Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

December 2, 2003

L-2003-276
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: St. Lucie Unit 2
Docket No. 50-389
Proposed License Amendment
WCAP-9272 Reload Methodology and
Implementing 30% Steam Generator Tube Plugging Limit

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating License NPF-16 for St. Lucie Unit 2. The purpose of this proposed license amendment is to allow operation of St. Lucie Unit 2 with a reduced reactor coolant system (RCS) flow, corresponding to a steam generator tube plugging level of 30% per steam generator. The re-analysis performed to support this reduction in RCS flow has used Westinghouse WCAP-9272, Westinghouse Reload Safety Evaluation Methodology. The implementation of these changes will require changes to the current Technical Specifications (TS).

The proposed amendment includes the following Technical Specifications changes: Revision to the Thermal Margin Safety Limit Lines TS Figure 2.1-1, reduction in RCS flow in TS Table 3.2-2 and in footnote to TS Table 2.2-1, changes to positive MTC in TS 3.1.1.4, changes to surveillance requirements for Linear Heat Rate TS 3/4.2.1, deletion of Fxy TS 3/4.2.2, relocation to COLR of DNB parameters in TS 3.2-5, changes to Design Features Fuel Assemblies TS 5.3.1, deletion of Design Features RCS Volume TS 5.4.2, COLR methodology list update in TS 6.9.1.11b and conforming changes to TS 1.38, TS 3.2.4, TS 3/4.10.2 and TS 6.9.1.11a.

To address expected increases in steam generator tube plugging (SGTP) for the current steam generators, analyses have been performed that support the operation of St. Lucie Unit 2 at 100% of rated thermal power (2700 MWt), with the following conditions:

1. Maximum SGTP of 30% in each of the two steam generators
2. Maximum tube plugging asymmetry of 7% between the two steam generators
3. A reduction in the Technical Specifications required minimum RCS flow from the current value of 355,000 gpm to 335,000 gpm.

The analyses are to be implemented for St. Lucie Unit 2 Cycle 15, which is planned to begin operation in December 2004. These analyses involve changes to the reload analysis methodology to improve and streamline the reload process related to cycle-

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information A001 specific physics calculations performed as part of the safety analysis checklist.

Attachment 1 is a description of the proposed changes and the supporting justification. Attachment 2 is the Determination of No Significant Hazards and Environmental Considerations. Attachment 3 contains marked up copies of the proposed Technical Specification changes. Attachment 4 contains information copies of the proposed changes to the TS Bases and Core Operating Limits Report (COLR). Attachment 5 contains copies of the retyped TS pages. Attachment 6 is a copy of the nonproprietary version of Westinghouse Licensing Report, St. Lucie Unit 2 30-Percent Steam Generator Tube Plugging and WCAP-9272 Reload Methodology Transition Project. Attachment 7 is a copy of Appendix C of the Westinghouse Licensing Report of Attachment 6, which contains the proprietary portions and the Westinghouse affidavit requesting that the information in Attachment 7 be withheld from public disclosure pursuant to 10 CFR 2.790 and the bases for the request.

Westinghouse Electric Company, LLC has determined that the information in Attachment 7 is proprietary in nature. Therefore, it is requested that this document be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790(a)(4). The Westinghouse reasons for the classification of this information as proprietary and the signed affidavit are included as part of Attachment 7.

The St. Lucie Facility Review Group and the Florida Power & Light Company Nuclear Review Board have reviewed the proposed amendment. In accordance with 10 CFR 50.91 (b)(1), a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

Approval of this proposed license amendment is requested by November 2004 to support the reload analyses for St. Lucie Unit 2 Cycle 15. Please issue the amendment to be effective on the date of issuance and to be implemented within 60 days of receipt by FPL. Please contact George Madden at 772-467-7155 if there are any questions about this submittal.

Very truly yours,

William Jefferson, Jr.
Vice President
St. Lucie Plant

WJ/GRM

Attachments

cc: Mr. William A. Passetti, Florida Department of Health

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

William Jefferson, Jr. being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

William Jefferson, Jr.

STATE OF FLORIDA

COUNTY OF ST LUCIE

Sworn to and subscribed before me

this 2nd day of Dec., 2003
by William Jefferson, Jr., who is personally known to me.

Name of Notary Public - State of Florida

Leslie J. Whitwell
MY COMMISSION # DD020212 EXPIRES
May 12, 2005
BONDED THRU TROY FAIN INSURANCE, INC.

(Print, type or stamp Commissioned Name of Notary Public)

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

ATTACHMENT 1

DESCRIPTION OF THE PROPOSED CHANGES AND JUSTIFICATION

1.0 Introduction/Background

The purpose of this proposed license amendment is to allow operation of St. Lucie Unit 2 with a reduced reactor coolant system (RCS) flow, corresponding to a steam generator tube plugging level of 30% per steam generator. The re-analysis performed to support this reduction in RCS flow has used Westinghouse WCAP-9272 methodology as described later in this report. The implementation of these changes will require amendment to the current Technical Specifications (TS).

To address expected increases in steam generator tube plugging (SGTP) for the current steam generators, analyses have been performed that support the operation of St. Lucie Unit 2 at 100% of rated thermal power (2700 MWt), with the following conditions:

1. Maximum SGTP of 30% in each of the two steam generators
2. Maximum tube plugging asymmetry of 7% between the two steam generators
3. A reduction in the Technical Specifications required minimum RCS flow from the current value of 355,000 gpm to 335,000 gpm.

The analyses are to be implemented for St. Lucie Unit 2 Cycle 15, which is planned to begin operation in December 2004. These analyses involve changes to the reload analysis methodology to improve and streamline the reload process related to cycle-specific physics calculations performed as part of the safety analysis checklist.

The Technical Specifications changes proposed in this license amendment include reducing the TS minimum RCS flow in TS Table 3.2-2 from 355,000 gpm to 335,000 gpm, changing in TS 3.1.1.4 the maximum upper limit for moderator temperature coefficient (MTC) at ≥ 70% of RATED THERMAL POWER so that MTC is not positive at full power, deletion of F_{xy} in TS 3/4.2.2, and deletion of design features in TS 5.4.2 (RCS volume). Deletion of F_{xy} required introduction of a cycle-dependent W(z) function in the surveillance of linear heat rate when using the excore detector monitoring system. Additionally, design features in TS 5.3.1 (fuel assemblies) is modified to be consistent with NUREG-1432, Revision 2 and core operating limits report (COLR) methodology list in TS 6.9.1.11b is updated to include new methodologies used in support of the reload analyses presented in this submittal.

The DNB parameter limits in TS Table 3.2-2 are proposed to be relocated to COLR, consistent with TSTF-339, Revision 2. The lower limit on the RCS flow will remain in TS, however, cycle-specific RCS flow limit is proposed to be in the COLR. TS Figure 2.1-1 is revised to be consistent with the reduction in RCS flow to 335,000 gpm. This change to the TS Figure 2.1-1 is accompanied with a change to the TS Bases Figure B2.1-1.

The deletion of F_{xy} is consistent with St. Lucie Unit 1 Technical Specifications and the deletion of RCS volume from the TS is consistent with NUREG-1432, Revision 2. The implementation of ZIRLO™ cladding is covered in the revised design features TS 5.3.1 (fuel assemblies). There are other changes to the Technical Specifications, which are dependent on the changes described above. The COLR linear heat rate limit in Figure 3.2-1 is reduced from 13 kW/ft to 12.5 kW/ft based on the large break loss-of-coolant accident (LOCA) analysis requirements. COLR Figure 3.2-2 is revised accordingly to comply with this linear heat rate limit. The part power Fr limit in the COLR Figure 3.2-3 is revised to make it consistent with the analysis.

There are several methodology changes implemented in the revised reload analyses. A summary table of methodologies used for St. Lucie Unit 2 analyses is presented in Section 1 of Attachment 6, and the conditional requirements for the use of each methodology are addressed in Appendix B of Attachment 6.

The reload analysis and methodology uses the approach and philosophy described in WCAP-9272 for the review and acceptability of cycle-specific reload designs. This reload safety evaluation methodology is applied to St. Lucie Unit 2 for the first time and is based on a bounding approach for the safety analysis that includes a reload safety analysis checklist (RSAC) to confirm the validity of the safety analysis for cycle-specific reload designs. VIPRE-W is used as the thermal-hydraulic subchannel analysis code with ABB-NV DNB correlation (WCAP-14565-P-A, Addendum 1). RETRAN (WCAP-14882-P-A) is used as the transient analysis code for the non-LOCA event analysis except for the steam generator tube rupture event. The application of the RETRAN model to St. Lucie Unit 2 required no coding changes and plant specific features (control and protection systems) are accommodated through proper modeling changes using features already available in the RETRAN code approved for application to Westinghouse pressurized water reactors. The RETRAN modeling for St. Lucie Unit 2 is described in Appendix C of Attachment 6.

The DNB analysis uses the Westinghouse revised thermal design procedure (RTDP) methodology (WCAP-11397-P-A) in place of the currently used extended statistical combination of uncertainties (ESCU) approach. The Westinghouse relaxed axial offset control (RAOC) methodology (WCAP-10216-P-A) is used to evaluate axial power distributions. The application of the RAOC analysis is presented in Appendix A of Attachment 6.

ANC/PHOENIX (WCAP-11596-P-A) code package, currently in use for St. Lucie Unit 2, continues to be the neutronics analysis methodology. FATES-3B (CEN-161-(B)-P, Supplement 1-P-A) will remain the fuel performance code. The implementation of ZIRLO™ cladding is in accordance with CENPD-404-P-A, which is generically approved by the NRC for Combustion Engineering (CE) plant applications.

The 99-Evaluation Model (CENPD-132, Supplement 4-P-A) is used for the large break LOCA analysis, whereas the S2M model (CENPD-137, Supplement 2-P-A) is used for the small break LOCA analysis. While these LOCA methods are new in application for St. Lucie Unit 2, they have been generically approved by the NRC for CE plant applications.

The methodology and assumptions used in dose consequence analysis, submitted previously in FPL letter L-2003-220 using the alternate source term methodology, remain applicable for the current reload analysis for 30% SGTP, except for the steam generator tube rupture. Steam generator tube rupture dose consequences are recalculated using the same methodology and assumptions as those in L-2003-220, except that the steam release information was adjusted to cover the 30% SGTP analysis.

First Application to St. Lucie Unit 2 of Generically Approved Methodologies for CE Plants

1. CENPD-132, Supplement 4-P-A, Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model, March 2001.

2. CENPD-137, Supplement 2-P-A, Calculative Methods for the ABB CE Small Break LOCA Evaluation Model, April 1998.

3. CENPD-404-P-A, Revision 0, Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November 2001.

Application of Methodologies Not Previously Approved for Application to St. Lucie Unit 2

1. WCAP-7588, Revision 1-A, An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, January 1975.

2. WCAP-7908-A, FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod, December 1989.

3. WCAP-7979-P-A/WCAP-8028-A, TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code, January 1975.

4. WCAP-9272-P-A/9273-NP-A, Westinghouse Reload Safety Evaluation Methodology, July 1985.

5. WCAP-10216-P-A, Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification, February 1994.

6. WCAP-11397-P-A, Revised Thermal Design Procedure, April 1989.

7. WCAP-14882-P-A, RETRAN Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses, February 1999.

Methodologies/Submittals Currently Under NRC Review for Application to St. Lucie Unit 2

1. WCAP-14565-P-A, Addendum 1, Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code, May 2003.

2. L-2003-220 dated September 18, 2003, Alternate Source Term Methodology and Conforming Amendments.

2.0 Description of the Proposed Changes

2.1 TS Page 1-7, TS Definition 1.37: Unrodded Planar Radial Peaking Factor – F_{xy}

Definition of Unrodded Planar Radial Peaking Factor is deleted from the Technical Specifications. TS Definition 1.38 is renumbered to 1.37 based on the above deletion.

2.2 TS Page 2-3, TS Safety Limit 2.1.1: (Reactor Core), Figure 2.1-1 Thermal Margin Safety Limit Lines and TS Bases Figure B2.1-1

TS Figure 2.1-1 is revised to reflect the reduction in RCS flow from 355,000 gpm to 335,000 gpm and the changes to the methodology. The revised TS basis Figure B2.1-1 reflects the axial shapes used in the generation of the DNB lines on TS Figure 2.1-1.

2.3 TS Page 2-5, TS Reactor Trip Setpoints Table 2.2-1: Reactor Protective Instrumentation Trip Setpoint Limits

The specification of flow in the footnote to this table is changed to refer to COLR limit for the minimum RCS flow in the COLR Table 3.2-2.

2.4 TS Page 2-9, TS Reactor Trip Setpoints Figure 2.2-3: Thermal Margin/Low Pressure Trip Setpoint Part 1 (Y_1 Versus A_1)

This figure is replaced by a more clear figure with no changes.

2.5 TS Page 3/4 1-5, TS Limiting Condition for Operation (LCO) 3.1.1.4: Moderator Temperature Coefficient (MTC)

This TS is modified to change the MTC for power levels greater than 70% of RATED THERMAL POWER from +3 pcm/^{0}F to a linear ramp from +5 pcm/^{0}F at 70% of RATED THERMAL POWER to 0 pcm/^{0}F at 100% of RATED THERMAL POWER.

2.6 TS Page 3/4 2-2, TS Surveillance Requirement (SR) 3/4.2.1: Linear Heat Rate

TS 4.2.1.3 is modified to replace reference to F_{xy} with reference to F_r^T in TS 4.2.1.3.c. Additionally, items 4.2.1.3.d, 4.2.1.3.e, and 4.2.1.3.f are added to include a function W(z) in the linear heat rate surveillance using the excore detector monitoring system. This function W(z), to be defined in COLR, is a cycle dependent function that accounts for power distribution transient encountered during normal operation.

2.7 TS Pages 3/4 2-7 and 3/4 2-8, TS 3/4.2.2: Total Planar Radial Peaking Factors - F^T_{xy}

This TS is deleted and TS pages 3/4 2-7 and 3/4 2-8 are denoted as deleted pages. Change 2.12 discusses conforming changes to TS 3.2.4 on TS Page 3/4 2-13, TS 3/4.10.2 on TS Page 3/4 10-2, and TS 6.9.1.11.a on TS Page 6-20.

2.8 TS Page 3/4 2-15, TS 3.2.5: DNB Parameters

TS Table 3.2-2 (DNB Margin Limits) is changed to relocate cold leg temperature, pressurizer pressure, and RCS flow rate limits to the COLR. The lower limit of RCS flow rate continues to remain in this TS. Additional editorial conforming changes are proposed for Table 3.2-2.

2.9 TS Page 5-3, TS Design Feature 5.3.1: Fuel Assemblies

The description of fuel assemblies in the Design Features, TS 5.3.1, is modified to include both Zircaloy and ZIRLO™ as fuel rod cladding material and reworded to be similar in level of detail as NUREG 1432, Revision 2.

2.10 TS Page 5-4, TS Design Feature 5.4.2: Reactor Coolant System Volume

This Design Feature, TS 5.4.2, is deleted.

2.11 TS Page 6-20d, TS Administrative Requirement 6.9.1.11: Core Operating Limits Report (COLR)

In TS 6.9.1.11.b the list of analytical methods that can be used to determine the core operating limits is expanded to include the following new methodologies used in the St. Lucie Unit 2 analysis.

56. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

57. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

58. CENPD-404-P-A, Revision 0, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.

59. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

60. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification," February 1994.

61. WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure," April 1989.

62. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

63. WCAP-14565-P-A, Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

64. 30% SGTP PLA submittal L-2003-276 and the associated NRC SER for this submittal.

2.12 TS Conforming Changes Related to the Above Changes

TS Index Page II is changed based on Change 2.1, TS Index Page V is changed based on Change 2.7, and TS Index Page XVII is changed based on Change 2.10.

TS Page 3/4 2-13, TS 3.2.4.a and 3.2.4.b are modified to delete F_{xy} requirements based on Change 2.7.

TS Page 3/4 10-2, TS 3/4.10.2 is revised to delete references to TS 3.2.2.

TS Page 6-20, TS 6.9.1.11a is modified to reflect the deletion of TS 3.2.2 and changes made to TS Table 3.2-2.

TS Page 3/4 2-2a is added due to typing overflow from current TS pages 3/4 2-2 and TS Page 6-20e was changed to reflect typing overflow from TS Page 6-20d.

3.0 Basis For Proposed Changes:

3.1 TS 1.37: Unrodded Planar Radial Peaking Factor – F_{xy}

The definition of Unrodded Planar Radial Peaking Factor is deleted consistent with the deletion of F_{xy} TS 3.2.2.

3.2 TS 2.1.1: (Reactor Core), Figure 2.1-1 Thermal Margin Safety Limit Lines and TS Bases Figure B2.1-1

These figures are updated to reflect the revised minimum RCS flow of 335,000 gpm and the changes to the analysis methodology. The DNB analysis shows that the TM/LP trip provides adequate protection against the violation of these limit lines. The accident analysis acceptance criteria continue to be met at the revised conditions.

3.3 TS Table 2.2-1: Reactor Protective Instrumentation Trip Setpoint Limits

The specification of flow in the footnote to this table is changed to refer to the COLR limit specified in COLR Table 3.2-2. Analyses and evaluations have been performed that support plant operation with low flow trip setpoint based on the reduced RCS flow.

3.4 TS Figure 2.2-3: Thermal Margin/Low Pressure Trip Setpoint Part 1 (Y_1 Versus A_1)

This is an administrative change to provide a more clear figure. There are no changes to the values that comprise this figure.

3.5 TS 3.1.1.4: Moderator Temperature Coefficient (MTC)

Detailed analyses and evaluations were performed at the revised operating conditions. The accident analysis acceptance criteria continue to be met at the revised conditions with linear MTC ramp from +5 pcm/^{0}F at 70% of RATED THERMAL POWER to 0 pcm/^{0}F at 100% of RATED THERMAL POWER.

3.6 TS 3/4.2.1: Linear Heat Rate

TS 4.2.1.3 is modified to replace reference to F_{xy} with reference to F_r^T in TS 4.2.1.3.c. Also, for excore detector monitoring, a penalty function W(z) is added in the linear heat rate surveillance. The accident analysis acceptance criteria continue to be met with these changes and revised operating conditions.

3.7 TS 3/4.2.2: Total Planar Radial Peaking Factors - F^t_{xy}

Constant monitoring for linear heat rate with the incore monitoring system makes the F_{xy} surveillance redundant. For excore detector monitoring of linear heat rate, additional surveillance requirements are added in new TS 4.2.1.3d - f using a cycle dependent penalty function W(z). Therefore F_{xy} surveillance has been eliminated, consistent with the previous changes to the St. Lucie Unit 1 Technical Specifications.

Also, F_{xy} does not meet any of the criteria set forth in 10 CFR 50.36(c)(2)(ii) for a technical specification limiting condition for operation.

(A) F_{xy} *is not used to indicate a significant abnormal degradation of the* reactor coolant pressure boundary,

(B) F_{xy} is not is an initial condition of a design basis accident or transient analysis that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier,

(C) F_{xy} is not a part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier, and,

(D) F_{xy} is not an item which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

3.8 TS 3.2.5: DNB Parameters

The relocation of DNB parameters to the COLR is consistent with TSTF-339, Revision 2, which provides operating and analysis flexibility from cycle to cycle. The minimum lower limit of RCS flow rate continues to remain in the TS. The detailed analyses performed at the revised RCS flow conditions showed that accident analysis acceptance criteria continue to be met.

3.9 TS 5.3.1: Fuel Assemblies

New fuel assemblies will utilize ZIRLO™ fuel rod cladding. The description of fuel assemblies in the design features section of TS is modified to include both Zircaloy and ZIRLO™ fuel rod cladding and to be consistent in level of detail with NUREG-1432, Revision 2.

3.10 TS 5.4.2: Reactor Coolant System Volume

The safety analyses use specific RCS volumes as inputs where required, consistent with the assumed steam generator tube plugging level. The *specification of the RCS volume in the Design Features TS is deleted, consistent* with NUREG-1432, Revision 2.

3.11 TS 6.9.1.11b: Core Operating Limits Report (COLR)

The COLR list of references is being updated to include additional analysis methodologies used in the St. Lucie Unit 2 revised safety analysis.

3.12 TS Changes Related to the Above Changes

Changes to TS 3.2.4.a, 3.2.4.b, 3/4.10.2, and 6.9.1.11a are made so that these specifications remain consistent with the proposed TS changes for deletion of Fxy TS 3.2.2 and relocation of DNB parameters to the COLR.

4.0 Analysis Of Effects On Safety:

4.1 Fuel Assembly Design

The fuel assembly design for Cycle 15 will include ZIRLO™ cladding on the fuel rods. The mechanical design criteria for the fuel assembly, fuel rods, and poison rods will continue to be met for all fuel region designs. This is described in Section 2 of Attachment 6.

4.2 Thermal Hydraulic Analysis

4.2.1 DNBR Analysis

Steady state DNBR analyses at the rated power level of 2700 MWt have been performed using the VIPRE computer code, the ABB-NV Critical Heat Flux (CHF) correlation, and the revised thermal design procedure (RTDP) methodology. More details of this procedure are presented in Section 4 of Attachment 6.

4.2.2 Effects of Fuel Rod Bowing on DNBR Margin

The fuel rod bowing has a potential impact on the bundle power distribution and on the DNB margin of fuel rods. The effects of fuel rod bowing on DNBR margin have been incorporated into the safety and setpoint analyses in the same manner as currently done. The penalty used for this analysis, 1.2% on minimum DNBR, remains valid for the ABB-NV DNB correlation as specified in Section 4 of Attachment 6.

4.3 LOCA Analysis

An Emergency Core Cooling System (ECCS) performance analysis was performed for St. Lucie Unit 2 to demonstrate conformance to the ECCS performance acceptance criteria for light water nuclear power reactors. For large break LOCA analysis, the linear heat rate limit was reduced from 13 kW/ft to 12.5 kW/ft. This limit will be reflected in the Cycle 15 COLR.

The large break LOCA ECCS performance analysis was performed using the NRC-approved 99-evaluation model (CENPD-132, Supplement 4-P-A). The small break LOCA ECCS performance analysis was performed using the NRC-approved S2M evaluation model (CENPD-137, Supplement 2-P-A). The post-LOCA boric acid precipitation analysis was performed using the NRC-approved post-LOCA long term cooling (LTC) evaluation model (CENPD-254-P-A). The details of these analyses are presented in Section 5.2 of Attachment 6.

The results showed that:

Parameter		Criterion
Peak Cladding Temperature	≤	2200 °F
Maximum Cladding Oxidation	≤	17 %
Core-wide Cladding Oxidation	≤	1 %

Additionally the long term cooling analysis demonstrated that the core decay heat removal capability is maintained and the boric acid concentration in the core is maintained below the solubility limit when the simultaneous hot and cold leg safety injection flow is initiated between two and six hours after the start of the LOCA. The results of the analysis demonstrate conformance to the ECCS acceptance criteria at an initial peak linear heat generation rate of 12.5 kW/ft.

4.4 Non-LOCA Analysis

The Non-LOCA safety analysis was performed for Unit 2 utilizing WCAP-9272 methodology and using the RETRAN (WCAP-14882-P-A) computer code to support plant operation up to 30% steam generator tube plugging. The Design Bases Events (DBE) considered in the safety analyses (listed in Table 4.4-1) are categorized into two major groups: Moderate Frequency events, referred to as Anticipated Operational Occurrences (AOO), and Postulated Accidents.

For events that are predicted to have fuel failures, acceptable values for fuel failure were derived from the dose consequence analysis previously submitted in L-2003-220. Cycle specific analyses presented in Section 5.1 of Attachment 6 have verified that the fuel failures for these events meet the failure limits of the dose consequence analysis.

For the post-trip steam line break event, conservative assumptions used for the hot zero power (HZP) case bound the case for hot full power. Also, as discussed in Appendix A of Attachment 6, the loss of offsite power case is bounded by the case without the loss of offsite power. As such, only the HZP case without the loss of offsite power is presented in Section 5.1.6 of Attachment 6 for the post-trip steam line break.

The key parameters assumed in the transient analysis, and the specific initial conditions for each event, along with the analysis results are provided in Section 5.1 of Attachment 6.

4.5 Setpoint Analysis

The Setpoint Analysis confirms that cycle-specific power distributions satisfy the DNB and LHR Limiting Conditions for Operation (LCO), the Limiting Safety System Settings

(LSSS), and the equipment setpoint requirements for St. Lucie Unit 2. In the WCAP-9272 RSAC methodology, the relaxed axial offset control (RAOC) method is used for power distribution confirmation. A representative RAOC analysis is presented in Appendix A of Attachment 6. Cycle specific evaluations are performed to verify that safety limits are met for reload design.

4.6 Dose Consequence Analysis

The dose events analyzed for St. Lucie Unit 2, with respect to off-site dose and control room dose, are the following:

Loss-of-Coolant Accident
Fuel Handling Accident
Main Steam Line Break
Steam Generator Tube Rupture
Reactor Coolant Pump Shaft Seizure
Control Element Assembly Ejection
Primary Line Break (Letdown Line)
Feedwater Line Break
Waste Gas Decay Tank Rupture

The dose analyses, submitted in L-2003-220, describe the analysis approach and assumptions used, which are based on the alternate source term methodology from Regulatory Guide (RG) 1.183. The methodology and analysis assumptions for dose calculations related to offsite dose and control room dose presented in L-2003-220 remain applicable to the 30% SGTP analysis. The dose consequences reported in L-2003-220 for all events cover 30% SGTP analysis, except for the steam generator tube rupture (SGTR). The dose calculations for SGTR are redone using the same methodology and assumptions, except for the steam release information from the two steam generators. The doses calculated using the steam release information from 30% SGTP steam generator tube rupture analysis are presented in Section 5.1.24 of Attachment 6.

The fuel failure limits for postulated non-LOCA accidents remain unchanged from those presented in L-2003-220. The fuel failure events are:

Main Steam Line Break
Reactor Coolant Pump Shaft Seizure
Control Element Assembly Ejection
Primary Line Break (Letdown Line) – no fuel failure assumed
Feedwater Line Break – no fuel failure assumed

The non-LOCA analyses described in Section 5.1 of Attachment 6 have verified that the fuel failure limits are not violated in the analyses of these events performed with 30% SGTP conditions. The dose consequences of AOOs remain bounded by those of the feedwater line break, which is analyzed to the same criteria as the AOOs. It is thus concluded that all the dose consequences meet the limits specified in RG 1.183 and 10 CFR 50.67.

4.7 Containment Pressure/Temperature Analysis

The analysis of containment mass and energy releases for main steam line break and LOCA and their impact on the containment pressure/temperature response are presented in Sections 5.3, 5.4, and 5.5 of Attachment 6. It is concluded that the peak containment pressure/temperature continues to meet the acceptance criteria for containment integrity.

Table 4.4-1
St. Lucie Unit 2,
Design Basis Events Considered in the Safety Analysis
(Section Numbers refer to those in Attachment 6)
*Postulated Accidents

Section	Sub-Section	Description
5.1	Increase in Heat Removal by the Secondary System	
	5.1.3	Decrease in Feedwater Temperature
	5.1.1	Increase in Feedwater Flow
	5.1.4	Increased Main Steam Flow
	5.1.2	Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve
	5.1.5*	Pre-Trip Steam System Piping failures
	5.1.6*	Post-Trip Steam System Piping failures
	5.1.7*	Steam System Piping failures Outside Containment
5.1	Decrease in Heat Removal by the Secondary System	
	5.1.8	Turbine Trip
	5.1.10	Loss of Condenser Vacuum
	5.1.9	Loss of Offsite Power
	5.1.9	Loss of Normal Feedwater
	5.1.12*	Feedwater Line Break Event
5.1	Decrease in Reactor Coolant Flowrate	
	5.1.13	Partial Loss of Forced Reactor Coolant Flow
	5.1.14	Total Loss of Forced Reactor Coolant Flow
	5.1.15*	Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft
5.1	Reactivity and Power Distribution Anomalies	
	5.1.17	Uncontrolled CEA Withdrawal from a Subcritical Condition
	5.1.16	Uncontrolled CEA Withdrawal at Power
	5.1.18	Control Element Assembly Drop Event
	5.1.19	CVCS Malfunction (Uncontrolled Boron Dilution)
	N/A	Startup of an Inactive Reactor Coolant System Pump Event
	5.1.20*	Control Element Assembly Ejection
5.1	Increase in Reactor Coolant System Inventory	
	5.1.21	CVCS Malfunction
	N/A	Inadvertent Operation of the ECCS During Power Operation
5.1	Decrease in Reactor Coolant System Inventory	
	5.1.22	Pressurizer Pressure Decrease Events
	5.1.23*	Primary Line Break Outside Containment

Table 4.4-1
St. Lucie Unit 2,
Design Basis Events Considered in the Safety Analysis
(Section Numbers refer to those in Attachment 6)
*Postulated Accidents

Section	Sub-Section	Description
	5.1.24*	Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power
5.1	Miscellaneous	
	5.1.11	Transients Resulting from the Malfunction of One Steam Generator

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Introduction

The purpose of this proposed license amendment is to allow operation of St. Lucie Unit 2 with a reduced reactor coolant system (RCS) flow, corresponding to a steam generator tube plugging level of 30% per steam generator. The analysis performed to support this reduction in RCS flow has used Westinghouse WCAP-9272, Westinghouse Reload Safety Evaluation Methodology. The implementation of these changes will require changes to the current Technical Specifications (TS).

The proposed amendment includes the following Technical Specifications changes: Revision to the Thermal Margin Safety Limit Lines TS Figure 2.1-1, reduction in RCS flow in TS Table 3.2-2 and in footnote to TS Table 2.2-1, changes to positive MTC in TS 3.1.1.4, changes to surveillance requirements for Linear Heat Rate TS 3/4.2.1, deletion of Fxy TS 3/4.2.2, relocation to COLR of DNB parameters in TS 3.2-5, changes to Design Features Fuel Assemblies TS 5.3.1, deletion of Design Features RCS Volume TS 5.4.2, COLR methodology list update in TS 6.9.1.11b and conforming changes to TS 1.38, TS 3.2.4, TS 3/4.10.2 and TS 6.9.1.11a.

To address expected increases in steam generator tube plugging (SGTP) for the current steam generators, analyses have been performed that support the operation of St. Lucie Unit 2 at 100% of rated thermal power (2700 MWt), with the following conditions:

1. Maximum SGTP of 30% in each of the two steam generators
2. Maximum tube plugging asymmetry of 7% between the two steam generators
3. A reduction in the Technical Specifications required minimum RCS flow from the current value of 355,000 gpm to 335,000 gpm.

These analyses involve changes to the reload analysis methodology to improve and streamline the reload process related to cycle-specific physics calculations performed as part of the safety analysis checklist.

Determination of No Significant Hazards Consideration

The standards used to arrive at a determination that a request for amendment involves a no significant hazards consideration are included in the Commission's regulation, 10 CFR 50.92, which states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; nor (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

None of the proposed changes to the Technical Specifications nor the reload methodology result in operation of the facility that would adversely affect the initiation of any accident previously evaluated. There is no adverse impact on any plant system. All systems will function as designed, and all performance requirements for these systems remain acceptable. The comprehensive engineering effort, performed to support the proposed changes, has included evaluations or analyses of all the accident analyses including the effects of ZIRLO™ fuel rod cladding. The DNBR and setpoint analyses have verified that the accident analyses criteria continue to be met. Dose consequences acceptance criteria have been verified to be met for all the events.

Therefore, the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any previously evaluated.

No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes to the Technical Specifications and the reload methodology. The proposed changes have no adverse effects on any safety-related systems and do not challenge the performance or integrity of any safety-related system. The DNBR limits and trip setpoints associated with the respective reactor protection system functions have verified that the accident analyses criteria continue to be met.

Therefore, this amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The safety analyses of all design basis accidents, supporting the proposed changes to the Technical Specifications and the reload methodology, continue to meet the applicable acceptance criteria with respect to the radiological consequences, specified acceptable fuel design limits (SAFDLs), primary and secondary overpressurization, and 10 CFR 50.46 requirements. The DNBR and the setpoint analyses are performed on a cycle-specific basis to verify that the reactor protection system functions continue to provide adequate protection against fuel design limits. Revised steam line break and LOCA mass and energy releases were determined and used to confirm the overall containment response remains acceptable. The performance requirements for all systems have been verified to be acceptable from design basis accidents' consideration. The proposed amendment, therefore, will not involve a significant reduction in the margin of safety.

Based on the preceding, FPL has determined that the proposed amendments do not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any previously evaluated; nor (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.

Environmental Impact Consideration Determination

The proposed license amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released off-site, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendment involves no significant hazards consideration, and therefore, meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendment.

ATTACHMENT 3

ST. LUCIE UNIT 2 MARKED-UP TECHNICAL SPECIFICATION PAGES

INDEX

DEFINITIONS (Continued)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

INDEX

DESIGN FEATURES

DEFINITIONS

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.36 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in the unrodded core, excluding tilt.

~~UNRODDED PLANAR RADIAL PEAKING FACTOR - F_{xy}~~

~~1.37 The UNRODDED PLANAR RADIAL PEAKING FACTOR is the maximum ratio of the peak to average power density of the individual fuel rods, in any of the unrodded horizontal planes, excluding tilt.~~

VENTILATION EXHAUST TREATMENT SYSTEM

~~1.38~~ 1.37 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Features (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

St. Lucie Unit 2
Docket No. 50-389
L-2003-276 Attachment 3 Page 6

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

Replace this figure with the one on the next page

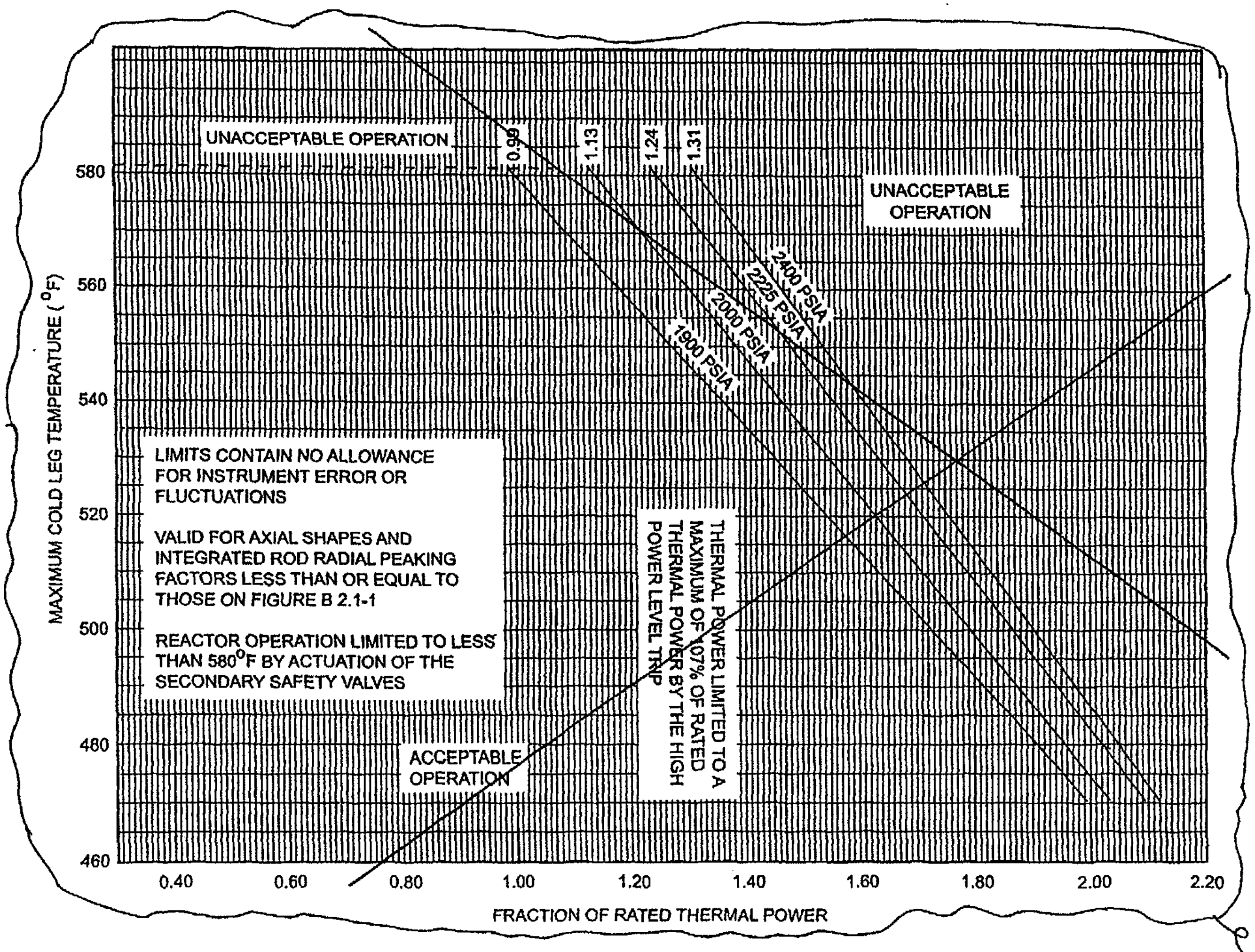

FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES FOUR REACTOR COOLANT PUMPS OPERATING

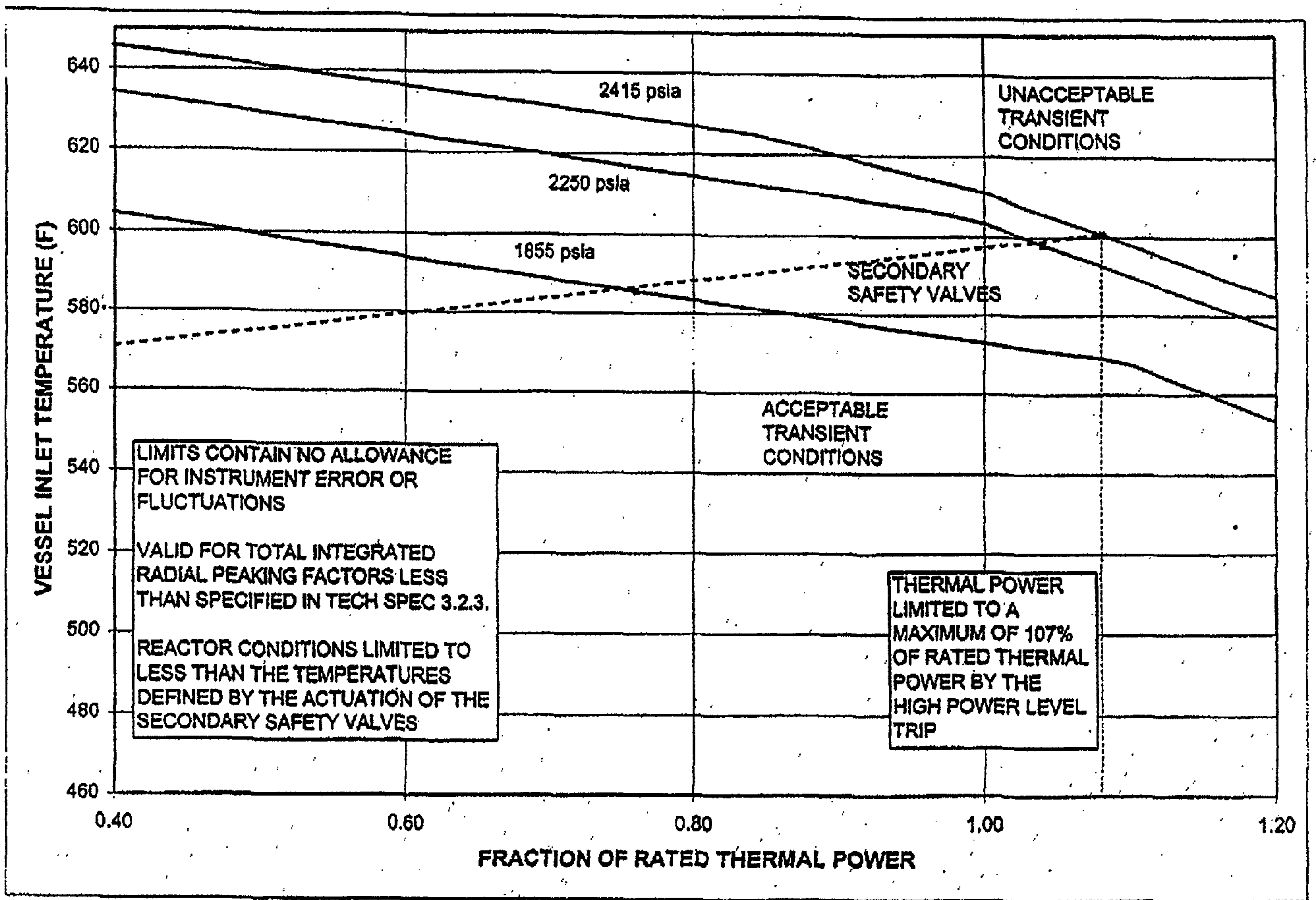

FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES
FOUR REACTOR COOLANT PUMPS OPERATING

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density – High[5] Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps – Low	≥ 636 gpm**	≥ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power – High[4]	≤ 2.49 decades per minute	≤ 2.49 decades per minute
14. Reactor Coolant Flow – Low[1]	≥ 95.4% of design Reactor Coolant flow with four pumps operating*	≥ 94.9% of design Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low[5]	≥ 800 psig	≥ 800 psig

* Design reactor coolant flow with four pumps operating is ~~355,000 gpm.~~

** 10-minute time delay after relay actuation.

the minimum RCS flow specified in the COLR Table 3.2-2.

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

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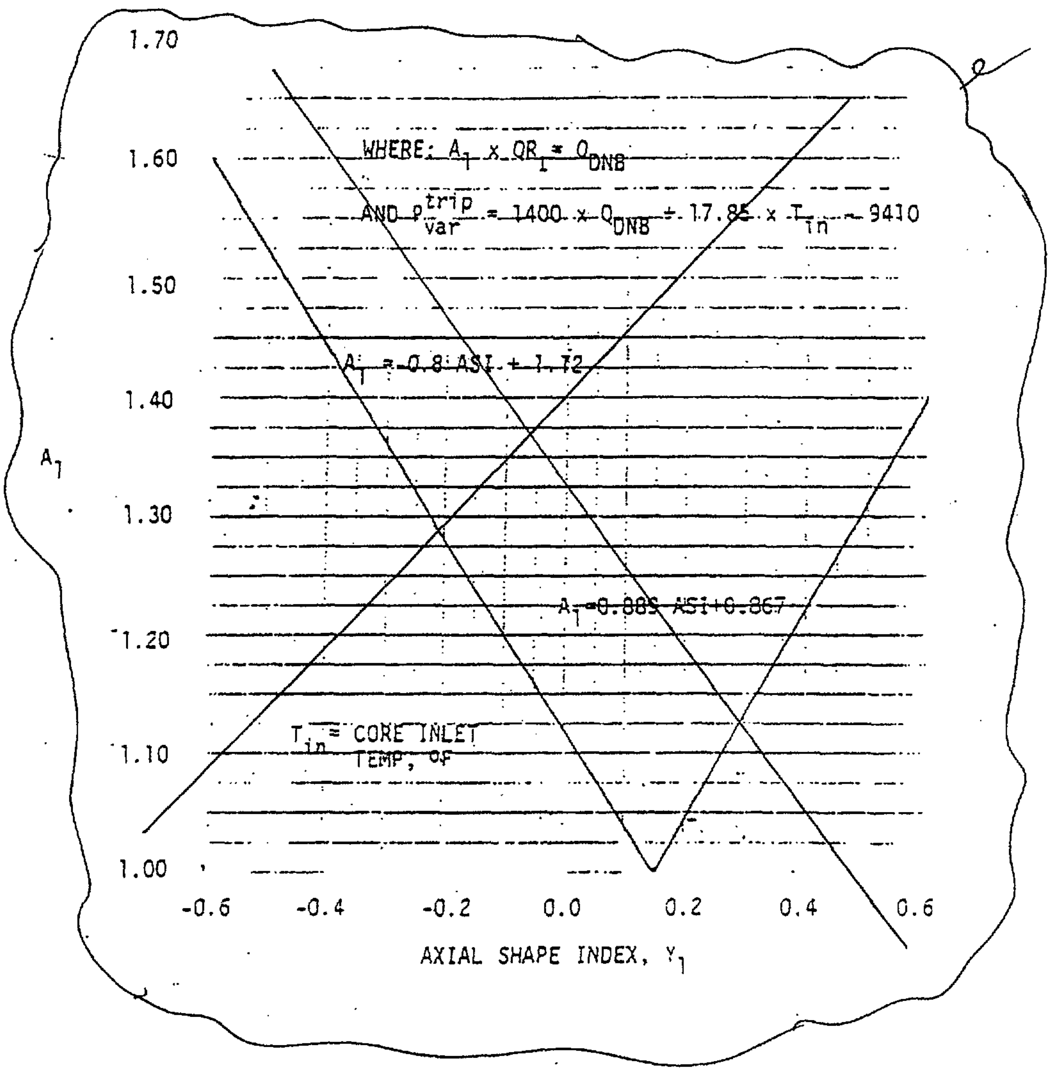


FIGURE 2.2-3

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 1 (Y_1 Versus A_1)

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

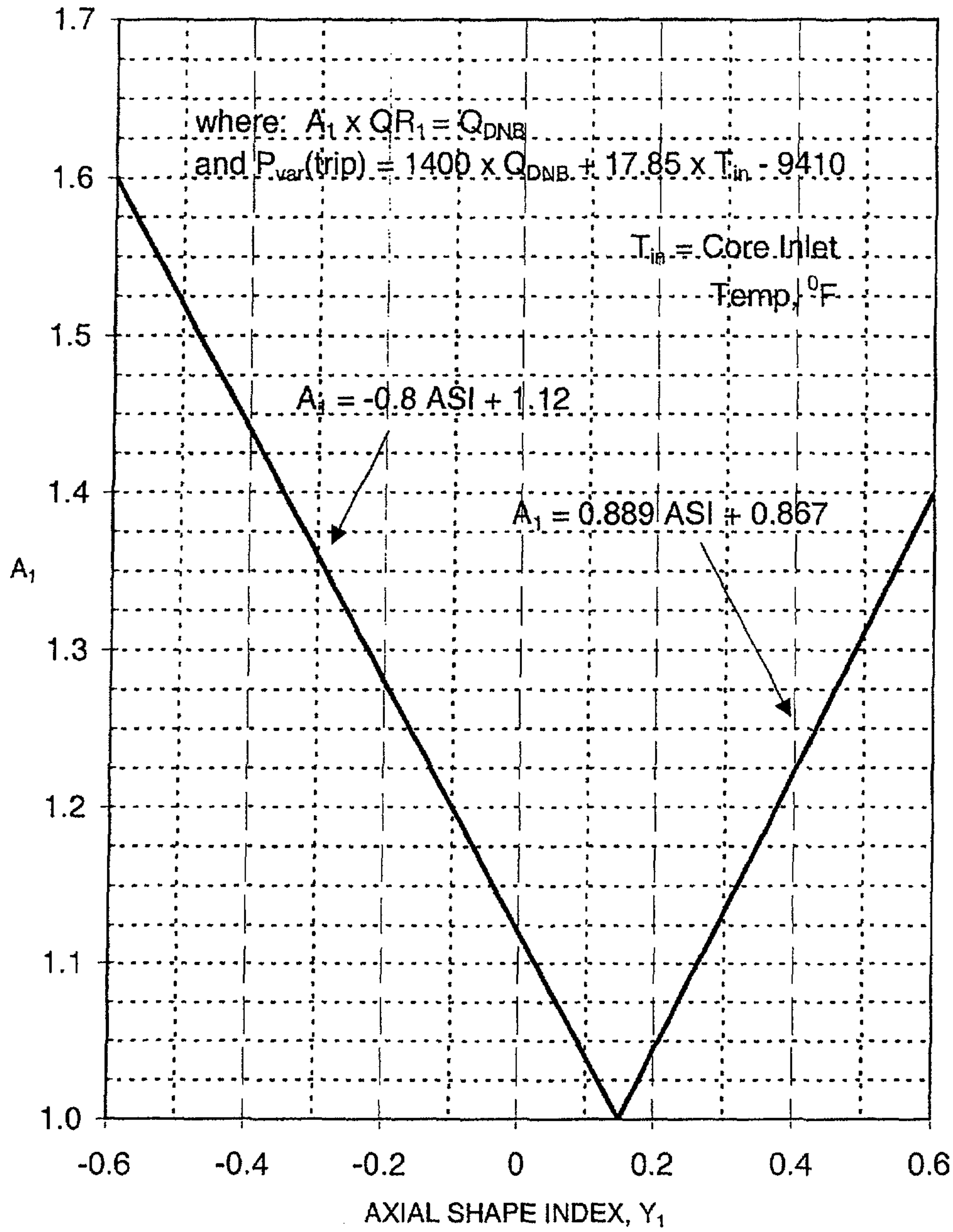


FIGURE 2.2-3
THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 1 (Y_1 Versus A_1)

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the COLR. ~~The maximum positive limit shall be:~~

~~a. Less positive than +5 pcm/°F at ≤ 70% RATED THERMAL POWER, and~~

~~b. Less positive than +3 pcm/°F at > 70% RATED THERMAL POWER.~~

The maximum upper limit shall be +5 pcm/^{0}F at ≤ 70% of RATED THERMAL POWER, with a linear ramp from +5 pcm/^{0}F at 70% of RATED THERMAL POWER to 0 pcm/^{0}F at 100% of RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 800 ppm.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

*With K_{eff} greater than or equal to 1.0.
#See Special Test Exceptions 3.10.2 and 3.10.5.

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of COLR Figure 3.2-2, where 100% of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:

M x N

where:

$\boxed{F_r^T}$

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the $\cancel{F_{xy}^T}$ curve of COLR Figure 3.2-3.

4.2.1.4 Incore Detector Monitoring System # - The incore detector monitoring system may be used for monitoring the linear heat rate by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on COLR Figure 3.2-1.

Insert A from the next page

If incore system becomes inoperable, reduce power to M x N within 4 hours and monitor linear heat rate in accordance with Specification 4.2.1.3.

Insert A

d. Verifying that the measured linear heat rate $LHR^M(z)$, obtained from a previous incore detector power distribution map, meets the following criteria:

$$LHR^M(z) \leq \frac{LHR}{W(z)}$$

W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. LHR and W(z) are specified in COLR Figure 3.2-1 and Table 3.2-3, respectively.

e. Operation is limited to the following:

1. The operation using excore detector monitoring system is limited to less ≤ 10% above the power level corresponding to the power level at which $LHR^M(z)$ is determined in Specification 4.2.1.3d.

2. Continuous operation using excore detector monitoring system is limited to 31 days from the time of the power distribution map used in Specification 4.2.1.3d.

f. The limit specified in Specification 4.2.1.3d above is not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive
2. Upper core region from 85 to 100%, inclusive

POWER DISTRIBUTION LIMITS

3/4.2.2 TOTAL PLANAR RADIAL PEAKING FACTORS - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F^T_{xy} shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} not within limits, within 6 hours either :

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} to within the limits of COLR Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}(1+T_q)$ when F_{xy} is calculated with a non-full core power distribution analysis code and shall be calculated as $F^T_{xy} = F_{xy}$ when calculations are performed with a full core power distribution analysis code. F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70% of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within 4 hours if the AZIMUTHAL POWER TILT (T_q) is > 0.03.

*See Special Test Exception 3.10.2.

Delete

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

~~4.2.2.3 F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing reactor coolant pump combination. *This determination shall be limited to core planes between 15% and 85% of full core height and shall exclude regions influenced by grid effects.*~~

~~4.2.2.4 T_q shall be determined each time a calculation of F^T_{xy} is made using a non full core power distribution analysis code. The value of T_q used in this case to determine F^T_{xy} shall be the measured value of T_q.~~

Delete

POWER DISTRIBUTION LIMITS

3/4.2.4 AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.03.

APPLICABILITY: MODE 1*.

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > .030 but ≤ .10, either correct the power tilt within 2 hours or determine within the next 2 hours and at least once per subsequent 8 hours, that ~~the TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) and~~ the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) ~~are~~ is within the limits of Specification~~s 3.2.2 and~~ 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) ~~and TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) are~~ is within the limits of Specification~~s 3.2.2 and~~ 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 7 days.

b. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER is > 75% of RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

TABLE 3.2-2

DNB MARGIN

LIMITS

PARAMETER	**FOUR REACTOR COOLANT PUMPS OPERATING**
Cold Leg Temperature (Narrow Range)	~~535°F* ≤ T ≤ 549°F~~ Within the limits specified in the COLR Table 3.2-2
Pressurizer Pressure *	~~2225 psia** ≤ P_{PZR} ≤ 2350 psia*~~ Within the limits specified in the COLR Table 3.2-2
Reactor Coolant Flow Rate	~~≥ 355,000 gpm~~ ≥ 335,000 gpm and ≥ the limit specified in the COLR Table 3.2-2
AXIAL SHAPE INDEX	COLR Figure 3.2-4

* Applicable only if power level ≥ 70% RATED THERMAL POWER

** * Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

SPECIAL TEST EXCEPTIONS

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion and power, distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.2.2~~ 3.2.3 and 3.2.4 may be suspended during performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.2.2~~ 3.2.3 and 3.2.4 are suspended, either:

a. *Reduce* THERMAL POWER *sufficiently to satisfy the requirements of* Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.2.2~~ 3.2.3, or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.2.2~~ 3.2.3, or 3.2.4 are suspended.

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with ~~each fuel assembly containing~~ 236 fuel and poison rod locations. ~~All~~ fuel and poison rods are clad ~~with Zircaloy-4~~. Each ~~fuel~~ rod shall have a nominal active fuel length of 136.7 inches ~~and contain~~ approximately 1700 grams uranium. The initial ~~core loading~~ shall have a ~~maximum enrichment~~ of 2.73 weight ~~percent U-235~~. Reload fuel shall be similar in ~~physical design to~~ the initial core loading.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 91 full-length control element assemblies and no part-length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

5.3.1 The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO clad fuel rods and/or poison rods, with fuel rods having an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

DESIGN FEATURES

~~VOLUME~~

~~5.4.2 The total water and steam volume of the reactor coolant system is 10,931 ± 275 cubic feet at a nominal T_{avg} of 572°F.~~

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel pool and spent fuel storage racks shall be maintained with:

1. A k_{eff} equivalent to less than 1.0 when flooded with unborated water, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

2. A k_{eff} equivalent to less than or equal to 0.95 when flooded with water containing 520 ppm boron, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

3. A nominal 8.96 inch center-to-center distance between fuel assemblies placed in the storage racks.

b. Fuel placed in Region I of the spent fuel storage racks shall be stored in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:

1. Fresh fuel shall have a nominal average U-235 enrichment of less than or equal to 4.5 weight percent.

2. The reactivity effect of CEAs placed in fuel assemblies may be considered.

3. The reactivity equivalencing effects of burnable absorbers may be considered.

4. The reactivity effects of fuel assembly burnup and decay time may be considered as specified in Figures 5.6-1c through 5.6-1e.

c. Fuel placed in Region II of the spent fuel storage racks shall be placed in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:

1. Fuel placed in Region II shall meet the burnup and decay time requirements specified in Figure 5.6-1a or 5.6-1b.

2. The reactivity effect of CEAs placed in fuel assemblies may be considered.

3. The reactivity equivalencing effects of burnable absorbers may be considered.

ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.11 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Movable Control Assemblies – CEA Position
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
~~Specification 3.2.2~~	~~Total Planar Radial Peaking Factors – F^T_{xy}~~
Specification 3.2.3	Total Integrated Radial Peaking Factors – F_r^T
Specification 3.2.5	DNB Parameters ~~– Axial Shape Index~~
Specification 3.9.1	Refueling Operations – Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in the following documents or any approved Revisions and Supplements thereto:

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary).

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. CENPD-199-P, Rev. 1-P-A, "C-E Setpoint Methodology: CE Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986.

4. CENPD-266-P-A, "The ROCS and DIT Computer Code for Nuclear Design," April 1983.

5. CENPD-275-P, Revision 1-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 1988.

6. CENPD-188-A, "HERMITE: A Multi-Dimensional Space – Time Kinetics Code for PWR Transients," July 1976.

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

46. CENPD-199-P, Rev. 1-P-A, Supplement 2-P-A, "CE Setpoint Methodology," June 1998.

47. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

48. CEN-396(L)-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KG for St. Lucie Unit 2," November 1989 (NRC SER dated October 18, 1991, Letter J.A. Norris (NRC) to J.H. Goldberg (FPL), TAC No. 75947).

49. CENPD-269-P, Rev. 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984.

50. CEN-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2," December 1984 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

51. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

52. CENPD-140-A, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," June 1976.

53. CEN-365(L), "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis," June 1988 (NRC SER dated March 13, 1989, Letter J.A. Norris (NRC) to W.F. Conway (FPL), TAC No. 69325).

54. DP-456, F.M. Stern (CE) to E. Case (NRC), dated August 19, 1974, Appendix 6B to CESSAR System 80 PSAR (NRC SER, NUREG-75/112, Docket No. STN 50-470, "NRC SER – Standard Reference System, CESSAR System 80," December 1975).

55. CENPD-387-P-A, Revision 000, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

Insert B from the next page

Insert B

56. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001

57. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998

58. CENPD-404-P-A, Rev. 0, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001

59. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology'" July 1985

60. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification," February 1994

61. WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure," April 1989

62. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999

63. WCAP-14565-P-A~~,~~ Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003

64. 30% SGTP PLA Submittal and the SER

ATTACHMENT 4

INFORMATION ONLY COPIES OF

ST. LUCIE UNIT 2 MARKED-UP TECHNICAL SPECIFICATION

BASES AND CORE OPERATING LIMITS REPORT PAGES

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

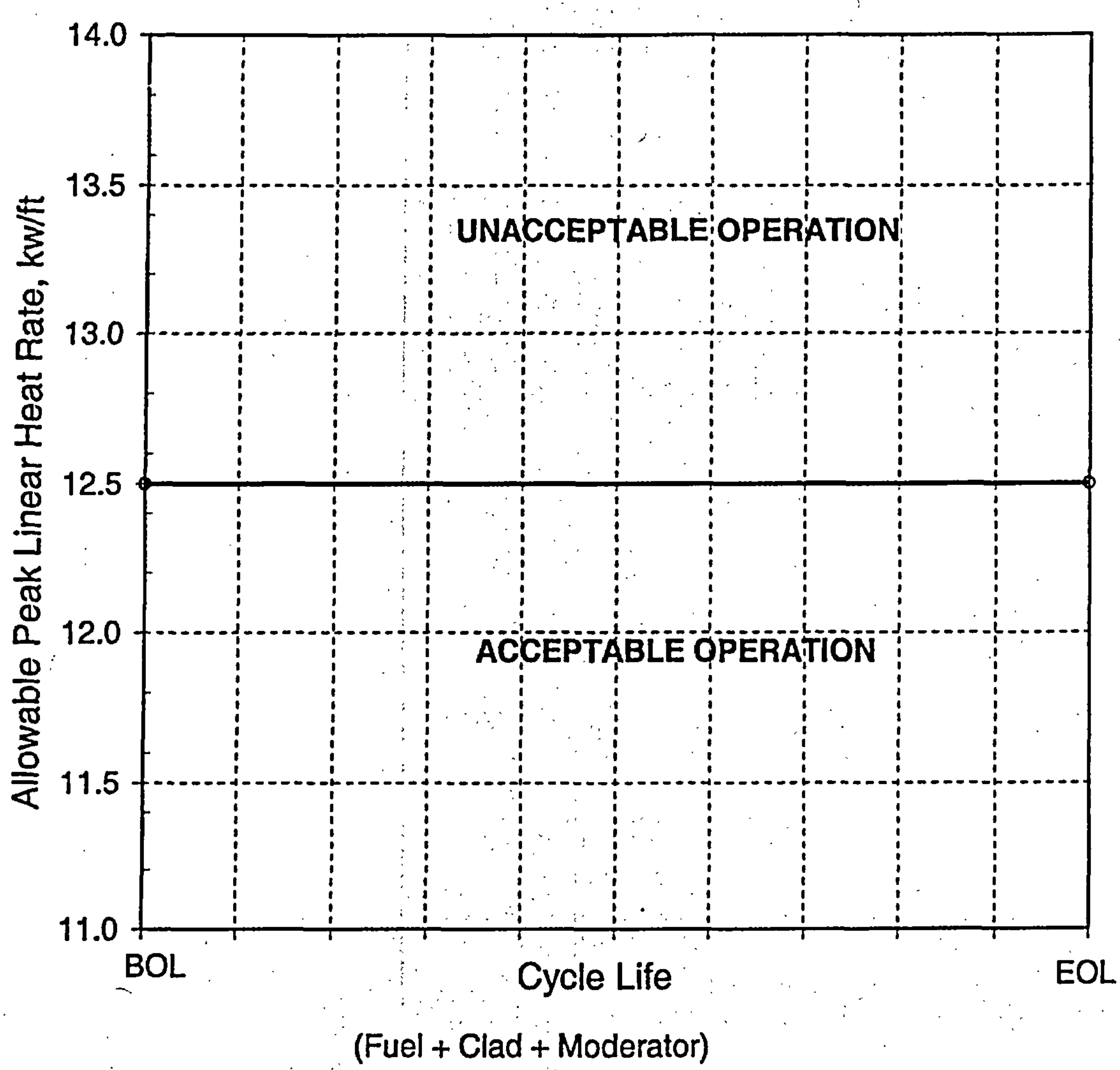
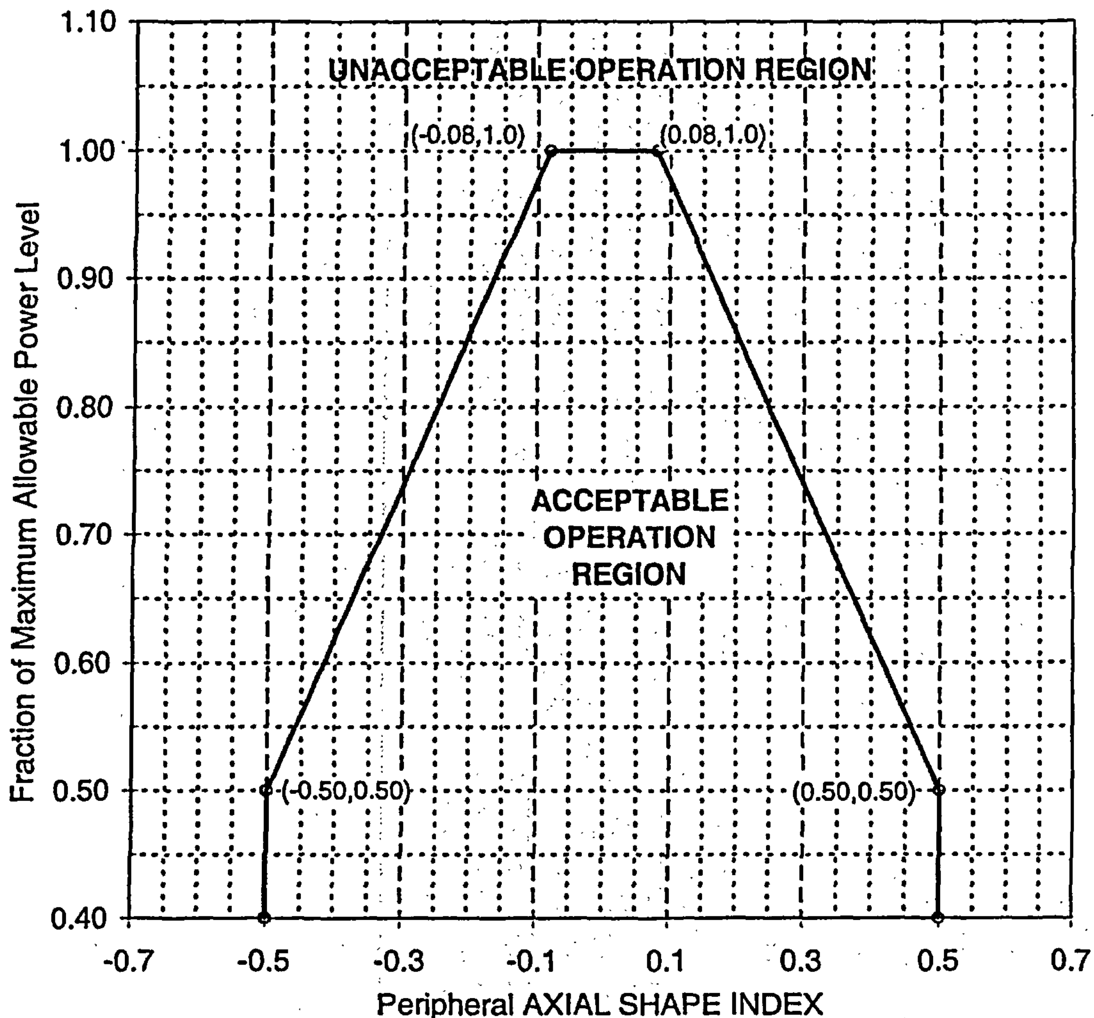


FIGURE 3.2-1
Allowable Peak Linear Heat Rate vs. Burnup



(Not Applicable Below 40% Power)

FIGURE 3.2-2
AXIAL SHAPE INDEX vs. Maximum Allowable Power Level

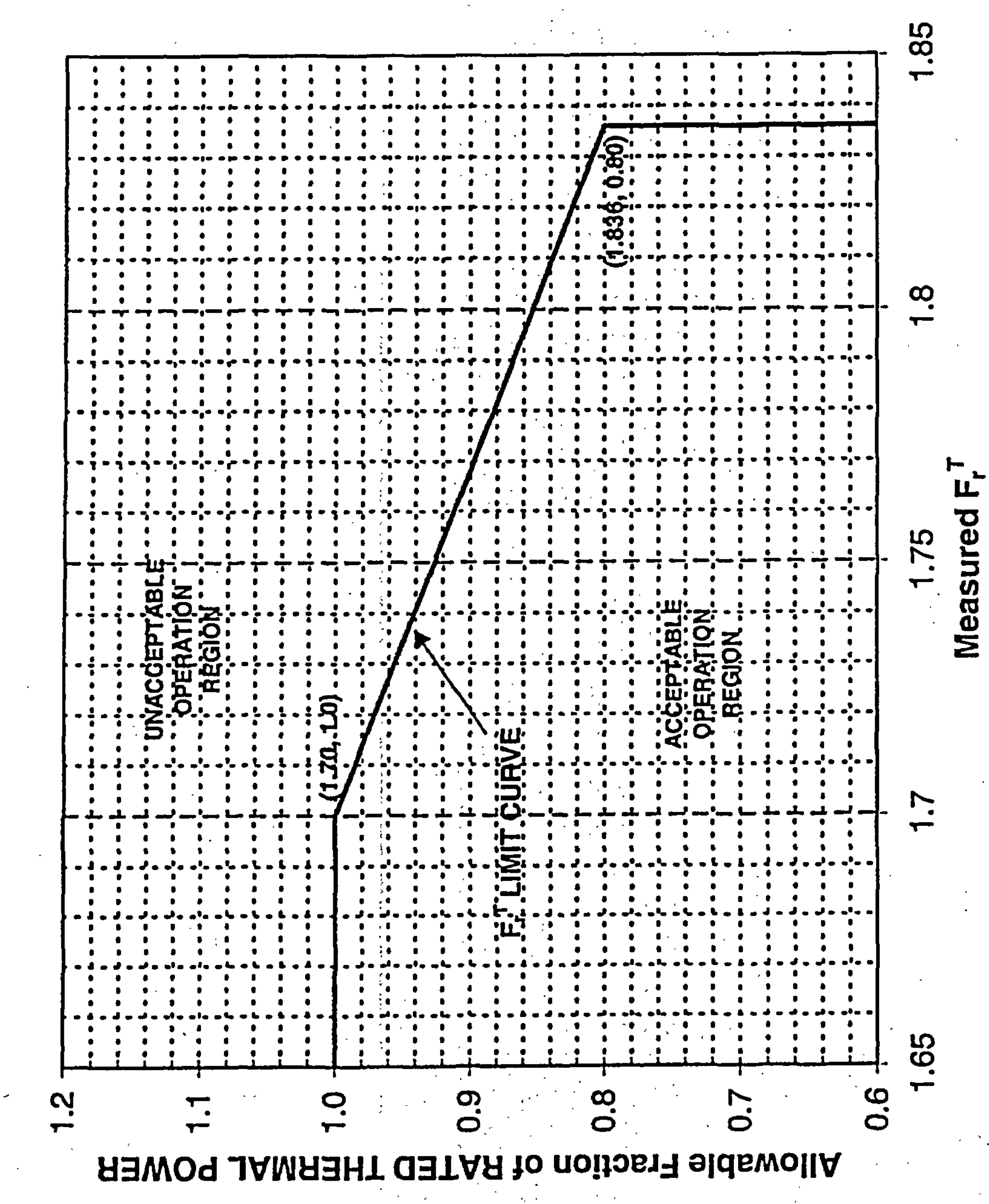


FIGURE 3.2-3
Allowable Combinations of THERMAL POWER and F_r^T

Table 3.2-2

DNB MARGIN LIMITS

PARAMETER	FOUR REACTOR COOLANT PUMPS OPERATING
Cold Leg Temperature (narrow Range)	$535^{0}F^{**} \leq T \leq 549^{0}F$
Pressurizer Pressure	2225 psia* $\leq P_{PZR} \leq$ 2350 psia**
Reactor Coolant Flow Rate	≥ 335,000 gpm
AXIAL SHAPE INDEX	Within the limits specified in Figure 3.2-4

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

** Applicable only if power level ≥ 70% of RATED THERMAL POWER.

TABLE 3.2-3. Load Follow W(Z) Factors at 150 MWD/MTU as a Function of Core Height

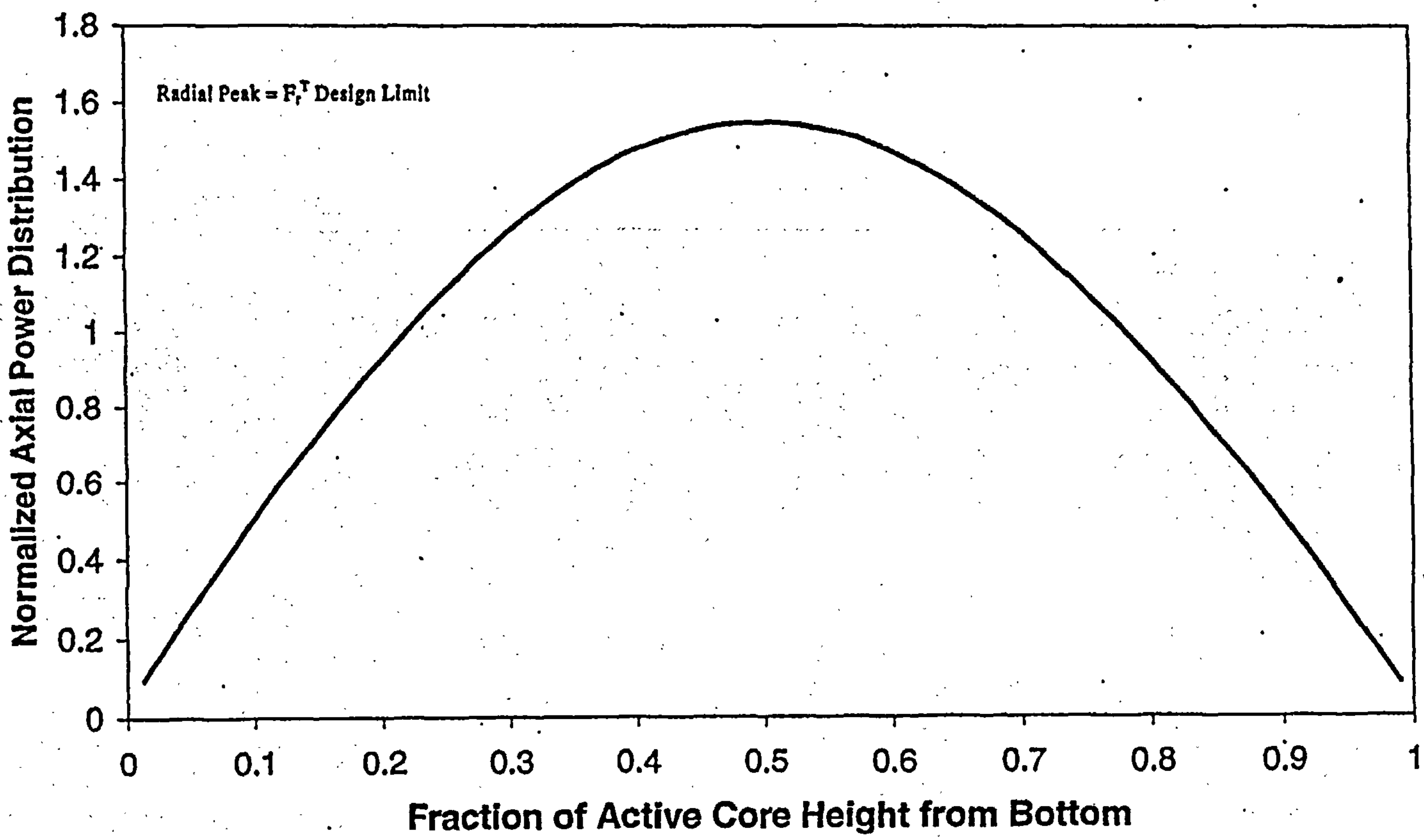
HEIGHT (FEET)	150 MWD/MTU W(Z)	HEIGHT (FEET)	150 MWD/MTU W(Z)
*0.0000		6.0000	
*0.2000		6.2000	
*0.4000		6.4000	
*0.6000		6.6000	
*0.8000		6.8000	
*1.0000		7.0000	
*1.2000		7.2000	
*1.4000		7.4000	
*1.6000		7.6000	
*1.8000		7.8000	
2.0000		8.0000	
2.2000		8.2000	
2.4000		8.4000	
2.6000		8.6000	
2.8000		8.8000	
3.0000		9.0000	
3.2000		9.2000	
3.4000		9.4000	
3.6000		9.6000	
3.8000		9.8000	
4.0000		10.0000	
4.2000		*10.2000	
4.4000		*10.4000	
4.6000		*10.6000	
4.8000		*10.8000	
5.0000		*11.0000	
5.2000		*11.2000	
5.4000		*11.4000	
5.6000		*11.6000	
5.8000		*11.8000	

NOTE: W(z) values are calculated on a cycle-specific basis. A range of burnup tables is considered.

* Top and Bottom 15% excluded

Figure B 2.1-1

AXIAL POWER DISTRIBUTIONS FOR THERMAL MARGIN SAFETY LIMITS



ATTACHMENT 5

ST. LUCIE UNIT 2 RETYPED TECHNICAL SPECIFICATION PAGES

The attached retyped pages reflect the currently issued version of the Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

TS Page

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3/4 2-2a
3/4 2-7
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3/4 2-15
3/4 10-2
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6-20d
6-20e

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

INDEX

DEFINITIONS (Continued)

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

INDEX

DESIGN FEATURES

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.36 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

VENTILATION EXHAUST TREATMENT SYSTEM

1.37 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). *Engineered Safety Features (ESF) atmospheric cleanup systems are not considered* to be VENTILATION EXHAUST TREATMENT SYSTEM components.

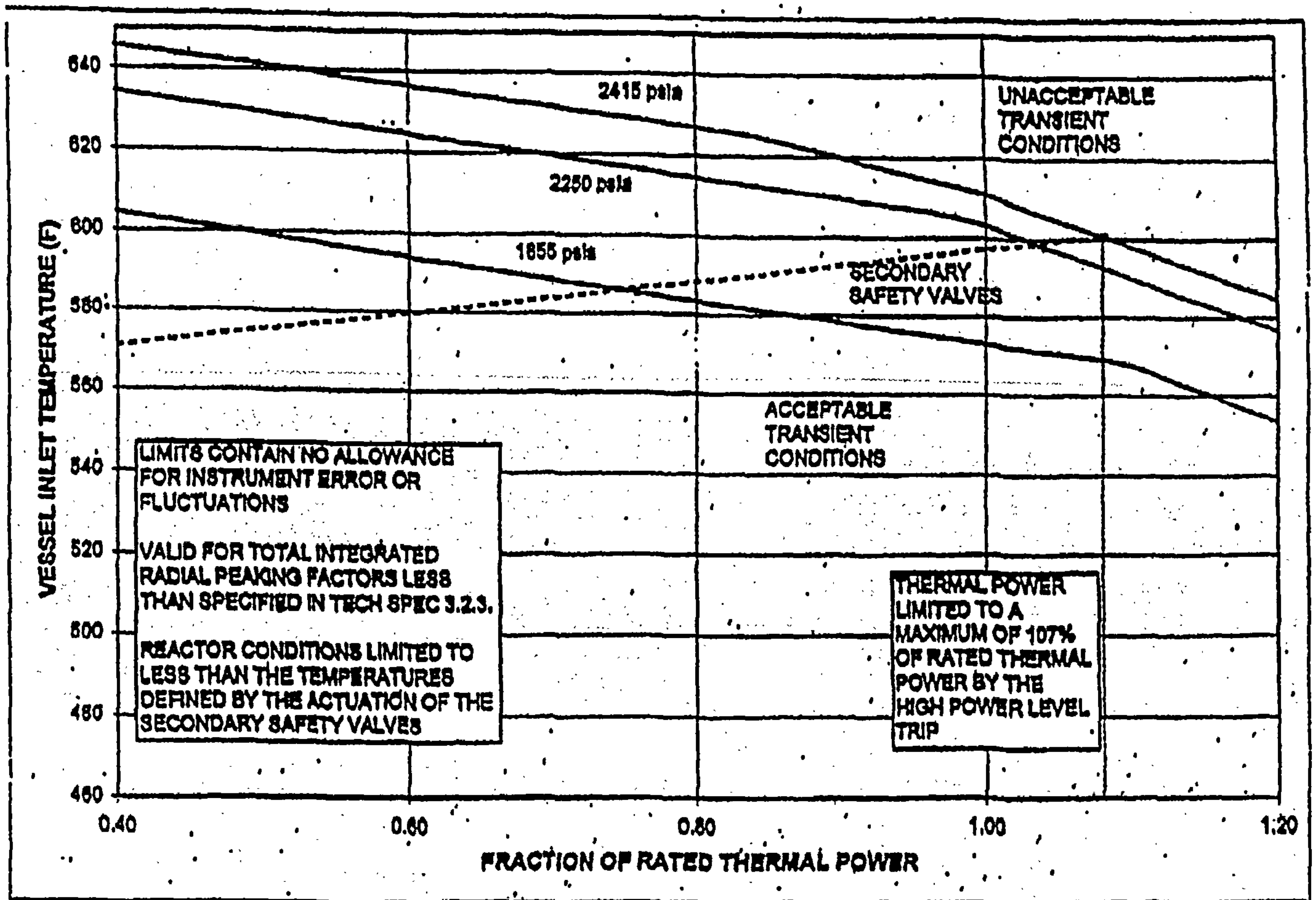

FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES
FOUR REACTOR COOLANT PUMPS OPERATING

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density – High(5) Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps – Low	$\geq$ 636 gpm**	$\geq$ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power – High(4)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute
14. Reactor Coolant Flow – Low(1)	$\geq$ 95.4% of design Reactor Coolant flow with four pumps operating*	$\geq$ 94.9% of design Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low(5)	$\geq$ 800 psig	$\geq$ 800 psig

* Design reactor coolant flow with four pumps operating is the minimum RCS flow specified in the COLR Table 3.2-2.

** 10-minute time delay after relay actuation.

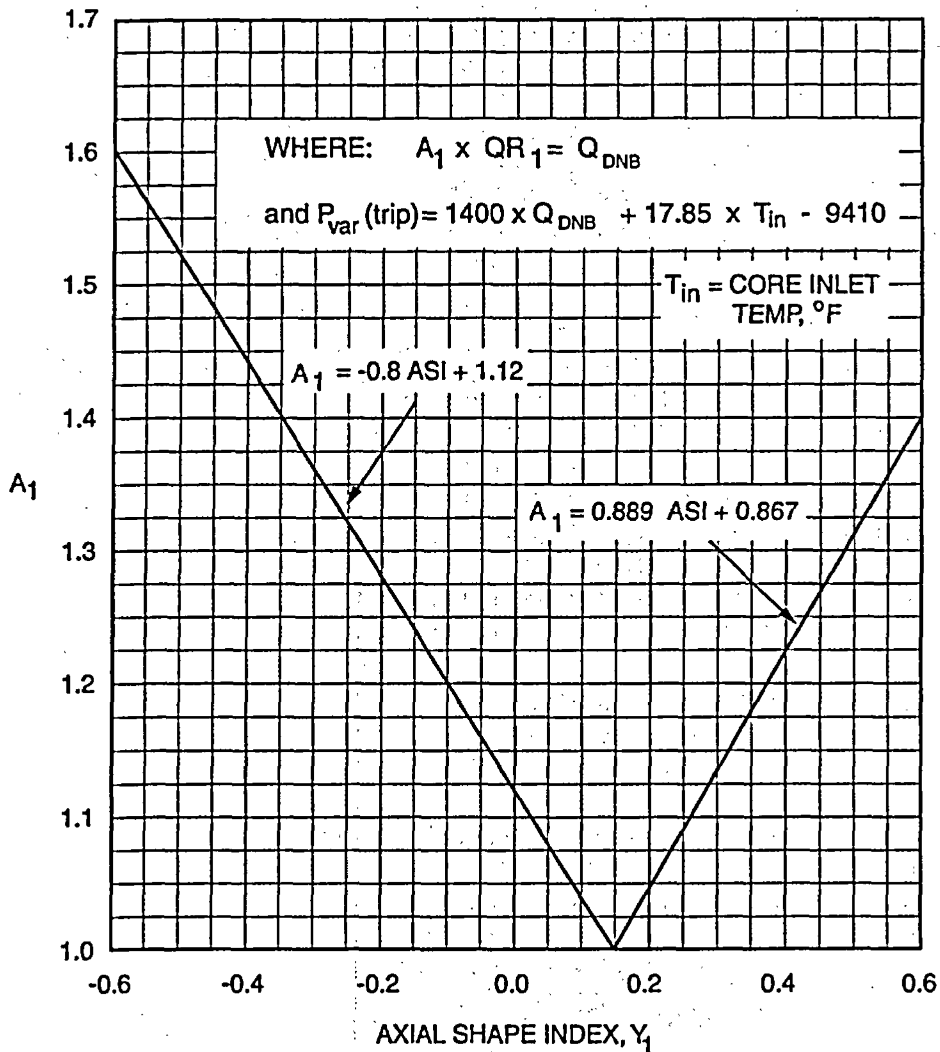

FIGURE 2.2-3

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT PART 1 (Y_1 Versus A_1)

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the COLR. The maximum upper limit shall be +5 pcm/°F at ≤ 70% of RATED THERMAL POWER, with a linear ramp from +5 pcm/°F at 70% of RATED THERMAL POWER to 0 pcm/°F at 100% RATED THERMAL POWER.

APPLICABILITY: MODES 1 AND 2*#.

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 800 ppm.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

---

* With K_{eff} greater than or equal to 1.0.

\# See Special Test Exception 3.10.2 and 3.10.5.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of COLR Figure 3.2-2, where 100% of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:

$$M \times N$$

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F_r^T curve of COLR Figure 3.2-3.

d. Verifying that the measured linear heat rate $LHR^M(z)$, obtained from a previous incore detector power distribution map, meets the following criteria:

$$LHR^M(z) \leq \frac{LHR}{W(z)}$$

W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. LHR and W(z) are specified in COLR Figure 3.2-1 and Table 3.2-3, respectively.

e. Operation is limited to the following:

1. The operation using excore detector monitoring system is limited to less ≤ 10% above the power level corresponding to the power level at which $LHR^M(z)$ is determined in Specification 4.2.1.3d.

2. Continuous operation using excore detector monitoring system is limited to 31 days from the time of the power distribution map used in Specification 4.2.1.3d.

f. The limit specified in Specification 4.2.1.3d above is not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive

2. Upper core region from 85 to 100%, inclusive

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.4 Incore Detector Monitoring System# – The incore detector monitoring system may be used for monitoring the linear rate by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on COLR Figure 3.2-1.

If incore system becomes inoperable, reduce power to M x N within 4 hours and monitor linear heat rate in accordance with Specification 4.2.1.3.

POWER DISTRIBUTION LIMITS

DELETED

POWER DISTRIBUTION LIMITS

DELETED

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

POWER DISTRIBUTION LIMITS

3/4.2.4 AZIMUTHAL POWER TILT – T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.03.

APPLICABILITY: MODE 1*.

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > .030 but ≤ 0.10, either correct the power tilt within 2 hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T) is within the limits of Specification 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T) is within the limits of Specification 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 7 days.

b. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER is > 75% of RATED THERMAL POWER.

* See Special Test Exception 3.10.2.

TABLE 3.2-2

DNB MARGIN

LIMITS

PARAMETER	**FOUR REACTOR COOLANT PUMPS OPERATING**
Cold Leg Temperature (Narrow Range)	Within the limits specified in the COLR Table 3.2-2
Pressurizer Pressure*	Within the limits specified in the COLR Table 3.2-2
Reactor Coolant Flow Rate	$\geq$ 335,000 gpm and $\geq$ the limit specified in the COLR Table 3.2-2
AXIAL SHAPE INDEX	COLR Figure 3.2-4

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

SPECIAL TEST EXCEPTIONS

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 may be suspended during performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended.

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO clad fuel rods and/or poison rods, with fuel rods having an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 91 full-length control element assemblies and no part-length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel pool and spent fuel storage racks shall be maintained with:

1. A k_{eff} equivalent to less than 1.0 when flooded with unborated water, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

2. A k_{eff} equivalent to less than or equal to 0.95 when flooded with water containing 520 ppm boron, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

3. A nominal 8.96 inch center-to-center distance between fuel assemblies placed in the storage racks.

b. Fuel placed in Region I of the spent fuel storage racks shall be stored in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:

1. Fresh fuel shall have a nominal average U-235 enrichment of less than or equal to 4.5 weight percent.

2. The reactivity effect of CEAs placed in fuel assemblies may be considered.

3. The reactivity equivalencing effects of burnable absorbers may be considered.

4. The reactivity effects of fuel assembly burnup and decay time may be considered as specified in Figures 5.6-1c through 5.6-1e.

c. Fuel placed in Region II of the spent fuel storage racks shall be placed in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:

1. Fuel placed in Region II shall meet the burnup and decay time requirements specified in Figure 5.6-1a or 5.6-1b.

2. The reactivity effect of CEAs placed in fuel assemblies may be considered.

3. The reactivity equivalencing effects of burnable absorbers may be considered.

ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.11 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Movable Control Assemblies – CEA Position
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factors – F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations – Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in the following documents or any approved Revisions and Supplements thereto:

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary).

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. CENPD-199-P, Rev. 1-P-A, "C-E Setpoint Methodology: CE Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986.

4. CENPD-266-P-A, "The ROCS and DIT Computer Code for Nuclear Design," April 1983.

5. CENPD-275-P, Revision 1-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 1988.

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ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

46. CENPD-199-P, Rev. 1-P-A, Supplement 2-P-A, "CE Setpoint Methodology," June 1998.

47. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

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57. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

58. CENPD-404-P-A, Rev. 0, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.

59. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

60. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification," February 1994.

ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

61. WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure," April1989.

62. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

63. WCAP-14565-P-A, Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

64. 30% SGTP PLA Submittal and the SER.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

(continued on page 6-21)

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

ATTACHMENT 6

WESTINGHOUSE LICENSING REPORT

ST. LUCIE UNIT 2 30-PERCENT STEAM GENERATOR

TUBE PLUGGING AND WCAP-9272 RELOAD METHODOLOGY

TRANSITION PROJECT

Attachment 7 contains 10 CFR 2.790(a)(4) Proprietary Information

October 2003

St. Lucie Unit 2 30-Percent Steam Generator Tube Plugging and WCAP-9272 Reload Methodology Transition Project

Licensing Report

St. Lucie Unit 2
30-Percent Steam Generator Tube Plugging and WCAP-9272 Reload Methodology Transition Project

Licensing Report

J. J. Akers
Nuclear Fuels Division

Francis Scapellato
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October 2003

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ACRONYMS

ADV	atmospheric dump valves
AFW	auxiliary feedwater
ANS	American Nuclear Society
AOO	anticipated operational occurrence
AOR	analysis of record
ASGT	asymmetric steam generator transient
ASI	axial shape index
ASME	American Society of Mechanical Engineers
BDAS	boron dilution alarm system
BOC	beginnning of cycle
BOL	beginning of life
BWR	boiling water reactor
CAW	customer application for withholding
CCW	component cooling water
CE	Combustion Engineering
CEA	control element assembly
CEDM	control element drive mechanism
CFC	containment fan coolers
CFR	Code of Federal Regulations
CLOF	complete loss of flow
COLR	core operating limits report
CS	containment spray
CVCS	chemical and volume control system
DEDLS	double-ended discharge and suction leg slot
DEG/PD	double-ended guillotine/pump discharge
DEHLS	double-ended hot leg slot
DER	double-ended rupture
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DPC	Doppler power coefficient
ECCS	emergency core cooling system
EDG	emergency diesel generator
EDMS	excore detector monitoring system
EFW	emergency feedwater
EM	evaluation model
EOC	end of cycle
EOL	end of life
EPRI	Electric Power Rersearch Institute
EQ	equipment qualification
ESCU	extended statistical combination of uncertainties

ACRONYMS (cont.)

FON	fraction of nominal
FPL	Florida Power and Light
GDC	General Design Criteria
HFP	hot full power
HPPT	high pressurizer pressure trip
HPSI	high pressure safey injection
HSGDP	high steam generator differential pressure
HZP	hot zero power
ICW	intake cooling water
IDMS	incore detector monitoring system
LBLOCA	large-break loss of coolant accident
LCO	limiting condition of operation
LHR	linear heat rate
LOCA	loss of coolant accident
LOCV	loss of condenser vacuum
LONF	loss of normal feedwater
LOOP	loss of offsite power
LP	low pressure
LPSI	low pressure safety injection
MDC	moderator density coefficient
MFIV	main feedwater isolation valve
MSIV	main steam isolation valve
MSIS	main steamline isolation signal
MSS	main steam system
MSLB	main steamline break
MSSV	main steam safety valve
MTC	moderator temperature coefficient
NCLO	no clad lift-off
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OD	outer diameter
PAC	plant analysis checklist
PCWG	Performance Capabilities Working Group
PHLA	pressurizer high level alarm
PMTC	positive moderator temperature coefficient

ACRONYMS (cont.)

PSV	pressurizer safety valve
PWR	pressurized water reactor
RAOC	relaxed axial offset control
RAS	recirculation actuation signal
RCP	reactor coolant pump
RCS	reactor coolant system
RPS	reactor protection system
RSAC	reload safety analysis checklist
RSE	reload safety evaluation
RTD	resistance thermal detector
RTDP	Revised Thermal Design Procedure
RWT	refueling water tank
SAFDL	specified acceptable fuel design limit
SAL	safety analyses limit
SBLOCA	small break loss of coolant accident
SCS	shutdown cooling system
SDC	shutdown cooling
SER	safety evaluation report
SG	steam generator
SGTP	steam generator tube plugging
SGTR	steam generator tube rupture
SI	safety injection
SIAS	safety injection actuation signal
SIS	safety injection system
SIT	safety injection tank
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SSFI	safety system functional inspection
STDP	Standard Thermal Design Procedure
SV	safety valve
TDF	thermal design flow
T/H	thermal-hydraulic
TM	thermal margin
TM/LP	thermal margin/low pressure
UFSAR	updated final safety analysis report
VHP	variable high power
VHPT	variable high power trip

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1 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

To address expected increases in tube plugging for the current steam generators, analyses and evaluations have been performed that support the operation of St. Lucie Unit 2 at 100% Rated Thermal Power (2700 MWt nuclear steam supply system (NSSS) Power), with the following conditions:

- Up to 30% of the steam generator tubes removed from service (2520 tubes/steam generator)
- Up to 7% Steam generator tube plugging (SGTP) asymmetry (up to 600 tubes),
- A reduction in the technical specification required minimum reactor coolant system (RCS) flow from the current value of 355,000 gpm to 335,000 gpm

The analyses are expected to be implemented for St. Lucie Unit 2 Cycle 15.

A parallel goal is to improve and streamline the reload analysis process to result in a reduction of effort related to the cycle-specific physics calculations to be performed by Florida Power & Light (FPL). To streamline the reload process, St. Lucie Unit 2 will transition to the Westinghouse Reload Methodology (WCAP-9272-P-A/WCAP-9273-NP-A; Reference 2). The proposed transition to this methodology will be implemented in combination with a 30% SGTP analysis for Cycle 15. FPL is familiar with the WCAP-9272 reload methodology (which includes the reload safety analysis checklist, or RSAC), which is currently in use at both Turkey Point units. The transition will establish a typical reload interface with FPL as designer, using the same core design methods currently at use for St. Lucie Units 1 & 2 and the Turkey point units, while maintaining reasonable margins. Where appropriate, the transition will include a transition to analysis methods typically associated with current era application of the WCAP-9272 reload methodology.

This report summarizes the detailed evaluations and analyses that were performed to confirm the acceptable use of these features for St. Lucie Unit 2 operations. Sections 2.0 through 5.0 of this report provide the results of the mechanical, nuclear, thermal-hydraulic (T/H), and accident analyses, respectively. Appendix A includes licensing information on some of the first-time application of some methods to St. Lucie Unit 2, and supplements the information found elsewhere in the report. Conditional requirements identified in the Nuclear Regulatory Commission (NRC) Safety Evaluation Reports (SERs) for methods and features not previously applied to St. Lucie Unit 2 are discussed in Appendix B.

This report serves as a reference evaluation and analysis report for the region-by-region reload transition to subsequent cores (Cycle 15 and beyond) containing the features described in Section 1.2. Thus, this report will be used as a basic reference document in support of future St. Lucie Unit 2 Reload Safety Evaluations (RSEs) for fuel reloads. Additional references that will be applicable for Cycle 15 will be the technical specifications with changes proposed in this submittal, and the UFSAR with changes to be incorporated consistent with the safety analyses described in this report.

For the analyses, key safety parameters have been chosen to maximize the applicability of the results for future reload cycle evaluations which will be performed utilizing the Westinghouse standard reload methodology (Reference 2). The objective of subsequent cycle-specific RSEs will be to verify that the applicable safety limits are not exceeded based on the reference analyses currently in the UFSAR (Reference 1) or as established in this report. Additional discussion on the selection of values for the key safety parameters is included in Section 3.4.

1.2 CYCLE 15 FEATURES AND CONDITIONS

While the main focus of the engineering analyses and evaluations was an increase in SGTP to 30% with a corresponding reduction in RCS flow, there are also other considerations that are being incorporated for Cycle 15, specifically:

- WCAP-9272 reload methodology implementation
- Relaxed Axial Offset Control (RAOC) implementation
- A single change in fuel design to implement ZIRLO™ cladding

1.2.1 Reload Methods

The WCAP-9272 Reload Methodology is an NRC-accepted means to evaluate reload safety. The underlying philosophy is independent of the details of the plant design or the application tools and methods, as long as these codes and methods are appropriately applied and technically sound. There is no fundamental constraint in the application of the WCAP-9272 Reload Methodology, per se, but attention is focused on ensuring the technically adequate modeling of the safety analyses and reload design.

The WCAP-9272 Reload Methodology represents a philosophy for the review of acceptability of plant reload designs. The philosophy is based on four considerations:

- A bounding approach is used.

- The acceptability review is a streamlined process based on the review of key parameters to which the analyses are known to be sensitive and which have a potential to change as a due of cycle-to-cycle fuel redistribution.

- Safety analyses and reload design evaluations are based on NRC-approved or accepted codes and methods, wherever such review is required.

- The review process is based on a well-defined procedure.

All elements of this philosophy will be addressed in the application of this process to St. Lucie Unit 2. The use of methods that differ from those normally applied in the WCAP-9272 Reload Methodology does not affect this philosophy, but may require that some adjustments be made in its application.

The process of identifying bounding values for the key reload parameters will be maintained by establishing a set of "baseline neutronics" for use in the safety analyses. These values and the key parameters themselves may be adjusted to accurately model the unique features of the St. Lucie Unit 2 plant, trips, and technical specifications. The values established for the baseline neutronics are chosen to be sufficiently conservative to reasonably preclude violations in the reload evaluation process without being overly conservative, resulting in challenges to analysis margins. For the key parameters, parameter limits have been established based on recent past operation of St. Lucie Unit 2 to identify representative parameter values. From these representative values, limit values for use in the safety analyses can be established with due consideration of the existing analysis assumptions and extensive plant and design experience, and accounting for changes in SGTP, minimum technical specification flow, and cladding material transition.

For reload applications, the baseline neutronics assumed in the safety analyses will be reviewed against the values obtained for the cycle-specific design for each cycle. Violations will be addressed via evaluations or reanalysis of the affected safety analyses to establish new limits, or through reload redesign to provide margin to the existing limits, or both.

1.2.2 Analysis Methods

RCS flow has a significant impact on the departure from nucleate boiling (DNB) ratio calculations. Reducing the technical specification RCS flow requirement would, therefore, reduce the available analysis margins to reactor protection system (RPS) setpoints and accident analysis acceptance criteria. The projected reduction in RCS flow is approximately 6%, which is sufficiently large to preclude evaluating the change solely with engineering judgment. Reanalysis of some of the licensing basis analyses will be required to show that the acceptance criteria are still met (departure from nucleate boiling ratio (DNBR), peak primary and secondary side pressure, etc.).

The first phase of this project is the development phase that includes the following basic types of tasks:

- Adjustment of the codes, models, and methods to facilitate modeling of St. Lucie Unit 2 and development of necessary WCAP-9272 reload materials (procedures, RSAC, etc.), and

- Assessments of margins (qualitative and/or quantitative, with a focus on the unique activities, or areas of known margin challenges), and assessment of licensing risks.

The basic operation of the Westinghouse and Combustion Engineering (CE) plants are similar. The dissimilarities can be addressed within the safety analyses and methodologies and will support the WCAP-9272 Reload Methodology applicability and philosophy.

The underlying philosophy in the development phase has been to identify those areas which are likely to create margin issues and/or create licensing risk; those areas where current and future methods are most different; and those areas in which differences in the plant design, technical specifications or operations represent new territory for the WCAP-9272 technologies.

The proposed efforts will be based on the following technologies for reload-related activities:

Discipline	Proposed Technology	Prior Technology	Comments
Reload Methodology	WCAP-9272/9273 (Reference 2)	Formerly PAC process	Streamlined, but effectively similar bounding approach
LBLOCA	99 Evaluation Model (99EM) (Reference 3)	85EM	Upgrade of prior technology
SBLOCA	Supplement 2 Model (S2M) (Reference 4)	S1M	Upgrade of prior technology
Post-LOCA	Consistent with current licensing basis	Consistent with current licensing basis	No change
Transient Analysis	RETRAN* (Reference 5)	CESEC	
Transient Analysis Methods	As described, TM/LP will be modeled explicitly	Consistent with CESEC / PAC	Various – must be confirmed on merit as applicable to both technology and application
Thermal-Hydraulics	VIPRE-W (References 6 through 8)	TORC	
DNB Correlations	ABB-NV (Reference 9)	CE-1	Upgrade of technology previously approved for this plant
Thermal Design Procedure	RTDP (Reference 10)	ESCU	Change to an effectively similar method
Mechanical Design	Consistent with current licensing basis	Consistent with current licensing basis	No change
Fuel Performance	FATES-3B Reference 11 through 15)	FATES-3B	Minor changes to accommodate automation
Fuel Performance Methods	Consistent with current licensing basis, augmented with high-duty drivers (more explicit corrosion calculation)	Consistent with current licensing basis	
Nuclear Design	ALPHA/PHOENIX/ANC (Reference 16)	ALPHA/PHOENIX/ ANC	No change
Nuclear Design Methods	Standard design methods and adjustments for non-standard technologies (LOCA and FP/MD)	Standard design methods and adjustments for nonstandard technologies	Minor refinements – standard METCOM interfaces restored for transients and T/H)
Power Distribution Confirmation Methods	RAOC (Reference 17)	Xenon Swing	Similar approaches
* CESEC has been used for the SGTR event			

The general approach is to use those tools and methods traditionally associated with the WCAP-9272 Reload Methodology. Key exceptions are noted for loss-of-coolant accident (LOCA), fuel performance, mechanical design, and DNB correlations. Implementation of ZIRLO™ cladding will be addressed in accordance with the approved methods of Reference 18.

Code utilization and applicability will be described. All code packages planned for use in this project have been previously licensed or will be re-licensed as necessary with the NRC. NRC SER conditional requirements for those applications not previously applied to St. Lucie Unit 2 are addressed in Appendix B. The selection of code packages is explained and justified. The selection of code packages and analysis methodology supports the reload methodology applicability.

LOCA models share initialization and rod performance routines with the related fuel performance tools. Thus the LOCA evaluation model (EM) and fuel performance model are linked. Severing this link is possible, but would likely result in intensive license activities for both disciplines. Having identified recommended EMs for LOCA, the associated fuel performance methods follow. Due to the close relationship between fuel performance modeling and mechanical design, the associated mechanical design methods also follow.

While this choice creates a mismatch between fuel performance models and transient analysis models, the link between these models is much looser owing to the less detailed modeling of fuel rod characteristics in the non-LOCA calculations.

Owing to differences in the plant features, trips and technical specifications of St. Lucie Unit 2, conventional wisdom associated with the methods for the various non-LOCA transients applicable to Westinghouse plants may not apply. Where such limited cases have been identified, necessary adjustments in methods (to provide a correct technical basis), have been identified.

The ANC/PHOENIX code package is currently in use for St. Lucie Unit 2 (i.e., ANC/PHOENIX is applicable to modeling CE core designs). Thus, there are no differences in methodology for the nuclear design.

As with WCAP-9272, the overall approach of the RAOC methodology described in WCAP-10216-P-A is effectively equivalent to the approach currently used.

VIPRE-01 (VIPRE) has been selected as the thermal-hydraulic subchannel analysis code. VIPRE is a sophisticated, robust thermal-hydraulic subchannel analysis code that can accurately model various core/fuel configurations with numerous correlations. The ABB-NV & ABB-TV correlations have been inserted into the VIPRE code for application to the St. Lucie Unit 2 fuel design. The use of this correlation in the VIPRE model is currently being reviewed.

RETRAN-02 (RETRAN) has been selected as the transient analysis code. The currently licensed RETRAN model will be modified for the St. Lucie Unit 2 control and protection systems and plant equipment operations. These features do not require coding updates and can be accommodated through proper modeling changes using features already available in the approved RETRAN code. Use of RETRAN will support the reload methodology.

Some of the methods for the various analyses are separately documented in NRC-approved topical reports. These reports may identify specific assumptions, including "generically" determined limiting cases, which may not be technically appropriate for the St. Lucie Unit 2 features. Any changes necessary to provide an appropriate technical basis for the St. Lucie Unit 2 design are defined in the appropriate analysis subsection, or elsewhere in this report, for licensing consideration.

Fuel performance information for the non-LOCA analyses has been provided from FATES-3B for this application. Note that, unlike the LOCA relationship to fuel performance, the non-LOCA methods do not rely on the specifics of the fuel performance model. Rather, adequately conservative fuel temperatures, pressures and geometric information are taken from the fuel performance calculations to set up simplified modeling internal to the RETRAN (and related) code calculations. Therefore, the exact nature of the modeling is not paramount, but it is necessary to ensure that the values for the key fuel performance parameters are conservative for this application. On the basis of these arguments, it is believed that conservative nature of the fuel performance data input bases included in the analyses will be sufficient to address licensing considerations in this regard.

The 99EM for large break has been used for the St. Lucie Unit 2 transition to the Westinghouse Reload Safety Evaluation Methodology. The S2M Evaluation Model for Small Break has been used for the St. Lucie Unit 2 transition to the Westinghouse Reload Safety Evaluation Methodology. Post-LOCA modeling maintains the methods currently used for the St. Lucie Unit 2 licensing basis.

While the proposed LOCA methods for large-break LOCA (LBLOCA) and small-break LOCA (SBLOCA) (99EM LBLOCA EM and the S2M SBLOCA EM) are new in application for St. Lucie Unit 2, they have been generically approved by the NRC for CE plant applications. Use of these evaluation models does not invalidate the reload methodology since it will be used to confirm key safety parameters.

FATES-3B will remain the fuel performance code for the transition. The only change anticipated in the application of FATES-3B is the application for ZIRLO™ cladding (Reference 18). Use of FATES-3B does not invalidate the reload methodology.

1.3 PEAKING FACTORS

The full-power F_r peaking factor design limit is 1.70 (without measurement uncertainties), and full-power peak linear heat rate is 13.0 kw/ft. For the initial implementation, the peak linear heat rate limit is reduced to 12.5 kw/ft to satisfy the acceptance criteria of 10 CFR 50.46 for LBLOCA (see Section 5.2.3). Higher limits for peak linear heat rate may be addressed in future on a cycle-specific basis.

1.4 RTDP UNCERTAINTIES

With Cycle 15, St. Lucie Unit 2 will transition from the current extended statistical combination of uncertainties (ESCU) methods to the Westinghouse Revised Thermal Design Procedure (RTDP) for DNB analysis. The method of uncertainty analysis is discussed in Reference 10 and is the same regardless of whether the application to the safety analysis is RTDP or non-RTDP methodology. The uncertainty analysis statistically combines the individual uncertainties using the square root of the sum of the squares (SRSS) method. The analysis includes uncertainties for:

- The method of measurement
- The type of field device (that is, RTDs, transmitters, special test measurements)
- The calibration of the instrumentation

These uncertainties for temperature, pressure, power, and flow are then used in the development of the reactor core limits and the DNBR limits. The thermal margin/low pressure (TM/LP) reactor trip setpoints are then confirmed from the new core limits for use in the technical specifications.

1.5 PCWG PARAMETERS

The analysis basis for this program is based on those parameters specified in the Performance Capabilities Working Group (PCWG) parameter sheet. This PCWG parameter sheet is used by all the analysis groups in performing their analyses to ensure a consistent reference level for the analyses. A copy of the St. Lucie Unit 2 parameter sheet is provided in Table 1-1. Each parameter sheet provides four cases for analysis purposes: two cases at 549°F T_{inlet}, consistent with expected plant operating conditions, for 0% and 30% SGTP, and two cases at 535°F T_{inlet}, for use in LOCA and non-LOCA analyses that bound on assumed 535°F $\leq T_{inlet} \leq$ 549°F, for 0% and 30% SGTP. The 30% SGTP limit represents the maximum allowable tube plugging for any single steam generator. Refer to the footnotes in Table 1-1 for more details of the cases.

Table 1-2 provides flow data, based on the Table 1-1 data, used for the evaluation of various asymmetric plugging configurations for the non-LOCA analyses.

1.6 GENERAL ANALYSIS ASSUMPTIONS

Part of the re-analysis performed for this program is to update and confirm many of the assumptions and inputs used in the analyses. These new or revised assumptions and input parameters form the basis upon which the analyses are performed and ultimately establish the St. Lucie Unit 2 licensing basis (that is, technical specifications and UFSAR analysis of record). The process is started by Westinghouse documenting those assumptions and input parameters originally expected to be used in the analysis. FPL reviews the list and provides confirmation and/or revisions, as appropriate. Once concurrence is obtained, the assumptions and input parameters are documented as final values, which are used in the analysis.

In general, the only changes are those defined by the program, those required for accurate technical modeling, and those chosen to improve analysis margins.

The changes identified by the program are:

- Incorporation of ZIRLO cladding

- Increase in SGTP up to 30% of the steam generator tubes removed from service (2520 tubes/steam generator), with up to 7% SGTP asymmetry (up to 600 tubes)

- Reduced minimum technical specification flow.

There are no changes in data, per se, associated with the technical modeling, but differences from historical modeling may require additional detail or alternate processing of data. The collection and confirmation of these data follow the process outlined in the first paragraph in this subsection.

While not guaranteed, the goal of the project is to maintain existing margins. Since the transition of methods is expected to result in similar margins, assumptions employed in the current analyses were re-evaluated for relaxation as a means to accommodate the anticipated margin effects of reduced flow. Following is a summary of some of the key changes from the assumptions employed in the current safety analyses:

- Neutronics parameters will no longer bound potential designs that include Erbia burnable absorbers.
- Neutronics parameters will no longer bound potential designs that include 24-month cycle operation.
- The current full-power positive moderator temperature coefficient (PMTC) will no longer be assumed (see technical specification changes proposed).
- The RTD time constant for analysis will be reduced from 14 seconds to 8 seconds based on previous replacement and testing of hardware.

In addition to these assumptions, results have required reductions in COLR limits for peak linear heat rate and the linear heat rate limiting condition of operation (LCO), as discussed in Section 3.6.

1.7 CONCLUSIONS

The results of evaluation/analysis described in this report lead to the following conclusions:

1. The design and safety analysis results documented in this report show the core's capability to operate safely with the conditions that have been assumed for St. Lucie Unit 2 Cycle 15 and beyond (Chapters 3, 4, and 5):
    - Incorporation of ZIRLO cladding
    - Increase in SGTP up to 30% of the steam generator tubes removed from service (2520 tubes/steam generator), with up to 7% SGTP asymmetry (up to 600 tubes)
    - Reduced minimum technical specification flow to 335,000 gpm
2. This report establishes a reference upon which to base Westinghouse reload safety evaluations for future reloads with the methodology described in WCAP-9272/9273 (Chapters 2, 3, 4, and 5).

1.8 REFERENCES

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5. Huegel, D. S., et al., "RETRAN Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A (Proprietary), February 1999.

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16. Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A (Proprietary), June 1988.

17. Miller, R. W., et al., "WCAP-10216-P-A (Proprietary), "Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification," February 1994.

18. CENPD-404-P-A, Rev. 0 (Proprietary), "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.

Table 1-1 Performance Capability Parameters

OWNER UTILITY:	Florida Power and Light Company	Attachment to PCWG-03-23	
PLANT NAME:	St. Lucie (STL2)		
UNIT NUMBER:	2		
BASIC COMPONENTS			
Reactor Vessel, ID, in.	172.4	Isolation Valves	No
Core		Number of Loops	2
Number of Assemblies	217	Steam Generator	
Rod Array	16 x 16 CE[1]	Model	Model 67
Rod OD, in.	0.382	Shell Design Pressure, psia	1000
Number of Grids	10/assembly[8]	Reactor Coolant Pump	
Active Fuel Length, in.	136.7	Model/Weir	3543/Yes
Number of Control Rods, FL	91	Pump Motor, hp	6500
Internals Type	PSL2	Frequency, Hz	60

	30% SGTP Program			
THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3	Case 4
NSSS Power, %	100	100	100	100
MWt	2720	2720	2720	2720
10^6 BTU/hr	9,281	9,281	9,281	9,281
Reactor Power, MWt	2700	2700	2700	2700
10^6 BTU/hr	9,213	9,213	9,213	9,213
Thermal Design Flow, Loop gpm	167,500[4]	167,500[4]	167,500[4]	167,500[4]
Reactor 10^6 lb/hr	126.1	126.1	128.4	128.4
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	3.7	3.7	3.7	3.7
Reactor Coolant Temperature, °F				
Core Outlet	605.9	605.9	592.9	592.9
Vessel Outlet	604.0	604.0	590.9	590.9
Core Average	578.6	578.6	564.9	564.9
Vessel Average	576.5	576.5	563.0	563.0
Vessel/Core Inlet	549.0	549.0	535.0[6]	535.0[6]
Steam Generator Outlet	548.6[7]	548.6[7]	534.6[7]	534.6[7]
Steam Generator				
Steam Temperature, °F	526.2	513.2	511.7	498.6
Steam Pressure, psia	857[2]	766[2]	755[2]	672[2]
Steam Flow, 10^6 lb/hr total	11.87	11.83	11.82	11.79
Feed Temperature, °F	435	435	435	435
Moisture, % max.	0.25	0.25	0.25	0.25
Design FF, hr. sq. ft. °F/BTU	0.00017[3]	0.00017[3]	0.00017[3]	0.00017[3]
Tube Plugging, %	0	30	0	30
Zero Load Temperature, °F	532	532	532	532
HYDRAULIC DESIGN PARAMETERS				
Pump Design Point, Flow (gpm)/Head (ft.)	87,750/296.75			
Mechanical Design Flow, gpm	406,000			
Minimum Measured Flow, gpm/total	349,500[5]			

Notes:

1. Fuel Features: Includes OPTIN cladding (transitioning to ZIRLO cladding only).
2. 12 psi SG internal pressure drop incorporated.
3. This fouling factor was determined by adding some conservatism to a BE fouling factor based on plant data. It is less than the warranted FF of 0.0003624 hr. sq. ft. °F/Btu which is overly conservative.
4. TDF=167,500 gpm/steam generator loop (each SG contains 1 hot leg and 2 cold legs). This is equivalent to the Tech Spec minimum flow in CE terminology.
5. Flow measurement uncertainty is 14,500 gpm total. The difference between MMF and TDF is 14,500 gpm total.
6. Cases with low Tcold=535°F to be used for LOCA and non-LOCA analyses.
7. The difference between SG Outlet and Vessel Inlet Temperature is due to RCP net heat input.
8. CE Fuel Grid: 1 Guardian, 8 HID-1L, and 1 Top HID-1L.

Table 1-2	Performance Capability Parameters – Asymmetric Flow for Non-LOCA Transient Evaluations – St. Lucie Unit 2 (STL) – Reactor Core and RCS Loop Flow at Various SGTP Levels			
Reactor Vessel Tavg (°F)	**SGTP Level (%)**	**Loop 1 RCS Flow (gpm)**	**Loop 2 RCS Flow (gpm)**	**Reactor Core RCS Flow (Based on Core Tavg, gpm)**
576.5	30% (Symmetric)	173,215	173,215	348,341
576.5	30%/25% (Asymmetric)	172,736	178,664	353,120
576.5	25% (Symmetric)	178,161	178,161	357,893
576.5	30%/20% (Asymmetric)	172,304	183,510	357,362

2 MECHANICAL DESIGN FEATURES

2.1 INTRODUCTION AND SUMMARY

This section evaluates the mechanical design of the 16x16 Combustion Engineering (CE) HID-1L fuel design and its compatibility with the same fuel assembly design with ZIRLO™ cladding (refer to Table 2-1). The evaluation also covers the increase in steam generator tube plugging (SGTP) to 30% peak (asymmetric plugging of ~ 7%), and a reduced reactor coolant system (RCS) flow rate of 335,000 gpm. Since there is no change in the fuel assembly design, other than the transition to ZIRLO™, there will be no impacts to the fuel handling equipment or refueling equipment interfaces. The reactor internals interfaces have been evaluated for the reduced RCS flow rate. References in this section are made to CENPD-404-P-A, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs" (Reference 1), and the current St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR), Amendment 14 (Reference 2).

The only change in the 16x16 CE HID-1L fuel assembly design requiring licensing action is the transition to ZIRLO™ cladding.

2.1.1 Background

Licensing Background

Westinghouse developed an advanced zirconium-based alloy, known as ZIRLO™, to enhance fuel performance and achieve extended fuel discharge burnup. This alloy provides a significant improvement in fuel rod and fuel assembly component corrosion resistance and dimensional stability under irradiation. Fuel incorporating ZIRLO™ clad fuel rods was first introduced into a commercial reactor, North Anna Unit 1, beginning in 1987, in 2 demonstration assemblies, AM1 and AM2. A conditional licensing approval for use of this advanced alloy in these assemblies was given in a U.S. Nuclear Regulatory Commission (NRC) letter dated May 13, 1987. The U.S. NRC granted an exemption (Reference 3) from the provisions of 10 CFR 51.52 with respect to the use of the North Anna demonstration fuel assemblies with the advanced cladding material ZIRLO™. Use in a full commercial region began in 1991 in the V. C. Summer reactor. The first CE nuclear steam supply system (NSSS) reactor to use ZIRLO™ was Fort Calhoun. The second and third applications of ZIRLO™ fuel rod cladding were in Palo Verde and Calvert Cliffs.

Use of the ZIRLO™ alloy in fabrication of guide thimble tubes or in other assembly components, such as mid-span grids, does not require technical specification changes. Further, as a result of federal regulation changes (Reference 4) the implementation of ZIRLO™ clad fuel rods to currently-approved fuel discharge burnup levels is justifiable under 10 CFR 50.92 and requires no exemptions.

Chemical/Mechanical Properties

The chemical composition (refer to Table 2-1) of the cladding with the ZIRLO™ alloy in the St. Lucie Unit 2 Cycle 15 assemblies are similar to that of Zircaloy-4 or OPTIN except for slight reductions in the tin (Sn) and iron (Fe) content, the elimination of chromium (Cr), and the addition of a nominal amount of niobium (Nb). This composition change and associated materials structure refinements are responsible for the improved corrosion resistance as compared to that of the standard Zircaloy-4, improved Zircaloy-4, or OPTIN components. The physical and mechanical properties of ZIRLO™ are comparable or better than those of Zircaloy-4 or OPTIN. However, the temperature at which the metallurgical phase change occurs differs, as discussed in Reference 1.

2.1.2 Related Experience

Demonstration Programs

ZIRLO™ alloy corrosion resistance has been evaluated in long-term, out-of-pile tests over a wide range of temperatures (up to 680°F in water tests and up to 932°F in steam tests). Additional tests have also been conducted in higher than current primary chemistry lithiated water environments at temperatures of 680°F. This includes modified and coordinated pH chemistries. ZIRLO™ Lead Test Assemblies (LTAs) planned in Comanche Peak at higher pH values under an EPRI-sponsored program.

The improved corrosion resistance of ZIRLO™ cladding under irradiation has also been demonstrated. Beginning in the early 1970s in the BR-3 Test Reactor Demonstration Program, fuel rods containing cladding fabricated from ZIRLO™ alloy were irradiated at linear power levels of up to 17 kW/ft to rod average burnups of 68 GWD/MTU (i.e., peak pellet burnups of approximately 80 GWD/MTU). Post-irradiation examinations have demonstrated that the ZIRLO™ alloy exhibited a reduction in corrosion rate and improved dimensional stability as compared to Zircaloy-4 rods having similar power histories, which were irradiated as controls in the same assemblies.

Full-length ZIRLO™ clad fuel rods were fabricated for a second demonstration program at the North Anna Unit 1 commercial reactor, with operation beginning in June 1987. The first post-irradiation examination of the assemblies was completed after 18 months of irradiation to a rod average burnup of over 21 GWD/MTU (complete in February 1989). Visual and dimensional inspection during refueling showed no abnormalities.

Continued corrosion resistance and dimensional stability with no abnormalities was shown in the visual and dimensional inspections performed on the North Anna assemblies with demonstration rods operated for a second cycle to a lead rod burnup approaching 46 GWD/MTU (completed in January 1991) and a third cycle, completed in the fall of 1994, to a lead rod burnup of approximately 55 GWD/MTU. Cladding corrosion measurements showed that the reduced corrosion exhibited by the ZIRLO™ rods continues to be better than that anticipated on the basis of licensing-basis analyses. Eight ZIRLO™ rods from the original North Anna demonstration program were irradiated for a fourth cycle of operation and were discharged in March 2001 with burnups in the range of 66 to 70 GWD/MTU. Examinations of these rods (including visual, oxide, and growth) were made and showed good performance to high burnup with margin to the design criteria.

Commercial Applications

Since the first commercial region application of VANTAGE + fuel in the V. C. Summer Cycle 7 core, experience with ZIRLO™ alloy components has continued to increase. A summary of ZIRLO™ operating experience through March of 2003 is provided in Table 2-2. A total of 46 plants are using ZIRLO™ with an experience base of nearly 13,000 assemblies. The number of active and discharged

ZIRLO™ assemblies plotted as a function of burnup interval are shown in Figure 2-1. More than 1,000 assemblies have achieved burnups above 48 GWD/MTU. Examinations of ZIRLO™ high-burnup assemblies have continued to show good performance.

In addition to the extensive experience in Westinghouse plants, there are currently 316 assemblies of ZIRLO™ clad fuel at the Palo Verde units, and 180 assemblies of ZIRLO™ clad fuel at the Calvert Cliffs units, all in the first cycle of operation.

2.2 COMPATIBILITY OF FUEL ASSEMBLIES

The only modifications to the assembly design requiring licensing action consists of the use of ZIRLO™ clad fuel rods. Since ZIRLO™ has less growth than Zircaloy-4/OPTIN, by maintaining the fuel rod dimensions the same as the current fuel, there will be no concerns with respect to irradiated growth of the fuel rod within the assembly.

2.2.1 Fuel Rods

The 16x16 CE HID-1L fuel rod will remain unchanged other than the cladding material will be ZIRLO™, which enhances fuel performance.

A factor that needs to be considered in looking at the pellet-to-cladding gap is the material of the cladding. The 16x16 CE HID-1L fuel product will use ZIRLO™ cladding versus the current Zircaloy-4/OPTIN cladding. The resultant effect on pellet-to-cladding contact due to the difference in the creep rate of ZIRLO™ compared to Zircaloy-4/OPTIN is negligible.

2.2.2 Grid Assemblies

The bottom Inconel (non-mixing vane) grid of the 16x16 CE HID-1L fuel assemblies is the GUARDIAN grid, which remains unchanged from the current design.

The 16x16 CE HID-1L spacer grids will remain Zircaloy-4/OPTIN.

2.2.3 Guide Thimble Tubes

The 16x16 CE HID-1L guide thimbles will remain unchanged. The current guide thimble tubes are fabricated from Zircaloy-4.

2.2.4 Upper and Lower End Fittings

The upper and lower end fittings of the 16x16 CE HID-1L will remain unchanged.

2.3 MECHANICAL PERFORMANCE

Material changes associated with the 16x16 CE HID-1L with ZIRLO™ cladding does not influence the fuel assembly structural characteristics that were determined by prior mechanical testing (Reference 1). Therefore, the 16x16 CE HID-1L fuel assembly, with expected structural behavior and projected performance, will meet design requirements throughout the fuel's life. The effect of a small increase in temperature due to reduced flow on the core non-heat flux structures has been evaluated. The effects are neglible compared to measured oxide thicknesses from testing.

Another effect of the reduction in of RCS flow relates to the potential for flow-induced vibration. As part of the initial assembly development, the 16x16 CE HID-1L assembly has been acceptably tested in a spectrum of flow rates that encompasses the current and anticipated flow rates that will be experienced by St. Lucie Unit 2.

2.4 FUEL ROD PERFORMANCE

Fuel rod performance for 16x16 CE HID-1L fuel is identical to the 16x16 CE HID-1L fuel in use at the Palo Verde Units, which has previously been shown to satisfy the NRC Standard Review Plan (SRP), Section 4.2, fuel rod design bases on a region-by-region basis. The design bases for Westinghouse 16x16 CE HID-1L fuel assembly with ZIRLO™ cladding are discussed in Reference 1.

Fuel rod design evaluations for the 16x16 CE HID-1L fuel were performed using NRC-approved models and design criteria methods (References 5 through 11) to demonstrate that all fuel rod design criteria are satisfied.

The fuel rod design criteria given below are verified by evaluating the predicted performance of the limiting fuel rod, defined as the rod which gives the minimum margin to the design limit. In general, no single rod is limiting with respect to all the design criteria. Generic evaluations have identified which rods are most likely to be limiting for each criterion, and exhaustive screening of fuel rod power histories to determine the limiting rod is typically not required.

The NRC-approved FATES3B model (References 5, 6, 7, and 11) for in-reactor behavior is used to calculate the fuel rod performance over its irradiation history. FATES3B is the principal design tool for evaluating fuel rod performance. FATES3B iteratively calculates the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power.

2.4.1 Fuel Rod Design Criteria

The criteria pertinent to the fuel rod design are as follows:

- Maximum internal hot gas pressure
- Excessive fuel rod DNB propagation
- Fuel rod stress
- Fuel rod strain
- Maximum fuel temperature
- Fuel rod fatigue damage
- Cladding creep collapse
- Shoulder gap
- Seismic and LOCA loads

The specific assumptions used in the verification of these criteria for the St. Lucie Unit 2 fuel include:

- St. Lucie Unit 2 30% SGTP and associated reduced RCS flow rate of 335,000 gpm
- ZIRLO™ cladding

- Fuel rod duty (steady-state powers, fuel rod axial power shapes, etc.)

Each of these key fuel rod design criteria have been evaluated for application of the 16x16 CE HID-1L fuel assembly with ZIRLO™ cladding in St. Lucie Unit 2. Based on these evaluations, it is concluded that each design criterion can be satisfied through transition cycles to a full core of the 16x16 CE HID-1L with ZIRLO™ cladding design. The design criteria are described in more detail below.

Maximum Internal Gas Pressure

Criterion: The fuel rod internal hot gas pressure shall not exceed the critical maximum pressure determined to cause an outward cladding creep rate that is in excess of the fuel radial growth rate anywhere locally along the entire active fuel length of the fuel rod.

Evaluation: Maximum internal gas pressure depends on fuel rod temperatures, fission gas release, and void volume. The critical pressure limit for no clad lift-off (NCLO) depends on fuel swelling rate and clad tensile creep during normal operation. An evaluation demonstrated that maximum predicted rod internal pressures will not exceed the critical pressure limit at any time in life for anticipated operation of St. Lucie Unit 2.

Fuel Rod DNB Propagation

Criterion: The radiological dose consequences of DNB failures shall remain within the specified limits.

Evaluation: Calculation of DNB propagation depends on rod internal gas pressure, high-temperature creep, and high-temperature rupture stress (burst stress). An evaluation demonstrated that with the maximum rod internal pressures predicted for St. Lucie Unit 2, no DNB propagation will occur. If conditions change, evaluation will be performed to verify the lack of DNB propagation or to demonstrate that with DNB propagation the radiological dose consequences of DNB failures will remain within the specified limits.

Fuel Rod Stress

Criteria: (1) During Conditions I and II, the primary tensile stress in the cladding and the end-cap welds must not exceed 2/3 of the minimum unirradiated yield strength of the material at the applicable temperature. During Condition III, the primary tensile stress limit is the yield strength and during Condition IV seismic and LOCA (mechanical excitation only) conditions the stress limit is the lesser of 0.7 Su or 2.4 Sm.

(2) During Conditions I, II and III, primary compressive stress in the cladding and the end-cap welds must not exceed the minimum unirradiated yield strength of the material at the applicable temperature. During Condition IV seismic and LOCA (mechanical excitation only) conditions, the stress limit is the lesser of 0.7 Su or 2.4 Sm.

Evaluation: The above fuel rod stress criteria have been evaluated for St. Lucie Unit 2 fuel design containing ZIRLO™ cladding and found to be satisfied. This evaluation considered differential cladding pressures, creep, cladding growth and oxide buildup. All of these parameters involve the material properties and capabilities of the cladding. An evaluation demonstrated that the application of ZIRLO™ properties and models for all properties and models, except corrosion, will have no appreciable impact on maximum stress.

Fuel Rod Strain

Criteria: (1) At any time during the fuel or integral-burnable-absorber rod lifetime, the net unrecoverable circumferential tensile cladding strain shall not exceed 1% based on beginning-of-life (BOL) cladding dimensions. This criterion is applicable to normal operating conditions, and following a single Condition II or III event or a single anticipated operational occurrence (AOO).

(2) For fuel or integral-burnable-absorber rods having axial average burnups greater than 52 MWD/KGU, the total (elastic + plastic) circumferential cladding strain increment produced as a result of a single Condition II or III event, or a single AOO, shall not exceed 1%.

Evaluation: The above fuel rod strain criteria have been evaluated for St. Lucie Unit 2 fuel design containing ZIRLO™ cladding and found to be satisfied. This evaluation considered differential cladding pressures, creep and cladding growth. Further, the evaluation demonstrated that the application of ZIRLO™ properties and models will have no impact on maximum cladding strain.

Maximum Fuel Temperature

Criterion: The fuel rod centerline temperature shall not exceed the fuel melt temperature, accounting for degradation due to burnup and addition of burnable absorbers.

Evaluation: An evaluation for St. Lucie Unit 2 demonstrated that maximum predicted fuel temperatures will not exceed the fuel melt temperature limit at any time in life for anticipated operation of St. Lucie Unit 2.

Fuel Rod Fatigue Damage

Criterion: For the number and types of transients which occur during Condition I reactor operation, end-of-life (EOL) cumulative fatigue damage factor in the cladding and in the end-cap welds must be less than 0.8.

Evaluation: The above fuel rod fatigue damage factor criterion has been evaluated for St. Lucie Unit 2 fuel design containing ZIRLO™ cladding and found to be satisfied. The evaluation considered rod temperature and pressure, cladding creep, thermal expansion, and pellet swelling. The evaluation demonstrated that the application of ZIRLO™ properties and models will have no impact on maximum cladding fatigue damage factor.

Cladding Creep Collapse

Criterion: The time required for the radial buckling of the cladding in any fuel or gadolinium absorber rod must exceed the reactor operating time necessary for the appropriate batch to accumulate its design average discharge burnup. This criterion must be satisfied for continuous reactor operation at any reasonable power level and during any Condition I, II, or III situation. It will be considered satisfied if it can be demonstrated that axial gaps longer than 0.125 inch will not occur between fuel pellets and the plenum spring radial support capacity is sufficient to prevent cladding collapse under all design conditions.

Evaluation: The above fuel rod cladding collapse criterion has been evaluated for St. Lucie Unit 2 fuel design containing ZIRLO™ cladding and found to be satisfied. This evaluation considered differential cladding pressures, creep, cladding growth, and oxide buildup. The evaluation demonstrated that the application of ZIRLO™ properties and models for all properties and models, except corrosion, will have no impact on maximum stress.

Shoulder Gap

Criterion: The axial length between end fittings must be sufficient to accommodate differential thermal expansion and irradiation-induced differential growth between fuel rods and guide tubes such that it can be shown with 95% confidence that no interference exists.

Evaluation: The above design criterion is commonly referred to as shoulder gap and is evaluated using the irradiation-induced and thermal growth characteristics of the fuel rod cladding. This criterion has been evaluated for St. Lucie Unit 2 fuel design containing ZIRLO™ cladding and found to be satisfied. The evaluation demonstrated that the application of ZIRLO™ properties and models will have no impact on the minimum predicted shoulder gap.

Seismic and LOCA Loads

Criterion: The fuel rod cladding shall be capable of withstanding the loads resulting from the mechanical excitations occurring during the seismic and/or LOCA without failure resulting from excessive primary stresses.

Evaluation: The analyses methodology is unaffected by the change to ZIRLO™ cladding. Minor changes to allowable stress margins may occur but there will be no impact since significant stress margins exist for cladding under the postulated loading conditions.

2.4.2 ZIRLO™ Corrosion

With the introduction of ZIRLO™ clad fuel in St. Lucie Unit 2, cycle-specific fuel rod corrosion evaluation will be performed for the ZIRLO™ clad fuel consistent with the requirements of Reference 1. Evaluations of waterside corrosion of Westinghouse ZIRLO™ clad fuel as anticipated for operation in St. Lucie Unit 2 with the 30% SGTP operating conditions show acceptable performance with the current and planned primary chemistry for St. Lucie Unit 2.

2.5 SEISMIC/LOCA IMPACT ON FUEL ASSEMBLIES

Since the 16x16 HID-1L fuel design with ZIRLO™ cladding has no change to the fuel assembly grid design or the guide thimble tube design, there will be no impact to the seismic/LOCA evaluation. The reduced RCS flow rate, assuming the same temperature window for T_{cold} and T_{hot}, will have no impact on the evaluation.

2.6 CORE COMPONENTS

The core components for St. Lucie Unit 2 are compatible with the 16x16 HID-1L fuel design with ZIRLO™ cladding. The 16x16 HID-1L guide thimble tubes design and spacer grid configurations remains unchanged.

2.7 REFERENCES

1. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A, Revision 0, November 2001.

2. "Updated Safety Analysis Report - St. Lucie Unit 2," Docket Nos. 50-389, Amendment 14, December 2001 USAR Update.

3. "Safety Evaluation by the Office of Nuclear Regulation Related to Amendment No. 94, Facility Operating License No. NPF-4 Virginia Electric and Power Company Old Dominion Electric Cooperative North Anna Power Station, Unit No. 1," Docket No. 50-338, May 13, 1987.

4. "Use of Fuel with Zirconium-Based (Other than Zircaloy) Cladding (10 CFR 50.44, 50.46, and Appendix K to Part 50)," Federal Register, Vol. 57, No. 169, Rules and Regulations, pg. 39353 and 39355, August 31, 1992.

5. CENPD-139-P-A, "Fuel Evaluation Model," July 1974.

6. CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.

7. CEN-161(B)-P, Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992.

8. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

9. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

10. CENPD-275-P, Revision 1-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 1988.

11. CENPD-275-P, Revision 1-P, Supplement 1-P-A, "C-E Methodology for PWR Cores Designs Containing Gadolinia-Urania Burnable Absorbers," April 1999.

Table 2-1 Nominal Composition of ZIRLO™ and Zircaloy-4 Cladding				
Element	Zircaloy-4 (wt %)	Improved Zircaloy-4 (wt %)	OPTIN (wt %)	ZIRLO™ (wt %)
Tin (Sn)	1.2 -1.7	1.2 -1.45	1.2-1.44	0.8 -1.2
Iron (Fe)	0.18 - 0.24	0.18 - 0.24	0.18-0.24	0.09 - 0.13
Chromium (Cr)	0.07 - 0.13	0.07 - 0.13	0.07-0.13	---
Niobium (Nb)	---	---	---	0.8 -1.2
Zirconium (Zr)	Balance	Balance	Balance	Balance

Table 2-2 ZIRLO™ Fuel Experience			
Lattice	Operated through March 31, 2003		
	Number of Plants	Number of Assemblies	Number of Regions
17x17, 0.360" OD Rod	15	5455	67
17x17, 0.374" OD Rod	11	3353	47
16x16, 0.374" OD Rod	1	157	5
15x15, 0.422" OD Rod	9	2583	40
14x14, 0.422" OD Rod	2	153	4
14x14, 0.400" OD Rod	3	714	15

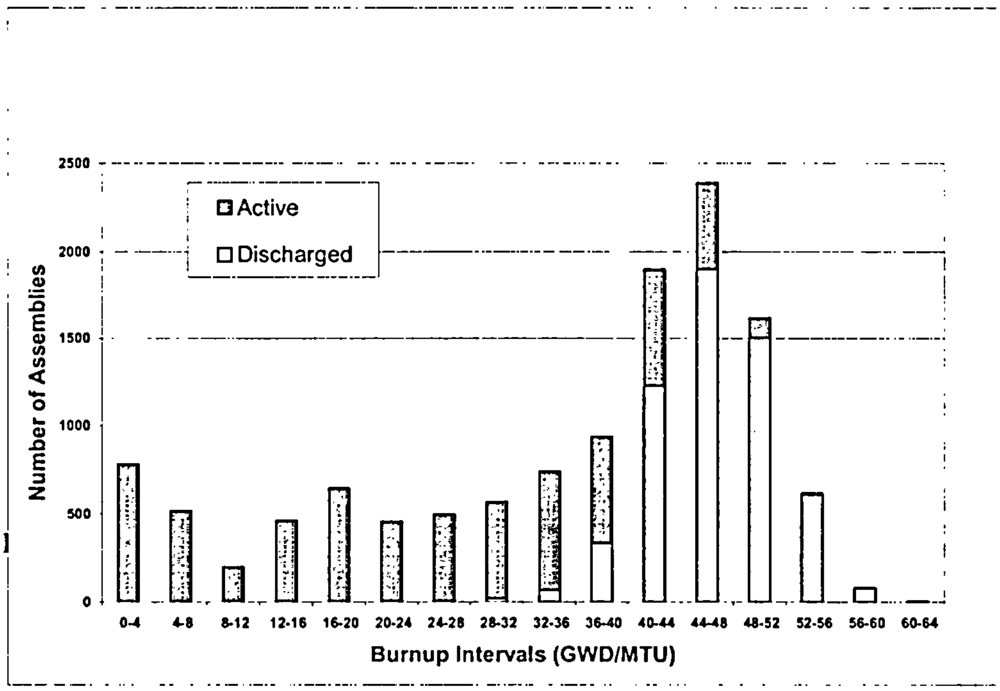

Figure 2-1 Total Number of Active and Discharged ZIRLO™ Assemblies (as of March 31, 2003, plotted as a function of burnup)

3 NUCLEAR DESIGN

3.1 INTRODUCTION AND SUMMARY

The effects of the following changes are evaluated in this section:

- Transitioning to ZIRLO™ cladding for the current 16x16 CE HID-1L fuel design
- An increase in steam generator tube plugging (SGTP) to 30% peak, producing a reduced reactor coolant system (RCS) minimum Technical Specification flow rate of 335,000 gpm, and
- Transitioning to Relaxed Axial Offset Control (RAOC) methodology for confirmation of axial power distributions.

The specific values of core safety parameters, e.g., power distributions, peaking factors, rod worths, and reactivity parameters, are primarily loading pattern dependent. The variations in the loading pattern dependent safety parameters are expected to be typical of the normal cycle-to-cycle variations for the standard fuel reloads. Standard nuclear design analytical models and methods (References 2, 3 and 4) accurately describe the neutronic behavior of the 16x16 CE HID-1L fuel design with ZIRLO™ cladding.

3.2 DESIGN BASIS

The specific design bases and their relation to the General Design Criteria (GDC) in 10 CFR 50, Appendix A for the current 16x16 CE HID-1L fuel design with ZIRLO™ cladding are the same as those of the current 16x16 CE HID-1L fuel design. The fuel burnup design will remain at the 60,000 MWD/MTU as currently licensed by the NRC (Reference 7).

3.3 METHODOLOGY

Consistent to the WCAP-9272 reload methodology (Reference 1), the purpose of evaluating the reload core analysis for the proposed changes is to ensure, prior to the cycle-specific reload design, that the values for the key safety parameters will remain applicable for the expected operating condition for which they will apply. This will allow the majority of any safety analysis re-evaluations/re-analyses to be completed prior to the cycle-specific design analysis.

No changes to the nuclear design philosophy, methods or models are necessary because of these changes. The reload design philosophy includes the evaluation of the reload core key safety parameters which comprise the nuclear design dependent input to the Updated Final Safety Analysis Report (UFSAR) safety evaluation for each reload cycle. These key safety parameters will be evaluated for each St. Lucie Unit 2 reload cycle. If one or more of the parameters fall outside the bounds assumed in the reference safety analysis, the affected transients will be re-evaluated/re-analyzed using standard methods and the results documented in the Reload Safety Evaluation (RSE) for that cycle.

ZIRLO™ material has also had extensive nuclear design and operating experience with other fuel assembly designs. These changes have a negligible effect on the use of standard nuclear design analytical models and methods to accurately describe the neutronic behavior of the 16x16 CE HID-1L fuel design.

3.4 DESIGN EVALUATION – PHYSICS CHARACTERISTICS AND KEY SAFETY PARAMETERS

The process of identification of bounding values for the key reload parameters has been maintained by establishing a set of "baseline neutronics" for use in the safety analyses. These values and the key parameters themselves have been adjusted only slightly from a standard application for a Westinghouse application to accurately model the unique features of the St. Lucie Unit 2 plant, trips, and technical specifications (References 5 and 6). The values established for the baseline neutronics were chosen to be sufficiently conservative to reasonably preclude violations in the reload evaluation process without being overly conservative (resulting in challenges to analysis margins). For the key parameters, limits have been established based on recent past operation of St. Lucie Unit 2 to identify representative parameter values. From these representative values, limit values for use in the safety analyses have been established with due consideration of the existing analysis assumptions, extensive plant and design experience, and accounting for changes in SGTP, minimum technical specification flow and cladding material transition.

The effect of coastdowns to extend the cycle length beyond nominal full power capability has been considered in determination of the key safety parameters.

Table 3-1 provides the key safety parameters ranges compared to the current limits.

3.5 DESIGN EVALUATION – POWER DISTRIBUTIONS AND PEAKING FACTORS

Beyond the power distribution impacts already mentioned, and discussed in Section 3.6, other changes to the core power distributions and peaking factors are the result of the normal cycle-to-cycle variations in core loading patterns. The normal methods of feed enrichment variation and insertion of fresh burnable absorbers will be employed to control peaking factors. Compliance with the peaking factor technical specifications can be assured using these methods.

3.6 TECHNICAL SPECIFICATION CHANGES RELATIVE TO NUCLEAR DESIGN

The plant technical specifications were reviewed. The technical specification changes which impact the nuclear design are:

- Reduced Core Operating Limits Report (COLR) limit for peak linear heat rate (large-break loss-of-coolant accident (LOCA) limitation),
- Introduction of a part power multiplier for F_r,
- Reduced COLR limit for the COLR linear heat rate (LHR) LCO when operating on the excore detector monitoring system (EDMS),
- Elimination of F_{xy} surveillance (consistent with St. Lucie Unit 1),
- Redefinition of linear heat rate surveillance when operating on the EDMS (only) to include linear heat rate surveillance with application of W(z) penalties, and

- Elimination of full-power positive moderator temperature coefficient (PMTC).

The reduced peak linear heat rate COLR limit reduces nuclear design flexibility, but was required to satisfy the acceptance criteria of 10 CFR 50.46 for large-break LOCA (see Section 5.2.3). Higher limits for peak linear heat rate are supportable from a neutronics design perspective, and may be addressed in future on a cycle-specific basis. The part power multiplier is based on St. Lucie Unit 2 plant-specific operating history information and conservatively bounds the effects of reduced power operation on radial peaking based on application of the standard nuclear design analytical models and methods. A revision of the COLR LHR LCO is required to accommodate the reduced peak LHR limit. Use of a linear ASI limit results in a non-linear relationship for axial flux difference, causing the most challenging heat rates to be determined at lower powers (approximately 70%), with relatively large values of axial shape index (ASI). Hence, the ASI limits have been reduced at lower powers where linear heat rates are challenged with the relatively large values of ASI. Constant monitoring for linear heat rate with the incore detector monitoring system (IDMS) makes F_{xy} surveillance redundant when performing surveillance with incore detectors. Therefore F_{xy} surveillance for incore monitoring has been eliminated, consistent with previous changes for St. Lucie Unit 1. For excore monitoring, F_{xy} surveillance is replaced by linear heat rate surveillance, with application of W(z) penalties. This is consistent with typical operation using Best Estimate Analyzer for Core Operations – Nuclear (BEACON) already in use at St. Lucie Unit 2. Because analytical margins are expected to be approximately the same for the application of the RETRAN-based non-LOCA methods compared to the current methods for evaluation of axial power distributions, the elimination of the full-power PMTC will permit less challenge to bounding neutronic assumptions for the non-LOCA analyses to accommodate margin reductions due to decreased RCS flow.

The following F_r and peak linear heat rate values have been considered in the evaluation to appropriately bound the defined changes:

$F_r = 1.70 \times [1 + 0.4(1-P)]$, PLHR = 13.0 kw/ft[1] Operating on IDMS

and

$F_r = 1.70 \times [1 + 0.4(1-P)]$, PLHR = 12.5 kw/ft[2] Operating on IDMS or EDMS

where P is the fraction of full power.

3.7 NUCLEAR DESIGN EVALUATION CONCLUSIONS

Except where specifically identified to accommodate margin considerations, the key safety parameters evaluated for St. Lucie Unit 2 as it transitions to ZIRLO™ cladding, reduced RCS flow, and RAOC axial power distribution methods are typical of the normal cycle-to-cycle variations experienced as loading patterns change. The usual methods of enrichment and burnable absorber usage will be employed to ensure compliance with the peaking factor technical specifications.

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[1] Conservative for confirmation of power distributions for DNBR event or initial conditions for non-LOCA transients.

[2] Conservative for evaluation of operational flexibility for reload requirements for linear heat rate.

3.8 REFERENCES

1. Davidson, S. L. (Ed.), et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-NP-A, July 1985.

2. Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June 1988.

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5. "Technical Specifications – St. Lucie Plant, Unit 2," Docket No. 50-389, Amendment 131.

6. "Updated Safety Analysis Report - St. Lucie Plant, Unit 2," Docket No. 50-389, Rev. 14.

7. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A, Revision 0, November 2001.

Table 3-1 Range of Key Safety Parameters		
Safety Parameter	**Current Design Values**	**Future Design Values**
Reactor Core Power (MWt)	2700	2700
Vessel Inlet Coolant Temp. HFP (°F)	548.4	549
Coolant System Pressure (psia)	2250	2250
Most Positive MTC (pcm/°F) at > 70% Reactor Thermal Power (RTP)	+ 3	+ 5 @ 70% power, ramping to + 0 @ 100% power
Most Positive MTC (pcm/°F) at $\leq$ 70% RTP	+ 5	+ 5
Most Positive MDC ($\Delta K/g/cm^3$)	0.43	0.43
Doppler Temperature Coefficient	-0.00253 to -0.00108 $\Delta\rho/\Delta(K^{1/2})$	-2.90 to -0.91 pcm/°F
Beta-Effective	0.0044 to 0.0070	0.0044 to 0.0070
Normal Operation F_r (without uncertainties)	1.70	1.70
Shutdown Margin (pcm)	3600	3600
Normal Operation Peak Linear Heat Rate (kw/ft)	13.0	12.5

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4 THERMAL AND HYDRAULIC DESIGN

4.1 INTRODUCTION AND SUMMARY

This section describes the thermal-hydraulic (T/H) departure from nucleate boiling (DNB) analysis, in support of Saint Lucie Unit 2 30% steam generator tube plugging (SGTP) and WCAP-9272, Reload Methodology Transition Project. The T/H analysis ensures that the reactor core meets the DNB design criterion.

The specific criterion for pressurized water reactor (PWR) core T/H design, as described in the Updated Final Safety Analysis Report (UFSAR) (Reference 1), is that there should be a 95% probability at a 95% confidence level (95/95) that the hot rod in the core does not experience DNB during normal operation or anticipated operational occurrences (AOOs). Uncertainties in the values of process parameters, core operating parameters, and fuel design parameters are also treated with at least a 95% probability at a 95% confidence level.

The T/H analysis is based on the 16x16 HID-1L fuel design that has been used in Saint Lucie Unit 2 core reloads. Table 4-1 illustrates a comparison between the previous T/H design parameters and the new T/H design parameters used in this analysis. A discussion of the T/H methodology and DNB ratio (DNBR) limits is provided in the subsequent sections.

4.2 METHODOLOGY

The T/H analysis of the 30% SGTP is based on the ABB-NV DNB correlation (Reference 2), the Revised Thermal Design Procedure (RTDP) (Reference 3), and the VIPRE-01 (VIPRE) code (References 4, 5 and 6). The W-3 correlation and the Standard Thermal Design Procedure (STDP) are used at the design conditions at which the ABB-NV DNB correlation and RTDP are not applicable. The STDP is the traditional design method with parameter uncertainties applied deterministically in the limiting direction.

4.2.1 ABB-NV DNB Correlation

The ABB-NV DNB correlation is based entirely on rod bundle data and reflects significant improvements in the accuracy of the critical heat flux predictions over previous DNB correlations for Combustion Engineering (CE) fuel designs. The Nuclear Regulatory Commission (NRC) has approved that a 95/95 correlation limit DNBR of 1.13 is appropriate for the CE 16x16 fuel assemblies (Reference 2). Furthermore, it has been shown that the ABB-NV 95/95 correlation limit DNBR of 1.13 remains valid with the VIPRE-01 (VIPRE) code (Reference 6).

4.2.2 Revised Thermal Design Procedure

The RTDP is a statistical DNB analysis method similar to the Extended Statistical Combination of Uncertainty (ESCU) (Reference 7) methodology. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor. This factor is used for defining the design limit DNBR that satisfies the 95/95 DNB design criterion. Since the parameter uncertainties are considered in determining the RTDP design limit, DNBR calculations in the plant safety analyses are performed using the parameter values without the uncertainties.

The parameter uncertainties considered in RTDP design limit DNBR are the same as those used previously in ESCU. The following uncertainties have been incorporated into the RTDP design limit:

- The nuclear enthalpy rise hot channel factor (F^N_r)
- The enthalpy rise engineering hot channel factor (F^E_r)
- Uncertainties in the VIPRE and transient codes
- Uncertainties in inlet flow distribution, cladding outside diameter, and rod pitch
- Uncertainties based on surveillance data associated with reactor coolant system (RCS) coolant flow, coolant temperature, pressure, and reactor core power.

4.2.3 VIPRE Code

VIPRE (Reference 4) is a subchannel code developed by the Battelle Northwest National Laboratories under the sponsorship of Electric Power Research Institute (EPRI). VIPRE was developed based on several versions of the COBRA code. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow, and momentum pressure drop. Together with a DNB correlation, VIPRE is used for predicting DNBR margin in the reactor core under steady-state conditions and in non-LOCA transients.

Westinghouse has made additions and enhancements in its version of the VIPRE-01 code, including the installation of the ABB-NV DNB correlation. However, the code modifications do not alter the fundamental VIPRE computational methods and functional capabilities. Westinghouse VIPRE modeling and qualification for PWR non-LOCA T/H safety analysis are described in Reference 5.

4.3 DNBR LIMITS

Table 4-2 provides a listing of parameter uncertainties that are statistically convoluted with the ABB-NV correlation uncertainty in defining an RTDP design limit DNBR. A sensitivity factor for each parameter was determined through VIPRE calculations representing the change in DNBR corresponding to a change in the parameter. The RTDP DNBR limit calculations are illustrated in Table 4-3. In the DNB safety analyses, the design limit DNBR is conservatively increased to provide DNB margin to offset the effect of rod bow and any other DNB penalties that may occur, and to provide flexibility in design and operation of the plant. The increased DNBR is referred to as the safety analysis limit (SAL) DNBR as shown in Table 4-4, along with the plant-specific margin retained between the design limit and the SAL. It should be noted that the DNBR margin summaries are cycle dependent and may vary from that documented here in future reload designs.

4.4 EFFECTS OF FUEL ROD BOW ON DNBR

The concerns about the fuel rod bow phenomenon are the potential effects on bundle power distribution and on the margin of fuel rods to DNB. Thus, the phenomenon of fuel rod bowing must be accounted for in the DNBR safety analysis of Condition I and Condition II events (i.e., normal operations and AOOs). The effects of fuel rod bowing on DNBR margin have been incorporated into the safety and setpoint analysis. The rod bow penalty for the 30% SGTP remains unchanged from the current value of 1.2% DNBR as discussed in Saint Lucie 2 Unit 2 UFSAR (Reference 1). The rod bow penalty is valid with the ABB-NV DNB correlation as discussed in Reference 2.

4.5 REFERENCES

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4. Stewart, C. W. et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Volume 1-3 (Revision 3, August 1989), Volume 4 (April 1987), NP-2511-CCM-A, Eletric Power Research Institute.

5. Sung, Y., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A (Proprietary), October 1999.

6. Sung, Y. et al., "Addendum 1 to WCAP-14565-P-A, Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," WCAP-14565-P-A, Addendum 1, May 2003.

7. CEN-348(B)-P-A-SUPPL1-P-A, "Extended Statistical Combination of Uncertainties," January 1997.

Table 4-1 Saint Lucie Unit 2 Thermal-Hydraulic Design Parameters Comparison		
Thermal-Hydraulic Design Parameters	**Cycle 14**	**30% SGTP**
Reactor Core Heat Output, MWt	2700	2700
Heat Generated in Fuel, %	97.5	97.5
RCS Pressure, psia	2250	2250
Integrated Radial Peaking Factor, $F^N r$[1] Part Power Multiplier[2]	1.70 COLR Figure 3.2-3	1.70 (1 + 0.4*(1 – P))
Vessel Thermal Design Flow Rate (including bypass), gpm	355,000	335,000
Core Inlet Temperature, °F	549.0	549.0
Design Core Bypass Flow, % of Vessel Flow	3.7	3.7
Core Flow Area, ft^2	54.82	54.82

1. Excluding 6% measurement uncertainty.
2. P = (Thermal Power/Rated Thermal Power)

Table 4-2 RTDP Parameter Uncertainties	
Parameter	**Uncertainty (One-Sigma)**
Power	±1.0% at 100% power ±2.2% at 50% power
Reactor Coolant System Flow	±1.92% flow +0.24% flow (bias)
Pressure	±22.5 psi
Inlet Temperature	±1.33°F
Radial Peaking Factor, $F^N r$	±3.65%
F^E_r	±0.015
F^E_Q	±0.015
Inlet Flow Distribution	±4.92% flow
Cladding Outside Diameter	±0.000244 inches
Rod Pitch	±0.000236 inches
VIPRE Code	±0.02
Transient Code	±0.005

Table 4-3a Calculation of RTDP DNBR Design Limit for Matrix Channel *
σ = standard deviation
μ = mean
s = sensitivity factor

RTDP Parameter	μ	σ	σ/μ	S	$S^2(\sigma/\mu)^2$
Power, fraction	1.000	0.0126	0.0126	-2.791	1.24E-03
Pressure, psia	2250	22.5	0.010	1.517	2.30E-04
Flow, fraction	1.000	0.0192	0.0192	1.980	1.45E-03
Tin, °F	549.0	1.333	0.0024	-7.566	3.38E-04
Inlet Flow Distribution, fraction	0.85	0.0492	0.0579	0.053	9.25E-06
$F^N r$	1.700	0.0621	0.0365	-2.831	1.07E-02
F^E_r	1.000	0.0150	0.0150	-0.679	1.04E-04
Pitch, in.	0.506	0.000236	0.0005	0.804	1.41E-07
Cladding OD, in.	0.382	0.000244	0.0006	0.911	3.39E-07
F^E_Q	1.000	0.0150	0.0150	1.000	2.25E-04
VIPRE Code	1.000	0.020	0.020	1.000	4.00E-04
Transient Code	1.000	0.005	0.005	1.000	2.50E-05
				$(\sigma_y/\mu_y)^2 = \Sigma =$	0.014687
$(\sigma_z/\mu_z)^2 = (\sigma_y/\mu_y)^2 + [k*s_{M/P}/(1.645*M_{M/P})]^2$ $= (\sigma_y/\mu_y)^2 + [1.760*0.066/(1.645*1.000)]^2$					
				Design Limit DNBR $= 1/[M_{M/P}(1 - 1.645(\sigma_z/\mu_z))] =$	1.300

* Subchannel surrounded by four fuel rods.

Table 4-3b Calculation of RTDP DNBR Design Limit for Side Thimble Channel *
σ = standard deviation
μ = mean
s = sensitivity factor

RTDP Parameter	μ	σ	σ/μ	S	$S^2(\sigma/\mu)^2$
Power, fraction	1.000	0.0126	0.0126	-2.448	9.52E-04
Pressure, psia	2250	22.5	0.010	1.177	1.38E-04
Flow, fraction	1.000	0.0192	0.0192	1.714	1.08E-03
Tin, °F	549.0	1.333	0.0024	-6.503	2.49E-04
Inlet Flow Distribution, fraction	0.85	0.0492	0.0579	0.040	5.33E-06
$F^N r$	1.700	0.0621	0.0365	-2.488	8.25E-03
F^E_r	1.000	0.0150	0.0150	-0.632	9.00E-05
Pitch, in.	0.506	0.000236	0.0005	0.407	3.61E-08
Cladding OD, in.	0.382	0.000244	0.0006	1.539	9.66E-07
F^E_Q	1.000	0.0150	0.0150	1.000	2.25E-04
VIPRE Code	1.000	0.020	0.020	1.000	4.00E-04
Transient Code	1.000	0.005	0.005	1.000	2.50E-05
				$(\sigma_y/\mu_y)^2 = \Sigma =$	0.011419
$(\sigma_z/\mu_z)^2 = (\sigma_y/\mu_y)^2 + [k*s_{M/P}/(1.645*M_{M/P})]^2$ $= (\sigma_y/\mu_y)^2 + [1.760*0.066/(1.645*1.000)]^2$					
				Design Limit DNBR $= 1/[M_{M/P}(1 - 1.645(\sigma_z/\mu_z))] =$	1.267

* Subchannel surrounded by two fuel rods and a guide thimble tube.

Table 4-3c	**Calculation of RTDP DNBR Design Limit for Corner Thimble Channel *** **σ = standard deviation** **μ = mean** **s = sensitivity factor**				
RTDP Parameter	μ	σ	σ/μ	S	$S^2(\sigma/\mu)^2$
Power, fraction	1.000	0.0126	0.0126	-2.353	8.79E-04
Pressure, psia	2250	22.5	0.010	1.702	2.90E-04
Flow, fraction	1.000	0.0192	0.0192	1.832	1.24E-03
Tin, °F	549.0	1.333	0.0024	-7.111	2.98E-04
Inlet Flow Distribution, fraction	0.85	0.0492	0.0579	-0.013	5.81E-06
Fr	1.700	0.0621	0.0365	-2.555	8.70E-03
F^E_r	1.000	0.0150	0.0150	-0.517	6.02E-05
Pitch, in.	0.506	0.000236	0.0005	0.404	3.55E-08
Cladding OD, in.	0.382	0.000244	0.0006	0.915	3.42E-07
F^E_Q	1.000	0.0150	0.0150	1.000	2.25E-04
VIPRE Code	1.000	0.020	0.020	1.000	4.00E-04
Transient Code	1.000	0.005	0.005	1.000	2.50E-05
				$(\sigma_y/\mu_y)^2 = \Sigma =$	0.012113
$(\sigma_z/\mu_z)^2 = (\sigma_y/\mu_y)^2 + [k{*}s_{M/P}/(1.645{*}M_{M/P})]^2$ $= (\sigma_y/\mu_y)^2 + [1.760{*}0.066/(1.645{*}1.000)]^2$					
				Design Limit DNBR = $1/[M_{M/P}(1 - 1.645(\sigma_z/\mu_z))] =$	1.274
* Subchannel surrounded by three fuel rods and a guide thimble tube.					

Table 4-4 DNBR Limits and Margin Summary	
Fuel Type	**16x16 HID-1L**
DNB Correlation	ABB-NV
DNBR Correlation Limit	1.13
RTDP DNBR Design Limit*	1.27 (side thimble) 1.28 (corner thimble) 1.30 (matrix)
DNBR SAL	1.39 (thimbles) 1.42 (matrix)
DNBR Retained Margin	8.6% (side thimble) 7.9% (corner thimble) 8.4% (matrix)
Rod Bow DNBR Penalty	-1.2%
Available DNBR Margin	7.4% (side thimble) 6.7% (corner thimble) 7.2% (matrix)

*DNBR Limits for are non-RTDP events are identified in the respective transient information contained in Section 5.

5 ACCIDENT ANALYSIS

5.1 NON-LOCA TRANSIENTS

The transient safety analyses discussed herein support an increase in steam generator tube plugging (SGTP) to 30%, as well as a transition to the Westinghouse WCAP-9272 reload approach. Although the analyses did support an increase in the SGTP level to 30%, the overall approach was to analyze the plant as currently designed. In some cases, changes were implemented which improved the analysis results, but were the result of overly conservative assumptions, such as the reduction in the assumed RTD response time. Changes that are inherent to an increase in SGTP, such as reduced RCS flow, were addressed.

The non-LOCA events are analyzed with NRC approved codes. The specific models and methods used in the implementation of the WCAP-9272 reload philosophy are based on the same principles as, and are similar to, those approved by the NRC. The non-LOCA phenomena have characteristics and behavior that are well understood. The results of the transients were found to be consistent with expectations and consistent with the observed behavior of other Westinghouse PWRs, in light of the given changes. For cases where adaptations of methodology or model were made, comparisons were made to the existing licensing basis analysis and were found to be consistent with those analyses. Differences that occurred were attributable to the differences in the assumptions and the details of the models used to assure bounding analyses were performed.

The transient safety analyses discussed herein support the transition of the 16x16 Combustion Engineering (CE) HID-1L fuel design to ZIRLO™ cladding. The transition to ZIRLO™ cladding is the only change in the 16x16 CE HID-1L fuel assembly design. In addition, safety analyses presented in Sections 5.1 through 5.6 also cover the increase in steam generator tube plugging to 30% peak (asymmetric plugging of ~ 7%), and a reduced reactor coolant system (RCS) flow rate of 335,000 gpm. These are justified with respect to the non-loss-of-coolant-accident (LOCA) and LOCA design bases in Sections 5.1 and 5.2, respectively. Sections 5.3 and 5.4 contain the mass and energy release for steamline break and LOCA, respectively. The containment design basis analysis is addressed in Sections 5.5. Key analysis assumptions or bases that differ from those specified below are identified as required.

Key fuel features applicable to all analyses that have been considered include:

- Gadolinia (2.5, 4, 6, or 8 weight percent [w/o]) fuel burnable absorber
- ZIRLO™ fuel cladding,
- 0.382-inch outer diameter (OD) fuel rod design,
- Solid, mid-enriched pellets in axial blankets
- Cold, undensified fuel stack height of 136.70 inches.

5.1.0 Non-LOCA Overview

5.1.0.1 Fuel Design Mechanical Features

The effects of fuel design mechanical features on the non-LOCA transient analyses are accounted for in fuel-related input assumptions, such as fuel and cladding dimensions, cladding material, fuel temperatures, and core bypass flow.

The transition to ZIRLO™ cladding is the only change in the fuel assembly design. Consequently, the current 16x16 CE HID-1L fuel design with OPTIN cladding and the 16x16 CE HID-1L fuel design with ZIRLO™ cladding are mechanically, thermal-hydraulically, and neutronically compatible as demonstrated in Sections 2, 3, and 4. Also, the non-LOCA transient analyses nuclear steam supply system (NSSS) models utilize only a limited amount of detail in the fuel-related input assumptions (see above). In addition, the non-fuel-related acceptance criteria parameter results of the non-LOCA analyses (for example, RCS pressure, main steam system (MSS) pressure, pressurizer does not become water solid, etc.) are not overly sensitive to the fuel-related input assumptions. Therefore, the results of the non-LOCA transient analyses that are not fuel-related are applicable to either cladding type. Furthermore, the thermal-hydraulic statepoints (RCS pressure, core inlet temperature, core flow, and core-average heat flux) generated in the non-LOCA transient analyses are suitable for use in the thermal-hydraulic DNBR analyses of the DNBR analyses of the 16x16 HID-IL fuel.

5.1.0.2 Peaking Factors, Kinetics Parameters

The power distribution is characterized by an enthalpy hot channel factor (radial peaking, F_r) of 1.70 (Revised Thermal Design Procedure [RTDP (Reference 10)])/1.802 (non-RTDP) and a peak linear heat rate of 13.0 kw/ft for the 16x16 CE HID-1L fuel design with ZIRLO™ cladding. F_r is important for transients that are departure from nucleate boiling (DNB) limited (Note that Table 5.1.0-2 identifies those events analyzed for DNB concerns as well as the DNB methodology used: (RTDP or non-RTDP)). Since F_r increases with decreasing power level (due to rod insertion), all transients that may be DNB limited are assumed to begin with an F_r consistent with the F_r defined in the Technical Specifications for the nominal power level. Peak linear heat rate is important for transients that may be overpower limited. Peak linear heat rate may increase with decreasing power level such that the full power hot-spot heat flux is not exceeded. Consequently, all non-LOCA transients for this program that may be overpower limited assume an initial hot full power peak linear heat rate 13.0 kw/ft.[1]

The analyses of events that are sensitive to minimum shutdown margin assumed to be 3600 pcm (T_{AV} greater than 200 °F), consistent with the current St. Lucie Unit 2 COLR limits.

5.1.0.3 Plant Characteristics and Initial Conditions

Plant Design Conditions

Table 5.1.0-1 lists the principal power rating value that is assumed in the analyses performed in this report. The guaranteed Nuclear Steam Supply System (NSSS) thermal power output includes the thermal

[1] Note that while initial implementation is limited to a linear heat rate of 12.5 kw/ft due to large-break LOCA limitations, all non-LOCA transient analyses were performed at, and support, a peak linear heat rate of 13.0 kw/ft.

power generated by the reactor coolant pumps. Where initial power operating conditions are assumed in accident analyses, the "NSSS thermal power output" is assumed. The values used for each transient analyzed are given in Table 5.1.0-2. The nominal values of pertinent plant parameters are given in Table 5.1.0-3.

Initial Conditions

For accidents that are departure from nucleate boiling (DNB) limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and included in the design limit DNB ratio (DNBR), as described in WCAP-11397 (Reference 2). This procedure is known as the "Revised Thermal Design Procedure," (RTDP) and is discussed in Section 4.2.

For accidents that are not DNB limited, or in which the RTDP is not employed, the initial conditions are obtained by adding the maximum steady-state errors to rated values. The following steady-state errors are considered:

Core power: ± 2 percent at 100% power for calorimetric error (Figure 5.1.0-1)
Average RCS temperature: ± 3.0°F temperature measurement error
Pressurizer pressure: ± 45 psi steady-state fluctuations and pressure measurement error

Table 5.1.0-2 summarizes initial conditions and computer codes used in the accident analyses and shows which accidents employ RTDP.

