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RECIP.NAME RECIPIENT AFFILIATION
MARTIN,J.B. Region 5 (Post 820201)

SUBJECT: Forwards results of analysis performed for justification for continued operation w/potential for single failure causing opening of all steam bypass control valves,per 901101 ltr.

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WILLIAM F. CONWAY
EXECUTIVE VICE PRESIDENT
NUCLEAR

December 31, 1990

161-03678-WFC-JST

Docket Nos. STN 50-528/529/530

Mr. John B. Martin
Regional Administrator, Region V
U. S. Nuclear Regulatory Commission
1450 Maria Lane, Suite 210
Walnut Creek, California 94596-5368

Reference: Letter from W. F. Conway, APS, to J. B. Martin, USNRC, (161-03569-WFC/MEP/RAB) dated November 1, 1990
Subject: Justification For Continued Operation - Potential for a Single Failure Causing the Opening of All Steam Bypass Control Valves

Dear Mr. Martin:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Results of Analysis Supporting Justification for Continued Operation - Potential for a Single Failure Causing the Opening of All Steam Bypass Control Valves
File: 90-056-026

The attachment to this letter provides the results of the safety analyses performed by Arizona Public Service (APS) in support of the referenced Justification for Continued Operation (JCO). The results of these analyses demonstrate the safety of continued operation of PVNGS under the provisions of the JCO. The three events analyzed to support the JCO were: (1) a parametric study of the inadvertent opening of all steam bypass control valves (SBCVs) from less than 100 percent power (the results of the 100 percent power case were transmitted in the referenced JCO), (2) control element assembly ejection (CEAE) with opening of all SBCVs considered as a concurrent single failure, (3) inadvertent opening of all SBCVs at full power with a loss of offsite power as a single failure. These were the Updated Final Safety Analysis Report (UFSAR) Chapter 15 events identified in the JCO as being affected by inadvertent opening of the SBCVs. In addition to the Chapter 15 events analyzed as a result of the JCO, APS performed an analysis of steam bypass control system (SBCS) failures concurrent with a reactor power cutback system (RPCS) actuation. This analysis was not required for the JCO but was identified as part of APS's comprehensive review of events potentially affected by SBCS failure.

The analysis of Event 1, the inadvertent opening of all SBCVs from 95 percent power, was determined to be the limiting case for excess steam demand anticipated operational occurrences. This event was more limiting than an event occurring from 100 percent power because this power level represented the point of minimum Core Operating Limit Supervisory System (COLSS) margin. The result of this

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/ 1

analysis demonstrated that the five percent additional reserve overpower margin (ROPM) placed in core operating limit supervisory system (COLSS) was conservative and that only one percent additional ROPM was required to ensure no Specified Acceptable Fuel Design Limits (SAFDL) would be exceeded. The analyses of Events 2 and 3 identified in the attachment were found to be bounded by the currently docketed analyses.

The fourth event analyzed, failure of the SBCS concurrent with a RPCS actuation, concluded that the additional five percent ROPM placed in COLSS as a result of the JCO will prevent exceeding a SAFDL. The analysis also demonstrated that a reduction in the control element assembly calculator (CEAC) reactor power cutback timer addressable constant, TCBP, and the core protection calculator (CPC) addressable constant, TCBSP, to 16 seconds would prevent exceeding a SAFDL even without the additional five percent ROPM.

Arizona Public Service intends to continue operation under the JCO until the end of the first quarter of 1991. At that time, the Failure Modes and Effects Analysis of the Steam Bypass Control System (SBCS) will be completed and a determination will be made as to whether the SBCS can be restored to its original design basis (no single failure will result in the opening of more than one SBCV). If there is reasonable assurance that the SBCS can be restored to its design basis, APS will continue operation under the provisions of the JCO and a schedule for completion of this project will be provided. If the SBCS cannot be restored to its design basis, the results of these analyses will be submitted for NRC review and incorporated into the UFSAR.

If you have any questions concerning this matter, please contact Michael E. Powell of my staff at (602) 340-4981.

Sincerely,

WFConway

WFC/JST

Attachment

cc: Document Control Desk (all w/attachment)
J. B. Martin
C. M. Trammell
D. H. Coe
A. H. Gutterman
A. C. Gehr

bcc: J. N. Bailey (1966)
J. M. Levine (6125)
E. C. Simpson (1962)
W. F. Quinn (1502)
M. E. Powell (1515)
R. A. Bernier (1515)
G. R. Overbeck (6102)
P. F. Crawley (1605)

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

EVENT 1: INADVERTENT OPENING OF ALL STEAM BYPASS CONTROL SYSTEM VALVES AT 95 PERCENT OF FULL POWER

The inadvertent opening of all the steam bypass control system valves (IOSBCSV) event at full power (i.e., 102% rated power) was presented in Reference (1). The determination of the COLSS margin requirements for excess load events involving the spurious opening of all steam bypass valves at other than full power is addressed in this report. The limiting IOSBCSV case for less than full power conditions is the 95% full power case, presented below.

1.1 Identification of Event and Causes

The inadvertent opening of all the steam bypass control system valves (IOSBCSV) event from 95% full power results in an increase in heat removal by the secondary system greater than that previously analyzed in the PVNGS UFSAR, Section 15.1, Revision 2. The IOSBCSV event is analyzed from a power operating limit at 95% full power to determine if the minimum DNBR resulting from this event violates the SAFDL (DNBR > 1.24), and to determine if additional margin is required.

For this event, the major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a fuel design limit has been violated and thus whether fuel cladding degradation might be anticipated. Those factors which cause a decrease in local DNBR are:

a. increasing coolant temperature
b. decreasing coolant pressure
c. increasing local heat flux (including radial and axial power distribution effects)
d. decreasing coolant flow

1.2 Sequence of Events and Systems Operation

The inadvertent opening of the steam bypass control system (SBCS) valves increases the rate of heat removal by the steam generators, causing a rapid cooldown of the reactor coolant system (RCS). Opening all eight SBCS valves increases the steam flow by 88% of full power steam flow. Due to the negative moderator temperature coefficient (MTC) assumed for this event, core power increases from the initial value of 95% of rated core power to a value of 110.6%, at which time the reactor trips on CPC variable overpower. The CPC variable overpower trip (VOPT) is

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

conservatively delayed by 0.3 seconds. The feedwater control system, which is assumed to be in the automatic mode, supplies feedwater to the steam generators such that steam generator water levels are maintained.

Following the generation of a turbine trip on reactor trip, the primary and secondary systems will respond essentially as described in Reference (1).

1.3 Analysis of Effects and Consequences

A. Mathematical Model

The nuclear steam supply system (NSSS) response to the IOSBCSV event was simulated using the CESEC-III computer program described in UFSAR section 15.0.3. The time-dependent thermal margin on DNBR in the reactor core was calculated using the CETOP-D computer program which uses the CE-1 critical heat flux correlation described in Chapter 4 of the UFSAR.

B. Input Parameters and Initial Conditions

Table 1-1 lists the assumptions and initial conditions used for this event in addition to those discussed in UFSAR Section 15.0. Conditions were chosen such that, for an event initiated at 95% of rated core power with a reserved overpower margin of 115%, the overpower condition caused by the increase in steam flow results in the maximum degradation of margin from the specified acceptable fuel design limits (SAFDL).

C. Results

The dynamic behavior of key NSSS parameters following the IOSBCSV is presented in Figures 1-1 through 1-5. Table 1-2 summarizes the major events, times and results for this transient.

The IOSBCSV increases the rate of heat removal by the steam generators, causing cooldown of the RCS. Due to the negative moderator reactivity coefficient, core power increases from 95% of rated core power to 110.6% at 6.32 seconds, at which time a reactor trip signal occurs on CPC variable overpower. At 7.2 seconds, the control rods begin inserting into the core. Power increases to a maximum of 120.0% at 8.35 seconds, before dropping in response to the reactor trip, as shown in Figure 1-1. At 8.85 seconds a minimum DNBR of 1.23 occurs, followed by a rapid increase in DNBR as shown on Figure 1-2. Since a minimum DNBR of 1.23

was unacceptable it was determined that a COLSS overpower margin of 115.4% (rounded up to 116%) was required to ensure that the IOSBCSV event does not violate the SAFDL for power levels below full power.

D. Conclusions

The margin requirements at 95% full power were analyzed because this power level represents the point of minimum COLSS margin. A COLSS overpower margin of 116% is sufficient to ensure that the SAFDL is not violated and no fuel failure will occur in the event of a IOSBCSV.

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

TABLE 1-1
ASSUMPTIONS AND INITIAL CONDITIONS FOR 95% POWER INADVERTENT OPENING OF THE STEAM BYPASS CONTROL SYSTEM VALVES (8)

Parameter	Value
Core Power Level, MWt[1]	3632
Core Inlet Coolant Temperature, °F	570
Core Mass Flow rate, 10^6 lbm/hr	147.2
Pressurizer Pressure, psia	2200
Pressurizer Water Volume, ft^3	918
Steam Generator Pressure, psia	1070
Steam Generator Inventory, lbms per SG	174,000
CEA Worth on Trip, 10^{-2} delta rho	-6.5
Core Burnup	End of Cycle
ASI	-0.32
Max. Radial Peaking Factor	1.56
MTC, 10^{-4} delta rho/°F	-3.5
FTC	Beginning of Cycle (min.)
Gap Conductance, 10^6 Btu/ft^2-hr-°F	1.814

1 The time-dependent thermal margin for calculating DNBR was initiated at the power operating limit, with a required overpower margin of 115%.

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

TABLE 1-2
SEQUENCE OF EVENTS FOR 95% POWER
INADVERTENT OPENING OF THE STEAM BYPASS CONTROL SYSTEM VALVES (8)

Time	Event	Setpoint or Value
0.0	SBCS Valves begin to open	--
0.1	SBCS valves fully open	--
6.32	Reactor trips, (VOPT)	110.6%
7.2	CEA's drop into core	--
8.35	Power peak occurs, % full power	120.0
8.85	Minimum DNBR occurs	1.23

