NMC
Committed to Nuclear Excellence

Kewaunee Nuclear Power Plant
N490, State Highway 42
Kewaunee, WI 54216-9511
920-388-2560

Operated by
Nuclear Management Company, LLC

October 12, 2000

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Ladies and Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Wisconsin Public Service Corporation Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3

Reference: 1) Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC), dated November 9, 1988, transmitting WPSRSEM-NP-A, Revision 2

This letter transmits topical report WPSRSEM-NP, Revision 3, "Reload Safety Evaluation Methods for Application to Kewaunee." This report supersedes WPSRSEM-NP-A, Revision 2, which was transmitted in Reference 1 and is intended to be applicable to all WPSC reload cycles after and including Cycle 25, presently scheduled to commence in the fall of 2001.

The enclosed report incorporates all of the reload safety evaluation and plant safety analysis changes that have occurred since the Reference 1 transmittal or that are being proposed for application to Kewaunee after and including Cycle 25. Revision 3 of the report reflects:

- Editorial changes, including corrections to the limiting directions of core physics parameters and clarification of the definition of core physics parameters.
- Changes made to incorporate the CONTEMPT code for containment analysis. CONTEMPT is currently described for this purpose in the Kewaunee USAR.
- The adoption of the GOTHIC code for containment analysis.
- Changes in Reload Safety Evaluation Methods due to Large Break Loss of Coolant Accident Upper Plenum Injection Analysis.
- The adoption of RETRAN- 3D for use in the 2D mode for system analysis.
- The extension of the VIPRE-01 code to reflect changes in fuel design.

Attachment 1 to this letter provides the description, safety evaluation, significant hazards determination and environmental considerations for the proposed changes.

Attachment 2 to this letter provides the revised Reload Safety Evaluation Methods Topical Report, WPSRSEM-NP, Revision 3.

Attachment 3 to this letter provides a summary of the DYNODE-P to RETRAN- 3D Benchmark Report.

Attachment 4 to this letter provides a summary of the CONTEMPT, COCO, and GOTHIC Containment Analysis Benchmark Report.

Should the staff have any further questions concerning the need, accuracy, applicability or justification of the proposed changes, please feel free to contact T. J. Webb (920) 388-8537, or J. T. Holly (920) 388-8296 of my staff.

In accordance with the requirements of 10 CFR 50.30(b), this submittal has been signed and notarized. A complete copy of this submittal has been transmitted to the State of Wisconsin as required by 10 CFR 50.91(b)(1).

Sincerely,

Kenneth H. Weinhauer
Assistant Site Vice President

JTH/eld

Attach.

cc - US NRC, Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

Subscribed and Sworn to
Before Me This 12th Day
of October 2000

Jeanne M. Ferries
Notary Public, State of Wisconsin

My Commission Expires:
June 8, 2003

ATTACHMENT 1

Letter from K. H. Weinhauer (NMC)

To

Document Control Desk (NRC)

Dated

October 12, 2000

WPSRSEM-NP, Revision 3

Description of Changes

Safety Evaluation

Significant Hazards Determination

Environmental Considerations

Introduction

The topical report WPSRSEM-NP, "Reload Safety Evaluation Methods for Application to Kewaunee", Revision 3, is being submitted for review and approval. This report supersedes WPSRSEM-NP-A, Revision 2 (reference A1) and is intended to be applicable to all Wisconsin Public Service Corporation (WPSC) Reload Cycles after and including Cycle 25, presently scheduled to commence in the fall of 2001.

Revision 3 of WPSRSEM-NP incorporates all of the reload safety evaluation and plant safety analysis methods changes that have occurred since the Reference 1 transmittal or that are being proposed for application to Kewaunee after and including Cycle 25.

The Revision 3 changes in WPSRSEM-NP will be identified by gray shading for additions or alterations of existing text and the presence of a □ symbol for deletions. The changes fall into three major categories:

- Analysis Methods Changes – Methodology
- Analysis Methods Changes – Software Tools
- Editorial Changes

This attachment will present a Description of Changes, a Safety Evaluation, a Significant Hazards Determination, and an Environmental Considerations section for each major category of change.

Analysis Methods Changes – Methodology

Description of Changes

a) Containment Safety Analysis Methods

The CONTEMPT Containment Analysis Methods have been added to Appendix F of the report. The analysis methods described in Appendix F are identical to the analysis methods described in Section 14.C and Section 14.2.5 of the KNPP USAR (reference A2). The CONTEMPT code has been part of KNPP design basis since original licensing (reference A2, Section 14. C).

WPSRSEM-NP, Revision 3 is changed to document the addition of containment analysis methods that can be applied to the design basis accidents, Main Steam Line Break (MSLB), and Large Break Loss of Coolant Accident (LBLOCA). The containment analysis methods are used to calculate the containment thermal hydraulic response during these design basis accidents.

WPSRSEM-NP-A, Revision 2, did not include containment analysis methods. At the time of WPSRSEM–NP–A, Revision 2 the containment integrity analysis of record was the analysis that had been performed by the NSSS vendor, Westinghouse, using the COCO and CONTEMPT codes.

Several significant plant issues; containment safeguards timing assumptions for the LBLOCA, main steam isolation valve closure time, and boric acid storage tank boron concentration reduction and removal from the safety injection system were the important drivers for the development of containment analysis methods. WPSC successfully applied the CONTEMPT containment analysis methodologies to address these plant issues (see reference A2 Section 14.C, reference A2 Section 14.2.5, and references A3 and A4).

These newly included Containment Safety Analysis Methods are described in Sections 3.14 (Main Steam Line Break), and Appendix F (CONTEMPT) of WPSRSEM-NP, Revision 3.

There are no additional reload safety core physics parameter evaluations that result from the addition of containment analysis methods.

This methodology has been previously reviewed and approved by the NRC. It is being included in the Safety Analysis Methodology Report, Revision 3, for completeness. Therefore, this is an administrative change and does not require prior NRC approval.

b) Large Break Loss of Coolant Analysis Upper Plenum Injection Reload Safety Evaluation Methods

WPSRSEM-NP, Revision 3 Reload Safety Evaluation Methods for LBLOCA is changed to reflect the implementation of Upper Plenum Injection (UPI) safety analysis methods for Kewaunee. The LBLOCA UPI safety analyses (the actual system analysis of the LBLOCA accident) are performed by Westinghouse using Westinghouse methods. These methods for LBLOCA were implemented to achieve better (more accurate) representation of the plant response to a design basis LBLOCA accident. WPSC implemented the upgraded ECCS Analysis Methods reflecting UPI for the LBLOCA design basis accident analysis via letter notification in reference A5. In WPSRSEM-NP, revision 3 the changes described are changes to the Reload Safety Evaluation Methods. These methods are within WPSC scope of responsibility for this accident.

WPSRSEM–NP–A, Revision 2, does not include LBLOCA UPI Safety Analysis Methods since at that time the analysis of record for LBLOCA was a non-UPI LOCA methodology. Therefore LBLOCA UPI Reload Safety Evaluation Methods were not required at that time. The changes in Reload Safety Evaluation Methods are necessary because of the more detailed and accurate modeling of the reactor core attained with the LBLOCA UPI Safety Analysis Methods relative to the non-UPI methods. These revised Reload Safety Evaluation Methods are implemented for the current fuel cycle (reference A6).

The Reload Safety Evaluation Methods changes for the LBLOCA accident are incorporated into Sections 2.12 through 2.15 (General Physics Methods) and in Section 3.16 (Loss of Coolant Accident) of WPSRSEM-NP, Revision 3.

This methodology has been previously reviewed and approved by the NRC. It is being included in the Safety Analysis Methodology Report, Revision 3, for completeness. Therefore, this is an administrative change and does not require prior NRC approval.

c) High Thermal Performance Departure from Nucleate Boiling Ratio Correlation

WPSRSEM-NP, Revision 3 is being changed to incorporate the High Thermal Performance (HTP) Departure from Nucleate Boiling Ratio (DNBR) correlation with a 1.14 MDNBR safety limit. WPSRSEM-NP-A, Revision 2 includes the Westinghouse W-3 DNBR correlation with a 1.30 MDNBR safety limit which was the appropriate DNBR correlation and DNBR limit for the fuel design in use at Kewaunee at that time.

In the current operating cycle, Cycle 24, WPSC is using a Siemens Power Corp. (SPC) fuel design that includes the HTP grid/spacer design. The HTP grid/spacer design was adopted to improve fuel thermal performance. The HTP grid/spacer design has a corresponding HTP DNBR correlation and DNBR limit. The HTP correlation has been qualified for use with the VIPRE-01 computer code. WPSC implemented the VIPRE-01 code for reload safety evaluation and safety analysis as it applies to the Kewaunee Siemens fuel design in the current operating cycle. The NRC reviewed and approved WPSC's use of the HTP DNBR correlation with the VIPRE-01 code in reference A7.

The HTP DNBR correlation change is included in the discussion of VIPRE-01 fuel thermal hydraulic analysis methods in Appendix C of WPSRSEM-NP, Revision 3.

This methodology has been previously reviewed and approved by the NRC. It is being included in the Safety Analysis Methodology Report, Revision 3, for completeness. Therefore, this is an administrative change and does not require prior NRC approval.

Analysis Methods Changes – Software Tools

Description of Changes

a) RETRAN- 3D

WPSRSEM-NP, Revision 3 documents the use of RETRAN-3D (the latest version of the RETRAN code) as an acceptable software tool for performing the safety analysis of integrated plant systems. WPSRSEM-NP-A, Revision 2 documents the use of RETRAN-02, which was the RETRAN code version in use at that time. RETRAN- 3D provides improved plant system analysis capabilities and is consistent with current industry best-practice safety analysis methods. For example, RETRAN 3D enables the modeling of the reactor core in three dimensions and it is more capable of accurately modeling the plant in shutdown conditions. RETRAN- 3D is an accepted industry standard for plant safety analyses as evidenced by the utility group submittal of RETRAN- 3D for generic approval (reference A8).

WPSC will run RETRAN-3D in a RETRAN-02 mode. RETRAN-3D and RETRAN-02 have been extensively compared as part of the RETRAN-3D verification and validation process. These comparisons are documented in Volume 4 (Applications Manual) of reference A9. Based on these verification and validation analyses it is concluded that when RETRAN-3D is run in a RETRAN-02 mode, which is the way WPSC intends to run RETRAN-3D, the two codes are essentially the same. Advanced RETRAN-3D features such as three dimensional core models will be implemented in a planned manner when appropriate for the issue being resolved and when appropriate review and approval are completed.

Benchmark analyses have been performed comparing the WPSC safety analysis computer codes, RETRAN-3D and DYNODE-P. The benchmark analyses were performed for the Kewaunee design basis Non-LOCA accidents using current WPSC Non LOCA safety analysis methodologies with similar analysis input assumptions. These benchmark analyses are documented in a separate benchmark report (reference A10) that is summarized in Attachment 3. From the benchmark analyses we conclude that DYNODE-P and RETRAN-3D yield similar results when applied to KNPP Design Basis Non-LOCA transients. The two computer codes are considered interchangeable computer codes for Non- LOCA accident analysis applications.

The use of RETRAN-3D for plant safety analysis is documented in Appendix E of WPSRSEM-NP, Revision 3. The safety analysis results for all of the transient events except the rod ejection, main steam line break and startup of an inactive loop accidents (which are DYNODE-P results) are generated by RETRAN-3D and are presented in Appendix G of WPSRSEM-NP, Revision 3.

b) GOTHIC

WPSRSEM-NP, Revision 3 documents the use of the GOTHIC computer code (reference A11) as an acceptable software tool for containment safety analysis. WPSRSEM-NP-A, Revision 3 documents both CONTEMPT and GOTHIC as acceptable software tools for containment safety analysis. CONTEMPT is the software tool that is integral to the current analysis of record containment analysis methods (see containment safety analysis methods discussion above). WPSC is developing the GOTHIC computer code for application to containment integrity design basis analyses. GOTHIC is a more versatile analysis tool than CONTEMPT. WPSC is including it in the safety analysis methodologies for increased analysis flexibility and capability in the containment thermal hydraulic analysis area. GOTHIC would be applied in the same way that CONTEMPT is currently applied, i.e. for containment integrity analysis of the design basis MSLB and LBLOCA accidents.

Benchmark analyses have been performed between the CONTEMPT and GOTHIC codes for the design basis accidents, MSLB and LBLOCA. These benchmark analyses are documented in a separate report (reference A12) that is summarized in Attachment 4. The benchmark analyses demonstrate consistency and similarity between the two computer codes when applied to design basis containment integrity analyses and with similar input assumptions.

The changes describing the use of the GOTHIC computer code are presented in Appendix H of WPSRSEM–NP, Revision 3.

Safety Evaluation

Significant Hazards Determination

The proposed changes were reviewed in accordance with the provisions of 10CFR 50.92 to show no significant hazards exist. The proposed changes will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

Analysis methods are not accident initiators, therefore, changes in analysis methods will not increase significantly the probability of occurrence or the consequences of an accident previously evaluated.

The changed analysis methods are conservative and conform to industry standards for analysis methods that are applied to design basis safety analyses. Benchmark analyses have demonstrated good agreement between the changed analysis methods and the current analysis of record (AOR) methods. The safety analysis results using the changed analysis methods are shown to satisfy all applicable design and safety analysis acceptance criteria. The demonstrated adherence to safety analysis acceptance criteria precludes new challenges to components and systems that could adversely affect the ability of existing components and systems to mitigate the consequences of any accident or adversely affect the integrity of any fission product barrier.

Analysis methods changes will not impact plant equipment important to safety. Equipment important to safety will continue to operate within its design capabilities. The analysis methods changes also do not affect the plant configuration or the overall plant performance capabilities.

Therefore, the changes will not increase probability or the consequences of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change is a change to the analysis methods, which are applied to Kewaunee. Analysis methods are not accident initiators. The changed analysis methods are applied to the accidents that are the established design basis accidents for Kewaunee. Analysis methods changes will not impact plant equipment important to safety. Equipment important to safety will continue to operate within its design capabilities. The analysis methods changes also do not affect the plant configuration or the overall plant performance capabilities.

As demonstrated by the benchmark reports the methodologies provide a more accurate but still conservative representation of expected plant response following a design basis accident. Since the new methodologies are conservative with respect to actual expected plant response the changes will not create the possibility of an accident of a different type than any previously evaluated.

3) Involve a significant reduction in the margin of safety.

The proposed changes are changes to the analysis methods, which are applied to Kewaunee design basis safety analyses. The revised analysis methods have been verified through benchmark analyses against the current Analysis of Record methods. The analysis methods are conservative and appropriate for application to Kewaunee design basis analyses. Safety analysis acceptance criteria are satisfied when the changed analysis methods are applied to the Kewaunee design basis safety analyses. Demonstrated adherence to safety analysis acceptance criteria using the new analysis methods assures that Technical Specification limits will be satisfied during operation with the changed analysis methods.

Therefore, the margin of safety as defined in the basis of any Technical Specification will not be reduced significantly because of these changes.

Environmental Considerations

WPSC has determined that the changes incorporated in Revision 3 of WPSRSEM-NP involve no significant hazard considerations 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. Accordingly, this revision meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(11). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this revision.

Editorial Changes

Description of Changes

The following editorial changes have been incorporated into the Reload Safety Analysis Methods Report using 10CFR50.59. They are provided here for completeness.

a) Corrections to Limiting Directions and Clarification of Core Physics Parameters

WPSRSEM-NP, Revision 3 incorporates the noted corrections for the accident analyses listed below.

The affected accident analyses and parameters are:

1. Uncontrolled Rod Withdrawal From a Subcritical Condition (Section 3.1.5):
   Changed: Limiting direction effective delayed neutron fraction.

2. Uncontrolled Rod Withdrawal at Power (Section 3.2.5):
   Changed: Limiting direction effective delayed neutron fraction

3. Chemical and Volume Control System Malfunction (formerly titled, "Uncontrolled Boron Dilution") (Section 3.5.5):
   Added: Critical boron concentration at power conditions
   Changed: Boron reactivity coefficient at power conditions
   Added: Scram reactivity curve at power conditions
   Added: Critical boron concentration startup conditions

4. Excessive Heat Removal Due to Feedwater System Malfunction (formerly titled, "Feedwater System Malfunction") (Section 3.7.5)
   Added: Scram reactivity curve

5. Excessive Load Increase (Section 3.8.5):
   Added: Scram reactivity curve

6. Loss of Reactor Coolant Flow-Locked Rotor (Section 3.12.5):
   Changed: Limiting direction effective delayed neutron fraction
   Changed: Number of fuel pins above $F_{\Delta H}$ (DNBR = 1.3) to $C_{F\Delta H}$, locked rotor

7. Fuel Handling Accident (Section 3.13.5):
   Changed: Nuclear enthalpy rise hot channel factor is now used instead of F_Q

It should be noted that the limiting directions and clarified physics parameters for the above-listed accidents were correctly considered in the safety analyses of the individual transients.

Section 2.7, Item 3, of the NRC SER (Reference A1) states the acceptance conditions for the Doppler and moderator temperature coefficients (in the case of the Controlled Rod Ejection Accident) specified in the topical report were in error. In the SER, the staff stated the bounding values for these coefficients should be the <u>most</u> negative bounding value.

The methodology of WPSRSEM-NP-A, Revision 2 compares the cycle specific Doppler and moderator temperature coefficients (compensated for reliability) to the bounding values for these coefficients. The appropriate bounding values for the Doppler and moderator temperature coefficients (for the Control Rod Ejection Accident) are the <u>least</u> negative values.

Selection of the least negative values will minimize the negative feedback during the rod ejection and will maximize the consequences of the accident. Our analysis has, and will continue to use, this conservative approach.

The changes incorporating these corrections appear in the sections listed above of WPSRSEM-NP, Revision 3.

<u>b) Other</u>

All of the WPSC safety analysis results have been removed from section 3.0 and placed in Appendix G. This change facilitates the documentation of future results' changes.

Appendix C no longer has detailed discussions on sensitivity studies performed with the VIPRE –01 code. These sensitivity studies were necessary for the initial WPSC VIPRE-01 fuel thermal hydraulic methodology development but are now considered as reference material and not applicable to the overall purpose of this topical report. Appendix C describes the current WPSC core thermal hydraulic methodology.

Appendix F of WPSRSEM-NP-A, Revision 2 that described best estimate safety analysis methods has been eliminated. This section is no longer applicable and is not necessary to present in this topical report.

References

A1. Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC), dated November 9, 1988, transmitting WPSRSEM-NP-A, Revision 2

A2. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Updated Safety Analysis Report, Revision 15

A3. Letter from W. O. Long NRC) to M. L. Marchi (WPSC) dated 4-15-98 transmitting "Basis Change to Technical Specifications for facility operating license DPR-43 Kewaunee Nuclear Power Plant TAC No. M98638

A4. Letter from R. L. Laufer (NRC) to M. L. Marchi (WPSC) dated 3-28-95 transmitting amendment no. 116 to facility operating license DPR –43 Kewaunee Nuclear Power Plant TAC no. M91072

A5. Letter from M. L. Marchi (WPSC) to Document Control Desk (NRC), dated 6/4/97, "Implementation of ECCS Analysis Methods Reflecting Upper Plenum Injection

A6. Kewaunee Nuclear Power Plant, Reload Safety Evaluation Report, Cycle 24, dated April 2000

A7. Letter from R. J. Laufer (NRC) to M. L. Marchi (WPSC) dated 12-30-97, "Safety Evaluation for the High Thermal Performance (HTP) Departure from Nucleate Boiling (DNB) Correlation and the Associated 1.14 MDNBR Safety Limit for Kewaunee Fuel"

A8. Letter from G. B. Swindlehurst (Duke Power Co.) to Mr. Timothy E. Collins (NRC) dated 7-8-98 transmitting Request for NRC Review of RETRAN –3D

A9. Paulson, M.P., et al., RETRAN-3D: A Program for Transient Thermal- Hydraulic Analysis of Complex Fluid Flow Systems EPRI NP –7450, October 1996

A10. Kewaunee Nuclear Plant Report dated April, 2000 entitled "Kewaunee Nuclear Plant DYNODE-P RETRAN 3D Benchmark"

A11. George, Thomas L., et al., GOTHIC version 6.0 Containment Analysis Package, December 1997 EPRI RP4444-1

A12. WCAP-15427, "Development and Qualification of a GOTHIC Containment Evaluation Model for the Kewaunee Nuclear Power Plant," R. Ofstun, July 2000

ATTACHMENT 2

Letter from K. H. Weinhauer (NMC)

To

Document Control Desk (NRC)

Dated

October 12, 2000

WPSRSEM-NP Revision 3

WPSRSEM-NP
REVISION 3

RELOAD SAFETY EVALUATION METHODS

FOR

APPLICATION TO KEWAUNEE

Prepared By: J. L. Voskuil
Nuclear Fuel Engineer

Date: 9-13-2000

Reviewed By: John T. Holly
Analytical Engineering Group Leader

Date: 9-14-00

Reviewed By: D. J. Warner
Director – Nuclear Fuel

Date: 9-14-00

ABSTRACT

This document is an updated Topical Report describing the Wisconsin Public Service Corporation (WPSC) reload safety evaluation and plant safety analysis methods for application to Kewaunee.

The report addresses the methods for the calculation of cycle specific physics parameters and their comparison to the bounding values used in the safety analyses.

□

TABLE OF CONTENTS

1.0 INTRODUCTION 1

2.0 GENERAL PHYSICS METHODS 3

2.1 Moderator Temperature Reactivity Coefficient, α_M 3
2.2 Power Reactivity Coefficient, α_P 3
2.3 Doppler Reactivity Coefficient, α_D 3
2.4 Boron Reactivity Coefficient, α_B 4
2.5 Shutdown Margin, SDM 4
2.6 Scram Reactivity Curve, $\Delta\rho_{scram(t)}$ 4
2.7 Nuclear Heat Flux Hot Channel Factor, F_Q 5
2.8 Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}$ 5
2.9 Effective Delayed Neutron Fraction, β_{eff} 5
2.10 Prompt Neutron Lifetime, ℓ^* 5
2.11 Fuel Temperature, T_f 6
2.12 Maximum Assembly Average Peaking Factor, P_{HA} 6
2.13 Axial Offset at 100% Power, AO_{HFP} 6
2.14 Maximum Core Average Power in Low Power Assemblies, P_{LP} 6
2.15 Maximum 95/95 Power for the Hot Rod, $P_{95/95}$ 7
2.16 Critical Boron Concentration, C_B 7
2.17 Fuel Rod Census, $C_{F\Delta H}$ 7

3.0 SAFETY EVALUATION METHODS 8

3.1 Uncontrolled RCCA Withdrawal from a Sub-Critical Condition 10
3.2 Uncontrolled RCCA Withdrawal at Power 14
3.3 Control Rod Misalignment 18
3.4 Control Rod Drop 20
3.5 Chemical and Volume Control System Malfunction 22
3.6 Startup of an Inactive Coolant Loop 26
3.7 Excessive Heat Removal Due to Feedwater System Malfunction 28
3.8 Excessive Load Increase 31
3.9 Loss of External Load 34
3.10 Loss of Normal Feedwater Flow 37
3.11 Loss of Reactor Coolant Flow - Pump Trip 39
3.12 Loss of Reactor Coolant Flow - Locked Rotor 42
3.13 Fuel Handling Accident 46
3.14 Main Steam Line Break 48
3.15 Control Rod Ejection 53
3.16 Loss of Coolant Accident 57
3.17 Power Distribution Control Verification 61

TABLE OF CONTENTS (continued)

4.0 REFERENCES 62

Appendix A - Computer Program Overview 63
Appendix B - NSSS Simulation, DYNODE-P 64
Appendix C - Core and Fuel Thermal-Hydraulic Analysis, VIPRE-01 68
Appendix D - Fuel Thermal-Hydraulic Analysis, TOODEE-2 71
Appendix E - NSSS Simulation, RETRAN-3D 73
Appendix F - Containment Thermal-Hydraulic Analysis, CONTEMPT LT-28 78
Appendix G - WPS Safety Analysis Results 79
Appendix H – Containment Thermal-Hydraulic Analysis, GOTHIC 204

□

1.0 INTRODUCTION

This report addresses the methods for the calculation of Kewaunee cycle specific physics parameters and their comparison to and establishment of the bounding values used in the Kewaunee □safety analyses. This document is an update of previous submittals, which were reviewed and approved by the NRC [1]. Calculation of core physics parameters for the purpose of performing reload safety evaluations requires an intimate knowledge of the safety analyses to which cycle specific comparisons are to be made. Specifically, one must understand the manner in which the bounding physics parameters have been used in each of the analyses and the conservatism inherent in the values chosen. Wisconsin Public Service (WPS) has acquired this understanding with over twenty years of experience in performing independent reload core design and safety analysis.

A brief description of the general physics calculational procedures is provided in Section 2. The general physics calculational procedures are consistent with the reactor physics methods of reference 2, which has been reviewed and approved by the NRC [3]. The specific detailed calculations are controlled by written procedures in accordance with the WPSC Operational Quality Assurance Program (OQAP). General methods are described for each of the key physics parameters of interest in reload safety evaluations.

A general description is given in Section 3 of each of the accidents analyzed in the Kewaunee Updated Safety Analysis Report (USAR) [4] that is sensitive to core physics parameters and is therefore of concern for a reload evaluation. □

In addition, cycle specific physics calculations and their comparisons to the safety analyses are described for each accident along with the specific applications of the reliability factors [2].

Section 3 discusses each applicable Kewaunee USAR transient as follows:

- A description of the transient is given describing the physical phenomena involved.
- A description of the analysis methodology to be applied, and the assumptions used in the analysis are given.
- The results (key parameter results figures) of the safety analysis are identified and are included in Appendix G.
- The reload specific physics parameters to which the transient analysis is sensitive are described and their conservative direction determined. □
- The set of reload specific physics parameters and their associated reliability factors and biases are presented with the corresponding safety analysis parameters in inequality tables that are used to determine whether or not an accident must be reanalyzed to accommodate the characteristics of a specific fuel reload. □

An updated list of bounding safety analyses applicable to Kewaunee is compiled for each Reload Safety Evaluation Report submitted to the NRC [5]. The specific bounding values for each analysis

are provided in the cycle specific Reload Safety Evaluation Report utilizing the most up-to-date analysis methodology.

The computer models applied to Kewaunee were developed in accordance with documented guidelines, which accompany each of the computer codes. The development of the computer models described in this report was controlled by procedures in accordance with the WPS Operational Quality Assurance Program (OQAP). WPSC Quality Programs □periodically audits the control of these computer models. A brief description of the computer models is provided in the appendices to this report as follows.

Appendix A gives an overview of the computer code package that is used to simulate the transients and accidents listed in this report.

Appendix B gives a description of the DYNODE-P computer code that is used to simulate the transient response of the Nuclear Steam Supply System. □

Appendix C gives a description of the VIPRE-01 computer code that is used to simulate the thermal-hydraulic response of the reactor core and hot fuel assembly subchannel. A discussion of the WPS thermal margin methodology is included in this section. □

Appendix D gives a description of the TOODEE 2 computer code that is used to simulate the thermal response of the hot fuel rod and associated coolant channel under transient conditions. A discussion of the WPS fuel thermal response methodology is also included in this appendix.

Appendix E gives a description of the RETRAN-3D computer code that is used to simulate the transient response of the Nuclear Steam Supply System. □

Appendix F provides a description of the CONTEMPT LT-28 computer code that is used to analyze the thermal-hydraulic response of containment to steam line break and loss of coolant accidents.

Appendix G provides representative results for the safety analyses described in this report.

Appendix H provides a description of the GOTHIC computer code that is used to analyze the thermal-hydraulic response of containment to steam line break and loss of coolant accidents.

2.0 GENERAL PHYSICS METHODS

In this section, the general physics calculational methods are described for application to reload safety evaluations for Kewaunee.

□The applications of reliability factors and comparisons to the safety analyses are discussed in Section 3 for each accident considered.

2.1 MODERATOR TEMPERATURE REACTIVITY COEFFICIENT, α_M

Definition: α_M is the change in core reactivity associated with a 1°F change in average moderator temperature at constant average fuel temperature.

Calculations of α_M are performed in three dimensions with the nodal model. □The average moderator temperature is varied while the independent core parameters such as core power level, control rod position and Reactor Coolant System (RCS) boron concentration are held constant. Dependent core parameters such as power distribution and moderator temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics. The average fuel temperature is held constant and no changes in nodal xenon inventory are permitted.

2.2 POWER REACTIVITY COEFFICIENT, α_p

Definition: α_p is the change in core reactivity associated with a 1% (of full power) change in core average power level.

Calculations of αp are performed in three dimensions with the nodal model. □Core power is varied while all other independent parameters such as rod position and RCS boron concentration are held constant. Dependent core parameters such as power distribution, average fuel and moderator temperatures and moderator temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics. No changes in nodal xenon inventory are permitted.

2.3 DOPPLER REACTIVITY COEFFICIENT, α_D

Definition: α_D is the change in core reactivity associated with a 1°F change in core average fuel temperature at constant average RCS moderator temperature.

□

Calculations of α_D are performed in three dimensions with the nodal model. Fuel temperature is varied at selected core power levels while other independent parameters are held constant. Core power distribution is permitted to change as dictated by the core neutronics and thermal hydraulics. Moderator temperature and nodal xenon inventory are not permitted to change.

2.4 BORON REACTIVITY COEFFICIENT, α_B

Definition: α_B is the change in reactivity associated with a 1PPM change in core average soluble boron concentration.

Calculations of α_B are performed in three dimensions with the nodal model. □The core average boron concentration is varied while the independent core parameters such as core power level and control rod position are held constant. Dependent core parameters such as power distribution and moderator temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics. No changes in xenon inventory are permitted.

2.5 SHUTDOWN MARGIN, SDM

Definition: SDM is the amount of reactivity by which the core would be subcritical following a reactor trip, assuming the most reactive control rod is stuck out of the core and no changes in xenon or RCS boron concentration.

Calculations of SDM are performed in three dimensions with the nodal model. □The general calculational sequence is given below.

Case #1 - At power condition with rods at the power dependent insertion limits.

Case #2 - Hot Zero power condition with all rods in except the stuck rod. No changes in xenon or boron are assumed.

Case #3 - Hot Zero power conditions with rods at the position of Case #1.

The dependent core parameters such as power distribution and temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics. All spatial effects and rod insertion allowances are explicitly accounted for in each calculation. The SDM is computed as the change in core reactivity between Case 1 and Case 2. This value is conservatively adjusted using Case 3 and model reliability factors, RF_i, and biases. □ These uncertainty factors are applied to the inserted rod worth, the moderator defect, and the Doppler defect.

2.6 SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}(t)$

Definition: $\Delta\rho_{scram}$ (t) is the rod worth inserted into the core as a function of time after rod release. The most reactive rod is assumed to remain fully withdrawn.

The independent core parameters such as power level, RCS boron concentration and xenon inventory are held constant during the insertion. Neutron flux, a dependent parameter, is assumed to redistribute instantaneously during rod insertion. However, the effects of moderator and Doppler feedback on the scram reactivity shape are not included.

The total scram reactivity insertion is conservatively normalized to the minimum N-1 rod worth reduced by the rod worth reliability factor.
The reactivity dependence on rod position calculated above is converted into a time dependent function using empirical data relating rod position to time after rod release. The empirical data is normalized such that the total time to full rod insertion is equal to or greater than the limits defined by the Technical Specifications [6].

2.7 NUCLEAR HEAT FLUX HOT CHANNEL FACTOR, F_Q

Definition: The maximum local fuel rod linear power density divided by the core average fuel rod linear power density.

Calculations of F_Q are based on three dimensional power distributions obtained with the nodal model □coupled with local peak pin to assembly power ratios obtained from the quarter core PDQ model. Statistical factors defined in Reference 2 are applied to increase the F_Q to a conservative value.

2.8 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Definition: The maximum integral linear power along a fuel rod divided by the core average fuel rod integral power.

Calculations of $F_{\Delta H}$ are based on three dimensional power distributions obtained with the nodal model □coupled with the local peak pin to assembly power ratios obtained from the quarter core PDQ model. □Statistical factors defined in Reference 2 are applied to increase $F_{\Delta H}$ to a conservative value.

2.9 EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

Values for β_i are determined by weighting the delayed neutron fractions from each fissile isotope group by the fission sharing of that isotope group as determined from PDQ. The importance factor I, applied as .97, conservatively accounts for the effects of reduced fast fissioning, increased resonance escape, and decreased fast leakage by the delayed neutrons. β_{eff} is the product of β and I, where

$$\beta = \sum_{i=1}^{6} \beta_i$$

2.10 PROMPT NEUTRON LIFETIME, ℓ^*

ℓ^* is calculated as a function of core exposure from two-dimensional PDQ calculations. This is accomplished by using two group flux weighted PDQ parameters to compute the slowing down time and the thermal diffusion time based on the 1/v nature of the boron absorption cross section.

2.11 FUEL TEMPERATURE, T_f

T_f is calculated as a function of linear heat generation rate. Conditions of maximum fuel densification, low oxide conductivity and low gap conductance are assumed in the analysis.

2.12 MAXIMUM ASSEMBLY AVERAGE PEAKING FACTOR, P_{HA}

The maximum assembly average peaking factor, P_{HA}, is a parameter related specifically to the LOCA analysis. It is defined as the maximum assembly relative power at hot full power, all rods out conditions. Calculations of P_{HA} are based on the three-dimensional nodal model. Statistical factors defined in Reference 2 are applied to increase P_{HA} to a conservative value. The statistical factor applied to $F_{\Delta H}$ is conservatively applied to P_{HA}. This is conservative, since the statistical factor applied to $F_{\Delta H}$ considers not only the uncertainty in the nodal assembly power calculation, but also an additional uncertainty to account for the PDQ pin-to-box calculational uncertainty.

2.13 AXIAL OFFSET AT 100% POWER, AO_{HFP}

The axial offset at 100% power, AO_{HFP}, is a parameter related specifically to the LOCA analysis. It is defined as the nominal best estimate axial offset at hot full power, all rods out conditions. Calculations of AO_{HFP} are based on the three-dimensional nodal model. Since this is a nominal best estimate parameter, no uncertainties are applied to the calculated value. The axial offset is a measure of the core-wide axial power distribution and is calculated as follows:

Axial Offset = (PT - PB)/(PT+ PB); where

PT = Relative power in the top half of the core; and
PB = Relative power in the bottom half of the core

2.14 MAXIMUM CORE AVERAGE POWER IN LOW POWER ASSEMBLIES, P_{LP}

The maximum core average power in low power assemblies, P_{LP} is a parameter related specifically to the LOCA analysis. It is defined as the nominal best estimate relative power in the low power assemblies at hot full power, all rods out conditions. The low power assemblies for the Kewaunee plant are defined as the assemblies in full core locations A-6, A-7, A-8, B-4, B-10, C-3, C-11, D-2, D-12, F-1, F-13, G-1, G-13, H-1, H-13, J-2, J-12, K-3, K-11, L-4, L-10, M-6, M-7, and M-8. Calculations of P_{LP} are based on the three-dimensional nodal model. Since this is a nominal best estimate parameter, no uncertainties are applied to the calculated value. The relative power predicted by the three-dimensional nodal model for each of the 24 low power assemblies is summed and divided by 24 to obtain the value of P_{LP}.

2.15 MAXIMUM 95/95 POWER FOR THE HOT ROD, $P_{95/95}$

The maximum 95/95 power for the hot rod, $P_{95/95}$, is a parameter related specifically to the LOCA analysis. For the purposes of reload core design, it is defined as the peak linear heat rate value below which 95% of the fuel rods fall. The maximum F_O of the 95th percentile rod for the cycle, including all PDC-II related uncertainties, is converted to a peak linear heat rate for this calculation. PDC-II is discussed in Section 3.17.

2.16 CRITICAL BORON CONCENTRATION, C_B

Calculations of C_B are based on three dimensional calculations obtained with the nodal model.

2.17 FUEL ROD CENSUS, $C_{F\Delta H}$

Number of fuel rods at or above a specified FΔH value.

3.0 SAFETY EVALUATION METHODS

This section addresses the evaluation of the cycle specific physics parameters with respect to the bounding values used in the safety analyses □ for each accident or transient by which the determination is made as to whether or not any reanalysis is required. For each accident or transient the following material is described:

a. Description of the Accident – a □ description of the transient event, its causes and consequences.

b. Method of Analysis – a □description of the typical methods employed and discussion of the sensitive physics parameters. Included is a list of the accident or transient acceptance criteria.

c. WPS Safety Analysis Results - a presentation of representative WPS safety analysis results for the most limiting case of each accident type. Results are presented in Appendix G.

d. Cycle Specific Physics Calculations - a description of the specific physics calculations performed each cycle for the purposes of a reload safety evaluation.

e. Reload Safety Evaluation - a description of the comparisons of the cycle specific physics characteristics and the bounding values used in the safety analysis. Specific applications of the model reliability factors and biases, which are determined, as described in Reference 2 are also addressed. Biases and reliability factors are to be applied in the following manner:

- Moderator Temperature Coefficient (α_M)
  Apply in a conservative direction as follows:
  $\alpha_M = \alpha_M$ (MODEL) + $B_M \pm RF_M$
  B_M = Moderator temperature coefficient bias (pcm/°F)
  RF_M = Moderator temperature coefficient reliability factor (pcm/°F)

- Doppler Coefficient (α_D)
  Apply in a conservative direction as follows:
  $\alpha_D = \alpha_D$ (MODEL) * $(1 \pm RF_D)$
  RF_D = Doppler coefficient reliability factor

- Boron Reactivity Coefficient (α_B)
  Apply in a conservative direction as follows:
  $\alpha_B = \alpha_B$ (MODEL) * $(1 \pm RF_B)$
  RF_B = Boron coefficient reliability factor

- Nuclear Heat Flux Hot Channel Factor (F_Q)
  Apply in a conservative direction as follows:
  $F_Q = (F_Q(\text{MODEL})) * (1+RFF_Q) * (1+T)$
  RFF_Q = Nuclear heat flux hot channel factor reliability
  T = Technical Specification Tilt Limit

- Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)
  Apply in a conservative direction as follows:
  $F_{\Delta H} = (F_{\Delta H}(MODEL)) * (1+RFF_{\Delta H}) * (1+T)$
  $RFF_{\Delta H}$ = Nuclear enthalpy rise hot channel factor reliability
  T = Technical Specification Tilt Limit

- Effective Delayed Neutron Fraction (β_{eff})
  Apply in a conservative direction as follows:
  $\beta_{eff} = \beta_{eff}$ (MODEL) * ($1 \pm RF_{\beta}$)
  $RF_{\beta} = \beta_{eff}$ □reliability factor

- Prompt Neutron Lifetime (ℓ^*)
  Apply in a conservative direction as follows:
  $\ell^* = \ell^*$ (MODEL) * ($1 \pm RF_{\ell^*}$)
  RF_{ℓ^*} = prompt neutron lifetime reliability factor

- Scram Reactivity ($\Delta\rho_{scram}(t)$)
  Apply in a conservative direction as follows:
  $\Delta\rho_{scram}(t) = \Delta\rho_{scram}(t)$ (MODEL) * ($1 - RF_R$)
  RF_R = Rod worth □reliability factor

- Rod Worth ($\Delta\rho_R$)
  Apply in a conservative direction as follows:
  $\Delta\rho_R = \Delta\rho_R$(MODEL) * ($1 \pm RF_R$)
  RF_R = Rod worth □reliability factor

- Fuel Temperature (T_f)
  Apply in a conservative direction as follows:
  $T_f = T_f$(MODEL) * ($1 \pm RFT_f$)
  RFT_f = Fuel temperature □reliability factor

- Maximum Assembly Average Peaking Factor, P_{HA}
  Apply in a conservative direction as follows:
  $P_{HA} = P_{HA}$ (MODEL) * ($1 + RFF_{\Delta H}$) * ($1 + T$)
  $RFF_{\Delta H}$ = Nuclear enthalpy rise hot channel factor reliability
  T = Technical Specification Tilt Limit

- Maximum 95/95 Power for the Hot Rod, $P_{95/95}$
  Apply in a conservative direction as follows:
  $P_{95/95} = P_{95/95}$ (MODEL) * ($1 + RFF_O$) * ($1 + T$) * V(Z)
  RFF_O = Nuclear heat flux hot channel factor reliability
  T = Technical Specification Tilt Limit
  V(Z) = PDC-II function (see Section 3.17)

The specific numerical values assigned as the bounding values for each accident for purposes of performing the Kewaunee reload safety evaluations are presented in the cycle specific Reload Safety Evaluation Report [5].

If an accident or transient requires reanalysis because any one of the cycle specific physics parameters exceeds the current bounding value, the reanalysis will be performed utilizing the safety analysis methodology as described herein for that specific event. □ If the parameter exceeded involves a Technical Specification Limit the reanalysis will be submitted to the NRC in support of the appropriate Technical Specification amendment.

3.1 UNCONTROLLED RCCA WITHDRAWAL FROM A SUB-CRITICAL CONDITION

3.1.1 DESCRIPTION OF THE ACCIDENT

An uncontrolled addition of reactivity due to an uncontrolled withdrawal of a Rod Cluster Control Assembly (RCCA) results in a power excursion. The nuclear power response is characterized by a very fast rise terminated by the reactivity effect of the negative fuel temperature coefficient. After the initial power burst, the reactor power is limited by this inherent feedback and the accident is terminated by a reactor trip, typically on high nuclear power. Due to the small amount of energy released to the core coolant, pressure and temperature excursions are minimal during this accident.

3.1.2 METHOD OF ANALYSIS

The uncontrolled RCCA withdrawal from a sub-critical condition is analyzed using a dynamic simulation incorporating point neutron kinetics, including delayed neutrons and decay heat; fuel, clad, and gap heat conduction; and channel coolant thermal-hydraulics. The reactivity effects due to moderator and fuel temperature effects, as well as that due to control rod insertion after trip, are included.

The core is assumed initially to be at Hot Zero Power, (HZP). Power is supplied to the RCCA drive mechanisms such that no more than two banks may be withdrawn simultaneously. The maximum reactivity insertion due to the rods is therefore conservatively assumed as that due to two banks of maximum worth moving simultaneously at maximum speed through the region of highest differential worth.

The magnitude of the power peak reached during the transient is strongly dependent upon the Doppler reactivity coefficient for a given rate of reactivity insertion. A value conservatively small in absolute magnitude, which generally occurs at Beginning of Cycle (BOC), is assumed for the accident analysis. The magnitude of the power spike is relatively insensitive to the value of moderator temperature reactivity coefficient chosen. A positive moderator temperature coefficient is used to maximize the calculated consequences of the accident.

In calculating reactivity due to control rod insertion by reactor trip, the most adverse combination of instrument and setpoint errors and time delays is assumed. The power range - low range trip setpoint is assumed to be 10% (of full power) above its

nominal value. The most reactive rod is assumed to stick in the fully withdrawn position when the trip signal is actuated.

As long as the reactivity insertion remains small compared to β_{eff}, the total delayed neutron yield, the shortest reactor period during the transient will remain large compared to ℓ^*, the prompt neutron lifetime. In this case, the transient core power response is relatively insensitive to the value of ℓ^* and is determined predominately by the yields and decay constants of the delayed neutron precursors. The transient power response is sensitive to ℓ^* in cases where ultra-conservative reactivity insertion rates, in which prompt criticality is achieved, are assumed. The postulated initial core pressure and temperature are conservatively taken as the minimum and maximum, respectively, consistent with the assumed rod and power configurations.

The □ acceptance criteria for this accident are:

a. The maximum pressures in the reactor coolant and main stream systems do not exceed 110% of design values.

b. Cladding integrity is maintained by limiting the minimum departure from nucleate boiling ratio (MDNBR) to a value greater than the MDNBR limit.

3.1.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results for this accident are presented in Figures 3.1-1 through 3.1-5 (see Appendix G). □

3.1.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed as a function of power level □ at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC for the Hot Zero Power (HZP), unrodded, no xenon core condition. This produces the least negative moderator coefficient due to both the unrodded and high critical boron concentration core conditions.

c. MAXIMUM REACTIVITY INSERTION RATE, $\Delta\rho/\Delta t$

In order to compare with the reactivity insertion rate assumed by the safety analysis for uncontrolled rod withdrawal transients, the assumption is made that two banks of highest worth will be withdrawn simultaneously at maximum speed. This value requires two components. First, the maximum withdrawal speed is required in inches per second. □The second component is the maximum differential reactivity insertion per inch for the two maximum worth rod banks moving in 100% overlap. This has been obtained by first calculating the two banks, which have the maximum worths, and then moving these two banks simultaneously at HZP conditions in an area of highest differential worth. These calculations are performed at both BOC and EOC. Finally the reactivity insertion rate is divided by the minimum β_{eff} determined according to (e) of this subsection to yield the maximum reactivity insertion rate in dollars.

d. SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}$ (t)

Calculations of the scram reactivity are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the zero power condition. Making the following assumptions generates a conservatively slow scram curve:

1. The integral of the scram curve is based on an initial rod position at or below the zero power insertion limits specified in the Technical Specification[6].

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant negative reactivity insertion.

3. The xenon distribution is that which causes the minimum N-1 rod worth.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

e. EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC.

f. PROMPT NEUTRON LIFETIME, ℓ^*

The value of ℓ^* is calculated in accordance with the general procedures given in Section 2. Cycle specific calculations are performed at BOC and EOC.

3.1.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above is adjusted to include the model reliability factors, RF_i □ and biases. These adjusted values are the cycle specific parameters that are then compared to the bounding values assumed in the safety analysis.

The cycle specific parameters are acceptable if the following inequalities are met:

Cycle Specific Parameters		Safety Analysis Parameters
a. $\alpha_D * (1\text{-}RF_D)$	$\leq$	α_D (least negative bounding value)
b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
c. $\left(\dfrac{\Delta\rho / \Delta t * (1 + RF_R)}{\beta eff * (1 - RF_\beta)}\right)$	$\leq$	$\left(\dfrac{\Delta\rho / \Delta t(\text{bounding})}{\beta eff(\text{maximum})}\right)$
d. $\Delta\rho_{scram}(t) * (1\text{-}RF_R)$	$\geq$	$\Delta\rho_{scram}(t)$ (bounding)
e. $\beta_{eff} * (1+RF_\beta)$	$\leq$	β_{eff} (maximum)
f. $\ell^* * (1\text{-}RF_{\ell^*})$	$\geq$	ℓ^* (minimum)

□

3.2 UNCONTROLLED RCCA WITHDRAWAL AT POWER

3.2.1 DESCRIPTION OF THE ACCIDENT

An uncontrolled RCCA withdrawal at power results in a gradual increase in core power followed by an increase in core heat flux. The resulting mismatch between core power and steam generator heat load results in an increase in reactor coolant temperature and pressure. The reactor core would eventually suffer departure from nucleate boiling if the power excursion were not checked by the reactor protection system. Depending on the initial power level and rate of reactivity insertion, the following reactor trips serve to prevent fuel damage or over-pressurization of the coolant system: power range high flux, over temperature/power ΔT and high pressurizer pressure. For the more rapid rates of reactivity insertion, the maximum power reached during the transient will exceed the power at the time the trip setpoint is exceeded by an amount proportional to the insertion rate and the time delay associated with trip circuitry and rod motion.

3.2.2 METHOD OF ANALYSIS

The uncontrolled RCCA withdrawal at a power condition is analyzed using a dynamic simulation incorporating point neutron kinetics, reactivity effects of moderator fuel and control rods, and decay heat. A simulation of the reactor vessel, steam generator tube and shell sides, pressurizer, and connecting piping is required to evaluate the coolant pressure and core inlet temperature response and their effect on core thermal margins. The reactor trip system, main steam and feedwater systems, and pressurizer control systems are also included in the model. This model calculates the response of the average core channel thermal-hydraulic conditions and heat generation and is coupled to a detailed model of the hot channel. This latter model calculates the departure from nucleate boiling ratio (DNBR) as a function of time during the accident.

In order to maximize the peak power during the transient the fuel and moderator temperature coefficients used in the analysis are the least negative values likely to be encountered. □ The least negative fuel and moderator temperature coefficients are normally encountered at BOC.

The reactivity insertion due to reactor trip is calculated by considering the most adverse combination of instrument and setpoint errors and time delays. The rate of reactivity insertion corresponding to the trip of the RCC assemblies is calculated assuming that the most reactive assembly is stuck in the fully withdrawn position.

Since the reactivity insertion rate determines which protective system function will initiate termination of the accident, a range of insertion rates must be considered. Relatively rapid insertion rates result in reactor trip due to high nuclear power. The maximum rate is bounded by that calculated assuming that the two highest worth banks, both in their region of highest incremental worth, are withdrawn at their maximum speed. Relatively slow rates of reactivity insertion result in a slower

transient that is terminated by an overtemperature ΔT trip signal, or in some cases, a high, pressurizer pressure signal. The minimum rate that need be considered in the analysis is determined by reducing the reactivity insertion rates until the analysis shows no further change in DNBR. The acceptance criteria for this accident are that the maximum pressures in the reactor coolant and main steam systems do not exceed 110% of design values and that cladding integrity be maintained by limiting the MDNBR greater than the MDNBR limit.

3.2.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.2-1 through 3.2-16 (see Appendix G). □

3.2.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed as a function of power level over the range 0 to 100% power at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC to determine the moderator coefficient at intermediate and full power under various conditions of xenon inventory. The model Bias, B_M, is applied to the moderator temperature coefficient as shown in Section 3.2.5.

c. MAXIMUM REACTIVITY INSERTION RATE, $\Delta\rho/\Delta t$

Calculations similar to those described in Section 3.1.4(c) are performed at intermediate and full power, equilibrium xenon conditions. The reactivity insertion rate is divided by the minimum cycle β_{eff} determined according to (f) of this subsection to yield the maximum reactivity insertion rate in dollars.

d. SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full and intermediate power conditions. Making the following assumptions generates a conservatively slow scram curve:

1. The integral of the scram curve is based on an initial rod position at or below the power dependent insertion limits specified in the Technical Specifications. [6]

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum N-1 rod worth.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

e. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$'s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II [7]. The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

f. EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC.

3.2.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above is adjusted to include the model reliability factors, RF_i and biases. These adjusted values are the cycle specific parameters that are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

Cycle Specific Parameters		Safety Analysis Parameters
a. $\alpha_D * (1 - RF_D)$	$\leq$	α_D (least negative bounding value)
b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
c. $\left(\frac{\Delta\rho/\Delta t * (1 + RF_R)}{\beta eff * (1 - RF_\beta)}\right)$	$\leq$	$\left(\frac{\Delta\rho/\Delta t(\text{bounding})}{\beta eff(\text{maximum})}\right)$

d. $\Delta\rho_{scram}(t) * (1-RF_R)$	$\geq$	$\Delta\rho_{scram}(t)$ (bounding)
e. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)
f. $\beta_{eff} * (1+RF_{\beta})$	$\leq$	β_{eff} (maximum)

□

3.3 CONTROL ROD MISALIGNMENT

3.3.1 DESCRIPTION OF THE ACCIDENT

In the analysis of this accident, one or more rod cluster control assemblies is assumed to be statically misplaced from the normal or allowed position. This situation might occur if a rod were left behind when inserting or withdrawing banks, or if a single rod were to be withdrawn. Full power operation under these conditions could lead to a reduction in DNBR and is subject to limitations specified in the plant Technical Specifications[6].

3.3.2 METHOD OF ANALYSIS

In the analysis of misaligned control rods, $F_{\Delta H}$ will be determined for the most limiting configuration. In general, the worst case is that with Bank D fully inserted except for one fully withdrawn rod cluster control assembly (RCCA). Bank D is the only bank that may be inserted at full power. In practice, multiple independent alarms would alert the operator well before the postulated conditions are approached.

The limiting value of $F_{\Delta H}$ is input to a steady state thermal-hydraulic sub-channel calculation to determine the departure from nucleate boiling ratio (DNBR). This calculation assumes the most adverse combination of steady state errors applied to core neutron flux level, coolant pressure, and coolant temperature at the core inlet.

The acceptance criterion for this accident is that fuel rod failure is not permitted and calculating the MDNBR to demonstrate that it is not less than the MDNBR limit insures this.

3.3.3 WPS SAFETY ANALYSIS RESULTS

There are no transient figures since control rod misalignment is a steady state accident condition. □

3.3.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

The nuclear enthalpy rise hot factor channel ($F_{\Delta H}$) is calculated for this accident consistent with the procedure described in Section 2. The maximum $F_{\Delta H}$ for a control rod misalignment at full power is calculated with Bank D fully inserted with one rod cluster control assembly (RCCA), fully withdrawn. □The rod misalignment calculations are performed for both BOC and EOC.

3.3.5 RELOAD SAFETY EVALUATIONS

The $F_{\Delta H}$ calculated above is conservatively adjusted to account for the model reliability factor, $RFF_{\Delta H}$. Additionally, a conservatism is applied to account for the maximum initial quadrant tilt condition (T) allowed by the Technical Specifications[6]. The resulting $F_{\Delta H}$ is then compared to the value used in the safety analysis as follows:

Cycle Specific Parameter		Safety Analysis Parameter
$F_{\Delta H}*(1+RFF_{\Delta H}) * (1+T)$	$\leq$	$F_{\Delta H}$ (Rod Misalignment)

3.4 CONTROL ROD DROP

3.4.1 DESCRIPTION OF THE ACCIDENT

In the analysis of this accident, a full-length RCCA is assumed to be released by the gripper coils and to fall into a fully inserted position in the core. The reactor is assumed to be operating in the manual mode of control.

A dropped rod cluster control assembly (RCCA) typically results in a reactor trip signal from the power range negative neutron flux rate circuitry. The core power distribution is not adversely affected during the short interval prior to reactor trip. The drop of a single RCCA assembly may or may not result in a reactor trip. If the plant is brought to full power with an RCCA fully inserted, a reduction in core thermal margins may result because of a possible increased hot channel peaking factor.

3.4.2 METHOD OF ANALYSIS

In the analysis of dropped RCCAs, $F_{\Delta H}$ will be determined for all possible dropped rod configurations and the most limiting configuration will be used in an analysis to determine the DNBR that would result if the core were returned to full power.

The limiting value of $F_{\Delta H}$ is input into a steady state thermal-hydraulic subchannel calculation to determine the departure from nucleate boiling ratio (DNBR). The calculation is performed assuming full power with the most adverse combination of steady state errors applied to core neutron flux level, coolant pressure, and coolant temperature at the core inlet.

The acceptance criteria for the accident are that pressures in the reactor coolant system and main steam system do not exceed 110% of the design pressures, and that fuel clad integrity is maintained, which is ensured by limiting the MDNBR to greater than the MDNBR limit.

3.4.3 WPS SAFETY ANALYSIS RESULTS

There are no transient figures since control rod drop is a steady state accident condition. □

3.4.4 CYCLE SPECIFIC PHYSICS CALCULATION

The nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) is calculated for all possible dropped rods, consistent with the procedure described in Section 2. Each rod is dropped at full power, equilibrium xenon conditions and the rod yielding the largest $F_{\Delta H}$ is determined. This rod is then dropped into the core assuming various initial xenon and flux distributions to determine the maximum $F_{\Delta H}$ under dropped rod conditions. Additionally, the peak $F_{\Delta H}$ occurring during the xenon transient

following the dropped rod is calculated and compared to the initial dropped rod $F_{\Delta H}$. These calculations are performed at both BOC and EOC.

3.4.5 RELOAD SAFETY EVALUATION

The nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) calculated above is conservatively adjusted to account for the model reliability factor, $RFF_{\Delta H}$. Additionally, a conservatism is applied to account for the maximum initial quadrant tilt condition allowed by the Technical Specifications. The resulting $F_{\Delta H}$ is then compared to the value used in the safety analysis as follows:

Cycle Specific Parameter		Safety Analysis Parameter
$F_{\Delta H} * (1+RFF_{\Delta H}) * (1+T)$	$\leq$	$F_{\Delta H}$ (Control Rod Drop)

3.5 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

3.5.1 DESCRIPTION OF THE ACCIDENT

The accident considered here is the malfunction of the chemical and volume control system in such a manner as to deliver unborated water at the maximum possible flowrate to the reactor coolant system under full power conditions. Dilution during refueling or startup is assumed to be recognized and terminated by operator intervention before loss of shutdown margin. With the reactor in automatic control, the power and temperature increase from boron dilution at power results in the insertion of the RCC assemblies and a decrease in shutdown margin. Rod insertion limit alarms would alert the operator to isolate the source of unborated water and initiate boration prior to the time that shutdown margin is lost. With the reactor in manual control, the power and temperature rise due to boron dilution would likely result in an overtemperature ΔT reactor trip if the operator did not intervene. After such a trip, the operator would be expected to isolate the unborated water source and initiate boration procedures

3.5.2 METHOD OF ANALYSIS

The system transient response to an uncontrolled boron dilution is simulated using a detailed model of the plant which includes the core, reactor vessel, steam generators, pressurizer, and connecting piping. The model also includes a simulation of the charging and letdown systems, rod control system, pressurizer control system, and the reactor protection system. Reactivity effects due to fuel and moderator feedback, coolant boron concentration, and control rod motion before and after trip are included in the analysis. This model provides the transient response of average core power, reactor coolant pressure, and coolant temperature at the core inlet which are applied as forcing functions to a thermal-hydraulic simulation of the hot channel. The hot channel model is used to calculate the departure from nucleate boiling ratio in the hot channel.

The reactivity inserted due to boron dilution is calculated by assuming the maximum possible charging flow, a bounding differential boron worth, and a zero ppm charging flow. The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, Doppler temperature coefficient and spatial power distribution. This is normally the BOC condition.

The minimum shutdown margin required by the Technical Specifications[6] is conservatively assumed to exist prior to the initiation of the transient. For at power conditions, the reactivity insertion due to reactor trip is calculated by considering the most adverse combination of instrument and setpoint errors and time delays. The rate of reactivity insertion corresponding to the trip of the RCC assemblies is calculated assuming that the most reactive assembly is stuck in the fully withdrawn position.

The acceptance criteria for this accident are that pressures in the reactor coolant system and main steam system do not exceed 110% of the respective design pressures, and that fuel clad integrity is maintained, which is ensured by limiting the MDNBR to greater than the MDNBR limit.

3.5.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.5-1 through 3.5-5 (see Appendix G). □

3.5.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the procedure described in Section 2. Cycle specific calculations are made as a function of power at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed using the methods described in Section 2. Cycle specific calculations are made at unrodded, full power conditions. □ The model bias, B_M, is included in the calculations.

c. BORON REACTIVITY COEFFICIENT, α_B

Calculations of α_B, the boron reactivity coefficient, are performed using the methods described in Section 2. Cycle specific calculations for these accidents are threefold: full power, all rods out; zero power, all rods in, less one stuck rod; and zero power, all rods in. These are performed at both BOC and EOC. □

d. SHUTDOWN MARGIN, SDM

For the refueling □mode (cold), the shutdown margin is calculated directly with all rods in rather than with one stuck rod, consistent with the assumptions made in the safety analysis.

e. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$'s are verified to remain within Technical Specification[6] limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II [7]. The cycle

specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

f. EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC.

g. CRITICAL BORON CONCENTRATION, C_B

Calculations of C_B are performed in accordance with the general procedures described in Section 2. For startup conditions, an ARI critical boron value is calculated and 100 ppm is added to the value for conservatism. For at power conditions, the BOC, HFP critical boron concentration is calculated and 100 ppm is added to the value for conservatism.

h. SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full power condition. Making the following assumptions generates a conservatively slow scram curve:

1. The integral of the scram curve is based on an initial rod position at or below the full power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum N-1 rod worth.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

3.5.5 RELOAD SAFETY EVALUATION

All the cycle specific parameters discussed above are adjusted to include model reliability factors RF_i and biases. These adjusted results are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

Cycle Specific Parameters		Safety Analysis Parameters
Refueling □ Conditions		
SDM (ARI)	≥	SDM (bounding)
At Power Conditions		
a) $\alpha_D*(1-RF_D)$	≤	α_D (least negative bounding value)
b) $\alpha_M+RF_M+B_M$	≤	α_M (least negative bounding value)
c) $\left(\frac{\alpha_B(1+RF_\beta)}{\beta eff(1-RF_\beta)}\right)$	≥	$\left(\frac{\alpha_B}{\beta eff}\right)$ (most negative bounding value)
d) SDM	≥	SDM (bounding)
e) $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	≤	Technical Specifications (Refer to Section 3.17)
f) $\beta_{eff} * (1+RF_\beta)$	≤	β_{eff} (maximum)
g) C_B (ARO)	≤	C_B (maximum, boron dilution at power)
h) $\Delta\rho_{scram}(t) * (1-RF_R)$	≥	$\Delta\rho_{scram}(t)$ (bounding)
Startup Conditions		
C_B (ARI)	≤	C_B (maximum, boron dilution at startup)

3.6 STARTUP OF AN INACTIVE COOLANT LOOP

3.6.1 DESCRIPTION OF THE ACCIDENT

Since there are no isolation valves or check valves in the Kewaunee reactor coolant system, operation of the plant with an inactive loop causes reverse flow through that loop. If there is a thermal load on the steam generator in the inactive loop, the hot leg coolant in that loop will be at a lower temperature than the core inlet temperature. The startup of the pump in the idle loop results in a core flow increase and the injection of cold water into the core, followed by a rapid reactivity and power increase. The resulting increase in fuel temperature limits the power rise due to Doppler feedback. Above 10% rated power, however, the reactor protection system prevents operation with an inactive loop, and consequently the temperature differential in an inactive loop would be small enough to minimize the accident consequences. Furthermore, the Kewaunee Technical Specifications[6] do not permit operation with a reactor coolant pump out of service except during low power physics testing.

3.6.2 METHOD OF ANALYSIS

The system transient response to an inactive loop startup is simulated using a detailed model which includes the core, reactor vessel, steam generators, main steam and reactor coolant piping, and the plant control and protection systems. This model calculates the time-dependent behavior of the average core power, coolant pressure, and core inlet flow and temperature which are supplied as forcing functions to a model of the hot channel for calculation of DNBR.

The accident is analyzed using the most negative moderator temperature coefficient and the least negative Doppler coefficient calculated to occur during the cycle. No credit is taken for reactivity reduction caused by reactor trip.

The reactor is initially assumed to be operating at 12% of rated power with reverse flow through the inactive loop. This includes a 2% uncertainty for calibration error above the 10% power setpoint in the protection system for single loop operation. The assumption of this high initial power level is conservative since it maximizes the temperature difference between the hot leg and cold leg in the inactive loop. The most adverse combination of initial coolant pressure and core inlet temperature is chosen to minimize the margin to core DNB limits.

The acceptance criteria for this accident are that the maximum pressures in the reactor coolant and main steam systems do not exceed 110% of design values and that cladding integrity be maintained, which is ensured by limiting the MDNBR greater than the MDNBR limit.

3.6.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in figures 3.6-1 through 3.6-5 (see Appendix G). □

3.6.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Specific evaluation of α_D for this accident is made assuming 12% power for both BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Specific calculations for this accident are performed □ at both BOC and EOC. The model bias, B_M, is included in the calculations.

c. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$'s are verified to remain within Technical Specification[6] limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II [7]. The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

3.6.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above is conservatively adjusted to include the model reliability factors RF_i and biases. These adjusted values are the cycle specific parameters that are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

Cycle Specific Parameters		Safety Analysis Parameters
a. $\alpha_D*(1-RF_D)$	$\leq$	α_D (least negative bounding value)
b. $\alpha_M-RF_M + B_M$	$\geq$	α_M (most negative bounding value)
c. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)

3.7 EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTION

3.7.1 DESCRIPTION OF THE ACCIDENT

Two classes of accidents are to be considered under this classification: Those that result in a decrease in feedwater temperature and those that result in an increase in feedwater flow. Either condition will result in an increased heat transfer rate in the steam generators, causing a decrease in the reactor coolant temperature and an increased core power level due to negative reactivity coefficients and/or control system action.

For the case of the decrease in feedwater temperature, the worst accident that may be postulated involves opening the bypass valve that diverts flow around the feedwater heaters. For the case of an increase in feedwater flow rate, the worst accident that may be postulated involves the full opening of the feedwater regulating valves. □

3.7.2 METHOD OF ANALYSIS

The feedwater system malfunction transient is analyzed using a dynamic simulation. This includes core kinetics and heat transfer, reactor vessel and coolant piping, steam generators, pressurizer, and control systems. Pertinent variables obtained from the NSSS simulation are then applied as forcing functions to a separate thermal-hydraulic model of the hot channel which calculates DNBR.

Two cases are analyzed. The first case is for a reactor in manual control and with a zero moderator temperature coefficient. This represents the situation where the reactor has the least inherent transient response capability. In this case, the core power slowly increases due to Doppler and moderator reactivity effects until the core power level again matches the load demand and a new steady state is achieved. The reactor does not trip. The coolant temperature decreases, which has the effect of increasing the margin to DNB. This increase in DNBR is larger than the decrease caused by the higher heat flux and the net effect is that MDNBR increases during the transient. The manual control case is analyzed for the feedwater temperature decrease class.

The second case analyzed assumes that the reactor automatic control system responds to the decreasing coolant temperature and matches reactor power to load demand. A conservatively large (in absolute value) negative moderator temperature coefficient is assumed to exist. □This case results in a somewhat higher final core power level than the manual control case without moderator feedback; this in turn results in a net decrease in DNBR but the decreased coolant temperature again maintains a significant margin above the MDNBR limit.

The core neutronic characteristics that exert a significant influence on the calculated results of this transient are the Doppler and moderator reactivity coefficients. A range of Doppler and moderator temperature coefficients that bound cycle operation is used in the analysis to maximize the power increase. For such slow rates of

reactivity addition as are encountered, the transient response is insensitive to the value of ℓ^*, the prompt neutron lifetime. Trip reactivity insertion characteristics are relevant for cases where the reactor automatic control system responds, since the reactor may trip. The automatic control case is analyzed for both the feedwater temperature decrease and the feedwater flow increase classes.

The acceptance criteria for the feedwater system malfunction transient are that cladding integrity is maintained, which is ensured by limiting the minimum DNBR to be greater than the MDNBR limit and that maximum pressure in the reactor coolant and main steam system not exceed 110% of the design pressure.

3.7.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.7-1 through 3.7-10 (see Appendix G). □

3.7.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at100% power at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC to determine the least negative α_M and the most negative α_M at full power conditions. The model bias is included in these calculations.

c. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$'s are verified to remain within Technical Specification[6] limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II [7]. The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

d. SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full power condition.

Making the following assumptions generates a conservatively slow scram curve:

1. The integral of the scram curve is based on an initial rod position at or below the full power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum N-1 rod worth.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

3.7.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above are adjusted to include the model reliability factors RF_i and biases. These adjusted values are the cycle specific parameters that are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to feed water malfunction transients if the following inequalities are met:

	Cycle Specific Parameters		Safety Analysis Parameters
BOC	a. $\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value) if $\alpha_M \leq 0$
	$\alpha_D * (1+RF_D)$	$\geq$	α_D (most negative bounding value) if $\alpha_M > 0$
	b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
EOC	a. $\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value)
	b. $\alpha_M - RF_M + B_M$	$\geq$	α_M (most negative bounding value)
	c. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)
	d. $\Delta\rho_{scram}(t) * (1-RF_R)$	$\geq$	$\Delta\rho_{scram}(t)$ (bounding)

3.8 EXCESSIVE LOAD INCREASE

3.8.1 DESCRIPTION OF THE ACCIDENT

An excessive load increase accident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between core heat generation and secondary side load demand. The ensuing decrease in reactor coolant temperature results in a core power increase due to fuel and moderator feedback and/or control system action. Only steam flow increases within the capability of the turbine control valves are considered here; larger flow increases are considered in connection with main steam line break accidents. □

3.8.2 METHOD OF ANALYSIS

The excessive load increase transient is analyzed using a dynamic simulation which includes the reactor core, reactor vessel, steam generators, pressurizer and connecting piping. The main steam and feedwater systems and control and protection systems are also modeled. The departure from nucleate boiling ratio is computed using a separate model of the hot fuel assembly channel thermal-hydraulic behavior. □This model is coupled to the NSSS simulation that supplies core power and flow and coolant temperature and pressure as a function of time.

The transient is initiated by imposing a rapid increase in steam flow to 120% of rated full power flow. Initial pressurizer pressure, reactor coolant temperature, and core power are assumed at their extreme steady-state values to minimize the calculated margin to DNB. Typically, four cases are analyzed: moderator reactivity coefficient at minimum and maximum values with manual and with automatic reactor control.

For the cases in manual control, the case with the least negative moderator coefficient shows a large coolant temperature decrease relative to the power increase and the net effect is to increase the DNBR. The case with the more negative moderator coefficient shows a larger increase in power and a decrease in DNBR. The cases with automatic reactor control show similar behavior but the control system acts to maintain average coolant temperature by increasing reactor power, so the DNBR decreases in both cases. However, all cases □exhibit a large margin to the MDNBR limit.

Reactor trip occurs during some of the transients considered, consequently scram reactivity insertion characteristics are □ factors in the evaluation of these transients. Moderator and Doppler reactivity coefficients are □significant kinetics parameters. □ The acceptance criteria for this accident are that the fuel cladding integrity be maintained, which is ensured by limiting the MDNBR to □greater than the MDNBR limit and reactor coolant and main steam system maximum pressures not be greater than 110% of the design pressures.

3.8.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.8-1 through 3.8-18 (see Appendix G). □

3.8.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at the full-power, equilibrium, xenon conditions at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC to determine the maximum and minimum values of the moderator coefficient at full-power, equilibrium, xenon conditions.

c. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification[6] limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II[7]. The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

d. SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}$ (t)

Calculations of the scram reactivity are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the zero power condition. Making the following assumptions generates a conservatively slow scram curve:

1. The integral of the scram curve is based on an initial rod position at or below the zero power insertion limits specified in the Technical Specification[6].

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant negative reactivity insertion.

3. The xenon distribution is that which causes the minimum N-1 rod worth.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

3.8.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above are adjusted to include the model reliability factors RF_i and biases. These adjusted values are the cycle specific parameters that are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to excessive load increase transients if the following inequalities are met:

	Cycle Specific Parameters		Safety Analysis Parameters
BOC	a. $\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value) if $\alpha_M \leq 0$
	$\alpha_D * (1+RF_D)$	$\geq$	α_D (most negative bounding value) if $\alpha_M > 0$
	b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
EOC	a. $\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value)
	b. $\alpha_M - RF_M + B_M$	$\geq$	α_M (most negative bounding value)
	c. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)
	d. $\Delta\rho_{scram}(t) * (1-RF_R)$	$\geq$	$\Delta\rho_{scram}(t)$ (bounding)

3.9 LOSS OF EXTERNAL LOAD

3.9.1 DESCRIPTION OF THE ACCIDENT

The most likely source of a complete loss of load is a turbine-generator trip. Above approximately 10% power, a turbine trip generates a direct reactor trip which is signaled from either of two diverse inputs: release of autostop oil or stop valve closure. If credit is taken for the steam bypass system and pressurizer control system, there is no significant increase in reactor coolant temperature or pressure. To provide a conservative assessment of the accident, however, no credit is taken for direct reactor trip, steam bypass actuation or pressurizer pressure control. Under these assumptions both secondary and primary pressures increase rapidly and a reactor trip is generated. □

3.9.2 METHOD OF ANALYSIS

The loss of external load accident is analyzed using a detailed model of the nuclear steam supply system and associated control and protection systems. Core kinetics heat transfer, reactor coolant and steam generator secondary side temperatures and pressures, steam and feedwater flowrates, and pressurizer liquid level are some of the variables computed by the model. No credit is taken for direct reactor trip caused by turbine trip, the steam bypass system or the pressurizer control system. The secondary side pressure rises to the safety valve setpoint and is limited to that pressure by steam relief through the safety valves. The scram mitigates the consequences of this accident and prevents water relief through the pressurizer relief and safety valves.

The worst case with respect to overpressurization assumes no control rod motion prior to reactor trip and no credit for pressurizer relief or spray valves. In this case, the magnitude of the moderator reactivity coefficient has only a very slight effect on the magnitude of the maximum reactor coolant pressure; and likewise very little effect on DNBR response. The peak pressure is likewise insensitive to the magnitude of the Doppler reactivity coefficient. □

The acceptance criteria for this accident are that the maximum main steam and reactor coolant system pressures not exceed 110% of their design pressures and DNBR calculations must demonstrate that the MDNBR is not less than the MDNBR limit at any time during the transient.

3.9.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.9-1 through 3.9-20 (see Appendix G). □

3.9.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at the full power equilibrium xenon condition at BOC and EOC.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC to determine the least negative value of the moderator coefficient at the full power condition. The model bias, B_M, is included in these calculations.

c. SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full power condition. Making the following assumptions generates a conservatively slow scram curve:

1. The integral of the scram curve is based on an initial rod position at or below the full power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum N-1 rod worth.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

d. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$'s are verified to remain within Technical Specification[6] limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II [7]. The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

3.9.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above was adjusted to include the model reliability factors RF_i and biases. These adjusted values are the cycle specific parameters that are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

	Cycle Specific Parameters		Safety Analysis Parameters
BOC	a. $\alpha_D * (1+RF_D)$	$\geq$	α_D (most negative bounding value) if $\alpha_M \leq 0$
	$\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value) if $\alpha_M > 0$
	b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
EOC	a. $\alpha_D * (1+RF_D)$	$\geq$	α_D (most negative bounding value)
	b. $\alpha_M - RF_M + B_M$	$\geq$	α_M (most negative bounding value)
	c. $\Delta\rho_{scram}(t) * (1-RF_R)$	$\geq$	$\Delta\rho_{scram}(t)$ (bounding)
	d. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specifications (Refer to Section 3.17)

3.10 LOSS OF NORMAL FEEDWATER FLOW

3.10.1 DESCRIPTION OF THE ACCIDENT

This accident is defined as a complete loss of normal feedwater. Realistically, the plant's auxiliary feedwater pumps would be actuated and would supply sufficient feedwater to both steam generators to dissipate residual and decay heat after reactor trip. To provide a margin of conservatism however, only one of the three auxiliary feedwater pumps is assumed to deliver feedwater to one of the two steam generators. Under this assumption, the steam generator not receiving auxiliary feedwater suffers a degradation of heat transfer capability and the reactor coolant system temperature and pressure increase as a result of decay heat following reactor trip. Traditionally, an additional conservatism has been applied to the analysis of the loss of feedwater accident by assuming that the reactor coolant pumps are tripped and coast down to natural circulation conditions, further degrading the heat transfer capability of both steam generators. When analyzed in this manner, the accident corresponds to a loss of non-emergency AC power.

3.10.2 METHOD OF ANALYSIS

The loss of normal feedwater accident is analyzed using a dynamic simulation model that includes the reactor and reactor coolant system and the secondary plant systems. The model includes a simulation of the natural circulation flow existing in the reactor coolant system subsequent to the assumed coast down of the reactor coolant pumps. The model also includes the heat source due to the decay of fission products since the reactor trips early in the transient, and this decay heat constitutes the main energy source thereafter.

The results of the analysis of the loss of normal feedwater accident are not sensitive to the values of the core neutronics parameters. The reactor trips very early in the transient. □Since this occurs well before steam generator heat transfer capability has been reduced, the margin to DNB is not reduced significantly prior to reactor trip. The maximum reactor coolant temperature occurs well after accident initiation and is not significantly affected by the core neutron power transient, since decay of fission products is the major energy source over most of this time interval. The decay heat is conservatively calculated by assuming that the fission products are initially in equilibrium at the existing core power level.

The acceptance criteria for this accident are that pressure in the reactor coolant and main steam systems not exceed 110% of design pressure and that the MDNBR occurring during the accident be not less than the MDNBR limit.

3.10.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.10-1 through 3.10-5 (see Appendix G). □

3.10.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

The loss of normal feedwater transient is not sensitive to core physics parameters since the reactor is assumed to trip in the initial stages □of the transient. This trip occurs □well before the heat transfer capability of the steam generator is reduced. The decay heat then drives the transient from the tripped reactor. Also, the loss of flow transient analyzed in Section 3.11 is considered a more severe transient of this type.

Therefore, no comparisons will be made for reload safety evaluations.

3.11 LOSS OF REACTOR COOLANT FLOW - PUMP TRIP

3.11.1 DESCRIPTION OF THE ACCIDENT

The accident considered here is the simultaneous loss of electrical power to both of the reactor coolant pumps. Two cases are analyzed: One in which the pumps are at nominal frequency and one in which they are at a degraded frequency. As a result of loss of driving head supplied by the pumps, the coolant flow rate decreases, but is retarded by the rotational inertia of the reactor coolant pump flywheel and by the hydraulic inertia of the fluid itself. Any one of several diverse and redundant signals which monitor coolant pump and coolant flow conditions trips the reactor. This trip results in a power reduction before the thermal-hydraulic conditions in the core approach those that could result in damage to the fuel. Loss of power to one of the pumps with both pumps initially operating may also occur, but the consequences are less severe than for the two-pump trip. Seizure of the reactor coolant pump shaft is considered in Section 3.12.

3.11.2 METHOD OF ANALYSIS

The loss of forced reactor coolant flow accident is analyzed using a detailed model of the reactor coolant system thermal-hydraulics. The conservation of momentum and continuity equations for the coolant, coupled to a representation of the pump hydraulics and speed coastdown, are solved to compute the system flowrate as a function of time. Reactor core neutron kinetics and heat transfer equations are coupled to the flow coastdown equations in order to compute heat flux and coolant temperatures in the reactor. A simulation of the steam generators and pressurizer is also included in the model. A separate model analyzes the transient response of the core hot channel, using conditions supplied by the NSSS model as input, and computes the departure from nucleate boiling ratio (DNBR).