Other Major Assumptions

Table 5.1.0-2 lists the non-LOCA initial condition assumptions used. Other major assumptions considered in the non-LOCA transient analyses are discussed below:

a. The pressurizer safety valves (PSVs) are modeled assuming a ± 3-percent setpoint tolerance.

b. The Main Steam Safety Valves are modeled with different opening pressures for the two banks. The first bank of valves are modeled using 3% tolerance and 3% accumulation. A maximum tolerance of 1.0236 or 2.3% is modeled for the second bank of main steam safety valves.

c. The fission product contribution to decay heat assumed in the non-LOCA analyses bounds the American Nuclear Society (ANS)-5.1-1979 residual decay heat model, increased by two standard deviations for conservatism.

d. In those events where a loss of offsite power is assumed, the loss of offsite power is modeled either as the initiating event, or as the result of a turbine trip with a 3-second delay.

5.1.0.4 Thermal Margin/Low Pressure Reactor Trip Setpoints

5.1.0.4.1 Design Basis

The current thermal margin/low pressure (TM/LP) reactor trip function ensures that the DNB design basis is satisfied for the St. Lucie Unit 2 plant. It also precludes hot leg boiling to ensure that the loop ΔT (hot leg temperature minus the cold leg temperature) is proportional to the power. The TM/LP reactor trip is a function of the RCS temperature, the pressurizer pressure, the core power as measured by the excore detector or the ΔT (hot leg temperature minus the cold leg temperature) and the axial power shape as defined by the bottom detector signal minus the top detector signal divided by the total power (axial shape index or ASI). The RCS pressure range that the TM/LP reactor trip is valid for is limited by the TM/LP floor (low pressurizer pressure trip) and the high pressurizer pressure trip (safety analysis values). The TM/LP is also limited by the locus of conditions defined by the main steam safety valves which will limit any increase in the RCS temperature and the variable high power reactor trip function (safety analysis value), which limits the overpower condition. The TM/LP is based on all of the reactor coolant pumps being in operation and accounts for changes in the axial power shape via the A_1 function, which adjusts the setpoint for variations in the axial power shapes. Highly skewed axial power shapes can be limiting with respect to ensuring that the DNB design basis is satisfied.

The current TM/LP reactor trip function, as presented in the St. Lucie Unit 2 Technical Specifications, is presented below.

$$Pvar = 1400 * QR_1 * A_1 + 17.85 * Tinlet - 9410$$

Where the QR_1 function and the A_1 function are defined in the St. Lucie technical specifications

The QR_1 function is a linear function which adjusts the TM/LP setpoint to account for the effects of an increase in the Fr at lower power levels. As noted above, the A_1 function adjusts the setpoint for variations in the axial power shapes.

5.1.0.4.2 Method of Evaluation

The current TM/LP reactor trip setpoint described above was confirmed to be valid for the St. Lucie Unit 2 30% steam generator tube plugging program. The confirmation of the TM/LP reactor trip setpoint was based on the approach presented in the approved WCAP-8745 (Reference 1). The approach presented in WCAP-8745 ensures that the overtemperature ΔT reactor trip setpoint (which is similar in functionality to the TM/LP reactor trip function) bounds the core thermal limits. Therefore, the TM/LP setpoint with an A_1 function of 1.0 (that is, corresponding to an axial power shape with an ASI of 15%) was demonstrated to bound a conservative set of core thermal limits, based on a limiting reference axial power shape for an ASI of 15%. These core limits are applicable to the St. Lucie fuel and are based on a rated core power of 2700 MWt, an RCS flowrate of 341,400 gpm and a design Fr of 1.70 with a part power multiplier of 0.4. The core thermal limits present a locus of conditions where the DNB design basis is satisfied and hot leg boiling is precluded and consider variations in the RCS inlet temperature with power and pressure. The relationship of the TM/LP setpoint with an A_1 of "1.0" (minimum value) corresponding to an ASI of 15% to the reference core thermal limits is illustrated in Figure 5.1.0-5. The locus of conditions for the TM/LP is obtained as a function of the inlet temperature and the pressurizer pressure. The area of permissible operation (power, temperature, and pressure) is bounded by the combination of the following.

- Variable high power reactor trip which is assumed to be 118% of rated thermal power.
- High pressurizer pressure reactor trip which is assumed to be 2415 psia.
- TM/LP floor setpoint, which is assumed to be 1855 psia.
- The main steamline safety valves assuming a pressure corresponding to 110% of design pressure (1100 psia)

It should be noted that the TM/LP "protection lines" in Figure 5.1.0-5 are drawn to include the uncertainties as defined by the approved Extended Statistical Combination of Uncertainties methodology (see CEN-348 (B)-P-A). The uncertainties include processing uncertainties, system parameter uncertainties, ASI uncertainties and critical heat flux correlation uncertainties. These uncertainties, as defined by the extended statistical combination of uncertainties (ESCU) Penalty Factors, have been incorporated into the TM/LP "protection lines." Under nominal conditions, a reactor trip on the TM/LP reactor trip function would occur well within the area bounded by these lines. The figure shows that the safety analysis limit DNBR is bounded by the TM/LP protection lines, the variable high power protection lines, with uncertainties, and the steam generator safety valve lines.

In addition to confirming that the TM/LP reactor trip function protects the core thermal limits, it is necessary to ensure that the current A_1 function is adequately adjusting the TM/LP reactor trip setpoint for skewed axial power shapes. The approach for confirming the A_1 function is similar to the approach performed for the overtemperature ΔT (OTDT) $f(\Delta I)$ function, as discussed in WCAP-8745. That is, a DNB analysis is performed on a large number of highly skewed axial power shapes, as generated via the RAOC methodology discussed in Section 4. A conservative penalty of Tinlet versus ASI is generated based on this DNB analysis. This curve is used adjust the core thermal limits for various ASI values. The resulting relationship is compared to the TM/LP A_1 function. The result is that for skewed axial power shapes, the TM/LP A_1 function continues to conservatively protect the locus of conditions corresponding to the safety analysis limit DNBR.

The TM/LP setpoint also includes margin to the core thermal limits, as defined by the gamma bias term, which is intended to account for the delays associated with fluid transport, instrumentation response time, protection system delays and delays associated with the time for the control rods to drop. This gamma bias term is used in the "steady-state" verification of the TM/LP since the dynamic compensation associated with the TM/LP reactor trip function is set to zero. The transient analysis of the Chapter 15 events, such as CEA Withdrawal at Power and RCS Depressurization, ensure that the gamma bias accounts for the above-mentioned delays. This step is necessary as the confirmation of the TM/LP setpoint against the core thermal limits is a "steady-state" confirmation of the setpoint. The Chapter 15 analyses demonstrate that under transient conditions, the TM/LP reactor trip function with uncertainties, provides protection such that the DNB design basis is satisfied and hot leg boiling is precluded.

Note that the TM/LP setpoint remains valid for the range of RCS average temperatures being considered for this program (that is, T_{avg} window between 563.0°F and 576.5°F).

5.1.0.4.3 Results/Conclusions

Based on the above discussed evaluation, the current TM/LP reactor trip function has been confirmed to provide the necessary protection to ensure that the DNB design basis is satisfied and to ensure that hot leg boiling is precluded, thereby ensuring that ΔT is proportional to the power.

5.1.0.4.4 References

1. WCAP-8745, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip functions," Approved September 1986.

5.1.0.5 Reactor Protection System and Engineered Safety Features Evaluation Functions Assumed in Analyses

Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 5.1.0-4. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are released. There are various instrumentation delays associated with each trip function including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. A reactor trip signal acts to open trip breakers that removes power from to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the control element assemblies (CEAs), which then fall by gravity into the core.

Figure 5.1.0-2 presents the relationship between the core thermal limits and selected protection functions. The core thermal limits are provided as reactor pressure vessel inlet temperatures for the design flow and reference power distribution (1.55 chopped cosine) for limiting pressurizer pressures and the nominal operating pressure value. The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high pressurizer pressure (fixed setpoint); low pressurizer pressure thermal margin/low-pressure ((TM/LP) floor/fixed setpoint); TM/LP (variable setpoints); variable high power trip (ceiling/fixed setpoint), and by a line defining conditions at which the steam generator safety valves open. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that, under nominal conditions, trip would occur well within the area bounded by these lines.

The utility of this diagram is that the limit imposed by any given DNBR can be represented as a series of lines. The core thermal limit lines represent the locus of conditions for which for DNBR equals the limit value and ensures that vessel exit boiling is precluded to ensure that ΔT remains a function of power. All points below and to the left of the core thermal limits for a given pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases based on the reference power shape if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line.

The limit value, which was used as the DNBR limit for all accidents analyzed with the RTDP, is conservative compared to the actual design DNBR value required to meet the DNB design basis as discussed in Section 4.3.

The difference between the limiting trip point assumed for the analysis and the nominal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant technical specifications. Additionally, protection system channels are calibrated and instrument response times determined periodically in accordance with the plant technical specifications.

5.1.0.6 CEA Reactivity Characteristics

The negative reactivity insertion following a reactor trip is a function of the position versus time of the CEAs and the variation in rod worth as a function of rod position. The CEA position versus time assumed in accident analyses is shown in Figure 5.1.0-3. Following a 0.74 second breaker opening delay, a CEA insertion time of 2.66 seconds is assumed in the safety analyses unless otherwise noted in the discussion of a specific event. The insertion times are specified in the plant technical specifications.

Figure 5.1.0-4 shows the fraction of total negative reactivity insertion versus normalized rod position for a core that the axial distribution is skewed to the lower region of the core. This curve is used to compute the negative reactivity insertion versus time following a reactor trip that is input to the point kinetics core models used in transient analyses. The bottom skewed power distribution itself is not an input into the point kinetics core model. There is inherent conservatism in the use of Figure 5.1.0-4 in that it is based on a skewed flux distribution which would exist relatively infrequently.

A total negative reactivity insertion following a trip of 5.4 percent Δk is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available. For Figure 5.1.0-3, the CEA drop time is normalized to 2.66 seconds. Figures 5.1.0-3 and 5.1.0-4 are used to define a conservative trip reactivity versus time relationship in those transient analyses for which a point kinetics core model is used.

Where special analyses require use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the CEA position versus time is based on Figure 5.1.0-3.

5.1.0.7 Reactivity Coefficients

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient (MTC) and the Doppler power coefficient. The values used in the analysis of each event are given in Table 5.1.0-2.

In the analysis of certain events, conservatism requires the use of maximum reactivity coefficient values whereas in the analysis of other events, conservatism requires the use of minimum reactivity coefficient values. Some analyses, such as loss of reactor coolant from cracks or ruptures in the RCS, do not depend on reactivity feedback effects. Figure 5.1.0-6 shows the upper and lower bound Doppler power coefficients as a function of power, used in the accident analysis. Figure 5.1.0-6 shows the maximum moderator temperature coefficient as a function of power level. The justification for use of maximum versus minimum reactivity coefficient values is treated on an event-by-event basis. In some cases conservative combinations of parameters are used for a given transient to bound the effects of core life, although these may represent hypothetical combinations.

5.1.0.8 Computer Codes Utilized

Summaries of the principal computer codes used in the transient analyses are given below. The codes used in the analyses of each non-LOCA transient have been listed in Table 5.1.0-2.

FACTRAN

The FACTRAN program calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which exhibits the following features simultaneously:

- A sufficiently large number of radial space increments to handle fast transients, such as rod ejection accidents
- Material properties, which are functions of temperature and a sophisticated fuel-to-cladding gap heat transfer calculation
- The necessary calculations to handle post DNB transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel

FACTRAN is further discussed in Reference 4.

RETRAN

RETRAN is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. This code simulates a multi-loop system by a lumped parameter model containing the reactor vessel, hot- and cold-leg piping, reactor coolant pumps, steam generators (tube and shell sides), steamlines, and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves may also be modeled. RETRAN includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the steam generator uses a detailed nodalization for the thermal transients. The reactor protection system (RPS) simulated in the code includes reactor trips on variable high power, thermal margin/low pressure (TM/LP), low reactor coolant system (RCS) flow, high pressurizer pressure, low pressurizer pressure (TM/LP floor), low steam generator pressure, and low steam generator water level. Control systems are also simulated including rod control and pressurizer pressure control. Parts of the safety injection system (SIS), including the safety injection tanks, may also be modeled. RETRAN approximates the transient value of DNBR based on input from the core thermal limits.

RETRAN is further discussed in References 5 and 6.

TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code that was patterned after steady-state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions.

The code uses six delayed neutron groups and contains a detailed multiregion fuel-cladding-coolant heat transfer model for calculating point-wise Doppler and moderator feedback effects. The code handles up to 8000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided; e.g., channel-wise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 7.

CESEC

The CESEC digital computer program provides for the simulation of a Combustion Engineering (CE) NSSS. The program can calculate the plant response for non-LOCA initiating events for a wide range of operating conditions.

The primary system components considered in the code include the reactor vessel, the reactor core, the primary coolant loops, the pressurizer, the steam generators, and the reactor coolant pumps. The secondary system components include the secondary side of the steam generators, the main steam system, the feedwater system, and the various steam control valves. In addition, the program models those plant control and protections systems needed to perform the analysis.

CESEC is further described in Reference 8.

CEFLASH-4A

CEFLASH-4A calculates the RCS thermal-hydraulic response during the blowdown phase of the LBLOCA. Its primary functions are to supply hot assembly boundary conditions to the STRIKIN-II computer code and to supply break flow mass and energy to the COMPERC-II computer code.

The RCS is modeled as a network of nodes (control volume) joined by flowpaths. In each of the nodes at each time step, CEFLASH-4A solves the equations for conservation of mass and energy and the fluid equation of state. In each of the flowpaths at each time step, the conservation of momentum equation is solved to yield the local, instantaneous flow rates. Additional models in CEFLASH-4A include a point kinetics model with moderator and Doppler reactivities, a core heat transfer model similar to that in STRIKIN-II, a cladding oxidation model, models for steam generator and metal wall heat transfer, a reactor coolant pump model, and a critical flow model for calculating break flow rate.

COMPERC-II

COMPERC-II calculates the RCS thermal-hydraulic response during the refill of reflood phases of the LBLOCA. COMPERC-II also calculates the containment pressure during the blowdown, refill, and reflood phases. Its major functions are to calculate the reflood heat transfer coefficients used in STRIKIN-II and to determine the boundary conditions used in PARCH.

COMPERC-II models the reactor vessel with the following six regions:

- Subcooled fluid in the annulus and lower plenum
- Subcooled fluid in the core
- Two-phase fluid in the core
- Steam region in the core
- Steam region above the core
- Core barrel / core baffle bypass region

The RCS piping, steam generators, and reactor coolant pumps are modeled as a resistance network for the purpose of calculating steam flow to the break.

In addition, COMPERC-II has component models for core-to-coolant heat transfer, cladding oxidation, safety injection flow rates, interaction of safety injection liquid and nitrogen with steam in the RCS cold legs, FLECHT-based reflood heat transfer coefficients, and containment pressure.

COMZIRC

COMZIRC calculates the amount of core-wide cladding oxidation for the LBLOCA. It is a derivative of the COMPERC-II code.

STRIKIN-II (LBLOCA)

STRIKIN-II calculates the transient response of the hot rod in the core during the LBLOCA. Its major function is to calculate the PCT and maximum cladding oxidation of the hot rod.

STRIKIN-II solves in the axial direction the one-dimensional conservation of energy equation and the equation of state for the fluid. In the fuel rod, STRIKIN-II solves in the radial direction the one-dimensional cylindrical heat conduction equation. The conduction model illustrates the fuel, gap, and cladding. The gap conductance model illustrates heat conduction through the pellet-to-cladding contact points and conduction and radiation through the gas gap.

Additional models in STRIKIN-II include models for core-to-coolant heat transfer for sub-cooled forced convection through film boiling as well as rod-to-rod radiation, cladding swelling and rupture, and cladding oxidation.

PARCH (LBLOCA)

PARCH calculates the steam cooling heat transfer coefficients that are used in STRIKIN-II to cool the hot rod at and above the elevation of cladding rupture after the core reflood rate falls below 1 inch/second.

The original purpose of PARCH was to calculate cladding temperatures for SBLOCAs under pool boiling conditions. The code was subsequently modified to enable PARCH to calculate steam cooling heat transfer coefficients for use in LBLOCA analyses. It has a three-region radial conduction model (fuel pellet, cladding, and coolant), and it also has models for pool boiling and steam cooling heat transfer, variable gap conductance, cladding rupture and swelling, and cladding oxidation. When used in a LBLOCA analysis, it calculates turbulent flow steam cooling heat transfer coefficients based on steam flows calculated by COMPERC-II and normalized blocked channel steam flow rates calculated by HCROSS.

HCROSS

HCROSS calculates the normalized steam flow in the hot channel at and above the elevation of cladding rupture. HCROSS solves a two-dimensional conservation of momentum equation for the blocked hot channel and a parallel unblocked channel including the crossflow between the two channels. The HCROSS results are input to the PARCH code.

CEFLASH-4AS

CEFLASH-4AS calculates the thermal-hydraulic response of the RCS during the small-break LOCA transient. Its primary function is to provide boundary conditions to the hot rod heatup analysis performed by the STRIKIN-II and PARCH computer codes. CEFLASH-4AS is a derivative of the CEFLASH-4A computer code that is used in the large-break LOCA analysis.

Like CEFLASH-4A, CEFLASH-4AS represents the RCS as a network of nodes connected by flow paths. Likewise, the CEFLASH-4AS thermal-hydraulic model is a three equation (mass, energy, and momentum) model.

Unlike CEFLASH-4A, which treats the fluid inventory in each node as a homogeneous mixture, CEFLASH-4AS treats the liquid in each of the RCS nodes as a heterogeneous mixture of two-phase liquid and steam. The phase separation in each node is calculated using a bubble rise model with a pressure dependent value for the bubble drift velocity.

In addition to the phase separation model, CEFLASH-4AS has other models appropriate for representing phenomenon important to small break LOCAs that are not in CEFLASH-4A. Examples include different core and steam generator heat transfer models and a heterogeneous flow path model. Also, unlike CEFLASH-4A, CEFLASH-4AS does not represent momentum flux effects in the solution of the momentum equation.

STRIKIN-II (SBLOCA)

STRIKIN-II calculates the fuel and cladding temperatures of the hot rod during the initial, forced convection period of the small break LOCA transient. The two major functions of STRIKIN-II are to determine whether the hot rod experiences departure from nucleate boiling and to provide the fuel and cladding temperatures to initialize the PARCH computer code. See the description of STRIKIN-II in the listing of the large-break LOCA evaluation model computer codes for additional detail.

PARCH (SBLOCA)

PARCH calculates the fuel and cladding temperatures of the hot rod during the pool boiling period of the small-break LOCA transient. The major function of PARCH is to calculate the hot rod peak cladding temperature and maximum cladding oxidation.

PARCH models a single fuel rod that is divided into a number of axial segments. The energy balance is solved at the end points of each segment. The energy balance is performed in two parts, first, in the core two-phase mixture region and, secondly, in the steam region of the core. The energy balance performed in the two-phase mixture region is used to obtain the core boil-off steam flow rate that is used in the energy balance for the steam region.

PARCH uses a simplified three-region radial conduction model. The three regions are the fuel pellet, the cladding, and the coolant. The heat conduction equation is solved at every time step for each region at each axial node. The three regions are represented by their average temperatures, however, the fuel pellet is modeled with a parabolic temperature distribution. Unlike STRIKIN-II, the fuel pellet-cladding gap is not represented as a separate region. Instead, the gap is represented as a thermal resistance for radial heat transfer between the fuel pellet and the cladding.

PARCH models the following heat transfer regimes in the two-phase region: nucleate boiling, transition boiling, and film boiling. In the steam region, PARCH models steam cooling for laminar flow, transition flow, and turbulent flow. Also, PARCH models thermal radiation heat transfer to steam in the steam region of the core.

PARCH has component models to calculate the gap conductance, cladding rupture and strain, and cladding oxidation.

VIPRE-W

VIPRE-W predicts the 3-D velocity, pressure, and thermal energy fields and fuel rod temperatures for single- and two-phase flow. It solves the finite difference equations for mass, energy, and momentum conservation for an interconnected array of channels, assuming an incompressible, thermally expandable homogeneous flow. The equations are solved with no time step or channel size restrictions for stability. Although the formulation is homogeneous, nonmechanistic models are included for subcooled boiling and vapor/liquid slip in two-phase flow.

Like most other core thermal-hydraulic codes, the VIPRE-W modeling structure is based on a subchannel analysis. The core or section of symmetry is defined as an array of parallel flow channels with lateral connections between adjacent channels. A channel may represent a true subchannel within a rod array, a closed tube, or a larger flow area representing several subchannels or rod bundles. The shape and size of the channels and their interconnections are essentially arbitrary. The user has a great deal of flexibility for modeling reactor cores or any other fluid flow geometry.

The VIPRE code has been updated to include the ABB-NV and ABB-TV critical heat flux correlations (Reference 9).

FATES3B

The FATES code calculates the radial and axial steady state temperature distribution through a single fuel rod using specified values of the rod linear heat rate and coolant flow rate. The effects of fission gas release, fuel swelling, densification and relocation, and clad creep are treated.

5.1.0.9 Classification of Events

The American Nuclear Society (ANS) classification of plant conditions divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public (Reference 1). The four categories are used to define acceptance criteria and are defined as follows:

- Condition I: Normal operation and operational transients
- Condition II: Faults of moderate frequency
- Condition III: Infrequent faults
- Condition IV: Limiting faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Functioning of the reactor trip system and engineered safeguards is assumed to the extent allowed by considerations, such as the single failure criterion, in fulfilling this principle.

5.1.0.9.1 Condition I – Normal Operation and Operational Transients

Condition I occurrences are those that are expected frequently or regularly in the course of normal plant operation, refueling, and maintenance. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter that would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

A typical list of Condition I events is listed below:

A. Steady-state and shutdown operations

1. Power operation (Keff ≥ 0.99, > 5 percent of rated thermal power, Tavg ≥ 325°F)
2. Startup (Keff ≥ 0.99, ≤ 5 percent of rated thermal power, Tavg ≥ 325°F)
3. Hot standby (Keff < 0.99, Tavg ≥ 325°F, shut down cooling (SDC) system isolated)
4. Hot shutdown (Keff < 0.99, 325°F ≥ Tavg ≥ 200°F, SDC system in operation)
5. Cold shutdown (Keff < 0.99, Tavg ≤ 200°F, SDC system in operation)
6. Refueling (Keff < 0.95, Tavg ≤ 140°F, SDC system in operation)

B. Operation with permissible deviations

Various deviations that may occur during continued operation as permitted by the plant technical specifications are considered in conjunction with other operational modes. These include:

1. Operation with components or systems out of service

2. Leakage from fuel with cladding defects

3. Radioactivity in the reactor coolant

    a. Fission products
    b. Corrosion products
    c. Tritium

4. Operation with steam generator leaks up to the maximum allowed by the technical specifications

5. Testing as allowed by the technical specifications

C. Operational transients

1. Plant heatup and cooldown (up to 100°F/hr heatup for the RCS; 200°F/hr cooldown for the pressurizer)

2. Step load changes (up to ± 10 percent)

3. Ramp load changes (up to 5 percent/min)

4. Load rejection up to and including the design load rejection transient

5.1.0.9.2 Condition II - Faults of Moderate Frequency

ANS Condition II occurrences are faults that may occur with moderate frequency during the life of the plant. They are accommodated with, at most, a reactor shutdown with the plant being capable of returning to operation after a corrective action. In addition, no ANS Condition II occurrence shall cause consequential loss of function of fuel cladding and reactor coolant system barriers.

Criteria established for Condition II events include the following:

- Condition II incidents shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action.

- A single Condition II incident shall not cause consequential loss of the function of any barrier to the escape of radioactive products.

- For a Condition II event, any release of radioactive materials in effluents to unrestricted areas shall be in conformance with the Code of Federal Regulations (CFR) Paragraph 20.1 of 10 CFR Part 20, "Standards for Protection Against Radiation."

- By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV classification without other incidents occurring independently.

The following faults are included in this category:

A. Feedwater system malfunction causing a reduction in feedwater temperature

B. Feedwater system malfunction causing an increase in feedwater flow

C. Excessive increase in secondary steam flow

D. Inadvertent opening of a steam generator relief or safety valve

E. Loss of external load

F. Turbine trip

H. Loss of condenser vacuum and other events resulting in turbine trip

G. Inadvertent closure of main steam isolation valves boiling water reactor ((BWR) event)

I. Steam pressure regulator failure

J. Loss of non-emergency AC power to the station auxiliaries

K. Loss of normal feedwater flow

L. Transients resulting from the malfunction of one steam generator

M. Partial loss-of forced reactor coolant flow

N. Uncontrolled CEA bank withdrawal from a subcritical or low-power startup condition

O. Uncontrolled CEA bank withdrawal at power

P. CEA misoperation - dropped assembly, dropped assembly bank, or statically misaligned CEA

Q. Startup of an inactive reactor coolant loop at an incorrect temperature

R. Chemical and volume control system (CVCS) malfunction that results in a decrease in boron concentration in the reactor coolant

S. Inadvertent operation of emergency core cooling system (ECCS) during power operation

T. CVCS malfunction that increases reactor coolant inventory

U. Inadvertent opening of a pressurizer safety or relief valve

V. Failure of small lines carrying primary coolant outside containment

5.1.0.9.3 Condition III - Infrequent Faults

ANS Condition III occurrences are faults that may occur very infrequently during the life of the plant. They may be accompanied by the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. An ANS Condition III occurrence will not, by itself, generate an ANS Condition IV fault or result in a consequential loss of function of the RCS or containment barriers.

Criteria established for Condition III events include the following:

- Condition III incidents shall not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude resumption of operation for a considerable outage time.

- A Condition III incident shall not, by itself, result in a consequential loss of function of the RCS or reactor containment barriers.

- A Condition III incident shall not, by itself, generate a Condition IV fault.

- The release of radioactive material due to Condition III incidents may exceed the guidelines of 10 CFR Part 20, "Standards for Protection Against Radiation", but shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. This is typically interpreted as a small fraction of the 10 CFR Part 100 guidelines.

The following faults are included in this category:

A. Minor steam system piping failures

B. Complete loss of forced reactor coolant flow

C. CEA misoperation - single CEA withdrawal at full power

D. Inadvertent loading and operation of a fuel assembly in an improper position

E. Loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuates the ECCS

F. Waste gas system failure

G. Radioactive liquid waste system leak or failure (atmospheric release)

H. Postulated radioactive releases due to liquid containing tank failure

I. Spent fuel cask drop accidents

5.1.0.9.4 Condition IV - Limiting Faults

ANS Condition IV occurrences are faults that are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences that must be designed against and represent limiting design cases.

Criteria established for Condition IV events include the following:

- Condition IV faults shall not cause a release of radioactive material that results in an undue risk to public health and safety exceeding the guidelines of 10 CFR 100, "Reactor Site Criteria."
- A single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault, including those of the reactor coolant system (RCS) and the reactor containment system.

The following faults have been classified in this category:

A. Major steam system piping failure

B. Feedwater system pipe break

C. Reactor coolant pump rotor seizure (locked rotor)

D. Reactor coolant pump shaft break

E. Spectrum of CEA ejection accidents

F. Steam generator tube rupture

G. LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary

H. Design basis fuel handling accident

5.1.0.10 RETRAN Code

Please refer to Appendix C.

5.1.0.11 References

1. ANSI/ANS N18.2-1973, "American National Standard for the Design or Stationary Pressurized Water Reactor Plants."

2. Friedland, A. J and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-Proprietary), April 1989.

3. Intentionally Left Blank.

4. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908-A, December 1989.

5. McFadden, J. H., et al., "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCMA.

6. Huegel, D. S., et al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A, April 1999.

7. Risher, D. H., Jr. and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

8. Letter, A. E. Scherer, Enclosure 1-P to LD-82-001, "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981.

9. Sung, Y. X., et al., "Addendum 1 to WCAP-14565-P-A Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," WCAP-14565, Addendum 1 (Proprietary), June 2002.

10. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989

Table 5.1.0-1 Nuclear Steam Supply System Power Ratings	
Rated Reactor Core Thermal Power Output, MWt	2700
NSSS Rated Thermal Power, MWt	2714.2
Thermal Power Generated by the Reactor Coolant Pumps, MWt	
Nominal	14.2
Maximum	20

Table 5.1.0-2 Summary of Initial Conditions and Computer Codes (Sheet 1 of 6)

	Event	Computer Codes Used	Moderator Density Coefficient (Δk/gm/cc)	Moderator Temperature Coefficient (pcm/°F)	Doppler Feedback	DNB Correlation	Revised Thermal Design Procedure	Initial Core Thermal Power (MWt)
15.1	**Increase in Heat Removal by the Secondary System**							
	Decrease in Feedwater Temperature	RETRAN	0.43	N/A	Upper Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	Increase in Feedwater Flow Rate	RETRAN	0.43	N/A	Upper Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	Excessive Increase in Main Steam Flow	Bounded by 5.1.6 and 5.1.5	N/A	N/A	N/A	N/A	N/A	N/A
	Inadvertent Opening of an SG Relief or Safety Valve	Bounded by 5.1.6 and 5.1.5	N/A	N/A	N/A	N/A	N/A	N/A
	Pre-Trip Steamline Break	RETRAN VIPRE, ANC	0.0, 0.1, 0.2, 0.3, 0.43	N/A	Upper Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	Post-Trip Steamline Break	RETRAN VIPRE, ANC	See Figure 5.1.6-1	N/A	See Figure 5.1.6-2	W-3	No	0 (subcritical)
15.2	**Decrease in Heat Removal by the Secondary System**							
	Loss of Condenser Vacuum – Overpressure Case	RETRAN	N/A	See Figure 5.1.0-7	Upper Curve of Figure 5.1.0-6	N/A	No	2754.4
	Loss of Condenser Vacuum – DNB Case	RETRAN VIPRE	N/A	See Figure 5.1.0-7	Upper Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	Loss of Non-Emergency AC to the Station Auxiliaries	Bounded by 5.1.10 and 5.1.14	N/A	N/A	N/A	N/A	N/A	N/A
	Loss of Normal Feedwater Flow	Bounded by 5.1.10 and 5.1.14	N/A	N/A	N/A	N/A	N/A	N/A
	Feedwater System Pipe Rupture – Overpressure Case	RETRAN	N/A	See Figure 5.1.0-7	Upper Curve of Figure 5.1.0-6	N/A	No	2754.4
	Feedwater System Pipe Rupture – DNB Case	RETRAN	N/A	See Figure 5.1.0-7	Upper Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	Asymmetric Steam Generator Transient	RETRAN VIPRE, ANC	N/A	See Figure 5.1.0-7	Upper Curve of Figure 5.1.0-6	ABB-NV	Yes	2700

Table 5.1.0-2 Summary of Initial Conditions and Computer Codes (Sheet 2 of 6) (cont.)

	Event	Reactor Coolant Pump Heat (MWt)	Reactor Vessel Flow Rate (gpm)	Vessel T-avg (°F)	Pressurizer Pressure (psia)	Pressurizer Water Level (%)	Feedwater Temp. (°F)	SG Tube Plugging Level (%)
15.1	**Increase in Heat Removal by the Secondary System**							
	Decrease in Feedwater Temperature	14.2	341,400	576.5	2225	63.0	435	0.0
	Increase in Feedwater Flow Rate	14.2	341,400	576.5	2225	63.0	435	0.0
	Excessive Increase in Main Steam Flow	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	Inadvertent Opening of an SG relief or Safety Valve	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	Pre-Trip Steamline Break	14.2	341,400	576.5	2225	63.0	435	0.0
	Post-Trip Steamline Break	0.0	335,000	532.0	2250	33.1	240	0.0
15.2	**Decrease in Heat Removal by the Secondary System**							
	Loss of Condenser Vacuum – Overpressure Case	20.0	335,000	579.5	2180	65.0	435	30
	Loss of Condenser Vacuum – DNB Case	14.2	341,400	576.5	2225	65.0	435	30
	Loss of Non-Emergency AC to the Station Auxiliaries	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	Loss of Normal Feedwater Flow	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	Feedwater System Pipe Rupture – Overpressure Case	14.2	335,000	560.0	2180	70.0	435	30
	Feedwater System Pipe Rupture – DNB Case	14.2	341,000	576.5	2225	70.0	435	30
	Asymmetric Steam Generator Transient	14.2	341,400	576.5	2225	65.0	435	30

Table 5.1.0-2 Summary of Initial Conditions and Computer Codes (Sheet 3 of 6) (cont.)

	Event	Computer Codes Used	Moderator Density Coefficient (Δk/gm/cc)	Moderator Temperature Coefficient (pcm/°F)	Doppler Feedback	DNB Correlation	Revised Thermal Design Procedure	Initial Core Thermal Power (MWt)
15.3	**Decrease in RCS Flow Rate**							
	Partial/Complete Loss of Forced Flow	RETRAN VIPRE	N/A	See Figure 5.1.0-7	Lower Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	Reactor Coolant Pump Seized Rotor/Shaft Break – DNB Case	RETRAN VIPRE	N/A	See Figure 5.1.0-7	Lower Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	Reactor Coolant Pump Seized Rotor/Shaft Break – Overpressure/PCT Case	RETRAN	N/A	See Figure 5.1.0-7	Lower Curve of Figure 5.1.0-6	N/A	No	2754.4
15.4	**Reactivity and Power Distribution Anomalies**							
	Uncontrolled CEA Bank Withdrawal from Subcritical	TWINKLE FACTRAN VIPRE	N/A	5.0	900 pcm	ABB-NV	No	0.0 (subcritical)
	Uncontrolled CEA Bank Withdrawal at Power	RETRAN	0.43	See Figure 5.1.0-7	Upper & Lower Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	CEA Misoperation (Dropped Rod)	RETRAN VIPRE, ANC	N/A	0.0 to –35.0	Lower Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	Startup of an Inactive Loop at an Incorrect Temperature	Precluded by Technical Specifications	N/A	N/A	N/A	N/A	N/A	N/A
	CEA Ejection	TWINKLE FACTRAN	N/A	See Note 1	900 pcm	N/A	No	0.0 & 2754.4

Table 5.1.0-2 Summary of Initial Conditions and Computer Codes (Sheet 4 of 6) (cont.)

	Event	Reactor Coolant Pump Heat (MWt)	Reactor Vessel Flow Rate (gpm)	Vessel T-avg (°F)	Pressurizer Pressure (psia)	Pressurizer Water Level (%)	Feedwater Temp. (°F)	SG Tube Plugging Level (%)
15.3	**Decrease in RCS Flow Rate**							
	Partial/Complete Loss of Forced Flow	14.2	341,400	576.5	2225	63.0	435	30
	Reactor Coolant Pump Seized Rotor/Shaft Break – DNB Case	14.2	341,400	576.5	2225	63.0	435	30
	Reactor Coolant Pump Seized Rotor/Shaft Break – Overpressure/PCT Case	14.2	335,000	579.5	2395	63.0	435	30
15.4	**Reactivity and Power Distribution Anomalies**							
	Uncontrolled CEA Bank Withdrawal from Subcritical	N/A	335,000	532.0	2205	N/A	N/A	N/A
	Uncontrolled CEA Bank Withdrawal at Power	14.2	341,400	576.5	2225	63.0	435	30
	CEA Misoperation (Dropped Rod)	20.0	341,400	576.5	2250	63.0	435	30
	Startup of an Inactive Loop at an Incorrect Temperature	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	CEA Ejection	N/A	335,000	532.0 579.5	2205.0	N/A	N/A	N/A

Table 5.1.0-2 Summary of Initial Conditions and Computer Codes (Sheet 5 of 6) (cont.)

	Event	Computer Codes Used	Moderator Density Coefficient (Δk/gm/cc)	Moderator Temperature Coefficient (pcm/°F)	Doppler Feedback	DNB Correlation	Revised Thermal Design Procedure	Initial Core Thermal Power (MWt)
15.5	**Increase in Coolant Inventory**							
	Inadvertent ECCS Operation at Power	Precluded by SIS Design	N/A	N/A	N/A	N/A	N/A	N/A
	CVCS Malfunction	RETRAN	0.43	N/A	Lower Curve of Figure 5.1.0-6	N/A	No	2754.4
15.6	**Decrease in Coolant Inventory**							
	Inadvertent RCS Depressurization	RETRAN	N/A	See Figure 5.1.0-7	Lower Curve of Figure 5.1.0-6	ABB-NV	Yes	2700
	Steam Generator Tube Rupture	CESEC	N/A	N/A	N/A	N/A	No	2754
	LOCAs	Section 5.2	Section 5.2	Section 5.2	Section 5.2	Section 5.2	Section 5.2	Section 5.2
	Primary Line Break Outside Containment	CESEC	N/A	N/A	N/A	N/A	No	2754

Table 5.1.0-2 Summary of Initial Conditions and Computer Codes (Sheet 6 of 6) (cont.)

	Event	Reactor Coolant Pump Heat (MWt)	Reactor Vessel Flow Rate (gpm)	Vessel T-avg (°F)	Pressurizer Pressure (psia)	Pressurizer Water Level (%)	Feedwater Temp. (°F)	SG Tube Plugging Level (%)
15.5	**Increase in Coolant Inventory**							
	Inadvertent ECCS Operation at Power	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	CVCS Malfunction	20.0	335,000	579.5	2180	61.0	435	30
15.6	**Decrease in Coolant Inventory**							
	Inadvertent RCS Depressurization	14.2	341,400	576.5	2225	63.0	435	30
	Steam Generator Tube Rupture	20.0	335,000	Tcold = 553	2400	70.0	435	30
	LOCAs	Section 5.2	Section 5.2	Section 5.2	Section 5.2	Section 5.2	Section 5.2	Section 5.2
	Primary Line Break Outside Containment	20.0	>335,000	Tcold < 554	<2410	N/A	N/A	<30

NOTES:

1. The following moderator temperature coefficient (MTC) values were assumed in the analysis of the CEA Ejection event:

   Beginning of life (BOL), hot zero power (HZP) MTC=5.0 pcm/°F

   BOL, hot full power (HFP) MTC=0.0 pcm/°F

   End of life (EOL), HZP ITC=-10.0 pcm/°F

   EOL, HFP ITC=-20.0 pcm/°F

Table 5.1.0-3 Nominal Values of Pertinent Plant Parameters Utilized in the Accident Analyses				
Equivalent Steam Generator Tube Plugging Level (all loops)	**0% Hi-T_{avg}**	**30% Hi-T_{avg}**	**0% Lo-T_{avg}**	**30% Lo-T_{avg}**
NSSS Thermal Power (MWt)[1]	2720	2720	2720	2720
Reactor Core Power (MWt)	2700	2700	2700	2700
HFP core inlet temperature (°F)	549.0	549.0	535.0	535.0
HFP vessel average temperature (°F)	576.5	576.5	563.0	563.0
Zero Load Temperature (°F)	532.0	532.0	532.0	532.0
Pressurizer Pressure (psia)	2250.0	2250.0	2250.0	2250.0
Total Reactor Vessel Inlet Flow (thermal design, gpm)[2]	335,000	335,000	335,000	335,000
Total Reactor Coolant Flow (10^6 lb/hr)	126.1	126.1	128.4	128.4
Steam Flow from NSSS (10^6 lb/hr)	11.87	11.83	11.82	11.79
Steam Pressure at Steam Generator Outlet (psia)	857	766	755	672
Maximum Steam Moisture Content (%)	0.25	0.25	0.25	0.25
Assumed Feedwater Temperature at Steam Generator Inlet (°F)	435	435	435	435

Notes:

1. Includes maximum reactor coolant pump heat (20 MWt). Nominal pump heat is 14.2 MWt.
2. Minimum measured flow is 341,400 gpm.

Table 5.1.0-4 Safety Analysis RPS and ESFAS Trip Setpoints and Delay Times (Sheet 1 of 2)		
RPS Trip Function	**Technical Specification Value**	**Analysis Setpoint**
Variable Power Level – High (% Above Initial Power Level)	9.61	10.2
Variable Power Level – Ceiling (% Rated Thermal Power)	107	112.2[(1)]
Variable Power Level – Floor (% Rated Thermal Power)	15	23.0
Pressurizer Pressure – High (psia) Normal Environment Harsh Environment[(2)]	2370	 2415 2460
Pressurizer Pressure – Low (Floor of Thermal Margin/Low Pressure), (psia) Normal Environment Harsh Environment[(2)]	1900	 1855 1810
Steam Generator Pressure – Low (psia) Normal Environment Harsh Environment[(2)]	626	 586 546
Steam Generator Pressure – High Difference, (psid)	120	230
Steam Generator Level – Low (% Narrow Range Tap Span)	20.5	1.0
Reactor Coolant Flow – Low (% of Design Flow ΔP)	N/A	85.678[(3)]
Containment Pressure – High (psig)	3.0	4.65
Thermal Margin/Low Pressure (psig)	Tech. Spec. Table 2.2-1	Variable, See Figure 5.1.0-2
Engineered Safety Feature Actuation System (ESFAS) Function		
Safety Injection Actuation Signal (SIAS) on Pressurizer Pressure – Low (psia) Normal Environment Harsh Environment[(2)]	1736	 1646 1578
Main Steamline Isolation Signal (MSIS) on Steam Generator Pressure – Low (psia) Normal Environment Harsh Environment[(2)]	600	 560 520
Main Feedwater Isolation on Steam Generator Pressure – Low (psia) Normal Environment Harsh Environment[(2)]	600	 560 520
Auxiliary Feedwater Isolation on Steam Generator Pressure Difference – High (psid)	275	360

1. Rod shadowing and downcomer temperature decalibration effects on excore detector signals are applied independently.
2. Harsh environment setpoints apply to inside containment steamline and feedwater line breaks.
3. Setpoint is equivalent to 91.9% of nominal flow for a single loop loss-of-flow event.

Table 5.1.0-4 Safety Analysis RPS and ESFAS Trip Setpoints and Delay Times (Sheet 2 of 2) (cont.)			
RPS Trip Function	**Sensor Response Time Constant (sec)**	**Processing Delay (sec)**	**Total Delay Time (sec)**
Variable High Power		0.4	
Excore Neutron Power detectors	<0.01[4]		0.4
Hot-Leg and Cold-Leg RTDs (Thermal power calculation)	8.0		8.4
Pressurizer Pressure – High	0.75	0.4	1.15
Thermal Margin/Low Pressure		0.4	
Pressurizer Pressure	0.75		1.15
Hot-Leg and Cold-Leg RTDs (Thermal power calculation)	8.0		8.4
Excore Neutron Power detectors (ASI calculation)	N/A		-
Steam Generator Pressure – Low	0.75	0.4	1.15
Steam Generator Pressure – High Difference	0.75	0.4	1.15
Steam Generator Level – Low	0.75	0.4	1.15
Reactor Coolant Flow – Low	0.25	0.4	0.65
Containment Pressure – High	0.75	0.4	1.15
ESFAS Function			**Total Delay Time (sec)**
Pressurizer Pressure – Low (SI) → To achieve full SI flowrate			30.0
Steam Generator Pressure – Low → To complete MSIV closure			6.75
Steam Generator Pressure – Low → To complete MFIV closure			5.15
Auxiliary Feedwater Isolation On Steam Generator Pressure Difference – High →To complete auxiliary feedwater isolation to affected SG			120
4. Typical response time is 50 μ seconds.			

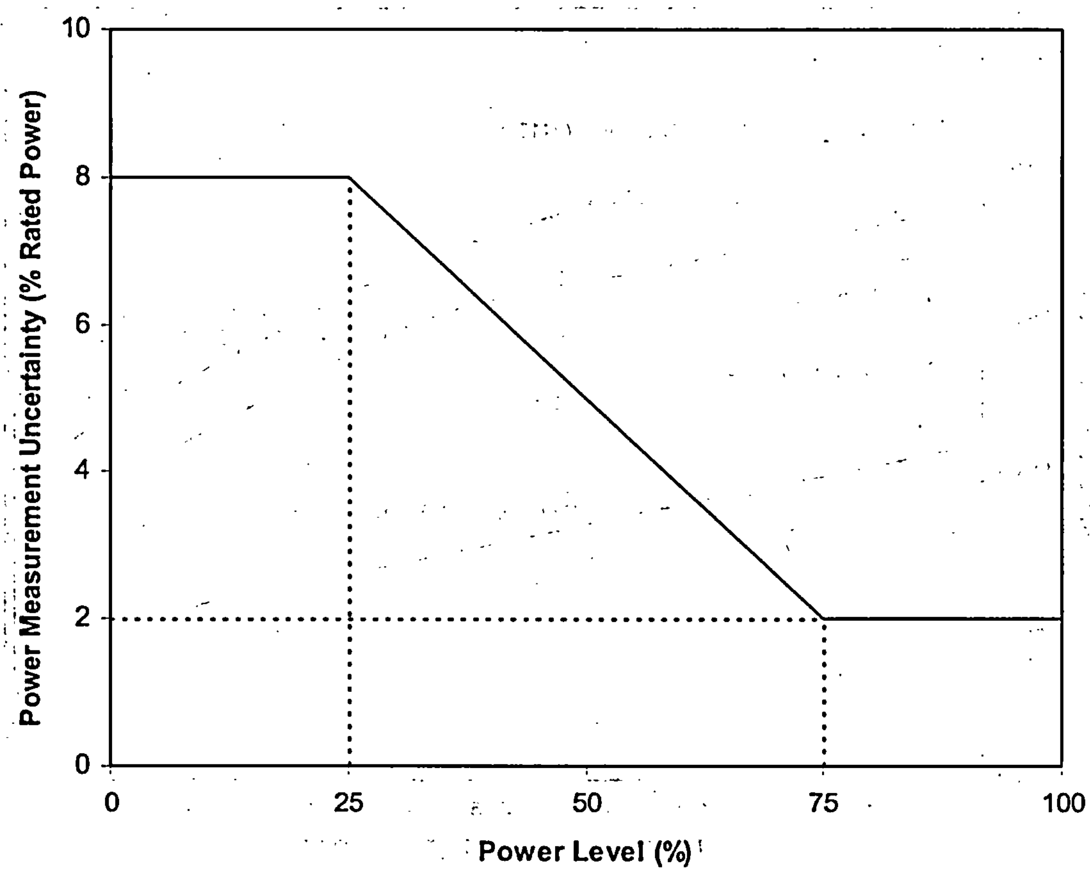

Figure 5.1.0-1 Power Measurement Uncertainty Used in Accident Analysis

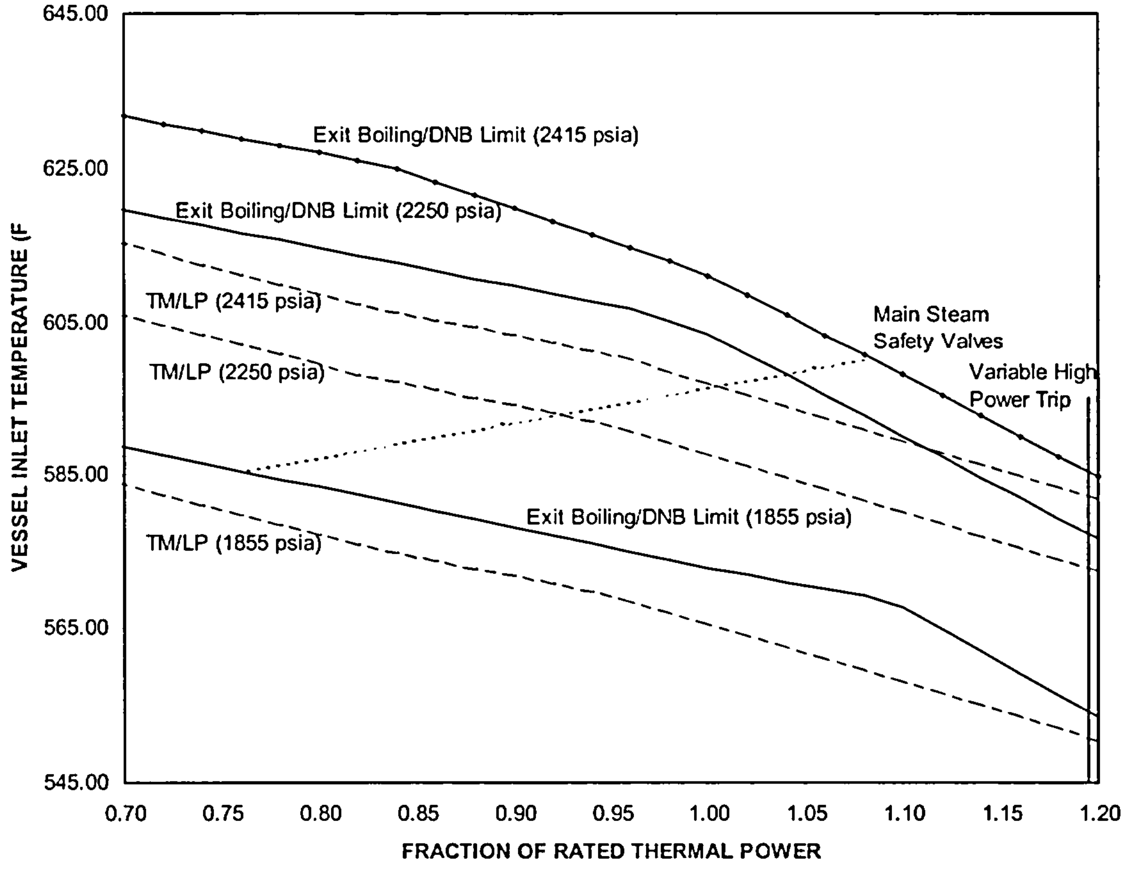

Figure 5.1.0-2 Core Thermal Limits and Protection Functions
(Axial Power Distribution = 1.55 Cosine, ASI – 0.0)

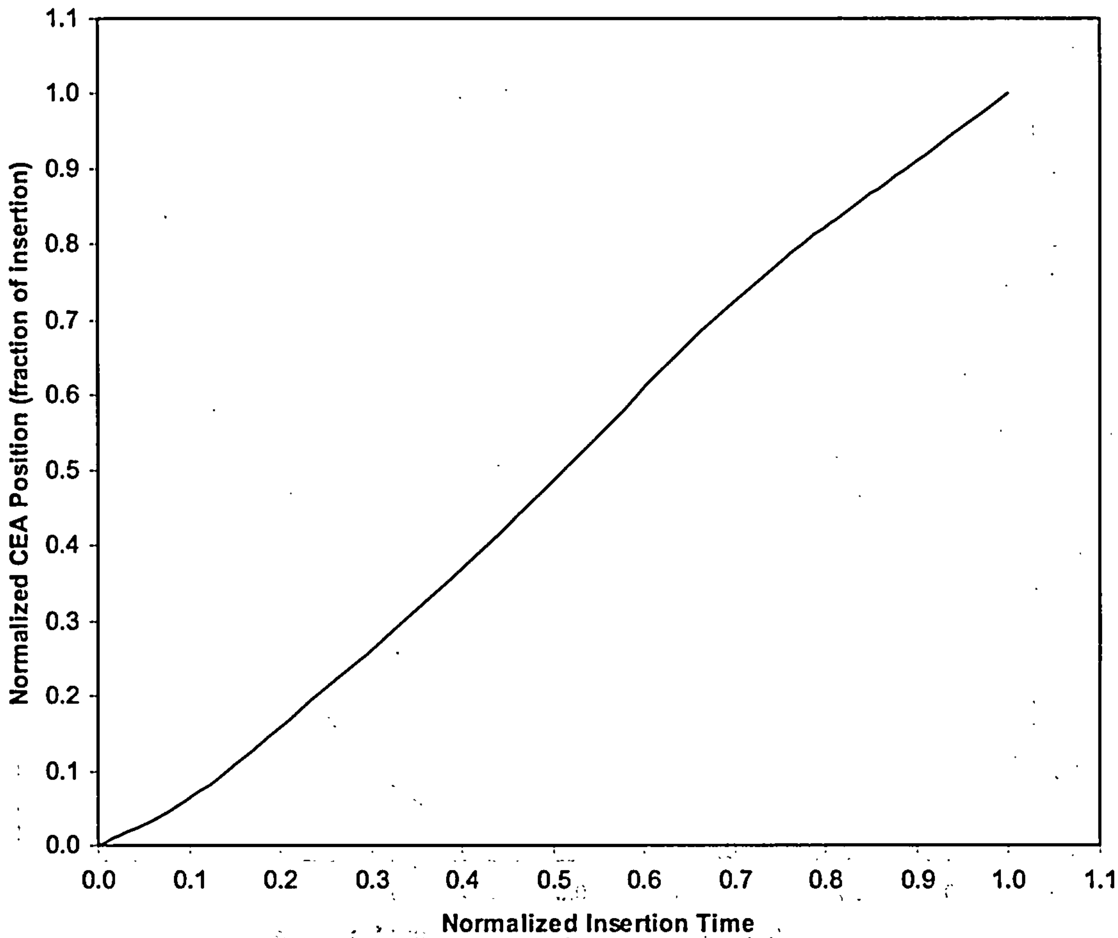

Figure 5.1.0-3 Normalized CEA Position vs. Time

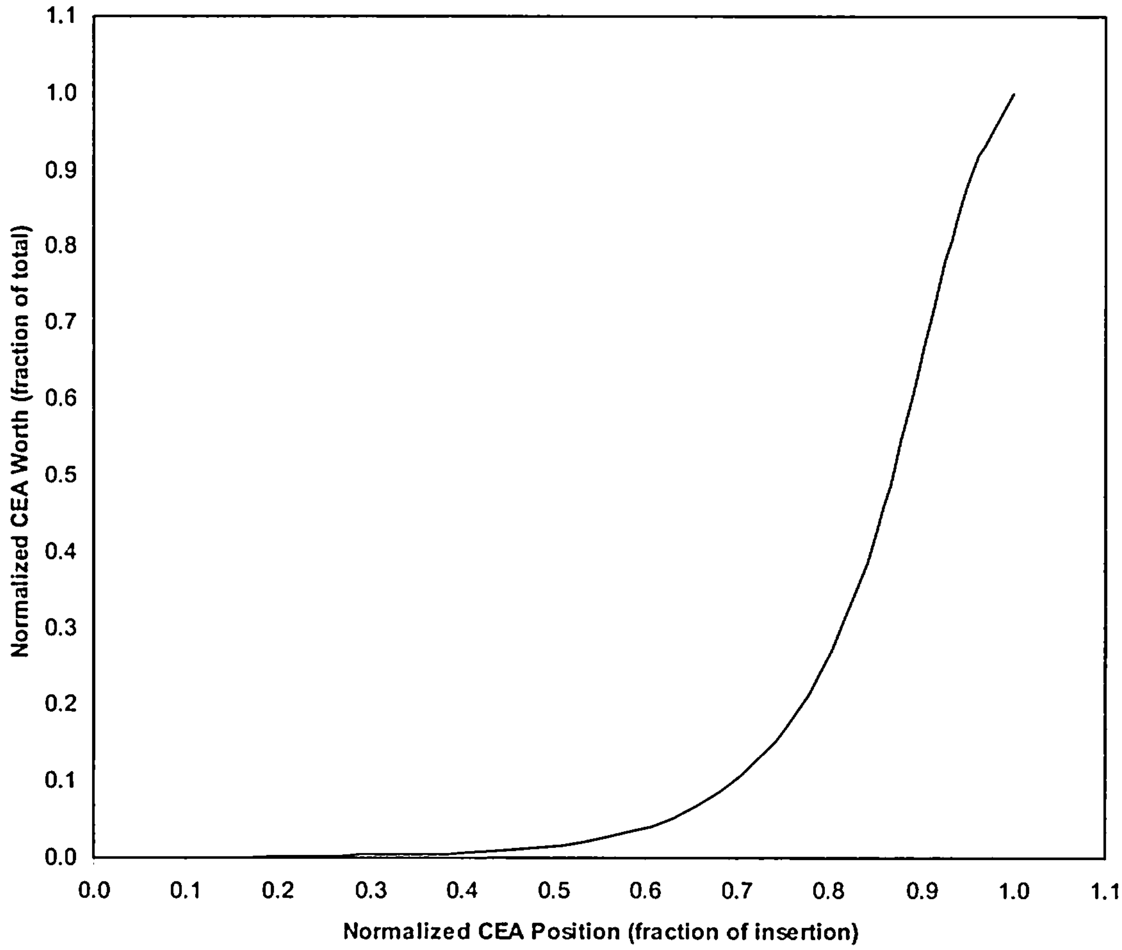

Figure 5.1.0-4 Normalized CEA Worth vs. Position

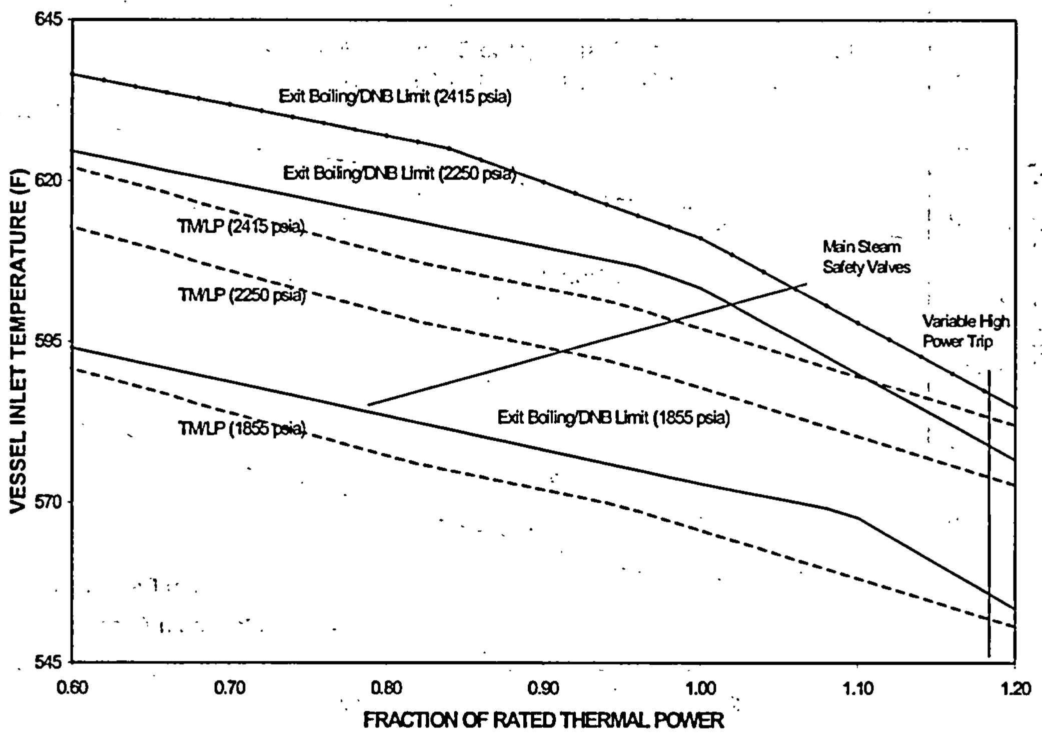

Figure 5.1.0-5 Illustration of the Thermal Margin/Low Pressure Reactor Trip Function and the Reference Core Thermal Limits

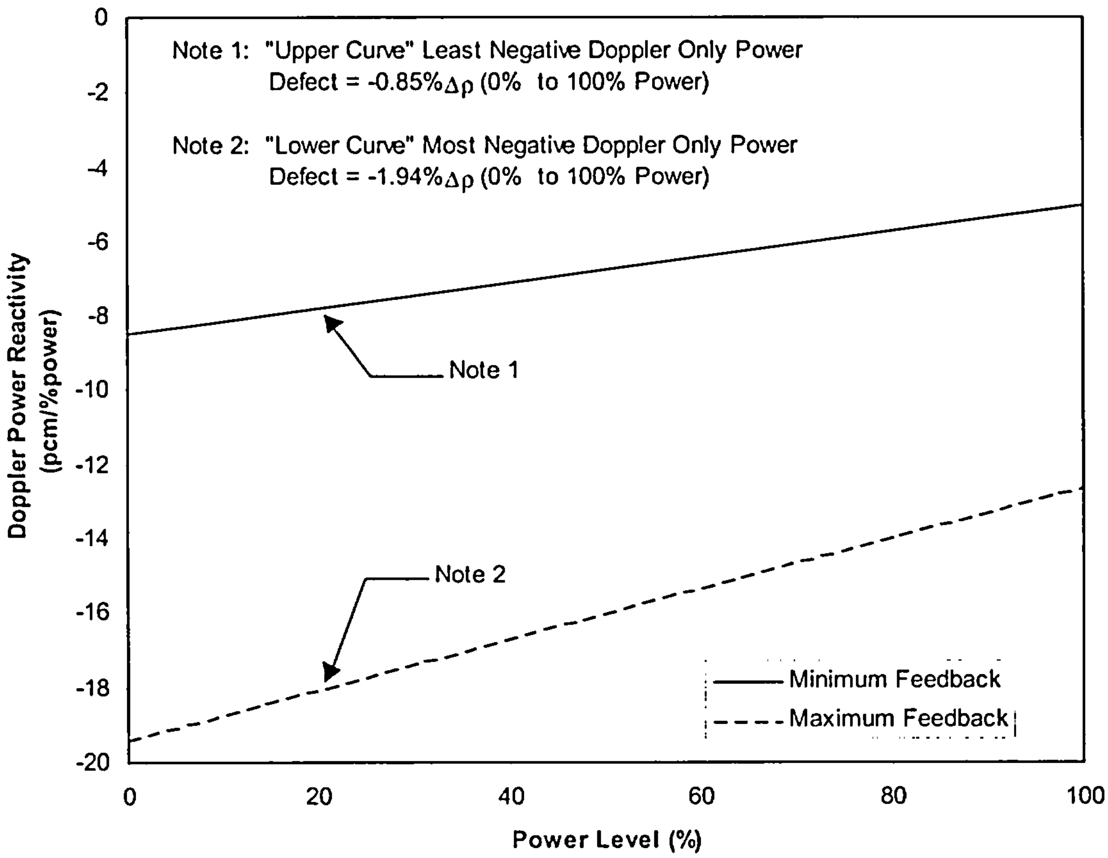

Figure 5.1.0-6 Doppler Power Coefficient Used in Accident Analyses

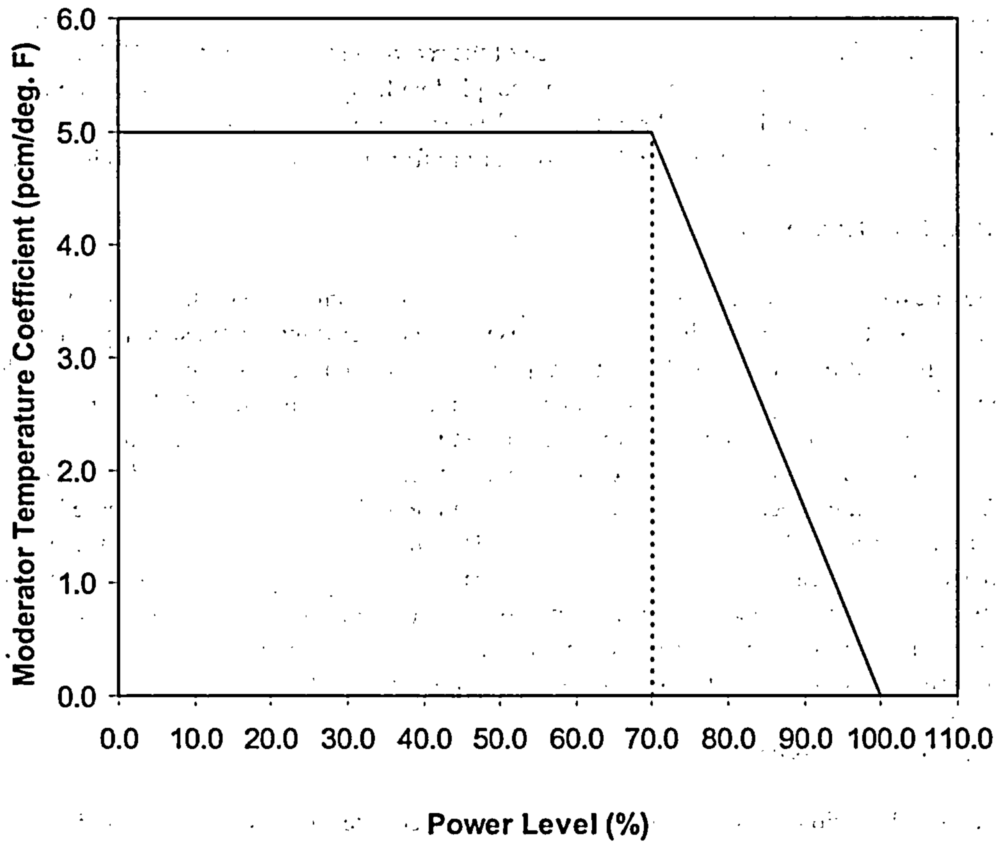

Figure 5.1.0-7 Moderator Temperature Coefficient Used in Accident Analyses

5.1.1 Increase in Feedwater Flow

A change in steam generator feedwater conditions that results in an increase in feedwater flow or a decrease in feedwater temperature could result in excessive heat removal from the plant primary coolant system. Such changes in feedwater flow or feedwater temperature are a result of a failure of a feedwater control valves, feedwater bypass valve, failure in the feedwater control system, or operator error.

The occurrence of these failures that result in an excessive heat removal from the plant primary coolant system cause the primary-side temperature and pressure to decrease significantly. The existence of a negative moderator and fuel temperature reactivity coefficients can cause core reactivity to rise, as the primary-side temperature decreases. In the absence of the reactor protection system (RPS) reactor trip or other protective action, this increase in core power, coupled with the decrease in primary-side pressure, can challenge the core thermal limits. Therefore, the Excessive Heat Removal Due to Feedwater System Malfunctions Event is analyzed to ensure that the departure from nucleate boiling (DNB) is not violated.

5.1.1.1 Accident Description

An example of excessive heat removal from the reactor coolant system (RCS) is the accidental opening of the feedwater regulating valves resulting in an increase of feedwater flow to both steam generators, causing excessive heat removal from the RCS. At power, excess feedwater flow causes a greater load demand on the primary side due to increased subcooling in the steam generator. With the plant at zero-power conditions, the addition of relatively cold feedwater may cause a decrease in primary-side temperature, and, therefore, a reactivity insertion due to the effects of the negative moderator temperature coefficient. The resultant decrease in the average temperature of the core causes an increase in core power due to moderator and control system feedback. This transient is attenuated by the thermal capacity of the primary and secondary sides. If the increase in reactor power is large enough, the primary RPS trip functions [e.g., high neutron flux, variable high-pressure trip (VHPT)] will prevent any power increase that can lead to a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit value. The RPS trip functions may not actuate if the increase in power is not large enough.

5.1.1.2 Method of Analysis

The feedwater malfunction analysis causing an increase in feedwater flow is performed to demonstrate that the DNB design basis is satisfied. This is accomplished by showing that the calculated minimum DNBR is greater than the DNBR safety analysis limit.

The feedwater system malfunction transient is analyzed using the RETRAN code. The RETRAN computer code is a flexible, transient thermal-hydraulic digital computer code, that has been reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) for pressurized water reactor licensing applications (References 1 and 2). The main features of the program include a point kinetics and one-dimensional kinetics model, one-dimensional homogeneous equilibrium mixture thermal-hydraulic model, control system models, and two-phase natural convection heat transfer correlations. The results from the RETRAN computer code are used to determine if the DNB safety analysis limits for the excessive heat removal due to feedwater malfunction event are met.

Feedwater system failures, including the accidental opening of the feedwater control valves, have the potential of allowing increased feedwater flow to each steam generator that will result in excessive heat removal from the RCS. Therefore, it is assumed that the feedwater control valves fail in the full-open position allowing the maximum feedwater flow to both steam generators. Also addressed is the initiation of a feedwater malfunction event from a hot zero-power (HZP) condition.

The following assumptions are made for the analysis of the feedwater malfunction event involving the accidental opening of the feedwater control valves:

1. The plant is operating at full-power (and no-load conditions for the HZP case) conditions with the initial reactor power, pressure, and RCS average temperatures assumed to be at the nominal values.

2. Uncertainties in initial conditions are included in the DNBR limit calculated using the Revised Thermal Design Procedure (RTDP) methodology (Reference 3), where applicable (full-power cases).

3. The feedwater temperature of 435°F for the full-power cases is consistent with normal plant conditions. The feedwater enthalpy assumed at no-load conditions corresponds to a feedwater temperature of 240°F.

4. The excessive feedwater flow event assumes accidental opening of the feedwater control valves with the reactor at full power and zero power while modeling a post-reactor-trip condition with minimum shutdown margin. The feedwater flow malfunction results in a step increase to 120% of the nominal full-power feedwater flow to both steam generators.

5. Maximum (end-of-life) reactivity feedback conditions with a minimum Doppler-only power defect is conservatively assumed.

6. The heat capacity of the RCS metal and steam generator shell are ignored, thereby maximizing the temperature reduction of the RCS coolant.

7. The feedwater flow resulting from a fully open control valve is terminated by the steam generator high-high water level signal or operator action.

The automatic rod control system is not modeled as it is disabled at the plant. The RPS functions to trip the reactor on the appropriate signal. No single active failure will prevent the RPS from functioning properly.

Protection against undesirable conditions is provided by steam generator water level alarms with automatic or manual control actions to reduce feedwater flow, and in extreme cases by reactor trips due to high power (VHPT), low pressurizer pressure, thermal margin/low pressure (TM/LP), or low steam generator pressure.

5.1.1.3 Results

The results of the analyses demonstrate that the hot full-power (HFP) case meets the applicable DNBR acceptance criterion.

The limiting case is the excessive feedwater flow from a full-power initial condition. This case gives the largest reactivity feedback and results in the greatest power increase. The power increases until reactor trip is actuated when the variable high power trip setpoint is reached. The peak core heat flux and minimum DNB condition is reached shortly after the control element assemblies (CEAs) begin to fall. The consequences of the event are mitigated prior to reaching a high steam generator water level condition in either steam generator. However, feedwater will be terminated after reaching the high-high steam generator water level setpoint. The limiting full-power feedwater flow increase conditions were analyzed and it was confirmed that the calculated minimum DNBR is above the safety analysis DNBR limit. Therefore, the applicable DNBR acceptance criterion is met.

The excessive feedwater flow from a zero power condition models a HZP post-trip condition (i.e., HZP stuck rod coefficients, minimum shutdown margin) with maximum reactivity feedback conditions (end of life). The effects of an increased feedwater flow and combined reactivity feedback effect at post-trip conditions is not sufficient enough to offset the impact of the minimum shutdown worth of the CEAs. As a result, there is no return to power, and therefore, no challenge to the minimum DNBR criterion.

Table 5.1.1-1 shows the time sequence of events for the full-power feedwater malfunction transient case resulting in an increase in feedwater flow. Figures 5.1.1-1 through 5.1.1-7 show transient responses for various system parameters during a feedwater system malfunction causing a feedwater flow increase initiated from full-power conditions. As shown in Table 5.1.1-2, the increase feedwater flow at zero power conditions does not result in any return to power condition.

5.1.1.4 Conclusions

The results of the analysis show that the VHPT or TM/LP RPS signal provides adequate protection against the feedwater malfunction transient. No fuel or cladding damage is predicted for this accident. All acceptance criteria are satisfied.

5.1.1.5 References

1. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April, 1999.

2. EPRI NP-1850-CCM, Rev. 6, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," December 1995.

3. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

Table 5.1.1-1 Feedwater System Malfunction Event at Full Power, Increased Feedwater Flow Sequence of Events and Transient Results	
Event	**Time (seconds)**
Main Feedwater Control Valves Fail Full Open	0.0
Variable High Power Trip Setpoint is Reached	38.43
Trip Breakers Open	38.83
CEAs Begin to Drop into Core	39.57
Minimum DNBR Occurs	40.00
Turbine Trip	40.83
Results	
Peak Nuclear Power, fraction of nominal	1.136
Peak Core Heat Flux, fraction of nominal	1.133
Minimum DNBR	1.97

Table 5.1.1-2 Feedwater System Malfunction Event at Zero Power, Increased Feedwater Flow Sequence of Events and Transient Results	
Event	**Time of event, sec**
Main Feedwater Control Valves Fail Full Open	0.0
Hi-Hi Steam Generator Water Level Trip Setpoint is Reached	16.98
Feedwater Isolation Valves Fully Closed	87.11
Results	
Return-to-Power Peak, fraction of nominal	0.0
Peak Core Heat Flux, fraction of nominal	0.011

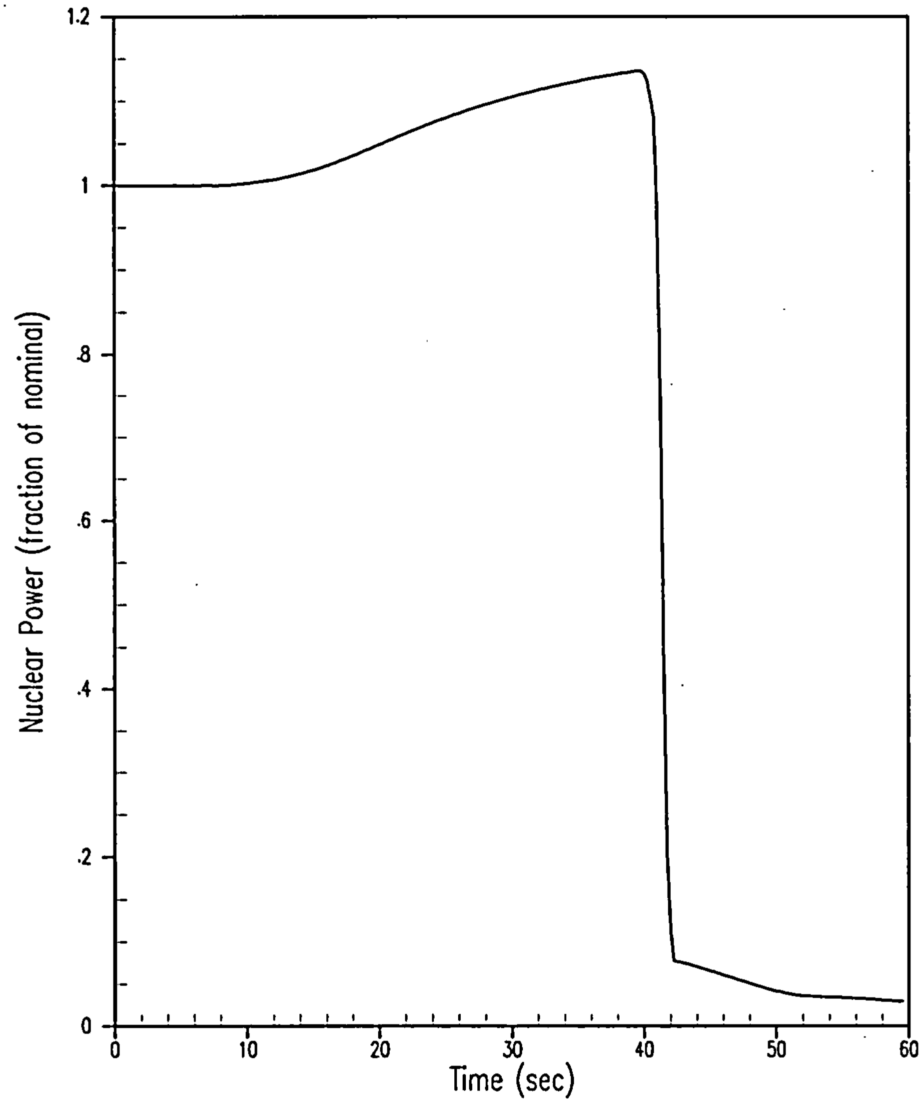

Figure 5.1.1-1 Nuclear Power for a Feedwater Malfunction Increased Feedwater Flow at HFP Initial Conditions

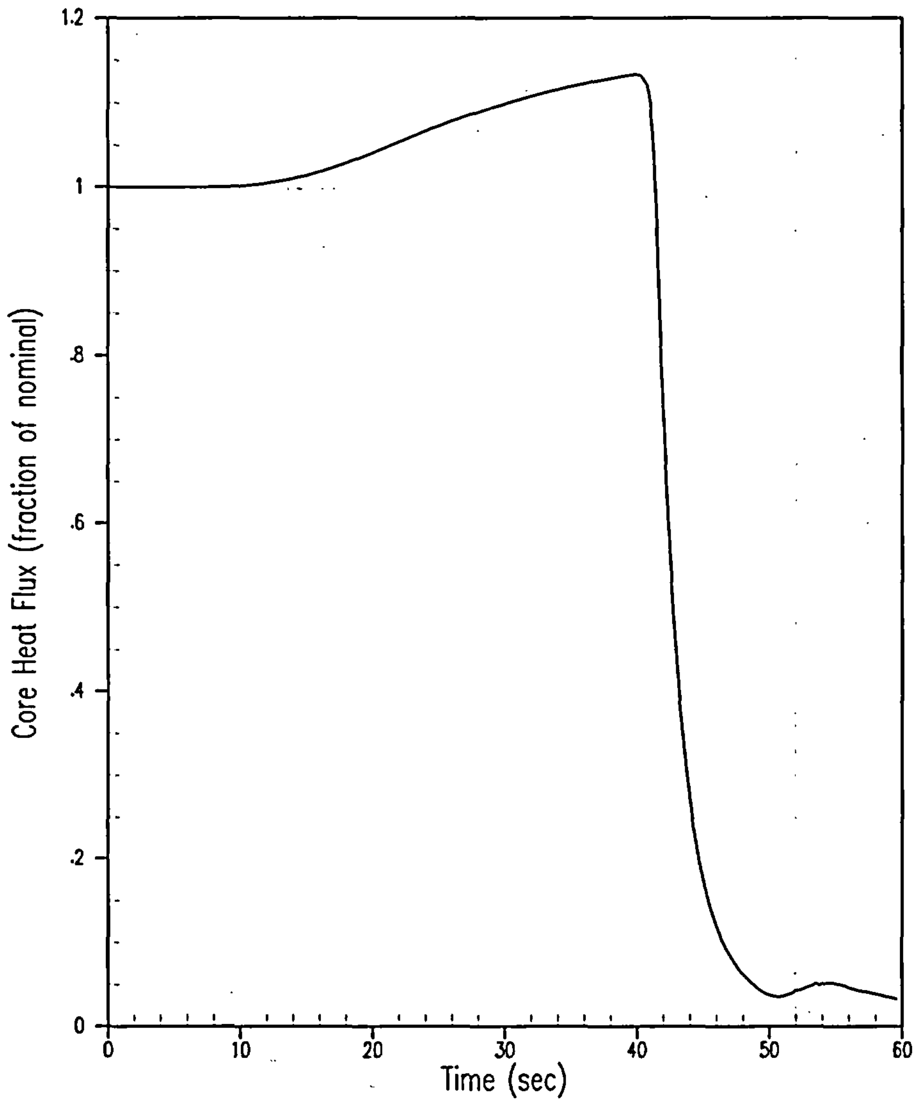

Figure 5.1.1-2 Core Heat Flux for a Feedwater Malfunction Increased Feedwater Flow at HFP Initial Conditions

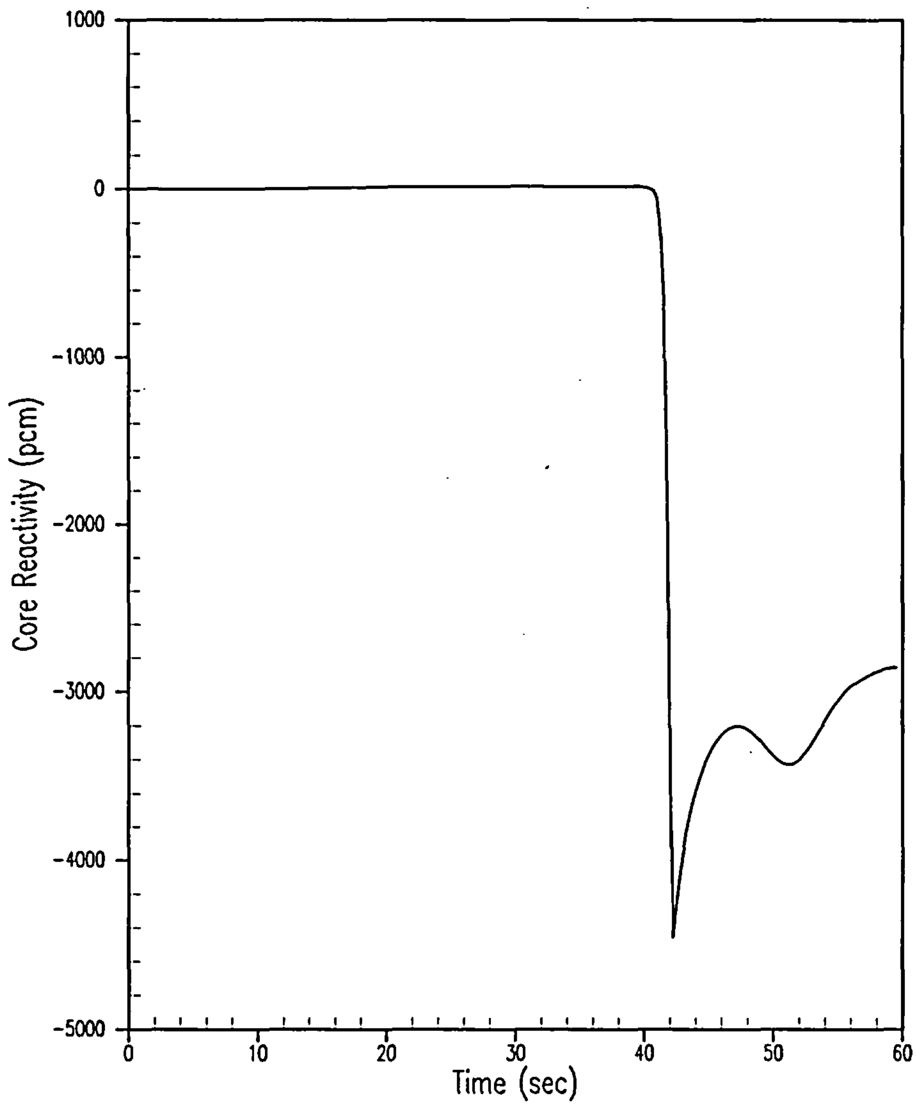

Figure 5.1.1-3 Core Reactivity for a Feedwater Malfunction Increased Feedwater Flow at HFP Initial Conditions

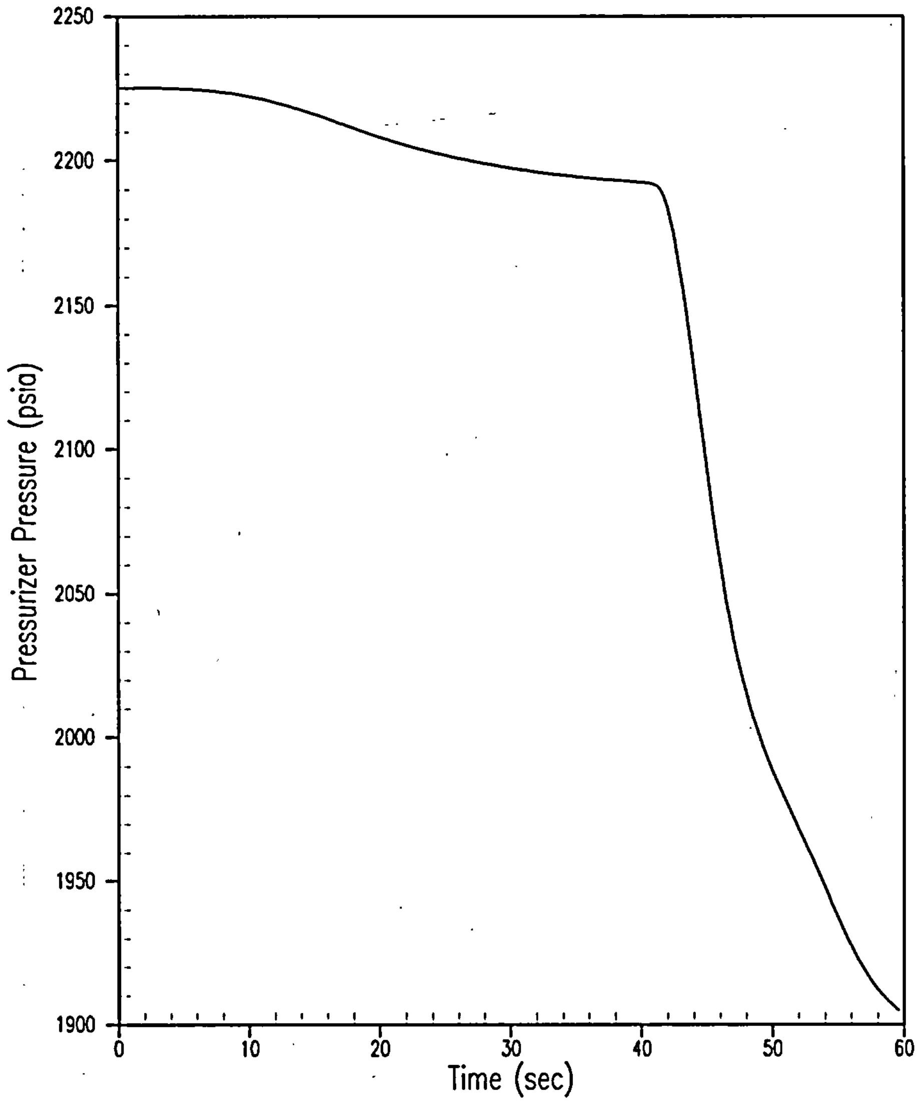

Figure 5.1.1-4 Pressurizer Pressure for a Feedwater Malfunction Increased Feedwater Flow at HFP Initial Conditions

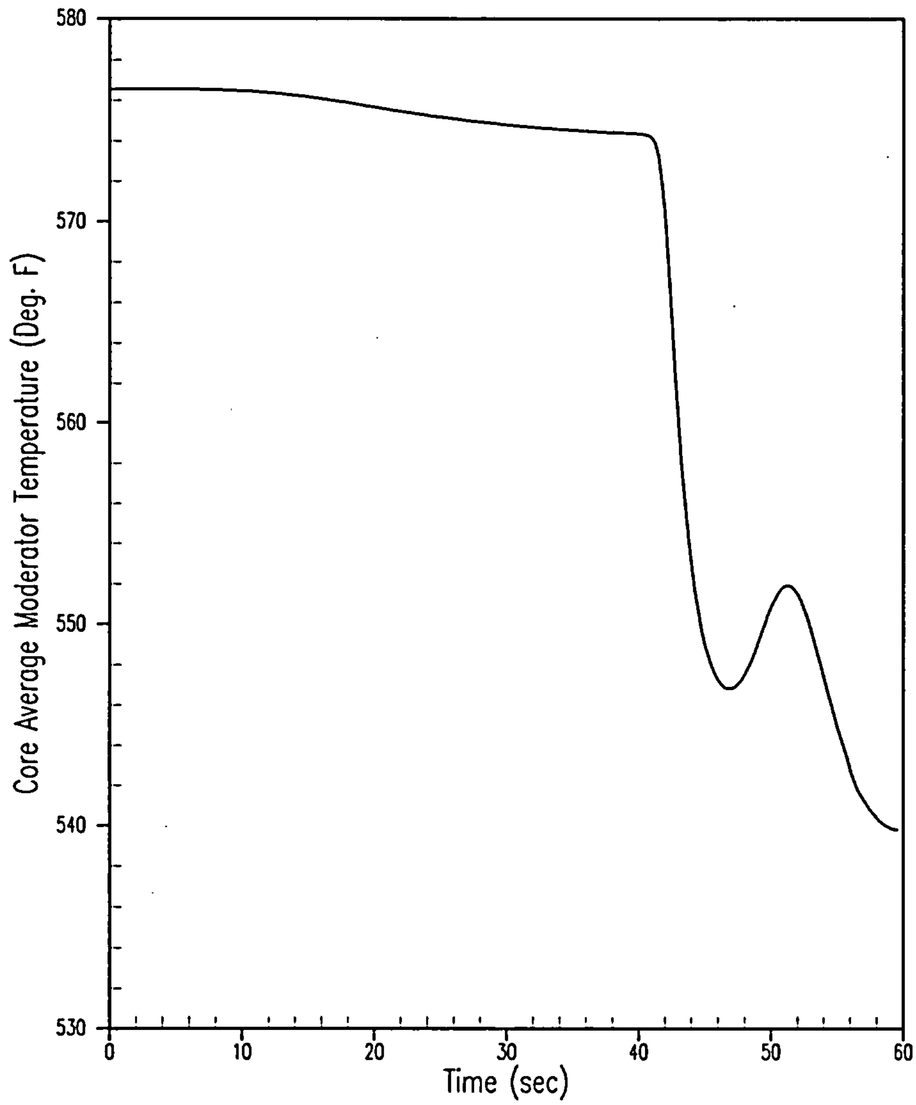

Figure 5.1.1-5 Core Average Moderator Temperature for a Feedwater Malfunction Increased Feedwater Flow at HFP Initial Conditions

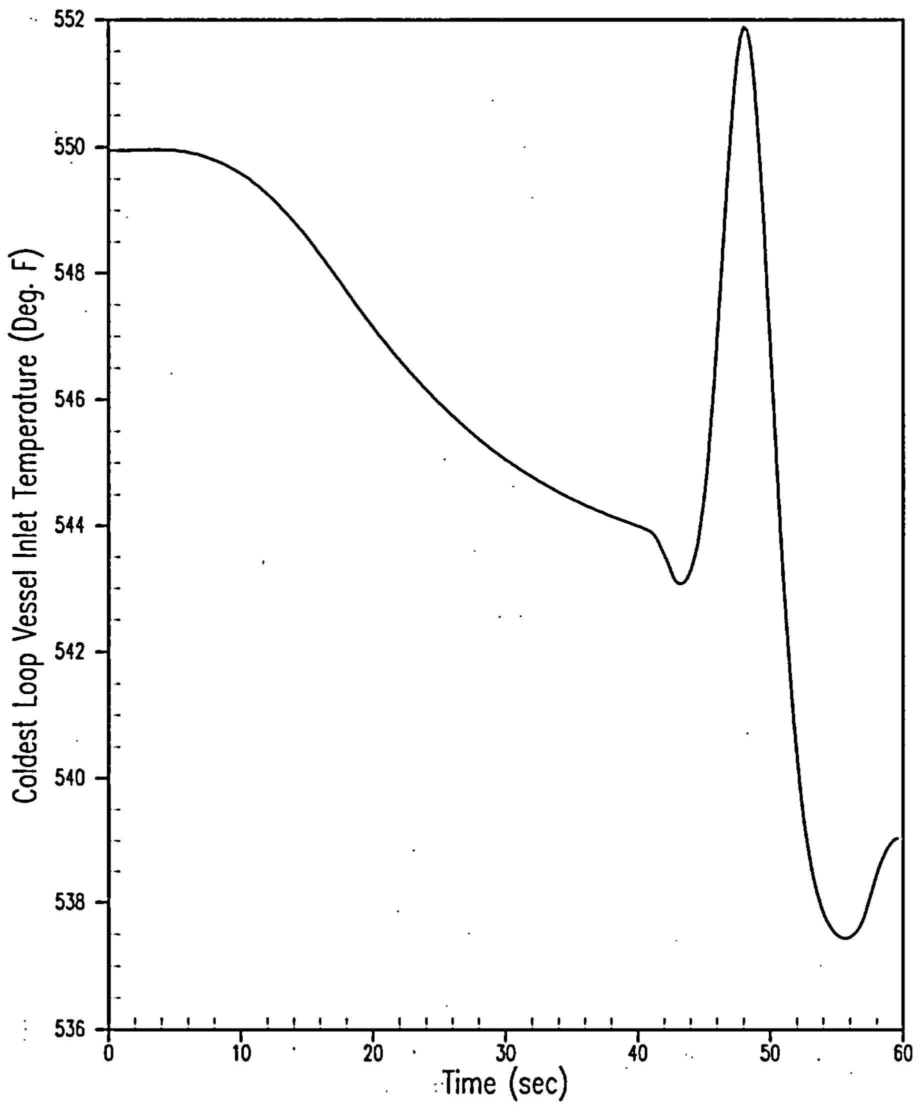

Figure 5.1.1-6 Vessel Inlet Temperature for a Feedwater Malfunction Increased Feedwater Flow at HFP Initial Conditions

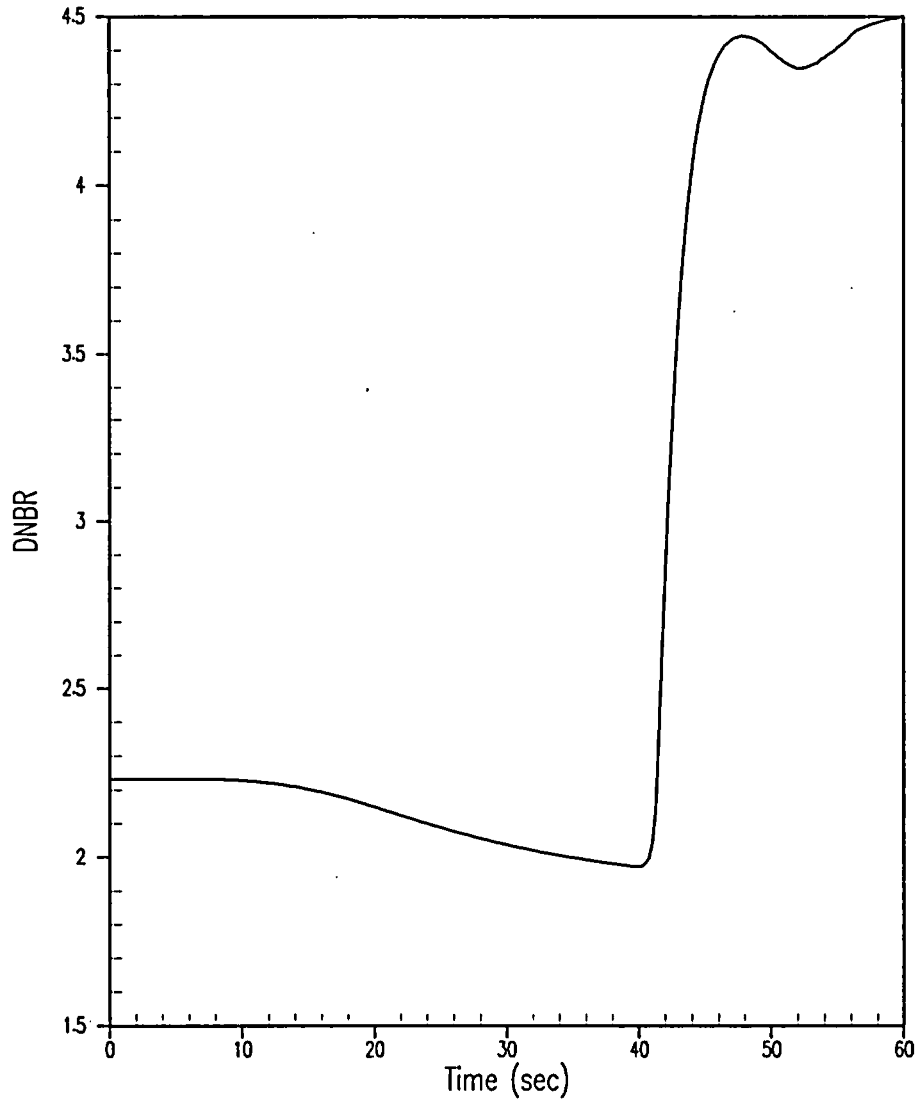

Figure 5.1.1-7 DNBR for a Feedwater Malfunction Increased Feedwater Flow at HFP Initial Conditions

5.1.2 Inadvertent Opening of Steam Generator Safety Valve/Atmospheric Dump Valve

5.1.2.1 Accident Description

The inadvertent opening of a main steam relief or safety valve (SV) results in a transient similar to a steamline break event. Upon the inadvertent opening of a main steam relief or safety valve, steam flow would increase causing a mismatch between the reactor core power and the steam generator load demand. The increased steam flow draws more heat from the primary side. This reduces the temperature of the water in the RCS. In the presence of a negative moderator temperature coefficient, the RCS temperature reduction can result in a nuclear power increase if analyzed from an at power condition. If analyzed from subcritical conditions, the assumption of a stuck CEA in conjunction with a negative moderator temperature coefficient could result in a reactivity transient that overcomes the shutdown margin and results in a subsequent return to power. In both cases, the reduced coolant temperature results in a reduction in the RCS pressure due to the increase in coolant density. Given an increase in core power and the reduction in the RCS pressure, the possible consequence of this accident is DNB with subsequent fuel damage.

5.1.2.2 Conclusions

Based on the fact that both the pre- and post-trip steamline break analyses (Sections 5.1.5 and 5.1.6) meet the acceptance criteria associated with an American Nuclear Society (ANS) Condition II event, the inadvertent opening of a main steam relief or safety valve event is bounded. Therefore, an explicit analysis of the inadvertent opening of a main steam relief or safety valve event is not required to support the St. Lucie Unit 2 transition to WCAP-9272 reload methodology.

5.1.3 Decrease in Feedwater Temperature

A change in steam generator feedwater conditions that results in an increase in feedwater flow or a decrease in feedwater temperature could result in excessive heat removal from the plant primary coolant system. Such changes in feedwater flow or feedwater temperature are a result of a failure of a feedwater control valves, feedwater bypass valve, failure in the feedwater control system, or operator error.

The occurrence of these failures that result in an excessive heat removal from the plant primary coolant system cause the primary-side temperature and pressure to decrease significantly. The existence of negative moderator and fuel temperature reactivity coefficients can cause core reactivity to rise, as the primary-side temperature decreases. In the absence of the RPS reactor trip or other protective action, this increase in core power, coupled with the decrease in primary-side pressure, can challenge the core thermal limits. Therefore, the Excessive Heat Removal Due to Feedwater System Malfunctions Event is analyzed to ensure that the DNB is not violated.

5.1.3.1 Accident Description

Another example of excessive heat removal from the RCS is the transient associated with loss of high-pressure feedwater heaters. In the event of a loss of high-pressure feedwater heaters, there could be an immediate reduction in feedwater temperature to the steam generators. At power, the increased subcooling will create a greater load demand on the RCS due to the increased heat transfer in the steam generator.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and, therefore, a reactivity insertion due to the effects of the negative moderator temperature coefficient. However, the rate of energy change is reduced as load and feedwater flow decrease, so that the transient is less severe than the full-power case.

The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow. If the increase in reactor power is large enough, the primary RPS trip functions (e.g., variable power level – high, VHPT) will prevent any power increase that can lead to a DNBR less than the safety analysis limit value.

5.1.3.2 Method of Analysis

The feedwater malfunction analysis resulting in a feedwater temperature reduction is performed to demonstrate that the DNB design basis is satisfied. This is accomplished by showing that the calculated minimum DNBR is greater than the safety analysis limit DNBR using the same methods described in the feedwater flow increase analysis (Section 5.1.1) with the following assumptions.

The following assumptions made for the analysis of the feedwater malfunction event involving the loss of the high pressure feedwater heaters are the same as those for increased feedwater flow, except:

1. At full-power conditions the reduced feedwater enthalpy is assumed to be 307.5 Btu/lbm which corresponds to a reduced feedwater temperature of 335°F.

2. The excessive feedwater temperature reduction assumes the nominal full-power feedwater flow is maintained to both steam generators.

3. The feedwater temperature reduction event resulting from the loss of the high-pressure feedwater heaters is terminated by the steam generator high water level signal that closes all main feedwater control and feedwater control bypass valves. High-high steam generator level protection will trip the turbine, stop the main feedwater pumps, and close the main feedwater pump discharge valves.

Protection against undesirable conditions is similar to that described for increased feedwater event.

5.1.3.3 Results

The loss of high-pressure feedwater heaters causes a reduction in feedwater temperature, which increases the thermal load on the primary system. The reduction in feedwater temperature generates a more limiting condition than the excessive feedwater flow increase case performed at full-power initial conditions. The power increases until reactor trip is actuated when the variable high-power trip setpoint is reached. The peak core heat flux and minimum DNB condition is reached shortly after the CEAs begin to drop into the core. The consequences of the event are mitigated prior to reaching a high steam generator water level condition in either steam generator. The limiting full-power feedwater temperature reduced conditions were analyzed and it was confirmed that the calculated minimum DNBR is above the DNBR safety analysis limit. Therefore, the applicable DNBR acceptance criterion is met.

Table 5.1.3-1 shows the time sequence of events for the full power feedwater malfunction transient case resulting in a reduction in feedwater temperature. Figures 5.1.3-1 through 5.1.3-7 show transient responses for various system parameters during a feedwater system malfunction causing a feedwater temperature reduction from full power conditions.

5.1.3.4 Conclusions

The results of the analysis show that the VHP trip or TM/LP RPS signal provides adequate protection against the feedwater malfunction transient. No fuel or cladding damage is predicted for this accident. All acceptance criteria are satisfied.

Table 5.1.3-1 Feedwater System Malfunction Event at Full Power, Reduced Feedwater Temperature Sequence of Events and Transient Results	
Event	**Time (seconds)**
Loss Of Feedwater Heater Occurs	0.0
Variable High Power Trip Setpoint is Reached	36.70
Trip Breakers Open	37.10
CEAs Begin to Drop into Core	37.84
Minimum DNBR Occurs	38.25
Turbine Trip	39.10
Results	
Peak Nuclear Power, fraction of nominal	1.144
Peak Core Heat Flux, fraction of nominal	1.140
Minimum DNBR	1.95

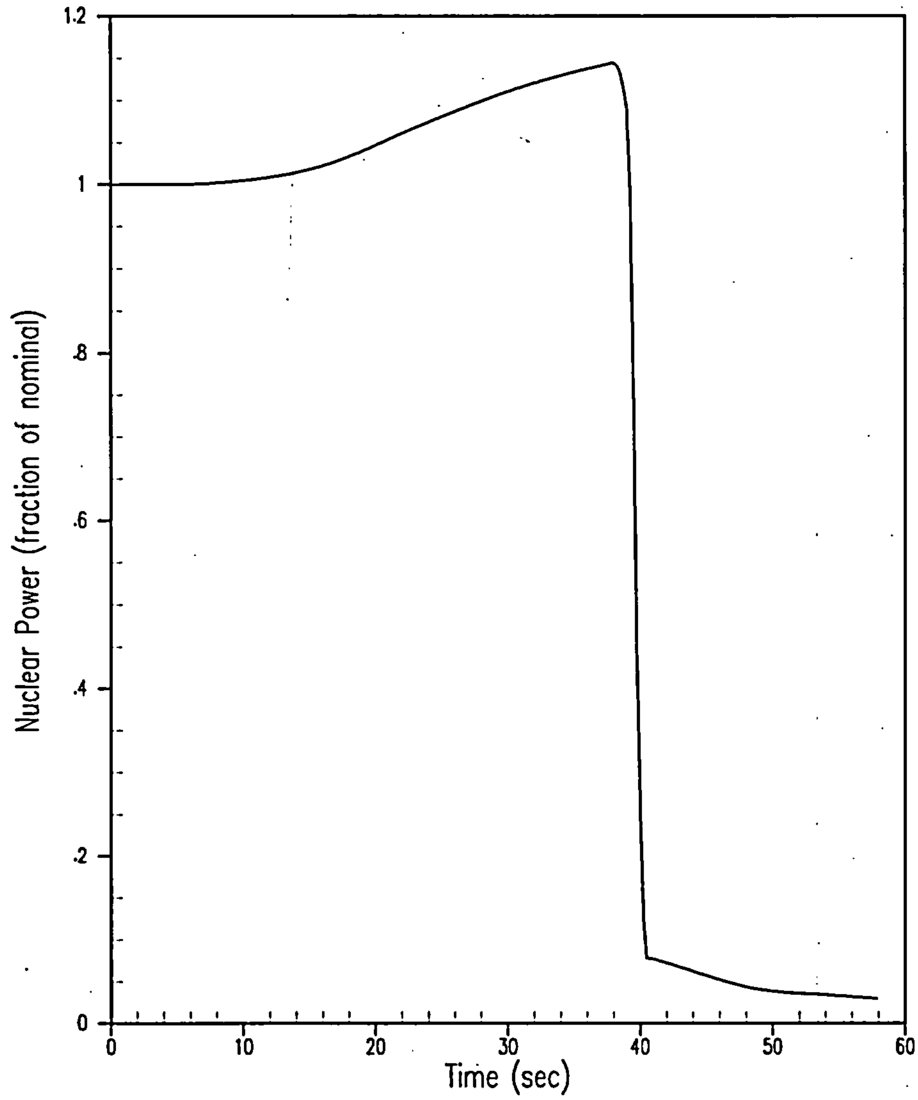

Figure 5.1.3-1 Nuclear Power for a Feedwater Malfunction Reduced Feedwater Temperature at HFP Initial Conditions

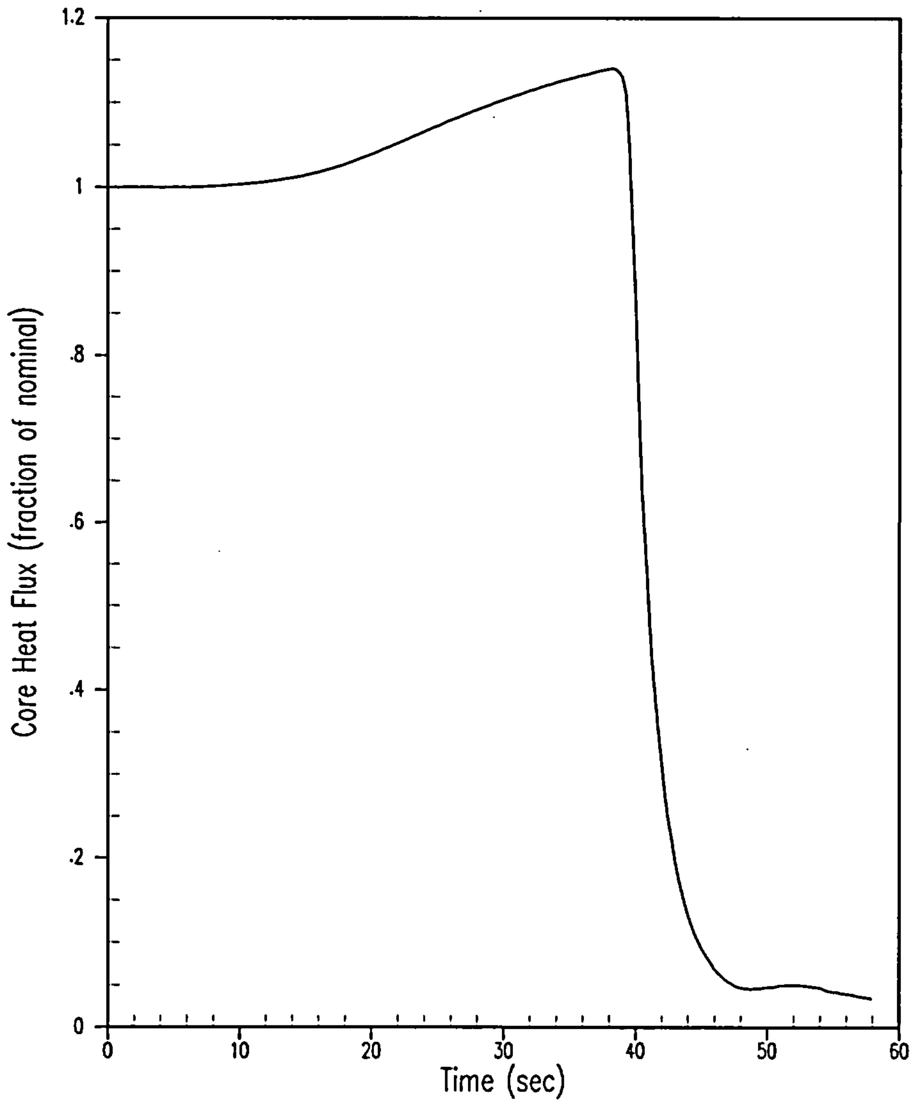

Figure 5.1.3-2 Core Heat Flux for a Feedwater Malfunction Reduced Feedwater Temperature at HFP Initial Conditions

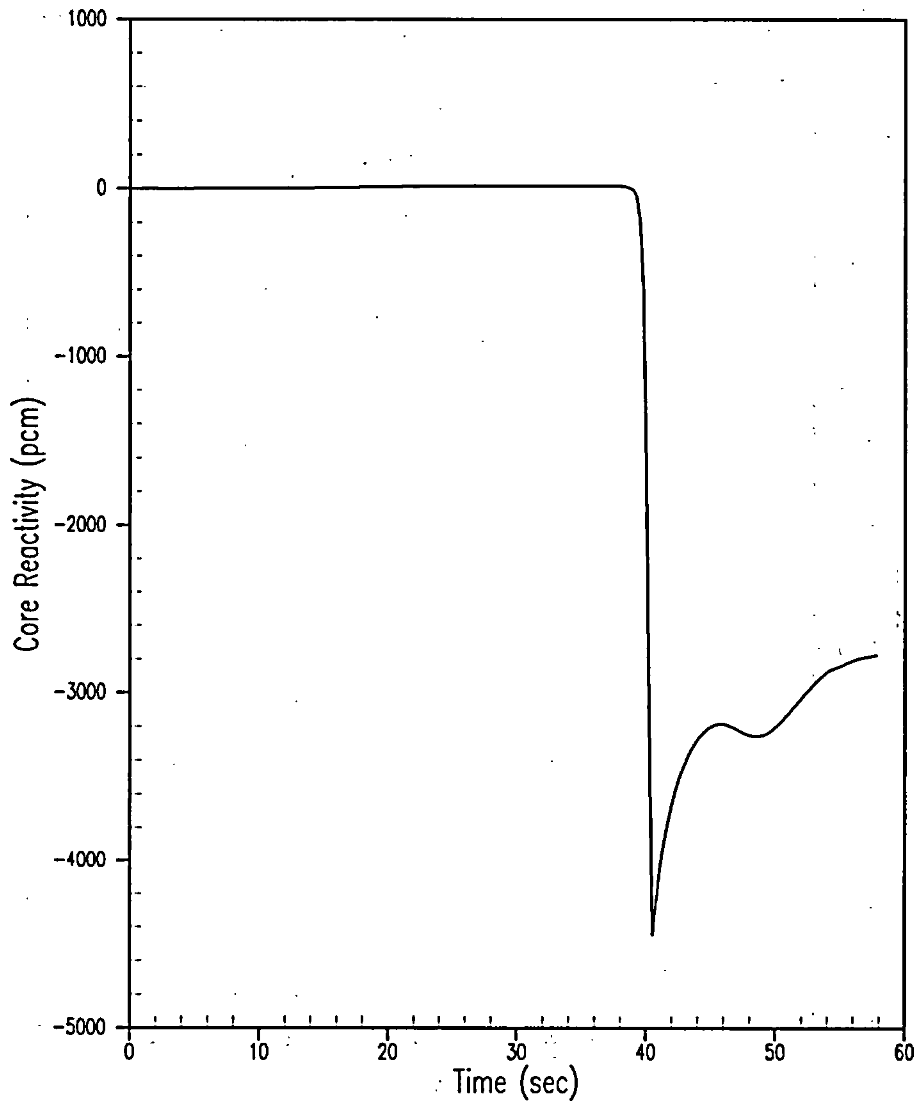

Figure 5.1.3-3 Core Reactivity for a Feedwater Malfunction Reduced Feedwater Temperature at HFP Initial Conditions

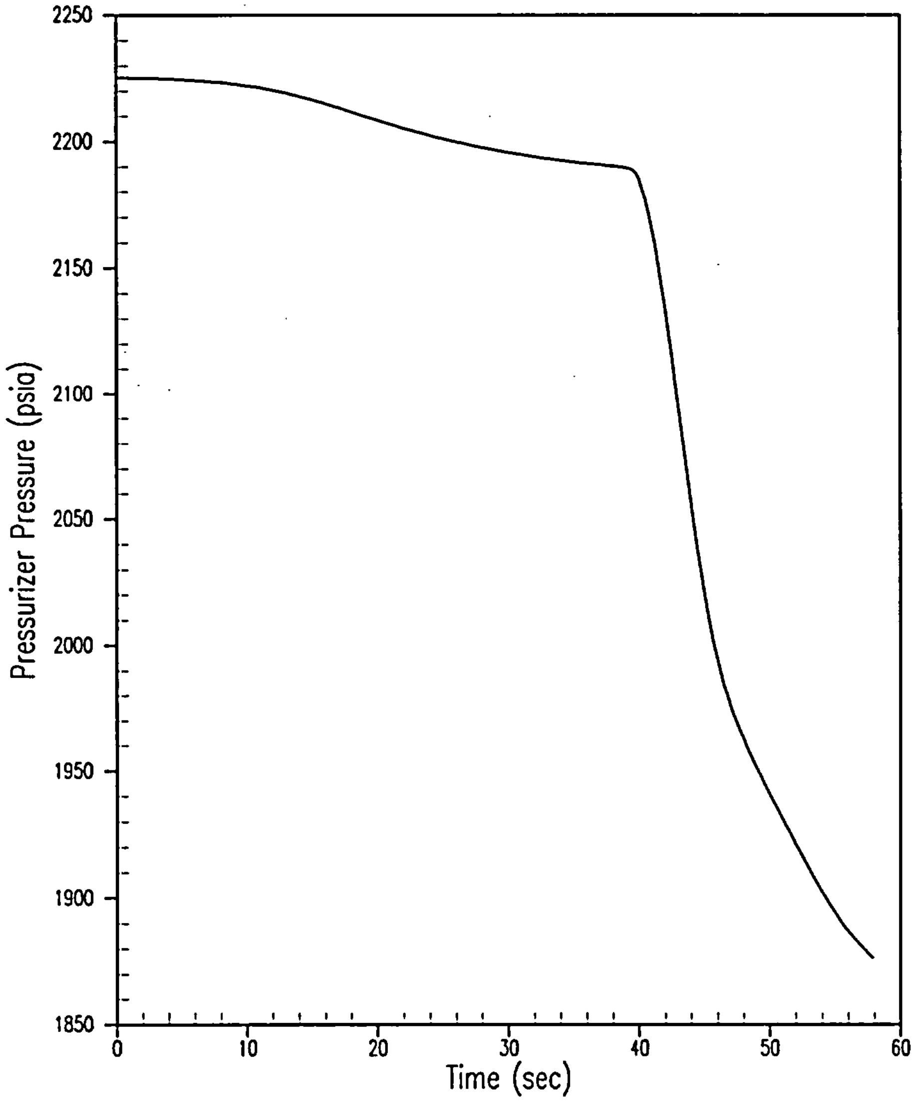

Figure 5.1.3-4 Pressurizer Pressure for a Feedwater Malfunction Reduced Feedwater Temperature at HFP Initial Conditions

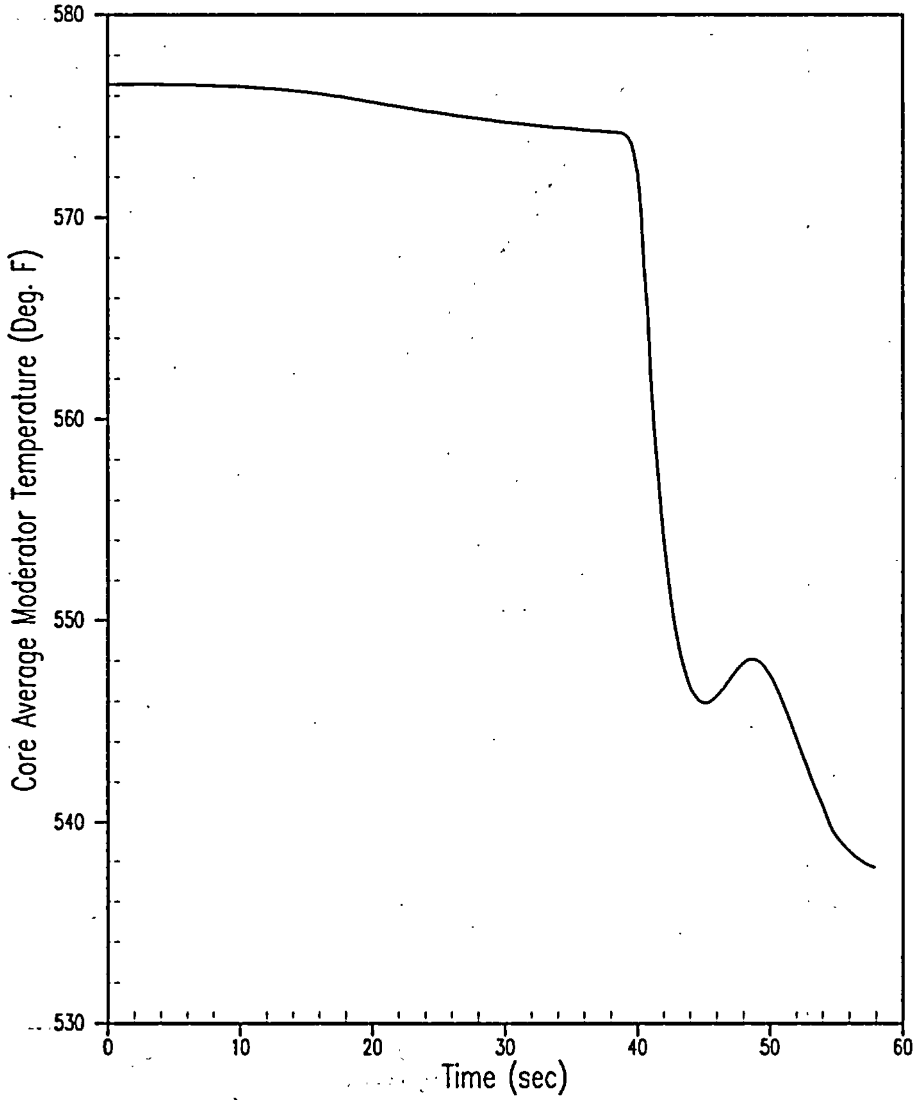

Figure 5.1.3-5 Core Average Moderator Temperature for a Feedwater Malfunction Reduced Feedwater Temperature at HFP Initial Conditions

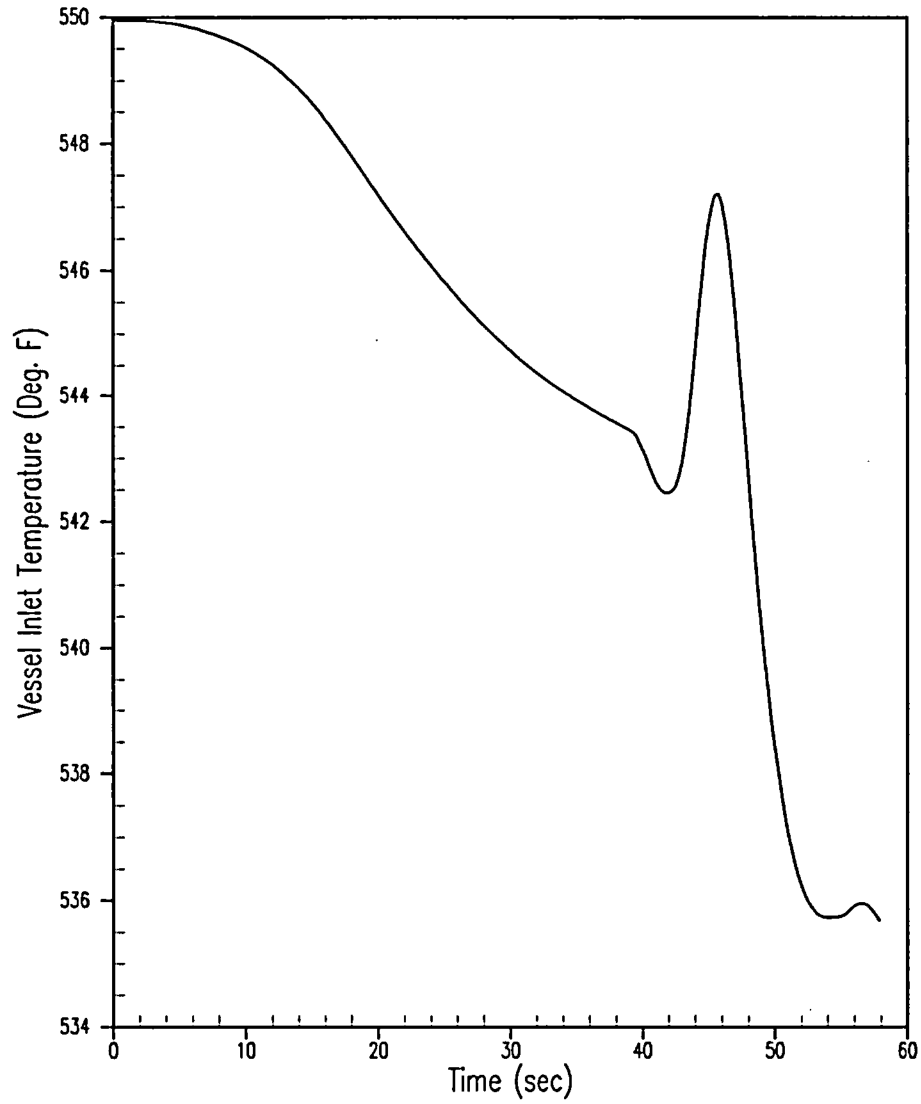

Figure 5.1.3-6 Vessel Inlet Temperature for a Feedwater Malfunction Reduced Feedwater Temperature at HFP Initial Conditions

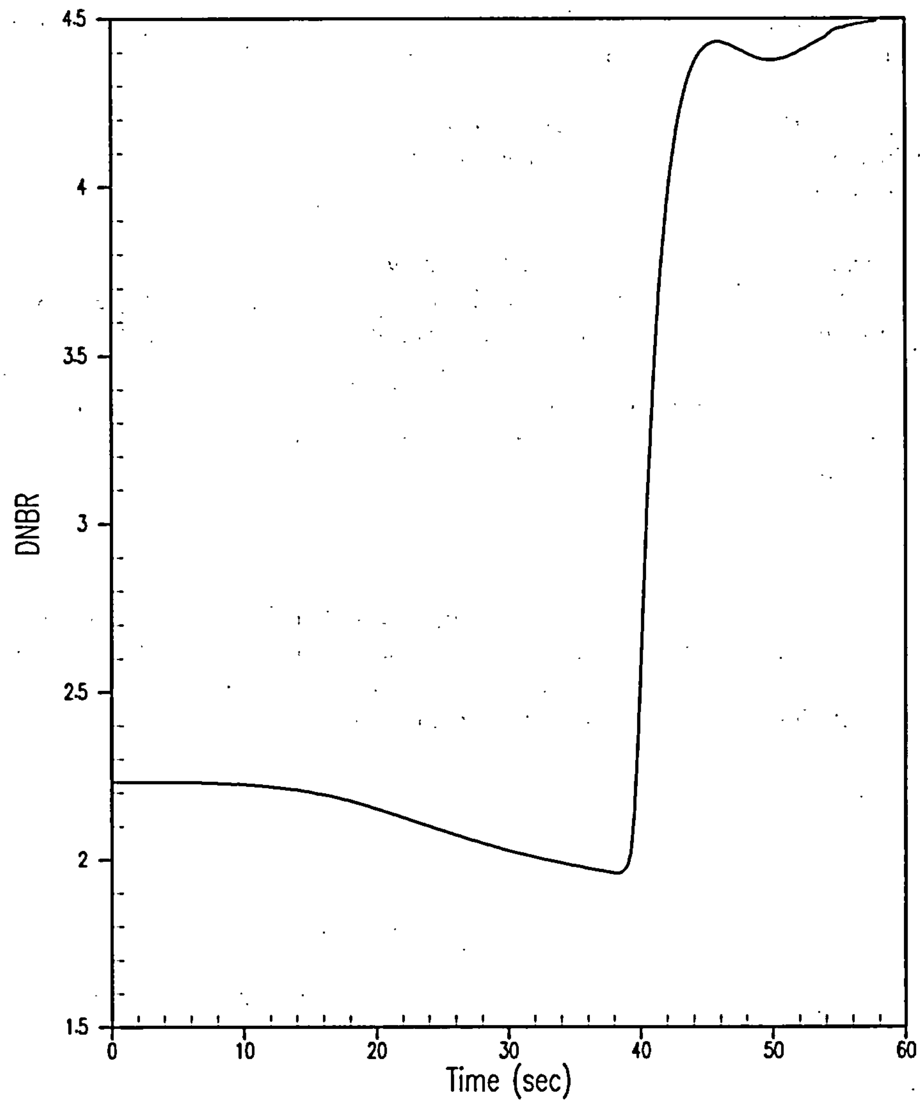

Figure 5.1.3-7 DNBR for a Feedwater Malfunction Reduced Feedwater Temperature at HFP Initial Conditions

5.1.4 Increase in Main Steam Flow

5.1.4.1 Accident Description

An excessive increase in main steam flow event (excessive load increase) is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. For this accident, the steam flow is typically assumed to increase by not more than 30% percent of the initial value. This accident could result from one of the following:

- An administrative violation such as excess loading by the operator
- Equipment malfunction in the steam dump control
- Turbine throttle valve control malfunction

The increased steam flow draws more heat from the primary side. This reduces the temperature of the water in the RCS. In the presence of a negative moderator temperature coefficient, the RCS temperature reduction can result in a nuclear power increase. The reduced coolant temperature also results in a reduction in the pressurizer water volume, due to the increased density of the cooler RCS water, and a reduction in the pressurizer pressure. Given an increase in core power and the reduction in the RCS pressure, the possible consequence of this accident (assuming no protective functions) is DNB with subsequent fuel damage.

5.1.4.2 Conclusions

Based on the fact that the pre-trip steamline break analysis of Section 5.1.5 meets the acceptance criteria associated with an ANS Condition II event, the excessive increase in main steam flow event is bounded. Therefore, an explicit analysis of the excessive increase in main steam flow event is not required to support the St. Lucie Unit 2 transition to WCAP-9272 reload methodology.

5.1.5 Pre-Trip Steam System Piping Failure

5.1.5.1 Accident Description

A rupture in the main steam system piping from an at-power condition creates an increased steam load, which extracts an increased amount of heat from the RCS via the steam generators. This results in a reduction in RCS temperature and pressure. In the presence of a strong negative moderator temperature coefficient, typical of end-of-cycle life conditions, the colder core inlet coolant temperature causes the core power to increase from its initial level due to the positive reactivity insertion. The power approaches a level equal to the total steam flow. Depending on the break size, a reactor trip may occur due to overpower conditions or as a result of low steam generator pressure.

The steam system piping failure accident analysis described in Section 5.1.6 is performed assuming a hot zero power initial condition with the control rods inserted in the core, except for the most reactive rod in the fully withdrawn position, out of the core. That condition could occur while the reactor is at hot shutdown at the minimum required shutdown margin or after the plant has been tripped manually or by the reactor protection system following a steamline break from an at-power condition. For an at-power break, the analysis of Section 5.1.6 represents the limiting condition with respect to core protection for the time period following reactor trip. The purpose of this section is to describe the analysis of a steam system piping failure occurring from an at-power initial condition and to demonstrate that core protection is maintained prior to and immediately following reactor trip.

Depending on the size of the break, this event is classified as either an ANS Condition III (infrequent fault) or Condition IV (limiting fault), as defined in Section 5.1.0.9. However, acceptance criteria applied to the analysis of this event is consistent with an ANS Condition II event to ensure that all small bore steam system pipe breaks, steam system valve malfunctions (Section 5.1.2) and uncontrolled increases in steam flow (Section 5.1.4) are bounded. The acceptance criteria for this event are defined in Section 5.1.0.9.2.

5.1.5.2 Method of Analysis

The analysis of the steamline rupture is performed in the following stages:

- The RETRAN code (References 1 and 2) is used to calculate the nuclear power, core heat flux, and RCS temperature and pressure transients resulting from the cooldown following the steamline break.

- The core radial and axial peaking factors are determined using the thermal-hydraulic conditions from the transient analysis as input to the nuclear core models. The VIPRE code (see Section 4.2) is then used to calculate the DNBR for the limiting time during the transient.

This accident is analyzed with the Revised Thermal Design Procedure. Plant characteristics and initial conditions are provided in Table 5.1.0-2.

The following assumptions are made in the transient analysis:

1. Initial Conditions – The initial core power, reactor coolant temperature, and RCS pressure are assumed to be at their nominal full-power values. The full-power condition is more limiting than part-power in terms of DNBR. The RCS minimum measured flow is used. Uncertainties in initial conditions are included in the DNBR limit as described in Reference 3.

2. Break size – A spectrum of break sizes is analyzed. Small breaks do not result in a reactor trip. Intermediate size breaks result in a reactor trip on variable overpower. Larger break sizes result in a reactor trip on low steam generator pressure.

3. Break flow – In computing the steam flow during a steamline break, the Moody curve (Reference 4) for fL/D = 0 is used.

4. Reactivity Coefficients – The analysis is performed over a range of moderator density coefficients (MDCs) and otherwise assumes end-of-cycle reactivity feedback coefficients with the minimum Doppler power feedback to maximize the power increase following the break.

5. Protection System – This analysis only considers the initial phase of the transient initiated from an at-power condition. Protection in this phase of the transient is provided by a reactor trip, as discussed in Item 2, above. Section 5.1.6 presents the analysis of the bounding transient following reactor trip, where other protection system features are actuated to mitigate the effects of the steamline break.

6. Control Systems – The results of the analysis would not be more severe as a result of control system actuation. Therefore, their effects have been ignored in the analysis. Control systems are not credited in mitigating the effects of the transient.

5.1.5.3 Results

A spectrum of breaks ranging in size from 0.1 ft^2 to 6.305 ft^2 was analyzed. Additionally, a spectrum of MDCs was considered, ranging from 0.0 Δk/gm/cc to 0.43 Δk/gm/cc. The results show that, for small break sizes up to approximately 0.4 ft^2, a reactor trip is not generated. In this case, the event is similar to an excessive load increase event as described in Section 5.1.4. The core reaches a new equilibrium condition at a higher power equivalent to the increased steam flow. For break sizes of approximately 0.5 ft^2 to 3.5 ft^2, the power increase results in a reactor trip on variable overpower (either excore power or ΔT power). For break sizes of approximately 3.5 ft^2 and larger, a reactor trip is generated within a few seconds of the break on the low steam generator pressure signal.

With respect to the parametric study performed on MDC and break size combinations, the most severe of the cases, assuming the most positive MDC value, resulted in a reactor trip on variable overpower (from the excore power calculation). This resulted in peak power levels below that of the most severe of the cases assuming a less positive MDC that generated a reactor trip on variable overpower (from the ΔT power calculation). Therefore, the most limiting MDC was identified as 0.30 Δk/gm/cc. This interaction between trip functions and MDC is shown in Figure 5.1.5-1.

The limiting case for demonstrating DNB protection is the 3.2 ft^2 break for an MDC of 0.30 Δk/gm/cc. The time sequence of events for this case is shown in Table 5.1.5-1. Figures 5.1.5-2 through 5.1.5-9 show the transient response for the limiting case.

5.1.5.4 Conclusions

A detailed analysis to assess the both minimum DNBR and peak linear heat rate was performed using radial and axial core peaking factors based on the statepoints generated from the limiting case. Because the radial and axial peaking factors are dependent on the cycle-specific loading pattern, the minimum DNBR and peak linear heat rate are verified to meet their respective limits on a cycle-specific basis through the WCAP-9272 reload process. The initial analysis supporting the implementation of the WCAP-9272 reload process for St. Lucie Unit 2 concludes that both the DNB design basis and the peak linear heat rate limit are met for the limiting case. Although the steamline break accident is classified as an ANS Condition III or IV event, the analysis demonstrates that the acceptance criteria for an ANS Condition II event are satisfied for all ruptures occurring from an at-power condition.

5.1.5.5 References

1. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April, 1999.

2. EPRI NP-1850-CCM, Rev. 6, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," December 1995.

3. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

4. Moody, F. S., "Transactions of the ASME Journal of Heat Transfer," Figure 3, page 134, February 1965.

Table 5.1.5-1 Steamline Break Analysis Assumptions and Sequence of Events		
Event	**Time (sec)**	**Value**
MSLB Transient Initiated	0.01	
Variable Overpower – ΔT Power Setpoint Reached	10.13	112.2 %
Reactor Trip Signal Generated	10.53	0.40 sec. delay from setpoint
Turbine Trip on Reactor Trip	10.78	0.25 sec. delay from reactor trip
CEA Release	11.27	0.74 sec. delay from reactor trip
Minimum DNBR Reached	12.60	1.442
Peak Linear Heat Rate Reached	12.60	21.23 kW/ft
Peak Heat Flux Reached	12.60	131.0 %
Loss of Offsite Power/RCP Trip	13.78	3.0 sec delay from turbine trip

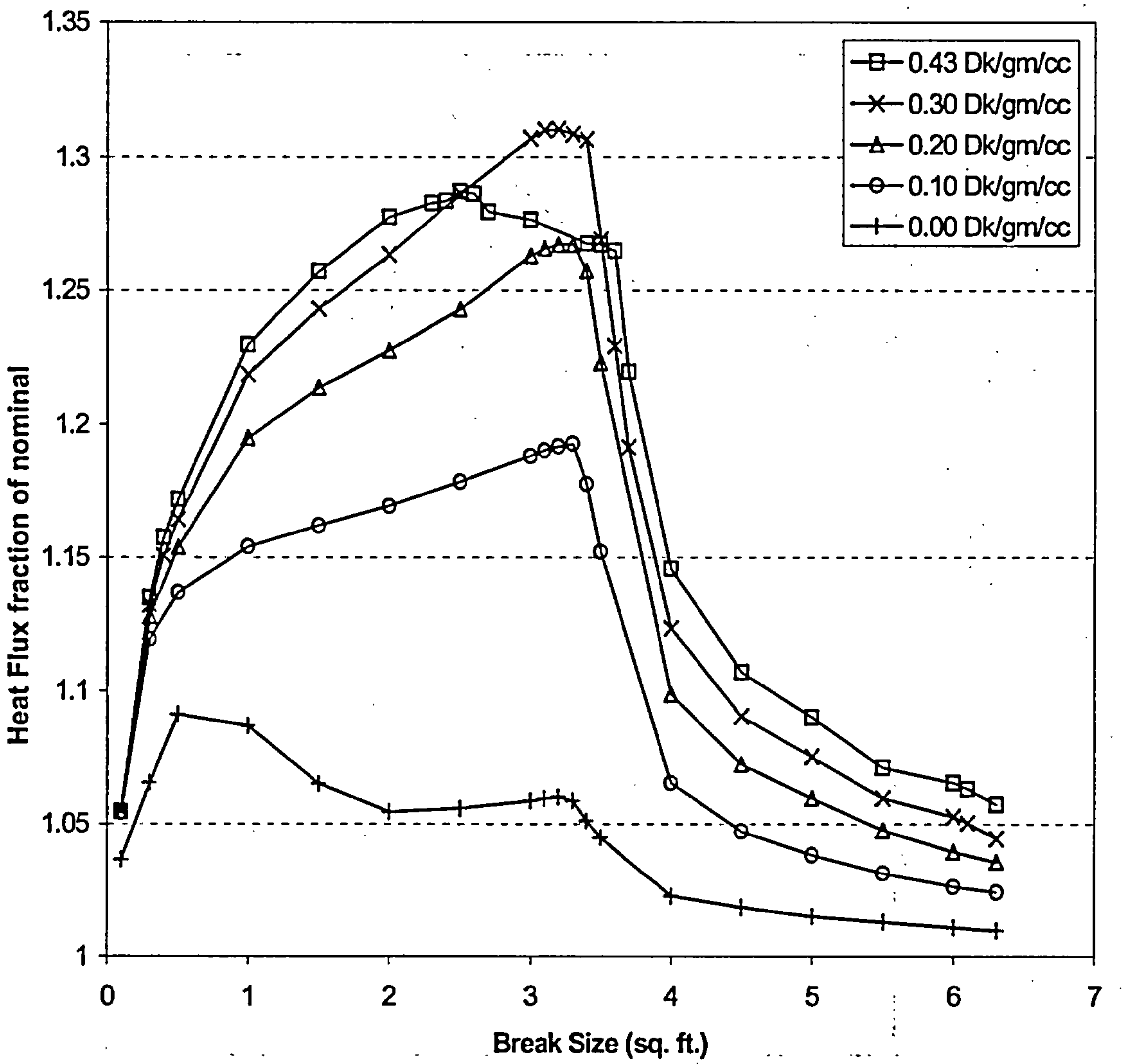

Figure 5.1.5-1 Summary of Peak Core Heat Flux vs. Break Size for a Spectrum of MDCs (0.0 to 0.43 Δk/gm/cc)

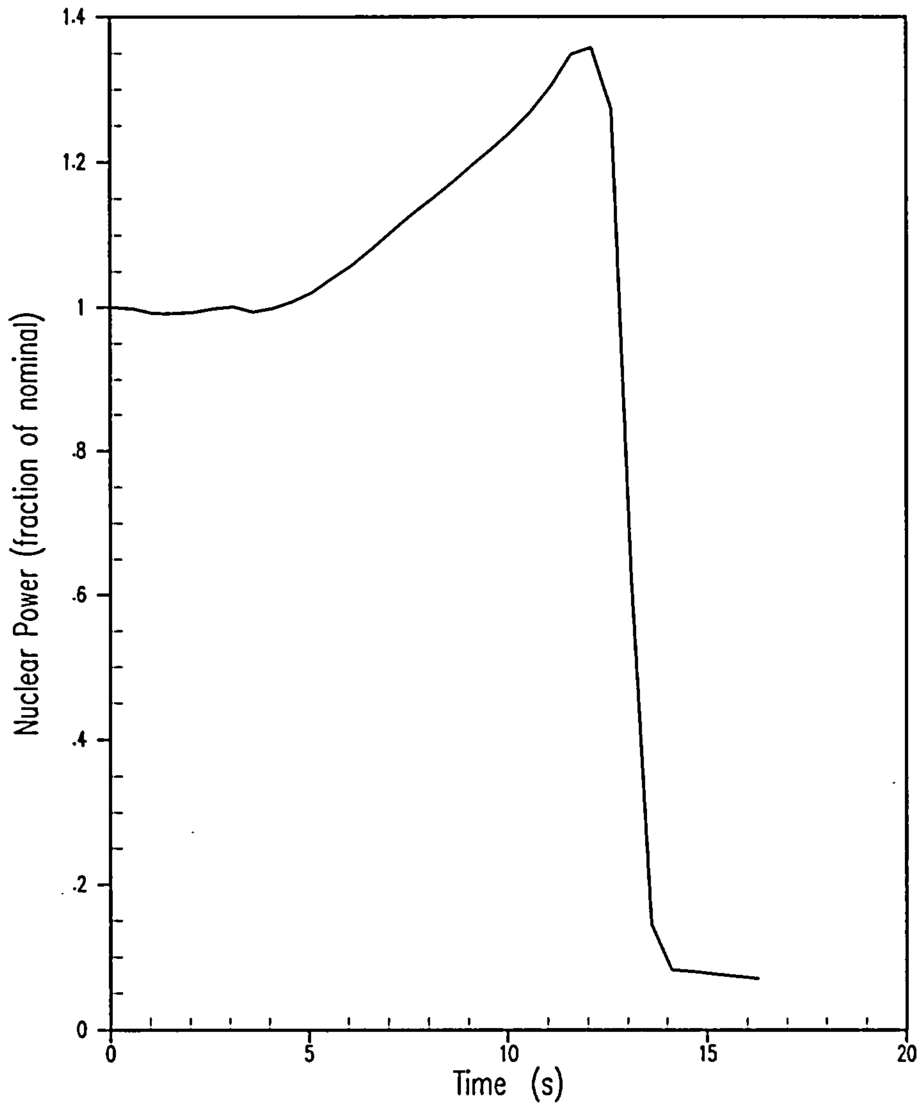

Figure 5.1.5-2 Pre-Trip Main Steamline Break – Nuclear Power versus Time

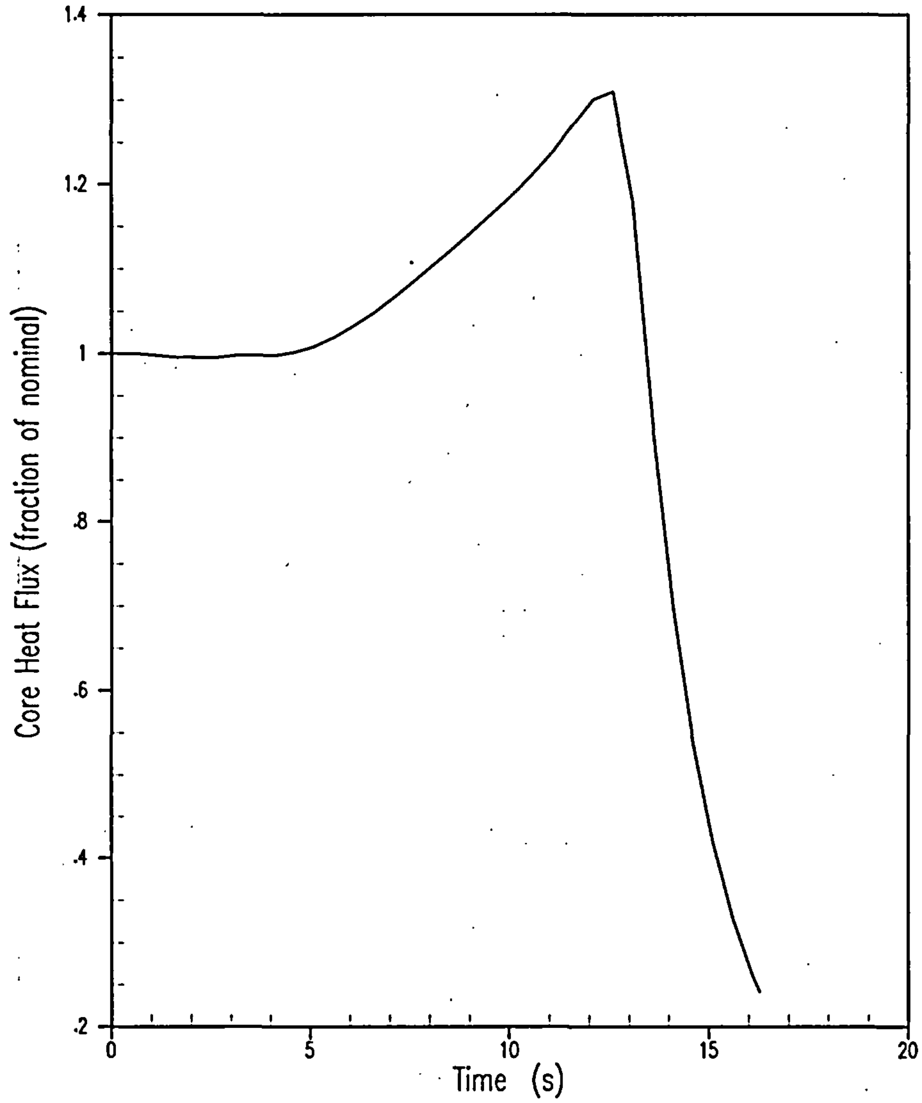

Figure 5.1.5-3 Pre-Trip Main Steamline Break – Core Heat Flux versus Time

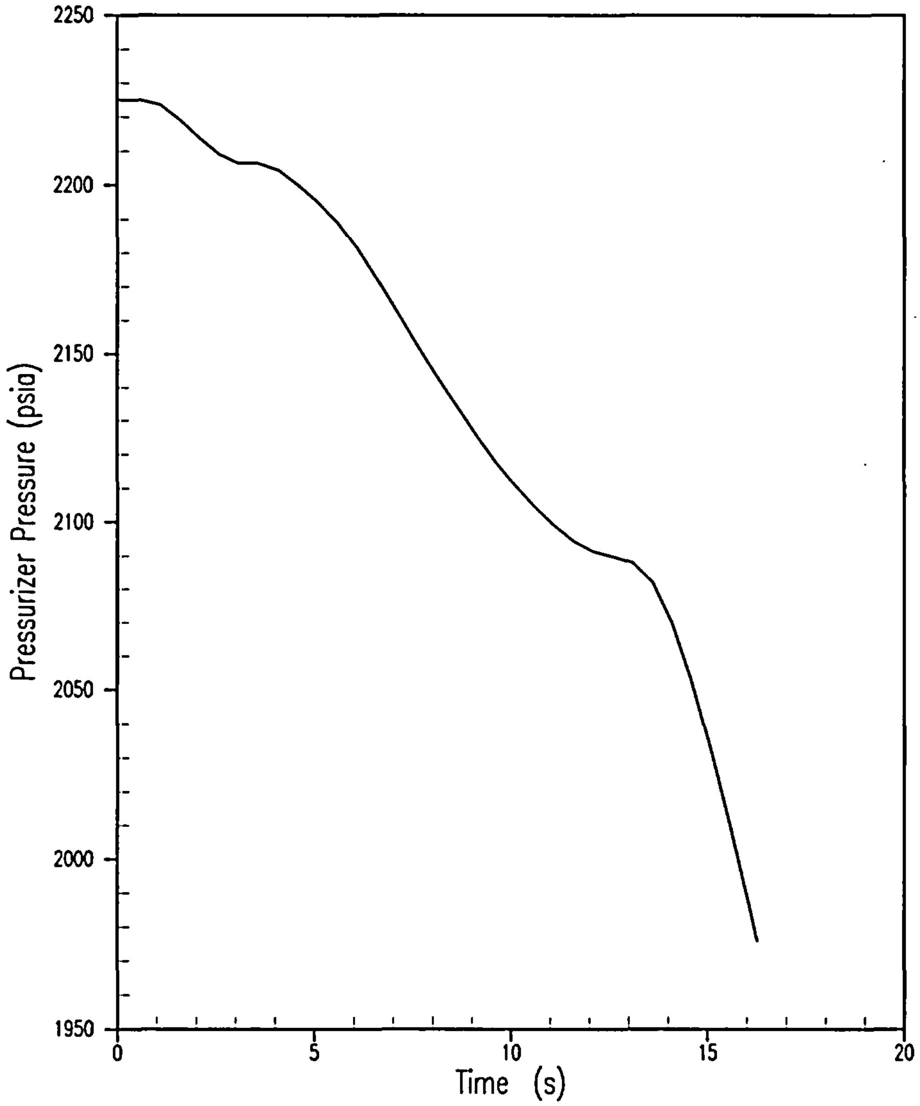

Figure 5.1.5-4 Pre-Trip Main Steamline Break – Pressurizer Pressure versus Time

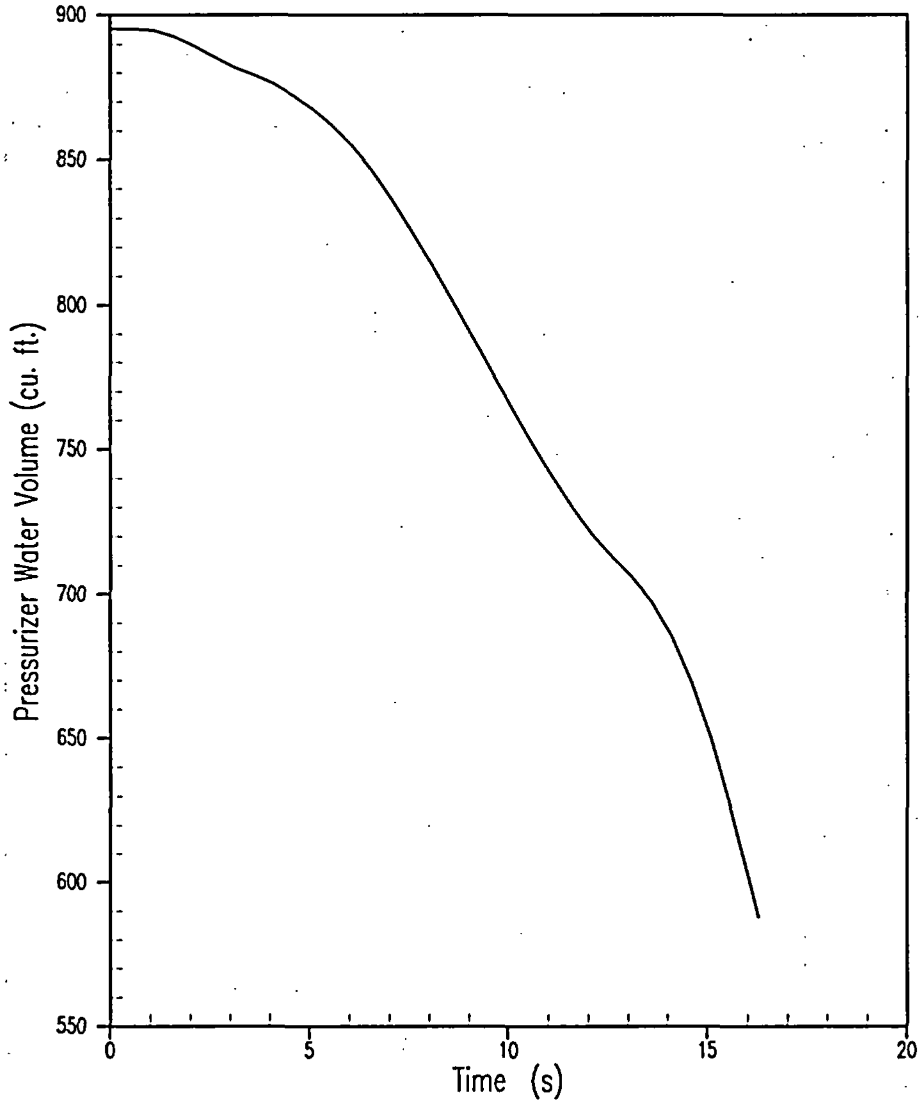

Figure 5.1.5-5 Pre-Trip Main Steamline Break – Pressurizer Water Volume versus Time

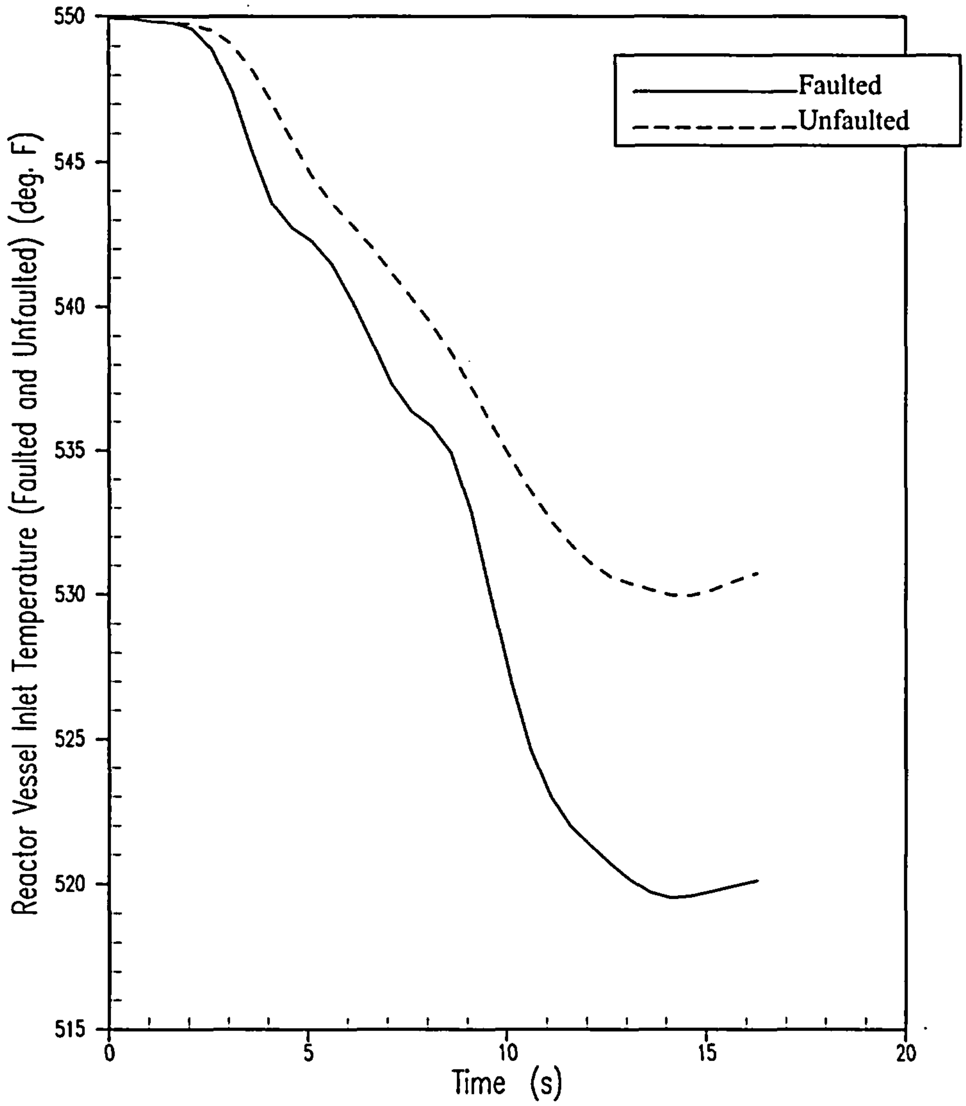

Figure 5.1.5-6 Pre-Trip Main Steamline Break – Reactor Vessel Inlet Temperature versus Time

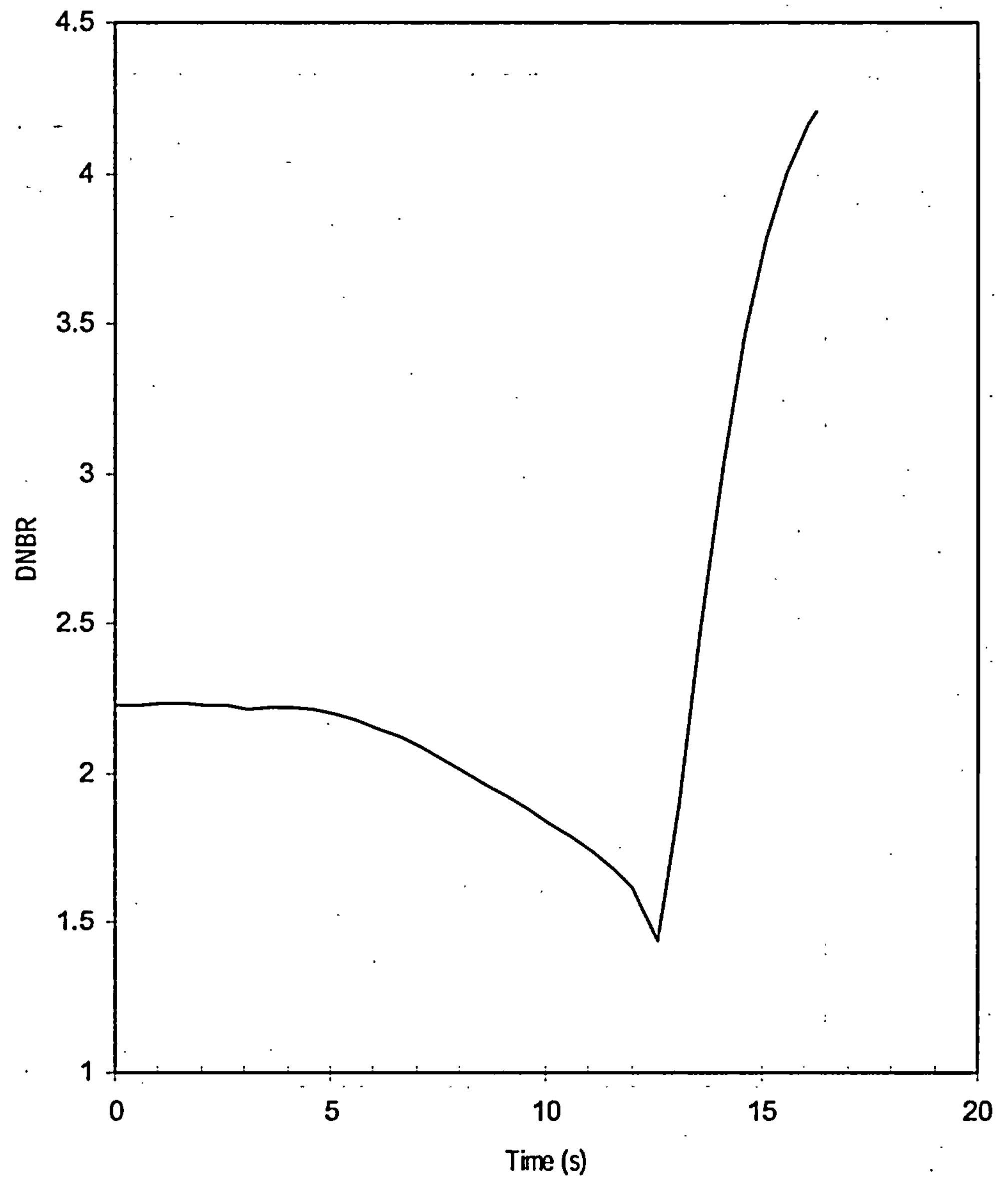

Figure 5.1.5-7 Pre-Trip Main Steamline Break – DNBR versus Time

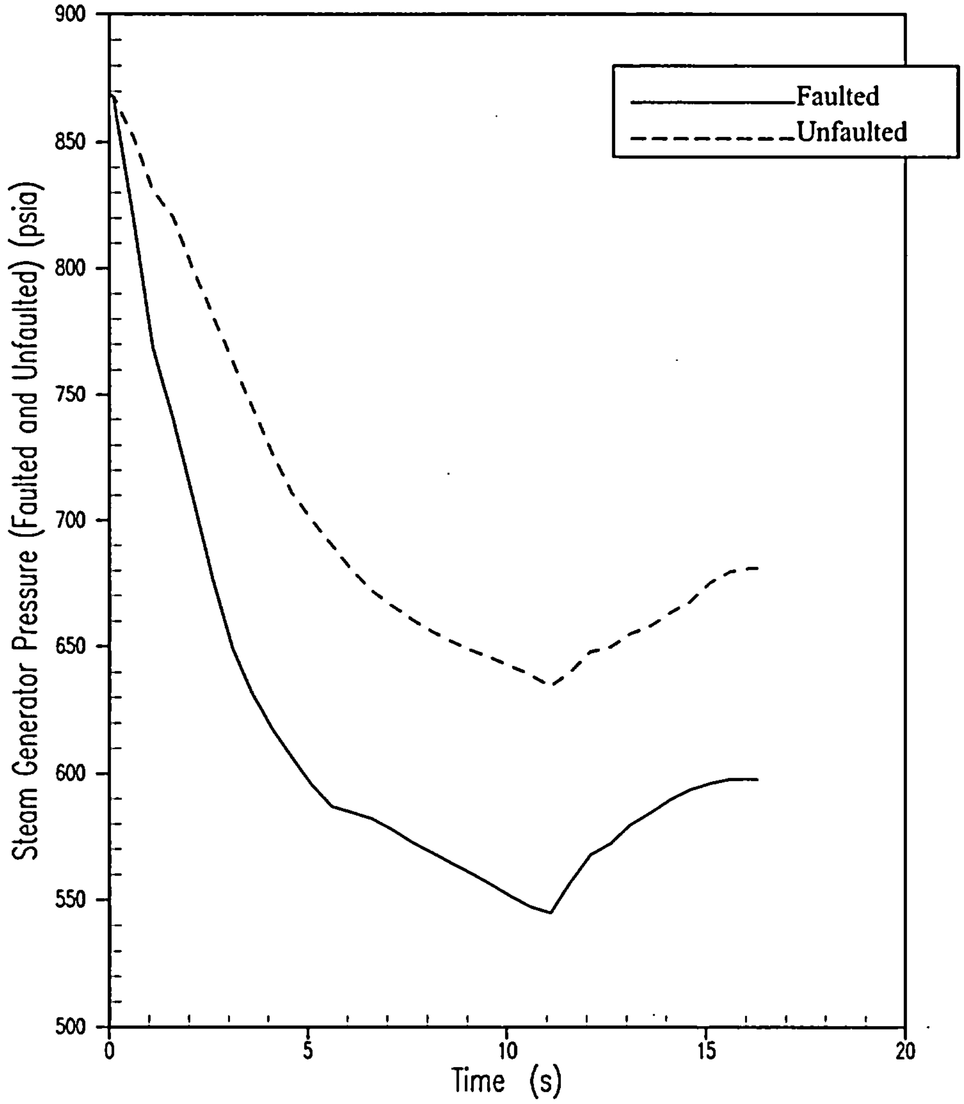

Figure 5.1.5-8 Pre-Trip Main Steamline Break – Steam Generator Steam Pressure versus Time

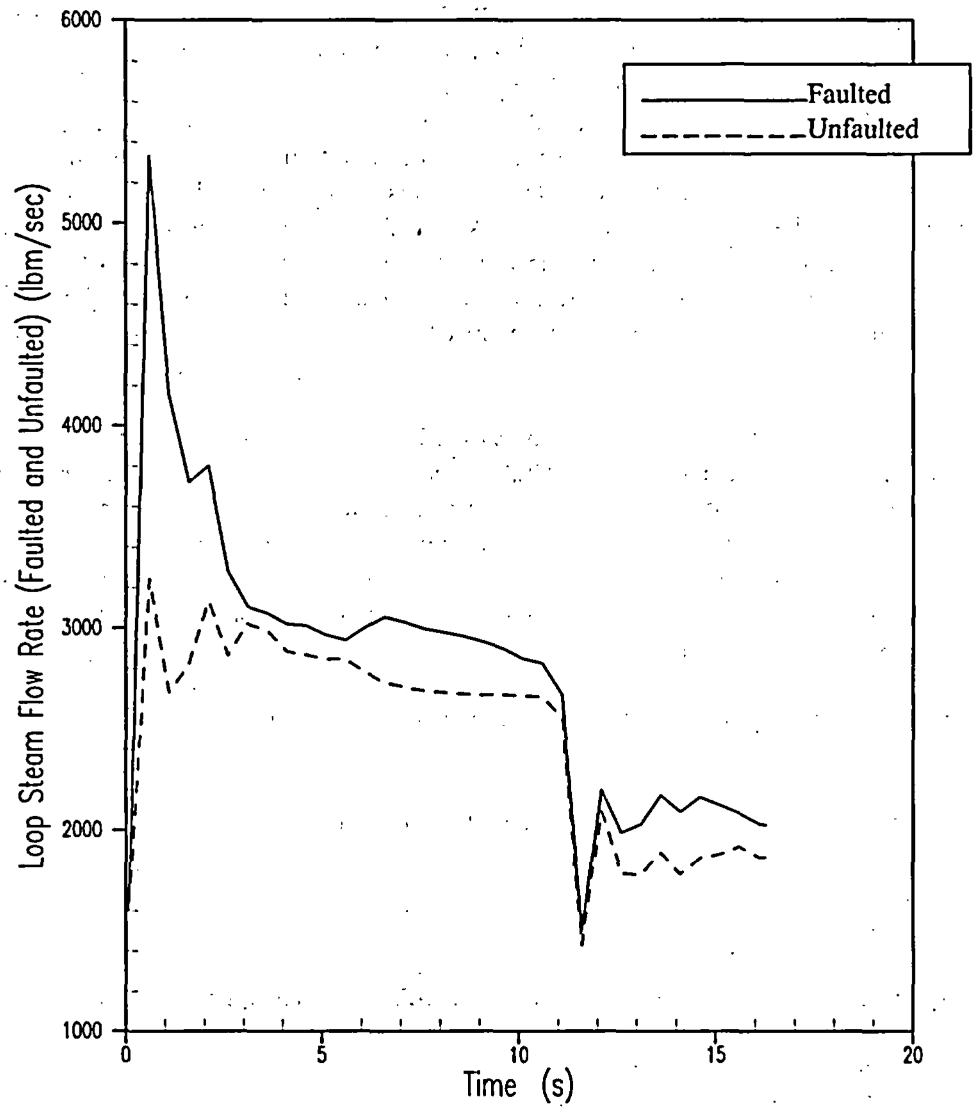

Figure 5.1.5-9 Pre-Trip Main Steamline Break – Loop Steam Flow Rate versus Time

5.1.6 Post-Trip Steam System Piping Failures

5.1.6.1 Accident Description

A steamline break transient results in an uncontrolled increase in steam flow release from the steam generators, with the flow decreasing as the steam pressure drops. This steam flow release increases the heat removal from the RCS, which decreases the RCS temperature and pressure. With the existence of a negative moderator temperature coefficient (MTC), the RCS cooldown results in a positive reactivity insertion, and consequently a reduction of the core shutdown margin for the post-trip condition. If the most reactive CEA is assumed stuck in its fully withdrawn position after reactor trip, the possibility is increased that the core may become critical and subsequently return to power. A return to power following a steamline break from the post-trip condition is a concern with the high-power peaking factors that may exist when the most reactive CEA is stuck in its fully withdrawn position. Following a steamline break, the core is ultimately shut down by the boric acid injected into the RCS by either the emergency core cooling system (safety injection) or the actuation of the Safety Injection Tanks (SITs). Additionally, the event may be terminated due to reaching a dry-out condition in the affected steam generator.

