

February 22, 1979
L-79-45

Director of Nuclear Reactor Regulation
Attention: Mr. Victor Stello, Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

Re: St. Lucie Unit 1
Docket No. 50-335
Proposed Amendment to
Facility Operating License DPR-67

In accordance with 10 CFR 50.30, Florida Power & Light Company submits herewith three (3) originals and forty (40) copies of a request to amend Appendix A of Facility Operating License DPR-67.

The purpose of this submittal is to support Cycle 3 operation of St. Lucie Unit 1. Technical Specification changes associated with the reload are described in Section 9 of the attached Reload Safety Evaluation (Attachment A) and shown on the attached Technical Specification pages bearing the date of this letter in the lower right hand corner (Attachment B).

The proposed amendment has been reviewed by the St. Lucie Facility Review Group and the Florida Power & Light Company Nuclear Review Board. They have concluded that it does not involve an unreviewed safety question. The RSE evaluates Cycle 3 design and performance and supports the proposed Technical Specification changes.

Florida Power & Light has determined that this is a Class III amendment in accordance with 10 CFR 170.22. A check in the amount of \$4,000.00 is enclosed.

Very truly yours,

Robert E. Uhrig
Vice President
Advanced Systems and Technology

REU/TCG:cf
Attachments (2)

cc: Mr. James P. O'Reilly, Region II
Harold F. Reis, Esquire

50-335
7903010298
Date 2/22/79 of Documents
REGULATORY DOCKET FILE

7903010298

Acc 1
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w/ check
\$4,000.00
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ATTACHMENT A

Re: St. Lucie Unit 1
Docket No. 50-335
Cycle 3 Operation

RELOAD SAFETY EVALUATION

Docket # 50-335
Control # 7903010298
Date 2/22/79 of Document:
REGULATORY DOCKET FILE

1. INTRODUCTION AND SUMMARY

This report provides an evaluation of the design and performance for the operation of St. Lucie I during its third fuel cycle at full rated power of 2560 MWT. Operating conditions remain generally the same as those for Cycle 2. The core will consist of presently operating Batch B, C and D assemblies together with fresh Batch E assemblies.

System requirements have created a need for flexibility in the Cycle 2 burnup length ranging from 8000 to 8600 MWD/T. The Cycle 3 loading pattern described in this report has been designed to accommodate this range of shutdown points. In performing analyses of postulated accidents, determining limiting safety system settings and establishing limiting conditions for operations, values of key parameters were chosen to assure that expected conditions are enveloped within the above Cycle 2 burnup range.

During scheduled shutdown maintenance after Cycle 1 operation, it was observed that CEA fingers caused wear in the CEA guide tubes as reported in Reference 1. The mechanical integrity of the worn areas was restored by installing stainless steel sleeves in the guide tubes. Sleeving was performed on those assemblies which were installed in CEA locations in Cycle 2 with the exception of the center assembly. For Cycle 3 operations, those assemblies will also be sleeved which were not placed under CEAs in cycles 1 and 2 but will be located under CEAs in Cycle 3. All 68 Batch E assemblies will be sleeved.

The evaluations of the reload core characteristics have been examined with respect to the Reference 2 safety analyses describing Cycle 2, hereafter referred to as the "reference cycle". In all cases, it has been concluded that either the reference cycle analyses properly envelope the new conditions, or that the revised analyses presented in this report continue to show acceptable results.

Where dictated by variations from the reference cycle, proposed modifications to the plant Technical Specifications are provided and are justified by the analyses reported herein.

2. OPERATING HISTORY OF THE REFERENCE CYCLE

St. Lucie Unit I is presently operating in its second fuel cycle utilizing Batch A, B, C and D fuel assemblies at a licensed core power level of 2560 MWt. Operation of Cycle 2 has continued at or near licensed power. The CEA guide tube problem has been mitigated by the introduction of sleeves into relevant assemblies.

It is presently estimated that Cycle 2 will terminate during April, 1979. To allow for flexibility in the Cycle 2 termination date, a range of burnups between 8000 and 8600 MWD/T has been anticipated. Operation of Cycle 3 is scheduled to commence in May, 1979.

3. GENERAL DESCRIPTION

The Cycle 3 core will consist of the numbers and types of assemblies from the various fuel batches as described in Table 3-1. The primary change to the core for Cycle 3 is the removal of the remaining Batch A assemblies, and removal of 59 Batch B assemblies. These assemblies will be replaced by 40 Batch E (3.03 w/o enrichment) and 28 Batch E* (2.73 w/o enrichment) assemblies. The fuel management pattern developed for Cycle 3 allows for flexibility in Cycle 2 burnup length between 8000 and 8600 MWD/T. The loading pattern is shown in Figure 3-1.

The Cycle 3 core loading pattern is 90 degrees rotationally symmetric. That is, if one quadrant of the core were rotated 90 degrees into its neighboring quadrant, each assembly would overlay a similar assembly. This similarity includes batch type, number of fuel rods, initial enrichment and beginning of cycle burnup.

Figure 3-2 shows the beginning of Cycle 3 assembly burnup distribution for a Cycle 2 burnup length of 8300 MWD/T. The initial enrichment of each assembly is also shown. The residual B-10 content of the burnable poison shims in the Batch B and C assemblies is very low in all assemblies at the beginning of Cycle 3.

Table 3-1

St. Lucie Unit 1

Cycle 3 Core Loading

<u>Assembly Designation</u>	<u>Number or Assemblies</u>	<u>Initial Enrichment w/o U-235</u>	<u>Beginning of Cycle 3 Batch Average Burnup MWD/MTU (EOC2= 8300 MWD/T)</u>	<u>Number of Shims</u>	<u>Initial Shim Loading w/o B₄C**</u>	<u>Total Shims</u>	<u>Total Fuel Rods</u>
B	21	2.33	21,200	12	2.7	252	3,444
C	40	2.82	17,600	0	---	0	7,040
C+	16	2.82	22,100	12	.83	192	2,624
C.	12	2.82	21,600	12	.46	144	1,968
D	40	3.03	6,700	0	---	0	7,040
D*	20	2.73	9,400	0	---	0	3,520
E	40	3.03	0	0	---	0	7,040
E*	28	2.73	0	0	---	0	4,928
	<u>217</u>		<u>10,000</u>			<u>588</u>	<u>37,604</u>

** Both original and replacement shims



						E	E	
				E	E	E*	D*	C
		E	E*	B	D	C+	E*	
	E	D	C	E*	C°	D	B	
E	E*	C	C°	C	D*	B	D	
E	B	E*	C	D	C	D	C	
E*	D	C°	D*	C	C	C+	D*	
E	D*	C+	D	B	D	C+	D	C
E	C	E*	B	D	C	D*	C	B

St. Lucie Nuclear Power Station Unit No. 1	CYCLE - 3 LOADING PATTERN	Figure 3-1
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XXXX
Y.YY

BOC 3 BURNUP (MWD/T)
INITIAL ENRICHMENT, WT% U-235

						0.0 3.03	0.0 3.03	
				0.0 3.03	0.0 3.03	0.0 2.73	8,700 2.73	18,700 2.82
		0.0 3.03	0.0 2.73	21,400 2.33	8,100 3.03	22,800 2.82	0.0 2.73	
	0.0 3.03	6,200 3.03	17,700 2.82	0.0 2.73	21,200 2.82	6,800 3.03	21,400 2.33	
0.0 3.03	0.0 2.73	17,700 2.82	22,300 2.82	18,700 2.82	9,700 2.73	21,000 2.33	6,800 3.03	
0.0 3.03	21,400 2.33	0.0 2.73	18,700 2.82	6,800 3.03	17,200 2.82	5,400 3.03	16,800 2.82	
0.0 2.73	8,100 3.03	21,200 2.82	9,700 2.73	17,200 2.82	17,700 2.82	21,400 2.82	10,300 2.73	
3.03	8,700	22,800	6,800	21,000	5,400	21,400	6,200	15,600
0.0	2.73	2.82	3.03	2.33	3.03	2.82	3.03	2.82
3.03	18,700	0.0	21,400	6,800	16,800	10,300	15,600	20,700
	2.82	2.73	2.33	3.03	2.82	2.73	2.82	2.33

<p>St. Lucie Nuclear Power Station Unit No. 1</p>	<p>CYCLE 3 - ASSEMBLY AVERAGE BURNUP AND INITIAL ENRICHMENT DISTRIBUTION</p>	<p>Figure 3-2</p>
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4.0 FUEL DESIGN

4.1 Mechanical Design

The fuel assembly complement for Cycle 3 is given in Table 3-1. The mechanical design of the reload fuel assemblies, Batch E, is essentially identical to that introduced with the St. Lucie-1 Batch D fuel.

C-E has performed analytical predictions of cladding creep-collapse time for all St. Lucie-1 fuel batches that will be irradiated during Cycle 3 and has concluded that the collapse resistance of all fuel rods is sufficient to preclude collapse during their design lifetime. This lifetime will not be exceeded by the Cycle 3 duration.

The analyses utilized the CEPAN computer code (Reference 3) and included as input conservative values of internal pressure, cladding dimensions, cladding temperature and neutron flux.

4.2 Hardware Modifications to Mitigate Guide Tube Wear

The mechanical design of the fuel assemblies is as described in the preceding section except that (1) those fuel assemblies which sustained substantial guide tube wear in Cycle 1 have had stainless steel sleeves installed in the guide tubes as a means of improving the mechanical strength margins in the worn areas, and (2) all Batch B, C, D and E fuel assemblies to be installed in CEA locations in Cycle 3 will also have stainless steel sleeves installed in the guide tubes in order to mitigate guide tube wear.

A detailed discussion of the design of the sleeves and its effects on reactor operation is contained in Reference 4.

4.3 Thermal Design

Using the FATES model (Reference 5), the thermal performance of the various types of fuel assemblies has been evaluated with respect to their Cycle 1 and Cycle 2 burnups, proposed burnups during Cycle 3, their respective fuel geometries and expected flux levels during Cycle 3. The Batch E fuel has been determined to be the limiting fuel batch with respect to stored energy.

