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Constellation Nuclear

Calvert Cliffs
Nuclear Power Plant

A Member of the
Constellation Energy Group

November 19, 2001

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: Reanalysis of the Loss of Feedwater Event

REFERENCE: (a) Letter from Mr. C. H. Cruse (CCNPP) to NRC Document Control Desk, dated December 20, 2000, License Amendment Request: Revision to the Technical Specifications to Support Steam Generator Replacement

Pursuant to 10 CFR 50.90, Calvert Cliffs Nuclear Power Plant, Inc. hereby requests an Amendment to Renewed Operating License Nos. DPR-53 and DPR-69 with the submittal of changes to the Loss of Feedwater Flow analysis in the Updated Final Safety Analysis Report (UFSAR). The current analysis contains several non-conservative assumptions, resulting in the need for reanalysis. Prior Nuclear Regulatory Commission review is required due to changes in the methodology and acceptance criteria that will be used for the Loss of Feedwater Flow analysis. Therefore, per 10 CFR 50.59(2)(c), we request the Nuclear Regulatory Commission review and approve this change through an amendment to our renewed operating licenses pursuant to 10 CFR 50.90. Upon approval of the analysis, the UFSAR will be updated. The revised analysis, performed with corrected assumptions, indicates that the results of the Loss of Feedwater Flow event previously evaluated in the safety analysis report continue to be acceptable.

BACKGROUND

During a review of the Loss of Feedwater Flow analysis in the UFSAR Chapter 14, some non-conservative assumptions were discovered. In addition, during a recent fire protection inspection the Nuclear Regulatory Commission noted that the treatment of steam generator blowdown in the Loss of Feedwater Flow analysis was also non-conservative. An operability evaluation was performed and both Units were determined to be operable, even accounting for these non-conservative assumptions.

The assumptions in question are:

- The single-failure treatment of the mitigating system – The Auxiliary Feedwater (AFW) System is required to mitigate the consequences of this event. In the current analysis, no single-failure was assumed. The requirement for considering a single-failure of the AFW System in this analysis is contained in TMI Action Item II.E.1.1. To partially compensate for this non-conservative assumption operator action to increase flow from the operating AFW pump is credited in this analysis.

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- The amount of blowdown assumed in the analysis – The current analysis assumes no blowdown from the steam generators. In fact, blowdown is operated on a routine basis and has an effect on steam generator inventory once feedwater flow is lost. To resolve this non-conservative assumption, Calvert Cliffs will install a modification to isolate steam generator blowdown upon receipt of an Auxiliary Feedwater Actuation System (AFAS) signal. This will prevent further steam generator inventory reduction in the event that a Loss of Feedwater Flow event were to occur.
- The inventory at the time of reactor trip assumed in the analysis – The method of calculating inventory at the time of reactor trip does not account for the change in density of the water once the feedwater flow has stopped.
- Sludge deposition in the steam generators - The sludge deposited in the steam generators during normal operation is not accounted for when determining the inventory of water remaining in the steam generators. This issue has a small effect and although it affects only the current steam generators, this will also account for any build up of sludge in the replacement steam generators.

These non-conservative assumptions primarily affect the portion of the Loss of Feedwater Flow analysis that addresses maximum steam generator inventory depletion criteria. The Loss of Feedwater Flow analysis has been performed with these assumptions corrected. The assumptions, methods, and results of the analysis are presented in Attachment (1). Note that this analysis applies to the replacement steam generators only. An operability evaluation was performed for the existing steam generators and both Units were determined to be operable, even accounting for these non-conservative assumptions. The new analysis also takes credit for installation of the modification to isolate blowdown on an AFAS, which will be installed prior to operation of Unit 1 (2002) and Unit 2 (2003) with the replacement steam generators.

SAFETY ANALYSIS

A Loss of Feedwater Flow event is defined as a reduction in feedwater flow to the steam generator without a corresponding reduction in steam flow from the steam generator. 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 a Loss of Feedwater Flow event. The most limiting Loss of Feedwater Flow event at full power is 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 steam generator inventory.

Three areas are evaluated during the Loss of Feedwater Flow event: Reactor Coolant System (RCS) pressure, secondary system pressure, and the depletion of steam generator inventory. Each of these areas is evaluated independently to allow the assumptions to be adjusted to maximize the effect of the event in these different areas. The analysis must demonstrate that the specified acceptable fuel design limits (SAFDLs) continue to be met. This ensures that the dose consequences of the event are controlled within the acceptance criteria given in UFSAR Section 14.1.4.4. The assumptions, methods of analysis, and results are given in Attachment (1).

Two areas are included in this analysis that had not been included in the previous analysis: operator action and closure of the steam generator blowdown valves by an AFAS signal. The operator actions assumed in the analysis have been evaluated with respect to the Emergency Operating Procedures to ensure that the actions would occur in a timeframe consistent with the analytical assumptions. Operators would have adequate indications to respond with the correct actions within the allotted timeframe. These assumptions are consistent with UFSAR Section 14.1.4.2. It should also be noted that the steam generator blowdown valves are safety-related valves. The closure signal that will be installed on these valves is also safety-related and is provided by the AFAS. Therefore, the valves can are assumed to perform their function and close following an AFAS signal during this event.

The results of the analysis, demonstrates that the action of the Reactor Protective System (RPS) prevents exceeding the SAFDL, and the RCS and secondary system peak pressure limits. The results of all three portions of the Loss of Feedwater Flow analysis are given here and are described in more detail in Attachment (1).

The new maximum calculated RCS pressure is 2620 psia. This portion of the analysis demonstrates that the RPS and the pressurizer safety valves prevent the RCS from exceeding 110% of RCS design pressure (i.e., 2750 psia). The previous analysis calculated a maximum RCS pressure of 2631 psia. The new maximum RCS pressure is lower than the previously calculated result.

The new maximum calculated steam generator pressure is 1107 psia. This portion of the analysis demonstrates that the RPS and the main steam safety valves prevent the secondary system from exceeding 110% of the steam generator design pressure (i.e., 1116.5 psia). The previous analysis calculated a maximum secondary side pressure of 1080 psia. The new secondary side pressure is higher than the previously calculated pressure, but remains below the steam generator design pressure.

For the steam generator depletion portion of the analysis, the actions of the RPS, AFAS, and AFW System are adequate to prevent exceeding the SAFDLs. This analysis considers an automatic isolation of steam generator blowdown and operator actions at 10 minutes to adjust AFW flow to provide acceptable results for this portion of the event. These two assumptions are differences between this analysis and the previous one. This analysis continues to show that the SAFDLs are protected.

The Loss of Feedwater Flow event has been analyzed with respect to RCS peak pressure, secondary system peak pressure, and steam generator depletion criteria. As noted above, the analysis of the event for all of these areas concludes with acceptable results. The response of the required safety systems is sufficient to prevent exceeding the SAFDLs and the RCS and secondary system pressure limits.

ASSESSMENT AND REVIEW

We have evaluated the significant hazards considerations associated with this proposed change, as required by 10 CFR 50.92, 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 impact statement or environmental assessment is needed in connection with the approval of the proposed amendment.

SAFETY COMMITTEE REVIEW

The Plant Operations and Safety Review Committee and Offsite Safety Review Committee have reviewed this proposed change and concur that operation with the proposed changes will not result in an undue risk to the health and safety of the public.

SCHEDULE

This change is requested to be approved and issued by February 15, 2002. This revised analysis is needed to support a previously submitted license amendment (Reference a). That amendment addresses several issues with the replacement steam generators that are scheduled to be installed during the upcoming Unit 1 outage. Therefore, delaying issuance of this amendment will impact startup from the Unit 1 outage, because it will delay the amendment requested in Reference (a). The Unit 1 2002 refueling outage is currently scheduled to begin February 15, 2002.

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

Very truly yours,

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

I, Charles H. Cruse, being duly sworn, state that I am Vice President - Nuclear Energy, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 19th day of November, 2001.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires: 02/01/02
Date

CHC/PSF/bjd

Attachment: (1) Loss of Feedwater Flow Analysis
(2) Determination of No Significant Hazards

cc: R. S. Fleishman, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
D. M. Skay, NRC
H. J. Miller, NRC
Resident Inspector, NRC
R. I. McLean, DNR

ATTACHMENT (1)

LOSS OF FEEDWATER FLOW ANALYSIS

Calvert Cliffs Nuclear Power Plant, Inc.
November 19, 2001

BACKGROUND

Updated Final Safety Analysis Report (UFSAR) Section 14.6 describes the Loss of Feedwater Flow (LOFW) event for Calvert Cliffs. A LOFW could occur due to pump failures, valve malfunctions, or a loss-of-offsite power. The most limiting LOFW event is an inadvertent closure of both feedwater regulating valves at 100% power. An instantaneous closure of the regulating valves would cause the largest steam and feedwater flow mismatch and result in the most rapid reduction in steam generator inventory. The LOFW event could potentially result in exceeding 110% of the design pressure for the Reactor Coolant System (RCS) and/or Main Steam System, and could result in violating the specified acceptable fuel design limits (SAFDLs) for the departure from nucleate boiling ratio and linear heat rate. The actions of the Reactor Protective System (RPS), Engineered Safety Feature Actuation System, main steam safety valves (MSSV), pressurizer safety valves (PSV), and the Auxiliary Feedwater (AFW) System prevent exceeding design pressures and SAFDLs.

