ABB

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February 9, 1993
LD-93-016

Docket No. 52-002

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

SUBJECT: Revisions to CESSAR-DC Chapter 15 Safety Analyses Results

Dear Sirs:

This letter submits a summary of several of the CESSAR-DC safety analyses reanalyzed to reflect resolution of DSER open items. A summary of the remaining reanalyses will be submitted by the end of February, 1993. All of these reanalyses will be incorporated into CESSAR-DC by the end of March, 1993.

The attachment describes the analysis assumptions and results for the following events:

1. Inadvertent Opening of an ADV (15.1.4)
2. Main Steam Line Breaks (15.1.5)
3. Main Feedwater Line Breaks (15.2.8)
4. Locked Rotor (15.3.3)
5. Inadvertent Deboration (15.4.6)
6. PLCS Malfunction (15.5.2)
7. Natural Circulation Cooldown (Appendix 5D)

The results conform to the applicable acceptance criteria in NUREG-0800 (Standard Review Plan) and are valid for a nominal full core power level of 3914 MWt. Several additional minor input data changes were also incorporated to represent the final CESSAR-DC plant design.

The offsite doses mentioned in the attachment were derived conservatively using TID-14844 based source terms. When the final reanalyses are incorporated in CESSAR-DC, revised source terms consistent with NUREG-1465 will be employed which will demonstrate additional margin regarding offsite doses.

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Should you have any questions on the enclosed material, please contact me or Mr. S. E. Ritterbusch of my staff at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

S. E. Ritterbusch for

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Acting Director
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T. Wambach (NRC)
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Enclosure I to
LD-93-016

REVISIONS TO CESSAR-DC CHAPTER 15

SAFETY ANALYSES RESULTS

SYSTEM 80+ DESIGN AT 3914 MWt

RESULTS FOR CESSAR-DC

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

A re-analysis of the inadvertent opening of a steam generator relief or safety valve event has demonstrated that the conclusions presented in Section 15.1.4.4 of CESSAR-DC for this event remain unchanged for the System 80+ power upgrade. The inadvertent opening of a steam generator atmospheric dump valve with loss of offsite power concurrent with turbine trip results in a transient minimum DNBR greater than the SAFDL and consequently no fuel pins are in DNB. The attached figure presents the transient minimum DNBR for this event as calculated by the CETOP code. The changes in the results and initial conditions due to the System 80+ power upgrade are shown in the attached mark-ups of Tables 15.1.4-1 and 3 of CESSAR-DC.

15.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT

A re-analysis of steam system piping failures inside and outside containment has demonstrated that:

a. the offsite doses are well within 10 CFR 100 guidelines and

b. potential fuel failure is sufficiently limited to ensure that the core will remain in place and intact with no loss of core cooling capability.

The conclusions which are presented in Section 15.1.5.4 of CESSAR-DC for these events, remain unchanged for the System 80+ power upgrade. The most limiting of these events have been shown to be

a. For cases chosen to maximize the potential for a post-trip return to power: a large steam line break (SLB) inside containment during full power operation with offsite power available, in combination with a single failure (SF) of a main steam isolation valve (MSIV) on one of the lines of the intact steam generator to close following MSIS and failure of the turbine control system to close both the turbine admission and turbine control valves and a stuck CEA (SLBFP) and

b. For cases chosen to maximize potential for degradation in fuel performance and dose at the site Exclusion Area Boundary: a large SLB outside containment, but upstream of the MSIV, during full power operation with loss of offsite power (LOOP) concurrent with the turbine trip which occurs when the reactor is tripped, in combination with a single failure of a MSIV on one of the lines of the intact steam generator to close following an MSIS and a stuck CEA (SLBFPLOPD).

For case A. (SLBFP) the maximum post-trip core reactivity is -0.889 percent delta rho. There is no return to power, thus, the values of DNBR remain above those for which fuel damage would be indicated. The attached figure presents the transient reactivities for this event. The changes in the initial conditions and results due to the System 80+ power upgrade are shown in the attached mark-ups of Tables 12-1 and 2 (taken from the ABB-CENP response to Item 12 of NRC RAI 440.96).

For case B. (SLBFPLOPD) the transient minimum DNBR is greater than 1.15. This results in less than 0.5% of the fuel rods being predicted to be in DNB. The consequent two-hour thyroid inhalation dose at the EAB was conservatively calculated using the TID-14844 source term to be less than 140 rem, which is within 10 CFR 100 guidelines. The doses will be even lower when NUREG-1465 sources are used. The attached figure presents the transient minimum DNBR for this event. The changes in the results and initial conditions due to the System 80+ power upgrade are shown in the attached mark-ups of Tables 15.1.5-5 and 10 of CESSAR-DC.

TABLE 15.1.4-1

SEQUENCE OF EVENTS FOR FULL POWER INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE (IOSGADV) with LOOP at trip

Time (sec)	Event	Setpoint or Value
0.0	One atmospheric dump valve opens fully	--
1800	Hot channel DNBR	~~1.24~~ 1.32
1800	Operator initiates manual trip	--
1800.4	Manual reactor trip signal generated	--
1800.55	Reactor trip breakers open, turbine trip initiated, Loss of offsite power assumed to occur, RCPs begin to coast down	--
1800.75 ~~1800.9~~	Steam generator water level reaches emergency feedwater actuation analysis setpoint, %WR	19.9
1909 ~~1805~~	Void begins to form in RV upper head	--
1805 ~~1808~~	Main steam safety valves open, psia	1212
1844 ~~1817~~	Main steam safety valves close, psia	1151
1860.75 ~~1860.9~~	Emergency feedwater delivered to generator	--
2049.4 ~~2032.9~~	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint, psia	719
2055.7 ~~2039.2~~	MFIVs close completely	--
2055.7 ~~2039.2~~	MSIVs close completely	--
2157.0 ~~2064.7~~	Pressurizer pressure reaches low pressure ~~pressurizer~~ safety injection actuation analysis setpoint, psia	1555
2199.0 ~~2104.7~~	Safety injection pumps reach full speed	--
2280 ~~2744~~	Affected steam generator dries out	--
3000	Operator manually closes ADV	--
3600	Operator initiates plant cooldown	--

TABLE 15.1.4-3

ASSUMPTIONS AND INITIAL CONDITION FOR FULL POWER INADVERTENT OPENING OF AN ATMOSPHERIC DUMP VALVE (IOSGADV) with LOOP at trip

Parameter	Value	
Initial Core Power Level, MWt	~~3876~~	3992
Initial Core Inlet Coolant Temperature, °F	~~563~~	561.
Initial Core Mass Flow rate, 10^6 lbm/hr	~~150.6~~	157.3
Initial Pressurizer Pressure, psia	~~1905~~	2175
Initial Pressurizer Water Volume, ft^3	500	
Initial Steam Generator Pressure, psia	~~1078~~	1074
Initial Steam Generator Inventory, lbm per SG	133,632	
CEA Worth on Trip, 10^{-2} $\Delta\rho$	~~-8.8~~	-8.86
Core Burnup	End of Cycle	
ASI	-.3	
Maximum Radial Peaking Factor	~~1.4~~	1.53

TABLE 12-1

ASSUMPTIONS AND INITIAL CONDITIONS FOR REANALYSIS OF A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFP)

Parameter	Assumed Value	
Initial Core Power Level, MWt	~~3876~~	3992.
Initial Core Inlet Coolant Temperature, °F	~~563~~	561.
Initial Core Mass Flow Rate, 10^6 lbm/hr	~~151.89~~	157.87
Initial Pressurizer Pressure, psia	~~2400~~	2325.
Initial Pressurizer Water Volume, ft^3	1350	
CEA Worth for Trip, 10^{-2} $\Delta\rho$	~~-8.86~~	-10.0 *
Initial Steam Generator Liquid Inventory, lbm	~~259277.~~	270,000
One Main Steam Isolation Valve on Intact Steam Generator	Inoperative	
Core Burnup	End of Cycle	
Blowdown Fluid	Saturated Steam	
Blowdown Area for Each Steam Line, ft^2	1.283	
Turbine Trip on Reactor Trip	Inoperative	

* INDICATIVE OF LOW LEAKAGE FUEL WITH ERBIUM SHIMS.

TABLE 12-2

SEQUENCE OF EVENTS FOR REANALYSIS OF A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFP)

Time (Sec)	Event	Setpoint or Value
0.0	Steam Line Break Occurs	--
7.29 ~~7.31~~	CPC Variable Overpower Trip Condition Reached (% Power)	115
7.69 ~~7.71~~	CPC Variable Overpower Trip Signal Generated	--
7.84 ~~7.86~~	Reactor Trip Breakers Open	--
16.52 ~~16.48~~	Steam Generator Pressure Reaches Main Steam Isolation Signal Analysis Setpoint, psia	719
14.80 ~~19.65~~	Voids Begin to Form in RV Upper Head	--
22.82 ~~22.83~~	MFIVs ~~Begin to~~ Close Completly	--
22.82 ~~22.83~~	MSIVs Close Completely	--
40.7 ~~38.8~~	Pressurizer Empties	--
65.8 ~~73.08~~	Pressurizer Pressure Reaches Safety Injection Actuation Signal Analysis Setpoint, psia	1555
105.8 ~~113.6~~	Safety Injection Flow Begins	--
164.8 ~~147.96~~	Safety Injection Boron Begins to Reach Reactor Core	--
227.0 ~~185.04~~	Maximum Transient Reactivity, $10^{-2}\ \Delta\rho$	~~-0.35~~ -0.889
1800	Operator Initiates Cooldown	--

TABLE 15.1.5-5

SEQUENCE OF EVENTS FOR A STEAM LINE BREAK OUTSIDE CONTAINMENT DURING FULL POWER OPERATION WITH ~~OFFSITE POWER AVAILABLE~~ (SLBFP LOPD)