4.4 Chemical Design

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for the Batch E fuel have not been changed from the original Cycle 1 and Cycle 2 designs. Therefore, the chemical or metallurgical performance of the Batch E fuel will be unchanged from that of the original core fuel and discussions in the FSAR, Reference 6 are still valid.

4.5 Operating Experience

Fuel assemblies incorporating the same design features as the St. Lucie Unit 1, Batch E fuel assemblies have had operating experiences at Calvert Cliffs 1, Fort Calhoun 1, Millstone II, and Maine Yankee. The operating experience has been successful except for the CEA guide tube wear problem which has been addressed in Section 4.2.

4.6 Inconel Irradiation Experiment

In order to further the basic material property data base on irradiated Inconel 625 CEA cladding, three empty CEA tubes will be placed in the center guide posts of selected high flux regions of the St. Lucie core at the start of Cycle 3. The overall design of the test assemblies will be similar to the standard C-E design for the neutron source assembly described in Section 4.2.2.2.9 of Reference 6. Basically, an upper end fitting and a lower end cap will be welded to an empty 14 x 14 or 16 x 16 CEA cladding tube and appropriate flow holes introduced to eliminate stresses from differential system pressure on the test cladding. The upper end fitting has 'ears' which fit into the recesses of the guide post and the upper guide structure compresses a spring loaded support to keep the specimens in place.

The intent of the program is to perform mechanical tests on the fully irradiated cladding (~ 1022 nvt) material at an appropriate hot cell.

5. NUCLEAR DESIGN

5.1 PHYSICS CHARACTERISTICS

5.1.1 Fuel Management

The Cycle 3 fuel management employs a mixed central region as described in Section 3, Figure 3-1. The fresh Batch E is comprised of two sets of assemblies, each having a unique enrichment in order to minimize radial power peaking. There are 40 assemblies with an enrichment of 3.03 wt% and 28 assemblies with an enrichment of 2.73 wt% U-235. With this loading, the Cycle 3 burnup capacity for full power operation is expected to be between 9700 MWD/T and 10,100 MWD/T, depending on the final Cycle 2 termination point. The Cycle 3 core characteristics have been examined for Cycle 2 terminations between 8000 and 8600 MWD/T and limiting values established. The loading pattern (see Section 3) is applicable to any Cycle 2 termination point between the stated extremes.

Physics characteristics including reactivity coefficients for Cycle 3 are listed in Table 5-1 along with the corresponding values from the reference cycle. It is noted that the values of parameters actually employed in safety analyses are typically chosen to conservatively bound predicted values with accommodation for appropriate uncertainties and allowances. Table 5-2 presents a summary of CEA shutdown worths and reactivity allowances for Cycle 3 with a comparison to reference cycle data. The power dependent insertion limit (PDIL) curve and CEA group identification are unchanged from the reference cycle (Reference 2). Table 5-3 shows the reactivity worths of various CEA groups calculated at full power conditions for Cycle 3.

5.2 PHYSICS ANALYSIS METHODS

5.2.1 Uncertainties in Measured Power Distributions

The power distribution measurement biases and uncertainties which are applied to Cycle 3 are:

	<u>Non Load-Following Mode</u>	<u>Load-Following Mode</u>
Fq:	= 7.0 percent	10.0 percent
	where $F_q = F_{xy} \times F_z$, local power density	
Fr:	= 6.0 percent	8.0 percent

These values are to be used for monitoring power distribution parameters during operation.

5.2.2 Nuclear Design Methodology

5.2.2.1 Use of Coarse Mesh Neutronics Calculations

The coarse mesh computer program ROCS (Reference 7) has been used along with the standard fine mesh design program PDQ (Reference 8) in the Cycle 3 safety analysis.

- a. ROCS was used to survey a variety of core configurations to determine limiting conditions.

5.1.2 Power Distribution

Figures 5-1 through 5-3 illustrate the all rods out (ARO) planar radial power distributions at BOC 3, MOC 3 and EOC 3 that are characteristic of the high burnup end of the Cycle 2 shutdown window. These planar radial power peaks are characteristic of the major portion of the active core length between about 20 and 80 percent of the fuel height.

Figure 5-4 illustrates the planar radial power distribution within the upper 20 percent of the core produced with the insertion of the first CEA regulating group, Bank 7. In this case, the power distribution shown is predicated on the low burnup end of the Cycle 2 shutdown window, providing an illustration of maximum power peaking expected for this configuration. Higher burnup Cycle 2 shutdown points tend to reduce power peaking in this upper region of the core with Bank 7 inserted. It is a characteristic of both ARO and Bank 7 inserted conditions that the Cycle 3 peaks are highest at BOC and decrease with cycle burnup.

The radial power distributions described in this section are calculated data without uncertainties or other allowances. However, single rod power peaking values do include the increased peaking that is characteristic of fuel rods adjoining the water holes in the fuel assembly lattice. For both DNB and kw/ft safety and setpoint analyses in either rodded or unrodded configurations, the power peaking values actually used are higher than those expected to occur at any time during Cycle 3. These conservative values, which are specified in Sections 7 and 9 of this document, establish the allowable limits for power peaking to be observed during operation.

The range of allowable axial peaking is defined by the limiting conditions for operation of the axial shape index (ASI). Within these ASI limits, the necessary DNBR and kw/ft margins are maintained for a wide range of possible axial shapes. The maximum three-dimensional or total peaking factor anticipated in Cycle 3 during normal base load, all rods out operation at full power is 1.79, not including uncertainty allowances and augmentation factors.

5.1.3 Safety Related Data

5.1.3.1 Ejected CEA

The maximum reactivity worths and planar radial power peaks associated with an ejected CEA event are shown in Table 5-4 for both BOC and EOC. These values encompass the worst conditions anticipated during Cycle 3 for the planned range of Cycle 2 termination points.

5.1.3.2 Dropped CEA

The limiting parameters of dropped CEA reactivity worth and maximum increase in radial peaking factor have been calculated for Cycle 3. The results indicate that there are no changes in either of these parameters when compared to the reference cycle results.

5.1.3.3 Augmentation Factors

The procedure outlined in Reference 6 for calculating augmentation factors has been re-examined for Cycle 3. This re-examination was necessitated because of the great disparity between the number of fuel pins subject to large degrees of densification as compared to the number of pins subject to a small densification and because of the geometrical separations involved.

The peak and near peak pins for Cycle 3 occur adjacent to water holes in fuel assemblies which are expected to have only a small degree of densification. A statistical combination of gaps due to small densification effects near the peak pin and gaps due to large densification effects for the peak pin was performed. The augmentation factors for Cycle 3 based on this analysis are presented in Table 5-5 and are compared to the reference cycle values.

- b. ROCS was used to obtain axial power shapes, to weight the relative importance of fine mesh PDQ planar power and burnup distributions in the determination of three-dimensional effects and to determine the impact of the three-dimensional gross power distributions on reactivity parameters.
- c. ROCS was used to compute selected safety parameters. The calculation of those limiting parameters, which require a knowledge of 1-pin peaking factors, continues to be based on the fine mesh PDQ program.
- d. Two- and three-dimensional ROCS calculations were used in conjunction with two-dimensional PDQ calculations to obtain best estimate core parameters such as those shown in Table 5-3.

TABLE 5-1

St. Lucie Unit 1
Cycle 3 Physics Characteristics

	<u>Units</u>	<u>Reference Cycle</u>	<u>Cycle 3</u>
<u>Dissolved Boron :</u>			
<u>Dissolved boron content for criticality, CEAs withdrawn:</u>			
hot full power, equilibrium xenon, BOC	PPM	650	850
<u>Boron Worth:</u>			
hot BOC	PPM/% $\Delta\rho$	88	90
hot EOC	PPM/% $\Delta\rho$	77	80
<u>Reactivity Coefficients (CEAs Withdrawn):</u>			
<u>Moderator temperature coefficients, hot operating :</u>			
Beginning of Cycle (Equilibrium XE)	$10^{-4} \Delta\rho/^\circ\text{F}$	-0.4	-0.2
End of Cycle	$10^{-4} \Delta\rho/^\circ\text{F}$	-1.8	-1.8
<u>Doppler coefficient:</u>			
hot BOC zero power	$10^{-5} \Delta\rho/^\circ\text{F}$	-1.44	-1.44
hot BOC full power	$10^{-5} \Delta\rho/^\circ\text{F}$	-1.13	-1.13
hot EOC full power	$10^{-5} \Delta\rho/^\circ\text{F}$	-1.22	-1.22
<u>Total Delayed Neutron Fraction, β_{eff}:</u>			
Beginning of Cycle		.0060	.0060
End of Cycle		.0052	.0051
<u>Neutron Generation Time, ℓ^*</u>			
BOC	10^{-6} sec	28	28
EOC	10^{-6} sec	32	33

TABLE 5-2

St. Lucie Unit 1
Limiting Values of Cycle 3 CEA
REACTIVITY WORTHS AND ALLOWANCES,
%Δρ

	BOC		EOC	
	Reference Cycle	Reload Cycle	Reference Cycle	Reload Cycle
<u>Worth Available *</u>				
Worth of all CEAs inserted	9.6	10.5	10.0	11.4
Stuck CEA allowance	3.3	2.7	2.7	3.1
Worth of all CEAs less highest worth CEA stuck out	6.3	7.8	7.3	8.3
<u>Worth Required (Allowances)</u>				
Power defect, HFP to HZP (Doppler, Tav _g , redistribution)	1.9	1.7	2.5	2.2
Moderator voids	0.0	0.0	0.1	0.1
CEA bite, boron deadband and maneuvering band	0.6	0.6	0.6	0.6
Required shutdown margin (%Δρ)	<3.3	3.3	3.3	3.3
Total reactivity required	<5.8	5.6	6.5	6.2
<u>Available Worth Less Allowances</u>				
Margin available	>0.5	2.2	0.8	2.1

*For every accident or A00 considered in the safety analysis, a calculational uncertainty of 10% is deducted from the worth available.