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

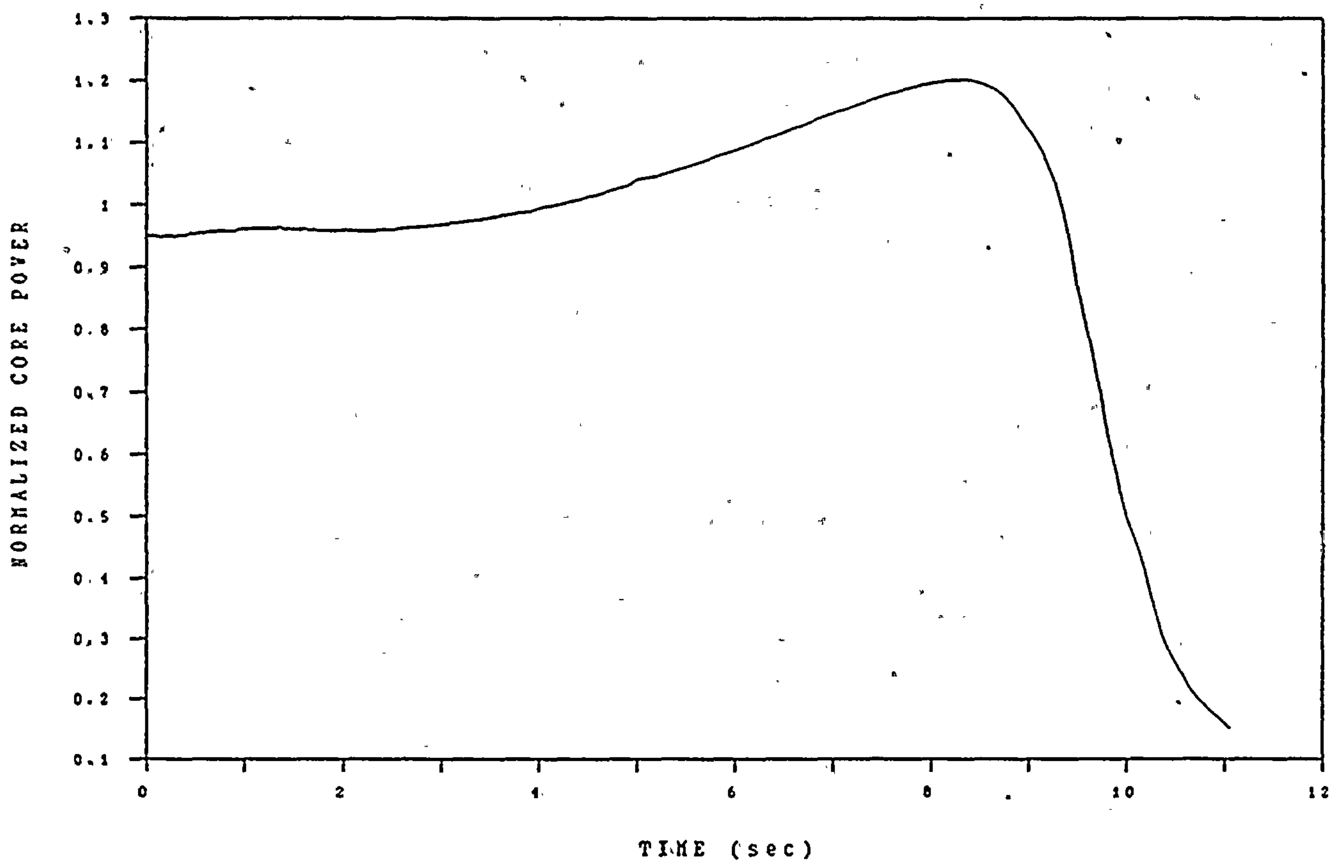

Figure 1-1: Normalized Core Power

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

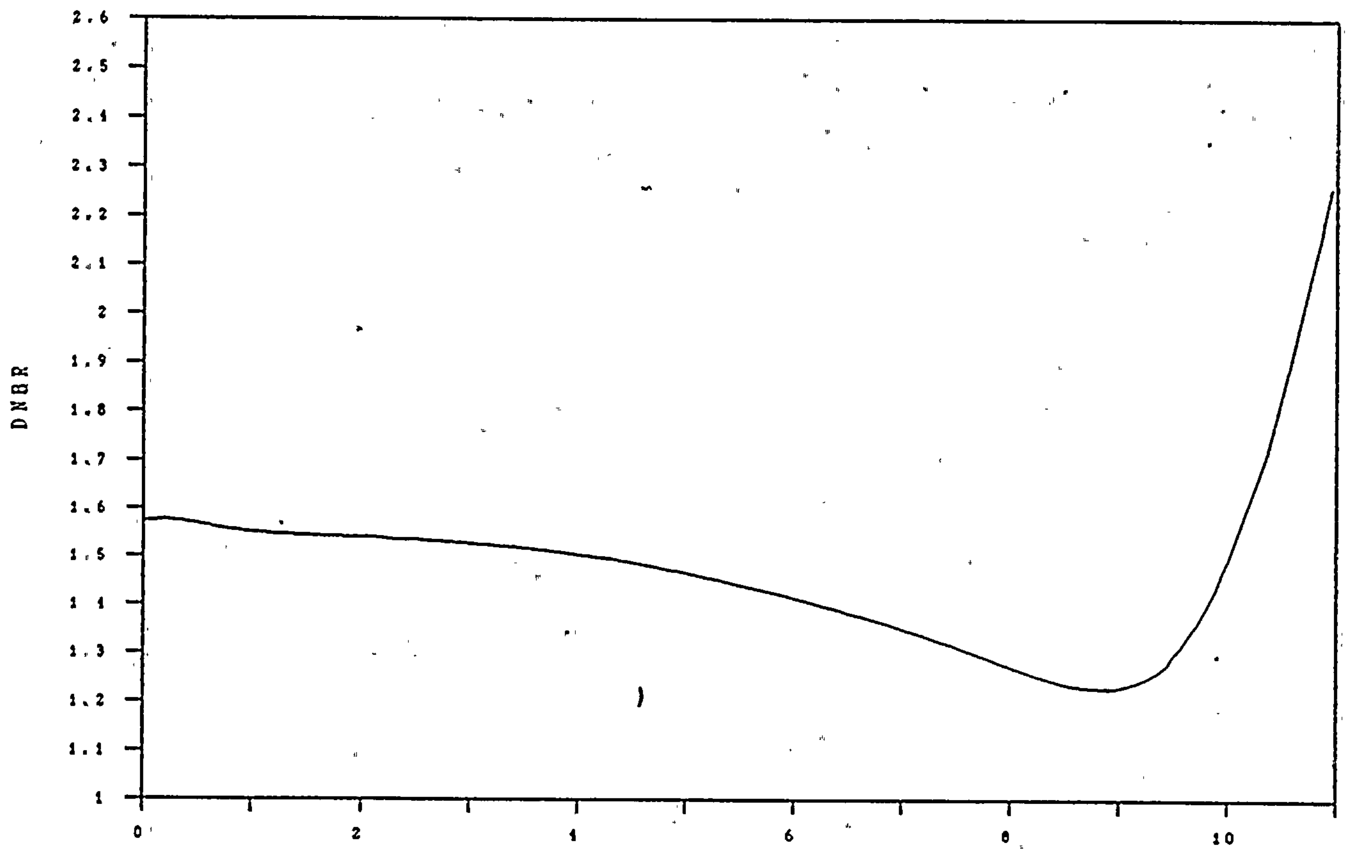

Figure 1-2: DNBR vs. Time

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

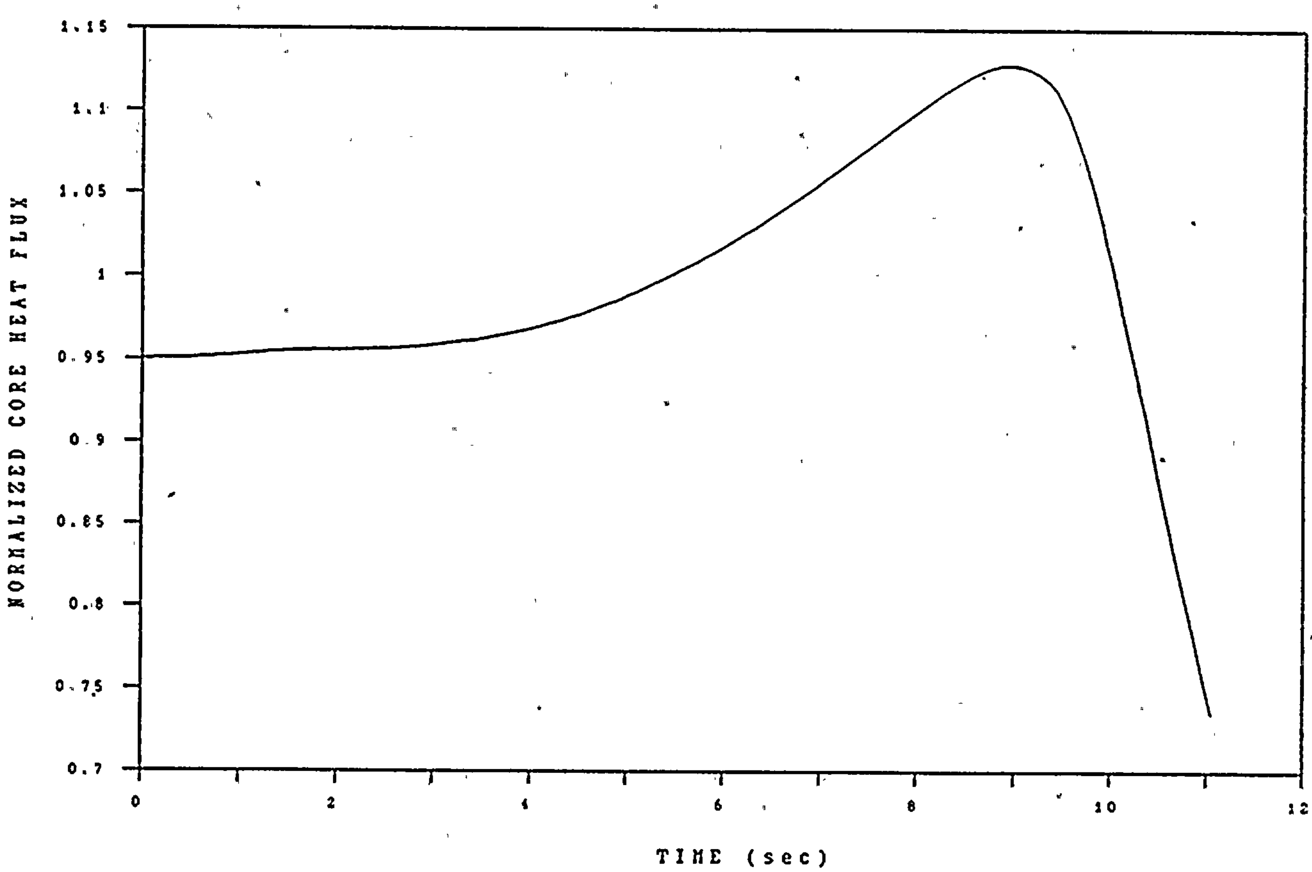

Figure 1-3: Normalized Core Heat Flux

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

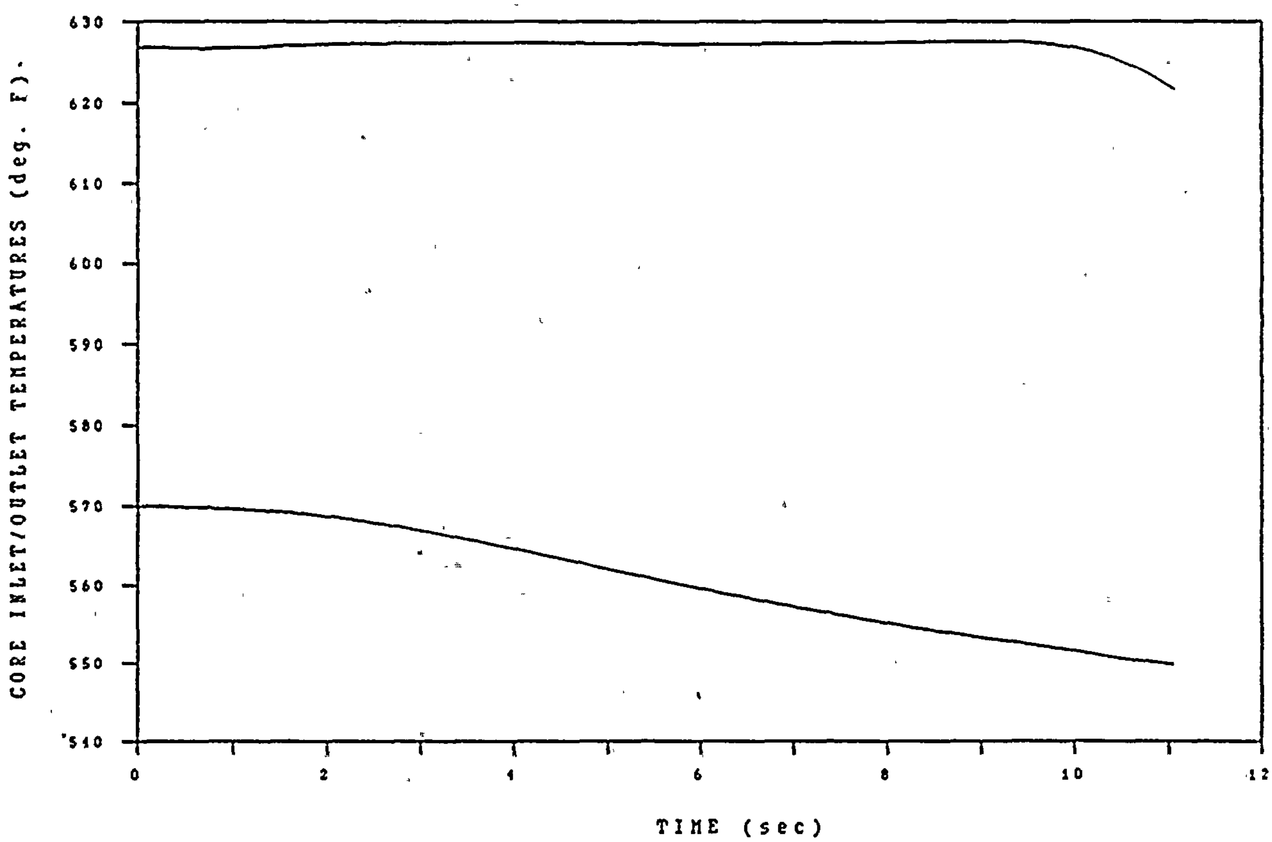

Figure 1-4: Core Inlet & Outlet Temperatures

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

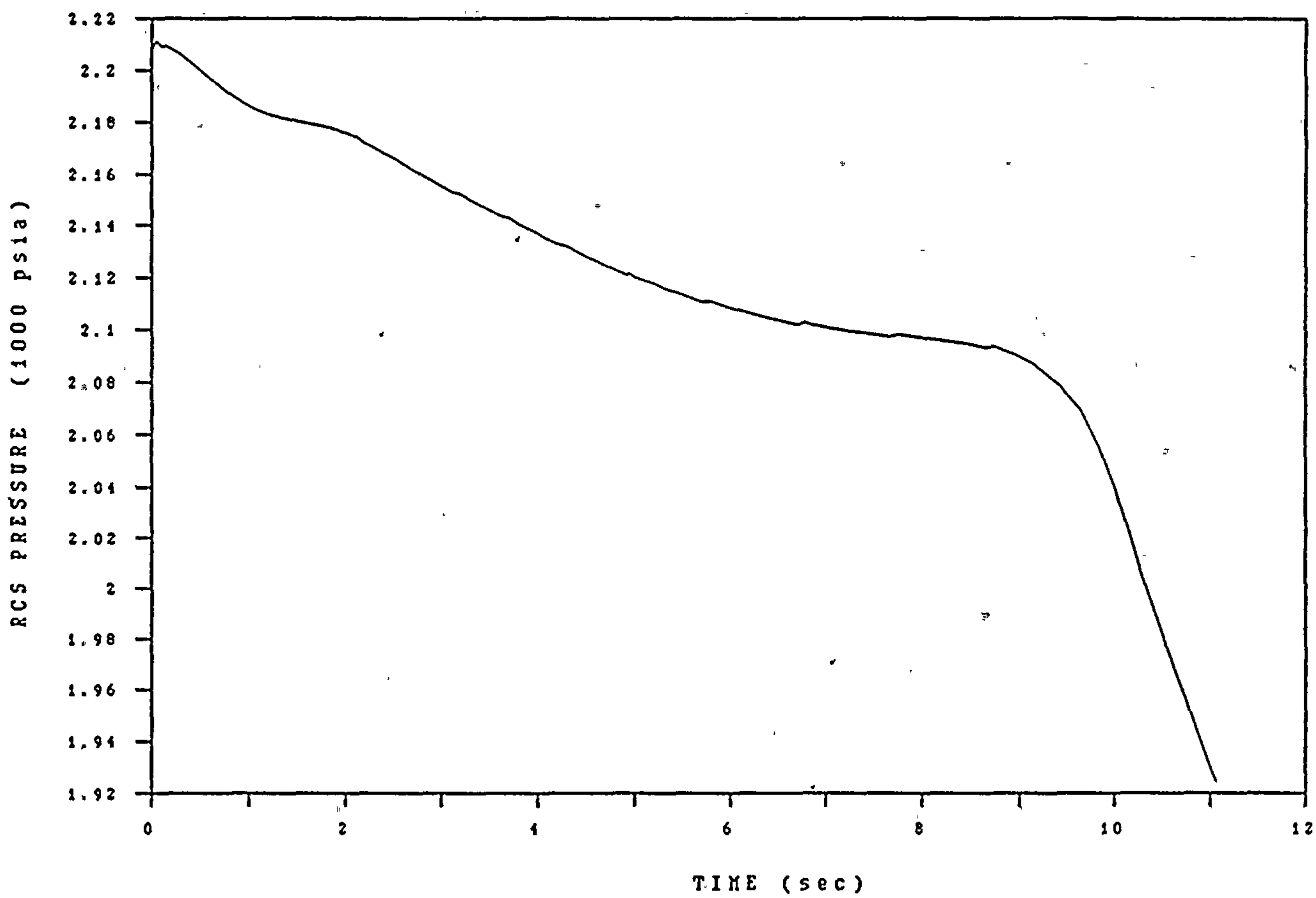

Figure 1-5: RCS Pressure

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

EVENT 2: <u>THE CEA EJECTION EVENT WITH INADVERTENT OPENING OF STEAM BYPASS CONTROL SYSTEM VALVES (IOSBCSV)</u>

2.1 <u>Identification of Event and Causes</u>

In accordance with Reference (1), the CEA ejection analysis was reanalyzed to evaluate the potential combined effects of a CEA ejection event with a postulated IOSBCSV.

The CEA ejection analysis presented in FSAR Section 15.4.8 results from a postulated circumferential rupture of the control element drive mechanism (CEDM) housing or the CEDM nozzle. This event assumes a loss of offsite power following turbine trip, resulting in a coastdown of all four RCPs. Because the RCP coastdown occurs after trip, the loss of offsite power has no effect on the calculated fuel failure, but does affect the calculated dose. Since the condenser is unavailable following loss of offsite power, the Atmospheric Dump Valves are used to cool the plant. Thus, steam release (and subsequent dose) occurs through the MSSVs and the atmospheric dump valves.

The primary objective of the reanalysis is to determine the impact of a combined, delayed trip CEAE with IOSBCSV on the corresponding fuel failure calculation. An examination of the limiting CEAE case presented in the UFSAR determined that the inadvertent opening of the SBCSVs during this event would not result in a higher fuel failure, nor would it significantly impact the dose calculation. Fuel failure is terminated by reactor trip, which occurs very early in the transient, before an excess steam demand could affect DNBR in the primary system. Additionally, the secondary system would be isolated by a Main Steam Isolation Signal very shortly after actuation of the SBCS. Due to the secondary system pressure drop, the resumption of steam release via the MSSVs would be delayed until the steam generators repressurized due to transfer of decay heat from the primary system. Thus, the combination of the IOSBCSV with the limiting CEAE full power (i.e., 102% power) case does not significantly impact the steam release or calculated dose.

In addition to the limiting case presented in the UFSAR, a spectrum of CEA ejection cases were examined at different power levels using various ejected CEA worths. As noted above, those cases which result in an immediate reactor trip would not be adversely impacted by the IOSBCSV. Some cases, however, do not generate an immediate reactor trip, such that the reactor could reach a quasi-steady state condition prior to initiation of the IOSBCSV portion of the transient.

Two delayed trip CEAE with IOSBCSV events were therefore analyzed to examine the effect of the SBCS failure on calculated fuel failure. These cases were initiated at 100% and 75% power levels, and the amount of fuel failure determined and compared to the fuel failure

calculated in the UFSAR for the limiting CEAE case. For both cases, the amount of predicted fuel failure was less than that documented in the UFSAR. The 100% power case is presented below.

2.2 Sequence of Events and Systems Operation

The delayed trip CEAE with IOSBCSV is initiated by a CEA ejection from 100% power. UFSAR Table 15.4.8-1 presents a chronological sequence of events which would occur during the CEAE transient without a IOSBCSV, from the time of ejection until operator action is initiated at 1800 seconds. The table shows the high power trip occurring in less than one second, at a high power trip setpoint of 117%.[1]

