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BGE

January 31, 1997

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
License Amendment Request; Change to Reactor Coolant System Flow Requirements to Allow Increased Steam Generator Tube Plugging

REFERENCE: (a) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated December 4, 1996, License Amendment Request; Conversion of the Calvert Cliffs Units 1 and 2 Technical Specifications to the Improved Standard Technical Specifications, NUREG-1432

Pursuant to 10 CFR 50.90, the Baltimore Gas and Electric Company hereby requests an amendment to Operating License Nos. DPR-53 and DPR-69 by incorporating the changes described below into the Technical Specifications and Updated Final Safety Analysis Report for Calvert Cliffs Unit Nos. 1 and 2. The proposed changes are necessary to support a larger number of plugged steam generator (SG) tubes for future operating cycles.

The proposed Technical Specification changes are to reduce the minimum Reactor Coolant System (RCS) total flow rate from 370,000 gpm to 340,000 gpm; reduce the Reactor Protective Instrumentation trip setpoint for Reactor Coolant Flow - Low from ≥ 95% to ≥ 92% of design reactor coolant flow; adjust the reactor core thermal margin safety limit lines to reflect the reduced RCS flow rate; and reduce the lift setting range for the eight Main Steam Safety Valves (MSSVs) with the highest allowable lift setting from the current range of 935 to 1065 psig to a more restrictive range of 935 to 1050 psig. In addition to the changes to the Technical Specifications necessary to support an increased number of plugged SG tubes, reanalysis of the accident analyses affected by this change identified an Unreviewed Safety Question (USQ) associated with these changes. The USQ results from the determination that the Main Steam Line Break (MSLB) Event and Seized Rotor Event analyses involve an increased percentage of failed fuel cladding. Finally, three reanalyzed events (MSLB, Loss of Coolant Flow, and Boron Dilution) will require Nuclear Regulatory Commission (NRC) approval due to changes to the methodology or assumptions used to analyze these events.

A001/1

BACKGROUND

Pressurized water reactor SGs have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, stress corrosion cracking, and crevice corrosion, along with other phenomena, such as denting and vibration wear. Tubes that experience excessive degradation reduce the integrity of the primary-to-secondary pressure boundary. Eddy current examination is used to measure the extent of tube degradation. When the reduction in tube wall thickness reaches the plugging or repair limit, as specified in the Technical Specifications, the tube is considered defective and a corrective action is taken.

Currently, the Calvert Cliffs Technical Specifications allow defective tubes to be plugged and removed from service, or to be repaired using a laser-welded sleeving technique developed by Westinghouse Electric Corporation. The most widely used tube maintenance technique at many pressurized water reactors, including Calvert Cliffs, is removal of the degraded tube from service by installing plugs at both ends of the tube. The installation of SG tube plugs removes the heat transfer surface of the plugged tube from service, and the increased flow resistance leads to a reduction in the primary coolant flow available for core cooling. The reduced heat transfer surface and RCS flow rate, in turn, result in reduced SG steam pressure. Another direct result of the reduced RCS flow rate is an increase in the temperature rise (delta T) between the core inlet (cold leg) and core outlet (hot leg). Cold leg temperature will remain constant, below the current Technical Specification limit of 548°F. Therefore, while maintaining the core power level as close to 100% rated thermal power as possible, hot leg temperature will increase.

The accident analyses for Calvert Cliffs Units 1 and 2 currently assume a maximum of 800 tubes plugged in each of the units' two SGs. The proposed amendment will revise the appropriate Technical Specifications and their Bases to account for the effects of plugging up to 2500 tubes in each SG. Attachment (1) summarizes the analyses and evaluations affected by the increased number of plugged SG tubes. In accordance with 10 CFR 50.71(e), the next regularly scheduled update of the Updated Final Safety Analysis Report following approval of these changes will include changes necessary to support plugging up to 2500 tubes in each SG.

REQUESTED TECHNICAL SPECIFICATION CHANGES

The proposed amendment will revise the appropriate Technical Specifications to permit operation of Calvert Cliffs Nuclear Power Plant Units 1 and 2 with a reduction in the RCS flow rate resulting from an increased number of plugged SG tubes. A summary description of the proposed Technical Specification changes follows. Attachment (1) provides a more detailed evaluation of these changes. Attachment (2) provides the Determination of No Significant Hazards Consideration. The marked-up current Technical Specification pages in Attachments (3) and (4) (for Units 1 and 2, respectively) depict the proposed changes. Additionally, the marked-up Improved Technical Specifications (ITS) pages in Attachment (5) show the effect of the requested changes on the proposed ITS submitted in Reference (a).

Technical Specification 2.1.1 - SAFETY LIMITS (ITS 2.1.1-1)

The thermal power limit lines in Figure 2.1-1, Reactor Core Thermal Margin Safety Limit, will be revised to reflect the reduced RCS flow rate. The lines represent the loci of points of thermal power and reactor coolant temperature and pressure for which the Departure from Nucleate Boiling Ratio (DNBR)

is no less than the DNBR limit for the axial shapes and peaking factors less than or equal to those shown in Technical Specification Figure B2.1-1, Axial Power Distribution for Thermal Margin Safety Limits. The minimum DNBR limit for steady-state operation, normal operational transients, and anticipated transients remains unchanged. This value assures with at least a 95 percent probability at a 95 percent confidence level that Departure from Nucleate Boiling will not occur.

Technical Specification 2.2 - LIMITING SAFETY SYSTEM SETTINGS (ITS 3.3.1)

In Table 2.2-1, Reactor Protective Instrumentation Trip Setpoint Limits (ITS Table 3.3.1-1, Reactor Protective System Instrumentation), the trip setpoint for Reactor Coolant Flow - Low will be changed from "≥ 95% of design reactor coolant flow" to "≥ 92% of design reactor coolant flow." This change is necessary to increase the operating margin such that the design reactor coolant flow may be achieved, while minimizing the potential for spurious low flow pre-trip alarms during normal plant operation and unnecessary plant trips during abnormal electrical system frequency reductions. These frequency reductions are within the operating guidelines established for the Pennsylvania-New Jersey-Maryland Interconnection.

Technical Specification 3.2.5 - POWER DISTRIBUTION LIMITS; DNB Parameters (ITS Limiting Condition for Operation 3.4.1.c)

The RCS total flow rate will be changed from ≥370,000 gpm to ≥340,000 gpm. This Technical Specification ensures adequate RCS flow is provided to remove core heat during normal operation, Anticipated Operational Occurrences, and Postulated Accidents. The installation of SG tube plugs leads to a reduction in the primary coolant flow available for core cooling. The minimum allowable RCS total flow rate must be reduced to 340,000 gpm to support plugging up to 2500 tubes per SG.

Technical Specification 4.7.1.1 - TURBINE CYCLE; Safety Valves (No effect on ITS)

In Table 4.7-1, Steam Line Safety Valves Per Loop, the highest allowable lift settings for eight MSSVs will be changed from 935 - 1065 psig to 935 - 1050 psig. Specifically, this will require a change to the lift settings identified for valves RV-3996/4004, RV-3997/4005, RV-3998/4006, and RV-3999/4007, and to the note on the bottom of the page specifying the setpoints for the eight valves with the highest lift settings. This Technical Specification ensures the MSSV lift settings are set so as to prevent steam pressure from exceeding the SG pressure upset limit. For the reanalysis of the Loss of Load and Loss of Feedwater Events, it was necessary to credit a more restrictive lift setting range for the highest set valves.

The change to the MSSV settings will not affect the ITS, as the ITS submittal proposes relocating this information from the Technical Specifications to the Inservice Testing Program. The Inservice Testing Program will include the revised lift settings.

TECHNICAL SPECIFICATION BASES CHANGES

The following changes to the Technical Specification Bases are necessary to support the proposed Technical Specifications changes. These changes were evaluated in accordance with the requirements of 10 CFR 50.59, and were determined not to involve an Unreviewed Safety Question.

Technical Specification Bases 3/4.7.1.4 - ACTIVITY (ITS B.3.7.14 - Secondary Specific Activity)

Currently, the Bases state that the limitations on Secondary System-specific activity ensure that the resultant off-site radiation dose will be limited to "a small fraction of" 10 CFR Part 100 limits in the event of a steam line rupture, including the effects of a coincident "1.0 GPM" primary-to-secondary tube leak. The analysis for the MSLB supporting this activity credits the current Technical Specification 3/4.4.6.2 primary-to-secondary leakage limit of 100 gallons per day through any one steam generator, so this Bases statement will be changed from 1.0 GPM to 100 gallons per day. In accordance with the Calvert Cliffs Updated Final Safety Analysis Report, the acceptance criteria for this Postulated Accident is that the site boundary doses do not exceed the 10 CFR Part 100 guidelines. Therefore, it is appropriate to change this Bases statement to reflect the correct acceptance criteria.

UNREVIEWED SAFETY QUESTION

During the performance of the analyses and evaluations supporting this change, it was concluded that implementation of the reduced RCS flow in two of these analyses involves a USQ. A brief description of the USQ follows. Attachment (1) provides a detailed presentation of the affected accident analyses.

Increased Fuel Pin Failure Results for the MSLB and Seized Rotor Event Analyses

The results of the MSLB analysis with reduced RCS flow show that the percentage of fuel pins that could fail increases from less than 1% to no more than 10%. The reactor coolant pump Seized Rotor Event analysis with reduced RCS flow also indicates that the resultant percentage of fuel pins that could fail increases from no more than 3% to no more than 5%. The increased number of fuel pin failures is considered an increase in the probability of malfunction of equipment important to safety (i.e., the fuel cladding), so the reduction in RCS flow was determined to involve a USQ. Therefore, NRC review and approval of the reduced RCS flow and these accident analyses results is required.

METHODOLOGY CHANGES

The proposed amendment resulted in the reanalysis of a number of Updated Final Safety Analysis Report Chapter 14 accident analyses. Three of these analyses involved the use of a different methodology, or different assumptions, than those used previously in the NRC-approved transient or accident analysis. A brief description of each of these methodology changes follows. Attachment (1) evaluates the revised methodologies in greater detail.

Loss of Coolant Flow Analysis - Use of HERMITE Methodology

This change involves the use of the current Asea Brown Boveri methodology (HERMITE) for analysis of the transient core power, heat flux, and hot bundle heat flux for the Loss of Coolant Flow Event (Updated Final Safety Analysis Report Section 14.9). Calvert Cliffs' current Loss of Coolant Flow analysis uses the CESEC and STRIKIN computer codes for the transient core response. The HERMITE methodology has been approved by the NRC for application at other nuclear plants for the Loss of Coolant Flow analysis, but has not previously been applied to Calvert Cliffs' Loss of Coolant Flow analysis.

MSLB - Incorporation of a Time Delay Between Reactor Trip and Resultant Loss of Offsite Power

This methodology change involves incorporating an assumption into the analysis that a three-second time delay exists between the reactor trip and the subsequent coastdown of the reactor coolant pumps due

to the resultant loss of offsite power. Credit for this time delay is necessary to demonstrate acceptable results for the MSLB analysis with reduced RCS flow. The occurrence of a time delay is based on the stable nature of the offsite electrical grid. Such a time delay has been approved in accident analyses at other nuclear plants, but has not been credited previously in Calvert Cliffs' MSLB analysis.

Boron Dilution Event - Methodology Change

This change involves the analysis of the Mode 2 Boron Dilution Event using the acceptance criterion applicable to Modes when the reactor is critical, as opposed to that when the reactor is shutdown. The currently used acceptance criterion compares the calculated minimum time to lose shutdown margin against an acceptance criterion of 15 minutes. However, in Mode 2, when the reactor is critical, shutdown margin is provided by the control element assemblies, not the boron concentration, so this acceptance criterion is not appropriate for Mode 2. Therefore, application of the acceptance criterion applicable to Modes when the reactor is critical, which compares the boron dilution reactivity addition rate to the reactivity addition rate analyzed in another critical transient (Control Element Assembly Withdrawal Event) is proposed. Nuclear Regulatory Commission approval of the proposed Mode 2 methodology is required.

SIGNIFICANT HAZARDS CONSIDERATIONS / ENVIRONMENTAL ASSESSMENT

We have evaluated significant hazards considerations associated with this change and have determined that there are none [see Attachment (2) for a complete discussion]. We have also determined that operation with the proposed amendment would not result in any significant change in the types, or significant increases in the amounts, of any effluents that may be released offsite, nor would it result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental effect statement or environmental assessment is needed in connection with the approval of the proposed amendment.

The Plant Operations and Safety Review Committee and the Offsite Safety Review Committee have reviewed the proposed changes to the Technical Specifications and the USQ associated with the supporting analyses for these Technical Specification changes, and concurred that operation with the proposed changes will not result in an undue risk to the health and safety of the public.

SCHEDULE

Baltimore Gas and Electric Company anticipates the possibility of exceeding the maximum assumed number of plugged SG tubes (800 per SG) during the next Unit 2 refueling outage, which is scheduled to begin on March 14, 1997, and end on April 30, 1997. If we do exceed 800 plugged tubes on either Unit 2 SG during the 1997 Refueling Outage, it will be necessary to implement this license amendment prior to start-up from that outage. Therefore, we request this change be approved on or before April 15, 1997.

The lift settings for the MSSVs must be reduced for each unit in order for the revised Loss of Load and Loss of Feedwater analyses to apply. The MSSV setpoint change is planned for Unit 2 during the 1997 Unit 2 Refueling Outage; however, this work cannot be performed on Unit 1 until the 1998 Unit 1 Refueling Outage. To facilitate implementation of the requested Technical Specifications changes, we plan to implement the changes to both units' Technical Specifications at the same time. Therefore, the

Unit 1 Technical Specifications will be annotated to indicate that the changes associated with this license amendment request do not apply until after the current Unit 1 operating cycle (Cycle 13). This change will also allow implementation into the common Improved Technical Specifications, which is currently undergoing NRC review.

Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

Charles H. Cruse

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I hereby certify that on the 31 day of January, 1997, before me, the subscriber, a Notary Public of the State of Maryland in and for Calvert County, personally appeared Charles H. Cruse, being duly sworn, and states that he is Vice President of the Baltimore Gas and Electric Company, a corporation of the State of Maryland; that he provides the foregoing response for the purposes therein set forth; that the statements made are true and correct to the best of his knowledge, information, and belief; and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal: Denise Snukis
Notary Public

My Commission Expires: 2/2/98
Date

CHC/NH/dlm

Attachments:
(1) Description and Evaluation of Changes in Support of Increased Steam Generator Tube Plugging Limit
(2) Determination of Significant Hazards
(3) Unit 1 Marked-Up Technical Specification Pages
(4) Unit 2 Marked-Up Technical Specification Pages
(5) Marked-Up Improved Technical Specification Pages

cc:
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J. E. Silberg, Esquire
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ATTACHMENT (1)

DESCRIPTION AND EVALUATION OF CHANGES IN SUPPORT OF INCREASED STEAM GENERATOR TUBE PLUGGING LIMIT

ATTACHMENT (1)

DESCRIPTION AND EVALUATION OF CHANGES IN SUPPORT OF INCREASED STEAM GENERATOR TUBE PLUGGING LIMIT

I. INTRODUCTION

This evaluation supports the proposed Technical Specification changes, Unreviewed Safety Question (USQ), and methodology changes required to allow plugging of up to 2500 tubes per steam generator (SG) for Calvert Cliffs Units 1 and 2. The primary effects of plugging SG tubes are to reduce SG heat transfer area and increase the Reactor Coolant System (RCS) flow resistance. These effects, in turn, result in reduced SG steam pressure, reduced RCS flow rate, and increased core outlet (hot leg) temperature. This evaluation considers the effects of these changes in plant conditions on the fuel and core, on the Design Basis Events (DBEs) evaluated in Updated Final Safety Analysis Report (UFSAR) Chapter 14, on corrosion of Alloy 600 components in the RCS, and on normal operation of the plant. In particular, this evaluation forms the basis for the determination that there is no significant hazards consideration with regard to the proposed Technical Specification changes and the USQ. This evaluation also establishes the basis for the acceptability of using different methodologies and assumptions than those currently approved for use in the Calvert Cliffs safety analyses.

II. SUMMARY OF CHANGES AND EVALUATIONS

In order to plug up to 2500 tubes per SG at Calvert Cliffs, approval of Technical Specification changes, revised accident analysis methodologies, and a USQ are required. This section summarizes the required Technical Specification and methodology changes, the USQ, the supporting analyses and evaluations, and the conclusions that support the determination that the changes to the Technical Specifications and the USQ do not involve a significant hazards consideration.

Required Technical Specification Changes

Technical Specification Figure 2.1-1, Reactor Core Thermal Margin Safety Limit:

Revise thermal limit lines to reflect reduced RCS flow rate.

The thermal limit lines in Technical Specification Figure 2.1-1, Reactor Core Thermal Margin Safety Limit, will be revised to reflect the reduced RCS flow rate. The marked-up Technical Specification pages indicate the changes to the endpoint values only; the thermal limit lines will be relocated to correspond with the revised endpoint values. The lines represent the loci of points of thermal power and reactor coolant temperature and pressure for which the departure from nucleate boiling ratio (DNBR) is no less than the DNBR limit for the axial shapes and peaking factors less than or equal to those shown in Technical Specification Figure B2.1-1, Axial Power Distribution for Thermal Margin Safety Limits. The minimum DNBR limit for steady-state operation, normal operational transients, and anticipated transients remains unchanged. This assures with at least a 95 percent probability at a 95 percent confidence level that Departure from Nucleate Boiling (DNB) will not occur.

For Unit 1 only, a note will be added to this Technical Specification indicating that the existing reactor core thermal margin safety limits (renamed Figure 2.1-1a) shall apply through Cycle 13. During the refueling outage following Cycle 13, the Main Steam Safety Valve (MSSV) lift setting changes affecting Technical Specification Table 4.7-1 will be implemented when the applicable Modes are entered, and the revised thermal limit lines will apply.

Technical Specification 2.2.1, Reactor Coolant Flow:

Change Reactor Coolant Flow - Low setpoint from ≥ 95% to ≥ 92%.

This setpoint provides core protection during events involving a loss or reduction of RCS flow. The current setpoint is ≥ 95% of the design reactor coolant flow, per Technical Specification 2.2.1. The current design reactor coolant flow is 370,000 gpm per Technical Specification 3/4.2.5. Therefore, this setpoint is equivalent to a fixed RCS flow rate of ≥ 351,500 gpm.

The Reactor Protective System Reactor Coolant Flow signal normally oscillates within approximately ± 3% of the steady-state flow. Operating margin must be maintained above the Reactor Coolant Flow-Low reactor trip setpoint (≥ 95% of design flow) and the associated pre-trip alarm (≥ 97% of design flow) to account for these oscillations. The current RCS flow rate is approximately 103% of design flow, so the 3% signal oscillations do not cause pre-trip alarm actuations. However, as a greater number of SG tubes are plugged, the reactor coolant flow rate will be closer to the design coolant flow rate. With the Reactor Coolant Flow - Low pre-trip alarm setpoint at ≥ 97% and the reactor coolant flow at the design value, the operating margin will not be adequate to prevent spurious low flow pre-trip alarms during normal plant operation. In addition, the 5% margin between the design coolant flow rate and the trip setpoint will not be adequate to prevent an unnecessary plant trip during an abnormal electrical grid frequency reduction, due to the reduced reactor coolant pump (RCP) speed during such a reduction. Therefore, in order to increase the operating margin such that operation at, or near, the design reactor coolant flow is achievable, it will be necessary to reduce the Reactor Coolant Flow - Low setpoint to ≥ 92% in Technical Specification 2.2.1. (Note: A corresponding 3% reduction in the pre-trip alarm setpoint to 94% of design flow will also be considered outside of this license amendment request.)

Analyses supporting this change were performed for a range of flow rates such that this change is justified for the current design flow (370,000 gpm), and for the reduced RCS flow discussed below (340,000 gpm).

As discussed below, for Unit 1 only, this Technical Specification will be annotated to indicate that the existing limit of ≥ 95% of design reactor coolant flow applies through Unit 1, Cycle 13.

Technical Specification 3.2.5.c, Power Distribution Limits - DNB Parameters:

Change RCS total flow rate from 370,000 gpm to 340,000 gpm.

This Technical Specification ensures adequate RCS flow is provided to remove core heat during normal operation, transients, and accidents. As discussed above, plugging SG tubes reduces the RCS flow rate. In order to support plugging up to 2500 tubes per SG, this Technical Specification must be changed from ≥ 370,000 gpm to ≥ 340,000 gpm.

As discussed below, for Unit 1 only, this Technical Specification will be annotated to indicate that the existing limit of ≥ 370,000 gpm applies through Unit 1, Cycle 13.

Technical Specification Table 4.7-1, Steam Line Safety Valves Per Loop:

Change the MSSV maximum lift settings for the eight valves with the highest pressure setpoint from 1065 psig to 1050 psig.

This Technical Specification ensures the MSSV lift settings are set so as to prevent steam pressure from exceeding the SG pressure upset limit. For the reanalysis of the Loss of Load and Loss of Feedwater Events discussed below, it is necessary to credit a more restrictive lift setting range for eight valves with the highest lift setting: RV-3996/4004, RV-3997/4005, RV-3998/4006, and RV-3999/4007. Therefore, for these valves, the allowable lift setting range must be reduced to 935-1050 psig from 935-1065 psig.

The MSSV setpoint change is planned for Unit 2 during the 1997 Unit 2 Refueling Outage; however, this work cannot be performed on Unit 1 until the 1998 Unit 1 Refueling Outage. Therefore, the Unit 1 Technical Specifications will be annotated to indicate that the revised Technical Specifications do not apply until after the current operating cycle, Cycle 13. Similar notes will be incorporated into the common ITS pages to indicate that, for Unit 1 only, the existing specification values will apply through Cycle 13.

Technical Specifications Bases Changes

The following changes to the Technical Specification Bases are necessary to support the proposed Technical Specifications changes. These changes were evaluated in accordance with the requirements of 10 CFR 50.59, and were determined not to involve an Unreviewed Safety Question.

Technical Specification Bases 3/4.7.1.4 - Activity (ITS B.3.7.14 - Secondary Specific Activity)

Currently, the Bases state that the limitations on Secondary System-specific activity ensure that the resultant off-site radiation dose will be limited to "a small fraction of" 10 CFR Part 100 limits in the event of a steam line rupture, including the effects of a coincident "1.0 GPM" primary-to-secondary tube leak. The analysis for the MSLB supporting this activity credits the current Technical Specification 3/4.4.6.2 primary-to-secondary leakage limit of 100 gallons per day through any one steam generator, so this Bases statement will be changed from 1.0 GPM to 100 gallons per day. In accordance with the Calvert Cliffs Updated Final Safety Analysis Report, the acceptance criteria for this Postulated Accident is that the site boundary doses do not exceed the 10 CFR Part 100 guidelines. Therefore, it is appropriate to change this Bases statement to reflect the correct acceptance criteria.

Unreviewed Safety Question

The proposed amendment required the reanalysis of a number of UFSAR Chapter 14 transient and accident analyses. It was concluded that the reduction of RCS flow to 340,000 gpm involves a USQ based on the results of the reanalysis of two of these events, the Main Steam Line Break (MSLB) Event and the Seized Rotor Event. However, the results of all of the reanalyzed accidents are within the acceptance criteria established by the NRC. A brief description of the USQ follows. The revised analyses are discussed in greater detail in Sections IV and VII.

Increased Fuel Pin Failures for MSLB and RCP Seized Rotor Analyses

The results of the MSLB analysis with reduced RCS flow show that the percentage of fuel pins that could fail has increased from less than 1% to no more than 10%. The results of the RCP Seized Rotor Event analysis with reduced RCS flow also show that the percentage of fuel pins that could fail has increased from no more than 3% to no more than 5%. The increased number of fuel pin failures for these events results in a USQ, as the probability of malfunction of equipment important to safety (i.e., the fuel cladding) has increased. Therefore, NRC approval of the RCS flow reduction and results of the Calvert Cliffs MSLB and Seized Rotor analyses is required.

Methodology Changes

Three of the revised analyses involved the use of either a different methodology or a different assumption than those used previously in the current NRC-approved transient or accident analysis. A brief description of each of these methodology changes follows. The methodology changes are discussed in greater detail in Sections IV and VII.

Loss of Coolant Flow Analysis - Use of HERMITE Methodology

This change involves the use of the current Asea Brown Boveri methodology (HERMITE) for analysis of the transient core power, heat flux, and hot bundle heat flux for the Loss of Coolant Flow Event (UFSAR, Section 14.9). Calvert Cliffs' current Loss of Coolant Flow analysis uses the CESEC and STRIKIN computer codes for the transient core response. Use of the HERMITE methodology provides additional margin to the Loss of Coolant Flow analysis with the reduced Reactor Coolant Flow - Low setpoint and reduced initial RCS flow. The HERMITE methodology (Reference 1) has been approved by the NRC for application in the Loss of Coolant Flow analyses for other nuclear plants, but has not previously been applied to Calvert Cliffs' Loss of Coolant Flow analysis; therefore, NRC approval for use of this methodology for the Calvert Cliffs Loss of Coolant Flow analysis is requested. The results of the revised Loss of Coolant Flow analysis using the HERMITE methodology meet the NRC acceptance criteria.

MSLB - Incorporation of Time Delay Between Reactor Trip and Resultant Loss of Offsite Power

This methodology change involves incorporating an assumption into the analysis that a time delay exists between the reactor trip and the subsequent coastdown of the RCPs due to the resultant loss of offsite power (LOOP). The occurrence of a time delay is based on the stable nature of the Baltimore Gas and Electric Company (BGE) grid, but this time delay has not been credited previously in Calvert Cliffs' pre-trip MSLB analysis. Credit for the time delay is necessary to demonstrate acceptable results for the pre-trip MSLB analysis with reduced RCS flow. Because the introduction of a time delay is not assumed in the current MSLB analysis, NRC approval of this analysis is requested. Application of a three-second time delay was justified for Combustion Engineering System 80 plants in Reference (2). The application of this time delay for the RCP Shaft Seizure Event received NRC approval in Reference (3). Baltimore Gas and Electric Company has concluded that the justification provided in Reference (2) is applicable to Calvert Cliffs' MSLB analysis, and that the time delay assumed in Calvert Cliffs' MSLB analysis is bounded by that approved by the NRC for CE System 80 plants in Reference (3).

Boron Dilution Event - Methodology Change

This change involves the analysis of the Mode 2 Boron Dilution Event using the acceptance criterion applicable to Modes when the reactor is critical, as opposed to that when the reactor is shutdown. The currently used acceptance criterion compares the calculated minimum time to lose shutdown margin against an acceptance criterion of 15 minutes. However, in Mode 2, when the reactor is critical, shutdown margin is provided by the control element assemblies (CEAs), not the boron concentration, so this acceptance criterion is not appropriate for Mode 2. Therefore, application of the acceptance criterion applicable to Modes when the reactor is critical, which compares the boron dilution reactivity addition rate to the reactivity addition rate analyzed in another critical transient (CEA Withdrawal Event) is proposed. Nuclear Regulatory Commission approval of the proposed Mode 2 methodology is required.

Supporting Analyses and Evaluations

The following analyses and evaluations have been performed to support the proposed Technical Specification changes required to allow plugging up to 2500 tubes per SG.

Fuel and Core Related Analyses:

The following fuel and core related analyses were performed in order to verify that the fuel and core performance will remain within acceptable limits and will be bounded by the assumptions for fuel performance made in the DBE analyses discussed below:

FUEL AND CORE ANALYSES	PURPOSE
Fuel Mechanical Design Evaluation	Verify that the dimensional effects of the new flow and temperature conditions on the fuel assembly are within acceptable limits.
Physics Evaluation	Verify core reactivity and power distribution parameters used to characterize expected operation and used to bound performance in the safety analyses are appropriate.
Fuel Performance Analysis	Verify internal fuel pin pressure and power-to-centerline melt limits are not exceeded, and that fuel rod conditions assumed in the transient analyses remain applicable.
Thermal Hydraulic Analysis	Verify thermal/hydraulic (TORC/CETOP) models used in transient and setpoint analyses remain bounding for the new flow/temperature condition. The input to the Loss-of-Coolant Accident (LOCA) analyses was revised, as necessary.
Fuel Rod Corrosion Evaluation	Verify fuel clad corrosion rates are acceptable at the new, higher temperature condition.

UFSAR Chapter 14, Design Basis Events:

The following is a listing of the UFSAR Chapter 14 DBEs and whether they required reanalysis or evaluation for this effort. Re-analysis of an event involves quantitative calculations using NRC-approved methodology, whereas evaluation involves qualitative arguments to demonstrate that NRC acceptance criteria for the event are met.