TABLE 5-3

ST. LUCIE UNIT I
CYCLE 3 REACTIVITY WORTH
OF CEA REGULATING GROUPS
AT HOT FULL POWER,
 $\% \Delta \rho$

<u>Regulating CEAs</u>	<u>Beginning of Cycle</u>	<u>End of Cycle</u>
Group 7	0.78	0.84
Group 6	0.52	0.56
Group 5	0.39	0.46

Note

Values shown assume sequential group insertion.

TABLE 5-4

ST. LUCIE UNIT I
CYCLE 3 CEA EJECTION
DATA

	<u>Limiting Value</u>	
	<u>Reference Cycle</u>	<u>Cycle 3</u>
<u>Maximum Radial Power Peak</u>		
Full power with Bank 7 inserted; worst CEA ejected	3.90	3.60
Zero power with Banks 7+6+5 inserted; worst CEA ejected	8.34	8.34
<u>Maximum Ejected CEA Worth ($\% \Delta \rho$)</u>		
Full power with Bank 7 inserted; worst CEA ejected	.29	.29
Zero power with Banks 7+6+5 inserted; worst CEA ejected	.65	.65

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Note

Uncertainties and allowances are included in the above data.

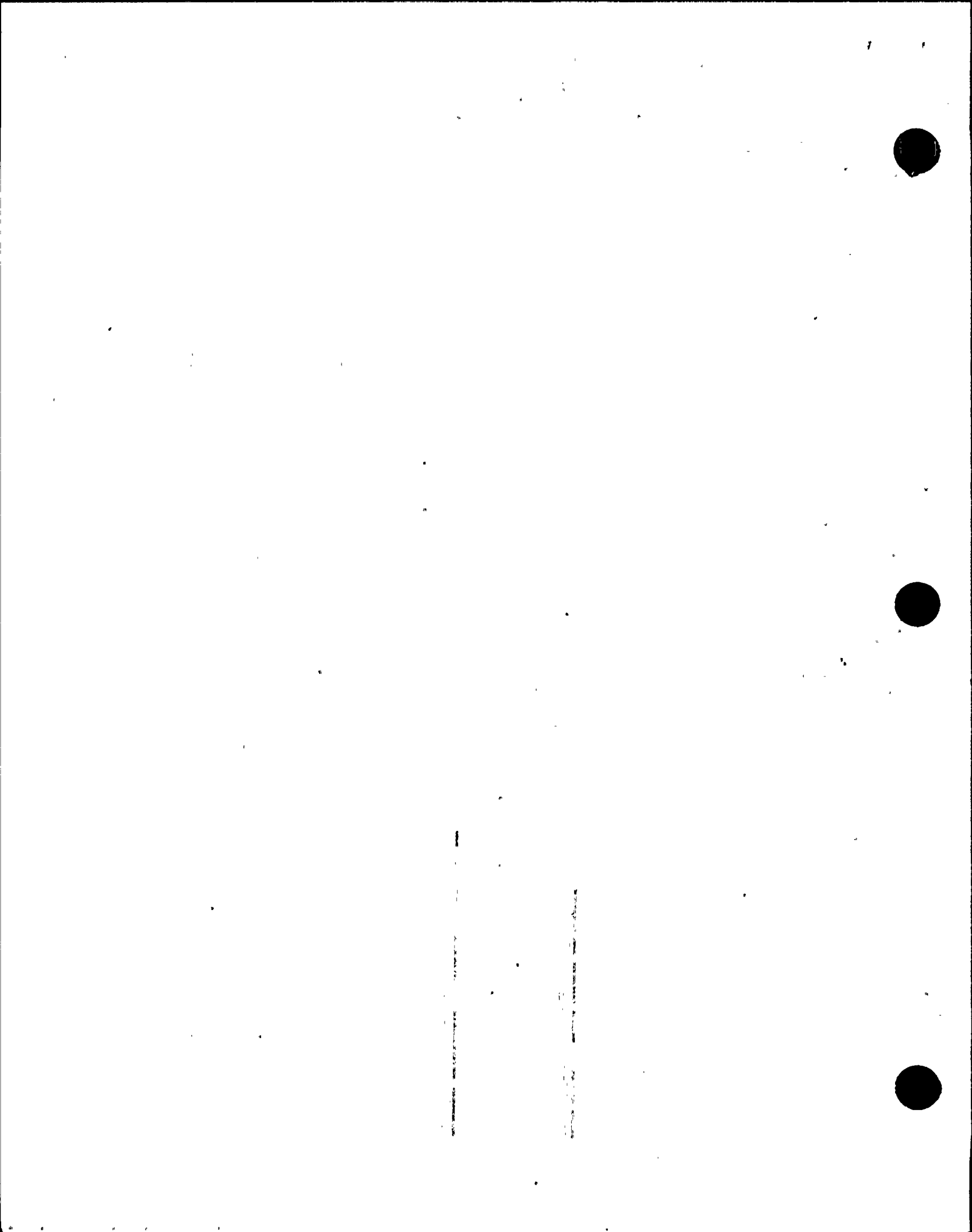


TABLE 5-5

ST. LUCIE UNIT 1
AUGMENTATION FACTORS AND GAP
SIZES FOR CYCLE 3
AND REFERENCE CYCLE

Core Height (percent)	Core Height (inches)	Reference Cycle		Reload Cycle	
		Noncollapsed Clad Augmentation Factor	Gap Size (inches)	Noncollapsed Clad Augmentation Factor	Gap Size (inches)
98.5	134.7	1.062	2.94	1.058	2.04
86.8	118.6	1.058	2.59	1.053	1.80
77.9	106.5	1.055	2.33	1.050	1.62
66.2	90.5	1.049	1.98	1.044	1.38
54.4	74.4	1.043	1.64	1.038	1.14
45.6	62.3	1.038	1.38	1.033	0.96
33.8	46.2	1.030	1.04	1.026	0.72
22.1	30.2	1.022	0.69	1.018	0.48
13.2	18.1	1.015	0.43	1.013	0.30
1.5	2.0	1.003	0.086	1.003	0.06

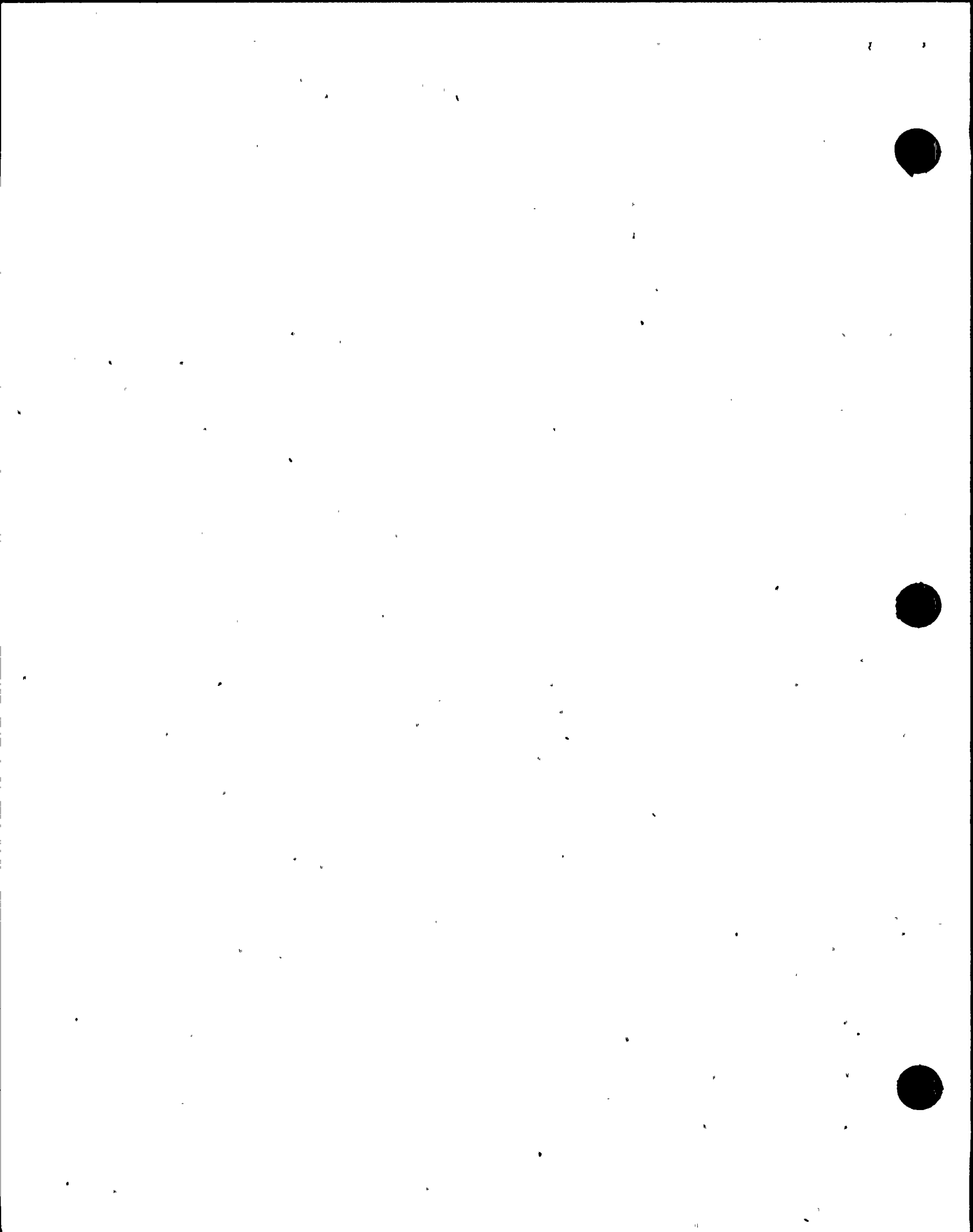
Notes

Values are based on approved model described in Reference 5.