In the reanalysis, initial conditions are selected such that no trip occurs due to the initial power spike from the CEA ejection. The high power trip is conservatively set to 132% power. This ensures that a reactor trip does not occur on the initial power spike, and conservatively bounds any trip delays due to temperature shadowing effects. All other trips are manually turned off. No loss of offsite power occurs, since the SBCS and condenser are assumed to be operable. An initial run was made to determine peak SG pressure and the time to initiate the IOSBCSV. The transient was then rerun, as described below.

The initial power spike due to the CEA ejection peaks at 118.4% power at 0.3 seconds. Power then begins to drop back to the original power level due to doppler feedback effects.

Steam generator pressure increases due to the initial power spike, reaching a maximum value of approximately 1067 psia at approximately 24 seconds. The SBCS valves opening setpoint is 1115 psia. Nevertheless, the SBCS valves are assumed to open at 24 seconds (with power at approx. 100.78%). The IOSBCSV increases the rate of heat removal by the steam generators, causing a rapid cooldown of the RCS. Opening all eight SBCS valves is conservatively assumed to increase the steam flow by 88% of full power steam flow. Due to the negative moderator temperature coefficient (MTC) assumed for this event, core power increases as shown in Figure 2-1. Based on the trip setpoint noted above, the high power trip should occur at 132%. However, for this analysis, the temperature decalibration option in the CESEC code delays the trip until power reaches 145.5% at approximately 43 seconds. Power peaks at approximately 146.6% rated power at 44 seconds. The minimum DNBR of 0.85 occurs at 45.35 seconds, as shown on Figure 2-2.

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[1] Actual high power trip setpoint is 110%. An additional 7% bias is included to bound temperature shadowing effects.

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

2.3 Analysis of Effects and Consequences

A. Mathematical Model

The nuclear steam supply steam (NSSS) response to the CEAE with IOSBCSV event was simulated using the CESEC-III computer program described in FSAR section 15.0.3. The time-dependent thermal margin on DNBR in the reactor core was calculated using the CETOP-D computer program which uses the CE-1 critical heat flux correlation described in UFSAR Chapter 4.

B. Input Parameters and Initial Conditions

Table 2-1 lists the assumptions and initial conditions used for this event in addition to those discussed in FSAR Section 15.0. Conditions were chosen such that, for an event initiated at 100% of rated core power, the overpower condition caused by the CEAE with IOSBCSV event results in the largest degradation in the thermal margin.

C. Results

The dynamic behavior of key NSSS parameters following the CEAE with IOSBCSV Event at 100% power is presented in Figures 2-1 through 2-5. Table 2-2 summarizes the major events, times and results for this transient.

Based on the minimum DNBR of 0.85, the fuel failure for the 100% power case was calculated at 9.38%. The corresponding fuel failure for the 75% power case was calculated at 9.5%. In both cases, fuel failure was less than the 9.8% fuel failure reported in UFSAR Section 15.4.8.

2.4 Conclusions

The CEAE with IOSBCSV event is bounded by the CEAE Event reported in UFSAR Section 15.4.8.

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

TABLE 2-1
ASSUMPTIONS AND INITIAL CONDITIONS FOR CEA EJECTION AT 100% POWER WITH INADVERTENT OPENING OF THE STEAM BYPASS CONTROL SYSTEM VALVES (8)

Parameter	Value
Core Power Level, MWt[1]	3822
Core Inlet Coolant Temp.,°F	560
Core Mass Flow rate,[2] 10^6 lbm/hr	--
Pressurizer Pressure, psia	2000
Steam Generator Pressure, psia	1060
Steam Generator Inventory, lbms per SG	174,000
Ejected CEA Worth, 10^{-2} delta rho	0.124
CEA Ejection Time, seconds	0.05
CEA Worth on Trip, 10^{-2} delta rho	-9.0
Core Burnup	End of Cycle
ASI	-0.33
Pre-Ejected Radial Peaking Factor	1.62
MTC, 10^{-4} delta rho/°F	-3.5
FTC	Beginning of Cycle (min.)
Gap Conductance, 10^6 Btu/ft²-hr-°F	1.94

[1] The time-dependent thermal margin for calculating DNBR was initiated at the power operating limit, with a required overpower margin of 115%.

