Westinghouse Non-Proprietary Class 3

WCAP-15996-NP-A, Revision 0
CENPD-282-NP-A, Revision 1

April 2004

# Technical Description Manual

# for the

# CENTS Code

## Volume 4

## April 2004

Westinghouse

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

December 1, 2003

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: FINAL SAFETY EVALUATION FOR TOPICAL REPORT WCAP-15996-P, "TECHNICAL DESCRIPTION MANUAL FOR THE CENTS CODE" (TAC NO. MB6982)

Dear Mr. Bischoff:

On December 13, 2002, and February 19, 2003, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-15996-P, "Technical Description Manual for the CENTS Code" to the staff. On October 6, 2003, an NRC draft safety evaluation (SE) regarding our approval of WCAP-15996-P was provided for your review and comments. By letter dated October 31, 2003, the WOG commented on the draft SE. The staff's disposition of the WOG's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter. The red-line and strikeout version of the SE, along with numbering the lines of the SE, was very helpful in reviewing your comments. In the future, it would be helpful if you numbered the comments in the table in addition to providing the SE line numbers.

The staff has found that WCAP-15996-P is acceptable for referencing in licensing applications for Westinghouse and Combustion Engineering designed pressurized water reactors to the extent specified and under the limitations delineated in the report and in the enclosed SE. The SE defines the basis for acceptance of the report. The dose model portion of the CENTS code is being reviewed separately and will be the subject of a separate SE.

Our acceptance applies only to material provided in the subject report. We do not intend to repeat our review of the acceptable material described in the report. When the report appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this topical report will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that the WOG publish an accepted version of this topical report within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in appendices historical review information, such as questions and accepted responses, draft SE comments, and original report pages that were replaced. The accepted version shall include a "-A" (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions in this letter, that the topical report is acceptable, is invalidated, the WOG and/or the licensees referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Safety Evaluation

cc w/encl:
Mr. J. S. Galembush, Acting Manager
Regulatory Compliance and Plant Licensing
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Pittsburgh, PA 15230-0355

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-15996-P, "TECHNICAL DESCRIPTION MANUAL FOR THE CENTS CODE"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

## 1.0 INTRODUCTION

By letters dated December 13, 2002, and February 19, 2003, the Combustion Engineering Owners Group (now part of the Westinghouse Owners Group [WOG]) submitted Topical Report (TR) WCAP-15996-P, "Technical Description Manual for the CENTS Code" (References 1 and 2), to the NRC staff for review and approval of the transient analysis methodology described therein for licensing applications with regard to both Combustion Engineering- (CE) and Westinghouse-designed pressurized water reactors (PWRs). By letter dated June 13, 2003, the WOG provided responses to the staff's request for additional information (RAI) (Reference 3). WCAP-15996-P is an update of CENPD-282-P-A (Reference 4); the latter was previously reviewed and approved by the staff for application to CE-designed PWRs (Reference 5), and subsequently the staff extended this approval to Westinghouse-designed PWRs (Reference 6). Central to the methodology described and discussed at length in both submittals is the CENTS computer code. This review focuses on, although is not limited to, the changes made to the CENTS code, between the approved version described in CENPD-282-P-A and the improved version described in WCAP-15996-P. The changes were made to more accurately model plant systems and transient behavior of the reactor system. To assist the staff in the review, the WOG prepared a "Roadmap" that identified the changes made to the original TR, CENPD-282-P-A, and the rationale for the changes. This review relies to a great extent, although not exclusively, on the submitted "Roadmap."

TR WCAP-15996-P will, on approval, supercede CENPD-282-P-A; the latter was previously found acceptable by the staff for referencing in licensing actions with respect to the calculation of transient behavior in PWRs. CENPD-282-P-A may continue to be utilized as it was originally approved by the NRC. In particular, the evaluation and approval of the models in the CENTS code, central to the CENPD-282-P-A methodology, are limited to non-loss-of-coolant accident (LOCA) licensing analyses. That is, CENTS is not approved for demonstrating compliance to 10 CFR 50.46 acceptance criteria. It is, however, acceptable when used to model small breaks in the primary system that can be classified as LOCAs for the purpose of demonstrating compliance to non-LOCA regulatory acceptance criteria. For example, CENTS is used to evaluate the dose consequences of steam generator tube rupture and letdown line break events. The qualification of the previous versions of the CENTS code was based on CENTS predictions of startup measurements, operating transients, and comparisons to calculations made with the staff-approved design codes CESEC, CEFLASH-4AS, and RELAP 5/MOD 3. Since model upgrades to the CENTS code are under review, the staff evaluated differences in the predictions of the originally approved code version and those of the upgraded CENTS version described in WCAP-15996-P for the most limiting design basis events. The basis for the approval of WCAP-15996-P is that any variance from previous results due to the model changes precludes exceeding the safety-related limits on which the approved CENPD-282-P-A methodology was based.

## 2.0 REGULATORY EVALUATION

Section 50.34 of Title10 of the *Code of Federal Regulations* (10 CFR) contains requirements for the analysis of abnormal plant operating events by licensees. NUREG-0800, "Review of Safety Analysis Reports for Nuclear Power Plants," provides guidelines to licensees and the staff for evaluating these types of events. Section 50.71 requires licensees to update the final safety analysis report (FSAR) for a given plant periodically. Included in the FSAR are the descriptions of abnormal events and accidents for which a given plant is analyzed. These are typically referred to as Chapter 15 analyses, corresponding to Chapter 15 of NUREG-0800.

The CENTS code is intended to provide analysis capability in the areas of engineering, operations and training. It also is intended to provide evaluation capabilities for transient events, accidents, operator actions, design and scoping studies. Under this review, it is specifically being evaluated for non-LOCA Chapter 15 analyses for PWRs.

## 3.0 SUMMARY OF WCAP-15996-P

The WOG submittals identified the specific changes that have been incorporated into the CENTS code since its previous approval. These modifications can be grouped into two classes: those that do not have an impact on the computed results and those that do affect the computed results.

In the former class are the editorial changes to the descriptions in CENPD-282-P-A with regard to the models of the bubble rise velocity used in the heat transfer coefficient for bubble condensation and the annulus bubble release rate. Both changes bring the text in the TR into conformance with the correct and previously approved coding in the CENTS code. The staff approves these changes. Westinghouse also requested a clarification of the restriction on the use of the CENTS code for application to control element assembly (CEA) ejection licensing analyses. With regard to CEA ejection licensing analyses, the safety evaluation for CENPD-282-P-A states, "... CENTS is not approved for performing CEA ejection licensing analyses." The rationale for this restriction is stated as "Benchmarking for the CEA ejection transient has not been provided...." A sui generis application of the CENTS code to a CEA ejection event has been reviewed and approved by the staff (Reference 7). The staff will continue to entertain, on a case-by-case basis, such analyses for review.

The WOG has added a new dose assessment model to the CENTS computer code that has the capability to calculate offsite dose due to an accident condition. Westinghouse has indicated that this model is essentially the same as the currently employed hand-calculated assessments used to determine dose consequences. The WOG has indicated that the benefit offered by the incorporation of the new dose model is the improved accuracy afforded by performing more exact iodine tracking and release calculations.

NRC review of this new dose assessment model is ongoing. Pending final approval, applicants may use the new model. Until such time as the new CENTS dose assessment model is approved by the NRC, the NRC will review each licensee's dose assessment on a case-by-case basis.

### 3.1 Model Changes

To technically justify those upgrades to the CENTS code that provide new modeling capabilities or provide more detail and accuracy for existing models, and, thereby have an impact on the computed results, the WOG performed benchmark testing. There are four such modifications to the CENTS code considered in this review; review of the fifth, a modification in the dose model, is ongoing as discussed in Section 3.0.

#### 3.1.1 Core Channel Heat Transfer Model Upgrade

The original channel enthalpy model ignores the heat capacity of the fluid, and is based on the assumption that the change in the enthalpy over a computational section is negligible relative to the transport-time constant over the section. The new version of the CENTS code allows for a time-dependent change in the enthalpy in a computational section by taking into account the heat capacity of the liquid. The differential equation for the rate of change of enthalpy in a computational section is solved analytically. Thus, this new option not only takes into account the heat capacity of the fluid, but also precludes any numerical instability that might be introduced through a finite-difference solution for large time steps.

#### 3.1.2 Steam Generator (SG) Tube Nodalization Model with Sectional Coolant Enthalpy

The updated SG model consists of an increase in the number of active-tube nodes per SG. Within each of the active-tube nodes of each SG tube, an internal calculation tracks a detailed temperature profile for the coolant and the tubes. For this purpose, each tube node is divided into multiple subsections; the number of sections in each tube node is specified via input. This more detailed nodalization of the primary side of the SG is provided as an option to support the enhanced tube heat transfer model described above.

#### 3.1.3 Multiple Node Reactor Pressure Vessel (RPV) Downcomer Model

The updated CENTS code contains an option for a more detailed nodalization in the reactor vessel downcomer. This modification, by introducing both axial and azimuthal nodalization, improves the simulation of the asymmetric effects in the loops of the reactor coolant system (RCS).

#### 3.1.4 Detailed Main Feedwater Model

For the previously approved simplified feedwater line model, the feedwater flowrate delivered by the pumps is specified directly by the control system for each SG. The model feeds the indicated flows to the SGs unless a feedwater line break has occurred. In the latter case, the break flow from the feedwater lines is calculated by the homogeneous equilibrium model, or if the flow is choked, by the Henry-Fauske correlation.

The updated CENTS code allows discrete main feedwater (MFW) and auxiliary feedwater (AFW) models. This capability enables accurate, time-dependent transient simulation of the MFW and AFW systems. The models are predicated on the availability of a network of discrete MFW and AFW components and piping through user developed and specified input. Thus, the system network is adaptable to different plant designs.

## 4.0 EVALUATION

Benchmark testing consisted of code comparisons for six events:

1. Main Steamline Break
2. Feedwater Line Break
3. Control Element Assembly Withdrawal from Sub-critical Conditions
4. Control Element Assembly Withdrawal from Hot Zero Power Conditions
5. Reactor Coolant Pump Seized Rotor
6. SG Tube Rupture

These are viewed as the most limiting design basis events in this review.

To test that all the minor code modifications and error corrections made since 1994 have not had a significant net effect, the above six cases were run with the new version of the CENTS code with the upgrade models described above deactivated. No significant variances in the results were observed when compared to the results from the previously approved version. They are judged to preclude exceeding the safety-related limits on which the approval of the CENPD-282-A methodology is based. The staff accepts that the new version of the CENTS code (with the model upgrades described above deactivated) is comparable to the previously accepted version and that it continues to be acceptable to use CENTS in this manner.

The model upgrades in the new version of the code consist, as a whole, of a more realistic description of physical phenomena and a more detailed description of system components. As such, they will lead to more realistic and accurate results. These results may be noticeably different from those obtained with the previously approved version. To demonstrate that the new models lead to correct results, a second set of comparisons for the same general scenarios was made with all the CENTS upgrade models activated. To isolate the effect of the individual upgrades and evaluate their phenomenological behavior, the upgrades were also separately activated.

The new models in the CENTS code induce the following main changes in the results of the six benchmark cases.

### 4.1 Main Steamline Break

The new models cause a slightly more severe and rapid blowdown of the affected SG which results in a deeper drop in the core temperatures. This drop in core temperature has a reactivity worth of +0.0023 Δρ compared to the upgraded version with model changes deactivated. This change in reactivity is far from sufficient to induce a return to power; it is conservative. The staff accepts that the CENTS code with the new models continues to give conservative results for this event.

4.2 Feedwater Line Break

The upgrade models, together and individually, result in greater system flow to the intact SG. This results in lower long-term RCS temperatures and pressures, and less swell into the pressurizer. The regulatory acceptance criterion for this event, with a limiting single failure, is that the peak RCS pressure must be less than 120 percent of the RCS design pressure. The staff accepts that the CENTS code with the new models continues to give conservative results with respect to this criterion for this event.

4.3 Control Element Assembly Withdrawal from Sub-critical Conditions

The only upgrade that has a significant effect on the results in this event is the channel heat transfer model. The improved modeling of the core fluid heat capacity reduces the positive moderator temperature reactivity feedback, and, thereby, lowers the peak power from ~119 percent to ~105 percent of nominal. The improved modeling reduces the code conservatism, however, it is physically based and is acceptable to the staff.

4.4 Control Element Assembly Withdrawal from Hot Zero Power Conditions

As in the previous event, the only upgrade that has a significant effect on the results is the channel heat transfer model. The improved modeling of the core fluid heat capacity reduces the positive moderator temperature reactivity feedback, and, thereby, lowers the peak power from ~106 percent to ~101 percent of nominal. It is acceptable to the staff as described above.

4.5 Reactor Coolant Pump Seized Rotor

The comparison of results between the upgraded CENTS code with models deactivated and activated shows good agreement, and, thereby, precludes exceeding safety-related limits on which the approved CENPD-282-P-A methodology was based. It is therefore acceptable to the staff.

4.6 SG Tube Rupture

The SG tube rupture event is a penetration of the barrier between the RCS and the main steam system due to the failure of a steam generator U-tube. The integrity of the barrier between the RCS and the main steam system is significant from the radiological release standpoint. The limiting event considered is a double-ended rupture of a SG tube with concurrent loss of alternating current (AC) power. Both phenomenologically and quantitatively, the comparison of thermal-hydraulic plant response parameters between the CENTS code with and without the upgraded model is excellent. The safety-related consequences for this event are mainly predicated on the dose model. The dose model portion of the CENTS code is presently under review by the staff as discussed in Section 3.0.

Although this review is based solely on the results of the above comparisons of benchmark calculations, the WOG has submitted results of a comparative analysis of a main steamline break event and a feedwater line break computed with CENTS with upgrades and RELAP5/MOD3 (Reference 8). The agreement is good, and, furthermore, gives some insight into the effectiveness of the model upgrades in the CENTS code.

## 5.0 CONCLUSIONS

The staff has reviewed TR WCAP-15996-P and the supporting documentation sent in response to the staff's request for additional information. On the basis of this review, the staff approves the transient methodology described in WCAP-15996-P for referencing in licensing actions with respect to the calculation of transient behavior in PWRs designed by CE and by Westinghouse, subject to the limitations stated below. These limitations were placed on the approval of the CENPD-282-P-A methodology and apply to WCAP-15996-P methodology, approved herein. This does not include approval of the CENTS code dose model. The dose model portion of the CENTS Code is under NRC review at this time. The CENTS code dose model will be evaluated separately.

1. CENTS departure from nucleate boiling ratio (DNBR) analysis: The CENTS DNBR calculation for determining overall trends in thermal margin should not be used for licensing analyses. The DNBR licensing analyses should be performed with the presently approved CE DNBR methods.

2. Limitation to CE and Westinghouse Type Plants: The application of CENTS is limited to PWRs of CE and Westinghouse design.

3. LOCA and Severe Accident Analysis: Adequate benchmarking of the CENTS LOCA and severe accident capabilities has not been provided. Consequently, CENTS should not be used for performing LOCA or severe accident licensing analyses. CENTS is not approved for demonstrating compliance to 10 CFR 50.46 criteria. It is; however, acceptable for use in modeling small breaks in the primary system that can be classified as LOCAs for the purpose of demonstrating compliance to non-LOCA regulatory acceptance criteria.

4. Three-Dimensional Core Neutronics: Benchmarking for the CENTS three-dimensional core neutronics capability has not been provided. Consequently, licensing applications of CENTS should be based on a point kinetics model.

5. CEA Ejection Analyses: This review does not give general approval for the application of CENTS simulations of a CEA ejection transient for licensing analyses. The staff will consider and review such requests on a case-by-case basis. This portion of the CENTS code remains under review at this time.

## 6.0 REFERENCES

1. Letter, G. S. Pavis (CEOG) to USNRC Document Control Desk, "Submittal of Combustion Engineering Owners Group Reports: WCAP-15996-P, and WCAP-15996-NP, entitled 'Technical Description Manual for the CENTS Code'," CEOG-02-256, December 13, 2002.

2. Letter, G. S. Pavis (CEOG) to USNRC Document Control Desk, "Submittal of Combustion Engineering Owners Group Reports: WCAP-15996-P, Volume 4 and WCAP-15996-NP, Volume 4 entitled 'Technical Description Manual for the CENTS Code'," WOG-03-76, February 19, 2003.

3. Letter, R. H. Bryan (WOG) to USNRC Document Control Desk, "Response to Request for Additional Information Related to the Westinghouse CENTS Topical Report (WCAP-15996-P)," WOG-03-305, June 13, 2003.

4. CENPD-282-A, Rev. 0, "Technical Description Manual for the CENTS Code."

5. Letter, M. J. Virgilio (NRC) to S. A. Toelle (ABB-CE), "Acceptance for Referencing of Licensing Topical Report CENPD-282-P, 'Technical Manual for the CENTS Code'," March 17, 1994.

6. Letter, R. C. Jones (NRC) to S. A. Toelle (ABB-CE), "Acceptance for Referencing of Licensing Topical Report CENPD-282-P, Vol. 4, 'Technical Manual for the CENTS Code'," February 24, 1995.

7. Letter, L. R. Wharton (NRC) to G. R. Overbeck (APS), "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Issuance of Amendments re: Various Administrative Controls (TAC Nos. ME1668, MB1669, and MB1670)," October 15, 2001.

8. RELAP5/MOD3 Code Manual, NUREG/CR5535, INEL-95/0174, Vol. 1.

Attachment: Disposition of Comments on Draft SE

Principal Contributor: Yuri Orechwa

Date: December 1, 2003

DISPOSITION OF COMMENTS RECEIVED FROM WOG ON
THE DRAFT SAFETY EVALUATION OF WCAP-15996

**This table identifies and tracks the resolution of comments received from the WOG on October 31, 2003.**

| Comment Number | Comment Description<br>A = comment accepted; PA = comment partially accepted;<br>R = comment rejected |
|---|---|
| 1 | Suggest incorporating reference for submittal of CENTS Volume 4.<br>A |
| 2 | Suggest incorporating reference for submittal of CENTS RAI responses.<br>A |
| 3 | Update reference numbers due to previous incorporation of additional references.<br>A |
| 4 | Request for clarification of the intent of the word "supercede" on Line #29. A |
| 5 | Suggest incorporating a more specific statement regarding CENTS usage with respect to analysis of LOCAs.<br>A |
| 6 | Editorial<br>R – staff prefers original wording. |
| 7 | Suggest incorporating a more complete statement regarding staff-approved design codes against which CENTS was originally benchmarked. Also, suggest deleting reference.<br>A |
| 8 | Editorial<br>A |
| 9 | Suggest incorporating a more specific statement regarding CENTS usage.<br>PA |
| 10 | Suggest incorporating specific reference to the sui generis application the SE refers to and also update reference number.<br>PA |
| 11 | Suggest incorporating a more specific constraint clarification regarding use of CENTS for a portion of a CEA ejection evaluation, as indicated.<br>R – This comment involves an item that is still under review. |
| 12 | Suggest incorporating a more specific statement regarding the dose model assessment.<br>A |
| 13 | Suggest incorporating a more specific section title.<br>A |
| 14 | Suggest changing section title to eliminate proprietary information.<br>A |
| 15 | Suggest deleting text to eliminate proprietary information.<br>A |
| 16 | Change "channel" to "tube" to clarify that it is steam generator and not core heat transfer that is being discussed.<br>A |

Attachment

DISPOSITION OF COMMENTS RECEIVED FROM WOG ON
THE DRAFT SAFETY EVALUATION OF WCAP-15996

| Comment Number | Comment Description<br>A = comment accepted; PA = comment partially accepted;<br>R = comment rejected |
|---|---|
| 17 | If SE statement is a limitation, it would seem to follow that it should be included in the Conclusion section where the NRC lists limitations.<br>A |
| 18 | Suggest incorporating a more specific clarification regarding use of updated CENTS version in a manner which replicates the previously accepted version of the code.<br>A |
| 19 | Editorial<br>PA |
| 20 | Suggest incorporating a more specific statement regarding the dose model assessment.<br>PA |
| 21 | Update reference number<br>A |
| 22 | Suggest Incorporating a more specific statement regarding the dose model assessment.<br>PA |
| 23 | Suggest incorporating a more specific statement regarding CENTS usage with respect to analysis of LOCAs.<br>A |
| 24 | Suggest incorporating a more specific constraint clarification regarding use of CENTS for a portion of a CEA ejection evaluation, as indicated.<br>R – This item involves an issue still under review. |
| 25 | Correction of submittal reference.<br>A |
| 26 | Incorporation of submittal reference for CENTS Volume 4.<br>A |
| 27 | Incorporation of submittal reference for CENTS RAI responses.<br>A |
| 28 | Update reference number.<br>A |
| 29 | Update reference number.<br>A |
| 30 | Update reference number and correct document number.<br>A |
| 31 | Incorporate reference for CEA ejection event sui generis application.<br>A |
| 32 | Incorporate reference for RELAP5/MOD3.<br>A |

DISPOSITION OF COMMENTS RECEIVED FROM WOG ON
THE DRAFT SAFETY EVALUATION OF WCAP-15996

| Comment Number | Comment Description<br>A = comment accepted; PA = comment partially accepted;<br>R = comment rejected |
|---|---|
| 33 | Incorporate reference for CENPD-190-A, CEA ejection methodology.<br>A |
| 34 | Delete incorrect reference to CEFLASH-4AS.<br>A |
| 35 | RELAP5/MOD3 reference is now Reference 8.<br>A |

Westinghouse Non-Proprietary Class 3

WCAP-15996-NP-A, Revision 0
CENPD-282-NP-A, Revision 1

# Technical Description Manual for the CENTS CODE

## VOLUME 4

April 2004

# ABSTRACT

CENTS is an interactive, faster-than-real-time computer code for simulation of the Nuclear Steam Supply System and related systems. CENTS is used to evaluate PWR behavior for normal and abnormal conditions including accidents.

WCAP-15996-P, Volumes 1 and 2 and 3 describe the various CENTS models, the input and output variables, and the data base and data dictionary.

WCAP-15996-P, Volume 4 provides a detailed comprehensive set of comparison benchmark cases. CENTS predictions used to support the originally approved code version (reported in CENPD-282-P, Volumes 3 and 4) are compared to the upgraded CENTS version (described in WCAP-15996-P, Rev. 0, Volumes 1 to 3) for an assortment of benchmark cases. These comparisons isolate the effects of code modifications made since CENTS original approval in 1994. The benchmark cases were run several times with various code options enabled/disabled so that the effect of each major change could be isolated. The results of the comparison show when the newly added model options are disabled by input, the results of the upgraded CENTS version are essentially unchanged from those of its predecessor. This demonstrates that the minor improvements and error corrections made to the original CENTS version have not had a significant net effect.

In some cases, the results of the upgraded CENTS version change noticeably when the newly added model options are enabled (typically via input). Enough information is provided to establish that the new models provide correct results.

This comparison demonstrates the effect of all the minor code modifications and error corrections made since 1994 which did not require NRC review and approval prior to their implementation. A second comparison for the same general scenarios is made when all the appropriate CENTS upgrade models are activated (i.e., those for which NRC review and approval is being sought). This comparison shows the impact of all upgrades collectively, with discussion of major impacts provided by individual models also included. Two scenarios, a Main Steam Line Break and Feedwater Line Break, allow comparison of the response for the CENTS detailed Main Feedwater model with a plant specific RELAP5.3 feedwater model. Lastly, a test case is provided for the Feedwater Line break which alters the event evaluation methodology that has been previously used for CE designed plants. This change in evaluation methodology simply uses CENTS input parameter specification to place the steam generator feedring at its actual physical elevation in the downcomer node and to deactivate a model which

forced full tube heat transfer until a low liquid inventory was reached in the secondary side of the steam generator. The previous evaluation methodology was used to generate limiting peak RCS pressures. However, realistic modeling of the actual plant configuration (via CENTS input parameters) still generates nearly equivalent peak RCS pressures, while providing more realistic simulation of the long term pressurizer level response which is important for determining the time available to plant operators to prevent pressurizer overfill.

## Table of Contents

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## List of Figures

# 1.0 Introduction

WCAP-15996-P, Volume 4 presents the results of a series of benchmark cases that evaluate the changes made to the CENTS computer code since it was approved for use by the NRC. WCAP-15996-P, Volume 4 supplements the information previously provided in CENPD-282-P-A, Volumes 3 and 4.

A number of modifications have been made to the CENTS code since approval in 1994. Most of the changes either do not affect CENTS computational results at all (e.g. changes to program input/output functionality) or have no effect on safety-related (i.e., SAR Chapter 15 licensing) transient analyses. A few changes were made to correct specific code errors or to bring the code into precise compliance with the description provided in CENPD-282-P-A. These changes did not have a significant effect on calculated results.

In addition, a few major upgrades were made to the code. These modifications either provide new modeling capabilities or provide more detail and accuracy for existing models. In some cases these modifications do result in noticeable differences in results. The changes for which benchmark cases are provided are:

(a) Core heat transfer model upgrade (Volume 1, Section 3.3.5)
(b) Four node SG tube model with sectional coolant enthalpy (Volume 1, Section 5.3.1)
(c) Multiple node Pressure Vessel (PV) downcomer model (Volume 1, Section 4.1 & Volume 2, Section 7.2.1)
(d) Dose model (Volume 1, Section 5.8).
(e) Detailed Main feedwater model (Volume 1, Section 5.5)

Benchmark testing was performed as follows:

- Six test cases were run to benchmark the upgraded CENTS version to the original version documented in CENPD-282-P-A, Volumes 3 and 4. These cases were run with all of the model upgrades deactivated via input. These events are the most severe design basis events and provide a comprehensive set of benchmark cases. The events presented are:
  1. Main Steam Line Break
  2. Feedwater Line Break
  3. Reactor Coolant Pump Seized Rotor
  4. Steam Generator Tube Rupture
  5. Control Element Assembly Withdrawal from Sub-critical conditions
  6. Control Element Assembly Withdrawal from hot zero power conditions

Sections 2.2 and 3.2 present the results of these cases. No significant differences in results were seen for these cases.

- The same six cases were run with upgrades (a), (b), (c) and (d) activated. The upgrades were individually enabled, if necessary, to isolate the effect of each upgrade separately. Sections 2.3 and 3.3 present the results of these cases.

- Two additional benchmark cases were run to evaluate the effect of the newly added detailed Main feedwater model. The cases are a Main Steam Line Break and Feedline Break from full power initial conditions. For these cases, the CENTS calculation of main and auxiliary feedwater flow and enthalpy are compared to the results of RELAP5.3. Section 3.4 presents the results of these cases.

- Section 3.4.2 presents a third case which includes a different treatment of the break enthalpy for the Feedline Break event. The analysis of the feedwater line break event discussed in CENPD-282-P-A, Volume 3, Sections 2.3.2 and 3.3.2, assumed that the fluid exiting the steam generator through the break consisted of only liquid water. This was accomplished by artificially locating (via input specification) the feedring at the top elevation of the tubesheet. This assumption is unnecessarily overly conservative for the analysis of plants which do not have economizer steam generators. For the analysis of the feedline break model presented in Section 3.4.2, the feedring is simply modeled (via input specification) to be at its actual physical elevation so that a steam blowdown commences when the steam generator downcomer water level drops below the feedring elevation. This change did not require a modification to CENTS algorithms since the feedring elevation is an input parameter.

- These are typical cases. Details of the analyses vary from plant to plant, due to differences in design and licensing history. The intent of the choice of cases was to provide good bases for comparison of the CENTS versions.

Two plant designs were chosen for the study based on:

(a) Availability of CENTS base decks,
(b) Availability of representative event case files,
(c) Availability of RELAP base decks for use in benchmarking the detailed CENTS Main Feedwater model,
(d) The plants do not have SG economizers; therefore, the feedring location is an applicable issue.

The comparison of the event results of the upgraded CENTS version with all model upgrades activated to the original CENTS version provides verification of the improvements. The results support the use of the upgraded code version with all model upgrades activated for the performance of licensing analysis for non-LOCA plant transients.

# 2.0 Benchmark Comparisons for Plant D, 3390 MWt

## 2.1 Discussion

Verification of the upgraded CENTS version, with all its model improvements, includes two (2) comparisons of plant behavior as predicted by CENTS. Section 2.2 provides a benchmark of the upgraded CENTS version (all model changes deactivated) to the original SER approved CENTS version (i.e., CENPD-282-P-A). Section 2.3 tests the results of various accident scenarios with the upgraded CENTS version (model changes activated).

### 2.1.1 Plant Description

Plant D is a Combustion Engineering PWR Design initially licensed to operate at a core thermal power output of 3390 MWt, which is the power level used in the benchmark analyses (with 2% uncertainty applied).

Plant Arrangement

The containment structure houses a nuclear steam supply system (NSSS), consisting of a reactor, two (2) steam generators, four (4) reactor coolant pumps, a pressurizer, and some reactor auxiliaries which do not normally require access during power operation.

Reactor

The reactor is a pressurized light water cooled and moderated design fueled by slightly enriched uranium dioxide ($UO_2$). The $UO_2$ is in the form of pellets and is contained in zirconium alloy (e.g., Zircaloy-4, ZIRLO™) tubes fitted with welded end caps. These fuel rods are arranged into fuel assemblies each consisting of 236 fuel rods arranged on a 16x16 rod square matrix. Each fuel assembly contains five (5) guide tubes for the insertion of control element assemblies (CEAs), if called for by management.

The reactor is controlled by a combination of chemical shim and solid absorbers. An integral fuel burnable absorber may be mixed into selected fuel rods, as appropriate. Five (5) CEA fingers of boron carbide ($B_4C$) in the form of pellets form a single CEA (i.e., four tubes in a square matrix plus a central tube). The individual CEA fingers are connected together at the top by a yoke, which is in turn connected to the control element drive mechanism (CRDM) extension shaft. Each CEA is aligned and is inserted into a guide tube in the fuel assembly.

Chemical shim is provided by boric acid dissolved in the reactor coolant system (RCS) coolant water. The concentration of boric acid is maintained and controlled as required by the chemical and volume control system.

Reactor Coolant System

The RCS consists of two (2) closed heat transfer loops in parallel with the reactor vessel. Each loop, moving outward from the core exit, contains one (1) hot leg, one (1) steam generator, two (2) coolant pump suction cold legs, two (2) reactor coolant pumps to circulate the coolant, and two (2) discharge cold legs returning the coolant to the reactor vessel. A pressurizer vessel is connected to one of the coolant hot legs. The RCS was originally designed to operate at a power level of 3390 MWt and to produce steam at 900 psia.

RCS pressure is maintained by electrical heater elements in the lower region of the pressurizer and by pressurizer spray nozzles in the upper steam region of the pressurizer. Over-pressure protection is provided by spring-loaded safety valves connected to the pressurizer. Safety valve discharge is released under water in the quench tank where the steam discharged is condensed.

The steam generators are of the vertical shell and U-tube design. Steam is generated on the shell side of the steam generator and flows upward through moisture separators.

The reactor coolant is circulated by four (4) electric motor driven, single suction, centrifugal pumps. Each pump is equipped with an anti-reverse mechanism to prevent reverse rotation of any pump that has power removed.

CENTS nodalization of the Plant D NSSS is shown in Figure 2.1.1

Engineered Safety Features

An engineered safety features system is provided to localize, control, mitigate and terminate postulated accidents which could potentially release radioactive fission products from the fuel rods.

The engineered safety features systems include the high pressure safety injection pumps (HPSIs), the low pressure safety injection pumps (LPSIs), the safety injection tanks (SIT), and the auxiliary feedwater system (AFWS).

For each unit, four (4) SITs are provided, each connected to one of the four (4) cold legs. In the event of a Loss-of-Coolant Accident (LOCA), the SIT borated water is forced into the RCS by the expansion of the nitrogen gas contained in the tank. The water from the SITs adequately cools the entire core. In addition, borated water is injected into the RCS by two (2) LPSIs and two (2) HPSIs taking suction from the refueling water storage tank (RWST). For maximum reliability, the design capacity from the combined operation of one (1) HPSI and one (1) LPSI provides adequate injection flow for any LOCA. In the event of an accident at least one (1) HPSI and one (1) LPSI will receive power from the emergency power sources even if normal power is lost and one of the emergency diesel generators fails to start.

The AFWS consists of three (3) pumps (two motor driven and one steam driven) which are capable of cooling the RCS in the event that normal feedwater is lost.

Reactor Protection System

Reactor parameters are maintained within acceptable limits by the inherent self-controlling characteristics of the reactor, by CEA positioning, by the boron content of the reactor coolant and by operating procedures. The function of the reactor protection system (RPS) is to initiate reactor shutdown when any reactor parameter approaches the preset limits for safe operation.

The RPS is divided into four (4) channels, each receiving trip signals from separate sensors when the parameter reaches preset levels. If any two (2) of these four (4) channels receives coincident signals, the power to the magnetic jack CRDMs is interrupted, allowing the CEAs to drop into the core to shut down the reactor. The RPS is completely independent of, and separate from, the normal plant operation control systems.

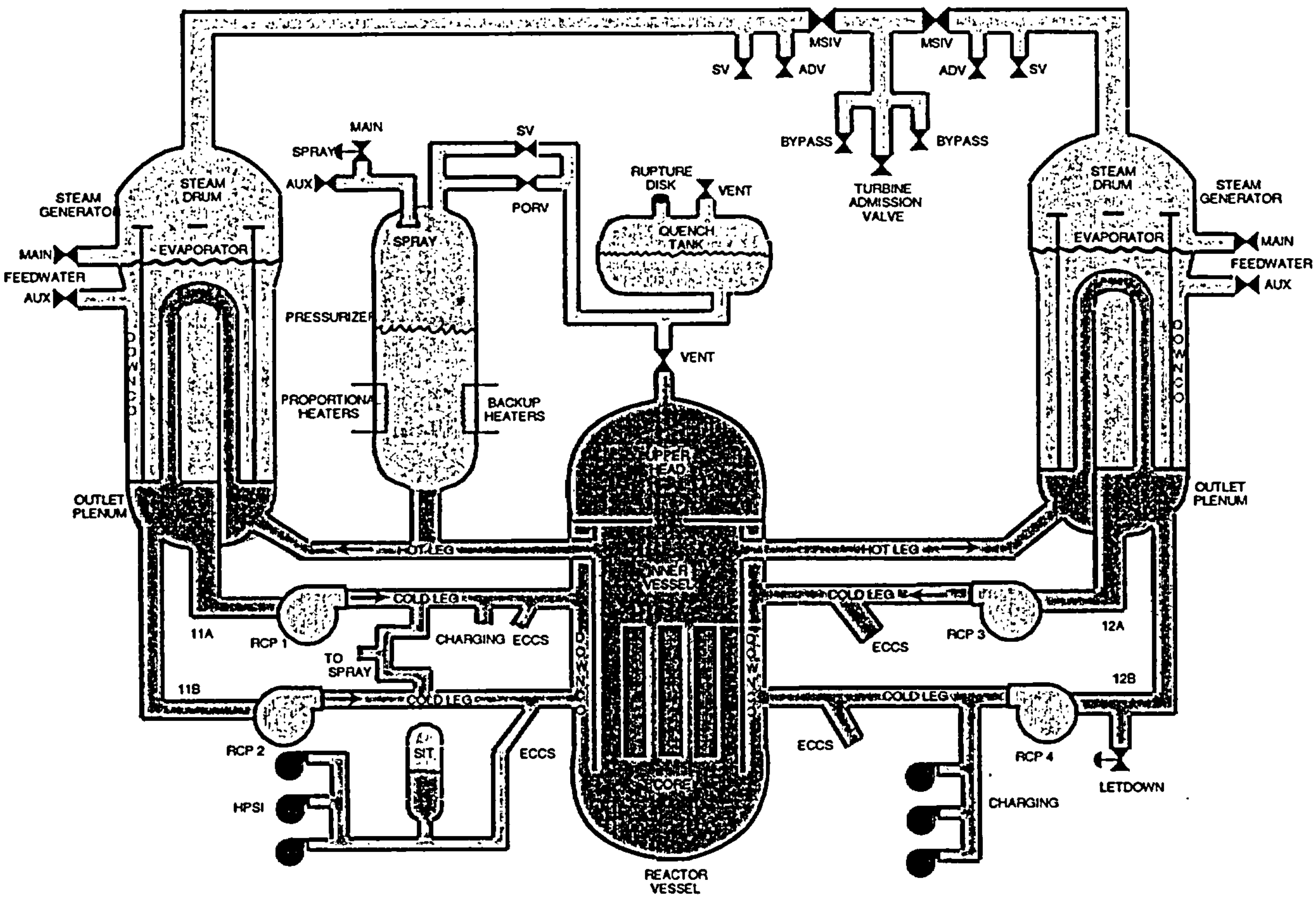
The RPS includes the digital Core Protection Calculators (CPC). The CPCs provide an online calculation of the approach to the Specified Acceptable Fuel Design Limits (SAFDLs). The calculation is compensated for the dynamic effects which would occur during plant Design Basis Events (DBEs). A reactor trip is generated if the CPCs predict that the thermal margin conditions of the core warrant it.

Operating Restrictions

Normal plant operation is restricted to the parameter limits included in the Technical Specifications (TSs). The limits are imposed to ensure that plant operation remains in compliance with the limits assumed in the safety analysis.

The TSs include restrictions such as the minimum number of safety injection pumps which must be operable, the slowest allowed response times of the containment isolation features, and restrictions on important process parameters such as RCS pressure and temperature and maximum allowed CEA insertion.

Figure 2.1.1
CENTS Model of a Two Loop Pressurizer Water Reactor



## 2.2 Comparison of Upgraded CENTS Code Version (Upgrades De-activated) to Original Version

Four (4) benchmarking events were analyzed for Plant D. The events are:

1. Main Steam Line Break (MSLB)
2. Feedwater Line Break (FWLB)
3. Control Element Assembly Withdrawal (CEAW) from Subcritical Conditions
4. Control Element Assembly Withdrawal (CEAW) from Critical Conditions

For each event, a comparison is made of the results from the upgraded (models de-activated) with the original (i.e., CENPD-282-P-A) CENTS versions.
The results for each of these cases show no significant differences. A detailed discussion of each of the event benchmarks is provided in the following sections.

### 2.2.1 Main Steam Line Break (MSLB)

Discussion of Event

A postulated Main Steam Line Break (MSLB) is analyzed in accordance with Section 15.1.5 of the Standard Review Plan, Reference 3. This analysis is performed to demonstrate that sufficient sources of negative reactivity are available to offset the insertion of positive reactivity added during the transient by the rapid cooldown of the moderator.

A single MSLB case was simulated using the two CENTS code versions, as discussed above. The case assumes that a double-ended guillotine break occurs in the main steam line inside the containment building from hot full power initial conditions. This case assumes a loss of AC power at the start of the event, so that the reactor coolant pumps coast down. Also commensurate with a loss of AC power, feedwater flow is assumed to ramp linearly to zero in 20 seconds and feedwater enthalpy ramps to 80 BTU/LBM in the same period.

Table 2.2.1.A contains a list of the important assumptions for this case. These assumptions were used in setting up the case specific CENTS input data.

The cooldown of the RCS continues until the affected steam generator empties. The MSLB case is run to the time at which the core is sub-critical and negative reactivity is being added.

Analysis Methods

A number of analysis assumptions affect the calculation of the maximum post-trip reactivity. These assumptions are the same as those used in the benchmark cases performed in CENPD-282-P-A, Volume 3. The CENTS code includes several options to ensure that the simulation of a MSLB provides conservative results. Important conservative analysis assumptions used include:

a) End of cycle core conditions are assumed. All appropriate uncertainties are applied to the reactivity components which are input to the point kinetics model.
b) The maximum worth CEA is assumed to stick in the fully withdrawn position after trip.
c) The MSLB is assumed to be initiated by an instantaneous double-ended rupture of one steam line upstream of the main steam isolation valve (MSIV).
d) Saturated steam blowdown with no moisture carryover from the steam generators is assumed. This assumption results in the maximum rate of energy removal from the RCS.
e) MSLB analyses for CE designed PWRs includes the reactivity feedback effect due to the local changes in the moderator density near the location of the assumed stuck CEA. Localized moderator feedback effects are important during a MSLB event if a return to power occurs.
f) For the MSLB analysis it is conservative to assume that the steam generator connected to the ruptured steam line maintains full heat transfer area until it is essentially empty. This modeling maximizes the rate of heat removal from the RCS and thus inserts the greatest positive reactivity due to moderator feedback.
g) Asymmetric heat removal during a MSLB event causes unequal cold leg temperatures at the reactor vessel inlets from the two steam generator loops. Unequal reactor inlet temperatures, in combination with incomplete mixing of coolant in the reactor vessel downcomer and lower plenum, results in a temperature distribution at the core inlet. Basing moderator reactivity on the core cold edge moderator density includes the effect of this temperature distribution.

During the early portions of the MSLB transient, from event initiation until about 20 seconds after reactor trip, the reactivity insertion due to moderator feedback is based on core average moderator conditions. After trip, moderator reactivity feedback is based on coolant conditions on the colder side of the inlet plenum.

Results

Table 2.2.1.B provides a comparison of the sequence of events for the MSLB with a Loss of AC power. Figures 2.2.1.A through 2.2.1.Q provide comparisons of important parameters as calculated by the original CENTS version and the upgraded version (with model changes de-activated). These plots show agreement consistent with expectations.

The transient trend is in general the same between the two code versions. The predicted termperatures of the affected loop and of the reactor vessel are essentially the same. Although the intact loop for the upgraded version shows higher temperatures (Figure 2.2.1.F, the intact loop minimum temperature is approximately 29 °F higher for the upgraded version than for the original version), the effect on the overall transient is small. The change that affects the intact side results is the correction of an error in the implementation of the steam generator heat transfer correlation for reverse heat transfer, which also affects the intact steam generator pressure (Figure 2.2.1.H).

The change in total reactivity is small. The upgraded version yields a change in reactivity of +0.0012 delta rho compared to the original version.

Note, Figures 2.2.1.B, C, O & P show a minor spike for the original CENTS version case at approximately 560 seconds. This spike is due to a discontinuity in a CENTS water properties table which has been corrected for the upgraded CENTS version, in which the spike no longer occurs.

## Table 2.2.1.A

**Important Assumptions**
**Steam Line Break**

| Parameter | Value |
|---|---|
| Break Size | 7.876 $Ft^2$ |
| Core Power | 102% of 3390 MWt |
| Core Inlet Temperature | 560 °F |
| Pressurizer Pressure | 2300 PSIA |
| Pressurizer Level | 19.675 Feet |
| Core Burnup | End of Cycle |
| Steam Generator Pressure | 960 PSIA |
| Steam Generator level | 38.55 Feet |
| Scram Worth | 7.88 % $\Delta\rho$ |
| Number of Operable High Pressure Safety Injection Pumps | 1 |
| All Control Systems | Manual Mode<br>(Note Pressurizer Pressure & Level Control lost on Loss of AC Power) |
| Loss of Offsite Power | Offsite Power is lost at commencement of the event. |

## Table 2.2.1.B

## Sequence of Events
## Steam Line Break

| Time(Sec) | | Event | Value | |
|---|---|---|---|---|
| Upgraded Version (Upgrades deactivated) | Original Version | | Upgraded Version (Upgrades deactivated) | Original Version |
| | | Main steam line break $Ft^2$ | | |
| | | Loss of AC power | | |
| | | Reactor Trip on Steam Generator low pressure, PSIA | | |
| | | Main steam isolation signal, PSIA | | |
| | | Main steam isolation valves fully closed | | |
| | | Safety injection actuation, PSIA | | |
| | | Safety injection flow begins | | |
| | | Affected steam generator empties (downcomer empty) | | |
| | | Minimum mixed core inlet temperature is reached, °F | | |
| | | Maximum Reactivity is reached, delta rho | | |

## 2.2.2 Feedwater Line Break

Discussion of Event

A feedwater system line break (FWLB) may produce a total loss of normal feedwater and a blowdown of one steam generator. If normal sources of AC electrical power were lost, there would also be a simultaneous loss of primary coolant flow, turbine load, pressurizer pressure and level control and steam bypass control. The result of these events would be a rapid decrease in the heat transfer capability of both steam generators and eventually the complete loss of the heat transfer capability of one steam generator.

The NSSS is protected during this transient by the pressurizer safety valves and the following reactor trips:

- Low steam generator level
- Low steam generator pressure
- High pressurizer pressure
- High containment pressure

Depending on the initial conditions, any one of these trips may terminate the transient. The NSSS is also protected by MSIVs, feedwater line check valves, steam generator safety valves and the auxiliary feedwater system, which serves to protect the integrity of the secondary heat sink following reactor trip.

The regulatory acceptance criterion for this event, with a limiting single failure, is that the peak RCS pressure must be less than 120% of RCS design pressure.

The FWLB case assumes the limiting break size, a relatively small break (.150 $ft^2$), occurs in the feedline to one of the steam generators, downstream of the feedwater check valve.

Table 2.2.2.A contains a listing of the important assumptions for this case.

## Analysis Methods

The CENTS includes the models necessary to implement the FWLB methodology presented in Section 15.E of CESSAR, Reference 1. These features include:

- Liquid Blowdown - The fluid exiting the steam generator through the break is modeled as consisting only of liquid water. Historically, this has been thought to conservatively underestimate the heat removal capability of this blowdown fluid. Steam generator blowdown is assumed to be frictionless critical flow as calculated by the Henry-Fauske correlation. See Note below.
- Steam Generator Heat Transfer Ramp-down - In order to conservatively model the RCS over-pressurization, the effective heat transfer area of the steam generator is assumed to decrease linearly from the design value to zero as the steam generator liquid mass decreases from a selected value to zero. See Note below.

No credit is taken for a low water level trip condition in the ruptured steam generator until the generator is emptied of water. This conservatively delays the time of reactor trip, prolonging the RCS heatup and over-pressurization.

No credit is taken for the high containment pressure trip.

*Note: Section 3.4.2 demonstrates that with adjustment to the limiting break size, the non-physical modeling described above is no longer necessary. The limiting case in that section shows that with realistic modeling assumptions, the resulting blowdown still produces peak pressures essentially the same as those resulting from a pure liquid blowdown and the SG heat transfer ramp down discussed above.*

## Results

Table 2.2.2.B provides a comparison of the sequence of events for the FWLB. Figures 2.2.2.A through 2.2.2.Q provide comparisons of important parameters as calculated by the original and upgraded (models deactivated) CENTS versions.

The comparison of results between the upgraded and original CENTS versions shows that peak RCS and steam generator pressures increase by small amounts as does short term pressurizer level. All key parameters follow essentially the same trend. The differences after ~400 seconds are a consequence of a slight delay in the system response that is first noticed in the Pressurizer pressure behavior at ~200 seconds (Figure 2.2.2.Q). This is a result of a slowdown of the steam generator blowdown in the original CENTS version (Figure 2.2.2.G) caused by flow oscillations between the two steam generators between ~150 and ~320 seconds. The pressurizer pressure in the original CENTS version, when emergency feedwater (EFW) is activated (~385 seconds), is ~33 psi lower than the pressure in the upgraded CENTS version when EFW is activated (~375 seconds), and it stays lower for the rest of the transient (Figure 2.2.2.Q). The upgraded CENTS version does not show the flow oscillations between the steam generators. The difference between the two models is the enhanced steam line model (Volume 1 Section 5.6). The double peak in pressurizer pressure (Figure 2.2.2.D) is due to the pressurizer safety valves opening which causes a short duration drop in pressure; however, insurge from the RCS continues and turns pressure around for the second and limiting peak.

## Table 2.2.2.A

### Important Assumptions
### Feed Line Break

| Parameter | Value |
|---|---|
| Break Size | 0.150 $Ft^2$ |
| Core Power | 102% of 3390 MWt |
| Core Inlet Temperature | 560 °F |
| Pressurizer Pressure | 2150 PSIA |
| Steam Generator Pressure | 929 PSIA |
| All Control Systems | Manual Mode |
| Loss of Offsite Power | Power is lost at the time a reactor trip signal is generated |

## Table 2.2.2.B

**Sequence of Events**
**Feed Line Break**

| Time(Sec) | | | Value | |
|---|---|---|---|---|
| Upgraded Version (Upgrades deactivated) | Original Version | Event | Upgraded Version (Upgrades deactivated) | Original Version |
| | | Feed line break, $Ft^2$ | | |
| | | Affected steam generator empties | | |
| | | Reactor trip condition occurs (High Pressurizer Pressure Trip), PSIA | | |
| | | Loss of AC power | | |
| | | Peak RCS Pressure, PSIA | | |
| | | Peak Pressurizer Liquid Volume (1st peak, 0 – 120 seconds), $Ft^3$ | | |
| | | Peak Steam Generator Pressure**, PSIA | | |
| | | Minimum Intact Steam Generator Liquid Mass, Lbm | | |
| | | Peak Pressurizer Liquid Volume (2nd peak, 120 – 1800 seconds), $Ft^3$ | | |

** Peak Steam Generator pressure shown above is for the scenario that emphasizes peak RCS pressure. Different initial conditions would be required to emphasize peak steam generator pressure for a Feedwater line break.

## 2.2.3 Control Element Assembly Withdrawal (CEAW) from Sub-Critical Conditions

### Discussion of Event

CEA withdrawal (CEAW) from sub-critical conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase together with corresponding increases in reactor coolant temperatures and RCS pressure. The withdrawal motion of CEAs also produces a time dependent redistribution of core power. These transient variations in core thermal parameters result in the system's approach to the specified fuel design limits, thereby requiring the protective action of the RPS.

Table 2.2.3.A contains a listing of the important assumptions for this case. These assumptions were used in setting up the input data for both versions of CENTS. The Doppler and scram reactivity tables were used.

### Analysis Methods

CENTS includes the models necessary to implement the uncontrolled CEAW from a sub-critical condition methodology presented in Section 15.4.1 of CESSAR, Reference 1.

### Results

Table 2.2.3.B provides a comparison of the sequence of events for the CEAW from a sub-critical condition. Plots of key parameters are provided in Figures 2.2.3.A through K.
Figures 2.2.3.A and 2.2.3.G & H provide comparisons of the total core power fraction and total reactivity, respectively, for this event. There is essentially no difference between the original and the upgraded (with model changes deactivated) CENTS versions.

## Table 2.2.3.A

## Important Assumptions
## CEA Withdrawal from Sub-critical Conditions

| Parameter | Value |
|---|---|
| Core Power | 11.66E-8% of 3390 MWt |
| Shutdown Margin | -.01%$\Delta\rho$ |
| Reactivity Addition Rate | 2.7 x $10^{-4}$ %$\Delta\rho$/sec |
| Core Inlet Temperature | 560 °F |
| Reactor Trip Set point | 4% of Rated Core Power |
| Reactor Trip System Response Time | 0.4 seconds |

## Table 2.2.3.B

### Sequence of Events
### CEA Withdrawal from Sub-critical Conditions

| Time(Sec) | | | Value | |
|---|---|---|---|---|
| Upgraded Version (Upgrades deactivated) | Original Version | Event | Upgraded Version (Upgrades deactivated) | Original Version |
| | | Withdrawal of CEA's Initiating Event | | |
| | | Reactor Trip Set point, % of rated core power | | |
| | | Trip Breakers Open (CEA withdrawal stopped) | | |
| | | CEAs begin to drop | | |
| | | Maximum Core Power, % of rated core power | | |
| | | Maximum Core average Heat Flux, % of full Power heat flux | | |

## 2.2.4 Control Element Assembly Withdrawal (CEAW) from Critical Condition

CEAW from low power conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase together with corresponding increases in reactor coolant temperatures and RCS pressure. The withdrawal motion of CEAs also produces a time dependent redistribution of core power. These transient variations in core thermal parameters result in the system's approach to the specified fuel design limits, thereby requiring the protective action of the RPS.

Table 2.2.4.A contains a listing of the important assumptions for this case. These assumptions were used in setting up the input data for both versions of CENTS. It is noted that moderator reactivity feedback effects were removed (due to minimized contribution based on the trip set point). Uniform Doppler and scram reactivity tables were used.

### Analysis Methods

CENTS includes the models necessary to implement the uncontrolled CEAW from a low power condition methodology presented in Section 15.4.1 of CESSAR, Reference 1.

### Results

Table 2.2.4.B provides a comparison of the sequence of events for the CEAW from a critical condition. Plots of key parameters are provided in Figures 2.2.4.A through K. Figures 2.2.4.A and H provide comparisons for this event of the total core power fraction and total reactivity, respectively. The table and figures show that there is negligible difference between the original and the upgraded (with upgrades deactivated) CENTS version.

## Table 2.2.4.A

### Important Assumptions
### CEA Withdrawal from a Critical Condition

| Parameter | Value |
|---|---|
| Core Power | 1.0 E-5% of 3390 MWt |
| Shutdown Margin | $-.102 \times 10^{-4}$ %Δρ |
| Reactivity Addition Rate | $2.0 \times 10^{-4}$ %Δρ/sec |
| Core Inlet Temperature | 560 °F |
| Reactor Trip Set point | 40% of Rated Core Power |
| Reactor Trip System Response Time | 0.4 seconds |

## Table 2.2.4.B

### Sequence of Events
### CEA Withdrawal from a Critical Condition

| Time(Sec) | | | Value | |
|---|---|---|---|---|
| Upgraded Version (Upgrades deactivated) | Original Version | Event | Upgraded Version (Upgrades deactivated) | Original Version |
| | | Withdrawal of CEA's Initiating Event | | |
| | | Reactor Trip Set point, % of rated core power | | |
| | | Trip Breakers Open (CEA withdrawal stopped) | | |
| | | CEAs begin to drop | | |
| | | Maximum Core Power, % of rated core power | | |
| | | Maximum Core average Heat Flux, % of full Power heat flux | | |

## 2.3 Comparison of Upgraded Code Version — Model Changes Activated

The same four (4) events that are analyzed in Section 2.2 were reassessed for each event using the upgraded CENTS version with the various model changes activated. The results from these benchmarks are compared to the cases presented in Section 2.2 as Upgraded Version (models de-activated). This comparison provides the impact that each model improvement had on the key results. The results of "turning on" each upgraded model is discussed in the following sections. The case results are cumulative. First, the upgraded Core Heat Transfer model (described in WCAP-15996-P, Volume I, Section 3.3.5) is activated, then the four (4) node steam generator model is added with the detailed enthalpy calculation (described in WCAP-15996-P, Volume I, Section 5.3.1), finally the detailed pressure vessel nodalization is activated (described in WCAP-15996-P, Volume II, Section 7.2.1). Main Feedwater model inputs have not been established and, therefore, were not assessed for Plant D. However, benchmarking of the Feedwater model was conducted for Plant E and is discussed in Section 3.4.

### 2.3.1 Main Steam Line Break (MSLB)

Discussion of Event – See Section 2.2.1 & Table 2.2.1A

Analysis Methods – See Section 2.2.1

Results

Table 2.3.1 provides a comparison of the sequence of events for the MSLB with a Loss of AC power. Figures 2.2.1.A through 2.2.1.Q provide comparisons of important parameters as calculated by the upgraded CENTS version (with model changes activated) and the upgraded version (with model changes de-activated).

The transient trend as each model is activated remains the same. The differences in timing and magnitude of change for various parameters remain relatively small. Activating the Core Heat Transfer (CHT) model has essentially no effect on the results in this transient. However, the four (4) node steam generator model does have an effect. In general, this model change causes the tube heat transfer to be enhanced for both the affected and intact steam generators. This causes the blowdown of the affected steam generator to proceed more rapidly, and in turn, the drop in RCS temperatures also occurs sooner and with slightly greater magnitude. The core inlet temperature reaches a minimum of ~354°F at approximately ~160 seconds vs. ~357.4°F in ~169 seconds with this model inactive.

The detailed pressure vessel downcomer model allows a radial distribution of downcomer temperatures (and density differences) which promotes slightly greater core flow, but with less flow to the intact steam generator and more to the affected loop which results in a moderately more severe cooldown. With this model activated, minimum core inlet temperature drops to ~340°F at approximately ~164 seconds.

Thus overall, the model improvements provide for a slightly more severe and rapid blowdown of the affected steam generator with resulting deeper drop in temperatures at the core. There is still no return to power, but there is an effect on core reactivity of which is summarized in Table 2.3.1. The change in total reactivity is small. The upgraded version with upgrades activated yields a change in reactivity of +0.0023 delta rho compared to the upgraded version with model changes de-activated.

## Table 2.3.1
## Sequence of Events
## Steam Line Break – Upgraded Version

| Time(Sec) | | | | Event | Value | | | |
|---|---|---|---|---|---|---|---|---|
| Upgrades deactivated | Upgrades Activated (Sequentially) | | | | Upgrades deactivated | Upgrades Activated (Sequentially) | | |
| | CHT | 4 SG Nodes | Detailed PV | | | CHT | 4 SG Nodes | Detailed PV |
| | | | | Main steam line break $Ft^2$ | | | | |
| | | | | Loss of AC power | | | | |
| | | | | Reactor Trip on Steam Generator low pressure, PSIA | | | | |
| | | | | Main steam isolation signal, PSIA | | | | |
| | | | | Main steam isolation valves fully closed | | | | |
| | | | | Safety injection actuation, PSIA | | | | |
| | | | | Safety injection flow begins | | | | |
| | | | | Affected steam generator empties (Downcomer empty) | | | | |
| | | | | Minimum mixed core inlet temperature is reached, °F | | | | |
| | | | | Maximum Reactivity is reached, delta rho | | | | |

### 2.3.2 Feedwater Line Break

Discussion of Event – See Section 2.2.2 & Table 2.2.2A

Analysis Methods – See Section 2.2.2

Results

Table 2.3.2 provides a comparison of the sequence of events and key limiting parameters for the FWLB. Figures 2.2.2.A through 2.2.2.Q provide comparisons of important parameters.

All key parameters follow essentially the same trend. The core heat transfer model upgrade has no appreciable effect on this transient. The four (4) node steam generator tube model and its associated detailed heat transfer model does have an effect. It allows for greater steam generator heat transfer during the blowdown of the affected steam generator. This means that heatup rate of the RCS is slightly less severe at the time that the steam generator empties and the RCS high pressure reactor trip signal occurs. With a lower heatup rate, the insurge rate to the pressurizer is lower and the overshoot in pressure after the pressurizer safety valves lift is also less, with a peak pressure of ~2687 psia. This same phenomenon results in a lower peak pressurizer level, for both the short term (0-120 seconds) and the long term. Also, the greater heat transfer to the steam generators causes secondary side peak pressure to be higher.

The detailed pressure vessel model also has an effect, though minor early in the event. Peak pressurizer pressure increases, from ~2687 to ~2701psia. The temperature distribution in the pressure vessel downcomer, allowed by the detailed modeling, causes higher natural circulation flow rates to the intact loop in the longer term portion of the event. This promotes greater steam flow and a slower long term buildup in secondary inventory (Figure 2.2.2.L). Overall, this greater system flow to the intact steam generator results in lower long term RCS temperatures, pressure and less swell into the pressurizer (Figures 2.2.2.P. & 2.2.2.Q).

## Table 2.3.2

**Sequence of Events**
**Feed Line Break – Upgraded Version**

| Time(Sec) | | | | Event | Value | | | |
|---|---|---|---|---|---|---|---|---|
| Upgrades deactivated | Upgrades Activated (Sequentially) | | | | Upgrades deactivated | Upgrades Activated (Sequentially) | | |
| | CHT | 4 SG Nodes | Detailed PV | | | CHT | 4 SG Nodes | Detailed PV |
| | | | | Feed line break, $Ft^2$ | | | | |
| | | | | Affected steam generator empties | | | | |
| | | | | High pressurizer pressure trip condition, PSIA | | | | |
| | | | | Loss of AC power | | | | |
| | | | | Peak RCS Pressure, PSIA | | | | |
| | | | | Peak Pressurizer Liquid Volume (1st peak, 0 – 120 seconds), $Ft^3$ | | | | |
| | | | | Peak Steam Generator Pressure**, PSIA | | | | |
| | | | | Minimum Intact Steam Generator Liquid Mass, Lbm | | | | |
| | | | | Peak Pressurizer Liquid Volume (2nd peak, 120 – 1800 seconds), $Ft^3$ | | | | |

** Peak Steam Generator pressure shown above is for the scenario that emphasizes peak RCS pressure. Different initial conditions would be required to emphasize peak steam generator pressure for a Feedwater line break.

## 2.3.3 Control Element Assembly Withdrawal (CEAW) from Sub-Critical Conditions

Discussion of Event - See Section 2.2.3 & Table 2.2.3A

Analysis Methods - See Section 2.2.3

Results

Table 2.3.3 provides the comparison of effects of the various CENTS upgrades. Figures 2.2.3.A through K provide a comparison for key NSSS and steam generator parameters showing the effects of the combined upgrades. The only upgrade which has a significant effect upon the results is the core heat transfer upgrade. The reduction in peak power from ~119% to ~105% is due to the improved modeling of the core fluid heat capacity, which reduces the moderator temperature feedback (Figure 2.2.3.I).

## Table 2.3.3

### Sequence of Events
### CEA Withdrawal, from Sub-critical Conditions – Upgraded Version

| Time(Sec) | | | | Event | Value | | | |
|---|---|---|---|---|---|---|---|---|
| Upgrades deactivated | Upgrades Activated (Sequentially) | | | | Upgrades deactivated | Upgrades Activated (Sequentially) | | |
| | CHT | 4 SG Nodes | Detailed PV | | | CHT | 4 SG Nodes | Detailed PV |
| | | | | Withdrawal of CEA's Initiating Event | | | | |
| | | | | Reactor Trip Set point, % of rated core power | | | | |
| | | | | Trip Breakers Open (CEA withdrawal stopped) | | | | |
| | | | | CEAs begin to drop | | | | |
| | | | | Maximum Core Power, % of rated core power | | | | |
| | | | | Maximum Core average Heat Flux, % of full Power heat flux | | | | |

## 2.3.4 Control Element Assembly Withdrawal (CEAW) from Hot Zero Power Conditions

Discussion of Event - See Section 2.2.3 & Table 2.2.3A

Analysis Methods - See Section 2.2.3

Results

Table 2.3.4 provides the comparison of effects of the various CENTS upgrades. Figures 2.2.4.A through O provide a comparison for key NSSS and steam generator parameters showing the effects of the combined upgrades. As with the sub-critical CEAW event, the only upgrade which has a significant effect upon the results is the core heat transfer upgrade. The reduction in peak power from ~106% to ~101% is due to the improved modeling of the core fluid heat capacity, which reduces the positive reactivity insertion due to moderator temperature feedback.

## Table 2.3.4.

## Sequence of Events
## CEA Withdrawal, from Hot Zero Power Conditions – Upgraded Version

| Time(Sec) | | | | Event | Value | | | |
|---|---|---|---|---|---|---|---|---|
| Upgrades deactivated | Upgrades Activated (Sequentially) | | | | Upgrades deactivated | Upgrades Activated (Sequentially) | | |
| | CHT | 4 SG Nodes | Detailed PV | | | CHT | 4 SG Nodes | Detailed PV |
| | | | | Withdrawal of CEA's Initiating Event | | | | |
| | | | | Reactor Trip Set point, % of rated core power | | | | |
| | | | | Trip Breakers Open (CEA withdrawal stopped) | | | | |
| | | | | CEAs begin to drop | | | | |
| | | | | Maximum Core Power, % of rated core power | | | | |
| | | | | Maximum Core average Heat Flux, % of full Power heat flux | | | | |