						0.76	0.96	
				0.73	0.99	1.15	1.12	0.99
			0.84	1.18	0.90	1.23	0.96	1.30
		0.83	1.13	0.98	1.30 _X	0.98	1.17	0.87
	0.72	1.18	0.98	0.88	0.94	1.04	0.85	1.14
	0.98	0.90	1.29	0.93	1.15	0.92	1.13	0.92
	1.14	1.22	0.97	1.03	0.93	0.86	0.84	0.96
0.75	1.11	0.95	1.17	0.86	1.15	0.85	1.05	0.89
0.95	0.97	1.29	0.87	1.15	0.97	0.96	0.83	0.71

NOTE: X=MAXIMUM I-PIN PEAK=1.49

<p>St. Lucie Nuclear Power Station Unit No. 1</p>	<p>CYCLE 3 - ASSEMBLY RELATIVE POWER DENSITY BOC, EQUILIBRIUM XENON</p>	<p>Figure 5-1</p>
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						0.73	0.92	
				0.71	0.95	1.09	1.06	0.95
			0.81	1.13	0.88	1.17	0.94	1.25
		0.81	1.08	0.96	1.26	0.97	1.17	0.89
	0.71	1.13	0.96	0.89	0.95	1.06	0.90	1.17
	0.95	0.88	1.25 X	0.95	1.17	0.97	1.19	0.98
	1.08	1.17	0.97	1.06	0.98	0.94	0.93	1.05
0.73	1.06	0.94	1.17	0.90	1.20	0.93	1.15	0.99
0.91	0.94	1.25	0.89	1.18	1.03	1.05	0.93	0.83

NOTE: X=MAXIMUM 1-PIN PEAK=1.41

						0.75	0.92	
				0.74	0.95	1.06	1.04	0.94
			0.83	1.11	0.89	1.14	0.94	1.21
		0.82	1.07	0.96	1.22	0.97	1.15	0.91
	0.74	1.11	0.96	0.91	0.95	1.06	0.92	1.16
	0.95	0.89	1.22 X	0.95	1.16	0.98	1.18	1.00
0.75	1.06	1.14	0.97	1.06	0.99	0.96	0.95	1.06
	1.04	0.94	1.15	0.92	1.19	0.95	1.16	1.02
0.92	0.93	1.21	0.91	1.16	1.03	1.06	0.96	0.88

NOTE: X=MAXIMUM 1-PIN PEAK=1.35

6. THERMAL-HYDRAULIC DESIGN

6.1 DNBR Analyses

Steady state DNBR analyses of Cycle 3 at the rated power level of 2560 Mwt have been performed using the same design codes as described in the FSAR, Reference 6. Appropriate adjustments were made to the input of these codes to reflect the Cycle 3 power distribution.

Table 6-1 contains a list of pertinent thermal-hydraulic design parameters used for both safety analyses and for generating reactor protective system setpoint information.

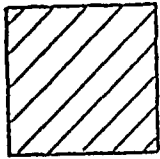
The analyses were performed in the same manner as for the reference cycle except for the following consideration on conservatism in DNBR limits due to statistical combinations.

In the DNBR limit analysis, the assumed uncertainties in various measured parameters are not combined in a single equation but are factored into functional relationships as biases at various points in the analysis. This biasing of functional relationships throughout the analysis is equivalent to adding the absolute power uncertainties (equivalent to the uncertainties in the various measured parameters) and applying the total power uncertainty to the best estimate calculation. The specific uncertainties along with their equivalent power uncertainties are given below.

ASI	0.06 ASIU	$\geq 2.2\%$
Pressure	22 PSI	$\geq 0.8\%$
Temperature	2°F	$\geq 0.9\%$
Flow	4%	$\geq 5.0\%$
Power	5% (LSSS)	$\geq 3.5\%$
	2% (LCO)	$\geq 1.4\%$

In the Cycle 3 analysis, the equivalent sum of these uncertainties is 12.4% for LSSS, 10.3% for LCO. Treating these measurement uncertainties as statistically independent, the proper method for combining them is Root Sum Square (RSS). The RSS combination yields 6.6% for LSSS and 5.8% for LCO, giving a net conservatism in the analysis of 5.8% for LSSS and 4.5% for LCO. For the Cycle 3 analysis, a partial credit of 3% has been taken for the LCO and LSSS.

Investigations have been made to ascertain the effect of the CEA guide tube wear problem and the sleeving repair on DNBR margins as established by this type of analysis. The findings were reported to NRC in Reference 4 which conclude that the wear problem and the sleeving repair do not adversely affect DNBR margin.



BANK 7 CEA INSERTIONS

						0.62	0.77	
				0.59	0.85	1.00	0.94	0.78
			0.53	0.93	0.87	1.17	0.86	0.77
		0.53	0.56	0.83	1.27	1.06	1.22	0.89
	0.59	0.93	0.83	0.89	1.08	1.24	1.06	1.37
	0.85	0.87	1.27	1.07	1.40	1.19	1.47	1.22
	1.00	1.17	1.06	1.24	1.20	1.15	1.15	1.26
0.62	0.93	0.86	1.22	1.07	1.48	1.15	1.34	1.09
0.76	0.77	0.76	0.90	1.38	1.27	1.27	1.03	0.58

NOTE: X=MAXIMUM 1-PIN PEAK=1.69

<p>St. Lucie Nuclear Power Station Unit No. 1</p>	<p>CYCLE 3 - ASSEMBLY RELATIVE POWER DENSITY WITH CEA BANK 7 INSERTED AT HFP BOC</p>	<p>Figure 5-4</p>
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6.2 Effects of Fuel Rod Bowing On DNBR Margin

Fuel rod bowing effects on DNBR margin for St. Lucie Unit 1 Cycle 3 have been evaluated within the guidelines set forth in Reference 9. Within the range of Cycle 2 termination points and Cycle 3 lifetimes identified in this document, no more than 89 assemblies will exceed the DNB reduction or penalty threshold burnup of 24,000 MWD/T. At EOC 3, the maximum burnup attained by any of these assemblies will be 33,800 MWD/T. From Reference 9, the corresponding DNB penalty for 34,200 MWD/T is 3.4 percent.

An examination of the power distributions show that the maximum radial peak at HFP in any of the assemblies that eventually exceed 24,000 MWD/T is at least ten percent less than the maximum radial peak in the entire core. Since the percent increase in DNBR has been confirmed to be never less than the percent decrease in radial peak, there exists at least ten percent DNBR margin for assemblies exceeding 24,000 MWD/T relative to the DNB limits established by other assemblies in the core.