[2] Calculated by CETOP-D code to initiate transient from a power operating limit.

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

TABLE 2-2
SEQUENCE OF EVENTS CEA EJECTION FROM 100% POWER WITH INADVERTENT OPENING OF THE STEAM BYPASS CONTROL SYSTEM VALVES (8)

Time	Event	Setpoint or Value
0.0	CEA ejection occurs	--
0.3	Initial power spike occurs, % full power	118.4%
24	Steam Generator pressure peaks, psia	1067
24	SBCSVs assumed to open, % excess load	88%
43	Reactor trips on high power,[1] % full power	145.5%
44	Second power peak occurs, % full power	146.6%
45.35	Minimum DNBR occurs	0.85

1 Reactor trip set to 132%. With temperature decalibration option turned on, trip is conservatively delayed until 145.5%.

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

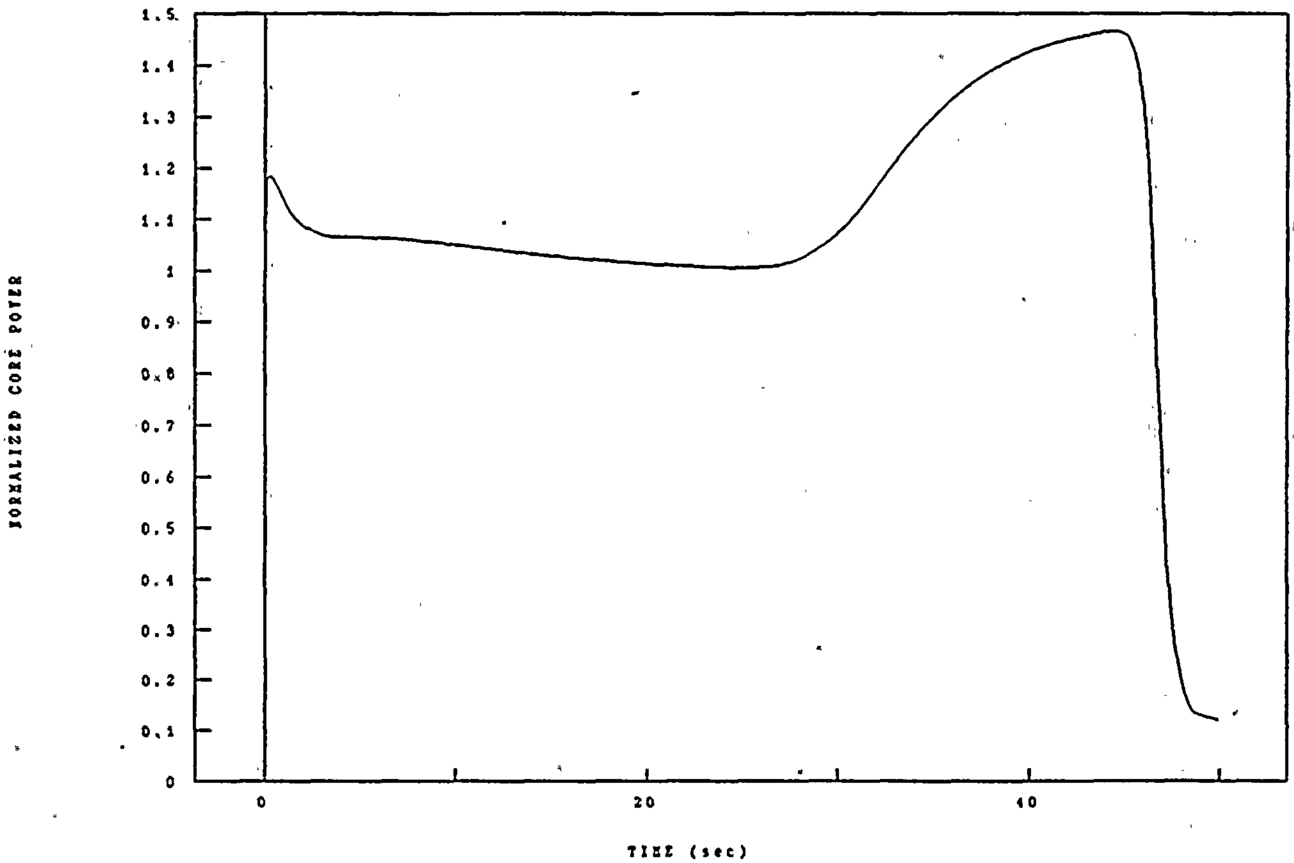

Figure 2-1: CEAE plus IOSBCSV, Normalized Core Power

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

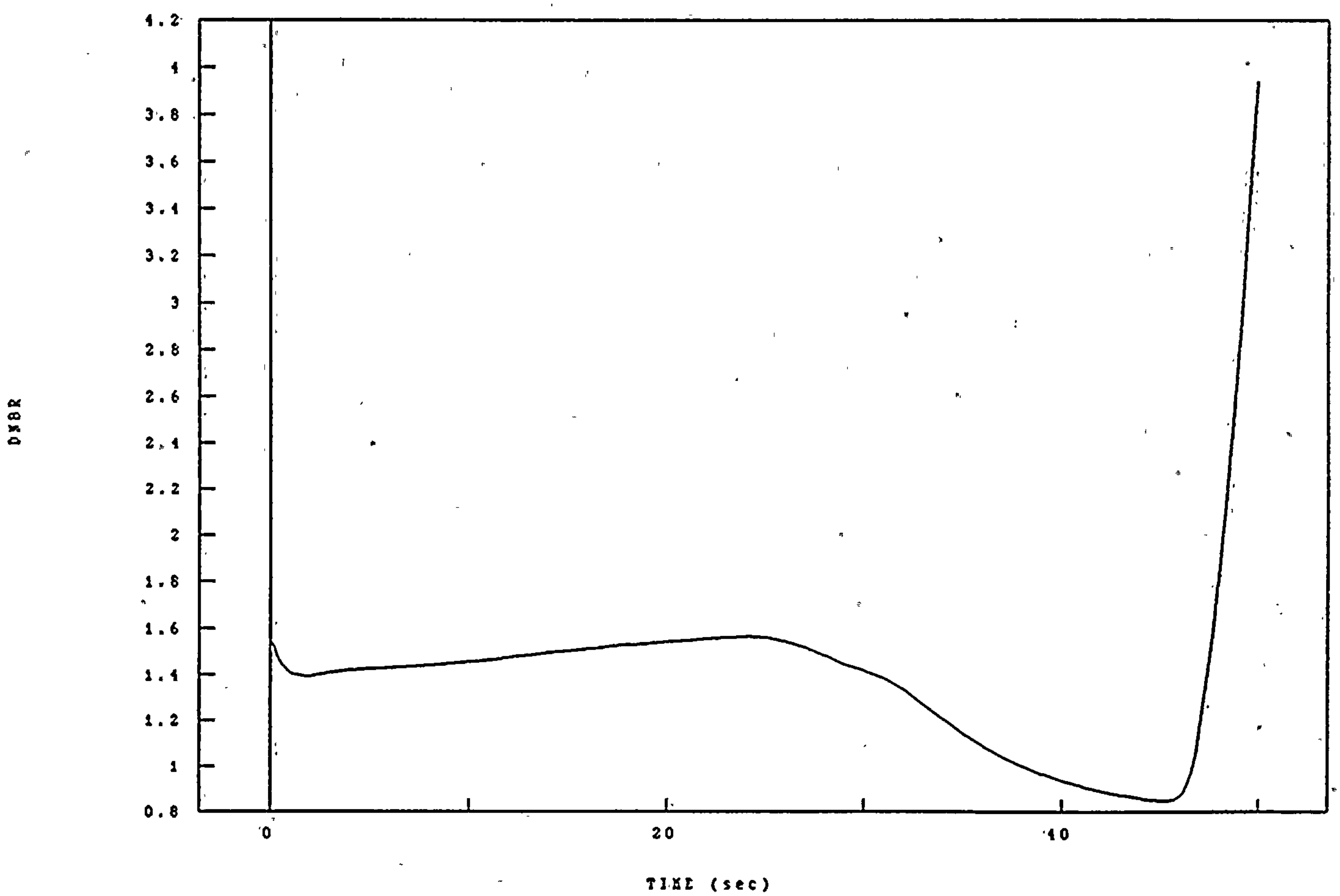

Figure 2-2: CEAE plus IOSBCSV, DNBR vs. Time

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

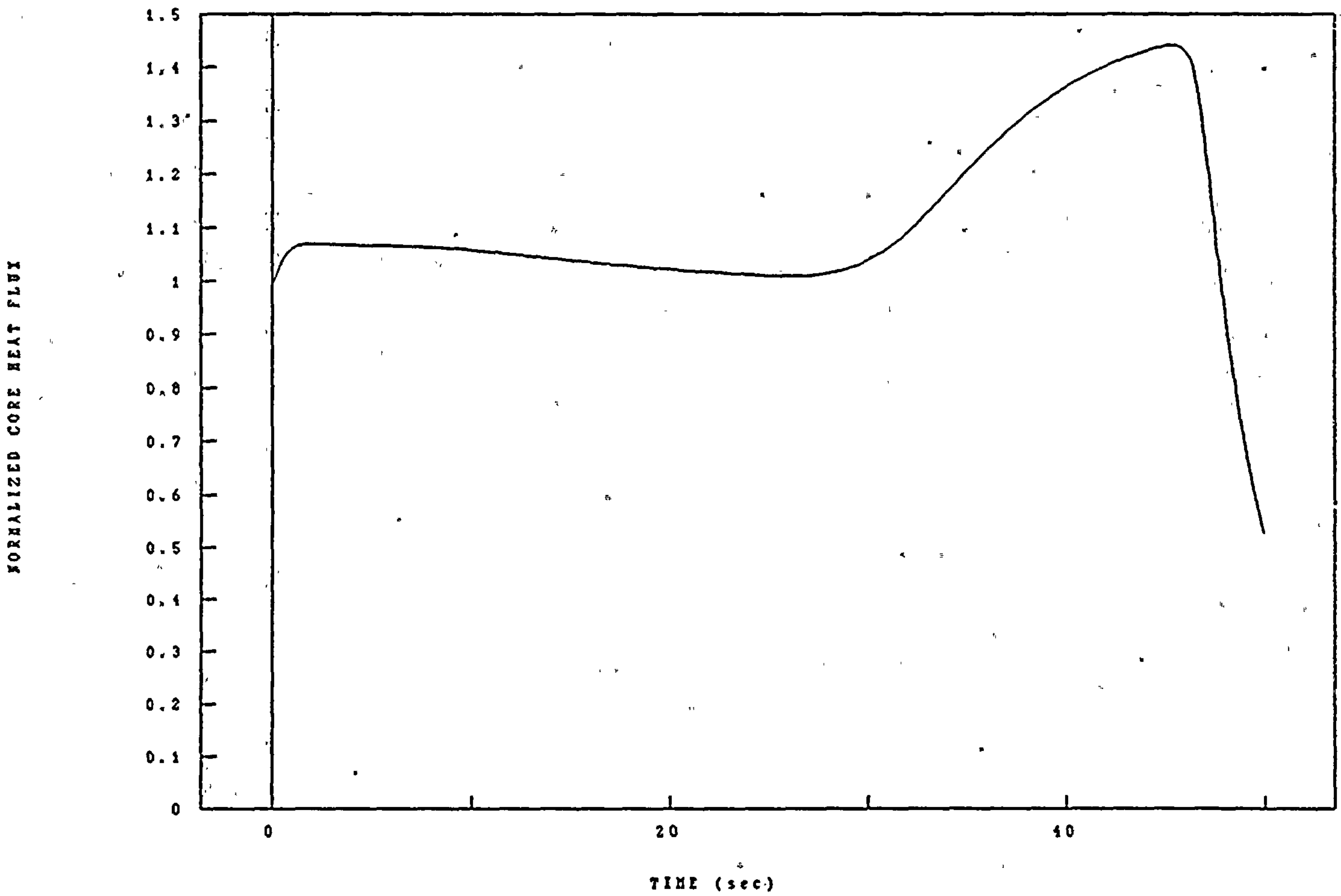

Figure 2-3: CEAE plus IOSBCSV, Normalized Core Heat Flux

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

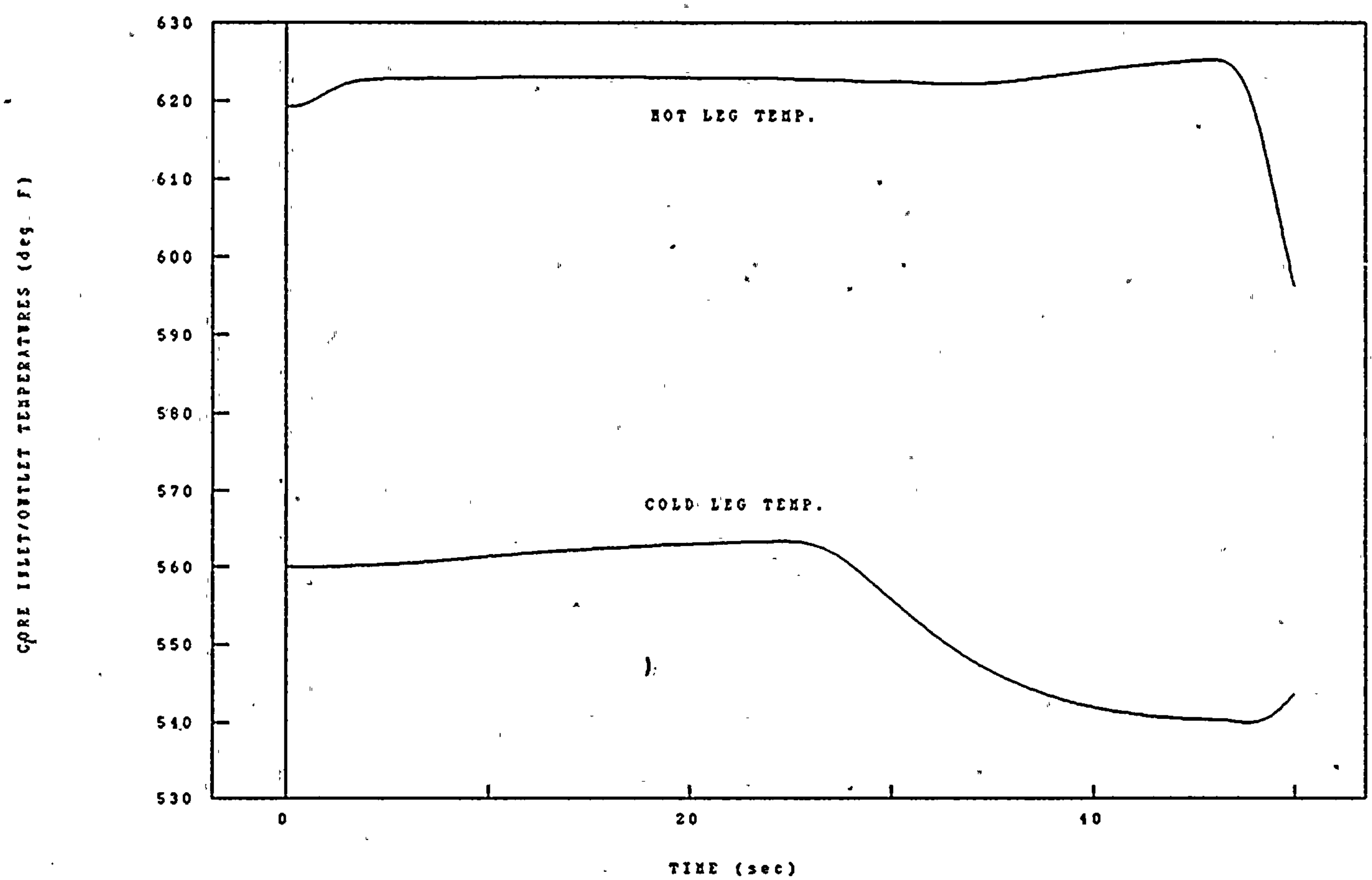

Figure 2-4: CEAE plus IOSBCSV, Hot/Cold Leg Temperatures

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

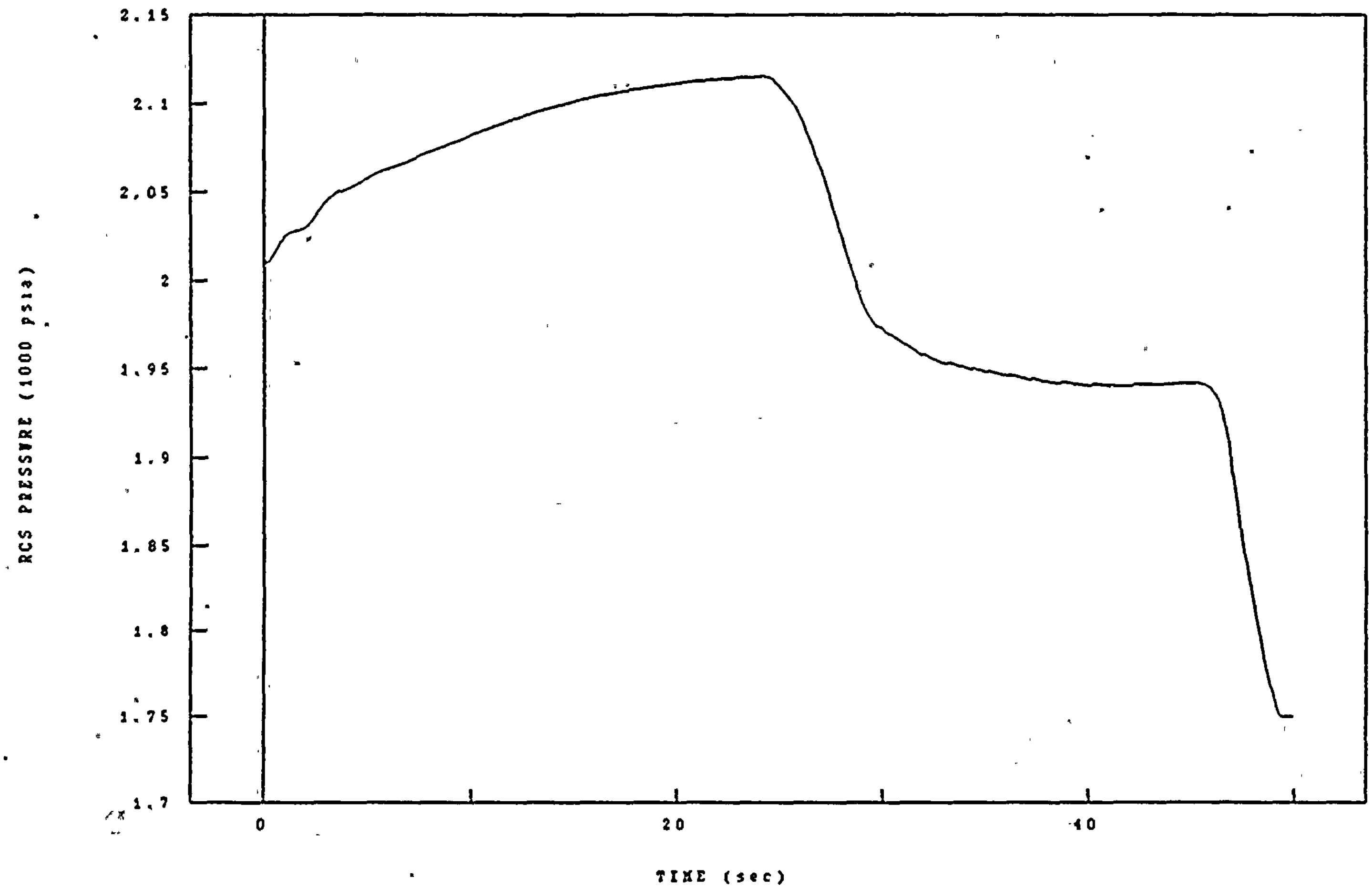

Figure 2-5: CEAE plus IOSBCSV, RCS Pressure

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

EVENT 3: INADVERTENT OPENING OF ALL STEAM BYPASS CONTROL SYSTEM VALVES AT FULL POWER WITH LOSS OF OFFSITE POWER

3.1 Identification of Event and Causes

The most limiting combination of an Anticipated Operational Occurance (AOO) with a single failure in terms of minimum DNBR that is currently reported in the UFSAR is the Inadvertent Opening of a Steam Generator Dump Valve plus Loss Of AC power (IOSGADV + LOAC). In this scenerio, an Atmospheric Dump Valve (ADV) or a Steam Bypass Control Valve (SBCV) may be inadvertently opened by an operator, or may open due to a failure of the control system which operates the valve. The opening of these valves results in similar consequences because they release steam at the same maximum flow rate. The single failure which yields the minimum transient hot channel DNBR is the single failure which combines the greatest reduction in DNBR after the initiation of a reactor trip signal with the lowest possible pre-trip DNBR. Previous analyses (refer to UFSAR and RAR) showed that the LOAC power is the most limiting single failure for the excess load (IOSGADV) event. The Loss Of Flow (LOF) due to the four pump coast down that results from the LOAC causes a greater decrease in DNBR after reactor trip than any other possible single failure.