Figure 2.2.1.A

Reactor Core Power

Steam Line Break Event for Plant D

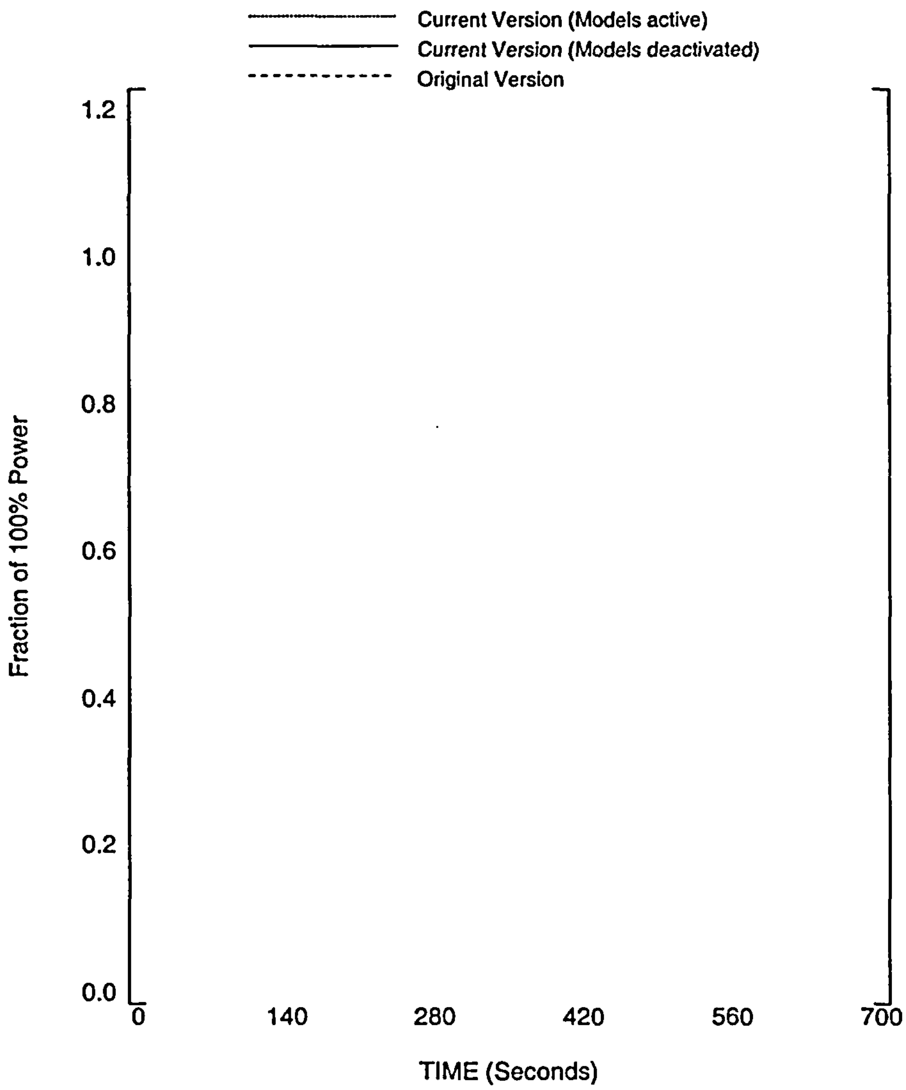


Figure 2.2.1.B

Reactor Core Heat Flux

Steam Line Break Event for Plant D

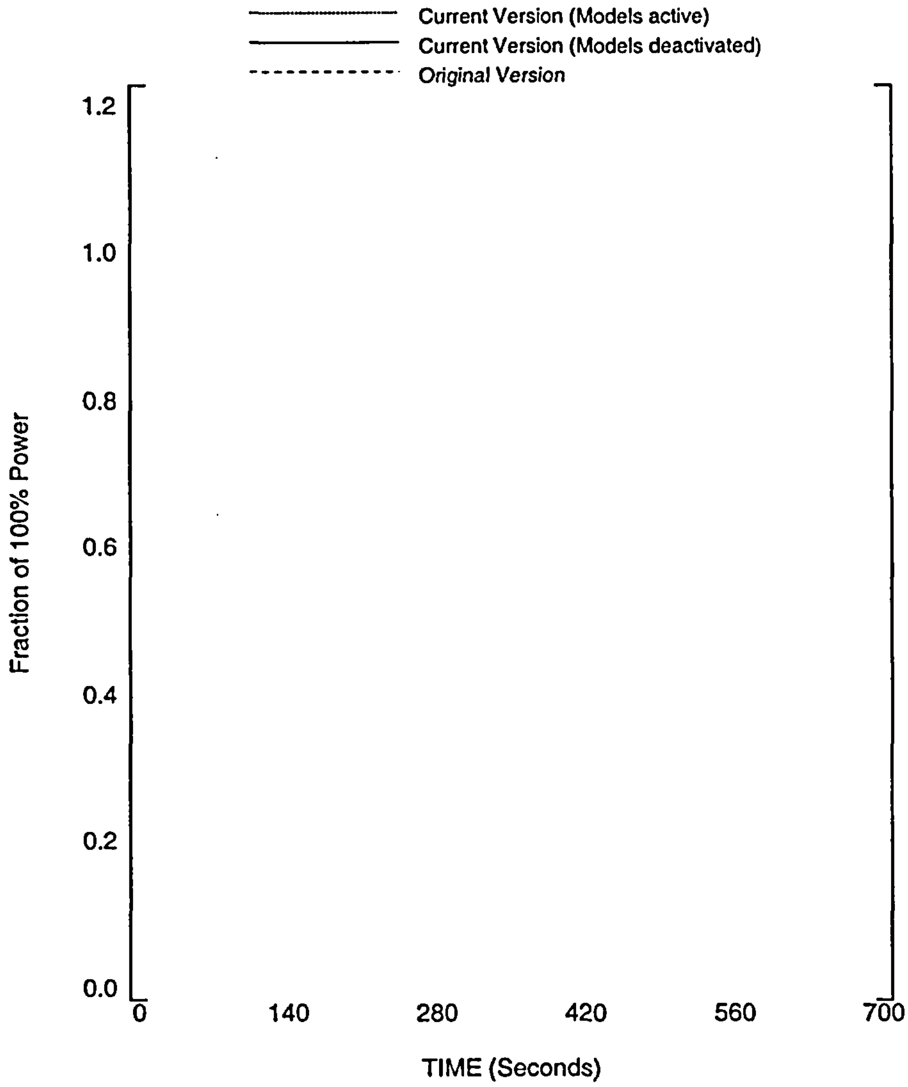


Figure 2.2.1.C

Reactor Coolant System Pressure

Steam Line Break Event for Plant D

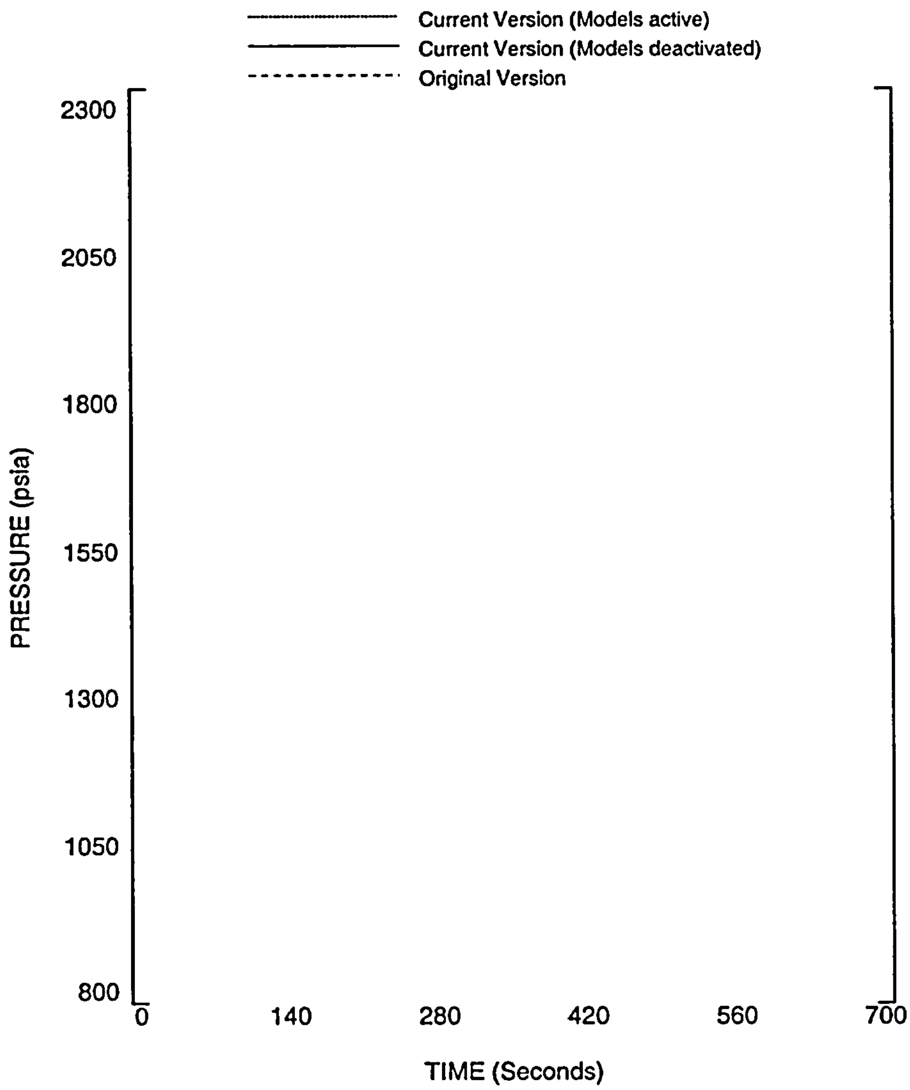


Figure 2.2.1.D

Pressurizer Pressure

Steam Line Break Event for Plant D

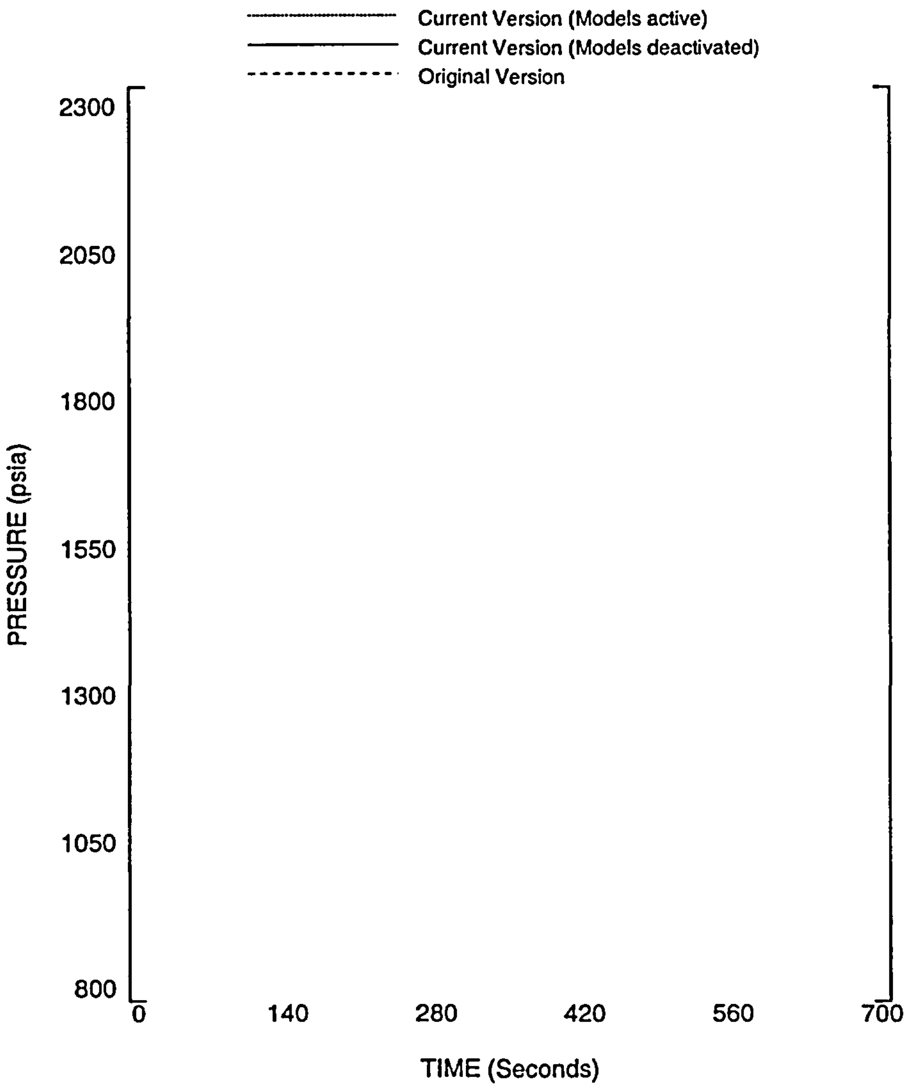


Figure 2.2.1.E

Cold Leg Temperature, Affected Loop

Steam Line Break Event for Plant D

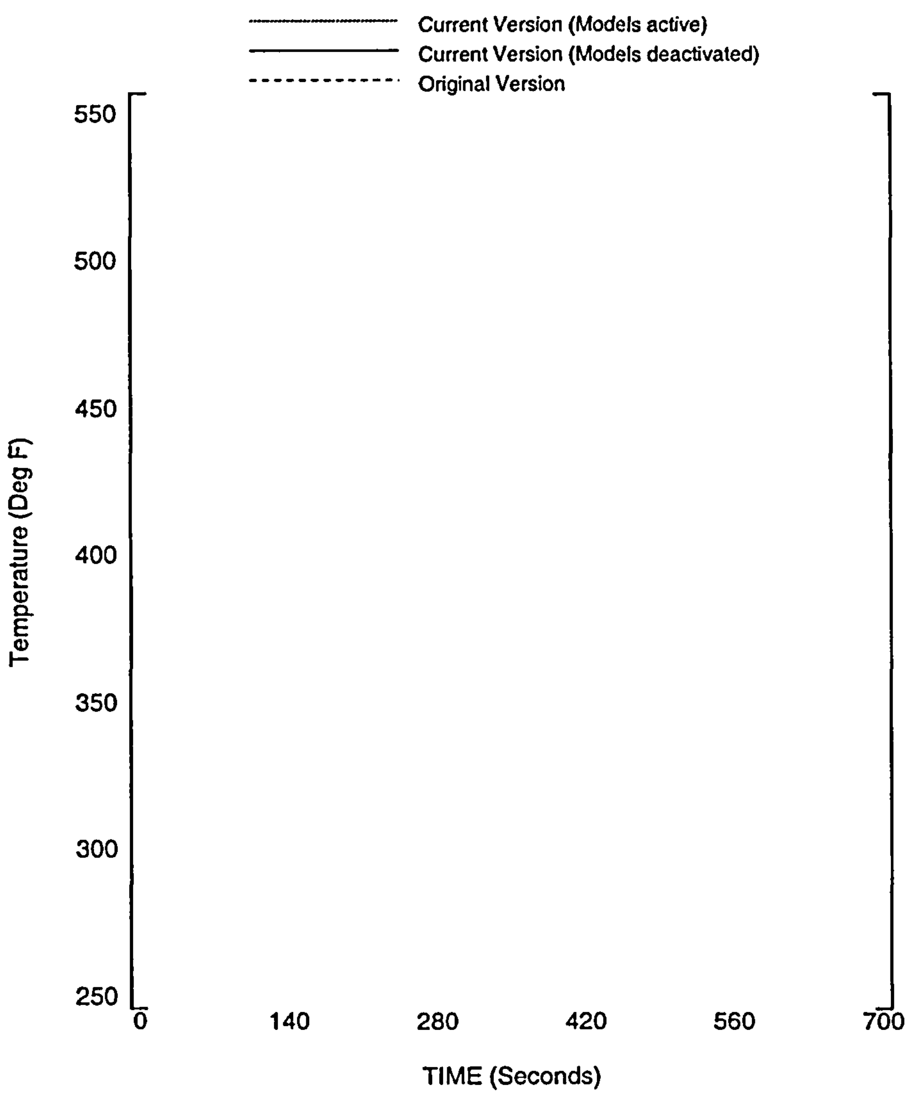


Figure 2.2.1.F

Cold Leg Temperature, Intact Loop

Steam Line Break Event for Plant D

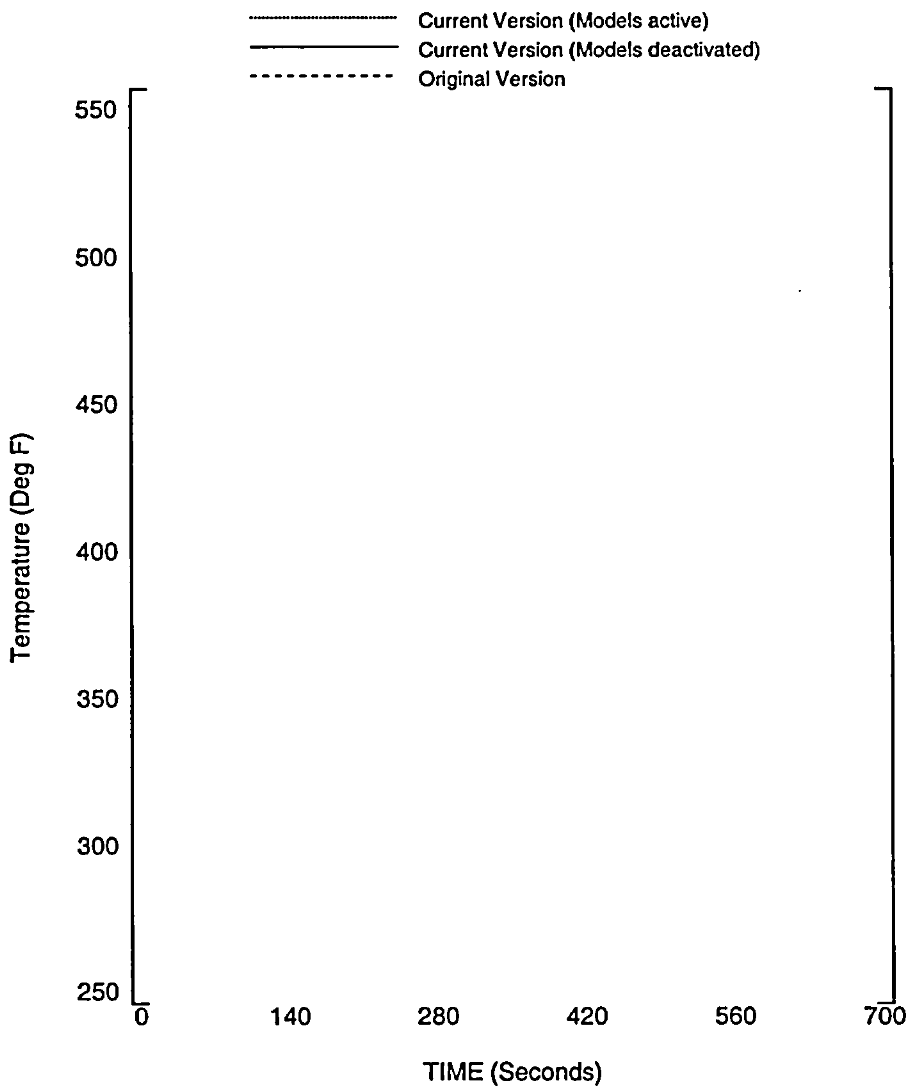


Figure 2.2.1.G

Steam Generator Pressure, Affected Steam Generator

Steam Line Break Event for Plant D

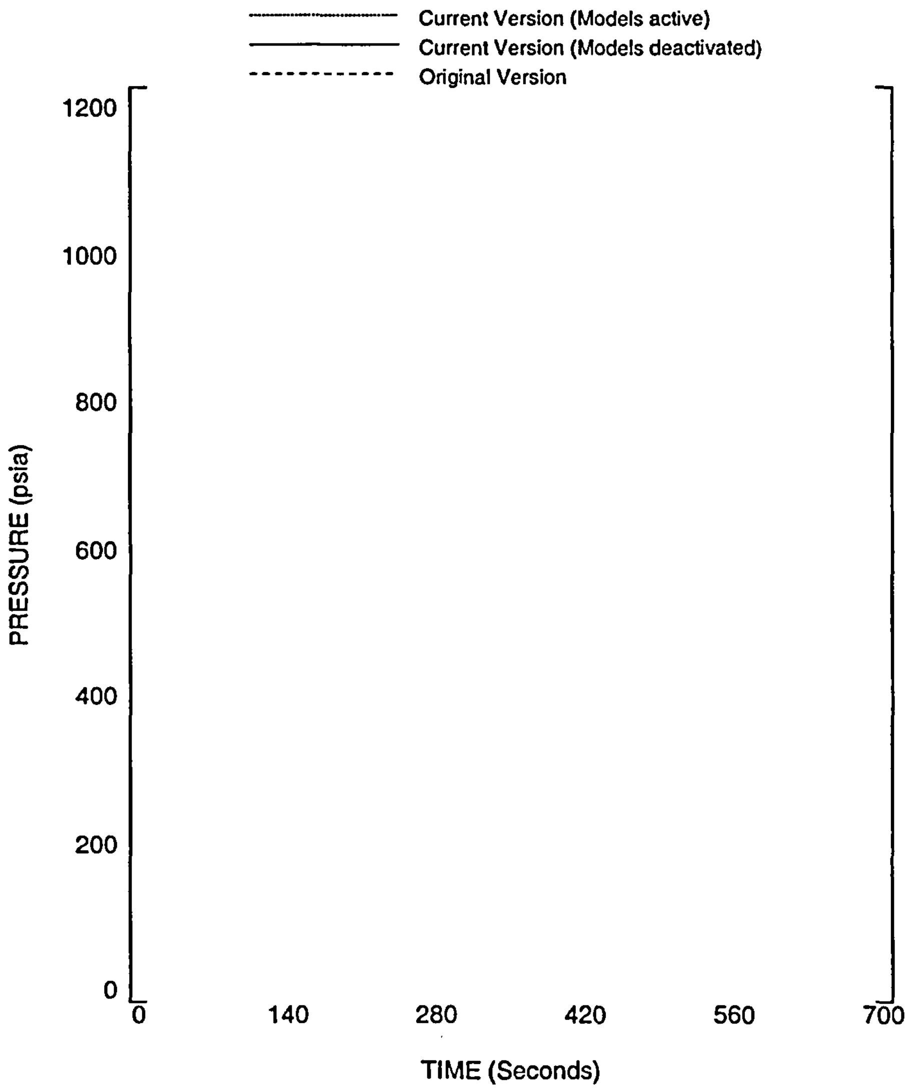


Figure 2.2.1.H

Steam Generator Pressure, Unaffected Steam Generator

Steam Line Break Event for Plant D

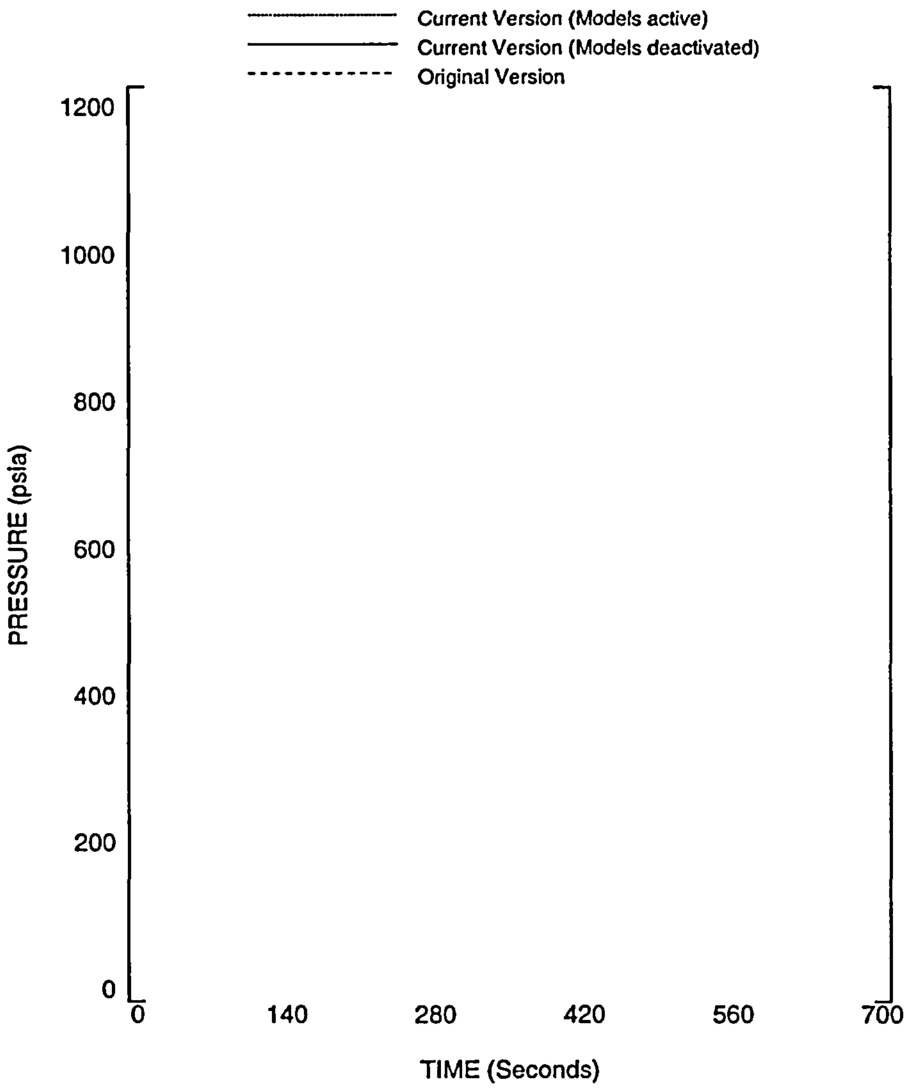


Figure 2.2.1.I

Steam Generator Steam Flow, Affected Steam Generator

Steam Line Break Event for Plant D

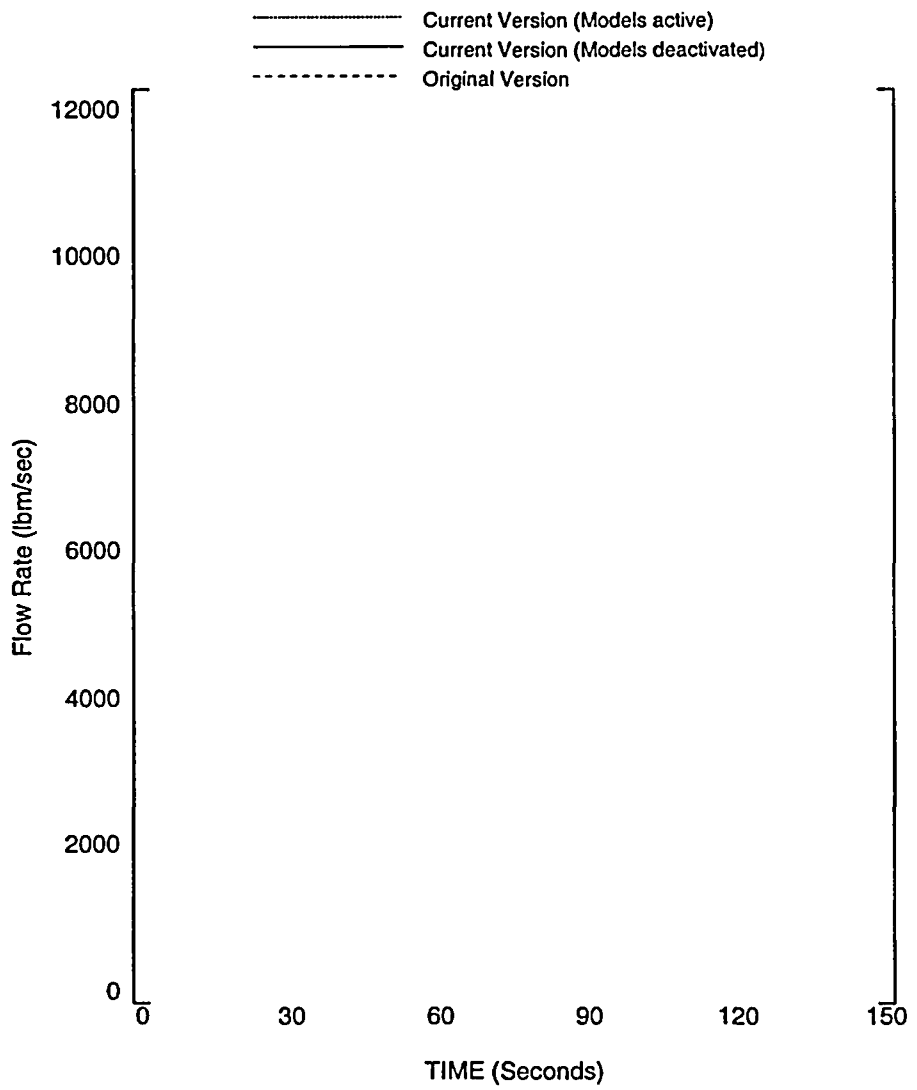


Figure 2.2.1.J

Steam Generator Steam Flow, Unaffected Steam Generator

Steam Line Break Event for Plant D

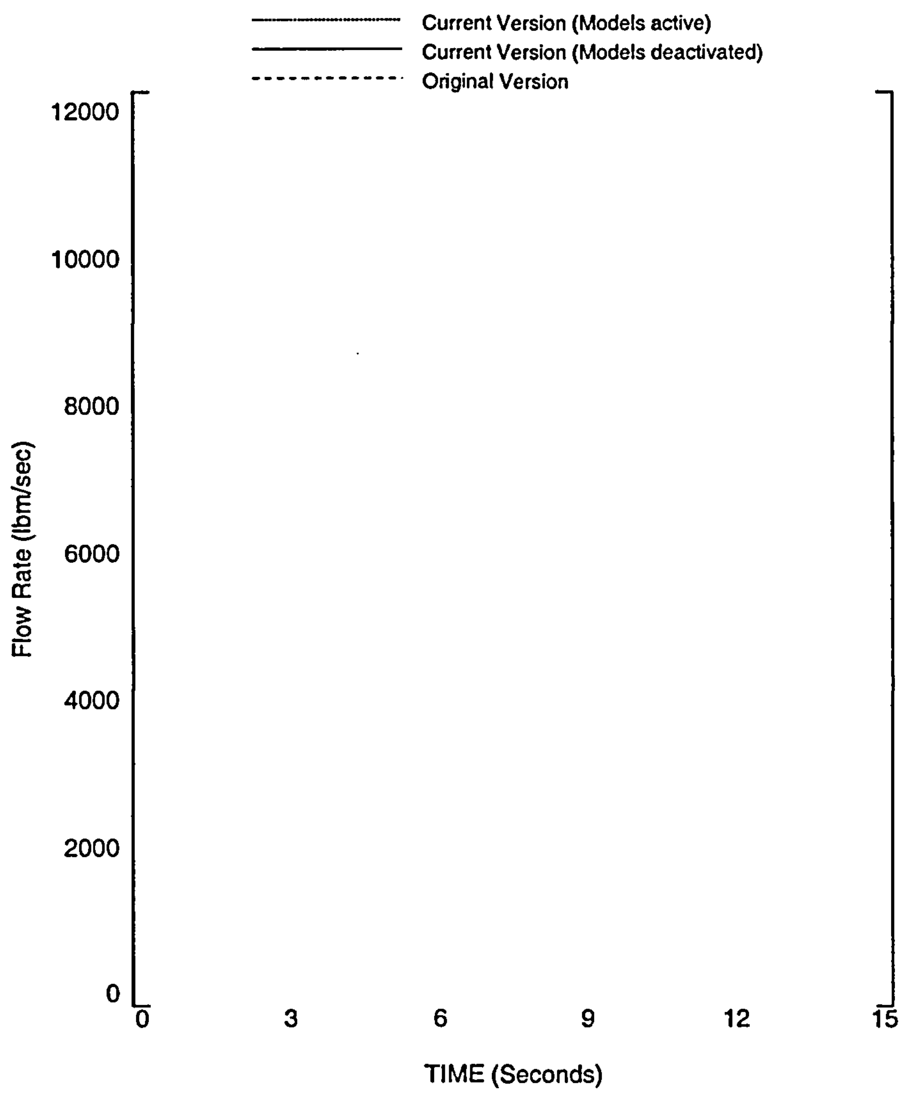


Figure 2.2.1.K

Steam Generator Liquid Mass, Affected Steam Generator

Steam Line Break Event for Plant D

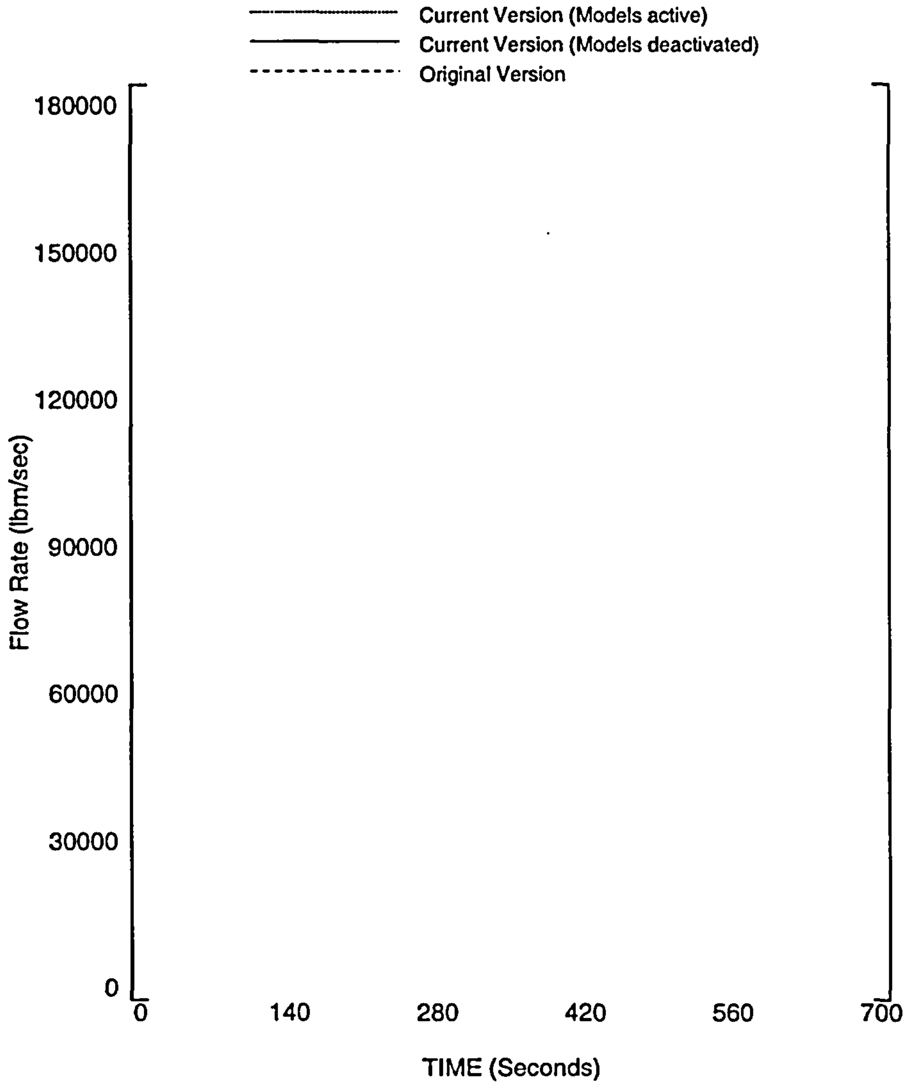


Figure 2.2.1.L

Scram Reactivity

Steam Line Break Event for Plant D

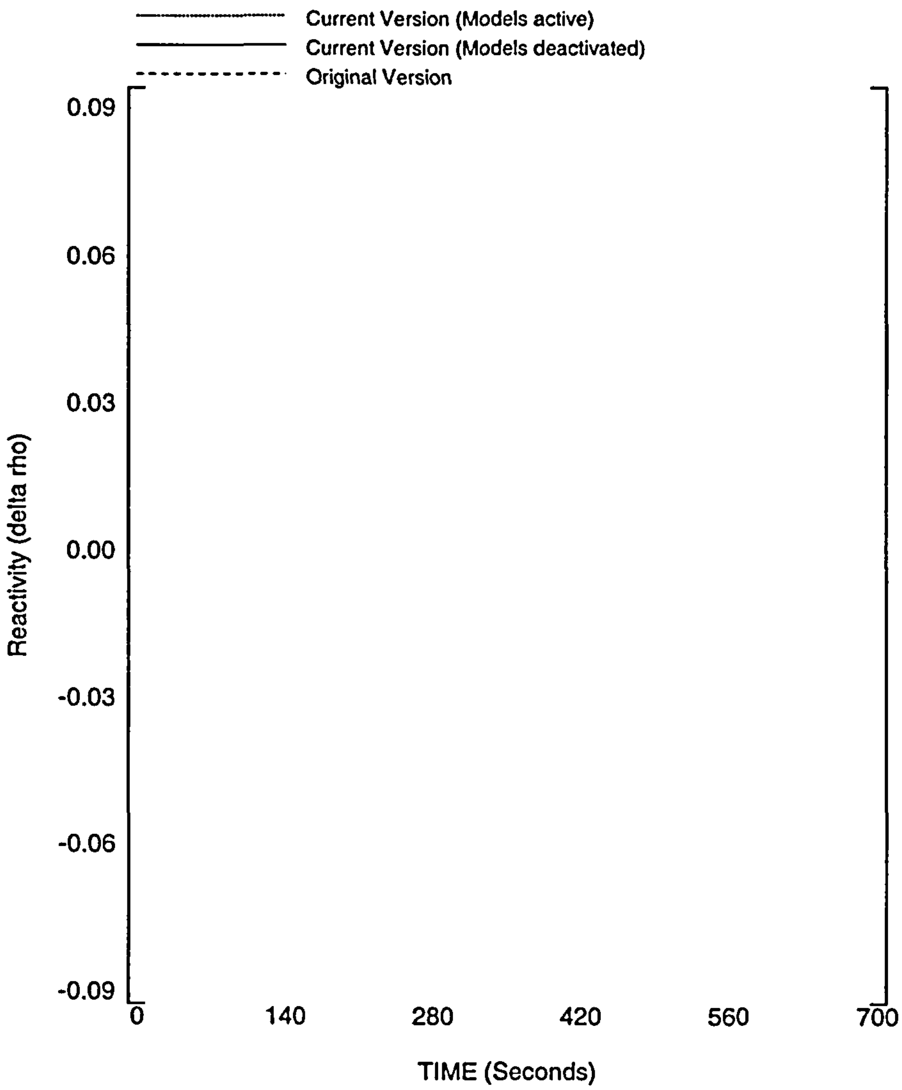


Figure 2.2.1.M

Doppler Reactivity

Steam Line Break Event for Plant D

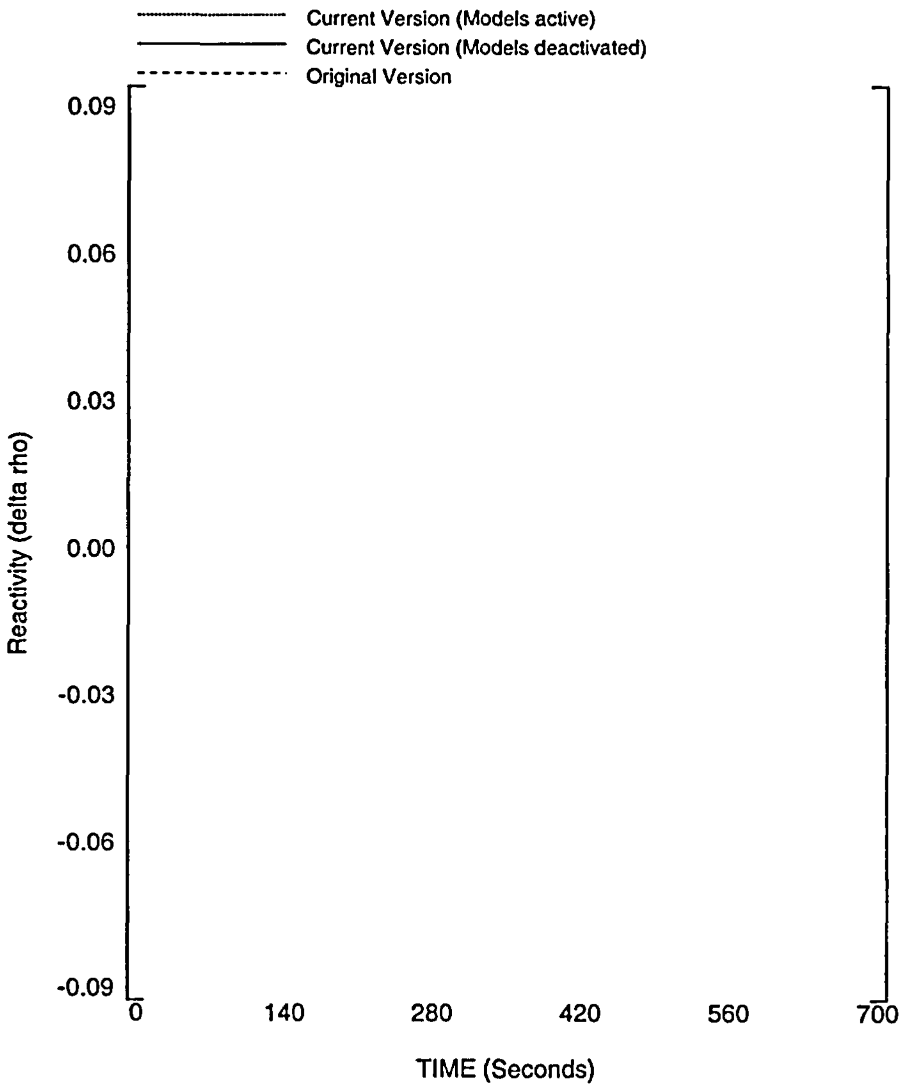


Figure 2.2.1.N

Boron Reactivity

Steam Line Break Event for Plant D

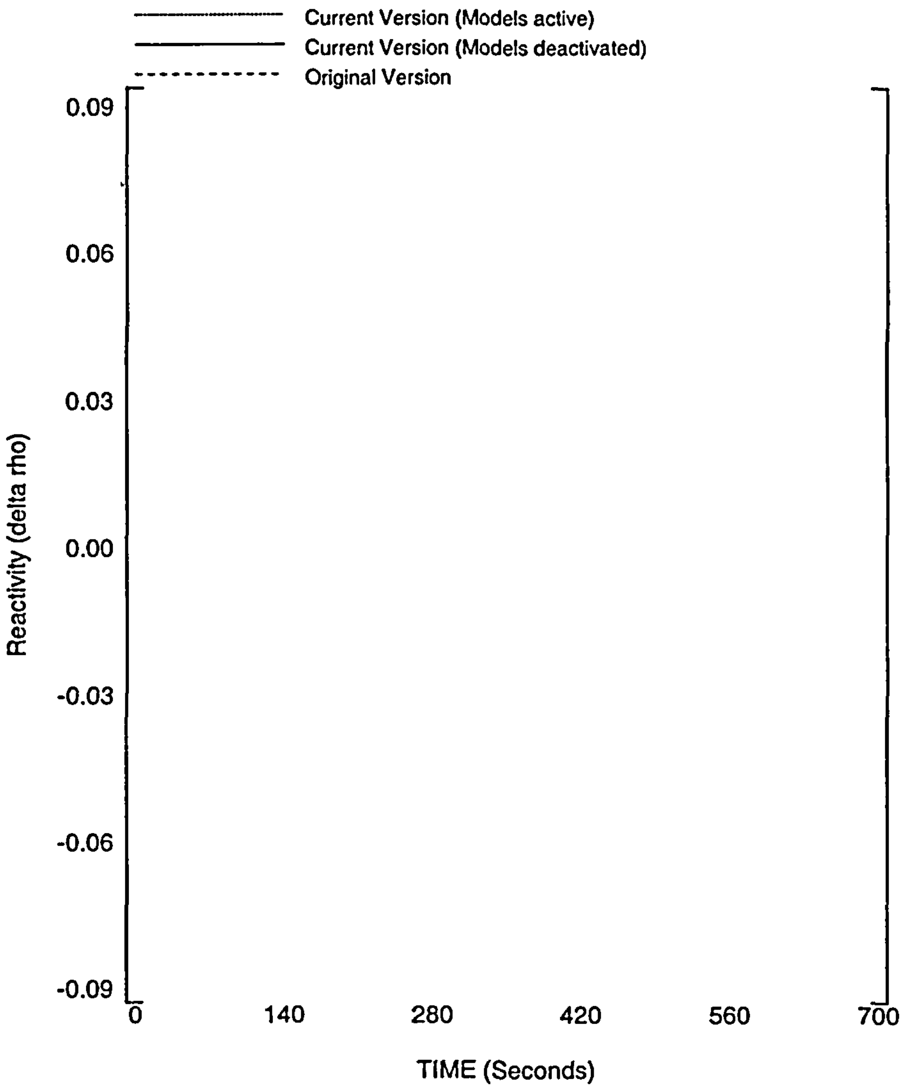


Figure 2.2.1.O

Moderator Reactivity

Steam Line Break Event for Plant D

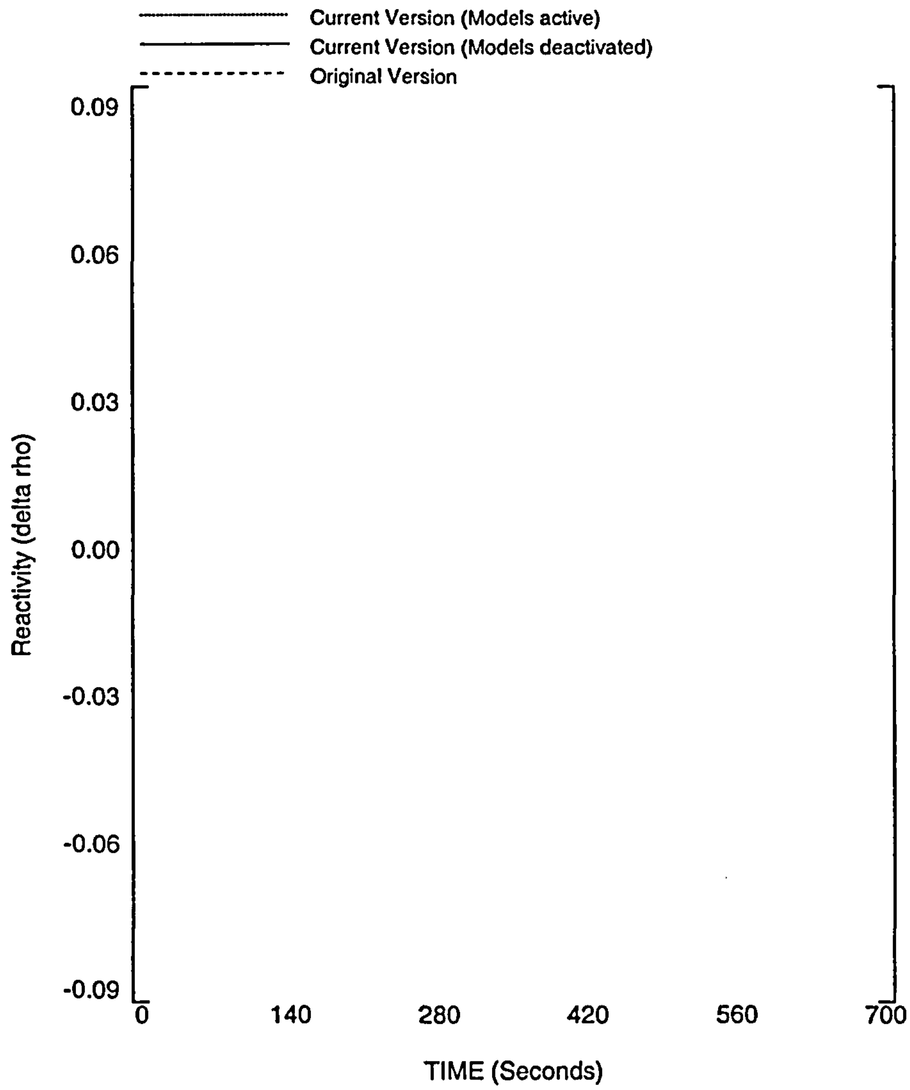


Figure 2.2.1.P

Total Reactivity

Steam Line Break Event for Plant D

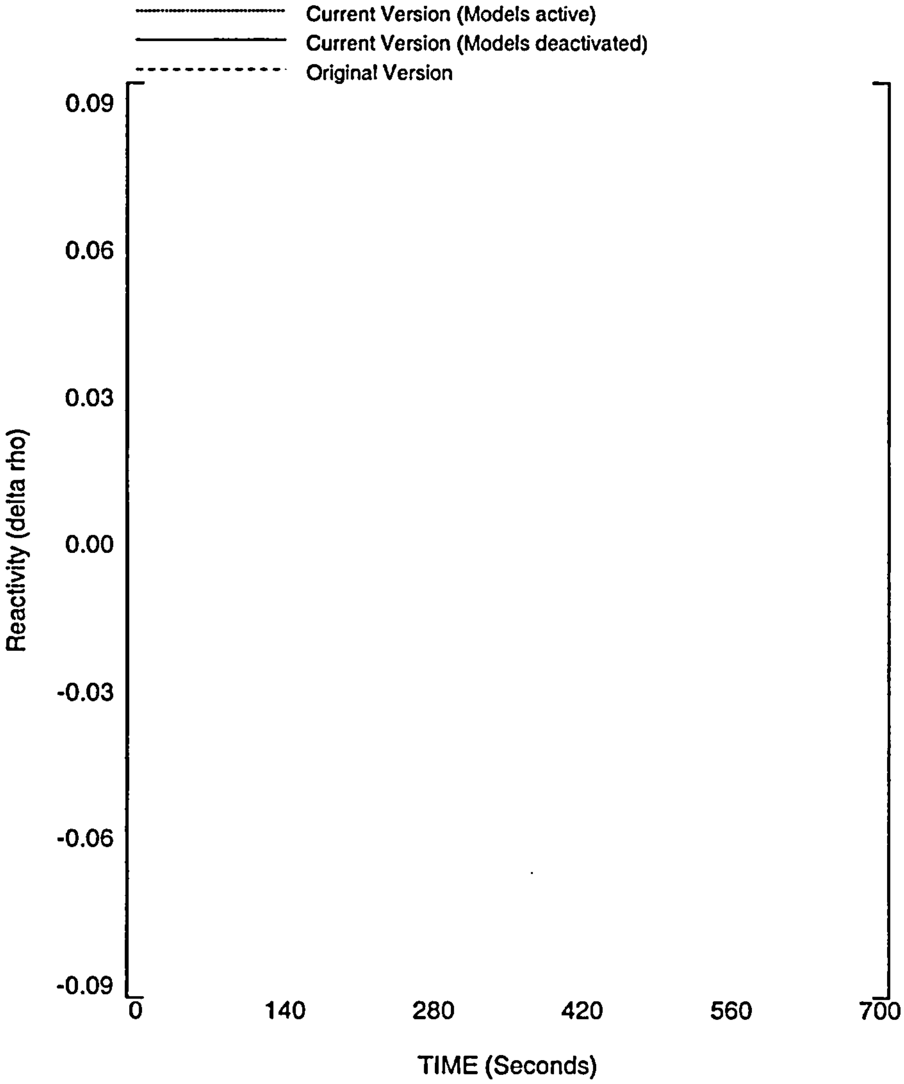


Figure 2.2.1.Q

HERMITE Credit Reactivity

Steam Line Break Event for Plant D

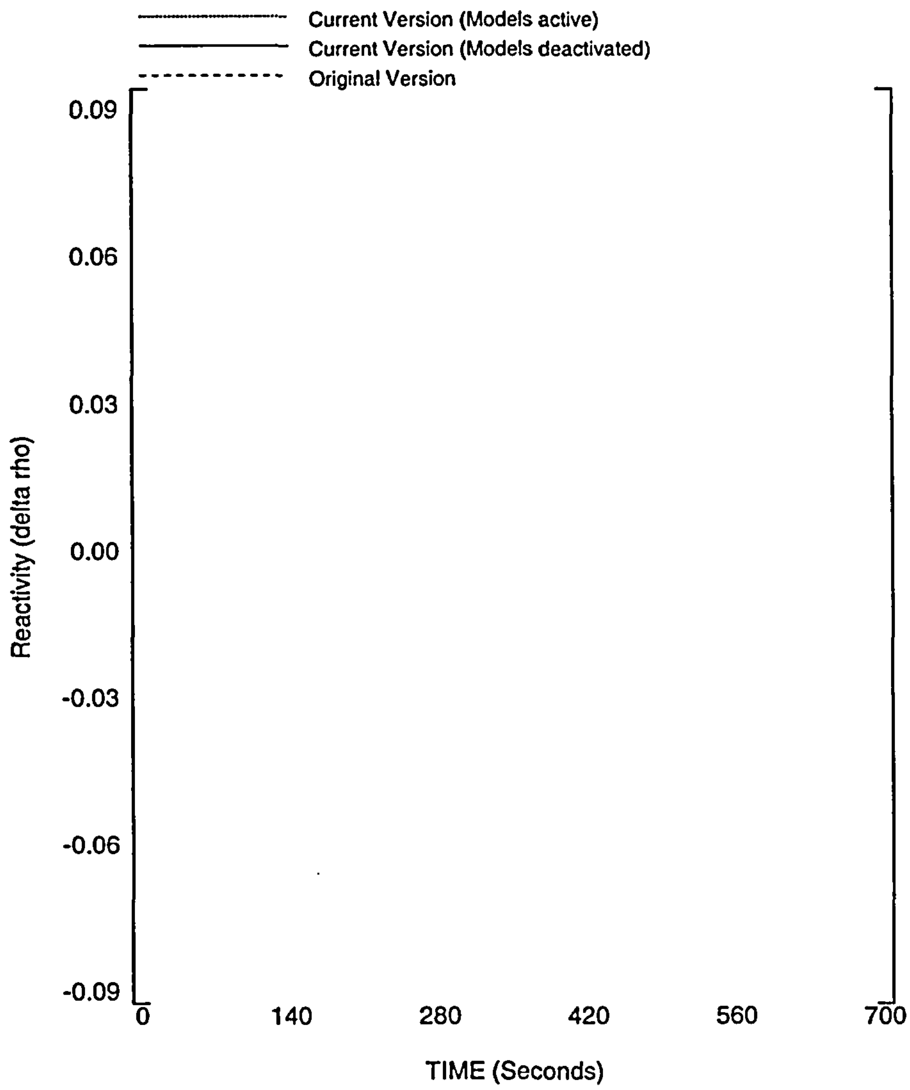


Figure 2.2.2.A

Reactor Core Power
Feedwater Line Break for Plant D

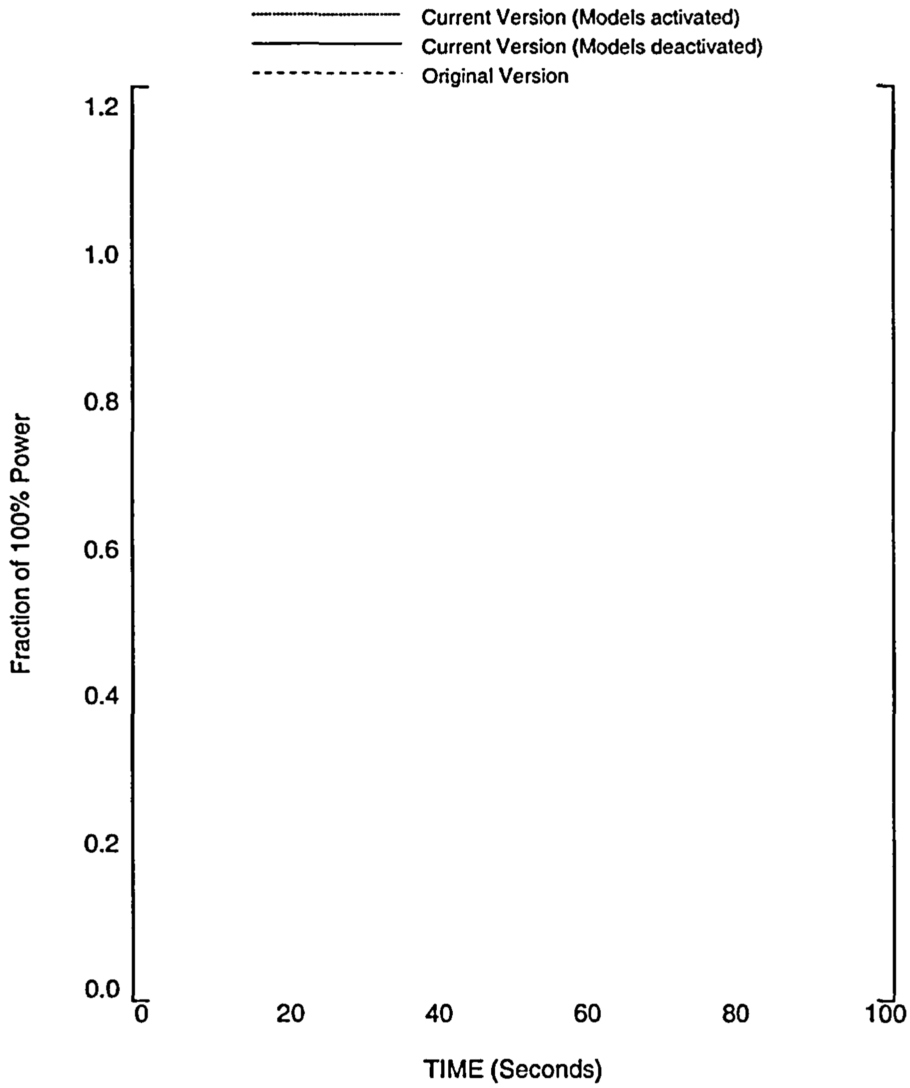


Figure 2.2.2.B

Reactor Core Heat Flux
Feedwater Line Break for Plant D

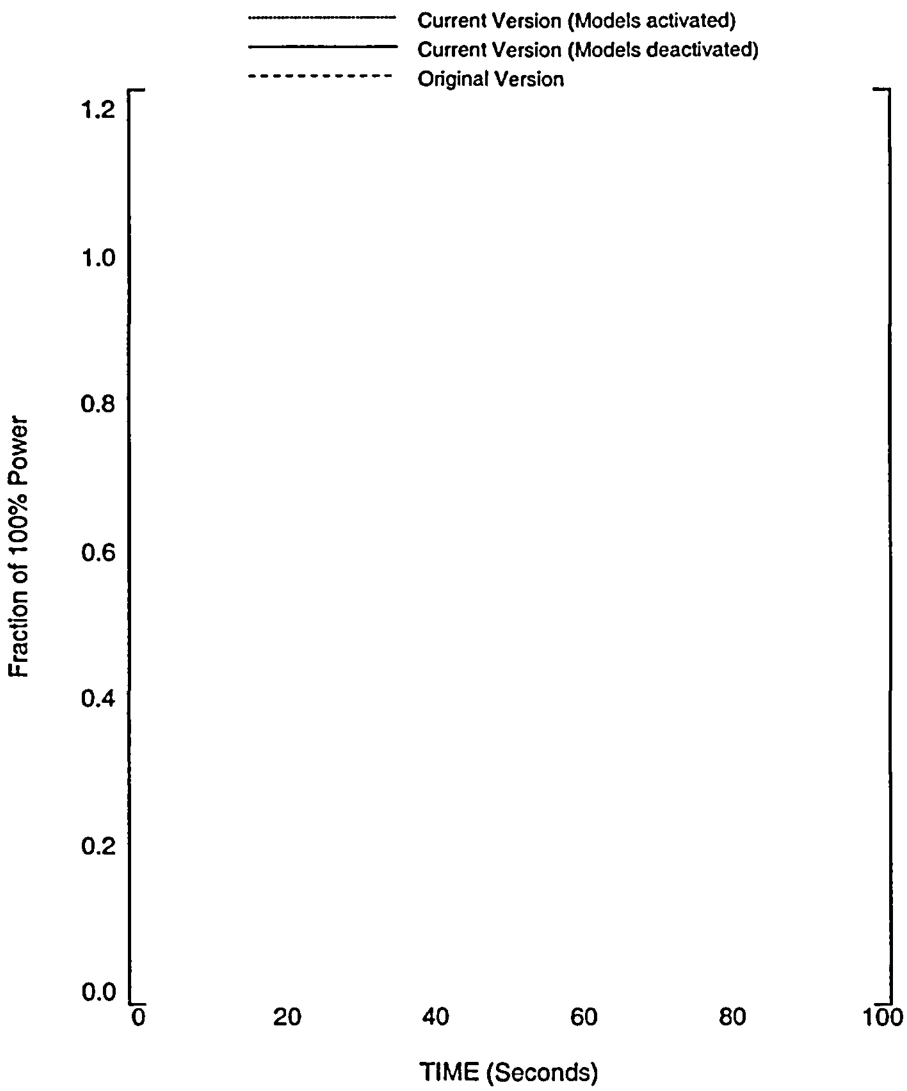


Figure 2.2.2.C

Reactor Coolant System Pressure
Feedwater Line Break for Plant D

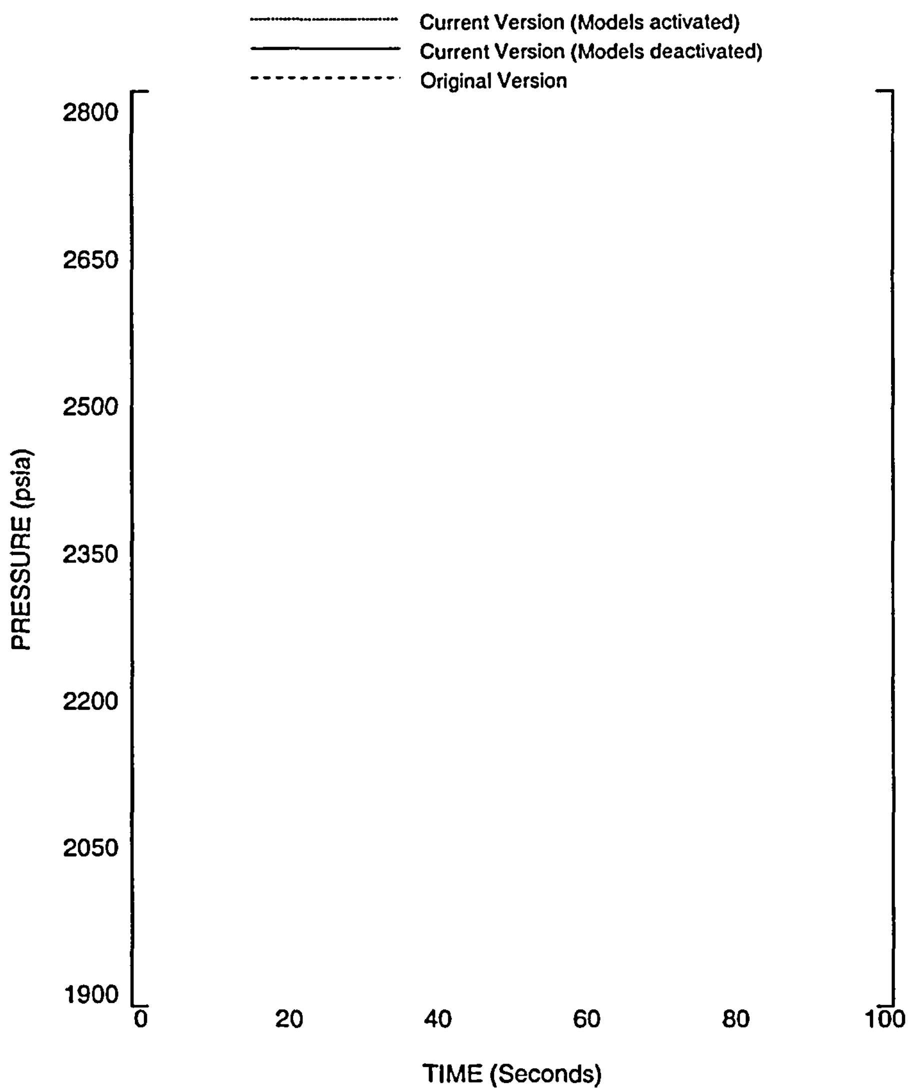


Figure 2.2.2.D

Pressurizer Pressure
Feedwater Line Break for Plant D

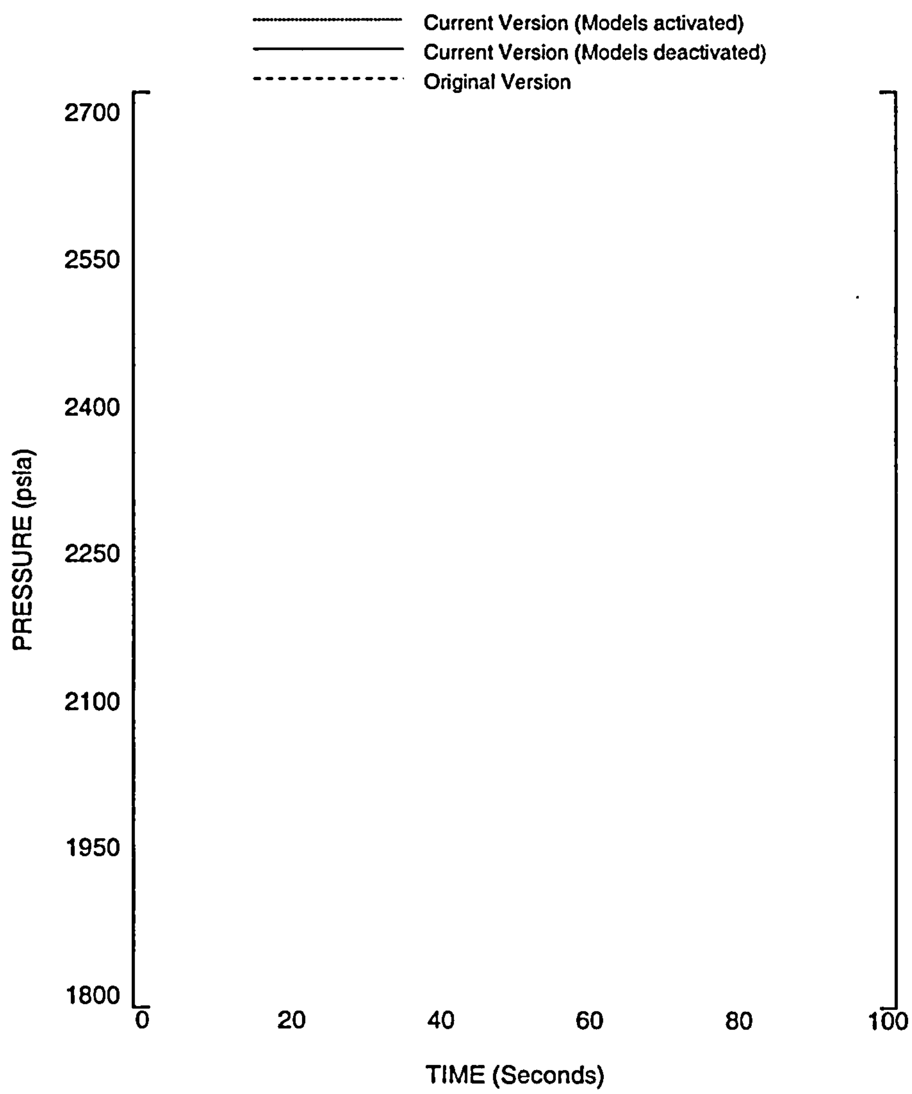


Figure 2.2.2.E

Cold Leg Temperature, Affected Loop
Feedwater Line Break for Plant D

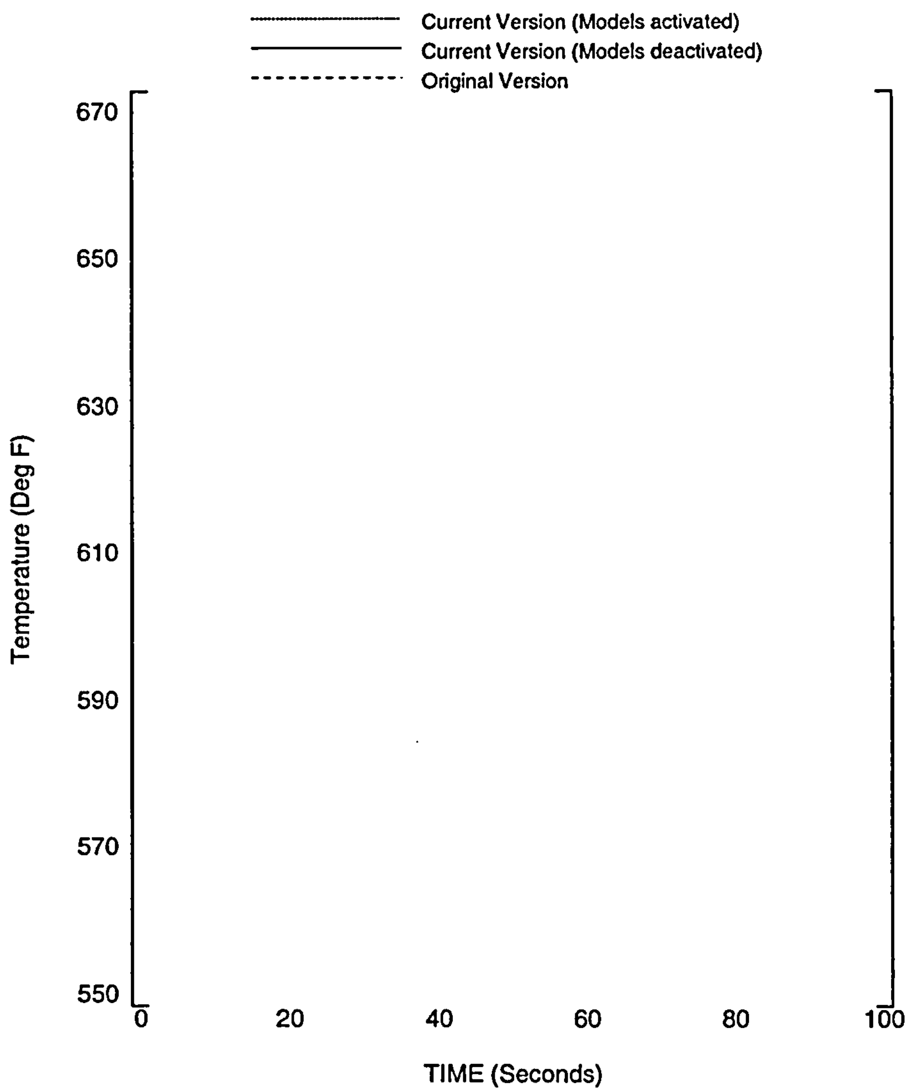


Figure 2.2.2.F

Cold Leg Temperature, Intact Loop
Feedwater Line Break for Plant D

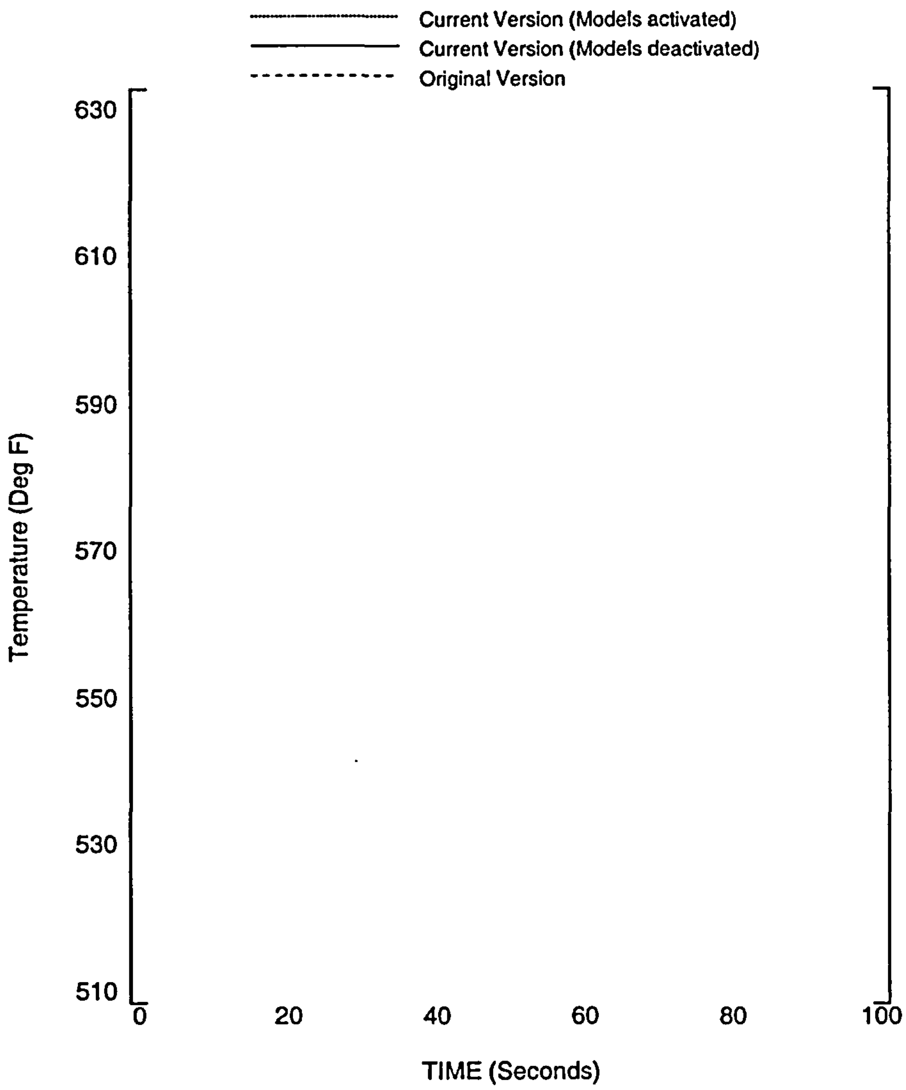


Figure 2.2.2.G

Steam Generator Pressure, Affected Steam Generator
Feedwater Line Break for Plant D

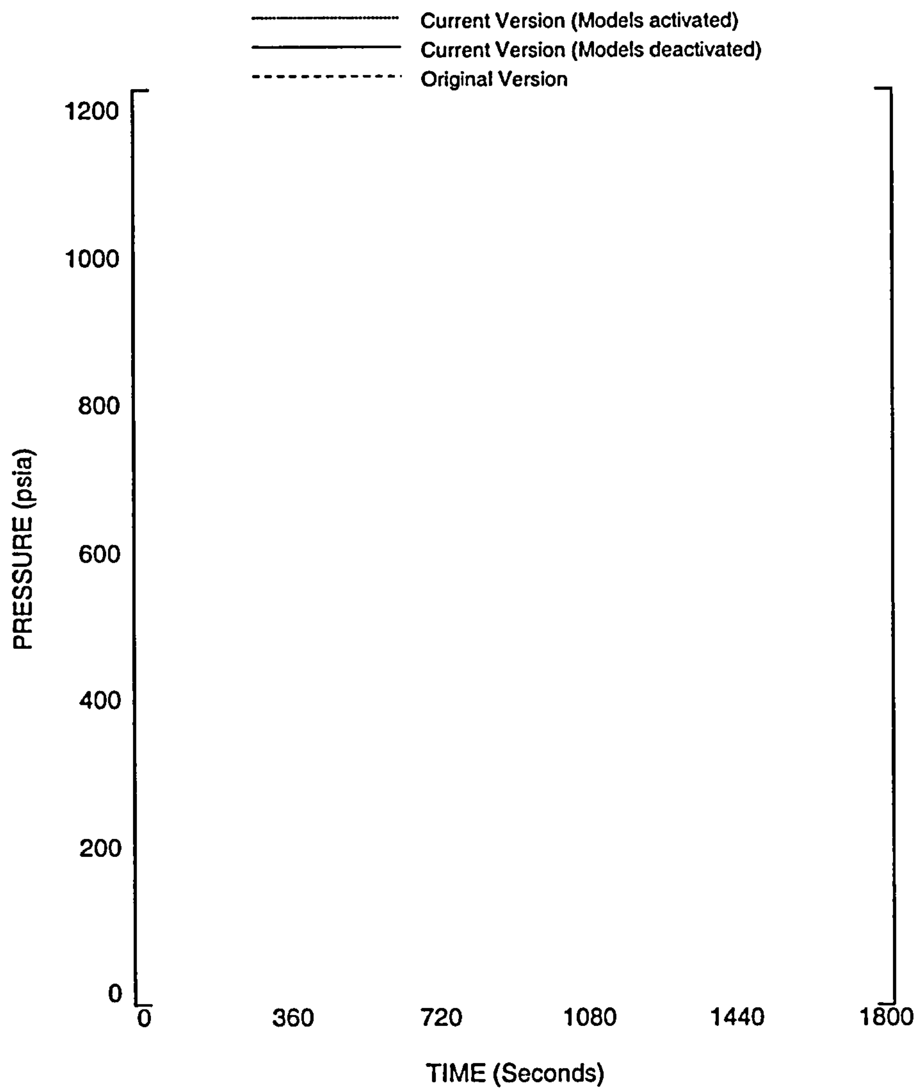


Figure 2.2.2.H

Steam Generator Pressure, Intact Steam Generator
Feedwater Line Break for Plant D

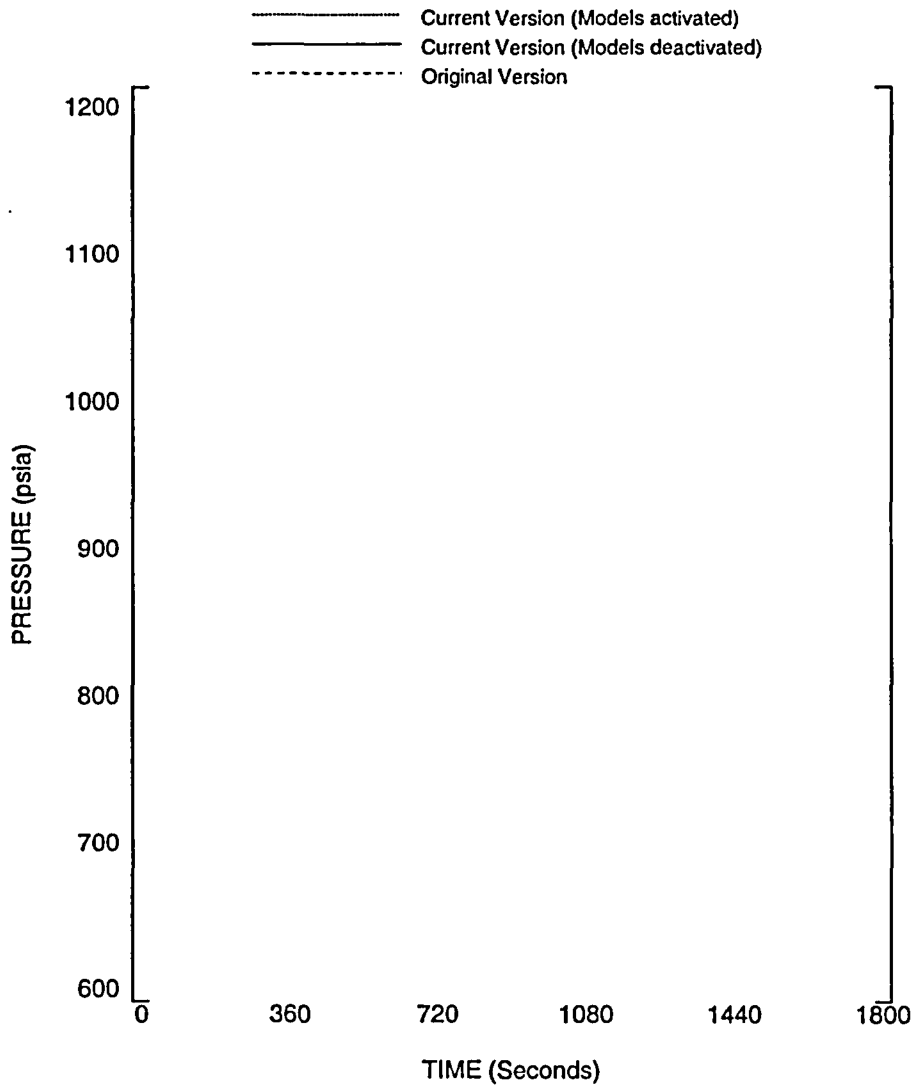


Figure 2.2.2.1

Total Steam Flow, Affected Steam Generator
Feedwater Line Break for Plant D

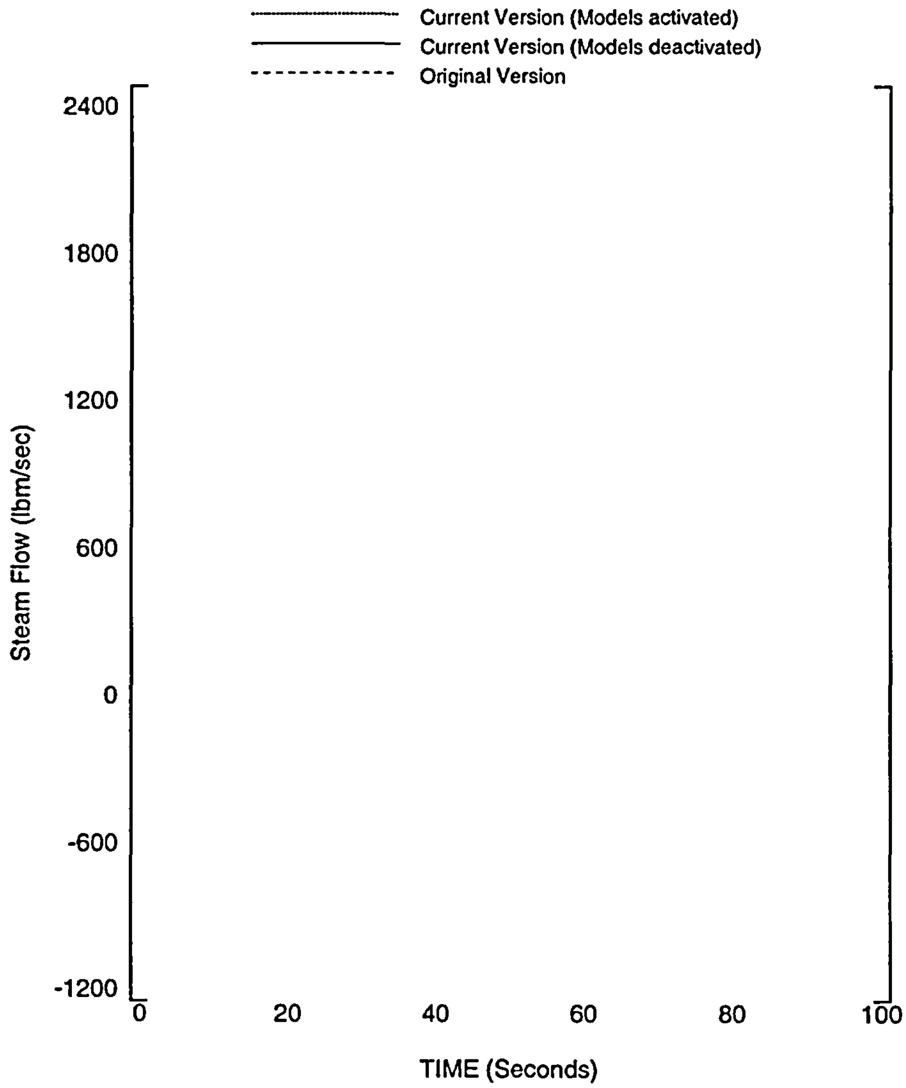


Figure 2.2.2.J

Total Steam Flow, Intact Steam Generator
Feedwater Line Break for Plant D

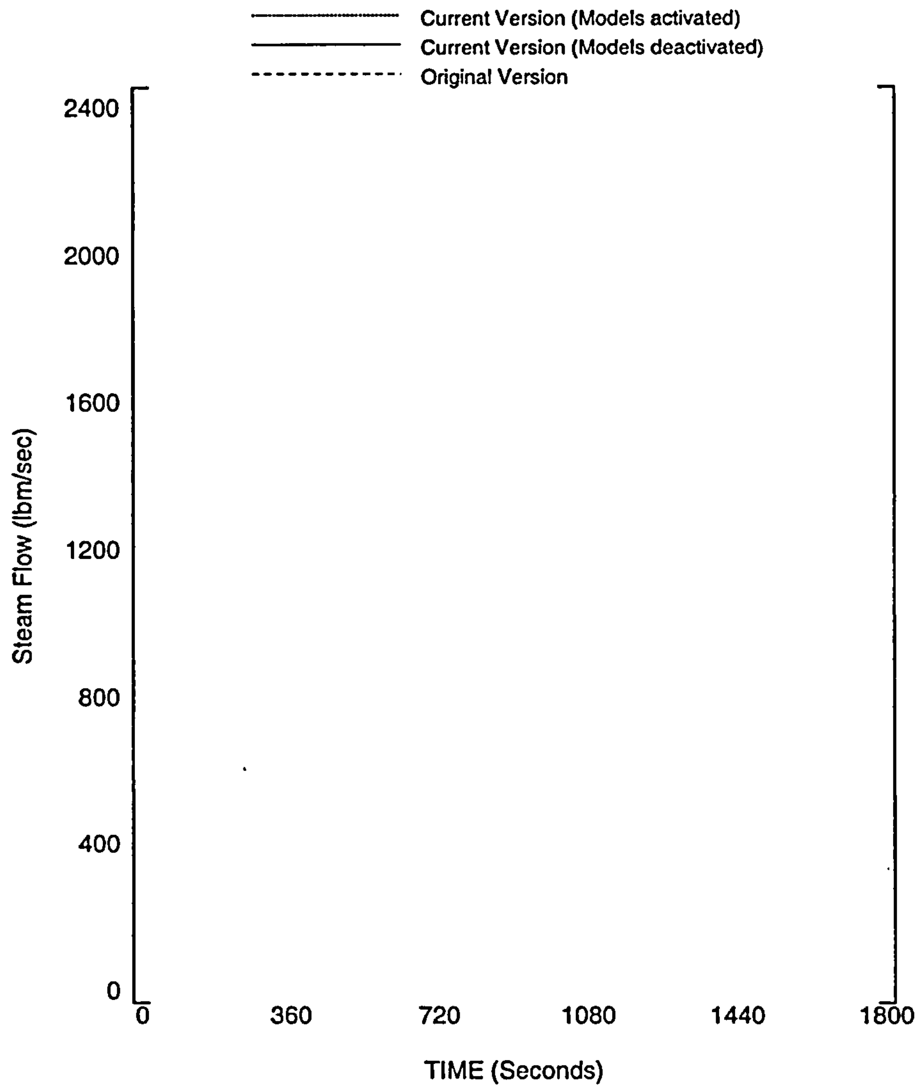


Figure 2.2.2.K

Steam Generator Liquid Mass, Affected Steam Generator
Feedwater Line Break for Plant D