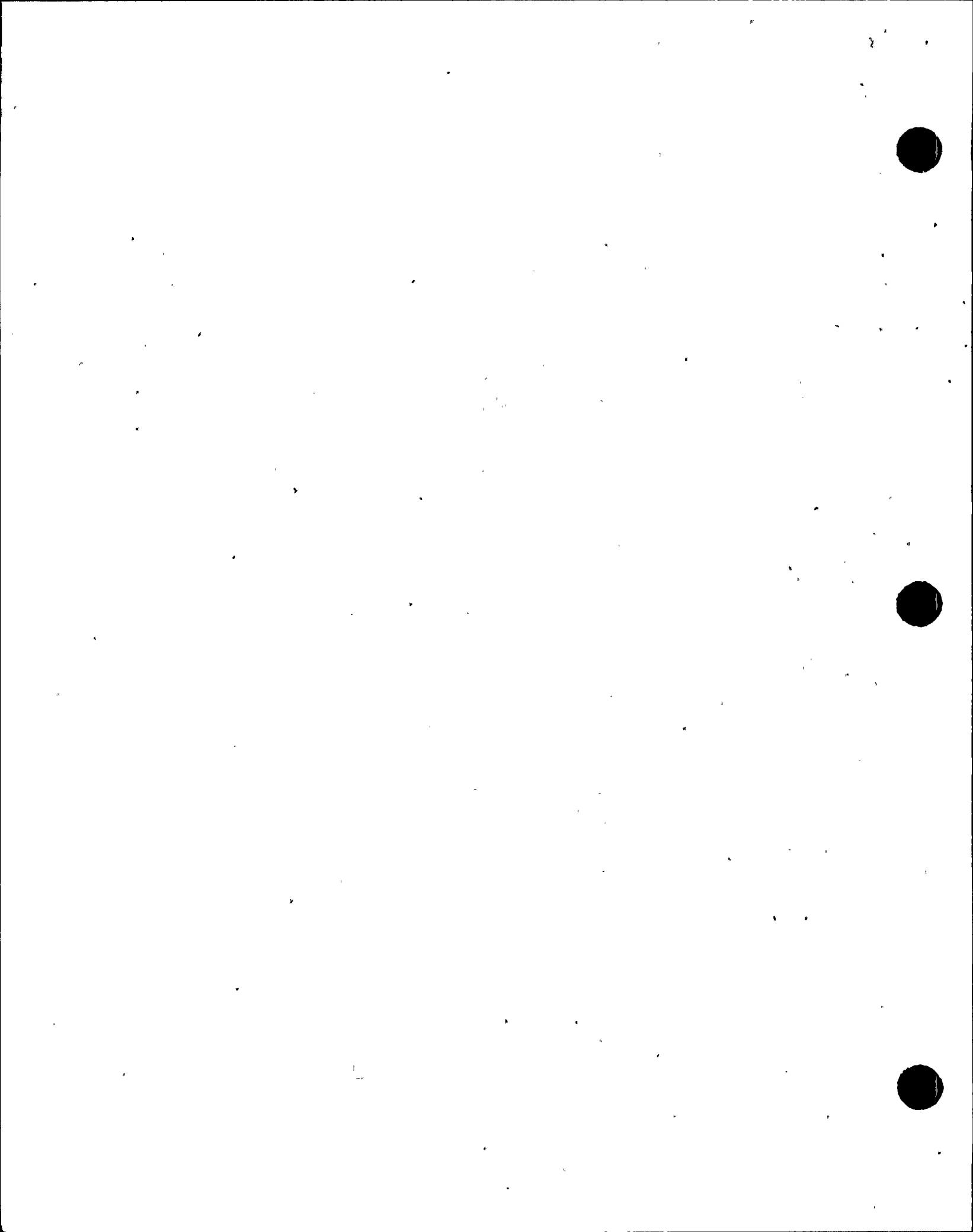


TABLE 6-1

St. Lucie Unit 1
Thermal-Hydraulic Parameters at Full Power

<u>General Characteristics</u>	<u>Unit</u>	<u>Reference Cycle 2</u>	<u>Cycle 3</u>
Total Heat Output (core only)	Mwt 10^6 BTU/hr	2560 8737	2560 8737
Fraction of Heat Generated in Fuel Rod		.975	.975
Primary System Pressure			
Nominal	PSIA	2250	2250
Minimum in steady state	PSIA	2200	2200
Maximum in steady state	PSIA	2300	2300
Design Inlet Temperature	°F	544	544
Total Reactor Coolant Flow (minimum steady state)	GPM 10^6 lb/hr	370,000 140.2*	370,000 140.2*
Coolant Flow Through Core	10^6 lb/hr	135.0*	135.0*
Hydraulic Diameter (nominal channel)	ft	0.044	0.044
Average Mass Velocity	10^6 lb/hr-ft ²	2.53*	2.53*
Pressure Drop Across Core (minimum steady state flow irreversible ΔP over entire fuel assembly)	PSI	10.2	10.3
Total Pressure Drop Across Vessel (based on nominal dimensions and minimum steady state flow)	PSI	33.3	33.5
Core Average Heat Flux (accounts for above fraction of heat generated in fuel rod and axial densification factor)	BTU/hr-ft ²	177,700	174,400
Total Heat Transfer Area (accounts for axial densification factor)	ft ²	47,940	48,860
Film Coefficient at Average Conditions	BTU/hr-ft ² -°F	5820	5820
Maximum Clad Surface Temperature	°F	657	657
Average Film Temperature Difference	°F	31	31
Average Linear Heat Rate of Undensified Fuel Rod (accounts for above fraction of heat generated in fuel rod)	kw/ft	5.94	5.83
Average Core Enthalpy Rise	BTU/lb	65*	65*

* Calculated at design inlet temperature, nominal primary system pressure.

TABLE 6-1 (continued)

<u>Calculational Factors</u>	<u>Reference Cycle</u>	<u>Cycle 2</u>
Engineering Heat Flux Factor	1.03	1.03
Engineering Factor on Hot Channel Heat Input	1.03	1.03
Inlet Plenum Nonuniform Distribution	1.05	1.05
Rod Pitch, Bowing and Clad Diameter	1.065	1.065
Fuel Densification Factor (axial)	1.01	1.01
Fuel Rod Bowing Augmentation Factor on Fr	1.018	1.018
Statistical Component of Fr ¹¹ @ 95/95 Confidence Level	1.0513	1.0513



References (Sections 1 through 6)

1. CEN-79-P, "Reactor Operation With Guide Tube Wear", February 3, 1978
2. Letter, Robert E. Uhrig (FP&L) to Victor Stello (NRC), dated March 22, 1978, "St. Lucie Unit 1 Docket No. 50-335 Proposed Amendment to Facility Operating License DPR-67"
3. CEIIPD-187, "CEPAH Method of Analyzing Creep Collapse of Oval Cladding", June 1975
4. CEN-80(N)-P, "Millstone Unit 2 Reactor Operation With Modified CEA Guide Tubes", February 8, 1978
5. CENPD-139, "C-E Fuel Evaluation Model Topical Report", July 1, 1974
6. St. Lucie Nuclear Power Plant (Formerly Hutchinson Island) Unit One, Final Safety Analysis Report, in support of Docket No. 50-335
7. Omaha Public Power District Ft. Calhoun Station Unit No. 1, Docket No. 50-285, Proposed Amendment to Facility Operating License DPR-40, August 1978
8. W. R. Cadwell, "PDQ-7 Reference Manual", WAPD-TM-678, January 1968
9. "Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors", Pages 21 and 26 (NRC Report)

7.0 ACCIDENT AND TRANSIENT ANALYSIS

The purpose of this section is to present the results of the safety analysis (other than LOCA) for St. Lucie Unit 1, Cycle 3, at 2560 Mwt. The events considered for this analysis are listed in Table 7-1. These are the design basis events for the plant. These events can be categorized into the following groups.

1. Anticipated Operational Occurrences for which the Reactor Protection System assures that the Specified Acceptable Fuel Design Limits (SAFDLs) will not be exceeded;
2. Anticipated Operational Occurrences for which the initial steady state overpower margin must be maintained in order to assure that the SAFDLs will not be exceeded;
3. Postulated Accidents.

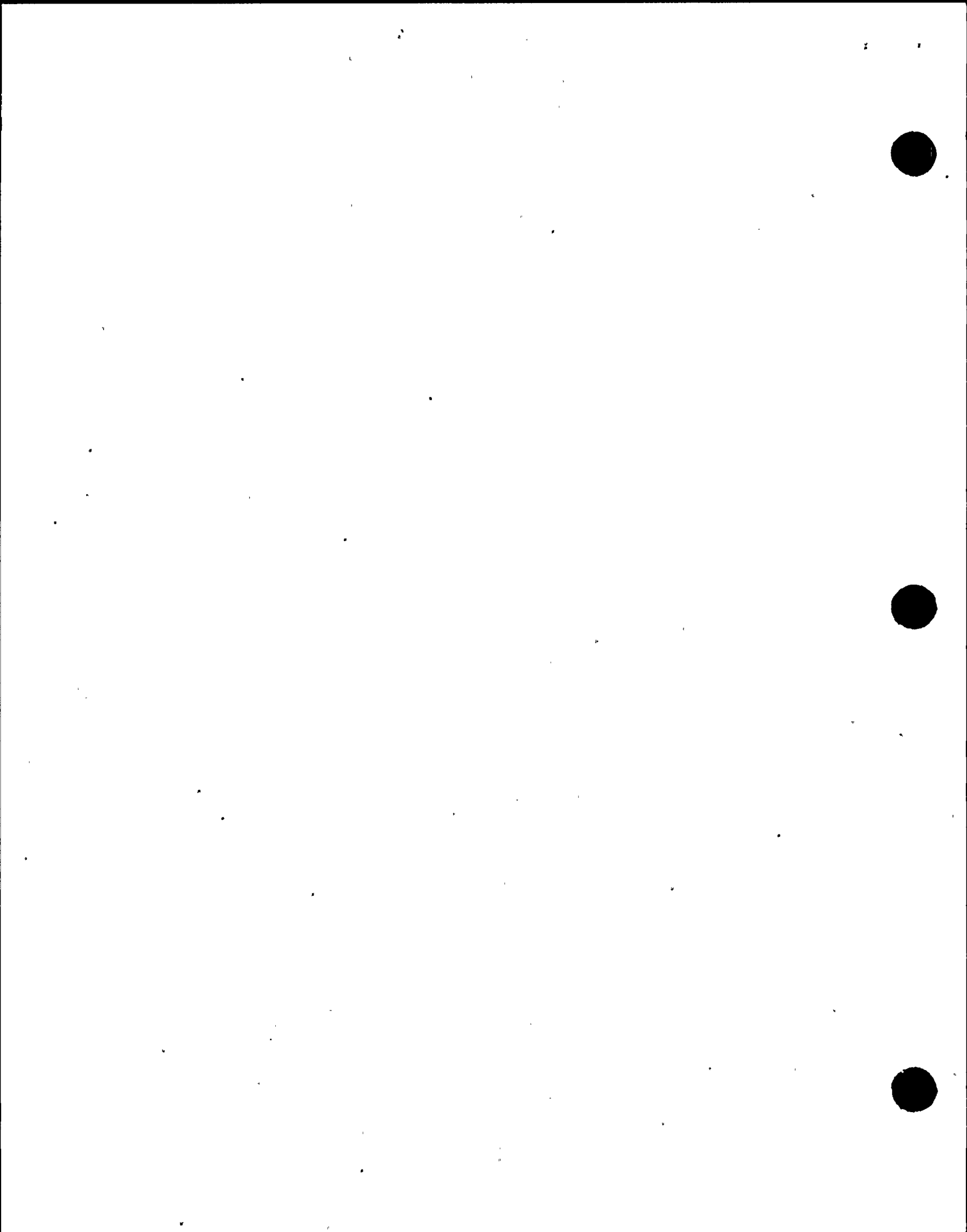
Each of the events listed in Table 7-1 has been reviewed for Cycle 3 to determine if an explicit reanalysis was required. Table 7-1 indicates the analysis status of each transient. The review of each design basis event (DBE) entails a comparison between all the current and reference cycle key transient parameters that significantly impact the results of the event. The reference cycle is one for which an explicit analysis was performed for the event in question. If all the current cycle values of key parameters for a particular event are bounded by (conservative with respect to) the reference cycle, no reanalysis is required. In some instances, a reanalysis is performed if it is deemed beneficial from the standpoint of enhanced operation flexibility or if it is desired to bound parameters which are expected to become more adverse in future cycles.

The reference cycle for this analysis is St. Lucie Unit 1, Cycle 2.

The results of the review were that the key input parameters to all the DBEs for Unit 1, Cycle 3, operation are less limiting than the specified reference cycle input parameters, except for the following:

1. Azimuthal Tilt Allowance
2. Radial Peaking Factors, F_{XY} , F_r
3. Seized Rotor Pin Census Data
4. Different Axial Power Distributions
5. CEA Ejection Pin Census Data

In addition to changes in these key parameters, it was deemed to be desirable to use augmentation factors and CEA scram position versus time curves which bound future cycles. Thus, a maximum augmentation factor of 1.071 was used



instead of 1.058 quoted in Table 5.5. Also, the scram rod insertion time to 90% inserted was increased from 3.0 to 3.1 seconds.

