using the VIPRE-D 21-channel production model for MPS3 with the 17x17 RFA-2 fuel design. Since the production model that Dominion intends for the MPS3 evaluations are used to develop the SDL, the licensee determined there is no need for additional model uncertainty. Therefore, the model uncertainty is set to zero. The NRC staff finds this approach acceptable.

3.1.5.2.3 Code Uncertainty

The NRC-approved VEP-NE-2-A methodology states that a code uncertainty (F_c) must be applied because of two factors: (i) the effect of analyzing a full core with a correlation which was based on steady state test bundle data and (ii) and the effect of performing the analyses with the Virginia Power (former licensee) COBRA code when the W-3 data were reduced by the use of a Westinghouse thermal-hydraulics code. These uncertainties are independent of the correlation. The code uncertainty was quantified at 5 percent; consistent with the factors specified for other thermal/hydraulic codes in VEP-NE-P-A and the basis of this uncertainty is described in the application of this methodology at the Surry power station (Reference 11). A one-sided 95 percent confidence level on the code uncertainty is then 3.04% (= (5.0%) /1.645). The use of the 1.645 divisor (the one-sided 95 percent tolerance interval multiplier) is conservative since the NRC staff considers the 5 percent uncertainty to be a 2σ value. Upon review of the references and the methodology described above, the NRC staff has determined that the selection of the value of the code uncertainty is acceptable.

3.1.5.3 Monte Carlo Calculations

For the Monte Carlo analysis, nine (9) nominal statepoints that cover full range of nominal operation and AOO transients are selected for both WRB-2M and ABB-NV CHF correlations. These nine statepoints cover a range of conditions, such as pressure, temperature, etc. over which the statistical methodology is applied and also cover the DNB limiting range of the Reactor Core Safety Limits (RCSL) and within the validation range of applicability of the associated correlations. Tables 3.6-1 and 3.6-2 of the Attachment 6 of the LAR lists pressurizer pressure, inlet temperature, power, flow $F_{\Delta H}{}^{N}$ and MDNBR for VIPRE-D/WRB-2M and VIPRE-D/ABB-NV, respectively, for the Westinghouse 17x17 RFA-2 fuel at MPS3.

The Monte Carlo calculations consisted of 2,000 calculations for each of the nine nominal statepoints for each CHF correlation. The DNBR standard deviation at each nominal statepoint was augmented by the code/correlation uncertainty, the small sample correction factor, and the code uncertainty to obtain a total DNBR standard deviation. Equation 3.3 of Attachment 6 of the LAR provides a relationship for the total DNBR standard deviation using the Root-Sum-Square method and the total standard deviation (s_{TOTAL}) is dependent on standard deviation of the randomized DNBR distribution, uncertainty in the standard deviation of the 2000 Monte Carlo simulations that provides a 95 percent upper confidence limit on standard deviation, the code uncertainty, and the model uncertainty.

$$s_{TOTAL} = \sqrt{s_{DNBR}^2 \left(1.0 + \sqrt{\left\{\sqrt{\frac{n-1}{x^2}} - 1.0\right\}^2 + \frac{1}{N}}\right)^2 + F_C^2 + F_M^2}$$

The limiting peak fuel rod SDL was calculated to be 1.225 for the VIPRE-D/WRB-2M code/correlation pair and 1.177 for the VIPRE-D/ABB-NV code/correlation pair. The Monte Carlo Statepoint analysis is summarized in Tables 3.6-3 and 3.6-4 of Attachment 6 of the LAR.

The use of the correction factor for the total DNBR standard deviation ensures that the SDL developed in accordance with the methodology of VEP-NE-2-A and using the calculated correlation statistics provides a 95 percent probability at a 95 percent confidence level (95/95) that the peak rod does not experience DNB. The NRC staff verified the use of the accepted methodology in the calculation of SDL and determined that the use of the calculated correlation statistics as input to SDL calculation is acceptable and reasonable.

3.1.5.4 Full Core DNB Probability Summation

The data statistics are used to determine the number of rods expected in DNB. The DNB sensitivity is estimated as partial derivative of DNBR divided by partial derivative of ($1/F_{\Delta H}$), and are listed in Tables 3.7-1 and 3.7-2 of Attachment 6 of the LAR for WRB-2M and ABB-NV correlations, respectively and are denoted by β. To ensure conservatism in the calculations, a one-sided tolerance limit of β is used. Variable $1/F_{\Delta H}$ is the most statistically significant independent variable in the linear regression model, yielding a regression coefficient greater than 99 percent. Table 3.7-3 of Attachment 6 of the LAR lists a representative rod census curve used for determining SDL; this table provides probable maximum limit $F_{\Delta H}$ versus maximum % of fuel rods in core. Tables 3.7-4 and 3.7-5 of Attachment 6 of the LAR provides full core DNB probability summations for VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs, respectively. After a review of the values presented in the above Tables, the NRC staff determined that the values listed in these tables are acceptable.

3.1.5.5 Verification of Nominal Statepoints Used in SDL Calculations

Condition 1 of the SER for VEP-NE-2-A requires the nominal statepoints used in the SDL analysis must be justified in providing a bounding DNBR standard deviation for any set of conditions. This justification is performed by demonstrating that s_{TOTAL} is maximized for any real set of conditions at which the core approaches the SDL. For this, a regression analysis is performed using the unrandomized DNBR standard deviations at each statepoints. Tables 3.8-1 and 3.8-2 of Attachment 6 of the LAR show the R^2 linear regression coefficients verifying the nominal statepoints for WRB-2M and ABB-NV, thus validating the independence. These values are listed in Table 3.1.5-2

Table 3.1.5-2
Regression Coefficients for the Verification of the Nominal Statepoints for MPS3 17x17 RFA-2 fuel with VIPRE-D/WRB-2M and VIPRE-D/ABB-NV Code Correlation Pairs

Statepoints	**R^2 Linear Regression For WRB-2M (%)**	**R^2 Linear Regression For ABB-NV (%)**
Pressure	2.75	23.62
Temperature	7.70	34.95
Flow Rate	1.27	24.3
Power	3.04	27.76

In a response to NRC RAI-6 (SNPB) (Reference 2), the licensee stated that the relatively large differences in R^2 values between the two correlations are expected since the correlations used to evaluate the behavior are based on different equation forms and experimental databases. The nominal statepoints at which the two CHF correlations are evaluated are also different (Tables 3.6-1 and 3.6-2 of Attachment 6 of the LAR). For a linear regression analysis that is performed for these two correlations, the analysis is expected to generate different values for R^2 since different statepoints are used for the two correlations. Also, for a given correlation, the R^2 values are similar and there is not a strong dependence on any one single input condition. The NRC staff has reviewed the statistical regression analysis as well as the inputs to the two correlation analyses and determined that the R^2 values are acceptable.

3.1.5.6 Scope of Applicability

Condition 4 of the SER for VEP-NE-2-A requires that for the relevant CHF correlations, justification of the 95/95 DNBR limit, and the normality of the M/P distribution, its mean and standard deviation must be included in the submission, unless there is an approved TR documenting these such as DOM-NAF-2-P-A.

Table 3.9-1 of Attachment 6 of the LAR lists the accidents to which the SDL methodology is applicable. These include all Condition I and II DNB events except the Rod Withdrawal from Subcritical (RWSC) and the complete Loss of Flow, the Locked Rotor Accident, the Single Rod Cluster Control Assembly Withdrawal at Power, and feedwater system pipe break. The Statistical DNBR Evaluation Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the bypass flow, $F_{\Delta H}^{N}$ (measurement component) and hot channel uncertainties which are statistically calculated into the DNBR limit.

3.1.5.7 Application of VIPRE-D/WRB-2M/ABB-NV/WLOP to MPS3

The VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs, together with the Statistical DNBR Evaluation Methodology, will be applied to all Condition I and II DNB events (except RWSC), and to the Complete Loss of Flow event and the Locked Rotor Accident. The WRB-2M, ABB-NV and WLOP CHF correlations are used for the DNBR calculation for Westinghouse RFA-2 fuel product. The WLOP CHF correlation is used for operating conditions outside the range of applicability of WRB-2M and ABB-NV CHF correlations, namely, for the MSLB accident analysis.