LOSS OF OFFSITE POWER

Time (Sec)	Event	Setpoint or Value
0.0	Steam Line Break Occurs	--
6.65 ~~5.76~~	CPC Variable Overpower Trip Condition Reached	115
6.65 ~~5.76~~	EFW Initiated to Both Steam Generators	
7.05 ~~6.16~~	CPC Variable Overpower Trip Signal Generated	--
7.20 ~~6.31~~	Reactor Trip Breakers Open Turbine Trip initiated, Loss of Offsite Power assumed to occur, RCPs begin to coast down	--
8.8 ~~8.87~~	Minimum Transient DNBR	~~1.18~~ 1.157
13.48 ~~13.38~~	Steam Generator Pressure Reaches Main Steam Isolation Signal Analysis Setpoint, psia	719
12.78 ~~17.07~~	Voids Begin to Form in RV Upper Head	--
19.78 ~~19.73~~	MFIVs Close Completely	--
19.78 ~~19.73~~	MSIVs Close Completely	--
42.7	Pressurizer Empties	--
177.9 ~~124.7~~	Maximum Post-trip Transient Reactivity, 10^{-2} $\Delta\rho$	~~-1.72~~ -1.80
297.15 ~~203.6~~	Pressurizer Pressure Reaches Safety Injection Actuation Signal (SIAS) Analysis Setpoint, psia	1555
337.15 ~~243.6~~	Safety Injection Flow Begins	--
448.65 ~~310~~	Safety Injection Boron Begins to reach Reactor Core	--
1800	Operator Initiates Cooldown	--

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TABLE 15.1.5-10

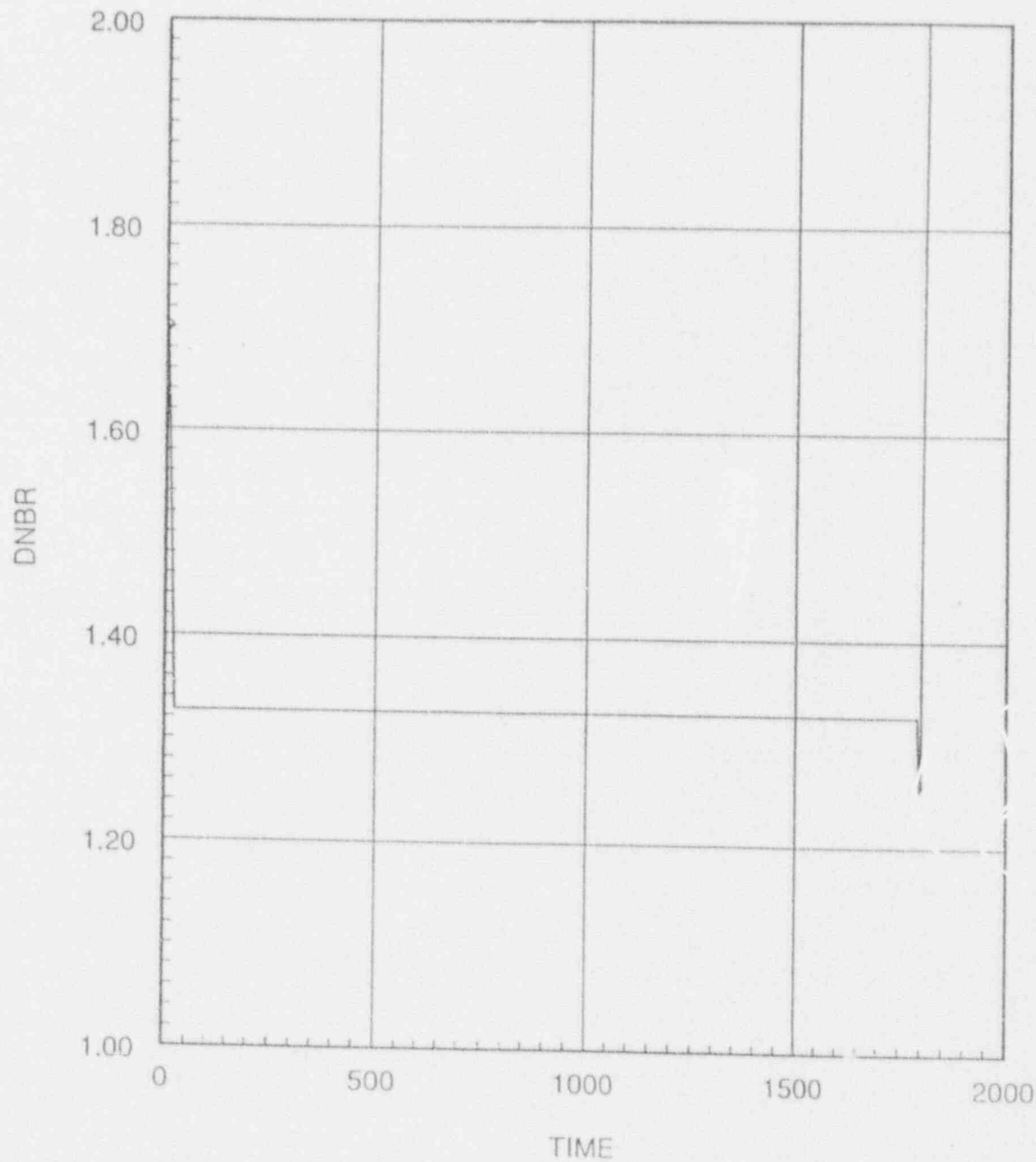
ASSUMPTIONS AND INITIAL CONDITIONS FOR THE STEAM LINE BREAK OUTSIDE CONTAINMENT DURING FULL POWER OPERATION WITH ~~OFFSITE POWER AVAILABLE~~ LOSS OF OFFSITE POWER (SLBFPLOPD)

Parameter	Assumed Value	
Initial Core Power Level, MWt	~~3876~~	3992
Initial Core Inlet Coolant Temperature, °F	~~563~~	561
Initial Core Mass Flow Rate, 10^6 lbm/hr	~~151.9~~	157.87
Initial Pressurizer Pressure, psia	2400	
Initial Pressurizer Water Volume, ft^3	1350.0	
Radial Peaking Factor, F_R	~~1.47~~	1.65
CEA Worth for Trip, 10^{-2} $\Delta\rho$	-8.86	
Initial Steam Generator Liquid Inventory, lbm	~~113205~~	108640
Core Burnup	End of Cycle	
Blowdown Fluid	Saturated Steam	
Blowdown Area for Each Steam Line, ft^2	1.283	