TABLE 1

UFSAR SECTION	DBE DESCRIPTION	ANALYSIS EFFORT
14.2	CEA Withdrawal Event	Reanalyzed
14.3	Boron Dilution Event (Modes 1-4)	Reanalyzed
	Boron Dilution Event (Modes 5 and 6)	Reanalysis not required
14.4	Excess Load Event	Reanalyzed
14.5	Loss of Load Event	Reanalyzed
14.6	Loss of Feedwater Flow (LOFW) Event (Overpressure)	Reanalyzed
	LOFW Event (SG Dry-Out)	Reanalysis not required
14.7	Excess Feedwater Heat Removal Event	Evaluated
14.8	RCS Depressurization Event	Reanalyzed
14.9	Loss of Coolant Flow Event	Reanalyzed
14.10	Loss of Non-Emergency AC Power Event	Evaluated
14.11	CEA Drop Event	Evaluated
14.12	Asymmetric SG Event	Reanalyzed
14.13	CEA Ejection Event	Evaluated
14.14	MSLB - Post-Trip	Reanalyzed
	MSLB - Pre-Trip	Reanalyzed
14.15	Steam Generator Tube Rupture (SGTR) Event	Evaluated
14.16	Seized Rotor Event	Reanalyzed
14.17	LOCA	Reanalyzed
	Small Break LOCA	Reanalyzed
14.18	Fuel Handling Incident	Reanalysis not required
14.19	Turbine-Generator Overspeed Incident	Reanalysis not required
14.20	Containment Response - LOCA	Reanalysis not required
	Containment Response - MSLB	Reanalysis not required
14.21	Hydrogen Accumulation in Containment	Reanalysis not required
14.22	Waste Gas Incident	Reanalysis not required
14.23	Waste Evaporator Incident	Reanalysis not required
14.24	Maximum Hypothetical Accident	Reanalysis not required
14.25	Excessive Charging Event	Reanalysis not required
14.26	Feedline Break Event	Reanalyzed

Setpoint Analyses:

To determine the thermal margin with respect to DNB criterion for several of the analyses and evaluations above, a new setpoint analysis was also performed. This analysis establishes the required Thermal Margin/Low Pressure (TM/LP) setpoints and DNB Limiting Conditions for Operation (LCOs).

RCS Alloy 600 Corrosion Evaluation:

An evaluation of the effect of increased core outlet temperature on Alloy 600 cracking in SG tubes and various RCS pressure boundary penetrations was performed.

RCS Structural Evaluation:

An evaluation of the effect of increased core outlet temperature on stress analyses for the RCS was performed.

Operational Evaluation:

An evaluation of the effect of reduced RCS flow, increased RCS average temperature, and reduced SG pressure on normal plant operation, including the effect on (non-safety-related) control systems (i.e., feedwater, pressurizer level, and pressurizer pressure) and the RCPs was performed.

Conclusions

The following conclusions were reached by the above reanalyses and evaluations supporting plugging up to 2500 tubes per SG:

➢ Fuel and core performance remains within acceptable limits, and is bounded by the inputs to the safety and setpoint analyses.

➢ The results of Postulated Accident analyses continue to meet NRC acceptance criteria. Offsite dose is within 10 CFR Part 100 guidelines and core geometry remains coolable. Loss-of-Coolant Accident results meet the acceptance criteria stipulated in 10 CFR 50.46(b).

➢ The results of analyses of Anticipated Operational Occurrences continue to meet NRC acceptance criteria. Fuel parameters do not exceed the specified acceptable fuel design limits (SAFDLs) and site boundary dose is a small fraction of 10 CFR Part 100 guidelines. Primary and secondary pressure remain below the pressure upset limits for the RCS and the SGs, 110% of the respective design pressures.

➢ The safety significance of primary water stress corrosion cracking of Alloy 600 RCS penetrations will be unchanged by the slightly elevated core exit temperature resulting from an RCS flow reduction. Primary water stress corrosion cracking cracks of RCS penetrations are expected to leak before breaking, thereby allowing them to be detected by periodic visual inspections mandated by Calvert Cliffs' Inservice Inspection Program. The SG Tube Inspection Program will continue to find and repair tube defects prior to these defects threatening tube structural integrity. The probability of a SGTR is not increased due to the elevated core exit temperature.

- The stress analyses performed for the reactor vessel and piping remain bounding for the slightly elevated core exit temperature.

- The performance of non-safety-related control systems (i.e., feedwater, pressurizer level, and pressurizer pressure) and the RCPs remains adequate to maintain RCS and SG parameters within appropriate operating limits with periodic adjustment.

III. FUEL AND CORE RELATED ANALYSES

The following analyses and evaluations were performed to verify fuel and core performance is acceptable given the reduced RCS flow rate and increased core exit temperature caused by plugging 2500 tubes per SG.

Fuel Mechanical Design Evaluation

Re-evaluation of the fuel mechanical design criteria was necessary to consider the conditions of reduced flow and increased coolant temperatures resulting from plugging additional SG tubes. The fuel mechanical design criteria which were affected by the RCS minimum flow reduction and coolant temperature increase were evaluated for the standard fuel and for the lead fuel assemblies for both Unit 1 and Unit 2. The cladding collapse evaluation required that the limiting transient conditions for collapse be evaluated. The newly revised Loss of Load Event and Loss of Feedwater Event provided these limiting RCS pressures to the cladding collapse calculation in the mechanical design evaluation. The evaluation determined that all of the fuel mechanical design criteria were met under the conditions of reduced flow and increased coolant temperatures resulting from plugging up to 2500 tubes per SG.

Physics Evaluation

The reduced RCS flow condition results in a slightly higher core exit temperature and average coolant temperature. The effect of this increase in coolant temperature on two sets of physics data (i.e., best estimate data characterizing the expected operation of the plant, and data used in the safety analyses) was evaluated. Explicit depletion calculations were performed for the higher temperature condition. The results of these calculations, when compared to the current data, showed that all physics parameters of interest were very close and would cause only minimal changes in core operating characteristics.

Following the explicit calculations noted above, an evaluation was performed to determine the potential effects of the higher coolant temperature on the physics data used in the safety analyses. This evaluation relied on the extensive database of calculated physics data versus generic data used in safety analyses. The evaluation determined that, for all parameters, sufficient margin was available to conclude that the current physics data used in the safety analyses remained valid for the reduced flow condition.

Fuel Performance Analysis

An evaluation was performed to determine whether the existing Fuel Performance Analysis remained valid for the reduced flow and higher coolant temperature conditions. This evaluation considered the three slightly different fuel assembly types now in Calvert Cliffs cores. Explicit analyses were performed for the two predominant fuel assembly types, and an evaluation was performed for the third, less predominant fuel assembly type. Data from the explicit analyses was used in the LOCA analysis.

The conclusions of this evaluation were that the maximum fuel pin internal pressure remained below the no-clad-lift-off pressure criterion, and that power-to-centerline fuel melt limits were not exceeded.

Thermal Hydraulic Analysis

The increase in coolant temperature and reduced RCS flow necessitated verification of the applicability of the thermal-hydraulic models to the new flow range. The analysis provided verification for use of TORC and CETOP models for use in the transient and setpoint analyses for DNB calculations. The Statistical Combination of Uncertainties input to the DNB LCO setpoint analysis was also verified. The new analysis produced updated hydraulic data for use in the LOCA analysis. The existing HRISE model used in the MSLB analysis DNB calculation, fuel assembly and fuel rod analyses, and the existing incore instrument and CEA heating analyses were determined to be valid. It was also determined that the maximum tube plugging asymmetry of 750 plugged tubes would not affect the limiting core inlet flow distribution.

Fuel Rod Corrosion Evaluation

An evaluation of the effect of reduced RCS flow and higher coolant temperature on fuel rod corrosion was performed. Fuel rod corrosion was predicted using the optimized low-tin corrosion model for the highest burnup fuel rods from a representative Calvert Cliffs core. Although the results of the evaluation indicate the maximum oxide thickness increases slightly, all of the predicted maximum oxide thicknesses remain well below the acceptable value of 120 microns.

IV. UFSAR CHAPTER 14, DESIGN BASIS EVENTS

This section summarizes the results and conclusions of the evaluations and analyses performed to address the UFSAR Chapter 14 DBEs. Each of the DBEs described in UFSAR Chapter 14 is discussed briefly below. Some of the DBEs are not affected by plugging SG tubes and decreasing RCS flow; for these events, no additional evaluation or analyses were performed. Some of the DBEs are not significantly affected by the change; for these events, evaluations were performed to verify that NRC acceptance criteria for the event are not exceeded. Some of the DBEs are significantly affected by the change; for these events, analyses were performed to verify NRC acceptance criteria were not exceeded. A tabulation of the evaluations and analyses performed for the UFSAR Chapter 14 DBEs is provided in Section II, Table 1.

These analyses and evaluations also considered an increase in the maximum full power core inlet temperature from 548°F to 550°F. Where conservative, the higher temperature was used in the analyses and evaluations discussed below. Therefore, these analyses and evaluations are bounding with respect to the current limitation on maximum core inlet temperature. No change to the current maximum allowed core inlet temperature specified in the Technical Specifications is requested as part of this submittal.

In addition to the summaries provided below, a detailed presentation of the analyses performed for six DBEs is provided in Section VII. A detailed presentation of these DBEs is provided since the analyses for these events involve a USQ (MSLB and Seized Rotor Events), a methodology change (Boron Dilution, Loss of Coolant Flow, and MSLB), or credit for a more restrictive Technical Specification for the lift settings of the MSSVs [Loss of Load and LOFW (overpressure) Events].

UFSAR Section 14.2, Control Element Assembly Withdrawal Event

The CEA Withdrawal Event involves withdrawal of a group of CEAs which causes an increase in core power and an approach to the fuel SAFDLs. In addition, since the RCS temperature increase during the transient is significant, this event also involves an increase in RCS pressure. Separate analyses are performed for the full-power, zero-power, and over-pressure cases. Plugging SG tubes reduces the heat transfer to the secondary system during this transient, which exacerbates the RCS temperature increase. Since the most positive moderator temperature coefficient (MTC) is assumed for the transient, the power increase is also made worse. Therefore, plugging SG tubes has a significant effect on this event, and new analyses were performed for the full-power, zero-power, and over-pressure cases to support this effort. The NRC acceptance criteria for this Anticipated Operational Occurrence are that fuel failure does not occur, and the RCS pressure upset limit (2750 psia) is not exceeded.

Analyses for this event were performed using the CESEC and CETOP computer codes, the current NRC-approved methodology for Calvert Cliffs. The analyses were performed for the reduced flow condition, with increased tube plugging, and increased coolant temperature. The analyses assumed an MTC of $+0.15 \times 10^{-4}$ $\Delta\rho/°F$, consistent with Reference (4). The results of the full-power case demonstrate the fuel SAFDLs for linear heat rate (LHR) and DNB are not exceeded. The results for the zero-power case demonstrate that although the LHR SAFDL is exceeded, the fuel centerline temperature melt limit is not exceeded, and the fuel will not fail. The results for the over-pressure case demonstrate the RCS pressure upset limit is not exceeded.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the CEA Withdrawal Event remain within the NRC acceptance criterion that fuel failure does not occur and the RCS pressure upset limit (2750 psia) is not exceeded.

UFSAR Section 14.3, Boron Dilution Event

The Boron Dilution Event involves dilution of the coolant in the RCS with a maximum flow of unborated water (0 ppm boron) from the Chemical and Volume Control System. Separate analyses for this event are performed for operating Modes 1 through 6. Plugging SG tubes reduces the RCS coolant volume for the Modes 1 through 4 condition, but does not affect the coolant volume for the Modes 5 and 6 condition. A smaller initial RCS coolant volume exacerbates the dilution affect of the pure water. Therefore, plugging SG tubes has a significant effect on the Modes 1 through 4 analyses, but the Modes 5 and 6 analyses are not affected. New analyses were performed for Modes 1 through 4 to address this change. This event is an Anticipated Operational Occurrence. The NRC acceptance criterion for Boron Dilution Events when the reactor is critical is that fuel failure does not occur. The NRC acceptance criterion for Boron Dilution Events when the reactor is shutdown is that the required shutdown margin is maintained for a given time frame (30 minutes for Mode 6, 15 minutes for other Shutdown Modes).

Analyses for the Mode 1 through 4 Boron Dilution Events were performed for an RCS coolant volume which corresponds to 2500 plugged tubes per SG. A hand calculation methodology, which is the same as the method currently approved by the NRC for Calvert Cliffs, was used. However, the method of analyzing the Mode 2 event has been changed such that it is now analyzed with the reactor critical, as opposed to assuming the reactor is shutdown, as previously analyzed. This is appropriate as the reactor is critical in Mode 2, and shutdown margin is provided by the CEAs, not the boron concentration. This

change is considered a methodology change that requires NRC review. Therefore, a detailed presentation of the Mode 1 and 2 Boron Dilution Events analyses is presented in Section VII for NRC review.

The results for the Mode 1 and 2 cases demonstrate that the maximum rate of reactivity addition is an order of magnitude slower than the worst case CEA Withdrawal Event. Therefore, the results of the Mode 1 and 2 Boron Dilution Events are bounded by the results of the CEA Withdrawal Event, which shows that fuel failure does not occur. The results for the Mode 3 and 4 Boron Dilution Event cases continue to demonstrate that at least 15 minutes is available before shutdown margin is lost.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the Boron Dilution Event remain within the NRC acceptance criteria that fuel failure does not occur when the reactor is critical, and that shutdown margin is maintained for the prescribed time frame when the reactor is shutdown.

UFSAR Section 14.4, Excess Load Event

The Excess Load Event involves an increase in steam demand caused by opening the atmospheric dump and turbine bypass valves, or fully opening the turbine control valves. The increase in steam demand due to the valve repositioning causes an increase in core power and an approach to the fuel SAFDLs. Separate analyses are performed for the full-power and zero-power conditions. Plugging SG tubes reduces the rate of heat transfer to the SG secondary side, which moderates the effect of this transient. However, since an increase in the maximum full-power core inlet temperature was also considered, a new analysis was performed to address the combined effects of plugging SG tubes and increasing coolant inlet temperature. The NRC acceptance criterion for this Anticipated Operational Occurrence is that fuel failure does not occur.

Analyses of the Excess Load Event were performed using the CESEC and CETOP computer codes, the current NRC-approved methodology for Calvert Cliffs. The analyses considered the reduced flow condition, with increased tube plugging, and increased coolant temperature. Analyses were performed for both the full-power and zero-power cases. The results of both cases demonstrate that the fuel SAFDLs are not exceeded; therefore, fuel failure does not occur.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the Excess Load Event remain within the NRC acceptance criterion that fuel failure does not occur.

UFSAR Section 14.5, Loss of Load Event

The Loss of Load Event involves a rapid reduction of steam demand due to closure of the turbine stop valves. This rapid steam demand reduction causes the RCS temperature and pressure to increase and approach to the RCS pressure upset limit (2750 psia). The SG pressure approaches the SG pressure upset limit, as well. Due to the increase in RCS pressure during the transient, this event is relatively benign in terms of approaching DNB. Plugging SG tubes has the effect of reducing the heat transfer rate to the SG secondary side. This effect exacerbates the RCS temperature and pressure increase. Therefore, plugging SG tubes has a significant effect on the Loss of Load Event and a new analysis of this event has been

performed. The NRC acceptance criteria for this Anticipated Operational Occurrence are that the RCS and SG pressure upset limits are not exceeded.

Analyses of the Loss of Load Event were performed using the CESEC computer code, the current NRC-approved methodology for Calvert Cliffs. The analyses were performed for two cases: one case maximizes RCS pressure, and the other case maximizes SG pressure. Both analyses considered the reduced flow condition, with increased tube plugging, and increased coolant temperature. The analyses assumed an MTC of $+0.15 \times 10^{-4}$ $\Delta\rho$/°F, consistent with Reference (4). In addition, the analyses assumed a reduced allowable range of lift settings for the MSSVs with the highest pressure lift settings (935 psig to 1050 psig). The reduced safety valve lift settings assumed in the analyses necessitated a corresponding change to the MSSVs allowable lift settings in Technical Specification Table 4.7-1. A detailed analysis of the Loss of Load Event is presented in Section VII, since a reduced MSSV lift setting has been assumed in the analysis. The results of the analyses demonstrate that the RCS and SG pressure upset limits are not exceeded.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the Loss of Load Event remain within the NRC acceptance criteria that the RCS and SG pressure upset limits are not exceeded.

UFSAR Section 14.6, Loss of Feedwater Flow Event

The LOFW Event involves a rapid reduction of the feedwater flow to the SGs (i.e., due to closure of the feedwater regulating valves) without a corresponding reduction of the steam demand. The reduction of feedwater flow causes the SG water inventory to decrease, ultimately causing a reactor trip on low SG water level. During the transient, RCS and SG pressure increase and approach the corresponding pressure upset limits due to degraded heat transfer to the secondary system. This event is analyzed to ensure peak RCS and SG pressures do not exceed the RCS and SG pressure upset limits, respectively. The event is also analyzed to ensure a complete loss of SG water inventory, i.e., "dryout," does not occur after the reactor trip, before the Auxiliary Feedwater System is able to restore SG inventory. Plugging SG tubes has the effect of reducing the heat transfer rate to the SG secondary side. This effect will exacerbate the RCS temperature and pressure increase. Therefore, a new analysis of the maximum RCS and SG pressure case was performed. However, the analysis to maximize SG inventory depletion is primarily dependent on two parameters; available SG inventory at the SG low level trip setpoint, and core decay heat. Since SG tube plugging has no effect on these parameters, no evaluation or analysis of the maximum inventory depletion case was performed. The NRC acceptance criteria for this Anticipated Operational Occurrence are that the RCS and SG pressure upset limits are not exceeded, and that the SGs do not dry out.

Analysis of the LOFW Event was performed with the CESEC computer code, the current NRC-approved methodology for Calvert Cliffs. The analysis considered the reduced flow condition, with increased tube plugging, and increased coolant temperature. The analysis assumed an MTC of $+0.15 \times 10^{-4}$ $\Delta\rho$/°F, consistent with Reference (4). In addition, the analysis credited a reduced allowable range of lift settings for MSSVs with the highest pressure setpoint (935 psig to 1050 psig). The MSSV lift setting change necessitated a corresponding change to the MSSVs allowable lift settings in Technical Specification Table 4.7-1. The results of this analysis demonstrate that the RCS and SG pressure upset limits are not exceeded.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the LOFW Event remain within the NRC acceptance criteria that the RCS and SG pressure upset limits are not exceeded, and that the SGs do not dry out.

UFSAR Section 14.7, Excess Feedwater Heat Removal Event

The Excess Feedwater Heat Removal Event involves a reduction in SG feedwater temperature without a corresponding reduction in steam flow from the SG. The most limiting Excess Feedwater Heat Removal Event involves a simultaneous loss of both high-pressure feedwater heaters. The reduction in SG feedwater temperature causes a decrease in RCS temperature and an increase in core power due to positive reactivity feedback. The resultant RCS temperature increase offsets the effect of the colder feedwater, and a new steady-state condition is reached without a reactor trip. The thermal margin degradation for this event is less severe than for the Excess Load Event. Plugging SG tubes reduces the rate of heat transfer to the SG secondary side, which has a beneficial effect on this transient. However, increasing the maximum allowed full-power core inlet temperature will tend to degrade the thermal margin. The combined effect of the two changes will not significantly affect the Excess Feedwater Heat Removal Event. Therefore, an evaluation of this event was performed for the new set of operational conditions. The NRC acceptance criterion for this Anticipated Operational Occurrence is that fuel failure does not occur. The evaluation concluded that more than adequate thermal margin is maintained by the DNB LCO to assure that the fuel SAFDLs will not be exceeded during the transient, and fuel failure will not occur. Therefore, no new analysis was performed.

Therefore, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the evaluation indicates that the results of the Excess Feedwater Heat Removal Event analysis remain within the NRC acceptance criterion that fuel failure does not occur.

UFSAR Section 14.8, Reactor Coolant System Depressurization Event

The RCS Depressurization Event involves a rapid decrease in RCS pressure without a loss of coolant liquid inventory. The most limiting event is caused by simultaneous opening of two power-operated relief valves (PORVs). The decrease in RCS pressure causes an approach to the DNB SAFDL. Plugging SG tubes reduces the RCS flow rate, which reduces the available margin to DNB. Therefore, plugging SG tubes may significantly affect this event, and a new analysis of the event was performed. The NRC acceptance criterion for this Anticipated Operational Occurrence is that fuel failure does not occur.

Analysis of the RCS Depressurization Event was performed with the CESEC and CETOP computer codes, the current NRC-approved methodology for Calvert Cliffs. The analysis was performed for the reduced flow condition, with increased tube plugging, and increased coolant temperature. The results of the analysis demonstrate that the DNB SAFDL is not exceeded and that fuel failure does not occur.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the RCS Depressurization Event remain within the NRC acceptance criterion that fuel failure does not occur.

UFSAR Section 14.9, Loss of Coolant Flow Event

The Loss of Coolant Flow Event involves a loss of forced coolant flow through the core with offsite power available. The most limiting Loss of Coolant Flow Event includes a concurrent loss of power to all four RCPs. Plugging SG tubes increases the RCS flow resistance and potentially increases the rate of RCS flow reduction after the loss of power to the RCPs. Plugging SG tubes also reduces the initial RCS flow rate, which reduces the available margin to DNB. Therefore, plugging SG tubes may significantly affect this event and a new analysis of the event was performed. The NRC acceptance criterion for this Anticipated Operational Occurrence is that fuel failure does not occur.

Analysis of the Loss of Coolant Flow Event was performed using the HERMITE code to calculate the transient core power, heat flux, and hot bundle heat flux, and the CETOP code to calculate the minimum DNBR. The RCS flow reduction is based on measured data for a SG with essentially no tubes plugged, adjusted for tube plugging using the CC code. The overall plant transient response analysis using the CESEC code was not performed to recalculate the peak RCS pressure, since this event is relatively benign in this regard, and is bounded by the reanalysis of the Loss of Load Event for peak RCS pressure. All of the above codes are NRC-approved. The HERMITE code was previously used at Calvert Cliffs to calculate a reactivity credit in the post-trip MSLB analysis, and to generate axial power distributions for the LCO setpoint analysis. However, the HERMITE code has not been used previously for the Loss of Coolant Flow analysis at Calvert Cliffs. Therefore, NRC approval for use of the HERMITE code for Calvert Cliffs Loss of Coolant Flow analysis is required. As a result, the Loss of Coolant Flow analysis is presented in detail in Section VII. To provide additional operating margin to the Reactor Coolant Flow - Low reactor trip, this trip setpoint was lowered to 90% in the analysis. The Loss of Coolant Flow analysis is the basis for changing the Technical Specification 2.2.1 Reactor Coolant Flow - Low reactor trip setpoint to ≥ 92%. The 2% margin is incorporated in the analysis Reactor Coolant Flow - Low reactor trip setpoint to account for instrument uncertainty.

The analysis for the Loss of Coolant Flow Event was performed for a range of coolant flows (from the maximum allowed to the minimum required flow) with increased tube plugging, and increased coolant temperature. The results of the analysis demonstrate that fuel failure does not occur.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the Loss of Coolant Flow Event remain within the NRC acceptance criterion that fuel failure does not occur.

UFSAR Section 14.10, Loss of Non-Emergency AC Power Event

The Loss of Non-Emergency AC Power Event involves loss of the plant's 500 kV/13 kV service transformers. This loss of power causes a coastdown of all four RCPs. Thus, the initial portion of this transient is essentially identical to the Loss of Coolant Flow Event discussed above. In fact, the analysis for the Loss of Coolant Flow Event demonstrates that no fuel failure will occur during the Loss of Non-Emergency AC Power Event.

The Loss of Non-Emergency AC Power Event is analyzed to demonstrate that the RCS pressure upset limit and the site boundary dose criteria in the 10 CFR Part 100 guidelines are met. Plugging SG tubes has the effect of decreasing the heat transfer rate to the secondary system. Although this should have the beneficial effect of reducing secondary system steam releases and offsite dose, there is the potential to

slightly increase the peak RCS pressure during the transient. Since the Loss of Non-Emergency AC Power Event is not as severe, with respect to the peak RCS pressure, as the Loss of Load Event, it was judged that the effect of plugging tubes on the Loss of Non-Emergency AC Power Event is not significant. Therefore, an evaluation of this event was performed. The NRC acceptance criteria for this Anticipated Operational Occurrence are that the RCS pressure upset limit is not exceeded, and the site boundary dose shall not exceed a small fraction of the 10 CFR Part 100 guidelines.

The evaluation of the Loss of Non-Emergency AC Power Event concluded that the margin to the RCS pressure upset limit for this event is more than adequate to compensate for the slight increase in peak RCS pressure due to plugging SG tubes. Therefore, the RCS pressure upset limit will not be exceeded for this event. The evaluation also concluded that the current limiting offsite dose calculation for this event is not affected by plugging SG tubes. Therefore, the site boundary dose does not exceed a small fraction of the 10 CFR Part 100 guidelines.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the evaluation indicates that the results of the Loss of Non-Emergency AC Power Event analysis remain within the NRC acceptance criteria that the RCS pressure upset limit is not exceeded, and the site boundary dose does not exceed a small fraction of the 10 CFR Part 100 guidelines.

UFSAR Section 14.11, Control Element Assembly Drop Event

The CEA Drop Event involves an uncontrolled insertion of a CEA at full power. The CEA insertion results in an increase in radial power peaking in the core, and a cooling of the RCS (at end of cycle) as the plant returns to equilibrium. The RCS cooling also causes a decrease in RCS pressure. This transient causes an approach to the DNB and LHR SAFDLs. Steam generator tube plugging does not affect the bounding core physics parameters assumed in the current limiting analysis for the CEA Drop Event, so the increase in radial peaking and the decrease in RCS pressure for the limiting CEA Drop Event are not affected by SG tube plugging. Therefore, it was judged that the effect SG tube plugging has on the limiting analysis for CEA Drop Event is not significant, and an evaluation of the CEA Drop Event was performed. The NRC acceptance criteria for this Anticipated Operational Occurrence are that fuel failure does not occur, and that the site boundary dose does not exceed a small fraction of the 10 CFR Part 100 guidelines.

The evaluation of the CEA Drop Event was performed for the reduced flow condition, with increased tube plugging, and increased coolant temperature. The evaluation concluded that the margin to the DNB and LHR SAFDLs, as determined in the current limiting analysis for the CEA Drop Event, is adequate to ensure the DNB and LHR SAFDLs will not be exceeded for the new conditions. Therefore, no fuel failure will occur during a CEA Drop Event. The evaluation also concluded that the current limiting offsite dose calculation for this event is not affected by plugging SG tubes. Therefore, the site boundary dose does not exceed a small fraction of the 10 CFR Part 100 guidelines.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the evaluation indicates that the results of the CEA Drop Event analysis remain within the NRC acceptance criteria that fuel failure does not occur, and the site boundary dose does not exceed a small fraction of the 10 CFR Part 100 guidelines.

UFSAR Section 14.12, Asymmetric Steam Generator Event

The most limiting Asymmetric SG Event for Calvert Cliffs involves a loss of load to one SG. This event is characterized by a non-uniform core inlet temperature distribution caused by the unbalanced load on the two SGs. The non-uniform temperature distribution in turn causes an increase in local power peaking and an approach to the DNB and LHR SAFDLs. Plugging SG tubes reduces the RCS flow rate and the margin to DNB. In addition, asymmetric tube plugging may affect the core inlet temperature distribution. As SG tube plugging may significantly affect this event, a new analysis of the event was performed. The NRC acceptance criteria for this Anticipated Operational Occurrence are that fuel failure does not occur, and that the site boundary dose does not exceed a small fraction of the 10 CFR Part 100 guidelines.

Analysis of the Asymmetric SG Event was performed using the CESEC and CETOP computer codes, the current NRC-approved methodology for Calvert Cliffs. The analysis was performed for the reduced flow condition, with increased tube plugging, increased coolant temperature, and a maximum tube plugging asymmetry of 750 tubes (to be controlled by the reload design control process). The results of the analysis demonstrate that the DNB and LHR SAFDLs are not exceeded; therefore, no fuel failure occurs.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the Asymmetric SG Event remain within the NRC acceptance criteria that fuel failure does not occur, and the site boundary dose does not exceed a small fraction of the 10 CFR Part 100 guidelines.

UFSAR Section 14.13, Control Element Assembly Ejection Event

The CEA Ejection Event involves a rapid ejection of the highest-worth CEA, combined with a rapid RCS depressurization caused by failure of the control rod drive mechanism pressure boundary. As a consequence, this event involves a rapid reactivity insertion and power excursion which can cause an approach to the fuel SAFDLs. For this event, analyses are performed to demonstrate that energy deposited in the fuel pellets is acceptable. This analysis is, therefore, only influenced by changes that affect the initial stored energy in the pellets and the rate of energy deposition during the transient. Steam generator tube plugging does not affect the bounding core physics parameters that determine the rate of energy deposition during the transient. Steam generator tube plugging reduces RCS flow rate and increases core exit temperature, which has a slightly adverse effect on the initial stored energy in the fuel pellet. It was judged that the slight increase in initial stored energy of the fuel pellet will not significantly affect the CEA Ejection Event. Therefore, an evaluation of the event was performed. The NRC acceptance criteria for this Postulated Accident are that the site boundary dose shall not exceed the 10 CFR Part 100 guidelines and that the core geometry remains coolable.