Thermal margin analyses evaluates the design and safety analysis limits and these limits are used to define the available DNBR margins for each application. The difference between the safety analysis limit (SAL) and the design limit is the available retained DNBR margin. For deterministic DNB analyses, the design DNBR limit is set equal to the applicable code/correlation limit and it is termed the DDL. For statistical DNB analyses, the design DNBR limit is set equal to the applicable SDL.

Table 4.2-1 of Attachment 6 to the LAR lists the DDLs and SDLs for the three CHF correlations and reproduced here as Table 3.1.5-3:

Table 3.1.5-3 DNBR Limits for WRB-2M, ABB-NV and WLOP

Correlations	VIPRE-D/WRB-2M	VIPRE-D/ABB-NV	VIPRE-D/WLOP
DDL	1.14	1.14	1.22
SDL	1.23	1.19	Deterministic
SAL	1.50	1.50	1.50

The SDL limit provides a peak fuel rod DNB protection with at least 95 percent probability at a 95 percent confidence level and a 99.9 percent DNB protection for the full core. A deterministic and statistical SAL equal to 1.50 has been selected for 17x17 RFA-2 fuel at MPS3 with the VIPRE-D/WRB-2M, VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs. This SAL is applicable for all deterministic analyses for a maximum peaking factor $F_{\Delta H}^{N}$ equal to 1.65 and for all statistical analyses for a maximum peaking factor $F_{\Delta H}^{N}$ equal to 1.587.

The difference between SAL and the design limit is available as retained DNBR margin:

$$\text{Retained DNBR Margin} = 100 * \left(\frac{SAL-DDL}{SAL} \right)$$

Retained DDNBR margins are listed in Table 3.1.5-4

Table 3.1.5-4 DNBR Limits and Retained DNBR Margins

Deterministic DNB Applications			
DNB Correlation	DDL	SAL_{DET}	Retained DNBR Margin (%)
WRB-2M	1.14	1.50	24.0
ABB-NV	1.14	1.50	24.0
WLOP	1.22	1.50	18.6
Statistical DNB Applications			
DNB Correlation	SDL	SAL_{DET}	Retained DNBR Margin (%)
WRB-2M	1.23	1.50	18.0
ABB-NV	1.19	1.50	20.6

The NRC staff has reviewed processes that calculated the retained DNBR margins, deterministic safety analysis limits, for the code/correlation pairs and determined that the licensee has applied the approved methodology in the calculations and are therefore acceptable.

3.1.5.8 NRC Staff Conclusion - Statistical DNBR Evaluation Methodology

The licensee has proposed to adopt the use of the Dominion methodology in TR VEP-NE-2-A for statistical DNBR evaluation for MPS3. Using a combination of Monte Carlo analysis using 2,000 random statepoints, standard deviation of randomized DNBR distributions which is the un-randomized standard deviation corrected for CHF correlation uncertainty, a combination of

Root Sum Square with code, and model uncertainty standard deviations, the licensee obtained a total DNBR standard deviation (Tables 3.6-3 and 3.6-4 of Attachment 6 of the LAR). The analysis resulted in the SDLs of 1.23 for VIPRE-D/WRB-2M and 1.19 for VIPRE-D/ABB-NV.

The NRC staff reviewed Attachment 6 of the LAR that used the Dominion methodology to calculate the SDLs for MPS3. The NRC staff determined that the licensee appropriately used the approved methodology to determine the SDL and provided sufficient margin through the use of statistical rather than deterministic uncertainty treatment. The staff finds the licensee's analysis satisfied the conditions that were listed in the SER for the methodology TR VEP-NE-2-A. Therefore, the NRC staff has determined that the SDLs that are listed in Table 3.1.5-3 of this SE constitute a design basis limit for a fission product barrier.

3.1.6 Reactor Core Thermal-Hydraulics using VIPRE-D Computer Code (TR DOM-NAP-2-A, Rev. 0.3)

The reactor core thermal-hydraulics code VIPRE-D described in Dominion TR DOM-NAF-2-A, Rev. 0.3, and Section 3.6 of Reference 16 is a Dominion-modified version of the VIPRE-01. VIPRE-D is used to calculate reactor coolant conditions to verify that the DNBR design safety limit is maintained. It has been adapted to accommodate the various fuel designs used at Dominion nuclear power stations by incorporating vendor proprietary CHF correlations.

VIPRE-D was approved by the NRC for PWR licensing calculations up to CHF using approved CHF correlations with the conditions and limitations listed in the SERs approving Dominion TR, DOM-NAF-2-A, and EPRI Report, NP-2511-CCM, "VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores," (Reference 34). The licensee showed in Section 3.6.3 of Reference 16 its compliance with each of the applicable conditions and limitations imposed in the NRC-SERs of TRs, DOM-NAF-2-A and NP-2511-CCM, for the use of VIPRE-D at MPS3. The licensee also indicated that the use of VIPRE-D for MPS3 would be in a manner consistent with the conditions and limitations relating to plant-specific and fuel-specific application discussed in Section 3.6.3 of Reference 16. In addition, as discussed in Section 3.1.5 above, the plant-specific and fuel-specific statistical design limit is determined within the context of the statistical DNBR evaluation methods. Therefore, the NRC staff determines that the use of VIPRE-D discussed in Dominion TR DOM-NAF-2-A and Section 3.6 of Reference 16 is acceptable for the core thermal-hydraulics analysis to support licensing application for MPS3.

3.2 Technical Specifications Changes

The proposed TS changes documented in Reference 21 intend to apply the Dominion reload methods to MPS3, and address the issues identified in Westinghouse NSAL-09-5, Rev. 1, NSAL-15-1 and Westinghouse Communication 06-IC-03. The NRC staff has reviewed the proposed TS changes and provided its evaluation as follows.

3.2.1 Deletion of TS 1.43 and 1.44 (ALLOWED POWER LEVEL) – Definitions

The licensee proposed to delete the definitions of APL^{ND} and APL^{BL} as entries 1.43 and 1.44 in the MPS3 TS. Both definitions would be replaced with the word "Deleted." The staff compared the proposed changes to the STS guidance as part of the review.

In the LAR (Reference 1), the licensee explains this as follows:

> Definition 1.43 specifies APL^{ND} as the minimum allowable nuclear design power level for base load operation. The value of APL^{ND} is specified in the COLR. This definition is being deleted since the base load operation mode is not supported by the Dominion methods.

and,

> Definition 1.44 specifies APL^{BL} as the maximum allowable power level when transitioning into base load operation. This definition is being deleted since the base load operation mode is not supported by Dominion methods.

According to the applicable STS for MPS3, NUREG-1431, Rev. 4, in Section 1.1 Definitions, "The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications." The NRC staff reviewed the licensee's reasons for deletion of these definitions. With the use of Dominion's methodology, definition of these terms is no longer needed because they will no longer appear elsewhere in the TS. The proposed changes are administrative in nature and consistent with use of defined terms in the NUREG-1431, Rev. 4, guidance; therefore, the NRC staff finds them to be acceptable and in accordance with 10 CFR 50.36. Additionally, the licensee proposed to use Dominion RPDC methods to replace the RAOC methods for future cycles of MPS3. Dominion RPDC methods do not support the base load operation mode. The proposed deletion is consistent the approved Dominion methods discussed in Section 3.1.2 of this safety evaluation (SE), and therefore, is acceptable.