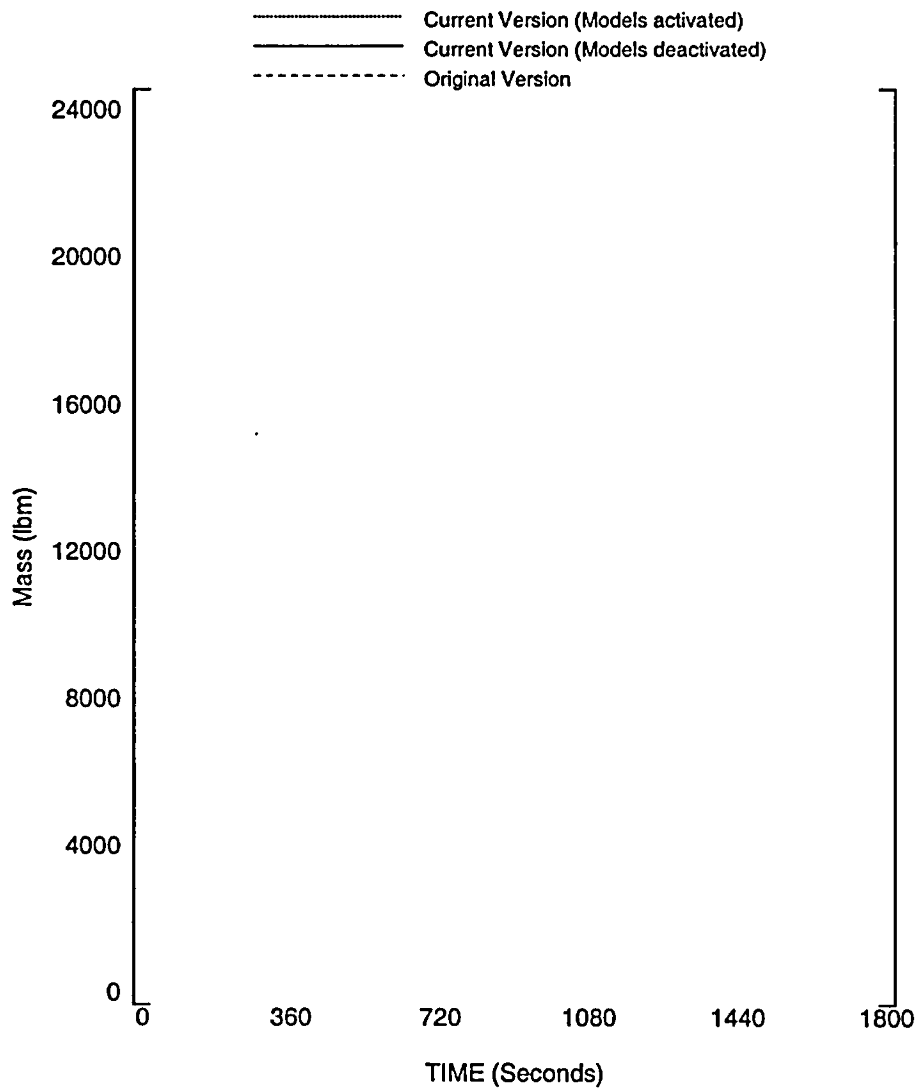


Figure 2.2.2.L

Steam Generator Liquid Mass, Intact Steam Generator
Feedwater Line Break for Plant D