The LOFW analysis is performed for three distinct, separate scenarios: peak RCS pressure, peak secondary system pressure, and maximum steam generator inventory depletion (to demonstrate protection of fuel SAFDLs). For each scenario, the initial conditions, inputs, and assumptions are selected to maximize the parameters of interest. Calvert Cliffs uses the computer code CESEC to analyze the Nuclear Steam Supply System response to this event. The CESEC computer code is described in UFSAR Section 14.1.4.

During the course of a reanalysis for the Replacement Steam Generators (RSGs), the following issues were discovered with the current LOFW analysis. While the issues were discovered during the RSG reanalysis, none of the issues were a result of the new generators. Not all of the issues impact all three scenarios.

1. The methodology for the LOFW event credits the RPS Steam Generator Low Level Trip by examining the steam generator mass vs. level. When the inventory depletion reaches the mass corresponding to the low-level trip setpoint, a trip is initiated. In the current analysis the inventory vs. level calculations were based on normal steady-state conditions. When the inventory vs. level calculations are performed under LOFW conditions, the mass left in the steam generators for a given low-level trip setpoint is less than at normal feedwater conditions. This is due to changes in density in the downcomer. Therefore, at the time of trip, less inventory would be remaining in the generators than in the current UFSAR analysis. This would also lead to a delay in the time of trip since more inventory must be depleted before the setpoint would be reached. This error impacts both the peak secondary pressure analysis and the inventory analysis.

2. The current UFSAR analysis did not account for steam generator blowdown. Operating with steam generator blowdown would further deplete the inventory in the generators. While the blowdown system does receive isolation signals, none of the LOFW event responses would trigger these isolation signals. Therefore, blowdown would have to be manually isolated by operators with the current plant configuration. This error impacts the inventory analysis only.

3. Over the course of operation, sludge is deposited in the steam generators. This was not accounted for in the current UFSAR analysis. The presence of sludge would lead to less liquid inventory in the generators. Although not previously accounted for, the impact in liquid inventory is minimal. This error impacts the inventory analysis only.

4. The current UFSAR analysis did not consider a single-failure of the AFW System. It has been determined that based on previous correspondence with the Nuclear Regulatory Commission (NRC) that this was an error and single-failure must be considered. A single-failure of the AFW System will only impact the inventory analysis since the pressure analyses do not credit this system.

As a result of the above issues, the LOFW analysis is submitted for NRC approval. The revised analysis accounts for several changes in inputs, assumptions, methodology, and acceptance criteria.

Peak Pressure Analyses (RCS and Secondary System)

The initial conditions and inputs used to analyze the LOFW event are presented in Table 1. These inputs are designed to maximize peak pressures at 100% power with no inoperable MSSVs. All values include appropriate uncertainties. Parametric cases were run on several parameters to determine the worst set of conditions. Parametric cases were run for the RCS temperature, RCS pressure, pressurizer volume, pressurizer spray, steam generator tube plugging, and with and without a loss of AC power (LOAC). Power-operated relief valves, atmospheric dump valves, and turbine bypass valves were assumed to be unavailable, as this would provide primary and secondary pressure relief. For the peak pressure events, steam generator blowdown was assumed to be isolated prior to the event in order to delay the low-level trip.

The RCS pressure must not exceed 110% of design (2750 psia). The sequence of events for the peak RCS pressure scenario is given in Table 2. For this scenario, only the High Pressurizer Pressure Trip was credited. The worst case RCS pressure analysis assumed a LOAC. The RCS pressure reaches the analysis trip setpoint of 2420 psia at 22.6 seconds, with the first PSV lifting at 25.9 seconds. Figures 1 through 4 present the transient behavior of core power, RCS temperature, RCS pressure, and steam generator pressure. The maximum calculated RCS pressure is 2620 psia, including elevation head. This demonstrates that the RPS and the PSVs prevent exceeding the RCS pressure limit of 2750 psia.

Steam generator pressure also must not exceed 110% of design (1116.5 psia) for the RSGs. The sequence of events for the peak secondary pressure is given in Table 3. This scenario assumed 100% power with no inoperable MSSVs. The peak secondary pressure scenario credits either the High Pressurizer Pressure Trip or the steam generator low-level trip. The worst-case secondary pressure analysis was with no LOAC. Steam generator inventory reaches the analysis trip setpoint of 55" below normal level at 27.4 seconds with the first bank of MSSVs opening at 29.7 seconds. Figures 5 through 8 present the transient behavior of core power, RCS temperature, RCS pressure, and steam generator pressure. The maximum calculated steam generator pressure is 1107 psia, including downcomer liquid head. This demonstrates that the RPS and the MSSVs prevent exceeding the steam generator pressure limit of 1116.5 psia.

Steam Generator Inventory Depletion Analysis

This scenario maximizes steam generator inventory depletion. Due to the errors listed in the Background section, the following changes were made to the LOFW calculation for this scenario.

- Calvert Cliffs will install a modification that will isolate steam generator blowdown upon receipt of an Auxiliary Feedwater Actuation System (AFAS) signal. Prior to the AFAS signal, steam generator blowdown flow is modeled in computer code CESEC. The steam generator blowdown valves are safety-related. The closure signal that will be installed on these valves is also safety-related and is provided by the AFAS. Therefore, the valves can be assumed to perform their function and close following an AFAS signal during this event.
- The LOFW analysis assumes the worst single active failure within the AFW system. This has been determined to be failure of the motor-driven AFW pump to deliver flow.
- Operator action is credited at 10 minutes to increase AFW flow from the steam-driven AFW pump. The operator actions assumed in the analysis have been evaluated with respect to the Emergency Operating Procedures to ensure that the actions would occur in a timeframe consistent with the analytical assumptions. Operators would have adequate indications to respond with the correct actions within the allotted timeframe. The assumption of a 10 minute operator response is consistent with UFSAR Section 14.1.

- The current version of the CESEC computer code utilizes the 1971 American Nuclear Society (ANS) decay heat standard. This standard uses 20% uncertainty prior to 1000 seconds and 10% uncertainty after 1000 seconds. The revised analysis replaced the 1971 ANS standard with the 1979 ANS standard, which uses a 2-sigma uncertainty and includes actinide decay. This results in a reasonably bounding, but not excessively conservative (approximately 8% uncertainty), decay heat assumption.

- The previous LOFW analyses assumed full, non-degraded, heat transfer from the primary system to the secondary system for the duration of the event. In reality, decreasing inventory in the steam generators would lead to degraded heat transfer which would slow down the steam generator inventory reduction. Therefore, the revised analysis will incorporate heat transfer degradation due to the steam generator inventory depletion. This methodology has been previously used in the Feedline Break event (UFSAR Section 14.26).

- Flow uncertainty for the steam-driven AFW pump is reduced. New uncertainties are determined in accordance with Instrumentation Systems and Automation (ISA)-S-67.04-1987, "Setpoints for Nuclear Safety-Related Instrumentation."

- The steam generator low-level trip setpoint was raised from conservative value of 116.4" below normal level, used in the current UFSAR analysis to 55" below normal level. The new setpoint supports the existing Technical Specification 3.3.1 limit of 50" below normal level. The new setpoint includes uncertainties determined in accordance with ISA-S-67.04-1987.

- These analyses are performed using the RSG characteristics, since the RSG analysis bounds the original steam generator analysis because at the low-level trip setpoint of 55" below normal level, less inventory is available in the RSGs as compared to the original steam generators. Therefore, the inventory depletion in the RSGs would be more severe.

- A change in the acceptance criteria from the current UFSAR analysis has been made for the inventory analysis portion only. The current UFSAR acceptance criteria shows that steam generator water mass is preserved above the minimum acceptable amount required for adequate heat removal (i.e., the steam generators do not dryout). The analyst, in fact, selects a conservative water mass that must be maintained to ensure that dryout does not occur. The revised analysis no longer uses this acceptance criterion. Although the revised analysis continues to show steam generator dryout does not occur, this is only because heat transfer degrades significantly such that liquid depiction is virtually halted. The new acceptance criteria will be to demonstrate that the existing steam generator inventory, in combination with the low-level trip setpoint, AFAS signal setpoint, and AFW flow, is sufficient to prevent exceeding the RCS and steam generator design pressure limits and the SAFDLs. This is done by demonstrating that all pertinent plant parameters (RCS pressure, temperature, pressurizer level, and steam generator pressure) stabilize within acceptable bounds, allowing adequate time for the operator to achieve control of the plant. With this acceptance criterion, steam generator inventory does not have to be maintained above a selected water mass in order to show acceptable performance. It should also be noted that transients involving addition of feedwater to dry steam generators are discussed in UFSAR Section 4.1.3.2.