3.2.2 TS 3.2.1.1, SR 4.2.1.1 - Axial Flux Difference and TS 3.2.2.1, SR 4.2.2.1 - Heat Flux Hot Channel Factor - $F_Q(Z)$

The proposed changes involve additions, deletions and revisions to existing content in the TS that are associated with TS 3.2.1.1, TS 3.2.2.1, SR 4.2.1.1, and SR 4.2.2.1. These changes accomplish three key objectives: (1) accommodate implementation of the Dominion RPDC method; (2) removal of base load operation; and (3) provide resolution of issues documented in Westinghouse letters NSAL-09-5, Rev. 1 (Reference 23), 06-IC-03 (Reference 24) and NSAL-15-1 (Reference 25). The NRC staff provided its evaluation of the specific proposed changes as follows:

3.2.2.1 TS LCO 3.2.1.1

- TS LCO 3.2.1.1.a states that "The limits specified in the CORE OPERATING LIMITS REPORT (COLR) for Relaxed Axial Offset Control (RAOC) operation, or"

- TS LCO 3.2.1.1 Action a. states that "For RAOC operation with the indicated AFD outside of the applicable limits specified in the COLR."

The proposed changes to above TS 3.2.1.1.a and TS 3.2.1.1 Action a. deletes a reference to RAOC operation. The changes are to revise the TS to be consistent with the licensee's intent to make the TS more general regarding specific axial power distribution control methodology.

Existing LCO 3.2.1.1 Action a. would be modified by removing the words "For Relaxed Axial Offset Control (RAOC) operation" since RAOC operation will no longer be employed. These words distinguish maintenance of the AFD within the limits specified in the COLR from optional part b which was to maintain the AFD within a target band about the flux difference during base load operation. Additionally the letter "w" in the word "With ..." is capitalized to begin revised Action a.

Since the RAOC methods are replaced by the approved Dominion RPDC methods discussed in Section 3.1.2 of this SE, the deletion of a reference to RAOC is acceptable.

- TS LCO 3.2.1.1.b states that "Within the target band about the target flux difference during base load operation, specified in the COLR."

- TS LCO 3.2.1.1 Action b.1 and b.2 states that "For base load operation above APL^{ND} with the indicted AFD outside of the applicable target band about the target flux differences:

    1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or

    2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue base load operation within 30 minutes."

The above TS 3.2.1.1.b, TS 3.2.1.1 Action b.1 and Action b.2 are being deleted, and replaced for each with the word "Deleted". The TSs are associated with the base load mode of operation, which is supported by the RAOC methods for the current cycle. The approved Dominion methods to replace the RAOC methods, as discussed in Section 3.1.2 for MPS3 future cycles, do not support the base load of operation. Therefore, the proposed TS deletions are acceptable.

3.2.2.2 Surveillance Requirements 4.2.1.1.3 and 4.2.1.1.4

SR 4.2.1.1.3 states that "When in base load operation, the target flux difference of each OPERABLE excore channel shall be determined by measurement at the frequency specified in the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable."

SR 4.2.1.1.4 states that "When in base load operation, the target flux difference shall be updated at the frequency specified in the Surveillance Frequency Control Program by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.1.3 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable."

The above SR 4.2.1.1.3 and SR 4.2.1.1.4 are being deleted and replaced with the word "Deleted". The SRs are related to the base load mode of operation, which is supported by the RAOC methods for the current cycle. The approved Dominion methods to replace the RAOC methods, as discussed in Section 3.1.2 above for MPS3 future cycles, do not support the base load of operation. The NRC staff has reviewed the proposed deletion of the above SRs and determined that they are consistent with the change to Dominion's RPDC methods and sufficient to continue to comply with 10 CFR 50.36(c)(3). Therefore, the proposed deletions of the SRs are acceptable.

3.2.2.3 TS LCO 3.2.2.1

TS LCO 3.2.2.1 is modified from:

$F_Q(Z)$ shall be limited by the following relationships:

$F_Q(Z) \leq (F_Q^{RTP}/P)\ K(Z)$ for $P > 0.5$

$F_Q(Z) \leq (F_Q^{RTP}/0.5)\ K(Z)$ for $P \leq 0.5$

F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) provided in the CORE OPERATING LIMITS REPORT (COLR).

Where: P = THERMAL POWER / RATED THERMAL POWER, and

K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

To:

$F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR.

Use of a relationship where $F_Q(Z)$ is approximated by $F_Q^M(Z)$ is consistent with the approved Dominion RPDC methodology and agrees with the guidance of NUREG-1431, Revision 4 where a similar relationship is used in RAOC-W(Z) Methodology or CAOC-W(Z) Methodology. Limits for the specification vary with each specific core reload, and are therefore, located in the COLR. The NRC staff finds that this is consistent with current technical specifications (CTS) and the NUREG-1431, Revision 4 guidance.

The relationships for $F_Q(Z)$ for $P > 0.5$ or $P \leq 0.5$ and a supporting description are relocated to the TS bases. The basic requirement of this LCO is that $F_Q(Z)$ remain within appropriate limits to prevent fuel damage. This requirement will remain in the TS as an LCO. The detail of the limits that vary with each specific core reload will be located in the COLR. The detail of the specific relationships was determined to not be needed in TS during formulation of the TS improvement program that resulted in this detail being relocated to documents under licensee control for creation of the improved TS of NUREG-1431. Therefore, these proposed changes are consistent with the guidance of NUREG-1431, Revision 4 and are acceptable.

LCO 3.2.2.1 Action a. is modified from:

a. For RAOC operation with Specification 4.2.2.1.2.b not being satisfied or for base load operation with Specification 4.2.2.1.4.b not being satisfied:

To:

a. With Specification 4.2.2.1.2.b not being satisfied:

LCO 3.2.2.1 Action a. is modified by removing the text associated with RAOC and base load operation since these power distribution control methods will no longer be used. The modified action statement is more generic and also applies to Dominion's RPDC methods; therefore, the changes are acceptable.

Existing LCO 3.2.2.1 Action b. states:

b. For RAOC operation with Specification 4.2.2.1.2.c not being satisfied, one of the following ACTIONS shall be taken:

   (1) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by at least 1% AFD for each percent $F_Q(Z)$ exceeds its limits. Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

   (2) Verify that the requirements of Specification 4.2.2.1.3 for base load operation are satisfied and enter base load operation.

   Where it is necessary to calculate the percent that $F_Q(Z)$ exceeds the limits for item (1) above, it shall be calculated as the maximum percent over the core height (Z), consistent with Specification 4.2.2.1.2.f, that $F_Q(Z)$ exceeds its limits by the following expression: [equation not shown]

In LCO 3.2.2.1 Action b., the text "For RAOC operation ..." is removed from the beginning of the action statement. This is acceptable because RAOC will no longer be used. The "W" in the word "with" is capitalized and the action continues to apply when $F_Q^M(Z)$ does not meet the equilibrium limits. Additionally, existing required actions in LCO 3.2.2.1 Action b(1) is split among new sub-actions b(1)a, b(1)b, b(1)c, b(1)d, b(1)e, and b(1)f. The equation for determining the percent by which $F_Q(Z)$ exceeds its limits in LCO 3.2.2.1 Action b. is relocated to the TS bases. This is acceptable because the detail of the specific relationships was determined to not be needed in the TSs during formulation of the TS improvement program that resulted in this detail being relocated to documents under licensee control for creation of the improved TS of NUREG-1431. The change is consistent with the NUREG-1431, Revision 4 guidance, and is therefore, acceptable.

LCO 3.2.2.1 Action b(1). would be revised as follows:

a. Within 4 hours, control the AFD to within the new reduced AFD limits specified in the COLR that restores $F_Q(Z)$ to within its limits, and

b. Reduce the THERMAL POWER by the amount specified in the COLR that restores $F_Q(Z)$ to within its limits within 4 hours, and

c. Reduce the Power Range Neutron Flux - High Trip Setpoints by ≥ 1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

d. Reduce the Overpower ΔT Trip Setpoints by ≥1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

e. Within 8 hours, reset the AFD Alarm Setpoints to the modified limits, and

f. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION b(1)(b) above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits.