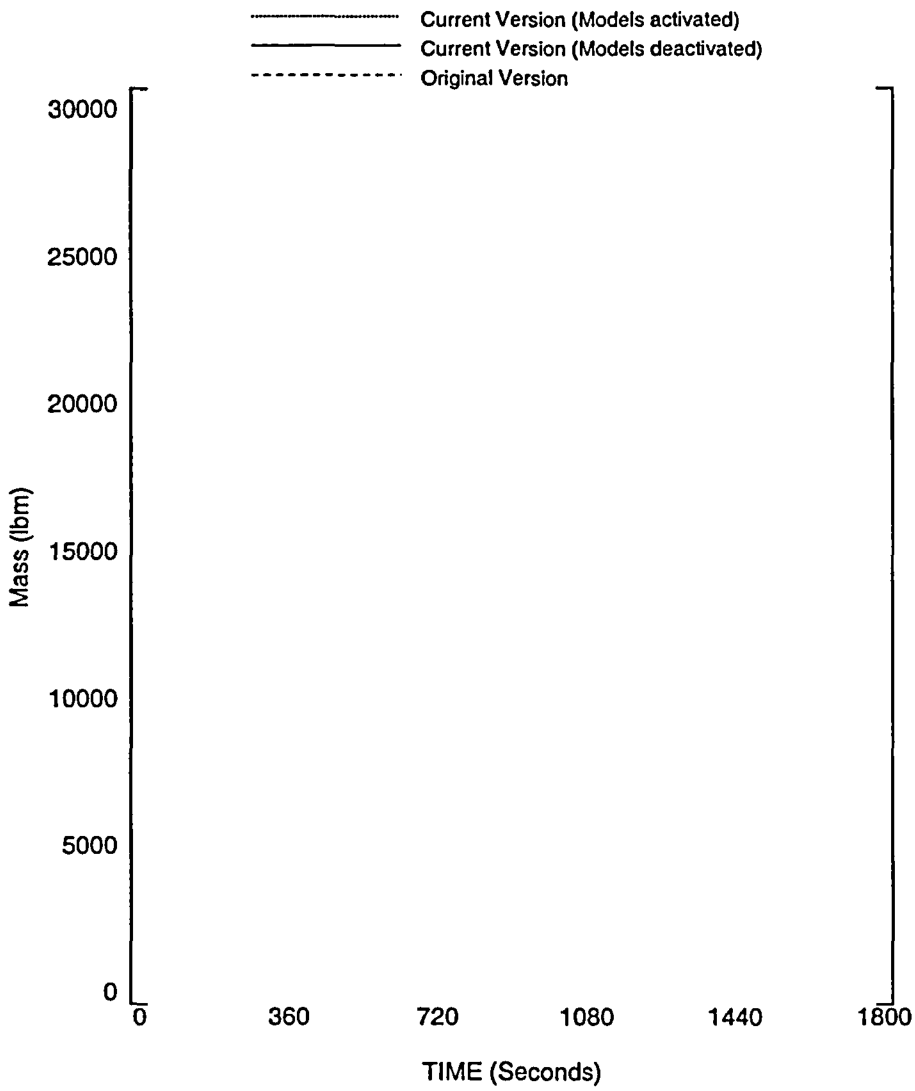


Figure 2.2.2.M

Core Flow
Feedwater Line Break for Plant D

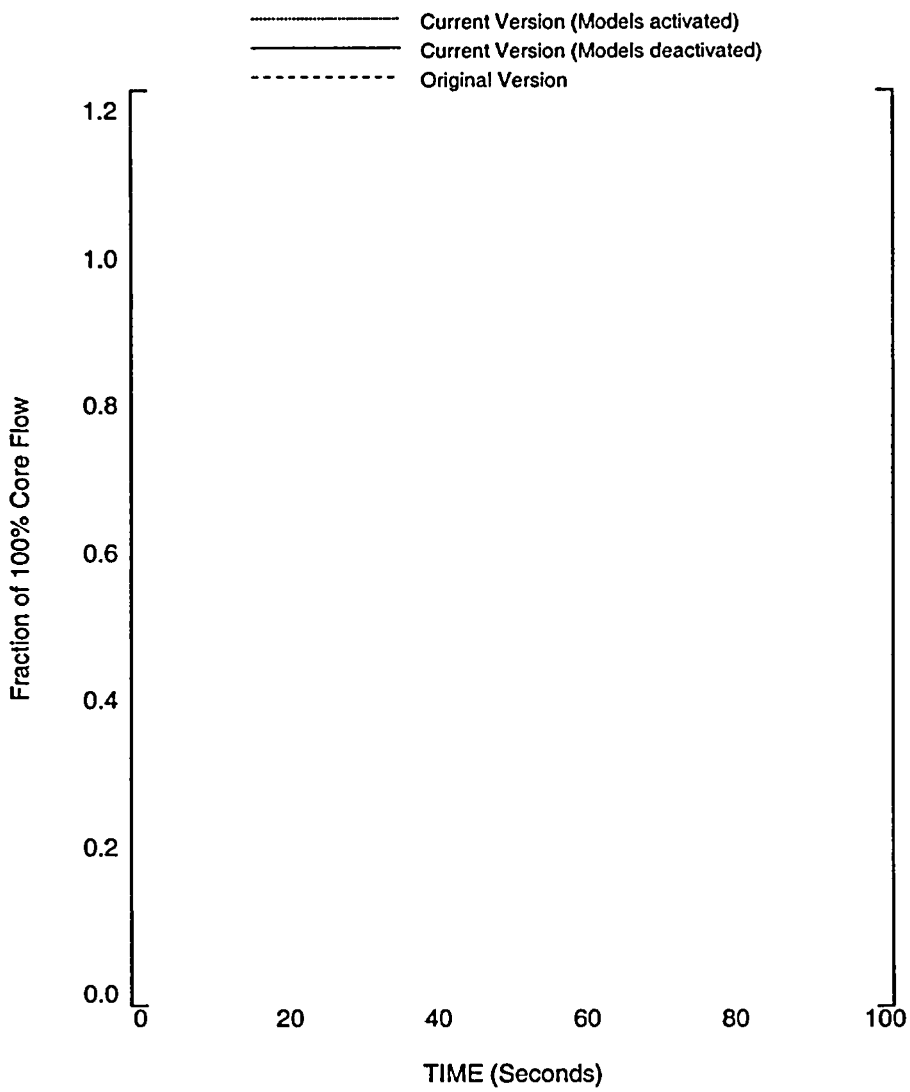


Figure 2.2.2.N

Affected Steam Generator, Back flow to Break
Feedwater Line Break for Plant D

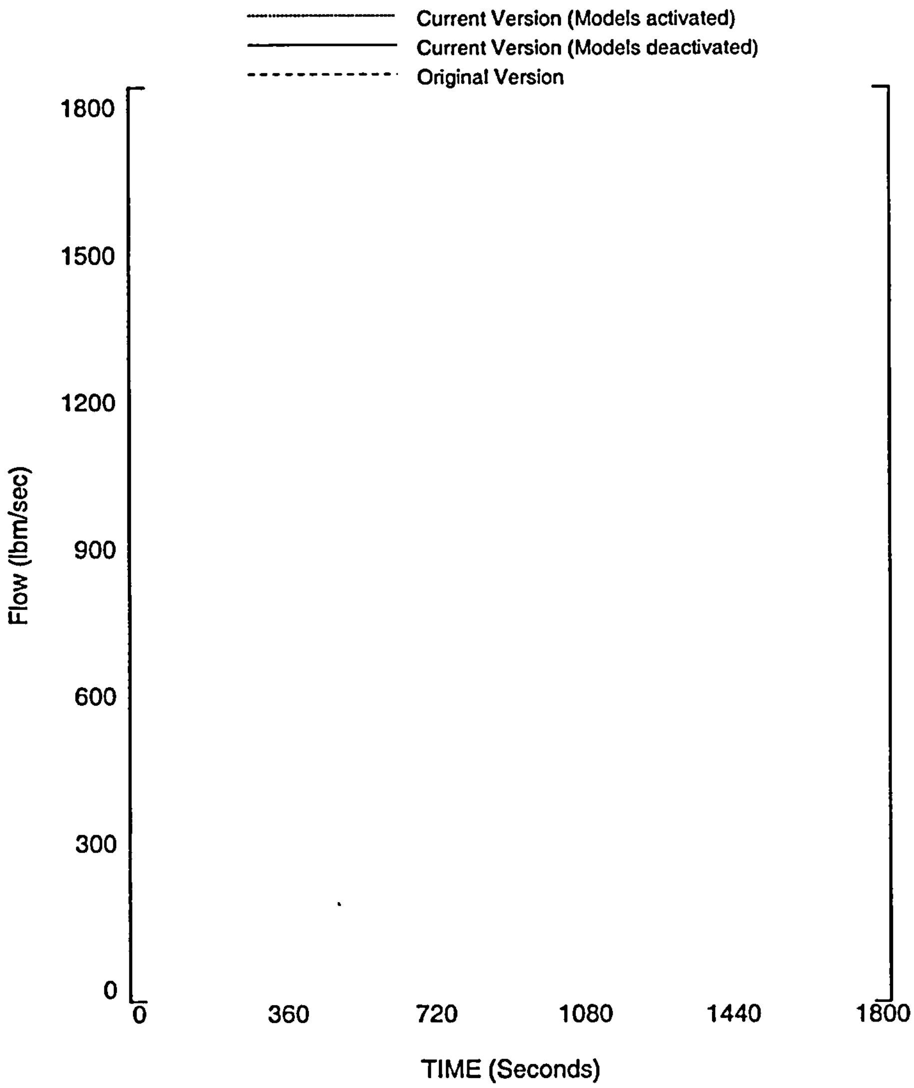


Figure 2.2.2.O

Pressurizer Safety Valve Flow
Feedwater Line Break for Plant D

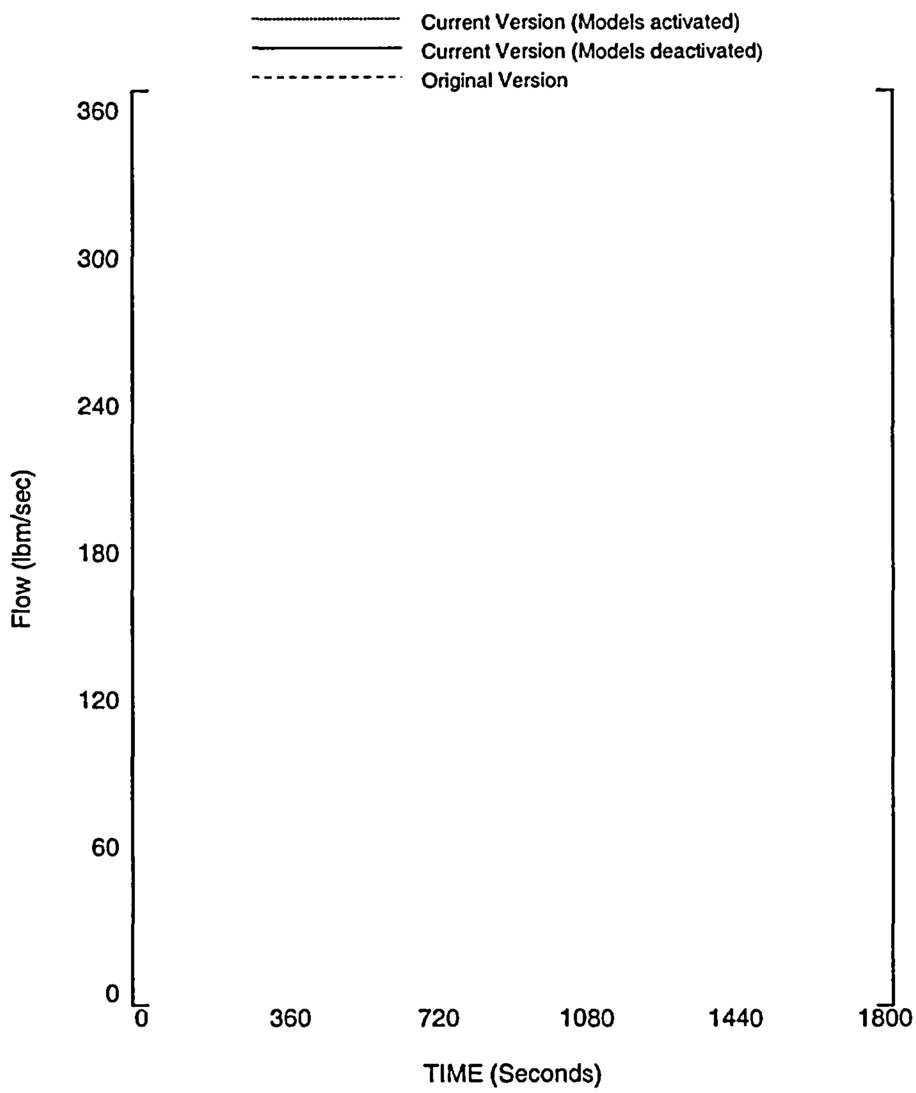


Figure 2.2.2.P

Pressurizer Two-Phase Volume
Feedwater Line Break for Plant D

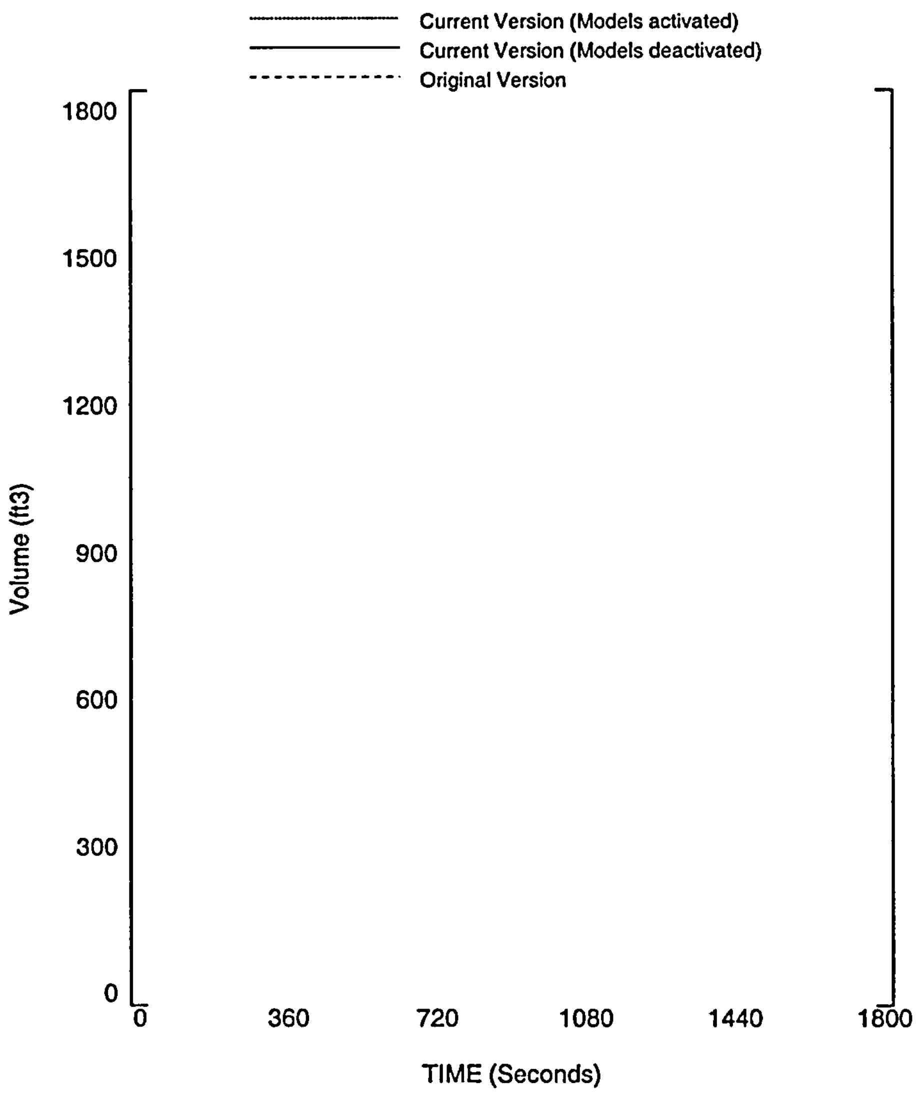


Figure 2.2.2.Q

Pressurizer Pressure
Feedwater Line Break for Plant D

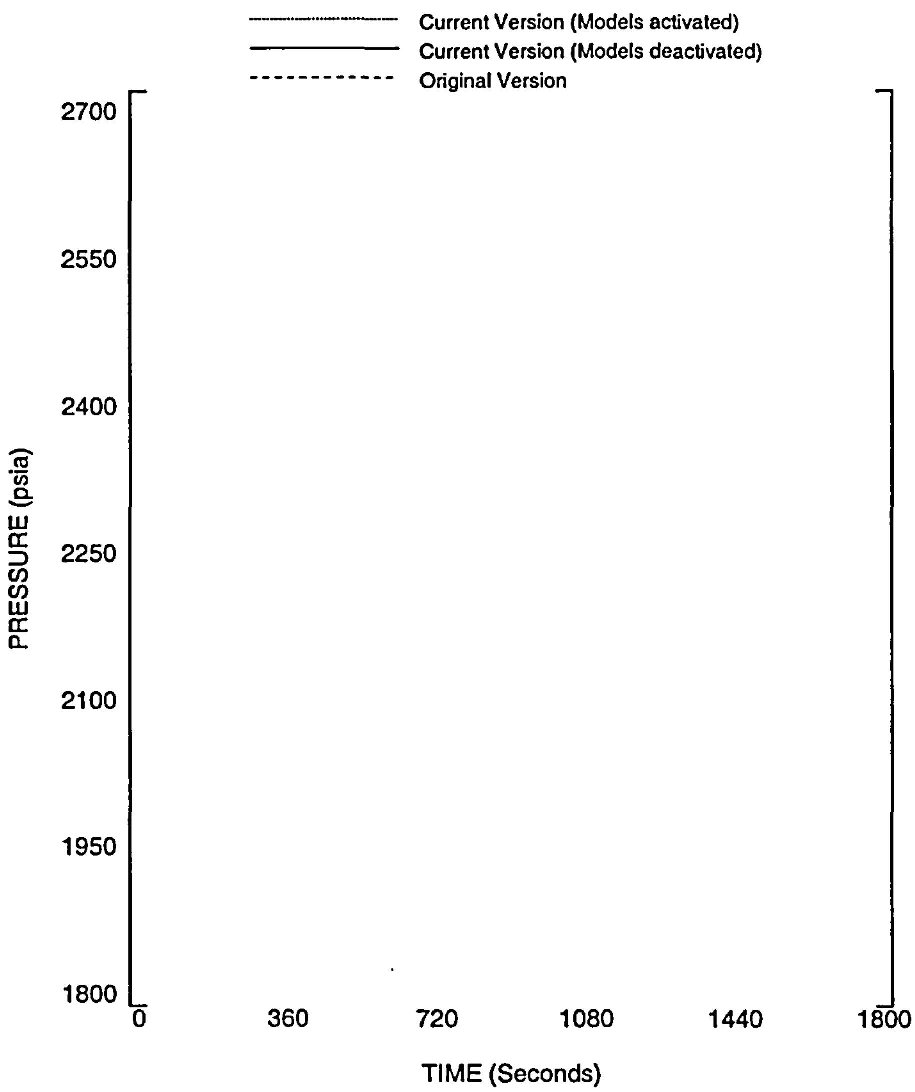


Figure 2.2.3.A

Reactor Core Power

Subcritical CEA Withdrawal Event for Plant D

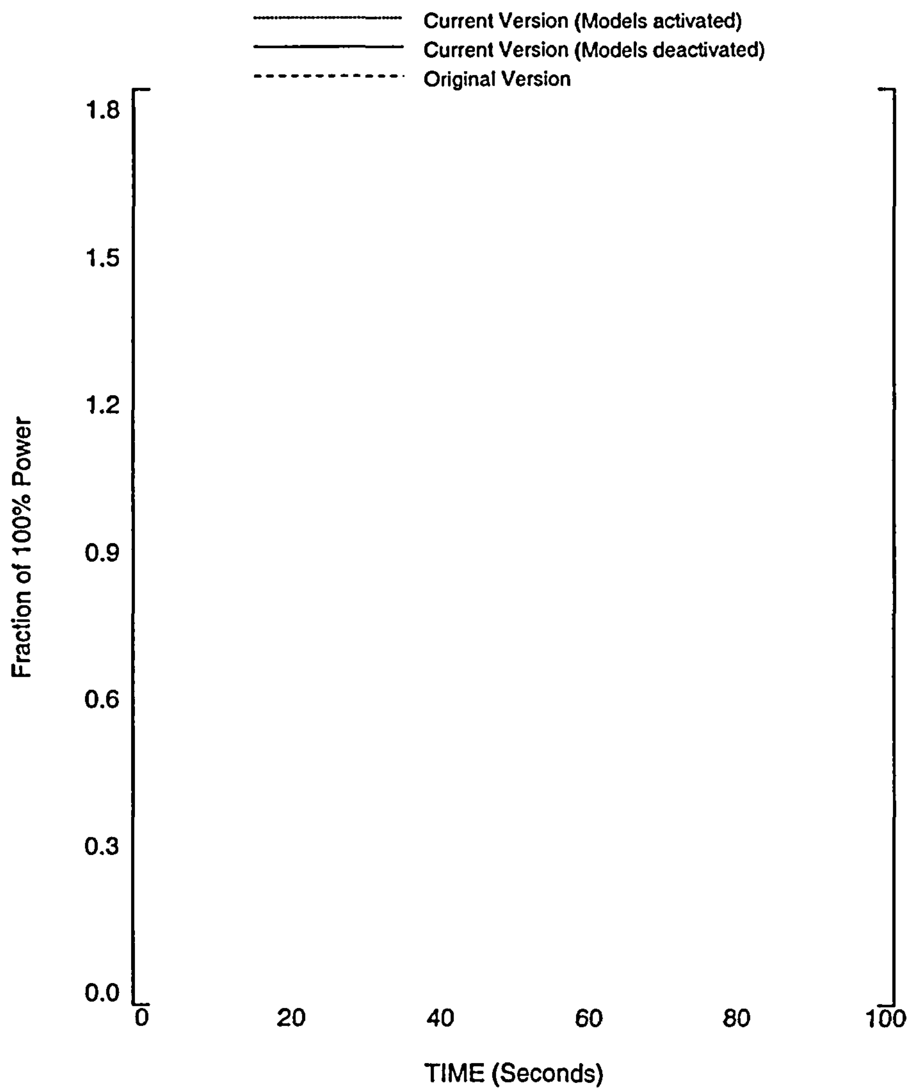


Figure 2.2.3.B

Reactor Core Heat Flux

Subcritical CEA Withdrawal Event for Plant D

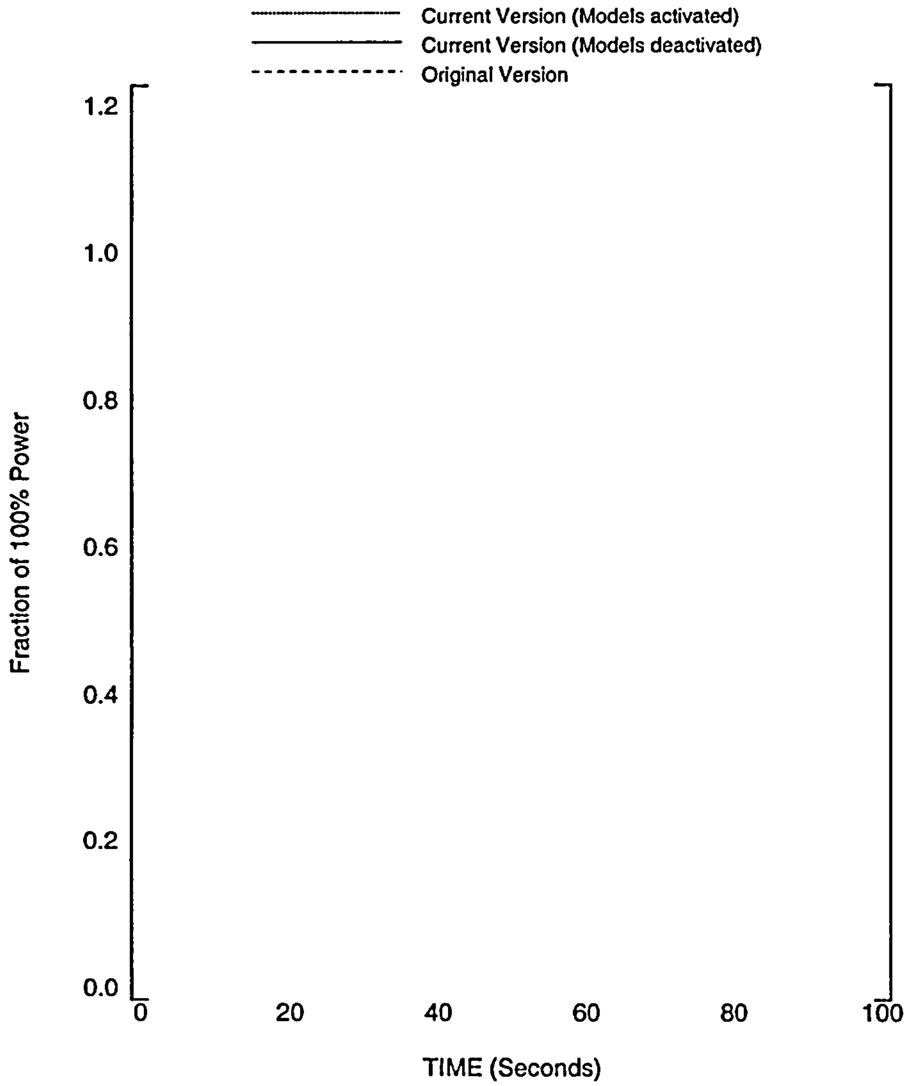


Figure 2.2.3.C

Pressurizer Pressure

Subcritical CEA Withdrawal Event for Plant D

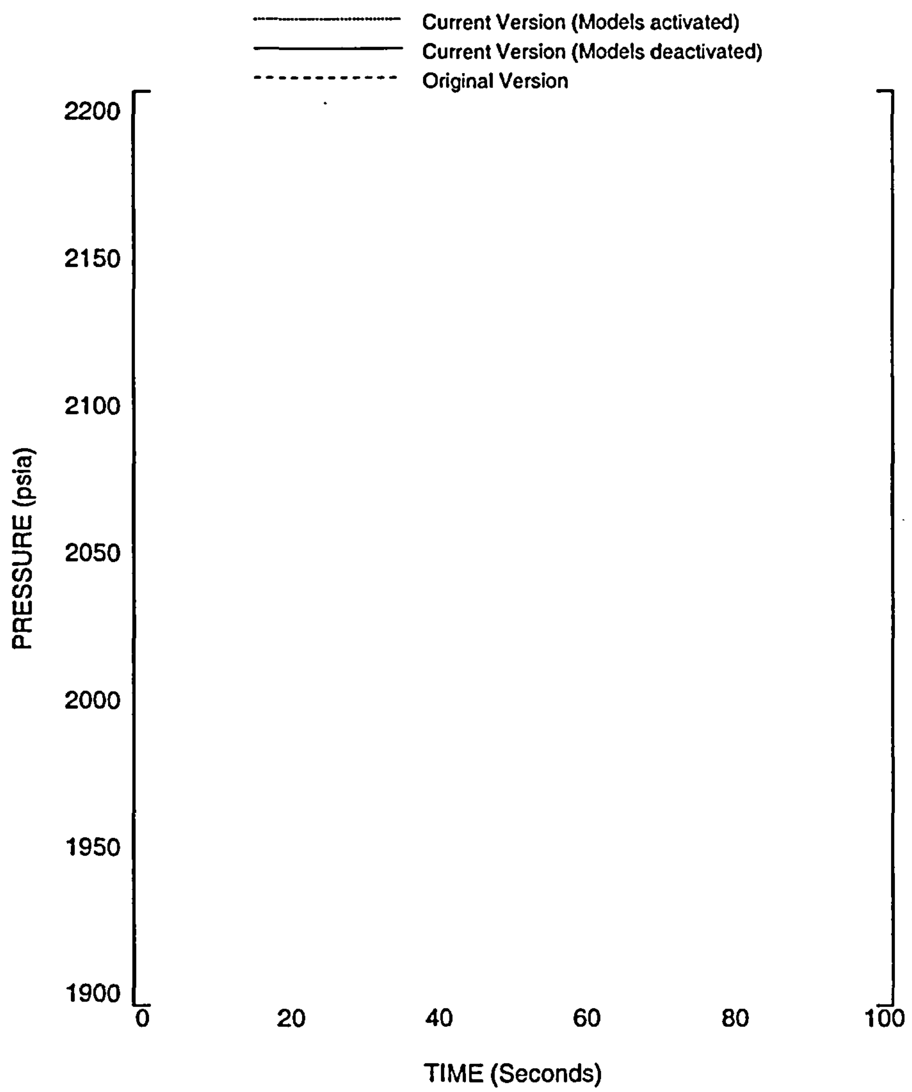


Figure 2.2.3.D

Core Average Temperature

Subcritical CEA Withdrawal Event for Plant D

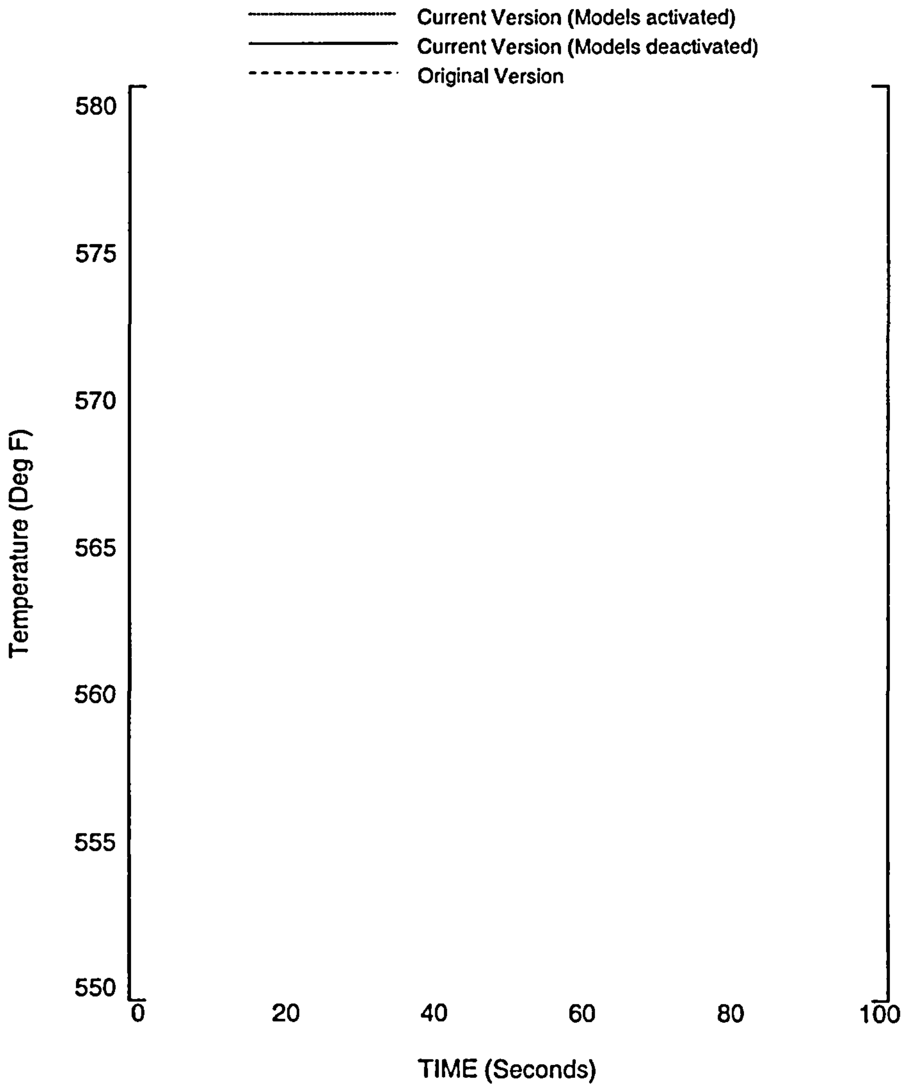


Figure 2.2.3.E

Hot Leg Temperature

Subcritical CEA Withdrawal Event for Plant D

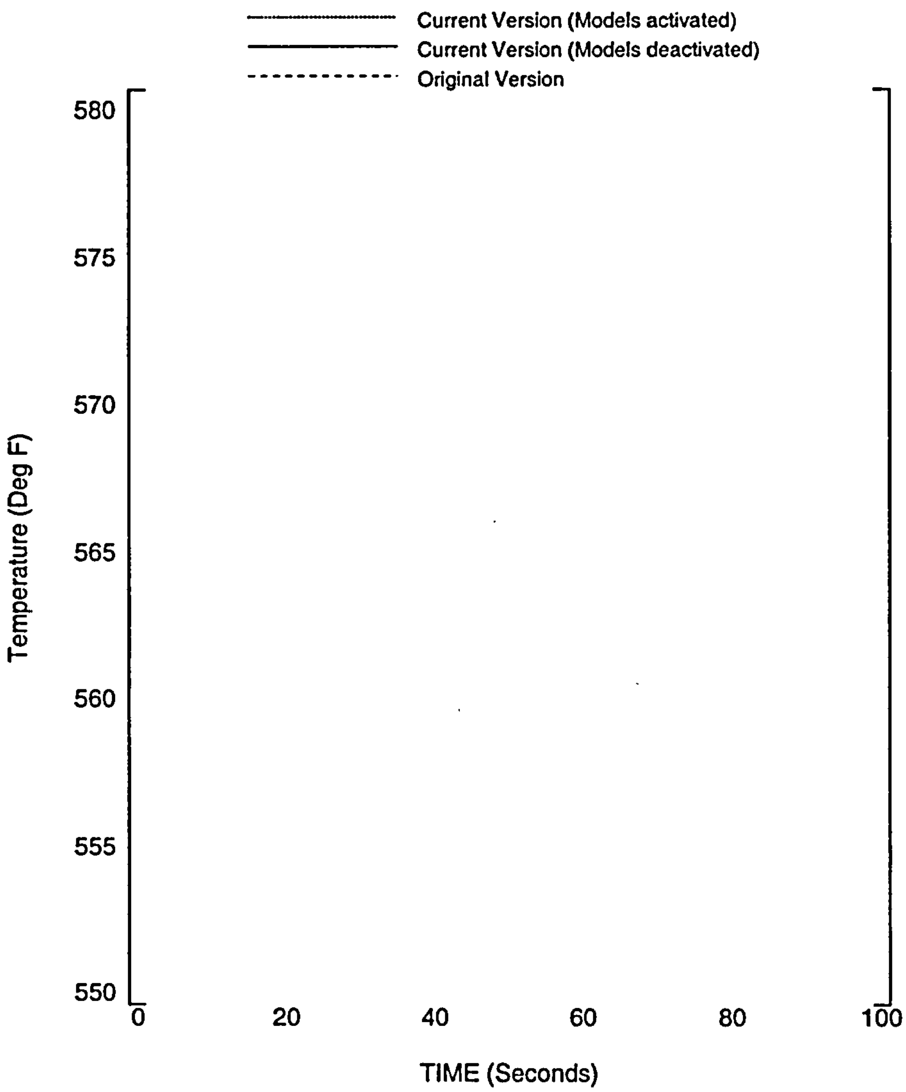


Figure 2.2.3.F

Steam Generator Pressure

Subcritical CEA Withdrawal Event for Plant D

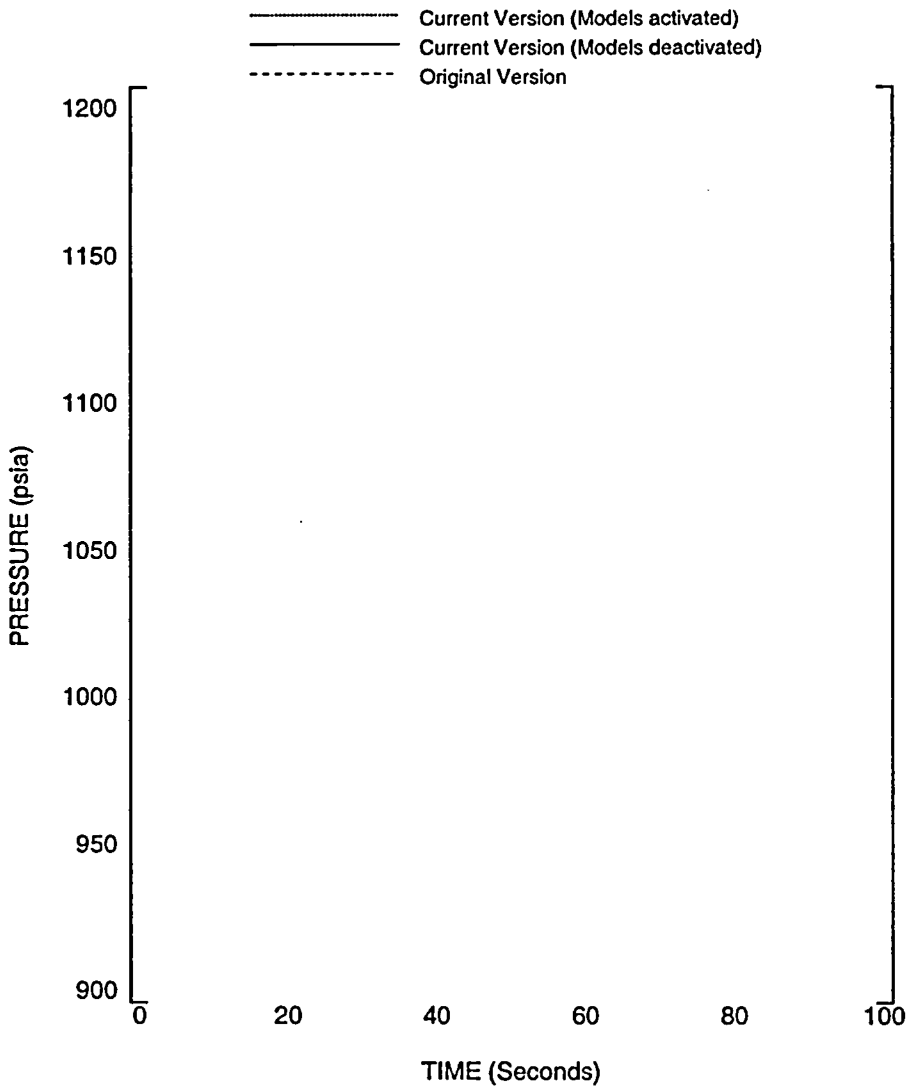


Figure 2.2.3.G

Total Reactivity

Subcritical CEA Withdrawal Event for Plant D

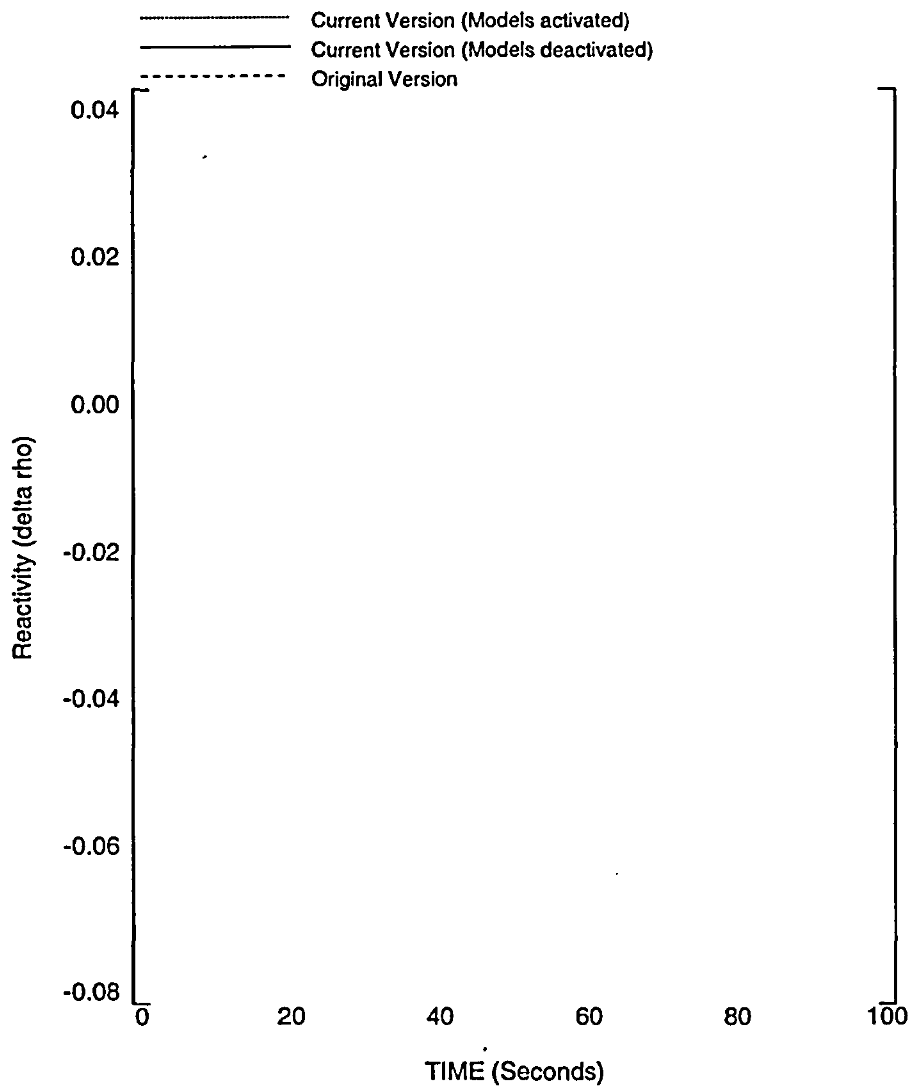


Figure 2.2.3.H

Total Reactivity

Subcritical CEA Withdrawal Event for Plant D

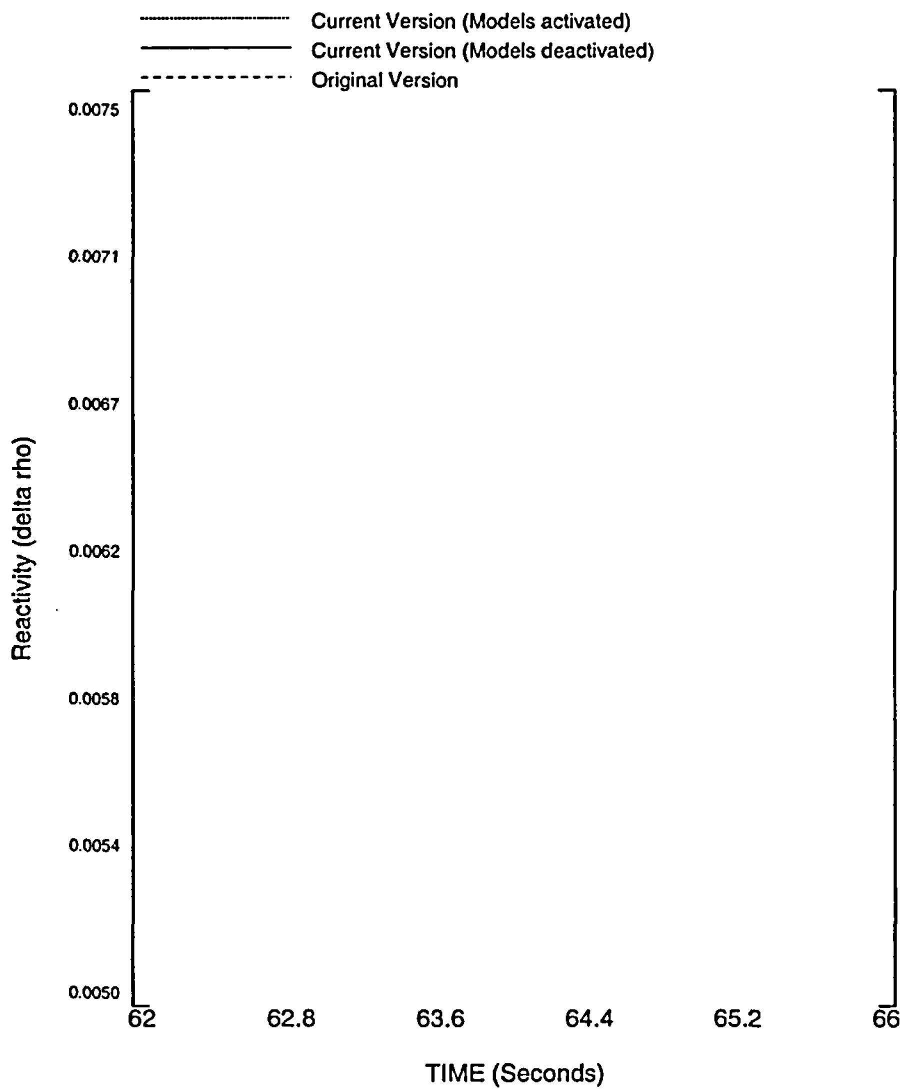


Figure 2.2.3.l

Moderator Reactivity

Subcritical CEA Withdrawal Event for Plant D

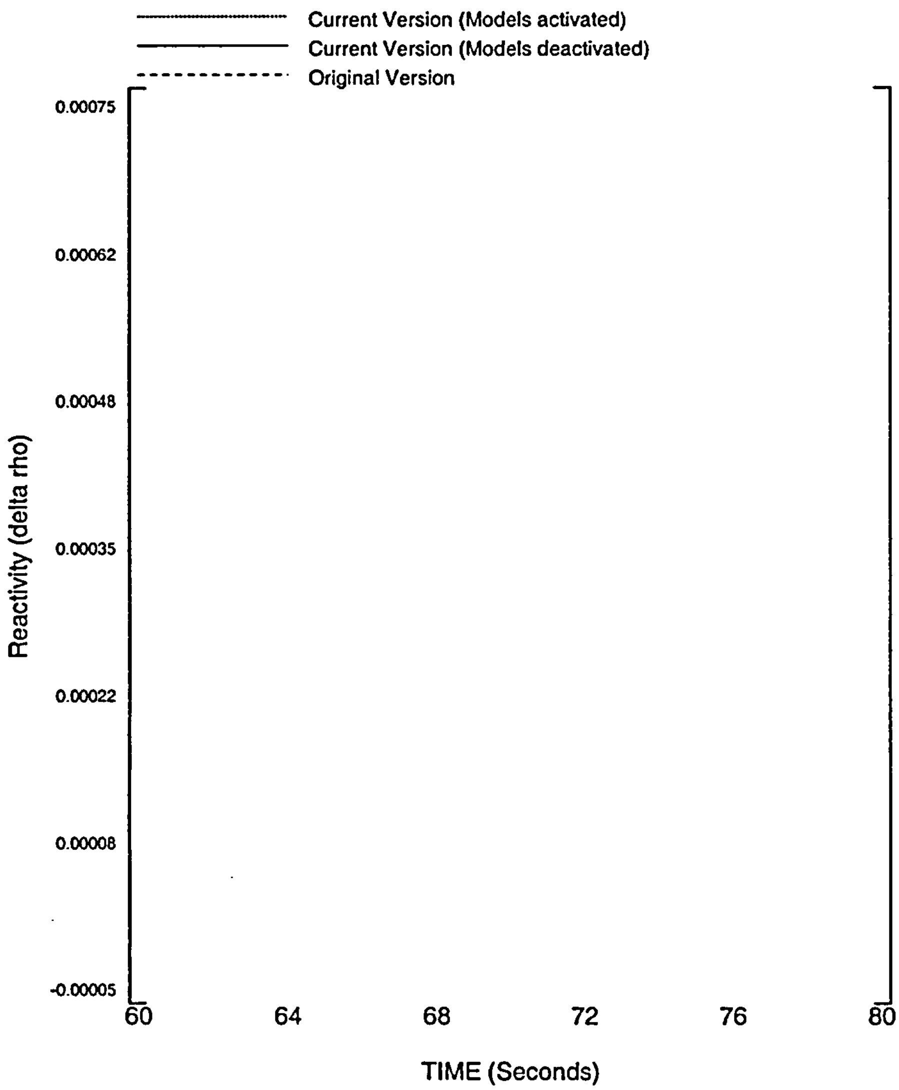


Figure 2.2.3.J

Doppler Reactivity

Subcritical CEA Withdrawal Event for Plant D

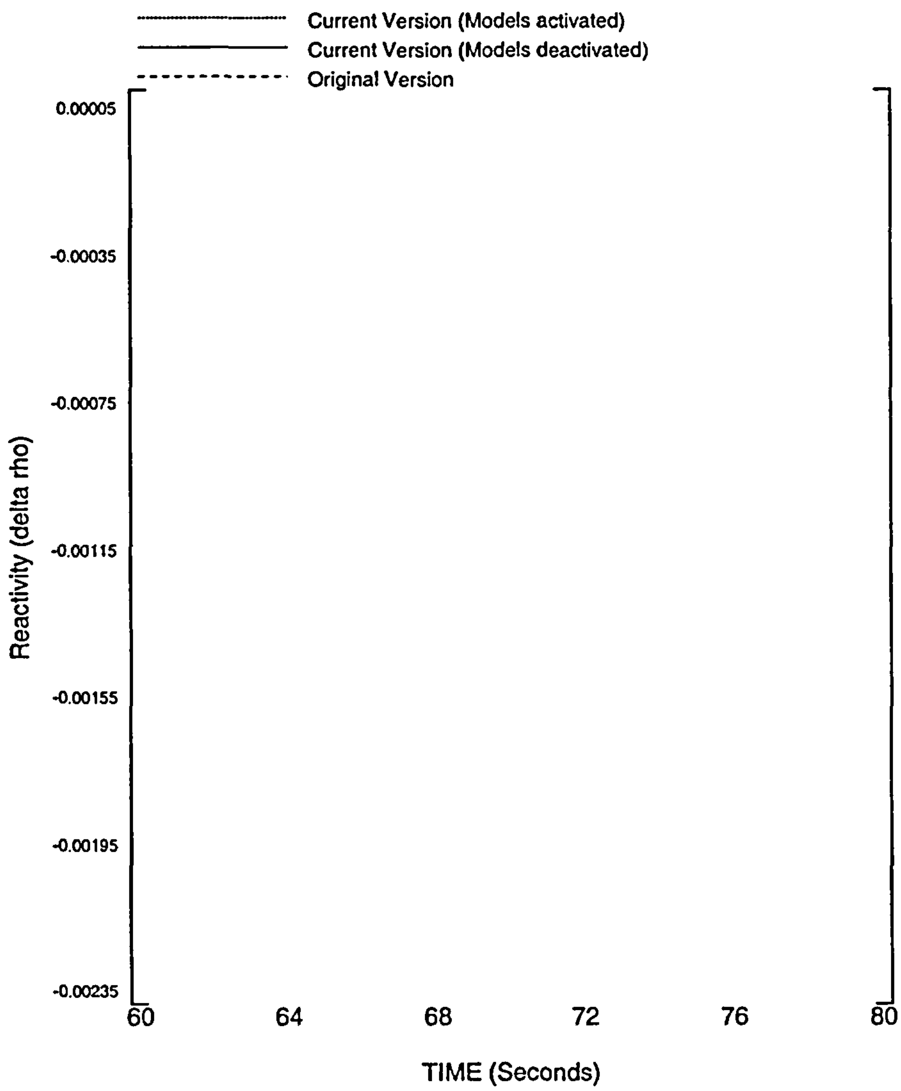


Figure 2.2.3.K

Reactor Core Power

Subcritical CEA Withdrawal Event for Plant D

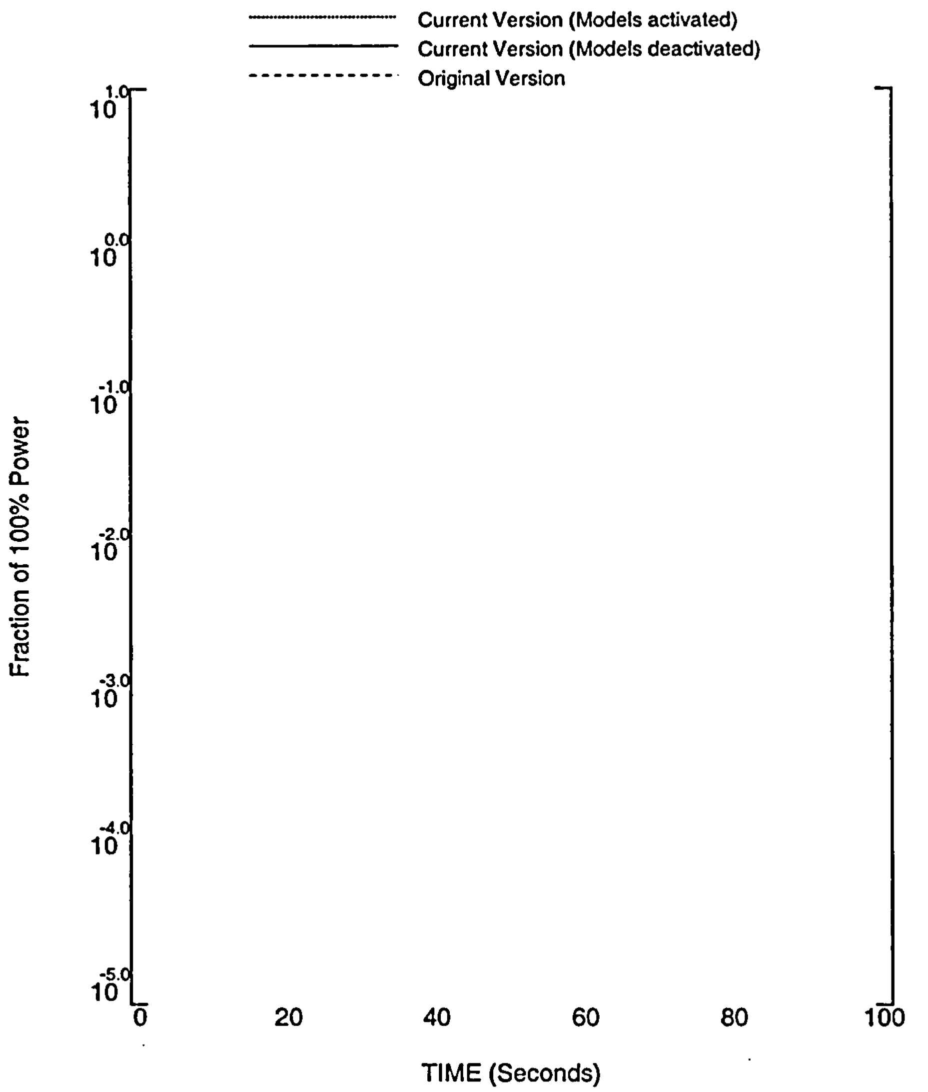


Figure 2.2.4.A

Reactor Core Power

CEA Withdrawal Event from Hot Zero Power for Plant D

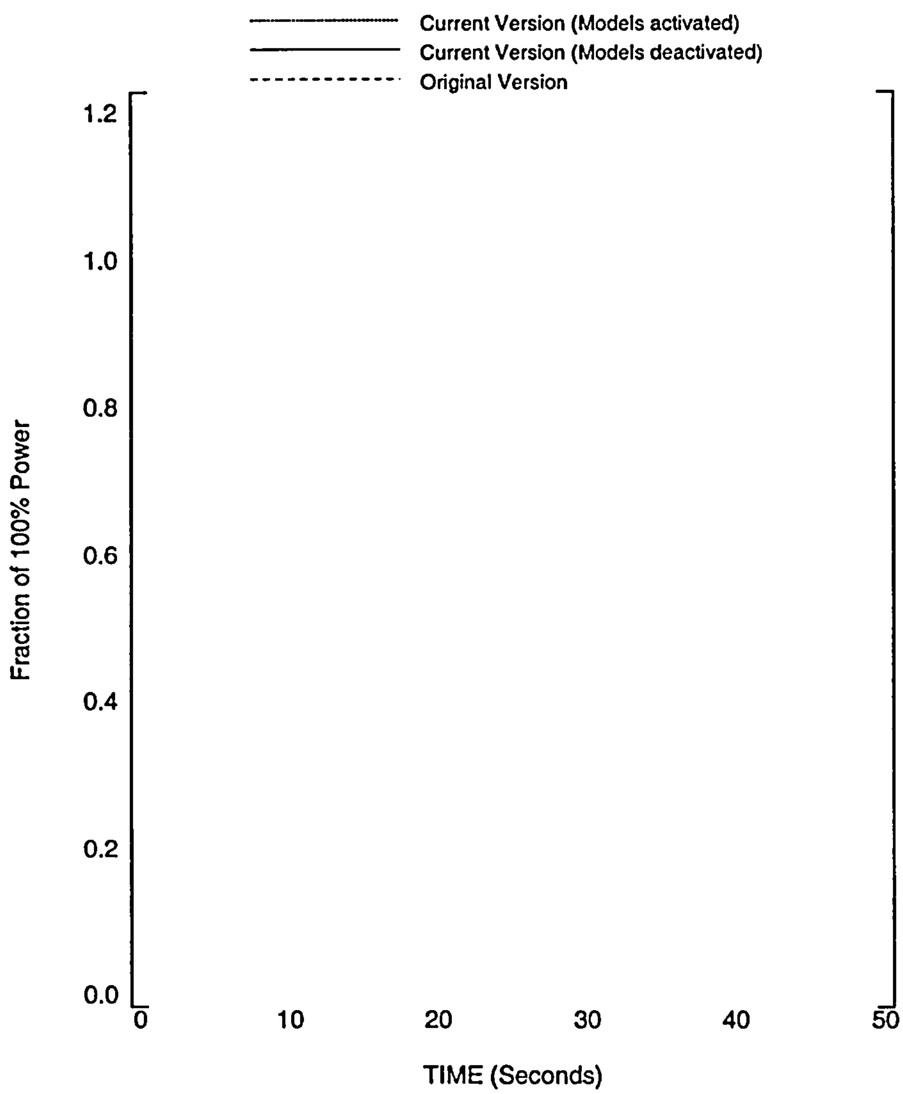


Figure 2.2.4.B

Reactor Core Heat Flux

CEA Withdrawal Event from Hot Zero Power for Plant D

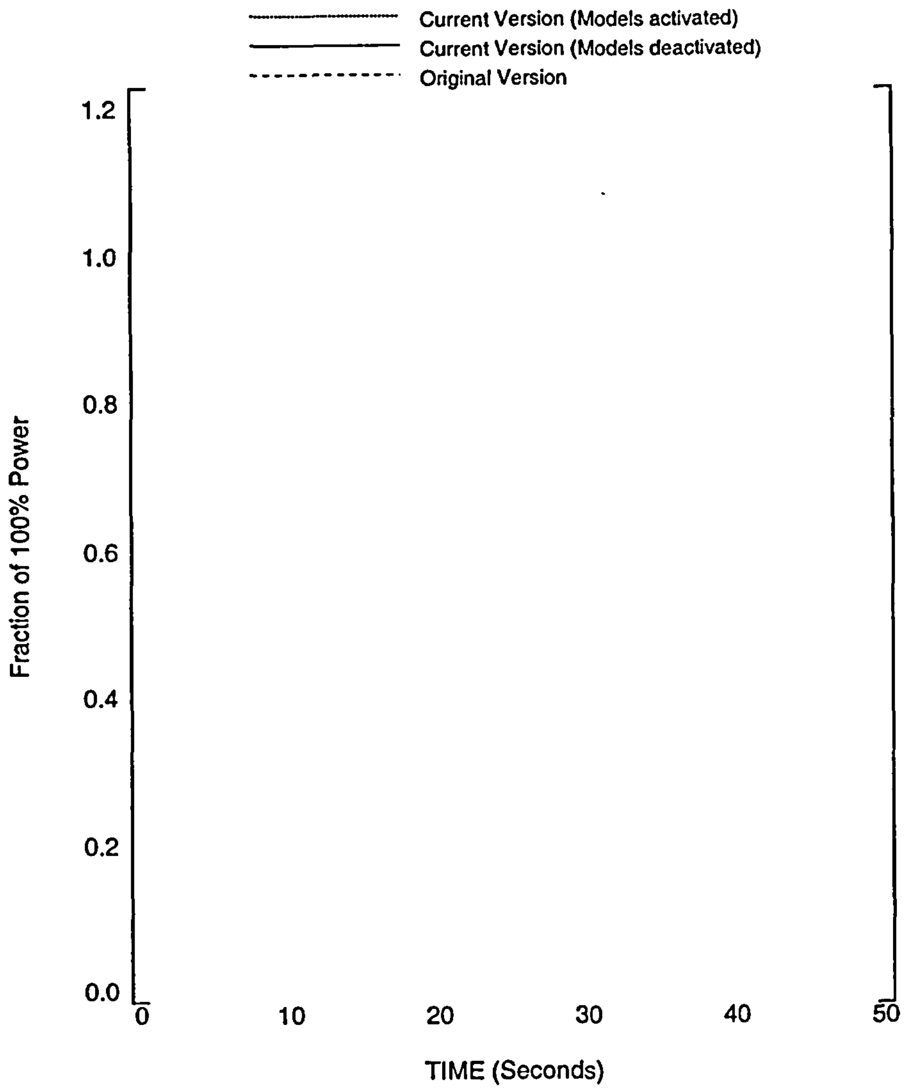


Figure 2.2.4.C

Pressurizer Pressure

CEA Withdrawal Event from Hot Zero Power for Plant D

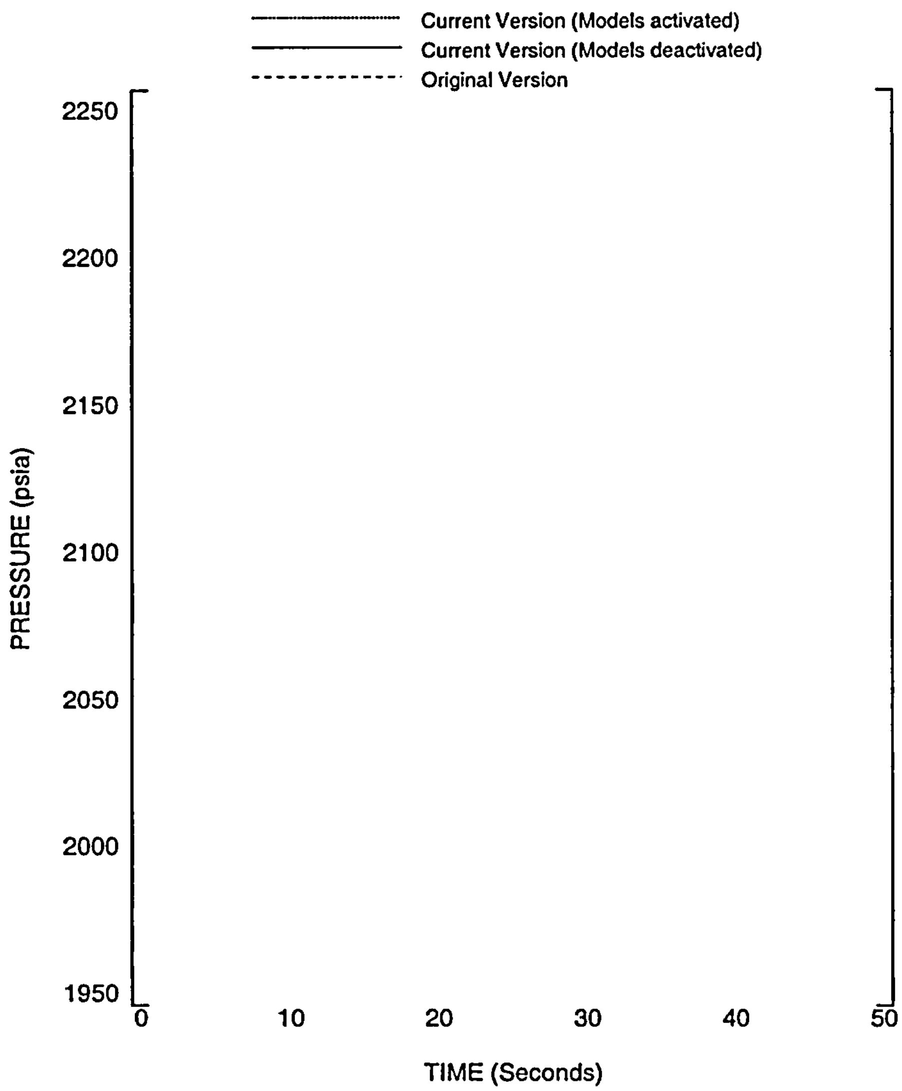


Figure 2.2.4.D

Core Average Temperature

CEA Withdrawal Event from Hot Zero Power for Plant D

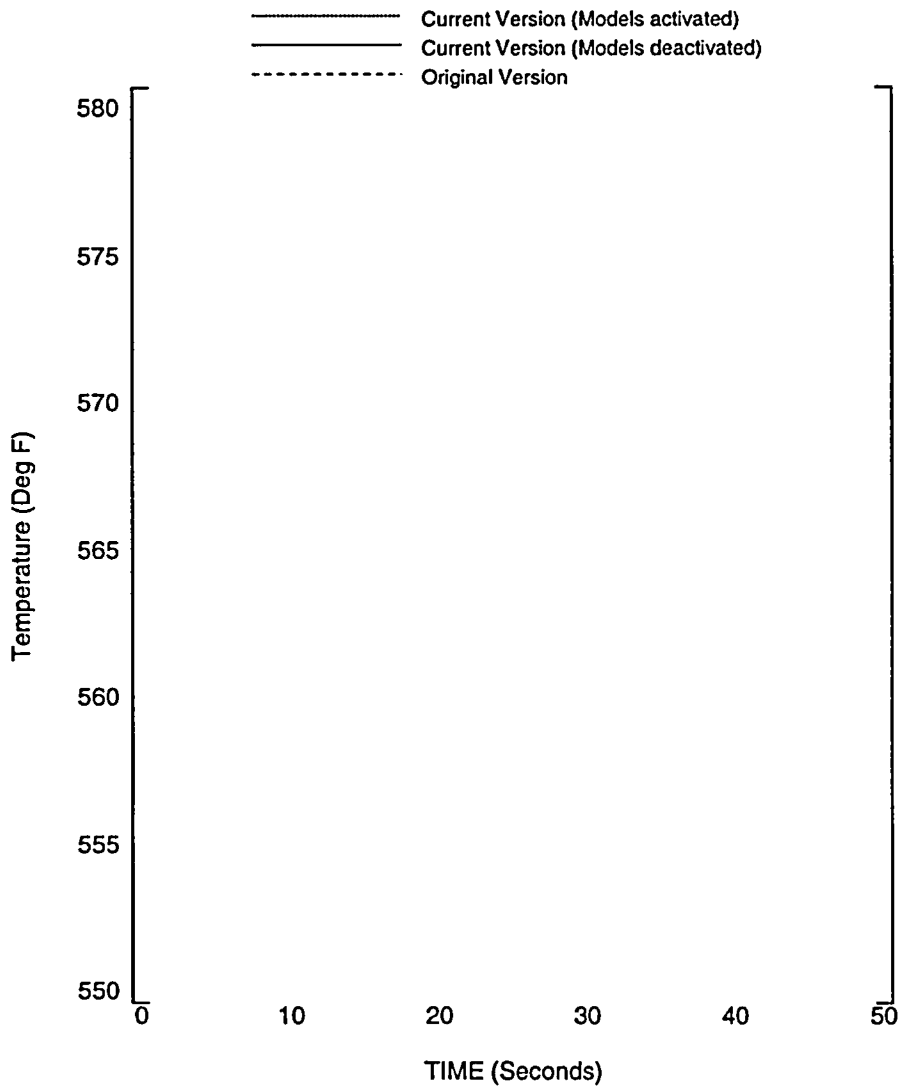


Figure 2.2.4.E

Hot Leg Temperature

CEA Withdrawal Event from Hot Zero Power for Plant D

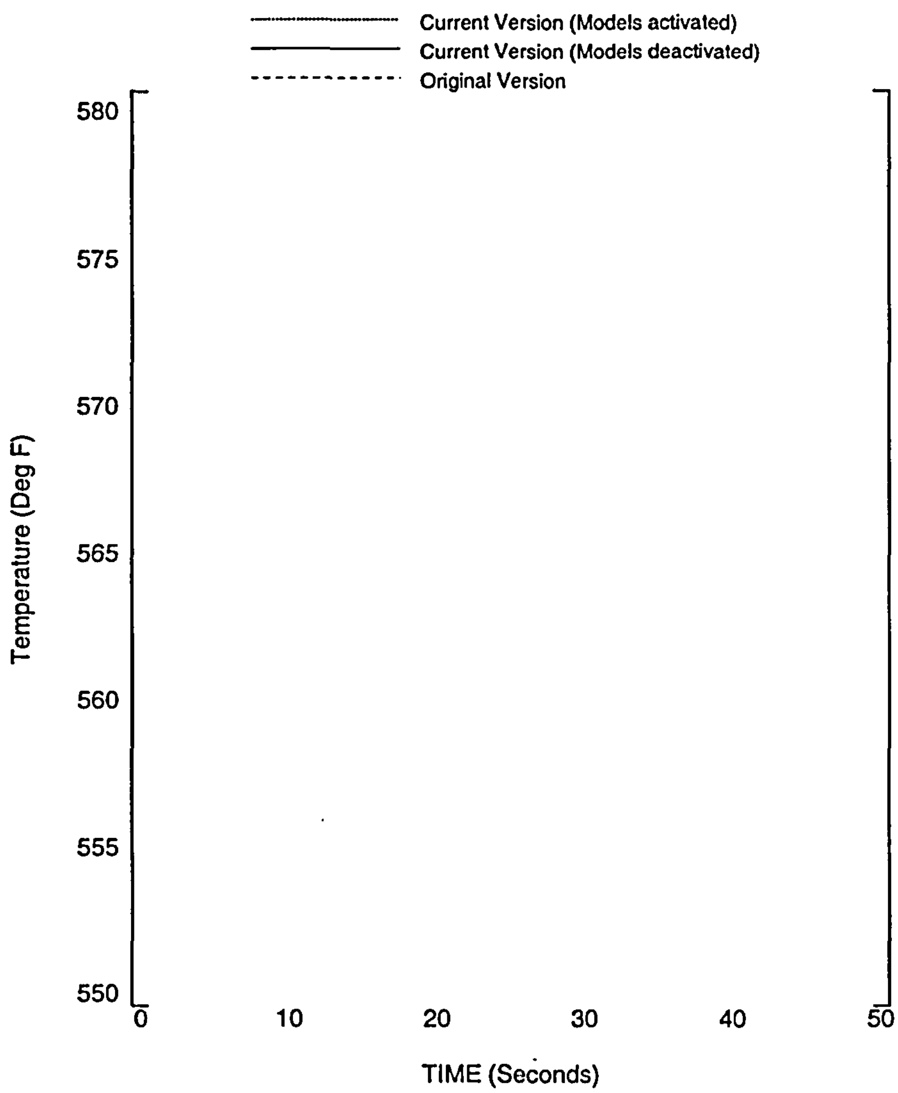


Figure 2.2.4.F

Steam Generator Pressure

CEA Withdrawal Event from Hot Zero Power for Plant D

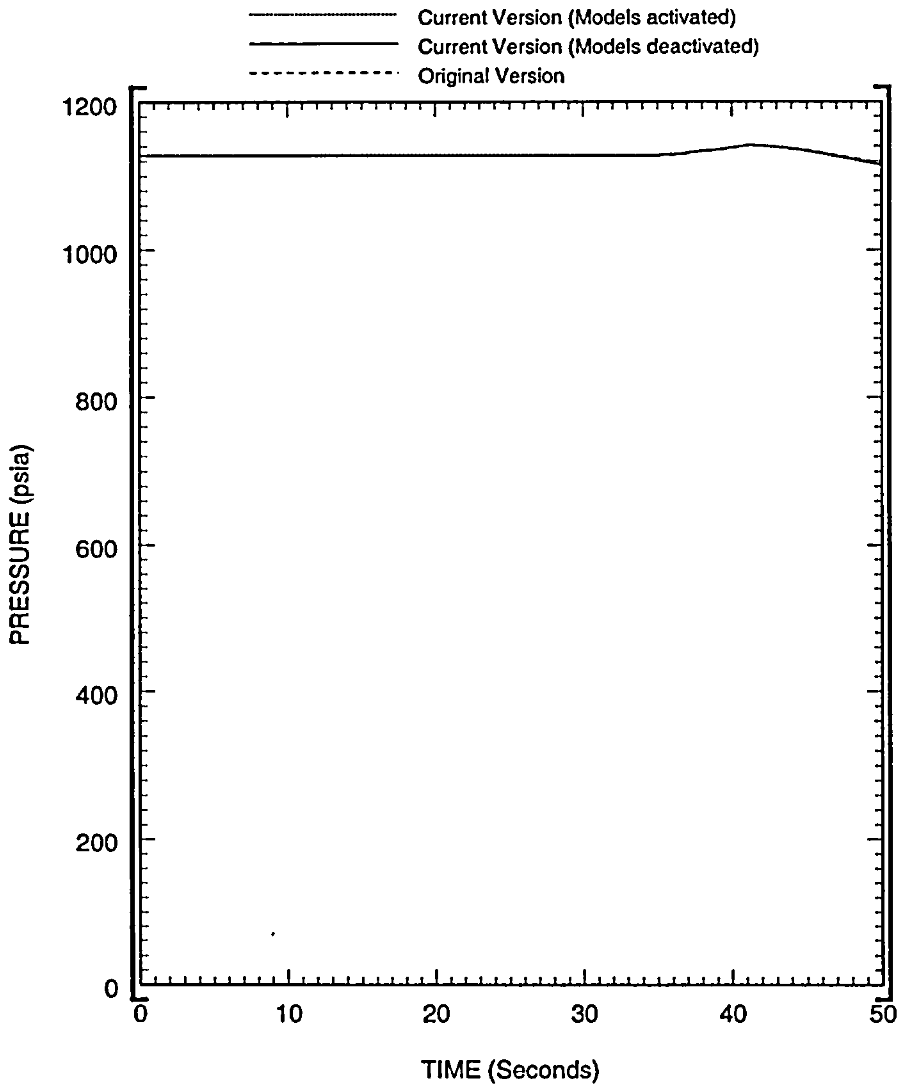


Figure 2.2.4.G

Total Core reactivity

CEA Withdrawal Event from Hot Zero Power for Plant D

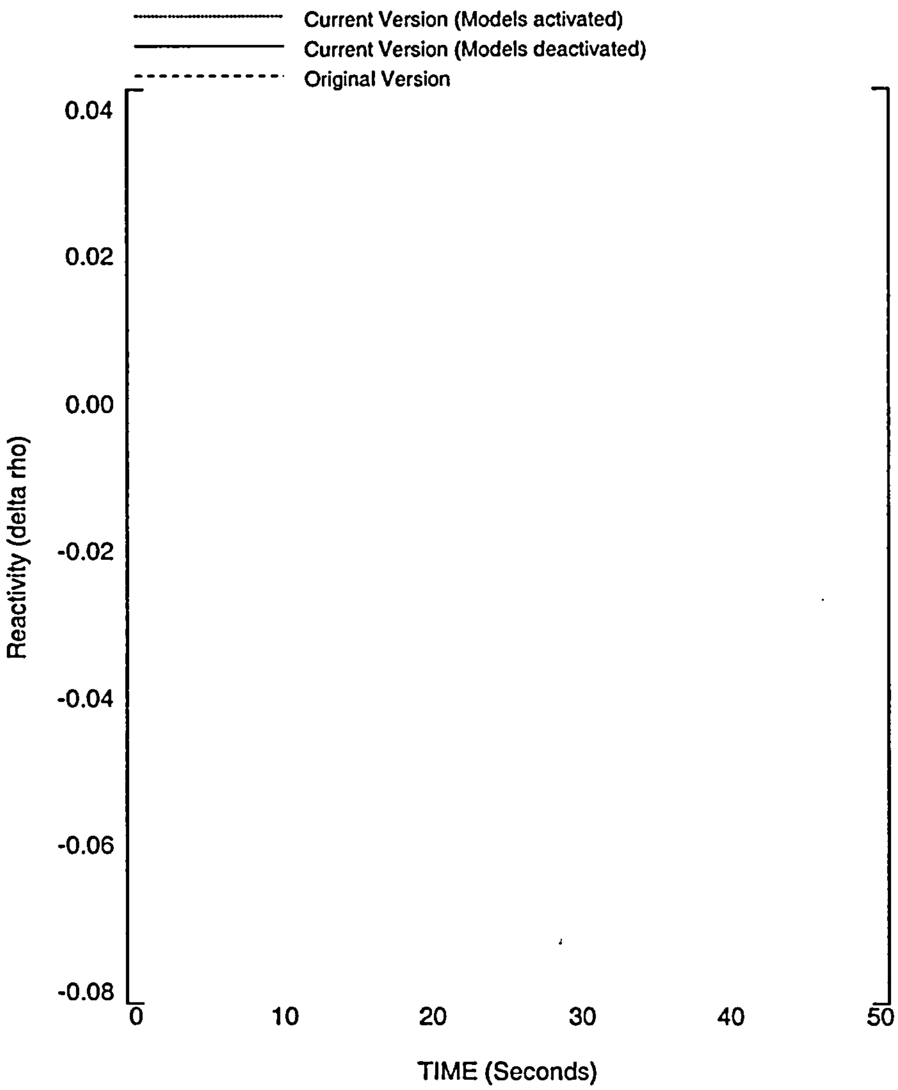


Figure 2.2.4.H

Total Core reactivity

CEA Withdrawal Event from Hot Zero Power for Plant D

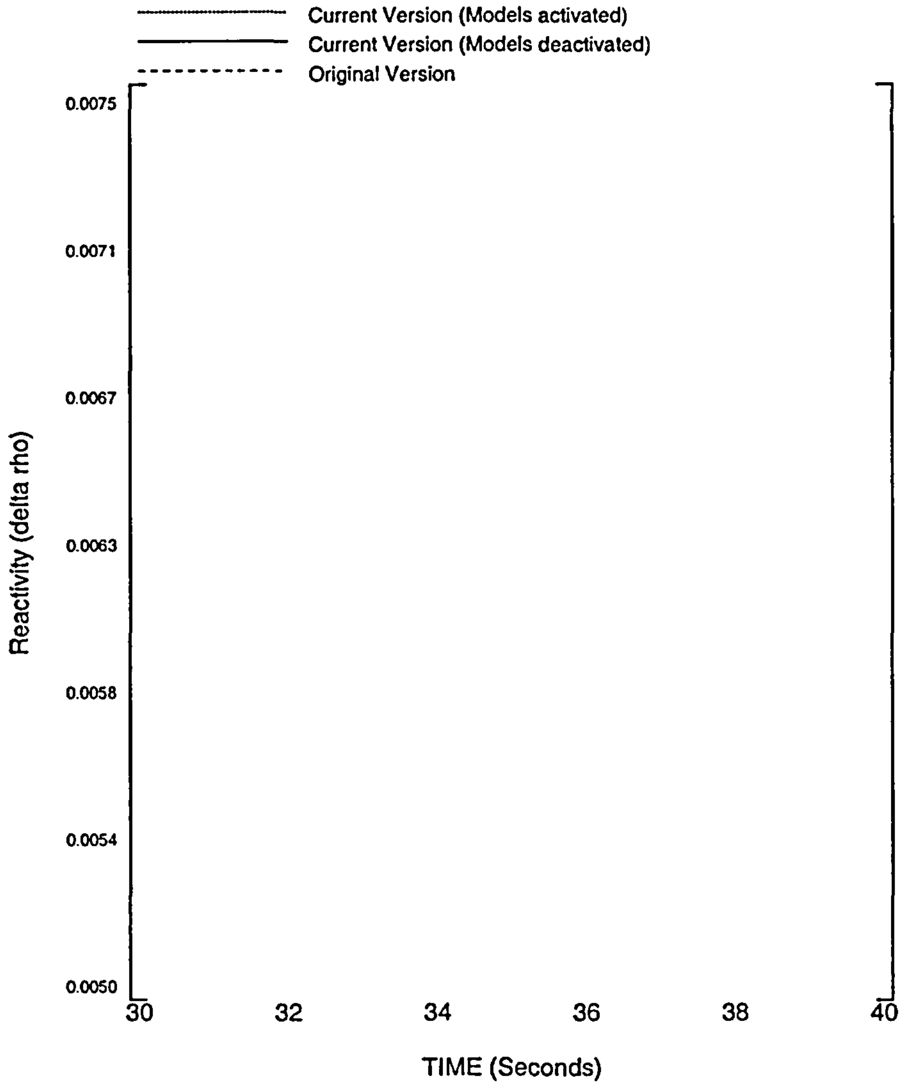


Figure 2.2.4.1

Moderator Reactivity

CEA Withdrawal Event from Hot Zero Power for Plant D

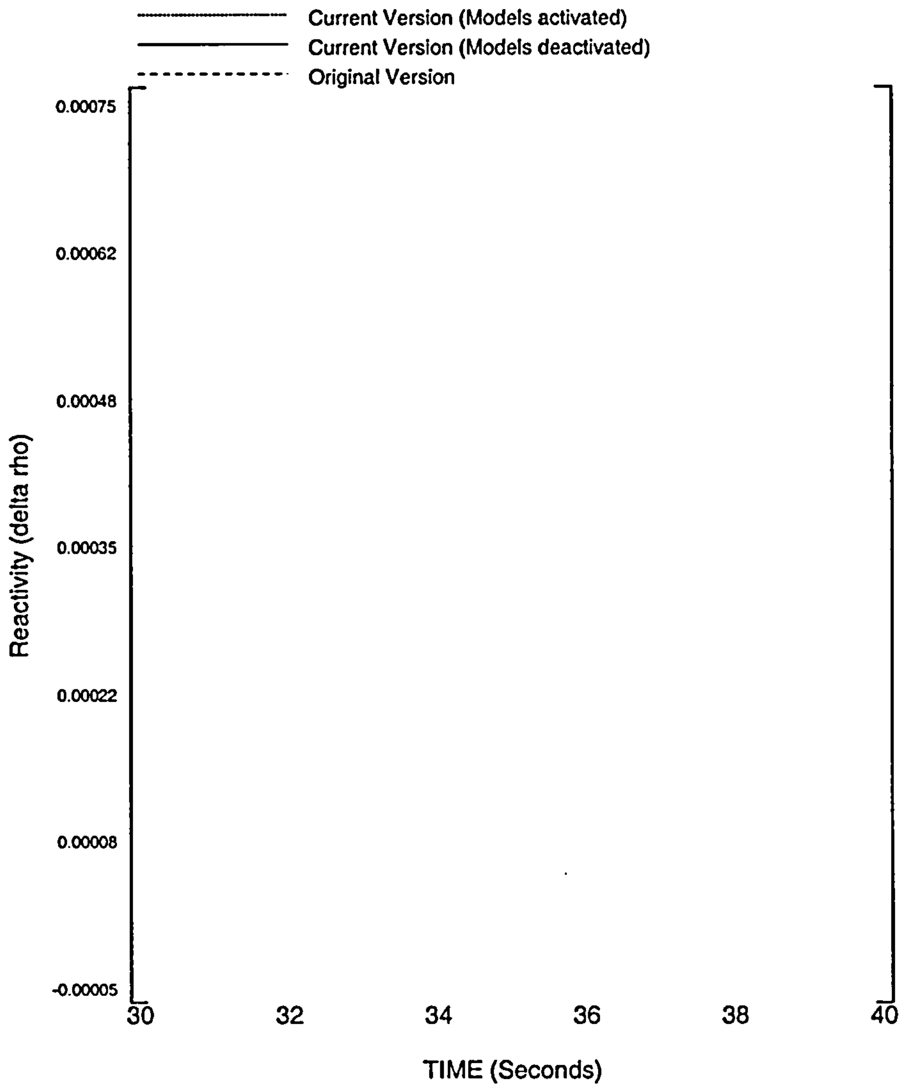


Figure 2.2.4.J

Doppler Reactivity

CEA Withdrawal Event from Hot Zero Power for Plant D

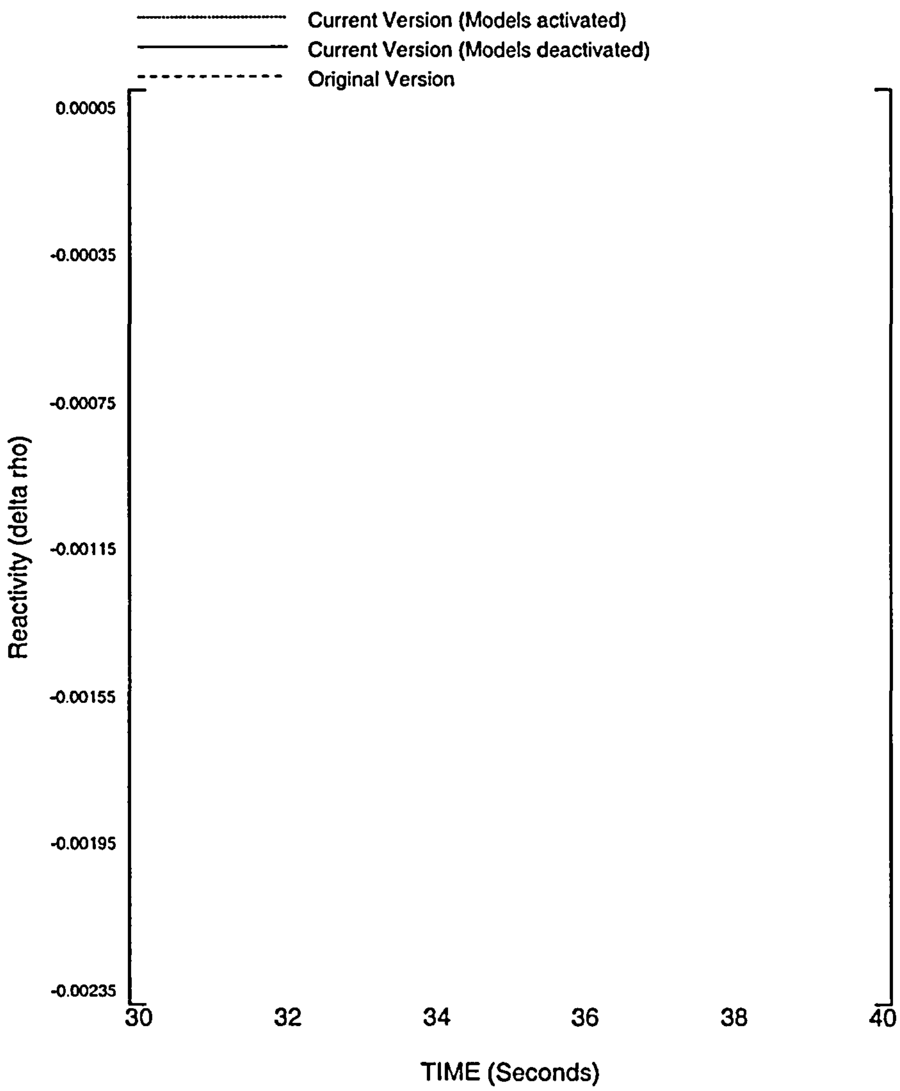
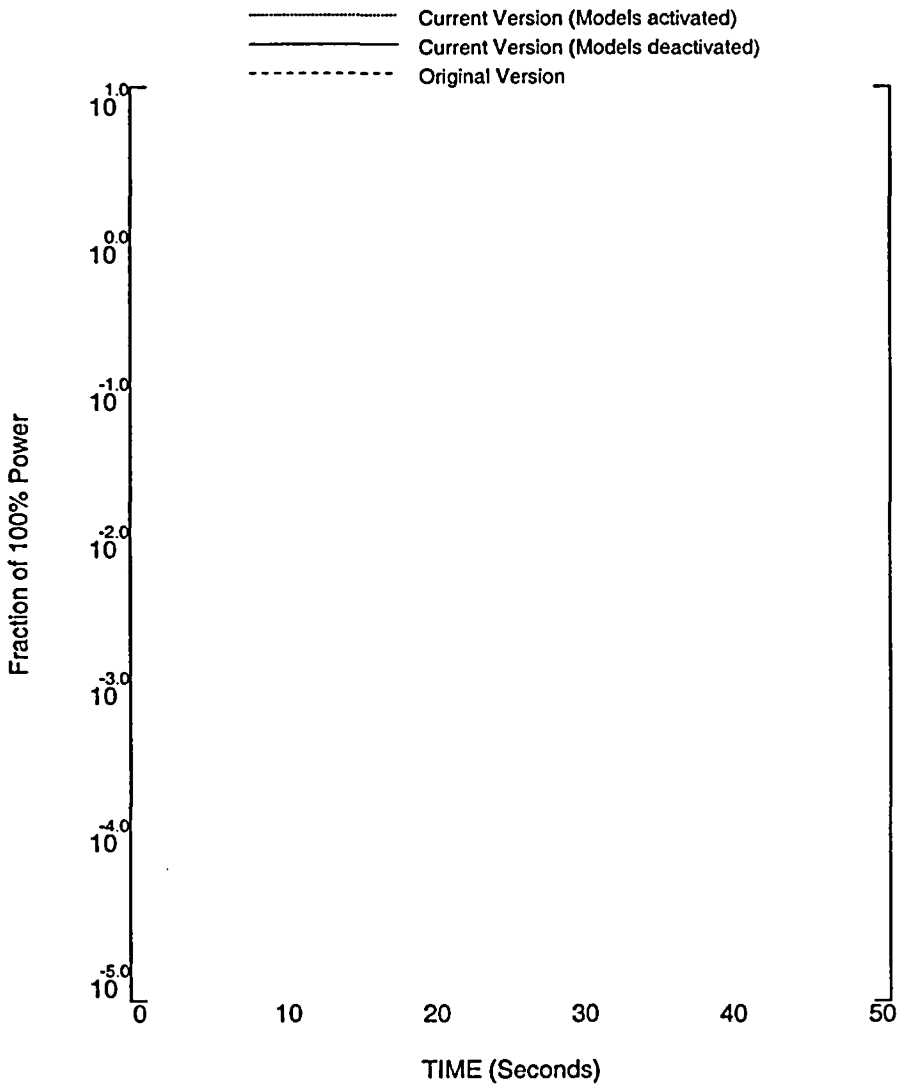


Figure 2.2.4.K

Reactor Core Power

CEA Withdrawal Event from Hot Zero Power for Plant D



# 3.0 Benchmark Comparisons for Plant E, 3026 MWt

## 3.1 Discussion

Four (4) events were analyzed for Plant E. For two (2) events, Seized Reactor Coolant Pump (RCP) Rotor and a Steam Generator Tube Rupture (SGTR), the first comparison made for each event compared the results of the upgraded CENTS version with the results of the original CENTS version. The discussions of these comparisons are in Sections 3.2.1 and 3.2.2. The figures for each of these two events show the three-way comparison of original code version to upgraded code version with upgrades deactivated and activated. These comparisons showed no significant differences. The presentation of the SGTR event includes a detailed discussion of the results of using the new CENTS dose model, which tracks radio-nuclides throughout the RCS and secondary plant to the atmosphere (WCAP-15996-P, Volume 1, Section 5.8).

The final two events analyzed for Plant E are a MSLB and a FWLB. A detailed set of inputs for the CENTS Main Feedwater (MFW) System model has been developed for this plant. In addition, Plant E also has a RELAP5.3 model of the feedwater and condensate systems. Thus, the CENTS MFW model has been benchmarked against RELAP5.3. This is discussed in Sections 3.4.1 and 3.4.2. Lastly, an methodology change in the FWLB event has been analyzed, namely placing the feedring at its actual physical elevation in the steam generator (instead of artificially at the tubesheet elevation), thereby allowing a steam blowdown once the water level drops below the elevation of the feedring.

### 3.1.1 Plant Description

Plant E is a Combustion Engineering NSSS design licensed to operate at 3026 MWt. Plant E has undergone an extended power uprate from it's originally licensed 2815 MWt power level to its current licensed power rating. As is typical for Combustion Engineering NSSS designs, Plant E has two (2) independent primary coolant loops each of which has two (2) reactor coolant pumps, a steam generator, a hot leg and two (2) cold legs (which includes the RCP suction and discharge legs). Thus in general arrangement the Plant E NSSS is identical to that of Plant D discussed in Section 2.1.1 and shown in Figure 2.1.1. In the discussion below, only the major differences between Plant E and Plant D are highlighted.

Reactor Core

The enriched $UO_2$ fuel is held in 177 fuel assemblies, each assembly consisting of a 16x16 matrix of fuel rods.

Steam Generators

The steam generators, like those of Plant D are of the vertical U-tube design. They are, however, replacement steam generators of Westinghouse design.

Reactor Protection System

The operation of the RPS is essentially the same as that for Plant D.

Operating Restrictions

The operating restrictions imposed by the Plant E TSs are similar in content to those imposed on Plant D.

## 3.2 Comparison of Upgraded (Upgrades Deactivated) to Original CENTS Code Versions

### 3.2.1 Seized Rotor

Discussion of Event

Following seizure of a reactor coolant pump shaft, the core flow rate rapidly decreases to the value which occurs with only three (3) of the RCPs in operation. The reduction in core flow with the associated increase in core coolant inlet temperature will reduce the margin to the DNB safety limit.

The analysis of the single Seized Rotor event assumes that a reactor coolant pump stops instantaneously. For this event, asymmetric steam generator tube plugging is assumed, as this provides the limiting condition for flow coastdown and minimum DNBR.

For Plant E, the event is terminated by a CPCS Low Primary Coolant Pump Shaft Speed Trip. Note that for this comparison, the same trip time was used for both the original and upgraded CENTS versions.

Table 3.2.1.A contains a listing of the important assumptions and initial conditions for this case. These assumptions were used in setting up the case data for CENTS.

Results

Table 3.2.1.B provides a comparison of the sequence of events for the Seized Rotor event. Figures 3.2.1.A through 3.2.1.I provide comparisons of important parameters calculated by the original, upgraded (models deactivated) and upgraded (models activated) CENTS versions.

The comparison of trends between the upgraded and original CENTS versions is excellent. The minor differences in the transient results are mostly a consequence of the slight differences in the initial conditions calculated at time zero, which are a result of establishing the steam generator asymmetric tube plugging conditions prior to commencing the event. See (Figures 3.2.1.E & F).

## Table 3.2.1.A

**Important Assumptions**
**Seized Rotor**

| Parameter | Initialization Value | Original CENTS Version: Asymmetric Steady State(61 seconds) | Upgraded CENTS Version: Asymmetric Steady State (58 seconds) |
|---|---|---|---|
| Core Power | 102% of 3026 Mwt | 1.0179 | 1.0168 |
| Moderator Temperature Coefficient | $-0.2 \times 10^{-4}\ \Delta\rho/°F$ | -.-.- | -.-.- |
| Scram Worth | $5.0 \times 10^{-2}\ \Delta\rho$ | -.-.- | -.-.- |
| Delayed Neutron Fraction | Cycle Maximum | -.-.- | -.-.- |
| Initial Core Inlet Temperature | 558°F | 557.8°F | 558.2°F |
| Pressurizer Pressure | 2324 psia | 2324.3 psia | 2324.6 psia |
| Pressurizer Pressure Control System | Manual | -.-.- | -.-.- |
| Turbine Bypass System | Inoperable | -.-.- | -.-.- |
| Steam Generator Tubes Plugged | Asymmetric Plugging<br>SG #1: 10%<br>SG #2: 0% | -.-.- | -.-.- |

## Table 3.2.1.B
## Sequence of Events for Seized Rotor

| Time(Sec) | | Event | Value | |
|---|---|---|---|---|
| Upgraded Version (Upgrades deactivated) | Original Version | | Upgraded Version (Upgrades deactivated) | Original Version |
| | | Seizure of a single reactor coolant pump shaft | | |
| | | CPCS Low reactor coolant pump speed reactor trip condition, Fraction of initial | | |
| | | Reactor trip breakers open | | |
| | | CEAs start to fall | | |
| | | Main steam safety valves begin to open, PSIA | | |
| | | Peak primary system pressure, PSIA | | |
| | | Peak Steam Generator Pressure, PSIA | | |

### 3.2.2 Steam Generator Tube Rupture

Discussion of Event

The Steam Generator Tube Rupture (SGTR) accident is a penetration of the barrier between the RCS and the main steam system, which results from the failure of a steam generator U-tube. Integrity of the barrier between the RCS and main steam system is significant from a radiological release standpoint. The radioactivity from the leaking steam generator mixes with the shell-side water in the affected steam generator. A fraction of the radioactive inventory which leaks into the affected steam generator is subsequently released to atmosphere.

A SGTR event results in a depressurization of the RCS. For this scenario, the SGTR is accompanied by a simultaneous Loss of AC power. A reactor trip is generated by a CPCS Low Primary Coolant Pump Shaft Speed Trip signal.

A single SGTR case was simulated. Table 3.2.2.A contains a listing of the important assumptions for this case. These assumptions were used in setting up the input data for CENTS. Note that the assumption of a Loss of AC at the time of the tube rupture (time = 0.0 sec) was chosen in this scenario to maximize the amount of steam release to the environment. If some other trip condition were used to trip the reactor (and turbine), then much of the steam (and radio-nuclides) would be routed to the condenser via the turbine, instead of to the atmosphere via the main steam safety valves.

This section provides the comparison of the original to the upgraded CENTS version (upgrade models deactivated). Note that an upgraded SGTR model (WCAP-15996-P, Volume 1, Section 5.7) is fully integrated into the upgraded CENTS version. Thus, that model is part of the comparison to the original CENTS version.

## Results

Table 3.2.2.B provides a comparison of the sequence of events for the SGTR Event. Figures 3.2.2.A through 3.2.2.S provide comparisons of important parameters as calculated by the original and upgraded versions of CENTS. The comparison of trends between the original and upgraded CENTS versions is excellent.

The slightly lower total mass and higher enthalpy of the fluid transferred to the ruptured steam generator is the result of the calculation with the new SGTR model which calculates flow from both the hot side and from the cold side of the tubes. The original CENTS model allowed flow from one node only. In this comparison, for the original CENTS version case, the hot side tube node was chosen. The enthalpy is the exit enthalpy, which is approximately the same as the tube average (or RCS average) temperature.

For the upgraded CENTS version with the new SGTR model, the flow is coming from the hot and cold side steam generator plenums. Thus the break flow enthalpy is the flow weighted average of the hot and cold side enthalpies. Since the break location chosen is at the hot side tube sheet, the break flow from the hot side is calculated to be approximately three (3) times that from the cold side, due to the extensive line losses from the cold side. Thus, the total break flow enthalpy is higher for the upgraded CENTS version, while the flow rate is slightly lower (Figures 3.3.2.J through M).

The total mass of steam released to atmosphere calculated by the two code versions differed by about 0.5%. As shown in Table 3.2.2.B, the steam released to atmosphere from the two steam lines differed more. This is due to different MSSV cycling response.

## Table 3.2.2.A
## Important Assumptions for the Steam Generator Tube Rupture

| Parameter | Value |
|---|---|
| Core Power | 102% of 3026 MWt |
| Moderator Temperature Coefficient | $-3.8 \times 10^{-4}$ $\Delta\rho$/ °F |
| Scram Worth | $5.0 \times 10^{-2}$ $\Delta\rho$ |
| Delayed Neutron Fraction | Cycle Maximum |
| Core Inlet Temperature | 556.7 °F |
| Pressurizer Pressure | 2300 PSIA |
| Pressurizer Level Control System | Lost with LOAC, Charging back on with SIAS |
| Pressurizer Pressure Control System | Lost with LOAC, Proportional Heaters back on after SIAS |
| Turbine Bypass System | Inoperable |
| Loss of Offsite Power | Concurrent with tube rupture |

## Table 3.2.2.B
## Sequence of Events for the Steam Generator Tube Rupture

| Time(Sec) | | Event | Value | |
|---|---|---|---|---|
| Upgraded Version (Upgrades deactivated) | Original Version | | Upgraded Version (Upgrades deactivated) | Original Version |
| | | Double ended rupture of a steam generator tube, $in^2$ with concurrent Loss of AC power | | |
| | | CPCS Low reactor coolant pump speed reactor trip condition, Fraction of initial | | |
| | | Reactor trip signal generated | | |
| | | Safety injection actuation, PSIA | | |
| | | Mass of primary coolant transferred to the ruptured SG, lbm | | |
| | | Mass of steam released from steam line 1 to atmosphere, lbm | | |
| | | Mass of steam released from steam line 2 to atmosphere, lbm | | |
| | | Total mass of steam released from steam lines 1 and 2 to atmosphere, lbm | | |

## 3.3 Comparison of Upgraded Code Version — Upgrades Activated

The Seized Rotor and SGTR events are again analyzed for Plant E. This time all the upgraded CENTS model improvements are activated. These cases can then be compared to the upgraded CENTS model with the improvements deactivated.

### 3.3.1 Seized Rotor

Discussion of Event - See Section 3.2.1 & Table 3.2.1A

Results

Table 3.3.1.B provides a comparison of the sequence of events for the Seized Rotor event. Figures 3.2.1.A through 3.2.1.I provide comparisons of important parameters as calculated by the original and upgraded CENTS versions with upgrade models deactivated and activated.

The comparison of trends between the upgraded CENTS version (models deactivated) and (models activated) shows excellent agreement. The differences in the transient results are very small. The Core Heat Transfer upgrade causes virtually no change in this event. The four node steam generator tube model allows for slightly better heat transfer during the coastdown in flow, which lowers swell into the pressurizer. This in turn reduces peak core pressure by about 12 psi. The extended pressure vessel nodalization has a very slight opposite effect by differentiating the cold temperatures from each loop and changing the average node density in the pressure vessel downcomer region. Peak core pressure due to pressurizer swell increases about 2 psi due to this model. Overall, there is about a 10 psi drop in peak core pressure (Figure 3.2.1.C) with all models activated. Core inlet temperature actually increases very slightly, from 565.5°F to 565.8°F, though the peak occurs later, at 13.1 seconds vs. 10.95 seconds with the upgrade models deactivated (Figure 3.2.1.E).

## Table 3.3.1.B

## Sequence of Events for the Seized Rotor

| Time(Sec) | | | | Event | Value | | | |
|---|---|---|---|---|---|---|---|---|
| Upgrades deactivated | Upgrades Activated (Sequentially) | | | | Upgrades deactivated | Upgrades Activated (Sequentially) | | |
| | CHT | 4 SG Nodes | Detailed PV | | | CHT | 4 SG Nodes | Detailed PV |
| | | | | Seizure of a single reactor coolant pump shaft | | | | |
| | | | | Low reactor coolant flow reactor trip condition, Fraction of initial | | | | |
| | | | | Reactor trip breakers open | | | | |
| | | | | CEA's start to fall | | | | |
| | | | | Main steam safety valves begin to open, PSIA | | | | |
| | | | | Peak RCS Pressure, PSIA | | | | |
| | | | | Peak Steam Generator Pressure, PSIA | | | | |

### 3.3.2 Steam Generator Tube Rupture (SGTR)

Discussion of Event –

See Section 3.2.2 and Table 3.2.2A for a discussion of the thermal-hydraulic aspects of this SGTR scenario. In this section, in addition to activating the upgraded models, the CENTS dose model is discussed. Note that where $I_{131}$ is discussed, it refers to equivalent $I_{131}$.

The transport of equivalent $I_{131}$ throughout the RCS and secondary system has been tracked. The objectives of this analysis show:

(a) that the quantity of radio-nuclide is being properly conserved
(b) that it is correctly transported to different nodes and or portions of the plant
(c) that the flashing model and stripping factors are correctly applied and
(d) that the 2 hour and 8 hour doses rates are correctly calculated.

These objectives can be reached by using simple static or spreadsheet calculations over the one hour time span that this event scenario is analyzed. While a plant cooldown to shutdown cooling entry conditions is not part of this analysis, as it would be to determine the total 2 hr and 8 hr doses, the objective of analyzing the proper performance of the dose model is achieved by reviewing the 1 hour contribution toward the 2 and 8 hour doses. Transport of radio-nuclide gases (needed for determination of whole body doses) is identical to that of $I_{131}$ with the exception that when it leaks into the steam generator secondary, it immediately is transported to the steam generator steam space.

The above dose model objectives are reached in several ways. First, conservation of the $I_{131}$ radio-nuclide is determined by totaling the initial quantity throughout the NSSS and secondary systems at the beginning of the event. Any releases during the event are then added in and finally the total quantity of $I_{131}$ at the end of the event is determined for comparison. For simplicity, decay of $I_{131}$ is ignored by setting the decay constant to a very large number. Also, the removal rate by the Chemical Volume And Control System (CVCS) purification system is set to zero. Concentrations of $I_{131}$ within selected nodes are tracked throughout the event to show that transport is occurring smoothly and buildup or dilution is correct. The flashing model determines the amount of $I_{131}$ which is transported directly to the steam space of the steam generator when the hot RCS fluid leaks into the steam generator secondary and a portion of the fluid flashes to steam, based upon its enthalpy in relation to the secondary side liquid enthalpy. The following equation is used in a spreadsheet calculation to check the model accuracy.

$$X = (h_p - h_{fs}) / (h_{gs} - h_{fs})$$

where X = Flashing Fraction

$h_p$ = Enthalpy of the primary coolant

$h_{gs}$ = Enthalpy of the steam in the secondary system

$h_{fs}$ = Enthalpy of the liquid in the secondary system

With the flashing fraction, the tube rupture leak rate and the iodine concentration in the RCS upstream node, the amount iodine flashing directly to the steam generator steam space can be calculated each time step and integrated over the time of the run by spreadsheet. Similarly, the amount of iodine boiling off (with a DF of 100) as the steam generator produces steam can also become a spreadsheet calculation. By comparing the total amount of iodine in the secondary steam space at the beginning and end of the event, plus the amount released to the environment, the results of the CENTS dose model can be benchmarked to these hand calculations.

Both an event generated Iodine spike (GIS) and a pre-existing Iodine spike (PIS) are analyzed. Table 3.3.2.A provides the key dose assumptions for both the PIS and GIS cases. Note that a leak to the intact steam generator, initially at 1 gpm, is also modeled in this event. It is treated as a small slot break tube rupture. Where appropriate, when discussing the results of $I_{131}$ tracking and dose, this leakage is at times ignored.

## Results

### Thermal- Hydraulic Plant Response

Table 3.3.2.B provides a comparison of the sequence of events for the SGTR Event. Figures 3.2.2.A through 3.2.2.S provide comparisons of important parameters as calculated by the upgraded version of CENTS, with and without the upgrades activated. The comparison of trends is excellent.

The core heat transfer upgrade has virtually no affect on results for this event. The four node steam generator model does have an effect. The improved tube heat transfer modeling causes the RCS and steam generator secondary to reach equilibrium more quickly. Thus, early in the event, the RCS pressure and temperature drop more quickly. This, in turn causes greater safety injection flow rate which re-pressurizes the RCS more quickly, causing greater leak flow after about 600 seconds. By approximately 900 seconds, all cases are in a quasi-steady-state with the main steam safety valves cycling to remove energy from the system. The detailed pressure vessel nodalization alters affected and intact loop flows such that the steam releases from each of the steam generators is somewhat closer to equal.

### Radio-nuclide and Dose Results

Table 3.3.2C (PIS case) and Table 3.3.2D (GIS case) plus Figures 3.2.2.T through X [GIS] and 3.2.2.Y through AC [PIS] provide details of the CENTS dose model results. In general, there is excellent agreement between the CENTS results and the manual spreadsheet calculations, as shown in the Tables. Any minor variations are due to round-off or the fact that the spreadsheets use data at 1.0 second intervals, whereas CENTS is calculating output from every time step. All the $I_{131}$ is properly conserved throughout the event. In addition, the amount of $I_{131}$ flashing directly to the steam space was verified by manual spreadsheet calculation. For the PIS case, this calculation showed about 80 curies of direct flashing and another 5 curies generated through boiling in the liquid. These hand calculations under-predict the total amount of flashing and boil-off predicted directly from CENTS by approximately 10 curies.

Figure 3.2.2.Y provides a graphic representation of $I_{131}$ dilution in the various RCS locations. As expected, the pressurizer and pressure vessel upper head dilute most slowly since flow rates through these areas is low, particularly in natural circulation. Conversely, the cold discharge legs and pressure vessel downcomer dilute most rapidly since they are closest to the source of Safety Injection makeup water.

Figures 3.2.2.Z, AA, and AB provide details of how the $I_{131}$ migrates to the secondary systems. In Figure 3.2.2.AB, the difference between the amount of Iodine exiting the RCS and residing in the steam generators and main steam header is the RCP seal leakage and the amount that enters the atmosphere via the MSSVs.

Lastly, Figure 3.2.2.AC shows the buildup of dose over the one hour time span presented in this scenario. Each step rise in dose corresponds to an opening of the MSSVs.

The same basic figures for the GIS case are presented in Figures 3.2.2.T through X. The major differences for Figure 3.2.2.T are that Iodine concentration builds up from low levels as the core releases Iodine to the coolant. This time, the low flow regions of the RCS buildup Iodine concentrations most slowly, whereas the core node has always the highest concentration as the source node for the Iodine. In general, after one hour, the GIS case results in much lower $I_{131}$ concentrations throughout the RCS and secondary. Doses are less than a third those for the PIS case.

## Table 3.3.2.A

## Important Dose Related Assumptions
## Steam Generator Tube Rupture

| Dose Related Parameter | GIS Case | PIS Case |
|---|---|---|
| Initial RCS $I_{131}$ concentration, $\mu$Ci/gm | 1.0 | 60.0 |
| Initial RCS Noble Gas concentration, 1/E-bar $\mu$Ci/gm | 1.667 | 100.0 |
| Initial SG steam space & Main Steam header $I_{131}$ concentration, $\mu$Ci/gm | .001 | .001 |
| Initial SG liquid $I_{131}$ concentration, $\mu$Ci/gm | .1 | .1 |
| SG decontamination (or stripping factor) | .01 | .01 |
| Breathing Rate, $m^3$/sec | $3.47 \times 10^{-4}$ | $3.47 \times 10^{-4}$ |
| $I_{131}$ decay constant (no decay assumed) | $1 \times 10^{10}$ | $1 \times 10^{10}$ |
| X/Q @ site boundary, sec/$m^3$ (2 hr) | $6.5 \times 10^{-4}$ | $6.5 \times 10^{-4}$ |
| X/Q @ low population zone, sec/$m^3$ (8 hr) | $3.1 \times 10^{-5}$ | $3.1 \times 10^{-5}$ |
| Core $I_{131}$ release rate, Ci/sec (500 x pre-accident release rate) | 2.7855 | 0.0 |

## Table 3.3.2.B

### Sequence of Events
### Steam Generator Tube Rupture

| Time(Sec) | | | | Event | Value | | | |
|---|---|---|---|---|---|---|---|---|
| Upgrades deactivated | Upgrades Activated (Sequentially) | | | | Upgrades deactivated | Upgrades Activated (Sequentially) | | |
| | CHT | 4 SG Nodes | Detailed PV | | | CHT | 4 SG Nodes | Detailed PV |
| | | | | Double ended rupture of a steam generator tube, $in^2$ with concurrent Loss of AC power | | | | |
| | | | | CPCS Low reactor coolant pump speed reactor trip condition, Fraction of initial | | | | |
| | | | | Reactor trip signal generated | | | | |
| | | | | Safety injection actuation, PSIA | | | | |
| | | | | Mass of primary coolant transferred to the ruptured SG, lbm | | | | |
| | | | | Mass of steam released from steam line 1 to atmosphere, lbm | | | | |
| | | | | Mass of steam released from steam line 2 to atmosphere, lbm | | | | |
| | | | | Total mass of steam released from steam lines 1 and 2 to atmosphere, lbm | | | | |

## Table 3.3.2.C

## Summary of Iodine Transport & Dose Results (PIS Case)
## Steam Generator Tube Rupture

| Hand Calculated (Spreadsheet) Output | Parameter Description | Direct CENTS Output |
|---|---|---|
| | Total RCS $I_{131}$ at time = 0.0, Curies | |
| | Total $I_{131}$ in SG liquid at time = 0.0, Curies | |
| | Total $I_{131}$ in Secondary Steam at time = 0.0, Curies | |
| | Total $I_{131}$ in Secondary Side at time = 0.0, Curies | |
| | **Total Global $I_{131}$ at time = 0.0, Curies** | |
| | Total $I_{131}$ exiting the RCS via the tube rupture during the event | |
| | Total $I_{131}$ exiting the RCS via RCP seal leakage during the event | |
| | **Total $I_{131}$ exiting the RCS during the event** | |
| | Total $I_{131}$ exiting to the environment during the event | |
| | Total RCS $I_{131}$ at time = 3600.0, Curies | |
| | Total $I_{131}$ in SG liquid at time = 3600.0, Curies | |
| | Total $I_{131}$ in Secondary Steam at time = 3600.0, Curies | |
| | Total $I_{131}$ in Secondary Side at time = 3600.0 Curies | |
| | **Total Global $I_{131}$ at time = 3600.0, Curies** | |
| | **1 hour contribution toward** 2 hour site boundary thyroid dose | |
| | **1 hour contribution toward** 8 hour LPZ thyroid dose | |
| | **1 hour contribution toward** 2 hour whole body dose*** | |
| | **1 hour contribution toward** 8 hour whole body dose*** | |