The most limiting initial conditions and inputs for this analysis are presented in Table 4. All values include appropriate uncertainties. Parametric cases were run on several parameters to determine the worst set of conditions. Parametric cases included pressurizer volume, pressurizer spray, MSSV characteristics, and with and without a LOAC. The atmospheric dump valves and turbine bypass valves are assumed to be in operation, since this maximizes the steam release from the steam generators. The power-operated relief valves were assumed to be unavailable, as this would provide primary pressure relief.

The sequence of events for maximum steam generator inventory reduction is listed in Table 5. The worst case is without a LOAC. A reactor trip is generated on low steam generator level at 24.4 seconds into the event. At 34.6 seconds, an AFAS signal is generated, causing steam generator blowdown isolation at 69.6 seconds, and initiation of the steam-driven AFW pump at 218.10 seconds.

Figures 9 through 14 present the transient behavior of core power, RCS inlet temperature, RCS pressure, pressurizer level, steam generator pressure, and steam generator inventory. As shown on the Figures, RCS temperature, RCS pressure, pressurizer level, and steam generator pressure initially peak before the trip. Due to the reactor trip, the RCS parameters rapidly decrease. After the initial decrease, RCS parameters begin to rise again due to the lack of adequate heat removal from the steam generators. Steam generator pressure initially remains steady and then begins to drop due to the decreasing steam generator level and decreased steam generator heat transfer. Reactor Coolant System temperature peaks at 550.3°F and RCS pressure peaks at 2307 psia. At 2376.0 seconds, steam generator inventory reaches a minimum of 2150 lbm. At this point, AFW flow is sufficient to remove decay heat. Steam generator inventory begins increasing and RCS parameters begin to stabilize. Within 45 minutes plant parameters have stabilized giving operators sufficient time to achieve control of the plant. Therefore, the actions of the RPS, AFAS, and AFW System are adequate to prevent exceeding the design pressures.

The SAFDLs are not explicitly determined for this event, since both SAFDLs criteria are bounded by other more limiting events. The LOFW event is an increasing pressure event and, therefore, the departure from nucleate boiling ratio SAFDL of 1.21 is not limiting for this event. The LOFW event also results in small power increases and, therefore, the linear heat rate limit of 22 kW/ft is not challenged. The LOFW event is a heat-up transient and is most limiting at the beginning-of-the-cycle. Therefore, extended fuel burnup has no adverse impact on the event. The radiological consequences of this event are bounded by the LOAC event, which remains within 10 CFR Part 100 limits, and therefore need not be explicitly determined in the LOFW analysis.

CONCLUSION

This analysis demonstrates that RCS and steam generator design pressures are not exceeded and SAFDLs are protected. Therefore, the results of the Loss of Feedwater analysis are acceptable.

TABLE 1

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

PARAMETER	UNITS	PEAK RCS PRESSURE	PEAK SECONDARY PRESSURE
Initial Core Power Level	MWth	2754	2754
RCP Pump Heat	MWth	17	17
Initial RCS Inlet Temperature	°F	546	550
Initial RCS Flow Rate	gpm	422,250	370,000
Initial Pressurizer Pressure	psia	2164	2164
Initial Pressurizer Liquid Volume	ft^3	975	425
Pressurizer Spray	---	no	no
Number of Plugged Tubes	---	0	0
Moderator Temperature Coefficient (MTC)	$\times 10^{-4}$ $\Delta\rho$/°F	+0.15	+0.15
Axial Shape Index (ASI)	---	+0.2	+0.2
Scram Worth	%$\Delta\rho$	-4.4	-4.4
High Pressurizer Pressure Analysis Trip Setpoint	psia	2420	2420
Steam Generator Low Level Analysis Trip Setpoint	inches below normal water level	N/A	55
Trip Delay Time	sec	0.9	0.9
LOAC assumed at time of trip	---	yes	no
PSV Setpoint – RC-200	psia	2550	2550
PSV Setpoint – RC-201	psia	2617	2617
MSSV Bank 1 Setpoint (2 per Steam Generator)	psia	1020.0	1020.0
MSSV Bank 2 Setpoint (2 per Steam Generator)	psia	1051.4	1051.4
MSSV Bank 3 Setpoint (4 per Steam Generator)	psia	1071.0	1071.0

TABLE 2

SEQUENCE OF EVENTS FOR THE LOFW EVENT TO MAXIMIZE CALCULATED PEAK RCS PRESSURE

TIME (SEC)	EVENT	SETPOINT OR VALUE
0.1	Loss of Main Feedwater (MFW)	---
22.6	High Pressurizer Pressure Trip Analysis Setpoint Reached	2420 psia
23.5	Trip Breakers Open, LOAC Occurs	---
24.0	Control Element Assemblies (CEAs) Begin to Drop into the Core	---
25.5	MSSVs Begin to Open	1020 psia
25.9	PSVs Begin to Open	2550 psia
27.5	Maximum RCS Pressure	2620 psia
31.4	PSVs Close	2448 psia

TABLE 3

SEQUENCE OF EVENTS FOR THE LOFW EVENT TO MAXIMIZE CALCULATED PEAK SECONDARY PRESSURE

TIME (SEC)	EVENT	SETPOINT OR VALUE
0.1	Loss of MFW	---
27.4	Steam Generator Low Level Trip Setpoint Reached	55" BNL
28.3	Trip Breakers Open	---
28.8	CEAs Begin to Drop Into the Core	---
29.7	MSSVs Begin to Open	1020 psia
35.2	Maximum Secondary Pressure	1107 psia

TABLE 4

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOFW EVENT TO MAXIMIZE STEAM GENERATOR INVENTORY DEPLETION

PARAMETER	UNITS	SETPOINT OR VALUE
Initial Core Power Level	MWth	2754
RCP Pump Heat	MWth	17
Initial RCS Inlet Temperature	°F	550
Initial RCS Flow Rate	gpm	370,000
Initial Pressurizer Pressure	psia	2164
Initial Pressurizer Liquid Volume	ft^3	975
Pressurizer Spray	---	no
Number of Plugged Tubes	---	0
MTC	$x\ 10^{-4}\ \Delta\rho/°F$	+0.15
ASI	---	+0.2
Scram Worth	%$\Delta\rho$	-4.4
Steam Generator Low Level Analysis Trip Setpoint	inches below normal water level	55
Trip Delay Time	sec	0.9
AFW Actuation Analysis Setpoint	inches below normal water level	204
Steam-Driven AFW Response Time	sec	180
Steam-Driven AFW Flow (per Steam Generator)	gpm	100
Steam-Driven AFW Flow Credited with Operator Action (per Steam Generator)	gpm	200
Steam Generator Blowdown Flow (total from both Steam Generator)	lbm/hr	150,000
Steam Generator Blowdown Isolation Response Time	sec	35
Atmospheric Dump Valves (begin to open/fully opened/fully closed)	°F	540/557/535
Turbine Bypass Valves (begin to open/fully opened)	psia	895/905

TABLE 5

SEQUENCE OF EVENTS FOR THE LOFW EVENT TO MAXIMIZE STEAM GENERATOR INVENTORY DEPLETION

TIME (SEC)	EVENT	SETPOINT OR VALUE
0.1	Loss of MFW	---
24.4	Steam Generator Low Level Trip Setpoint Reached	55" BNL
25.3	Trip Breakers Open	---
25.8	CEAs Begin to Drop Into the Core	---
34.6	AFAS Setpoint is Reached	204" BNL
69.6	Steam Generator Blowdown Isolation Valves Close	---
218.1	Steam-Driven AFW Flow Enters Both Steam Generators	100 gpm/SG
624.4	Operators Take Action to Increase AFW Flow	200 gpm/SG
1866.0	Maximum RCS Pressure Occurs (second peak)	2307 psia
2080.0	Maximum RCS Inlet Temperature Occurs (second peak)	550.3°F
2376.0	Minimum Steam Generator Inventory is Reached	2150 lbm