The initial conditions for the accident analysis assume the most adverse combination of power, core inlet temperature, and pressurizer pressure including allowances for steady state error so that the initial margin to DNB is the minimum expected during steady state operation.

The power transient is analyzed using a least negative value of moderator reactivity coefficient that bounds the cycle. □ The most negative value of Doppler reactivity coefficient that bounds the cycle is used in the analysis since this value has been shown to result in the maximum hot spot heat flux at the time of minimum DNBR. The reactivity reduction due to control rod insertion after a trip is calculated by assuming the most adverse delay time expected to occur between loss of power to the pump and the initiation of rod motion. Upon reactor trip, it is assumed that the most reactive RCC assembly is stuck in its fully withdrawn position, resulting in a minimum insertion of negative reactivity. The trip reactivity insertion dominates the power response and is the most important neutronics input parameter.

The acceptance criteria for the loss of reactor coolant flow accident are that the MDNBR not fall below the MDNBR limit and that the maximum reactor coolant and main steam system pressures not exceed 110% of their design values.

3.11.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.11-1 through 3.11-8 (see Appendix G). □

3.11.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC to determine the most negative value at full power conditions.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC to determine the least negative value of the moderator coefficient at the full power condition. The model bias, B_M, is included in the calculations.

c. SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full power condition.

Making the following assumptions generates a conservatively low scram curve:

1. The integral of the scram curve is based on an initial rod position at or below the full power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum N-1 rod worth.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

d. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$'s are verified to remain within Technical Specification[6] limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control Scheme PDC-II[7]. The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

e. FUEL TEMPERATURE, T_f

Fuel temperature is calculated at conditions to maximize fuel temperature as a function of linear heat generation rate. The maximum cycle specific linear heat generation rate is used to derive the maximum cycle specific fuel temperature.

3.11.5 RELOAD SAFETY EVALUATIONS

Each of the physics parameters calculated above are adjusted to include the model reliability factors RF_i □and biases. These adjusted values are the cycle specific parameters that are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to loss of reactor coolant flow pump trip transients if the following inequities are met:

Cycle Specific Parameters		Safety Analysis Parameters
a. $\alpha_D * (1+RF_D)$	$\geq$	α_D (most negative bounding value) □
b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
c. $\Delta\rho_{scram}(t) * (1-RF_R)$	$\geq$	$\Delta\rho_{scram}(t)$ (bounding)
d. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	Technical Specification (Refer to Section 3.17)
e. $T_f * (1+RFT_f)$	$\leq$	T_f (bounding value Loss of Flow)

□

3.12 LOSS OF REACTOR COOLANT FLOW - LOCKED ROTOR

3.12.1 DESCRIPTION OF THE ACCIDENT

The accident postulated is the instantaneous seizure of the rotor of a single reactor coolant pump. Flow through the affected loop is rapidly reduced, leading to a reactor trip initiation due to low flow. The sudden decrease in core flow while the reactor is at power results in a degradation of core heat transfer and departure from nucleate boiling in some of the fuel rods. The sudden degradation in steam generator heat transfer associated with the coolant flow transient causes an increase in reactor coolant temperature and a pressurizer insurge. The pressurizer safety valves are actuated and maintain the reactor coolant system pressure within acceptable limits. This accident is classified as a condition IV limiting fault.

3.12.2 METHOD OF ANALYSIS

The analysis of the locked reactor coolant pump rotor is performed using a detailed model of the reactor coolant system thermal-hydraulics. The conservation of momentum and continuity equations for the coolant, coupled to a representation of the pump hydraulic characteristics, are solved to compute the system flow rates as a function of time. Reactor core neutron kinetics and transient heat transfer equations are coupled to the flow equations in order to compute the core heat flux and coolant temperatures in the reactor. A simulation of the pressurizer and steam generators is also included in the model. Separate models compute the thermal-hydraulic response of the coolant hot channel and fuel hot spot using conditions supplied by the NSSS model as input. These models compute heat flux, fuel and clad temperatures, and MDNBRs for a conservative evaluation of the extent of fuel damage which could occur during a locked rotor accident.

For the purposes of evaluating the minimum DNBR, the initial conditions for the accident analysis assume the most adverse combination of power, core inlet temperature, and pressurizer pressure including allowances for steady state errors so that the initial margin to DNB is the minimum expected during steady state operation. For purposes of evaluating the reactor coolant system pressure transient, the initial pressure is assumed as the maximum expected during normal operation including allowances for instrumentation error and controller tolerances.

The power transient is analyzed using a least negative value of moderator reactivity coefficient that bounds the cycle. The most negative Doppler reactivity coefficient is used in the analysis since this results in maximum hot spot heat flux at the time of minimum DNBR. Trip reactivity insertion characteristics are calculated by assuming the maximum time delay between a low flow signal and control rod motion. It is further assumed that the most reactive RCC assembly is stuck in a fully withdrawn position.

The acceptance criteria for the locked rotor analysis are as follows:

1. The maximum reactor coolant and main steam system pressures must not exceed 110% of the design values.

2. The number of fuel rods calculated to experience a DNBR of less than the MDNBR limit should not exceed the number of fuel rods required to fail in order to yield doses due to released activity which will exceed the limits of 10CFR100.

3. The maximum clad temperature calculated to occur at the core hot spot must not exceed 2700°F.

3.12.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.12-1 through 3.12-5 (see Appendix G). □

3.12.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC to determine the most negative value at full power conditions.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC to determine the least negative value at the full power condition. The model bias, B_M, is included in the calculations.

c. SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full power condition. Making the following assumptions generates a conservatively slow scram curve:

1. The integral of the scram curve is based on an initial rod position at or below the full power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum N-1 rod worth.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

d. EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2. Cycle specific calculations are performed at BOC and EOC. □

e. FUEL ROD CENSUS, $C_{F\Delta H}$

Calculation of the percentage of fuel rods at or above a specific $F_{\Delta H}$ value is performed in accordance with the general procedures described in Section 2. The calculations determine the percentage of fuel rods, which exceed the limiting value of $F_{\Delta H}$ and which are expected to experience DNB ratios less than the MDNBR limit. Cycle specific fuel rod $C_{F\Delta H}$ census histograms are determined at the full power conditions for both BOC and EOC with the control rods at or above the power dependent insertion limits.

f. NUCLEAR HEAT FLUX HOT CHANNEL FACTOR, F_Q

The maximum core F_Q's are assumed to remain within the current limits as defined in the Technical Specifications[6] for allowable combinations of axial offset and power level. The safety analysis value for F_Q bounds the current Technical Specification[6] limit.

g. FUEL TEMPERATURE, T_f

Fuel temperature is calculated at conditions to maximize fuel temperature as a function of linear heat generation rate. The maximum cycle specific linear heat generation rate is used to derive the maximum cycle specific fuel temperature.

3.12.5 RELOAD SAFETY EVALUATION

Each of the physics parameters calculated is adjusted to include the model reliability factors, RF_i and biases. These adjusted values are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to the locked rotor accident if the following inequalities are met:

Cycle Specific Parameters		Safety Analysis Parameters
a. $\alpha_D*(1+RF_D)$	$\geq$	α_D (most negative bounding value) □
b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
c. $\Delta\rho_{scram}(t) * (1-RF_R)$	$\geq$	$\Delta\rho_{scram}(t)$ (bounding)
d. $\beta_{eff} * (1+RF_\beta)$	$\leq$	β_{eff} (maximum value)
e. $C_{F\Delta H}$, locked rotor	$\leq$	$C_{F\Delta H}$ (upper bound)
f. $F_Q * (1+RFF_Q) * (1+T)$	$\leq$	F_Q (bounding value Locked Rotor)
g. $T_f * (1+RFT_f)$	$\leq$	T_f (bounding value Locked Rotor)

3.13 FUEL HANDLING ACCIDENT

3.13.1 DESCRIPTION OF THE ACCIDENT

The accident considered is the sudden release of the gaseous fission products held in the plenum between the pellets and cladding of one fuel assembly. The activity associated with this accident would be released either inside the Containment Building or the Auxiliary Building. A high radiation level alarm in the Containment Building would close the purging supply and exhaust ducts. A high radiation level on the Auxiliary Building vent monitor would automatically activate the special ventilation system with subsequent absolute and charcoal filtration. In calculating the offsite exposure from the accident, however, it is assumed that the activity is discharged to the atmosphere at ground level from the Auxiliary Building since this maximizes the offsite doses.

3.13.2 METHOD OF ANALYSIS

The gap activity is calculated based on fission gas buildup in the fuel and subsequent diffusion to the fuel rod gap at rates dependent upon the operating temperature. The calculation assumes that the assembly with the maximum gap activity is the one that is damaged. Only that fraction of fission gases which has diffused into the gap and plenum regions of the fuel pin would be available for immediate release. This fraction is calculated based on a conservative evaluation of the temperature and power distribution in the highest-powered assembly prior to shutdown. This activity is further reduced by decay during the time elapsing after shutdown before removal of the vessel head.

The activity present in the fuel rod gaps consists predominately of halogens and noble gases. Decontamination factors are applied to account for halogen depletion by the pool water; all the noble gas inventory is assumed to escape from the pool water surface. □Using conservative radiological formulae, the activity concentrations at the site exclusion boundary are converted to integrated whole body and thyroid doses. These doses are then compared to the acceptance criteria set forth in 10CFR100.

3.13.3 WPS SAFETY ANALYSIS RESULTS

There are no transient figures since the fuel handling accident does not impose a transient on the NSSS.

3.13.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

Calculations of the nuclear enthalpy rise hot channel factor $F_{\Delta H}$, are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$'s are verified to remain within Technical Specification limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-

core detectors using a Power Distribution Control Scheme PDC-II[7]. The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

3.13.5 RELOAD SAFETY EVALUATIONS

The $F_{\Delta H}$ calculated above is conservatively adjusted to account for the model reliability factor, $RFF_{\Delta H}$. Additionally, a conservatism is applied to account for the maximum initial quadrant tilt condition (T) allowed by the Technical Specifications[6]. The resulting $F_{\Delta H}$ is then compared to the value used in the safety analysis as follows:

Cycle Specific Parameters		Safety Analysis Parameters
$F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	$F_{\Delta H}$ (bounding value for the fuel handling accident)

3.14. MAIN STEAM LINE BREAK

3.14.1 DESCRIPTION OF THE ACCIDENT

3.14.1.a Core Response

The accidents considered include a spectrum of break sizes and locations inside containment. The plant is initially at no load conditions with both reactor coolant pumps running. The resulting uncontrolled steam release causes a rapid reduction in reactor coolant temperature and pressure as the secondary side is depressurized. If the most reactive RCC assembly is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power due to the negative moderator coefficient. A return to power is potentially a problem mainly because of the high hot channel factors that exist with a stuck RCC assembly. The core is ultimately restored to a subcritical condition by boric acid injection via the Emergency Core Cooling System. The zero power case is considered because the stored energy of the system is at a minimum and steam generator secondary inventory is at a maximum under these conditions, thus increasing the severity of the transient. Similarly, the case with both reactor coolant pumps running is analyzed because this assumption maximizes the cool down rate of the reactor coolant system.

3.14.1.b Containment Response

The accidents considered include a spectrum of break sizes and locations inside containment. Parameters that influence the plant response are conservatively selected to maximize the mass and energy release into containment. Containment thermal hydraulic response is analyzed to determine the maximum pressure and temperature inside containment. Actions of engineered safeguards systems are critical in mitigating the consequences of the accident and keep containment pressure and temperature within the acceptance criteria.

3.14.2 METHOD OF ANALYSIS

3.14.2.a Core Response

The analysis of the steam line break accident is performed using a detailed, multi-loop model of the core, reactor coolant system and pressurizer, steam generators, and main steam system. The steam flow through the broken steam line is calculated using a critical flow model. Conservation equations for the steam generator shell side mass and energy inventory are solved to predict the temperatures and

pressures existing throughout the transient. Heat transfer from the reactor coolant system to the steam generators is calculated based on instantaneous fluid conditions and empirical correlations. The analytical model includes a representation of the reactor vessel upper head volume in order to predict the transient response of the reactor coolant pressure subsequent to draining the pressurizer. A simulation of the safety injection system and boron injection allows calculation of the core coolant boron concentration and its influence on core neutron kinetics. The representation of core moderator density reactivity effects must include allowances for the large change in density that the coolant undergoes as the system temperature falls. A detailed thermal-hydraulic model of the hot channel is coupled to the system simulation and provides a calculation of the departure from nucleate boiling ratio during the transient.

The core neutronics parameters input to the model are evaluated at the core conditions which yield the most limiting values of moderator and Doppler reactivity coefficients, spatial power distribution, and shutdown margin. This is normally the EOC condition, since the moderator temperature coefficient is most negative and the shutdown margin is minimum. Trip reactivity insertion characteristics need not be input to the analysis, since the reactor is assumed to be initially shutdown with minimum shutdown margin. The moderator reactivity coefficient is also calculated assuming the most reactive rod is stuck in its fully withdrawn position, and includes the local reactivity feedback from the high neutron flux in the vicinity of the stuck rod.

An important parameter that is input to the model is the boron concentration in the Refueling Water Storage Tank. The value used in the model corresponds to the minimum value required by the Technical Specifications[6].

The acceptance criteria for the main steam line break core response accident are that reactor coolant and main steam system pressures do not exceed 110% of design pressure and that the MDNBR be not less than the main steam line break MDNBR limit.

3.14.2.b Containment Response

Containment response analysis methods build upon the core response methods. Additional analytical requirements for the containment response accident include containment thermal-hydraulic analysis using the steam line break mass and energy release data, entrainment calculations for the steam generator blowdown, and greater modeling detail for the engineering safeguards that are important to the accident. The acceptance criteria for steam line break containment

response are that containment temperature and pressure remain below safety analysis acceptance criteria.

There are four major factors that influence the release of mass and energy following a steam line break. These are the initial steam generator fluid inventory, primary to secondary heat transfer, engineered safeguard system's operation, and the state of the secondary fluid blowdown. Those factors are most sensitive to the following:

- Plant Power Level
- Main Feedwater System Design
- Auxiliary Feedwater System Design
- Break Type, Area, Location
- Availability of Offsite Power
- Steam Generator Design
- Safety System Failures
- SG Reverse Heat Transfer and Reactor Coolant System Metal Heat Capacity

All of the above variables are considered in the analyses and are conservatively considered.

The safety analyses are based on a set of five parameters:

1. Power Level.

2. Break Size and Location.

3. Single Failures.

4. Offsite Power.

5. Entrainment.

3.14.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.14-1 through 3.14-12 (see Appendix G). □

3.14.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Specific calculations □ of α_M are made as a function

of core average temperature with all rods in, except for the most reactive RCCA, at 1000 psia. Using this functional value of $\alpha_M(T)$, k_{eff} is calculated versus temperature assuming an initial condition at 547°F with a shutdown amount equal to the shutdown margin limit.

b. SHUTDOWN MARGIN, SDM

The shutdown margin is calculated consistent with the description given in Section 2 and is calculated for □EOC, HZP and HFP conditions.

c. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

The maximum $F_{\Delta H}$ is calculated consistent with the description given in Section 2 and is calculated for reactor conditions expected during the cooldown.

d. DOPPLER TEMPERATURE COEFFICIENT, α_D

Calculations of α_D are performed in accordance with the procedures described in Section 2. Cycle specific calculations are made as a function of power at BOC and EOC.

e. BORON REACTIVITY COEFFICIENT, α_B

Calculations of α_B, the boron reactivity coefficient, are performed using the methods described in Section 2. Cycle specific calculations for EOC and HZP conditions are performed.

3.14.5 RELOAD SAFETY EVALUATION

Each parameter calculated above is conservatively adjusted to include the model reliability factors, RF_i, and biases. □These results are then compared to the bounding values assumed in the safety analysis. For k_{eff} versus temperature during cooldown, the reliability factors are applied to the calculation of the moderator temperature coefficient in the determination of k_{eff}.

Uncertainties for the rod worth, moderator temperature defect, and Doppler temperature defect are applied to the shutdown margin (SDM) as discussed in Section 2.5. The cycle specific parameters are acceptable if the following inequalities are met:

Cycle Specific Parameters		Safety Analysis Parameters
a. $k_{eff}(T)$	$\leq$	$k_{eff}(T)$ (bounding)

b. SDM	$\geq$	SDM (bounding)
c. $F_{\Delta H}*(1+RFF_{\Delta H})*(1+T)$	$\leq$	$F_{\Delta H}$ (bounding value steam line break)
d. $\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value)
e. $\alpha_B * (1-RF_B)$	$\leq$	α_B (least negative bounding value)

3.15 CONTROL ROD EJECTION

3.15.1 DESCRIPTION OF THE ACCIDENT

This accident is postulated to result from the unlikely failure of a control rod pressure housing followed by ejection of an RCC assembly by the reactor coolant system pressure. If a rod inserted in a high worth region of the core were to be ejected, the rapid reactivity insertion and unfavorable power distribution, which would result might cause localized fuel rod damage.

3.15.2 METHOD OF ANALYSIS

The analysis of the control rod ejection accident requires a model of the neutron kinetics coupled to models of the fuel and clad transient conduction and the thermal-hydraulics of the coolant channel. In industry practice, model sophistication has varied from point kinetics to three-dimensional spatial kinetics. When three-dimensional calculations are not employed, the reactivity feedbacks must be corrected using weighting factors to account for the spatial dimensions not included in the model. The thermal-hydraulic model used includes a multi-nodal radial model of fuel, gap, and clad conduction; and a multi-nodal axial model of the coolant channel. Since the calculations result in maximum fuel enthalpies less than those corresponding to catastrophic fuel failures, the system pressure surge is calculated on the basis of conventional heat transfer from the fuel. The pressure surge model includes prompt heat generation in the coolant (so called "direct moderator heating"), fluid transport in the system, heat transfer in the steam generators, and the action of relief and safety valves. No credit is taken for pressure reduction caused by the assumed failure of the control rod pressure housing.

The maximum ejected rod worth is calculated with all control banks at their maximum permissible insertion for the power level of interest.

The moderator reactivity effect is included in the model by correlating reactivity with moderator density, thereby including effects of coolant temperature, pressure, and voiding. The Doppler reactivity effect is typically correlated as a function of either fuel temperature or power. The highest boron concentration corresponding to the initial reactor state is assumed in the calculation of moderator feedback. The largest temperature rise during the transient, and hence the largest reactivity effects, occurs in channels where the power is higher than average. This means that the reactivity feedback is larger than that predicted by a single average channel analysis. As a result, when a three-dimensional space-time kinetics calculation is not performed, weighting factors are applied as multipliers to the average channel Doppler feedback reactivity to account for spatial reactivity feedback effects. □A one-dimensional kinetics model is used in which the axial dimension effects such as power distribution, scram insertion rate, and temperature distribution are accounted for.

The results of the accident analysis are relatively insensitive to β_{eff}. □The minimum value of β_{eff} calculated for the assumed, initial, reactor state is used in the accident analysis.

The results are also relatively insensitive to ℓ^*, the prompt neutron lifetime, in the range of values normally encountered in commercial pressurized water reactors. Minimum values of ℓ^* are used in the accident analysis.

Control rod reactivity insertion during trip is obtained by combining a differential rod worth curve with a rod velocity curve, based on maximum design values for scram insertion times. The reactor trip delay time is calculated by combining the maximum time delays involved in the instrumental and actuation circuitry.

The acceptance criteria for the control rod ejection accident are as follows:

* The average hot spot fuel enthalpy must be less than 200 calories/gram.
* The maximum reactor coolant system pressure must be less than 2900 psia.
* The maximum clad temperature calculated to occur at the core hot spot must not exceed 2700°F.

3.15.3 WPS SAFETY ANALYSIS RESULTS

WPS safety analysis results are presented in Figures 3.15-1 through 3.15-12 (see Appendix G). □

3.15.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

a. DOPPLER TEMPERATURE COEFFICIENT, α_D

The values of α_D are calculated in accordance with the general procedures described in Section 2. Full and zero power core conditions in rodded, unrodded, and ejected rod configurations are considered at BOC and EOC in order to determine the least negative value of α_D.

b. MODERATOR TEMPERATURE COEFFICIENT, α_M

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Cycle specific values are computed at full and zero power, BOC and EOC core conditions to determine the least negative moderator temperature coefficients.

c. EFFECTIVE DELAYED NEUTRON FRACTION, β_{eff}

The value of β_{eff} is calculated in accordance with the general procedures given in Section 2. Cycle specific calculations are performed at BOC and EOC. □

d. MAXIMUM EJECTED ROD WORTH, $\Delta\rho_{eject}$

Calculations of the ejected rod worth are performed with the nodal model in three dimensions. No credit is taken for either moderator or Doppler reactivity feedback mechanisms. All calculations are performed with the control rods at or below their power dependent insertion limits (PDIL). Cycle specific calculations are performed at BOC and EOC for both the full power and zero power conditions. The search for the highest worth ejected rod includes all rods initially inserted to the PDIL. The maximum worth of the ejected rod includes consideration of transient xenon conditions such as maximum positive or negative axial offsets.

e. SCRAM REACTIVITY CURVE, $\Delta\rho_{scram}(t)$

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2. Cycle specific calculations for this accident are performed at BOC and EOC for the full and zero power conditions. Making the following assumptions generates a conservatively slow scram curve:

1. The integral of the scram curve is based on an initial rod position at or below the power insertion limits.

2. The shape of the scram curve is based on an initial rod position of full out. This provides the longest possible delay to significant reactivity insertion.

3. The xenon distribution is that which causes the minimum N-1 rod worth.

4. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

f. NUCLEAR HEAT FLUX HOT CHANNEL FACTOR, F_Q

F_Q is calculated for each of the cases investigated as described above for the determination of the maximum $\Delta\rho_{eject}$. The maximum value of F_Q does not necessarily correspond to the maximum value of $\Delta\rho_{eject}$. As described above, the calculations of F_Q for the ejected rod do not take credit for the moderator or Doppler feedback mechanisms.

g. PROMPT NEUTRON LIFETIME, ℓ^*

The value of ℓ^* is calculated in accordance with the general procedures given in Section 2. Cycle specific calculations are performed at BOC and EOC.

3.15.5 RELOAD SAFETY EVALUATION

Each of the physics parameters calculated for this accident that is adjusted to include the model reliability factors, RF_i, and biases. □ These adjusted values are the cycle specific parameters to be compared to the bounding values used in the safety analysis.

The cycle specific parameters are acceptable with regard to the ejected rod accident if the following inequalities are met:

Cycle Specific Parameters		Safety Analysis Parameters
a. $\alpha_D * (1-RF_D)$	$\leq$	α_D (least negative bounding value)
b. $\alpha_M + RF_M + B_M$	$\leq$	α_M (least negative bounding value)
c. $\beta_{eff} * (1-RF\beta)$	$\geq$	β_{eff} (minimum)
d. $\Delta\rho_{eject}*(1+RF_R)$	$\leq$	$\Delta\rho_{eject}$ (bounding)
e. $\Delta\rho_{scram}(t)*(1-RF_R)$	$\geq$	$\Delta\rho_{scram}(t)$ (bounding)
f. $F_Q*(1+RFF_Q)*(1+T)$	$\leq$	F_Q (bounding value rod ejection accident)
g. $\ell^* * (1-RF\ell_*)$	$\geq$	ℓ^* (minimum)

3.16 LOSS OF COOLANT ACCIDENT

3.16.1 DESCRIPTION OF THE ACCIDENT

The loss of coolant accident is defined as the rupture of the reactor coolant system piping or any line connected to the system, up to and including a double-ended guillotine rupture of the largest pipe. Ruptures of small flow area would cause coolant expulsion at a rate that would allow replacement at the same rate via the charging pumps and an orderly shutdown would be possible. A larger rupture would result in a net loss of reactor coolant inventory and a decreasing pressurizer water level and pressure. A reactor trip occurs and safety injection is actuated resulting in the injection of borated water into the reactor coolant system, isolation of the normal feedwater, and initiation of the auxiliary feedwater supply. When the reactor coolant system depressurizes to the Technical Specification [6] accumulator pressure value, the nitrogen bubble in the accumulator tanks expands, forcing additional water into the reactor coolant system. For large breaks, void formation in the core coolant during the initial blowdown phase results in almost immediate power reduction down to decay heat levels.

3.16.2 METHOD OF ANALYSIS

An analysis of the loss of coolant accident is performed to demonstrate the effectiveness of the emergency core cooling system (ECCS) to meet the criteria of 10CFR50.46 and in preventing radioactive releases which would violate the criteria of 10CFR100. This analysis includes the following:

1. A system blowdown analysis is performed to obtain the time-dependent behavior of core power, system pressure, flowrates, and other relevant variables. The blowdown phase lasts from break initiation until the time when the reactor coolant system pressure equals the containment pressure. The digital model employed in this calculation is a detailed representation of the primary and secondary systems. This includes the hot fuel assemblies and the remainder of the core; the reactor vessel downcomer, upper plenum, upper head, and lower plenum regions; the steam generators, pressurizer, and associated piping; and the safety injection systems. The model uses a flexible noding scheme to compute the space and time variations of the thermal-hydraulic conditions of the primary and secondary systems. Some of the phenomena which must be considered in the blowdown analysis are coolant flows between regions; heat transfer between primary and secondary fluids, and between system metal surfaces and fluids contacting them; the hydraulic interactions of system components such as reactor coolant pumps; fuel rod swelling and rupture; and the behavior of emergency core coolant as it is injected into a system undergoing rapid decompression.

2. An analysis of the core hot channel during blowdown is conducted using a detailed thermal-hydraulic model. □ These calculations must consider cross

flow between regions and any flow blockage calculated to occur as a result of clad swelling or rupture. □

3. The refill/reflood model continues the analysis from the end of blowdown until the entire core has been recovered. Due to the complexity of the phenomena occurring, empirical correlations of experimental data are used to define such variables as carryover fraction, heat transfer coefficients, natural convection in the secondary side of the steam generators, and slip flow in the ruptured loop cold leg nozzle.

4. A thermal calculation of the temperature transient in the hot fuel rod during blowdown and refill/reflood is also performed. □ Empirical correlations of measured data are employed to represent complex phenomena such as flow blockage due to clad swelling and rupture. Metal - water chemical reaction and radiation from the fuel rod surface are included in the hot rod model.

Detailed requirements for ECCS evaluation models are described in 10CFR Part 50, Appendix K. □

The spatial power distribution used in the ECCS evaluation analysis is chosen as the most limiting from those calculated to occur over the lifetime of the core. □ The initial hot spot peaking factor, F_Q, plays an important role in determining the severity of the worst cladding temperature response in the core. Because of the rapid degradation in heat transfer following the break, the radial temperature profile within the fuel rod tends to flatten out, thereby increasing the cladding temperature. In addition, larger values of F_Q will result in less effective heat transfer during the reflood period at the hot spot. Thus, a larger value of F_Q will produce a more severe cladding temperature response.

3.16.3 WPS SAFETY ANALYSIS RESULTS

WPS has not analyzed this accident. The current large break and small break LOCA analyses using the approved Westinghouse analysis methodologies for Kewaunee [8,9] are reviewed for each reload to determine their applicability to the current core design.

3.16.4 CYCLE SPECIFIC PHYSICS CALCULATIONS

□

a. NUCLEAR HEAT FLUX HOT CHANNEL FACTOR, F_Q

The maximum core F_Qs are assumed to remain within the current limits as defined in the Technical Specifications[6] for allowable combinations of axial offset and power level. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using the Power Distribution Control (PDC-II) scheme [7]. The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the F_Q limits are described in Section 3.17.

b. NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR, $F_{\Delta H}$

Calculations of $F_{\Delta H}$ are performed in accordance with the general procedures described in Section 2. Maximum core $F_{\Delta H}$s are verified to remain within Technical Specification[6] limits for allowable combinations of axial offset, power level, and control rod insertion. For Kewaunee, the continuous surveillance of the power distribution is accomplished with the ex-core detectors using a Power Distribution Control scheme, PDC-II [7]. The cycle specific physics calculations performed for the verification of the PDC-II scheme with respect to the $F_{\Delta H}$ limits are described in Section 3.17.

c. MAXIMUM ASSEMBLY AVERAGE PEAKING FACTOR, P_{HA}

Calculations of P_{HA} are performed at each depletion step throughout the cycle in accordance with the general procedures described in Section 2. The calculated value of P_{HA} is increased to include the model reliability factors and core tilt penalties. The value of P_{HA} is verified to remain below the value assumed in the current LOCA analysis.

d. AXIAL OFFSET AT 100% POWER, AO_{HFP}

Calculations of the axial offset at 100% power are performed at each depletion step throughout the cycle in accordance with the general procedures described in Section 2. The axial offset at 100% power must remain within the range assumed by the current LOCA analysis.

e. MAXIMUM CORE AVERAGE POWER IN LOW POWER ASSEMBLIES, P_{LP}

Calculations of the maximum core average power in the low power assemblies are performed at each depletion step throughout the cycle in accordance with the general procedures described in Section 2. The maximum core average power in the low power assemblies must remain below the value assumed by the current LOCA analysis.

f. MAXIMUM 95/95 POWER FOR THE HOT ROD, $P_{95/95}$

Calculations of the maximum 95/95 power for the hot rod are performed in accordance with the general procedures described in Section 2. The maximum 95/95 power for the hot rod must remain below the value assumed by the current LOCA analysis.

3.16.5 RELOAD SAFETY EVALUATION

The calculated values of F_Q and $F_{\Delta H}$ are increased to include the model reliability factors and core tilt penalties. These adjusted cycle specific values are compared to the current peaking factor limits defined in the Technical Specifications[6]. The details of this comparison are described in Section 3.17.

The calculated value of P_{HA} is increased to include the model reliability factors and core tilt penalties. The adjusted cycle specific value is compared to the current LOCA analysis value. The axial offset at 100% power is evaluated without additional uncertainty at all rods out depletion conditions throughout the cycle, since non-baseload conditions are accounted for by the F_O calculated using the PDC-II methodology. The axial offset at 100% power is compared to the current LOCA analysis value. The maximum core average power in the low power assemblies is evaluated at all rods out depletion conditions throughout the cycle with no uncertainties applied, since this parameter is a relative best-estimate comparison of the nominal power in the peripheral assemblies to those in the rest of the core. The maximum core average power in the low power assemblies is compared to the current LOCA analysis value. The maximum 95/95 power for the hot rod is determined based on the PDC-II methodology. The calculated value is increased to include the model reliability factors and core tilt penalties. The maximum 95/95 power for the hot rod is compared to the current LOCA analysis value.

Cycle Specific Parameters		Safety Analysis Parameters
a. $F_Q * (1+RFF_Q) * (1+T)$	$\leq$	Tech Spec Limit (Refer to Section 3.17)
b. $F_{\Delta H} * (1+RFF_{\Delta H}) * (1+T)$	$\leq$	Tech Spec Limit (Refer to Section 3.17)
c. $P_{HA} * (1+RFF_{\Delta H}) * (1+T)$	$\leq$	P_{HA} (LOCA bounding value)
d. $AO_{HFP,POS}$	$\leq$	AO_{HFP} (LOCA upper bound value)
$AO_{HFP,NEG}$	$\geq$	AO_{HFP} (LOCA lower bound value)
e. P_{LP}	$\leq$	P_{LP} (LOCA bounding value)
f. $P_{95/95}$	$\leq$	$P_{95/95}$ (LOCA bounding value)

3.17 POWER DISTRIBUTION CONTROL VERIFICATION

Calculations are performed at exposures ranging from beginning to end of cycle to verify the applicability of the power distribution control (PDC-II) scheme as defined in the Technical Specifications[6]. Specifically, the core peaking factors, $F_{\Delta H}$ and $F_Q(Z)$ are calculated at full power equilibrium core conditions and multiplied by conservative factors to verify that they remain within the limits as defined in the Technical Specification[6].

3.17.1 PEAKING FACTOR CALCULATIONS

Calculations of F_Q and $F_{\Delta H}$ are performed in accordance with the general procedures described in Sections 2.7 and 2.8. Cycle specific calculations are performed at full power equilibrium core conditions at exposures ranging from BOC to EOC. Statistical uncertainty factors derived from measured to predicted power distribution comparison analyses are conservatively applied to the calculated peaking factors [2].

Variations in the axial power distribution cause variations in $F_Q(Z)$ distribution, while the associated control rod motion causes variations in the $F_{\Delta H}$ distribution. To account for potential axial power distribution variations allowed by the Power Distribution Control (PDC-II) procedures, conservative factors called the V(Z) function are applied to the calculated full power equilibrium $F_Q(Z)$. The V(Z) function is determined by investigating the changes in $F_Q(Z)$ during core axial power perturbations, most of which are induced with combinations of power level and rod insertions changes. □

The maximum $F_{\Delta H}$ is chosen from a range of power distributions resulting from core maneuvers allowed by PDC-II in combination with control rod insertions allowed by Technical Specification Rod Insertion limits.

3.17.2 RELOAD SAFETY EVALUATIONS

The calculated peaking factors, $F_Q(Z)$ and $F_{\Delta H}$ are increased by statistical uncertainty factors and conservative reliability factors. □ $F_Q(Z)$ is further increased by the V(Z) function. □ Calculated cycle specific $F_Q(Z)$ and $F_{\Delta H}$ are compared to the current Technical Specification limits. □

CYCLE SPECIFIC PARAMETERS		SAFETY ANALYSIS PARAMETERS
a. $F_Q(Z) * (1+RFF_Q) * V(Z) * (1+T)$	$\leq$	F_Q (Technical Specification Limits)
b. $F_{\Delta H} * (1+RFF_{\Delta H}) * (1+T)$	$\leq$	$F_{\Delta H}$ (Technical Specification Limits)

4.0 REFERENCES

1. Letter from C. R. Steinhardt to U.S. NRC Document Control Desk, transmitting WPSC Topical Report WPSRSEM-NP-A, Rev. 2, "Reload Safety Evaluation Methods for Application to Kewaunee," dated 11/9/88.

2. Letter from E. W. James (WPS) to A. Schwencer (NRC) transmitting Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, topical report entitled, "Qualification of Reactor Physics Methods for Application to Kewaunee," dated October 1978.

3. Letter from A. Schwencer (NRC) to E. R. Mathews (WPS) transmitting Safety Evaluation Report by the Office of Nuclear Reactor Regulation: "Qualification of Reactor Physics Methods for Application to Kewaunee," dated October 22, 1979.

4. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Updated Safety Analysis Report. □

5. Kewaunee Nuclear Power Plant, Reload Safety Evaluation, Cycle 24, dated April 2000.

6. Wisconsin Public Service Corporation, Technical Specification for the Kewaunee Nuclear Power Plant, Docket Number 50-305.

7. R. J. Burnside and J. S. Holm, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II," XN-NF-77-57, Exxon Nuclear Company, Inc., dated January 1978.

8. Letter from M. L. Marchi (WPSC) to Document Control Desk (NRC), dated 6/4/97, "Implementation of ECCS Analysis Methods Reflecting Upper Plenum Injection."

9. Lee H., et al., "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code," WCAP-10054-P-A, dated August 1985.

□

APPENDIX A

Computer Program Overview

This section describes the WPS computer programs that are used to simulate the response of the Nuclear Steam Supply System (NSSS) and predict the thermal-hydraulic response of the hot coolant channel and hot spot in the core for the transients and accidents listed in Section 3.0.

The DYNODE-P □and/or RETRAN-3D programs are used to analyze the transient response of the Nuclear Steam Supply System (NSSS). □

The VIPRE-01 program is used to analyze the transient response of the hot channel in the core. VIPRE-01 provides a simulation of the thermal-hydraulic response of the coolant channels and associated fuel rods within the core.

The TOODEE-2 program is used to compute the transient temperature response of the hot fuel spot for certain accidents. TOODEE-2 provides a simulation of the hot fuel rod and associated coolant channel. This program is used only if the VIPRE-01 hot channel analysis yields a departure from nucleate boiling ratio (DNBR) which is less than the DNBR correlation MDNBR limit.

The sequence of calculations and interfaces of these programs are as follows: DYNODE-P and/or RETRAN-3D are run to obtain the core average heat flux and the reactor coolant system (RCS) thermal-hydraulic responses. The transient core average heat flux, core inlet coolant temperature, □RCS pressure, and core average flow responses along with the appropriate core spatial power distribution are input into VIPRE-01 to obtain the hot channel transient DNBR. Similar information is also input into TOODEE-2 to analyze the thermal response of the fuel rod, in those cases requiring fuel rod analysis.

CONTEMPT LT-28 and GOTHIC are used to analyze the containment thermal-hydraulic response to loss of coolant and main steam line break accidents. Mass and energy release data are provided to CONTEMPT LT-28 and GOTHIC from DYNODE-P or RETRAN-3D for the main streamline break accident. In the case of the loss of coolant accident, mass and energy data are provided from the Westinghouse LBLOCA codes.

APPENDIX B

NSSS Simulation, DYNODE-P

NSSS Simulation

The response of the NSSS of a PWR under transient and accident conditions is analyzed with the DYNODE-P program [B1]. This program includes a simulation of the components of a PWR NSSS which significantly influence the response of the system to transient conditions. □

The major features of the DYNODE-P □program are:

- Point and one-dimensional kinetics model for core power transients with major feedback mechanisms and decay heat represented. Initial subcritical core conditions can be modeled.
- Power forced mode option for hot channel analyses.
- Multinode radial fuel rod and multinode axial coolant channel representations in the core.
- Conservation of mass, energy, volume and boron concentration for the reactor coolant system (RCS). Conservation of momentum is optional.
- Detailed non-equilibrium pressurizer model including spray and heater systems and safety and relief valves.
- Explicit representation of the shell side of the steam generators including conservation of mass, energy, and volume.
- Representation of heat transfer with structural metal components of the NSSS.
- Explicit representation of the main steam system with isolation, check, dump, bypass, and turbine valves including conservation of mass, energy, momentum, and volume.
- Representation of the reactor protection and high-pressure safety injection systems.
- Representation of the major control systems.
- Provisions for stimulating a variety of transients and accidents including a break in the main steam system and asymmetric loop transients.

APPENDIX B (continued)

The base input parameters relating to the initial conditions are:

- Core geometry and initial thermal-hydraulic characteristics.
- Initial RCS pressure and pressurizer level, core inlet enthalpy, RCS flow distribution, and RCS boron concentration.
- Initial core power level and distribution.
- RCS, steam generator, and main stream system volume distributions and hydraulic characteristics.
- Initial steam generator pressures and levels and heat transfer data.

The input parameters required to obtain the transient responses are:

- Core kinetics characteristics including control rod motion.
- Reactor coolant system inertias, pressure loss coefficients, and pump hydraulic and torque characteristics.
- Control system characteristics.
- Main and auxiliary feedwater characteristics.
- Valve characteristics.
- Safety system characteristics.
- Transient power demand.

APPENDIX B (continued)

The major output consists of the following list of parameters that are edited at select time points during the transient:

- Core variables

  Average power - Fuel rod temperatures and heat flux - Coolant enthalpies, temperature, and mass - Kinetics variables including keff.

- RCS variables

  Mass, energy, and boron distribution of the coolant - Loop flow rates - Pressurizer pressure and level - Safety system variables - Pressure control system variables - Reactor coolant pump speeds, torques, and developed heads.

- Steam generator variables

  Pressure and levels - Masses - Heat loads - Feedwater and steam flows.

- Main steam system variables

  Pressure and mass distributions - Steam Flows

APPENDIX B (continued)

References

B1. R. C. Kern, L. W. Cress, D. L. Harrison, "DYNODE-P: A Nuclear Steam Supply System Transient Simulator for Pressurized Water Reactors."

APPENDIX C

Core and Fuel Thermal-Hydraulic Analysis, VIPRE-01

A thermal-hydraulic hot channel analysis is performed for transients in which DNB is a parameter of concern. VIPRE-01 is the WPS fuel thermal-hydraulic analysis computer code. □

VIPRE-01 [C1], a thermal-hydraulic computer code developed by Battelle Northwest under the sponsorship of EPRI, computes the flow and enthalpy distribution in the fuel assembly sub-channel for steady state or transient conditions. VIPRE-01 has undergone generic review by the NRC [C2] at the request of the Utility Group for Regulatory Applications (UGRA) and has been found acceptable for use in licensing applications. □ Utility specific submittals containing VIPRE-01 analyses have also been reviewed. □

The coolant regions analyzed by VIPRE-01 are divided into computational cells in which the conservation equations for mass, energy, and momentum for the fluid are solved. The independent variables; enthalpy, pressure, void fraction and velocity are averaged for each cell considering heat and momentum sources and sinks due to fixed solids such as fuel rods and grid spacers.

Heat transfer regimes from sub-cooled to super-heated forced convection including departure from nucleate boiling (DNB), and turbulent and diversion cross-flows are considered in the sub-channel analysis.

The basic input parameters are:

- Fuel rod and channel geometries
- Fluid thermal-hydraulic parameters
- Heat flux or power distribution
- Turbulent mixing parameters
- Transient forcing functions: Core inlet temperature, Core inlet flow, Core pressure, Core □heat flux.

The major time-dependent output parameters are:

- Sub-channel DNBR
- Sub-channel flow distribution
- Sub-channel fluid properties
- Fuel rod temperature distribution

The WPS safety analysis methodology establishes thermal margin on the basis of Kewaunee transient system analyses performed with DYNODE-P or RETRAN-3D and fuel thermal-hydraulic analyses performed with VIPRE-01. VIPRE-01 evaluates the MDNBR response in the hot channel using system analysis results for core heat flux, inlet temperature, pressure, and flow as input forcing functions. WPS uses the high thermal performance (HTP) [C3,C4] critical heat flux (CHF) correlation for all transients except main steam line break. For main steam line break the Westinghouse W-3 CHF correlation is used. □

APPENDIX C (continued)

A 1/8 core VIPRE-01 model is created to accurately account for assembly sub-channel and core-wide flow distribution effects. This model provides an overall analysis of the thermal-hydraulic behavior of the core. Individual or a limited number of fuel rods are modeled as the hot quarter assembly in the 1/8 core model. The remainder of the core in the 1/8 core model is modeled on an assembly-by-assembly basis. Each channel is divided axially into increments of equal lengths. Resistance to cross-flow and coolant mixing between adjacent channels is considered. Flow redistribution due to localized hydraulic resistances (e.g. spacer grids) is also predicted. The effects of local variations in power, fuel rod and fuel pellet fabrication and fuel rod spacing are also considered.

Power distributions in the 1/8 core VIPRE-01 models are derived from the WPS reload core design neutronic analysis. □

Once the highest-powered rod has been identified in the reload core design neutronic analysis, single sub-channels are represented in the hot quarter assembly, which contains this hottest rod. The hot assembly fuel rod power distribution is increased to a pre-determined $F_{\Delta H}$ limit. The hot sub-channel is identified as the one having the lowest MDNBR with that fuel rod power distribution.

In valid sub-channel analysis, sufficient detail of the regions surrounding the hot channel must be considered. □A □geometry must be described such that the hot channel is interior to a region of equivalently sized sub-channels. If the hot channel occurs on the edge of the hot quarter assembly, the geometry is modeled by representing the adjacent quarter assembly on a detailed sub-channel basis, similar to the hot quarter assembly, rather than a lumped channel basis.

□

APPENDIX C (continued)

References

C1. C. W. Stewart, et. al. "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores," BNWL Research Project 1584-1, Rev. 1, dated November 1983.

C2. Letter from Charles Rossi (NRC) to John Blaisdell (UGRA) dated May 1, 1986, "Safety Evaluation Report on EPRI NP-2511-CCM VIPRE-01."

C3. Letter from ML Marchi to NRC Document Control Desk, dated 9-4-97, "Justification for Use of High Thermal Performance Departure from Nucleate Boiling Correlation in the Kewaunee Reload Safety Evaluation Methods."

C4. Letter from R. J. Laufer (NRC) to M. L. Marchi (WPSC) dated 12/30/97, "Safety Evaluation for the High Thermal Performance (HTP) Departure from Nucleate Boiling, (DNB) Correlation and the Associated 1.14 MDNBR Safety Limit for Kewaunee Fuel".

APPENDIX D

Fuel Thermal-Hydraulic Analysis, TOODEE-2

In those cases where the MDNBR limit may be exceeded, transient clad and fuel temperatures of the hot fuel rod must also be obtained to assure that design criteria are met. The hot fuel rod analysis is performed using the TOODEE-2 code.

TOODEE-2 [D1] computes the thermal response of a fuel rod and associated coolant channel under transient conditions.

TOODEE-2 solves the conservation of energy equation in the fuel rod and the conservation of mass and energy in the coolant channel over the entire length of the core. The fuel-cladding gap model is the same as in the GAPCON [D2] programs. Material properties are computed based on local conditions. Cladding deformation is taken into account.

Zr-H_2O reaction is also considered as part of the total heat source. Heat transfer regimes from subcooled to superheated forced convection are considered. The major input parameters are:

- Fuel rod and coolant channel geometries and properties
- Initial power level and distribution
- Initial temperature distribution
- Time-dependent forcing functions: power, inlet flow, inlet temperature, saturation temperature

The major time-dependent output parameters are:

- Temperature distribution in fuel rod - Fuel rod surface and gap conditions - Energy in the fuel

Transient forcing functions for saturation temperature, power, inlet flow, and temperature are obtained from the system analysis performed with DYNODE-P or RETRAN-3D. TOODEE-2 provides a conservative calculation of the transient hot spot fuel temperature.

APPENDIX D (continued)

References

D1. G. N. Lauben, "TOODEE-2: A Two-Dimensional Time Dependent Fuel Element Thermal Analysis Program," NUREG-75/057, dated May 1975.

D2. G. R. Horn, F. E. Panisco, "Users Guide for GAPCON: A Computer Program to Predict Fuel-to-Cladding Heat Transfer Coefficient in Oxide Fuel Pins," HEDL-TME72-128, September 1972.

APPENDIX E

NSS Simulation, RETRAN-3D

RETRAN-3D[E1] is a best-estimate transient thermal-hydraulic code designed to analyze operational transients, small break loss-of-coolant accidents, and anticipated transients without scram, natural circulation, long-term transients, and events involving limited non-equilibrium conditions in light water reactors. It can also be used to analyze the steady state and transient response of any thermal-hydraulic system using water as the cooling fluid.

The field equations solved include the integral form of the one-dimensional, homogeneous equilibrium mixture equations for the conservation of continuity, momentum and energy, with options to also use a slip equation based on either dynamic or algebraic models, and a slip equation and a vapor mass equation.

Input models for the code are developed by assembling the basic building blocks consisting of fluid control volumes, flow paths or junctions, and components (e.g., heat conductors, pumps, energy sources, valves, and control systems) into a representative model of the system to be analyzed.

RETRAN-3D has options to use the following features:

- iterative solution of the steady-state field equations and the control system and other component equations
- an implicit, two-surface heat conduction model that allows internal power generation
- models for one-dimensional and point reactor kinetics
- trip logic
- control system models
- two sets of heat transfer correlations
- flow and pressure boundary conditions
- component models for pressurizes, steam separators, centrifugal pumps, valves, and accumulators
- special purpose models for modeling the movement of a temperature front or impurities

Implicit solution methods are used for the steady state and transient form of the field equations. Both linear and iterative nonlinear solutions of the transient field equations are available. The iterative transient solution method includes a number of algorithms used to provide automated time-step size control. APPENDIX E (continued)

The equation-of-state properties are generally valid between 100 and 6000 psi, allowing for the analysis over a wide range of operating conditions. Separate numerical algorithms (algebraic and finite difference in form) are also used for the solution of other equations (e.g., equation of state, heat conduction, neutron kinetics, control system, and pump behavior) as required.

The documentation for RETRAN-3D is provided in a four-volume code manual 1[E1]. The content of these volumes is:

1. Theory and Numerics Manual	discusses the theoretical development of the general equations, the constitutive relationships, and numerical solution techniques.
2. Programmer's Manual	presents the general coding philosophy, code installation and maintenance instructions, descriptions of the code modules, subroutines and data files, and information on auxiliary programs that can be used with the code.
3. User's Manual	provides the input data requirements and sample problem input and typical output data for each of the RETRAN modules.
4. Applications Manual	describes the verification and validation tasks for the code development phase and presents results of analyses used to evaluate and qualify the code for various applications.

RETRAN-02 was a predecessor to RETRAN-3D.

A Safety Evaluation Report (SER) has been issued by the NRC in which it was determined that RETRAN-02 is an acceptable program for use in licensing applications [E2].

However, from both theoretical considerations and a code user viewpoint, RETRAN-02 has limitations relative to performing some reactor transient analyses. The major theoretical limitations are:

- the requirement (excluding the auxiliary neutron void model) that the phase temperatures are equal for two-phase conditions,
- the numerical solution method in RETRAN-02 is limited to time-step sizes that are smaller than the Courant transport time,
- the limited nature of the vector momentum relative to multidimensional flow problems, and
- the effect of non-condensable gases on the fluid behavior cannot be accounted.

Some of the practical limitations of RETRAN-02 include:

- calculations of countercurrent flow behavior,
- the explicit nature of the numerical solution method,
- the discontinuous nature of some heat transfer and friction models which reduces the reliability of the code, and
- steady-state initialization of once-through steam generators and the two-phase region of some BWR models.
- insufficient qualification for some model applications (e.g., subcooled void model, dynamic slip, pressurizer, and bubble rise);
- improving existing models such as fluid properties for low pressure (<100 psia) and temperature (<100°F); and
- updating a model to reflect current practices such as using the 1979 ANS decay heat model.

The RETRAN-3D development program was designed to ease many of these limitations. The main objectives of the RETRAN-3D development program 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.

RETRAN-3D includes:

- 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 cocurrent 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 nonequilibrium phenomena (five-equation model),
- an option to include noncondensable gas flow,
- the option to analyze multidimensional neutron kinetics conditions, and
- an improved model to calculate steady-state initial conditions for low power situations.

Although all of these new features and capabilities have been designed into the code, RETRAN-3D still has the capability to perform RETRAN-02 analysis. The relationship between the two code versions is such that when RETRAN-3D is run in a RETRAN-02 mode, results from the two code versions are virtually identical. WPS intends to run RETRAN-3D in a RETRAN-02 mode, and will not implement the advanced capabilities of the code at this time.

A utility group has initiated licensing action for the purpose of obtaining generic review and approval of the RETRAN-3D code [E3]. WPS will consider utilizing some of the new capabilities of RETRAN-3D following this generic review and based on specific plant analysis needs and requirements.

APPENDIX E (continued)

References

E1. Paulson, M.P., et al., RETRAN-3D: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems EPRINP-7450, October 1996

E2. Letter from Cecil O. Thomas (NRC) to Dr. Thomas W. Schnatz (UGRA), Safety Evaluation Report: RETRAN-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," dated September 2, 1984.

E3. Letter from G. B. Swindlehurst (Duke Power Co.) to Mr. Timothy E. Collins (NRC) dated 7/8/98, transmitting request for review of RETRAN-3D.

APPENDIX F

Containment Thermal-Hydraulic Analysis, CONTEMPT-LT/28

The transient response of the containment during main steam line break and loss of coolant accident is analyzed with the CONTEMPT-LT/28 program [F1]. CONTEMPT-LT/28 can model a PWR dual dry containment with an annulus region. The program calculates the time variation of compartment pressures, temperatures, mass and energy inventories, heat structure temperature distributions, and energy exchange with adjacent compartments. Models are provided to describe fan cooler and cooling spray engineered safety systems.

The basic input parameters are:

- Containment compartment geometries and initial conditions
- Heat structure geometries and material
- Mass and energy addition rates
- Engineered safety system description
- Leakage rates
- Heat transfer coefficients

The major time-dependent output parameters are:

- Compartment pressure, temperature, mass and energy distributions
- Heat structure temperature distributions

The fluid mass and energy release from DYNODE-P or RETRAN-3D is used in conjunction with the CONTEMPT-LT/28 code to perform containment thermal hydraulic calculations for the main steam line break transient. CONTEMPT-LT28 is also used to analyze the containment thermal hydraulic response to a Loss of Coolant Accident. Loss of Coolant Accident mass and energy release data are calculated by Westinghouse LOCA codes.

References

F.1 L. L. Wheat, et al. "CONTEMPT-LT/28: A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," dated March 1979.

APPENDIX G
WPS Safety Analysis Results
[Consider all of Appendix G as highlighted]

FIGURE	TITLE
3.1-1	Uncontrolled RCCA Withdrawal From A Sub-Critical Condition Reactor Power vs. Time
3.1-2	Uncontrolled RCCA Withdrawal From A Sub-Critical Condition Heat Flux vs. Time
3.1-3	Uncontrolled RCCA Withdrawal From A Sub-Critical Condition Fuel Temperature vs. Time
3.1-4	Uncontrolled RCCA Withdrawal From A Sub-Critical Condition Tave vs. Time
3.1-5	Uncontrolled RCCA Withdrawal From A Sub-Critical Condition Hot Spot Clad Temperature vs. Time
3.2-1	Uncontrolled RCCA Withdrawal - Fast Rate 100% Power - High Pressure Reactor Power vs. Time
3.2-2	Uncontrolled RCCA Withdrawal - Fast Rate 100% Power - High Pressure Pressurizer Pressure vs. Time
3.2-3	Uncontrolled RCCA Withdrawal - Fast Rate 100% Power - High Pressure Tave vs. Time
3.2-4	Uncontrolled RCCA Withdrawal - Fast Rate 100% Power Minimum DNBR vs. Time
3.2-5	Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure Reactor Power vs. Time
3.2-6	Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure Pressurizer Pressure vs. Time
3.2-7	Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure Tave vs. Time
3.2-8	Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure Minimum DNBR vs. Time
3.2-9	Uncontrolled RCCA Withdrawal - Fast Rate 60% Power - High Pressure Reactor Power vs. Time

APPENDIX G
WPS Safety Analysis Results
[Consider all of Appendix G as highlighted]

FIGURE	TITLE
3.2-10	Uncontrolled RCCA Withdrawal - Fast Rate 60% Power - High Pressure Pressurizer Pressure vs. Time
3.2-11	Uncontrolled RCCA Withdrawal - Fast Rate 60% Power - High Pressure Tave vs. Time
3.2-12	Uncontrolled RCCA Withdrawal - Fast Rate 60% Power Minimum DNBR vs. Time
3.2-13	Uncontrolled RCCA Withdrawal - Slow Rate 60% Power - High Pressure Reactor Power vs. Time
3.2-14	Uncontrolled RCCA Withdrawal - Slow Rate 60% Power - High Pressure Pressurizer Pressure vs. Time
3.2-15	Uncontrolled RCCA Withdrawal - Slow Rate 60% Power - High Pressure Tave vs. Time
3.2-16	Uncontrolled RCCA Withdrawal - Slow Rate 60% Power Minimum DNBR vs. Time
3.5-1	CVCS Malfunction – Dilution at Power - Manual Control - High Pressure Reactor Power vs. Time
3.5-2	CVCS Malfunction – Dilution at Power - Manual Control - High Pressure Pressurizer Pressure vs. Time
3.5-3	CVCS Malfunction – Dilution at Power - Manual Control - High Pressure Tave vs. Time
3.5-4	CVCS Malfunction – Dilution at Power - Manual Control - High Pressure Heat Flux vs. Time
3.5-5	CVCS Malfunction – Dilution at Power - Manual Control - High Pressure Minimum DNBR vs. Time
3.6-1	Startup Of An Inactive Reactor Coolant Loop Tinlet vs. Time
3.6-2	Startup Of An Inactive Reactor Coolant Loop Tave vs. Time

APPENDIX G
WPS Safety Analysis Results
[Consider all of Appendix G as highlighted]

FIGURE	TITLE
3.6-3	Startup Of An Inactive Reactor Coolant Loop Reactor Power vs. Time
3.6-4	Startup Of An Inactive Reactor Coolant Loop Pressurizer Pressure vs. Time
3.6-5	Startup Of An Inactive Reactor Coolant Loop Heat Flux vs. Time
3.7-1	Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control Reactor Power vs. Time
3.7-2	Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control Pressurizer Pressure vs. Time
3.7-3	Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control Tave vs. Time
3.7-4	Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control Delta T Loop vs. Time
3.7-5	Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control Minimum DNBR vs. Time
3.7-6	Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control Reactor Power vs. Time
3.7-7	Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control Pressurizer Pressure vs. Time
3.7-8	Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control Tave vs. Time
3.7-9	Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control Delta T Loop vs. Time
3.7-10	Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control Minimum DNBR vs. Time
3.8-1	Excessive Load Increase – BOC Manual Control Reactor Power vs. Time

APPENDIX G
WPS Safety Analysis Results
Consider all of Appendix G as highlighted

FIGURE	TITLE
3.8-2	Excessive Load Increase – BOC Manual Control Pressurizer Pressure vs. Time
3.8-3	Excessive Load Increase – BOC Manual Control Delta T Loop vs. Time
3.8-4	Excessive Load Increase – BOC Manual Control Minimum DNBR vs. Time
3.8-5	Excessive Load Increase – EOC Manual Control Reactor Power vs. Time
3.8-6	Excessive Load Increase – EOC Manual Control Pressurizer Pressure vs. Time
3.8-7	Excessive Load Increase – EOC Manual Control Delta T Loop vs. Time
3.8-8	Excessive Load Increase – EOC Manual Control Minimum DNBR vs. Time
3.8-9	Excessive Load Increase – BOC Auto Control Reactor Power vs. Time
3.8-10	Excessive Load Increase – BOC Auto Control Pressurizer Pressure vs. Time
3.8-11	Excessive Load Increase – BOC Auto Control Delta T Loop vs. Time
3.8-12	Excessive Load Increase – BOC Auto Control Tave vs. Time
3.8-13	Excessive Load Increase – BOC Auto Control Minimum DNBR vs. Time
3.8-14	Excessive Load Increase – EOC Auto Control Reactor Power vs. Time
3.8-15	Excessive Load Increase – EOC Auto Control Pressurizer Pressure vs. Time

APPENDIX G
WPS Safety Analysis Results
[Consider all of Appendix G as highlighted]

FIGURE	TITLE
3.8-16	Excessive Load Increase – EOC Auto Control Delta T Loop vs. Time
3.8-17	Excessive Load Increase – EOC Auto Control Tave vs. Time
3.8-18	Excessive Load Increase – EOC Auto Control Minimum DNBR vs. Time
3.9-1	Loss of External Electrical Load - BOC Auto Control Reactor Power vs. Time
3.9-2	Loss of External Electrical Load - BOC Auto Control Tinlet vs. Time
3.9-3	Loss of External Electrical Load - BOC Auto Control Pressurizer Pressure vs. Time
3.9-4	Loss of External Electrical Load - BOC Auto Control Pressurizer Water Level vs. Time
3.9-5	Loss of External Electrical Load - BOC Auto Control Minimum DNBR vs. Time
3.9-6	Loss of External Electrical Load - EOC Auto Control Reactor Power vs. Time
3.9-7	Loss of External Electrical Load - EOC Auto Control Tinlet vs. Time
3.9-8	Loss of External Electrical Load - EOC Auto Control Pressurizer Pressure vs. Time
3.9-9	Loss of External Electrical Load - EOC Auto Control Pressurizer Water Level vs. Time
3.9-10	Loss of External Electrical Load - EOC Auto Control Minimum DNBR vs. Time
3.9-11	Loss of External Electrical Load - BOC Manual Control - High Pressure Reactor Power vs. Time

APPENDIX G
WPS Safety Analysis Results
[Consider all of Appendix G as highlighted]

FIGURE	TITLE
3.9-12	Loss of External Electrical Load - BOC Manual Control - High Pressure Tinlet vs. Time
3.9-13	Loss of External Electrical Load - BOC Manual Control - High Pressure Pressurizer Pressure vs. Time
3.9-14	Loss of External Electrical Load - BOC Manual Control - High Pressure Pressurizer Water Level vs. Time
3.9-15	Loss of External Electrical Load - BOC Manual Control Minimum DNBR vs. Time
3.9-16	Loss of External Electrical Load - EOC Manual Control - High Pressure Reactor Power vs. Time
3.9-17	Loss of External Electrical Load - EOC Manual Control - High Pressure Tinlet vs. Time
3.9-18	Loss of External Electrical Load - EOC Manual Control - High Pressure Pressurizer Pressure vs. Time
3.9-19	Loss of External Electrical Load - EOC Manual Control - High Pressure Pressurizer Water Level vs. Time
3.9-20	Loss of External Electrical Load - EOC Manual Control Minimum DNBR vs. Time
3.10-1	Loss of Normal Feedwater - High Pressure Tave vs. Time
3.10-2	Loss of Normal Feedwater - High Pressure Pressurizer Liquid Volume vs. Time
3.10-3	Loss of Normal Feedwater - High Pressure SG A Wide Range Level vs. Time
3.10-4	Loss of Normal Feedwater - High Pressure SG B Wide Range Level vs. Time
3.10-5	Loss of Normal Feedwater - High Pressure Pressurizer Pressure vs. Time

APPENDIX G
WPS Safety Analysis Results
Consider all of Appendix G as highlighted

FIGURE	TITLE
3.11-1	Loss of Reactor Coolant Flow - Two Pump Trip Core Flow vs. Time
3.11-2	Loss of Reactor Coolant Flow - Two Pump Trip Reactor Power vs. Time
3.11-3	Loss of Reactor Coolant Flow - Two Pump Trip Heat Flux vs. Time
3.11-4	Loss of Reactor Coolant Flow - Two Pump Trip Minimum DNBR vs. Time
3.11-5	Loss of Reactor Coolant Flow – Underfrequency Trip Core Flow vs. Time
3.11-6	Loss of Reactor Coolant Flow – Underfrequency Trip Reactor Power vs. Time
3.11-7	Loss of Reactor Coolant Flow – Underfrequency Trip Heat Flux vs. Time
3.11-8	Loss of Reactor Coolant Flow – Underfrequency Trip Minimum DNBR vs. Time
3.12-1	Loss of Reactor Coolant Flow - Locked Rotor - High Pressure Core Flow vs. Time
3.12-2	Loss of Reactor Coolant Flow - Locked Rotor - High Pressure Heat Flux vs. Time
3.12-3	Loss of Reactor Coolant Flow - Locked Rotor - High Pressure Pressurizer Pressure vs. Time
3.12-4	Loss of Reactor Coolant Flow - Locked Rotor Minimum DNBR vs. Time
3.12-5	Loss of Reactor Coolant Flow - Locked Rotor - High Pressure Hot Spot Clad Temperature vs. Time
3.14-1	Main Steam Line Break – Limiting Core Case 14NYY0 Tave vs. Time

APPENDIX G
WPS Safety Analysis Results
[Consider all of Appendix G as highlighted]

FIGURE	TITLE
3.14-2	Main Steam Line Break – Limiting Core Case 14NYY0 Pressurizer Pressure vs. Time
3.14-3	Main Steam Line Break – Limiting Core Case 14NYY0 Heat Flux vs. Time
3.14-4	Main Steam Line Break – Limiting Core Case 14NYY0 SG B Break Flow vs. Time
3.14-5	Main Steam Line Break – Limiting Core Case 14NYY0 Reactivity vs. Time
3.14-6	Main Steam Line Break – Limiting Containment Pressure Case 14RYY2 Containment Pressure vs. Time
3.14-7	Main Steam Line Break – Limiting Containment Temperature Case 11NYY0 Containment Temperature vs. Time
3.15-1	RCCA Ejection - BOC Full Power Reactor Power vs. Time
3.15-2	RCCA Ejection - BOC Full Power Integral Reactor Power vs. Time
3.15-3	RCCA Ejection - BOC Full Power Tave vs. Time
3.15-4	RCCA Ejection - BOC Zero Power Reactor Power vs. Time
3.15-5	RCCA Ejection - BOC Zero Power Integral Reactor Power vs. Time
3.15-6	RCCA Ejection - BOC Zero Power Tave vs. Time
3.15-7	RCCA Ejection - EOC Full Power Reactor Power vs. Time
3.15-8	RCCA Ejection - EOC Full Power Integral Reactor Power vs. Time

APPENDIX G
WPS Safety Analysis Results
[Consider all for Appendix G as highlighted]

FIGURE	TITLE
3.15-9	RCCA Ejection - EOC Full Power Tave vs. Time
3.15-10	RCCA Ejection - EOC Zero Power Reactor Power vs. Time
3.15-11	RCCA Ejection - EOC Zero Power Integral Reactor Power vs. Time
3.15-12	RCCA Ejection - EOC Zero Power Tave vs. Time

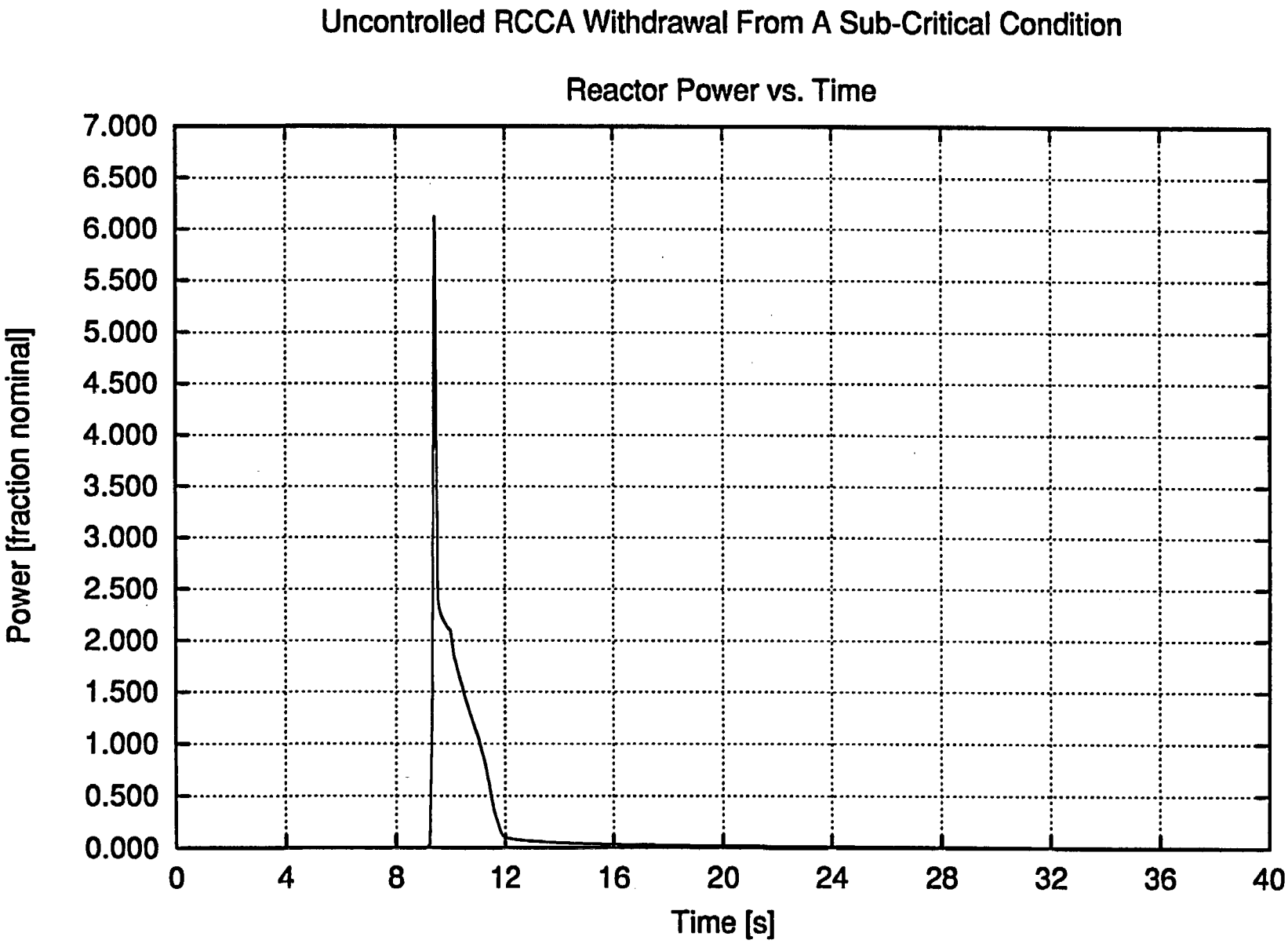
Uncontrolled RCCA Withdrawal From A Sub-Critical Condition
Reactor Power vs. Time
Power [fraction nominal]
7.000
6.500
6.000
5.500
5.000
4.500
4.000
3.500
3.000
2.500
2.000
1.500
1.000
0.500
0.000
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.1-1

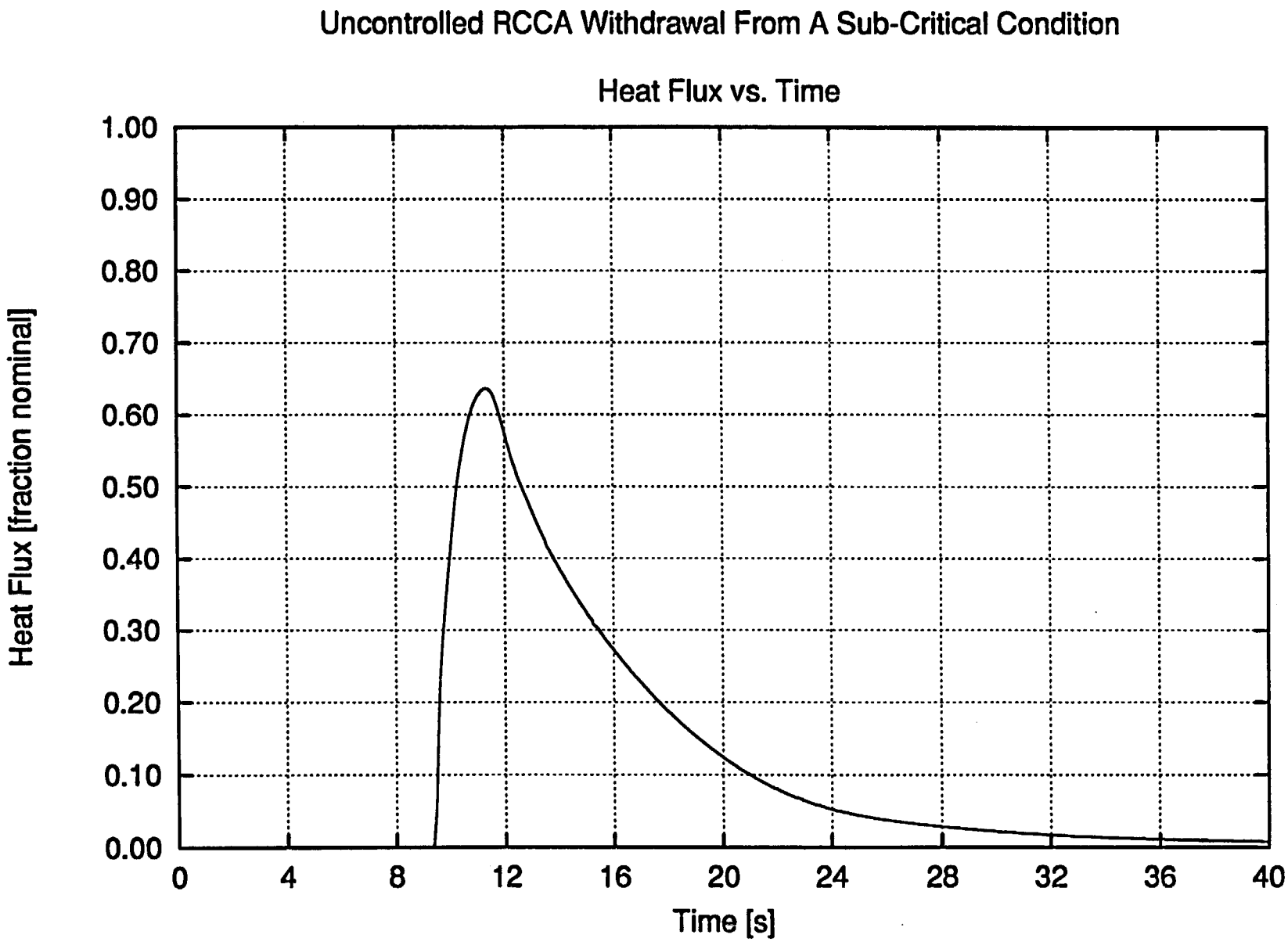
Uncontrolled RCCA Withdrawal From A Sub-Critical Condition
Heat Flux vs. Time
Heat Flux [fraction nominal]
1.00
0.90
0.80
0.70
0.60
0.50
0.40
0.30
0.20
0.10
0.00
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.1-2

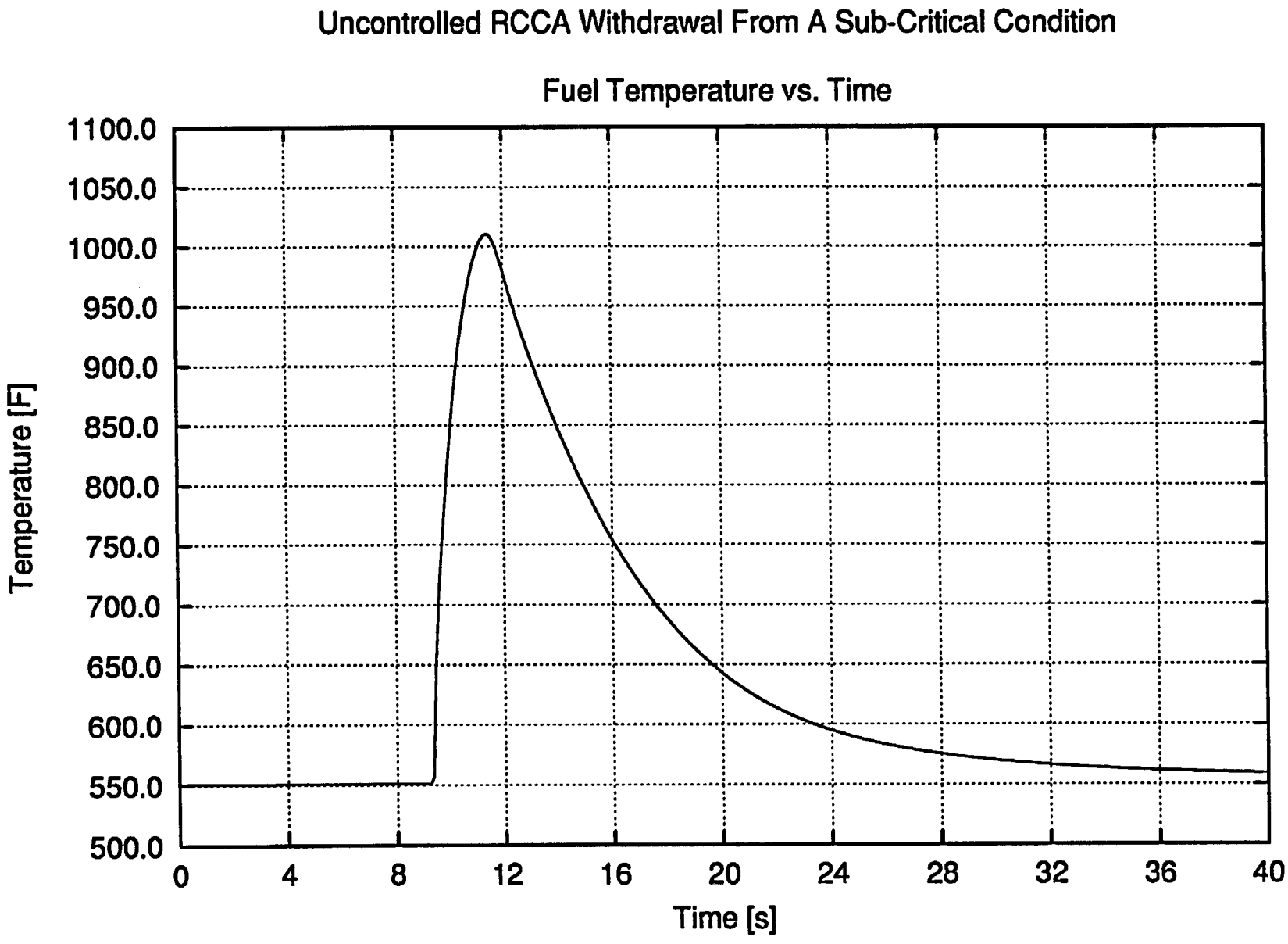
Uncontrolled RCCA Withdrawal From A Sub-Critical Condition
Fuel Temperature vs. Time
Temperature [F]
1100.0
1050.0
1000.0
950.0
900.0
850.0
800.0
750.0
700.0
650.0
600.0
550.0
500.0
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.1-3

