RELOAD ANALYSIS REPORT FOR SONGS 2 CYCLE 3

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1.0 Introduction and Summary

This report provides an evaluation of the design and performance of San Onofre Nuclear Generating Station Unit 2 during its third cycle of operation at 100% rated core power of 3390 MWt and NSSS power of 3410 MWt. Operating conditions for Cycle 3 have been assumed to be consistent with those of the previous cycle and are summarized as full power operation under base load conditions. The core will consist of irradiated Batch A, C, and D assemblies, along with fresh Batch E assemblies. The Cycle 2 termination burnup has been assumed to be between 9,800 and 10,200 MWD/T.

The second cycle of SONGS-2 will hereafter be referred to in this report as the "Reference Cycle."

The safety criteria (trip setpoints, margins of safety, dose limits, etc.) applicable for SONGS-2 were established in the Cycle 1 FSAR (Reference 1-1) and the Reference Cycle (Reference 1-2). A review of all postulated accidents and anticipated operational occurrences has shown that the Cycle 3 core design meets these safety criteria.

The evaluations of the Cycle 3 reload core characteristics have been examined with respect to the Reference Cycle. Specific differences in core fuel loadings have been accounted for in the present analysis. The status of the postulated accidents and anticipated operational occurrences for Cycle 3 can be summarized as follows:

1. transient data are less severe than those of the Reference Cycle analysis, therefore, no reanalysis is necessary, and

2. transient data are not bounded by those of the Referance Cycle analysis, therefore, reanalysis is required.

For those transients requiring reanalysis (Type 2), analyses are presented in Sections 7 and 8 showing results that meet the established safety criteria.

The Technical Specification changes needed for Cycle 3 are described both in Section 10 and in separate license amendment applications.

Modifications to the Core Protection Calculator (CPC) System and to the Core Operating Limit Supervisory System (COLSS) are being made to improve performance and reflect the Cycle 3 core configuration. The data base changes are a result of the Extended Cycles Program (ECP), are applicable to Cycle 3 and should be applicable to future cycles of SONGS-2. Algorithm changes are a result of the CPC Improvement Program (CIP) and are summarized in Section 9. A description of the ECP and CIP and their relationship to Cycle 3 are discussed in Reference 1-3.

2.0 Operating History of the Reference Cycle

SONGS-2 Unit 2 is currently in its second fuel cycle which began with initial criticality on April 12, 1985. Low Power Physics Testing was satisfactorily completed on April 19, 1985, and on May 2, 1985 the unit reached full power.

It is presently estimated that Cycle 2 will terminate on or about January 15, 1986. The Cycle 2 termination point can vary between 9800 MWD/T and 10,200 MWD/T to accommodate the plant schedule and still be within the assumptions of the Cycle 3 analyses.

As of June 24, 1985 the unit has had no major outages. The Cycle 2 average burnup achieved to this date is 2260 MWD/T.

3.0 General Description

The Cycle 3 core will consist of those assembly types and numbers listed in Table 3-1. Eighty Batch B assemblies and eight Batch C will be removed from the Cycle 2 core to make way for 88 fresh, Batch E assemblies. Fifty-six Batch C and all Batch D assemblies now in the core will be retained. One Batch A assembly now in the core will be replaced with one Batch A assembly discharged after Cycle 1.

The reload batch will consist of 40 type E0 assemblies, 8 type E1 assemblies with 4 burnable poison shims per assembly, 28 type E2 assemblies with 8 burnable poison shims per assembly and 12 type E3 assemblies with 16 burnable poison shims per assembly. These sub-batch types are zone-enriched and their configurations are shown in Figure 3-1.

The loading pattern for Cycle 3, showing fuel type and location, is displayed in Figure 3-2.

Figure 3-3 displays the beginning of Cycle 3 assembly average burnup distribution along with the initial assembly average fuel enrichment. The burnup distribution is based on a Cycle 2 length of 10,000 MWD/T.

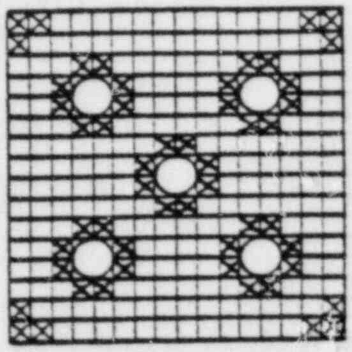
Control element assembly patterns and in-core instrument locations will remain unchanged from Cycle 2 and are shown in Figure 3-4 and Figure 3-5.

TABLE 3-1

San Onofre Nuclear Generating Station Unit 2

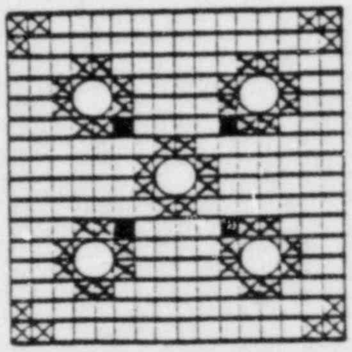
Cycle 3 Core Loading

Assembly Designation	Number of Assemblies	Fuel Rods per Assembly	Initial Enrichment (w/o U-235)	Number Shims/ Assembly	Initial Shim Loading (gm B_{10}/in)	Total Number of Fuel Rods	Total Number of Shim Rods
A	1	236	1.87	0	0	236	0
C	40	224	2.91	0	0	8960	0
		12	2.41			480	
C.	8	212	2.91	12	.01034	1696	96
		12	2.41			96	
C+	8	208	2.91	16	.01034	1664	128
		12	2.41			96	
D	56	184	3.65	0	0	10304	0
		52	2.78			2912	
D*	16	224	2.78	0	0	3584	0
		12	1.92			192	
E0	40	184	4.05	0	0	7360	0
		52	3.40			2080	
E1	8	180	4.05	4	.0192	1440	32
		52	3.40			416	
E2	28	216	3.40	8	.0242	6048	224
		12	2.78			336	
E3	12	208	3.40	16	.0192	2496	192
		12	2.78			144	
Total	217					50540	672



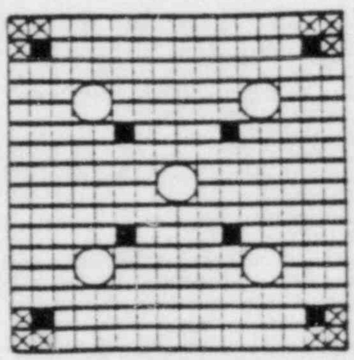
SUB-BATCH E0 40 ASSEMBLIES

- ☐ 4.05 w/o U-235
- ⊠ 3.40 w/o U-235



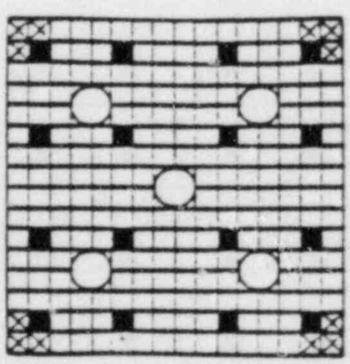
SUB-BATCH E1 - 8 ASSEMBLIES

- ☐ 4.05 w/o U-235
- ⊠ 3.40 w/o U-235
- ■ B_4C - AL_2O_3 SHIM PIN



SUB-BATCH E2 - 28 ASSEMBLIES

- ☐ 3.40 w/o U-235
- ⊠ 2.78 w/o U-235
- ■ B_4C - AL_2O_3 SHIM PIN



SUB-BATCH E3 - 12 ASSEMBLIES

- ☐ 3.40 w/o U-235
- ⊠ 2.78 w/o U-235
- ■ B_4C - AL_2O_3 SHIM PIN

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2	ENRICHMENT ZONING PATTERN FOR SONGS- BATCH E FUEL ASSEMBLIES	Figure 3-1

						E0	E0	E0	E0							
				E0	E0	E1	D	D*	D	E1	E0	E0				
			E0	D	C.	D	C	E2	C	D	C.	D	E0			
		E0	E2	D	E3	C	E2	D*	E2	C	E3	D	E2	E0		
	E0	D	D	C+	D*	E2	C	E3	C	E2	D*	C+	D	D	E0	
	E0	C.	E3	D*	C	D	D	C	D	D	C	D*	E3	C.	E0	
E0	E1	D	C	E2	D	C+	D	E2	D	C+	D	E2	C	D	E1	E0
E0	D	C	E2	C	D	D	C	C	C	D	D	C	E2	C	D	E0
E0	D*	E2	D*	E3	C	E2	C	A	C	E2	C	E3	D*	E2	D*	E0
E0	D	C	E2	C	D	D	C	C	C	D	D	C	E2	C	D	E0
	E1	D	C	E2	D	C+	D	E2	D	C+	D	E2	C	D	E1	
	E0	C.	E3	D*	C	D	D	C	D	D	C	D*	E3	C.	E0	
	E0	D	D	C+	D*	E2	C	E3	C	E2	D*	C+	D	D	E0	
		E0	E2	D	E3	C	E2	D*	E2	C	E3	D	E2	E0		
			E0	D	C.	D	C	E2	C	D	C.	D	E0			
				E0	E0	E1	D	D*	D	E1	E0	E0				
						E0	E0	E0	E0							

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2	SAN ONOFRE NUCLEAR GENERATING STATION - UNIT 2 CYCLE 3 CORE MAP	Figure 3-2

XXX
YYYY

INITIAL ENRICHMENT w/o U-235

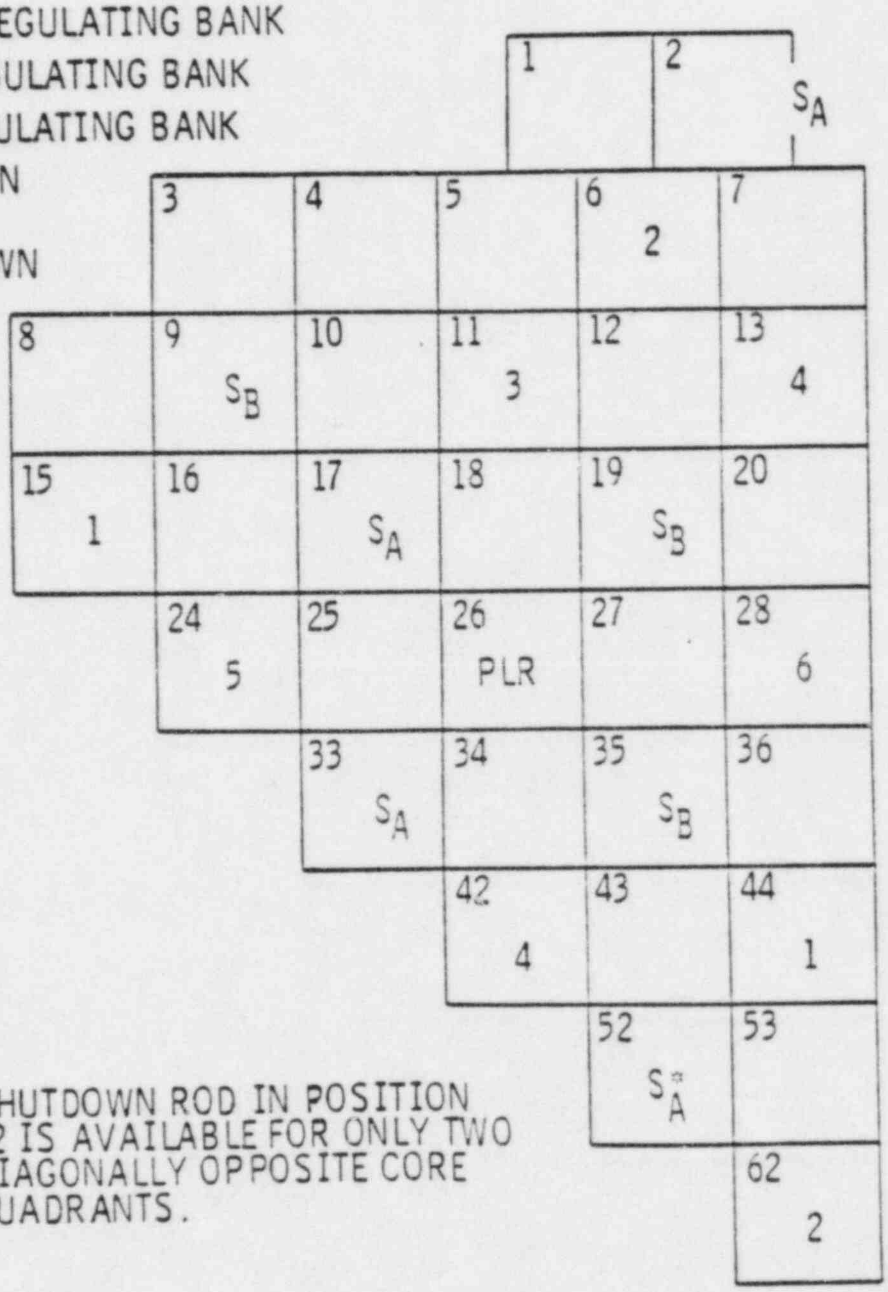
BOC ASSEMBLY AVERAGE BURNUP (MWD/T)
EOC2=10000 MWD/T

							3.91 0	3.91 0
				3.91 0	3.91 0	3.90 0	3.46 11504	2.74 12368
			3.91 0	3.46 6753	2.88 24081	3.46 9641	2.88 19876	3.37 0
		3.91 0	3.37 0	3.46 7357	3.37 0	2.88 20355	3.37 0	2.74 12714
	3.91 0	3.46 6753	3.46 7357	2.88 22233	2.74 12526	3.37 0	2.88 22188	3.37 0
	3.91 0	2.88 24081	3.37 0	2.74 12526	2.88 18857	3.46 8461	3.46 10967	2.88 20411
	3.90 0	3.46 9641	2.88 20355	3.37 0	3.46 8461	2.88 24697	3.46 7559	3.37 0
3.91 0	3.46 11504	2.88 19876	3.37 0	2.88 22188	3.46 10967	3.46 7559	2.88 20305	2.88 21247
3.91 0	2.74 12368	3.37 0	2.74 12714	3.37 0	2.88 20411	3.37 0	2.88 21247	1.87 11407

℄ ℄ ℄ ℄

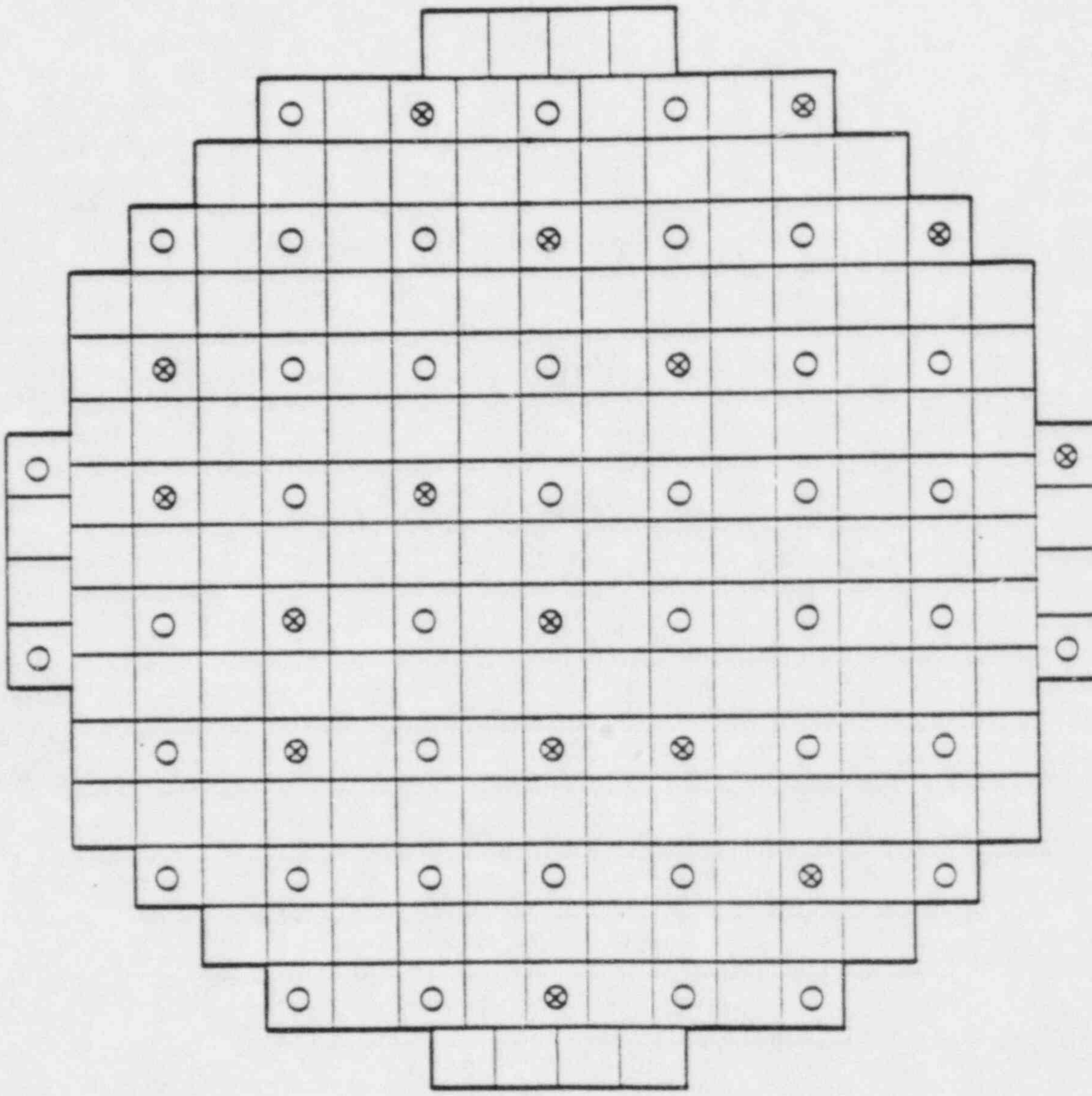
SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY AVERAGE BURNUP AND INITIAL ENRICHMENT DISTRIBUTION	Figure 3-3

6 - LEAD REGULATING BANK
5 - SECOND REGULATING BANK
4 - THIRD REGULATING BANK
3 - FOURTH REGULATING BANK
2 - FIFTH REGULATING BANK
1 - LAST REGULATING BANK
S_B - SHUTDOWN BANK B
S_A - SHUTDOWN BANK A



*SHUTDOWN ROD IN POSITION 52 IS AVAILABLE FOR ONLY TWO DIAGONALLY OPPOSITE CORE QUADRANTS.

SAN ONOFRE NUCLEAR GENERATING STATION Unit 2
CEA BANK IDENTIFICATION
Figure 3-4



○ IN-CORE INSTRUMENT ASSEMBLY LOCATION

⊗ IN-CORE INSTRUMENT ASSEMBLY WITH BACKGROUND DETECTOR LOCATION

SAN ONOFRE
NUCLEAR GENERATING STATION
Unit 2

IN-CORE INSTRUMENT ASSEMBLIES - CORE LOCATIONS

Figure 3-5

4 - System Design

4.1 Mechanical Design

The mechanical design for the standard Batch E reload fuel is essentially identical to that of Batch D fuel used in SONGS-2 Cycle 2 and described in the reload analysis report for the Reference Cycle (Reference 4-1), with the following exception:

> The designs of the CEA guide tubes and wear sleeves have been modified to permit installation of the wear sleeves completely within the guide tubes. This permits a design in which the sleeve is expanded along its entire length, thereby eliminating the need for vent holes in the sleeve and facilitates, when necessary, fuel bundle reconstitution. Reference 4-2 is C-E's submittal discussing the CEA guide tube wear sleeve modification and Reference 4-3 is the NRC's acceptance of the design change.

C-E has performed analytical predictions of cladding creep-collapse time for all SONGS-2 fuel batches that will be irradiated in Cycle 3 and has concluded that the collapse resistance of all fuel pins is sufficient to preclude collapse during Cycle 3. These analyses utilized the CEPAN computer code (Reference 4-4) and the procedures described in Reference 4-7 and included as input conservative values of internal pressure, cladding dimensions, cladding temperatures and neutron fluence.

4.2 Mitigation of Guide Tube Wear

All fuel assemblies which will be placed in CEA locations in Cycle 3 will have stainless steel sleeves installed in the guide tubes to prevent guide tube wear. The design of the sleeves for the Batch E fuel is discussed in Section 4.1 above. For all other batches of fuel a detailed discussion of the design of the sleeves and their effect on reactor operation is contained in Reference 4-12.

4.3 Thermal Design

The thermal performance of composite fuel pins that envelope the various pins of the various fuel batches present in Cycle 3 (fuel batches A, C, D

and E) have been evaluated using the FATES3A version of the fuel evaluation model (References 4-5 and 4-8) as approved by the NRC (Reference 4-9). The analysis was performed using a power history that enveloped the power and burnup levels representative of the peak pin at each burnup interval, from beginning of cycle to end of cycle burnups. The burnup range analyzed is in excess of that expected at the end of Cycle 3.

Results of these burnup dependent fuel performance calculations were used in the Transient Analysis presented in Section 7 and in the ECCS Analysis presented in Section 8.

4.4 Chemical Design

The metallurgical design specifications of the fuel cladding and the fuel assembly structural members for the Batch E fuel are identical to those of the Batches A, B and C fuel as described in Reference 4-6 and the Batch D fuel as described in Reference 4-1.

4.5 Shoulder Gap Adequacy

Calculations using the methods described in Reference 4-10 indicate that adequate shoulder gap can be provided for all fuel assemblies that will be irradiated in Cycle 3. The NRC review conducted on these methods (Reference 4-11) concluded that additional data were necessary before the methods were useable on fuel accumulating fluences exceeding 6.5×10^{21} nvt. Therefore, an inspection program and an evaluation will be performed to ensure that adequate shoulder gap remains on fuel scheduled for its third cycle of service in Cycle 3.

5.0 Nuclear Design

5.1 Physics Characteristics

5.1.1 Fuel Management

The Cycle 3 loading pattern is characterized by loading approximately half of the fresh fuel on the core periphery and shuffling to the interior the fuel assemblies previously located on the periphery in Cycle 2. Forty fresh fuel assemblies have a lower assembly average enrichment than those on the periphery and are mixed with the previously burned fuel in the central region of the core in a pattern which minimizes power peaking. With this loading and a Cycle 2 endpoint at 10,000 MWD/T, the Cycle 3 reactivity lifetime for full power operation is expected to be 14,500 MWD/T. Explicit evaluations have been performed to assure applicability of all analyses to a Cycle 2 termination burnup of between 9,800 and 10,200 MWD/T and for a Cycle 3 length up to 16,000 MWD/T.

Characteristic physics parameters for Cycle 3 are compared to those of the Reference Cycle in Table 5-1. The values in this table are intended to represent nominal core parameters. Those values used in the safety analysis (see Sections 7 and 8) contain appropriate uncertainties, or incorporate values from the Extended Cycles Program (Reference 5-1) to bound future operating cycles, and in all cases are conservative with respect to the values reported in Table 5-1.

Table 5-2 presents a summary of CEA reactivity worths and allowances for the end of Cycle 3 full power steam line break transient with a comparison to the Reference Cycle data. The full power steam line break was chosen to illustrate differences in CEA reactivity worths for the two cycles.

The CEA core locations and group identifications remain the same as in the Reference Cycle. The power dependent insertion limit (PDIL) for regulating groups and part length CEA groups remains the same as in the Reference Cycle and is shown in Figures 5-1 and 5-2 respectively. Table 5-3 shows the reactivity worths of various CEA groups calculated at full power conditions for Cycle 3 and the Reference Cycle.

5.1.2 Power Distribution

Figures 5-3 through 5-5 illustrate the calculated All Rods Out (ARO) planar radial power distributions during Cycle 3. The one-pin planar radial power peaks presented in these figures represent the maximum that could be expected between about 20 and 80 percent of core height. Power peaks outside this axial region were examined and found not to be limiting at any time during the cycle. Time points at the beginning, middle, and end of cycle were chosen to display the variation in maximum planar radial peak as a function of burnup.

Radial power distributions for rodded configurations are given for BOC and EOC in Figures 5-6 through 5-11. The rodded configurations shown are those allowed by the PDIL at full power: part length CEAs (PLCEAs), Bank 6, and Bank 6 plus the PLCEAs. As is the case for unrodded configurations, the largest planar radial peak for each of these rodded configurations occurs at beginning of Cycle 3.

The radial power distributions described in this section are calculated data which do not include any uncertainties or allowances. The calculations performed to determine these radial power peaks explicitly account for augmented power peaking which is characteristic of fuel rods adjacent to the water holes.

Nominal axial peaking factors are expected to range from 1.24 at BOC3 to 1.09 at EOC3.

5.2 Safety Related Data

5.2.1 Augmentation Factors

A recently completed analysis performed by C-E for EPRI, Reference 5-2, demonstrated that the increased power peaking associated with the small interpellet gaps found in C-E's modern fuel rods (non-densifying fuel in pre-pressurized tubes) is insignificant compared to the uncertainties in the safety analyses. The report concluded that augmentation factors can be eliminated from the reload analyses of any reactor loaded exclusively with this type of fuel. This discussion of the elimination of the augmentation

factors was used by BG&E in Reference 5-3 and accepted by the NRC in Reference 5-4. Since the manufacturing process of C-E's modern fuel is the same for both BG&E and SCE, and the fuel differs only in dimensions, it is C-E's conclusion that the peaking factor penalty due to fuel densification is insignificant compared to the uncertainties incorporated into COLSS and CPC and thus the augmentation factors have been eliminated for Cycle 3.

5.3 Physics Analysis Methods

5.3.1 Analytical Input to In-Core Measurements

In-core detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in accordance with Reference 5-5. As in the Reference Cycle, ROCS-DIT with the MC module will be used. ROCS-DIT and the MC module have been approved for this application in Reference 5-6.

5.3.2 Uncertainties in Measured Power Distributions

The planar radial power distribution measurement uncertainty of 5.3%, based on Reference 5-5, will be applied to the Cycle 3 COLSS and CPC on-line calculations which use planar radial power peaks. The axial and three dimensional power distribution measurement uncertainties are determined in conjunction with other monitoring and protection system measurement uncertainties, as was done for Cycle 2.

5.3.3 Nuclear Design Methodology

As in the Reference Cycle, the Cycle 3 nuclear design was performed with two and three dimensional core models using the ROCS computer code and employing DIT calculated cross sections. The ROCS-DIT and the MC module was described in Reference 5-6.

TABLE 5-1

SONGS-2 CYCLE 3

NOMINAL PHYSICS CHARACTERISTICS

Dissolved Boron	Units	Reference Cycle	Cycle 3
Dissolved Boron Concentration for Criticality, CEAs Withdrawn, Hot Full Power Equilibrium Xenon, BOC	PPM	845	1186
Boron Worth			
Hot Full Power, BOC	PPM/% $\Delta\rho$	98	114
Hot Full Power, EOC	PPM/% $\Delta\rho$	84	94
Moderator Temperature Coefficients			
Hot Full Power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\Delta\rho/^{o}F$	-0.4	-0.2
End of Cycle	$10^{-4}\Delta\rho/^{o}F$	-2.4	-2.6
Doppler Coefficient			
Hot Zero Power, BOC	$10^{-5}\Delta\rho/^{o}F$	-1.73	-1.68
Hot Full Power, BOC	$10^{-5}\Delta\rho/^{o}F$	-1.25	-1.21
Hot Full Power, EOC	$10^{-5}\Delta\rho/^{o}F$	-1.38	-1.41
Total Delayed Neutron Fraction, β_{eff}			
BOC	-------	0.0066	0.0066
EOC	-------	0.0046	0.0051
Neutron Generation Time, ℓ^*			
BOC	10^{-6} sec	23.7	22.3
EOC	10^{-6} sec	34.1	27.1

TABLE 5-2

SONGS-2 CYCLE 3 LIMITING VALUES OF

REACTIVITY WORTHS AND ALLOWANCES FOR HOT

FULL POWER STEAM LINE BREAK, %Δρ END-OF-CYCLE (EOC)

	Reference Cycle	Cycle 3
1. Worth of all CEAs Inserted	-10.4	-11.4
2. Stuck CEA Allowance	+2.45	+1.9
3. Worth of all CEAs Less Highest Worth CEA Stuck Out	-7.95	-9.5
4. Full Power Dependent Insertion Limit CEA Bite	+0.2	+0.2
5. Calculated Scram Worth	-7.75	-9.3
6. Physics Uncertainty	+.65	+.80
7. Other Allowances (worth losses due to voiding and moderator temperature axial redistribution)	+0.2	+0.2
8. Net Available Scram Worth	-6.9	-8.3

TABLE 5-3

SONGS-2 CYCLE 3

REACTIVITY WORTH OF CEA REGULATING GROUPS

AT HOT FULL POWER, %Δρ

Regulating CEAs	Beginning of Cycle: Reference Cycle	Beginning of Cycle: Cycle 3	End of Cycle: Reference Cycle	End of Cycle: Cycle 3
Group 6	0.4	0.3	0.4	0.5
Group 5	0.3	0.3	0.4	0.4

Note:

Values shown assume sequential group insertion.

FIGURE 5-1 SONGS-2 Cycle 3 PDIL for Regulating Groups

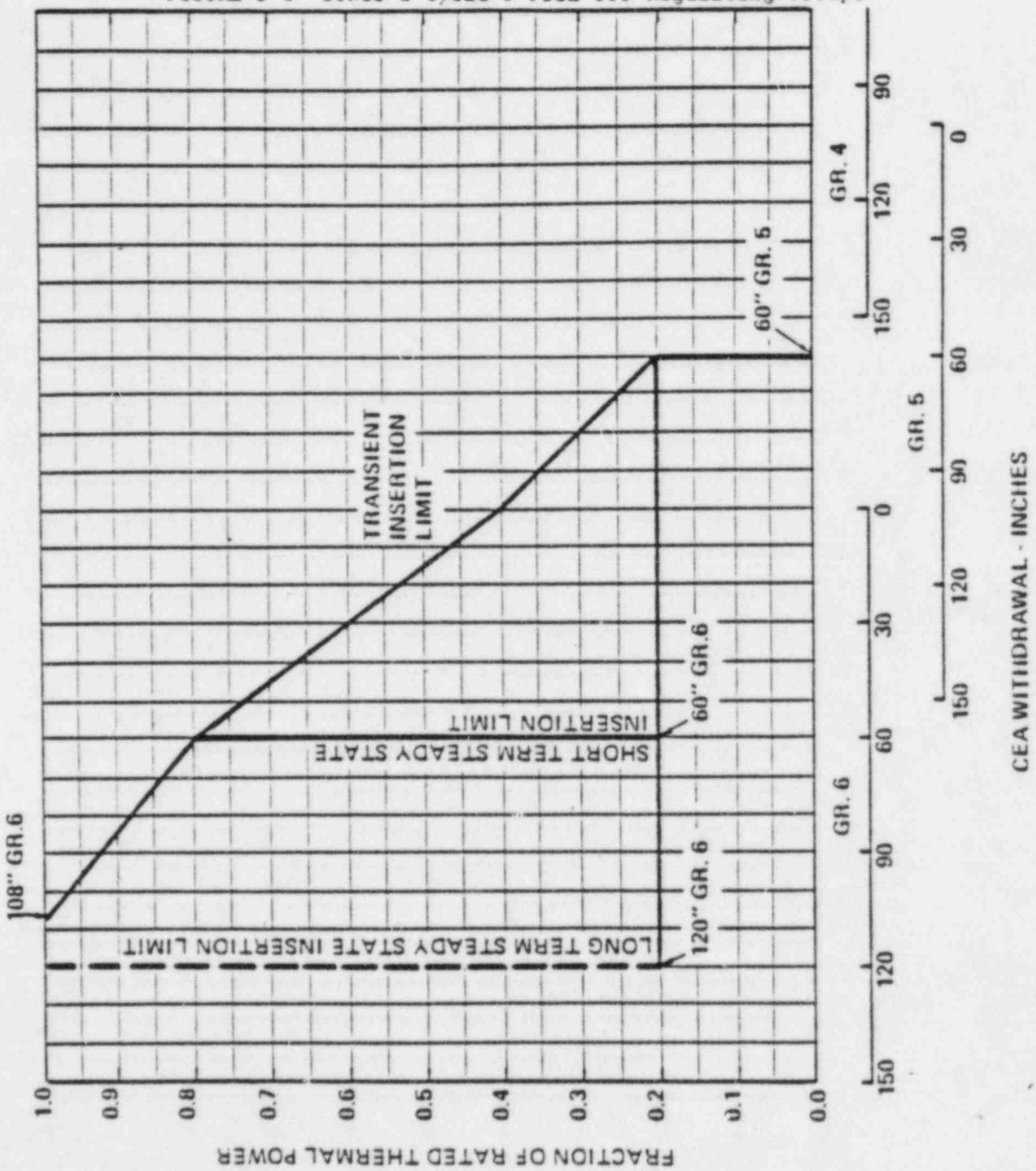
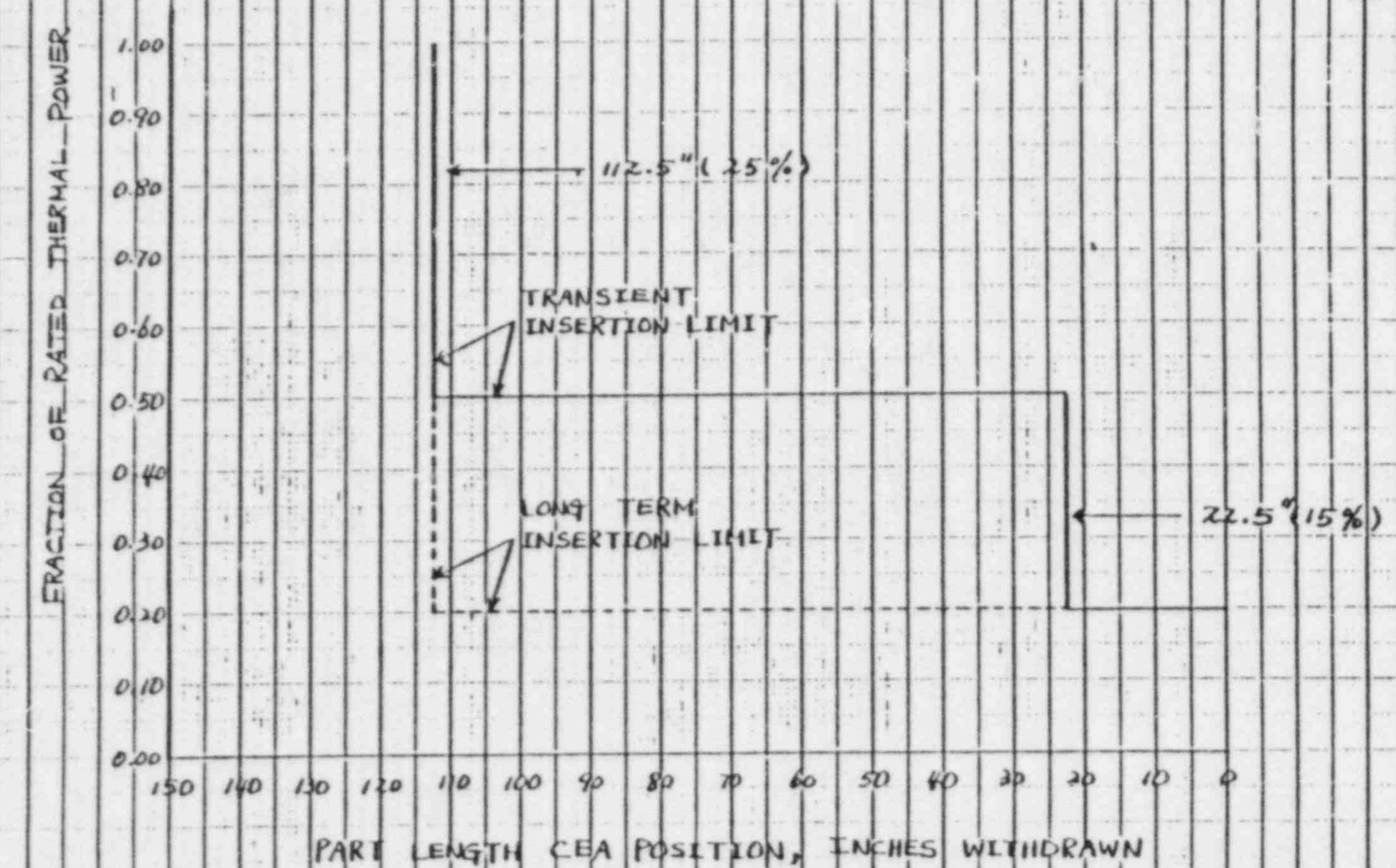


FIGURE 5-2

SONGS-2 CYCLE-3

PART LENGTH CEA INSERTION LIMIT VS. THERMAL POWER



							0.78	1.00
				0.69	0.94	1.16	X 1.12	1.03
			0.83	1.05	0.82	1.10	0.94	1.24
		0.83	1.16	1.17	1.09	0.87	1.18	0.99
	0.69	1.05	1.17	0.87	0.93	1.20	0.87	1.09
	0.94	0.82	1.09	0.93	0.90	1.19	1.14	0.90
	1.16	1.10	0.87	1.20	1.19	0.90	1.16	1.18
0.73	1.12	0.94	1.17	0.87	1.13	1.15	0.78	0.70
0.99	1.02	1.24	0.99	1.08	0.89	1.17	0.69	0.55

ASSEMBLY RELATIVE POWER DENSITY

℄

X = LOCATION OF MAXIMUM 1-PIN PEAK = 1.48

SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY RELATIVE POWER DENSITY, HFP AT BOC, EQUILIBRIUM XENON, ARO	Figure 5-3

ASSEMBLY RELATIVE POWER DENSITY

℄

							0.71	0.88
				0.67	0.89	1.07	1.02	0.95
			0.81	1.01	0.84	1.08	0.93	1.24
		0.81	1.14	1.16	1.22	0.93	1.25	1.05
	0.67	1.01	1.15	0.91	0.98	1.26	0.94	X 1.25
	0.89	0.84	1.22	0.98	0.93	1.16	1.13	0.94
	1.07	1.08	0.93	1.26	1.16	0.90	1.13	1.20
0.71	1.02	0.93	1.25	0.94	1.13	1.12	0.81	0.76
0.88	0.95	1.24	1.05	1.25	0.93	1.19	0.75	0.64

℄ ℄ ℄

X = LOCATION OF MAXIMUM 1-PIN PEAK = 1.42

SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY RELATIVE POWER DENSITY, HFP AT 8 GWD/T, EQUILIBRIUM XENON, ARO	Figure 5--

ASSEMBLY RELATIVE POWER DENSITY

						0.70	0.86	
				0.69	0.89	1.06	0.99	0.93
			0.81	1.00	0.87	1.06	0.94	1.24
		0.81	1.15	1.14	1.28	0.96	1.27	1.06
	0.69	1.00	1.14	0.92	0.99	1.26	0.96	X 1.31
	0.89	0.87	1.28	0.99	0.91	1.10	1.08	0.94
	1.06	1.06	0.96	1.26	1.10	0.88	1.08	1.19
0.70	0.99	0.94	1.27	0.96	1.08	1.08	0.83	0.79
0.36	0.93	1.25	1.06	1.31	0.94	1.19	0.79	0.70

℄ ℄ ℄ ℄

X = LOCATION OF MAXIMUM 1-PIN PEAK = 1.46

SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY RELATIVE POWER DENSITY, HFP AT EOC, EQUILIBRIUM XENON, ARO	Figure 5-5

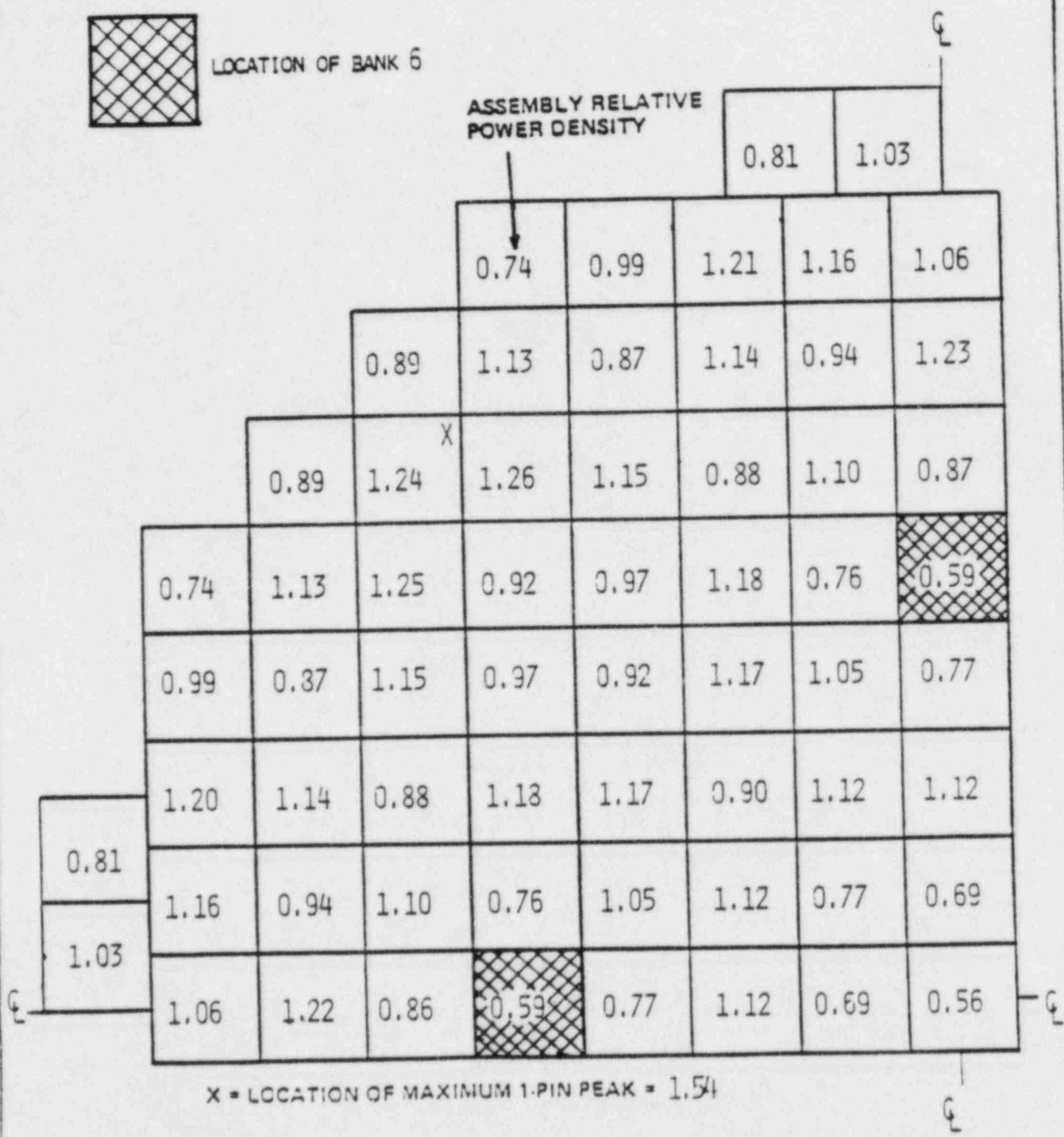
LOCATION OF PLCEAS

ASSEMBLY RELATIVE POWER DENSITY

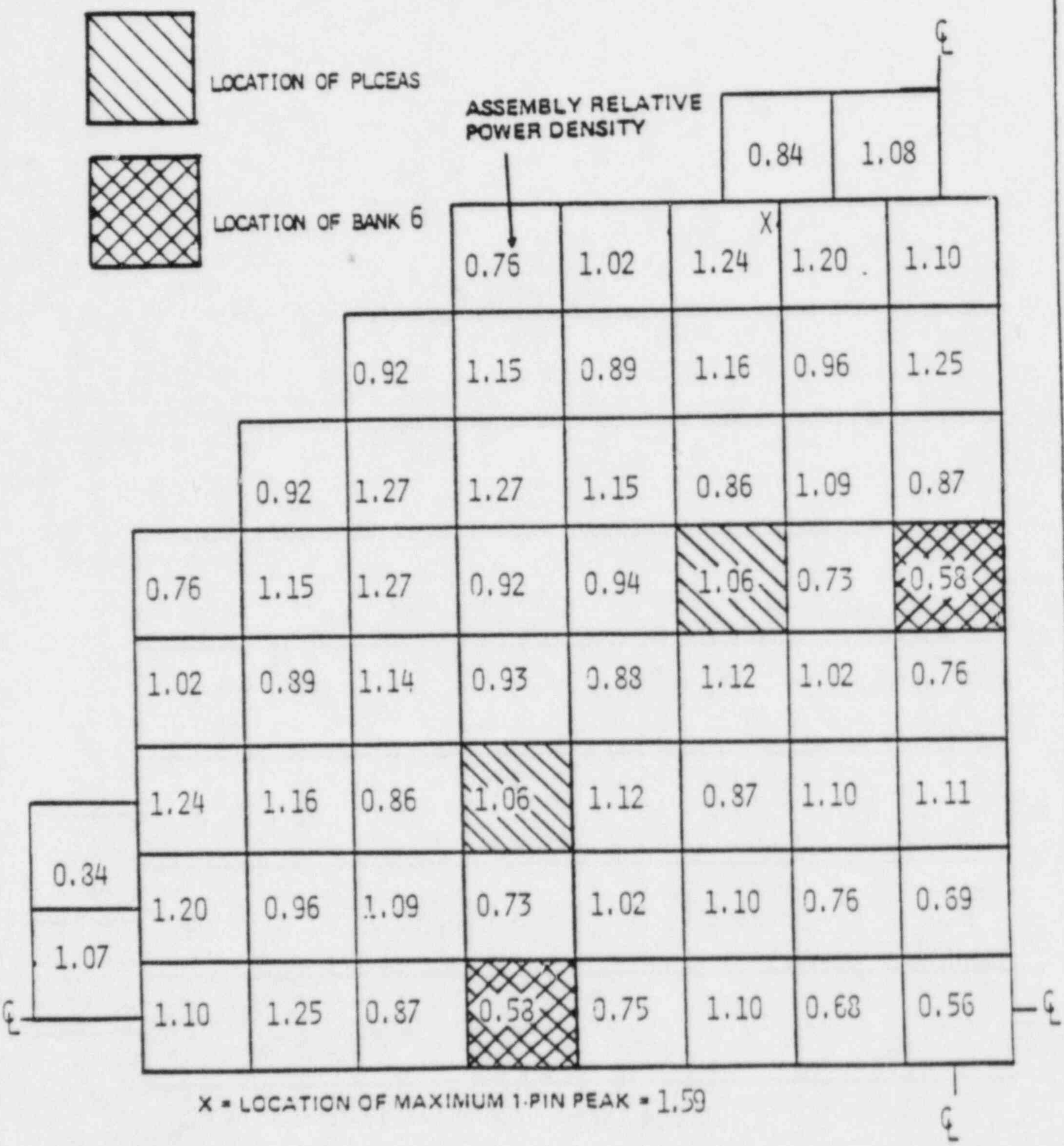
							0.79	1.01
				0.70	0.95	X 1.17	1.14	1.05
			0.84	1.06	0.83	1.11	0.95	1.26
		0.84	1.16	1.18	1.09	0.86	1.17	1.00
	0.70	1.06	1.18	0.89	0.91	1.09	0.86	1.09
	0.94	0.83	1.09	0.91	0.89	1.17	1.14	0.91
	1.17	1.11	0.86	1.09	1.16	0.92	1.18	1.20
0.79	1.14	0.95	1.17	0.85	1.13	1.17	0.80	0.73
1.01	1.04	1.25	0.99	1.09	0.90	1.19	0.72	0.58