SYS80+ : IOSGADV + LOOP

DNBR

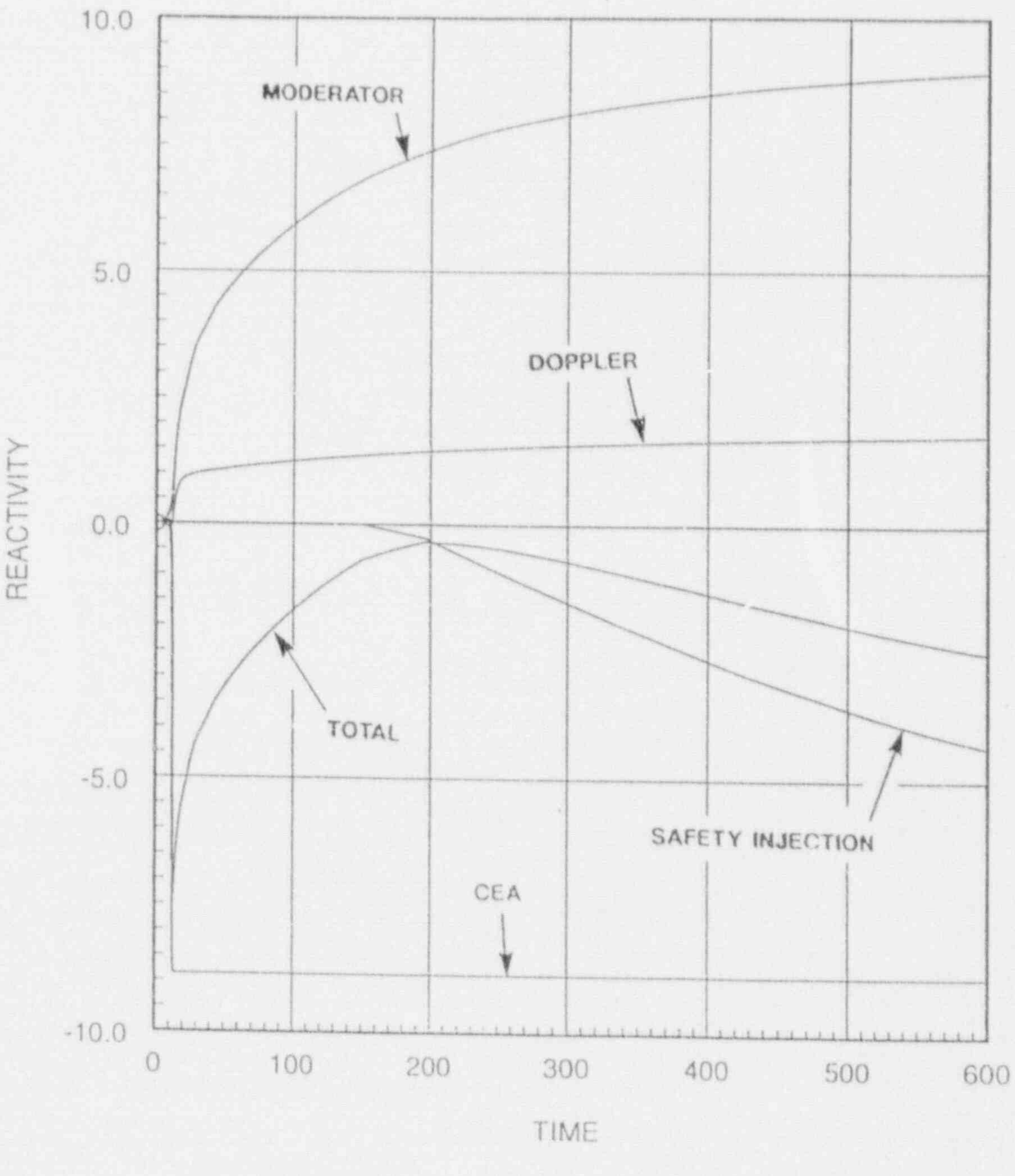
C89300M2 1993/02/01



SYS80+ : SLBFP post trip no LOOP

REACTIVITY

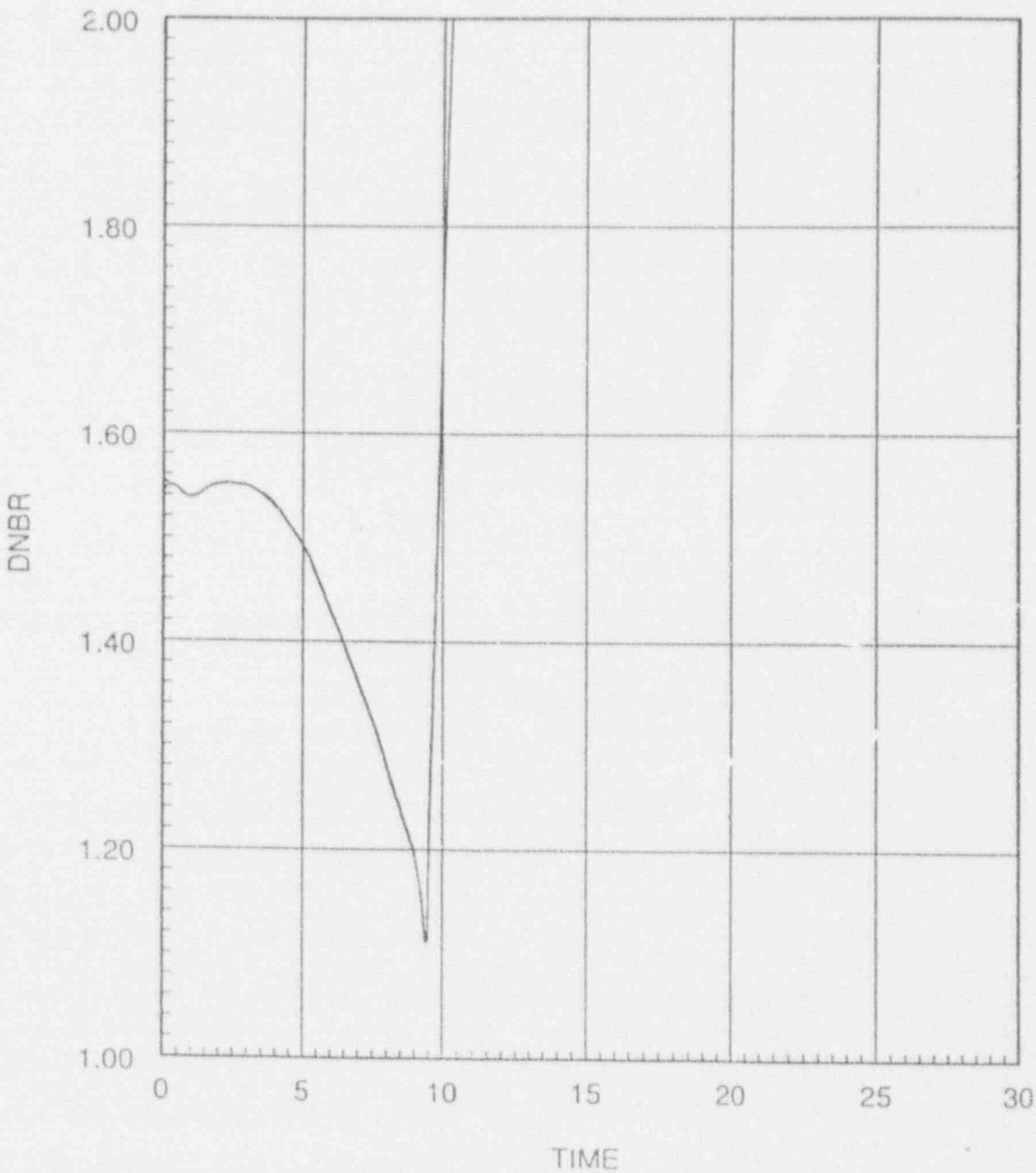
C89300M2 1993/02/01



SYS80+ : SLBFP pre trip with LOOP

DNBR

C89300M2 1993/02/01



SYSTEM 80+ DESIGN AT 3914 MWt

RESULTS FOR CESSAR-DC

15.2.8 FEEDWATER SYSTEM PIPE BREAKS

The feedwater line break analysis for System 80+ has been reanalyzed to demonstrate acceptable results with the following design changes:

1. A 3 percent increase in core thermal power.
2. An increase in the assumed maximum pressurizer safety valves lift setpoint to 2540 psia from 2525 psia.
3. An increase in maximum surge line length from 100 ft. to 125 ft.
4. A more realistic operating range for the pressurizer pressure of 2175 to 2325 psia.

The first three changes result in higher maximum RCS pressures than found in previous analyses. The revised pressurizer pressure operating range results in lower maximum RCS pressures for small breaks, therefore, the limiting break size increases from the previous analysis.

The maximum pressure never exceeded 2750 psia (110% design pressure) for small breaks. Small breaks are as defined in Reference 15.2.8-1 (6 inch break = 0.2 sq.ft). For large breaks the pressure never exceeded the 3000 psia (120% of design pressure) established in Section 15.2.8 of the Standard Review Plan for the probability of occurrence of breaks of this size and as previously documented in Reference 15.2.8-1. The minimum DNBR never fails below the SAFDL for this analysis resulting in no fuel failure.

A parametric study covering break size and initial pressurizer pressure was performed. Figure 15.2.8-1 shows the maximum RCS pressure results for break sizes of 0.5 to 0.9 square feet as a function of initial operating pressurizer pressure. As can be seen for any given break size, there is a peak pressure for some initial pressurizer pressure. This is the initial pressure at which the affected steam generator dry out coincides with the pressurizer high pressure trip. If the initial pressurizer pressure is lower than that where the maximum transient peak pressure occurs the affected steam generator will dry out prior to the pressurizer high pressure trip, resulting in a lower pressure at the time of reactor trip. Similarly, if the initial pressure is higher than that where the maximum peak pressure occurs, the pressurizer high pressure setpoint is reached prior to affected steam generator dry out. This results in a lower peak pressure during the accident due to residual water inventory in the ruptured steam generator providing greater heat removal after the trip, thus limiting the peak pressure. For all break sizes, when the initial pressurizer pressure is far from the initial pressure which results in the maximum transient peak pressure for that break size the peak pressure is approximately 2680 psia.

The maximum peak pressure achieved as a function of break size is shown in Figure 15.2.8-2. As can be seen for break sizes below 0.5 square feet, the peak pressure is well below 110% of the design as required for those breaks. The highest pressure reached is still within 120% of design pressure for large breaks.

Reference:

15.2.8-1 "Response to NRC Round One Question 440.42 on the CESSAR-FSAR," enclosure to letter LD-81-069, A. E. Scherer to D. G. Eisenhut, dated October 8, 1981.

TABLE 15.2.8-1

ASSUMPTIONS FOR THE LIMITING CASE FEEDWATER LINE BREAK EVENT

Parameter	Nominal Value	Assumed Value
Initial Core Power, MWt	3914 ~~3800~~	3992 ~~3876~~
Initial Core Inlet Temperature, °F	558	556
Initial Reactor Vessel Flow Rate, gpm	445600	418718 ~~422200~~
Initial Pressurizer Pressure, psia	2250	2250 ~~1971.3~~
Fuel Gas Gap Heat Transfer Coefficient, Btu/Hr-ft^2-°F	>375	540 ~~375~~
Doppler Coefficient Multiplier	1.0	1.0
Pressurizer Safety Valves Rated Flow Rate per Valve, lbm/hr	525000 ~~≥460000~~	525000 ~~460000~~
Initial Pressurizer Liquid Volume, ft^3	1200	1350
Initial Steam Generator Inventory, lbm	212185 ~~213600~~	212185 ~~213600~~
Initial Feedwater Enthalpy, Btu/lbm	430	365.5
Steam Bypass Control System	Automatic	Manual
Normal Onsite or Offsite Electrical Power After Turbine Trip	Available	Unavailable
Feedwater Pipe Break Area, ft^2	--	0.7 ~~0.3~~
CEA Worth at Trip, 10^{-2} $\Delta\rho$	-15.36	-8.86

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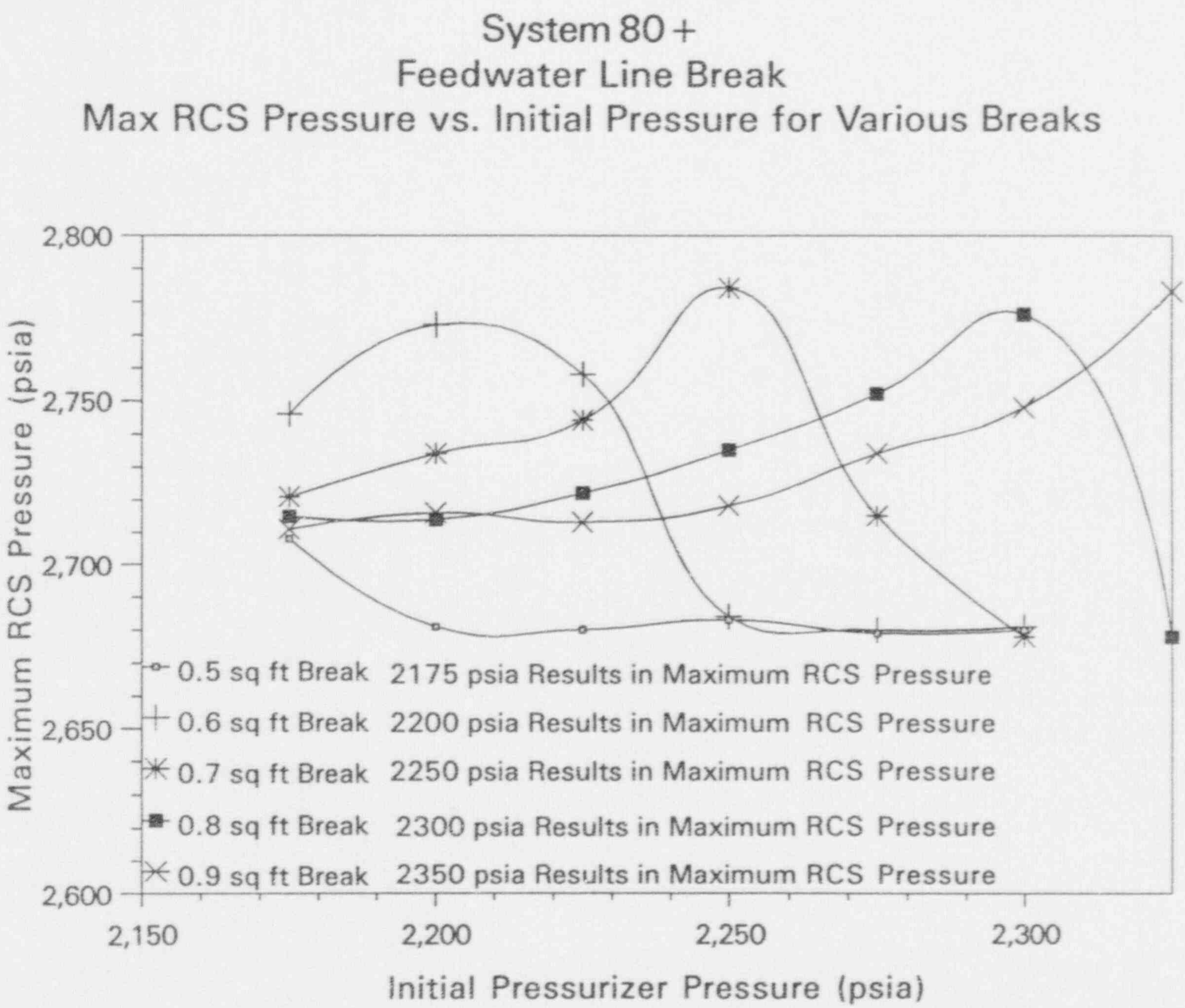