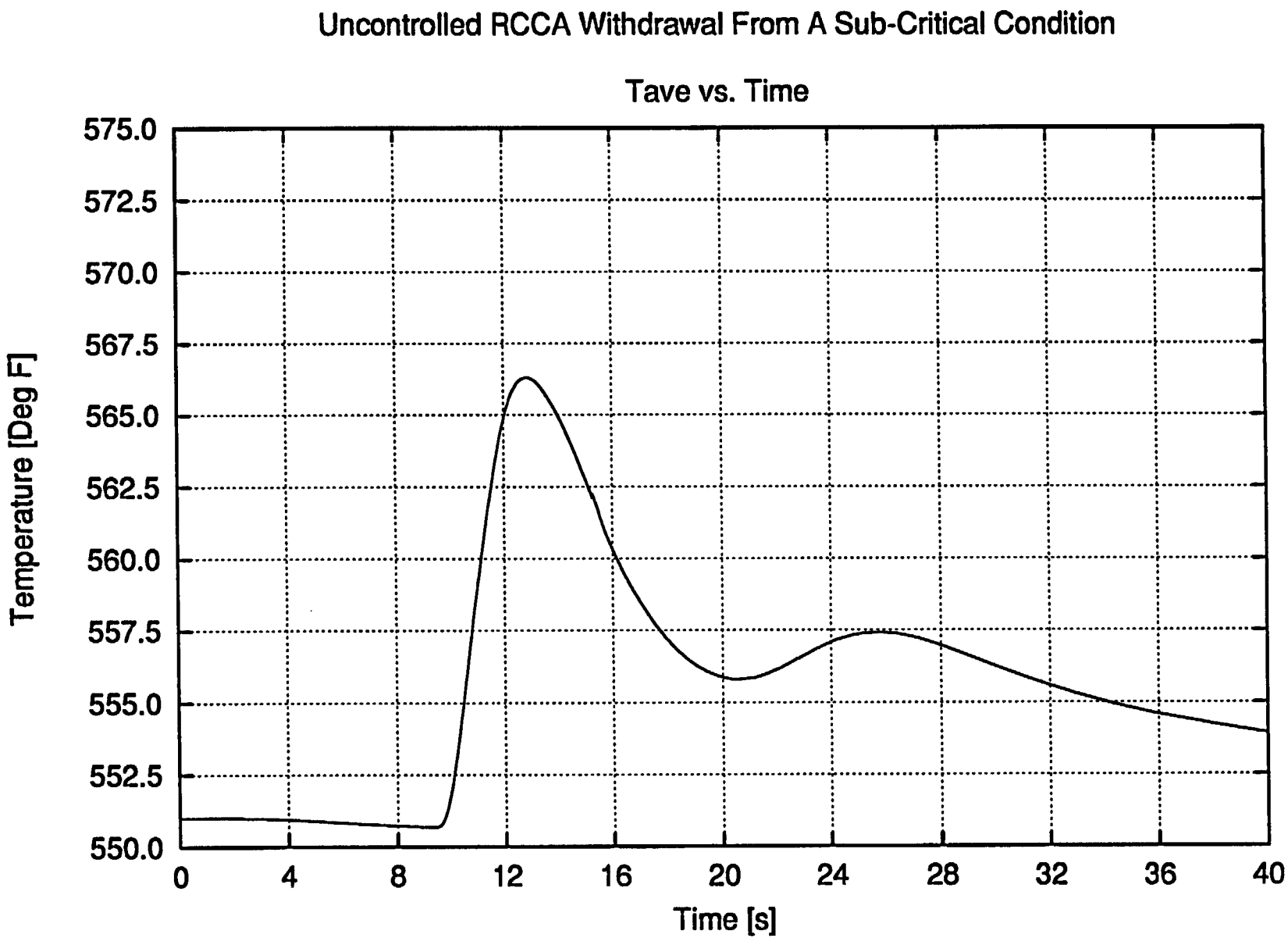
Uncontrolled RCCA Withdrawal From A Sub-Critical Condition
Tave vs. Time
Temperature [Deg F]
575.0
572.5
570.0
567.5
565.0
562.5
560.0
557.5
555.0
552.5
550.0
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.1-4

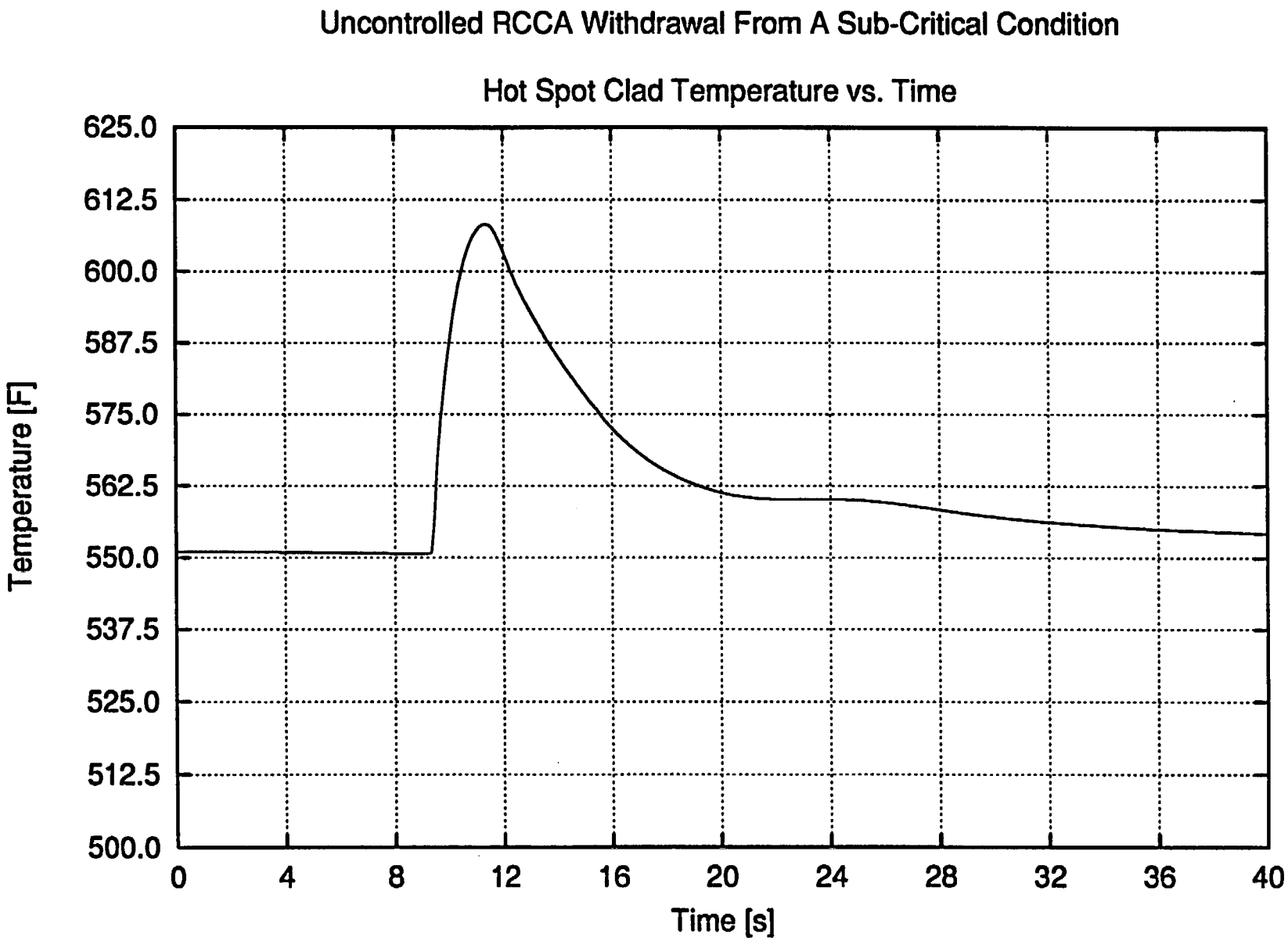
Uncontrolled RCCA Withdrawal From A Sub-Critical Condition
Hot Spot Clad Temperature vs. Time
Temperature [F]
625.0
612.5
600.0
587.5
575.0
562.5
550.0
537.5
525.0
512.5
500.0
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.1-5

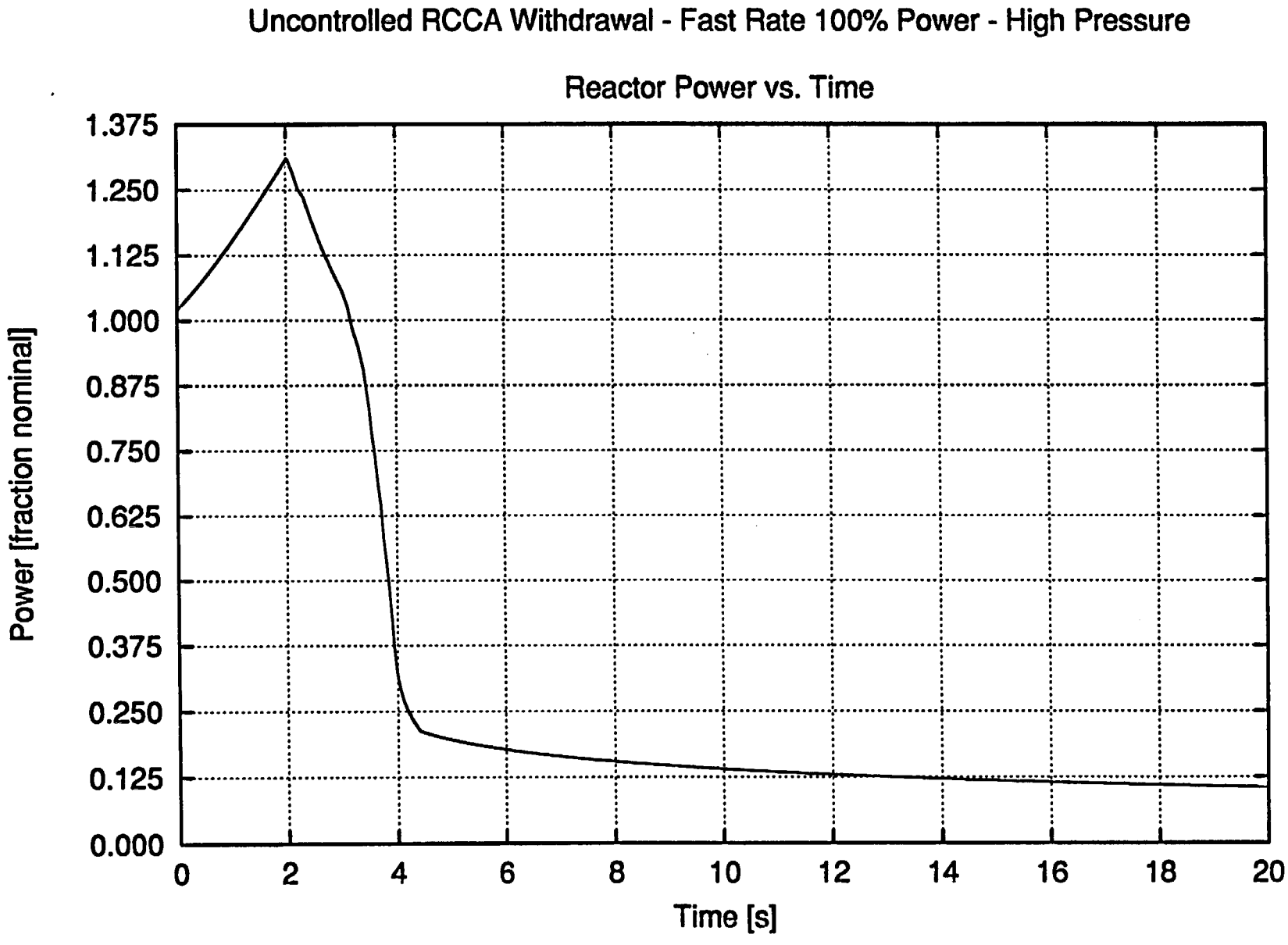
Uncontrolled RCCA Withdrawal - Fast Rate 100% Power - High Pressure
Reactor Power vs. Time
Power [fraction nominal]
1.375
1.250
1.125
1.000
0.875
0.750
0.625
0.500
0.375
0.250
0.125
0.000
0
2
4
6
8
10
12
14
16
18
20
Time [s]

Figure 3.2-1

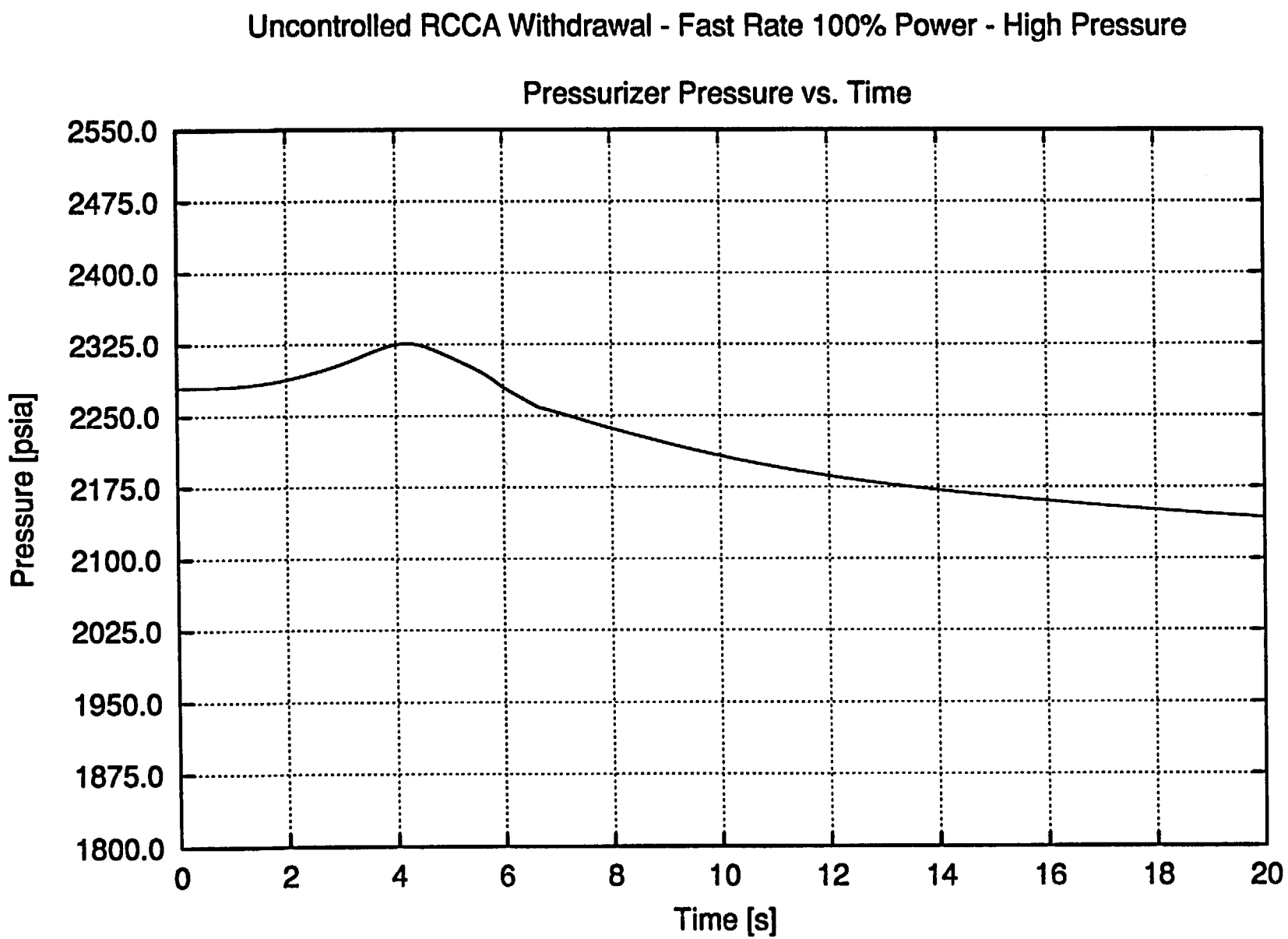
Uncontrolled RCCA Withdrawal - Fast Rate 100% Power - High Pressure
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
2
4
6
8
10
12
14
16
18
20
Time [s]

Figure 3.2-2

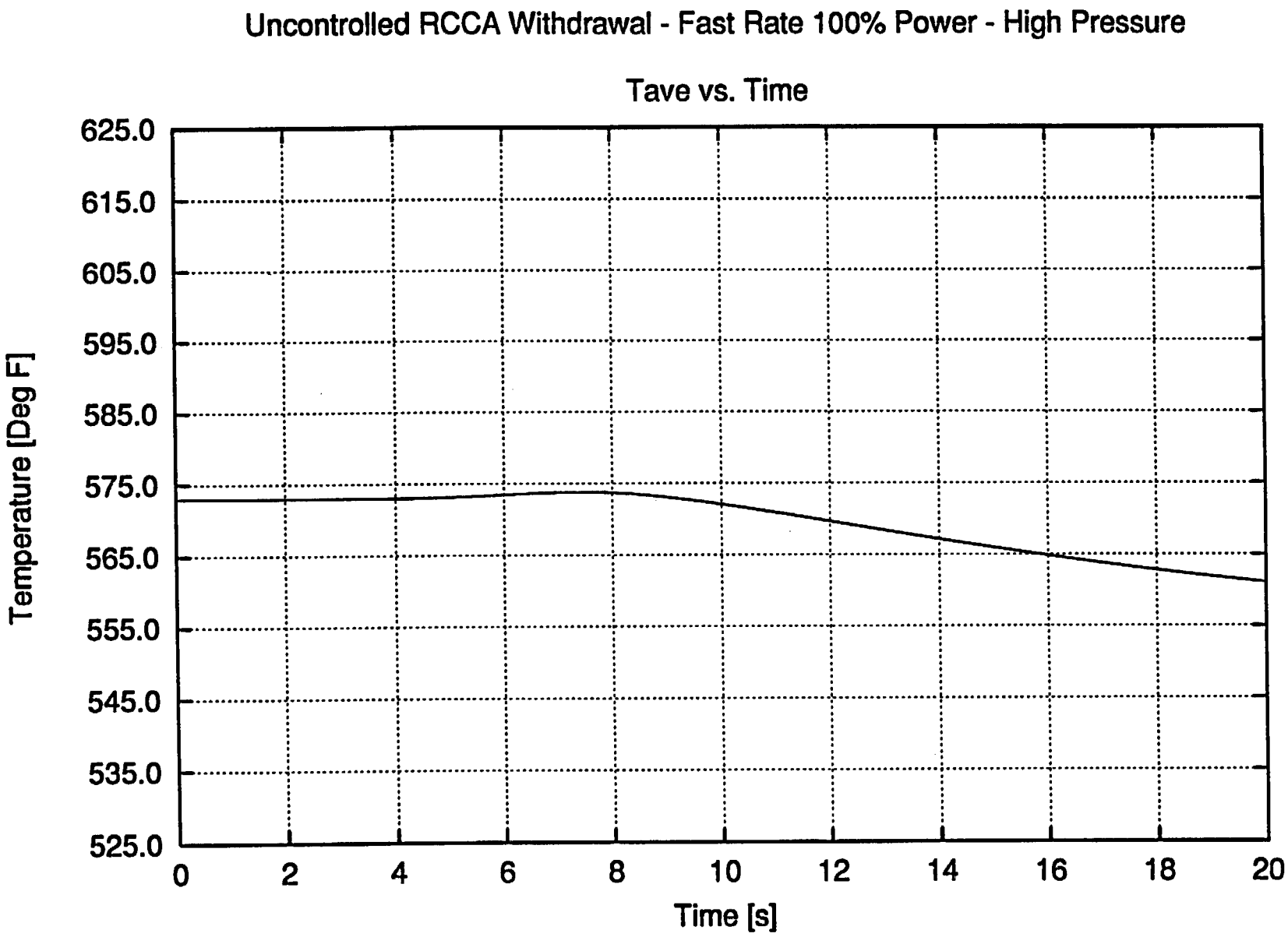
Uncontrolled RCCA Withdrawal - Fast Rate 100% Power - High Pressure
Tave vs. Time
Temperature [Deg F]
625.0
615.0
605.0
595.0
585.0
575.0
565.0
555.0
545.0
535.0
525.0
0
2
4
6
8
10
12
14
16
18
20
Time [s]

Figure 3.2-3

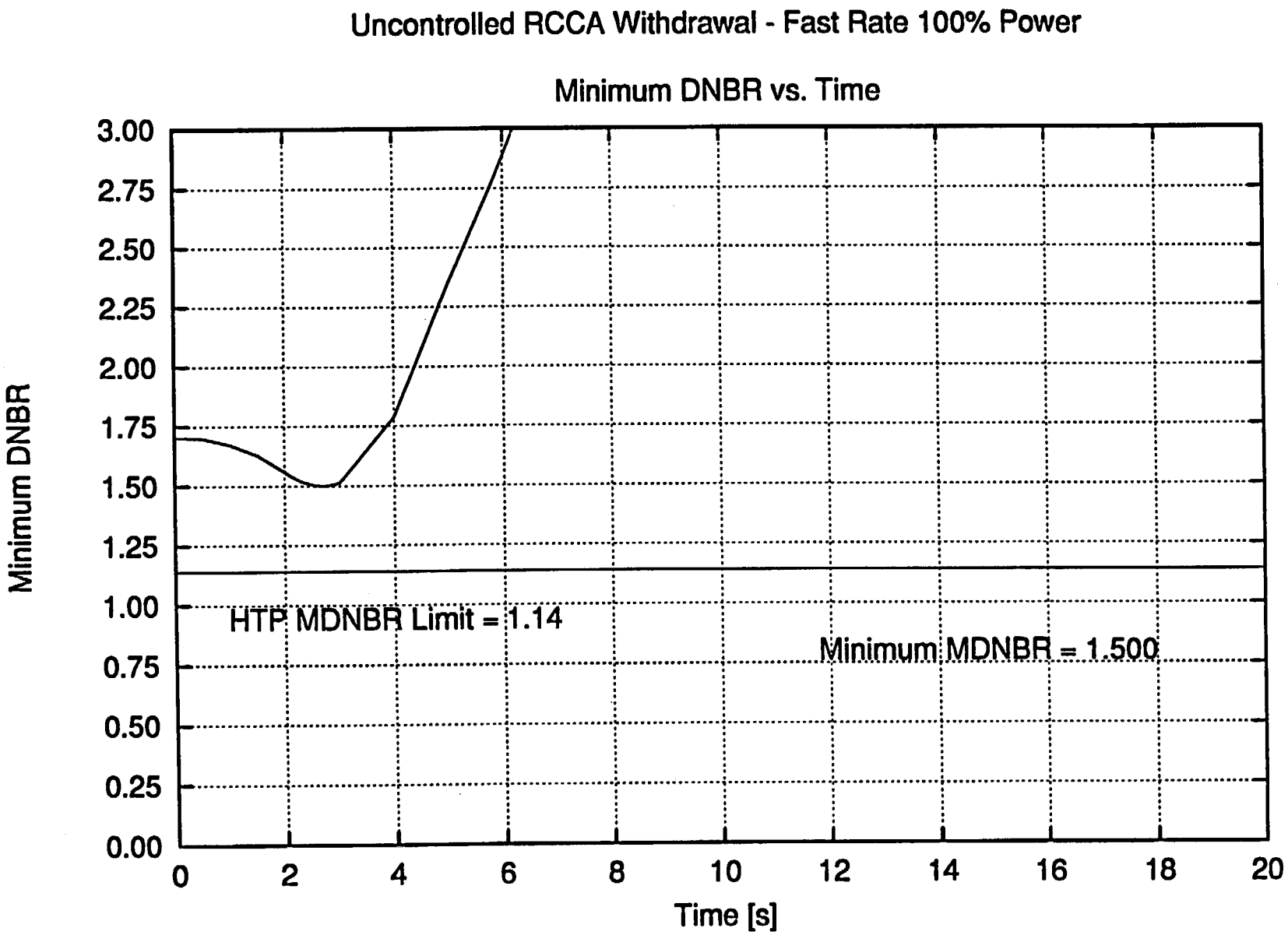
Uncontrolled RCCA Withdrawal - Fast Rate 100% Power
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0
2
4
6
8
10
12
14
16
18
20
Time [s]
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.500

Figure 3.2-4

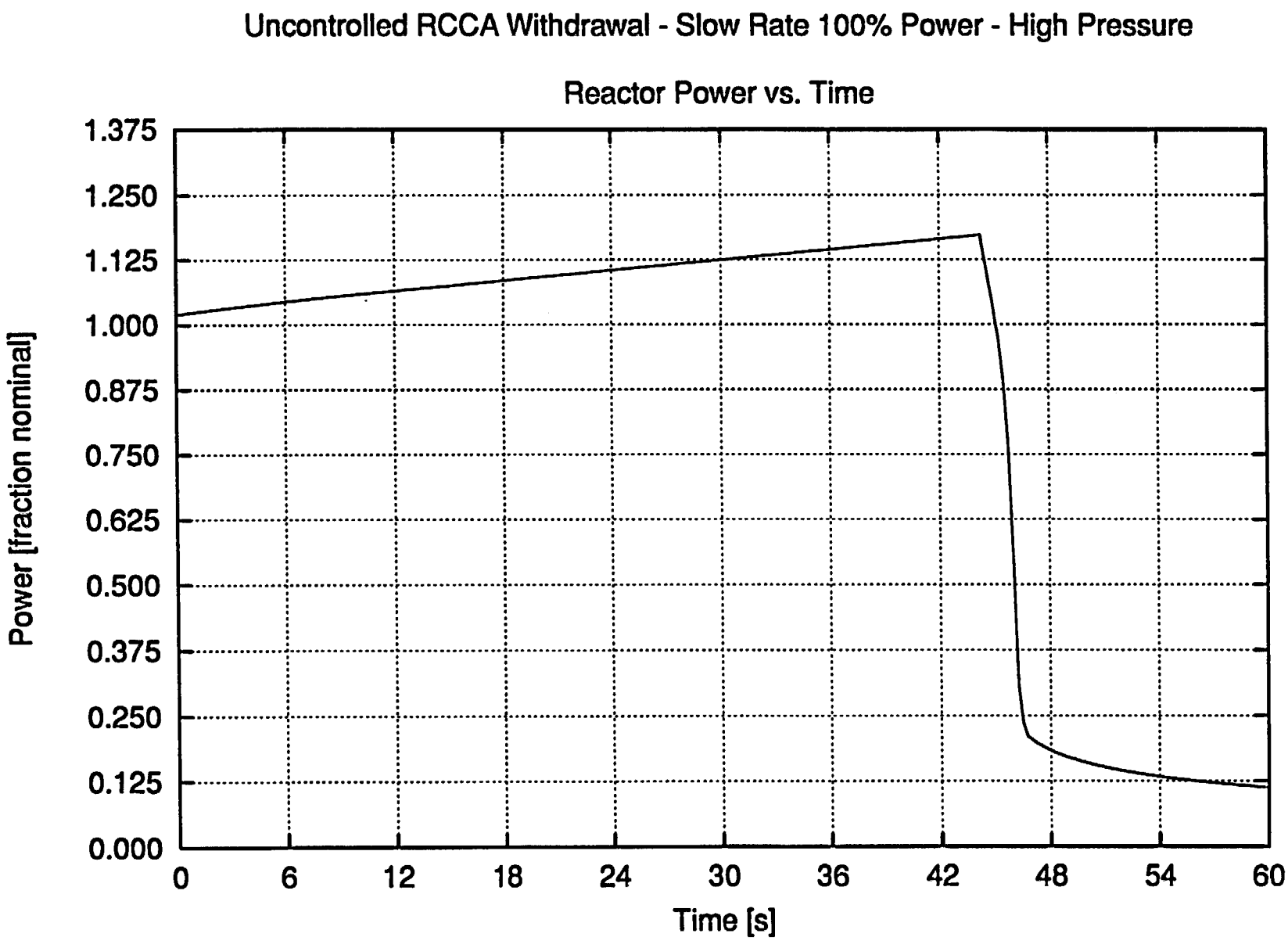
Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure
Reactor Power vs. Time
Power [fraction nominal]
1.375
1.250
1.125
1.000
0.875
0.750
0.625
0.500
0.375
0.250
0.125
0.000
0
6
12
18
24
30
36
42
48
54
60
Time [s]

Figure 3.2-5

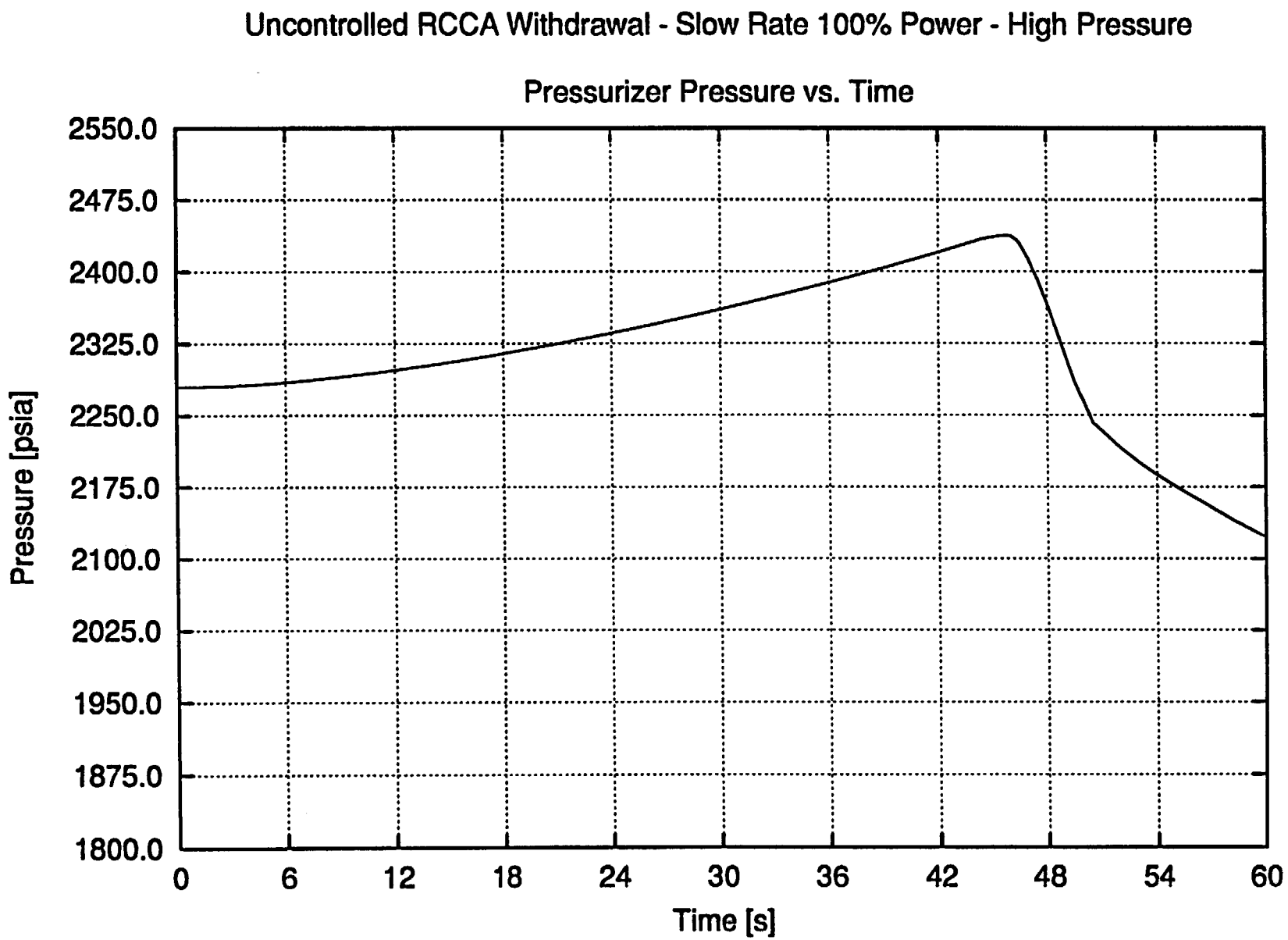
Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
6
12
18
24
30
36
42
48
54
60
Time [s]

Figure 3.2-6

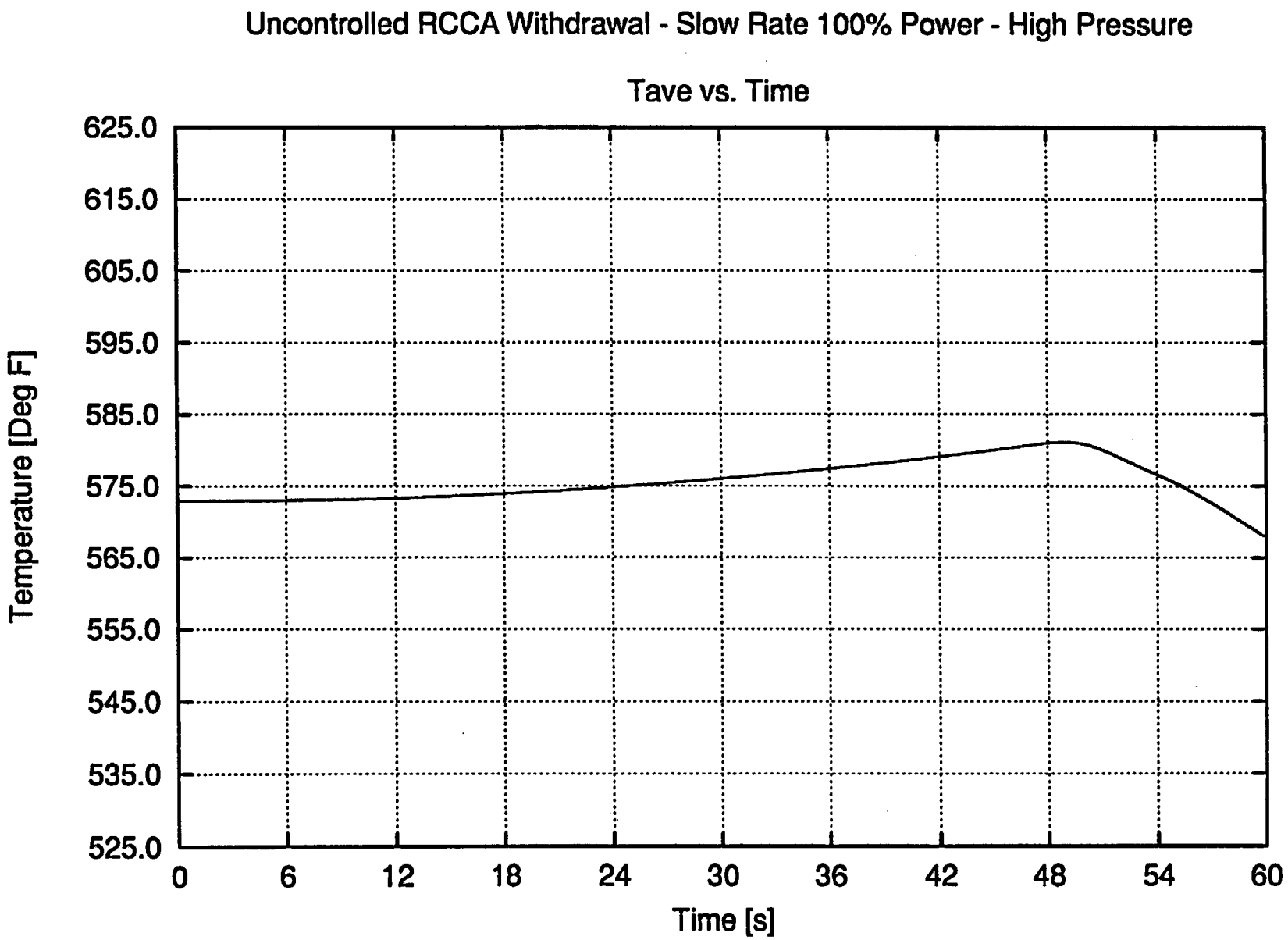
Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure
Tave vs. Time
Temperature [Deg F]
625.0
615.0
605.0
595.0
585.0
575.0
565.0
555.0
545.0
535.0
525.0
0
6
12
18
24
30
36
42
48
54
60
Time [s]

Figure 3.2-7

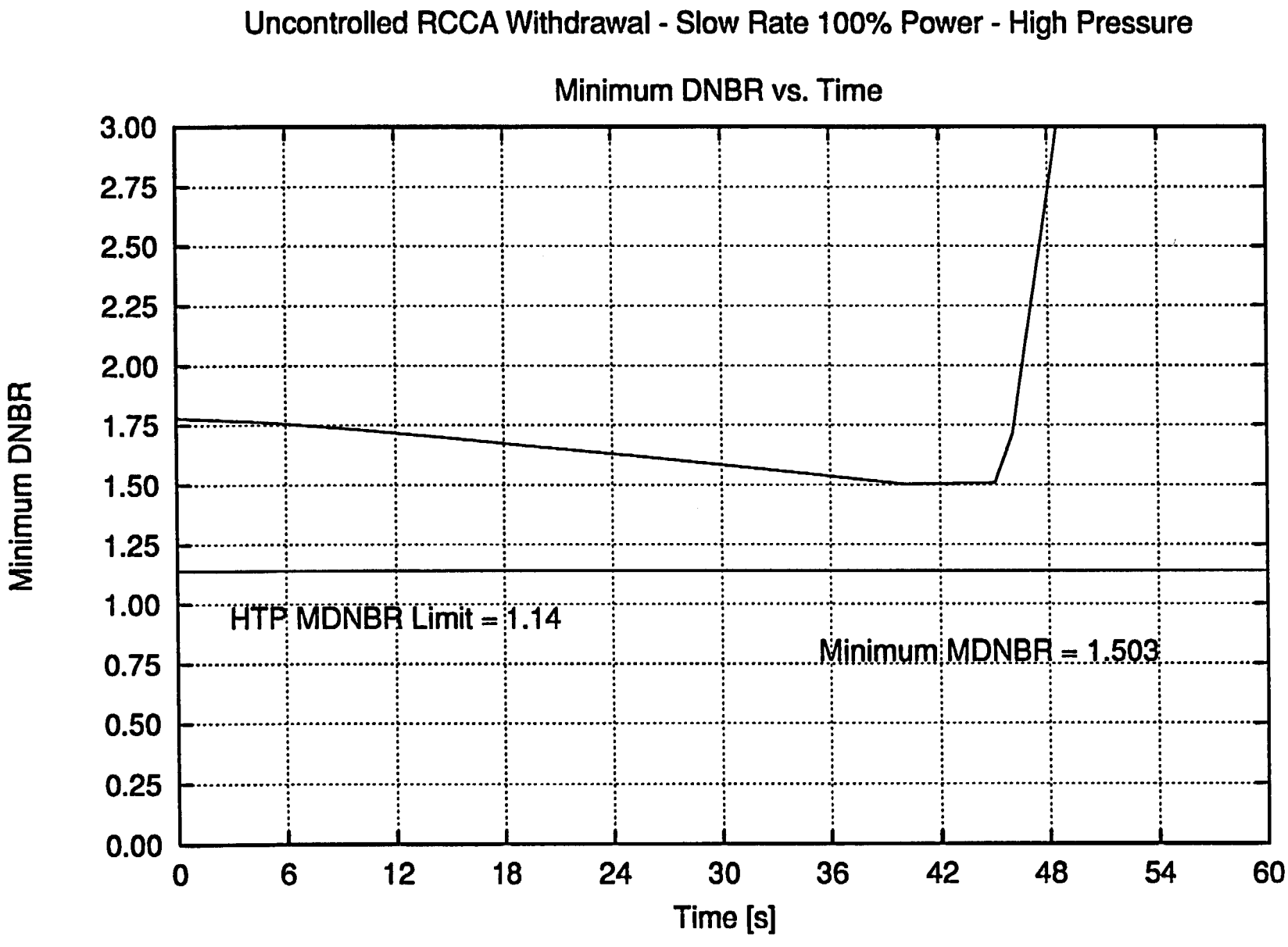
Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0
6
12
18
24
30
36
42
48
54
60
Time [s]
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.503

Figure 3.2-8

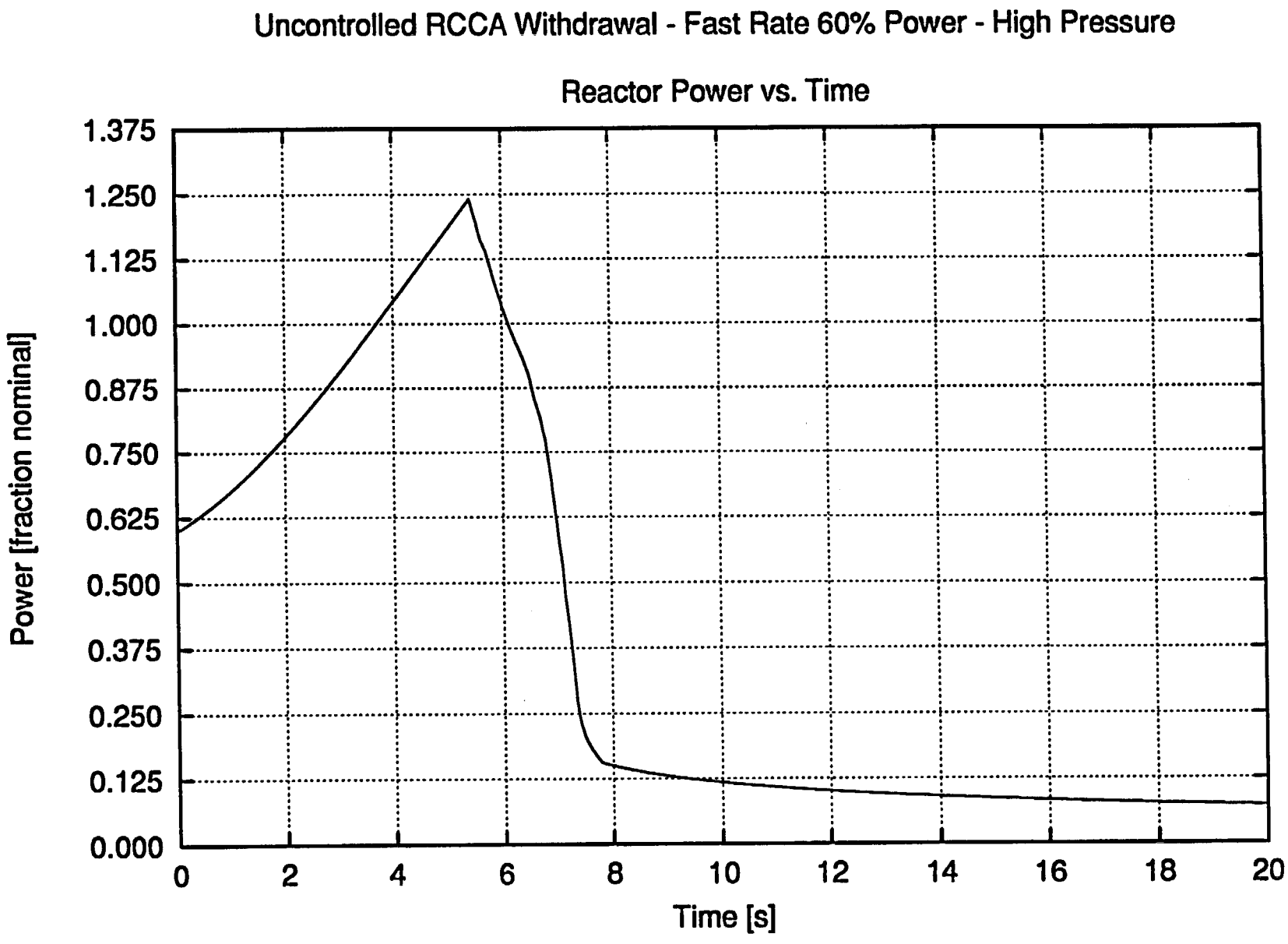
Uncontrolled RCCA Withdrawal - Fast Rate 60% Power - High Pressure
Reactor Power vs. Time
Power [fraction nominal]
1.375
1.250
1.125
1.000
0.875
0.750
0.625
0.500
0.375
0.250
0.125
0.000
0
2
4
6
8
10
12
14
16
18
20
Time [s]

Figure 3.2-9

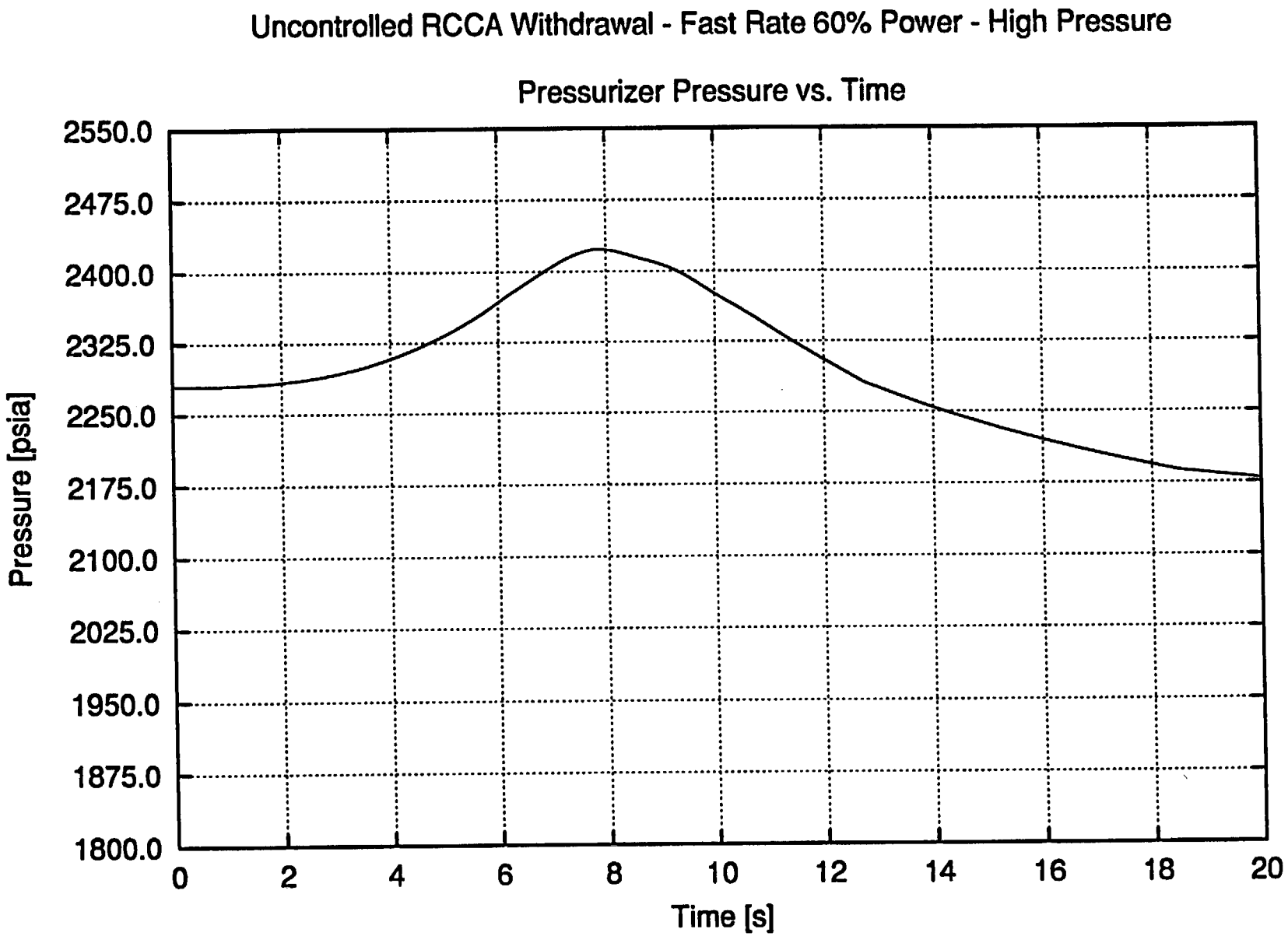
Uncontrolled RCCA Withdrawal - Fast Rate 60% Power - High Pressure
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
2
4
6
8
10
12
14
16
18
20
Time [s]

Figure 3.2-10

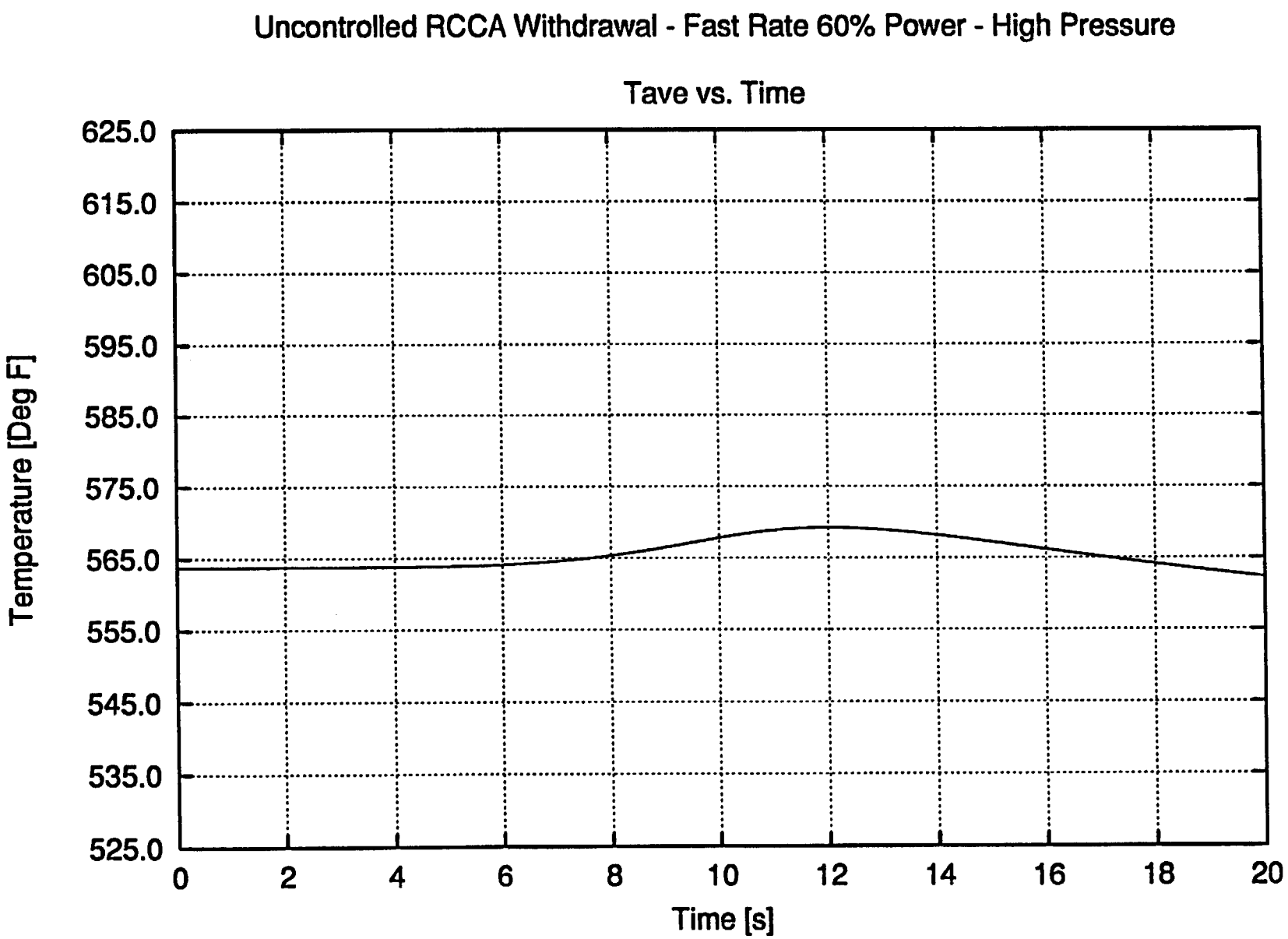
Uncontrolled RCCA Withdrawal - Fast Rate 60% Power - High Pressure
Tave vs. Time
Temperature [Deg F]
625.0
615.0
605.0
595.0
585.0
575.0
565.0
555.0
545.0
535.0
525.0
0
2
4
6
8
10
12
14
16
18
20
Time [s]

Figure 3.2-11

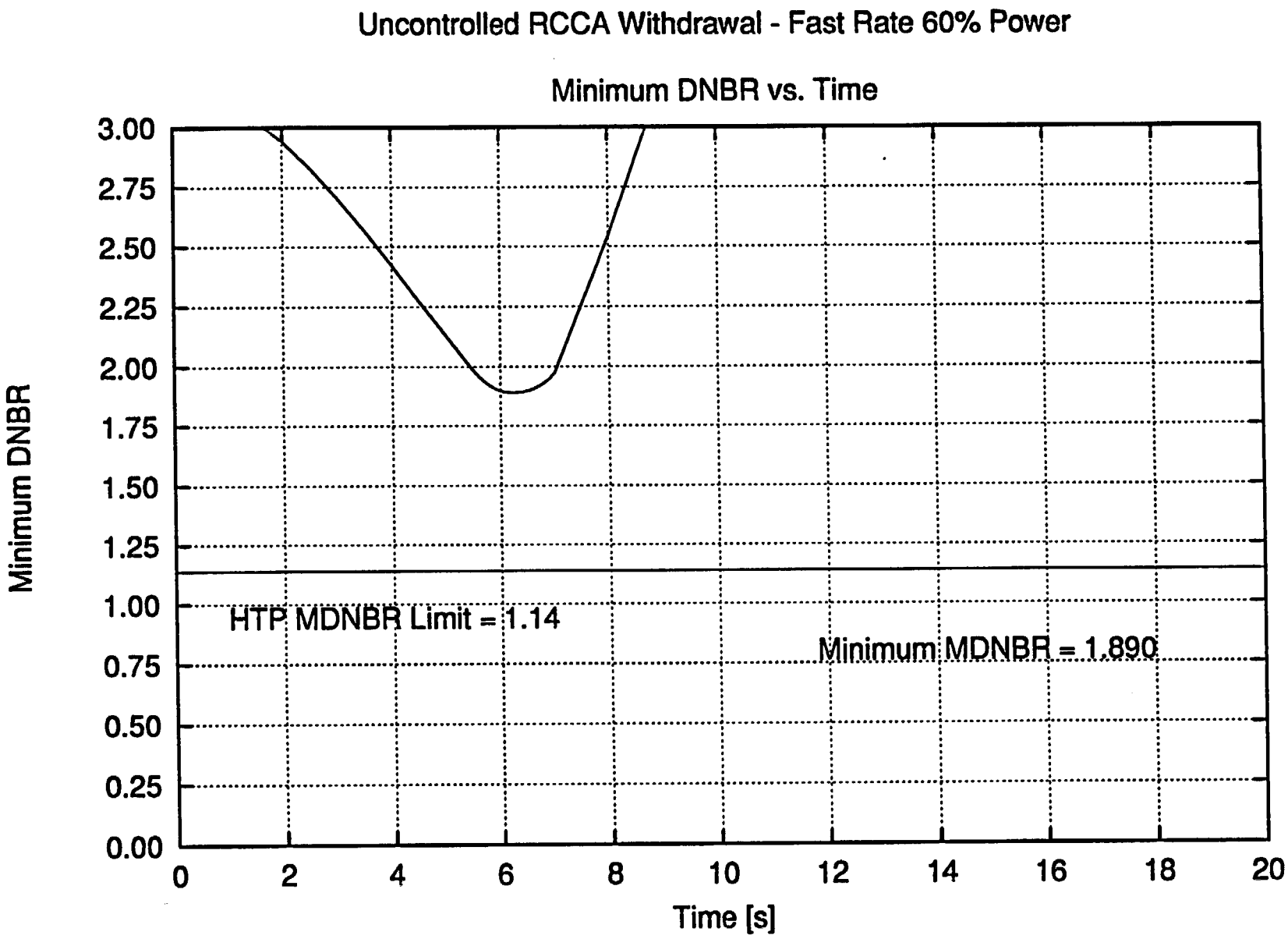
Uncontrolled RCCA Withdrawal - Fast Rate 60% Power
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.890
0
2
4
6
8
10
12
14
16
18
20
Time [s]

Figure 3.2-12

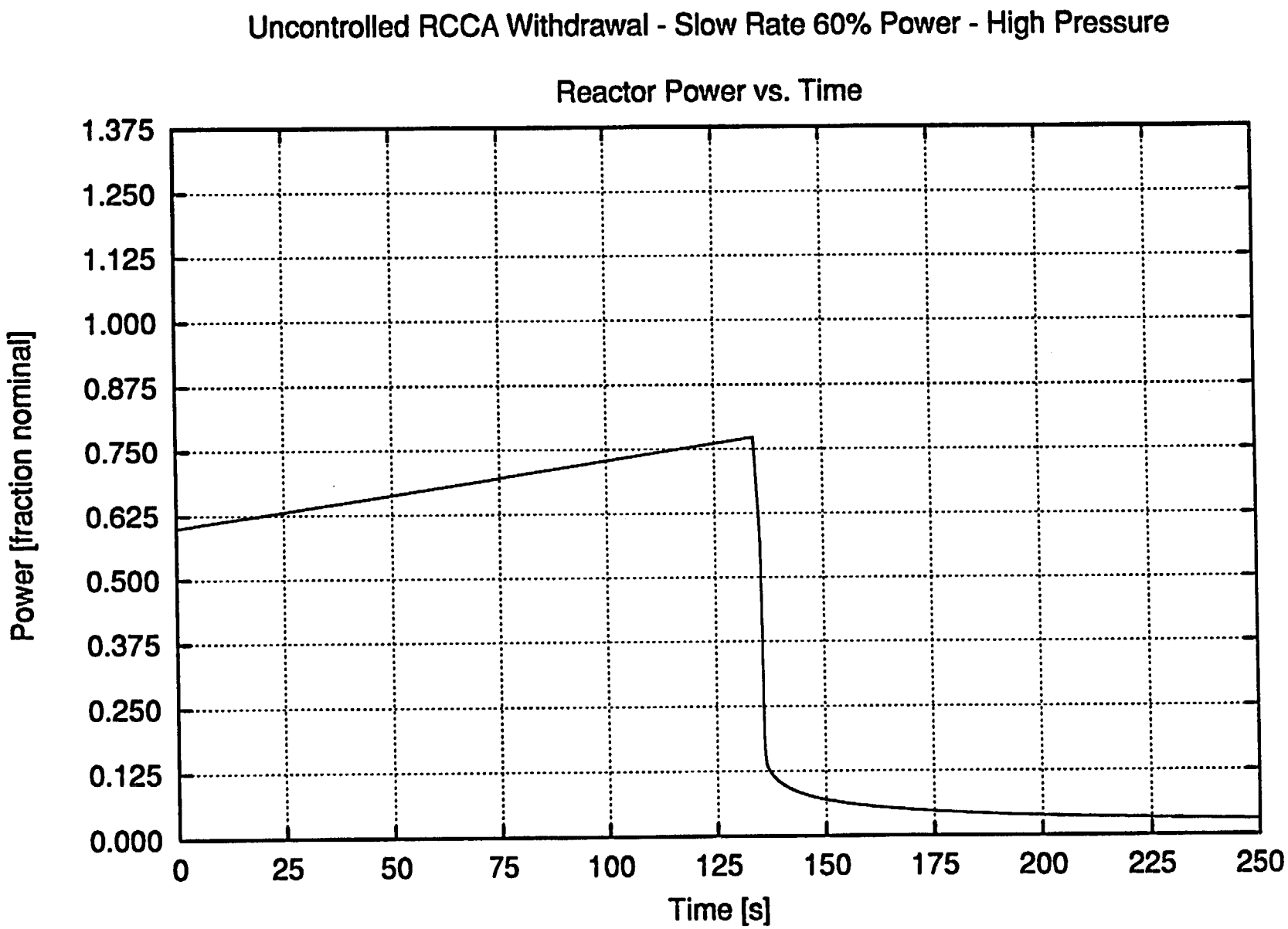
Uncontrolled RCCA Withdrawal - Slow Rate 60% Power - High Pressure
Reactor Power vs. Time
Power [fraction nominal]
1.375
1.250
1.125
1.000
0.875
0.750
0.625
0.500
0.375
0.250
0.125
0.000
0
25
50
75
100
125
150
175
200
225
250
Time [s]

Figure 3.2-13

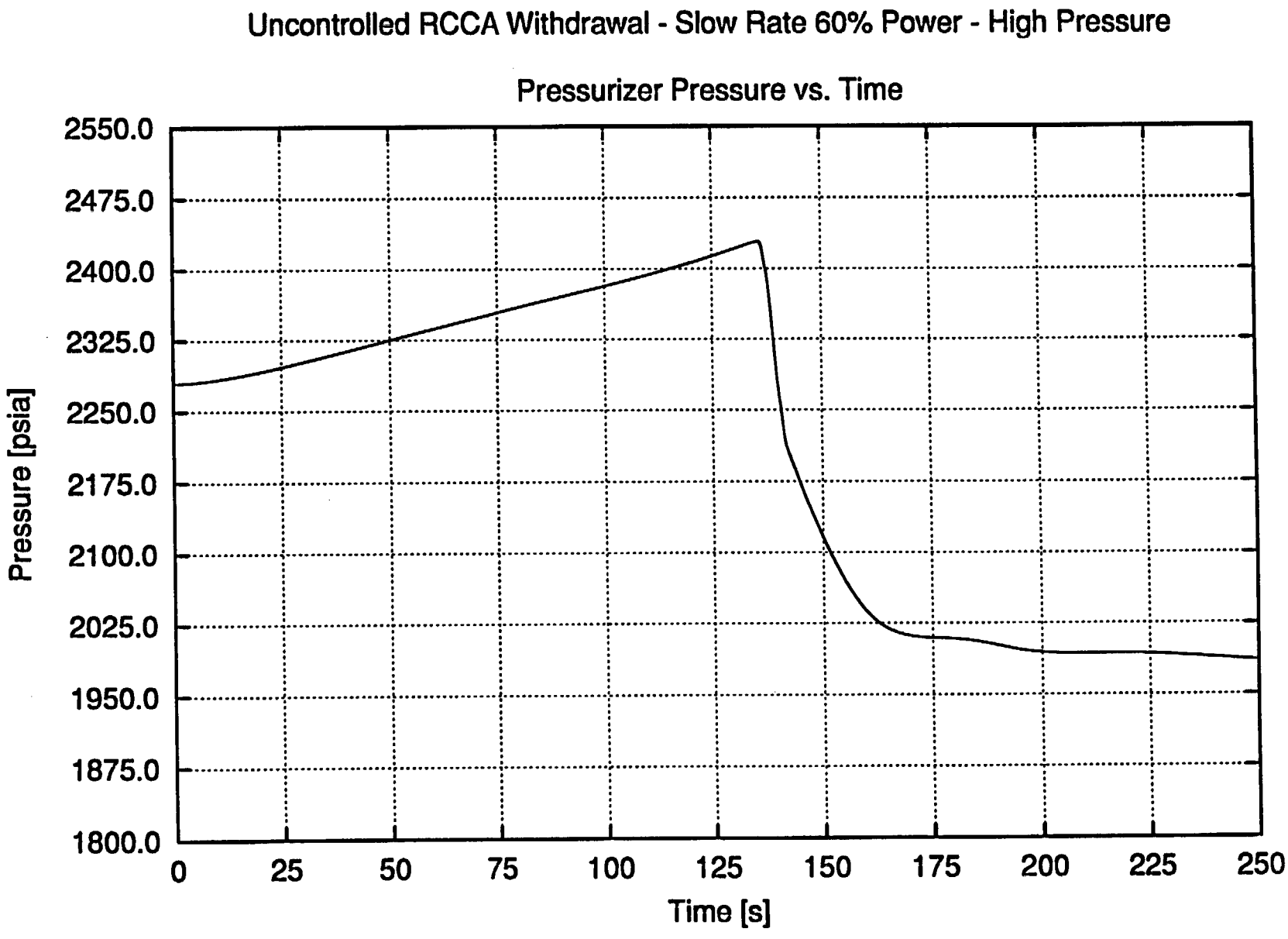
Uncontrolled RCCA Withdrawal - Slow Rate 60% Power - High Pressure
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
25
50
75
100
125
150
175
200
225
250
Time [s]

Figure 3.2-14

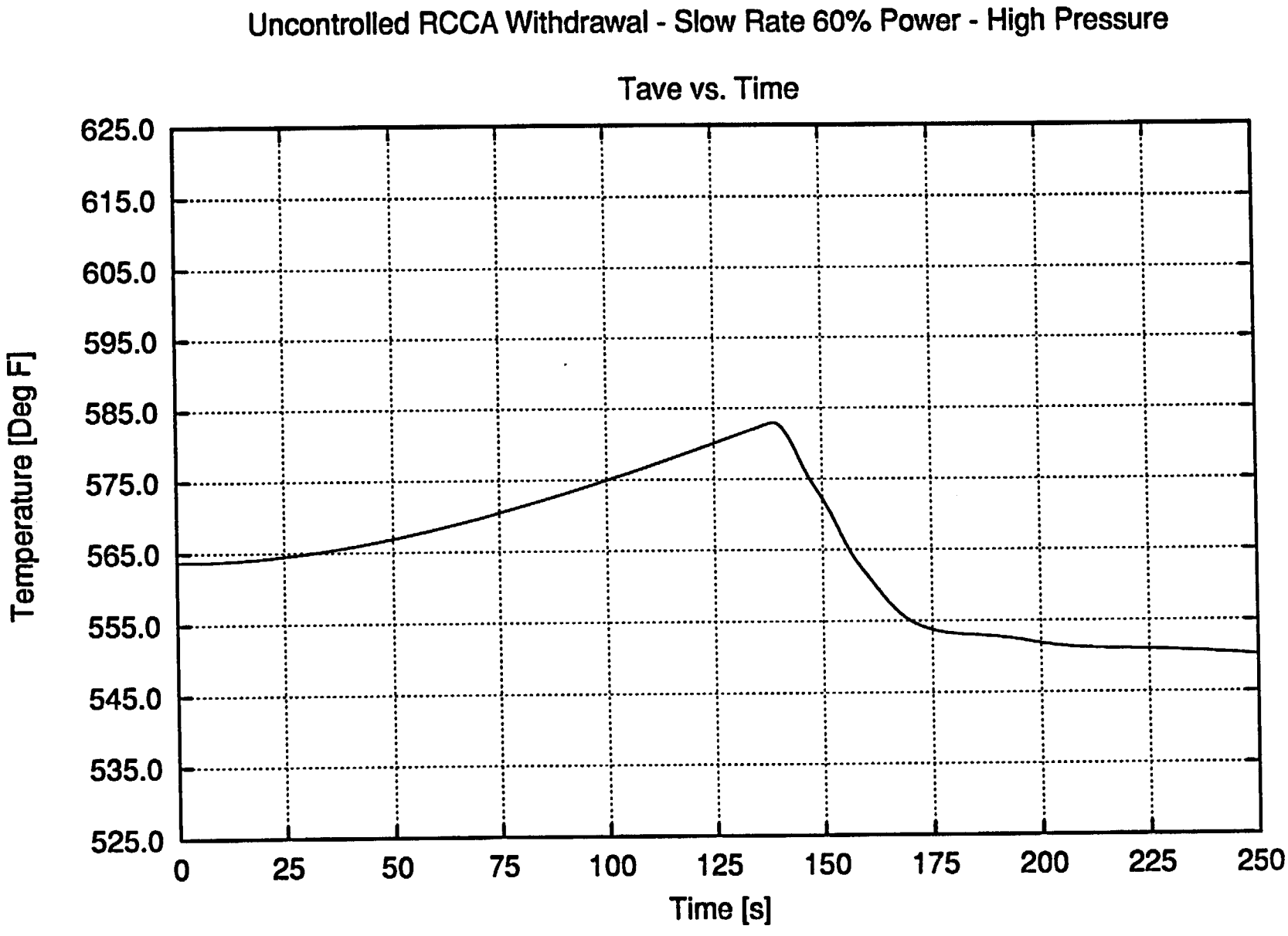
Uncontrolled RCCA Withdrawal - Slow Rate 60% Power - High Pressure
Tave vs. Time
Temperature [Deg F]
625.0
615.0
605.0
595.0
585.0
575.0
565.0
555.0
545.0
535.0
525.0
0
25
50
75
100
125
150
175
200
225
250
Time [s]

Figure 3.2-15

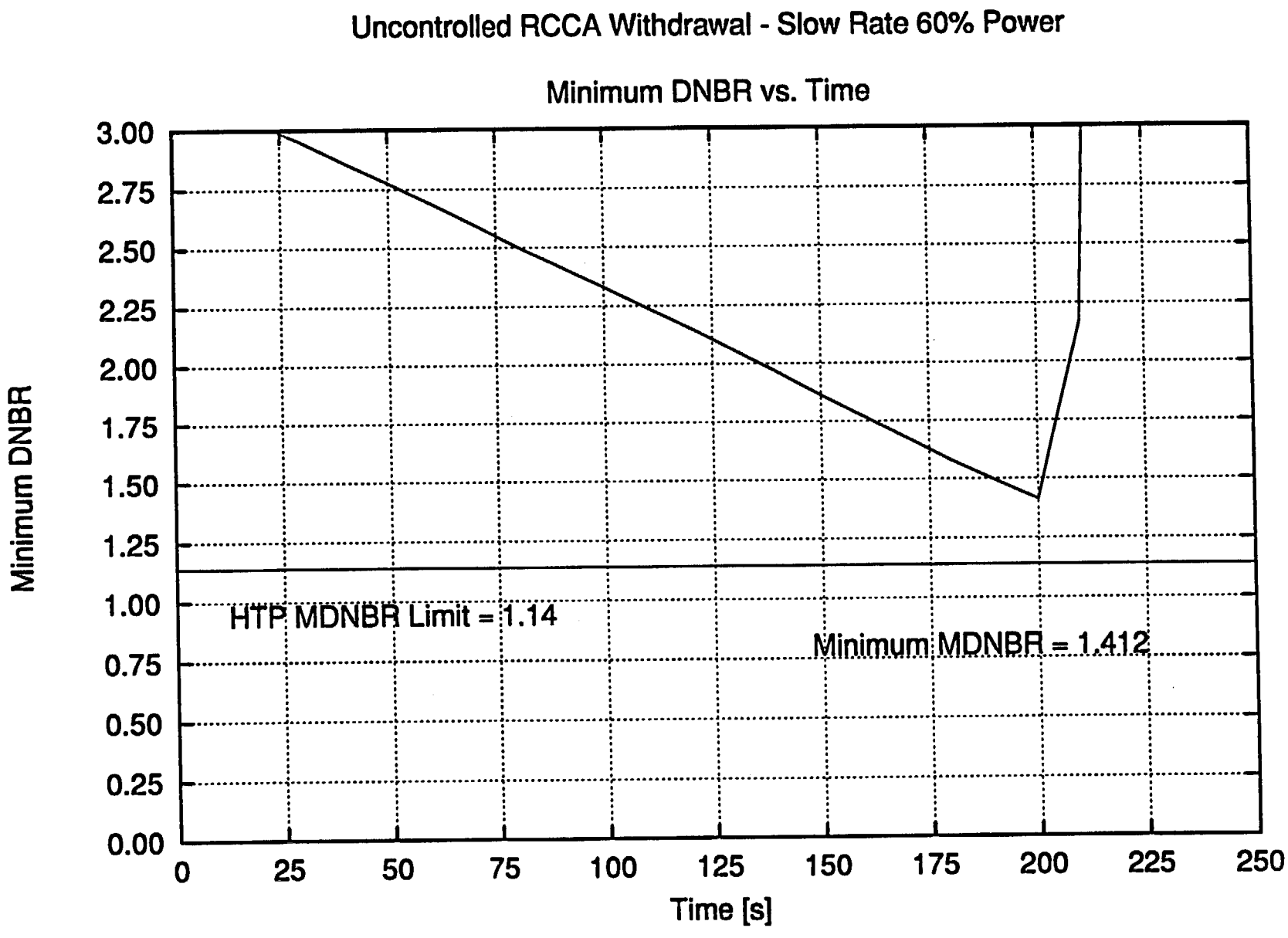
Uncontrolled RCCA Withdrawal - Slow Rate 60% Power
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.412
0
25
50
75
100
125
150
175
200
225
250
Time [s]

Figure 3.2-16

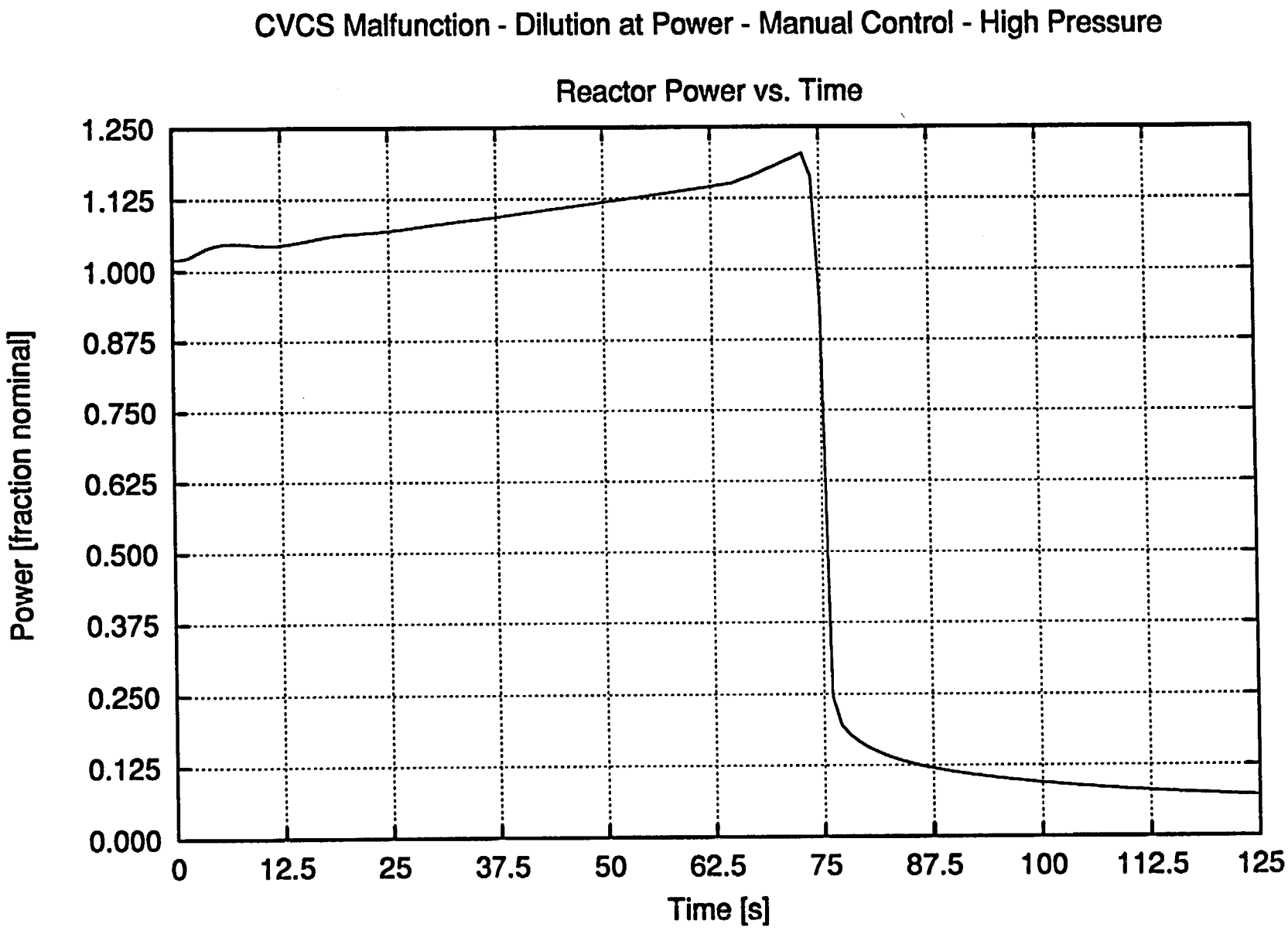
CVCS Malfunction - Dilution at Power - Manual Control - High Pressure
Reactor Power vs. Time
Power [fraction nominal]
1.250
1.125
1.000
0.875
0.750
0.625
0.500
0.375
0.250
0.125
0.000
0
12.5
25
37.5
50
62.5
75
87.5
100
112.5
125
Time [s]

Figure 3.5-1

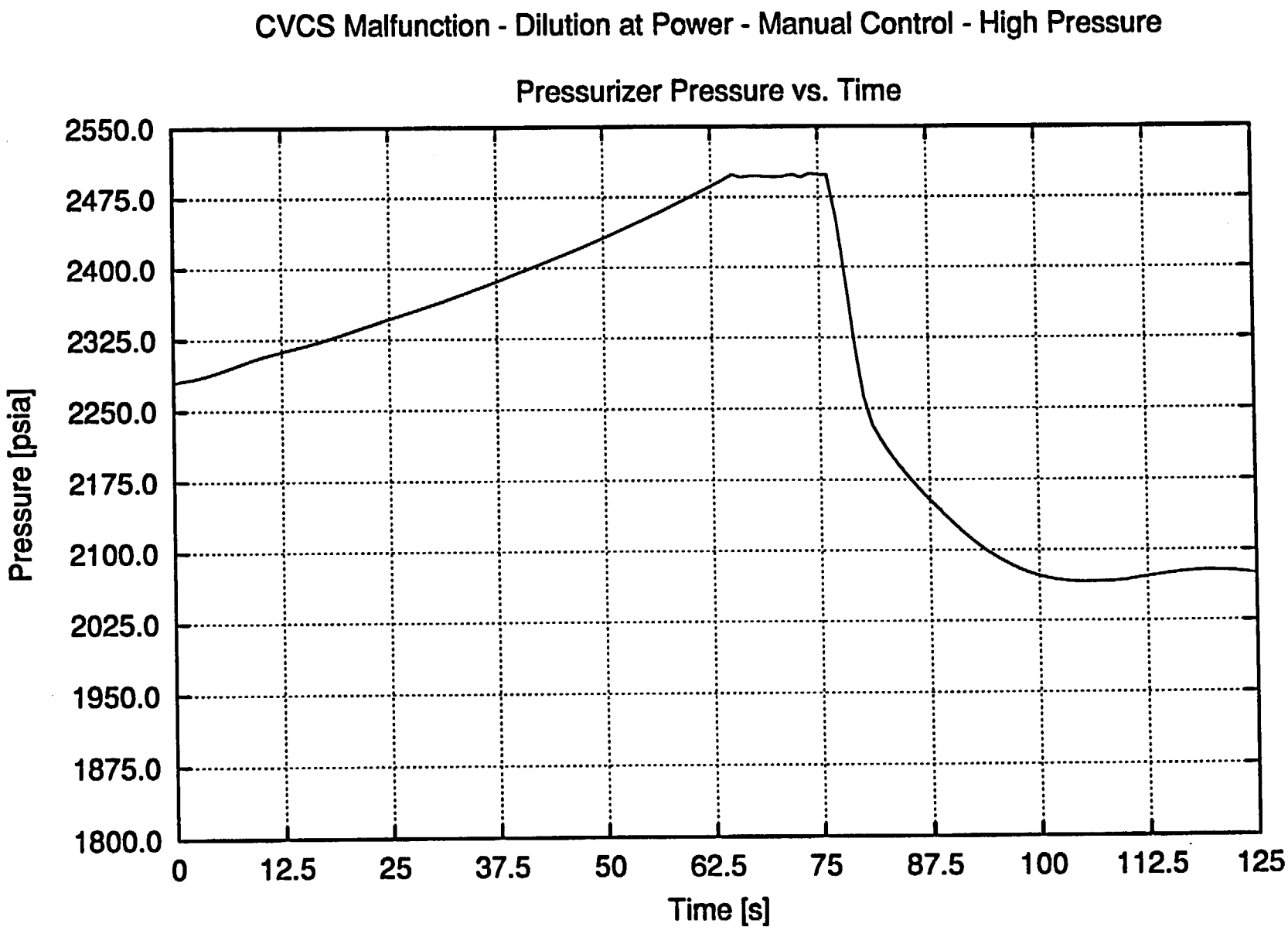
CVCS Malfunction - Dilution at Power - Manual Control - High Pressure
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
12.5
25
37.5
50
62.5
75
87.5
100
112.5
125
Time [s]