The evaluation of the CEA Ejection Event was performed to address the effect of SG tube plugging which causes reduced RCS flow and elevated core exit temperature. The evaluation concluded that the margin to the limit on the total average enthalpy of the hottest fuel pellet was adequate to accommodate the slight increase in coolant core exit temperature. Therefore, fuel failure does not occur, and site boundary dose does not exceed 10 CFR Part 100 guidelines.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the evaluation indicates that the results of the CEA Ejection Event analysis remain within the NRC

acceptance criteria that the site boundary dose does not exceed the 10 CFR Part 100 guidelines, and that the core geometry remains coolable.

UFSAR Section 14.14, Main Steam Line Break (Core Response) Event

The MSLB Event is a Postulated Accident involving the rupture of a main steam line. The steam released through the break causes a decrease in core inlet temperature which, assuming a negative MTC, causes core power to increase. The increase in core power may cause fuel to exceed the SAFDLs and fail. For large breaks, a significant RCS temperature reduction continues after the reactor trip. This temperature reduction adds additional positive reactivity, which challenges the negative reactivity inserted by the control rods. The result is that the core approaches criticality and a return to power may occur. Plugging SG tubes reduces the RCS flow rate and the margin to DNB. As a result, the initial core power increase may be significantly affected; therefore, the pre-trip MSLB was reanalyzed. Plugging SG tubes reduces the heat transfer rate to the SG secondary side, which should be beneficial with regard to the RCS temperature reduction transient after the reactor trip. However, other changes to inputs and assumptions to the post-trip analysis were required so a new analysis for the post-trip event was also performed. The NRC acceptance criteria for this Postulated Accident are that the site boundary dose shall not exceed the 10 CFR Part 100 guidelines and that the core geometry remains coolable.

Analysis of the pre-trip MSLB was performed using the CESEC and CETOP computer codes, the current NRC-approved methodology for this event. The analysis was performed for the reduced flow condition, with increased tube plugging, and increased coolant temperature. The MSLB event was analyzed for cases with and without a LOOP. For the pre-trip analysis case involving a LOOP, a three-second time delay between the reactor trip and RCP coastdown was assumed, based on the stability of the BGE grid. This time delay was considered a change to the NRC-approved methodology for the pre-trip MSLB analysis. Therefore, NRC approval of the pre-trip MSLB analysis is required.

In addition, the results of the pre-trip MSLB analysis with reduced RCS flow demonstrate that the percentage of fuel pins that may fail increases from that previously reviewed by the NRC. Although the site boundary dose does not exceed 10 CFR Part 100 guidelines and the core geometry remains coolable, the increase in fuel pin failures results in a USQ. A USQ results because the probability of malfunction of equipment important to safety (i.e., the fuel pin cladding) increases. Because the pre-trip MSLB analysis involves a USQ and a change to the NRC-approved methodology, it is presented in detail in Section VII for NRC review and approval.

The current NRC-approved methodology for Calvert Cliffs, the CESEC and HRISE computer codes, were used for the reanalysis of the post-trip MSLB. The analysis was performed for the reduced flow condition, with increased tube plugging, and increased coolant temperature. In addition, main feedwater isolation valve and auxiliary feedwater isolation valve leakage were conservatively added to the analysis assumptions, and the boron concentration of the Chemical and Volume Control System fluid delivered to the RCS was reduced to support plant operation. These changes, however, were evaluated in accordance with 10 CFR 50.59, and it was determined that they do not involve a USQ. Therefore, the post-trip MSLB analysis is not presented in detail in Section VII. The results of the post-trip MSLB analysis demonstrate that the site boundary dose is less than 10 CFR Part 100 guidelines and that the core remains coolable.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the pre-trip and post-trip MSLB Event remain within the NRC acceptance criteria that the site boundary dose does not exceed the 10 CFR Part 100 guidelines and that the core geometry remains coolable. However, it was determined that the pre-trip MSLB Event, with reduced RCS flow, involves a USQ due to the increased number of fuel pins which may fail as a result of this event. Additionally, reanalysis of the pre-trip MSLB Event includes a new assumption that a time delay exists between the reactor trip and subsequent RCP coastdown; therefore, this analysis is discussed in greater detail in Section VII.

UFSAR Section 14.15, Steam Generator Tube Rupture

The SGTR Event is a Postulated Accident involving the double-ended rupture of a SG tube. As a consequence, this event involves an RCS depressurization and approach to the DNB fuel SAFDL. However, the RCS depressurization rate is not as severe as for the RCS Depressurization Event, so the new analysis of the RCS Depressurization Event is used to verify that the fuel SAFDLs are not exceeded during the SGTR Event. The SGTR Event also involves transport of activity from the RCS to the SG, and then offsite through the available steam paths. Therefore, the SGTR Event is analyzed to demonstrate site boundary doses are a small fraction (no more than 10%) of the 10 CFR Part 100 guidelines. Plugging SG tubes has the effect of decreasing the SG steam pressure, which will slightly increase the leak rate through the ruptured tube. However, the flow through the ruptured tube is choked for a significant portion of the transient. As a result, it was judged that SG tube plugging will not significantly affect the SGTR Event, and an evaluation of the event was performed. The NRC acceptance criterion for this Postulated Accident is that the site boundary dose does not exceed 10% of the 10 CFR Part 100 guidelines.

An evaluation of the SGTR Event was performed for the reduced flow condition, with increased tube plugging, and increased coolant temperature. The evaluation concluded that the margin to the NRC acceptance criterion for the site boundary dose was adequate to accommodate the slight increase in the site boundary dose caused by SG tube plugging, and the site boundary dose will not exceed 10% of the 10 CFR Part 100 guidelines.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the evaluation indicates that the results of the SGTR Event analysis remain within the NRC acceptance criterion that the site boundary dose does not exceed 10% of the 10 CFR Part 100 guidelines.

UFSAR Section 14.16, Seized Rotor Event

The Seized Rotor [illegible] a Postulated Accident involving the instantaneous seizure of a single RCP shaft. The reacto [illegible] flow through the core is asymmetrically reduced to three-pump flow as a result of the shaft s [illegible] An instantaneous step change in RCS flow from four-pump to three-pump flow is assumed in the [illegible]ysis. The rapid RCS flow reduction causes RCS temperature and core power to increase. The increase in core power and decrease in RCS flow rate may cause core fuel to exceed the SAFDLs and fail. Plugging SG tubes causes RCS flow to decrease both for the four-pump and three-pump combinations. As the Seized Rotor Event may be significantly affected by SG tube plugging, a new analysis of the event was performed. The NRC acceptance criteria for this Postulated Accident are that the site boundary dose shall not exceed the 10 CFR Part 100 guidelines and that the core geometry remains coolable.

Analysis of the Seized Rotor Event was performed using the TORC and CETOP computer codes. This methodology is the current NRC-approved methodology for this event for Calvert Cliffs. The analysis was performed for the reduced flow condition, with increased tube plugging, and increased coolant temperature. In order to bound the results of the Seized Rotor Event analysis, the percentage of fuel pins that may fail is increased from that previously reviewed and accepted by the NRC. Although the site boundary dose does not exceed 10 CFR Part 100 guidelines and the core geometry remains coolable, the increased number of fuel pin failures is an increase in the probability of malfunction of equipment important to safety (i.e., the fuel pin cladding), and it was determined that this event involves a USQ. For this reason, the Seized Rotor analysis is presented in detail in Section VII for NRC review.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the Seized Rotor Event remain within the NRC acceptance criteria that the site boundary dose does not exceed the 10 CFR Part 100 guidelines, and that the core geometry remains coolable. However, it was determined that the Seized Rotor Event, with reduced RCS flow, involves a USQ due to the increased number of fuel pins which may fail as a result of this event.

UFSAR Section 14.17, Loss-of-Coolant Accident

The LOCA is a Postulated Accident involving a rupture of the RCS piping. The RCS piping rupture causes a loss of RCS inventory and rapid RCS depressurization. The safety injection tanks and Emergency Core Cooling System pumps respond to recover the core. The NRC acceptance criteria for this Postulated Accident are specified in 10 CFR 50.46(b).

For a small break LOCA, tube plugging is significant for two reasons: (1) The available RCS inventory is reduced, and (2) the heat transfer to the secondary system is reduced. The reduced inventory results in a more rapid core uncovery during the course of the accident. The reduced SG heat transfer results in elevated RCS pressure, which reduces flow from the Emergency Core Cooling System and prolongs the core uncovery period. Therefore, the small break LOCA has been reanalyzed to support this license amendment.

For a large break LOCA, SG tube plugging is significant since it affects the available flow area through the SGs. This flow area is important for cold leg breaks (the limiting break location at Calvert Cliffs), since it affects the pressure drop caused by steam flow from the core, through the SG, and out the break. A smaller flow area results in a larger pressure drop, which reduces the core reflood rate. This effect will elevate the peak clad temperature for the event. Therefore, the large break LOCA analysis has been reanalyzed to support this license amendment.

Analysis of small break LOCA was performed using the CEFLASH-4AS and PARCH computer codes. Analysis of the large break LOCA was performed using the CEFLASH-4A, COMPERC-II, PARCH, and STRIKIN-II computer codes. These are the NRC-approved LOCA methodologies for Calvert Cliffs. The analyses were performed for the reduced flow condition, with increased tube plugging. The results of the new analyses of large break and small break LOCAs demonstrate that the NRC acceptance criteria specified in 10 CFR 50.46(b) are met. The maximum peak cladding temperature of 2181°F would occur during a large break LOCA.

In summary, reanalysis of both the small break and large break LOCAs, with reduced flow conditions associated with plugging up to 2500 tubes per SG, demonstrates that the NRC acceptance criteria specified in 10 CFR 50.46(b) for LOCA are met.

UFSAR Section 14.18, Fuel Handling Incident

The Fuel Handling Incident is a Postulated Occurrence involving a dropped fuel assembly. The analysis for this event is not affected by plugging SG tubes. Therefore, no additional analysis or evaluation has been performed to support reduced flow conditions associated with plugging up to 2500 tubes per SG.

UFSAR Section 14.19, Turbine-Generator Overspeed Incident

The Turbine-Generator Overspeed Incident is a Postulated Occurrence involving the failure of the turbine-generator. The analysis for this event is not affected by plugging SG tubes. Therefore, no additional analysis or evaluation has been performed to support reduced flow conditions associated with plugging up to 2500 tubes per SG.

UFSAR Section 14.20, Containment Response

The Containment Response Event is a Postulated Occurrence involving a rapid increase in containment temperature and pressure as a result of an MSLB or reactor coolant pipe rupture. Steam generator tube plugging decreases the RCS inventory, reduces the SG heat transfer rate, and reduces the initial SG pressure and temperature. These effects have an overall beneficial effect on the mass and energy release from the RCS and the SGs. These effects completely offset the effect of the slightly elevated RCS average coolant temperature. Therefore, the analysis for the Containment Response Event is not affected by reduced flow conditions associated with plugging up to 2500 tubes per SG, so no additional analysis or evaluation has been performed.

UFSAR Section 14.21, Hydrogen Accumulation in Containment

The Hydrogen Accumulation in Containment Event is a Postulated Occurrence involving the build-up and removal of hydrogen gas in containment following a LOCA. The analysis for this event is not affected by plugging SG tubes. Therefore, no additional analysis or evaluation has been performed to support reduced flow conditions associated with plugging up to 2500 tubes per SG.

UFSAR Section 14.22, Waste Gas Incident

UFSAR Section 14.23, Waste Evaporator Incident

UFSAR Section 14.24, Maximum Hypothetical Accident

All three of these events are Postulated Occurrences involving calculation of site boundary doses due to activity released from the Auxiliary Building or Containment. Plugging SG tubes does not affect the amount of activity or release rates for the activity for any of these analyses. Therefore, no additional analyses or evaluations have been performed for these events to support reduced flow conditions associated with plugging up to 2500 tubes per SG.

UFSAR Section 14.25, Excessive Charging Event

The Excessive Charging Event is an Anticipated Operational Occurrence involving an increase in RCS pressure as a result of inadvertent charging system operation. The time available for operators to respond to the event is determined by the available pressurizer bubble volume and the charging system capacity. Plugging SG tubes does not affect these parameters. Therefore, no additional analysis or evaluation has been performed to support reduced flow conditions associated with plugging up to 2500 tubes per SG.

UFSAR Section 14.26, Feedline Break Event

The Feedline Break Event is a Postulated Accident involving rupture of the main feedline between the SG and the reverse flow check valve in Containment. The most limiting break size and location results in rapid uncovery of the SG tubes, leading to an RCS temperature increase. The RCS temperature increase causes an RCS pressure increase that approaches the RCS pressure upset limit (2750 psia). Plugging SG tubes has the effect of reducing the heat transfer rate to the secondary side of the SG. This effect exacerbates the RCS temperature and pressure increase. Therefore, plugging SG tubes has a significant effect on the Feedline Break Event, and a new analysis has been performed supporting this license amendment. The NRC acceptance criterion for this Postulated Accident is that the RCS pressure upset limit is not exceeded.

Analysis of the Feedline Break Event was performed using the CESEC computer code, the current NRC-approved methodology for Calvert Cliffs. The analysis was performed for the reduced flow condition, with increased tube plugging, and increased coolant temperature. The analysis assumed an MTC of $+0.15 \times 10^{-4}\ \Delta\rho/°F$, consistent with Reference (4). The results of the analysis demonstrate that the RCS pressure upset limit is not exceeded.

In summary, considering the reduced flow conditions associated with plugging up to 2500 tubes per SG, the results of the reanalysis of the Feedline Break Event remain within the NRC acceptance criterion that the RCS pressure upset limit is not exceeded.

Summary of Setpoint Analyses

The effect of the increase in SG tube plugging, decrease in RCS flow rate, and increase in coolant temperature have been incorporated in the setpoint analyses supporting the TM/LP reactor trip setpoints and the Technical Specification LCO for DNB, as documented in the Core Operating Limits Report. The TM/LP setpoints, in concert with the DNB LCOs, were verified to provide adequate thermal margin for all Anticipated Operational Occurrences to prevent failure of fuel. A change to the setpoints in the Core Operating Limits Report associated with the DNB LCO when monitoring via the excore nuclear instruments, and associated with Total Integrated Radial Peaking Factors (F_r^T) will be required and will be implemented prior to implementation of the license amendment for each unit. The LHR protection system setpoints and LCOs are not affected by SG tube plugging or RCS flow changes.

V. RCS CORROSION AND STRUCTURAL EVALUATIONS

RCS Alloy 600 Corrosion Evaluation

An evaluation of the effect of increased core outlet coolant temperature on Alloy 600 corrosion was performed. Components in the reactor head, RCS hot leg, and SG were considered. Due to the decrease in RCS flow rate caused by plugging SG tubes, the RCS hot leg and reactor vessel head coolant temperatures will increase slightly. Considering a decrease in the Technical Specification minimum allowed RCS flow rate from 370,000 gpm to 340,000 gpm, the average RCS hot leg temperature will increase approximately 4°F to a maximum average hot leg temperature of 601°F. An evaluation of the effect of this temperature increase on the rate of primary water stress corrosion cracking on the RCS primary side, and of intergranular stress corrosion cracking on the SG secondary side, was performed. Note that since the normal operating pressure for the RCS will not change, there is no increase in pressurizer temperature.

The evaluation concluded that previous safety assessments to address the consequences of primary water stress corrosion cracking are not affected by the hot leg temperature increase resulting from the proposed RCS flow reduction. For the control element drive mechanism and incore instrument nozzles in the reactor vessel head, previous safety assessments, References (5) and (6), have concluded that cracks would penetrate the wall long before reaching critical size for unstable crack propagation. Such cracks would result in boric acid leakage, which would be detected by periodic visual inspections. The safety assessments were performed generically for Combustion Engineering nuclear steam supply systems for which the maximum hot leg temperature considered was 621°F. The conclusions of these safety assessments, therefore, remain valid for Calvert Cliffs with the requested RCS flow reduction. For the RCS instrument and resistance temperature detector nozzles in the hot leg, safety assessments, References (7) and (8), performed previously for pressurizer instrumentation nozzles with a similar configuration, assuming a coolant temperature of 653°F, are bounding for the temperature increase associated with this amendment As a result, the probability of a LOCA due to failure of an Alloy 600 RCS penetration will not increase.

Calvert Cliffs' Alloy 600 SG tubes are high-temperature mill-annealed (as opposed to low-temperature mill-annealed), which makes the tubes much less susceptible to primary water stress corrosion cracking. However, Calvert Cliffs' SG tubes have experienced intergranular stress corrosion cracking originating on the secondary side (outside) of the tubes. The span of intergranular stress corrosion cracking growth rates for Alloy 600 SG tubes throughout the industry is 3-6% through-wall, per effective full-power year. This data includes plant operation for pressurized water reactors at temperatures significantly higher than those for Calvert Cliffs. In addition, the temperature increase on the outside of the SG tubes is mitigated by the decrease in the average SG secondary side temperature caused by tube plugging. Therefore, the temperature increase associated with this amendment will not affect the probability of identification and isolation of SG tube cracks before they fail in service. As a result, the probability of a SGTR will not increase.

RCS Structural Evaluation

An evaluation of the effect of increased core outlet coolant temperature on the RCS structural integrity was performed. The evaluation involved review of the stress reports of the Calvert Cliffs reactor vessels and RCS coolant piping. A design temperature of 650°F and a normal operating hot leg temperature of 604°F were used in the original analyses of the reactor vessel and RCS coolant piping. These

temperatures bound the temperature increase associated with the RCS flow reduction requested in this amendment. The evaluation concluded that the expected temperature increase remains within the design conditions already analyzed for Calvert Cliffs.

VI. OPERATIONAL EVALUATION

An operational evaluation was performed to assess the effect of plugging up to 2500 tubes per SG on normal plant operation. Best estimate methods were used to determine the RCS and SG parameters during normal plant operation with up to 2500 tubes plugged in each SG. Best estimate methods were also used to analyze normal maneuvering transients on the plant. These results were then used to evaluate the effect of plugging up to 2500 tubes per SG on the operation of the main turbine and various non-safety-related control systems. A summary of the effects on these systems is discussed below.

Steam Generator Secondary Side

One of the primary effects of plugging SG tubes is that it reduces the heat transfer area in the SGs. This effect causes SG steam pressure to decrease, in order to transfer the same amount of heat. The effect of the reduced full-power steam pressure in the SG on the operating margin to the Low SG Pressure reactor trip, and on SG instrumentation, was considered. The evaluation concluded that the operating margin to the Low SG Pressure trip was adequate. However, periodic calibration of the main steam flow and SG water level instrumentation may be required due to the lower steam density at the reduced SG steam pressure.

Main Turbine

Plugging 2500 tubes per SG will result in SG steam pressure below the main turbine design full-power steam pressure. As a result, the main turbine may not be able to operate at full power. Alternatives are being considered to maximize power output of the main turbine.

Pressurizer Level Control

As previously discussed, the full-power core outlet temperature increases as a result of plugging SG tubes. Therefore, the full-power average coolant temperature increases. As a result, the change in RCS average temperature after a reactor trip will increase slightly. This evaluation confirmed that the program pressurizer level at full power is adequate to prevent draining the pressurizer after a reactor trip. In general, the evaluation concluded that the functions of the pressurizer level control system will continue to be performed with up to 2500 plugged tubes per SG.

Pressurizer Pressure Control

Plugging SG tubes reduces the RCS flow rate, which results in a decrease in the pressure drop across the reactor vessel during normal operation. This pressure drop provides the driving force for pressurizer spray flow. The evaluation concluded that pressurizer spray flow for the minimum RCS flow rate requested in this amendment is sufficient to meet the minimum design flow for pressurizer spray. In addition, the evaluation determined that the pressurizer pressure control system functions adequately to prevent reactor trips on expected load rejections and maneuvering transients.

Feedwater Control System

The reduced SG steam pressure due to plugging SG tubes will result in an increase in the feedwater regulating valve differential pressure. The increased pressure drop across the valve may necessitate periodic adjustments to the feedwater control system.

Steam Dump and Bypass System

Due to the reduced heat transfer rate to the SG secondary side as a result of an increased number of plugged tubes, the post-trip demands on the Steam Dump And Bypass System (SDBS) are reduced. Therefore, the evaluation concluded that the SDBS will continue to perform its design functions.

Reactor Coolant Pump Operation

As a result of SG tube plugging, the RCP seal pressure drop decreases for the same pressurizer pressure. Therefore, adjustments to the RCP operating curves will be required to ensure the minimum pressure for RCP seal operation is maintained.

VII. DETAILED DESIGN BASIS EVENT ANALYSES

UFSAR Section 14.3, Boron Dilution Event

A. Event Description

An inadvertent Boron Dilution Event at power and startup (Mode 1 and 2) can be postulated as a result of various malfunctions or inadvertent operation of the Chemical and Volume Control System. The sequence of events starts with the reduction in the boron concentration in the RCS. All three charging pumps are on, adding a maximum flow of unborated (demineralized) water into the RCS. The effect of reducing the boron concentration is to add positive reactivity. With the reactor initially critical, the core power, heat flux, and RCS temperatures will increase the pressurizer pressure and level. The combination of the pressurizer sprays and RCS letdown will accommodate those slow increases in pressure and level. However, no credit is taken for the pressurizer pressure and level control systems in the analysis. The SG temperature and pressure will slowly increase with the increasing average RCS temperature. This sequence is similar to the slow reactivity addition due to a CEA withdrawal.

If the dilution is not secured, the reactor will shut down by either the TM/LP or the Variable High Power trip. The action of the Pressurizer Pressure Control System (PPCS) will prevent the pressure from exceeding the Pressurizer Pressure-High trip setpoint. Operator action is required to secure the dilution.

B. Analysis

The method of evaluating the Mode 2 case has been changed such that NRC review is required. Mode 2 is declared during a startup when RCS boron concentration is within 100 ppm of critical boron concentration, and dilution is continuing for normal startup. The plant is in Mode 2 until reaching 5% rated thermal power. Mode 2 is declared during a shutdown when power has decreased to 5%. Therefore, in Mode 2, shutdown margin is provided by the CEAs, not the boron concentration.

Because Mode 2 includes power operation up to 5%, the method used for evaluating the Mode 2 Boron Dilution Event has been changed such that it is now identical to that used for the Mode 1 analysis (comparison with the CEA Withdrawal Event). The present UFSAR description includes a Mode 2

analysis which determines the minimum time to lose prescribed shutdown margin based on an initial boron concentration which is actually applicable to Mode 3. This case is actually a Mode 3 case, and should not be described as a Mode 2 case. The Mode 2 case should be evaluated by comparison of the reactivity insertion rate to the reactivity insertion rate of the CEA Withdrawal Event.

C. Equation Derivation - Reactivity Insertion Rate

The reactivity insertion rate during Modes 1 and 2 is simply the rate of concentration change divided by the Inverse Boron Worth. The reactivity insertion rate during Modes 1 and 2 is determined by performing a simple hand calculation.

$$\text{Reactivity Insertion Rate: } = \frac{dC}{dt} \bullet \frac{1}{IBW} = \frac{W}{M} \bullet \frac{\rho_{chg}}{\rho_{RCS}} \bullet \frac{C}{IBW \bullet 100} \, \Delta\rho \, / \, \text{sec}$$

where:

M	=	RCS active volume:	8861 ft^3
ρ_{RCS}	=	RCS density:	44.28 lb_m/ft^3 (Mode 1)
			46.97 lb_m/ft^3 (Mode 2)
W	=	Charging volume flow rate:	150 ft^3/sec
ρ_{chg}	=	Charging water density:	62.4 lb_m/ft^3
C	=	Boron concentration:	1800 ppm
IBW	=	Inverse Boron Worth:	70 ppm/% $\Delta\rho$ (Mode 1)
			65 ppm/% $\Delta\rho$ (Mode 2)

$$\text{Reactivity Insertion Rate: } = \frac{150 \text{ gpm}}{7.48 \text{ gal/ft}^3} \bullet \frac{\text{min}}{60 \text{ sec}} \bullet \frac{62.4}{\rho_{RCS}} \bullet \frac{\%\Delta\rho}{\text{IBW}} \bullet \frac{1800}{8861 \text{ ft}^3 \bullet 100}$$

$$= 1.37 \times 10^{-5} \, \Delta\rho/\text{sec (Mode 1)}$$

$$= 1.39 \times 10^{-5} \, \Delta\rho/\text{sec (Mode 2)}$$

D. Results

For Modes 1 and 2, an inadvertent charging of unborated water at the maximum rate would result in a maximum rate of reactivity addition of 1.39×10^{-5} $\Delta\rho$/sec. This is an order of magnitude slower than a CEA Withdrawal Event (UFSAR Section 14.2), and is, therefore, not as limiting as that event. Analysis of the most limiting CEA Withdrawal Event demonstrates that fuel failure does not occur. Therefore, fuel failure will not occur during the Modes 1 and 2 Boron Dilution Events.

UFSAR Section 14.5, Loss of Load Event

A. Event Description

A Loss of Load Event is defined as any event that results in a reduction in the SG's heat removal capacity through the loss of secondary steam flow. Closure of all main steam isolation valves, turbine stop valves, or turbine control valves will cause a Loss of Load Event.

The most limiting Loss of Load Event is a turbine trip without a concurrent reactor trip, or an inadvertent closure of the turbine stop valves at hot full power. A turbine trip would result in the closure of the turbine stop valves.

Reanalysis of the Loss of Load Event was necessary to consider changes associated with an increased number of plugged SG tubes and a reduced minimum core flow. The reanalysis also included a reduced maximum setpoint for the third bank of MSSVs (from 1065 psig to 1050 psig).

B. Analysis

The objectives in analyzing this event are to demonstrate that: (1) the primary pressure relief capacity is sufficient to limit the RCS pressure to less than the RCS upset pressure limit (2750 psia); (2) the peak secondary pressure stays below the secondary pressure upset limit (1100 psia); (3) the minimum DNBR remains above the safety limit; and (4) the peak LHR remains below the SAFDL.

No credit is taken for the reactor trip on turbine trip for the event. To maximize the pressure and temperature increase for the event, no credit is taken for the SDBS which would reduce the pressure transient.

Three primary cases are analyzed for this event; the first case determines the peak RCS pressure, the second case determines the peak secondary pressure, and the third case determines the minimum DNBR for the event. The limiting case in terms of power and temperature determines the peak LHR for the event.

C. Results

Parametric analyses were run to determine the most limiting conditions for the peak RCS pressure and peak secondary pressure cases. The peak RCS pressure case used the inputs detailed in Table 14.5-1. The peak secondary pressure case used the inputs detailed in Table 14.5-2. The minimum DNBR case used the inputs detailed in Table 14.5-3.

The peak RCS pressure case resulted in a peak pressure below the acceptance criteria of 110% of design pressure. The peak RCS pressure reached 2675 psia, below the RCS upset limit of 2750 psia. Table 14.5-4 details the sequence of events for the peak RCS pressure case; and Figures 14.5-1 through 14.5-6 detail the core power, core average heat flux, RCS pressure, RCS temperature, SG pressure, and pressurizer water volume versus time for the peak RCS pressure case.

The peak secondary pressure case resulted in a peak SG pressure below the acceptance criteria of 110% of design pressure. The peak secondary pressure reached 1086 psia, below the secondary pressure upset limit of 1100 psia. Table 14.5-5 details the sequence of events for the peak secondary pressure case. No plots are provided for the peak secondary pressure case because this transient behaves similarly to the peak RCS pressure transient case.

The minimum DNBR and peak LHR both remain below the specified safety limits for the Loss of Load Event.

TABLE 14.5-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOSS OF LOAD EVENT TO CALCULATE MAXIMUM RCS PRESSURE

PRESSURE	UNITS	UNIT 1	UNIT 2
Initial Core Power Level	MWt	2754[b]	2754[b]
Initial Core Inlet Coolant Temperature	°F	546	546
Initial RCS Vessel Flow Rate	gpm	340,000	340,000
Initial Pressurizer Pressure	psia	2165[a]	2165[a]
Initial Pressurizer Liquid Volume at Full Power	ft^3	975	975
Initial SG Pressure	psia	770	770
MTC	$x\ 10^{-4}\ \Delta\rho/^{o}F$	+0.15	+0.15
Doppler Coefficient Multiplier	---	0.85	0.85
No. of Plugged Tubes per SG	---	2500	2500
Axial Shape Index	---	+0.4	+0.4
CEA Worth at Trip	% $\Delta\rho$	-5.0	-5.0
Time to 90% Insertion of SCRAM Rods	seconds	3.1	3.1
RRS [Reactor Regulating System]	Operating Mode	Manual	Manual
SDBS	Operating Mode	Inoperative	Inoperative
MSSV Setpoints			
Bank 1	psig	995	995
Bank 2	psig	1035	1035
Bank 3	psig	1050	1050
PPCS [Pressurizer Pressure Control System]	Operating Mode	Manual	Manual
PLCS [Pressurizer Level Control System]	Operating Mode	Manual	Manual

(a) Corresponds to Technical Specification minimum indicated pressure of 2200 psia. The value includes an uncertainty of 35 psi.