Figure 15.2.8.1

System 80+
Feedwater Line Break
Maximum Peak Pressure vs. Break Size

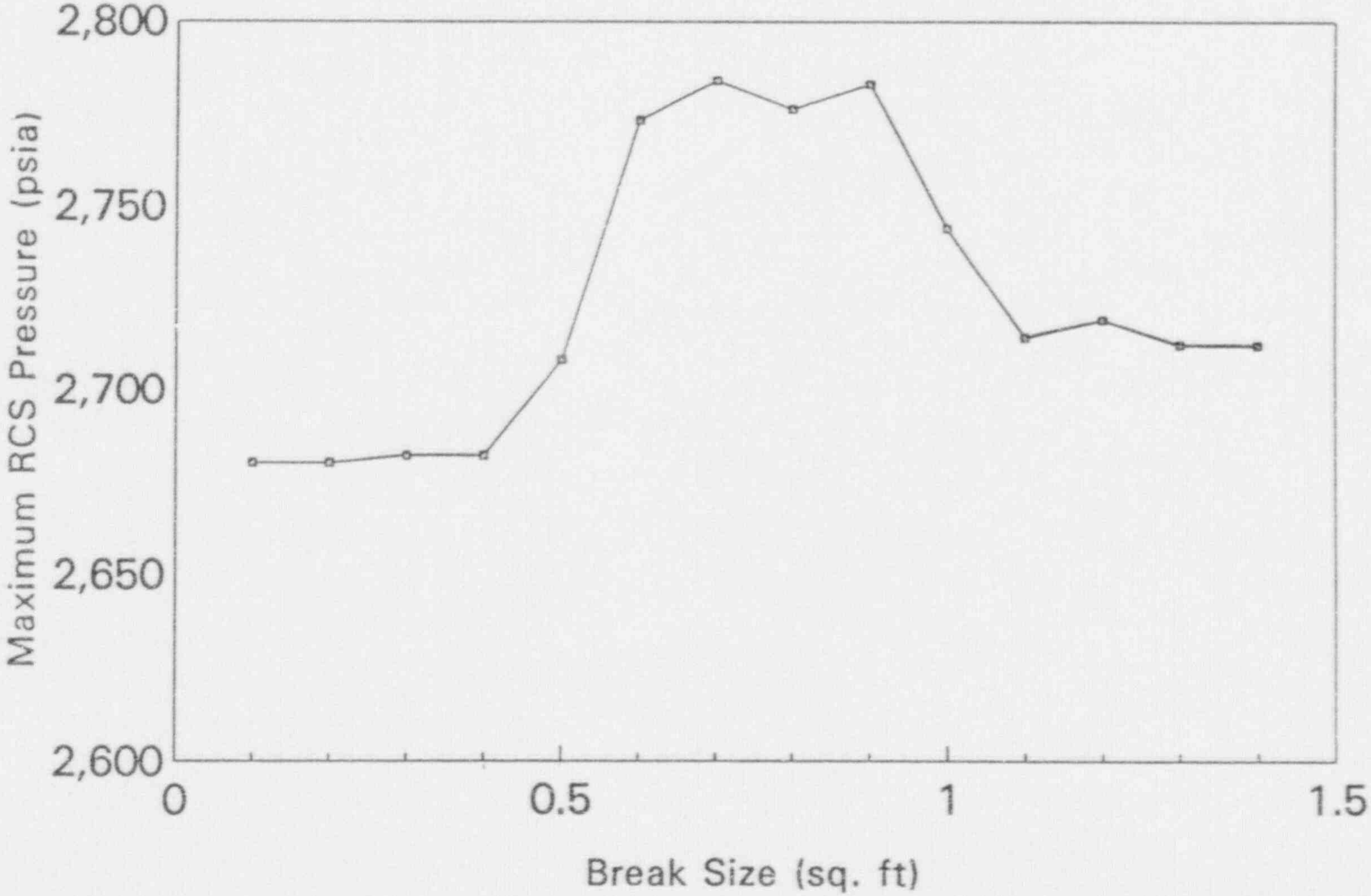


Figure 15.2.8.2

SYSTEM 80+ DESIGN AT 3914 MWt

RESULTS FOR CESSAR-DC

15.3.3 SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER

The single RCP rotor seizure analysis incorporates the design changes pertaining to the System 80+ power upgrade to 3914 MWt as well as the deletion of the 3 second time delay for loss of offsite power. The initial conditions and assumptions are presented in Table 15.3.3-2.

The minimum DNBR/fuel failure aspect of the analysis has been performed to demonstrate that the resultant offsite doses remain within 10CFR100 guidelines. The enclosed Figure 15.3.3-11 presents the minimum DNBR versus time for the most limiting initial conditions. It has been conservatively determined (based on the convolution method, Reference 1) that no more than 2.5 percent of the fuel should fail due to the RCP rotor seizure event with loss of offsite power coincident with the turbine trip. A brief summary of the results are presented below.

PRELIMINARY ANALYSIS RESULTS

Parameter	Value
Time of Turbine Trip, seconds	1.43
Time of Loss of Offsite Power, seconds	1.43
Minimum Transient DNBR	0.996
Time of Minimum DNBR, seconds	4.18
Percent of Failed Fuel	2.5*

(*) - the resultant doses are within the 10CFR100 guidelines

Reference:

1. "C-E Methods for Loss of Flow Analysis," CENPD-183, July 1976.

TABLE 15.3.3-2

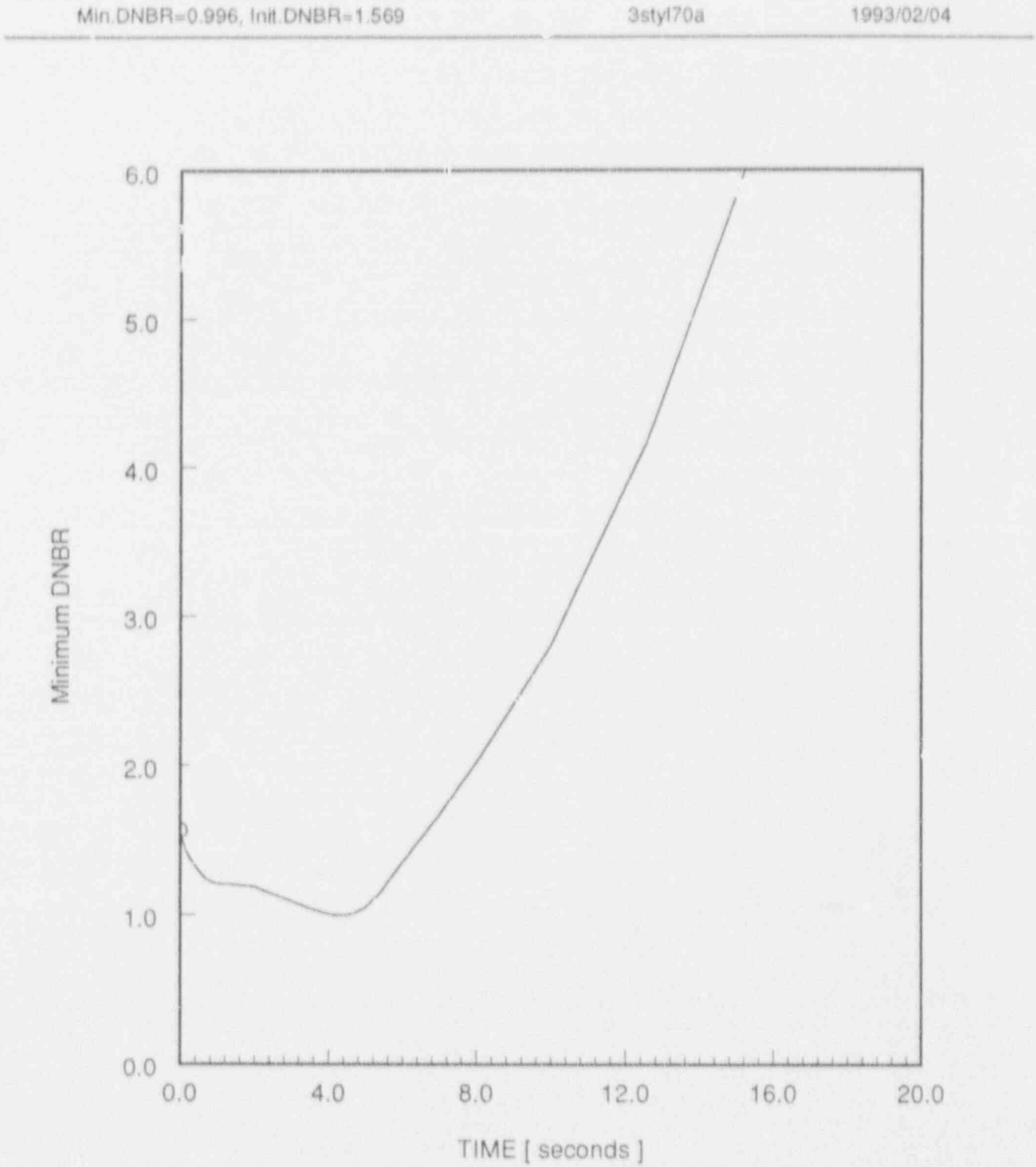
ASSUMPTIONS AND INITIAL CONDITIONS FOR THE ANALYSIS OF A SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER RESULTING FROM TURBINE TRIP