The steamline break analysis discussed herein was performed to demonstrate that core coolable geometry is maintained. Assuming the most reactive CEA is stuck in its fully withdrawn position, and applying the most limiting single failure of one safety injection train, steamline break core response cases were examined. Although DNB and fuel cladding damage are not necessarily unacceptable consequences of a steamline break transient, the analysis described herein demonstrates that there is no consequential damage to the primary system, and that the core remains in place and intact, by showing that the DNB design-basis is satisfied following a steamline break.

The systems and components that provide the necessary protection against a steamline break are listed as follows.

- Safety injection system actuation by any of the following:
  - Pressurizer pressure - low
  - Containment pressure - high
- Closure of the main steam isolation valves (MSIVs) after receipt of any of the following:
  - Steam generator pressure - low
  - Containment pressure - high
  - Closure of the main feedwater isolation valves.

Each main steamline is connected to a steam generator through an exit nozzle with an effective flow area of 6.305 ft^2 and contains a venturi-type flow restrictor located upstream of the MSIV and inside containment. These flow restrictors limit the steam release rate during a steamline break transient. The nozzle flow restrictors limit the effective maximum steamline break size to 2.27 ft^2 per steam generator.

5.1.6.2 Method of Analysis

The analysis of the steamline break transient has been performed to demonstrate that the DNB design basis is satisfied and that the peak linear heat rate does not exceed the limit value. This is accomplished by showing that the calculated minimum DNBR is greater than the safety analysis limit DNBR of 1.45 (W-3 low pressure DNB correlation limit) and that the peak linear heat rate remains below 22 kW/ft. The overall analysis process is described as follows.

Using the RETRAN code (References 1 and 2), transient values of key plant parameters identified as statepoints (core average heat flux, core pressure, core inlet temperature, RCS flow rate, and core boron concentration) were calculated first. Next, the advanced nodal code (ANC) core design code was used to:

- Evaluate the nuclear response to the RCS cooldown so as to verify the RETRAN transient prediction of the average core power/reactivity
- Determine the peaking factors associated with the return to power in the region of the stuck CEA

Finally, using the RETRAN-calculated statepoints and the ANC-calculated peaking factors, the detailed thermal and hydraulic computer code VIPRE was used to calculate the minimum DNBR based on the W-3 DNB correlation. The peak linear heat rate is calculated based on the results of the ANC analysis.

The following assumptions were made in the analysis of the main steamline break:

1. A hypothetical double-ended rupture (DER) of a main steamline was postulated at HZP/hot shutdown conditions. The maximum break size seen by the faulted steam generator is limited to the flow area of the steam generator outlet nozzle (6.305 ft2). The maximum break size seen by the unfaulted steam generator is limited to the flow area of the inline flow restrictor (2.27 ft2). The initial conditions correspond to a subcritical reactor, an initial vessel average temperature at the no-load value of 532°F, and no core decay heat. These conditions are conservative, compared to hot full power, for a steamline break transient because the resultant RCS cooldown does not have to remove any latent heat. Also, the steam generator water inventory is greatest at no-load conditions, which increases the capability for cooling the RCS. Thus, the analysis of the hot zero power case bounds the case of a post-trip analysis from hot full power.

2. One DER case was analyzed. A case assuming a loss of offsite power is bounded by the case assuming that offsite power is maintained throughout the event (see Appendix A). Steamline break transients associated with the inadvertent opening of a steam dump or relief valve were not analyzed because the resultant RCS cooldown. Therefore, the minimum DNBR would be less limiting compared to the DER cases.

3. Perfect moisture separation within the steam generators was conservatively assumed.

4. An end-of-life shutdown margin of 3.6 %Δk corresponding to no-load, equilibrium xenon conditions, with the most reactive CEA stuck in its fully withdrawn position was assumed. The stuck CEA was assumed to be in the core location exposed to the greatest cooldown; that is, related to the faulted loop. The reactivity feedback model included a positive MDC

corresponding to an end-of-life rodded core with the most reactive CEA in its fully withdrawn position. The variation of the MDC due to changes in temperature and pressure was accounted for in the model. Figure 5.1.6-1 presents the keff versus temperature relationship at 900 psia corresponding to the assumed negative MTC plus the Doppler temperature feedback effect. The Doppler power feedback corresponding to the stuck CEA conditions is provided in Figure 5.1.6-2.

The reactivity and power predicted by RETRAN were compared to those predicted by the ANC core design code. The ANC core analysis considered the following:

- Doppler reactivity feedback from the high fuel temperature near the stuck CEA
- Moderator feedback from the water conditions near the stuck CEA
- Power redistribution effects
- Non-uniform core inlet temperature effects

The ANC core analysis confirmed that the RETRAN-predicted reactivity is acceptable.

5. The Moody critical flow curve was applied to conservatively maximize the break flow rate assuming no frictional losses.

6. The closure of the MSIV of the intact/unfaulted loop was conservatively modeled to be complete at 6.75 seconds after receipt of a low steam generator pressure (520 psia) signal from the same loop.

7. The safety injection pumps were assumed to provide flow to the RCS at 20 seconds after receipt of a safety injection signal on low pressurizer pressure (1646 psia). This delay accounts for signal processing and pump startup delays.

8. The minimum capability for the injection of boric acid solution, corresponding to the most restrictive single active failure in the SIS, was assumed. The assumed safety injection flow (see Figure 5.1.6-3) corresponds to the operation of one high-head safety injection pump. Boric acid solution from the refueling water tank (RWT), with a minimum concentration of 1720 ppm and a minimum temperature of 51°F, was the assumed source of the safety injection flow. The safety injection lines downstream of the RWT were assumed to initially contain unborated water to conservatively maximize the time it takes to deliver the highly concentrated RWT boric acid solution to the reactor coolant loops.

9. Main feedwater flow equal to the nominal (100% power) value was assumed to initiate coincident with the postulated break. Feedwater flow to the unfaulted loop was maintained until a feedwater isolation signal was generated. Feedwater isolation to the faulted loop was delayed an additional 90 seconds beyond the receipt of a feedwater isolation signal. The feedwater isolation to the faulted steam generator was assumed to be complete at 92.9 seconds after the steam generator pressure in the faulted loop reaches the low setpoint.

10. A minimum SGTP level of 0% was assumed to maximize the heat transfer capabilities of the steam generators and therefore maximize the cooldown of the RCS.

11. Due to the design of the auxiliary feedwater system which automatically isolates the auxiliary feedwater from the broken loop, no auxiliary feedwater was assumed to be delivered during the event.

5.1.6.3 Results

The results of the statepoint evaluation demonstrate that the post-trip steamline break event meets the applicable DNBR and peak linear heat rate acceptance criteria. The time sequence of events is presented in Table 5.1.6-1.

Figures 5.1.6-4 through 5.1.6-11 show the steam pressure, steam flow, pressurizer pressure, pressurizer water volume, reactor vessel inlet temperature, core heat flux, core boron concentration, and core reactivity following a double-ended rupture of a main steamline at initial no-load conditions with offsite power available (full reactor coolant flow). The break size was limited to 6.305 ft^2 on the faulted steam generator and to 2.27 ft^2 on the unfaulted steam generator by the inline flow restrictors. Both steam generators were assumed to discharge through the break until steamline isolation had occurred.

5.1.6.4 Conclusions

The main steamline break transient was conservatively analyzed with respect to the reactor core response. Key analysis assumptions were made to conservatively maximize the cooldown of the RCS, so as to maximize the positive reactivity insertion, and thus maximize the peak return to power. Other key assumptions include: end-of-life shutdown margin with the most-reactive CEA stuck in its fully withdrawn position, maximum delays in actuating engineered safeguard features such as safety injection, main steam isolation and feedwater isolation, and minimum safety injection flow with a minimum boron concentration.

A DNBR statepoint analysis was performed for the post-trip steamline break event with offsite power. The case with offsite power available—that is, the case with full reactor coolant flow—was found to be the limiting case. The minimum DNBR was determined to be greater than the DNBR safety analysis limit, and thus the DNBR design basis is met. Additionally, the peak linear heat rate was demonstrated not to exceed the limit value.

5.1.6.5 References

1. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April, 1999.

2. EPRI NP-1850-CCM, Rev. 6, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," December 1995.

Table 5.1.6-1 Post-Trip Steamline Break Analysis Assumptions and Sequence of Events		
Event	**Time (sec)**	**Value**
MSLB (6.305 ft^2 DER) Transient Initiated	0.01	---
Manual Reactor Trip Initiated	0.01	---
MSIV/MFIV Closure Signal on Low Steam Generator Pressure	3.36	520 psia
Feedwater Isolation (MFIV Closure) on Loop 2	8.51	5.15 sec. delay
Steamline Isolation (MSIV Closure) on Loops 1 and 2	10.11	6.75 sec. delay
SI Actuation Signal on Low Pressurizer Pressure	13.71	1646 psia
Core Criticality Attained	48.05	---
Feedwater Isolation (MFIV Closure) on Loop 1	92.92	90.0 sec. delay
Peak Heat Flux Reached	305.50	18.25%
Minimum DNBR Reached	305.50	1.50
Peak Linear Heat Rate Reached	305.50	21.08 kW/ft

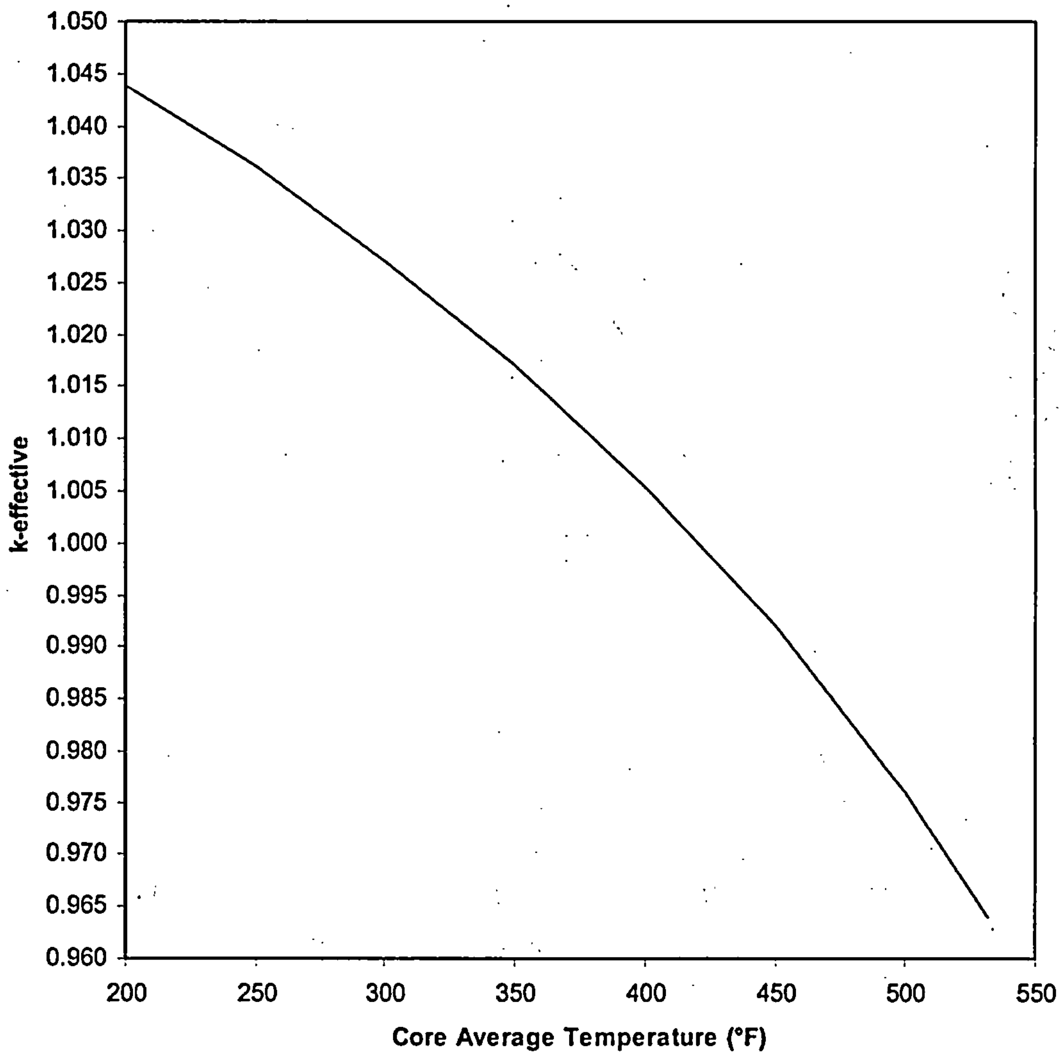

Figure 5.1.6-1 Variation of K_{eff} with Core Temperature (3.6 %Δk SDM, 900 psia)

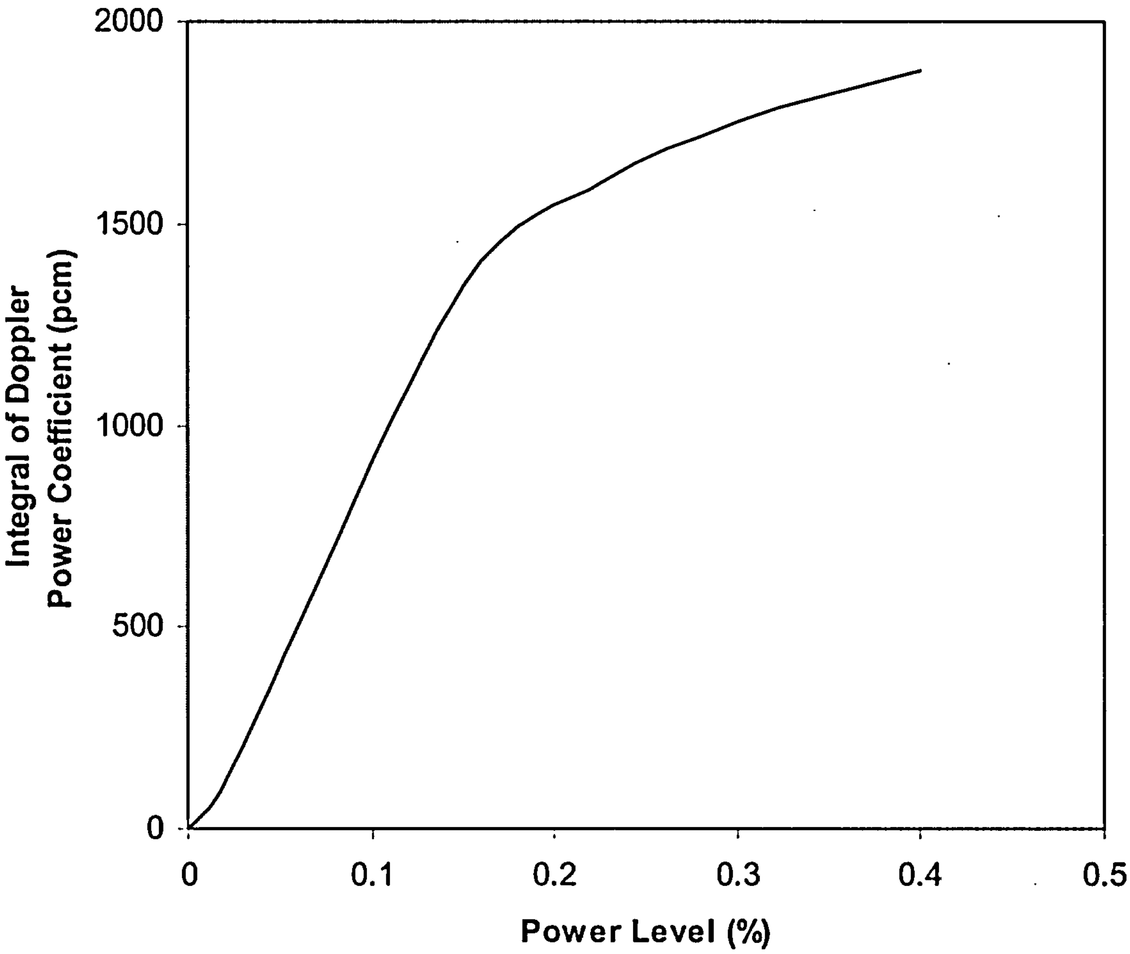

Figure 5.1.6-2 Stuck CEA Doppler Power Feedback

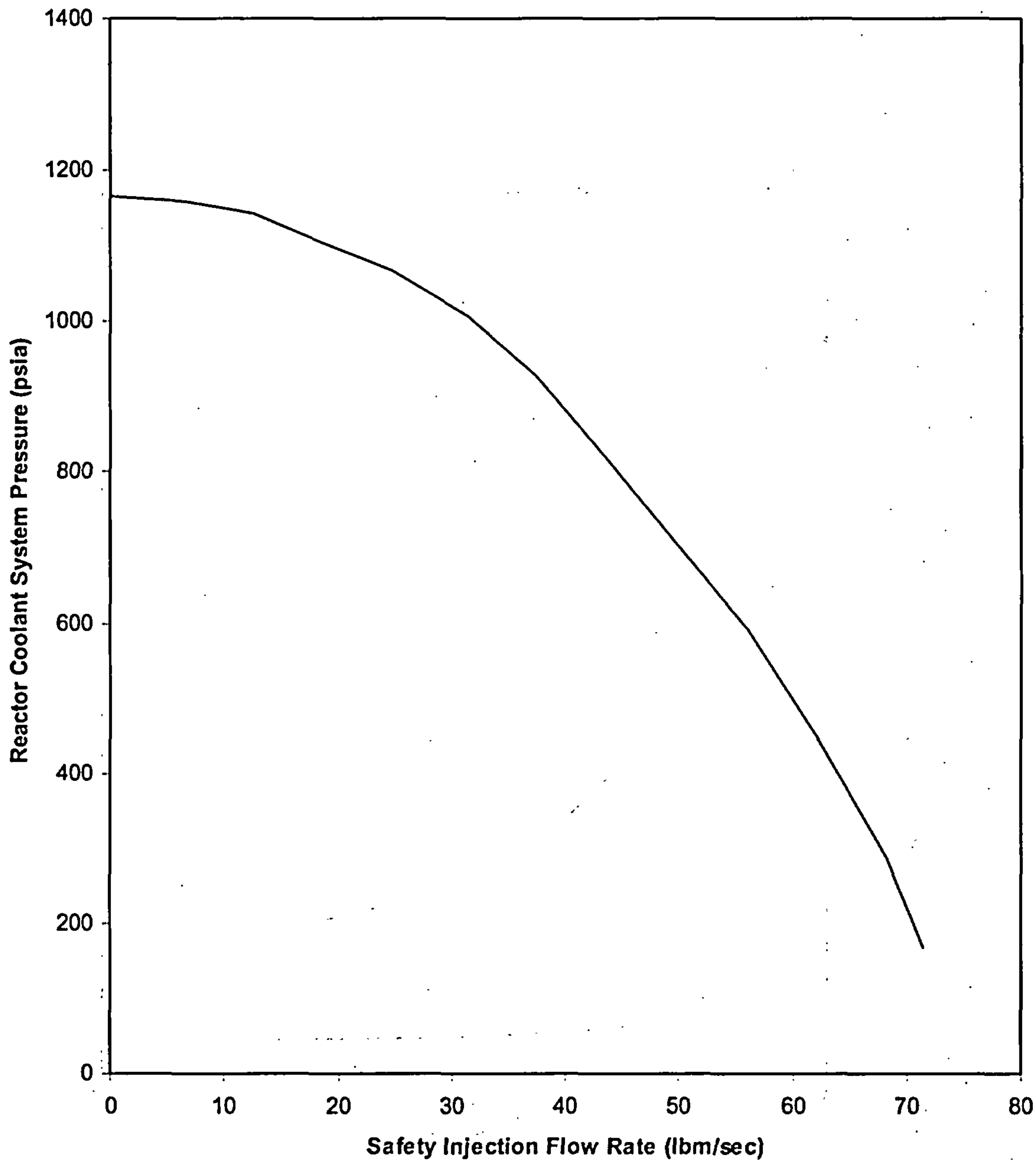

Figure 5.1.6-3 Safety Injection Curve

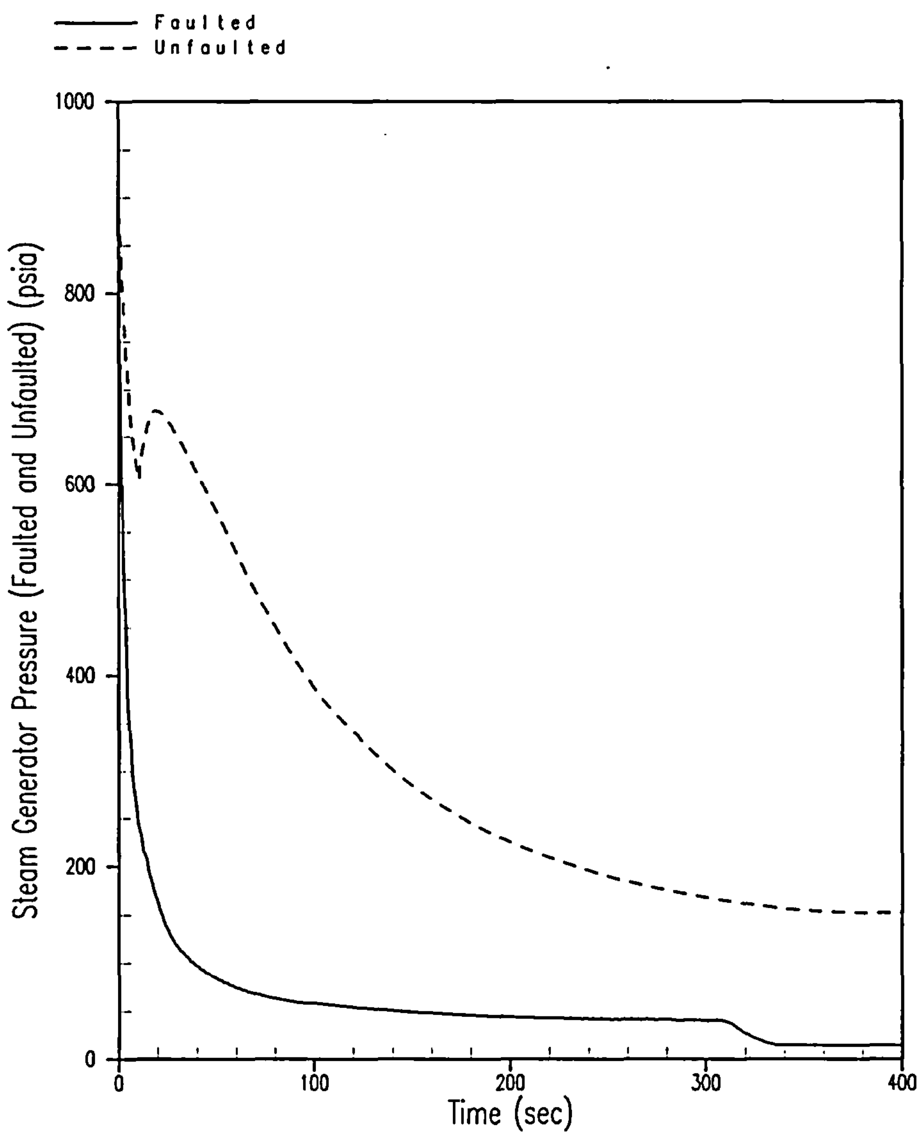

Figure 5.1.6-4 Post-Trip Main Steamline Break – Steam Generator Steam Pressure versus Time

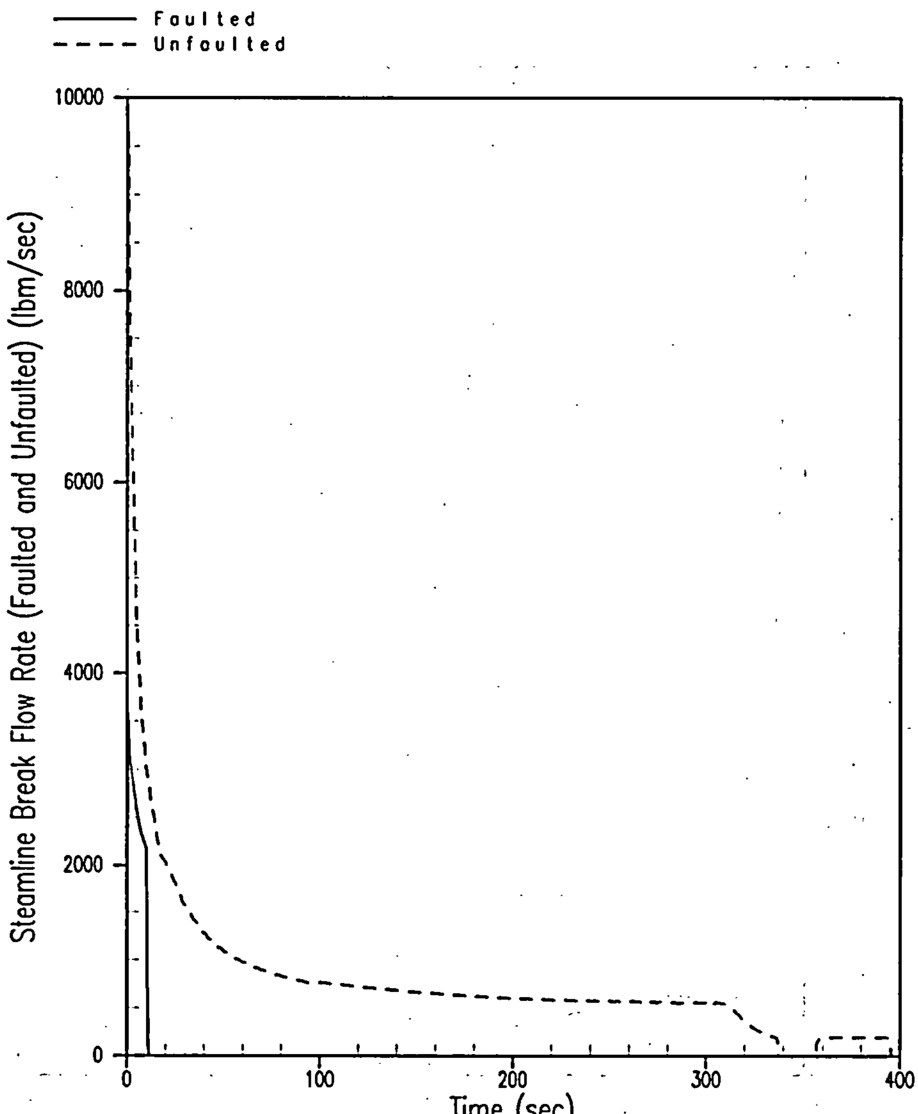

Figure 5.1.6-5 Post-Trip Main Steamline Break – Break Mass Flow Rate versus Time

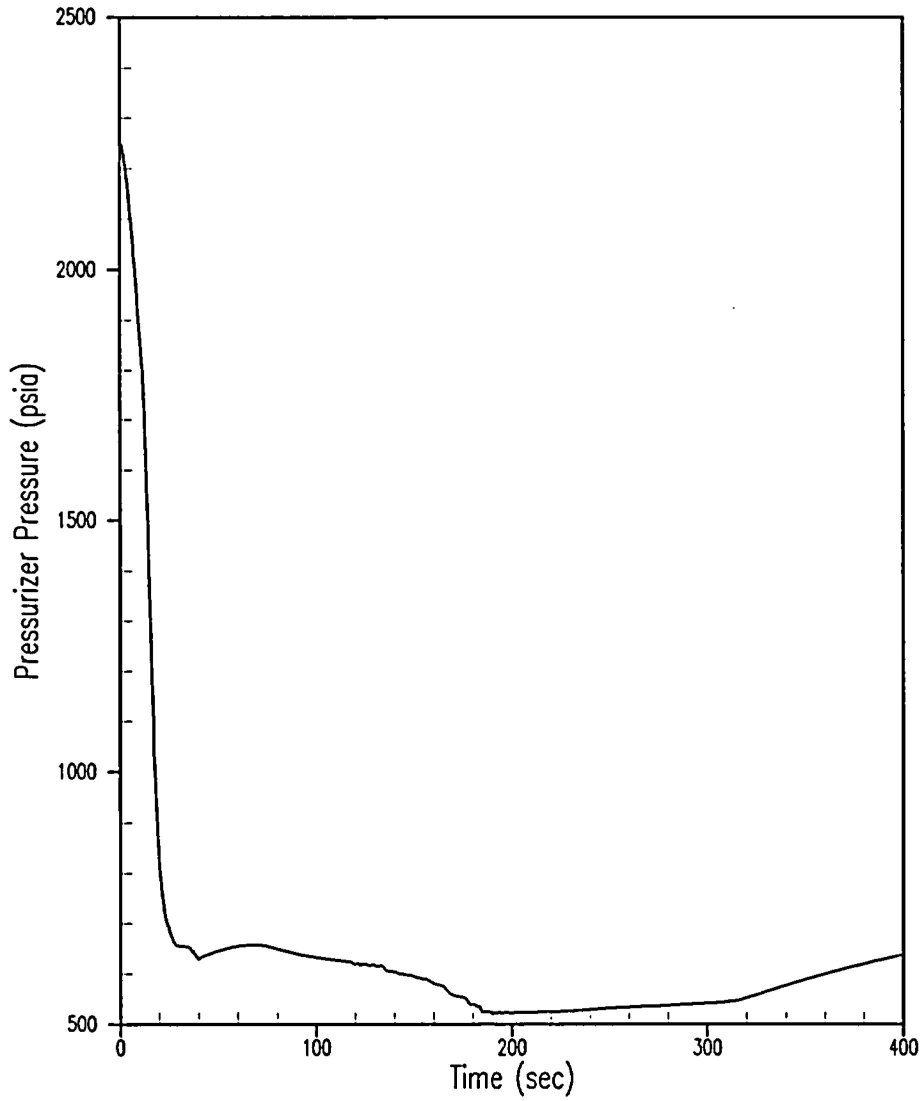

Figure 5.1.6-6 Post-Trip Main Steamline Break – Pressurizer Pressure versus Time

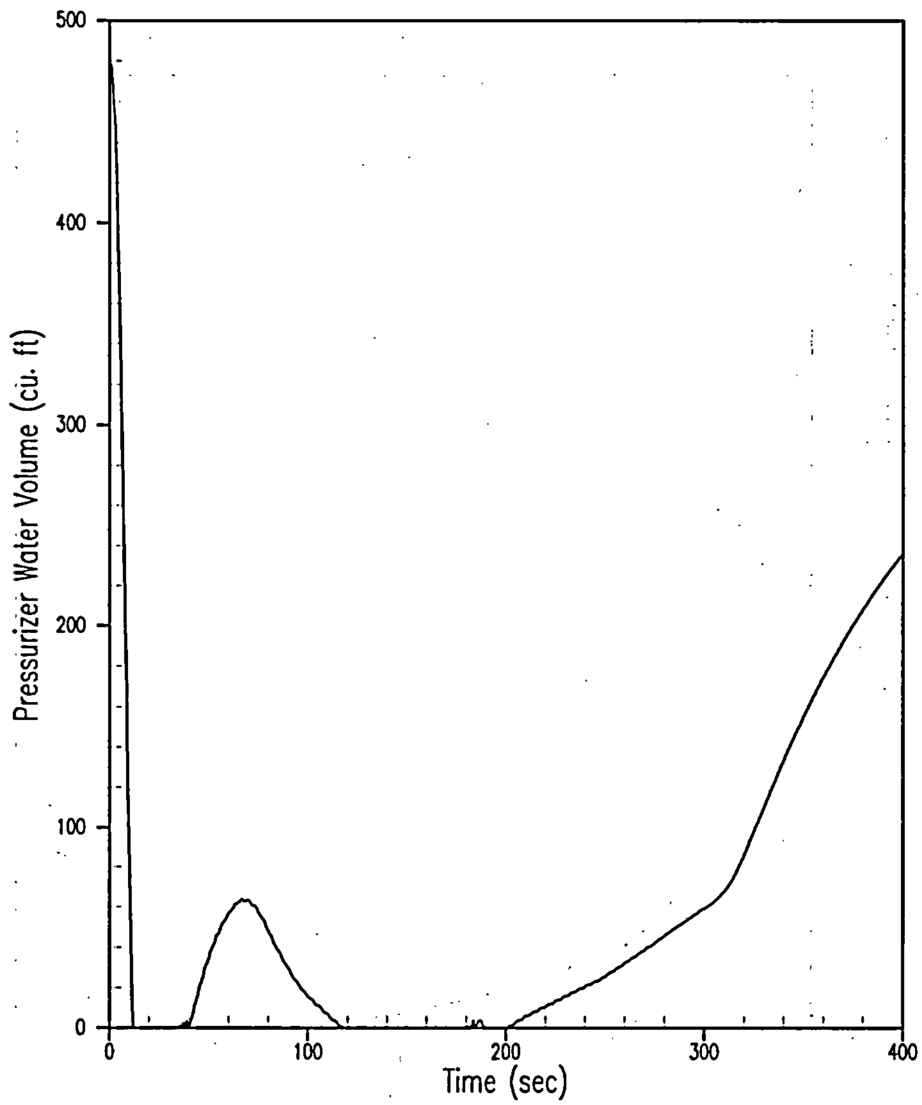

Figure 5.1.6-7 Post-Trip Main Steamline Break – Pressurizer Water Volume versus Time

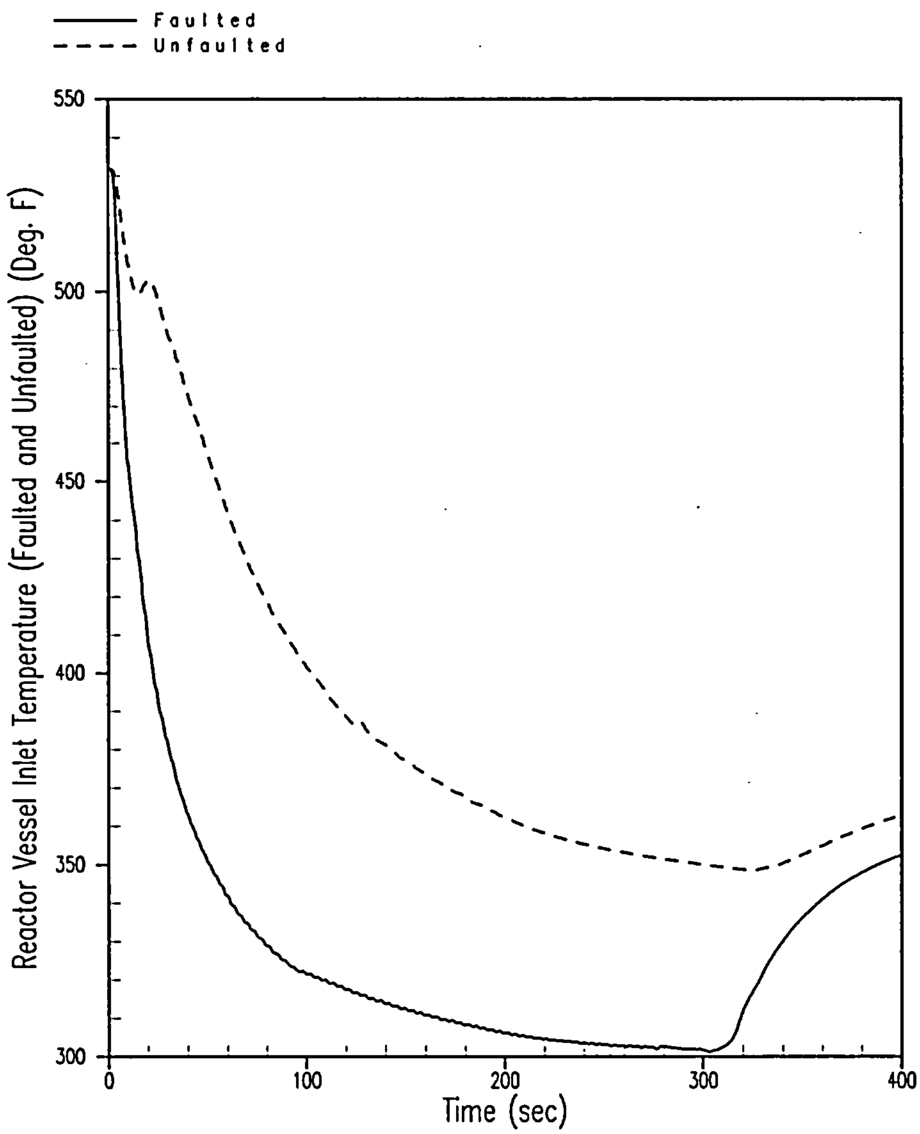

Figure 5.1.6-8 Post-Trip Main Steamline Break – Reactor Vessel Inlet Temperature versus Time

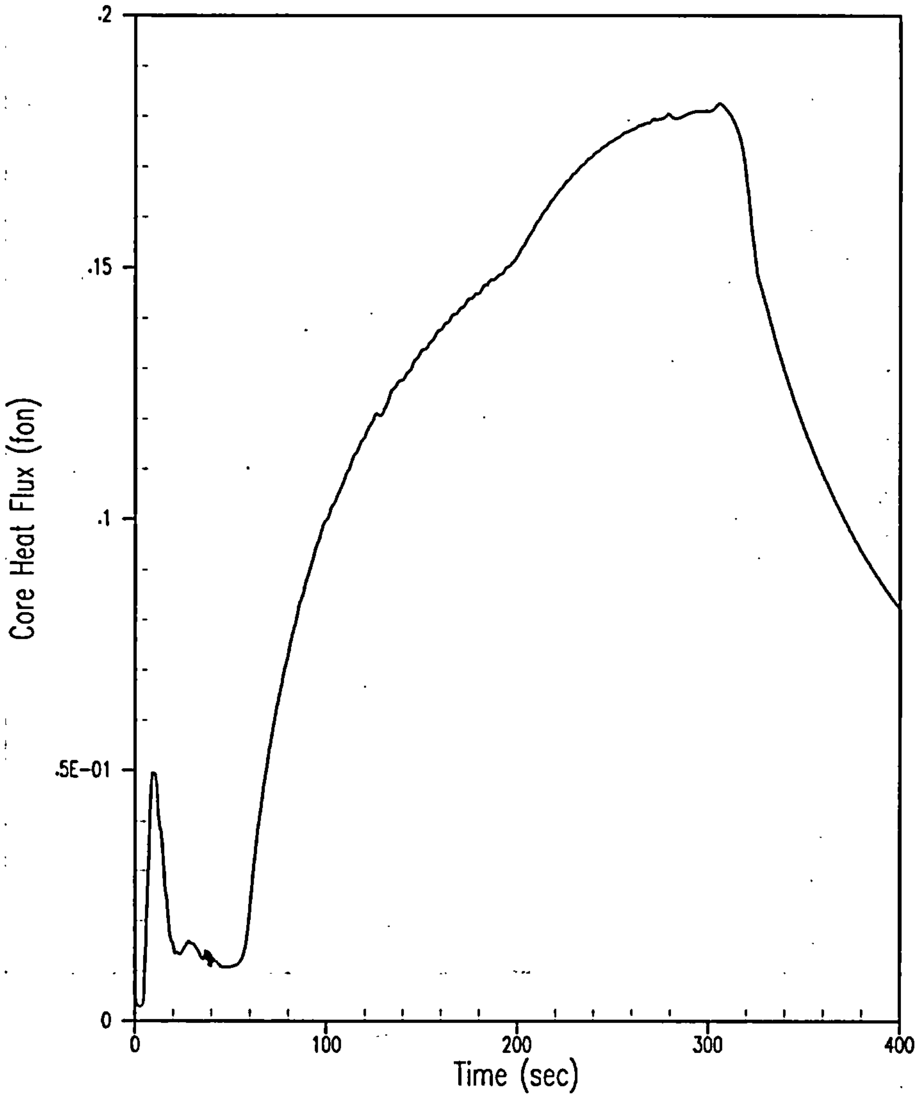

Figure 5.1.6-9 Post-Trip Main Steamline Break – Core Heat Flux versus Time

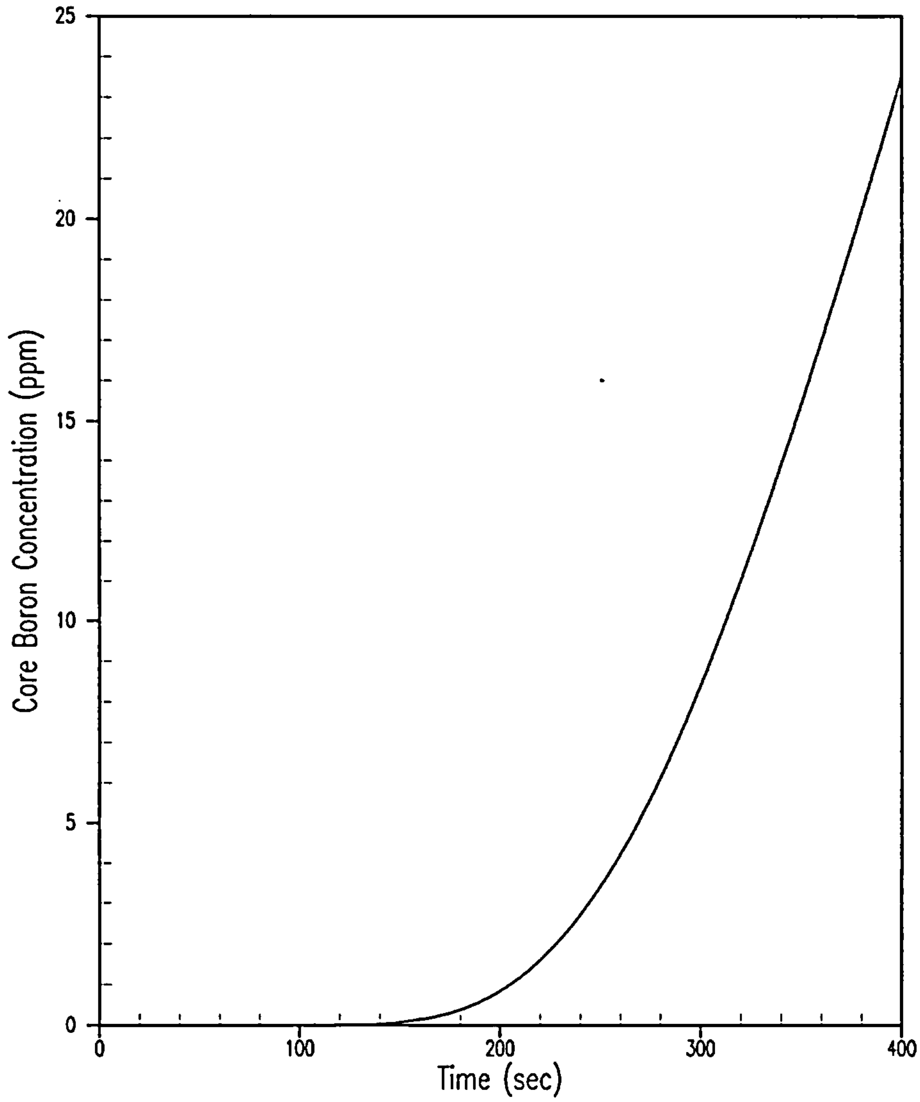

Figure 5.1.6-10 Post-Trip Main Steamline Break – Core Averaged Boron Concentration versus Time

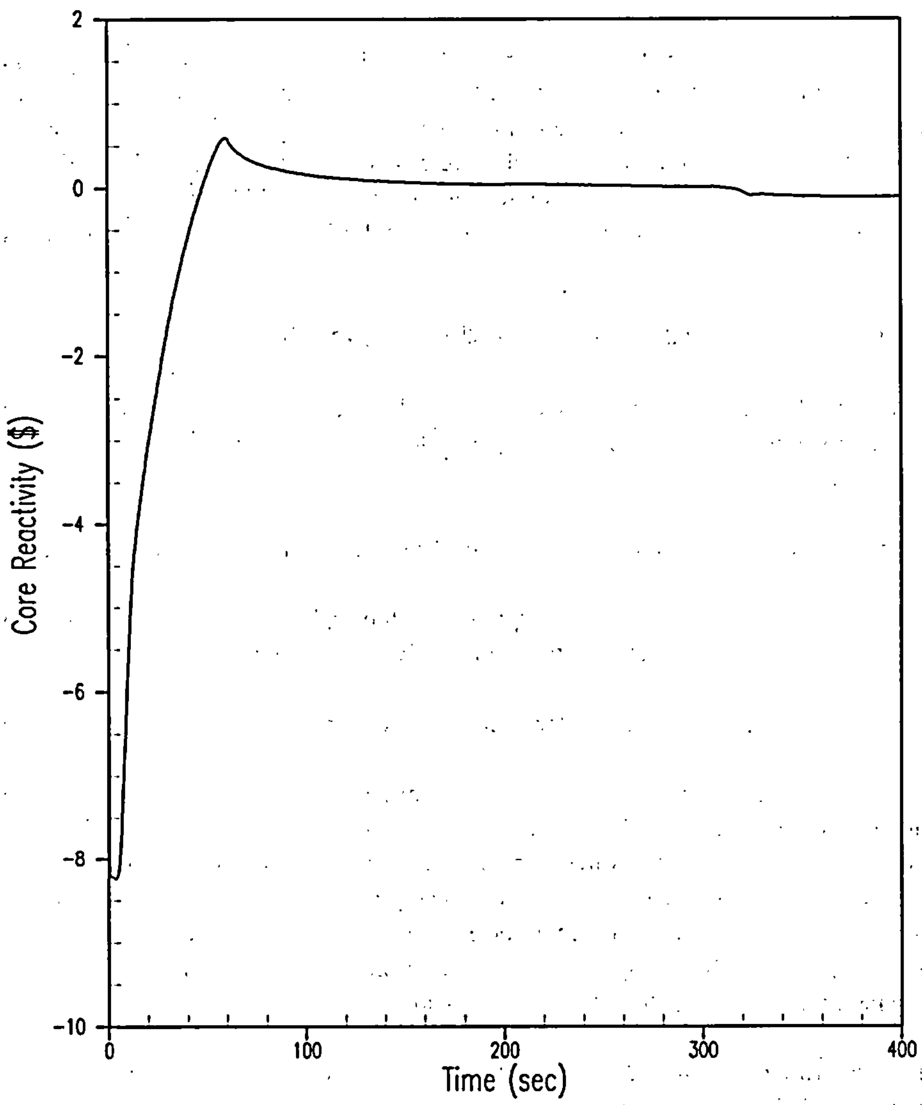

Figure 5.1.6-11 Post-Trip Main Steamline Break – Reactivity versus Time

5.1.7 Steam System Piping Failures Outside Containment

The steam system piping failures outside containment are bounded by the steam system piping failures inside containment discussed in Section 5.1.5 and 5.1.6.

5.1.8 Turbine Trip

A Turbine Trip is caused by an electrical or mechanical malfunction of the turbine, which produces a reduction of steam flow from the steam generators to the turbine due to the closure of the turbine stop valves. The core and system performance following a Turbine Trip would be no more adverse than those following a Loss of Condenser Vacuum, which is described in Section 5.1.10. The radiological consequences due to steam releases from the secondary system would be less severe than the consequences of the Feedwater Line Break (see Section 5.1.12). Therefore, a detailed analysis of the Turbine Trip event was not necessary.

5.1.9 Loss of Normal Feedwater Flow and Loss of Offsite Power

The Loss of Normal Feedwater Flow (LONF) event is defined as a complete loss of main feedwater flow while the reactor is operating at the maximum power level. A loss of main feedwater flow may occur due to the following causes:

- Breaks in the main feedwater system piping upstream of the main feedwater check valves
- Failure or trip of the main feedwater pumps, including loss of power (for motor-driven feedwater pumps) or loss of motive steam (for turbine-driven feedwater pump).
- Spurious closure of main feedwater isolation valves or the main feedwater regulating valves.

The immediate consequence of a loss of main feedwater flow is a reduction in the steam generator water level, which if left unmitigated, will ultimately result in a reactor trip and auxiliary feedwater (AFW) actuation on a low steam generator water level signal. Following reactor trip, the rate of heat generation in the RCS (decay heat plus reactor coolant pump heat input) may exceed the heat removal capability of the secondary system. In this case, there will be an increase in the steam generator pressure and an increase in RCS pressure, RCS temperature, and pressurizer water level. This trend continues until the RCS heat generation rate falls below the secondary-side heat removal capability.

At that time, the primary pressure and temperature begin to decrease, thereby terminating the transient in terms of potential challenges to the applicable safety criteria. It is assumed that if such a transient were to occur at the plant, emergency operating procedures would be followed to bring the plant to a stable condition.

A Loss-of-Offsite-Power (LOOP) event is identical to the LONF event except that a loss of offsite power is assumed to occur simultaneously with or shortly after reactor trip. All systems that are not powered by the emergency diesel generators would not be available after the LOOP. This includes the reactor coolant pumps. Therefore, the reactor coolant pumps are assumed to trip shortly after the reactor trip, and the post-trip heat removal from the core relies upon natural circulation in the RCS loops.

The LONF and LOOP events, in conjunction with the Feedline Break event, impose the greatest demand on the AFW system and are considered to be the design-basis events for evaluating the performance of the AFW system.

The consequences of these events are bounded by other analyzed events as follows:

- With respect to core consequences, the LONF and LOOP events are not as limiting as the Loss of Flow event, which is analyzed to demonstrate that the DNB design basis is satisfied. The LONF and LOOP events result in a slight increase in the RCS temperature prior to reactor trip and there is no appreciable increase in the core power. In the case of the LOOP event, it is assumed that the reactor is tripped prior to the LOOP and thus, the loss of flow occurs when the core power has effectively been turned around. The Loss of Flow event is limiting with respect to demonstrating that the DNB design basis is satisfied since it results in a significant reduction in the RCS flow. The Loss of Flow event bounds the LONF and LOOP events since the effect of the reduction in RCS flow is more significant than the effect of the increase in the RCS temperatures observed for the LONF and LOOP events, prior to reactor trip. In addition, since there is no appreciable power increase in either the LONF or LOOP event and since the fuel centerline melting is primarily driven by the core power, that is, kW/ft, the fuel centerline melt limits will not be challenged. Therefore, the DNB and fuel centerline melting criteria continue to be satisfied for the LONF and LOOP events.

- With respect to overpressurization, the Loss of Condenser Vacuum/Turbine Trip event (LOCV) will be more limiting than either the LONF or LOOP events. The LOCV presents a much more significant reduction in the heat removal capability of the steam generators than the LONF or LOOP events because the LOCV event combines the loss of normal feedwater with a turbine trip. This causes the LOCV to have a faster pressurization of the RCS compared to the LONF and LOOP events. In addition, the LOOP does not credit the early reactor trip on the low RCS flow reactor trip. The net result for the LOCV event is a total loss of secondary heat sink, which results in the greatest challenge to primary and secondary overpressurization. Therefore, the LOCV remains the most limiting event with respect to primary and secondary overpressurization.

- With respect to long-term cooling, the ability of the auxiliary feedwater system to remove decay heat following reactor trip is demonstrated by the analyses presented in Chapter 10 of the Updated Final Safety Analysis Report (UFSAR).

It is the conclusion of this evaluation that for both the LONF and LOOP events, all criteria are bounded by other events. Therefore, no new analysis is required to support the transition to the WCAP-9272 reload methodology or to support the 30% steam generator tube plugging analysis assumption.

5.1.10 Loss of Condenser Vacuum

5.1.10.1 Accident Description

The Loss of Condenser Vacuum event is defined as a complete loss of steam load or a turbine trip from full power without a direct reactor trip. This anticipated transient is analyzed as a turbine trip from full power with a simultaneous loss of feedwater to both steam generators due to low suction pressure on the feedwater pumps. The atmospheric dump valves and the steam dump and bypass system valves are assumed to be unavailable, which minimizes the amount of cooling and maximizes the RCS and secondary peak pressures during the event.

In the event of a large loss of load in which the steam dump valves fail to open, the main steam safety valves (MSSVs) may lift and the reactor may be tripped by either of the following signals: high pressurizer pressure, or thermal margin / low pressure (TM/LP). No credit is taken for the Loss of Load trip as it is a control grade trip only. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. However, the pressurizer safety valves (PSVs) and MSSVs are sized to protect the RCS and steam generators against overpressure for all load losses without assuming the operation of the steam dump system. The RCS and main steam system (MSS) steam relieving capacities were designed to ensure safety of the unit without requiring pressurizer pressure control, steam bypass control systems, or a reactor trip on turbine trip.

5.1.10.2 Method of Analysis

The methodology for the Loss of Condenser Vacuum event is similar to that in WCAP-9272 with the exceptions noted below. The RETRAN code is used in the analysis. Input parameters from Table 5.1.0-2 have been incorporated into the analysis. The exceptions are:

1. An initial core power of 2714.2 MWt, based on a rated power of 2700 MWt and nominal pump heat for the DNB case. A core power of 2768.5 MWt, based on a rated power of 2700 MWt, nominal pump heat, and 2% uncertainty is assumed for the overpressure cases.

2. The maximum core average temperature is assumed since this maximizes the expansion of the primary system fluid.

3. A maximum of 30% of the steam generator U-tubes are assumed to be plugged.

The loss of condenser vacuum event is analyzed using the RETRAN computer code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and MSSVs. The code computes pertinent plant variables including temperatures, pressures, and power levels.

The loss of condenser vacuum is analyzed: (1) to confirm that the PSVs and MSSVs are adequately sized to prevent overpressurization of the primary RCS and MSS, respectively; and (2) to ensure that the increase in RCS temperature does not result in a DNB in the core.

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power with no credit taken for a direct reactor trip on turbine trip. This assumption will delay reactor trip until conditions in the RCS cause a trip on some other signal. Thus, the analysis assumes a worst-case transient and demonstrates the adequacy of the pressure relieving devices and plant-specific RPS setpoints assumed in the analysis for this event.

Of the three cases analyzed, one is performed to address DNB concerns, one ensures that the peak primary RCS pressure remains below the design limit (2750 psia), and the final case confirms that the peak MSS pressure remains below 110% of the steam generator shell design pressure (1100 psia). The major assumptions for these cases are summarized as follows.

For the case analyzed to demonstrate that the core thermal limits are adequately protected, minimum reactivity feedback conditions with automatic pressurizer pressure control are modeled. For this case, initial core power, reactor coolant temperature, and reactor coolant pressure are assumed to be at the nominal values consistent with steady-state full power operation. Uncertainties in initial conditions are included in determining the DNBR limit value. For the case analyzed to demonstrate the adequacy of the primary pressure-relieving devices minimum reactivity feedback conditions without automatic pressurizer pressure control are modeled. For this case, initial core power and reactor coolant temperature are assumed at the maximum values consistent with steady-state full-power operation, including allowances for calibration and instrument errors. Initial pressurizer pressure is assumed at the minimum value for this case, since it delays reactor trip on high pressurizer pressure and results in more severe primary-side temperature and pressure transients. The MSS overpressurization case differs from the primary RCS overpressurization case in that automatic pressurizer pressure control is assumed in order to delay reactor trip.

The loss of condenser vacuum event results in a primary system heatup and is therefore conservatively analyzed assuming minimum reactivity feedback. This includes assuming an MTC value consistent with HFP conditions and a least negative Doppler power coefficient (DPC).

Three cases were analyzed:

1. For the case analyzed for DNB, automatic pressurizer pressure control is assumed. Thus, full credit is taken for the effect of the pressurizer spray in reducing or limiting the primary coolant pressure. Safety valves are also available and are modeled assuming a - 3% setpoint tolerance.

2. For the case analyzed for primary RCS overpressure concerns, it is assumed that automatic pressurizer pressure control is not available. Therefore, no credit is taken for the effect of the pressurizer spray or power-operated relief valve (PORVs) in reducing or limiting the primary coolant pressure. Pressurizer safety valves are assumed operable, but are modeled assuming a + 3% setpoint tolerance.

3. For the case analyzed for MSS overpressure concerns, it is assumed that automatic pressurizer pressure control is available. Credit is taken for the effect of the pressurizer spray in reducing or limiting the primary coolant pressure, thus conservatively delaying the actuation of the reactor trip signal. Delaying the reactor trip ensures that the energy input to the secondary system, and subsequently the MSS pressure, is maximized. The PORVs are modeled with one valve aligned to the pressurizer and one valve locked out. The PORVs are actuated upon the receipt of the high pressurizer pressure trip signal and serve to protect the PSVs against spurious actuation by limiting the primary pressure increase post-trip.

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for AFW flow since a stabilized plant condition will be reached before AFW initiation is normally assumed to occur for full-power cases. However, the AFW pumps would be expected to start on a trip of the main feedwater pumps. The AFW flow would remove core decay heat following plant stabilization.

The analysis is performed for operation with a maximum steam generator tube plugging level (uniform) for St. Lucie Unit 2 of ≤30%.

5.1.10.3 Results

The transient responses for a total loss of condenser vacuum from full-power operation are shown in Figures 5.1.10-1 through 5.1.10-5 for the RCS overpressure case and Figures 5.1.10-6 through 5.1.10-10 for the DNB case.

The total loss of condenser vacuum event was analyzed assuming the plant to be initially operating at full power at BOC with no credit taken for the pressurizer spray or PORVs to maximize the primary RCS pressure response. Figures 5.1.10-1 through 5.1.10-5 show the transients for this case. The neutron flux remains relatively constant prior to reactor trip, while pressurizer pressure, pressurizer water volume, and RCS average temperature increase due to the sudden reduction in primary to secondary heat transfer. The reactor is tripped on the high pressurizer pressure trip signal. In this case, the PSVs are actuated and maintain the primary RCS pressure below 110% of the design value. Table 5.1.10-1 summarizes the sequence of events and limiting conditions for this case.

Figures 5.1.10-6 through 5.1.10-10 show the transient responses for the event with minimum feedback reactivity coefficients and assuming full credit for the pressurizer spray to calculate the transient DNBR response. Following event initiation, the pressurizer pressure and average RCS temperature increase due to the rapidly reduced steam flow and heat removal capacity of the secondary side. The peak pressurizer pressure and water volume and RCS average temperature are reached shortly after the reactor is tripped on the high pressurizer pressure trip signal. The DNBR initially increases slightly, then decreases until the reactor trip is tripped, and finally, following reactor trip, increases rapidly. The minimum DNBR remains well above the safety analysis limit value. Table 5.1.10-2 summarizes the sequence of events and limiting conditions for this case.

Table 5.1.10-3 summarizes the transient response for the total loss of steam load with minimum feedback reactivity coefficients assuming full credit for the pressurizer spray to maximize the MSS pressure response. Following event initiation, the pressurizer pressure and average RCS temperature increase due to the rapidly reduced steam flow and heat removal capacity of the secondary side. The peak pressurizer pressure and water volume and RCS average temperature are reached shortly after the reactor is tripped on the high pressurizer pressure trip signal. The MSS pressure increases, resulting in the actuation of the MSSVs, and then decreases rapidly following reactor trip. The MSSVs actuate to limit the MSS pressure below 110% of the steam generator shell design pressure. The transient response for the MSS pressure case is similar to that shown for the peak RCS pressure case (Figures 5.1.10-1 through 5.1.10-5).

5.1.10.4 Conclusions

The results of the analyses show that the plant design is such that a loss of condenser vacuum without a direct or immediate reactor trip presents no hazard to the integrity of the primary RCS or MSS. Pressure-relieving devices that have been incorporated into the plant design are adequate to limit the maximum pressures to within the safety analysis limits, i.e., 2750 psia for the primary RCS and 1100 psia for the MSS. The integrity of the core is maintained by operation of the RPS, i.e., the minimum DNBR is maintained above the safety analysis limit value of 1.42. Therefore, the WCAP-9272 methodology demonstrates that no core safety limit will be violated as a result of implementing up to 30% steam generator tube plugging.

Table 5.1.10-1 Sequence of Events and Transient Results Loss of Condenser Vacuum Without Pressurizer Pressure Control (for Primary RCS Overpressure)	
Event	**Time (seconds)**
Turbine Trip	10.1
Main Feedwater Terminates (both loops)	10.1
Reactor trip on High Pressurizer Pressure	18.9
Rod Motion Begins	19.6
Time of Peak RCS Pressure	20.9
Peak RCS Pressure	2691 psia
RCS Pressure Limit	2750 psia

Table 5.1.10-2 Sequence of Events and Transient Results Loss of Condenser Vacuum With Pressurizer Pressure Control (for Minimum DNB)	
Event	**Time (seconds)**
Turbine Trip	10.1
Main Feedwater Terminates (both loops)	10.1
Reactor Trip on High Pressurizer Pressure	20.3
Rod Motion Begins	21.0
Time of Minimum DNBR	22.1
Minimum DNBR Value	2.19
DNBR Limit	1.42

Table 5.1.10-3 Sequence of Events and Transient Results Loss of Condenser Vacuum With Pressurizer Pressure Control (for Main Steam System Overpressure)	
Event	**Time (seconds)**
Turbine Trip	10.1
Main Feedwater Terminates (both loops)	10.1
Reactor Trip on High Pressurizer Pressure	20.2
Rod Motion Begins	21.0
Time of Peak MSS Pressure	21.4
Peak MSS Pressure	1088 psia
MSS Pressure Limit	1100 psia

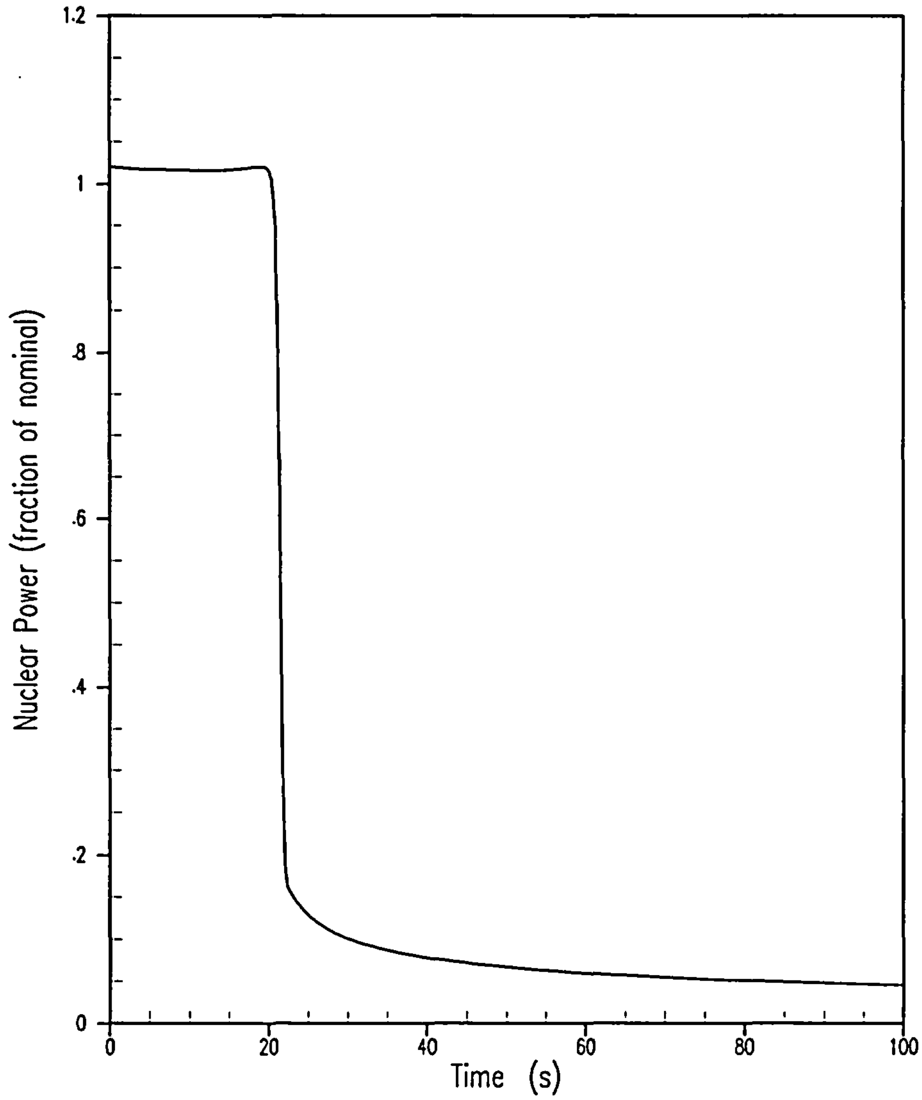

Figure 5.1.10-1 Loss of Condenser Vacuum (RCS Overpressure Case) Nuclear Power

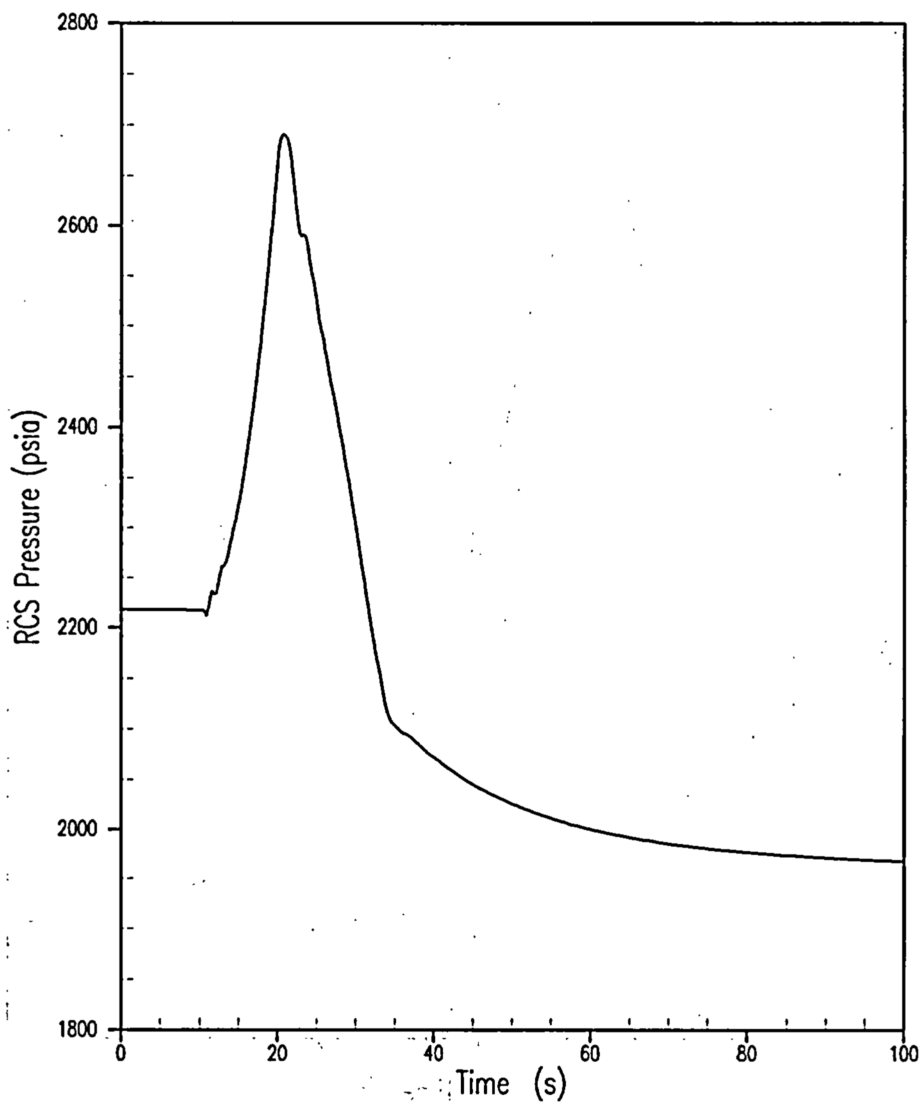

Figure 5.1.10-2 Loss of Condenser Vacuum (RCS Overpressure Case) RCS Pressure (RCP Outlet Pressure)

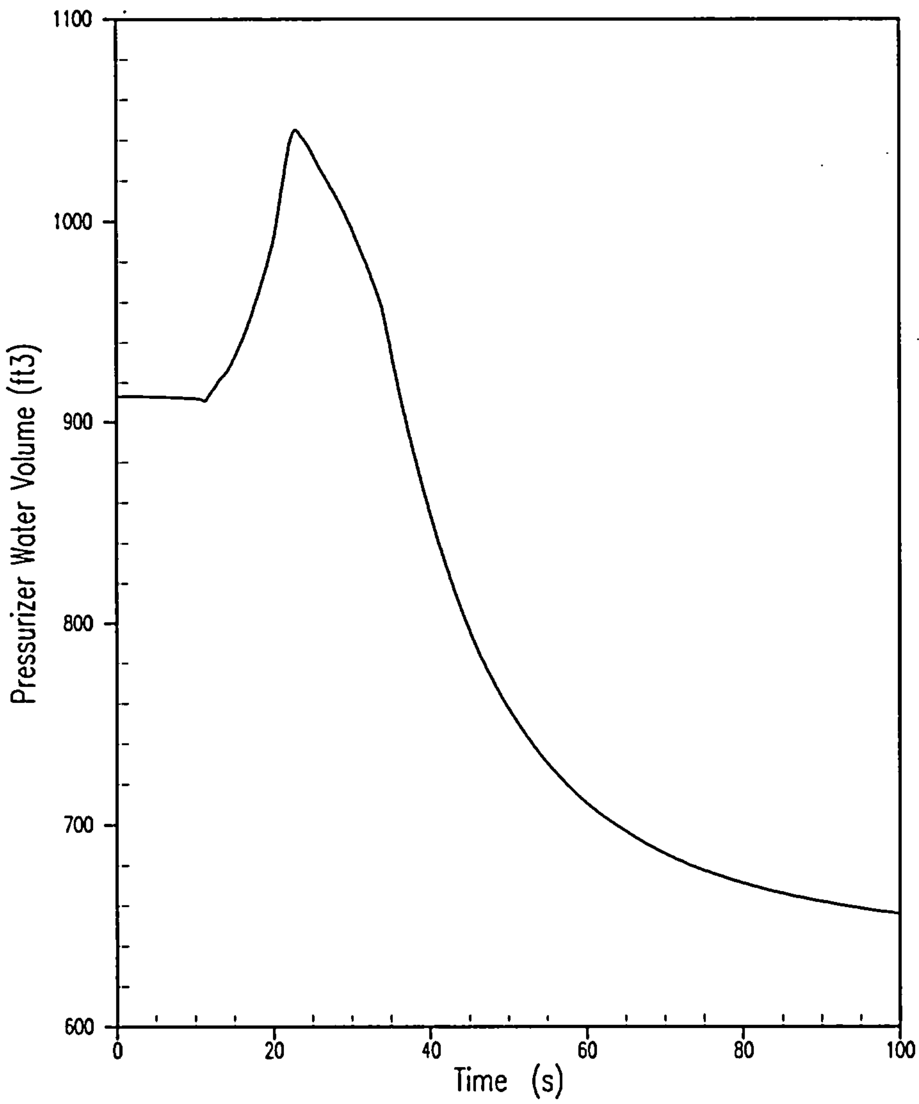

Figure 5.1.10-3 Loss of Condenser Vacuum (RCS Overpressure Case) Pressurizer Water Volume

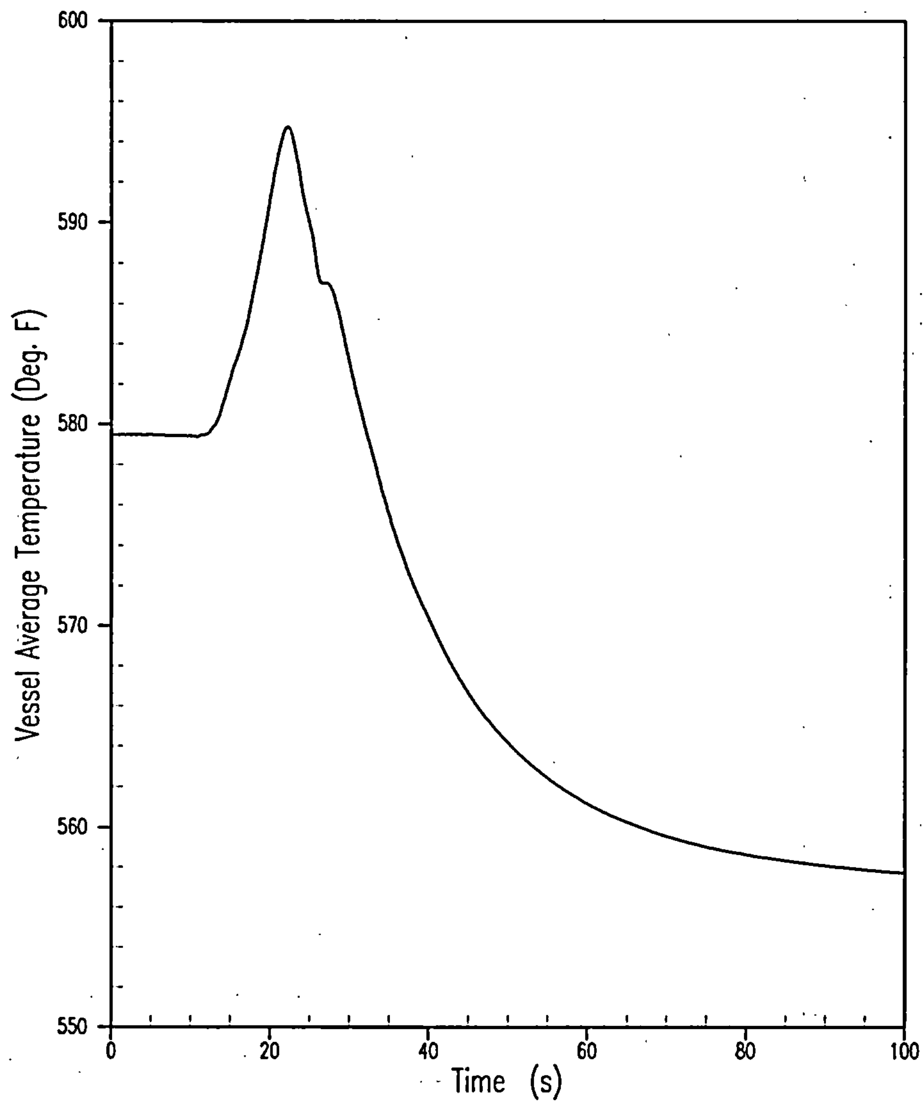

Figure 5.1.10-4 Loss of Condenser Vacuum (RCS Overpressure Case) Vessel Average Temperature

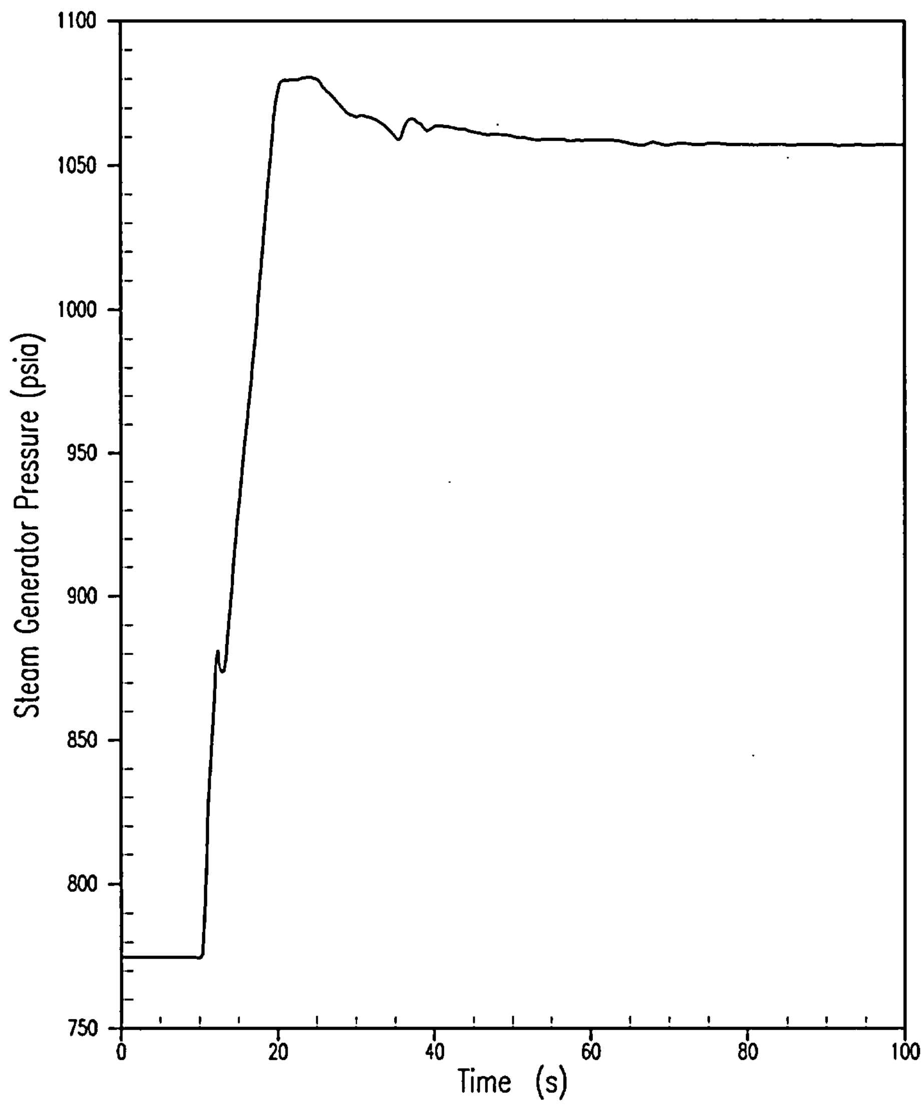

Figure 5.1.10-5 Loss of Condenser Vacuum (RCS Overpressure Case) Steam Generator Pressure

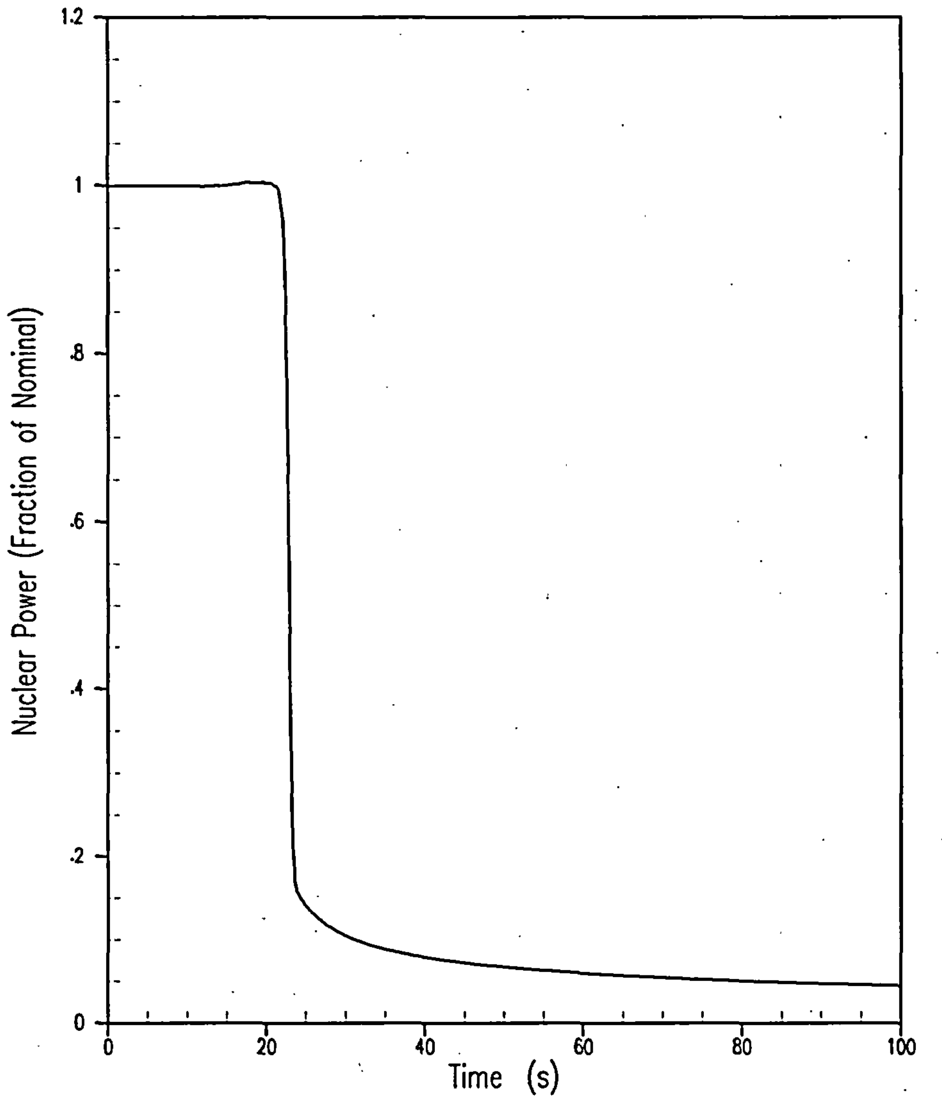

Figure 5.1.10-6 Loss of Condenser Vacuum (DNB Case) Nuclear Power

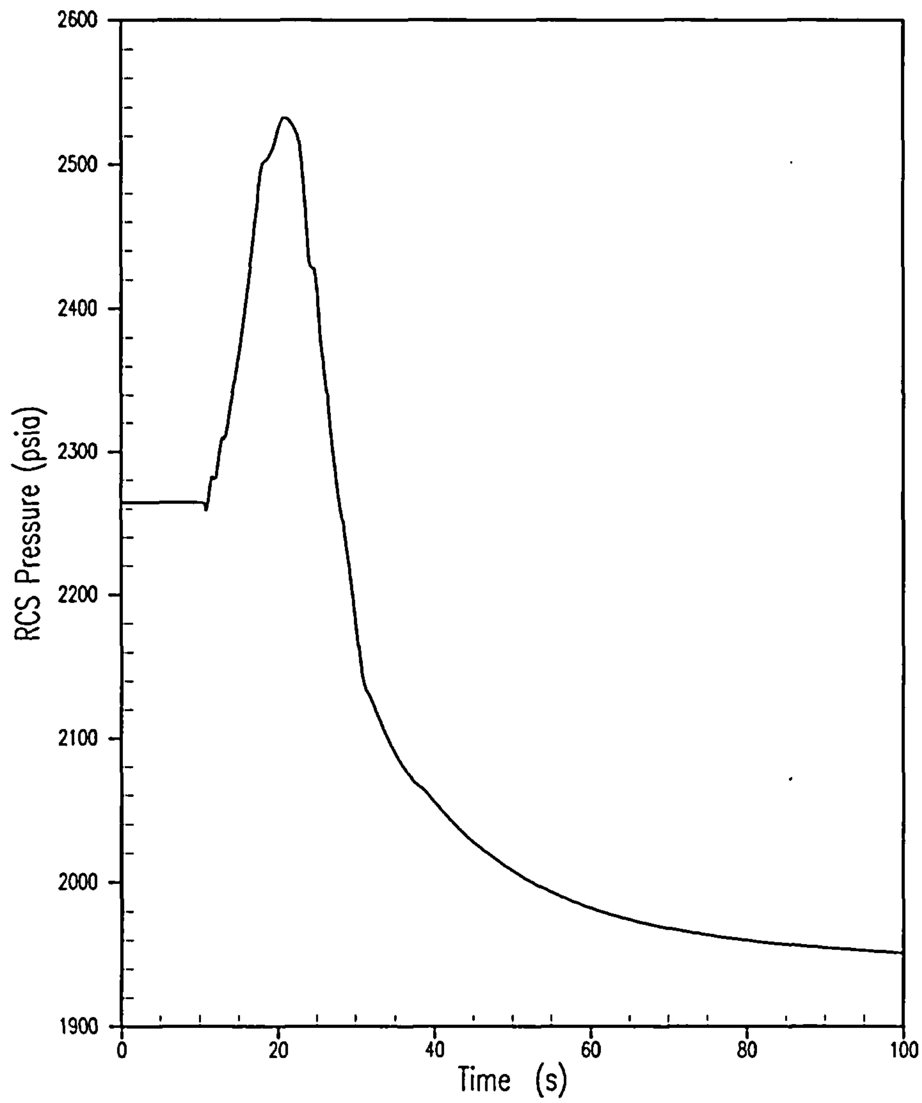

Figure 5.1.10-7 Loss of Condenser Vacuum (DNB Case) RCS Pressure (RCP Outlet Pressure)

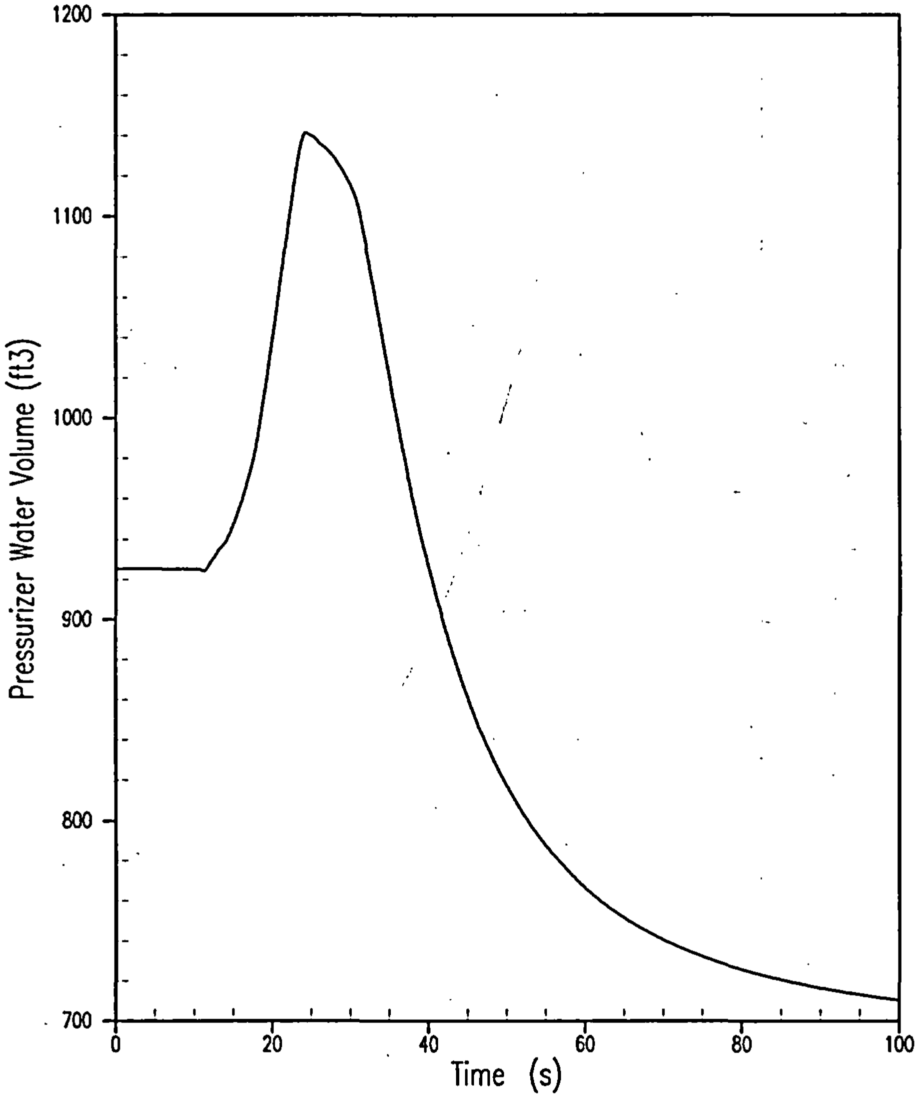

Figure 5.1.10-8 Loss of Condenser Vacuum (DNB Case) Pressurizer Water Volume

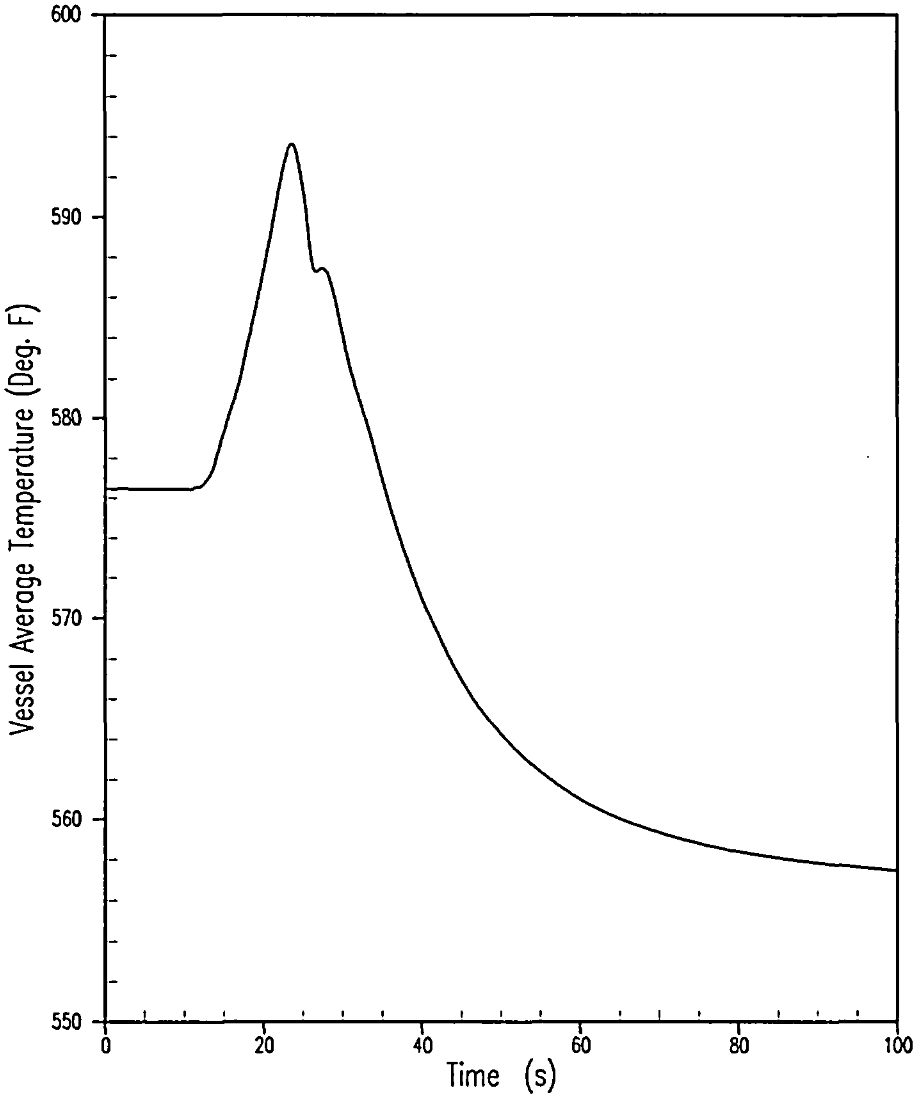

Figure 5.1.10-9 Loss of Condenser Vacuum (DNB Case) Vessel Average Temperature

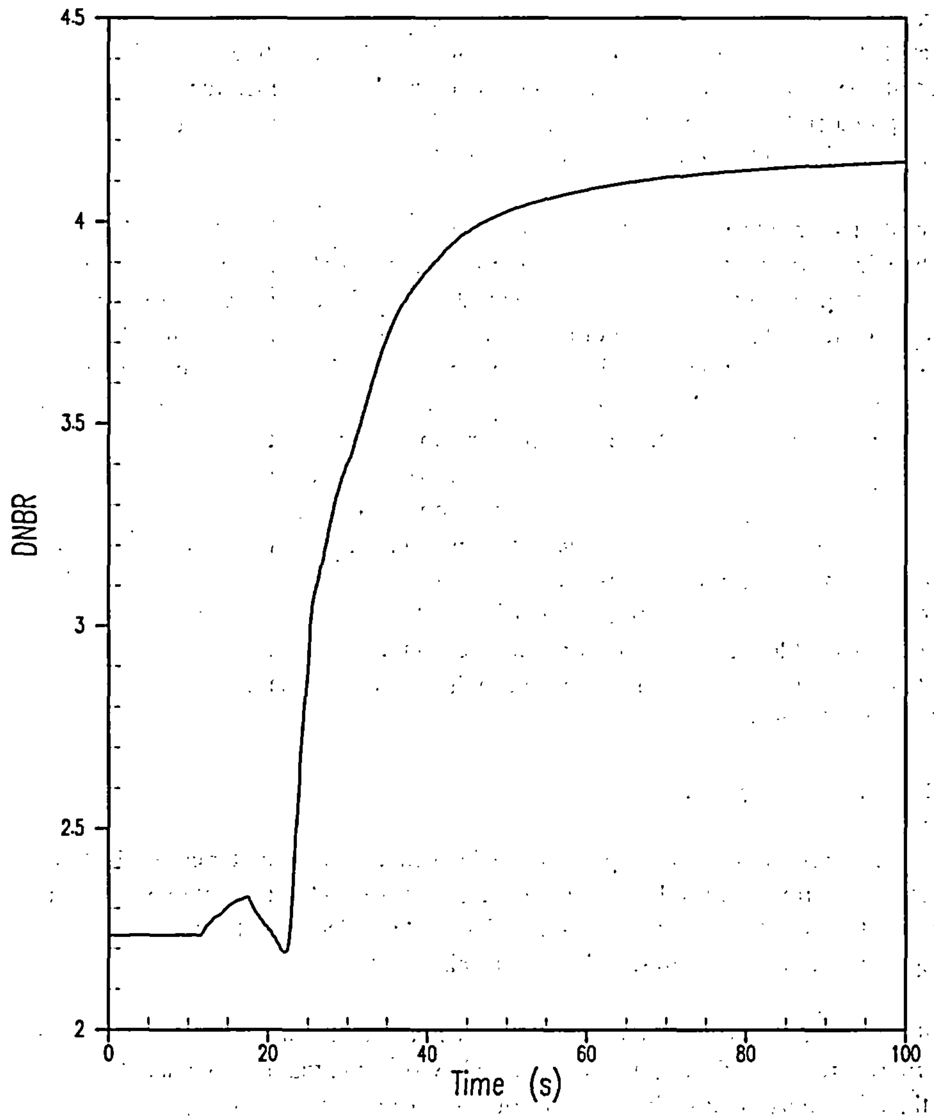

Figure 5.1.10-10 Loss of Condenser Vacuum (DNB Case) DNBR

5.1.11 Transients Resulting from the Malfunction of One Steam Generator

5.1.11.1 Accident Description

The Asymmetric Steam Generator Transient (ASGT) event is defined as a complete loss of steam load to one steam generator from a full-power condition. This transient is modeled as an inadvertent closure of the main steamline isolation valve to one steam generator. A concurrent termination of feedwater flow to the affected steam generator is assumed in the analysis to conservatively bound any potential response of the feedwater system. Feedwater isolation to the affected steam generator will result in an increase the vessel inlet temperature asymmetry during the transient, which is conservative with respect to demonstrating that the DNB design basis is satisfied.

In the event of a large loss of load to a single steam generator, the MSSVs may lift and the reactor may be tripped by a high steam generator differential pressure (HSGDP) reactor trip. This trip function is specifically designed to provide protection against an ASGT. Upon the loss of load to a single steam generator, the affected steam generator pressure increases to the opening setpoint (including tolerances) of the MSSVs. Once relief flow is established through the MSSVs, the pressure begins to decrease in the affected steam generator and settles to a value corresponding to the MSSV setpoint pressure. The unaffected steam generator continues to supply steam to the turbine and attempts to replace the steam load previously supplied by the affected steam generator because the turbine demand is assumed to be maintained at 100%. The increase in steam flow from the unaffected steam generator results in an overcooling of the cold legs associated with the unaffected loop. Additionally, the steam pressure in the unaffected steam generator decreases due to the increased steam flow in that loop. The increase in the core inlet temperature from the affected loops in combination with the decrease in core inlet temperature from the unaffected loops results in a large core temperature asymmetry. The asymmetric core temperature distributions result in an increase in the radial and axial peaking in the core, resulting in a challenge to the DNB design basis.

5.1.11.2 Method of Analysis

The ASGT event is analyzed by employing the detailed digital computer code RETRAN. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The analysis of the ASGT has been performed to demonstrate that the DNB design basis and the peak rod power criteria are satisfied. This is accomplished by showing that the calculated minimum DNBR is greater than the safety analysis limit DNBR of 1.42. The overall analysis process is described as follows.

The analysis of the ASGT event, as determined by the RETRAN code, calculates transient values of key plant parameters identified as statepoints (core average heat flux, core pressure, core inlet temperature, and RCS flow rate). The core radial and axial peaking factors are determined using the thermal-hydraulic conditions from the transient analysis as input to the nuclear core models. The detailed thermal and hydraulic computer code VIPRE was then used to calculate the DNBR response for the transient based on the core radial and axial peaking factors, and based on the limiting statepoints for the event.

The following assumptions were made in the analysis of the ASGT:

1. The initial reactor power and RCS temperature are assumed to be at their nominal values, the initial RCS flow rate is assumed at a value consistent with the minimum measured flow rate and the initial RCS pressure is assumed at a value consistent with minimum value allowed by the plant technical specifications. Uncertainties in initial conditions are statistically included in the calculation of the DNBR limit as described in the Revised Thermal Design Procedures.

2. Two cases were analyzed; one assuming 0% of the steam generator U-tubes to be plugged and one assuming 30% of the steam generator U-tubes to be plugged. These cases will cover any asymmetry withing these limits.

3. A bounding range of fuel to coolant heat transfer characteristics was evaluated to assure that the limiting statepoints are generated by the RETRAN code.

4. The initiating event is an inadvertent closure of a single MSIV with an assumed simultaneous termination of feedwater flow to the same steam generator.

5. The ASGT event results in a loss of steam flow and associated main feedwater flow to the affected steam generator. This causes a heatup in the associated primary RCS loop. The turbine demand is assumed to be maintained at 100% demand by the unaffected loop. This causes a cooldown to occur in the primary loop associated with the unaffected steam generator. The reactivity feedback is weighted to the unaffected loop since end-of-life reactivity feedback is assumed, which results in an increase in the core power due to the colder RCS temperature conditions.

6. The model assumes reactivity feedback coefficients that maximize the increase in nuclear power prior to reactor trip. These reactivity coefficients were weighted to the RCS loop associated with the unaffected steam generator to maximize the power increase. The affects associated with the asymmetric vessel inlet distribution caused by the transient were used to calculate conservative radial and axial peaking factors.

7. The cases are analyzed with the automatic pressurizer pressure control system assumed to be operable. Thus, full credit is taken for the effect of the pressurizer spray in limiting any primary coolant pressure increase above the initial pressure.

5.1.11.3 Results

The results of the statepoint evaluation demonstrate that the ASGT event meets the applicable DNB and the peak rod power (kW/ft) acceptance criteria. Table 5.1.11-1 summarizes the sequence of events and limiting conditions for the 0% steam generator tube plugging case and Table 5.1.11-2 correspondingly summarizes the results of the 30% steam generator tube plugging case.

The transient response for the ASGT from full-power operation are shown in Figures 5.1.11-1 through 5.1.11-12 assuming 0% SGTP, maximum reactivity feedback conditions with automatic pressurizer pressure control (pressurizer spray). Figures 5.1.11-13 through 5.1.11-24 are based on 30% SGTP, maximum reactivity feedback conditions, and automatic pressurizer pressure control (pressurizer spray).

The overall transient response for the ASGT evaluated herein provides a similar system response as reported in the current analysis.

5.1.11.4 Conclusions

The ASGT event was conservatively analyzed with respect to the reactor core response. Key analysis assumptions were made to conservatively maximize the asymmetry in vessel inlet temperatures, so as to maximize the core power and peaking factors.

Two cases were performed to assess the both minimum DNBR and peak rod power (kW/ft) for 0% SGTP and for 30% SGTP. The case with 30% of the steam generator tubes plugged was found to the most limiting case. Both the DNB design basis and the peak rod power limit are met for both cases analyzed.

Table 5.1.11-1 Sequence of Events and Transient Results Asymmetric Steam Generator Transient 0% Steam Generator Tube Plugging	
Event	**Time (seconds)**
Main Steam Isolation (Loop Two)	10.1
Manual Feedwater Termination (Loop Two)	10.1
Reactor Trip on HSGDP	15.4
Rod Motion Begins	16.1
Time of Minimum DNBR	17.6
Minimum DNBR Value	1.78
DNBR Limit	1.42

Table 5.1.11-2 Sequence of Events and Transient Results Asymmetric Steam Generator Transient 30% Steam Generator Tube Plugging	
Event	**Time (seconds)**
Main Steam Isolation (Loop Two)	10.1
Manual Feedwater Termination (Loop Two)	10.1
Reactor Trip on HSGDP	15.5
Rod Motion Begins	16.3
Time of Minimum DNBR	17.4
Minimum DNBR Value	1.77
DNBR Limit	1.42

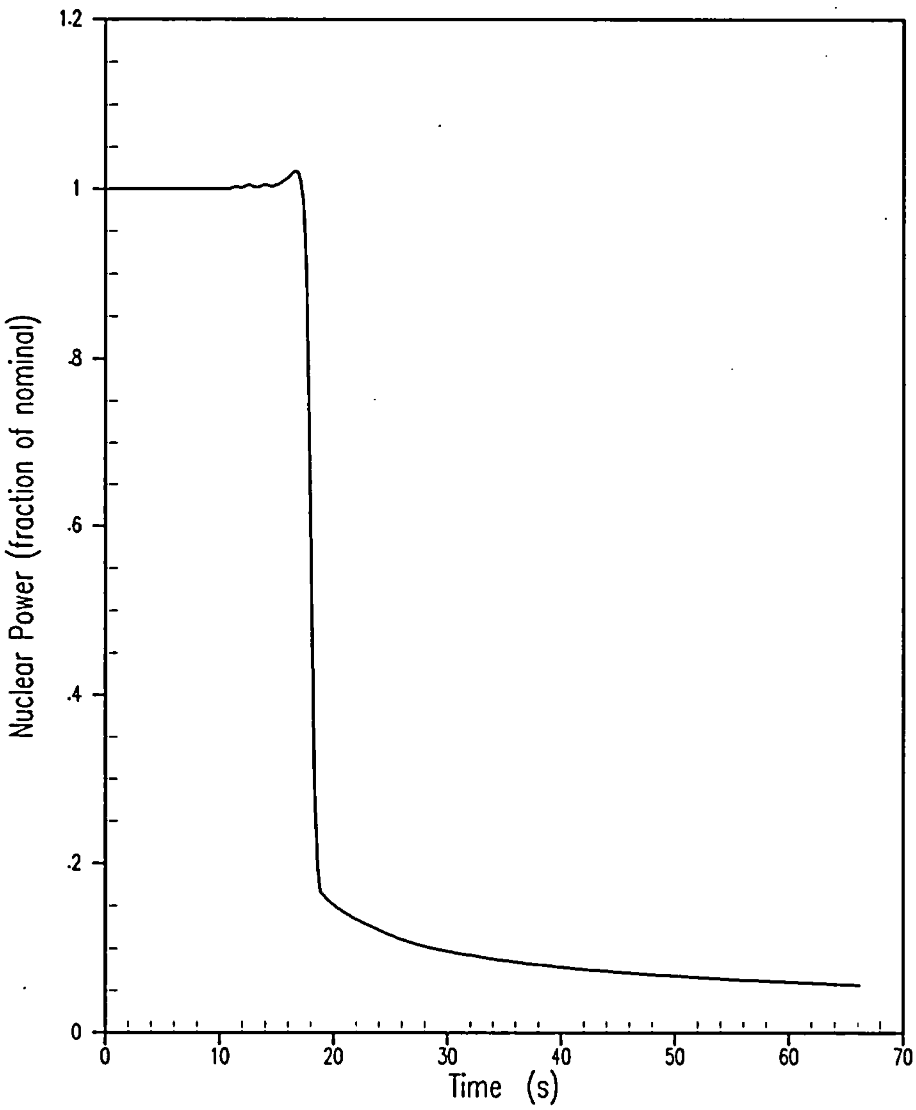

Figure 5.1.11-1 Nuclear Power for Asymmetric Steam Generator Transient 0% Tube Plugging

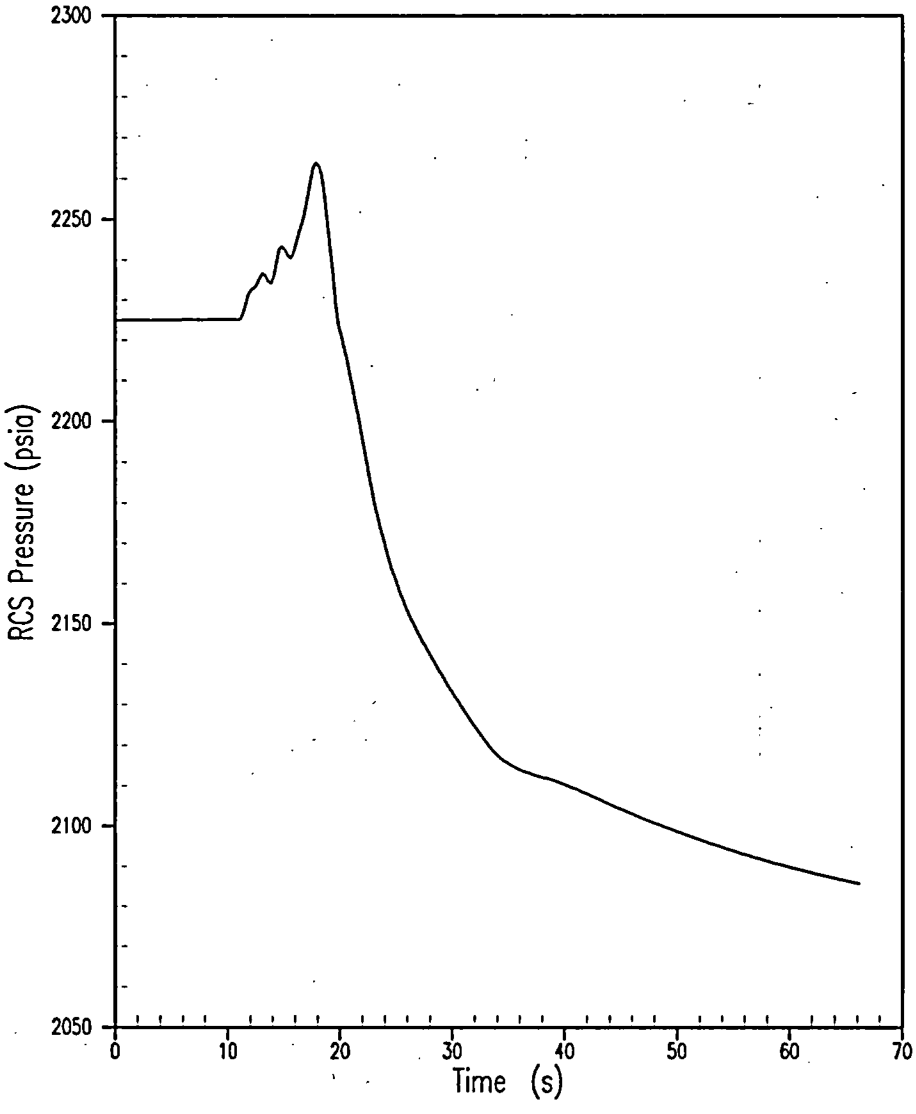

Figure 5.1.11-2 RCS Pressure for Asymmetric Steam Generator Transient 0% Tube Plugging

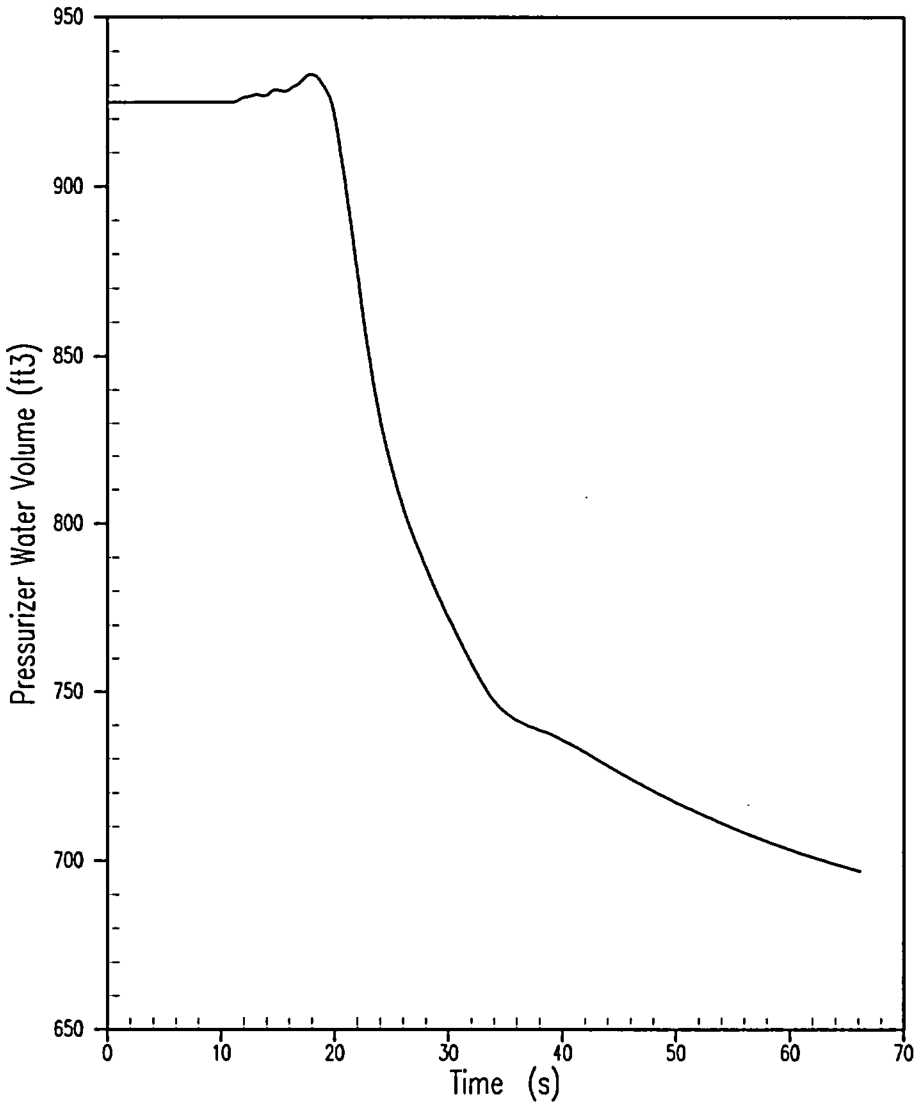

Figure 5.1.11-3 Pressurizer Water Volume for Asymmetric Steam Generator Transient 0% Tube Plugging

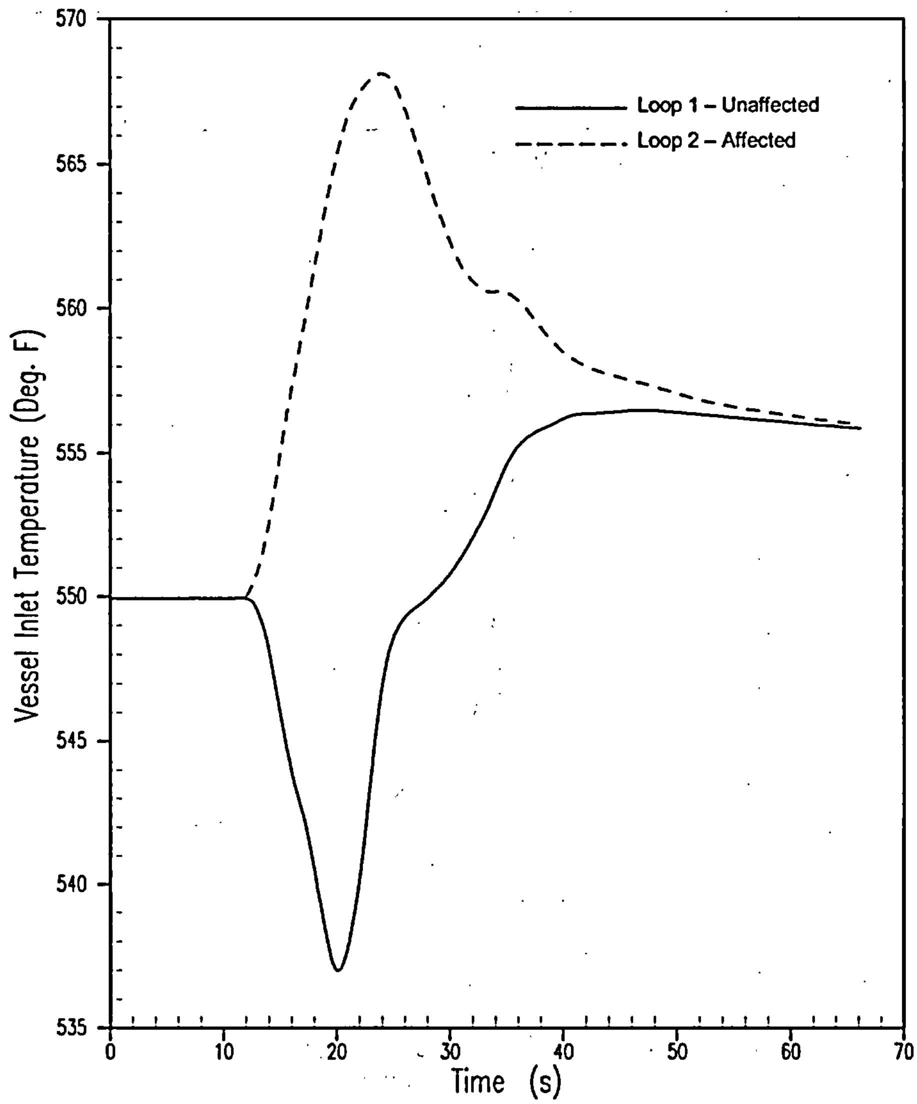

Figure 5.1.11-4 Vessel Inlet Temperature for Asymmetric Steam Generator Transient 0% Tube Plugging

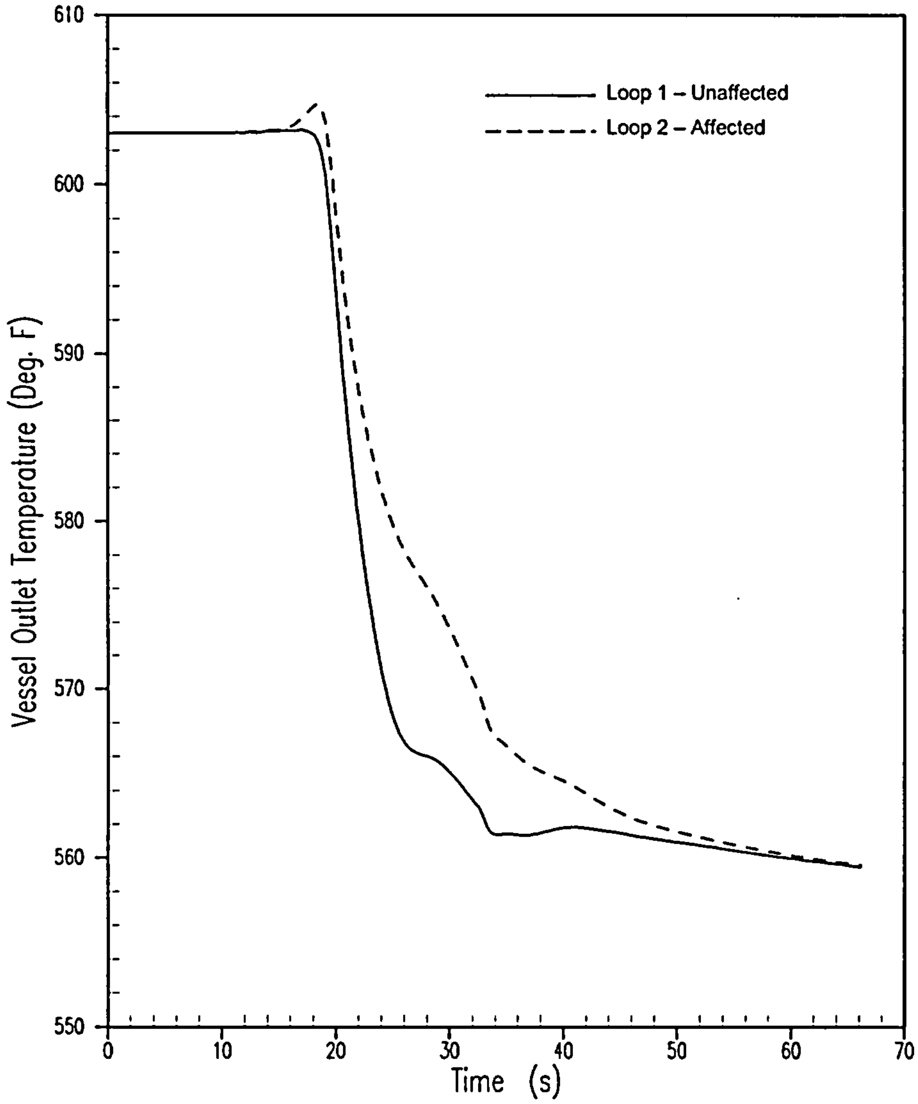

Figure 5.1.11-5 Vessel Outlet Temperature for Asymmetric Steam Generator Transient 0% Tube Plugging

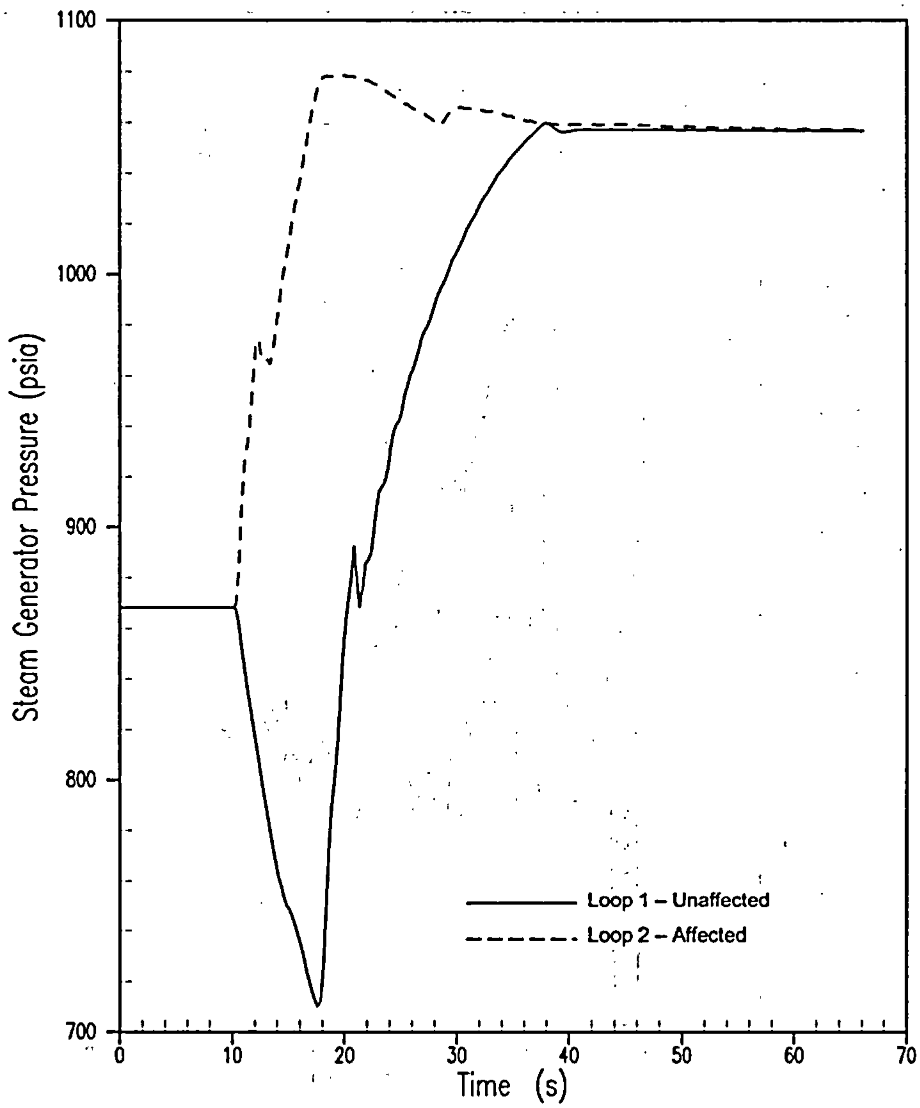

Figure 5.1.11-6 Steam Generator Pressure for Asymmetric Steam Generator Transient 0% Tube Plugging

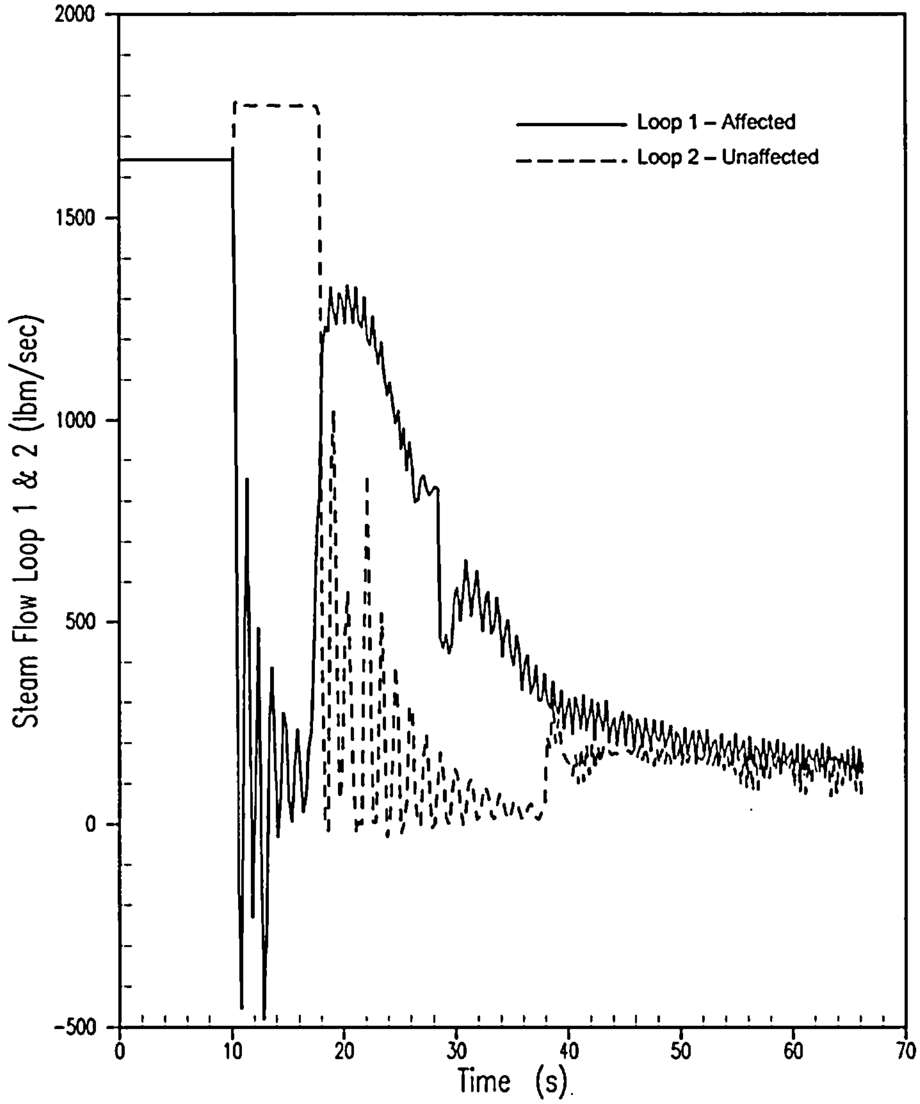

Figure 5.1.11-7 Steam Flow for Asymmetric Steam Generator Transient 0% Tube Plugging

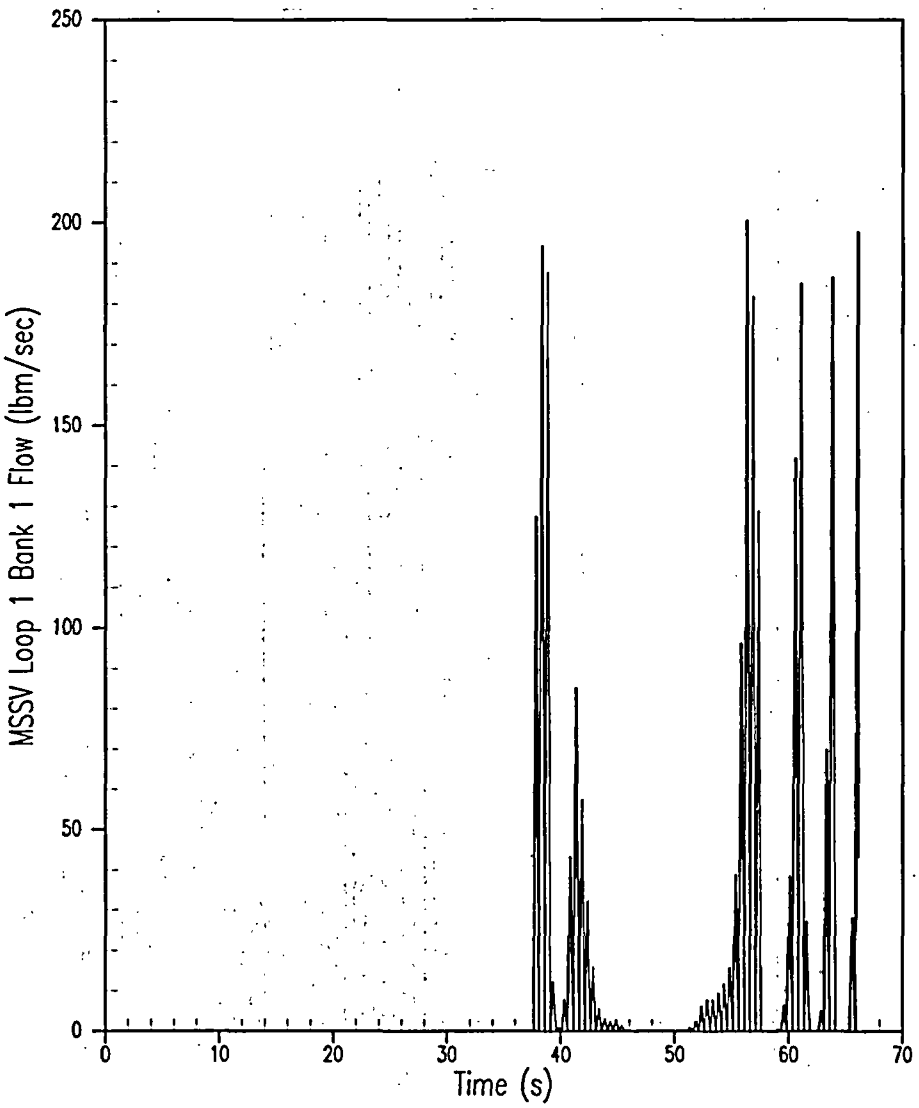

Figure 5.1.11-8 Main Steam Safety Valve Flow (Loop 1 – Bank 1 Unaffected Steam Generator) for Asymmetric Steam Generator Transient 0% Tube Plugging

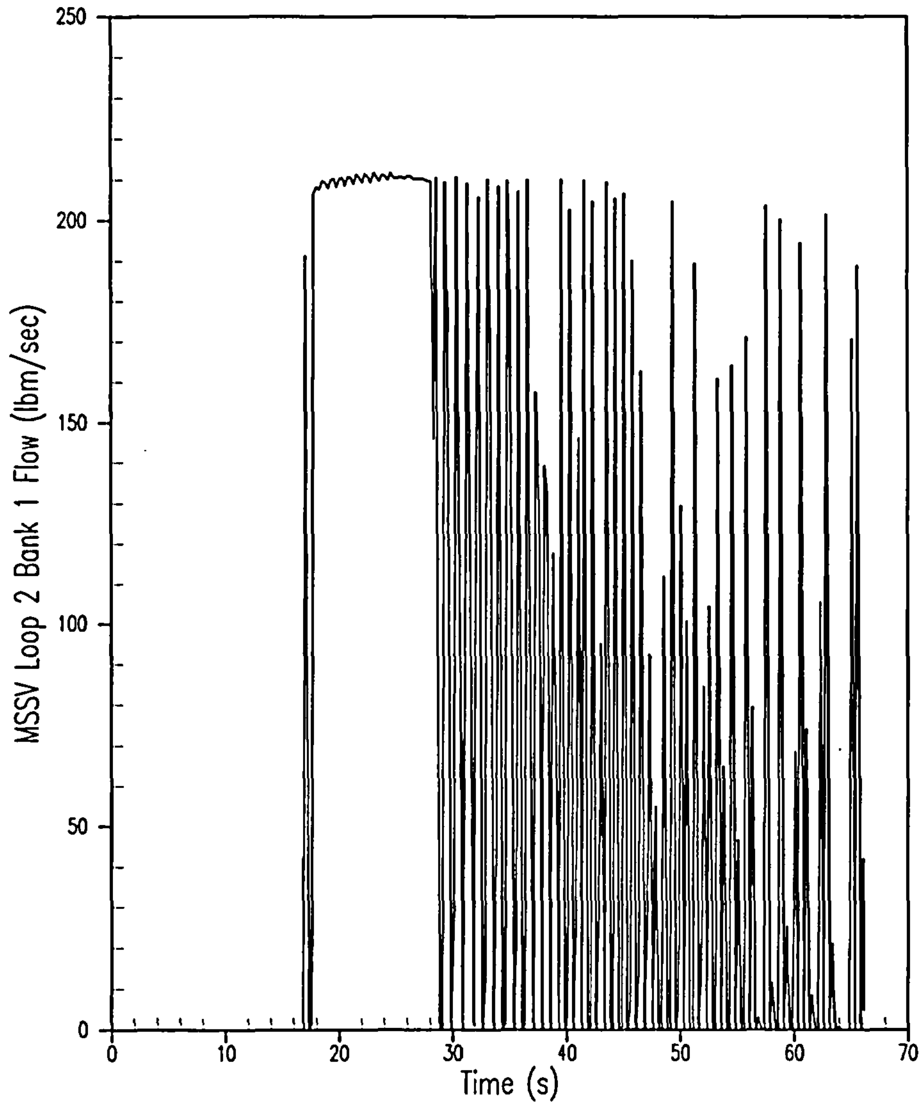

Figure 5.1.11-9 Main Steam Safety Valve Flow (Loop 2 – Bank 1 Affected Steam Generator) for Asymmetric Steam Generator Transient 0% Tube Plugging

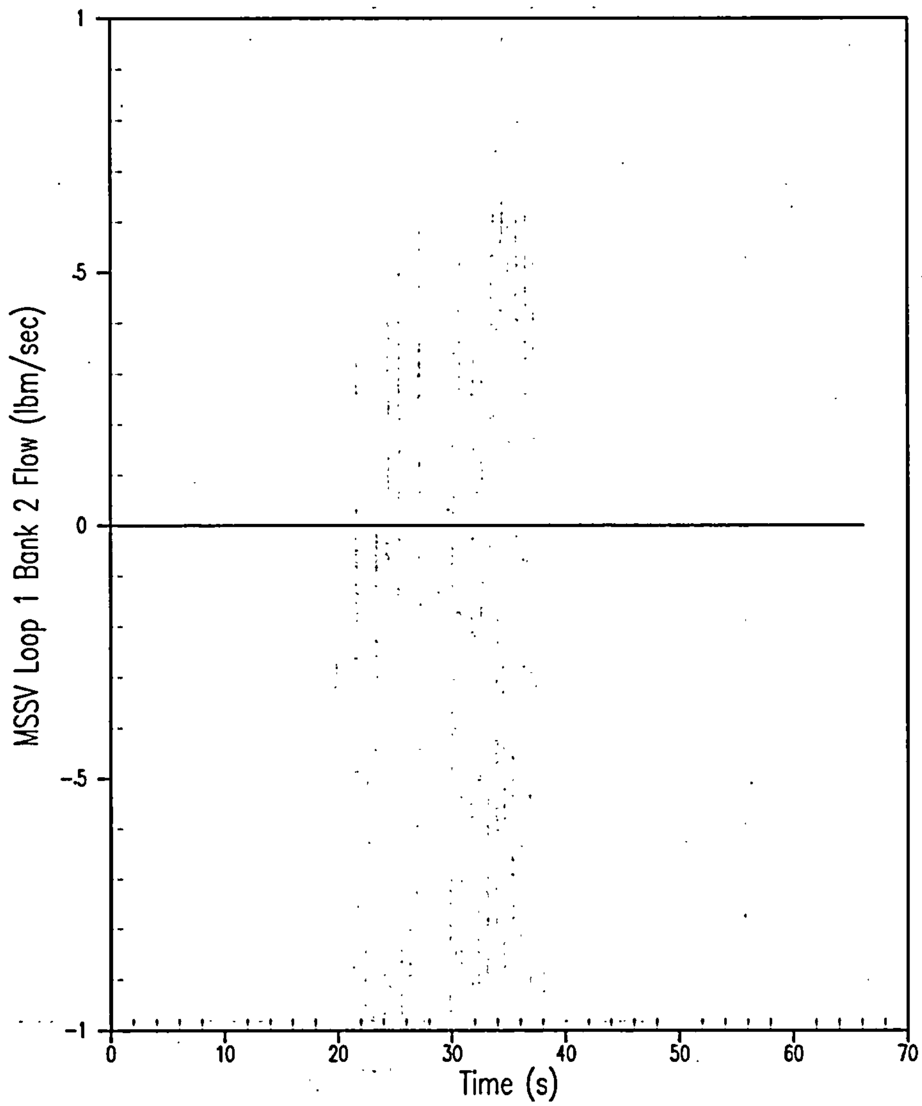

Figure 5.1.11-10 Main Steam Safety Valve Flow (Loop 1 – Bank 2 Unaffected Steam Generator) for Asymmetric Steam Generator Transient 0% Tube Plugging

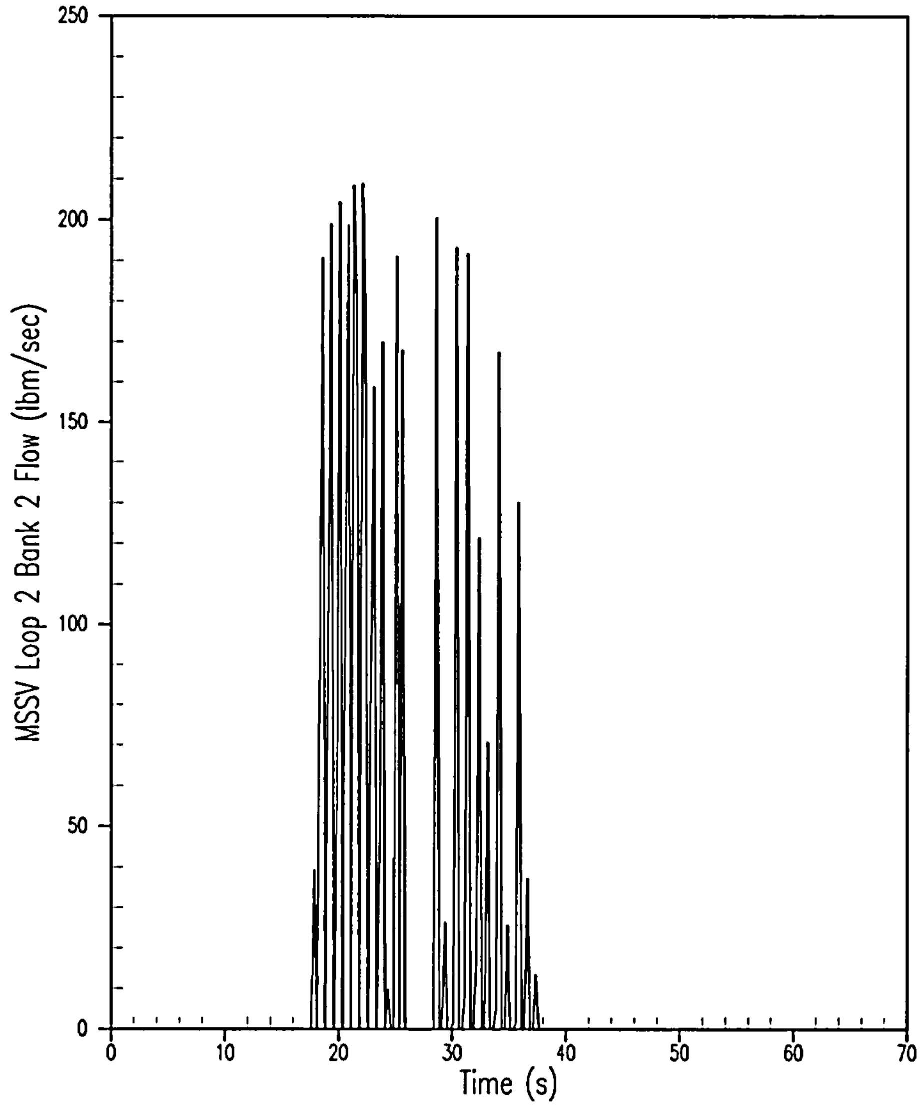

Figure 5.1.11-11 Main Steam Safety Valve Flow (Loop 2 – Bank 2 Affected Steam Generator) for Asymmetric Steam Generator Transient 0% Tube Plugging

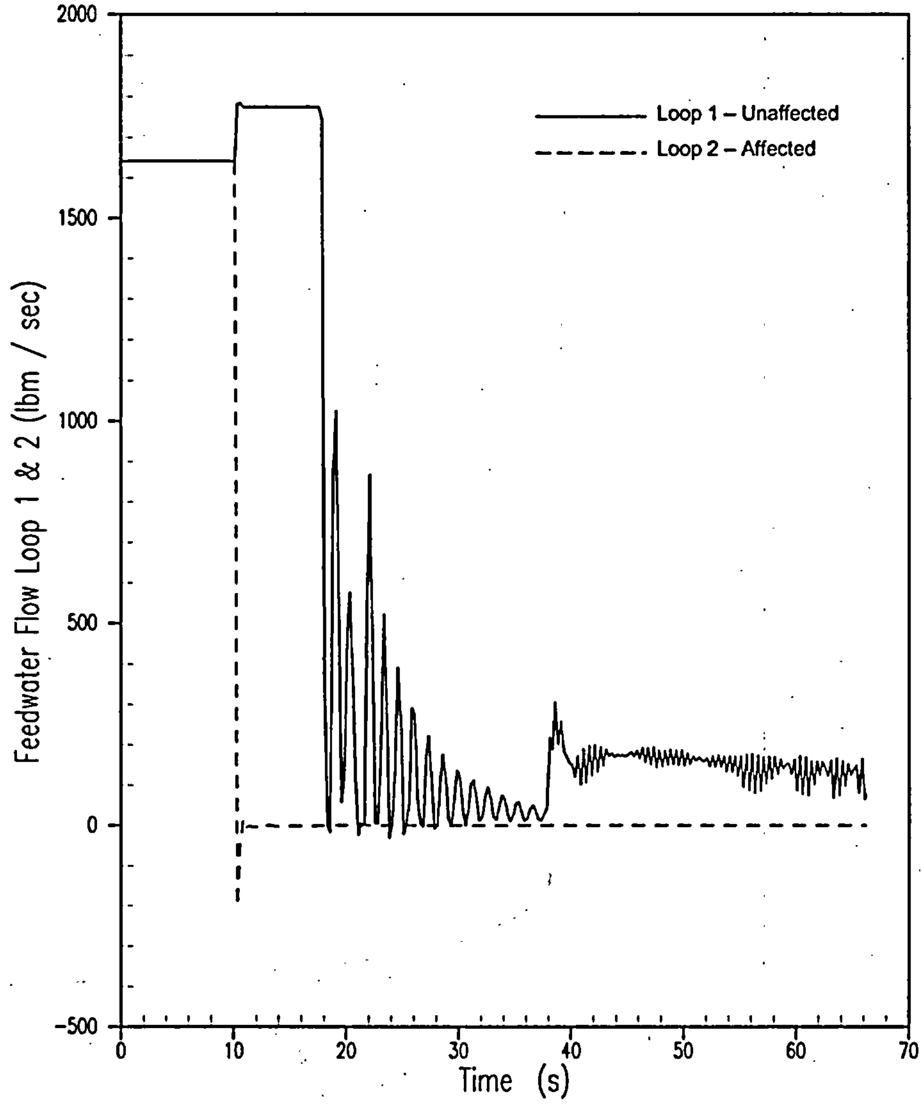

Figure 5.1.11-12 Feedwater Flow for Asymmetric Steam Generator Transient 0% Tube Plugging

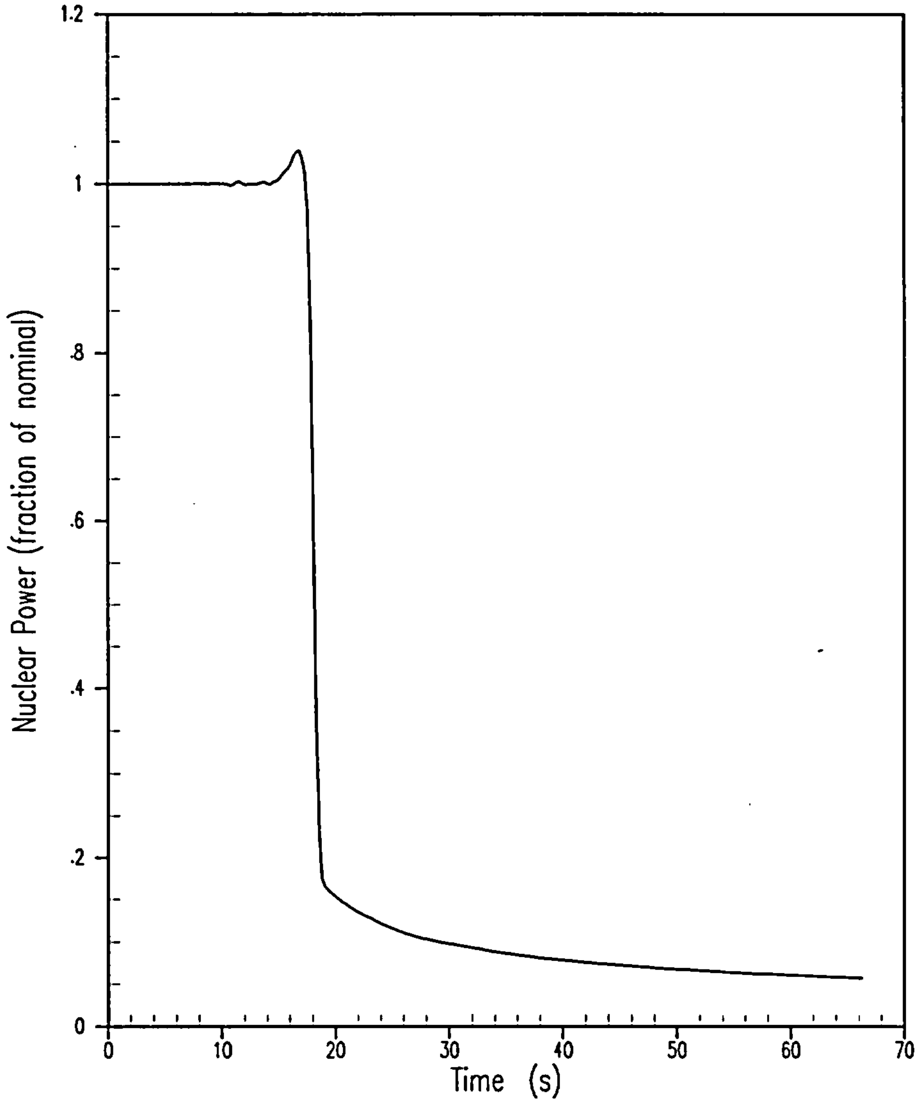

Figure 5.1.11-13 Nuclear Power for Asymmetric Steam Generator Transient 30% Tube Plugging

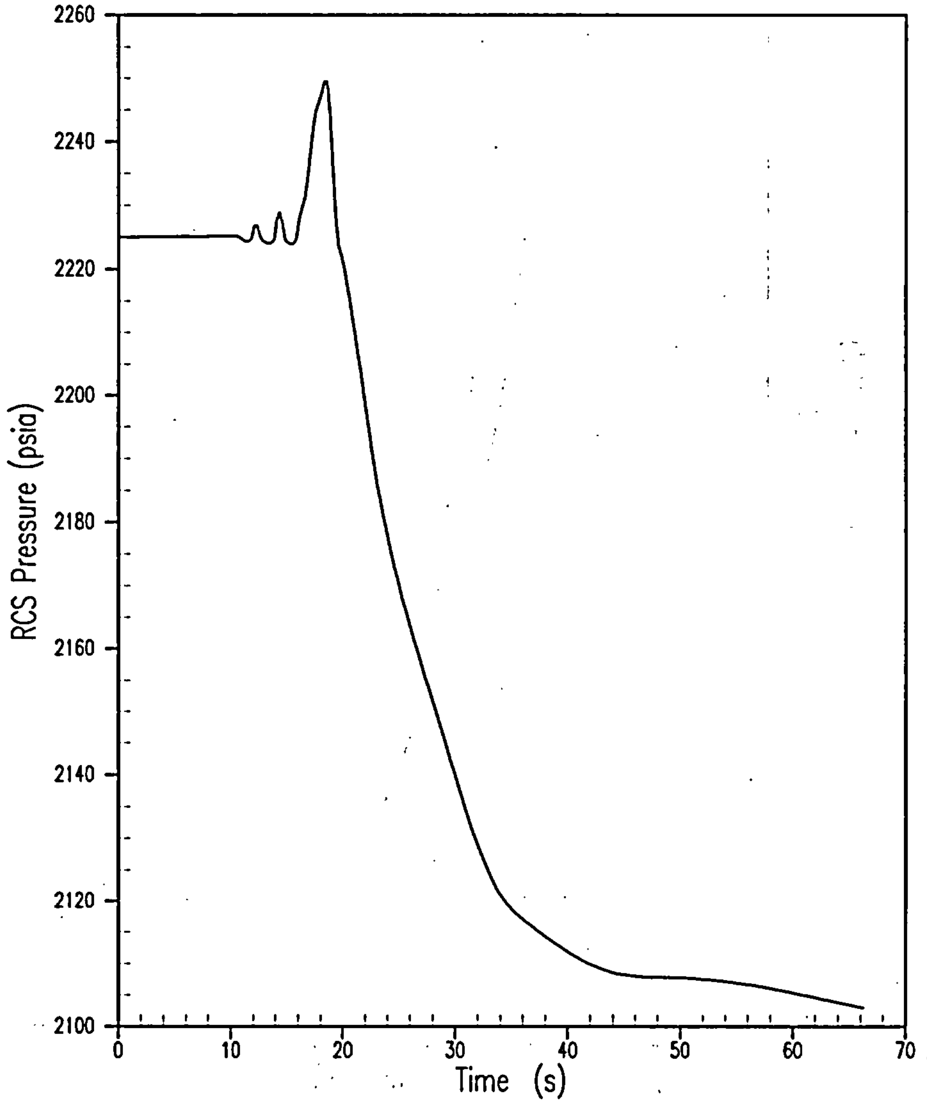

Figure 5.1.11-14 RCS Pressure for Asymmetric Steam Generator Transient 30% Tube Plugging

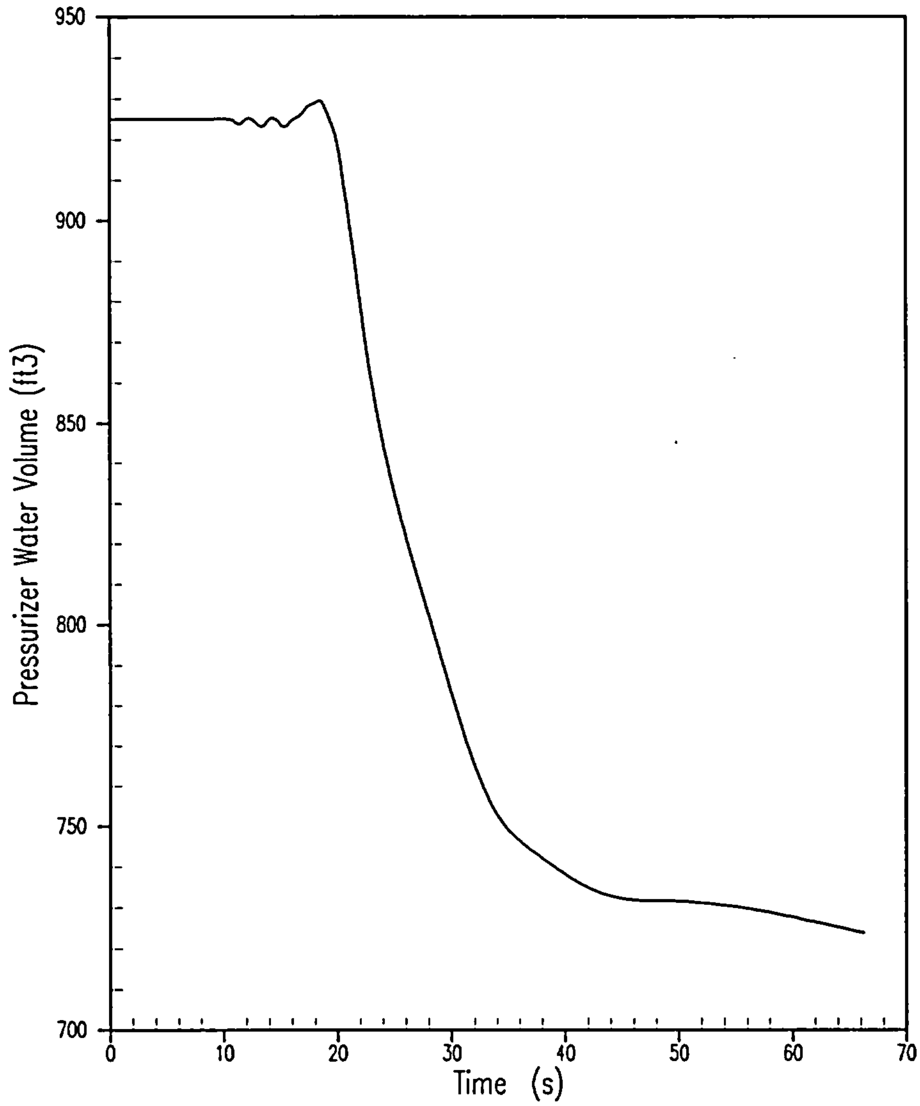

Figure 5.1.11-15 Pressurizer Water Volume for Asymmetric Steam Generator Transient 30% Tube Plugging

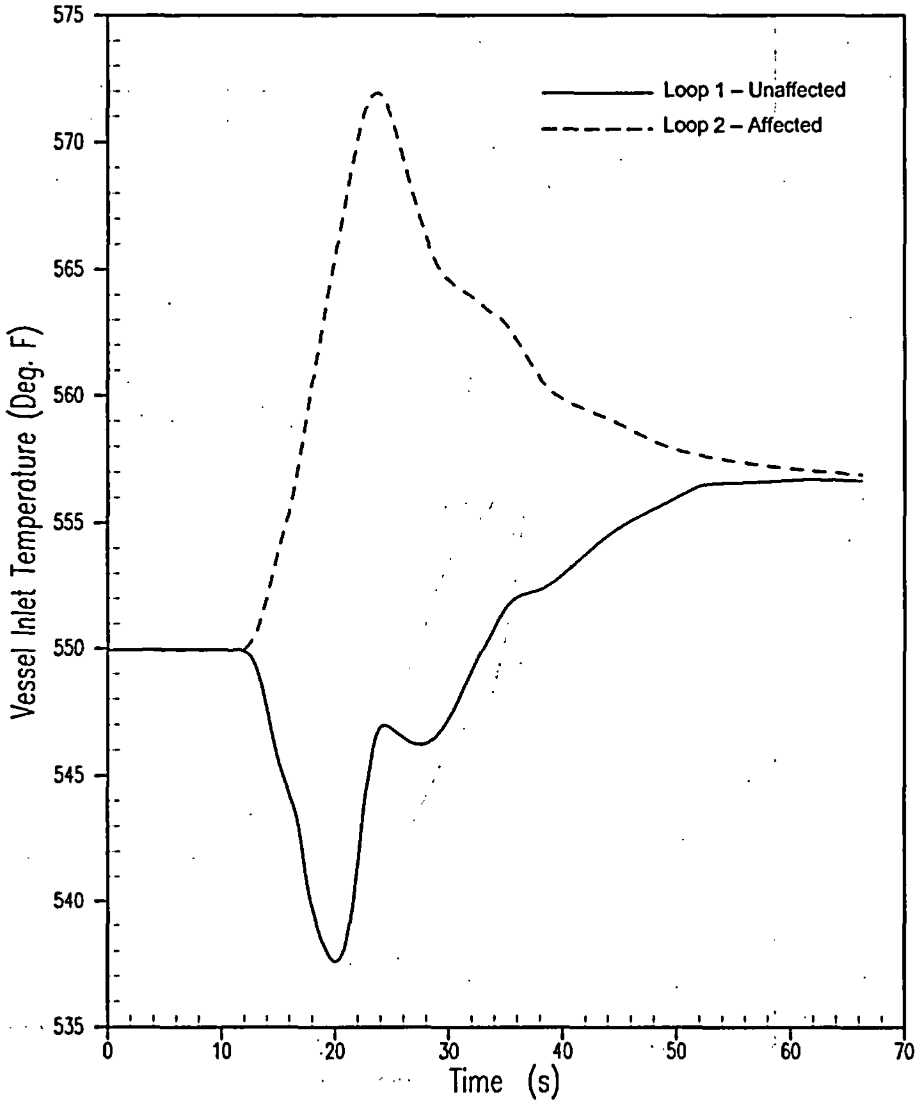

Figure 5.1.11-16 Vessel Inlet Temperature for Asymmetric Steam Generator Transient 30% Tube Plugging

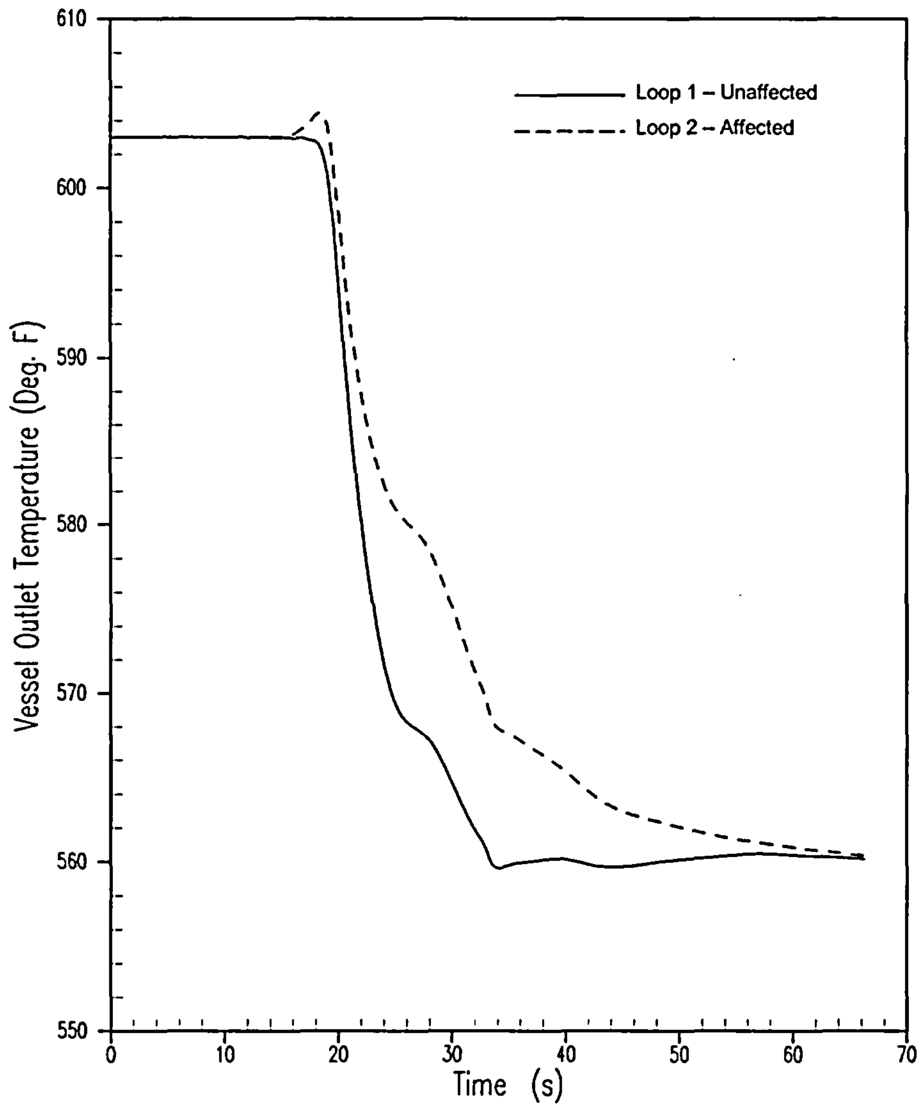

Figure 5.1.11-17 Vessel Outlet Temperature for Asymmetric Steam Generator Transient 30% Tube Plugging

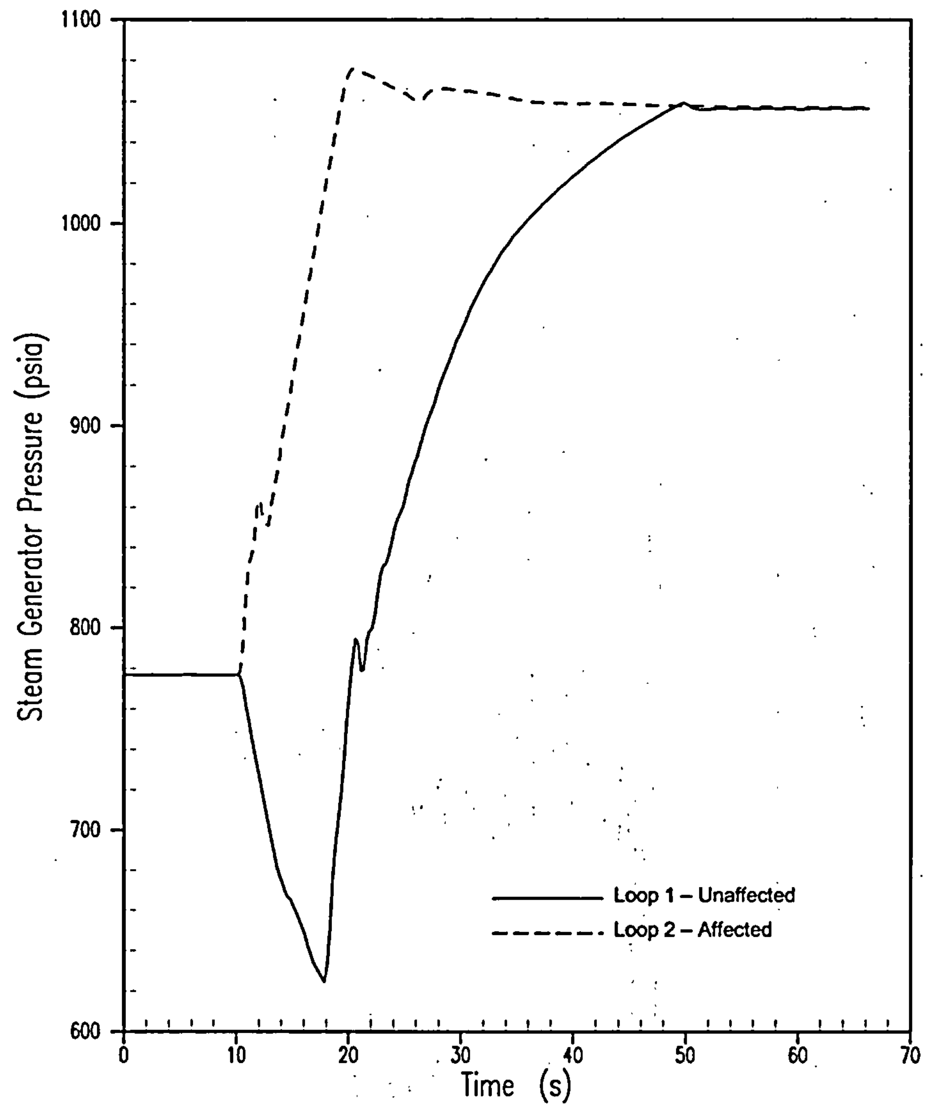

Figure 5.1.11-18 Steam Generator Pressure for Asymmetric Steam Generator Transient 30% Tube Plugging

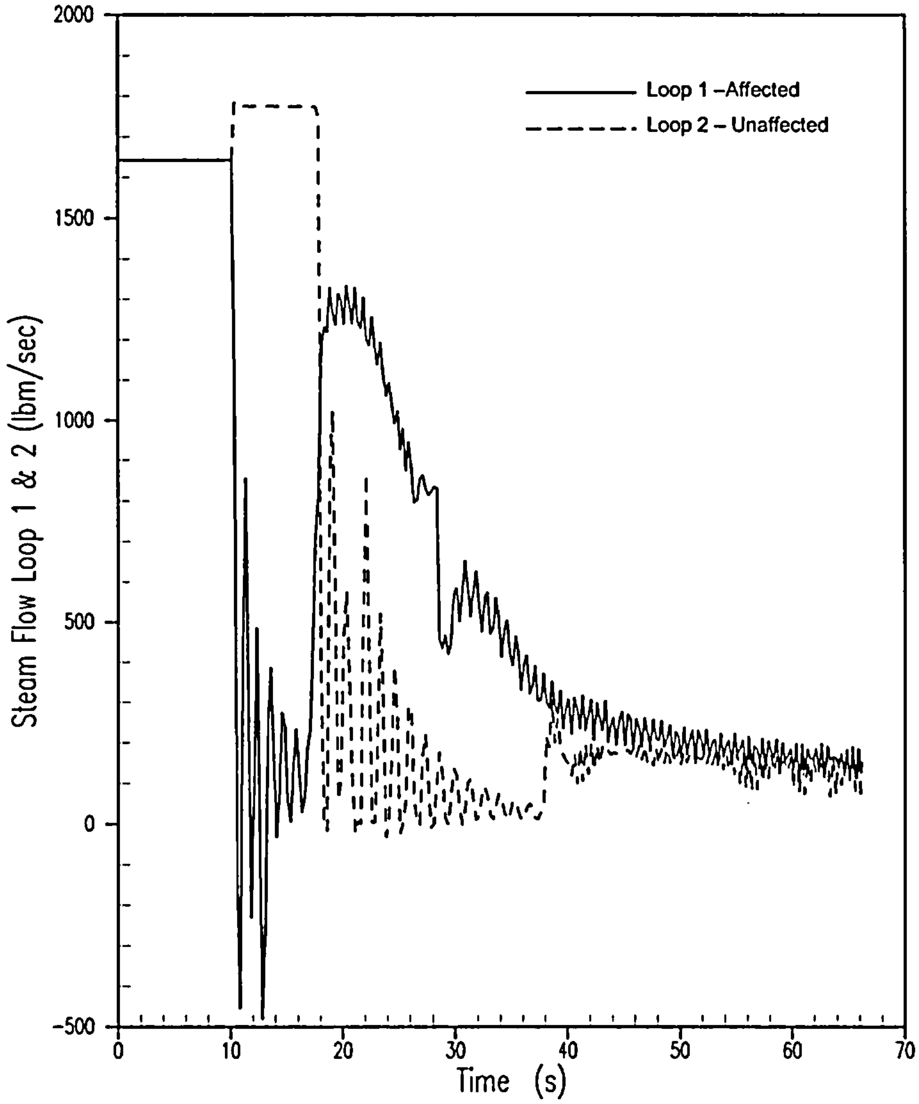

Figure 5.1.11-19 Steam Flow for Asymmetric Steam Generator Transient 30% Tube Plugging

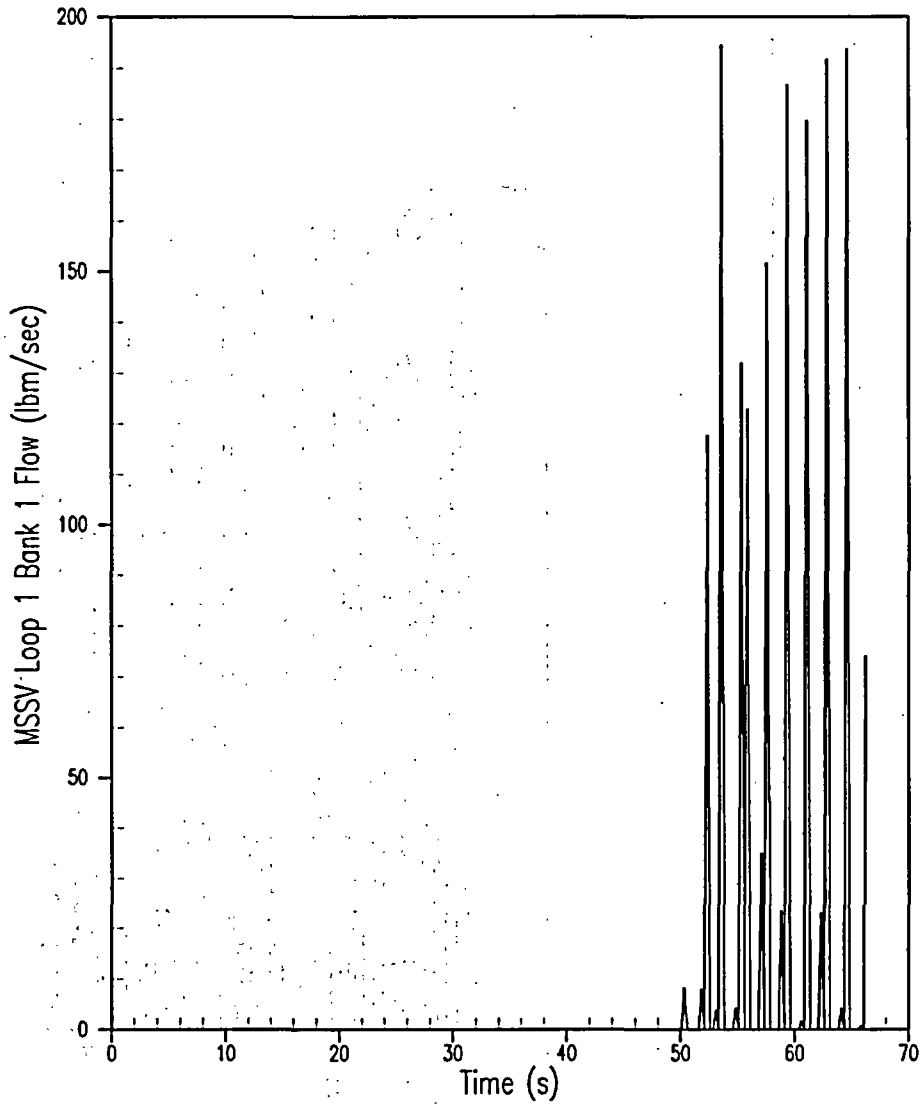

Figure 5.1.11-20 Main Steam Safety Valve Flow (Loop 1 – Bank 1 Unaffected Steam Generator) for Asymmetric Steam Generator Transient 30% Tube Plugging

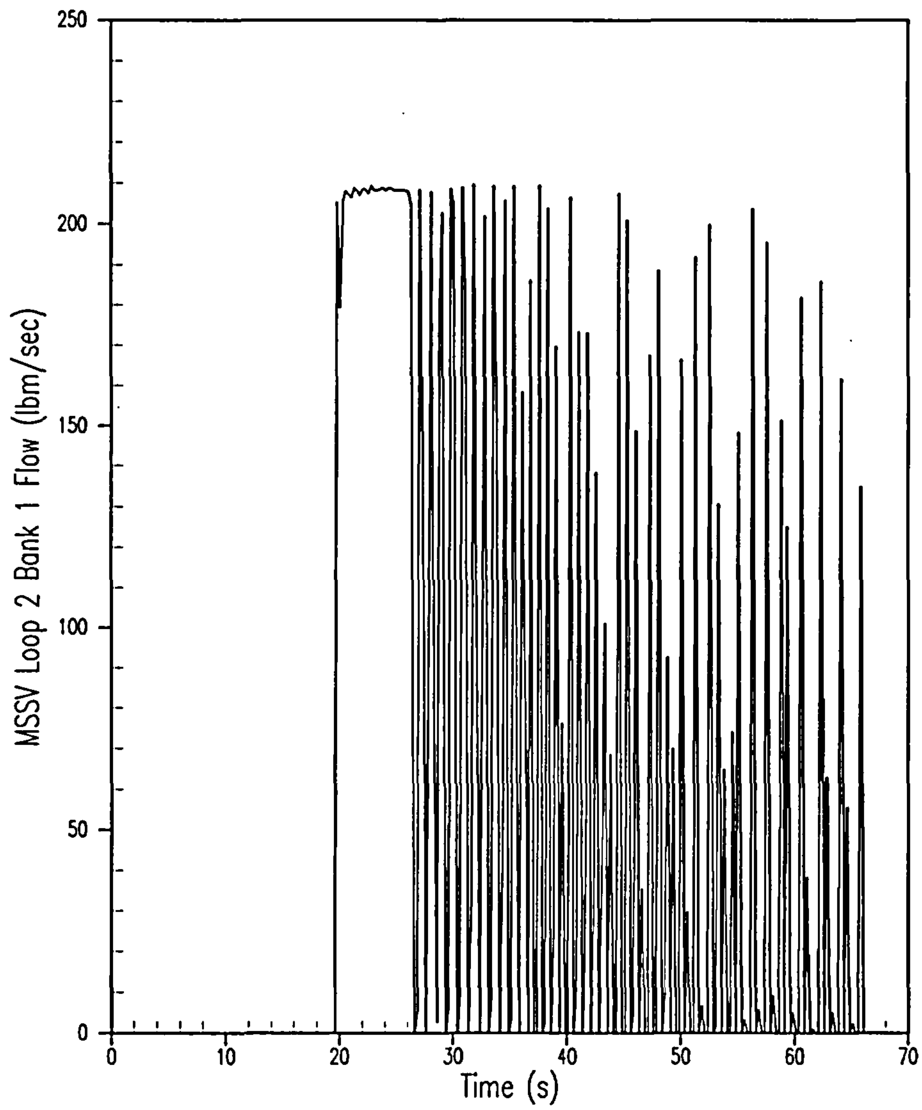

Figure 5.1.11-21 Main Steam Safety Valve Flow (Loop 2 – Bank 1 Affected Steam Generator) for Asymmetric Steam Generator Transient 30% Tube Plugging

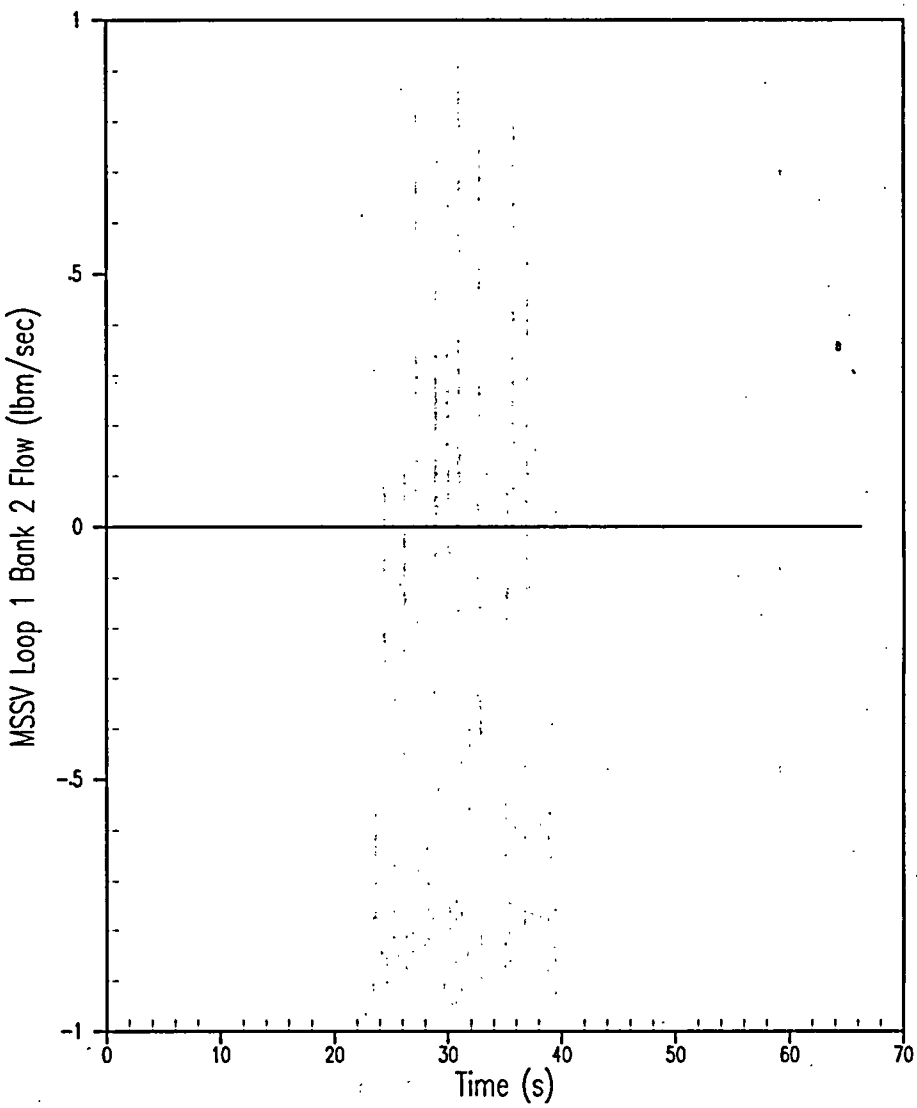

Figure 5.1.11-22 Main Steam Safety Valve Flow (Loop 1 – Bank 2 Unaffected Steam Generator) for Asymmetric Steam Generator Transient 30% Tube Plugging

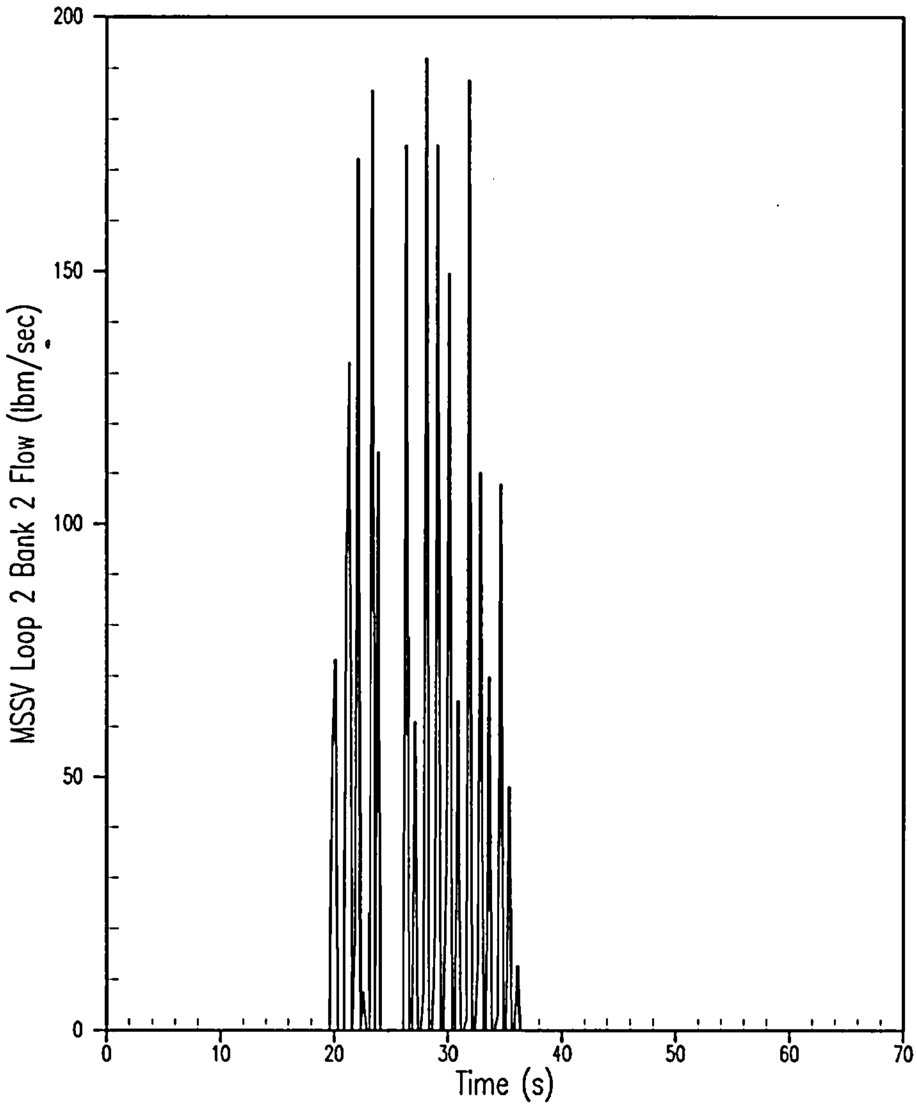

Figure 5.1.11-23 Main Steam Safety Valve Flow (Loop 2 – Bank 2 Affected Steam Generator) for Asymmetric Steam Generator Transient 30% Tube Plugging

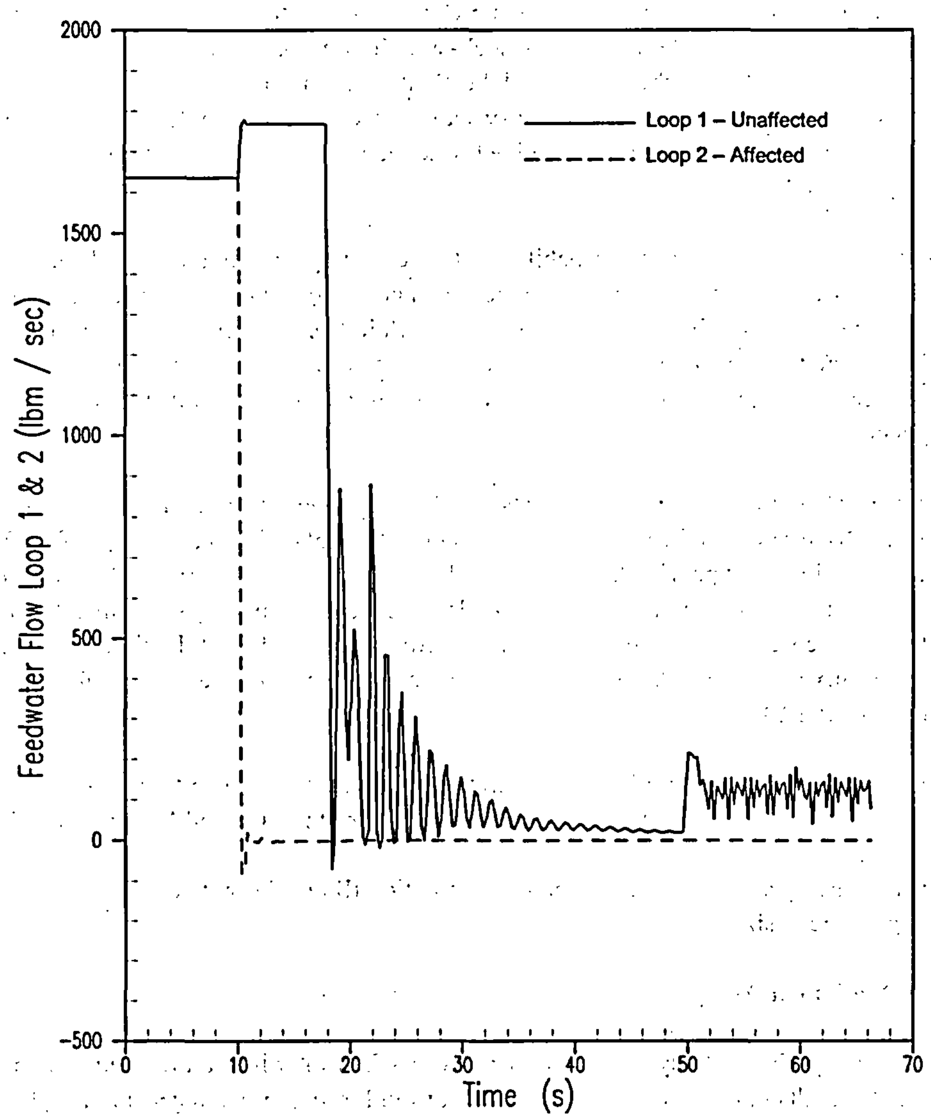

Figure 5.1.11-24 Feedwater Flow for Asymmetric Steam Generator Transient 30% Tube Plugging

5.1.12 Feedwater Line Break

5.1.12.1 Accident Description

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to maintain shell-side fluid inventory in the steam generators. Depending upon the size and location of the rupture and the plant operating conditions, the event can cause either a cooldown or a heatup of the reactor coolant system. Since the RCS cooldown resulting from a secondary system pipe break is covered by the steamline break event, only the RCS heatup aspects are emphasized for the case of feedwater line break.