X = LOCATION OF MAXIMUM 1-PIN PEAK = 1.51

SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY RELATIVE POWER DENSITY, HFP AT BOC, EQUILIBRIUM XENON, WITH PLCEAS	Figure 5-6



SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY RELATIVE POWER DENSITY, HFP AT BOC, EQUILIBRIUM XENON, WITH BANK 6	Figure 5-7



SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY RELATIVE POWER DENSITY, HFP AT BOC, EQUILIBRIUM XENON, WITH BANK 6 AND PLCEAS	Figure 5-8

LOCATION OF PLCEAS

ASSEMBLY RELATIVE POWER DENSITY

							0.72	0.88	℄
				0.70	0.91	1.08	1.02	0.95	
			0.83	1.02	0.88	1.08	0.95	X 1.27	
		0.83	1.17	1.15	1.28	0.94	1.27	1.07	
	0.70	1.02	1.15	0.92	0.96	1.15	0.93	1.30	
	0.91	0.88	1.28	0.96	0.88	1.07	1.07	0.94	
	1.08	1.08	0.94	1.15	1.07	0.87	1.08	1.20	
0.72	1.02	0.96	1.27	0.93	1.07	1.08	0.83	0.80	
℄ 0.88	0.95	1.27	1.07	1.30	0.94	1.20	0.79	0.71	℄
								℄	

X = LOCATION OF MAXIMUM 1-PIN PEAK = 1.46

SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY RELATIVE POWER DENSITY, HFP AT EOC, EQUILIBRIUM XENON, WITH PLCEAS	Figure 5-9

LOCATION OF BANK 6

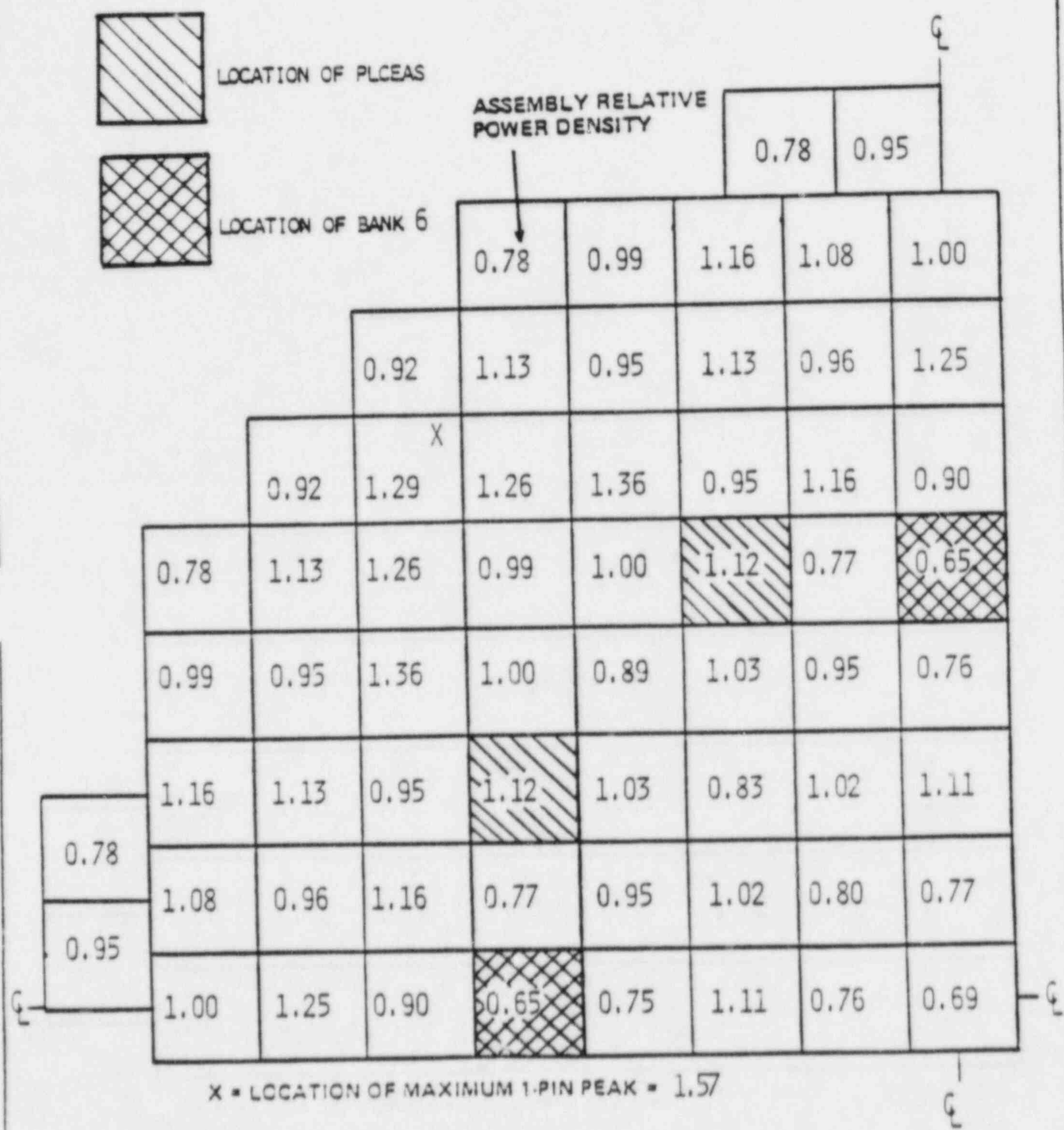
ASSEMBLY RELATIVE POWER DENSITY

								0.76	0.91
					0.76	0.97	1.13	1.05	0.97
				0.90	1.10	0.94	1.11	0.95	1.23
			0.90	1.27	1.24	1.36 X	0.97	1.17	0.90
		0.76	1.10	1.24	0.99	1.03	1.24	0.80	0.65
		0.97	0.94	1.36	1.03	0.93	1.07	0.97	0.76
		1.13	1.11	0.97	1.24	1.08	0.85	1.03	1.12
	0.76	1.05	0.95	1.17	0.80	0.97	1.02	0.80	0.76
℄	0.92	0.97	1.23	0.90	0.65	0.76	1.12	0.76	0.68 — ℄

℄

X = LOCATION OF MAXIMUM 1-PIN PEAK = 1.54

SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY RELATIVE POWER DENSITY, HFP AT EOC, EQUILIBRIUM XENON, WITH BANK 6	Figure 5-10



SOUTHERN CALIFORNIA EDISON CO. SONGS-2	SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2 CYCLE 3 ASSEMBLY RELATIVE POWER DENISTY, HFP AT EOC, EQUILIBRIUM XENON, WITH BANK 6 AND PLCEAS	Figure 5-1

6.0 Thermal-Hydraulic Design

6.1 DNBR Analysis

Steady state DNBR analyses of Cycle 3 at the rated power level of 3390 MWT have been performed using the TORC computer code described in Reference 6-1, the CE-1 critical heat flux correlation described in Reference 6-2, the simplified TORC modeling methods described in Reference 6-3, and the CETOP code described in Reference 6-4.

Table 6-1 contains a list of pertinent thermal-hydraulic design parameters. The Statistical Combination of Uncertainties (SCU) methodology presented in Reference 6-5 was applied with the calculational factors listed in Table 6-1 and other uncertainty factors at the 95/95 confidence/probability level to define a design limit of 1.31 on CE-1 minimum DNBR which was approved for use in the Reference Cycle. This limit has been verified for Cycle 3.

Information on the HID-1 and HID-2 grids is provided in References 6-6 and 6-7. The use of both HID-1 and HID-2 grids has already been approved by NRC for the SONGS 2 and 3 cores (Reference 6-8). A penalty of 0.01 was imposed by NRC on the CE-1 correlation DNBR limit for SONGS-2 and 3 to address NRC concern about the effect of the HID-1 and HID-2 spacer grids and a larger grid spacing. This penalty is included in the 1.31 DNBR limit, along with other penalties imposed by NRC in the review of previous SCU analyses (Reference 6-10).

6.2 Effects of Fuel Rod Bowing on DNBR Margin

Effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analyses in the manner discussed in References 6-5 and 6-9. The penalty used for this analysis, 1.75% MDNBR, is valid for bundle burnups up to 30,000 MWD/MTU. This penalty is included in the 1.31 DNBR limit.

For assemblies with burnup greater than 30 GWD/T sufficient available margin exists to offset rod bow penalties due to the lower radial power peaks in these higher burnup batches. Hence the rod bow penalty based upon Reference 6-9 for 30 GWD/T is applicable for all assembly burnups expected for Cycle 3.

TABLE 6-1

SONGS-2 Cycle 3
Thermal Hydraulic Parameters at Full Power

General Characteristics	Unit	Reference Cycle	Cycle 3
Total Heat Output (Core only)	MWt 10^6 Btu/hr	3390 11,570	3390 11,570
Fraction of Heat Generated in Fuel Rod	--	0.975	0.975
Primary System Pressure Nominal	psia	2250	2250
Inlet Temperature (Nominal)	^oF	553.0	553.0
Total Reactor Coolant Flow (Minimum Steady State)	gpm 10^6 lb/hr	396,000 148.0	396,000 148.0
Coolant Flow Through Core (Minimum)	10^6 lb/hr	143.9	143.6^+
Hydraulic Diameter (Nominal Channel)	ft	0.039	0.039
Average Mass Velocity	10^6 lb/hr-ft	2.63	2.63
Pressure Drop Across Core (Minimum steady state flow irreversible P over entire fuel assembly)	psi	20.0	19.9
Total Pressure Drop Across Vessel (Based on nominal dimensions and minimum steady state flow)	psi	43.6	43.6
Core Average Heat Flux (Accounts for fraction of heat generated in fuel rod and axial densification factor)	BTU/hr-ft^2	182,400***	178,900*
Total Heat Transfer Area (Accounts for axial densification factor)	ft^2	62,000***	63,000*
Film Coefficient at Average Conditions	BTU/hr-ft^2 ^oF	6200	6200
Average Film Temperature Difference	^oF	29.4	28.8
Average Linear Heat Rate of Undensified Fuel Rod (Accounts for fraction of heat generated in fuel rod)	kw/ft	5.34***	5.23*
Average Core Enthalpy Rise	BTU/lb	80.4	80.6

TABLE 6-1 (continued)

Calculational Factors	Unit	Reference Cycle	Cycle 3
Maximum Clad Surface Temperature	^oF	656.7	656.7
Engineering Heat Flux Factor		1.03**	1.03**
Engineering Factor on Hot Channel Heat Input		1.03**	1.03**
Rod Pitch, Bowing and Clad Diameter Factor		1.05**	1.05**
Fuel Densification Factor (Axial)		1.002	1.002

NOTES:

* Based on 672 shims.

** These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level to define a new design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6-5.

*** Based on 1632 shims.

+ Design bypass flowrate has increased from 2.8% to 3.0% of total reactor coolant flow.

7.0 Non-LOCA Safety Analysis

7.0.1 Introduction

This section presents the results of the San Onofre Nuclear Generating Station (SONGS) Unit 2, Cycle 3 Non-LOCA safety analyses at 3410 MWt.

The Design Bases Events (DBEs) considered in the safety analyses are listed in Table 7.0-1. These events are categorized into three groups: Moderate Frequency, Infrequent and Limiting Fault events. For the purpose of this report, the Moderate Frequency and Infrequent Events will be termed Anticipated Operational Occurrences. The DBEs were evaluated with respect to four criteria: Offsite Dose, Reactor Coolant System Pressure, Fuel Performance (DNBR and Centerline Melt SAFDLs) and Loss of Shutdown Margin. Tables 7.0-2 through 7.0-5 present the list of events analyzed for each criterion. All events were re-evaluated to assure that they meet their respective criterion with the Cycle 3 fuel design. The DBEs chosen for analysis for each criterion are the limiting events with respect to that criterion.

The write-ups for those events presented are broken down into a discussion of the reason(s) for the reanalysis, a discussion of the cause(s) of the event, a description of the analyses performed, the results and conclusions. In the Reference Cycle (Reference 7-1), some events previously analyzed with and without a single failure in the Cycle 1 FSAR (Reference 7-2) had been combined into the same section for presentation. This practice is repeated for Cycle 3.

7.0.2 Methods of Analysis

The analytical methodology used is consistent with the Reference Cycle analysis methods (Reference 7-1) unless otherwise stated in the event presentation.

7.0.3 Mathematical Models

The mathematical models and computer codes used in the Cycle 3 Non-LOCA safety analysis are identical to those used in the Reference Cycle (Reference 7-1). The exceptions to this are the application of the TORC code to the sheared shaft event and the HERMITE code to the Total Loss of Forced Reactor Coolant Flow and the Asymmetric Steam Generator Transient.

Plant response for Non-LOCA Events was simulated using the CESEC III computer code (Reference 7-10).

The STRIKIN II computer code (Section 15.0.4.1.2 of Reference 7-2 and Reference 7-3) was also used in the analysis of the CEA Ejection Event.

Simulation of the fluid conditions within the hot channel of the reactor core and calculation of DNBR was performed using the CETOP-D computer code (Section 6.1 of this report and References 7-7 and 7-8). The number of fuel pins predicted to experience DNB was calculated by the statistical convolution method described in Reference 7-4.

The TORC computer program is used to simulate the fluid conditions within the reactor core and to calculate fuel pin DNBR for the sheared shaft event. The TORC program is described in References 7-14 and 7-15.

Determination of DNBR for the post trip return to power portion of the steam piping failure events is based on the correlation developed by R. V. Macbeth (Reference 7-5) with corrections developed by Lee (Reference 7-6) to account for non-uniform axial heat flux. This methodology is consistent with that employed in the Reference Cycle analysis.

The HERMITE code (Reference 7-11) was used to simulate the reactor core for analyses which required more spatial detail than provided by a point kinetics model.

7.0.4 Input Parameters and Analysis Assumptions

Table 7.0-6 summarizes the core parameters assumed in the Cycle 3 transient analysis and compares them to the values used in the Reference Cycle. Specific initial conditions for each event are tabulated in the section of the report summarizing that event. For some of the DBEs presented, certain initial core parameters were assumed to be more limiting than the actual calculated Cycle 3 values (i.e., CEA worth at trip, moderator temperature coefficient). Those values and ranges used for the core parameters resulted from the Extended Cycles Program (ECP) (Reference 7-16) for SONGS Units 2 and 3. The data base generated for the future, extended burnup cycles yielded parameters and range that not only bound the Cycle 3 generated data, but also should be applicable to future cycles as well.

7.0.5 Conclusion

For all DBEs that have results bounded by the Reference Cycle, the margin of safety has not degraded from that of the Reference Cycle. Those events whose results were not bounded by the Reference Cycle are presented herein. All of these events have results within NRC acceptance criteria.

Table 7.0-1

SONGS Unit 2, Design Basis
Events Considered in the Cycle 3 Safety Analysis

7.1 Increase in Heat Removal by the Secondary System

- 7.1.1 Decrease in Feedwater Temperature
- 7.1.2 Increase in Feedwater Flow
- 7.1.3 Increased Main Steam Flow
- 7.1.4 Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve
- 7.1.5* Steam System Piping Failures

7.2 Decrease in Heat Removal by the Secondary System

- 7.2.1 Loss of External Load
- 7.2.2 Turbine Trip
- 7.2.3 Loss of Condenser Vacuum
- 7.2.4 Loss of Normal AC Power
- 7.2.5 Loss of Normal Feedwater
- 7.2.6* Feedwater System Pipe Breaks

7.3 Decrease in Reactor Coolant Flowrate

- 7.3.1 Partial Loss of Forced Reactor Coolant Flow
- 7.3.2 Total Loss of Forced Reactor Coolant Flow
- 7.3.3* Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

7.4 Reactivity and Power Distribution Anomalies

- 7.4.1 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition
- 7.4.2 Uncontrolled CEA Withdrawal at Power
- 7.4.3 CEA Misoperation Events
- 7.4.4 CVCS Malfunction (Inadvertent Boron Dilution)
- 7.4.5 Startup of an Inactive Reactor Coolant System Pump
- 7.4.6* Control Element Assembly Ejection

7.5 Increase in Reactor Coolant System Inventory

- 7.5.1 CVCS Malfunction
- 7.5.2 Inadvertent Operation of the ECCS During Power Operation

Table 7.0-1 (continued)

7.6 Decrease in Reactor Coolant System Inventory

- 7.6.1 Pressurizer Pressure Decrease Events
- 7.6.2* Small Primary Line Break Outside Containment
- 7.6.3* Steam Generator Tube Rupture

7.7 Miscellaneous

- 7.7.1 Asymmetric Steam Generator Events

* Categorized as Limiting Fault Events

Table 7.0-2

DBEs Evaluated with Respect to Offsite Dose Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.1.4	1) Inadvertent Opening of a Steam Generator Atmospheric Dump Valve or Safety Valve	Bounded by Reference Cycle
7.2.4	2) Loss of Normal AC Power	Bounded by Reference Cycle
	B) Limiting Fault Events	
	1) Steam System Piping Failures;	Presented
7.1.5a	a) Pre-Trip Power Excursions	
7.1.5b	b) Post Trip Analysis	
7.2.6	2) Feedwater System Pipe Breaks	Bounded by Reference Cycle
7.3.3	3) Single Reactor Coolant Pump Shaft Seizure	Presented
7.6.2	4) Small Primary Line Break Outside Containment	Bounded by Reference Cycle
7.6.3	5) Steam Generator Tube Rupture	Bounded by Reference Cycle

Table 7.0-3

DBEs Evaluated with Respect to RCS Pressure Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.2.1	1) Loss of External Load	Bounded by Reference Cycle
7.2.2	2) Turbine Trip	Bounded by Reference Cycle
7.2.3	3) Loss of Condenser Vacuum	Bounded by Reference Cycle
7.2.4	4) Loss of Normal AC Power	Bounded by Reference Cycle
7.2.5	5) Loss of Normal Feedwater	Bounded by Reference Cycle
7.4.1	6) Uncontrolled CEA Withdrawal from A Subcritical or Low Power Condition	Presented
7.4.2	7) Uncontrolled CEA Withdrawal at Power	Bounded by Reference Cycle
7.5.1	8) CVCS Malfunction	Bounded by Reference Cycle
7.5.2	9) Inadvertent Operation of ECCS During Power Operation	Bounded by Reference Cycle
	B) Limiting Fault Events	
7.2.6	1) Feedwater System Pipe Breaks	Presented

Table 7.0-4

DBEs Evaluated with Respect to Fuel Performance

Section	Event			Results
	A)	Anticipated Operational Occurrences		
7.1.1		1)	Decrease in Feedwater Temperature	Bounded by Reference Cycle
7.1.2		2)	Increase in Feedwater Flow	Bounded by Reference Cycle
7.1.3		3)	Increased Main Steam Flow	Presented
7.3.1		4)	Partial Loss of Forced Reactor Coolant Flow	Bounded by Reference Cycle
7.3.2		5)	Total Loss of Forced Reactor Coolant Flow	Presented*
7.4.1		6)	Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition	Presented
7.4.2		7)	Uncontrolled CEA Withdrawal at Power	Bounded by Reference Cycle
7.4.3		8)	CEA Misoperation Events	Bounded by Reference Cycle
7.6.1		9)	Pressurizer Pressure Decrease Events	Bounded by Reference Cycle
7.7.1		10)	Asymmetric Steam Generator Events	Presented*
	B)	Limiting Fault Events		
		1)	Steam System Piping Failures;	
7.1.5a			a) Pre-Trip Power Excursions	Presented
7.1.5b			b) Post Trip Analysis	Presented
7.3.3		2)	Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	Presented
7.4.6		3)	Control Element Assembly Ejection	Bounded by Reference Cycle

*The results of this event remain bounded by the Reference Cycle. The event is presented due to a change in analytical methodology.

Table 7.0-5

DBEs Evaluated with Respect to Shutdown Margin Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.1.4	1) Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.4.4	2) CVCS Malfunction (Inadvertent Boron Dilution)	Presented
	B) Limiting Fault Events	
7.1.5b	1) Steam System Piping Failure, Post Trip Analysis	Presented

Table 7.0-6

SONGS Unit 2, Cycle 3
Core Parameters Input to Safety Analyses

Safety Parameters	Units	Reference Cycle Values	Cycle 3 Values
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	3478	3478
Core Inlet Steady State Temperature	^oF	542 to 560 (70% power and above) 530 to 560 (below 70% power)	542 to 560 (70% power and above) 530 to 560 (below 70% power)
Steady State RCS Pressure	psia	2000 - 2300	2000 - 2300
Rated Reactor Coolant Flow	gpm	396,000 to 410,000	396,000 to 410,000
Axial Shape Index LCO Band Assumed for All Powers	ASI Units	-.3 to +.3	-.3 to +.3
Maximum CEA Insertion at Full Power	% Insertion of Lead Bank	28	28
	% Insertion of Part-Length	25	25
Maximum Initial Linear Heat Rate for Transient	KW/ft	13.9	13.9
Steady State Linear Heat Rate for Fuel CTM Assumed in the Safety Analysis	KW/ft	21.0	21.0
CEA Drop Time from Removal of Power to Holding Coils to 90% Insertion	sec	3.0	3.0
Minimum DNBR			
CE-1		1.31	1.31
Macbeth		1.30	1.30

Table 7.0-6 (continued)

Safety Parameters	Units	Reference Cycle Values (Cycle 2)	Cycle 3 Values
Moderator Temperature Coefficient	$10^{-4}\ \Delta\rho/^{o}F$	-2.5 to +.5 (below 70% power) -2.5 to 0.0 (70% power and above)	-3.3 to +.5 (below 70% power) -3.3 to 0.0 (70% power and above)
Shutdown Margin (Value Assumed in Limiting EOC Zero Power SLB)	$\%\Delta\rho$	-5.15	-5.15

7.1 Increase in Heat Removal by the Secondary System

7.1.1 Decrease in Feedwater Temperature

The results are bounded by the Reference Cycle.

7.1.2 Increase in Feedwater Flow

The results are bounded by the Reference Cycle.

7.1.3 Increased Main Steam Flow

The Increased Main Steam Flow Event is analyzed to ensure that the Departure from Nucleate Boiling Ratio (DNBR) and Fuel Centerline Melt (CTM) Specified Acceptable Fuel Design Limits (SAFDLs) are not violated. This event was reanalyzed due to a more adverse pin census and an increased Doppler multiplier, and the availability of the Variable Overpower Trip (VOPT).

7.1.3.1 Identification of Causes

An Increased Main Steam Flow Event is defined as any rapid increase in steam generator steam flow other than a steam line rupture (discussed in Section 7.1.5) or an inadvertent opening of a secondary safety valve (discussed in Section 7.1.4). Such rapid increases in steam flow result in a power mismatch between core power and steam generator load demand. Consequently, there is a decrease in reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient of reactivity, the decrease in reactor coolant temperature causes an increase in core power.

The High Power Level and Core Protection Calculators (CPCs) trips provide primary protection during this event. Additional protection is provided by other trip signals including Low Steam Generator Water Level and Low Steam Generator Pressure. The approach to the CTM limit is terminated by either the DNB/Local Power Density (LPD) related trip, the Variable Overpower Trip (VOPT) or the High Power Level Trip. In this analysis, credit is taken only for the action of the CPC Low DNBR Trip or the VOPT in the determination of the minimum transient DNBR and maximum local linear heat generation rate. The Variable Overpower Trip is described in Reference 7-17.

The following Increased Main Steam Flow Events have been examined:

A. An inadvertent increased opening of the turbine admission valves caused by operator error or turbine load limit malfunction. This can result in an additional 10% flow.

B. Failure in the turbine bypass control system which would result in an opening of one or more of the turbine bypass valves. The flowrate of each valve is approximately 11% of the full power turbine flowrate. There are four turbine bypass valves for a total of 45% at full power steam flow.

C. An inadvertent opening of an atmospheric dump valve or steam generator safety valve (see Section 7.1.4) caused by operator error or failure within the valve itself. Each atmospheric dump and safety valve can release approximately 5% of the full power turbine flowrate.

7.1.3.2 Analysis of Effects and Consequences

As in the Reference Cycle analysis (Reference 7-1), the opening of the four steam bypass valves at HFP produces the most adverse results. The opening of the four bypass valves at full power was initiated at the conditions given in Table 7.1.3-1. A moderator temperature coefficient (MTC) of $-3.3 \times 10^{-4} \Delta\rho /^{o}F$ was used in the analysis. This MTC, in conjunction with the decreasing coolant inlet temperature, results in an increase in the core heat flux. The most negative fuel temperature coefficient (FTC) with a bias of 25%, was used in the analysis. The minimum CEA worth for shutdown at the time of reactor trip for full power operation is $-6.0\% \Delta\rho$. The pressurizer pressure control system was assumed to be inoperable to minimize the RCS pressure during the event and reduce the calculated DNBR. All other control systems were assumed to be in manual mode of operation and have no significant impact on the results for this event. The Reference Cycle cited a coincident loss of AC power as the limiting single failure for this event. The loss of AC power and subsequent reactor coolant pump coastdown occurs such that a coincident CPC Low Flow/VOPT occurs. This timing maximizes both the degradation in DNBR and the quantity of predicted fuel failure.

7.1.3.3 Results

The Increased Main Steam Flow Event plus a single failure (loss of AC power) resulted in a CPC VOPT Trip/Low Flow Trip at 9.75 seconds. The minimum DNBR calculated for the event initiated from the conditions specified in Table 7.1.3-1 was 1.16 compared to the design limit of 1.31. This corresponds to a calculated fuel pin failure of less than 8%. A maximum allowable initial linear heat generation rate of 16.0 kW/ft could exist as an initial condition without exceeding the Acceptable Fuel to Centerline Melt Limit of 21.0 kW/ft during this transient. This amount of margin is assured by setting the linear heat rate LCO based on the more limiting allowable linear heat rate for LOCA (13.9 kW/ft, see Table 7.0-6).

NSSS cooldown is two hours in duration resulting in offsite doses of less than 300 REM thyroid and a whole body dose of less than 25 REM. These results are more limiting than those presented in the Reference Cycle for Increased Main Steam Flow Events with a single failure.

Table 7.1.3-2 presents the sequence of events for the event initiated at HFP conditions. Figures 7.1.3-1 to 7.1.3-5 present the NSSS response of core power, core heat flux, RCS pressure, RCS temperatures and steam generator pressure. The DNBR response for Cycle 3 as a function of time is presented in Figure 7.1.3-6.

The results of the Increased Main Steam Flow without a single failure would be no more adverse than those presented in the Reference Cycle.

7.1.3.4 Conclusions

For the Increased Main Steam Flow Events with a single failure, the radiological doses are less than the 10CFR100 limits of 300 REM for thyroid and 25 REM for whole body. For the Increased Main Steam Flow Event without a single failure, the DNBR and CTM limits are not exceeded.

7.1.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

The results are bounded by the Reference Cycle.

Table 7.1.3-1

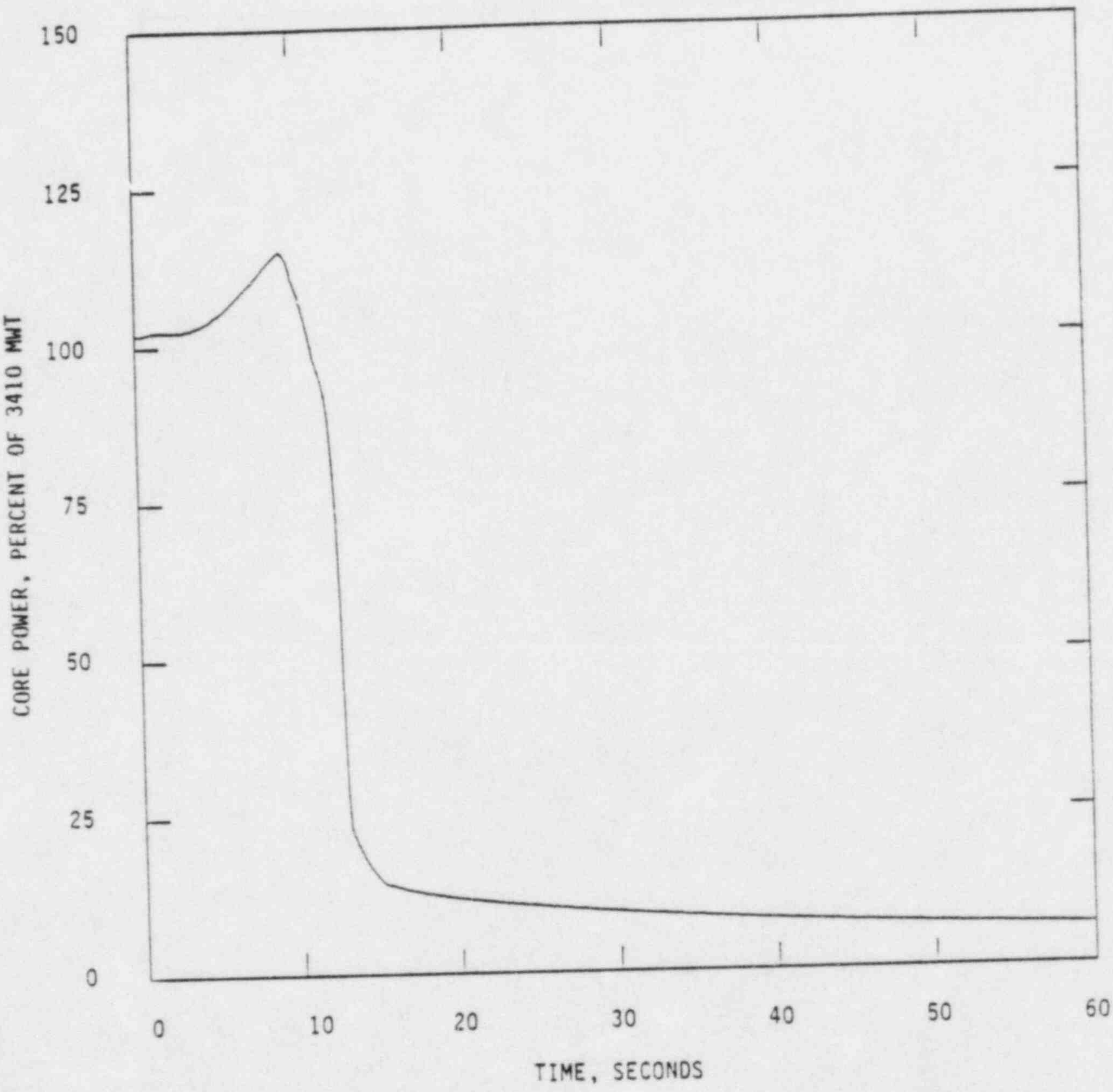
Key Parameters Assumed for the Increased Main Steam Flow Event

Parameter	Units	Reference Cycle Value	Cycle 3 Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	3478	3478
Initial Core Coolant Inlet Temperature	^oF	560	560
Initial Reactor Coolant System Pressure	psia	2200	2200
Initial RCS Vessel Flow Rate	gpm	396,000	396,000
Moderator Temperature Coefficient	$x10^{-4} \Delta\rho/^oF$	-3.3	-3.3
CEA Worth at Trip	$\%\Delta\rho$	-4.5	-6.0
Doppler Coefficient Multiplier		1.15	1.25

Table 7.1.3-2

Sequence of Events for the Increased Main Steam Flow Event Plus a Single Failure

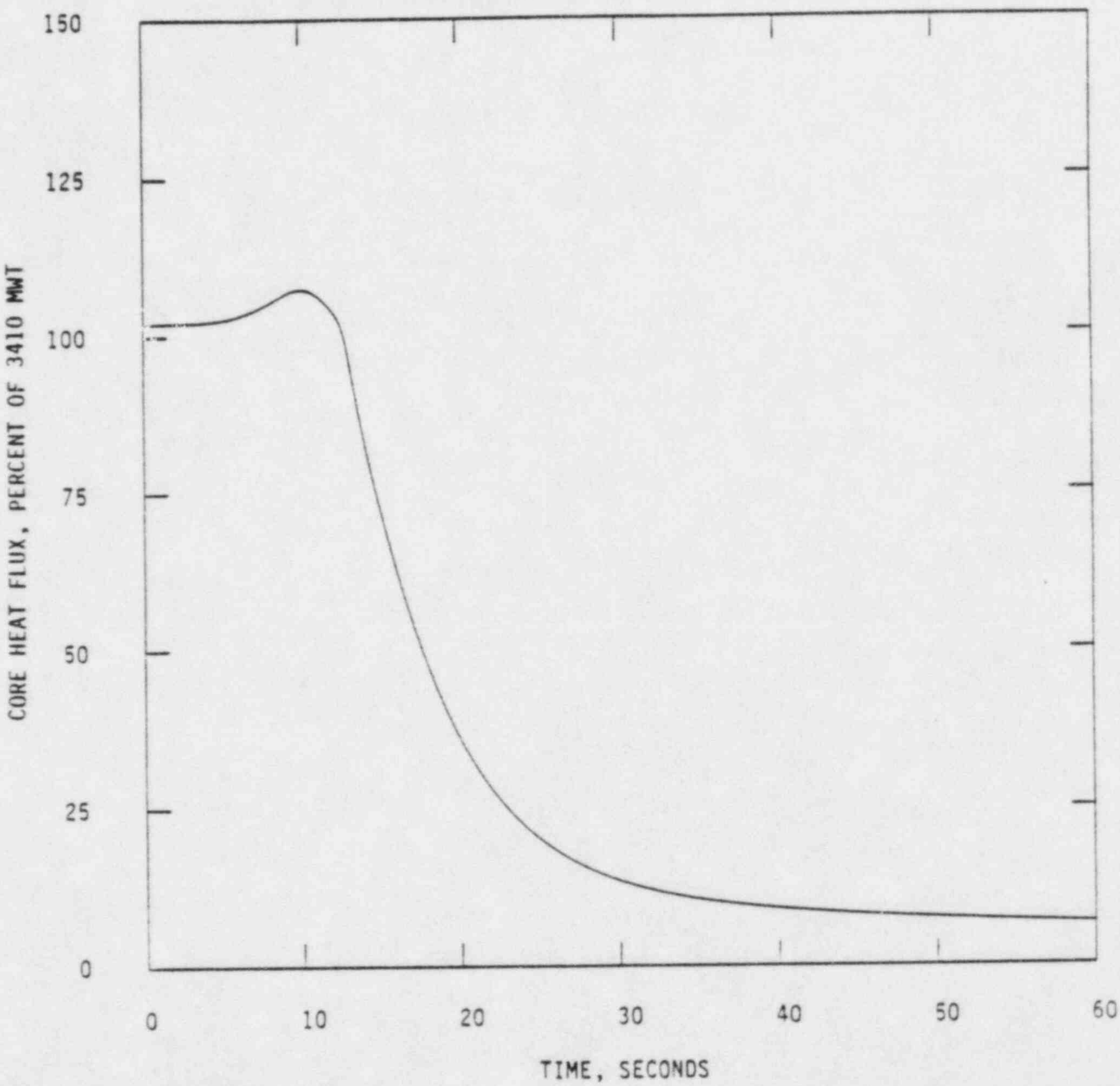
Time (sec)	Event	Setpoint or Value
0.0	Quick Open Signal Generated, Four Bypass Valves Start to Open	---
1.0	Four Bypass Valves Full Open	145% of full steam flow
8.95	Loss of All On and Offsite Power, Turbine Admission Valves and Bypass Valves Start to Close, Feedwater Begins to Coast Down, Reactor Coolant Pumps Begin to Coast Down	---
9.75	CPC VOPT Trip/Low Flow Signal Generated	116% of 3410 MWt, 95% of shaft speed
10.0	Reactor Trip Breakers Open, Turbine Trip	---
10.3	CEAs Begin to Drop in the Core	---
10.7	Maximum Core Power	117.6% of 3410 MWt
12.1	Maximum Core Heat Flux	110.6% of 3410 MWt
12.3	Minimum DNBR Occurs (CE-1)	1.16
12.75	Turbine Admission Valves and Bypass Valves Closed	---
16.7	Steam Generator Safety Valves Open	1100 psia
28.05	Feedwater Flow Reaches 5% of Full Power	



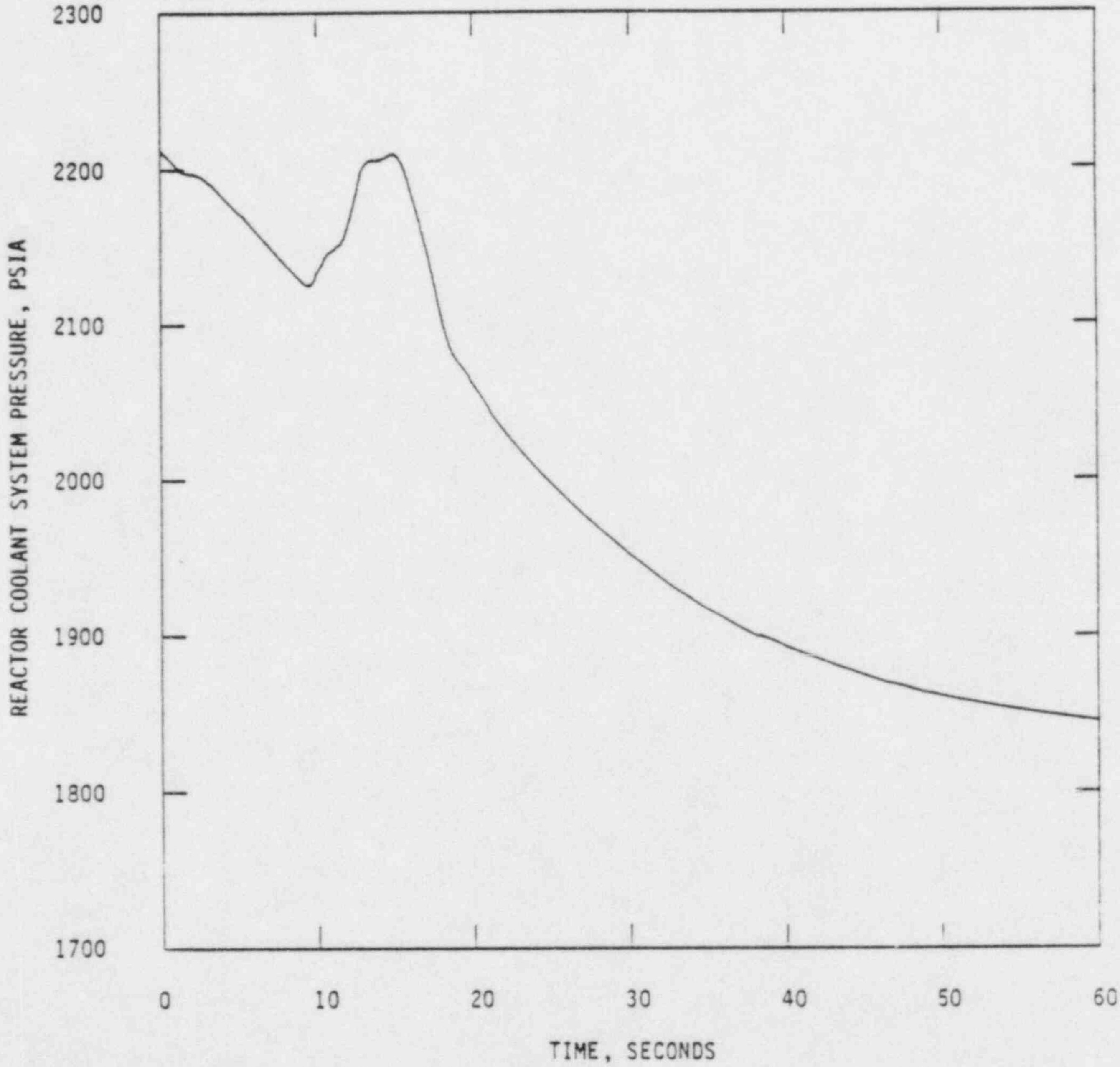
SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW
PLUS SINGLE FAILURE
CORE POWER VS TIME