FIGURE 1

LOFW EVENT FOR PEAK RCS PRESSURE CORE POWER VS. TIME

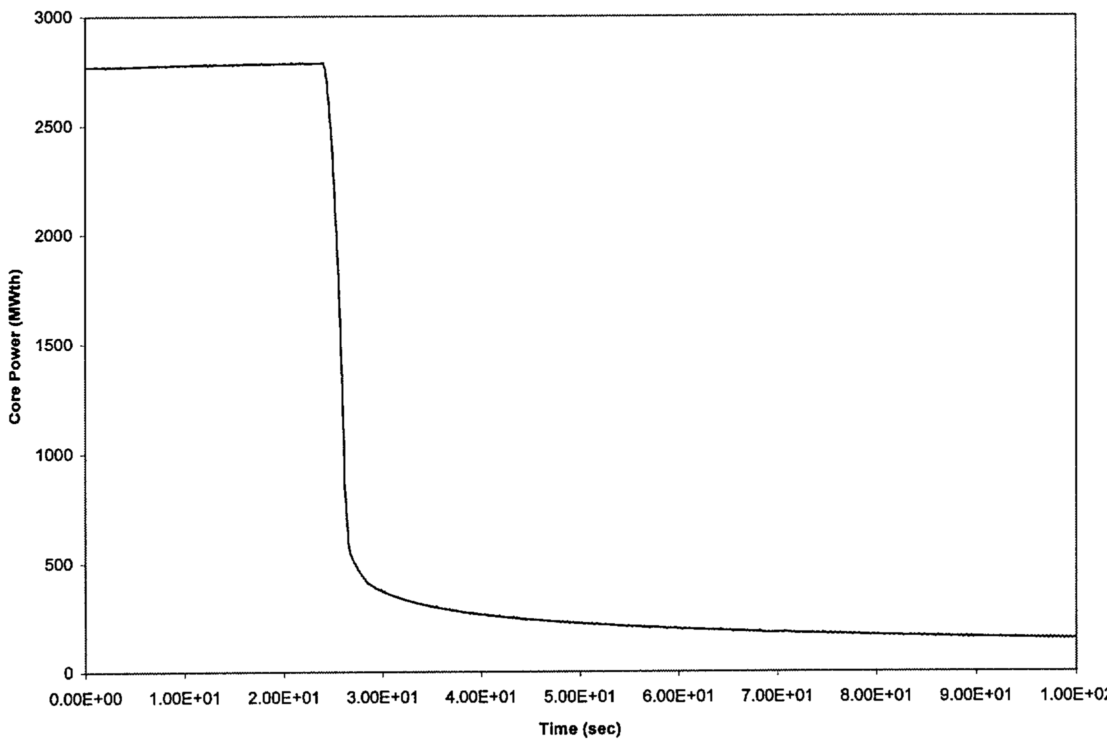


FIGURE 2

LOFW EVENT FOR PEAK RCS PRESSURE
RCS TEMPERATURE VS. TIME

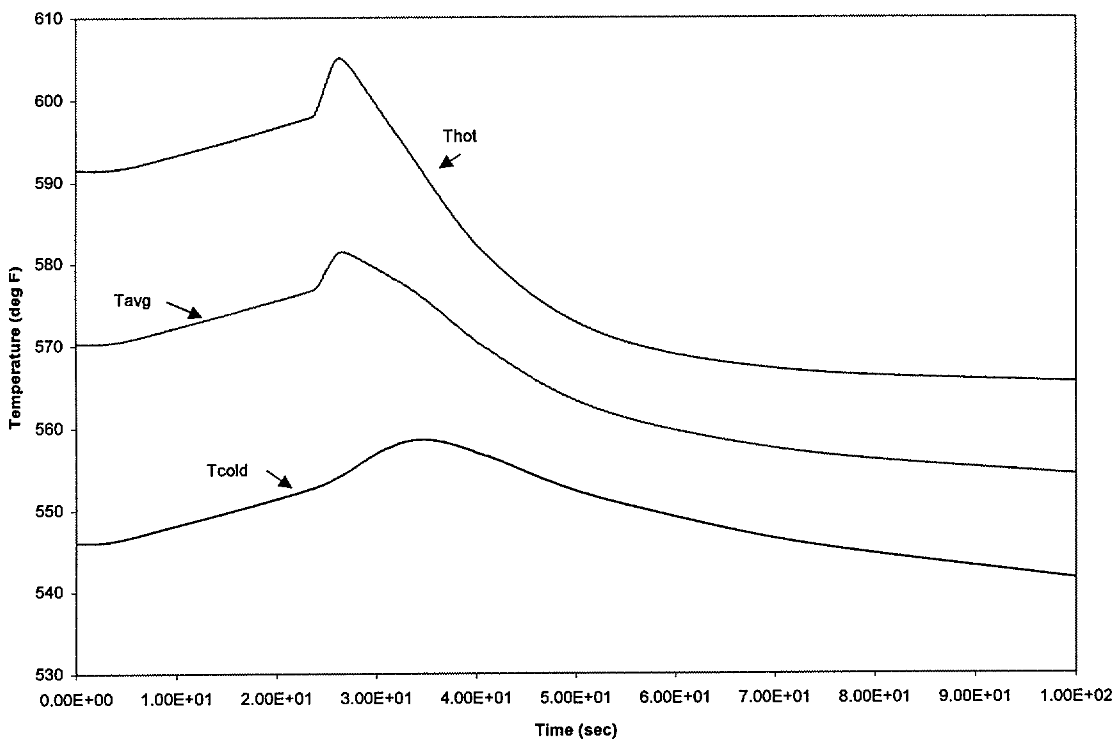


FIGURE 3

LOFW EVENT FOR PEAK RCS PRESSURE
RCS PRESSURE VS. TIME

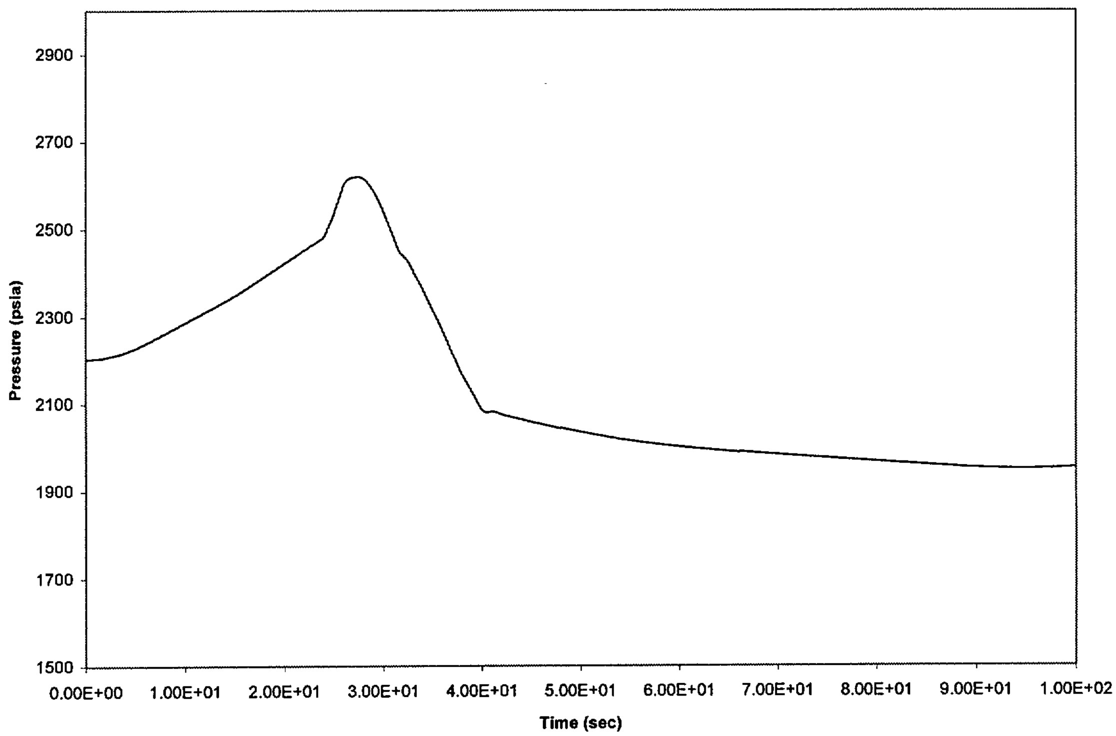


FIGURE 4

LOFW EVENT FOR PEAK RCS PRESSURE STEAM GENERATOR PRESSURE VS. TIME

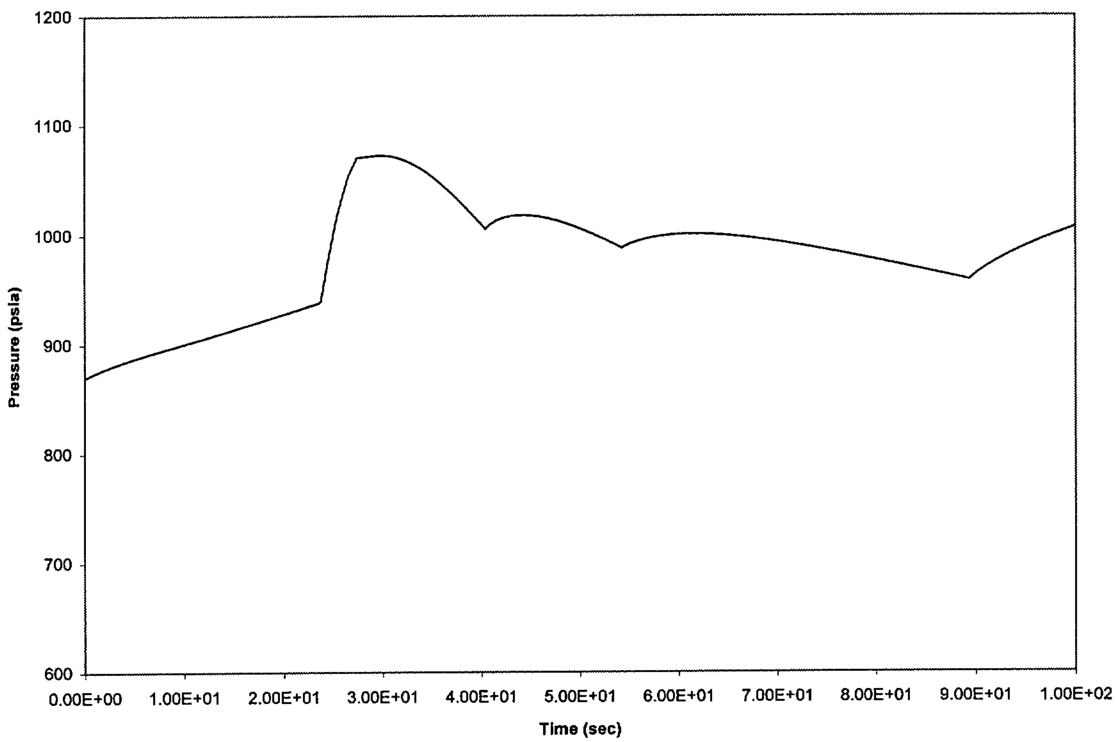


FIGURE 5

LOFW EVENT FOR PEAK SECONDARY PRESSURE CORE POWER VS. TIME

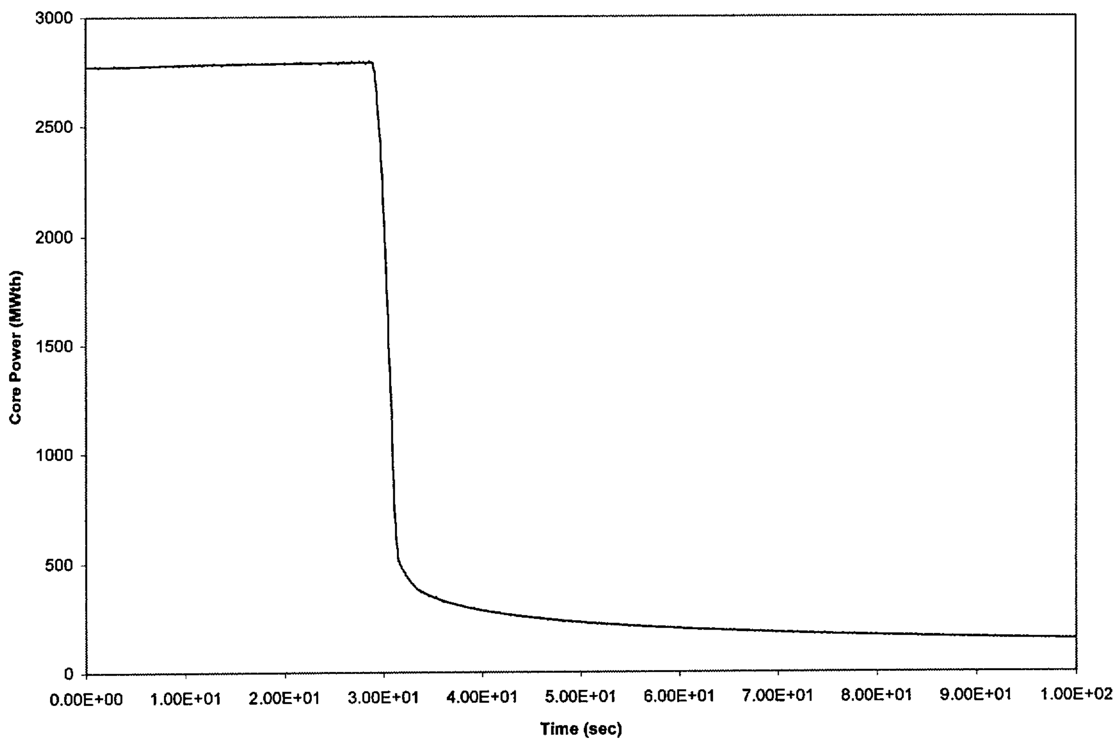


FIGURE 6

LOFW EVENT FOR PEAK SECONDARY PRESSURE RCS TEMPERATURE VS. TIME

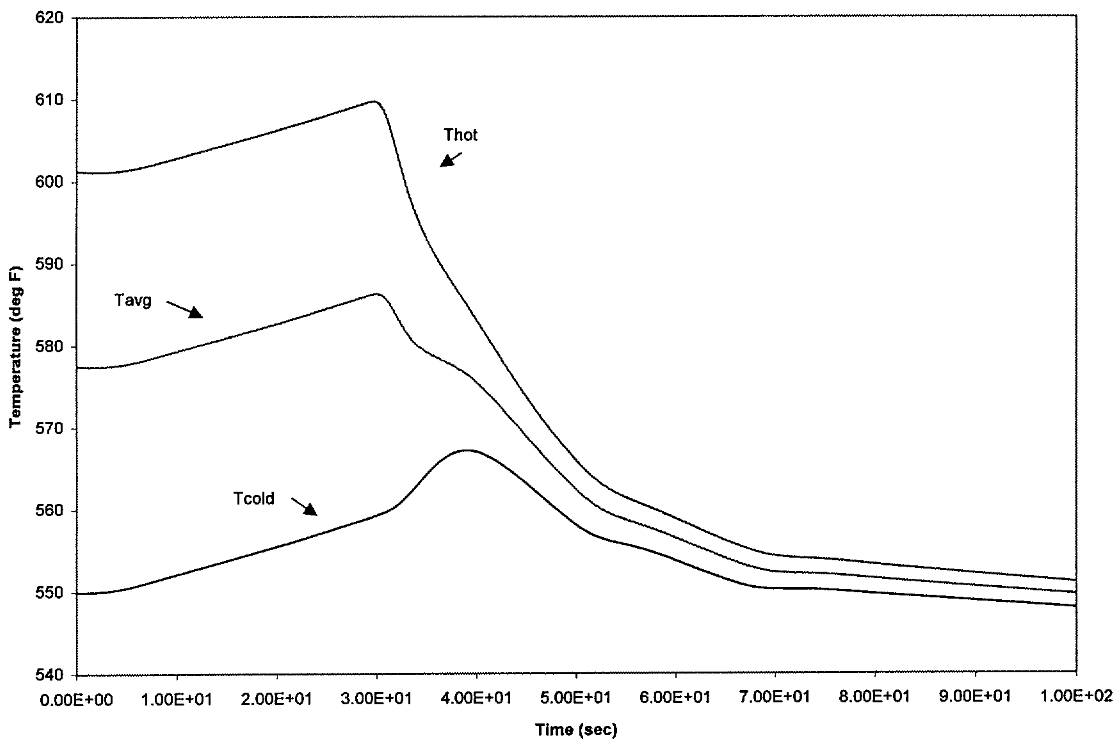


FIGURE 7

LOFW EVENT FOR PEAK SECONDARY PRESSURE
RCS PRESSURE VS. TIME

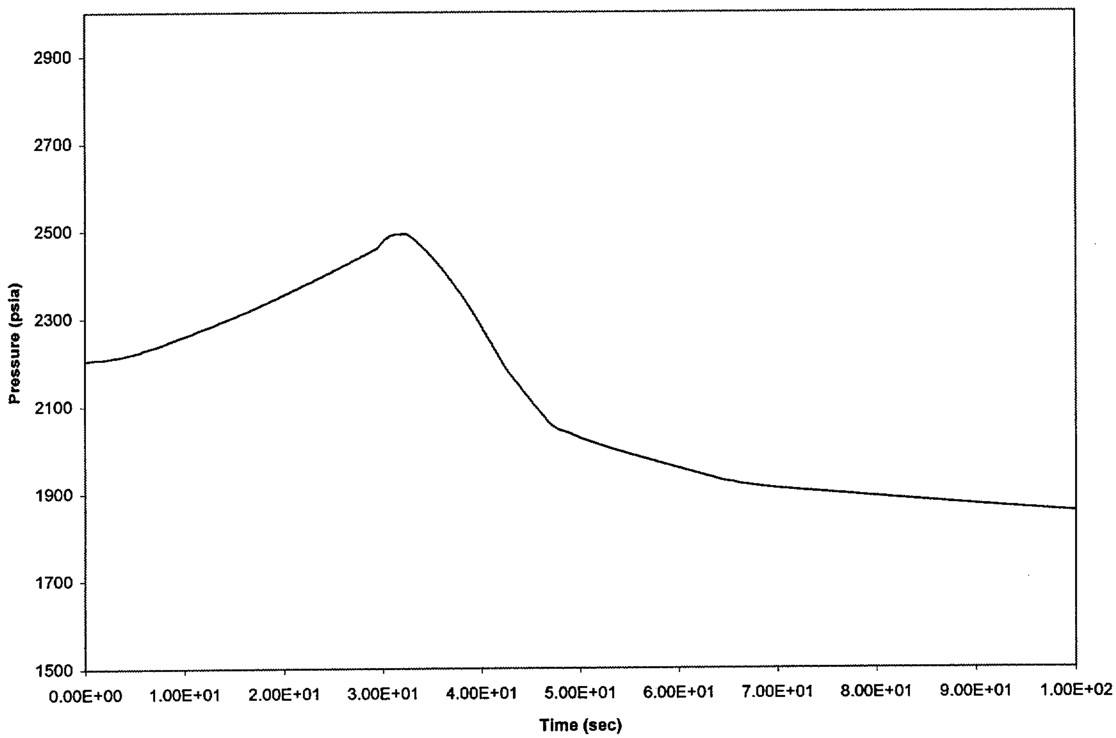


FIGURE 8

LOFW EVENT FOR PEAK RCS PRESSURE STEAM GENERATOR PRESSURE VS. TIME

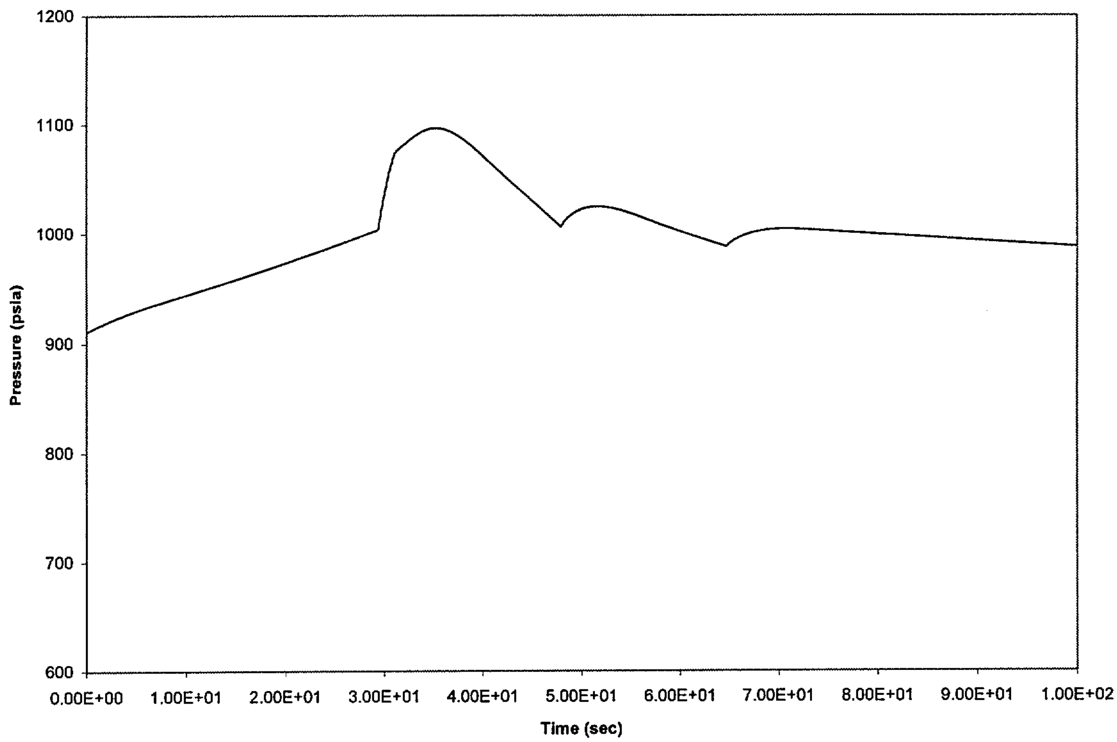


FIGURE 9

LOFW EVENT FOR MAXIMUM STEAM GENERATOR INVENTORY DEPLETION CORE POWER VS. TIME

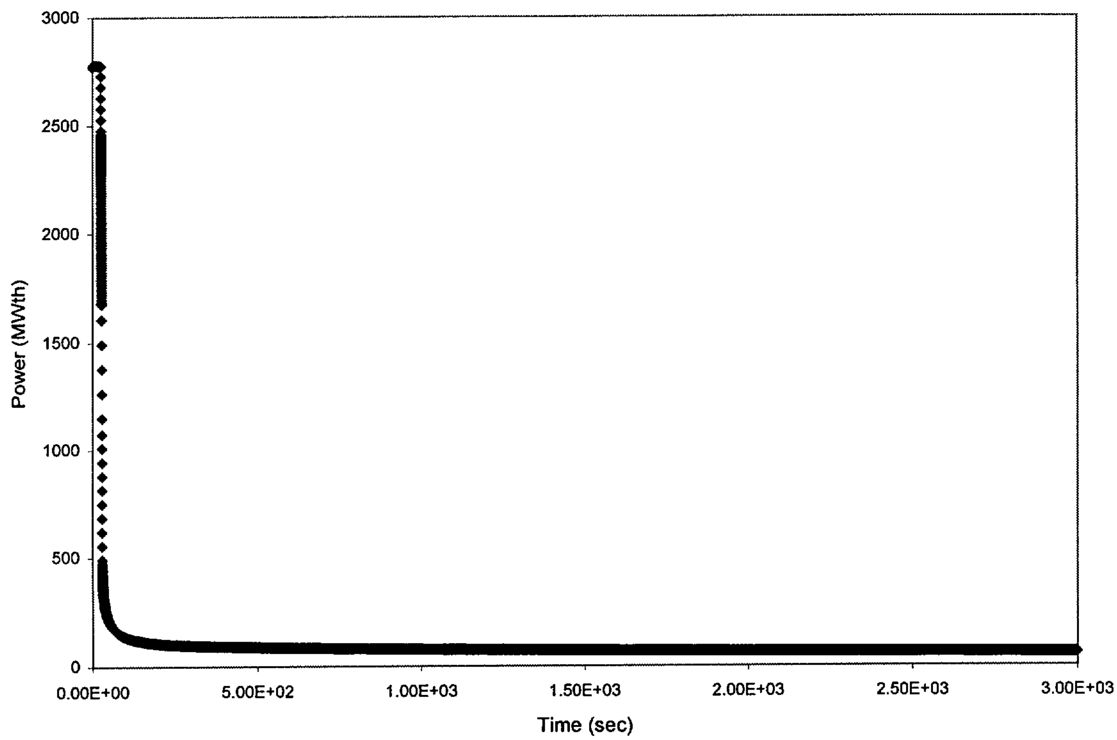


FIGURE 10

LOFW EVENT FOR MAXIMUM STEAM GENERATOR INVENTORY DEPLETION RCS PRESSURE VS. TIME

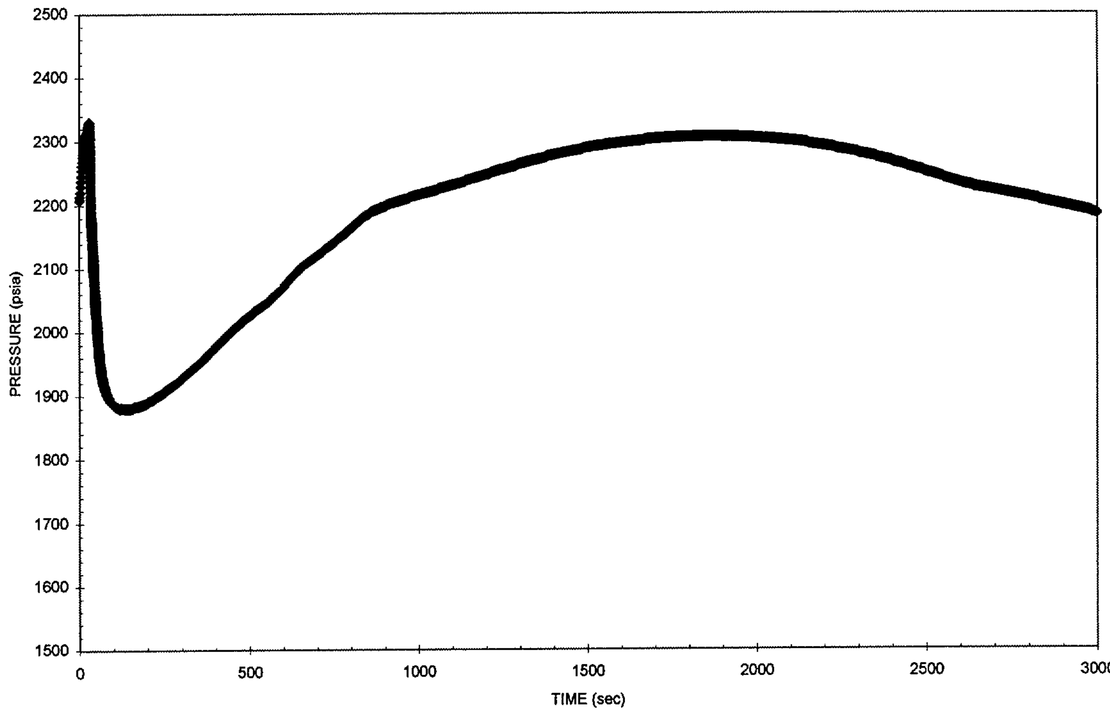


FIGURE 11

LOFW EVENT FOR MAXIMUM STEAM GENERATOR INVENTORY DEPLETION RCS INLET TEMPERATURE VS. TIME

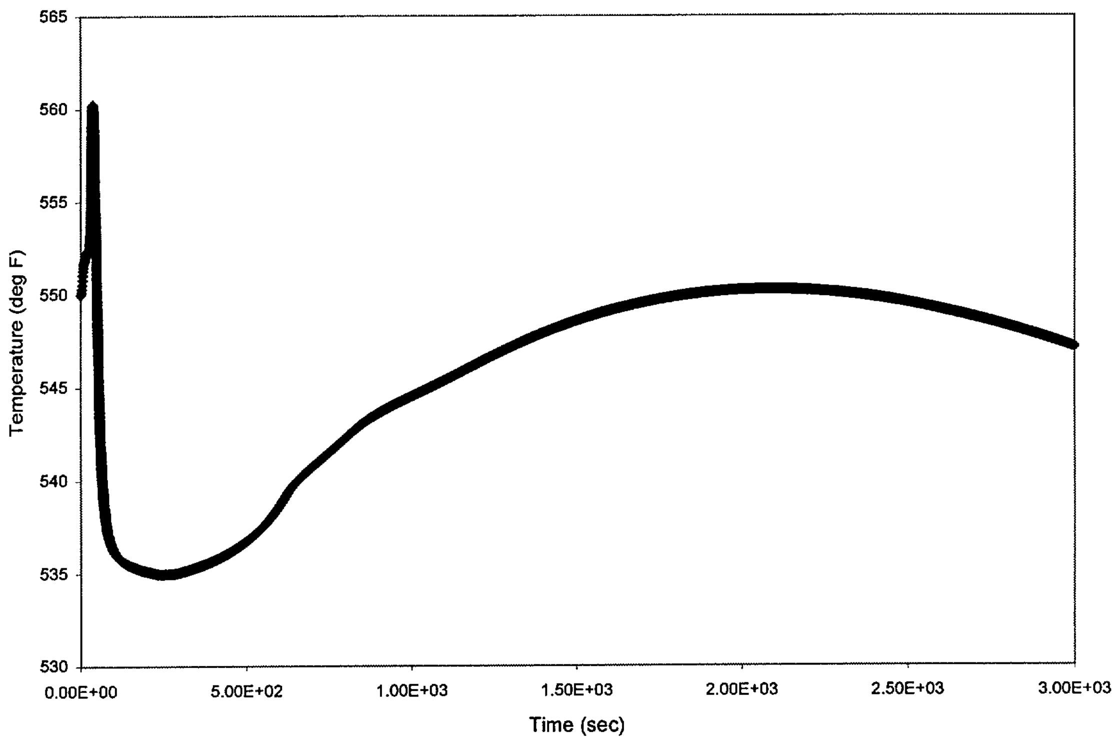


FIGURE 12

LOFW EVENT FOR MAXIMUM STEAM GENERATOR INVENTORY DEPLETION PRESSURIZER LEVEL VS. TIME

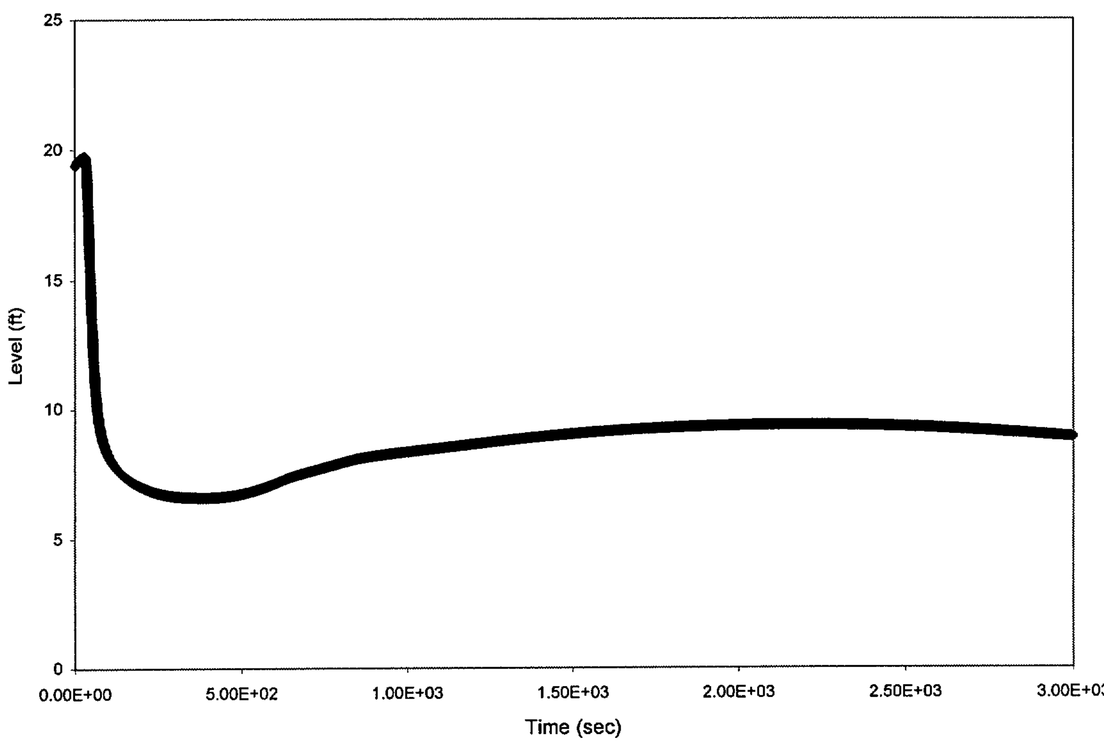


FIGURE 13

LOFW EVENT FOR MAXIMUM STEAM GENERATOR INVENTORY DEPLETION STEAM GENERATOR PRESSURE VS. TIME

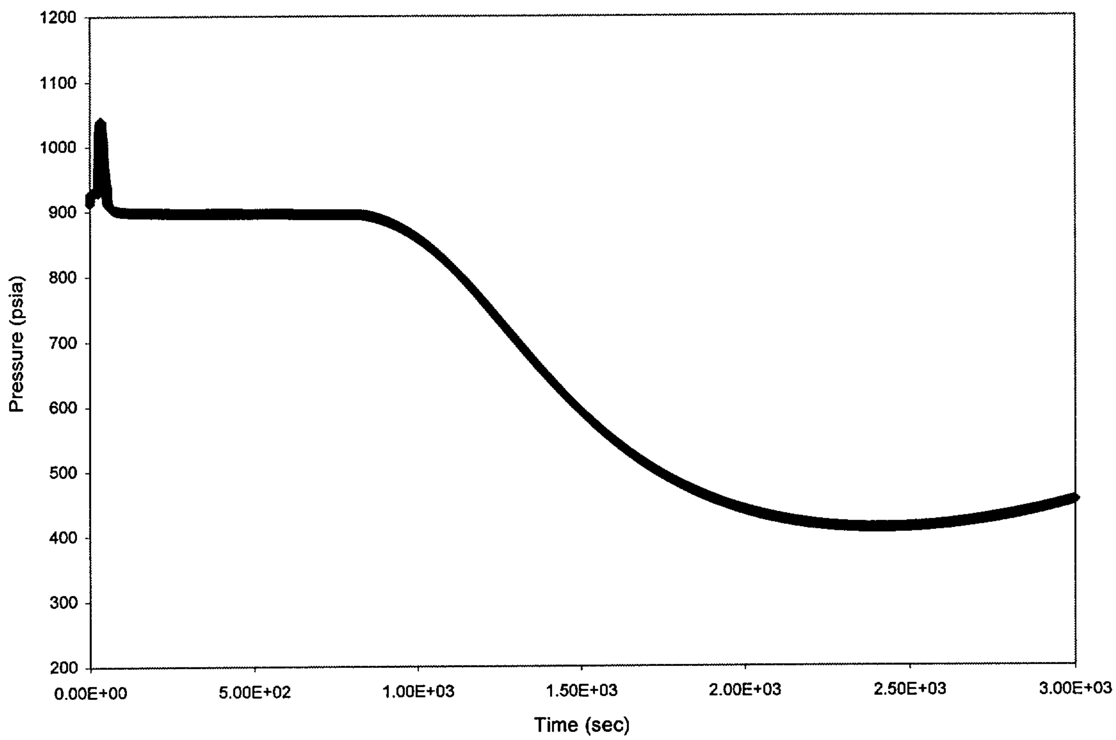
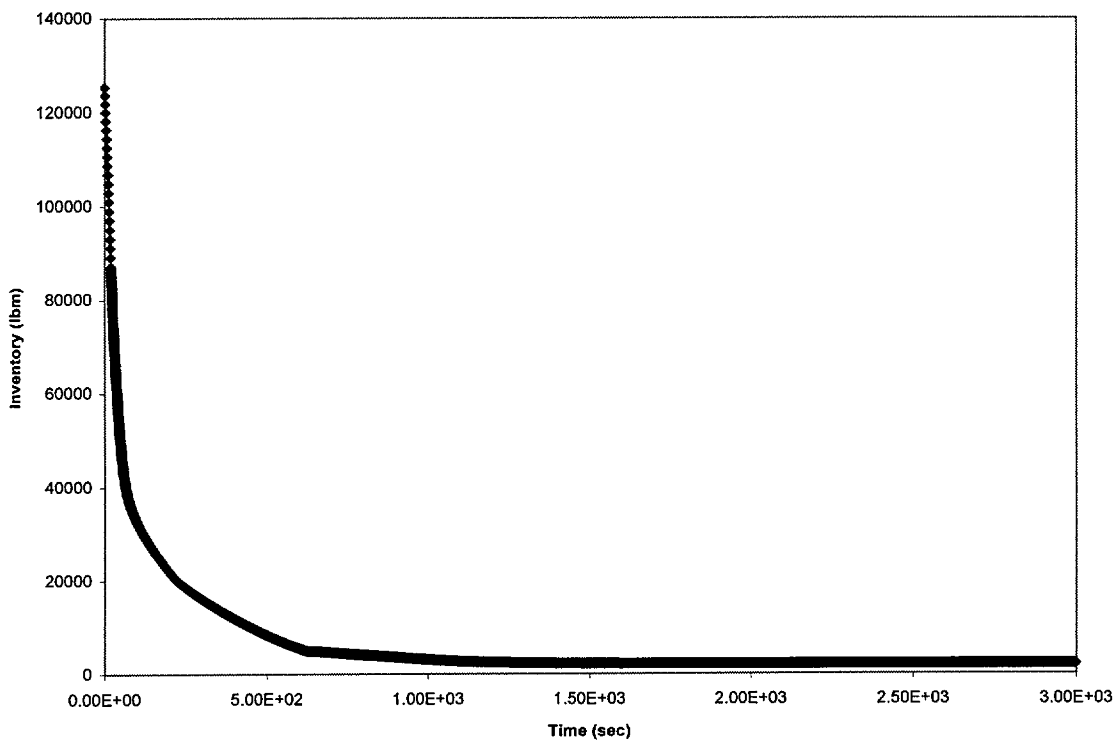


FIGURE 14

LOFW EVENT FOR MAXIMUM STEAM GENERATOR INVENTORY DEPLETION STEAM GENERATOR INVENTORY VS. TIME