A feedwater line break reduces the capability of the secondary system to remove heat generated by the core from the RCS. The feedwater flow to the steam generators is reduced or terminated, resulting in a decrease in the shell-side fluid inventory. Moreover, fluid from the faulted steam generator can be expelled through the broken pipe, thereby eliminating the capability of the steam generator to remove heat from the RCS. A broken feedwater line may also prevent the addition of main feedwater to the intact steam generator.

The feedwater line break is one of the events which defines the required minimum capacity of the auxiliary feedwater system for removing core residual heat following reactor trip. If sufficient heat removal capability is not provided, core residual heat following reactor trip could raise the RCS coolant temperature to the extent that the resulting fuel damage would compromise the maintenance of a coolable geometry of the core, and result in potential radioactive releases. For St. Lucie Unit 2, the analysis used to justify the auxiliary feedwater requirements for a postulated feedwater line break is presented in UFSAR Chapter 10.4.9A.

A feedwater line break during full-power operation may also cause a short-term pressure increase in both the RCS and main steam system challenging the integrity of the RCS and MSS pressure boundaries.

A feedwater line break is classified as an ANS Condition III or IV event, an infrequent or limiting fault, depending on break size.

5.1.12.2 Method of Analysis

The feedwater line break analysis assumes a break in a feedwater line at the steam generator inlet nozzle. Such a break results in an uncontrolled discharge of fluid from the steam generator. A break upstream of the feedwater line check valve would affect the RCS only as a loss of normal feedwater.

This accident is analyzed: (1) to confirm that the pressurizer safety valves (PSVs) and MSSVs are adequately sized to prevent overpressurization of the primary RCS and MSS, respectively; and (2) to ensure that the DNB design basis is satisfied. Chapter 10.4.9A of the UFSAR demonstrates the adequacy of the auxiliary feedwater system in removing long-term decay heat.

The feedwater line break transient is analyzed by employing the detailed digital computer code RETRAN (References 1 and 2). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and MSSVs. The code computes pertinent plant variables including temperatures, pressures, and power level.

The event is analyzed to conservatively meet Condition II acceptance criteria. Three separate cases are analyzed, one ensures that the peak primary RCS pressure remains below the design limit (2750 psia), one confirms that the peak MSS pressure remains below 110% of the steam generator shell design pressure (1100 psia) and the final case is performed to address DNB concerns. The major assumptions for these cases are summarized as follows.

In order to give conservative results in calculating the maximum RCS and MSS pressures during the transient, the following assumptions are made:

1. The initial reactor power is assumed to be at its maximum value plus uncertainty, the initial RCS flow rate is assumed at a value consistent with the thermal design flow rate and the initial RCS pressure is assumed at a value consistent with minimum value allowed by the plant technical specifications minus the pressure measurement uncertainty.

2. For maximum RCS pressure, the RCS temperature is assumed to be at Low-Tavg conditions minus uncertainty. For maximum MSS pressure, the RCS temperature is assumed to be at High-Tavg conditions plus uncertainty.

3. For maximum RCS pressure, the initial steam generator tube plugging level is assumed to be at the maximum plugging level. For maximum MSS pressure, the initial steam generator tube plugging level is assumed to be at the minimum plugging level.

4. The initial steam generator water level is assumed to be at the minimum water level, consistent with the low-level alarm setpoint minus the steam generator level measurement uncertainty.

5. The High Pressurizer Pressure and Low Steam Pressure reactor trip setpoints for adverse conditions are assumed. The Low-Low Steam Generator Level reactor trip is not credited.

6. The feedline break is assumed to occur at the physical inlet nozzle location on the steam generator.

7. An fL/D of 0 (zero) is assumed for the break and the blowdown quality is calculated by the RETRAN code.

8. A break size spectrum is analyzed to determine the limiting size with respect to RCS and MSS overpressurization.

9. Minimum reactivity feedback is assumed to maximize the energy input to the primary coolant.

10. No credit is taken for the effect of the pressurizer spray in reducing or limiting primary coolant pressure. Pressurizer Safety Valves are available, but are modeled assuming a +3% setpoint tolerance. Finally, the PORV is not considered since it would actuate after reactor trip on High Pressurizer Pressure.

The initial conditions are summarized in Table 5.1.0-2.

In order to give conservative results in calculating the minimum DNBR during the transient, the following assumptions are made:

1. The initial reactor power and RCS temperature are assumed to be at their nominal values, the initial RCS flow rate is assumed at a value consistent with the minimum measured flow rate and the initial RCS pressure is assumed at a value consistent with the lowest nominal value allowed by the plant technical specifications. Uncertainties in initial conditions are included in determining the DNBR limit value consistent with the use of RTDP (Reference 3).

2. The initial steam generator tube plugging level is assumed to be at the maximum plugging level. The initial steam generator water level is assumed to be at the minimum water level, consistent with the low-level alarm setpoint minus the steam generator level measurement uncertainty.

3. The High Pressurizer Pressure and Low Steam Pressure reactor trip setpoints for adverse conditions are assumed. The Low-Low Steam Generator Level reactor trip is not credited.

4. The feedline break is assumed to occur at the physical inlet nozzle location on the steam generator.

5. An fL/D of 0 (zero) is assumed for the break and the blowdown quality is calculated by the RETRAN code.

6. A break size spectrum is analyzed to determine the limiting size with respect to minimum DNBR.

7. Minimum reactivity feedback is assumed to maximize the energy input to the primary coolant.

8. Credit is taken for the effect of the pressurizer spray in reducing primary coolant pressure and delaying reactor trip on High Pressurizer Pressure. Pressurizer safety valves are also available and are modeled assuming a -3% setpoint tolerance. The PORV is assumed to actuate once reaching the High Pressurizer Pressure reactor trip setpoint.

The initial conditions are summarized in Table 5.1.0-2.

The feedline break methodology also considers the possibility of a Loss-of-Offsite-Power (LOOP) event. For this analysis, the LOOP is assumed to occur 3 seconds after turbine trip following reactor trip, however, assuming a loss of offsite power does not adversely impact the RCS or MSS overpressurization results. For the RCS pressure cases, peak pressure occurs immediately after reactor trip on High Pressurizer Pressure. By the time the reactor coolant pumps (RCPs) begin coastdown, the limiting point in the transient has already occurred. For MSS pressure cases, losing RCPs retards heat transfer to the intact steam generator, leading to a lower peak secondary side pressure. For DNBR, the results of the Complete Loss of Flow analysis are bounding, since the conditions of this event prior to reactor trip are more limiting.

5.1.12.3 Results

The Feedwater Line Break event was analyzed assuming the plant to be initially operating at full power at beginning of cycle (BOC) (minimum feedback reactivity coefficients) with no credit taken for the pressurizer spray to determine the primary RCS pressure response. Further, the low-low steam generator level reactor trip function was not credited. The break spectrum from 0.25 ft^2 to 0.375 ft^2 was analyzed to assure that the maximum RCS pressure case would be captured. Figures 5.1.12-1 through 5.1.12-7 show the transient results for the limiting break case, 0.28 ft^2. In this case, the PSVs are actuated and maintain the primary RCS pressure below 110% of the design value. Table 5.1.12-1 summarizes the results of the break spectrum analysis and Table 5.1.12-4 provides the sequence of events and limiting conditions for the 0.28 ft^2 case.

Table 5.1.12-2 summarizes the break spectrum results for the Feedwater Line Break event at BOC (minimum feedback reactivity coefficients) assuming 0% SGTP to determine the secondary MSS pressure response. Further, the low-low steam generator level reactor trip function was not credited. The break spectrum was analyzed from 0.005 ft^2 to 0.375 ft^2 to assure that the maximum MSS pressure case would be captured. The limiting break size was found to be 0.05 ft^2. The MSS pressure increases, resulting in opening the first five MSSVs, then decreases rapidly following reactor trip. The MSSVs actuate to limit the MSS pressure below 110% of the steam generator shell design pressure. Table 5.1.12-5 provides the sequence of events and limiting conditions for the 0.05 ft^2 case, and Figures 5.1.12-8 through 5.1.12-14 show the transient results. (Note: Due to the small break size, the MSS pressure and break flow response for the 0.05 ft^2 case is much different from those presented for the limiting RCS Overpressurization and DNB cases.)

The Feedwater Line Break DNB case is analyzed at BOC (minimum feedback reactivity coefficients) assuming full credit for the pressurizer spray to calculate the transient DNBR response. Further, the low-low steam generator level reactor trip function was not credited. The break spectrum was analyzed from 0.20 ft^2 to 0.375 ft^2 to assure that the limiting DNBR case would be captured. The limiting break size was found to be 0.25 ft^2. The minimum DNBR remains well above the safety analysis limit value. Table 5.1.12-3 summarizes the break spectrum results, which demonstrates this conclusion.
Table 5.1.12-6 summarizes the sequence of events and limiting conditions for the limiting 0.25 ft^2 case. Figures 5.1.12-15 through 5.1.10-22 show the transient responses for the 0.25 ft^2 case.

5.1.12.4 Conclusion

The results of the analyses show that the plant design is such that a feedwater line break presents no hazard to the integrity of the primary RCS or MSS by meeting all applicable Condition II acceptance criteria. Pressure relieving devices that have been incorporated into the plant design are adequate to limit the maximum pressures to within the safety analysis limits, i.e., 2750 psia for the primary RCS and 1100 psia for the MSS. The integrity of the core is maintained by operation of the RPS, i.e., the minimum DNBR is maintained above the safety analysis limit value of 1.42. Thus, no core safety limit will be violated as a result of implementing up to 30% steam generator tube plugging or transitioning to the WCAP-9272 methodology.

5.1.12.5 References

1. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses."

2. EPRI NP-1850-CCM, "Validation and Verification of the MTR-PC Thermohydraulic Package."

3. WCAP-11397, "Revised Thermal Design Procedure," April 1989.

Table 5.1.12-1 Feedwater Line Break RCS Overpressurization Case Results	
Break Size (ft^2)	Max RCS Pressure (psia)
0.375	2640
0.35	2651
0.325	2670
0.30	2723
0.29	2736
0.28	2739
0.27	2733
0.25	2706
110% of Design Pressure Limit	2750

Table 5.1.12-2 Feedwater Line Break MSS Overpressurization Case Results	
Break Size (ft^2)	Max MSS Pressure (psia)
0.375	991
0.300	1039
0.250	1063
0.200	1070
0.150	1079
0.100	1085
0.050	1090
0.010	1089
0.005	1089
110% of Design Pressure Limit	1100

Table 5.1.12-3 Feedwater Line Break DNBR Case Results	
Break Size (ft^2)	**Min DNBR**
0.375	1.74
0.30	1.64
0.25	**1.58**
0.20	1.64
MDNBR Limit	1.42

Table 5.1.12-4 Sequence of Events and Transient Results Feedwater Line Break Limiting Break Size = 0.28 ft^2	
Without Pressurizer Pressure Control (for Primary RCS Overpressure)	
Event	**Time (seconds)**
Initiation of Event	0.01
Manual Feedwater Isolation (both loops)	0.01
Reactor Trip on Low Steam Pressure	30.8
Rod Motion Begins	31.5
Time of Peak RCS Pressure	33.2
Peak RCS Pressure	2739 psia
RCS Pressure Limit	2750 psia

Table 5.1.12-5 Sequence of Events and Transient Results Feedwater Line Break Limiting Break Size = 0.05 ft^2	
Without Pressurizer Pressure Control (for Main Steam System Overpressure)	
Event	**Time (seconds)**
Initiation of Event	0.01
Manual Feedwater Isolation (both loops)	0.01
Reactor Trip on High Pressurizer Pressure	37.1
Rod Motion Begins	37.8
Time of Peak MSS Pressure	41.2
Peak MSS Pressure	1090 psia
MSS Pressure Limit	1100 psia

Table 5.1.12-6 Sequence of Events and Transient Results Feedwater Line Break Break Size = 0.05 ft^2	
With Pressurizer Pressure Control (for Minimum DNB))	
Event	**Time (seconds)**
Initiation of Event	0.01
Manual Feedwater Isolation (both loops)	0.01
Reactor Trip on Low Steam Pressure	40.8
Rod Motion Begins	41.5
Time of Minimum DNBR	60.9
Minimum DNBR Value	1.58
DNBR Limit	1.42

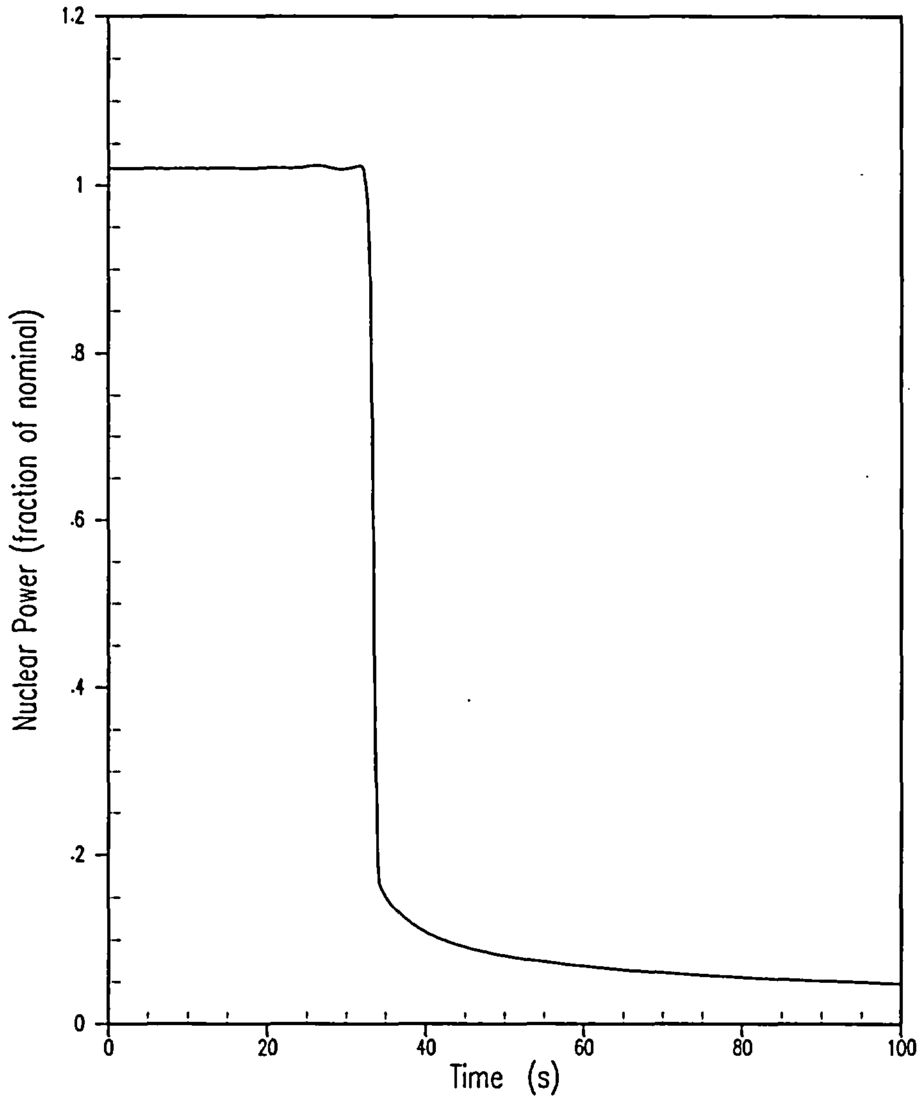

Figure 5.1.12-1 Feedwater Line Break RCS Overpressure Case Limiting Break Size = 0.28 ft^2 Nuclear Power

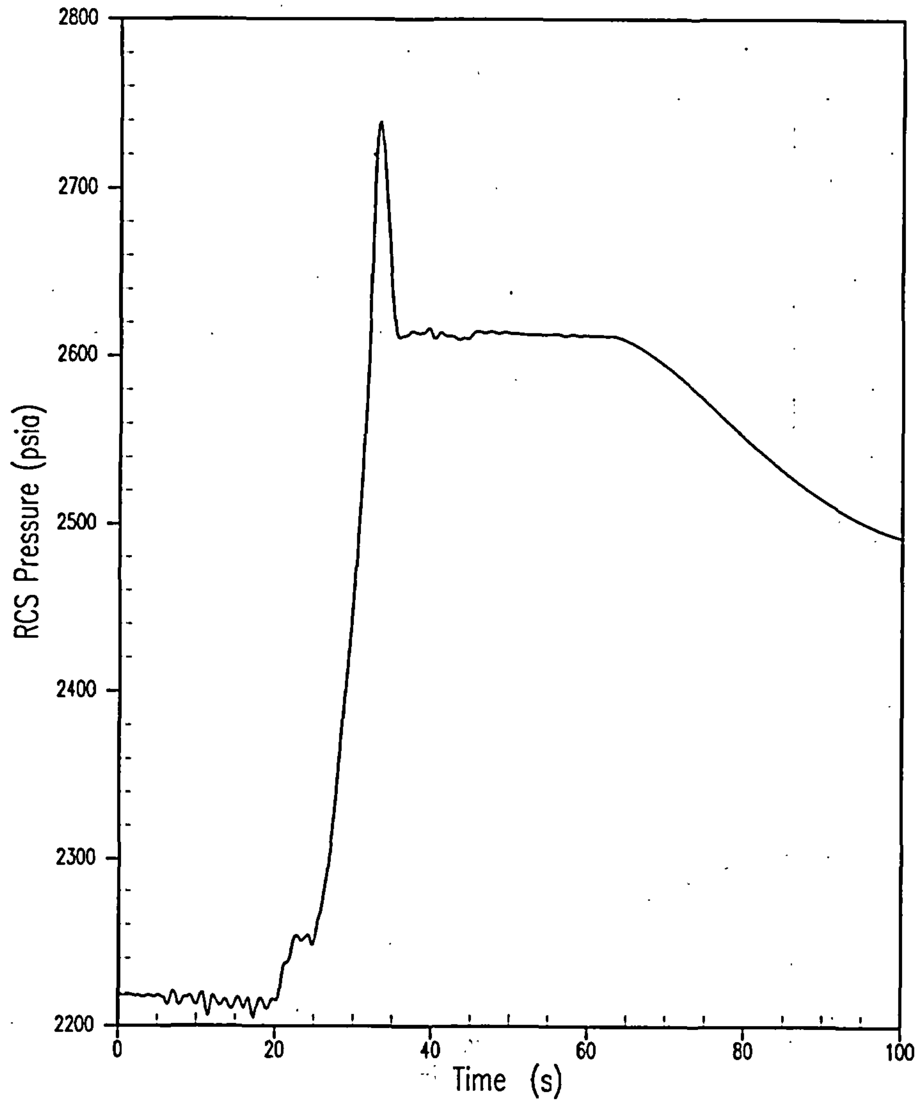

Figure 5.1.12-2 Feedwater Line Break RCS Overpressure Case Limiting Break Size = 0.28 ft^2 RCS Pressure

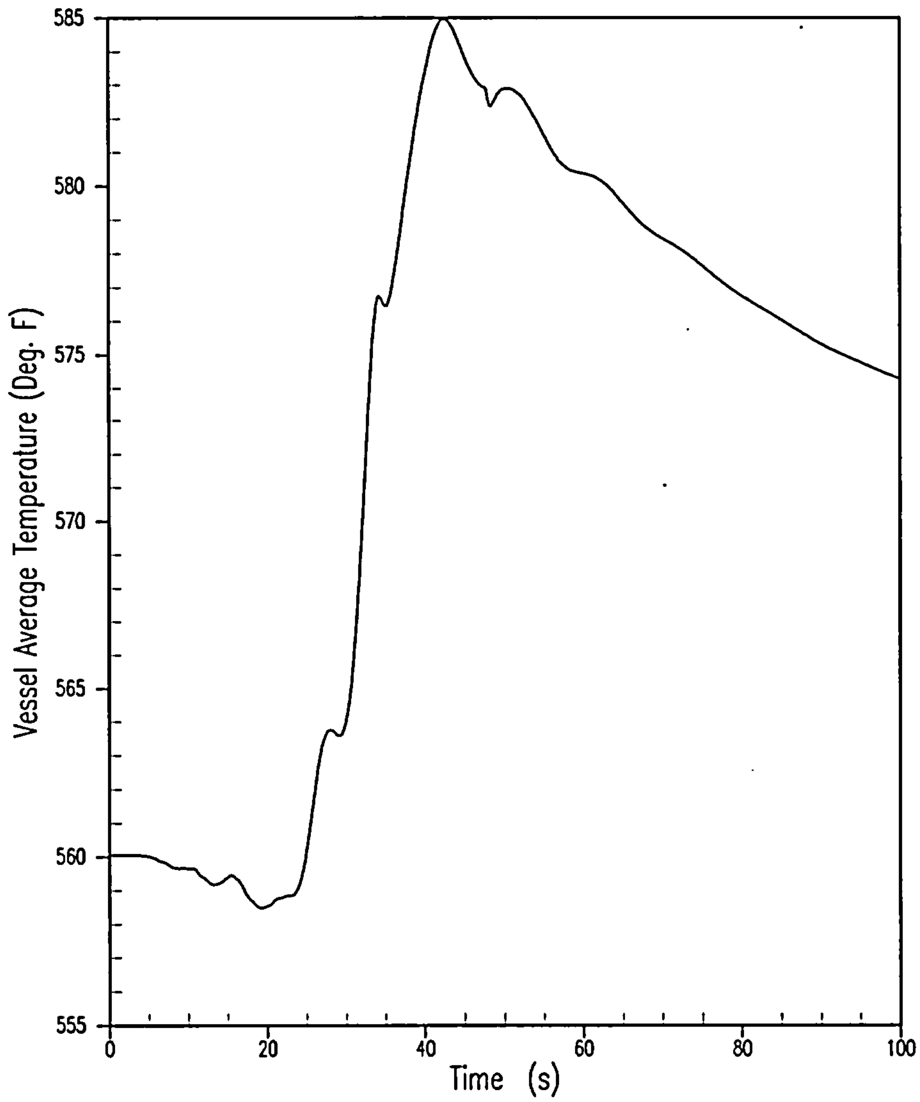

Figure 5.1.12-3 Feedwater Line Break RCS Overpressure Case Limiting Break Size = 0.28 ft^2, Vessel Average Temperature

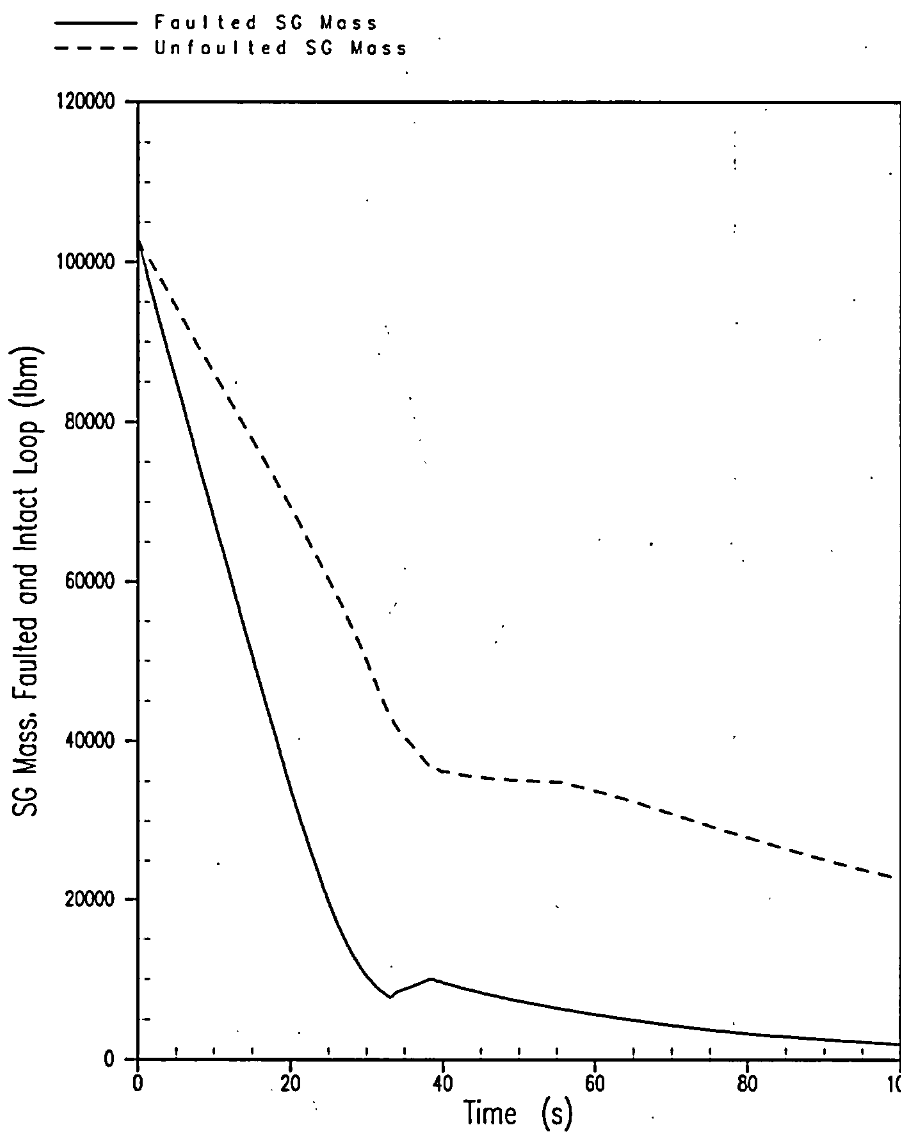

Figure 5.1.12-4 Feedwater Line Break RCS Overpressure Case Limiting Break Size = 0.28 ft² SG Mass, Faulted and Intact Loop

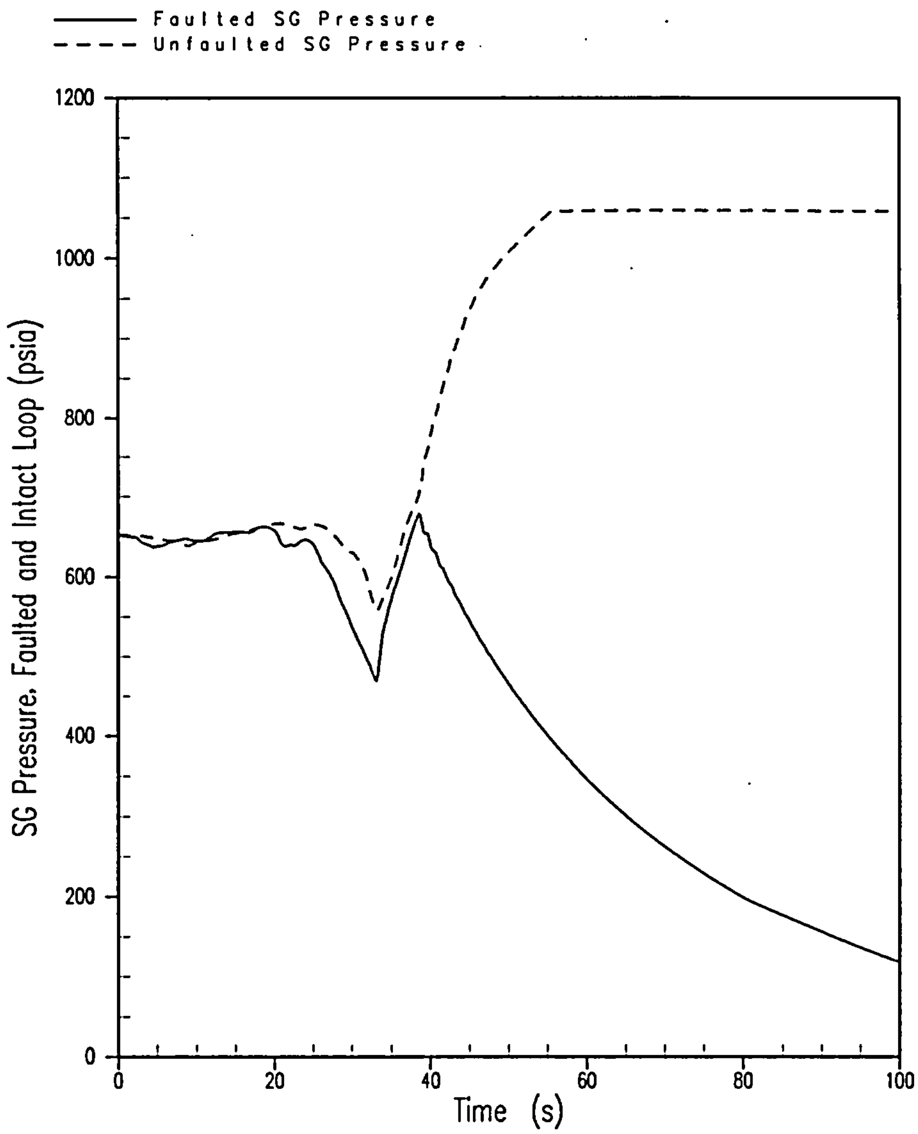

Figure 5.1.12-5 Feedwater Line Break RCS Overpressure Case Limiting Break Size = 0.28 ft^2 SG Pressure, Faulted and Intact Loop

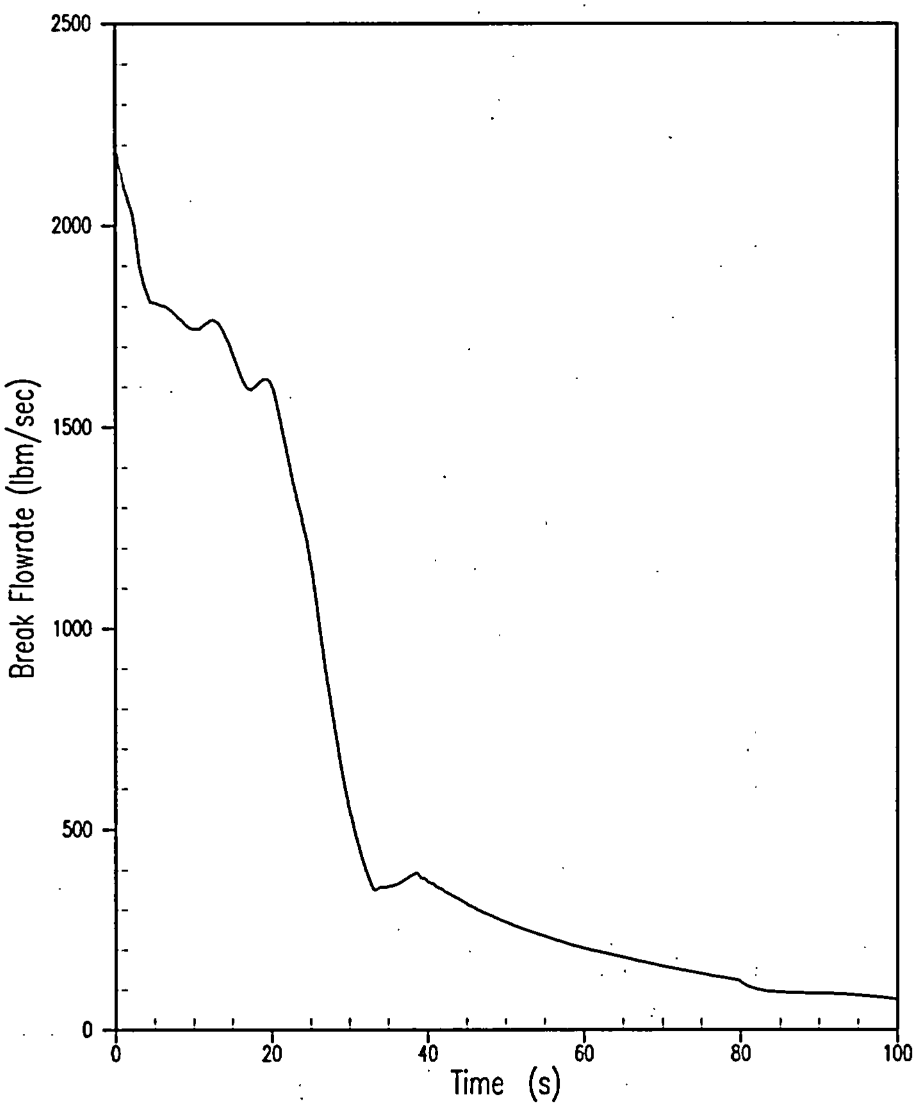

Figure 5.1.12-6 Feedwater Line Break RCS Overpressure Case Limiting Break Size = 0.28 ft^2 Break Flowrate

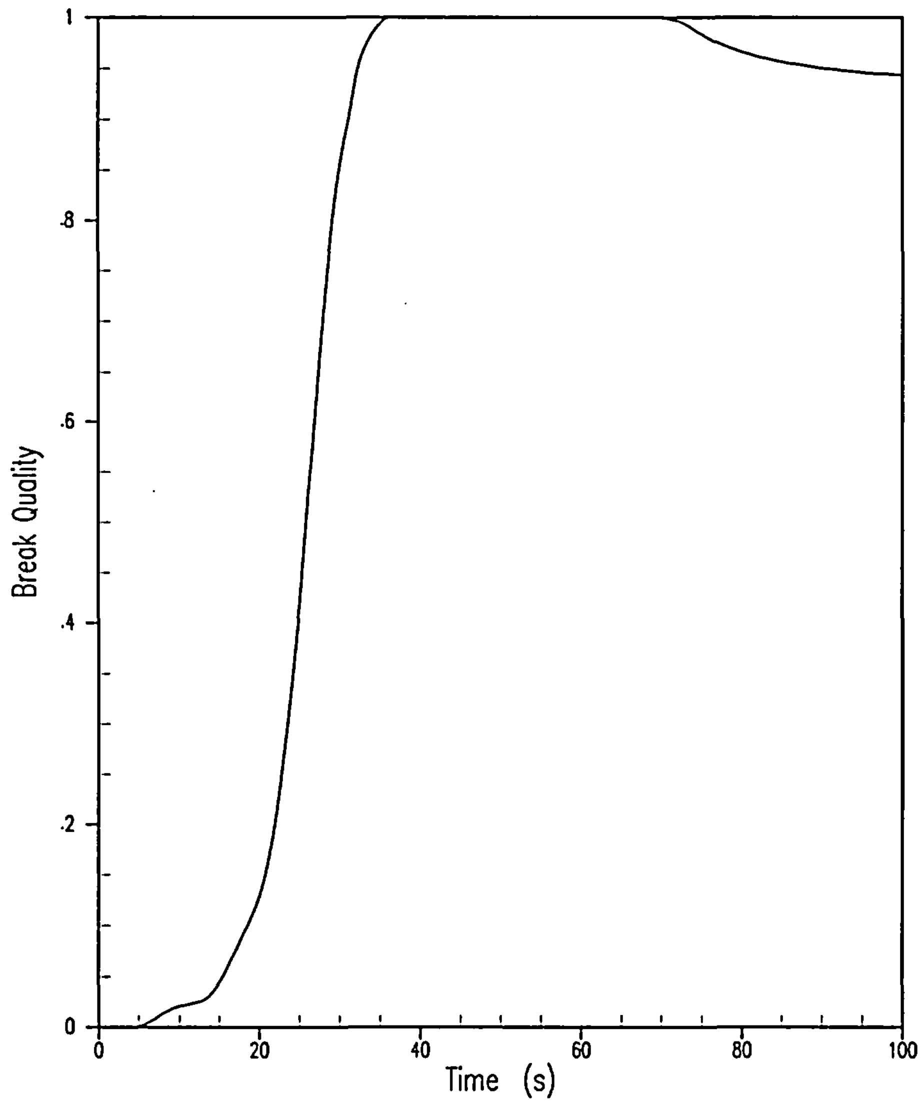

Figure 5.1.12-7 Feedwater Line Break RCS Overpressure Case Limiting Break Size = 0.28 ft² Break Quality

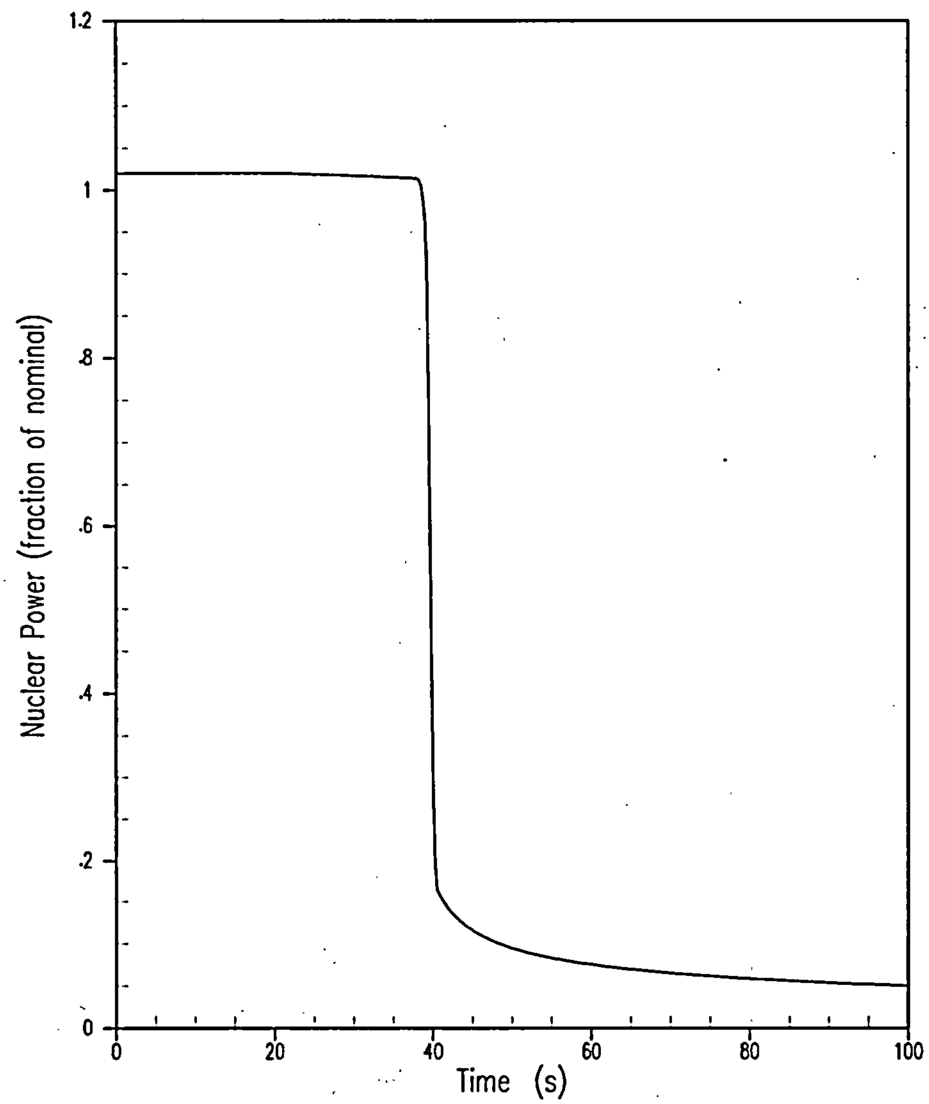

Figure 5.1.12-8 Feedwater Line Break MSS Overpressure Case Limiting Break Size = 0.05 ft^2 Nuclear Power

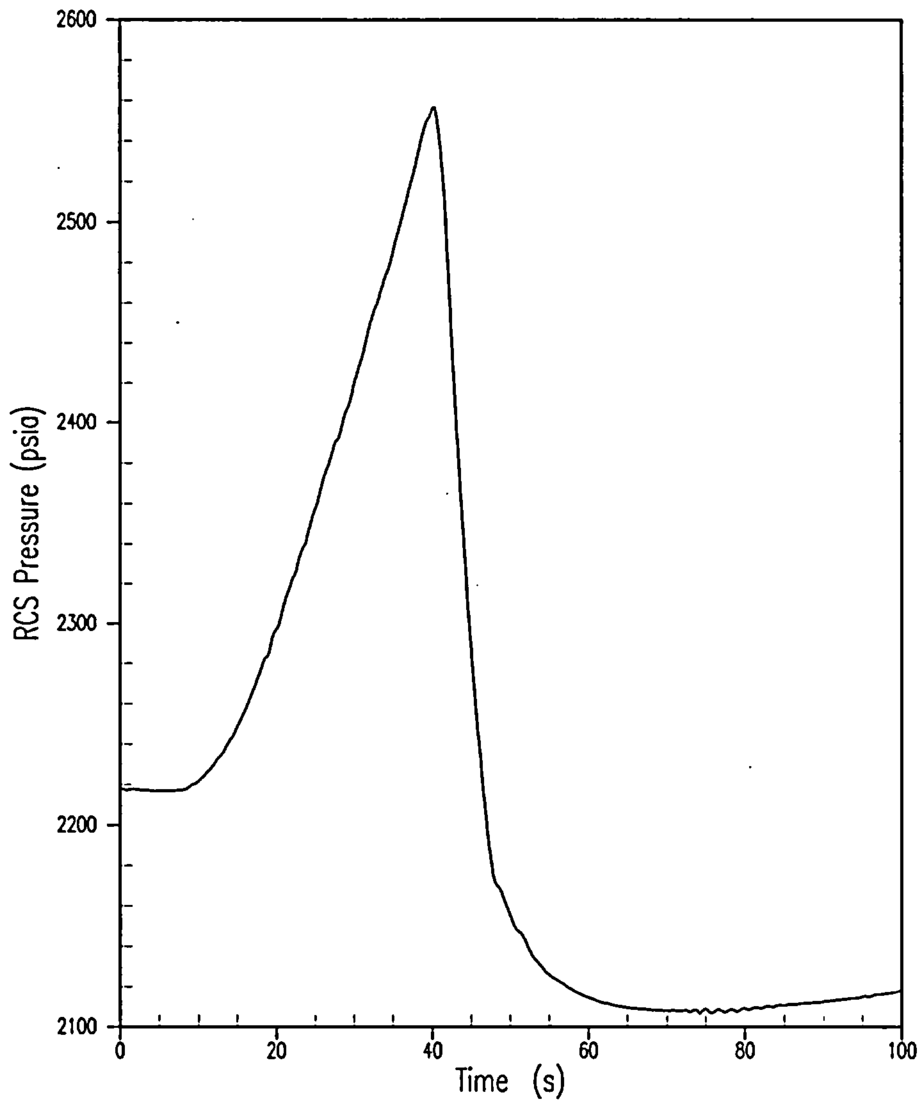

Figure 5.1.12-9 Feedwater Line Break MSS Overpressure Case Limiting Break Size = 0.05 ft² RCS Pressure

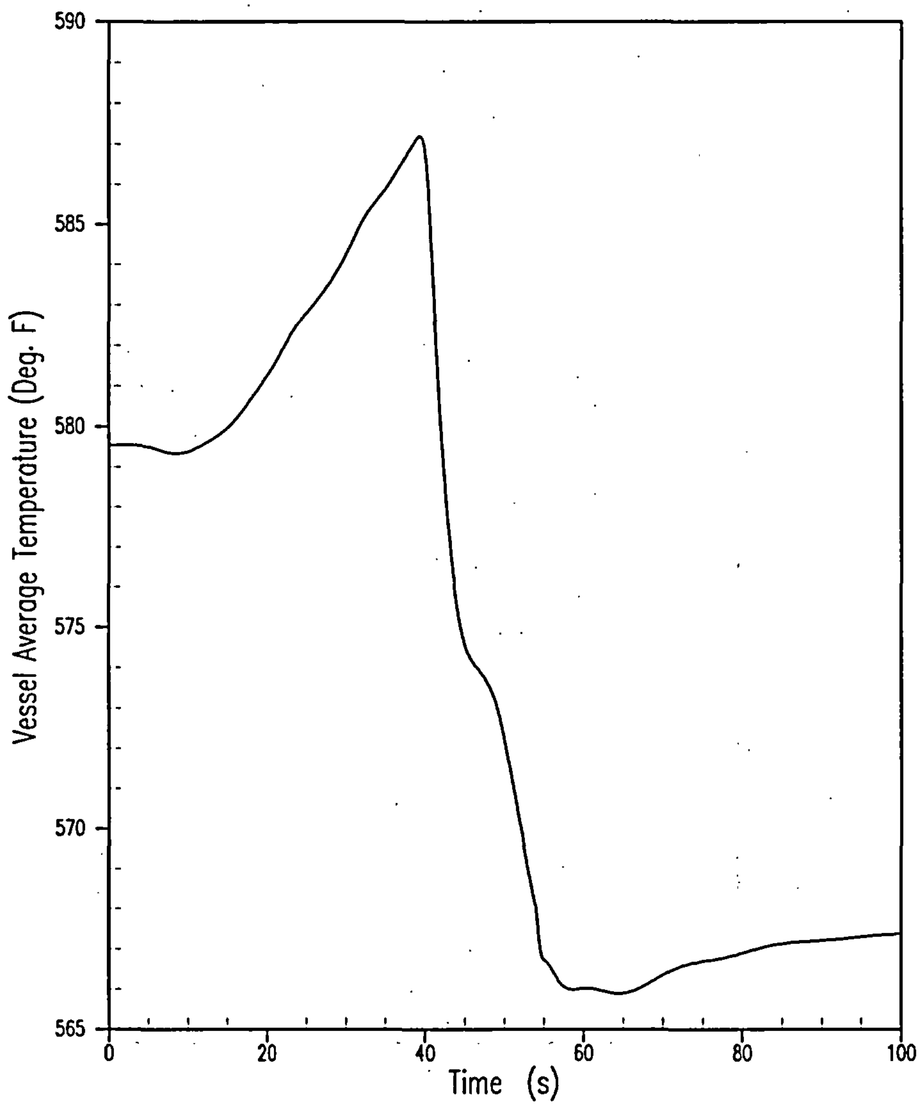

**Figure 5.1.12-10 Feedwater Line Break MSS Overpressure Case Limiting Break Size = 0.05 ft²
Vessel Average Temperature**

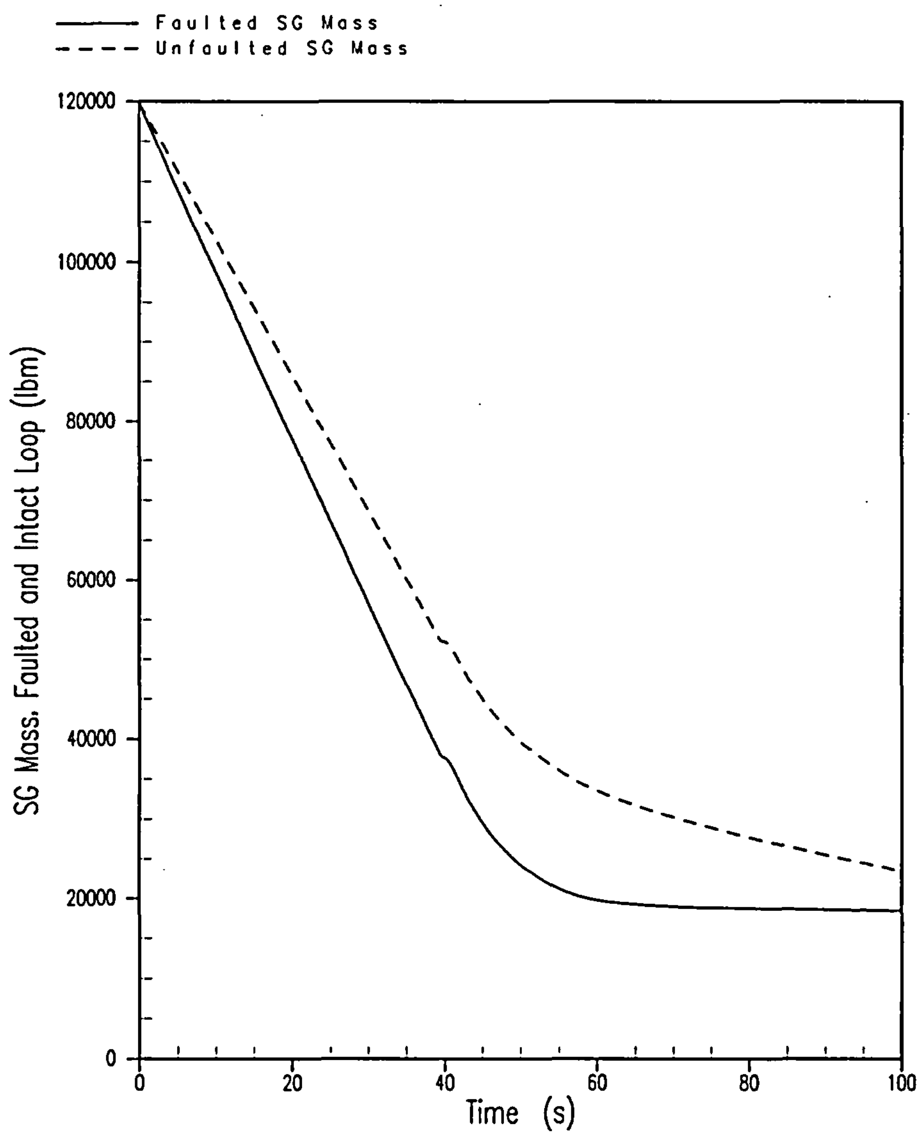

Figure 5.1.12-11 Feedwater Line Break MSS Overpressure Case Limiting Break Size = 0.05 ft² SG Mass, Faulted and Intact Loop

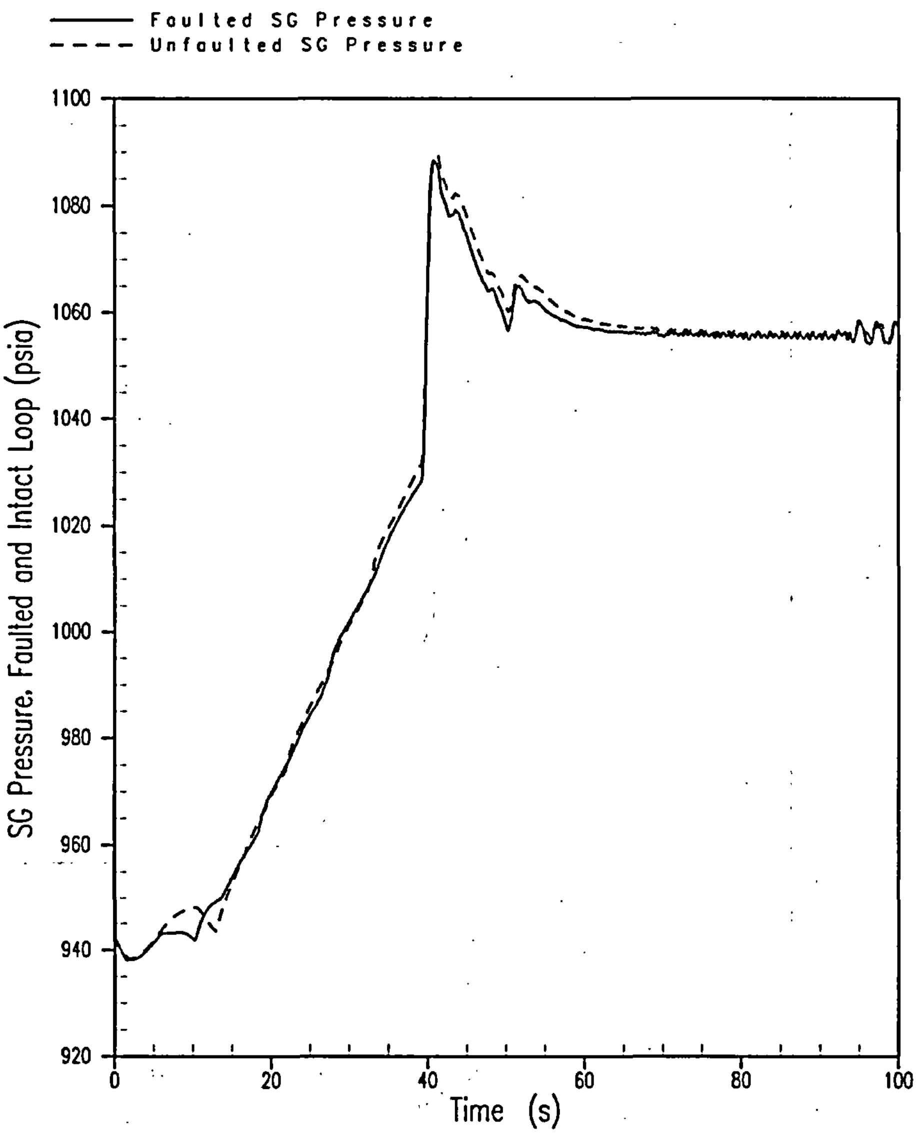

Figure 5.1.12-12 Feedwater Line Break MSS Overpressure Case Limiting Break Size = 0.05 ft^2 SG Pressure, Faulted and Intact Loop

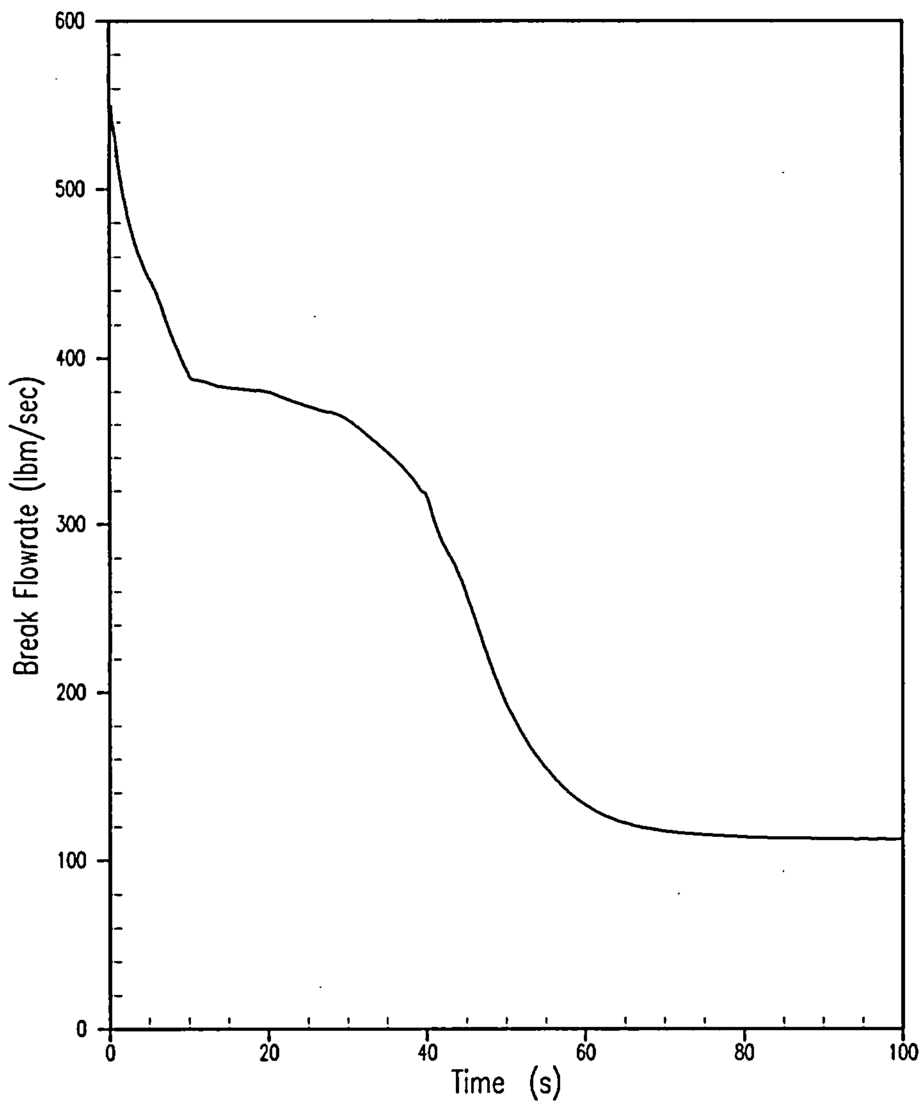

Figure 5.1.12-13 Feedwater Line Break MSS Overpressure Case Limiting Break Size = 0.05 ft^2 Break Flowrate

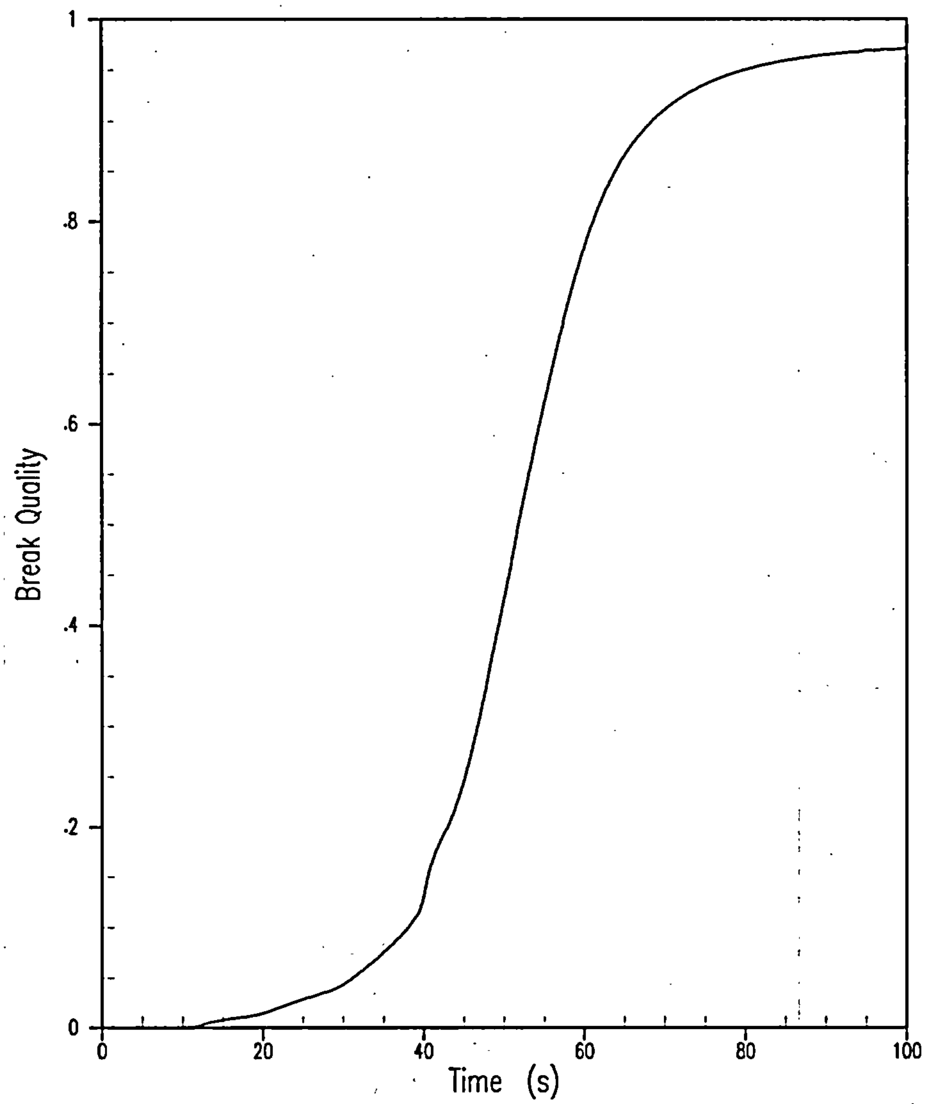

Figure 5.1.12-14 Feedwater Line Break MSS Overpressure Case Limiting Break Size = 0.05 ft^2 Break Quality

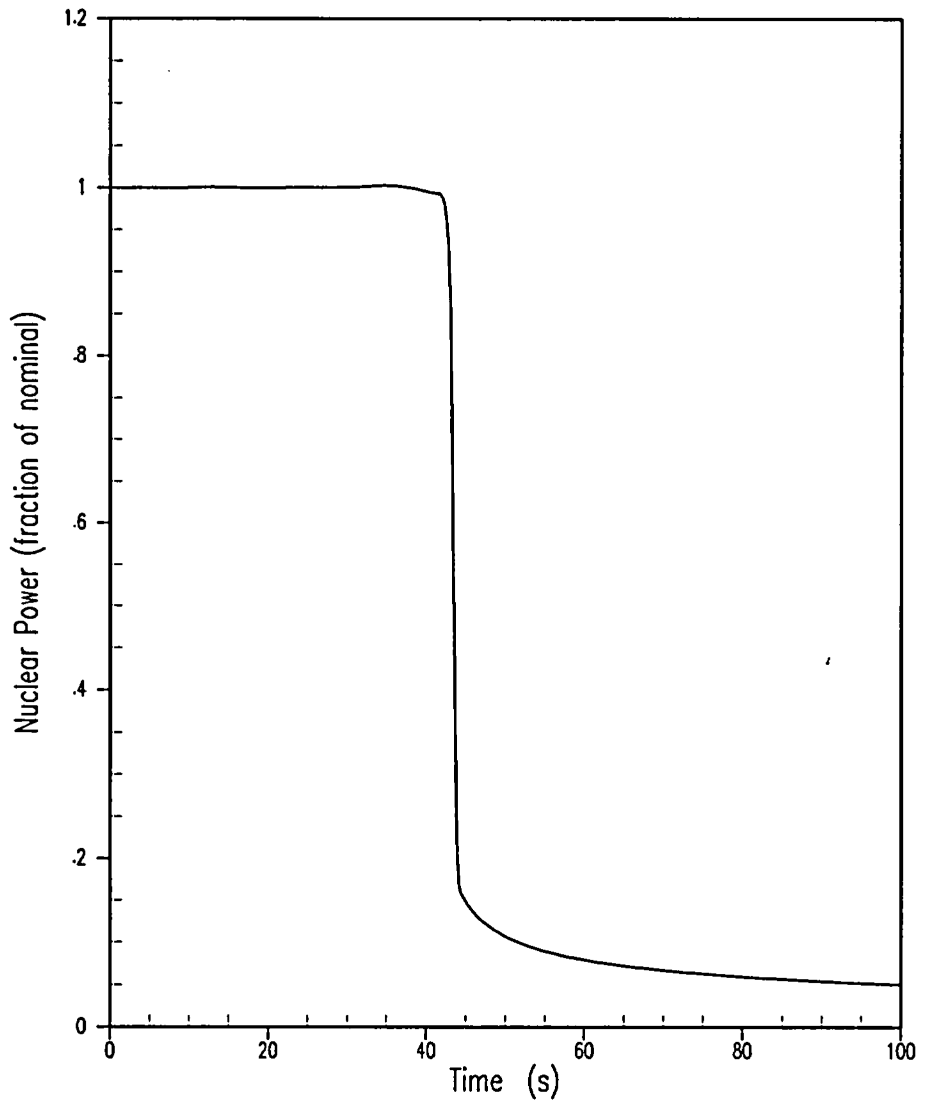

Figure 5.1.12-15 Feedwater Line Break DNB Case Limiting Break Size = 0.25 ft^2 Nuclear Power

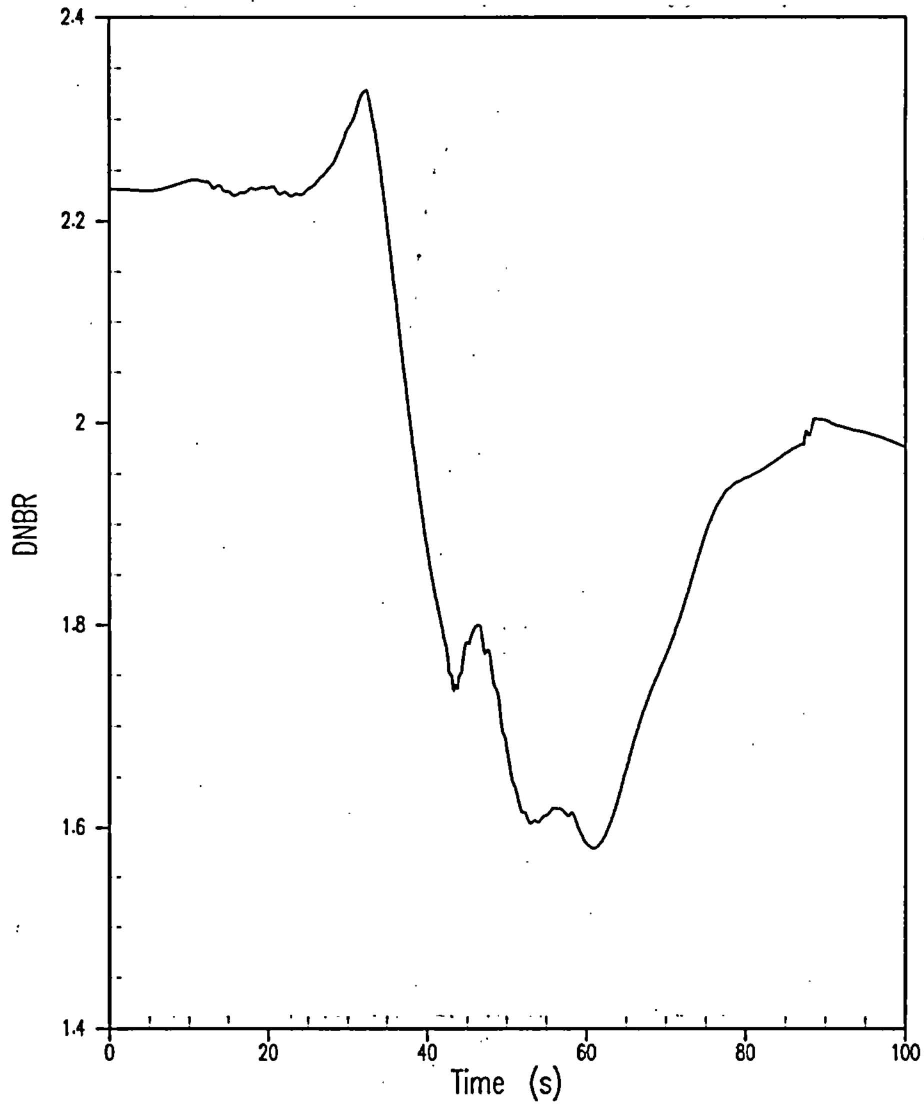

Figure 5.1.12-16 Feedwater Line Break DNB Case Limiting Break Size = 0.25 ft^2 DNBR

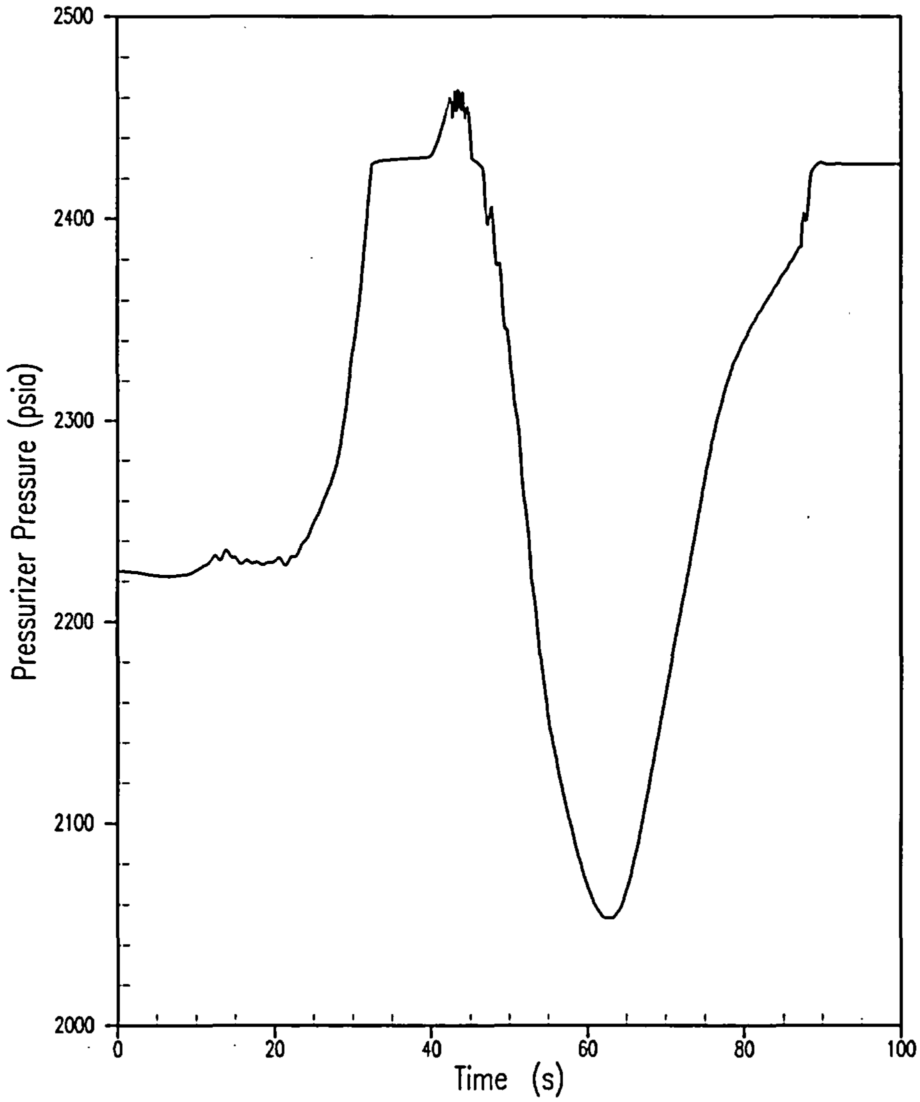

Figure 5.1.12-17 Feedwater Line Break DNB Case Limiting Break Size = 0.25 ft² Pressurizer Pressure

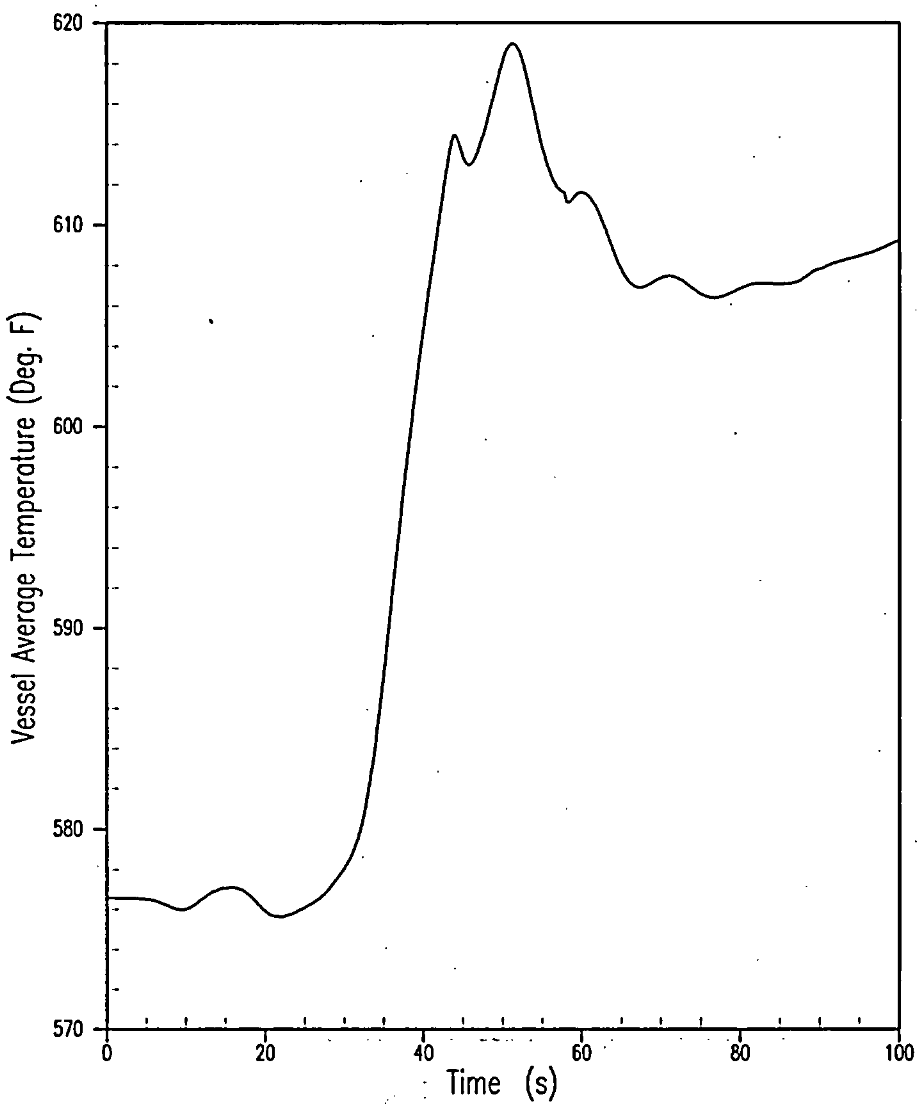

Figure 5.1.12-18 Feedwater Line Break DNB Case Limiting Break Size = 0.25 ft^2 Vessel Average Temperature

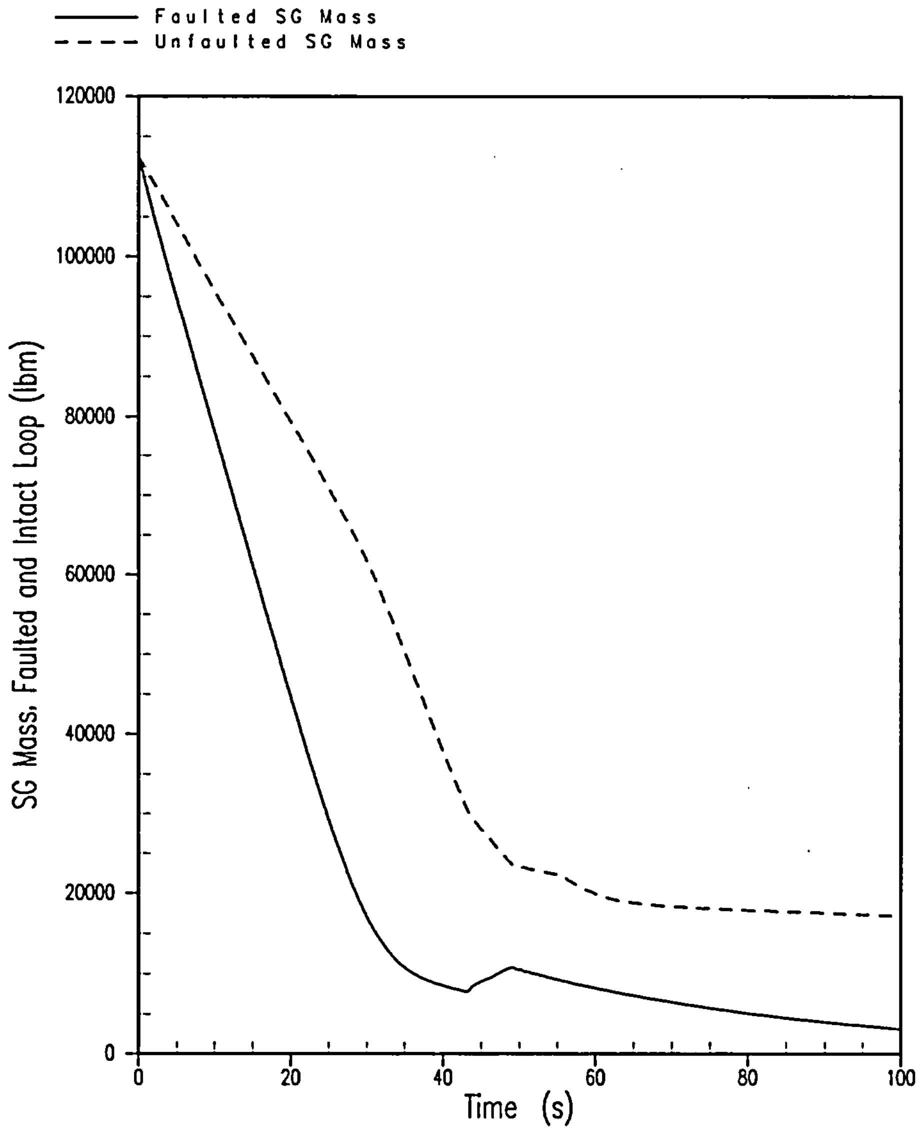

Figure 5.1.12-19 Feedwater Line Break DNB Case Limiting Break Size = 0.25 ft^2 SG Mass, Faulted and Intact Loop

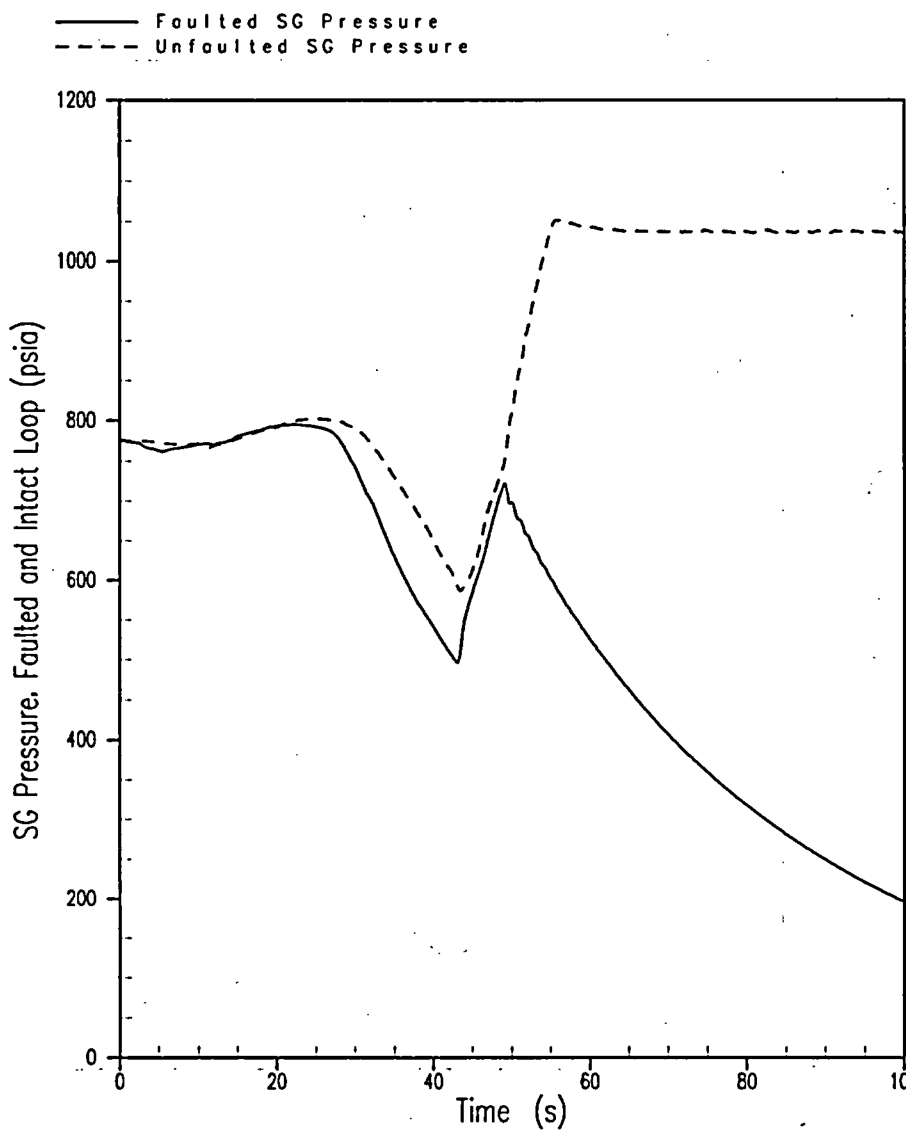

Figure 5.1.12-20 Feedwater Line Break DNB Case Limiting Break Size = 0.25 ft^2 SG Pressure, Faulted and Intact Loop

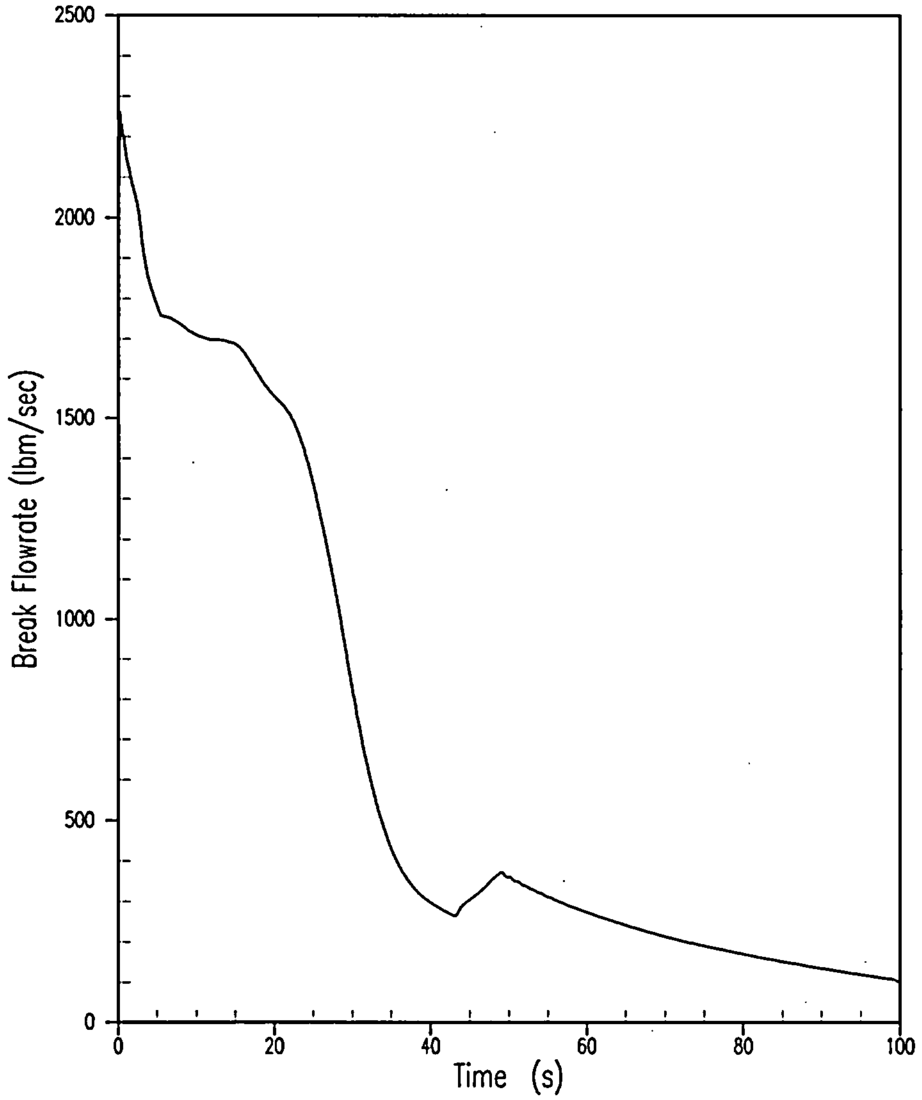

Figure 5.1.12-21 Feedwater Line Break DNB Case Limiting Break Size = 0.25 ft^2 Break Flowrate

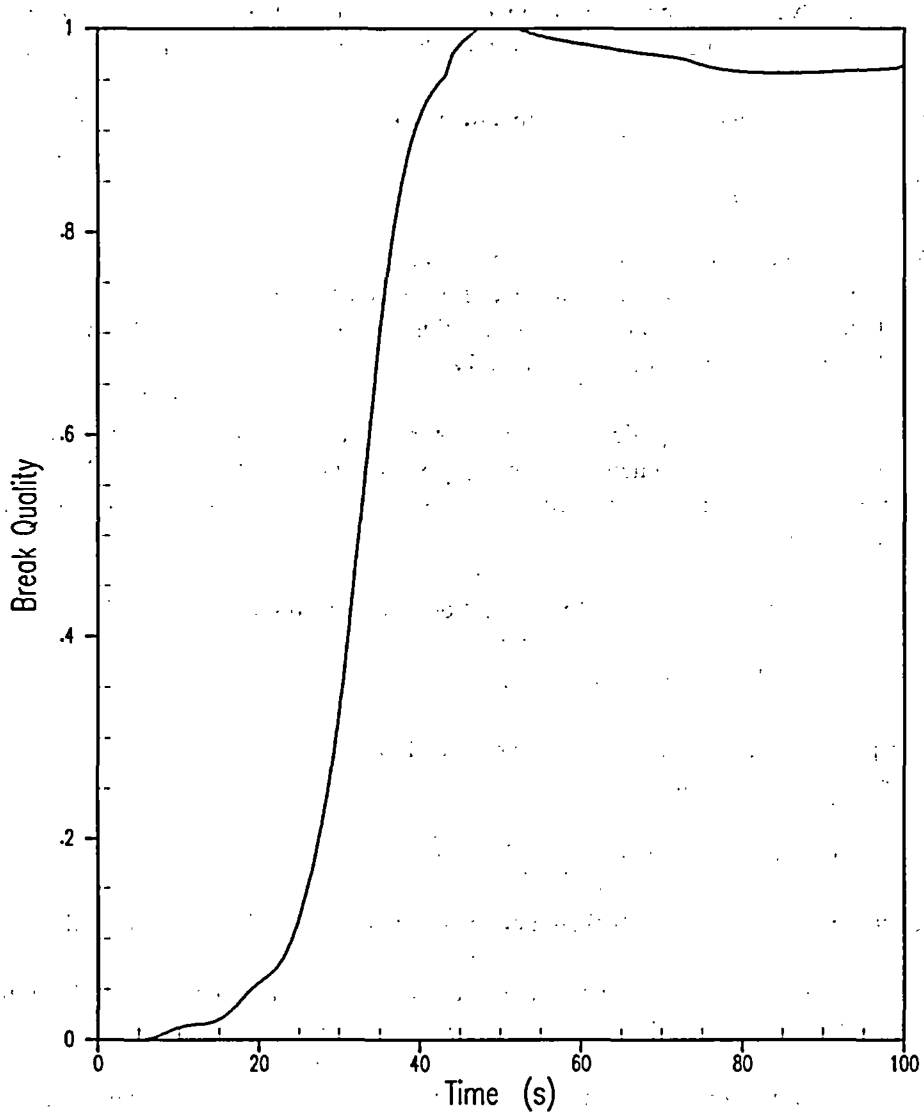

Figure 5.1.12-22 Feedwater Line Break DNB Case Limiting Break Size = 0.25 ft^2 Break Quality

5.1.13 Decrease in Reactor Coolant Flow Rate

Partial loss of forced reactor flow is caused by loss of electrical power to one or more of the reactor coolant pumps (RCPs). This is caused by the opening of an RCP power supply circuit breaker or the loss of a 6.9 kV bus. The core and system performance following a partial loss of forced reactor coolant flow would be no more adverse than those following a total loss of forced reactor coolant flow discussed in the Section 5.1.14. Therefore, an explicit analysis of the Partial Loss of Flow event is not presented herein.

5.1.14 Total Loss of Forced Reactor Coolant Flow

5.1.14.1 Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all RCPs. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly.

Normal power for the RCPs is supplied through buses from a transformer connected to the generator and the offsite power system. Two diametrically opposed pumps are on a separate bus. When a generator trip occurs, the buses continue to be supplied from external power lines and the pumps continue to supply coolant flow to the core.

The following signal provides the necessary protection against a complete loss-of-flow accident:

- Low reactor coolant loop flow reactor trip

The reactor trip on low primary coolant flow is provided to protect against loss-of-flow conditions that affect one or both reactor coolant loops.

This event is conservatively analyzed to the following acceptance criteria:

- Pressure in the RCS and MSS should be maintained below 110 percent of the design values.
- Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the limit value.
- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

5.1.14.2 Method of Analysis

The complete loss-of-flow transient is analyzed as a loss of four RCPs with both loops in operation. The event is analyzed to show that the integrity of the core is maintained as the DNBR remains above the safety analysis limit value. The loss-of-flow event does result in an increase in RCS and MSS pressures, but these pressure increases are generally not severe enough to challenge the integrity of the RCS and MSS. Since the maximum RCS and MSS pressures do not exceed 110 percent of their respective design pressures for the loss-of-condenser vacuum event, it is concluded that the maximum RCS and MSS pressures will also remain below 110 percent of their respective design pressures for the loss-of-flow events.

The limiting case analyzed is a complete loss-of-flow transient due to a loss of power to four pumps.

The transient is analyzed with two computer codes. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary-system pressure and temperature transients. The VIPRE computer code is then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The DNBR transients presented represent the minimum of the typical or thimble cell for the fuel.

This event is analyzed with the Revised Thermal Design Procedure (RTDP) (Reference 1). Initial reactor power, pressurizer pressure and RCS temperature are assumed to be at their nominal values as shown in Table 5.1.0-2. Minimum measured flow is also assumed. A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive MTC limit for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the minimum DNBR is reached.

A limiting DNB axial power shape is assumed in VIPRE for the calculation of DNBR. This shape provides the most limiting minimum DNBR for the loss-of-flow events.

A conservatively low trip reactivity value (5.4-percent $\Delta\rho$) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (2.341 seconds from release to full insertion). The trip reactivity versus rod position curve is confirmed to be valid as part of the reload safety analysis checklist (RSAC) verification process.

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics. Also, it is based on conservative estimates of system pressure losses.

A maximum, uniform, steam generator tube plugging level of 30% was assumed in the RETRAN analysis. Reactor coolant system loop flow asymmetry due to a loop-to-loop steam generator tube plugging imbalance does not need to be considered for transients in which all RCPs experience a coastdown.

5.1.14.3 Results

Figures 5.1.14-1 through 5.1.14-8 illustrate the transient response for the complete loss-of-flow case. All RCPs decelerate at a constant rate until a reactor trip on low flow is initiated. The minimum DNBR is

1.444 (typical cell) / 1.399 (thimble cell), which occurred at 3.55 seconds (Safety Analysis DNBR limit: 1.42 (typical cell) / 1.39 (thimble cell)).

The calculated sequence of events for the complete loss-of-flow case is shown on Table 5.1.14-1. Following reactor trip, the RCPs will continue to coast down, and natural circulation flow will eventually be established. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

5.1.14.4 Conclusions

The analysis performed has demonstrated that for the complete loss-of-flow event, the DNBR does not decrease below the design limit value at any time during the transient. Therefore, no fuel or cladding damage is predicted and all applicable acceptance criteria are met.

5.1.14.5 References

1. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), April 1989.

Table 5.1.14-1 Sequence of Events – Complete Loss of Reactor Coolant Flow	
Complete Loss of Flow	
Event	**Time (seconds)**
All Operating RCPs Lose Power and Coastdown Begins	0.0
Low Flow Reactor Trip Setpoint is Reached	0.941
Rods Begin to Drop	2.081
Minimum DNBR Occurs	3.55

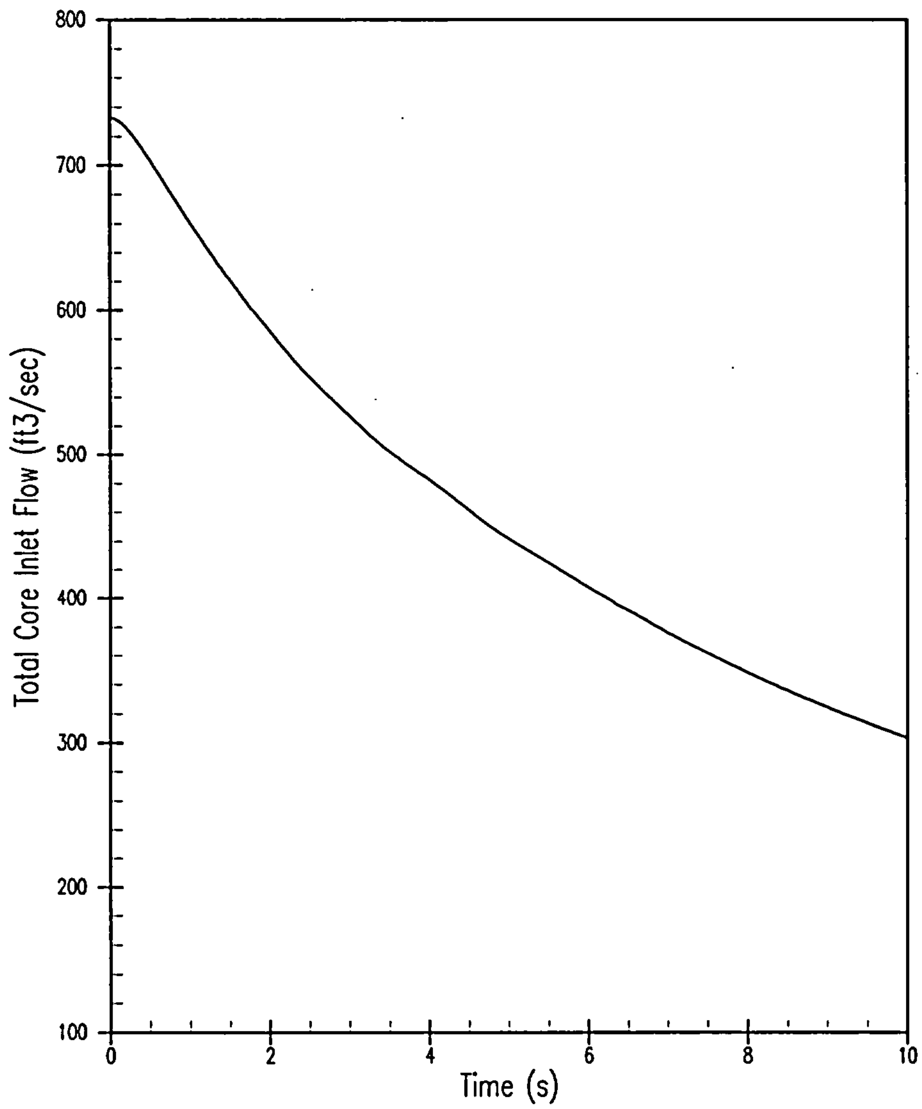

Figure 5.1.14-1 Total Core Inlet Flow versus Time – Complete Loss of Flow – Four Pumps Coasting Down (CLOF)

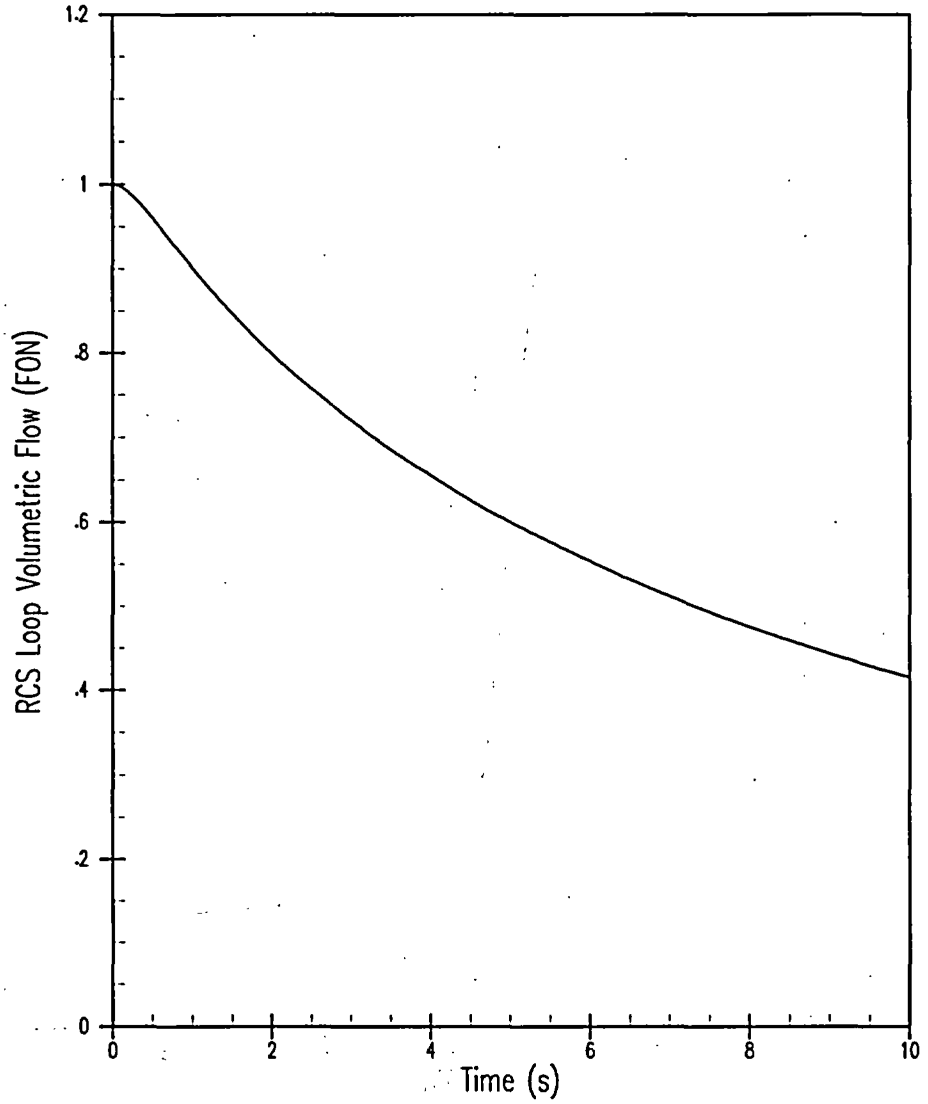

Figure 5.1.14-2 RCS Loop Flow versus Time – Complete Loss of Flow – Four Pumps Coasting Down (CLOF)

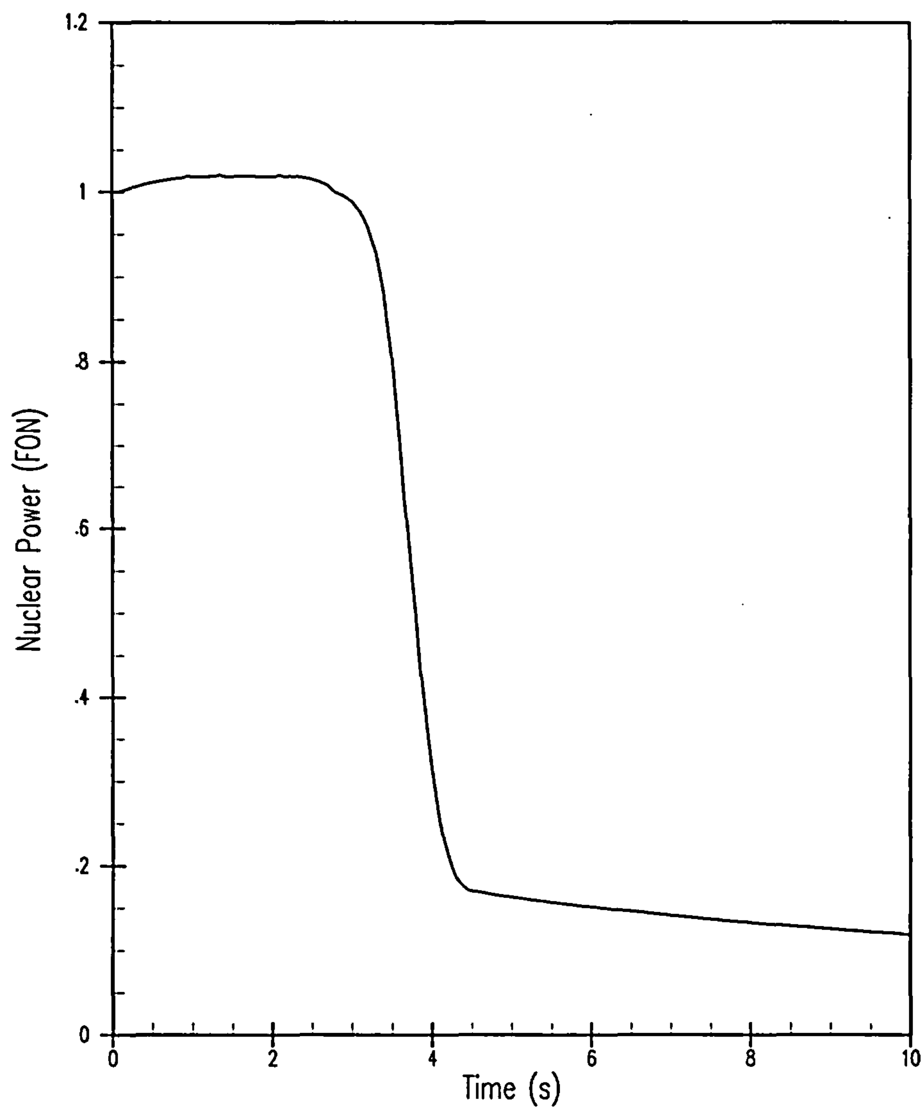

Figure 5.1.14-3 Nuclear Power versus Time – Complete Loss of Flow – Four Pumps Coasting Down (CLOF)

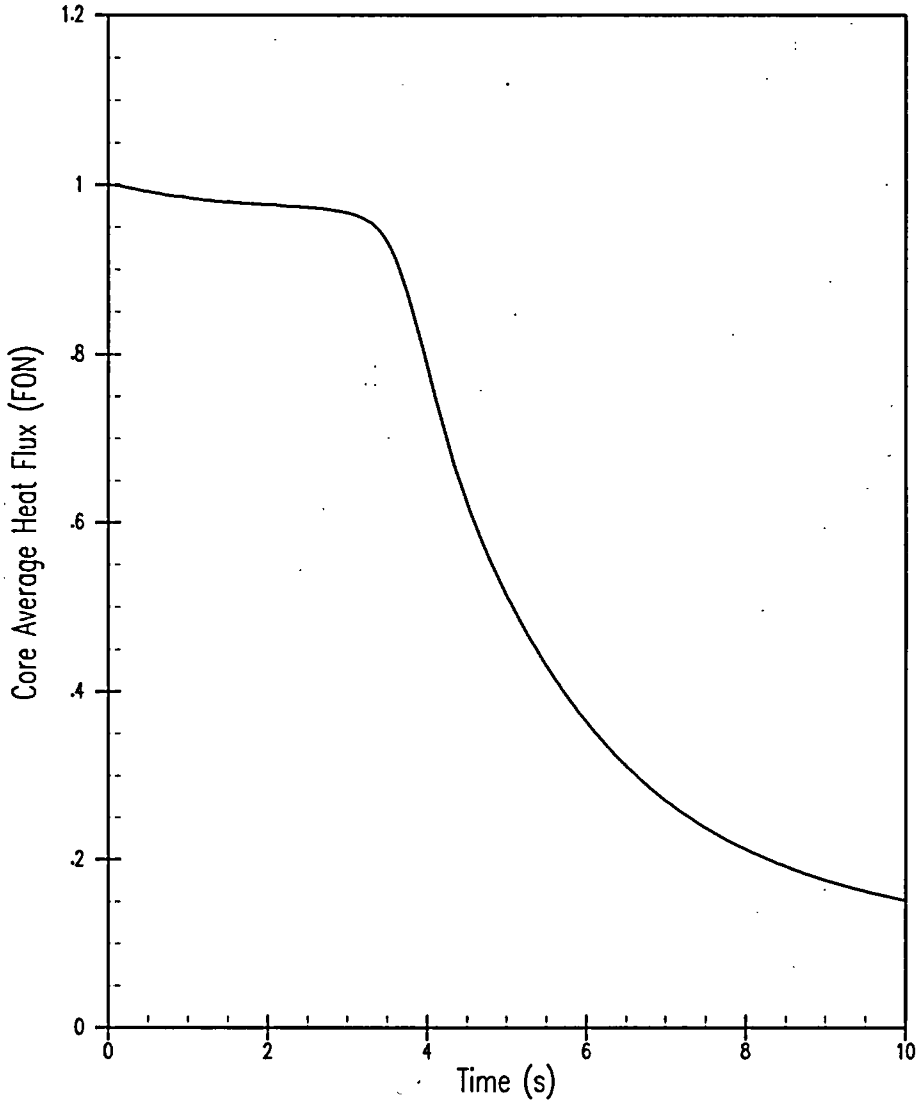

Figure 5.1.14-4 Core Average Heat Flux versus Time – Complete Loss of Flow – Four Pumps Coasting Down (CLOF)

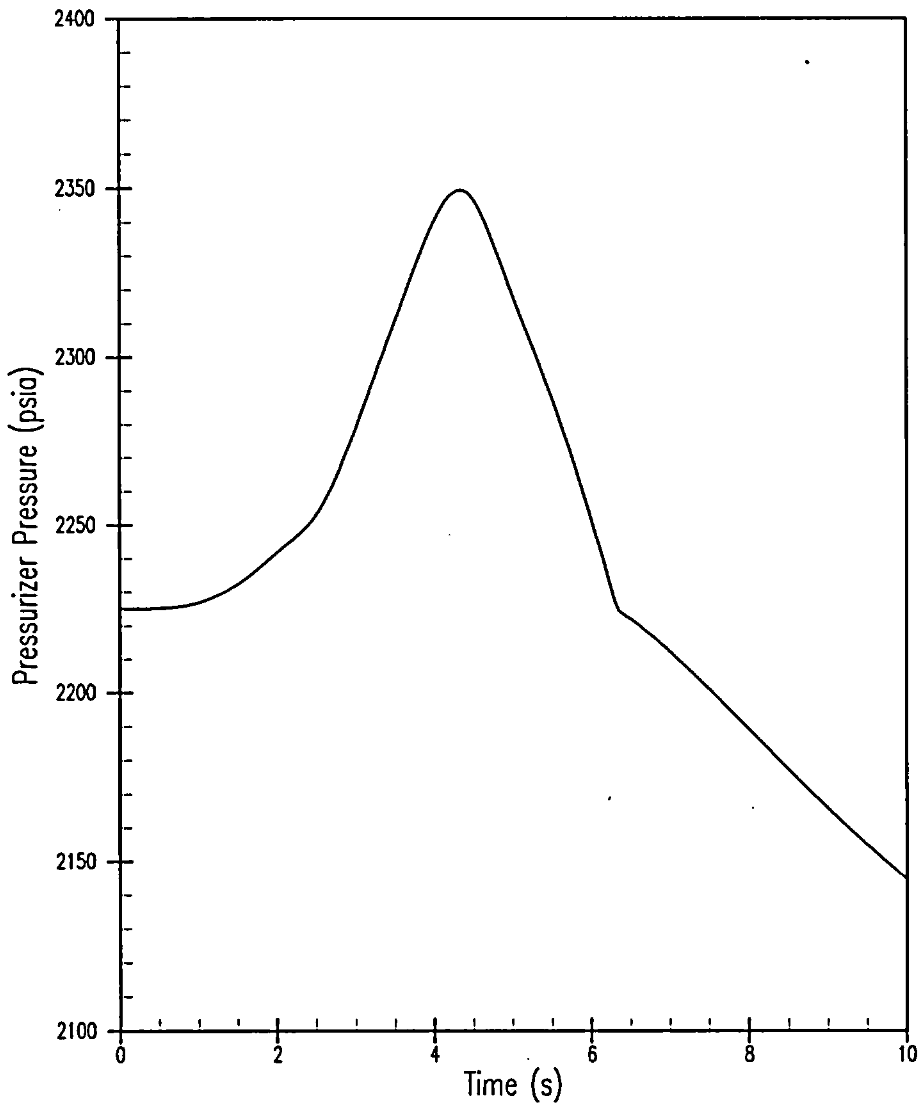

Figure 5.1.14-5 Pressurizer Pressure versus Time – Complete Loss of Flow – Four Pumps Coasting Down (CLOF)

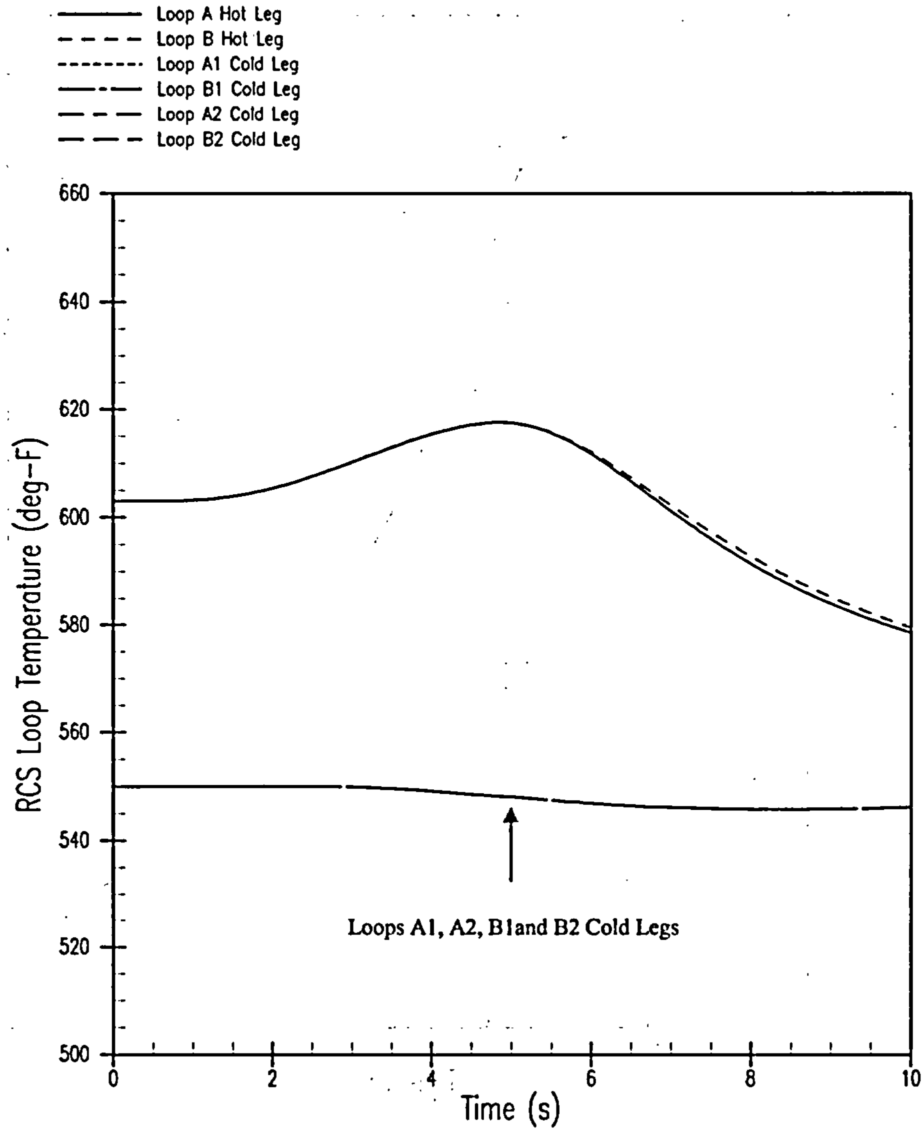

Figure 5.1.14-6 RCS Loop Temperature versus Time – Complete Loss of Flow – Four Pumps Coasting Down (CLOF)

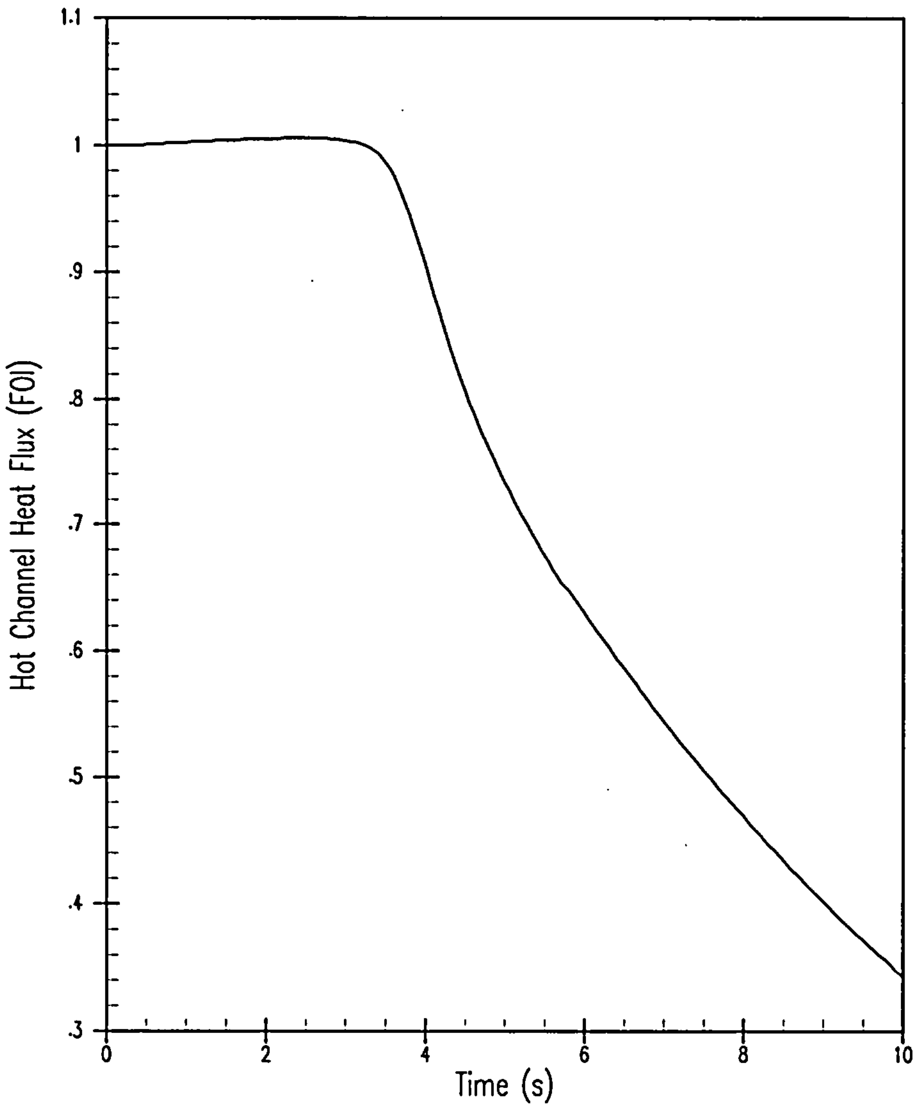

Figure 5.1.14-7 Hot Channel Heat Flux versus Time – Complete Loss of Flow, Four Pumps Coasting Down (CLOF)

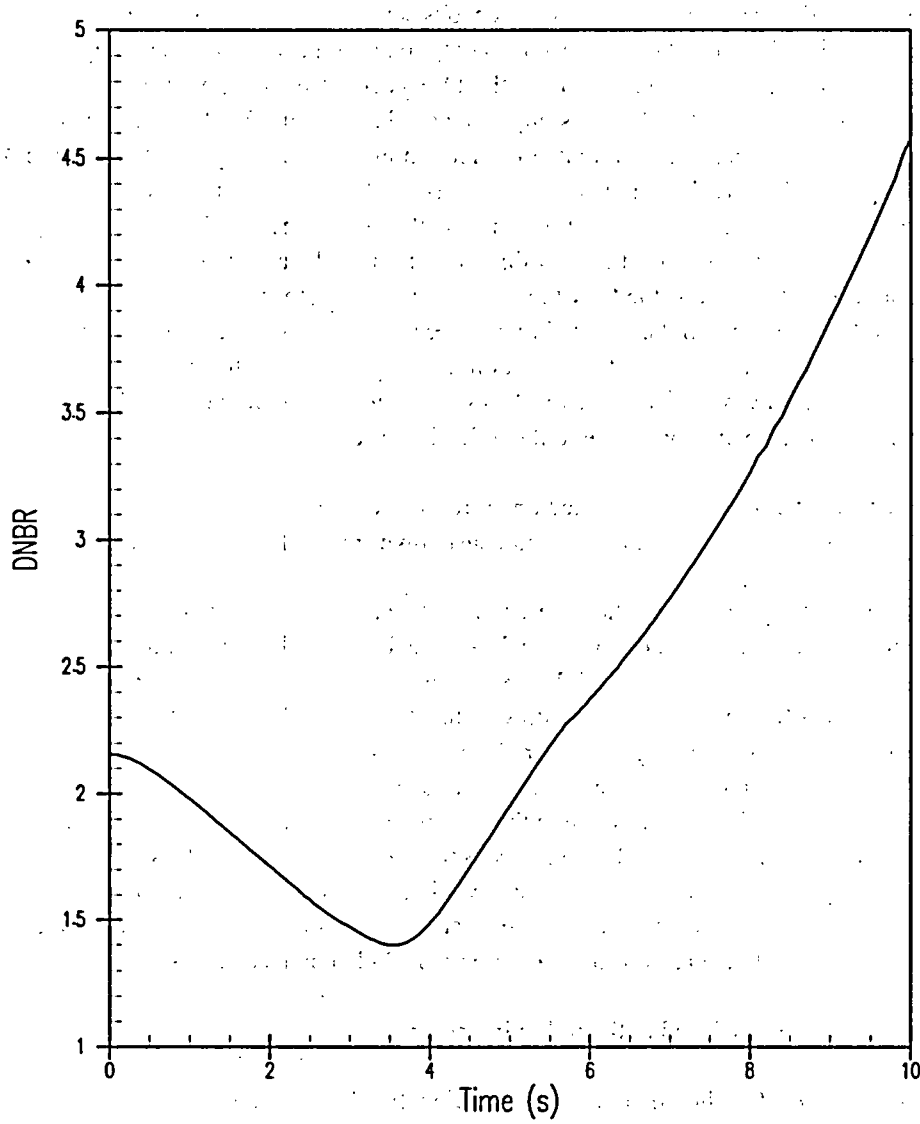

Figure 5.1.14-8 DNBR versus Time – Complete Loss of Flow, Four Pumps Coasting Down (CLOF)

5.1.15 Total Single RCP Shaft Seizure/Sheared Shaft

5.1.15.1 Accident Description

The postulated locked-rotor accident is an instantaneous seizure of an RCP rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low-flow signal. The consequences of a postulated pump shaft break accident are similar to the locked-rotor event. With a broken shaft, the impeller is free to spin, as opposed to it being fixed in position during the locked-rotor event. Therefore, the initial rate of reduction in core flow is greater during a locked-rotor event than in a pump shaft break event because the fixed shaft causes greater resistance than a free-spinning impeller early in the transient, when flow through the affected loop is in the positive direction. As the transient continues, the flow direction through the affected loop is reversed. If the impeller is able to spin free freely, the flow to the core will be less than that available with a fixed-shaft during periods of reverse flow in the affected loop. Because peak pressure, cladding temperature, and DNB occur very early in the transient, the reduction in core flow during the period of forward flow in the affected loop dominates the severity of the results. Consequently, the bounding results for the locked-rotor transients also are applicable to the RCP shaft break.

After the locked rotor, reactor trip is initiated on an RCS low-flow signal. At the time of turbine trip, the unaffected RCPs are assumed to lose power and coast down freely after a 3-second delay.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced. This is because, first, the reduced flow results in a decreased tube-side film coefficient; and then because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the PORVs, and opens the pressurizer safety valves, in that sequence. The two PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism in the peak-pressure evaluation, their pressure-reducing effect and the pressure-reducing effect of the pressurizer sprays are not included in the analysis.

The locked-rotor event is analyzed to the following criteria:

- Pressure in the RCS should be maintained below the designated limit.
- Coolable core geometry is ensured by showing that the peak cladding temperature and maximum oxidation level for the hot spot are below 2700°F and 16.0 percent by weight, respectively.
- Activity release is such that the calculated doses meet 10 CFR Part 100 guidelines.

For St. Lucie Unit 2, the locked-rotor RCS pressure limit is equal to 110 percent of the design value, or 2750 psia. For the secondary side, the locked-rotor pressure limit is also assumed to be equal to 110 percent of design pressure, or 1100 psia. Since the loss of condenser vacuum analysis bounds the locked rotor with respect to MSS overpressurization, a specific MSS overpressurization analysis is not performed.

A hot-spot evaluation is performed to calculate the peak cladding temperature and maximum oxidation level. Finally, a calculation of the "rods-in-DNB" is performed for input to the radiological dose analysis.

5.1.15.2 Method of Analysis

The locked-rotor transient is analyzed with two computer codes. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary-system pressure and temperature transients. The VIPRE computer code is then used to calculate the thermal behavior of the fuel located at the core hot spot including the rods-in-DNB using the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN.

For the case analyzed to determine the maximum RCS pressure and peak cladding temperature, the plant is assumed to be in operation under the most adverse steady-state operating conditions; that is, a maximum steady-state thermal power, maximum steady-state pressure, and maximum steady-state coolant average temperature. The case analyzed to determine the rods-in-DNB utilizes the RTDP methodology. Initial reactor power, pressurizer pressure and RCS temperature are assumed to be at their nominal values as shown in Table 5.1.0-2. Minimum measured flow is also assumed.

A maximum, uniform, steam generator tube plugging level of 30 percent was assumed in the RETRAN analysis. The effect of a flow asymmetry resulting from asymmetric tube plugging is addressed in the DNB analysis of the locked rotor statepoints.

A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive MTC limit for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the peak RCS pressures and hot-spot results are reached.

A conservatively low trip reactivity value (5.4-percent $\Delta\rho$) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (2.66 seconds from release to full insertion). The trip reactivity versus rod position curve is confirmed to be valid as part of the RSAC verification process.

A loss-of-offsite-power is assumed with the unaffected RCPs losing power 3 seconds after turbine trip.

For the peak RCS pressure evaluation, the initial pressure is conservatively set as shown in Table 5.1.0-2 to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. The peak RCS pressure occurs in the lower plenum of the vessel. The pressure transient in the lower plenum is shown in Figure 5.1.15-6.

For this accident, an evaluation of the consequences with respect to the fuel rod thermal transient is performed. The evaluation incorporates the assumption of rods going into DNB as a conservative initial condition to determine the cladding temperature and zirconium water reaction resulting from the locked rotor. Results obtained from the analysis of this hot-spot condition represent the upper limit with respect to cladding temperature and zirconium water reaction. In the evaluation, the rod power at the hot spot is assumed to be 2.9 times the average rod power (that is, $F_Q = 2.90$) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient is calculated in the VIPRE code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature. The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and RCS flow rate as a function of time are based on the RETRAN results.

Fuel Cladding Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between the pellet and cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient was assumed to increase from a steady-state value consistent with initial fuel temperature to 10,000 BTU/hr-ft^2-°F at the initiation of the transient. Therefore, the large amount of energy stored in the fuel because of the small initial value is released to the cladding at the initiation of the transient.

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (cladding temperature). The Baker-Just parabolic rate equation is used to define the rate of zirconium-steam reaction. The effect of the zirconium-steam reaction is included in the calculation of the hot-spot cladding temperature transient.

5.1.15.3 Results

Figures 5.1.15-1 through 5.1.15-9 illustrate the transient response for the locked-rotor event. The peak RCS pressure is 2596 psia and is less than the acceptance criterion of 2750 psia. Also, the peak cladding temperature is 1637.1°F, which is considerably less than the limit of 2700°F. The zirconium-steam reaction at the hotspot is 0.14 percent by weight, which meets the criterion of less than 16-percent zirconium-steam water reaction. For the radiological dose evaluation, the total percentage of fuel rods calculated to experience DNB is less than 1 percent (rods-in-DNB case). The sequence of events for the peak RCS pressure/peak cladding temperature case is given in Table 5.1.15-1. This transient trips on a low primary reactor coolant flow trip setpoint, which is assumed to be 91.9 percent.

5.1.15.4 Conclusions

The analysis performed has demonstrated that for the locked-rotor event, the RCS pressure remains below 110 percent of the design pressure and the hot-spot cladding temperature and oxidation levels remain below the limit values. Therefore, all applicable acceptance criteria are met. In addition, the total percentage of rods calculated to experience DNB is less than 1 percent.

Table 5.1.15-1 Sequence of Events – Reactor Coolant Pump Locked Rotor	
Event	**Time (seconds)**
Rotor on One Pump Locks	0.000
Low Flow Reactor Trip Setpoint Reached	0.232
Rods Begin to Drop	1.372
Maximum Cladding Temperature Occurs	3.300
Maximum RCS Pressure Occurs	3.400
Remaining Active Pumps Begin to Coastdown	3.634

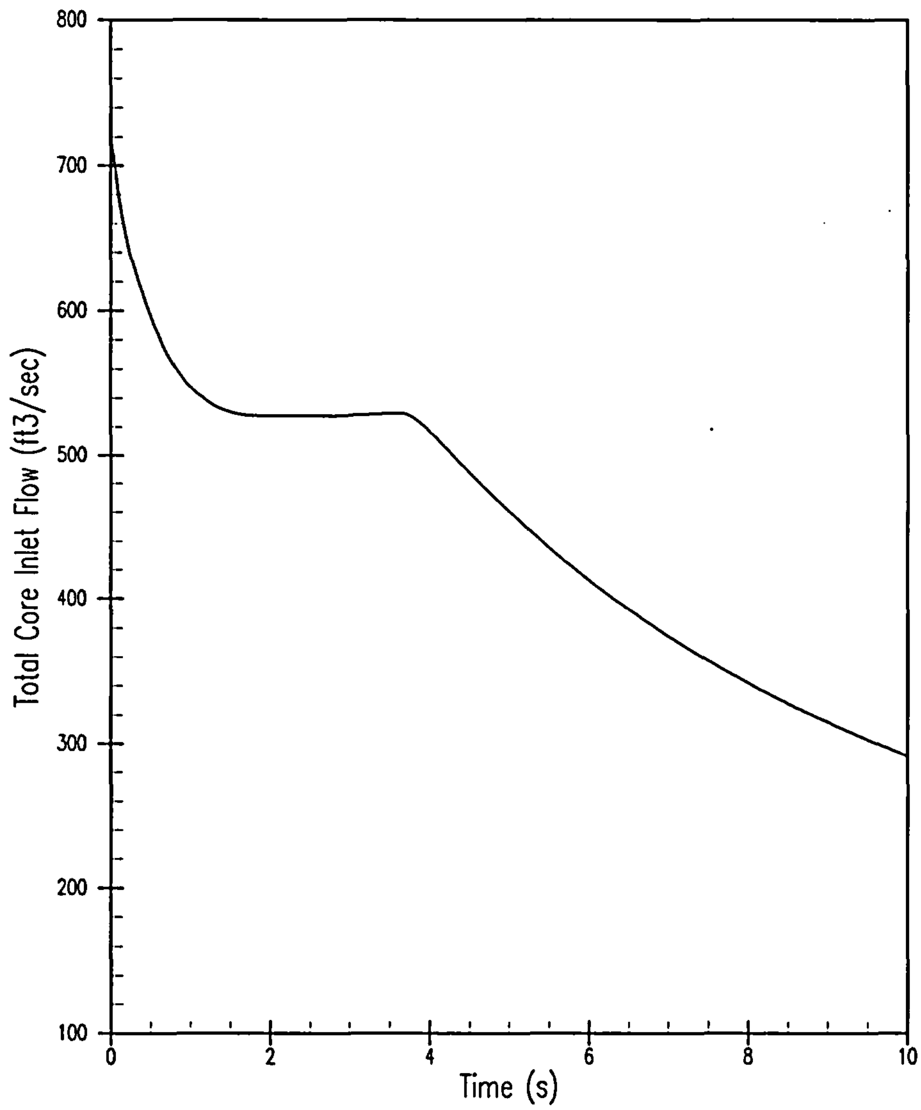

Figure 5.1.15-1 Total Core Inlet Flow versus Time – Locked Rotor/Shaft Break

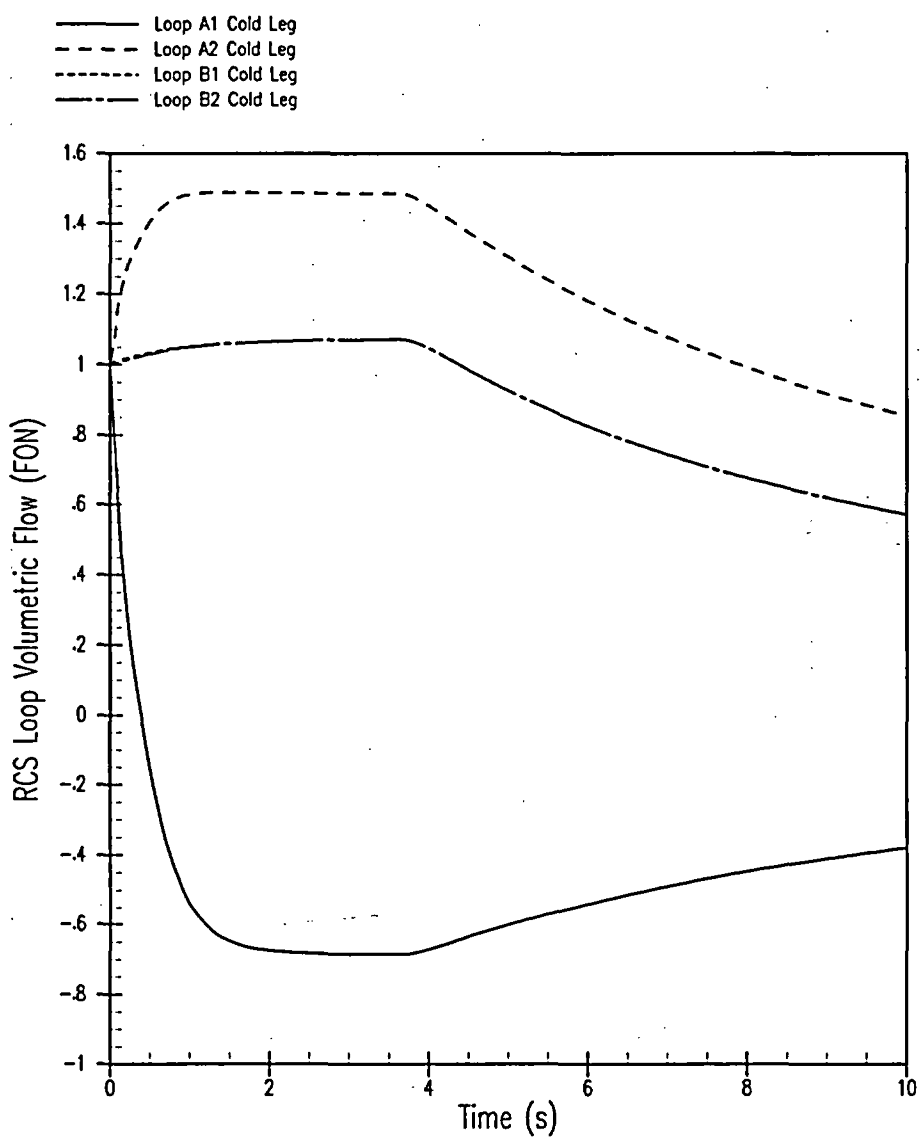

Figure 5.1.15-2 RCS Loop Flow versus Time – Locked Rotor/Shaft Break

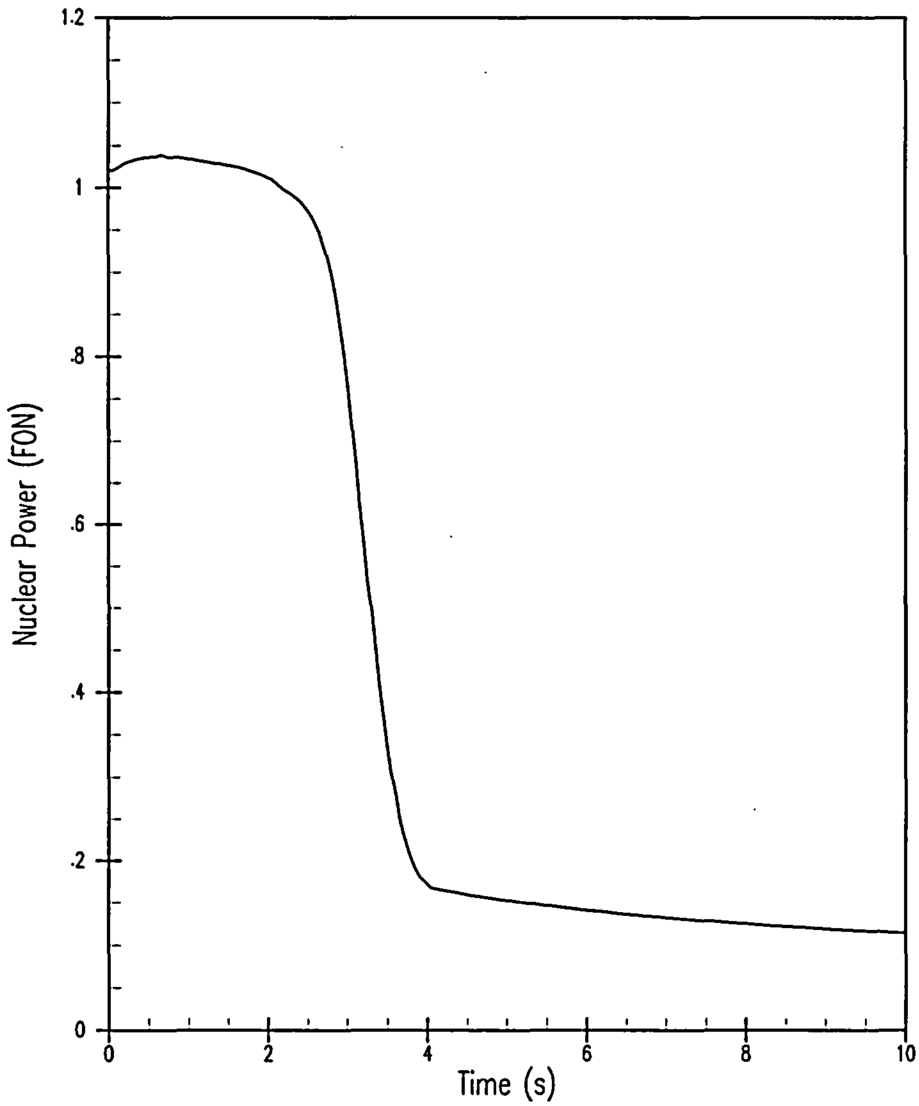

Figure 5.1.15-3 Nuclear Power versus Time – Locked Rotor/Shaft Break

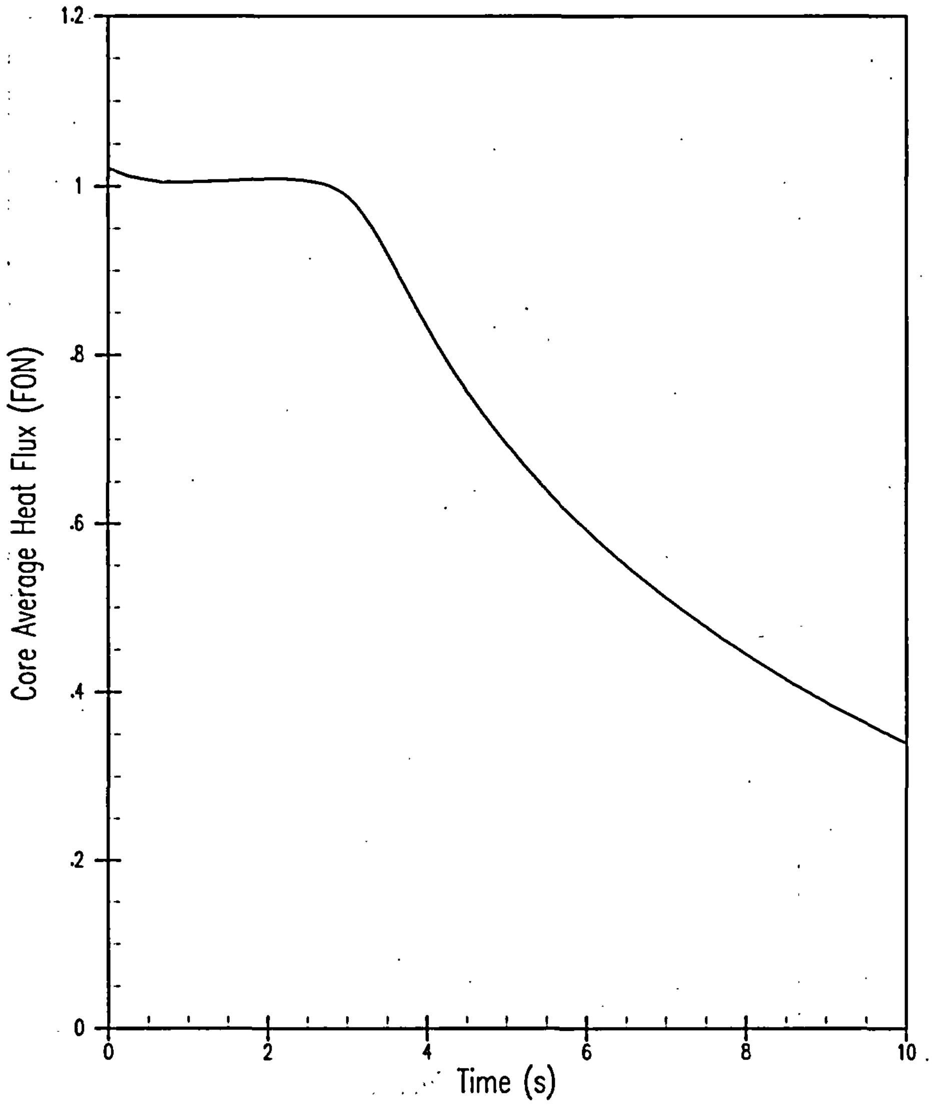

Figure 5.1.15-4 Core Average Heat Flux versus Time – Locked Rotor/Shaft Break

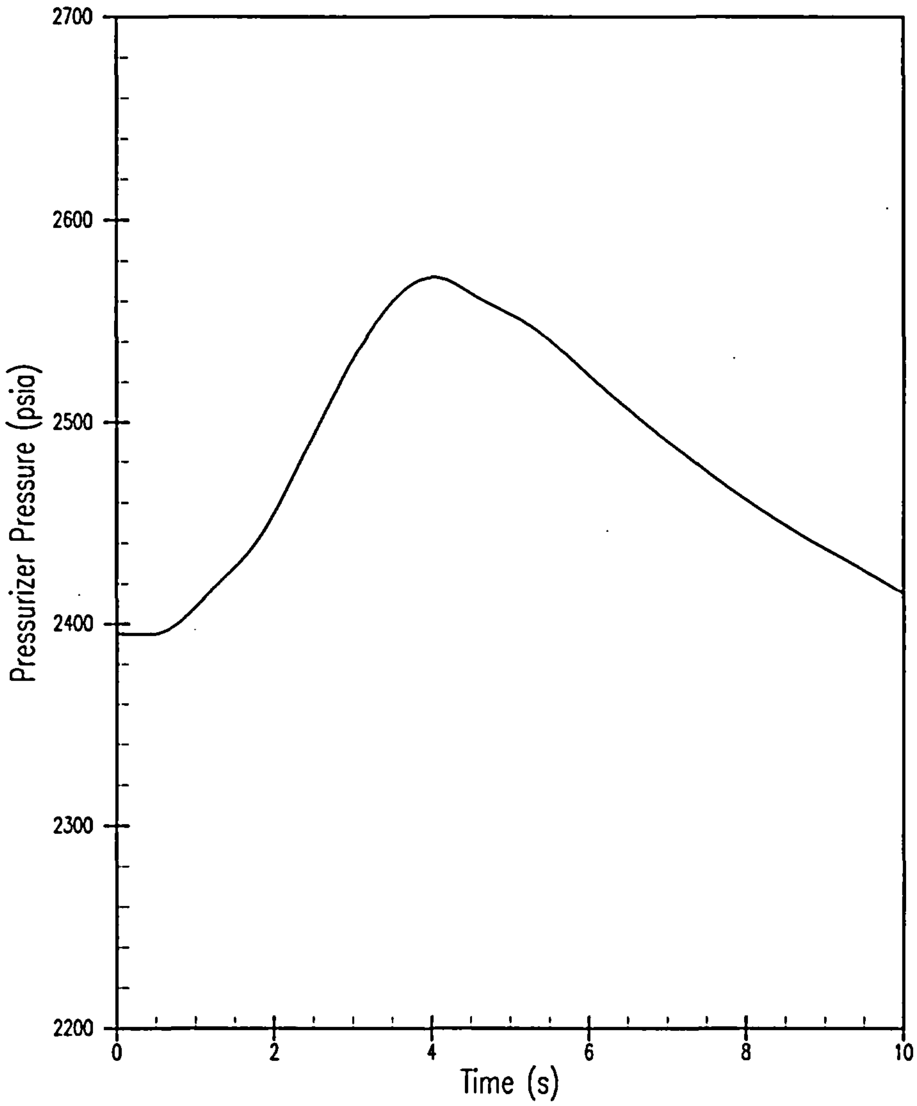

Figure 5.1.15-5 Pressurizer Pressure versus Time – Locked Rotor/Shaft Break

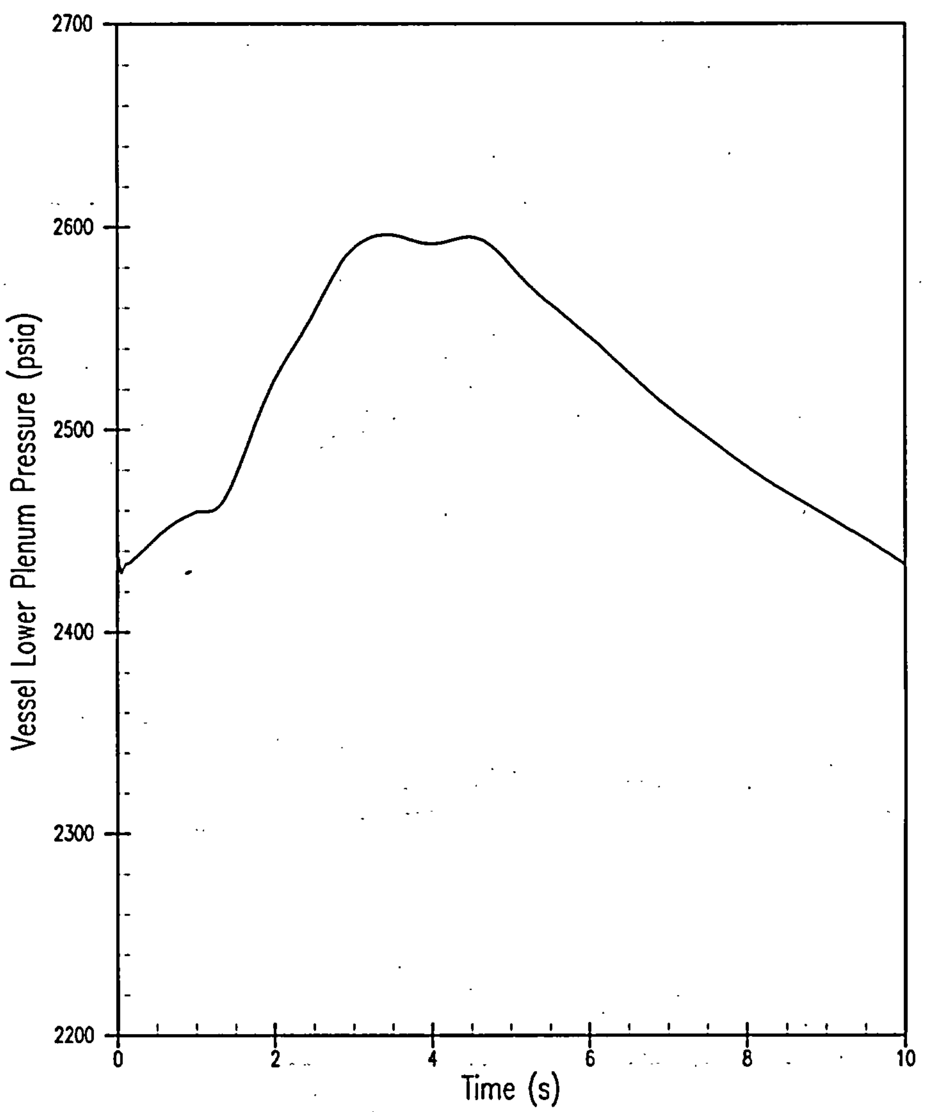

Figure 5.1.15-6 Vessel Lower Plenum Pressure versus Time – Locked Rotor/Shaft

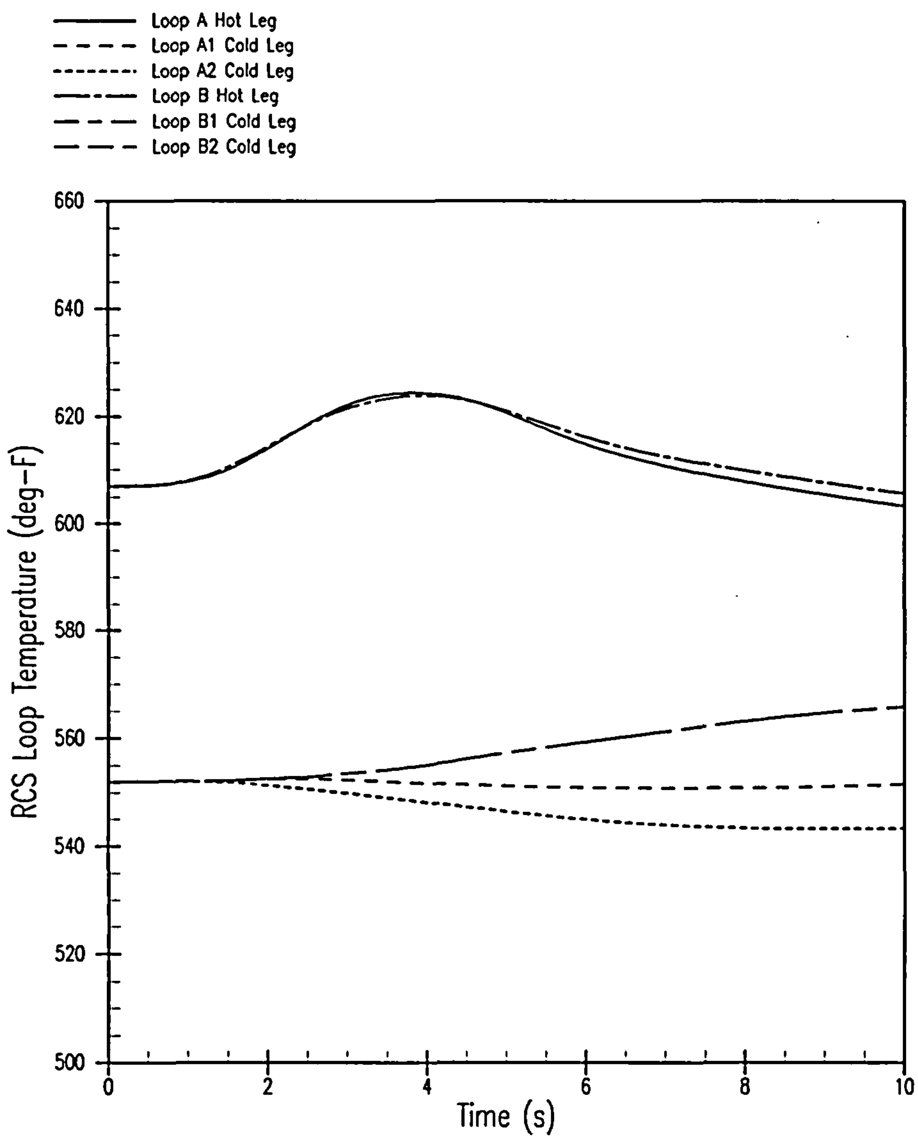

Figure 5.1.15-7 RCS Loop Temperature versus Time – Locked Rotor/Shaft Break

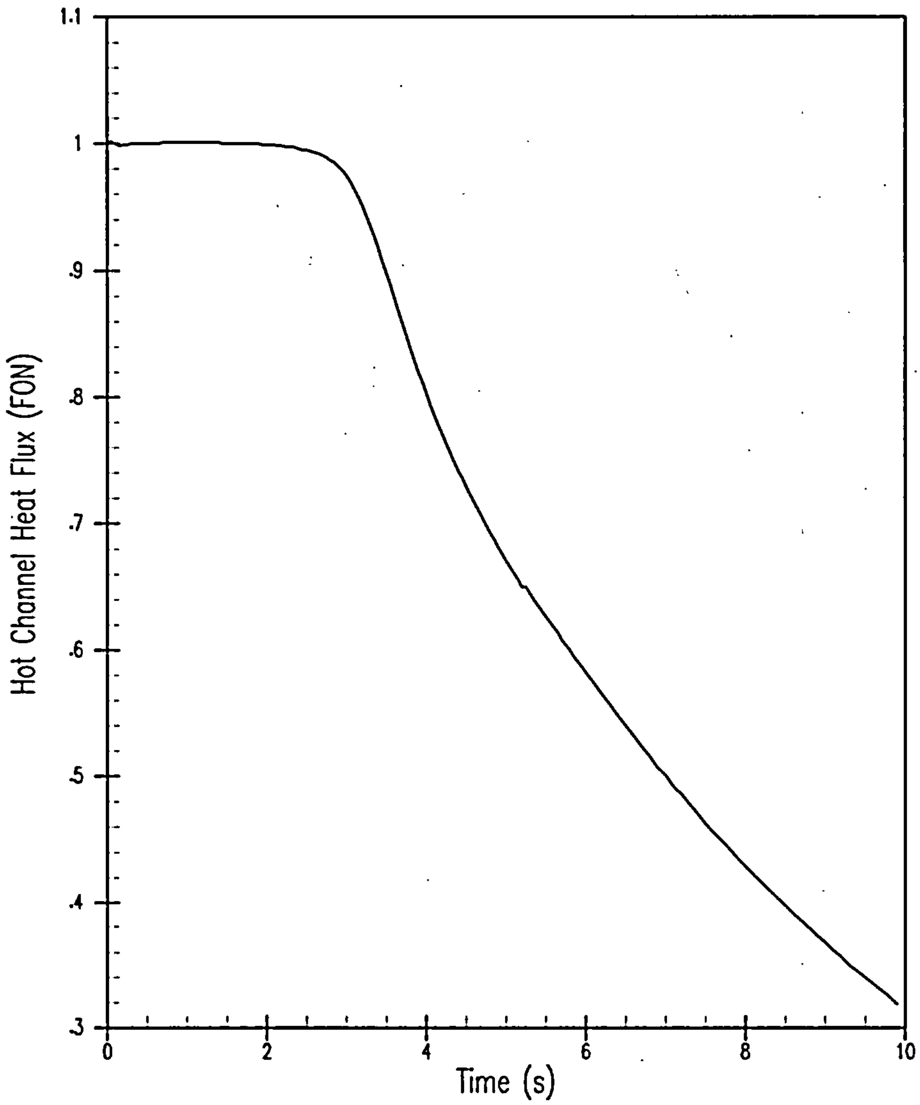

Figure 5.1.15-8 Hot Channel Heat Flux versus Time – Locked Rotor/Shaft Break

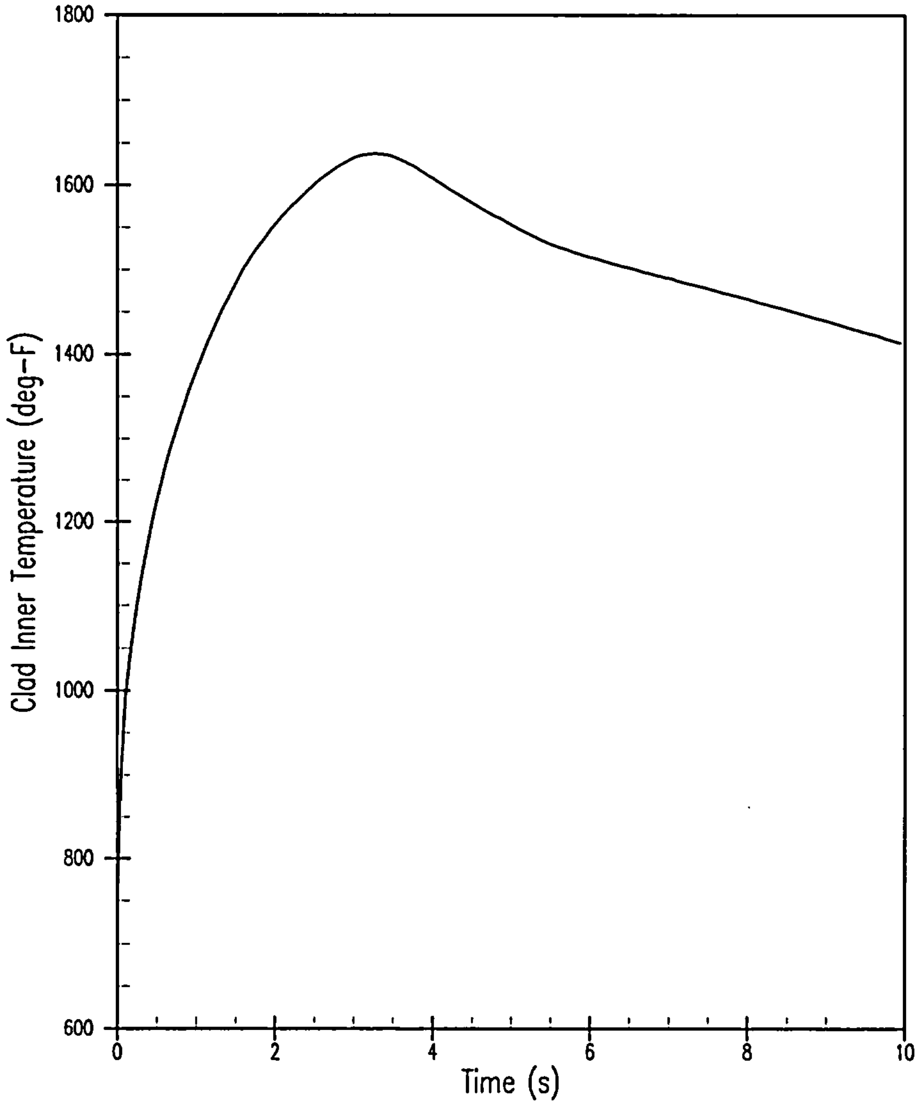

Figure 5.1.15-9 Hot-Spot Cladding Inner Temperature versus Time – Locked Rotor/Shaft Break

5.1.16 Uncontrolled Control Element Assembly Bank Withdrawal at Power

5.1.16.1 Accident Description

The Uncontrolled CEA Withdrawal At-Power event is defined as the inadvertent addition of reactivity to the core caused by the withdrawal of CEA banks when the core is above the no-load condition. The reactivity insertion resulting from the bank (or banks) withdrawal will cause an increase in core nuclear power and a subsequent increase in core heat flux and RCS temperature. A CEA bank withdrawal can occur with the reactor subcritical, at HZP, or at power. The uncontrolled CEA bank withdrawal at power event is analyzed for Mode 1 (power operation). The uncontrolled CEA bank withdrawal from a subcritical or low-power condition is considered as an independent event in Section 5.1.17.