The licensee states in Reference 1 that the proposed changes in TS LCO 3.2.2.1 Action b. will "...incorporate a modified version of the interim actions identified in NSAL-09-5, Rev. 1 [Reference 23], in the event that SR 4.2.2.1.2.c is not satisfied." Additionally, the licensee asserts the following:

> This approach was determined by Dominion analysis to most appropriately address the issues in NSAL-09-5, Rev. 1 for MPS3. The allowable operating space that applies to TS 3.2.2.1 - ACTION, step b, is relocated to the COLR. A new table, entitled "Required Operating Space Reductions for $F_Q(Z)$ Exceeding its Non-Equilibrium Limits," will be added to the COLR to quantify the required THERMAL POWER and AFD limits associated with different amounts of non-equilibrium $F_Q(Z)$ margin improvement (1%, 2%, etc.). If TS 3.2.2.1 - ACTION, step b is entered, the operating space as defined in the new COLR table will ensure that sufficient margin exists. Including the numerical specification of the operating space in the COLR provides greater assurance that the recommended actions are acceptable without regard to the specific power distribution control methodology. The proposed change can be applied under either the Westinghouse (RAOC) or Dominion (RPDC) power distribution control methodologies for a given reload cycle.

For Action b(1)a, the former action to control AFD to within the new AFD limits was originally proposed with a completion time of 15 minutes and was to be retained as new Action b(1)a except that with the change, the limit relationship would now be specified in the COLR. The relocation of the relationship to the COLR is allowed since the Dominion RPDC strategy uses a calculated AFD (delta-I) band that varies with each core reload. This is acceptable because the detail of the specific relationships was determined to not be needed in the TSs during formulation of the TS improvement program that resulted in this detail being relocated to documents under licensee control for creation of the improved TS of NUREG-1431. Therefore, this is consistent with the NUREG-1431, Revision 4 guidance. The original completion time of 15 minutes for control of AFD to within limits was proposed to be changed in the licensee's response to RAIs (Reference 4).

Action b(1)b is a newly proposed action to reduce thermal power to the amount specified in the COLR that restores $F_Q(Z)$ to within its limits. The licensee determined that reduction of thermal power to the amount specified in the COLR that restores $F_Q(Z)$ to within its limits appropriately addresses the issues in NSAL-09-5, Rev. 1. However, a 15 minute time is allowed for reduction of $F_Q(Z)$ elsewhere in the CTS (e.g. LCO 3.2.2.1, Action a(1)). Proposed Actions b(1) differ from associated with proposed Action a(1). The proposed Actions b(1) are for $F_Q(Z)$ exceeding its non-equilibrium limits and the proposed Actions a(1) are for $F_Q(Z)$ exceeding its equilibrium limits. The NRC staff issued RAI #1 (Reference 26) dated February 24, 2016, requesting additional justification for the 4-hour completion time proposed for LCO 3.2.2.1 Action b(1)(b).

The licensee responded via letter dated March 23, 2016 (Reference 4), stating that the augmentation of the $F_Q(Z)$ by the cycle dependent function is mathematically equivalent to the $F_Q^W(Z)$ nomenclature described in NUREG-1431 for LCO 3.2.1B, Action B and its associated technical basis. DNC responded that the augmentation of $F_Q(Z)$ by the cycle dependent function was mathematically equivalent to the $F_Q^W(Z)$ nomenclature in NUREG-1431 for LCO 3.2.1B, Action B and its basis. DNC supplemented its response by letter dated May 2, 2016, (Reference 6). In the supplement, the licensee provided the below technical justification for the use of the 4-hour completion time instead of the 15-minute completion time:

> The technical justification for a 4-hour completion time for Action b to LCO 3.2.2.1 Action b(1), instead of the 15-minute completion time in LCO 3.2.2.1 Action a(1), can be explained through a comparison of the different scenarios under which the LCOs are entered:
>
> Action a(1) of LCO 3.2.2.1 is entered when surveillance requirement 4.2.2.1.2.b is not met. This surveillance requirement is to address an active violation of $F_Q(Z)$ limits. When measured $F_Q(Z)$ is above its limit, a 15-minute action time is appropriate to return FQ(Z) within the limit as quickly as possible.
>
> In contrast, Action b(1) is entered when surveillance requirement 4.2.2.1.2.c is not met. This surveillance requirement is to address the condition when the non-equilibrium (or transient) $F_Q(Z)$ limit has not been met. In this case, measured $F_Q(Z)$ is not currently above its limit but could exceed its limit if a normal operation transient occurs. A 4-hour completion time is appropriate because a normal operation transient would occur based upon fission product (Xe) time scales and 4 hours is sufficient time to restrict [AFD] limits and thermal power so that core peaking factors are not exceeded.
>
> In addition, reducing power and controlling/reducing AFD to be within new limits (and any resultant actions such as insertion of control rods) within a 15-minute time frame could lead to the initiation of a normal operation transient and make it

more likely that core peaking factors could be violated. A 4-hour completion time allows for deliberate operator actions to minimize the initiation of a normal operation transient.

The NRC staff reviewed the proposed changes and determined that the licensee's justification for the 4-hour completion time for proposed Actions b(1)a and b(1)b is acceptable. If a power reduction is necessary as proposed in new Action b(1)b then proposed Actions b(1)c and b(1)d reasonably follow to reduce the power range neutron flux – high trip setpoint and overpower ΔT trip setpoint respectively. It is reasonable to reduce these RPS trip setpoints to the new allowable power level so that automatic trip can occur at that lower level if necessary to protect the fuel. For proposed Action b(1)d, the completion time of 72 hours to reduce thermal power agrees with the completion time of 72 hours allowed in CTS action a(1) for evaluation of $F_Q(Z)$. The requirement to reduce the power range neutron flux high trip setpoints is additionally equivalent to that in CTS action a(1); however, only a 4-hour completion time is allowed versus the 72 hours proposed.

The NRC staff issued RAI #2 dated February 24, 2016 (Reference 26) requesting additional technical justification for the 72-hour completion time in new proposed LCO 3.2.2.1 Action b(1)(c). In its response (Reference 4), as supplemented by letter dated May 2, 2016, (Reference 6), the licensee stated that a 72-hour completion time is appropriate for this action because of the very low probability of a severe accident occurring during this time as opposed to a normal operational transient and because Action b(1)a (AFD limit reduction) and Action b(1)b (thermal power reduction) will be performed under a 4-hour completion time which reduces possible initial conditions that form the starting point for a severe accident. Additionally, minimizing or reducing possible initial conditions that form the starting point for a severe accident increases the likelihood that achievable power shapes that could occur during a severe accident have already been considered in the safety analysis calculation. The NRC staff has reviewed the proposed changes and the licensee's RAI responses and finds the 72-hour completion time for proposed LCO 3.2.2.1 Action b(1)c is acceptable.

Proposed Action b(1)e relocates part of the existing Action b(1) to reduce the AFD alarm setpoint within 8 hours. The proposed change provides for administrative relocation of the applicable requirement and is editorial in nature; the NRC staff finds this proposed change acceptable.

Proposed Action b(1)(f) is a restorative action to correct the cause of the out of limit condition prior to increasing thermal power. This action applies when heat flux hot channel factor non-equilibrium limits are exceeded versus the equilibrium limits of Action a. The proposed change is a restorative action and consistent with the CTS; the NRC staff finds this proposed change acceptable.

Existing LCO 3.2.2.1 Action b(2) is deleted because it applies only to base load operation which will no longer be used. The proposed change provides is administrative in nature; the NRC staff finds it acceptable.

The proposed changes to LCO 3.2.2.1, Actions a. and Action b. are acceptable and consistent with the corresponding LCO 3.2.1B, Action B, of NUREG-1431, Rev. 4, guidance. Therefore, the NRC staff finds that LCO 3.2.2.1 as proposed will meet the requirements of 10 CFR 50.36(c)(2).

3.2.2.4 SR 4.2.2.1.2, SR 4.2.2.1.3, SR 4.2.2.1.4, and SR 4.2.2.1.5

SR 4.2.2.1.2 is modified by deleting the words "For RAOC operation" from the beginning of the paragraph. This is an administrative change due to MPS3 changing to Dominion methodology therefore this is acceptable.

SR 4.2.2.1.2.c is modified by inserting "Verify $F_Q^M(Z)$ satisfies the non-equilibrium limits specified in the COLR." The added text more adequately and succinctly describes the intent of the SR. The existing text is removed and the relationship equations are relocated to the TS bases. The licensee states that the limit equations and associated description are already described in the COLR making the information in the TS redundant. The NRC staff finds that these proposed changes adequately retain the requirements of the existing SR while more closely aligning with NUREG-1431, Rev. 4, and are therefore, acceptable.