As a result of the above noted changes in key parameters, the Control Element Assembly Withdrawal, RCS Depressurization, Loss of Coolant Flow, CEA Ejection, and Seized Rotor transients have been reanalyzed as indicated in Table 7-1.

For all DBEs other than those reanalyzed, reference cycle safety analyses, bound the results that would be obtained for Cycle 3 and demonstrate safe operation of the St. Lucie Unit 1, Cycle 3 at 2560 MWt.

The CEA Ejection event was reanalyzed for Cycle 3 due to a more adverse pin census, an increased azimuthal tilt allowance, and the use of a more conservative CEA drop time to 90% insertion relative to Cycle 2.

The CEA Withdrawal event has been reanalyzed due to the use of a more conservative CEA drop time to 90% insertion.

For Cycle 3, both the Planar Radial Peaking Factor (F_{xy}) and the Integrated Radial Peaking Factor (F_r) have increased in comparison to Cycle 2. The limiting radials, (F_{xy}) and (F_r), for all steady state and routine operating transients (i.e., limiting conditions for operations) are explicitly included in the shape analysis on which setpoints are based. In the evaluation of all DBEs (Table 7.1), the higher initial radials have no adverse impact on the safety analysis except for the Loss of Coolant Flow and Seized Rotor events. Besides the increase in (F_{xy}) and (F_r) and azimuthal tilt allowance relative to Cycle 2, a more conservative CEA drop time to 90% insertion was used to bound future cycles. Therefore, the Loss of Coolant Flow and Seized Rotor events have been reanalyzed.

The Reactor Coolant System Depressurization event was also reanalyzed for Cycle 3 due to the assumption of a longer CEA drop time to 90% insertion.

TABLE 7-1

ST. LUCIE UNIT 1, CYCLE 2
EVENTS CONSIDERED IN TRANSIENT AND ACCIDENT ANALYSIS

	<u>Analysis Status</u>
Anticipated Operational Occurrences for which the RPS Assures no Violation of SAFDLs:	
Control Element Assembly Withdrawal	Reanalyzed
Boron Dilution	Not Reanalyzed
Startup of an Inactive Reactor Coolant Pump	Not Reanalyzed
Excess Load	Not Reanalyzed
Loss of Load	Not Reanalyzed
Loss of Feedwater Flow	Not Reanalyzed
Excess Heat Removal due to Feedwater Malfunction	Not Reanalyzed
Reactor Coolant System Depressurization	Reanalyzed
Loss of Coolant Flow ¹	Reanalyzed
Loss of AC Power	Not Reanalyzed
Anticipated Operational Occurrences which are Dependent on Initial Overpower Margin for Protection Against Violation of SAFDLs:	
Loss of Coolant Flow	Reanalyzed
Loss of AC Power	Not Reanalyzed
Full Length CEA Drop	Not Reanalyzed
Part Length CEA Drop	Not Reanalyzed
Part Length CEA Malpositioning	Not Reanalyzed
Transients Resulting from Malfunction of One Steam Generator	Not Reanalyzed
Postulated Accidents:	
CEA Ejection	Reanalyzed
Steam Line Rupture	Not Reanalyzed
Steam Generator Tube Rupture	Not Reanalyzed
Seized Rotor	Reanalyzed

¹Requires Low Flow Trip.

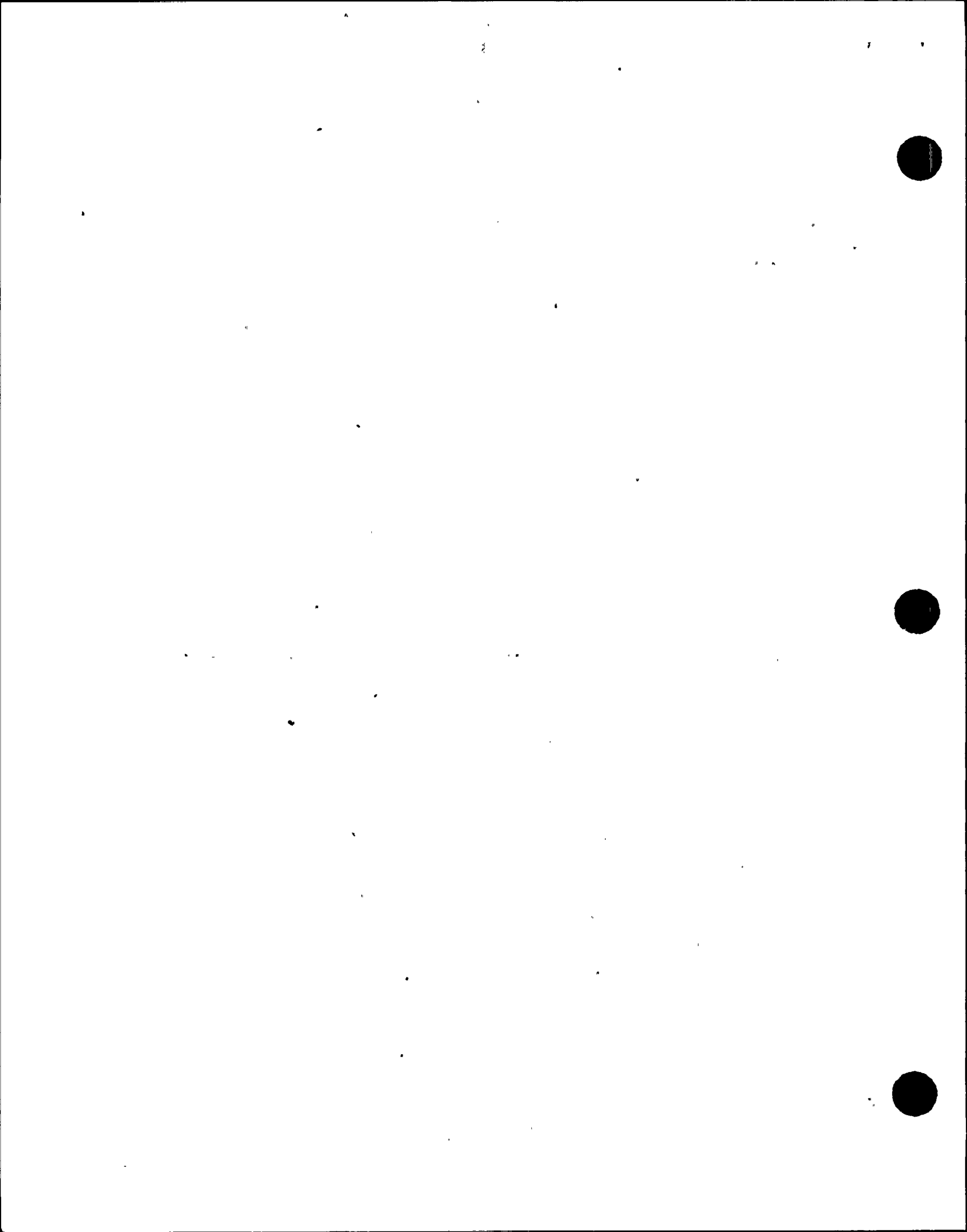


TABLE 7-2

ST. LUCIE 1
CORE PARAMETERS INPUT TO SAFETY ANALYSES

<u>Physics Parameters</u>	<u>Units</u>	<u>Reference Cycle Values</u>	<u>Cycle 3 Values</u>
<u>Planar Radial Peaking Factors</u>			
For DNB Margin Analyses			
Unrodded Region		1.53	1.59
Bank 7 Inserted		1.70	1.80
For Planar Radial Component of 3-D Peak (kw/ft Limit Analyses)			
Unrodded Region		1.56	1.58
Bank 7 Inserted		1.76	1.82
Peak Augmentation Factor		1.062	1.071 *
Moderator Temperature Coefficient	$10^{-4} \Delta\rho/^{\circ}\text{F}$	-2.5 \rightarrow +.5	-2.5 \rightarrow +.5
Shutdown Margin (Value used in Zero Power (SLB) (1 loop/2 loop)		-4.1/-3.3	-4.1/-3.3
<u>Safety Parameters</u>			
Power Level	MWt	2611	2611
Maximum Steady State Core Inlet Temperature	$^{\circ}\text{F}$	544	544
Minimum Steady State RCS Pressure	psia	2200	2200
Reactor Coolant Core Flow	10^6 lb/hr	134.9	134.9
Full Power Axial Shape Index Limit	Ip	-.23	-.23
Maximum CEA Insertion at Full Power	% Insertion of Group 7	25	25
<u>Minimum Allowable Initial Peak'</u>			
Linear Heat Rate for transients other than LOCA	kw/ft	16.0	16.0
Steady State Linear Heat Rate to Fuel Centerline Melt	kw/ft	21.0	21.0
CEA Drop Time from Removal of Power to Raising Coils to 90% Insertion	Sec	3.0	3.1
Three Pump Plenum Factor		1.09	1.09

*A conservative value was used in the safety analysis to bound later cycles.



7.1 CEA WITHDRAWAL EVENT

The CEA withdrawal event was reanalyzed for Cycle 3 due to the increase in the CEA drop time to 90% insertion.

As stated in CENPD-199-P (Reference 1), the CEA Withdrawal event initiated at rated thermal power is one of the DBEs analyzed to determine a bias factor used in establishing the TM/LP setpoints. This bias factor, along with conservative temperature, pressure, and power readings assures that the TM/LP trip prevents the DNBR from dropping below the SAFDL limits (DNBR = 1.30 based on W-3 correlation) for a CEA Withdrawal event.