(b) Value does not include 17 MWt of pump heat added to core power in the CESEC code.

TABLE 14.5-2

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOSS OF LOAD EVENT TO CALCULATE MAXIMUM SECONDARY PRESSURE

PRESSURE	UNITS	UNIT 1	UNIT 2
Initial Core Power Level	MWt	2754[(b)]	2754[(b)]
Initial Core Inlet Coolant Temperature	°F	552	552
Initial RCS Vessel Flow Rate	gpm	340,000	340,000
Initial Pressurizer Pressure	psia	2165[(a)]	2165[(a)]
Initial Pressurizer Liquid Volume at Full Power	ft^3	800	800
Initial SG Pressure	psia	841.7	841.7
MTC	$\times 10^{-4}\ \Delta\rho/°F$	+0.15	+0.15
Doppler Coefficient Multiplier	---	0.85	0.85
CEA Worth at Trip	% $\Delta\rho$	-5.0	-5.0
No. of Plugged Tubes per SG	---	0	0
Axial Shape Index	---	+0.4	+0.4
Time to 90% Insertion of SCRAM Rods	seconds	3.1	3.1
RRS	Operating Mode	Manual	Manual
SDBS	Operating Mode	Inoperative	Inoperative
MSSV Setpoints			
Bank 1	psig	995	995
Bank 2	psig	1035	1035
Bank 3	psig	1050	1050
PPCS [Pressurizer Pressure Control System]	Operating Mode	Manual	Manual
PLCS [Pressurizer Level Control System]	Operating Mode	Manual	Manual

(a) Corresponds to Technical Specification minimum indicated pressure of 2200 psia. The value includes an uncertainty of 35 psi.

(b) Value does not include 17 MWt of pump heat added to core power in the CESEC code.

TABLE 14.5-3

INITIAL CONDITIONS AND INPUT PARAMETERS FOR LOSS OF LOAD EVENT TO CALCULATE TRANSIENT MINIMUM DNBR

PARAMETER	UNITS	UNIT 1	UNIT 2
Initial Core Power Level	MWt	2700[a]	2700[a]
Initial Core Inlet Coolant Temperature	°F	550[a]	550[a]
Initial RCS Vessel Flow Rate	gpm	340,000	340,000
Initial RCS Pressure	psia	2200[a]	2200[a]
Initial SG Pressure	psia	830	830
Integrated Radial Peaking Factors, F_r^T	---	1.75[a]	1.75[a]
MTC	x 10^{-4} $\Delta\rho$/°F	+0.15	+0.15
Doppler Coefficient Multiplier	---	0.85	0.85
CEA Worth at Trip	% $\Delta\rho$	-5.0	-5.0
Time to 90% Insertion of SCRAM Rods	sec	3.1	3.1
RRS	Operating Mode	Manual	Manual
SDBS	Operating Mode	Inoperative	Inoperative

[a] Effects of uncertainties on these parameters were accounted for statistically.

TABLE 14.5-4

SEQUENCE OF EVENTS FOR LOSS OF LOAD EVENT TO MAXIMIZE CALCULATED RCS PEAK PRESSURE

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Loss of Secondary Load	---
7.1	High Pressurizer Pressure Trip Signal Generated	2420 psia
7.6	CEAs Begin to Drop Into the Core	---
7.7	Pressurizer Safety Valves (PSVs) Begin to Open	2550 psia
8.7	SG Safety Valves Begin to Open	1010 psia
9.2	Maximum RCS Pressure	2675 psia[a]
13.0	Maximum SG Pressure	1084 psia[b]
14.7	PSVs are Fully Closed	2448 psia

(a) RCS pressure includes elevation head.
(b) SG pressure includes downcomer liquid head.

TABLE 14.5-5

SEQUENCE OF EVENTS FOR LOSS OF LOAD EVENT TO MAXIMIZE CALCULATED SECONDARY PEAK PRESSURE

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Loss of Secondary Load	---
4.1	SG Safety Valves Begin to Open	1010 psia
8.2	High Pressurizer Pressure Trip Signal Generated	2420 psia
8.7	CEAs Begin to Drop Into the Core	---
10.0	Pressurizer Safety Valves Begin to Open	2550 psia
10.2	Maximum RCS Pressure	2598 psia[a]
12.6	PSVs are Fully Closed	2448 psia
15.2	Maximum SG Pressure	1086 psia[b]

[a] RCS pressure includes elevation head.
[b] SG pressure includes downcomer liquid head.

FIGURE 14.5-1

LOSS OF LOAD EVENT
CORE POWER VS TIME

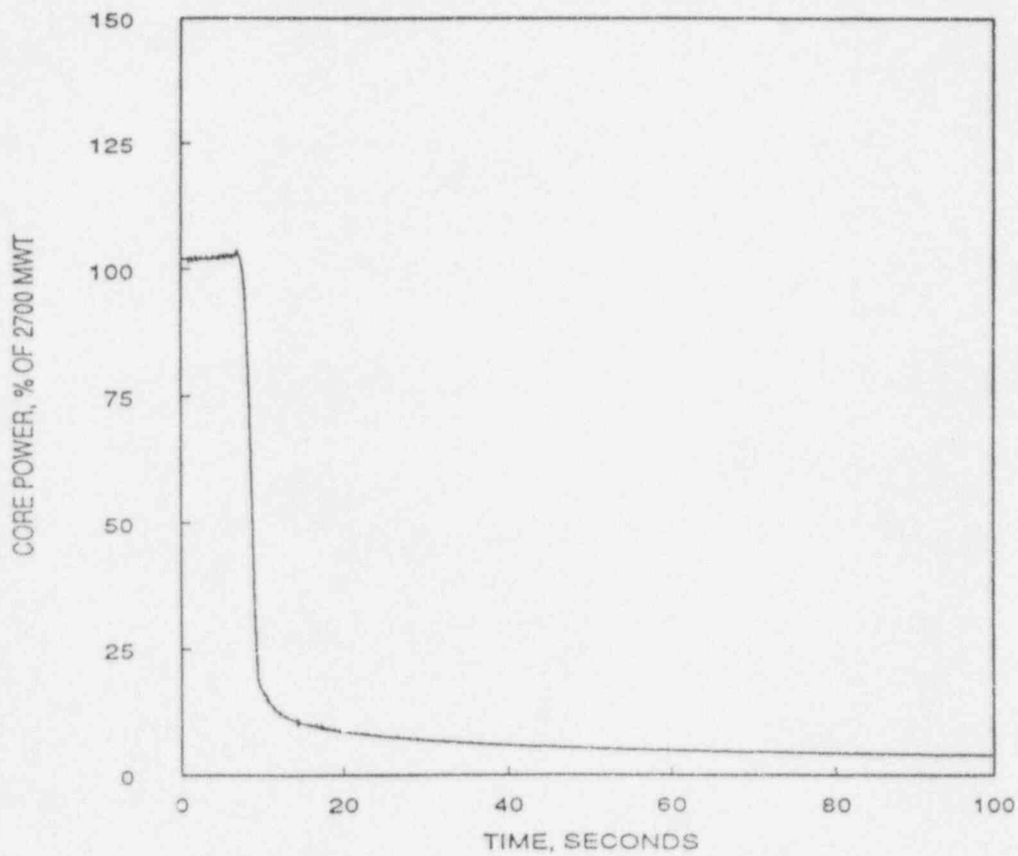


FIGURE 14.5-2

LOSS OF LOAD EVENT
CORE AVERAGE HEAT FLUX VS TIME

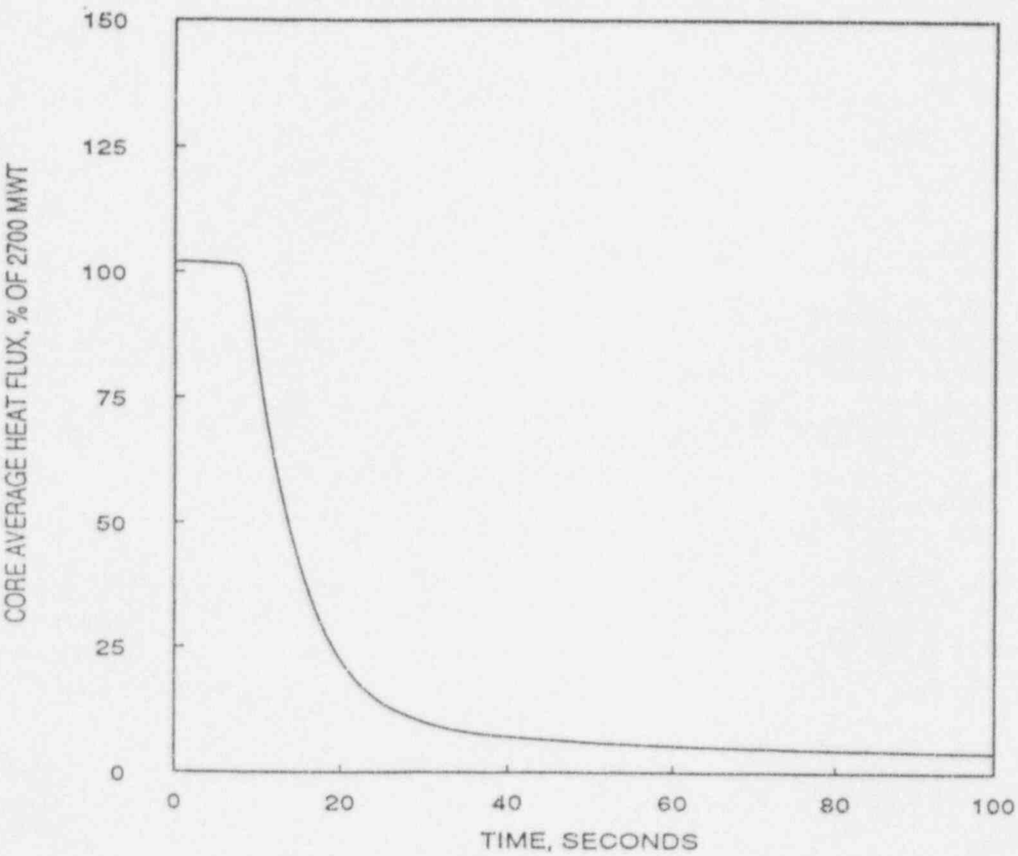
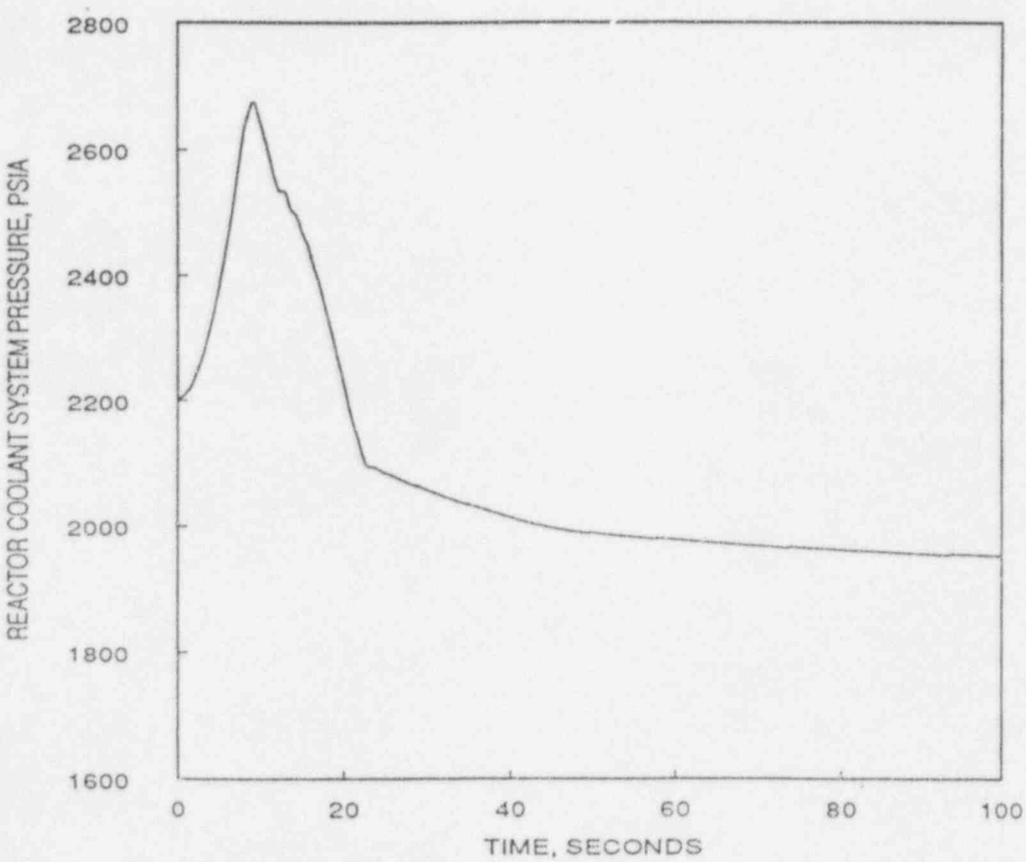


FIGURE 14.5-3

LOSS OF LOAD EVENT
RCS PRESSURE VS TIME



Includes elevation head.

FIGURE 14.5-4

LOSS OF LOAD EVENT
RCS TEMPERATURES VS TIME

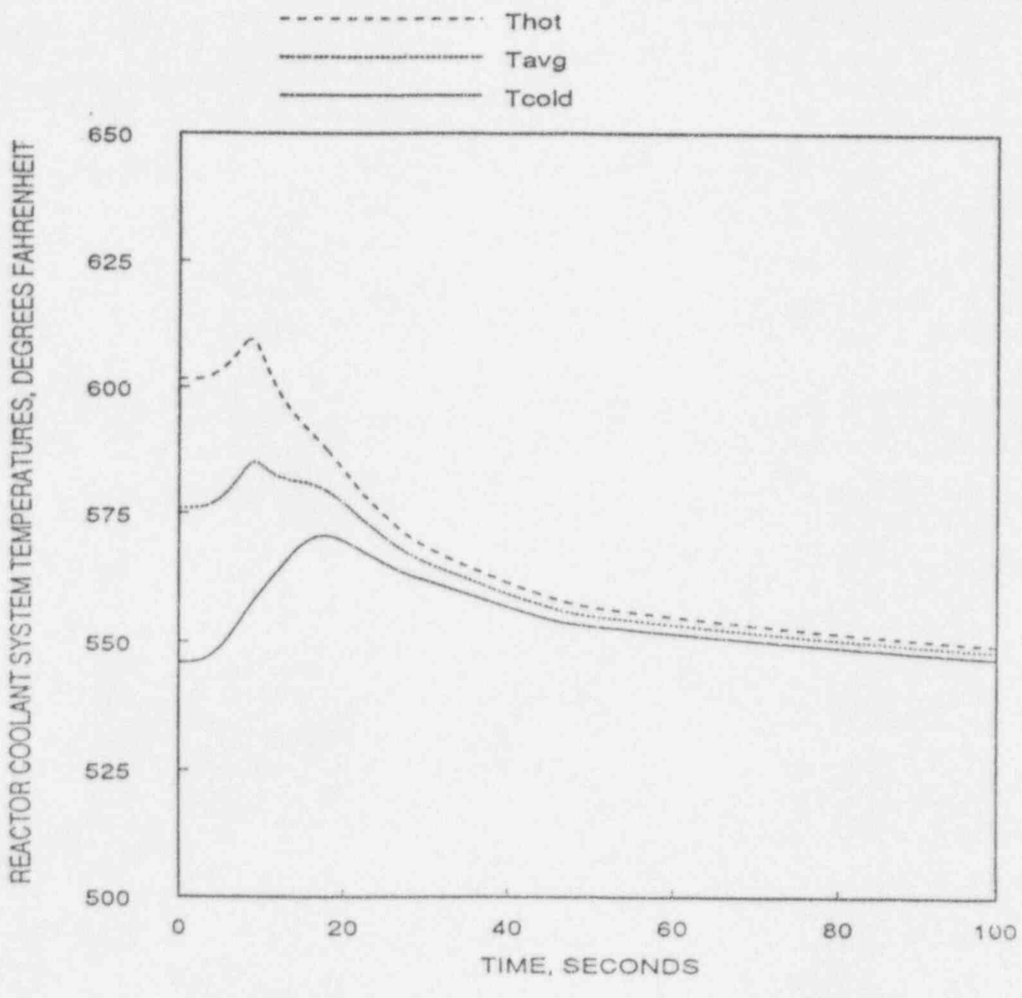
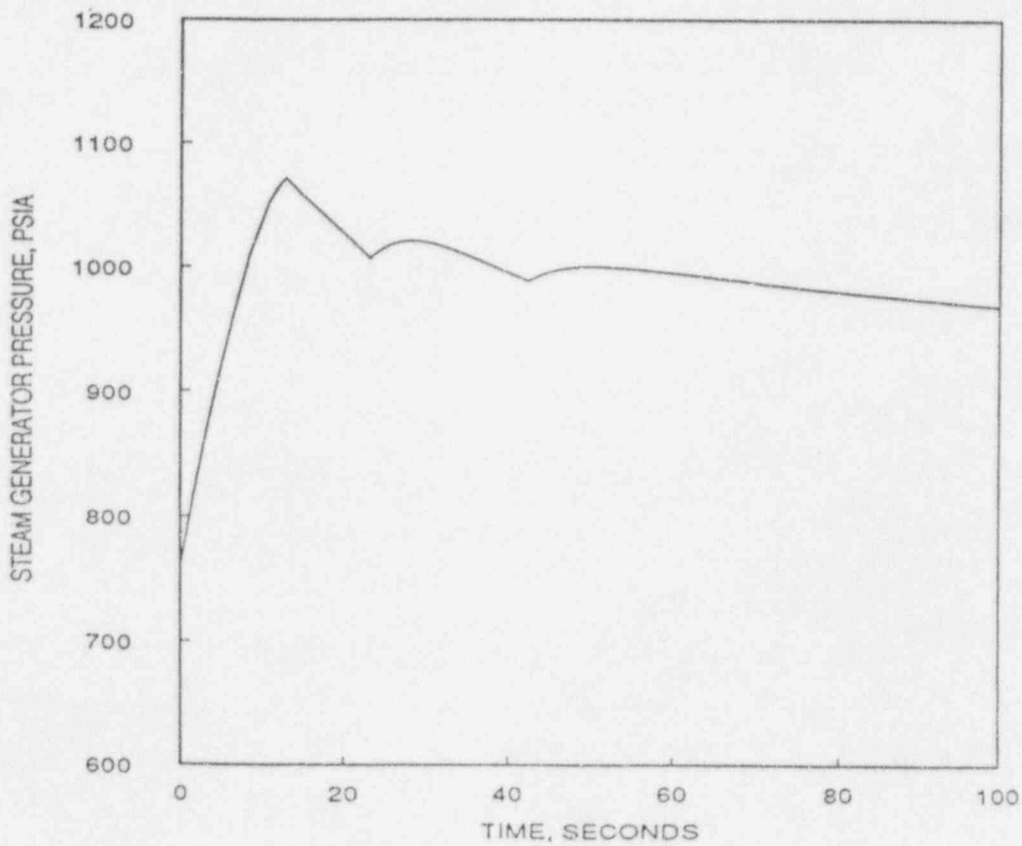


FIGURE 14.5-5

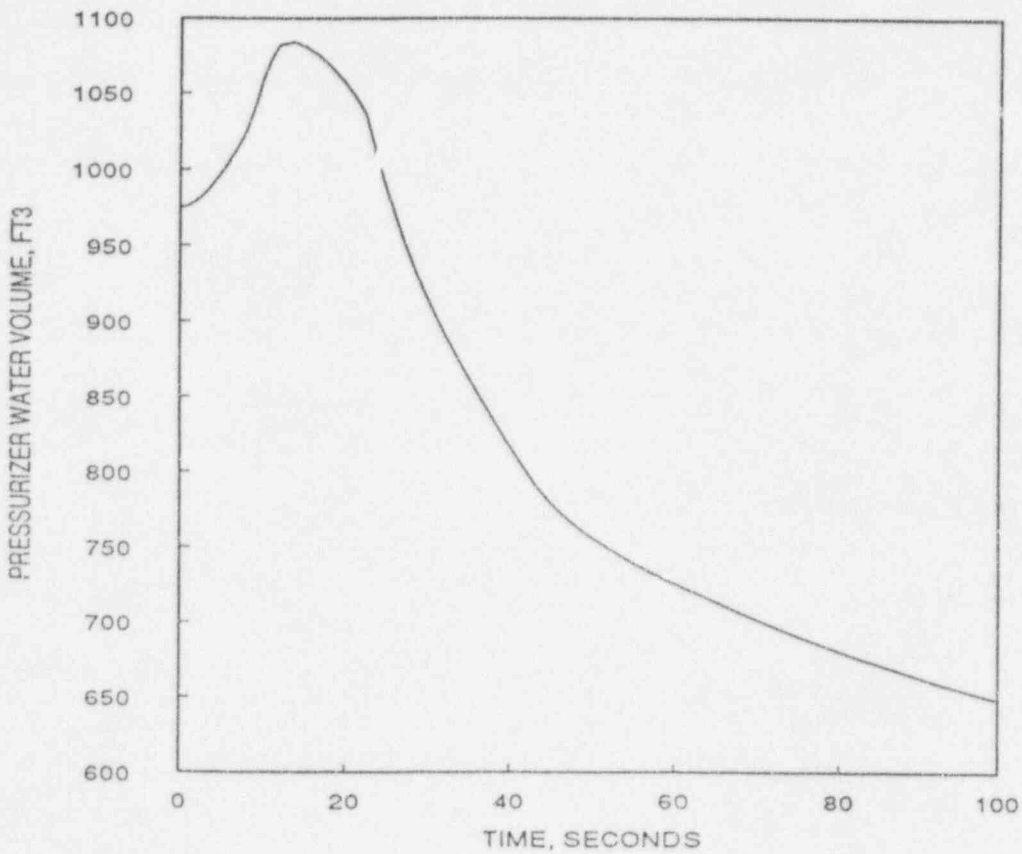
LOSS OF LOAD EVENT
STEAM GENERATOR PRESSURE VS TIME



* Does not include downcomer liquid head.

FIGURE 14.5-6

LOSS OF LOAD EVENT
PRESSURIZER WATER VOLUME VS TIME



UFSAR Section 14.6, Loss of Feedwater Flow Event

A. Event Description

An LOFW Event is defined as a reduction in feedwater flow to the SG without a corresponding reduction in steam flow from the SG. The closure of the feedwater regulating valves, the loss of condensate or feedwater pumps, or a pipe break in the condensate or feedwater systems during steady-state operation would result in an LOFW Event.

The most limiting LOFW Event at hot full power involves an inadvertent closure of both feedwater regulating valves. An instantaneous closure of the regulating valves would cause the largest steam and feedwater flow mismatch and result in the most rapid reduction in the SG inventory.

The LOFW Event primarily challenges the acceptance criteria on primary system over-pressurization and SG depletion. The over-pressurization analysis was the only analysis revised due to reduced RCS flow and increased RCS inlet temperature. Upon closure of both feedwater regulating valves, the immediate system response is a steady decrease in SG liquid inventory. The temperature in the SG will increase due to the loss of subcooled feedwater and cause the SG pressure to increase correspondingly. The turbine bypass valves will respond to the steady increase in main steam pressure by opening at 895 psia. This analysis assumes that the turbine bypass valves are inoperable, as this increases secondary pressure. With the SG inventory depleting, a Low SG Water Level trip would be initiated. This trip is assumed to be disabled in the analysis, as this assumption increases the heat added to the RCS, which causes the primary and secondary pressure to increase. The SG pressure will continue to increase until the final MSSVs analysis setpoint is reached.

The RCS temperature and pressure will increase due to the inability of the SG to remove all of the heat from the RCS. A reactor trip will occur upon reaching a High Pressurizer Pressure trip setpoint. The RCS is protected against over-pressurization by the PORVs and PSVs. However, the PORVs are assumed inoperable in this analysis. Actuation of the PSVs and MSSVs limits the magnitude of the primary system temperature and pressure increase.

With a positive MTC, increasing RCS temperature results in an increase in core power. The increasing RCS temperature and power reduce the margin to the thermal limits (i.e., DNBR limits) and challenge the DNBR acceptance criteria.

B. Analysis

The Nuclear Steam Supply System response to the LOFW Event was analyzed with the NRC-approved CESEC computer code. The transient minimum DNBR degradation calculation was performed with the CETOP computer code, which uses the CE-1 Critical Heat Flux correlation.

The inputs in Table 14.6-1 were used to maximize RCS over-pressurization.

Parametric cases were analyzed using various RCS inlet temperatures, SG pressures, number of plugged SG tubes, and RCS flow conditions. A minimum RCS flow of 340,000 gpm was considered in conjunction with the maximum SG tube plugging condition. The most conservative peak RCS pressure case included an initial SG pressure of 826 psia, zero plugged tubes, initial RCS temperature of 550°F, and RCS flow of 370,000 gpm.

The analysis included an MTC of $+0.15 \times 10^{-4}$ $\Delta\rho/°F$. With an increase in RCS temperature, the power will increase accordingly. The SDBS, PPCS, and PLCS, and the PORVs are assumed to be inoperable. These assumptions enhance the RCS pressure increase since the automatic operation of these systems mitigate the RCS pressure increase. A loss of AC power was assumed to occur at the time of the reactor trip. This will cause the RCPs to coast down and prevent operation of the SDBS, which will contribute to the primary heatup and pressurization.

The High Pressurizer Pressure trip analysis setpoint was 2420 psia. A final change from the previous analysis was a reduction of the third bank MSSV analysis setpoint from 1065 psig to 1050 psig.

C. Results

The sequence of events for the LOFW Event for peak pressure with a loss of AC power on reactor trip, is shown in Table 14.6-2. Figures 14.6-1 through 14.6-5 present the transient behavior of core power, core average heat flux, RCS temperatures, RCS pressure, and SG pressure.

The results show that the high pressurizer pressure trip analysis setpoint is reached at 22.8 seconds. The loss of AC power and reactor trip occur at 23.7 seconds. The PSVs begin to open at 26.2 seconds. The peak primary pressure was 2631 psia at 28.2 seconds. The peak secondary pressure, including elevation head, was 1080 psia at 29.2 seconds.

This analysis demonstrates that the peak primary pressure is well below the over-pressurization acceptance criteria of 110 % of design pressure (2750 psia). The peak secondary pressure is below the secondary over-pressurization acceptance criteria of 110 % of design pressure (1100 psia).

The DNBR margin degradation for this event is bounded by the Loss of Coolant Flow Event (UFSAR Section 14.9) DNBR margin.

TABLE 14.6-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOFW EVENT TO MAXIMIZE CALCULATED RCS PEAK PRESSURE

PARAMETER	UNITS	UNIT 1	UNIT 2
Initial Core Power Level	MWt	2754	2754
Initial Core Coolant Inlet Temperature	^{o}F	550[(b)]	550[(b)]
Initial RCS Vessel Flow Rate	gpm	370,000[(b)]	370,000[(b)]
Initial RCS Pressure	psia	2165	2165
Initial SG Pressure	psia	826[(b)]	826[(b)]
Initial Pressurizer Liquid Volume	ft^3	975	975
MTC	$x\ 10^{-4}\ \Delta\rho/^{o}F$	+0.15	+0.15
Doppler Coefficient Multiplier	---	0.85	0.85
Number of Plugged Tubes per SG	---	0[(b)]	0[(b)]
High Pressurizer Pressure Analysis Trip Setpoint	psia	2420	2420
MSSV Setpoints			
Bank 1	psig	995	995
Bank 2	psig	1035	1035
Bank 3	psig	1050	1050
RRS	Operating Mode	Manual[(a)]	Manual[(a)]
SDBS	Operating Mode	Manual[(a)]	Manual[(a)]

TABLE 14.6-1 (Continued)

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOFW EVENT TO MAXIMIZE CALCULATED RCS PEAK PRESSURE

PARAMETER	UNITS	UNIT 1	UNIT 2
PPCS	Operating Mode	Manual[a]	Manual[a]
PLCS	Operating Mode	Manual[a]	Manual[a]

(a) These modes of control system operation maximize the peak RCS pressure.

(b) Parametric cases were analyzed using various RCS inlet temperatures, SG pressures, SG plugged tubes, and RCS flow conditions. A minimum RCS flow of 340,000 gpm was considered in conjunction with the maximum SG tube plugging condition.

TABLE 14.6-2

SEQUENCE OF EVENTS FOR LOFW EVENT TO MAXIMIZE CALCULATED RCS PRESSURE WITH LOSS OF AC POWER

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Loss of Main Feedwater	---
22.8	High Pressurizer Pressure Trip Setpoint Reached	2420 psia
23.7	Trip Breakers Open, Loss of AC Power	---
24.0	Turbine Stop Valves Close	---
24.2	CEAs Begin to Drop into Core	---
26.2	PSVs Begin to Open	2550 psia
26.7	SG Safety Valves Begin to Open	1010 psia
28.2	Maximum RCS Pressure	2631 psia[a]
29.2	Maximum SG Pressure	1080 psia[b]
32.5	PSVs Close	2448 psia

(a) Pressure includes elevation head.
(b) SG pressure includes downcomer liquid head.