SR 4.2.2.1.2.e is modified by replacing the existing language with "Compliance with the non-equilibrium limits shall be conservatively accounted for during intervals between $F_Q^M(Z)$ measurements by performing either of the following." Westinghouse NSAL-15-1 identified certain conditions in which the required actions associated with this SR may not provide assurance that the non-equilibrium $F_Q(Z)$ LCO limit will be met between the performance of the required surveillance intervals. The NRC staff finds that the proposed change is applicable to MPS3 SR 4.2.2.1.2.e, conservatively addresses the deficiency identified in Westinghouse NSAL-15-1, and is therefore, acceptable.

SR 4.2.2.1.2.f is modified: (1) so that the definition of the non-applicable core regions are moved to the bases, and (2) to narrow the regions to satisfy the Westinghouse guidance of 06-IC-03. The licensee stated that "The intent of 06-1C-03 was to inform utilities that it is probable for the minimum F_Q margin to occur near the top or bottom of the core. In response to the information in 06-1C-03, the proposed change increases the core plane regions for which the limits apply and reduces the "not applicable" portion at the top and bottom to 8 percent. Reload methodology confirms transient F_Q margin over the entire core height and the 'not applicable' region can be adjusted larger or smaller, as necessary, to ensure peak transient F_Q is not in this region. Moving the 'not applicable' region to the Bases allows this adjustment. Relocating the defined core planar regions (where SRs 4.2.2.1.2.c and 4.2.2.1.2.e are applicable) to the bases matches the guidance in NUREG-1431, Rev. 4, and is therefore, acceptable.

SRs 4.2.2.1.3 and 4.2.2.1.4 will be deleted since they only applied to base load operations which will no longer be used. SR 4.2.2.1.5 will be revised to delete the reference to SR 4.2.2.1.4. The NRC staff has reviewed the proposed changes to the above SRs and determined that they are consistent with the change to Dominion's RPDC methods and sufficient to continue to comply with 10 CFR 50.36(c)(3). Therefore, the proposed changes to the SRs are acceptable.

3.2.3 TS 3.2.3.1.b(5) – Deletion of the Value of the RCS Flow Rate Measurement Uncertainty

TS 3.2.3.1.b(5) currently states that "The measured value of RCS total flow rate shall be used since uncertainties of 2.4 percent for flow measurement have been included in Specification 3.2.3.1.a". The licensee proposed to delete the value (2.4 percent) of the flow measurement uncertainty from TS 3.2.3.1.b(5).

The licensee justified the proposed deletion by stating that the value of the RCS flow uncertainty does not meet any of the following categories defined in 10 CFR 50.36 for items required to be in TS. TS categories are: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) Initial notification; or (8) written reports. Paragraph (c)(2)(ii) of 10 CFR 50.36 states that a TS must be established for each item meeting one or more of the following criteria:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

RCS total flow rate, which is already part of TS 3.2.3.1.a, meets Criterion 2. In the FSAR Chapter 15 analyses, the RCS total flow rate is used as an input value based on the nominal conditions (for statistical events) with consideration of the RCS flow uncertainty, while the thermal design flow rate is used as an input value based on the deterministic conditions (nominally biased for uncertainty). The difference between the RCS total flow rate and the thermal design flow rate bounds the RCS flow uncertainty. This approach is consistent with reload evaluation methods of VEP-FRD-42-A (Reference 7) and WCAP-9272-P-A (Reference 17), as well as the statistical DNB methodologies of VEP-NE-2-A (Reference 11) and WCAP-11397-P-A (Reference 28).

The NRC staff finds that the value of the flow measurement uncertainty does not meet: (1) criterion 1 discussed above since it is not used to detect radiological releases; (2) criterion 2 since it is not used as an initial condition in a design basis accident and or transient analysis; (3) criterion 3 since is not part of the primary success path used for mitigating a design basis accident or transient; and (4) criterion 4 since it is not a component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Therefore, NRC staff determines that the proposed deletion of the value (2.4 percent) of the flow measurement uncertainty from the TSs meets the requirements of Paragraph (c)(2)(ii) in 10 CFR 50.36. Also, the NRC staff finds that the proposed TS deletion is consistent with NUREG-1431 (Reference 15), since the STSs do not contain values of the RCS flow uncertainties.

Based on the above discussion, the NRC staff determines that the proposed deletion of the value (2.4 percent) of the flow measurement uncertainty from the TS meets the 10 CFR 50.36 requirements and is consistent with Westinghouse STS documented in NUREG-1431. Therefore, the NRC staff determined that the proposed TS change is acceptable.

3.2.4 SR 4.2.3.1.3.a - RCS Flow Rate Measurement Time

Current SR 4.2.3.1.3.a states that "Verifying by precision heat balance that the RCS total flow rate is > 363,200 gallons per minute (gpm) and greater than or equal to the limit specified in the COLR within 24 hours after reaching 90 percent of RATED THERMAL POWER after each fuel loading, and..." The proposed changes to SR 4.2.3.1.3.a relax the time requirement to perform the precision heat balance from 24 hours to 7 days after reaching 90 percent of the rated thermal power. The licensee justified the completion time relaxation by stating that 7 days will allow the establishment of stable operating conditions, installation of the test equipment, performance of the test, and completion of the analysis. The proposed 7-day period is consistent with that approved for SPS. The NRC SER (Reference 27) approving the flow rate measurement time for SPS provided a basis for acceptance stating that the 7-day period is adequate to complete an accurate flow rate measurement, which increases the quality of confirmation available to the licensee that it is operating its plants within safety analysis limits. Both SPS and MPS3 are owned and operated by Dominion; since their nuclear steam supply systems were designed and manufactured by Westinghouse, the differences in the design features of 4-RCS loops for MPS3 and 3-RCS loops for SPS will not affect the basis for establishing the 7-day flow rate measurement time. The NRC staff determines that the basis of its acceptance of the flow rate measurement time for SPS is applicable to MPS3, and thus, the proposed measurement time of 7 days is acceptable for MPS3.

3.2.5 TS 6.9.1.6.a - Core Operating Limits Report (COLR)

TS 6.9.1.6.a lists 12 cycle-specific core operating limits to be included in the COLR. The proposed changes to Items 7 and 8 in the core operating limits list of TS 6.9.1.6.a are to delete the "target band, and APL^{ND}" from Item 7 and "K(z), W(z), and APL^{ND}" from Item 8. The proposed deletions are terminologies associated with specific power distribution control methodology and base load operation. The revised Items 7 and 8 are as follows:

7. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1.1.
8. Heat Flux Hot Channel Factor Limits for Specification 3/4.2.2.1.

These proposed changes are acceptable, since they are consistent with the changes described in Section 2.2 of Reference 21 and the NRC staff's evaluation discussed in above Section 3.2.2 of the SER, where the proposed changes to TS 3/4.2.1.1 and 3/4.2.2.1 eliminate base load operation and relocate from TS the equations and terminology for either RAOC or RPDC transient multiplication factors.

The proposed change to Item 9 is also acceptable, since the change, renumbering TS 3/4.2.3 to TS 3/4.2.3.1 to align the associated TS subsection with that specified in the COLR, is an editorial change and does not change the technical content of the TS requirements.

3.2.6 TS 6.9.1.6.b – NRC-Approved Methods Referenced in COLR

3.2.6.1 Statement Preceding TS 6.9.1.6.b Reference List

The licensee proposed changes to the statement preceding the reference list in TS 6.9.1.6.b. The added wording requires that the cycle-specific COLR identify the full reference citation of the TRs used to support that cycle, including citations of the report number, title, revision, date, and any supplements. The NRC staff determines that the change is acceptable, since it is consistent with guidance of GL 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," (Reference 14). Specifically, the GL states, in part, that the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC. The number, title, and date of the TRs documenting the methodologies used for determining the core operating limits should be identified. The TS change is also consistent with the "Reviewer's Note" in TS 5.6.3.b of NUREG-1431 (Reference 15), which provides clarification that the reference methodologies used for a reload core should be specifically identified in the cycle-specific COLR.