FIGURE 7.1.3-1



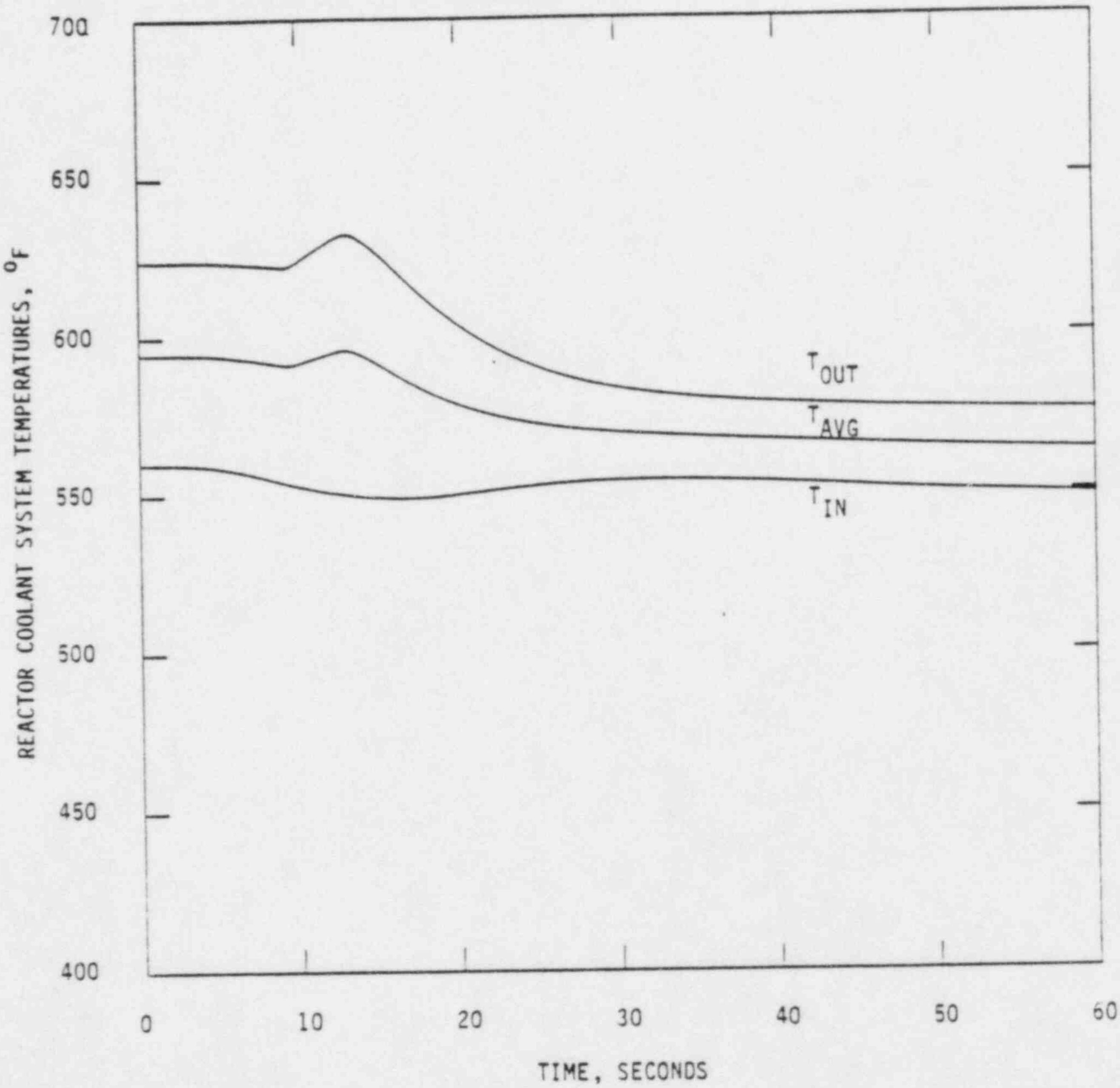
SAN ONOFRE NUCLEAR GENERATING STATION **Units 2 & 3**
INCREASED MAIN STEAM FLOW PLUS SINGLE FAILURE CORE HEAT FLUX VS TIME
FIGURE 7.1.3-2



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW
PLUS SINGLE FAILURE
RCS PRESSURE VS TIME

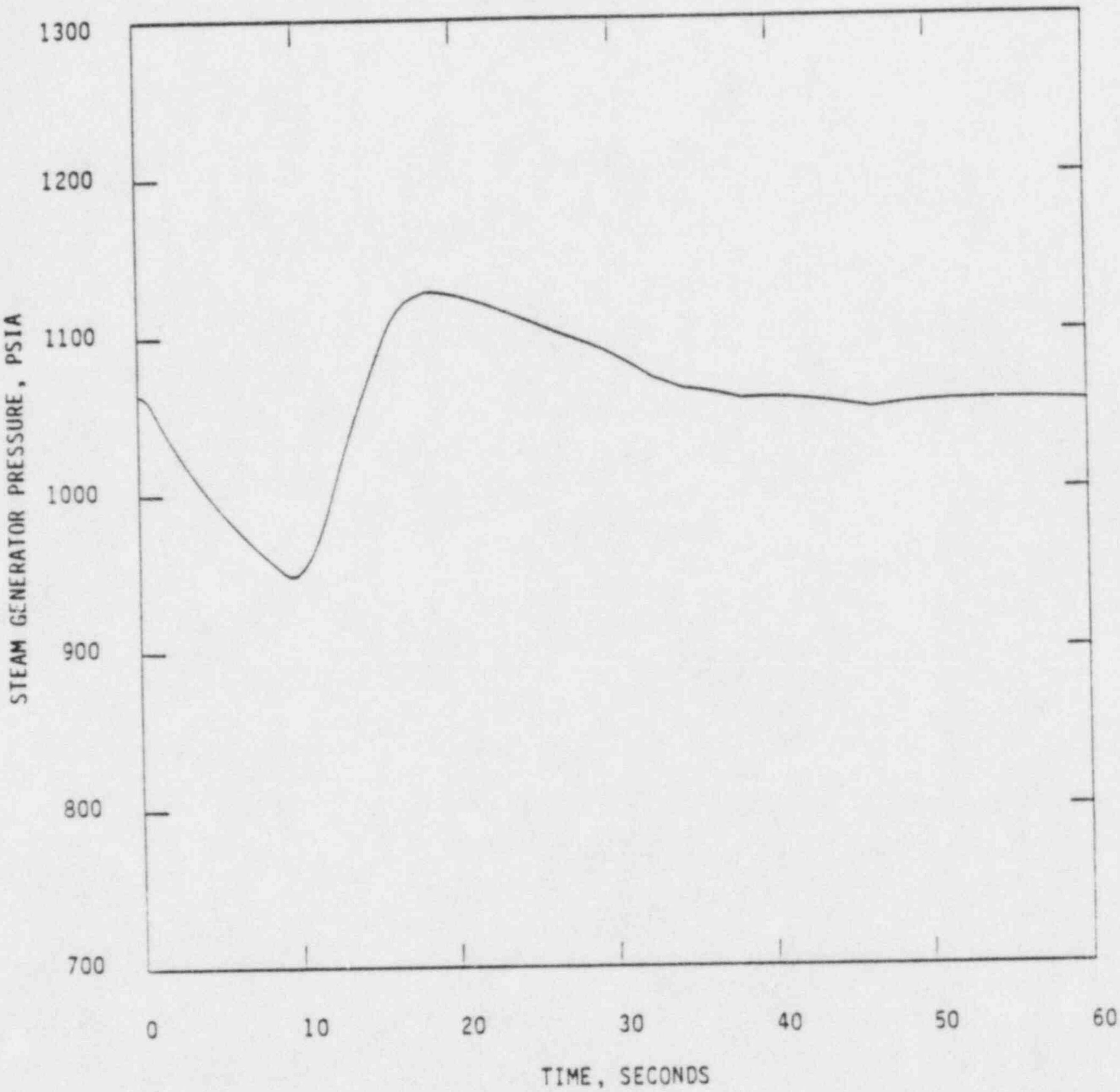
FIGURE 7.1.3-3



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW
PLUS SINGLE FAILURE
RCS TEMPERATURES VS TIME

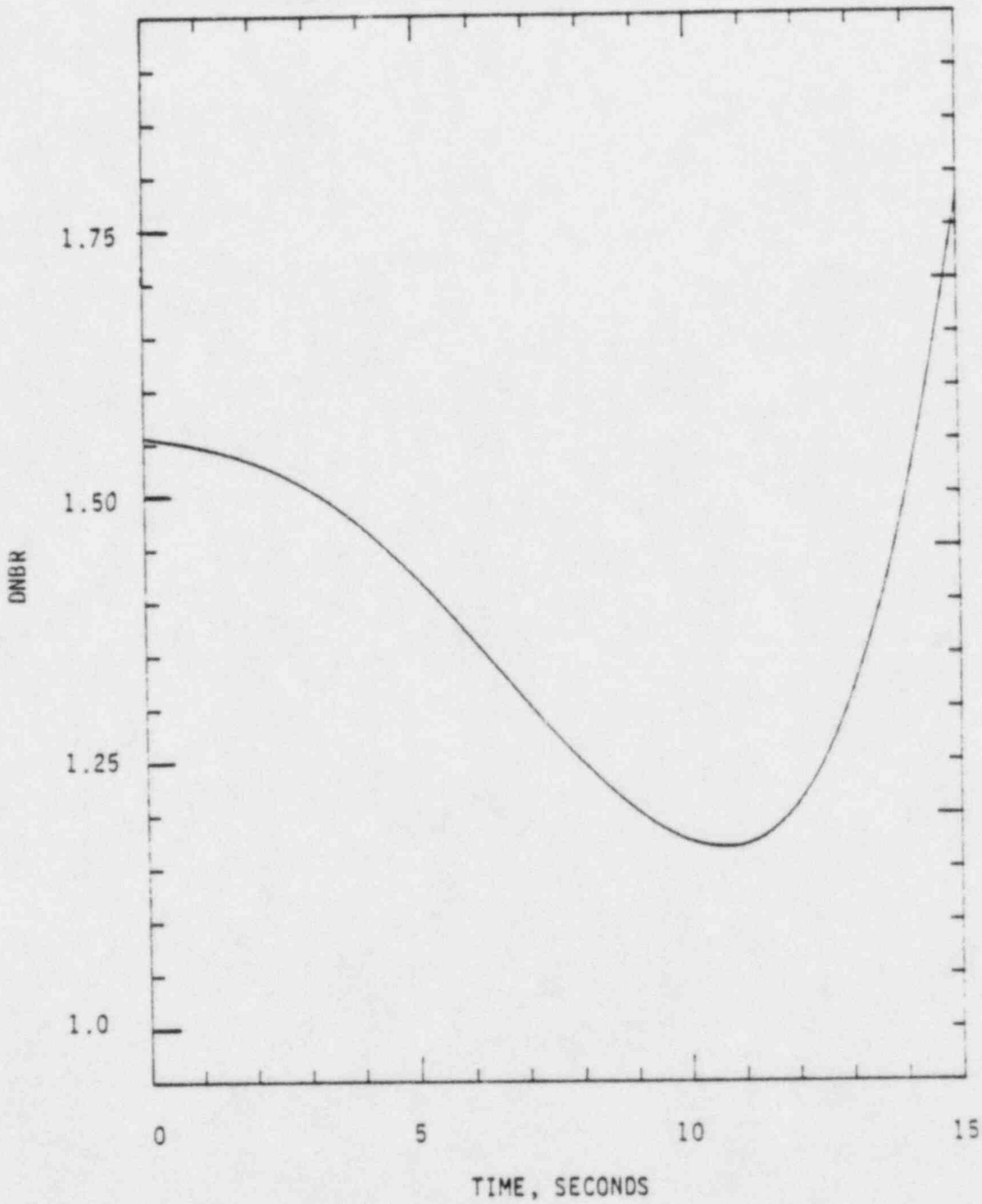
FIGURE 7.1.3-4



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW
PLUS SINGLE FAILURE
STEAM GENERATOR PRESSURE VS TIME

FIGURE 7.1.3-5



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
INCREASED MAIN STEAM FLOW PLUS SINGLE FAILURE DNBR VS TIME
FIGURE 7.1.3-6

7.1.5 Steam System Piping

Failures in the main steam system piping were analyzed to ensure that a coolable geometry is maintained and that the site boundary doses do not exceed 10CFR100 guidelines.

7.1.5a Steam System Piping Failures: Inside and Outside Containment Pre-Trip Power Excursions

This event was analyzed to evaluate the maximum number of calculated fuel pin failures for the site boundary dose calculation.

7.1.5a.1 Identification of Causes

A rupture in the main steam system piping increases steam flow from the steam generators. This increase in steam flow increases the rate of RCS heat removal by the steam generators and causes a decrease in core coolant inlet temperature. In the presence of a negative moderator temperature coefficient of reactivity (MTC), this decrease in temperature causes core power to increase.

The excursion in core power is terminated by the action of one of the following Reactor Protection System (RPS) trips: Core Protection Calculators (CPCs), Low Steam Generator Pressure (LSGP), High Linear Power Level, or High Containment Pressure.

7.1.5a.2 Analysis of Effects and Consequences

Steam Line Breaks (SLBs) inside containment may be postulated to have break areas up to the cross section of the largest main steam pipe ($7.41\ ft^2$). Those SLBs occurring outside the containment building have break areas limited by the areas of the flow restrictors ($4.13\ ft^2$) which are located upstream of the containment penetrations.

Inside containment SLBs may cause environmental degradation of sensor input to the CPCs and pressure measurement systems. Additionally, the higher linear power level trip undergoes temperature decalibration due to RCS cooldown. No credit is taken for CPC action during this event. Trips which are credited for inside containment SLBs are: LSGP, High Linear Power Level or High Containment Pressure. Additionally, the environmentally degraded value of the Delta Pressure Low Flow trip is used to determine the most adverse timing of a Loss of AC Power (LOAC).

Outside containment SLBs are not subject to the same environmental effects as the inside containment breaks. Therefore, the full array of RPS trips including the CPC Low DNBR trip, are credited for these breaks.

In the Reference Cycle, an extensive parametric analysis in both MTC and break area was performed on the inside containment SLB event. This parametric analysis identified the limiting inside containment SLB event in terms of fuel pin failure caused by the pre-trip power excursion. Table 7.1.5a-1 of the Reference Cycle (Reference 7-1) lists the values of key parameters used in the parametric analysis.

The inside containment SLB event was reanalyzed in Cycle 3 to accommodate a more adverse pin census changes in other Key Parameters for Cycle 3 are within the ranges used for the Reference Cycle Parametric Study. The Reference Cycle results (heat flux, RCS temperatures, pressure and flow rate) were combined with the pin census to yield a value for predicted fuel failure.

7.1.5a.3 Results

The outside containment SLBs are bounded by the Reference Cycle, since they are subject to a rapid RPS trip on Low DNBR. This trip provides timely termination of the power excursion preventing the fuel design limits from being exceeded. The radiation release accompanying these outside containment breaks are less severe than the outside containment Double Ended Guillotine Break examined in Section 7.1.5b for the post-trip return to power.

Based on the transient response of the Reference Cycle parametric for the limiting break, the number of calculated fuel pin failures for the inside containment SLB event is less than 8%.

The inside containment SLB event resulted in site boundary doses less than 300 REM to the thyroid and less than 25 REM whole body.

7.1.5a.4 Conclusions

The results of this analysis demonstrate that a coolable geometry is maintained during this event as the number of fuel pins calculated to fail is less than 8 percent. Site boundary doses are calculated to be less than the 10CFR100 guidelines.

7.1.5b Steam System Piping Failure, Post-Trip Return to Power

The Hot Full Power (HFP) Steam Line Break (SLB) Event was reanalyzed due to a more adverse moderator cooldown curve and an increase in maximum inverse boron worth. The HFP SLB with Loss of AC (LOAC) power was reanalyzed to ensure that a coolable geometry is maintained and that the site boundary doses do not exceed 10CFR100 guidelines.

7.1.5b.1 Identification of Causes

A break in the main steam system piping will cause an increase in steam flow. This increase in flow results in increased heat removal from the Reactor Coolant System (RCS). In the presence of a negative moderator temperature coefficient of reactivity (MTC) the cooldown will cause positive reactivity to be added to the core. Highly negative MTCs and large break sizes can combine to degrade shutdown margin and may cause a return-to-power.

This approach to criticality is terminated by the addition of safety injection boron and the increase in temperature following either,

1. Termination of steam flow and heat removal by the action of the MSIVs in both steam lines,

or

2. Termination of steam flow from the unaffected steam generator by the MSIV action and dryout of the affected steam generator.

The Hot Full Power (HFP) and Hot Zero Power (HZP) Steam Line Break (SLB) Events were analyzed to determine that critical heat fluxes are not exceeded during this event and site boundary doses do not exceed 10CFR100 guidelines.

7.1.5b.2 Analysis of Effects and Consequences

The analytical basis for the HFP simulation are discussed below.

A. A Double-Ended Guillotine break (7.41 ft^2) causes the greatest cooldown of the RCS and the most severe degradation of shutdown margin.

B. A break inside the containment building, upstream of the MSIVs causes a non-isolatable condition in the affected steam generator. This results in continued shutdown margin degradation until the affected steam generator blows dry.

C. A reactor trip is initiated by either Low Steam Generator Pressure, Low Steam Generator Water Level, High Linear Power Level, Low DNBR, or Delta-Pressure Low Flow Trip (Loss of AC Power).

D. The cooldown following a steam line break results in contraction of the reactor coolant. For this analysis, if the pressurizer empties, the reactor coolant pressure is set equal to the saturation pressure corresponding to the highest temperature in the reactor coolant system.

E. A safety injection actuation signal (SIAS) is actuated when the pressurizer pressure drops below the setpoint. Time delays associated with the safety injection pump acceleration, valve opening, and flushing of the unborated

safety injection lines are taken into account. Additionally, the event was initiated from the highest pressure allowed by the technical specifications to delay the effect of safety injection boron.

F. The cooldown of the RCS is terminated when the affected steam generator blows dry. As the coolant temperatures begin increasing, positive reactivity insertion from moderator reactivity feedback decreases. The decrease in moderator reactivity combined with the negative reactivity inserted via boron injection cause the total reactivity to become more negative.

The conservative assumptions included in the HFP simulation are discussed below.

The Moderator Temperature Coefficient (MTC) of reactivity assumed in the analysis corresponds to the most negative value allowed by the Technical Specifications. This negative MTC results in the greatest positive reactivity addition during the RCS cooldown caused by the steam line break. Since the reactivity change associated with moderator feedback varies significantly over the range of moderator density covered in the analysis, a curve of reactivity insertion versus moderator density rather than a single value of MTC is assumed in the analysis. The moderator cooldown curve used in the analysis was conservatively calculated assuming that on reactor trip, the highest worth control element assembly is stuck in the fully withdrawn position.

The reactivity defect associated with fuel temperature decrease is also based on a most negative Fuel Temperature Coefficient (FTC). This FTC, in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the steam line break event. The bias on the FTC assumed in the analysis is given in Table 7.1.5b-1. The delayed neutron fraction assumed is the maximum absolute value including uncertainties for end-of-life conditions. This too maximizes subcritical multiplication and thus enhances the potential for Return-to-Power (R-T-P).

The minimum CEA worth assumed to be available for shutdown at the time of reactor trip at the maximum allowed power level is -8.28%$\Delta\rho$. This available scram worth corresponds to the moderator cooldown curve and stuck rod worth used in the analysis.

During the return-to-power, negative reactivity credit was assumed in the analysis. This negative reactivity credit is due to the local heatup of the inlet fluid in the hot channel, which occurs near the location of the stuck CEA. This credit is based on three-dimensional coupled neutronic-thermal-hydraulic calculations performed with the HERMITE/TORC code (References 7-11 and 7-12) for Calvert Cliffs Unit 1 Cycle 7 (Reference 7-13). Only a fraction of the negative reactivity credit justified for Calvert Cliffs Unit 1 Cycle 7 was used.

The analysis assumed that, on a safety injection actuation signal, one high pressure safety injection pump fails to start. A maximum inverse boron worth of 110 ppm/%$\Delta\rho$ was conservatively assumed for safety injection. A conservative MSIV closure time of 10.0 seconds was assumed in this analysis.

7.1.5b.3 Results

The Hot Zero Power SLB events are not presented since the Reference Cycle results bound Cycle 3 for radiological releases and post-trip criticality. The Hot Full Power SLB with no Loss of AC results are bounded by the Hot Full Power SLB with concurrent LOAC presented herein.

Table 7.1.5b-2 presents the sequence of events for the HFP SLB with concurrent LOAC. The key plant parameters of core power, core heat flux, RCS pressure, RCS temperatures, steam generator pressure and reactivity are shown in Figures 7.1.5b-2 through 7.1.5b-7.

The minimum post-trip DNBR experienced during the transient was 1.36 using the Macbeth low flow DNBR correlation. This value results in no calculated fuel failure during the course of this transient.

7.1.5b.4 Conclusions

The results of this analysis demonstrate that since there is no calculated fuel failure, a coolable geometry is maintained, and the Cycle 3 radiological release is bounded by the Reference Cycle. In addition, since the return-to-power is negligible, sufficient shutdown margin exists to terminate the event.

Table 7.1.5b-1

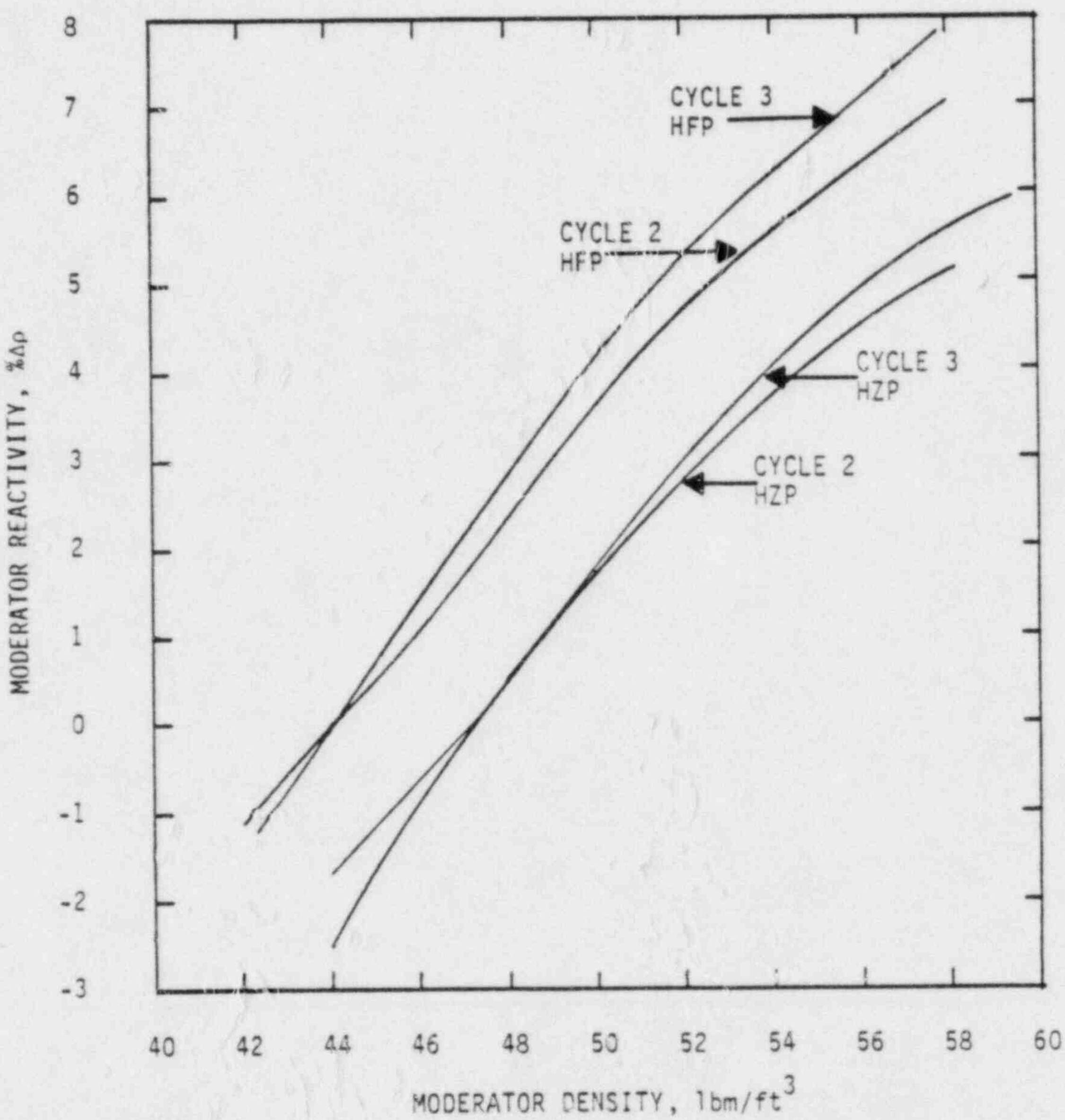
Key Parameters Assumed for the Steam Line Break Event

Parameter & Units	Reference Cycle Value	Cycle 3 Value
Hot Full Power		
Total RCS Power, MWt (Core Thermal Power + Pump Heat)	3478	3478
Initial Core Coolant Inlet Temperature, ^{o}F	560	560
Initial RCS Vessel Flow Rate, GPM	356,400	356,400
Initial Reactor Coolant System Pressure psia	2300	2300
Doppler Coefficient Multiplier	1.15	1.15
Moderator Temperature Coefficient, 10^{-4} $\Delta\rho/^{o}F$	-2.5	-3.3
CEA Worth at Trip, %$\Delta\rho$	-6.9	-8.28
Inverse Boron Worth, ppm/%$\Delta\rho$	95	110
Initial Steam Generator Pressure, psia	976	976
Steam Bypass Control System	Inoperable	Inoperable
Pressurizer Pressure Control System	Inoperable	Inoperable
High Pressure Safety Injection Pumps	One Pump Inoperable	One Pump Inoperable
Break Area, ft^2	7.41	7.41
Moderator Cooldown Curve	Figure 7.1.5b-1	Figure 7.1.5b-1

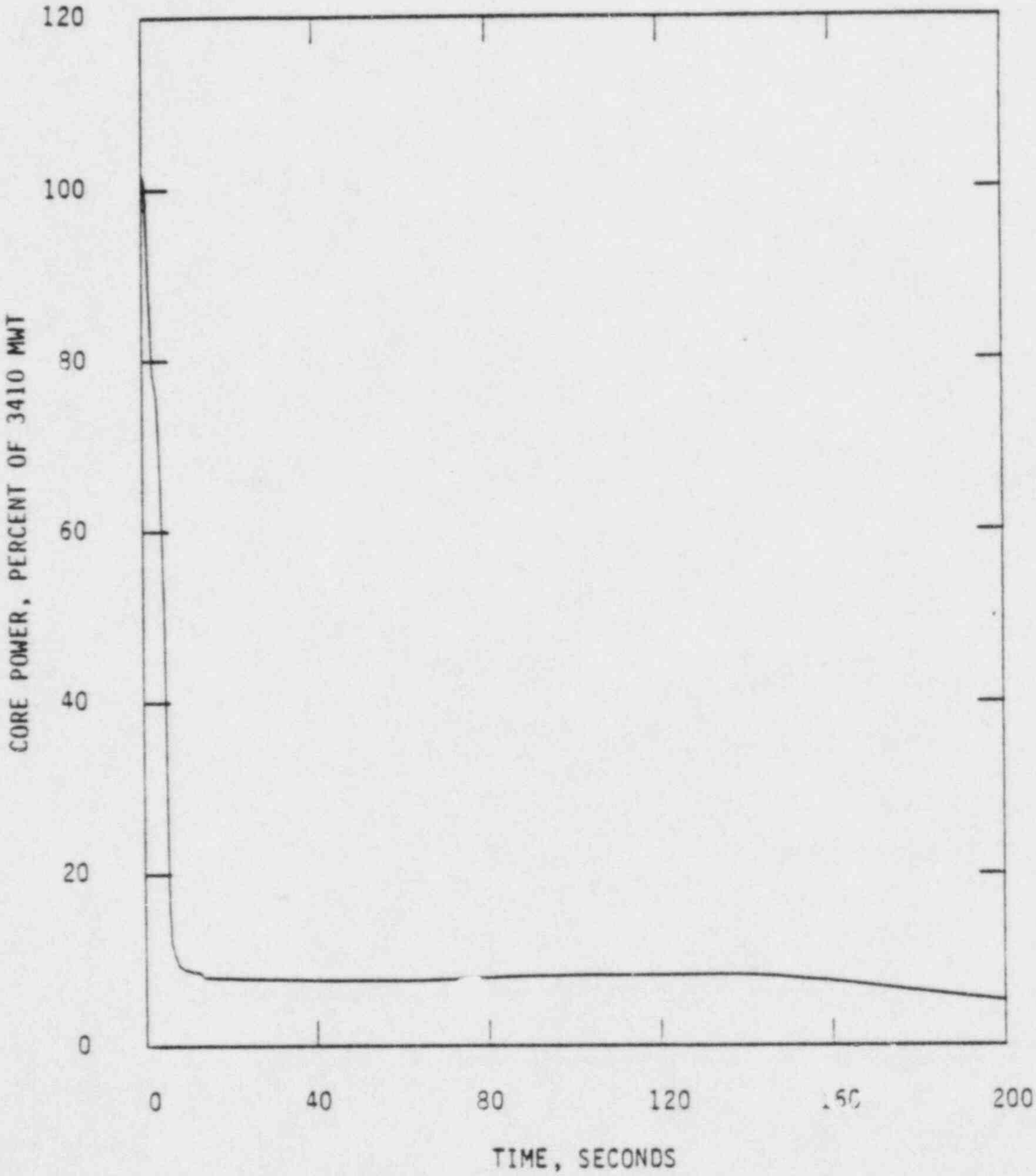
Table 7.1.5b-2

Sequence of Events for the Hot Full Power, 7.41 ft^2, Inside Containment Steam Line Break with Loss of Offsite Power

Time (sec)	Event	Setpoint or Value
0.0	Double-Ended Guillotine Break in a Main Steam Line with Concurrent LOAC, Reactor Coolant Pumps Begin to Coast Down	7.41 ft^2
2.4	Reactor Trip Signal Generated on Low Steam Generator Pressure, Main Steam Isolation Signal	675 psia
2.8	Trip Breakers Open	----
3.1	CEAs Begin to Drop	----
3.3	MSIVs Begin to Close	----
13.3	MSIVs are Completely Closed	----
17.3	Pressurizer Empties	----
17.7	Safety Injection Actuation Signal Generated on Low Pressurizer Pressure	1560 psia
48.9	Safety Injection Pumps Reach Full Speed	----
109.8	Affected Steam Generator Empties	----
132.6	Maximum Post-Trip Power	8.3% of 3410 MWt
139.1	Minimum Post-Trip McBeth DNBR	>1.30
141.2	Maximum Post-Trip Reactivity	-.078% $\Delta\rho$
1800.0	Plant Cooldown Initiated by Manual Control of the Atmospheric Steam Dump Valves for the Intact Steam Generator	----



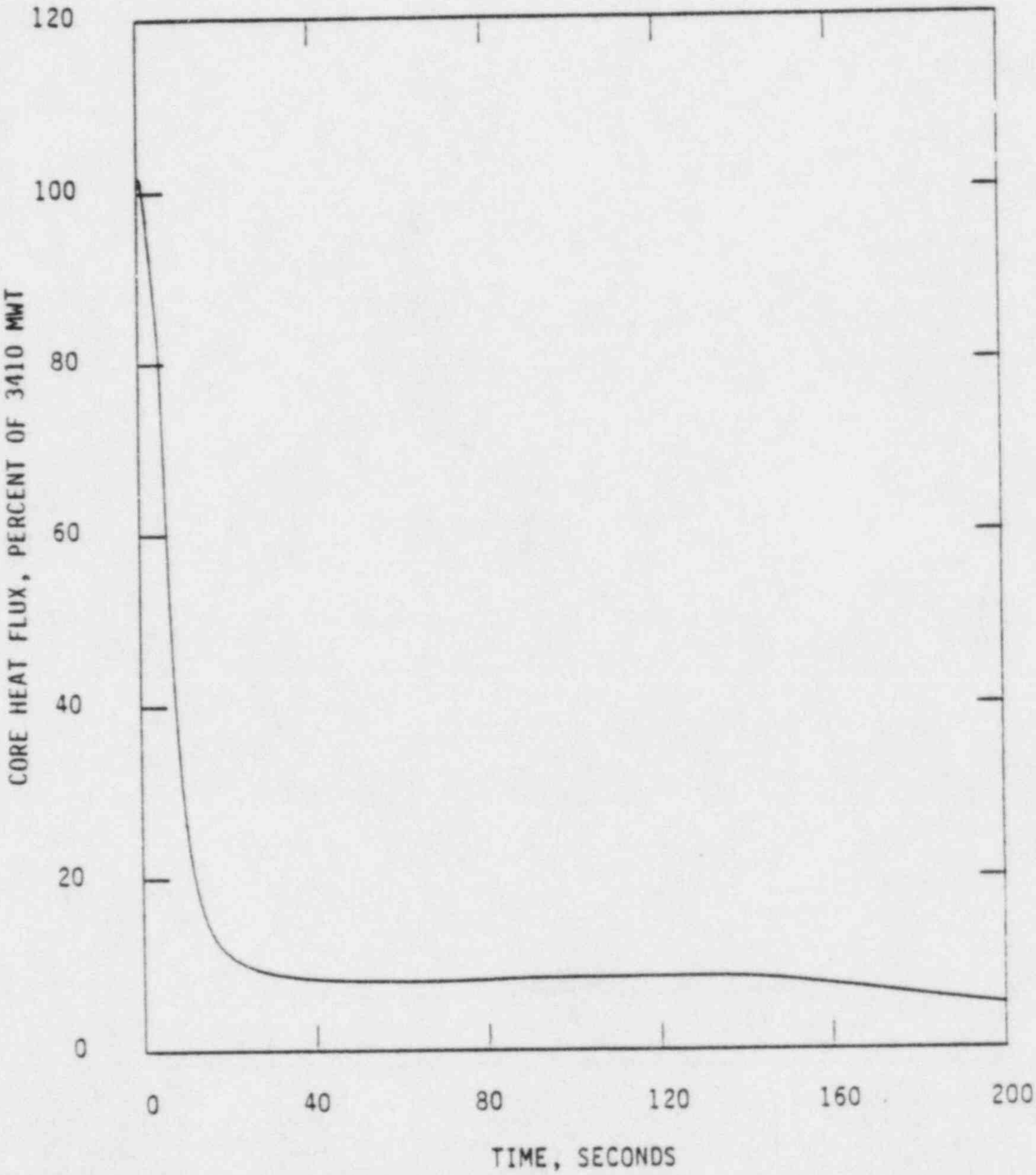
SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
STEAM LINE BREAK COMPARATIVE MODERATOR REACTIVITIES
FIGURE 7.1.5b-1



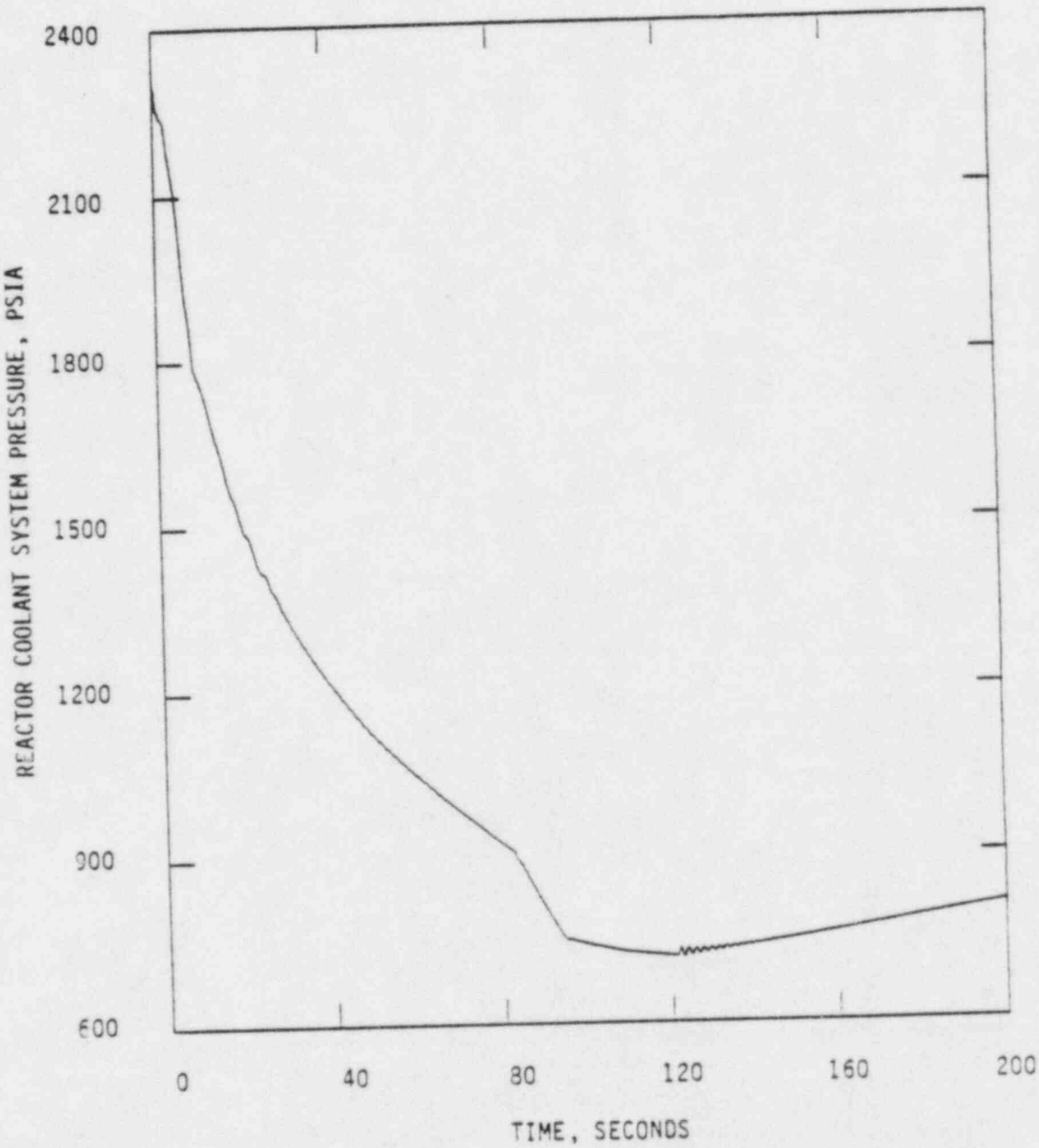
SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
WITH LOSS OF AC POWER
CORE POWER VS TIME

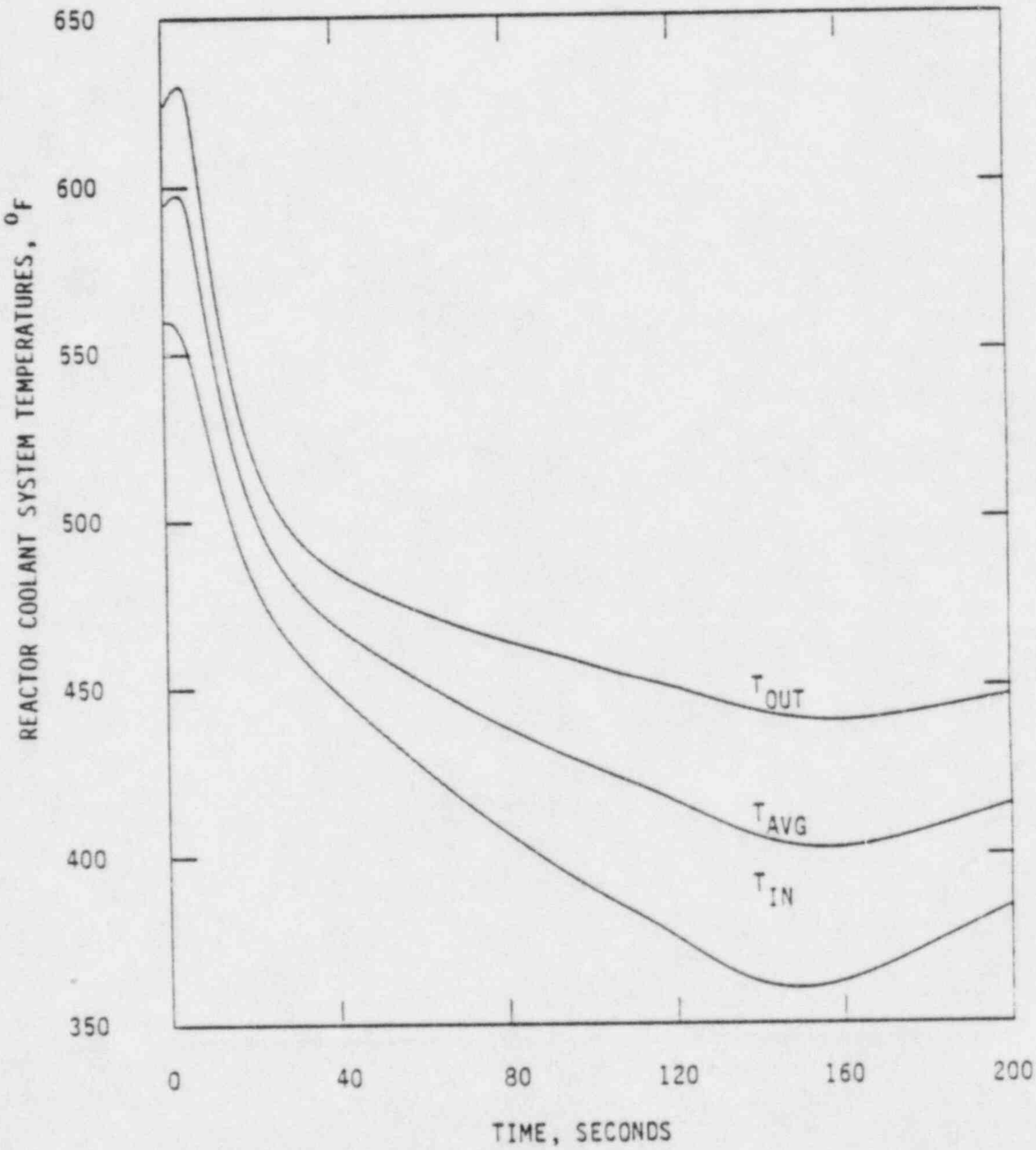
FIGURE 7.1.5b-2



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER CORE HEAT FLUX VS TIME
FIGURE 7.1.5b-3



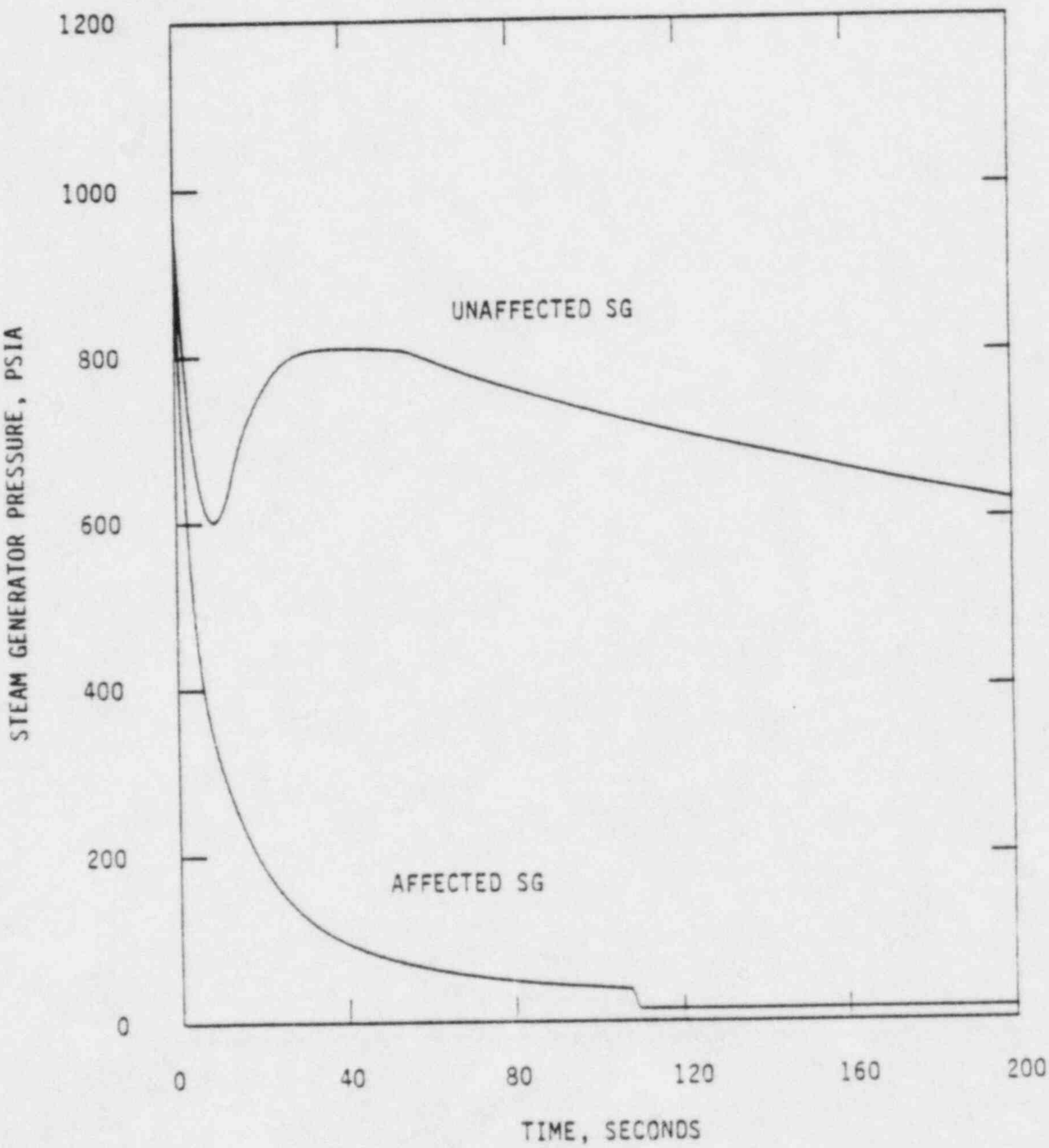
SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER RCS PRESSURE VS TIME
FIGURE 7.1.5b-4



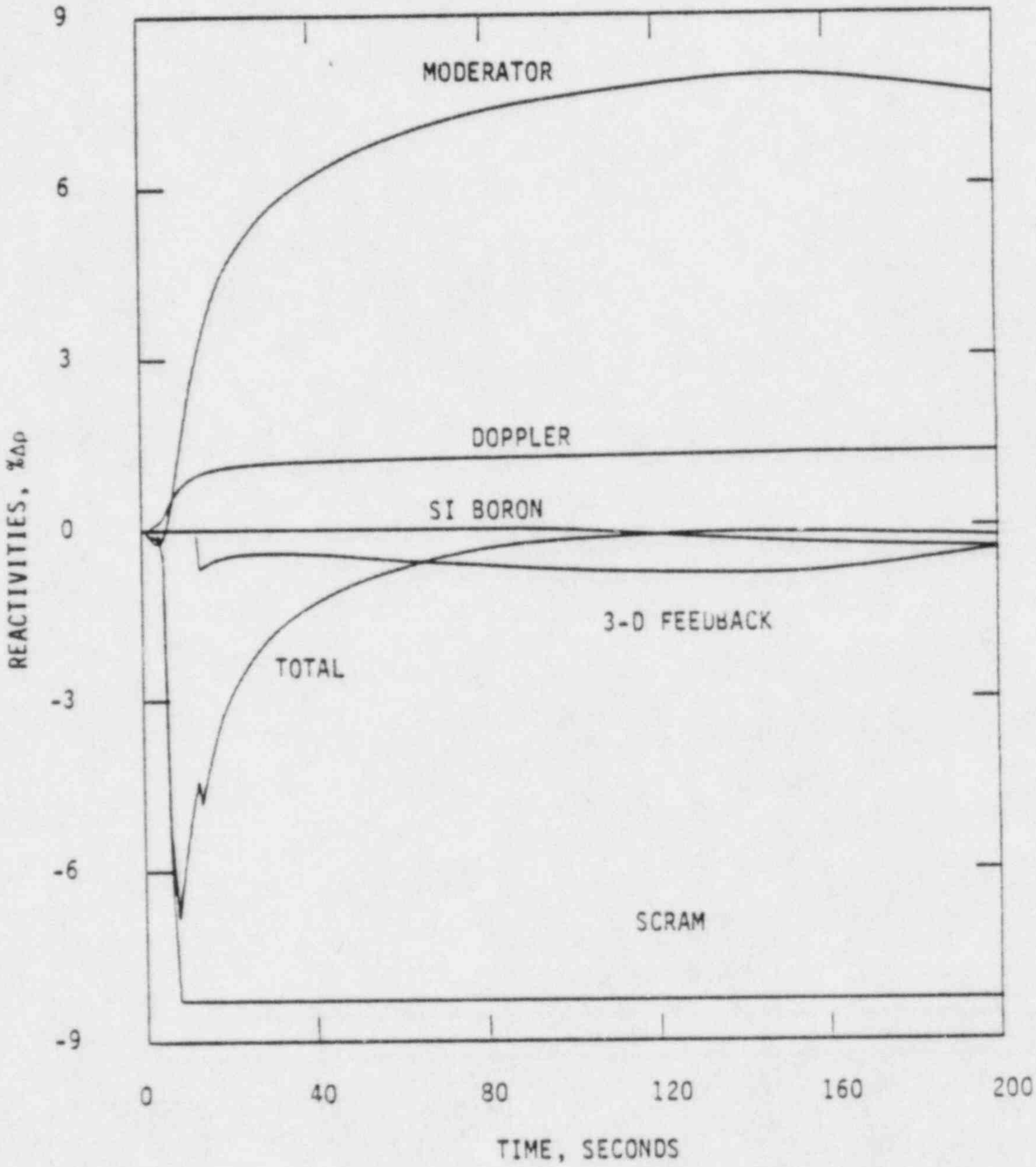
SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
WITH LOSS OF AC POWER
RCS TEMPERATURES VS TIME

FIGURE 7.1.5b-5



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER STEAM GENERATOR PRESSURE VS TIME
FIGURE 7.1.5b-6



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER REACTIVITIES VS TIME
FIGURE 7.1.5b-7

7.2 Decrease in Heat Removal by the Secondary System

7.2.1 Loss of External Load

The results are bounded by the Reference Cycle.