Parameter	Assumed Value	
Core Power Level, Mwt	3992 ~~3876~~	
Core Inlet Coolant Temperature, °F	561 ~~570~~	
Reactor Coolant System Pressure, psia	2188 ~~1911~~	
Pressurizer Pressure, psia	2175 ~~1900~~(a) 2325 ~~2400~~(b)	
Steam Generator Pressure, psia	1054 ~~1114~~	
Core Mass Flow, 10^6 lbm/hr	151.8 ~~149.0~~	
Maximum Radial Power Peaking Factor	1.43 ~~1.65~~	
Core Minimum DNBR	1.57	
Doppler Coefficient Multiplier	0.85	H
CEA Worth for Trip, 10^{-2} $\Delta\rho$ (most reactive CEA Stuck out)	-8.0(c)	
Moderator Temperature Coefficient, $\Delta\rho \times 10^{-4}/°F$	0.0	

NOTES:
a) Selected to minimize core thermal margin and maximize radiological release.
b) Selected to maximize RCS pressure.
c) Value is conservatively less negative than the available design value of -8.86 used in other Chapter 15 analyses.

FIG.15.3.3-11 SYSTEM 80+ SEIZED ROTOR LOOP 102%POWER

Minimum DNBR vs. Time



SYSTEM 80+ DESIGN AT 3914 MWt

RESULTS FOR CESSAR-DC

15.4.6 INADVERTENT DEBORATION

The analysis of the Inadvertent Deboration (ID) event initiated during each of the six operational modes defined in the Technical Specifications was re-evaluated to address the impact of increasing the core power from 3800 MWt to 3914 MWt. The critical parameters, selected to make the analysis conservative, are presented in the revised assumptions table (Table 15.4.6-1 of CESSAR-DC).

The results of the analysis demonstrate that the limiting case occurs during Mode 5, mid-loop conditions. The limiting case assumes that dilution occurs at the maximum possible rate and credits only the boron dilution alarm to mitigate the consequences of the event. The results of the analysis determined that 67 minutes elapses from the initiation of the event to the point at which the core becomes critical. This elapsed time period for the limiting ID event provided more than 30 minutes for the operator to terminate the deboration.

The operator is alerted to a decrease in the reactor coolant system (RCS) boron concentration either through a high neutron flux alarm on the startup flux channel, the reactor makeup water flow alarm (Mode 6 only), sampling, boronometer indications, or boric acid flow rate. He turns off the charging pump and closes the letdown control valves in order to halt further dilution. Next, he increases the RCS boron concentration by implementing the emergency boration procedure for achieving cold shutdown boron concentration.

TABLE 15.4.6-1

ASSUMPTIONS FOR THE INADVERTENT DEBORATION ANALYSIS

Parameter	Assumptions
Cold RCS Volume* (~~partially drained~~ mid-loop operation), ft^3	~~4,400~~ — 3,961
RCS Mass (~~partially drained~~ mid-loop operation), lbm	~~263,480~~ 237,185
Volumetric Charging Rate, gpm	~~180~~ — 150
Mass Charging Rate, lbm/sec	~~25~~ — 22.3
Dilution Time Constant, τ, sec^{-1}	~~10512~~ — 10650
Initial boron Concentration - C_0, ppm	~~1012~~ — 1193
Critical Boron Concentration - C_{crit}, ppm	814

* Includes the reactor vessel up to the mid-plane of the hot legs, half of a single hot leg, half of two cold discharge legs and a shutdown cooling system. RCS volume, and correspondingly RCS mass were reduced to reflect the current System 80+ plant design.

SYSTEM 80+ DESIGN AT 3914 MWt

RESULTS FOR CESSAR-DC

15.5.2 CVCS MALFUNCTION - PRESSURIZER LEVEL CONTROL SYSTEM MALFUNCTION WITH LOSS OF OFFSITE POWER

The event was analyzed in order to incorporate the following changes:

1. A 3% increase in core power

2. Reductio[illegible] he inlet coolant temperature from 558°F to 555.8°F.

3. Reduction in the maximum charging pump flow from 250 gpm to 150 gpm.

4. A zero time delay between turbine trip and loss of offsite power.

The event was analyzed assuming a failure in the pressurizer pressure and level control systems such that the pressurizer heaters failed to turn off and the charging flow was maximized and the letdown flow minimized. This resulted in the continuous filling of the pressurizer. The maximum RCS pressure was determined to be 2680 psia. The maximum steam generator pressure was determined to be 1267 psia. These values are less than 110% of the design pressures for the primary and secondary systems, respectively. The minimum DNBR was determined to remain above the SAFDL; thus, no fuel failures need be assumed. Table 15.5.2-1 presents the sequence of events and Table 15.5.2-2 lists the assumptions and initial conditions used for this analysis. Figures 15.5.2-1 through 15.5.2-3 present the dynamic response of the system to this event.

TABLE 15.5.2-1

SEQUENCE OF EVENTS FOR THE PLCS MALFUNCTION WITH A LOSS OF OFFSITE POWER ~~2~~ 0 SECONDS AFTER TURBINE TRIP

Time (Sec)	Event	Setpoint or Value
0.0	Charging Flow Maximized & Letdown Flow Minimized	--
~~635.6~~ 921.8	Pressurizer Pressure Reaches Reactor Trip Analysis Setpoint, psia	2475
~~636.6~~ 922.8	High Pressurizer Pressure Trip Signal Generated, Turbine Trip Occurs	--
~~636.75~~ 922.95	Trip Breakers Open	--
~~639.75~~ 922.95	Loss of Offsite Power	--
~~639.14~~ 924.9	Pressurizer Safety Valves open, psia	~~2525~~ 2540
~~639.14~~ 925.49	Maximum ~~Pressurizer~~ RCS Pressure, psia	~~2525~~ 2680
~~651.1~~ 931.7	Pressurizer Safety Valves Close, psia	~~2058~~ 2070
~~638.37~~ 928.7	Main Steam Safety Valves Open, psia	1212
~~655.03~~ 936.3	Maximum Steam Generator Pressure, psia	~~1247~~ 1267.0
1800.0	Operator Initiates Plant Cooldown	--

TABLE 15.5.2-2

ASSUMPTIONS AND INITIAL CONDITIONS FOR THE PLCS MALFUNCTION WITH A LOSS OF OFFSITE POWER ~~7~~ 10 SECONDS AFTER TURBINE TRIP

Parameter	Value
Initial Core Power Level, MWt	3992 ~~3876~~
Initial/ Core Inlet Coolant Temperature, °F	555.8 ~~550~~
Initial/ Core Mass Flow, 10^6 lbm/hr	152.3 ~~151.8~~
Initial/ Pressurizer Pressure, psia	1905
Initial Pressurizer Water Volume, ft^3	1400 ~~1464~~
CEA Worth on Trip, 10^{-2} $\Delta\rho$	-8.86

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FIGURE 15.5.2-1

RCS PRESS

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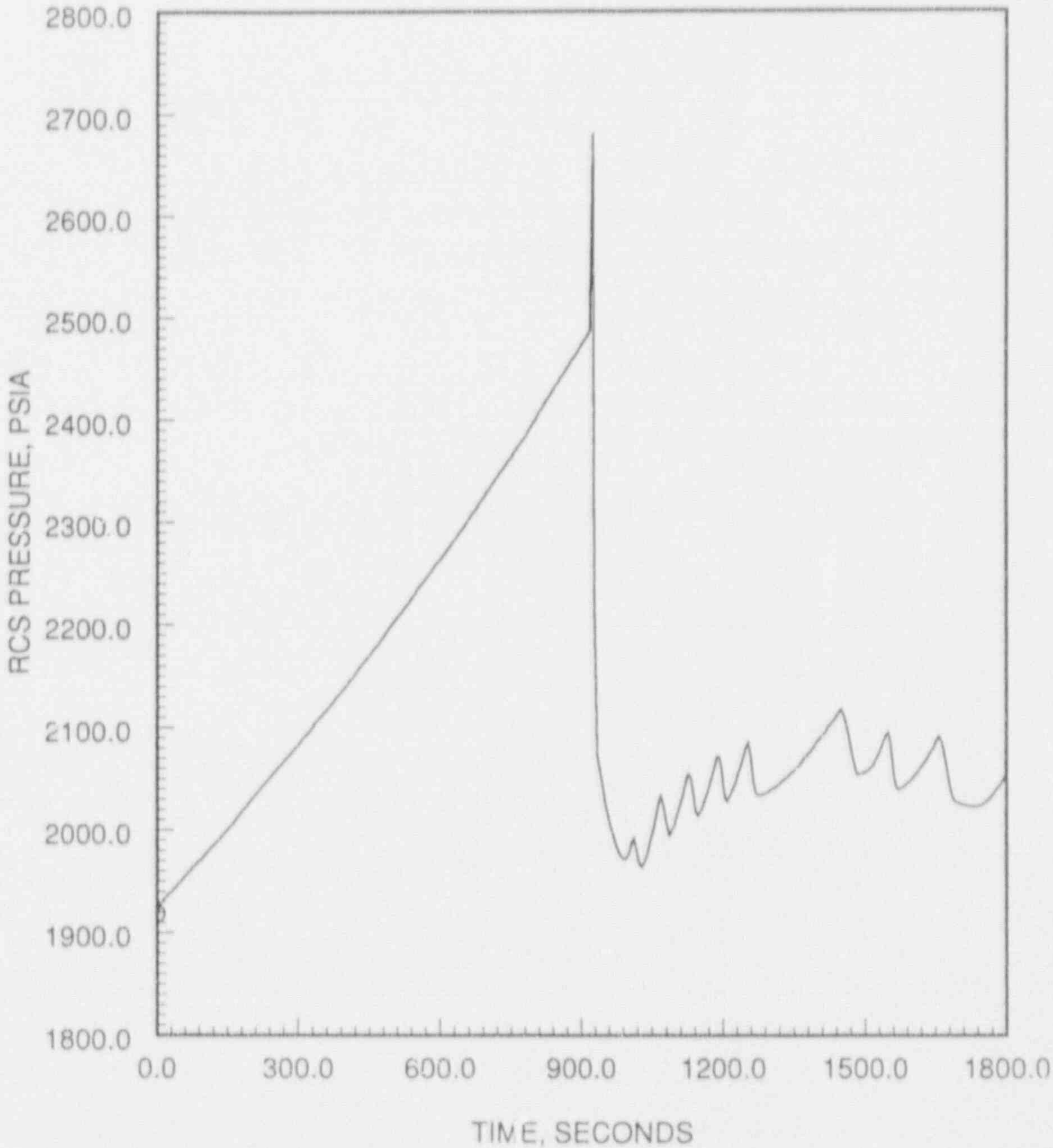