3.2.6.2 Addition of Approved Dominion Methodologies to the TS 6.9.1.6.b Reference List

MPS3 TS 6.9.1.6.b currently states that: "The analytical methods used to determine core operating limits shall be those previously reviewed and approved by the NRC." This TS also provides a list of NRC-approved analytical methods for MPS3. In order to use the analytical methods described in Reference 16, the licensee proposed to add the NRC-approved methodologies documented in TR VEP-FRD-42, TR VEP-NE-1, TR VEP-NE-2, and TR DOM-NAF-2 to the list in TS 6.9.1.6.b as References 20 through 23, respectively. The revised Items 20 through 23 are as follows:

20. VEP-FRD-42-A, "Reload Nuclear Design Methodology." Methodology for Specifications

    - 2.1.1 Reactor Core Safety Limits
    - 3.1.1.1.1 SHUTDOWN MARGIN – MODE 1 and 2
    - 3.1.1.1.2 SHUTDOWN MARGIN – MODES 3, 4 and 5 Loops Filled
    - 3.1.1.2 SHUTDOWN MARGIN – Cold Shutdown – Loops Not Filled
    - 3.1.1.3 Moderator Temperature Coefficient
    - 3.1.3.5 Shutdown Rod Insertion Limit
    - 3.1.3.6 Control Rod Insertion Limits
    - 3.2.2.1 Heat Flux Hot Channel Factor
    - 3.2.3.1 Nuclear Enthalpy Rise Hot Channel Factor
    - 3.3.5 Shutdown Margin Monitor
    - 3.9.1.1 REFUELING Boron Concentration

21. VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications." Methodology for Specifications:

    - 3.2.1.1 AXIAL FLUX DIFFERENCE
    - 3.2.2.1 Heat Flux Hot Channel Factor

22. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology." Methodology for Specifications:

    - 3.2.3.1 RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
    - 3.2.5 DNB Parameters

23. DOM-NAF-2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code." Methodology for Specifications:

    - 3.2.3.1 RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
    - 3.2.5 DNB Parameters

As discussed in Sections 3.1 above, the added TRs are acceptable for use in the analysis supporting licensing applications for MPS3. Therefore, the NRC staff determines that TS 6.9.1.6.b with the added TRs is acceptable.

TRs (a) and (b) below also document the methodologies that are acceptable for use in the reload analysis at MPS3.

(a) DOM-NAF-1-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations."
(b) VEP-FRD-41-P-A, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code."

Dominion does not include the above two TRs in TS 6.9.1.6.b. In response to NRC RAI-18 (SRXB) (Reference 2) regarding the adequacy of Dominion's approach, Dominion indicated that its approach of not including TR (a) and TR (b) above in TS 6.9.1.6.b is consistent with the application of Dominion's approved reload methods for North Anna (TS 5.6.5.b) and Surry (TS 6.2.C). Also, the MPS3 currently does not contain the Westinghouse equivalent references to above TR (a) and TR (b).

In addition, Dominion indicated (in its response to RAI-18 (SRXB), Reference 2) that the added Reference 10, VEP-FRD-42-A, "Reload Nuclear Design Methodology", contains in Section 2.2 the methodology (discussed in TR (a), DOM-NAF-1-P-A) for calculating reload core physical parameters. Sections 2.1.3 and 3.3 contains the methodology (discussed in TR (b), VEP-FRD-41-P-A) for use of RETRAN in the reload analysis. Section 2.3 and Appendix B of the added Reference 10, VEP-FRD-42-A, discusses the process by which either analytical models or methods could achieve approved status for use in Dominion's reload methodology. The process is based on the following NRC requirements and industry guidance:

- GL 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications" (Reference 14)
- NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation" (Reference 31)
- Regulatory Guide (RG) 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments" (endorses NEI 96-07, Rev. 1) (Reference 32)
- GL 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses" (Reference 33)
- 10 CFR 50.59, "Changes, tests and experiments." and in particular, 10 CFR 50.59(c)(2)(viii) states, in part, that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would 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.

Based on the above discussion, the NRC staff finds that: (1) the approach of not including TR (a) and TR (b) listed above is consistent with the application of Dominion's approved reload methods to North Anna (TS 5.6.5.b) and Surry (TS 6.2.C); (2) MPS3 currently does not contain the Westinghouse equivalent references to the above TR (a) and TR (b); and (3) the Dominion existing process of changes to approved methodologies is based on the NRC and industry guidance, including the 10 CFR 50.59(c)(2)(viii) requirements. Therefore, the NRC staff determines that the licensee's approach of not including TR DOM-NAF-1-P-A and TR VEP-FRD-41-P-A is acceptable.

3.2.6.3 Changes of the TS 6.9.1.6.b Reference List

The licensee proposed various changes to the TS 6.9.1.6.b Reference List. The NRC staff has reviewed the changes and provides its evaluation as follows:

3.2.6.3.1 Changes for Readability Improvement

The licensee proposed to reformat the specifications listed under each reference in TS 6.9.1.6.b using bullets to improve readability. Also, it made minor changes to these specifications to reflect conformance to the usage that is appropriate for either Westinghouse or Dominion references. Since the proposed changes are editorial and do not change the technical content of the TS requirements, the NRC staff determines the proposed changes are acceptable.

3.2.6.3.2 Modifications to Reference 1 of TS 6.9.1.6.b

The licensee proposed Reference 1 (WCAP-9272-P-A) of TS 6.9.1.6.b that uses TS 2.1.1 in the list to substitute TS 2.1.1.1, "Departure from Nuclear Boiling Ratio", and 2.1.1.2, "Peak Fuel Centerline Temperature". The proposed change is to align with Item 1 in TS 6.9.1.6.a for which the reference is applicable. In addition, the licensee proposed to add TS 3.2.5, "DNB Parameter", and 3.3.5, "Shutdown Margin Monitor", to the list since both TSs did not previously appear in Reference 1 of TS 6.9.1.6.b. The NRC staff finds that the proposed changes provide additional clarification for the appropriate use of the cited Reference 1, and therefore, are acceptable.

3.2.6.3.3 Retaining of References 1 and 4 in TS 6.9.1.6.b

The licensee proposed to retain the following References 1 and 4 in TS 6.9.1.6.b:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," (W Proprietary). Methodology for Specifications:

    - 2.1.1 Reactor Core Safety Limits
    - 3.1.1.1.1 SHUTDOWN MARGIN – MODE 1 and 2
    - 3.1.1.1.2 SHUTDOWN MARGIN – MODES 3, 4 and 5 Loops Filled
    - 3.1.1.2 SHUTDOWN MARGIN – Cold Shutdown – Loops Not Filled
    - 3.1.1.3 Moderator Temperature Coefficient
    - 3.1.3.5 Shutdown Rod Insertion Limit
    - 3.1.3.6 Control Rod Insertion Limits
    - 3.2.1.1 AXIAL FLUX DIFFERENCE
    - 3.2.2.1 Heat Flux Hot Channel Factor
    - 3.2.3.1 RCS Total Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
    - 3.9.1.1 REFUELING Boron Concentration
    - 3.2.5 DNB Parameters
    - 3.3.5 Shutdown Margin Monitor

4. WCAP-10216-P-A-R1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," (W Proprietary). (Methodology for Specifications 3.2.1.1--AXIAL FLUX DIFFERENCE and 3.2.2.1--Heat Flux Hot Channel Factor)

Reference items 1 and 4 of TS 6.9.1.6.b pertain to Westinghouse reload methods. The proposed list of reference methodologies in TS 6.9.1.6.b would contain both Westinghouse and Dominion references. As stated by the licensee in the LAR, the two reference items listed above "are being retained in TS 6.9.1.6.b since these methodologies are applicable to Westinghouse for establishing core operating limits and may be used for a specific core during the transition to Dominion methods." Additionally, the licensee states "the references used in a cycle-specific COLR will be a subset of the TS 6.9.1.6.b methodologies that are applicable to the specific reload cycle. If Westinghouse reload methods are used, then Westinghouse reload methods shall be listed in the cycle-specific COLR. If Dominion methods are used, then Dominion reload methods shall be listed."