** Spreadsheet Hand calculations are based on one second data intervals. Some round-off and truncating of values may occur and cause some deviation from CENTS output.

* Calculated using CENTS output value for RCS_DOSE_TOTAL_CURIE = 82.59

*** Whole body dose is hand calculated using the migration of $I_{131}$ calculated by CENTS as a basis. The calculation below provides the details of this hand calculation.

## Table 3.3.2.D

### Summary of Iodine Transport & Dose Results (GIS Case)
### Steam Generator Tube Rupture

| Hand Calculated (Spreadsheet) Output ** | Parameter Description | Direct CENTS Output |
|---|---|---|
| | Total RCS $I_{131}$ at time = 0.0, Curies | |
| | Total $I_{131}$ in SG liquid at time = 0.0, Curies | |
| | Total $I_{131}$ in Secondary Steam at time = 0.0, Curies | |
| | Total $I_{131}$ in Secondary Side at time = 0.0, Curies | |
| | **Total Global $I_{131}$ at time = 0.0, Curies** | |
| | Total $I_{131}$ exiting the RCS via the tube rupture during the event | |
| | Total $I_{131}$ exiting the RCS via RCP seal leakage during the event | |
| | **Total $I_{131}$ exiting the RCS during the event** | |
| | Total $I_{131}$ exiting to the environment during the event | |
| | Total RCS $I_{131}$ at time = 3600.0, Curies | |
| | Total $I_{131}$ released from the Core by 3600 seconds | |
| | Total $I_{131}$ in SG liquid at time = 3600.0, Curies | |
| | Total $I_{131}$ in Secondary Steam at time = 3600.0, Curies | |
| | Total $I_{131}$ in Secondary Side at time = 3600.0, Curies | |
| | **Total Global $I_{131}$ at time = 3600.0, Curies** | |
| | **1 hour contribution toward** | |
| | 2 hour site boundary thyroid dose | |
| | 8 hour LPZ thyroid dose | |
| | 2 hour whole body dose *** | |
| | 8 hour whole body dose *** | |

*** Whole body dose is hand calculated using the migration of $I_{131}$ calculated by CENTS as a basis. The calculation below provides the details of this hand calculation.

For the PIS case, noble gas concentrations in the RCS at the initiation of the event is assumed at 100/E-bar μCuries/gm. Since all gases exit the steam generators as they leak from the RCS, the total amount of gases entering the atmosphere during the event is assumed equal to the amount transferred from RCS to the steam generator.

CENTS tracks the $I_{131}$ that passes through the tube rupture (leak); therefore, it can be used to calculate the noble gases also.

[

]

Total integrated $I_{131}$ from RCS to SG can be determined from the CENTS output by using a spreadsheet calculation to integrate the leak flow rate x the $I_{131}$ concentration at the break. From this, the integrated noble gases leaking to the RSG are determined and subsequent doses are also determined. A summary is provided below.

$I_{131}$, Integrated flow from RCS to RSG = RCS_IOD_REL_TOT – 23.2 = 2848. –23.2 = 2824.8e6 μCuries

$$D_\gamma = 0.25(X/Q)(\#\ \text{Curies})(\text{E-bar}) \qquad \text{Ref.(9)}$$

Where X/Q (2 hr) = 6.5E-04 sec/m$^3$
X/Q (8 hr) = 3.1E-05 sec/m$^3$

| Time (seconds) | $I_{131}$, Integrated Flow from RCS to RSG (μCuries) | Initial noble gas conc./ Initial $I_{131}$ conc. (fraction/E-bar) | Noble gas, Integrated flow from RCS to RSG (μCuries) | Whole Body Dose (REM) due Noble gases |
|---|---|---|---|---|
| [ | | | | ] |

The GIS case is similarly calculated.

## 3.4 Comparison of CENTS MFW Model to RELAP Model

Two (2) events are analyzed for Plant E which assess the response of all the CENTS upgrades, much as was accomplished for Plant D in Section 2.3. In addition, a major objective for the benchmark cases in this section is to review the simulation response of the detailed CENTS Main Feedwater (MFW) Model. A comparison is made between the CENTS MFW model and a RELAP5.3 feedwater model developed specifically for Plant E. The two events analyzed in this section are a MSLB (from full power and without loss of AC power) and a FWLB (with a Loss of AC power at the time of the reactor trip).

### 3.4.1 Main Steam Line Break (MSLB)

Discussion of Event

A MSLB was analyzed in accordance with Section 15.1.5 of the Standard Review Plan, Reference 3. The analysis is performed to demonstrate that sufficient sources of negative reactivity are available to offset the insertion of positive reactivity added during the transient by the rapid cooldown of the moderator.

For all cases where the detailed CENTS MFW model is deactivated, the RELAP5.3 feedwater model is used to simulate the Plant E feedwater system response. This must be accomplished in an iterative process. A CENTS MSLB preliminary case is run with an assumed feedwater flow and enthalpy response. The steam generator pressure responses from this case output are then provided as input to the steam generator time dependent control volumes in the RELAP5.3 model. The event is then run in RELAP5.3 to determine its system response. The output includes the feedwater flow rates and enthalpies to both steam generators. This data is then used to adjust the feedwater flow and enthalpy input to CENTS. This iteration in cases continues until the resulting feedwater flows and steam generator pressures reach convergence from CENTS to RELAP5.3.

A single MSLB scenario was simulated using the upgraded CENTS code version with upgraded models deactivated, with all but MFW activated, and finally with MFW activated, as discussed above. The case assumes that a double-ended guillotine break occurs in the main steam line inside the containment building from hot full power initial conditions. This case assumes that AC power is maintained, and that the limiting single failure for the event is failure of a Main Feedwater Pump to trip upon receiving a Steam Generator Isolation Signal (SGIS). Thus feedwater flow to the affected steam generator continues until the feedwater isolation valves shut. Flashing of the hot feedwater in the unisolable section of feedwater piping is also modeled and analyzed. Table 3.4.1.A contains a listing of the important assumptions for this case. These assumptions were used in setting up the case data for CENTS.

The cooldown of the RCS continues until the affected steam generator empties. The MSLB case is run to the time at which the core is sub-critical and negative reactivity is being added.

Analysis Methods

The same analysis methods discussed in Section 2.2.1 also apply to the scenario analyzed in this section. For the explicit feedwater models employed in this scenario, the feedwater control system is assumed to "freeze" once the event begins. This means that Main Feedwater Pump speed remains at its initial setting required for 102% power operation. Likewise, the Feedwater Regulating Valves also remain at their initial opening value. No credit is taken for a feedwater system coastdown with a reactor trip. Only when a SGIS occurs does the feedwater system respond by tripping pumps and shutting the feedwater isolation valves.

Results

Table 3.4.1.B provides a comparison of the sequence of events for the MSLB with a failure of a Main Feedwater Pump to trip. Figures 3.4.1.A through 3.4.1.S provide comparisons of important parameters as calculated by the upgraded CENTS version (models deactivated) and the upgraded version (models activated) (other than the detailed MFW model) and lastly with CENTS MFW model active. These plots show excellent agreement.

The transient trend is in general the same in all three cases. In the first comparison between CENTS (with models deactivated) and with all but MFW activated, the RELAP model supplies the feedwater input data for the cases. The difference in these two cases can be attributed to the four node steam generator model with detailed tube heat transfer and the detailed pressure vessel downcomer. The four node steam generator model enhances the steam generator response, lowering RCS pressure more quickly. Also, the temperatures used for the cold edge algorithm are from the lower ring of nodes in the detailed pressure vessel model. Since there is some mixing in the pressure vessel, the temperatures are higher for the detailed pressure vessel model, which results in less moderator reactivity feedback and lower power for most of the event. However, the value of the peak total reactivity is not much affected except that the timing of the peak is later in the event by approximately 40 seconds.

The comparison of the actual feedwater system response between RELAP5.3 and CENTS was very similar as seen in Figure 3.4.1.S. The only major difference between the codes' response was due to a modeling assumption that concerned the behavior of the feedwater heaters during the event. Neither model has detailed cascading heater drain system modeled. Therefore, the cooldown of the feedwater heaters once a turbine trip has occurred is necessarily simplistic. In the RELAP model, the heater tube temperatures were held constant throughout the event. Thus, no heater cooldown was permitted. This supplies the maximum amount of heat to feedwater, particularly since the feedwater flow rate increases in the early the event. This means that the heaters are actually supplying greater than full power heat load when steam drain flow rate is dropping off. The CENTS model does not have the option of keeping tube temperature constant. As steam flow to the turbine varies, the heat load generated by the heaters varies proportionately with a lag function tuned to actual plant heater response during transients. For this scenario, to match the RELAP model as closely as possible, the lag time, $\tau$, was set to a very large number so that the heat rate provided by the heaters was essentially constant throughout the event.

The result of this modeling difference causes the feedwater temperature to decrease more in the CENTS model than in the RELAP5.3 model. Therefore, once the feedwater isolation valves are shut, flow to the affected steam generator stops immediately and does not recommence until steam generator pressure drops below point at which flashing will occur. CENTS MFW begins flashing at ~250 seconds or when steam generator pressure is about 122 psia. At that time, the feedwater in the unisolable line completely empties into the steam generator over the next 100 seconds. The overall amount of flashing makes the integrated amount of feedwater reaching the affected steam generator greater than that predicted by RELAP, due to the fact the water sitting in the unisolable line was cooler (therefore, more mass). For the RELAP5.3 model, the water reaching the line adjacent to the affected steam generator is hotter. Therefore, once the isolation valves shut, the hot feedwater immediately flashes as steam generator pressure drops. This occurs for about 7 seconds beyond the time of isolation. At about 300 seconds, small amounts of feedwater flash into the steam generator for the rest of the event. The differences in the feedwater models' response are quite small when reviewing the overall effect on maximum core reactivity.

One additional effect of the detailed MFW model occurs when emergency feedwater (EFW) is activated. Pressure in the intact steam generator is higher in the CENTS MFW case than it is in the RELAP5.3 case. This causes less of the cold EFW flow to reach the intact steam generator, which in turn, maintains the steam generator pressure higher for the CENTS MFW case. Thus the liquid inventory in the RELAP case recovers more quickly due to the higher EFW flow rates. However, this does not have any significant effect on the overall RCS reactivity or other RCS parameters.

## Table 3.4.1.A

## Important Assumptions for Steam Line Break

| Parameter | Value |
| --- | --- |
| Break Size (Equivalent to SG nozzle area) | 1.887 $Ft^2$ |
| Core Power | 102% of 3026 MWt |
| Core Inlet Temperature | 556.7 °F |
| Pressurizer Pressure | 2300 PSIA |
| Pressurizer Level | 22.04 Feet |
| Core Burnup | End of Cycle |
| Steam Generator Pressure | 1000 PSIA |
| Steam Generator level | 39.64 |
| Scram Worth | 6.84 % $\Delta\rho$ |
| Moderator Temperature Coefficient | $-3.8 \times 10^{-4}\ \Delta\rho/°F$ |
| Fuel Temperature Coefficient | End of Cycle |
| Loss of Offsite Power | None assumed |

## Table 3.4.1.B

### Sequence of Events
### Steam Line Break – Upgraded Version

| Time(Sec) | | | Event | Value | | |
|---|---|---|---|---|---|---|
| Models deactivated (RELAP MFW model) | Upgrades Activated (Sequentially) | | | Models deactivated (RELAP MFW model) | Upgrades Activated (Sequentially) | |
| | CHT, 4 SG Nodes, Detailed PV (RELAP MFW model) | CHT, 4 SG Nodes, Detailed PV (CENTS MFW model) | | | CHT, 4 SG Nodes, Detailed PV (RELAP MFW model) | CHT, 4 SG Nodes, Detailed PV (CENTS MFW model) |
| | | | Main steam line break $Ft^2$ | | | |
| | | | Reactor Trip on Containment High Pressure, PSIA | | | |
| | | | Main steam isolation signal on SG Low Pressure, PSIA | | | |
| | | | Main steam isolation valves fully closed | | | |
| | | | Safety injection actuation, PSIA | | | |
| | | | Safety injection flow begins | | | |
| | | | Affected steam generator empties (<1000 lbm liquid in SG) | | | |
| | | | Minimum mixed core inlet temperature is reached, °F | | | |
| | | | Maximum Reactivity is reached, delta rho | | | |
| | | | Maximum Return to Power, fraction | | | |
| | | | Total Integrated MFW flow to the affected SG, lbm | | | |

### 3.4.2 Feedwater Line Break

Discussion of Event

A FWLB may produce a total loss of normal feedwater and a blowdown of one steam generator. If normal sources of AC electrical power were lost, there would also be a simultaneous loss of primary coolant flow, turbine load, pressurizer pressure and level control and steam bypass control. The result of these events would be a rapid decrease in the heat transfer capability of both steam generators and eventually the complete loss of the heat transfer capability of one steam generator.

FWLB sizes which cause the most limiting RCS peak pressure are relatively small, compared to a full guillotine break of the feedwater line. For these small breaks, the feedwater system response can be an important consideration. Feedwater flow to the intact steam generator does not decrease to zero, but remains at a sizeable fraction of the initial flow rate, even when the feedwater control system is "frozen" at initial pump speeds and valve positions. This scenario simulates the effects of a detailed MFW model and compares the response of a RELAP5.3 feedwater model to the detailed CENTS MFW system.

Where the RELAP5.3 feedwater model is used to simulate the Plant E feedwater system response, an iterative process is employed. A preliminary CENTS FWLB case is run with an assumed feedwater flow and enthalpy response. The steam generator pressure response from this case output is then provided as input to the RELAP5.3 model. The feedline break is then run in RELAP. The output includes the flow rates from the affected steam generator to the break. This data is then used to adjust CENTS input. This iterative process continues until convergence of resulting flows and steam generator pressures in both codes occurs.

The NSSS is protected during this transient by the pressurizer safety valves and the following reactor trips:

- Low steam generator level
- Low steam generator pressure
- High pressurizer pressure
- High containment pressure

Depending on the initial conditions, any one of these trips may terminate the transient. The NSSS is also protected by MSIVs, feedwater line check valves, steam generator safety valves and the auxiliary feedwater system which serves to protect the integrity of the secondary heat sink following reactor trip.

For this event scenario, limiting peak RCS pressures are attained when the high pressurizer pressure trip and steam generator low level trip occur close to simultaneously. This is verified by parametric cases where break size is varied to adjust the relative times of the trip signals. Only those cases which cause a limiting peak pressure are presented herein.

The NRC criterion for this event, with a limiting single failure, is that the peak RCS pressure must be less than 120% of RCS design pressure.

This FWLB scenario assumes the limiting break size. As the case initial conditions change from the upgraded (models deactivated) to the upgraded (models activated) CENTS versions, the break size is adjusted to ensure a limiting peak RCS pressure condition. See Table 3.4.2.B for the size in each of the cases. These breaks are assumed to occur in the feedline to one of the steam generators, downstream of the feedwater check valve.

Table 3.4.2.A contains a listing of the important assumptions for this case.

Analysis Methods

The models necessary to incorporate the feedwater system pipe break methodology, presented in Section 15.E of CESSAR, Reference 1 (listed in Section 2.2.2), are used in all but the last test cases of this scenario. In those last cases (results in Section 3.4.2.1), the location (elevation) of the feedring is set at its actual physical elevation of ~32 ft above the tubesheet. This change means that the liquid blowdown to the feedwater line break will become steam when the steam generator downcomer level drops below the feedring. In addition, the non-physical modeling of steam generator heat transfer ramp down discussed in Section 2.2.2 is also deactivated in this last test case. The break size is again adjusted to create a limiting RCS peak pressure. The objectives of these cases are:

(a) to determine the overall effect on peak pressure when compared to the CESSAR methodology,
(b) to determine the effect on the long term pressurizer level response as the plant heats up to the quasi-steady state condition, where the cycling steam generator safety valves are relieving system heat, and
(c) to compare the feedwater system response using the RELAP5.3 model and the CENTS MFW model.

There are a total of five cases that are part of this analysis. All use the upgraded CENTS version.

a.) All upgrade models deactivated, with RELAP5.3 MFW model
b.) All upgrade models activated, with RELAP5.3 MFW model
c.) All upgrade models activated, with CENTS MFW model

See Section 3.4.2.1 for discussion of Case results for

d.) All upgrade models activated, with RELAP5.3 MFW model, feedring at actual height
e.) All upgrade models activated, with CENTS MFW model, feedring at actual height

<u>Results</u>

Table 3.4.2.B provides a comparison of the sequence of events for the FWLB. Figures 3.4.2.A through 3.4.2.Q provide comparisons of important parameters for three cases as calculated by the upgraded CENTS version (models deactivated), (models activated), both using the RELAP5.3 MFW model and (models activated, using the CENTS MFW model), all with the feedring at the bottom of the steam generator downcomer.

The first comparison of results to be reviewed is between the upgraded CENTS version (models deactivated) and (models activated), using the RELAP5.3 feedwater model in both cases. Thus, this comparison shows the effects of the core heat transfer model upgrade, the four node steam generator model upgrade and the detailed PV model. With these upgrades activated:

- Peak RCS pressure decreased by ~16 psia.
- Peak steam generator secondary pressure increased by ~8 psia.
- Maximum pressurizer liquid volume decreased by ~18 ft$^3$ (0- 120 sec)
- Long term pressurizer liquid volume decreased by about ~3 ft$^3$

The cause for these changes is very similar to that discussed for Plant D in Section 2.3.2. In summary, the core heat transfer upgrade does not effect this event. The 4 node steam generator model enhances the heat transfer of the steam generator tubes which limits RCS heat heatup leading up to reactor trip, thus lowering the RCS peak pressure transient. Long term, the detailed pressure vessel model promotes more natural circulation flow in the intact loop, promoting better heat transfer to the intact steam generator and helps to minimize pressurizer fill, though not significantly.

The next comparison to be made involves a benchmark of the CENTS MFW model to a RELAP5.3 model for Plant E. The results are expected to be similar, but with some differences based on different correlations or methods employed by the codes. The major differences are the choke flow correlation and flow regime determination in RELAP5.3 for the feedwater system node just upstream of the break.

The choke flow correlation employed by CENTS in the feedwater line break is either Henry-Fauske (HF) or Homogeneous Equilibrium (HEM). RELAP has its own theoretically calculated critical flow determination (Ransom & Trapp) (Reference 4, Section 3.4.1). During the single phase (subcooled) portion of the blowdown, the CENTS employed HF & HEM correlations both predict greater choked flow than the RELAP calculated flow (Figure 3.4.2.R). Thus, time till the steam generator empties is shorter with the CENTS MFW model by approximately 5 seconds (Figure 3.4.2.K). Since break flow is higher with the CENTS MFW, it means that available feedwater flow to the intact steam generator is lower (Figure 3.4.2.M). The effects of the differences in the feedwater models upon the rest of the NSSS is minimal in most respects. The timing for the reactor trip and peak RCS and steam generator pressures is shorter by 4 to 6 seconds, with the peak RCS pressure being about 6 psi higher with the CENTS MFW model. Long term, the peak pressurizer liquid volume is lower by about 10 $ft^3$ for the CENTS MFW model case.

## Table 3.4.2.A

## Important Assumptions
## Feed Line Break – Upgraded CENTS Version

| Parameter | Value | | |
|---|---|---|---|
| | Models deactivated (RELAP MFW model used) | Models activated (2 cases, one with RELAP, one with CENTS MFW model) | Models activated, Feedring at actual height (2 cases, one with RELAP, one with CENTS MFW model) |
| Core Power, MWt | 102% of 3026 | | |
| Core Inlet Temperature, °F | 556.7 | | |
| Moderator Temperature Coefficient, %Δρ/ °F | $-0.2 \times 10^{-4}$ | | |
| Pressurizer Level, ft | 22.0 | | |
| Pressurizer Pressure, psia | 2300 | | |
| Steam Generator Pressure, psia | ≈1000 | | |
| Feedwater Control System | Manual Mode | | |
| Loss of Offsite Power | Power is lost at the time a reactor trip signal is generated | | |
| Break Size, $ft^2$ | .215 | .218 | 0.120 |
| Feedring Height above tube sheet, ft | 0.0 | 0.0 | 31.6 |
| Feedwater model Employed | RELAP | (1) RELAP<br>(2) CENTS | (1) RELAP<br>(2) CENTS |
| Linear SG heat transfer ramp down methodology employed | Yes | Yes | No |
| High Pressurizer Pressure Trip Setpoint, psia | 2422 | 2422 | 2422 |
| Steam Generator Low Level Trip setpoint (affected SG) | 40,000 lbm | 40,000 lbm | .06 of Narrow range level indication |

## Table 3.4.2.B

### Sequence of Events
### Feed Line Break - Upgraded CENTS Version

| Time(Sec) | | | Event | Value | | |
|---|---|---|---|---|---|---|
| Models Deactivated Feedring @ 0.0 ft (RELAP) | Models Activated Feedring @ 0.0 ft (RELAP/CENTS) | Models Activated Feedring @ 31.6 ft (RELAP/CENTS) | | Models Deactivated Feedring @ 0.0 ft (RELAP) | Models Activated Feedring @ 0.0 ft (RELAP/CENTS) | Models Activated Feedring @ 31.6 ft (RELAP/CENTS) |
| | | | Feed line break, $Ft^2$ | | | |
| | | | Reactor Trip Signal on HPPT or SGLL | | | |
| | | | Loss of AC power | | | |
| | | | Peak RCS Pressure, PSIA | | | |
| | | | Peak Pressurizer Liquid Volume (1st peak, 0 – 120 seconds), $Ft^3$ | | | |
| | | | Peak Steam Generator Pressure, PSIA** | | | |
| | | | Affected steam generator empties (0 liquid in evaporator) | | | |
| | | | Peak Pressurizer Liquid Volume (2nd peak, 120 seconds to end of case), $Ft^3$ | | | |
| | | | Minimum Intact Steam Generator Liquid Mass, Lbm | | | |

** The objective of the scenario presented herein is to determine peak RCS pressure. Though peak SG pressure is reported in the table, this is not the case which determines limiting SG pressure.

### 3.4.2.1.1 Feedwater Line Break - with Feedring Modeled at Actual Height

Results

The last column of Table 3.4.2 provides a summary of the sequence of key events for an FWLB when the feedring is placed at its actual elevation and the CENTS realistic tube heat transfer model is employed (instead of forcing full heat transfer until a set liquid inventory in the steam generator is reached).
Figures 3.4.2.1.A through Q provide a comparison of key parameters. Two comparisons can be made with the figures, first a comparison of the feedring at the bottom and then with the feedring at its actual elevation, both using the CENTS MFW model. A second comparison reviews the differences between the CENTS MFW model and the RELAP5.3 feedwater model, both with the feedring modeled at its actual elevation.

Placing the feedring at its actual elevation changes the FLB event significantly (after RCS peak pressure has occurred). Early in the event, a general NSSS and secondary heatup occurs, just as it does when the feedring is artificially located at the bottom of the steam generator. Moreover, the results show that if the break size is adjusted until the limiting peak RCS pressure is attained, then placement of the feedring and the tube heat transfer modeling employed have very little effect on the magnitude of the peak pressure. With the RELAP5.3 feedwater model being used, peak RCS pressure rose by ~5 psi to ~2641 psia when the feedring was placed at its actual elevation. With the CENTS MFW model, the peak RCS pressure dropped by ~2 psi to ~2640 psia (Figure 3.4.2.1.C). Peak steam generator pressures dropped by ~12 to 14 psi with the feedring at its actual elevation, due to the steam being relieved via the break (Figures 3.4.2.1.G & H).

The significant effect of placing the feedring at its actual elevation is in the more realistic estimation of the long-term peak pressurizer liquid volume (Figure 3.4.2.1.P). With the feedring placed at the bottom of the steam generator, the entire affected steam generator empties before steam escapes via the break (Figure 3.4.2.1.K). Thus, after the reactor trip, the amount RCS coolant contraction (cooldown) is minimized. With a more realistic placement of the feedring location, a steam blowdown commences when the steam generator downcomer water level drops below the feedring. This relieves more energy per lbm via the break with considerably slower loss of steam generator inventory. After the reactor trip there is much greater steam generator mass available to blow down as steam. This causes a cooldown of the RCS until the liquid inventory in the affected steam generator is depleted (Figures 3.4.2.1.E & F). This cooldown / contraction in RCS liquid volume delays the RCS heatup to the quasi-steady state condition cycling the steam generator safety valve. The early swell due to core decay heat has been reduced, thus the final equilibrium pressurizer level is much lower (Figure 3.4.2.1.P). In addition to the inventory in the affected steam generator blowing down as steam, the intact steam generator also contributes steam to the break (Figures 3.4.2.1.J & I). The steam generator isolation signal, due to affected steam generator low pressure (905 psia), is delayed significantly. This allows a contribution from the intact steam generator for a longer period of time than was available when the feedring was artificially placed at the bottom of the steam generator.

These test cases, with the feedring at its actual elevation and realistic steam generator tube heat transfer modeling, provide ample justification for allowing these changes in future feedwater line break events for those plants that do not have steam generator economizers. Peak RCS pressures have not been significantly affected, but the accuracy of the long-term pressurizer level response has increased significantly.