To assure that the results of the IOSGADV + LOAC event is the most limiting event, the event of an Inadvertent Opening of all eight SBCVs (IOSBCSV) with a concurrent LOAC has been analyzed and is presented in this section. Like the IOSGADV + LOAC event, the RCS pressure increase is limited by the pressurizer safety valves so that there is no challenge to the RCS pressure limit criteria. Hence, fuel performance is the major concern for the IOSBCSV + LOAC event.

For this event, the major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a fuel design limit has been violated and thus whether fuel cladding degradation might be anticipated. Those factors which cause a decrease in local DNBR are:

a. increasing coolant temperature
b. decreasing coolant pressure
c. increasing local heat flux (including radial and axial power distribution effects)
d. decreasing coolant flow

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

3.2 Sequence of Events and Systems Operation

It is assumed that the sequence of events in the IOSBCSV + LOAC transient is identical to the IOSBCSV event discussed in Reference (1) up to the time of minimum DNBR, at which time the LOAC is assumed to occur. The initial conditions for the IOSBCSV transient were selected to maximize degradation of the initial COLSS overpower margin. The previously analyzed IOSBCSV event from full power conditions does not result in a violation of the SAFDL. It has further been determined (see Event 1 of this report), that a IOSBCSV event from less than full power conditions does not violate the SAFDL if a 116% ROPM is preserved by either COLSS or CPC penalty factors. Hence, the LOAC portion of the transient is assumed to initiate at the conditions existing at the time of minimum DNBR. This ensures that the LOAC portion of the transient begins at the worst possible set of initial conditions. This is consistent with the methodologies used earlier for the IOSGADV + LOAC event (refer to UFSAR). Since this falls into the category of infrequent events, violation of SAFDL and fuel failure is permissible.

The inadvertent opening of the steam bypass control system (SBCS) valves increases the rate of heat removal by the steam generators, causing a rapid cooldown of the reactor coolant system (RCS). Opening all eight SBCS valves increases the steam flow by 88% of full power steam flow. Due to the negative moderator temperature coefficient (MTC) assumed for this event, core power increases from the initial value of 102% of rated core power to a value of 118%, at which time the reactor trips on CPC variable overpower. The CPC variable overpower trip (VOPT) is conservatively delayed by 0.3 seconds. At 8.95 seconds, the time of minimum DNBR, LOAC is initiated. The feedwater control system, which is assumed to be in the automatic mode, supplies feedwater to the steam generators such that steam generator water levels are maintained.