7.2.2 Turbine Trip

The results are bounded by the Reference Cycle.

7.2.3 Loss of Condenser Vacuum

The results are bounded by the Reference Cycle.

7.2.4 Loss of Normal AC Power

The results are bounded by the Reference Cycle.

7.2.5 Loss of Normal Feedwater

The results are bounded by the Reference Cycle.

7.2.6 Feedwater System Pipe Break Event

The feedwater system pipe break event is analyzed for Cycle 3 to demonstrate that the RCS pressure faulted stress limit of 3000 psia is not exceeded during the transient. This event was reanalyzed on the basis of an assumed increase in the number of plugged steam generator tubes and a change in the Doppler multiplier.

7.2.6.1 Identification of Causes

The rupture of a feedline will cause rapid reduction of the liquid inventory in the affected steam generator and therefore partial loss of the secondary heat sink. This leads to the heatup of the RCS and an increase in primary pressure. Depending on initial conditions, break size, break locations and steam generator inventory, any of the several Plant Protective System (PPS) actions may occur. A decrease in the steam generator water level will initiate a reactor trip on low steam generator water level. The decrease in the steam generator pressure may result in a low steam generator pressure trip signal and cause the main steam isolation valves and the main feedwater isolation valves to close. The partial loss of the secondary heat sink causes the RCS to heat up. This may result in a high pressurizer pressure trip. Additional protection against complete loss of secondary heat sink is provided by automatic initiation of emergency feedwater to the intact steam generator.

7.2.6.2 Analysis of Effects and Consequences

The feedwater line break analyzed was assumed to occur during full power operation with concurrent loss of non-emergency AC power at time of trip. This is limiting from the standpoint of potential RCS pressure increase, since this results in the maximum initial stored energy and minimum steam generator inventory. In addition, in response to loss of non-emergency AC power upon trip, the following were assumed to occur to maximize the RCS pressure increase:

1. Turbine stop valves close immediately:

2. Reactor coolant pumps begin to coastdown; and

3. Pressurizer control systems are lost.

The limiting break size was established by the parametric study reported in the FSAR. The initial RCS pressure and initial steam generator inventory are selected such that the low steam generator water level trip and the high pressurizer pressure trip occur simultaneously. This results in the maximum peak RCS pressure after trip. A MSIV closure time of 10.0 seconds is conservatively assumed for this analysis.

7.2.6.3 Results

The feedwater line break event was initiated at the conditions shown in Table 7.2.6-1. This combination of parameters maximizes the calculated RCS peak pressure. Table 7.2.6-2 presents the sequence of events for this event. Figures 7.2.6-1 through 7.2.6-6 present the NSSS response for core power, core heat flux, RCS temperatures, RCS pressure, pressurizer pressure, and steam generator pressure.

The results indicate that the reduction of the secondary heat sink due to the discharging of saturated water through the feedwater line break and the subsequent emptying of the affected steam generator cause the RCS pressure to increase to 2943 psia compared to the Reference Cycle reported value of 2930 psia. Following reactor trip on high pressurizer pressure/low steam generator water level, the decay in core power and the action of the primary and secondary safety valves result in a reduction of the RCS pressure. The RCS pressure continues to decrease until low steam generator pressure initiates the closure of Main Steam Isolation Valves (MSIV). The MSIV closure terminates the blowdown of steam through the break thus causing the RCS to heat up once more. Eventually, the heatup is terminated by the opening of secondary safety valves.

7.2.6.4 Conclusions

The results of this analysis demonstrate that the Feedwater System Pipe Break Event will not result in a peak RCS pressure which exceeds the faulted stress pressure limit of 3000 psia.

Table 7.2.6-1

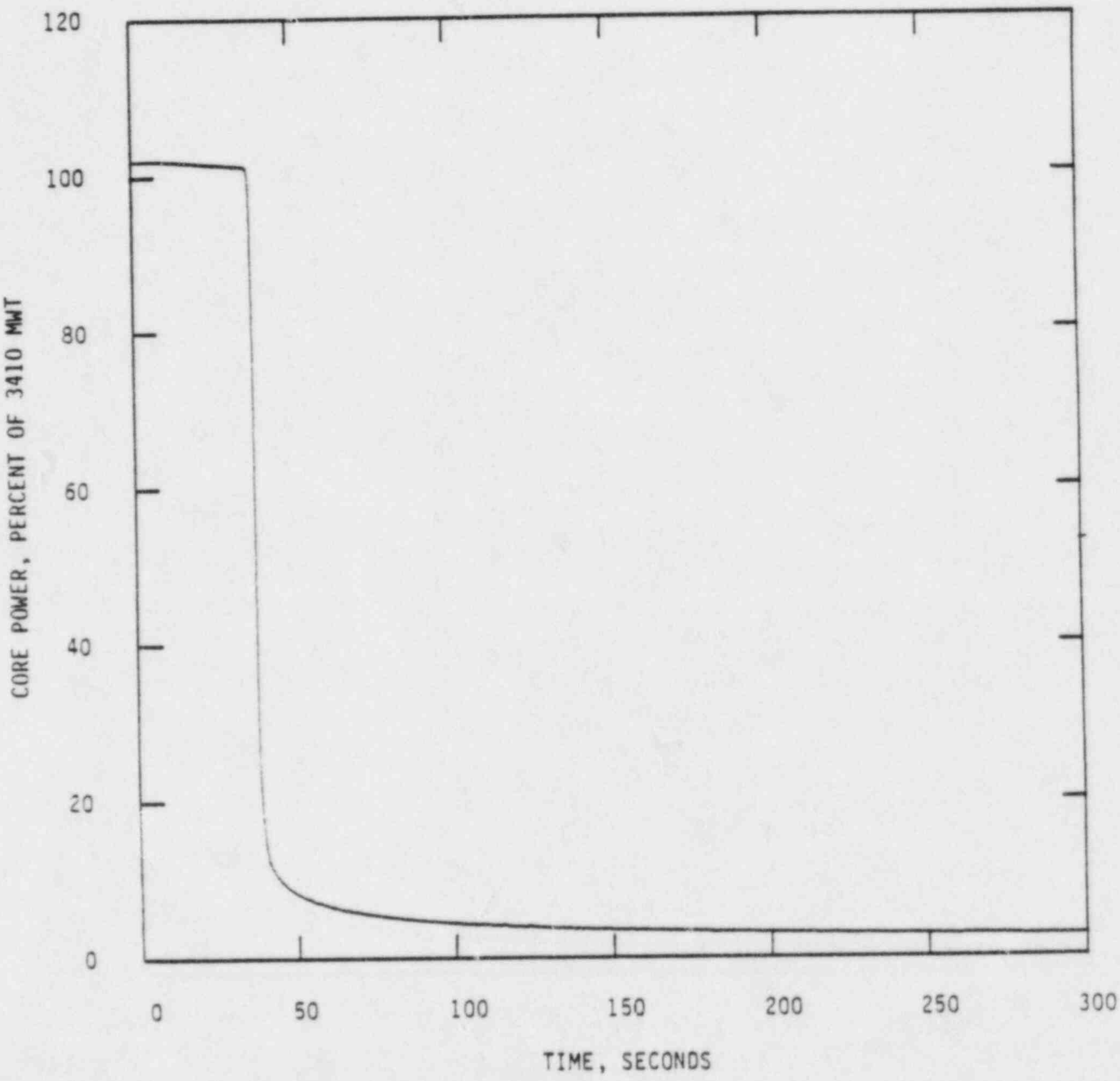
Key Parameters Assumed for the Feedwater System Pipe Break Event

Parameter	Reference Cycle	Cycle 3
Initial Core Power Level, MWt	3478	3478
Initial Inlet Coolant Temperature, ^{o}F	560	560
Initial Core Mass Flow Rate, 10^6 lbm/hr	132.2	132.2
Initial Steam Generator Pressure, psia	971	949
Initial RCS Pressure, psia	2240	2240
Moderator Temperature Coefficient (10^{-4} $\Delta\rho/^{o}F$)	0.0	0.0
Fuel Temperature Coefficient Multiplier	0.85	0.75
Minimum CEA Worth at Trip, $\%\Delta\rho$	-6.00	-6.00
Steam Bypass Control System	Inoperative	Inoperative
Pressurizer Pressure Control System	Automatic Mode	Automatic Mode
Pressurizer Level Control System	Inoperative	Inoperative
Feedwater Line Break Area, ft^2	0.2	0.2
Initial Intact Steam Generator Inventory, lbm	169,830	169,830
Auxiliary Feedwater Capacity assuming one failed pump, gpm	700.	700.
Number of Assumed Plugged Steam Plugged Steam Generator Tubes	200.	1000.

Table 7.2.6-2

Sequence of Events for the Feedwater System Pipe Break Event

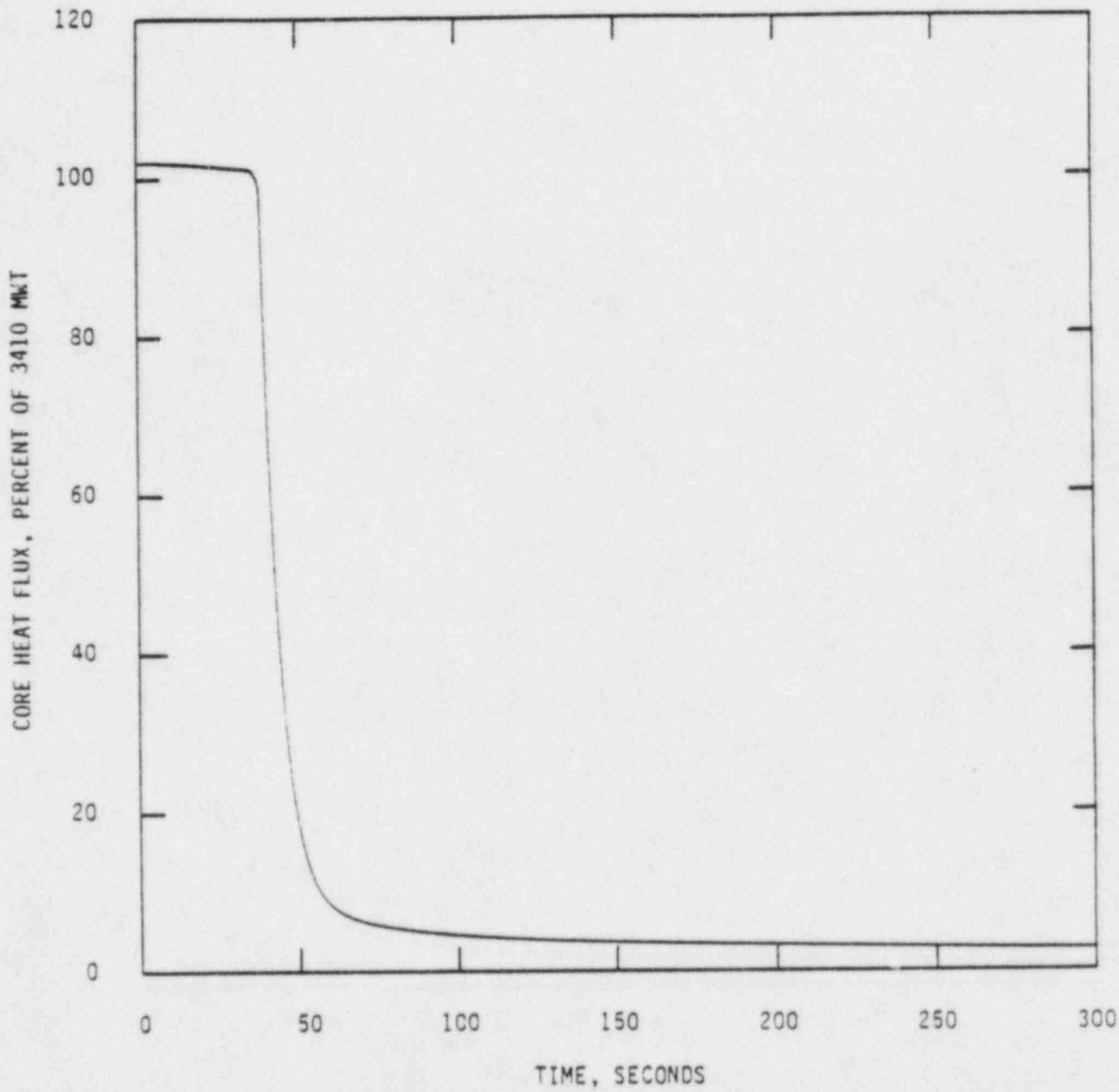
Time (sec)	Event	Setpoint or Value
0.0	Rupture of Main Feedwater Line	---
34.8	Affected Steam Generator Empties	---
35.0	Low Steam Generator Water Level Trip Condition Occurs in Intact Steam Generator	27.03 ft
	High Pressurizer Pressure Trip Condition Occurs	2475 psia
35.6	Pressurizer Safety Valves Open	2525 psia
35.9	Trip Breakers Open; Normal Onsite and Offsite Power Lost	--- ---
36.2	CEAs Begin to Drop into Core	---
40.0	Steam Generator Safety Valves Open	1100 psia
40.2	Peak RCS Pressure Occurs	2943 psia
43.1	Peak Steam Generator Pressure Occurs	1140 psia
48.4	Pressurizer Safety Valves Close	2400 psia
48.6	Maximum Pressurizer Liquid Volume	1270 ft^3
70.4	Steam Generator Safety Valves Close	1056 psia
88.9	Emergency Feedwater Enters Intact Steam Generator	---
212.7	Steam Generator Low Pressure Trip Condition and MSIS Initiated	675 psia
223.6	Complete Closure of Main Steam Isolation Valves Terminating Blowdown from the Intact Steam Generator	---
234.9	Minimum liquid Mass in the Steam Generator Connected to Intact Feedline	7527 lbm
1800.	Operator Opens the Atmospheric Steam Dump Valves to begin Plant Cooldown to Shutdown Cooling	---



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
CORE POWER VS TIME

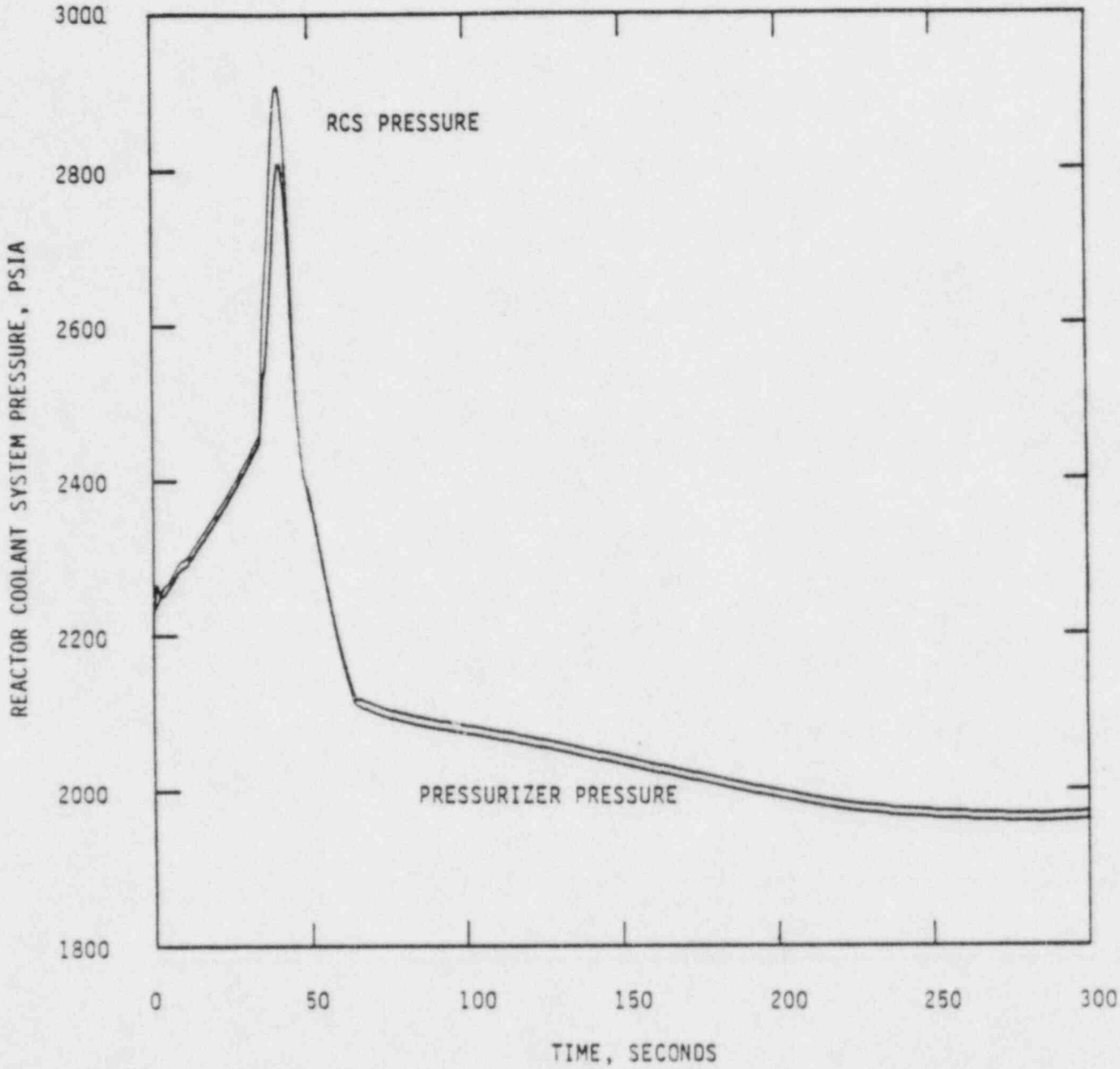
FIGURE 7.2.6-1



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
CORE HEAT FLUX VS TIME

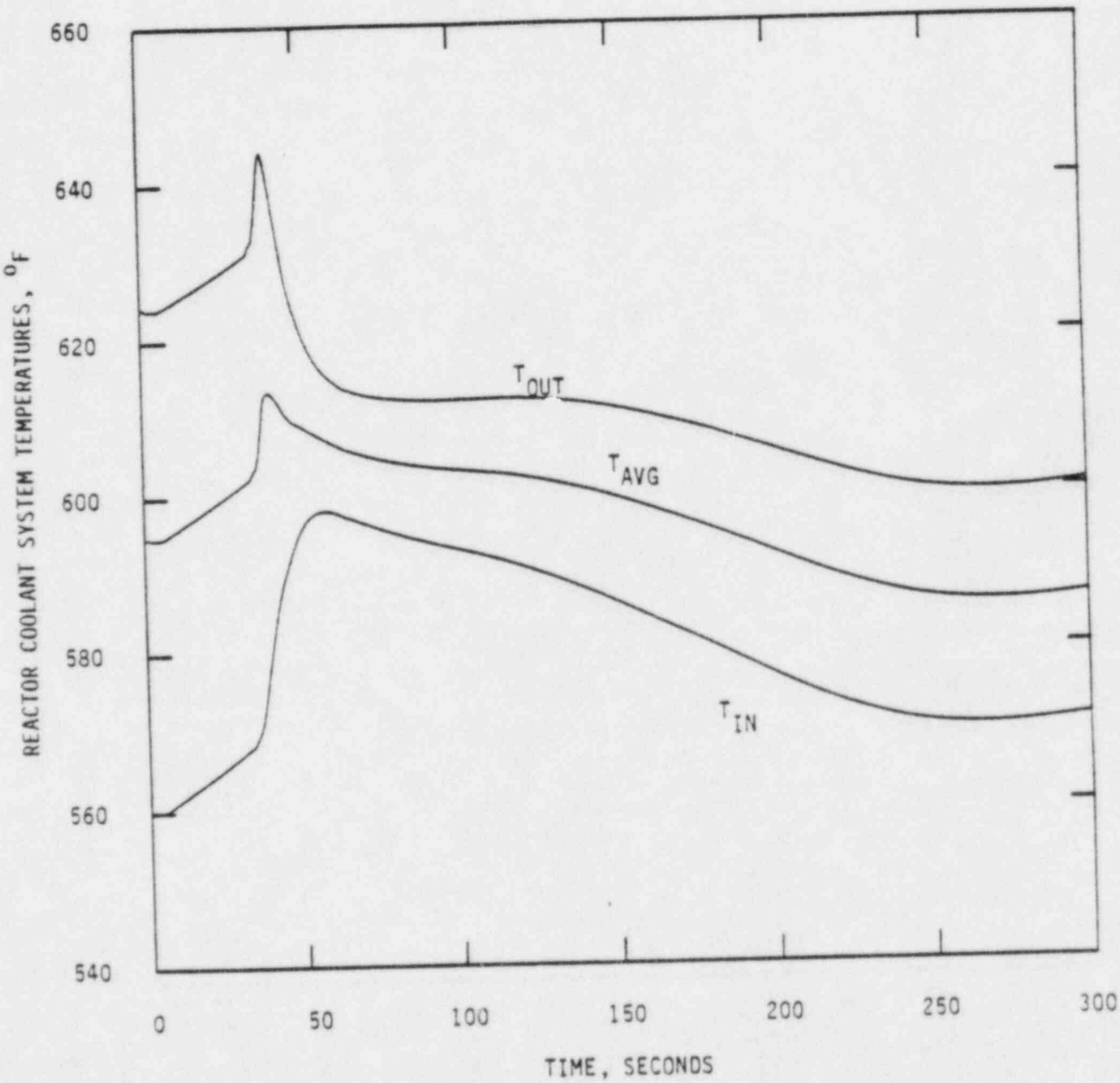
FIGURE 7.2.6-2



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
RCS PRESSURE VS TIME

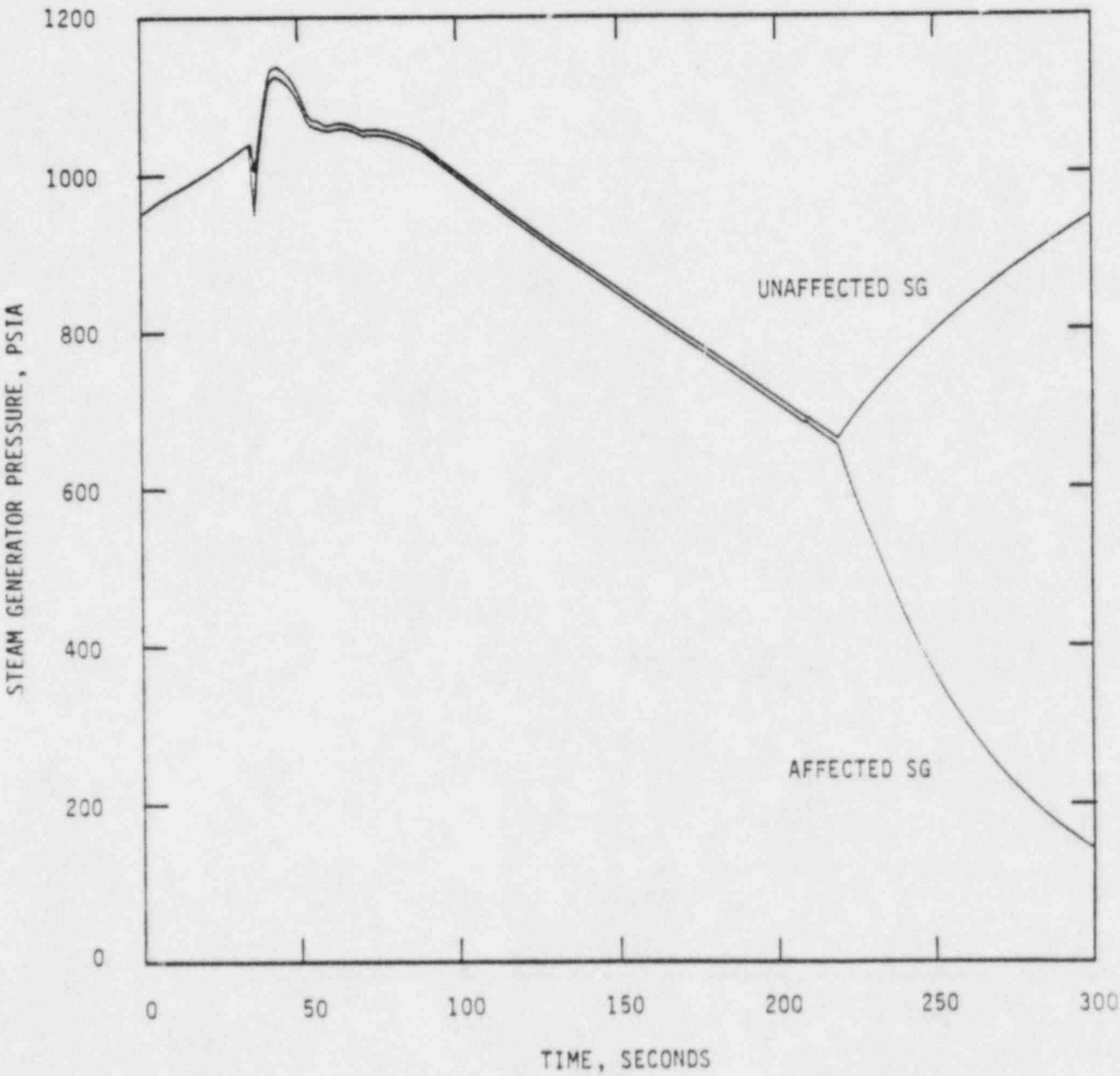
FIGURE 7.2.6-3



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
RCS TEMPERATURES VS TIME

FIGURE 7.2.6-4



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
STEAM GENERATOR PRESSURE VS TIME

FIGURE 7.2.6-5

7.3 Decrease in Reactor Coolant Flowrate

7.3.1 Partial Loss of Forced Reactor Coolant Flow

The results are bounded by the Reference Cycle.

7.3.2 Total Loss of Forced Reactor Coolant Flow

The Loss of Coolant Flow (LOF) Event is analyzed to determine the minimum initial margin that must be maintained by the Limiting Conditions for Operations (LCOs) such that in conjunction with the Reactor Protection System (RPS) the DNBR SAFDL will not be exceeded. This event was reanalyzed due to a reduction in CEA worth at trip. The method used to analyze this event is the same as the method described in Reference 7-14, Appendix A.

7.3.2.1 Identification of Causes

A loss of normal coolant flow may result either from a loss of electrical power to one or more of the four reactor coolant pumps or from a mechanical failure, such as a pump shaft seizure. Simultaneous mechanical failure of two or more pumps is not considered credible. If the RCP shaft speed reduction from either cause is greater than the CPC low pump speed trip setpoint, a reactor trip is initiated.

Reactor trip on loss of coolant flow is initiated by the CPC's on low RCP shaft speed. For a loss of flow at full power operating conditions, a trip will be initiated when the RCP shaft speed drops to 95 percent of its initial speed. For conservatism, the safety analysis assumes that the CPC's initiate a reactor trip when the reactor coolant flow reaches 95 percent. The reduction in core flow lags the decrease in RCP shaft speed.

7.3.2.2 Analysis of Effects and Consequences

The transient is characterized by the flow coastdown curve given in Figure 7.3.2-1. Table 7.3.2-1 presents the initial conditions assumed in this event.

7.3.2.3 Results

Table 7.3.2-2 presents the sequence of events for the 4-pump Loss of Flow Event. This is a representative case and is initiated at a shape index of zero. The low flow trip setpoint is reached at .80 seconds and the scram CEAs start dropping into the core 0.52 seconds later. A minimum CE-1 DNBR of 1.31 is reached at 2.7 seconds. Figures 7.3.2-2 to 7.3.2-5 present the core power, heat flux, RCS pressure, and RCS temperatures as a function of time.

7.3.2.4 Conclusions

The event initiated from the Technical Specification LCOs in conjunction with the CPC low RCP shaft speed trip will not exceed the DNBR SAFDL.

Table 7.3.2-1

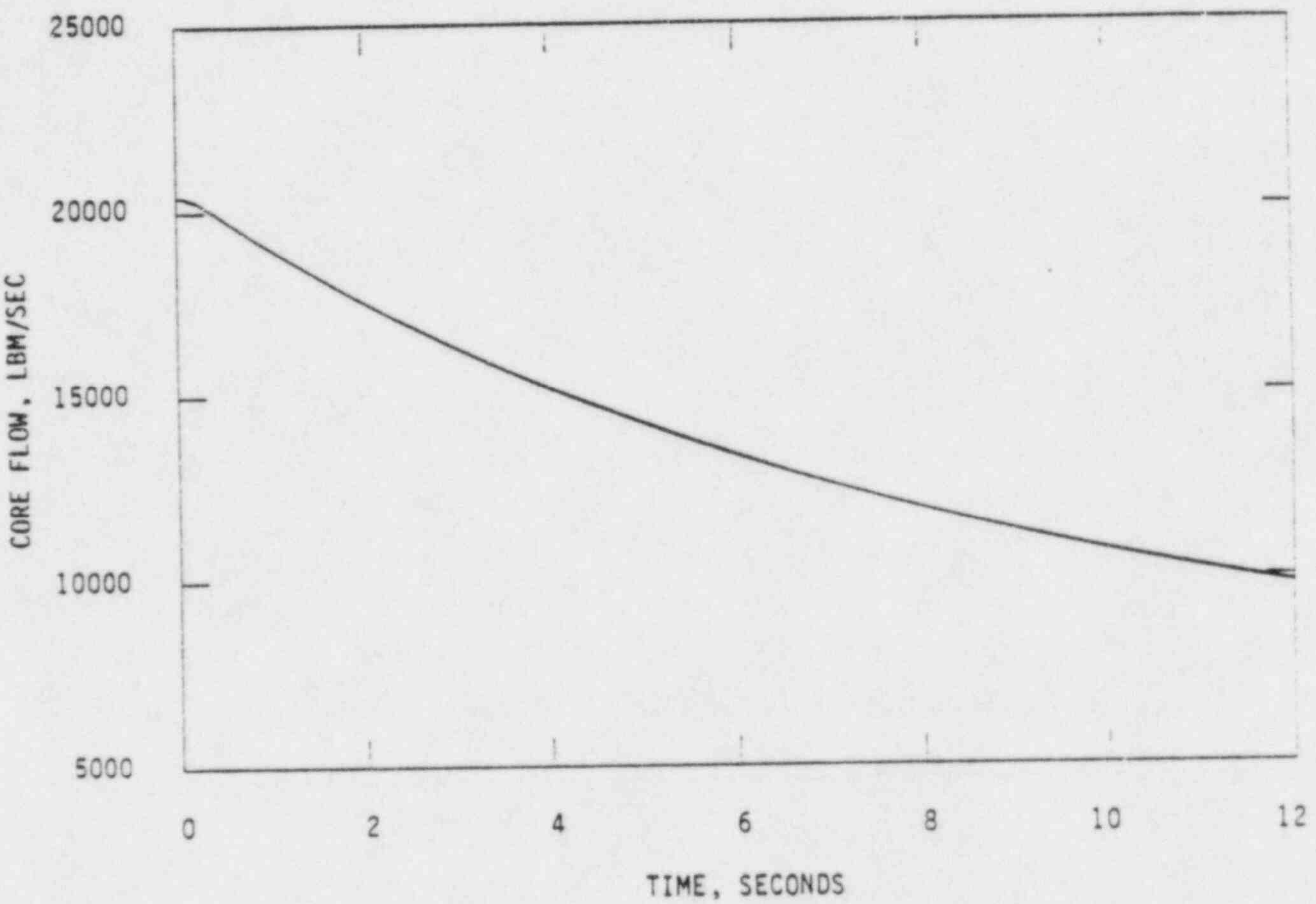
Key Parameters Assumed for the Total Loss of Forced Reactor Coolant Flow Event

Parameter	Units	Reference Cycle Value	Cycle 3 Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	3478	3478
Initial Core Coolant Inlet Temperature	^oF	560	560
Initial RCS Vessel Flow Rate	gpm	396,000	396,000
Initial Reactor Coolant System Pressure	psia	2325	2325
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/^oF$	+.5	+.5
Doppler Coefficient Multiplier	--	.85	.75
Low Pump Speed Trip Setpoint (Reactor Coolant Pump Shaft Speed Setpoint)		0.95	0.95
Low Pump Speed Trip Response Time	sec	0.22	0.22
CEA Holding Coil Delay	sec	0.3	0.3
CEA Time to 90% Insertion (Including Holding Coil Delay)	sec	3.0	3.0
CEA Worth at Trip (all rods out)	$\%\Delta\rho$	-6.25	-6.0
4-Pump RCS Flow Coastdown		Figure 7.3.2-1	Figure 7.3.2-1

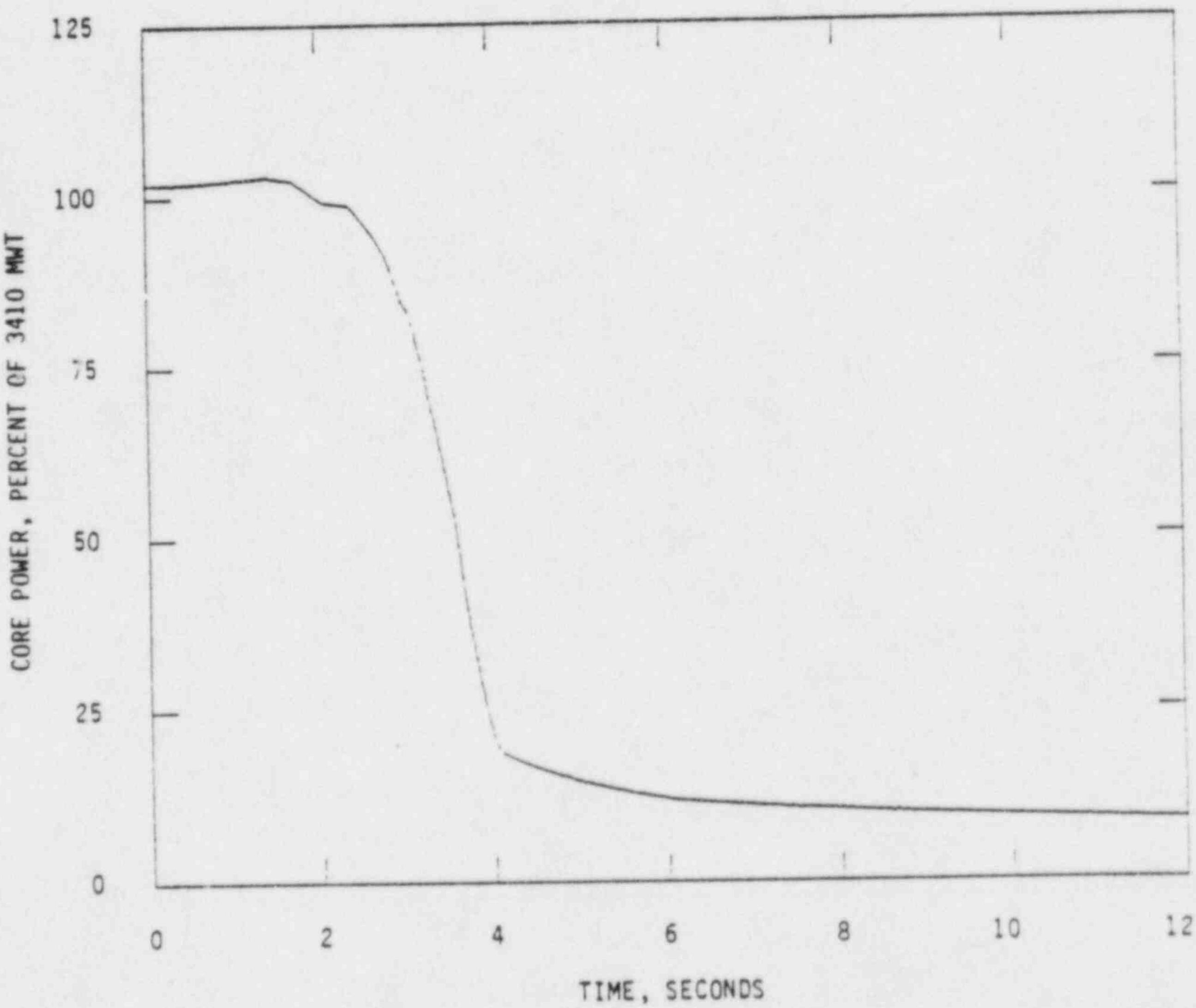
Table 7.3.2-2

Sequence of Events for Total Loss of Forced Reactor Coolant Flow Event

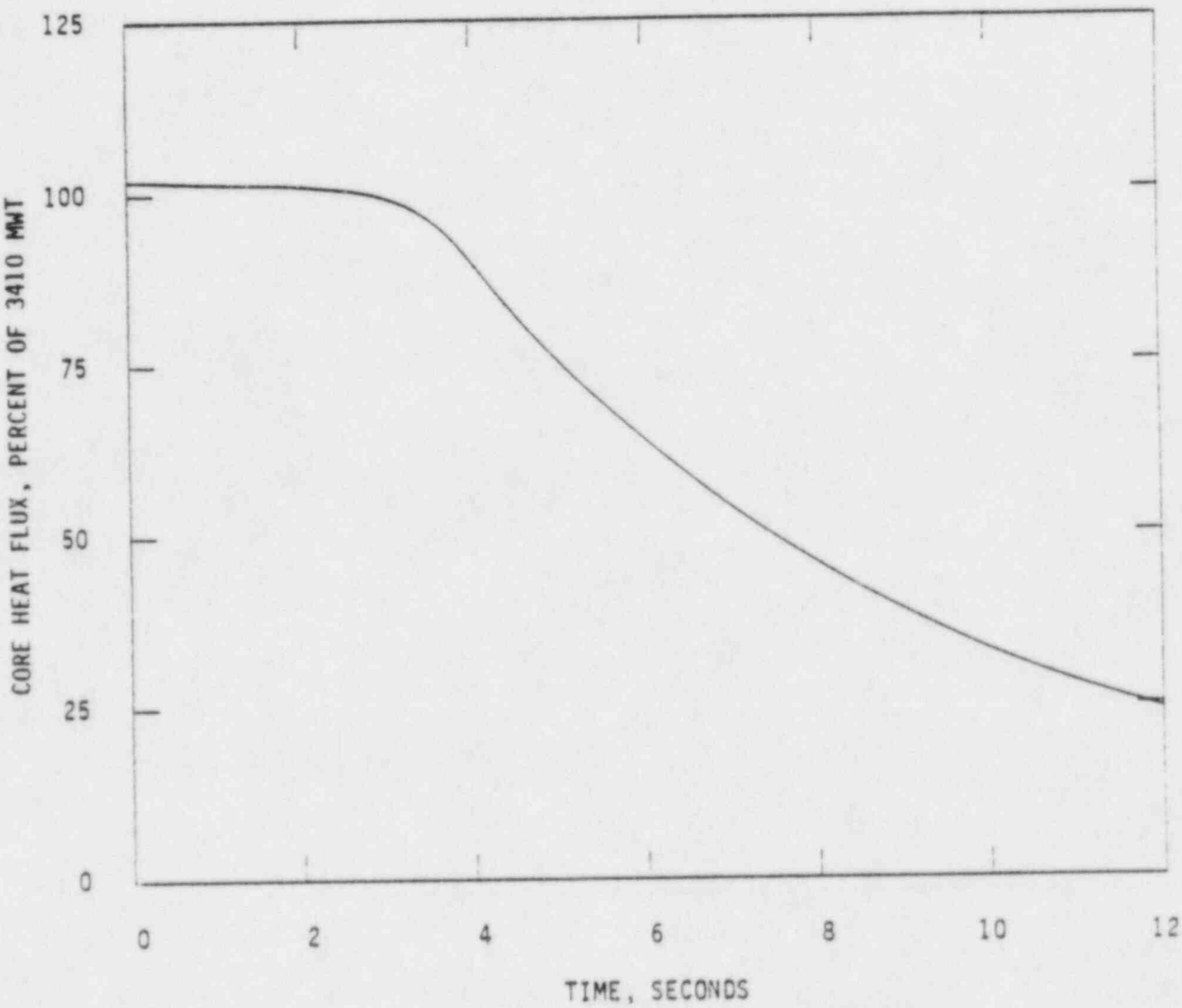
Time (sec)	Event	Setpoint or Value
0.0	Loss of Power to all Four Reactor Coolant Pumps	--
0.80	Low Reactor Coolant Pump Shaft Speed Trip Signal Generated	95% of shaft speed
1.02	Trip Breakers Open	--
1.32	CEAs Begin to Drop into Core	--
2.70	Minimum CE-1 DNBR	$\geq$ 1.31
4.7	Maximum RCS Pressure	2523 psia