The CEA Withdrawal At-Power event is simulated by modeling a constant reactivity insertion rate starting at time zero and continuing until an automatic reactor trip occurs or, for very low reactivity insertion rates, sufficient time has passed to credit a manual reactor trip. The analysis assumes a spectrum of possible reactivity insertion rates up to a maximum positive reactivity insertion rate greater than that occurring for the simultaneous withdrawal, at maximum speed, of two sequential CEA banks having the maximum differential rod worth.

The transient RCS response to the CEA Bank Withdrawal event is terminated by manual or automatic action to preclude the power mismatch and resultant temperature rise from resulting in DNB and/or fuel centerline melt. Additionally, the increase in RCS temperature caused by this event will increase the RCS pressure, and if left unchecked, could challenge the integrity of the RCS pressure boundary or the MSS pressure boundary.

To avert the core damage that might otherwise result from this event, the RPS is designed to automatically terminate any such event before the DNBR falls below the limit value, the fuel rod kW/ft limit is reached, or the peak primary and secondary pressures exceed their respective limits. Depending on the initial power level and the reactivity insertion rate, the reactor may be tripped and the CEA withdrawal terminated by any of the following trip signals:

- Variable High Power (VHP)
- High Pressurizer Pressure
- Thermal Margin/Low Pressure
- High Local Power Density

5.1.16.2 Method of Analysis

The Uncontrolled CEA Bank Withdrawal At-Power event is analyzed to show that: (1) the integrity of the core is maintained by the RPS because the DNBR and peak kW/ft remain within the safety analysis limit values and (2) the peak RCS and MS system pressures remain below 110 percent of the corresponding design limits. Of these criteria, the most limiting are the need to ensure that the DNBR and peak kW/ft limits are met.

The CEA Bank Withdrawal At-Power transient is analyzed with the RETRAN computer program (Reference 1). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator relief and safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

Selected initial conditions used in the safety analysis are summarized in Table 5.1.0-2. To obtain conservative values for minimum DNBR and ensure that the peak kW/ft limits are met, the following analysis assumptions are made:

1. Cases analyzed to assess the acceptability with respect to DNBR limits are analyzed with the RTDP DNB methodology (Reference 2). Therefore, the initial reactor power, RCS pressure, RCS flow and RCS temperatures are assumed to be at their nominal values. Uncertainties in these initial conditions are included in the limit DNBR.

2. Reactivity coefficients - Two feedback conditions are analyzed:

    a. Minimum reactivity feedback - A zero MTC of reactivity (0 pcm/°F) is assumed at full power. For power levels less than or equal to 70-percent power, a positive MTC of reactivity (+5 pcm/°F) is conservatively assumed, corresponding to the beginning of core life. A conservatively small (in absolute magnitude) Doppler power coefficient is used in the analysis (see Figure 5.1.0-5).

    b. Maximum reactivity feedback - A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed (Figure 5.1.0-5).

3. The variable high power reactor trip on high neutron flux or high indicated power from temperature measurement (ΔT-power) is actuated at a conservative safety analysis ceiling value of 112.2 percent of nominal full power. The decalibration of the excore detectors is conservatively modeled as the CEAs are withdrawn to account for the effect of the reduced indicated excore detector power which delays the reactor trip on the neutron flux signal.

4. The Δ-power feature of the variable high power trip is simulated in the analysis using two different approaches:

    a. For the cases initiated from less than full power (20, 50, and 65% of nominal full power) a conservative safety analysis setpoint of 30% of nominal full power on high neutron flux or high indicated ΔT-power is modeled. This 30% Δ-power trip setpoint includes setpoint uncertainties, power measurement uncertainties and accounts for excore decalibration due to CEA withdrawal.

    b. For the cases initiated from 100% power, a Δ-power trip setpoint of 11.0% of nominal full power based on the highest of the excore or ΔT-power signals is modeled. Decalibration of the excore detectors as the CEAs withdraw is explicitly modeled since this effect tends to reduce the indicated excore detector power and thereby delay the reactor trip.

5. In all cases, the thermal margin/low pressure trip is modeled without taking credit for any reduction in the calculated trip setpoint pressure associated with any skewed axial shape index

(ASI). Two different approaches are employed with regard to the indicated core power signal that is used by the thermal margin/low pressure function. The initial power level dictates the approach selected for any particular case.

a. For the cases initiated from less than full power (20, 50, and 65% of nominal full power), excore decalibration due to CEA withdrawal was not modeled. No such modeling was required because all of these cases produced an earlier reactor trip on signals other than the thermal margin/low pressure trip function.

b. For cases initiated from 100% power, the highest of the excore or ΔT-power signals is used by the thermal margin/low pressure trip logic. The decalibration of the excore detectors is conservatively modeled as the CEAs are withdrawn to account for the effect of the reduced indicated excore detector power which delays the reactor trip on the neutron flux signal input to the TM/LP.

6. The high pressurizer pressure reactor trip setpoint assumed in the safety analysis is 2415 psia.

7. All of the reactor trip functions modeled in the CEA withdrawal at-power analysis include appropriate instrumentation and setpoint errors. The delays for trip actuation are assumed to be the maximum values (see Table 5.1.0-4).

8. The CEA trip insertion characteristics are based on the assumption that the highest worth assembly is stuck in its fully withdrawn position (see Figure 5.1.0-4).

9. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined differential rod worth at a conservative speed. The maximum bounding reactivity insertion rate that is required for consideration is 53 pcm/sec. However, the plots associated with Figure 5.1.16-3 include results out to 60 pcm/sec.

10. The analysis includes consideration of up to 30% steam generator tube plugging in both steam generators.

11. Power levels of 20, 50, 65, and 100 percent of full power are considered.

5.1.16.3 Results

Selected results are reported in Table 5.1.16-1 for the CEA bank withdrawal at power transients analyzed to support 30% steam generator tube plugging and implementation of reload methodology in accordance with WCAP-9272 (Reference 3). The limiting results are summarized in Table 5.1.16-2.

For all of the cases analyzed, including reactivity insertion rates of up to 53 pcm/second, the RCS pressure never exceeds the 110% of design limit of 2750 psia. The reactivity insertion rate of 53 pcm/sec bounds that calculated for the simultaneous withdrawal, at maximum speed, of two sequential CEA banks having the maximum differential rod worth.

Figure 5.1.16-1 shows the response of nuclear power, pressurizer pressure, RCS vessel T_{avg}, and DNBR to a rapid (53 pcm/sec) CEA withdrawal incident starting from full power with minimum reactivity feedback conditions. Reactor trip on the variable high power function occurs shortly after the start of the accident. Since this case results in a rapid increase in the nuclear power with the core heat flux lagging behind because of the thermal time constants of the plant, small changes in the reactor core T_{avg} and pressurizer pressure result. Therefore, a large margin to the safety analysis limit DNBR is maintained.

The response of nuclear power, pressurizer pressure, RCS vessel T_{avg}, and DNBR for a slow (2 pcm/sec) CEA withdrawal from 100% power with minimum reactivity feedback conditions is shown in Figure 5.1.16-2. Reactor trip on the variable high power function occurs after a longer period of time compared to the rapid CEA withdrawal mentioned above and thus, the rise in temperature is consequently larger. Again, the minimum DNBR is greater than the safety analysis limit value.

Figure 5.1.16-3 shows the minimum predicted DNBR as a function of the reactivity insertion rate for the four initial power levels analyzed (100%, 65%, 50% and 20%) for both minimum and maximum reactivity feedback conditions. It can be seen that the combination of reactor trip functions modeled provides protection over the entire range of reactivity insertion rates because the minimum DNBR is never less than the safety analysis limit value.

In the referenced figures, the shape of the curves of minimum DNBR versus reactivity insertion rate is a function of both the reactor core and coolant system transient response and the reactor trips assumed to provide protection for this event.

Referring to Figure 5.1.16-3 (Sheet 1) for example, it is noted that:

1. For every minimum reactivity feedback cases initiated from 100% power, reactor trip was on the variable high power trip. For these cases, even with excore decalibration effects modeled, the variable high power reactor trip on the excore detector indicated power signal produces the reactor trip. This demonstrates that even for cases with low reactivity insertion rates, in the presence of a minimum reactivity feedback condition, the neutron flux level in the core rises relatively rapidly compared to coolant temperature changes that lag behind due to the thermal capacity of the fuel and coolant system fluid. For the full power minimum feedback cases over the entire range of reactivity insertion rates considered, the minimum predicted DNBR values remain well above the safety analysis limit value of 1.42.

2. When modeling maximum reactivity feedback for cases initiated from 100% power, protection is provided by a combination of the thermal margin/low pressure, high pressurizer pressure and variable high power reactor trip functions. For the range of low reactivity insertion rates (from 1 to 12 pcm/sec), reactor trip occurs on the thermal margin/low pressure reactor trip. With these relatively low reactivity insertion rates and maximum reactivity feedback, the core power increase is limited. However, as the reactivity insertion continues, the RCS temperature rises until a thermal margin/low pressure reactor trip is generated. For intermediate rates of reactivity insertion (from 13 to 30 pcm/sec) a reactor trip occurs on the high pressurizer pressure reactor trip function. Finally, for the cases with higher reactivity insertion rates (40 to 53 pcm/sec) the resulting rate of increase in the neutron flux is rapid and produces a reactor trip on a variable high

power signal. The minimum DNBR values predicted for all the maximum feedback cases for the entire range of reactivity insertions are all well above the safety analysis limit value of 1.42.

5.1.16.4 Conclusions

The results for the uncontrolled CEA Bank Withdrawal At-Power transient demonstrate that the combination of the variable high power, thermal margin/low pressure, and high pressurizer pressure reactor trip functions provide adequate protection over the entire range of possible reactivity insertion rates, expected initial power levels and for different times in life. That is, the minimum calculated DNBR is always greater than the safety analysis limit value. In addition, it was demonstrated that the peak kW/ft is less than the limit value for fuel melting and that the peak pressures in the RCS and secondary steam system do not exceed 110 percent of their respective design pressures.

Thus, all pertinent safety analysis criteria are met for the uncontrolled CEA bank withdrawal at power event in support of the 30% steam generator tube plugging program and the implementation of Westinghouse reload methodology as defined by WCAP-9272.

5.1.16.5 References

1. McFadden, J. H., et al. "RETRAN-02- A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCMA.

2. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-Proprietary), April 1989.

3. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

Table 5.1.16-1 Time Sequence of Events for Uncontrolled CEA Withdrawal at Power (100% Initial Power & Minimum Reactivity Feedback)	
Event	**Time (Seconds)**
Case A:	
Initiation of Uncontrolled CEA Withdrawal at Full Power with Minimum Reactivity Feedback (53 pcm/sec)	0
Variable High Power Trip Setpoint Reached	1.96
Rods Begin to Fall into Core	3.10
Minimum DNBR Occurs	4.0
Case B:	
Initiation of Uncontrolled CEA Withdrawal at Full Power with Minimum Reactivity Feedback (2 pcm/sec)	0
Variable High Power Trip Setpoint Reached	42.54
Rods Begin to Fall into Core	43.68
Minimum DNBR occurs	44.1

Table 5.1.16-2 Limiting Results for CEA Bank Withdrawal at Power Transient			
Criterion	**Limiting Value**	**Analysis Limit**	**Case**
DNBR	1.50	1.42	65% power, maximum reactivity feedback, 2 pcm/second reactivity insertion rate
Core Heat Flux (FON)	1.18	1.20	100% power, maximum reactivity feedback 40 pcm/second reactivity insertion rate
MSS Pressure (psia)	1085.4	1100.0	100% power, maximum reactivity feedback, 14 pcm/second reactivity insertion rate

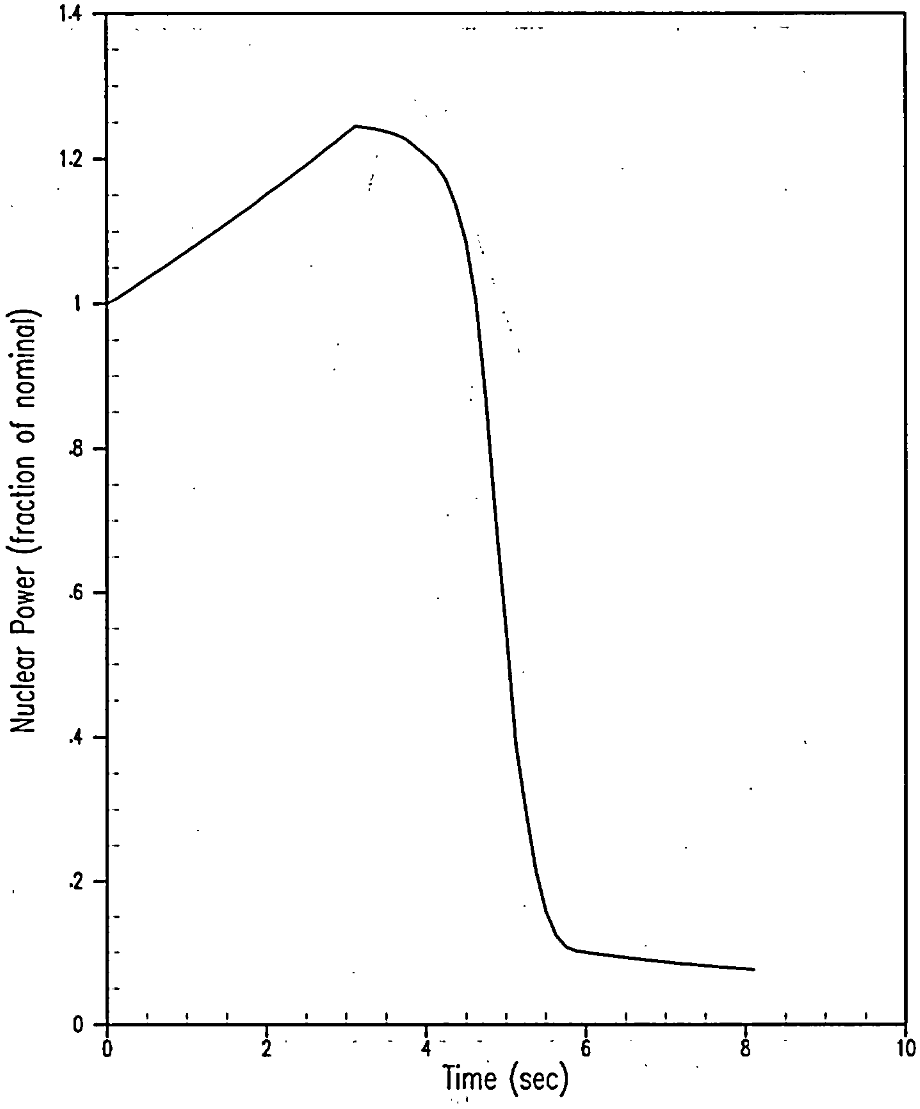

Figure 5.1.16-1 Uncontrolled CEA Bank Withdrawal at Power 100% Power, 53 pcm/sec Reactivity Insertion Rate & Minimum Feedback Variable High Power Trip & Maximum Nominal RCS Vessel T_{avg} (Sheet 1 of 4)

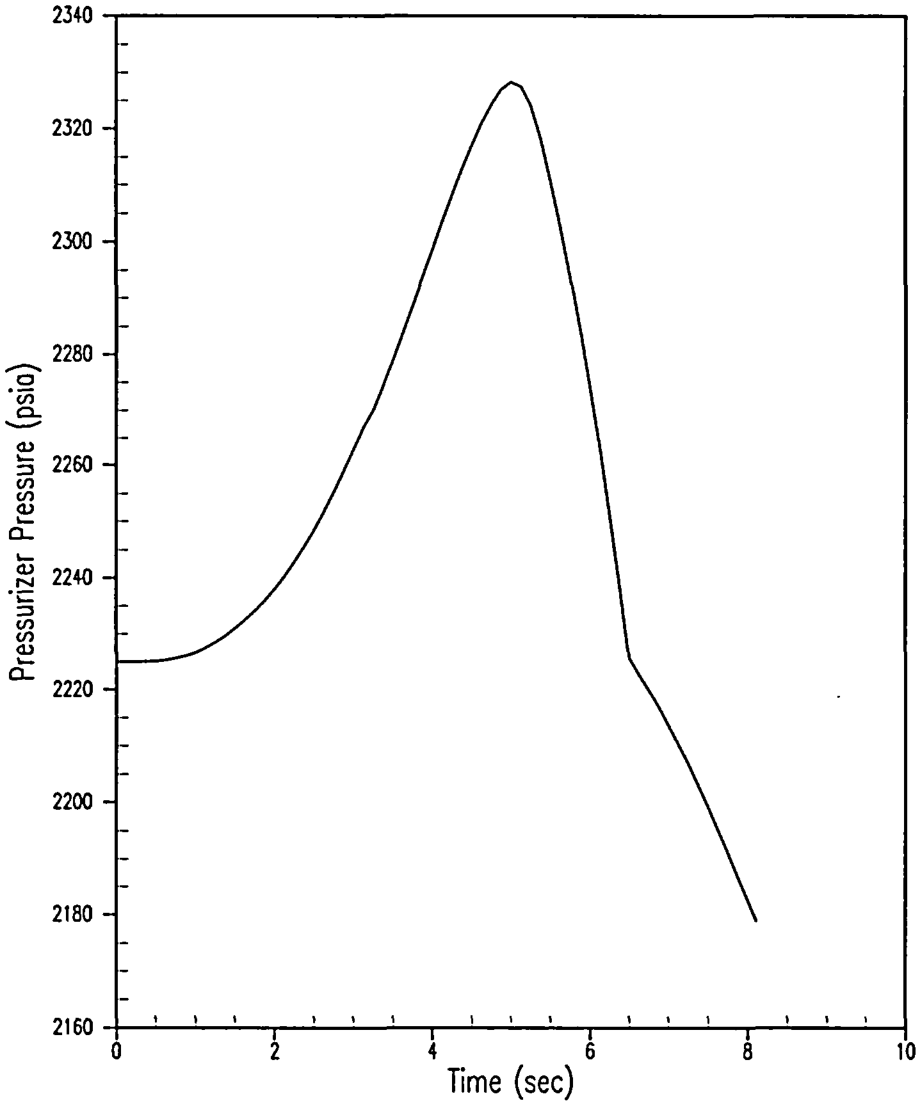

Figure 5.1.16-1 Uncontrolled CEA Bank Withdrawal at Power 100% Power, 53 pcm/sec Reactivity Insertion Rate & Minimum Feedback Variable High Power Trip & Maximum Nominal RCS Vessel T_{avg} (Sheet 2 of 4)

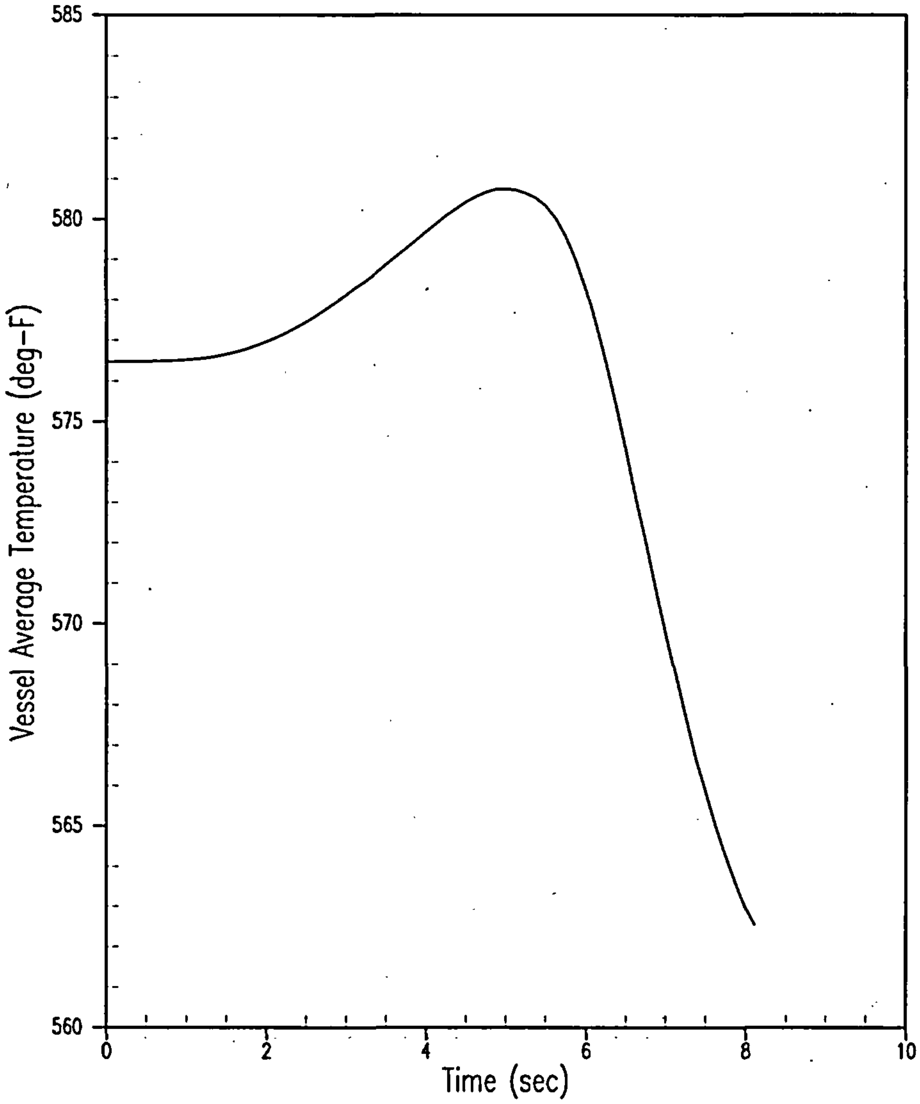

Figure 5.1.16-1 Uncontrolled CEA Bank Withdrawal at Power 100% Power, 53 pcm/sec Reactivity Insertion Rate & Minimum Feedback Variable High Power Trip & Maximum Nominal RCS Vessel T_{avg} (Sheet 3 of 4)

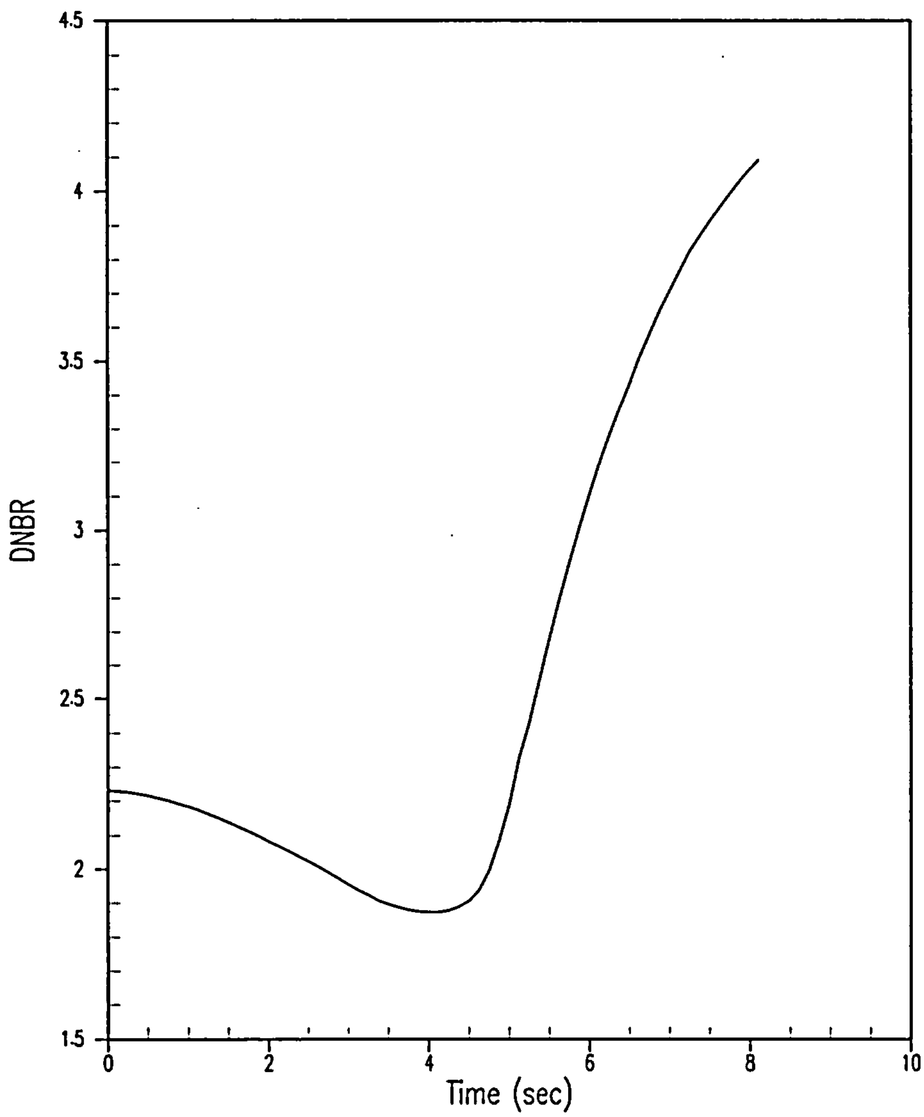

Figure 5.1.16-1 Uncontrolled CEA Bank Withdrawal at Power 100% Power, 53 pcm/sec Reactivity Insertion Rate & Minimum Feedback Variable High Power Trip & Maximum Nominal RCS Vessel T_{avg} (Sheet 4 of 4)

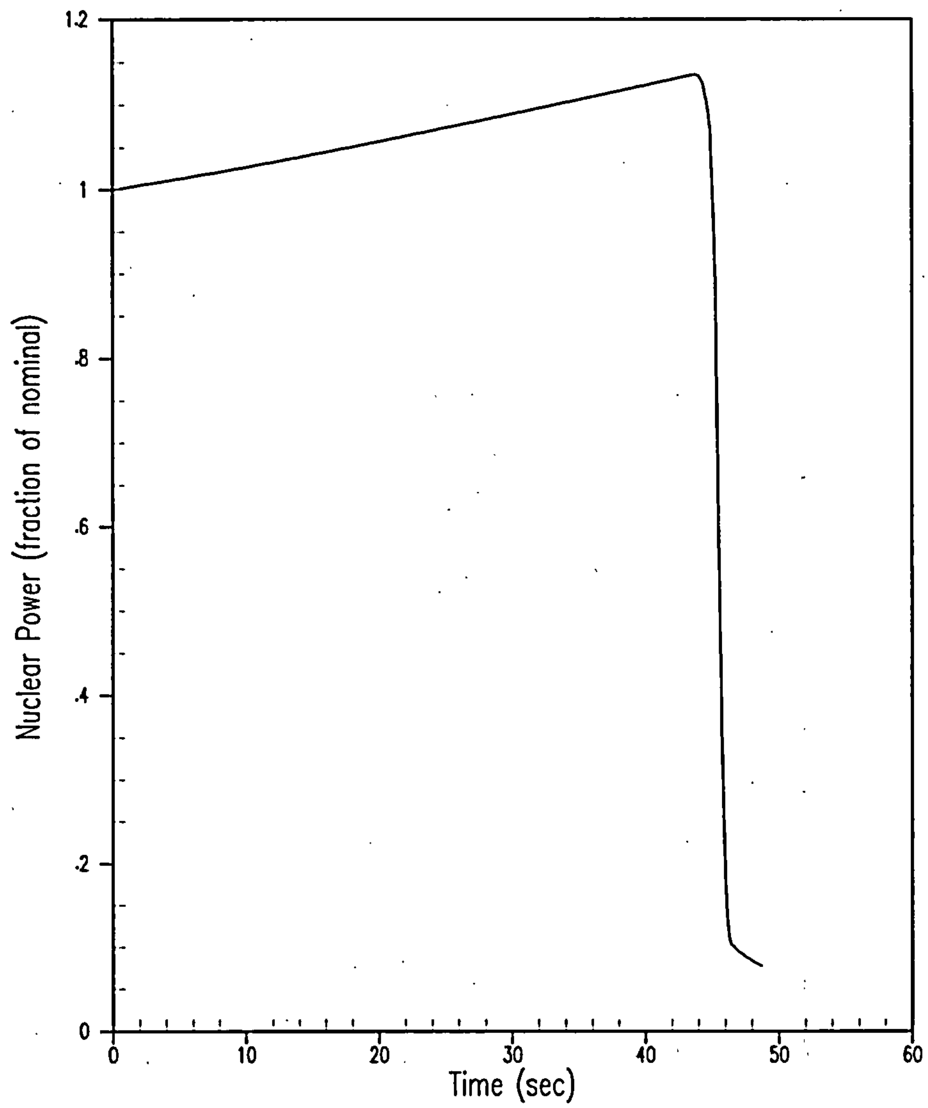

Figure 5.1.16-2 Uncontrolled CEA Bank Withdrawal at Power 100% Power, 2 pcm/sec Reactivity Insertion Rate & Minimum Feedback Variable High Power Trip & Maximum Nominal RCS Vessel T_{avg} (Sheet 1 of 4)

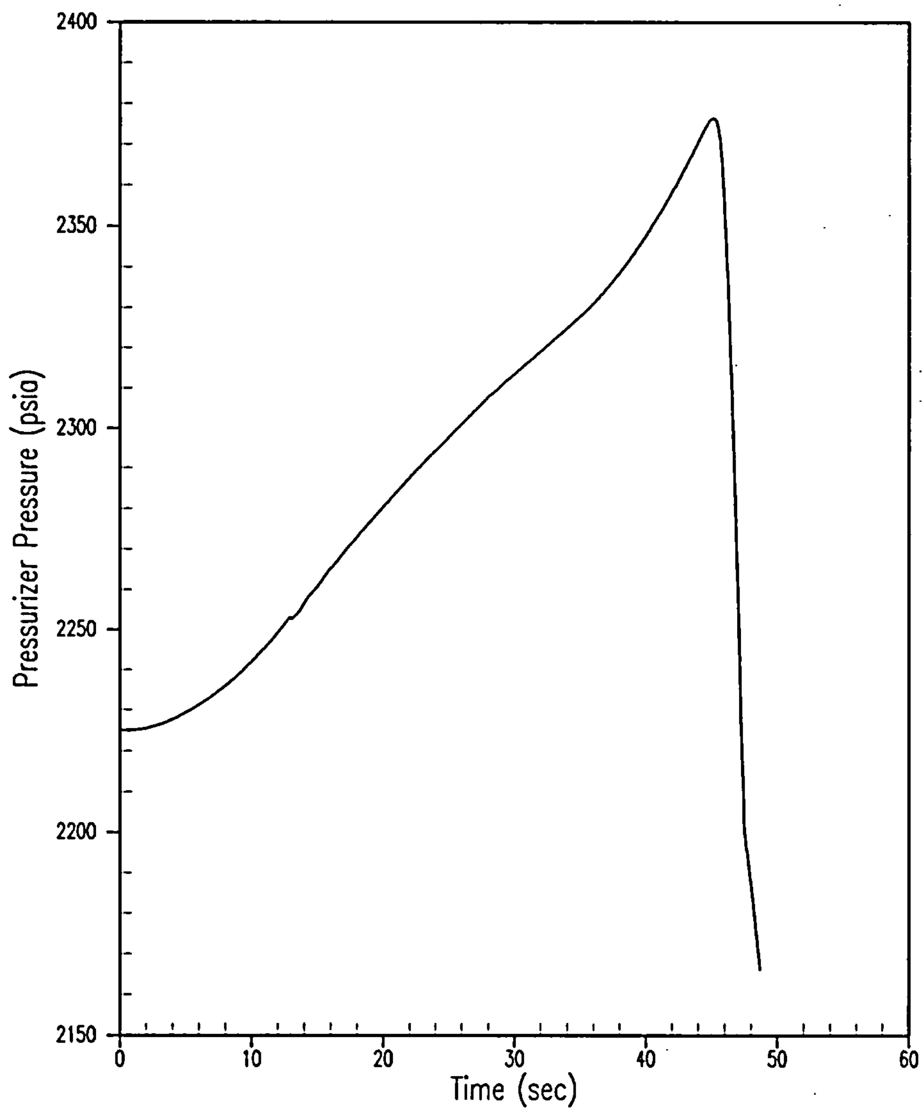

Figure 5.1.16-2 Uncontrolled CEA Bank Withdrawal at Power 100% Power, 2 pcm/sec Reactivity Insertion Rate & Minimum Feedback Variable High Power Trip & Maximum Nominal RCS Vessel T_{avg} (Sheet 2 of 4)

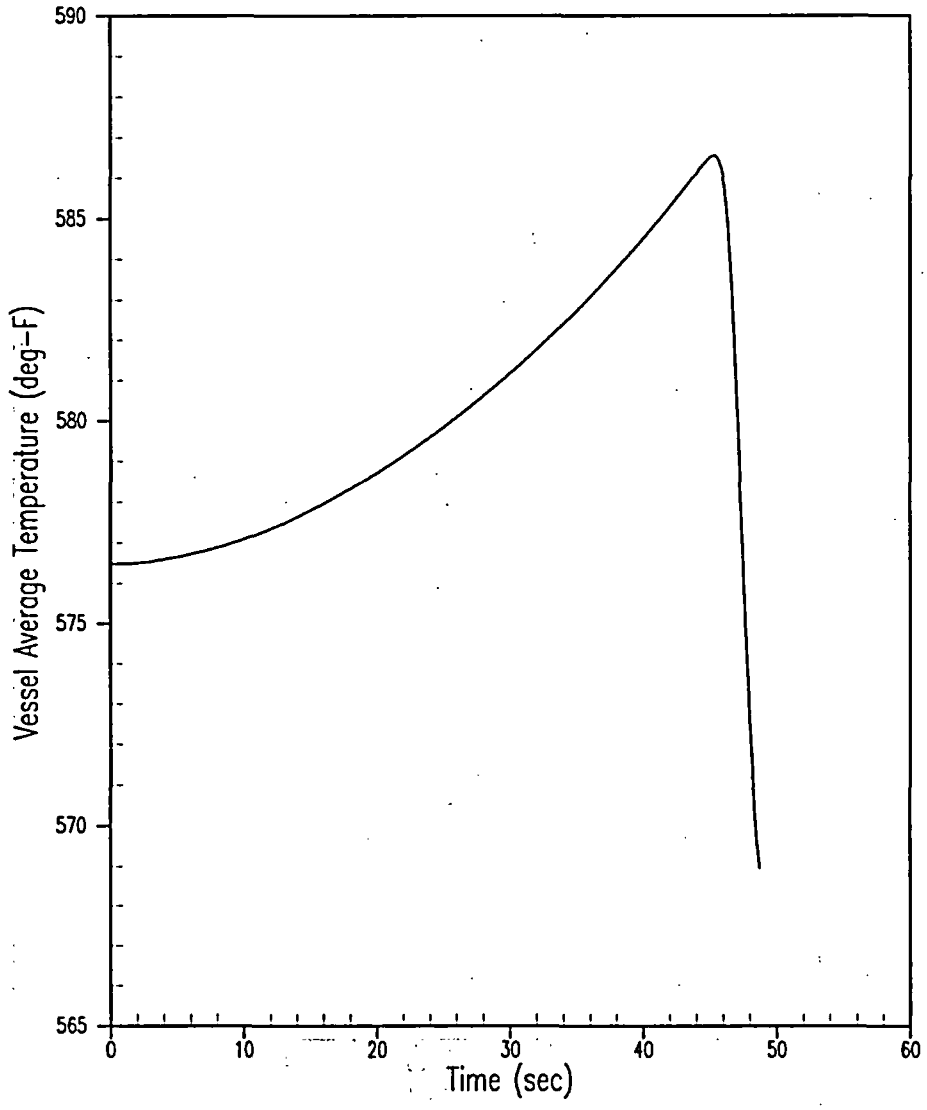

Figure 5.1.16-2 Uncontrolled CEA Bank Withdrawal at Power 100% Power, 2pcm/sec Reactivity Insertion Rate & Minimum Feedback Variable High Power Trip & Maximum Nominal RCS Vessel T_{avg} (Sheet 3 of 4)

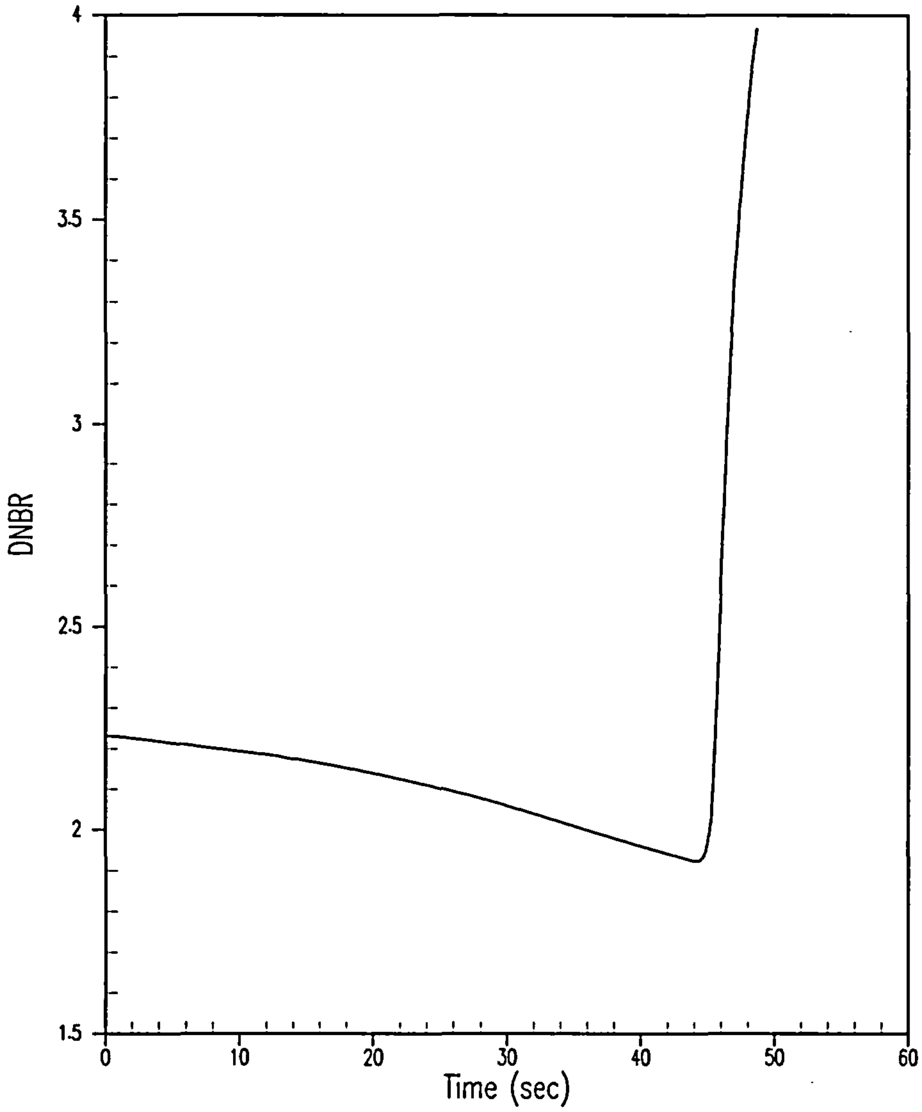

Figure 5.1.16-2 Uncontrolled CEA Bank Withdrawal at Power 100% Power, 2 pcm/sec Reactivity Insertion Rate & Minimum Feedback Variable High Power Trip & Maximum Nominal RCS Vessel T_{avg} (Sheet 4 of 4)

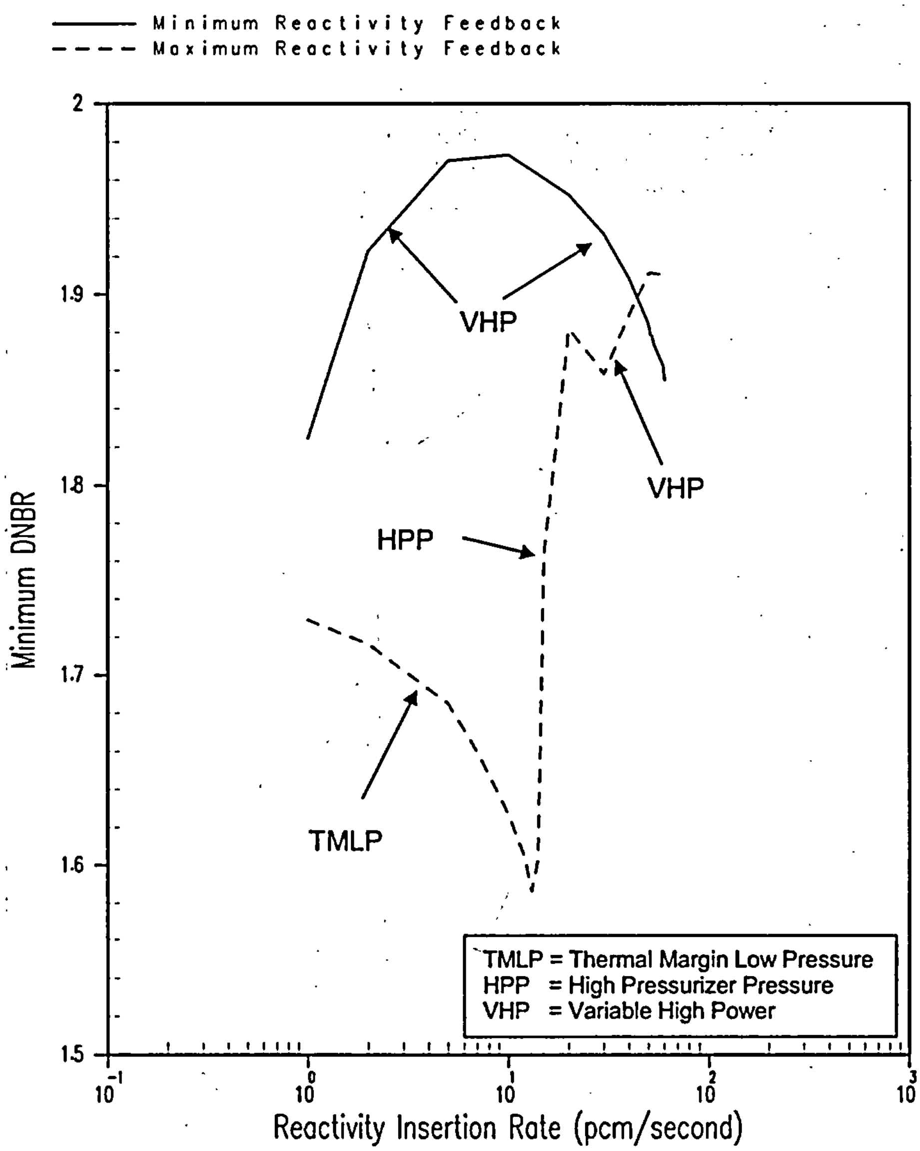

Figure 5.1.16-3 Uncontrolled CEA Bank Withdrawal at Power Initial Power Level of 100% (Sheet 1 of 4)

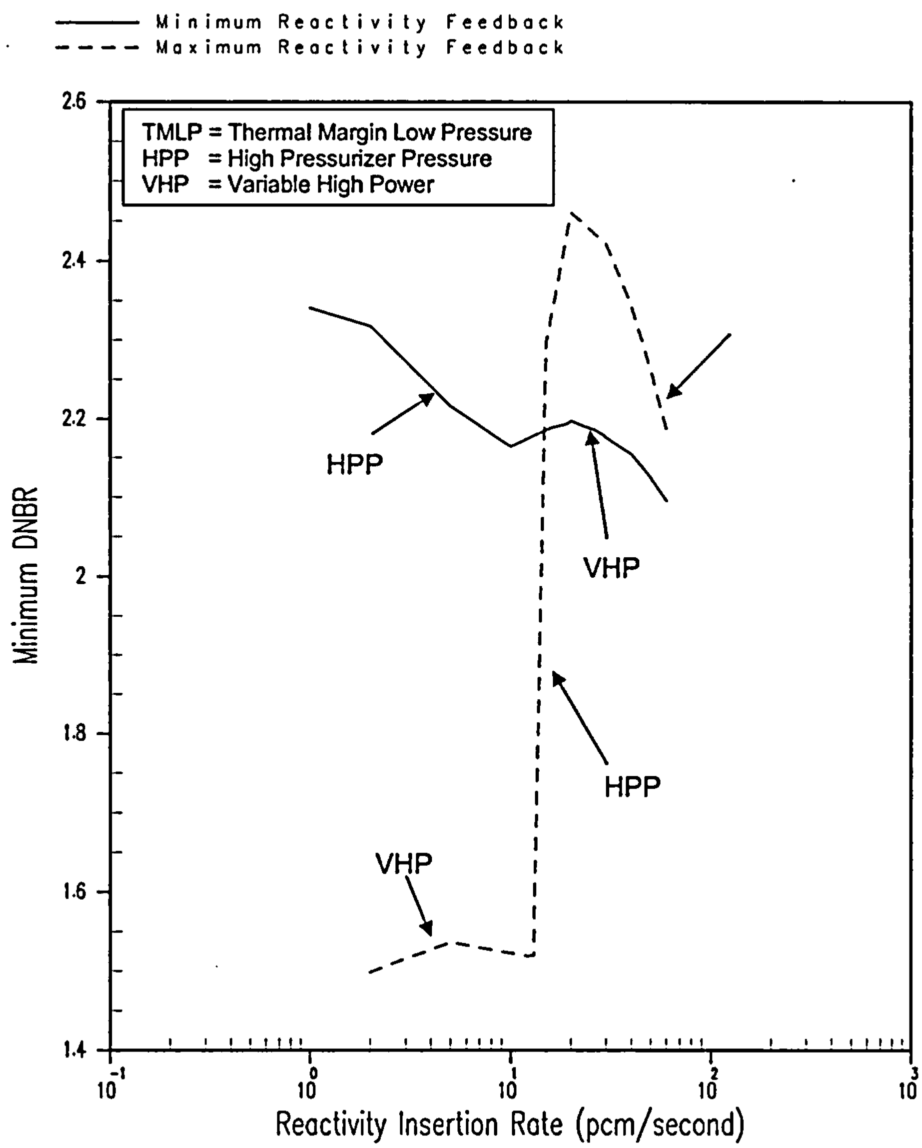

Figure 5.1.16-3 Uncontrolled CEA Bank Withdrawal at Power Initial Power Level of 65% (Sheet 2 of 4)

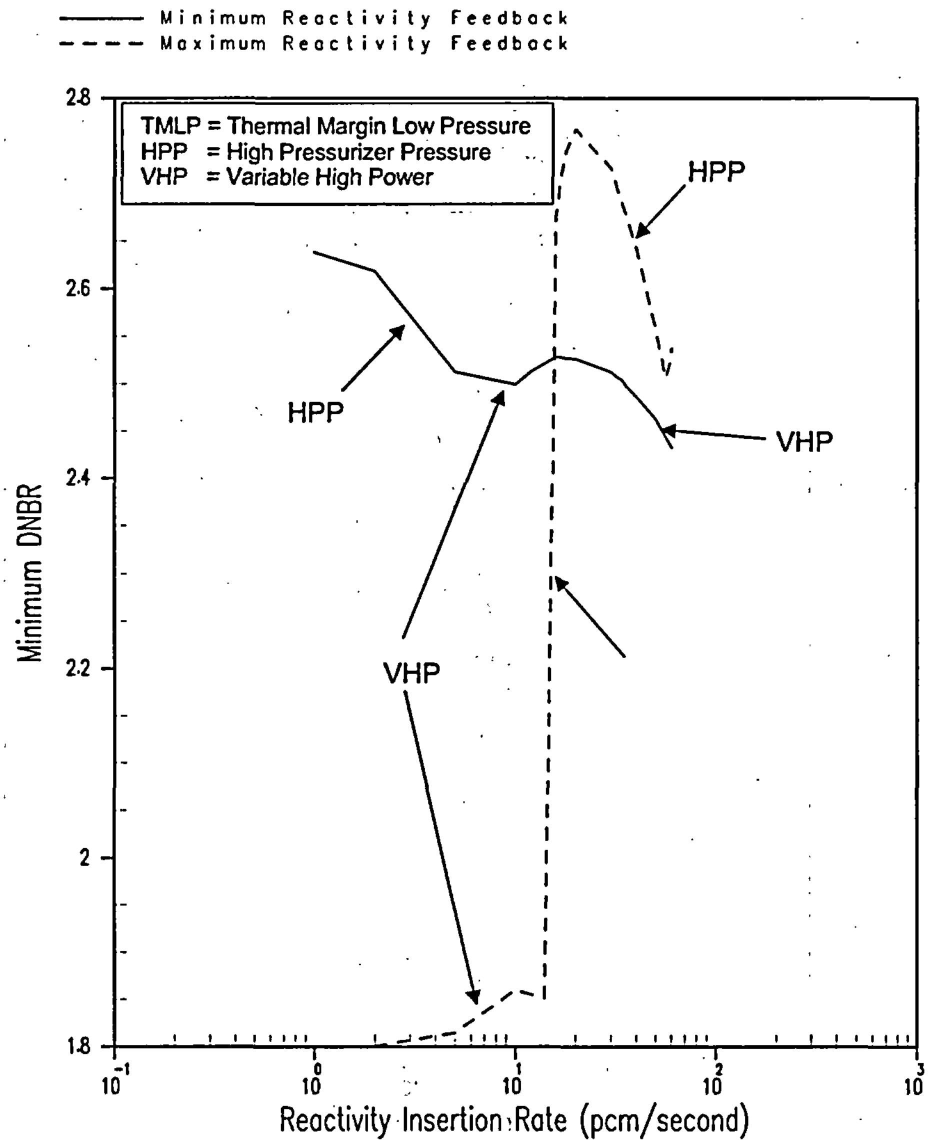

Figure 5.1.16-3 Uncontrolled CEA Bank Withdrawal at Power Initial Power Level of 50% (Sheet 3 of 4)

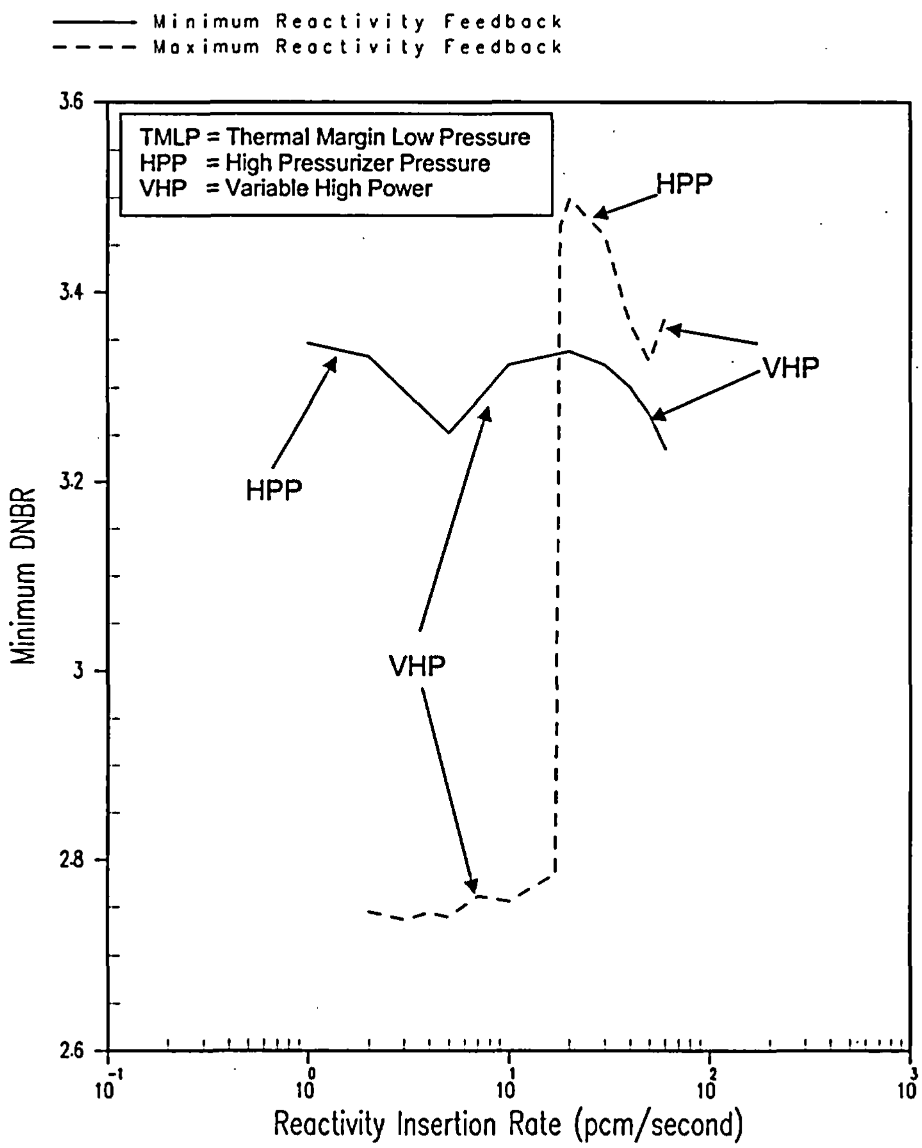

Figure 5.1.16-3 Uncontrolled CEA Bank Withdrawal at Power Initial Power Level of 20% (Sheet 4 of 4)

5.1.17 Uncontrolled CEA Withdrawal from a Subcritical Condition

5.1.17.1 Accident Description

The CEA withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of CEA banks resulting in a power excursion. While the occurrence of a transient of this type is unlikely, such a transient could be caused by a malfunction of the reactor control or the control element drive system. This could occur with the reactor either subcritical, at HZP, or at power. The "at power" case is discussed in Section 5.1.16.

Withdrawal of a CEA bank adds reactivity at a prescribed and controlled rate to bring the reactor from a subcritical condition to a low-power level during startup. Although the initial startup procedure typically uses the method of boron dilution, the normal startup is with CEA bank withdrawal. A CEA bank movement can cause much faster changes in reactivity than can be made by changing boron concentration (see Section 5.1.19, Chemical and Volume Control System Malfunction).

The neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous control element assembly withdrawal event occur, the following automatic features of the reactor protection system are available to terminate the transient:

- The Variable Power Level – High trip is provided to trip the reactor when the reactor power reaches a high preset value. This setpoint is set to a fixed increment (≤ 9.61% technical specification value) above the existing reactor power level, with a minimum setpoint of 15% of rated thermal power and a maximum of ≤ 107% of rated thermal power. This trip is actuated when two-out-of-four power range channels indicate a power level above the setpoint.

- The Rate-of-Change of Power – High trip is provided to trip the reactor when the rate-of-change of neutron flux power reaches a high preset value (≤ 2.49 decades per minute technical specification value). It is actuated when two-out-of-four wide-range logarithmic neutron flux monitoring channels indicate a rate above the preset setpoint. This trip function may be bypassed below 10^{-4}% and above 15% of rated thermal power. Bypass is automatically removed when wide-range logarithmic neutron flux power is ≥ 10^{-4}% and power range neutron flux power is ≤ 15% of rated thermal power.

5.1.17.2 Method of Analysis

The analysis of the uncontrolled CEA bank withdrawal from subcritical accident is performed in three stages. First, the spatial neutron kinetics computer code TWINKLE (Reference 1) is used to calculate the core average nuclear power transient including the various core feedback effects; that is, Doppler and moderator reactivity. FACTRAN (Reference 2) uses the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine the average heat flux and temperature transients. Finally, the peak core-average heat flux calculated by FACTRAN is used in VIPRE for transient DNBR calculations.

In order to give conservative results for a startup accident, the following assumptions are made:

1. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler power defect, a conservatively low (absolute magnitude) value is used (900 pcm).

2. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the neutron flux response time constant. However, after the initial neutron flux peak, the MTC can affect the succeeding rate of power increase. The effect of moderator temperature changes on the rate of nuclear power increase is calculated in TWINKLE based on temperature-dependent moderator cross-sections. The MTC value used in this event analysis is + 5 pcm/°F at HZP.

3. The analysis assumes the reactor to be at HZP nominal temperature of 532°F. This assumption is more conservative than that of a lower initial system temperature (that is, shutdown conditions). The higher initial system temperature yields a larger fuel-to-water heat transfer coefficient, a larger specific heat of the water and fuel, and a less negative (smaller absolute magnitude) DPC. The less negative DPC reduces the Doppler feedback effect, thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel-specific heat and larger heat transfer coefficient yields a larger peak heat flux. The analysis assumes the initial effective multiplication factor (k_{eff}) to be 1.0 since this results in the maximum neutron flux peak.

4. Reactor trip is initiated by the Variable High Power trip at a conservative trip setpoint of 35%. This increase from the nominal setpoint of 15 percent accounts for uncertainties. Figure 5.1.17-1 shows that the rise in nuclear flux is so rapid that the effect of error in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the total reactor trip reactivity is based on the assumption that the highest worth CEA is stuck in its fully withdrawn position. Further, the delays for trip signal actuation and CEA release are accounted for in the analysis.

5. A very conservative maximum positive reactivity insertion rate of 53 pcm/sec was assumed, which is greater than that for the simultaneous withdrawal of the two sequential CEA banks having the greatest combined worth at the maximum speed (30 in/min). This is confirmed for each reload cycle.

6. The DNB analysis assumes the most-limiting axial and radial power shapes possible during the fuel cycle associated with having the two highest combined worth banks in their high worth position.

7. The analysis assumes the initial power level to be below the power level expected for any shutdown condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

8. The accident analysis employs the Standard Thermal Design Procedure (STDP) methodology. Use of the STDP stipulates that the RCS flow rate will be based on the thermal design flow (TDF) and that the RCS pressure is the nominal pressure minus the uncertainty. Since the event

is analyzed from HZP, the steady-state STDP uncertainties on core power and RCS average temperature are not used in defining the initial conditions.

9. The fuel rod heat transfer calculations performed to determine the maximum fuel temperature during this event assume a total peaking factor or hot channel factor, FQ, that is a function of the axial and radial power distributions. The conservatively high value used in this analysis is presented in Table 5.1.17-1.

10. Both UO_2-only fuel and fuel with up to 8 weight-percent (w/o) Gadolinia content were considered in the transient analysis.

5.1.17.3 Results

Figures 5.1.17-1 through 5.1.17-5 show the transient behavior for a reactivity insertion rate of 53 pcm/sec, with the accident terminated by the reactor trip at 35 percent of nominal power. The rate is greater than that calculated for the two highest worth sequential control banks, with both assumed to be in their highest incremental worth region.

Figure 5.1.17-1 shows the neutron flux transient. The neutron flux overshoots the full-power nominal value for a very short period of time. Therefore, the energy release and fuel temperature increase are relatively small. The heat flux response of interest for the DNB considerations is shown in Figure 5.1.17-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux of much less than the nominal full power value. Figures 5.1.17-3 through 5.1.17-5 show the transient response of the hot-spot fuel centerline, fuel average, and cladding temperatures, respectively. DNBR calculations indicate that the minimum DNBR remains above the safety analysis limit value at all times.

Table 5.1.17-1 presents the assumptions and results of the analysis. Table 5.1.17-2 presents the calculated sequence of events. After reactor trip, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal shutdown procedures.

5.1.17.4 Conclusions

In the event of a CEA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected since the combination of thermal power and coolant temperature result in a DNBR greater than the limit value. Therefore, no fuel or cladding damage is predicted as a result of this transient. In addition, the RCS pressure will not approach the limit since the total amount of excess energy deposited in the reactor coolant is relatively small and there is no prolonged power mismatch between the primary and secondary side that could cause a significant RCS pressure increase. In addition, any insurge during this event would not be nearly as severe as for the Loss of Load/Turbine Trip event. Therefore, all acceptance criteria for this event are met.

5.1.17.5 References

1. Risher, D. H., Jr. and Barry, R. F., "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

2. Hargrove, H. G., "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908-A, December 1989.

Table 5.1.17-1 Assumptions and Results – Uncontrolled CEA Withdrawal from a Subcritical Condition	
Initial Power Level, %	0
Reactivity Insertion Rate, pcm/sec	53
Delayed Neutron Fraction	0.0070
Doppler Power Defect, pcm	900
Trip Reactivity, % Δk	2.0
Hot Channel Factor	7.82
Number of RCPs Operating	4

Results		
	Calculated Value	**Limit**
Peak Fuel Centerline Temperature, °F	3060	4717
Peak Fuel Average Temperature, °F	2493	4717
Minimum DNBR (small thimble cell)	1.361	1.29
Minimum DNBR (large thimble cell)	1.366	1.29
Minimum DNBR (typical cell)	1.523	1.29

Table 5.1.17-2 Sequence of Events – Uncontrolled CEA Withdrawal from a Subcritical Condition

Event	Time (seconds)
Initiation of Uncontrolled CEA Bank Withdrawal	0
Variable High Power Trip Setpoint (Lower Limit Setting) is Reached	13.5
Peak Nuclear Power Occurs	13.7
Rod Motion Begins	14.6
Peak Heat Flux Occurs	15.0
Minimum DNBR Occurs	15.0
Peak Cladding Temperature Occurs	15.3
Peak Fuel Average Temperature Occurs	15.8
Peak Fuel Centerline Temperature Occurs	16.7

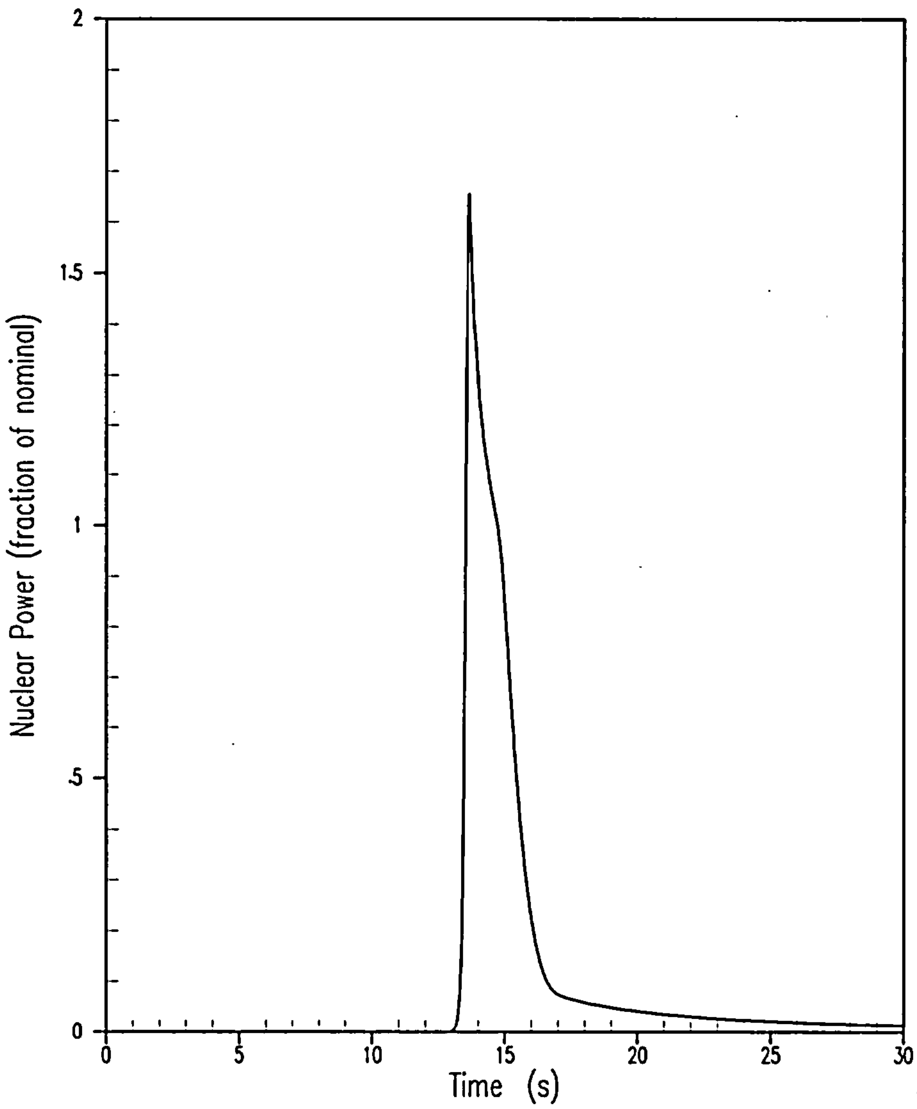

Figure 5.1.17-1 Uncontrolled CEA Withdrawal from a Subcritical Condition – Reactor Power versus Time

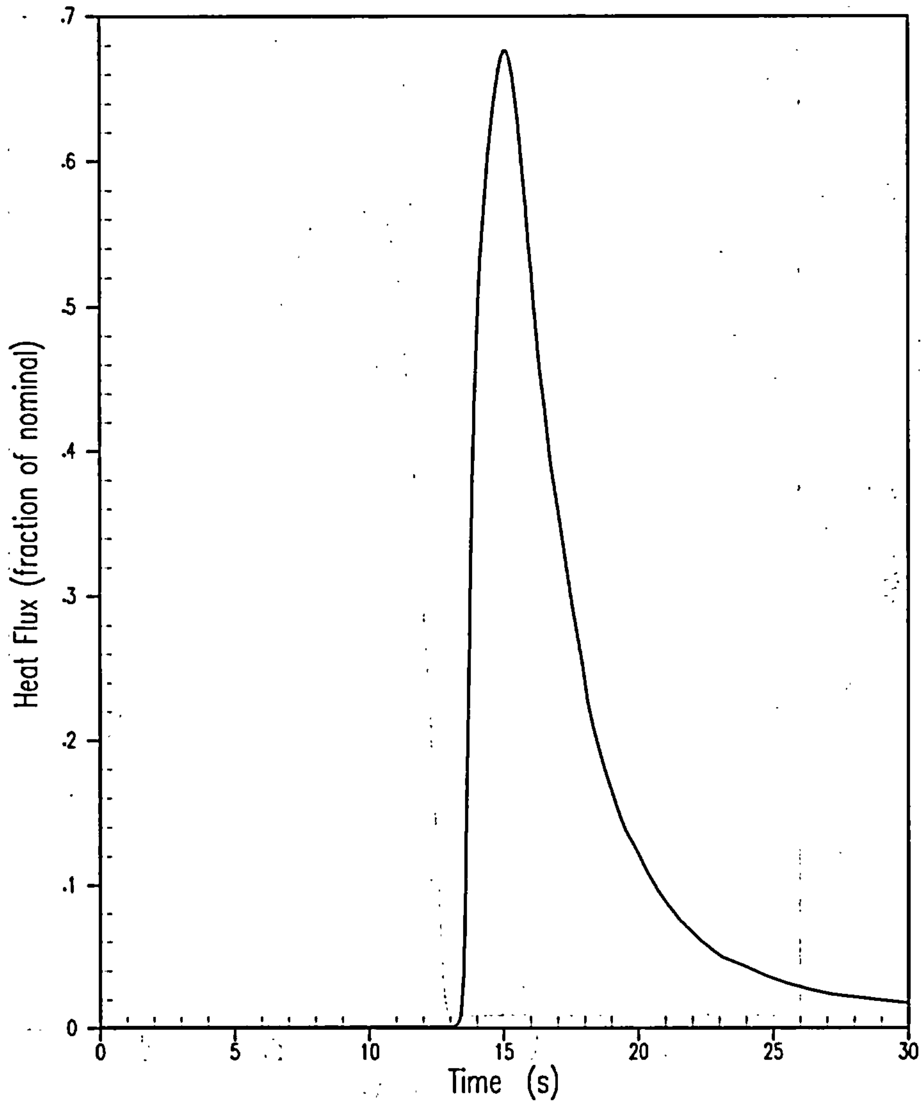

Figure 5.1.17-2 Uncontrolled CEA Withdrawal from a Subcritical Condition – Heat Flux versus Time

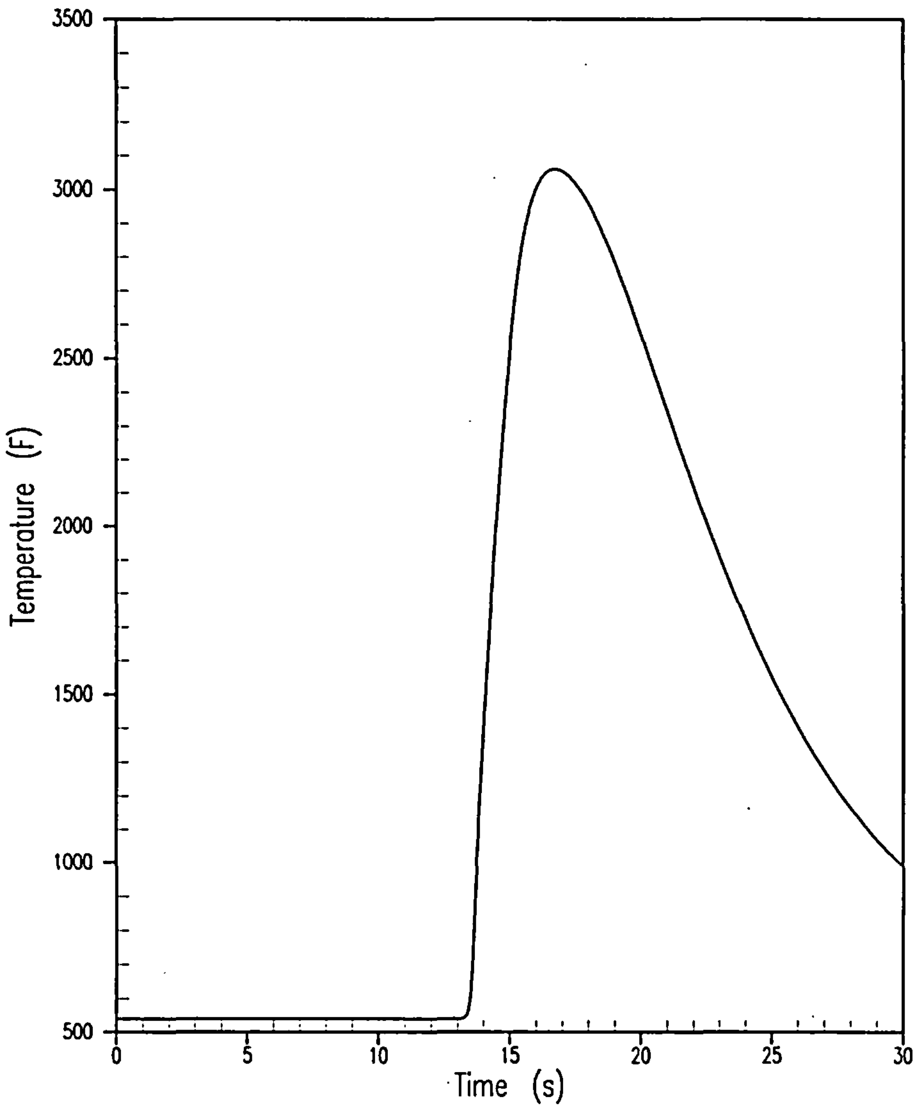

Figure 5.1.17-3 Uncontrolled CEA Withdrawal from a Subcritical Condition – Hot Spot Fuel Centerline Temperature versus Time

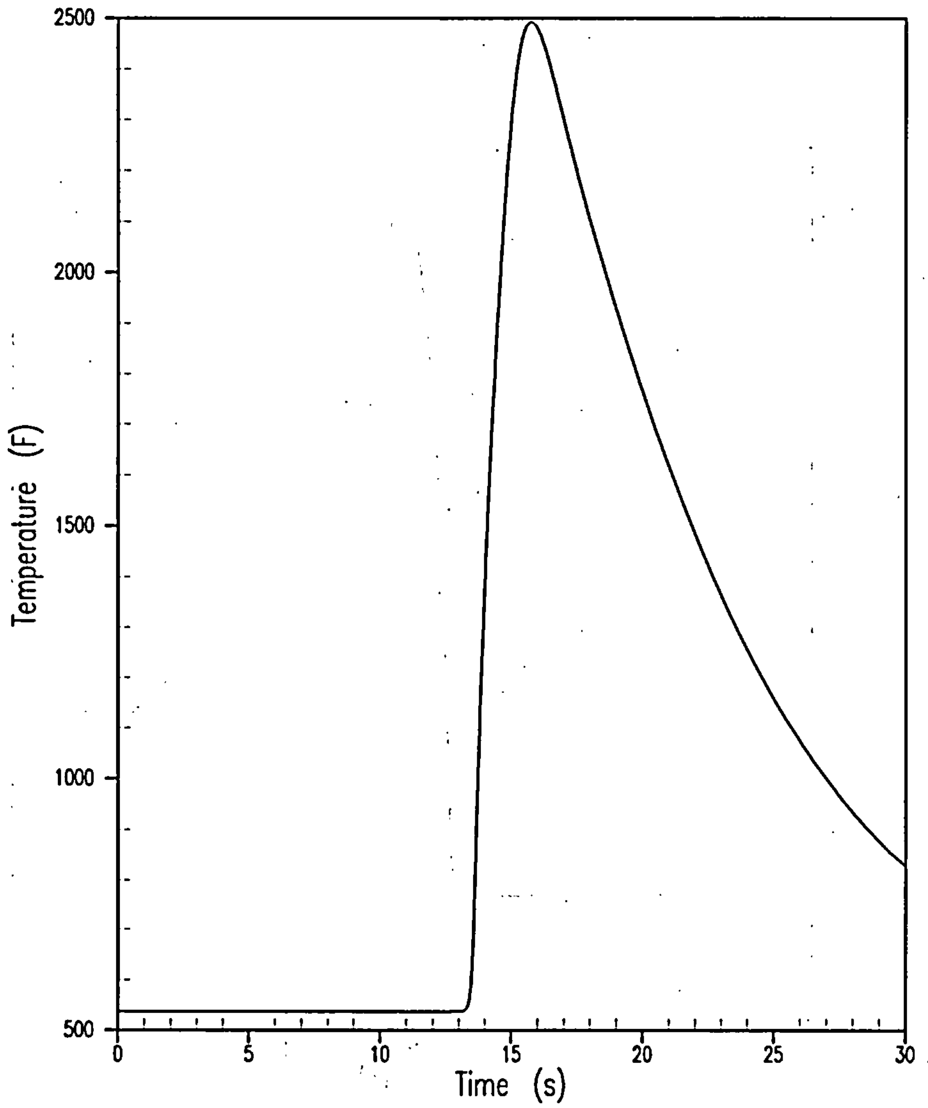

Figure 5.1.17-4 Uncontrolled CEA Withdrawal from a Subcritical Condition – Hot Spot Fuel Average Temperature versus Time

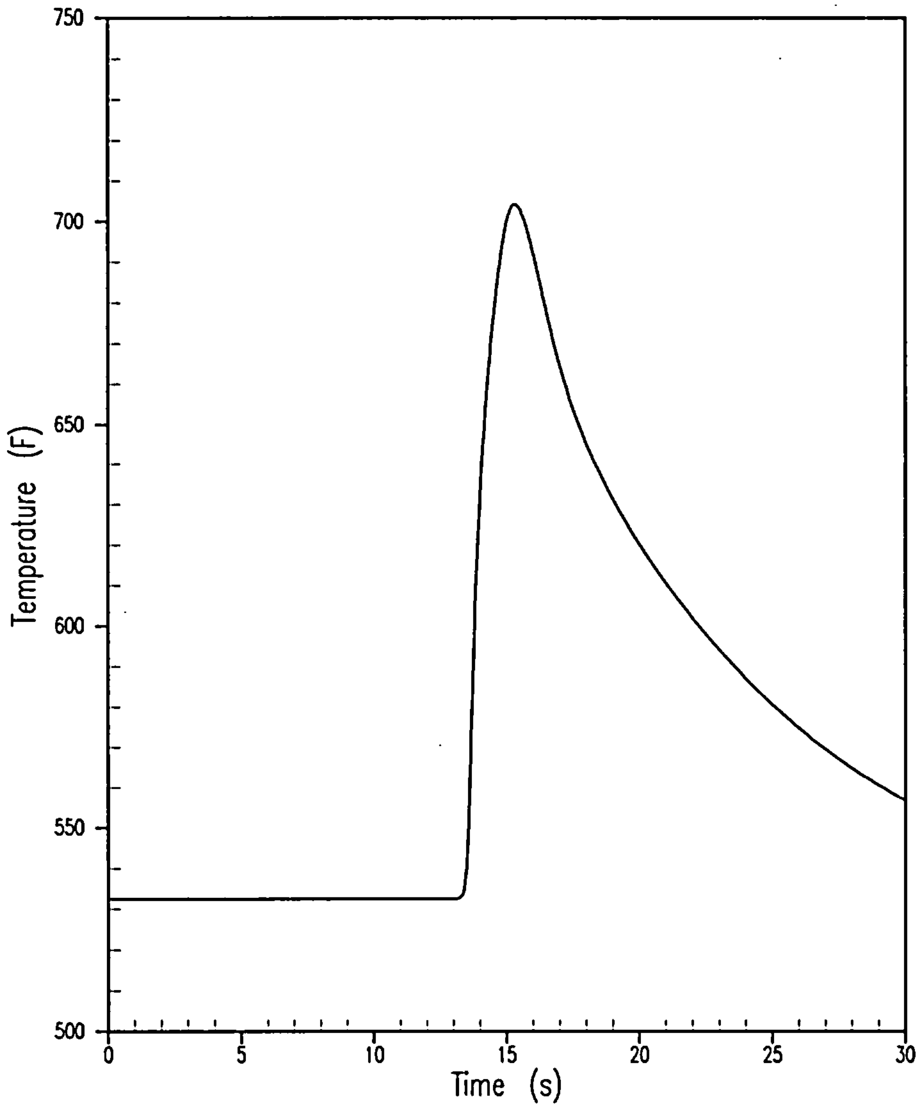

Figure 5.1.17-5 Uncontrolled CEA Withdrawal from a Subcritical Condition – Hot Spot Cladding Temperature versus Time

5.1.18 Control Element Assembly Drop Event

5.1.18.1 Identification of Causes and Accident Description

A CEA Drop Event is a Condition II event that is assumed to be initiated by a single electrical or mechanical failure which causes any number and combination of rods from a CEA subgroup to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to rapidly decrease. An increase in the hot channel factor may occur due to the skewed power distribution representative of a CEA drop configuration. Since this is a Condition II event, it must be shown that the DNB design basis is met for the combination of power, hot channel factor, and other system conditions which exist following a CEA Drop Event.

The CEA Drop Event accident includes:

- Full-length CEA drop
- Full-length CEA subgroup drop

The St. Lucie Unit 2 CEA drop detection system is assumed inoperable with no credit taken for the turbine runback feature. With a decrease in reactor power, the turbine load is not reduced, but is assumed to remain the same prior to the dropped CEA. This results in a power mismatch between the primary and secondary system, which leads to a cooldown of the RCS. In addition, the automatic withdrawal capability of the control element drive mechanism is disabled. Following a CEA drop, the plant will establish a new equilibrium condition at the original power level but as a reduced RCS temperature and pressure.

5.1.18.2 Method of Analysis

Full-length CEA Subgroup and Full-length CEA Drop

The transient following a CEA Drop Event is determined by a detailed digital simulation of the plant using the RETRAN-02 computer code as described in Reference 1. The RETRAN-02 computer code is a digital computer code, developed to simulate transient behavior in light water reactor systems. This program includes point kinetics and one-dimensional kinetics model, one-dimensional homogenous equilibrium mixture thermal-hydraulic model, control system models, two-phase natural convection heat transfer correlation, a non-equilibrium pressurizer model, etc. The code computes pertinent plant variables including temperatures, pressures, and power levels. Since RETRAN employs a point kinetics model, a CEA drop is modeled as a negative reactivity insertion corresponding to the reactivity worth of the dropped CEA regardless of the configuration of the CEA(s) that drop. The system transient is calculated by assuming a constant turbine load demand at the initial value (no turbine runback) and no bank withdrawal. A spectrum of dropped CEA worths from 100 pcm to 1000 pcm was analyzed.

Transient conditions are calculated that are then analyzed with nuclear models to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met.

5.1.18.3 Results

Full-length CEA Subgroup and Full-length CEA Drop

Full-length CEA drops result in a negative reactivity insertion. Full-length CEA subgroup drops typically result in a negative reactivity insertion greater than 500 pcm. The core is not adversely affected during this period since power is decreasing rapidly.

Following a CEA Drop Event, power may be reestablished by reactivity feedback. In cases where reactivity feedback does not offset the worth of the dropped CEA, a cooldown condition exists until a reactor trip is reached on a TM/LP (floor) or a low steam generator pressure signal. Figures 5.1.18-1 through 5.1.18-4 show typical transient response of a dropped CEA of 500 pcm worth at an MTC of 0 pcm/°F. In cases where reactivity is large enough to offset the worth of the dropped CEA, reactor power is reestablished at the original power level at a reduced RCS temperature and pressure condition. Figures 5.1.18-5 through 5.1.18-8 show a typical transient response of a dropped CEA of 500 pcm worth at a moderator temperature coefficient of -25 pcm/°F. In all cases, the minimum DNBR remains greater than the limit value.

Following plant stabilization, the operator may manually retrieve the CEA by following approved operating procedures.

5.1.18.4 Conclusions

Following a limiting CEA Drop condition event, the plant will return to a stabilized condition at the initial power level. Results of the analysis show that a CEA Drop Event does not adversely affect the core, since the DNBR remains above the safety analysis limit value for a bounding range of dropped CEA worths.

5.1.18.5 References

1. WCAP-14882-P-A, Revision 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurizer Water Reactor Non-LOCA Safety Analysis," April 1999.

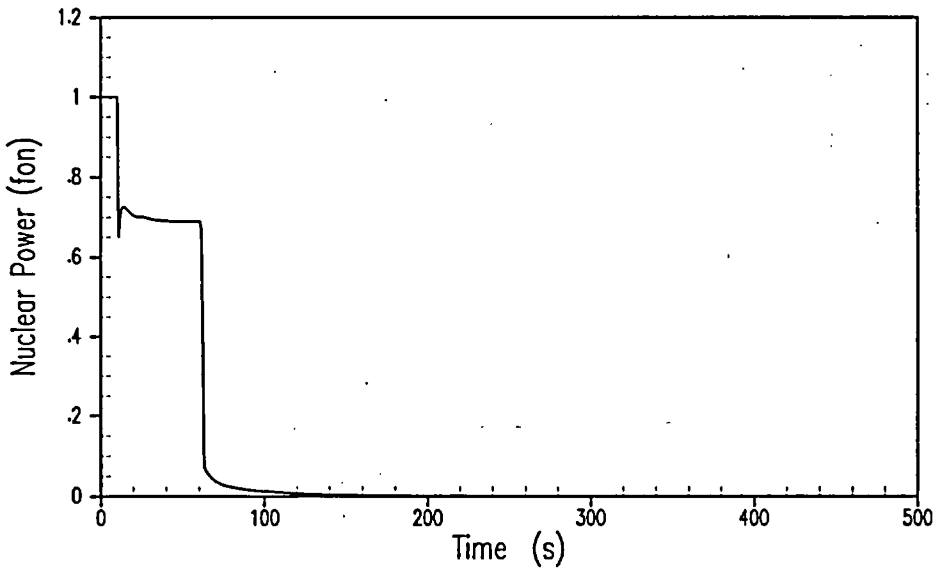

Figure 5.1.18-1 Representative Transient Response to a Dropped CEA Worth of 500 pcm at a Moderator Temperature Coefficient of 0 pcm/°F – Nuclear Power versus Time

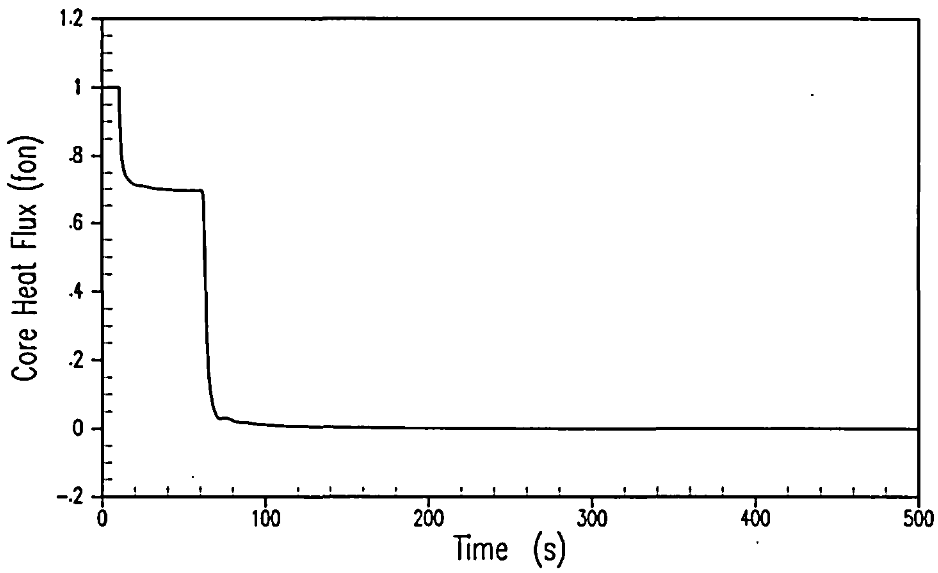

Figure 5.1.18-2 Representative Transient Response to Dropped CEA Worth of 500 pcm at a Moderator Temperature Coefficient of 0 pcm/°F – Core Heat Flux versus Time

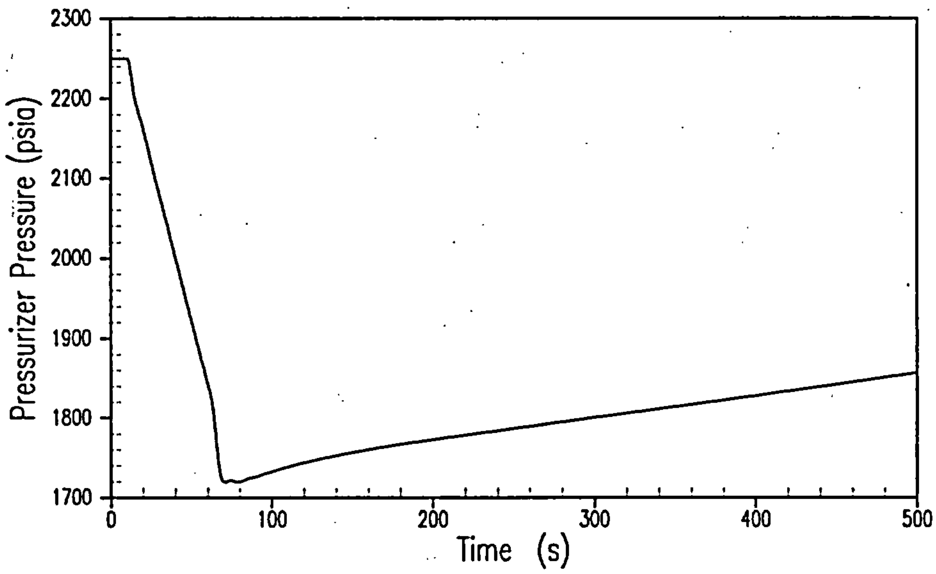

Figure 5.1.18-3 Representative Transient Response to Dropped CEA Worth of 500 pcm at a Moderator Temperature Coefficient of 0 pcm/°F – Pressurizer Pressure versus Time

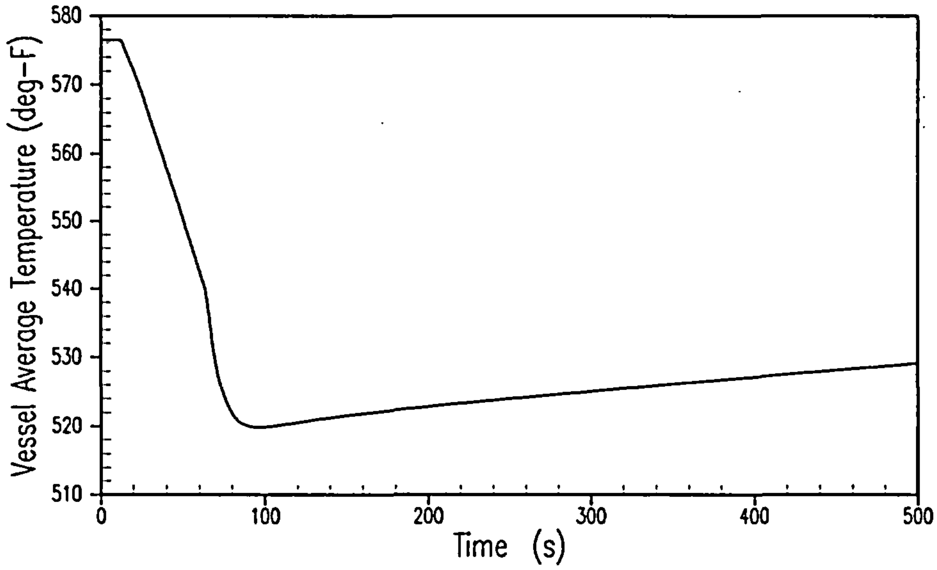

Figure 5.1.18-4 Representative Transient Response to Dropped CEA Worth of 500 pcm at a Moderator Temperature Coefficient of 0 pcm/°F – Vessel Average Temperature versus Time

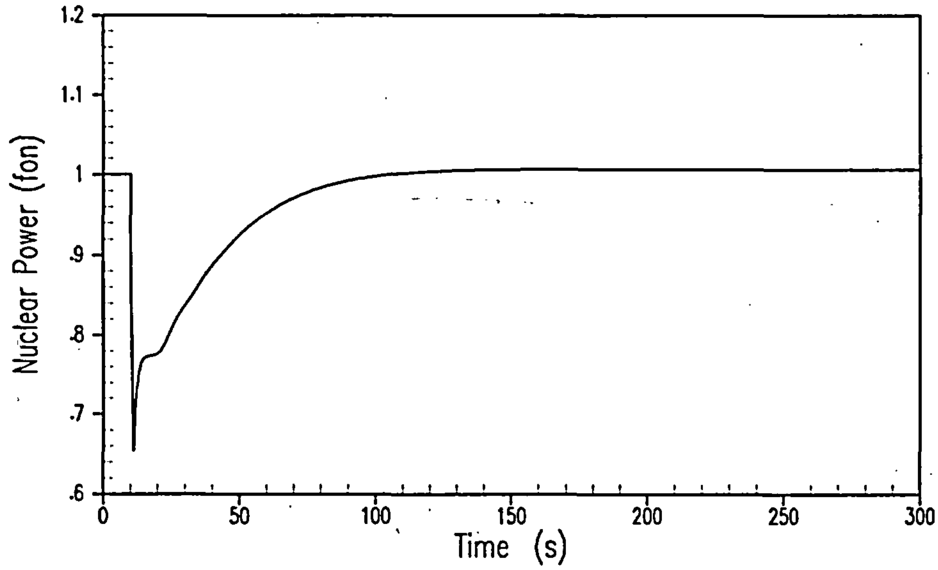

Figure 5.1.18-5 Representative Transient Response to a Dropped CEA Worth of 500 pcm at a Moderator Temperature Coefficient of -25 pcm/°F – Nuclear Power versus Time

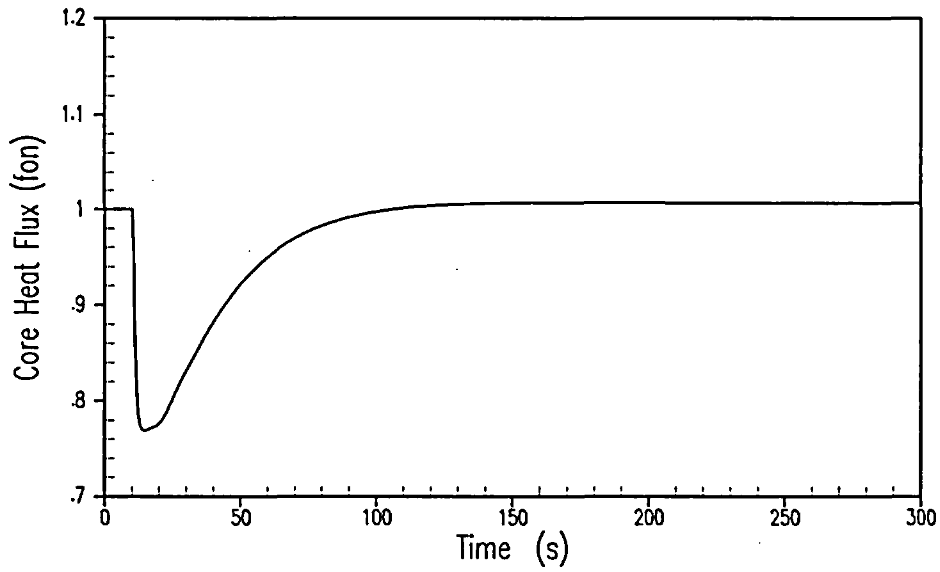

Figure 5.1.18-6 Representative Transient Response to Dropped CEA Worth of 500 pcm at a Moderator Temperature Coefficient of -25 pcm/°F – Core Heat Flux versus Time

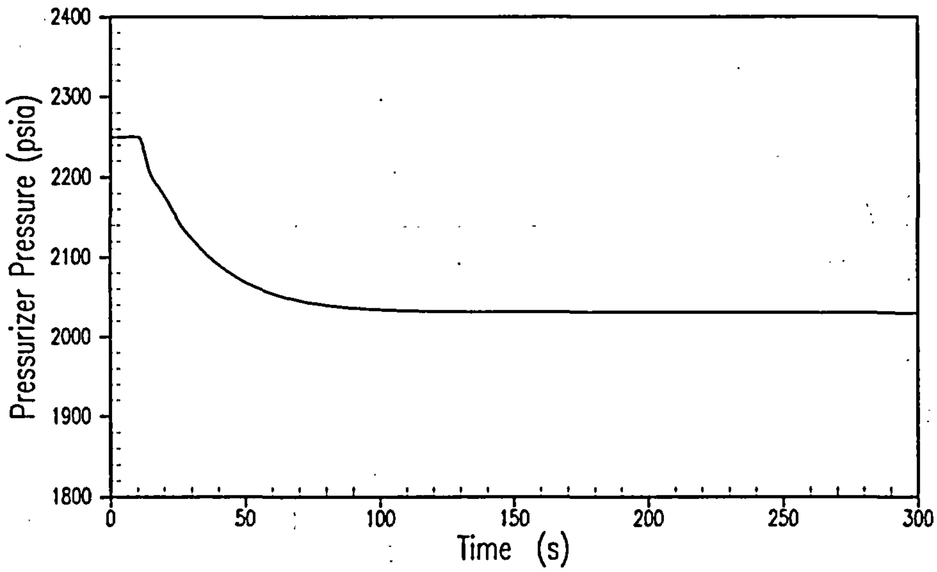

Figure 5.1.18-7 Representative Transient Response to Dropped CEA Worth of 500 pcm at a Moderator Temperature Coefficient of -25 pcm/°F – Pressurizer Pressure versus Time

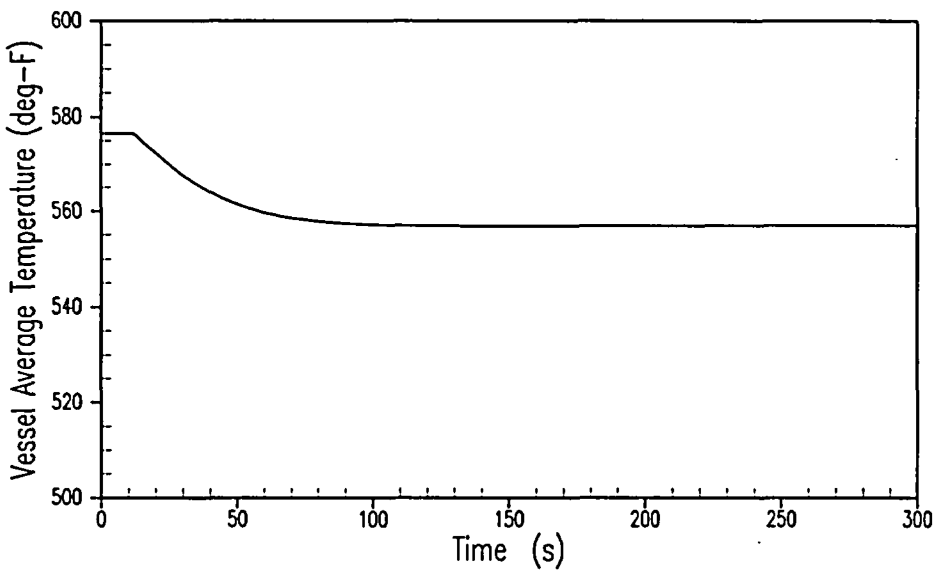

Figure 5.1.18-8 Representative Transient Response to Dropped CEA Worth of 500 pcm at a Moderator Temperature Coefficient of -25 pcm/°F – Vessel Average Temperature versus Time

5.1.19 Chemical and Volume Control System Malfunction (Uncontrolled Boron Dilution)

5.1.19.1 Accident Description

An Uncontrolled Boron Dilution event is defined as any event caused by a malfunction or an inadvertent operation of the CVCS that results in a dilution of the active portion of the RCS. The active portion of the RCS is defined as that volume of water that circulates through the core. For example, when in shutdown cooling (SDC), no credit is allowed for the volume of water in the stagnant portions of the RCS. A dilution of the RCS can be the result of adding water, which has a boron concentration that is less than the system boron concentration.

The CVCS regulates both the chemistry and the quality of coolant in the RCS. Changing the boron concentration in the RCS is a part of normal plant operation, compensating for long-term reactivity effects such as fuel burnup, xenon buildup and decay, and plant startup. During refueling operations, borated water is supplied from the refueling water tank (RWT).

Boron concentration in the RCS can be decreased by controlled addition of demineralized water. During normal operation, concentrated boric acid solution and demineralized water is introduced into the volume control tank in concentrations corresponding to the required concentration for proper plant operation. A purification ion exchanger with a deborating resin is normally used for boron removal when the boron concentration in the RCS is low and the feed and bleed method becomes inefficient.

The following provide a direct indication of a boron dilution in process:

- BDAS – Boron Dilution Alarm System (UFSAR 7.7.1.1.11), which provides alarm,
- and sampling.

The CVCS malfunction analysis (boron dilution) is performed to ensure that the analysis results meet the acceptance criteria for all modes, and remain consistent with the BDAS setpoint and sampling frequency for Modes 3, 4, 5, and 6.

To cover all phases of plant operation, boron dilution during refueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation is considered in this analysis. Assumptions used in the analysis result in conservative determinations of the time available for operator or system response after detection of a dilution transient in progress. Dilution flow rates listed for each mode are based on the dilution source fluid conditions for reactor makeup water at 40°F and 14.7 psia. The analysis results are based on calculations which account for density compensation between the dilution source conditions and the mode-specific RCS conditions listed.

5.1.19.2 Method of Analysis

Boron dilutions during all six modes of operation (refueling, cold shutdown, hot shutdown, hot standby, startup, and power operation) are considered in this analysis.

Dilution during Refueling (Mode 6)

The plant is normally maintained in Mode 6 at the beginning of cycle when fuel is being loaded and arranged in the core and at the end of cycle for the removal of spent fuel. Mode 6 is also used for the performance of plant maintenance. In Mode 6, when the head bolts are being tensioned or detensioned for the replacement or removal of the vessel head, the water level in the vessel is maintained below the top of the flange. The primary coolant forced flow is provided by the shutdown cooling system (SCS).

The following conditions are assumed for the limiting Mode 6 analysis with the RCS drained to the hot leg centerline:

- Dilution flow is the maximum capacity of one charging pump, 49 gpm.
- A minimum RCS water volume of 3412 ft^3 is assumed, which is more conservative (that is, smaller) than the volume necessary to fill the reactor vessel up to the mid-plane of the nozzles plus the volume of one SCS train.
- The minimum boron concentration during refueling ($k_{eff} < 0.95$) and minimum change in boron concentration from initial to critical conditions during cold shutdown are plant-specific values that are determined and verified every cycle as part of the reload process.

Dilution during Cold Shutdown (Mode 5)

In this mode, the plant is being taken from a long-term mode of operation, refueling (Mode 6), to a short-term mode of operation, hot shutdown (Mode 4). Typically, the plant is maintained in the cold shutdown mode when reduced RCS inventory is necessary or ambient temperatures are required. The water level can be dropped to the mid-plane of the hot leg for maintenance work that requires the steam generators to be drained. The plant is maintained in Mode 5 at the beginning of cycle for startup testing of certain systems. The limiting scenario for Mode 5 is typically the case where the vessel is drained and the reactor is shut down by boron to the technical specifications requirement. The boron dilution event is analyzed assuming the following conditions:

- Dilution flow is the maximum capacity of one charging pump, 49 gpm.
- A minimum RCS water volume of 3712 ft^3 corresponding to the active RCS volume for the filled case, and 3412 ft^3 corresponding to the active RCS volume for the partially drained case.
- The maximum critical boron concentration and minimum change in boron concentration from initial to critical conditions during cold shutdown are plant-specific values that are determined and verified every cycle as part of the reload verification process.

Dilution during Hot Shutdown (Mode 4)

In Mode 4, the plant is being taken from a short-term mode of operation, cold shutdown (Mode 5), to a long-term mode of operation, hot standby (Mode 3). Typically, the plant is maintained in the hot shutdown mode to achieve plant heatup before entering Mode 3. In Mode 4, the primary coolant forced flow can be provided by either the SCS or an RCP, depending on the system pressure. The boron dilution event in Mode 4 is analyzed assuming the following conditions:

- Dilution flow is the maximum capacity of two charging pumps, 98 gpm for the case analyzed with the plant on shutdown cooling system, and 147 gpm for the case analyzed with the plant operating with at least one RCP running.
- A minimum RCS water volume of 3712 ft^3 corresponding to the active RCS volume (not including the pressurizer volume) for the case analyzed with the plant on shutdown cooling system and 7368.6 ft^3 (not including the pressurizer volume and including the effects of 30% SGTP) for the case analyzed with the plant operating with at least one RCP running.
- The maximum critical boron concentration and minimum change in boron concentration from initial to critical conditions during hot shutdown are plant-specific values that are determined and verified every cycle as part of the reload verification process.

Dilution during Hot Standby (Mode 3)

In Mode 3, the plant is being taken from one short-term mode of operation, hot shutdown (Mode 4), to another, startup (Mode 2). The plant is maintained in Mode 3 at the beginning of cycle for startup testing of certain systems and to achieve plant heatup before entering Mode 2 and going critical. During cycle operation, the plant will enter Mode 3 following a reactor trip or as the result of a technical specification action statement. In Mode 3, all reactor coolant pumps may not be in operation. In the approach to Mode 2, the operator must manually withdraw the control rods and may initiate a limited dilution according to shutdown margin requirements. If the control rods are withdrawn to the HZP insertion limits, the inadvertent dilution scenario is similar to the Mode 2 analysis. The dilution scenario is more limiting if the control rods are not withdrawn and the reactor is shutdown by boron to the technical specifications minimum requirement for Mode 3. Conditions assumed for the analysis are:

- Dilution flow is the maximum capacity of three charging pumps, 147 gpm.
- A minimum RCS water volume of 7368 ft^3 corresponding to the active RCS volume (not including the pressurizer volume), including the effects of 30% steam generator tube plugging.
- The maximum critical boron concentration and minimum change in boron concentration from initial to critical conditions during hot shutdown are plant-specific values that are determined and verified every cycle as part of the reload verification process.

Dilution during Startup (Mode 2)

In this mode, the plant is being taken from one long-term mode of operation, hot standby (Mode 3), to another, power (Mode 1). All normal actions required to change power level require operator initiation. For a normal approach to criticality, the operator manually initiates a limited dilution and manually withdraws the control rods. Conditions assumed for the analysis are:

- Dilution flow is the maximum capacity of all three charging pumps, 147 gpm.

- A minimum RCS water volume of 7368 ft^3 corresponding to the active RCS volume (not including the pressurizer), including the effects of 30% steam generator tube plugging.

- The maximum critical boron concentration and minimum change in boron concentration from initial to critical conditions during startup are plant-specific values that are confirmed to be valid every cycle as part of the reload verification process.

This mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual rod control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently withdraw the control rods. This process takes several hours. The technical specifications require that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source-range reactor trip. For inadvertent boron dilution with slow reactivity additions, this event is bounded by the CEA withdrawal event. For fast reactivity additions, this event is protected by high rate of change of power reactor trip. Once dilution has been identified, the operator terminates the flow of non-borated water.

Dilution at Power (Mode 1)

The plant is operated at power with the rod control system in the manual mode. The analysis is performed assuming three charging pumps are in operation. Conditions assumed for this mode are:

- Dilution flow is the maximum capacity of all three charging pumps, 147 gpm.

- A minimum RCS water volume of 7368 ft^3 at 579.5°F. This is a very conservative estimate of the active RCS volume (not including the pressurizer), including the effects of 30% steam generator tube plugging.

- The maximum critical boron concentration (corresponding to the rods inserted to the insertion limits) and the minimum change in boron concentration from this initial condition to an HZP critical condition with all rods inserted are plant-specific values that are confirmed to be valid every cycle as part of the reload verification process. Full rod insertion, minus the most reactive stuck rod, is assumed to occur due to reactor trip.

If the dilution is not secured, the reactor will be shut down by either the TM/LP reactor trip, the high pressurizer pressure (HPP) reactor trip or the variable high power trip (VHPT).

Once dilution has been identified, the operator terminates the flow of non-borated water.

5.1.19.3 Operator Action Time Requirements

Analyses to determine the extent of fuel cladding damage and the overpressurization of the RCS are not done for this event. Instead, a calculation is performed to determine the time to alert the operator, either by BDAS or RCS sampling, so that the time available for operator action prior to the loss of the plant shutdown margin meets the respective acceptance criteria for each mode. Fifteen minutes for the at-power and startup conditions and thirty minutes for the refueling condition of plant operation are the criteria outlined in the Standard Review Plan (SRP), Section 15.4.6. If these operator action times are met, it can be concluded that the fuel cladding damage and RCS overpressurization criteria are also satisfied.

5.1.19.4 Results

Dilution during Refueling (Mode 6)

The results provided below are based on using representative St. Lucie Unit 2 boron concentrations, which will be verified every cycle. For dilution during refueling, the maximum time available for alarm annunciation, such that the 30-minute operator action criterion remains satisfied, is 90.82 minutes.

Dilution during Cold Shutdown (Mode 5 – filled)

For dilution during cold shutdown (filled), the maximum time available for alarm annunciation, such that the 15-minute operator action criterion remains satisfied, is 62.57 minutes.

Dilution during Cold Shutdown (Mode 5 – drained)

For dilution during cold shutdown (drained), the maximum time available for alarm annunciation, such that the 15-minute operator action criterion remains satisfied, is 56.29 minutes.

Dilution during Hot Shutdown (Mode 4 – at least one RCP operating)

For dilution during hot shutdown (at least one RCP operating), the maximum time available for alarm annunciation, such that the 15-minute operator action criterion remains satisfied, is 60.43 minutes.

Dilution during Hot Shutdown (Mode 4 – with Shutdown Cooling System)

For dilution during hot shutdown (with SCS), the maximum time available for alarm annunciation, such that the 15-minute operator action criterion remains satisfied, is 41.94 minutes.

Dilution during Hot Standby (Mode 3)

For dilution during hot standby, the maximum time available for alarm annunciation, such that the 15-minute operator action criterion remains satisfied, is 43.03 minutes.

For Dilution during Startup (Mode 2)

For dilution during startup, the maximum time available for alarm annunciation, such that the 15-minute operator action criterion remains satisfied is 78.75 minutes.

For Dilution during Full-Power Operation (Mode 1)

With the reactor in manual control, if no operator action is taken, the power and temperature rise causes the reactor to reach the HPP reactor trip setpoint. The boron dilution accident in this case is essentially identical to a CEA withdrawal accident at power. Prior to the HPP trip, an HPP alarm would be actuated. There is time available (~87 minutes) after a reactor trip for the operator to determine the cause of dilution, isolate the reactor makeup water source, and initiate reboration before the reactor can return to criticality.

5.1.19.5 Monitoring Frequency

Should the automatic boron dilution alarm be inoperable, UFSAR Section 13.7.2.4 contains requirements for the maximum frequency of RCS chemistry sampling. These sampling frequencies ensure that the specified criterion are met to ensure sufficient time is available to the operators, from the detection of dilution until criticality is achieved to mitigate the consequences of this event. The backup boron dilution detection monitoring frequencies provided in UFSAR Table 13.7.2-3 are verified every cycle and changed as necessary.

5.1.19.6 Conclusions

The time sequence of events is provided in Table 5.1.19-1. The boron dilution analyses at refueling, cold shutdown, hot shutdown, hot standby, startup, and full-power conditions show the acceptability of the increase to 30% steam generator tube plugging.

Table 5.1.19-1 Uncontrolled Boron Dilution Sequence of Events			
Mode		**Event**	**Time (minutes)**
Refueling (Mode 6)		Dilution begins	0
		Maximum available time from initiation of boron dilution to alarm annunciation	cycle-dependent
		Time from alarm annunciation to criticality	30
Cold Shutdown (Mode 5)	Filled	Dilution begins	0
		Maximum available time from initiation of boron dilution to alarm annunciation	62.57
		Time from alarm annunciation to criticality	15
	Partially Drained	Dilution begins	0
		Maximum available time from initiation of boron dilution to alarm annunciation	56.29
		Time from alarm annunciation to criticality	15
Hot Shutdown (Mode 4)	At least one RCP operating	Dilution begins	0
		Maximum available time from initiation of boron dilution to alarm annunciation	60.43
		Time from alarm annunciation to criticality	15
	Shutdown Cooling System (SCS)	Dilution begins	0
		Maximum available time from initiation of boron dilution to alarm annunciation	41.94
		Time from alarm annunciation to criticality	15
Hot Standby (Mode 3)		Dilution begins	0
		Maximum available time from initiation of boron dilution to alarm annunciation	43.03
		Time from alarm annunciation to criticality	15
Startup (Mode 2)		Dilution begins	0
		Maximum available time from initiation of boron dilution to alarm annunciation	78.75
		Time from alarm annunciation to criticality	15
At Power (Mode 1)		Dilution begins	0
		HPP reactor trip signal reached	172.02
		Rod motion begins	173.16
		Shutdown margin is lost (if dilution continues after trip)	> 15

5.1.20 Control Element Assembly Ejection

5.1.20.1 Accident Description

This accident is the result of the postulated mechanical failure of a control element drive mechanism (CEDM) pressure housing such that the RCS pressure would eject the CEA and drive shaft. The consequences of this mechanical failure, in addition to being a minor LOCA, may also be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Rapid ejection of a CEA from the core would require a complete circumferential break of the CEDM housing or the CEDM nozzle on the reactor vessel head. The CEDM housing and CEDM nozzle are an extension of the reactor coolant system boundary and designed and manufactured to Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Hence, the occurrence of such a failure is considered highly unlikely.

If a CEA ejection accident were to occur, a fuel rod thermal transient that could cause a DNB may occur together with limited fuel damage. The amount of fuel damage that can result from such an accident will be governed mainly by the worth of the ejected CEA and the power distribution attained with the remaining control element pattern. The transient is limited by the Doppler reactivity effects of the increase in fuel temperature and is terminated by reactor trip actuated by the high-power level trip. The transient is terminated before conditions are reached that can result in damage to the reactor coolant pressure boundary, or significant disturbances in the core, its support structures or other reactor pressure vessel internals that would impair the capability to cool the core.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative DPC. This self limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a CEA ejection accident occur, the following automatic features of the reactor protection system are available to terminate the transient:

- The Variable Power Level – High trip is provided to trip the reactor when the reactor power reaches a high preset value. This setpoint is set to a fixed increment (≤ 9.61% technical specification value) above the existing reactor power level, with a minimum setpoint of 15% of rated thermal power and a maximum of ≤ 107% of rated thermal power. This trip is actuated when two-out-of-four power range channels indicate a power level above the setpoint.

- The Rate-of-Change of Power – High trip is provided to trip the reactor when the rate-of-change of neutron flux power reaches a high preset value (≤ 2.49 decades per minute technical specification value). It is actuated when two-out-of-four wide-range logarithmic neutron flux monitoring channels indicate a rate above the preset setpoint. This trip function may be bypassed below 10^{-4}% and above 15% of rated thermal power. Bypass is automatically removed when wide-range logarithmic neutron flux power is ≥ 10^{-4}% and power range neutron flux power is ≤ 15% of rated thermal power.

The ultimate acceptance criteria for this event is that any consequential damage to either the core or the RCS must not prevent long-term core cooling, and that any offsite dose consequences must be within the guidelines of 10 CFR 100. To demonstrate compliance with these requirements, it is sufficient to show that the RCS pressure boundary remains intact, and that no fuel dispersal in the coolant, gross lattice distortions, or severe shock waves will occur in the core. Therefore, the following acceptance criteria are applied to the CEA ejection accident:

- Maximum average fuel pellet enthalpy at the hot spot must remain below 200 cal/g (360 Btu/lbm).
- Peak RCS pressure must remain below that which would cause the stresses in the RCS to exceed the faulted condition stress limits.
- Maximum fuel melting must be limited to the innermost 10 percent of the fuel pellet at the hot spot, independent of the above pellet enthalpy limit.

5.1.20.2 Method of Analysis

The calculation of the CEA ejection transient is performed in two stages: a core neutron kinetic analysis and a hot-spot fuel heat transfer analysis. The spatial neutron kinetics code TWINKLE (Reference 1) is used in a 1-D axial kinetics model to calculate the core nuclear power including the various total core feedback effects; that is, Doppler reactivity and moderator reactivity. The average core nuclear power is multiplied by the post-ejection hot-channel factor, and the fuel enthalpy and temperature transients at the hot spot are calculated with the detailed fuel and cladding transient heat transfer computer code, FACTRAN (Reference 2). The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. Additional details of the methodology are provided in WCAP-7588 (Reference 3).

In calculating the nuclear power and hot-spot fuel rod transients following CEA ejection, the following conservative assumptions are made:

1. The RTDP is not used for the CEA ejection analysis. Instead, the STDP (maximum uncertainties in initial conditions) is employed. The analysis assumes uncertainties of 2.0 percent in nominal core power, 3.0°F in nominal vessel T_{avg}, and 45 psi in nominal pressurizer pressure.

2. A minimum value for the delayed neutron fraction for beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions is assumed, which increases the rate at which the nuclear power increases following CEA ejection.

3. A minimum value of the Doppler power defect is assumed, which conservatively results in the maximum amount of energy deposited in the fuel following CEA ejection. A minimum value of the moderator feedback is also assumed. A positive MTC is assumed for the BOC, zero-power case. The analysis-specific values are presented in Table 5.1.0-2.

4. Maximum values of ejected CEA worth and post-ejection total hot-channel factors are assumed for all cases considered. These parameters are calculated using standard nuclear design codes for

the maximum allowed bank insertion at a given power level as determined by the rod insertion limits. No credit is taken for the flux flattening effects of reactivity feedback.

5. The total time for CEA ejection is assumed to be 0.05 seconds.

6. For the HZP cases, the reactor is assumed to be tripped by the nuclear power signal of the variable high-power (VHP) trip at the lower limit (floor) setpoint. For the HFP cases, the reactor trip is assumed to occur on the variable power level function of the VHP trip at a conservative setpoint. Appropriate error allowances are added to the technical specification setpoints to determine the analysis trip point. A trip time delay of 0.4 seconds for reactor trip breaker opening is used, with the control rods assumed to start moving 0.74 seconds after breaker opening.

7. The analysis conservatively assumes the trip rods are inserted starting from the fully withdrawn position, using a conservative rod position versus time curve. Also, the total trip reactivity is based on the conservative assumption that the highest worth adjacent CEA is stuck in its fully withdrawn position in addition to the ejected rod.

8. Both UO2-only fuel and 8 weight-percent (w/o) gadolinium-doped fuel were modeled in the analysis.

5.1.20.3 Results

Figures 5.1.20-1 through 5.1.20-8 present the nuclear power and hot-spot fuel rod thermal transients for the CEA ejection cases analyzed. The transient results of the analysis are summarized in Table 5.1.20-1. A time sequence of events is provided in Table 5.1.20-2. For all cases, the maximum fuel pellet enthalpy remained below 200 cal/g, and the peak hot-spot fuel centerline temperature remained below the fuel melting temperature (4900°F at BOC and 4800°F at EOC for UO_2-only; 4816°F at BOC and 4717°F at EOC for 8 w/o gad-doped fuel). The UO_2-only fuel was found to be more limiting for the HZP cases. The 8 w/o gad-doped fuel (without taking credit for the power suppression due to the gadolinium) was more limiting for the HFP cases.

5.1.20.4 Conclusions

The analysis performed has demonstrated that, for the CEA ejection event, the fuel thermal criteria are not exceeded. Based on the generic assessment in WCAP-7588, Revision 1-A (Reference 3) and the peak pressure results documented in the current UFSAR (Reference 4), the peak reactor coolant pressure will be less than that which would cause stresses to exceed the faulted condition stress limits. In addition, based on the generic assessment in Reference 3 and the rods-in-DNB results documented in the current UFSAR (Reference 4), the number of rods in DNB is expected to not exceed 9.5%. Therefore, all acceptance criteria for this event have been met.

5.1.20.5 References

1. Risher, D. H., Jr. and Barry, R. F., "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

2. Hargrove, H. G, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod," WCAP-7908-A, December 1989.

3. D. H. Risher, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

4. "St. Lucie Unit 2 Updated Final Safety Analysis Report," Amendment 14, Docket No. 50-389.

Table 5.1.20-1 Assumptions and Results – CEA Ejection		
Beginning of Cycle	**Full Power**	**Zero Power**
Initial Power Level, %	102	0
Ejected RCCA Worth, % Δk	0.25	0.60
Delayed Neutron Fraction	0.0050	0.0050
Doppler Power Defect, % Δk	0.900	0.900
Feedback Reactivity Weighting	1.262	2.633
Trip Reactivity, % Δk	3.0	2.0
F_Q Before Ejection	2.809	N/A
F_Q After Ejection	5.25	15.0
Number of RCPs Operating	4	4
Maximum Fuel Pellet Enthalpy, cal/g	151.1	70.8
Maximum Fuel Melted, %	None	None
End of Cycle	**Full Power**	**Zero Power**
Initial Power Level, %	102	0
Ejected RCCA Worth, % Δk	0.25	0.60
Delayed Neutron Fraction	0.0044	0.0044
Doppler Power Defect, % Δk	0.900	0.900
Feedback Reactivity Weighting	1.262	3.802
Trip Reactivity, % Δk	3.0	2.0
F_Q Before Ejection	2.809	N/A
F_Q After Ejection	5.25	26.25
Number of RCPs Operating	4	4
Maximum Fuel Pellet Enthalpy, cal/g	141.7	77.4
Maximum Fuel Melted, %	None	None

Table 5.1.20-2 Sequence of Events – CEA Ejection	
Beginning of Cycle – Hot Zero Power	**Time (seconds)**
CEA Ejection Occurs	0.00
Variable High Power Trip Setpoint (Lower Limit Setting) is Reached	0.49
Peak Nuclear Power Occurs	0.56
Rods Begin to Fall Into the Core	1.63
Peak Cladding Average Temperature Occurs	2.4
Peak Fuel Average Temperature Occurs	2.6
Beginning of Cycle – Hot Full Power	**Time (seconds)**
CEA Ejection Occurs	0.00
Variable High Power Trip Setpoint is Reached	0.03
Peak Nuclear Power Occurs	0.09
Rods Begin to Fall Into the Core	1.17
Peak Fuel Average Temperature Occurs	2.3
Peak Cladding Average Temperature Occurs	2.4
End of Cycle – Hot Zero Power	**Time (seconds)**
CEA Ejection Occurs	0.00
Variable High Power Trip Setpoint (Lower Limit Setting) is Reached	0.33
Peak Nuclear Power Occurs	0.38
Rods Begin to Fall Into the Core	1.47
Peak Cladding Average Temperature Occurs	1.9
Peak Fuel Average Temperature Occurs	2.0
End of Cycle – Hot Full Power	**Time (seconds)**
CEA Ejection Occurs	0.00
Variable High Power Trip Setpoint is Reached	0.02
Peak Nuclear Power Occurs	0.09
Rods Begin to Fall Into the Core	1.16
Peak Fuel Average Temperature Occurs	2.3
Peak Cladding Average Temperature Occurs	2.4

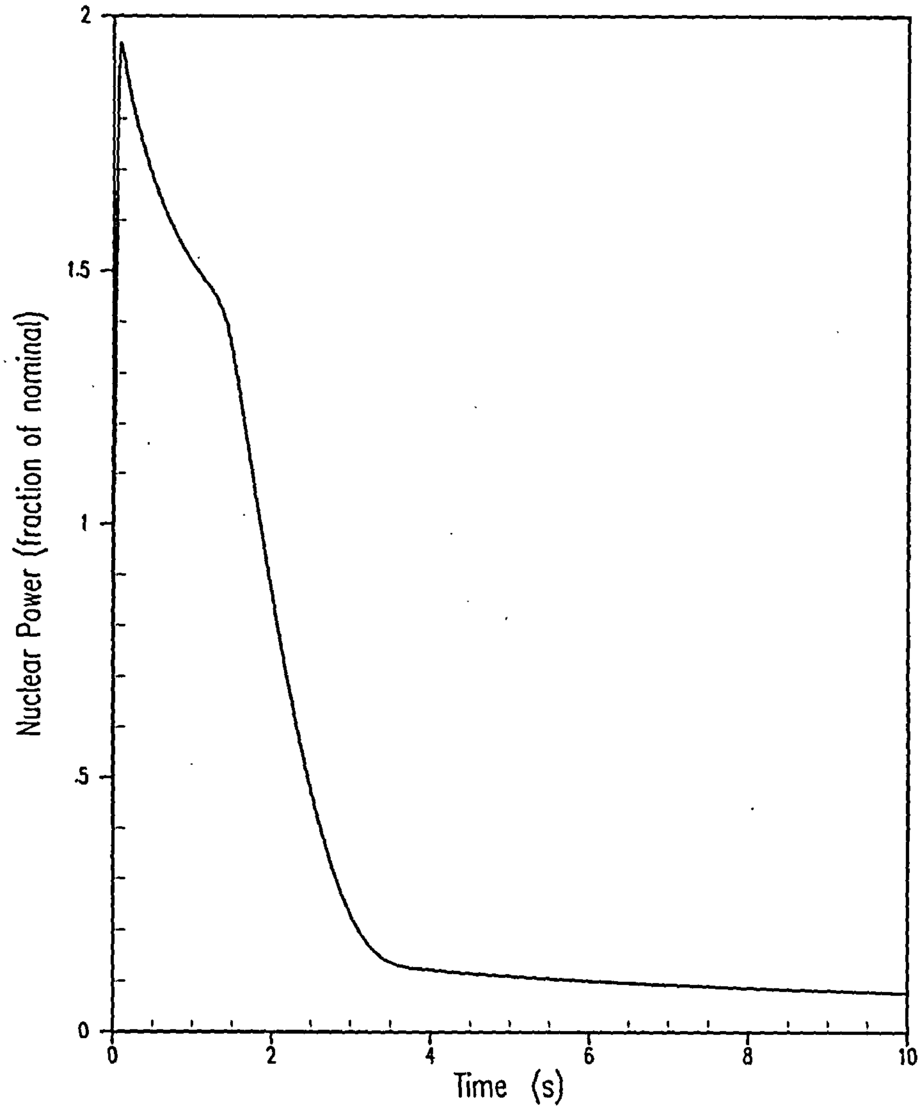

Figure 5.1.20-1 CEA Ejection Accident from Full Power Beginning of Cycle – Reactor Power versus Time

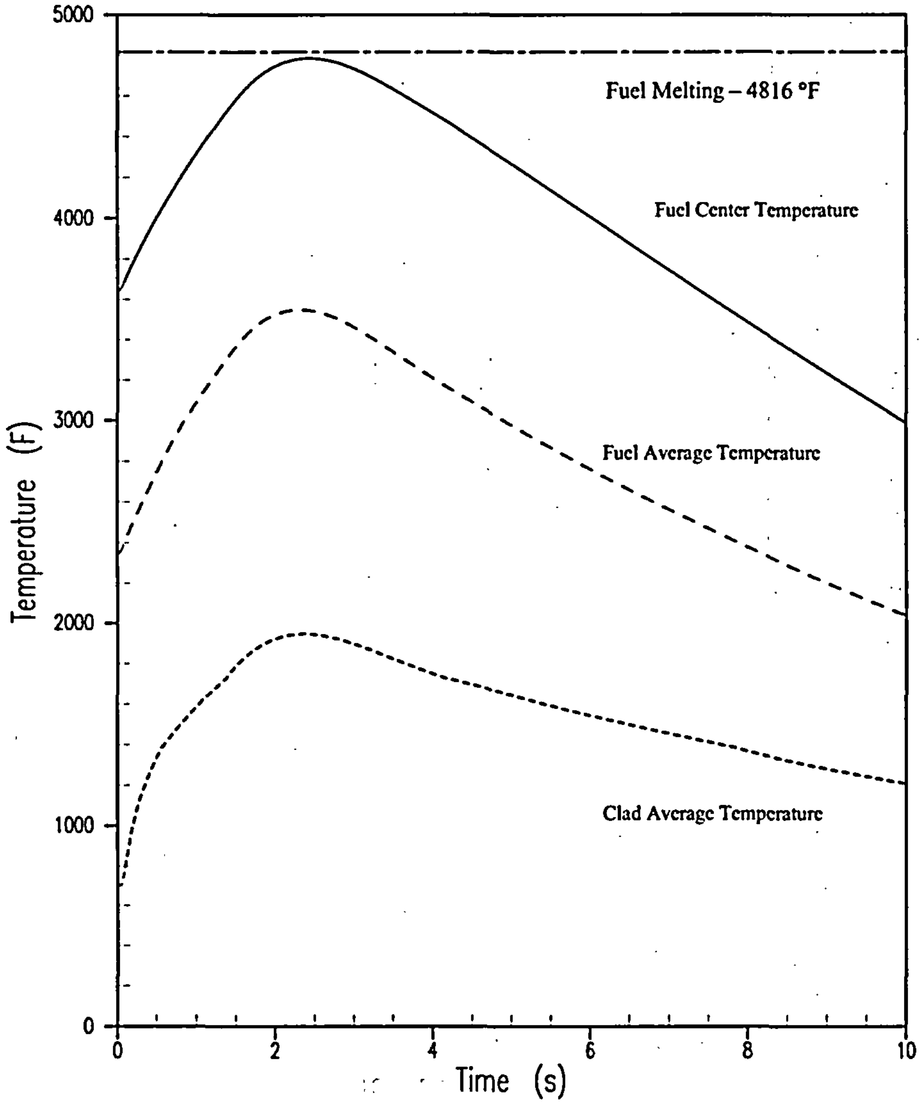

Figure 5.1.20-2 CEA Ejection Accident from Full Power Beginning of Cycle – Fuel and Cladding Temperatures versus Time

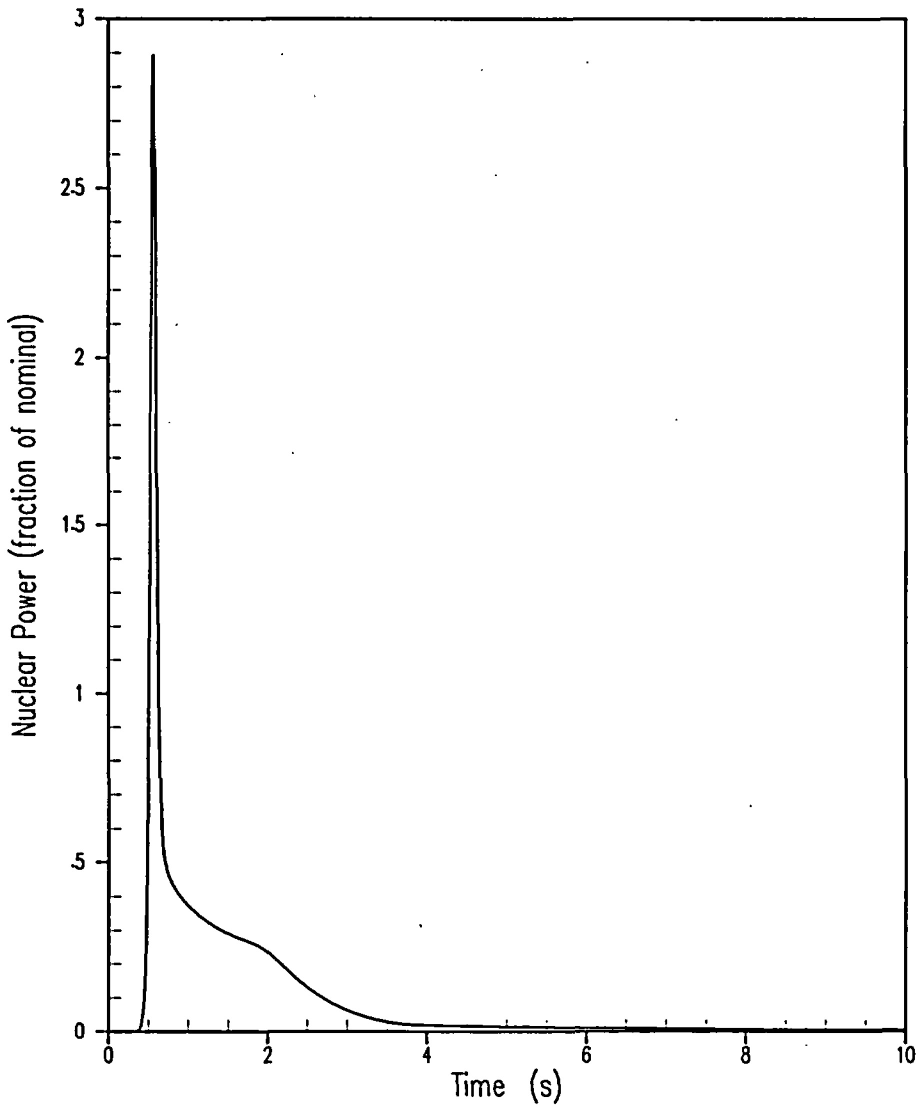

Figure 5.1.20-3 CEA Ejection Accident from Zero Power Beginning of Cycle – Reactor Power versus Time

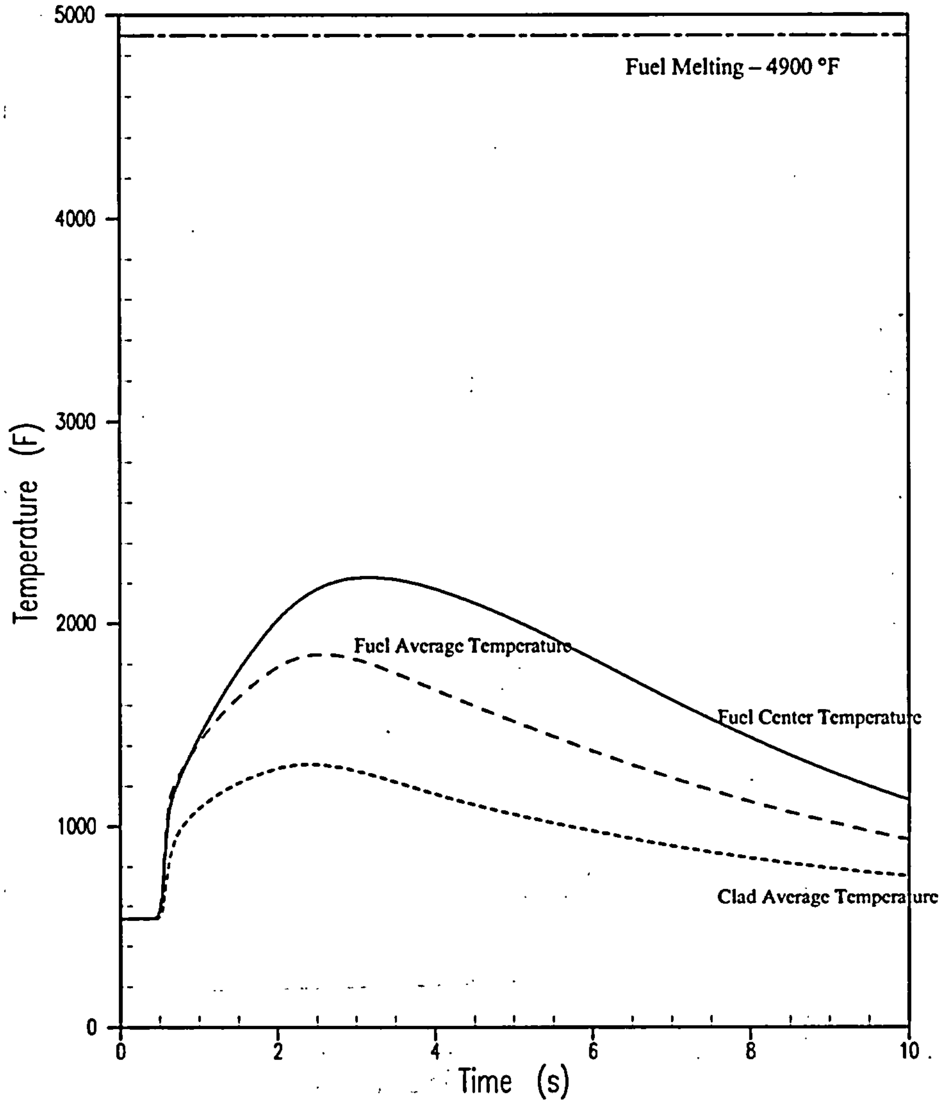

Figure 5.1.20-4 CEA Ejection Accident from Zero Power Beginning of Cycle – Fuel and Cladding Temperatures versus Time

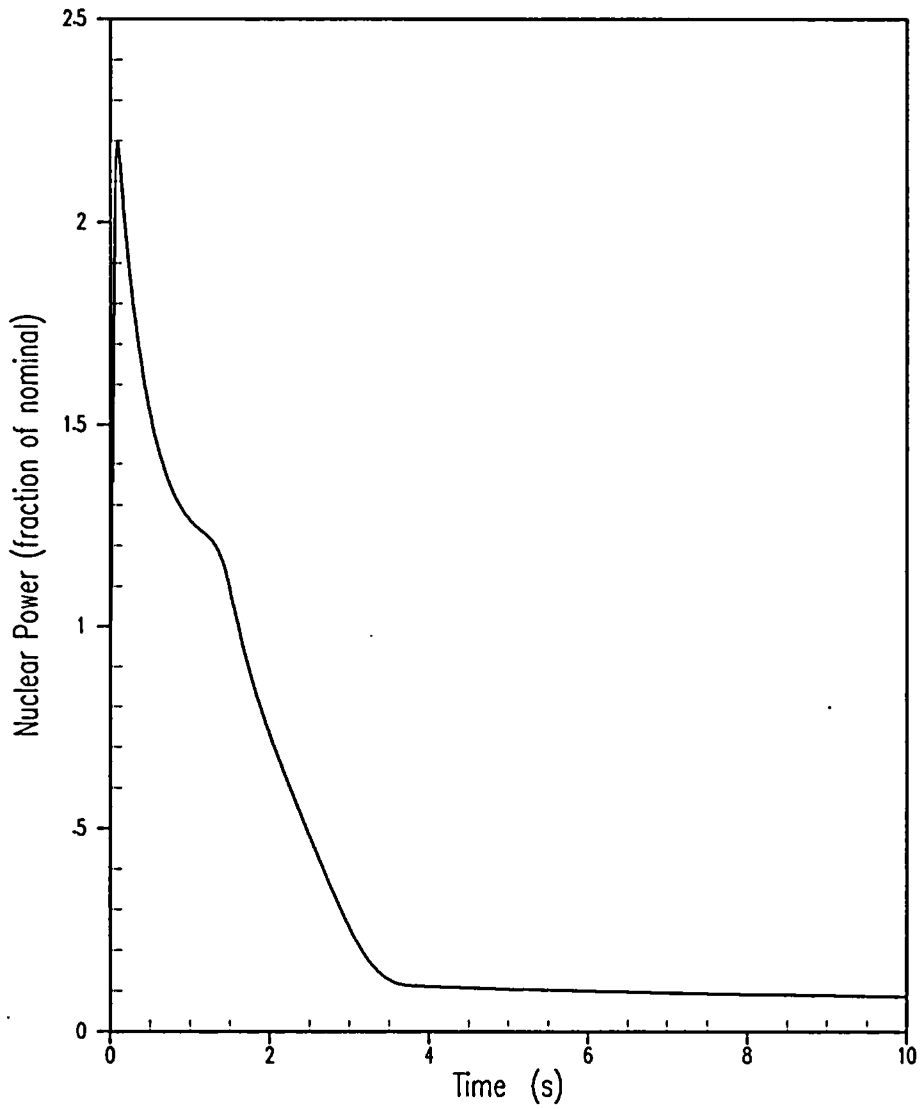

Figure 5.1.20-5 CEA Ejection Accident from Full Power End of Cycle – Reactor Power versus Time

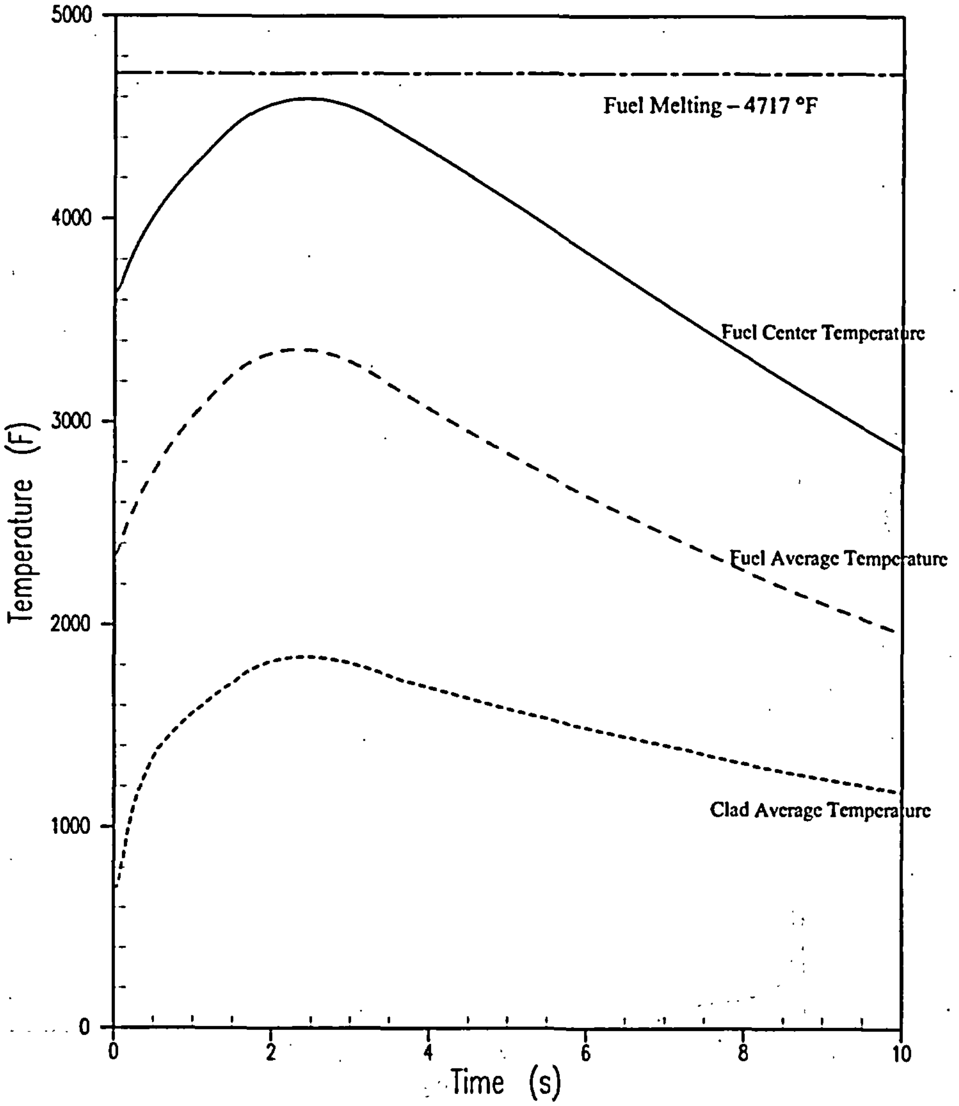

Figure 5.1.20-6 CEA Ejection Accident from Full Power End of Cycle – Fuel and Cladding Temperatures versus Time

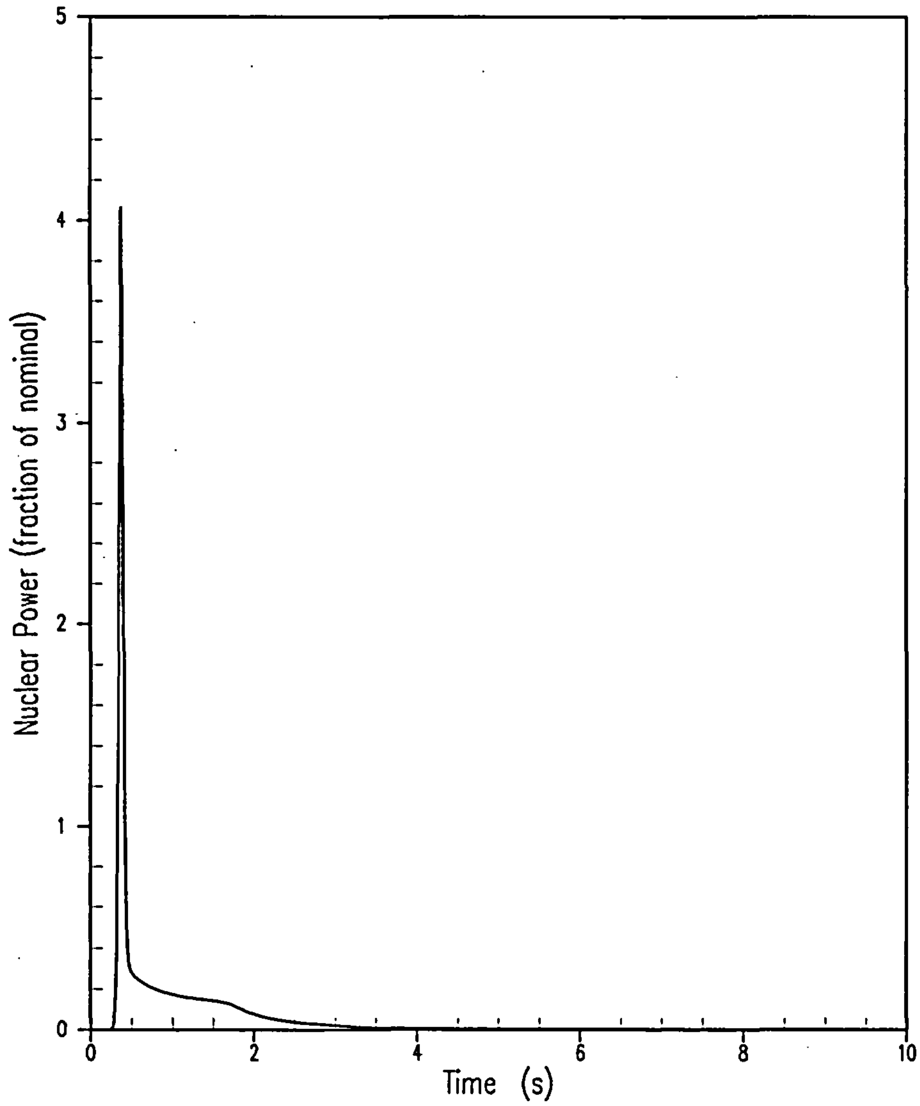

Figure 5.1.20-7 CEA Ejection Accident from Zero Power End of Cycle – Reactor Power versus Time

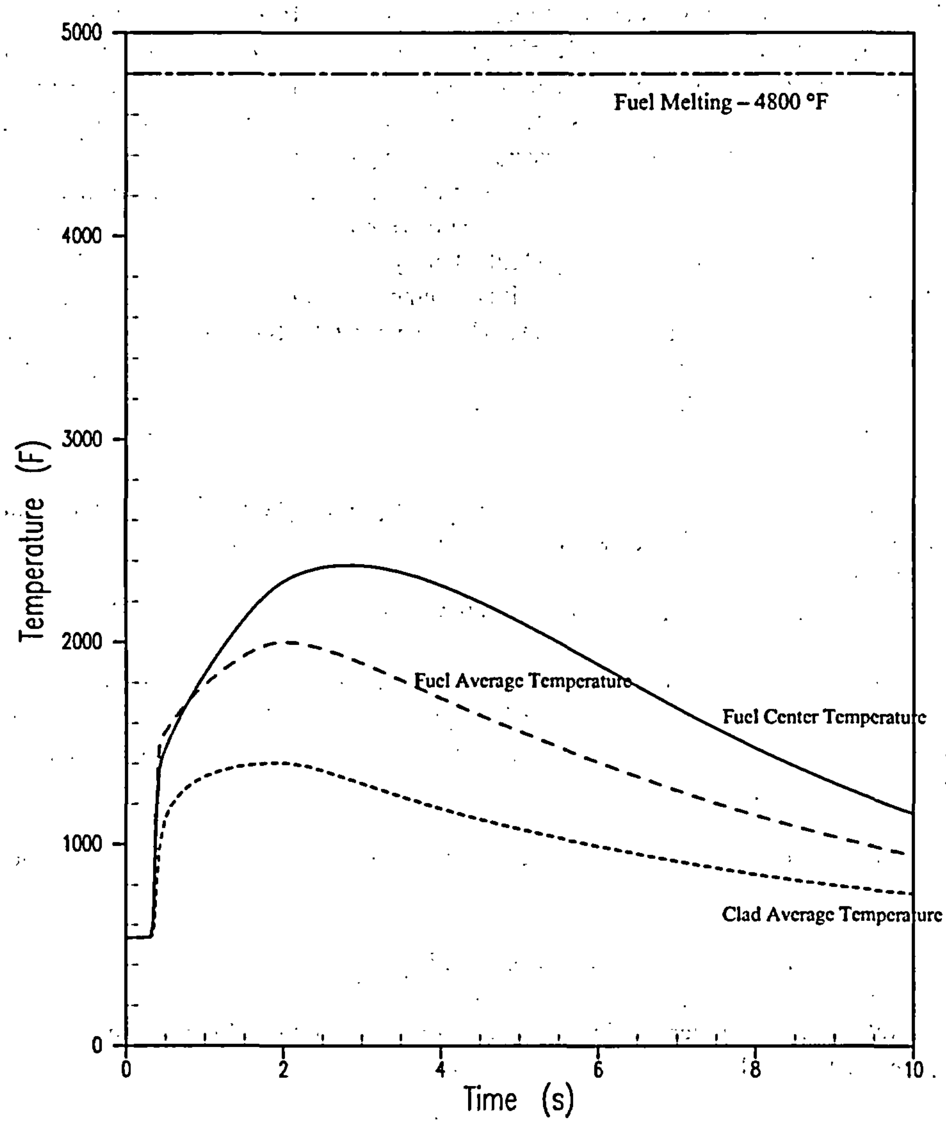

Figure 5.1.20-8 CEA Ejection Accident from Zero Power End of Cycle – Fuel and Cladding Temperatures versus Time

5.1.21 CVCS Malfunction

5.1.21.1 Accident Description

A CVCS Malfunction event that produces an unplanned increase in RCS inventory may be caused by operator error or a failure in the pressurizer level transmitter, which causes an erroneous low-low level signal. The generated signal will be transmitted to the controller, which responds by actuating a second charging pump and closing the letdown flow control valve to its minimum flow position. The CVCS Malfunction is assumed to occur without increasing or diluting the primary coolant initial boron concentration. With the mismatch between letdown and charging flow, the pressurizer mixture level and pressure increase. The pressurizer sprays mitigate the pressure increase. The operators are alerted to the event either by a high pressurizer pressure trip (HPPT) or by the pressurizer high level alarm (PHLA). Twenty minutes after either HPPT or the PHLA, it is assumed that the operators mitigate the event by reducing charging flow or restoring letdown flow. The case of a CVCS malfunction that produces a boron dilution is presented in Section 5.1.19.

5.1.21.2 Method of Analysis

The CVCS malfunction at-power transient is analyzed by employing the detailed digital computer code RETRAN (References 2 and 3). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The analysis of the CVCS Malfunction event has been performed to demonstrate that operators have a sufficient amount of time to preclude pressurizer filling following a high pressurizer level alarm. To maximize the peak pressurizer mixture volume following the high pressurizer level alarm, the following assumptions are made:

1. An initial core power of 2754.4 MWt, based on rated power of 2700 MWt and 2% uncertainty is assumed.

2. 30% of the steam generator U-tubes are assumed to be plugged.

3. Initial values of pressurizer pressure, vessel average temperature (Tavg) and pressurizer level are provided in Table 5.1.21-1.

4. Maximum charging flow is 49 gpm per pump for a total of 98 gpm for two charging pumps. This is reduced by 4 gpm for the reactor coolant pump bleedoff flow and results in a total charging flow assumption of 94 gpm.

5. The initiating event is an erroneous low-low level signal that actuates a second charging pump and closes the letdown flow control valve to the minimum position. This is consistent with the current event description in Section 15.5.3.2.2 of the UFSAR (Reference 1).

6. The assumed single failure is the complete closure of the letdown flow control valve that occurs concurrently with the start of the second charging pump and is consistent with the current event description in Section 15.5.3.2.2 of the UFSAR.

7. The charging flow boron concentration is assumed to be equal to the initial RCS boron concentration.

8. The pressurizer high level alarm setpoint is assumed to be 70% of tap span.

9. The pressurizer safety valves are assumed to have a -2% tolerance that corresponds to an opening setpoint of 2450.3 psia.

10. Operator action is assumed to occur at 20 minutes after the PHLA actuates and is consistent with the current UFSAR, Section 15.5.3.2.2.

11. Maximum reactivity feedback conditions are assumed.

12. Pressurizer sprays and heaters are assumed in the automatic mode.

The CVCS Malfunction is a Condition II event and is analyzed to show that the Condition II limits (specifically, generation of a more serious plant condition, pressurizer fill) are not exceeded. The applicable Condition II acceptance criteria are discussed as follows.

With respect to peak RCS and main steam system pressures, the CVCS Malfunction at-power event is bounded by the Loss of Condenser Vacuum event described in Section 15.1.10 of the UFSAR, which is analyzed with assumptions that are made to conservatively calculate the RCS and main steam system pressure transients.

With respect to the fuel damage acceptance criterion, the CVCS Malfunction at-power event is bounded by the CEA Bank Withdrawal at-power event described in Section 15.1.16 of the UFSAR. During the CVCS Malfunction there are very small increases in core power and RCS temperatures, and a very small change in RCS mass flow. The RCS pressure increase is limited by the pressurizer sprays and offsets any negative effects due to the minimal changes in other DNB-related parameters.

With respect to the acceptance criterion of not generating a more serious plant condition, this is satisfied by demonstrating that the PSVs do not discharge water. Due to the mismatch between charging and letdown flows, the pressurizer water volume increases during the CVCS Malfunction at-power event. Operator action is required to preclude water discharge through the PSVs.

With respect to the fission product barrier failure criterion, this is met by demonstrating that the DNB design basis is satisfied. Thus, there is no loss of function of any fission product barrier for the CVCS Malfunction at-power event.

5.1.21.3 Results

The RETRAN code analysis assumptions are listed in Table 5.1.0-2 and in the section above. Table 5.1.21-1 lists the Sequence of Events. Figures 5.1.21-1 to 5.1.21-5 present the key transient parameters during the event.

The transient is assumed to start at 10 seconds, and the pressurizer high level alarm alerts the operators at 337.7 seconds. At 1537.7 seconds (20 minutes later), the pressurizer mixture volume is 1508.3 ft^3, which corresponds to 10.7 ft^3 of margin to filling the pressurizer water solid.

The maximum pressurizer pressure is 2222.1 psia at 967.5 seconds. This is well below the pressurizer safety valves opening setpoint of 2450.3 psia.

5.1.21.4 Conclusions

The results demonstrate that the pressurizer volume does not become water solid prior to 20 minutes after the PHLA is actuated. The CVCS Malfunction at-power event is assumed to be mitigated by operator action prior to the pressurizer filling and no water is discharged through the PSVs. Thus, it is concluded that this transient does not generate a more serious plant condition.

5.1.21.5 References

1. St. Lucie Unit 2 Updated Final Safety Analysis Report," Amendment 14, Docket No. 50-389.

2. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April, 1999.