FIGURE 14.6-1

LOSS OF FEEDWATER FLOW EVENT
CORE POWER VS TIME

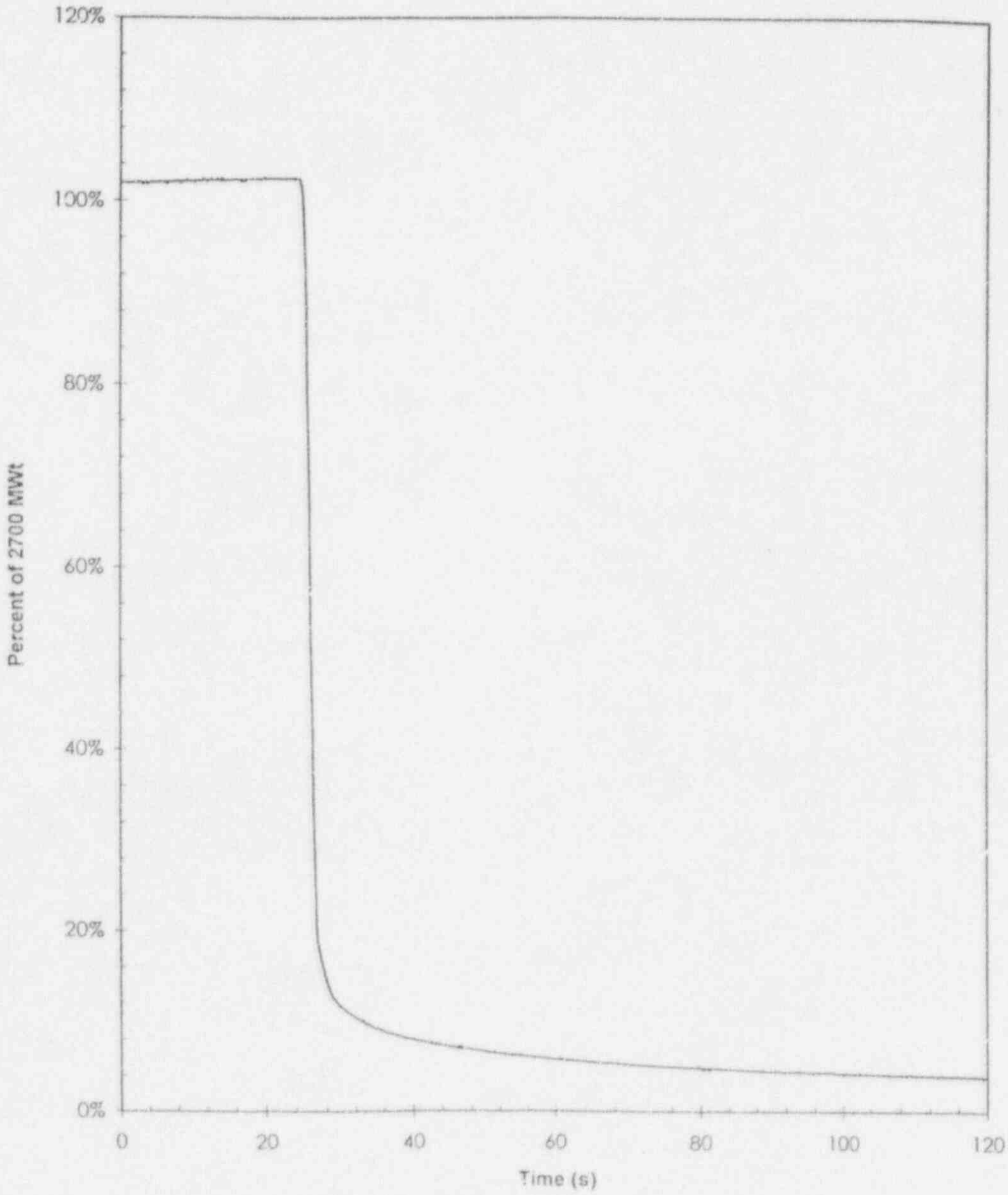


FIGURE 14.6-2

LOSS OF FEEDWATER FLOW EVENT
CORE HEAT FLUX VS TIME

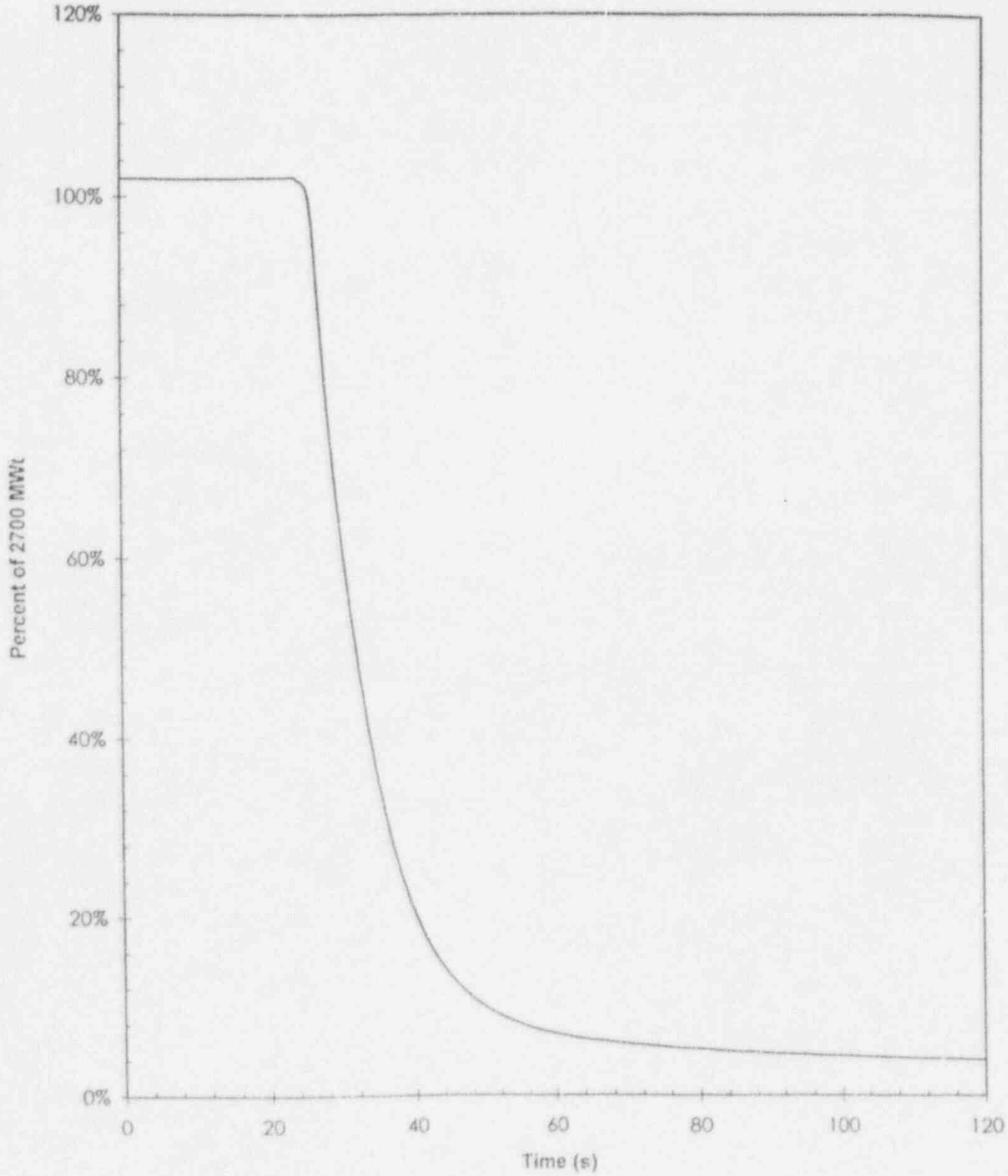


FIGURE 14.6-3

LOSS OF FEEDWATER FLOW EVENT
RCS COOLANT TEMPERATURES VS TIME

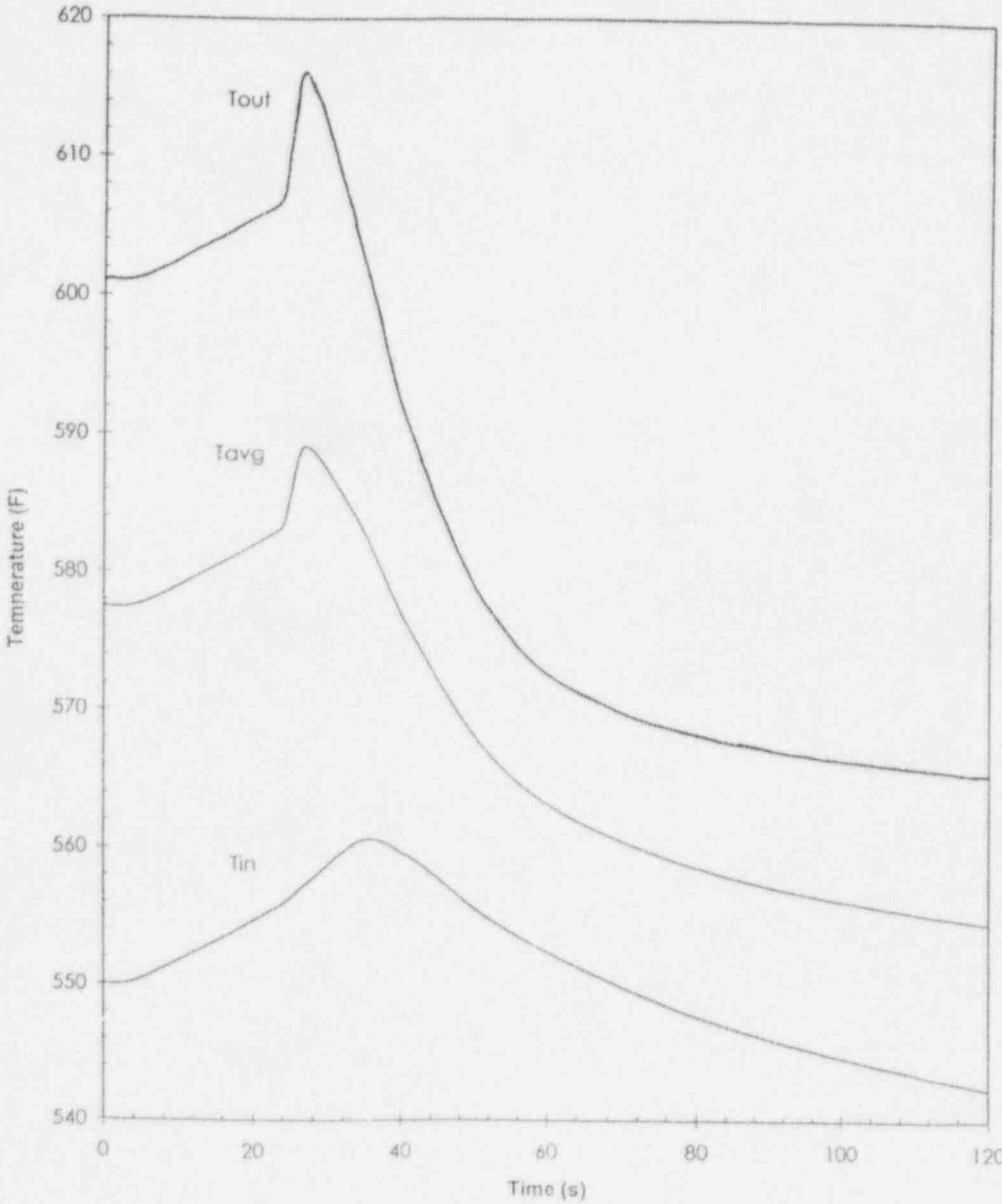


FIGURE 14.6-4

LOSS OF FEEDWATER FLOW EVENT
MAXIMUM RCS PRESSURE VS TIME

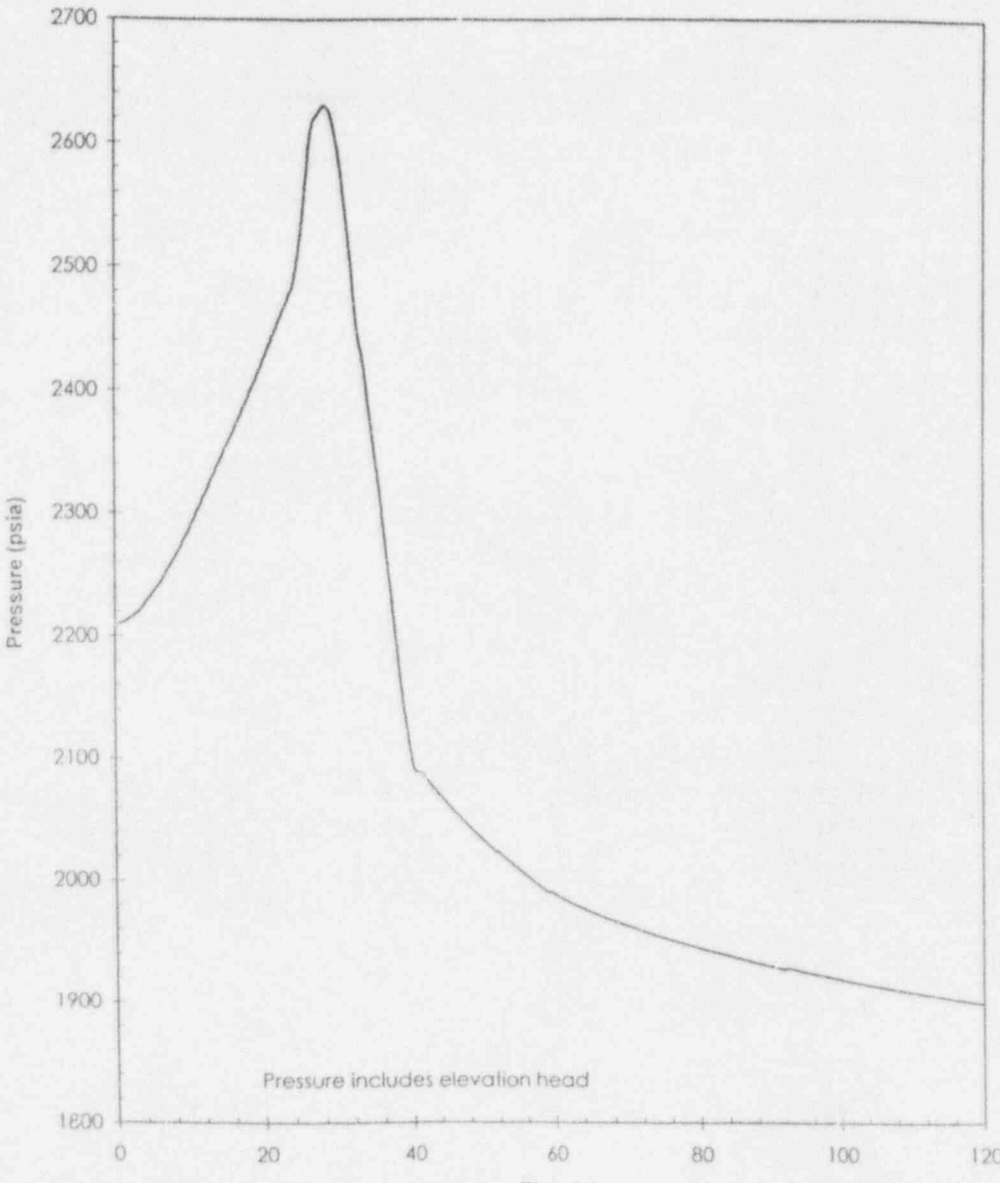
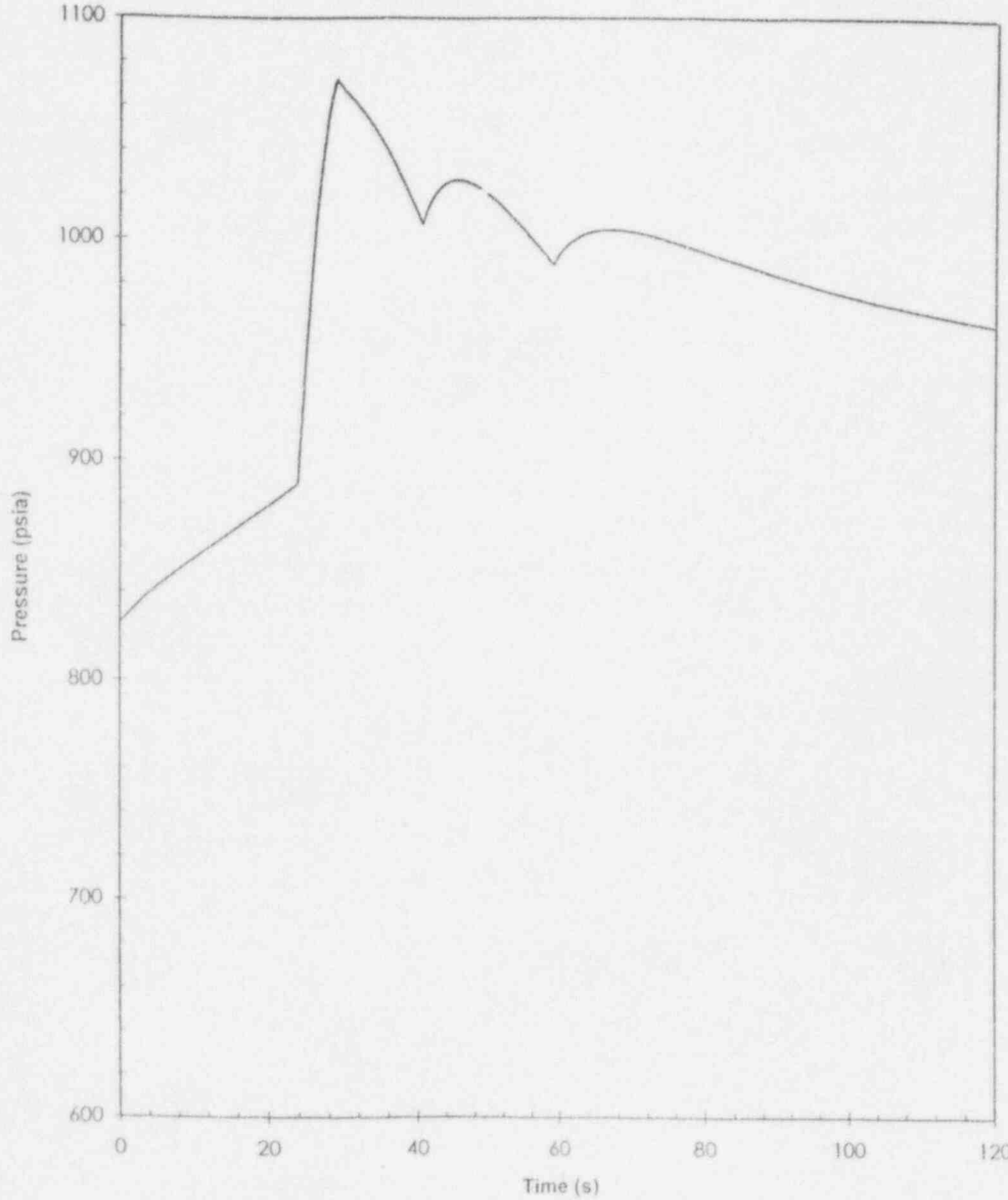


FIGURE 14.6-5

LOSS OF FEEDWATER FLOW EVENT
SG PRESSURE VS TIME



UFSAR Section 14.9, Loss of Coolant Flow Event

A. Event Description:

The Loss of Coolant Flow Event is initiated by a loss of forced reactor coolant flow through the core with offsite power available, but without a seized RCP rotor. The most limiting Loss of Coolant Flow Event involves a concurrent loss of power to all four RCPs. The immediate system response to the coastdown of all four RCPs is a rapid decrease in coolant mass flow rate through the core. The decreasing mass flow rate causes an increase in core and coolant temperatures, and an increase in core power due to a positive MTC. The fuel SAFDLs for DNB and LHR are approached.

Depending on the initial core flow, the Low Coolant Flow analysis trip setpoint, as sensed by the SG differential pressure transmitters, is reached in approximately one second. After an appropriate delay time for trip signal processing and holding coil decay, the CEAs are inserted into the core and the core power increase is terminated. The core heat flux continues to increase for a short time after the power peak due to the fuel time constant and increased enthalpy. The DNBR transient will terminate within three to five seconds of event initiation. Since the RCS loop cycle time is 10 seconds, SG parameters remain essentially constant during the DNBR transient. After the SDBS actuates, the RCS will stabilize at a hot-zero-power condition.

B. Analysis:

The Loss of Coolant Flow Event was analyzed for the reduced initial coolant flow condition using the HERMITE code to determine the reactor core response. Reference (1) provides a detailed description of HERMITE and has been reviewed and approved by the NRC. However, the HERMITE methodology has not been previously used at Calvert Cliffs for analysis of this event and, therefore, will require NRC approval. Previous analyses of the Loss of Coolant Flow Event at Calvert Cliffs have used the STRIKIN and CESEC codes to calculate the core power and heat flux. A brief description of HERMITE is provided below.

HERMITE Computer Code

The HERMITE computer code is used to determine the reactor core response during the postulated Loss of Coolant Flow Event. HERMITE can accept as input the transient boundary conditions of coolant flow rate, inlet coolant temperature, RCS pressure, and CEA position. In this application, HERMITE solves the few-group, space, and time-dependent neutron diffusion equation, including the feedback effects of fuel temperature, coolant temperature, coolant density, and control rod motion for a one-dimensional average fuel bundle. The hot bundle fuel pin power density is related to the average bundle fuel pin power density by the time-dependent planar radial peaking factors. For the calculation of heat flux, heat conduction equations are solved by a finite difference method. Continuity and energy conservation equations are solved to determine the coolant temperature and density for the average and hot bundles.

The synthesis of the axial power distribution and the planar radial power peaking factors provide a conservative representation of the hottest fuel assembly during the Loss of Coolant Flow transient, including maximum three-dimensional power peaking effects. This technique yields a conservative prediction of the minimum DNBR which can occur during the Loss of Coolant Flow transient.

The use of HERMITE provides both a reactivity and axial power distribution credit when applied to Loss of Coolant Flow-type transients. The upper region of the core experiences a reduction in density with the decreasing RCS flow prior to the reactor trip. This decrease in density results in a global negative reactivity insertion, even before the reactor trip occurs.

This same decrease in density in the upper regions of the core acts to drive the axial power distribution of the core to a more bottom-peaked configuration. This has the effect of reducing the degradation in thermal margin, which would occur relative to modeling with a constant axial power distribution.

HERMITE is used to calculate the time-dependent hot bundle and core average axial heat flux for the transient. The time-dependent coolant flow data used as input to the HERMITE calculation is developed based on measured pump coastdown data. The data is conservatively adjusted for the effect of SG tube plugging using the COAST code.

The static thermal-hydraulic method of calculating the DNBR was used, as described in Reference (9), except that the CETOP code was used in place of the COSMO code. The CETOP code uses the CE-1 Critical Heat Flux correlation described in Reference (10) to calculate the limiting channel DNBR. This method has been previously approved by the NRC for Calvert Cliffs and is used in the current Loss of Coolant Flow analysis. CETOP receives the core average fuel bundle heat flux, core inlet coolant mass flux, core inlet coolant temperature, and RCS pressure at selected times during the transient. CETOP is then used to calculate DNBR at these times. No credit is taken for the RCS pressure increase caused by the RCS temperature increase when calculating DNBR.

The CESEC transient analysis was not performed to recalculate the peak RCS pressure, because this event is relatively benign in this regard and is bounded by a Loss of Load Event for peak RCS pressure.

The input parameters and initial conditions used in the Loss of Coolant Flow analysis are listed in Table 14.9-1. Figure 14.9-1 is the four-pump coastdown curve used in the analysis. The axial power distributions and associated control rod reactivity used in the analysis are based on the most limiting distributions postulated to occur within the Technical Specifications LCOs. A parametric analysis is performed for rodded and unrodded axial power distributions at beginning of cycle and end of cycle conditions.

C. **Results:**

Table 14.9-2 contains the sequence of events for the Loss of Coolant Flow Event. Figures 14.9-2 through 14.9-5 present the transient core power, core average heat flux, RCS temperatures, and RCS pressure behavior based on previous analysis using the CESEC code.

The results of the reanalysis show that the DNBR SAFDL is not exceeded. Therefore, analysis of the Loss of Coolant Flow Event for the reduced RCS flow condition demonstrates that the action of the Reactor Protective System, in conjunction with the Technical Specifications LCOs, will prevent exceeding the fuel SAFDLs, and no fuel failure will occur. As a result, the radiological consequences of the event are negligible compared to the 10 CFR Part 100 guidelines.

TABLE 14.9-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR LOSS-OF-COOLANT FLOW EVENT

PARAMETER	UNITS	UNIT 1	UNIT 2
Minimum Analysis RCS Flow Trip Setpoint	gpm	90% of 340,000	90% of 340,000
Initial Core Power Leve	MWt	2700[b]	2700[b]
Initial Core Inlet Coolant Temperature	°F	550[b]	550[b]
Initial Core Mass Flow Rate	x 10^6 lbm/hr	123.36	123.36
RCS Pressure	psia	2250[b]	2250[b]
MTC	x 10^{-4} Δρ/°F	+0.15	+0.15
Doppler Coefficient Multiplier	---	1.3[a]	1.3[a]
Loss Flow Trip Resp...se Time	sec	0.5	0.5
CEA Holding Coil Delay	sec	0.5	0.5
CEA Time to 90% Insertion (Including Holding Coil Delay)	sec	3.1	3.1
CEA Worth at Trip (all rods out)	%Δρ	-5.60	-5.60
Unrodded Radial Peaking Factor (F_r^T)	---	1.4[c]	1.4[c]
Axial Shape Index	I_p	+0.25	+0.25
4-Pump RCS Flow Coastdown	---	Figure 14.9-1	Figure 14.9-1

(a) Consistent with the HERMITE methodology, a beginning of cycle Doppler with an end of cycle Doppler multiplier is used.

(b) For DNBR calculations, effects of uncertainties on the parameters were combined statistically.

(c) Consistent with the HERMITE methodology, a small value of F_r^T is used.

TABLE 14.9-2

SEQUENCE OF EVENTS FOR LOSS-OF-COOLANT FLOW EVENT

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.00	Loss of Power to all Four RCPs	---
1.23	Low Flow Trip Signal Generated	306,000[(a)] gpm
1.73	Trip Breakers Open	---
2.23	CEAs Begin to Drop Into Core	---
3.35[(b)]	Minimum CE-1 DNBR	> 1.21
5.70	Maximum RCS Pressure	2308[(c)] psia

(a) Value is 90% of 340,000 gpm.

(b) Time of minimum DNBR is based on an Axial Shape Index of +0.25. This time value may vary with Axial Shape Index.

(c) Value is based on previous CESEC Nuclear Steam Supply System simulation.