Following the generation of a turbine trip on reactor trip, the feedwater control system enters the reactor trip override mode and reduces feedwater flow to 5% of nominal, full power flow. If the low steam generator (SG) level setpoint is reached due to continued steaming through the SBCS, an auxiliary feedwater actuation signal (AFAS) is generated and the auxiliary feedwater pumps will actuate to provide additional feedwater. The steam generators will continue to blowdown until the main steam isolation valves close on low secondary system pressure (820 psia). Thereafter, the RCS and steam generators will heat up and repressurize until the main steam safety valve (MSSV) opening set pressures are reached. Steaming will then resume through the MSSVs to remove heat stored in the core and RCS. If required, the RCS pressure will be limited by the primary safety valves (PSVs), such that RCS pressure will remain within 110% of design pressure.

Attachment

Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

3.3 Analysis of Effects and Consequences

A. Mathematical Model

The nuclear steam supply system (NSSS) response to the IOSBCSV section of the event was simulated using the CESEC-III computer program described in UFSAR section 15.0.3. The HERMITE 1-D code discussed in UFSAR section 15D.2.4 was used to simulate the behavior of the core from the time of minimum DNBR in the CESEC code to the time of minimum DNBR recovered above the SAFDL conditions during the second phase of the IOSBCSV + LOAC event. The time-dependent thermal margin on DNBR in the reactor core was calculated using the CETOP-D computer program during both phases of the simulation which uses the CE-1 critical heat flux correlation described in Chapter 4 of the UFSAR.

B. Input Parameters and Initial Conditions

Table 3-1 lists the assumptions and initial conditions used for this event in addition to those discussed in UFSAR Section 15.0. Conditions were chosen such that, for an event initiated at 102% of rated core power, the overpower condition caused by the increase in steam flow results in the closest approach to the specified acceptable fuel design limits (SAFDL). Note these were the same set of initial conditions reported in the JCO (Reference (1)) for the IOSBCSV event.

For the HERMITE portion of the run, in addition to using the initial conditions obtained in the CESEC run at the time of minimum DNBR (such as core power, core flow, system pressure, etc.), additional conservatism was introduced by setting two key parameters to result in more adverse DNBR conditions as the simulation evolves. The first parameter, Moderator Temperature Coefficient (MTC), is set at $-1.0*10^{-4}$ delta rho/°F. Although a $-3.5*10^{-4}$ delta rho/°F was used for the CESEC run, a parametric study has shown that use of $-1*10^{-4}$ delta rho/°F produces the most adverse conditions for the combined excess load + LOF event. A much less adverse set of initial conditions would have been obtained if the $-1*10^{-4}$ delta rho/°F instead of $-3.5*10^{-4}$ delta rho/°F had been used for the excess load portion of the event.

The second parameter that has been reset is the Axial Shape Index (ASI). An ASI of -0.33 was used for the previous IOSCBSV event, since this is the most adverse ASI for the excess load event. The ASI was redefined to + 0.2995 for the HERMITE run, since this is the most adverse ASI for a loss of flow event. While this is an artificial change in the ASI parameter, it will nevertheless result in a conservative analysis.

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Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

C. Results

The dynamic behavior of key NSSS parameters during the IOSBCSV presented in Figures 1 through 5 of Reference (1) are generally applicable to this case up until the time loss of flow occurs. Figure 3-1 presents the DNBR plot versus time for the IOSBCSV + LOAC event. Table 3-2 summarizes the major events, times and results for this transient.

The IOSBCSV increases the rate of heat removal by the steam generators, causing a rapid cooldown of the RCS. Due to the negative moderator reactivity coefficient, core power increases from 102% of rated core power to 118% at 6.41 seconds, at which time a reactor trip signal occurs on CPC variable overpower. At 7.3 seconds, the control rods begin inserting into the core. Power increases to a maximum of 126.6% at 8.45 seconds, before dropping in response to the reactor trip, as shown in Figure 1 of Reference (1). At 8.95 seconds, the DNBR reaches a value of 1.27. The HERMITE results show that the DNBR continues to decrease to 1.14 at 10.75 seconds (the minimum DNBR starts at 1.2516 in the HERMITE run instead of 1.27, which is again conservative) before climbing back to 1.35 at 11.95 seconds.

Fuel failure was calculated to be at most 7.0311%. The level of failure was smaller than the 8.0% fuel failure reported for PVNGS-1 Cycle 3 (Reference (2)), and PVNGS-2 Cycle 3 (Reference (3)), as well as the 12.0% fuel failure reported for PVNGS-3 Cycle 2 (Reference (4)).

Conclusions

The fuel failure for this infrequent event is bounded by the fuel failure reported earlier for the three units for the IOSGADV + LOAC event.

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

TABLE 3-1
ASSUMPTIONS AND INITIAL CONDITIONS FOR FULL POWER INADVERTENT OPENING OF THE STEAM BYPASS CONTROL SYSTEM VALVES (8)

Parameter	Value
Core Power Level, MWt[1]	3876
Core Inlet Coolant Temperature, °F	570
Core Mass Flow rate, 10^6 lbm/hr	147.2
Pressurizer Pressure, psia	2200
Pressurizer Water Volume, ft^3	918
Steam Generator Pressure, psia	1070
Steam Generator Inventory, lbms per SG	174,000
CEA Worth on Trip, 10^{-2} delta rho	-6.5
Core Burnup	End of Cycle
ASI	-.32[2]
Max. Radial Peaking Factor	1.56
MTC, 10^{-4} delta rho/°F	-3.5[3]
FTC	Beginning of Cycle (min.)
Gap Conductance, 10^6 Btu/ft^2-hr-°F	1.814

1 The time-dependent thermal margin for calculating DNBR was initiated at the power operating limit, with a required overpower margin of 115%.

2 The ASI was changed to 0.2995 for the HERMITE section of the run.

3 The MTC was changed to $-1.0 * 10^{-4}$ delta rho /° F for the HERMITE run.

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

TABLE 3-2
SEQUENCE OF EVENTS FOR FULL POWER
INADVERTENT OPENING OF THE STEAM BYPASS CONTROL SYSTEM
VALVES (8)PLUS LOSS OF AC (LOAC)

Time	Event	Setpoint or Value
0.0	SBCS Valves begin to open	--
0.1	SBCS valves fully open	--
6.41	Reactor trips, (VOPT)	118%
7.3	CEA's drop into core	--
8.45	Power peak occurs, % full power	126.6
8.95	Minimum DNBR in CESEC run	1.27
10.75	Minimum DNBR in HERMITE run	1.14

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

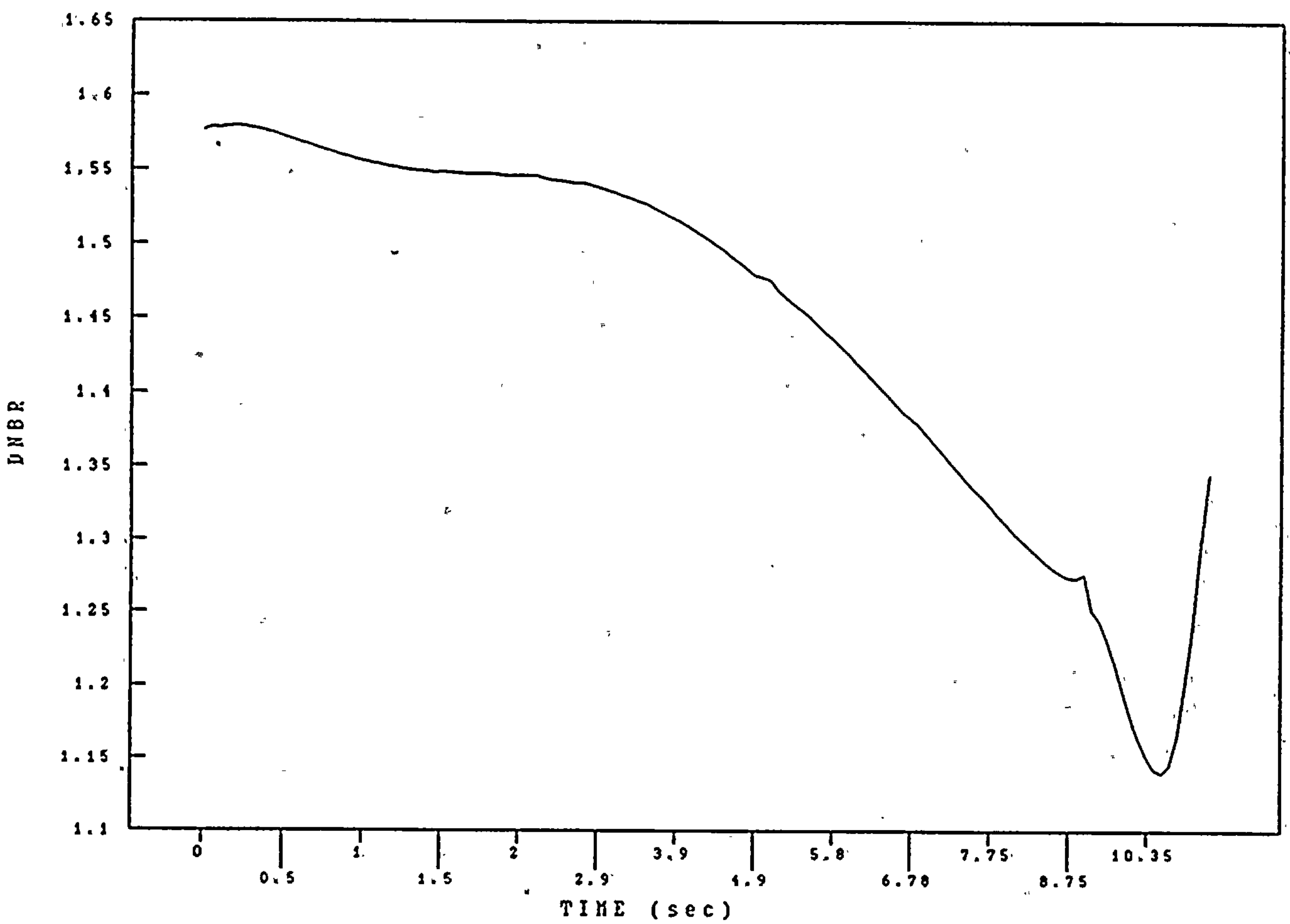

Figure 3-1: IOSGSBV + LOAC, DNBR vs. Time

Attachment
Analyses Performed for Justification for Continued Operation With Potential for a Single Failure Causing the Opening of All SBCS Valves

REFERENCES

1. 161-03569-WFC/MEP/RAB, November 1, 1990, letter to NRC, "PVNGS Units 1, 2 and 3, Justification for Continued Operation With Potential for a Single Failure Causing the Openinig of All SBCS Valves."

2. Reload Analysis Report, Unit 1, Cycle 3

3. Reload Analysis Report, Unit 2, Cycle 3.

4. Reload Analysis Report Unit 3, Cycle 2.

Attachment
Impact of Opening All Eight Steam Bypass Control System Valves
During a Reactor Power Cutback Event

EVENT 4: REACTOR POWER CUTBACK WITH FAILURE OF THE STEAM BYPASS CONTROL SYSTEM

4.1 Identification of Event and Causes

The Reactor Power Cutback (RPC) System (Reference (1)) is designed to reduce the power imbalance between the primary and secondary sides during a Loss of Load Event or a Loss of One Feedwater Pump Event. The RPC System reduces the primary system power to an acceptable level by dropping selected CEA subgroups. The remaining feedwater pump will automatically increase in speed and flow rate in response to decreasing steam generator level. The turbine will runback to about 60% power and remain at this level. Any temporary mismatch in primary, secondary power will be accommodated by the actuation of the Steam Bypass Control System.