FIGURE 15.5.2-2

PZR PRESS

1993/02/05

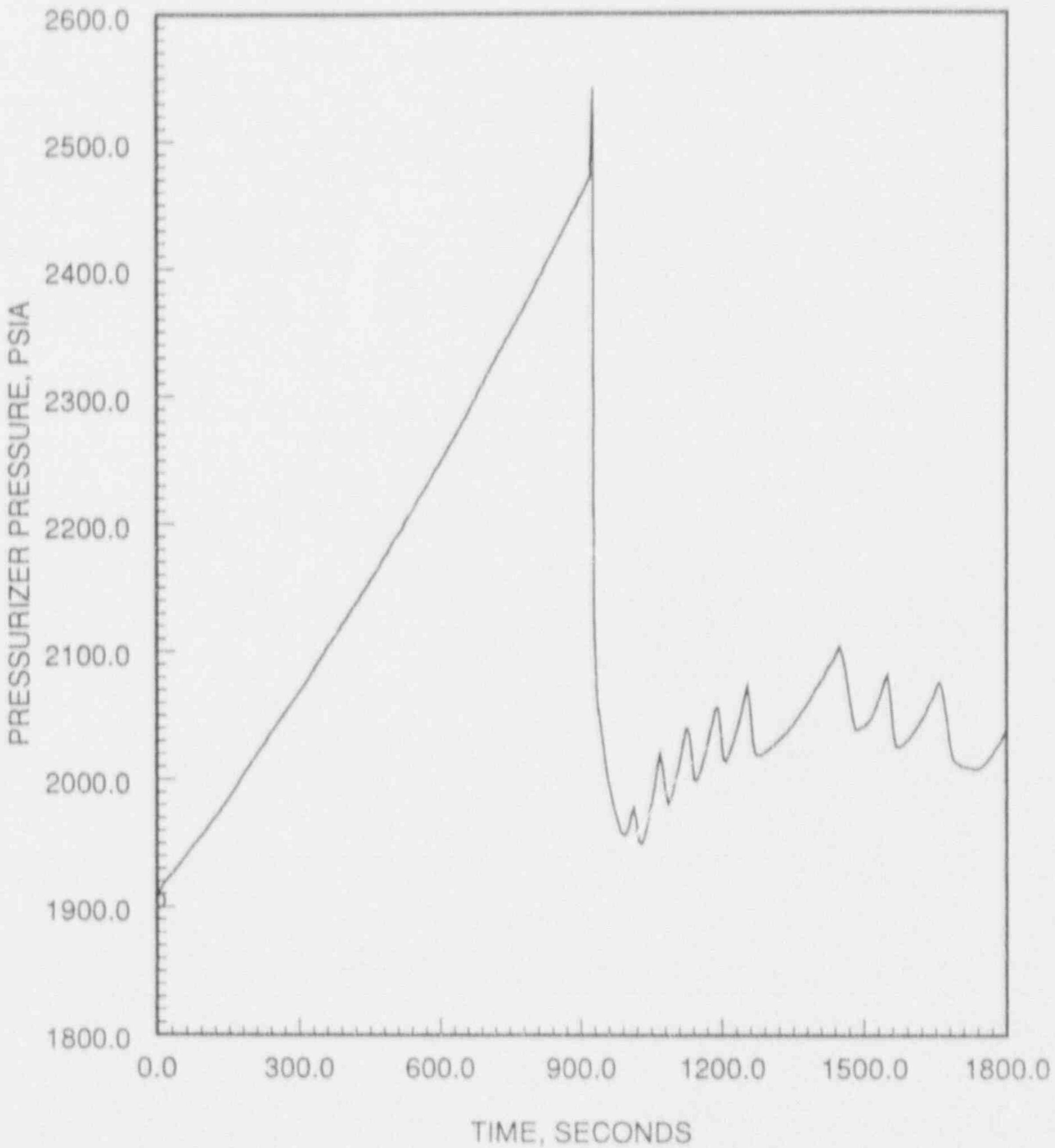
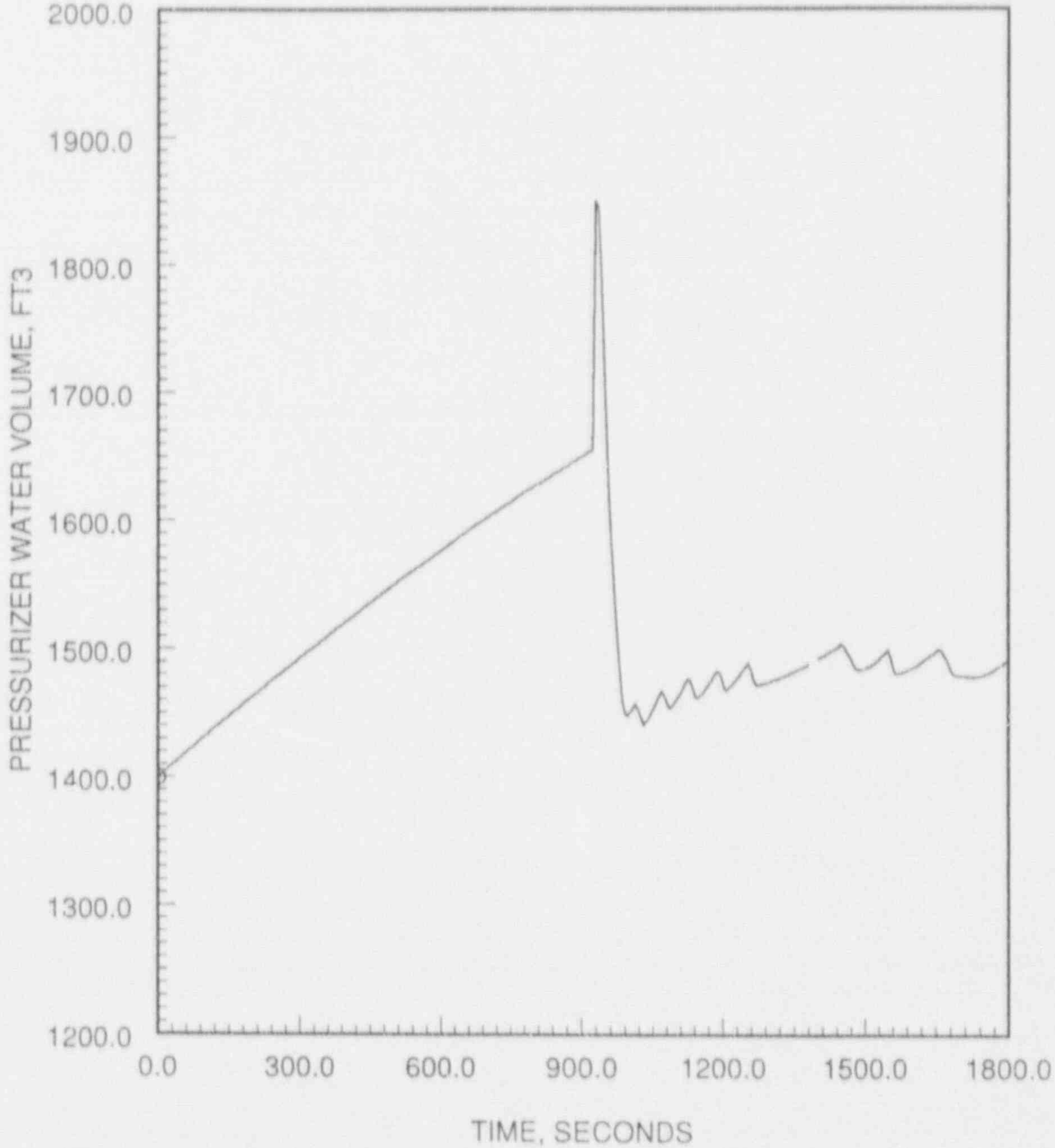


FIGURE 15.5.2-3

PZR LiqVol

1993/02/05



SYSTEM 80+ DESIGN AT 3914 MWt

RESULTS FOR CESSAR-DC APPENDIX 5D

NATURAL CIRCULATION COOLDOWN ANALYSIS

The results of a computer simulation of a natural circulation cooldown of the System 80+ NSSS from normal full power operation to shutdown cooling entry conditions is presented. The simulation utilizes the methods and assumptions approved by the NRC (see Reference 1) and is performed in conformance with the NRC Branch Technical Position RSB 5-1. The analysis includes the use of only safety-grade equipment, the concurrent loss of offsite power and a single failure. The study concludes that the System 80+ NSSS can be cooled and depressurized to shutdown cooling entry conditions in conformance with BTP RSB 5-1 requirements. RCS inventory control and boration is provided by the HPSI pumps. RCS pressure and reactor vessel upper head voiding are controlled by use of the Reactor Coolant Gas Vent System (RCGVS) in conjunction with the HPSI pumps. Significant NSSS parameters during the natural circulation cooldown are shown in Figures 5.D.1 through 5.D.5. The significant assumptions during a natural circulation cooldown are shown in Table 5.D.1. The sequence of events during a natural circulation cooldown is shown in Table 5.D.2.