As stated in the licensee's LAR, Section 3.2 (page 10 of 20), "the proposed changes are structured in a manner that is independent of specific power distribution control methodology (RAOC or RPDC). Relocating the specific equations associated with either the Westinghouse or Dominion power distribution control methodologies to the Bases is consistent with the guidance contained in NUREG-1431, Rev. 4 (Reference 3).

The NRC staff notes that the proposed changes to the affected LCOs and associated SRs involved replacement of terminology that applies only to the current Westinghouse-based methodology (RAOC or RAOC) with language that is not associated with a particular methodology. Additionally, specific equations containing terms that are unique to either methodology would be relocated to the Bases. This makes the final resultant TS LCOs independent of the particular methodology, so that either the Westinghouse RAOC or Dominion RPDC may be used. Also, this TS structure is not dependent upon any required or implied timeframe in which to transition to Dominion methods. Therefore, no issues or unintended consequence would be created from a postulated delay in transition to Dominion methods.

The listed references document the NRC-approved Westinghouse methodologies that are applicable to MPS3 for establishing core operating limits and are used for a specific core during the transition to Dominion methods; therefore, the NRC staff determines that the proposed TS 6.9.1.6.b is acceptable during the transition to Dominion methods.

Additionally, in Reference Item 4, the specification number 3.2.1 would be revised to 3.2.1.1; the text "[Relaxed Axial Offset Control]" is removed, specification number 3.2.2 is revised to 3.2.2.1, and the text "[W(z) surveillance requirements for FQ Methodology]" is removed. The proposed change to Reference item 4 is acceptable, since the revision provides additional clarity by adding the correct specification to which the methodology applies.

3.2.6.3.4 Retaining of References 4 through 10, 16, 17, 18 and 19 in TS 6.9.1.6.b

The licensee proposed to retain and revise for clarity of specification the following References in TS 6.9.1.6.b.: 4, 5, 6, 7, 8, 9, 10, 16, 17, 18 and 19:

4. WCAP-10216-P-A-R1A, "RELAXATION OF CONSTANT AXIAL OFFSETCONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," (W Proprietary). (Methodology for Specifications 3.2.1.1--AXIAL FLUXDIFFERENCE and 3.2.2.1--Heat Flux Hot Channel Factor)

5. WCAP-12945-P-A, "CODE QUALIFICATION DOCUMENT FOR BEST ESTIMATE LOCA ANALYSIS," (W Proprietary). (Methodology for Specification 3.2.2.1--Heat Flux Hot Channel Factor.)

6. WCAP-16009-P-A, "REALISTIC LARGE-BREAK LOCA EVALUATION METHODOLOGY USING THE AUTOMATED STATISTICAL TREATMENT OF UNCERTAINTY METHOD (ASTRUM)," (W Proprietary). (Methodology for Specification 3.2.2.1--Heat Flux Hot Channel Factor.)

7. WCAP-11946, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Millstone Nuclear Power Station Unit 3," (W Proprietary). Methodology for Specification:

• 3.1.1.3 - Moderator Temperature Coefficient

8. WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

9. WCAP-10079-P-A, "NOTRUMP - A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

10. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

16. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis." Methodology for Specification:

    • 3.2.2.1 - Heat Flux Hot Channel Factor

17. WCAP-10054-P-A, Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model." Methodology for Specification:

    • 3.2.2.1 - Heat Flux Hot Channel Factor

18. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature DT Trip Functions," (Westinghouse Proprietary Class 2) (Methodology for Specifications 2.2.1 - Overtemperature ΔT and Overpower ΔT Setpoints.)

19. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

In Items 5, 6, 8, 9, and 10, and 19 specification number 3.2.2 is replaced with 3.2.2.1. In Item 7 the text: "Methodology for Specification:" and bulleted item "3.1.1.3 – Moderator Temperature Coefficient" are added to the end of the reference. For Items 15 and 16, the specification 3.2.2.1 is included as the applicable methodology. For Item 17, the applicable specification for 2.2.1, "Overtemperature ΔT and Overpower ΔT Setpoints" is added to the end of the reference.

The licensee states in its LAR that references 5, 6, 7, 8, 9, 10, 16, 17, 18 and 19, are Westinghouse TRs which document methodologies that are independent of the scope for nuclear safety analysis and core design methods which Dominion is applying. Dominion asserts that these references remain applicable to either Westinghouse or Dominion reload core design methods.

The NRC finds that the above references document Westinghouse methodologies are unrelated to the proposed application of Dominion reload methods to MPS3. Therefore, these references remain applicable for use with either Westinghouse or Dominion reload methods. The proposed changes to the above items are also acceptable, since the revisions provide additional clarity by adding the correct specification to which the methodology applies, is editorial in nature, and does not involve any changes to the technical content of the TS requirements.

3.2.6.3.5 Deletion of References from TS 6.9.1.6.b Reference List

The licensee proposed to delete the following References 2, 3, 11, 12, 13, 14 and 15 from TS 6.9.1.6.b and replace each with the word “deleted”. :

2. T. M. Anderson to K. Kniel (Chief of Core performance Branch, NRC), January 31, 1980 – Attachment: Operation and Safety Analysis Aspects of Improved Load Follow Package.

3. NUREG-800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, “Nuclear Design, July 1981 Branch Technical Position CPB-4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Revision 2, July 1981.

11. Letter from V. L. Rooney (USNRC) to J. F. Opeka, "Safety Evaluation for Topical Report, NUSCO-152, Addendum 4, 'Physics Methodology for PWR Reload Design,' TAC No. M91815," July 18, 1995.

12. Letter from E. J. Mroczka to the USNRC, "Proposed Changes to Technical Specifications, Cycle 4 Reload Submittal - Boron Dilution Analysis," B13678, December 4, 1990.

13. Letter from D. H. Jaffe (USNRC) to E. J. Mroczka, "Issuance of Amendment (TAC No. 77924)," March 11, 1991.

14. Letter from M. H. Brothers to the USNRC, "Proposed Revision to Technical Specification, SHUTDOWN MARGIN Requirements and Shutdown Margin Monitor OPERABILITY for MODES 3, 4, and 5 (PTSCR 3-16-97), B16447, May 9, 1997.

15. Letter from J. W. Anderson (USNRC) to M. L. Bowling (NNECO), "Issuance of Amendment - Millstone Nuclear Power Station, Unit No.3 (TAC No. M98699)," October 21, 1998.

The above references were proposed to be deleted from TS 6.9.1.6.b, because they do not describe a methodology that establishes core operating limits. The licensee clarified the proposed deletion by stating that a methodology in the COLR reference list is to satisfy two conditions: (1) the methodology is used to determine core operating limits; and (2) it has been previously approved by the NRC. The NRC staff finds that the licensee's clarification is consistent with the guidance discussed in GL 88-16, “Removal of Cycle-Specific Parameter Limits from Technical Specifications” (Reference 14), which states, in part, that: "generally, the methodology for determining cycle-specific parameter limits is documented in an NRC-approved Topical Report or in a plant-specific submittal." Additionally, the NRC staff agrees with the licensee's reasoning that these items contain guidance or other information but are not analytical methods used in determining core operating limits, and therefore, may be removed. Therefore, the NRC determines the proposed deletion of the cited references is acceptable.

3.5 NRC Staff Conclusion

The NRC staff has reviewed the licensee's submittals and supporting documentation and finds that the proposed use of Dominion nuclear core design and safety analysis methods discussed in Section 3.1 and the proposed TS changes discussed in Section 3.2 are acceptable for use in licensing applications at MPS3.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified on May 2, 2016, of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* (FR) on June 13, 2016 (81 FR 38226). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission 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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Dominion Nuclear Connecticut, Inc. (DNC) Letter, Mark D. Sartain to NRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03," May 8, 2015, (ADAMS Accession No. ML15134A244).