Unless all CEAs in the selected bank(s) are released simultaneously, the CPCs would normally interpret the inserted CEAs as dropped rods and generate a reactor trip. To avert this action during a reactor power cutback, the CEA information provided by the CEACs and CEA deviation penalties are overridden for a short period of time.

A postulated failure of the turbine runback upon actuation of the RPCS has been the bounding analysis for determination of this time. The failure of the runback would result in bank 5, or banks 4 and 5 to be fully inserted into the core with the secondary system remaining at the initial power level. The RCS would cooldown due to the power to load mismatch. In the presence of a negative MTC, the cooldown would begin to increase power back to the original value. Additionally, the radial peaking within the core would increase due to CEA insertion and could result in violation of SAFDL. To prevent the SAFDL violation, an analysis for the above scenario determines the amount of time the CPCs may override the CEAC and penalty factor information without violating the SAFDL. This 'override time' determines the values of the CPC addressable constant, TCBSP and the CEAC addressable constant, TCBP, the so called reactor power cutback timer limits. Currently, these timer limits are set at 20 seconds for Units 1 and 2 and 25 seconds for Unit 3. Until the TCBSP/TCBP timer limits are reached after a RPC signal is received, the CPC DNBR trip is effectively disabled.

Since a demand on the steam bypass system is one of the expected responses to a RPC, a postulated failure in the SBCS was examined which replaced the failure of the turbine

Attachment
Impact of Opening All Eight Steam Bypass Control System Valves During a Reactor Power Cutback Event

to runback with the opening of all 8 turbine bypass valves as the most limiting single failure. The RPC configures the core to be at 60-68% power based upon reactivity considerations. The failure of the SBCS results in a secondary system power demand totaling roughly 150-155% of rated power rather than the 100% for the case of the failure of the turbine runback. The core responds to this increased excess load by approaching a higher power more rapidly than the design basis scenario. This scenario was examined to determine initial thermal margin requirements and acceptable values of the CPCs/CEACs RPCs timer limits during the interim until further decisions are made regarding the integrity of the SBCS design basis.

For this event, the major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a fuel design limit has been violated and thus whether fuel cladding degradation might be anticipated. Those factors which cause a decrease in local DNBR are:

a. increasing coolant temperature
b. decreasing coolant pressure
c. increasing local heat flux (core average heat flux combined with radial and axial power distribution effects)
d. decreasing coolant flow

4.2 <u>Sequence of Events and Systems Operation</u>

The sequence of events for this RPC + IOSBCV event is as follows: The initiating event is a Loss of Single Feedwater Pump from full power conditions. This results in a RPC followed by turbine runback to 60% power. The RPC signal also causes the insertion of CEA banks 5 and 4. An SBCS actuation signal is generated once the steam generator pressure exceeds 1115 psia. This actuation occurs shortly before the turbine demand is reduced to about 60-68%. At the time of SBCS actuation, all eight SBCVs are assumed to fail open, resulting in the secondary system experiencing an 88% excess load. Hence, the final steam demand is 155.4%.

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Impact of Opening All Eight Steam Bypass Control System Valves During a Reactor Power Cutback Event

4.3 Analysis of Effects and Consequences

A. Mathematical Model

The nuclear steam supply steam (NSSS) response to the RPC with IOSBCV event was simulated using the CESEC-III computer program described in FSAR section 15.0.3. The time-dependent thermal margin on DNBR in the reactor core was calculated using the CETOP-D computer program which uses the CE-1 critical heat flux correlation described in UFSAR Chapter 4.

B. Input Parameters and Initial Conditions

Table 4-1 lists the assumptions and initial conditions used for this event in addition to those discussed in FSAR Section 15.0. Conditions were chosen such that, for an event initiated at 102% of rated core power, operating at a power operating limit, the overpower condition caused by the RPC with IOSBCV event results in the largest degradation in the thermal margin.

C. Results

The dynamic behavior of key parameters following the RPC + IOSBCV is presented in Figure 4-1 through 4-5. Note that the Figure 4-2 for DNBR vs. time starts at 5.0 seconds instead of 0.0 seconds. During the first 5.0 seconds the DNBR will remain above SAFDL since power is decreasing. This was discussed in Reference (1). Table 4-2 summarizes the major events, times and results for this transient.

A normal RPC event due to Loss of a Feedwater Pump is first simulated using CESEC to obtain the time of the steam bypass control system actuation signal. The time of SBCS actuation is taken as the time at which the steam generator pressure in either of the secondary loops first exceed 1115 psia, the setpoint pressure for the SBCS actuation. The RPC + IOSBCV event is then simulated by repeating the RPC CESEC case, only this time an 88% excess load is introduced into the secondary system demand at the time of the SBCS actuation signal. CETOP is then executed using the data of CESEC output to obtain the DNBR vs.

time. The time at which DNBR drops below the limit of 1.24 is noted.

Cases were run to confirm that the current timer limits for the three units are adequate to avoid SAFDL violation with the 5% margin that has been set aside per Reference (2). Thus, the timer limits of 20 seconds for Units 1 and 2 and 25 seconds for Unit 3 need not be changed if this additional COLSS penalty of 5% is retained.

The current values for TCBSP and TCBP of 20 seconds for Unit 1 Cycle 3, and Unit 2, Cycle 3 and 25 seconds for Unit 3, Cycle 2 had been determined based on the methodology given in Reference (1). The Unit 2, Cycle 3 case presented in Figure 4-2 for DNBR shows that DNBR SAFDL of 1.24 will be violated at 20.15 seconds. A conservative value for TCBSP and TCBP is 16.0 seconds. This value would ensure SAFDLs would not be violated with a ROPM of 1.15.

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Impact of Opening All Eight Steam Bypass Control System Valves During a Reactor Power Cutback Event

TABLE 4-1
ASSUMPTIONS AND INITIAL CONDITIONS FOR RPC AT 100% POWER WITH INADVERTENT OPENING OF THE STEAM BYPASS CONTROL SYSTEM VALVES

Parameter	Value
Core Power Level, Mwt	3893
Core Inlet Coolant Temp.,°F	565
Core Mass Flow rate, 10^6 lbm/hr	147.2
Pressurizer Pressure, psia	2325
Steam Generator Pressure, psia	1070
Worth of Reactor Power Cutback Subgroups, 10^{-2} delta rho	-0.44
Core Burnup	End of Cycle
ASI	-0.299
MTC, 10^{-4} delta rho/°F	-3.5
FTC	End of Cycle (min.)
Minimum Gap Conductance, 10^6 Btu/ft^2-hr-°F	0.15
Initially Preserved Power Operating Limit, (% Full Power)	115.0

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Impact of Opening All Eight Steam Bypass Control System Valves During a Reactor Power Cutback Event

TABLE 4-2
SEQUENCE OF EVENTS RPC WITH
INADVERTENT OPENING OF ALL EIGHT STEAM BYPASS CONTROL SYSTEM VALVES

Time	Event	Setpoint or Value
0.0	Reactor Power Cutback signal generated due to a loss of a single feedwater pump - Steam generator feedwater flow rate reduction begins. - Turbine begins runback to 60% power. - Power removed to holding coils for RPC CEA Banks (Banks 5+4).	--
2.0	Steam Generator feedwater flow rate reaches its minimum value (fraction of nominal flow)	0.80
4.12	SBCS actuation pressure is reached (psia). - All SBCVs fail fully open - Minimum turbine demand level (% of full power)	1115.0 67.5
6.12	Secondary system demand due to SBCVs failure (%)	155.4
20.15	DNBR SAFDL reached.	1.24
26.85	Low Steam Generator pressure trip condition reached (psia) - MSIS actuates. - MSIVs begin closing.	820.0
28.00	CEAs begin dropping.	--
31.44	MSIVs fully closed.	--
32.00	CEAs fully inserted	--

Attachment
Impact of Opening All Eight Steam Bypass Control System Valves
During a Reactor Power Cutback Event

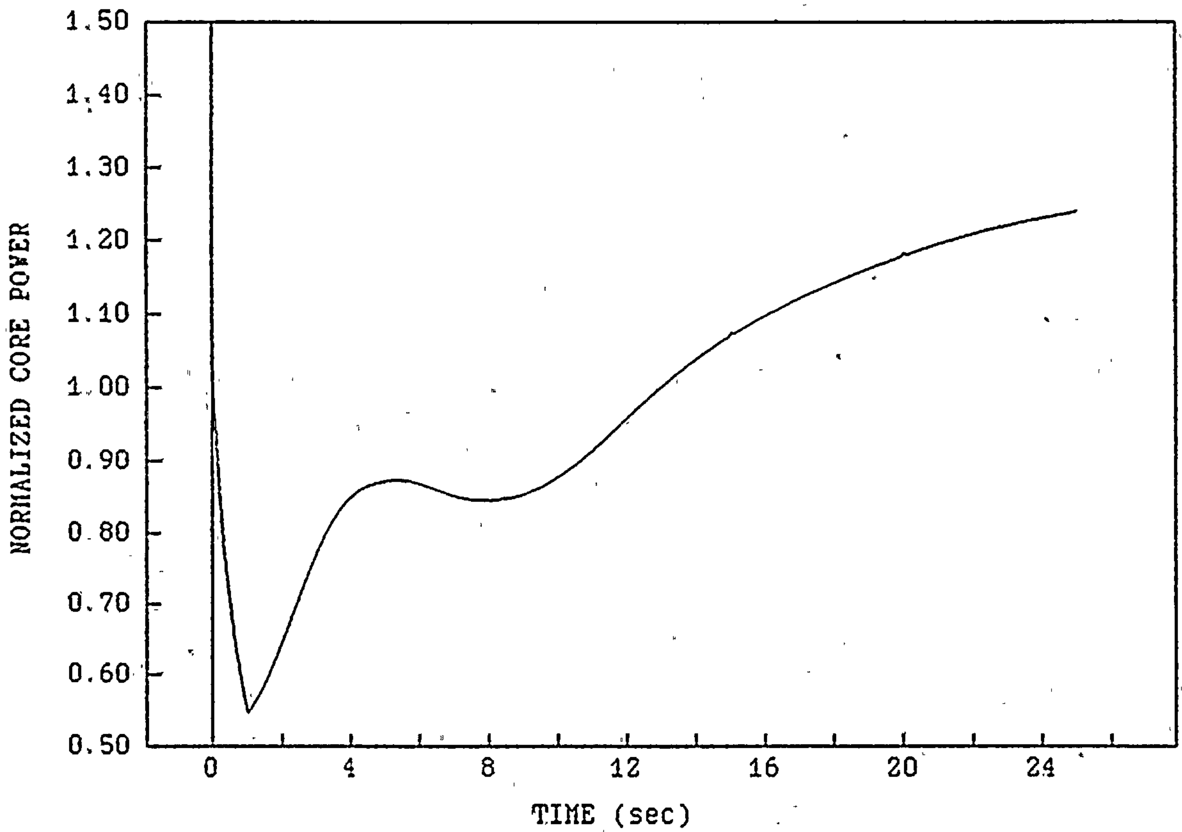

Figure 4-1: RPC plus IOSBCSVs, Normalized Core Power

Attachment
Impact of Opening All Eight Steam Bypass Control System Valves During a Reactor Power Cutback Event