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
TOTAL LOSS OF FORCED REACTOR COOLANT FLOW CORE FLOW VS TIME
FIGURE 7.3.2-1



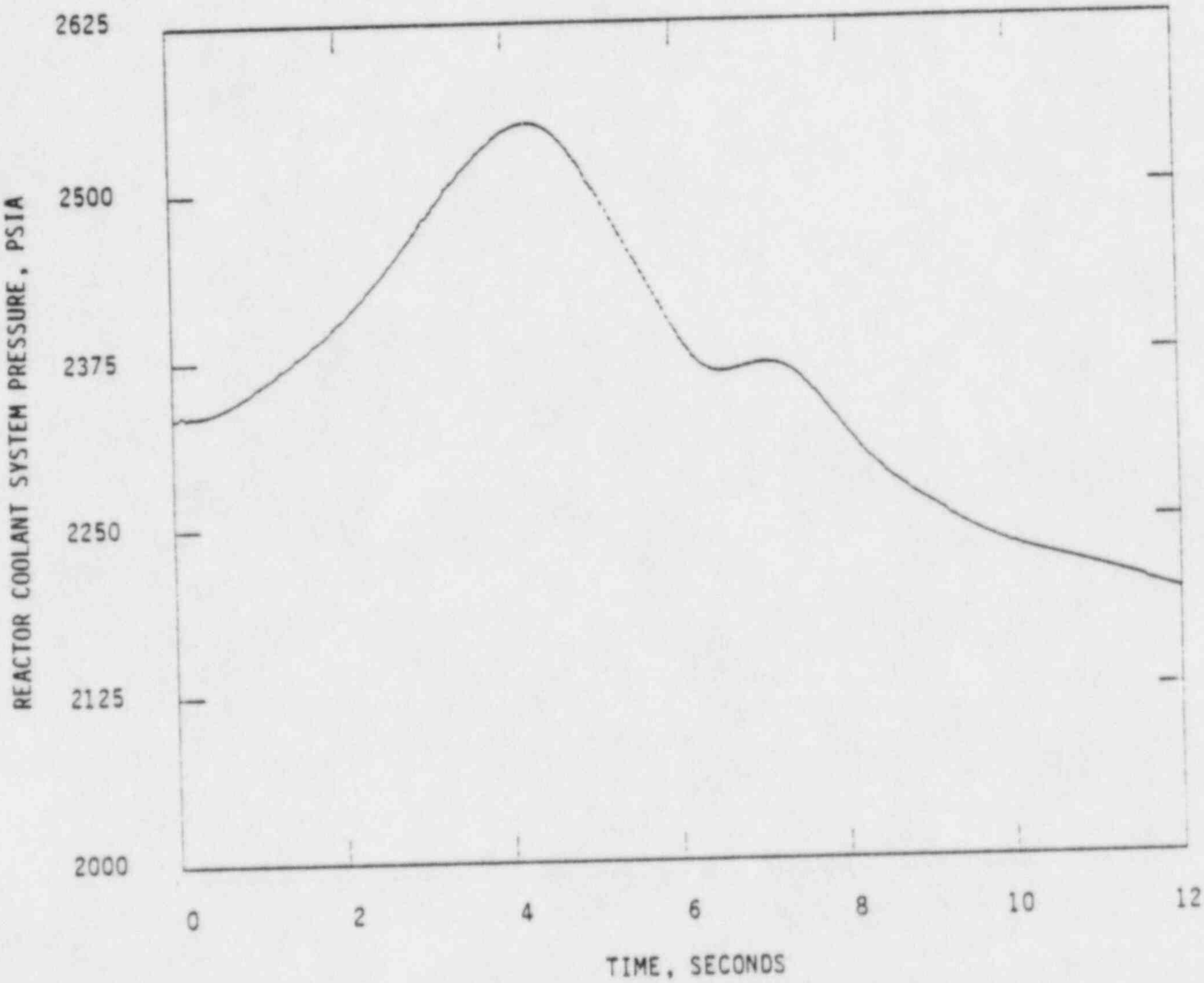
SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
TOTAL LOSS OF FORCED REACTOR COOLANT FLOW CORE POWER VS TIME
FIGURE 7.3.2-2



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

TOTAL LOSS OF FORCED REACTOR
COOLANT FLOW
CORE HEAT FLUX VS TIME

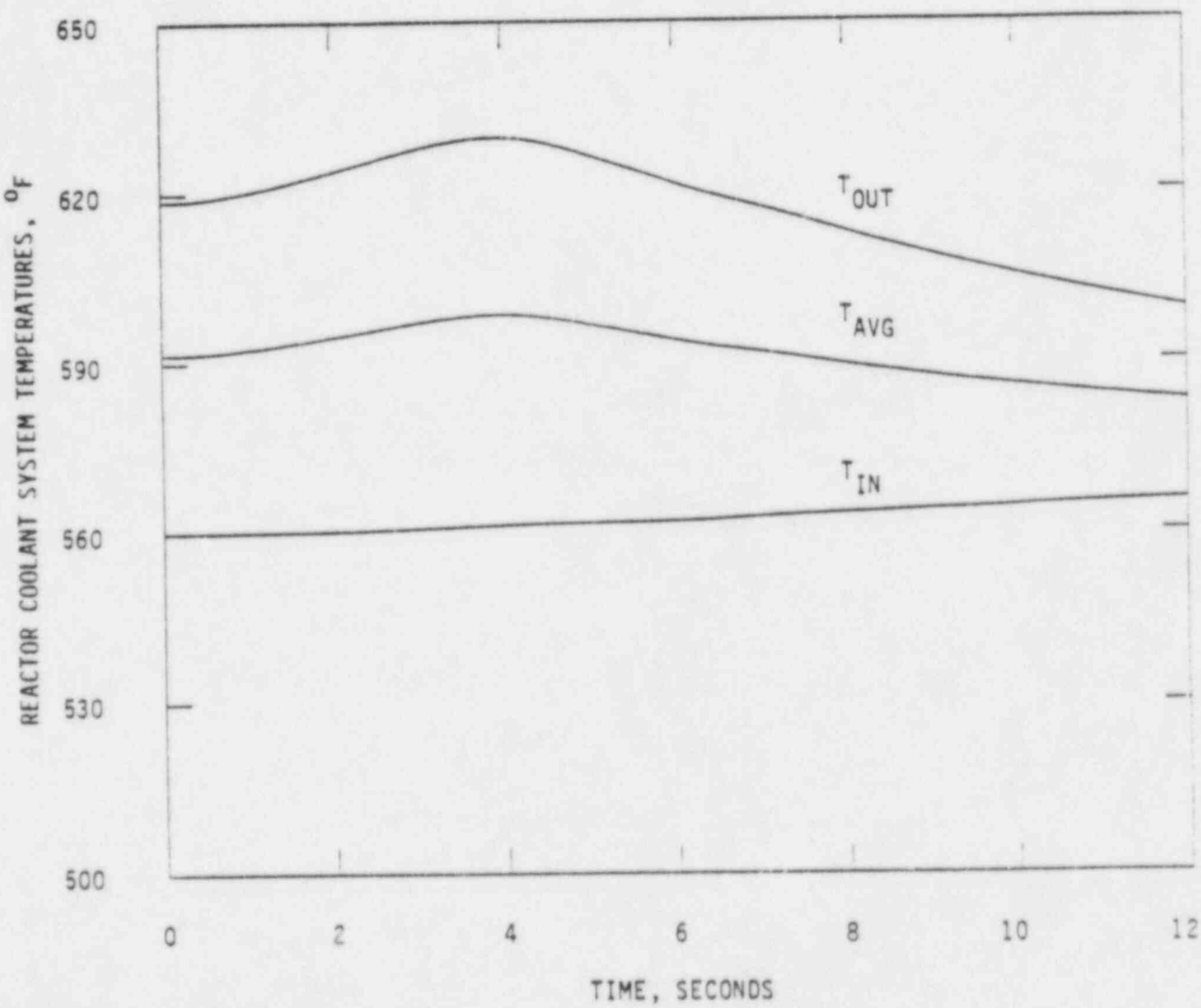
FIGURE 7.3.2-3



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

TOTAL LOSS OF FORCED REACTOR
COOLANT FLOW
RCS PRESSURE VS TIME

FIGURE 7.3.2-4



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

TOTAL LOSS OF FORCED REACTOR COOLANT FLOW
RCS TEMPERATURES VS TIME

FIGURE 7.3.2-5

7.3.3 Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

The single reactor coolant pump sheared shaft (SS) was reanalyzed due to a change in the fuel failure pin census. The SS was reanalyzed to ensure that a coolable geometry is maintained and that the site boundary doses do not exceed 10CFR100 guidelines.

7.3.3.1 Identification of Causes

A single reactor coolant pump sheared shaft is caused by mechanical failure of the pump shaft. Following the shearing of a reactor coolant pump shaft, the core flowrate rapidly decreases to the value that would occur with only three reactor coolant pumps operating. The reduction in coolant flowrate causes an increase in the average coolant temperature in the core and may produce a departure from nucleate boiling (DNB) condition in some portions of the core. A reactor trip is generated when the rapid flow reduction across the steam generator in the affected loop decreases the delta-pressure below the trip setpoint. The reactor trip produces an automatic turbine trip. Following turbine trip, offsite power is available to provide AC power to the auxiliaries. The operator can initiate a controlled system cooldown using the turbine bypass valves any time after reactor trip. The steam release to the atmosphere, even if operator action is delayed for 30 minutes following first indication of the event, would be no more than that following a loss of all normal AC power.

7.3.3.2 Analysis of Effects and Consequences

The sheared shaft was assumed to occur at hot full power and at core thermal hydraulic conditions such that the minimum thermal margin is being reserved by the Core Operating Limit Supervisory System (COLSS). Table 7.3.3-1 contains the initial conditions for Cycle 3 and Reference Analysis (Reference 7-2). No credit was taken for heat flux decay upon reactor trip. This method essentially trades the initial reserved margin off against a reduction of core flow to 75% of its initial value. This method is extremel/ conservative. The minimum DNBR for this event was calculated with the TORC computer code.

7.3.3.3 Results

The sheared shaft results in a minimum calculated DNBR of 1.12 compared to the design limit of 1.31. This results in a predicted fuel failure of less than 9%. The Acceptable Fuel to Centerline Melt of 21 kw/ft is not violated. The resultant offsite doses are less than 300 REM thyroid and less than 25 REM whole body. Additionally, the peak RCS pressure is less than 2750 psia.

7.3.3.4 Conclusions

For the sheared shaft the radiological doses are less than the 10CFR100 limits of 300 REM thyroid and less than 25 REM whole body. As in the FSAR (Reference 7-2), the consequences of the sheared shaft are more limiting than the seized rotor event.

Table 7.3.3-1

Key Parameters Assumed for the
Single Reactor Coolant Pump Sheared Shaft Event

Parameter	Reference Cycle Value	Cycle 3 Value
Initial Core Power Level, MWt	3478	3478
Core Inlet Coolant Temperature, $^{\circ}F$	560	560
Core Mass Flowrate, 10^6 lbm/hr	136.8	156.1
Reactor Coolant System Pressure, lb/in^{2}a	2,000	2350
Maximum Radial Power Peaking Factor	1.59	1.7

7.4 Reactivity and Power Distribution Anomalies

7.4.1 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition

The uncontrolled CEA withdrawal (CEAW) from subcritical or low power conditions is analyzed to ensure that the departure from nucleate boiling ratio (DNBR) and the fuel centerline melt (CTM) specified acceptable fuel design limits (SAFDLs) are not violated. Additionally, the CEAW from subcritical and low powers is analyzed to verify that the peak RCS pressure is less than the design limit of 2750 psia.

7.4.1.1 Identification of Causes

An uncontrolled withdrawal of CEAs is assumed to occur as a result of a single failure in the control element drive mechanism (CEDM), control element drive mechanism control system (CEDMCS), reactor regulating system, or as a result of operator error.

7.4.1.2 Analysis of Effects and Consequences

The withdrawal of CEAs from subcritical or low power conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase together with corresponding increases in reactor coolant temperatures and reactor coolant system (RCS) pressure. The withdrawal motion of CEAs also produces a time dependent redistribution of core power. These transient variations in core thermal parameters result in the system's approach to the specified fuel design limits and RCS and secondary system pressure limits, thereby requiring the protective action of the Reactor Protection System (RPS).

The reactivity insertion rate accompanying the uncontrolled CEA withdrawal is dependent primarily upon the CEA withdrawal rate and the CEA worth since, at subcritical and lower power conditions, the normal reactor feedback mechanisms do not occur until power generation in the core is large enough to cause changes in the fuel and moderator temperatures. The reactivity insertion rate determines the rate of approach to the fuel design limits. Depending on the system initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a high logarithmic power trip, high power level trip, high pressurizer pressure trip, low departure from nucleate boiling ratio (DNBR) trip, high local power density trip, or variable overpower trip (VOPT).

A CEA withdrawal from subcritical was initiated from the conditions in Table 7.4.1-1. A moderator temperature coefficient (MTC) of $+0.5\times10^{-4}\ \Delta\rho/^{o}F$ was used in this analysis. This MTC, in conjunction with the increasing core coolant temperatures, yields an increase in core heat flux. The least negative fuel temperature coefficient (FTC) with a bias is used in this analysis. The minimum CEA worth assumed for shutdown at time of reactor trip for zero power operation is 5.15%$\Delta\rho$ and 4.0%$\Delta\rho$ for subcritical (Mode 2) operation.

7.4.1.3 Results

The uncontrolled CEA withdrawal from subcritical conditions resulted in a reactor trip on high logarithmic power at 75.2 seconds. The minimum DNBR calculated for this event initiated from the conditions of Table 7.4.1-1 was greater than the design limit of 1.31. The peak linear heat generation rate

(PLHGR) was calculated to be 26 kw/ft which is in excess of the steady state acceptable fuel to centerline melt (CTM) limit of 21 kw/ft. However, the fuel centerline temperature does not exceed 4900^{o}F and the fuel is not predicted to melt. Additionally, the peak RCS pressure is less than the design limit of 2750 psia. Table 7.4.1-2 presents the sequence of events for this event. Figures 7.4.1-1 through 7.4.1-5 present the NSSS response for core power, core heat flux, RCS temperatures, RCS pressure and steam generator pressure.

The results of the uncontrolled CEA withdrawal from low power is presented due to a change in the RPS. The VOPT added to the CPCs is credited to mitigate the consequences of this event. The low power CEAWs were analyzed to maximize the RCS pressure increase and to maximize the potential for fuel degradation. The initial conditions for the CEAW that maximizes peak RCS pressure are listed in Table 7.4.1-3. A parametric on the reactivity addition rate was performed to yield a coincident VOPT/high pressurizer pressure trip in order to maximize the peak RCS pressure. A high pressurizer pressure/VOPT is generated at 151.4 seconds and the scram CEA's begin to drop at 151.7 seconds. The peak RCS pressure is 2640 psia and occurs at 152.9 seconds. The sequence of events is presented in Table 7.4.1-4. Figures 7.4.1-6 through 7.4.1-11 present the NSSS response for this event. Since the CEAW from low power is a CPC Design Basis Event (DBE) core thermal limits are not exceeded.

7.4.1.4 Conclusions

An uncontrolled CEA withdrawal from either subcritical or low power conditions will not exceed the DNBR or CTM limits. The RCS pressure limit of 2750 psia will not be exceeded during this event.

7.4.2 Uncontrolled CEA Withdrawal at Power

The results are bounded by the Reference Cycle.

7.4.3 CEA Misoperation Event

The results are bounded by the Reference Cycle.

Table 7.4.1-1

Key Parameters Assumed in the CEA Withdrawal From Subcritical Conditions Event

Parameter	Reference Cycle	Cycle 3
Initial Core Power Level, MWt	3478×10^{-10}	3478×10^{-10}
Initial Inlet Coolant Temperature, $^{\circ}F$	530.5	520
Initial Core Mass Flow Rate, 10^{6} lbm/hr	128.6	150.2
Initial RCS Pressure, psia	2000	2000
Moderator Temperature Coefficient (10^{-4} $\Delta\rho/^{\circ}F$)	0.5	0.5
Fuel Temperature Coefficient Multiplier	0.85	0.75
Minimum CEA Worth at Trip, $\%\Delta\rho$	-4.45	-4.0
Maximum Reactivity Addition Rate, (x 10^{-4} $\Delta\rho$/sec)	0.8	1.9

Table 7.4.1-2

Sequence of Events for the CEA Withdrawal

From Subcritical Conditions Event

Time (sec)	Event	Setpoint or Value
0.0	Initiation of Uncontrolled Sequential CEA Withdrawal	----
53.0	Reactor Reaches Criticality	----
74.8	Reactor Reaches High Logarithmic Power Trip Setpoint	2% of Rated
75.2	Reactor Trip Generated	----
75.5	CEAs Begin to Drop	----
75.6	Peak Reactor Core Power Reached	65% of 3410 MWt
75.7	Peak Reactor Core Heat Flux Reached	9.7% of 3410 MWt
75.7	Minimum DNBR Occurs	$\geq$ 1.31

Table 7.4.1-3

Key Parameters Assumed in the CEA Withdrawal From Low Powers Event

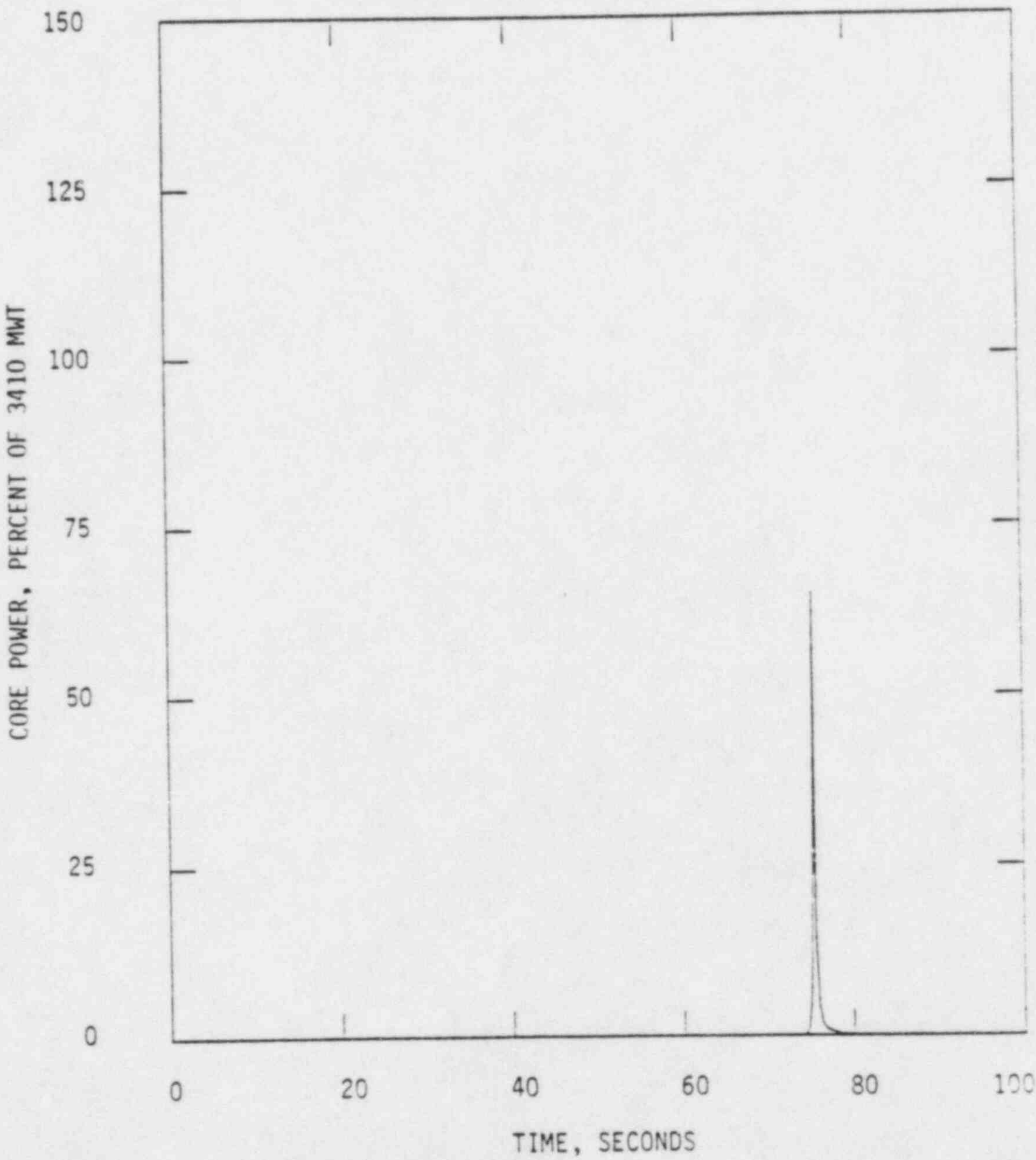
Parameter	Reference Cycle	Cycle 3
Initial Core Power Level, MWt	0.3478	34.78
Initial Inlet Coolant Temperature, ^{o}F	530.5	520
Initial Core Mass Flow Rate, 10^{6} lbm/hr	128.6	150.2
Initial RCS Pressure, psia	2000	2000
Moderator Temperature Coefficient ($10^{-4}\Delta\rho/^{o}F$)	0.5	0.5
Fuel Temperature Coefficient Multiplier	0.85	0.75
Minimum CEA Worth at Trip, $\%\Delta\rho$	-4.45	-5.15
Maximum Reactivity Addition Rate, ($x\ 10^{-4}\Delta\rho$/sec)	0.8	1.1

Table 7.4.1-4

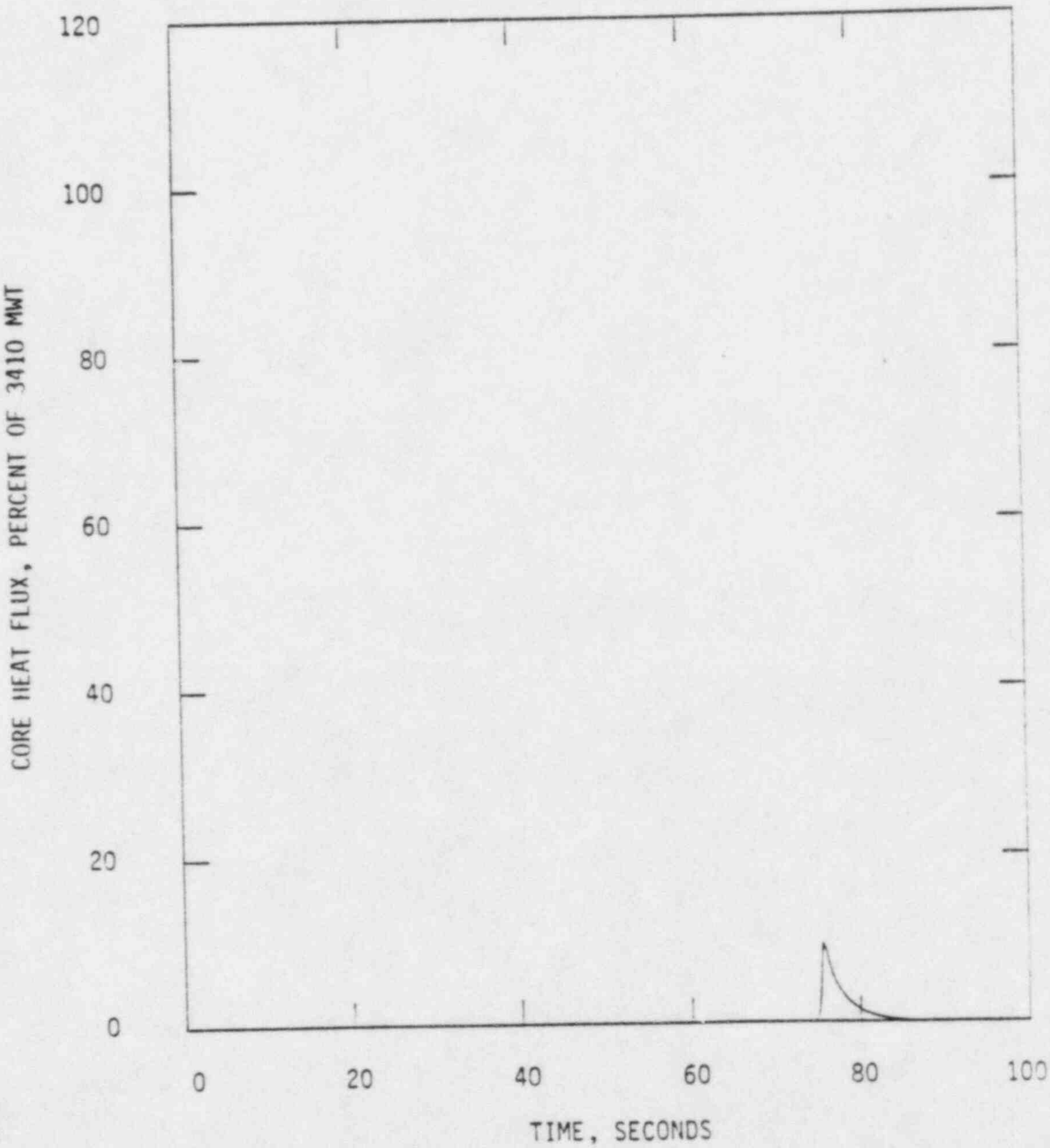
Sequence of Events for the CEA

Withdrawal from Low Powers Event

Time (sec)	Event	Setpoint or Value
0.0	CEAW Initiated	----
150.5	High Pressurizer Pressure Trip Condition	2475 psia
151.4	High Pressurizer Pressure/VOPT Reactor Trip Occurs	----
151.7	Scram CEAs Begin to Drop	----
152.1	Pressurizer Safety Valves Open	2525 psia
152.9	Peak RCS Pressure	2640 psia
153.1	Peak Core Power	75.4% of 3410 MWt
153.8	Peak Core Heat Flux	61.8% of 3410 MWt
153.8	Minimum DNBR	$\geq$ 1.31
155.7	Pressurizer Safety Valves Close	2400 psia



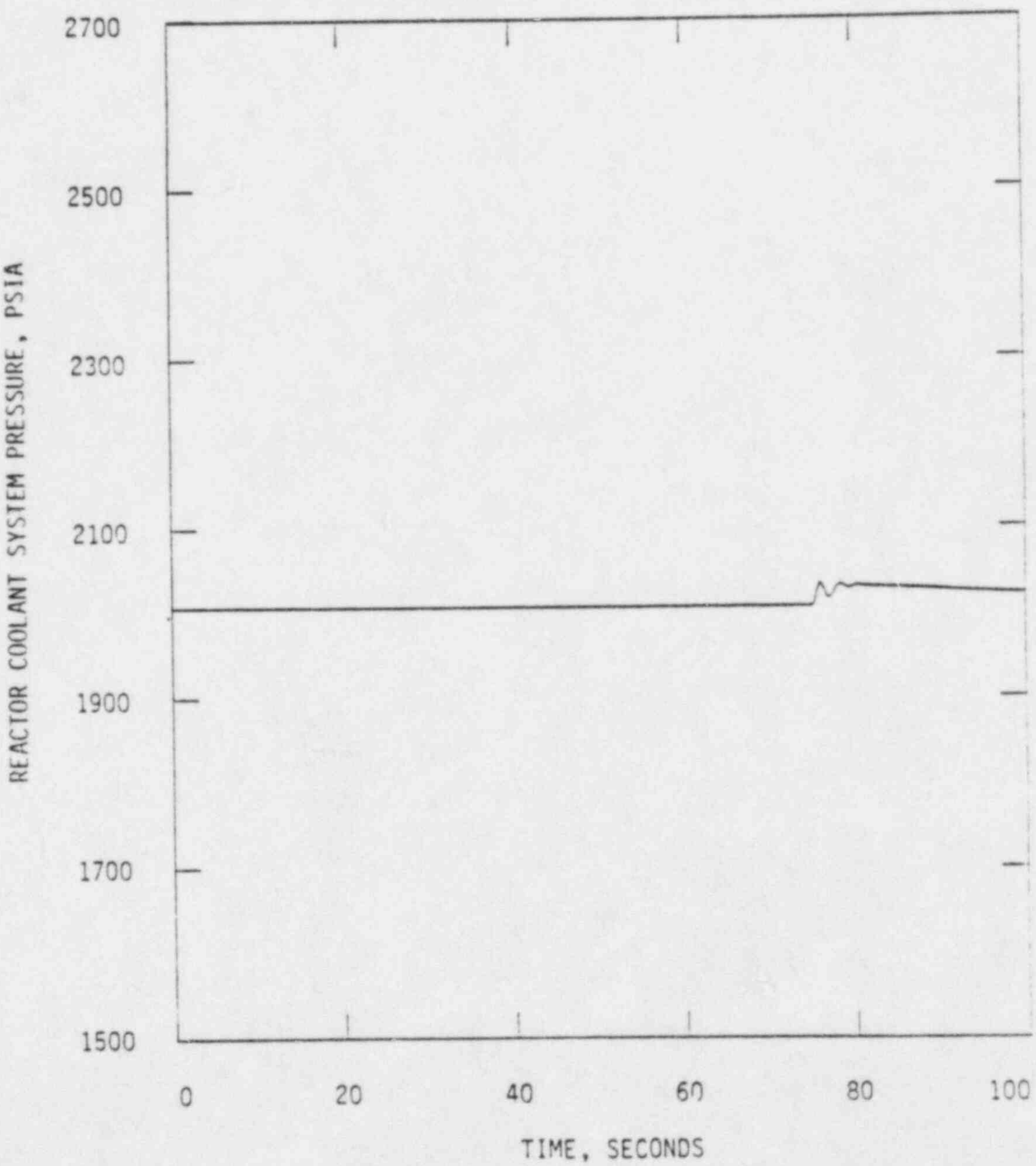
SAN ONOFRE **NUCLEAR GENERATING STATION** **Units 2 & 3**
CEA WITHDRAWAL FROM SUBCRITICAL CORE POWER VS TIME
FIGURE 7.4.1-1



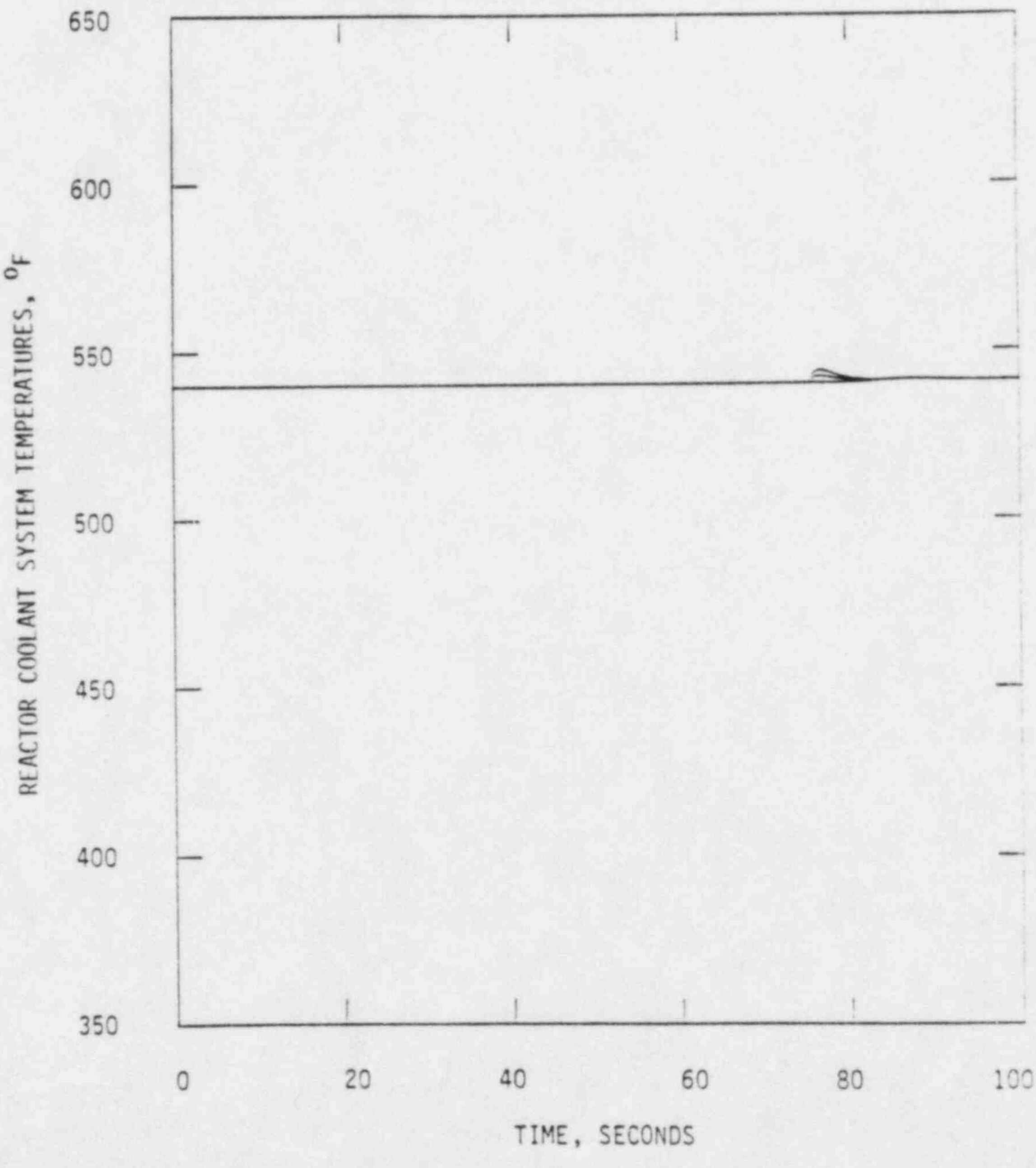
SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CEA WITHDRAWAL FROM SUBCRITICAL
CORE HEAT FLUX VS TIME

FIGURE 7.4.1-2



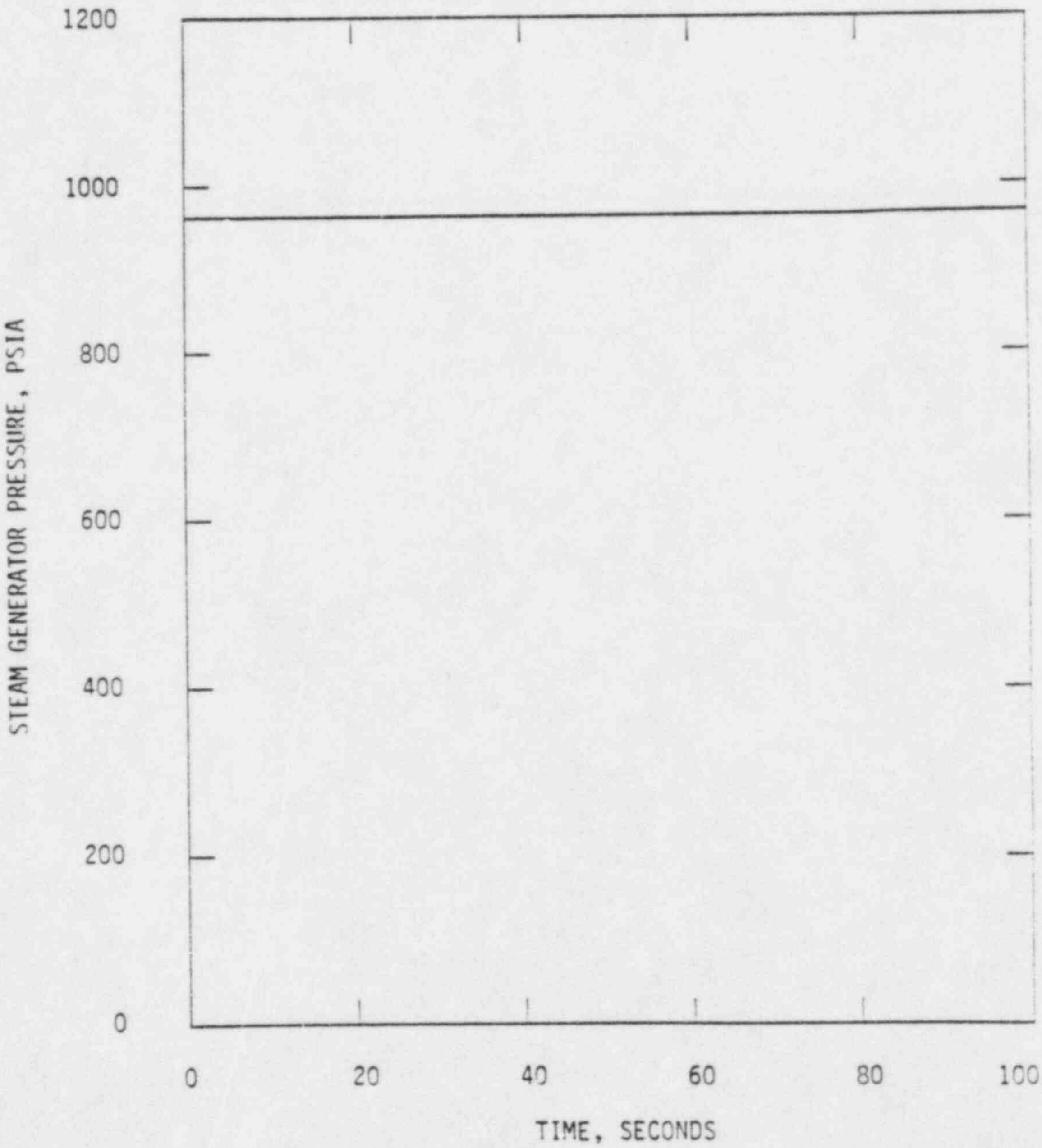
SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
CEA WITHDRAWAL FROM SUBCRITICAL RCS PRESSURE VS TIME
FIGURE 7.4.1-3



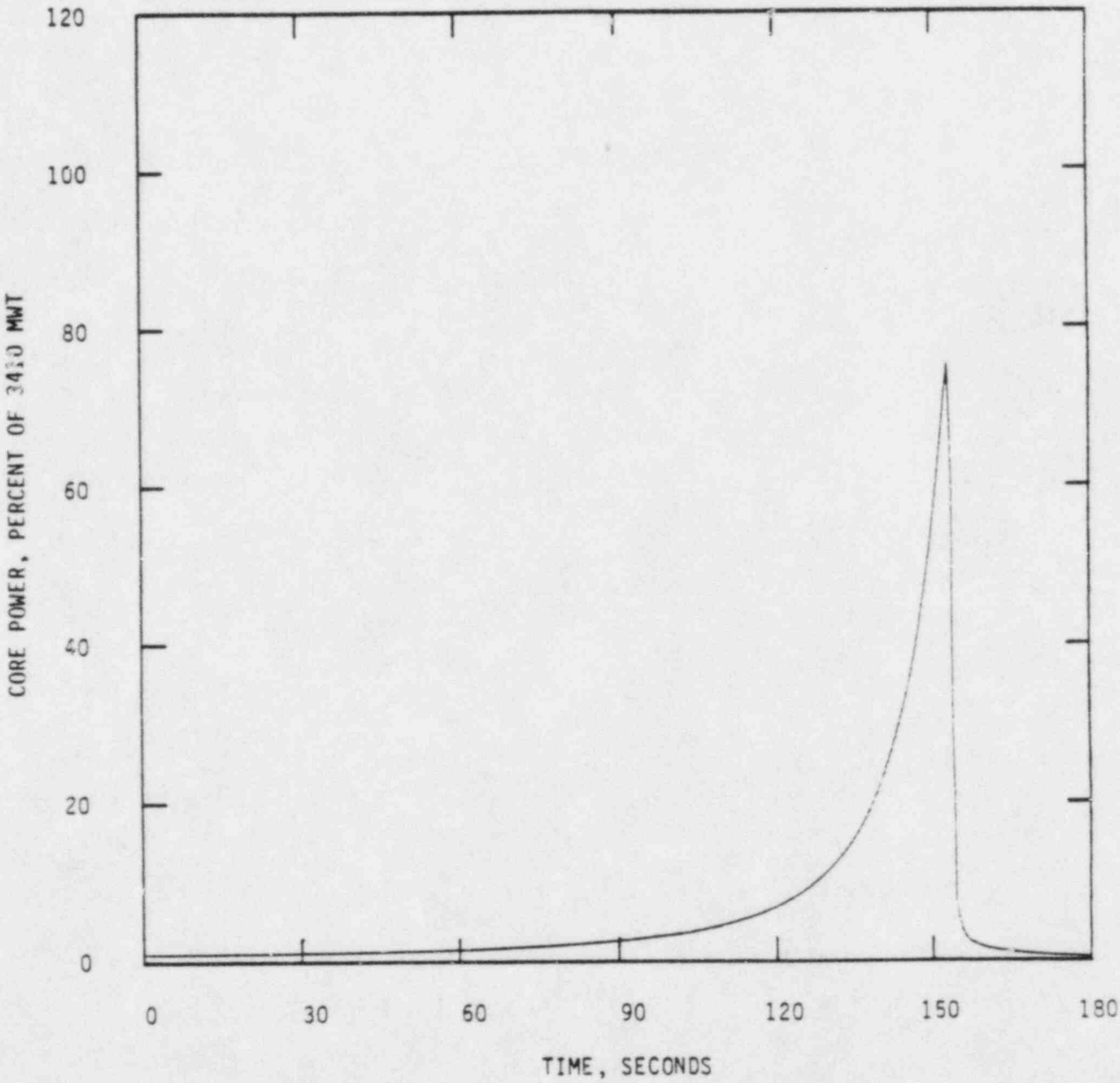
SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CEA WITHDRAWAL FROM SUBCRITICAL
RCS TEMPERATURES VS TIME