3. McFadden, J. H., et al, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Systems," EPRI NP-1850-CCMA.

Table 5.1.21-1 Sequence of Events for the CVCS Malfunction Event		
Time	**Event**	**Setpoint or Value**
10.0	Erroneous low-low level pressurizer level control system signal, Second charging pump starts, Letdown flow is isolated	---
337.7	Pressurizer high level alarm occurs	70% of tap span
967.5	Maximum pressurizer pressure	2222.1 psia
1537.7	Operator action occurs to mitigate the event.	---

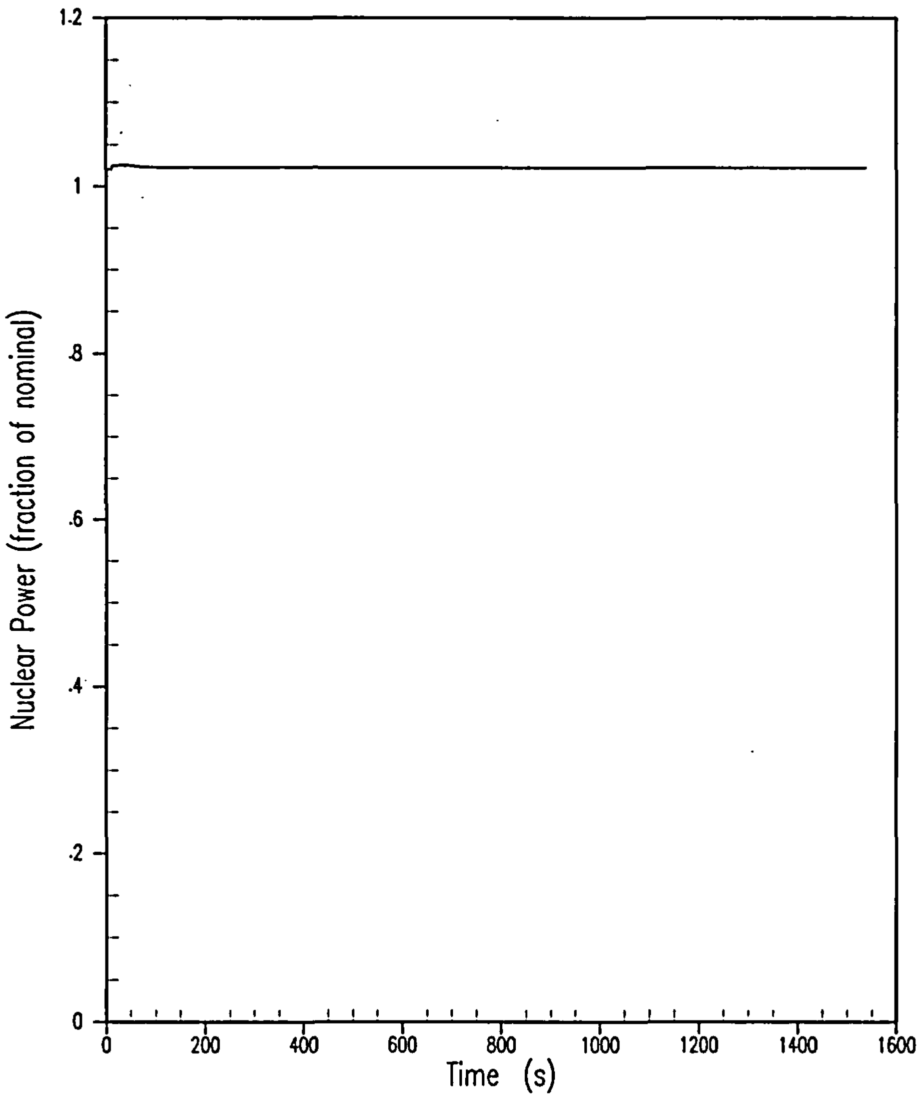

Figure 5.1.21-1 CVCS Malfunction at Power – Nuclear Power

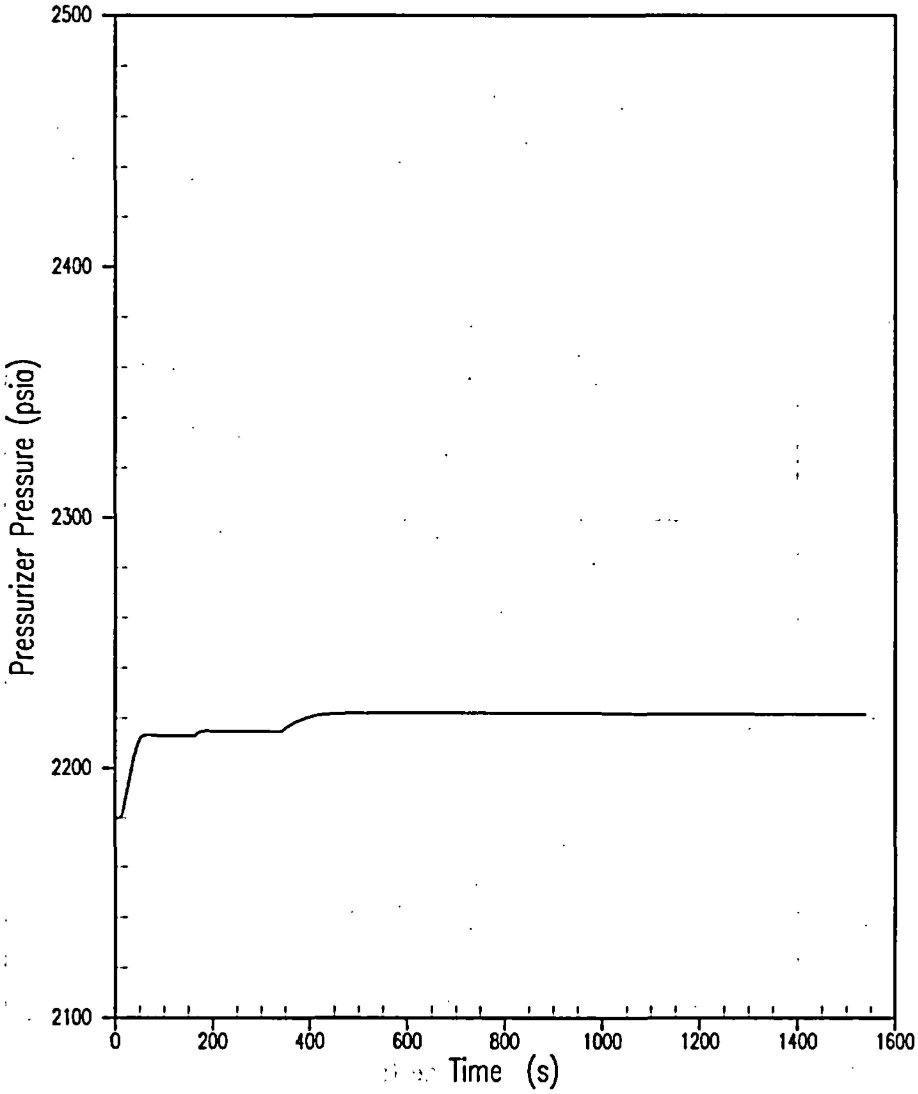

Figure 5.1.21-2 CVCS Malfunction at Power – Pressurizer Pressure

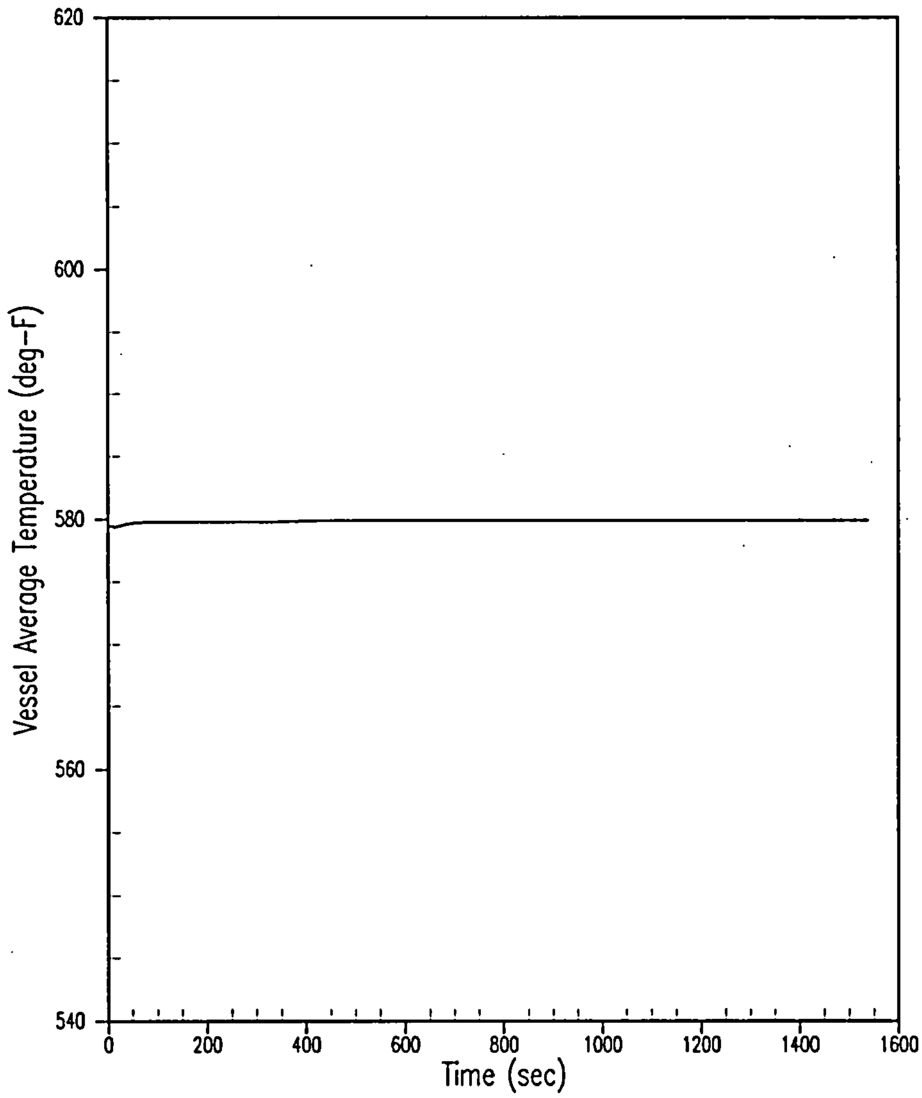

Figure 5.1.21-3 CVCS Malfunction at Power – Vessel Tavg

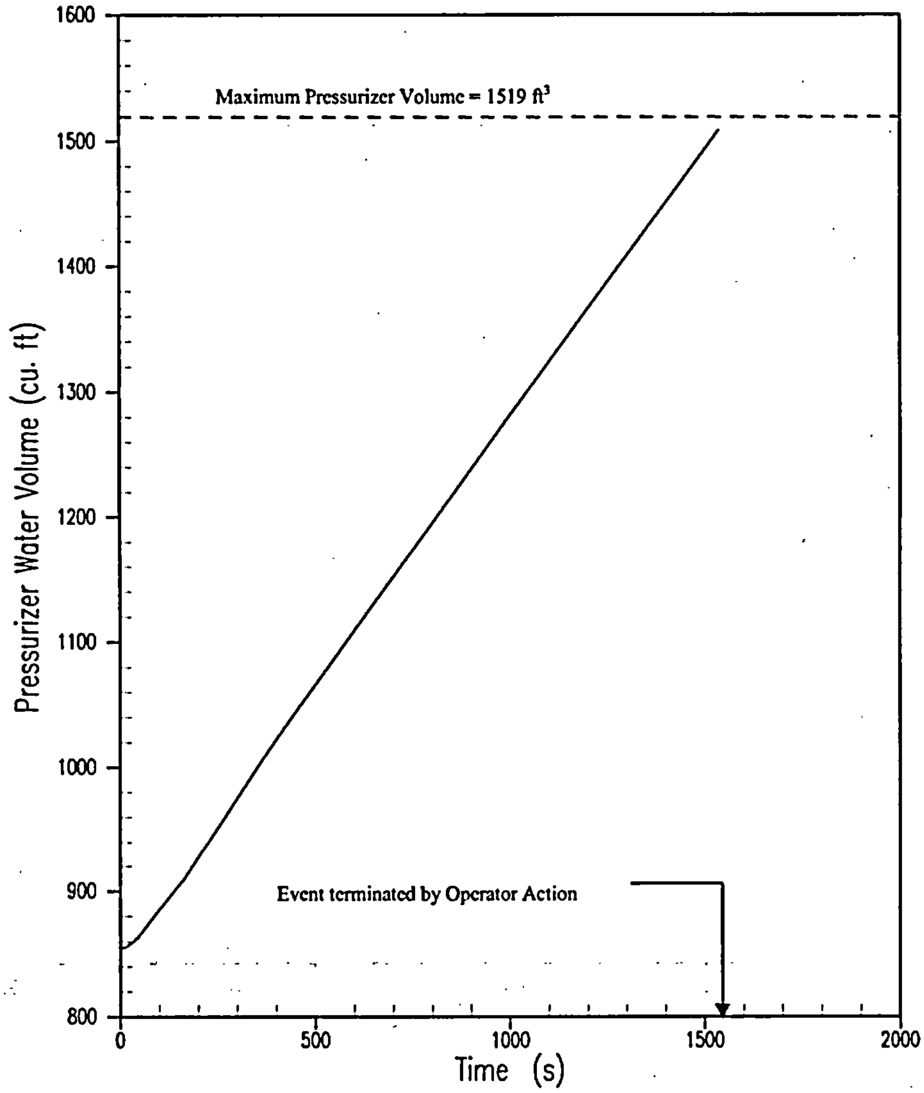

Figure 5.1.21-4 CVCS Malfunction at Power – Pressurizer Water Volume

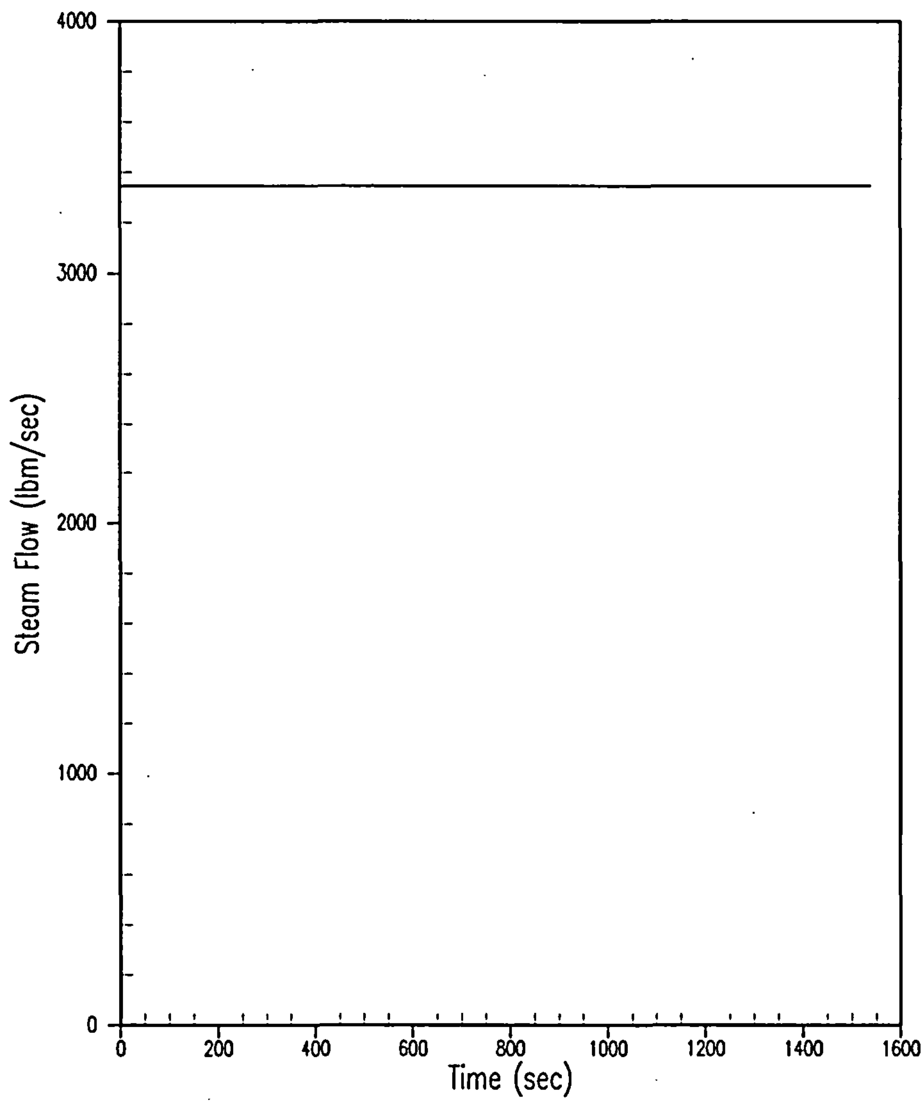

Figure 5.1.21-5 CVCS Malfunction at Power – Steam Flow

5.1.22 Pressurizer Pressure Decrease – Inadvertent Opening of the Pressurizer Relief Valves

5.1.22.1 Accident Description

An accidental depressurization of the RCS could occur as a result of one of the following: an inadvertent opening of both of the pressurizer PORVs, an inadvertent opening of a single PSV, or a malfunction of the pressurizer spray system. Since a PSV is sized to relieve approximately half the steam flow rate of a PORV, and the pressurizer spray system, even if fully open, cannot depressurize the RCS at the rate of two open PORVs, the most severe core conditions are associated with an inadvertent opening of both of the PORVs. It is assumed that a mechanical failure, spurious actuation signal, or unanticipated operator action will cause the opening of both PORVs. Initially, the event results in a rapidly decreasing RCS pressure which could reach the hot-leg saturation pressure if a reactor trip did not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease power via the moderator density feedback. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor core is protected against fuel damage by the TM/LP trip function.

An inadvertent opening of both of the pressurizer relief valves is classified as an ANS Condition II event, a fault of moderate frequency. It should be noted that a stuck open PSV is considered a more serious Condition III event. However, in the case of St. Lucie Unit 2, the relief capacity of a PORV far exceeds that of a PSV. Thus, the assumption of the opening of both PORVs and demonstration that the more restrictive Condition II acceptance criteria are met removes the PSV scenario from concern.

5.1.22.2 Method of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code RETRAN (References 1 and 2). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the revised thermal design procedure described in Reference 3.

Plant characteristics and initial conditions are discussed in Section 5.1.0. In order to give conservative results in calculating the DNBR during the transient, the following assumptions are made:

1. The initial reactor power and RCS temperature are assumed to be at their nominal values, the initial RCS flow rate is assumed at a value consistent with the minimum measured flow rate and the initial RCS pressure is assumed at a value consistent with minimum value allowed by the plant technical specifications. Uncertainties in initial conditions are statistically included in the calculation of the DNBR limit as described in Reference 3.

2. A least negative moderator temperature coefficient of reactivity corresponding to 0.0 pcm/°F is assumed. The spatial effect of void due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

3. A least negative Doppler power coefficient is assumed such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

Plant systems and equipment which are necessary to mitigate the effects of the accident are discussed in Section 5.1.0.

The automatic rod control system is not modeled as it is disabled at the plant. The RPS functions to trip the reactor on the appropriate signal. No single active failure will prevent the RPS from functioning properly.

5.1.22.3 Results

The system response to an inadvertent opening of both PORVs is shown in Figures 5.1.22-1 through 5.1.22-4. Figure 5.1.22-1 illustrates the nuclear power transient following the depressurization. Nuclear power remains relatively constant while pressurizer pressure decreases from the initial value until reactor trip occurs on the floor of the TM/LP trip. The pressure transient and average coolant temperature transient following the accident are given in Figures 5.1.22-2 and 5.1.22-3, respectively. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 5.1.22-4. The DNBR remains above the limit value of 1.42 throughout the transient.

The calculated sequence of events for the inadvertent opening of both PORVs is shown in Table 5.1.22-1.

5.1.22.4 Conclusion

The results of the analysis show that the TM/LP RPS signal provides adequate protection against the accidental depressurization of the RCS. No fuel or clad damage is predicted for this accident. All acceptance criteria are satisfied.

5.1.22.5 References

1. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April, 1999.

2. McFadden, J. H., et al, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCMA.

3. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

Table 5.1.22-1 Pressurizer Pressure Decrease – Inadvertent Opening of the Pressurizer Relief Valves Sequence of Events and Transient Results	
Event	**Time (seconds)**
Inadvertent Opening of Both PORVs	10.1
Reactor Trip on TM/LP	26.2
Rod Motion Begins	26.9
Time of Minimum DNBR	27.5
Minimum DNBR	1.73

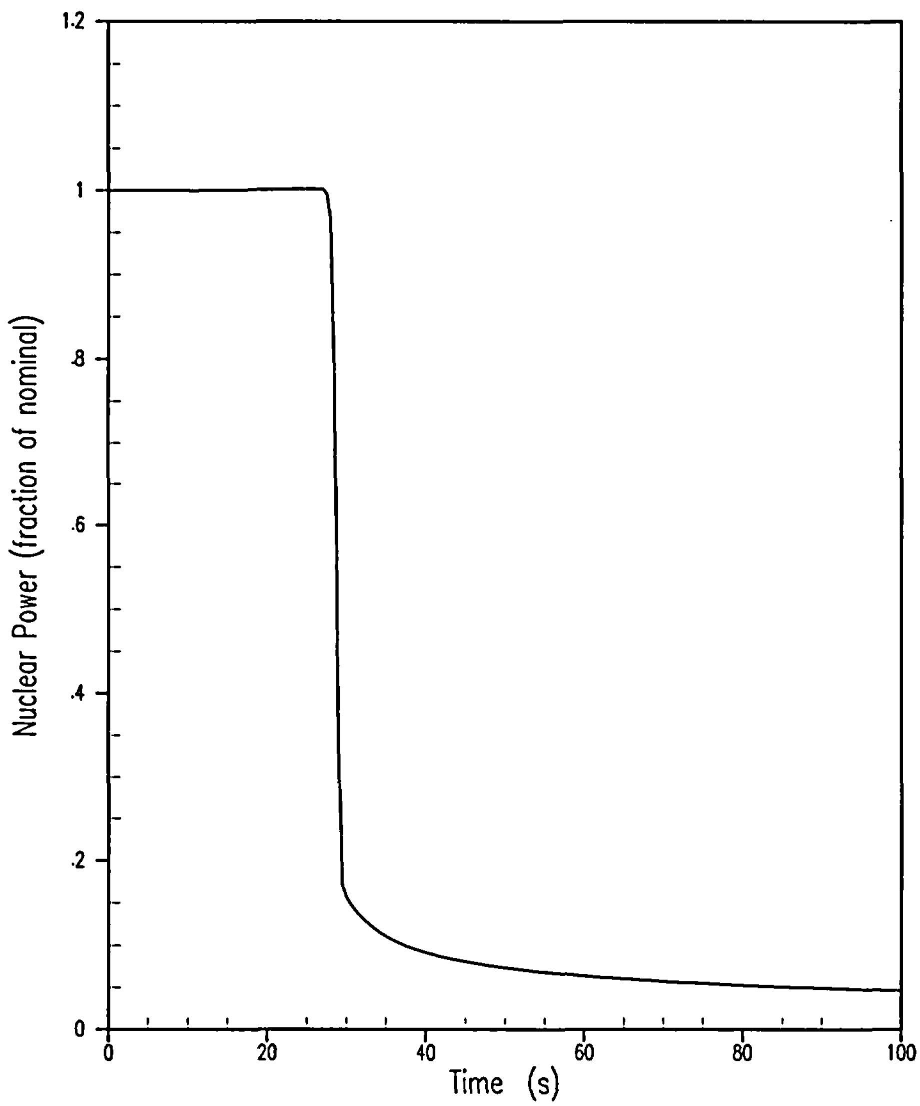

Figure 5.1.22-1 Nuclear Power for an Accidental Depressurization of the Reactor Coolant System Event

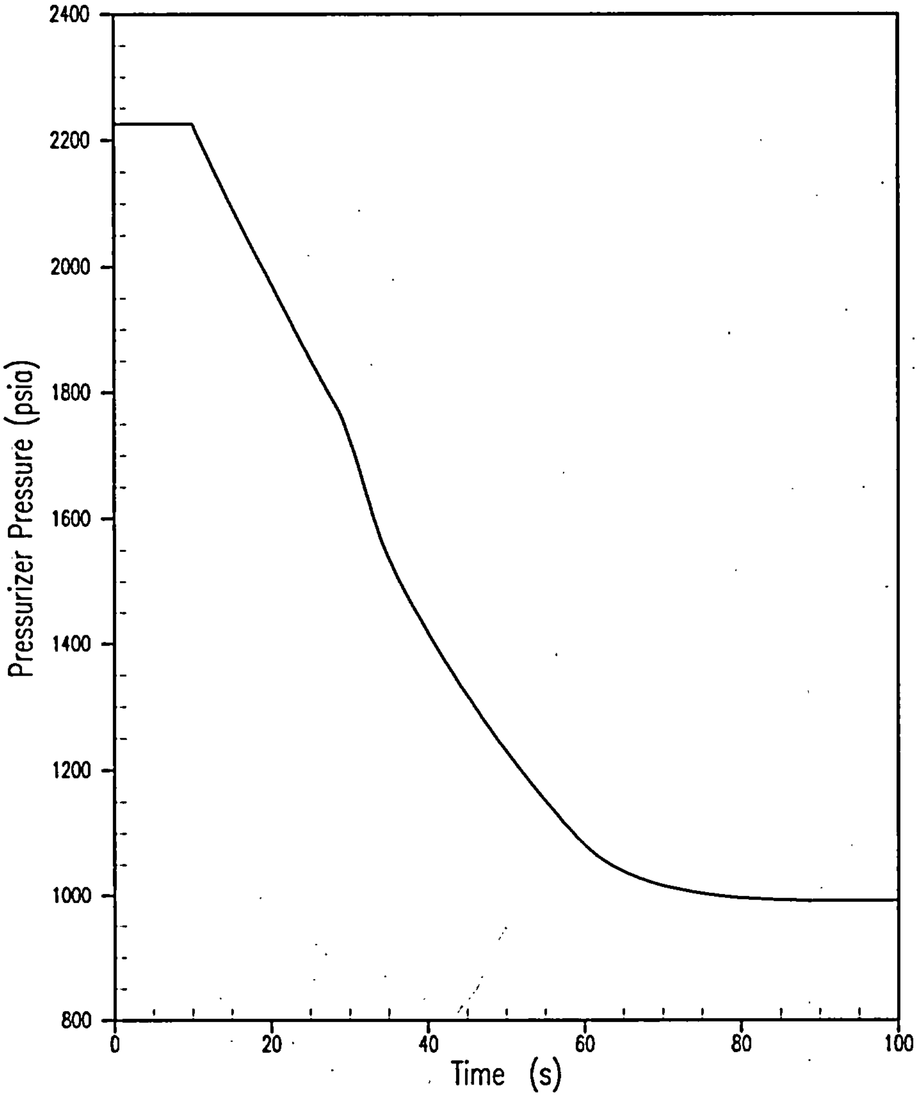

Figure 5.1.22-2 Pressurizer Pressure for an Accidental Depressurization of the Reactor Coolant System Event

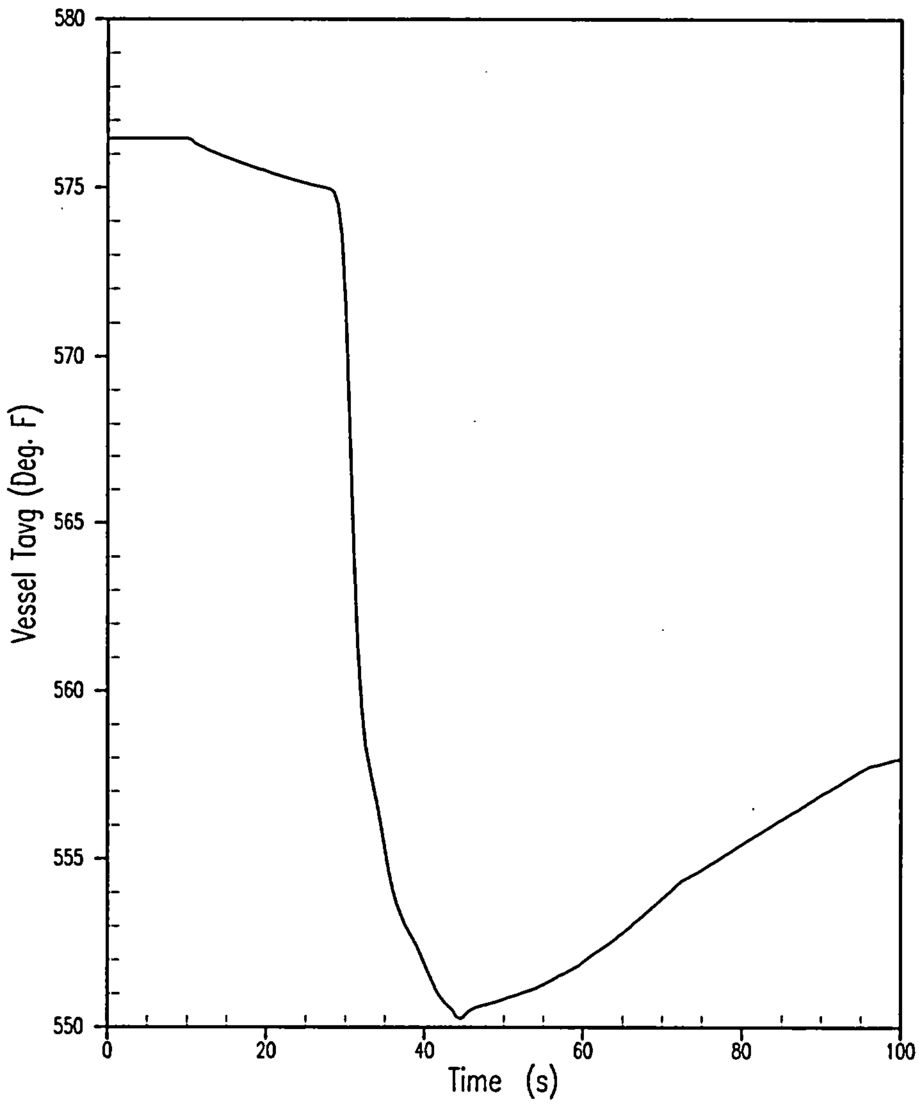

Figure 5.1.22-3 Vessel Tavg for an Accidental Depressurization of the Reactor Coolant System Event

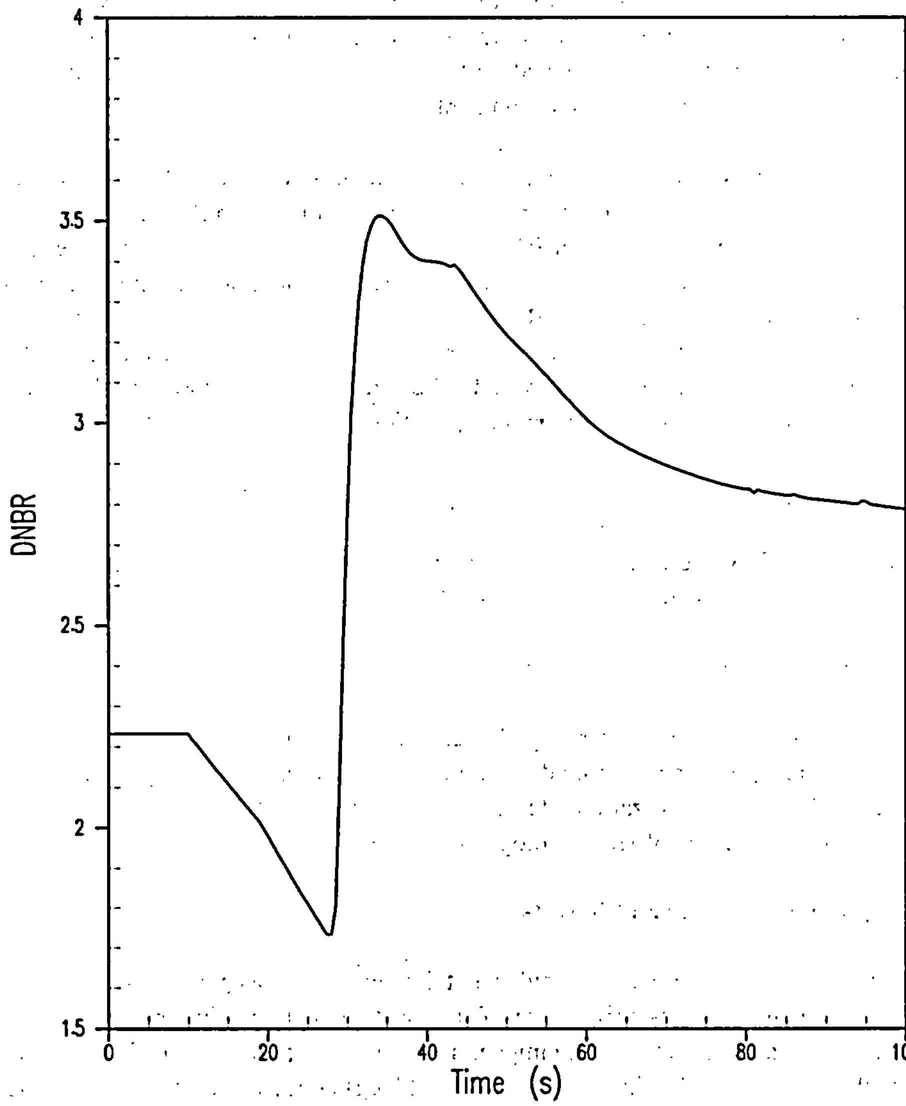

Figure 5.1.22-4 DNBR for an Accidental Depressurization of the Reactor Coolant System Event

5.1.23 Primary Line Break Outside Containment

5.1.23.1 Introduction

A Primary Line Break Outside Containment Event may result from a break in a letdown line, instrument line, or sample line. The double-ended break of the letdown line outside containment upstream of the outside containment isolation valve was selected for this analysis since it is the largest line and results in the largest release of reactor coolant to the environment.

The Primary Line Break Outside Containment provided in the St. Lucie 2 UFSAR (Reference 1), Section 15.6.3.1.7, was evaluated to account for a decrease in the minimum RCS flow from 355,000 gpm to 335,000 gpm, and an increase in steam generator tube plugging to 30%. Because this event analysis is not within the scope of the WCAP-9272 reload methodology, this evaluation remains based on the methods of the current analysis of record (AOR) as presented in Reference 1, Section 15.6.3.1.7 which included use of the CESEC transient analysis code. Also, the scope of this evaluation is limited to assessing the impact on the mass releases that become the source terms for dose calculations previously determined for this event as presented in Reference 1, Section 15.6.3.1.7.

5.1.23.2 Input Assumptions

Other than reduced RCS flow and increased tube plugging, there are no adverse plant changes relative to the key parameters identified in the AOR, and Reference 1, Table 15.6.3.1-7.

5.1.23.3 Acceptance Criteria

The Primary Line Break event does not directly challenge peak linear heat rate, peak RCS pressure, or peak secondary pressure criteria; and is bounded by the inadvertent opening of a pressurizer relief valve event with respect to RCS depressurization rate, and approach to the DNB criterion. Primary Line Break has been evaluated with respect to offsite radiological doses.

5.1.23.4 Description of Analyses and Results

An evaluation of the Primary Line Break Outside Containment event was performed to determine the impact of the two plant changes presented above. This was done by reviewing the key input and assumptions of the AOR and determining the impact of any adverse changes. The leak rate from the letdown line break is determined by the upstream (i.e., cold leg) temperature and pressure, and neither will be affected by an RCS flow change or tube plugging. Temperature is constant prior to reactor trip and the range of initial cold leg temperatures is not being changed by the increase in SGTP. The pressure decreases prior to reactor trip on low pressurizer pressure, with the pressure determined by the pressurizer conditions and the decrease in reactor coolant volume caused by the break. The pressurizer conditions, including the range of initial pressure and liquid level, charging flow, and heater capacity are not adversely changed by the increase in SGTP.

With no change to the pre-trip sequence of events, the time of trip will not be affected since the low pressurizer pressure trip setpoint is not being changed by an increase in SGTP.

Following reactor trip the leak rate is conservatively held constant at 45 lbm/sec for 10 minutes, and therefore, is independent of small variations in cold leg temperature and pressure. Any impact of initial RCS flow and tube plugging on post-trip cold leg temperature and pressure would be small, since cold leg temperature drops and equilibrates to a value determined by the MSSV opening setpoint (which is not changed by an increase in SGTP). The post-trip pressure decrease is caused by the coolant contraction as the RCS average temperature also drops to a value determined by the MSSV opening setpoint. The amount of contraction / depressurization will be slightly greater as initial average temperature increases due to reduced RCS flow, and will diminish slightly if the reactor coolant mass is decreased due to tube plugging. Both effects are small, and offsetting.

The evaluation determined that the decrease in the minimum RCS flow from 355,000 gpm to 335,000 gpm, and an increase to 30% steam generator tube plugging have a negligible impact on this event.

5.1.23.5 Conclusions

With regard to coolant leakage, this evaluation showed that the previously reported UFSAR results (Reference 1, Section 15.6.3.1.7) remain valid.

5.1.23.6 References

1. St. Lucie Unit 2 Updated Final Safety Analysis Report, Amendment 14, Docket No. 50-389.

5.1.24 Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power

5.1.24.1 Introduction

The Steam Generator Tube Rupture (SGTR) with a Concurrent Loss of Offsite Power event provided in the St. Lucie Unit 2 UFSAR (Reference 5.1.24-1), Section 15.6.2.1.7 was evaluated to account for increased steam generator tube plugging to 30%.

5.1.24.2 Input Assumptions

The SGTR Event was analyzed assuming the initial conditions in Table 5.1.24-1. In addition, the following assumptions were made:

1. While the reactor is operating at full power prior to trip, the steam mixture containing reactor coolant fission products passes through the turbine and the condenser.

2. Following the reactor and turbine trip, all fission product activity that is released is via the MSSVs until operator action at 1800 seconds.

3. Within thirty minutes (1800 seconds) after the tube rupture occurs, the operator identifies the problem and isolates the affected steam generator. At this time, plant cooldown is initiated using the unaffected steam generator atmospheric dump valves (ADVs).

This analysis assumes that following the SGTR, the RCS pressure drops until reaching the TM/LP low pressurizer pressure trip setpoint, thus preventing violation of the DNBR specified acceptable fuel design limit (SAFDL). Crediting the earliest trip maximizes the offsite doses. Offsite power is assumed to be lost at time of trip. The RCS continues to depressurize and the pressurizer empties.

Subsequent to reactor trip, stored and fission product decay heat is dissipated by the reactor coolant and main steam systems. In the absence of forced reactor coolant flow (due to a loss of offsite power), convective heat transfer is supported by natural circulation. The increasing steam generator pressure results in steam release to atmosphere via the MSSVs. At 1800 seconds, the operators are assumed to isolate the affected steam generator and initiate plant cooldown through the unaffected steam generator atmospheric dump valves. With the availability of stand-by power, emergency feedwater (EFW) is automatically initiated on a low steam generator water level signal. To maximize the offsite dose results this analysis conservatively assumes that the emergency feedwater will be unavailable until 1800 seconds and the normal feedwater flow ramps down immediately following loss of offsite power.

All other input remained the same as that in Reference 1, Section 15.6.2.1.7. Table 5.1.24-1 documents the changes in input parameters due to the reduction in the minimum RCS flow.

5.1.24.3 Acceptance Criteria

The SGTR event was analyzed to assure that the following dose criteria are not exceeded:

- With an assumed preaccident iodine spike in the reactor coolant and with the highest worth control rod assumed to stick in the fully withdrawn position on scram, the calculated doses do not exceed the values of 10 CFR 100, and
- With an equilibrium iodine concentration corresponding to full power operation and an assumed accident generated iodine spike, the calculated doses do not exceed 10% of 10 CFR 100.

Steam generator or RCS pressure limits are not challenged by SGTR and the TM/LP trip assures the DNBR SAFDL is met for SGTR.

5.1.24.4 Description of Analysis/Evaluation And Results

This analysis was performed using the same version of CESEC-III NSSS code as was previously reported in the license submittal.

Table 5.1.24-2 provides the radiological exposure for the SGTR event.

Table 5.1.24-3 contains the sequence of events for the SGTR Event with a Loss of Offsite Power. Figures 5.1.24-1 through 5.1.24-13 show the NSSS response for the system parameters important in dose calculation.

5.1.24.5 Conclusions

The radiological releases calculated for the SGTR Event with a concurrent loss of offsite power are provided in Table 5.1.24-2.

Primary and secondary system pressures are well below upset pressure limits thus assuring the integrity of these systems.

5.1.24.6 References

1. St. Lucie Unit 2 Updated Final Safety Analysis Report, Amendment 14, Docket No. 50-389.

Table 5.1.24-1 Key Parameters Assumed for the Steam Generator Tube Rupture Event With a Loss Of Offsite Power		
Parameter	**Units**	**Value**
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	°F	553
Initial RCS Vessel Flow Rate	gpm	335,000
Number of SG Plugged Tubes	---	30%
Initial Steam Generator Pressure	psia	770
Steam Generator U-Tube Break Size	in^2	0.336
CEA Worth at Trip	%Δρ	-5.4
Initial Pressurizer Pressure	psia	2395

Table 5.1.24-2 Radiological Exposures as a Result of a Steam Generator Tube Rupture Event With A Loss of Offsite Power			
Case	**EAB Dose (1) (rem TEDE)**	**LPZ Dose (2) (rem TEDE)**	**Control Room Dose (2) (rem TEDE)**
SGTR pre-accident iodine spike	0.28	0.27	3.33
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	0.08	0.08	0.96
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) RG 1.183, Table 6
(4) 10CFR50.67

Table 5.1.24-3 Sequence of Events for the Steam Generator Tube Rupture with Loss of Offsite Power		
Time (sec)	**Event**	**Setpoint / Value**
0.0	Tube Rupture Occurs	---
379.2	Reactor Trip Signal on Floor of TM/LP, psia	2142
380.84	CEAs Begin to Drop	----
384.8	MSSVs open,* psia	970
391.8	Maximum SG Pressure, psia	996
487.0	Pressurizer Empties	----
490.8	SIAS Generated on Low Pressurizer Pressure, psia	1578
520.8	HPSI Pumps Reach Full Speed	----
1800	1. Operator Borates to Cold Shutdown Concentration	
	2. Operator Isolates Affected SG by Closing MSIV	
	3. Operator Activates ADVs (Intact SG) to Commence Cooldown of RCS	
	Steam Releases via Turbine to Condenser Prior to Trip, lbm	1255628
	MSSVs Close, Affected/Intact SG Steam Releases, lbm	78786 / 77767
	Leakage Before Trip, lbm	20319
	Leakage from Reactor Trip to 1800 seconds, lbm	53021
	RCS to Affected SG Total Tube Leakage, lbm	73340
	RCS to Intact SG Total 2 hour T/S Leakage, lbm	723.2
	Total Intact SG Steam Releases via ADV, lbm	572,026
	RCS to Intact SG Total 8 hour T/S Leakage, lbm	2,929
	Total Intact SG Steam Releases via ADV, lbm	1,479,854

* MSSVs Cycle Until Operator Activates ADVs @ 1800 Sec

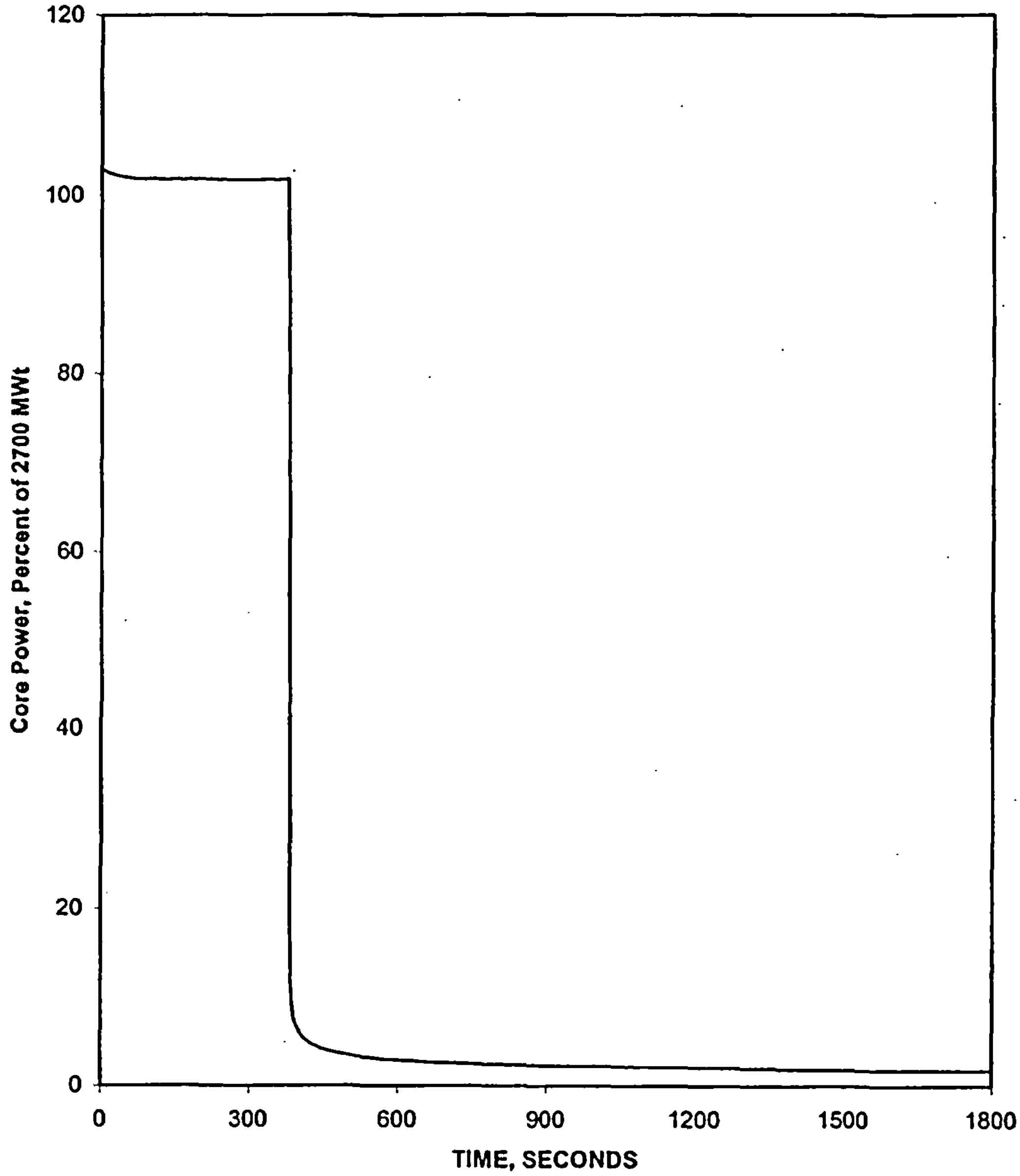

Figure 5.1.24-1 Steam Generator Tube Rupture with Loss of Offsite Power – Core Power vs. Time

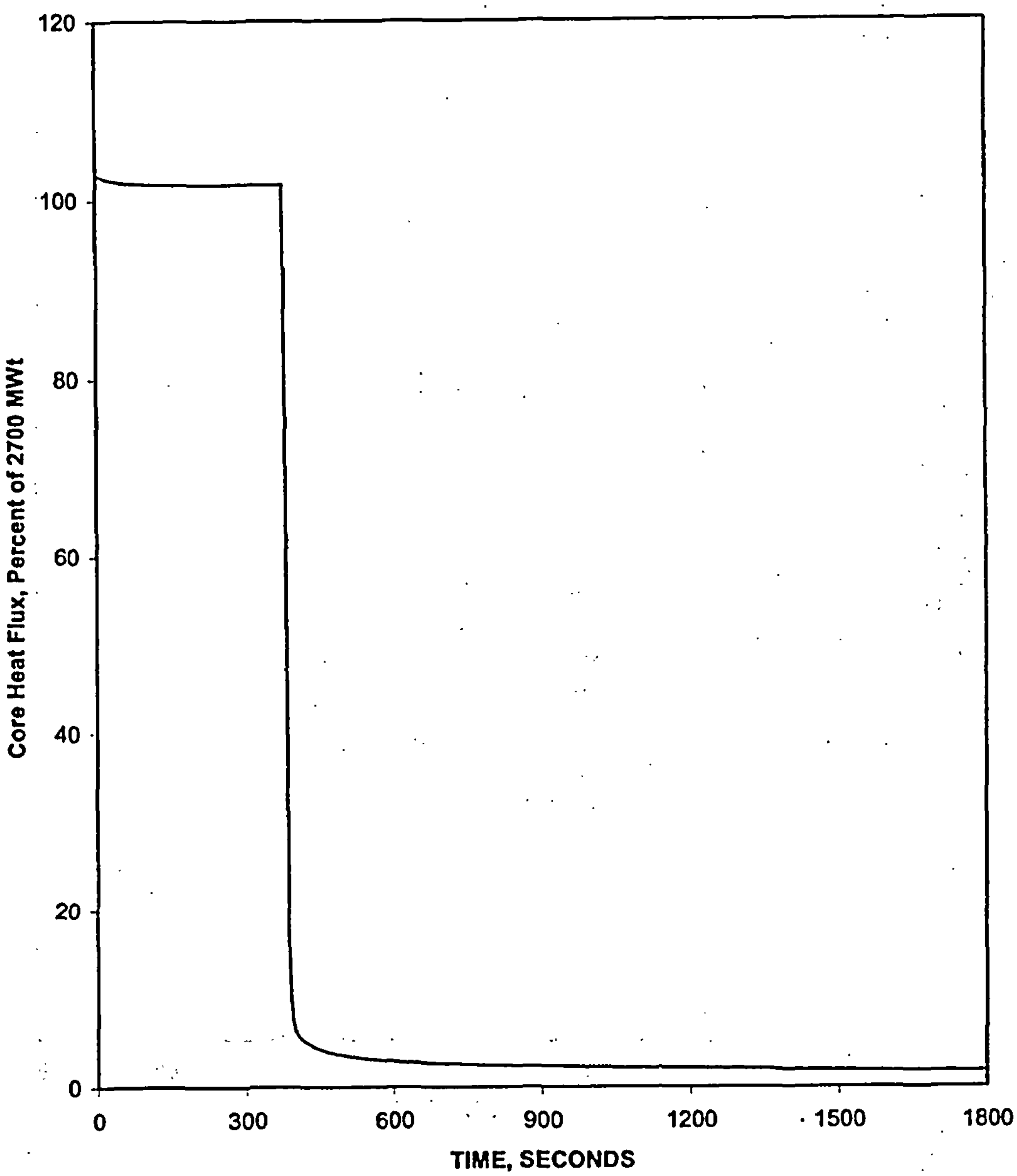

Figure 5.1.24-2 Steam Generator Tube Rupture with Loss of Offsite Power – Core Heat Flux vs. Time

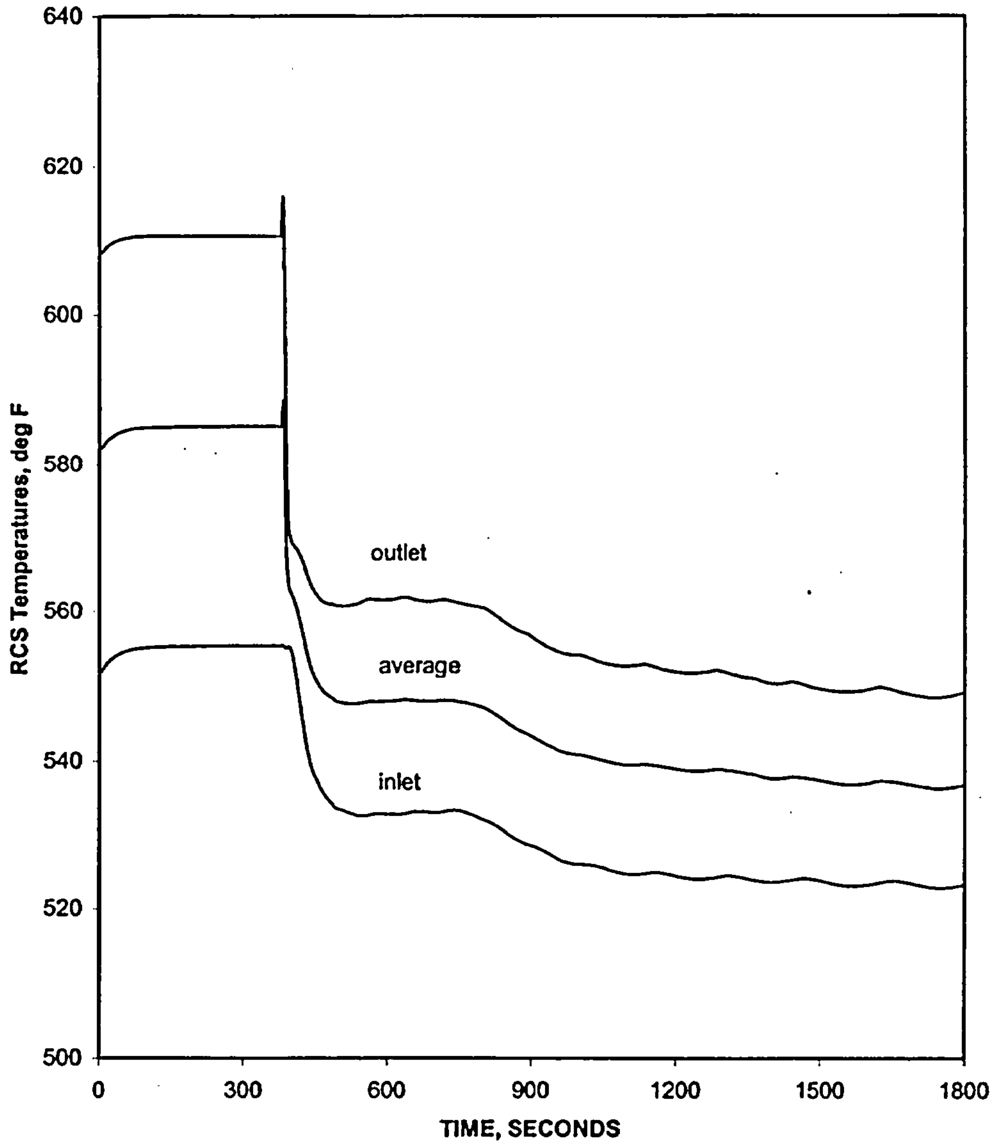

Figure 5.1.24-3 Steam Generator Tube Rupture with Loss of Offsite Power – Reactor Coolant System Temperatures vs. Time

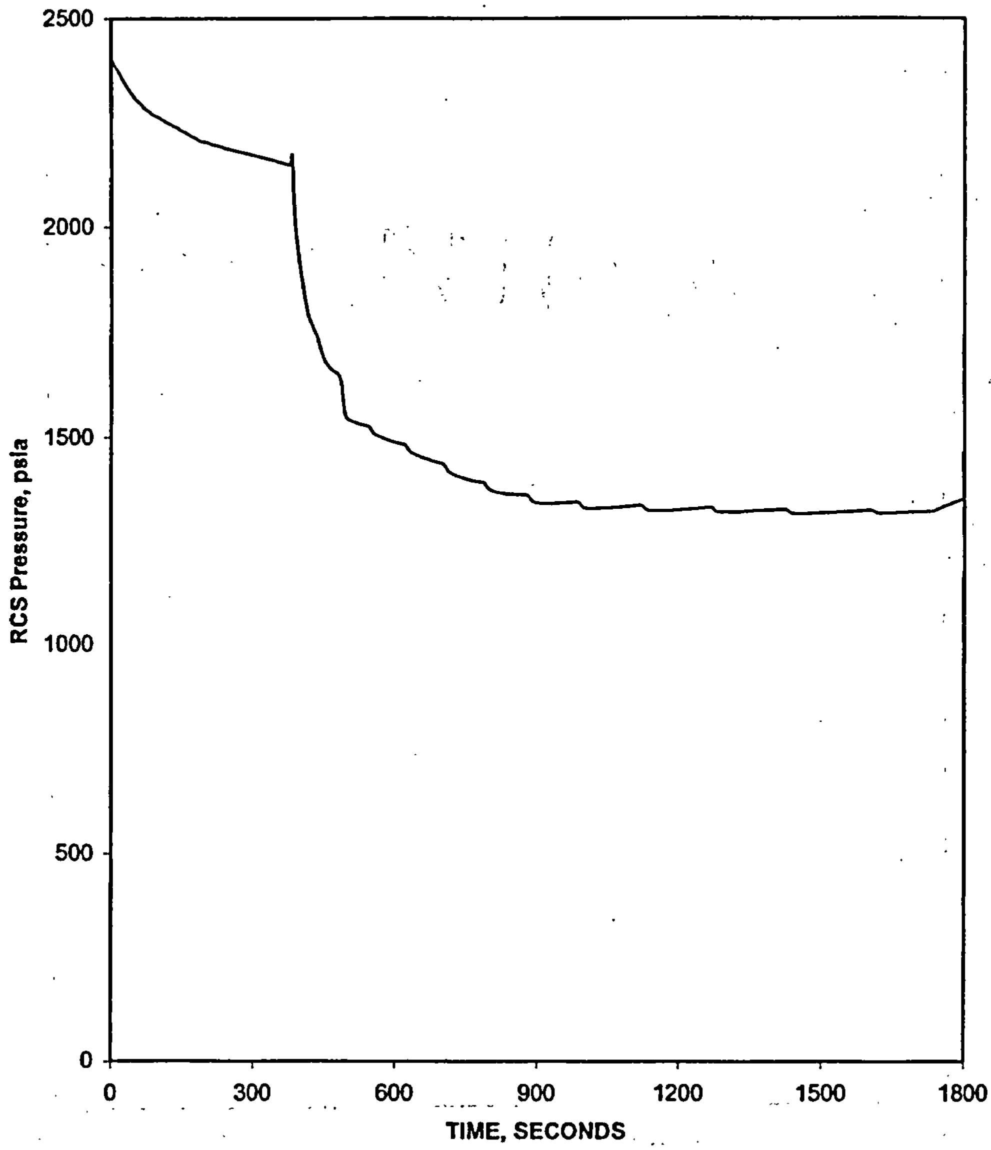

Figure 5.1.24-4 Steam Generator Tube Rupture with Loss of Offsite Power – Reactor Coolant System Pressure vs. Time

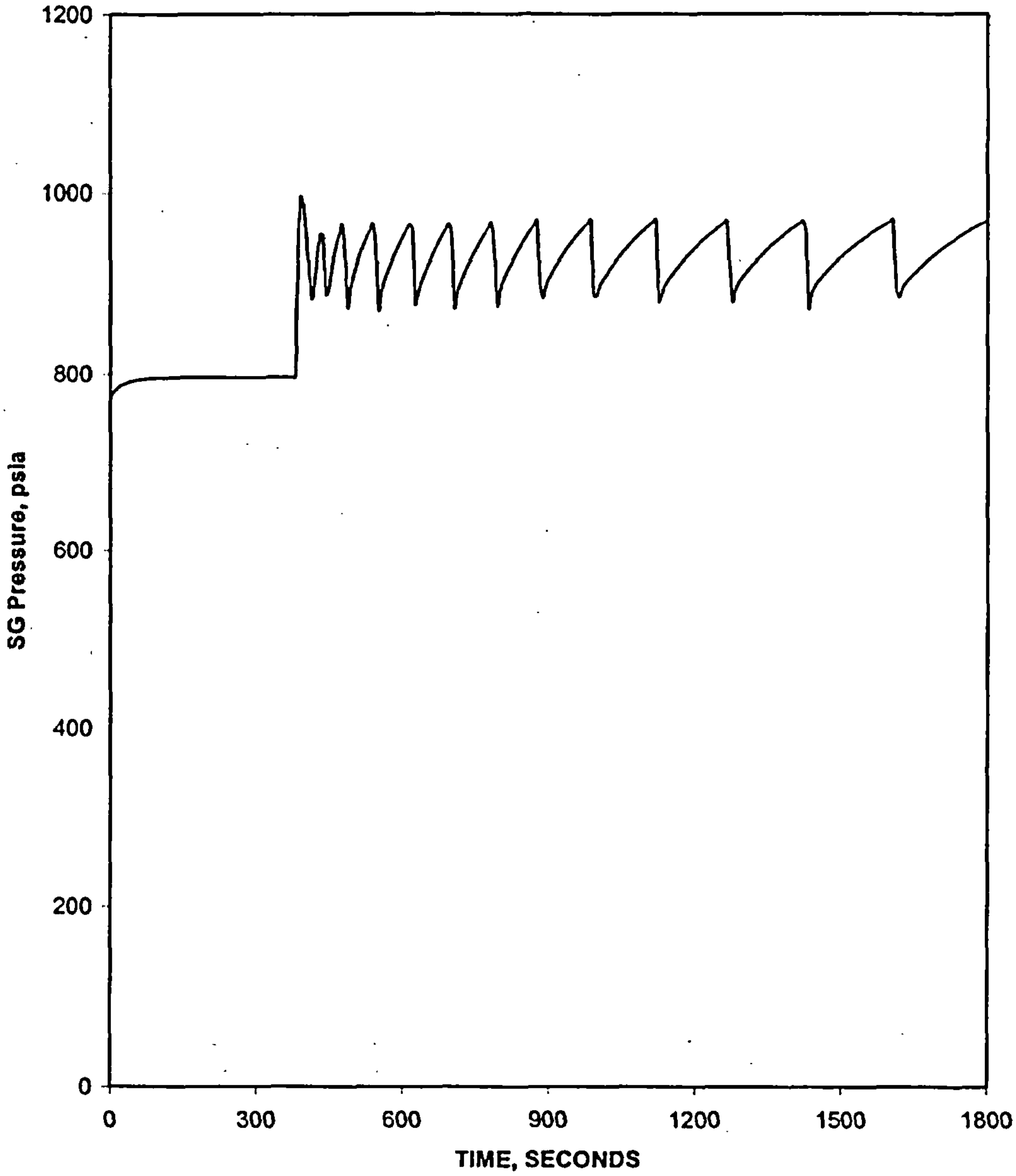

Figure 5.1.24-5 Steam Generator Tube Rupture with Loss of Offsite Power – Steam Generator Pressure vs. Time

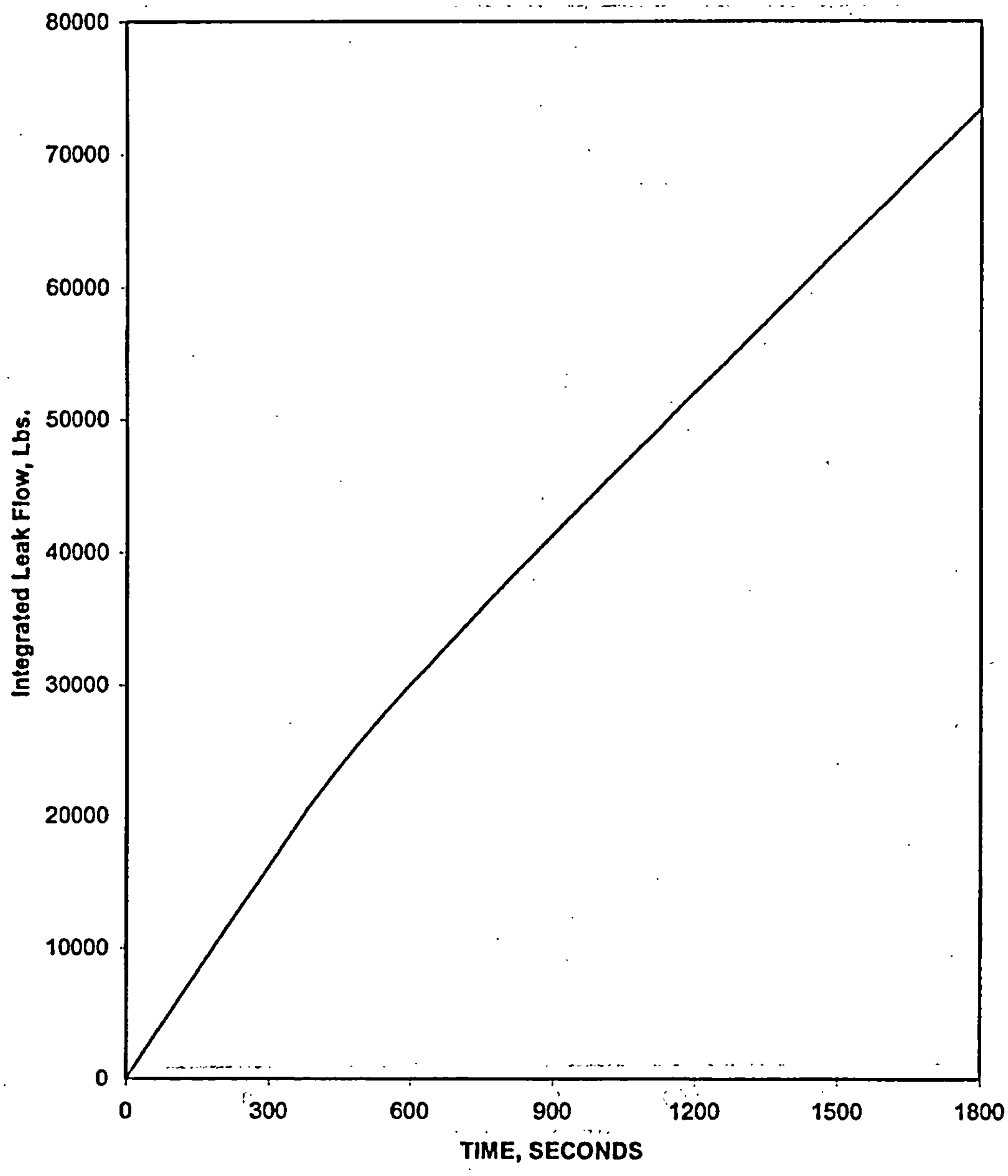

Figure 5.1.24-6 Steam Generator Tube Rupture with Loss of Offsite Power – Integrated Leak Flow vs. Time

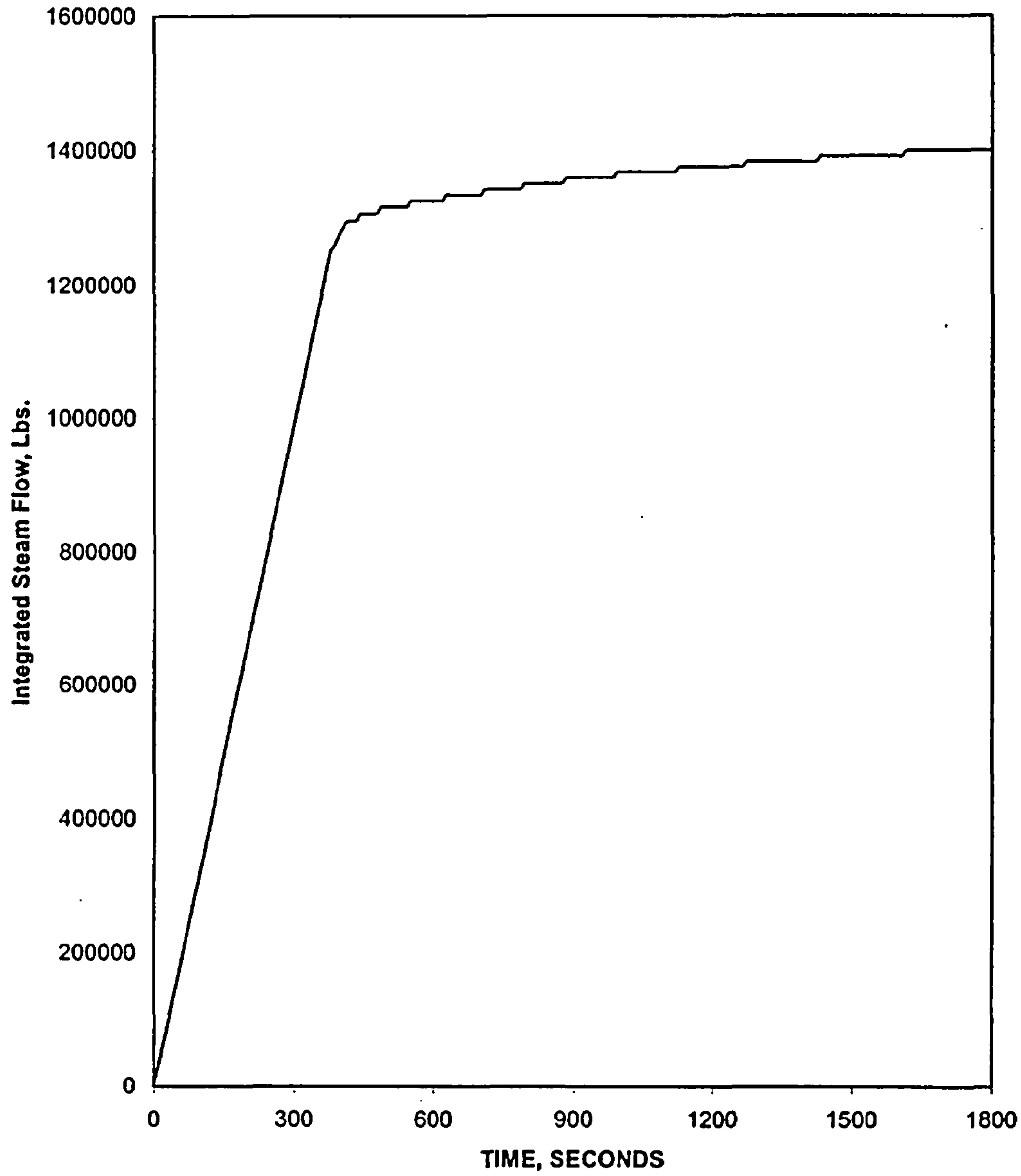

Figure 5.1.24-7 Steam Generator Tube Rupture with Loss of Offsite Power – Integrated Steam Flow vs. Time

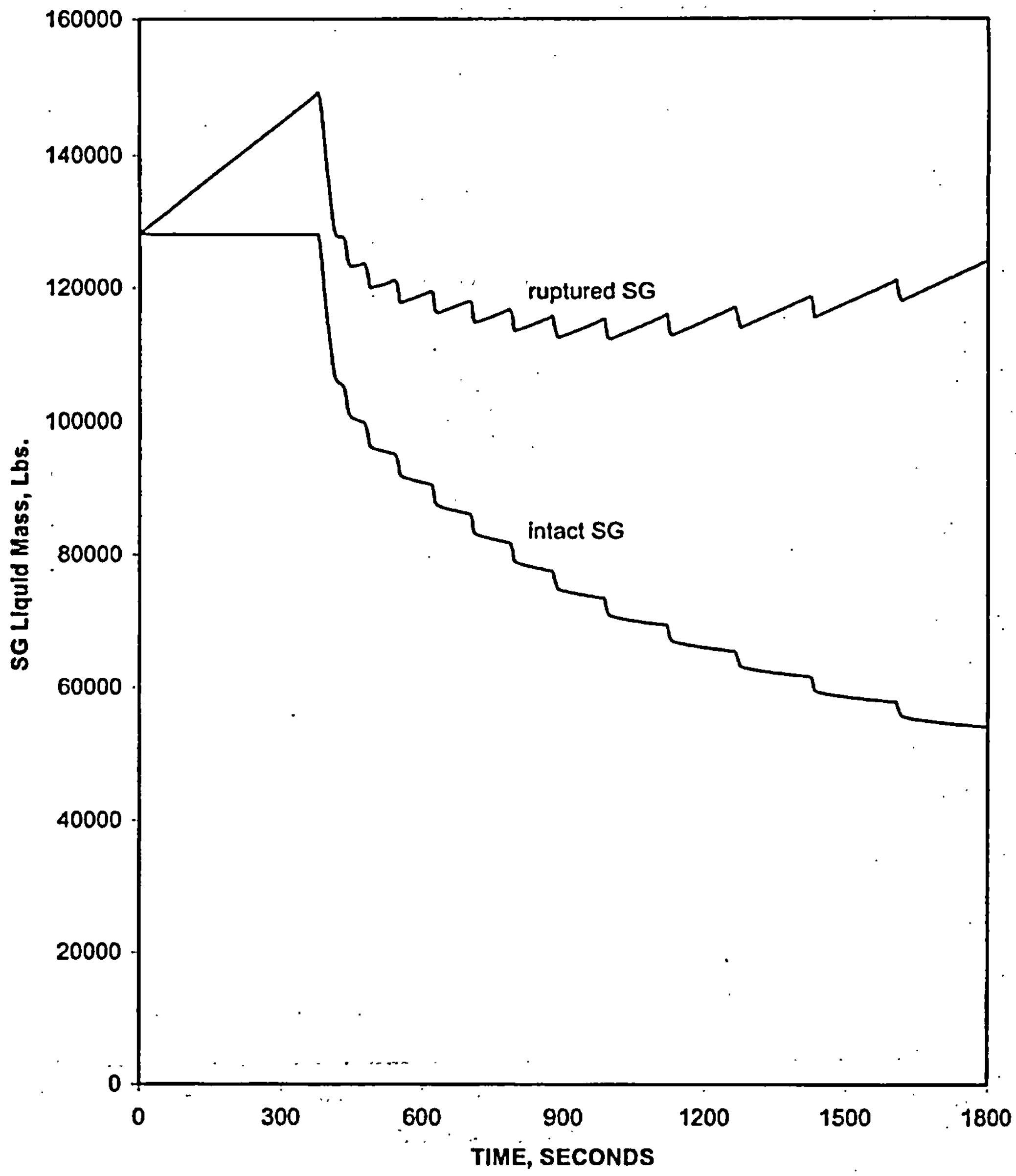

Figure 5.1.24-8 Steam Generator Tube Rupture with Loss of Offsite Power – Water Mass vs. Time

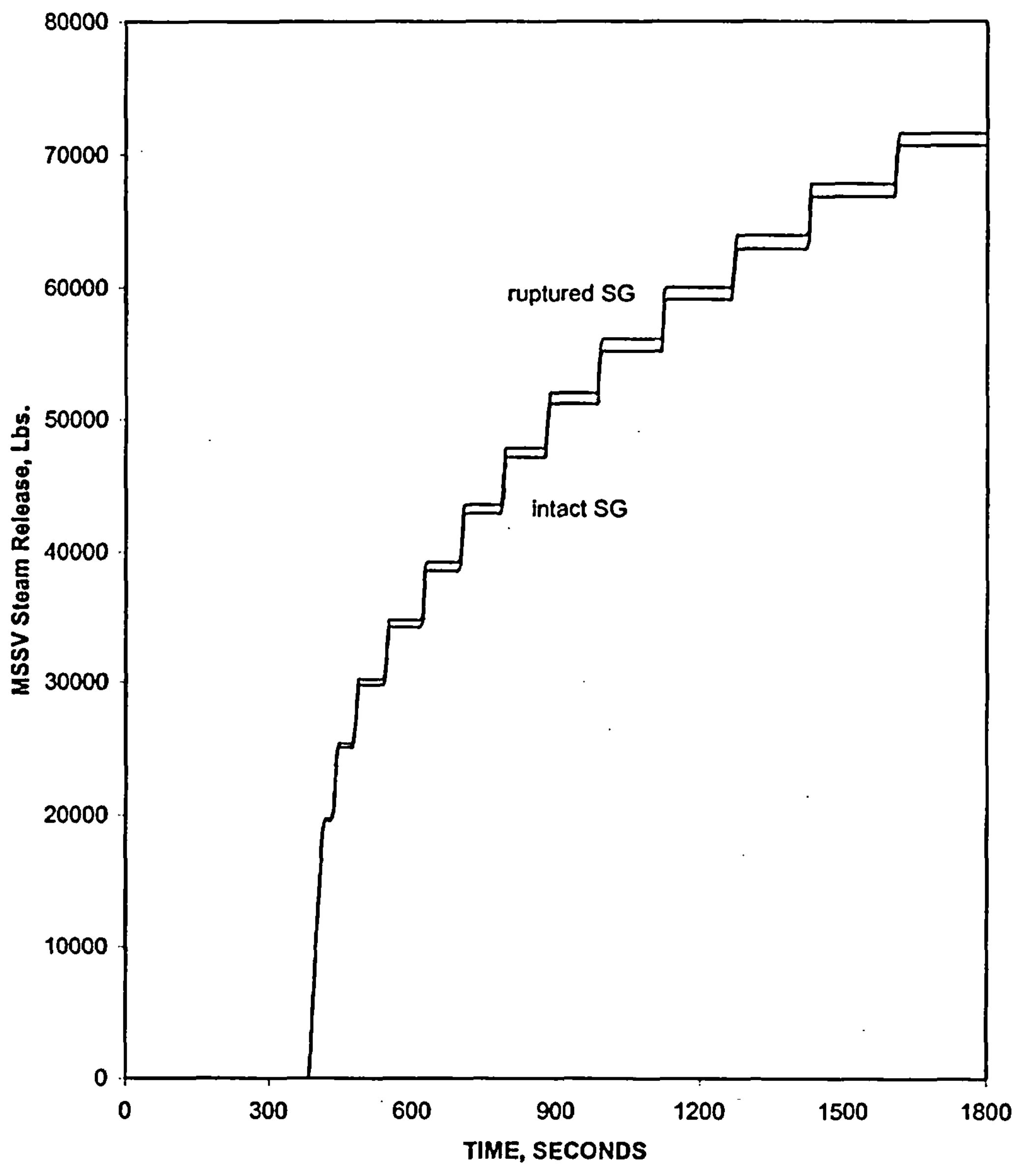

Figure 5.1.24-9 Steam Generator Tube Rupture with Loss of Offsite Power – Steam Release vs. Time

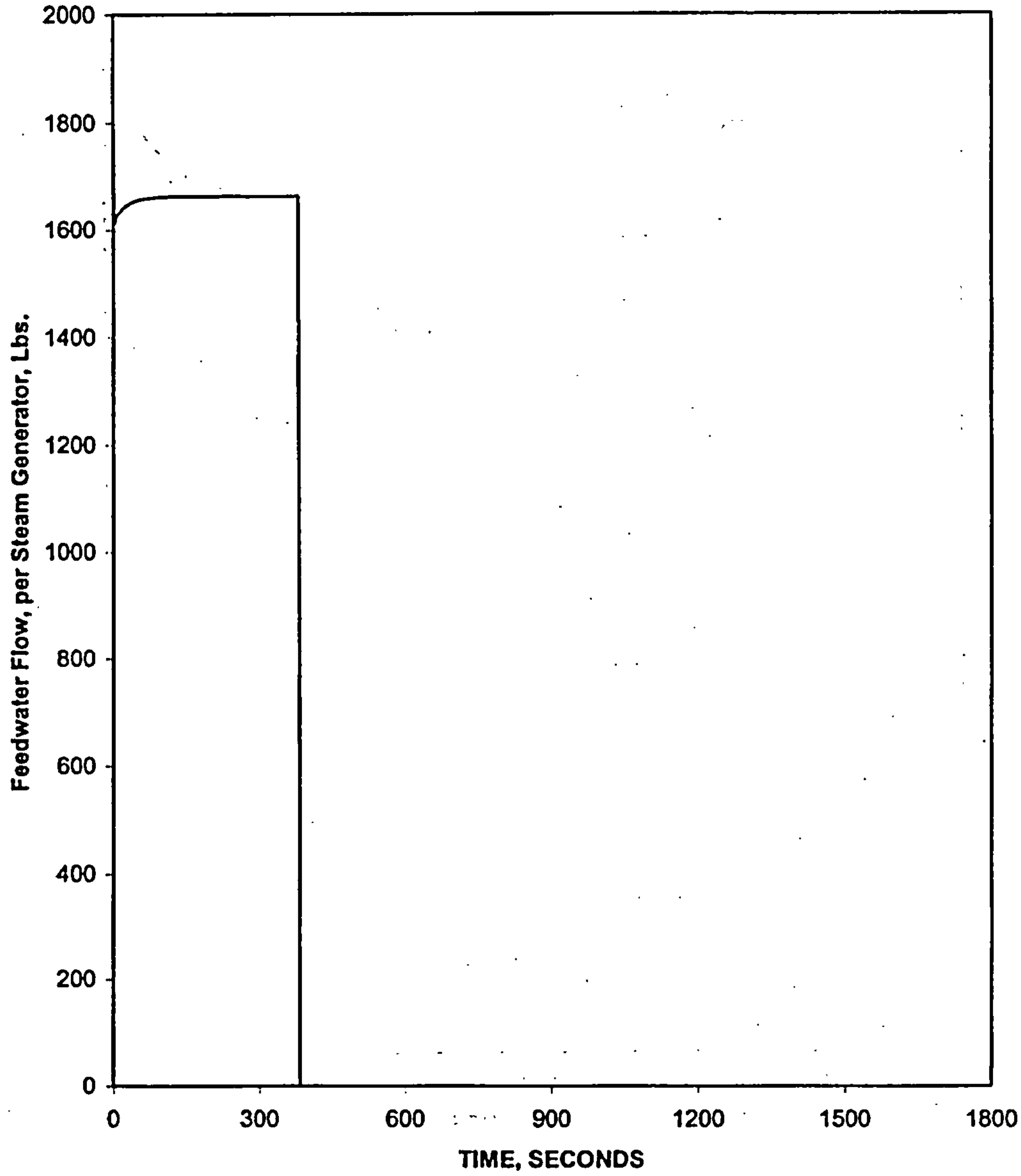

Figure 5.1.24-10 Steam Generator Tube Rupture with Loss of Offsite Power – Feedwater Flow vs. Time

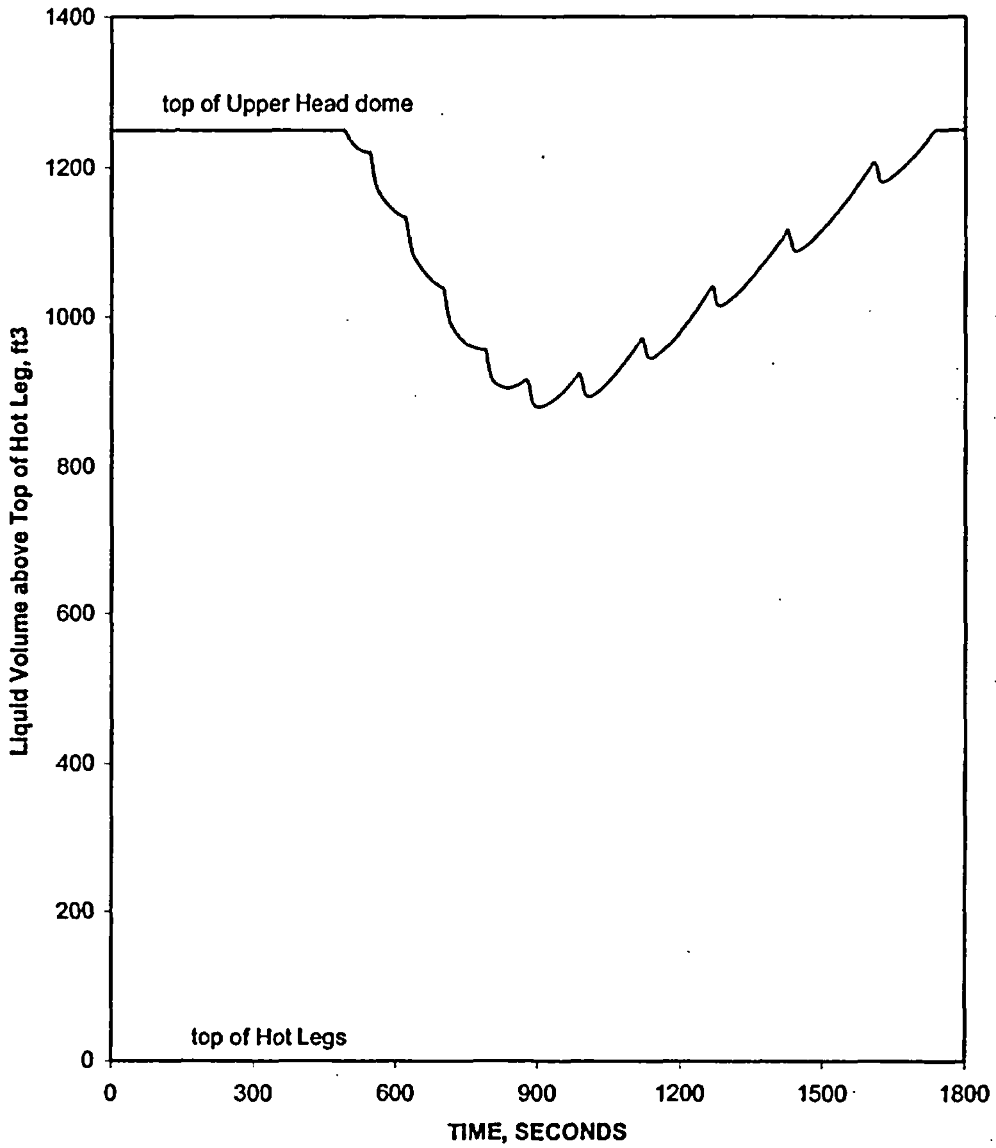

Figure 5.1.24-11 Steam Generator Tube Rupture with Loss of Offsite Power – Liquid Volume Above Top of Hot Leg vs. Time

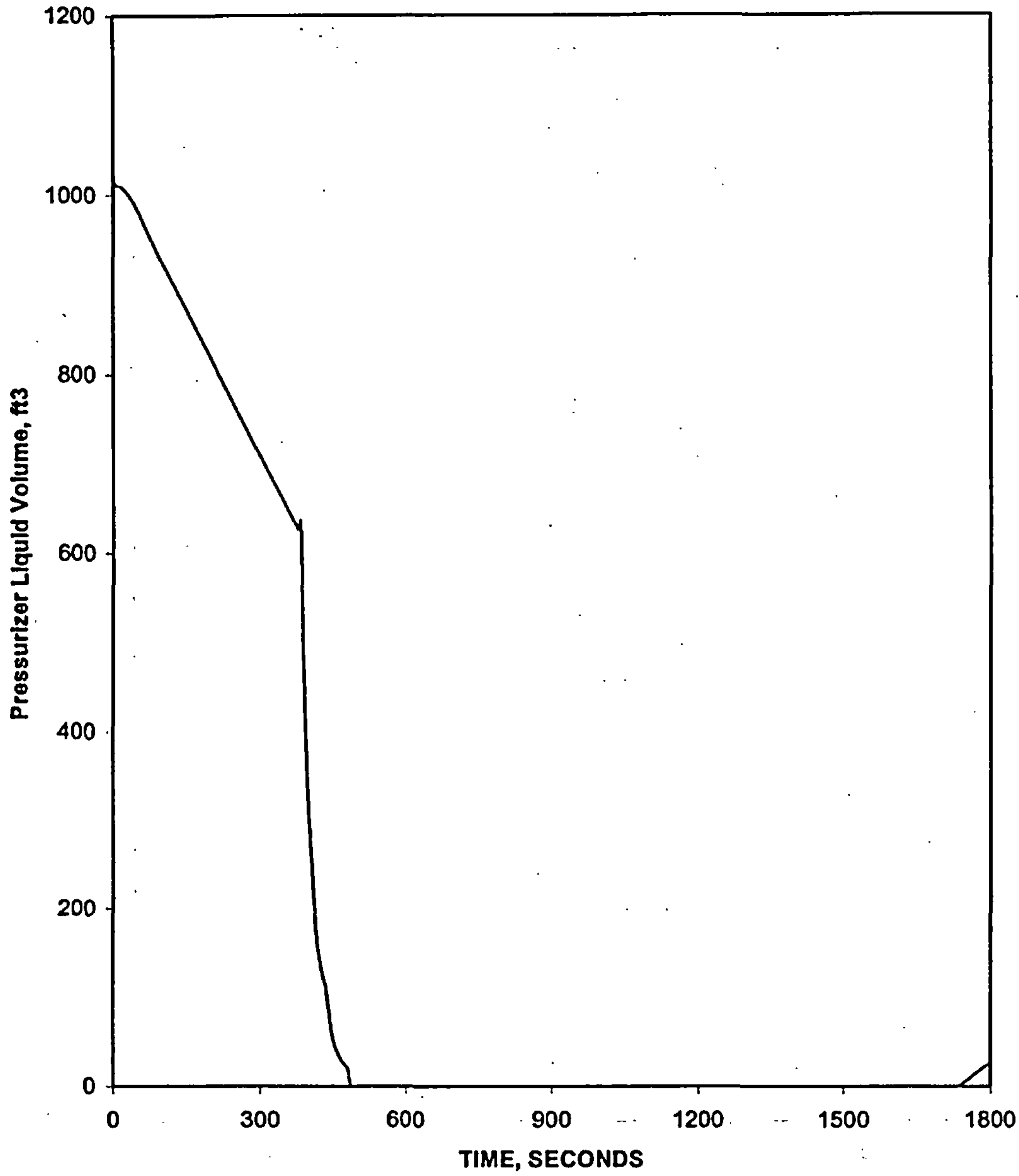

Figure 5.1.24-12 Steam Generator Tube Rupture with Loss of Offsite Power –PZR Water Volume vs. Time

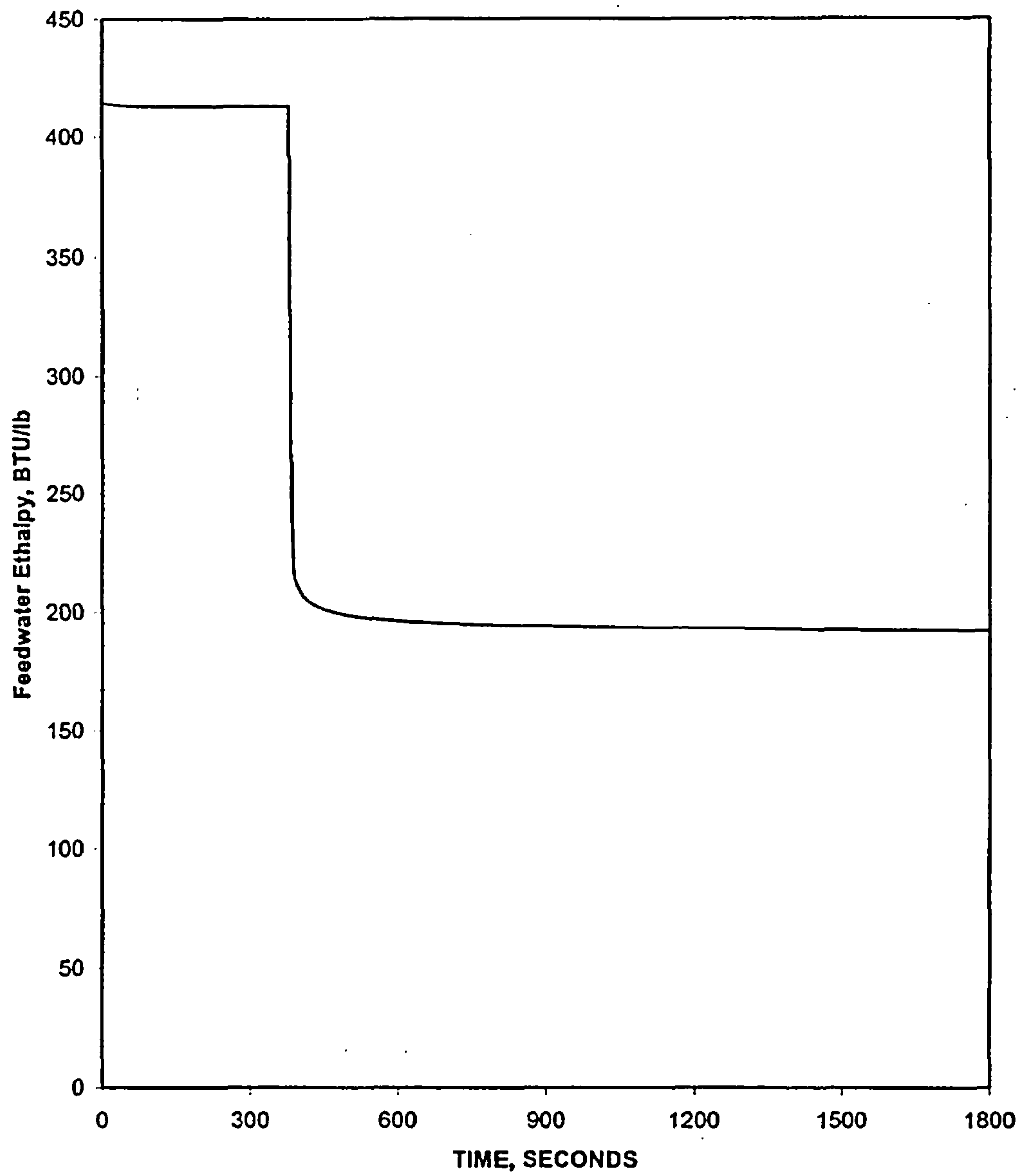

Figure 5.1.24-13 Steam Generator Tube Rupture with Loss of Offsite Power – Feedwater Enthalpy vs. Time

5.2 ECCS PERFORMANCE

5.2.1 Introduction

This section provides the emergency core cooling system (ECCS) performance analysis for St. Lucie Unit 2 with 30% steam generator tube plugging (SGTP) and a reduced technical specification minimum reactor coolant system (RCS) flow rate of 335,000 gpm. The objective of the analysis is to demonstrate conformance to the ECCS acceptance criteria in 10 CFR 50.46(b) for St. Lucie Unit 2 with 30% SGTP and the associated reduction in RCS flow rate. The ECCS performance analysis consists of three individual analyses, namely, the large-break loss-of-coolant accident (LBLOCA), small-break loss-of-coolant accident (SBLOCA) and post-LOCA long-term cooling analyses.

Section 5.2.2 identifies the acceptance criteria for the ECCS performance analysis. Sections 5.2.3 through 5.2.5 summarize the LBLOCA, SBLOCA, and post-LOCA long-term cooling analyses. The summaries include a description of the methodology, the plant design data, and the results of the analyses. The conclusions of the ECCS performance analysis are presented in Section 5.2.6.

5.2.2 Acceptance Criteria

The five acceptance criteria for ECCS performance analyses are specified in 10 CFR 50.46(b) (Reference 1). They are as follows.

Criterion 1: Peak Cladding Temperature: The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

Criterion 2: Maximum Cladding Oxidation: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

Criterion 3: Maximum Hydrogen Generation: The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Criterion 4: Coolable Geometry: Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Criterion 5: Long-Term Cooling: After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Additionally, ECCS performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. The evaluation model may either be a realistic evaluation model as described in 10 CFR 50.46(a)(1)(i) or must conform to the required and acceptable features of Appendix K ECCS Evaluation Models (Reference 2). The evaluation models used to perform the St. Lucie Unit 2 ECCS performance analysis for 30% SGTP are Appendix K evaluation models.

5.2.3 Large-Break Loca

5.2.3.1 Methodology

The LBLOCA ECCS performance analysis used the 1999 evaluation model (EM) version of the Westinghouse LBLOCA evaluation model for Combustion Engineering designed pressurized water reactors (PWRs) (Reference 3). The current St. Lucie Unit 2 LBLOCA analysis (Reference 5) used the June 1985 version of the Westinghouse LBLOCA evaluation model for Combustion Engineering designed PWRs (Reference 4).

Several computer codes are used in the 1999 EM. The computer codes are described in the references cited below with additional descriptive information provided in the 1999 EM topical report (Reference 3). The CEFLASH-4A computer code (Reference 6) is used to perform the blowdown hydraulic analysis of the RCS and the COMPERC-II computer code (Reference 7) is used to perform the RCS refill/reflood hydraulic analysis and to calculate the minimum containment pressure. It is also used in conjunction with the methodology described in Reference 8 to calculate the FLECHT-based reflood heat transfer coefficients used in the hot rod heatup analysis. The HCROSS (Reference 9) and PARCH (Reference 10) computer codes are used to calculate steam cooling heat transfer coefficients. The hot rod heatup analysis, which calculates the peak cladding temperature and maximum cladding oxidation, is performed with the STRIKIN-II computer code (Reference 11). Core-wide cladding oxidation is calculated using the COMZIRC computer code (Appendix C of Supplement 1 of Reference 7). The 1999 EM uses the models for ZIRLO™ cladding that are described in Section 6 of Reference 12. The initial steady state fuel rod conditions used in the LBLOCA analysis are determined using the FATES3B computer code (Reference 13).

The Safety Evaluation Reports for the evaluation model and computer code topical reports that comprise the 1999 EM are documented in References 14 through 22.

The limiting initial fuel rod conditions used in the LBLOCA analysis (i.e., the conditions that result in the highest calculated peak cladding temperature) were determined by performing burnup dependent calculations with STRIKIN-II using initial fuel rod conditions calculated by FATES3B. The calculations included the analysis of both UO_2 fuel rods and gadolinia burnable absorber fuel rods and Zircaloy-4 and ZIRLO™ cladding.

The analysis included a study to determine the most limiting single failure of ECCS equipment. The study analyzed no failure, failure of an emergency diesel generator, and failure of a low pressure safety injection (LPSI) pump. Maximum safety injection pump flow rates were used in the no failure case; minimum safety injection pump flow rates were used in the emergency diesel generator and LPSI pump failure cases. The pumps were actuated on a safety injection actuation signal (SIAS) generated by low pressurizer pressure with a startup delay of 30 seconds. The most limiting single failure (i.e., the failure that resulted in the highest calculated peak cladding temperature) was no failure of ECCS equipment. No failure is the worst condition because it maximizes the amount of safety injection that spills into the containment. This acts to minimize containment pressure, which in turn minimizes the rate at which the core is reflooded. The failure of either an emergency diesel generator or an LPSI pump is not the most damaging failure because, in both cases, there is sufficient safety injection pump flow to keep the reactor vessel downcomer filled to the cold leg nozzles. This maintains the same driving force for reflooding the core as for no failure, but results in less spillage into the containment. A study was also performed to investigate the impact of variations in safety injection tank (SIT) temperature, pressure, and water volume and refueling water tank temperature on peak cladding temperature. The combination of minimum temperature and pressure and maximum water volume for the SITs and minimum refueling water tank temperature was determined to result in the highest peak cladding temperature.

A spectrum of guillotine breaks in the reactor coolant pump discharge leg was analyzed. As described in Section 3.4 of Reference 3, the discharge leg is the most limiting break location and a guillotine break is more limiting than a slot break. In particular, the 1.0, 0.8, 0.6, and 0.4 Double-Ended Guillotine breaks in the reactor coolant Pump Discharge leg (DEG/PD) were analyzed. The 0.6 DEG/PD break was determined to be the limiting large break LOCA (i.e., the break that results in the highest calculated peak cladding temperature).

5.2.3.2 Input Assumptions

Important core, RCS, ECCS, and containment design data used in the large break LOCA analysis are listed in Table 5.2.3.2-1. The fuel rod conditions listed in Table 5.2.3.2-1 are for the hot rod burnup that produced the highest calculated peak cladding temperature. Plant design data for the containment (e.g., data for the containment initial conditions, containment volume, containment heat removal systems, and containment passive heat sinks) were selected to minimize the transient containment pressure.

5.2.3.3 Results

Table 5.2.3.3-1 lists the peak cladding temperature and oxidation percentages for the spectrum of large break LOCAs. Times of interest are listed in Table 5.2.3.3-2. The variables listed in Table 5.2.3.3-3 are plotted as a function of time for each break size in Figures 5.2.3.3-1 through 5.2.3.3-27 and Figures 5.2.3.3-39 through 5.2.3.3-47. The additional variables listed in Table 5.2.3.3-4 are plotted for the 0.6 DEG/PD break, the limiting large break LOCA, in Figures 5.2.3.3-28 through 5.2.3.3-38. The results demonstrate conformance to the ECCS acceptance criteria as summarized below.

Parameter	Criterion	Result
Peak Cladding Temperature	≤2200°F	2130°F
Maximum Cladding Oxidation	≤17%	16.10%
Maximum Core-Wide Oxidation	≤1%	<1%
Coolable Geometry	Yes	Yes

The results are applicable to St. Lucie Unit 2 with up to 30% tube plugging in each steam generator, a tube plugging differential between the two steam generators of up to 800 tubes and a minimum RCS flow rate of 335,000 gpm.

5.2.4 Small-Break LOCA

5.2.4.1 Methodology

The SBLOCA ECCS performance analysis used the Supplement 2 version (referred to as the S2M or Supplement 2 Model) of the Westinghouse SBLOCA evaluation model for Combustion Engineering designed PWRs (Reference 23). The current St. Lucie Unit 2 SBLOCA analysis (Reference 5) used the S1M version of the Westinghouse SBLOCA evaluation model for Combustion Engineering designed PWRs (Reference 24).

In the S2M evaluation model, the CEFLASH-4AS computer program (Reference 25) is used to perform the hydraulic analysis of the RCS until the time the SITs begin to inject. After injection from the SITs begins, the COMPERC-II computer program (Reference 7) is used to perform the hydraulic analysis. The hot rod cladding temperature and maximum cladding oxidation are calculated by the STRIKIN-II computer program (Reference 11) during the initial period of forced convection heat transfer and by the PARCH computer program (Reference 10) during the subsequent period of pool boiling heat transfer. Core-wide cladding oxidation is conservatively represented as the rod-average cladding oxidation of the hot rod. The S2M uses the models for ZIRLO™ cladding that are described in Section 6 of Reference 12. The initial steady state fuel rod conditions used in the SBLOCA analysis are determined using the FATES3B computer program (Reference 13).

The Safety Evaluation Reports for the evaluation model and computer code topical reports that comprise the S2M are documented in References 26 and 27 as well as several of the Safety Evaluation Reports identified in Section 5.2.3.1.

The COMPERC-II computer code was not run for this analysis because flow from the SITs was not credited in the analysis even if the RCS was calculated to depressurize below the minimum SIT gas pressure. Table 5.2.3.3-2 identifies when SIT injection would have begun if it were credited. As was done in the previous SBLOCA analysis, STRIKIN-II was not run because the peak cladding temperature, which is calculated by the PARCH code, is not significantly impacted by the portion of the hot rod heatup transient calculated by STRIKIN-II.

The break spectrum analysis was performed for the fuel rod conditions at the burnup that results in the maximum initial stored energy in the fuel. In addition, the rod internal pressure was adjusted to cause cladding rupture to occur at the time that resulted in the highest peak cladding temperature. The calculations included the analysis of both UO_2 fuel rods and gadolinia burnable absorber fuel rods and Zircaloy-4 and ZIRLO™ cladding.

The analysis was performed using the failure of an emergency diesel generator as the most limiting single failure of the ECCS. This failure causes the loss of both a high pressure safety injection (HPSI) pump and a LPSI pump and results in a minimum of safety injection water being available to cool the core. Based on this failure and the design of the St. Lucie Unit 2 ECCS, 75% of the flow from one HPSI pump is credited in the SBLOCA analysis. The LPSI pumps are not explicitly credited in the SBLOCA analysis since the RCS pressure never decreases below the LPSI pump shutoff head during the portion of the transient that is analyzed. However, 50% of the flow from one LPSI pump is available to cool the core given a failure of an emergency diesel generator and a break in the reactor coolant pump discharge leg.

A spectrum of three break sizes in the reactor coolant pump discharge leg was analyzed. The reactor coolant pump discharge leg is the limiting break location because it maximizes the amount of spillage from the ECCS. In particular, the 0.04, 0.05, and 0.06 ft^2/PD breaks were analyzed. The 0.05 ft^2/PD break was determined to be the limiting SBLOCA (i.e., the break that results in the highest calculated peak cladding temperature). The 0.04, 0.05, and 0.06 ft^2/PD breaks are at the upper end of the range of break sizes for which the hot rod cladding heatup transient is terminated solely by injection from a HPSI pump. It is within this range of break sizes that the limiting SBLOCA resides. Smaller breaks are too small to experience as much core uncovery as these breaks. For larger breaks, injection from the SITs and a HPSI pump recovers the core and terminates the heatup of the cladding before the cladding temperature approaches the peak cladding temperature of the limiting SBLOCA.

5.2.4.2 Input Assumptions

Important core, RCS, and ECCS design data used in the SBLOCA analysis are listed in Tables 5.2.4.2-1 and 5.2.4.2-2.

5.2.4.3 Results

Table 5.2.4.3-1 lists the peak cladding temperature and oxidation percentages for the spectrum of SBLOCAs. Times of interest are listed in Table 5.2.4.3-2. The variables listed in Table 5.2.4.3-3 are plotted as a function of time for each break size in Figures 5.2.4.3-1 through 5.2.4.3-24. The results for the 0.05 ft^2/PD break, the limiting SBLOCA, demonstrate conformance to the ECCS acceptance criteria as summarized below.

Parameter	Criterion	Result
Peak Cladding Temperature	≤2200°F	1943°F
Maximum Cladding Oxidation	≤17%	9.80%
Maximum Core-Wide Oxidation	≤1%	<0.64%
Coolable Geometry	Yes	Yes

The results are applicable to St. Lucie Unit 2 with up to 30% tube plugging in each steam generator, a tube plugging differential between the two steam generators of up to 800 tubes and a minimum RCS flow rate of 335,000 gpm.

5.2.5 Post-LOCA Long-Term Cooling

5.2.5.1 Methodology

The post-LOCA long-term cooling analysis used the Westinghouse post-LOCA long-term cooling evaluation model for Combustion Engineering designed PWRs, CENPD-254-P-A (Reference 28). This is the same methodology used in the current St. Lucie Unit 2 long-term cooling analysis (Reference 5). The Safety Evaluation Report documenting NRC acceptance of CENPD-254-P-A is contained in Reference 29.

The long-term cooling analysis consists of two separate analyses, namely, a boric acid precipitation analysis and a decay heat removal analysis. These two analyses are referred to as the large break analysis and the small break analysis in CENPD-254-P-A.

The purpose of the boric acid precipitation analysis is to demonstrate that the maximum boric acid concentration in the core remains below the solubility limit, thereby preventing the precipitation of boric acid in the core. If boric acid was to precipitate in the core region, the precipitate could prevent water from remaining in contact with the fuel cladding and, consequently, result in the core temperature not being maintained at an acceptably low value.

The boric acid precipitation analysis used a boric acid concentration of 27.6 wt% as the solubility limit of boric acid. This is the solubility limit of boric acid in saturated water at atmospheric pressure. Atmospheric pressure is a conservative minimum value for the core pressure following a LBLOCA.

The boric acid precipitation analysis was performed with the BORON computer code (Reference 28, Appendix C). An important parameter in the boric acid precipitation analysis is the volume within which the boric acid accumulates in the reactor vessel, i.e., the mixing volume. As stated in a footnote on page 20 of Amendment 1 to CENPD-254-P-A, the BORON code uses a constant, input specified value for the mixing volume that is conservatively determined. In the St. Lucie Unit 2 boric acid precipitation analysis, the mixing volume consists of the volume from the top of the core support plate to the bottom elevation of the hot legs that is inside the core baffle and, above the core baffle, that is inside the core barrel.

The purpose of the decay heat removal analysis is to demonstrate that, regardless of break size, the core remains covered with two-phase liquid in the long-term, thereby ensuring that the core temperature is maintained at an acceptably low value. If the break is small enough for the RCS to refill, the RCS is cooled down via the steam generators to the shutdown cooling entry temperature and shutdown cooling is initiated. Decay heat is then removed by the shutdown cooling system. For breaks that are too large for the RCS to refill, the break flow is sufficient to remove decay heat from the RCS in the long-term.

The decay heat removal analysis was performed with the CELDA, NATFLOW, and CEPAC computer codes (Reference 28, Appendices A, B, and D).

5.2.5.2 Input Assumptions

Important plant design data used in the post-LOCA long-term cooling analysis are listed in Table 5.2.5.2-1.

5.2.5.3 Results

The post-LOCA boric acid precipitation analysis determined that a minimum flow rate of 275 gpm from a HPSI pump to both the hot and cold legs of the RCS, initiated between two and six hours post-LOCA, maintains the boric acid concentration in the core below the solubility limit of 27.6 wt% for the limiting break, i.e., a large cold leg break. The analysis also determined that the potential for entrainment of the hot leg injection by the steam flowing in the hot legs ends prior to two hours post-LOCA.

Figure 5.2.5.3-1 compares the core boiloff rate with the minimum simultaneous hot and cold leg injection flow rate of 275 gpm. It shows that the initiation of 275 gpm of hot and cold leg injection at six hours post-LOCA provides a substantial and time-increasing flushing flow through the core. Figure 5.2.5.3-2 presents the core boric acid concentration as a function of time for the limiting break. It shows that without simultaneous hot and cold leg injection, the boric acid concentration in the core exceeds the solubility limit at approximately 7.8 hours post-LOCA. When 275 gpm of simultaneous hot and cold leg injection is initiated at six hours post-LOCA, the maximum boric acid concentration in the core is 24.2 wt%, a margin of 3.4 wt% to the solubility limit of 27.6 wt%. Figure 5.2.5.3-2 also shows that a flushing flow rate of 25 gpm started by six hours post-LOCA is sufficient to prevent the core boric acid concentration from reaching the solubility limit.

Figure 5.2.5.3-3 presents the sequence of events and time schedule for the operator actions that comprise the St. Lucie Unit 2 long-term cooling plan. The plan summarizes the key elements of the decay heat removal analysis as well as the boric acid precipitation analysis. The decay heat removal analysis shows that, regardless of break size, decay heat can be removed for the long-term and that in doing so, the core remains covered with two-phase liquid, thereby ensuring that core temperatures are maintained at acceptably low values. The analysis identified a decision time of 16 hours and a decision pressure of 130 psia. At the decision time, for breaks as large as 0.038 ft^2, the RCS has refilled and shutdown cooling may be used as the long-term decay heat removal method. For breaks as small as 0.007 ft^2, decay heat may be removed in the long-term by simultaneous hot and cold leg injection. The overlap in these two break ranges ensures that an appropriate long-term decay heat removal method is possible.

Figure 5.2.5.3-4 is a plot of break area versus RCS refill time. Figure 5.2.5.3-5 is a plot of RCS pressure at the decision time of 16 hours post-LOCA versus break area. Figure 5.2.5.3-6 tabulates break size and RCS pressure at the decision time. It also indicates the range of break sizes that are large breaks (i.e., simultaneous hot and cold leg injection is acceptable for long-term decay heat removal) and the range of break sizes that are small breaks (i.e., shutdown cooling is acceptable for long-term decay heat removal).

In summary, the results of the post-LOCA long-term cooling analysis demonstrate conformance to Criterion 5 of the ECCS acceptance criteria. The results are applicable to St. Lucie Unit 2 with up to 30% tube plugging in each steam generator, a tube plugging differential between the two steam generators of up to 800 tubes and a minimum RCS flow rate of 335,000 gpm.

5.2.6 Conclusions

An ECCS performance analysis was performed for St. Lucie Unit 2 with 30% SGTP and a reduced technical specification minimum RCS flow rate of 335,000 gpm. The analysis included consideration of LBLOCA, SBLOCA, and post-LOCA long-term cooling. The limiting break size, i.e., the break size that resulted in the highest peak cladding temperature, was determined to be the 0.6 DEG/PD break.

The results of the analysis demonstrate conformance to the ECCS acceptance criteria at a rated core power of 2700 MWt (2754 MWt including a 2% power measurement uncertainty) and a peak linear heat generator rate (PLHGR) of 12.5 kW/ft as follows:

Criterion 1: Peak Cladding Temperature: The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

Result: The ECCS performance analysis calculated a peak cladding temperature of 2130°F for the 0.6 DEG/PD break.

Criterion 2: Maximum Cladding Oxidation: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

Result: The ECCS performance analysis calculated a maximum cladding oxidation of 0.1610 times the total cladding thickness before oxidation for the 0.6 DEG/PD break.

Criterion 3: Maximum Hydrogen Generation: The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Result: The ECCS performance analysis calculated a maximum hydrogen generation (i.e., a maximum core-wide cladding oxidation) of less than 0.01 times the hypothetical amount for the 0.6 DEG/PD break.

Criterion 4: Coolable Geometry: Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Result: The cladding swelling and rupture models used in the ECCS performance analysis account for the effects of changes in core geometry that would occur if cladding rupture is calculated to occur. Adequate core cooling was demonstrated for the changes in core geometry that were calculated to occur as a result of cladding rupture. In addition, the transient analysis was performed to a time when cladding temperatures were decreasing and the RCS was depressurized, thereby precluding any further cladding deformation. Therefore, a coolable geometry was demonstrated.

Criterion 5: Long-Term Cooling: After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Result: The large-break and small-break LOCA ECCS performance analyses demonstrated that the St. Lucie Unit 2 ECCS successfully maintains the fuel cladding temperature at an acceptably low value in the short-term. Subsequently, for the extended period of time required by the long-lived radioactivity remaining in the core, the ECCS continues to supply sufficient cooling water from the refueling water tank and then from the sump to remove decay heat and maintain the core temperature at an acceptably low value. In addition, at the appropriate time, the operator realigns a HPSI pump for simultaneous hot and cold leg injection in order to maintain the core boric acid concentration below the solubility limit.

5.2.7 References

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Table 5.2.3.2-1 LBLOCA ECCS Performance Analysis Core and Plant Design Data		
Quantity	Value	Units
Reactor power level (102% of rated power)	2754	MWt
PLHGR of the hot rod	12.5	kW/ft
PLHGR of the average rod in assembly with hot rod	11.2	kW/ft
Gap conductance at the PLHGR(1)	1660	BTU/hr-ft^2-°F
Fuel centerline temperature at the PLHGR(1)	3255.4	°F
Fuel average temperature at the PLHGR(1)	2061.55	°F
Hot rod gas pressure(1)	1002.8	psia
Moderator temperature coefficient at initial density	0.3×10^{-4}	Δρ/°F
RCS flow rate	128.9×10^{6}	lbm/hr
Core flow rate	124.1×10^{6}	lbm/hr
RCS pressure	2250	psia
Cold leg temperature	532	°F
Hot leg temperature	588	°F
Number of plugged tubes per steam generator	2520	—
Low pressurizer pressure SIAS setpoint	1646	psia
Safety injection tank pressure (min/max)	499.7 / 679.7	psia
Safety injection tank water volume (min/max)	1388 / 1588	ft^3
LPSI+HPSI pump flow rate (min, 1 train/max, 2 trains)	2843 / 7480	gpm
Containment pressure	13.782	psia
Containment temperature	90.0	°F
Containment humidity	100.0	%
Containment net free volume	2.6313×10^{6}	ft^3
Containment spray pump flow rate	3450	gpm/pump
Refueling water tank temperature	51.0	°F
(1) These quantities correspond to the rod average burnup of the hot rod (500 MWD/MTU) that yields the highest peak cladding temperature.		

Table 5.2.3.3-1 LBLOCA ECCS Performance Analysis Results			
Break Size	Peak Cladding Temperature (°F)	Maximum Cladding Oxidation (%)	Maximum Core-Wide Cladding Oxidation (%)
1.0 DEG/PD	2091	14.26	<1.0
0.8 DEG/PD	2097	14.48	<1.0
0.6 DEG/PD	2130	16.10	<1.0
0.4 DEG/PD	2092	14.85	<1.0

Table 5.2.3.3-2 LBLOCA ECCS Performance Analysis Times of Interest (seconds after break)					
Break Size	SITs On	Time of Annulus Downflow	Start of Reflood	SITs Empty	Hot Rod Rupture
1.0 DEG/PD	10.7	21.8	38.0	111.0	53.6
0.8 DEG/PD	11.9	23.2	39.1	112.4	52.0
0.6 DEG/PD	14.2	25.5	41.0	114.7	47.2
0.4 DEG/PD	18.8	30.9	45.4	119.9	50.8

Table 5.2.3.3-3 LBLOCA ECCS Performance Analysis Variables Plotted as a Function of Time for Each Break
Variable
Core Power
Pressure in Center Hot Assembly Node
Break Flow Rate
Hot Assembly Flow Rate (Below Hot Spot)
Hot Assembly Flow Rate (Above Hot Spot)
Hot Assembly Quality
Containment Pressure
Mass Added to Core During Reflood
Peak Cladding Temperature[1]
(1) The cladding temperature at the elevation of cladding rupture is also shown for the limiting break.

Table 5.2.3.3-4 LBLOCA ECCS Performance Analysis Variables Plotted as a Function of Time for the Limiting Break

Variable
Mid Annulus Flow Rate
Quality Above and Below the Core
Core Pressure Drop
Safety Injection Flow Rate into Intact Discharge Legs
Water Level in Downcomer During Reflood
Hot Spot Gap Conductance
Maximum Local Cladding Oxidation Percentage
Fuel Centerline, Fuel Average, Cladding, and Coolant Temperature at the Hot Spot
Hot Spot Heat Transfer Coefficient
Hot Rod Internal Gas Pressure
Core Bulk Channel Flow Rate

Table 5.2.4.2-1. SBLOCA ECCS Performance Analysis Core and Plant Design Data		
Quantity	**Value**	**Units**
Reactor power level (102% of rated power)	2754	MWt
Peak linear heat generation rate	13.0	kW/ft
Axial shape index	-0.15	asiu
Moderator temperature coefficient at initial density	0.3×10^{-4}	$\Delta\rho/^{\circ}F$
RCS flow rate	335,000	gpm
RCS pressure	2250	psia
Cold leg temperature	552.0	°F
Hot leg temperature	607.9	°F
Number of plugged tubes per steam generator	2520	—
MSSV first bank opening pressure	1029	psia
Low pressurizer pressure reactor trip setpoint	1810	psia
Low pressurizer pressure SIAS setpoint	1646	psia
HPSI pump flow rate	Table 5.2.4.2-2	gpm
Safety injection tank pressure	500	psia

Table 5.2.4.2-2 High Pressure Safety Injection Pump Minimum Delivered Flow to RCS (Assuming Failure of an Emergency Diesel Generator)

RCS Pressure (psia)	Flow Rate (gpm)
1198	0
1177	100
1104	200
1035	250
943	300
829	350
699	400
551	450
393	500
217	550
0	604

(1) The flow is assumed to be split equally to each of the four discharge legs.
(2) The flow to the broken discharge leg is assumed to spill out the break.

Table 5.2.4.3-1 SBLOCA ECCS Performance Analysis Results			
Break Size	Peak Cladding Temperature (°F)	Maximum Cladding Oxidation (%)	Maximum Core-Wide Cladding Oxidation (%)
0.04 ft^2/PD	1672	3.26	<0.28
0.05 ft^2/PD	1943	9.80	< 0.64
0.06 ft^2/PD	1818	5.61	<0.42

Table 5.2.4.3-2 SBLOCA ECCS Performance Analysis Times of Interest (seconds after break)				
Break Size	HPSI Flow Delivered to RCS	LPSI Flow Delivered to RCS	SIT Flow Delivered to RCS	Peak Cladding Temperature Occurs
0.04 ft^2/PD	168	(1)	2282(2)	2113
0.05 ft^2/PD	135	(1)	1587(2)	1700
0.06 ft^2/PD	114	(1)	1341(2)	1739

(1) Calculation completed before LPSI flow delivery to RCS begins.

(2) SIT injection calculated to begin but not credited in analysis.

Table 5.2.4.3-3 SBLOCA ECCS Performance Analysis Variables Plotted as a Function of Time for Each Break
Variable
Core Power
Inner Vessel Pressure
Break Flow Rate
Inner Vessel Inlet Flow Rate
Inner Vessel Two-Phase Mixture Level
Heat Transfer Coefficient at Hot Spot
Coolant Temperature at Hot Spot
Cladding Temperature at Hot Spot

Table 5.2.5.2-1 Post-LOCA Long-Term Cooling Analysis Core and Plant Design Data		
Quantity	**Value**	**Units**
Reactor power level (102% of rated power)	2754	MWt
Number of plugged tubes per steam generator	2520	—
SG/RCS cooldown rate	75	°F/hr
Shutdown cooling entry temperature	300	°F
RCS pressure measurement uncertainty	±90	psi
Number of atmospheric dump valves/SG	1	—
Atmospheric dump valve flow rate, at 55 psia	51,300	lbm/hr/valve
Condensate storage tank volume	262,400	gal
Reactor coolant system		
liquid mass	456,000	lbm
boron concentration	2440	ppm
Boric acid makeup tanks		
number	2	—
liquid volume per tank	19,950	gal
boric acid concentration	3.57	wt%
Refueling water tank		
liquid mass	4,325,565	lbm
boron concentration	2200	ppm
Safety injection tanks		
number	4	—
liquid volume per tank	1588	ft^3
boron concentration	2200	ppm
Charging pumps		
number	3	—
flow rate per pump	49	gpm
Flow rates for emptying the RWT		
HPSI pump flow rate	548	gpm
LPSI pump flow rate	2426	gpm
CS pump flow rate	2700	gpm

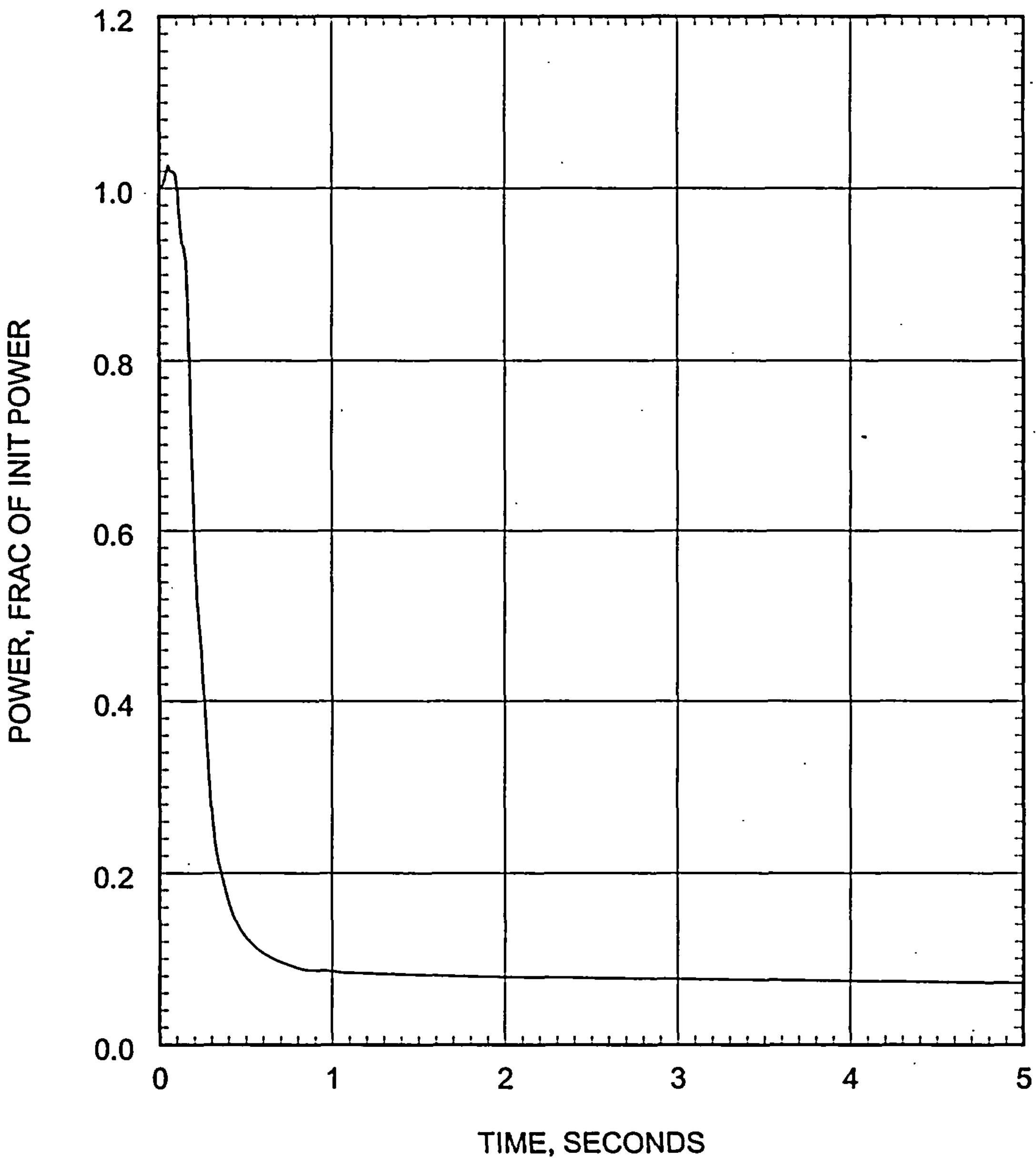

Figure 5.2.3.3-1 Large Break LOCA ECCS Performance Analysis 1.0 DEG/PD Break Core Power

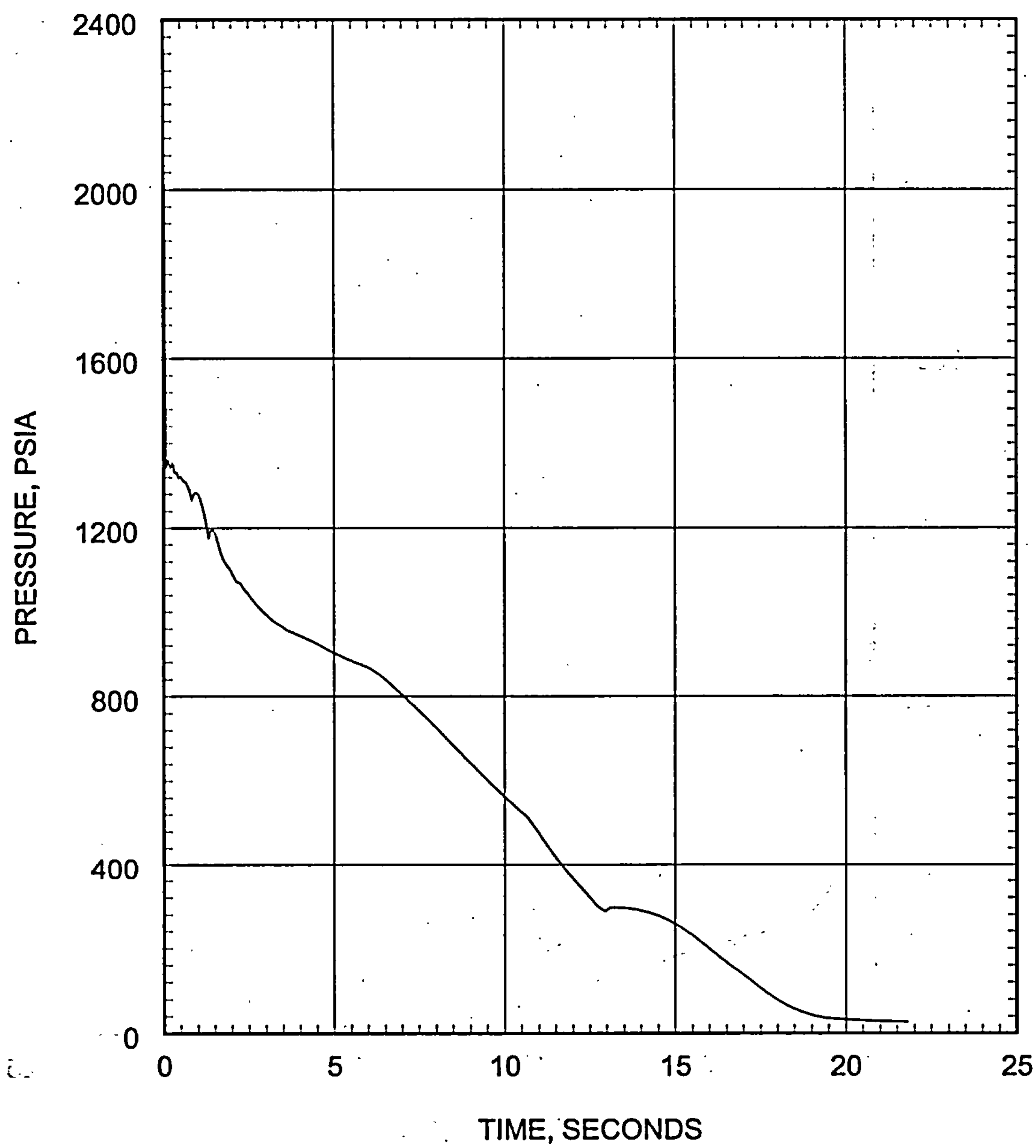

Figure 5.2.3.3-2 Large Break LOCA ECCS Performance Analysis 1.0 DEG/PD Break Pressure in Center Hot Assembly Node

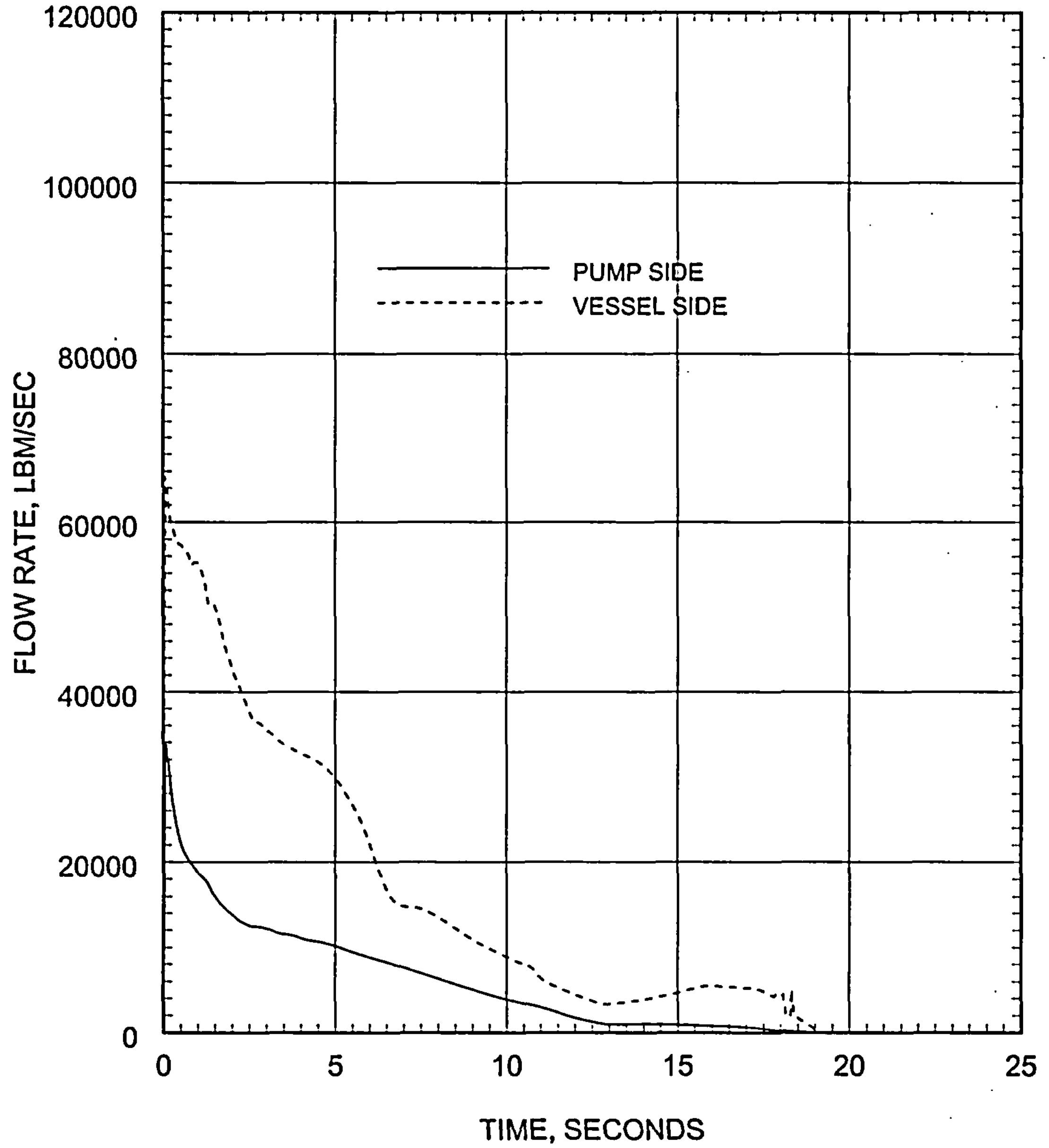

Figure 5.2.3.3-3 Large Break LOCA ECCS Performance Analysis 1.0 DEG/PD Break Leak Flow Rate

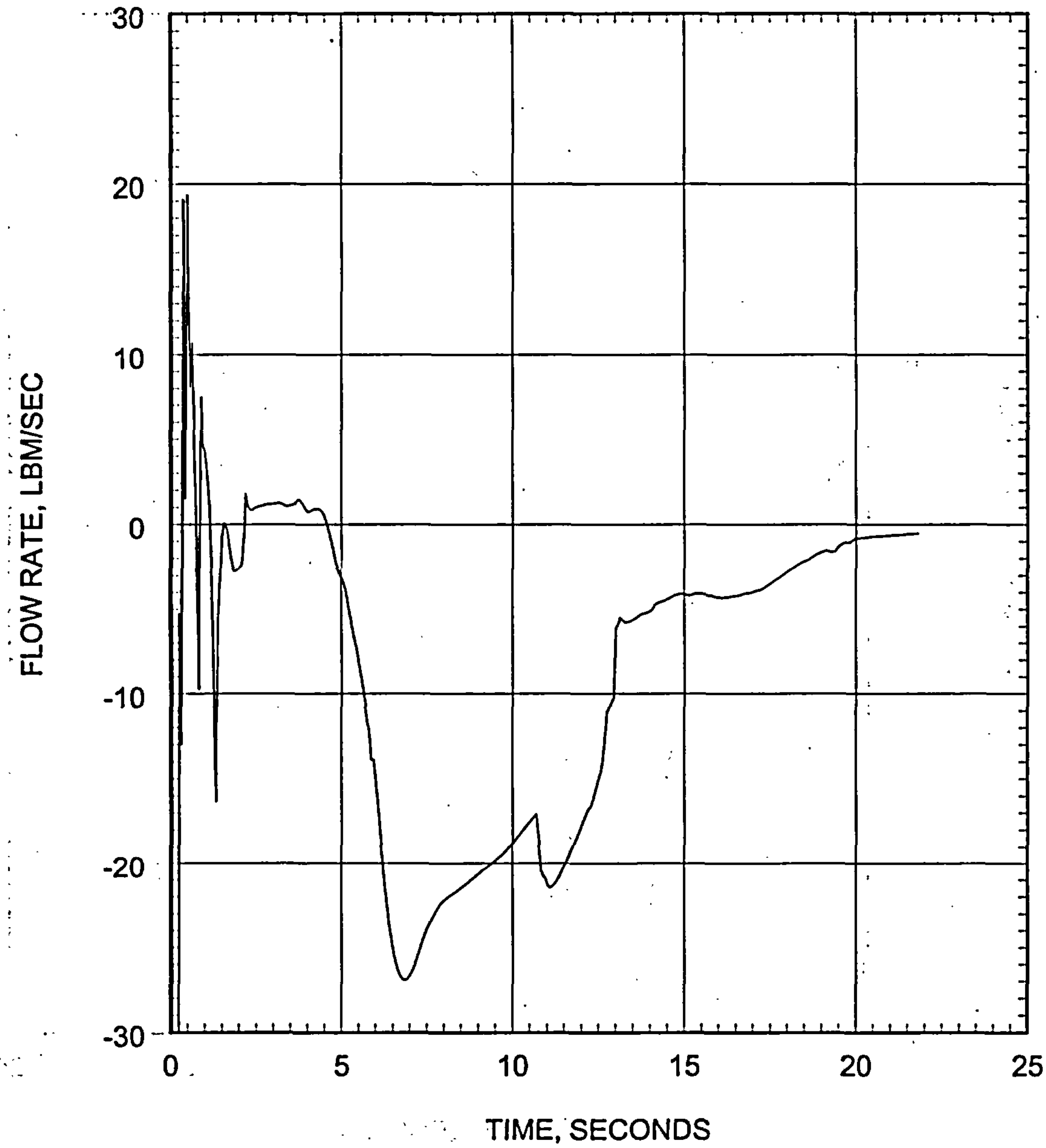

Figure 5.2.3.3-4 Large Break LOCA ECCS Performance Analysis 1.0 DEG/PD Break Hot Assembly Flow Rate (Below Hot Spot)

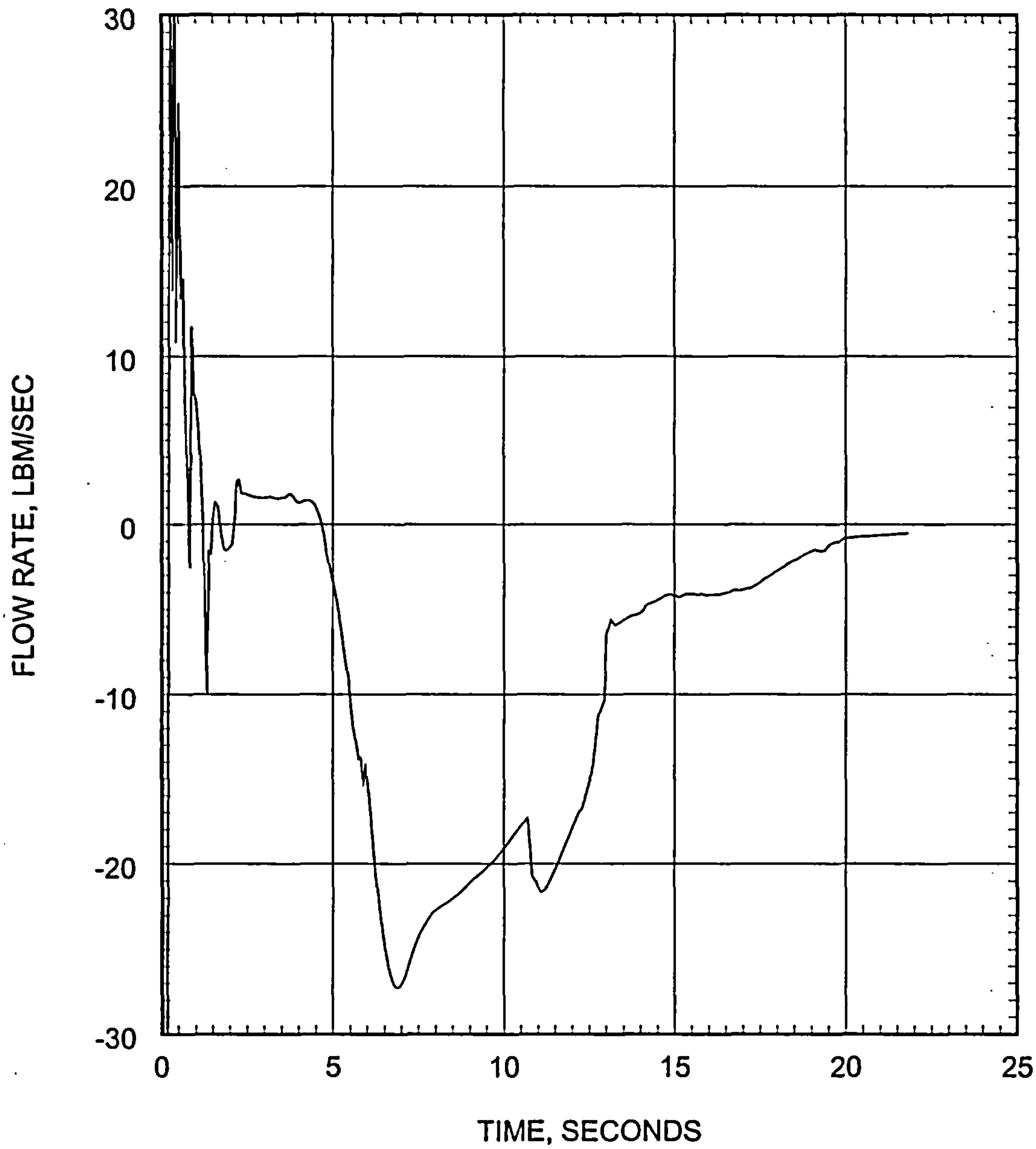

Figure 5.2.3.3-5 Large Break LOCA ECCS Performance Analysis 1.0 DEG/PD Break Hot Assembly Flow Rate (Above Hot Spot)

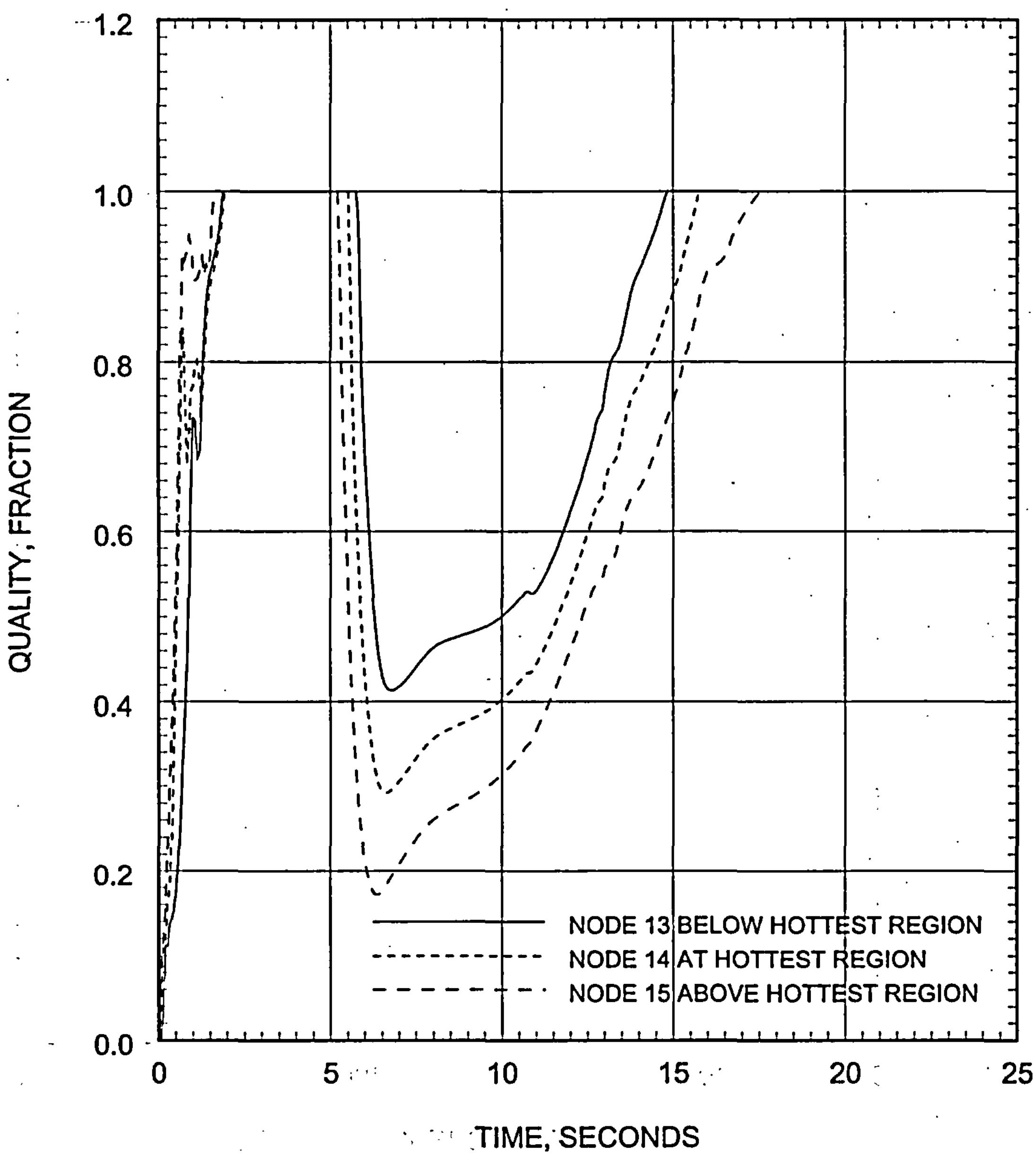

Figure 5.2.3.3-6 Large Break LOCA ECCS Performance Analysis 1.0 DEG/PD Break Hot Assembly Quality

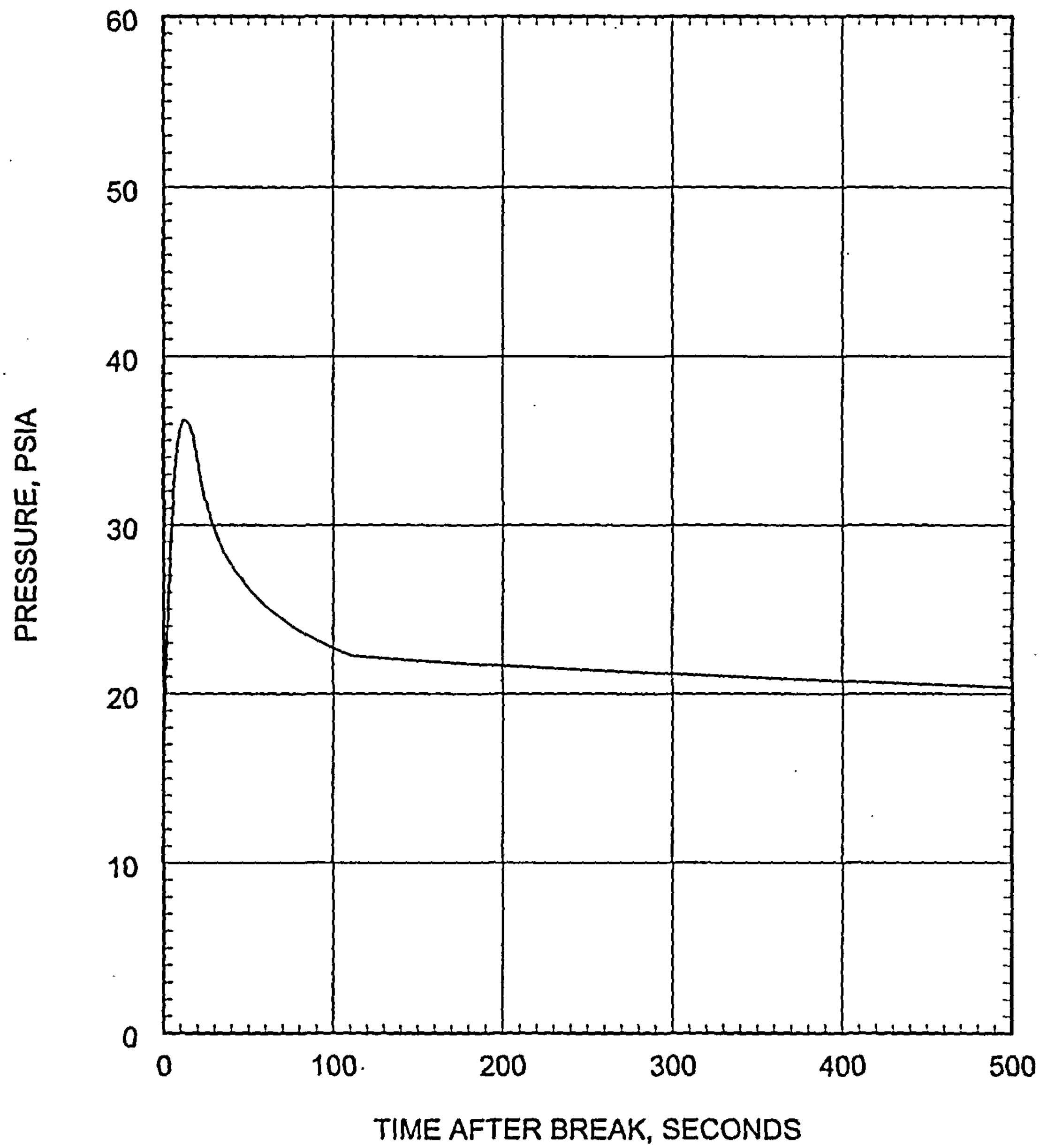

Figure 5.2.3.3-7 Large Break LOCA ECCS Performance Analysis 1.0 DEG/PD Break Containment Pressure

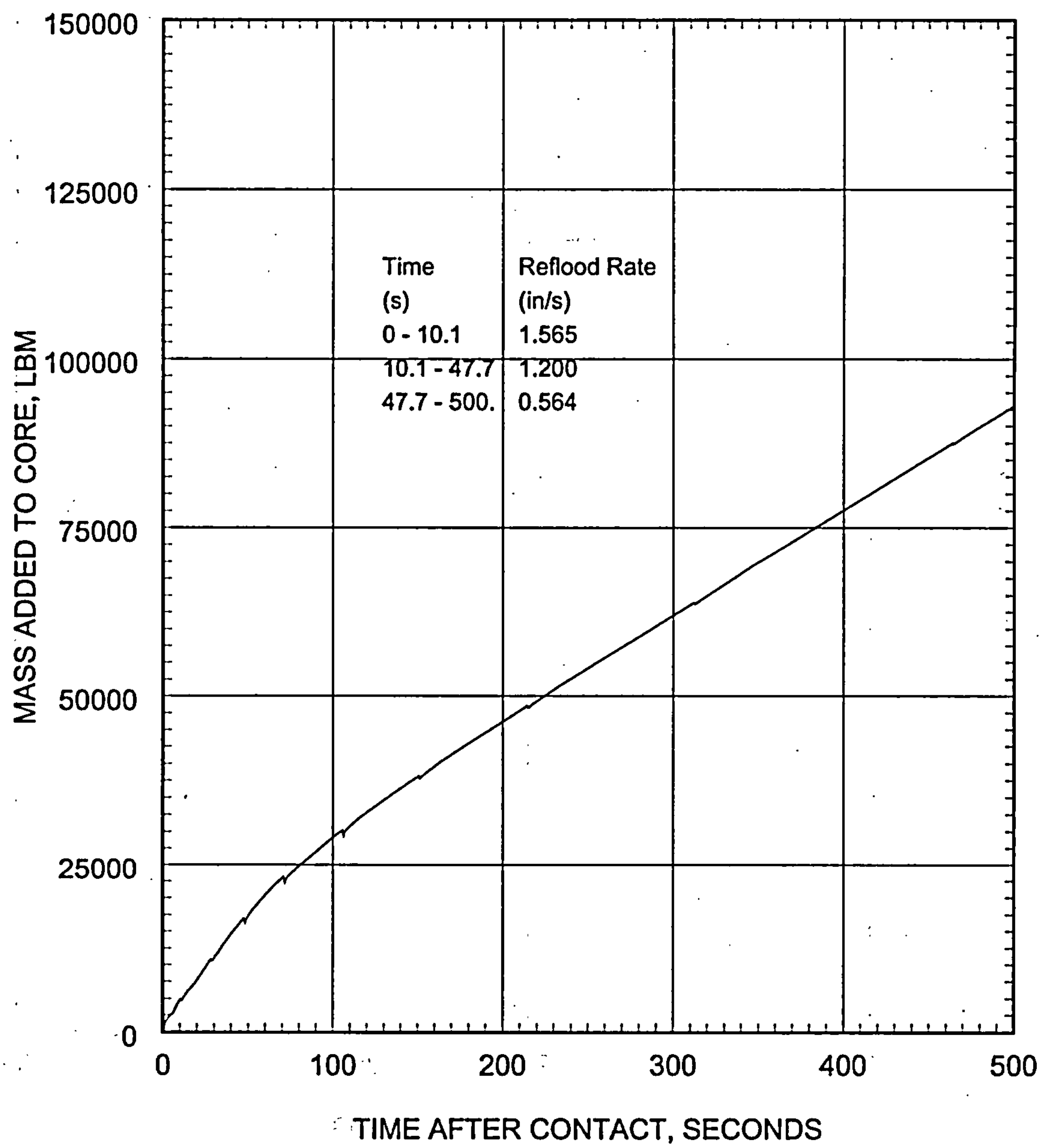

Figure 5.2.3.3-8 Large Break LOCA ECCS Performance Analysis 1.0 DEG/PD Break Mass Added to Core During Reflood

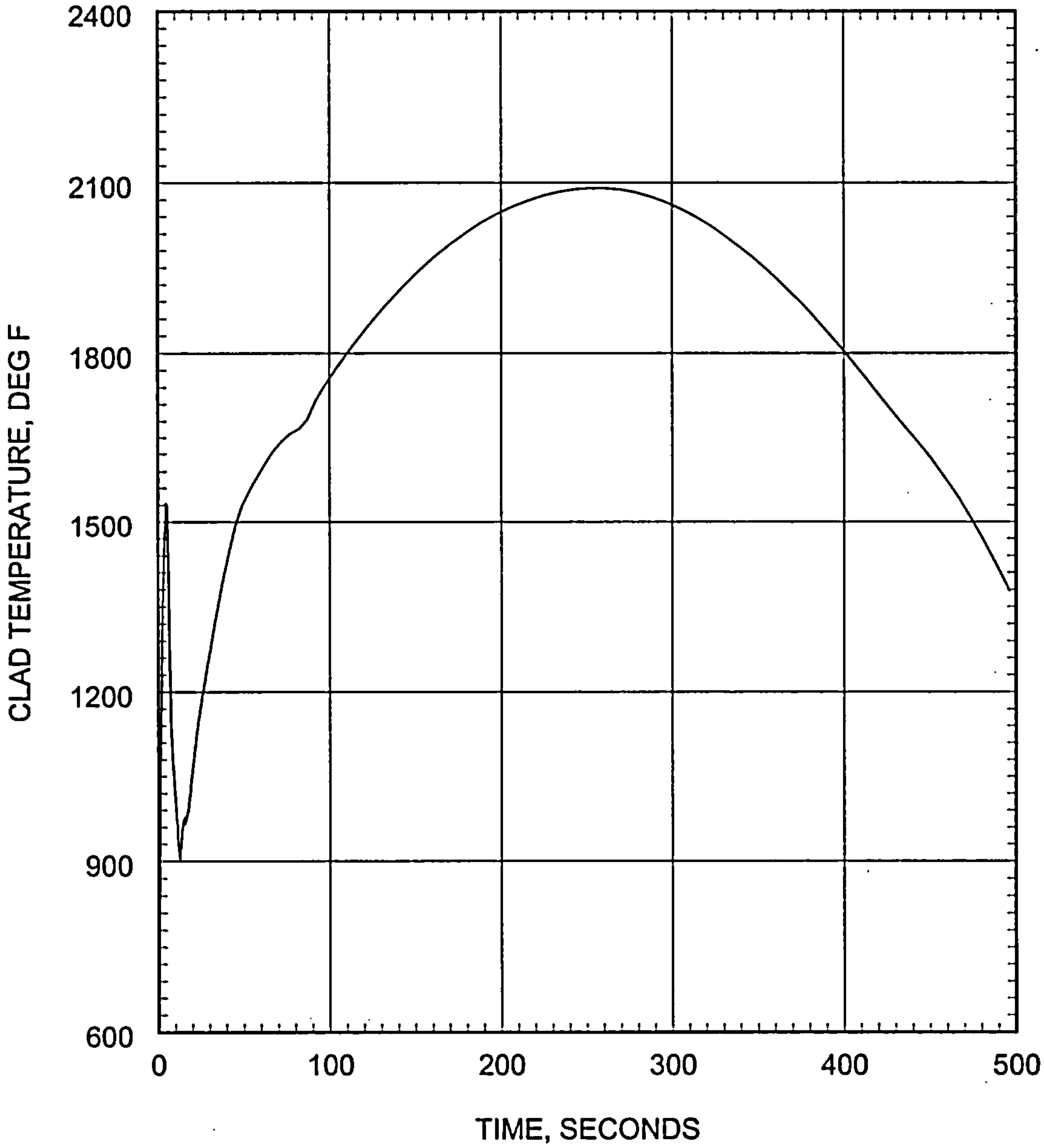

Figure 5.2.3.3-9 Large Break LOCA ECCS Performance Analysis 1.0 DEG/PD Break Peak Cladding Temperature

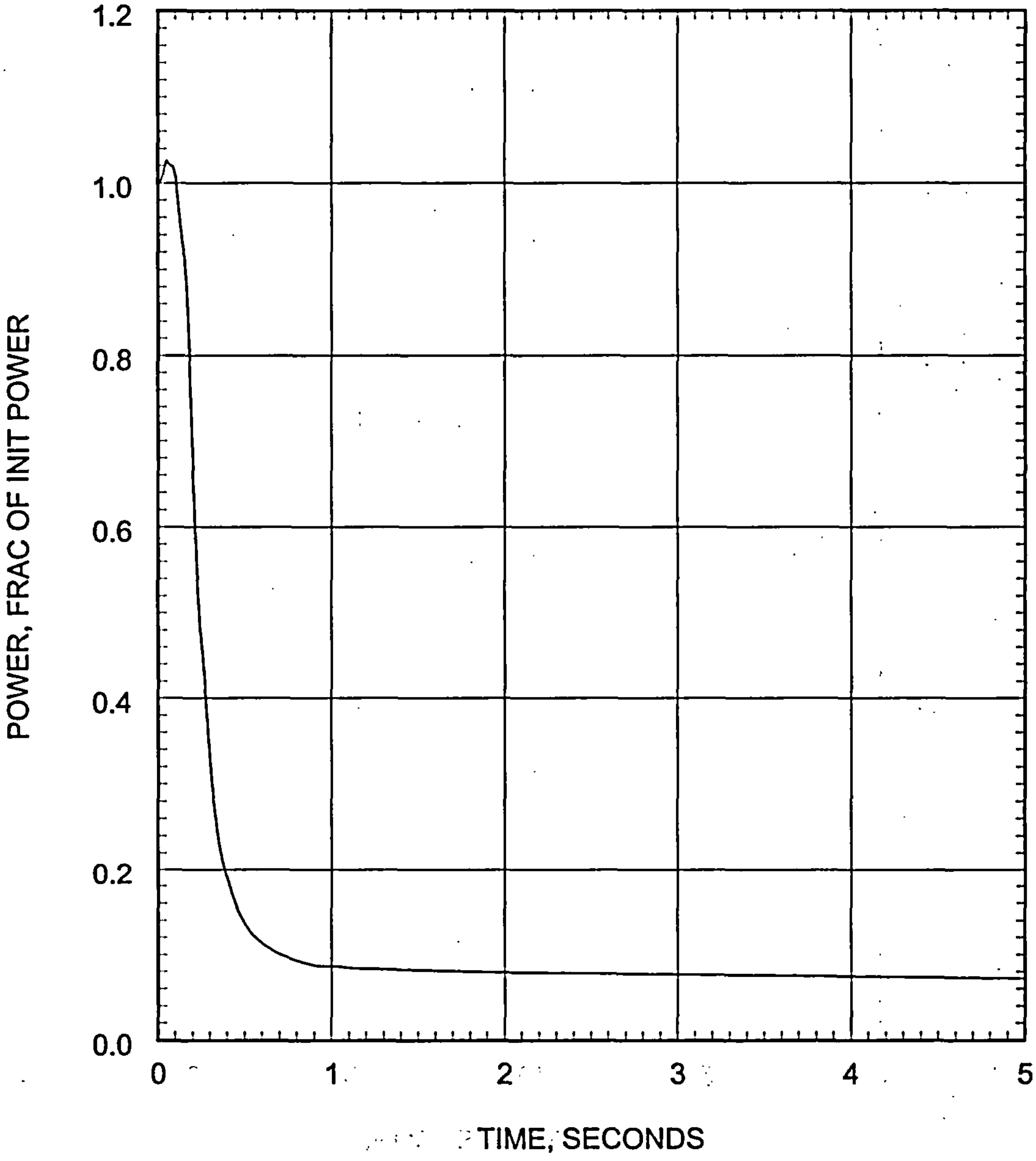

Figure 5.2.3.3-10 Large Break LOCA ECCS Performance Analysis 0.8 DEG/PD Break Core Power

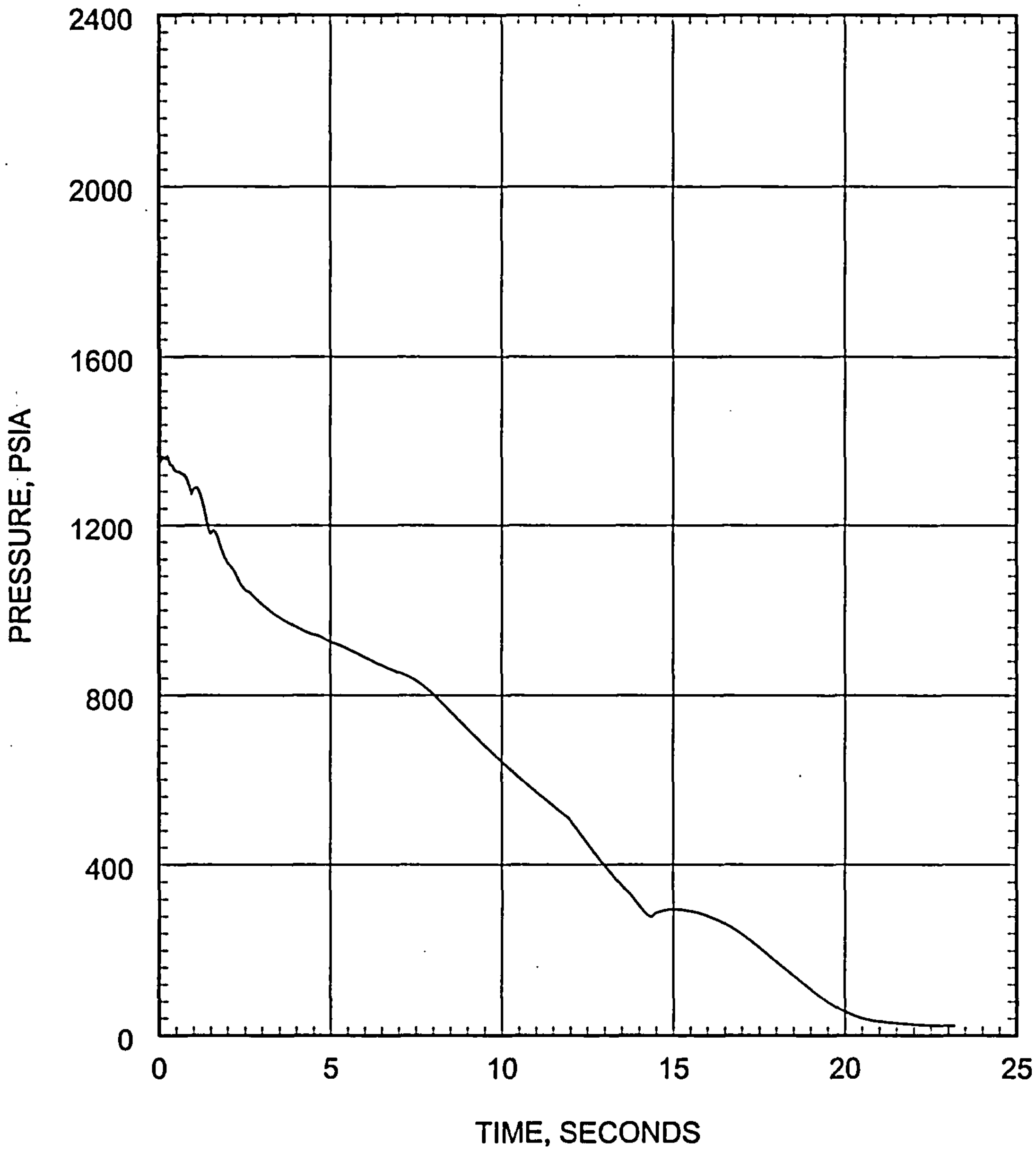

Figure 5.2.3.3-11 Large Break LOCA ECCS Performance Analysis 0.8 DEG/PD Break Pressure in Center Hot Assembly Node

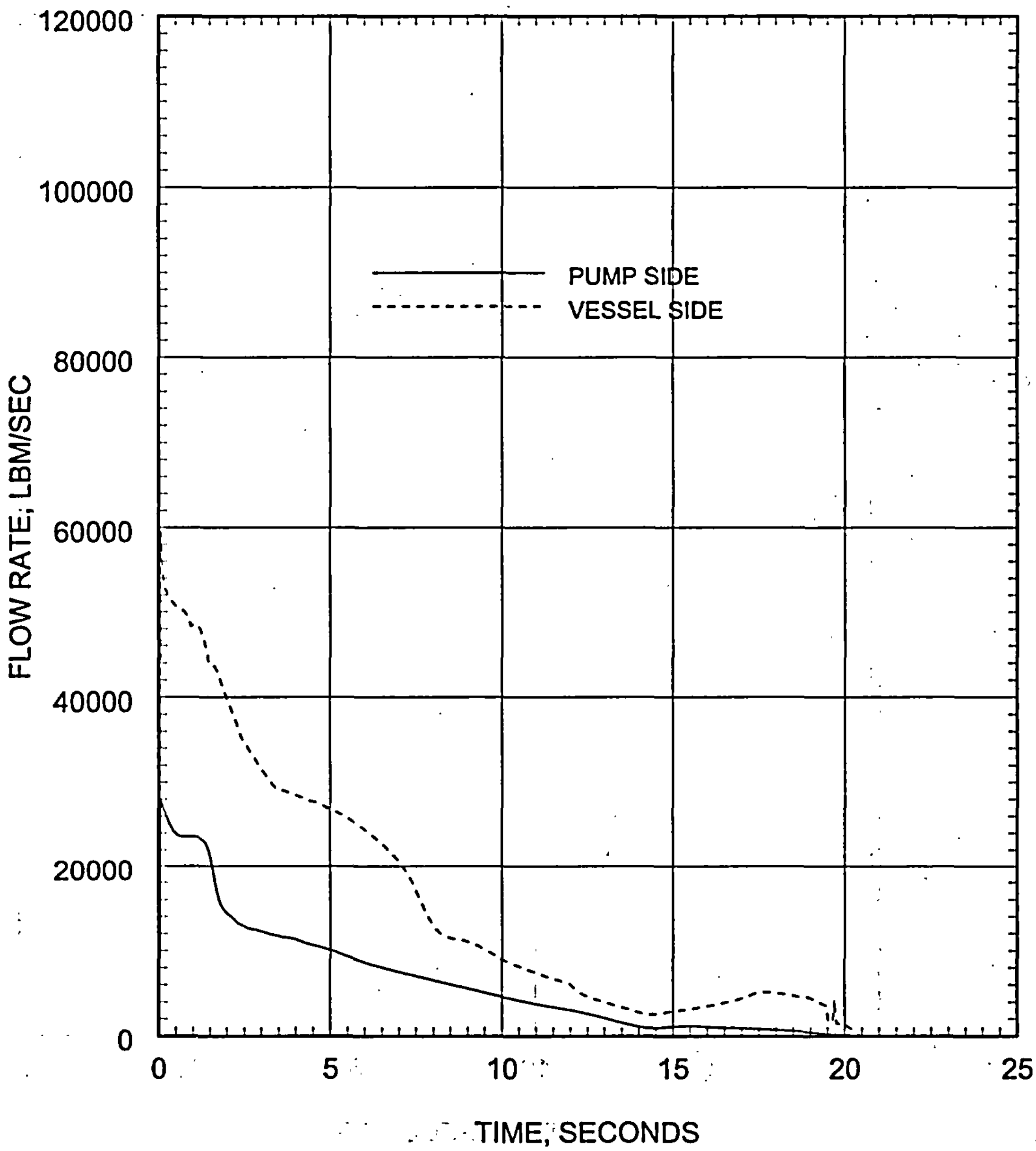

Figure 5.2.3.3-12 Large Break LOCA ECCS Performance Analysis 0.8 DEG/PD Break Leak Flow Rate

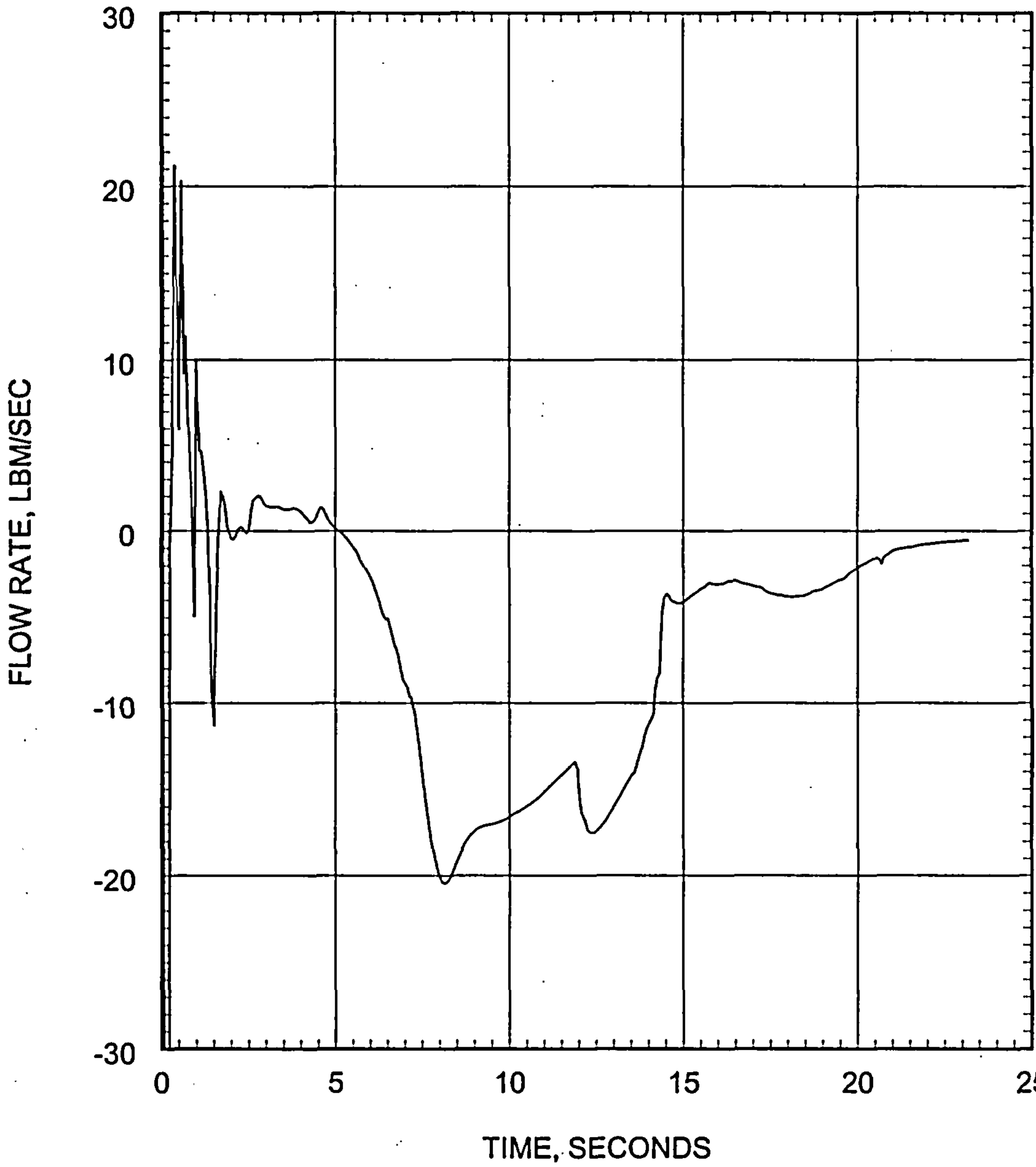

Figure 5.2.3.3-13 Large Break LOCA ECCS Performance Analysis 0.8 DEG/PD Break Hot Assembly Flow Rate (Below Hot Spot)