Figure 3.2.1.A

Reactor Core Power

Seized Rotor Event for Plant E

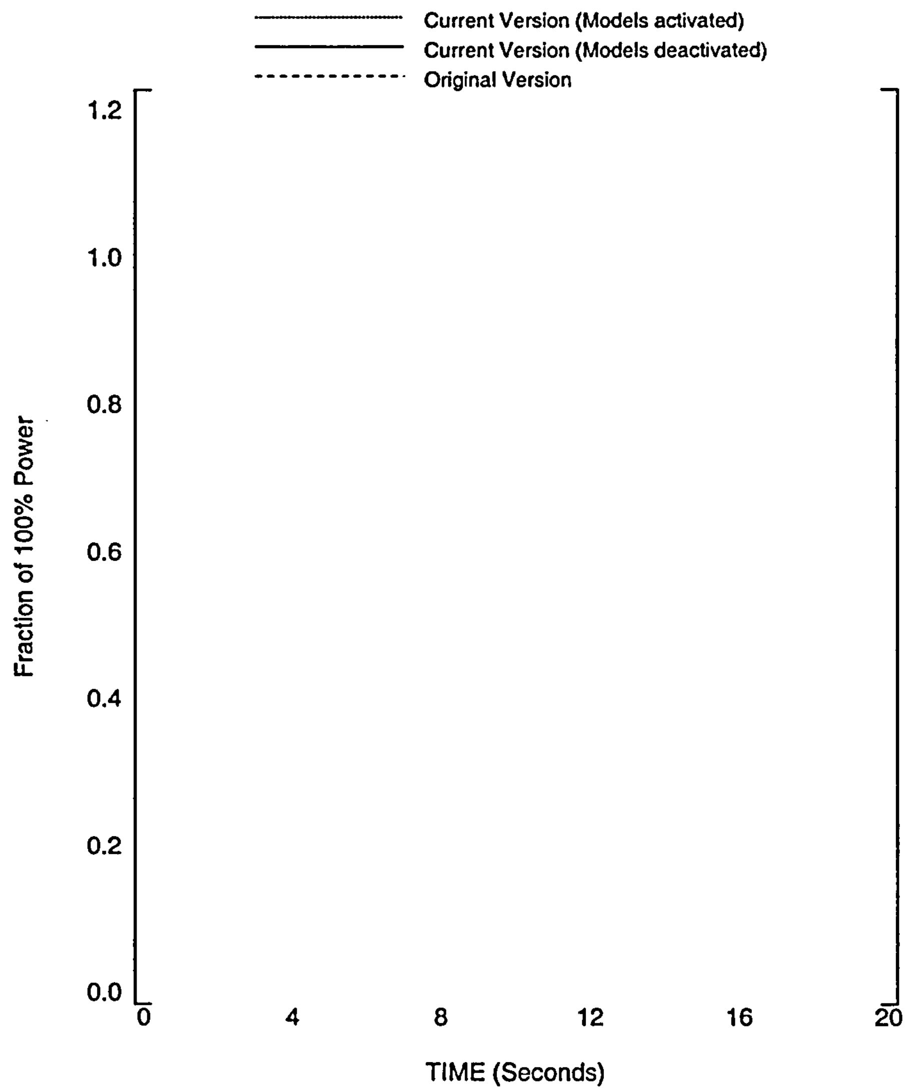


Figure 3.2.1.B

Reactor Core Heat Flux

Seized Rotor Event for Plant E

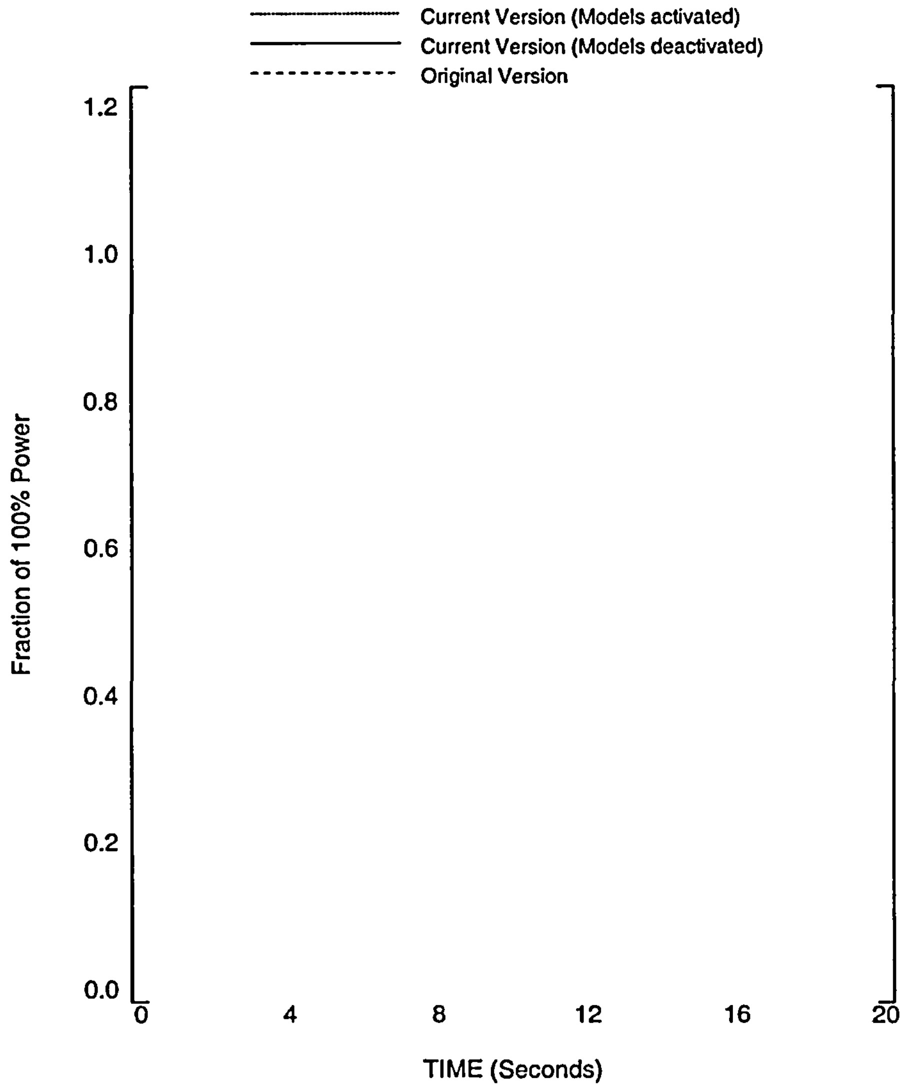


Figure 3.2.1.C

Core Pressure

Seized Rotor Event for Plant E

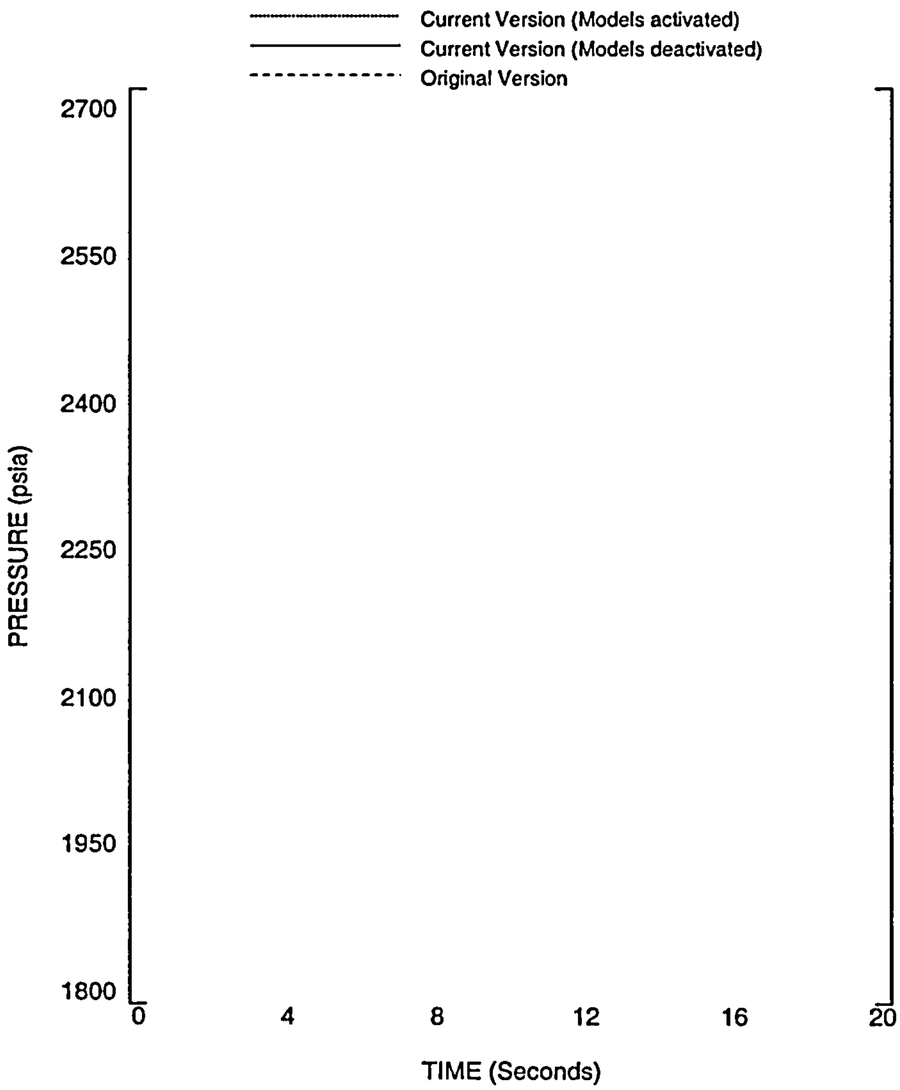


Figure 3.2.1.D

Pressurizer Pressure

Seized Rotor Event for Plant E

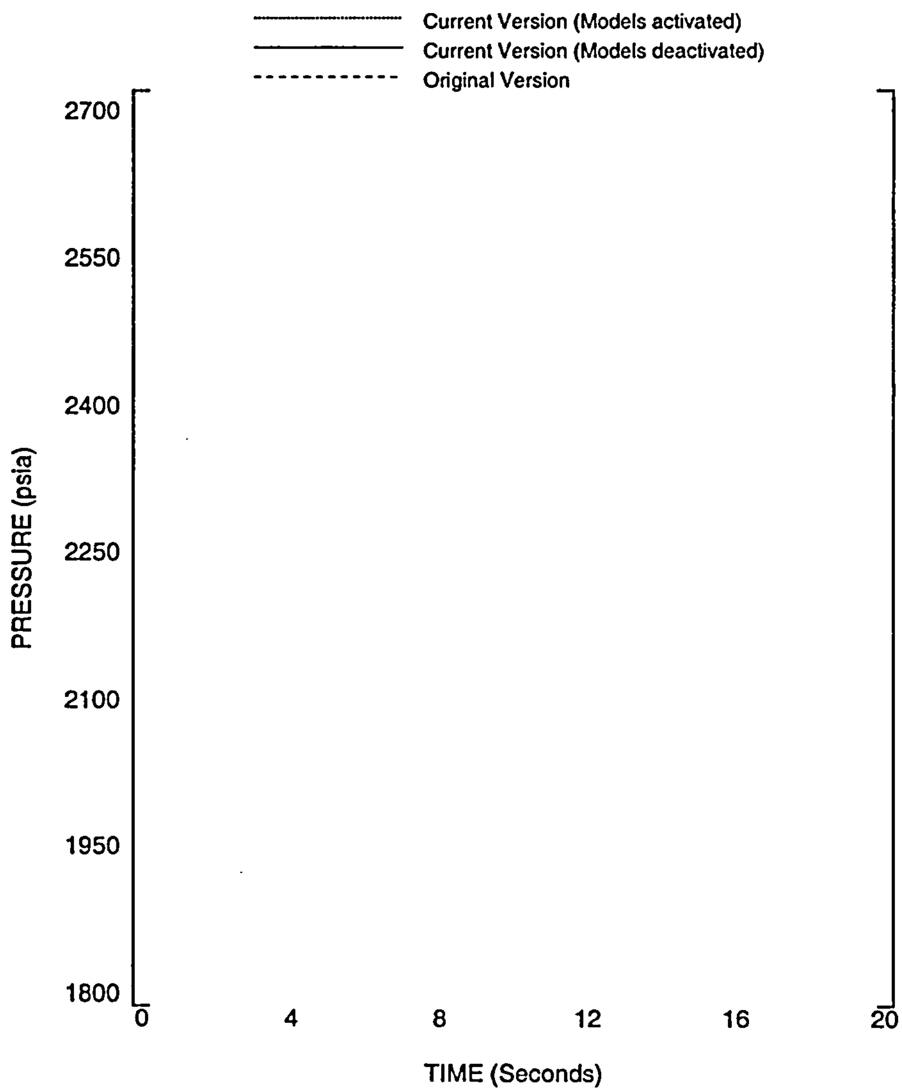


Figure 3.2.1.E

Core Inlet Temperature

Seized Rotor Event for Plant E

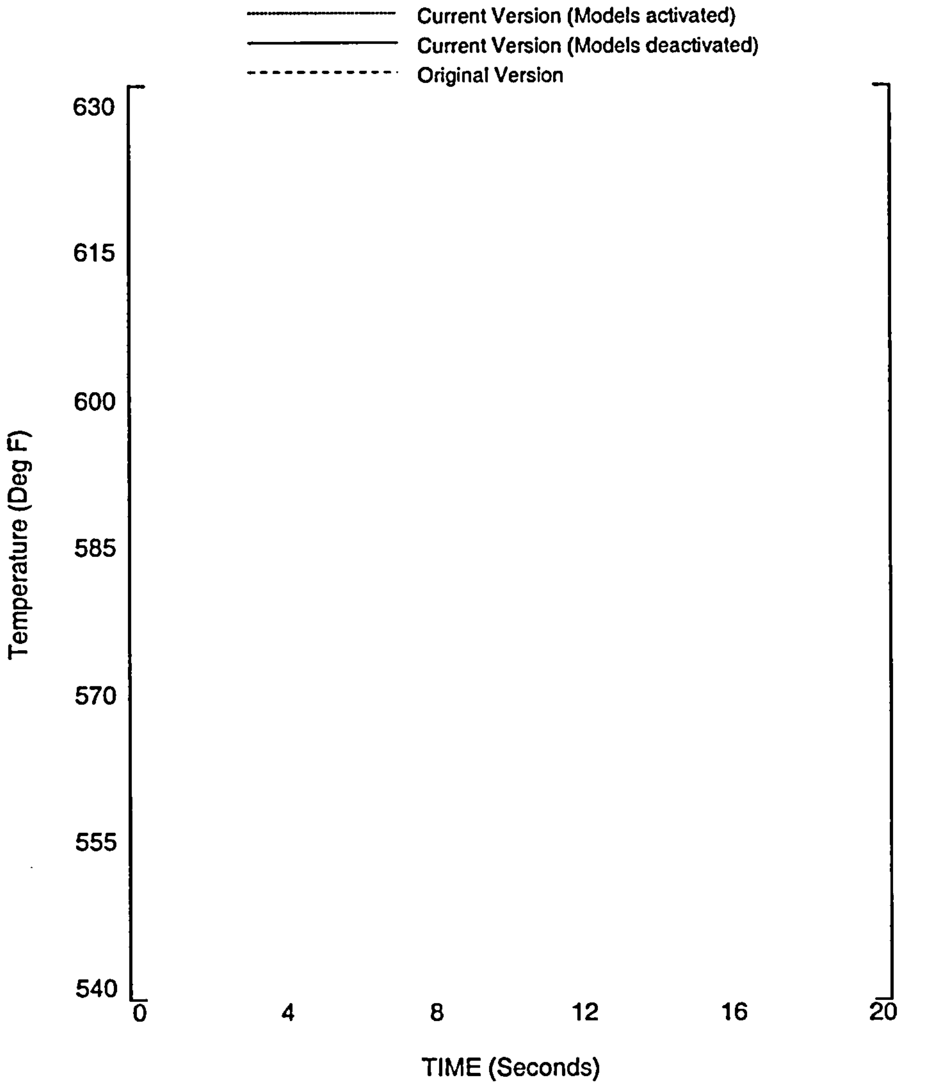


Figure 3.2.1.F

Steam Generator Pressure, Affected Steam Generator

Seized Rotor Event for Plant E

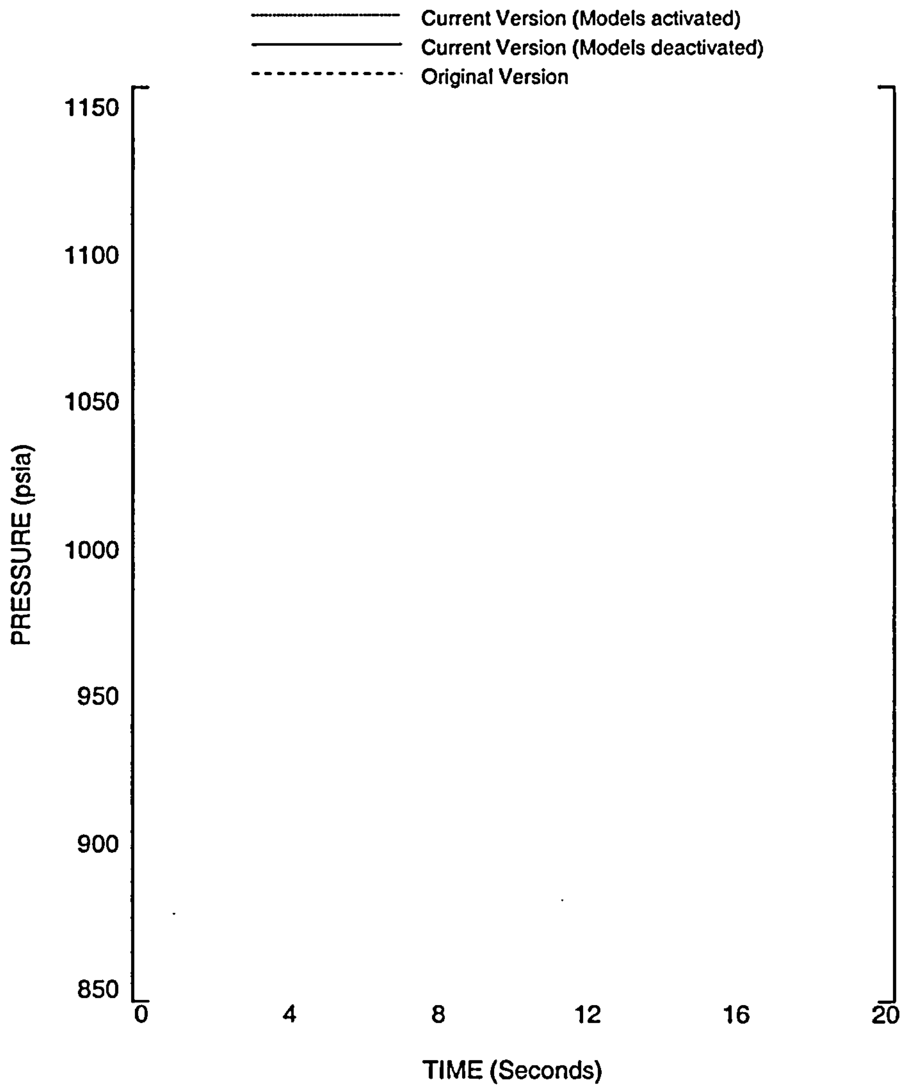


Figure 3.2.1.G

Secondary Steam Flow, Intact Steam Generator

Seized Rotor Event for Plant E

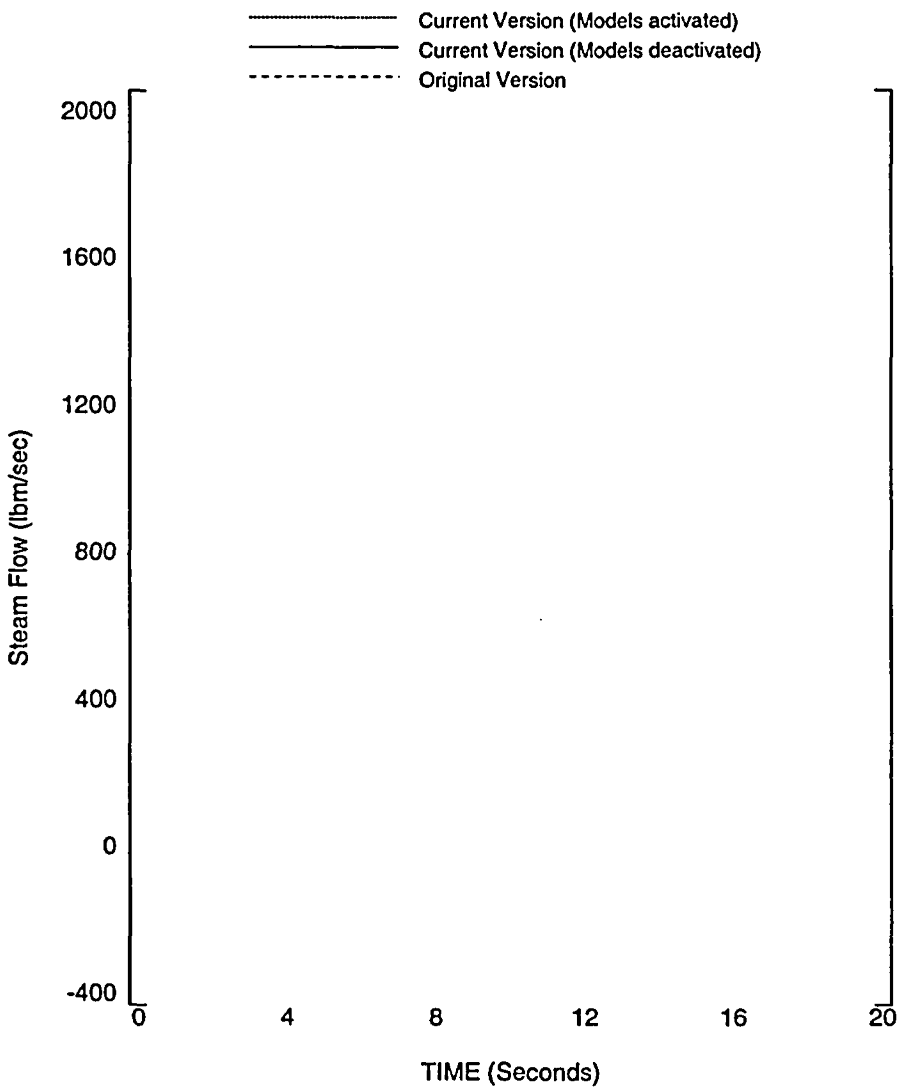


Figure 3.2.1.H

Secondary Steam Flow, Affected Steam Generator

Seized Rotor Event for Plant E

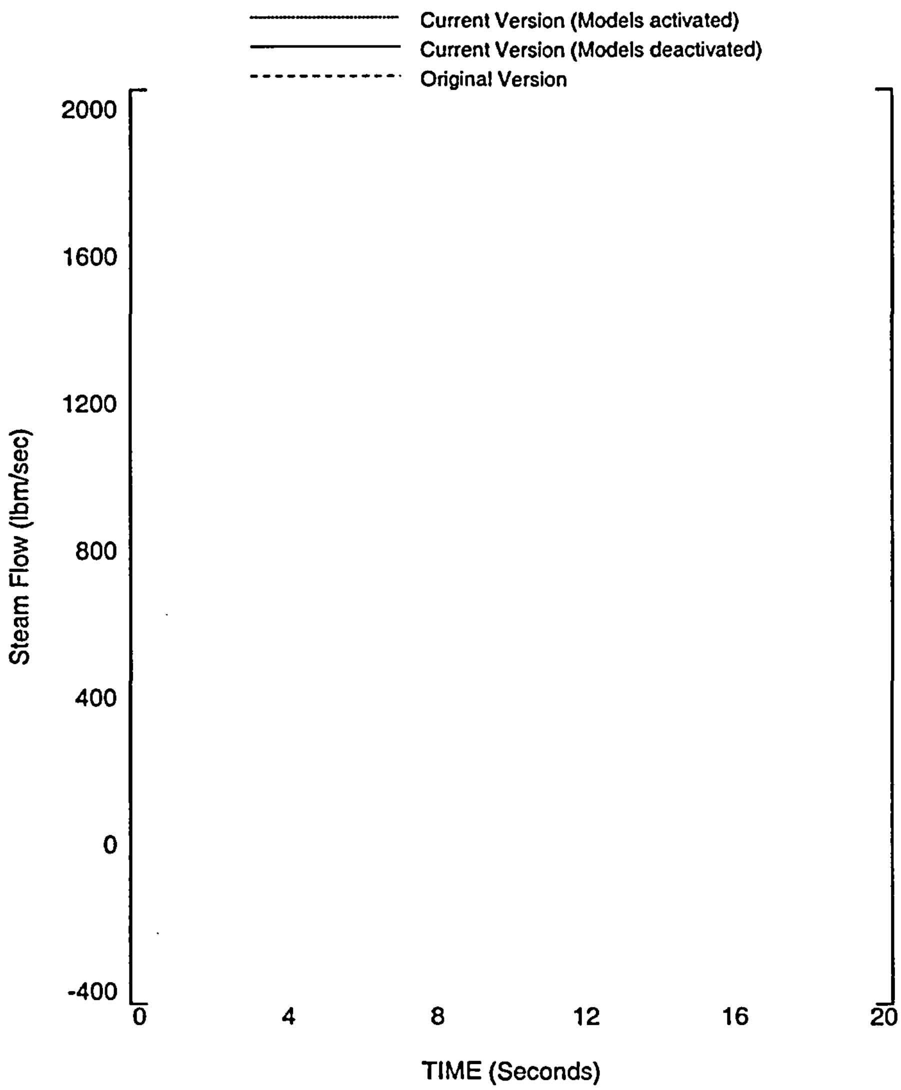


Figure 3.2.1.I

Core Mass Flow

Seized Rotor Event for Plant E

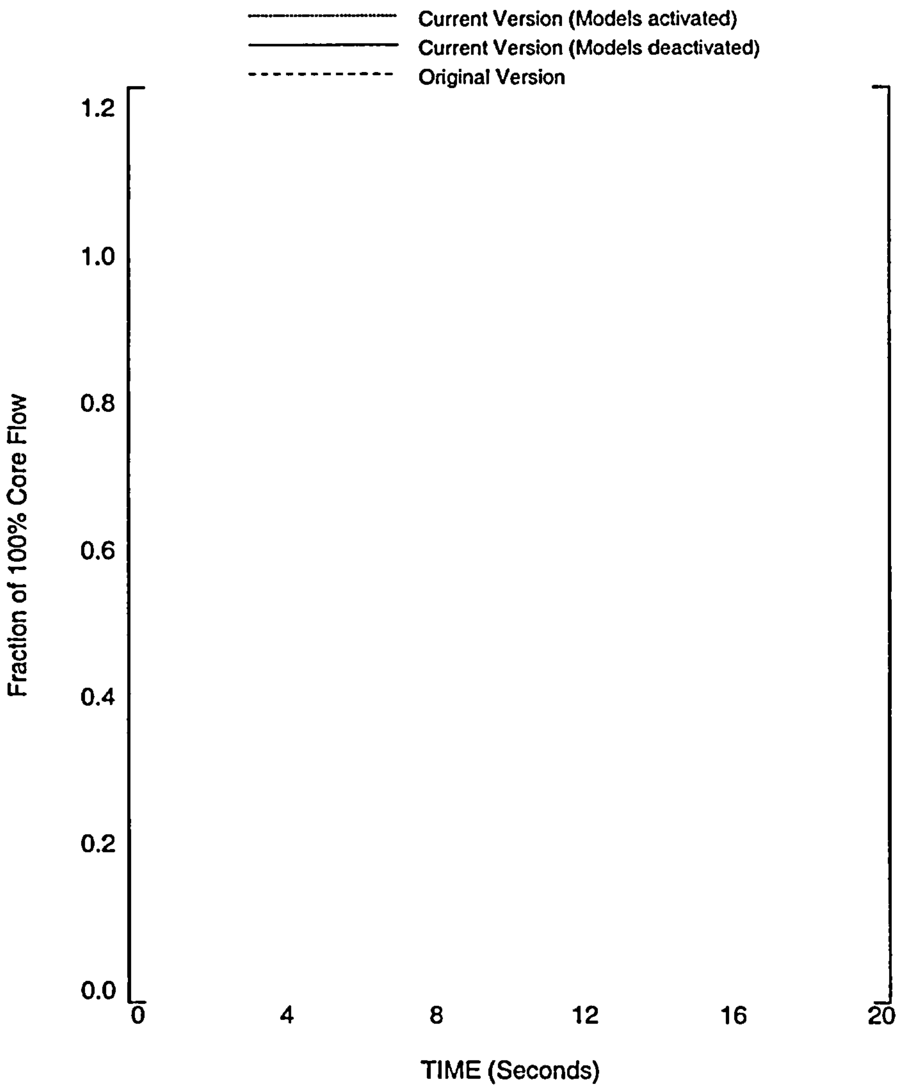


Figure 3.2.2.A

Reactor Core Power

Steam Generator Tube Rupture for Plant E

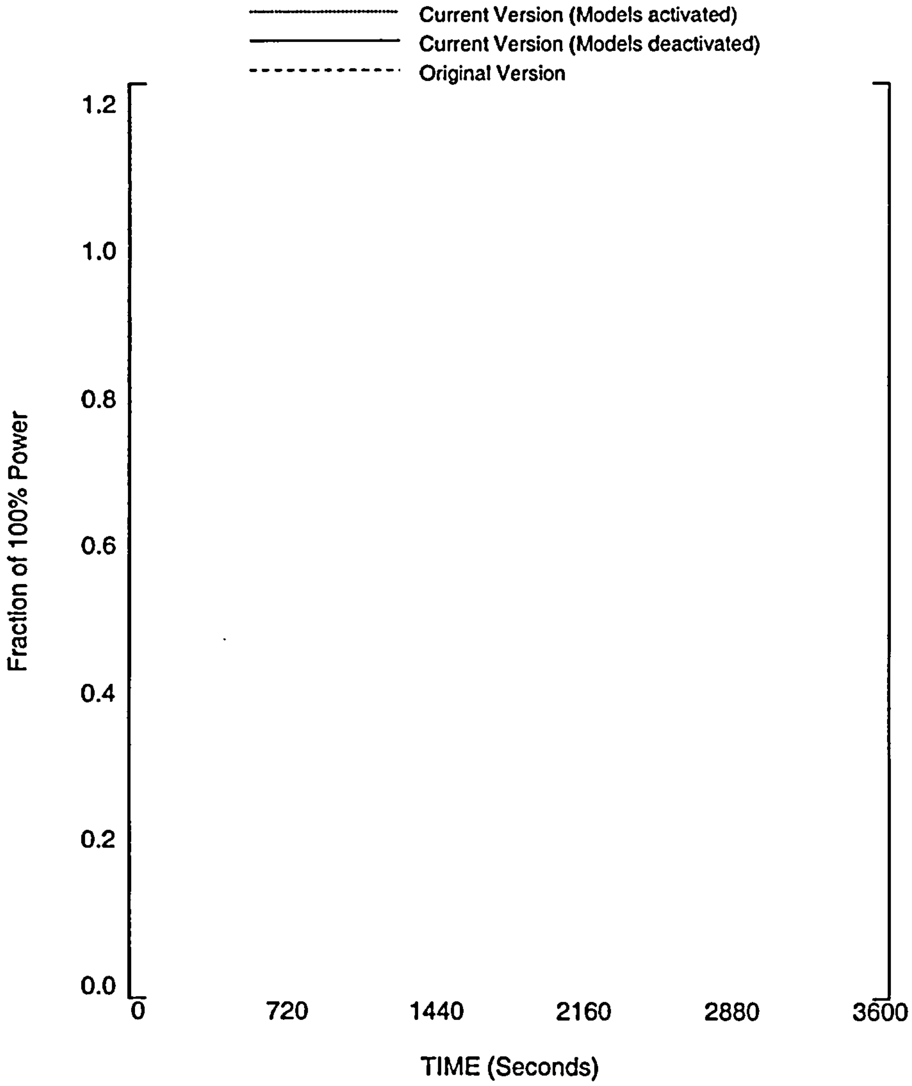


Figure 3.2.2.B

Reactor Core Heat Flux

Steam Generator Tube Rupture for Plant E

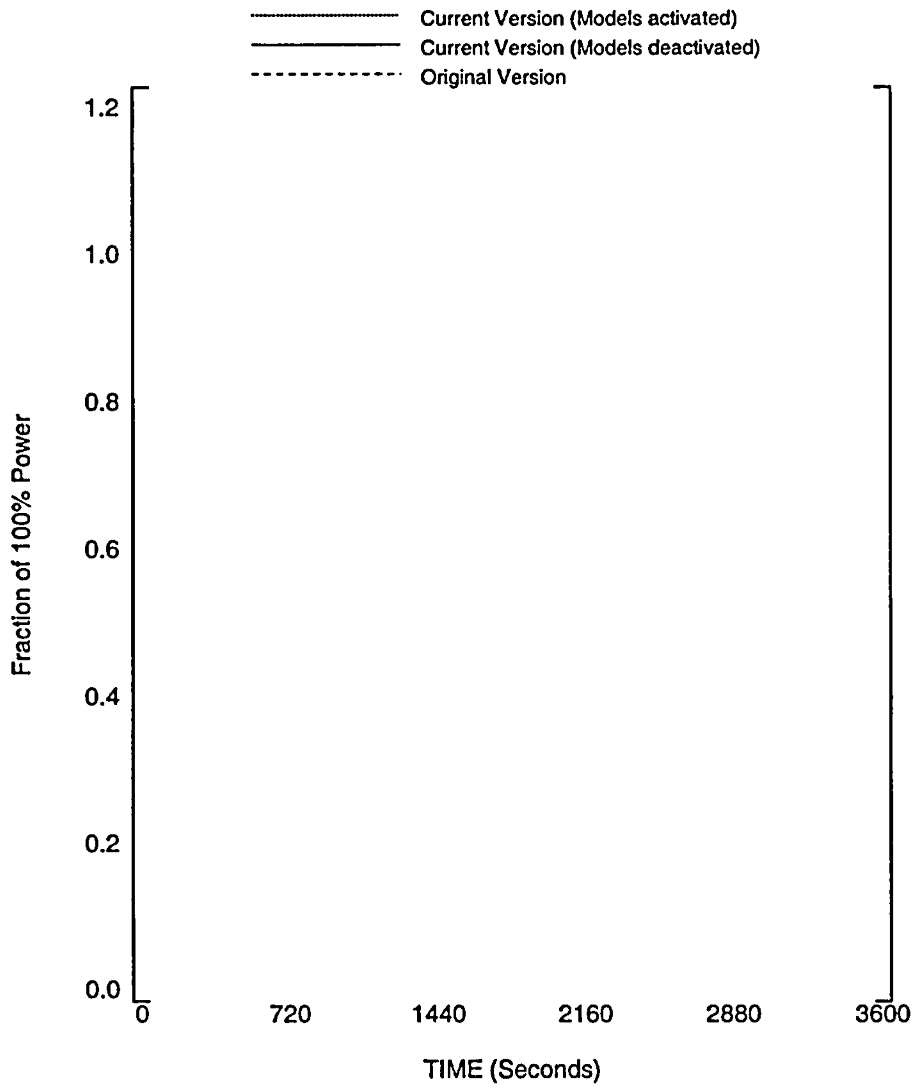


Figure 3.2.2.C

Core Pressure

Steam Generator Tube Rupture for Plant E

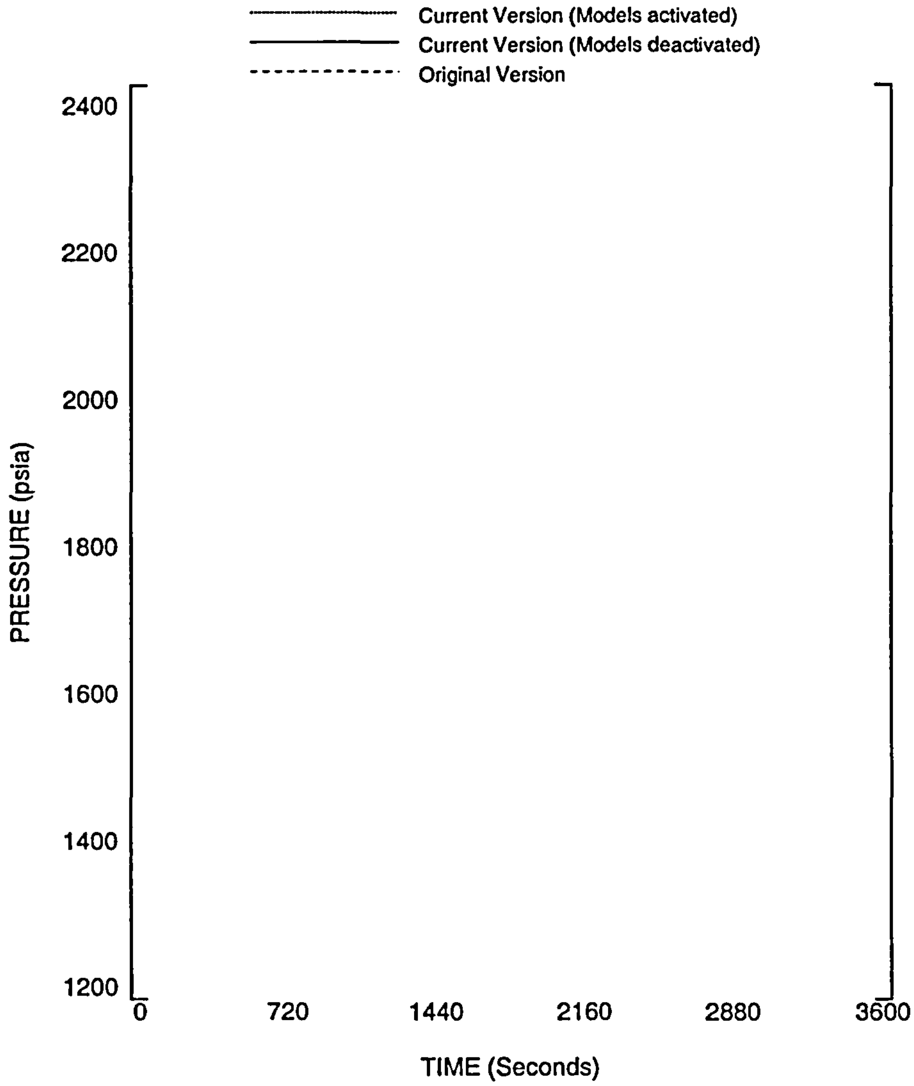


Figure 3.2.2.D

Pressurizer Pressure

Steam Generator Tube Rupture for Plant E

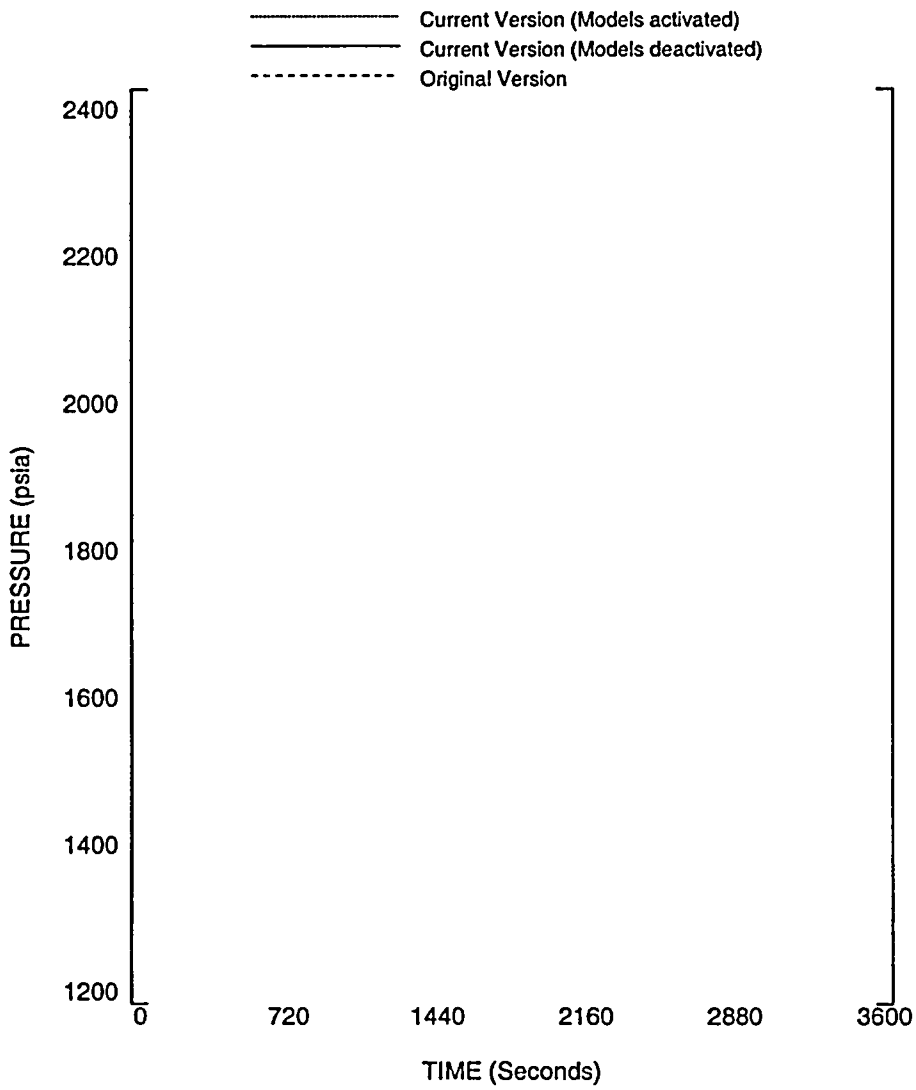


Figure 3.2.2.E

Core Inlet Temperature

Steam Generator Tube Rupture for Plant E

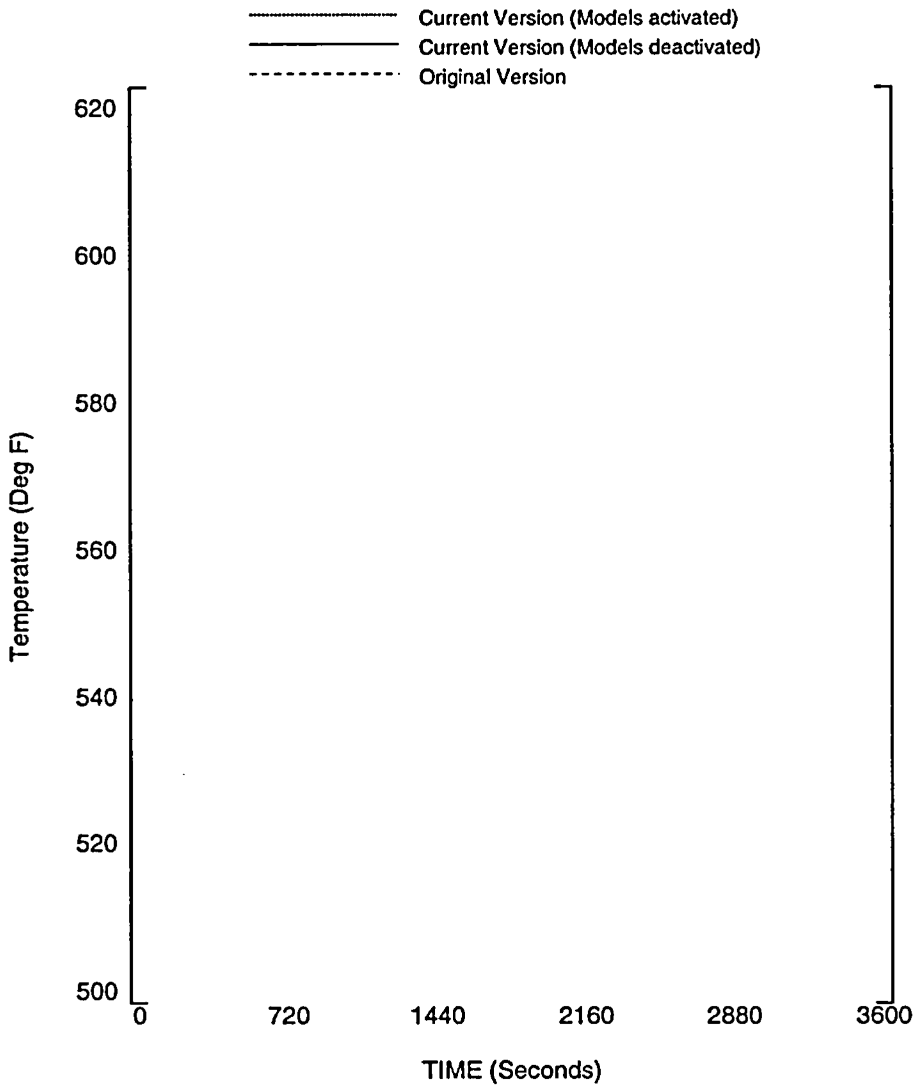


Figure 3.2.2.F

Steam Generator Pressure, Affected Steam Generator

Steam Generator Tube Rupture for Plant E

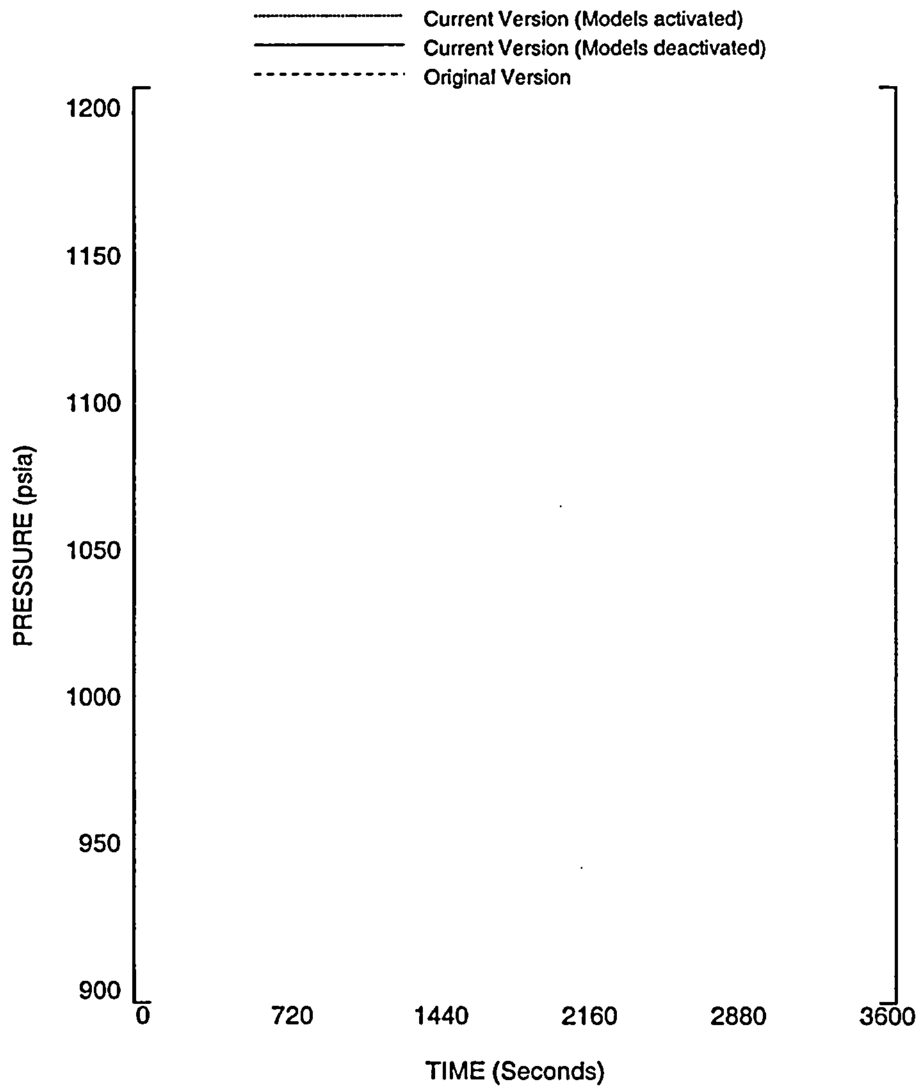


Figure 3.2.2.G

Steam Generator Pressure, Intact Steam Generator

Steam Generator Tube Rupture for Plant E

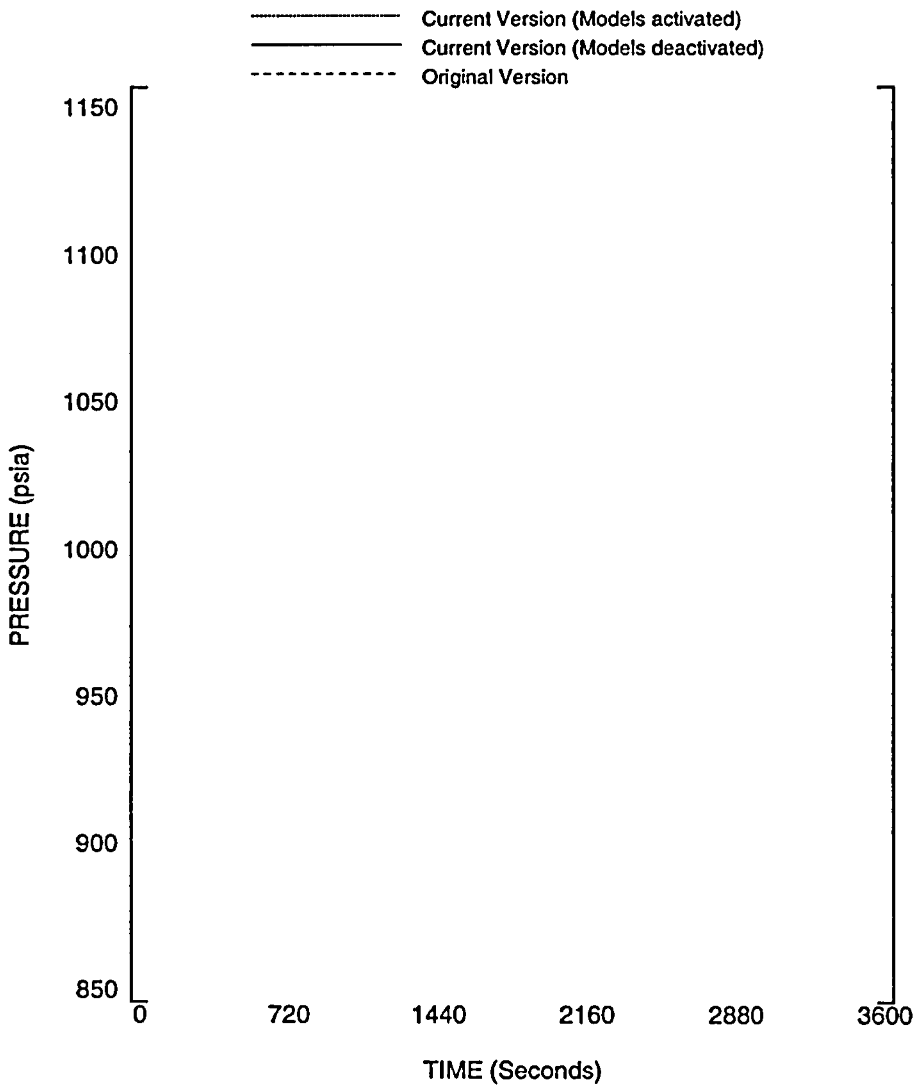


Figure 3.2.2.H

Pressurizer Level

Steam Generator Tube Rupture for Plant E

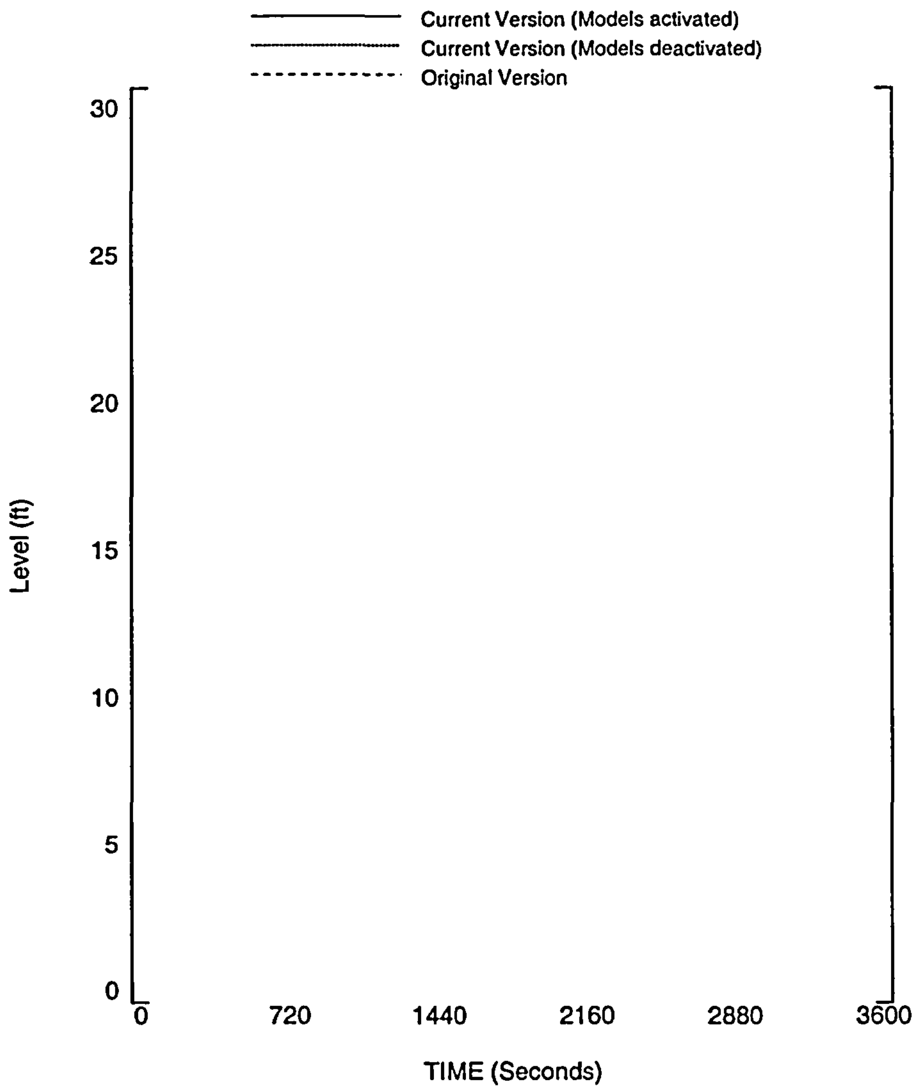


Figure 3.2.2.l

Core Mass Flow

Steam Generator Tube Rupture for Plant E

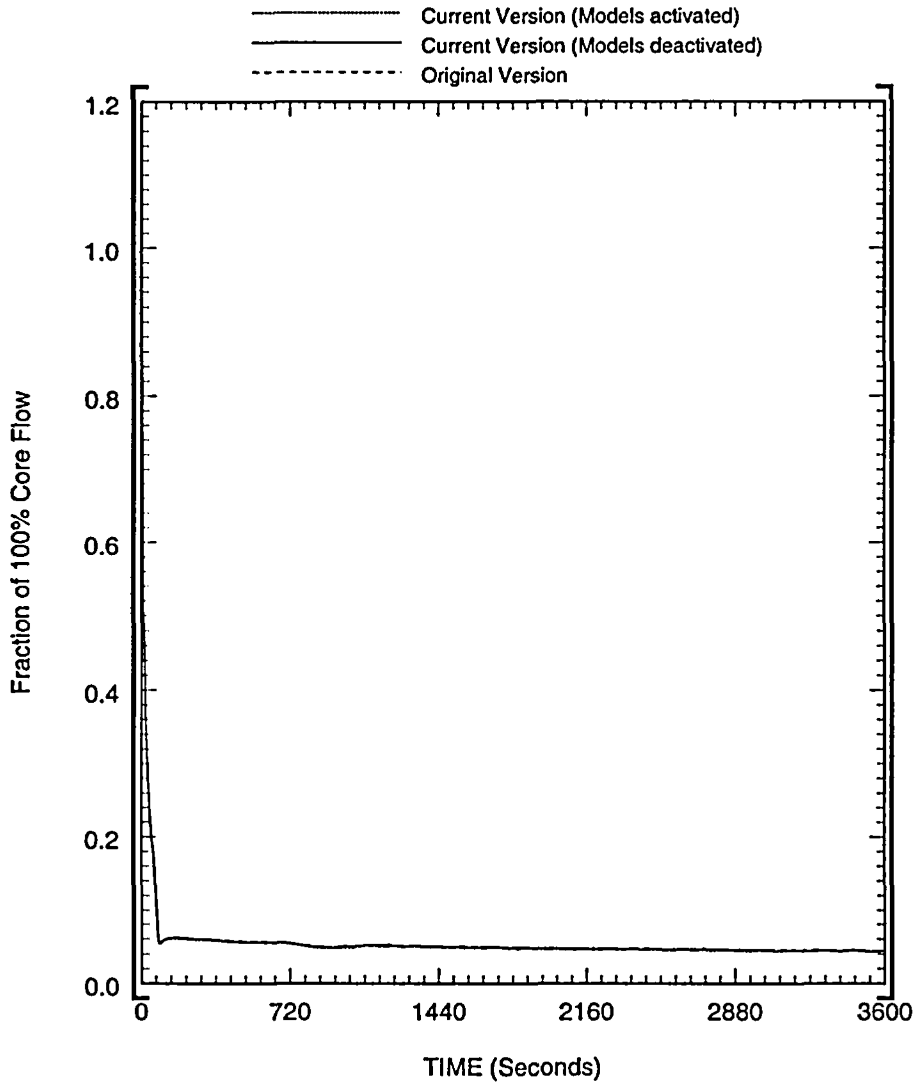


Figure 3.2.2.J

Break Flow, Weighted Average Enthalpy

Steam Generator Tube Rupture for Plant E

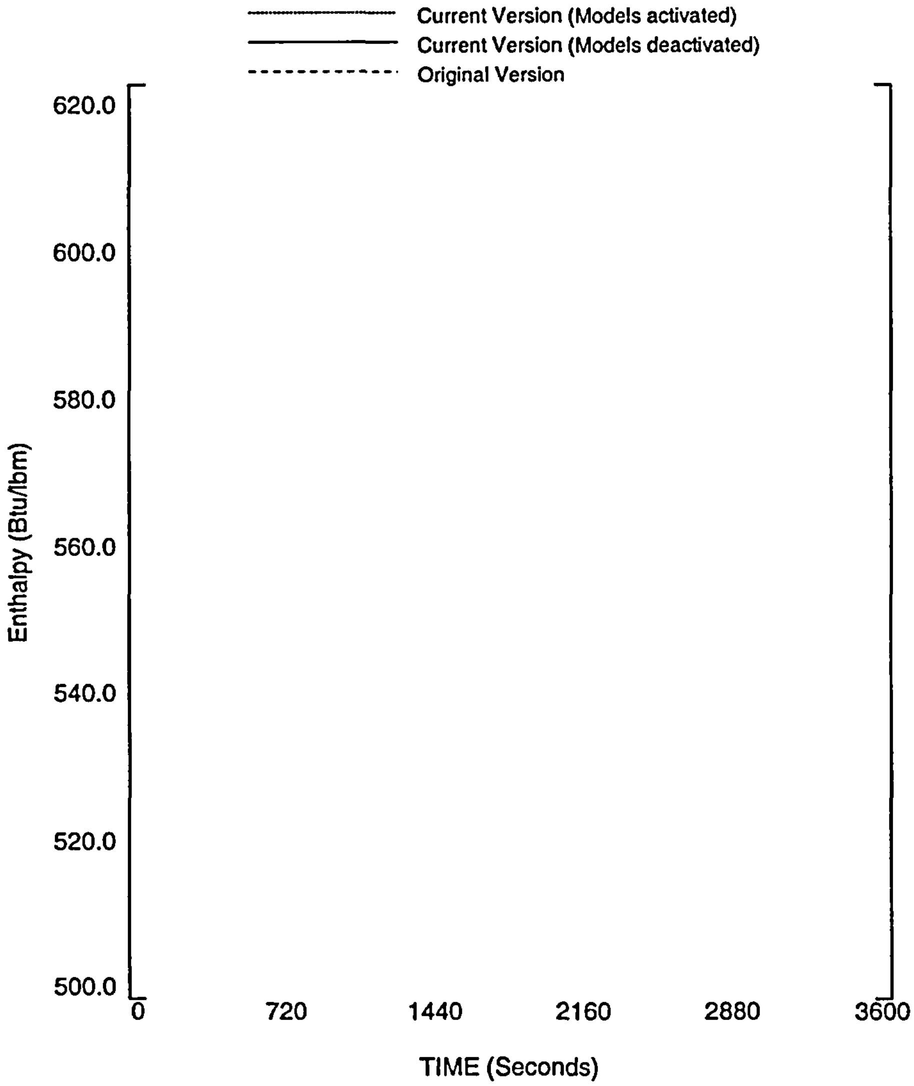


Figure 3.2.2.K

Break Flow, Hot Side

Steam Generator Tube Rupture for Plant E

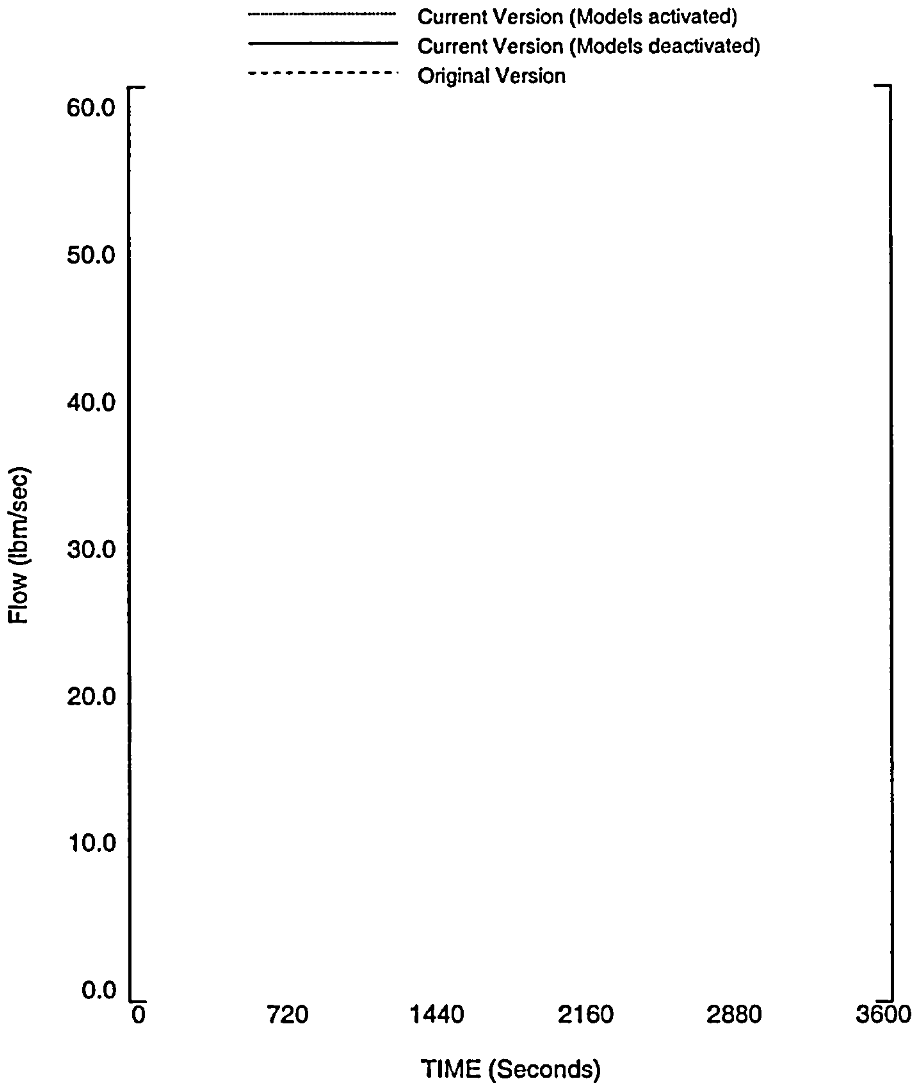


Figure 3.2.2.L

Break Flow, Cold Side

Steam Generator Tube Rupture for Plant E

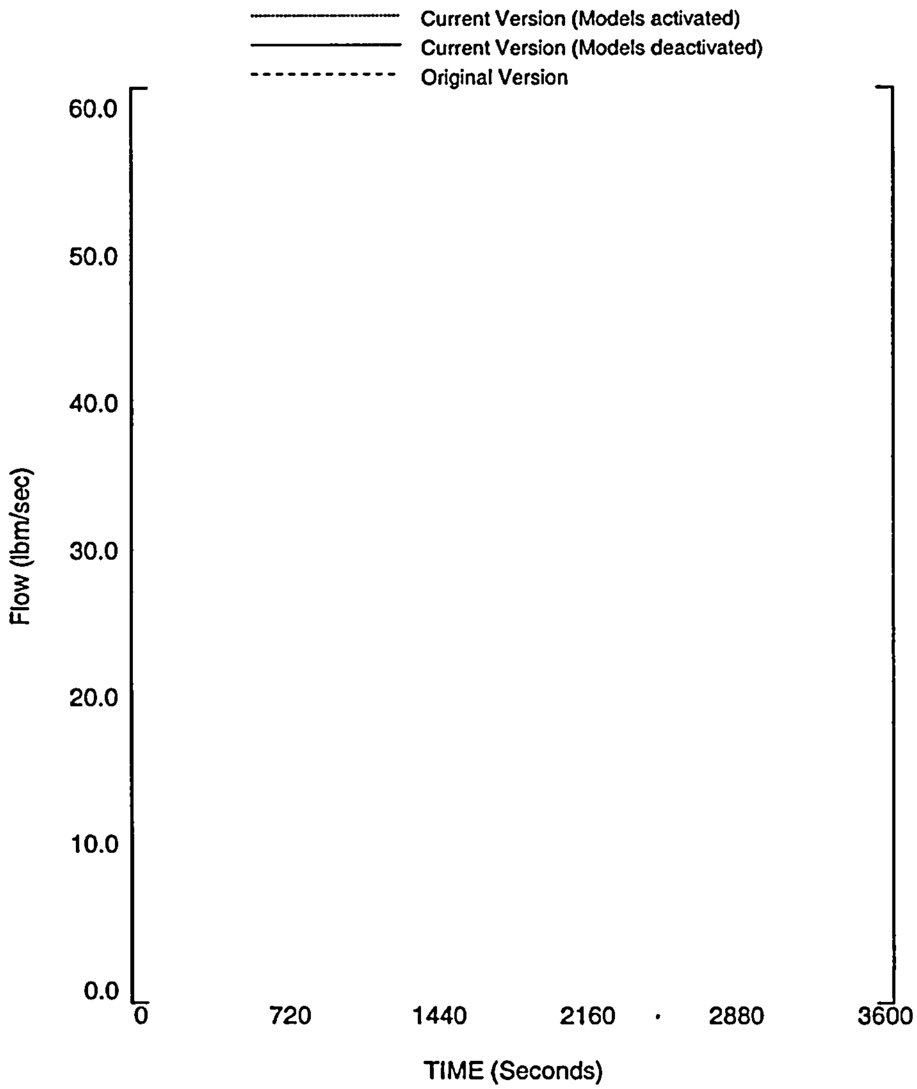


Figure 3.2.2.M

Break Flow, Total Flow

Steam Generator Tube Rupture for Plant E

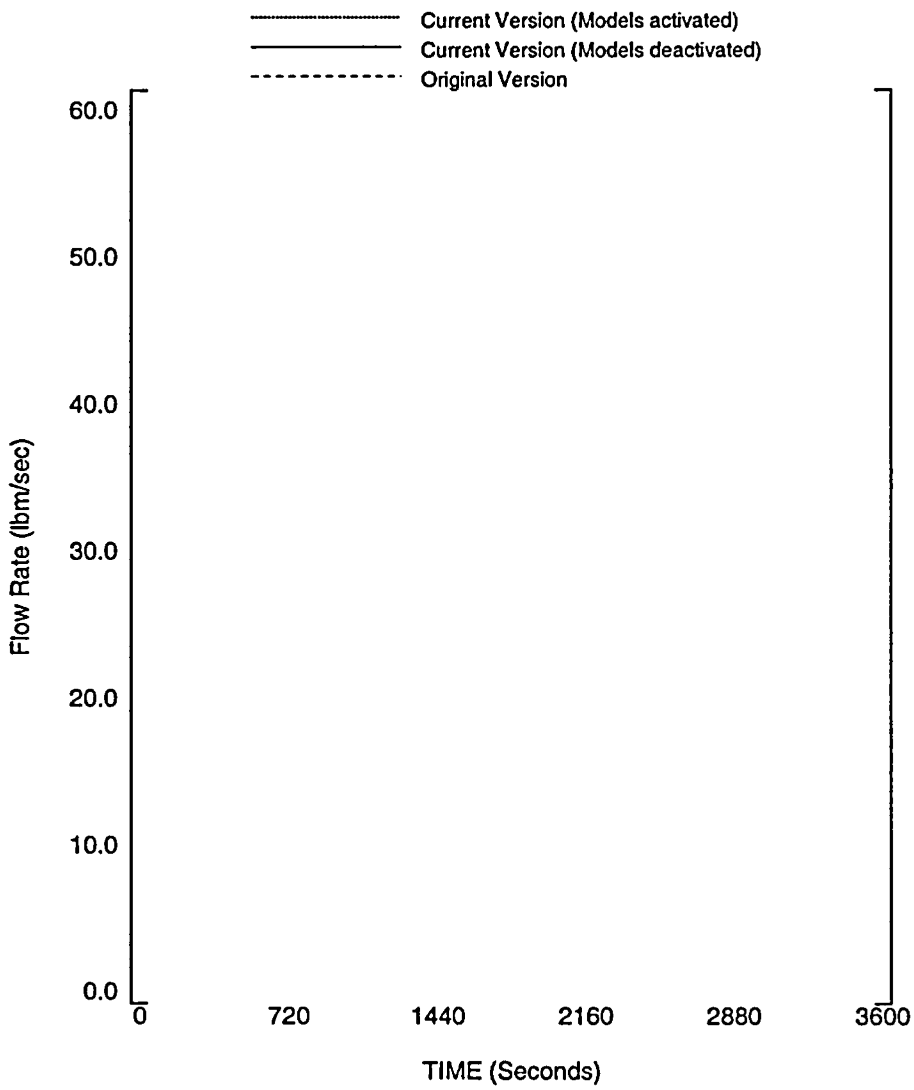


Figure 3.2.2.N

Steam Generator Liquid Mass, Affected Steam Generator

Steam Generator Tube Rupture for Plant E

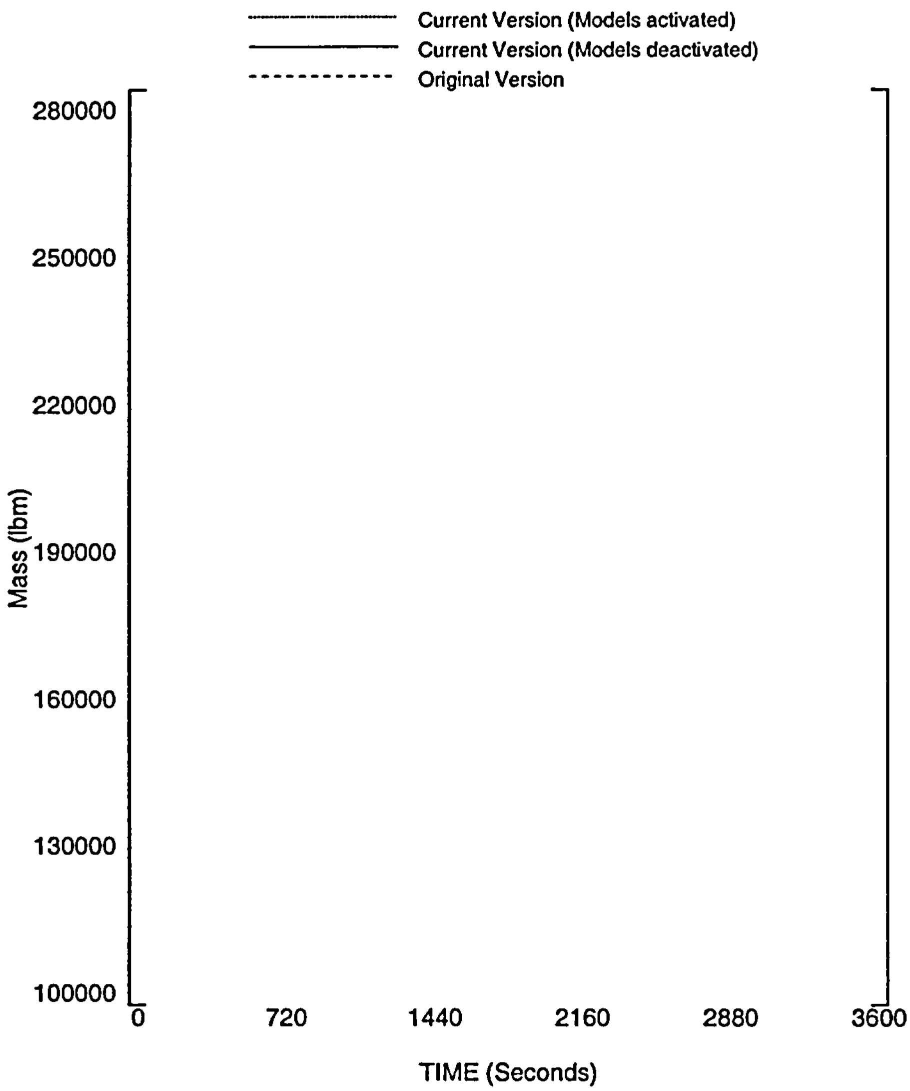


Figure 3.2.2.O

Steam Generator Liquid Mass, Intact Steam Generator

Steam Generator Tube Rupture for Plant E

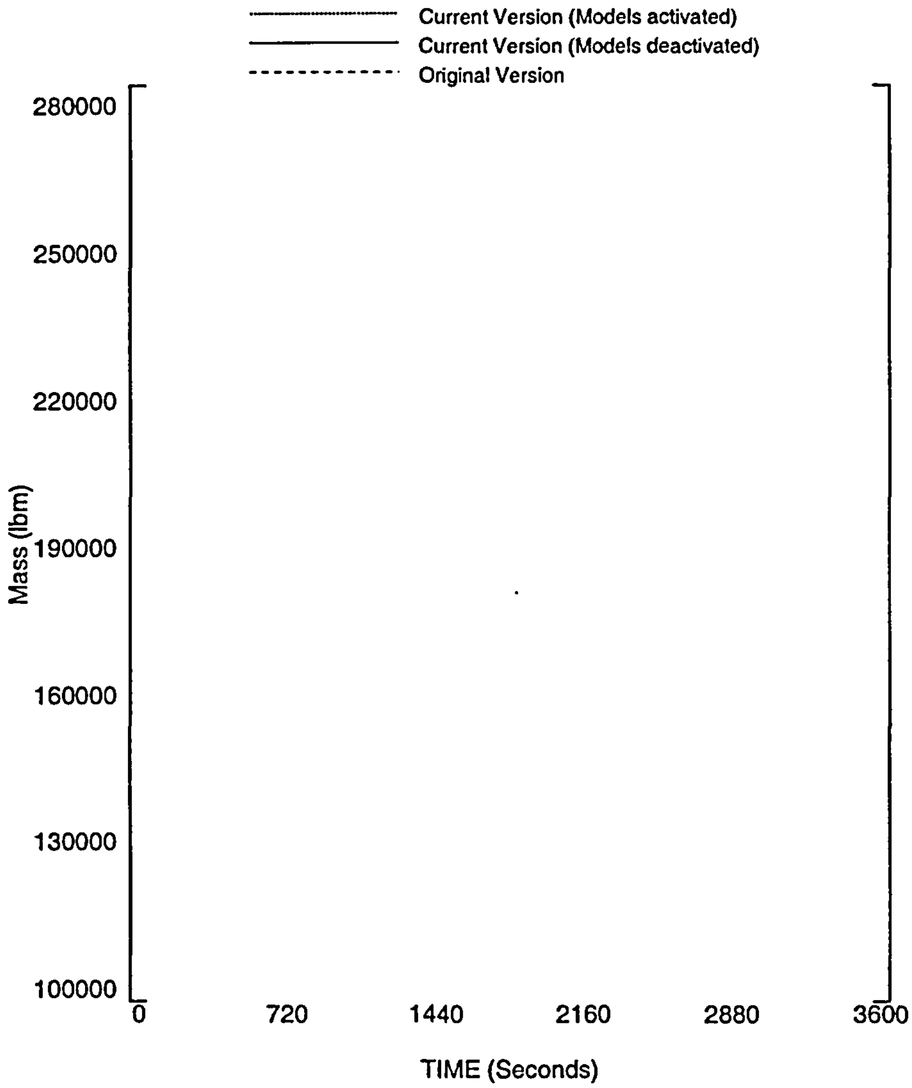


Figure 3.2.2.P

Steam Generator Steam Flow, Affected Steam Generator

Steam Generator Tube Rupture for Plant E

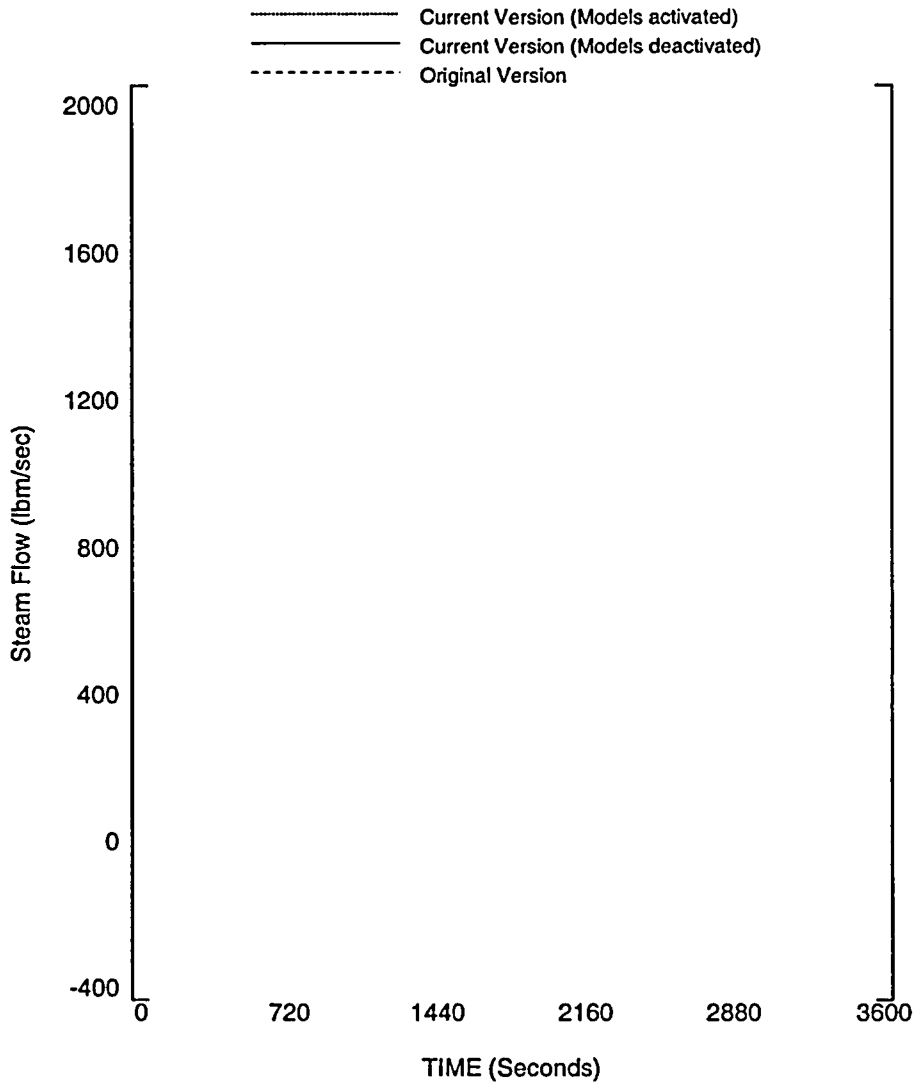


Figure 3.2.2.Q

Steam Generator Steam Flow, Intact Steam Generator

Steam Generator Tube Rupture for Plant E

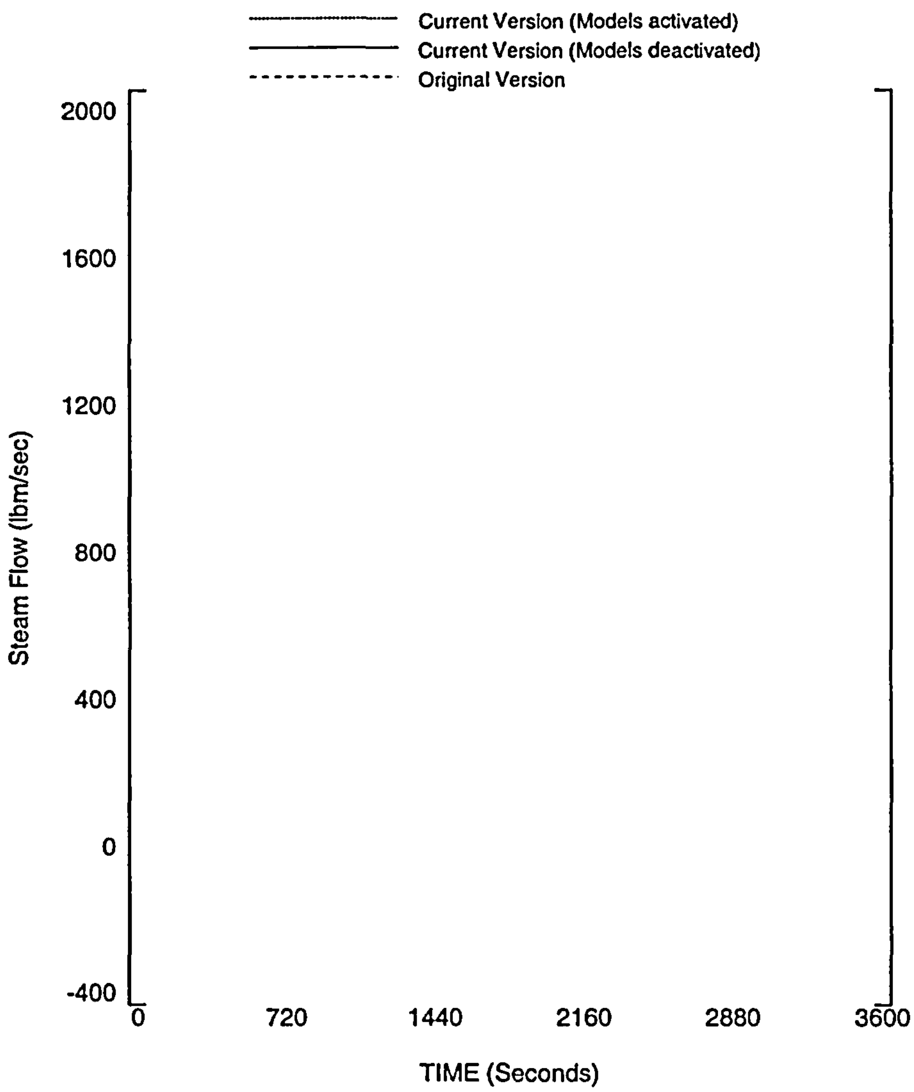


Figure 3.2.2.R

Affected Steam Generator, Safety Valve Flow

Steam Generator Tube Rupture for Plant E

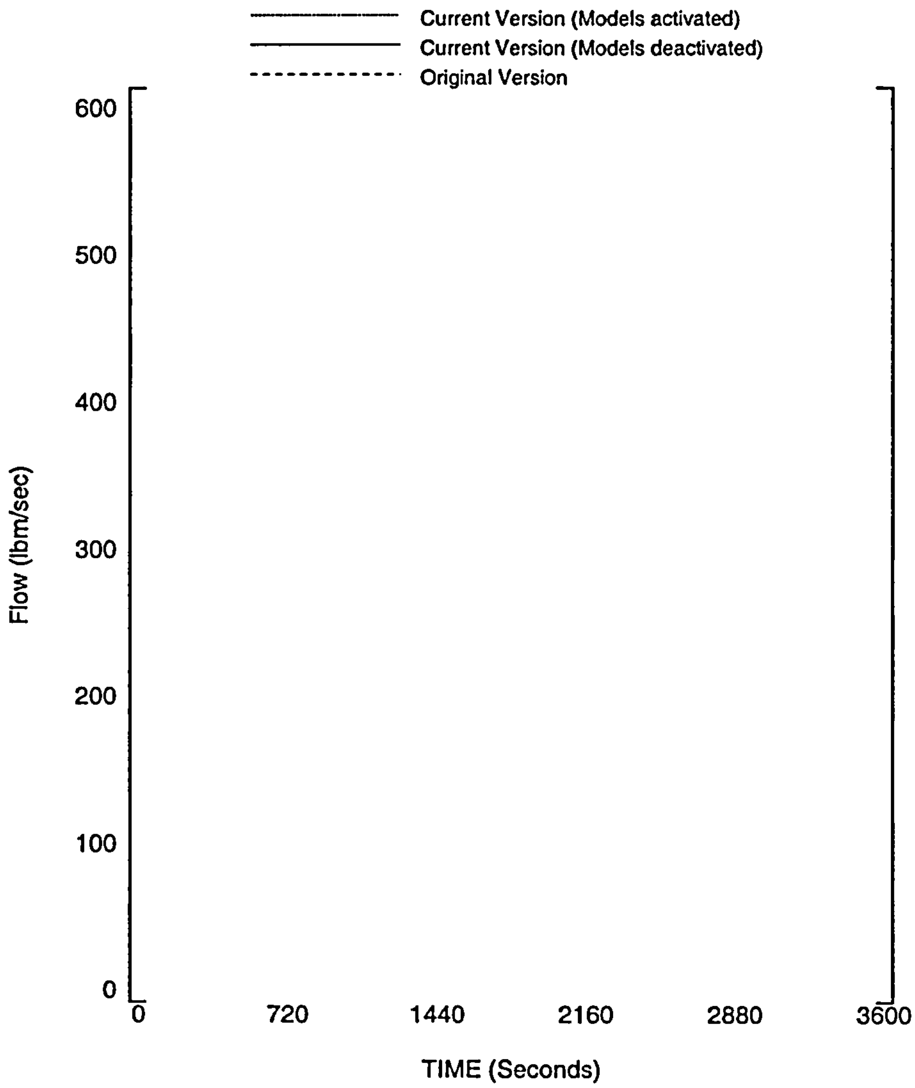


Figure 3.2.2.S

Intact Steam Generator, Safety Valves Flow

Steam Generator Tube Rupture for Plant E

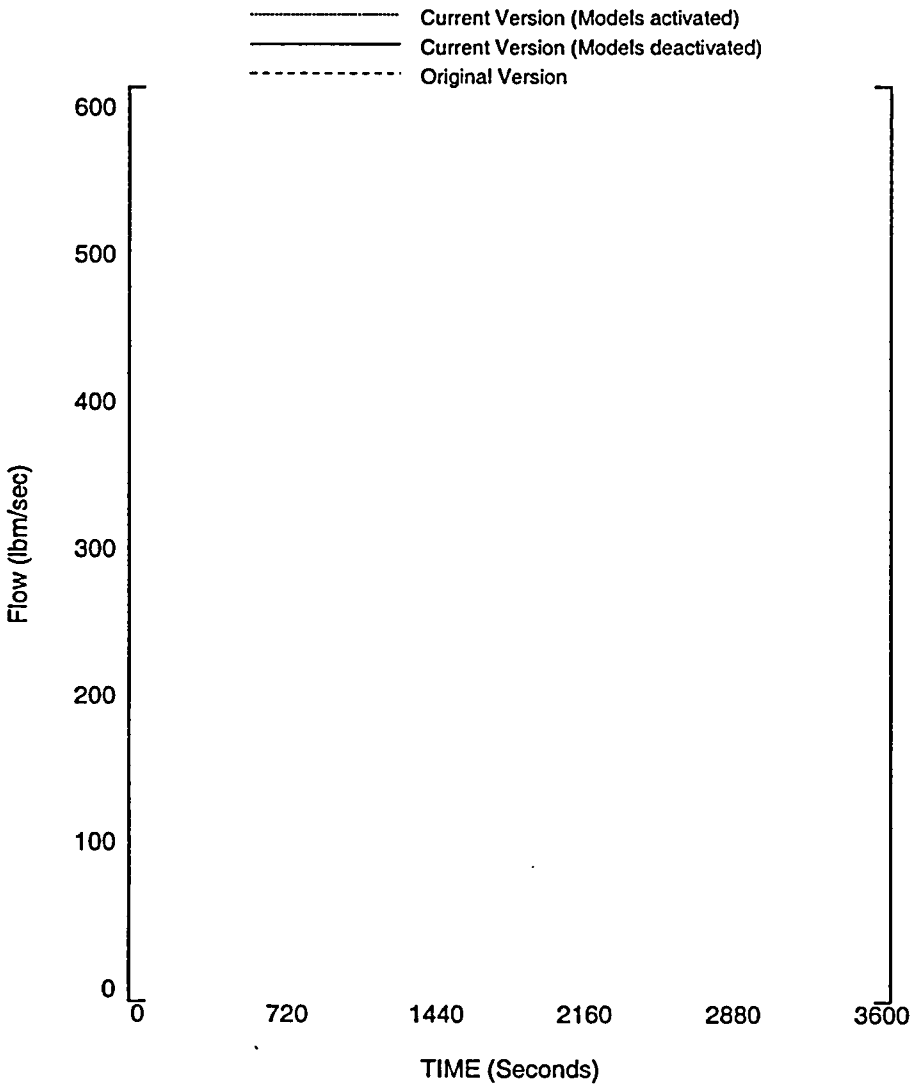


Figure 3.2.2.T
RCS Node Iodine Concentrations
Steam Generator Tube Rupture for Plant E - GIS Case

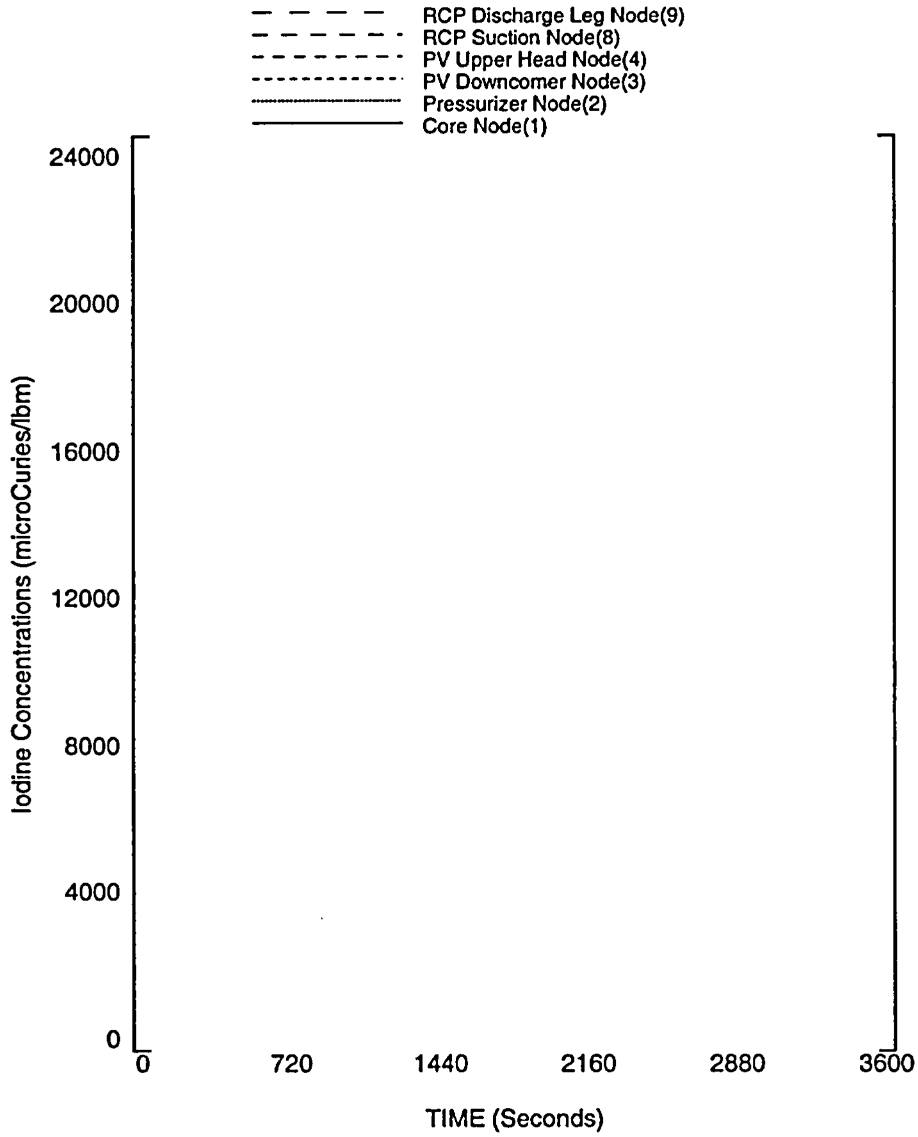


Figure 3.2.2.U

Total Iodine in RCS, Secondary & Atmosphere

Steam Generator Tube Rupture for Plant E - GIS Case

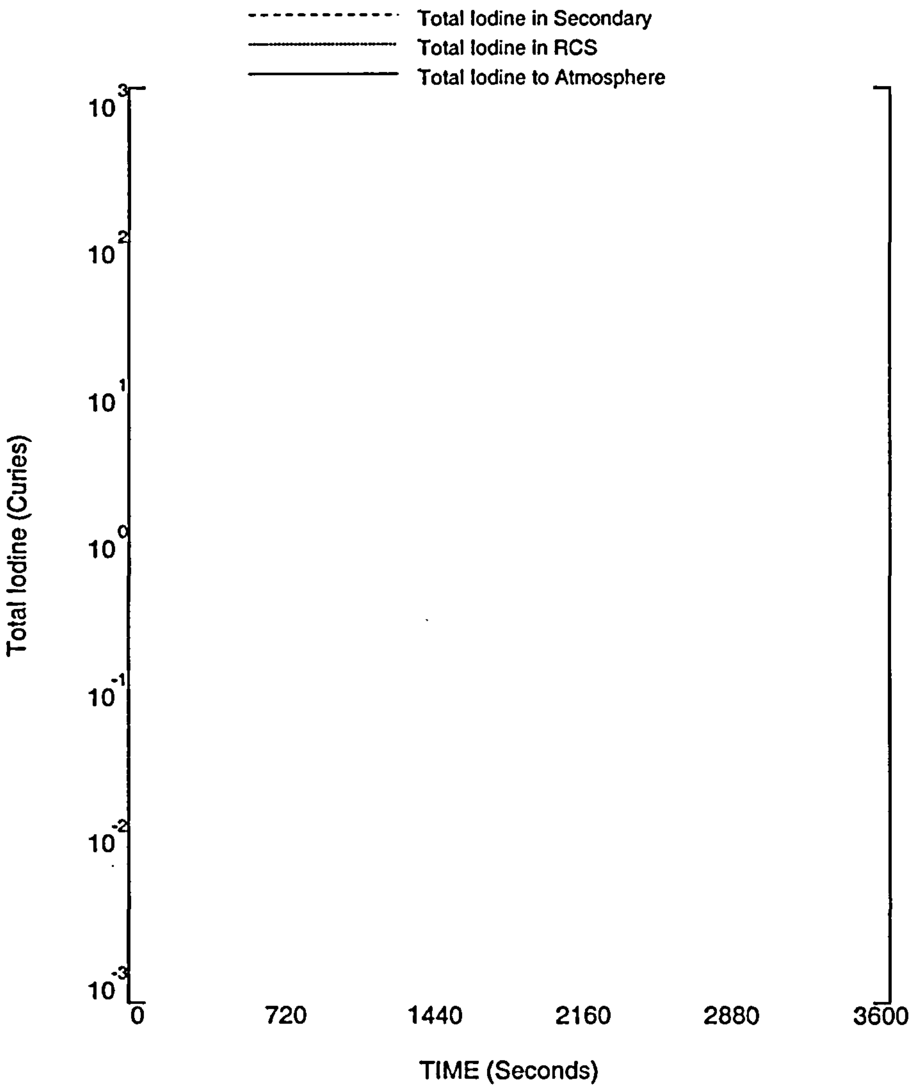


Figure 3.2.2.V
Secondary Side Iodine Concentrations
Steam Generator Tube Rupture for Plant E - GIS Case

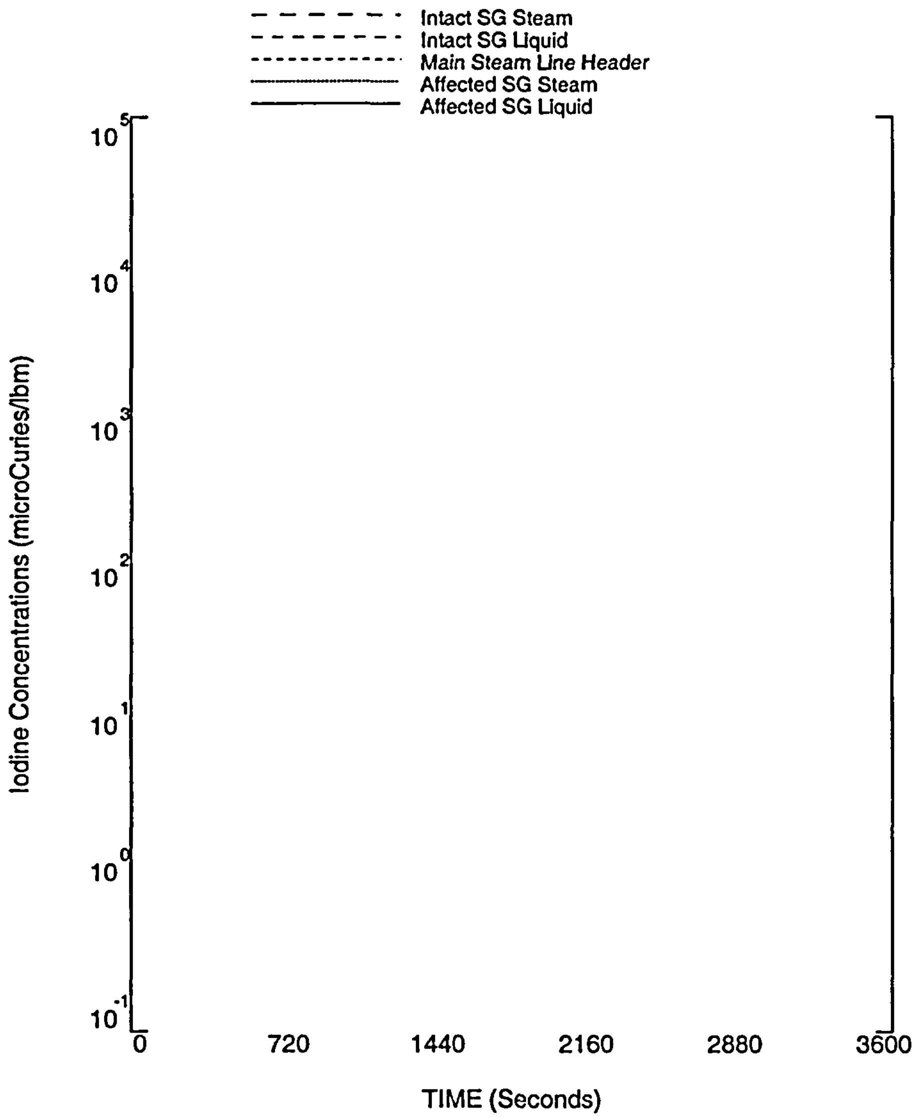


Figure 3.2.2.W

Iodine Transport from RCS to Secondary

Steam Generator Tube Rupture for Plant E - GIS Case

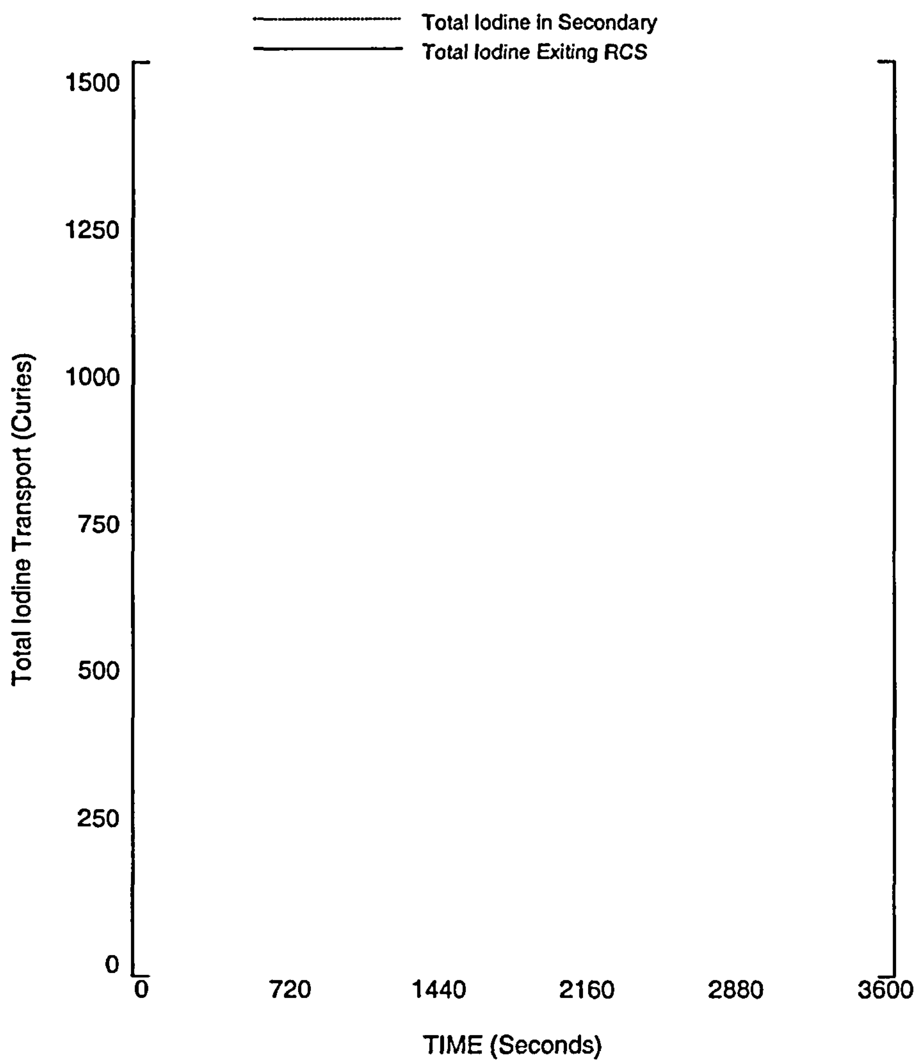


Figure 3.2.2.X

Accumulated Thyroid Doses (to 1 Hour)

Steam Generator Tube Rupture for Plant E - GIS Case

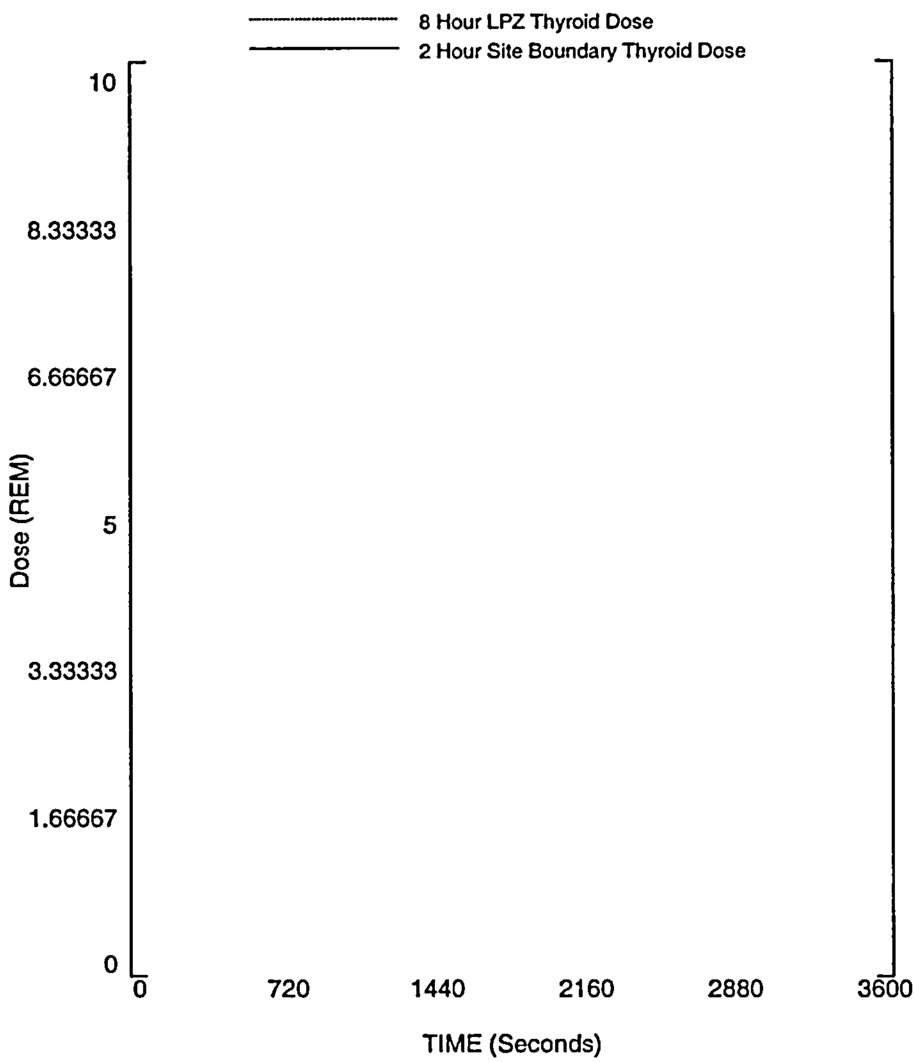


Figure 3.2.2.Y
RCS Node Iodine Concentrations
Steam Generator Tube Rupture for Plant E - PIS Case

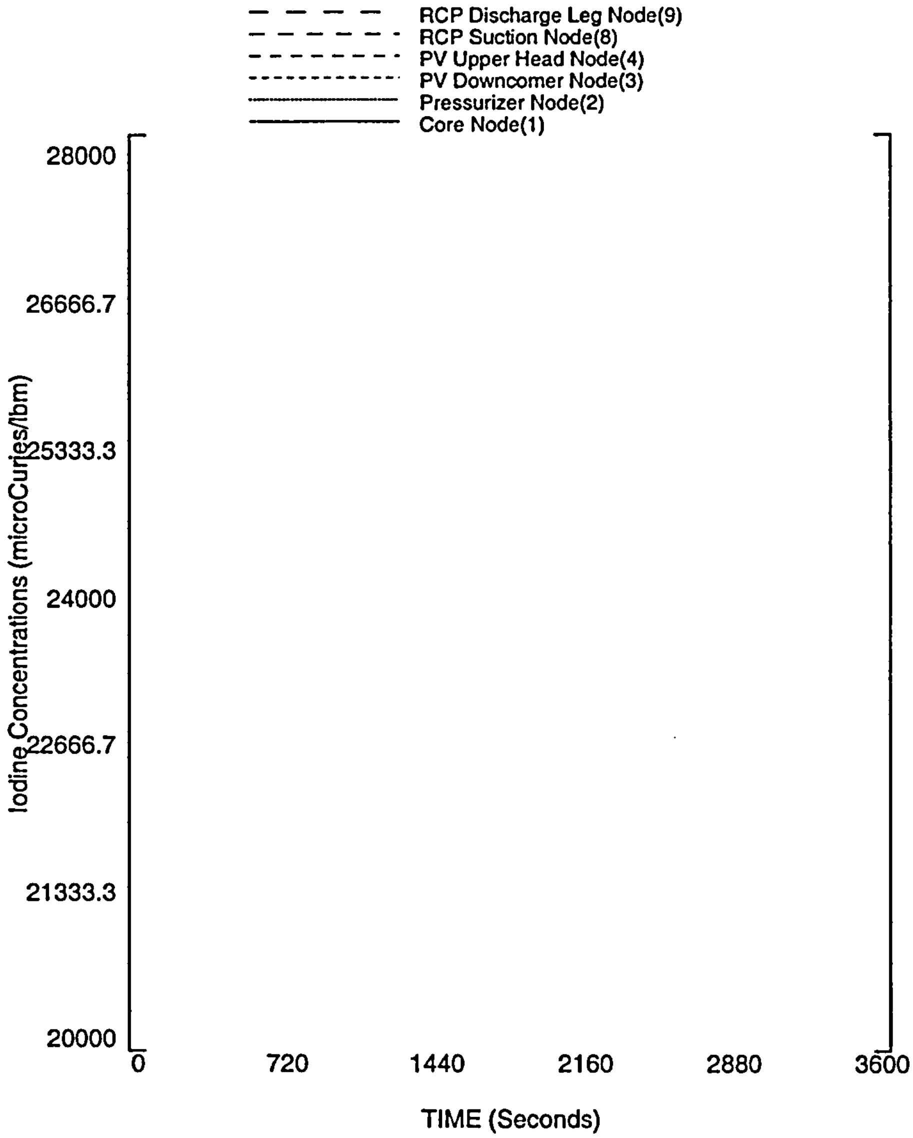


Figure 3.2.2.Z

Total Iodine in RCS, Secondary & Atmosphere

Steam Generator Tube Rupture for Plant E - PIS Case

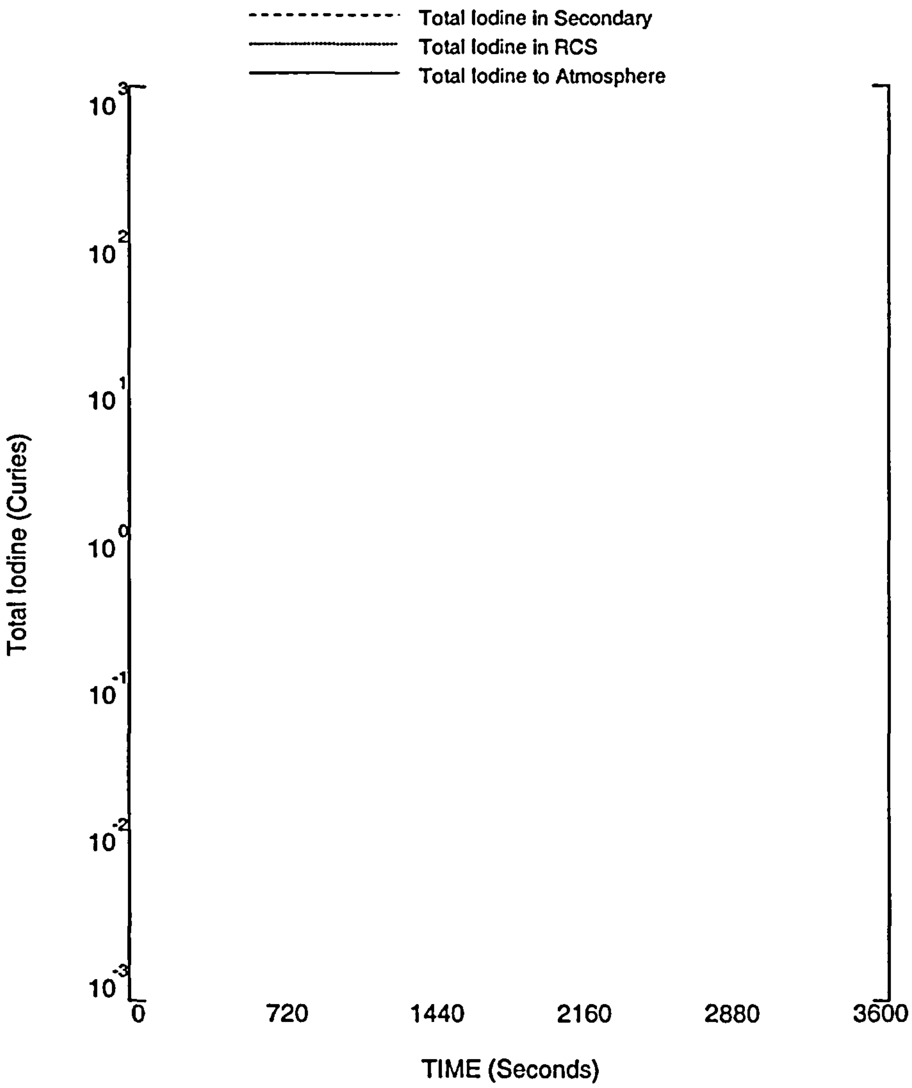


Figure 3.2.2.AA
Secondary Side Iodine Concentrations
Steam Generator Tube Rupture for Plant E - PIS Case

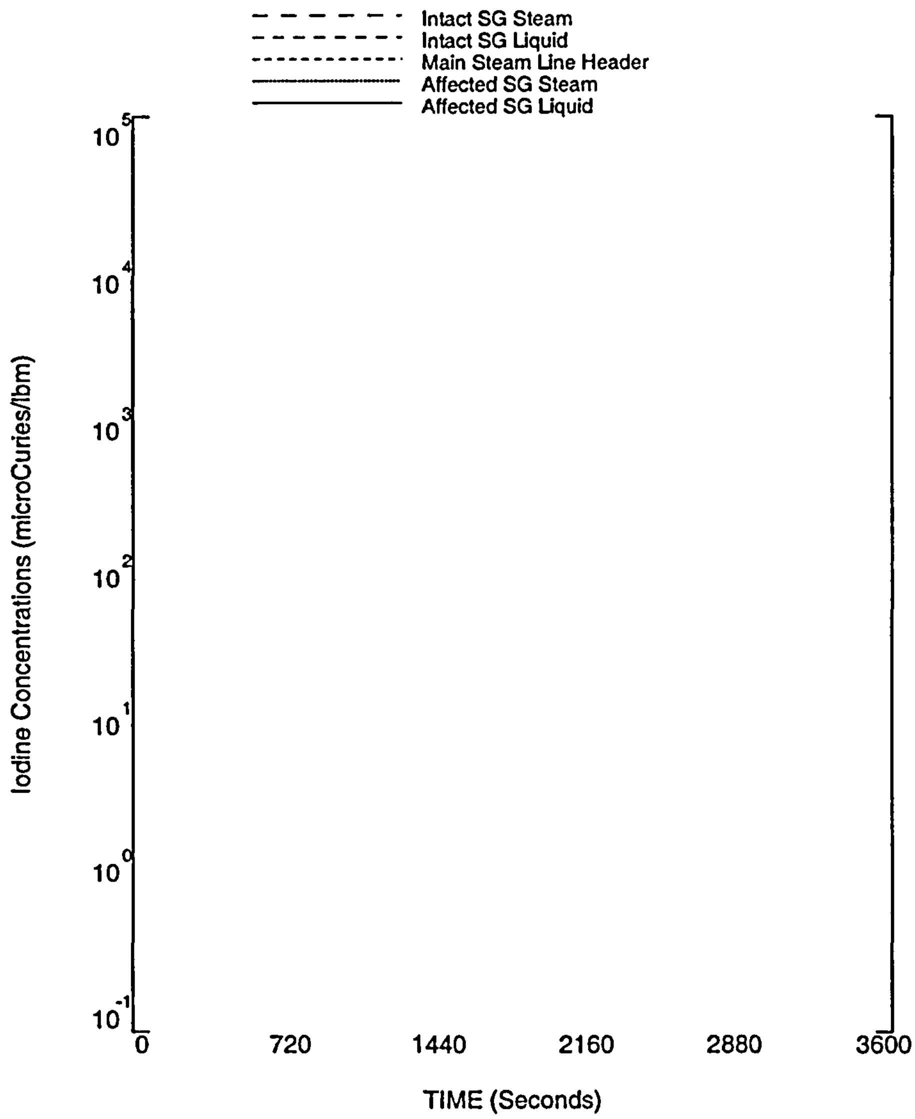


Figure 3.2.2.AB

Iodine Transport from RCS to Secondary

Steam Generator Tube Rupture for Plant E - PIS Case

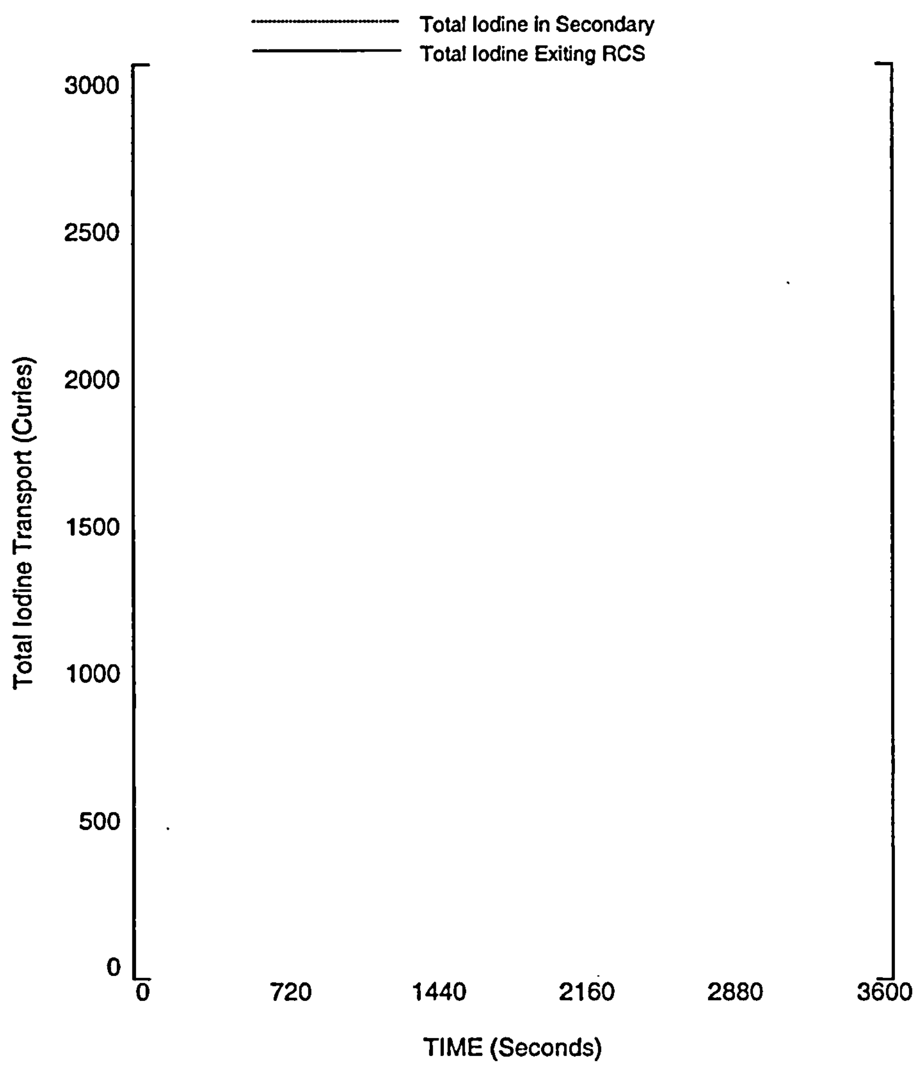


Figure 3.2.2.AC

Accumulated Thyroid Doses (to 1 Hour)