Reference:

(1) "Natural Circulation Cooldown Re-Analysis for CESSAR;F," LD-83-074, A.E. Scherer (ABB-CE) to D.G. Eisenhut (USNRC), August 12, 1983.

TABLE 5.D.1

SYSTEM 80+

Natural Circulation Cooldown Assumptions

System/Component	Status
Offsite power	Unavailable
Reactor coolant pumps	Unavailable
Pressurizer pressure control system	Unavailable
Pressurizer level control system	Unavailable
Letdown	Unavailable
Charging pumps	Unavailable
Auxiliary spray	Unavailable
Pressurizer heaters	Unavailable
Steam bypass control system	Unavailable
Main feedwater system	Unavailable
CEDM cooling fan	Unavailable
Reactor protection system	Available, all rods fully inserted following reactor trip.
Emergency diesel generators	Only one available
Safety injection system	Only two of four trains available (Minimum flowrate, maximum temperature)
Atmospheric dump valves	One available per steam generator
Emergency feedwater system	Only one train available (Minimum flowrate, maximum temperature)
Emergency feedwater storage tanks	Available (minimum inventory assumed)
In-containment refueling water storage tank	Available with a boron concentration of 4000 ppm.
Reactor vessel upper head gas vent	Available (minimum flowrate assumed)
Pressurizer gas vent	Available (minimum flowrate assumed)

TABLE 5.D.2

Sequence of Events for the Natural Circulation Cooldown Analysis

Time, Hrs*	Event	Setpoint or Value
0	Loss of offsite power	-----
3 seconds	Turbine, reactor and RCPs have tripped	-----
60 seconds	Operator begins to control EFW to maintain secondary heat sink	-----
90 seconds	Operator begins to control ADVs to maintain no load conditions	1100 psia
0.2	Natural circulation flow is established	4%
4.0	Operator opens pressurizer vent of RCGVS to depressurize RCS to SIS shut off head (stops when subcooling reaches minimum allowable)	1615 psia
4.1	SIS flow delivered to RCS	-----
4.2	Operator begins 50^{0}F/hr cooldoown using ADVs	-----
4.7	Operator opens pressurizer vent to further depressurize RCS to allow more SIS flow	-----
5.0	Operator closes pressurizer vent after reactor vessel upper head (RVUH) void increases pressurizer level to maximum allowable	-----
5.01	Operator opens RVUH vent to reduce steam void volume, reduce pressurizer level and reduce RCS pressure	-----
5.30	Operator controls pressurizer level between minimum and maximum allowable level with SIS	30 - 70%
5.32	Operator closes RVUH vent when steam void collapses	-----

TABLE 5.D.2

Sequence of Events for the Natural Circulation Cooldown Analysis

Time, Hrs*	Event	Setpoint or Value
6.5	Operator opens RVUH vent	-----
6.8	Operator closes RVUH vent	-----
6.9	Operator opens pressurizer vent to further reduce RCS pressure	-----
7.0	Operator opens RVUH vent to reduce steam void	-----
8.5	Operator closes RVUH vent as steam void collapses	-----
8.52	Operator begins sequence of pressurizer vents, RVUH openings and SI flows to RCS to reduce pressure and collapse RVUH void	-----
8.83	Shutdown Cooling System (SCS) entry conditions for normal conditions reached (PP, T_{HOT})	$\leq$ 390 psia, $\leq$ 345^{o}F
9.75	SCS entry conditions for accident conditions reached (PP, T_{HOT})	$\leq$ 290 psia, $\leq$ 360^{o}F
11.7	EFW usage of 235,000 gal within minimum available capacity	700,000 gal