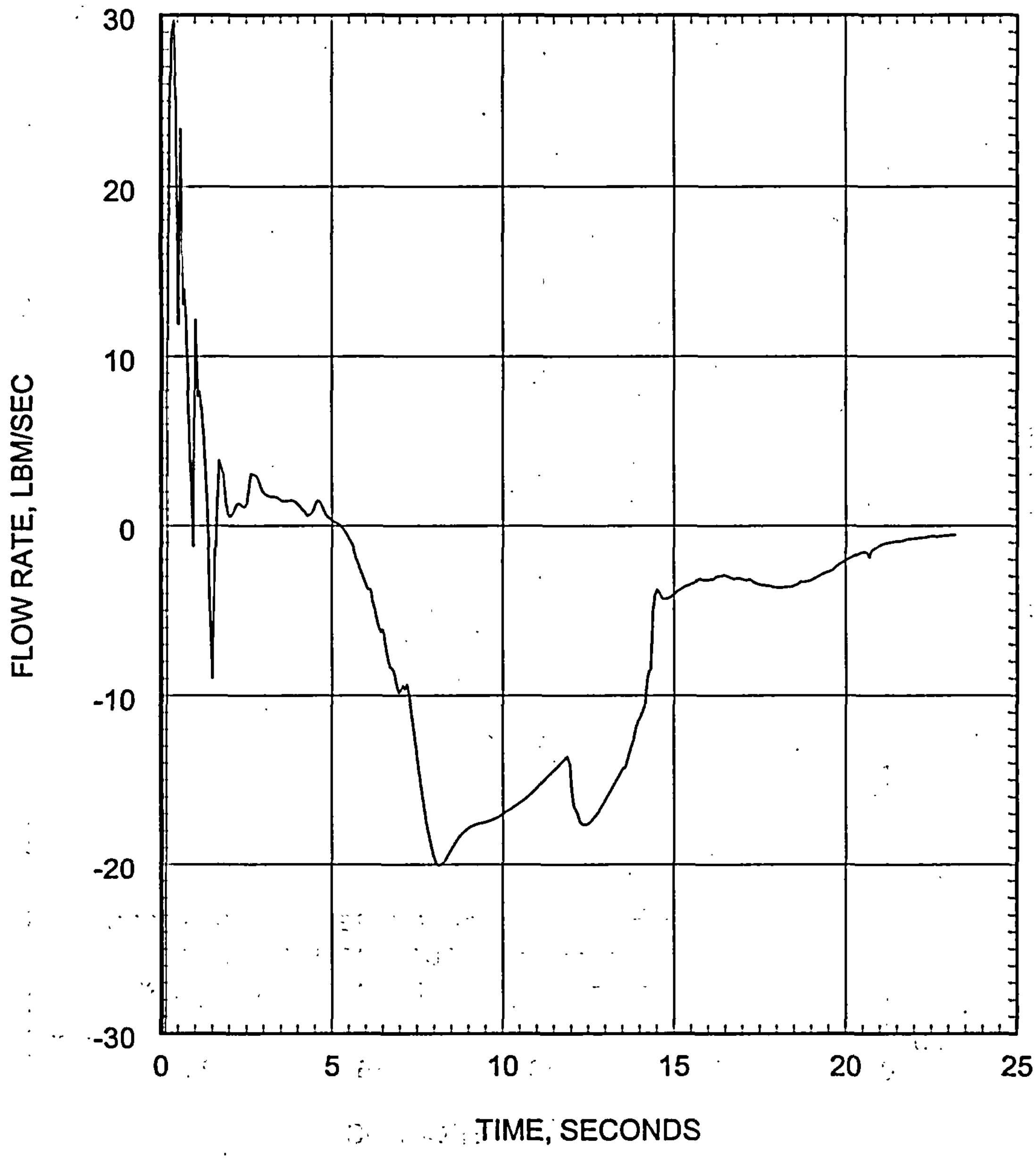

Figure 5.2.3.3-14 Large Break LOCA ECCS Performance Analysis 0.8 DEG/PD Break Hot Assembly Flow Rate (Above Hot Spot)

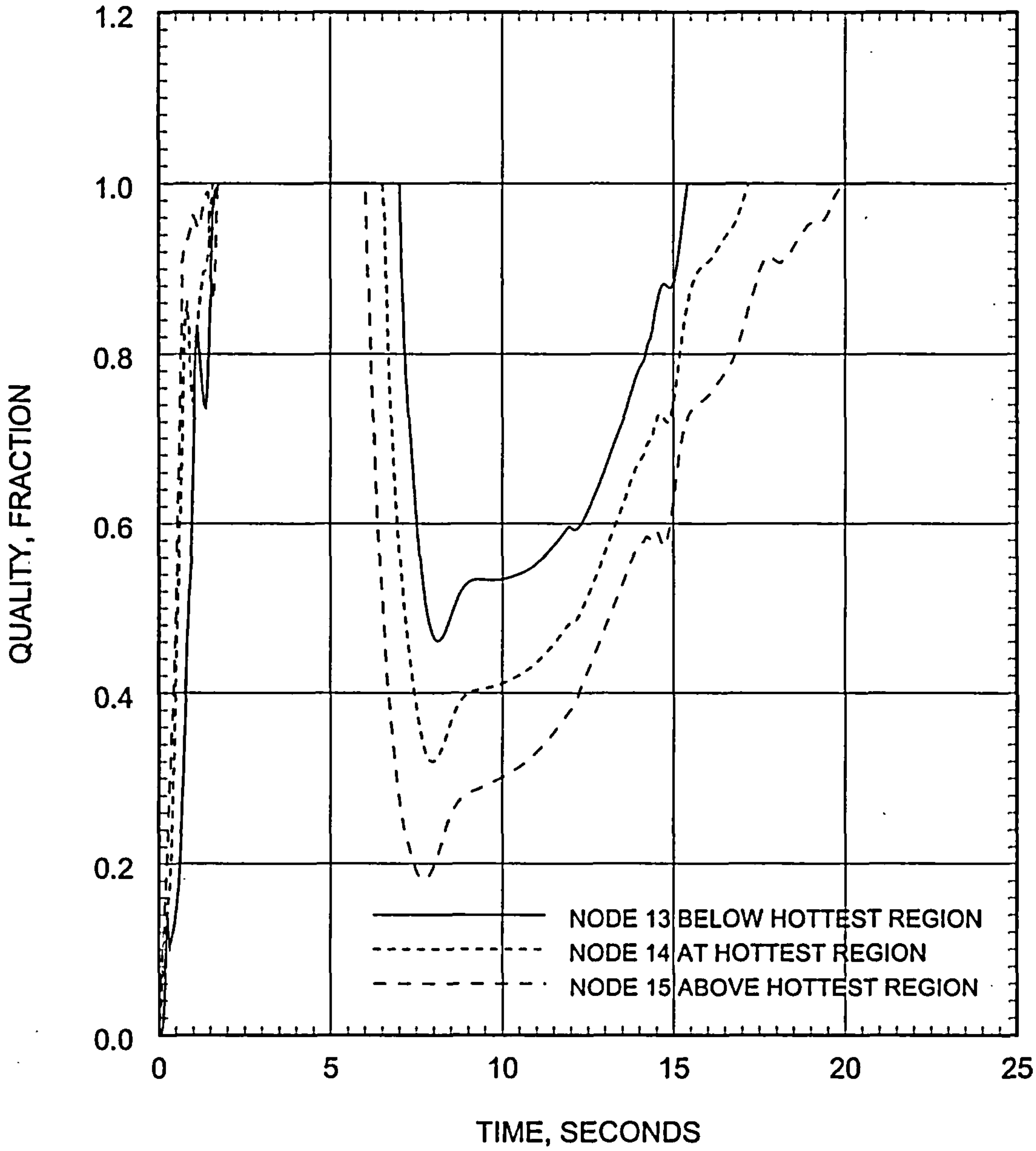

Figure 5.2.3.3-15 Large Break LOCA ECCS Performance Analysis 0.8 DEG/PD Break Hot Assembly Quality

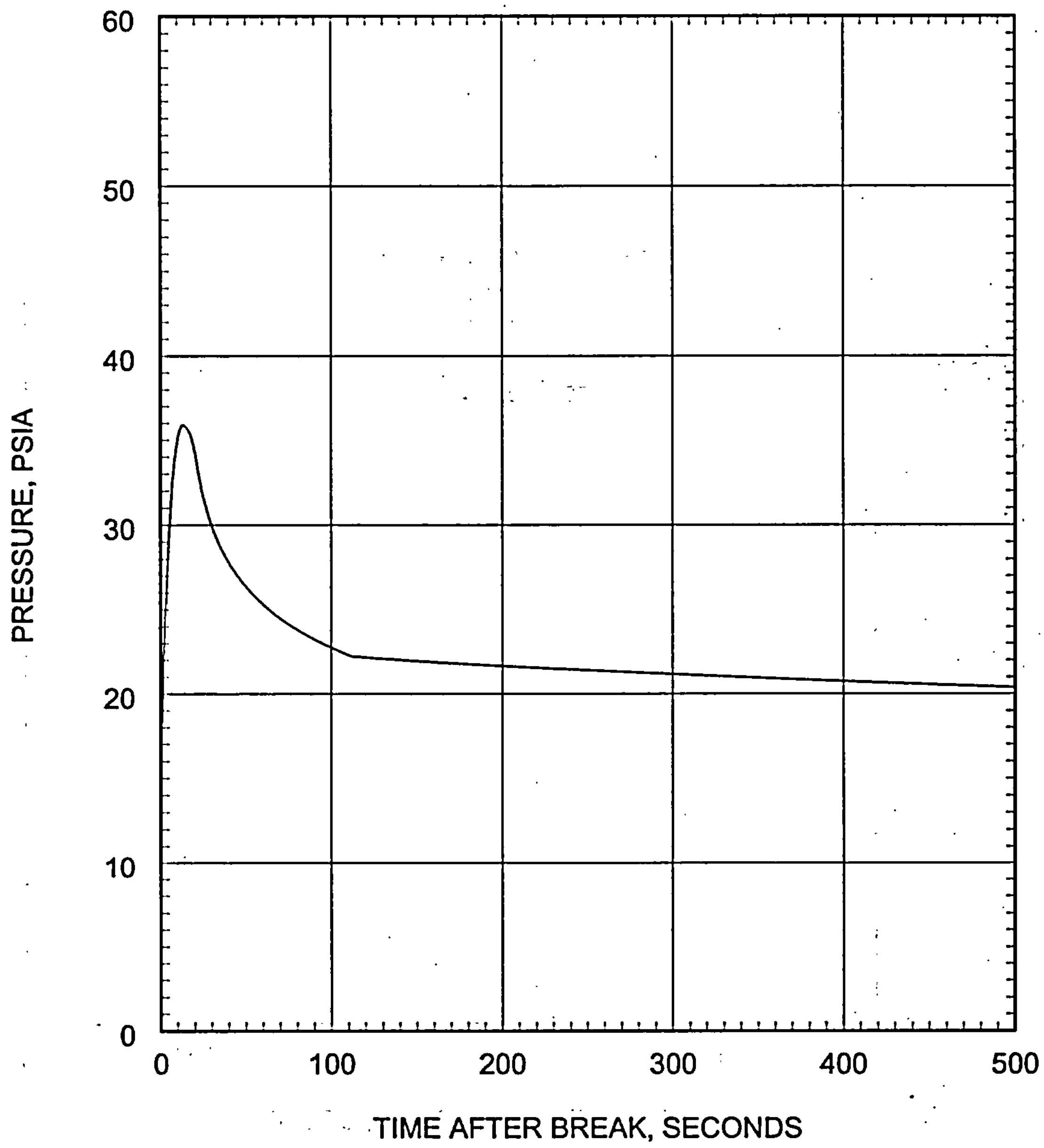

Figure 5.2.3.3-16 Large Break LOCA ECCS Performance Analysis 0.8 DEG/PD Break Containment Pressure

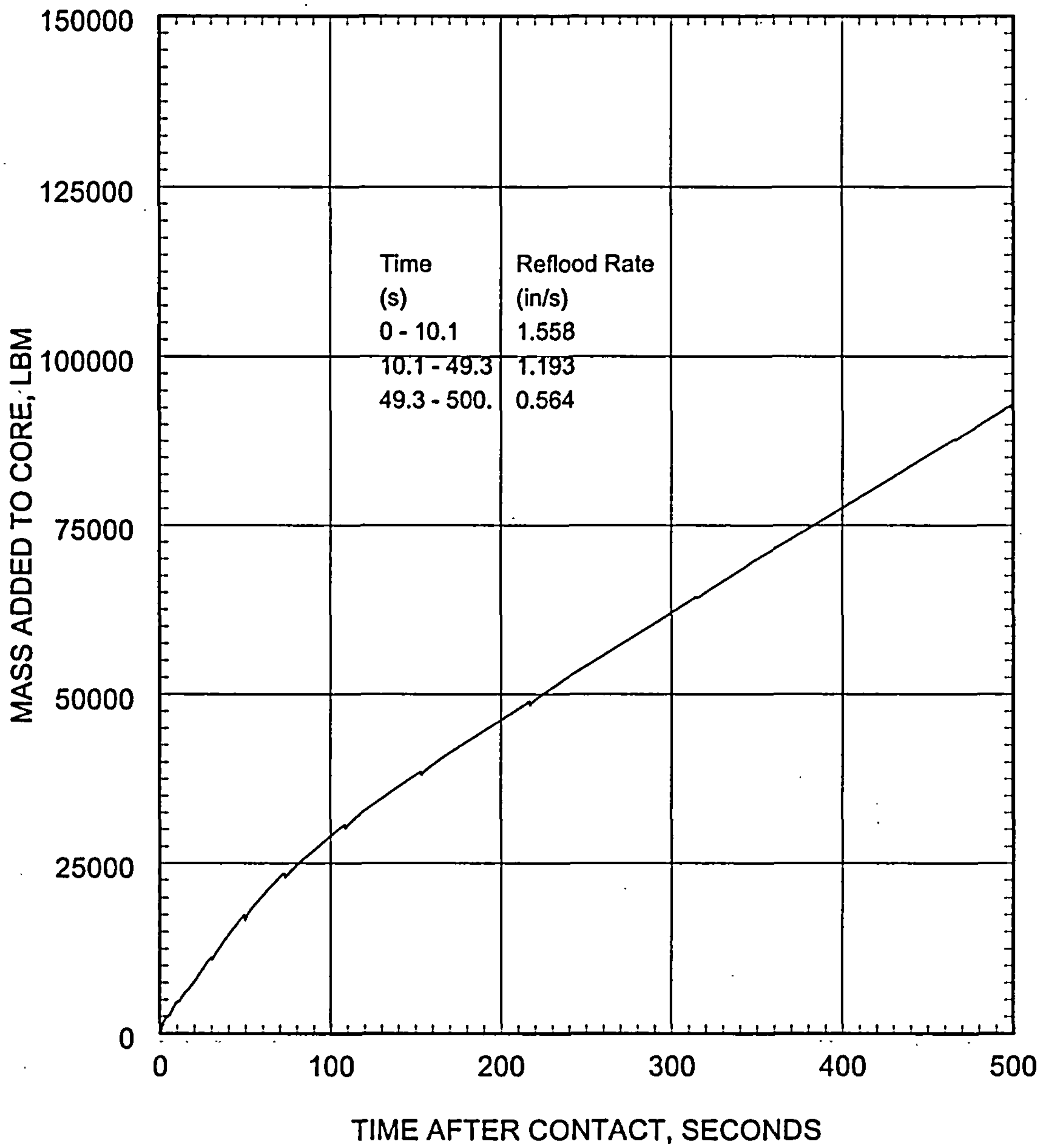

Figure 5.2.3.3-17 Large Break LOCA ECCS Performance Analysis 0.8 DEG/PD Break Mass Added to Core During Reflood

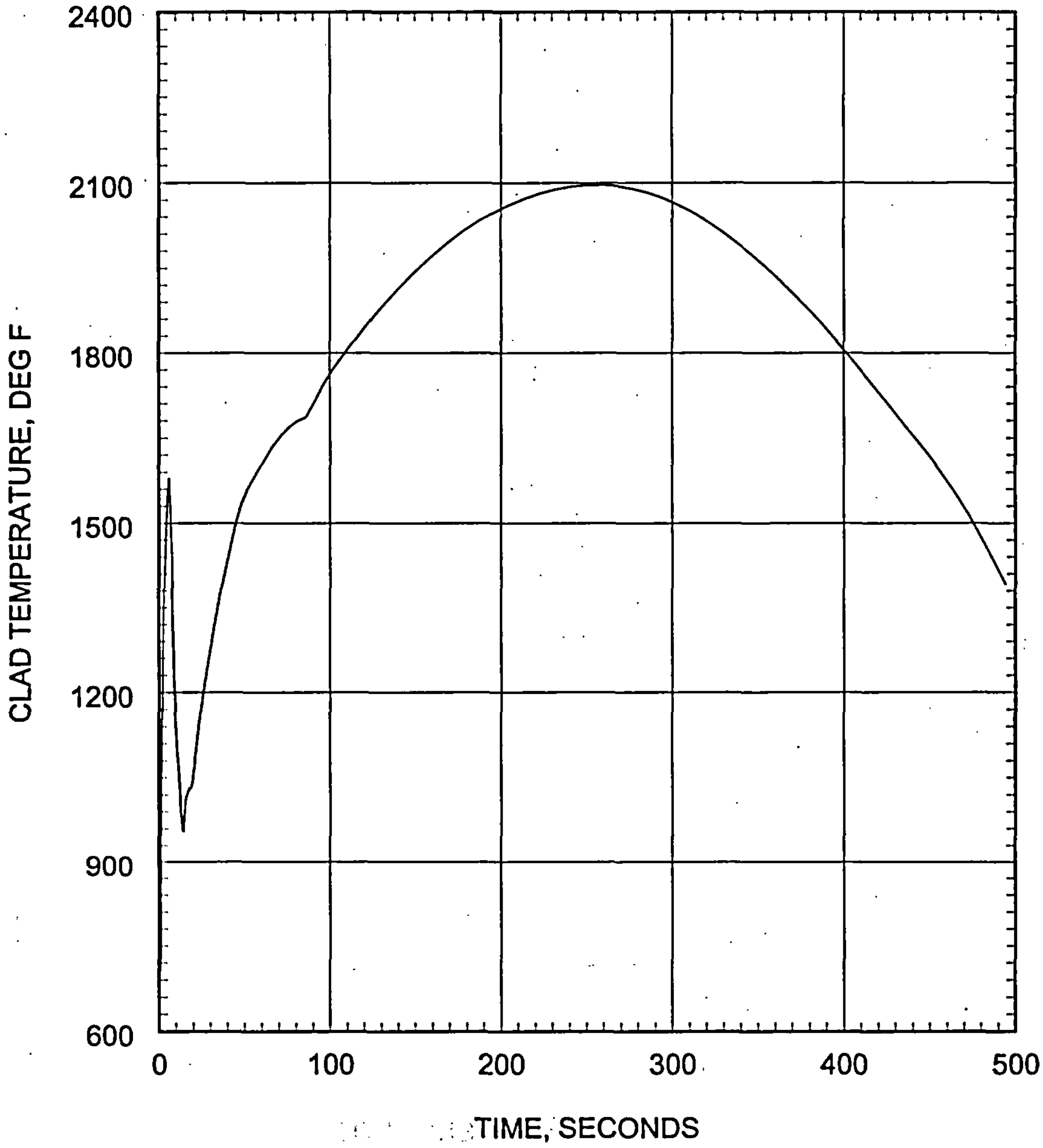

Figure 5.2.3.3-18 Large Break LOCA ECCS Performance Analysis 0.8 DEG/PD Break Peak Cladding Temperature

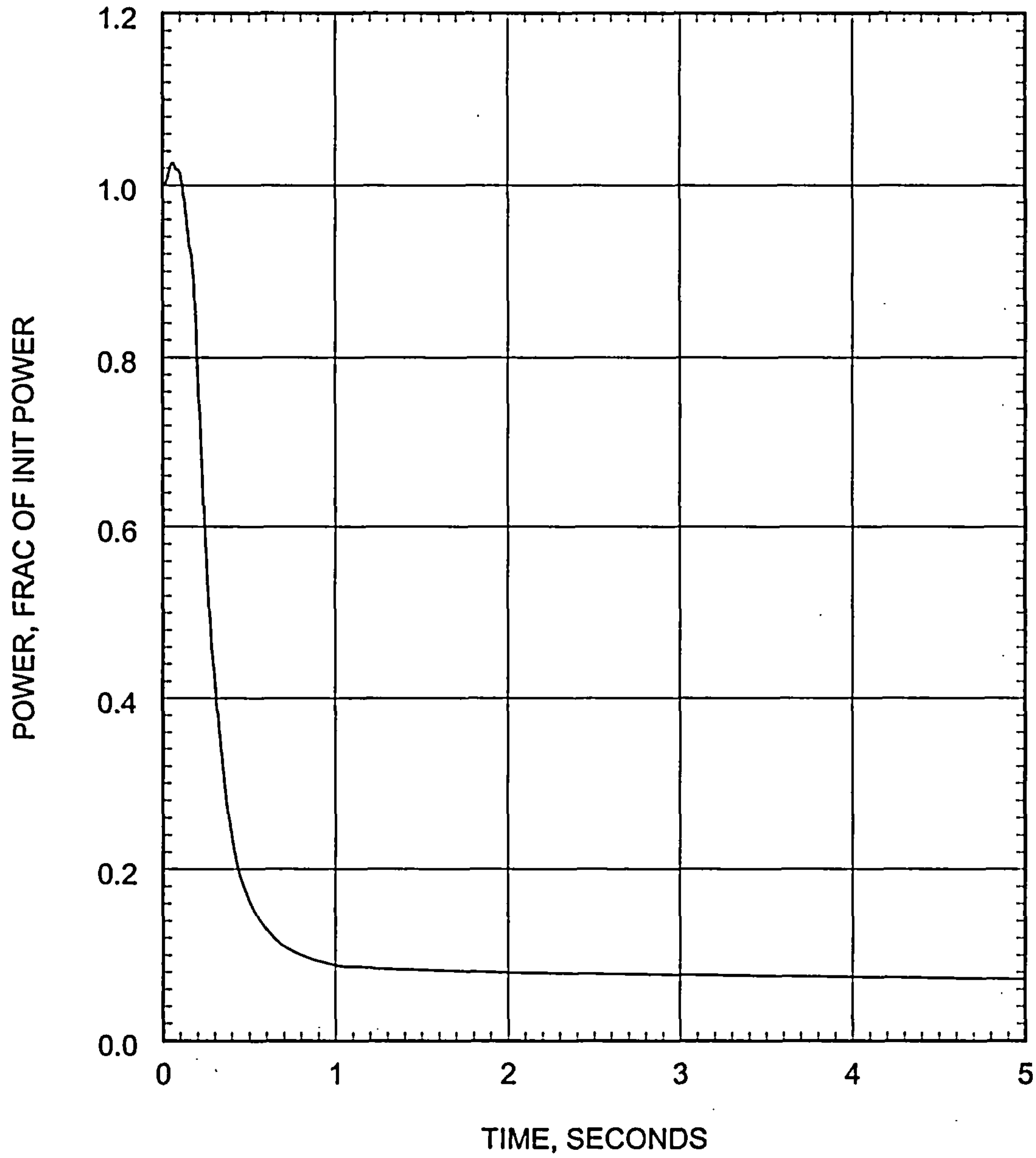

Figure 5.2.3.3-19 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Core Power

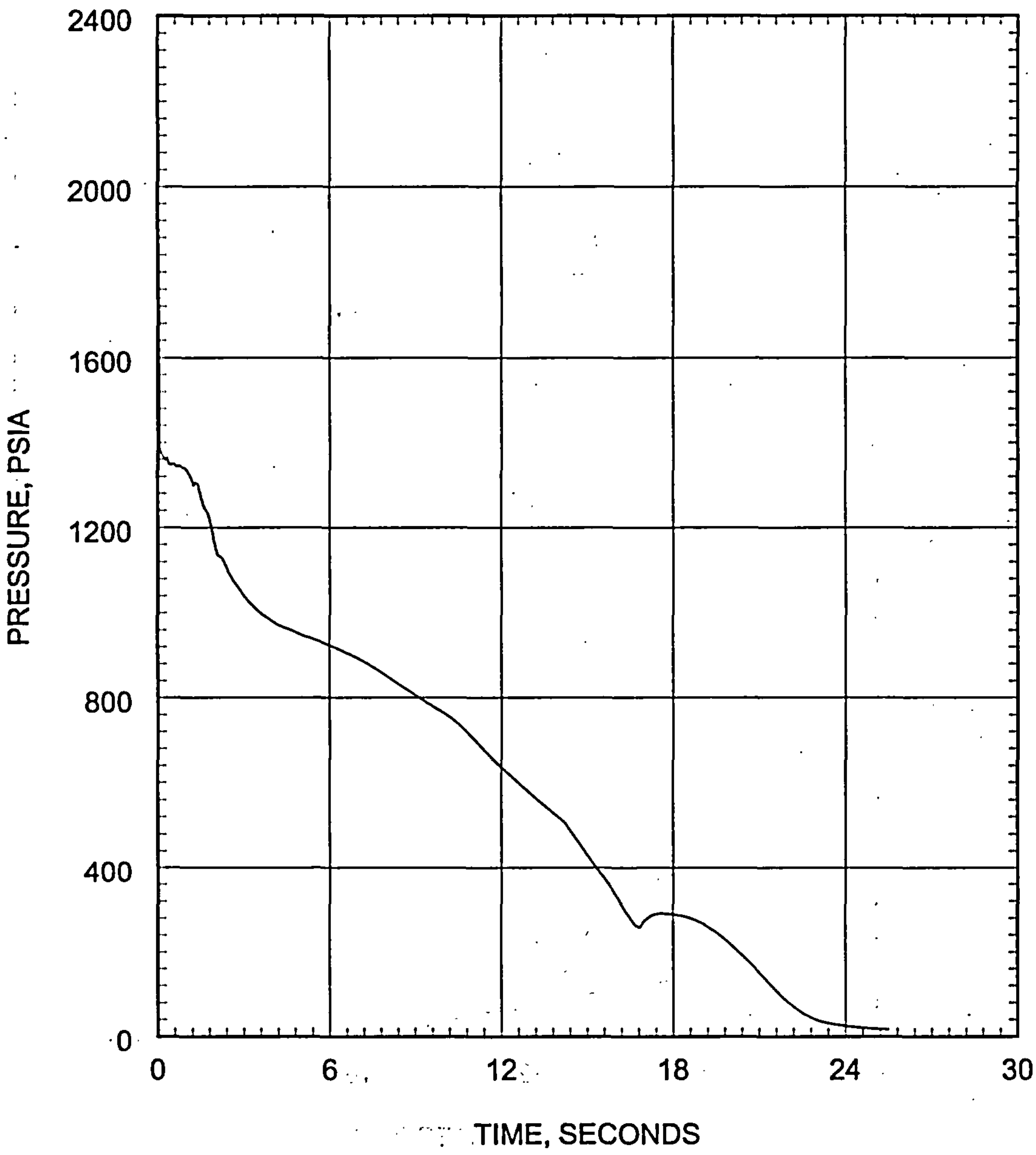

Figure 5.2.3.3-20 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Pressure in Center Hot Assembly Node

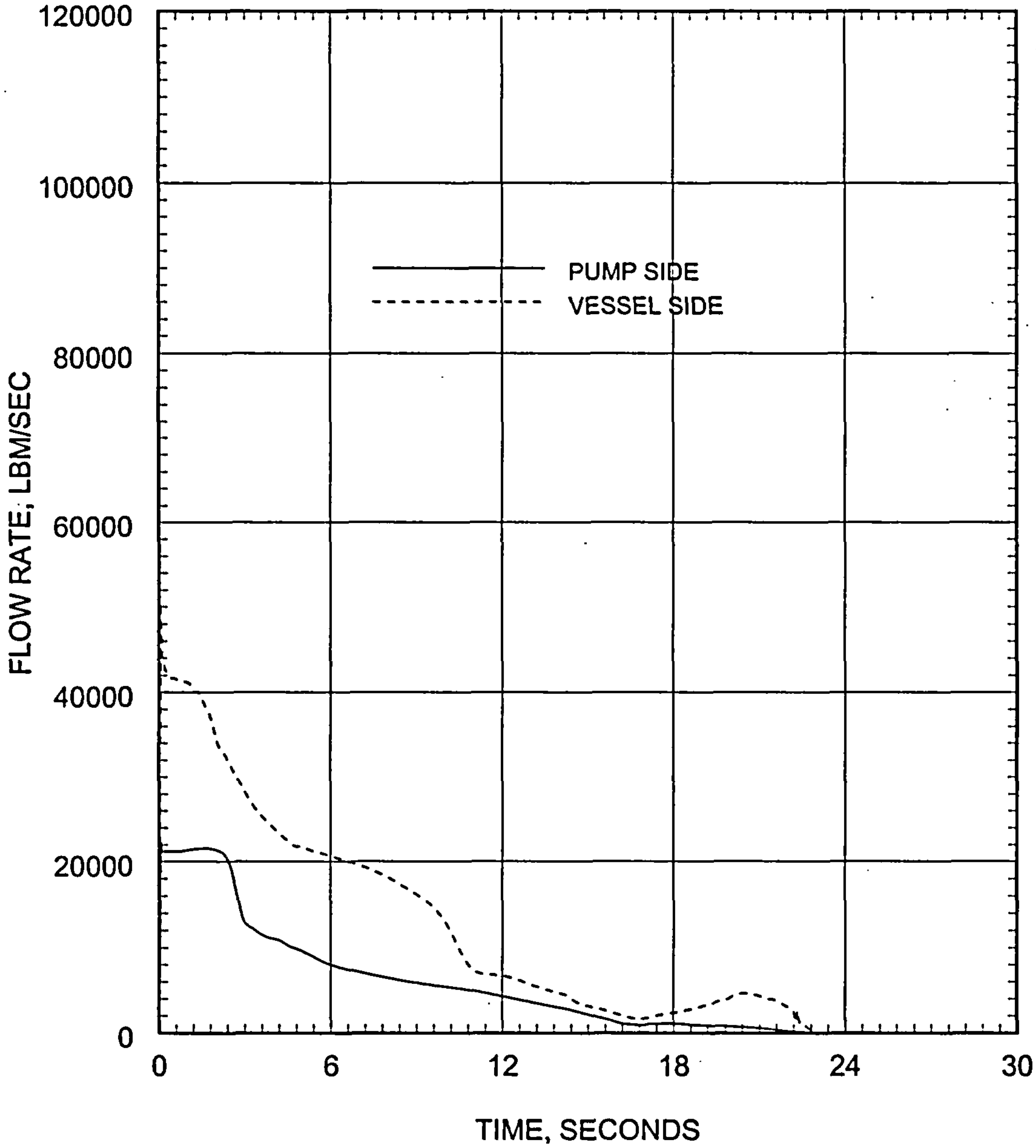

Figure 5.2.3.3-21 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Leak Flow Rate

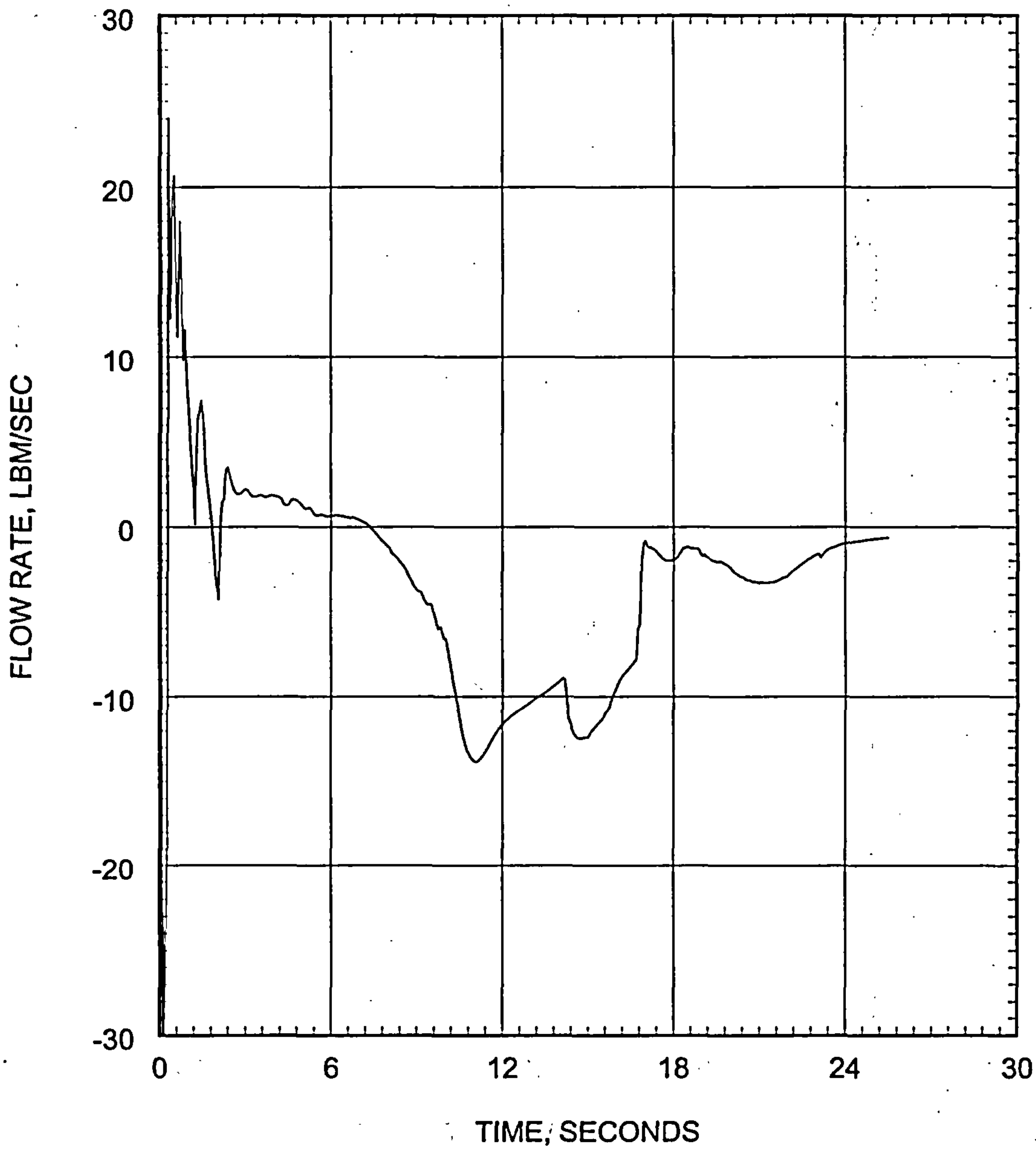

Figure 5.2.3.3-22 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Hot Assembly Flow Rate (Below Hot Spot)

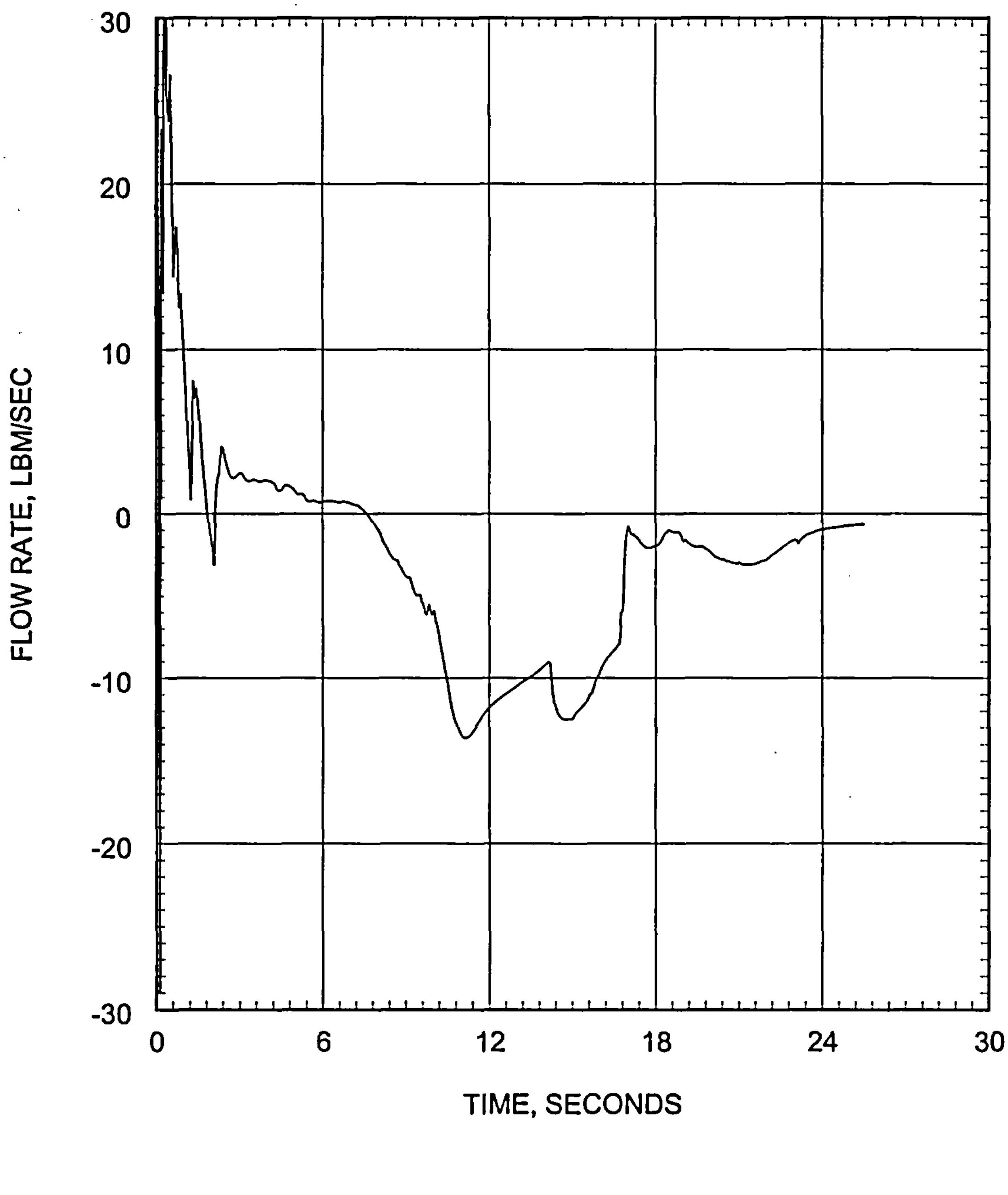

Figure 5.2.3.3-23 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Hot Assembly Flow Rate (Above Hot Spot)

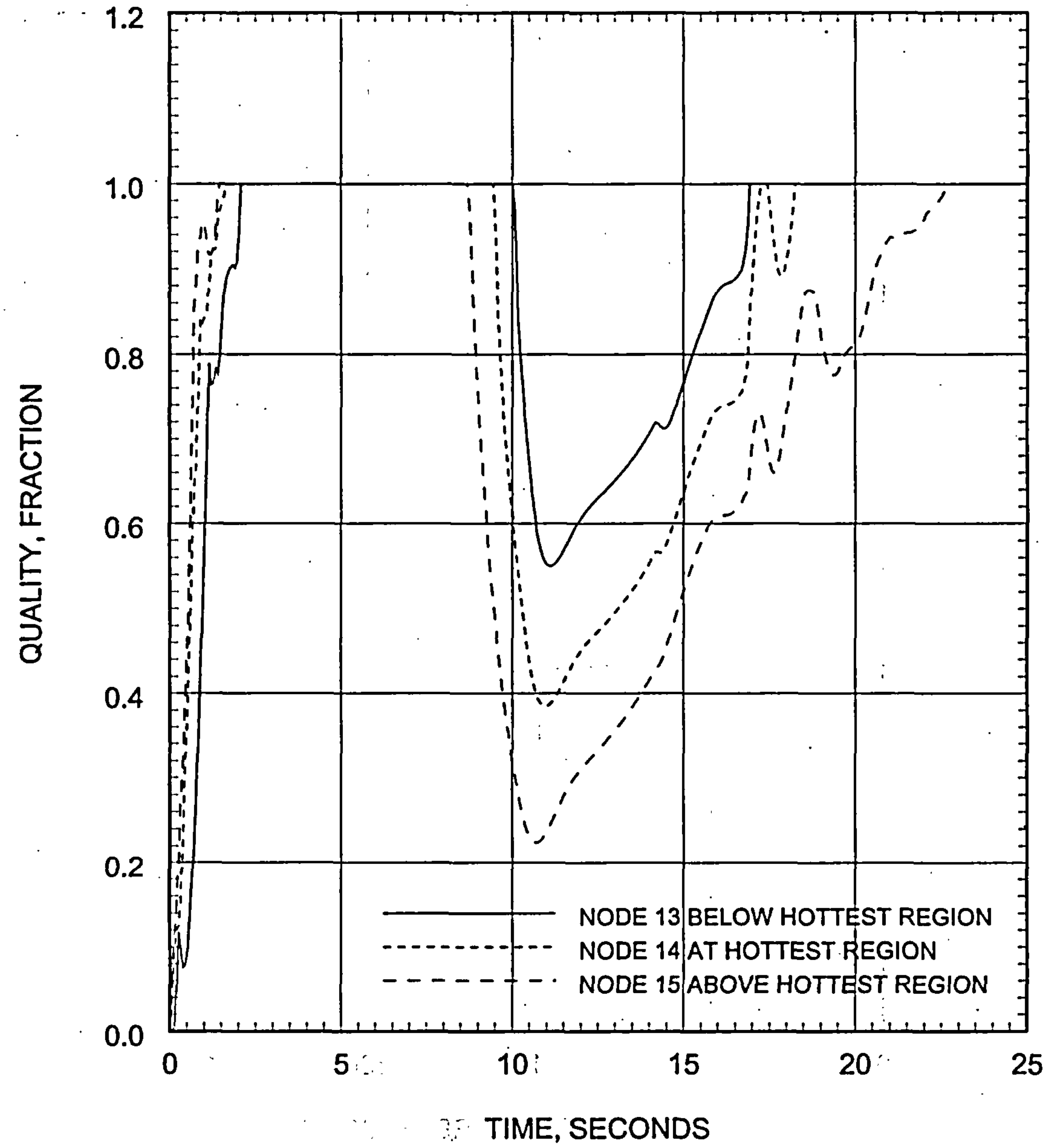

Figure 5.2.3.3-24 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Hot Assembly Quality

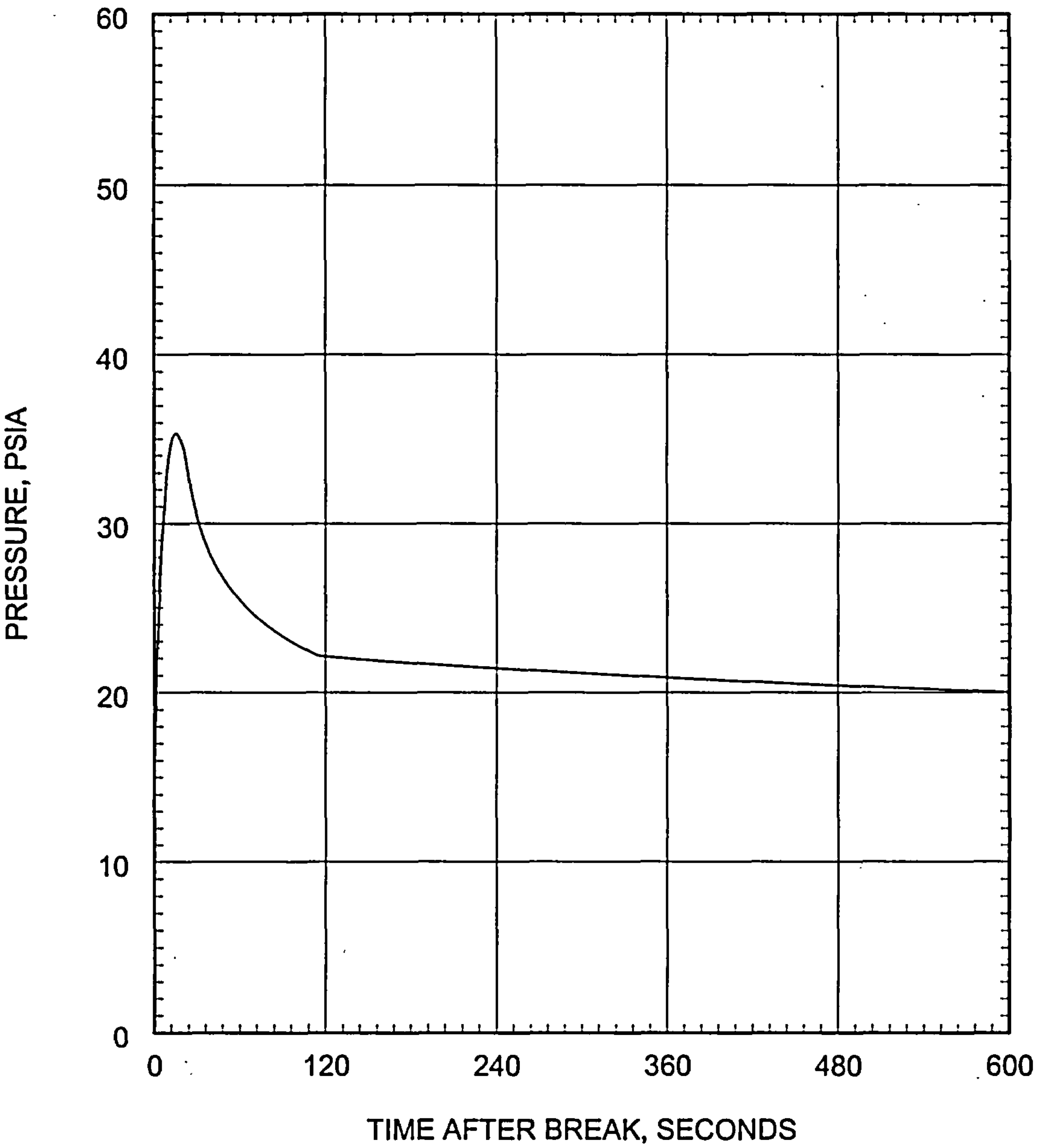

Figure 5.2.3.3-25 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Containment Pressure

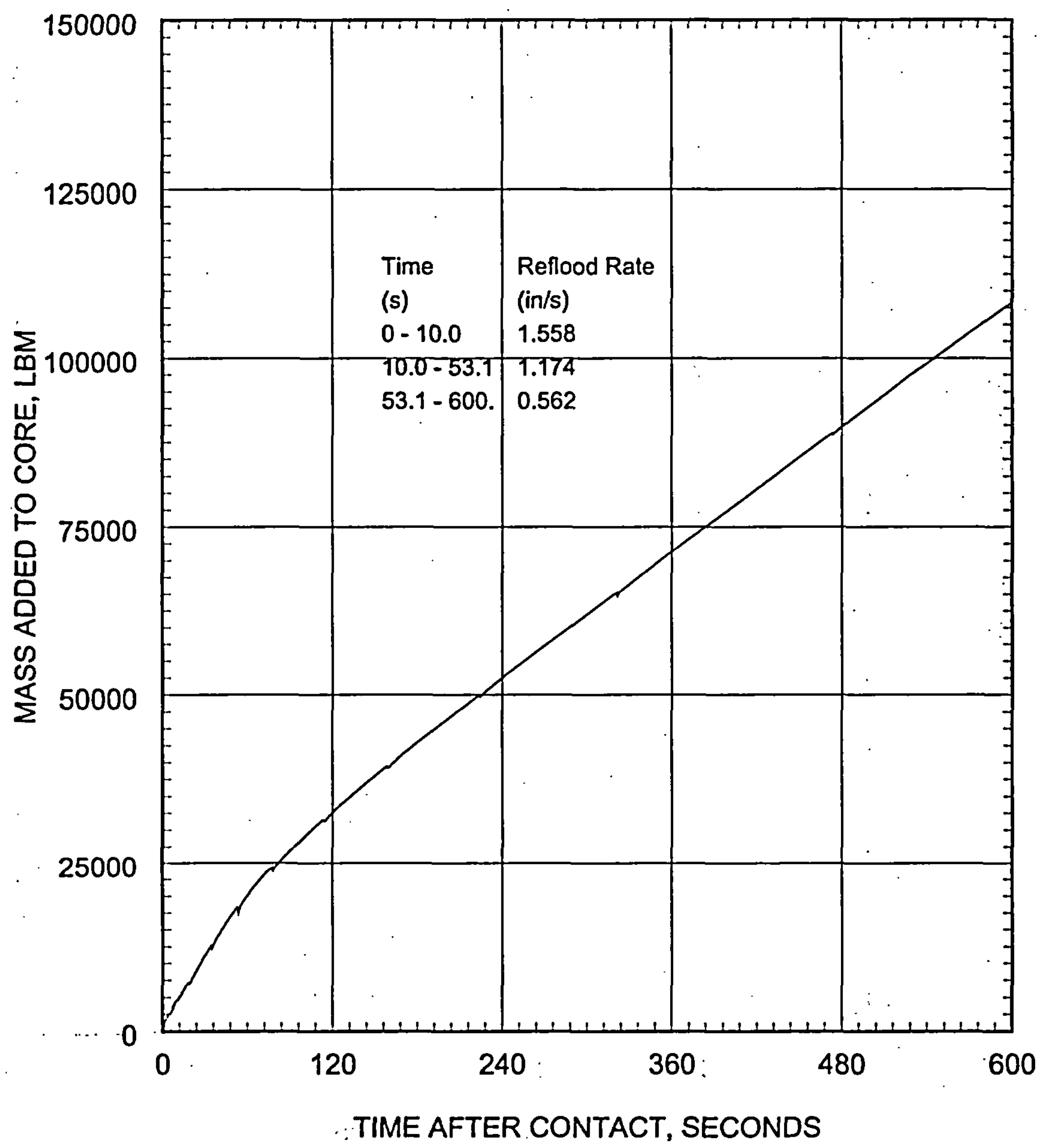

Figure 5.2.3.3-26 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Mass Added to Core During Reflood

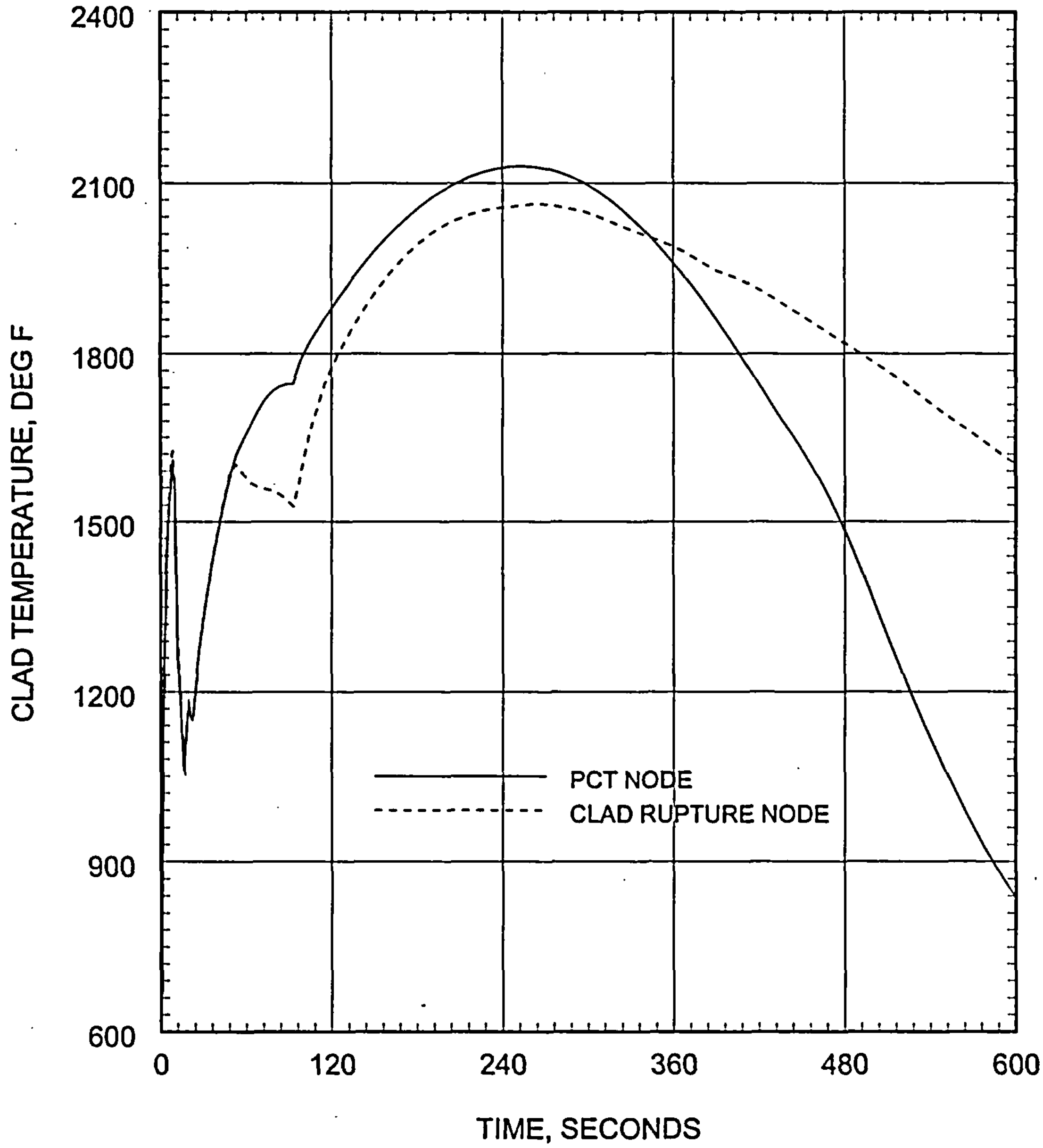

Figure 5.2.3.3-27 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Peak Cladding Temperature

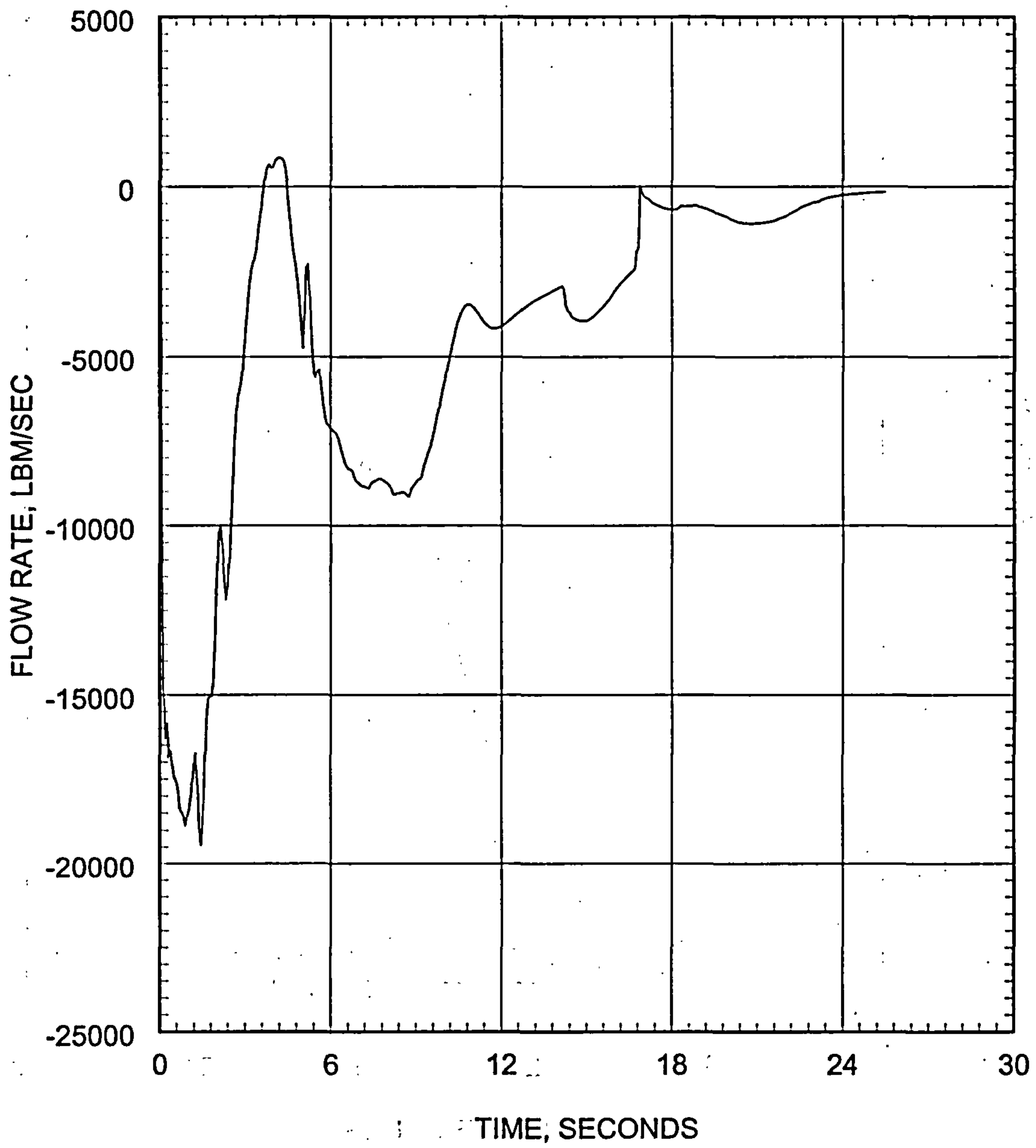

Figure 5.2.3.3-28 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Mid Annulus Flow Rate

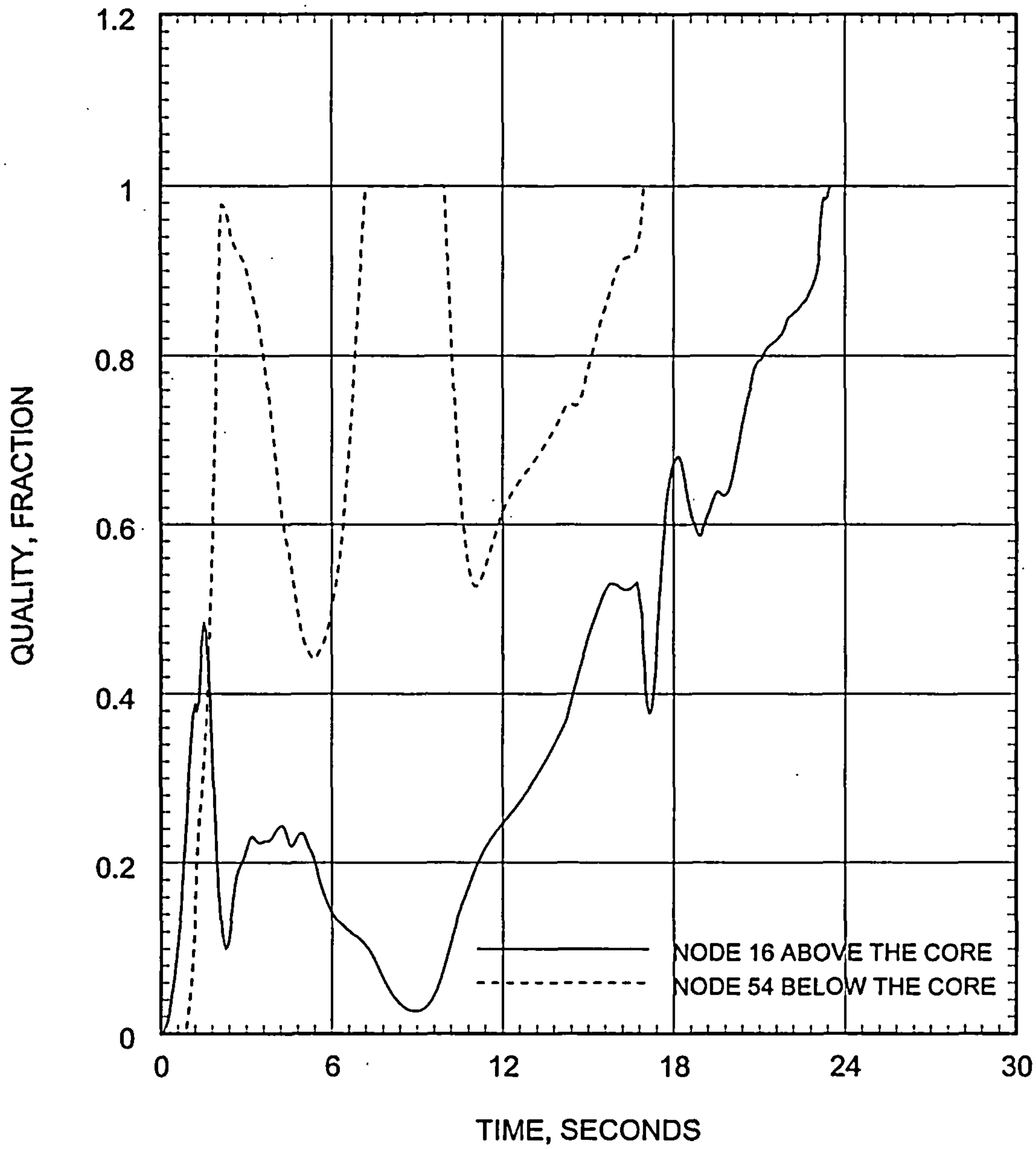

Figure 5.2.3.3-29 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Quality Above and Below the Core

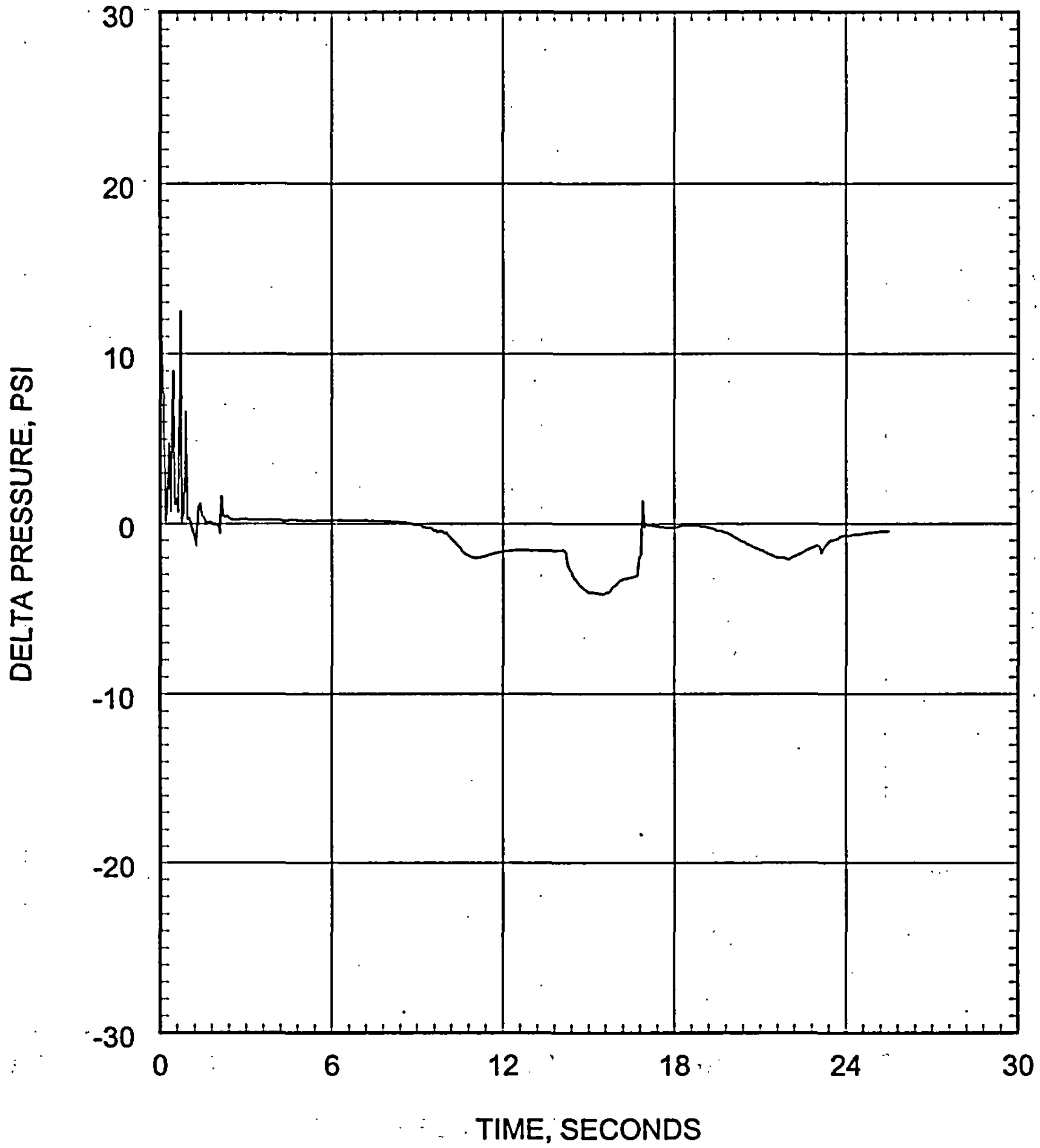

Figure 5.2.3.3-30 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Core Pressure Drop

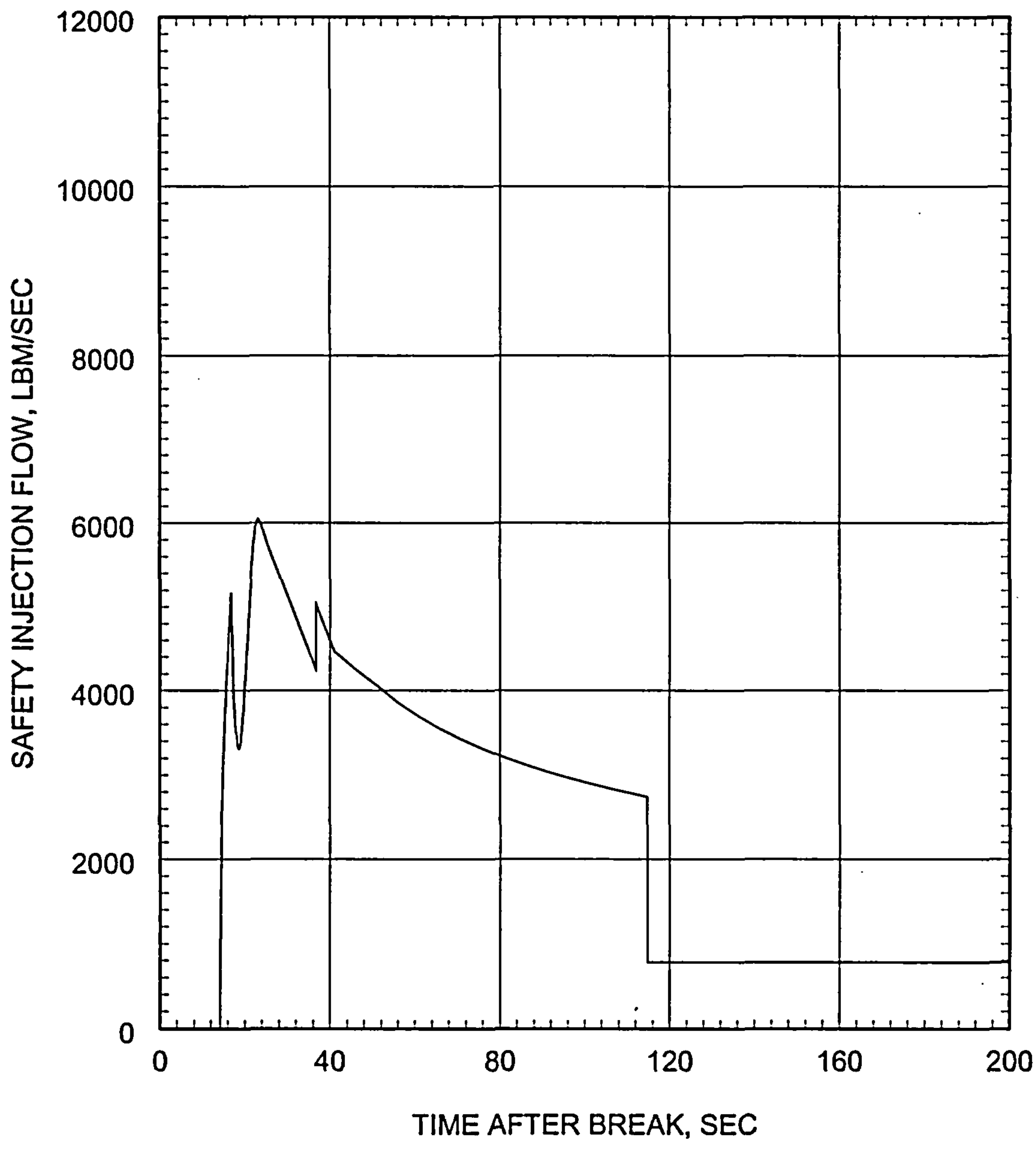

Figure 5.2.3.3-31 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Safety Injection Flow Rate into Intact Discharge Legs

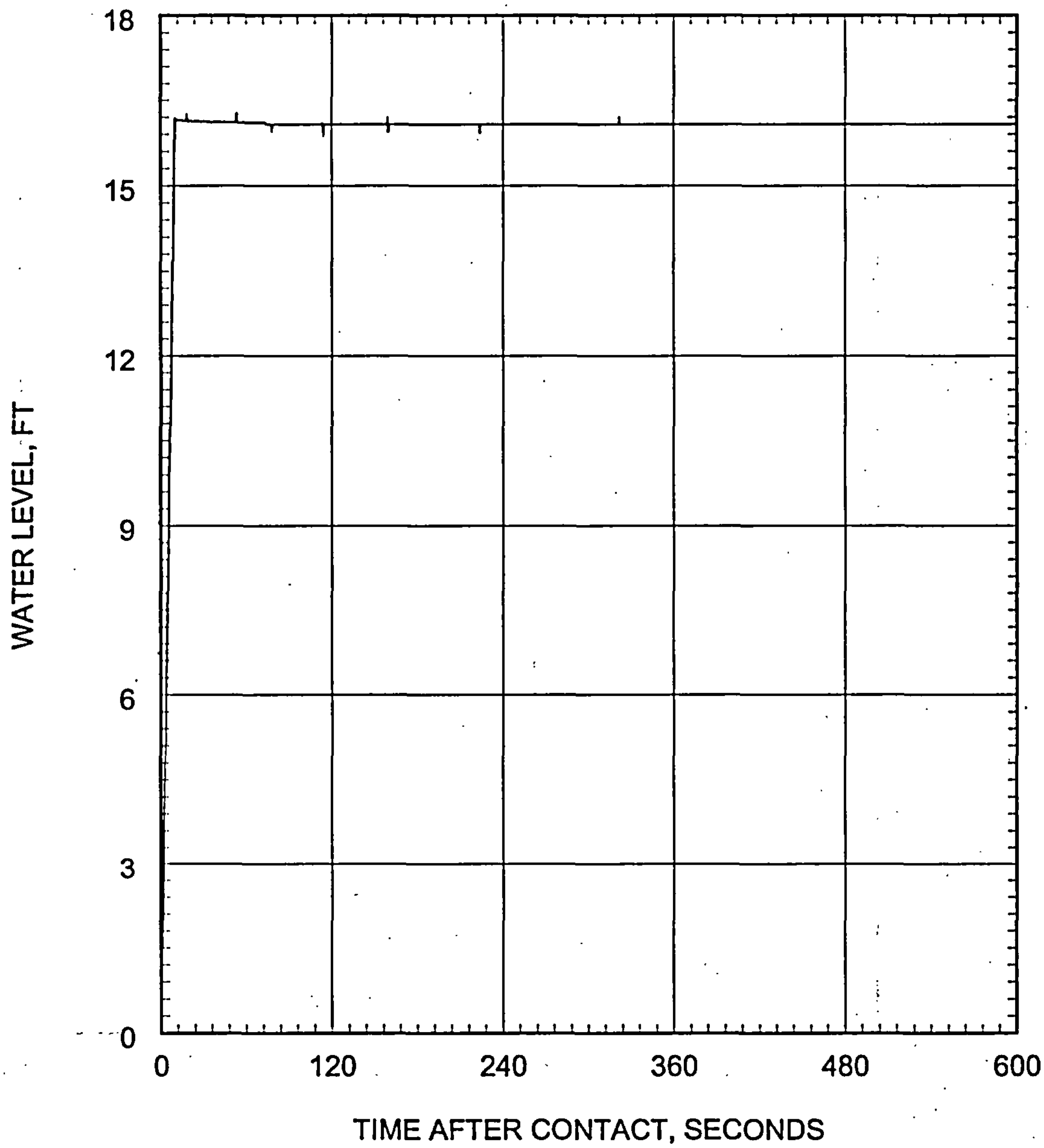

Figure 5.2.3.3-32 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Water Level in Downcomer During Reflood

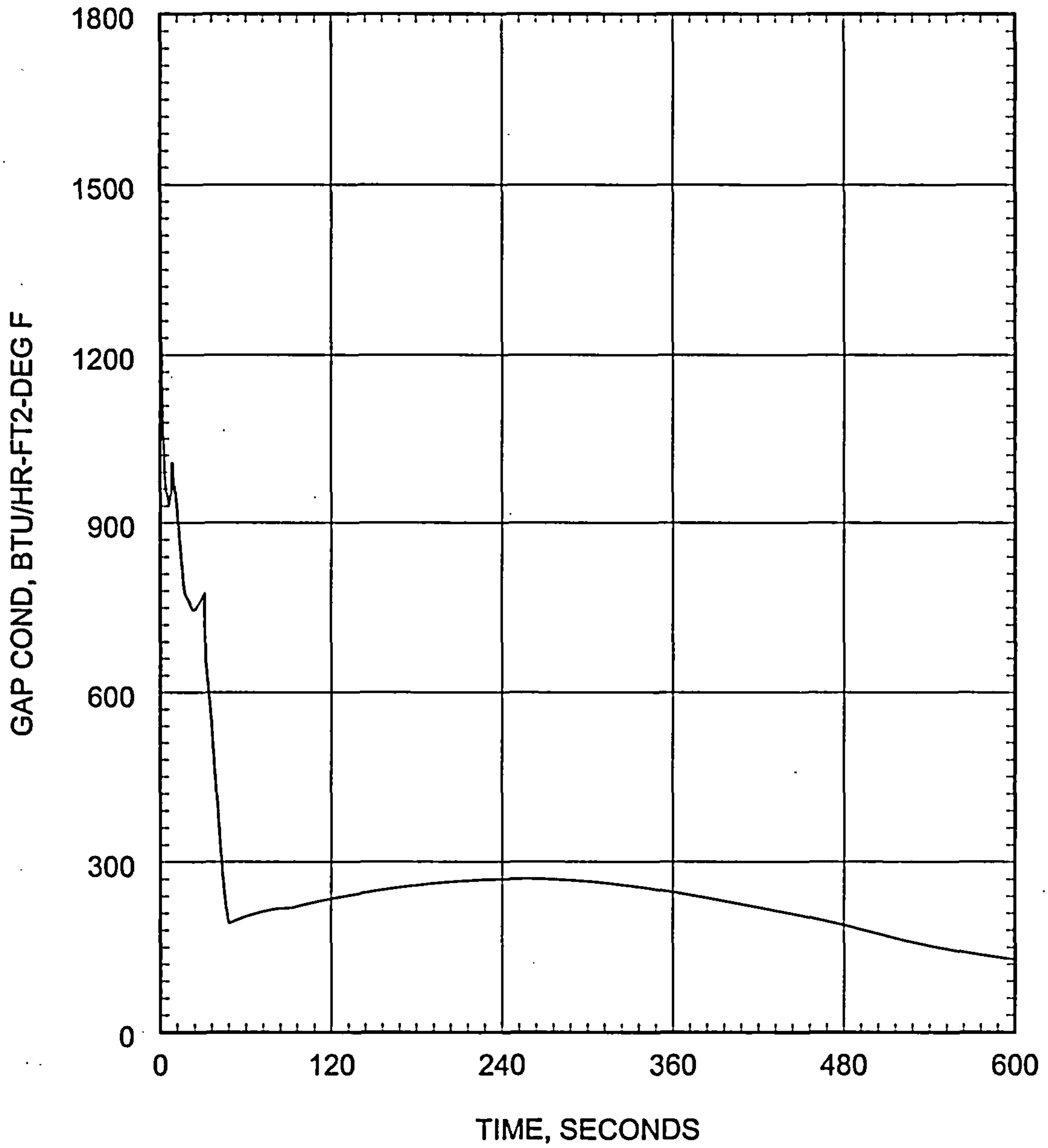

Figure 5.2.3.3-33 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Hot Spot Gap Conductance

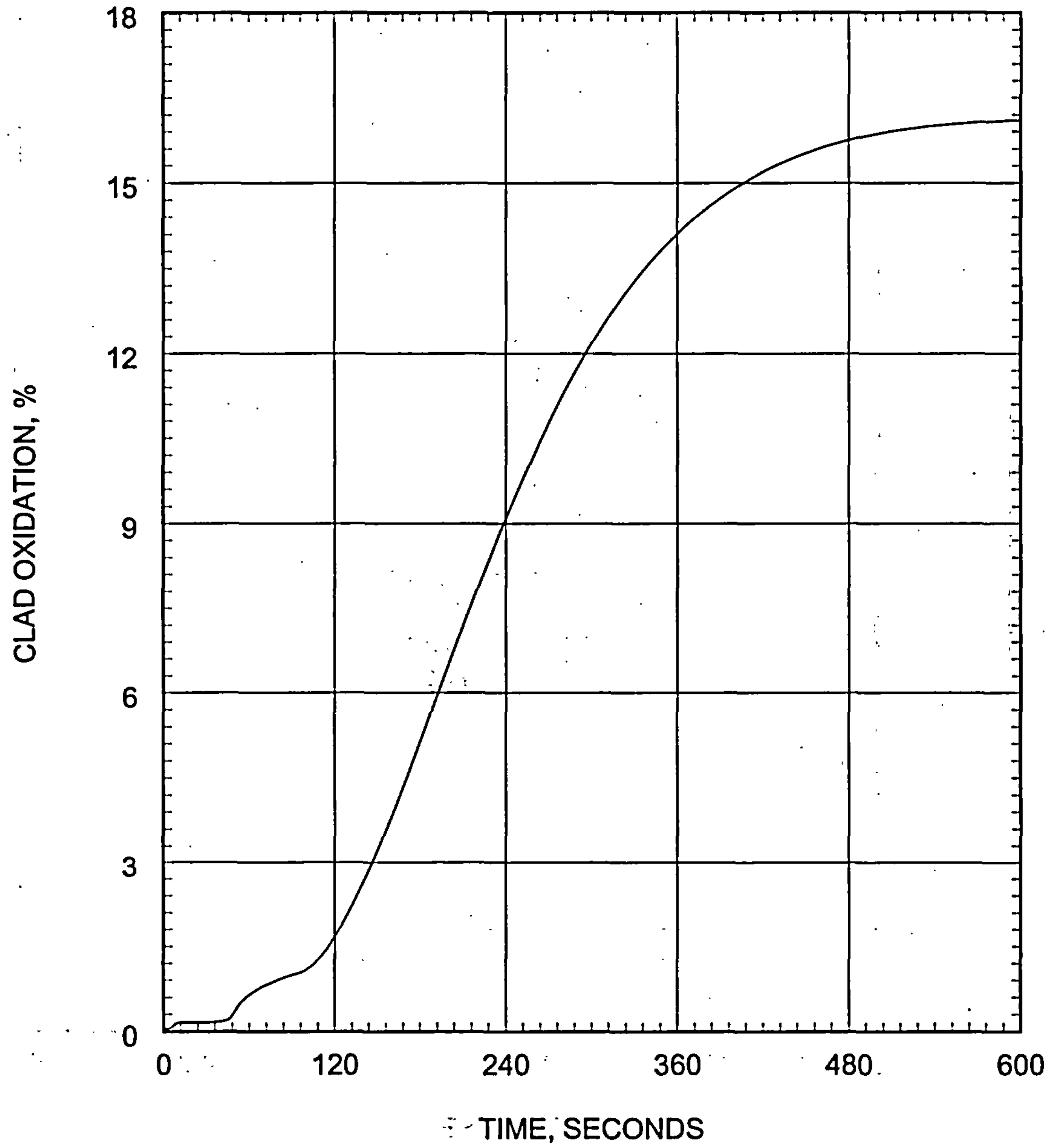

Figure 5.2.3.3-34 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Maximum Local Cladding Oxidation Percentage

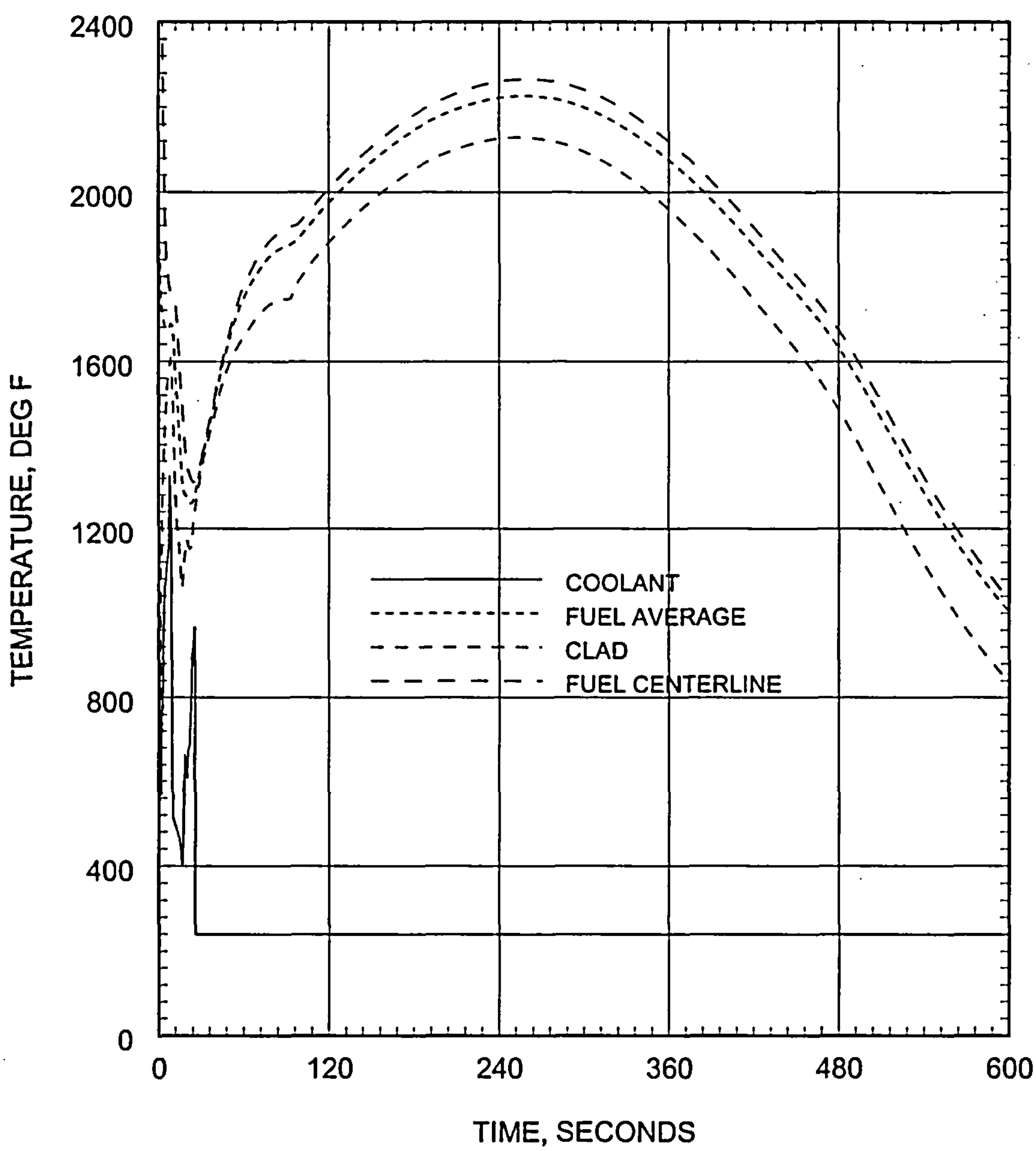

Figure 5.2.3.3-35 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Fuel Centerline, Fuel Average, Cladding, and Coolant Temperature at the Hot Spot

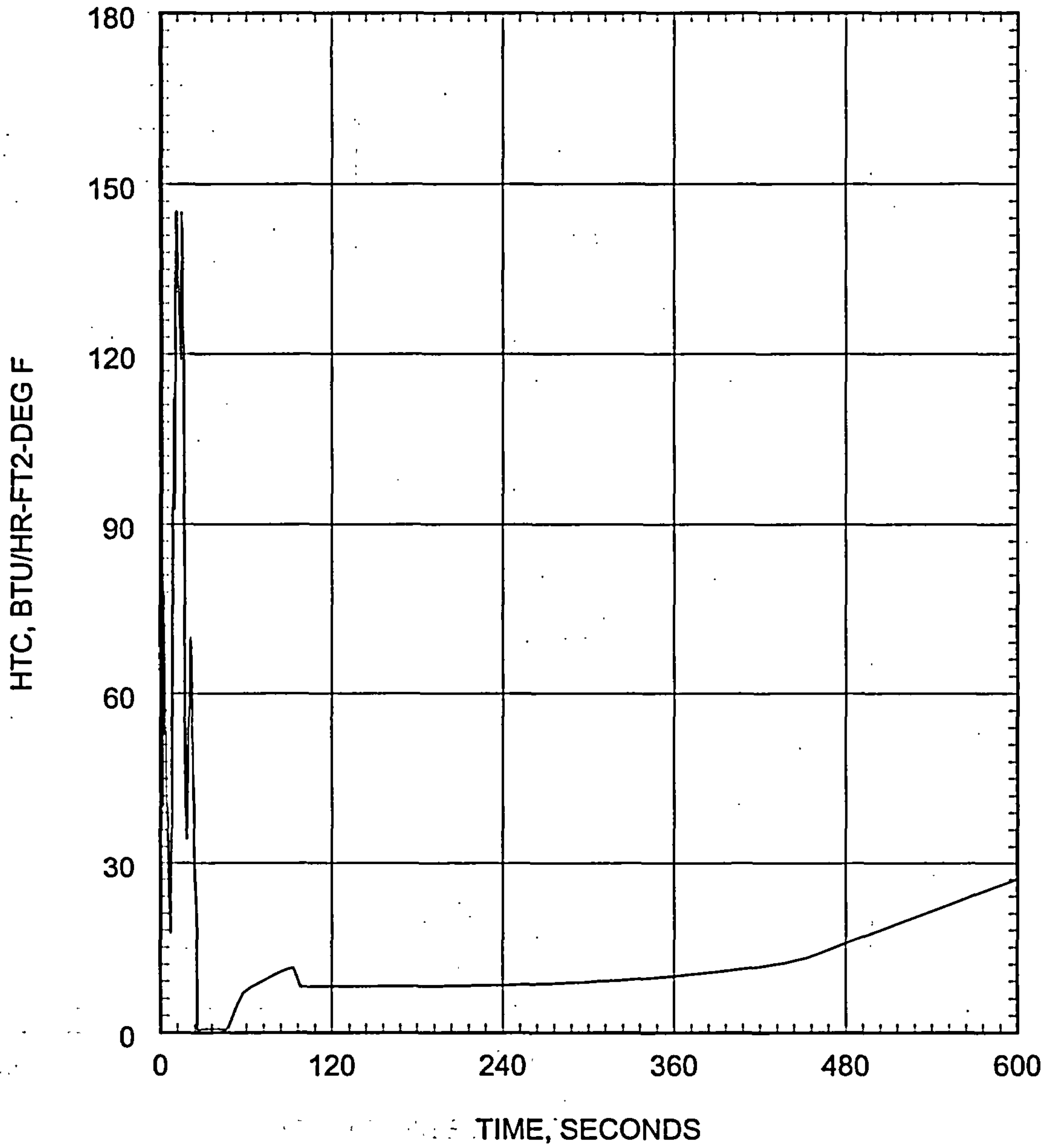

Figure 5.2.3.3-36 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Hot Spot Heat Transfer Coefficient

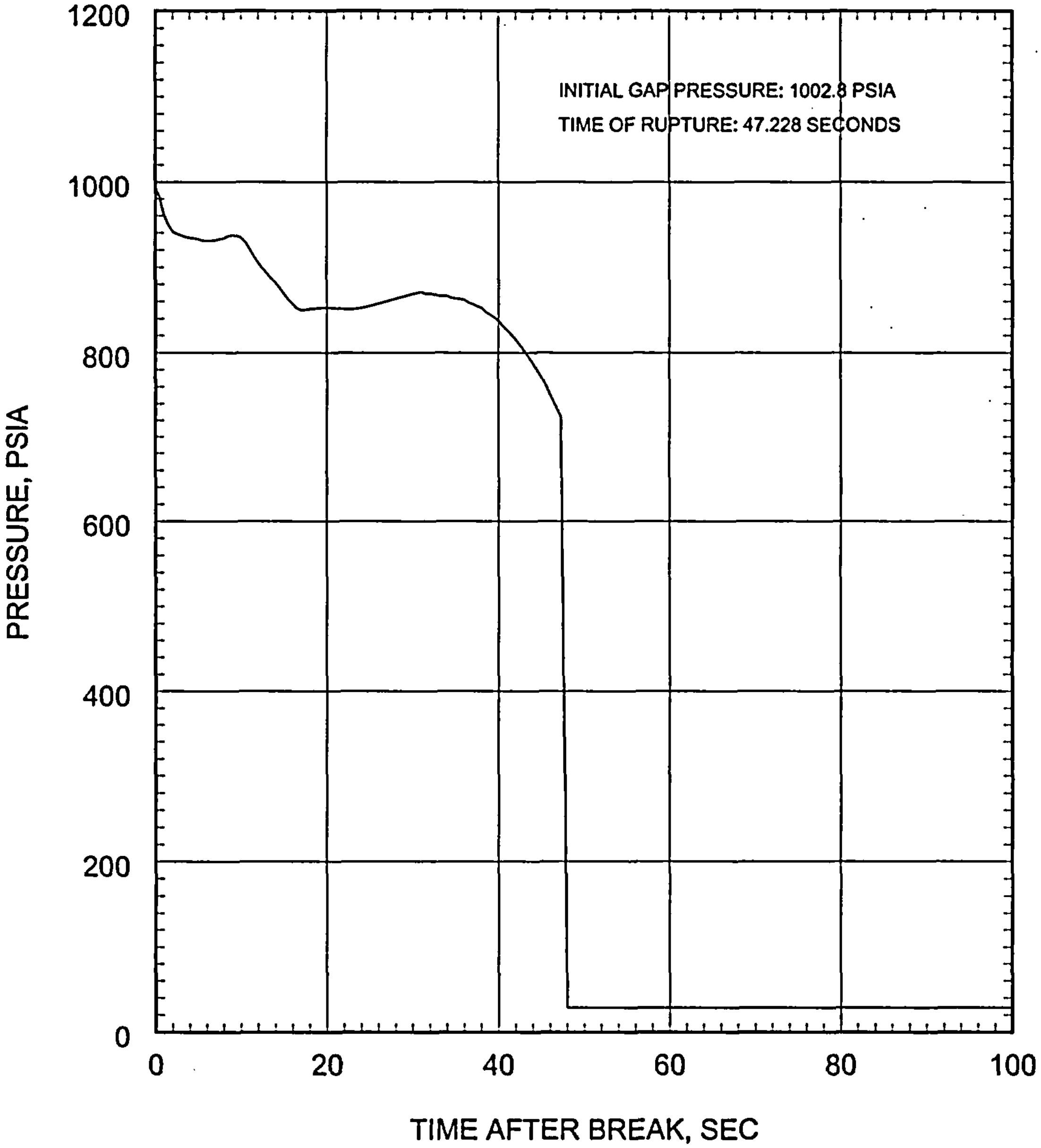

Figure 5.2.3.3-37 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Hot Rod Internal Gas Pressure

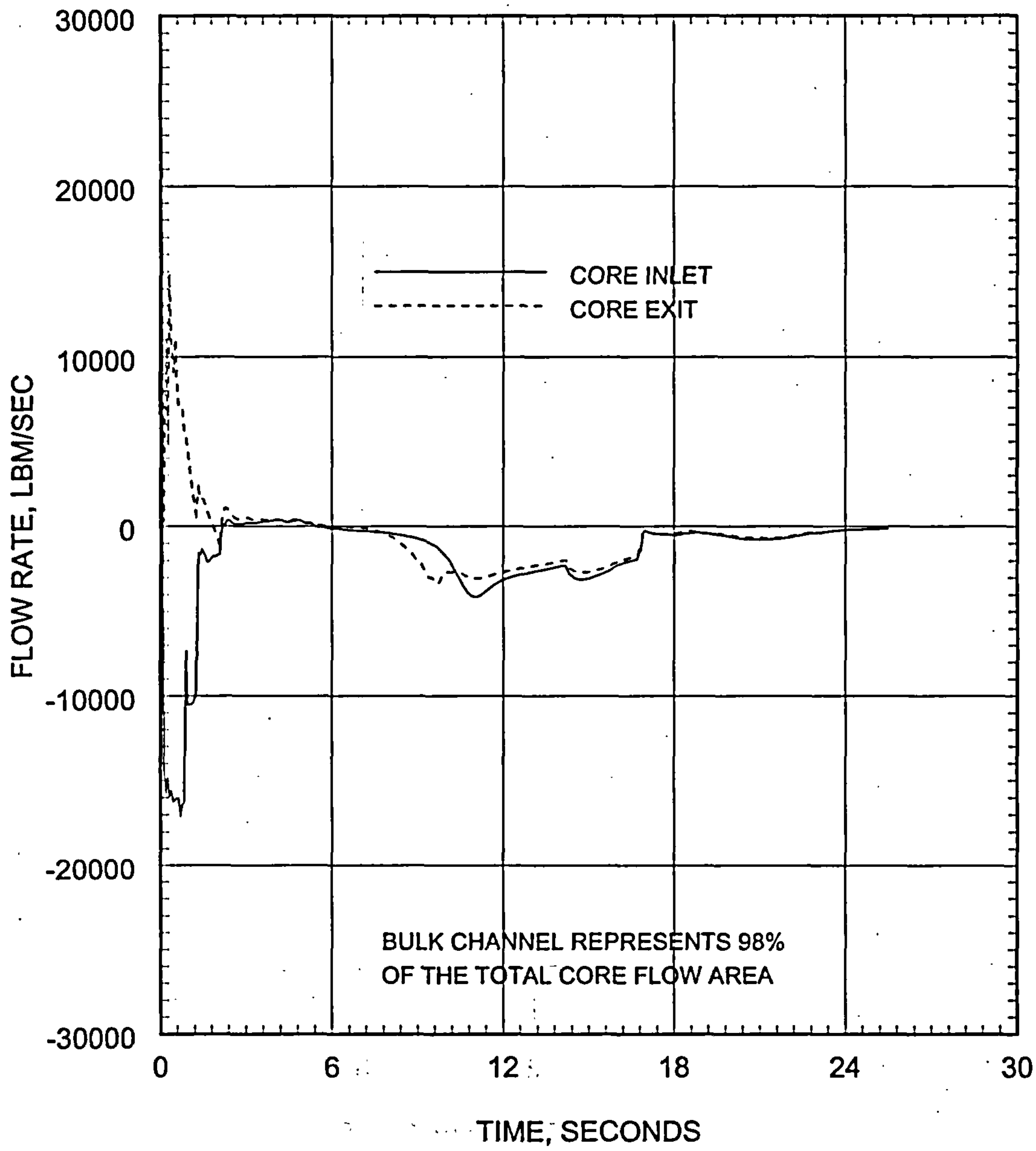

Figure 5.2.3.3-38 Large Break LOCA ECCS Performance Analysis 0.6 DEG/PD Break Core Bulk Channel Flow Rate

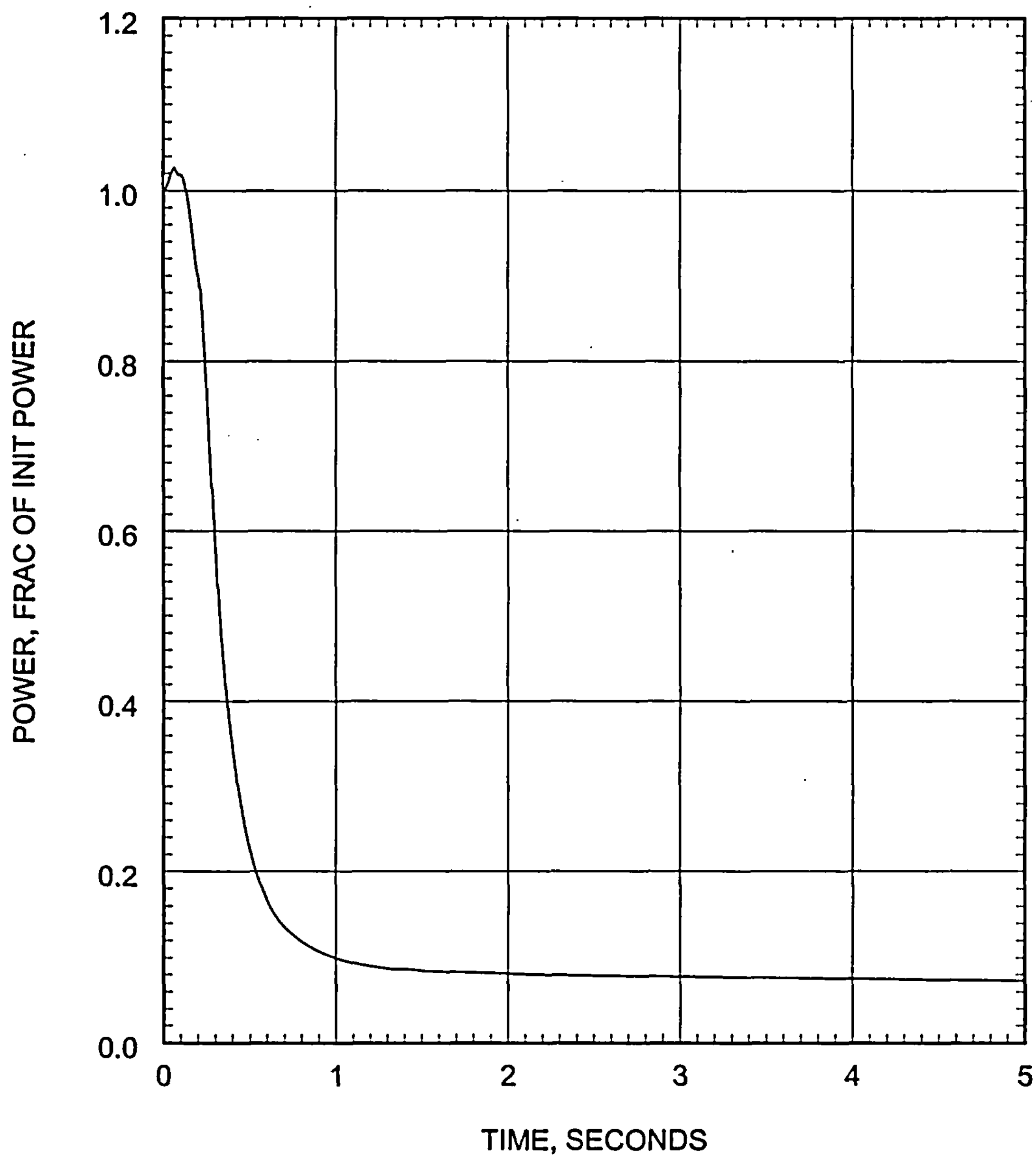

Figure 5.2.3.3-39 Large Break LOCA ECCS Performance Analysis 0.4 DEG/PD Break Core Power

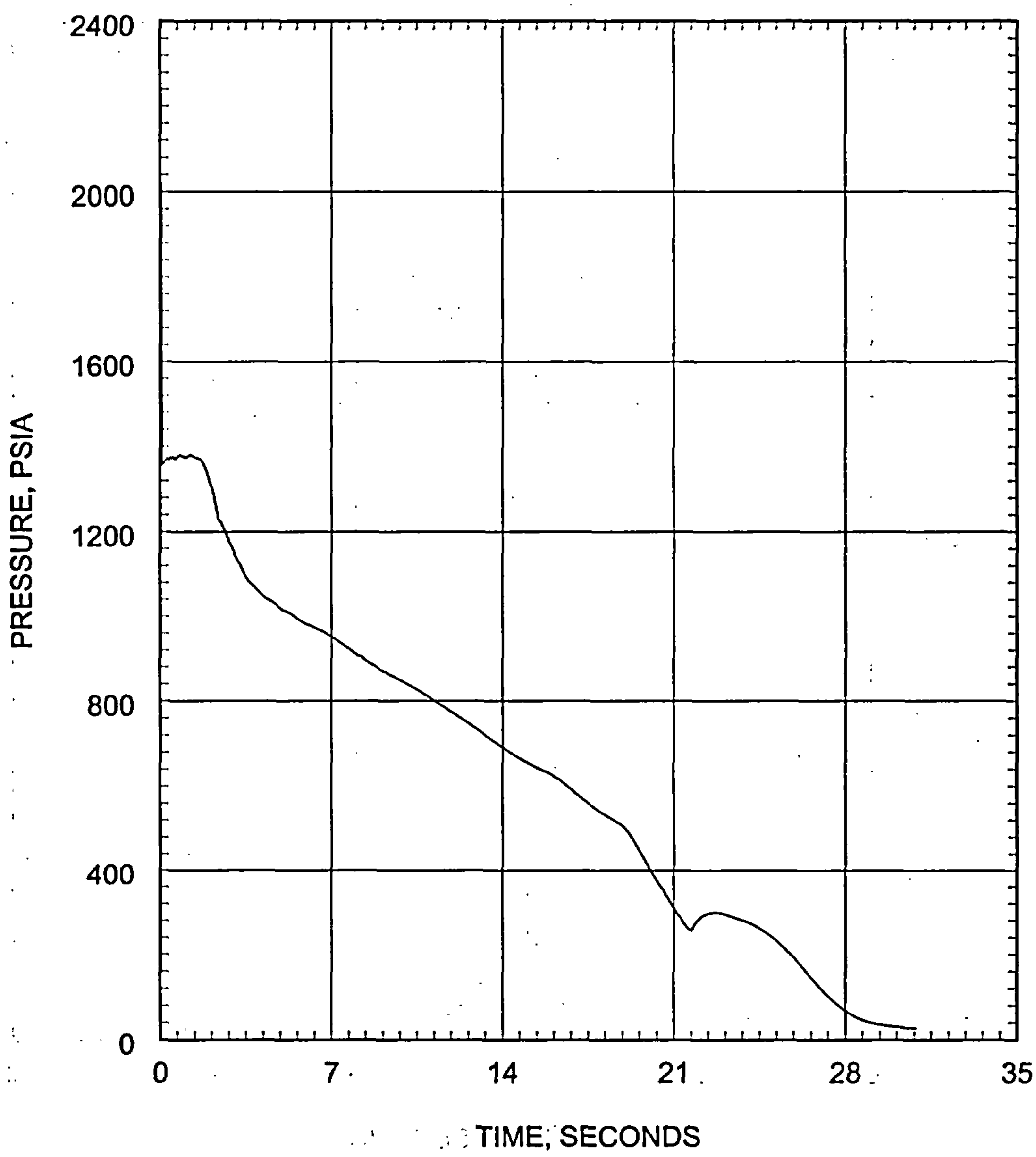

Figure 5.2.3.3-40 Large Break LOCA ECCS Performance Analysis 0.4 DEG/PD Break Pressure in Center Hot Assembly Node

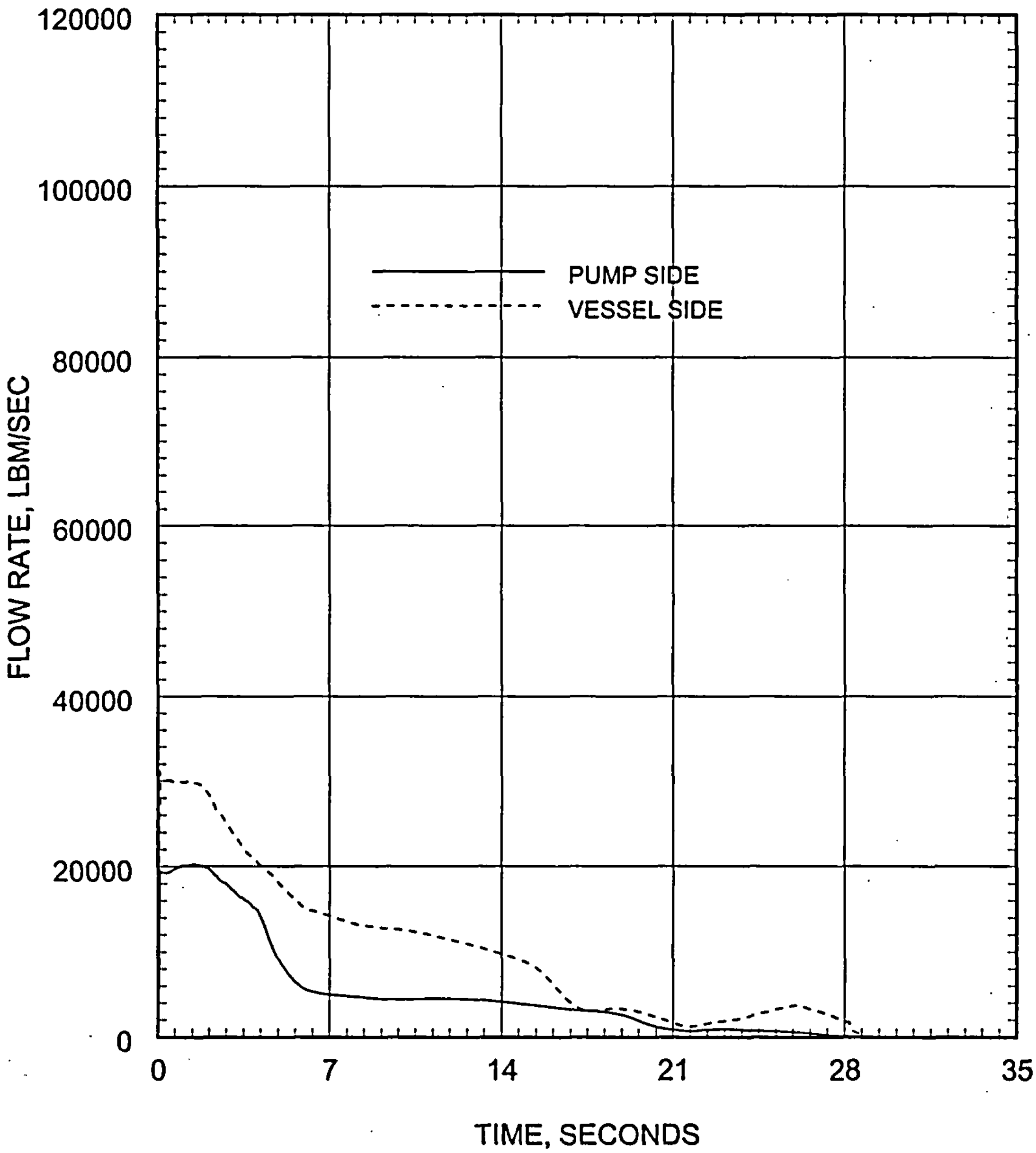

Figure 5.2.3.3-41 Large Break LOCA ECCS Performance Analysis 0.4 DEG/PD Break Leak Flow Rate

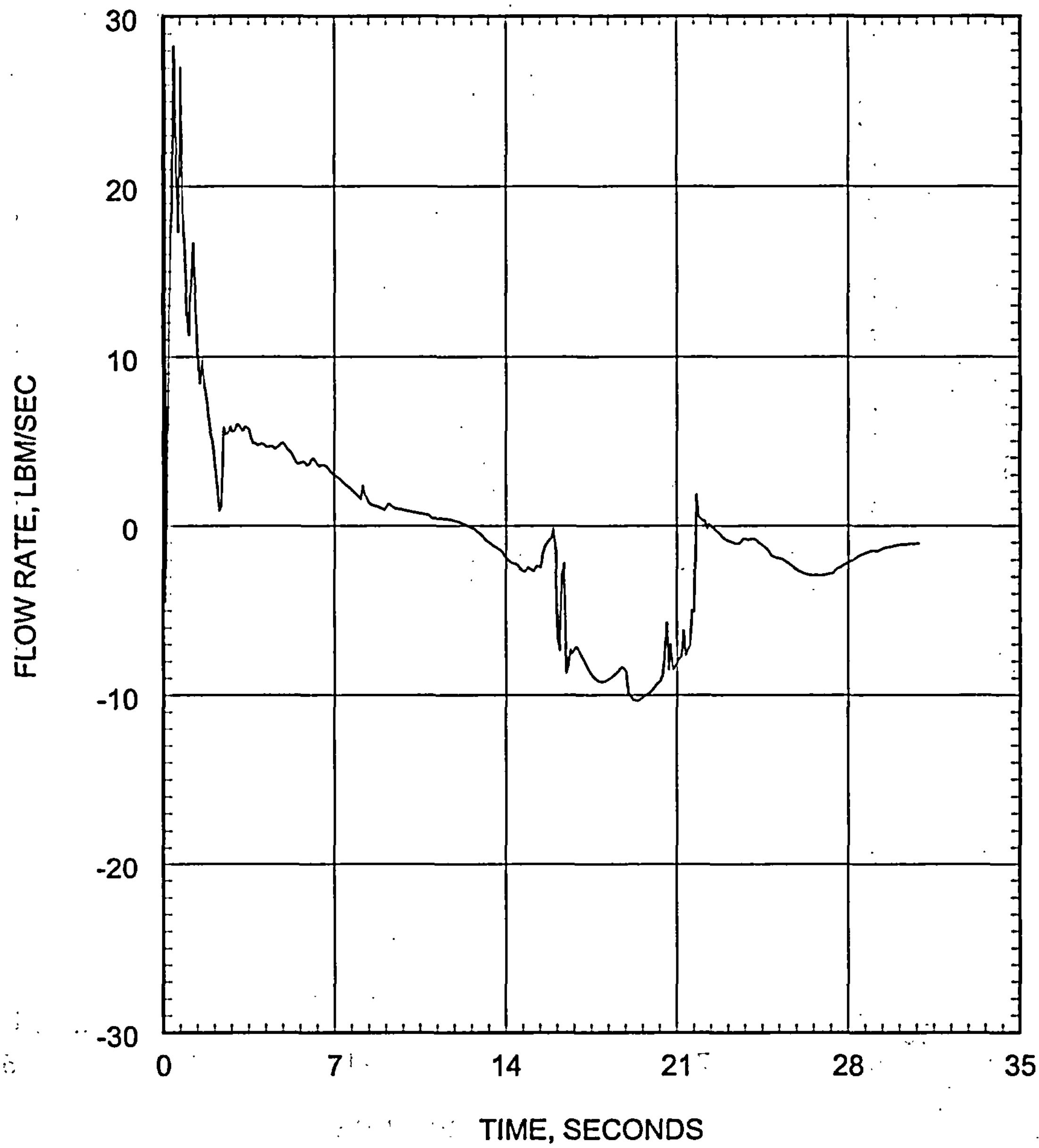

Figure 5.2.3.3-42 Large Break LOCA ECCS Performance Analysis 0.4 DEG/PD Break Hot Assembly Flow Rate (Below Hot Spot)

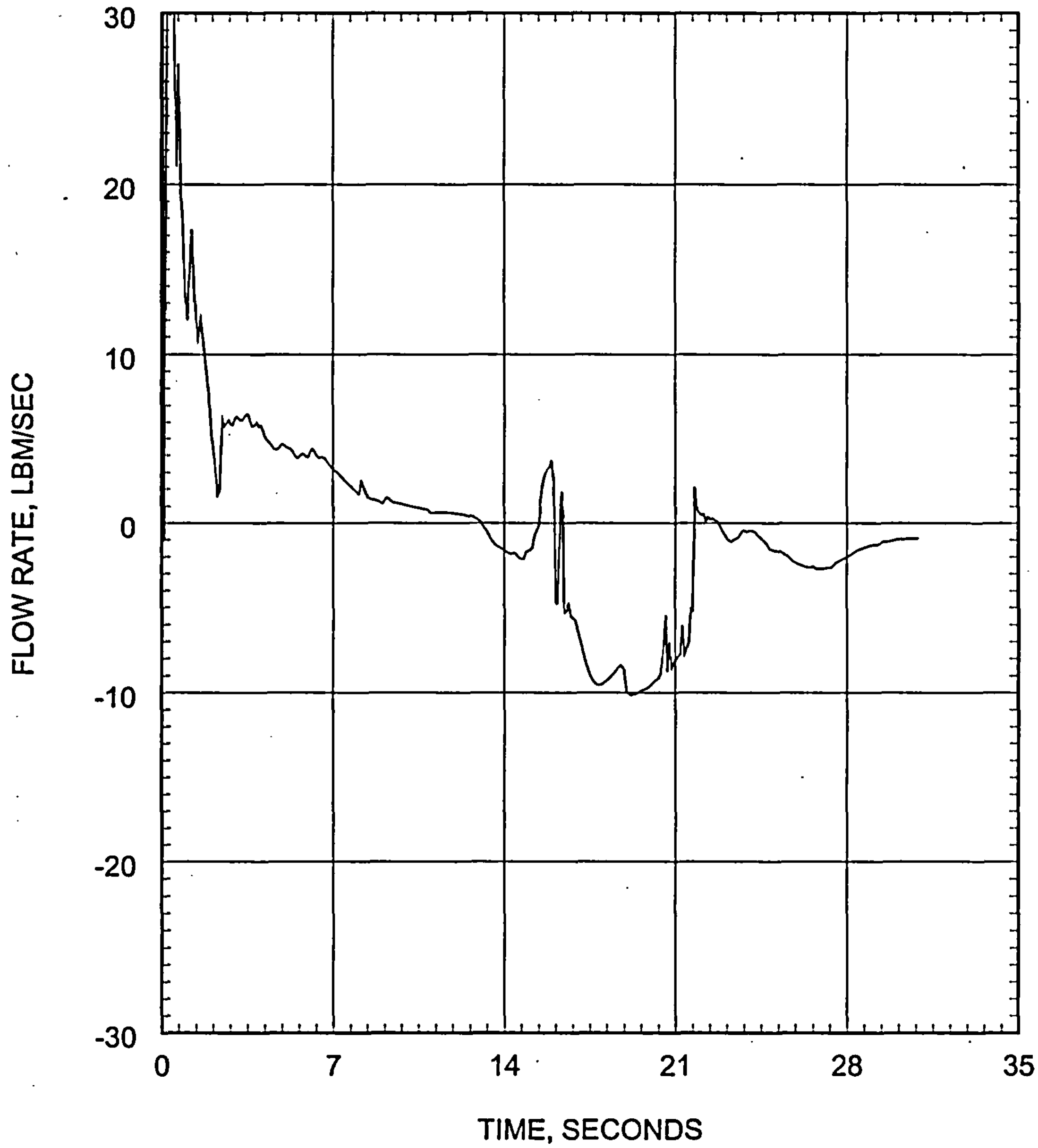

Figure 5.2.3.3-43 Large Break LOCA ECCS Performance Analysis 0.4 DEG/PD Break Hot Assembly Flow Rate (Above Hot Spot)

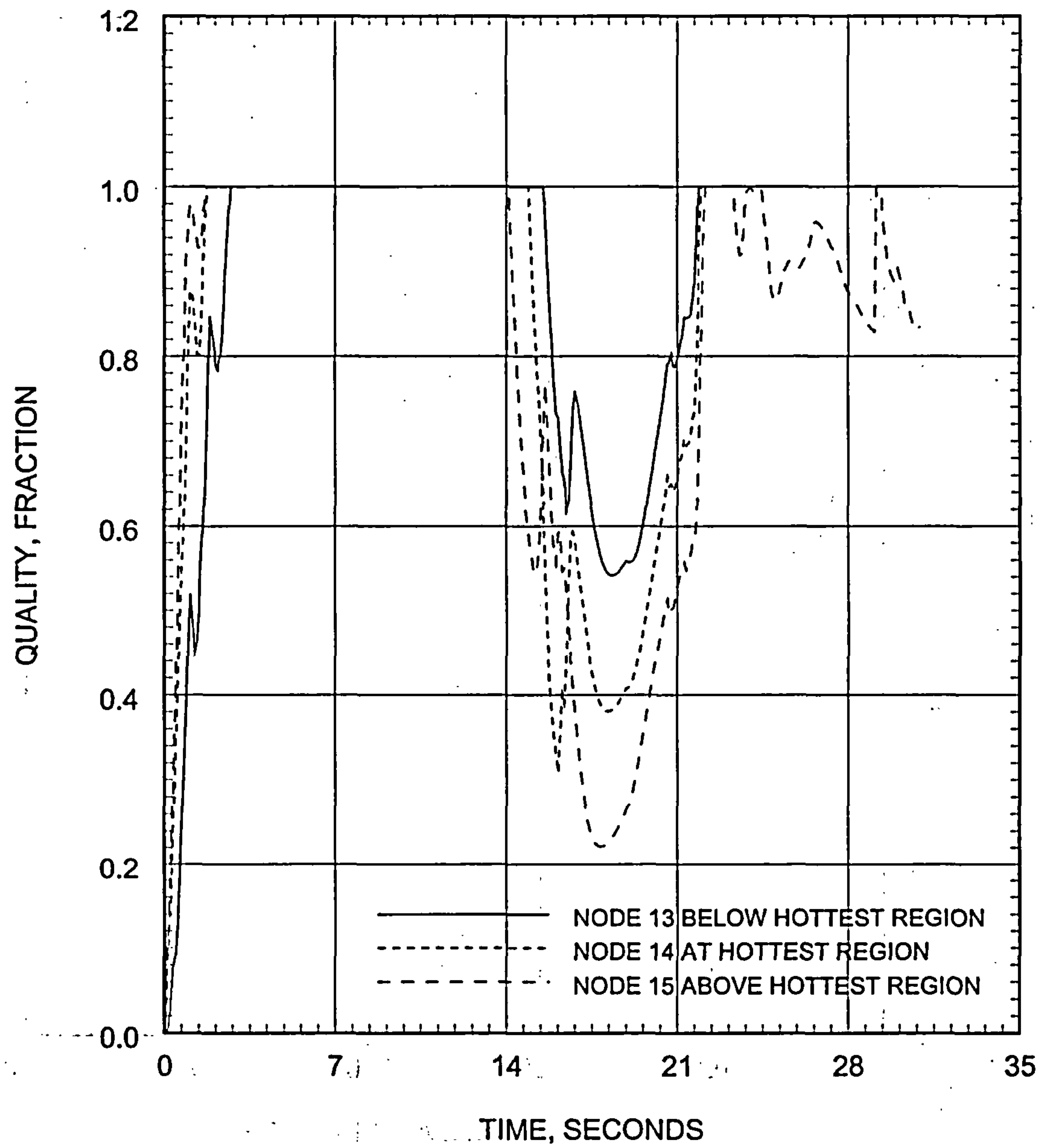

Figure 5.2.3.3-44 Large Break LOCA ECCS Performance Analysis 0.4 DEG/PD Break Hot Assembly Quality

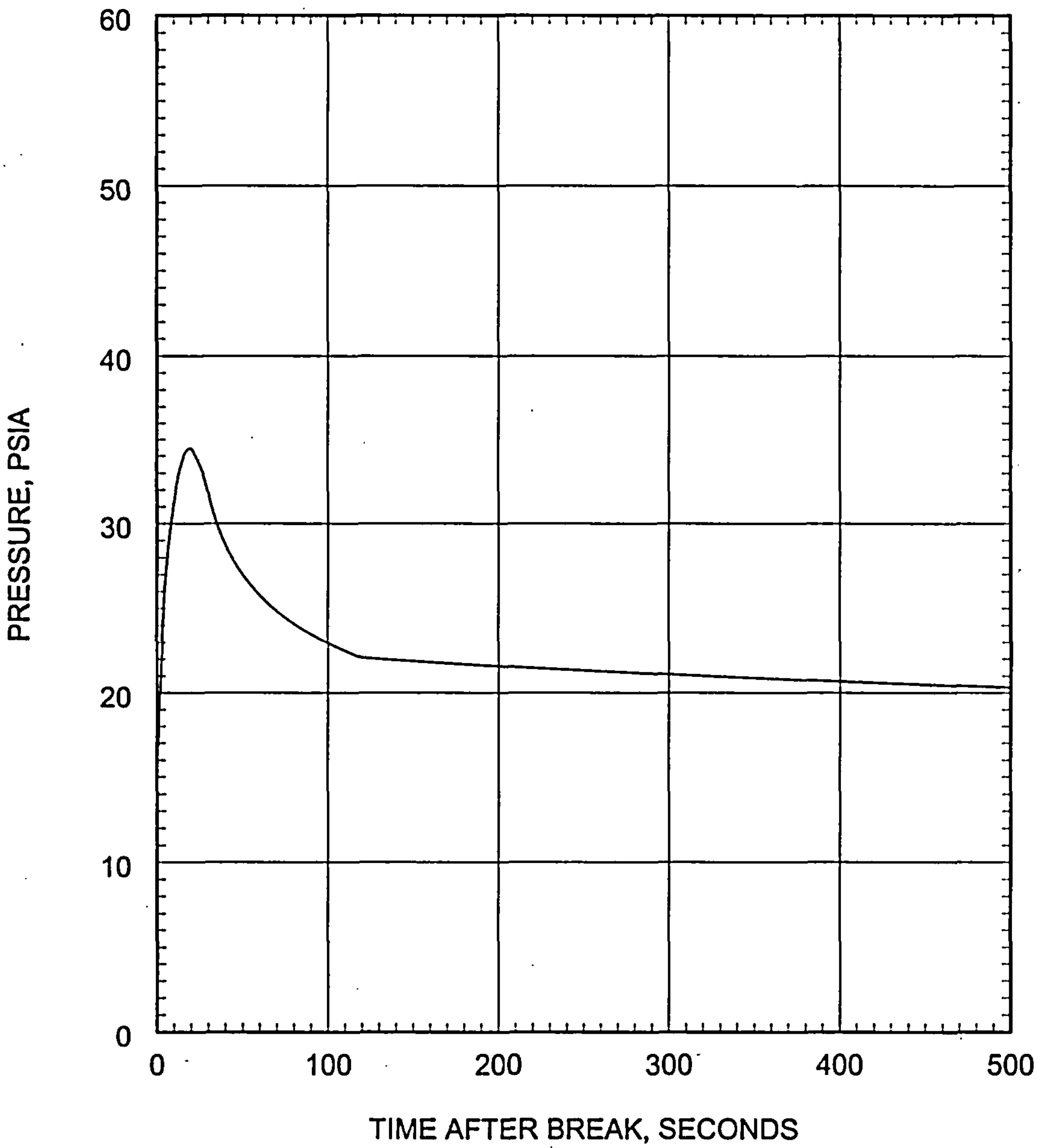

Figure 5.2.3.3-45 Large Break LOCA ECCS Performance Analysis 0.4 DEG/PD Break Containment Pressure

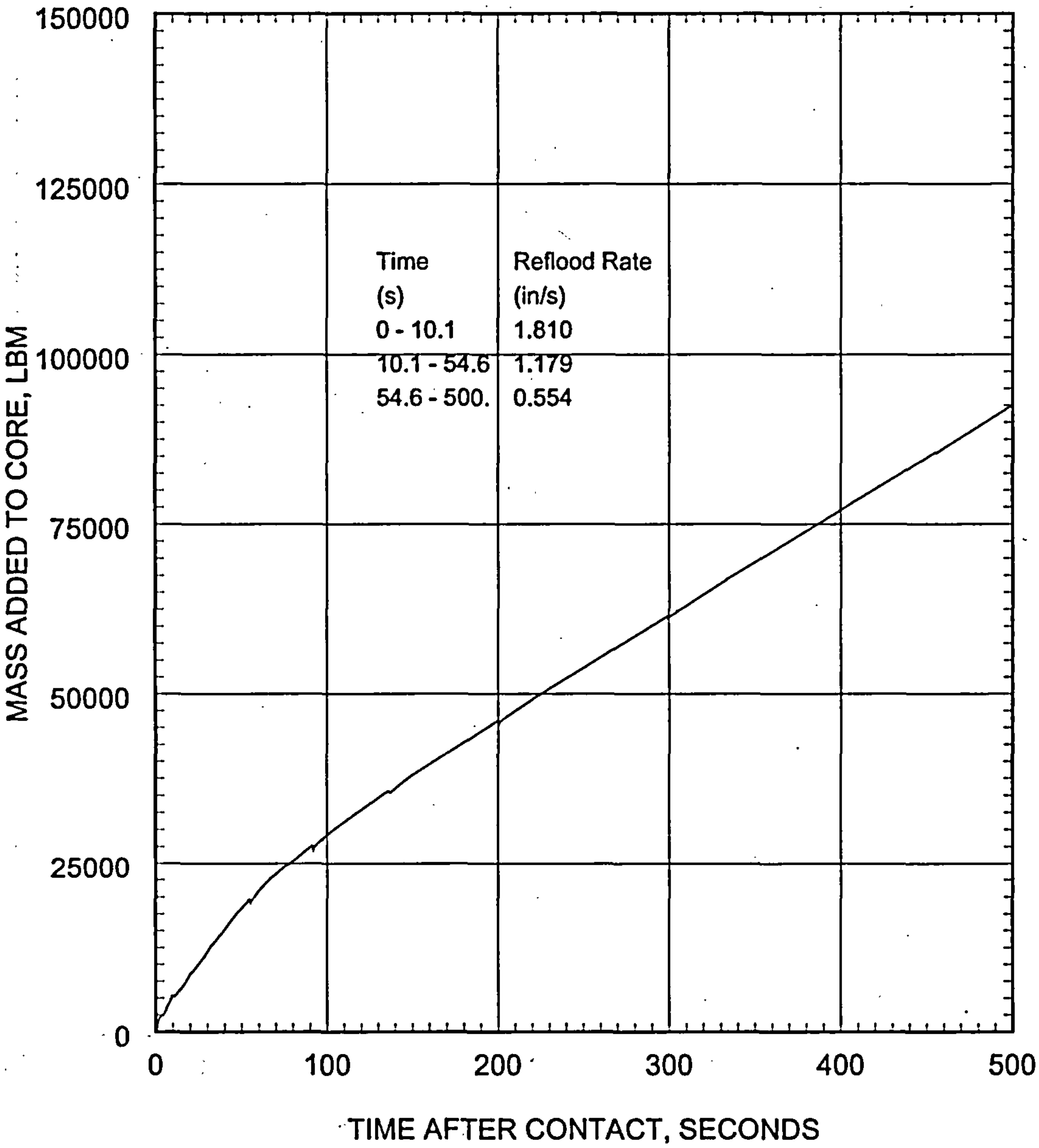

Figure 5.2.3.3-46 Large Break LOCA ECCS Performance Analysis 0.4 DEG/PD Break Mass Added to Core During Reflood

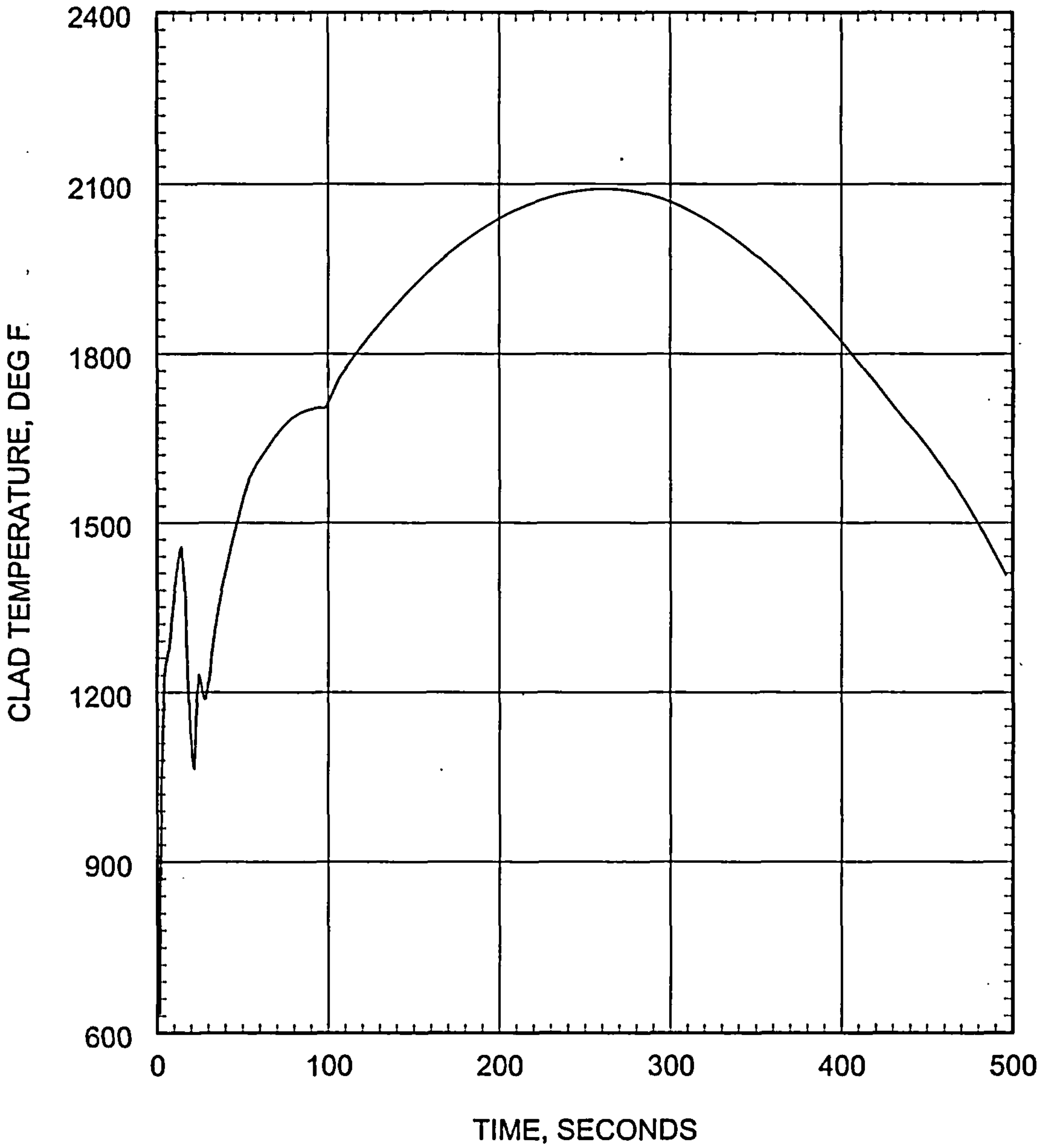

Figure 5.2.3.3-47 Large Break LOCA ECCS Performance Analysis 0.4 DEG/PD Break Peak Cladding Temperature

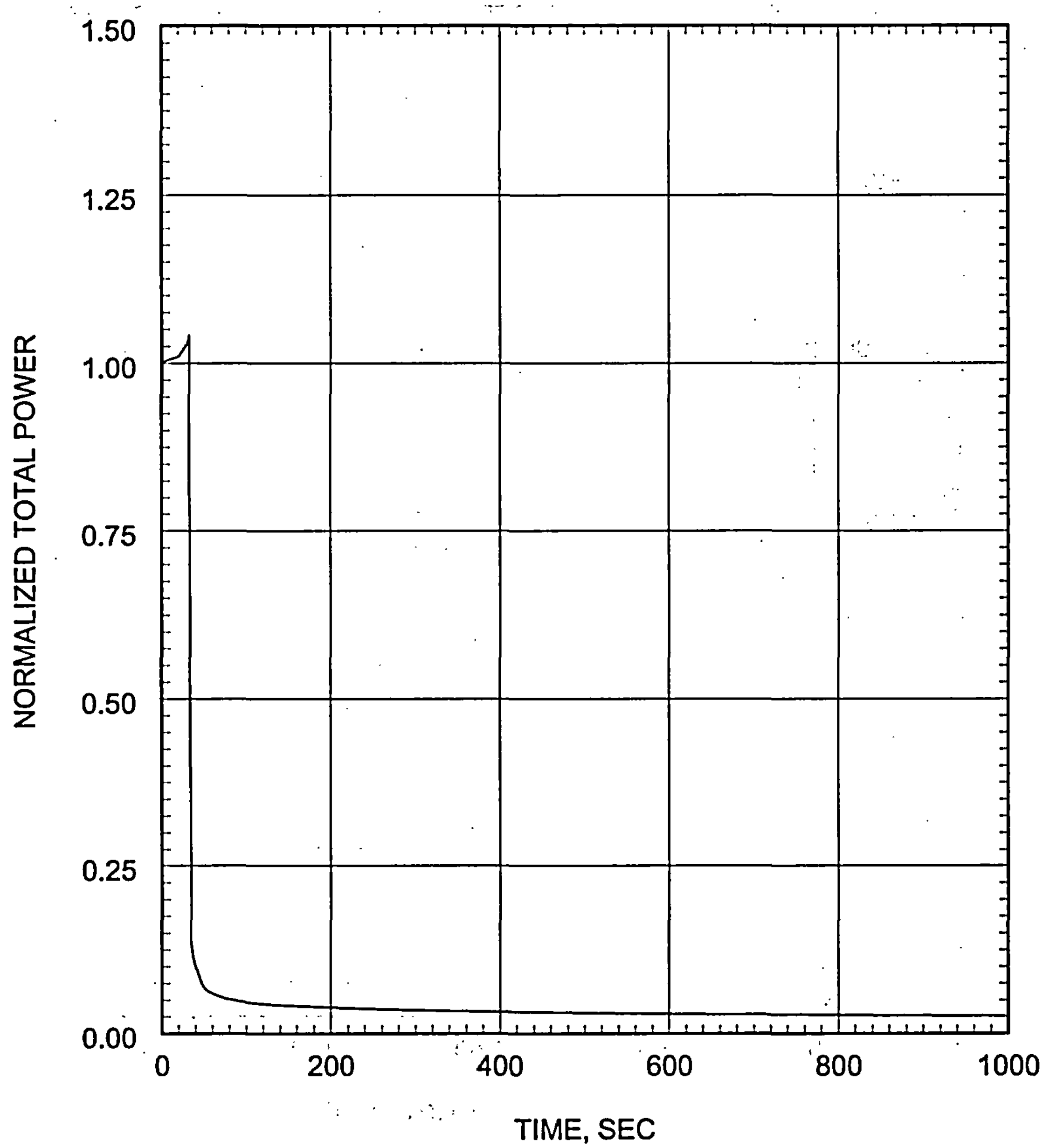

Figure 5.2.4.3-1 Small Break LOCA ECCS Performance Analysis 0.04 ft^2/PD Break Core Power

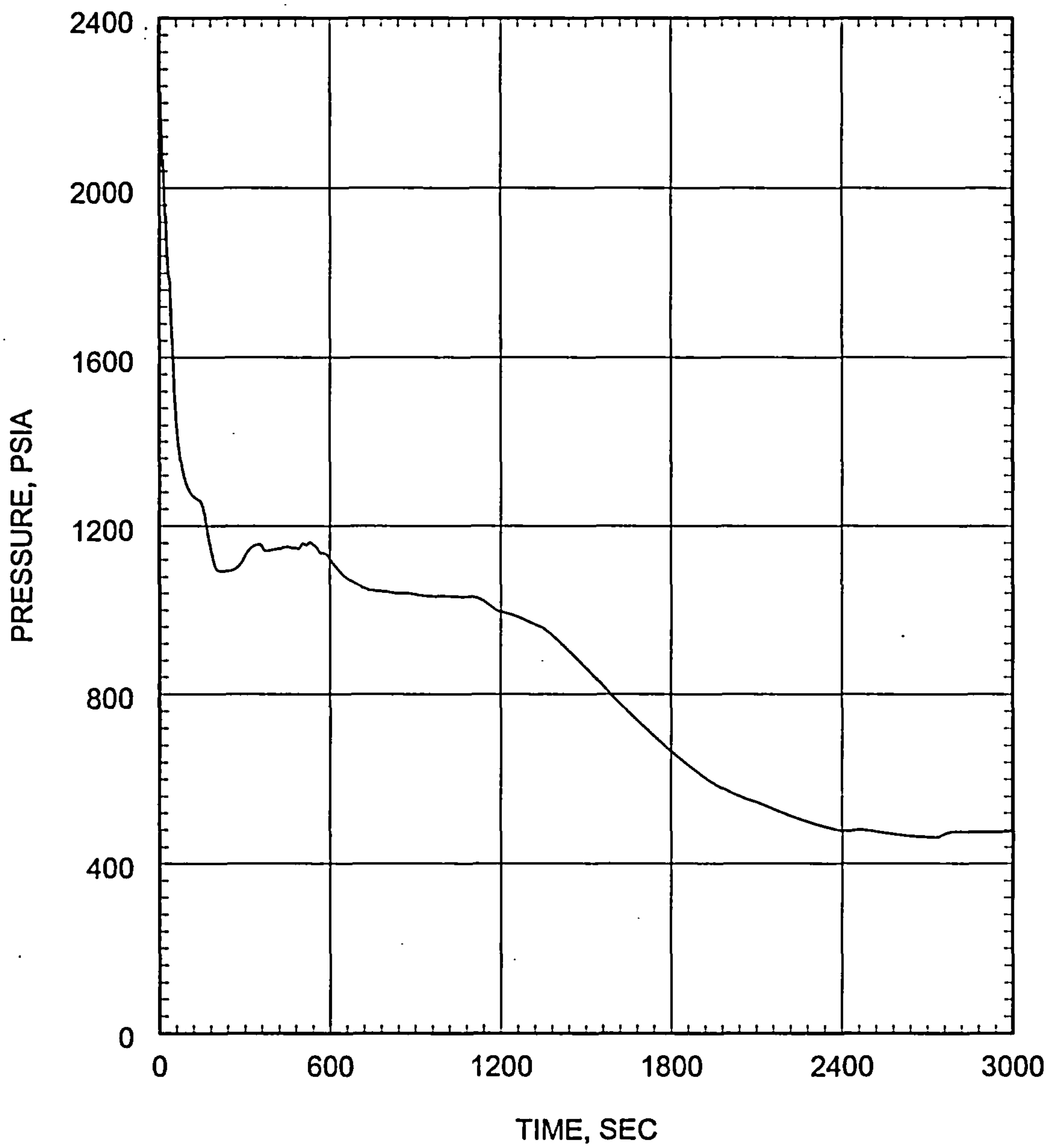

Figure 5.2.4.3-2 Small Break LOCA ECCS Performance Analysis 0.04 ft²/PD Break Inner Vessel Pressure

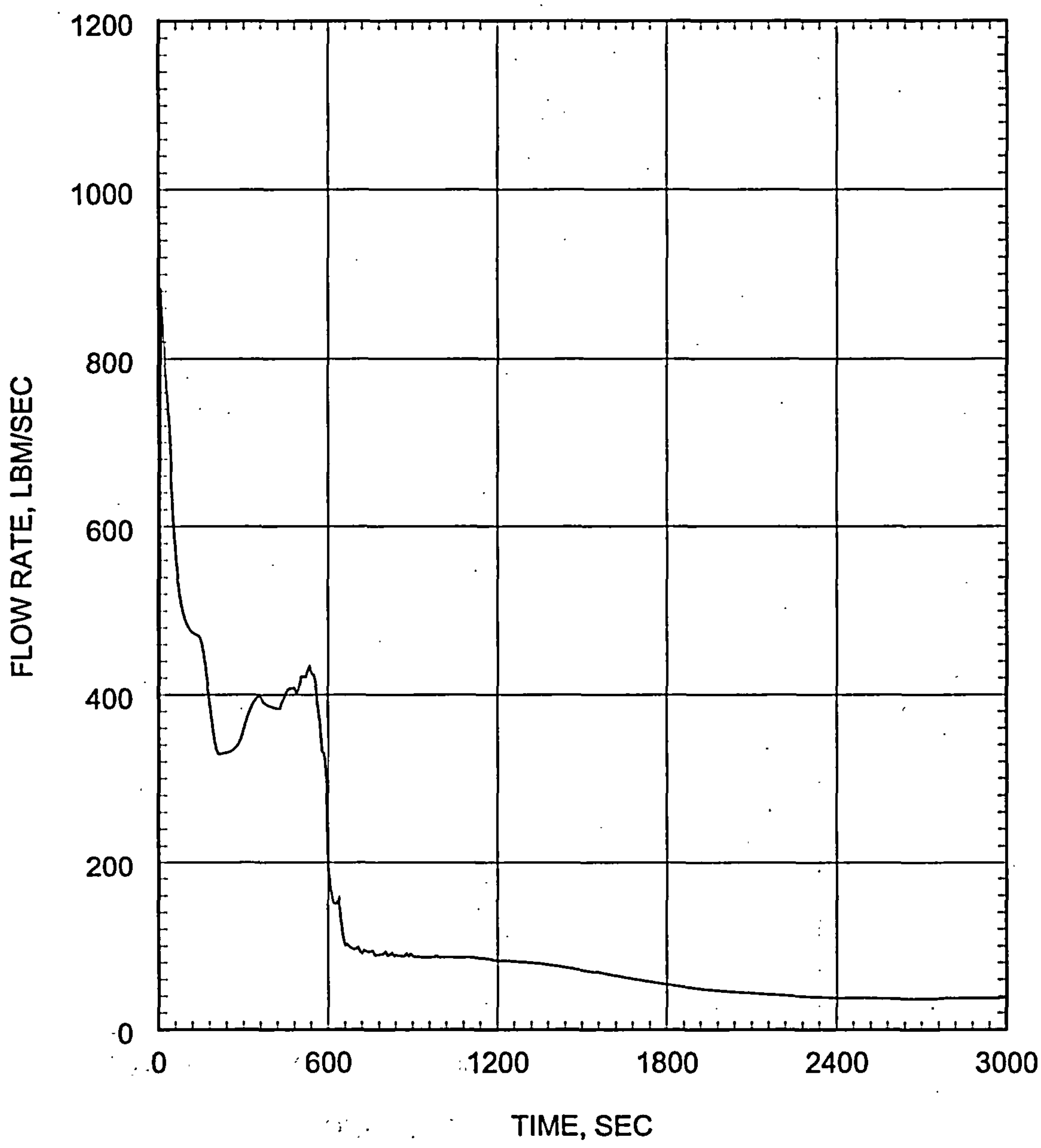

Figure 5.2.4.3-3 Small Break LOCA ECCS Performance Analysis 0.04 ft²/PD Break Break Flow Rate

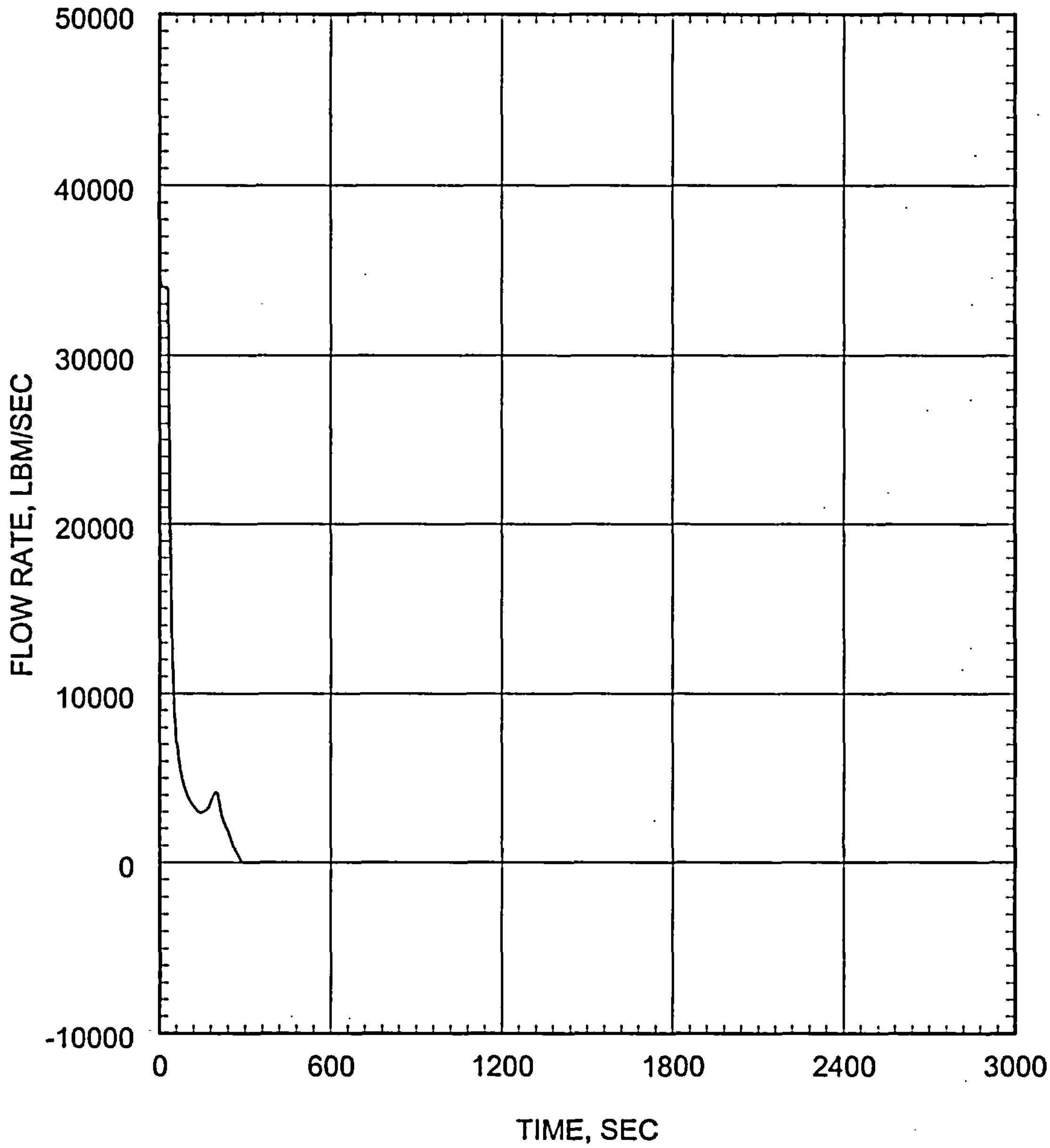

Figure 5.2.4.3-4 Small Break LOCA ECCS Performance Analysis 0.04 ft^2/PD Break Inner Vessel Inlet Flow Rate

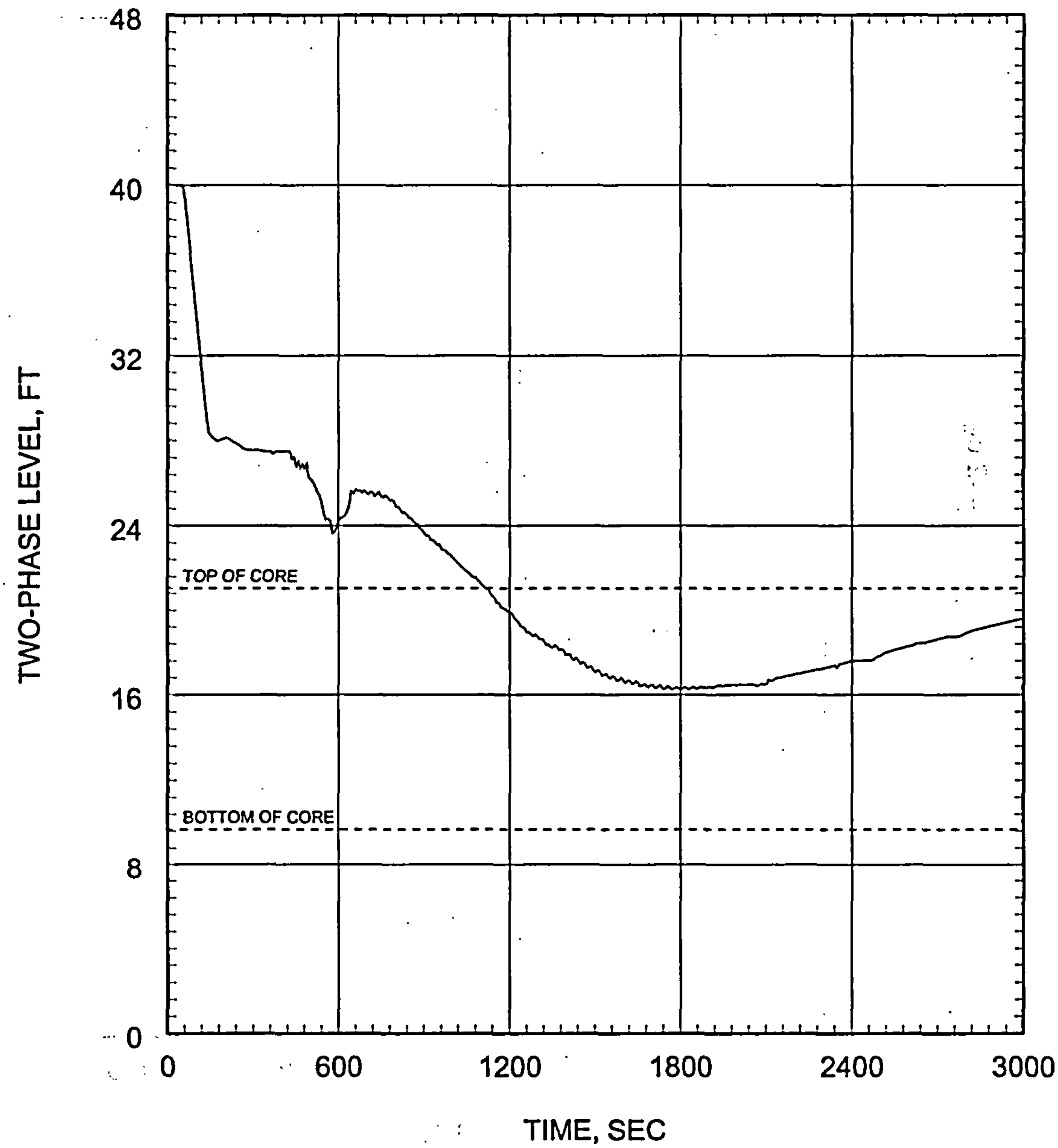

Figure 5.2.4.3-5 Small Break LOCA ECCS Performance Analysis 0.04 ft^2/PD Break Inner Vessel Two-Phase Mixture Level

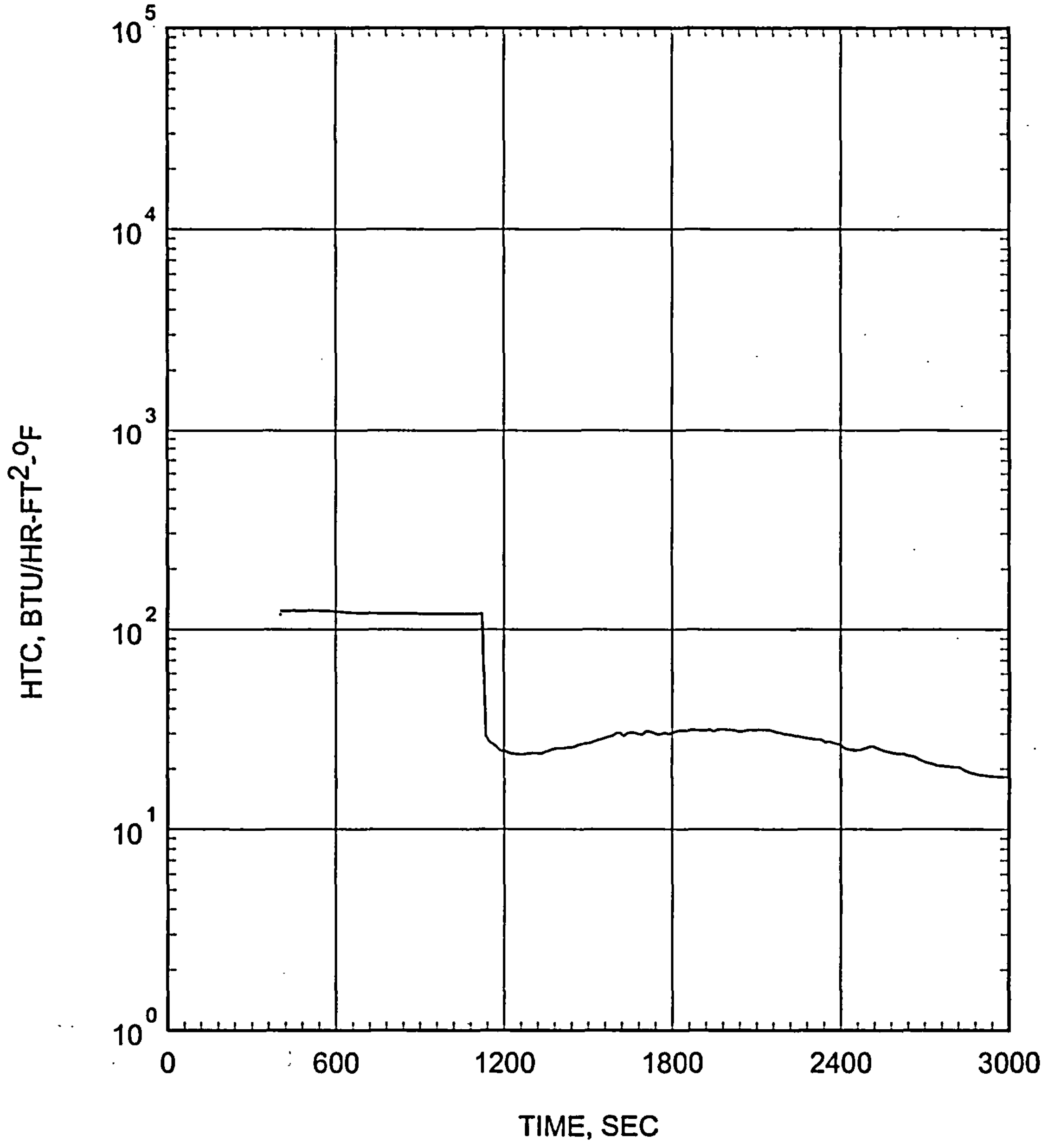

Figure 5.2.4.3-6 Small Break LOCA ECCS Performance Analysis 0.04 ft²/PD Break Heat Transfer Coefficient at Hot Spot

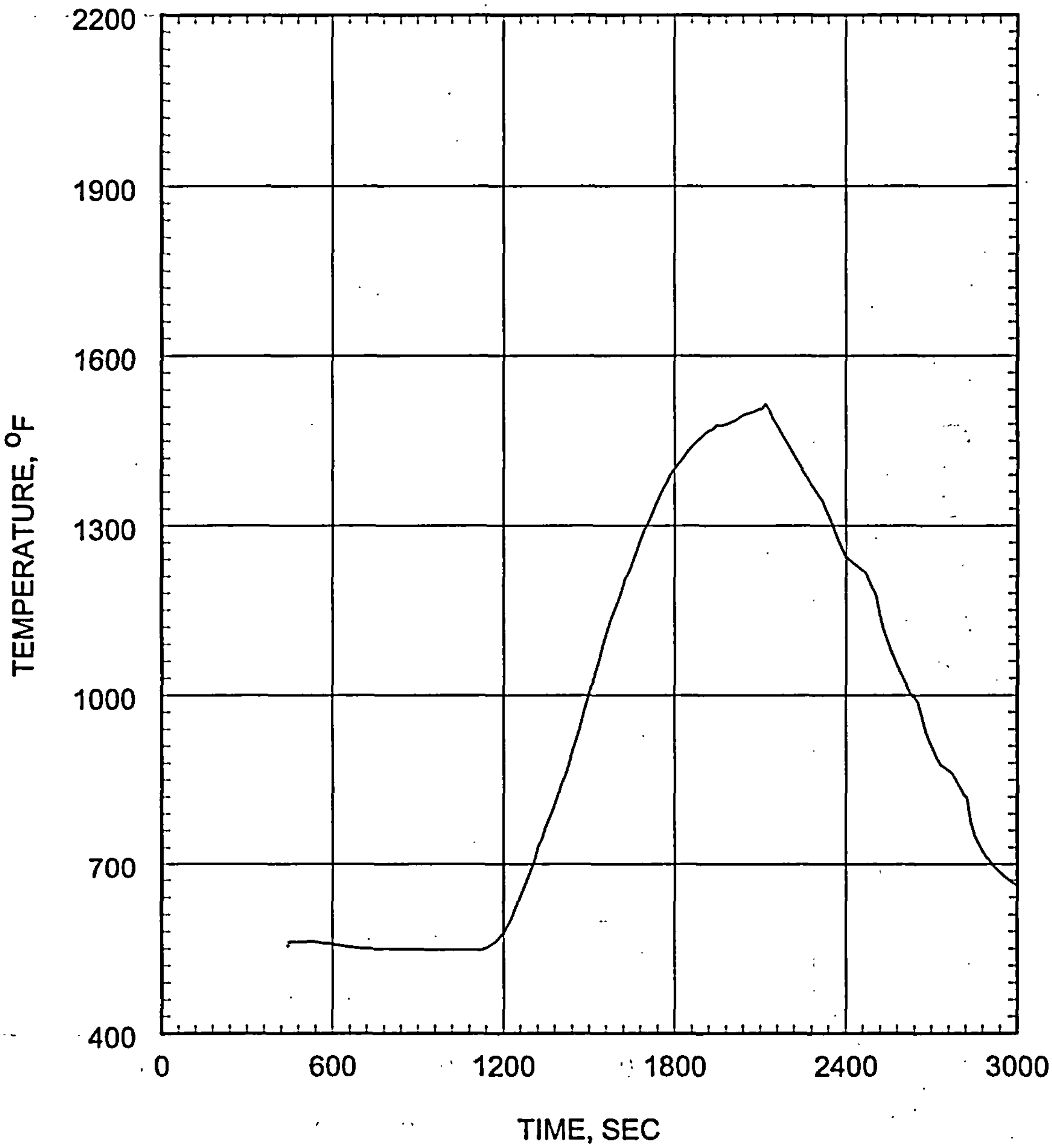

Figure 5.2.4.3-7 Small Break LOCA ECCS Performance Analysis 0.04 ft^2/PD Break Coolant Temperature at Hot Spot

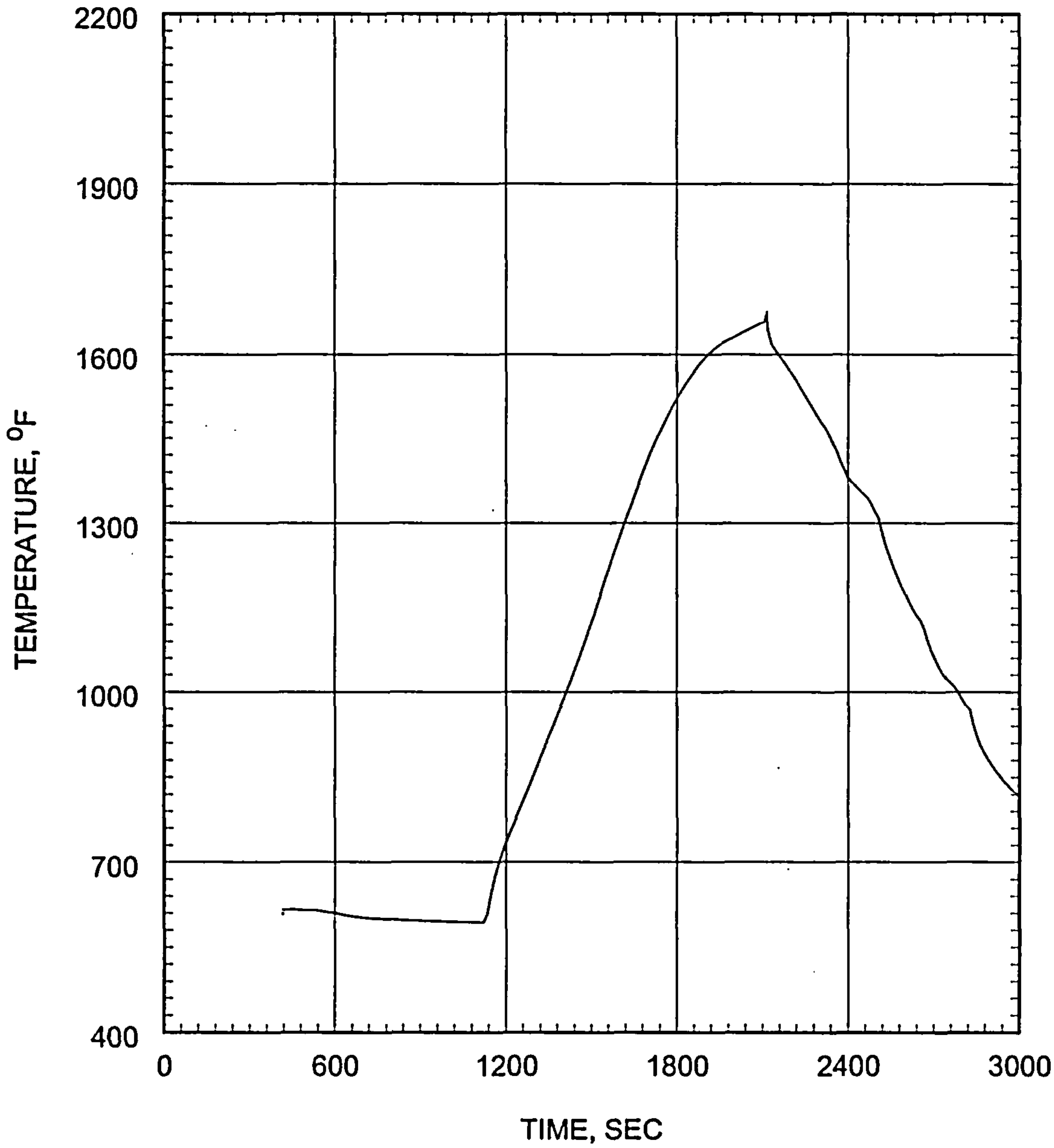

Figure 5.2.4.3-8 Small Break LOCA ECCS Performance Analysis 0.04 ft^2/PD Break Cladding Temperature at Hot Spot

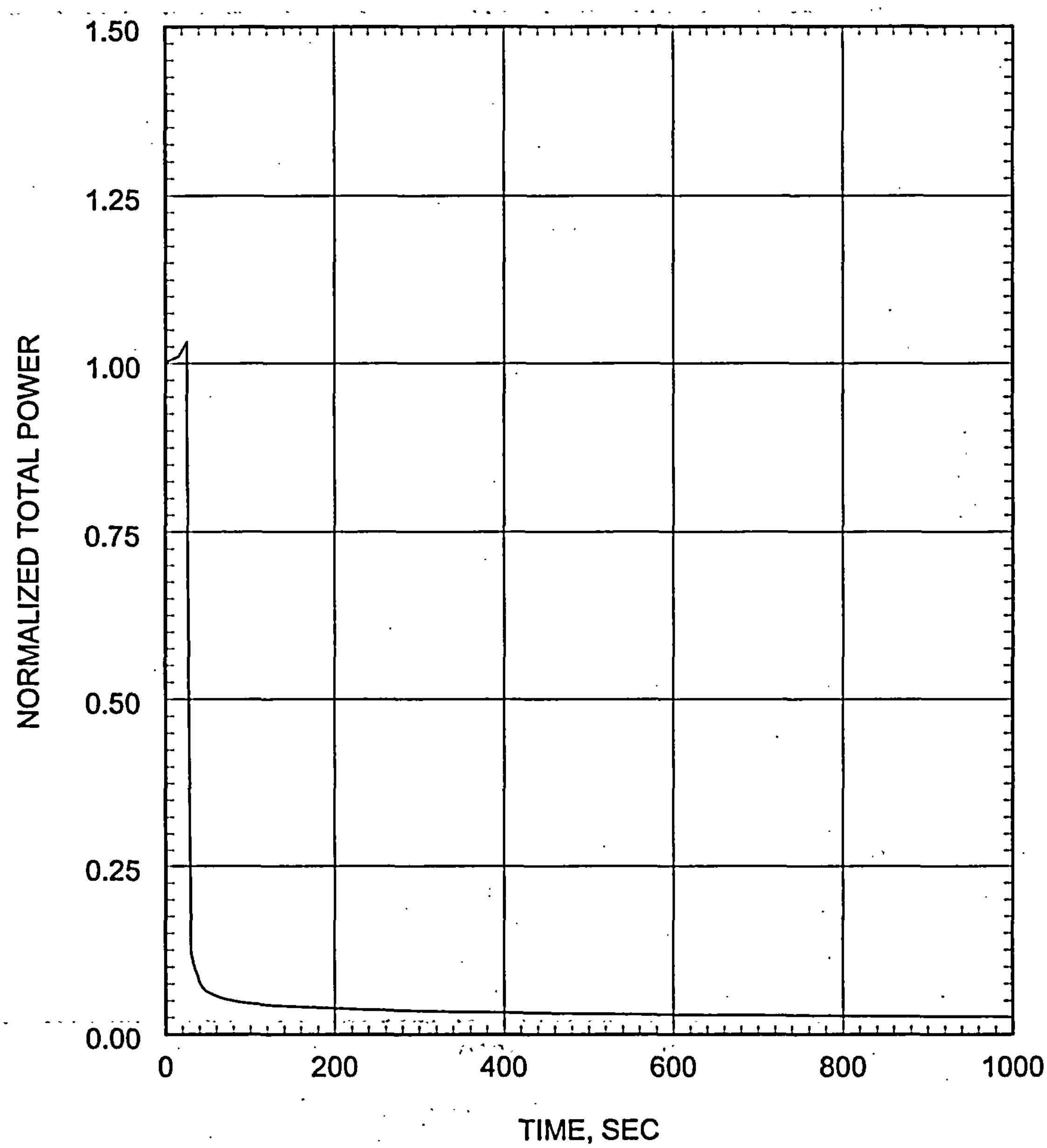

Figure 5.2.4.3-9 Small Break LOCA ECCS Performance Analysis 0.05 ft²/PD Break Core Power