FIGURE 7.4.1-4



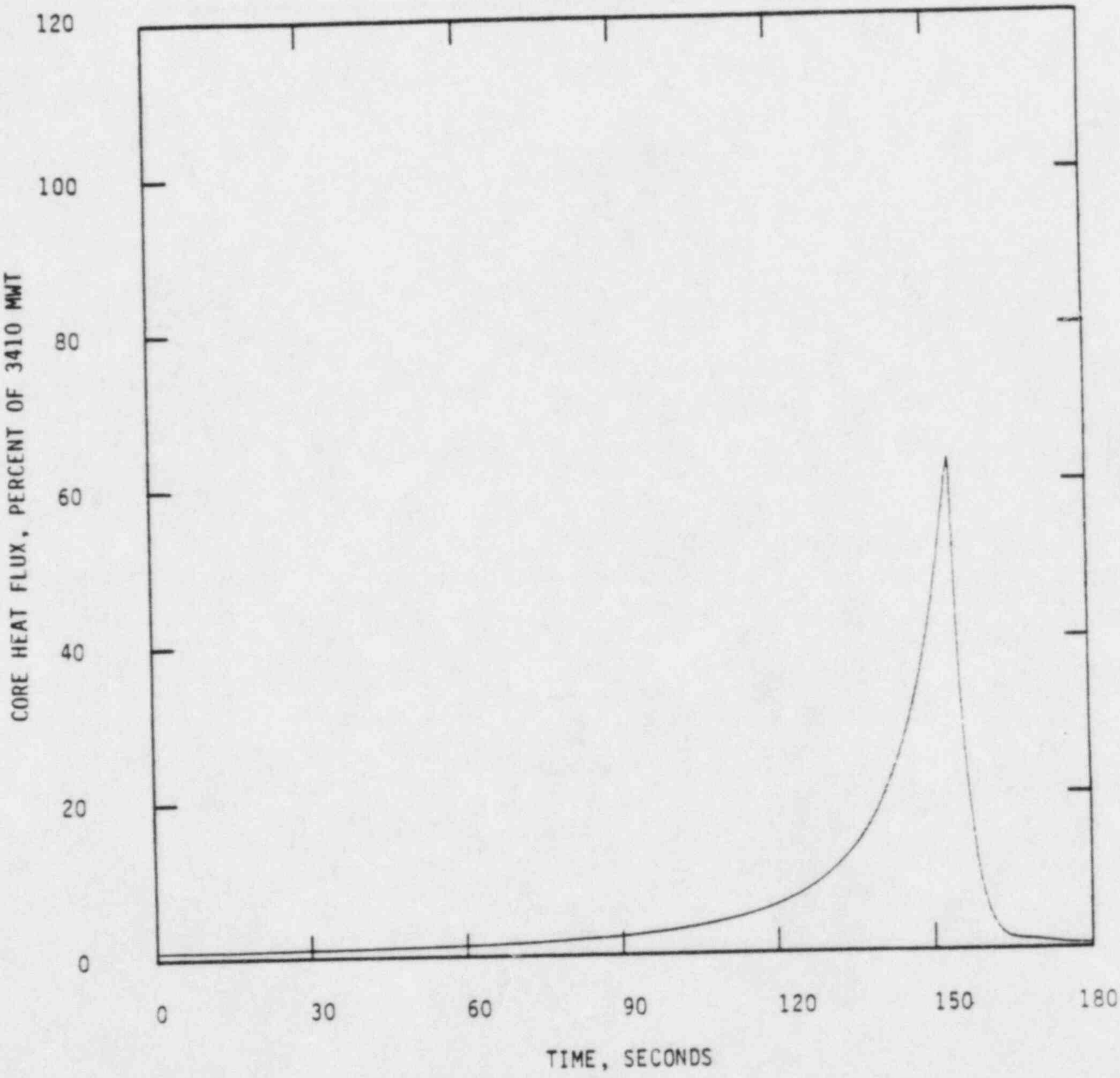
SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
CEA WITHDRAWAL FROM SUBCRITICAL STEAM GENERATOR PRESSURE VS TIME
FIGURE 7.4.1-5



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CEA WITHDRAWAL AT LOW POWER
CORE POWER VS TIME

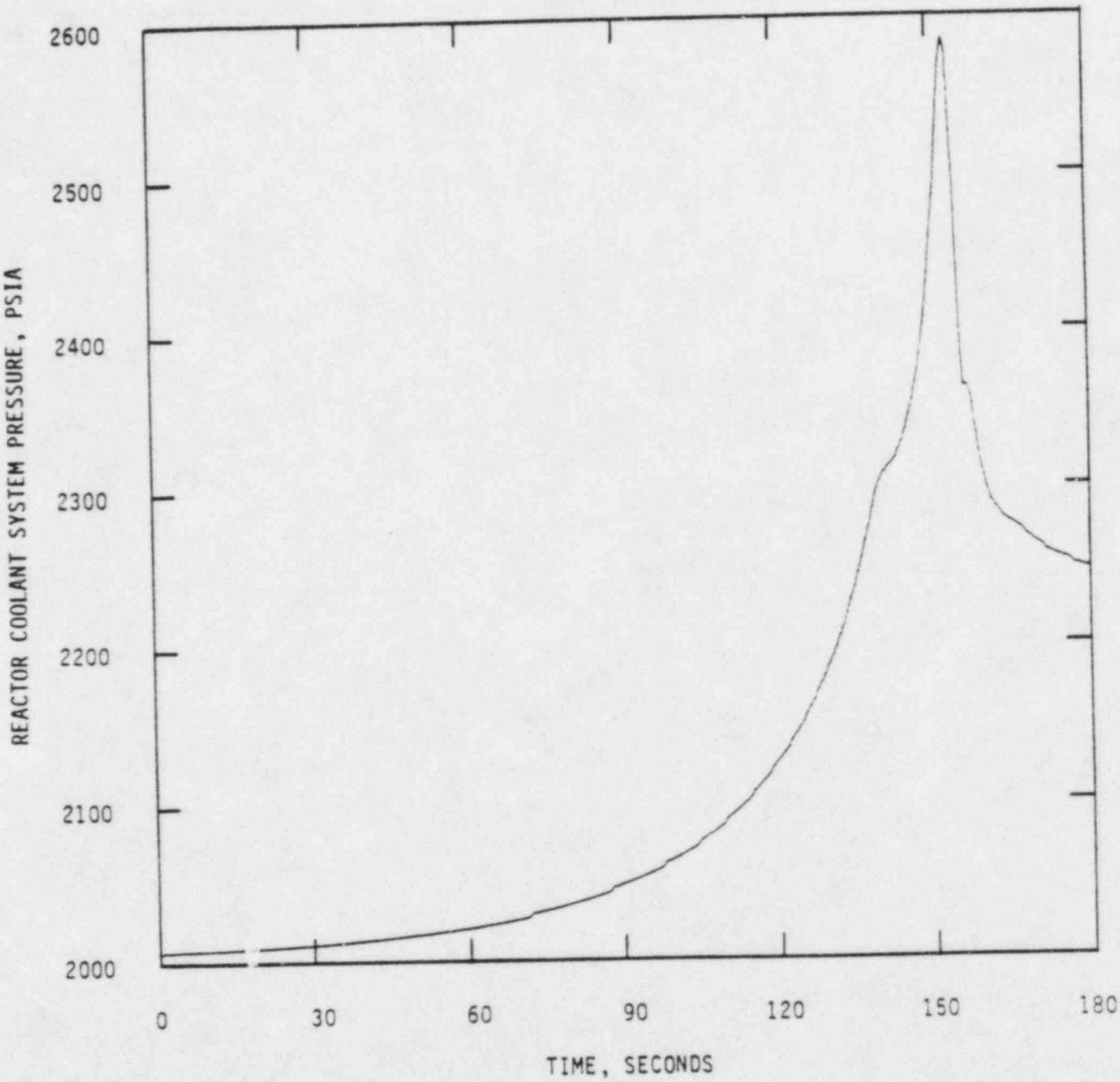
FIGURE 7.4.1-6



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CEA WITHDRAWAL AT LOW POWER
CORE HEAT FLUX VS TIME

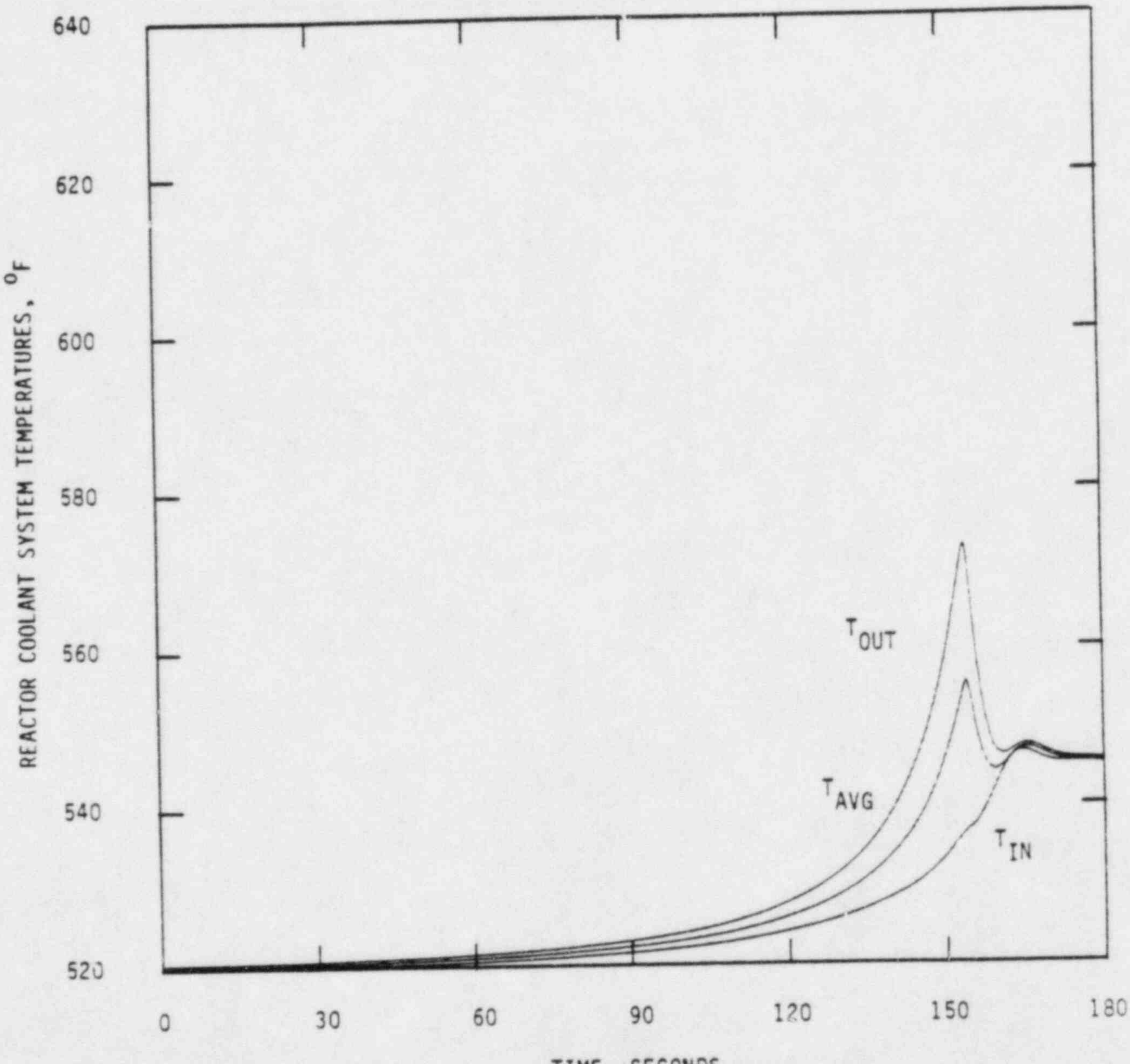
FIGURE 7.4.1-7



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CEA WITHDRAWAL AT LOW POWER
RCS PRESSURE VS TIME

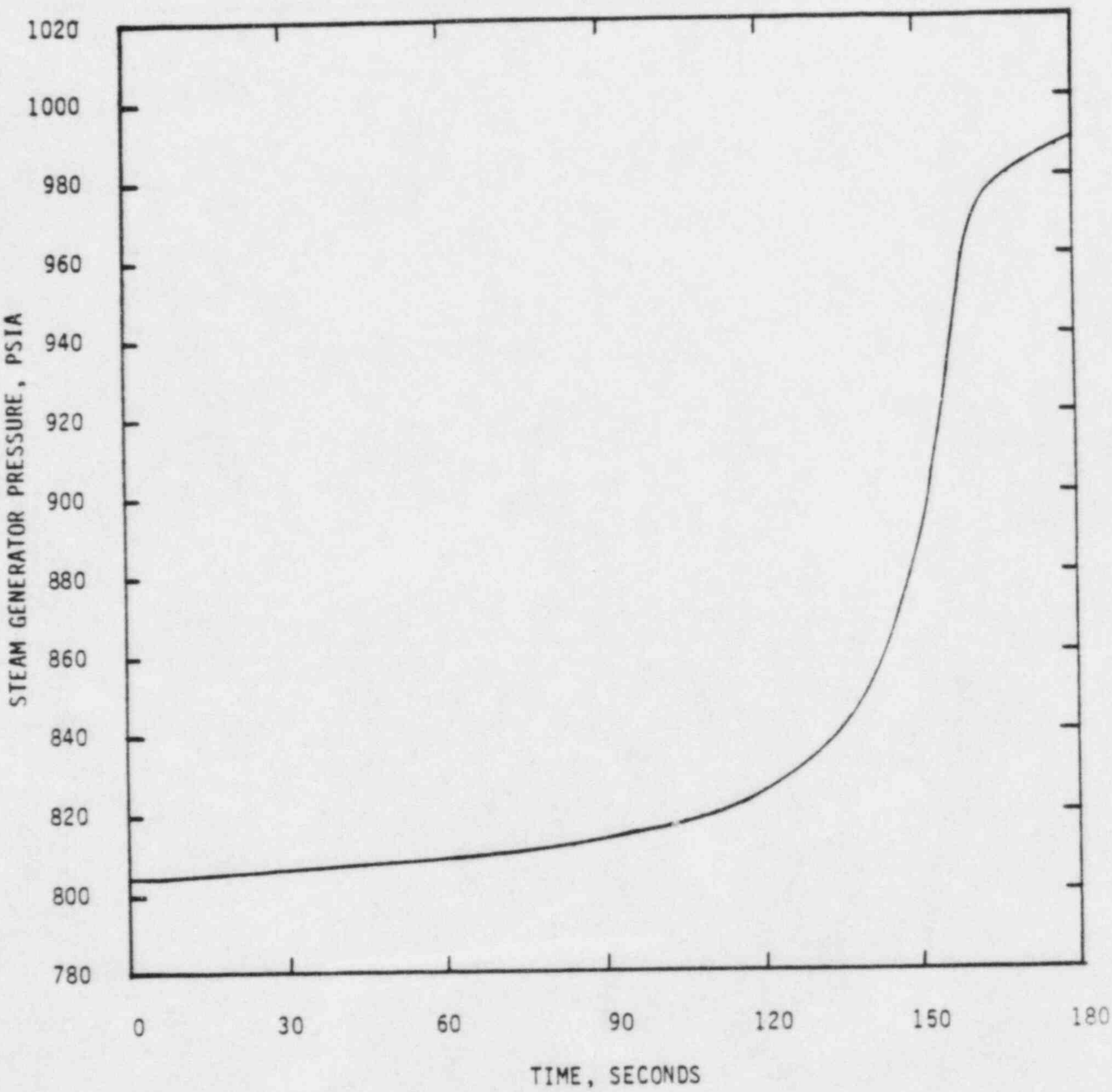
FIGURE 7.4.1-8



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CEA WITHDRAWAL AT LOW POWER
RCS TEMPERATURES VS TIME

FIGURE 7.4.1-9



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CEA WITHDRAWAL AT LOW POWER
STEAM GENERATOR PRESSURE VS TIME

FIGURE 7.4.1-10

7.4.4 CVCS Malfunction (Inadvertent Boron Dilution)

The Inadvertent Boron Dilution event is analyzed for Cycle 3 to demonstrate that sufficient time is available for an operator to identify the cause of and to terminate an approach to criticality for all subcritical modes of operation. The results of the analyses establish corresponding shutdown margin requirements for Modes 3 through 5. This event was reanalyzed on the basis of an increase in critical boron concentrations as shown in Table 7.4.4-1.

7.4.4.1 Identification of Causes

During operation at power (i.e., Modes 1 and 2), an inadvertent boron dilution adds positive reactivity and can cause an approach to the DNBR and CTM limits. The Core Protection Calculator (CPC) trip system monitors the transient behavior of pertinent safety parameters and will generate a reactor trip if necessary to prevent the DNBR and CTM limits from being exceeded. The high pressurizer pressure trip will prevent reaching the RCS pressure upset limit. The trip which is actuated depends on the rate of reactivity addition. For a boron dilution initiated from the low power portion of Mode 2, the power transient resulting from the reactivity insertion would be terminated by the high logarithmic power level trip prior to approaching these limits. For the subcritical modes (i.e., Modes 3 through 6), the time required to achieve criticality due to boron dilution is dependent on the initial and critical boron concentrations, the inverse boron worth, and the rate of dilution.

7.4.4.2 Analysis of Effects and Consequences

Table 7.4.4-1 compares the values of the key transient parameters assumed in each mode of operation for Cycle 3 and the Reference Cycle. The analysis conservatively assumed higher critical boron concentrations and lower inverse boron worths than expected for Cycle 3. These choices decrease the calculated times to criticality in initially subcritical modes. The time to criticality was determined by using the same mathematical expression as in the FSAR. (Reference 7-2, Section 15.4.1.4.3).

7.4.4.3 Results

Table 7.4.4-2 compares the results of the analysis for Cycle 3 with those for the Reference Cycle. The key results are the minimum times required to lose the prescribed negative reactivity in each operational mode. The Cycle 3 results are bounded by the Reference Cycle analysis for Modes 1 through 4. The time to criticality for Modes 5 and 6 have decreased due to an increase in critical boron concentration.

7.4.4.4 Conclusion

The results of this analysis demonstrate that sufficient time exists for the operator to take appropriate action to identify and mitigate the consequences of the Inadvertent Boron Dilution Event.

Table 7.4.4-1

Key Parameters Assumed in the Inadvertent Boron Dilution Event

Parameter	Reference Cycle Value	Cycle 3# Value
Critical Boron Concentration, PPM (All Rods Out, Zero Xenon)		
Power Operation (Mode 1)	1500	2050
Startup (Mode 2)	1500	2050
Hot Standby (Mode 3)	1500	2050
Hot Shutdown (Mode 4)	1500	2050
Cold Shutdown (Mode 5)	1300	2050
Refueling (Mode 6)	1150	1650(1445)##
Inverse Boron Worth, PPM/%Δρ		
Power Operation	70	80
Startup	60	80
Hot Standby	60	80
Hot Shutdown	60	65
Cold Shutdown	60	80
Refueling	N/A	N/A
Minimum Shutdown Margin Assumed, %Δρ		
Power Operation	5.15	5.15
Startup	5.15	4.0
Hot Standby	5.15	4.0
Hot Shutdown	5.15	4.0
Cold Shutdown	3.0	3.0
Refueling	*	*

* For Cycle 3, Technical Specification minimum refueling concentration of 1720 ppm with uncertainty is assumed. Extended Cycle Program (ECP) analysis assumes a refueling boron concentration of 2000 ppm.

\# Values assumed are ECP bounding values unless otherwise indicated.

\## Cycle 3 specific.

Table 7.4.4-2

Results of the Inadvertent Boron Dilution Event

Mode	Time to Lose Minimum Shutdown Margin (Minutes)		Acceptance Criterion To Terminate the Event (Minutes)
	Reference Cycle	Cycle 3	
Startup (Mode 2)	>73	>60	15
Hot Standby (Mode 3)	>73	>60	15
Hot Shutdown (Mode 4)	>73	>60	15
Cold Shutdown (Mode 5)			
RCS Full	>60	>60	15
RCS Partially Drained*	>60	>60	15
Refueling (Mode 6)*	>60	>60	30

*Assumes only one charging pump is operable.

7.4.5 Startup of an Inactive Reactor Coolant Pump Event

The results are bounded by the Reference Cycle.

7.4.6 Control Element Assembly Ejection

The results are bounded by the Reference Cycle.

7.5 Increase in Reactor Coolant System Inventory

7.5.1 Chemical and Volume Control System

The results are bounded by the Reference Cycle.

7.5.2 Inadvertent Operation of the ECCS During Power Operation

The results are bounded by the Reference Cycle.

7.6 Decrease in Reactor Coolant System Inventory

7.6.1 Pressurizer Pressure Decrease Events

The results are bounded by the Reference Cycle.

7.6.2 Small Primary Line Pipe Break Outside Containment

The results are bounded by the Reference Cycle.

7.6.3 Steam Generator Tube Rupture

The results are bounded by the Reference Cycle.

7.7 Miscellaneous

7.7.1 Asymmetric Steam Generator Events

The transients resulting from the malfunction of one steam generator are analyzed to determine the initial margins that must be maintained by the LCO's such that in conjunction with the RPS (CPC high differential cold leg temperature) the DNBR and Fuel Centerline Melt (CTM) limits are not exceeded. This event is presented due to a change in moderator temperature coefficient and a change in analytical methodology.

7.7.1.1 Identification of Causes

The four events which affect a single generator are identified below:

a) Loss of Load to One Steam Generator (LL/1SG)

b) Excess Load to One Steam Generator (EL/1SG)

c) Loss of Feedwater to One Steam Generator (LF/1SG)

d) Excess Feedwater to One Steam Generator (EF/1SG)

Of the four events described above, it has been determined that the Loss of Load to One Steam Generator (LL/1SG) Event is the limiting asymmetric event. Hence, only the results of this transient are reported.

The event is initiated by the inadvertent closure of a Single Main Steam Isolation Valve (MSIV), which results in a loss of load to the affected steam generator. Upon the loss of load to the single steam generator, its pressure and temperature increase to the opening pressure of the secondary safety valves and its water level decreases. The core inlet temperature of the loop with the affected steam generator increases resulting in an asymmetric temperature tilt across the core. The intact steam generator "picks up" the lost load, which causes its temperature and pressure to decrease, and its water level to increase, thus causing the core average inlet temperature to decrease and enhancing the asymmetry in the reactor inlet temperatures. In the presence of a negative moderator temperature coefficient the radial peaking increases in the cold side of the core, resulting in a condition which potentially could cause an approach to DNB and CTM limits. The CPC high differential cold leg temperature trip serves as the primary means of mitigating this transient. Additional protection is provided by the steam generator low level trip.

7.7.1.2 Analysis of Effects and Consequences

The most negative value of the moderator temperature coefficient is assumed to maximize the calculated severity of the asymmetry.

The LL/1SG is initiated at the initial conditions presented in Table 7.7.1-1 and is analyzed parametric on axial shape index to determine the maximum initial margin needed to ensure the SAFDLs are not violated.

The NSSS response is generated with the CESEC code. The resulting core parameters (core flow, RCS inlet temperature, RCS pressure, and reactor trip time) are the input into a 2-D simulation of the core using the HERMITE code. HERMITE is used to model both the effects of the temperature tilt on radial power distribution and the space-time impact of the scram. The thermal margin changes are evaluated with the CETOP code. Information from both HERMITE and CESEC is used to determine the resultant DNBR.

7.7.1.3 Results

A reactor trip is generated by the CPC's at 6.0 seconds based on high differential cold leg temperature between the cold legs associated with the steam generators.

Table 7.7.1-2 presents the sequence of events for the loss of load to one steam generator. Figures 7.7.1-1 to 7.7.1-5 show the NSSS response for core power, core heat flux, RCS temperatures, RCS pressure, and steam generator pressure. The minimum transient DNBR calculated for the LL/1SG Event is greater than 1.31.

A maximum allowable initial linear heat generation rate of 17.0 kW/ft could exist as an initial condition without exceeding the Acceptable Fuel to Centerline Melt Limit of 21.0 kW/ft during this transient. This amount of margin is assured by setting the linear heat rate LCO based on the more limiting allowable linear heat rate for LOCA (13.9 kW/ft, see Table 7.0-6).

7.7.1.4 Conclusions

This event initiated from the Technical Specification LCO's will not exceed the DNBR and CTM limits.

Table 7.7.1-1

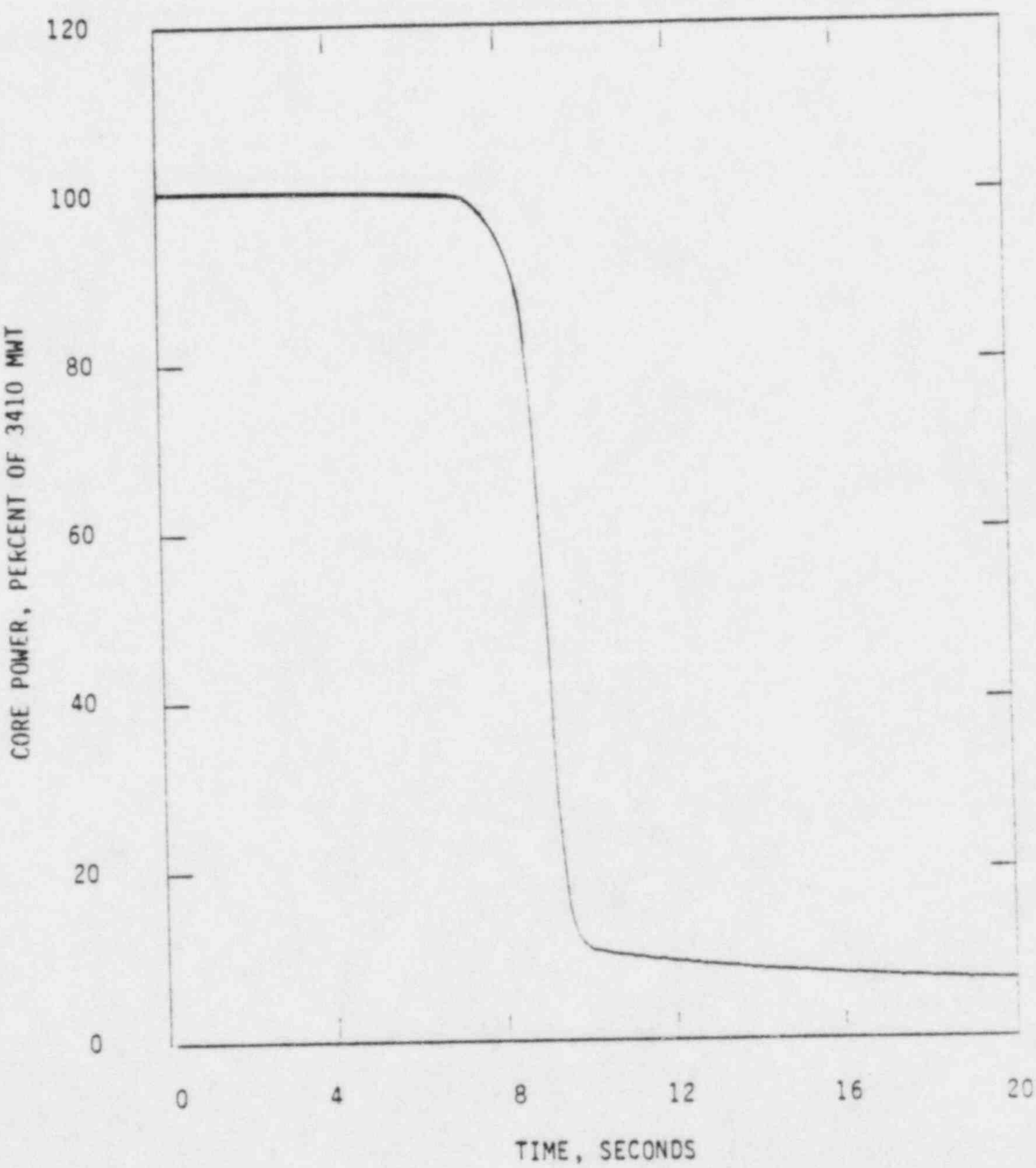
Key Parameters Assumed for the
Loss of Load to One Steam Generator Event

Parameter	Units	Reference Cycle Value	Cycle 3 Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	3478	3478
Initial Core Inlet Temperature	^{o}F	553	553
Initial Reactor Coolant System Pressure	psia	2250	2250
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/^{o}F$	-2.5	-3.3
Doppler Coefficient Multiplier		0.85	0.75
Radial Distortion Factor for a $18^{o}F$ Core Inlet Temperature Asymmetry		1.158	1.130

Table 7.7.1-2

Sequence of Events for the Loss of Load to One Steam Generator Event

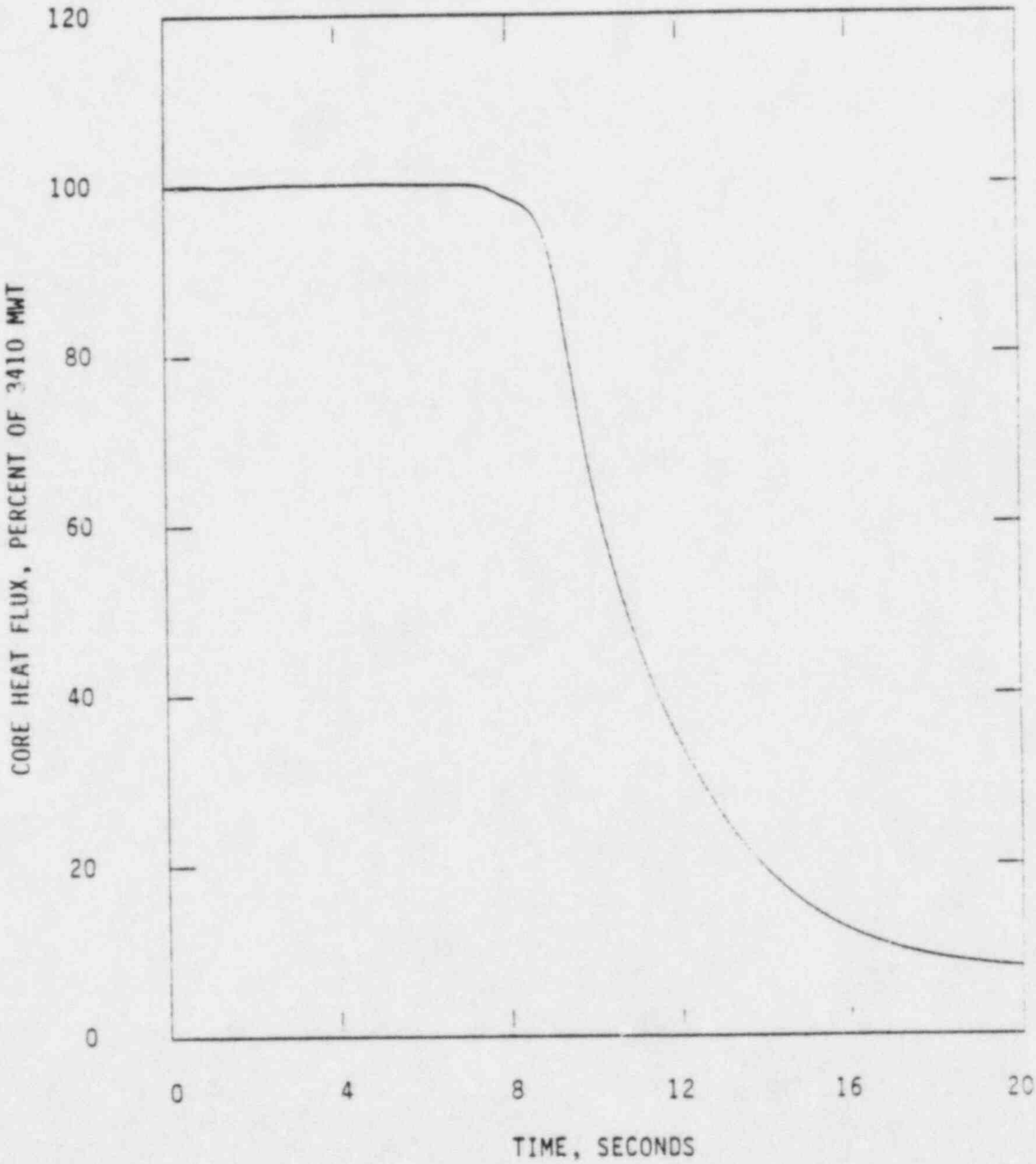
Time (sec)	Event	Setpoint or Value
0.0	Spurious Closure of a Single Main Steam Isolation Valve (MSIV)	----
0.1	MSIV on Affected Steam Generator is Closed	----
0.1	Steam Flow from Unaffected Steam Generator Increases to Maintain Turbine Power	----
6.0	CPC Delta-T Setpoint Reached (Differential Cold Leg Temperature)	18°F
6.1	Safety Valves Open on Isolated Steam Generator	1100 psia
6.25	Trip Breakers Open	---
6.55	CEAs Begin to Drop into Core	----
7.15	Minimum DNBR Occurs	$\geq$ 1.31
10.9	Maximum Steam Generator Pressure	1135 psia



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

ASYMMETRIC STEAM GENERATOR EVENT
CORE POWER VS TIME

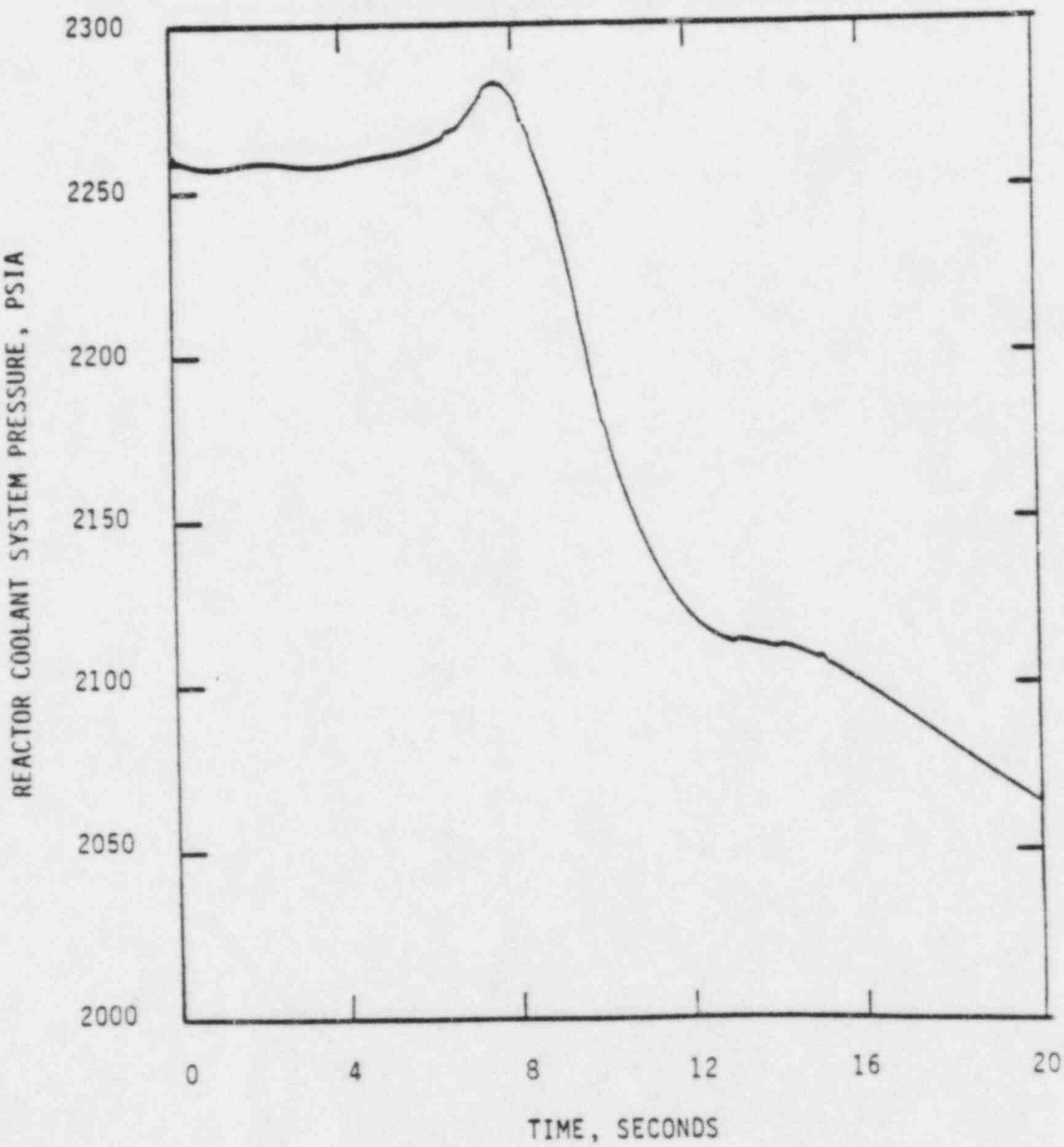
FIGURE 7.7.1-1



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3

ASYMMETRIC STEAM GENERATOR EVENT
CORE HEAT FLUX VS TIME

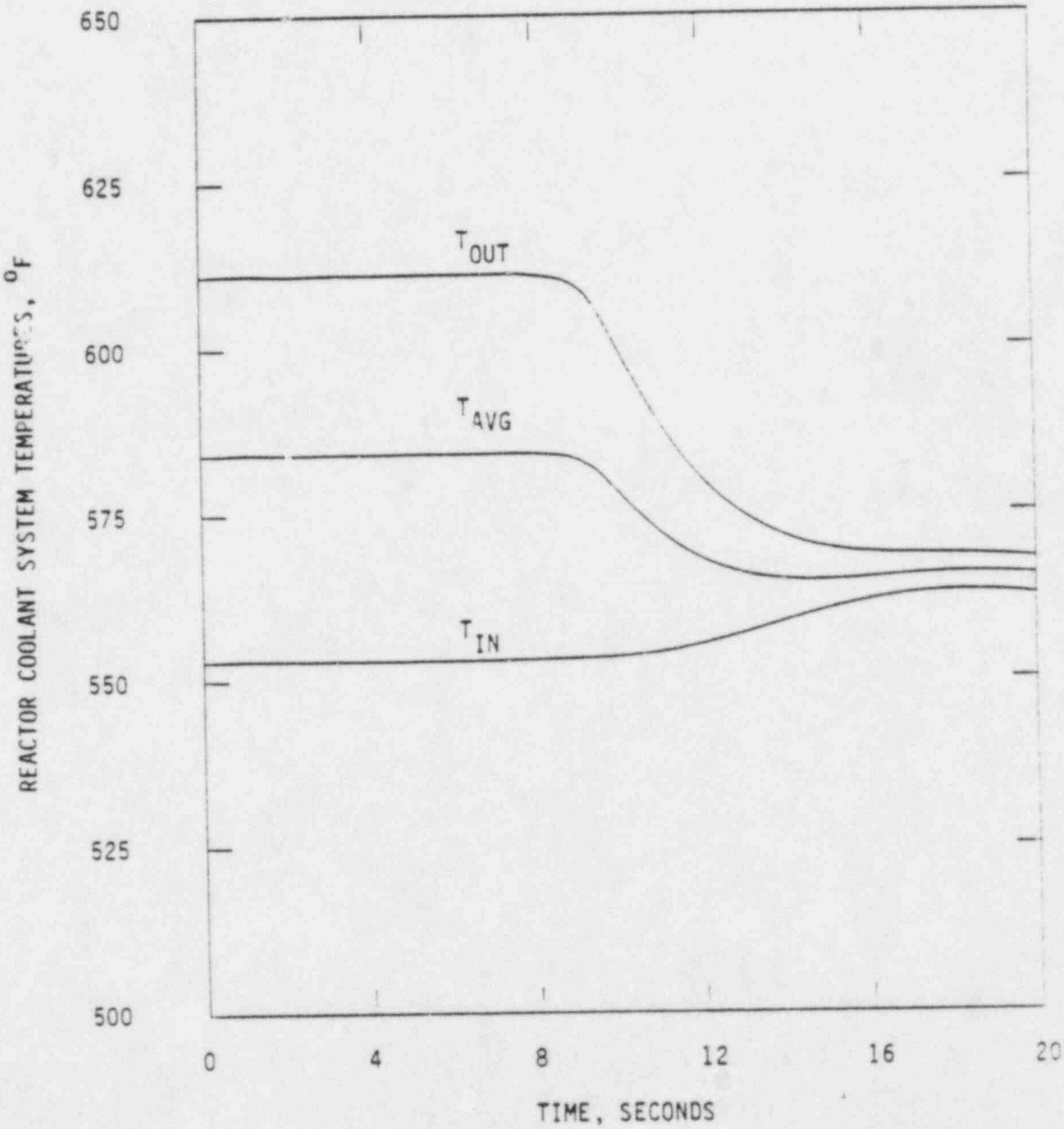
FIGURE 7.7.1-2



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

ASYMMETRIC STEAM GENERATOR EVENT
RCS PRESSURE VS TIME

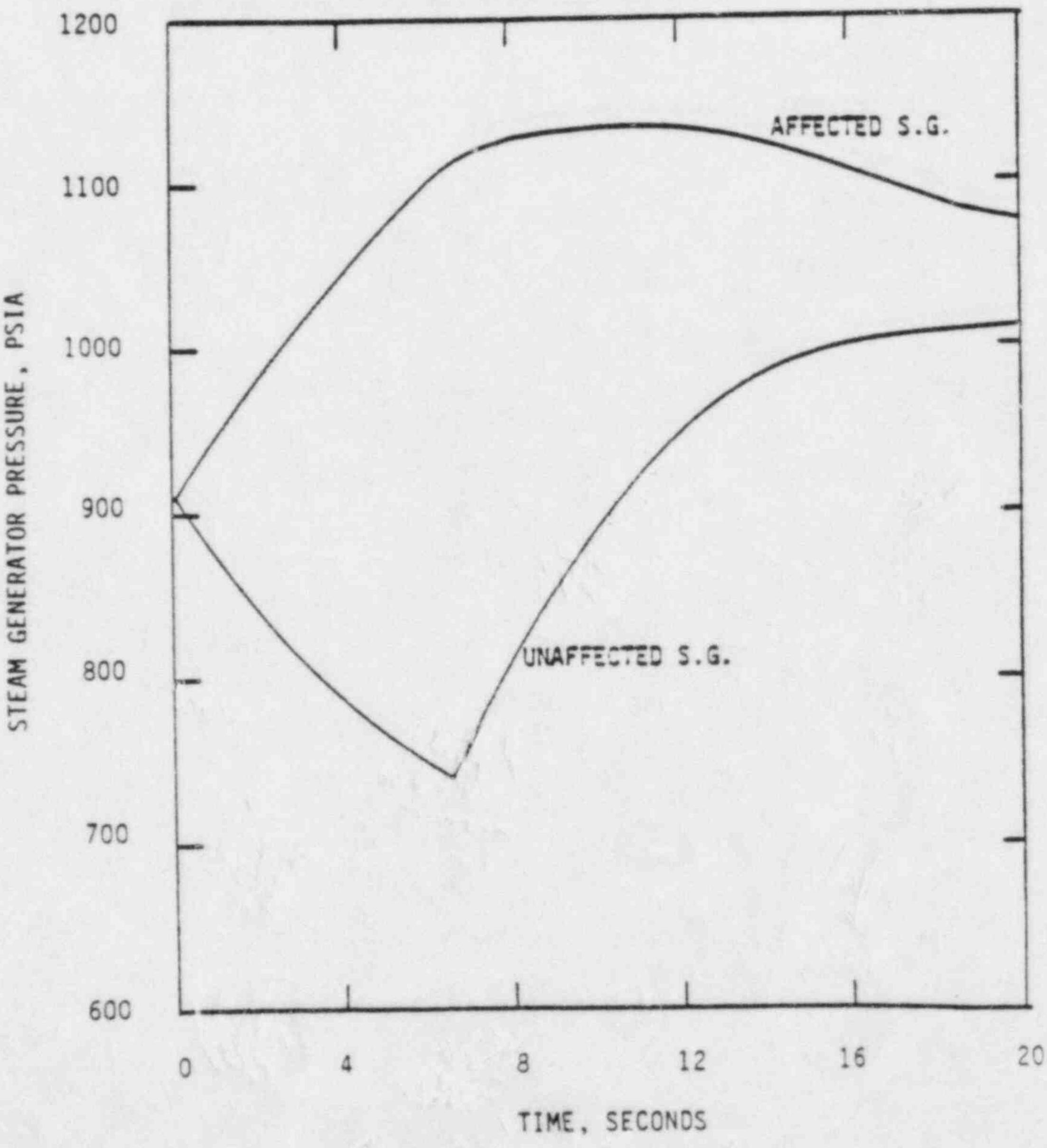
FIGURE 7.7.1-3



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

ASYMMETRIC STEAM GENERATOR EVENT
RCS TEMPERATURES VS TIME

FIGURE 7.7.1-4



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

ASYMMETRIC STEAM GENERATOR EVENT
STEAM GENERATOR PRESSURE VS TIME

FIGURE 7.7.1-5

8.0 ECCS Analysis

8.1 Introduction and Summary

An ECCS performance analysis was performed for SONGS-2 Cycle 3 to demonstrate compliance with 10CFR50.46 which presents the NRC Acceptance Criteria for Emergency Core Cooling Systems for Light Water-Cooled reactors (Reference 8-1). The analysis justifies an allowable peak linear heat generation rate (PLHGR) of 13.9 kw/ft. This PLHGR is equal to the existing limit for SONGS Unit 2. The method of analysis and detailed results which support this value are presented herein.