2. DNC Letter, Mark D. Sartain to NRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03 (CAC No. MF6521)," January 28, 2016, (ADAMS Accession No. ML16034A216).

3. DNC Letter, Mark D. Sartain to NRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03 (CAC No. MF6521)," February 25, 2016, (ADAMS Accession No. ML16057A812).

4. DNC Letter, Mark D. Sartain to NRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03 (CAC No. MF6521)," March 23, 2016, (ADAMS Accession No. ML16088A140).

5. DNC Letter with two Attachments, Daniel G. Stoddard to NRC, "Dominion Nuclear Connecticut Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03 (CAC No. MF6521)," March 29, 2016, (ADAMS Accession No. ML16095A233).

6. DNC Letter, Mark D. Sartain to NRC, "Dominion Nuclear Connecticut, Inc. Millstone Power Station Unit 3, Supplement to Response to Request for Additional Information Regarding License Amendment Request to Adopt Dominion Core Design and Safety Analysis Methods and to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1, NSAL-15-1, and 06-IC-03 (CAC No. MF6521)," May 2, 2016, (ADAMS Accession No. ML16130A563).

7. Topical Report, VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003, (ADAMS Accession No. ML15313A149).

8. Topical Report, VEP-NE-1, Rev. 0.1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," August 2003, (ADAMS Accession No. ML15313A154).

9. Topical Report, DOM-NAF-1-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations," June 2003, (ADAMS Accession No. ML031690108).

10. Topical Report, VEP-FRD-41-P-A, Rev. 0.2, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code," March 2015, (ADAMS Accession No. ML15313A141).

11. Topical Report, VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987 (Proprietary), (ADAMS Accession No. ML101330527).

12. Fleet Report, DOM-NAF-2-P-A, Rev. 0.3, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," Appendix D & Attachments, September 2014 (Proprietary), (ADAMS Accession Nos. ML14294A517 for the Fleet Report and ML14294A516 for the SER approving the Report).

13. NUREG-0800, Standard Review Plan, Section 16, Revision 3.0, "Technical Specification," dated March 2010, (ADAMS Accession No. ML100351425).

14. Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988, (ADAMS Accession No. ML031200485).

15. NUREG-1431, Revision 4, Vol. 1 and 2, "Standard Technical Specifications – Westinghouse Plants," (ADAMS Accession No. ML12100A222).

16. Attachment 4 to Reference 1, "Application of Dominion Nuclear Core Design and Safety Analysis Methods," (ADAMS Accession No. ML15134A244).

17. Westinghouse Topical Report WCAP-9272-P-A, 'Westinghouse Reload Safety Evaluation Methodology," (Proprietary), March 1978, (ADAMS Accession No. ML051390150).

18. Dominion Nuclear Connecticut, Inc. (DNC) e-mail, Wanda D. Craft to NRC, "RAI-12 Supplement RAI Response Adopt Core Design and Safety Analysis Methods – MPS3," dated March 3, 2016, (ADAMS Accession No. ML16069A250).

19. Attachment 5 to Reference 1, "RETRAN Benchmarking Information," (ADAMS Accession No. ML15134A244).

20. Attachment 6 to Reference 1, "Development of Statistical Design Limits," (ADAMS Accession No. ML15134A244).

21. Attachment 1 to Reference 1, "Evaluation of Technical Specifications Changes", and Attachment 2 to Reference 1, "Marked-up Technical Specifications pages," (ADAMS Accession No. ML15134A244).

22. USNRC Information Notice (IN) 2014-01, "NRC Information Notice 2014-01: Fuel Safety Limit Calculations Inputs Were Inconsistent With NRC-Approved Fuel Design," dated February 21, 2014, (ADAMS Accession No. ML13325A966)

23. Westinghouse Nuclear Safety Advisory Letter, NSAL-09-5, Rev. 1, "Relaxed Axial Offset Control FQ Technical Specification Actions," September 23, 2009.

24 Westinghouse Notice "06-IC-03, F_Q and F_{xy} Surveillance Zone Issue," February 21, 2006.

25. Westinghouse Nuclear Safety Advisory Letter, NSAL-15-1, "Heat Flux Hot Channel Factor Technical Specification Surveillance," February 3, 2015, (ADAMS Accession No. ML15105A102).

26. E-mail from R.Guzman to W.Craft, Re: Request for Additional Information - LAR to Adopt Dominion Core Design and Safety Analysis Methods (MF6251). Millstone Power Station, Unit 3 RAI #2 dated February 24, 2016, (ADAMS Accession No, ML16055A530).

27. Letter from K. Cotton (USNRC) to D. A. Heacock (Dominion), "Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Request for Technical Specification Revisions Related to the Core Operating Limits Report (TAC Nos. ME2591 and ME2592)," dated October 19, 2010 (ADAMS Accession No. ML102530115); corrected by Letter from K. Cotton (USNRC) to D. A. Heacock (Dominion), "Surry Power Station, Unit Nos. 1 and 2, Correction to Amendments Regarding Technical Specification Revisions Related to the Core Operating Limits Report (TAC Nos. ME2591 and ME2592)," October 21, 2010, (ADAMS Accession No. ML102930565).

28. WCAP-11397-A, "Revised Thermal Design Procedure" Proprietary, (ADAMS Accession No. ML080630437).

29. Attachment 5 of DNC Letter 07-0450, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3 License Amendment Request Stretch Power Uprate," July 13, 2007, (ADAMS Accession No. ML072000386).

30. Letter from S. A. Richard (NRC) to G. L. 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, (TAC No. MA4311). (ADAMS Accession No. ML010470342).

31. Nuclear Energy Institute (NEI) 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," (ADAMS Accession No. ML003771157).

32. USNRC Regulatory Guide (RG) 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments" (endorses NEI 96-07, Rev. 1), (ADAMS Accession No. ML003759710).

33. GL 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses" (NUDOCS Accession Number 9906210103).

34. Electric Power Research Institute (EPRI), "VIPRE-01: A Thermal Hydraulic Code for Reactor Cores," NP-2511-CCM-A, Revision 4, Palo Alto, CA, June 2007 (ADAMS Accession Nos. ML102090545, ML102090544, ML102090543, and ML102070202; Non-Publicly Available).

Principal Contributors: Summer Sun
Pete Snyder
Mathew M. Panicker

Date: July 28, 2016

July 28, 2016

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT ADOPTING DOMINION CORE DESIGN AND SAFETY ANALYSIS METHODS AND ADDRESSING THE ISSUES IDENTIFIED IN THREE WESTINGHOUSE COMMUNICATION DOCUMENTS (CAC NO. MF6251)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 268 to Renewed Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3. This amendment is in response to your application dated May 8, 2015, as supplemented by letters dated January 28, February 25, March 23, March 29, and May 2, 2016.

The amendment revises the Technical Specifications (TSs) to (1) allow the use of Dominion nuclear safety and reload core design methods; (2) allow the use of applicable departure from nucleate boiling ratio design limits for VIPRE-D; (3) update the approved reference methodologies cited in TS 6.9.1.6.b; (4) remove the base load mode of operation that is not a feature of the Dominion Relaxed Power Distribution Control power distribution control methodology; and (5) address the issues identified in Westinghouse Nuclear Safety Advisory Letter (NSAL-09-5), Rev. 1, NSAL-15-1, and Westinghouse Communication 06-IC-03. Additionally, the amendment relocates certain equations, supporting descriptions and surveillance requirements from the TSs to licensee-controlled documents.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures:
1. Amendment No. 268 to NPF-49
2. Safety Evaluation

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OFFICE	NRR/DORL/LPLI-1/PM	NRR/DORL/LPLI-1/LA	DSS/SRXB/BC	DSS/STSB/BC
NAME	RGuzman	KGoldstein	EOesterle*	AKlein*
DATE	5/23/2016	5/27/2016	5/02/2016	5/11/2016
OFFICE	DSS/SNPB/BC	OGC	DORL/LPLI-1/BC	DORL/LPLI-1/PM
NAME	JDean*	BHarris	TTate	RGuzman
DATE	5/10/2016	7/18/2016	7/28/2016	7/28/2016

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