Hence, this event was analyzed for Cycle 3 to generate the bias term input to the TM/LP trip. The CEA Withdrawal transient may require protection against exceeding both the DNBR and fuel centerline melt (kw/ft) SAFDLs. Depending on the initial conditions and the reactivity insertion rate associated with the CEA withdrawal, either the Variable High Power Level or Thermal Margin/Low Pressure (TM/LP) trip reacts to prevent exceeding the DNBR SAFDL. An approach to the kw/ft limit is terminated by either the Variable High Power Level trip or the Local Power Density trip.

The zero power case was analyzed to demonstrate that SAFDLs are not exceeded. For the zero power case, a reactor trip, initiated by the variable high power trip at 25% (15% + 10% uncertainty) of rated thermal power, was assumed in the analysis.

The key parameters for the cases analyzed are reactivity insertion rate due to rod motion and moderator temperature feedback effects, and initial axial power distribution. The Resistance Temperature Detector (RTD) response time is also important in determining the pressure bias factor.

The range of reactivity insertion rates considered in the analysis is given in Table 7.1-1, along with the values of other key parameters used in the analysis of this event. These parameters were chosen to produce the most severe rate of change of DNBR at the time a trip is encountered, thereby producing the most limiting case in terms of SAFDL protection requirements. The initial axial power shape and the corresponding scram worth versus insertion used in the analysis of both cases is a bottom peaked shape. This power distribution maximizes the time required to terminate the decrease in DNBR following a trip.

The CEA Withdrawal transient initiated at 102% of 2560 MWt results in the maximum pressure bias factor of 52.0 psia. This bias factor accounts for measurement system processing delays during the CEA Withdrawal event. The pressure bias factor for this cycle has not increased from the reference cycle, since the decrease in maximum reactivity insertion rate compensates for the increased CEA drop time and helps improve the results. This pressure bias factor is used in generating TM/LP trip setpoint and will prevent the SAFDLs from being exceeded during a CEA withdrawal event.

The zero power case initiated at the limiting conditions of operation results in a minimum W-3 DNBR of 2.4. Also, the analysis shows that the fuel centerline temperatures are well below those corresponding to the fuel centerline melt SAFDL.

The Sequence of Events for the zero power case is presented in Table 7.1-2. Figures 7.1-1 to 7.1-4 presents the transient behavior of core power, core average heat flux, the coolant temperatures, and the RCS pressure.

The analysis of the CEA Withdrawal event presented herein, shows that the DNB and fuel centerline melt SAFDLs will not be exceeded during a CEA Withdrawal transient for Cycle 3, and that the pressure bias for the TM/LP generated for Cycle 2 is conservative.

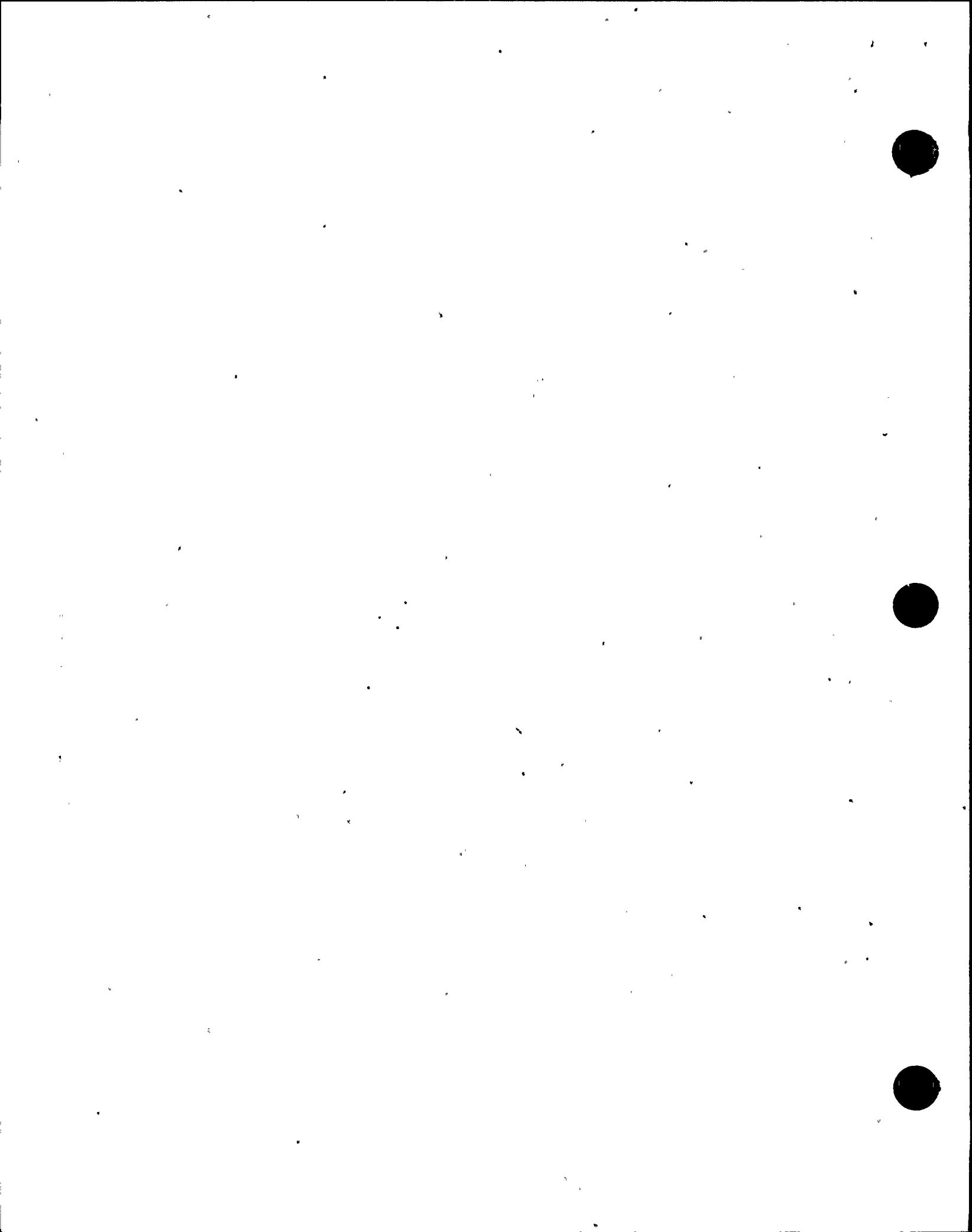
TABLE 7.1-1

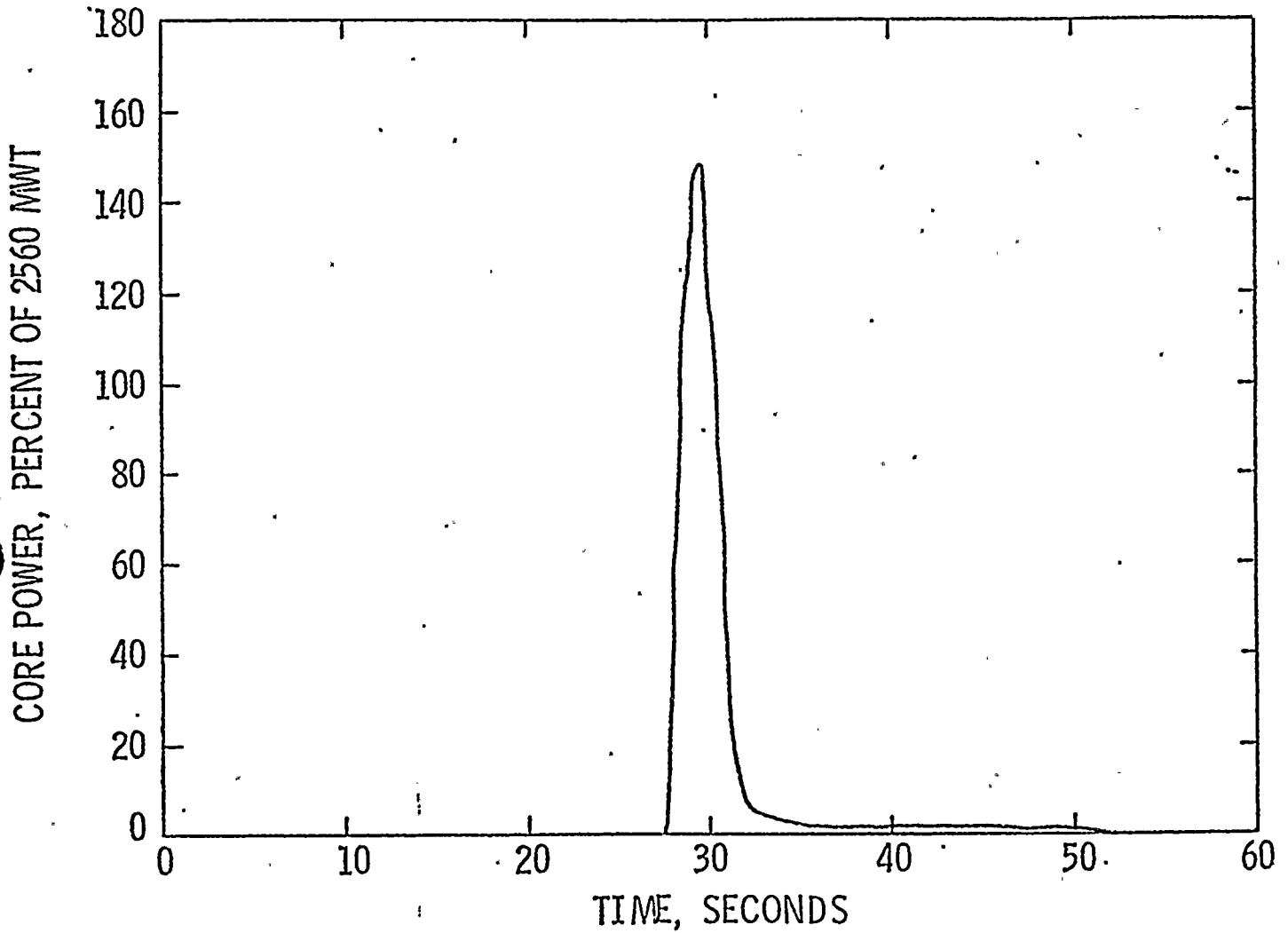
KEY PARAMETERS ASSUMED IN THE CEA WITHDRAWAL ANALYSIS

<u>Parameter</u>	<u>Units</u>	<u>Cycle 2</u>	<u>Cycle 3</u>
Initial Core Power Level	MWt	0-102% of 2560	0-102% of 2560
Core Inlet Coolant Temperature	$^{\circ}\text{F}$	532-544	532-544
Reactor Coolant System Pressure	psia	2200	2200
Moderator Temperature Coefficient	$10^{-4} \Delta\rho/^{\circ}\text{F}$	+5	+5
Doppler Coefficient Multiplier		.85	.85
CEA Worth at Trip - FP	$10^{-2} \Delta\rho$	-4.32	-4.32
CEA Worth at Trip - ZP	$10^{-2} \Delta\rho$	-3.3	-3.3
Reactivity Insertion Rate	$\times 10^{-4} \Delta\rho/\text{sec}$	0 to 2.0	0 to 1.3
Holding Coil Delay Time	sec	0.4	0.5
CEA Time to 90 Percent Insertion (Including Holding Coil Delay)	sec	3.0	3.1
Resistance Temperature Detector Response Time	sec	8.0	8.0