FIGURE 14.9-1

LOSS OF COOLANT FLOW EVENT
CORE FLOW FRACTION VS TIME

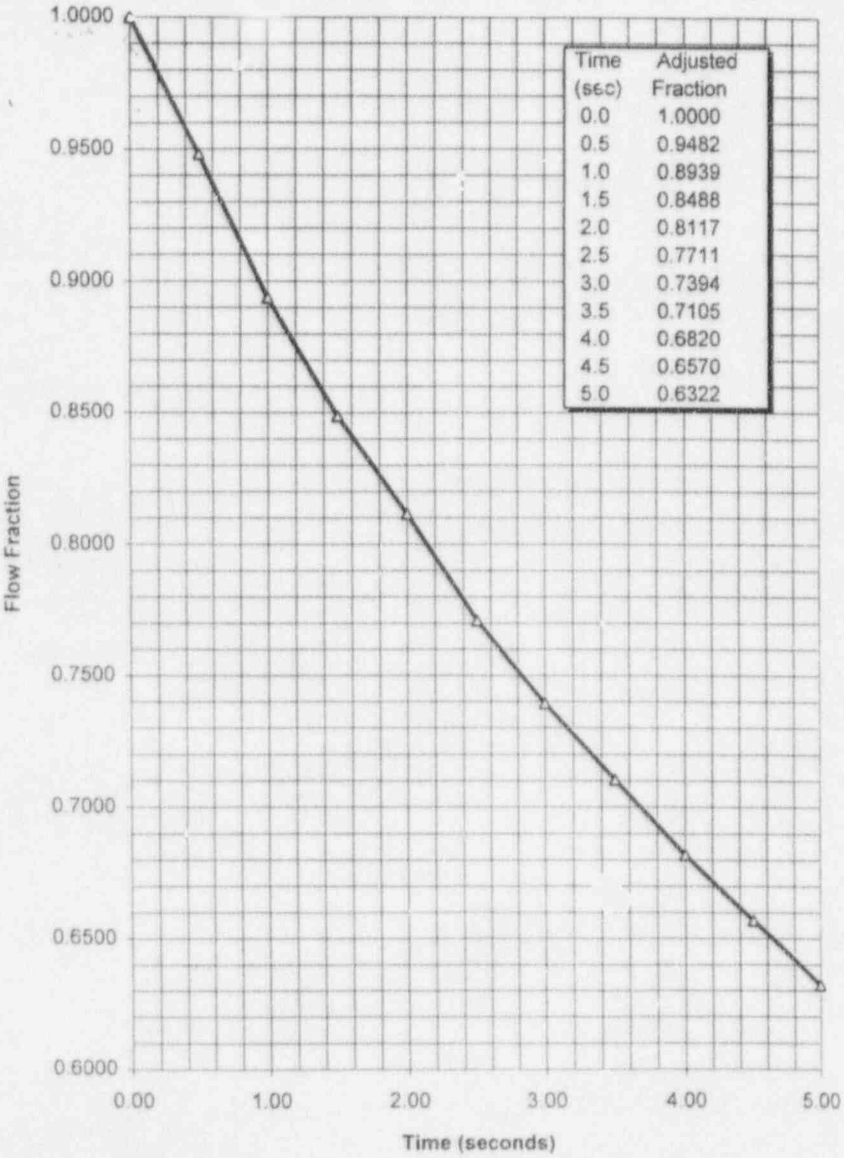


FIGURE 14.9-2

LOSS OF COOLANT FLOW EVENT
CORE POWER VS TIME

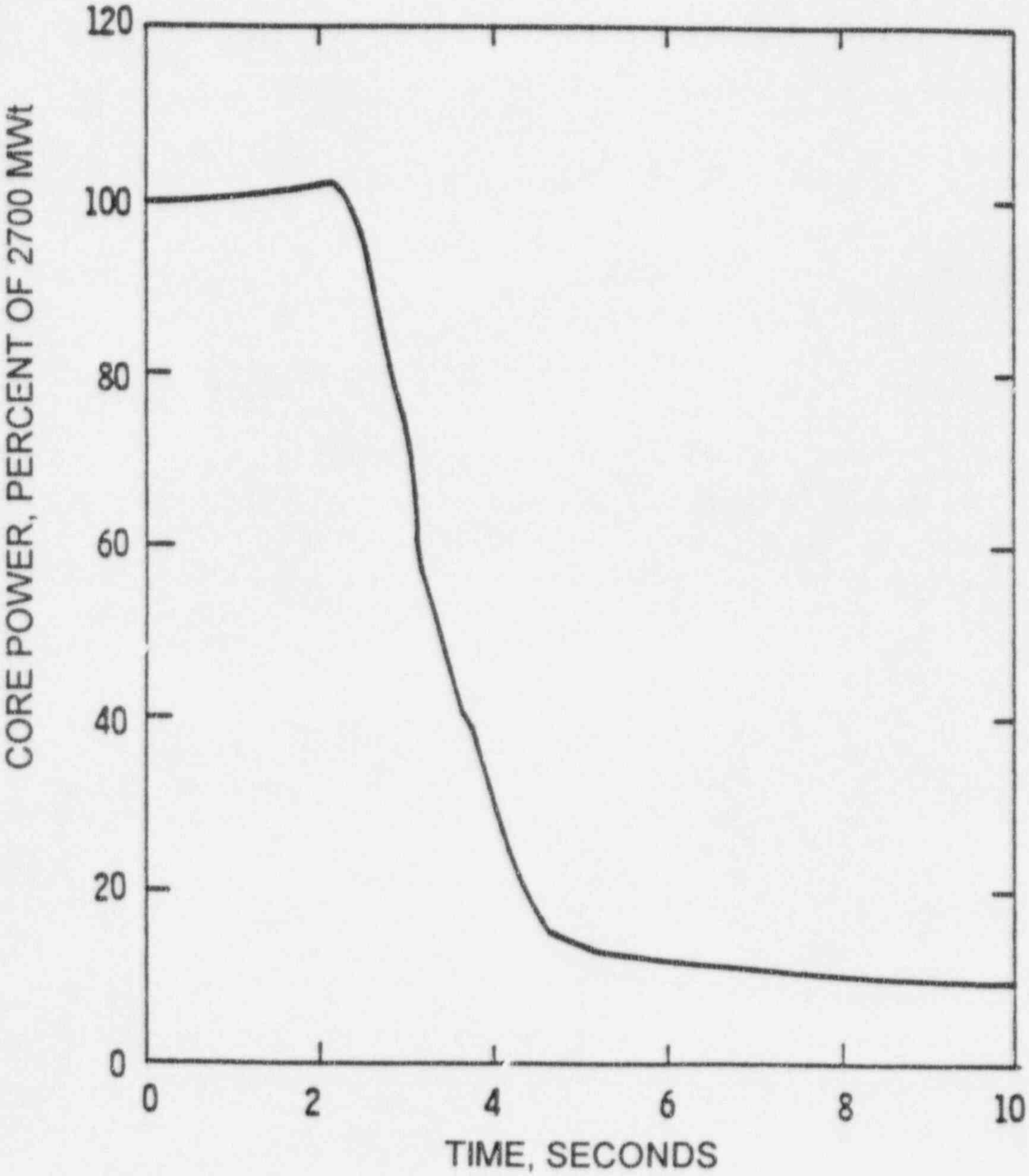


FIGURE 14.9-3

LOSS OF COOLANT FLOW EVENT
CORE HEAT FLUX VS TIME

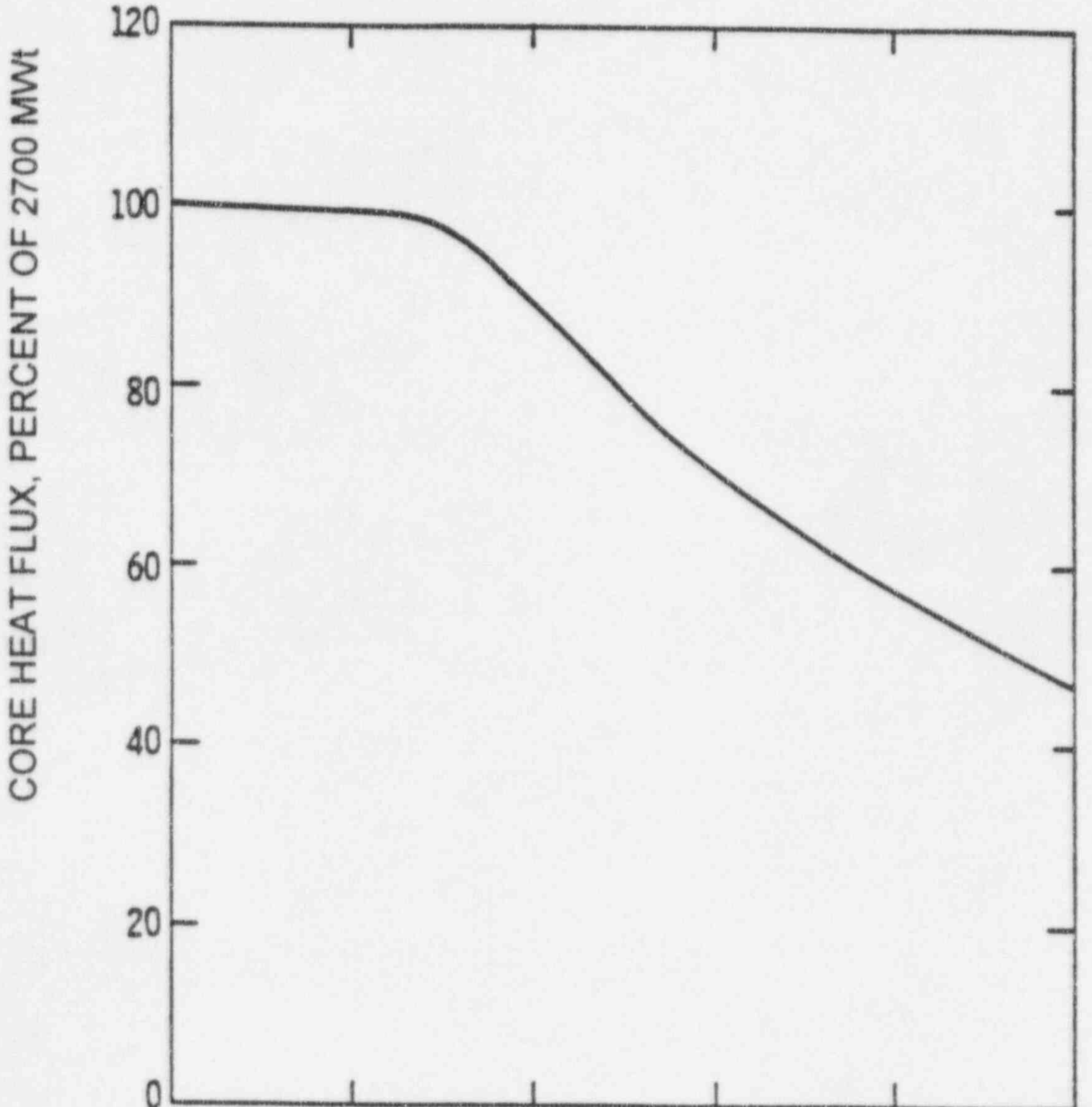


FIGURE 14.9-4

LOSS OF COOLANT FLOW EVENT
RCS TEMPERATURES VS TIME

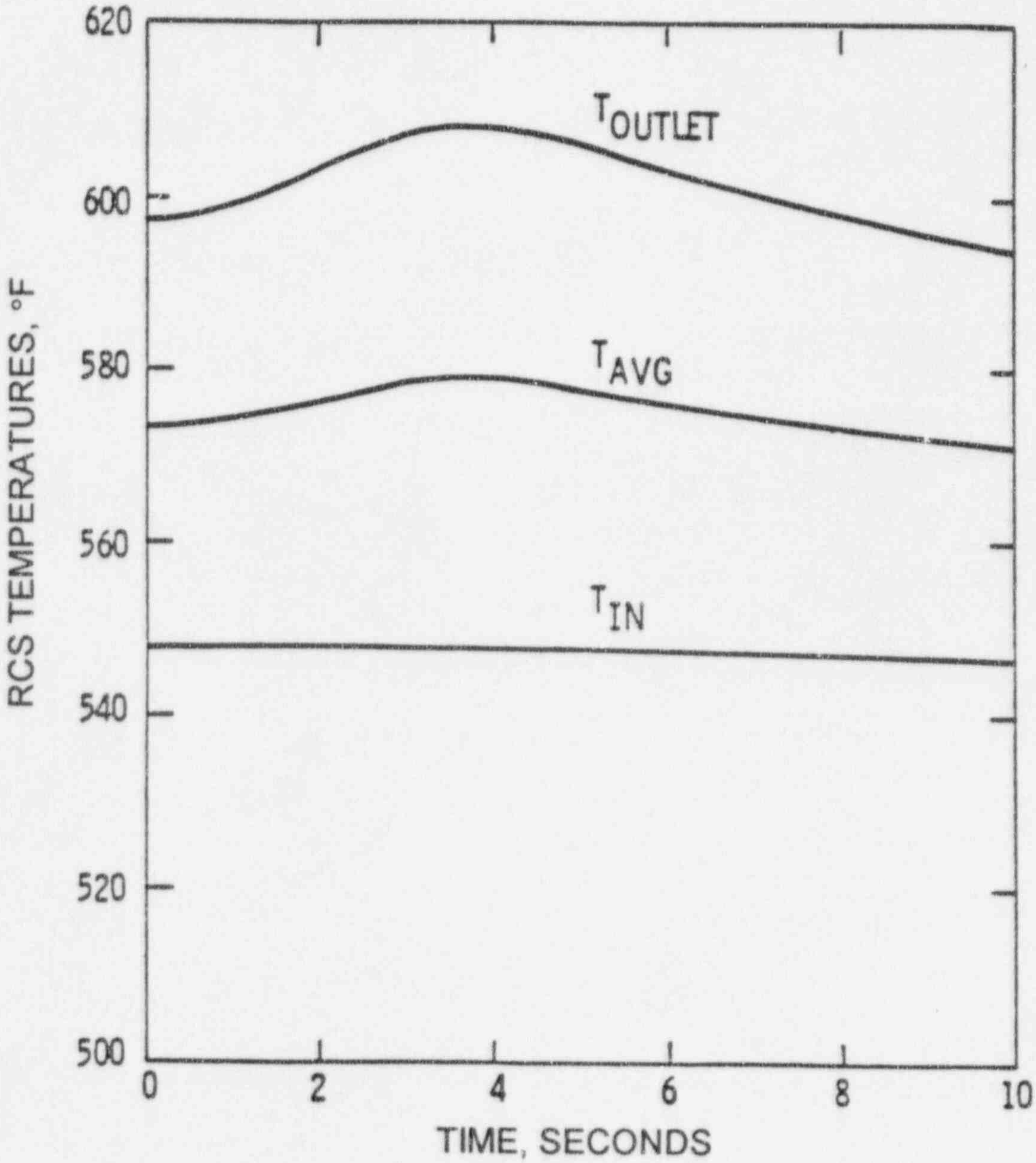
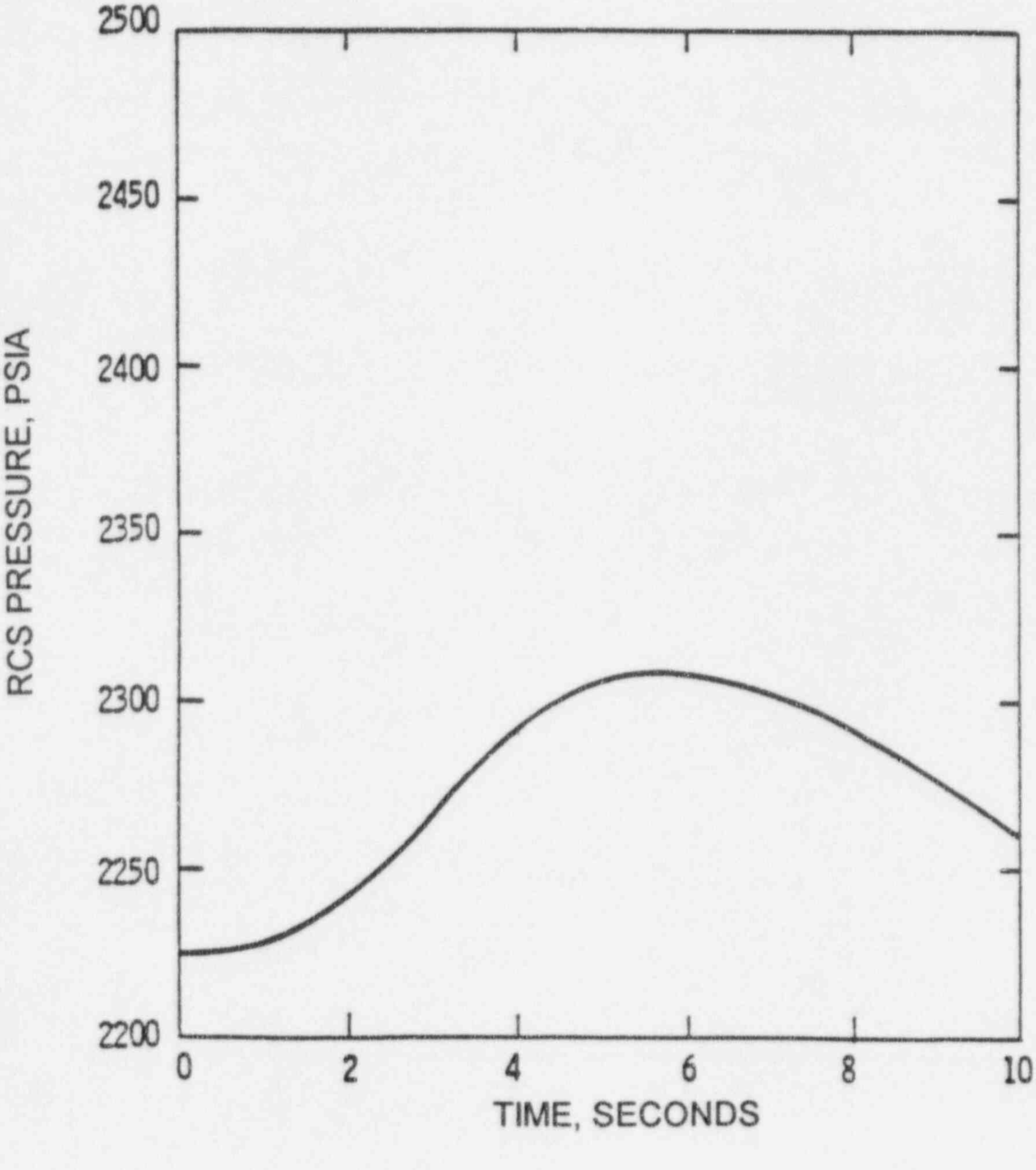


FIGURE 14.9-5

LOSS OF COOLANT FLOW EVENT
RCS PRESSURE VS TIME



UFSAR Section 14.14, Main Steam Line Break Event

A. Event Description

The MSLB Event analyzes a break in a main steam line. An MSLB may occur as a result of thermal stress or cracking. The break increases the rate of steam extraction by the SGs and causes an RCS temperature reduction. With a negative MTC, the RCS temperature reduction will produce a positive reactivity insertion. Two analyses are performed to evaluate the reactivity insertion: the post-trip analysis and the pre-trip analysis.

The post-trip analysis evaluates the system response and core reactivity balance long after insertion of the CEAs. The cooldown associated with a large break, combined with the most negative MTC, may result in enough positive reactivity insertion to overcome the negative reactivity inserted by the CEAs, and present the possibility of a return-to-power. The main steam line flow restrictors limit the steam flow and resultant cooldown for a break outside of the containment. Therefore, the post-trip MSLB analysis is performed for a main steam line break inside the containment. A post-trip MSLB analysis was performed to analyze the effect of reduced primary coolant mass flow associated with an increase in the number of plugged SG tubes. An increase of 2°F in the core inlet temperature was considered. The results of the post-trip MSLB analysis demonstrate that the site boundary dose is less than 10 CFR Part 100 guidelines, and the core remains coolable; therefore, the post-trip MSLB results do not constitute a USQ.

The pre-trip MSLB analysis evaluates DNBR at or near the time of trip. The pre-trip MSLB analysis involves parametric analysis on a range of break sizes and MTC values to determine the combination that leads to the most limiting DNBR and highest fuel failure rate. Breaks inside containment and outside containment are considered. Inside containment breaks are considered due to the instrument uncertainty degradation associated with a harsh environment. Degradation of instrument uncertainties may result in an additional delay between event initiation and reactor trip.

The MSLB Event is classified as a Postulated Accident. The action of the Limiting Safety System Settings, in conjunction with the LCOs, will limit the number of fuel pins that experience DNB. Site boundary dose must not exceed the 10 CFR Part 100 limits. Since the event involves an RCS temperature reduction, primary system pressure decreases, and the RCS pressure upset limit is not approached.

B. Analysis

The objectives of the pre-trip MSLB Event are to demonstrate that a coolable core geometry is maintained and that the site boundary dose is within the 10 CFR Part 100 limits.

The pre-trip MSLB transient analyses include a parametric analysis to determine the limiting break size and MTC combination that lead to the highest power at the time of reactor trip. The trips that are credited are the Variable High Power trip, Low SG Pressure trip, Low SG Level trip, and High Containment Pressure trip. As the Variable High Power trip signal is based upon the excore detectors, the decalibration of the detectors that occurs as the reactor coolant temperature decreases is considered.

The core mass flow used in the analysis reflects the reduction in flow associated with an increase in the number of plugged SG tubes. The core inlet temperature was increased by 2°F. A loss of AC power was

assumed to occur three seconds after the time of reactor trip. The primary effect of a loss of AC power is that primary flow decreases. If the flow reduction begins prior to CEA insertion, the power (heat flux) to flow ratio will increase, and a lower DNBR will result. A delay of three seconds between the reactor trip and the beginning of flow reduction allows the heat flux to begin to decrease before the flow reduction begins. The three second delay is based upon the time that it takes for the grid voltage to decay to the point where power to the RCPs is lost. The three-second time delay is considered conservative as an evaluation performed by BGE determined that the actual grid decay time would be at least 20 seconds.

The BGE 500 kV network is configured to withstand the worst single contingency (i.e., unit trip), without jeopardizing system stability, under any operating condition. Enough spinning reserve is maintained on the system such that it will not break up into islands. Therefore, the BGE 500 kV network is not operated under conditions where the trip of a Calvert Cliffs unit would challenge system stability. In addition, following a unit trip at Calvert Cliffs, the turbine generator remains connected to the 500 kV network for 20 seconds after a reverse power relay detects power flow from the 500 kV network to the turbine generator. This feature ensures the turbine generator will not accelerate when the generator output breakers are opened. Therefore, for any 500 kV bus configuration, the RCP motors will continue to run for at least 20 seconds following the unit trip caused by an MSLB.

The calculation of the percentage of fuel failures used convolution methodology. This methodology was previously approved for use for the MSLB Event in Reference (11). The potential for DNB propagation during the pre-trip MSLB was also evaluated. Departure from Nucleate Boiling propagation would result in increased offsite dose due to additional fuel failure. The evaluation concluded that the strain prediction was less than previously calculated. Since DNB propagation is not expected to occur, no additional fuel failures due to DNB propagation are postulated. As the number of fuel failures is limited, and DNB propagation is postulated not to occur, a coolable core geometry is maintained.

The calculation of the offsite dose uses the Technical Specification primary-to-secondary leakage rate limit of 100 gallons per day per SG. The site boundary atmospheric dispersion coefficient used is 1.3×10^{-4} second/m^3, and the breathing rate is 3.47×10^{-4} m^3/second. The dose conversion factors from [International Committee on Radiation Protection] ICRP-30 were used. These dose conversion factors have previously been approved for the Calvert Cliffs Fuel Handling Incident (Reference 9).

C. Results

The results of the pre-trip MSLB analysis show that the predicted number of fuel pin failures is limited to ≤ 10%. The potential for DNB propagation was evaluated, and was determined not to occur. As the number of fuel failures is limited, and DNB propagation is postulated not to occur, a coolable geometry is maintained. The resultant site boundary doses, based on 10% fuel failures, are:

0 - 2 Hr Thyroid Dose	=	40 REM
0 - 2 Hr Whole Body Dose	=	1.3 REM

Table 14.14-1 provides the initial conditions and input parameters. Table 14.14-2 provides the assumptions used for the MSLB Event dose calculation. Table 14.14-3 provides the sequence of events. Figures 14.14-1 through 14.14-7 plot the following parameters for the MSLB Event: Moderator Reactivity versus Moderator Density, Reactor Power and Heat Flux versus Time, Core Average Heat Flux and Core Average Flow versus Time, RCS Pressure versus Time, RCS Temperatures versus Time, Reactivities versus Time, and SG Pressures versus Time.

TABLE 14.14-1

INITIAL CONDITIONS AND INPUT PARAMETERS ASSUMED THE OUTSIDE-CONTAINMENT MAIN STEAM LINE BREAK EVENT INITIATED FROM HOT FULL POWER

PARAMETER	UNITS	VALUE
Initial Core Power	MWt	2754
Initial Core Inlet Temperature	°F	550
Initial RCS Pressure	psia	2300
Initial SG Pressure	psia	865
Low SG Pressure Analysis Trip Setpoint	psia	640
Minimum CEA Worth Available at Trip	% $\Delta\rho$	-5.4
Doppler Multiplier	---	0.85
Moderator Cooldown Curve	% $\Delta\rho$ vs density	Figure 14.14-2
Most Negative MTC	x 10^{-4} $\Delta\rho$/°F	-3.0
Beta Fraction (Including Uncertainty)	---	.0044
Number of Plugged Tubes per SG	---	0[a]

[a] Up to 2500 plugged tubes per SG was considered; zero plugged tubes is more limiting.

TABLE 14.14-2

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION OF THE MAIN STEAM LINE BREAK EVENT

PARAMETER	UNITS	VALUE
RCS Maximum Allowable Concentration (DEQ I-131)[a]	μCi/gm	1.0
Secondary Maximum Allowable Concentration (DEQ I-131)[a]	μCi/gm	0.1
Partition Factor Assumed for All Doses	---	1.0
Atmospheric Dispersion Coefficient[b]	sec/m^3	1.3×10^{-4}
Breathing Rate	m^3/sec	3.47×10^{-4}
Dose Conversion Factors	REM/Ci	[C]

(a) Technical Specification limits.
(b) 0-2 hour accident condition.
(c) Dose conversion factors obtained from ICRP-30.

TABLE 14.14-3

SEQUENCE OF EVENTS FOR OUTSIDE CONTAINMENT STEAM LINE BREAK

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	Steam Line Break Occurs	0.75 ft^2
58.2	Minimum DNBR Occurs	0.967
59.9	Variable High Power Trip Generated	112.2%
60.8	Trip Breakers Open	---
61.3	CEAs Begin to Drop into Core	---
61.3	Maximum Core Power Occurs	138.6% of rated power
70.1	Main Steam Isolation Valves Close	---