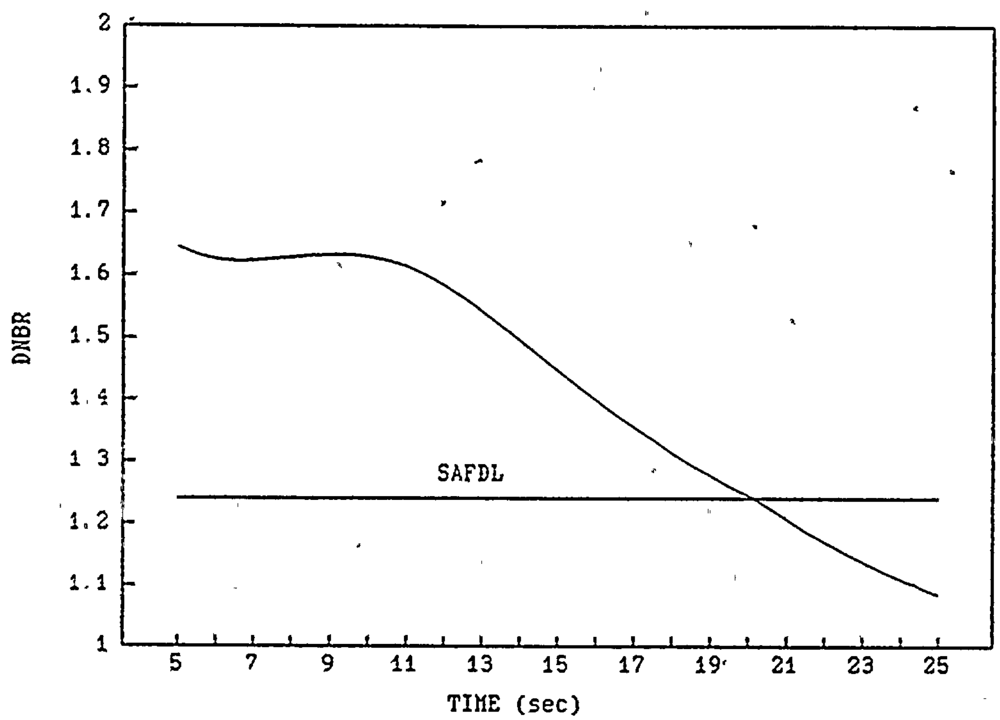

Figure 4-2: RPC plus IOSBCSVs, DNBR vs. Time[1]

[1] DNBR will remain above the SAFDL during the first 5 seconds of the transient since power is decreasing as discussed in Reference (1).

Attachment
Impact of Opening All Eight Steam Bypass Control System Valves
During a Reactor Power Cutback Event

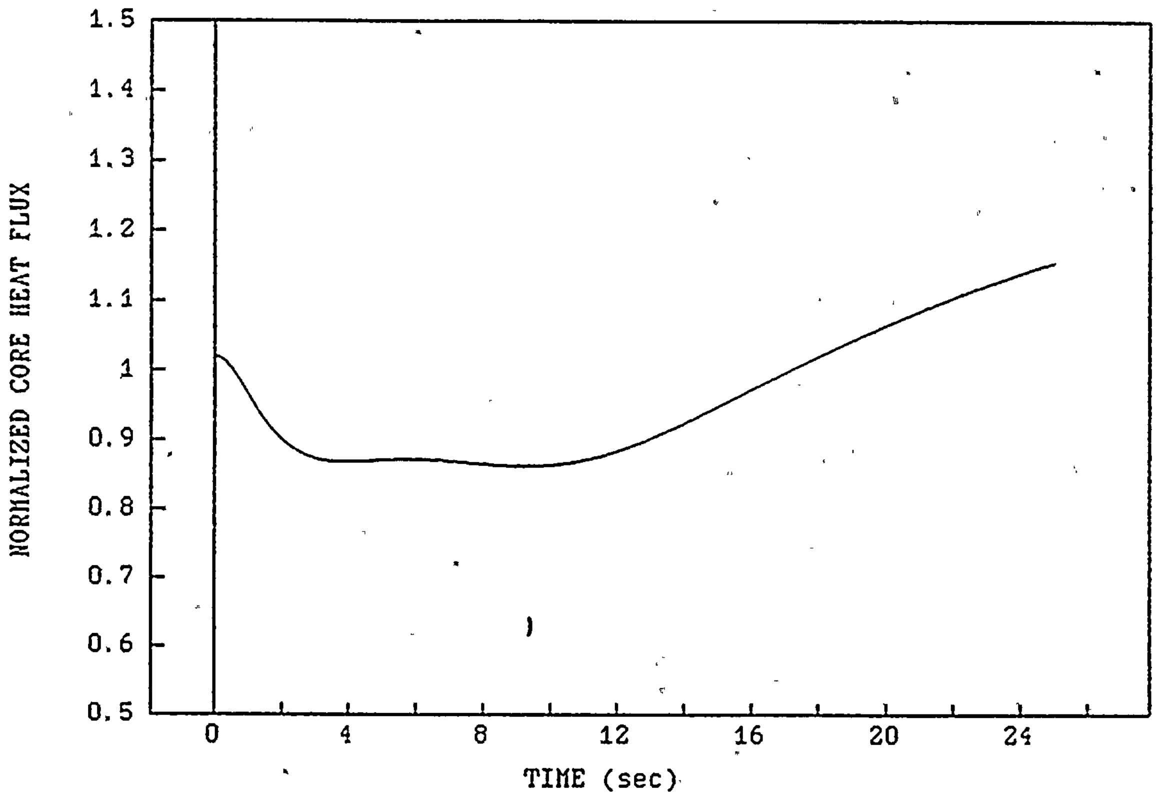

Figure 1-3: RPC plus IOSBCSVs, Normalized Core Heat Flux

Attachment
Impact of Opening All Eight Steam Bypass Control System Valves
During a Reactor Power Cutback Event

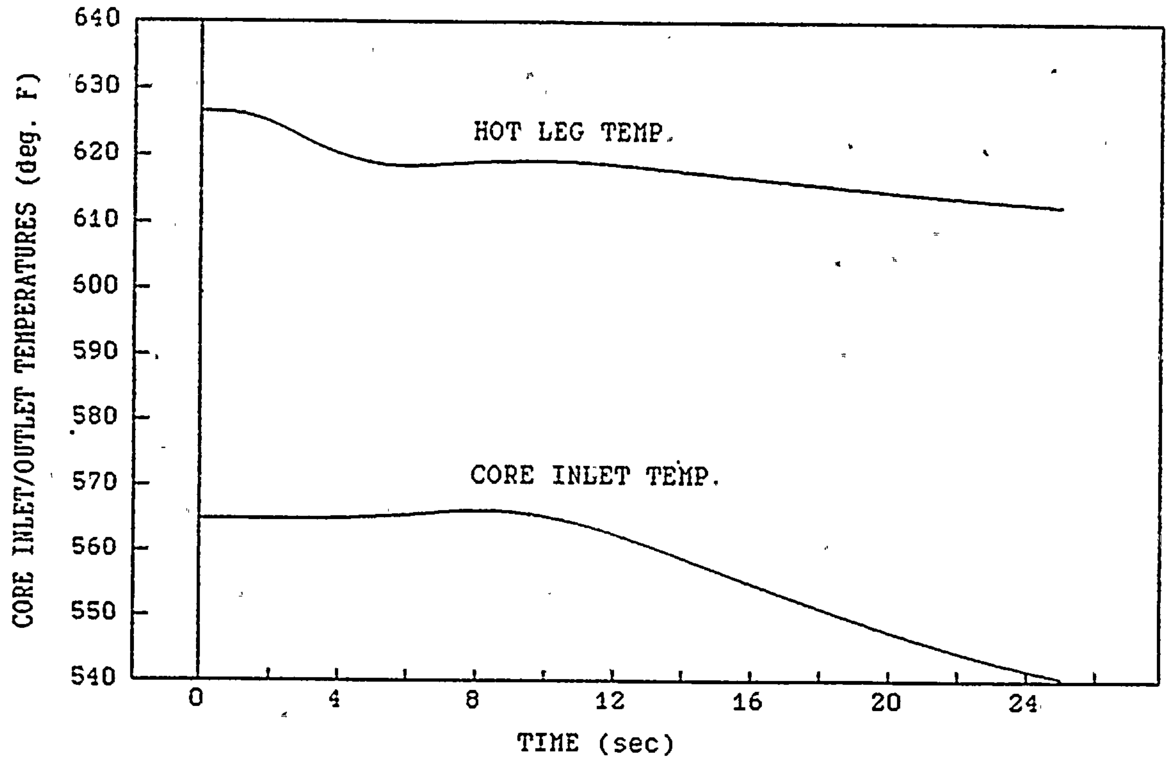

Figure 1-4: RPC plus IOSBCSVs, Hot/Cold Leg Temperatures

Attachment
Impact of Opening All Eight Steam Bypass Control System Valves During a Reactor Power Cutback Event

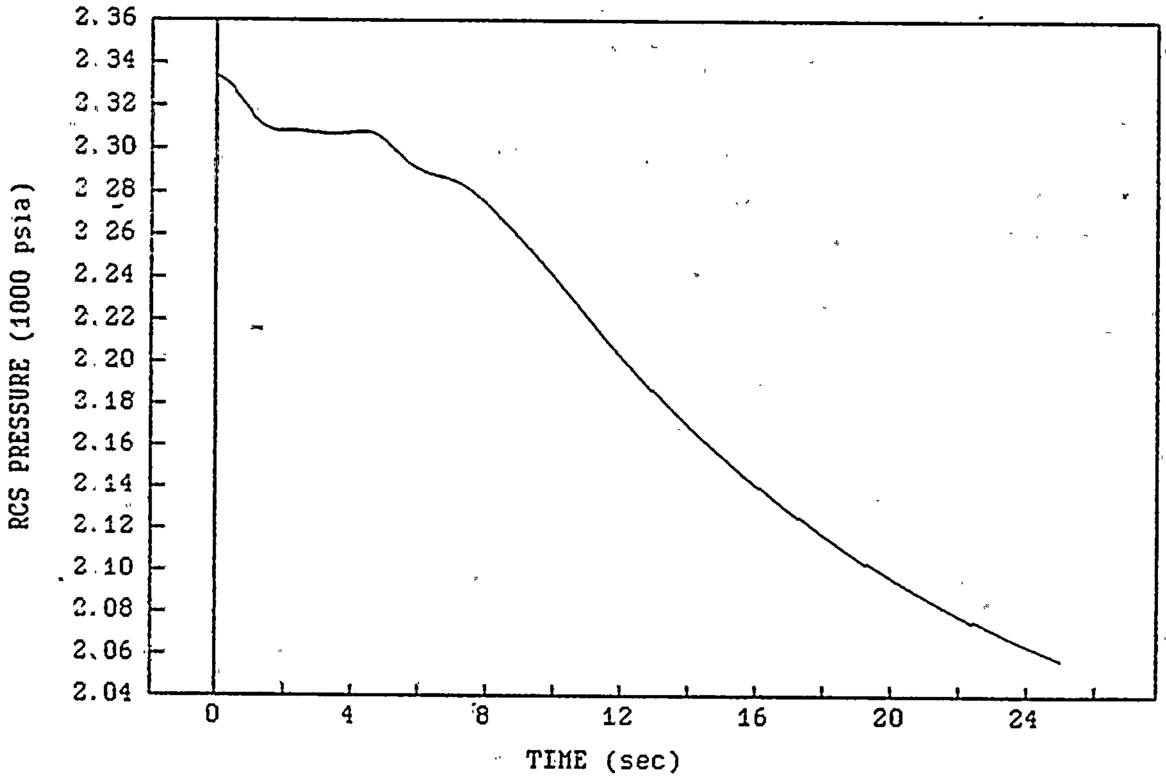

Figure 1-5: RPC plus IOSBCSVs, RCS Pressure

Attachment
Impact of Opening All Eight Steam Bypass Control System Valves During a Reactor Power Cutback Event

REFERENCES

1. "Safety Evaluation of the Reactor Power Cutback System," Enclosure 3-P to LD-82-039, March, 1982.

2. 161-03569-WFC/MEP/RAB, November 1, 1990, letter to NRC, "PVNGS Units 1, 2 and 3, Justification for Continued Operation with Potential for a Single Failure Causing the Opening of All SBCS Valves."