Figure 3.5-2

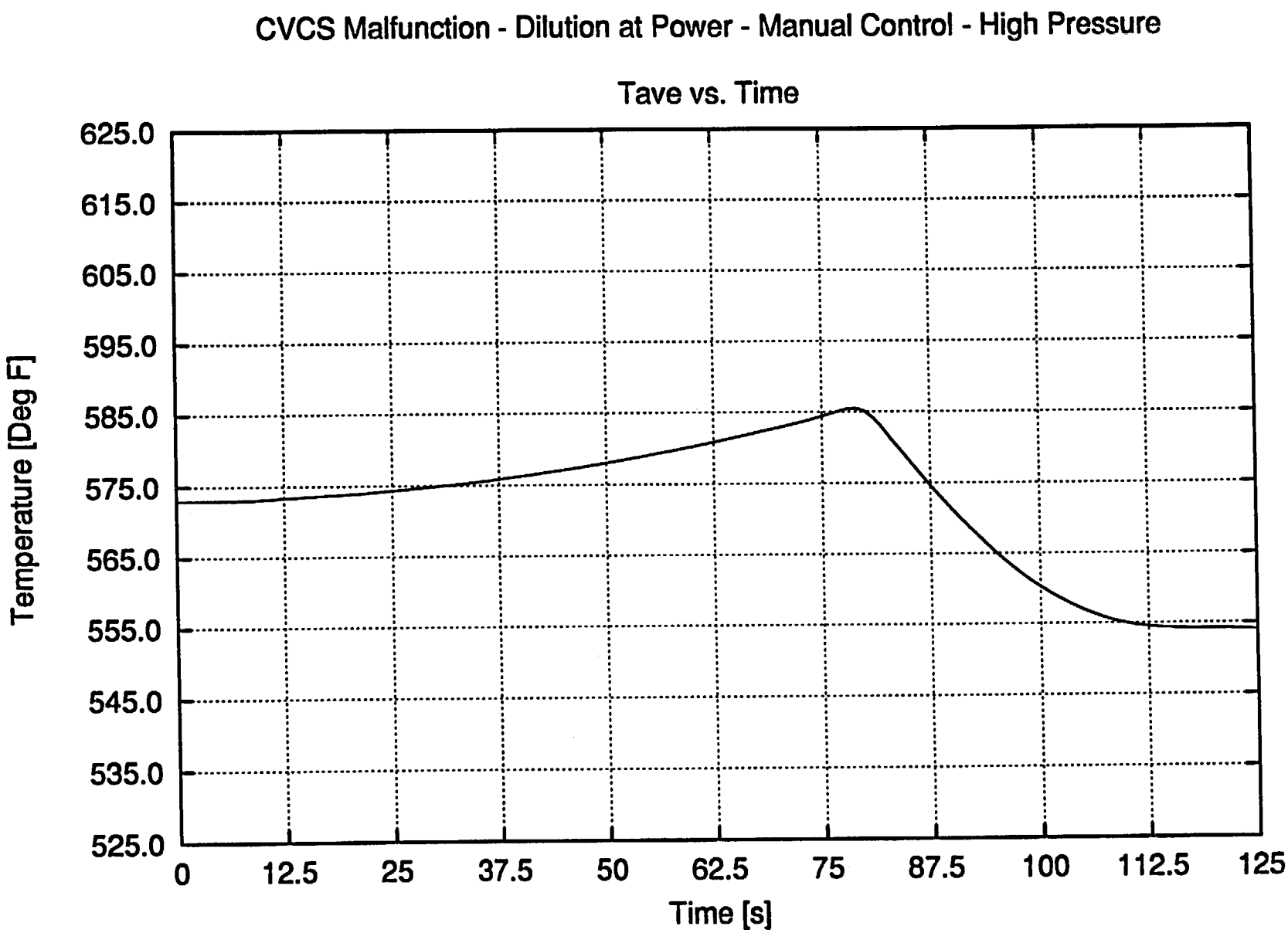
CVCS Malfunction - Dilution at Power - Manual Control - High Pressure
Tave vs. Time
Temperature [Deg F]
625.0
615.0
605.0
595.0
585.0
575.0
565.0
555.0
545.0
535.0
525.0
0
12.5
25
37.5
50
62.5
75
87.5
100
112.5
125
Time [s]

Figure 3.5-3

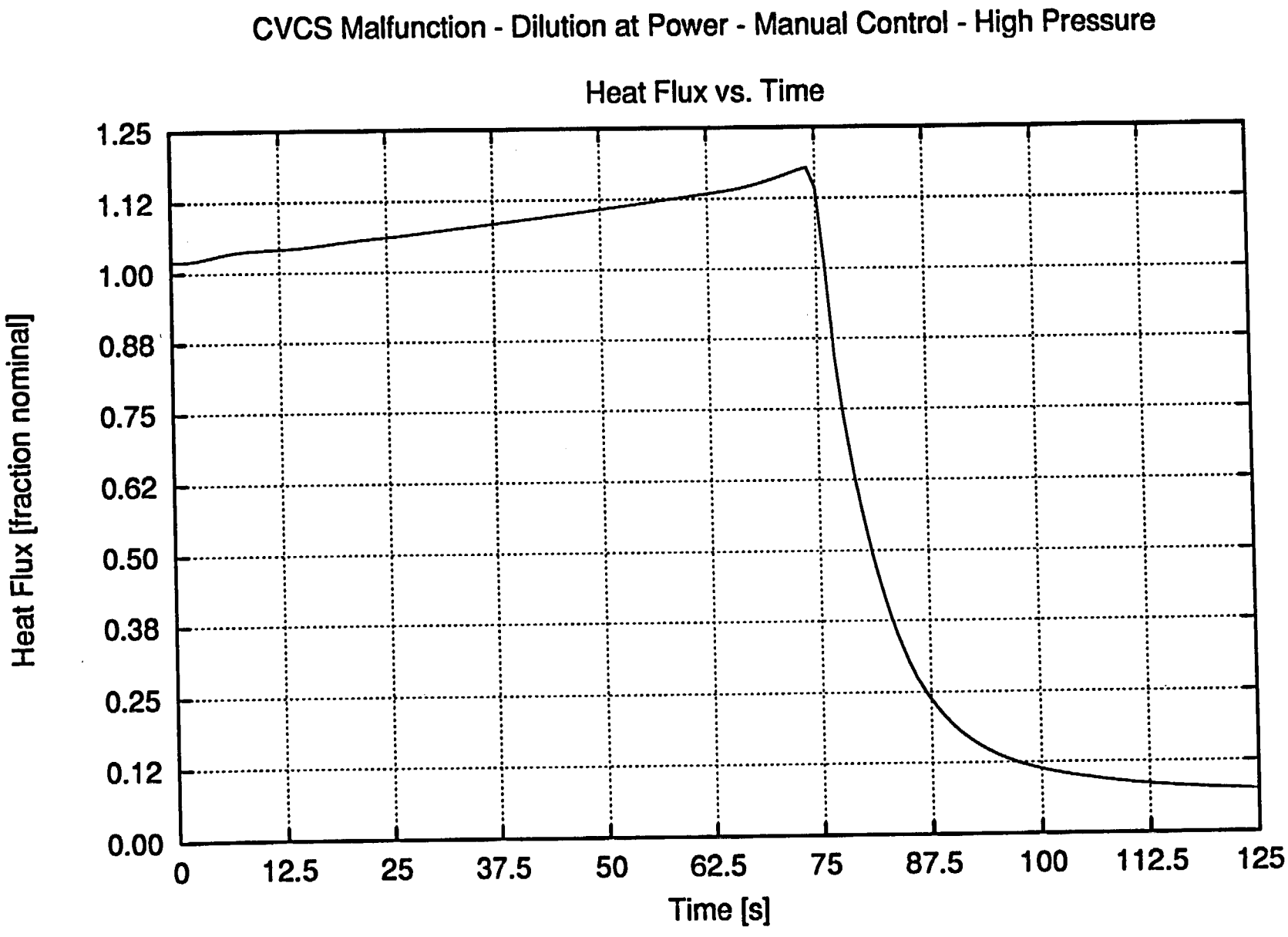
CVCS Malfunction - Dilution at Power - Manual Control - High Pressure
Heat Flux vs. Time
Heat Flux [fraction nominal]
1.25
1.12
1.00
0.88
0.75
0.62
0.50
0.38
0.25
0.12
0.00
0
12.5
25
37.5
50
62.5
75
87.5
100
112.5
125
Time [s]

Figure 3.5-4

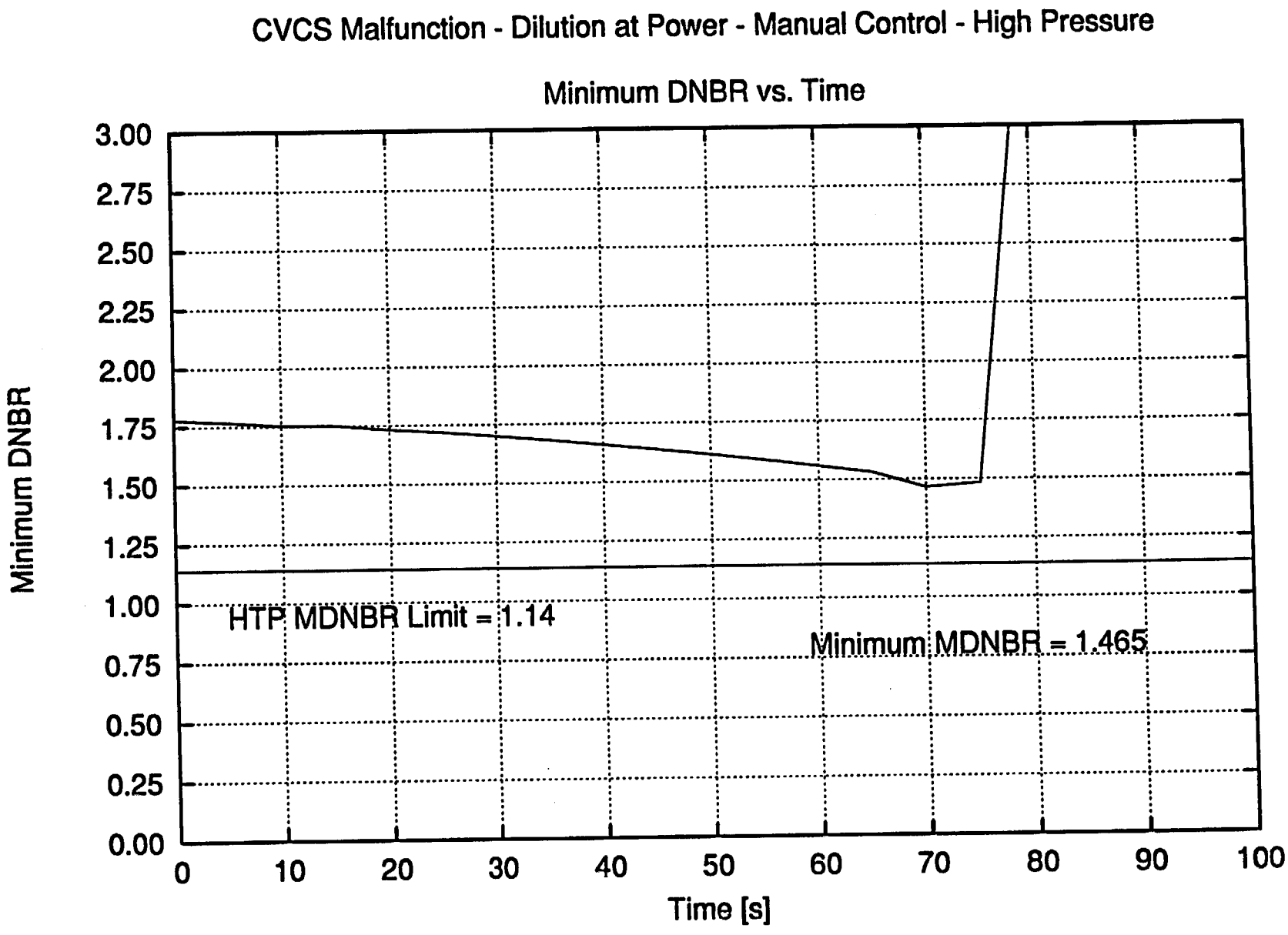
CVCS Malfunction - Dilution at Power - Manual Control - High Pressure
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0
10
20
30
40
50
60
70
80
90
100
Time [s]
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.465

Figure 3.5-5

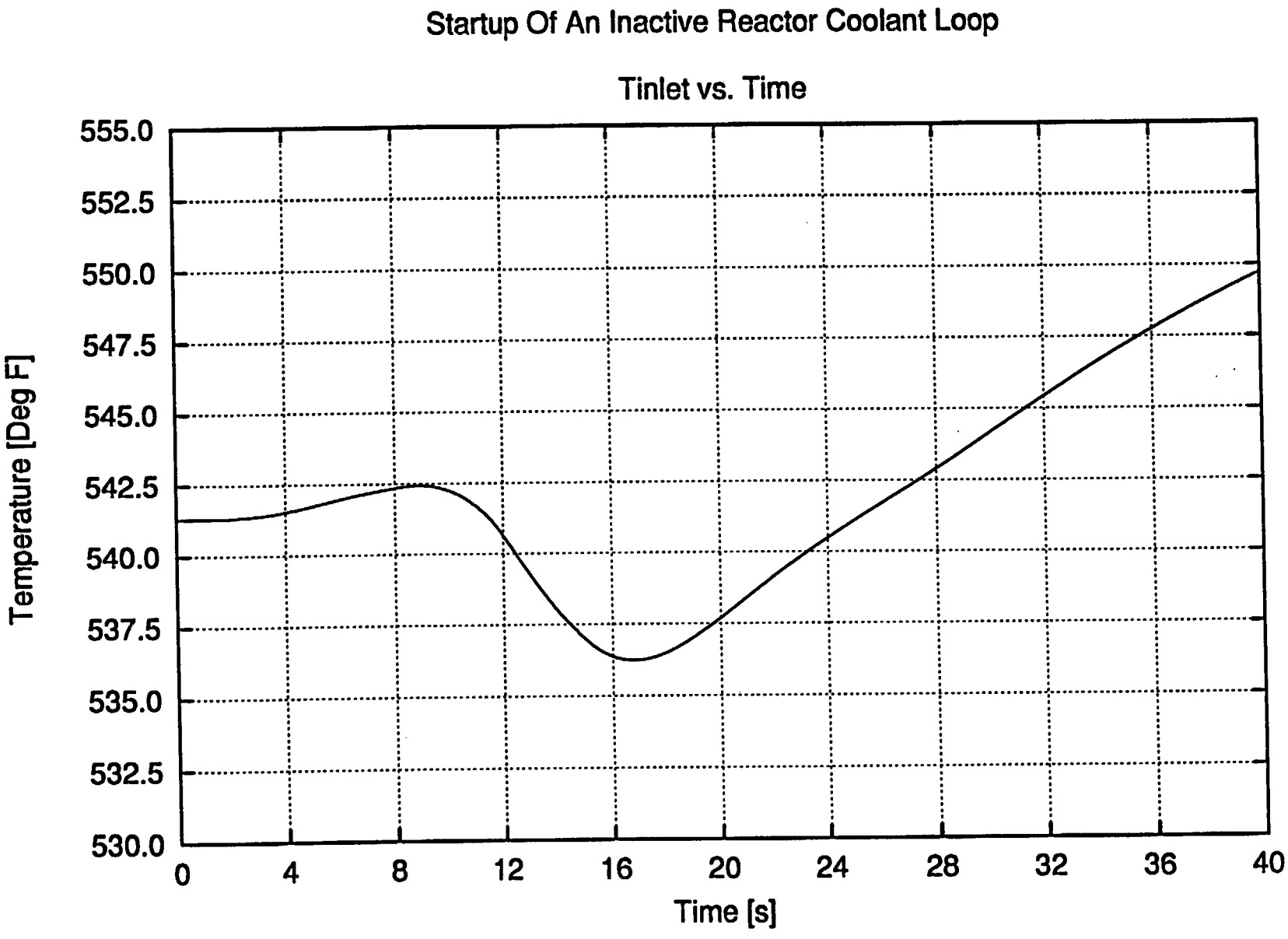
Startup Of An Inactive Reactor Coolant Loop
Tinlet vs. Time
Temperature [Deg F]
555.0
552.5
550.0
547.5
545.0
542.5
540.0
537.5
535.0
532.5
530.0
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.6-1

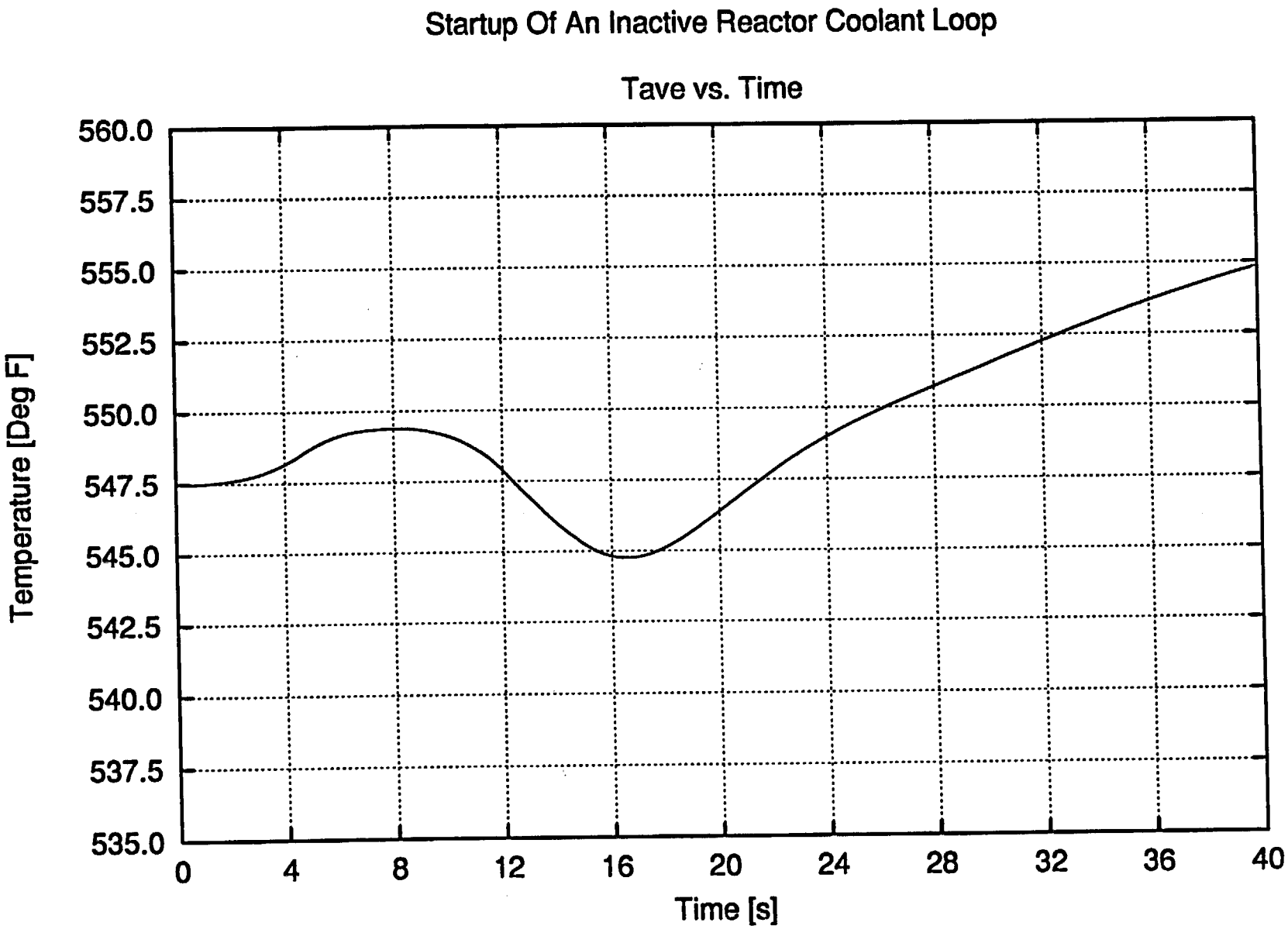
Startup Of An Inactive Reactor Coolant Loop
Tave vs. Time
Temperature [Deg F]
560.0
557.5
555.0
552.5
550.0
547.5
545.0
542.5
540.0
537.5
535.0
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.6-2

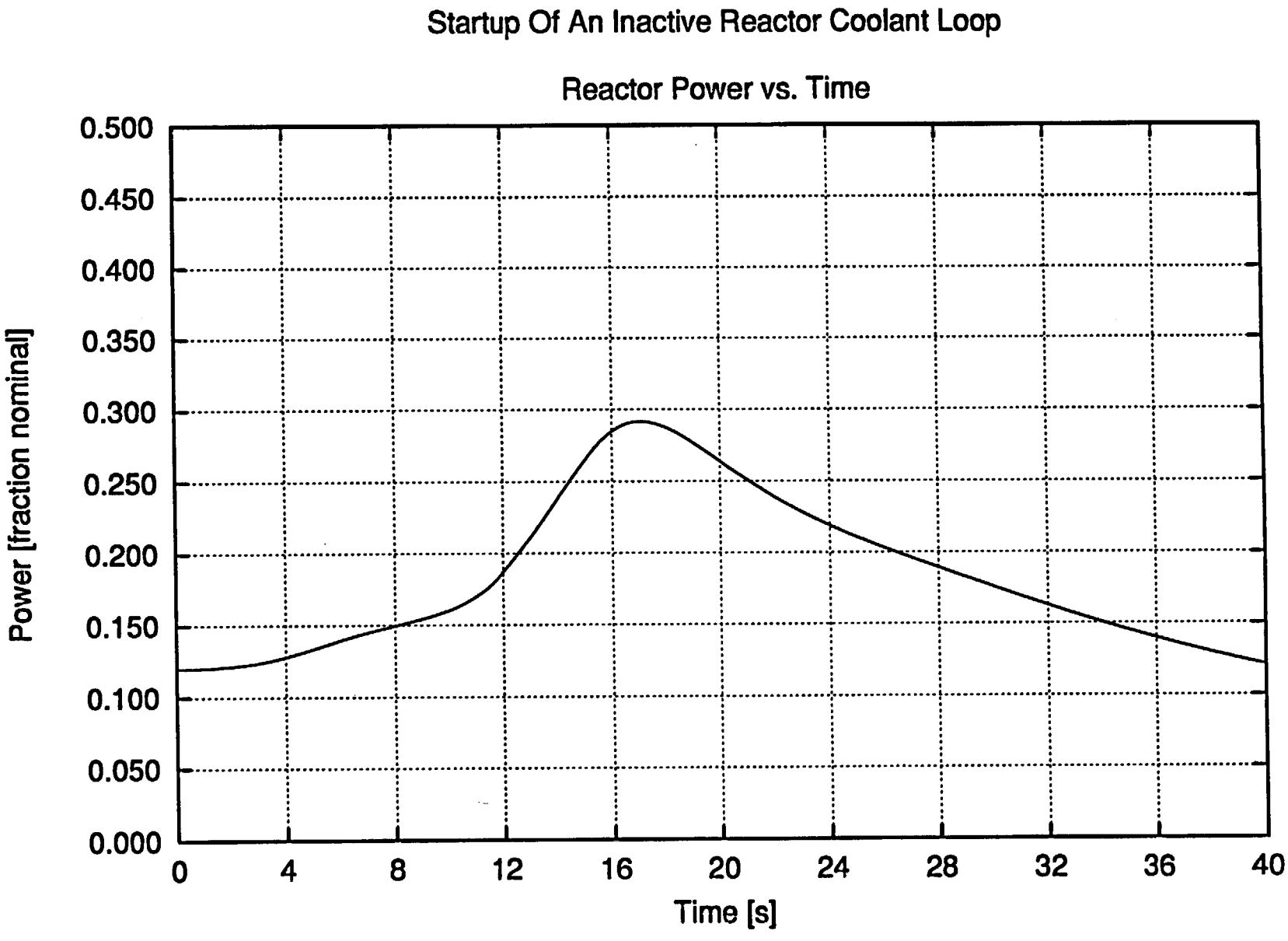
Startup Of An Inactive Reactor Coolant Loop
Reactor Power vs. Time
Power [fraction nominal]
0.500
0.450
0.400
0.350
0.300
0.250
0.200
0.150
0.100
0.050
0.000
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.6-3

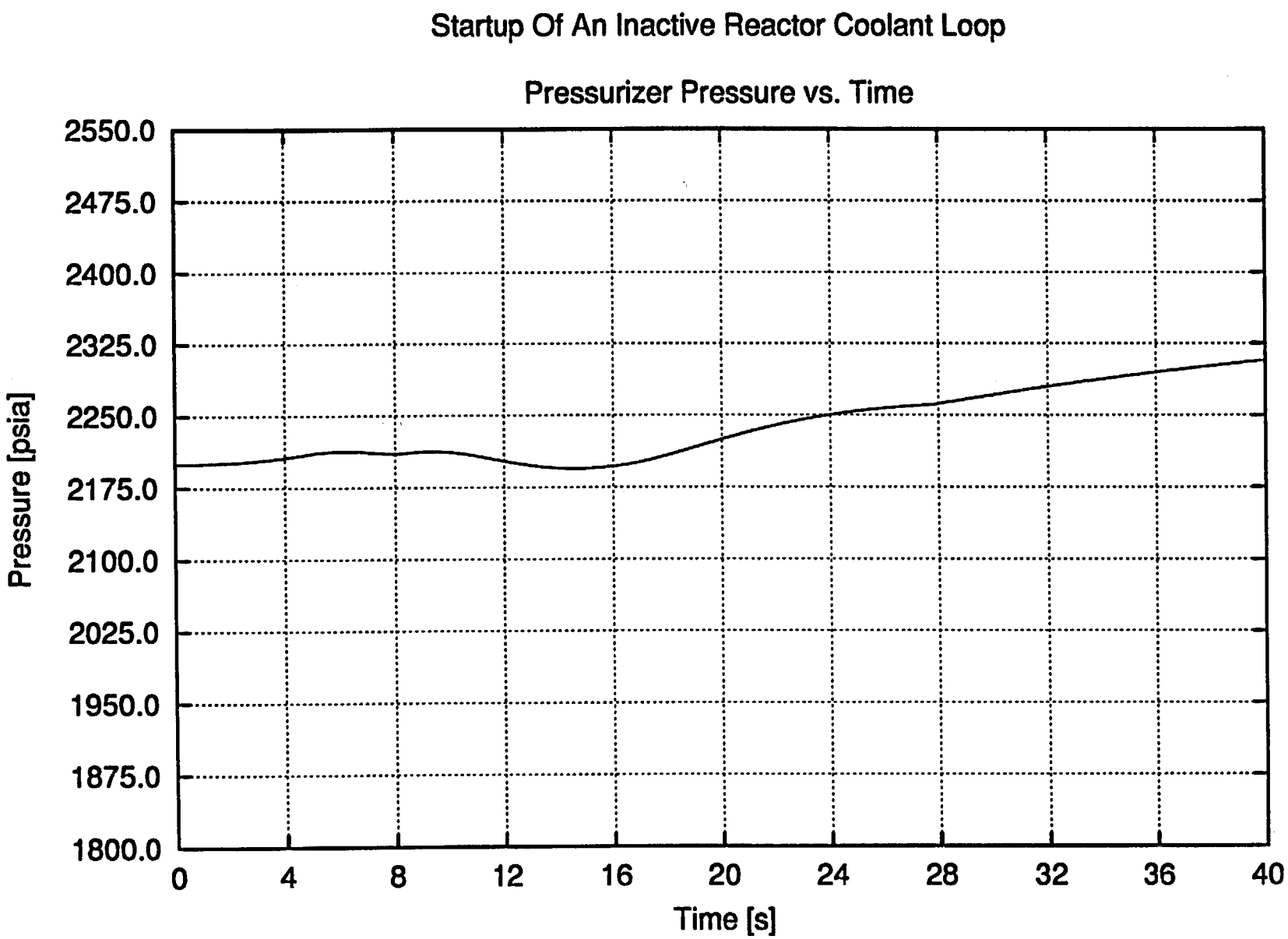
Startup Of An Inactive Reactor Coolant Loop
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.6-4

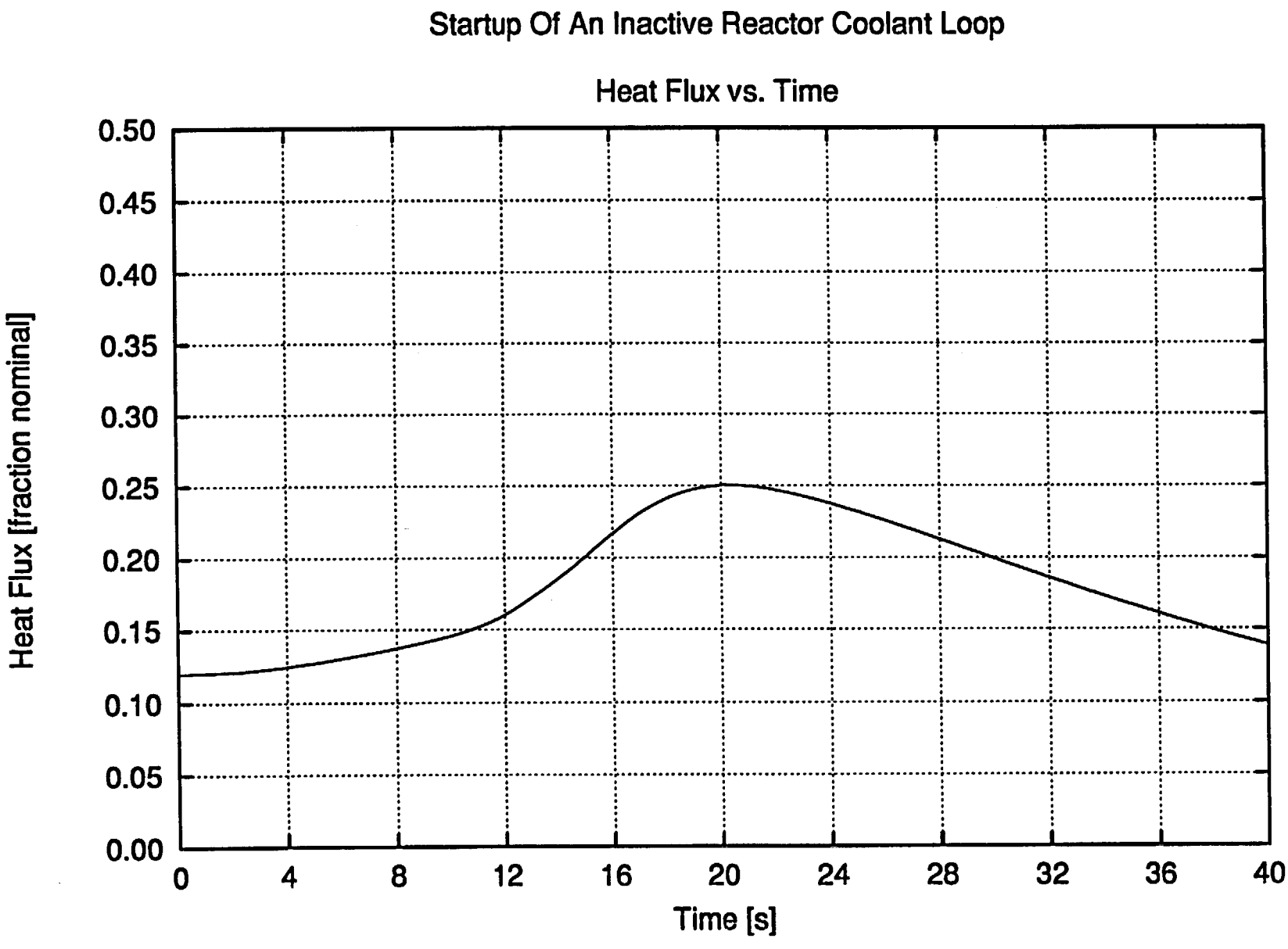
Startup Of An Inactive Reactor Coolant Loop
Heat Flux vs. Time
Heat Flux [fraction nominal]
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
0
4
8
12
16
20
24
28
32
36
40
Time [s]

Figure 3.6-5

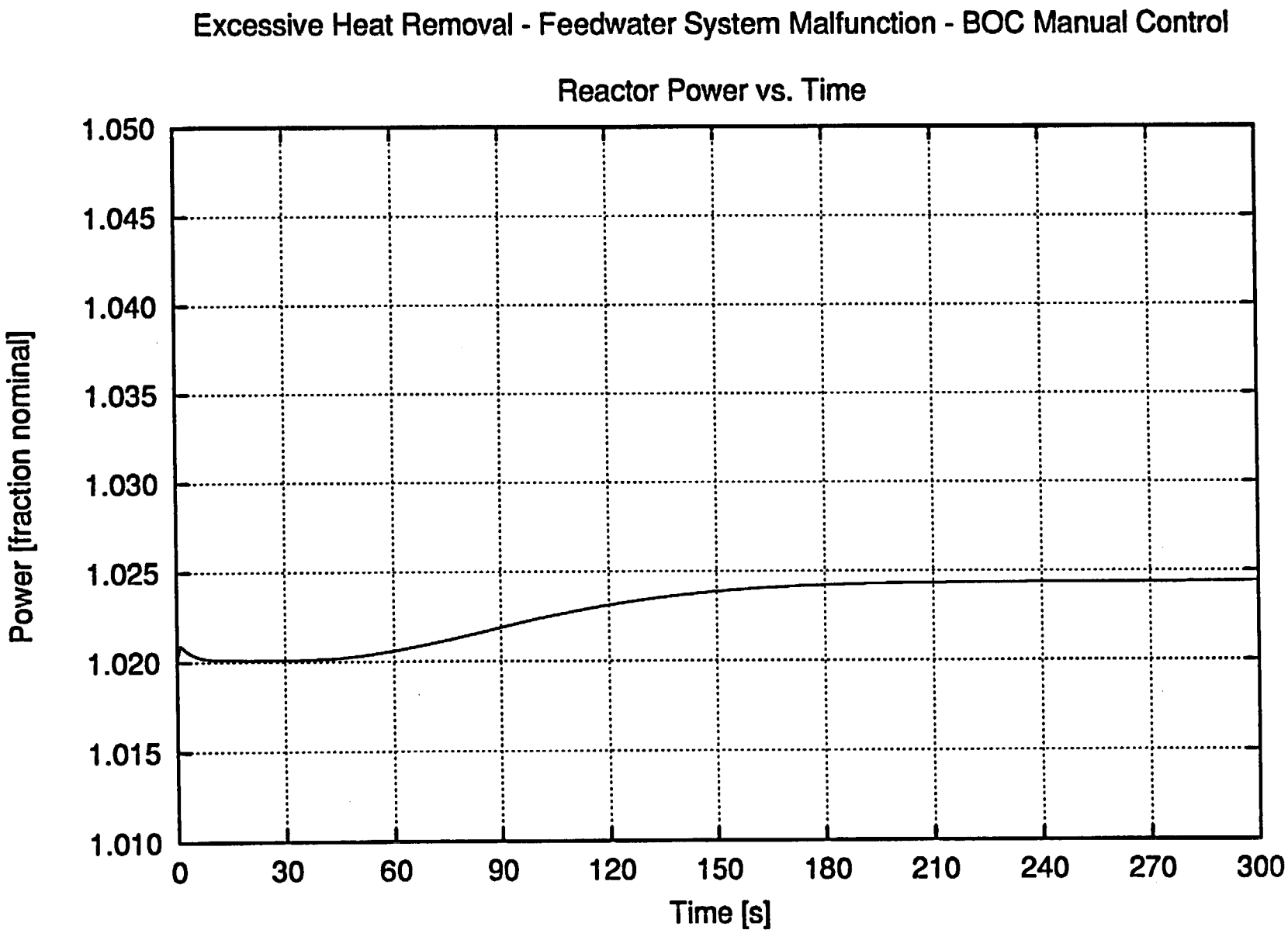
Excessive Heat Removal - Feedwater System Malfunction - BOC Manual Control
Reactor Power vs. Time
Power [fraction nominal]
1.050
1.045
1.040
1.035
1.030
1.025
1.020
1.015
1.010
0
30
60
90
120
150
180
210
240
270
300
Time [s]

Figure 3.7-1

Excessive Heat Removal - Feedwater System Malfunction - BOC Manual Control

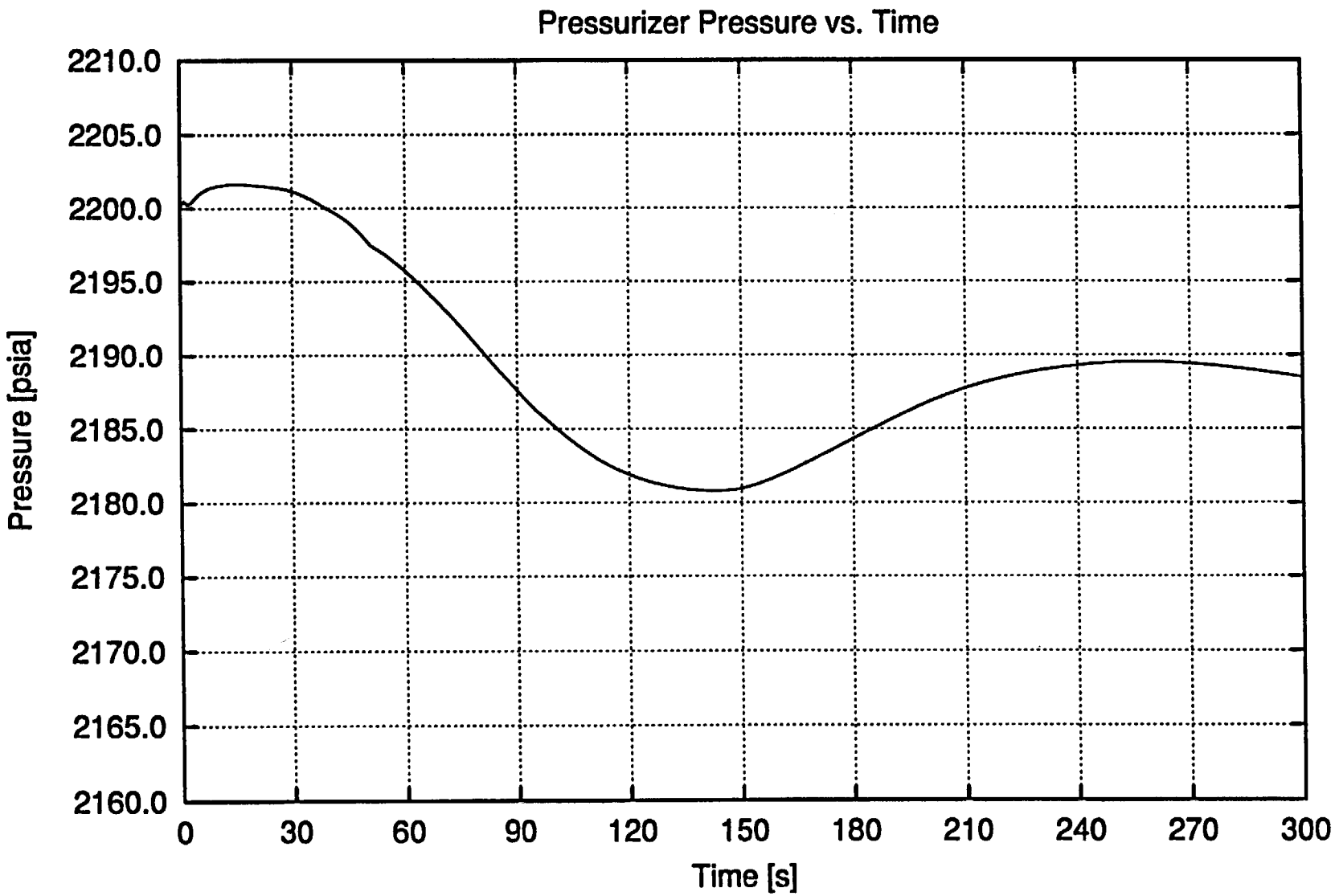

Figure 3.7-2

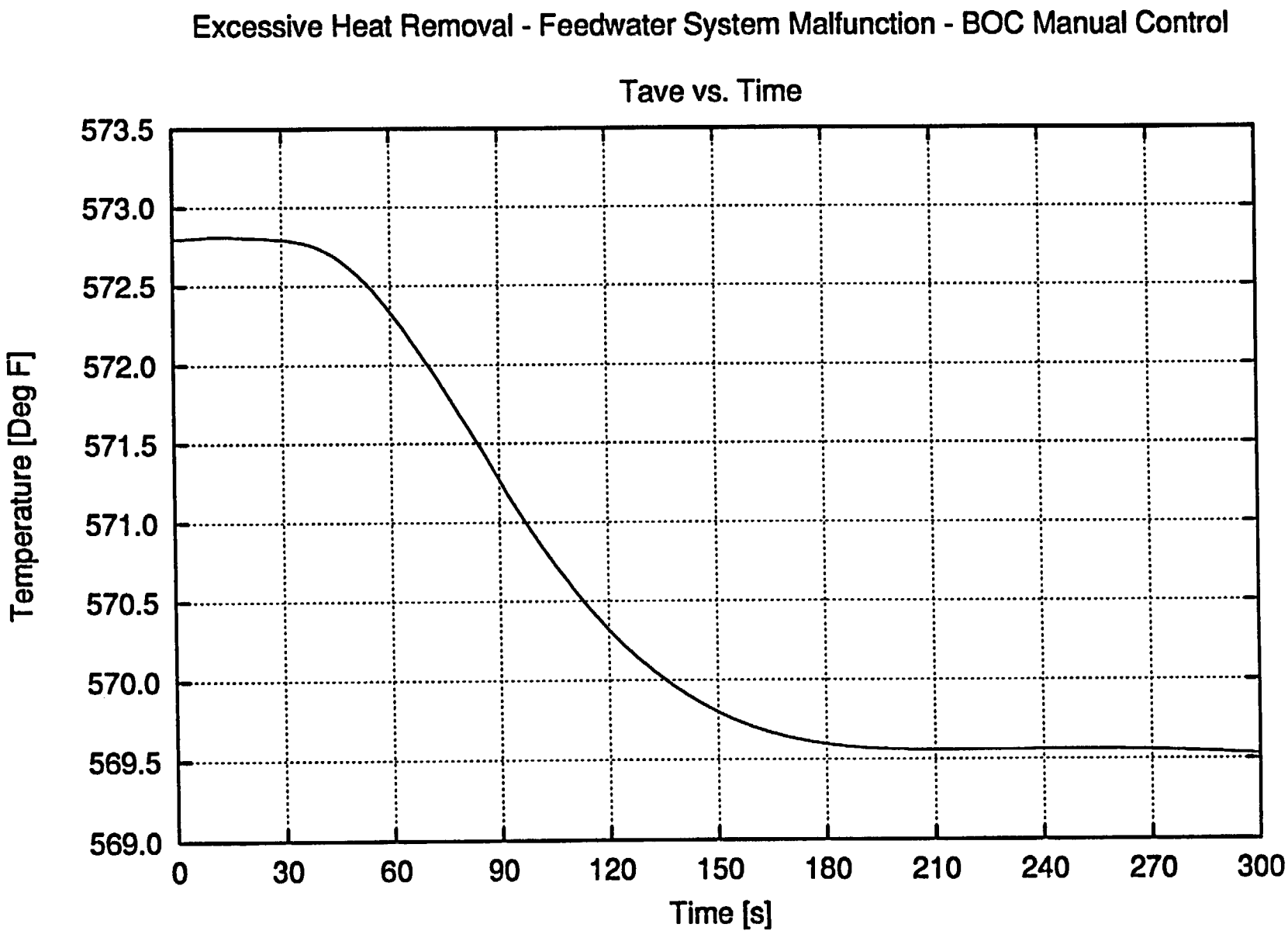
Excessive Heat Removal - Feedwater System Malfunction - BOC Manual Control
Tave vs. Time
Temperature [Deg F]
573.5
573.0
572.5
572.0
571.5
571.0
570.5
570.0
569.5
569.0
0
30
60
90
120
150
180
210
240
270
300
Time [s]

Figure 3.7-3

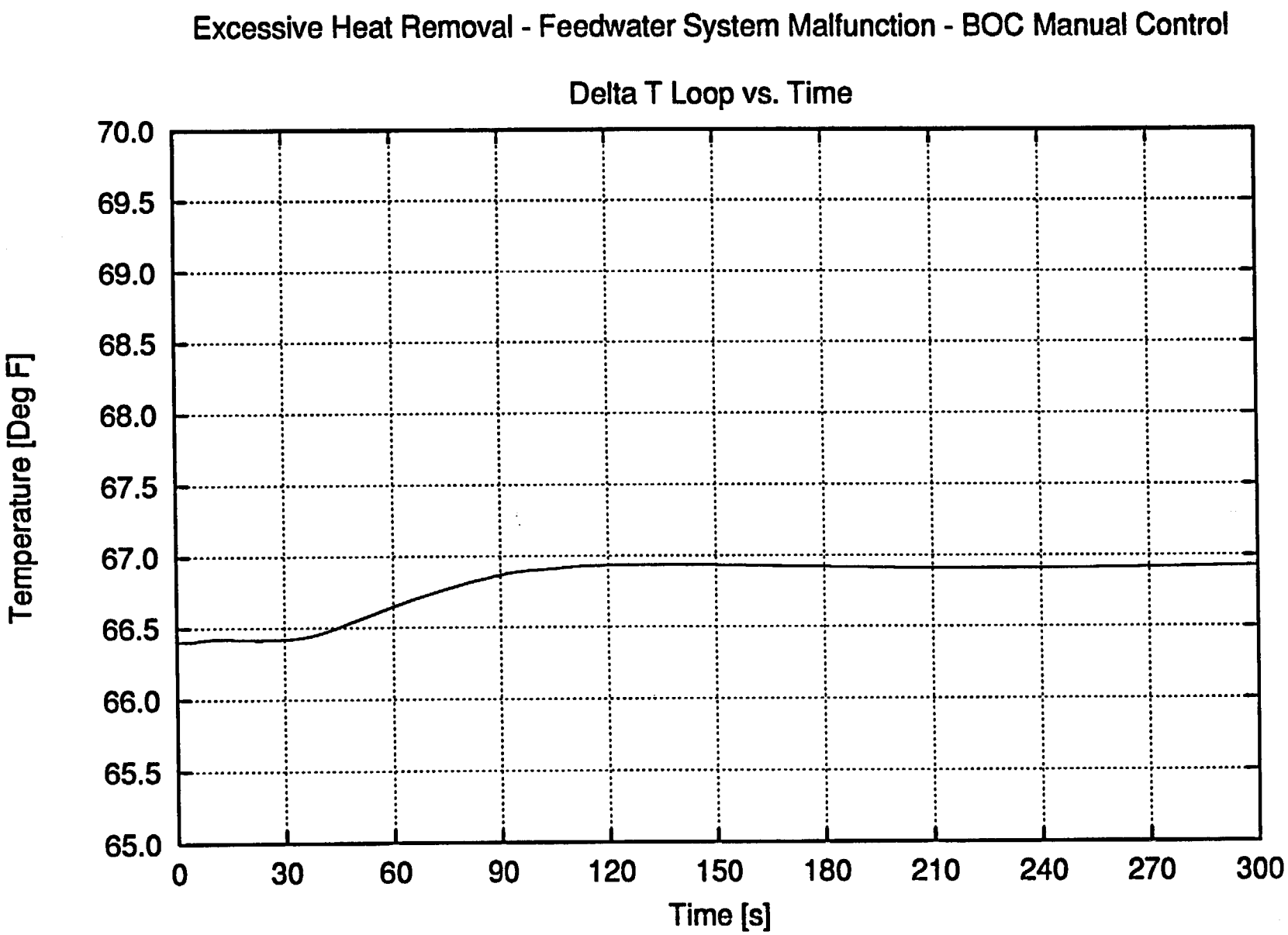
Excessive Heat Removal - Feedwater System Malfunction - BOC Manual Control
Delta T Loop vs. Time
Temperature [Deg F]
70.0
69.5
69.0
68.5
68.0
67.5
67.0
66.5
66.0
65.5
65.0
0
30
60
90
120
150
180
210
240
270
300
Time [s]

Figure 3.7-4

Excessive Heat Removal - Feedwater System Malfunction - BOC Manual Control

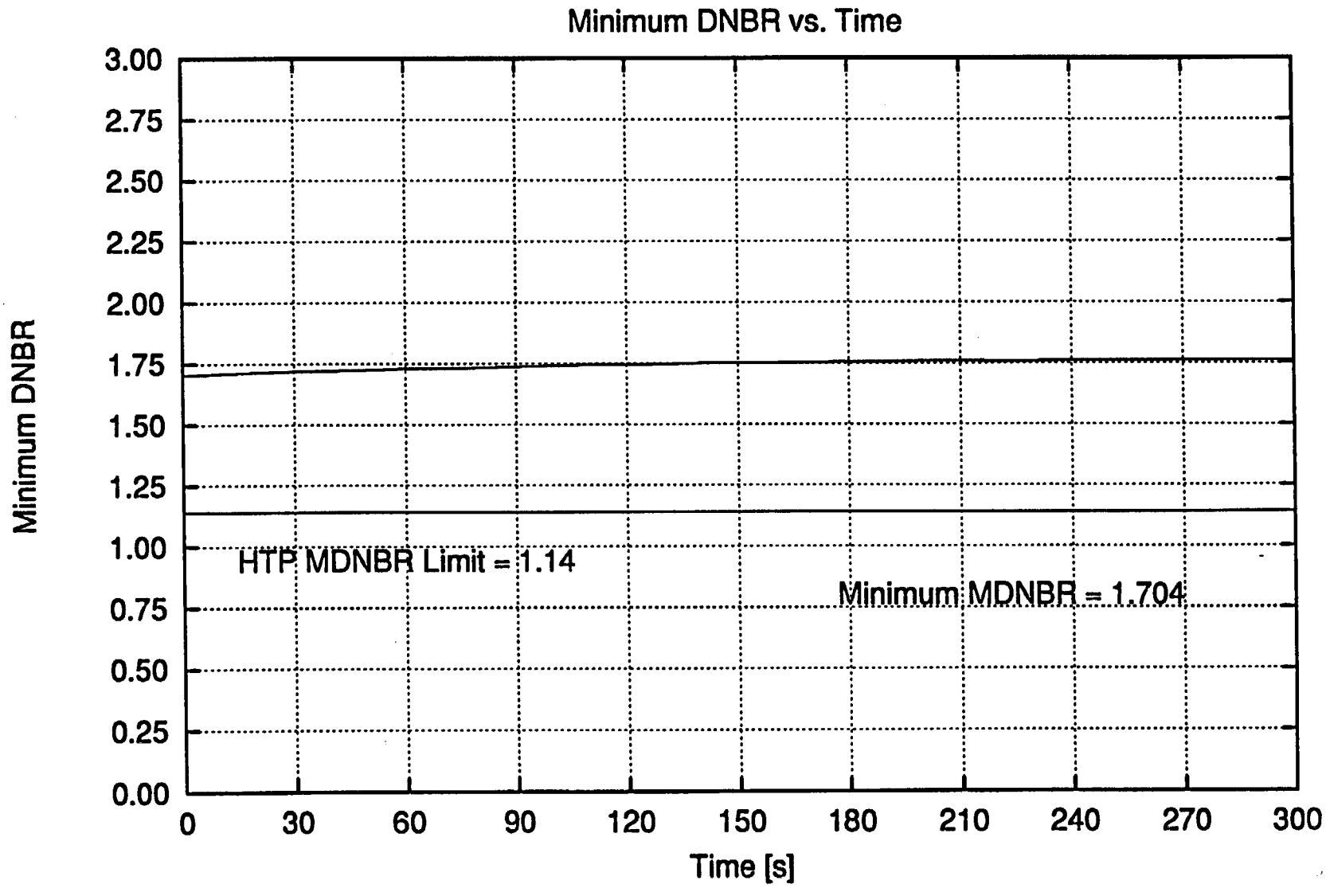

Figure 3.7-5

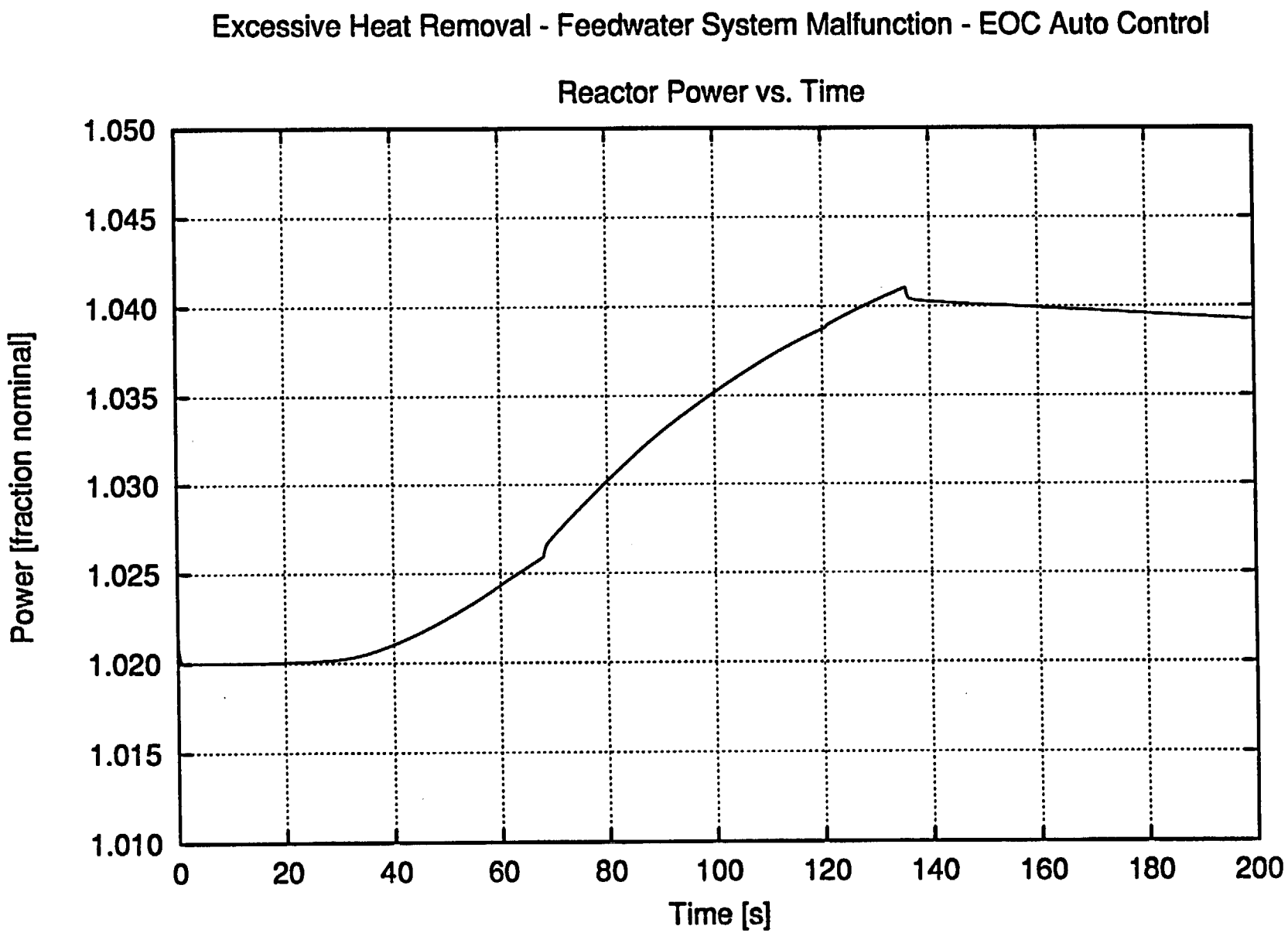
Excessive Heat Removal - Feedwater System Malfunction - EOC Auto Control
Reactor Power vs. Time
Power [fraction nominal]
1.050
1.045
1.040
1.035
1.030
1.025
1.020
1.015
1.010
0
20
40
60
80
100
120
140
160
180
200
Time [s]

Figure 3.7-6

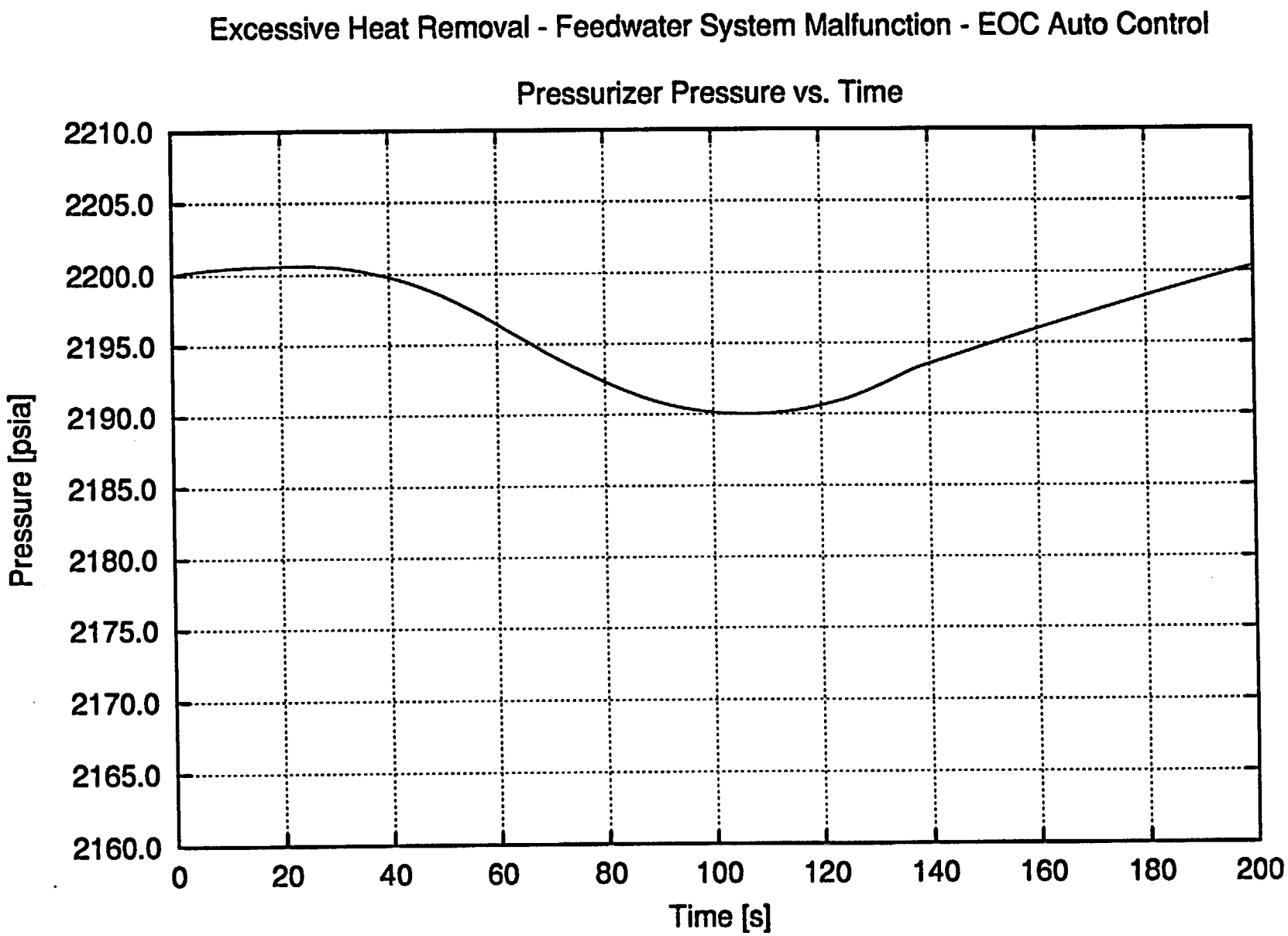
Excessive Heat Removal - Feedwater System Malfunction - EOC Auto Control
Pressurizer Pressure vs. Time
Pressure [psia]
2210.0
2205.0
2200.0
2195.0
2190.0
2185.0
2180.0
2175.0
2170.0
2165.0
2160.0
0
20
40
60
80
100
120
140
160
180
200
Time [s]

Figure 3.7-7

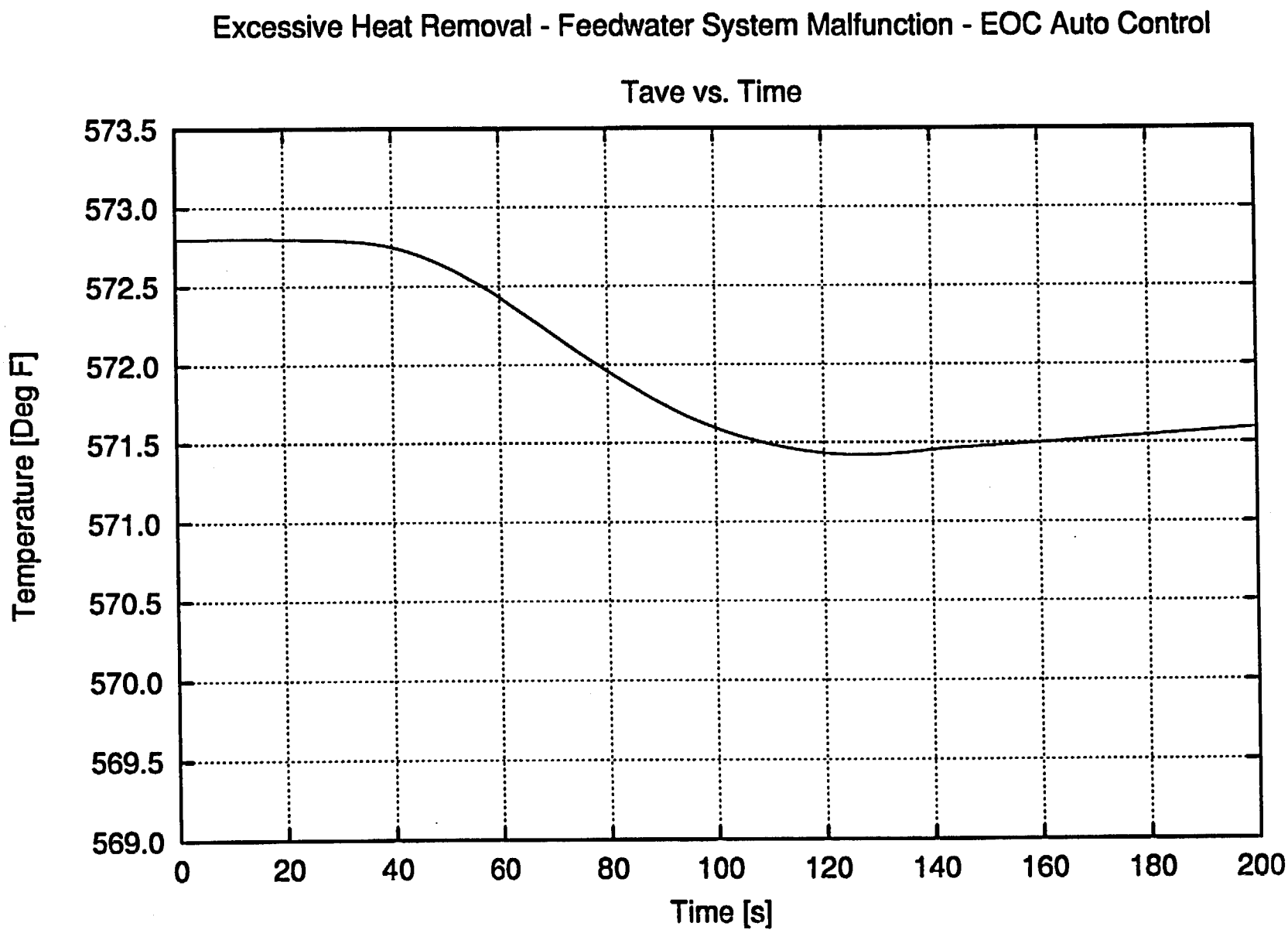
Excessive Heat Removal - Feedwater System Malfunction - EOC Auto Control
Tave vs. Time
Temperature [Deg F]
573.5
573.0
572.5
572.0
571.5
571.0
570.5
570.0
569.5
569.0
0
20
40
60
80
100
120
140
160
180
200
Time [s]

Figure 3.7-8

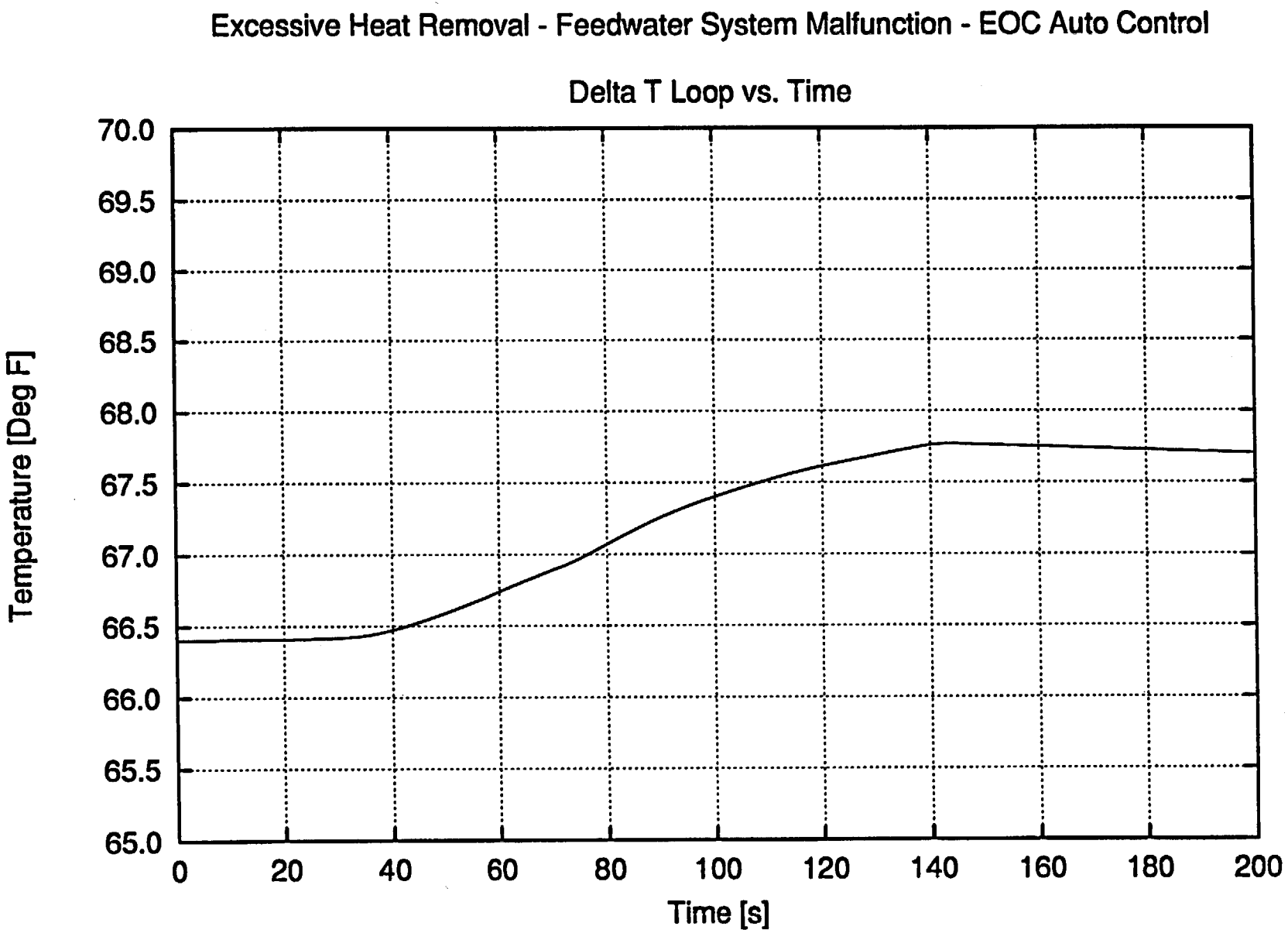
Excessive Heat Removal - Feedwater System Malfunction - EOC Auto Control
Delta T Loop vs. Time
Temperature [Deg F]
70.0
69.5
69.0
68.5
68.0
67.5
67.0
66.5
66.0
65.5
65.0
0
20
40
60
80
100
120
140
160
180
200
Time [s]

Figure 3.7-9

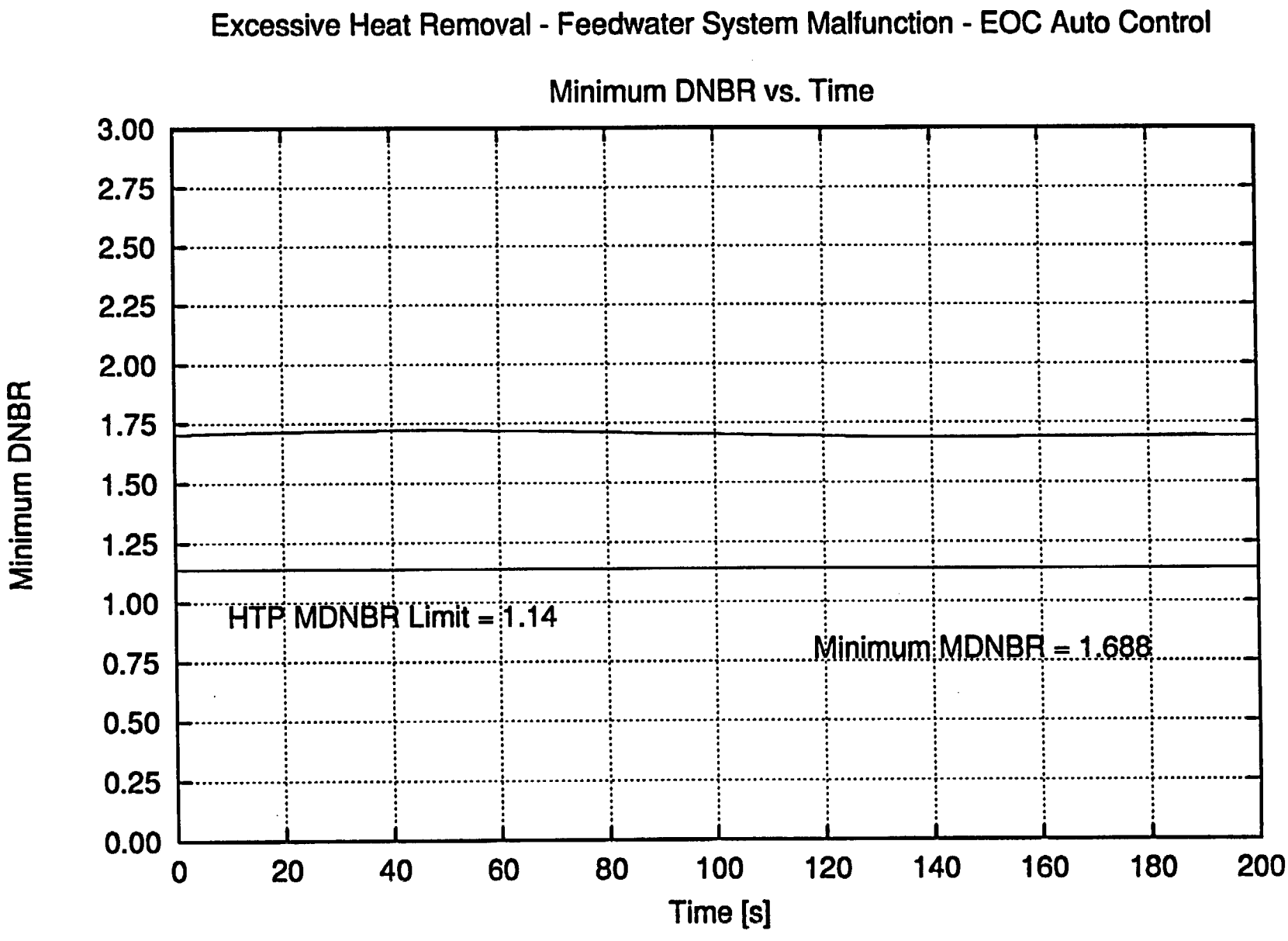
Excessive Heat Removal - Feedwater System Malfunction - EOC Auto Control
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.688
0
20
40
60
80
100
120
140
160
180
200
Time [s]

Figure 3.7-10

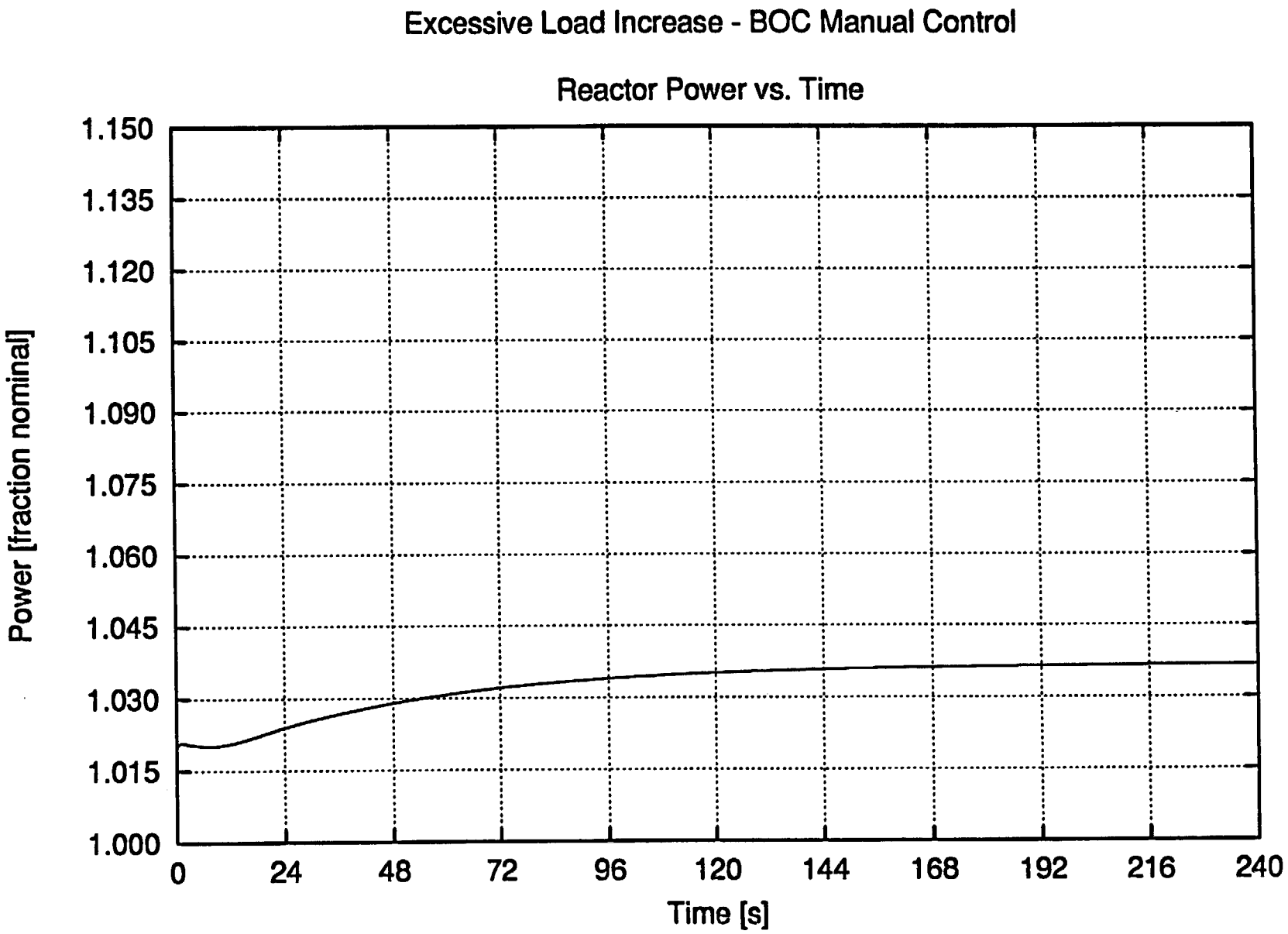
Excessive Load Increase - BOC Manual Control
Reactor Power vs. Time
Power [fraction nominal]
1.150
1.135
1.120
1.105
1.090
1.075
1.060
1.045
1.030
1.015
1.000
0
24
48
72
96
120
144
168
192
216
240
Time [s]