FIGURE 14.14-1

MSLB EVENT OUTSIDE CONTAINMENT
MODERATOR REACTIVITY VS MODERATOR DENSITY

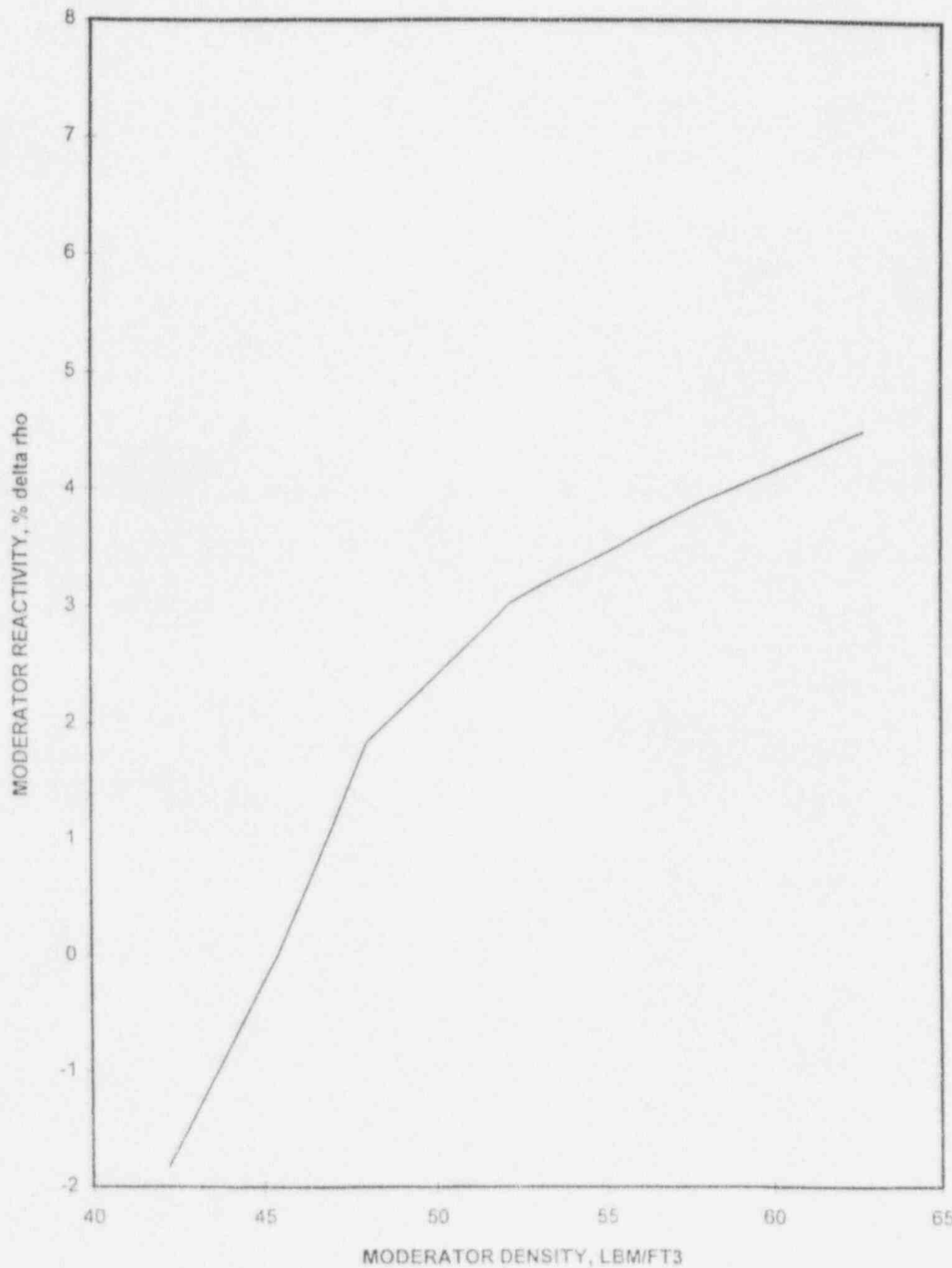


FIGURE 14.14-2

MSLB BREAK EVENT OUTSIDE CONTAINMENT
CORE POWER AND HEAT FLUX VS TIME

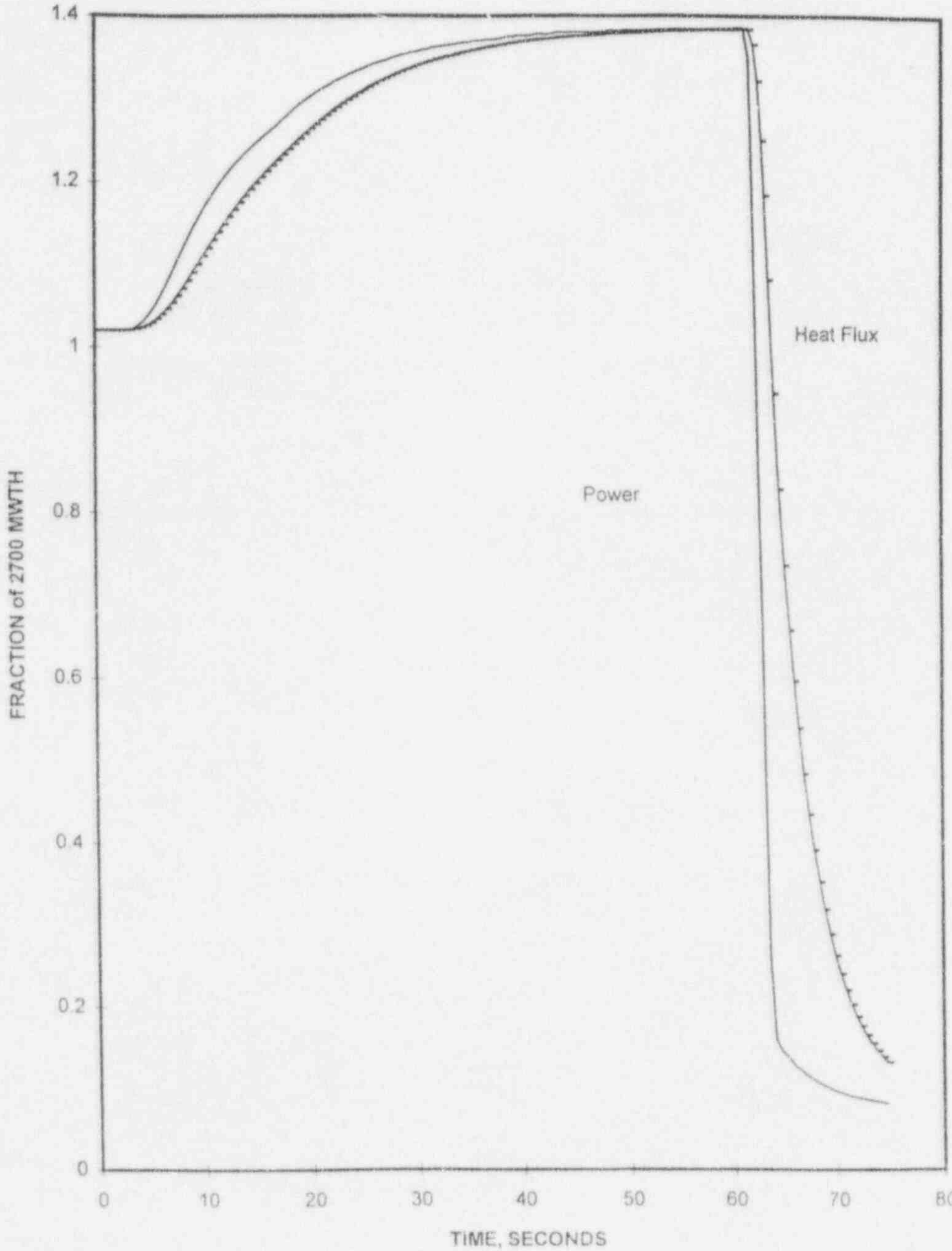


FIGURE 14.14-3

MSLB BREAK EVENT OUTSIDE CONTAINMENT
CORE FLOW AND HEAT FLUX VS TIME

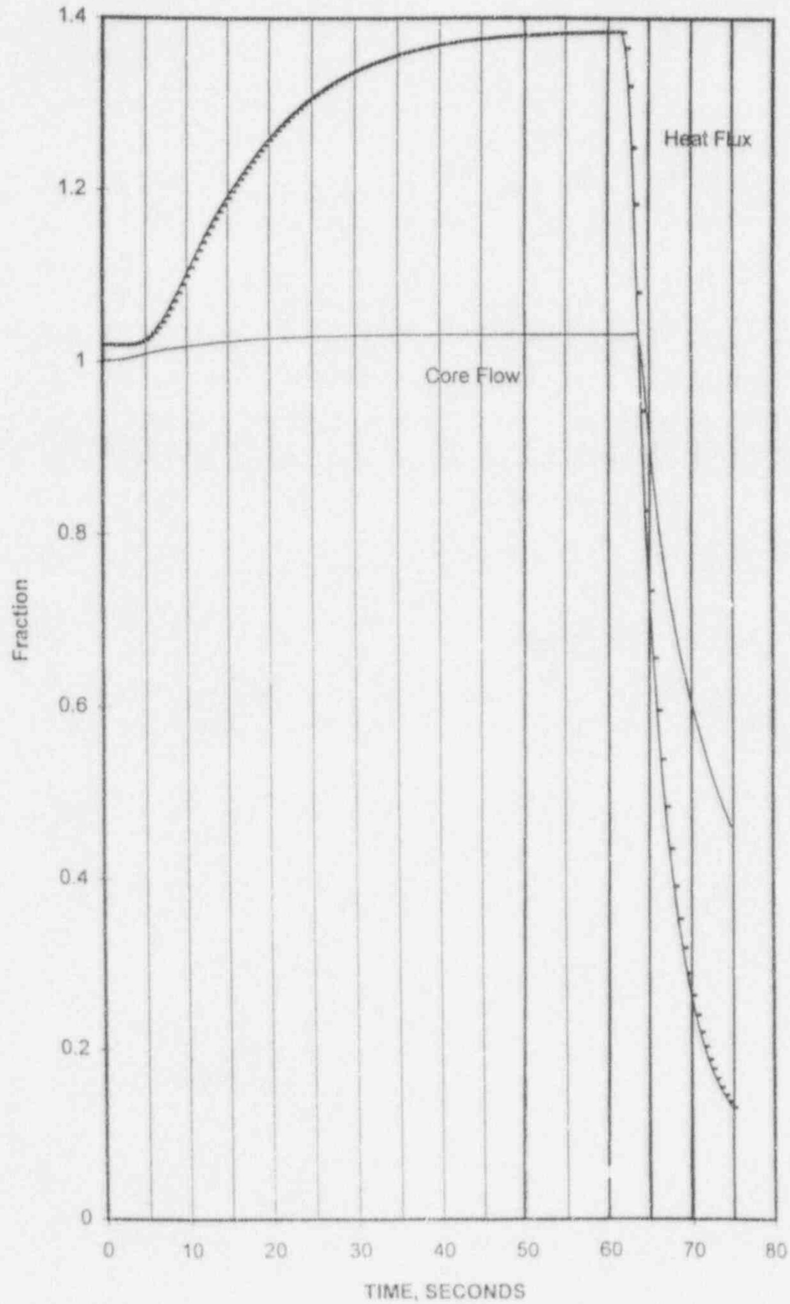


FIGURE 14.14-4

MSLB BREAK EVENT OUTSIDE CONTAINMENT
RCS PRESSURE VS TIME

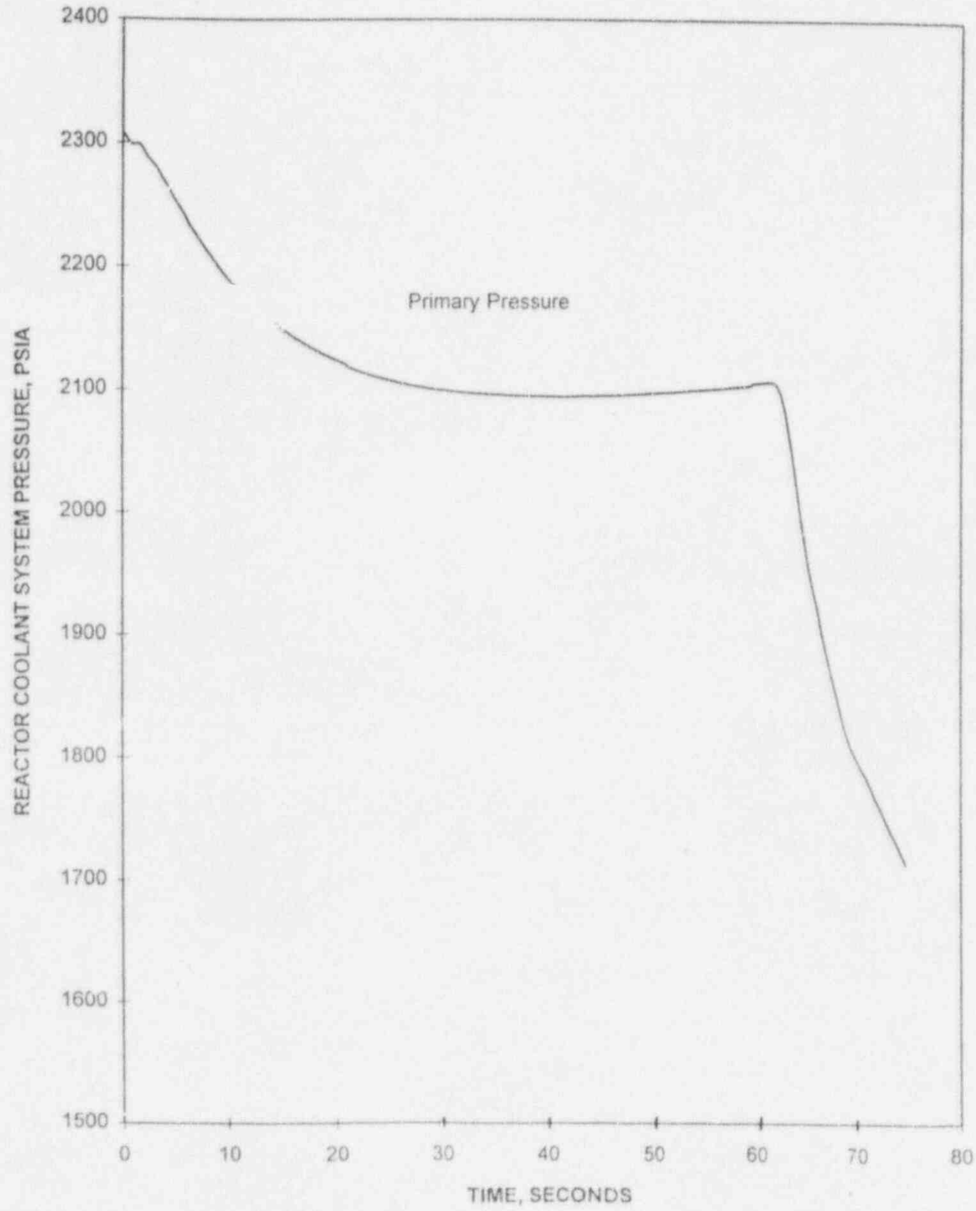


FIGURE 14.14-5

MSLB EVENT OUTSIDE CONTAINMENT
RCS TEMPERATURES VS TIME

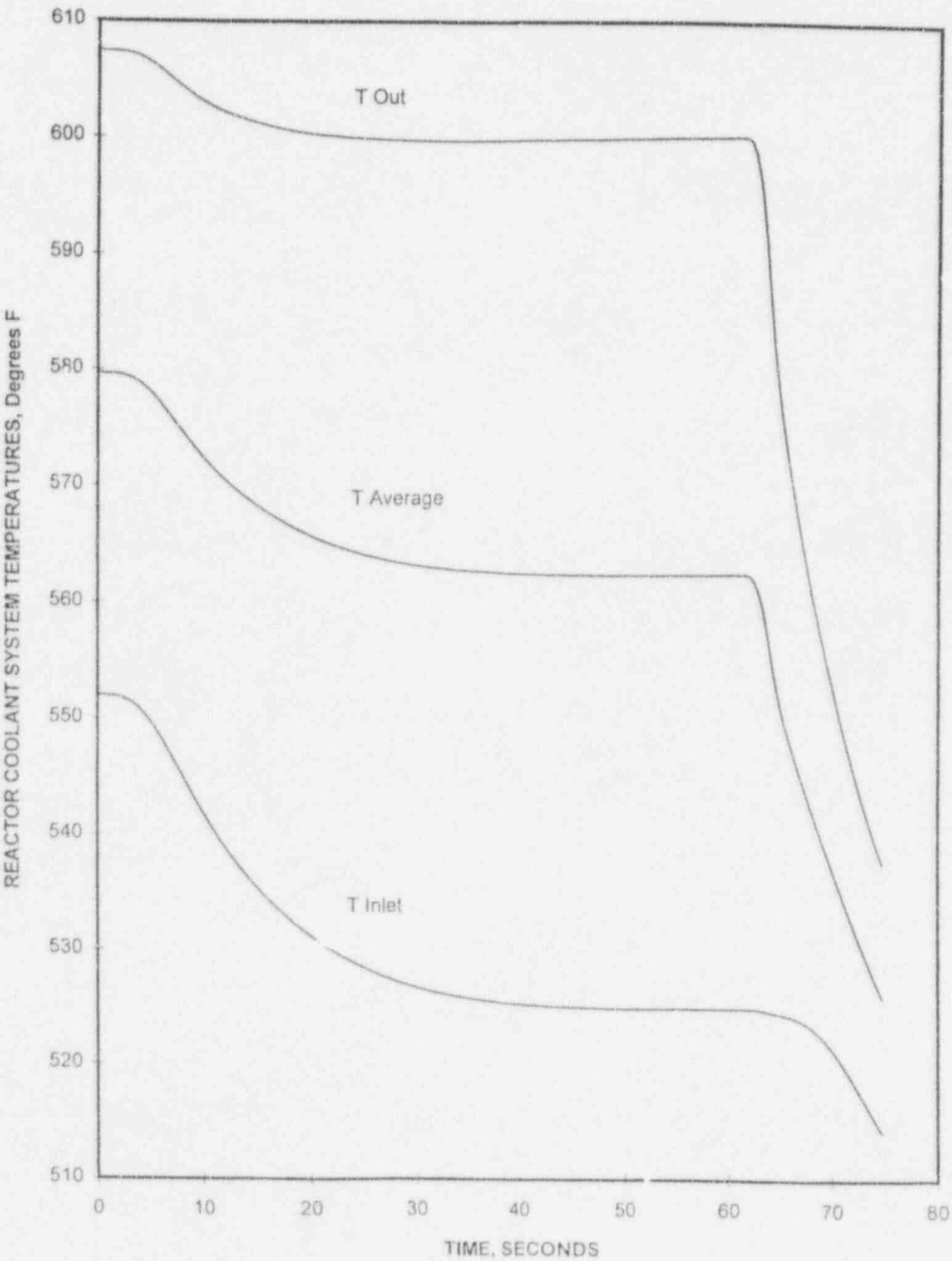


FIGURE 14.14-6

MSLB EVENT OUTSIDE CONTAINMENT
REACTIVITIES VS TIME

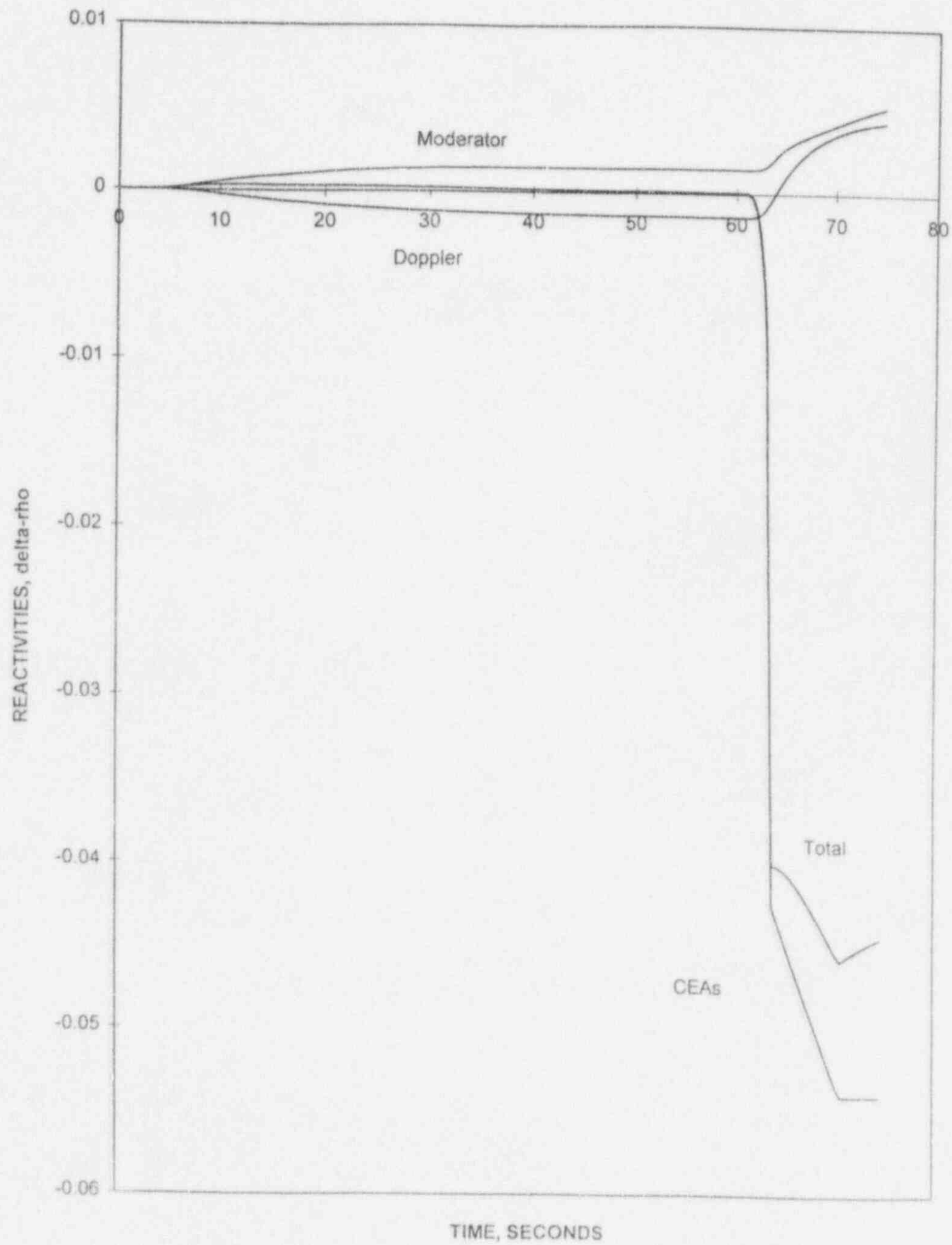
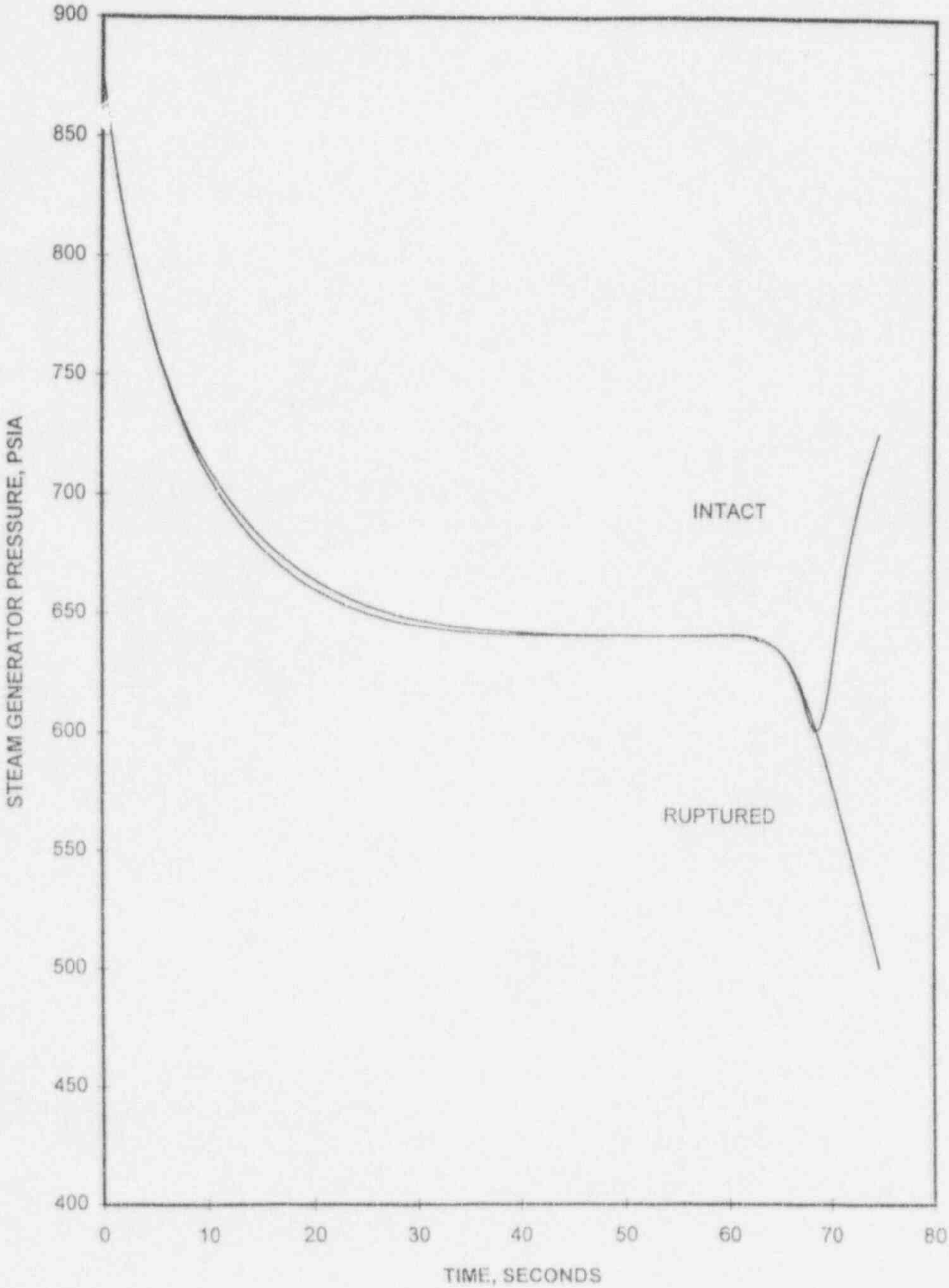


FIGURE 14.14-7

MSLB EVENT OUTSIDE CONTAINMENT
SG PRESSURE VS TIME



UFSAR Section 14.16, Seized Rotor Event

A. Event Description

A Seized Rotor Event is defined as a complete seizure of a single RCP shaft. The seizure is postulated to occur due to a mechanical failure or a loss of component cooling to the pump shaft seals. The most limiting Seized Rotor Event is an instantaneous RCP shaft seizure at hot full power. The reactor coolant flow through the core would be asymmetrically reduced to three-pump flow as a result of a shaft seizure on one pump. Due to the rapid reduction in core flow, the coolant temperatures will increase. Assuming a positive MTC, the core power and the heat flux will also increase. After appropriate delays, the insertion of the CEAs will terminate the power rise. The heat flux will then be reduced to the decay heat level.

The Seized Rotor Event primarily challenges the acceptance criteria on primary system over-pressurization and DNBR. The action of the Limiting Safety System Setting, in conjunction with the LCOs, will limit the number of fuel pins that experience DNB. Since minimum DNBR occurs within several seconds of event initiation, the secondary side (i.e., SGs and steam systems) does not affect the DNBR analysis.

B. Analysis

The objectives of the analysis are to demonstrate that the primary pressure is limited to less than 110% of the design pressure (110% of 2500 psia) and that the site boundary doses are within the 10 CFR Part 100 limits, and that a coolable core geometry is maintained. Two transient analyses are normally performed. The first analysis determines the DNBR and calculates the number of fuel pins that fail. The second analysis determines peak primary pressure and provides steam release data for the site boundary dose calculation.

The transient simulation of the Seized Rotor Event analysis that provides input to the DNBR analysis was not performed. A conservative DNBR analysis was performed that used the core axial shape from the previous transient analysis, and did not credit a reduction in core heat flux that was shown to occur in the previous analysis. The initial margin was based upon the minimum thermal margin reserved by the LCOs. The initial core mass flow considered the decrease in flow associated with an increase in the number of plugged SG tubes. The transient mass flow was instantaneously reduced to three-pump flow and the core inlet flow distribution associated with three-pump flow was used. The initial core inlet temperature was increased by 2°F. Overall, the use of the previous transient analysis results combined with neglecting the reduction in core heat flux is a conservative approach to the DNBR calculation.

The convolution methodology was used to calculate the percentage of fuel failure. This methodology was previously approved for use for the Seized Rotor Event in Reference (12).

The number of fuel pins predicted to fail was limited to $\leq 5\%$. The potential for DNB propagation during the Seized Rotor Event was also evaluated. Departure from nucleate boiling propagation would result in increased offsite dose due to additional fuel failure. The minimum period of time in DNB (< 5 seconds) experienced during the Seized Rotor transient effectively minimizes the potential for DNB propagation, so no additional fuel failure is postulated to occur. Since the fuel failure is limited and DNB propagation is postulated not to occur, a coolable core geometry is maintained.

The transient analysis to determine the peak primary pressure and to provide steam release data for the site boundary dose calculation included the decrease in core mass flow and the increase in core inlet temperature. The uncertainties were selected to maximize the steam released.

The offsite dose calculation uses the Technical Specification primary-to-secondary leakage rate limit of 100 gallons per day per SG. The site boundary atmospheric dispersion coefficient used is 1.3×10^{-4} second/m^3 and the breathing rate is 3.47×10^{-4} m^3/second. The dose conversion factor in equivalent I-131 is 1.48×10^6 REM/Ci.

C. Results

The results of the DNBR analysis show that the Reactor Coolant Flow - Low trip, in conjunction with the DNB LCO, limits the predicted number of fuel pin failures to ≤ 5%. The resultant site boundary doses are:

0 - 2 Hr Thyroid Dose	=	12 REM (DEQ I-131)
0 - 2 Hr Whole Body Dose	=	0.06 REM (DEQ Xe-133)

The potential for DNB propagation was evaluated, and it was postulated that DNB propagation would not occur. Since the number of fuel pin failures is limited, and DNB propagation is postulated not to occur, a coolable core geometry is maintained.

The maximum RCS pressure experienced during the event is well below the upset pressure limit of 2750 psia.

Table 14.16-1 provides the initial conditions and input parameters. Table 14.16-2 provides the sequence of events. Table 14.16-3 delineates the assumptions used for the Seized Rotor Event dose calculation. Figures 14.16-1 through 14.16-4 plot the following parameters for this event: Reactor Power; Core Average Heat Flux; RCS Temperatures; and RCS Pressure versus Time.

TABLE 14.16-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR SEIZED ROTOR EVENT

PARAMETER	UNITS	UNIT 1	UNIT 2
Initial Core Power Level	MWt	2754[a] 2700[b]	2754[a] 2700[b]
Core Inlet Coolant Temperature	°F	552[a] 550[b]	552[a] 550[b]
4-Pump RCS Flow Rate[a], [b]	gpm	340,000	340,000
3-Pump RCS Mass Flow Fraction[b]	---	0.749	0.749
RCS Pressure	psia	2300[a] 2200[b]	2300[a] 2200[b]
MTC	$\times 10^{-4}\ \Delta\rho/^{\circ}F$	+0.15	+0.15
Doppler Coefficient Multiplier	---	0.85	0.85
CEA Worth at Trip (All Rods Out)	% $\Delta\rho$	-5.0	-5.0
Unrodded Integrated Radial Peaking Factor with Tilt; F_r^T	---	1.7	1.7
Axial Shape Index[b]	---	-0.23	-0.23

(a) Input parameters used in the CESEC code steam release/site boundary dose evaluation.

(b) Nominal input values were employed to calculate the minimum DNBR based upon the Extended Statistical Combination of Uncertainties conditions.