* Except as noted

C:\LPS\POWERUPO

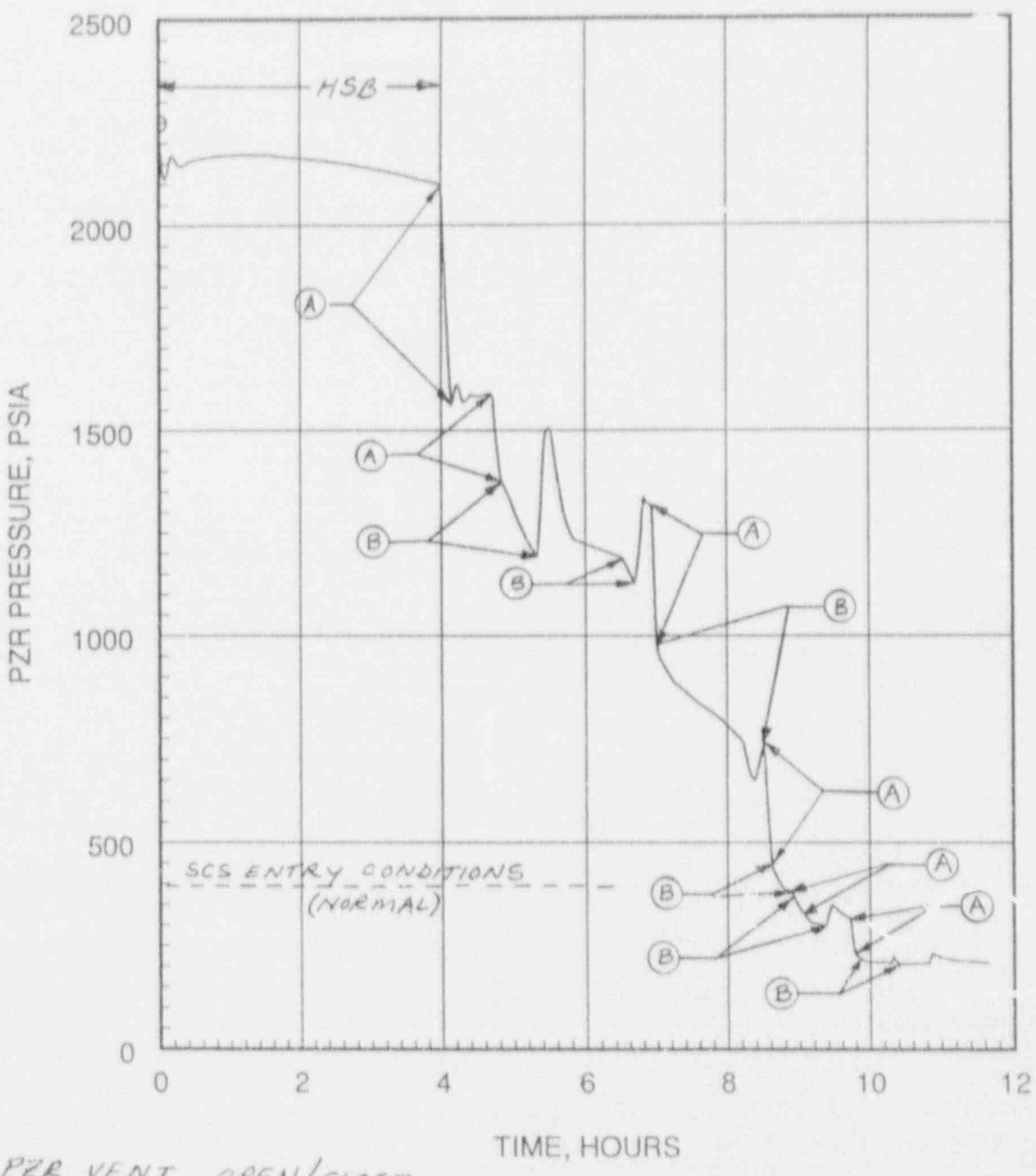
SYSTEM 80+

Natural Circulation Cooldown

LIST OF FIGURES

FIGURE 5.D.1

System 80+ NCC Analysis
PZR PRESSURE, PSIA



A = PZR VENT OPEN/CLOSE

B = RVUH VENT OPEN/CLOSE

HSB = HOT STANDBY CONDITIONS

FIGURE 5.D.2

System 80+ NCC Analysis

HOT LEG TEMPERATURE, DEG F

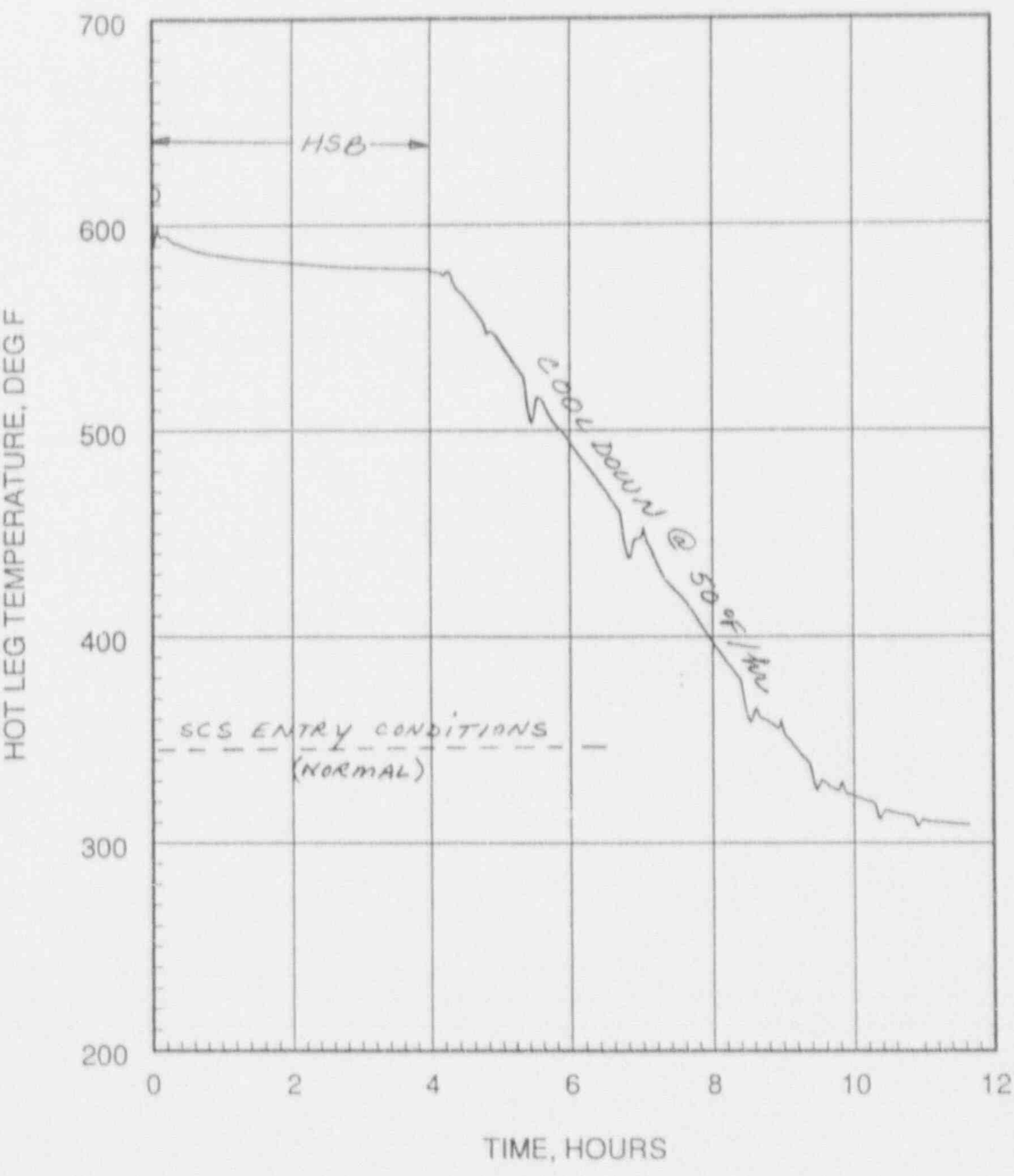


FIGURE 5.D.3

System 80+ NCC Analysis

RVUH STEAM VOLUME, FT^3

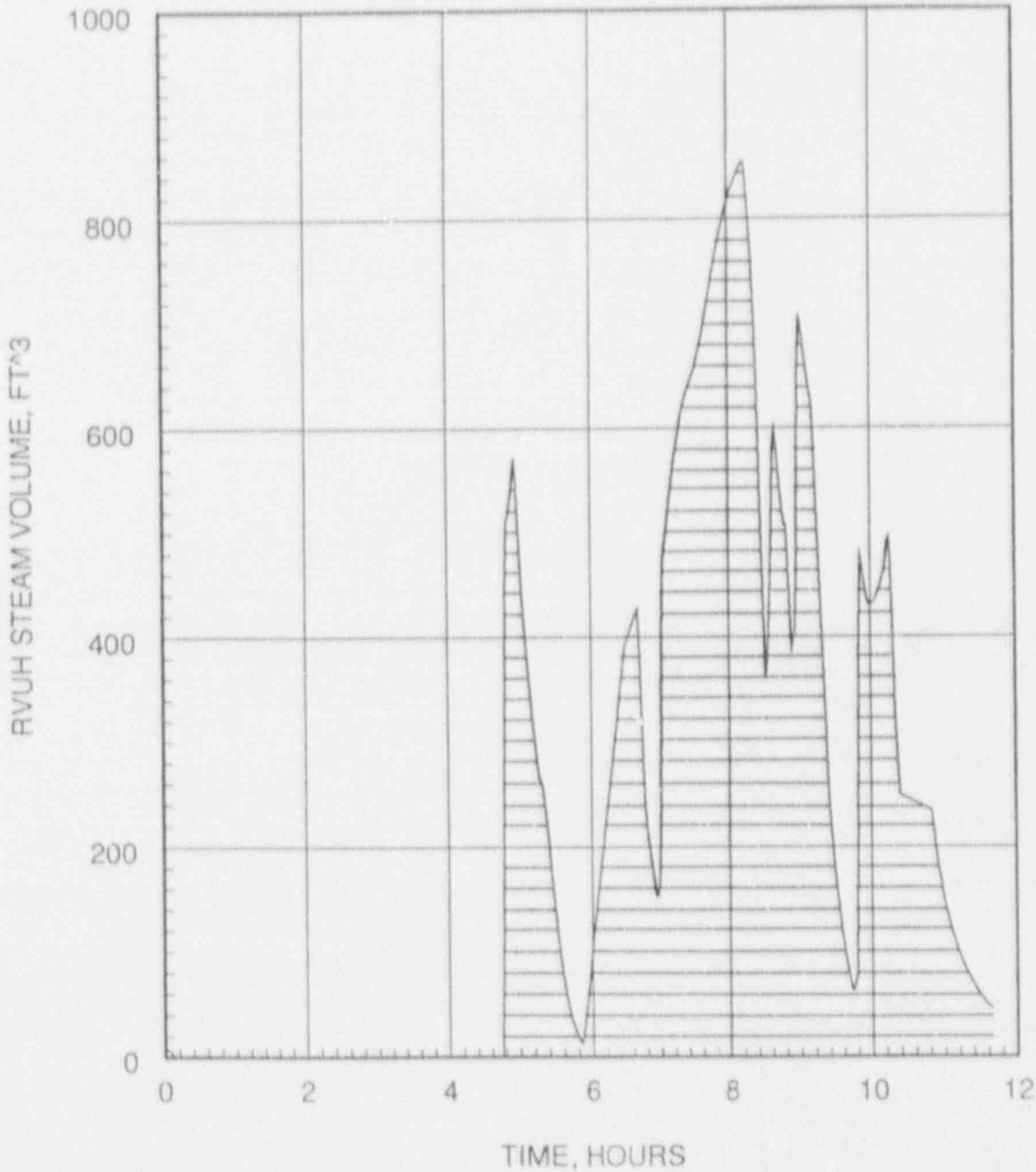


FIGURE 5.D.4

System 80+ NCC Analysis
SIS FLOWRATE (TOTAL), GPM

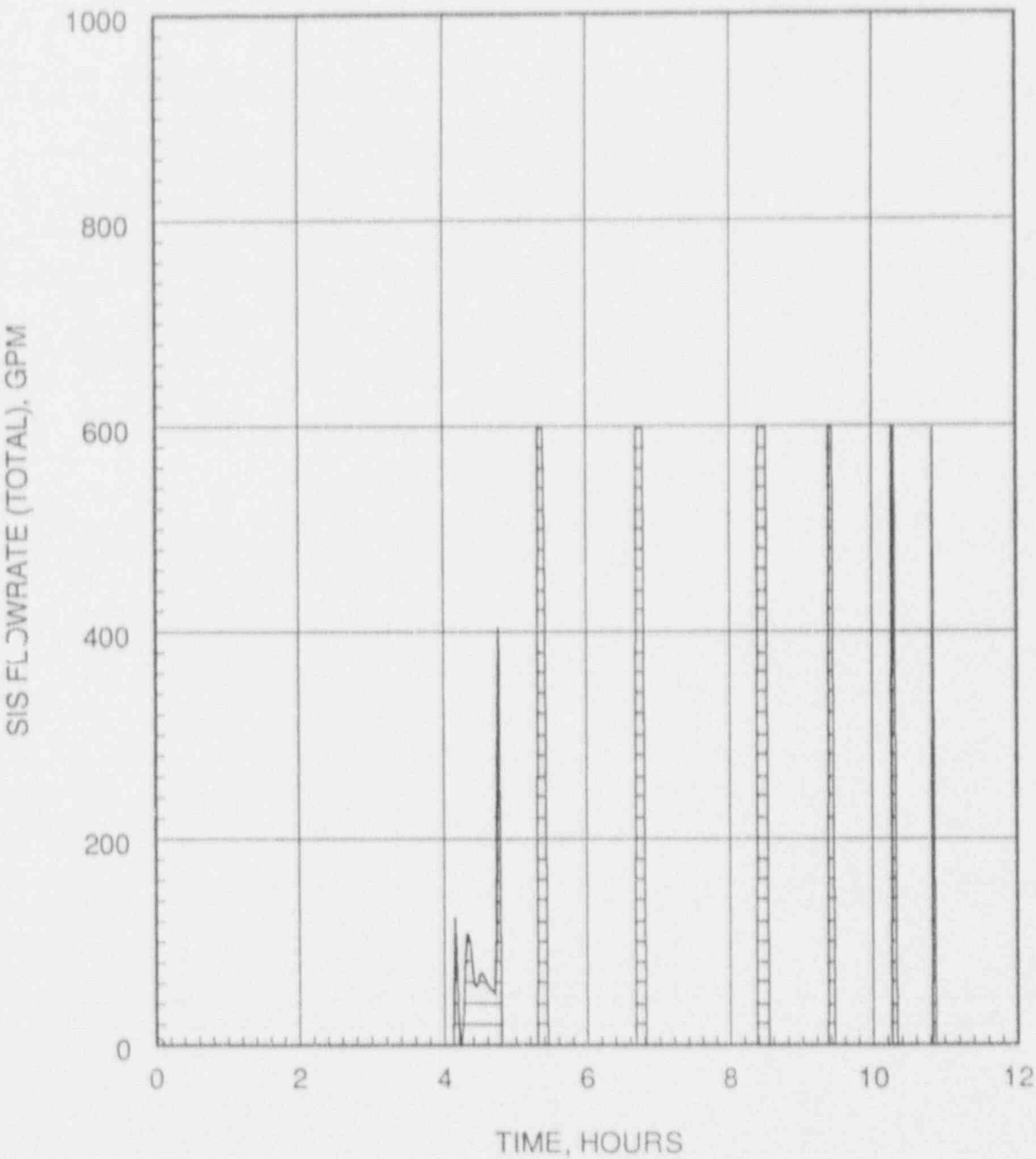


FIGURE 5.D.5

System 80+ NCC Analysis

EFW FLOW INTEGRAL, GAL* 1000