Steam Generator Tube Rupture for Plant E - PIS Case

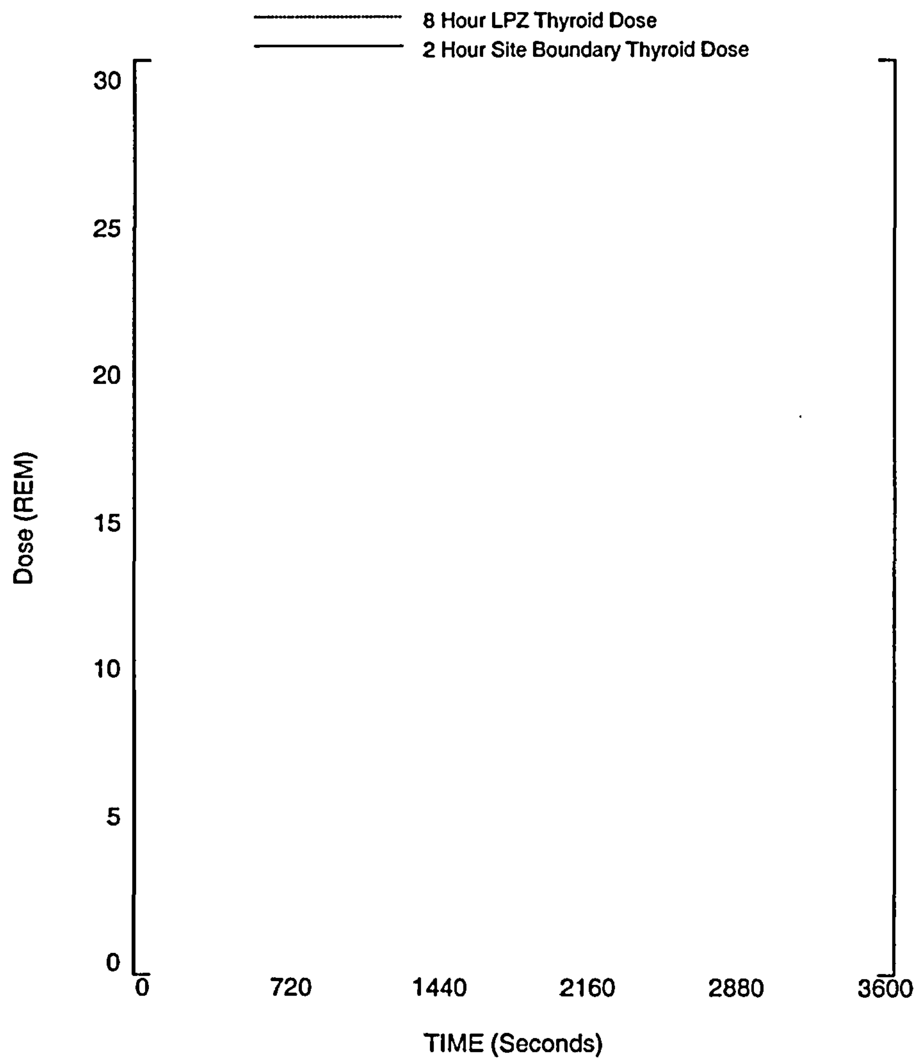


Figure 3.4.1.A

Reactor Core Power

Steam Line Break Event for Plant E

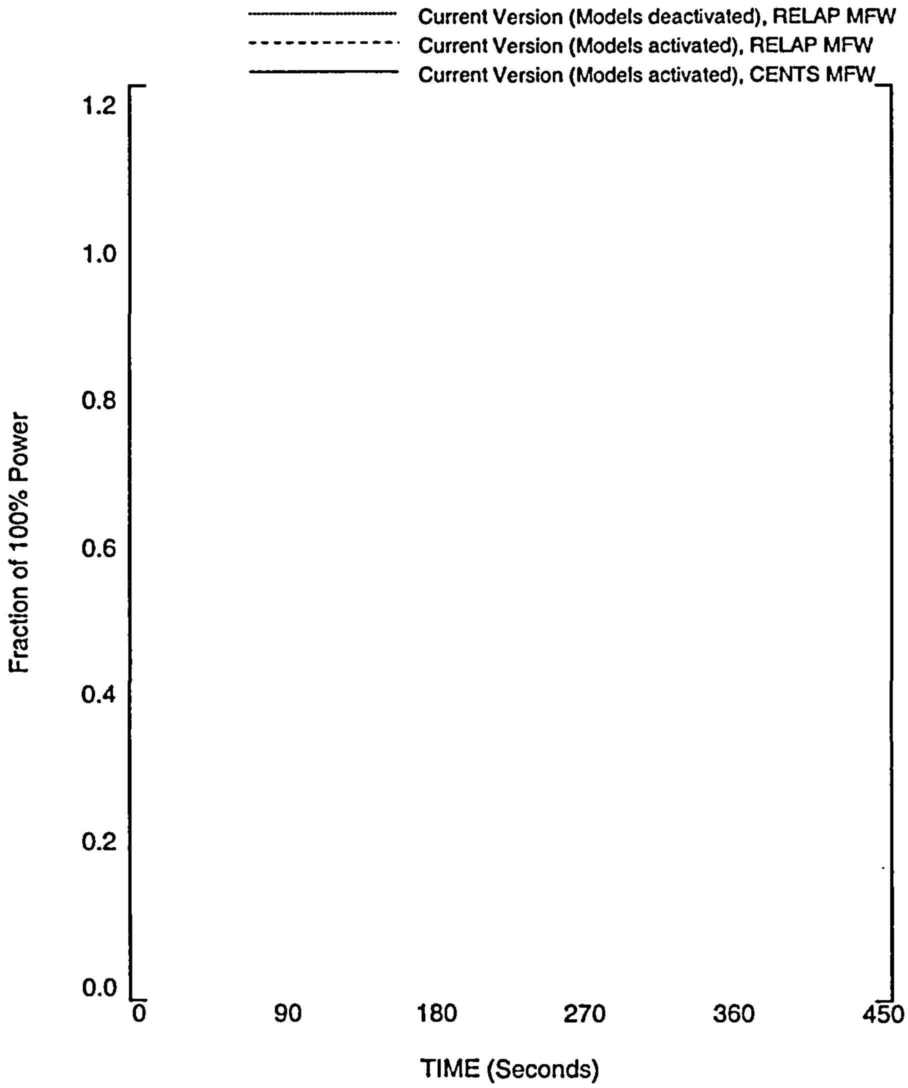


Figure 3.4.1.B

Reactor Coolant System Pressure

Steam Line Break Event for Plant E

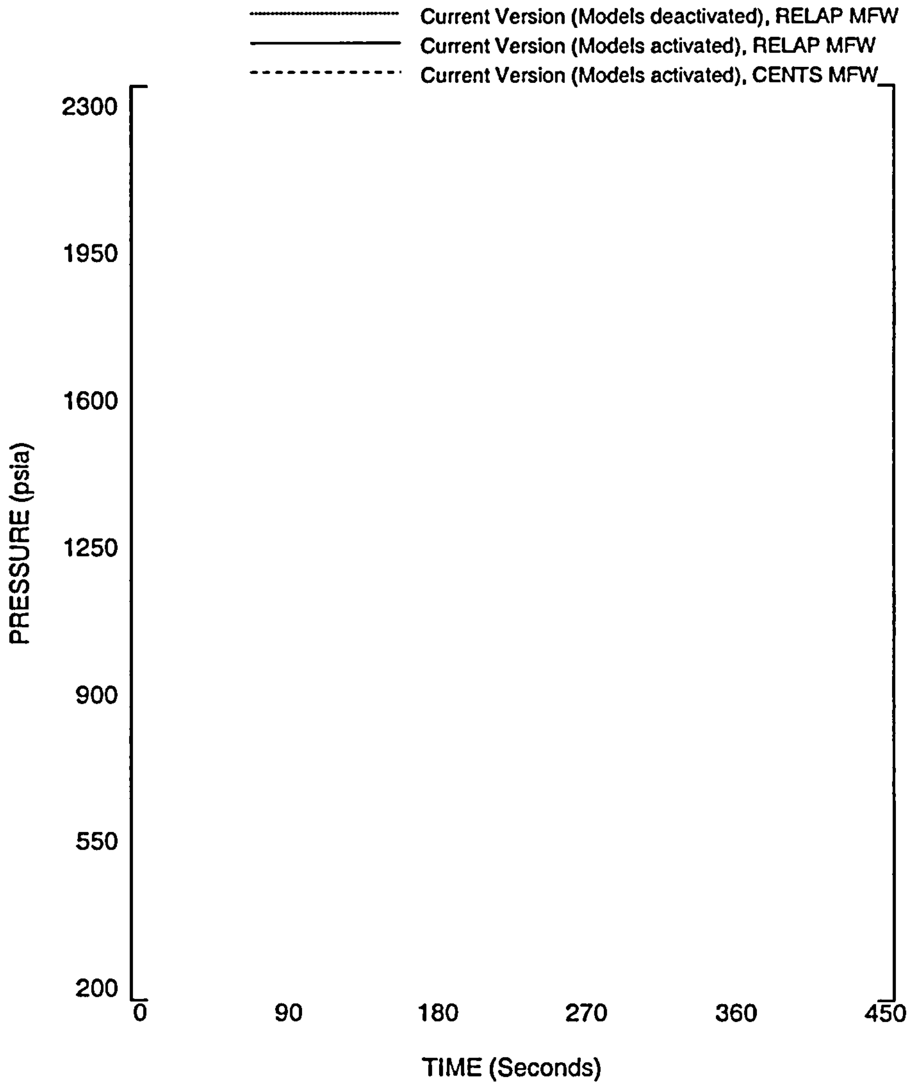


Figure 3.4.1.C

Pressurizer Pressure

Steam Line Break Event for Plant E

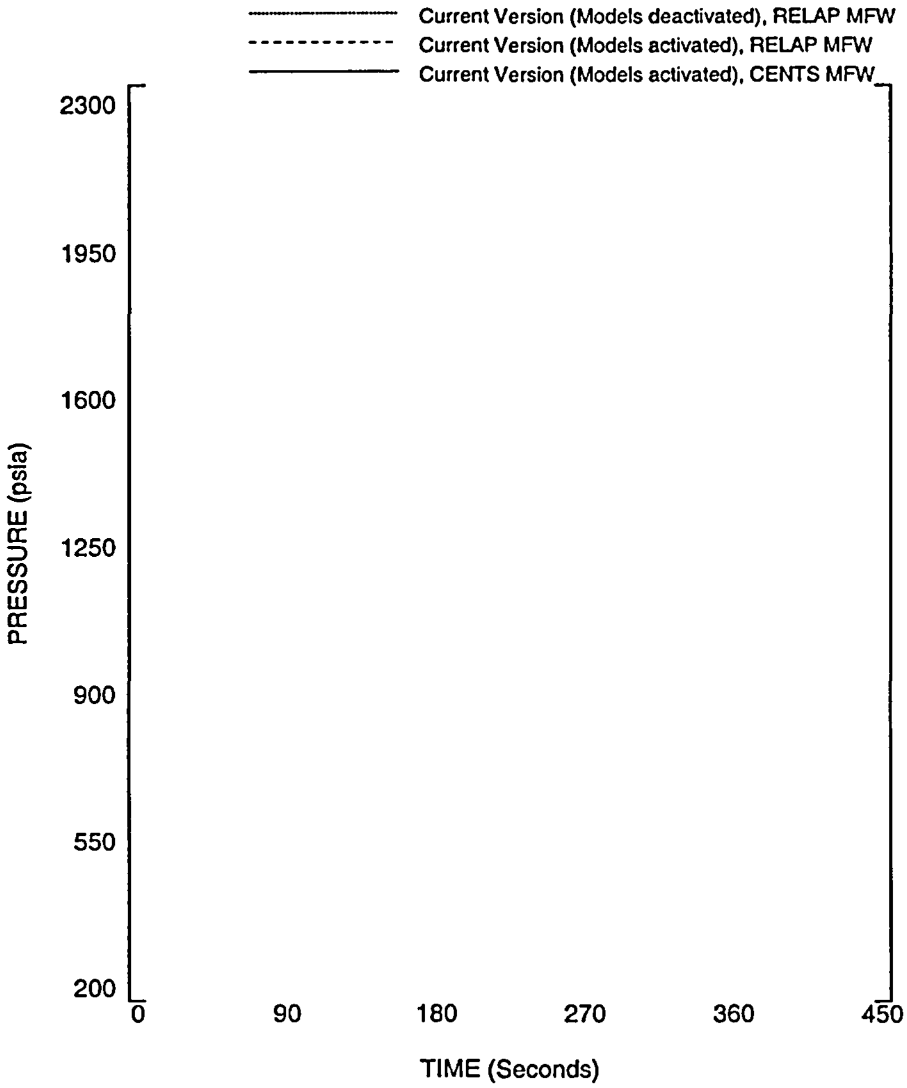


Figure 3.4.1.D

Cold Leg Temperature, Affected Loop

Steam Line Break Event for Plant E

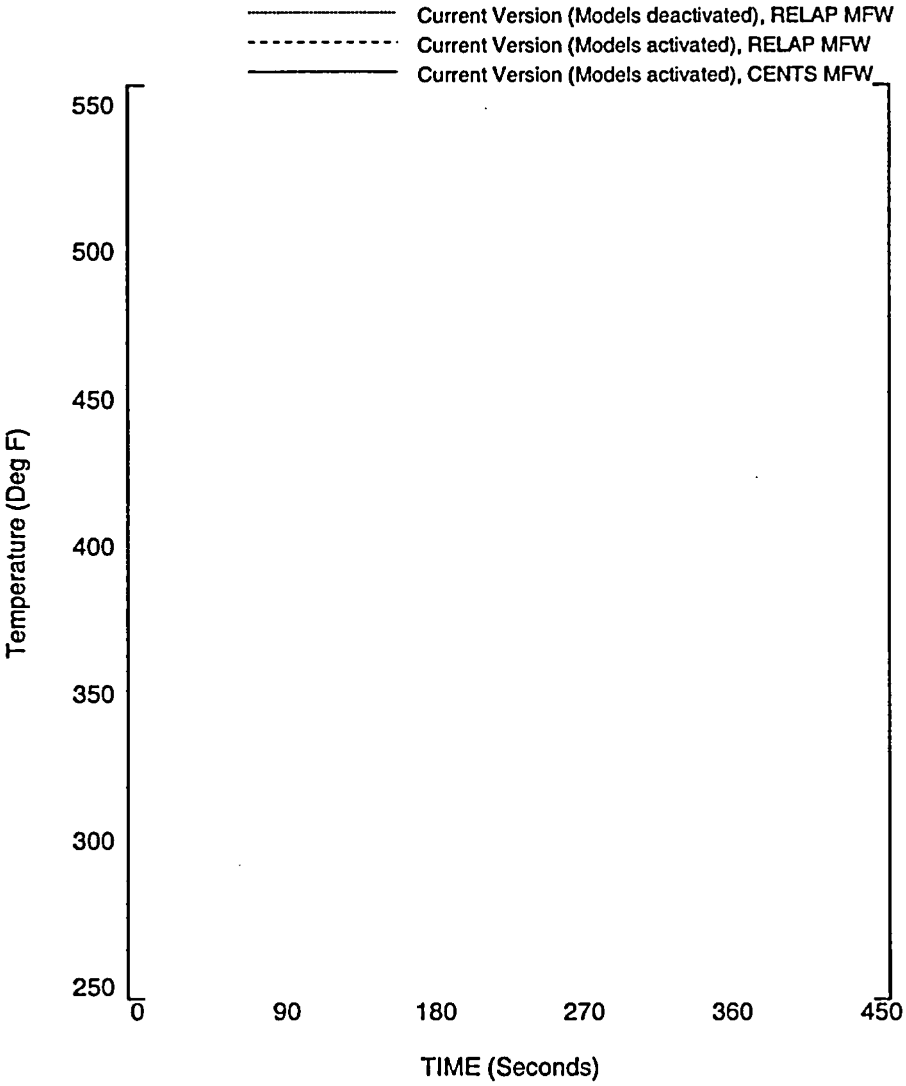


*Figure 3.4.1.E*

Cold Leg Temperature, Intact Loop

Steam Line Break Event for Plant E

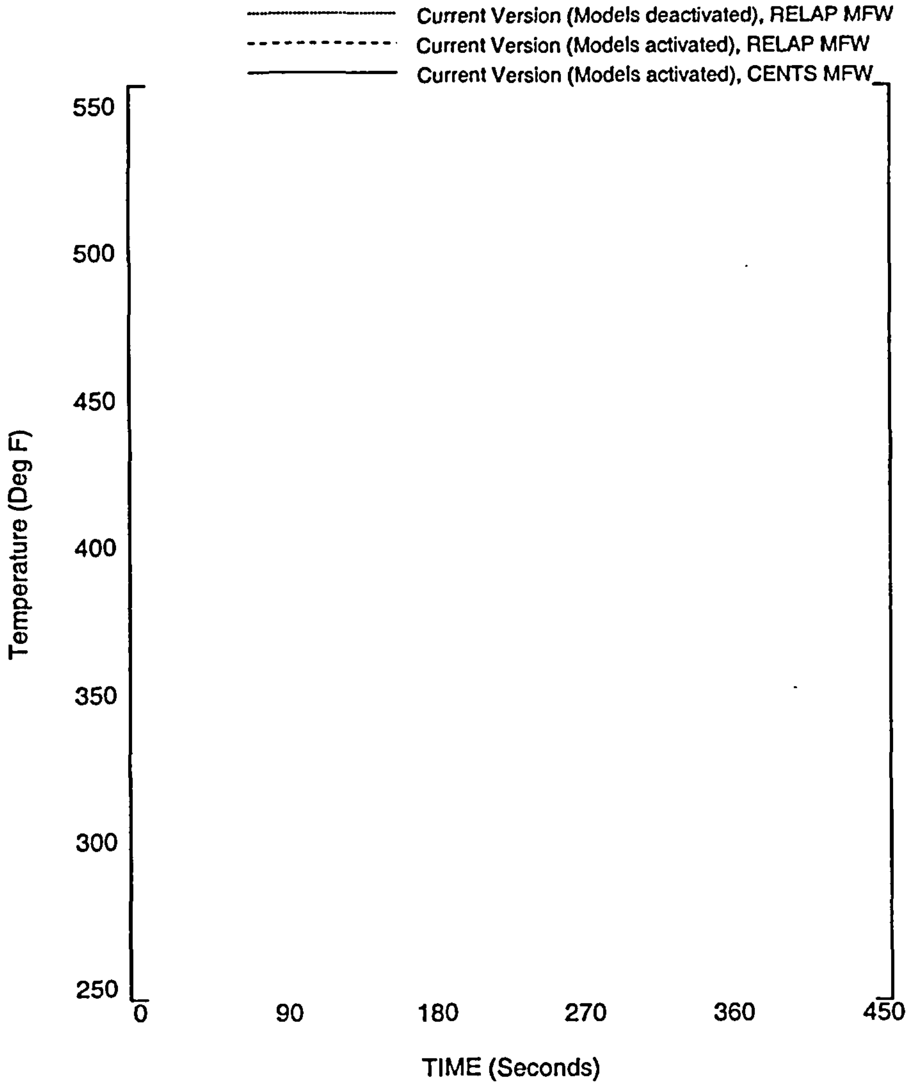


Figure 3.4.1.F

Mixed Core Inlet Temperature

Steam Line Break Event for Plant E

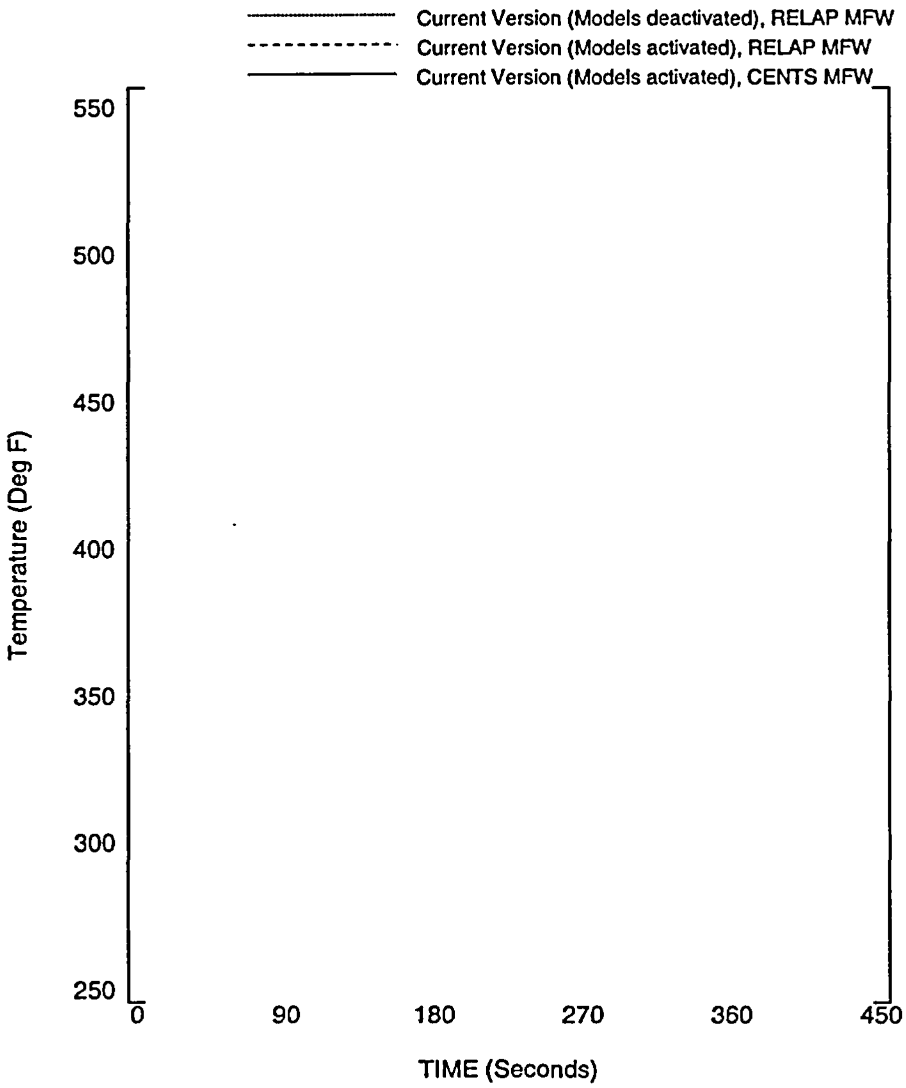


Figure 3.4.1.G

Steam Generator Pressure, Affected Steam Generator

Steam Line Break Event for Plant E

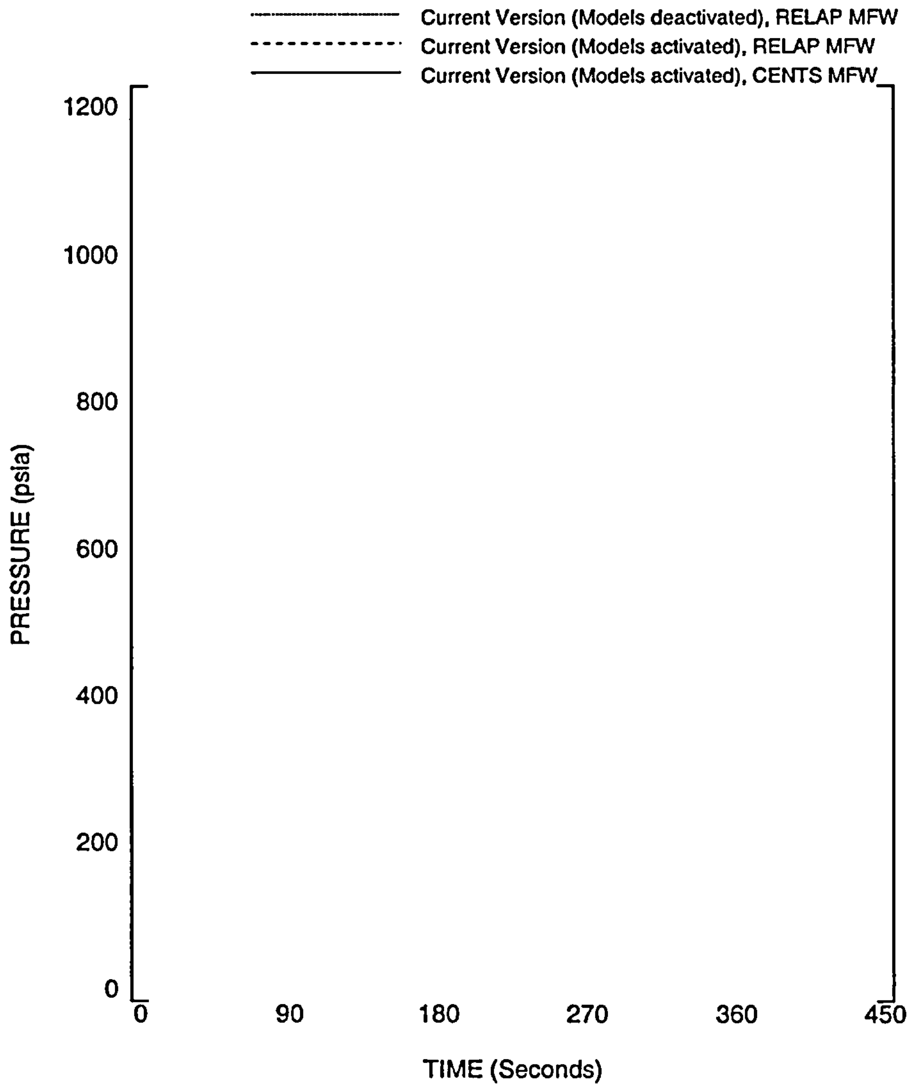


Figure 3.4.1.H

Steam Generator Pressure, Intact Steam Generator

Steam Line Break Event for Plant E

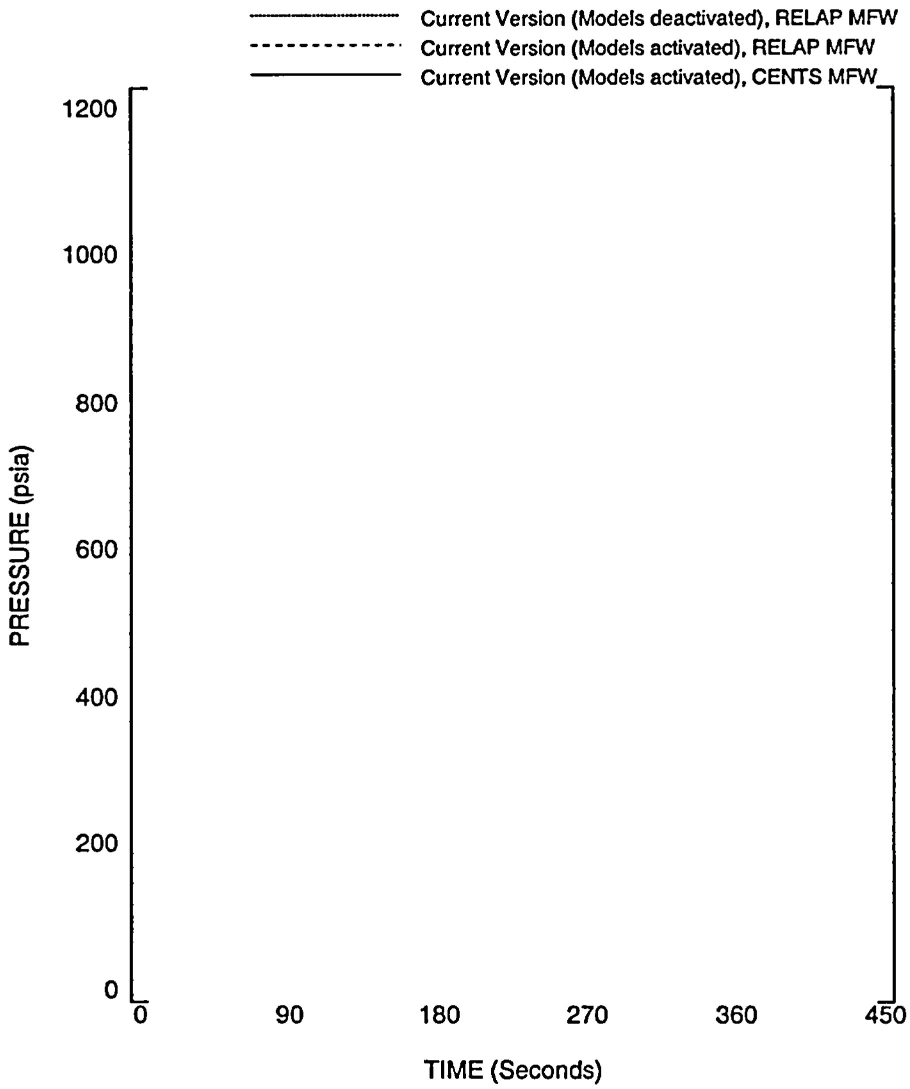


Figure 3.4.1.l

Total Steam Flow, Affected Steam Generator

Steam Line Break Event for Plant E

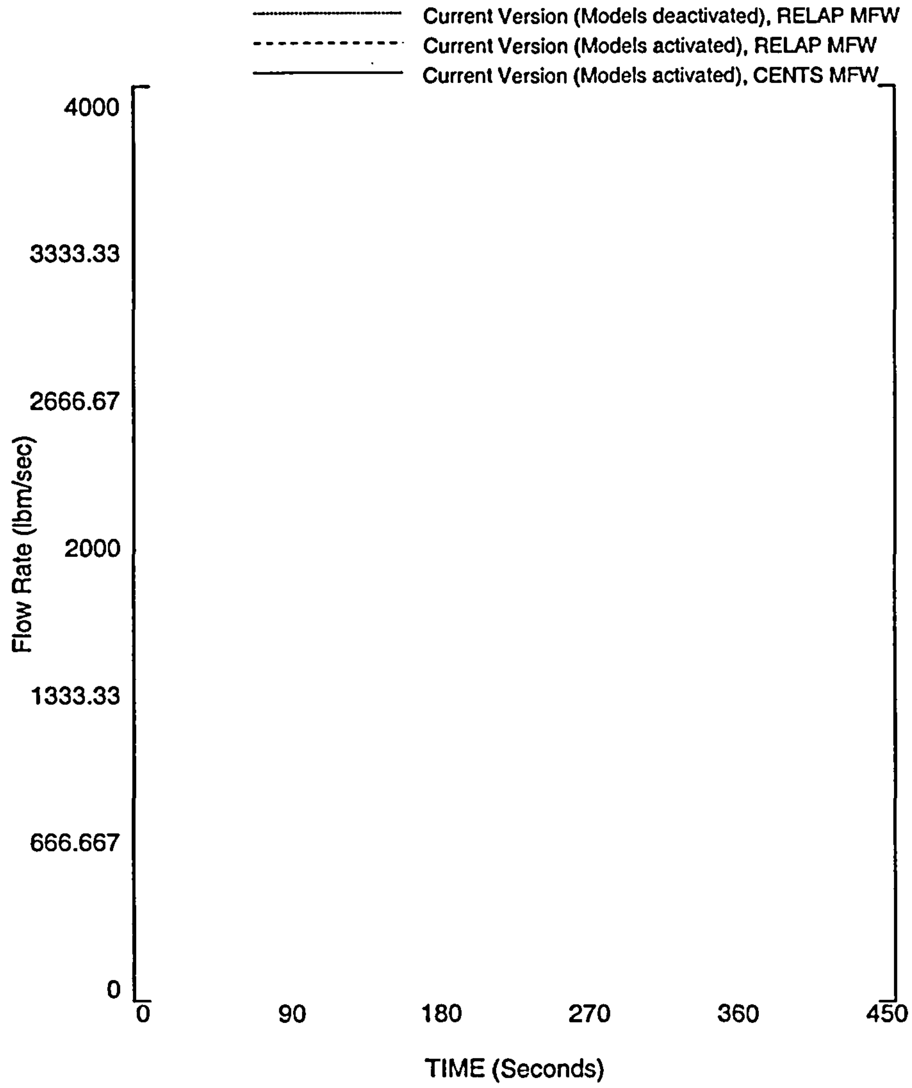


Figure 3.4.1.J

Total Steam Flow, Intact Steam Generator

Steam Line Break Event for Plant E

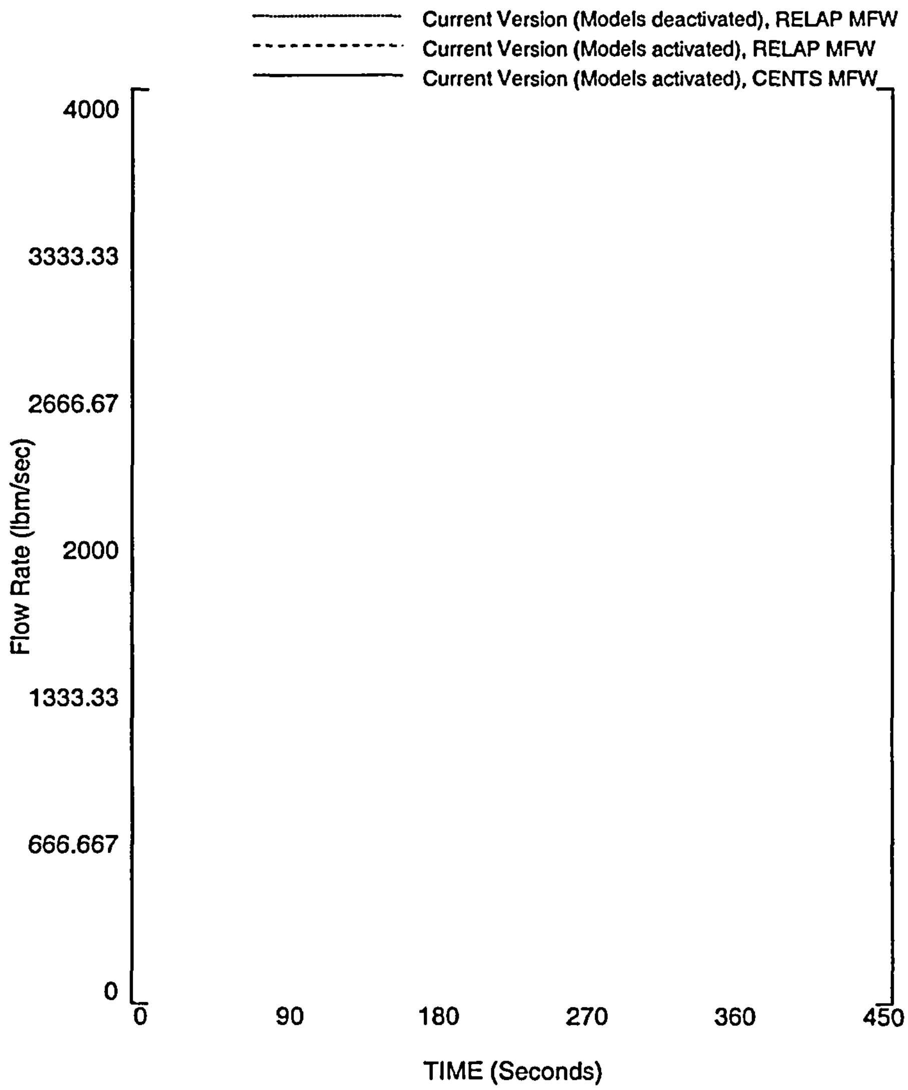


Figure 3.4.1.K

Steam Generator Liquid Mass, Affected Steam Generator

Steam Line Break Event for Plant E

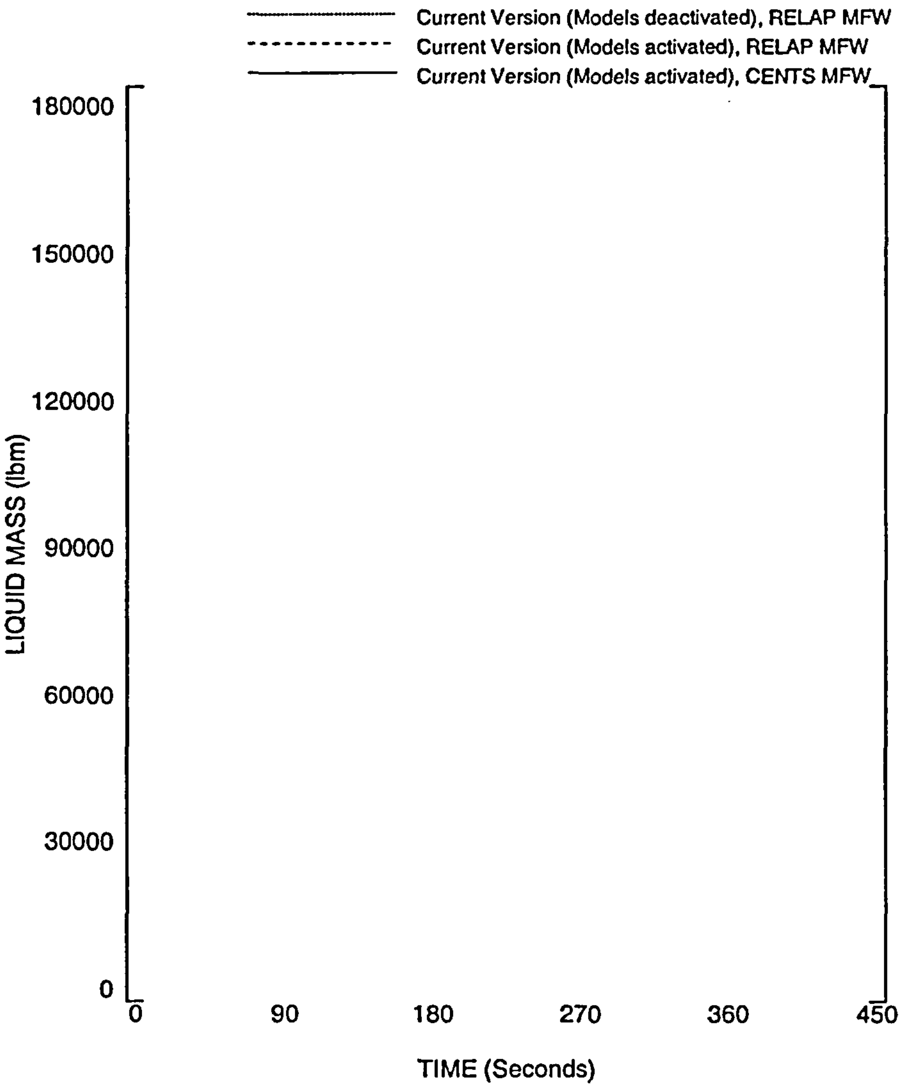


Figure 3.4.1.L

Steam Generator Liquid Mass, Intact Steam Generator

Steam Line Break Event for Plant E

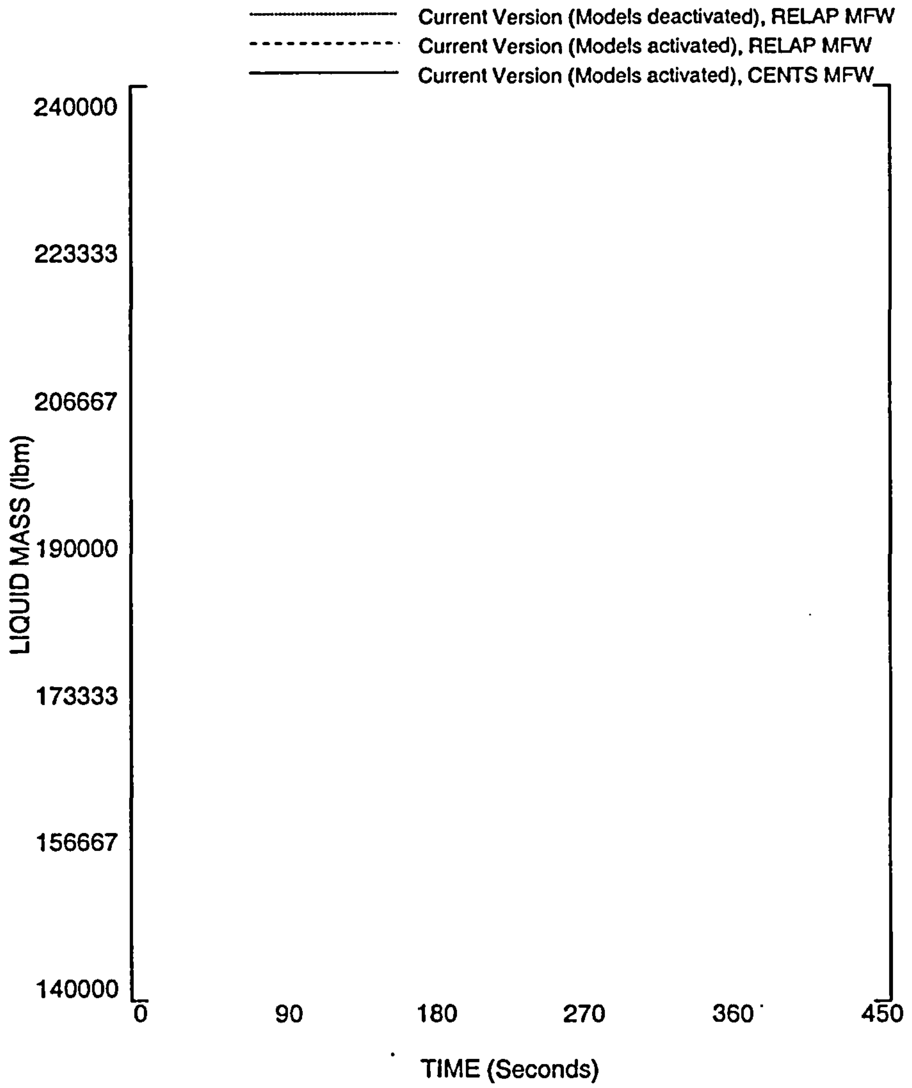


Figure 3.4.1.M

Scram Reactivity

Steam Line Break Event for Plant E

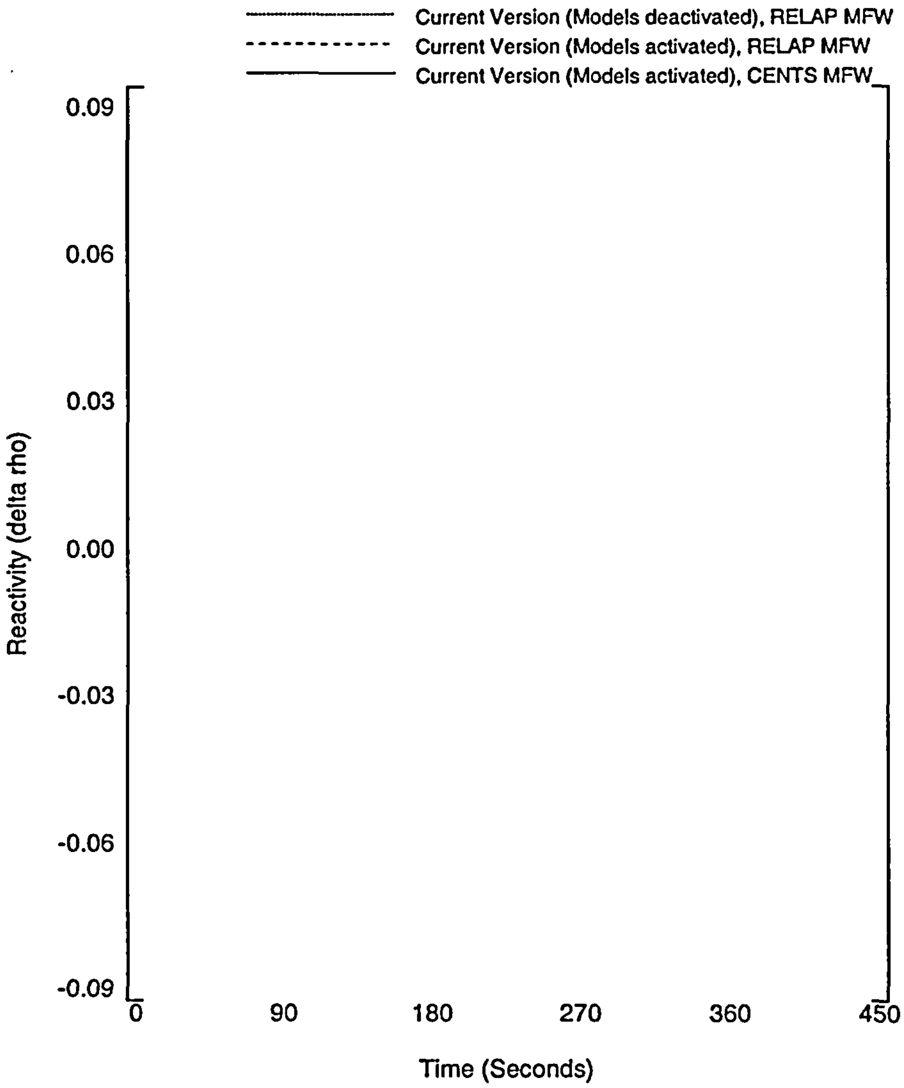


Figure 3.4.1.N

Doppler Reactivity

Steam Line Break Event for Plant E

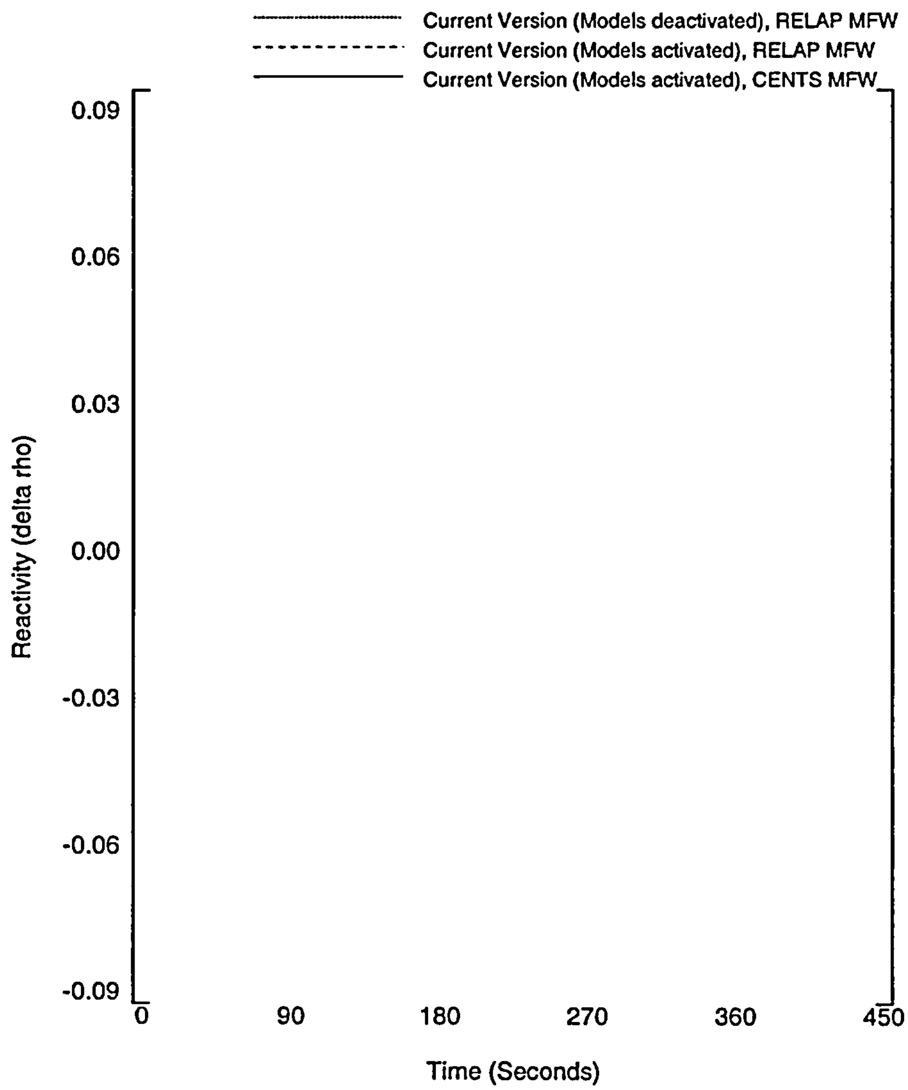


Figure 3.4.1.O

Boron Reactivity

Steam Line Break Event for Plant E

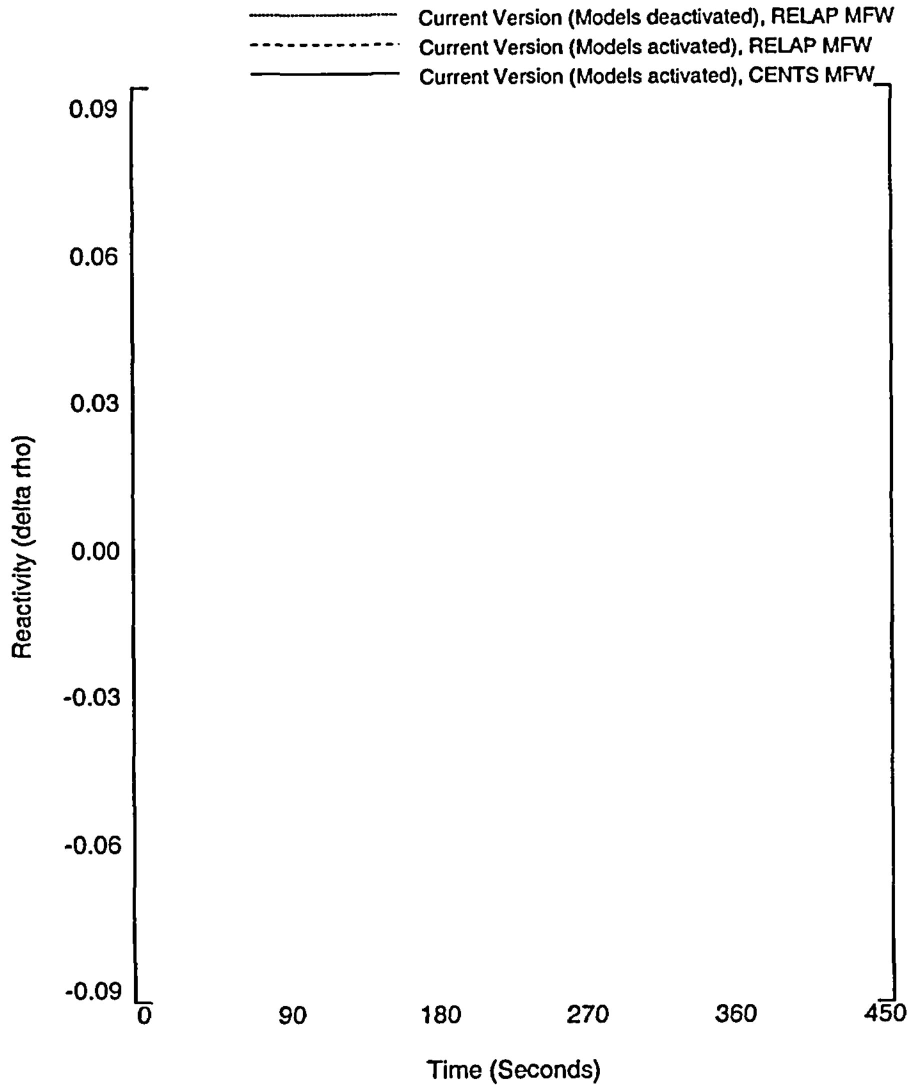


Figure 3.4.1.P

Moderator Reactivity

Steam Line Break Event for Plant E

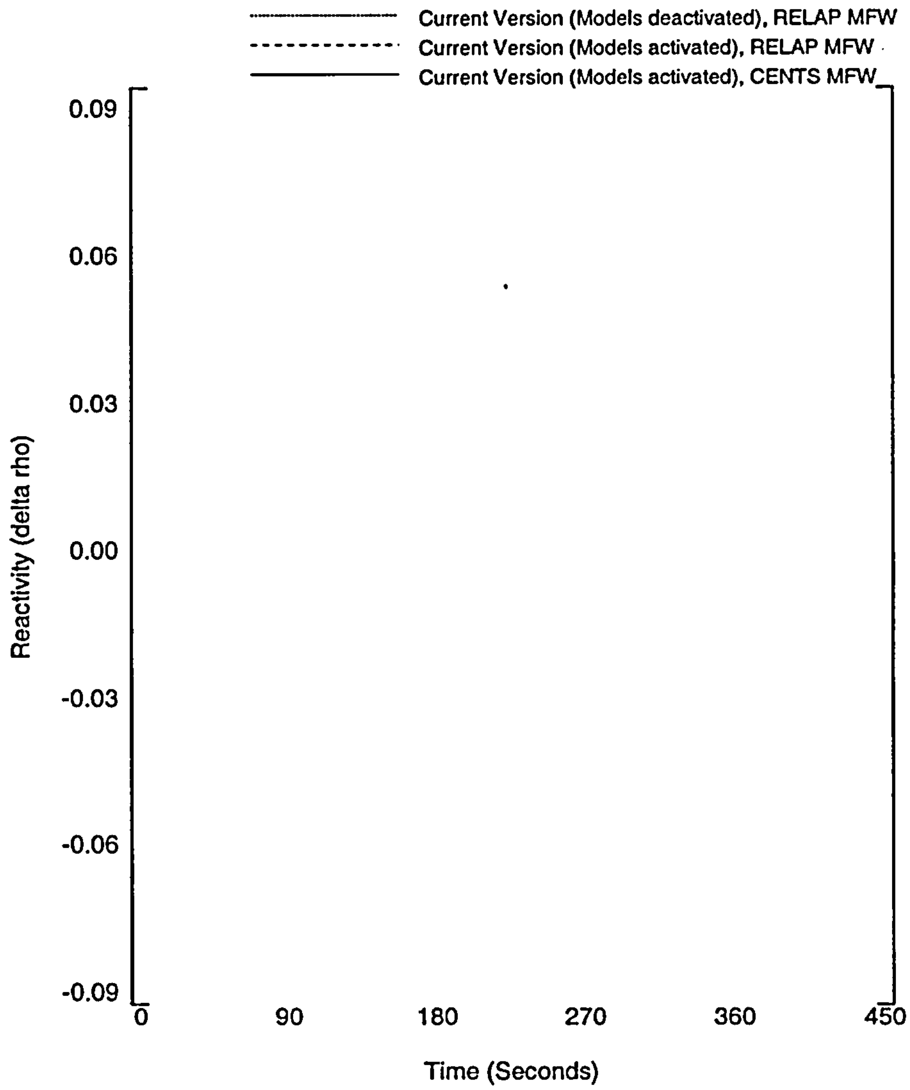


Figure 3.4.1.Q

Total Reactivity

Steam Line Break Event for Plant E

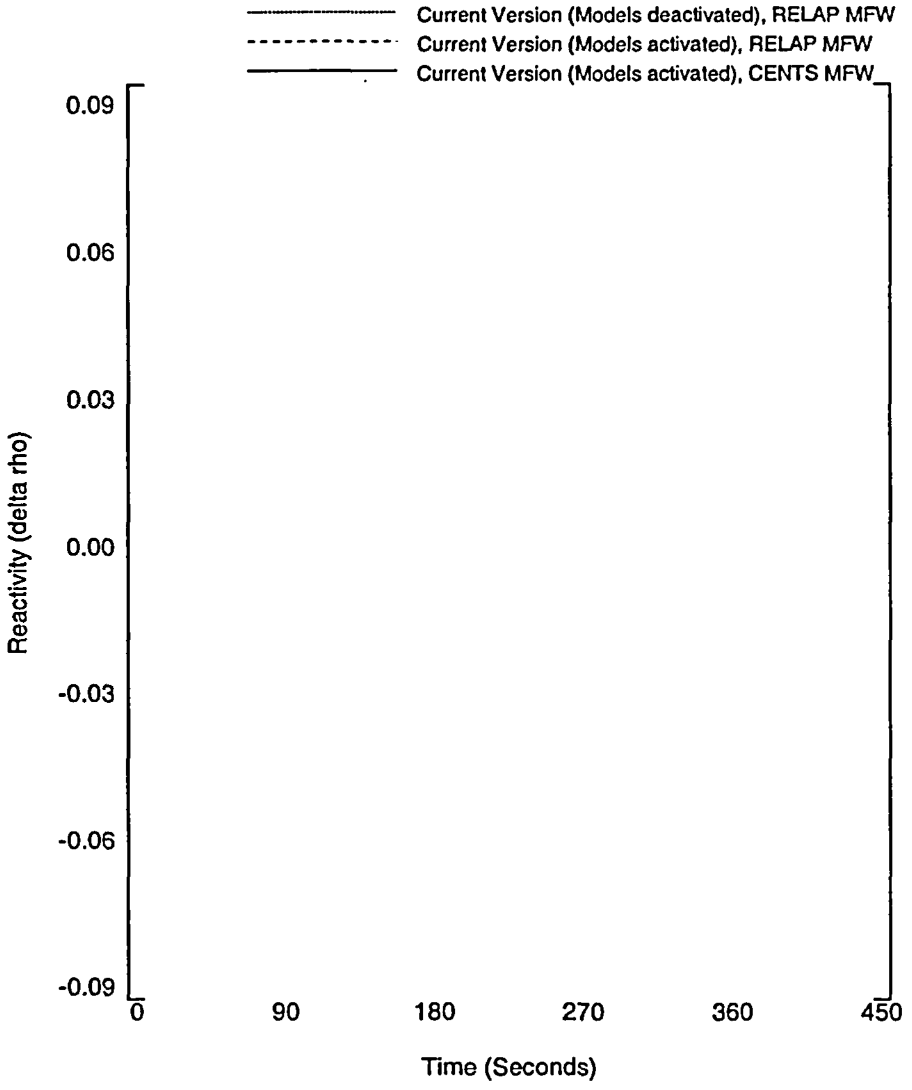


Figure 3.4.1.R

HERMITE Credit Reactivity

Steam Line Break Event for Plant E

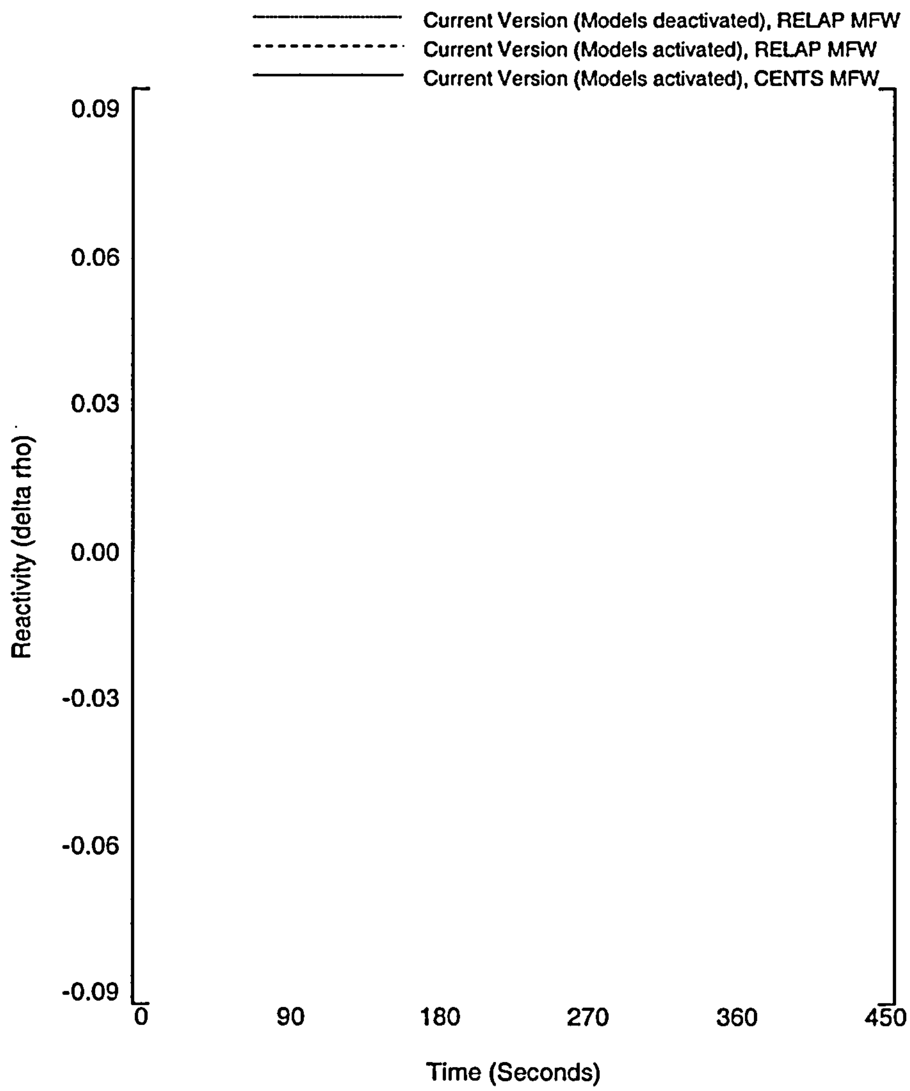


Figure 3.4.1.S

Feedwater Flow to Affected Steam Generator

Steam Line Break Event for Plant E

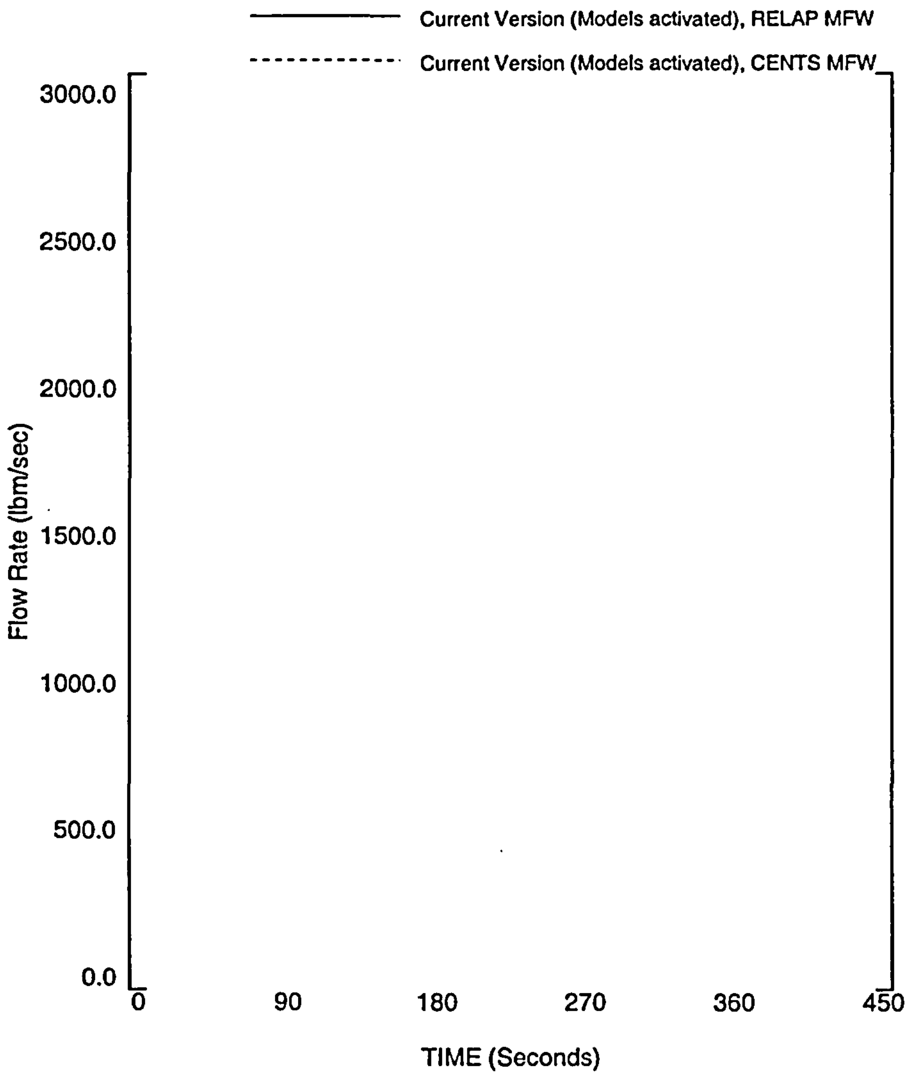


Figure 3.4.2.A

Reactor Core Power

Feedwater Line Break for Plant E

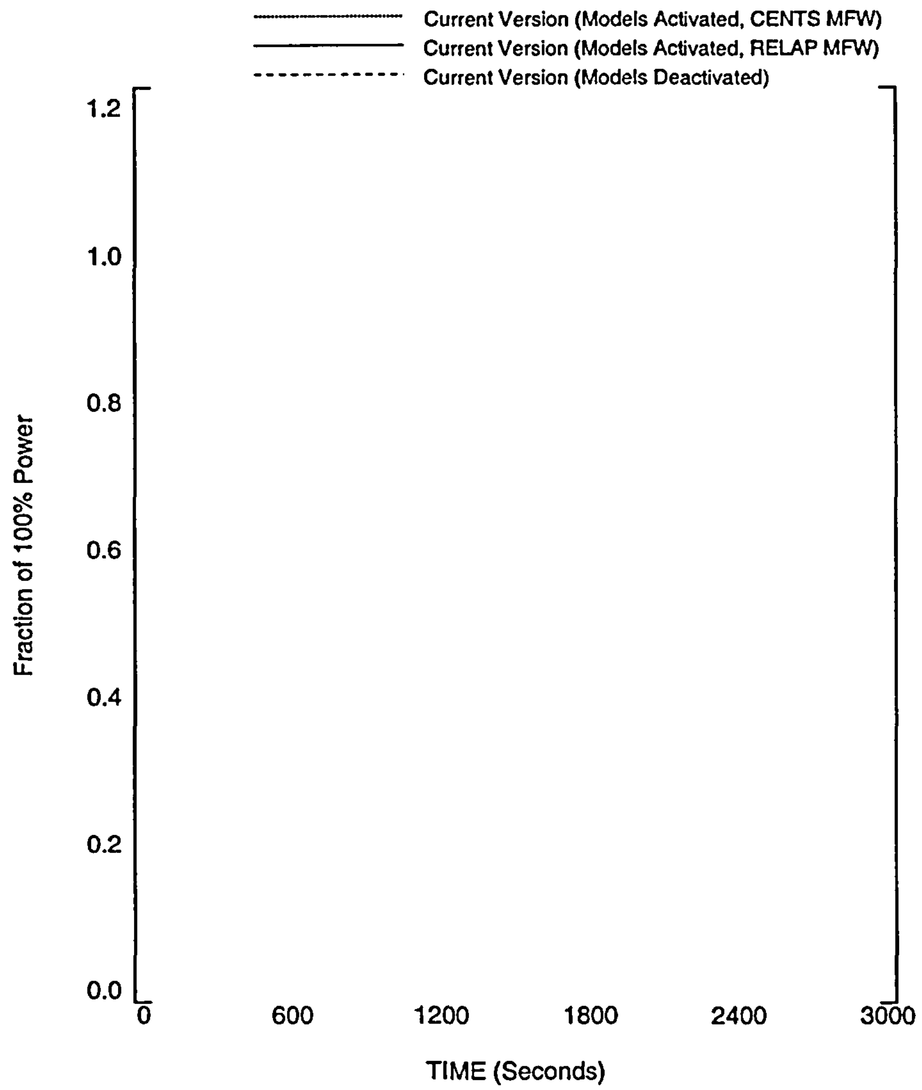


*Figure 3.4.2.B*

Reactor Core Heat Flux

Feedwater Line Break for Plant E

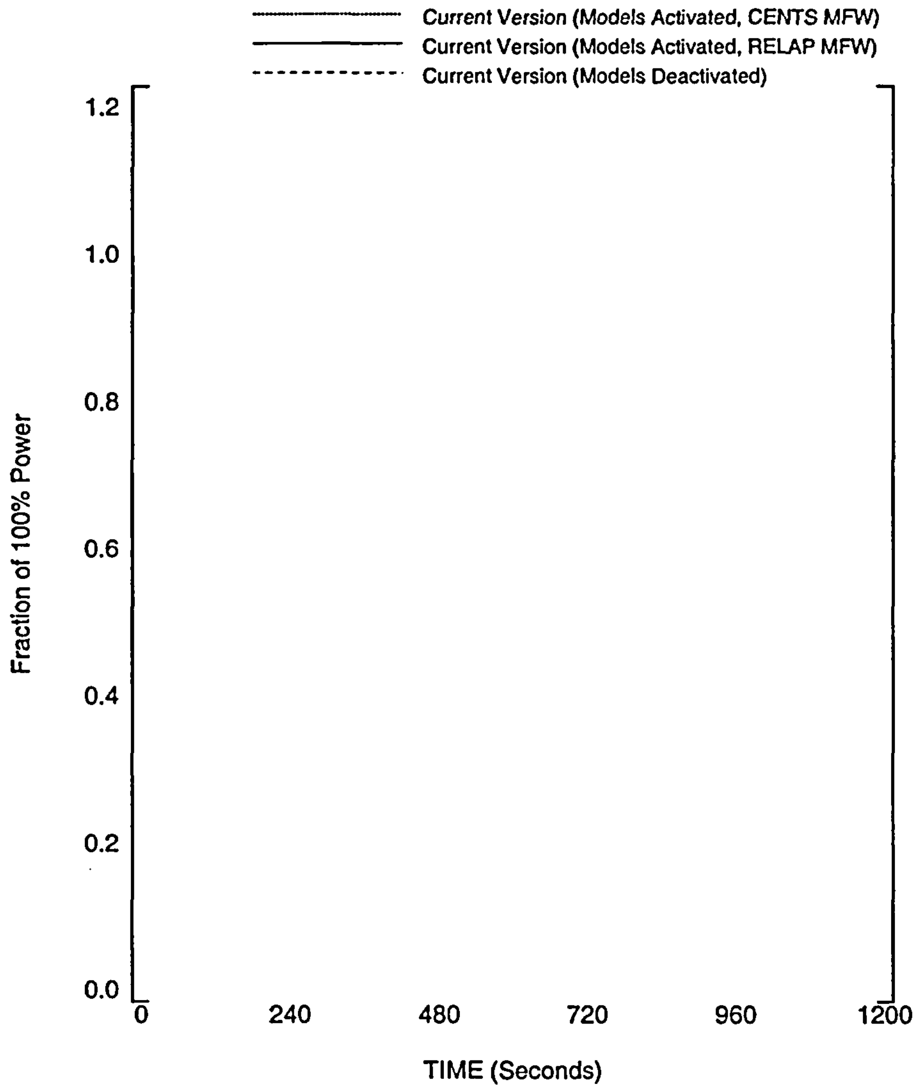


Figure 3.4.2.C

Reactor Coolant System Pressure

Feedwater Line Break for Plant E

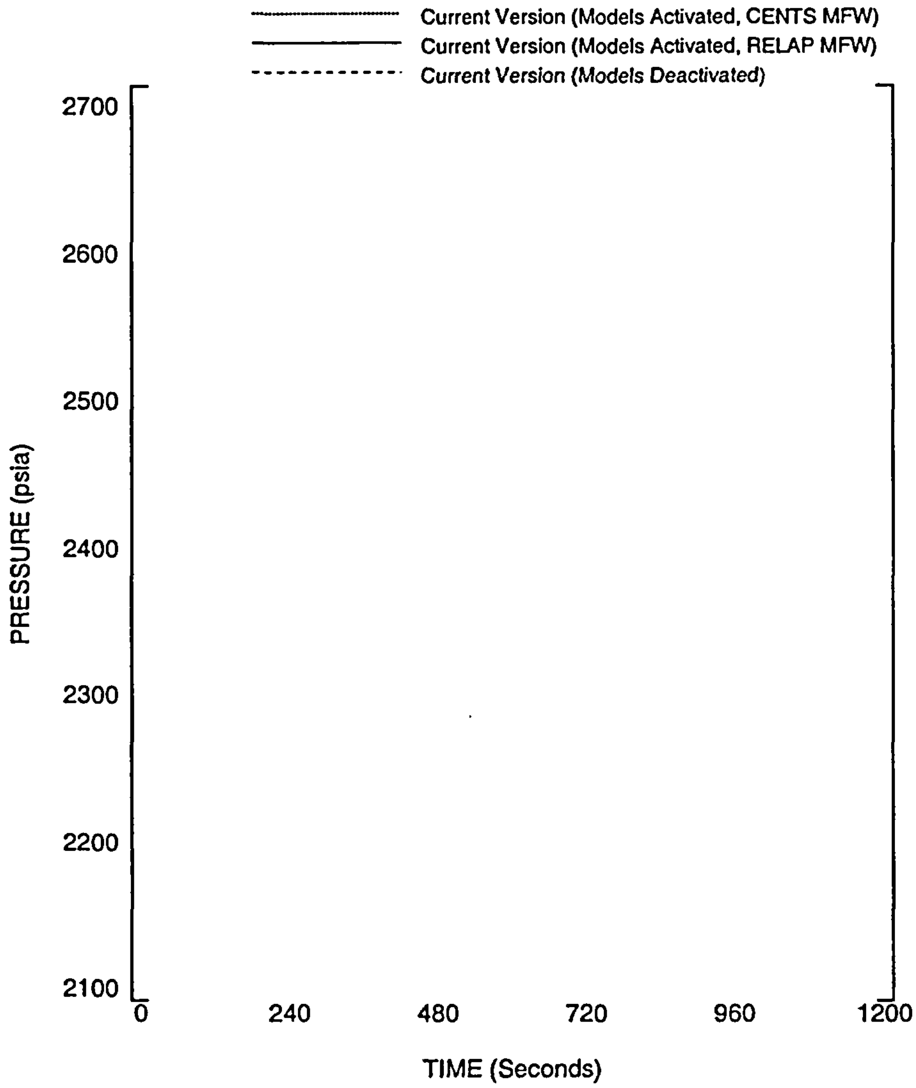


Figure 3.4.2.D

Pressurizer Pressure

Feedwater Line Break for Plant E

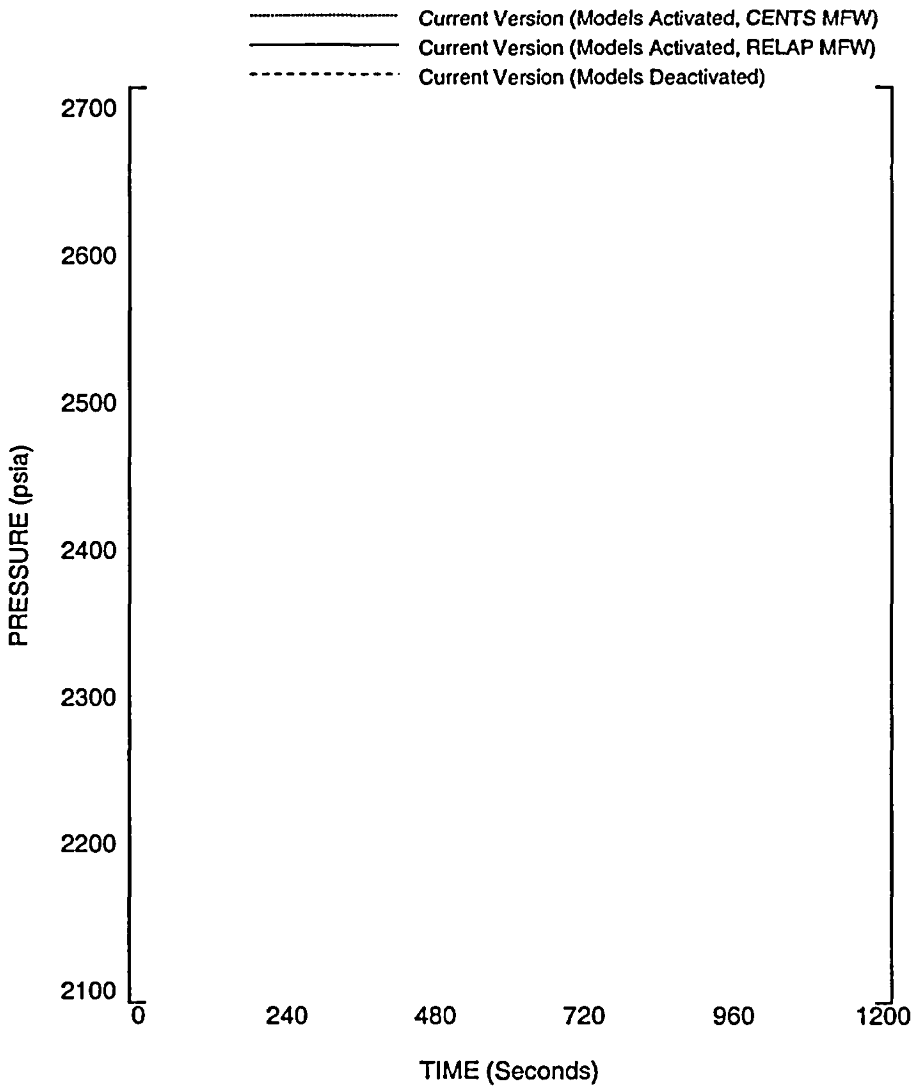


Figure 3.4.2.E

Cold Leg Temperature, Affected Loop

Feedwater Line Break for Plant E

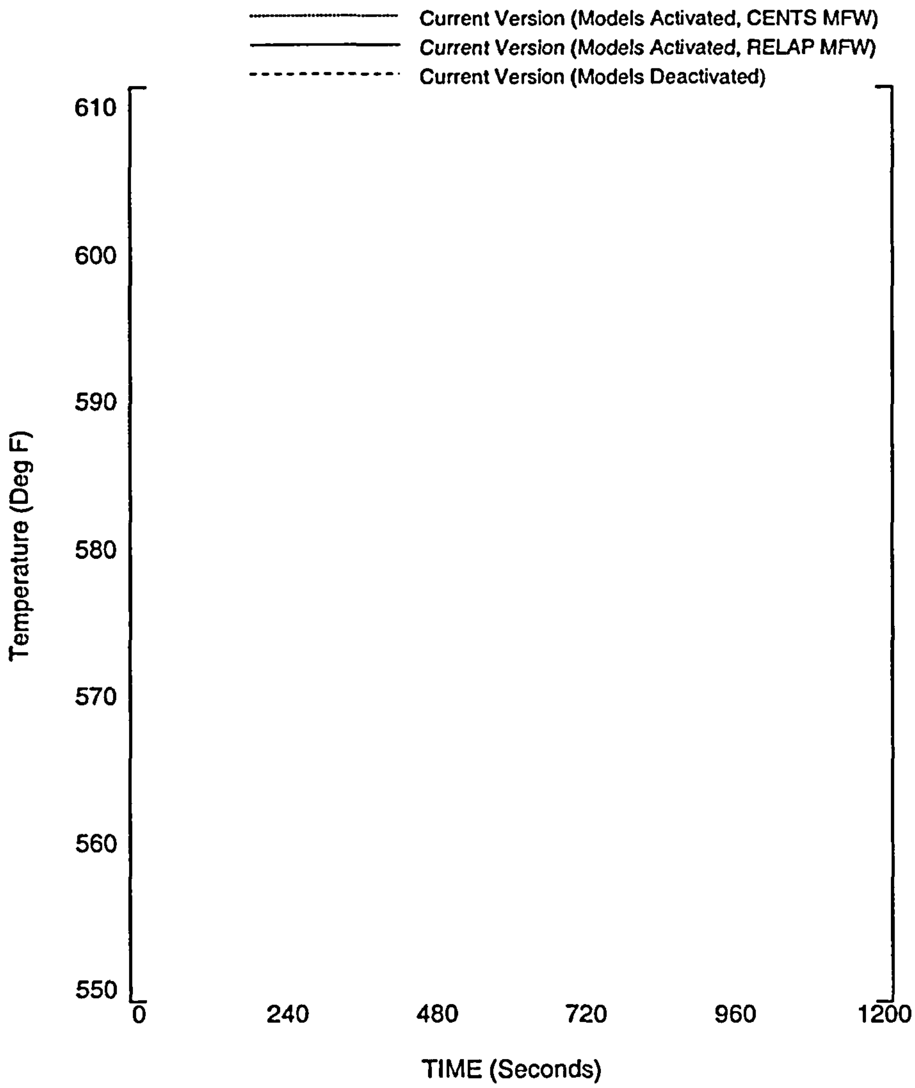


Figure 3.4.2.F

Cold Leg Temperature, Intact Loop

Feedwater Line Break for Plant E

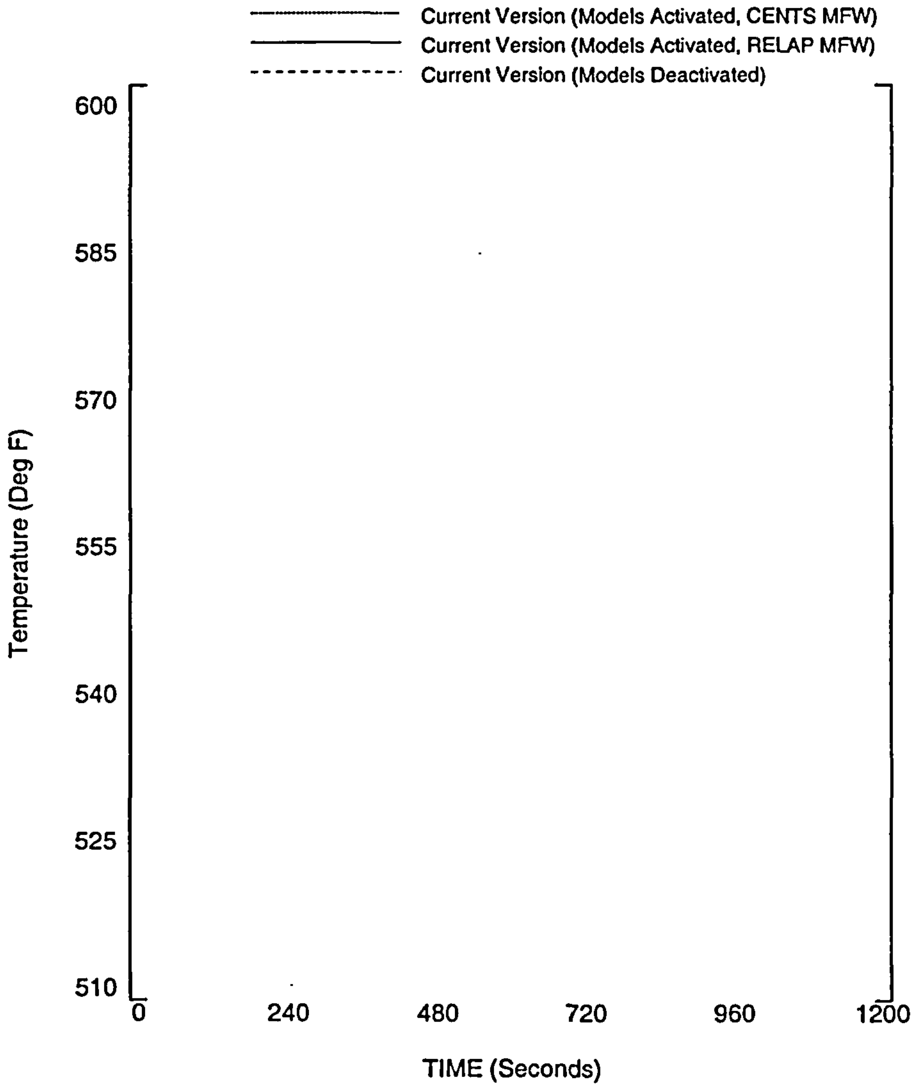


Figure 3.4.2.G

Steam Generator Pressure, Affected Steam Generator

Feedwater Line Break for Plant E

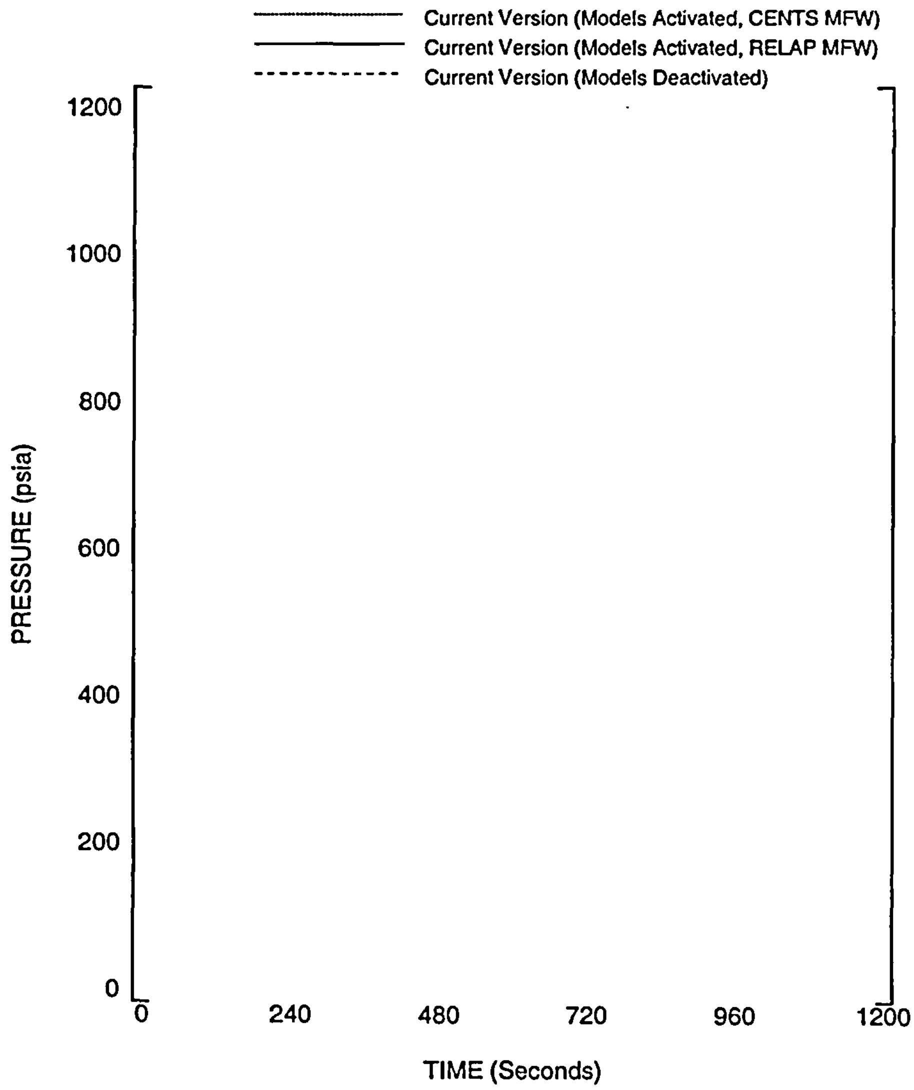


Figure 3.4.2.H

Steam Generator Pressure, Intact Steam Generator

Feedwater Line Break for Plant E

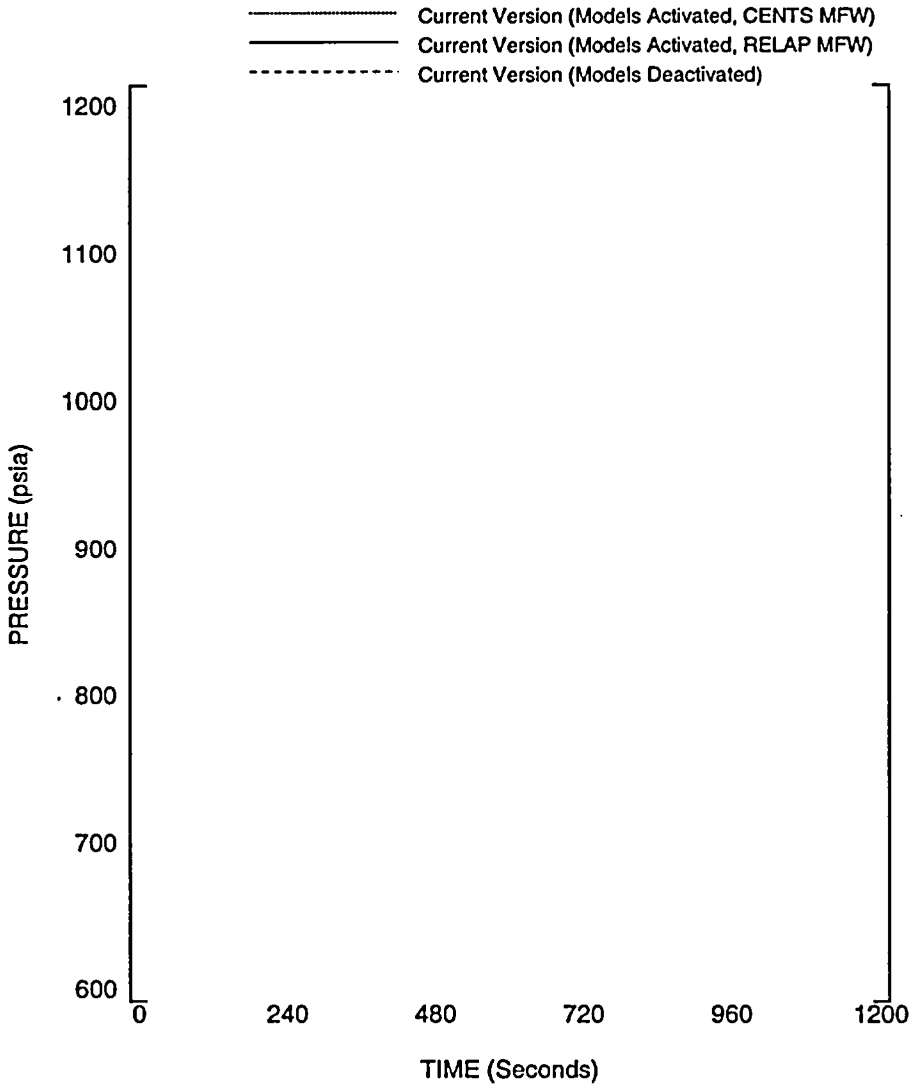


Figure 3.4.2.I

Total Steam Flow, Affected Steam Generator

Feedwater Line Break for Plant E

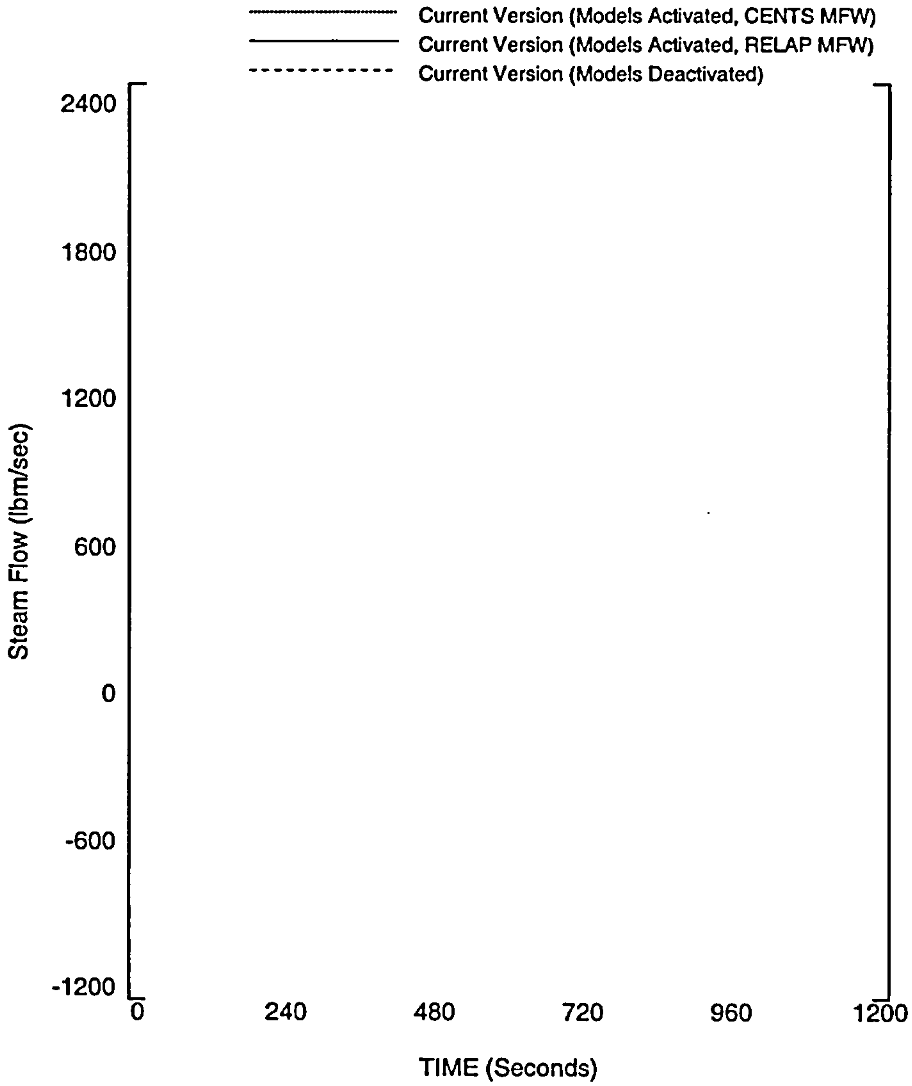


Figure 3.4.2.J

Total Steam Flow, Intact Steam Generator

Feedwater Line Break for Plant E

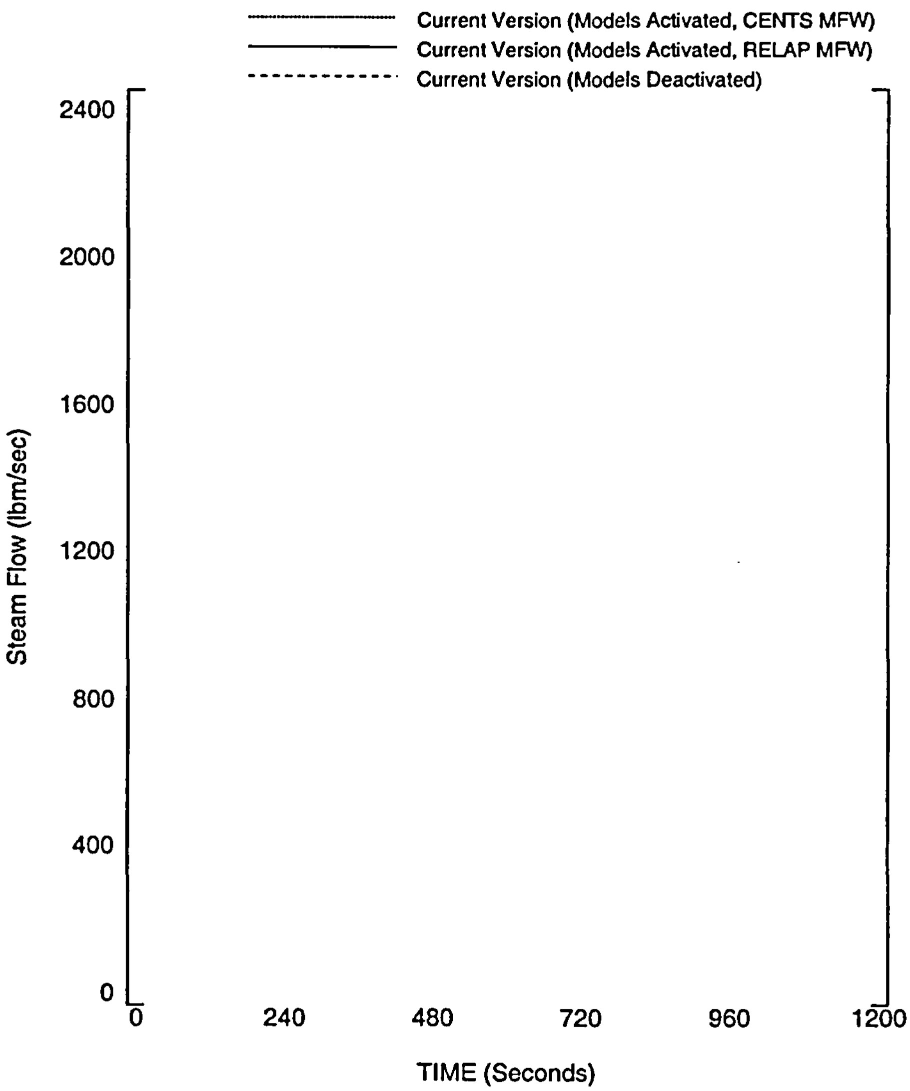


Figure 3.4.2.K

Steam Generator Liquid Mass, Affected Steam Generator

Feedwater Line Break for Plant E

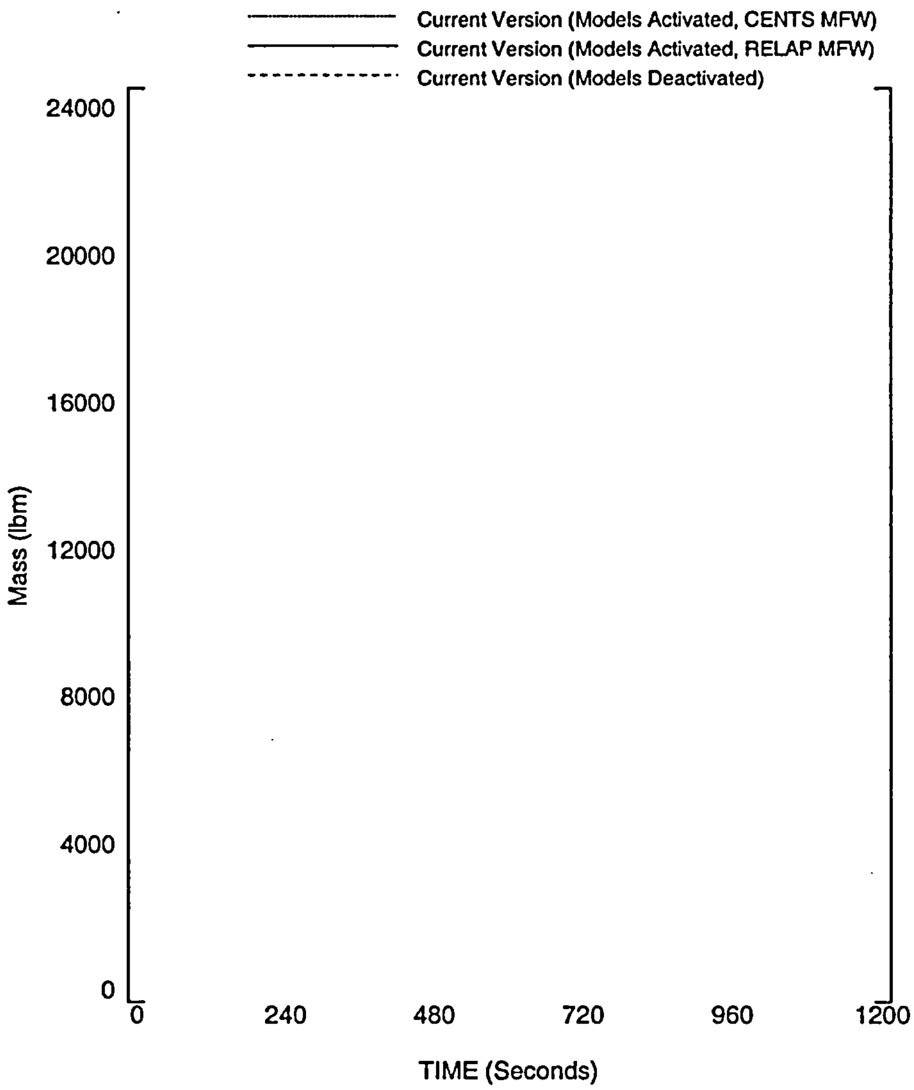


Figure 3.4.2.L

Steam Generator Liquid Mass, Intact Steam Generator

Feedwater Line Break for Plant E

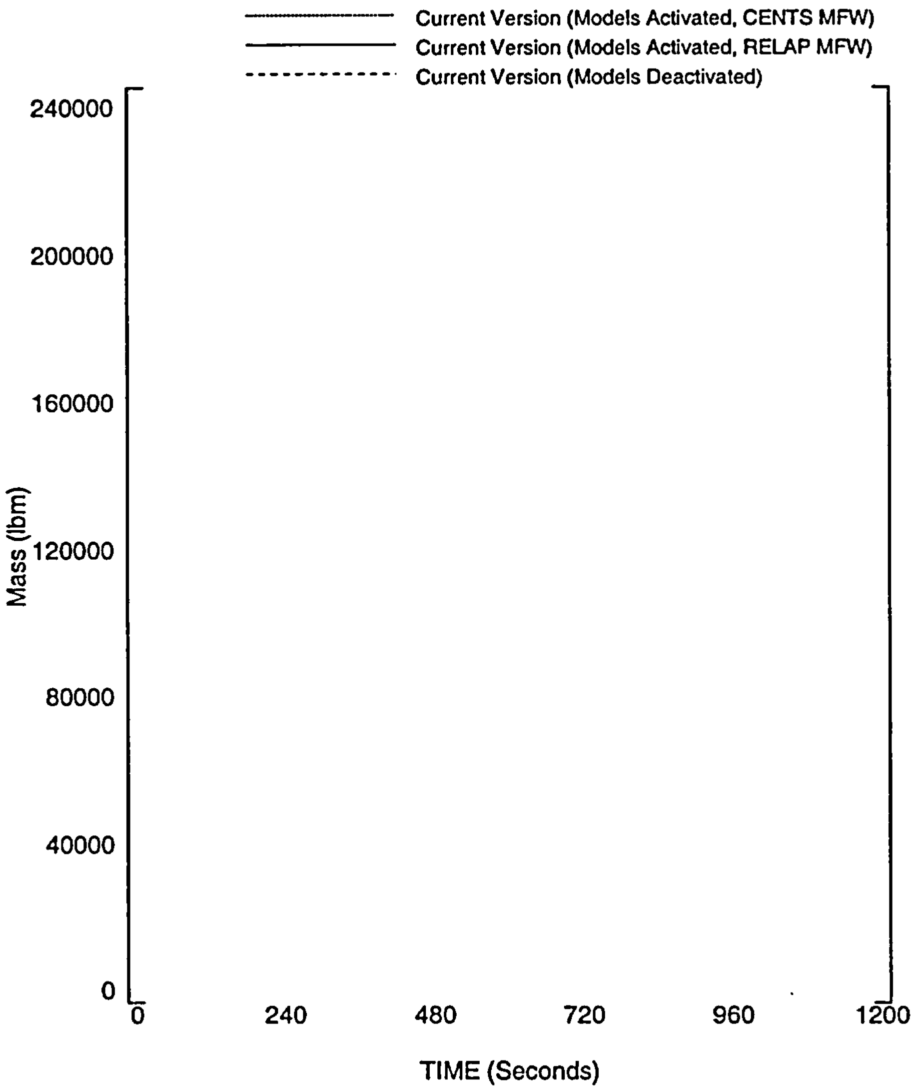


Figure 3.4.2.M

Feedwater Flow to Intact Steam Generator

Feedwater Line Break for Plant E

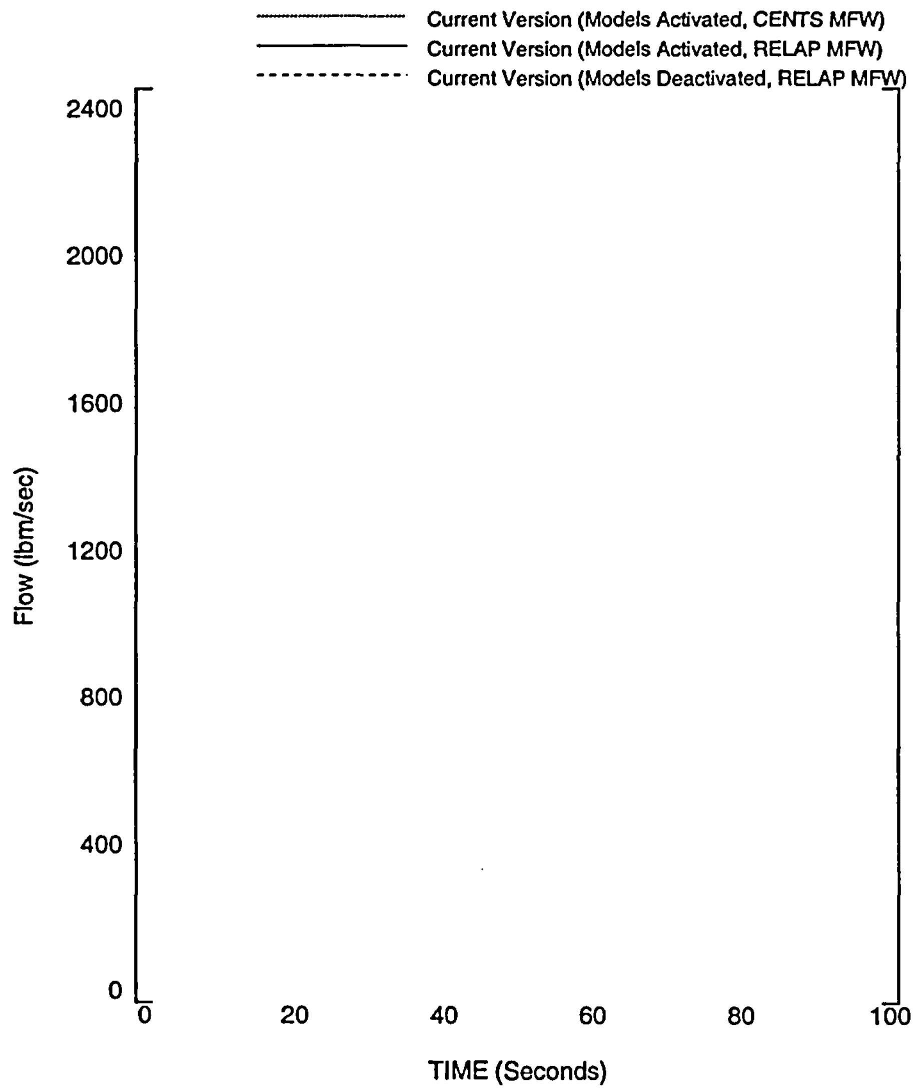


Figure 3.4.2.N

Back Flow to Break from Affected Steam Generator

Feedwater Line Break for Plant E

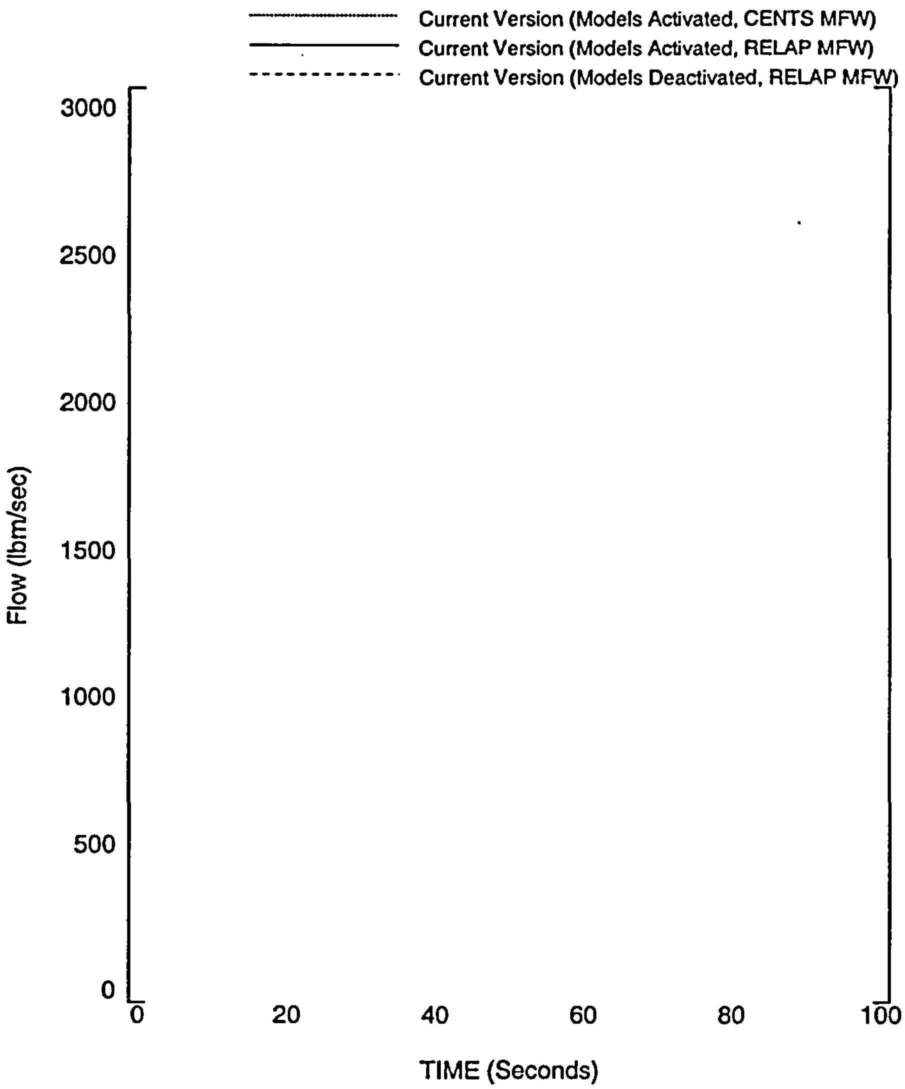


Figure 3.4.2.O

Pressurizer Safety Valve Flow

Feedwater Line Break for Plant E

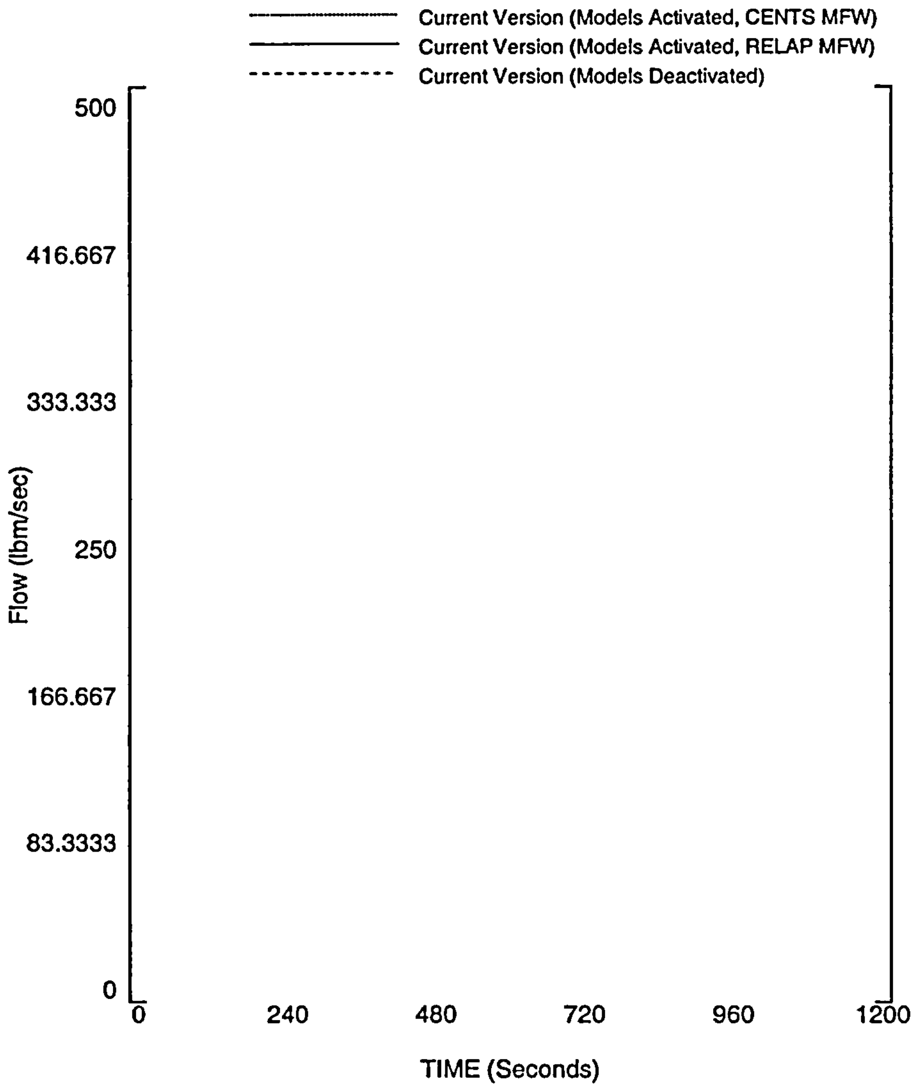


Figure 3.4.2.P

Pressurizer Two-Phase Volume

Feedwater Line Break for Plant E