Figure 3.8-1

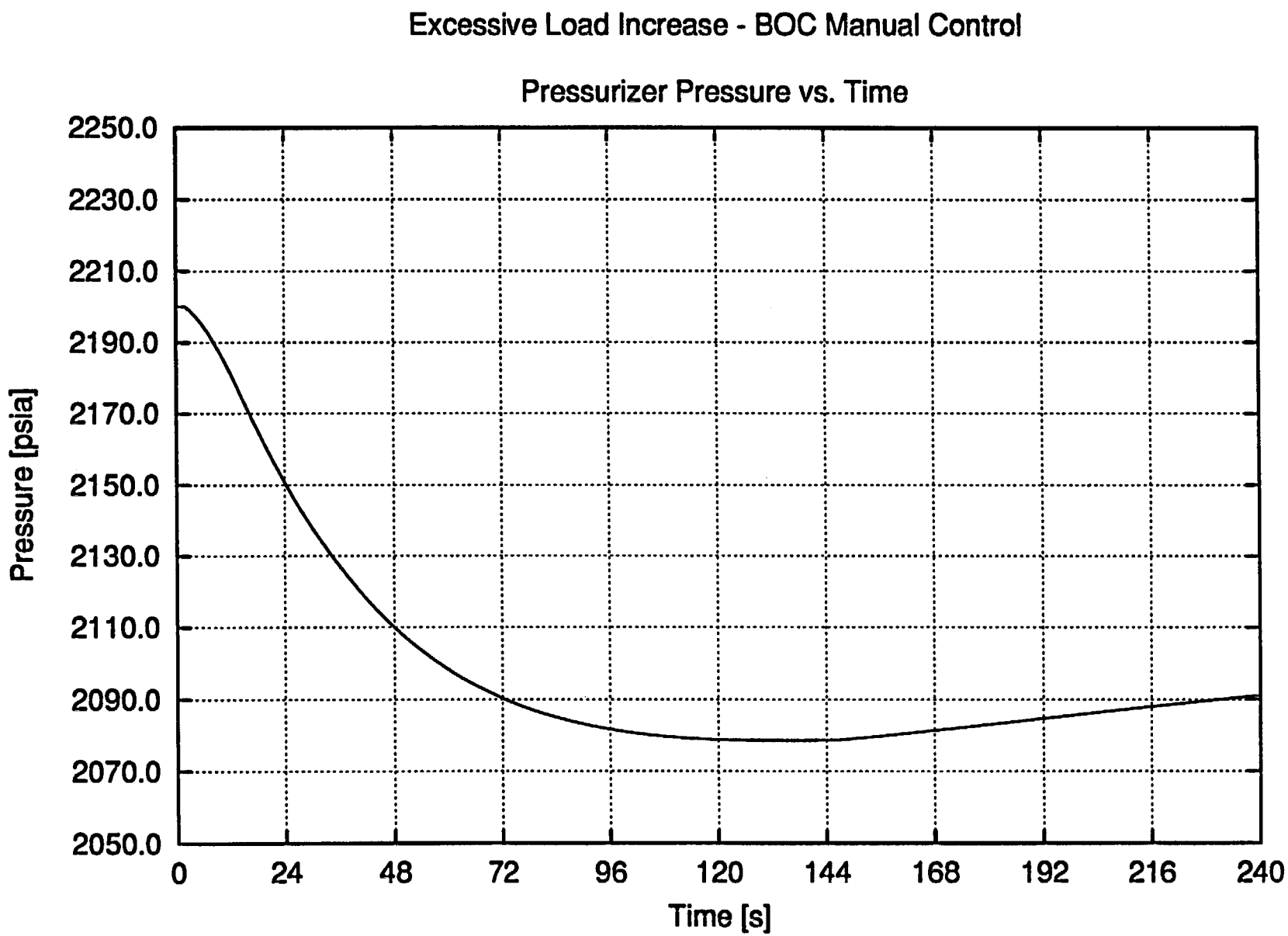
Excessive Load Increase - BOC Manual Control
Pressurizer Pressure vs. Time
Pressure [psia]
2250.0
2230.0
2210.0
2190.0
2170.0
2150.0
2130.0
2110.0
2090.0
2070.0
2050.0
0
24
48
72
96
120
144
168
192
216
240
Time [s]

Figure 3.8-2

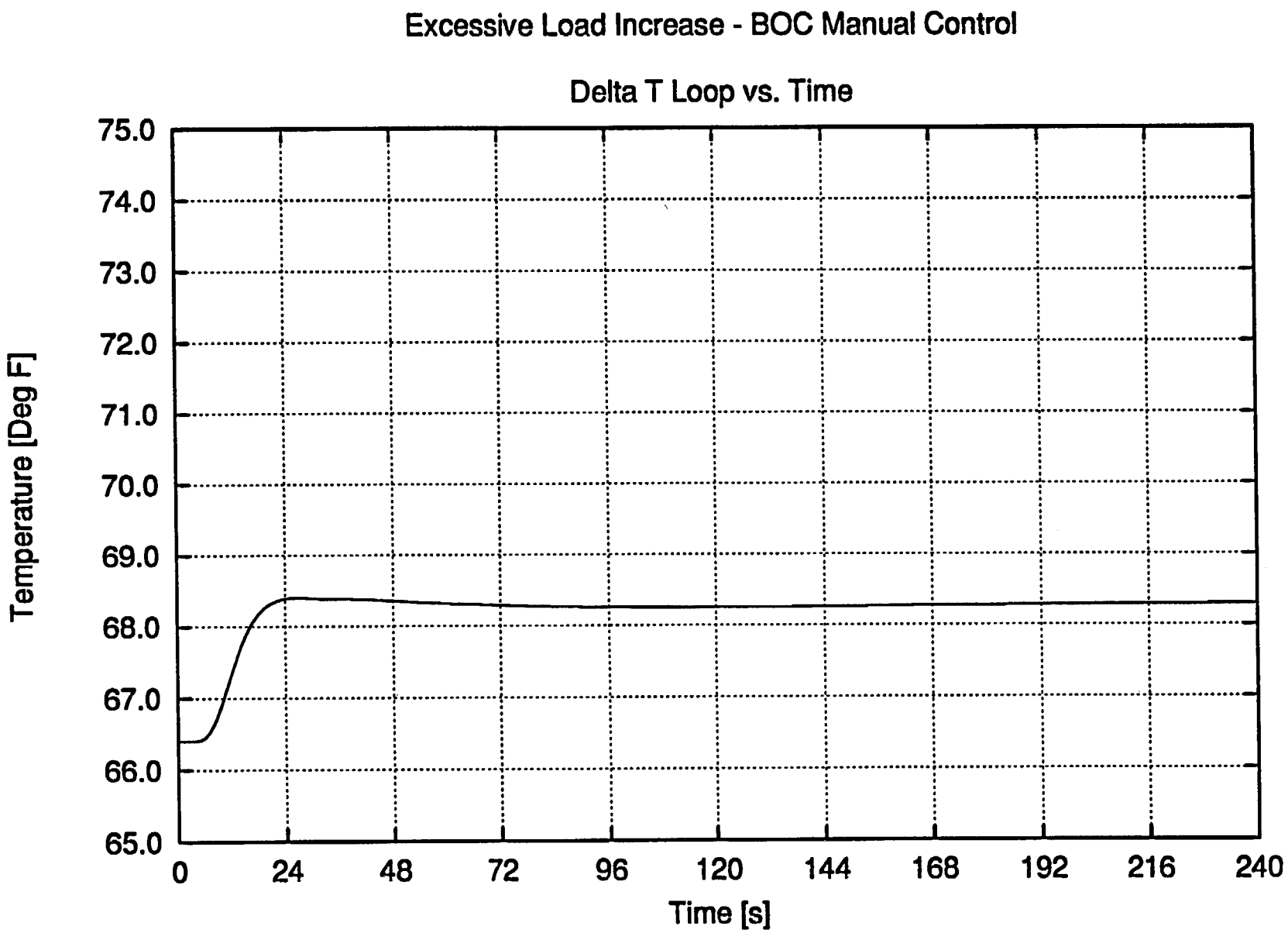
Excessive Load Increase - BOC Manual Control
Delta T Loop vs. Time
Temperature [Deg F]
75.0
74.0
73.0
72.0
71.0
70.0
69.0
68.0
67.0
66.0
65.0
0
24
48
72
96
120
144
168
192
216
240
Time [s]

Figure 3.8-3

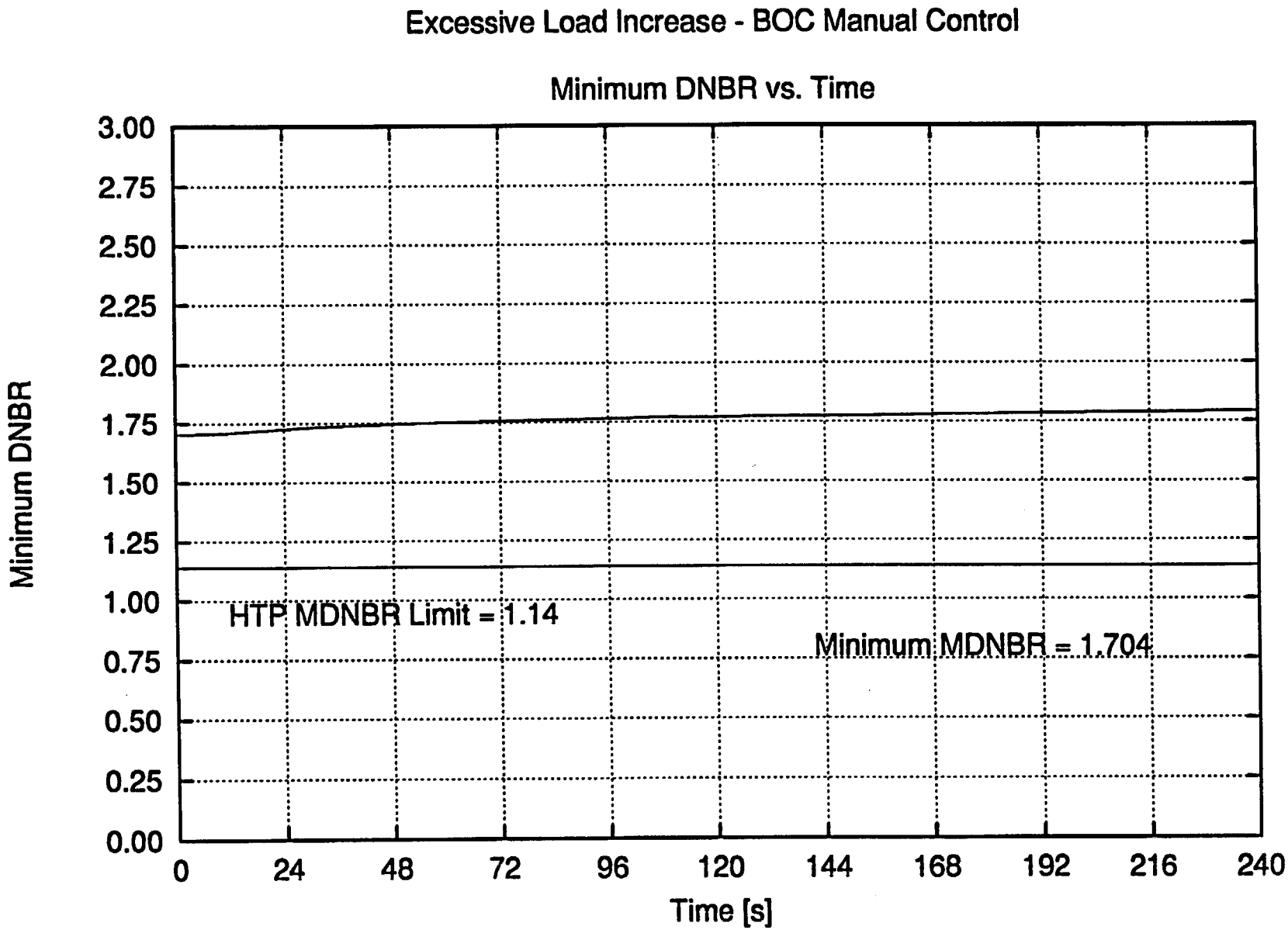
Excessive Load Increase - BOC Manual Control
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.704
0
24
48
72
96
120
144
168
192
216
240
Time [s]

Figure 3.8-4

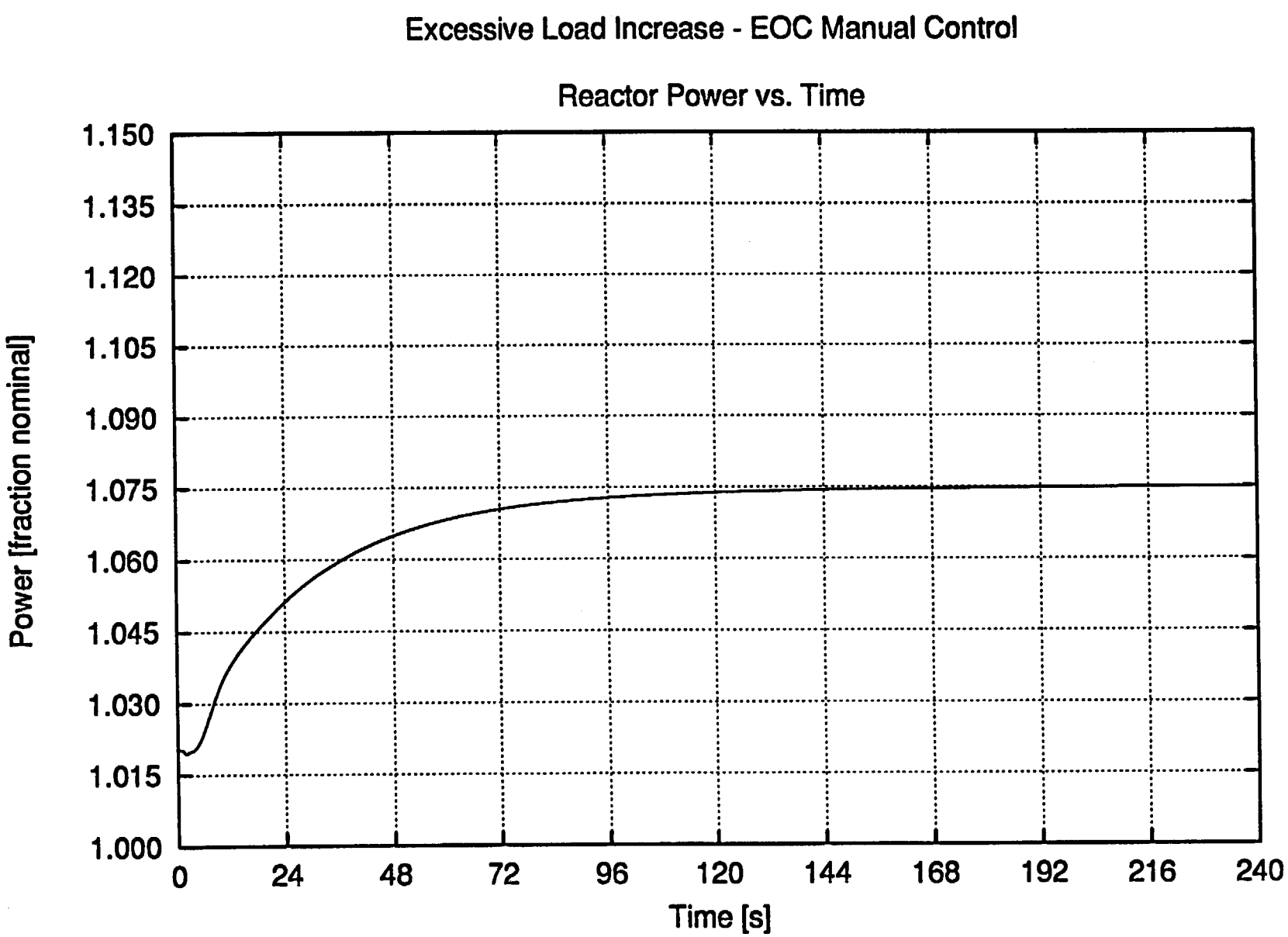
Excessive Load Increase - EOC Manual Control
Reactor Power vs. Time
Power [fraction nominal]
1.150
1.135
1.120
1.105
1.090
1.075
1.060
1.045
1.030
1.015
1.000
0
24
48
72
96
120
144
168
192
216
240
Time [s]

Figure 3.8-5

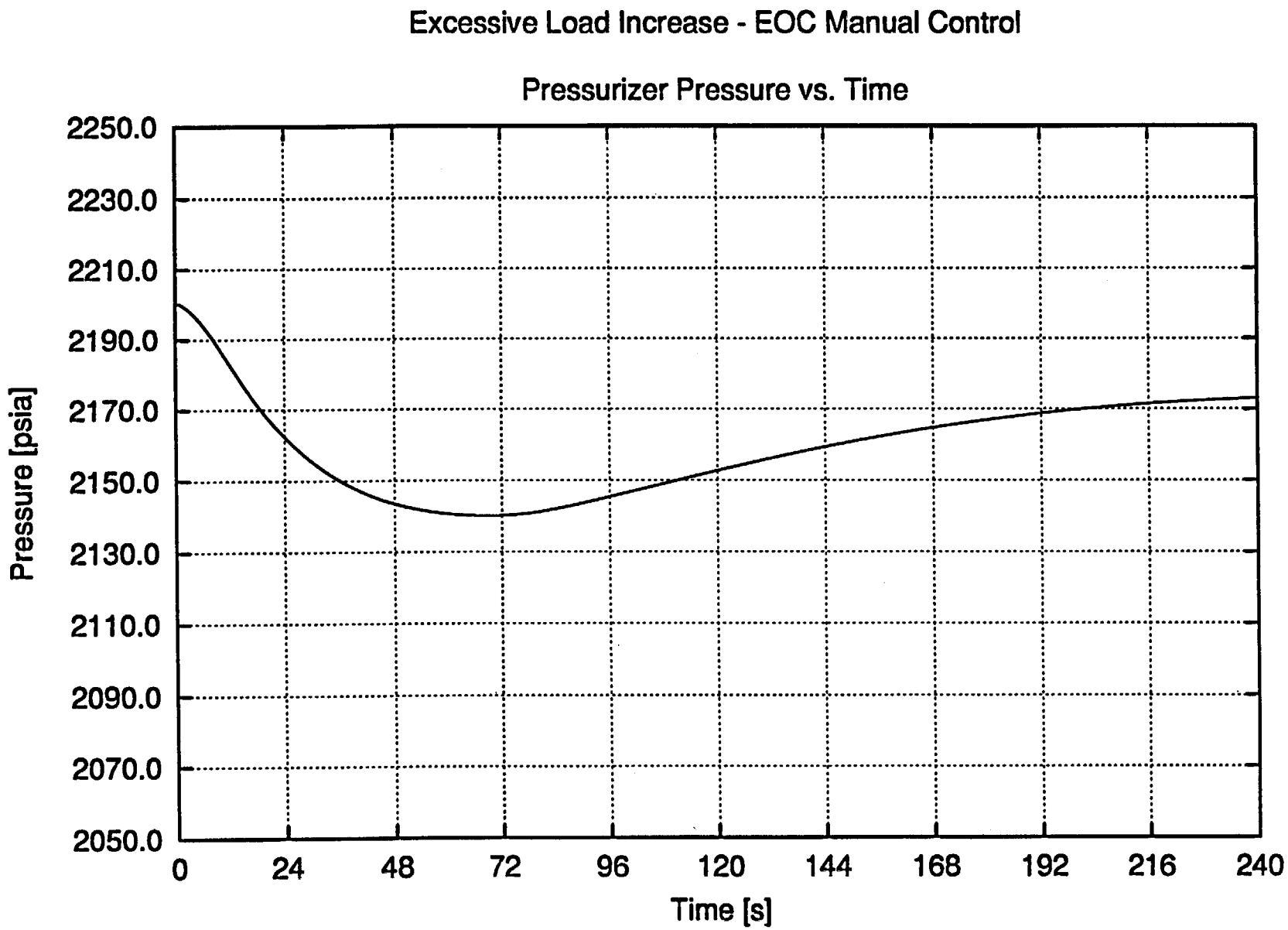
Excessive Load Increase - EOC Manual Control
Pressurizer Pressure vs. Time
Pressure [psia]
2250.0
2230.0
2210.0
2190.0
2170.0
2150.0
2130.0
2110.0
2090.0
2070.0
2050.0
0
24
48
72
96
120
144
168
192
216
240
Time [s]

Figure 3.8-6

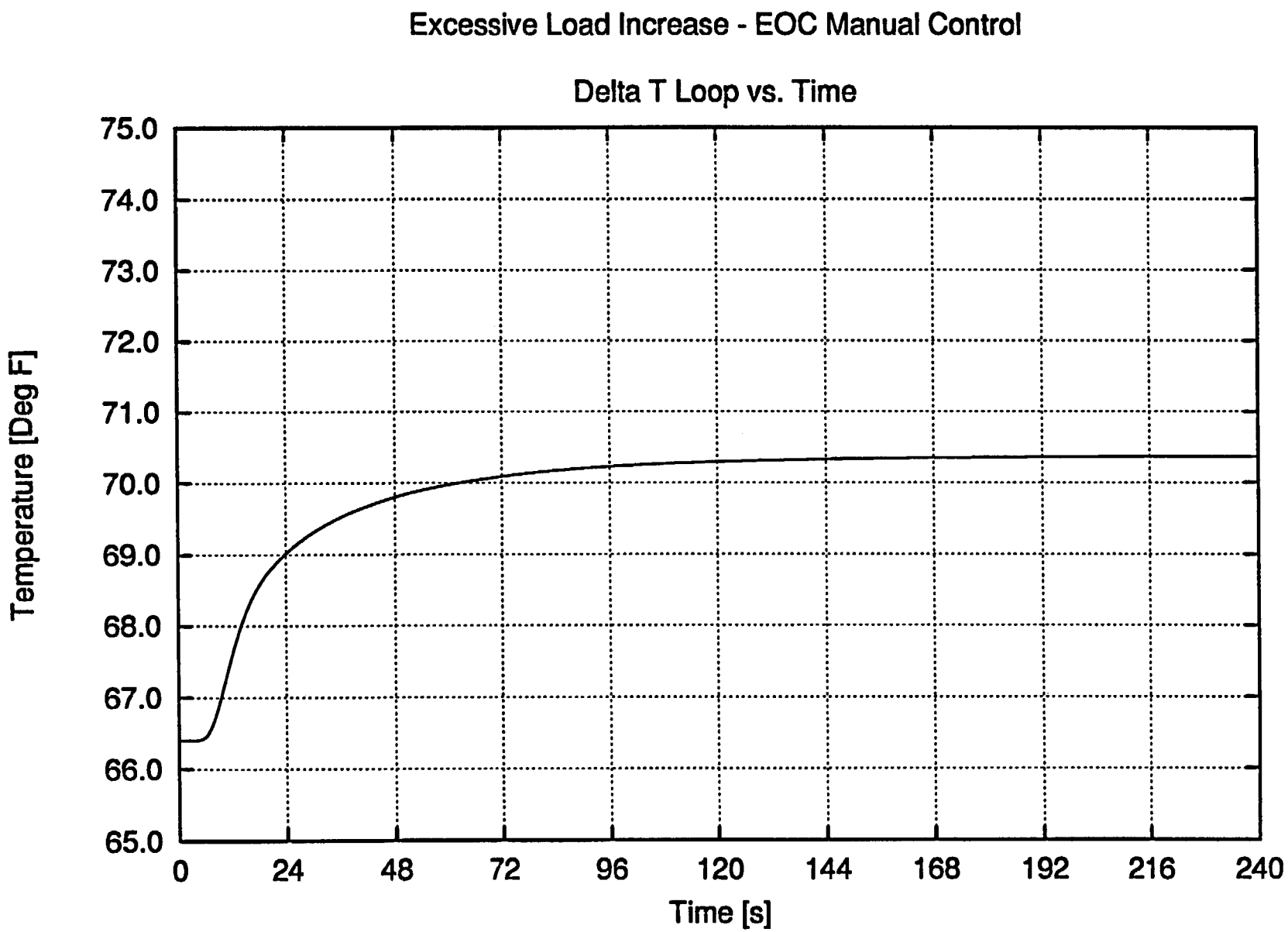
Excessive Load Increase - EOC Manual Control
Delta T Loop vs. Time
Temperature [Deg F]
75.0
74.0
73.0
72.0
71.0
70.0
69.0
68.0
67.0
66.0
65.0
0
24
48
72
96
120
144
168
192
216
240
Time [s]

Figure 3.8-7

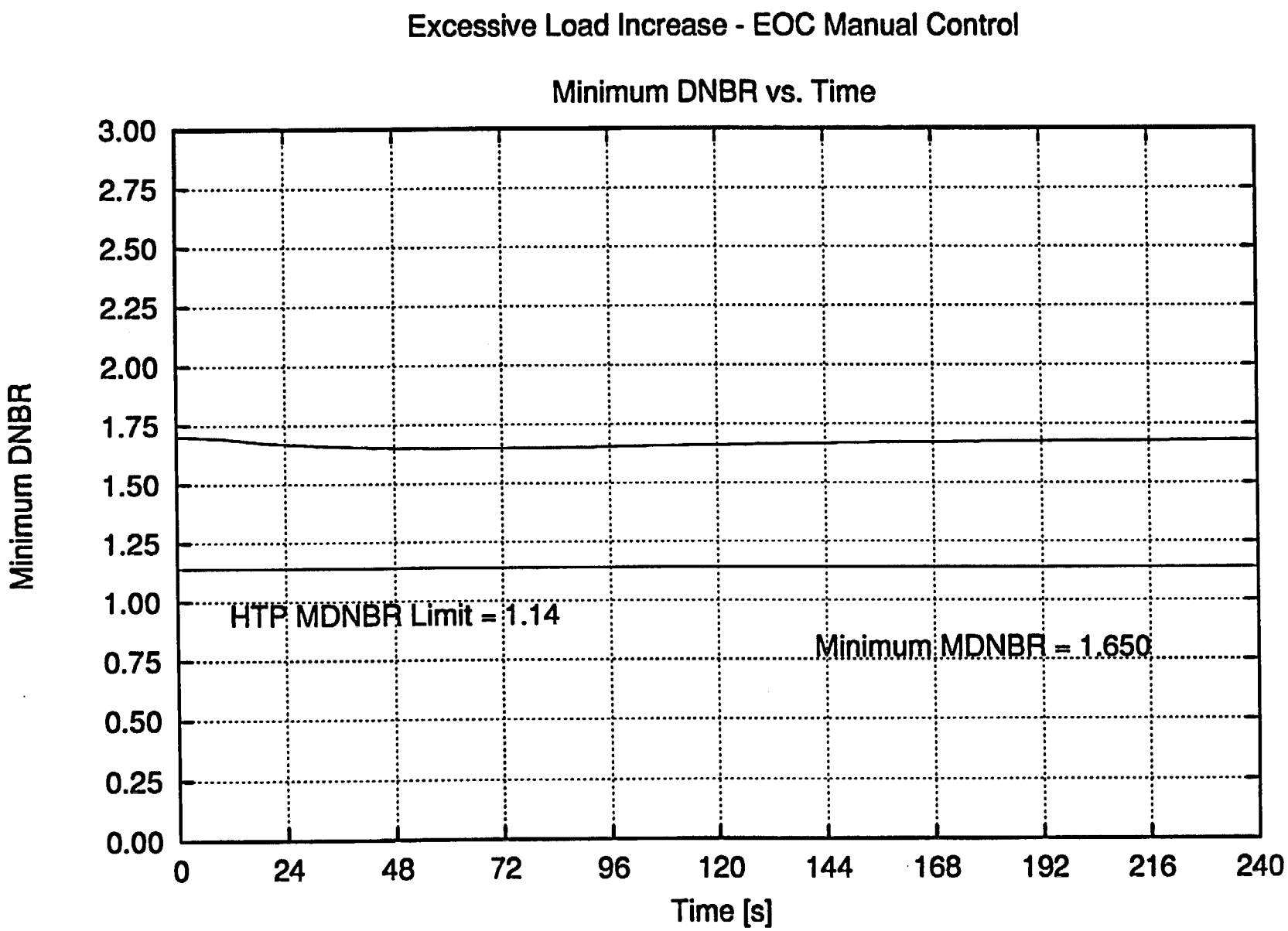
Excessive Load Increase - EOC Manual Control
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.650
0
24
48
72
96
120
144
168
192
216
240
Time [s]

Figure 3.8-8

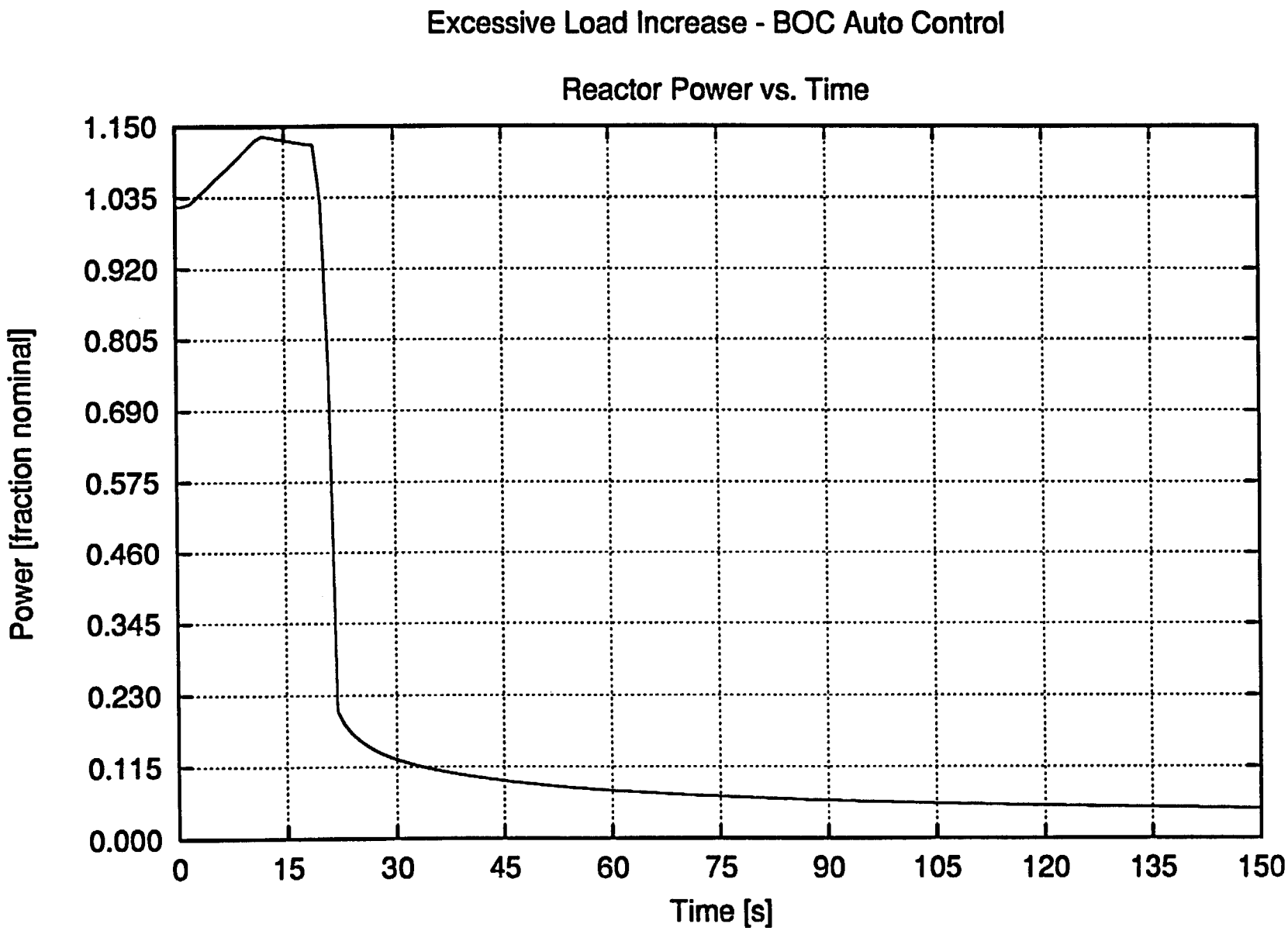
Excessive Load Increase - BOC Auto Control
Reactor Power vs. Time
Power [fraction nominal]
1.150
1.035
0.920
0.805
0.690
0.575
0.460
0.345
0.230
0.115
0.000
0
15
30
45
60
75
90
105
120
135
150
Time [s]

Figure 3.8-9

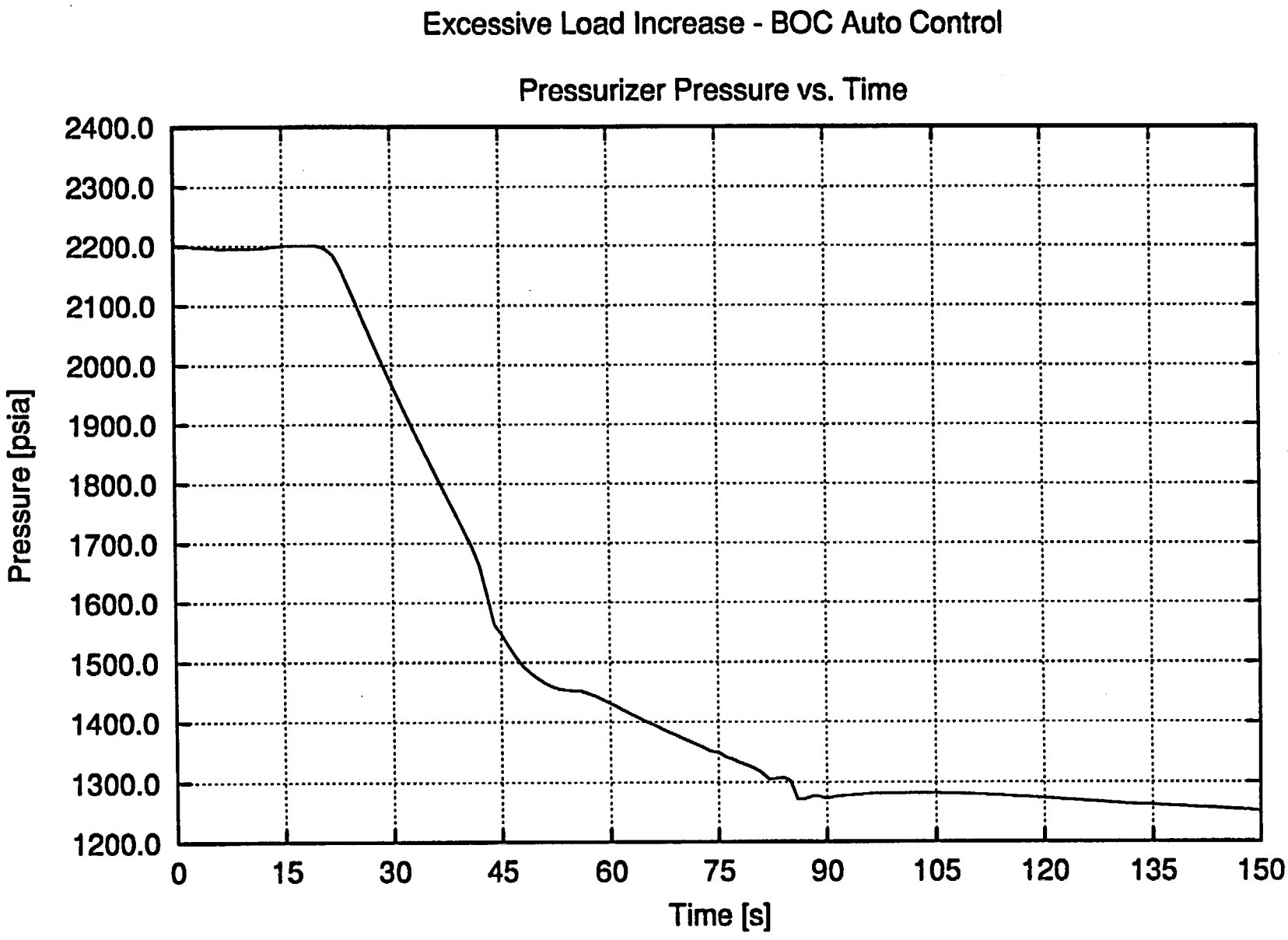
Excessive Load Increase - BOC Auto Control
Pressurizer Pressure vs. Time
Pressure [psia]
2400.0
2300.0
2200.0
2100.0
2000.0
1900.0
1800.0
1700.0
1600.0
1500.0
1400.0
1300.0
1200.0
0
15
30
45
60
75
90
105
120
135
150
Time [s]

Figure 3.8-10

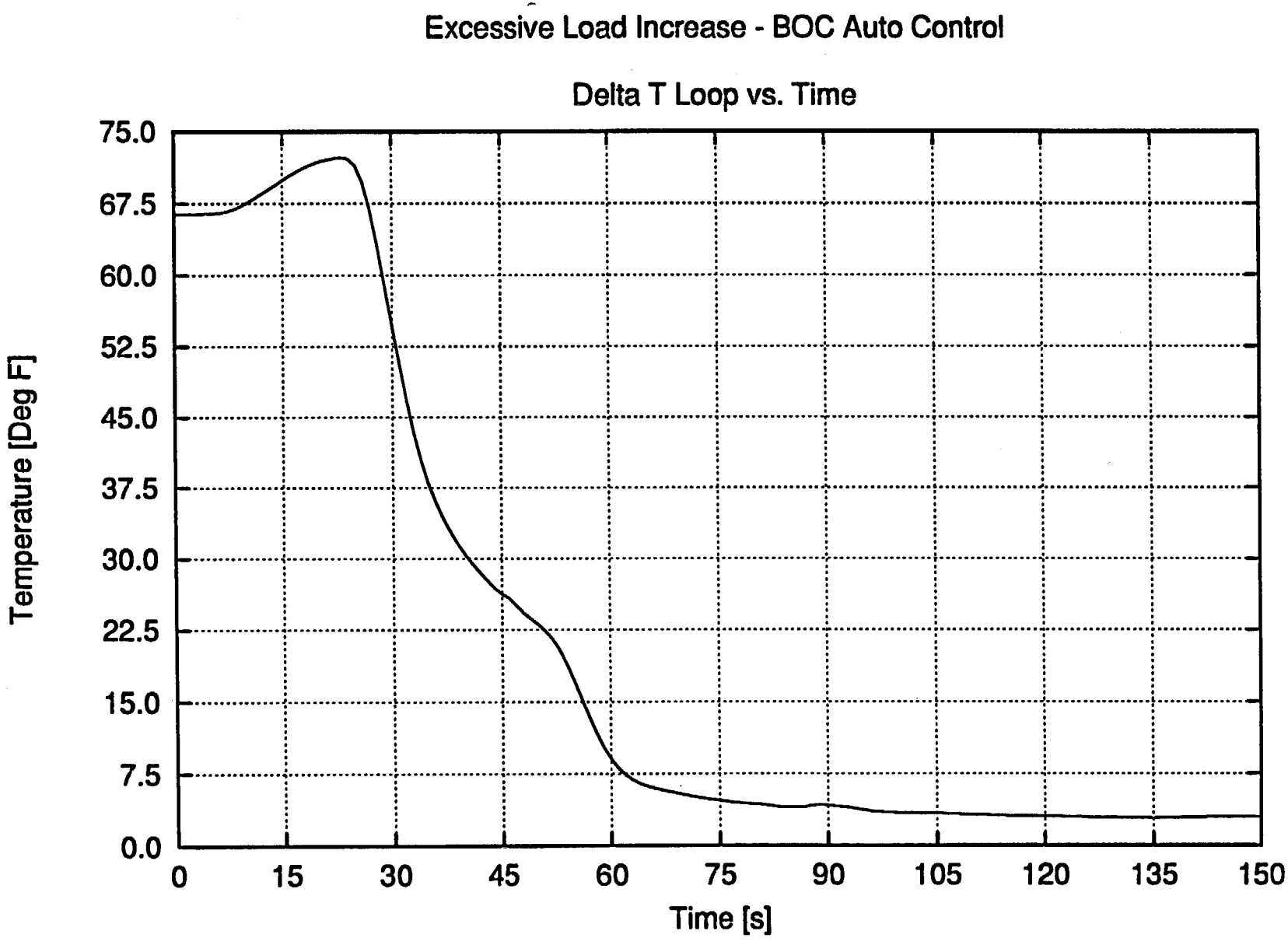
Excessive Load Increase - BOC Auto Control
Delta T Loop vs. Time
Temperature [Deg F]
75.0
67.5
60.0
52.5
45.0
37.5
30.0
22.5
15.0
7.5
0.0
0
15
30
45
60
75
90
105
120
135
150
Time [s]

Figure 3.8-11

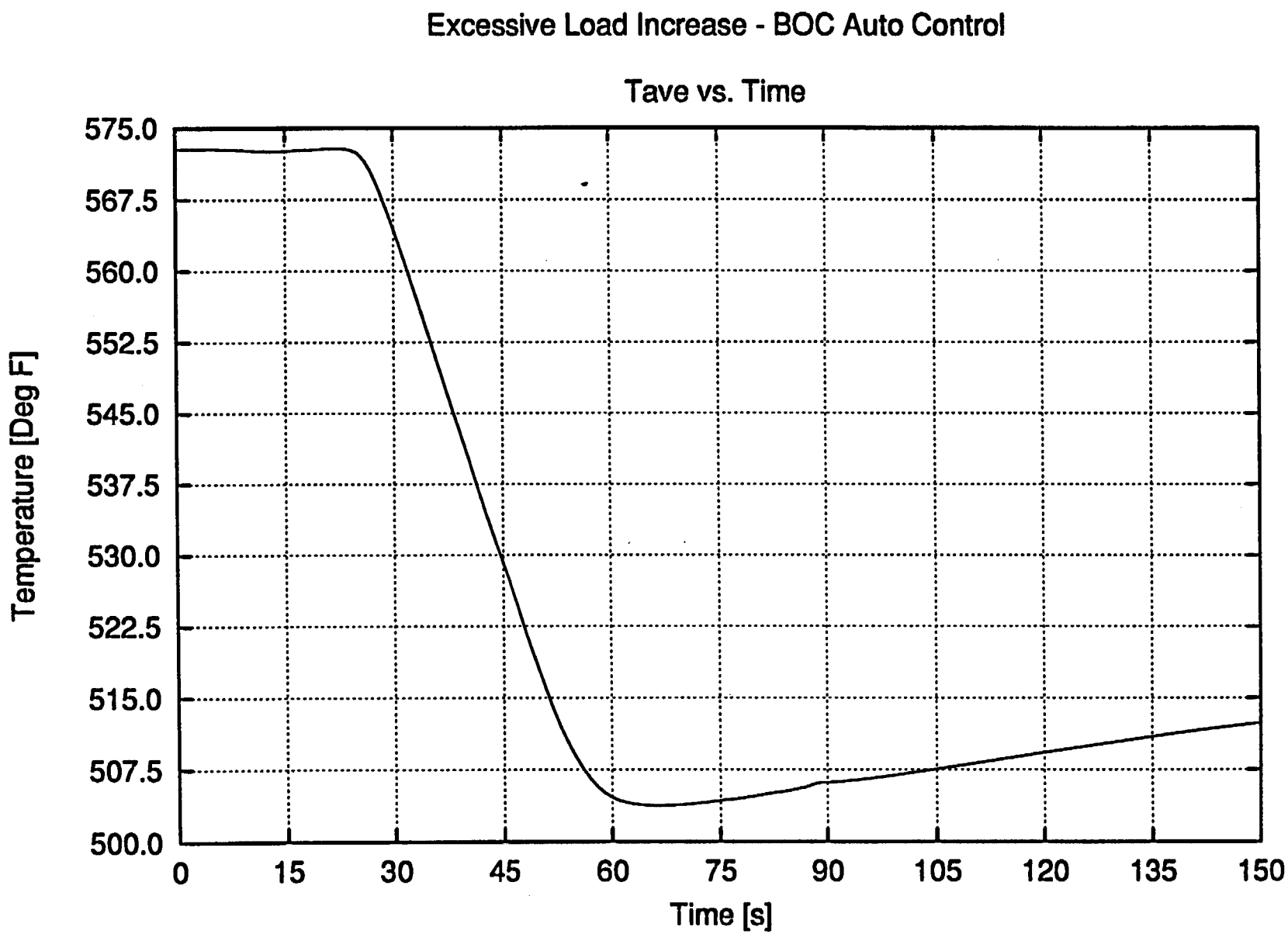
Excessive Load Increase - BOC Auto Control
Tave vs. Time
Temperature [Deg F]
575.0
567.5
560.0
552.5
545.0
537.5
530.0
522.5
515.0
507.5
500.0
0
15
30
45
60
75
90
105
120
135
150
Time [s]

Figure 3.8-12

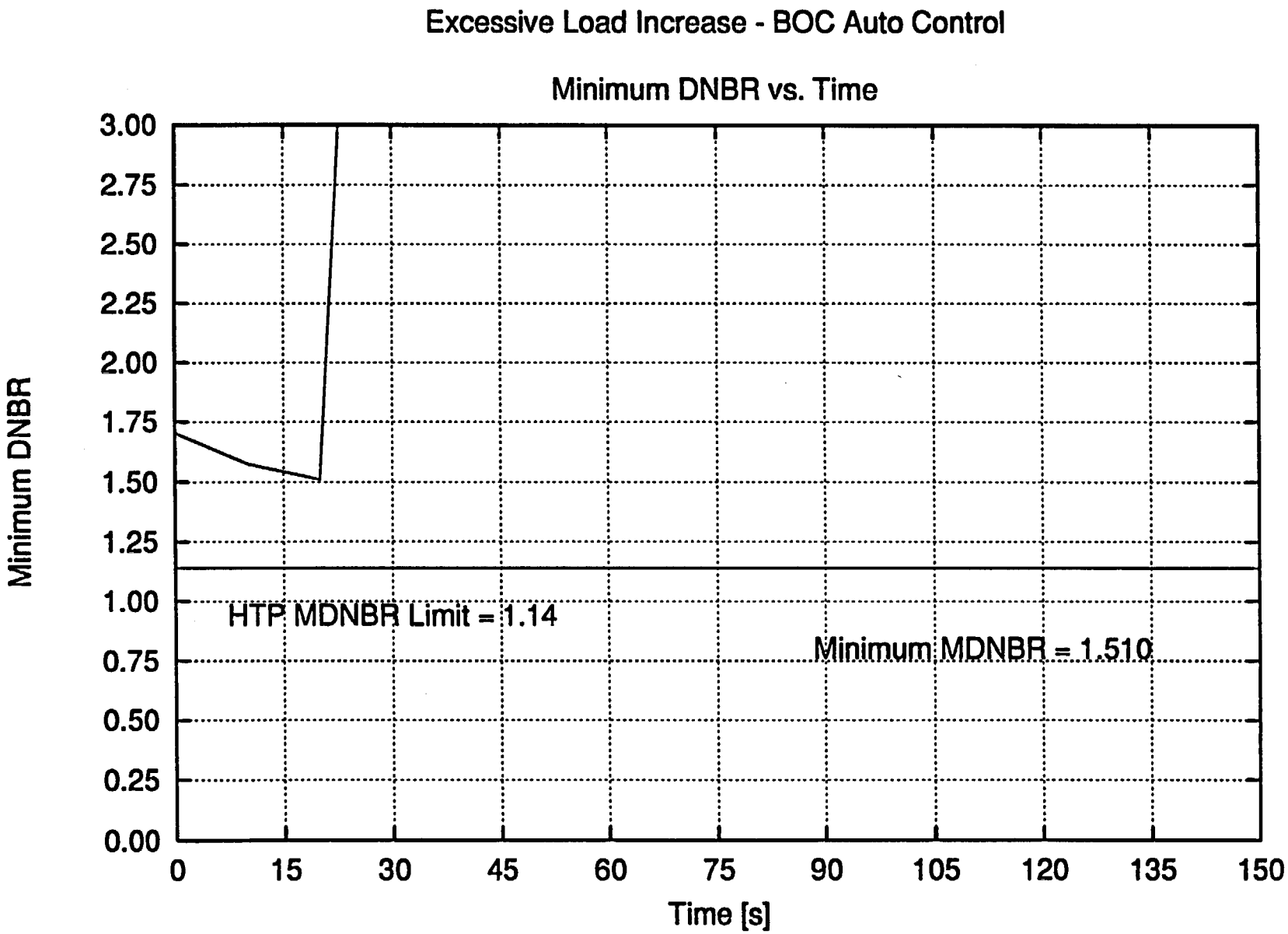
Excessive Load Increase - BOC Auto Control
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.510
0
15
30
45
60
75
90
105
120
135
150
Time [s]

Figure 3.8-13

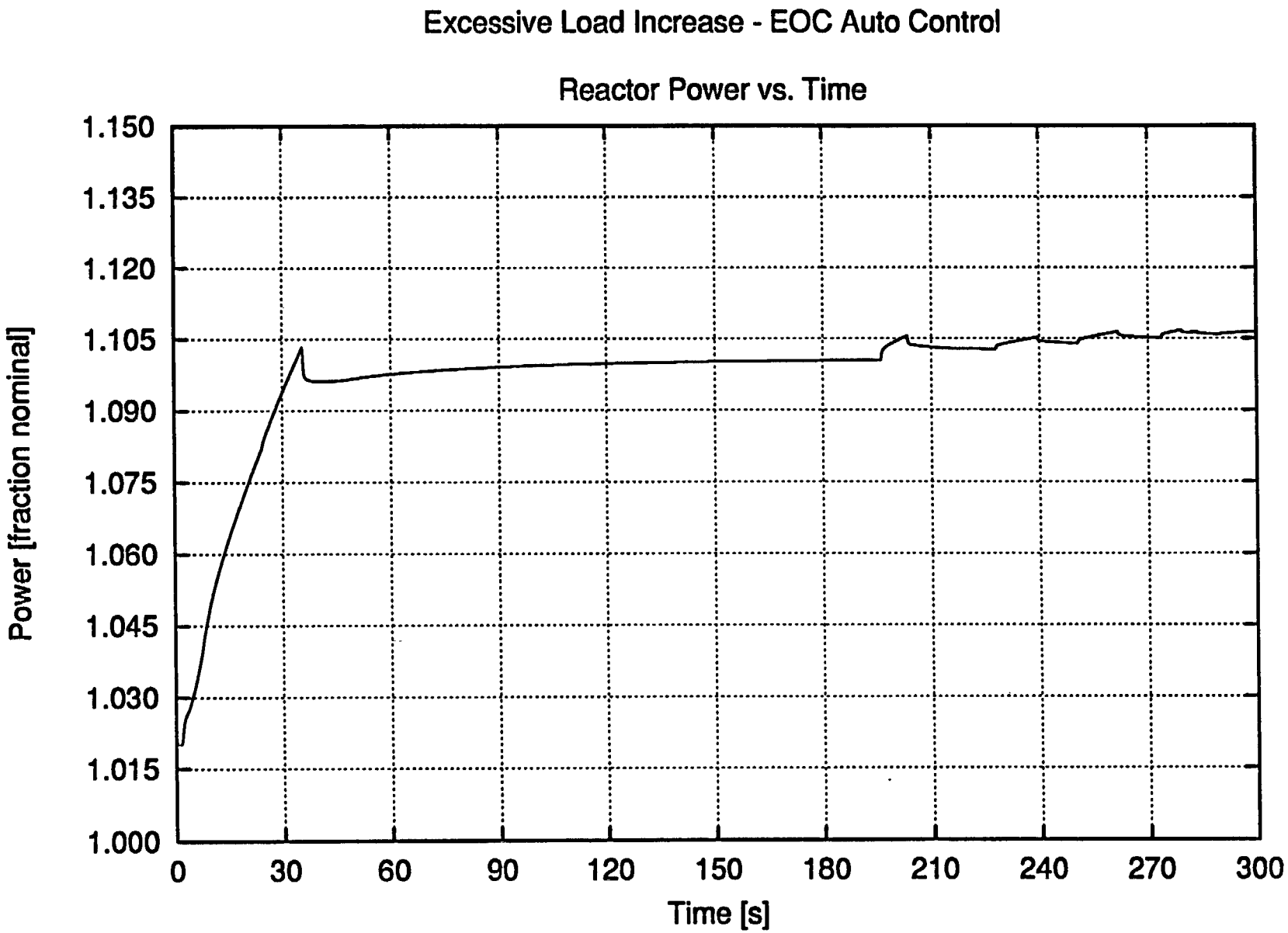
Excessive Load Increase - EOC Auto Control
Reactor Power vs. Time
Power [fraction nominal]
1.150
1.135
1.120
1.105
1.090
1.075
1.060
1.045
1.030
1.015
1.000
0
30
60
90
120
150
180
210
240
270
300
Time [s]

Figure 3.8-14

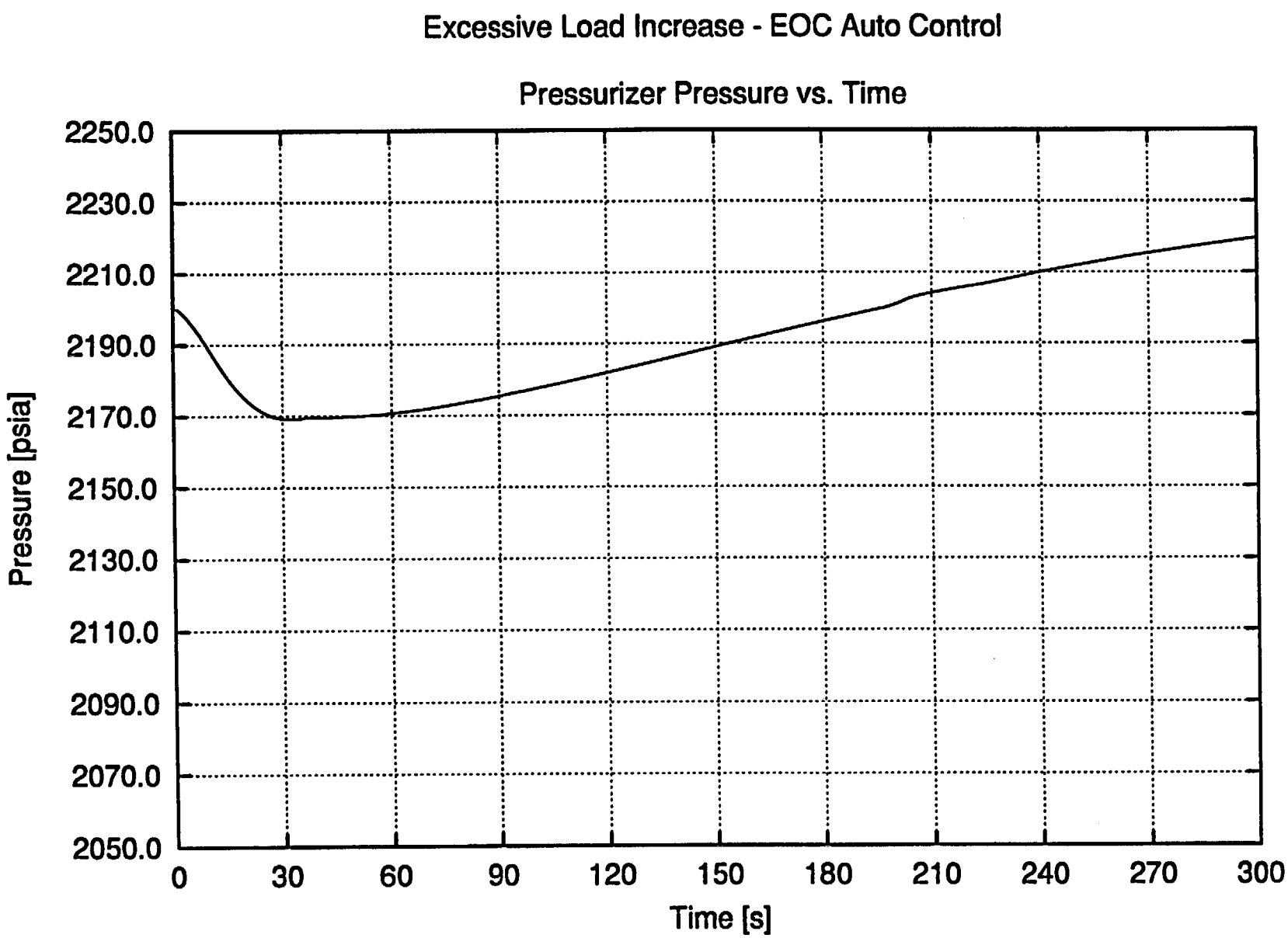
Excessive Load Increase - EOC Auto Control
Pressurizer Pressure vs. Time
Pressure [psia]
2250.0
2230.0
2210.0
2190.0
2170.0
2150.0
2130.0
2110.0
2090.0
2070.0
2050.0
0
30
60
90
120
150
180
210
240
270
300
Time [s]

Figure 3.8-15

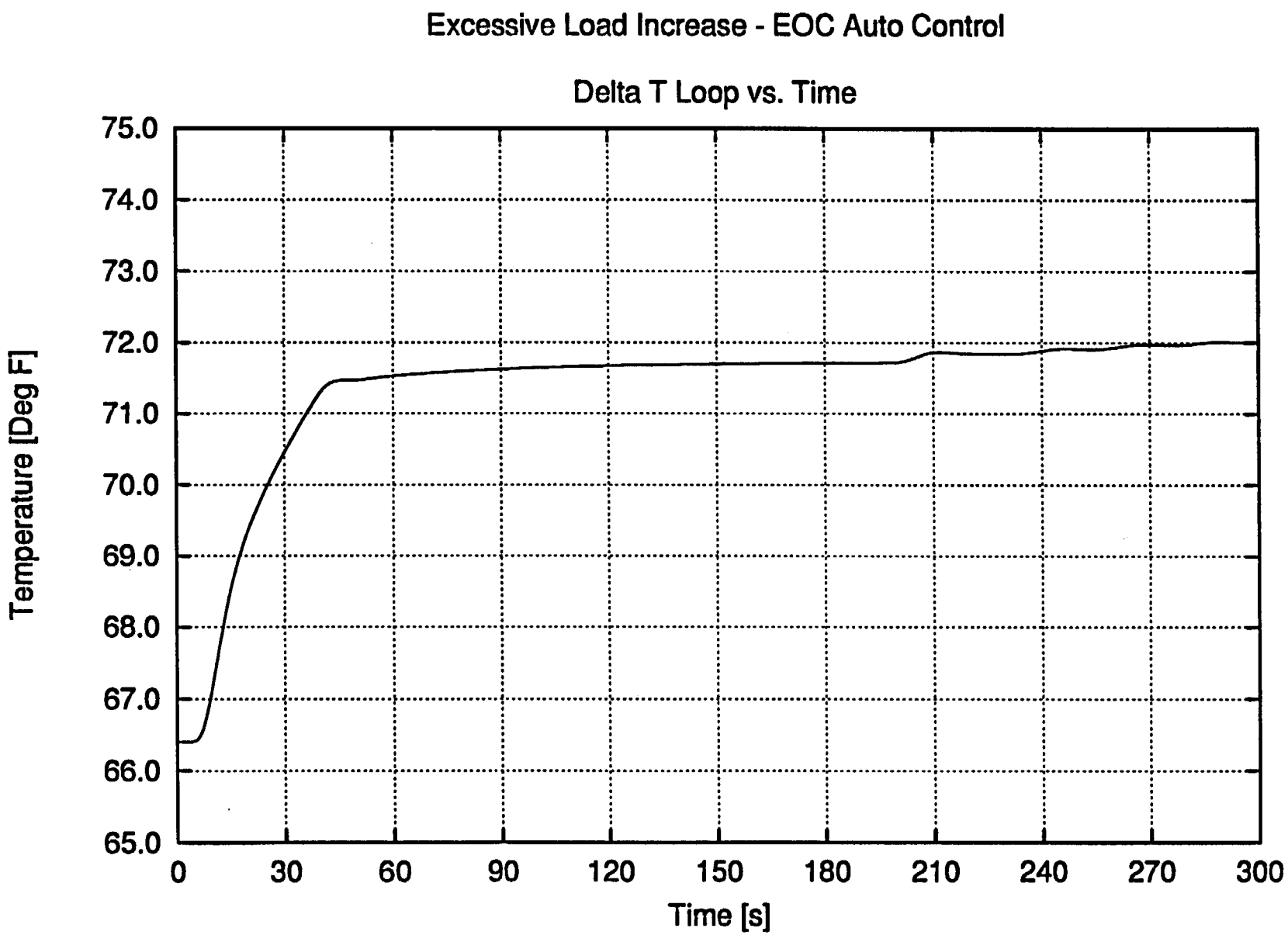
Excessive Load Increase - EOC Auto Control
Delta T Loop vs. Time
Temperature [Deg F]
75.0
74.0
73.0
72.0
71.0
70.0
69.0
68.0
67.0
66.0
65.0
0
30
60
90
120
150
180
210
240
270
300
Time [s]

Figure 3.8-16

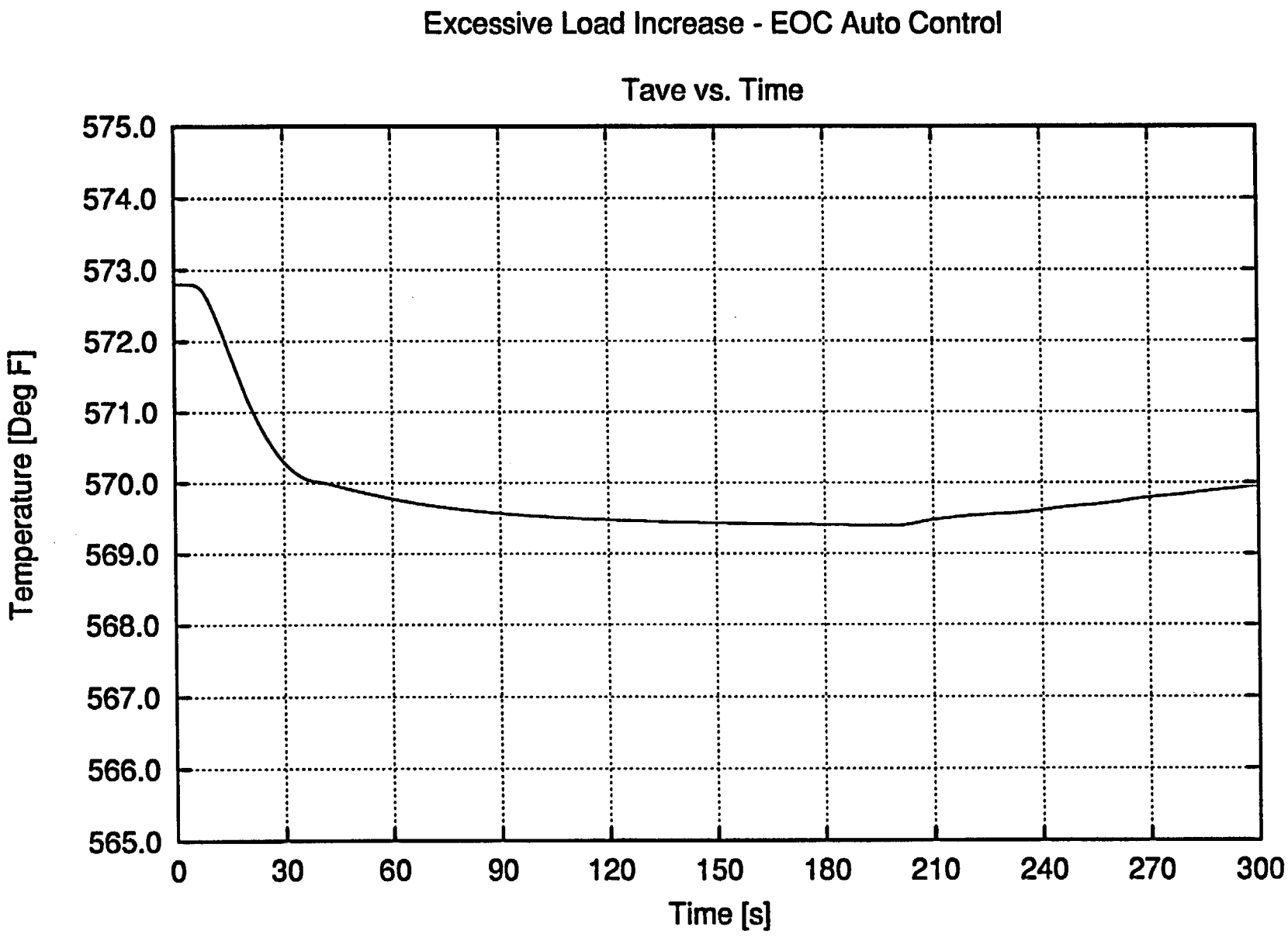
Excessive Load Increase - EOC Auto Control
Tave vs. Time
575.0
574.0
573.0
572.0
571.0
570.0
569.0
568.0
567.0
566.0
565.0
Temperature [Deg F]
0
30
60
90
120
150
180
210
240
270
300
Time [s]

Figure 3.8-17

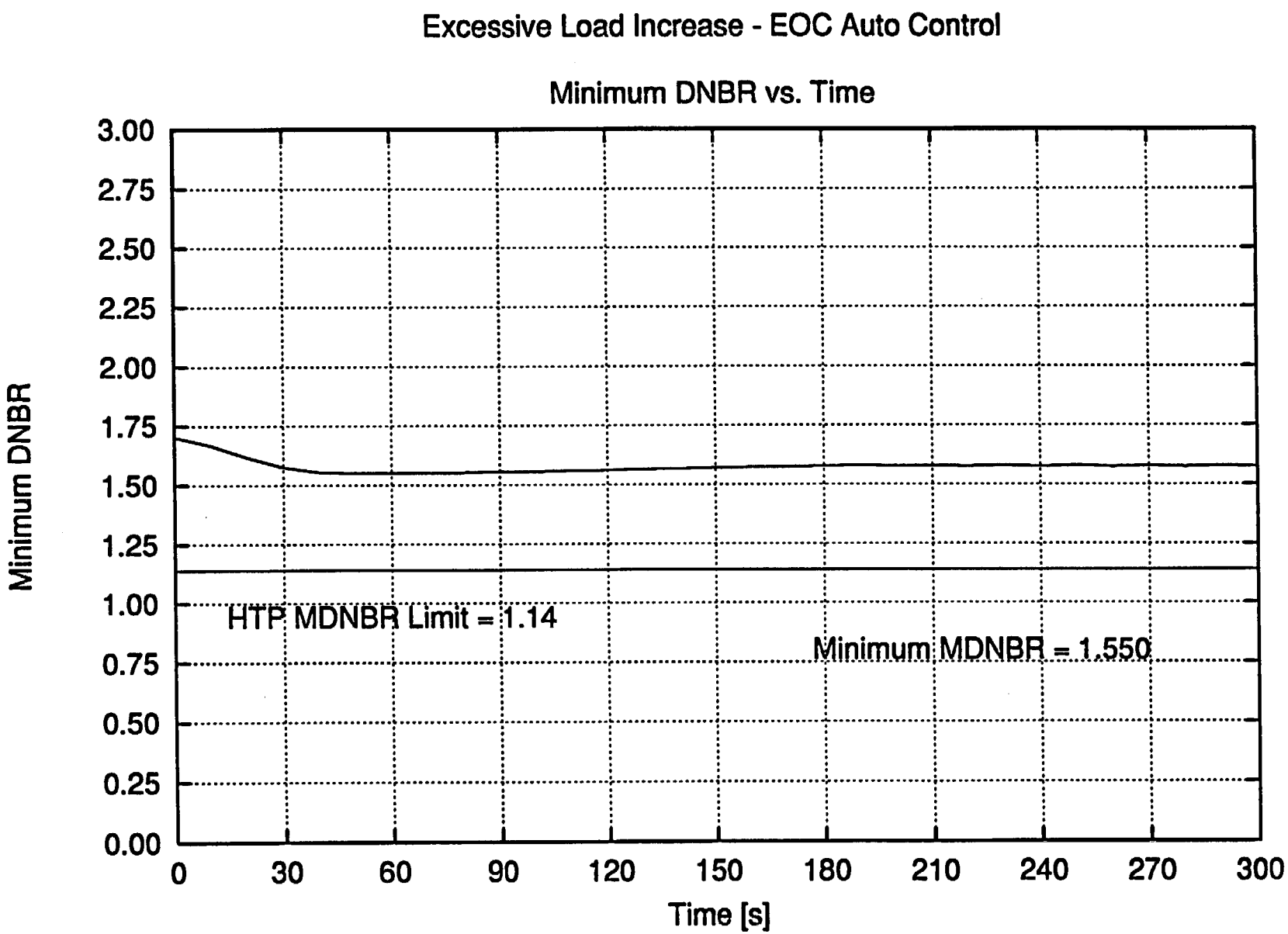
Excessive Load Increase - EOC Auto Control
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0
30
60
90
120
150
180
210
240
270
300
Time [s]
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.550

Figure 3.8-18

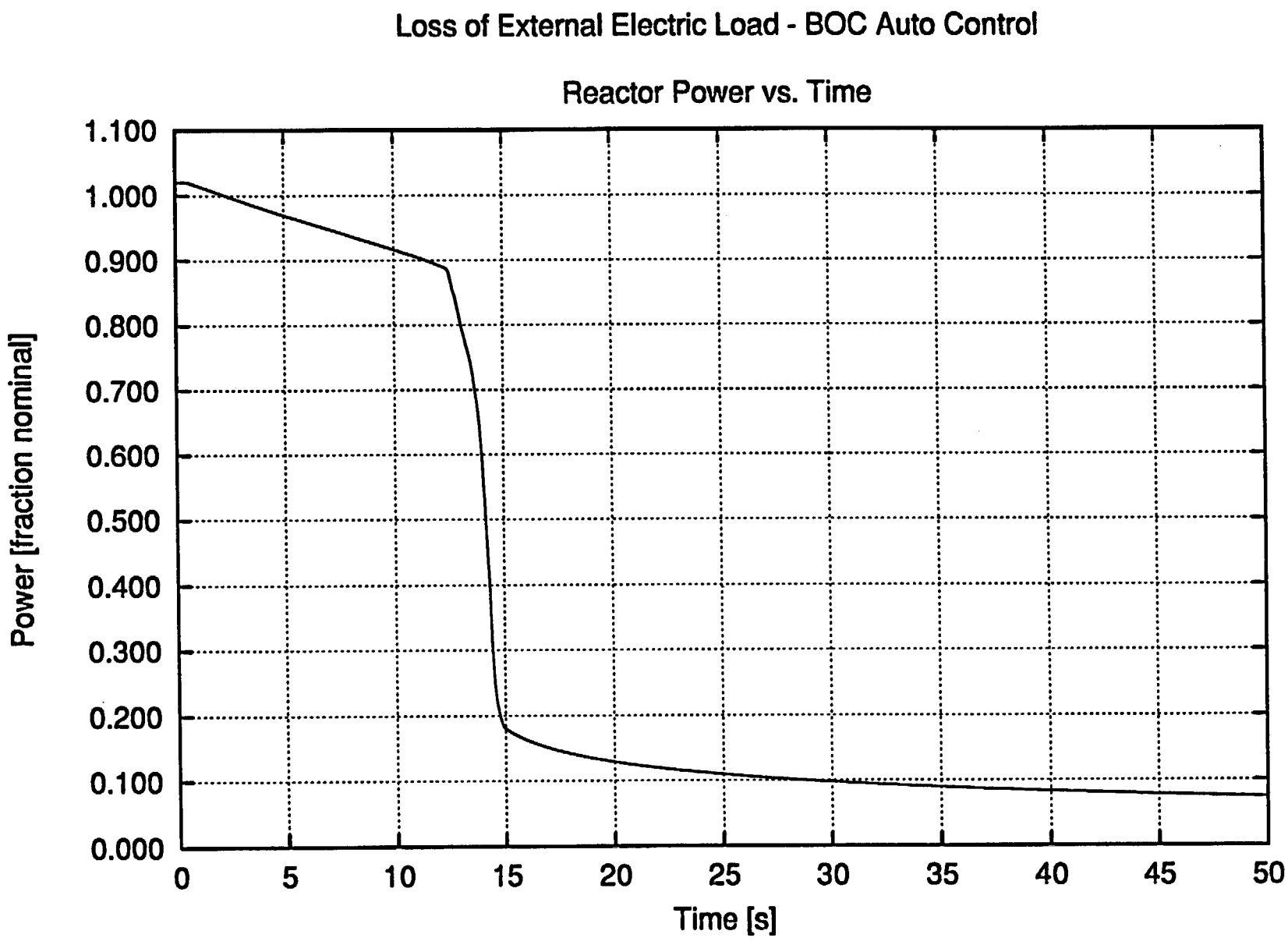
Loss of External Electric Load - BOC Auto Control
Reactor Power vs. Time
Power [fraction nominal]
1.100
1.000
0.900
0.800
0.700
0.600
0.500
0.400
0.300
0.200
0.100
0.000
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-1

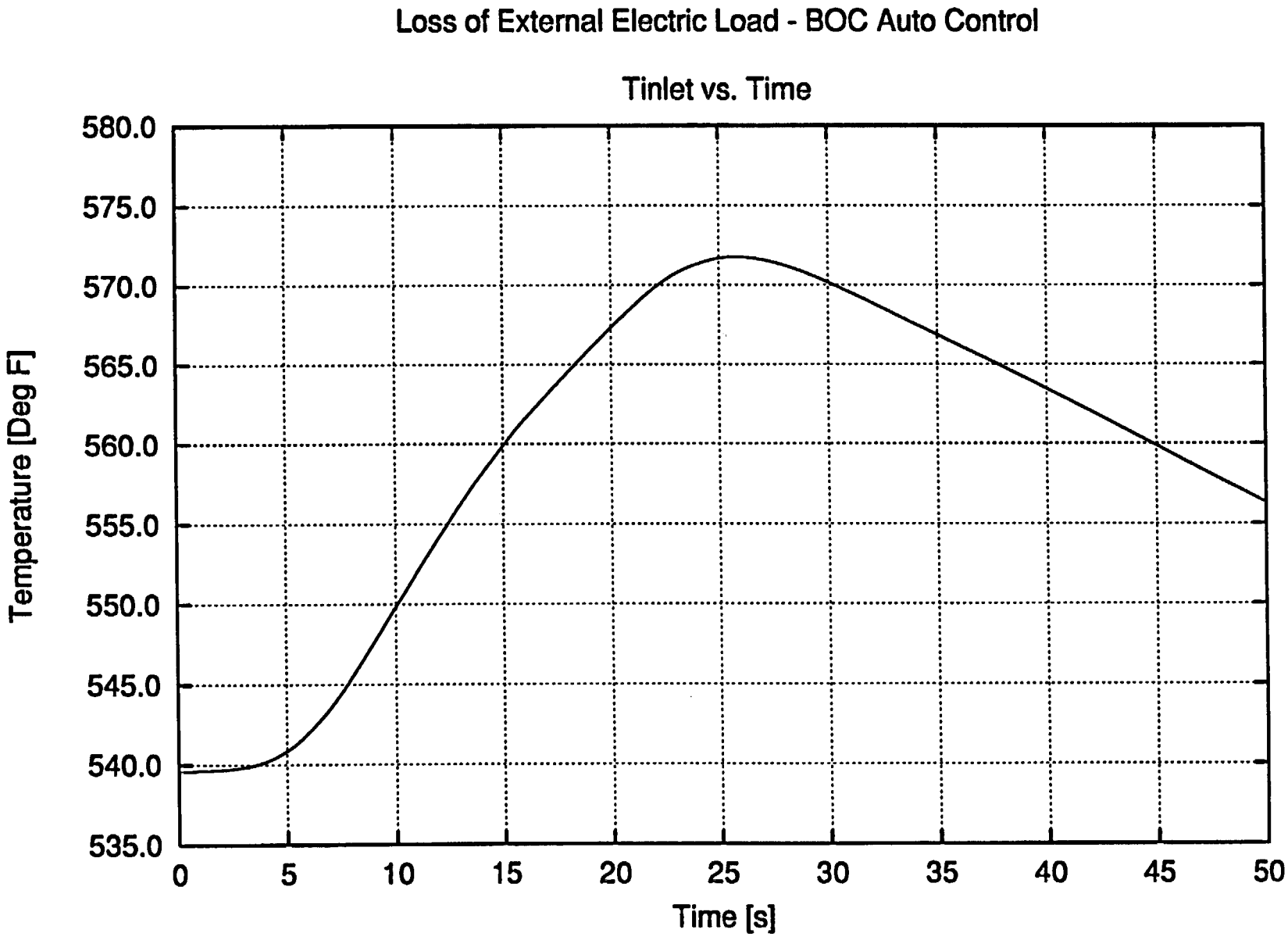
Loss of External Electric Load - BOC Auto Control
Tinlet vs. Time
580.0
575.0
570.0
565.0
560.0
555.0
550.0
545.0
540.0
535.0
Temperature [Deg F]
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-2