ATTACHMENT (2)

DETERMINATION OF SIGNIFICANT HAZARDS

DETERMINATION OF SIGNIFICANT HAZARDS

This proposed change to the Renewed Operating Licenses submits changes to the Loss of Feedwater Flow analysis in the Updated Final Safety Analysis Report (UFSAR). The current analysis contained several non-conservative assumptions, resulting in the need for reanalysis. Prior Nuclear Regulatory Commission review is required due to changes in the methodology and acceptance criteria that will be used for the Loss of Feedwater Flow analysis.

The proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to not involve a significant hazards consideration in 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.*

   A Loss of Feedwater Flow event is defined as a reduction in feedwater flow to the steam generator without a corresponding reduction in steam flow from the steam generator. 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 a Loss of Feedwater Flow event. The most limiting Loss of Feedwater Flow event at full power is 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 steam generator inventory.

   Three areas are evaluated during the Loss of Feedwater Flow event: Reactor Coolant System (RCS) pressure, secondary system pressure, and the depletion of steam generator inventory. Each of these areas is evaluated independently to allow the assumptions to be adjusted to maximize the effect of the event in these different areas. The analysis must demonstrate that the specified acceptable fuel design limits (SAFDLs) continue to be met. This ensures that the dose consequences of the event are controlled within the acceptance criteria given in the UFSAR.