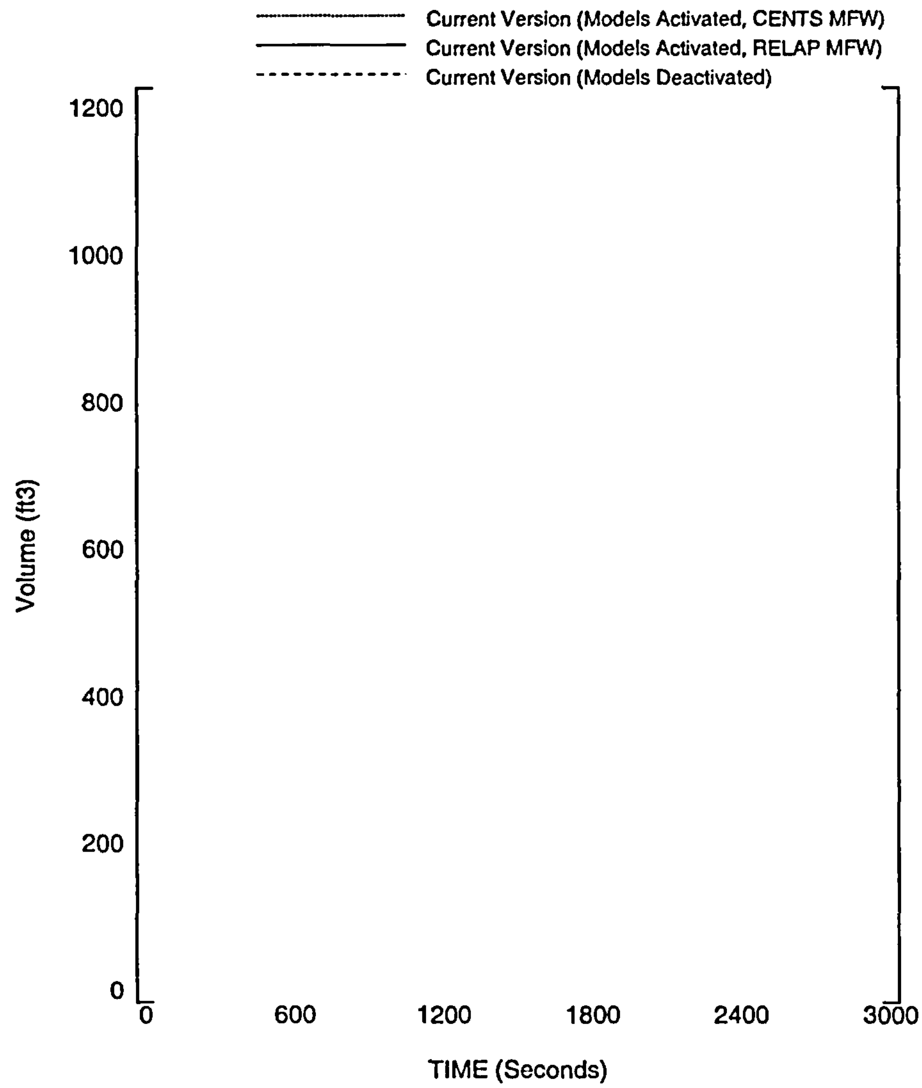


Figure 3.4.2.Q

Feedwater Line Break Flow

Feedwater Line Break for Plant E

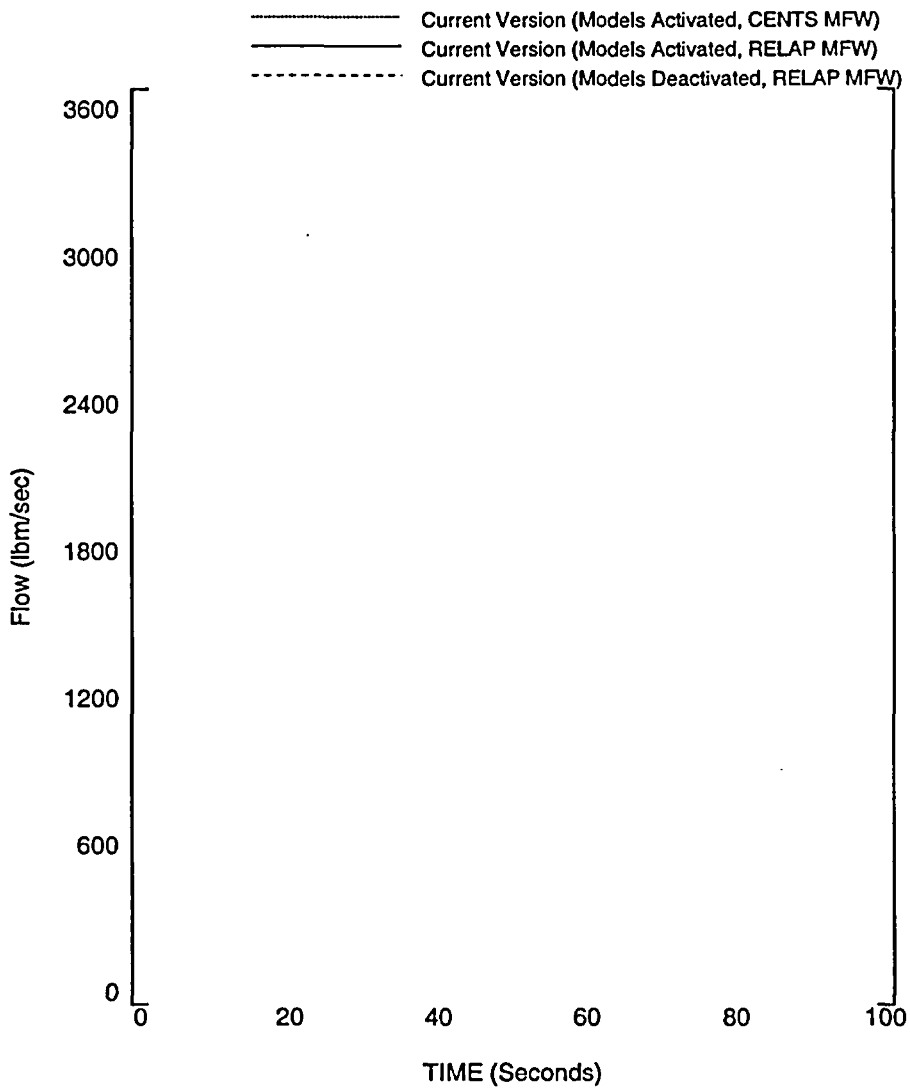


Figure 3.4.2.1.A

Reactor Core Power

Feedwater Line Break for Plant E

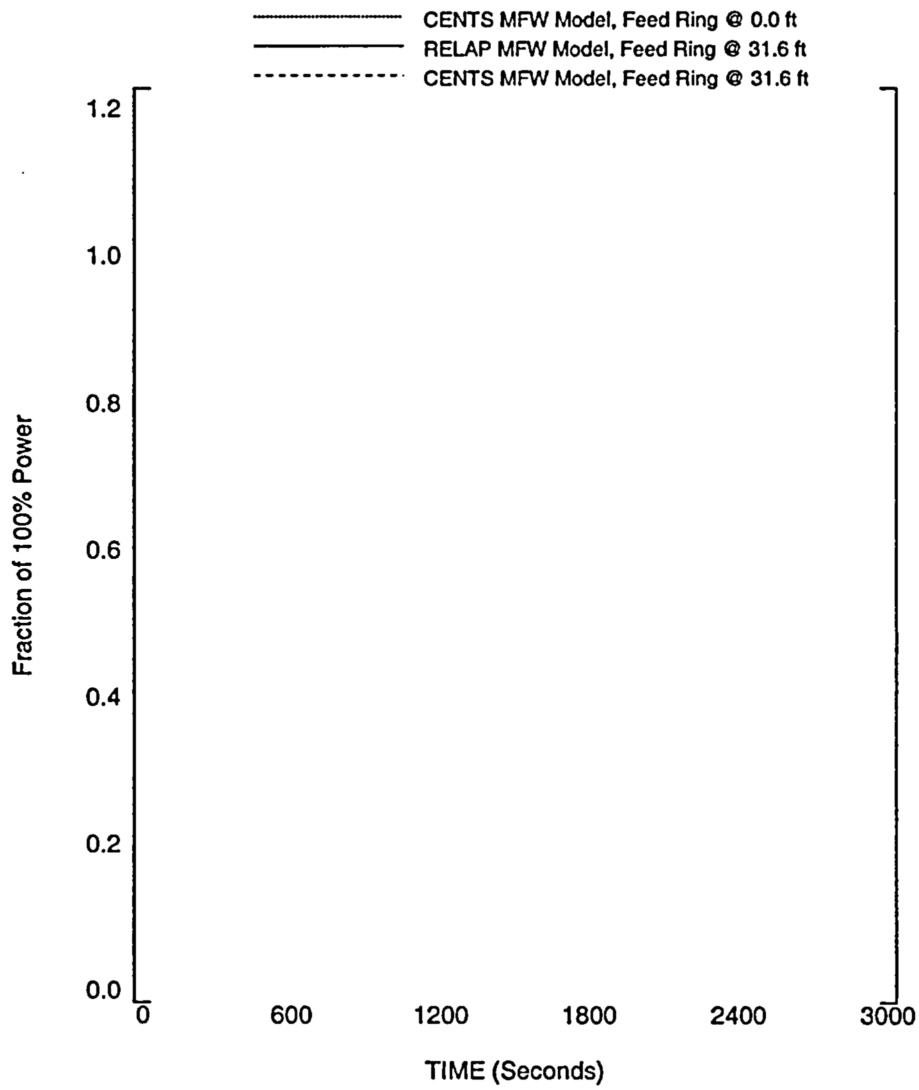


Figure 3.4.2.1.B

Reactor Core Heat Flux

Feedwater Line Break for Plant E

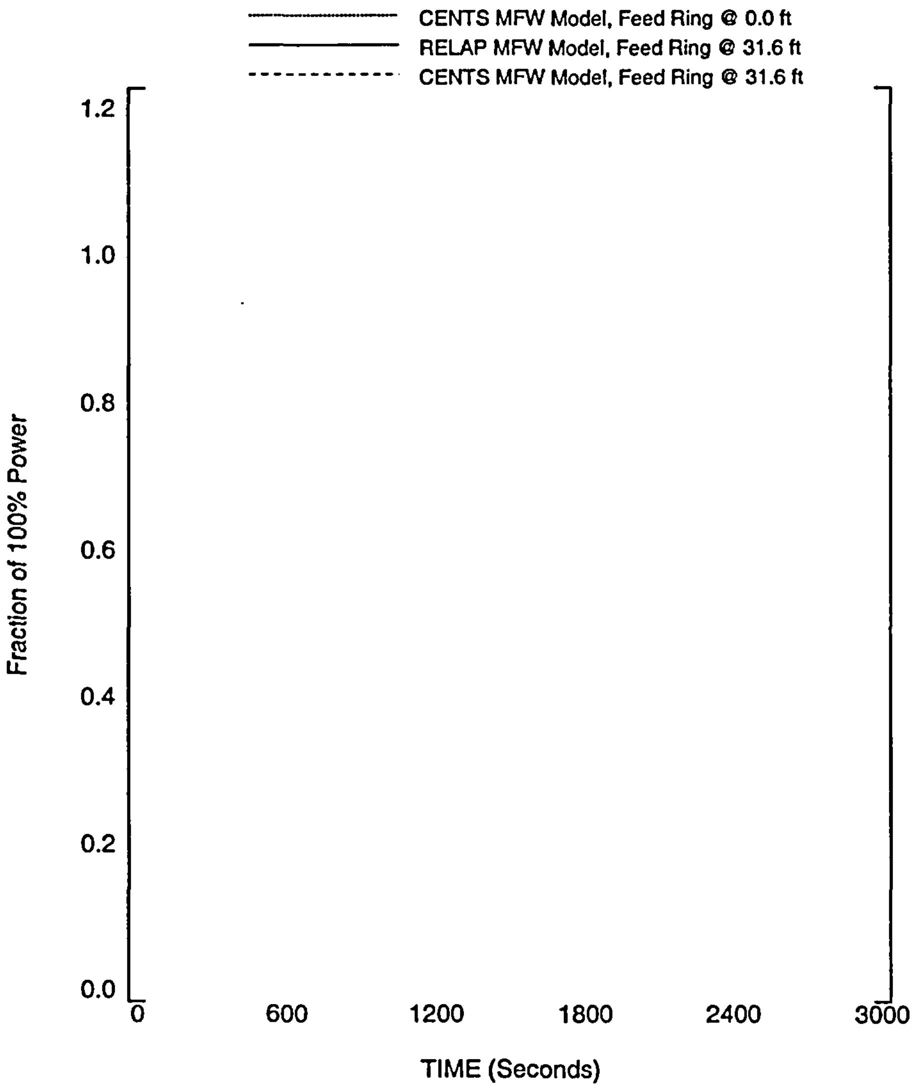


Figure 3.4.2.1.C

Reactor Coolant System Pressure

Feedwater Line Break for Plant E

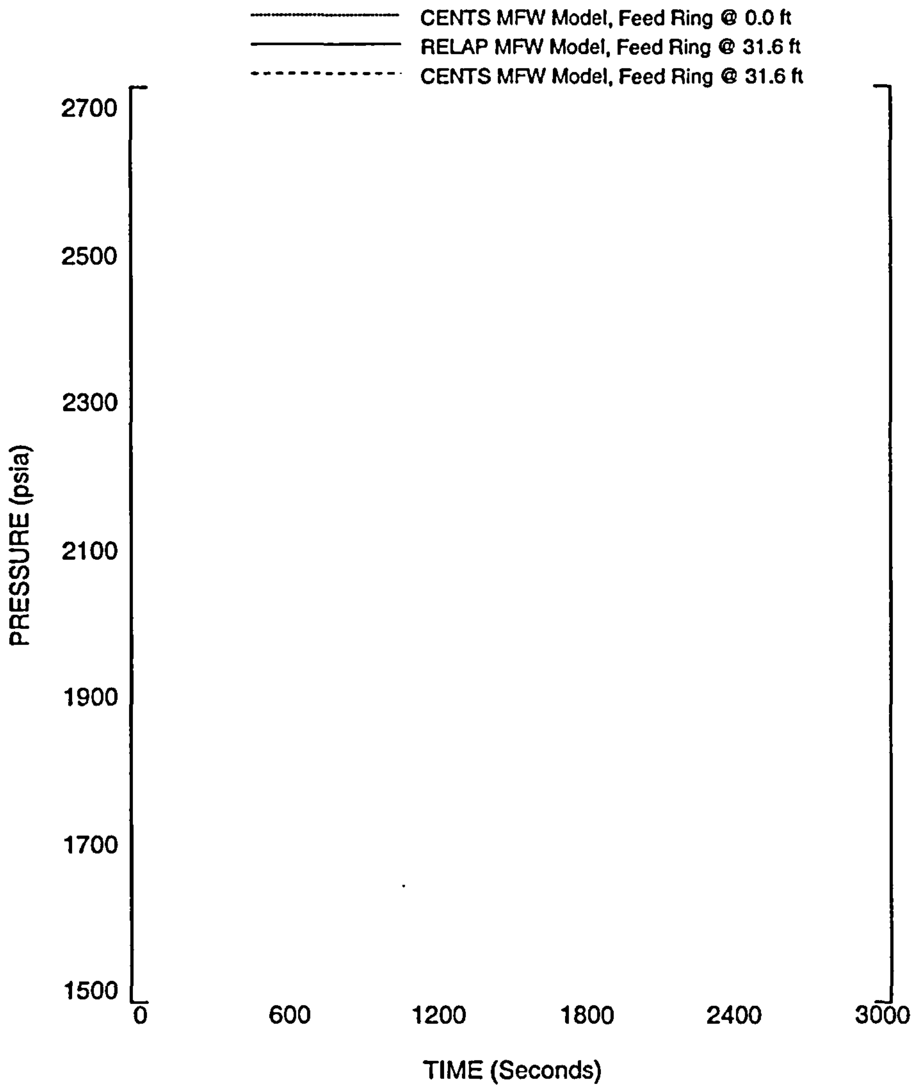


Figure 3.4.2.1.D

Pressurizer Pressure

Feedwater Line Break for Plant E

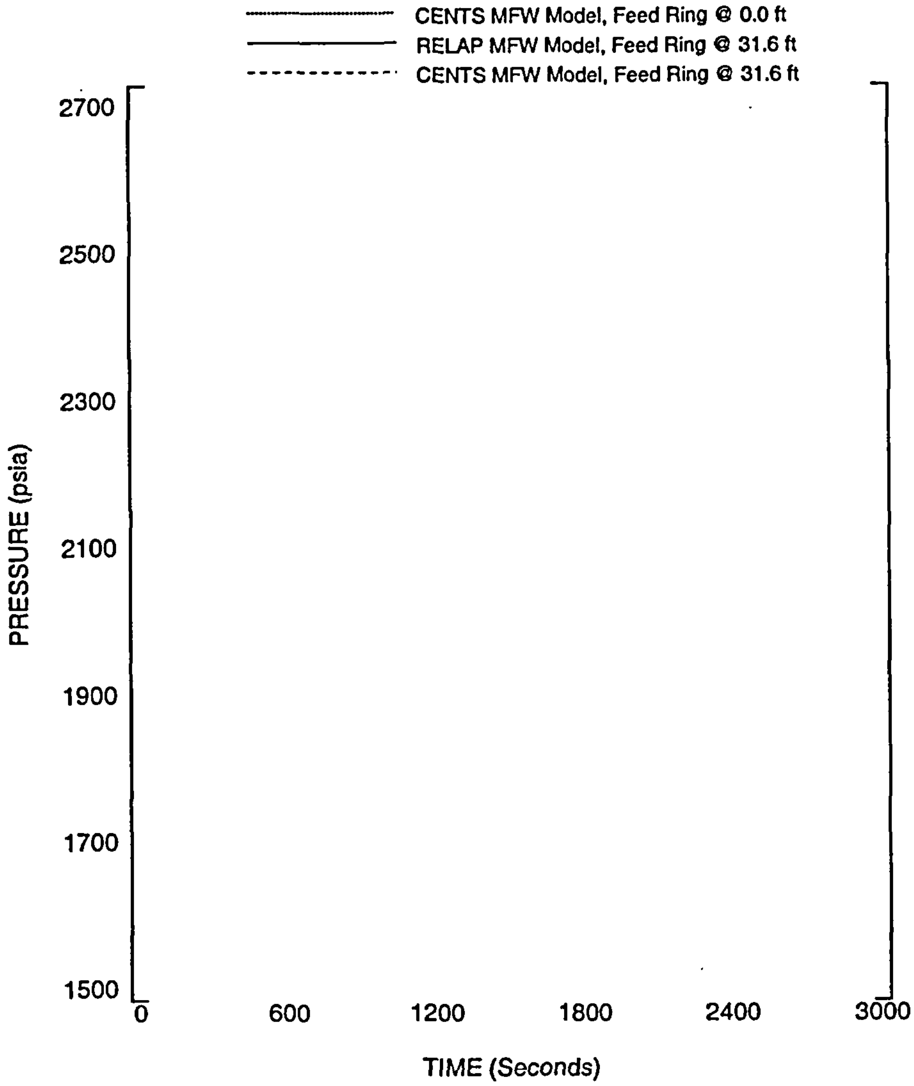


Figure 3.4.2.1.E

Cold Leg Temperature, Affected Loop

Feedwater Line Break for Plant E

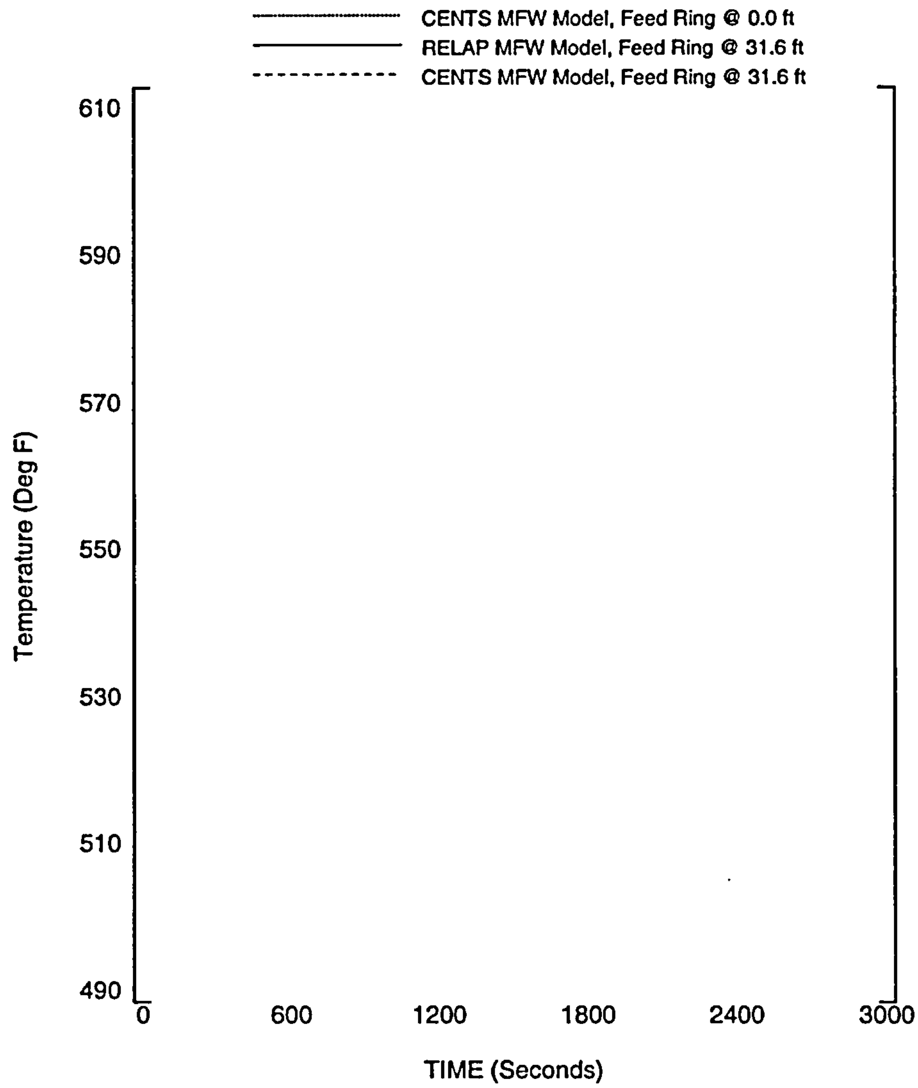


Figure 3.4.2.1.F

Cold Leg Temperature, Intact Loop

Feedwater Line Break for Plant E

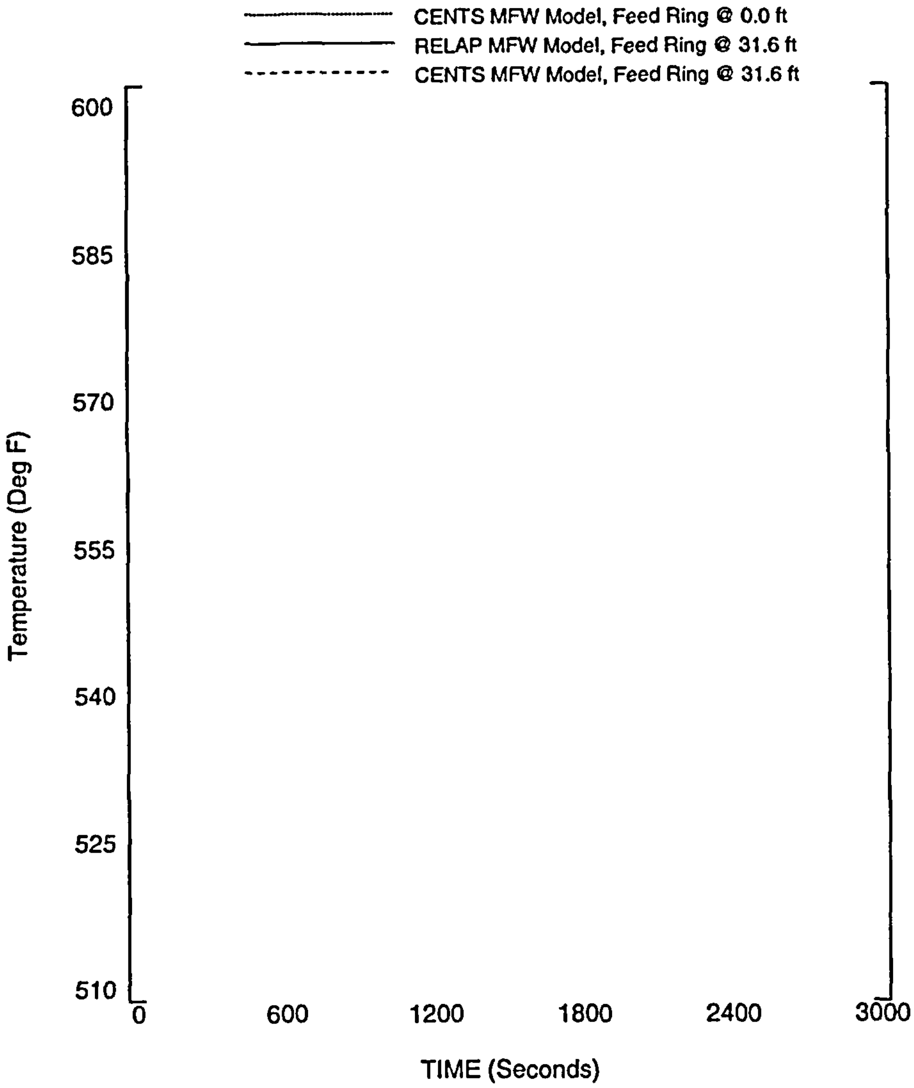


Figure 3.4.2.1.G

Steam Generator Pressure, Affected Steam Generator

Feedwater Line Break for Plant E

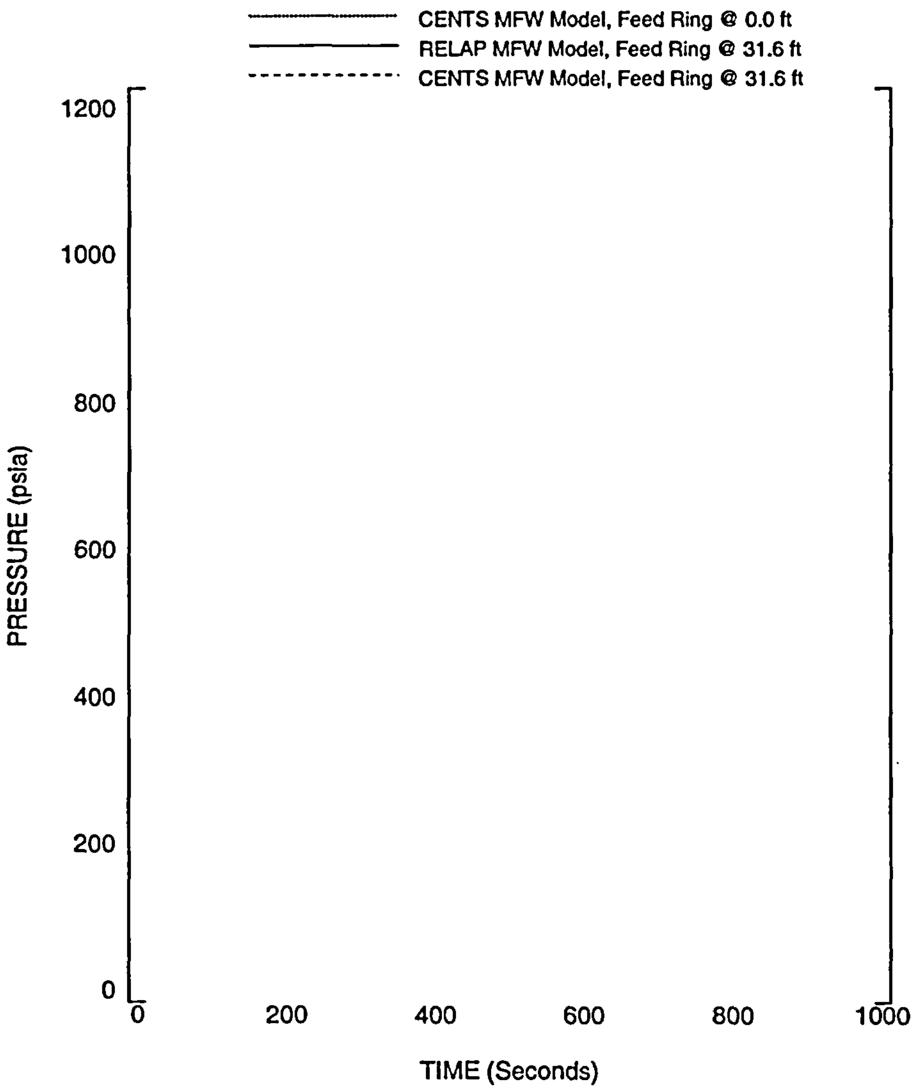


Figure 3.4.2.1.H

Steam Generator Pressure, Intact Steam Generator

Feedwater Line Break for Plant E

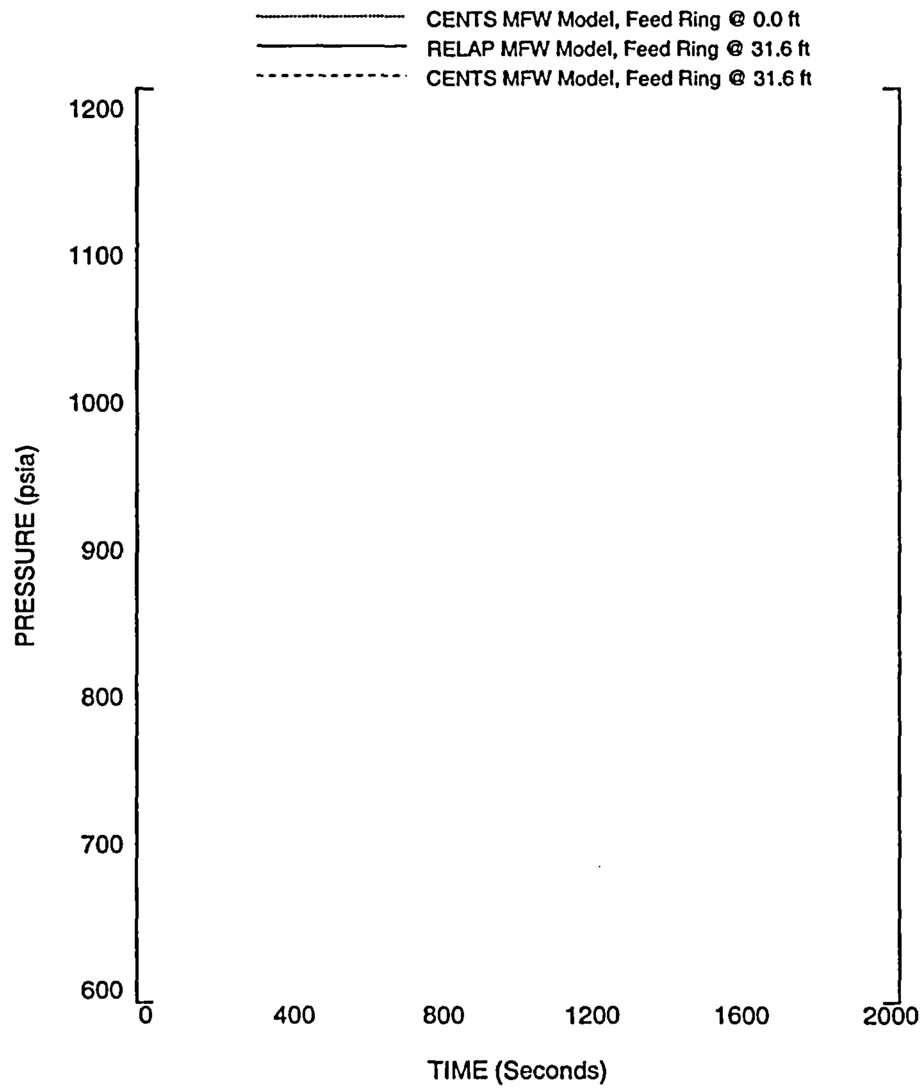


Figure 3.4.2.1.l

Total Steam Flow, Affected Steam Generator

Feedwater Line Break for Plant E

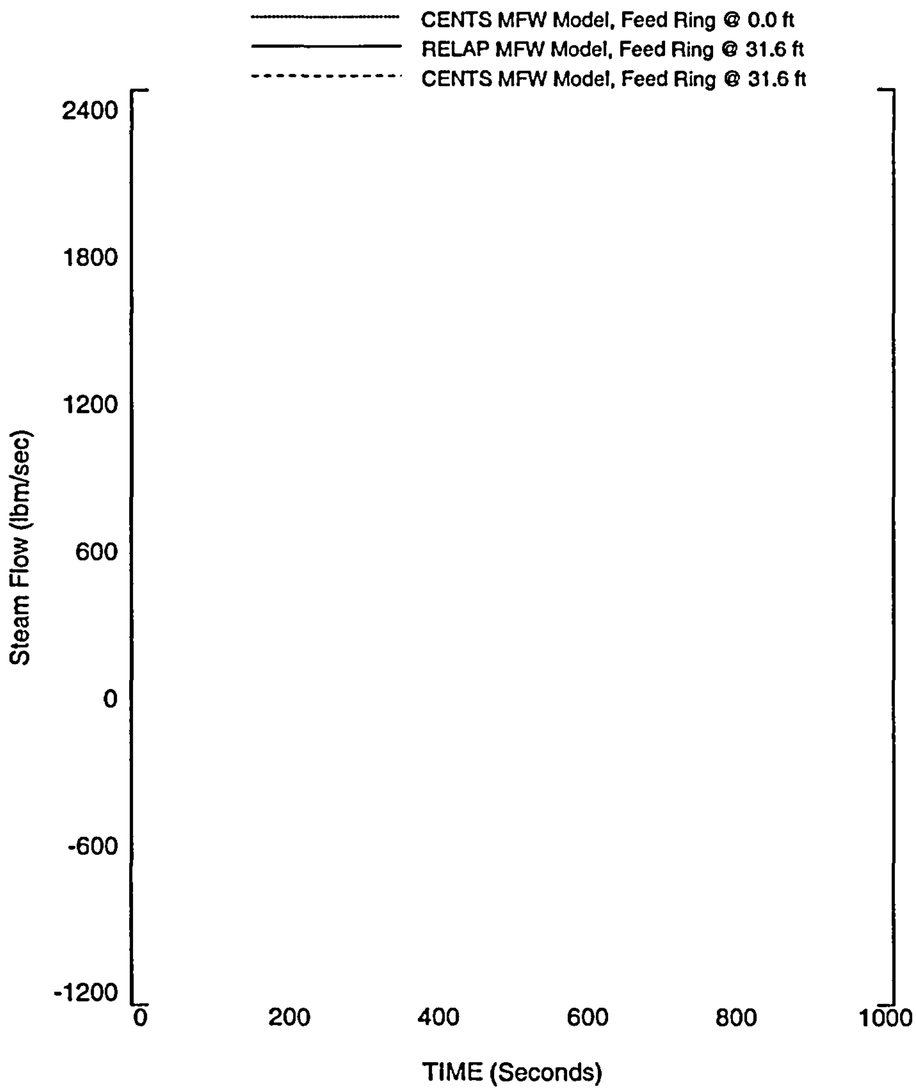


Figure 3.4.2.1.J

Total Steam Flow, Intact Steam Generator

Feedwater Line Break for Plant E

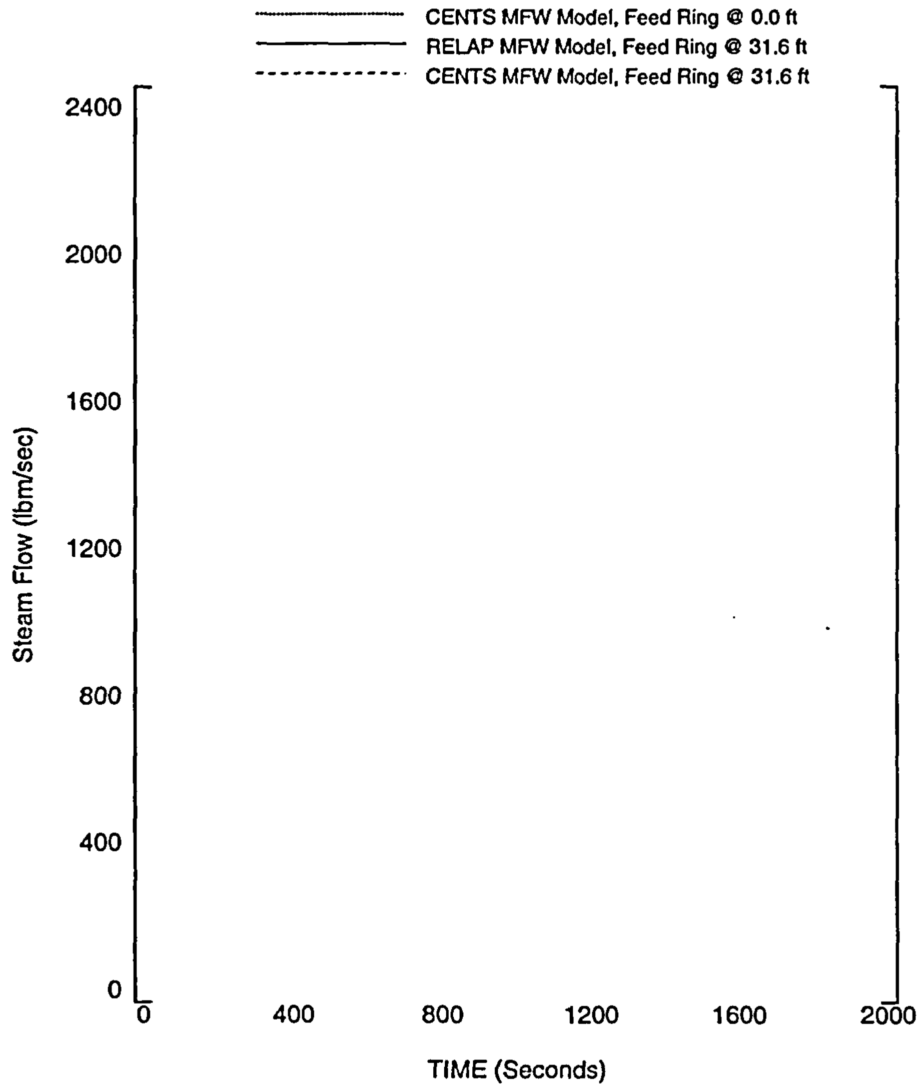


Figure 3.4.2.1.K

Steam Generator Liquid Mass, Affected Steam Generator

Feedwater Line Break for Plant E

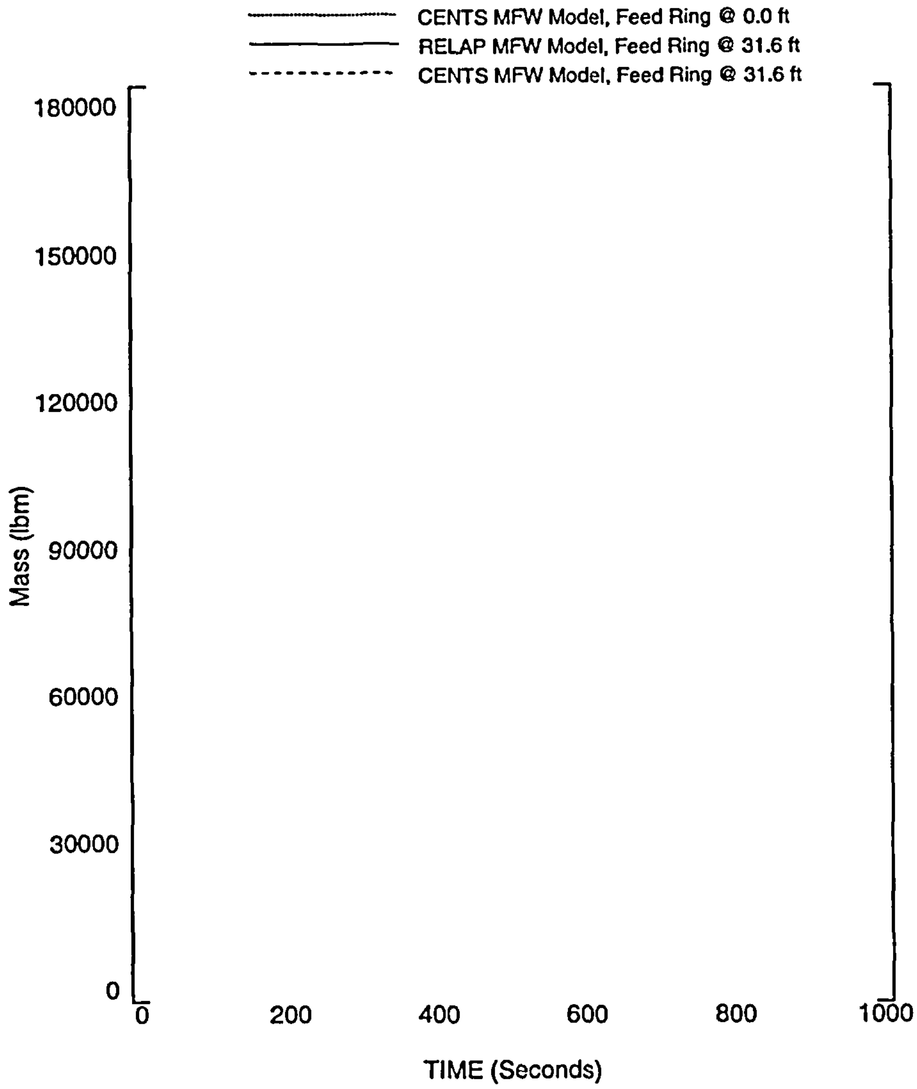


Figure 3.4.2.1.L

Steam Generator Liquid Mass, Intact Steam Generator

Feedwater Line Break for Plant E

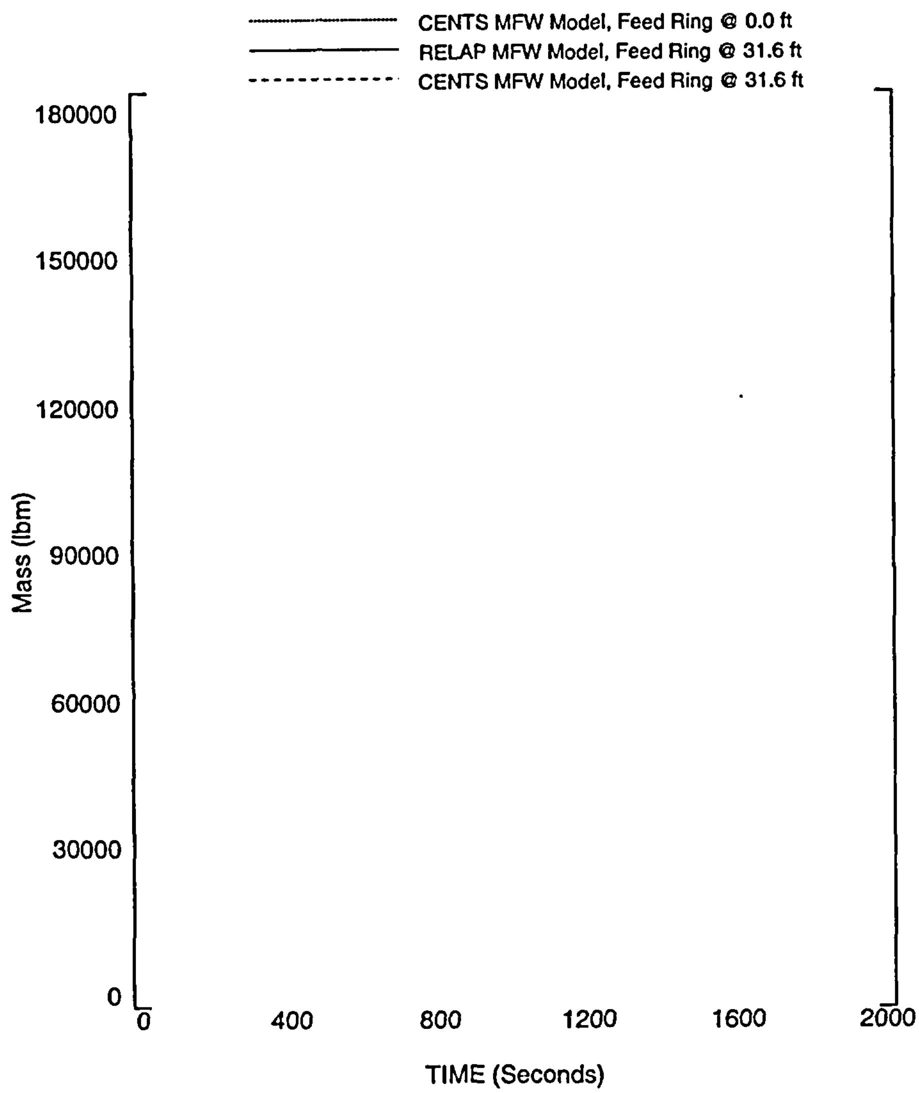


Figure 3.4.2.1.M

Feedwater Flow to Intact Steam Generator

Feedwater Line Break for Plant E

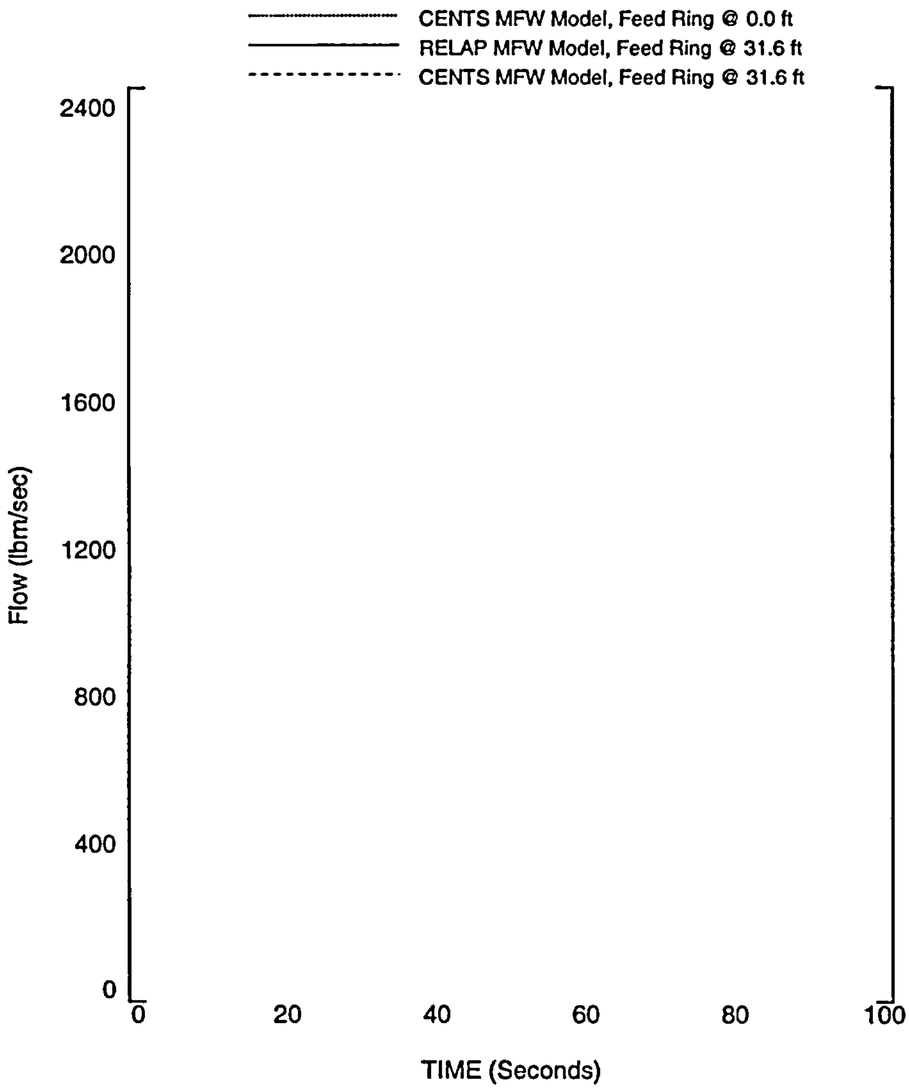


Figure 3.4.2.1.N

Back Flow to Break from Affected Steam Generator

Feedwater Line Break for Plant E

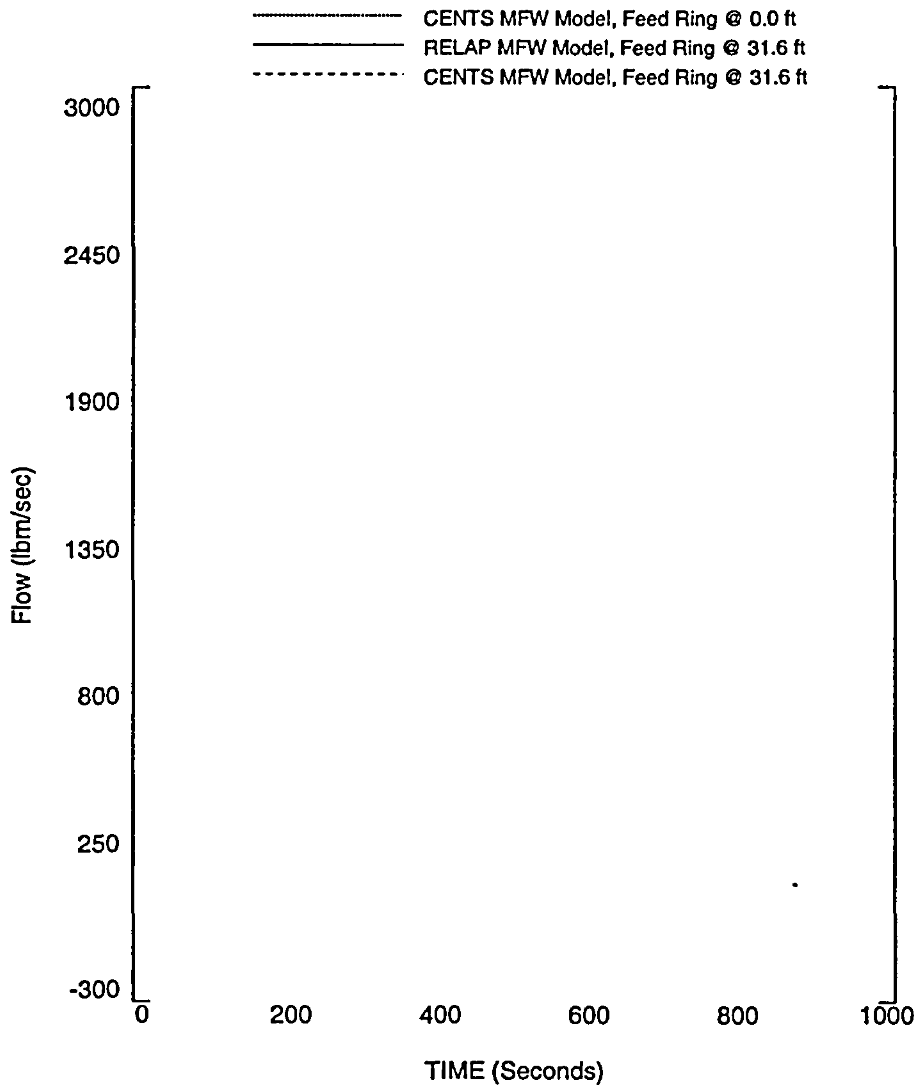


Figure 3.4.2.1.O

Pressurizer Safety Valve Flow

Feedwater Line Break for Plant E

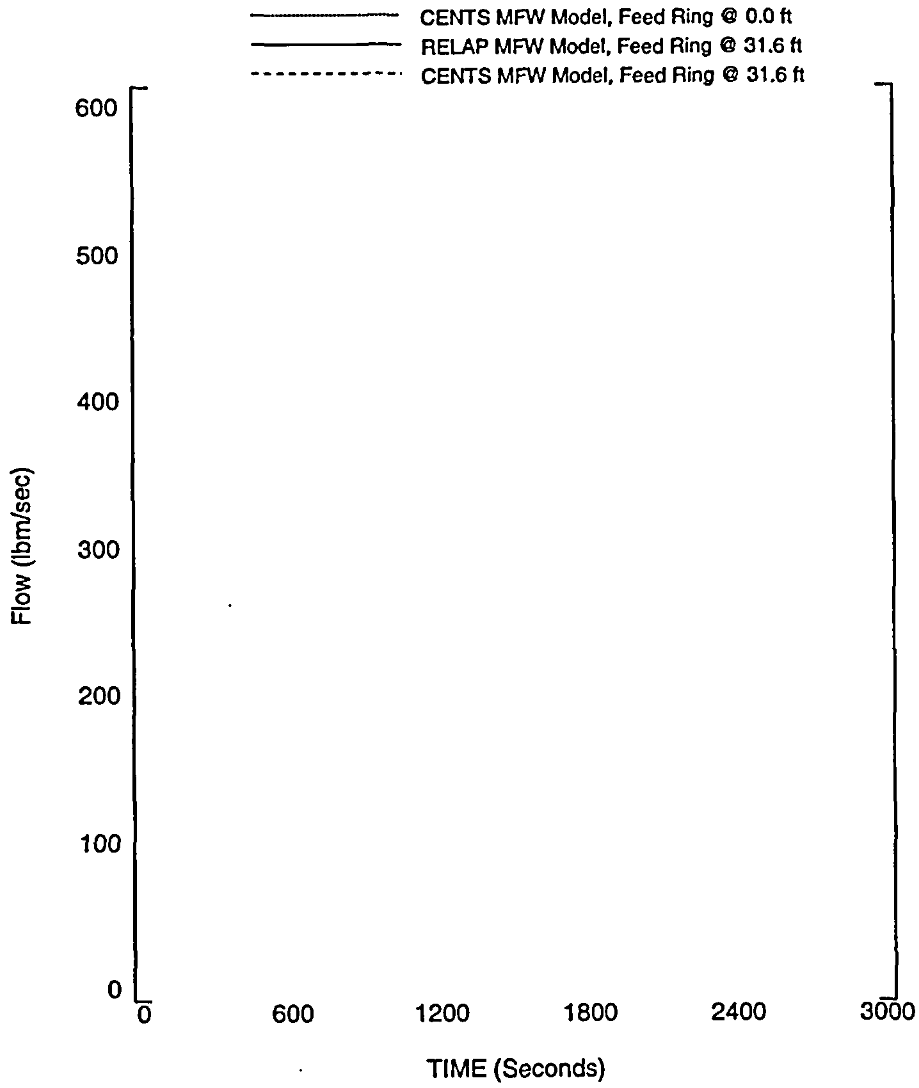


Figure 3.4.2.1.P

Pressurizer Two-Phase Volume

Feedwater Line Break for Plant E

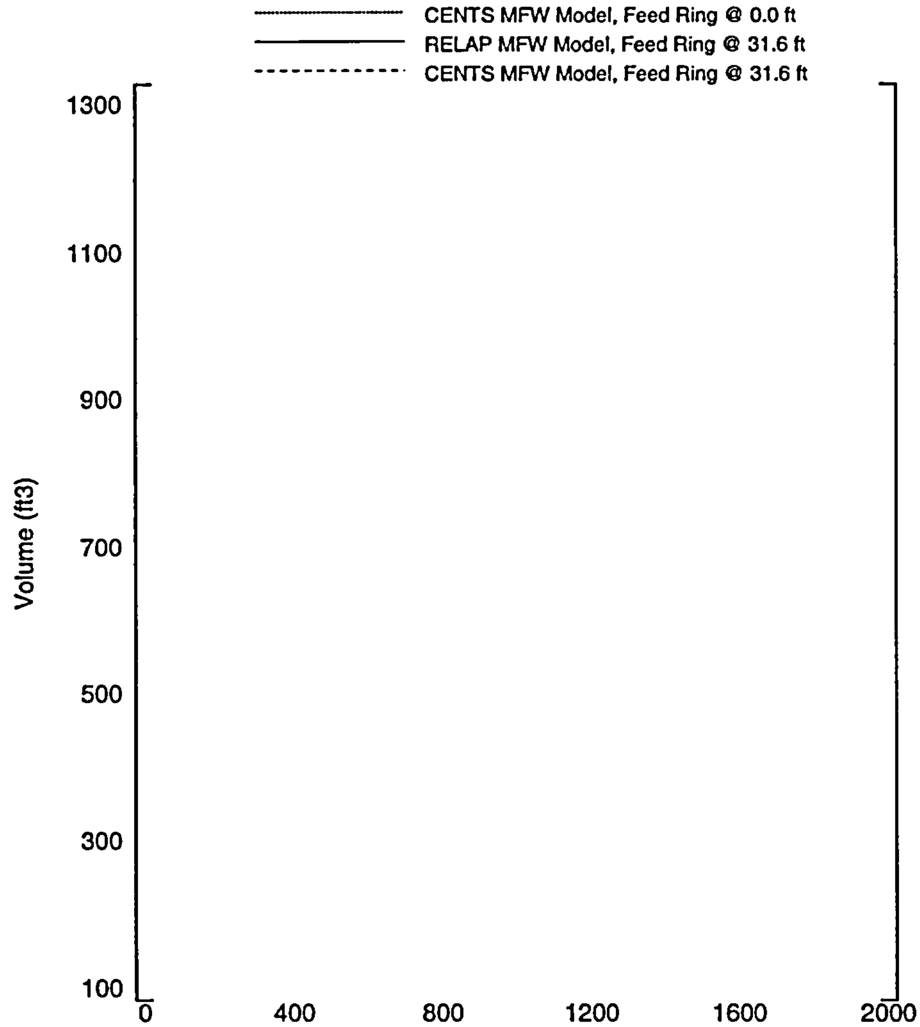
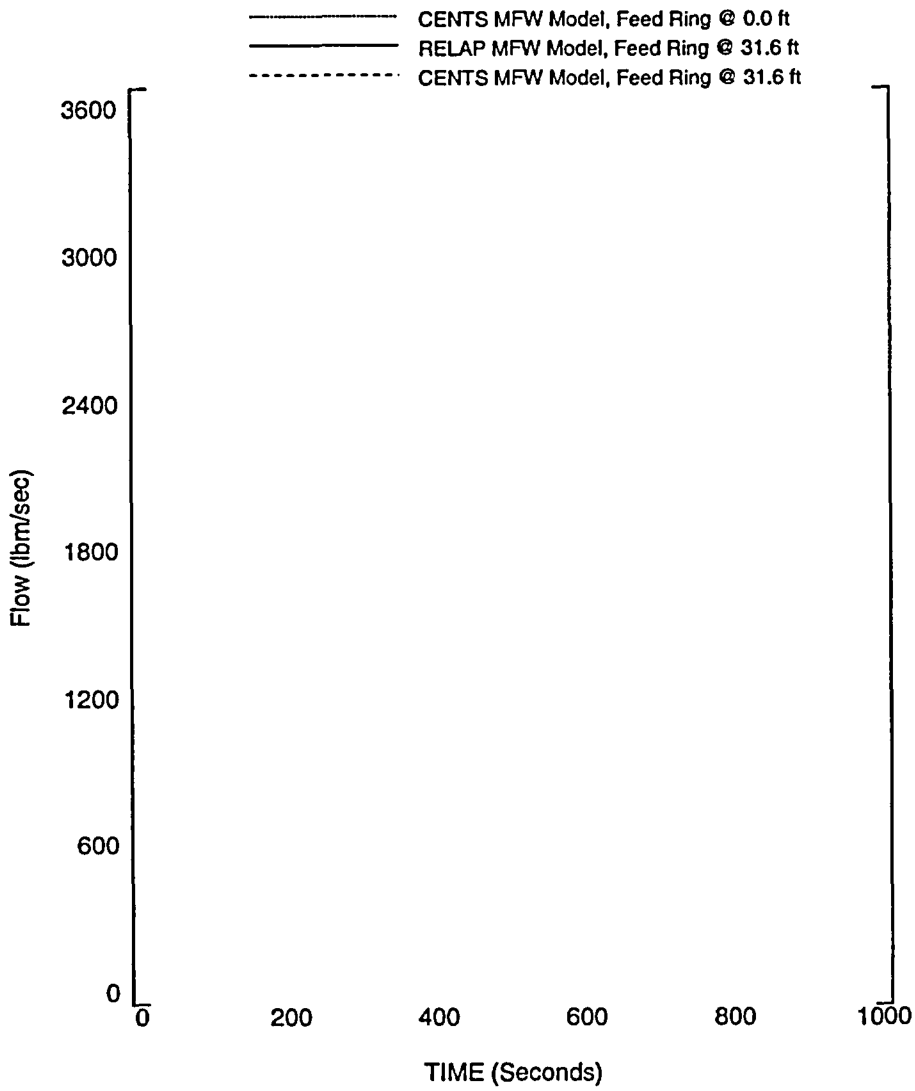


Figure 3.4.2.1.Q

Feedwater Line Break Flow

Feedwater Line Break for Plant E



## 4.0 Conclusions

WCAP-15996-P, Volume 4 presents a comprehensive set of benchmark cases for the CENTS computer code. The cases demonstrate that the CENTS upgraded models can accurately predict PWR plant response to upset conditions. The verification effort supports the following conclusions:

1. The upgraded CENTS version has a numerically stable solution methodology with a proper conservation of mass, momentum, and energy.
2. The upgraded CENTS version reproduces measured plant behavior for a range of different events. Deviations from plant behavior are generally within the uncertainty of the measurement.
3. The upgraded CENTS version satisfactorily reproduces the plant behavior as predicted by the original version of CENTS (i.e., CENPD-282-P-A). Differences between the predictions of the upgraded CENTS version (with all models activated) and the original CENTS code can be generally ascribed to the more detailed models used in the upgraded CENTS version.
4. The upgraded CENTS version remains an accurate NSSS simulation code. Appropriate conservatism of licensing analyses of non-LOCA design basis events is introduced primarily through code input.
5. For the FWLB, using methodology which places the feedring at its actual elevation within the steam generator downcomer and using the CENTS simulation of tube heat transfer provides acceptable simulation of limiting peak pressures within the RCS and steam generators. It also provides more accurate simulation of the long term pressurizer level response.

The upgraded CENTS version is shown herein to be capable of predicting NSSS response for PWR non-LOCA design basis events for a range of operating conditions. Thus, the upgraded CENTS version can be effectively used as a predictive tool for licensing analyses of non-LOCA events for Combustion Engineering and Westinghouse PWR designs.

# 5.0 References


1. "System 80 Standard Safety Analysis Report (CESSAR)", through Amendment 11, Combustion Engineering, Inc., 1985.
2. "Safety Evaluation Report for CESSAR System 80", NUREG 0852 through Supplement 3, STN 50-470.
3. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.
4. "RELAP5/MOD3 Code Manual", NUREG/CR5535, INEL-95/0174, Vol.1.