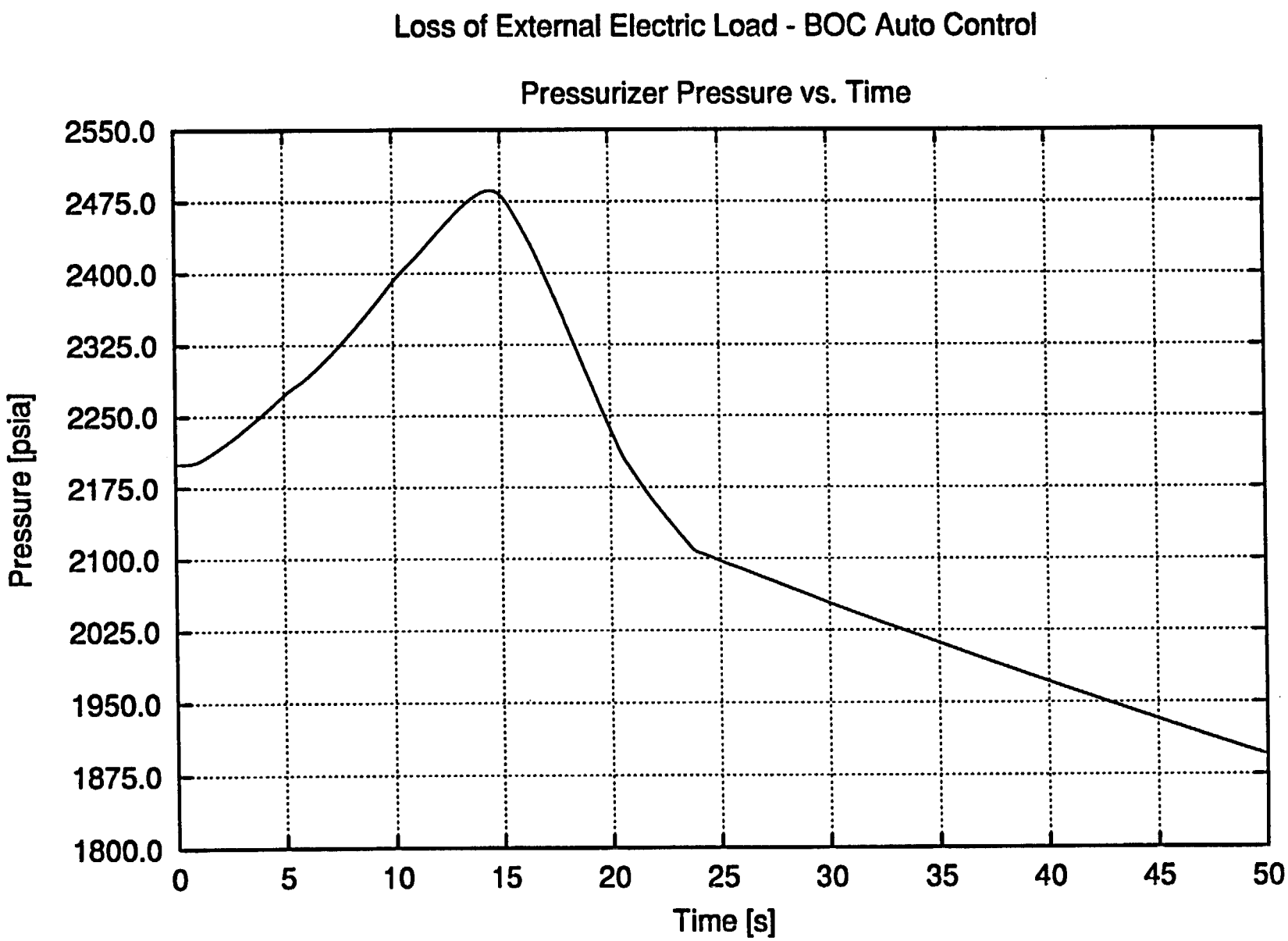
Loss of External Electric Load - BOC Auto Control
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-3

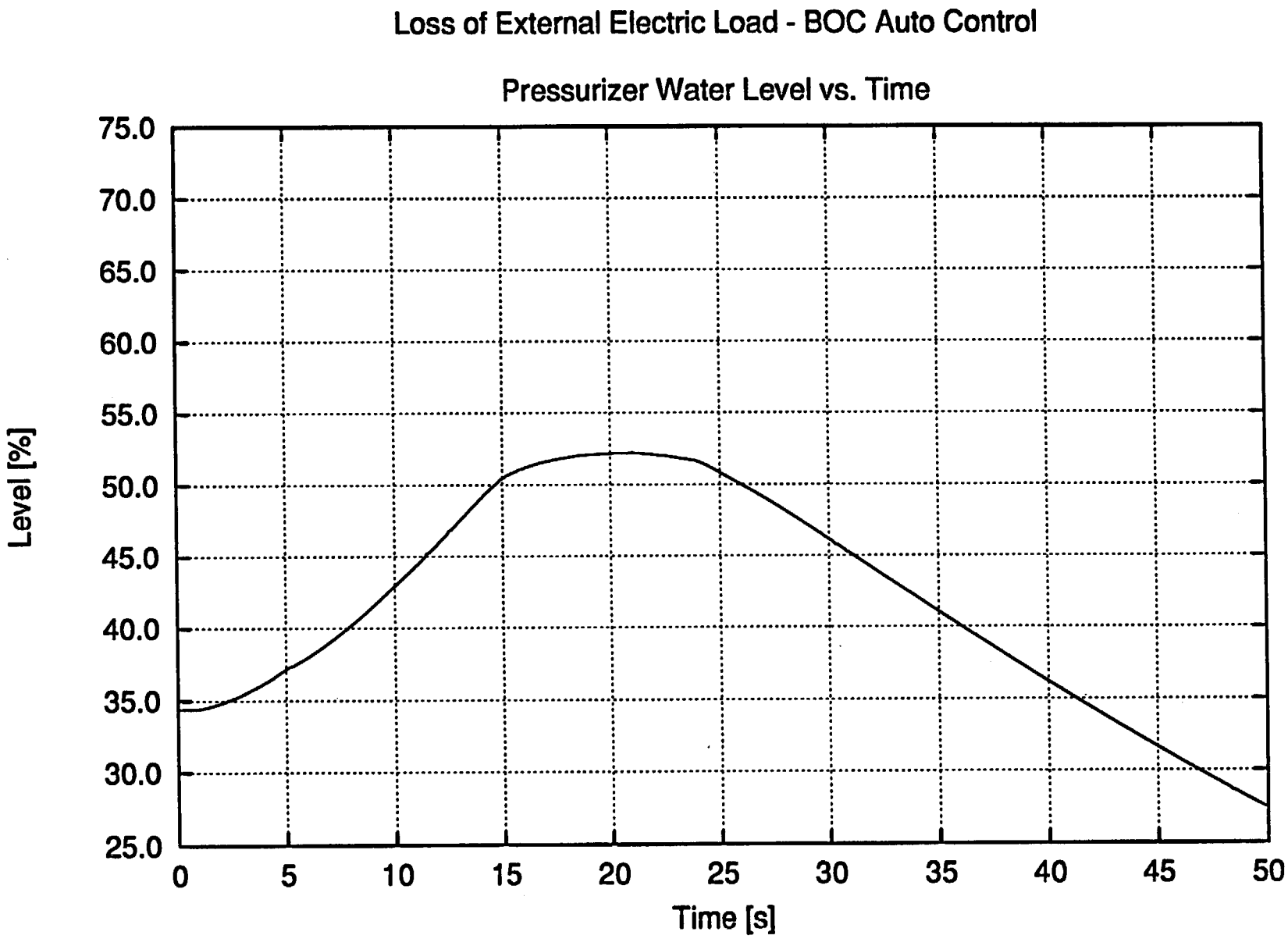
Loss of External Electric Load - BOC Auto Control
Pressurizer Water Level vs. Time
Level [%]
75.0
70.0
65.0
60.0
55.0
50.0
45.0
40.0
35.0
30.0
25.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-4

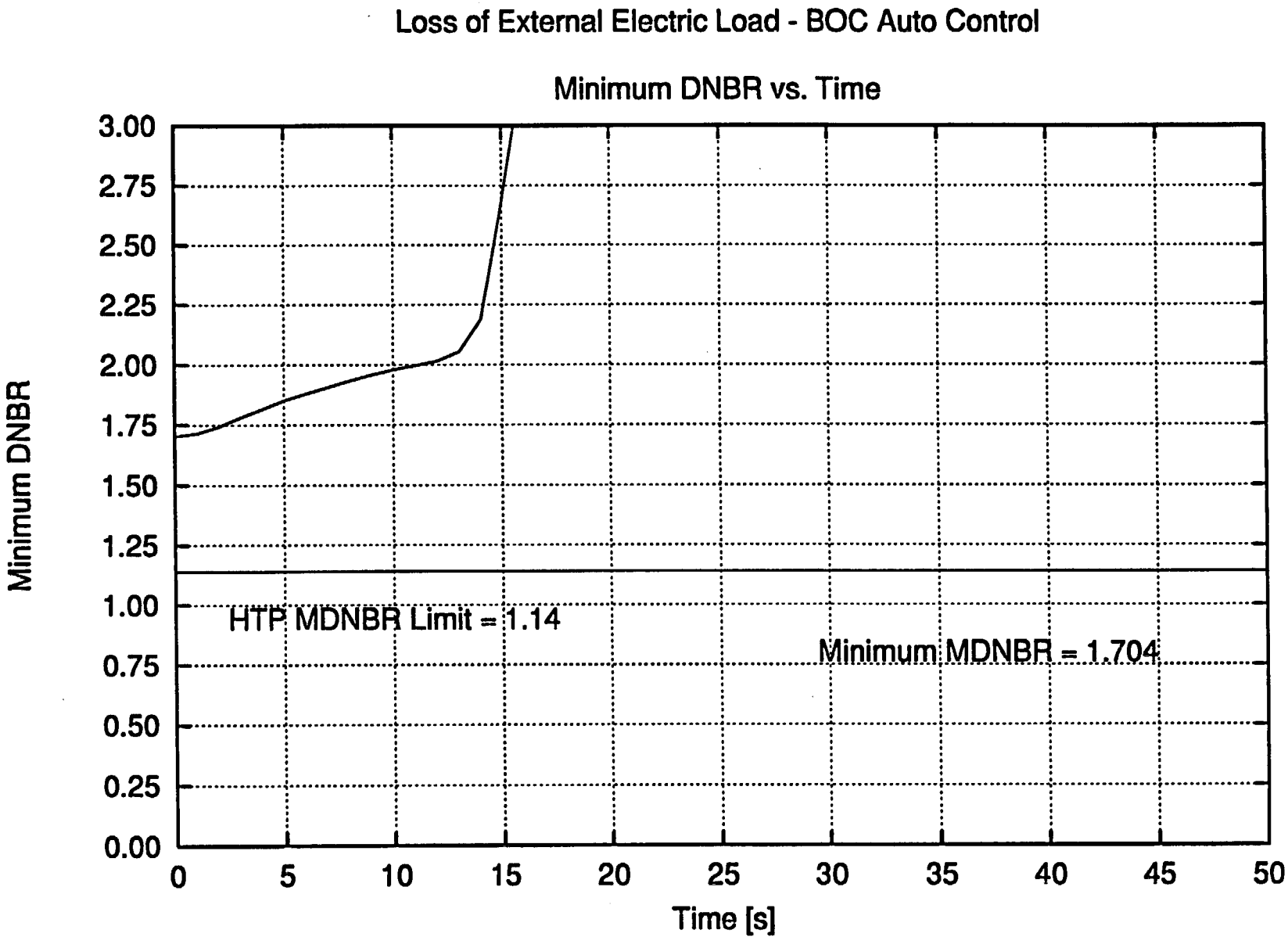
Loss of External Electric Load - BOC Auto Control
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.704
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-5

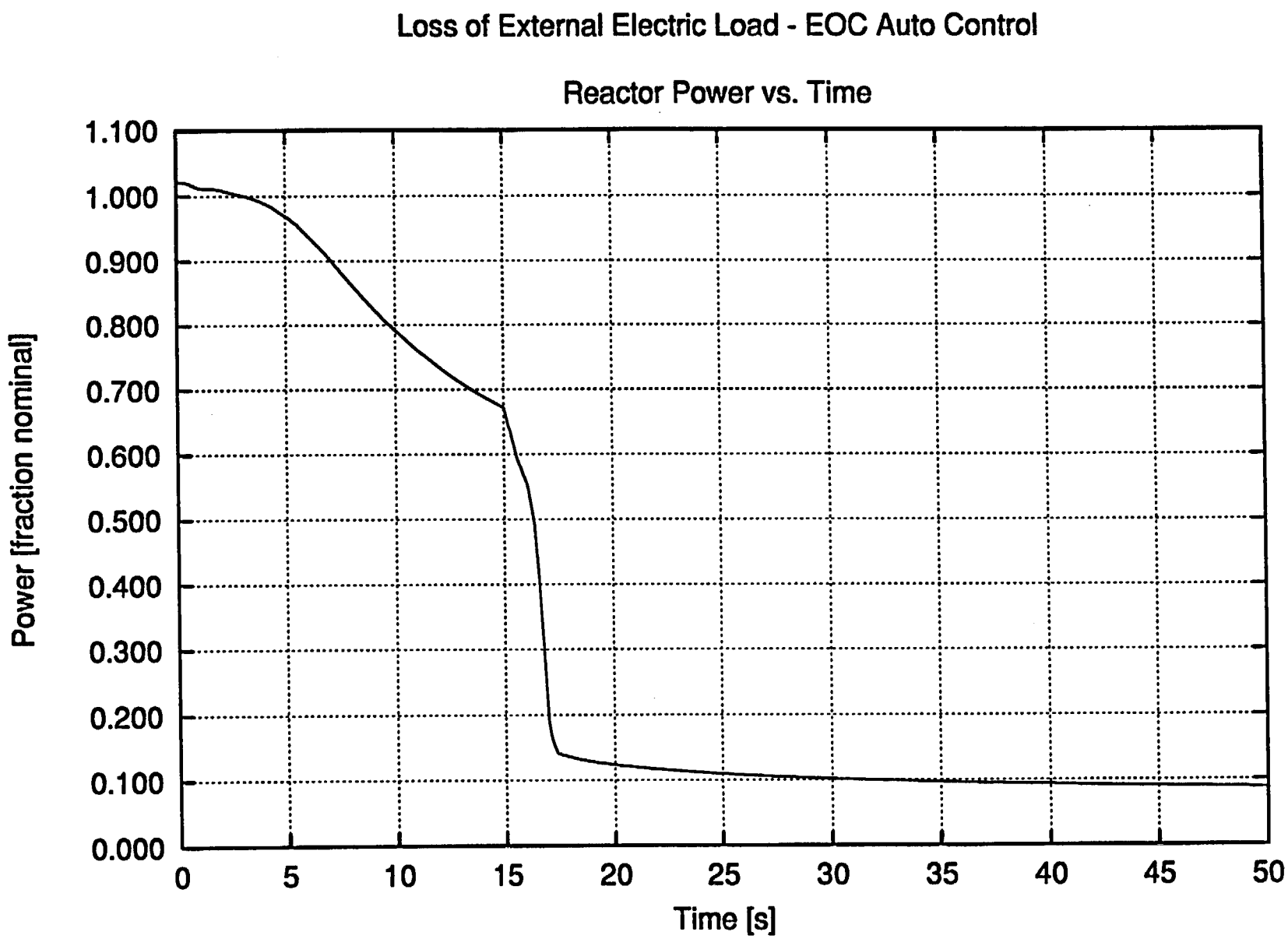
Loss of External Electric Load - EOC Auto Control
Reactor Power vs. Time
Power [fraction nominal]
1.100
1.000
0.900
0.800
0.700
0.600
0.500
0.400
0.300
0.200
0.100
0.000
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-6

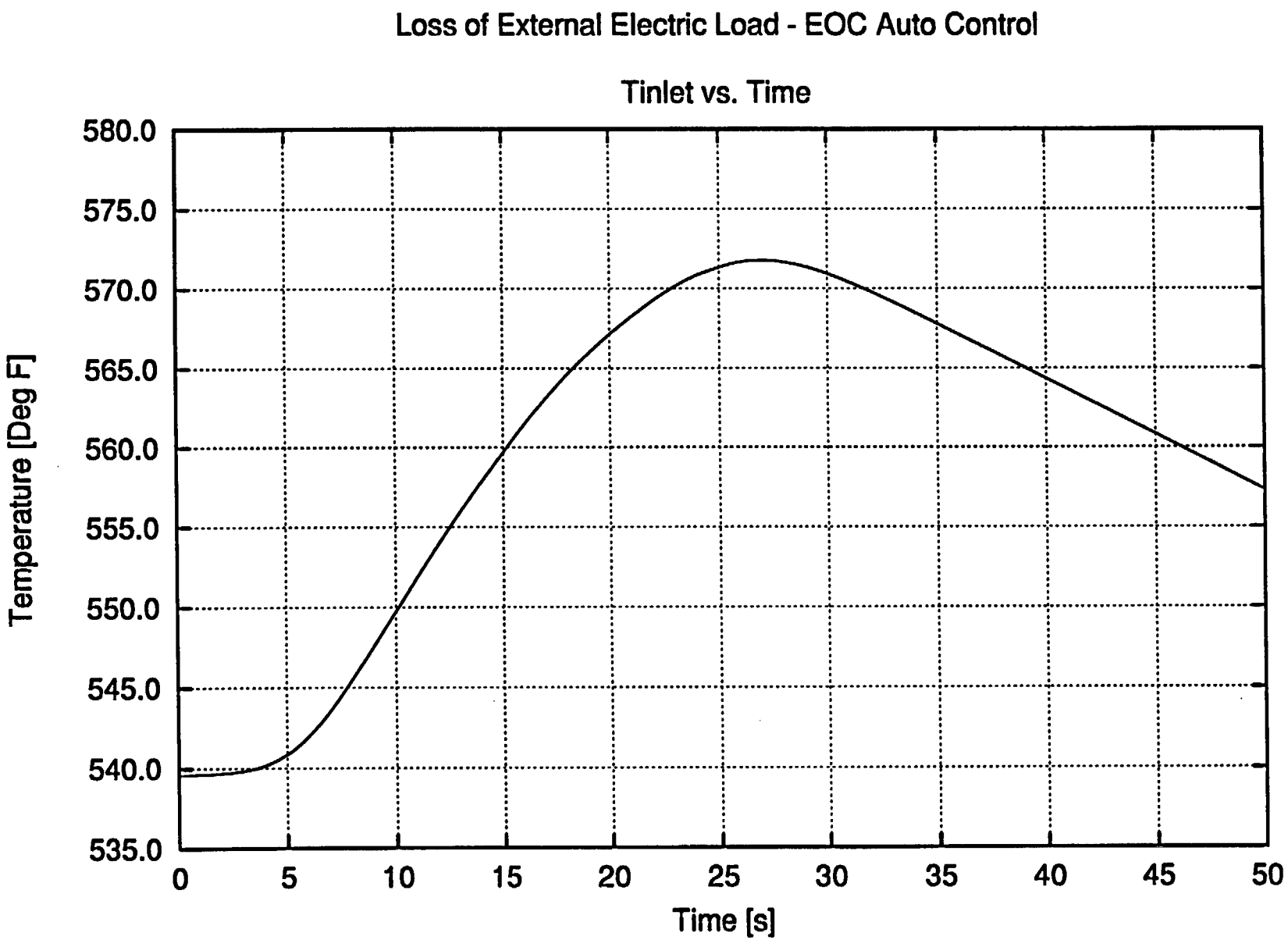
Loss of External Electric Load - EOC Auto Control
Tinlet vs. Time
580.0
575.0
570.0
565.0
560.0
555.0
550.0
545.0
540.0
535.0
Temperature [Deg F]
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-7

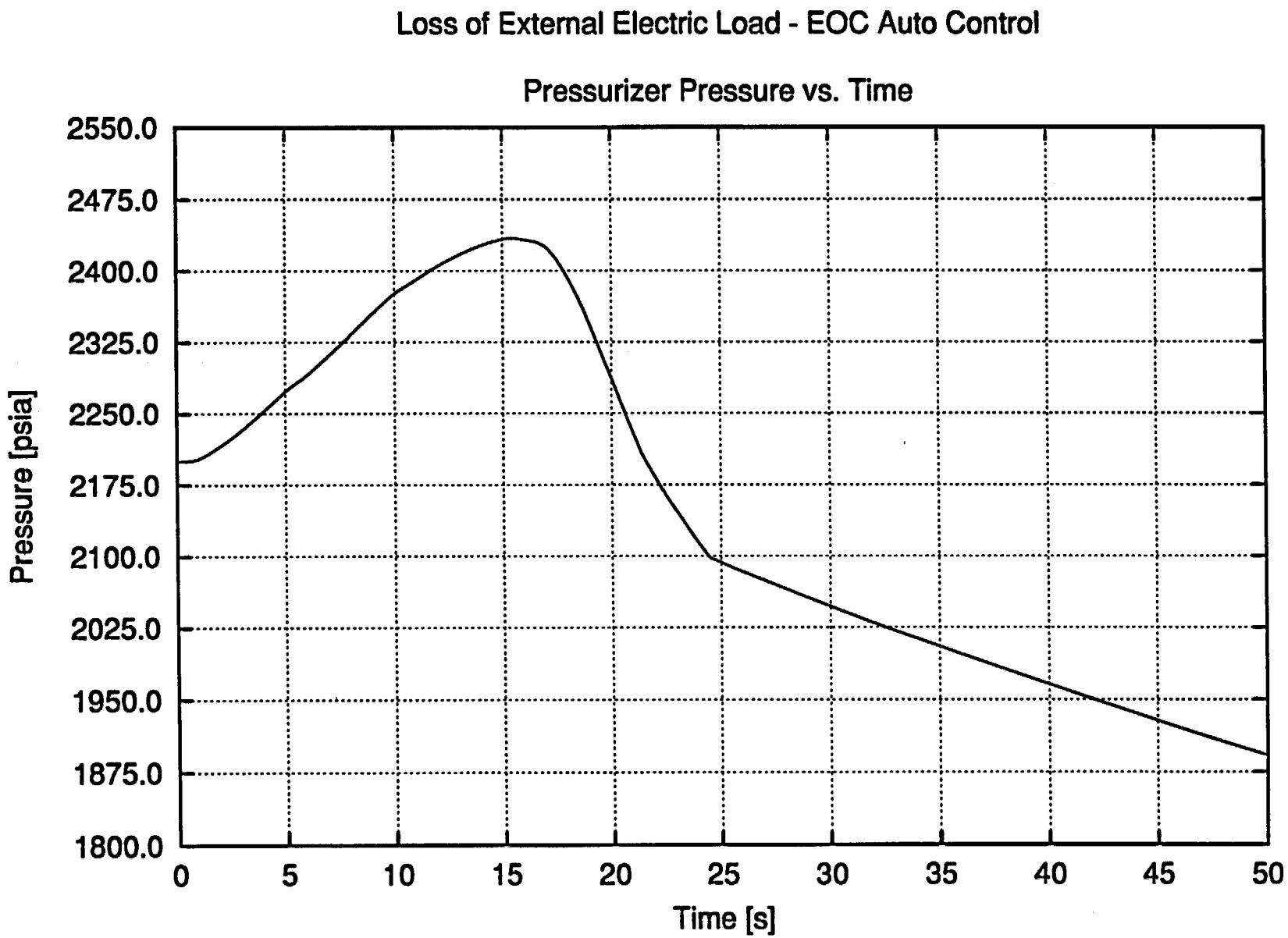
Loss of External Electric Load - EOC Auto Control
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-8

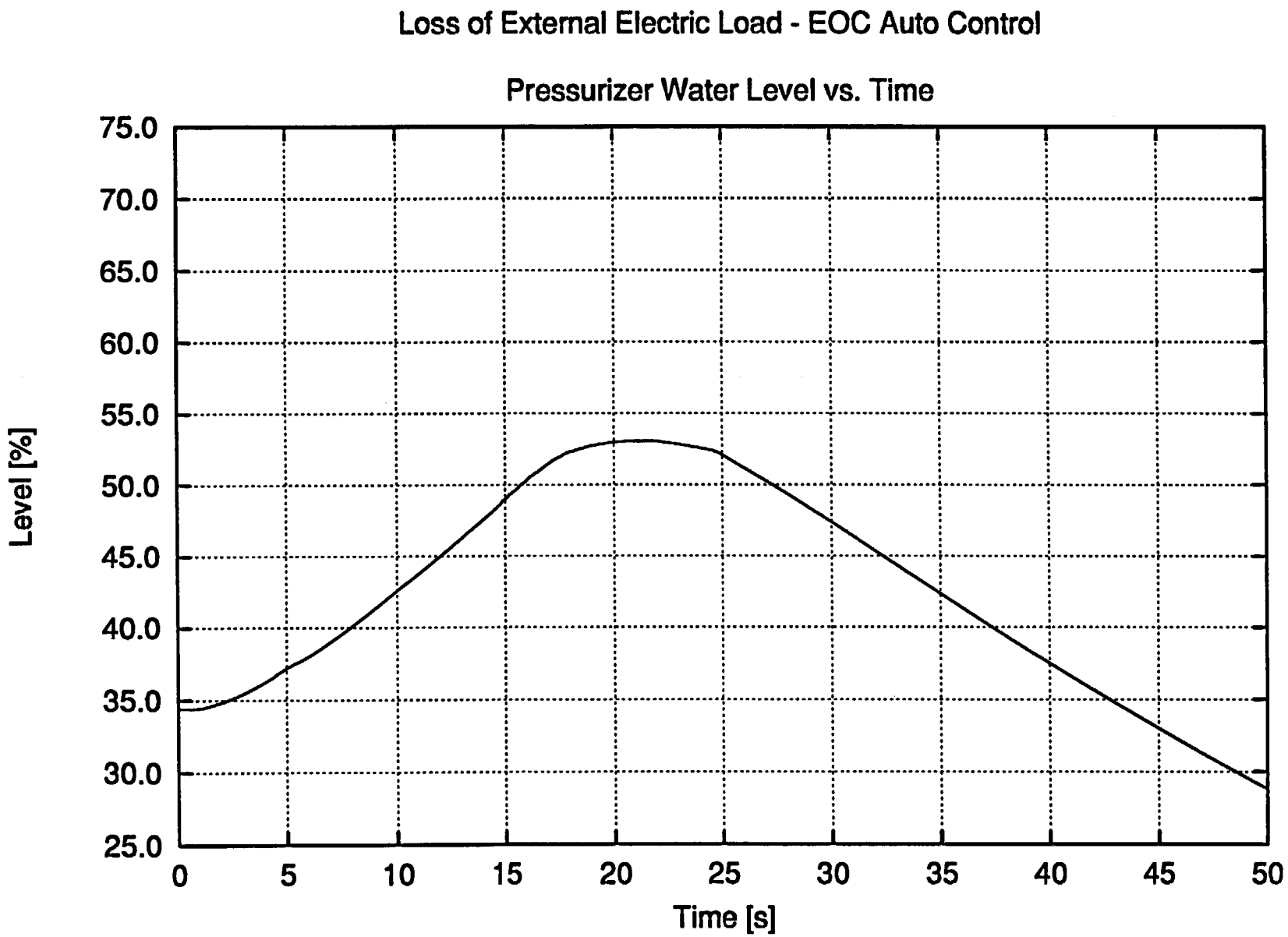
Loss of External Electric Load - EOC Auto Control
Pressurizer Water Level vs. Time
Level [%]
75.0
70.0
65.0
60.0
55.0
50.0
45.0
40.0
35.0
30.0
25.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-9

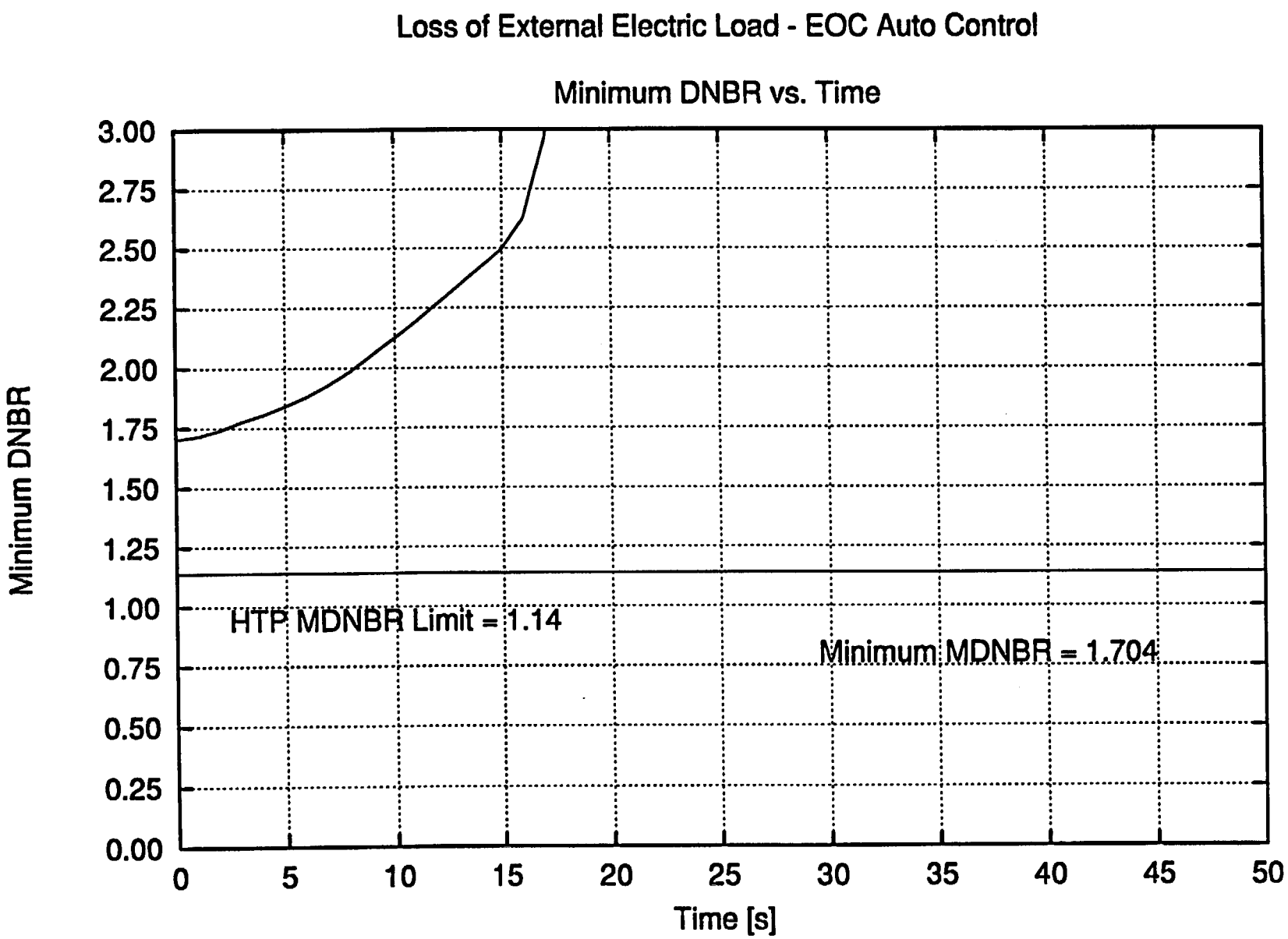
Loss of External Electric Load - EOC Auto Control
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.704
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-10

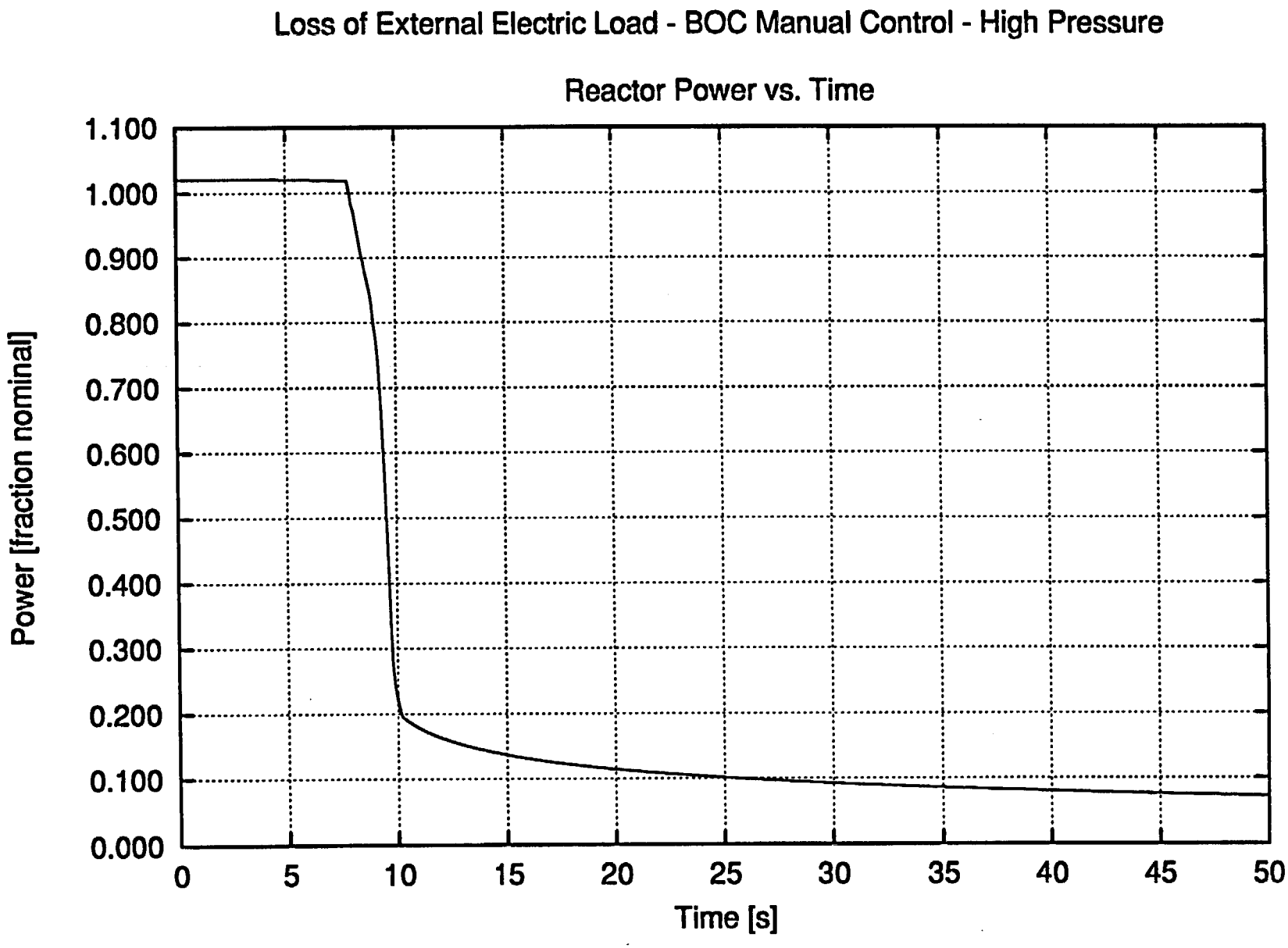
Loss of External Electric Load - BOC Manual Control - High Pressure
Reactor Power vs. Time
Power [fraction nominal]
1.100
1.000
0.900
0.800
0.700
0.600
0.500
0.400
0.300
0.200
0.100
0.000
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-11

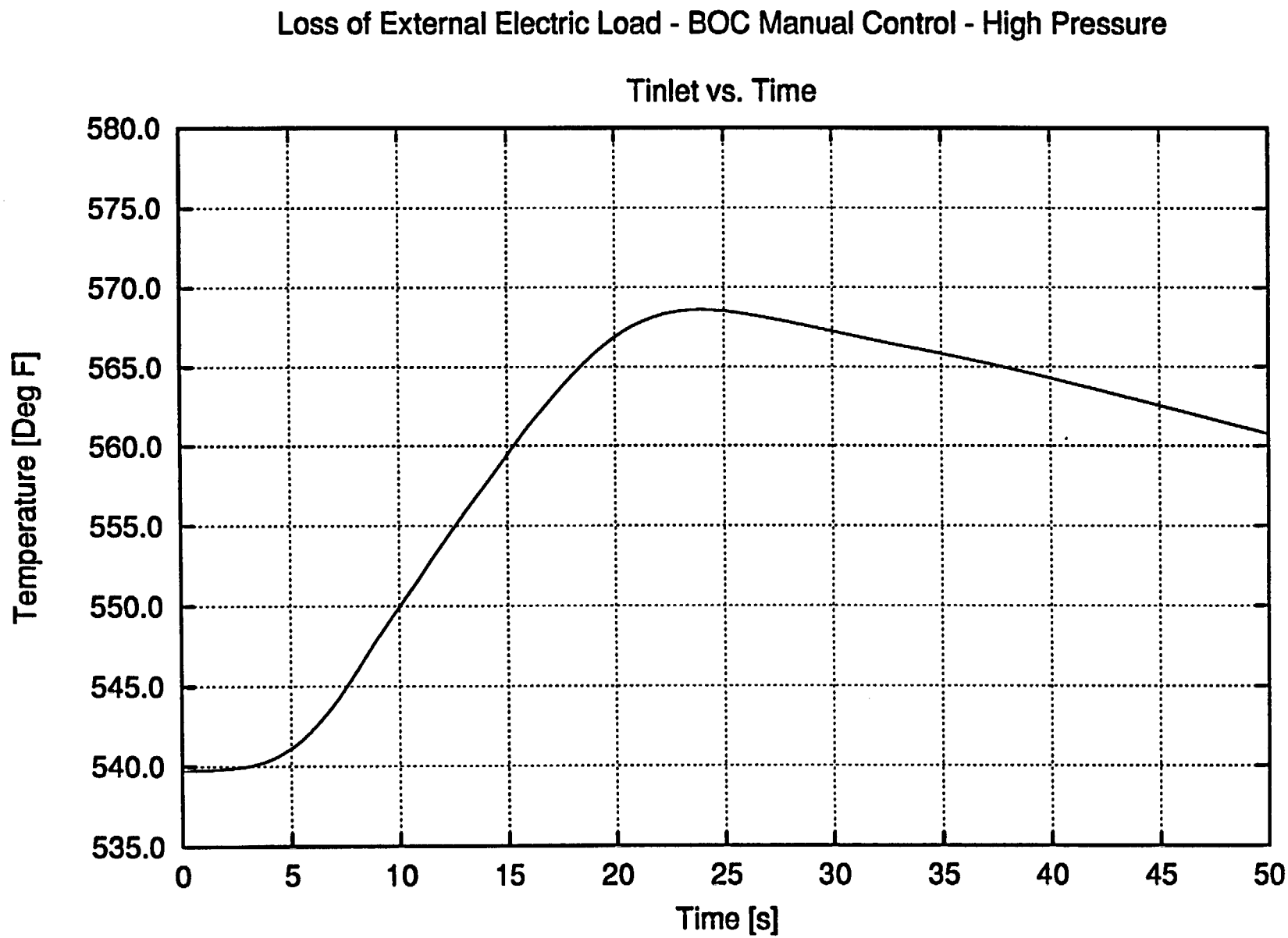
Loss of External Electric Load - BOC Manual Control - High Pressure
Tinlet vs. Time
Temperature [Deg F]
580.0
575.0
570.0
565.0
560.0
555.0
550.0
545.0
540.0
535.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-12

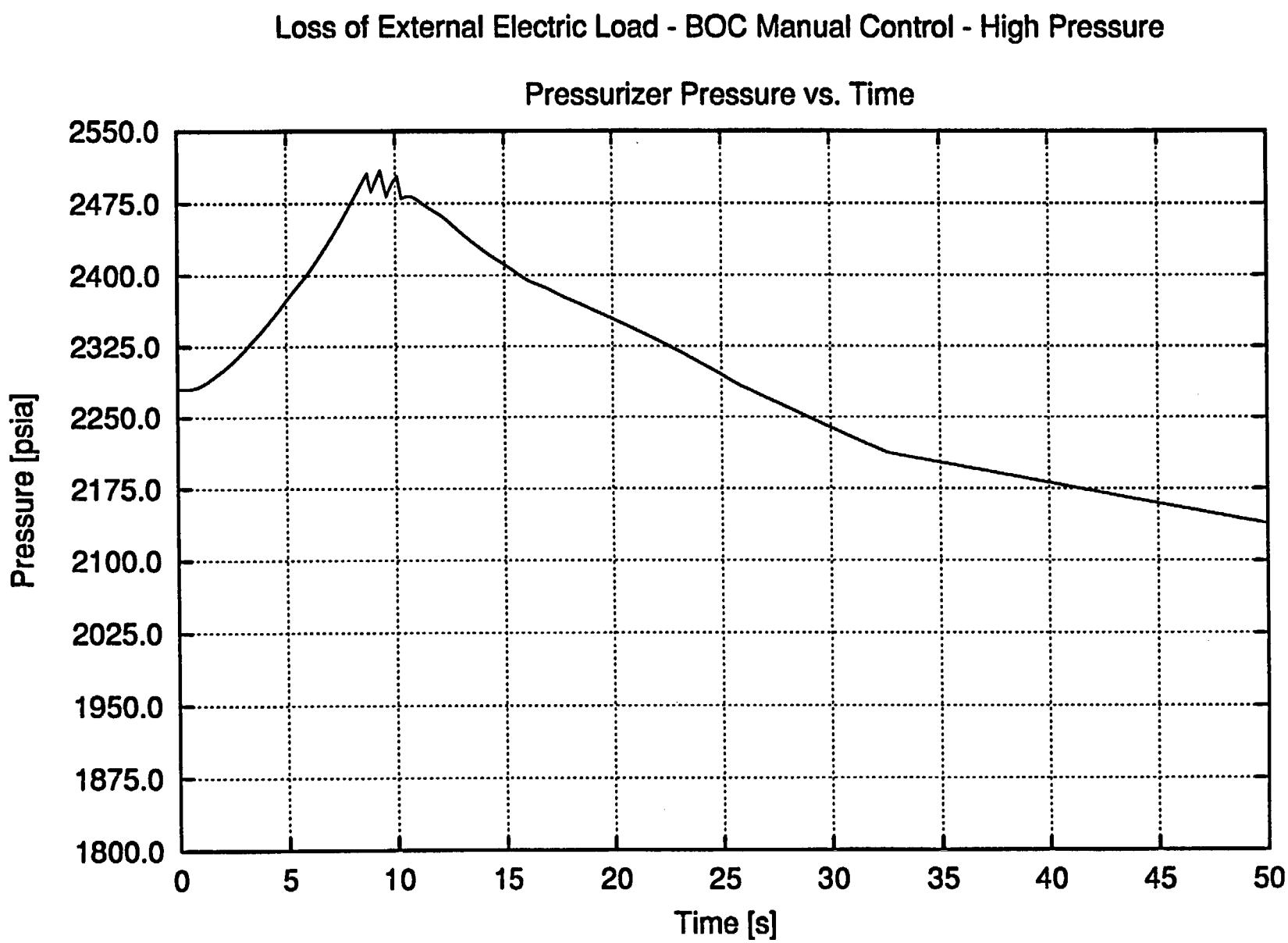
Loss of External Electric Load - BOC Manual Control - High Pressure
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-13

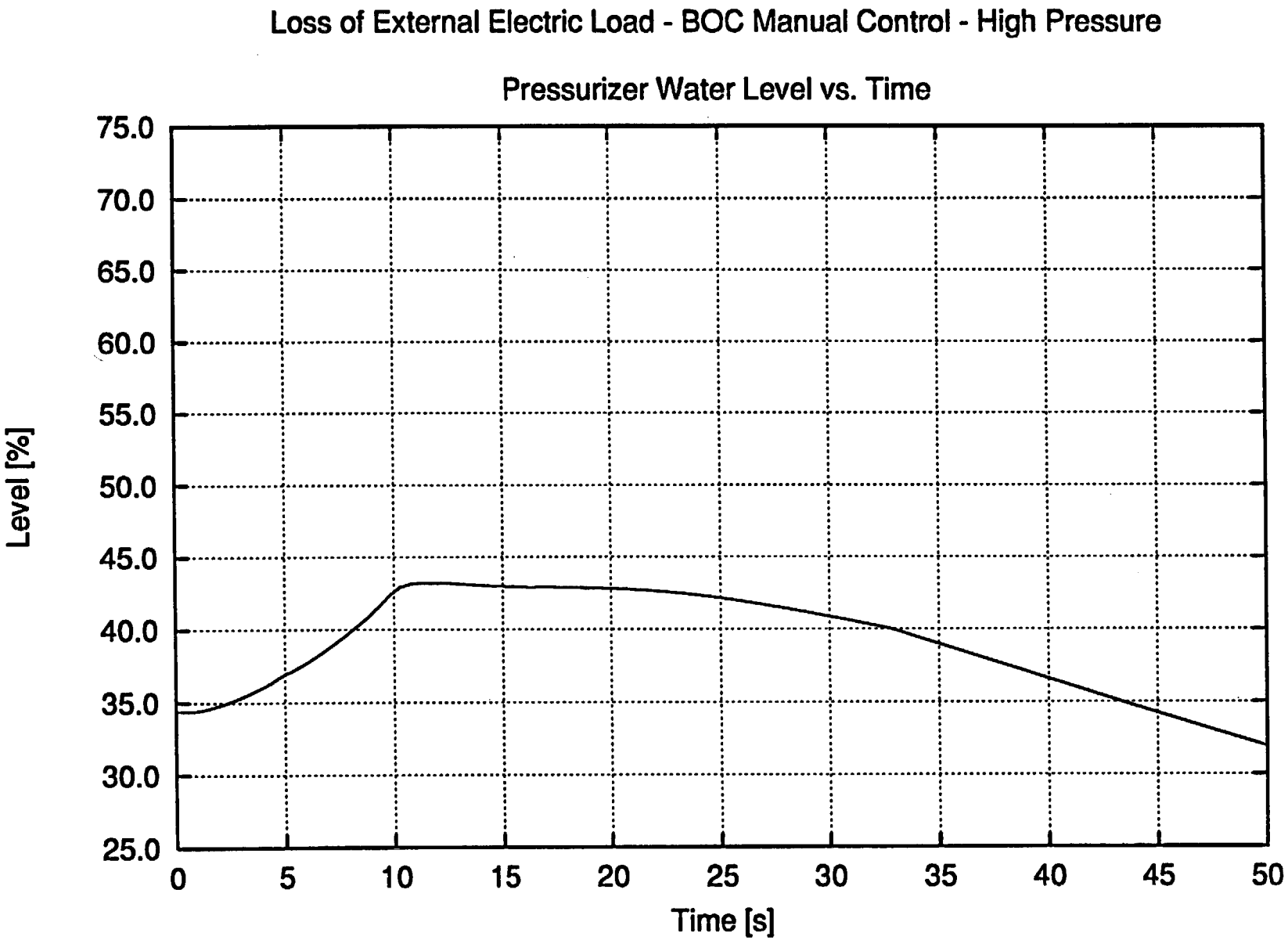
Loss of External Electric Load - BOC Manual Control - High Pressure
Pressurizer Water Level vs. Time
Level [%]
75.0
70.0
65.0
60.0
55.0
50.0
45.0
40.0
35.0
30.0
25.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-14

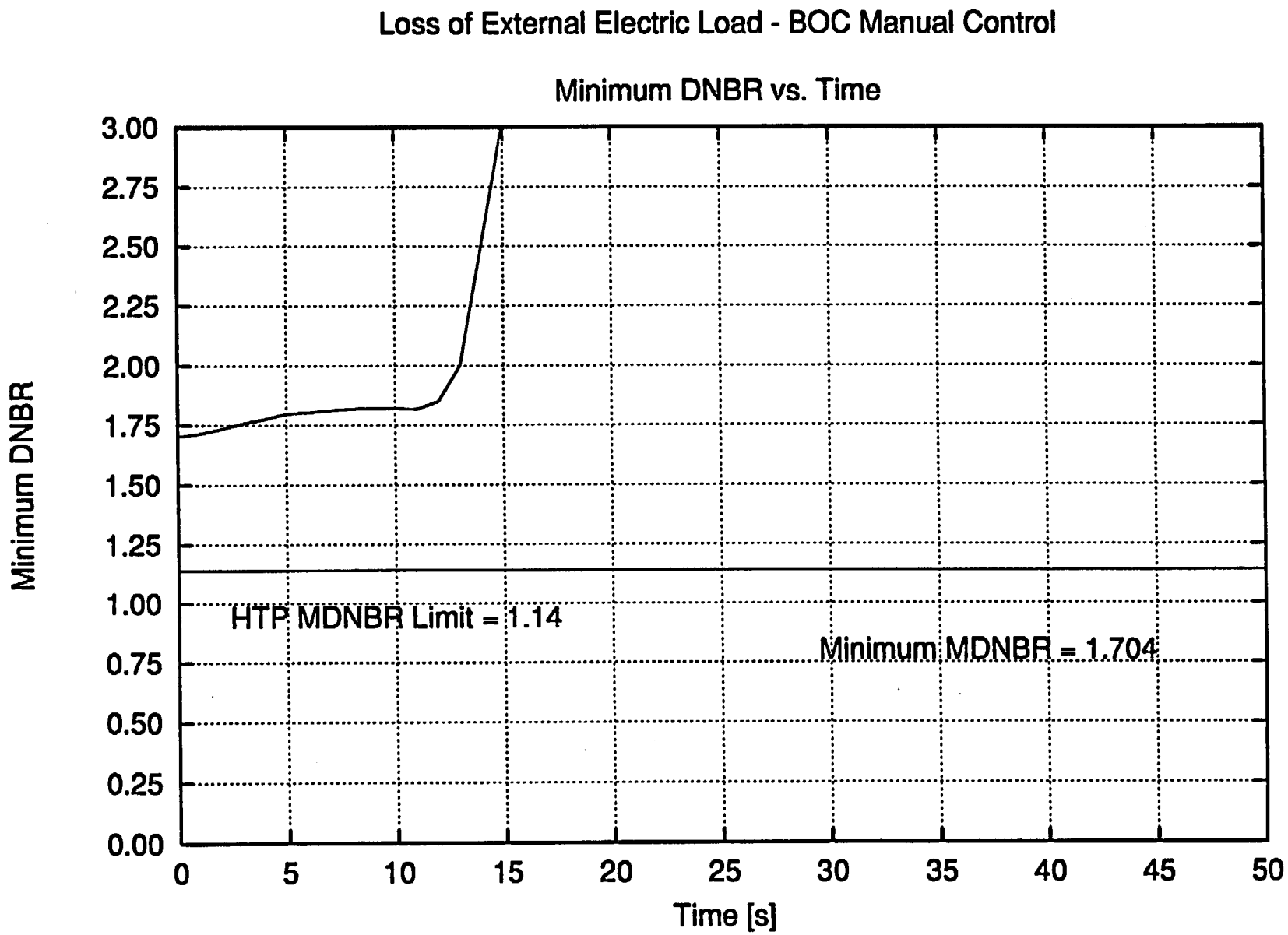
Loss of External Electric Load - BOC Manual Control
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.704
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-15

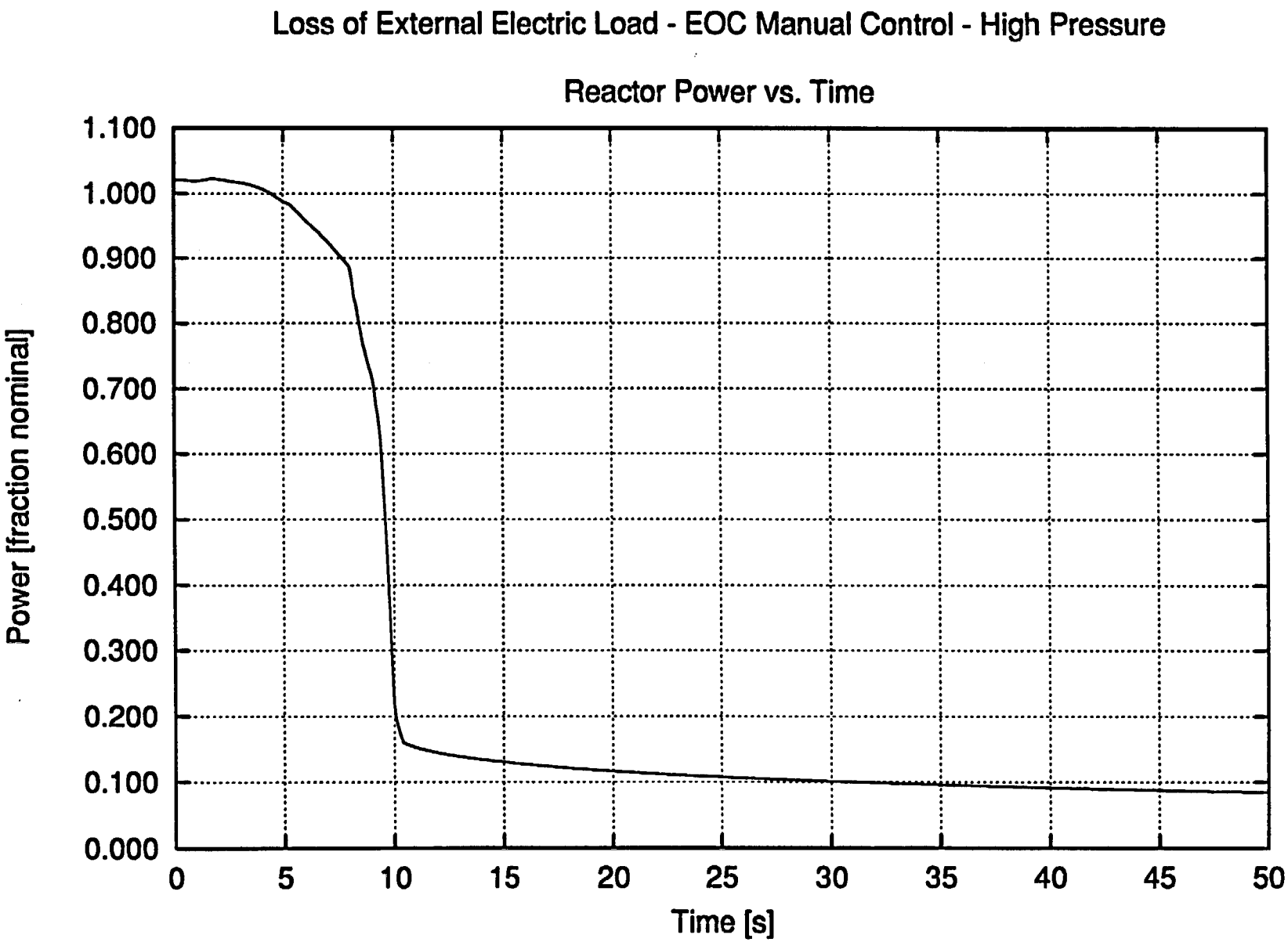
Loss of External Electric Load - EOC Manual Control - High Pressure
Reactor Power vs. Time
Power [fraction nominal]
1.100
1.000
0.900
0.800
0.700
0.600
0.500
0.400
0.300
0.200
0.100
0.000
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-16

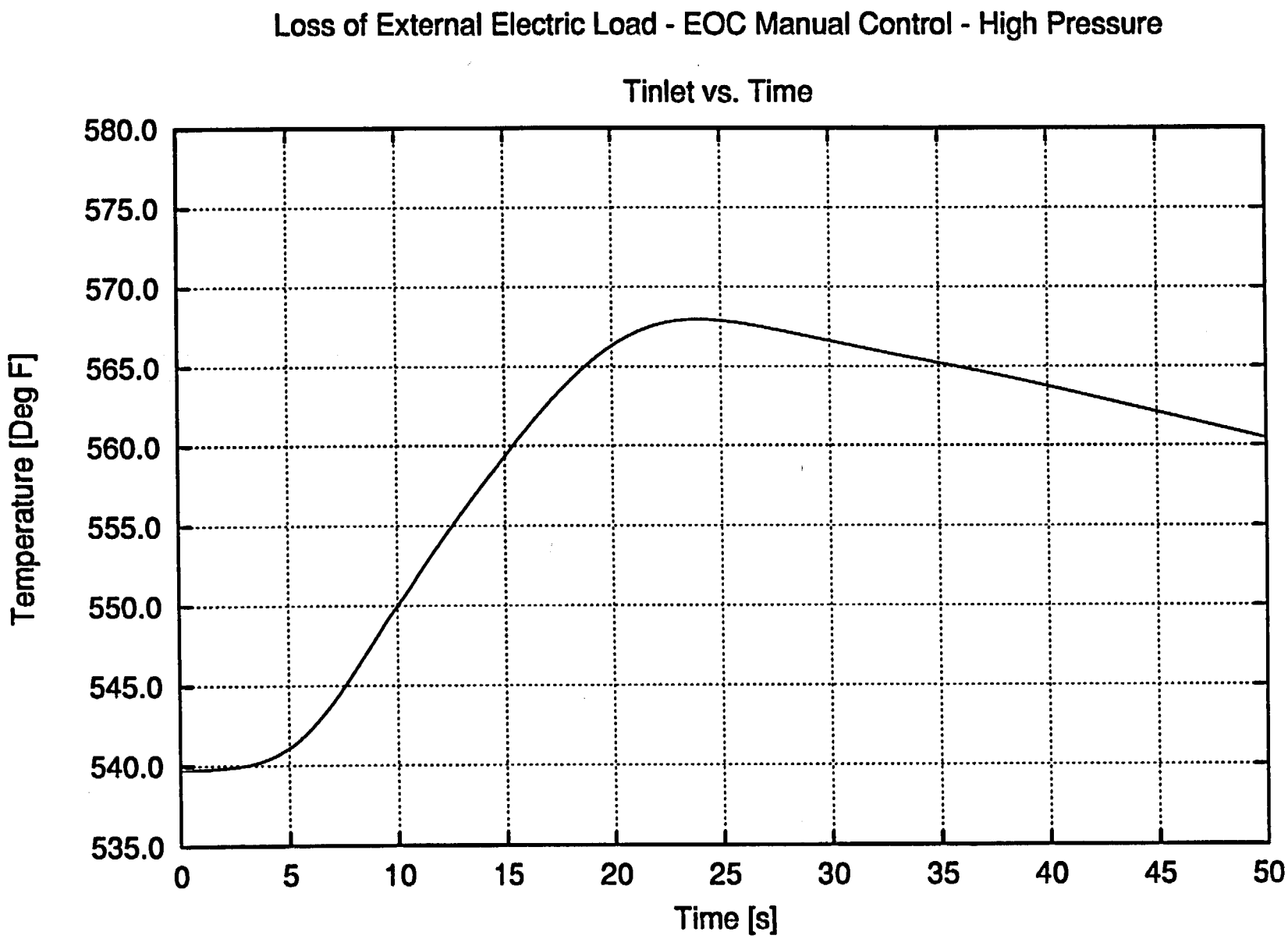
Loss of External Electric Load - EOC Manual Control - High Pressure
Tinlet vs. Time
Temperature [Deg F]
580.0
575.0
570.0
565.0
560.0
555.0
550.0
545.0
540.0
535.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-17

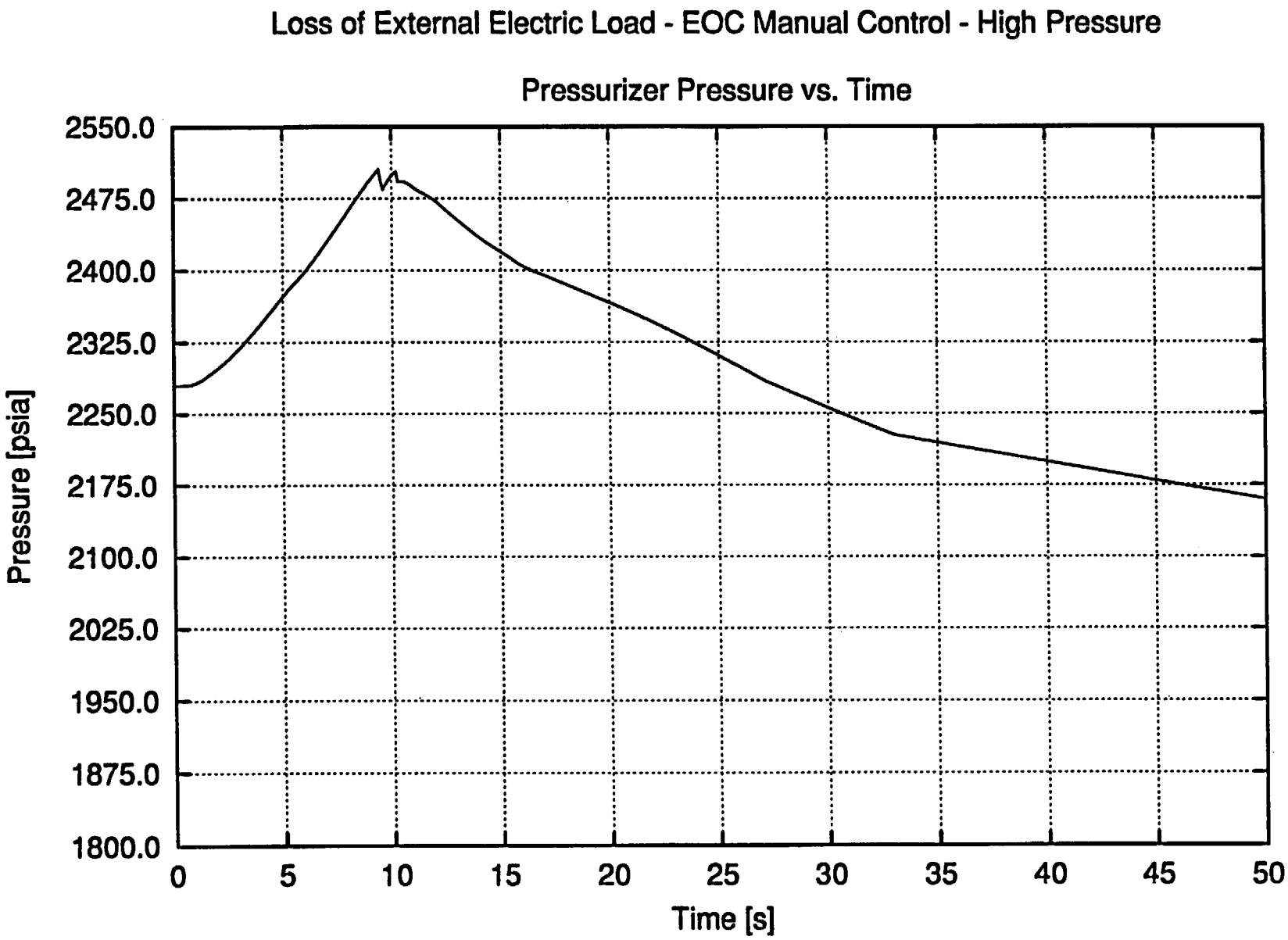
Loss of External Electric Load - EOC Manual Control - High Pressure
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-18

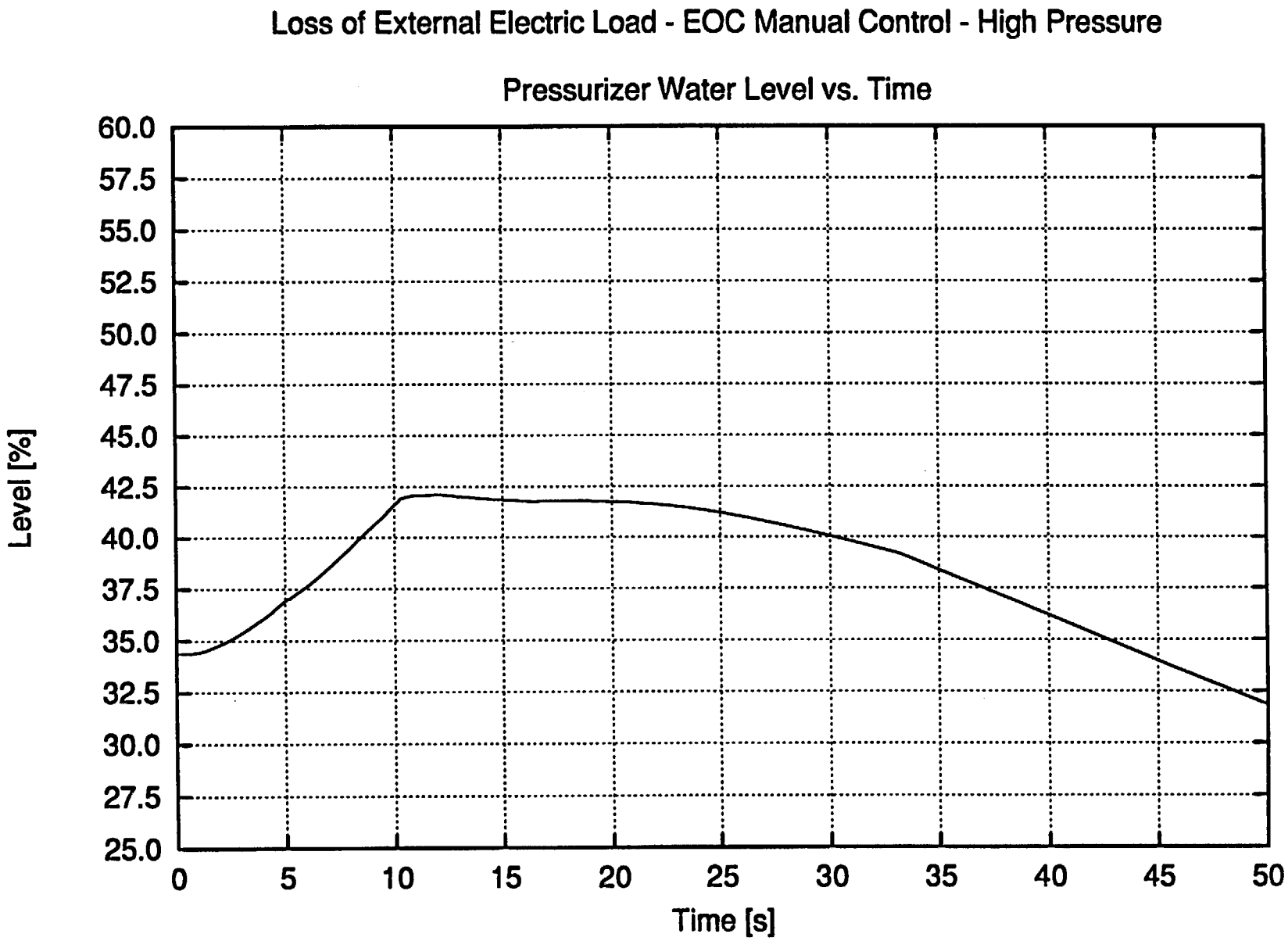
Loss of External Electric Load - EOC Manual Control - High Pressure
Pressurizer Water Level vs. Time
Level [%]
60.0
57.5
55.0
52.5
50.0
47.5
45.0
42.5
40.0
37.5
35.0
32.5
30.0
27.5
25.0
0
5
10
15
20
25
30
35
40
45
50
Time [s]

Figure 3.9-19

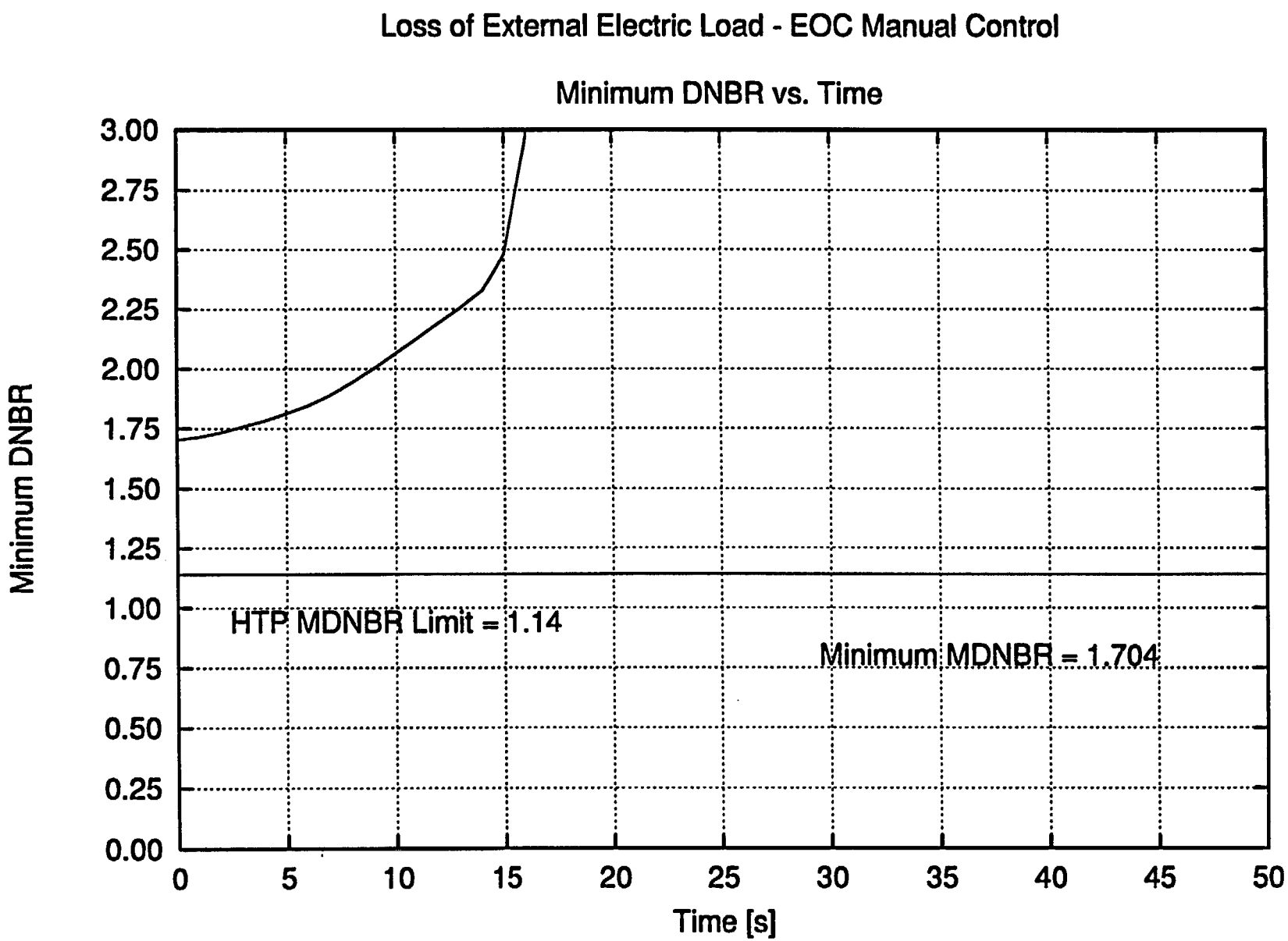
Loss of External Electric Load - EOC Manual Control
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0
5
10
15
20
25
30
35
40
45
50
Time [s]
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.704

Figure 3.9-20

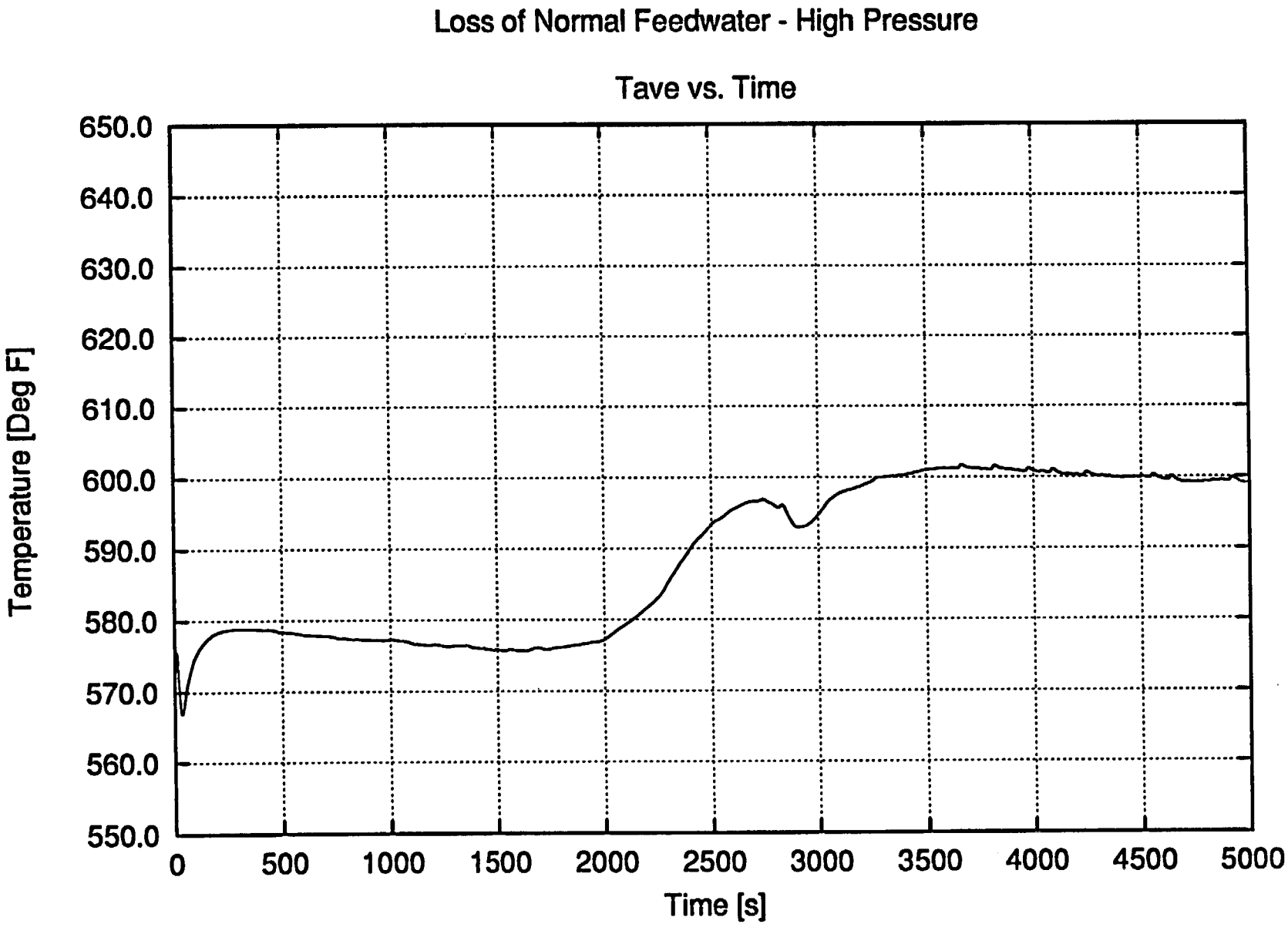
Loss of Normal Feedwater - High Pressure
Tave vs. Time
Temperature [Deg F]
650.0
640.0
630.0
620.0
610.0
600.0
590.0
580.0
570.0
560.0
550.0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
Time [s]

Figure 3.10-1

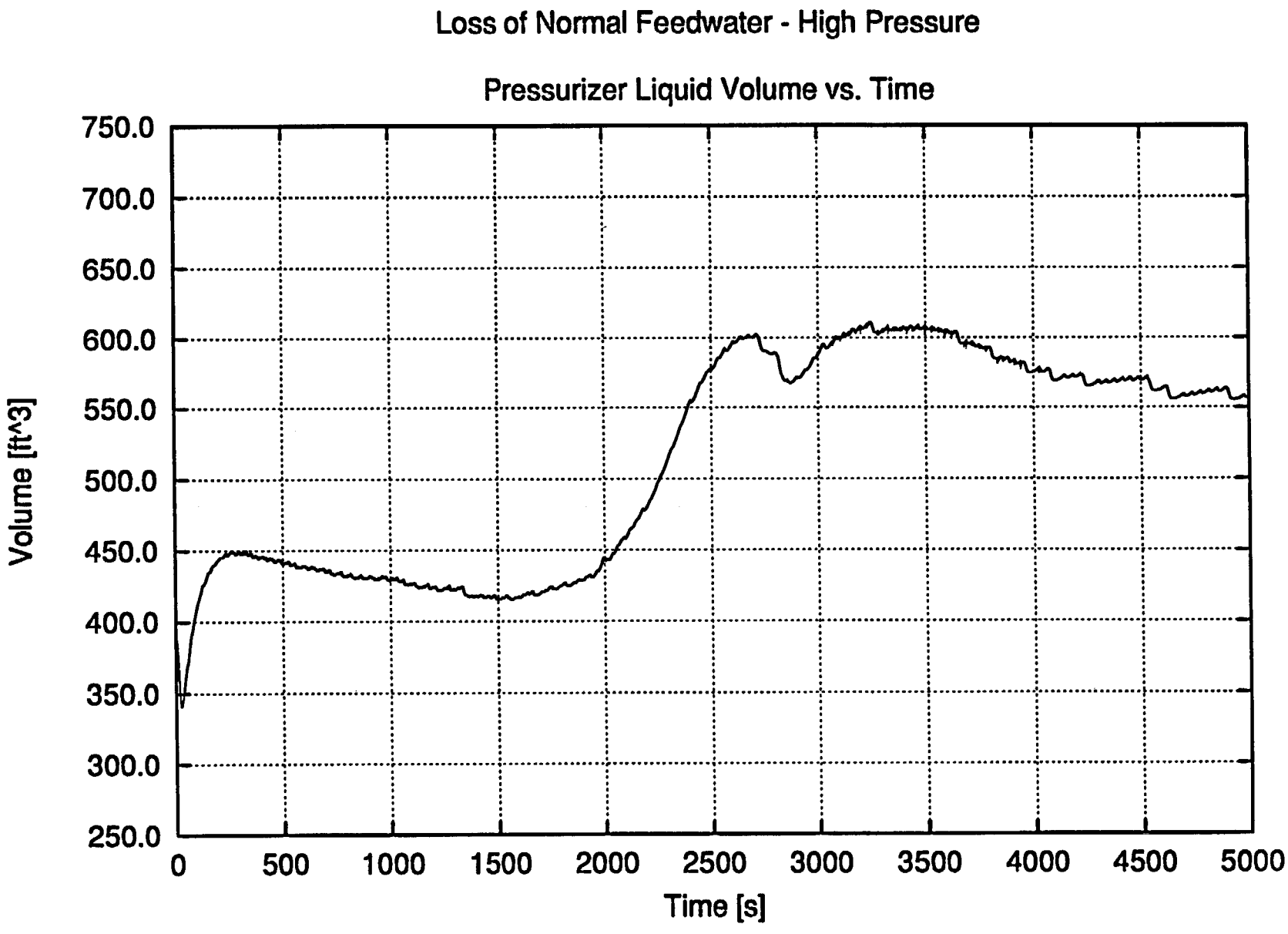
Loss of Normal Feedwater - High Pressure
Pressurizer Liquid Volume vs. Time
Volume [ft^3]
750.0
700.0
650.0
600.0
550.0
500.0
450.0
400.0
350.0
300.0
250.0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
Time [s]