   This reanalysis modifies several of the assumptions in the UFSAR analysis to more accurately reflect the plant response to the event, to account for a single-failure, and to update the assumptions to more recent standards. These changes do not reflect a change in the probability of the event. In fact, the initiating event (closure of the feedwater regulating valves) has not changed. Therefore, the proposed change does not increase the probability of an accident previously evaluated.

   The results of all three portions of the Loss of Feedwater Flow analysis are given here. The new maximum calculated RCS pressure is 2620 psia. This portion of the analysis demonstrates that the Reactor Protective System (RPS) and the pressurizer safety valves prevent the RCS from exceeding 110% of RCS design pressure. The previous analysis calculated a maximum RCS pressure of 2631 psia. The new maximum RCS pressure is lower than the previously calculated result.

   The new maximum calculated steam generator pressure is 1107 psia. This portion of the analysis demonstrates that the RPS and the main steam safety valves prevent the secondary system from exceeding 110% of the steam generator design pressure. The previous analysis calculated a maximum secondary side pressure of 1080 psia. The new secondary side pressure is higher than the previously calculated pressure, but remains below the limit.

   For the steam generator depletion portion of the analysis, the actions of the RPS, Auxiliary Feedwater Actuation System (AFAS), and Auxiliary Feedwater (AFW) System are adequate to prevent exceeding the SAFDLs. This analysis considers an automatic isolation of steam generator blowdown and operator actions at 10 minutes to adjust AFW flow to provide acceptable results for this portion of the event. These two assumptions are differences between this analysis and the previous one. This analysis continues to show that the SAFDLs are protected.