8.2 Method of Analysis

The method of analysis is identical to the Reference Cycle large break LOCA ECCS performance analysis (Reference 8-2). As in the Reference Cycle, the calculations performed for this evaluation used the NRC approved C-E large break ECCS performance evaluation model which is described in References 8-3 through 8-8. Blowdown and refill/reflood hydraulics and hot rod temperature calculations were performed with the fuel parameters which bound Cycle 3 at a reactor power level of 3458 Mwt. The blowdown hydraulic calculations were performed with the CEFLASH-4A (Reference 8-5) code while the refill/reflood hydraulic calculations were performed with the COMPERC-II (Reference 8-6) code. The hot rod clad temperature and clad oxidation calculations were performed with the STRIKIN-II (Reference 8-7) and PARCH (Reference 8-8) codes. Fuel performance calculations were performed using the FATES 3A version of C-E's fuel performance code (Reference 8-9 and 8-10) as approved by the NRC (Reference 8-11) with the fuel grain size restriction. Core wide clad oxidation calculations were also per d in this analysis.

The significant core and system parameters for Cycle 3 and the Reference Cycle are shown in Table 8-1. The Reference Cycle used the C-E generic blowdown analysis for the 3400 Mwt class plants which conservatively bound the SONGS blowdown characteristics. However, a SONGS specific blowdown analysis was performed for Cycle 3 to account for the steam generator tube plugging. This resulted in additional input parameter

differences between Cycle 3 and the Reference Cycle as shown in Table 8-1. The major differences between the Reference Cycle and the Cycle 3 analysis are the fuel performance characteristics, steam generator tube plugging, lowering of minimum initial containment pressure, initial core inlet temperature and the core bypass flow. The other ECCS analysis input parameters are essentially the same as those of the Reference Cycle.

The Cycle 3 analysis accounts for steam generator U-tube plugging of 1000 average length tubes per steam generator. Steam generator U-tube plugging increases system resistance to flow and hence the ability of the Reactor Coolant System (RCS) to vent steam during reflood. The analysis also accounts for the decreased heat transfer area and primary side coolant volume caused by the tube plugging.

Additionally, to provide operational flexibility the minimum containment pressure used was lowered from 14.40 psia to 13.7 psia.

8.3 Results

Table 8-2 presents the analysis results for the 1.0 DEG/PD* break which produces the highest peak clad temperature. For comparison the results of the Reference Cycle are also presented. The results of the evaluation confirm that 13.9 kw/ft is an acceptable value for the PLHGR LCO in Cycle 3. The peak clad temperature and maximum local and core wide clad oxidation values, as shown in Table 8-2, are well below the 10CFR50.46 acceptance limits of 2200°F, 17%, and 1%, respectively. Table 8-3 presents a list of the significant parameters displayed graphically for the limiting 1.0 DEG/PD break.

*DEG/PD = Double-Ended Guillotine at Pump Discharge

Burnup dependent hot rod calculations were performed with STRIKIN-II to determine the initial fuel conditions which results in the highest peak clad temperature (PCT). This study demonstrated that the burnup with the highest initial fuel stored energy results in the highest PCT. This occurred at a hot rod burnup of 1000 MWD/MTU.

The 1.0 DEG/PD break produced the highest peak clad temperature of 2116°F. For the 1.0 DEG/PD break the peak local oxidation (PLO) was calculated to be 10.08%. The 1.0 DEG/PD also resulted in the highest core wide clad oxidation of less than 0.68% which is well below the 1% NRC acceptance criterion.

A review of the effects of initial operating conditions on these results was performed. It was determined that over the ranges of initial operating conditions as specified in the Technical Specifications (Section 10), operation of the plant at a PLHGR of 13.9 kw/ft is acceptable for Cycle 3.

8.4 Conclusion

The results of the ECCS performance evaluation for SONGS Unit 2, Cycle 3 demonstrated a peak clad temperature of 2116°F, a peak local clad oxidation percentage of 10.08% and a peak core wide clad oxidation percentage of less than 0.68% compared to the acceptance criteria of 2200°F, 17% and 1%, respectively. Therefore, operation of SONGS Unit 2 Cycle 3 at a core power level of 3458 Mwt (102% of 3390 MWt) and a PLHGR of 13.9 kw/ft is in conformance with 10CFR50.46.

Table 8-1

SONGS Unit 2 Cycle 3 Core and System Parameters

Parameter (Units)	Reference Cycle	Cycle 3
Average Linear Hear Rate @ 102% of of Nominal (kw/ft)	5.6	5.76
Peak Linear Heat Generation Rate (kw/ft)	13.9	13.9
Core Inlet Temperature (^{o}F)	557.5	553
Core Outlet Temperature (^{o}F)	618.6	613.5
System Flow Rate (lbm/hr)	$148.0X10^{6}$	$148.0X10^{6}$
Core Flow Rate (lbm/hr)	$142.8X10^{6}$	$143.6X10^{6}$
Gap Conductance[1] (BTU/hr-ft^{2}-^{o}F)	1590.0	1639.0
Fuel Centerline Temperature[1] (^{o}F)	3411.0	3429.0
Fuel Average Temperature[1] (^{o}F)	2154.0	2155.8
Hot Rod Gas Pressure[1] (^{o}F)	1131.0	1141.40
Hot Rod Burnup (MWD/MTU)	998.0	1000
Number of Steam Generator Tubes Plugged per S.G.	100	1000
Minimum Initial Containment Pressure (psia)	14.40	13.70

(1) Initial value at the limiting hot rod burnup as calculated by STRIKIN-II at 13.9 kw/ft.

Table 8-2

SONGS Unit 2 Cycle 3

Limiting Break Size (1.0 DEG/PD)

	Reference Cycle	Cycle 3
Peak Linear Heat Generation Rate (kw/ft)	13.9	13.9
Peak Clad Temperature (^{o}F)	2015.0	2116.0
Time of Peak Clad Temperature (Seconds)	257.0	264.0
Time of Clad Rupture (Seconds)	70.50	68.80
Peak Local Clad Oxidation (%)	10.46	10.08
Total Core-Wide Clad Oxidation (%)	<0.68	<0.68

Table 8-3

SONGS Unit 2 Cycle 3

Variables Plotted as a Function of Time

for the Limiting Large Break

Variable	Figure Designation
Core Power	8-1
Pressure in Center Hot Assembly Node	8-2
Leak Flow	8-3
Hot Assembly Flow (below hot spot)	8-4
Hot Assembly Flow (above hot spot)	8-5
Hot Assembly Quality	8-6
Containment Pressure	8-7
Mass Added to Core During Reflood	8-8
Peak Clad Temperature	8-9
Hot Spot Gap Conductance	8-10
Peak Local Clad Oxidation	8-11
Clad Temperature, Centerline Fuel Temperature, Average Fuel Temperature and Coolant Temperature for Hottest Node	8-12
Hot Spot Heat Transfer Coefficient	8-13
Hot Rod Internal Gas Pressure	8-14

FIGURE 8-1
CORE POWER
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

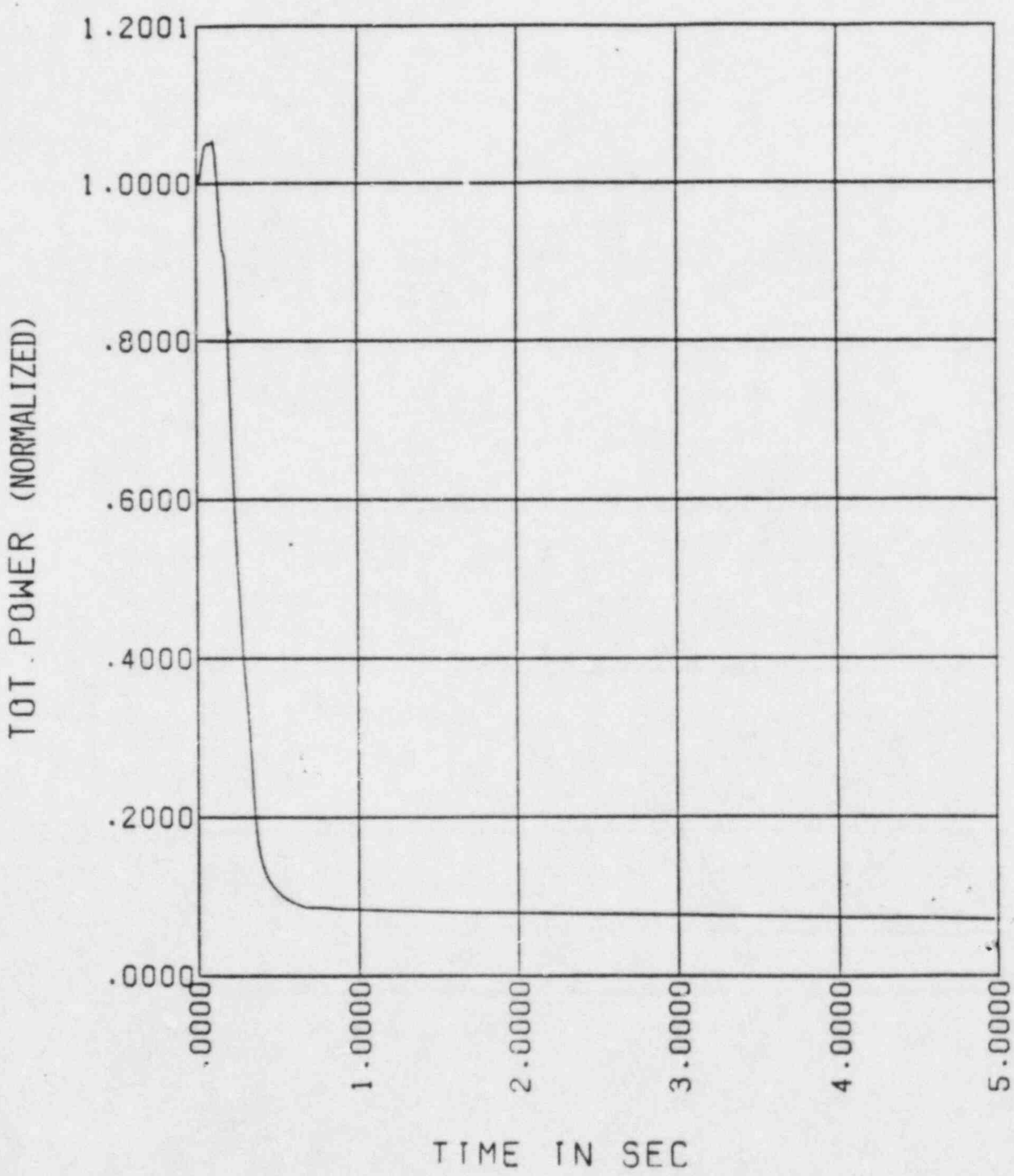


FIGURE 8-2
PRESSURE IN CENTER HOT ASSEMBLY NODE
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

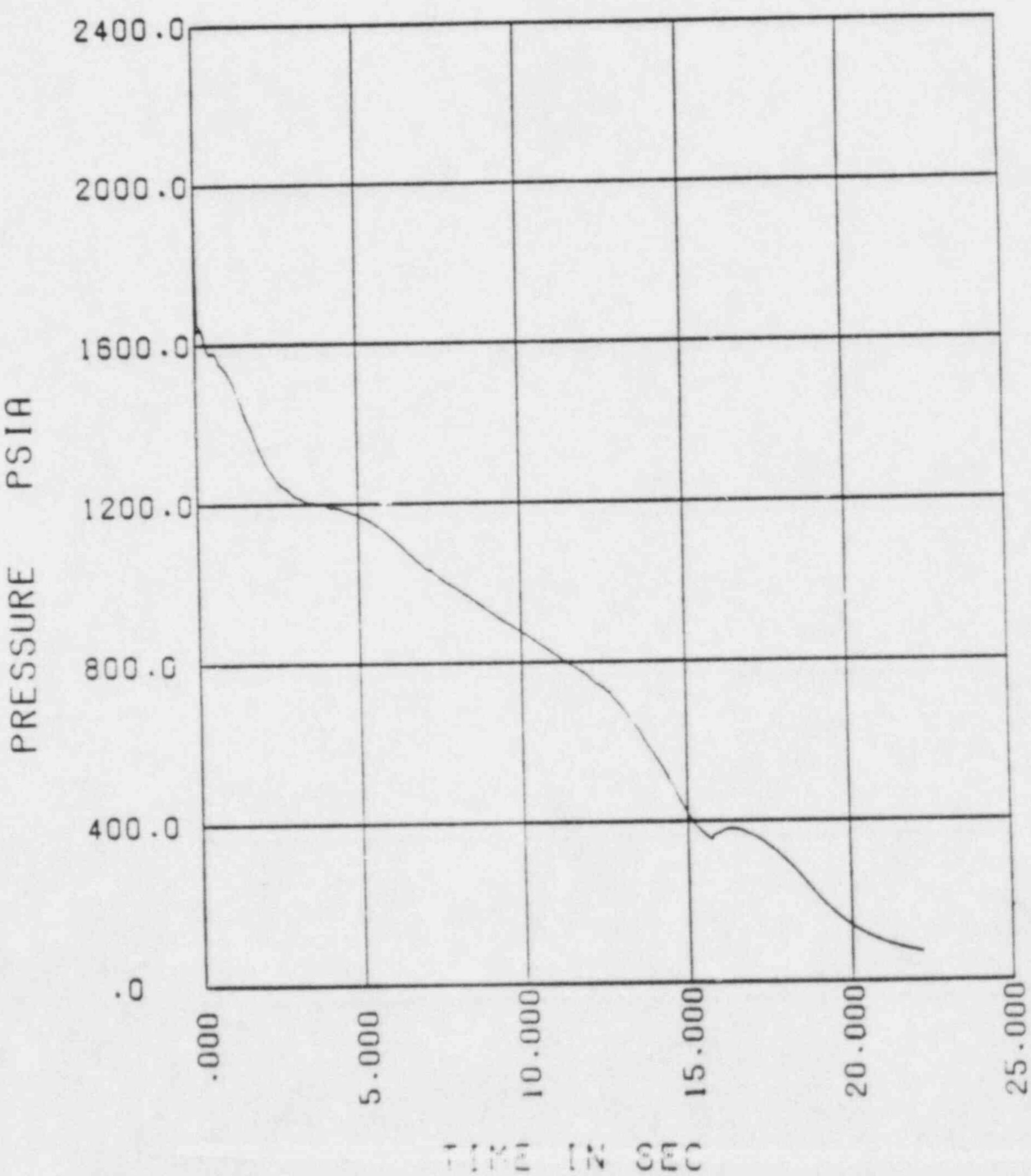


FIGURE 8-3

LEAK FLOW
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

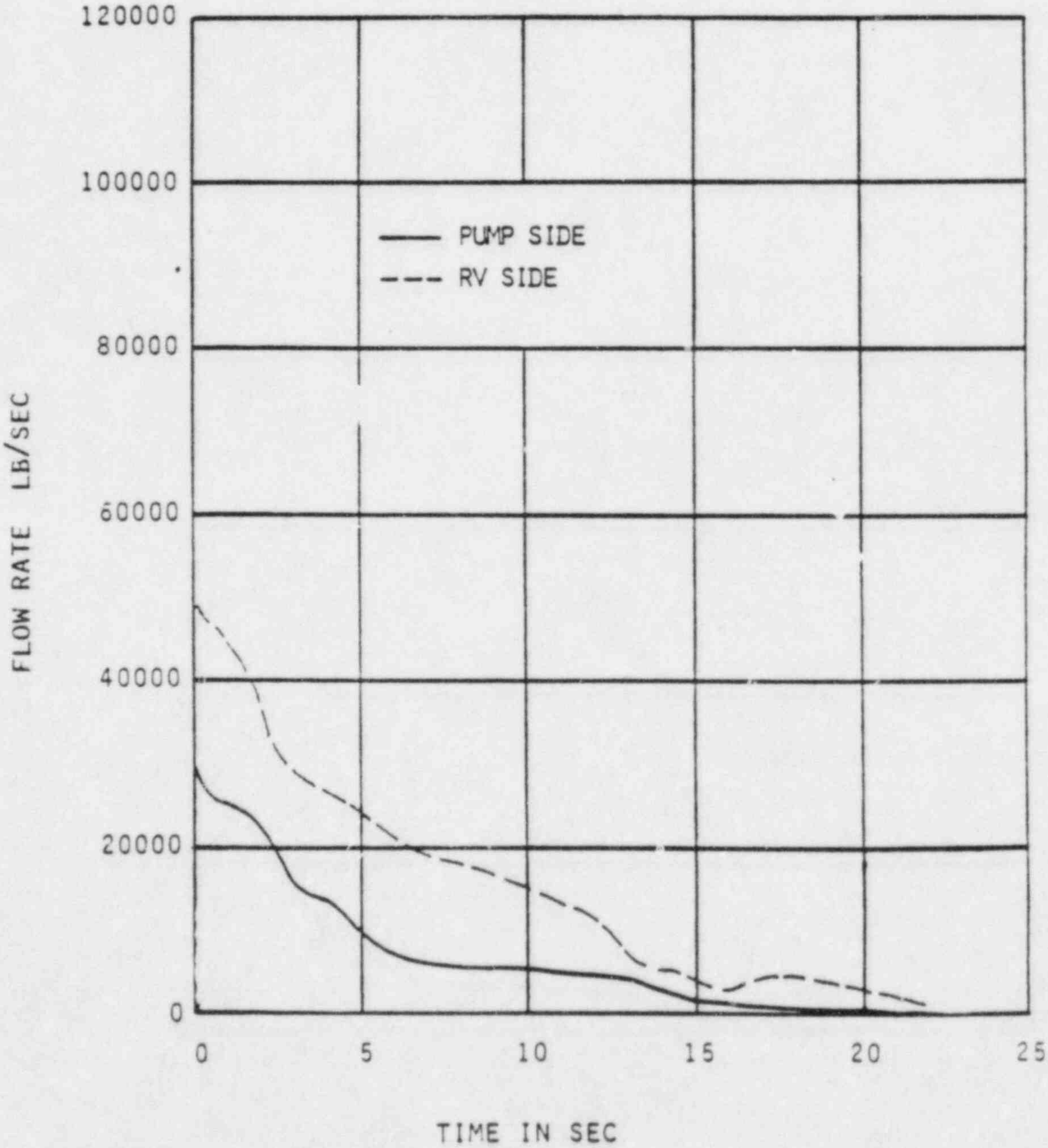


FIGURE 8-4

FLOW IN HOT ASSEMBLY-PATH 16, BELOW HOT SPOT
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

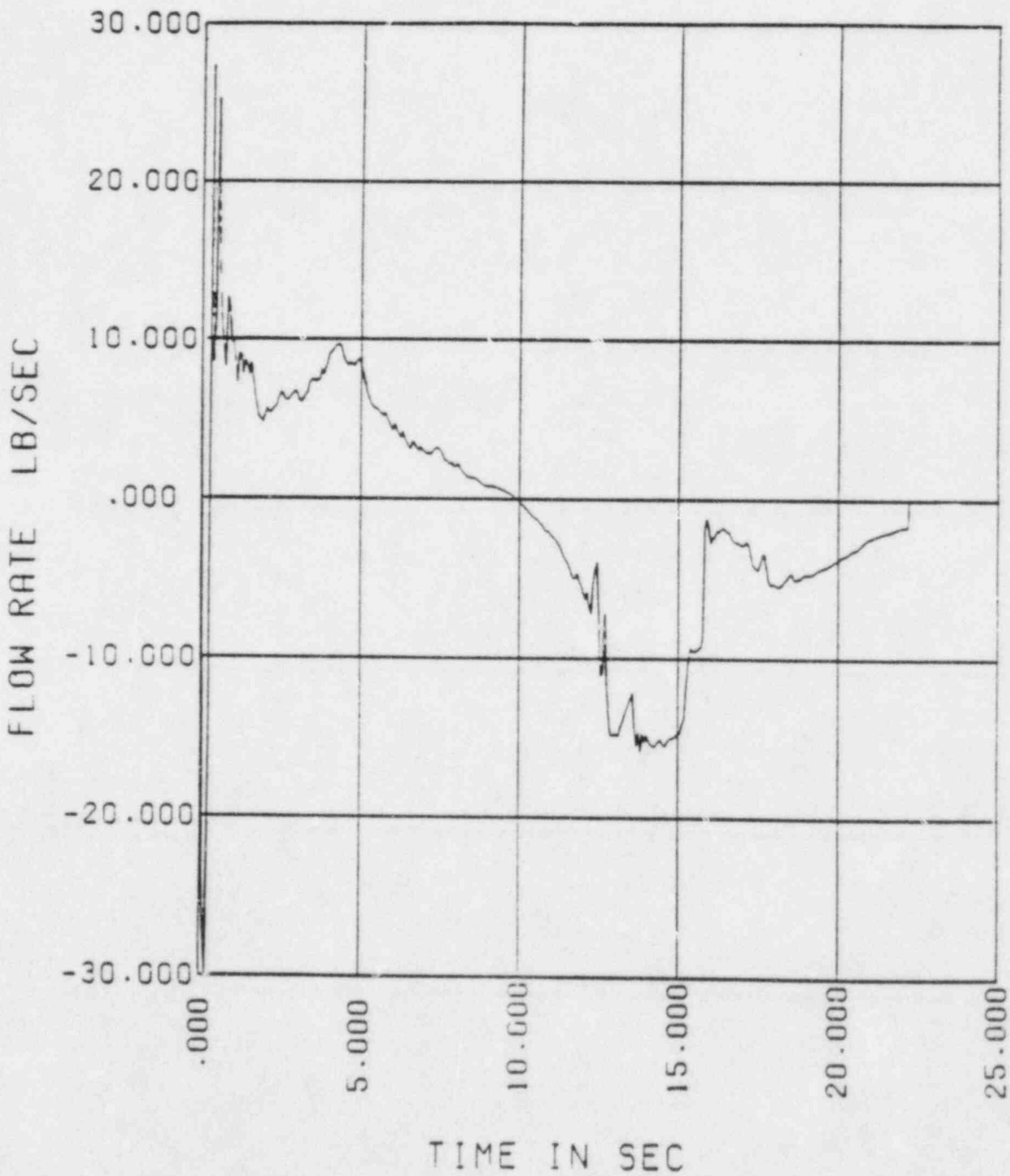


FIGURE 8-5

FLOW IN HOT ASSEMBLY-PATH 17, ABOVE HOT SPOT
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

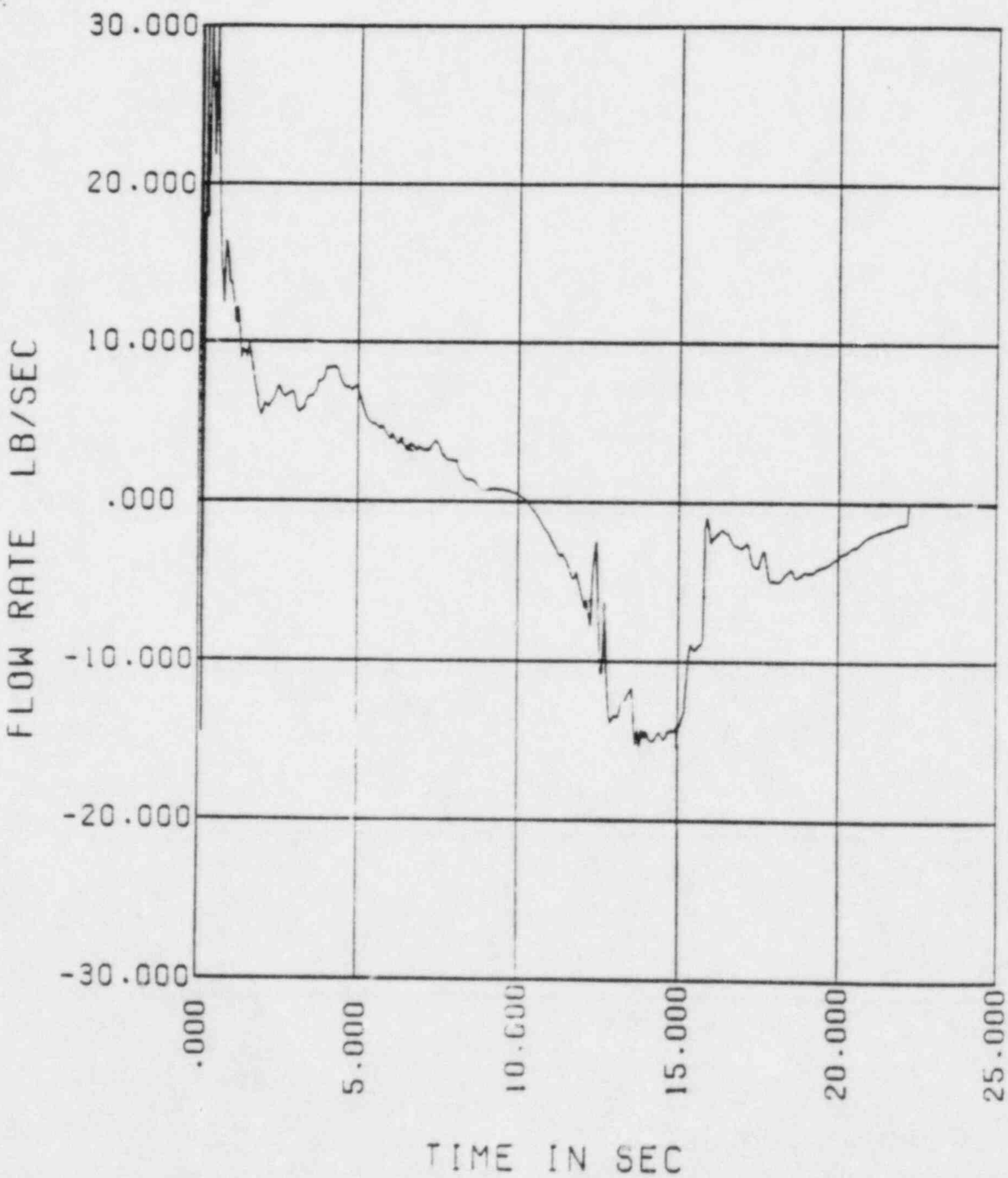


FIGURE 8-6

HOT ASSEMBLY QUALITY

1.0 X DOUBLE ENDED GUILLOTINE BREAK

IN PUMP DISCHARGE LEG

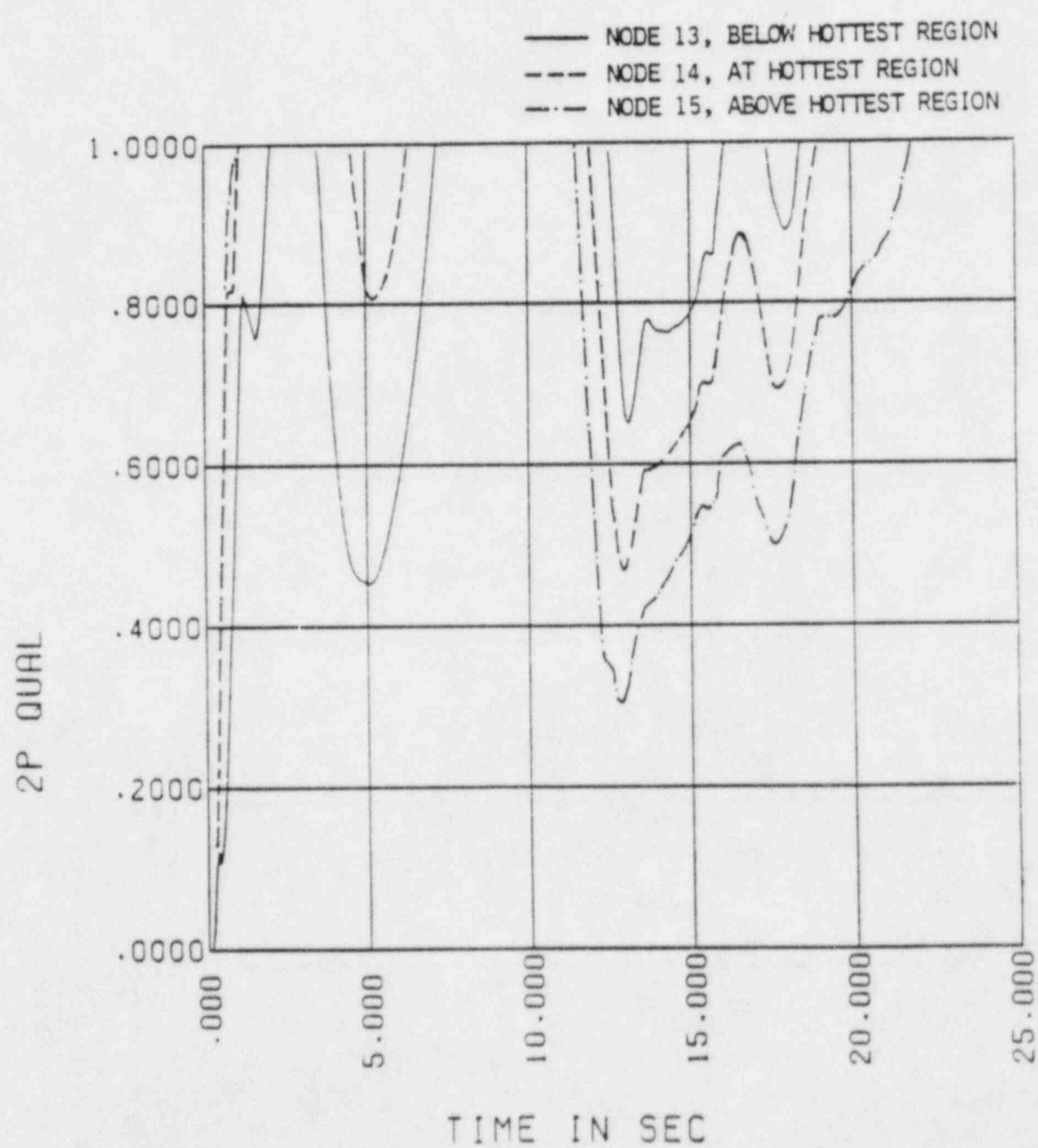


FIGURE 8-7

CONTAINMENT PRESSURE

1.0 X DOUBLE ENDED GUILLOTINE BREAK

IN PUMP DISCHARGE LEG

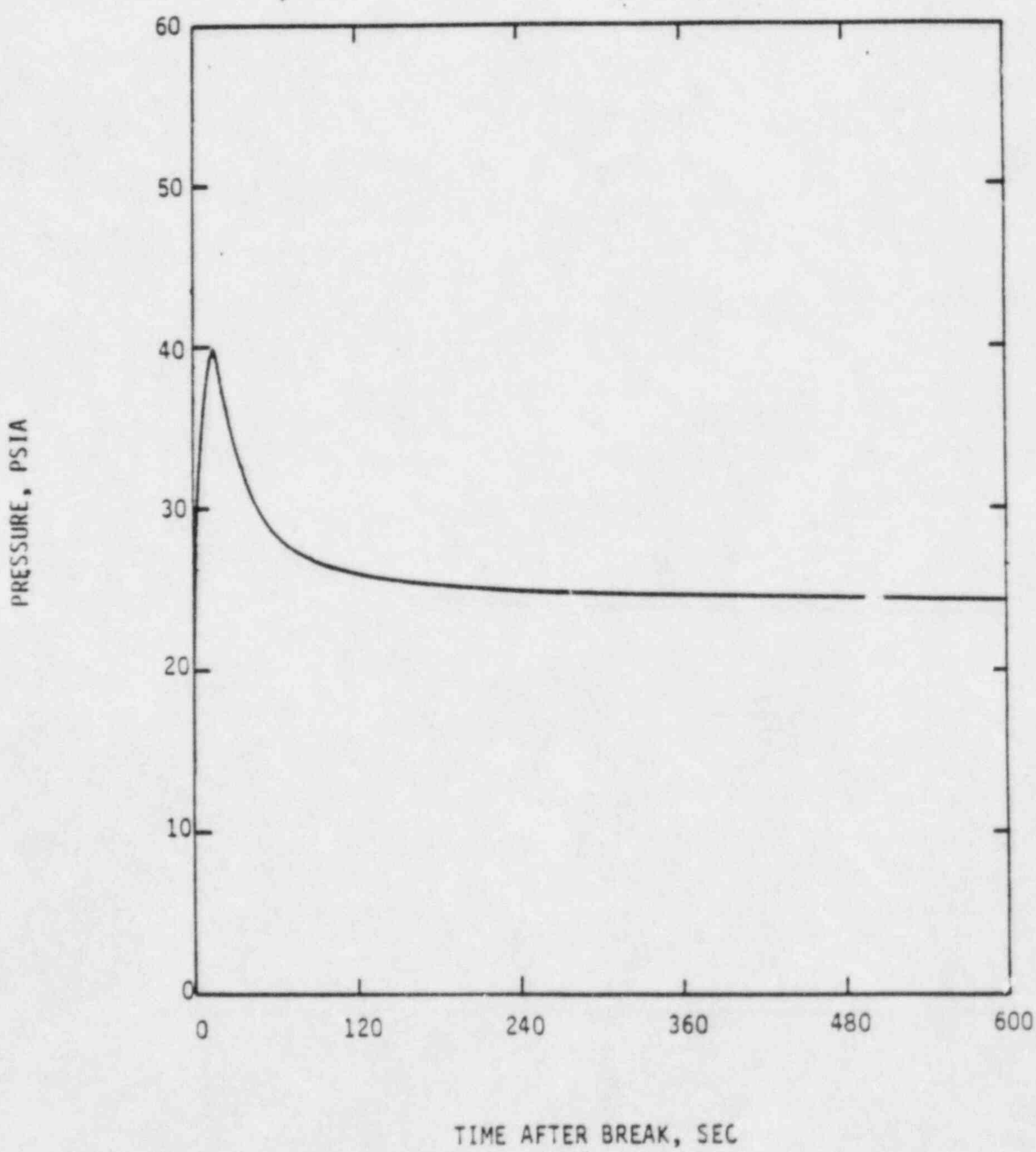
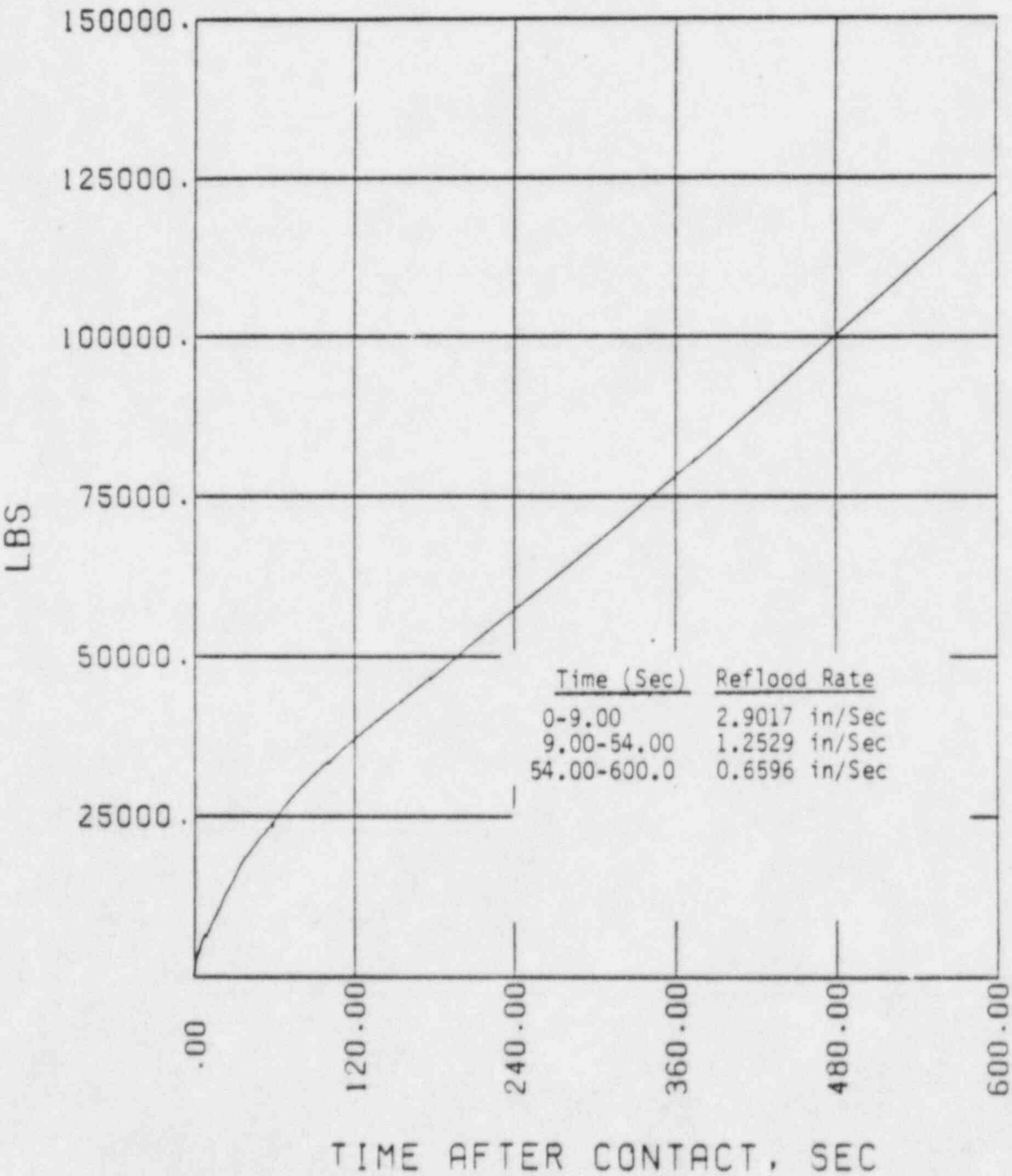


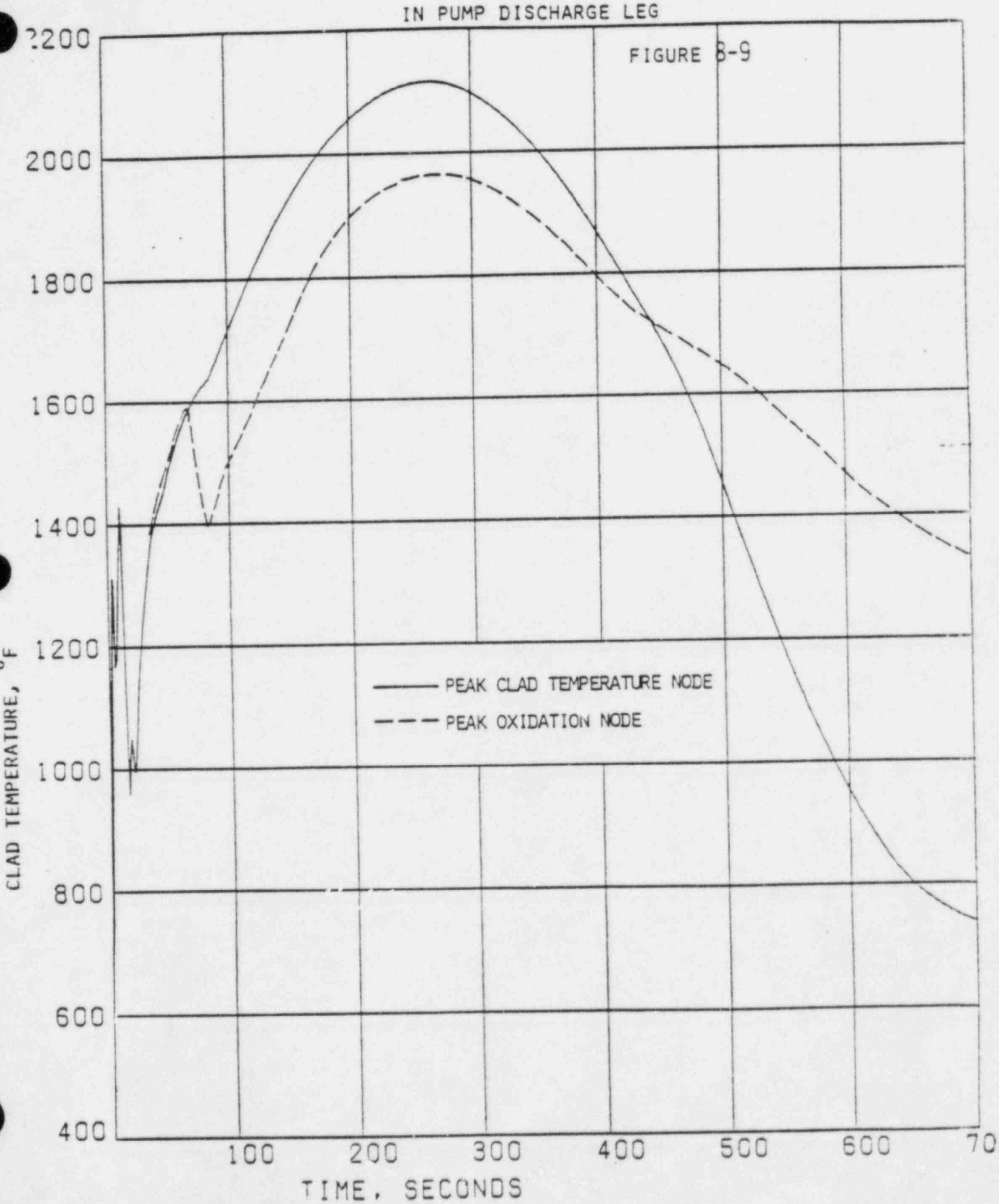
FIGURE 8-8

MASS ADDED TO CORE DURING REFLOOD
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG



PEAK CLAD TEMPERATURE
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

FIGURE 8-9



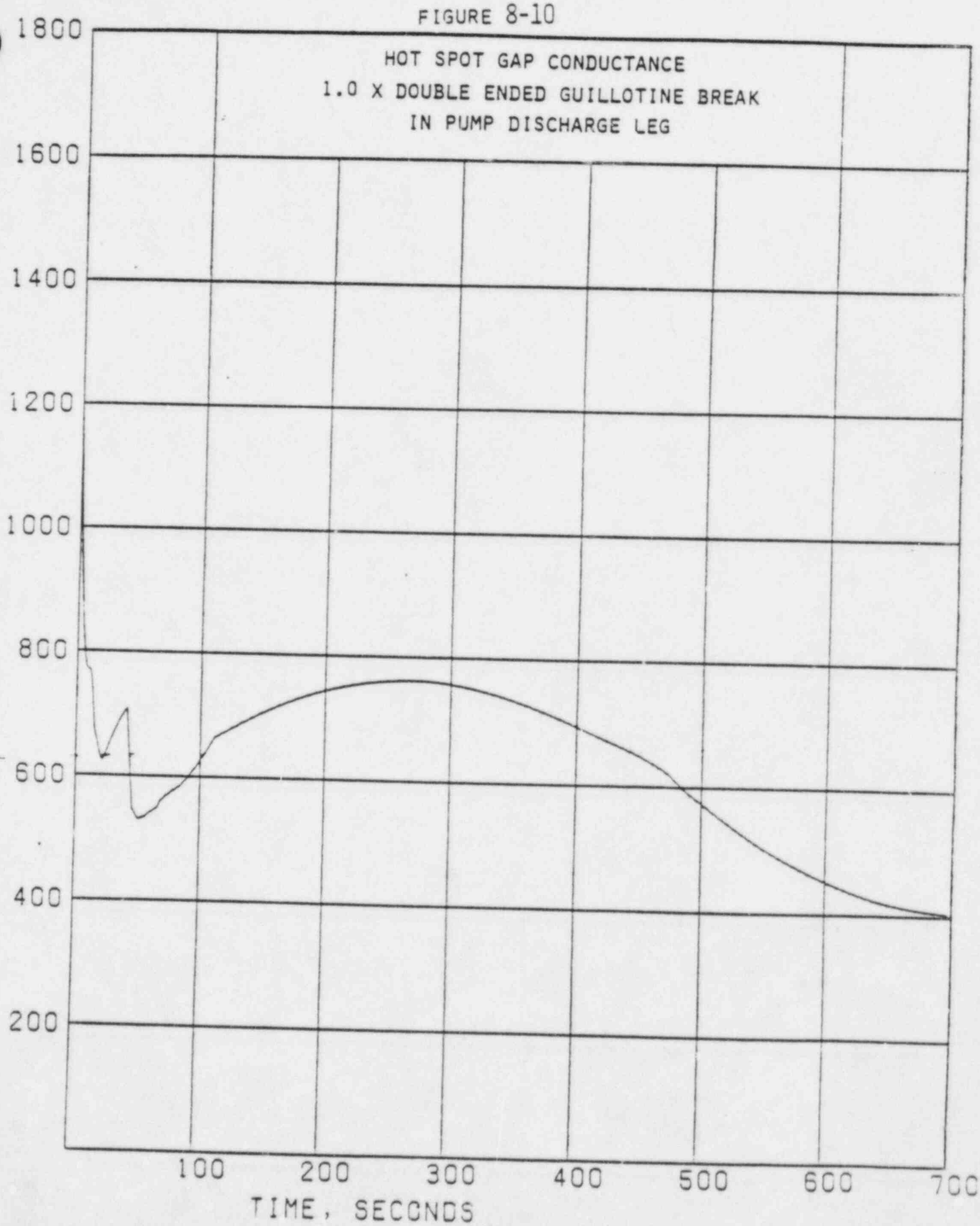
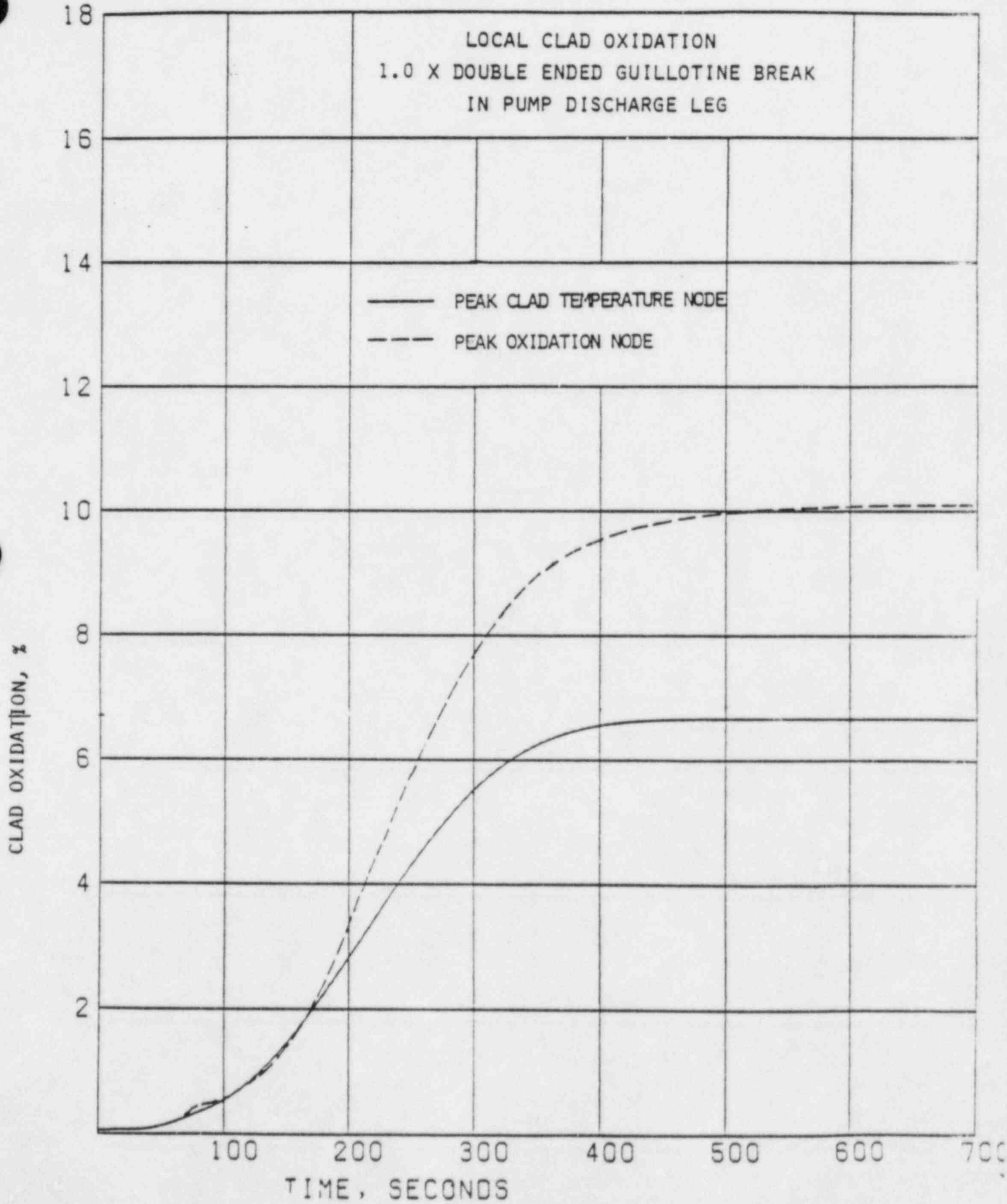


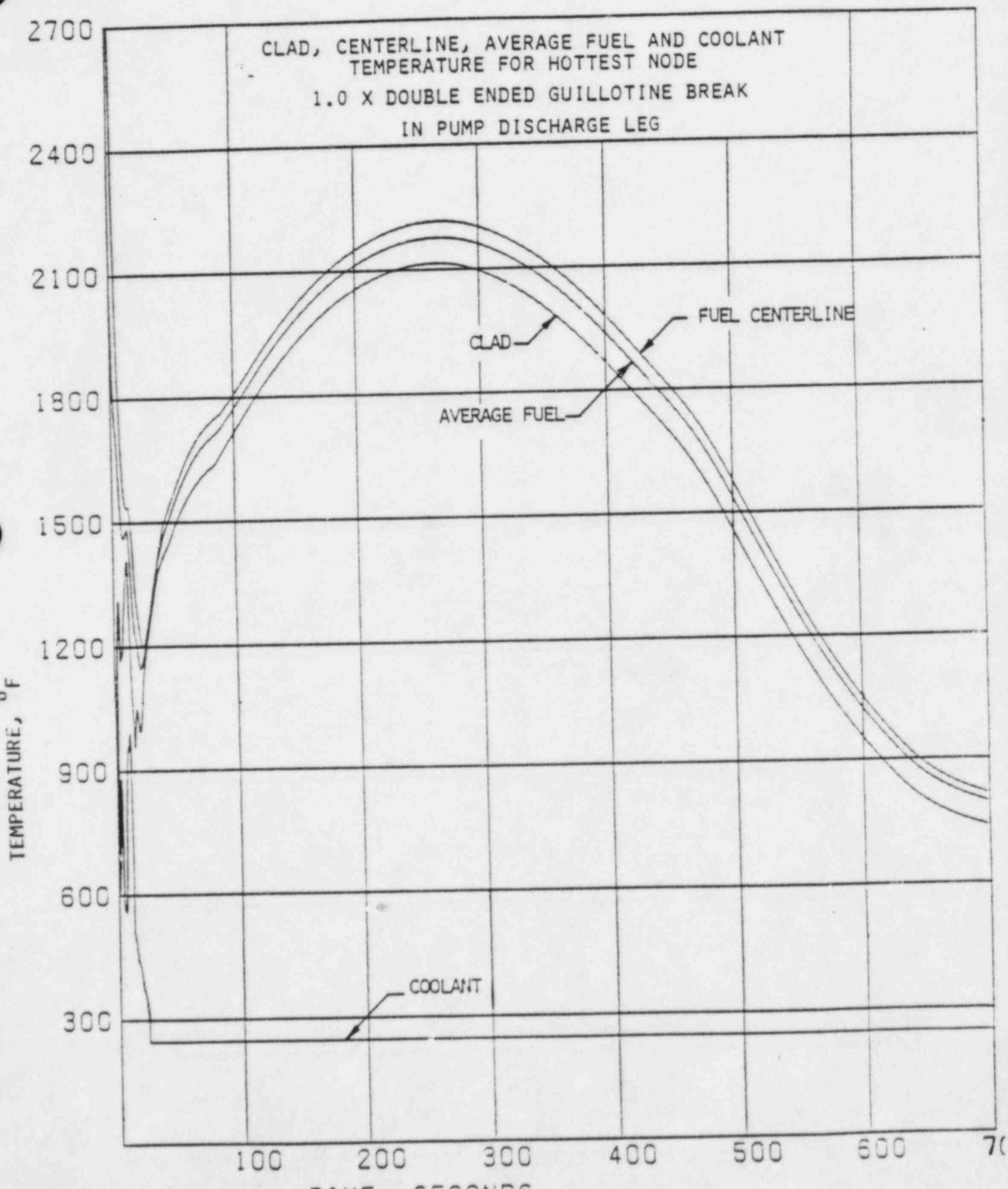
FIGURE 8-10
HOT SPOT GAP CONDUCTANCE
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG
1800
1600
1400
1200
1000
800
600
400
200
100
200
300
400
500
600
700
TIME, SECONDS

FIGURE 8-11



LOCAL CLAD OXIDATION
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE NODE
PEAK OXIDATION NODE
CLAD OXIDATION, %
TIME, SECONDS
18
16
14
12
10
8
6
4
2
100
200
300
400
500
600
700

FIGURE 8-12



CLAD, CENTERLINE, AVERAGE FUEL AND COOLANT
TEMPERATURE FOR HOTTEST NODE
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG
2700
2400
2100
1800
1500
1200
900
600
300
TEMPERATURE, °F
FUEL CENTERLINE
CLAD
AVERAGE FUEL
COOLANT
100
200
300
400
500
600
70
TIME, SECONDS

FIGURE 8-13

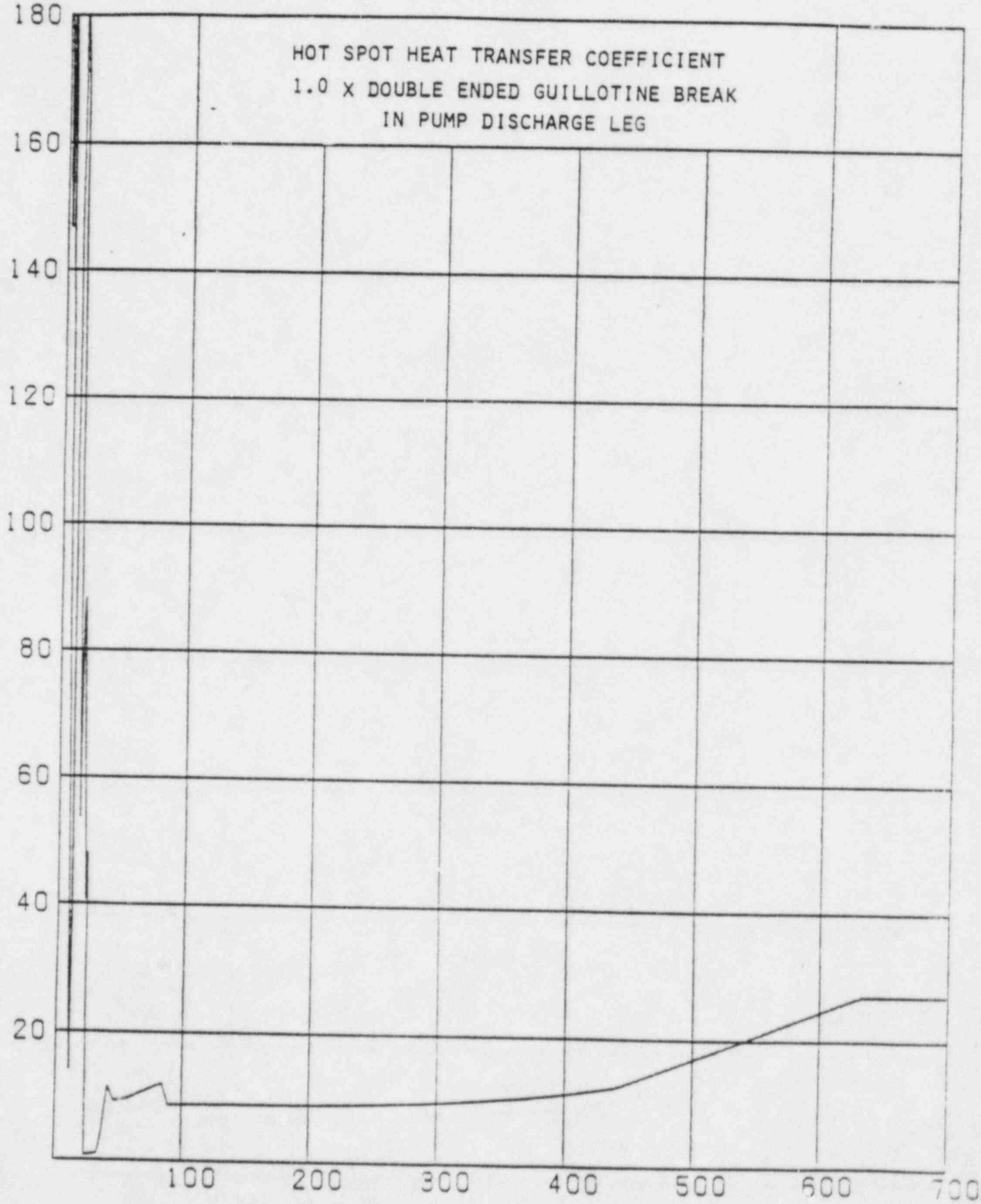
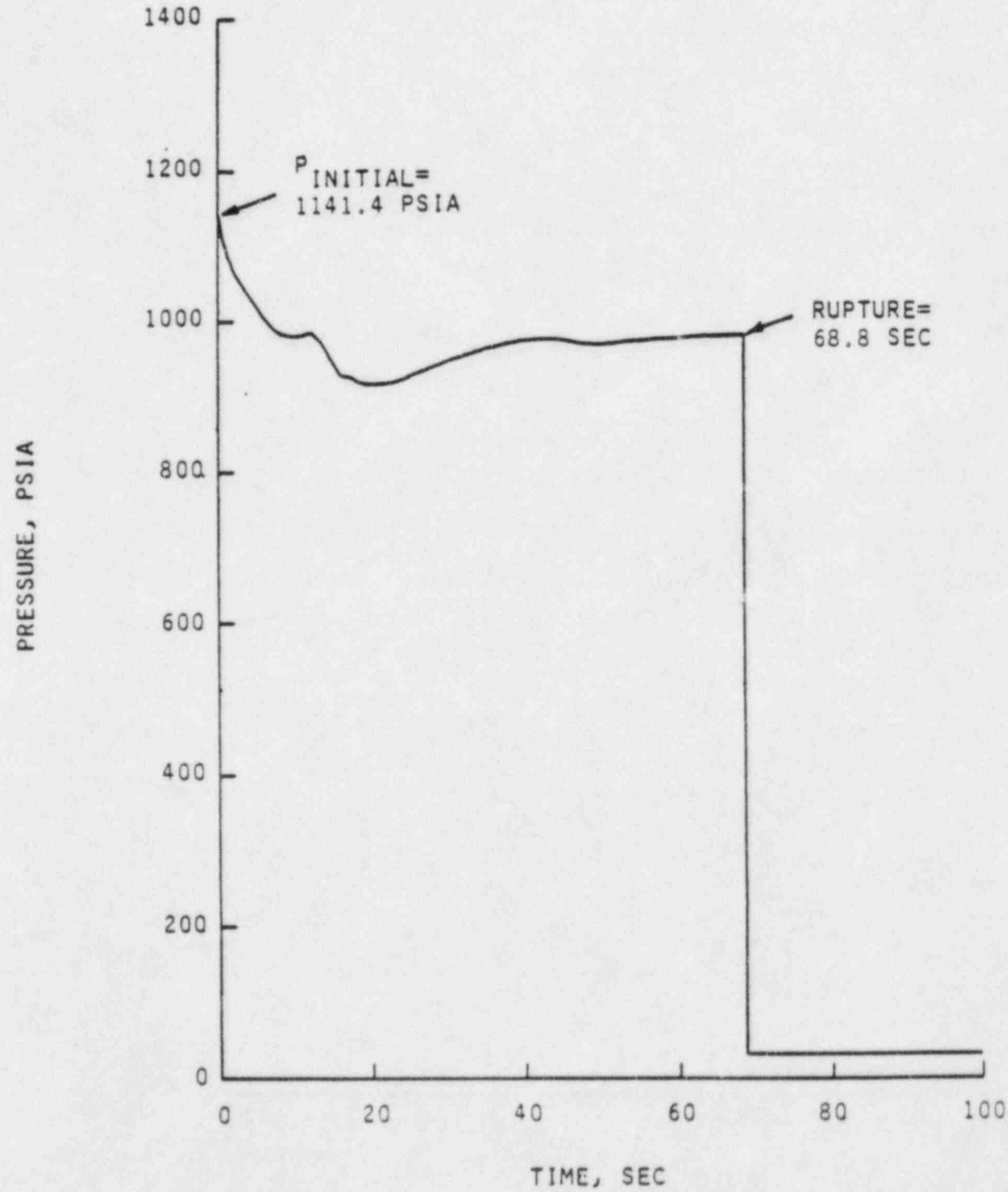

HOT SPOT HEAT TRANSFER COEFFICIENT
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG
HEAT TRANSFER COEFFIC[...]T, BTU/HR-FT²-°F
180
160
140
120
100
80
60
40
20
100
200
300
400
500
600
700
TIME, SECONDS

FIGURE 8-14

HOT ROD INTERNAL GAS PRESSURE
1.0 X DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG



9.0 Reactor Protection and Monitoring System

9.1 Introduction

The Core Protection Calculator System (CPCS) is designed to provide the low DNBR and high Local Power Density (LPD) trips to (1) ensure that the specified acceptable fuel design limits on departure from nucleate boiling and centerline fuel melting are not exceeded during Anticipated Operational Occurrences (AOOs) and (2) assist the Engineered Safety Features System in limiting the consequences of certain postulated accidents.

The CPCS in conjunction with the remaining Reactor Protection System (RPS) must be capable of providing protection for certain specified design basis events, provided that at the initiation of these occurrences the Nuclear Steam Supply System, its sub-systems, components and parameters are maintained within operating limits and Limiting Conditions for Operation (LCOs).

9.2 CPCS Software Modifications

The CPC/CEAC software for SONGS Unit 2 and 3 is being modified for operation in Cycle 3. This modification is being made by taking the SONGS Cycle 2 CPC/CEAC software (Reference 9-4) as a basis since it is the latest NRC approved software. The modifications for SONGS Units 2 and 3 Cycle 3 relative to the Reference Cycle software include algorithm changes derived from the implementation of the CPC Improvement Program (CIP). These modifications have been presented in detail in References 9-3 and 9-5 and are summarized in Table 9-1.

In addition to the algorithm changes, the CPCS data base and uncertainties will be updated from the Reference Cycle. All changes being made to the CPCS will be done in accordance with the NRC-approved software change procedure (References 9-1 and 9-2).

9.3 Addressable Constants

Certain CPC constants are addressable so that they can be changed as required during operation. Addressable constants include (1) constants that are measured during startup (e.g., shape annealing matrix, boundary point power correlation coefficients, and adjustments for CEA shadowing and planar radial peaking factors), (2) uncertainty factors to account for processing and measurement uncertainties in DNBR and LPD calculations (BERR0 through BERR4), and (3) miscellaneous items (e.g., pre-trip and trip setpoints, CEAC inoperable flag, calibration constants, etc.).

9.3.1 Changes to Addressable Constants

As a result of the CPCS software modifications discussed in Section 9.2 above, changes have been made to the list of addressable constants. These changes are listed in Table 9-2 and summarized as follows:

a. Addressable constants for maximum value of Variable Over Power Trip (VOPT) setpoint and offset between VOPT setpoint and FOLLOW will be needed.

b. As a result of the simplification of the flow calculations, addressable constant FC2, core coolant mass flow rate calibration constant, will be deleted. The pump speed trip setpoint will be made addressable.

c. An ASGT trip setpoint will be added as an addressable constant.

d. As a result of the CEAC densensitization changes in UPDATE, a CEAC penalty factor time delay will be added as an addressable constant.

e. Combination of the penalty factor multipliers for DNBR and LPD into a single multiplier will result in the deletion of addressable constant PFMLTL.

f. As a result of power synthesis algorithm changes in the POWER program, addressable constants ARM6, ARM7, EOL, ASM6 and ASM7 will be deleted.

g. The DNBR trip setpoint will be made an addressable constant.

9.4 Digital Monitoring System (COLSS)

The Core Operating Limit Supervisory System (COLSS) is a monitoring system that initiates alarms if the LCO on DNBR, peak linear heat rate, core power, or core azimuthal tilt are exceeded. The COLSS data base and uncertainties will be updated to reflect the Cycle 3 core design.

Table 9-1

CPC System Software Algorithm Changes for Cycle 3

A. FLOW Program

1. Simplification of flow calculations.*

2. Removal of the DNBR flow projection modules.

B. UPDATE Program

1. Addition of variable overpower trip.*

2. Removal of redundant thermal power compensation filters.

3. Enhancement of ASGT delta-T compensation filter.*

4. Changes for CEAC desensitization.*

5. Removal of pressure projection.

6. Combination of PFMLTD and PFMLTL into a single penalty factor multiplier.*

*Require additions to or modification of Addressable Constants.

Table 9-1 (c.ntinued)

CPC System Software Algorithm Changes for Cycle 3

C. POWER Program

1. Base low power ASI calculation on actual axial shape.

2. Revise power synthesis calcuations.*

3. Removal of flow projection calcualtions and DNBR operating limit.

4. Incorporation of an ASI dependent power peaking adjustment.

5. Changes for CEAC desensitization - CEA Withdrawal Prohibit (CWP) flag for misoperation.

D. TRIPSEQ Program

1. Removal of comparison to flow projected DNBR and pressure projected DNBR.

2. Redefinition of J_{trip}.

3. Changes for CEAC desensitization.

4. Addition of DNBR trip setpoint to addressable constants.*

E. CEAC Program

1. Changes for CEAC desensitization - Set flag to initiate CWP.

*Require additions to or modification of Addressable Constants.

Table 9-2

CPC System Addressable Constant Changes for Cycle 3

Point ID	Previous A/C	New A/C
061	FC2	RCP Speed Trip Setpoint
073	EOL	DNBR Trip Setpoint
079	ARM6	Maximum VOPT Setpoint
080	ARM7	VOPT Setpoint Offset
091	PFMLTL	CEAC PF Time Delay
096	ASM6	ASGT ΔT Trip Setpoint
097	ASM7	---

10.0 Technical Specifications

This section provides a summary of recommended changes that should be made to the SONGS-2 Technical Specifications in order to update the Technical Specifications for Cycle 3 operation. A description of each change and the corresponding technical specification change pages are presented in Reference 10-1.

SONGS UNIT 2 CYCLE 3
PROPOSED TECH SPEC CHANGES

CHANGE PACKAGE NUMBER: 1

SECTION	NATURE OF CHANGE FOR CYCLE 3
3.1.2.1	Boric acid concentration range and associated heat tracing requirements to change. Figure 3.1-1 to change.
3.1.2.2	Boric acid concentration range and associated heat tracing requirements to change. Figure 3.1-1 to change.
3.1.2.7	RWST minimum water volume of 9970 gals above the ECCS suction connection may change.
3.1.2.8	RWST requirements: 2300 ppm may increase.
3.5.1	SIT requirements: 2300 ppm may increase.
3.5.4	RWST requirements: 2300 ppm may increase.

REASONS FOR CHANGE: Boric acid concentration reduced to allow elimination of heat tracing. RWST and SIT concentration ranges increased for added flexibility.

SONGS UNIT 2 CYCLE 3
PROPOSED TECH SPEC CHANGES

CHANGE PACKAGE NUMBER : 2

SECTION	NATURE OF CHANGE FOR CYCLE 3
5.3.1	Enrichment limit of 3.7 w/o must be raised

REASON FOR CHANGE : Longer fuel cycles require higher fuel enrichment. Cycle 3 will contain 4.05 w/o fuel.

SONGS UNIT 2 CYCLE 3
PROPOSED TECH SPEC CHANGES

CHANGE PACKAGE NUMBER : 4

SECTION	NATURE OF CHANGE FOR CYCLE 3
3.1.1.3	Neg. MTC limit will get more neg.

REASON FOR CHANGE : Negative limit will be close for cycle 3 and is expected to be more negative than current Tech Spec value in later cycles. The MTC range is used as an analysis ground rule.

SONGS UNIT 2 CYCLE 3
PROPOSED TECH SPEC CHANGES

CHANGE PACKAGE NUMBER : 2

SECTION	NATURE OF CHANGE FOR CYCLE 3
5.3.1	Enrichment limit of 3.7 w/o must be raised

REASON FOR CHANGE : Longe fuel cycles require higher fuel enrichment.
Cycle 3 will contain 4.05 w/o fuel.

SONGS UNIT 2 CYCLE 3
PROPOSED TECH SPEC CHANGES

CHANGE PACKAGE NUMBER : 4

SECTION	NATURE OF CHANGE FOR CYCLE 3
3.1.1.3	Neg. MTC limit will get more neg.

REASON FOR CHANGE : Negative limit will be close for cycle 3 and is expected to be more negative than current Tech Spec value in later cycles. The MTC range is used as an analysis ground rule.

SONGS UNIT 2 CYCLE 3
PROPOSED TECH SPEC CHANGES

CHANGE PACKAGE NUMBER : 5

SECTION	NATURE OF CHANGE FOR CYCLE 3
3.3.1	Revise Note in Table 3.3-2 to change RTD response time to 8 sec. Delete Table 3.3-2a&b.

REASON FOR CHANGE : Safety and CPC analyses will be done using RTD response times of 8 sec. Penalty factors for greater response times will not be verified. Thus Tables 3.3-2a&b will not be supported and must be deleted.

SONGS UNIT 2 CYCLE 3
PROPOSED TECH SPEC CHANGES

CHANGE PACKAGE NUMBER : 6

SECTION	NATURE OF CHANGE FOR CYCLE 3
Table 2.2-1	LPD Trip limit w/o dynamic terms: 21 kw/ft

REASON FOR CHANGE : Install generic LPD trip limit.

SONGS UNIT 2 CYCLE 3
PROPOSED TECH SPEC CHANGES

CHANGE PACKAGE NUMBER : 7

SECTION	NATURE OF CHANGE FOR CYCLE 3
2.2.2	Delete the T/S Section.
Table 2.2-2	Delete the Table containing the Addressable Constants.

REASON FOR CHANGE : CIP will change, add and/or delete addressable constants

11.0 Startup Testing

The planned startup test program associated with core performance is outlined below. These tests verify that core performance is consistent with the engineering design and safety analysis. Some of the tests also provide the data needed for adjustment of addressable constants in the Core Protection Calculators (CPC's) and in determining constants for the Core Operating Limit Supervisory System (COLSS).

11.1 Precritical Test

11.1.1 Control Element Assembly (CEA) Trip Test

Precritical CEA drop times are recorded for all 91 CEA's at hot, full flow conditions before criticality following refueling. Acceptance criteria state that the CEA drop time from fully withdrawn to 90% inserted shall be less than 3.0 seconds at the stated conditions.

11.2 Low Power Physics Tests

11.2.1 Criticality

Criticality is obtained by withdrawing the Shutdown CEA Groups, diluting to the estimated critical boron concentration, then withdrawing the Regulating CEA Groups to the estimated critical position corresponding to the boron concentration already established.

11.2.2 Critical Boron Concentration

Once criticality is achieved, the equilibrium, all CEA's withdrawn boron concentration is obtained. Comparison to the reference critical boron concentration is performed by adding the boron equivalent of the residual CEA worth (from the actual CEA position to the reference CEA position) to the actual boron concentration. Acceptance criteria states that the critical boron concentration shall be within the equivalent of $\pm$ 1% $\Delta K/K$ of the design prediction.

11.2.3 Temperature Reactivity Coefficient

The isothermal temperature coefficient is measured at the Essentially All Rods Out configuration and at a partially rodded configuration. The average coolant temperature is varied and the reactivity feedback associated with the temperature change is measured. Acceptance criteria state that the measured value shall not differ from the predicted value by more than $\pm$ $0.3 \times 10^{-4} \Delta K/K/^{o}F$.

The moderator temperature coefficient (MTC) of reactivity is calculated by subtracting a predicted value of the fuel temperature coefficient of reactivity. The moderator temperature coefficient (MTC) value is then verified to be within the following Technical Specification criteria:

-3.3×10^{-4} $\Delta K/K/^{o}F$ < MTC < 0.0×10^{-4} $\Delta K/K/^{o}F$; Power > 70% Rated Thermal Power

-3.3×10^{-4} $\Delta K/K/^{o}F$ < MTC < 0.5×10^{-4} $\Delta K/K/^{o}F$; Power $\leq$ 70% Rated Thermal Power

11.2.4 CEA Reactivity Worth

CEA worths will be measured using the CEA Exchange technique. This technique consists of measuring the worth of a "Reference Group" via standard boration/dilution techniques, then exchanging this group with other groups to measure their worths. Due to the large differences in relative CEA group worths, two reference groups (one with very high worth and one with medium worth) will be used. The groups to be measured by exchange will be "assigned" to a specific reference group, depending on their predicted worth. This measurement technique provides verification that individual group CEA reactivities are within the engineering design safety analysis prediction for all CEA groups. Acceptance criteria state that the measured individual group worths shall be within $\pm$15% or $\pm$0.1% $\Delta K/K$ (whichever is larger of predicted values, and the total worth of all the groups shall be within $\pm$10% of the predicted values.

11.3 Power Ascension Tests

Following completion of the Low Power Physics Test sequence, reactor power will be increased in accordance with normal operating procedures. The power ascension will be monitored by an off-line NSSS performance and data processing computer algorithm. This computer code will be continously executed in parallel with the power ascension to monitor CPC and COLSS performance relative to the processed plant data against which they are normally calibrated. If necessary, the power ascension will be suspended while necessary data reduction and equipment calibrations are performed. Thus the monitoring algorithm continuously ensures conservative CPC and COLSS operation while optimizing overall efficiency of the test program.

11.3.1 Reactor Coolant Flow

Reactor coolant flow will be measured by calorimetric methods at steady state conditions in accordance with Technical Specifications. Acceptance criteria will require that the measured flow be within allowable limits and that both COLSS and the CPC's reactor coolant flow rates are within calibration requirements relative to the measured calorimetric flow rate.

11.3.2 Core Power Distribution

Core power distribution data using fixed incore neutron detectors is used to verify proper core fuel loading and consistency between the as-built and engineering design models. This is accomplished using measurement data from three power plateaus.

The first power distribution measurement is performed after the turbine is synchronized. The objective of this measurement is primarily to identify any fuel misloading which results in power assymetries or deviations from the reactor physics design. Because of the decreased signal to noise ratio at low powers and the absence of

xenon stability requirements, radial and azimuthal symmetry criteria are emphasized whereas pointwise absolute and statistical acceptance criteria are relaxed.

At the intermediate power plateau (between 40 and 70% reactor power) a core power distribution analysis is performed to again verify proper fuel loading and consistency with design predictions. The intermediate power acceptance criteria ensure that the power distribution is consistent with predictions and that reactor power may be increased to 100% and remain within the design limits.

The final power distributions comparison is performed with equilibrium xenon at approximately 100% power. At this plateau axial and radial power distributions are compared to design predictions as a final verification that the core is operating in a manner consistent with its design within the associated design incertainties.

The measured results are compared to predicted values in the following manner for the intermediate and full power distribution analysis:

A. The measured radial power distribution is compared to the predicted power distribution utilizing a root mean squared statistical error comparison of the relative radial power density distribution for each of the 217 fuel assemblies. The acceptance criteria states that the comparison of the measured radial power distribution shall satisfy the following expression:

$$RMS = \left[\frac{\sum_{i=1}^{217} Z_i^2}{217} \right]^{1/2} \leq 0.05$$

where Z_i is the difference between the predicted and measured relative power density distribution for the i^{th} fuel assembly.

B. The measured radial power distribution is additionally compared to the predicted power distribution utilizing a box-by-box comparison of the relative radial power density distribution for each of the 217 fuel assemblies. The acceptance criteria states

that for each assembly with a predicted relative power density >=0.9, the measured and predicted relative power density values must agree within $\pm$ 10%, and for each assembly with a predicted relative power density <0.9, the measured and predicted relative power density values must agree within $\pm$ 15%.

C. The measured axial power distribution is compared to the predicted power distribution utilizing a root mean squared statistical error comparison of the relative axial power distribution for each of the 51 axial nodes. The acceptance criteria states that the comparison of the measured axial power distribution with the predicted axial power distribution shall satisfy the following expression:

$$RMS = \left[\frac{\sum_{i=1}^{51} h_i^2}{51} \right]^{1/2} \leq 0.05$$

where h_i is the difference between the predicted and measured relative power density distribution for the i^{th} axial % of core height.

D. The measured values of total planar radial peaking factor (F_{xy}), total integrated radial factor (F_r), core average axial peak (F_z), and 3-D power peak (F_q) are compared to predicted values. The acceptance criteria states that the measured values of F_{xy}, F_r, F_z, and F_q shall be within $\pm$ 10% of the predicted values.

11.3.3 Shape Annealing Matrix (SAM) and Boundary Point Power Correlation Coefficients (BPPCC) Verification

The SAM matrix and BPPC coefficients are determined from a linear regression analysis of the measured excore detector readings and corresponding core power distribution determined from the incore detector signals. Since these values must be representative for a

rodded and unrodded core throughout life, it is desirable to use as wide a range of core axial power shapes as are available to establish their values. The spectrum of axial shapes encountered during the power ascension has been demonstrated to be adequate for the calculation of the matrix elements. Incore, excore, and related data are recorded and incore analysis is performed which relates the incore detector signals to power distribution and summarizes the necessary power distribution and excore detector data in a form and format which can be easily input to programs used to perform the least squares fitting. The data is processed and compiled throughout the power ascension by the off-line NSSS performance and data processing algorithm

The analysis results include:

A. Core peripheral power fractions for the upper, middle, and lower third of the core for each quadrant;

B. Core average power fractions for the upper, middle, and lower third of the core; and

C. Upper and lower core boundary average power.

Appropriate CPC constants are modified, if needed, based upon the measured values.

11.3.4 Radial Peaking Factor and CEA Shadowing Factor Verification

The performance of this test involves establishing the following CEA configurations:

All CEA's Out

Group 6 at LEL (Lower Electrical Limit)

Group 6 at LEL, Group P at 37.5 inches withdrawn

Group P at 37.5 inches withdrawn

As the various CEA configurations are established, incore detector data and excore detector data are taken after allowance of sufficient time for stabilization of the incore instrument signals. This data is analyzed and planar radial peaking factors (F_{xy}) and CEA shadowing factors are determined for each CEA configuration. Appropriate CPC and/or COLSS constants are modified, if needed, based on the measured values.

11.3.5 Reactivity Coefficients 100% Full Power

(1) Isothermal Temperature Coefficient - With the reactor at steady state and near equilibrium Xenon, CEA's are moved a specified amount. This reactivity change produces a change in reactor power which in turn causes a change in coolant temperature. The change in coolant temperature results in a reactivity feedback to counter the rod movement if the ITC is negative. The system eventually stabilizes at a new coolant temperature. Core power is kept essentially constant by adjustments made to turbine loading. ITC is calculated knowing the power and temperature changes along with the CEA integral worth and by using the prediction for the Power Coefficient. The MTC is calculated as described previously.

(2) Doppler Power Coefficient - Reactivity changes are made using CEA's, resulting in a change in reactor power. Average coolant temperature is held constant by changing turbine load. The reactor stabilizes at a new power when the reactivity feedback due to change in power is equal and opposite to the CEA reactivity insertion. The Doppler power coefficient is calculated in a manner similar to the ITC calculation.

Acceptance Criteria state the following:

a. The measured ITC shall agree with the predicted values within $\pm 0.3 \times 10^{-4} \Delta K/K/^{o}F$;

b. The measured power coefficient should agree with the predicted values within $\pm 0.3 \times 10^{-4} \Delta K/K/\%$ power; and

c. The MTC shall satisfy the following criteria:

$-3.3 \times 10^{-4} \Delta K/K/^{o}F < MTC < 0.0 \times 10^{-4} \Delta K/K/^{o}F$;
Power > 70% Rated Thermal Power

$-3.3 \times 10^{-4} \Delta K/K/^{o}F < MTC < 0.5 \times 10^{-4} \Delta K/K/^{o}F$;
Power ≤ 70% Rated Thermal Power

11.4 Procedure If Acceptance Criteria Are Not Met

If the acceptance criteria for any test are not met, an evaluation is performed before the test program is continued. The results of all tests will be reviewed by the plant's core analysis engineering group. If the acceptance criteria of the startup physics tests are not met, an evaluation will be performed by the plant's core analysis engineering group with assistance from the fuel vendor, as needed. The results of this evaluation will be presented to the Onsite Review Committee. Resolution will be required prior to power escalation. If an unreviewed safety question is involved, the NRC would be notified.

12.0 References

12.1 Section 1.0 References

(1-1) "San Onofre Nuclear Generating Station Units 2 and 3 Final Safety Analysis Report," Southern California Edison Co., Docket Nos. 50-361 and 50-362.

(1-2) Letter from M. O Medford (SCE) to G. W. Knighton (NRC), "Reload Analysis Report, San Onofre Nuclear Generating Station Units 2 and 3," September 28, 1984.

(1-3) Presentation to NRC on May 31, 1985

12.2 Section 2.0 References

None

12.3 Section 3.0 References

None

12.4 Section 4.0 References

(4-1) Letter from M. O. Medford (SCE) to G. W. Knighton (NRC), "Reload Analysis Report, San Onofre Nuclear Generating Station Units 2 and 3," September 28, 1984.

(4-2) LD-84-054, "CEA Guide Tube Wear Sleeve Modification," August 3, 1984.

(4-3) NRC Letter, C. O. Thomas to A. E. Scherer, "Acceptance for Referencing of Licensing Special Report LD-84-043, CEA Guide Tube Wear Sleeve Modification," September 7, 184.

(4-4) CENPD-187, "CEPAN Method of Analyzing Oval Cladding," June 1975

(4-5) CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report," July, 1974.

(4-6) "San Onofre Nuclear Generating Station Units 2 and 3 Final Safety Analysis Report," Southern California Edison Co., Docket Nos. 50-361 and 50-362.

(4-7) CEN-182(B)-P, "Statistical Approach to Analyzing Creep Collapse of Oval Fuel Rod Cladding Using CEPAN," September 1981.

(4-8) CEN-161(B)-P, "Improvement to Fuel Evaluation Model," July 1981.

(4-9) Letter, R. A. Clark (NRC) to A. E. Lundvall, Jr. BG&E), "Safety Evaluation of CEN-161 (FATES 3)," March 31, 1983.

(4-10) CENPD-269-P Revision 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984.

(4-11) E. J. Butcher (NRC) to A. E. Lunduall (BG&E), "Safety Evaluation of Topical Report CENPD-269-P," To be issued.

(4-12) CEN-96-(A)-P Rev. 1. "ANO-2 Reactor Operation with Modified CEA Guide Tubes and Lengthened Upper Guide Structure Flow Channels," July 12, 1978.

12.5 Section 5.0 References

(5-1) Presentation to NRC on May 31, 1985.

(5-2) "CEPAN Method of Analyzing Creep Collapse of Oval Cladding, Volume 5: Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," EPRI NP-3966-CCM, Volume 5, Project 2061-6, Computer Code Manual, April 1985.

(5-3) A. E. Lundvall (BG&E) to J. R. Miller (NRC), "Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317 Eighth Cycle License Application," February 22, 1985.

(5-4) "Safety Evaluation of the Office of Nuclear Reactor Regulation Related to Amendment No. 104 to Facility Operating License No. DPR-53 Baltimore Gas and Electric Company Calvert Cliffs Nuclear Power Plant Unit No. 1, Docket No. 50-317."

(5-5) CENPD-153-P, Rev. 1-P-A, "INCA/CECOR Power Peaking Uncertainty," May 1980.

(5-6) CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," April 1983.

12.6 Section 6.0 References

(6-1) CENPD-161-P, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.

(6-2) CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," April 1975.

(6-3) CENDP-206-P, "TORC Code, Verification and Simplified Modeling Methods," January 1977.

(6-4) CEN-160(S)-P, Rev. 1-P, "CETOP Code Structure and Modeling Methods for San Onofre Nuclear Generating Units 2 and 3," September 1981.

(6-5) CEN-283(S)-P, "Statistical Combination of Uncertainties, Part I: Combination of System Parameter Uncertainties in Thermal Margin Analyses for San Onofre Nuclear Generating Stations Units 2 and 3," June 1984.

(6-6) CEN-155-(S)-P, "CE-1 Applicability to San Onofre Units 2 and 3 HID-2 Grids, Response to NRC Questions," March 1981.

(6-7) CEN-165(S)-P, "Response to NRC Concerns on Applicability of the CE-1 Correlation to the SONGS Fuel Design," March 1981.

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(6-9) CENPD-255-P-A, "Fuel and Poison Rod Bowing," June 1983.

(6-10) Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 24 to Facility Operating License No. NPF-6, Arkansas Power and Light Company, Arkansas Nuclear One, Unit No. 2, Docket No. 50-368.

12.7 Section 7.0 References

(7-1) Letter from M. O. Medford (SCE) to G. W. Knighton (NRC), "Reload Analysis Report San Onofre Nuclear Generating Station, Units 2 and 3." September 28, 1984.

(7-2) "San Onofre Nuclear Generation Station Units 2 and 3, Final Safety Analysis Report".

(7-3) "STRIKIN II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," CENPD-135-P, August 1984.

(7-4) "Loss of Flow - C-E Methods for Loss of Flow Analysis," CENPD-183, July, 1975.

(7-5) Macbeth, R. V., "An Appraisal of Forced Convection Burnout Data," Proc. Instn. Mech. Engrs., Vol. 180, Pt3C, PP 37-50, 1965-66.

(7-6) Lee, D. H., "An Experimental Investigation of Forced Convection Burnout in High Pressure Water - Part IV, Large Diameter Tubes at about 1600 psia," A.E.E.W. Report R 479, 1966.

(7-7) "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs 1 and 2," CEN-191(B)-P, December 1981.

(7-8) CPC/CEAC Software Modifications for Arkansas Unit 2, CEN-143-A-P, Appendix A, December 1980.

(7-9) Not used.

(7-10) "CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981, Enclosure 1-P to LD-82-001, January 6, 1982.

(7-11) CENPD-188-A, "HERMITE Space-Time Kinetics," July 1976.

(7-12) CENPD-161-P, "TORC Code, A Computer Code for Determining the Thermal Margin for a Reactor Core," July 1975.

(7-13) Calvert Cliffs Nuclear Power Plant Unit 1 Docket No. 50-317, "Amendment to Operating License DPR 53 Supplement 1 to Seventh Cycle License Application," September 1, 1983.

(7-14) CENPD-161-P, "TORC Code - A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.

(7-15) CENPD-206-P, "TORC Code - Verification and Simplified Modeling Methods," January 1977.

(7-16) Presentation to NRC on May 31, 1985.

(7-17) CEN-308-P Revision 00-P, "CPC/CEAC Software Modifications for the CPC Improvement Program," July 1985.

12.8 Section 8.0 References

(8-1) Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3, Friday, January 4, 1974.

(8-2) Letter from M. O. Medford (SCE) to G. W. Knighton (NRC), "Reload Analysis Report San Onofre Nuclear Generating Station, Units 2 and 3." September 28, 1984.

(8-3) CENPD-132, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

CENPD-132, Supplement 1, "Updated Calculative Methods for the C-E Large Break LOCA Evaluation Model," December 1974.

(8-4) CENPD-132, Supplement 2, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975.

(8-5) CENPD-133, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis," April 1974.

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(8-8) CENPD-138, and Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup,"February 1975.

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(8-10) CENPD-161(B)-P, "Improvements to Fuel Evaluation Model Topical Report," July 1981.

(8-11) Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BG&E), dated March 31, 1983.

12.9 Section 9.0 References

(9-1) CEN-39(A)-P, Revision 02, "The CPC Protection Algorithm Software Change Procedure," December 21, 1978.

(9-2) CEN-39(A)-P, Supplement 1-P, Revision 01, January 5, 1979.

(9-3) CEN-308-P Revision 00-P, "CPC/CEAC Software Modifications for the CPC Improvement Program," July 1985.

(9-4) Safety Evaluation Related to Amendment No. 32 to NPF-10 and Amendment No. 21 to NPF-15 for San Onofre Nuclear Generating Station, Units 2 and 3, Docket Nos. 50-361 and 50-362, Southern California Edison Company, March 1985.

(9-5) CEN-310-P, "CPC Methodology Changes for the CPC Improvement Program," September 1985.

12.10 Section 10.0 References

(10-1) To be issued.

12.11 Section 11.0 References

None