Figure 3.10-2

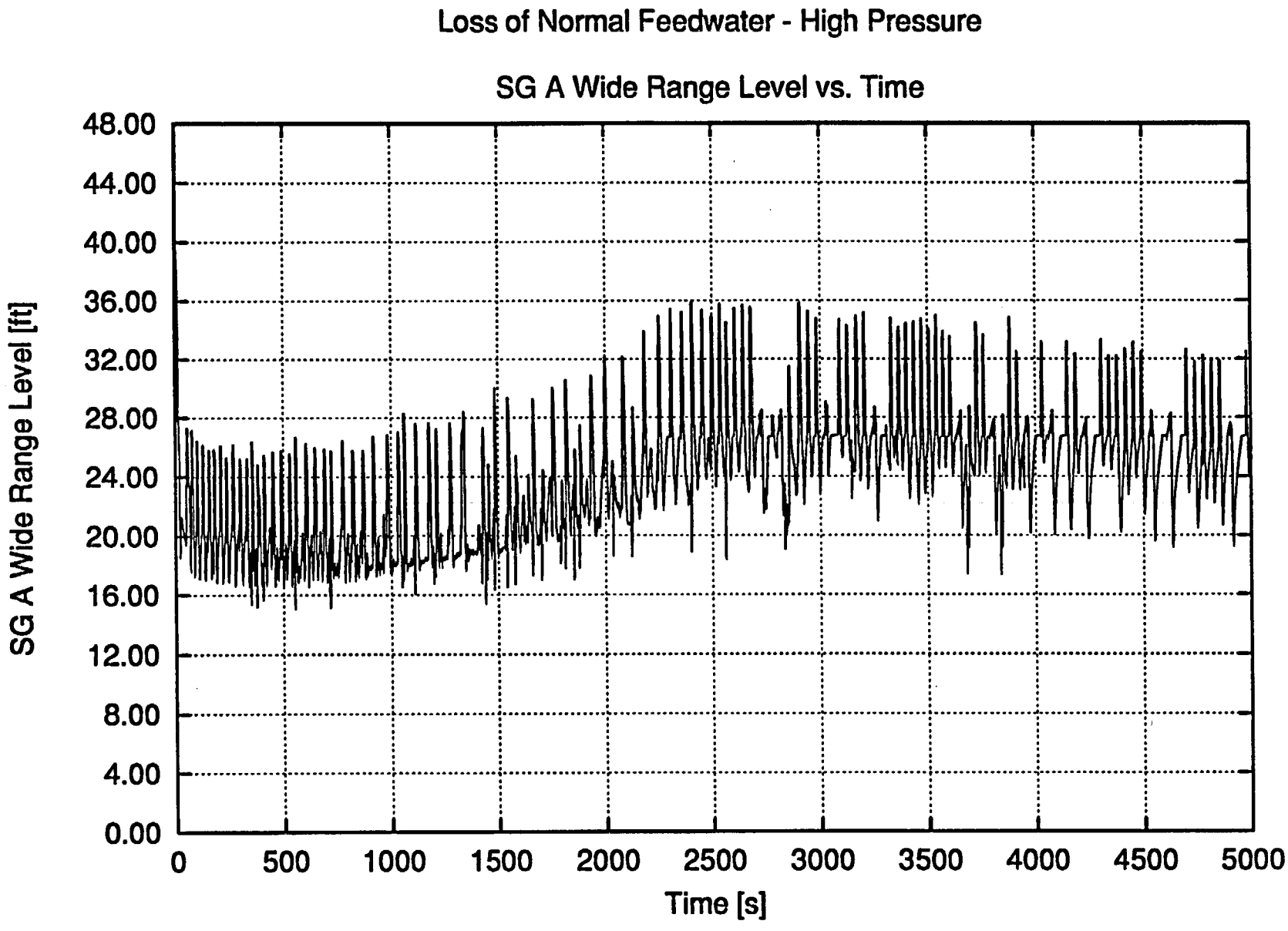
Loss of Normal Feedwater - High Pressure
SG A Wide Range Level vs. Time
SG A Wide Range Level [ft]
48.00
44.00
40.00
36.00
32.00
28.00
24.00
20.00
16.00
12.00
8.00
4.00
0.00
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
Time [s]

Figure 3.10-3

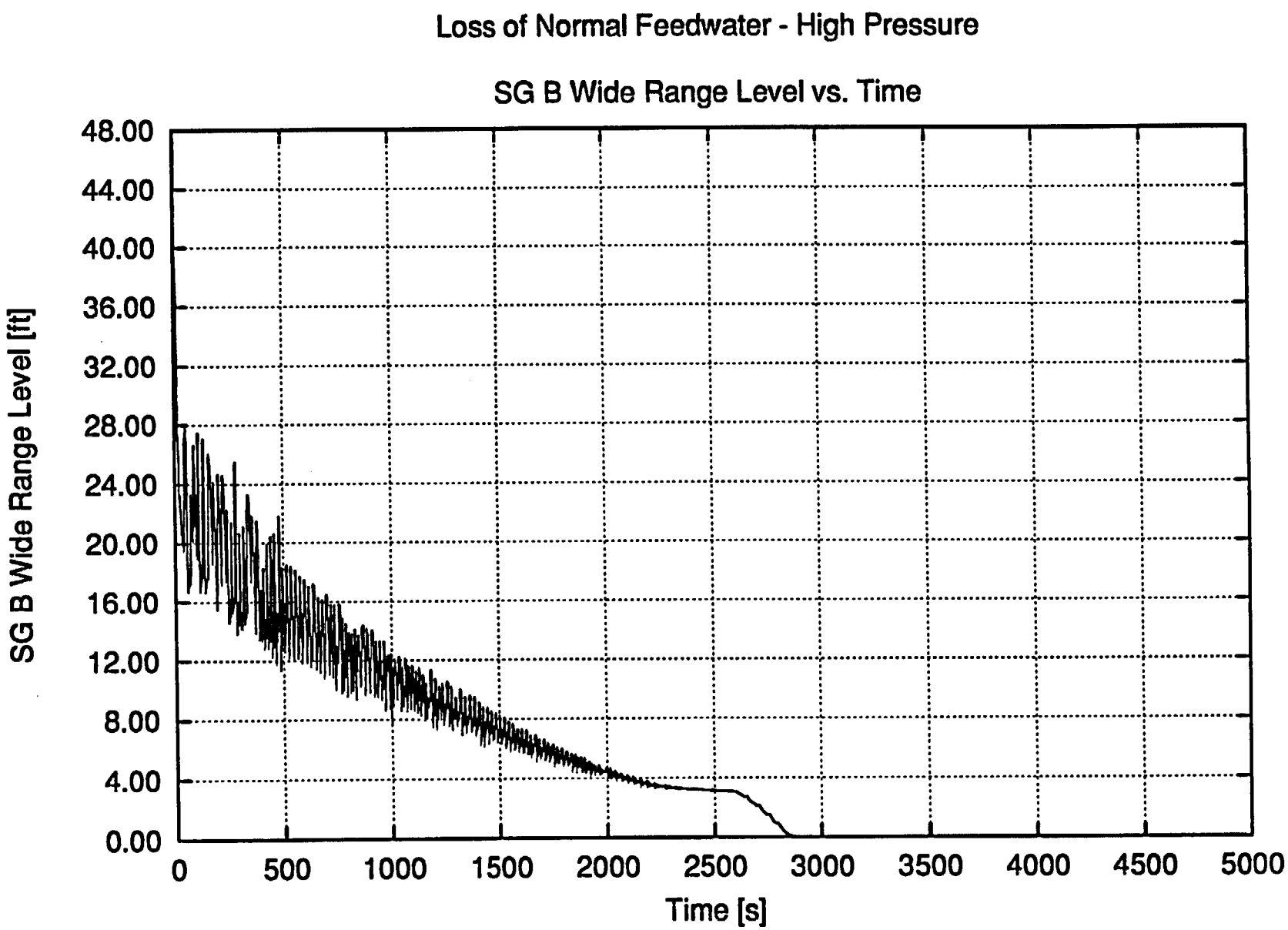
Loss of Normal Feedwater - High Pressure
SG B Wide Range Level vs. Time
SG B Wide Range Level [ft]
48.00
44.00
40.00
36.00
32.00
28.00
24.00
20.00
16.00
12.00
8.00
4.00
0.00
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
Time [s]

Figure 3.10-4

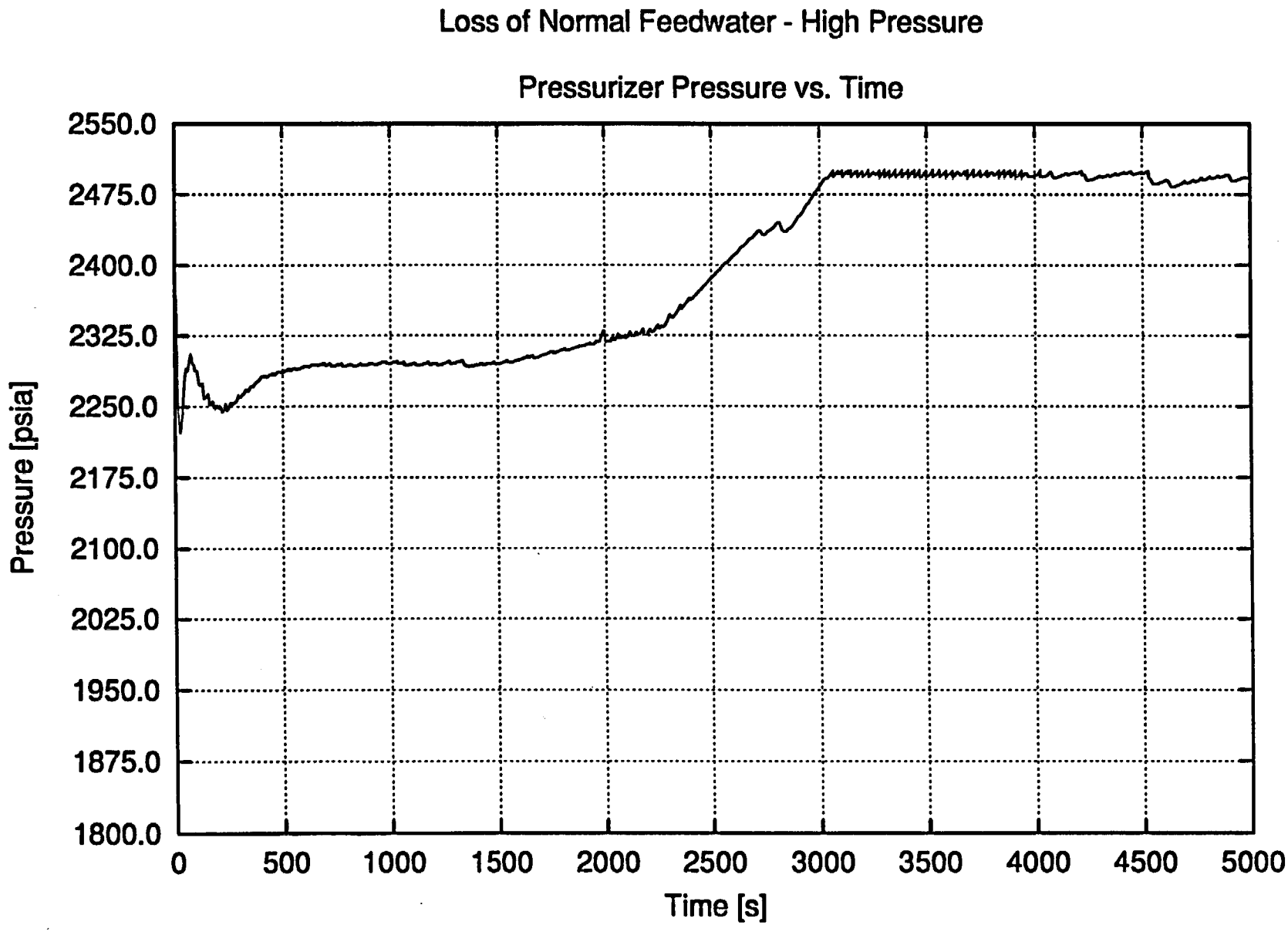
Loss of Normal Feedwater - High Pressure
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
Time [s]

Figure 3.10-5

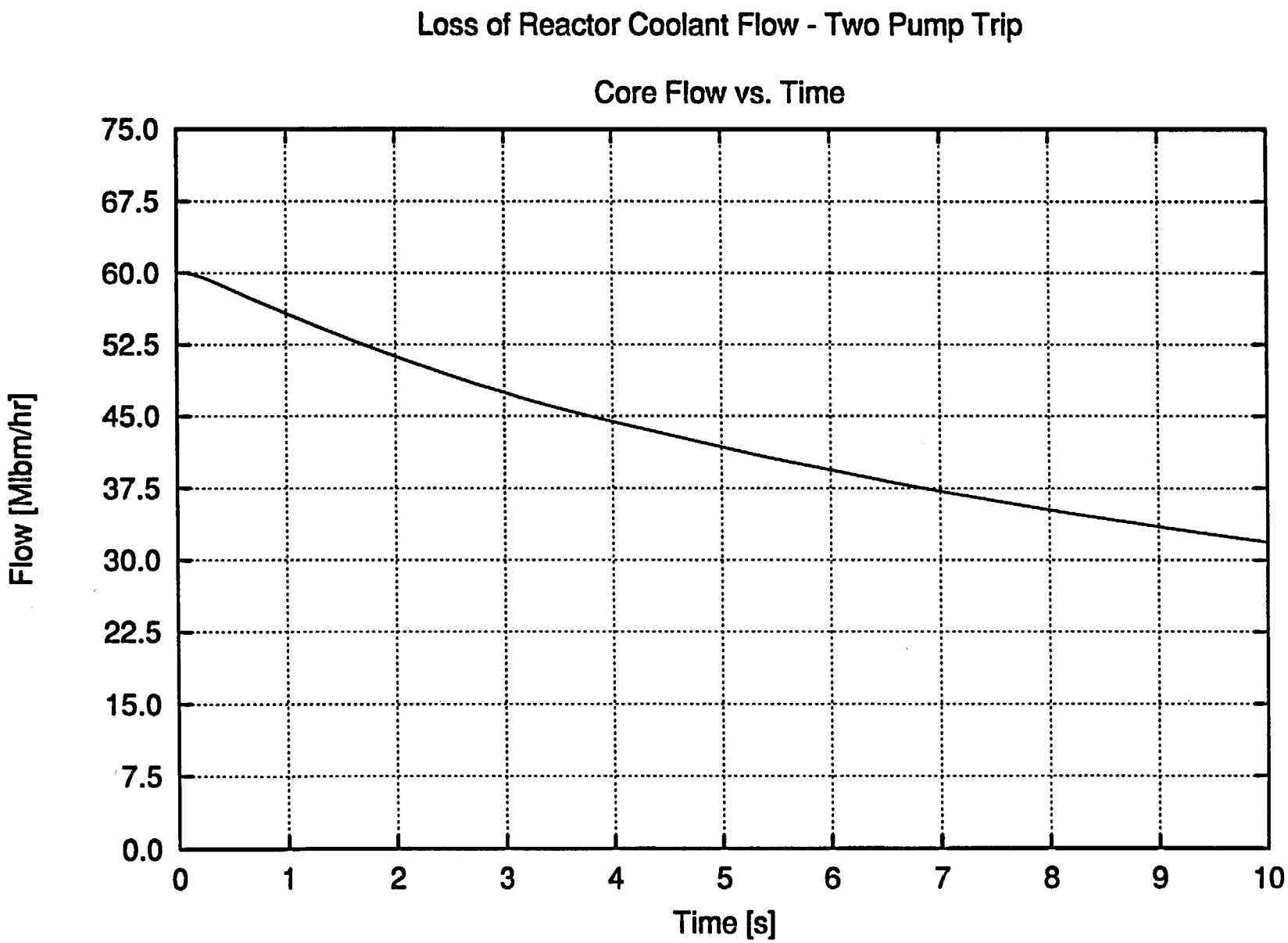
Loss of Reactor Coolant Flow - Two Pump Trip
Core Flow vs. Time
Flow [Mlbm/hr]
75.0
67.5
60.0
52.5
45.0
37.5
30.0
22.5
15.0
7.5
0.0
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.11-1

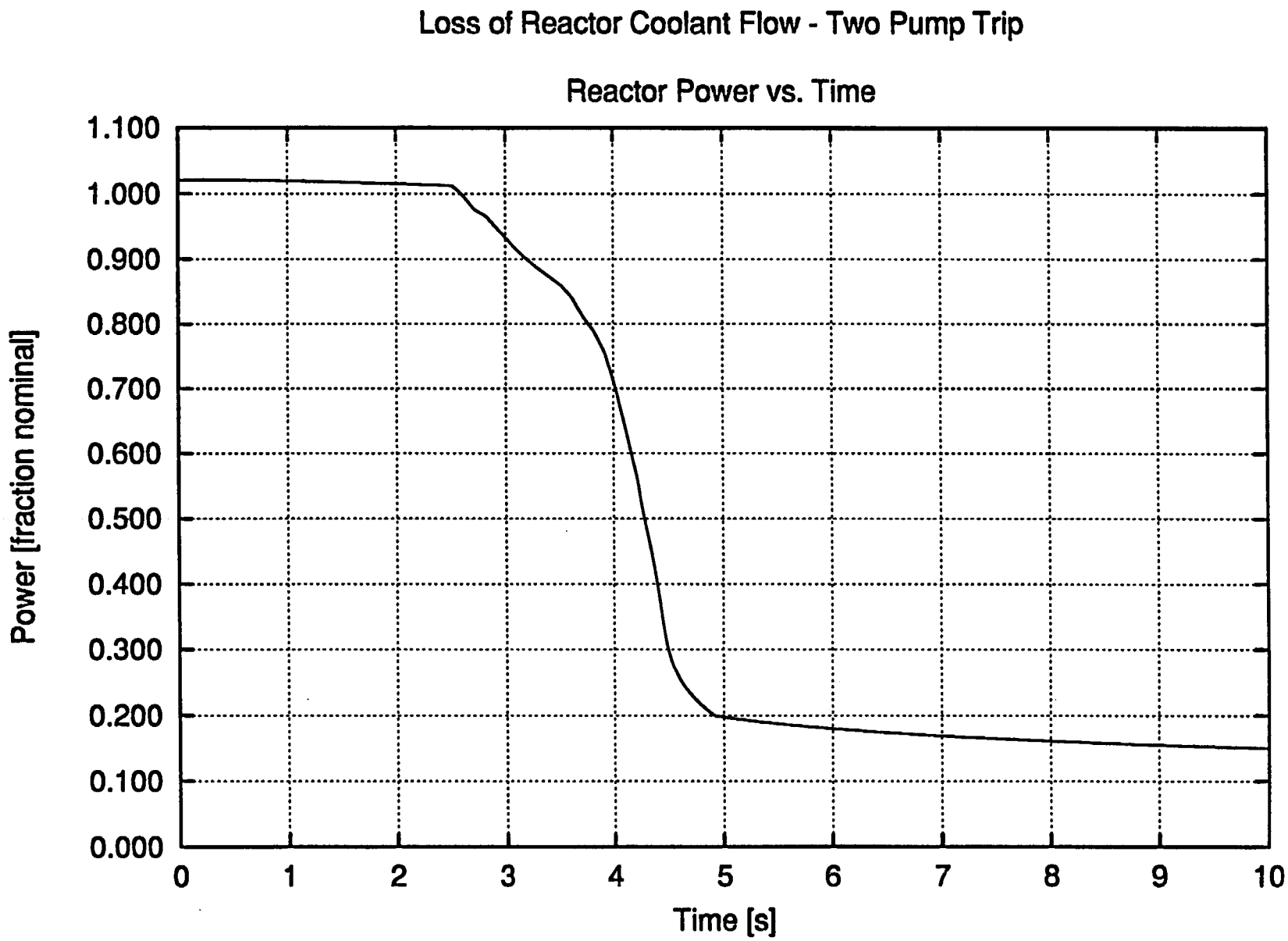
Loss of Reactor Coolant Flow - Two Pump Trip
Reactor Power vs. Time
Power [fraction nominal]
1.100
1.000
0.900
0.800
0.700
0.600
0.500
0.400
0.300
0.200
0.100
0.000
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.11-2

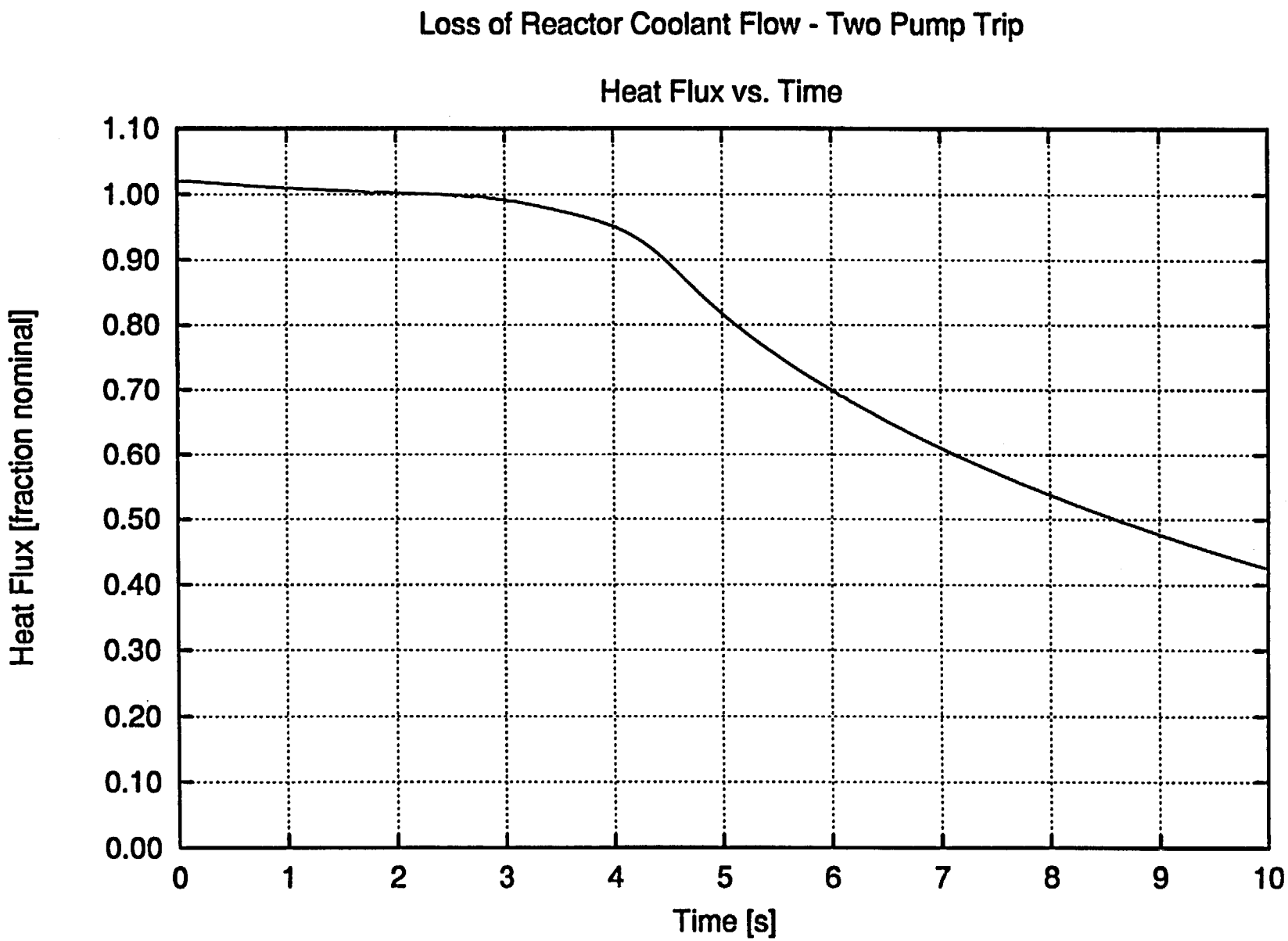
Loss of Reactor Coolant Flow - Two Pump Trip
Heat Flux vs. Time
Heat Flux [fraction nominal]
1.10
1.00
0.90
0.80
0.70
0.60
0.50
0.40
0.30
0.20
0.10
0.00
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.11-3

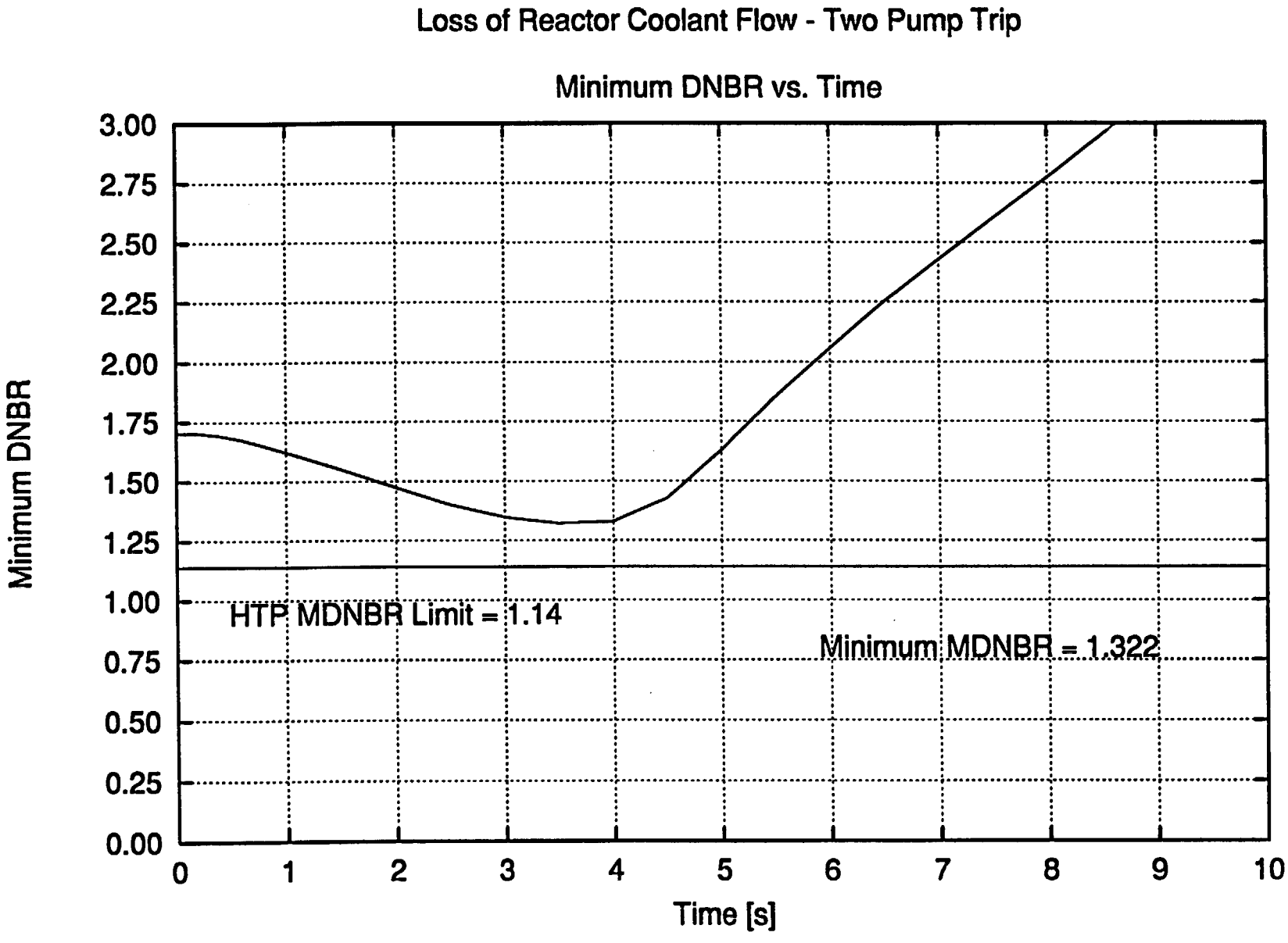
Loss of Reactor Coolant Flow - Two Pump Trip
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0
1
2
3
4
5
6
7
8
9
10
Time [s]
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.322

Figure 3.11-4

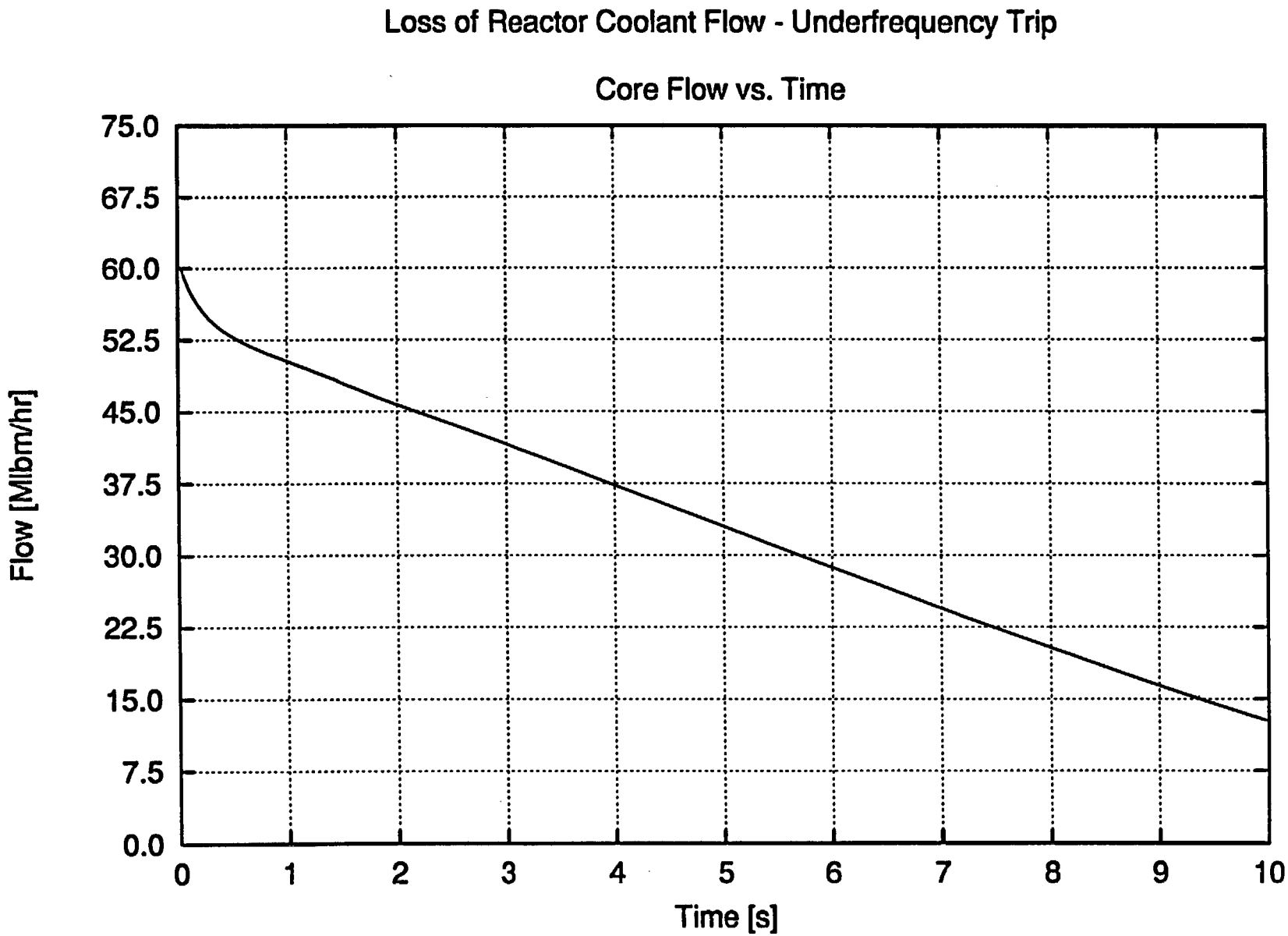
Loss of Reactor Coolant Flow - Underfrequency Trip
Core Flow vs. Time
Flow [Mlbm/hr]
75.0
67.5
60.0
52.5
45.0
37.5
30.0
22.5
15.0
7.5
0.0
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.11-5

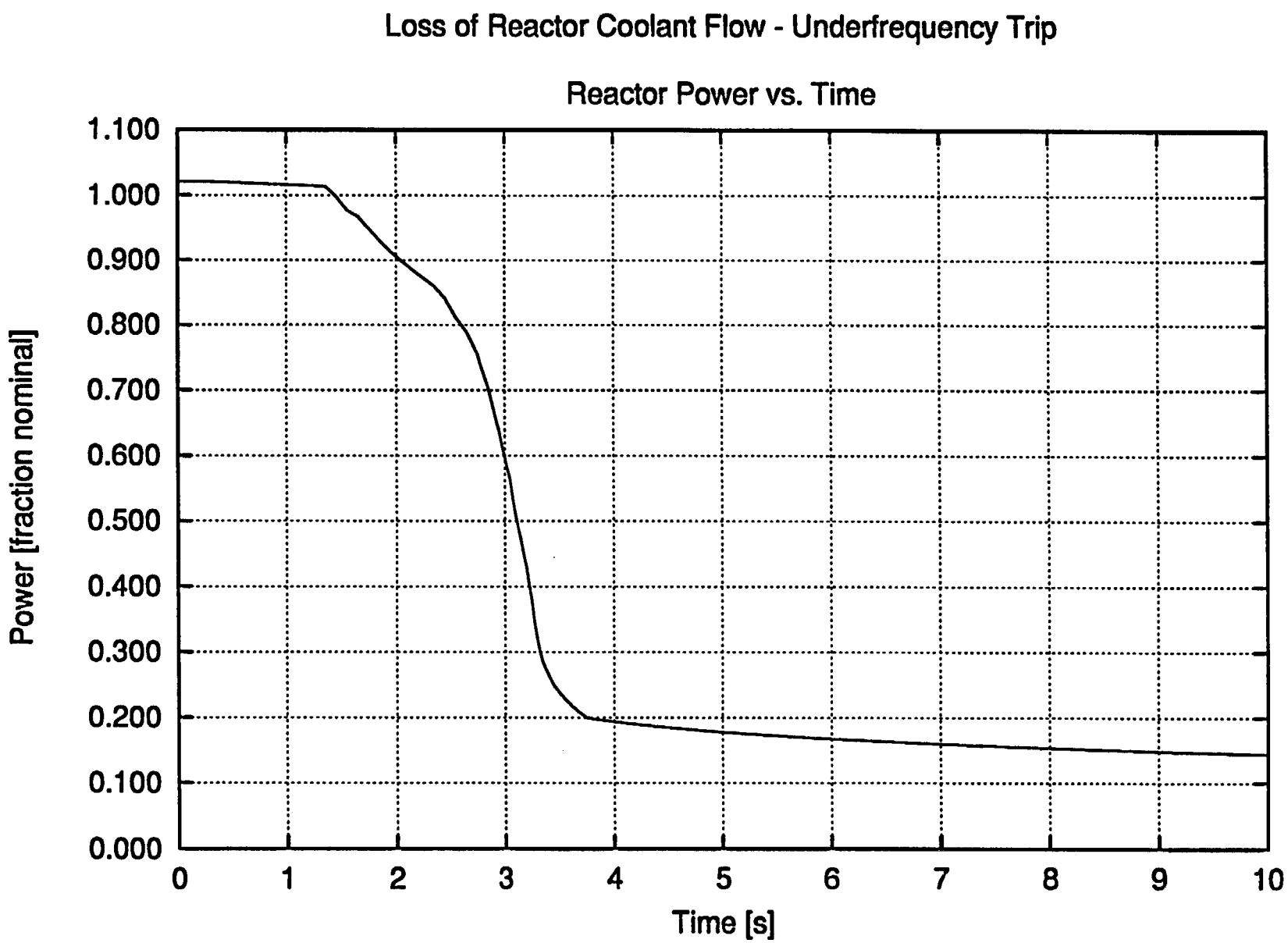
Loss of Reactor Coolant Flow - Underfrequency Trip
Reactor Power vs. Time
Power [fraction nominal]
1.100
1.000
0.900
0.800
0.700
0.600
0.500
0.400
0.300
0.200
0.100
0.000
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.11-6

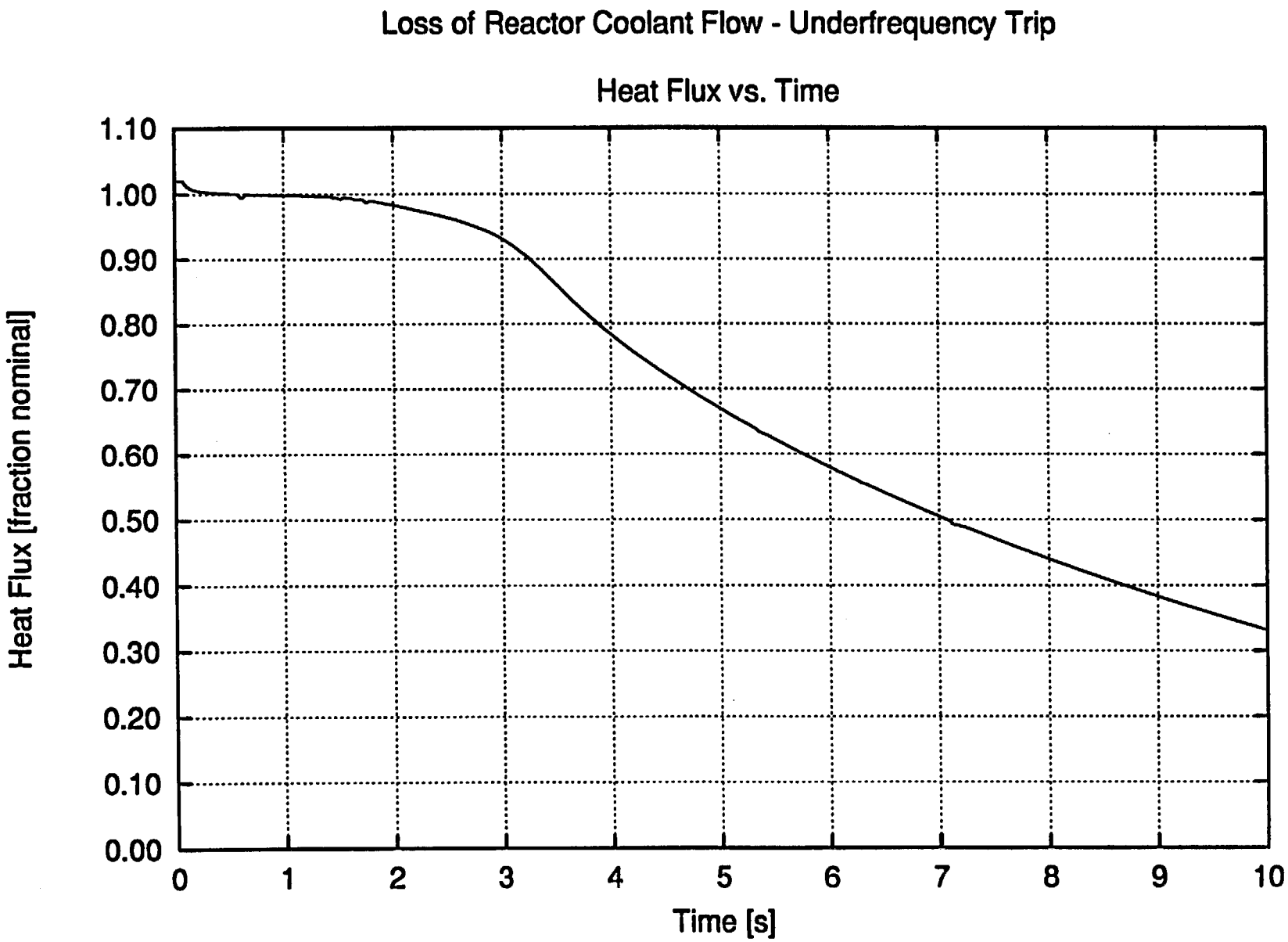
Loss of Reactor Coolant Flow - Underfrequency Trip
Heat Flux vs. Time
Heat Flux [fraction nominal]
1.10
1.00
0.90
0.80
0.70
0.60
0.50
0.40
0.30
0.20
0.10
0.00
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.11-7

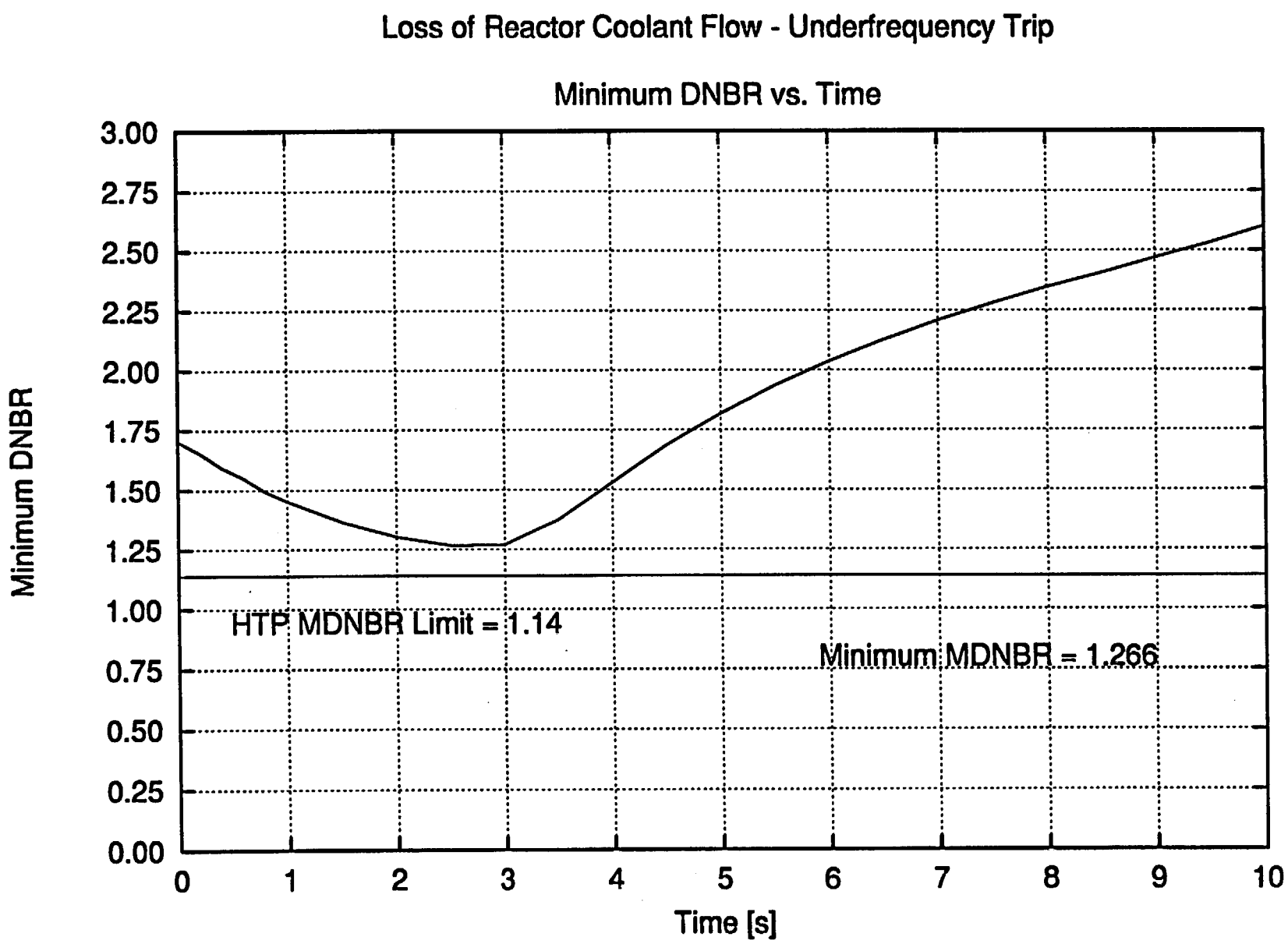
Loss of Reactor Coolant Flow - Underfrequency Trip
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0
1
2
3
4
5
6
7
8
9
10
Time [s]
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.266

Figure 3.11-8

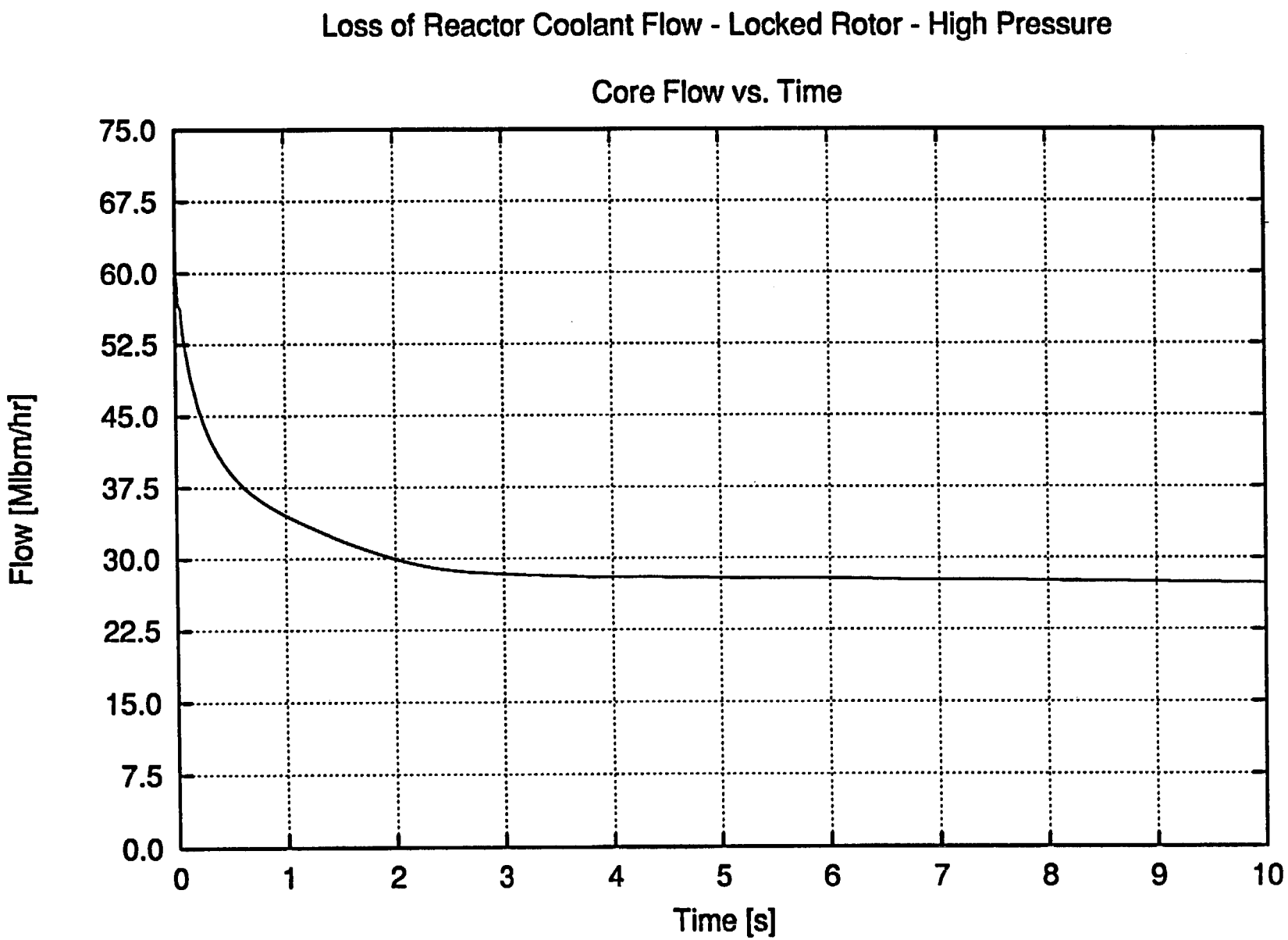
Loss of Reactor Coolant Flow - Locked Rotor - High Pressure
Core Flow vs. Time
Flow [Mlbm/hr]
75.0
67.5
60.0
52.5
45.0
37.5
30.0
22.5
15.0
7.5
0.0
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.12-1

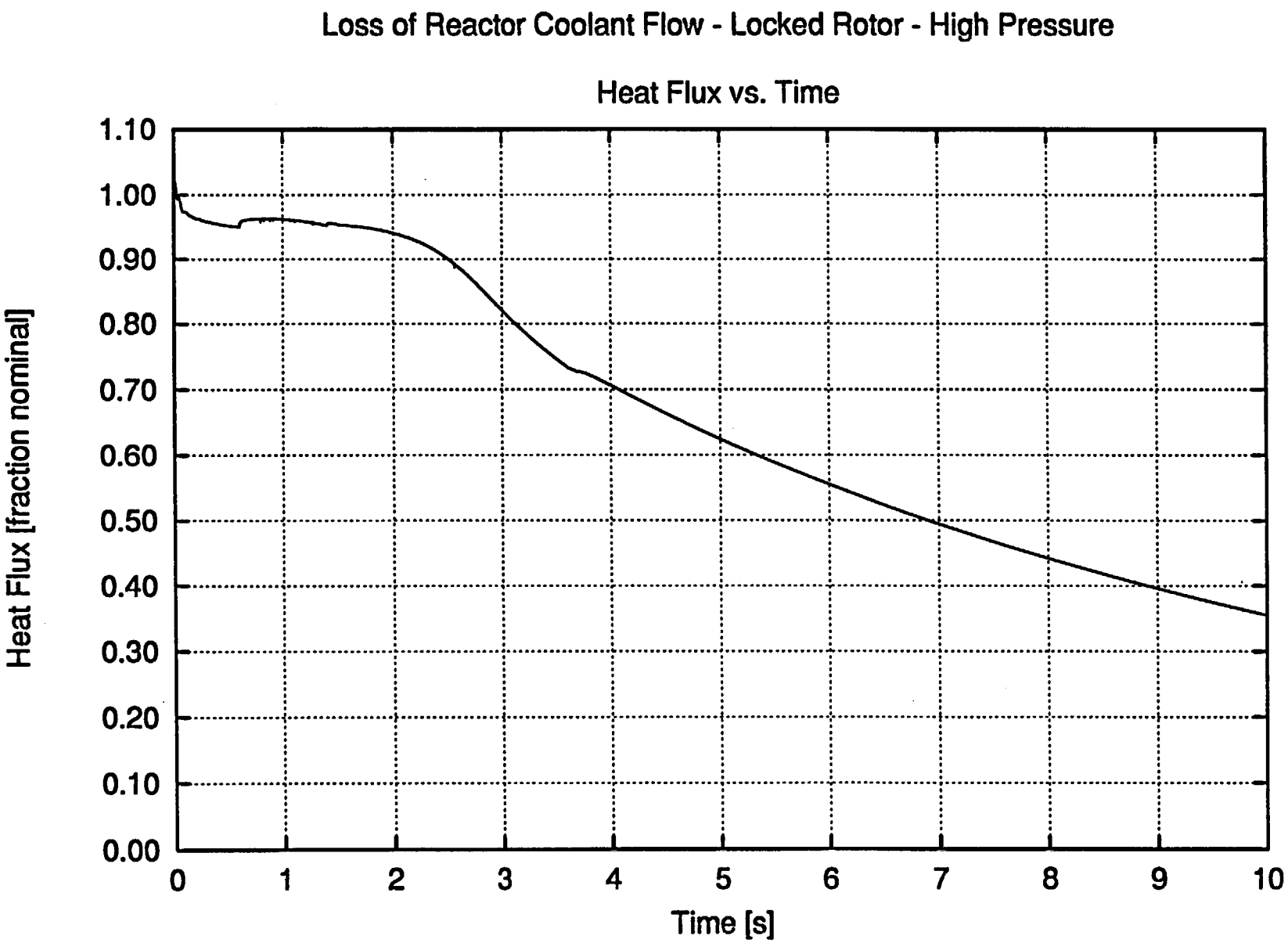
Loss of Reactor Coolant Flow - Locked Rotor - High Pressure
Heat Flux vs. Time
Heat Flux [fraction nominal]
1.10
1.00
0.90
0.80
0.70
0.60
0.50
0.40
0.30
0.20
0.10
0.00
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.12-2

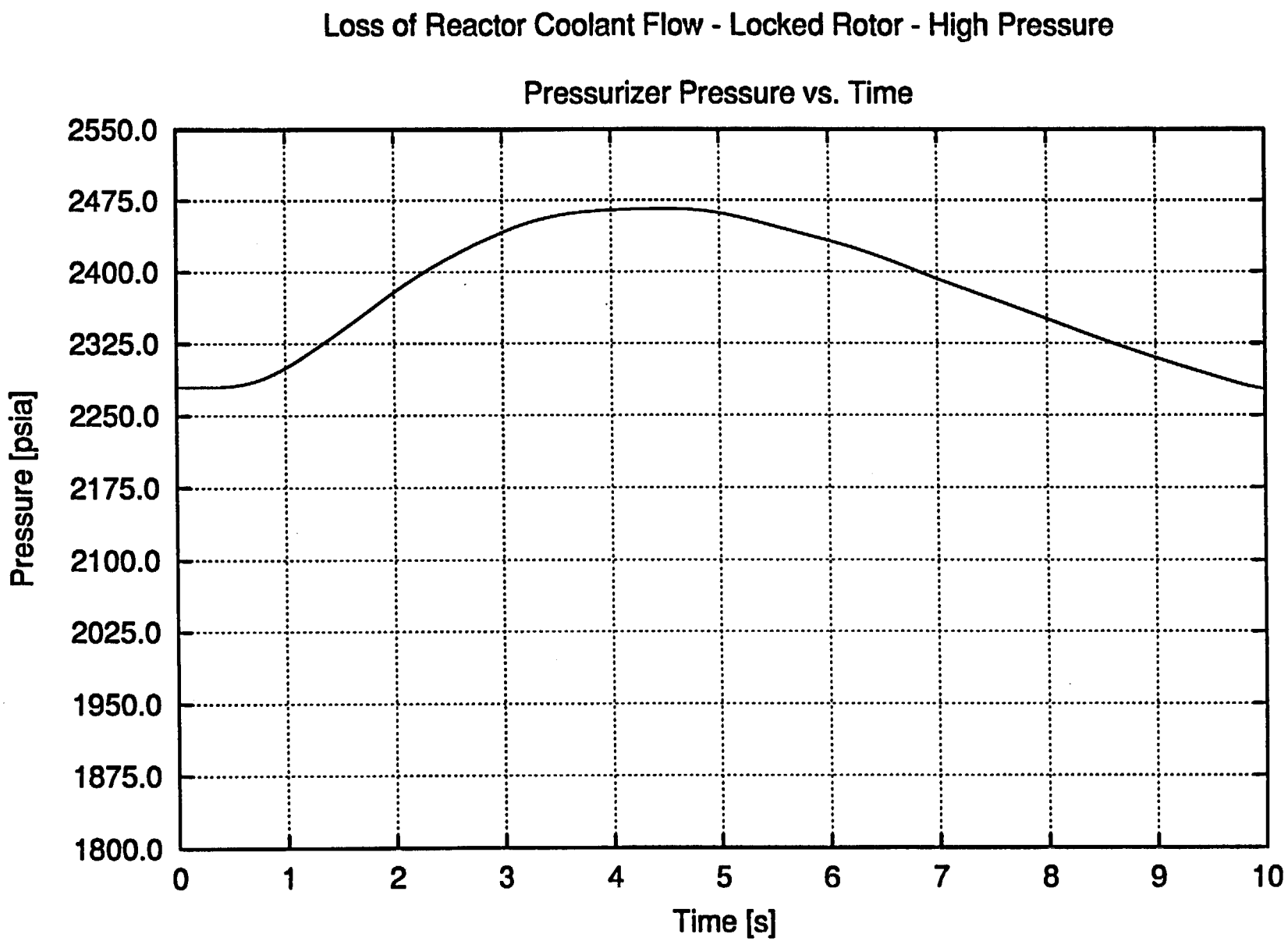
Loss of Reactor Coolant Flow - Locked Rotor - High Pressure
Pressurizer Pressure vs. Time
Pressure [psia]
2550.0
2475.0
2400.0
2325.0
2250.0
2175.0
2100.0
2025.0
1950.0
1875.0
1800.0
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.12-3

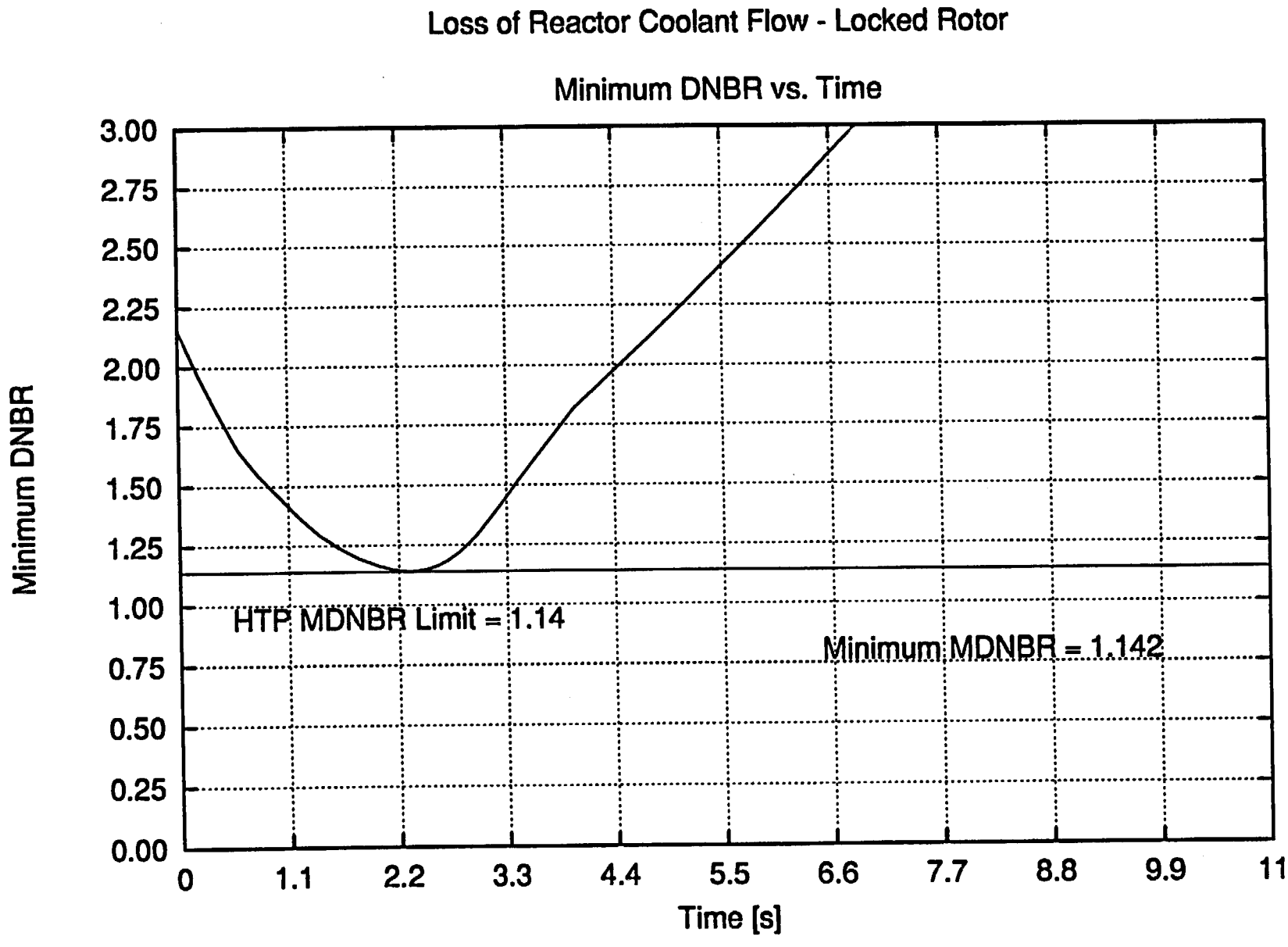
Loss of Reactor Coolant Flow - Locked Rotor
Minimum DNBR vs. Time
Minimum DNBR
3.00
2.75
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.00
HTP MDNBR Limit = 1.14
Minimum MDNBR = 1.142
0
1.1
2.2
3.3
4.4
5.5
6.6
7.7
8.8
9.9
11
Time [s]

Figure 3.12-4

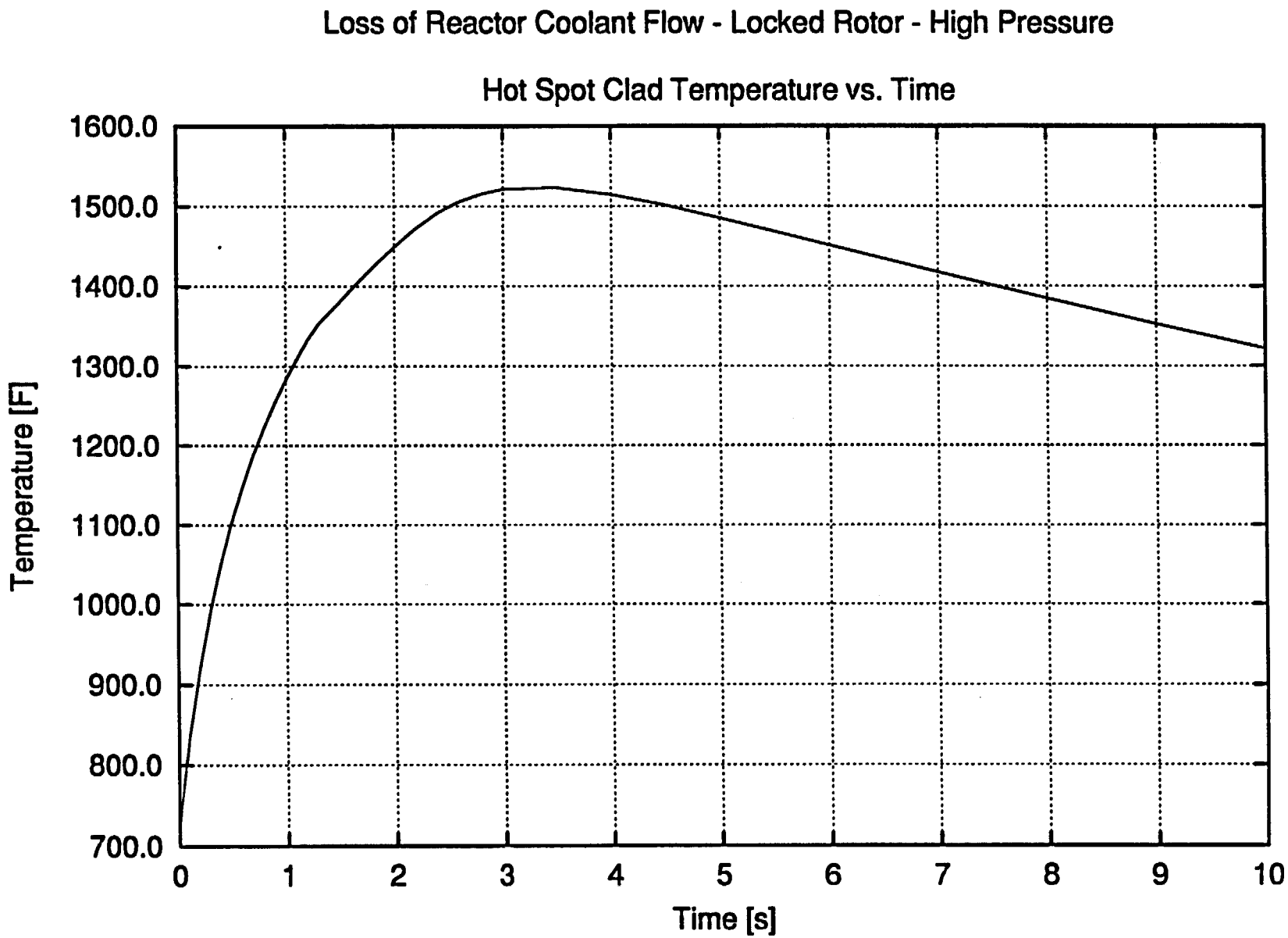
Loss of Reactor Coolant Flow - Locked Rotor - High Pressure
Hot Spot Clad Temperature vs. Time
Temperature [F]
1600.0
1500.0
1400.0
1300.0
1200.0
1100.0
1000.0
900.0
800.0
700.0
0
1
2
3
4
5
6
7
8
9
10
Time [s]

Figure 3.12-5

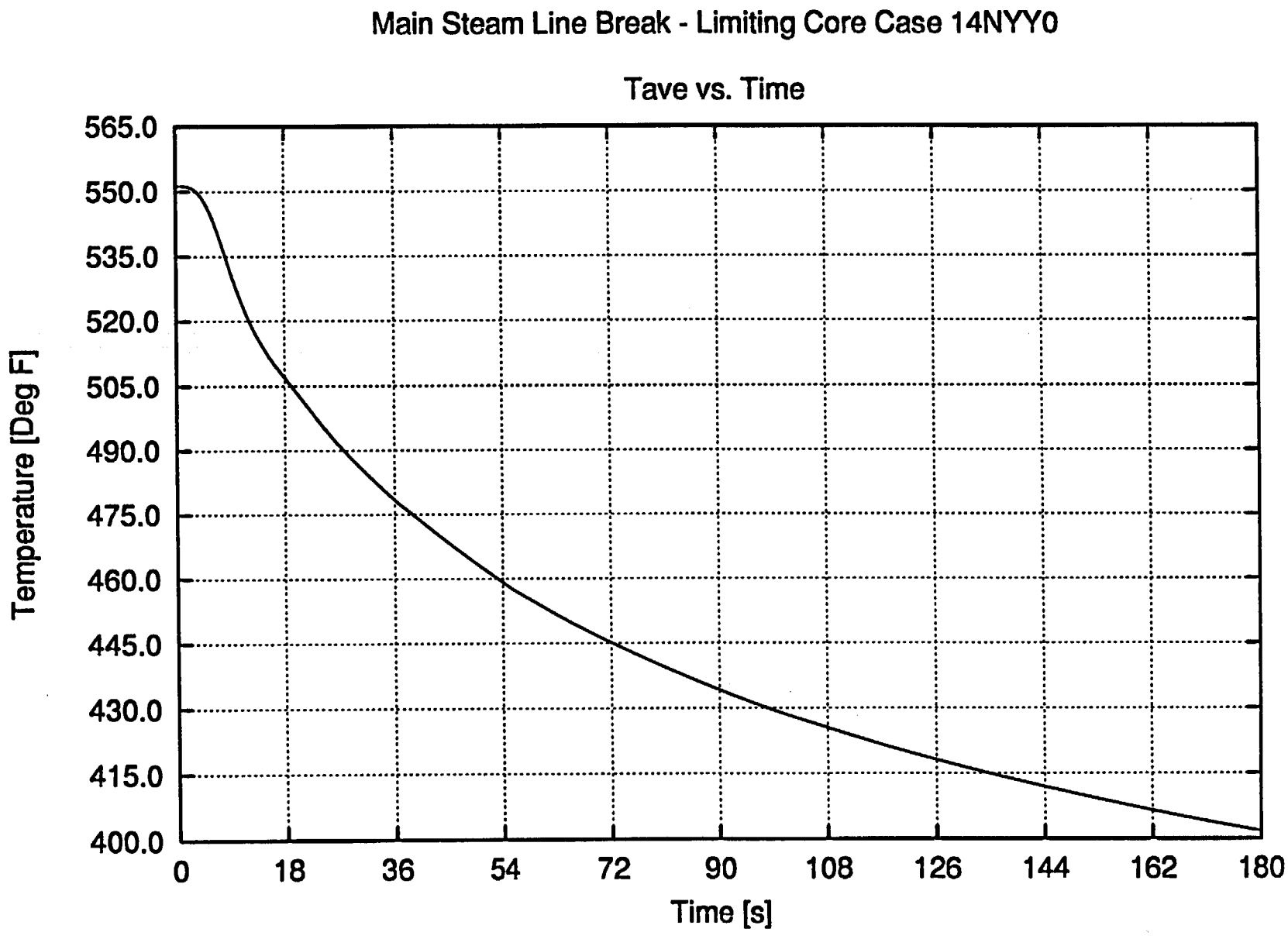
Main Steam Line Break - Limiting Core Case 14NYY0
Tave vs. Time
Temperature [Deg F]
565.0
550.0
535.0
520.0
505.0
490.0
475.0
460.0
445.0
430.0
415.0
400.0
0
18
36
54
72
90
108
126
144
162
180
Time [s]

Figure 3.14-1

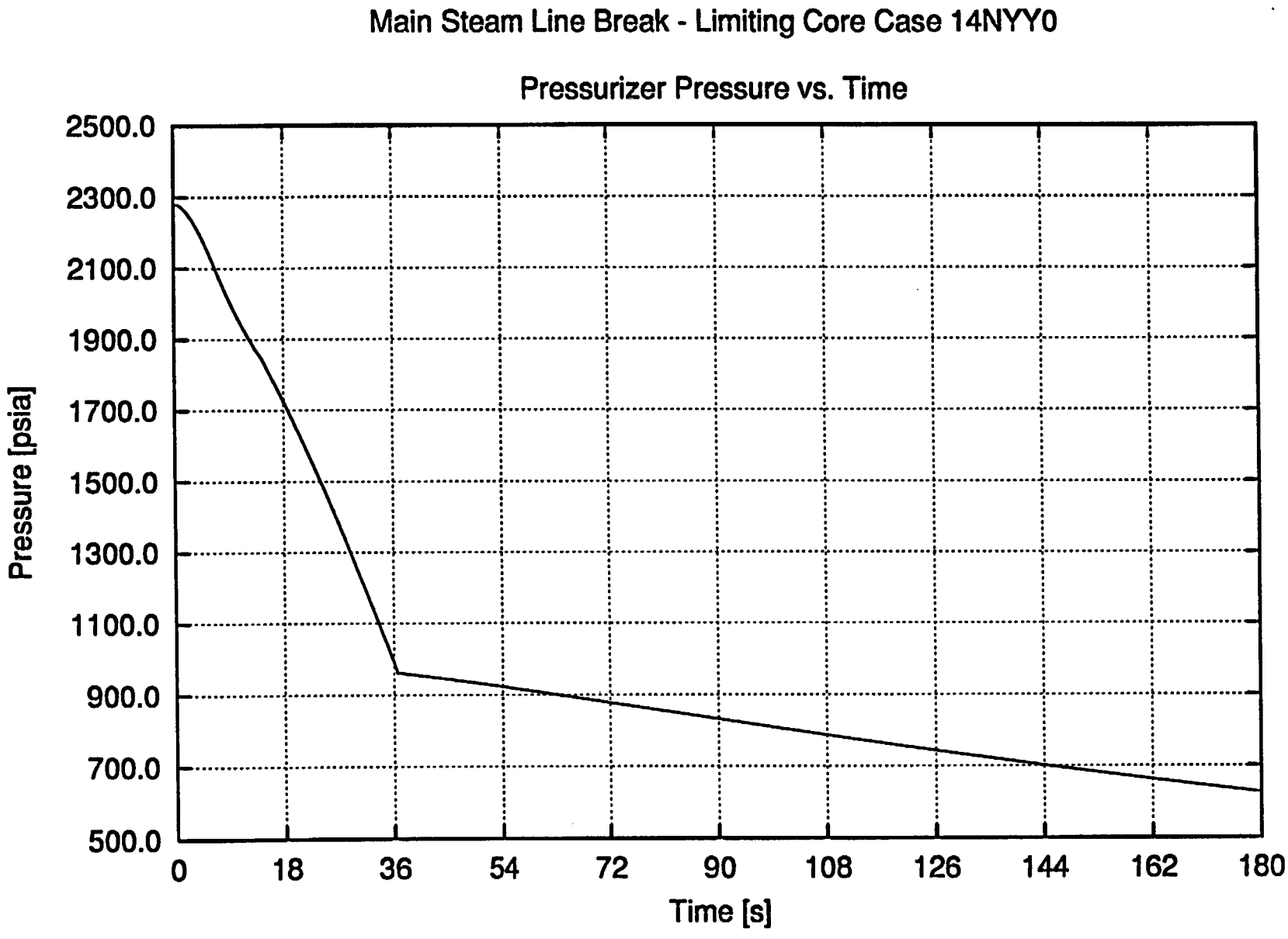
Main Steam Line Break - Limiting Core Case 14NYY0
Pressurizer Pressure vs. Time
Pressure [psia]
2500.0
2300.0
2100.0
1900.0
1700.0
1500.0
1300.0
1100.0
900.0
700.0
500.0
0
18
36
54
72
90
108
126
144
162
180
Time [s]

Figure 3.14-2

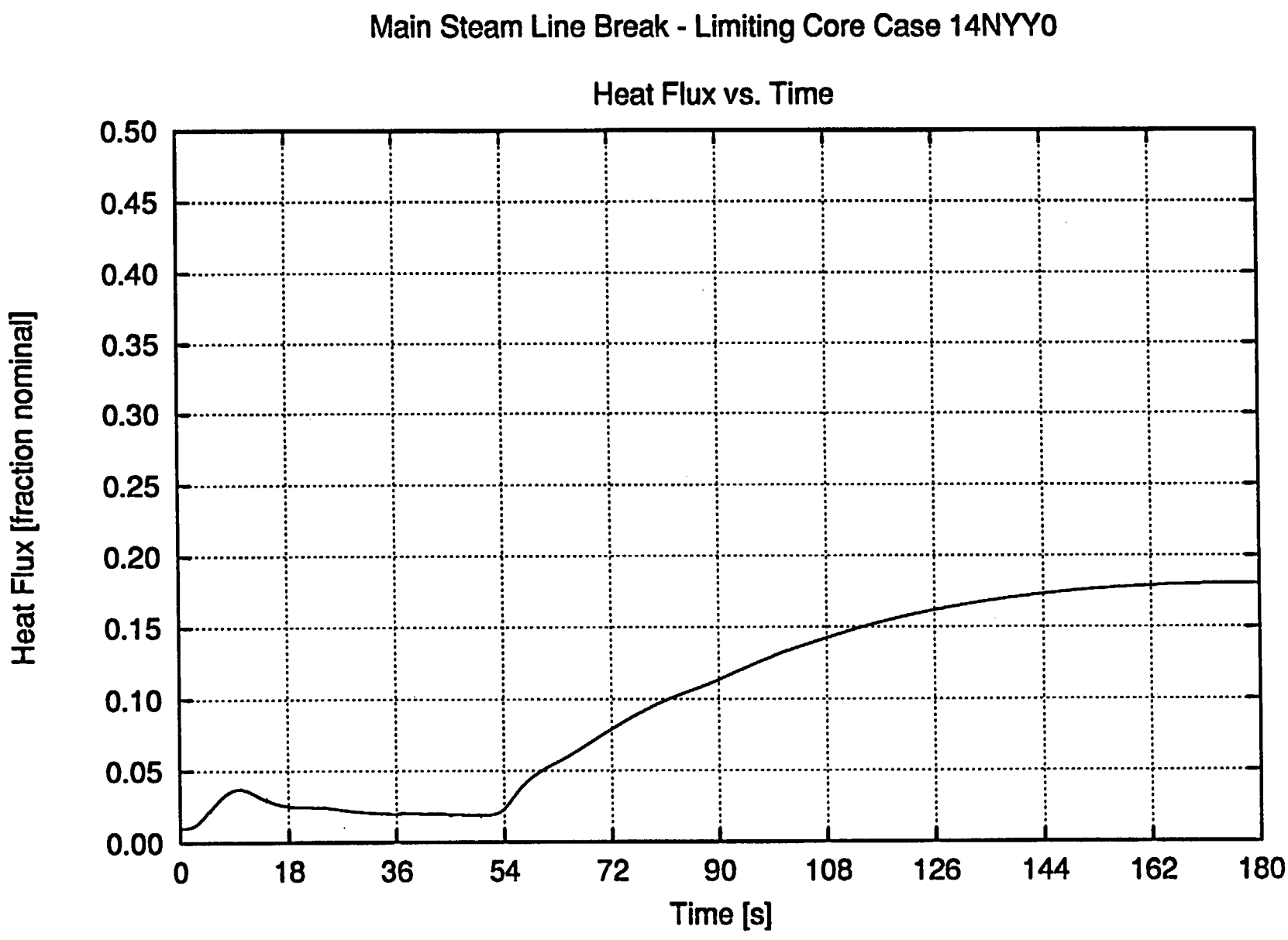
Main Steam Line Break - Limiting Core Case 14NYY0
Heat Flux vs. Time
Heat Flux [fraction nominal]
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
0
18
36
54
72
90
108
126
144
162
180
Time [s]