The Loss of Feedwater Flow event has been analyzed with respect to RCS peak pressure, secondary system peak pressure, and steam generator depletion criteria. As noted above, the analysis of the event for all of these areas concludes with acceptable results. The response of the required safety systems is sufficient to prevent exceeding the SAFDLs and/or the RCS and secondary system pressure limits. Since the SAFDLs continue to be met, the dose consequences of the event are controlled within the acceptance criteria given in the UFSAR. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

Based on the above discussion, the proposed change does not increase 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 change affects a previously evaluated accident, but creates no new or different type of accident. Two areas are included in this analysis that had not been included in the previous analysis: operator action and closure of the steam generator blowdown valves by an AFAS signal. The operator actions assumed in the analysis have been evaluated with respect to the Emergency Operating Procedures to ensure that the actions would occur in a timeframe consistent with the analytical assumptions. Operators would have adequate indications to respond with the correct actions within the allotted timeframe. These assumptions are consistent with the UFSAR.

It should also be noted that the steam generator blowdown valves are safety-related valves. The closure signal that will be installed on these valves is also safety-related and is provided by the AFAS. Therefore, the valves can be assumed to perform their function and close following an AFAS signal during this event. This is a mitigating system and therefore would not initiate any new or different accident. The steam generator blowdown system is currently designed to isolate on a high radiation signal and a containment spray actuation signal, and therefore adding the AFAS isolation signal does not change the normal operation of the blowdown system. Inadvertent isolation of blowdown does not create the possibility of a new or different type of accident.

We are not proposing to alter the function of any other equipment or have it operate differently than it was designed to operate. All other equipment required to mitigate the consequences of an accident would continue to operate as before. Therefore, this 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 the margin of safety.*

The margin of safety defined by 10 CFR Part 100 has not been significantly reduced. The actions of the Reactor Protective System, AFAS, and AFW System are adequate to prevent exceeding the design pressure limits and the SAFDLs. The results of the event do not result in exceeding any pressure limit for the RCS or the secondary system, or any SAFDL. Although the secondary pressure is higher than previously analyzed, it remains within the pressure limit for the system. An automatic closure of the blowdown isolation valves on an AFAS signal is now required, along with operator action to ensure that the results of the analysis remain below the acceptance limits. These changes are within the acceptance criteria for anticipated operational occurrences as defined in the UFSAR. Since the analysis demonstrates that the SAFDLs continue to be met, the dose consequences of the event continue to be controlled within the acceptance criteria given in the UFSAR.

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