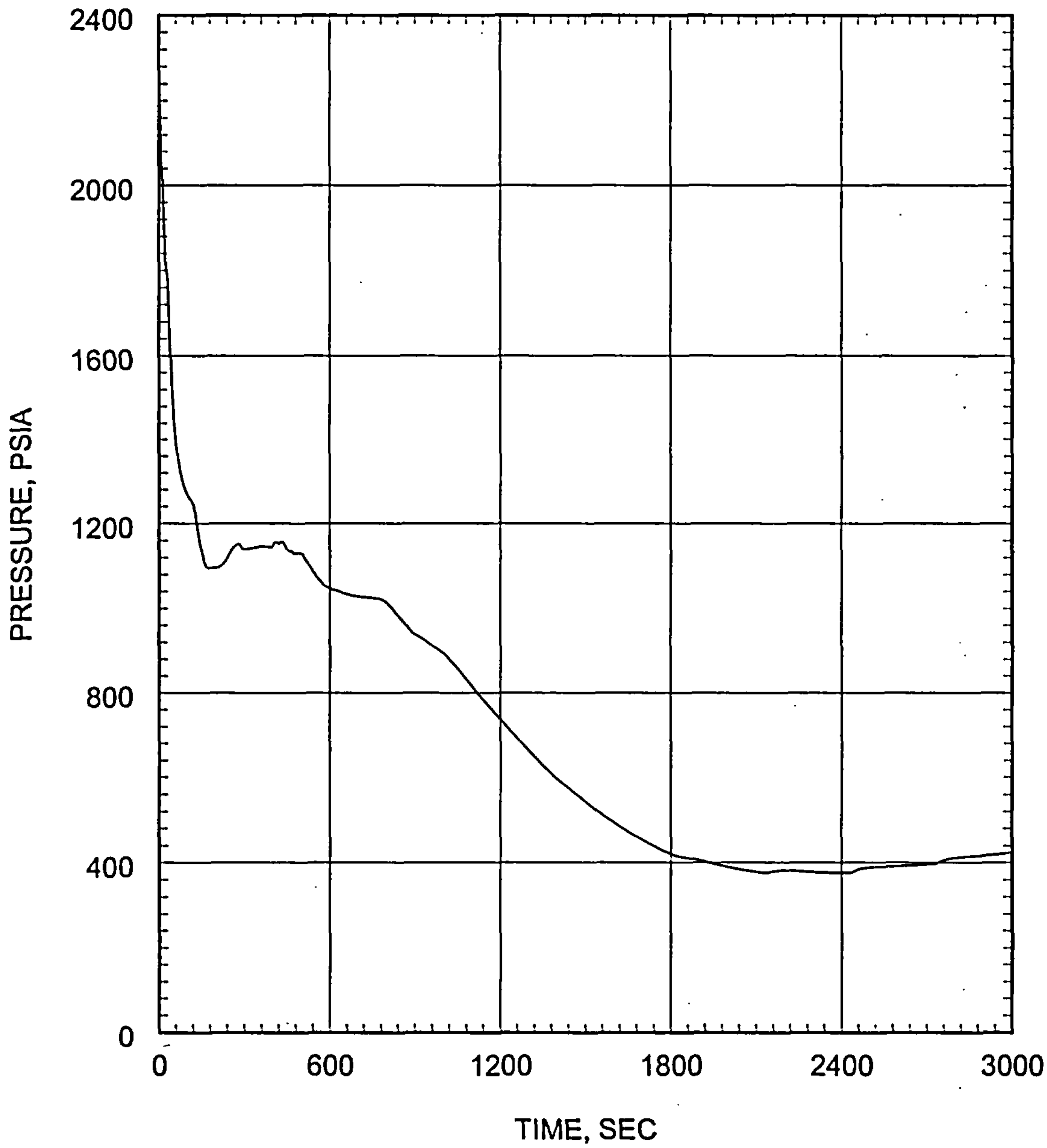

Figure 5.2.4.3-10 Small Break LOCA ECCS Performance Analysis 0.05 ft^2/PD Break Inner Vessel Pressure

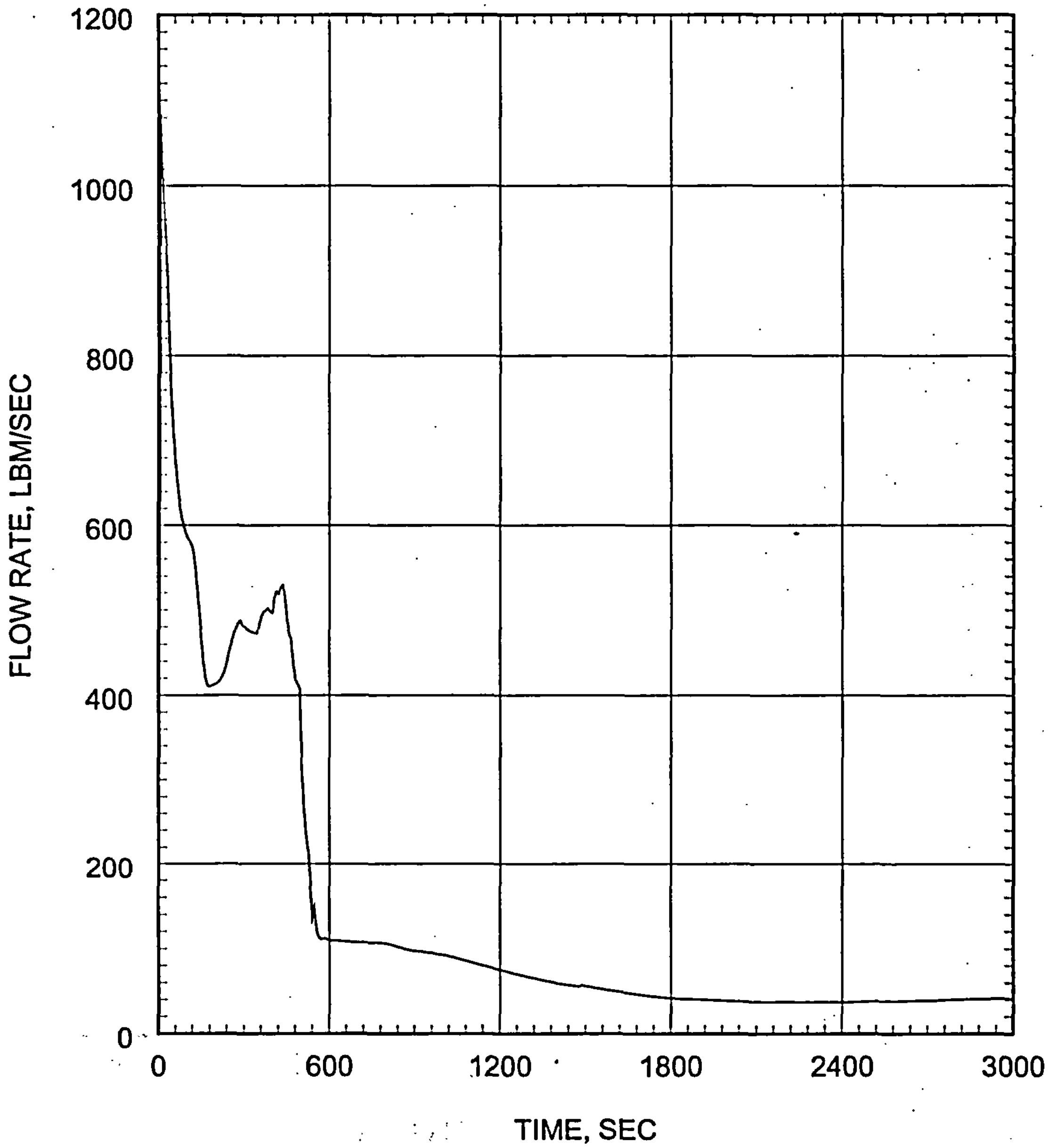

Figure 5.2.4.3-11 Small Break LOCA ECCS Performance Analysis 0.05 ft²/PD Break Break Flow Rate

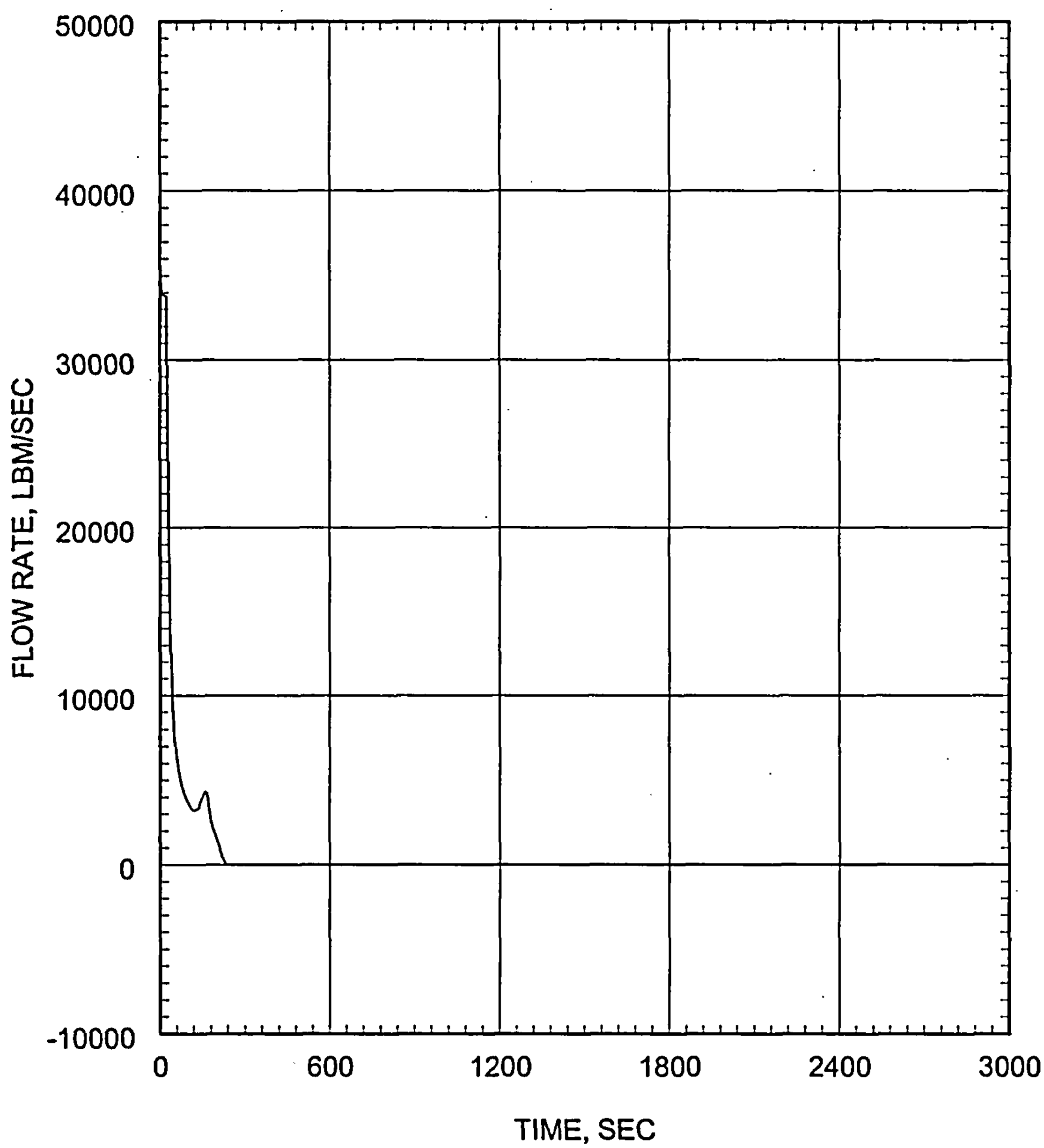

Figure 5.2.4.3-12 Small Break LOCA ECCS Performance Analysis 0.05 ft^2/PD Break Inner Vessel Inlet Flow Rate

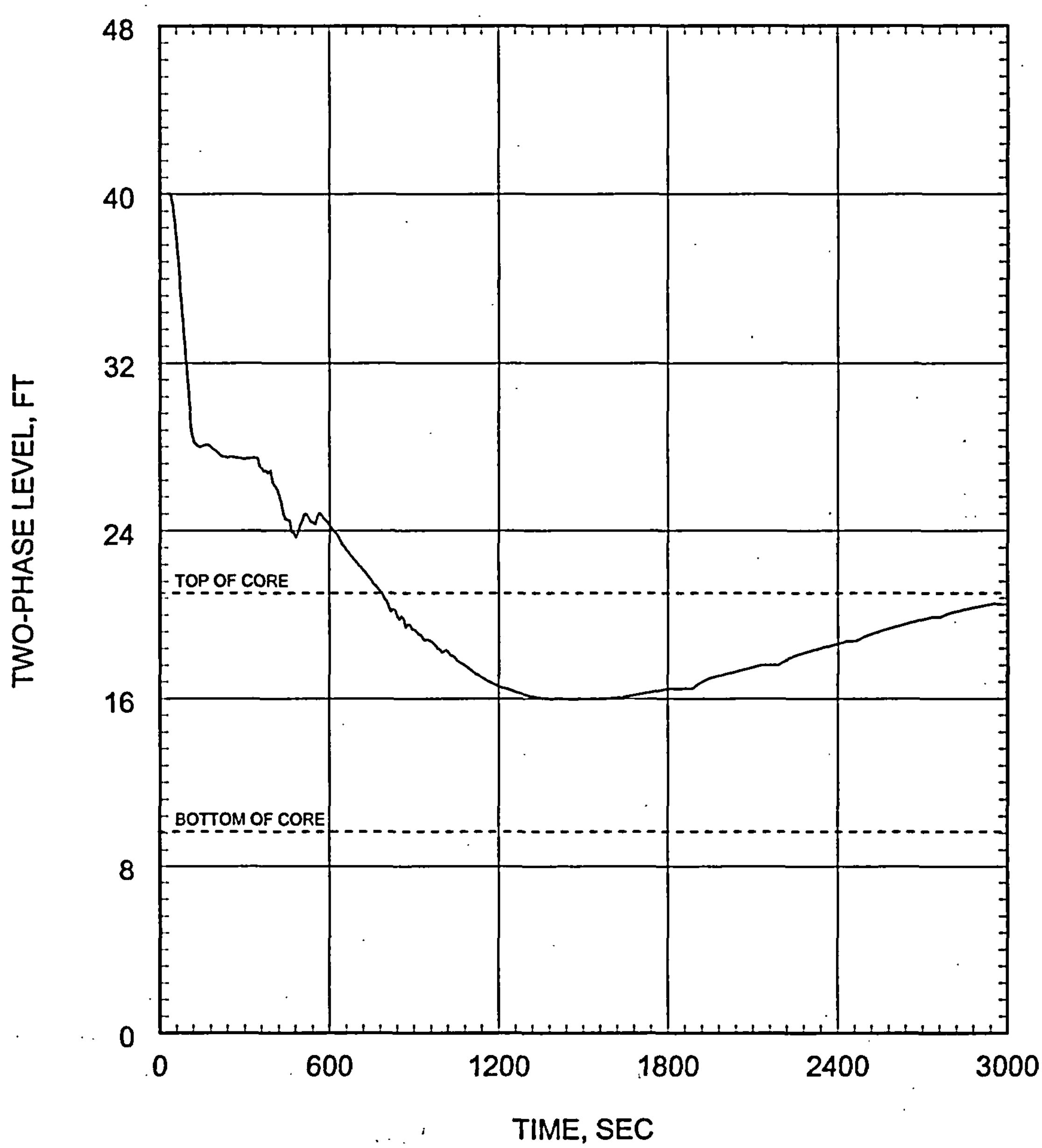

Figure 5.2.4.3-13 Small Break LOCA ECCS Performance Analysis 0.05 ft^2/PD Break Inner Vessel Two-Phase Mixture Level

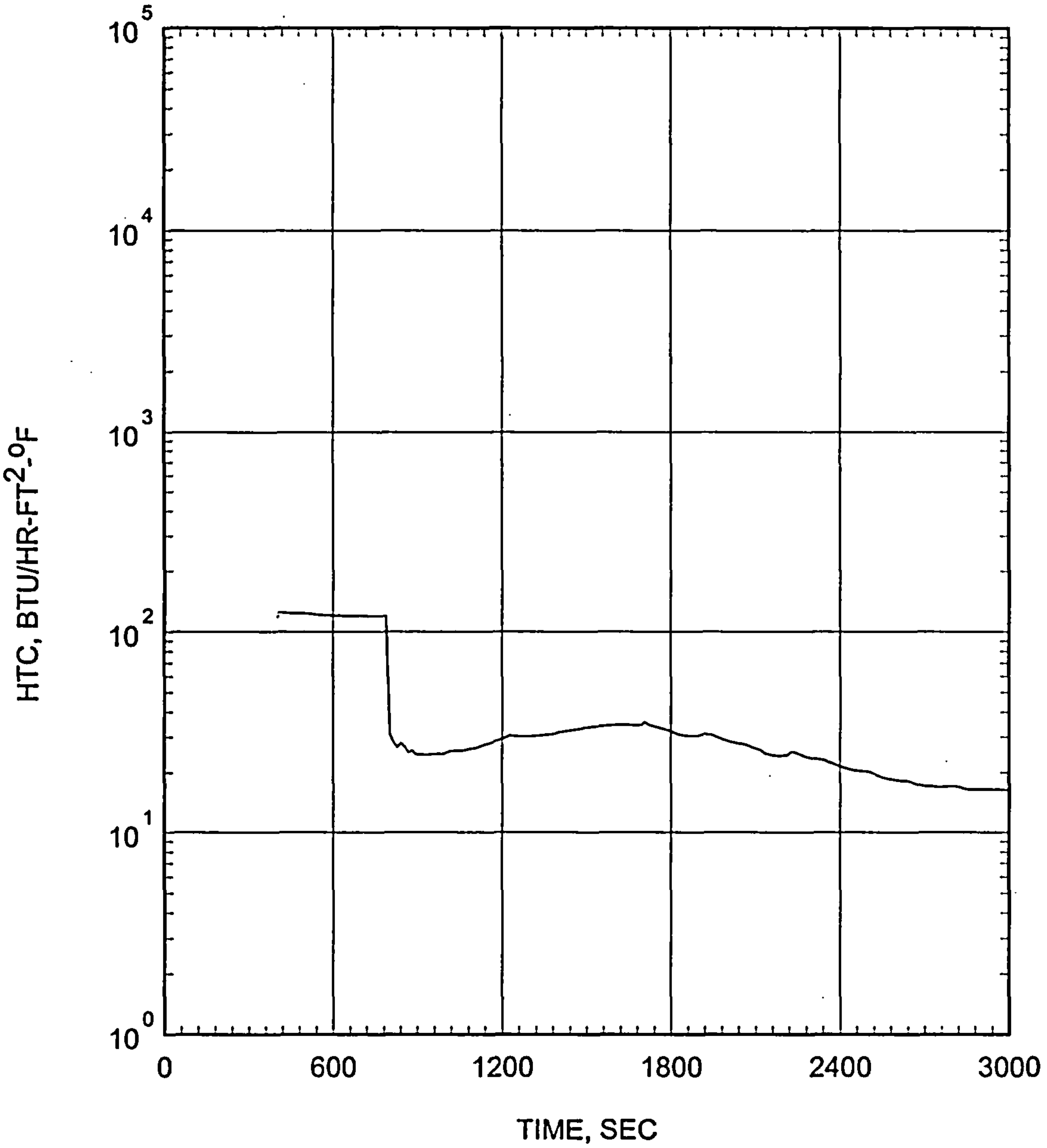

Figure 5.2.4.3-14 Small Break LOCA ECCS Performance Analysis 0.05 ft^2/PD Break Heat Transfer Coefficient at Hot Spot

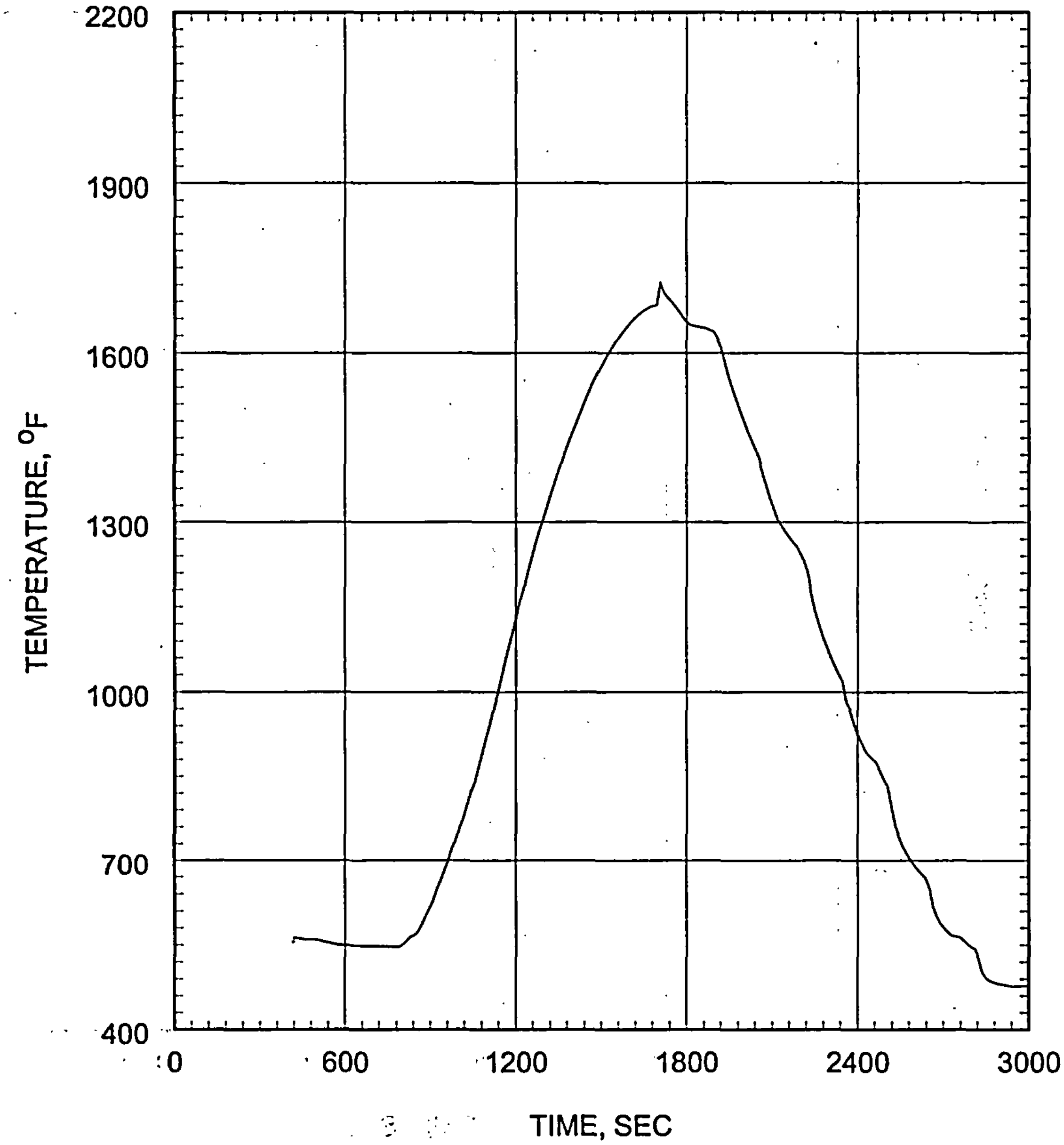

Figure 5.2.4.3-15 Small Break LOCA ECCS Performance Analysis 0.05 ft²/PD Break Coolant Temperature at Hot Spot

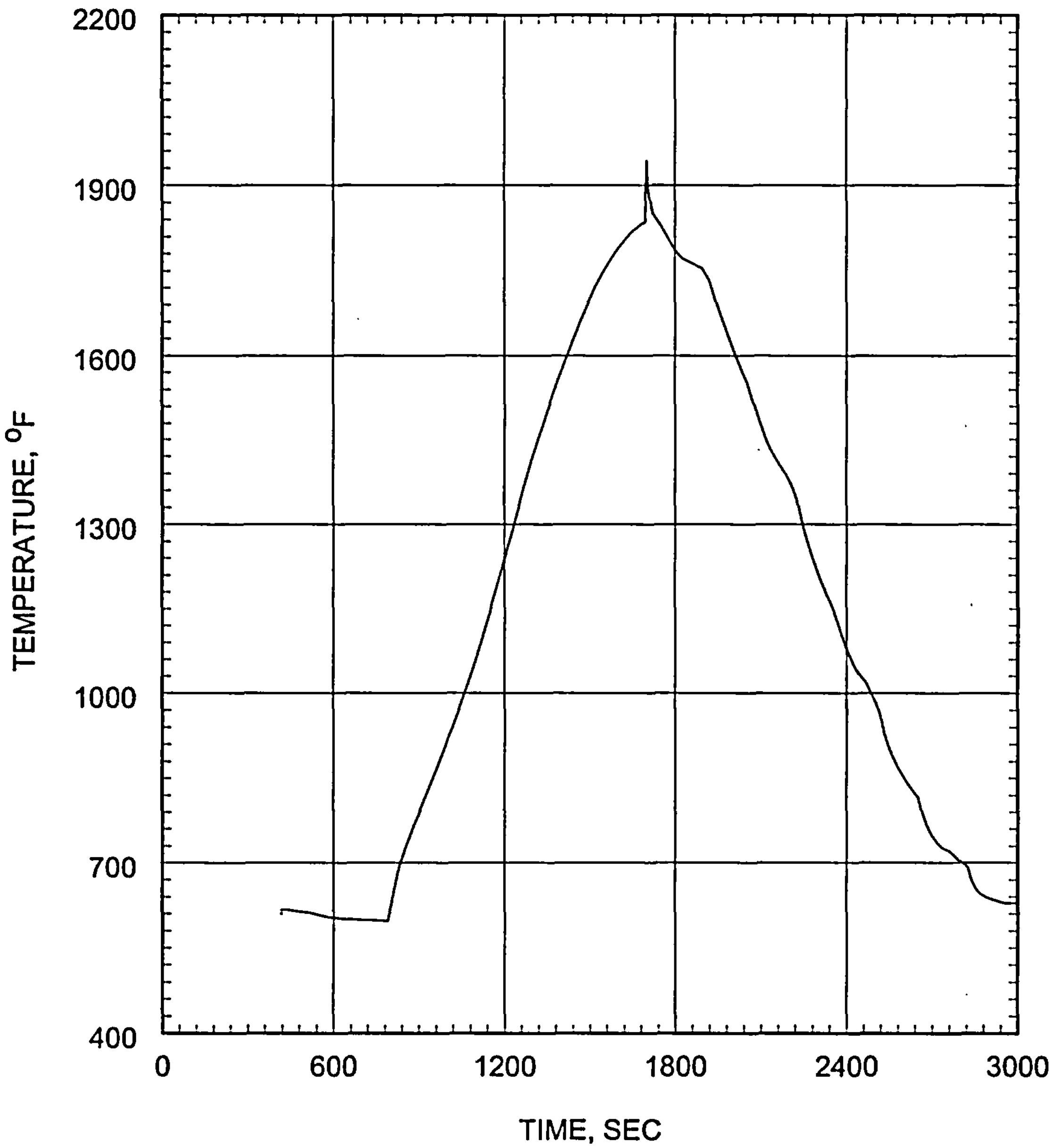

Figure 5.2.4.3-16 Small Break LOCA ECCS Performance Analysis 0.05 ft^2/PD Break Cladding Temperature at Hot Spot

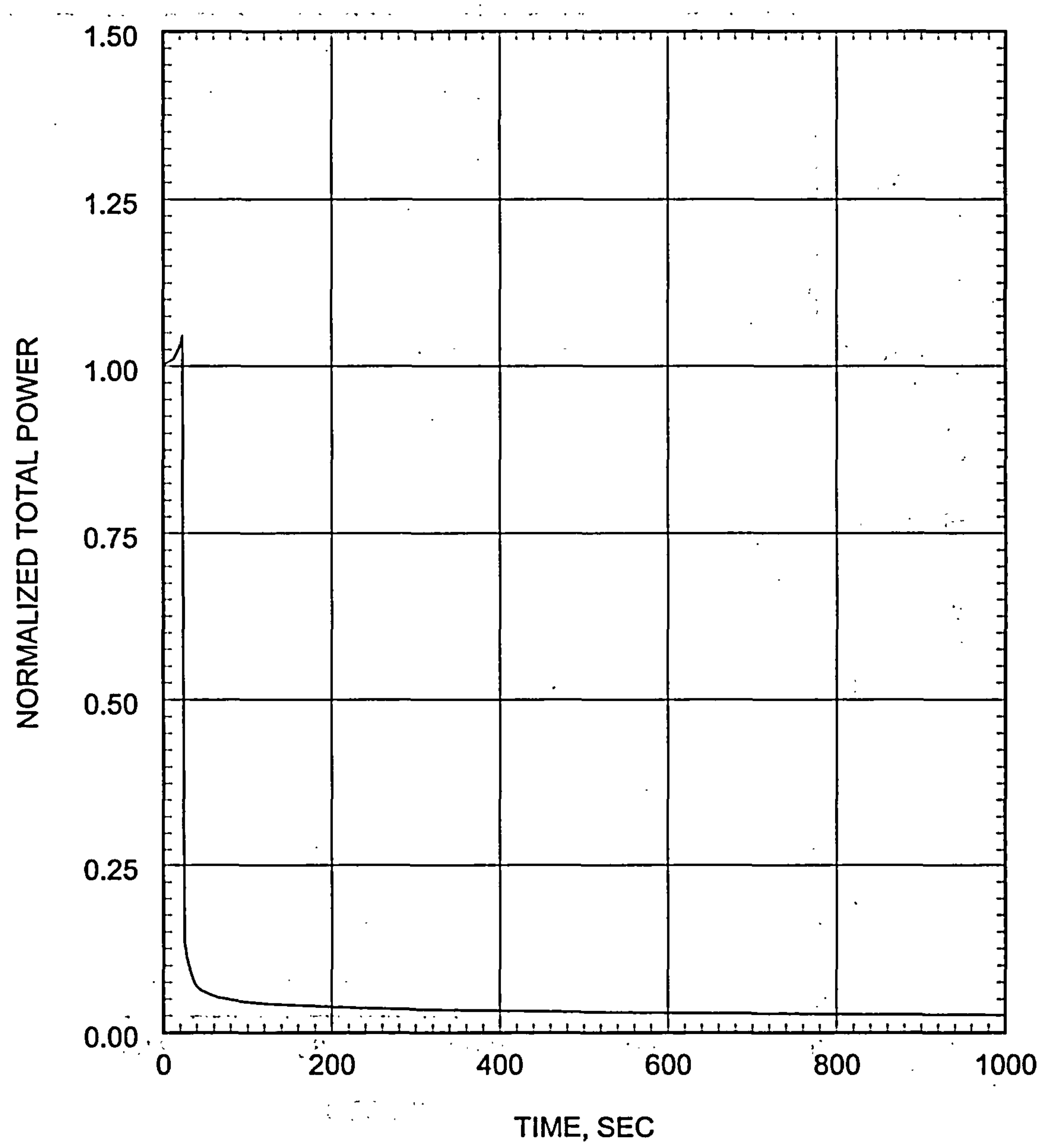

Figure 5.2.4.3-17 Small Break LOCA ECCS Performance Analysis 0.06 ft²/PD Break Core Power

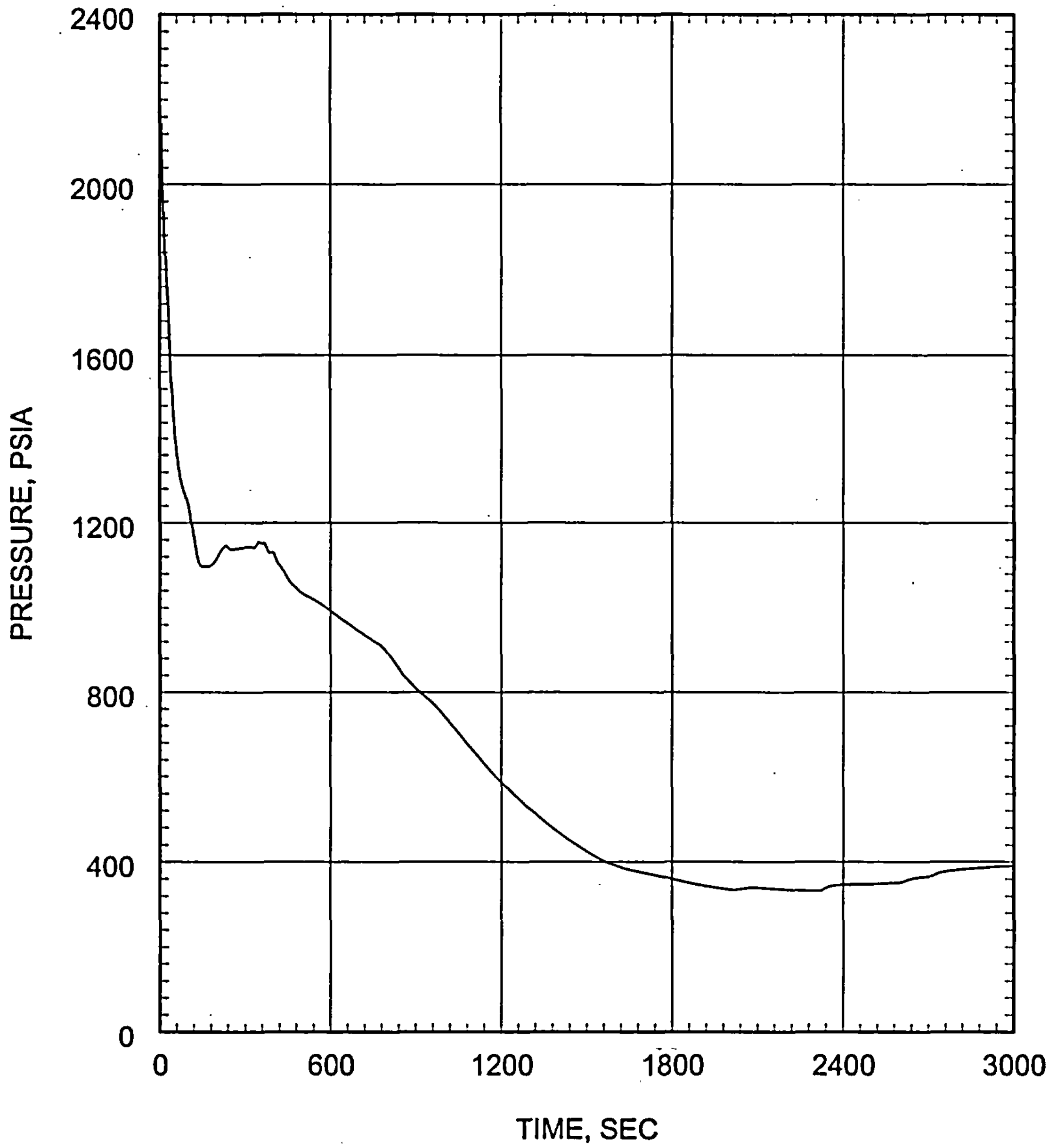

Figure 5.2.4.3-18 Small Break LOCA ECCS Performance Analysis 0.06 ft²/PD Break Inner Vessel Pressure

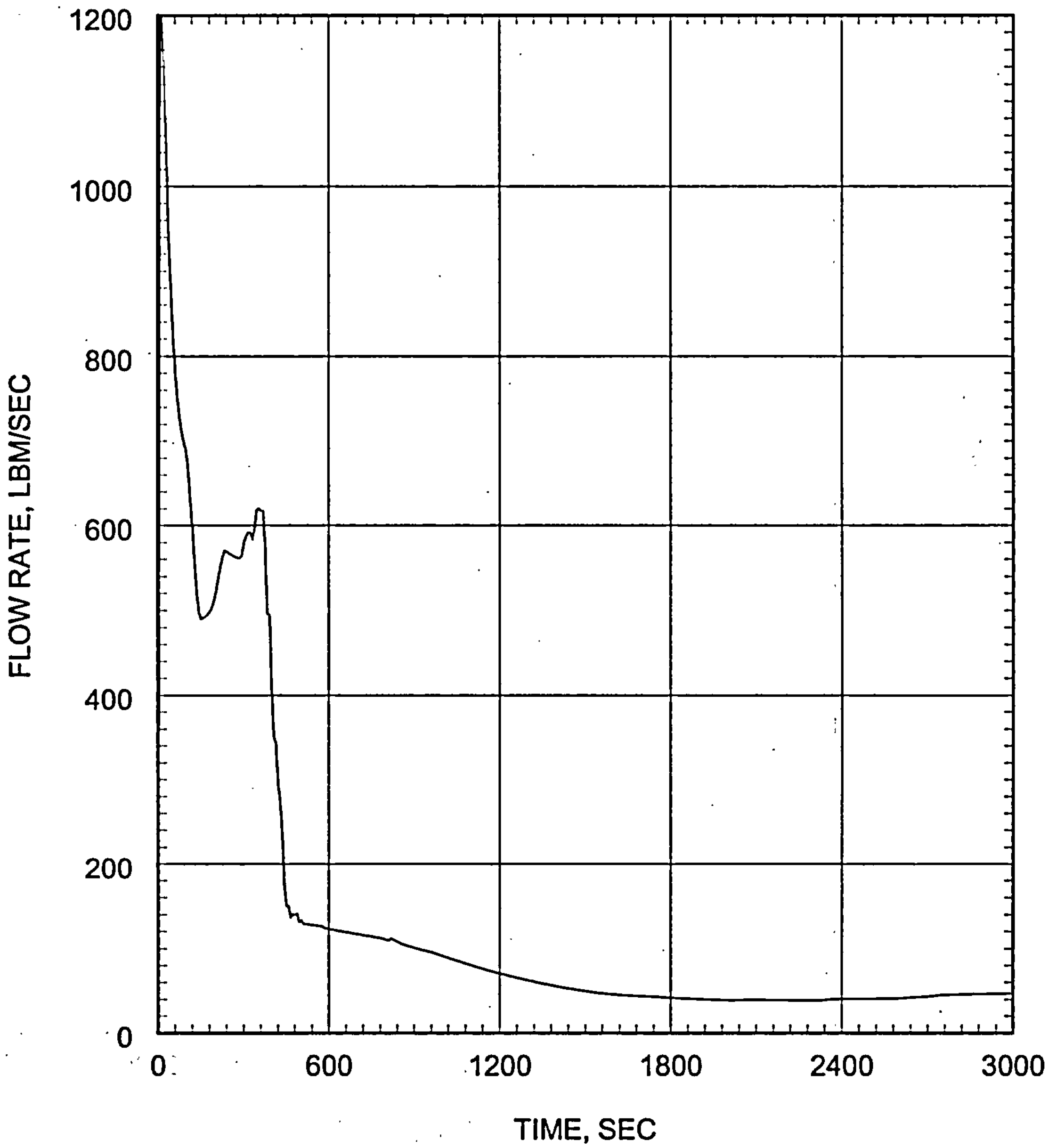

Figure 5.2.4.3-19 Small Break LOCA ECCS Performance Analysis 0.06 ft^2/PD Break Break Flow Rate

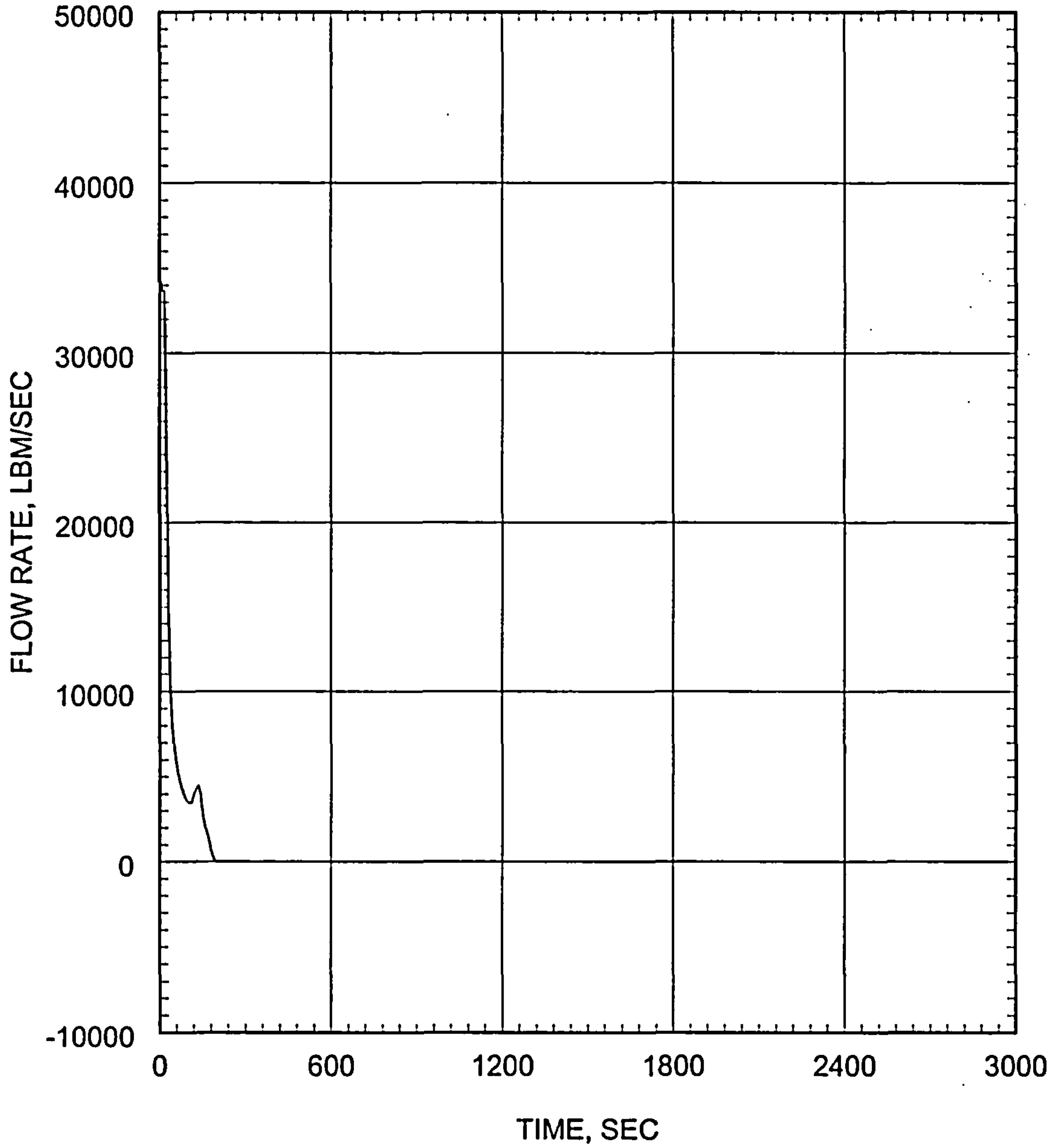

Figure 5.2.4.3-20 Small Break LOCA ECCS Performance Analysis 0.06 ft²/PD Break Inner Vessel Inlet Flow Rate

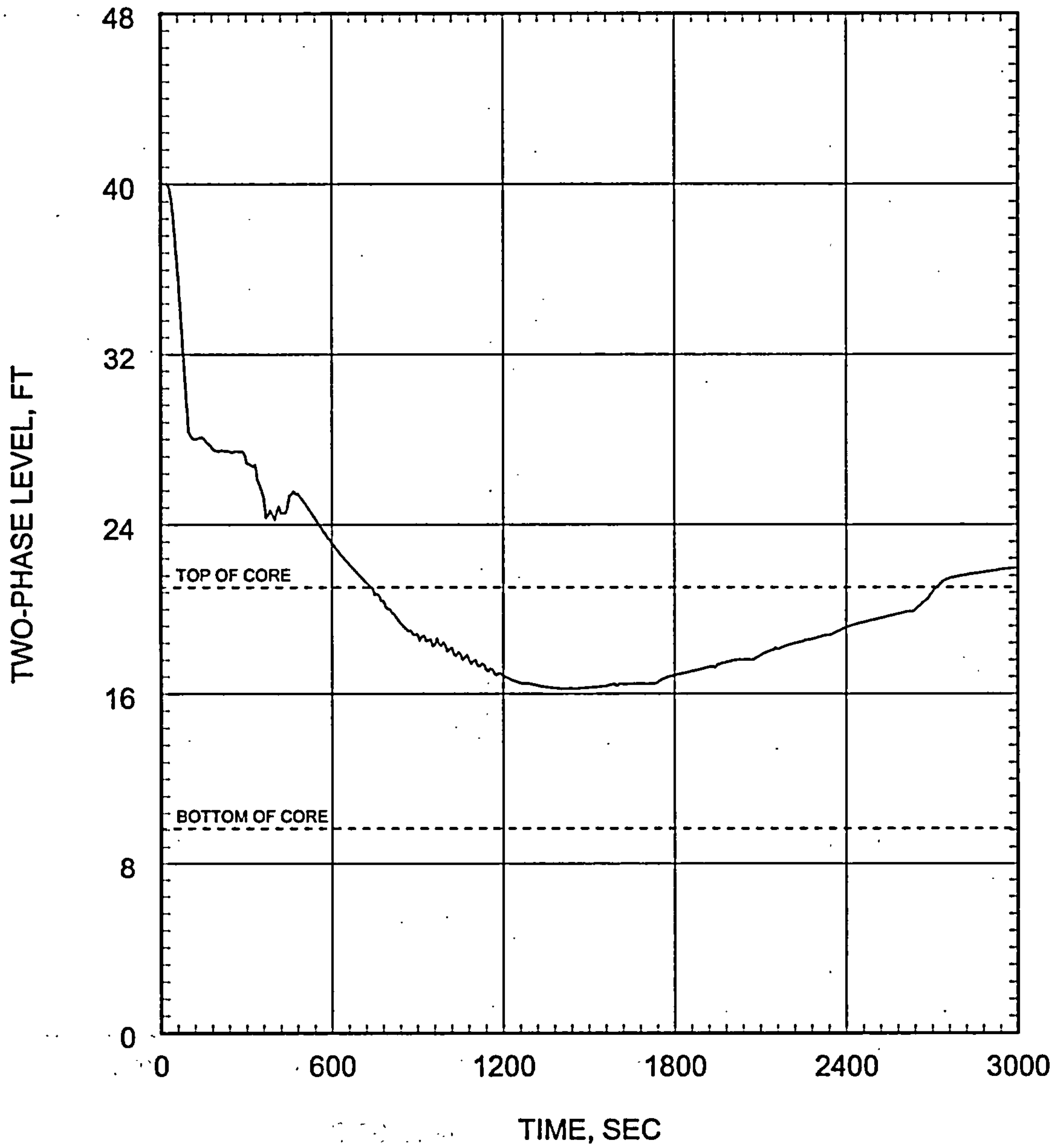

Figure 5.2.4.3-21 Small Break LOCA ECCS Performance Analysis 0.06 ft^2/PD Break Inner Vessel Two-Phase Mixture Level

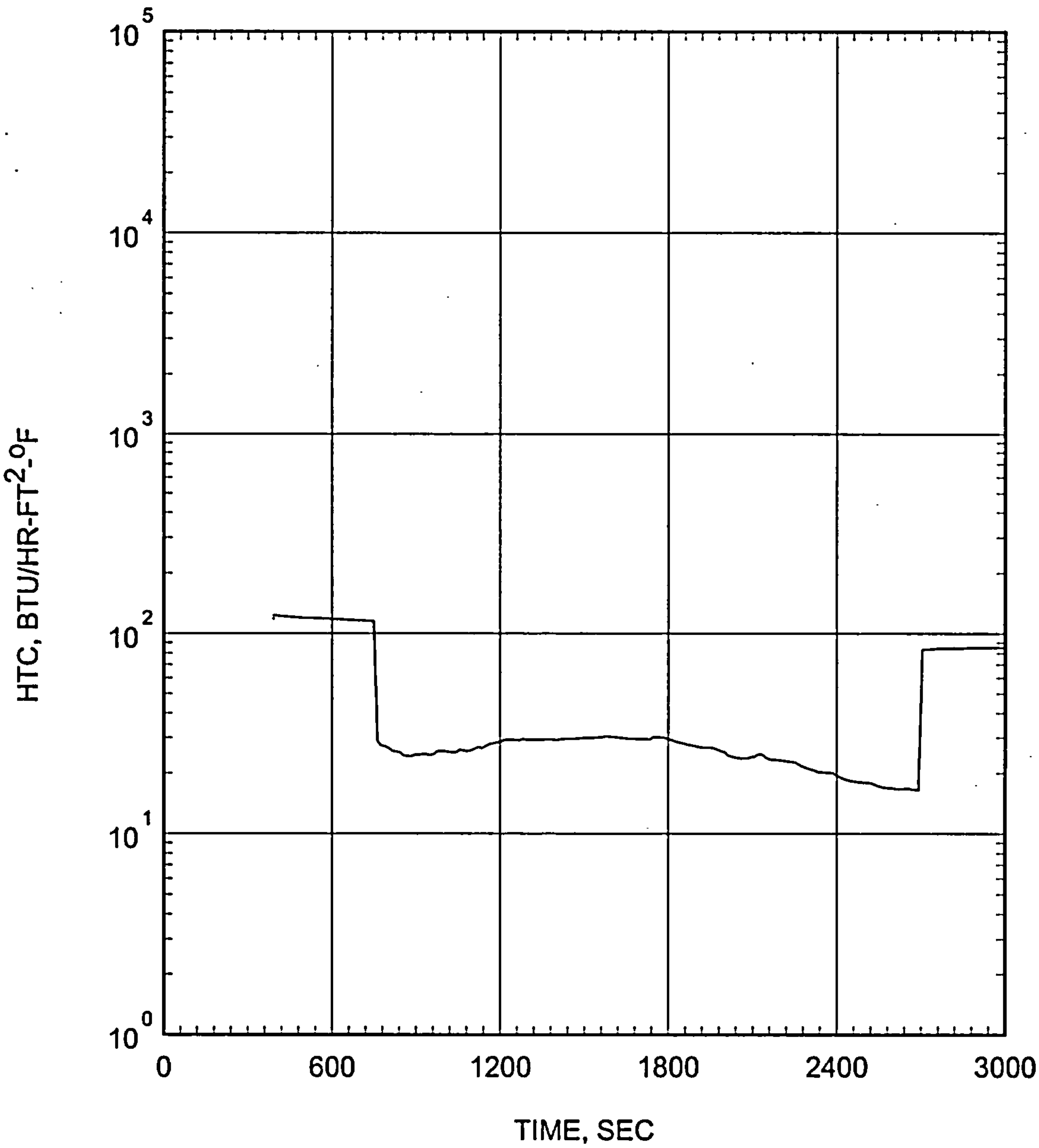

Figure 5.2.4.3-22 Small Break LOCA ECCS Performance Analysis 0.06 ft^2/PD Break Heat Transfer Coefficient at Hot Spot

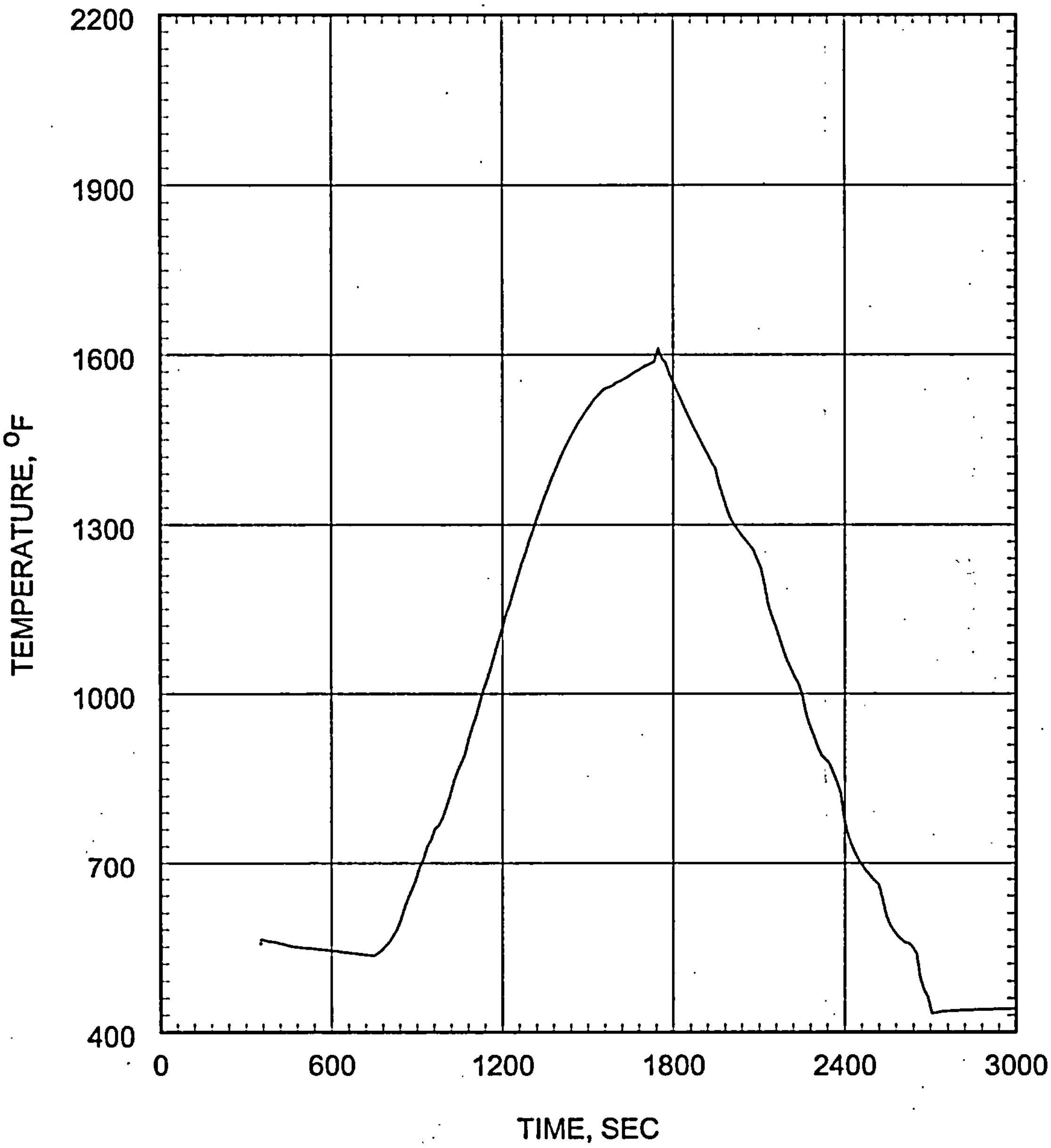

Figure 5.2.4.3-23 Small Break LOCA ECCS Performance Analysis 0.06 ft²/PD Break Coolant Temperature at Hot Spot

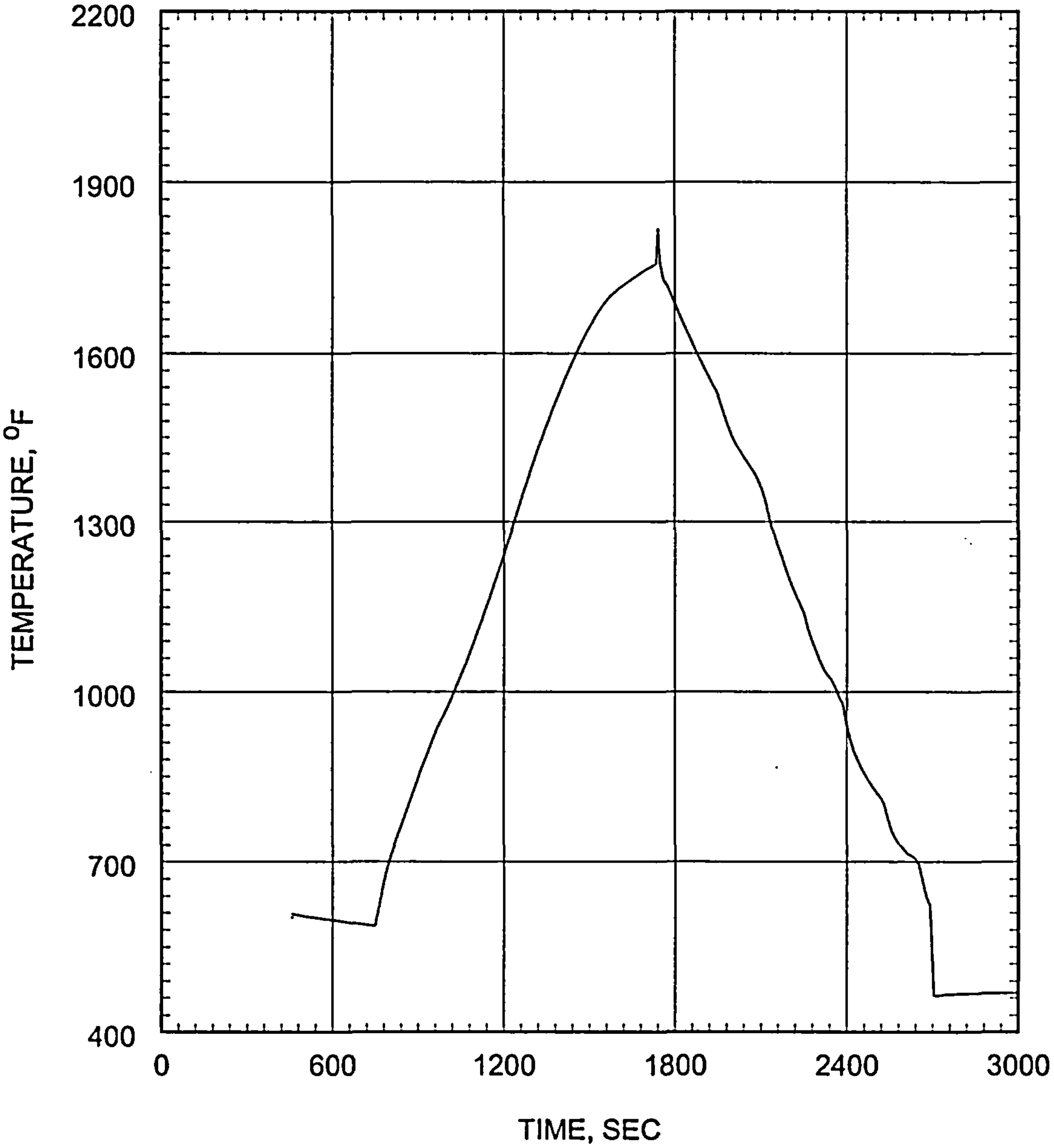

Figure 5.2.4.3-24 Small Break LOCA ECCS Performance Analysis 0.06 ft^2/PD Break Cladding Temperature at Hot Spot

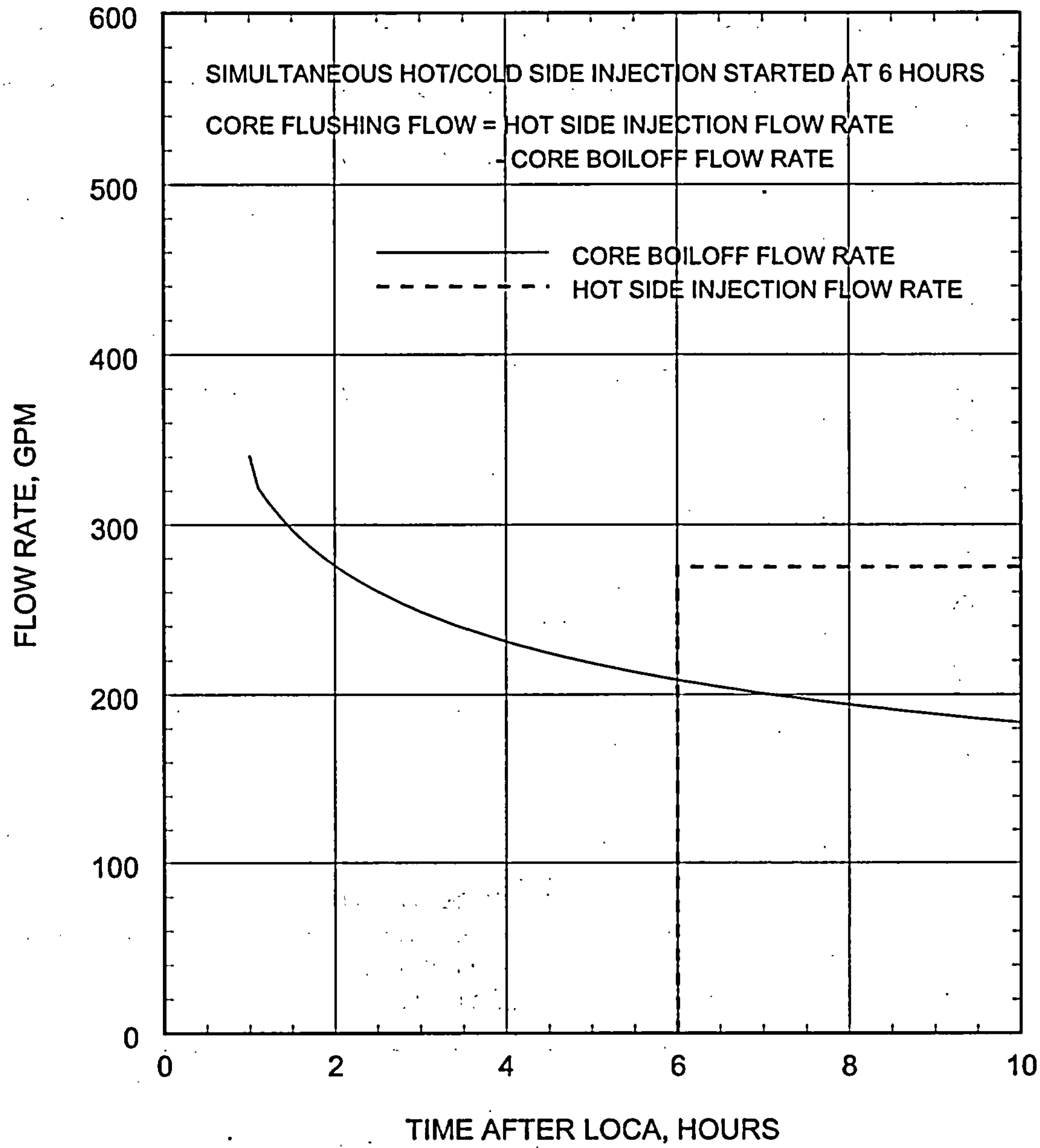

Figure 5.2.5.3-1 Long-Term Cooling Analysis Comparison of Core Boiloff Rate and the Minimum Simultaneous Hot and Cold Side Injection Flow Rate

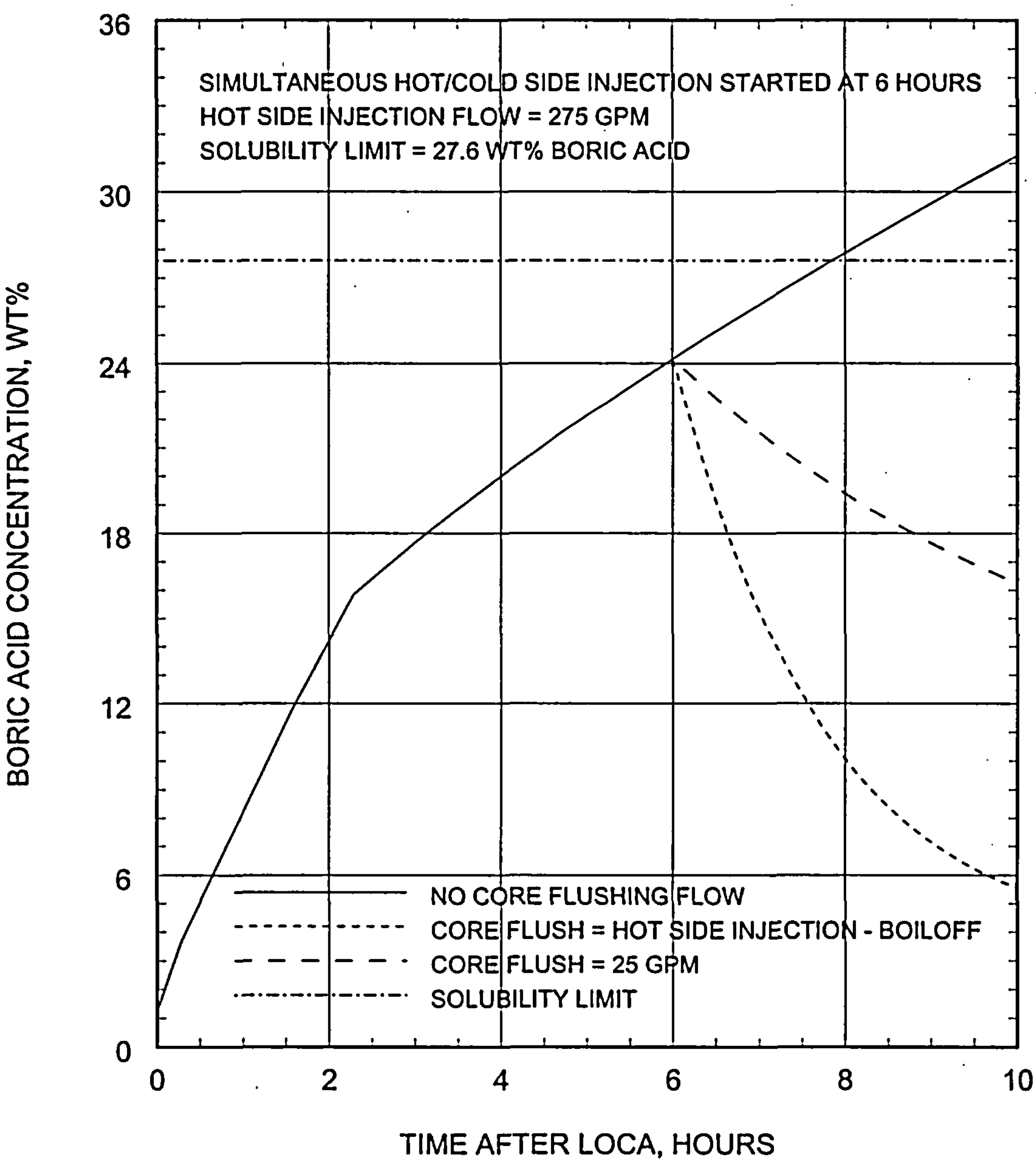

Figure 5.2.5.3-2 Long-Term Cooling Analysis Boric Acid Concentration in the Core Versus Time

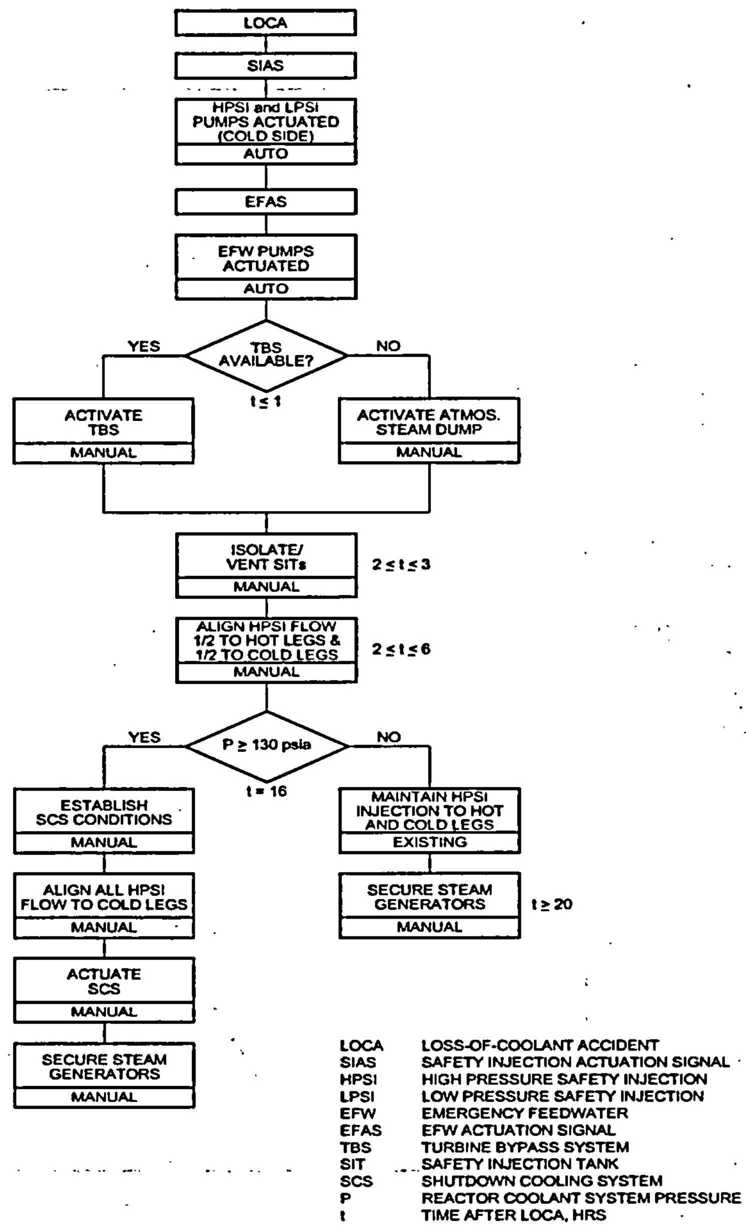

Figure 5.2.5.3-3 Long-Term Cooling Analysis Long-Term Cooling Plan

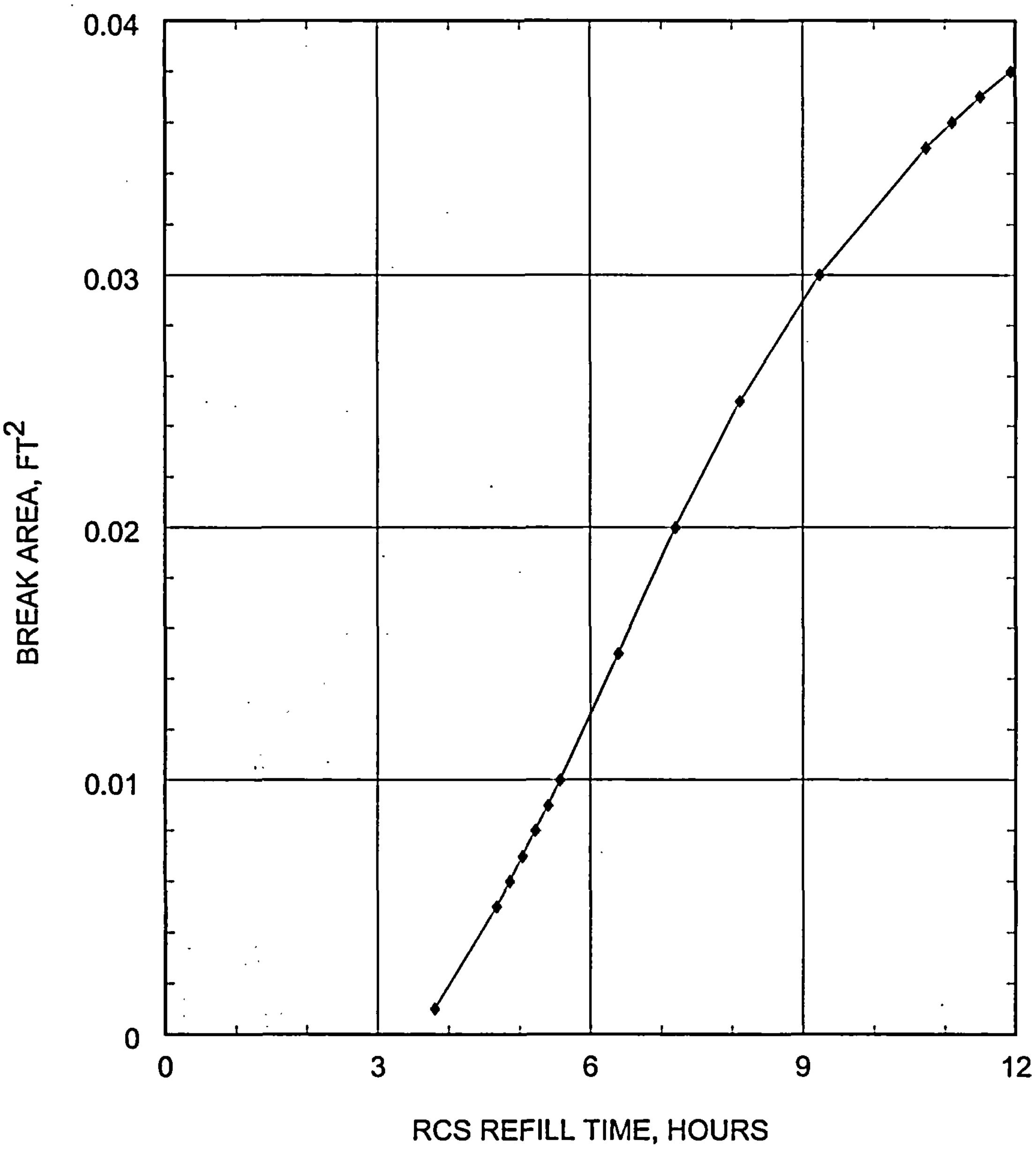

Figure 5.2.5.3-4 Long-Term Cooling Analysis Break Area Versus RCS Refill Time

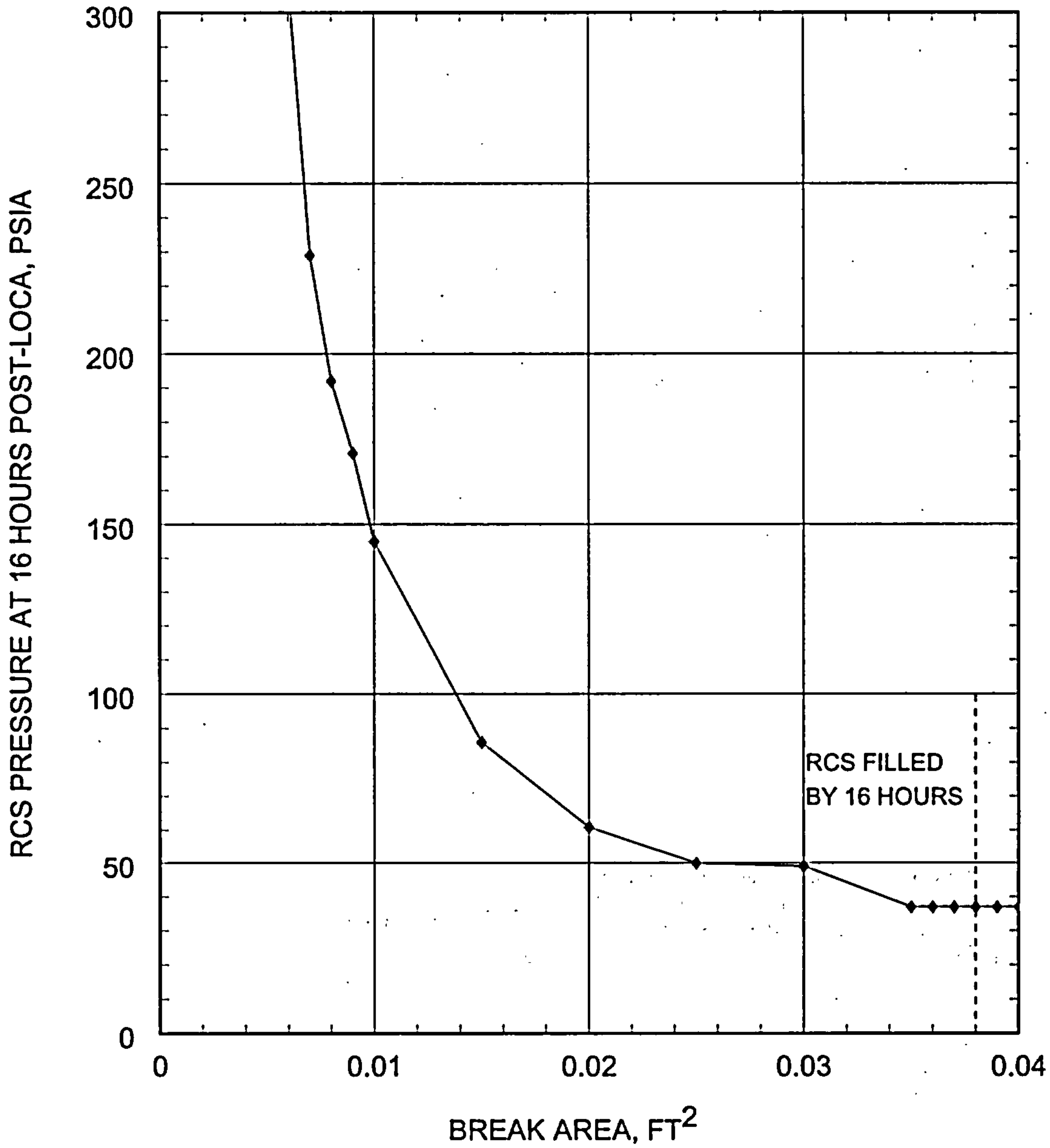

Figure 5.2.5.3-5 Long-Term Cooling Analysis RCS Pressure at the Decision Time Versus Break Area

For break areas 0.007 ft^2 and larger, simultaneous hot/cold leg injection cools the core and flushes boric acid from the core.

For break areas 0.038 ft^2 and smaller, refill of the RCS disperses boric acid throughout the RCS and the SGs cool the RCS to the SDC entry temperature.

Break Size (ft^2)	RCS Pressure at 16 Hours Post-LOCA (psia)
0.040	37
0.039	37
0.038	37
0.037	37
0.036	37
0.035	37
0.030	49
0.025	50
0.020	61
0.015	86
0.010	145
0.009	171
0.008	192
0.007	229
0.006	309
0.005	320
0.001	1073

Figure 5.2.5.3-6 Long-Term Cooling Analysis Overlap of Acceptable Procedures in Terms of Break Size

5.3 STEAMLINE BREAK MASS AND ENERGY RELEASES

5.3.1 Introduction

The containment-related Main Steamline Break (MSLB) event is characterized by the rapid blowdown of steam into the containment due to a rupture in a main steamline. The initial portion of the transient is characterized by the blowdown of both steam generators, including the main steamlines downstream of the main steam isolation valves (MSIVs), until the MSIVs close. The mass and energy releases terminate when the ruptured side steam generator dries out.

5.3.2 Input Assumptions

The planned increased steam generator tube plugging (SGTP), with a corresponding reduction in the reactor coolant system (RCS) flow rate, for St. Lucie 2 results in the following changes to the input:

- SGTP level up to 30% per steam generator
- Minimum RCS flow rate of 335,000 gpm

5.3.3 Acceptance Criteria

No acceptance criteria have been established for the mass and energy releases resulting from a postulated MSLB. The mass and energy releases following the MSLB events are used to determine the containment pressure and temperature response, and the long term equipment qualification (EQ) temperature profile.

5.3.4 Description of Analysis/Evaluation and Results

An engineering evaluation was made to assess the impact of the increased SGTP up to 30% per steam generator with the corresponding reduction in the RCS flow rate. The most limiting mass and energy releases for an inside containment MSLB are generated when assuming that there are no steam generator tubes plugged and using the maximum RCS flow rate. Since mass and energy release data generated in the analysis of record were based on more conservative inputs than current SGTP and reduced RCS flow, the analysis of record mass and energy release data would bound those with increased SGTP and reduced RCS flow rate.

5.3.5 Conclusions

The mass and energy release data for the inside containment MSLB contained in the analysis of record bounds the mass and energy release data with increased SGTP with corresponding reduction in the RCS flow rate.

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5.4 LOSS-OF-COOLANT-ACCIDENT MASS AND ENERGY RELEASES

5.4.1 Introduction

The containment-related Loss-of-Coolant-Accident (LOCA) event is characterized by the rapid discharge of the reactor coolant system (RCS) inventory into the containment. The analytical simulation of the LOCA event is initiated from 102% of rated core power and is characterized by four distinct phases. These are blowdown, reflood, post-reflood and long term cooldown.

The LOCA mass and energy and containment response analysis has been performed in general accordance with Sections 6.2.1.1.A and 6.2.1.3 of the NRC's Standard Review Plan (SRP), Reference 1.

5.4.2 Input Assumptions

The planned increased steam generator tube plugging (SGTP), with a corresponding reduction in the RCS flow rate, has a measurable effect on the LOCA mass and energy releases to the containment. The following list identifies those changes that have the most impact:

1. The increase in the steam generator tube plugging generally results in a decrease of steam generator (SG) heat transfer area which, in turn, results in less heat transfer between the primary to secondary side of the steam generators. Additionally, the increased tube plugging increases the resistance to the break flow. Therefore, the effect of increased tube plugging is a slight decrease in both the mass and energy release rates.

2. The reduced RCS flow rate results in a slight increase in the core/reactor vessel outlet temperature and, thus, slightly higher energy release rates to the containment.

Other assumptions and significant inputs are summarized in Table 5.4-1.

5.4.3 Acceptance Criteria

No acceptance criteria have been established for the mass and energy releases resulting from a postulated LOCA. The mass and energy releases following the LOCA events are used to determine the containment pressure and temperature response.

5.4.4 Description of Analysis/Evaluation and Results

5.4.4.1 Description

The analytical simulation of the LOCA event is characterized by four distinct phases. These are blowdown, reflood, post-reflood and long term cooldown. The mass and energy release data for the first three phases were generated using Nuclear Regulatory Commission (NRC)-approved methodologies and computer codes.

Three break locations were investigated, the double-ended discharge and suction leg slot (DEDLS and DESLS) cold leg breaks and the double-ended hot leg slot (DEHLS) break. All three break locations were analyzed assuming both maximum and minimum safety injection pump flows. The maximum safety injection flow refers to maximum flow from 2 high pressure safety injection (HPSI) and 2 low pressure safety injection (LPSI) pumps. The minimum safety injection flow refers to maximum flow from 1 HPSI and 1 LPSI pumps.

The blowdown phase of the LOCA was simulated with the NRC approved methodology and the CEFLASH-4A computer code cited in Reference 2. The blowdown phase analysis also conforms to the requirements of Appendix K (Reference 3), however some input and nodalization modifications are made to maximize the mass and energy releases to the containment. Note that CEFLASH-4A supercedes an earlier version of the code, CEFLASH-4, cited in the Standard Review Plan (SRP), Section 6.2.1.3. The reflood and post-reflood phases of the LOCA are simulated in accordance with the NRC-approved methodology cited in the SRP, Section 6.2.1.3, and using the FLOOD3 computer code (Reference 4).

The FLOOD3 simulation of the reflood and post-reflood phases is only used for the cold leg breaks. The reflood and post-reflood phases are not simulated for hot leg breaks since there is no viable means for the break flow to pass through the steam generators prior to exiting to containment.

These three phases comprise the short-term LOCA response for the cold leg breaks. For the hot leg break, the short-term response terminates at the end of the blowdown phase with the CEFLASH-4A code. The long-term phase of the LOCA completes the transient simulation of this event. In this phase, the analysis accounts for all residual energy, including decay heat, in the primary and secondary systems. The NRC approved CONTRANS2 containment code (Reference 5), as described in Section 5.5, was run for this long-term phase to calculate the time dependent energy addition due to this sensible heat. The long-term phase was performed for all three break locations.

5.4.4.2 Results

Based on the review of the results of the containment response analysis presented in Section 5.5, the most limiting LOCA event with respect to the maximum (peak) pressure/temperature was determined to be the DEHLS break with 8% steam generator tubes plugged. Table 5.4-2 provides the mass and energy release data for the limiting LOCA case through the time of peak pressure and temperature, which occurs during the blowdown phase. Note that since there is no viable means for the break flow to pass through the steam generators prior to exiting to containment, the reflood and post-reflood phases are not simulated for hot leg breaks. Therefore, only the mass and energy releases during the blowdown phase are presented for this limiting case.

5.4.5 Conclusions

Based on the review of the results of the containment response analysis presented in Section 5.5, the containment pressure and temperature response due to the mass and energy release data for the containment-related LOCA contained in the analysis of record bounds that due to the mass and energy release data with increased SGTP with a corresponding reduction in the RCS flow rate.

5.4.6 References

1. U.S. Nuclear Regulatory Commission NUREG-0800, Revision 1, Standard Review Plan, July 1981.

2. Stuart A. Richards (NRC) to P. W. Richardson (WEC), "Safety Evaluation of Topical Report CENPD-132, Supplement 4, Revision 1, 'Calculative Methods for the CE Engineering Technology Large Break LOCA Evaluation Model,' (TAC No. MA5660)," December 15, 2000.

3. Code of Federal Regulations, 10 CFR §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and 10 CFR part 50, Appendix K, "ECCS Evaluation Models."

4. Calculation No. CODES-FS-C-006, Rev. 03, "Software Verification and Validation Report for FLOOD3, Mod 2," July 7, 1999.

5. Calculation No. VV-OA-02-6, Rev. 0, "CONTRANS2 Revision ctn2m1.0702 Software Change Specification and Validation," August 30, 2002.

Table 5.4-1 Summary of Assumed Initial Conditions and Inputs for LOCA Mass & Energy Release Analysis	
Parameter	**Value / Assumption**
Initial NSSS Parameters	
Power Level (maximum)	2754 MWt (102% of rated core power)
Initial Primary Pressure (maximum)	2250 psia
Initial RCS Inlet Temperature (maximum)	552.0 °F
Initial Secondary Pressure (maximum)	930.0 psia
Initial Secondary Liquid Inventory per SG	134,130 lbm
Steam Generator Tube Plugging	Up to 30% per SG
Reactor Shutdown	On voids
Reactor Coolant Pumps	
Time Tripped	Tripped at initiation of the event
Total Initial RCS Flow rate (minimum)	335,000 gpm
Safety Injection Pump Flow*	
Maximum (No Failure case)	2 HPSIs / 2 LPSIs
Minimum (Loss of One Emergency Diesel Generator case)	1 HPSI / 1 LPSI
Safety Injection Tanks	
Initial Pressure (maximum)	650 psia
Initial Temperature (maximum)	120°F
Main Feedwater	
Enthalpy (maximum)	410.9 Btu/lbm
* Note that the analysis has conservatively omitted the safety injection pump flow during the blowdown phase of the LOCA event.	

Table 5.4-2	Loss-of-Coolant-Accident Limiting Hot Leg Slot Break Mass and Energy Release Data	
Break Type:	Double-Ended Hot Leg Slot (DEHLS)	
Pipe ID:	42 inches	
Break area:	19.24 square feet	
ECCS:	See Note (1)	
SG Tubes Plugged	8% per SG	
1:	BLOWDOWN MASS AND ENERGY RELEASE DATA	
TIME (sec)	MASS RATE (lbm/sec)	ENERGY RATE (Btu/sec)
0.0000000E+00	0.0000000E+00	0.0000000E+00
5.0282104E-02	1.3553761E+05	8.3750844E+07
1.0028211E-01	1.3546292E+05	8.3694896E+07
2.0028211E-01	1.2926662E+05	8.0204579E+07
3.0028212E-01	1.1447722E+05	7.0875610E+07
4.0028212E-01	1.0814796E+05	6.6808626E+07
5.0028213E-01	1.0230087E+05	6.3154209E+07
6.0028213E-01	9.7416651E+04	6.0064659E+07
7.0028213E-01	9.1469634E+04	5.6393501E+07
8.0028214E-01	8.7099604E+04	5.3914167E+07
9.0028214E-01	8.2427082E+04	5.1337237E+07
1.0002821E+00	7.8354602E+04	4.9085376E+07
1.5006598E+00	6.9877471E+04	4.3378654E+07
2.0006598E+00	7.0404778E+04	4.2426032E+07
2.5006598E+00	6.5055867E+04	3.9343054E+07
3.0006598E+00	5.7552528E+04	3.5928080E+07
3.5006598E+00	5.2573956E+04	3.3180409E+07
4.0006597E+00	5.0303002E+04	3.1516033E+07
5.0006597E+00	4.3391864E+04	2.7627820E+07
6.0006597E+00	3.3723227E+04	2.2788723E+07
7.0006597E+00	1.8783249E+04	1.5890902E+07
8.0022945E+00	1.4919700E+04	1.2387470E+07
9.0022944E+00	1.0129803E+04	9.0101953E+06
1.0002294E+01	6.1150884E+03	5.8969869E+06
1.1001362E+01	3.1099286E+03	3.7970003E+06

Table 5.4.4.2-1 Loss-of-Coolant-Accident Limiting Hot Leg Slot Break Mass and Energy Release Data (cont.)		
TIME (sec)	MASS RATE (lbm/sec)	ENERGY RATE (Btu/sec)
1.2000121E+01	2.9851657E+03	3.6414891E+06
1.2100121E+01	2.7650875E+03	3.3685126E+06
1.2200021E+01	2.6068649E+03	3.1766744E+06
1.2299921E+01	2.3809213E+03	2.8996514E+06
1.2399921E+01	2.1812468E+03	2.6549393E+06
1.2499921E+01	2.0210397E+03	2.4623501E+06
1.2599721E+01	1.8563435E+03	2.2676729E+06
1.2704121E+01	1.7035027E+03	2.0830555E+06
1.2803621E+01	1.5809766E+03	1.9357624E+06
1.2903121E+01	1.4577781E+03	1.7867476E+06
1.3002621E+01	1.3572104E+03	1.6646832E+06
1.3102221E+01	1.2749713E+03	1.5650602E+06
1.3201821E+01	1.2062421E+03	1.4816014E+06
1.3301421E+01	1.1434998E+03	1.4052442E+06
1.3401121E+01	1.0871355E+03	1.3366741E+06
1.3500821E+01	1.0321852E+03	1.2696225E+06
1.3600421E+01	9.8508463E+02	1.2122058E+06
1.3700121E+01	9.3859763E+02	1.1557747E+06
1.3799821E+01	8.9182906E+02	1.0986794E+06
1.3904421E+01	8.5744610E+02	1.0567023E+06
1.4004021E+01	8.2134695E+02	1.0126736E+06
1.4103621E+01	7.9569109E+02	9.8131589E+05
1.4203139E+01	0.0000000E+00	0.0000000E+00
Integrated Mass = 4.5591015E+05 lbm Integrated Energy = 3.0170474E+08 Btu		

Note:
(1) Note that the analysis has conservatively omitted the safety injection pump flow during the blowdown phase of the LOCA event.

5.5 CONTAINMENT INTEGRITY

5.5.1 Introduction

The containment building encloses the primary and secondary plant and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding the pressure and temperature conditions resulting from a postulated loss-of-coolant-accident (LOCA) and main steamline break (MSLB). The containment response analysis is performed to demonstrate that the design pressure and temperature conditions for the containment structure are not exceeded. The Equipment Qualification (EQ) analysis defines a temperature envelope within which the operability of all inside containment Class 1E safety-related equipment is ensured.

The St. Lucie Unit 2 plant has a current containment design pressure of 44 psig.

The containment response analysis has been performed in general accordance with the Nuclear Regulatory Commissions (NRC's) Standard Review Plan (SRP), Reference 1, which invokes the requirements of the General Design Criteria (GDC) 50.

5.5.2 Input Assumptions

The planned increased steam generator tube plugging (SGTP), with a corresponding reduction in the reactor coolant system (RCS) flow rate, has a negligible impact on the MSLB mass and energy releases to the containment and provides less limiting conditions. Consequently, the containment pressure and temperature conditions resulting from a postulated MSLB event will be bounded by the current conditions. However, the SGTP and reduced RCS flow have a measurable impact on the LOCA mass and energy releases to the containment. The combined effect of the SGTP and reduced RCS flow rate is an increase in the energy release to the containment and, consequently, the containment pressure and temperature conditions resulting from a postulated LOCA will be more adverse. Therefore, the discussion in this report is limited to the postulated LOCA.

Table 5.5-1 provides an overview of the significant input parameters for the containment response calculations.

5.5.3 Acceptance Criteria

The acceptance criteria are that the calculated peak containment atmosphere pressure and temperature should remain below the original design basis values. However, Florida Power and Light (FPL) has set a conservative limit on the pressure of 41.8 psig. This criterion corresponds to the current integrated leak rate test (ILRT) pressure. Another criterion set by FPL is to remain within the existing St. Lucie Unit 2 long term equipment qualification (EQ) temperature. This criterion is based on the integrated approach for the EQ temperature.

5.5.4 Description of Analysis/Evaluation and Results

5.5.4.1 Description

The LOCA containment analysis is performed in two parts. As described in Section 5.4, the NRC approved CEFLASH-4A and FLOOD3 computer codes were used to generate the mass and energy discharged from the RCS into the containment. This information was then utilized to calculate the containment response using the NRC approved CONTRANS computer code (References 2 and 3). This subsection provides both an overview of the analysis methodology and a summary of the important analysis results.

Three break locations were investigated – the reactor coolant pump (RCP) discharge and suction (cold) legs and the hot leg. All breaks analyzed were double-ended slot breaks.

Offsite power was assumed to be lost at the initiation of the LOCA. This provided the maximum delay in starting of the containment heat removal systems. Additionally, two types of scenarios were considered. The first represented loss of offsite power with no coincidental failure of emergency diesel generators (EDGs), the second represented loss of offsite power with failure of an EDG to start. These scenarios could be represented by assuming two trains of safety injection (SI) flow (i.e., maximum SI) and one train of SI flow (i.e., minimum SI), respectively. The CONTRANS analysis showed that the failure of one EDG was limiting.

5.5.4.2 Results

Table 5.5-2 lists the computer cases that were run to determine the post-LOCA containment pressure and temperature results for all the LOCA cases analyzed. As shown, the resulting maximum pressure and temperature were 39.53 psig and 262.12°F, respectively, for a double-ended slot break in the hot leg with an assumed failure of one EDG (minimum SI). Note that the peak pressure and temperature occur during the blowdown phase and, therefore, there is no impact due to single failure on these values. These values are less than the maximum pressure and temperature of 41.8 psig and 271.4°F, respectively, cited in the St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR) for the original analysis for the double-ended slot break in the RCP suction (cold) leg with no failure (maximum SI). Additionally, Figures 5.5-1 and 5.5-2 present the containment pressure and temperature response for the limiting hot leg break case.

The chart below compares the results from the limiting case to the results from original analysis documented in the UFSAR and includes the acceptance limits.

Criterion	Current Analysis	Original Analysis	Acceptance Limit
Peak Pressure < Maximum Design Pressure	39.53 psig	41.8 psig	< 44 psig
Peak Temperature < EQ Peak Temperature	262.12 °F	271.4 °F	-

5.5.5 Conclusions

Based on the review of the results of the containment response analysis presented in Section 5.5.4, the containment pressure and temperature response following postulated LOCA events with increased SGTP, and corresponding reduction in the RCS flow rate, meet the acceptance criteria documented in the UFSAR.

5.5.6 References

1. U.S. Nuclear Regulatory Commission NUREG-0800, Revision 1, Standard Review Plan, July 1981.

2. CENPD-140A, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," June 1976.

3. Calculation No. VV-OA-02-6, Rev. 0, "CONTRANS2 Revision ctn2m1.0702 Software Change Specification and Validation," August 30, 2002.

Table 5.5-1 Summary of Assumed Initial Conditions and Inputs for LOCA Containment Response Analysis		
Parameter	**Value / Assumption**	
Containment Parameters		
Free Volume	2.506×10^6 ft^3	
Initial Pressure	14.7 psia	
Initial Temperature (maximum)	120 °F	
Initial Relative Humidity	45%	
Refueling Water Tank		
Temperature (maximum)	100 °F	
Usable Water Volume (minimum)	305,600 gallons	
Safety Injection Pump Flow		
No EDG Failure	2 HPSIs / 2 LPSIs	
Loss of One EDG	1 HPSI / 1 LPSI	
Containment Fan Cooler Parameters		
Number of Fan Cooler Trains	2	
Number Fan Coolers per Train	2	
Actuation Setpoint	6.0 psig	
Actuation Delay Time	25.65 sec.	
CCW Flow Rate / Base Temperature	1500 gpm / 105 °F	
Heat Removal (per Fan Cooler)	Air Temperature, °F	Heat Removal, Btu/sec
	120	988.33
	200	10163.83
	280	22488.67
	500	22488.67
Containment Spray Parameters		
Number of CS Trains	2	
Number CS Pumps per Train	1	
Actuation Setpoint	6.0 psig	
Actuation Delay Time	60.0 sec.	
Spray Flow Rate per Pump prior to Recirculation Actuation Signal (pre-RAS)	3450 gpm	
Spray Flow Rate per Pump (post-RAS)	3600 gpm	

Table 5.5-2	Maximum Post-LOCA Containment Pressures and Temperatures		
		Maximum Containment @ time (sec)	
Case #	Case Description	Pressure (psig)	Temperature (°F)
1	Double Ended Discharge Leg (DEDLS) Break with 30%SGTP, Maximum SI	34.78 @ 13.31	254.39 @ 13.31
2	Double Ended Discharge Leg (DEDLS) Break with 30%SGTP, Minimum SI	34.78 @ 13.31	254.39 @ 13.31
3	Double Ended Discharge Leg (DEDLS) Break with 8%SGTP, Maximum SI	36.14 @ 13.31	256.70 @ 13.31
4	Double Ended Discharge Leg (DEDLS) Break with 8%SGTP, Minimum SI	36.14 @ 13.31	256.70 @ 13.31
5	Double Ended Discharge Leg (DEDLS) Break with 20%SGTP, Maximum SI	35.54 @ 13.31	255.68 @ 13.31
6	Double Ended Discharge Leg (DEDLS) Break with 20%SGTP, Minimum SI	35.54 @ 13.31	255.68 @ 13.31
7	Double Ended Suction Leg (DESLS) Break with 30%SGTP, Maximum SI	35.04 @ 18.31	254.86 @ 18.31
8	Double Ended Suction Leg (DESLS) Break with 30%SGTP, Minimum SI	35.04 @ 18.31	254.86 @ 18.31
9	Double Ended Suction Leg (DESLS) Break with 8%SGTP, Maximum SI	36.93 @ 18.81	258.03 @ 18.81
10	Double Ended Suction Leg (DESLS) Break with 8%SGTP, Minimum SI	36.93 @ 18.81	258.03 @ 18.81
11	Double Ended Suction Leg (DESLS) Break with 20%SGTP, Maximum SI	35.88 @ 18.31	256.29 @ 18.31
12	Double Ended Suction Leg (DESLS) Break with 20%SGTP, Minimum SI	35.88 @ 18.31	256.29 @ 18.31
13	Double Ended Hot Leg (DEHLS) Break with 30%SGTP, Maximum SI	37.49 @ 15.80	258.89 @ 15.80
14	Double Ended Hot Leg (DEHLS) Break with 30%SGTP, Minimum SI	37.49 @ 15.80	258.89 @ 15.80
15	Double Ended Hot Leg (DEHLS) Break with 8%SGTP, Maximum SI	39.53 @ 17.80	262.12 @ 17.80
16	Double Ended Hot Leg (DEHLS) Break with 8%SGTP, Minimum SI	39.53 @ 17.80	262.12 @ 17.80
17	Double Ended Hot Leg (DEHLS) Break with 20%SGTP, Maximum SI	38.47 @ 17.30	260.46 @ 17.30
18	Double Ended Hot Leg (DEHLS) Break with 20%SGTP, Minimum SI	38.47 @ 17.30	260.46 @ 17.30

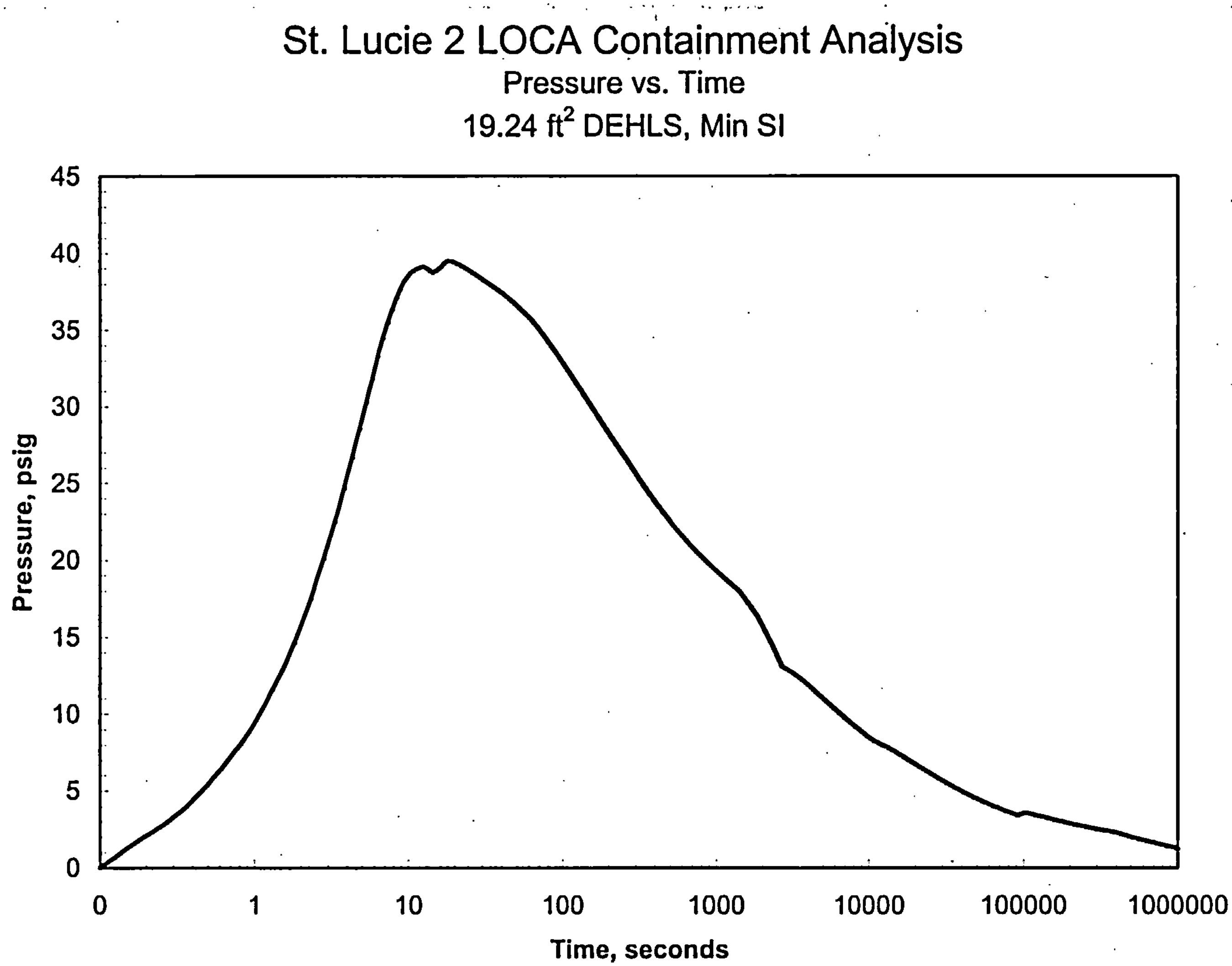

Figure 5.5-1 St. Lucie 2 LOCA Containment Analysis – Pressure vs. Time – 19.24 ft² DEHLS, Min SI

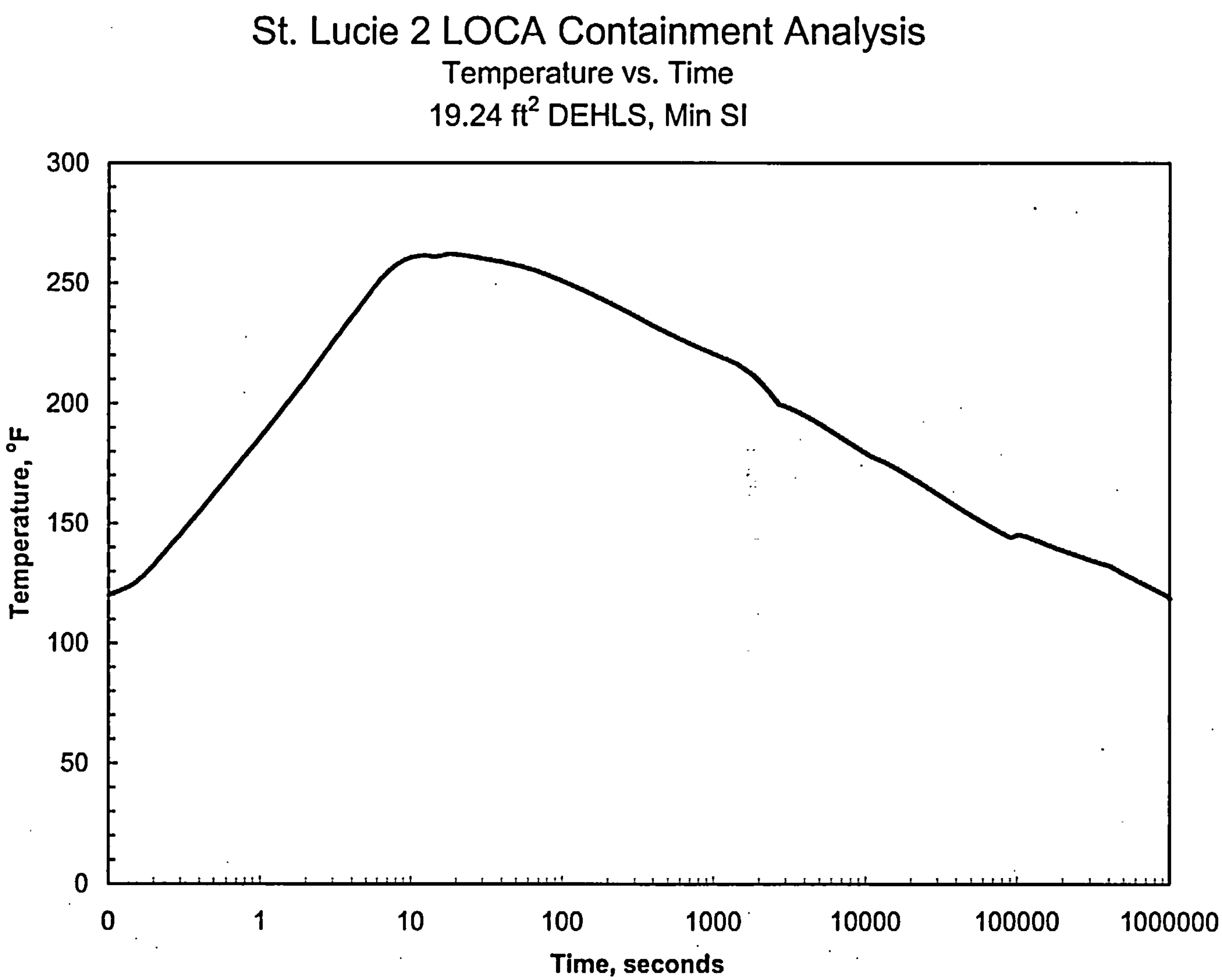

Figure 5.5-2 St. Lucie 2 LOCA Containment Analysis – Temperature vs. Time – 19.24 ft^2 DEHLS, Min SI

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APPENDIX A
SUPPLEMENTAL LICENSING INFORMATION

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This appendix provides supplemental licensing information to support the NRC review of the methods and application of methods used to support the St. Lucie Unit 2 30% Steam Generator Tube Plugging and WCAP-9272 Reload Methodology transition. The information included here is not intended to be a permanent part of the St. Lucie Unit 2 license, put provides an additional level of detail – sample applications, historical perspective, etc., to supplement the body of the licensing information contained elsewhere in this report.

Relaxed Axial Offset Control (RAOC)

Cycle 15 will represent the first application of the RAOC methodology to St. Lucie Unit 2. While the actual calculations will be performed as part of the reload evaluation process for Cycle 15 (and subsequent cycles), a representative design for Cycle 15 was used as a "test" of the expected performance of the RAOC methodology in advance of the Cycle 15 design.

With the reduction in peak linear heat rate required to satisfy the requirements of 10 CFR 50.46 for Large Break LOCA, the studies were performed with this reduced linear heat rate COLR limit which reduces operating margins for a given reload design. In this case, the representative design for Cycle 15 was run with a peak linear heat rate limit of 12.5 kw/ft. The results demonstrated non-trivial violations especially for the wide ASI limits at ~ 70% power (see Section 3.6 for additional discussion). Constant monitoring for linear heat rate with the Incore Detector Monitoring System (IDMS) addresses linear heat rate considerations when the IDMS is used (COLR Figure 3.2-1). However, when using the Excore Detector Monitoring System (EDMS), the linear heat rate reduction requires restriction of the allowable ASI (COLR Figure 3.2.2) based on this study. To address the peak linear heat rate reduction, the allowable ASI limits were reduced by 1) shifting the positive ASI "wing" inward and reducing the breakpoint (power for maximum allowable ASI), and 2) reducing the breakpoint for the negative ASI. Starting from this set of assumed ASI limits, the RAOC calculations were run based on these assumptions:

- Use of the revised LHR LCO, as a proposed replacement for the current LHR LCO,
- Application of a very conservative 8% uncertainty on ASI[1], and
- A peak linear heat rate limit of 12.5 kw/ft.

These calculations for a representative Cycle 15 reload design resulted in peak linear heat rates less than 12.5 kw/ft.

Figure A-1 presents the revised LHR LCO (COLR Figure 3.2.2) proposed for Cycle 15. Figure A-2 shows the flyspeck results (locus of limiting linear heat rate data) of the final RAOC "test" calculations compared with the 12.5 kw/ft peak linear heat rate limit currently proposed for Cycle 15. These results confirm the acceptability of the revised ASI limits for linear heat rate when using the EDMS, COLR Figure 3.2-2, as discussed in Section 3.6.

[1] An 8% uncertainty was a value chosen in early stages of the project based on high confidence that the final approach relative to uncertainties, including confirmation of ASI uncertainties, would support application of an ASI basis for Anticipated Operational Occurrences well inside the assumed envelope. Due to process considerations, the very conservative 8% has been used throughout the RAOC calculations, and is the basis for linear heat rate confirmations, despite subsequent information confirming that much less limiting conditions are supportable. FPL may choose to reclaim all or part of this margin in future based on cycle-specific analyses and/or evaluations justifying this margin reclamation.

Note that future activities may result in relaxation of 12.5 kw/ft peak linear heat rate limit. Recalculation of the RAOC cases under a relaxed peak linear heat rate assumption could then serve as the basis for relaxation of the ASI limits for linear heat rate when using the EDMS, which would be implemented through the COLR on a cycle-specific basis.

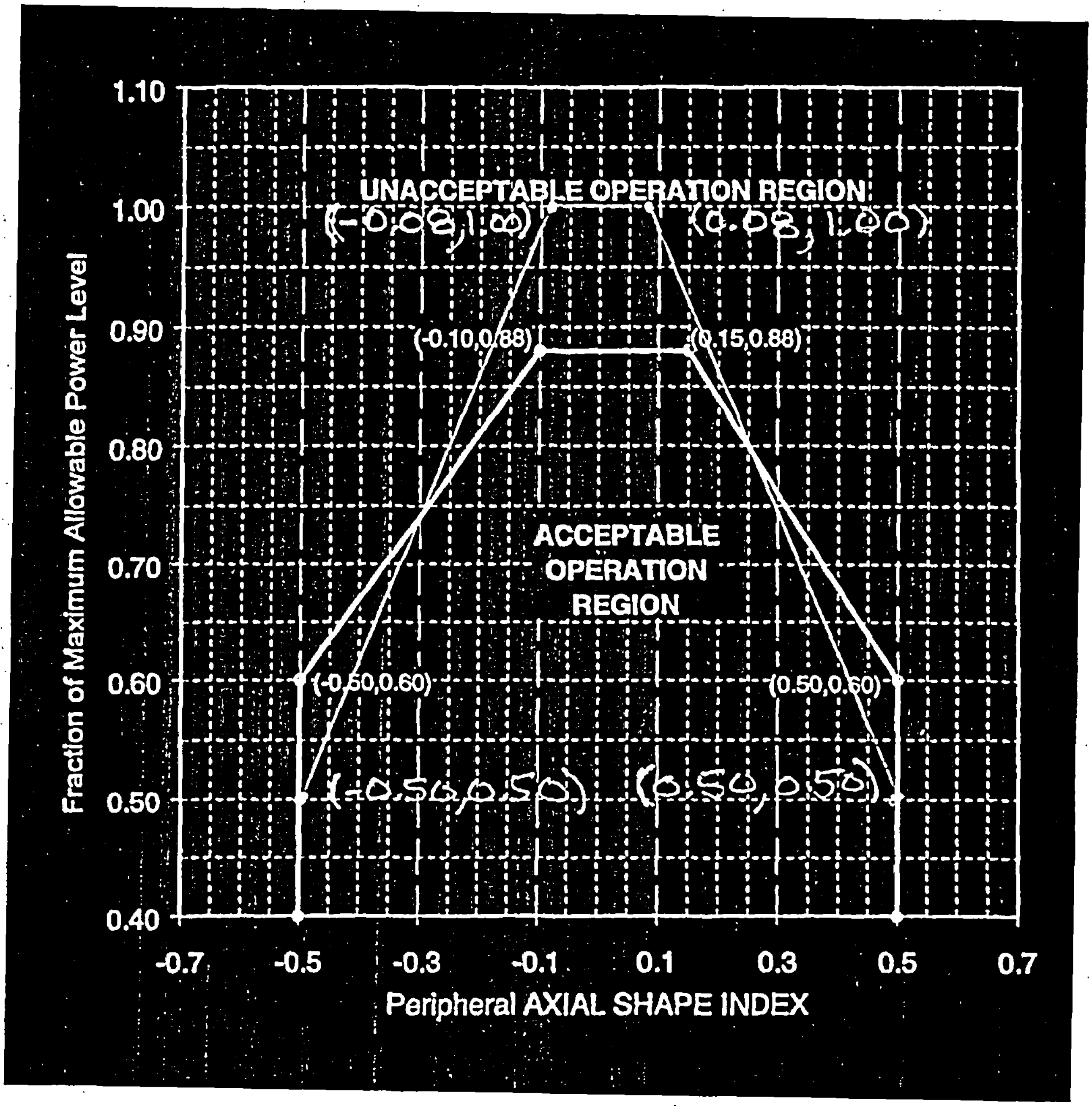

Figure A-1 Proposed AXIAL SHAPE INDEX Limits for Linear Heat Rate when Using Excore Detector Monitoring System (COLR Figure 3.2.2)

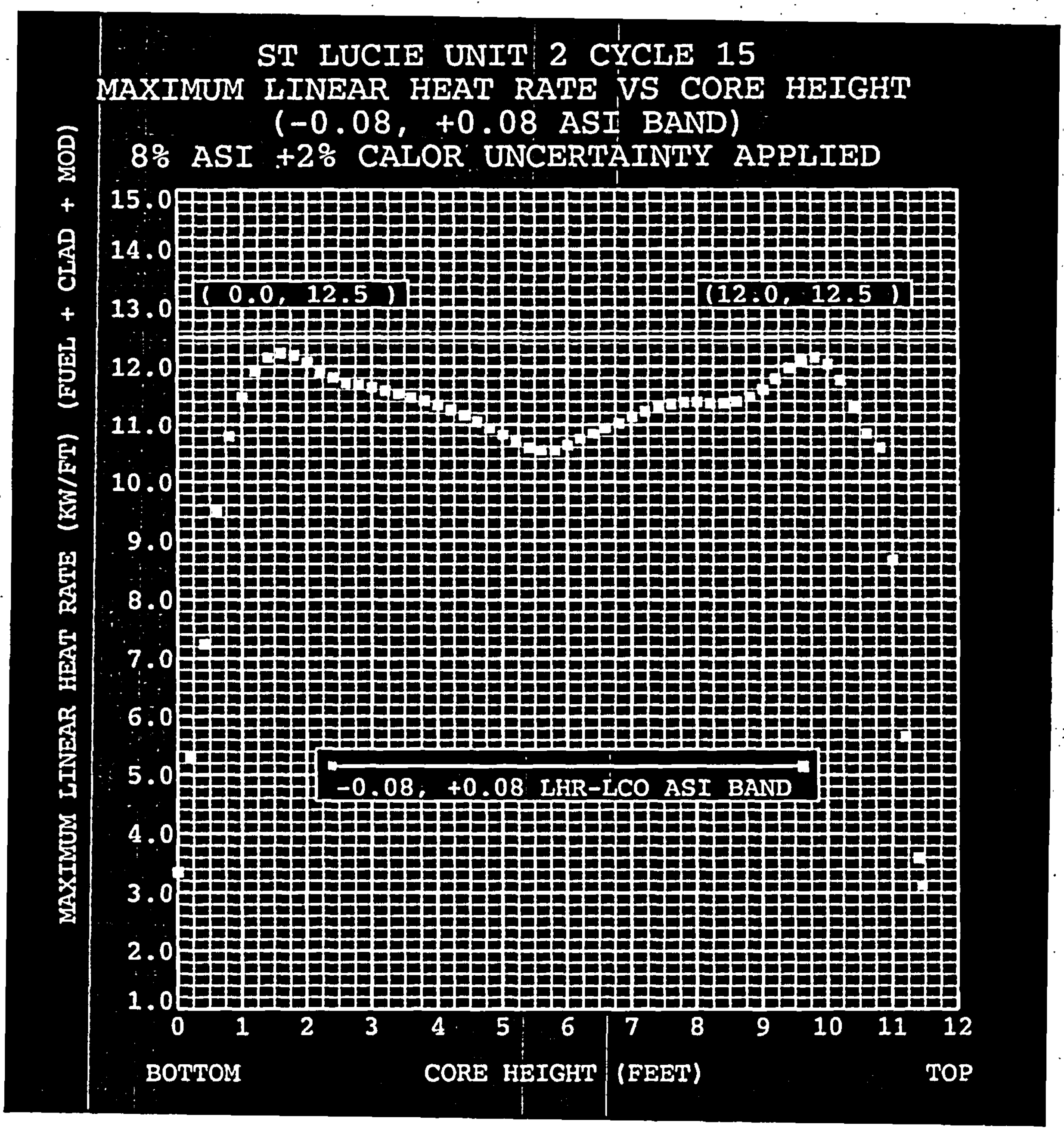

Figure A-2 RAOC Flyspeck Based on Limits of Figure D-1 and a Peak Linear Heat Rate Limit of 12.5 kw/ft

Post-Trip Steamline Break with Loss of AC

I. Background

The post-trip steamline break event is an ANS Condition IV event that is analyzed to ANS Condition II acceptance criteria. In particular the Westinghouse analysis methodology demonstrates that the departure from nucleate boiling (DNB) design basis is satisfied and that the peak linear heat rate (PLHR) limit is not exceeded. The most limiting conditions for the SLB are defined in the Westinghouse "Reactor Core Response to Excessive Secondary Steam Flow" topical (WCAP-9226-P-A, Revision 1). Although this topical is written for Westinghouse designed PWRs, it is applicable to plants of similar design, such as St. Lucie Unit 2, because of the basic and well understood nature of the transient. WCAP-9226 concludes that a large break in the secondary piping causes a rapid and severe drop in the primary side temperature and pressure. The analysis assumes that the most reactive Control Element Assembly (CEA) is stuck in the fully withdrawn position. In the presence of a large end-of-life moderator temperature coefficient, the rapid cooldown in the RCS causes an erosion of the shutdown margin with the potential for a return to power, especially at the location of the stuck CEA. The largest possible breaks are limiting since they cause the most significant cooldown and thus, the largest return to power. In addition, the case with full RCS flow has been demonstrated to be more limiting than the case without offsite power (AC) available for the Westinghouse fleet as stated in WCAP-9226-P-A. This conclusion is equally applicable to CE-designed plants as explained in the following sections.

II. Limiting Steamline Break Core Phenomena

As noted above, the primary criterion of interest for the post-trip SLB is demonstrating that the DNB design basis is satisfied. The DNB design basis is driven by many factors, most of which are related to the thermal-hydraulic conditions in the location of the core with the highest integrated axial power, that is, the hot channel. The conditions that are most significant to the SLB event include the core inlet temperature from both the faulted and the intact steam generators, the RCS pressure, the core power/heat flux in the hot channel, and the RCS flow in the hot channel. The hot channel will typically be in the location of the stuck CEA. The following paragraphs discuss the effects of assuming forced RCS flow (with offsite power available) versus assuming natural circulation conditions (with a loss of offsite power) in the steamline break event.

During a steamline break event, a significant drop in the RCS temperature and pressure occur for both the case with forced RCS flow (with offsite power) and for the natural circulation case (loss of offsite power). For large steamline breaks, the drop in the core inlet temperature causes a return to power and an increase in the power in the stuck rod location/hot channel. Under natural circulation conditions, as the coolant is heated in the hot channel, the flow in the hot channel increases as it picks up flow from the surrounding assemblies. The effect of the increased flow (cross flow) in the hot channel would affect the reactivity feedback in the hot channel, influence the axial power shape and affect margins to DNB at different elevations in the hot channel. Conversely, the effect of cross flow from the surrounding assemblies to the hot channel would not be significant for the case of forced RCS flow. Given all the different interdependent effects, it is not possible to conclude which condition, that is, forced RCS flow or natural circulation, would be limiting with respect to DNB. However, it is safe to conclude that one needs to consider the above discussed effects, such as cross flow from the surrounding assemblies, in order to perform a meaningful comparison of the forced RCS flow versus the natural circulation flow cases with respect to demonstrating margins to DNB.

III. Westinghouse Steamline Break Analysis Studies

The steamline break topical report, WCAP-9226 specifically examined the post-steamline break without offsite power. These studies took into account the effects that would be expected to occur during a steamline break event, and in particular, the effects of cross flow mentioned previously.

These studies involved several different plant specific analyses, including three and four loop plants, and concluded that an assumed loss of offsite power resulted in lower peaking factors and lower powers levels compared to the case with forced RCS flow. This effect, combined with an axial power shape skewed towards the bottom of the core and the lower peaking resulted in a higher absolute value of the calculated DNBR for the cases without offsite power compared to the case with forced RCS flow. These analyses substantiated the existing Westinghouse position of analyzing only the post-trip steamline break event with offsite power available.

Later in the 1990s, similar studies were repeated for the Westinghouse advanced AP-600 plant and found that the minimum DNBR values were significantly greater than the limit value of 1.45 for pressures between 500 psia and 100 psia, which would be expected for large steamline break transients. This is noteworthy in that the configuration of the AP-600 plant is very similar in design to the St. Lucie Unit 2 plant. That is, one hot leg and two cold legs per steam generator, a total of two steam generators and an overall steam generator design similar to the CE design. Similar studies (confirmatory calculations) were also performed for the St. Lucie Unit 2 plant and reached the same conclusion as presented in WCAP-9226-P-A.

Note that the different studies discussed above were performed at different times and used different code suites, yet all reached the same conclusion, that is, the steamline break with offsite power available is the limiting case for licensing basis purposes.

IV. Current Licensing Basis Steamline Break Analysis

The current licensing basis steamline break analysis demonstrates that the case without offsite power is a more limiting case compared to the case with forced RCS flow. The analysis of the case without offsite power (natural circulation flow) is very conservative because cross flow effects are not specifically modeled. In particular the event is analyzed at a high enough core flow such that cross flows can be ignored. Then partial credit is used to account for the effects of cross flow. Without modeling the effects of cross flow, the analysis will over-predict the return to power and the peaking factors associated with the stuck rod location/hot channel. The DNBR calculation is carried out with a single channel model, which does not include the increased flow, due to cross flow, towards the top of the channel. This results in a more limiting, and very conservative analysis, for the post-trip steamline break case without offsite power.

APPENDIX B
CONDITIONAL REQUIREMENTS

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This Appendix addresses the conditional requirements of NRC SERs for those topicals for which this report represents the first application of the topical methods or tools for St. Lucie Unit 2. For each conditional requirement, statements are provided describing the basis for satisfying the requirements for this application. No information is presented for those reports for which no conditional requirements exist.

WCAP-9272-P-A

1. The methodology of WCAP-9272 should not be used by a licensee other than Westinghouse.

Compliance: All methods employed in establishing the reference analyses and supporting the application of WCAP-9272-P-A are standard methods employed by Westinghouse for other plant applications. The methodology of WCAP-9272 is used by qualified Westinghouse personnel in the 30% SGTP analyses for St. Lucie 2. Any adjustment to the methods have been identified as necessary to assure proper technical modeling of the unique features of the Unit 2 plant and have been justified within this report

2. Any significant change in methods or codes used by Westinghouse must be evaluated for its impact on the reload safety evaluation methodology of WCAP-9272.

Compliance: This report provides the vehicle by which evaluation of the impact of changes in methods or codes and its impact on the reload safety evaluation methodology of WCAP-9272 can be assessed. All methods and codes employed in establishing the reference analyses are described explicitly, through reference to topical reports previously approved by the NRC, and/or other standard references (CFR, SRP, NUREG, etc.). The single exception is the application of the ABB-NV Critical Heat Flux correlation using the VIPRE-01 code. The incorporation of the ABB-NV correlation into the VIPRE-01 code has been separately documented in the topical report identified below, which is current under review by the NRC.

Sung, Y. et al., "Addendum 1 to WCAP-14565-P-A, Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," WCAP-14565-P-A, Addendum 1, May 2003.

3. Since quantitative criteria are not available for determining when an accident re-evaluation rather than a reanalysis can be performed, justification for any reevaluation should be presented in individual Reload Safety Evaluation reports.

Compliance: This report focuses on the reference analyses which will be used as the basis for future application of the WCAP-9272-P-A reload methodology. At such time as this methodology is applied to a St. Lucie Unit 2 reload, Westinghouse will include justification for any reevaluation in the Reload Safety Evaluation report for the cycle-specific application of the reload methodology, as is standard practice pursuant to this conditional requirement. It should be noted that the Reload Safety Evaluation reports are typically documented under the provisions of 10 CFR 50.59.

WCAP-10216-P-A, Rev. 1A, WCAP-10216-P-A, NS-EPR-2649

4. The approved version of WCAP-10216-P, Rev. 1 must be included in the Administrative Reporting Requirements Section of the TS for those plants incorporating the penalty factor in the COLR.

Compliance: The approved version of WCAP-10216-P, Rev. 1 has been included in the Administrative Reporting Requirements Section of the TS for St. Lucie Unit 2.

5. [Also,] TS Surveillance 4.2.2.2.e.1 must be modified to reflect inclusion of this parameter in the PFLR or COLR.

Compliance: The surveillance requirement equivalent to TS Surveillence 4.2.2.2.e.1 has been modified to reflect inclusion of this parameter in the St. Lucie Unit 2 COLR. Note that the nature of this specification uses linear heat rate (LHR(z)) rather than $F_Q(z)$ as its basis. These definitions are functionally equivalent (directly proportional based on core average linear power), and allow consistency across the Technical Specifications, and are consistent with the standard output units of the Incore Monitoring Detection System and Excore Monitoring Detection System in use at St. Lucie Unit 2.

CENPD-132, Supplement 4-P-A

6. The 1999 EM is applicable to LBLOCA licensing applications for CE designed pressurized water reactors.

Compliance: St. Lucie Unit 2 is a CE designed pressurized water reactor.

7. The use of the 1999 EM Automated Integrated Code System (AICS) without replacement of the Dougall-Rohsenow correlation for the 1985 EM simulation for licensing applications is not NRC reviewed or approved.

Compliance: This St. Lucie Unit 2 design analysis was performed using the 1999 EM AICS with Dougall-Rohsenow replaced for the 1999 EM, and was not performed for a 1985 EM simulation.

8. The 1999 EM is applicable to all CE designed pressurized water reactors with Zircaloy clad fuel.

Compliance: St. Lucie Unit 2 is a CE designed PWR. The 30% SGTP LBLOCA analysis was performed for Zircaloy and ZIRLO™ clad fuel. CENPD-404-P-A extends the applicability of the 1999 EM to ZIRLO™ cladding.

9. Each licensee that uses the 1999 EM must ensure that the choice of the RWST temperature for safety injection and containment spray provides a bounding PCT result for LBLOCA events.

Compliance: The 30% SGTP LBLOCA analysis included a RWST temperature sensitivity study that ensured that the choice of the RWST temperature for safety injection and containment spray provided a bounding PCT result.

10. The 1999 EM will continue to use the 1985 EM specified inputs for the SG secondary side initial pressure (nominal), SG secondary side initial inventory (nominal), and SG tube plugging (maximum).

Compliance: The 30% SGTP LBLOCA analysis used nominal SG secondary side initial pressure, nominal SG secondary side initial inventory and maximum SG tube plugging.

11. SI actuation in the 1999 EM calculation is based on the SIAS plus delay time.

Compliance: The SI actuation used in the 30% SGTP LBLOCA analysis is based on the SIAS plus delay time.

12. Each applicant referencing the 1999 EM topical report must perform a plant-specific, ECCS component, worst single failure assessment, including consideration of the most limiting value of the RWST temperature.

Compliance: The 30% SGTP LBLOCA analysis included a worst single failure assessment of ECCS components. The assessment included consideration of the most limiting value of the RWST temperature.

CENPD-404-P-A

LOCA

13. If the CENP LOCA methodologies and/or constituent models are changed in the future, documentation supporting the changes(s) should include justification of the continued applicability of the methodology or model for ZIRLO.

Compliance: The CENP LOCA methodologies and constituent models used in the 30% SGTP LOCA analyses did not change from the methodologies and constituent models of the evaluation models described in CENPD-404-P-A.

14. All CENP methodologies will be used only within the range for which ZIRLO™ data was acceptable and for which the verifications discussed in CENPD-404-P and responses to requests for additional information were performed.

Compliance: The 30% SGTP LOCA analyses used NRC-accepted evaluation models within their ranges of acceptability.

Fuel Performance

15. The corrosion limit as predicted by the best-estimate model will remain below 100 microns for all locations of the fuel.

Compliance: The maximum allowable corrosion limit of 100 microns will be added to the St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR). The corrosion thickness will be calculated using the best estimate models and methods described in CENPD-404-P-A on a cycle-specific basis. Contained in CENPD-404-P-A, Revision 0 is a letter from P. W. Richardson (Westinghouse) to J. S. Cushing (NRC), "Response to Requests for Additional Information on Topical Report CENPD-404-P, Rev. 0", LD-2001- 0045, Rev. 0, August 10, 2001. This letter specifically addresses the best estimate models for predicting corrosion limits.

16. All the conditions listed in the SEs for all the CENPD methodologies used for ZIRLO fuel analysis continue to be met, except that the use of ZIRLO cladding in addition to Zircaloy-4 cladding is now approved.

Compliance: All CENPD methods and codes employed in establishing the reference analyses for use of the WCAP-9272-P-A reload methodology are described explicitly and/or through reference to topical reports previously approved by the NRC.

17. All CENP methodologies will be used only within the range for which ZIRLO data was acceptable and for which the verifications discussed in CENPD-404-P and responses to requests for additional information were performed.

Compliance: ZIRLO™ data ranges for the methodologies in which they are used will be verified through FPL's and Westinghouse's Quality Assurance (QA) process that is employed for use of methodologies.

18. Until data is available demonstrating the performance of ZIRLO cladding in CENP designed plants, the fuel duty will be limited for each CENP designed plant with some provision for adequate margin to account for variations in core design (e.g., cycle length, plant operating conditions, etc). Details of this condition will be addressed on a plant specific basis during the approval to use ZIRLO in a specific plant.

Compliance: FPL will limit the fuel duty for St. Lucie Unit 2 with a provision for adequate margin to account for variations in core design (e.g., cycle length, plant operating conditions, etc). This limit will be applicable until data is available demonstrating the performance of ZIRLO™ cladding at CENP 16x16 plants.

FPL will restrict the modified Fuel Duty Index (FDIm) of each ZIRLO™ clad fuel pin to 110% of a maximum fuel pin value previously experienced. FPL proposes to use the same base maximum FDIm value (approximately 600) previously determined for APS's PVNGS plants, which employ a 16x16 fuel design similar to that used at St. Lucie Unit 2. Furthermore, since PVNGS plants have already implemented ZIRLO clad fuel, their operational experience and data collection will precede that of FPL's St. Lucie Unit 2 plant. Like PVNGS, for a fraction of the fuel pins in a limited number of assemblies (8), FPL will restrict the fuel duty of ZIRLO™ clad fuel pins to 120% of the maximum fuel pin value previously experienced at the PVNGS plants in the aggregate.

If the modified Fuel Duty Index and measured oxide thickness for CENP 16x16 fuel correlate as expected or is conservative relative to predictions, FPL would no longer restrict the FDIm except as required to meet the 100 micron oxide limit. Alternatively, if the measured oxide for CENP 16x16 fuel is significantly greater than predicted, FPL will provide justification to the NRC prior to an increase to the limits on FDIm. If the NRC lifts the FDIm condition based on sufficient accumulation of data from CENP designed plants, FPL would no longer restrict the FDIm except as required to meet the 100 micron oxide limit.

19. The burnup limit for this approval is 60 GWD/MTU.

Compliance: The maximum radial integrated rod burnup limit for ZIRLO™ clad fuel assemblies of 60 GWD/MTU has been added to the St. Lucie Unit 2 UFSAR.

CENPD-387-P-A

20. The ABB-NV and ABB-TV correlations indicate a minimum DNBR limit of 1.13 will provide a 95 percent probability with 95 percent confidence of not experiencing CHF on a rod showing the limiting value.

Compliance: The 95/95 DNBR limit of the ABB-NV correlation is not lower than the current NRC-approved limit of 1.13 for the St. Lucie 2 application.

21. The ABB-NV and ABB-TV correlations must be used in conjunction with the TORC code since the correlations were developed on the basis of the TORC and the associated TORC input specifications. The correlations may also be used in the CETOP-D code in support of reload design calculations.

Compliance: The ABB-NV correlation is used with the VIPRE code for the 30% SGTP DNBR analyses. The equivalence of the VIPRE code to TORC for DNBR calculations is described in Addendum 1 to WCAP-14565-P-A.

22. The ABB-NV and ABB-TV correlations must also be used with the ABB-CE optimized Fc shape factor to correct for non-uniform axial power shapes.

Compliance: The ABB-NV correlation in the VIPRE code is used with the ABB-CE optimized Fc shape factor to correct for non-uniform axial power shapes (WCAP-14565-P-A, Addendum 1).

23. Range of applicability for the ABB-NV and ABB-TV Correlations:

Parameter	ABB-NV Range	ABB-TV Range
Pressure (psia)	1750 to 2415	1500 to 2415
Local Mass Velocity (Mlbm/hr-ft2)	0.8 to 3.16	0.9 to 3.40

Heated Length, Inlet to CHF Location (in)	48 to 150	48 to 136.7
Grid Spacing (in)	8 to 18.86	8 to 18.86
Heated Hydraulic Diameter Ratio, Dhm/Dh	0.679 to 1.08	0.679 to 1.000

Compliance: The current range of applicability for the ABB-NV correlation remains applicable with the VIPRE code (WCAP-14565-P-A, Addendum 1).

24. The ABB-NV and ABB-TV correlation will be implemented in the reload analysis in the exact manner described in Section 7.1 of Topical Report CENPD-387-P, Revision 00-P.

Compliance: For future St. Lucie 2 reload analyses, the ABB-NV correlation will be used with the NRC-approved reload methodology in WCAP-9272-P-A and the statistical RTDP methodology in WCAP-11397-P-A, in addition to the NRC-approved topical reports listed in Section 7.1 of CENPD-387-P, Revision 00-P.

25. Technology transfer will be accomplished only through the process described in Reference 5 (of the SER) which includes ABB-CE performing an independent benchmarking calculation for comparison to the licensee generated results to verify that the new CHF correlations are properly applied for the first application by the licensee.

Compliance: At the present, there is no technology transfer of the ABB-NV correlation and DNB methodology from Westinghouse to FPL as part of the 30% SGTP project. Any future technology transfer will be accomplished through a process that meets the requirements specified in Generic Letter (GL) 83-11 Supplement 1, "Licensee Qualification for Performing Safety Analyses".

26. Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal.

Compliance: The ABB-NV correlation was used in the DNBR analyses. Justification of the ABB-NV correlation limit of 1.13 with the VIPRE code is provided in WCAP-14565-P-A, Addendum 1.

For the 30% SGTP DNBR analyses, the plant specific hot channel factors for enthalpy rise and other fuel-dependent parameters that have been previously used and approved for St. Lucie 2 have been assumed in these analyses.

27. Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE.

Compliance: The core boundary conditions for the VIPRE calculations are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE as discussed in the safety evaluation. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272/9273.

28. The NRC Staff's generic SER for VIPRE (Reference 2 of the SER) set requirements for use of new CHF correlations with VIPRE.. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification.

Compliance: Justification on use of the ABB-NV correlation with the VIPRE code is provided in Addendum 1 to WCAP-14565-P.

29. Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE (Reference 38) did not extent to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained.

Compliance: For the St. Lucie 2 30% SGTP analysis, the usage of VIPRE in the post-CHF region is limited to the peak clad temperature calculation for the seized (locked) rotor transient. The calculation demonstrated that the peak clad temperature in the reactor core is well below the allowable limit to prevent clad embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565-P and included the following conservative assumptions:

DNB was assumed to occur at the beginning of the transient;
Film boiling was calculated using the Bishop-Sandberg-Tong correlation;
The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction.

Conservative results were further ensured with the following input:

Fuel rod input based on the maximum fuel temperature at the given power;
The hot spot power factor was equal to or greater than the design linear heat rate;
Uncertainties were applied to the initial operating conditions in the limiting direction.

30. Sensitivity factors for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal.

Compliance: Sensitivity factors were evaluated using the ABB-NV DNB correlation and the VIPRE code for parameters previously considered in ESCU at the St. Lucie 2 plant specific design conditions as presented in Section 4.3 of the submittal. These sensitivity factors were used to determine the maximum Design Limit DNBR for the FSAR Chapter 15 DNB-limiting events analyzed using RTDP. The resultant Design Limit DNBR will be included in the St. Lucie 2 Unit 2 UFSAR.

31. Any changes in DNB correlation, THINC-IV correlations, or parameter values listed in Table 3-1 of WCAP-11397 outside of previously demonstrated acceptable ranges require re-evaluation of the sensitivity factors and of the use of Equation (2-3) of the topical report.

Compliance: As described in the response to Condition 1 above, sensitivity factors have been generated at the plant specific conditions using the ABB-NV DNB correlation and the VIPRE-01 code. VIPRE has been demonstrated to be equivalent to the THINC-IV code in WCAP-14565-P-A. Equation (2-3) of WCAP-11397-P-A has been evaluated to be valid for the St. Lucie 2 application.

32. If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC code, then the use of an uncertainty allowance for application of Equation (2-3) must be re-evaluated and the linearity assumption made to obtain Equation (2-17) of the topical report must be validated.

Compliance: Equation (2-3) of WCAP-11397-P-A and the linearity approximation made to obtain Equation (2-17) have been evaluated to be valid for the combination of the ABB-NV correlation and the VIPRE code for the St. Lucie 2 application. Furthermore, the sensitivity factors, operating parameters, and the VIPRE model used in this application do not differ significantly from those used in WCAP-11397-P-A.

33. Variances and distributions for input parameters must be justified on a plant-by-plant basis until generic approval is obtained.

Compliance: The plant specific variances and distributions for the RTDP input parameters are the same as those used previously in ESCU for St. Lucie 2. The RTDP parameter uncertainties are presented in Table 4-2 of the submittal.

34. Nominal initial condition assumptions apply only to DNBR analyses using RTDP. Other analyses, such as overpressure calculations, require the appropriate conservative initial condition assumptions.

Compliance: Nominal initial conditions were only applied to the DNBR analyses using RTDP.

35. Nominal conditions chosen for use in analyses should bound all permitted methods of plant operation.

Compliance: Bounding nominal conditions were used in the DNBR analyses using RTDP, consistent with the proposed methods of plant operation in support of 30% SGTP.

36. The code uncertainties specified in Table 3-1 (of WCAP-11397-p) (± 4 percent for THINC-IV and ± 1 percent for transients) must be included in the DNBR analyses using RTDP.

Compliance: The code uncertainties specified in Table 3-1 of WCAP-11397-P-A remained unchanged and were included in the DNBR analyses using RTDP.

WCAP-7588, Revision 1-A

The Westinghouse methodology for the analysis of the Rod Ejection Accident is described in the Topical Report WCAP-7588 Rev.1-A. The report was reviewed and approved by the Commission, and a Topical Report Evaluation was issued by the staff on August 2, 1973. The report describes the calculational basis using a one-dimensional spatial kinetics computer code (TWINKLE) and a transient fuel heat transfer code (FACTRAN) to determine the hot spot fuel temperature vs. time. The Regulatory evaluation concluded that the Westinghouse analysis limit of 225 cal/g peak fuel average enthalpy at the hot spot (for unirradiated fuel) and 200cal/g (for irradiated fuel) was acceptable. (Westinghouse has subsequently imposed a limit of 200 cal/g for all fuel.) The Regulatory position was that this Topical described an acceptable analysis basis for evaluating the rod ejection accident in Westinghouse pressurized water reactors, with no further requirements or restrictions. This methodology has also been applied and licensed in the analysis of the CEA Ejection event on the Millstone 2 and Fort Calhoun Unit 1 plants.

WCAP-9226

37. Only those codes which have been accepted by the NRC should be used for licensing applications.

Compliance: The RETRAN computer code has been accepted by the NRC for licensing applications.

38. For the pressure between 500 and 1000 psia, the 95/95 DNBR limit for the W-3 correlation is 1.45.

Compliance: The W-3 correlation was used and the 95/95 DNBR limit of 1.45 was applied.

EPRI NP-1850-CCM-A (RETRAN-02)

The responses contained in Reference 1 (Included in Appendix B of Reference 2), and Section 4 (Model Verification) of Reference 2 are applicable to the use of RETRAN-02 for St. Lucie Unit 2 analyses with the following exception:

In the response to Question 5, General Limitations, Item j, the heat conduction characteristics are verified through the use of either the PAD code or the FATES code as discussed in Reference 3.

1. NSD-NRC-98-5813, "Responses to Supplemental Request or Additional Information on WCAP-14882, 'RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,' (Proprietary)", December 10, 1998
2. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
3. Westinghouse RETRAN model description in Appendix C.

WCAP-14882-P-A

39. The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification.

Compliance: Only the asymmetric steam generator transient (ASGT) event is not directly listed in Table 1 of the WCAP-14882-P-A SER. The ASGT event may be caused by a multiple number of initiators, but the limiting ASGT event is identified as the sudden closure of a MSSV on one steam generator (or loss of load to one steam generator). As such, the dynamic response of the system does not introduce any phenomena that have not been seen in events currently analyzed with RETRAN, such as the Loss of Load/Turbine Trip or the Steamline Break events. The use of RETRAN to model the ASGT event is therefore deemed acceptable. The St. Lucie Unit 2 ASGT event analysis is presented in Section 5.1.11 of the licensing report.

40. WCAP-14882 describes modeling of Westinghouse designed 4-, 3-, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification.

Compliance: Appendix F of this report provides a detailed description of the RETRAN model developed to support a Combustion Engineering (CE) designed plant with an analog protection system. As noted in Appendix F, the only two changes required are the accomodation of the two-cold-legs per hot leg configuration of the CE design and the general replacement of the control and protection system processing logic. Although the CE analog plant control and protection system design is similar in functionality to the Westinghouse control and protection design, some changes were required to the system logic. The changes in logic and signal processing are reflected in the safety analyses and are addressed in Section 5 of this licensing report.

41. Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 (WCAP-9272-P-A). Licensing applications using RETRAN should include the source and justification for the input data used in the analysis.

Compliance: This report presents the basis for transitioning St. Lucie Unit 2 to a reload philosophy based on WCAP-9272-P-A. As such, the selection of conservative input used in the RETRAN models is based on approach presented in WCAP-9272-P-A and the discussion presented in WCAP-14882-P-A therefore remains applicable.

WCAP-7908-A

42. The fuel rod is divided into a number of concentric rings. The minimum number of fuel rod rings used to represent the fuel must be 6.

Compliance: This requirement is met since the analyses model at least six rings.

43. If the core average conditions such as calculated from the LOFTRAN code are used, care should be taken to ensure that the overall input to the FACTRAN code is more conservative than those dictated by the local conditions. There must be sufficient conservatism in the input value to make the overall method conservative.

Compliance: For St. Lucie, the FACTRAN code is not used with the RCS loop codes LOFTRAN or RETRAN. It is only used in conjunction with the TWINKLE neutron kinetics code to calculate the peak heat flux in the CEA Withdrawal from Subcritical or Low Power event, or the peak fuel and clad temperatures in the CEA Ejection event. For the CEA Withdrawal event, and the CEA Ejection event at hot zero power (HZP), the local conditions are the same as the average conditions since the reactor is at zero power. For the CEA Ejection at hot full power (HFP) event, the initial fuel temperatures are normalized to the hot spot values predicted by the fuel rod design code. Once DNB is initiated, the FACTRAN code uses the saturation temperature as the local temperature. For both of these events, the nuclear power transient is sufficiently brief that the core inlet conditions do not change over the time of interest.

44. The FACTRAN Code does not include a model for fuel mechanical behavior. Therefore, another code should be used to predict mechanical behavior. The FACTRAN code can be used to show compliance with a) DNBR, b) fuel melting, and c) pellet enthalpy criteria.

Compliance: The FACTRAN code is not used to model the mechanical behavior of the fuel. It is only used to show compliance with fuel melting and pellet enthalpy criteria.

APPENDIX C
THE WESTINGHOUSE RETRAN PLANT MODEL FOR COMBUSTION ENGINEERING DESIGN PWRs WITH ANALOG REACTOR PROTECTION SYSTEMS

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THE WESTINGHOUSE RETRAN PLANT MODEL
FOR COMBUSTION ENGINEERING DESIGN PWRs
WITH ANALOG REACTOR PROTECTION SYSTEMS

REFERENCES

1. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses", April, 1999.

2. EPRI NP-1850-CCMA, "RETRAN-02 – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems".

3. WCAP-10851-P-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," August 1988.

4. CENPD-139-P-A, "Fuel Evaluation Model," July 1974.
CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.
CEN-161(B)-P, Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992.
CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

5. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

6. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis" April 1997.

1 MODEL OVERVIEW

In defining an overall RETRAN plant model for a Combustion Engineering (CE) designed PWR with an analog reactor protection system (RPS), the RETRAN plant model developed for a standard Westinghouse Pressurized Water Reactor (PWR) documented in WCAP-14482 (Reference 1) was used as a starting point. The major differences between a CE-designed analog plant and a Westinghouse designed PWR are:

- the processing of the signals in the control and protection systems, and
- the configuration of the loops in the reactor coolant system (two cold legs per hot leg).

With minor exceptions, there are no fundamental differences in:

- the parameters which are monitored by the control and protection systems,
- the locations of the monitored parameters for the control and protection systems,
- control and protection system capabilities to affect the reactor coolant system (control rods, relief and safety valves, pressurizer spray and heaters, feedwater to steam generators, etc.),
- the thermal-hydraulic phenomenon that occur in the primary or secondary coolant of the reactor coolant system (RCS), or
- the neutron kinetics phenomenon which occur in the reactor core.

The exceptions to the items above are:

- The CE-designed plant flow measurement is based upon a total pressure drop across the two steam generators whereas the Reference 1 model is based upon independent loop measurements where (typically) the pressure drop measured only in the cold leg which does not involve any significant variations in volumetric flowrates between the tap points.

- The location of the taps for the pressurizer level measurement for a CE-designed plant are located in the end caps of the pressurizer whereas the Reference 1 model is based upon tap locations in the straight, cylindrical section of the pressurizer. This results in a slight variations to be addressed in the pressurizer liquid volume versus height relationship for indicated pressurizer level measurement.

As a result, with relatively minor adjustments of the base plant nodalization and a general replacement of the control and protection system processing logic, the modeling conventions applied to the standard Westinghouse RETRAN model will be applicable to a CE-designed PWR with an analog RPS. In addition, the Reference 1, Section 4.0 discussions addressing RETRAN SER (See Reference 2) limitations are applicable to the CE-designed plant model.

The following sections present the details of the Westinghouse developed RETRAN plant model for the CE-designed plant with an analog RPS, including models for the fuel rod, the core, the vessel, the RCS components, the steam generator (SG), the secondary side, the RPS, the control system, and other models. Each section presents the background for the individual models, applicable figures/diagrams and the basis for the models. Each model is implemented based on the plant specific design, data and operating characteristics.

2 CORE MODEL

2.1 Fuel Rod Model

The Fuel Rod Model is the same as developed and discussed in WCAP-14882 (Reference 1). Implementation of this Fuel Rod Model accounts for the geometric characteristics of the design of the fuel rods used in the plant. The description of the fuel rod model in Reference 1 is fully applicable with the exception that Westinghouse currently has two codes available to define the heat transfer characteristics of the fuel and cladding. Depending on the accident being analyzed, either minimum or maximum fuel-to-coolant UAs are implemented within the RETRAN model [ ] a, c

2.2 Thermal-Hydraulic Nodalization

a, c

2.3 Power Distribution

The power distribution for the core is the same as described in Reference 1. A fraction of the power produced in the core conductors is defined to be produced directly in the coolant. [ ] a, c

a, c

[ ]

a, c

2.4 Neutron Kinetics and Decay Heat

a, c

a, c

The decay heat model used in RETRAN [ ] For some analyses, decay heat is a benefit and may be conservatively removed by using a very small decay heat multiplier.

3 VESSEL MODEL

a, c

The CE vessel thermal-hydraulic nodalization is similar to the Reference 1 RETRAN model for a 4-loop reactor pressure vessel (RPV). [ ]

a, c

The choice of the vessel nodalization is [ ] the geometric, thermal hydraulic conditions, form loss coefficients, etc., are based on plant specific data.

In defining the Westinghouse RETRAN RPV model, it was necessary to consider the range of non-LOCA transients that will be analyzed and the varying phenomena occurring during these events. Of significant importance were the asymmetric events. These include asymmetric flow events, e.g., partial loss of flow, and asymmetric heat load events, e.g., steam line breaks, feedwater malfunction, and feedline break events. For asymmetric events, it is important to have a model which will adequately capture the effects of mixing (and non-mixing) that can take place within the lower and upper plenums of the RPV. In light of the above, the model shown in Figure 3-1 was developed and is consistent with the RETRAN models developed for Westinghouse designed PWRs documented in Reference 1. The characteristics in the RPV lower plenum and RPV upper plenum can be independently varied to match any desired mixing behavior from nearly-perfect mixing to virtually-zero mixing with all reactor coolant pumps in operation. The basic elements of the RPV model are discussed in detail below.

3.1 RPV Entry and Downcomer Volumes

[ ] a, c

3.2 Lower Plenum Volumes

[ ] a, c

3.3 Core and Associated Volumes

[ ] a, c

a, c

Figure 3-1. St. Lucie Unit 2 Reactor Vessel Nodalization

a, c

3.4 Upper Plenum and RPV Exit Volumes

a, c

3.5 Upper Head

a, c

4 RCS COMPONENT MODELS

4.1 RCS Piping

The model for RCS piping thermal-hydraulic nodalization is similar to the Reference 1 RETRAN model with two exceptions:

a, c

The nodalization developed for the St. Lucie Unit 2 RETRAN model is depicted in Figure 4-1.

4.2 Pressurizer

The pressurizer, as shown in Figure 4-1, is modeled as a two-region, non-equilibrium volume (volume 1). The pressurizer model includes the pressurizer heaters (both the backup and the proportional heaters), the pressurizer sprays, and the pressurizer relief (PORVs) and safety valves, which are shown in Figure 4-1. The pressurizer surge line, spray line and pressurizer relief tank (PRT) are all modeled as separate volumes. The PRT is modeled as a time dependent volume (that is, conditions in the PRT are initially defined and remain unchanged as flow is delivered to the volume) to ensure that the relief/safety valves which empty to the PRT operate under choked flow conditions. The model also includes the calculation of an indicated water level, based on the measured pressure difference over the span of the pressurizer tap locations and an indicated pressure signal which are used by the control and protection system models.

Plant specific information is used to define the physical dimensions, form loss coefficients, etc., for the pressurizer, pressurizer spray line and for the surge line. Plant specific information is also used to define heater, spray, and relief and safety valve behavior as well as reactor trip and ESFAS setpoints. The pressurizer relief valves and safety valves are discussed further in Sections 4.3 and 8.2.

a, c

Figure 4-1. St. Lucie Unit 2 General RCS Nodalization

The pressurizer proportional and backup heaters are controlled by the indicated pressurizer pressure and indicated pressurizer level signals. The indicated pressurizer level is defined by the differential pressure between the upper and lower taps. Below a defined level, the heaters are turned off, consistent with actual plant operation. Likewise, the pressurizer spray actuates based on indicated pressure. As the indicated pressure increases, the spray flow from the cold leg increases. Note that the operation of the pressurizer heaters and spray are only assumed to operate in the safety analyses if their operation results in a more limiting transient, otherwise their operation is not assumed.

A reactor trip is also included in the pressurizer model for the high pressurizer pressure trip function. Additionally, a low pressurizer pressure safety injection function is also included based on the plant specific ESFAS functions. A low pressurizer pressure reactor trip function is accommodated through a minimum pressure (floor) value modeled in the TM/LP reactor trip function. These functions are discussed further in Sections 6, 7, and 8.

4.3 Relief and Safety Valves

The pressurizer model includes two PORVs and three pressurizer safety valves [ ].[a, c] The relief characteristics for these valves are based on the rated valve relief capacity for each and the corresponding design pressure. [ ] a, c

4.4 Reactor Coolant Pumps

a, c

Each reactor coolant pump (RCP) is modeled as a single node with a corresponding set of pump performance data, applicable to all pumps. The RCPs are shown in Figure 4-1 [ ]

Depending upon the event in question, the pump data may be modified to reflect conservative accident conditions such as would occur during a locked rotor event. [ ] a, c

4.5 Steam Generators

a, c

A multi-node RETRAN model has been developed for the CE-designed feedring steam generators. [

]

a, c

The standard Westinghouse RETRAN feedring steam generator model is presented in Figure 4-2. The RETRAN steam generator model developed for CE-designed feedring steam generator is presented in Figure 4-3. [

a, c

]

a, c

Figure 4-2. Typical Westinghouse Feedring Steam Generator Nodalization

a, c

Figure 4-3. St. Lucie Unit 2 Feedring Steam Generator Nodalization

5 "BALANCE OF PLANT" MODELS

5.1 Steam and Feedwater System

The "balance of plant" models include the main piping from the steam generator outlet up to the turbine stop valves and a portion of the main feedwater piping. As shown in Figure 5-1, the steam system is divided into a separate volume for a steam line associated with each steam generator and one volume for the common steam header. Each of the main steam system safety valves are modeled individually, as discussed in Section 5.2. Main steam line isolation valves (MSIVs) are modeled on each steam line and the turbine control valves are modeled via [ ]^{a,c} flow area control on the steam header volume. The physical dimensions for the steam system piping are based on plant specific data. The main feedwater pumps are modeled as fill junctions which provide flow to the feedwater piping. The feedwater and steam flow controls are further discussed in Sections 8.3 and 8.4.

In addition to the main feedwater system, the auxiliary feedwater system is also modeled. Consistent with the modeling for the main feedwater system, the auxiliary feedwater pumps are modeled as fill junctions providing flow to the auxiliary feedwater piping. The auxiliary feedwater is typically defined as either a constant flow rate or as a function of the steam generator pressure, both before and after steam line isolation. A simple steam generator level control capability is also provided.

5.2 Main Steam System Safety Valves

Each of the main steam system safety valves is modeled along with one power operated atmospheric relief valve on each steam line. The junctions for these valves are modeled to be connected directly to the steamline. The relief characteristics for each of these valves are set to match the rated valve relief capacity at the corresponding rated pressure [ ] a, c

a, c

Figure 5-1. Main Steam System Nodalization

6 REACTOR PROTECTION SYSTEM MODELS

The Westinghouse RETRAN model of the control block logic for selected reactor protection system functions for a CE-designed plant with an analog RPS is discussed below. [ ] In addition, the electronic delays from the time a setpoint is reached until the rods begin falling into the core have been implemented in the CE-designed RETRAN plant model. a, c

6.1 Power Calculation

Consistent with the CE analog reactor protection system design, two calculations of reactor power are made:

1) Excore Neutron Detector Power (Excore Power)

The excore detectors monitor the fast neutron flux leakage from the reactor core. Since the fast neutron generation is directly related to the nuclear power of the core, this forms the beginning point for the excore power signal. Two modifications to this signal are then included.

- Downcomer Temperature Shadowing: The fast neutron leakage from the core to the detectors can be attenuated by the water in the reactor pressure vessel downcomer. Increasing the density of this water results in a greater attenuation of the neutrons reaching the detector. [ ] a, c

- Control Rod Shadowing: Control rods at the periphery of the core and nearest the excore neutron detectors, can distort, or shadow, the amount of fast neutron leakage from the core to the detector which can affect the calibration of the excore detectors to the actual core power. [ ] a, c

2) Thermal Power

The hot leg and cold leg temperature measurements are also used to calculate the core power level. In the CE analog reactor protection system the calculation follows the following form:

a, c

where:

a, c

[

]

The calculated excore power and thermal power are then auctioneered to determine the power level used for the TM/LP and Variable High Power reactor trip functions.

6.2 Thermal Margin/Low Pressure (TM/LP) Reactor Trip Function

The TM/LP reactor trip function is based on the form of the equation presented in the plant specific Technical Specifications:

TM/LP:

$$P_{var}^{trip} = \mathrm{Max}[1400 \times Q_{DNB} + 17.85 \times Tin - 9410, \mathrm{Floor}]$$

$$Q_{DNB} = A_1 \times QR_1$$

Where

P_{var}^{trip} = TM/LP reactor trip setpoint (psia) (compared to measured pressurizer pressure)
A_1 = Axial Shape Index (ASI) penalty function
QR_1 = Measured power conversion function
Floor = Minimum allowed pressure for this reactor trip function

a, c

[

]

6.3 Variable High Power Reactor Trip Functions

The variable high power reactor trip function is included in the CE-designed plant RETRAN model. a, c

[

]

Where,

[ ] a, c

6.4 Pressurizer Reactor Trip Functions

Reactor trip based upon high pressurizer is modeled. The [ ]a, c reactor trip function provides a low pressurizer pressure trip.

6.5 RCS Flow Related Reactor Trip Function

The CE-designed plant RETRAN model includes the capability to model the low RCS flow reactor trip function. For the analog protection system, the RCS flow signal is actually the sum of the pressure differential (delta-P) measurements across the primary side of the two steam generators and the trip setpoint is input [ ] a, c

6.6 Steam Generator Level Trip Functions

[ ] a, c

[ ] a, c

6.7 Turbine Trip/Manual Reactor Trips

A turbine trip signal will result in a reactor trip signal. In addition, a "manual" reactor trip is defined to allow the user to simulate a reactor trip at a specified time.

6.8 Rate of change of Power Reactor Trip Function

The High Rate-of-Change (ROC) of Power Reactor trip is modeled. The ROC Reactor trip function can be manually bypassed above a specified low power limit and below a very low power limit. [ ] a, c

The ROC measurement is modeled as the logarithm (base 10) of the indicated nuclear power. The transfer function for the dynamic compensation is:

a, c

6.9 Asymmetric Steam Generator Steam Pressure Reactor Trip Function

In the plant, the calculation of the variable TM/LP setpoint includes the Asymmetric Steam Generator Trip function (ASGT). If a high steam generator steam pressure difference exists between the two steam generators, the ASGT trip function forces a high TM/LP setpoint and thus a TM/LP trip. [ ][a, c]

6.10 High Local Power Density (LPD) Reactor Trip

The High Local Power Density (LPD) Reactor trip [ ] This trip is automatically bypassed below a specified minimum power level; the bypass resets above the specified limit. a, c

6.11 Low SG Pressure Reactor Trip

A reactor trip signal results from a low steam pressure for either steam generator.

7 ENGINEERED SAFETY FEATURES SYSTEM MODELS

The CE-designed plant RETRAN model of the control block logic and nodalization for the major engineered safety features systems are discussed below.

7.1 Safety Injection Signals

Two safety injection signals are defined which are based upon low pressurizer pressure and manual actuation signals.

a, c

7.2 Low Steam Pressure Signal

a, c

7.3 Safety Injection System

The safety injection (SI) system model is divided into three subsystems: SI pump train, "loop-specific" SI piping, and accumulators. These are modeled the same as in Reference 1.

a, c

The following describes the model development of the first two subsystems.

SI Pump Train Model

The purpose of the SI pump train model is to determine the flow generated by each independent SI pump train and the transport delay of cold borated refueling water tank (RWT) water reaching the loop-specific SI flow distribution point, if applicable. Figure 7-1 provides the RETRAN nodalization for the SI pump train model. [ ] a, c

Loop-Specific SI Purge Volume Model

a, c

a, c

Figure 7-1. RETRAN Nodalization of the St. Lucie Unit 2 Safety Injection System

7.4 Safety Injection Tanks

The safety injection tank (SIT) model is based on the available geometric data and the experience gained by benchmarking [ ] a, c

7.5 Low Steam Generator Level ESFAS Functions

The Low Steam Generator Level Reactor trip signal will also initiate the start of the auxiliary feedwater flow system, including any plant specific delays.

7.6 Turbine Trip Functions

A reactor trip or high steam generator level signal will initiate a turbine trip.

7.7 Auxiliary Feedwater System (AFW)

The definition of the AFW flow and enthalpy provides a boundary condition to the Westinghouse RETRAN model. Due to the wide variety of modeling assumptions that are used in safety analyses, only a "base" model will be described here. This "base" model provides a reasonable model which can be adjusted or modified (e.g., varying the forcing functions on AFW flow rates) easily. The following describes the "base" model.

a, c

a, c

7.8 Manual Actuations

Manual actuations are defined for the following actions:

- Reactor trip
- Turbine Trip
- Steamline Isolation
- Feedline Isolation
- Safety Injection
- Auxiliary Feedwater Start
- Turbine Trip
- RCP Trip (each loop RCP trip individually defined)

8 CONTROL SYSTEM MODELS

The RETRAN model simulates several control systems using the available control block logic. In general, the control block logic is similar to the models described in Reference 1 and in some cases, the models are unchanged from Reference 1. Each of the major control systems is discussed below.

8.1 Rod Control

a, c

a, c

8.2 Pressurizer Pressure Control

The pressurizer pressure control system maintains the reactor coolant system pressure at a predetermined value (nominal pressure) by employing heaters, spray and power-operated relief valves (PORVs). The heaters increase the pressurizer water temperature and increase pressure by boiling water. Spray, located at the top of the pressurizer operates to condense steam and decrease pressure.

Pressurizer pressure control is accomplished in the Westinghouse RETRAN model through the simulation of spray and heaters.

The operational characteristics of the heaters are discussed below.

a, c

a, c

a, c

a, c

8.3 Feedwater Flow Control

Typically, automatic feedwater control is not used in safety analyses since normal operation of the feedwater control system would tend to mitigate an event or provide non-bounding variations of a transient. As a result, safety analyses will typically define a bounding forcing function for the main feedwater flow. However, an automatic feedwater control has been included in the design of the CE-designed plant RETRAN model to provide additional capabilities in the areas of better-estimate transients, transition of initial conditions, and sensitivity studies. [ ]

a, c

The automatic feedwater control modeling includes the following features.

a, c

8.4 Turbine Control

The turbine is modeled as a single junction between the steam system steam header node and a time-dependent volume which is modeled at atmospheric conditions. The turbine load is controlled through adjustments of the junction (valve) flow area. The following capabilities are provided to control turbine load during the transient:

a, c

a, c

8.5 Pressurizer Level Control

The pressurizer level control model includes a simplified charging/letdown control capability. This provides a capability to evaluate the net effect of inventory addition/removal from the control system operation. It uses a single fill junction to the pressurizer loop cold leg to add or remove inventory from the reactor coolant system. The following controls are implemented:

a, c

8.6 High Steam Generator Level Functions

A high steam generator level signal is actuated when the indicated level signal, discussed in Section 3.8.2, is above the defined setpoint. Any high steam generator level actuation will initiate feedwater isolation (for all steam generators) and a turbine trip. The RETRAN model includes the capability to account for plant specific delays for both of these functions.

8.7 DNBR Model

a, c

The RETRAN model includes the capability to calculate a conservative approximation of the DNBR during a transient. [

]

The RETRAN model is used with the subchannel code (e.g., VIPRE code - see Reference 6) for calculation of DNBR for flow reduction and significant asymmetric power distribution events mentioned above.