Figure 3.14-3

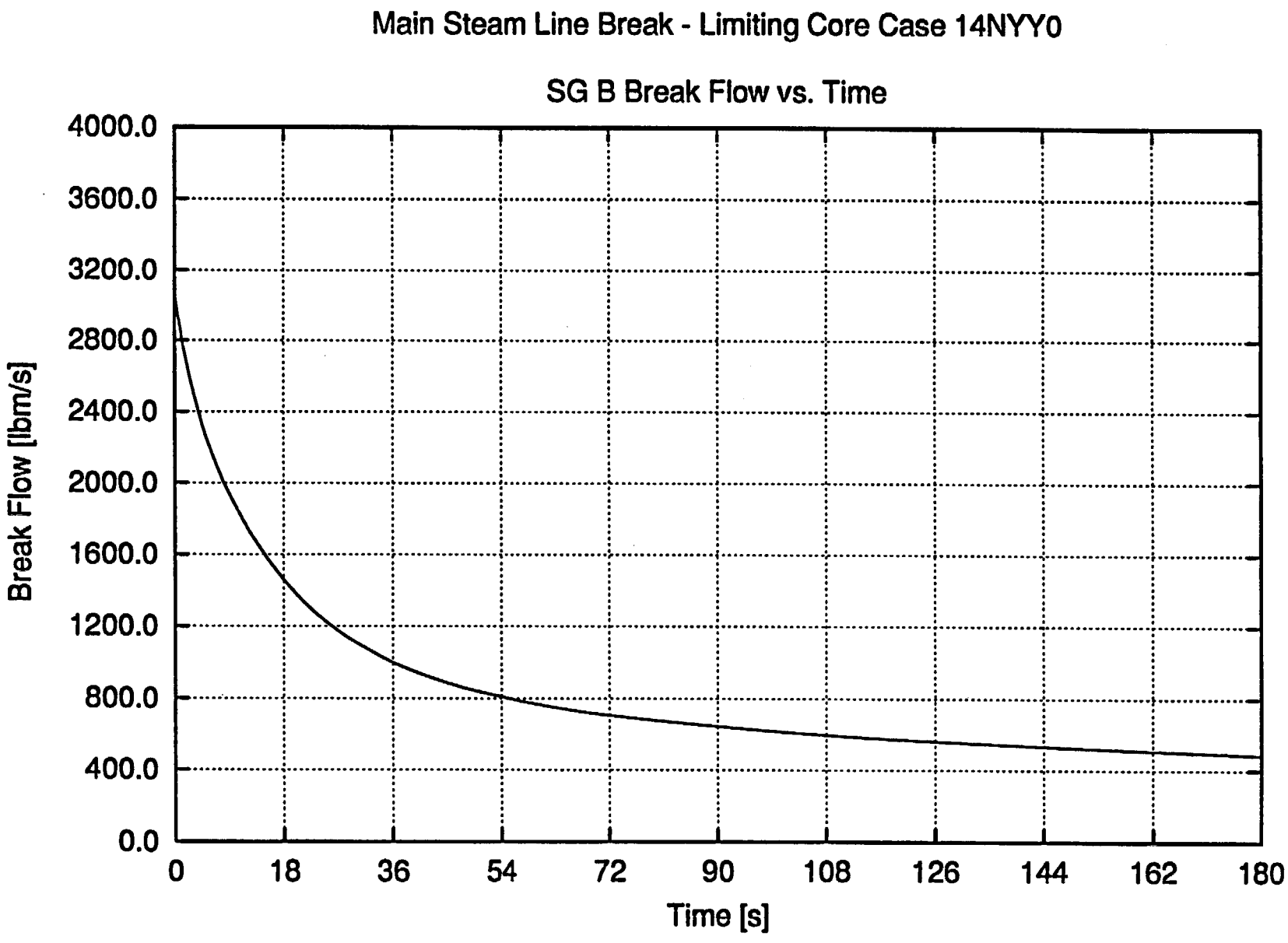
Main Steam Line Break - Limiting Core Case 14NYY0
SG B Break Flow vs. Time
Break Flow [lbm/s]
4000.0
3600.0
3200.0
2800.0
2400.0
2000.0
1600.0
1200.0
800.0
400.0
0.0
0
18
36
54
72
90
108
126
144
162
180
Time [s]

Figure 3.14-4

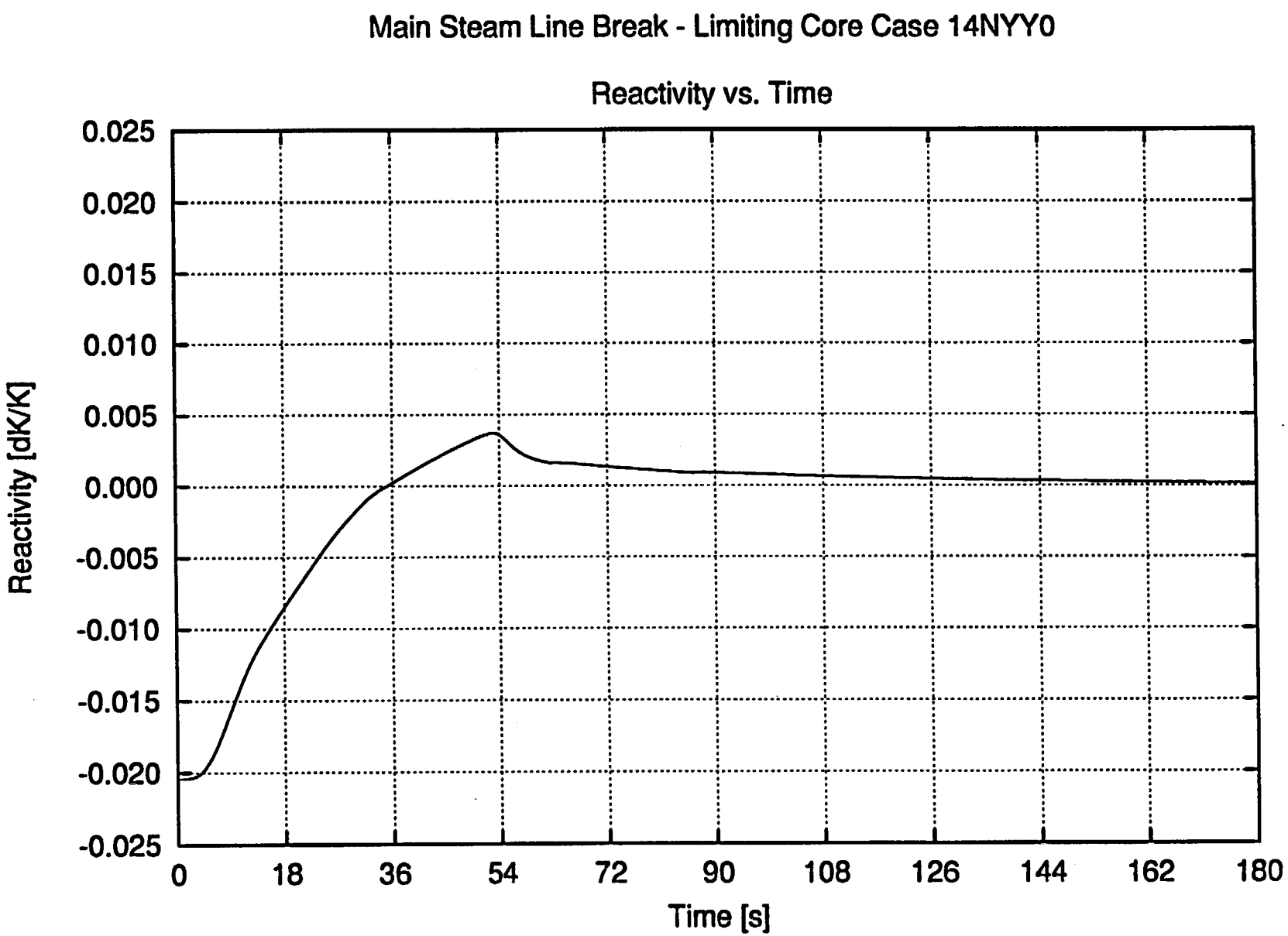
Main Steam Line Break - Limiting Core Case 14NYY0
Reactivity vs. Time
Reactivity [dK/K]
0.025
0.020
0.015
0.010
0.005
0.000
-0.005
-0.010
-0.015
-0.020
-0.025
0
18
36
54
72
90
108
126
144
162
180
Time [s]

Figure 3.14-5

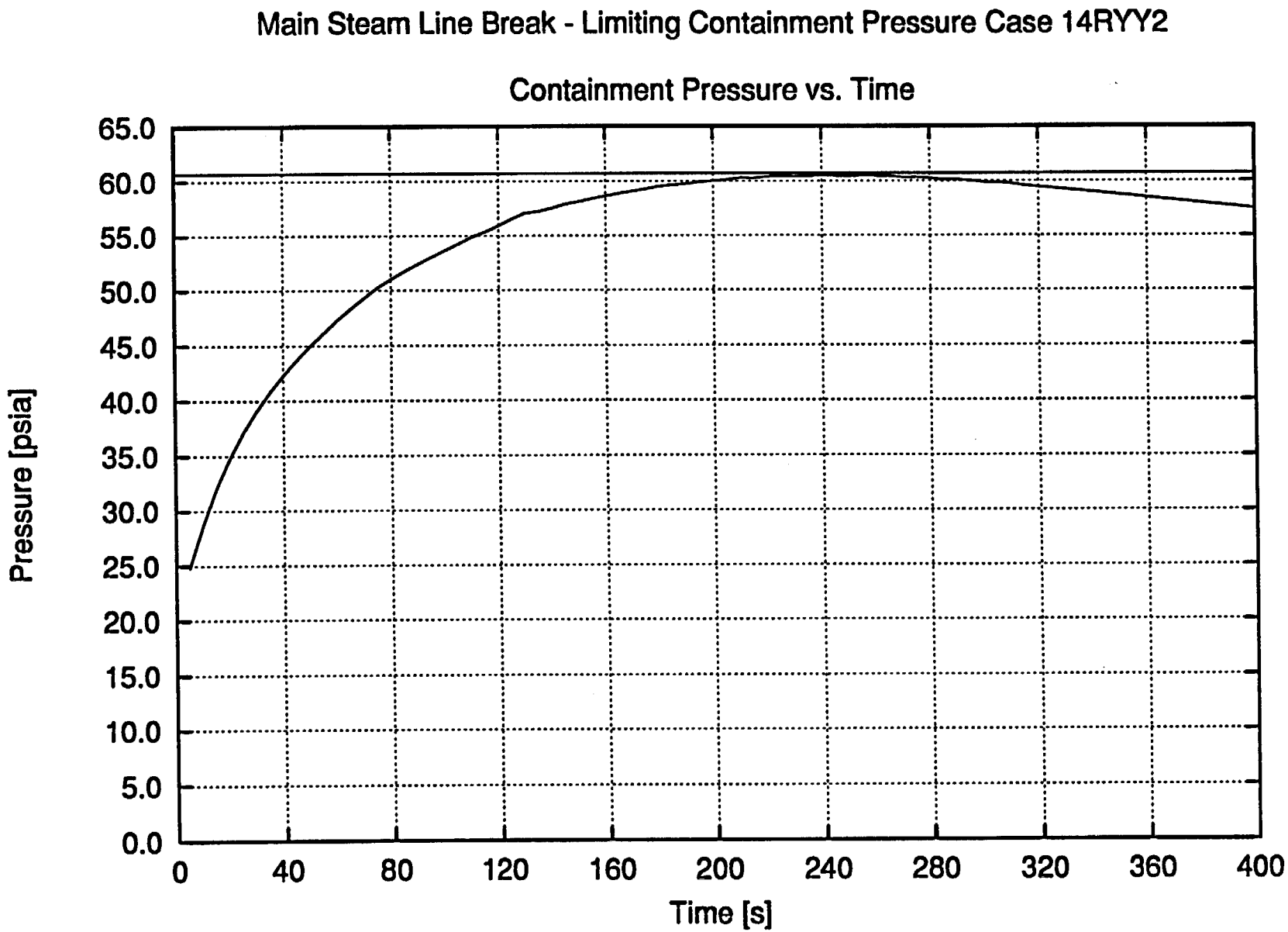
Main Steam Line Break - Limiting Containment Pressure Case 14RYY2
Containment Pressure vs. Time
Pressure [psia]
65.0
60.0
55.0
50.0
45.0
40.0
35.0
30.0
25.0
20.0
15.0
10.0
5.0
0.0
0
40
80
120
160
200
240
280
320
360
400
Time [s]

Figure 3.14-6

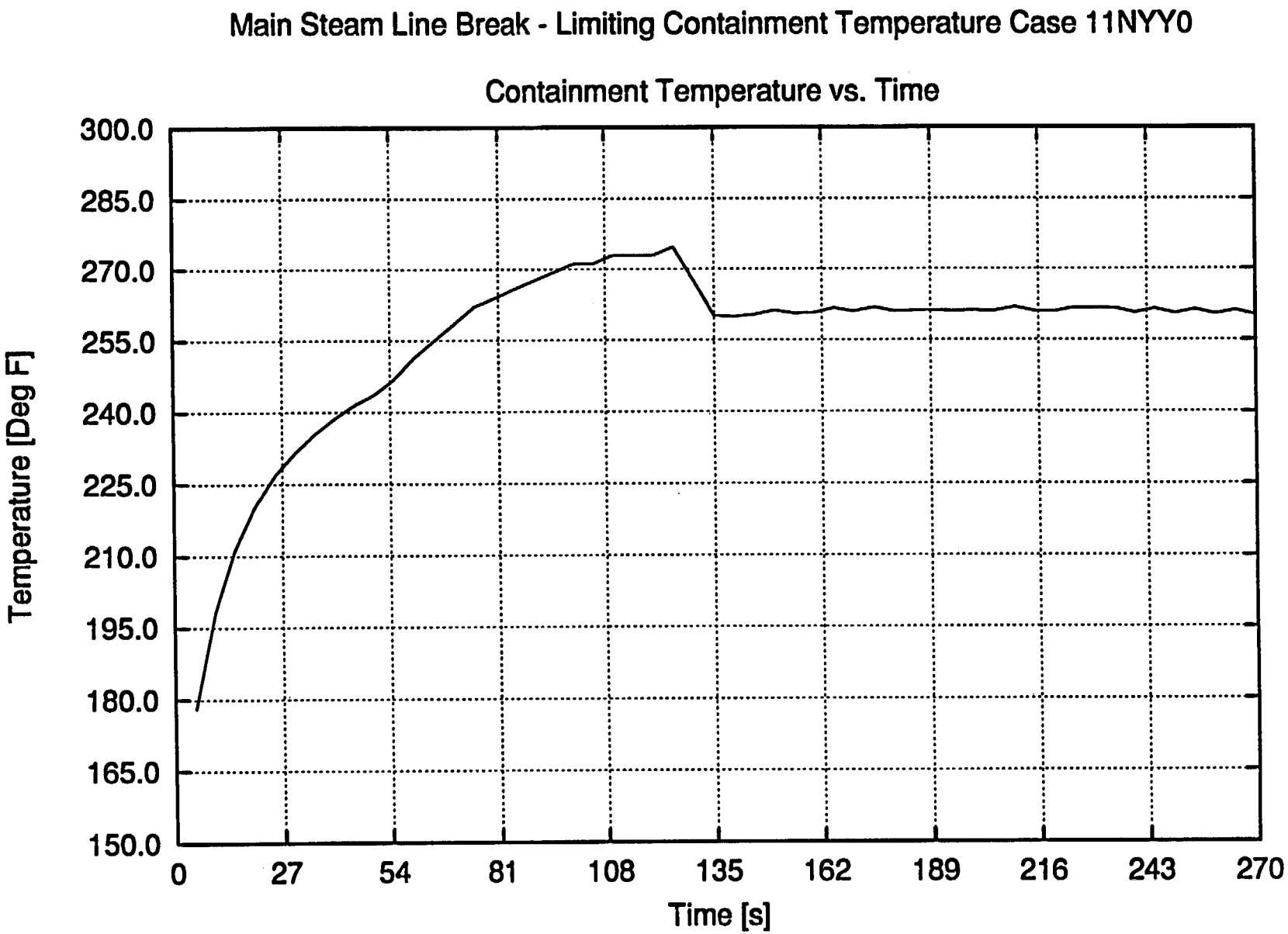
Main Steam Line Break - Limiting Containment Temperature Case 11NYY0
Containment Temperature vs. Time
Temperature [Deg F]
300.0
285.0
270.0
255.0
240.0
225.0
210.0
195.0
180.0
165.0
150.0
0
27
54
81
108
135
162
189
216
243
270
Time [s]

Figure 3.14-7

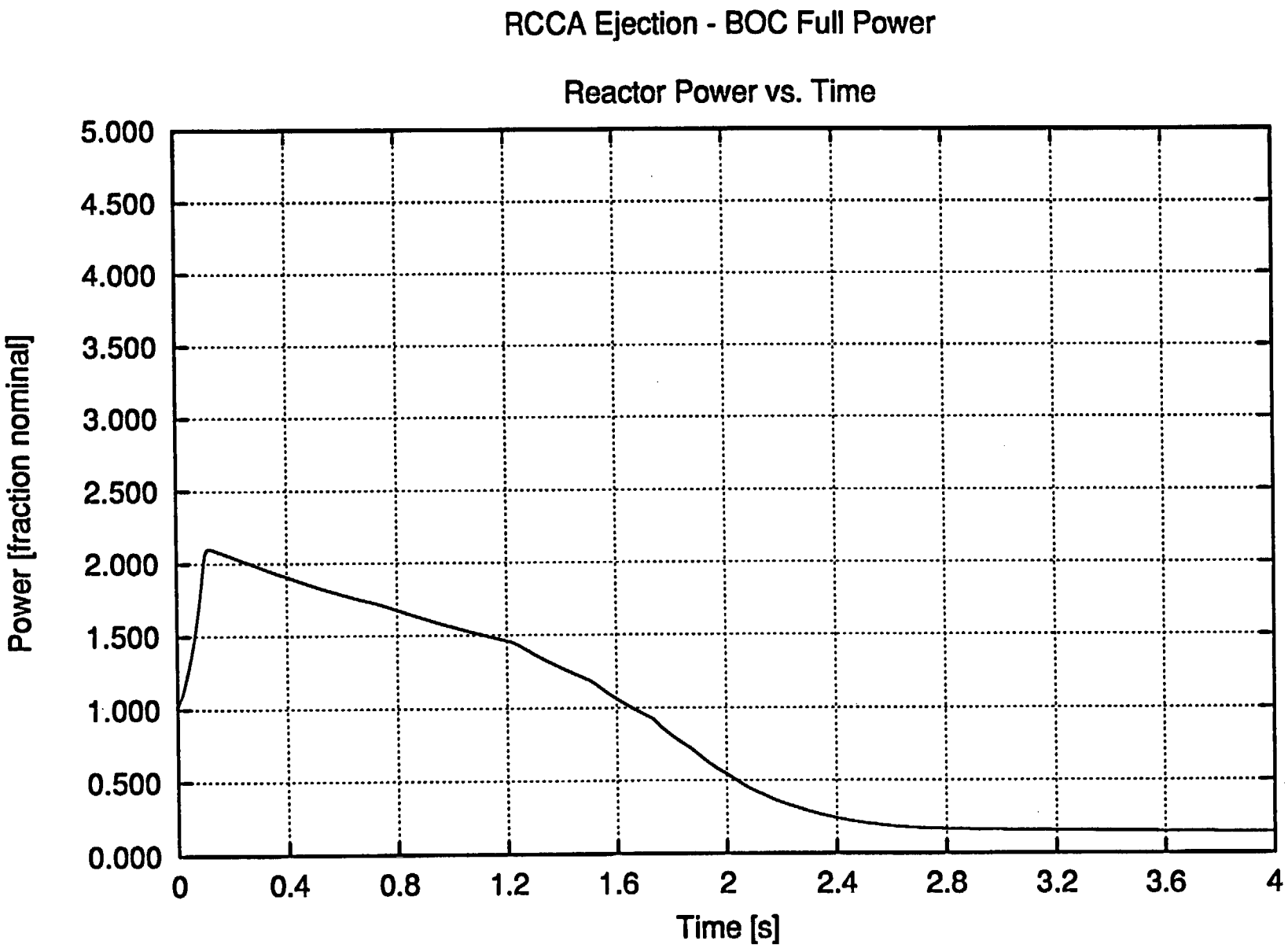
RCCA Ejection - BOC Full Power
Reactor Power vs. Time
Power [fraction nominal]
5.000
4.500
4.000
3.500
3.000
2.500
2.000
1.500
1.000
0.500
0.000
0
0.4
0.8
1.2
1.6
2
2.4
2.8
3.2
3.6
4
Time [s]

Figure 3.15-1

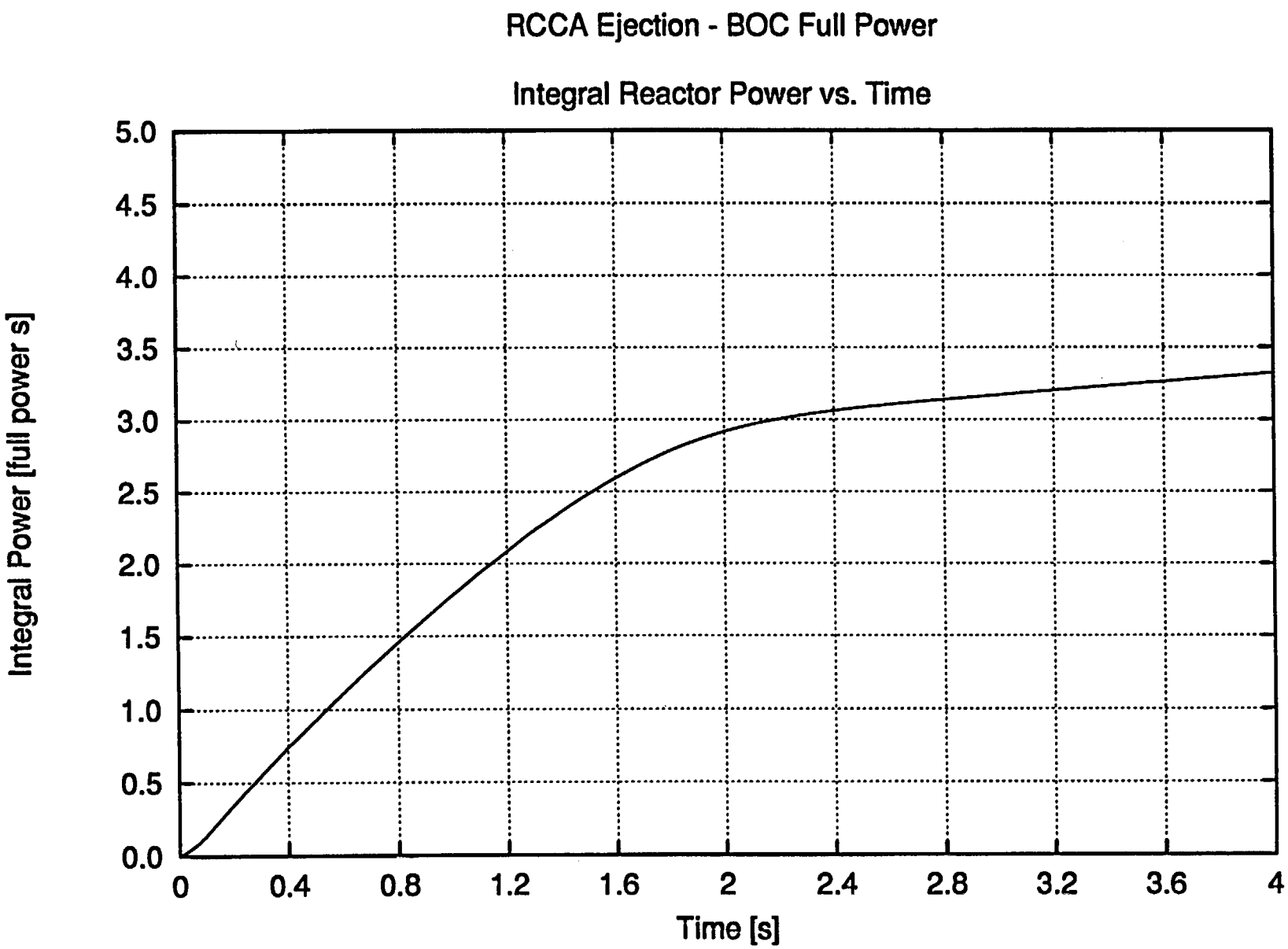
RCCA Ejection - BOC Full Power
Integral Reactor Power vs. Time
Integral Power [full power s]
5.0
4.5
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
0
0.4
0.8
1.2
1.6
2
2.4
2.8
3.2
3.6
4
Time [s]

Figure 3.15-2

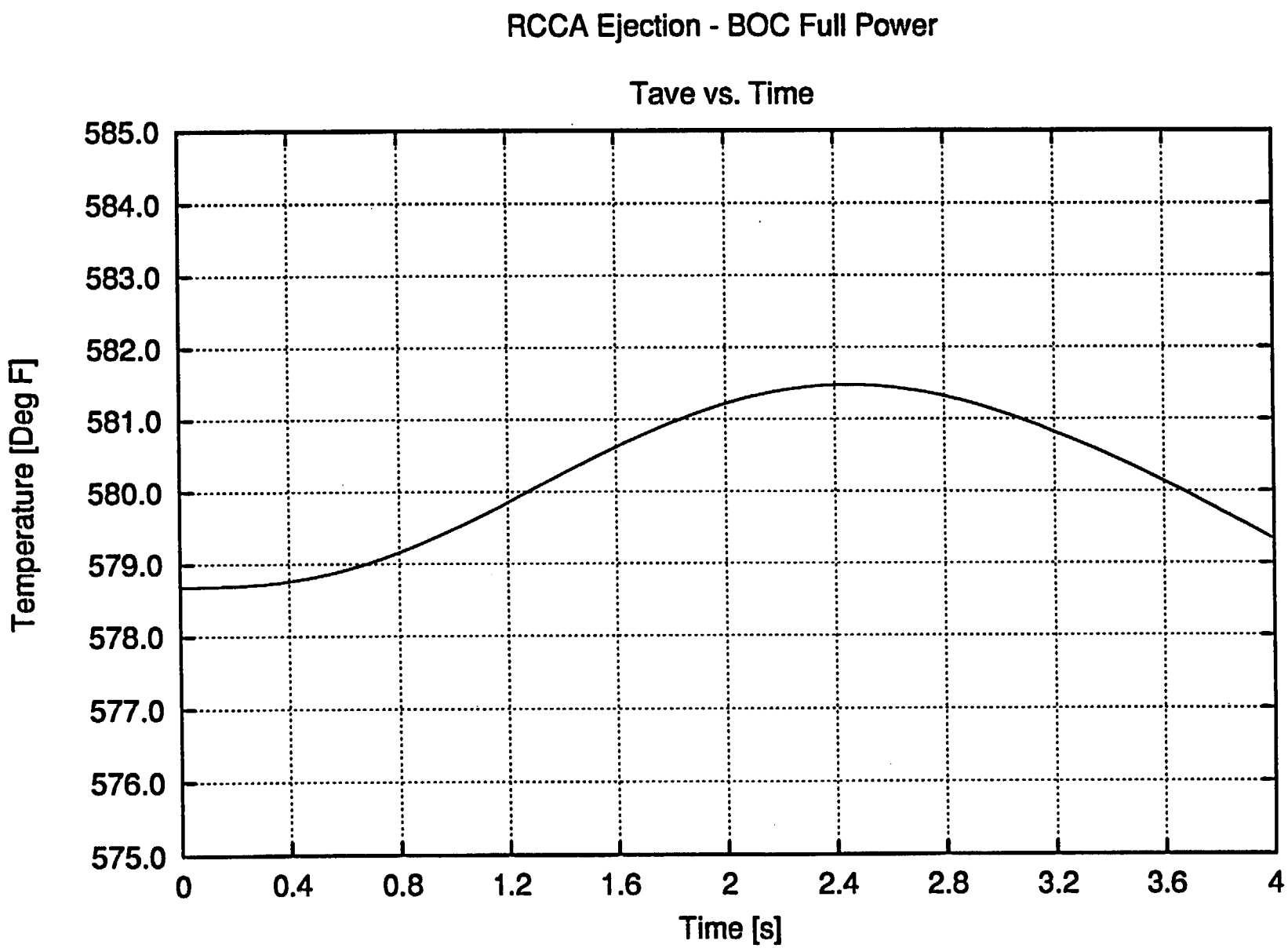
RCCA Ejection - BOC Full Power
Tave vs. Time
Temperature [Deg F]
585.0
584.0
583.0
582.0
581.0
580.0
579.0
578.0
577.0
576.0
575.0
0
0.4
0.8
1.2
1.6
2
2.4
2.8
3.2
3.6
4
Time [s]

Figure 3.15-3

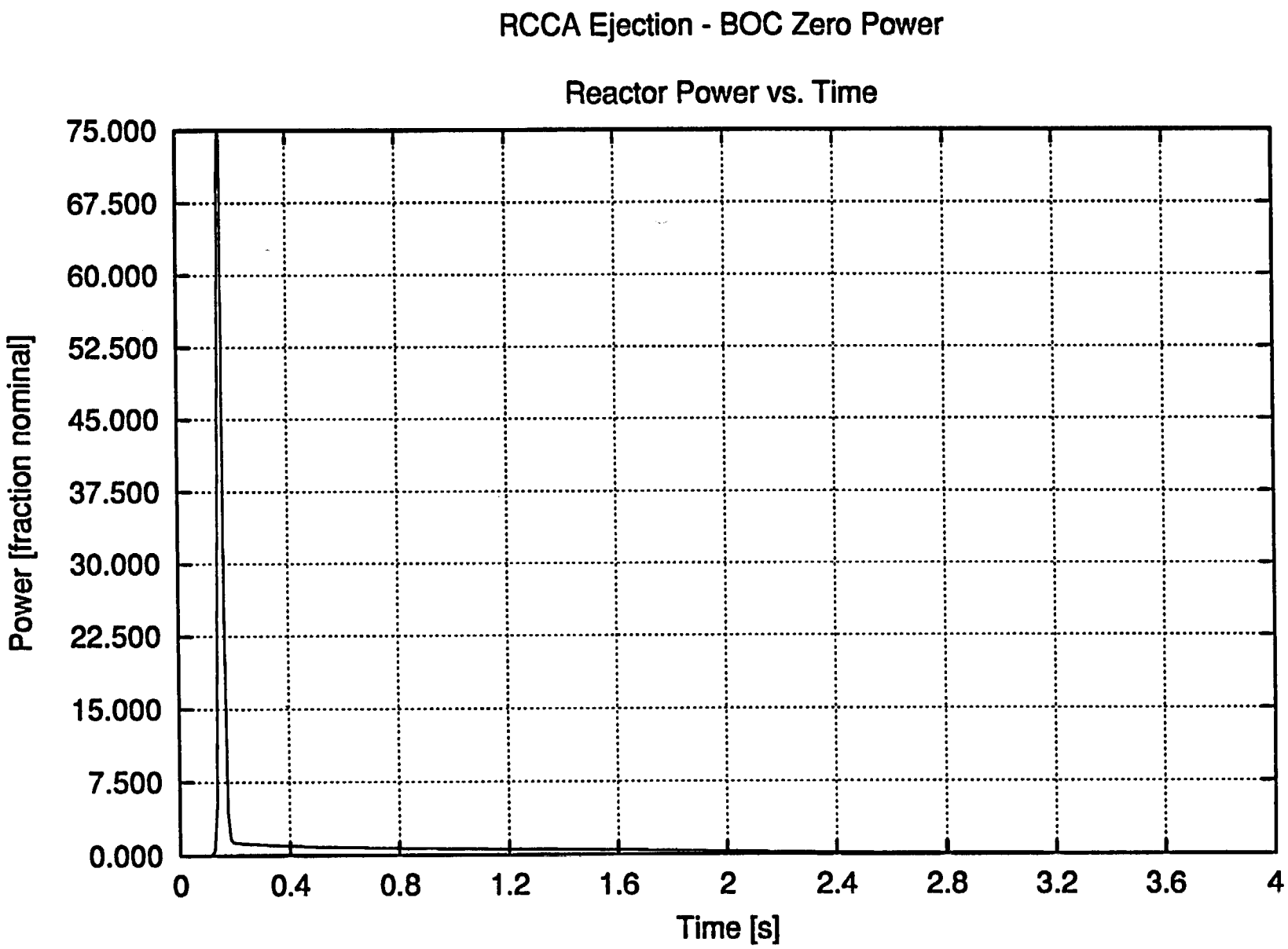
RCCA Ejection - BOC Zero Power
Reactor Power vs. Time
Power [fraction nominal]
75.000
67.500
60.000
52.500
45.000
37.500
30.000
22.500
15.000
7.500
0.000
0
0.4
0.8
1.2
1.6
2
2.4
2.8
3.2
3.6
4
Time [s]

Figure 3.15-4

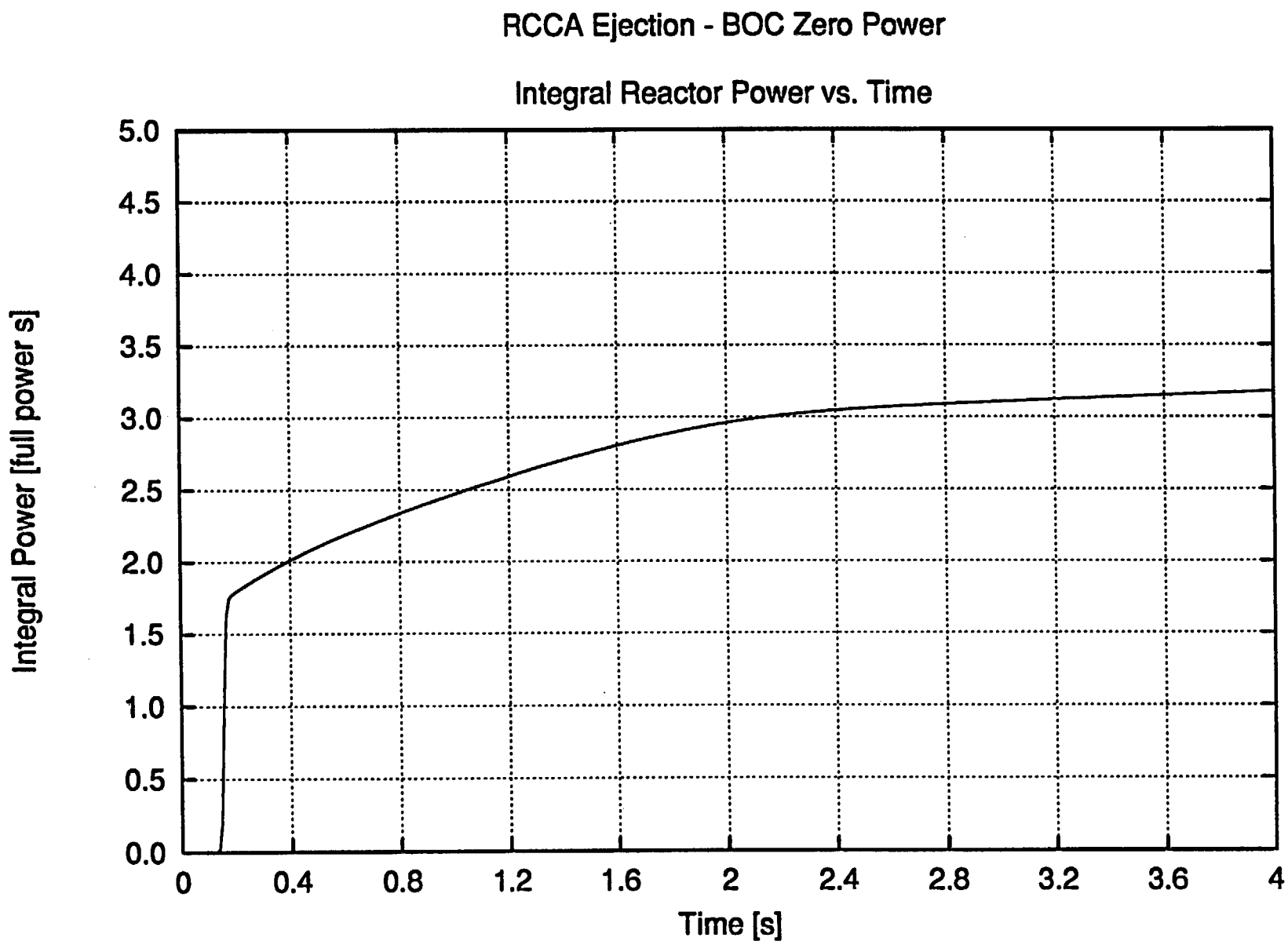
RCCA Ejection - BOC Zero Power
Integral Reactor Power vs. Time
Integral Power [full power s]
5.0
4.5
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
0
0.4
0.8
1.2
1.6
2
2.4
2.8
3.2
3.6
4
Time [s]

Figure 3.15-5

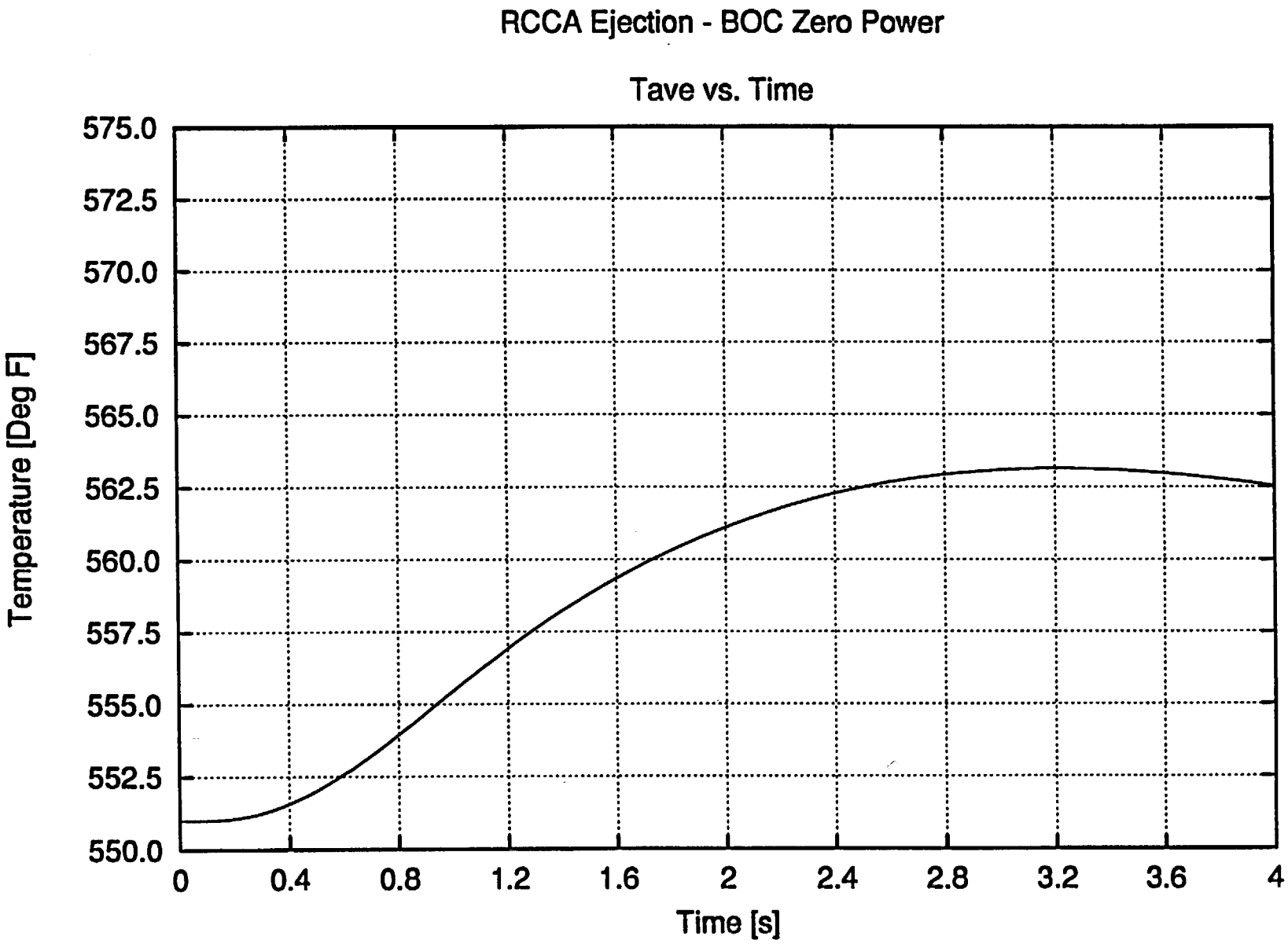
RCCA Ejection - BOC Zero Power
Tave vs. Time
Temperature [Deg F]
575.0
572.5
570.0
567.5
565.0
562.5
560.0
557.5
555.0
552.5
550.0
0
0.4
0.8
1.2
1.6
2
2.4
2.8
3.2
3.6
4
Time [s]

Figure 3.15-6

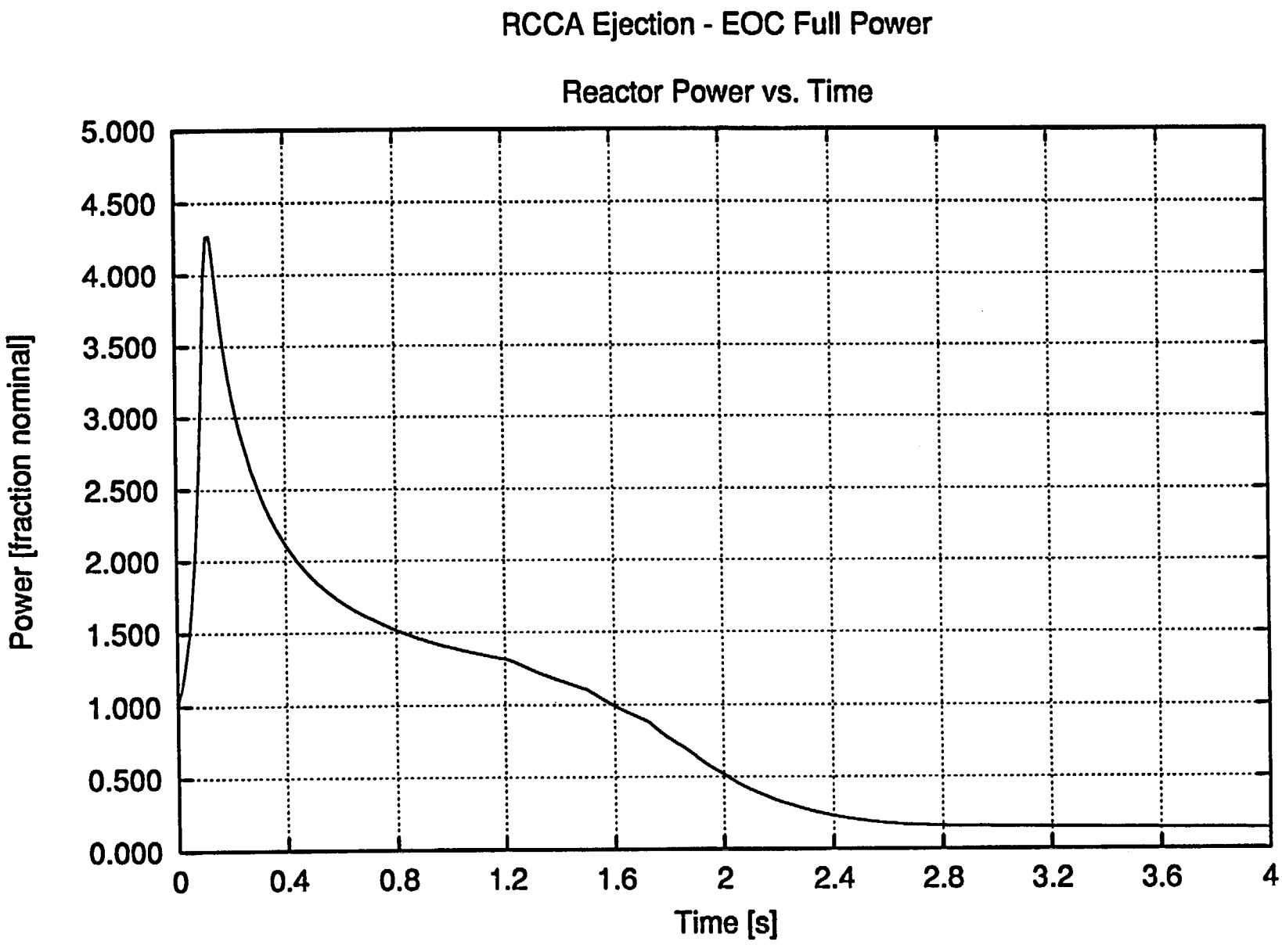
RCCA Ejection - EOC Full Power
Reactor Power vs. Time
Power [fraction nominal]
5.000
4.500
4.000
3.500
3.000
2.500
2.000
1.500
1.000
0.500
0.000
0
0.4
0.8
1.2
1.6
2
2.4
2.8
3.2
3.6
4
Time [s]

Figure 3.15-7

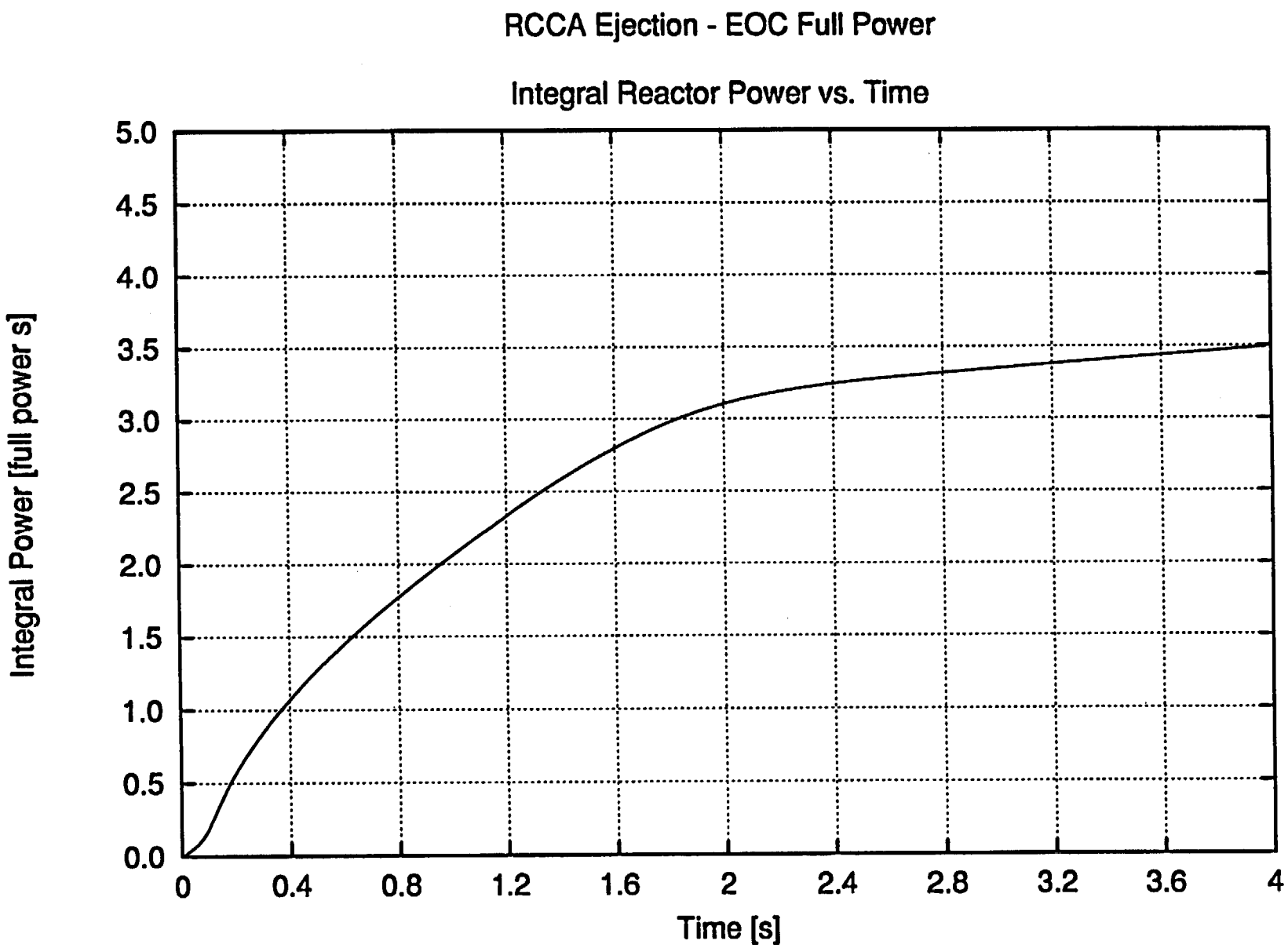
RCCA Ejection - EOC Full Power
Integral Reactor Power vs. Time
Integral Power [full power s]
5.0
4.5
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
0
0.4
0.8
1.2
1.6
2
2.4
2.8
3.2
3.6
4
Time [s]

Figure 3.15-8

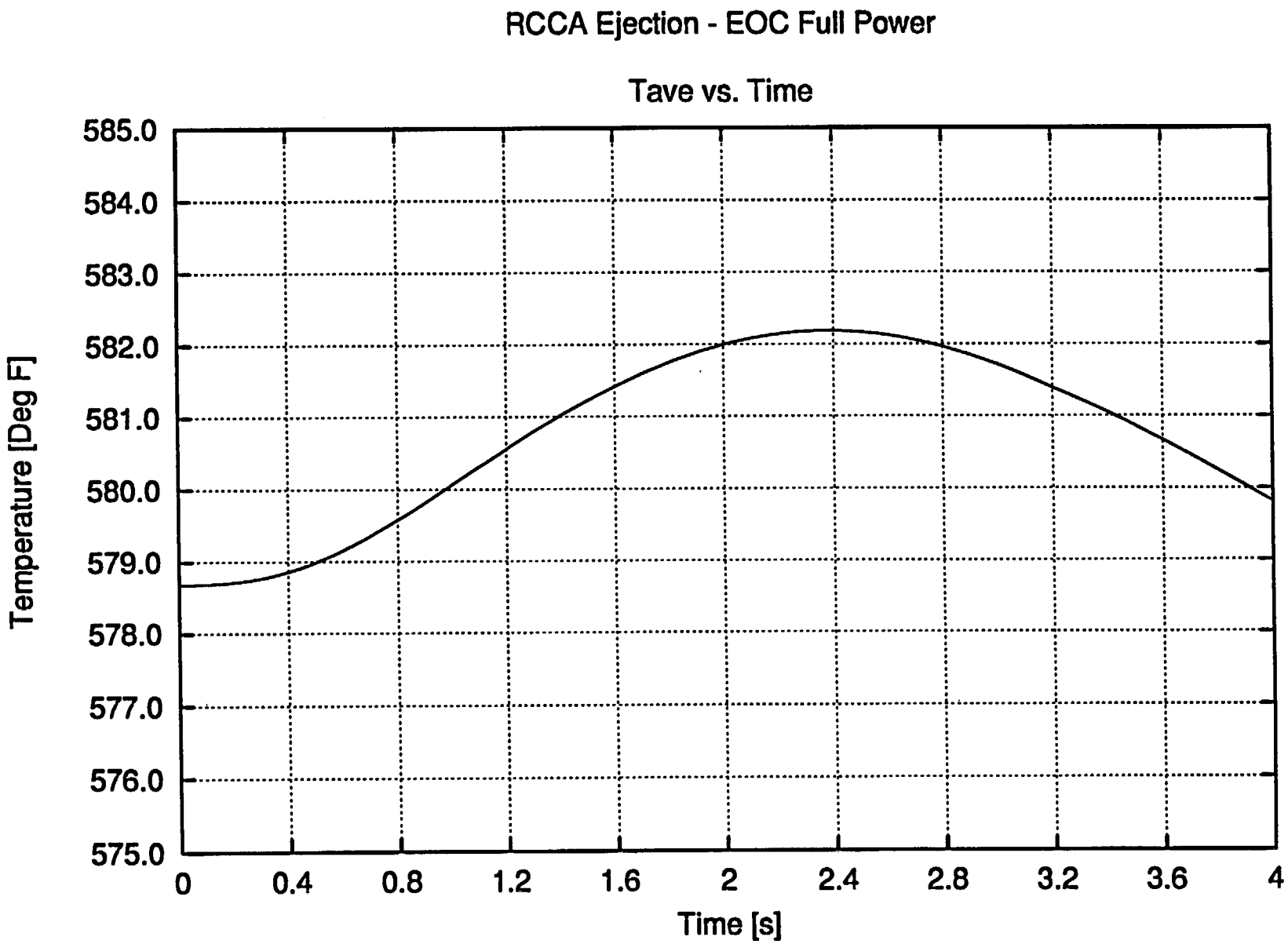
RCCA Ejection - EOC Full Power
Tave vs. Time
Temperature [Deg F]
585.0
584.0
583.0
582.0
581.0
580.0
579.0
578.0
577.0
576.0
575.0
0
0.4
0.8
1.2
1.6
2
2.4
2.8
3.2
3.6
4
Time [s]

Figure 3.15-9

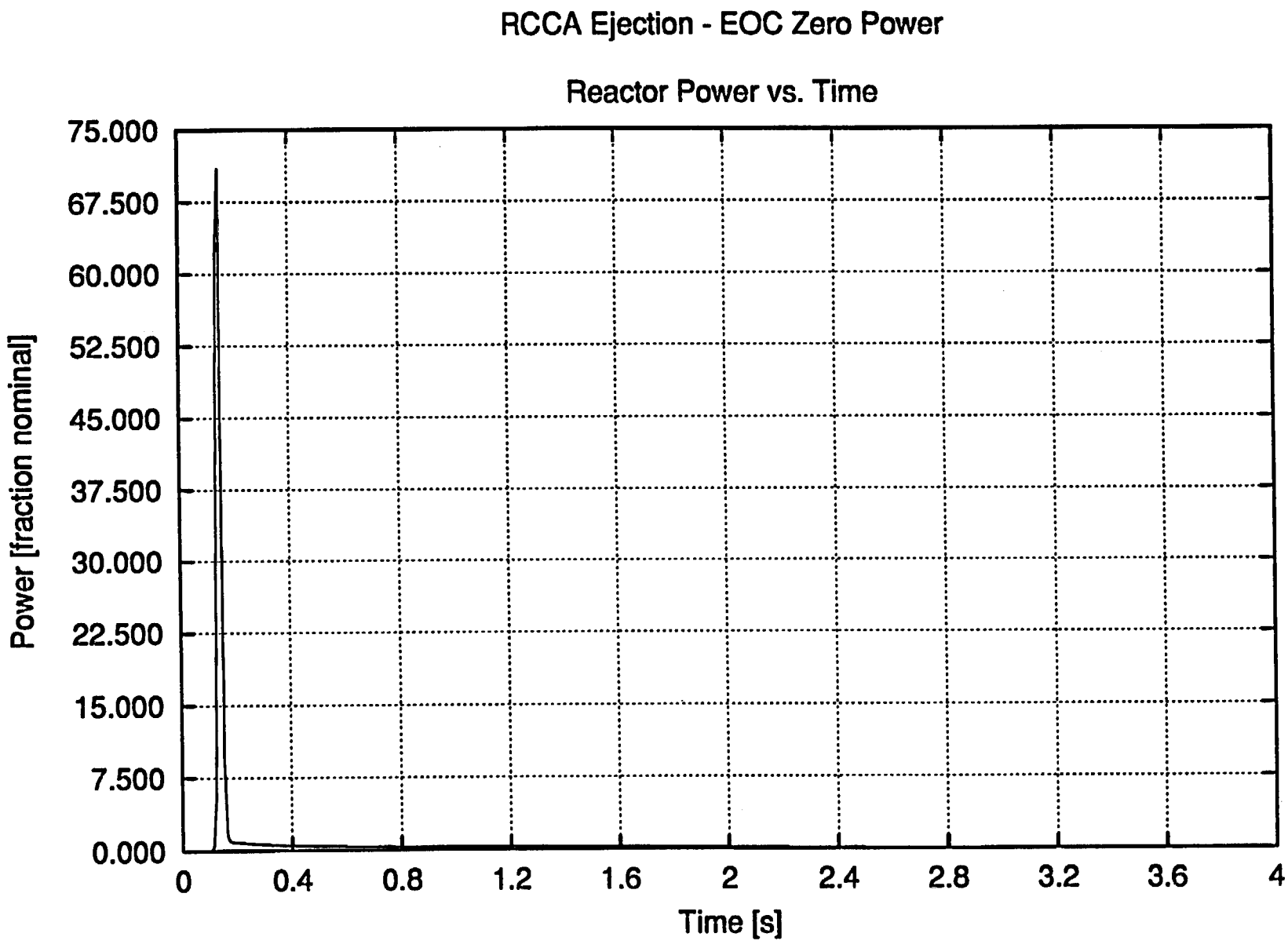

Figure 3.15-10

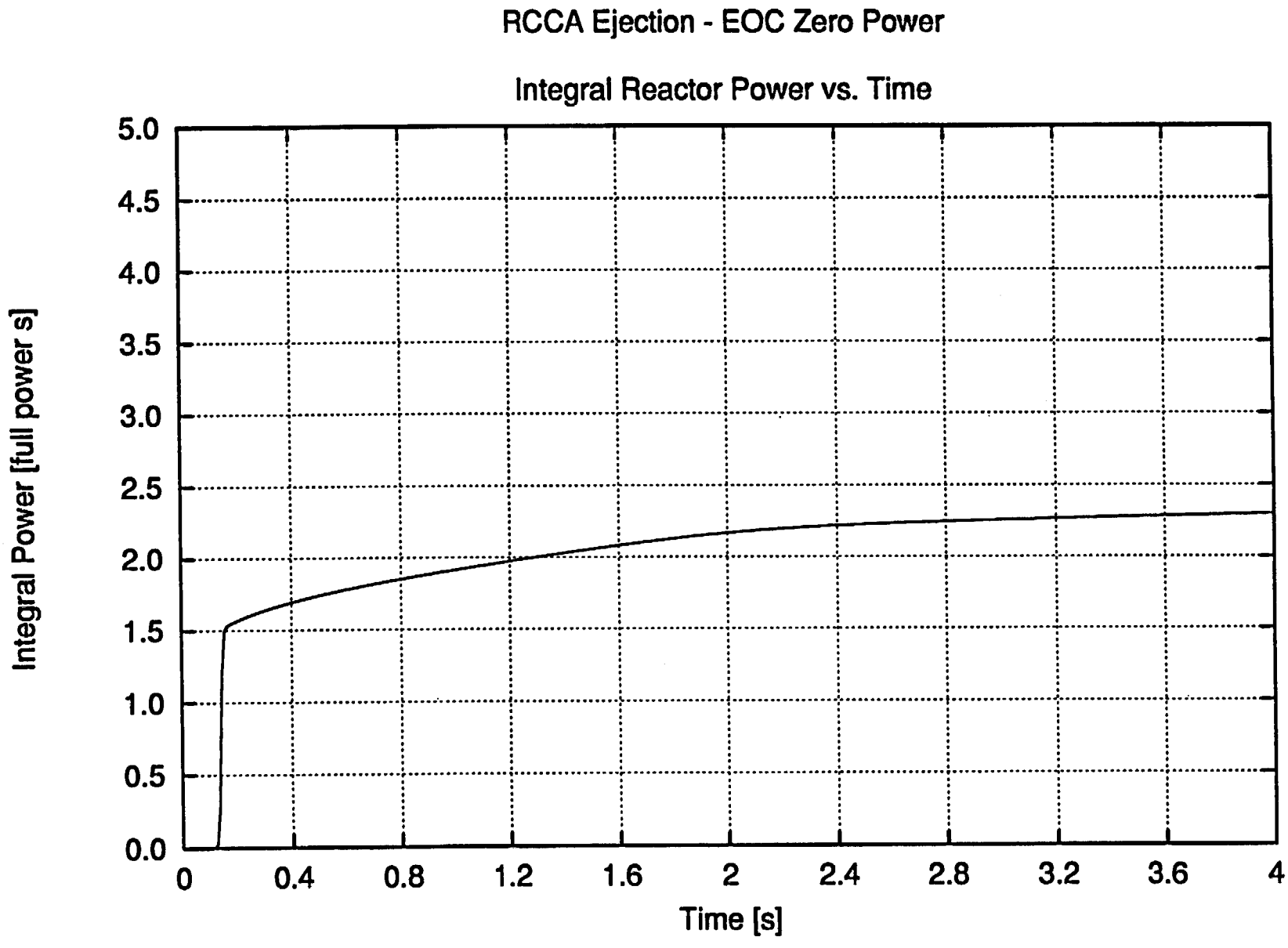

Figure 3.15-11

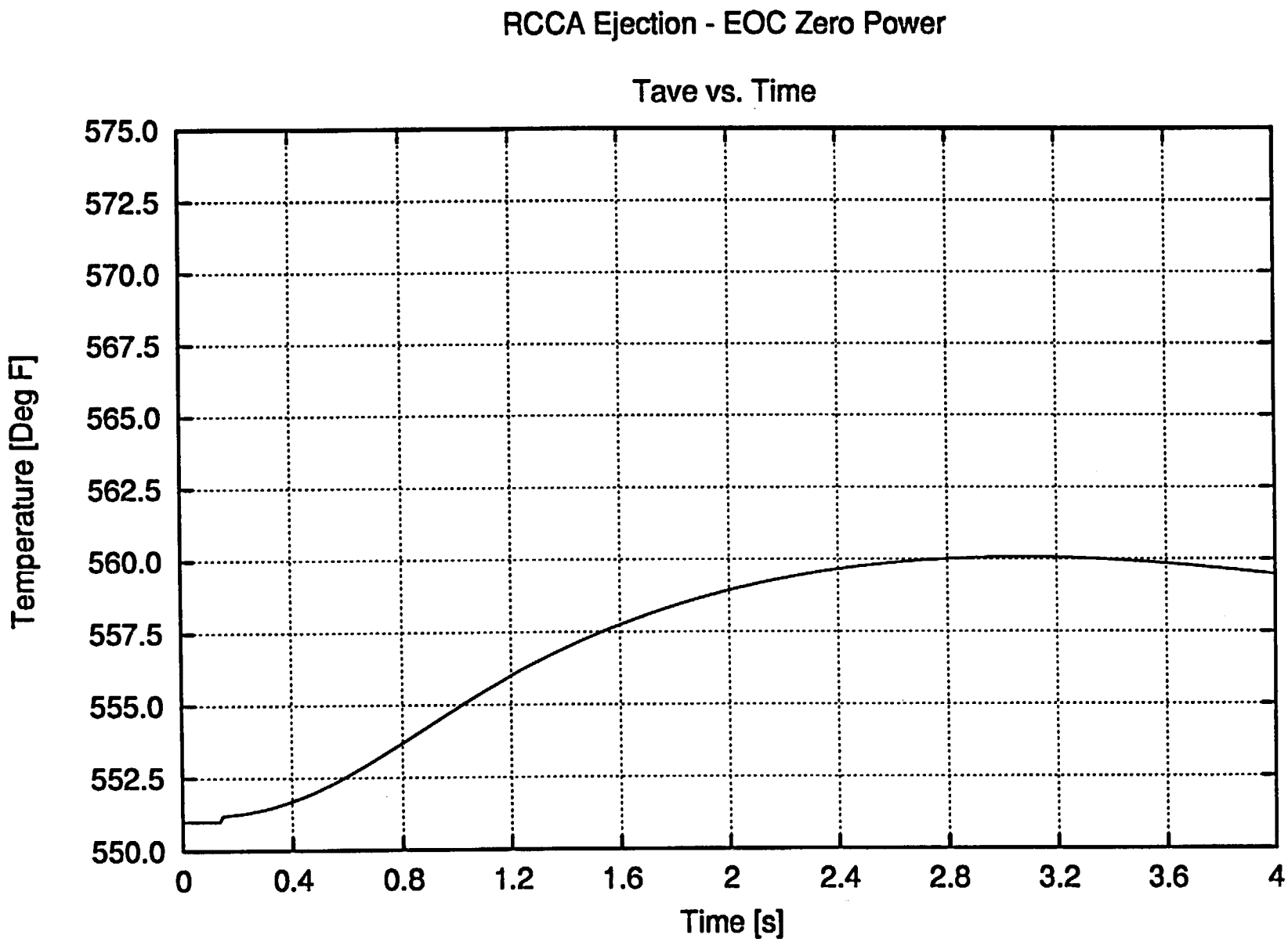

Figure 3.15-12

APPENDIX H

Containment Thermal-Hydraulic Analysis, GOTHIC

GOTHIC (Generation of Thermal-Hydraulic Information for Containments) is a general-purpose thermal-hydraulics computer program for design, licensing, safety and operating analysis of nuclear power plant containment, and other confinement buildings. Applications of GOTHIC include evaluation of containment, containment sub-compartment, and auxiliary building compartment responses to the full spectrum of high-energy line breaks within the design basis envelope. Applications may include pressure and temperature determination, equipment qualification profiles, and inadvertent system initiation, and degradation or failure of engineered safety features.

GOTHIC is a state-of-the-art program that solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow. The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer that cover the entire flow regime from bubbly flow to film/drop flow, as well as single phase flows. The interface models allow for the possibility of thermal nonequilibrium between phases and unequal phase velocities. GOTHIC includes full treatment of the momentum transport terms in multi-dimensional models, with optional models for turbulent shear and turbulent mass and energy diffusion. Conservation equations are solved for three fields:

- Steam/gas mixture
- Continuous liquid
- Liquid droplet

The principal element of a model is a computational volume. GOTHIC features a flexible noding scheme that allows computational volumes to be treated as lumped parameter (single node) or one-, two- or three-dimensional, or any combination of these within a single model. Separate volumes communicate through what are referred to as junctions or flow paths. A separate set of momentum equations is solved for junctions. Mass, momentum and energy can be added or removed at boundary conditions, which are connected to volumes by flow paths.

Solid structures are referred to in GOTHIC as thermal conductors. GOTHIC includes a general model for heat transfer between thermal conductors and the steam/gas mixture or the liquid.

Conduction models in GOTHIC provide that thermal conductors can be modeled by one of the following geometrical shapes:

- Flat plate (wall)
- Cylindrical tube
- Solid rod

Several surface heat transfer models are available for modeling condensation and convention. Boundary conditions that may be specified for a conductor surface include:

- Convection
- Heat flux
- Temperature

GOTHIC includes an extensive set of models for operating equipment. These items, referred to collectively as components, include the following:

- Pumps and fans
- Valves and doors
- Heat exchangers and fan coolers
- Vacuum breakers
- Spray nozzles
- Coolers and heaters
- Volumetric fans (annular fans, deck fans, etc.)
- Hydrogen recombiners (forced and natural convection)
- Ignitors (spark device used to ignite hydrogen burns)

Trip logic is used to control the on/off status of these components. Sensed variables used to activate trips include simulation time, relative time, or the value of certain computed variables such as pressure, pressure differential and temperature.

The GOTHIC code package (H1) includes the following documentation:

- User Manual
- Qualification Report
- Technical Manual

The User Manual describes how to use the GOTHIC code. The Qualification Report provides documentation of facility descriptions, GOTHIC models and predictions from GOTHIC compared to experimental data. The Qualification Report also provides documentation of GOTHIC predictions compared to analytic solutions. The Technical Manual provides a description of the governing equations, constitutive models and solution methods used in GOTHIC.

APPENDIX H (continued)

References

H.1 George, Thomas L., et al., GOTHIC Version 6.0, Containment Analysis Package, December 1997, EPRI RP4444-1

ATTACHMENT 3

Letter from K. H. Weinhauer (NMC)

To

To Document Control Desk (NRC)

Dated

October 12, 2000

Summary of DYNODE-P RETRAN 3-D

Benchmark Report

DYNODE-P RETRAN-3D Benchmark Report Summary

I. Introduction

DYNODE-P has been the main Nuclear Steam Supply System (NSSS) analysis computer code in the Kewaunee Nuclear Power Plant (KNPP) safety analysis methodology since the decision was made in 1978 to independently perform Kewaunee plant transient and accident analyses. At that time the DYNODE-P code was benchmarked to the design basis transient analysis results in Chapter 14 of the KNPP Final Safety Analysis Report to demonstrate the safety analysis capability and knowledge base of Wisconsin Public Service Corporation (WPSC) (reference 1). The DYNODE-P code has been used since 1979 to address plant operating, licensing, and design change issues and to support the reload core design and reload safety evaluation processes.

In the mid-1980's it was evident that RETRAN was fast becoming the industry standard for transient and accident analysis, leading WPSC to adopt it as an additional system analysis computer code. RETRAN was included in Revision 2 to the reload safety evaluation methods topical report (reference 2) to complement the DYNODE-P analyses and to provide independent verification of the DYNODE-P calculations. As RETRAN has continued to develop and improve, it is now beneficial to shift some or all of the system analyses from DYNODE-P to RETRAN. Having both codes operational through the years has provided analysis diversity and enabled the DYNODE-P to RETRAN-3D change to be made at the most appropriate time.

The need for safety analyses to support the steam generator replacement (SGR) project provides an excellent opportunity to make the change in computer codes. The KNPP steam generators are to be replaced in the fall of 2001 with safety analyses being completed in 2000. All of the design basis transients need to be reanalyzed for the SGR project. WPSC also has access to detailed design and thermal hydraulic information for the replacement steam generator components, which facilitated the RETRAN-3D steam generator modeling. In addition, RETRAN-3D became available, has a utility submittal for generic review by the Nuclear Regulatory Commission, and offers potential for improved analytical capability, e.g. the 3D core model, in the future.

WPSC has built up considerable knowledge and experience with RETRAN-3D both with conservative safety analysis calculations and with best-estimate plant response calculations. RETRAN-3D design inputs for the Updated Safety Analysis Report (USAR) (reference 3) Chapter 14 non-LOCA design basis transient events have also been developed.

The DYNODE-P RETRAN-3D Benchmark Report (reference 4) documents the comparisons between the DYNODE-P and RETRAN-3D computer codes for the KNPP non-LOCA design basis accidents with the existing steam generators. The benchmark analyses support the change process for the analysis methods that are being used for the SGR project. It also supports the WPSC topical report on reload safety evaluation methods, WPSRSEM –NP Rev.3 (see attachment 2) which has been revised to document, among other things, the RETRAN-3D methods and results.

II. Methodology and Acceptance Criteria

DYNODE-P inputs to the design basis accidents of Chapter 14 of the USAR are well understood and well documented. The latest plant safety analyses (PSA) of the non-LOCA transients were performed with DYNODE-P in 1998 to support the implementation of the Siemens Power Corp. heavy fuel design and the increased steam generator tube plugging to a 30% level. These 1998 safety analyses are the current analyses of record for the non-LOCA transients and are documented in the KNPP USAR. Detailed inputs for these analyses are documented in PSA DYNODE-P case setup guides and are controlled on the WPSC computer system. Results of the DYNODE–P analyses to which RETRAN-3D results are benchmarked are documented in the USAR and in the 1998 PSA notebook. All case outputs are on microfiche.

For the purpose of benchmarking the DYNODE-P and RETRAN-3D computer codes, the inputs to the two codes are set as close to the same as possible. RETRAN-3D base case and transient case inputs are developed using the DYNODE-P base case and transient case inputs as documented in the PSA case setup guides. The benchmark strategy is to compare the codes with the same inputs so that only the software computational differences are present.

The following acceptance criteria are used for the comparison.

- General trends in parameters plotted in the USAR shall be consistent.
- Parameter trend acceptance criteria: 5% for normalized power or heat flux, 5% of full span for levels, 50 psi for pressures, and 5°F for temperatures. Any difference of greater than these acceptance criteria shall be explainable in terms of code modeling assumptions.
- Key parameters acceptance criteria: 30 psi for steam generator or pressurizer pressure, 0.14 for MDNBR, 4°F for peak clad temperature. Any differences in parameters used for USAR acceptance criteria greater than the Safety Analysis uncertainty values for those parameters (30 psi steam generator or pressurizer pressure, 0.14 MDNBR, 4°F peak clad temperature) shall be explainable in terms of code modeling assumptions.

III. Results

Comparison plots for all of the key parameters for each of the transients and the comparisons of the USAR acceptance criteria are given in the results section of the Benchmark report. This section also presents a discussion of the results and an explanation of the differences observed.

IV. Conclusions

In general comparisons of key parameters between the two codes, RETRAN-3D and DYNODE-P, show that parameter trends are similar and that parameter differences are small. Several transient cases had results that were slightly greater than the acceptance criteria. For each of these cases the differences are explainable in terms of code modeling assumptions. For all other design basis accidents analyzed the RETRAN-3D results agree with DYNODE-P results to within the established acceptance criteria. Therefore it is concluded that RETRAN–3D is an acceptable computer code for

KNPP safety analysis and can be applied to KNPP design basis NON-LOCA transient events. Design basis accident analysis using RETRAN-3D will be conservative ensuring adequate plant safety margins.

V. References

1. Letter from E. W. James (WPSC) to Document Control Desk (NRC) dated January 1979 transmitting Wisconsin Public Service Corporation (WPSC), Kewaunee Nuclear Power Plant (KNPP), topical report entitled "Reload Safety Evaluation Methods for Application to Kewaunee"

2. Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC) dated 11-9-88 transmitting WPSC KNPP topical report WPSRSEM-NP-A Revision 2 entitled "Reload Safety Evaluation Methods for Application to Kewaunee"

3. Kewaunee Updated Safety Analysis Report, Revision 15, dated May, 1999

4. Kewaunee Nuclear Power Plant DYNODE-P RETRAN-3D Benchmark, Edward D. Coen, June 12, 2000

ATTACHMENT 4

Letter from K. H. Weinhauer (NMC)

To

Document Control Desk (NRC)

Dated

October 12, 2000

Summary of GOTHIC, CONTEMPT and COCO

Benchmark Report

GOTHIC Benchmark Report Summary

I. Introduction

The CONTEMPT code has been used to perform the design basis containment analysis for main steamline breaks (MSLB) at the Kewaunee Nuclear Power Plant (KNPP). The Westinghouse COCO code and the CONTEMPT code have been used to perform the design basis containment analysis for the loss of reactor coolant accidents (LOCA).

The GOTHIC code is rapidly becoming the industry standard for performing both inside and outside containment pressure and temperature design basis analyses. The code is being developed by Numerical Applications Incorporated (NAI) with funding by EPRI.

Wisconsin Public Service Corporation (WPSC) approved a plan to develop a GOTHIC containment evaluation model for the Kewaunee Nuclear Power Plant (KNPP). This work was done as part of the replacement steam generator program. The purpose of this work was to demonstrate the modeling capabilities with GOTHIC and to benchmark the GOTHIC model against both the existing CONTEMPT containment analysis model and the revised CONTEMPT and COCO containment analysis models created for the replacement SG program.

The GOTHIC model development program was done in two phases. The purpose of the first phase was to identify differences in modeling methodology between WPSC and Westinghouse and to compare the GOTHIC model with the CONTEMPT model results for the LOCA and MSLB transients. Several modeling differences and discrepancies were discovered and a number of changes were made to create the final LOCA and MSLB containment evaluation models. The purpose of the second phase was to compare results from the revised GOTHIC model to results from the final LOCA and MSLB containment evaluation models.

WCAP-15427 (reference 1) documents the development and benchmark testing of the KNPP GOTHIC containment model. Transient results from the GOTHIC model are compared to results calculated by CONTEMPT and COCO. The benchmark analyses support the SGR project. They also support the WPSC topical report on reload safety evaluation methods, WPSRSEM –NP Rev.3 which has been revised to document, among other things, the GOTHIC methods.

II. Methodology and Acceptance Criteria

For the purpose of benchmarking the GOTHIC computer code, the inputs were set as close to the same as the COCO and CONTEMPT input as possible. The GOTHIC inputs were developed using the CONTEMPT and COCO input as documented in WCAP-15247. The benchmark strategy was to compare the codes with the same inputs so that only the software computational differences were present.

The following acceptance criteria were used for the comparison:

- General trends in key transient parameters (pressure, vapor temperature, sump temperature) should be consistent.
- Peak vapor pressure and temperature values should be within 1 psi and 5 F respectively.

Differences in the output trend or peak values should be readily explainable by differences in the software computational differences.

IV. Results

This section of the Benchmark Report shows the comparison plots for all of the key parameters for both the MSLB and LOCA transients. This section also presents a discussion of the results and an explanation of the differences observed. A drop size sensitivity case is also provided.

IV. Conclusions

In general comparisons of key parameters between the codes show that parameter trends are similar and that parameter differences are small. Several transient cases had results that were slightly greater than the acceptance criteria. For each of these cases the differences are explainable in terms of code modeling assumptions. For all other design basis accidents analyzed, the GOTHIC results agree with COCO or CONTEMPT results to within the established acceptance criteria. Therefore it is concluded that GOTHIC is an acceptable computer code for KNPP safety analysis and can be applied to KNPP containment design basis accident analyses. Design basis accident analysis using GOTHIC will be conservative ensuring adequate plant safety margins.

V. References

1. WCAP-15427, "Development and Qualification of a GOTHIC Containment Evaluation Model for the Kewaunee Nuclear Power Plant," R. Ofstun, July 2000