TABLE 14.16-2

SEQUENCE OF EVENTS FOR SEIZED ROTOR EVENT

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	Seizure of One RCP	---
0.0	Low Coolant Flow Signal Generated	90% of Initial 4-Pump Flow
0.50	Trip Breakers Open	---
1.00	CEAs Begin Dropping Into Core	---
3.59	Maximum RCS Pressure	2385 psia

TABLE 14.16-3

ASSUMPTIONS FOR SEIZED ROTOR DOSE CALCULATION

1.	During first 1800 seconds, there is no operator action. Thus, the steam releases are to the condenser via the turbine bypass valves, and to the atmosphere via the atmospheric dump valves.	
2.	Beyond 1800 seconds, steam dump is to the condenser via operator action.	
3.	Primary activity released to secondary is assumed to be released without mixing within the SG secondary liquid.	
4.	Primary-to-Secondary Leak Rate[a]	100 gal/day per SG
5.	Initial RCS Maximum Allowable Technical Specification Concentration (DEQ I-131)[a]	$1.0 \frac{micro\ Ci}{gm}$
6.	Initial Secondary Maximum Allowable Concentration (DEQ I-131)[a]	$0.1 \frac{micro\ Ci}{gm}$
7.	RCS Maximum Allowable Concentration of Noble Gases (DEQ Xe-133)[a]	$100 / \overline{E} \frac{micro\ Ci}{gm}$
8.	Partition Factor for Primary Release Through Atmospheric Dump Valves	1.0
9.	Partition Factor for Secondary Release Through Atmospheric Dump Valves	0.1
10.	Feed Pump Turbine Air Ejector Partition Factor	0.0005

TABLE 14.16-3 (Continued)

ASSUMPTIONS FOR SEIZED ROTOR DOSE CALCULATION

11.	Atmospheric Dispersion Coefficient	1.3×10^{-4} sec/m^3
12.	Breathing Rate	3.47×10^{-4} m^3/sec
13.	Dose Conversion Factor (I-131)	1.48×10^6 REM/Ci
14.	All activity in the gap is assumed to be released to the coolant	
15.	Calculated activity in the Gap Corresponds to a burnup of 75,000 MWD/MT	
16.	Percent Fuel Failure	5%

(a) Technical Specification Limits

FIGURE 14.16-1

SEIZED ROTOR EVENT
CORE POWER VS TIME

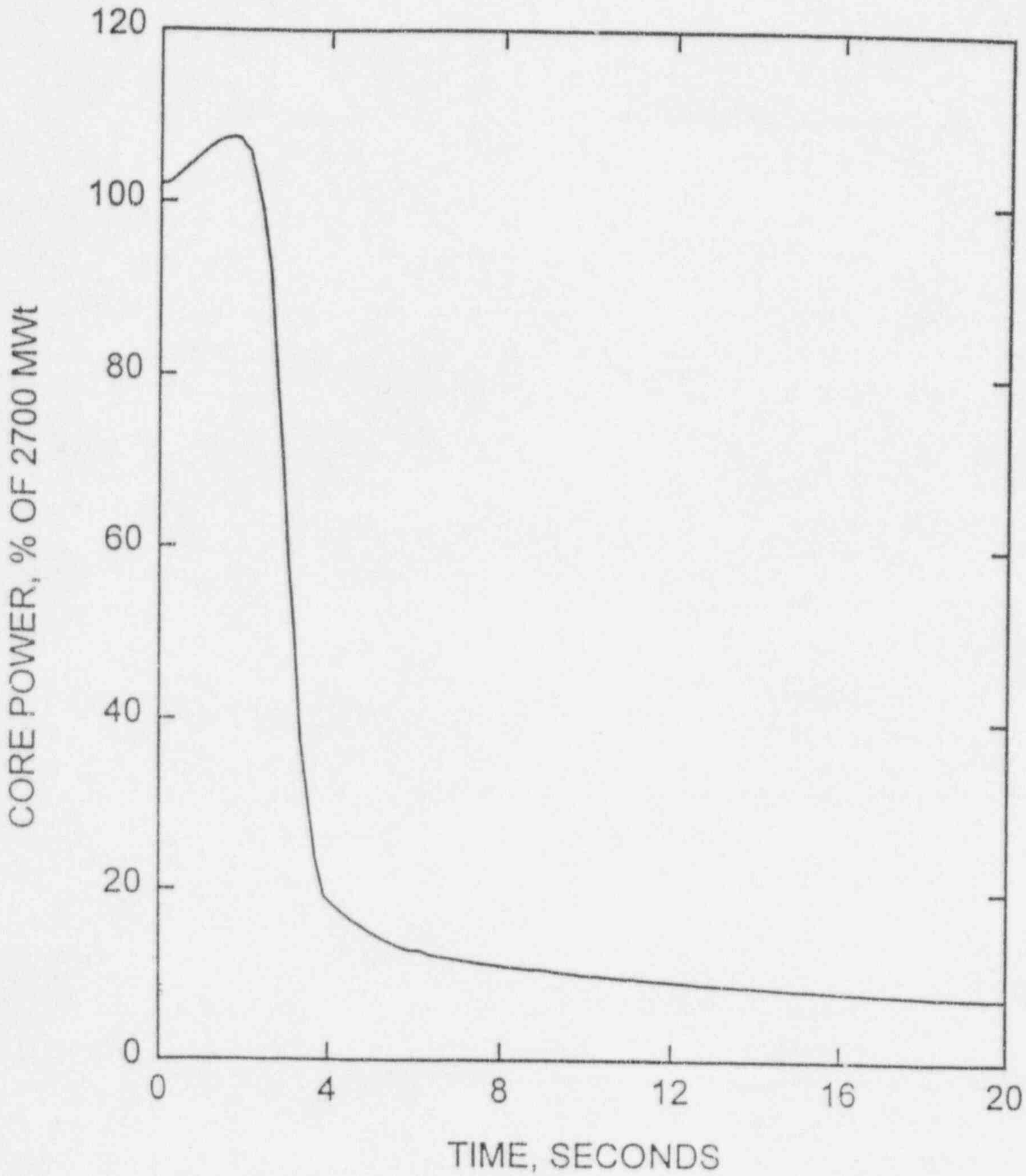


FIGURE 14.16-2

SEIZED ROTOR EVENT
CORE AVERAGE HEAT FLUX VS TIME

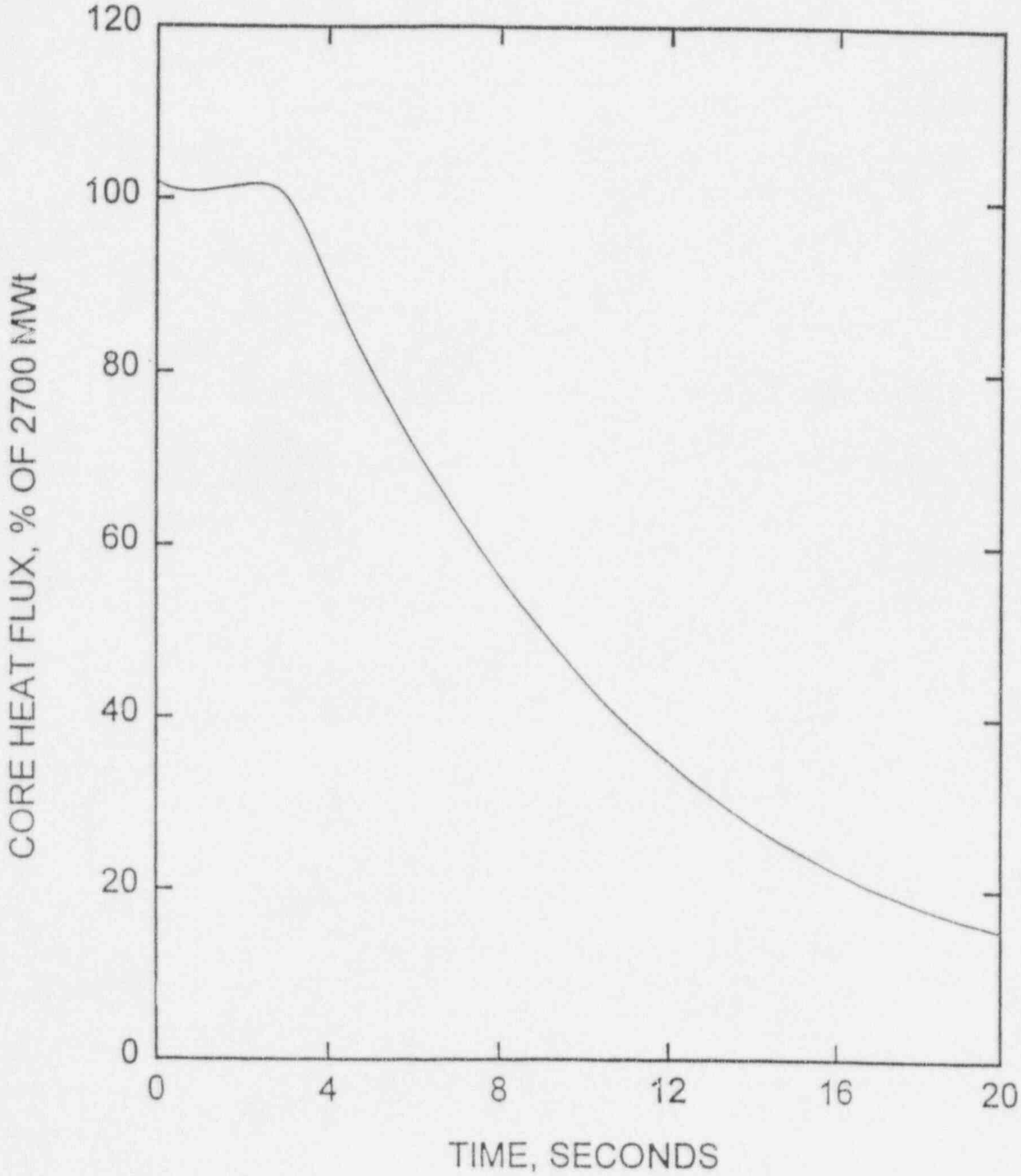


FIGURE 14.16-3

SEIZED ROTOR EVENT
RCS TEMPERATURES VS TIME

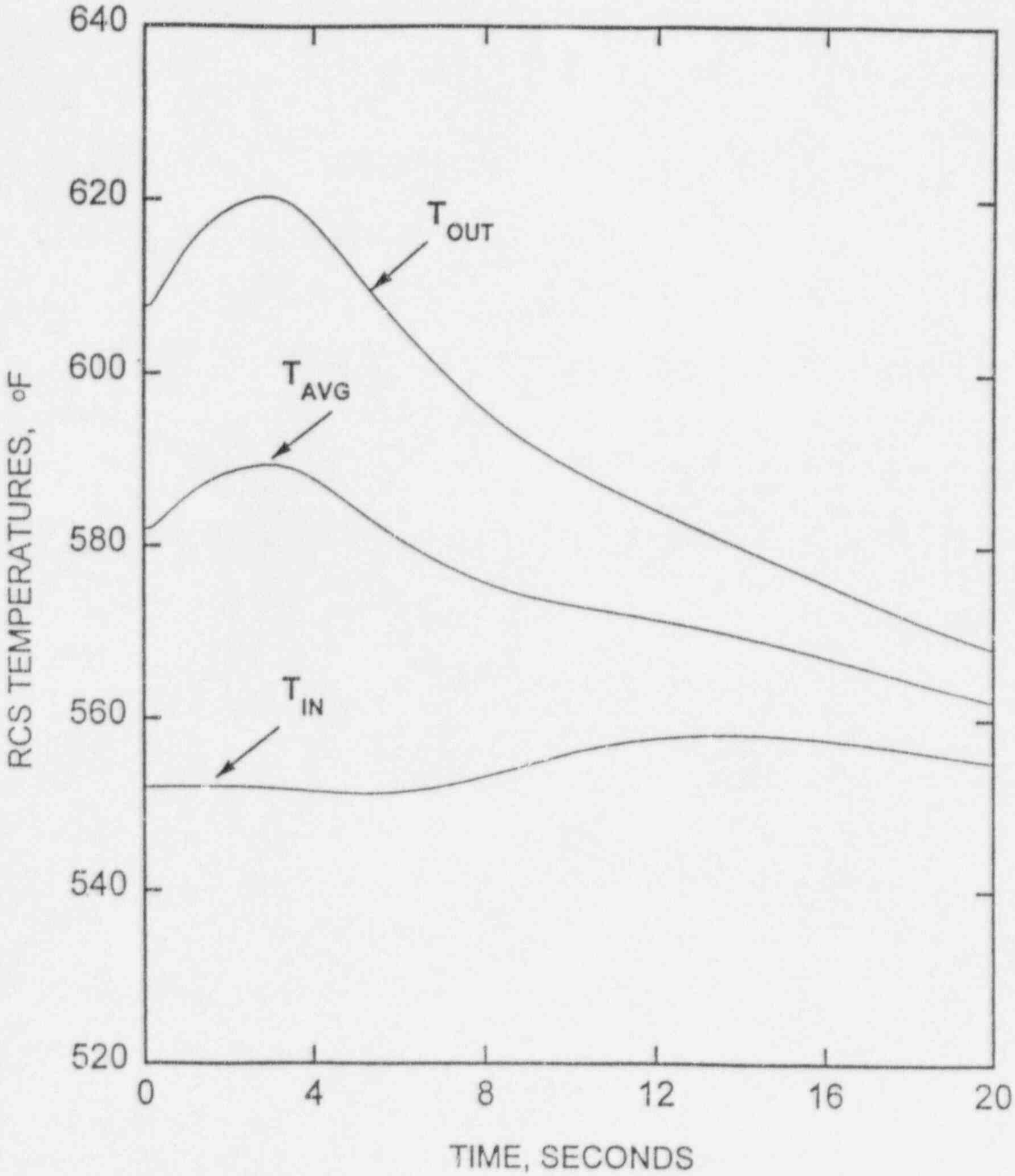
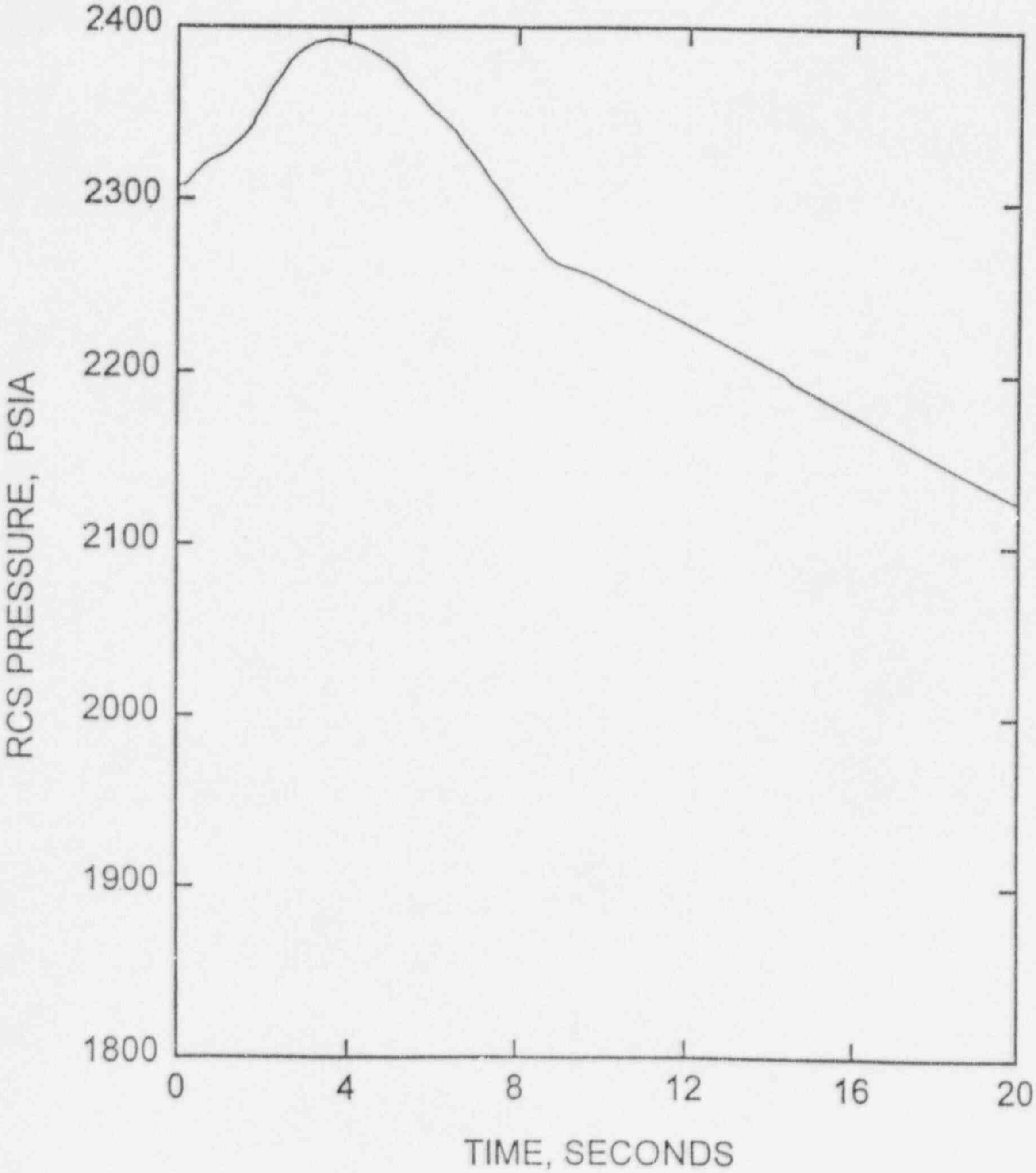


FIGURE 14.16-4

SEIZED ROTOR EVENT
RCS PRESSURE VS TIME



VIII. REFERENCES

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

2. Letter from Mr. A. E. Scherer (CE) to Mr. D. G. Eisenhut (NRC), dated March 31, 1982, Turbine Trip Time Delay

3. Safety Evaluation Report related to the final design of the Standard Nuclear Steam Supply Reference System; CESSAR System 80, Docket No. STN 50-470, dated September 1983

4. Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated March 28, 1996, License Amendment Request: Change to the Moderator Temperature Coefficient

5. "Safety Evaluation of the Potential for and Consequence of Reactor Vessel Head Penetration Alloy 600 ID Initiated Nozzle Cracking, CEN-607", May 1993, ABB-Combustion Engineering [submitted in Letter from Mr. A. Marion (NUMARC) to NRC Document Control Desk, dated January 31, 1994, "Alloy 600 CRDM Penetrations"]

6. "Safety Evaluation of the Potential for and Consequences of Reactor Head Alloy 600 Penetration O.D. Initiated Circumferential Cracking", CEN-614, December 31, 1994, ABB-Combustion Engineering

7. Letter from Mr. J. J. Hutchinson (Chairman C-E Owners Group) to Mr. J. T. Wiggins (NRC), dated February 26, 1992, "C-E Owners Group Report CE NPSD-690-P, 'Evaluation of Pressurizer Penetrations and Evaluation of Corrosion After Unidentified Leakage Develops'"

8. Letter from Mr. G. C. Creel (BGE) to NRC Document Control Desk, dated September 20, 1989, "Submittal of Basis for Determination"

9. Letter from Mr. M. J. Case (NRC) to Mr. R. E. Denton (BGE), dated August 31, 1994, "Issuance of Amendments for Calvert Cliffs Nuclear Power Plant, Unit No. 1 (TAC No. M88193) and Unit No. 2 (TAC No. M88194)

10. CENPD-162-P-A, Latest Approved Revision, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution"

11. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated May 11, 1995, "Approval to Use Convolution Technique in Main Steam Line Break Analysis - Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M90897 and M90898)"

12. CENPD-183-A, "Loss of Flow, C-E Methods for Loss of Flow Analysis," July 1975

ATTACHMENT (2)

DETERMINATION OF SIGNIFICANT HAZARDS

DETERMINATION OF SIGNIFICANT HAZARDS

Baltimore Gas and Electric Company is proposing a change to the Technical Specifications for Calvert Cliffs Units 1 and 2 to accommodate a larger number of plugged steam generator tubes for future operating cycles. Specifically, these changes will: (1) reduce the minimum Reactor Coolant System (RCS) total flow rate from 370,000 gpm to 340,000 gpm; (2) reduce the Limiting Safety System Setting for Reactor Coolant Flow - Low trip function from ≥ 95% to ≥ 92% of design reactor coolant flow; (3) revise the Reactor Core Thermal Safety Limit lines to indicate operation at the lower reactor coolant flow rate; and (4) decrease the maximum allowable lift settings for the eight highest set Main Steam Safety Valves from 1065 psig to 1050 psig. Additionally, it was determined through the reanalysis of the accident analyses affected by these changes, that lowering the RCS total flow rate limit to 340,000 gpm involves an Unreviewed Safety Question (USQ). The USQ is associated with the potential for an increased fuel cladding failure rate for the Main Steam Line Break and Seized Rotor Events. For both of these events, the established Nuclear Regulatory Commission acceptance criteria were not exceeded.

The proposed Technical Specification changes and USQ have been evaluated against the standards in 10 CFR 50.92 and have been determined to not involve a significant hazards consideration, in that operation of the facility in accordance with the proposed amendments:

1. *Would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The proposed amendment defines changes to the operating licenses for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, necessary to support increased steam generator tube plugging. The effects of increased steam generator tube plugging include reduced steam generator pressure and RCS flow rate, and increased core outlet (hot leg) temperature. The Technical Specification changes necessary to account for these effects are reducing the minimum RCS total flow rate from 370,000 gpm to 340,000 gpm; reducing the Limiting Safety System Setting for reactor coolant low flow trip function from ≥ 95% to ≥ 92% of design reactor coolant flow; revising the Reactor Core Thermal Safety Limit lines to indicate operation at the lower reactor coolant flow rate; and decreasing the maximum allowable lift settings for the eight highest set Main Steam Safety Valves from 1065 psig to 1050 psig. The Design Basis Events (DBEs) affected by these changes were reanalyzed to determine if the effects of increased steam generator tube plugging, and the associated changes to the Technical Specifications, could result in exceeding the acceptance criteria applicable to each of these events. Although it was determined that the DBE acceptance criteria would not be exceeded as a result of increased steam generator tube plugging, the analyses for the Main Steam Line Break and Seized Rotor Events indicated an increased percentage of fuel cladding failure as a result of the lower RCS total flow rate; therefore, it was determined that this activity involves a USQ.

   Technical Specification 2.1.1 will be changed to establish more restrictive limits on core thermal power and reflect a lower minimum RCS flow of 340,000 gpm. Making the core thermal power limits more restrictive does not initiate a change to plant conditions that would affect other plant components. Therefore, the probability of a previously evaluated accident is not significantly increased. Additionally, the Limiting Conditions for Operation and Limiting Safety System Settings based on these limits remain adequately conservative or will be changed in the Core Operating Limits Report, as appropriate. Therefore, the consequences of a previously evaluated accident are not significantly increased.

Technical Specification 2.2 will be changed to reduce the Reactor Coolant Flow - Low reactor trip setpoint from ≥ 95% to ≥ 92%, thereby providing additional operating margin to this trip setpoint and the associated pre-trip alarm. Reducing this setpoint does not initiate a change to plant conditions that would affect other plant components. Therefore, the probability of a previously evaluated accident is not significantly increased.

As demonstrated by the revised Loss of Coolant Flow analysis, the proposed Reactor Coolant Flow - Low reactor trip setpoint will continue to provide adequate core protection. A trip setpoint of ≥ 92% ensures fuel is not damaged, and the site boundary dose remains a small fraction of the 10 CFR Part 100 guidelines. Therefore, the consequences of a previously evaluated accident are not significantly increased.

Technical Specification 3.2.5.c will be changed to reduce the minimum RCS total flow rate from 370,000 gpm to 340,000 gpm. This change reduces the core heat removal rate and slightly increases the core outlet and average coolant temperatures. This change involves a USQ, as the Main Steam Line Break and Seized Rotor Event analyses have indicated an increase in the number of failed fuel pins during these events as a result of reducing the initial RCS flow rate. The probability of malfunction of equipment important to safety (i.e., fuel pin cladding) during these accidents increases. However, this malfunction is not an accident initiator. Rather, it is a consequence of an accident. Therefore, the probability of a previously evaluated accident is not significantly increased. The consequences of the Main Steam Line Break and Seized Rotor Events are not significantly increased, as the results of the analyses of these events are within the current acceptance criteria established by the NRC.

Analyses and evaluations have been performed to demonstrate that the new flow and temperature conditions are acceptable:

> Fuel and core performance remain within acceptable limits. Analysis and evaluation of fuel mechanical design, core physics parameters, fuel pin performance, fuel assembly thermal/hydraulic performance, and fuel pin corrosion all demonstrate acceptable results.
>
> The effect of the slightly elevated core outlet and average coolant temperature on the structural integrity of the RCS is acceptable. The RCS penetration inspection program and the steam generator tube inspection program will continue to identify and repair or isolate Alloy 600 cracks prior to inservice failure of these components. The stress analysis for the reactor vessel and piping remain bounding.
>
> The performance of control systems (i.e., feedwater, pressurizer level, and pressurizer pressure) will maintain RCS and steam generator parameters within appropriate limits by periodic adjustment, as necessary. Reactor coolant pump operation will be maintained within acceptable limits by periodic adjustment of the operating curves.

Therefore, the probability of a previously evaluated accident is not significantly increased.

Analyses and evaluations of the DBEs have been performed demonstrating that the NRC acceptance criteria for these events are met. The revised analyses and evaluations consider reduced RCS flow, increased reactor coolant temperature, and increased steam generator tube plugging conditions.

> The results of analyses and evaluations of the Postulated Accidents demonstrate that the site boundary dose is within 10 CFR Part 100 guidelines and the core geometry remains coolable. Loss-of-Coolant Accident analysis results meet the acceptance criteria stipulated in 10 CFR 50.46(b).
>
> The results of analyses and evaluations of Anticipated Operational Occurrences demonstrate that fuel parameters do not exceed the specified acceptable fuel design limits and site boundary dose is a small fraction of 10 CFR Part 100 guidelines. Primary and secondary system pressure remain below the pressure upset limits for the RCS and steam generators, respectively.

Therefore, the consequences of a previously evaluated accident are not significantly increased.

Technical Specification 4.7.1.1 will be changed to reduce the maximum allowable lift setting for the eight Main Steam Safety Valves with the highest lift setpoint. This change will place more restrictive limits on the allowable range of lift settings for these eight valves. The allowable range of lift settings for the proposed change is also allowed by the current Technical Specification. Therefore, the probability of a previously evaluated accident occurring is not significantly increased.

The revised safety analyses will credit the highest lift setting for these eight valves as being 1050 psig. The more restrictive limit on the maximum lift setting is required in order to make this Technical Specification consistent with the revised safety analyses. Analyses performed assuming the proposed maximum lift setting for these valves demonstrate that secondary system pressure does not exceed 110% of the system design pressure. Therefore, the consequences of a previously evaluated accident are not significantly increased.

Therefore, operation of the facility in accordance with this amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Would not create the possibility of a new or different type of accident from any accident previously evaluated.*

The proposed amendment revises limiting parameters to assure sa[illegible] commensurate with the effects of steam generator tube plugging, and will not cha[illegible] les of operation defined in the facility license. The analysis of transients associate[illegible] team generator malfunctions are part of the design and licensing bases. This chang[illegible] ot add any new equipment, modify any interfaces with any existing equipment, or change the equipment's function, or the method of operating the equipment. The proposed change does not change plant conditions in a manner which could affect other plant components. Reactor core, RCS, and steam generator parameters remain within appropriate design limits during normal operation.

Therefore, the proposed change could not cause any existing equipment to become an accident initiator.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. *Would not involve a significant reduction in a margin of safety.*

The margins of safety associated with this change are defined in the fuel and core-related analyses, the Alloy 600 stress corrosion cracking evaluation, the RCS structural evaluation, the operational evaluation, and in each of the transient and accident analyses affected by the increased steam generator tube plugging.

Reanalysis of the fuel and core-related analyses for fuel mechanical design, core physics, fuel performance, thermal hydraulics, and fuel rod corrosion verified that the fuel and core performance will remain within acceptable limits and will be bounded by the current assumptions for fuel performance in the transient and accident analyses. The Alloy 600 RCS penetration inspection program and the steam generator tube inspection program will continue to find and repair Alloy 600 cracks at the slightly elevated core exit temperature prior to any postulated inservice failure of these components. The stress analyses performed for the reactor vessel and piping remain bounding for the slightly elevated core exit temperature. Additionally, the performance of non-safety-related control systems remains adequate to maintain RCS and steam generator parameters within appropriate operating limits. Therefore, the margins of safety associated with the physical and operational effects of this change will not be significantly reduced.

An evaluation of the affected DBEs confirmed that the established acceptance criteria for specified acceptable fuel design limits, primary and secondary system over-pressurization, 10 CFR 50.46(b), Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, and potential radiation dose during accidents have been completed in support of this license amendment request. The evaluation concludes that, when considering the proposed Limiting Safety System Setting for the Reactor Coolant Flow - Low trip, Limiting Conditions for Operation for RCS total flow rate, and reduced lift settings for eight Main Steam Safety Valves per unit, all applicable acceptance limits are met. Furthermore, the USQ resulting from the reduced RCS total flow rate does not represent a reduction in the margin of safety, as the site boundary dose calculated in the affected DBE analyses is within the current established radiation dose limits and the core geometry remains coolable. Therefore, the margins of safety associated with the transient and accident analyses affected by this change will not be significantly reduced.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.