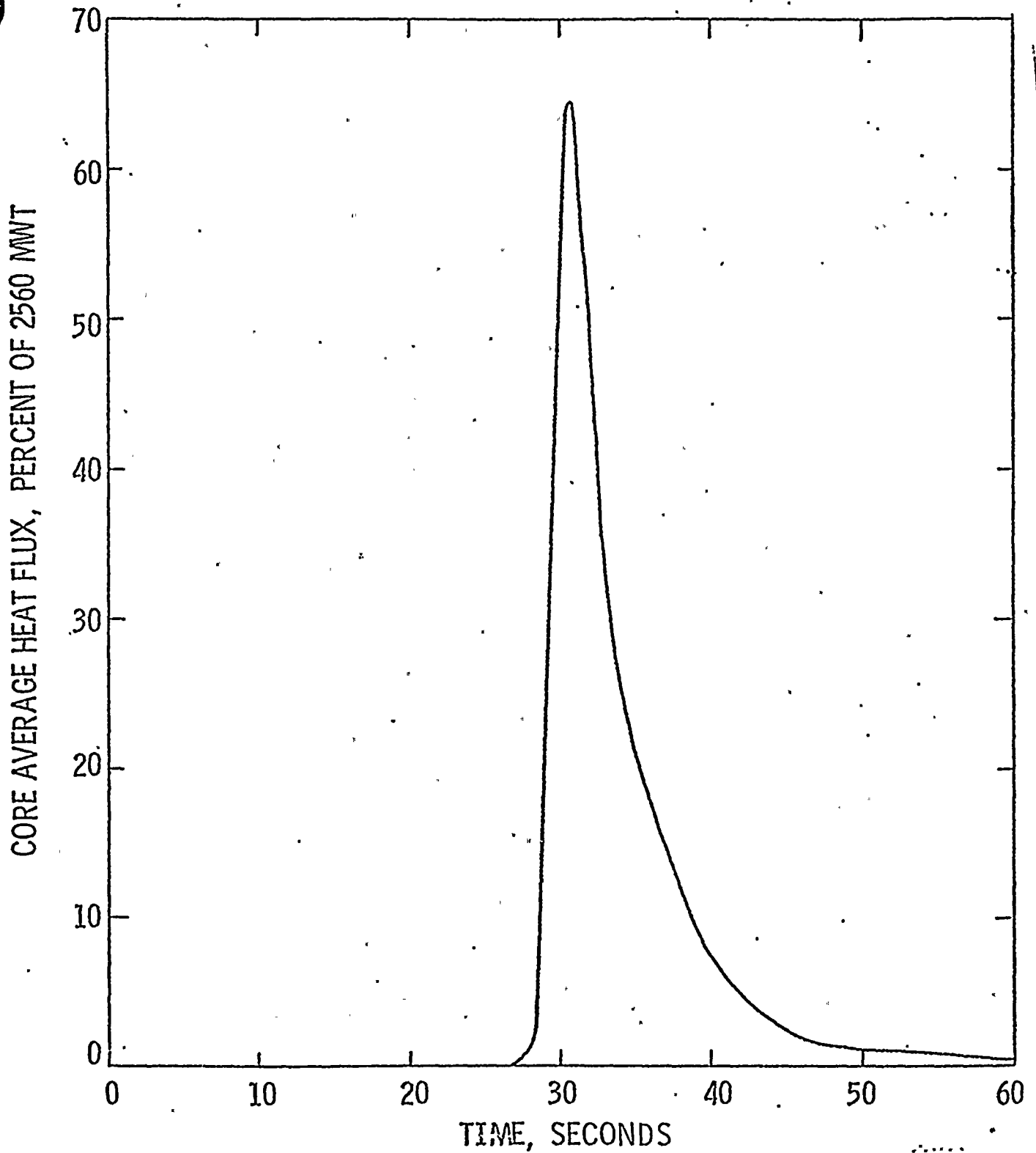
TABLE 7.1-2SEQUENCE OF EVENTS FOR
CEA WITHDRAWAL FROM ZERO POWER

<u>Time (Sec)</u>	<u>Event</u>	<u>Setpoint or Value</u>
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	--
28.1	High Power Trip Signal Generated	25% of 2560 MWt
28.5	Reactor Trip Breakers Open	--
29.0	CEAs Begin to Drop into Core	--
29.3	Maximum Power Reached	148% of 2560 MWt
30.6	Maximum Heat Flux Reached	64.6% of 2560 MWt
30.6	Minimum W-3 DNBR Occurs	2.40
32.7	Maximum Pressurizer Pressure Reached	2358 psia





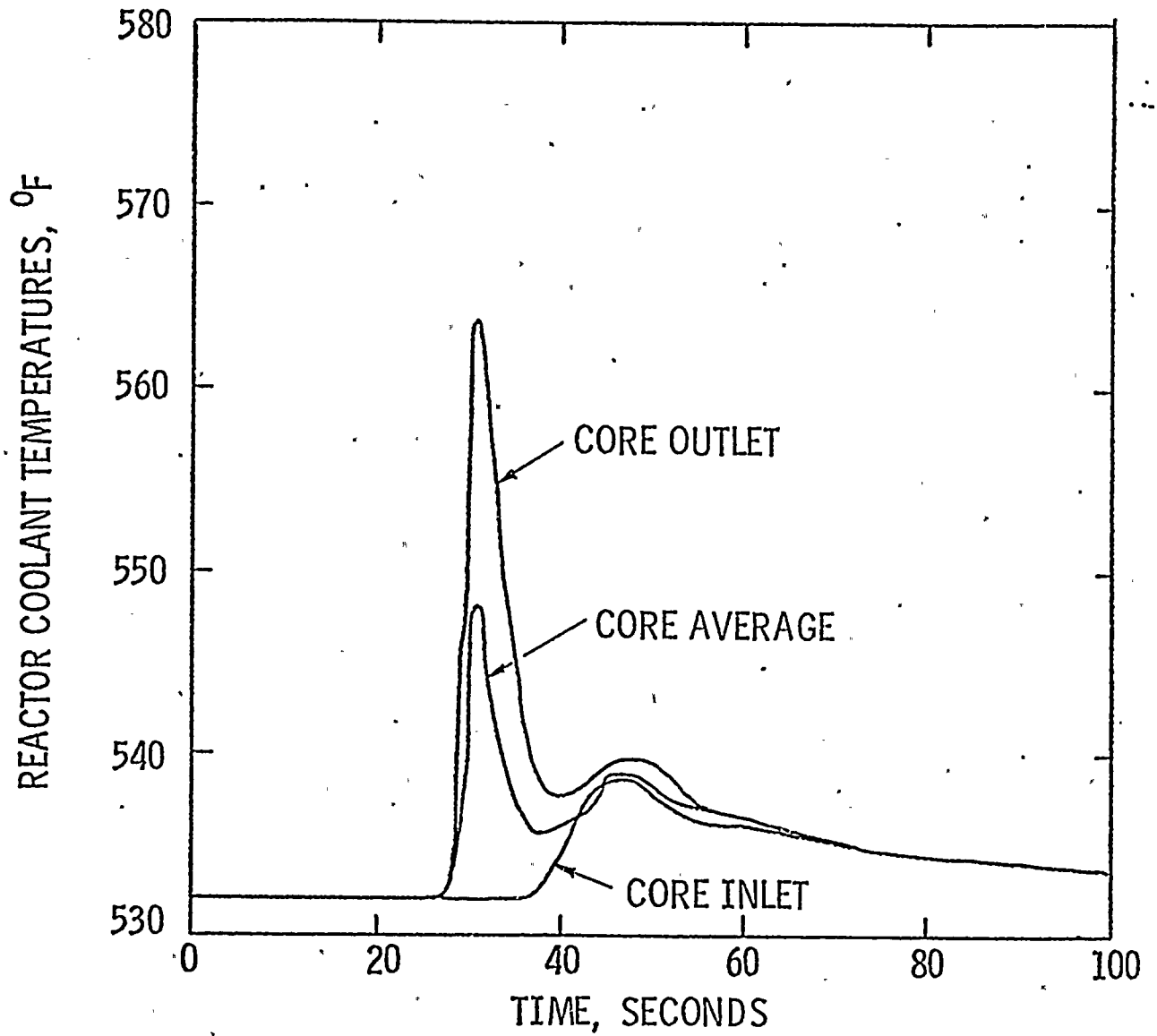
FLORIDA POWER & LIGHT CO. St. Lucie Plant	CEA WITHDRAWAL. CORE POWER VS TIME	Figure 7.1-1
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CEA WITHDRAWAL
CORE AVERAGE HEAT FLUX VS TIME

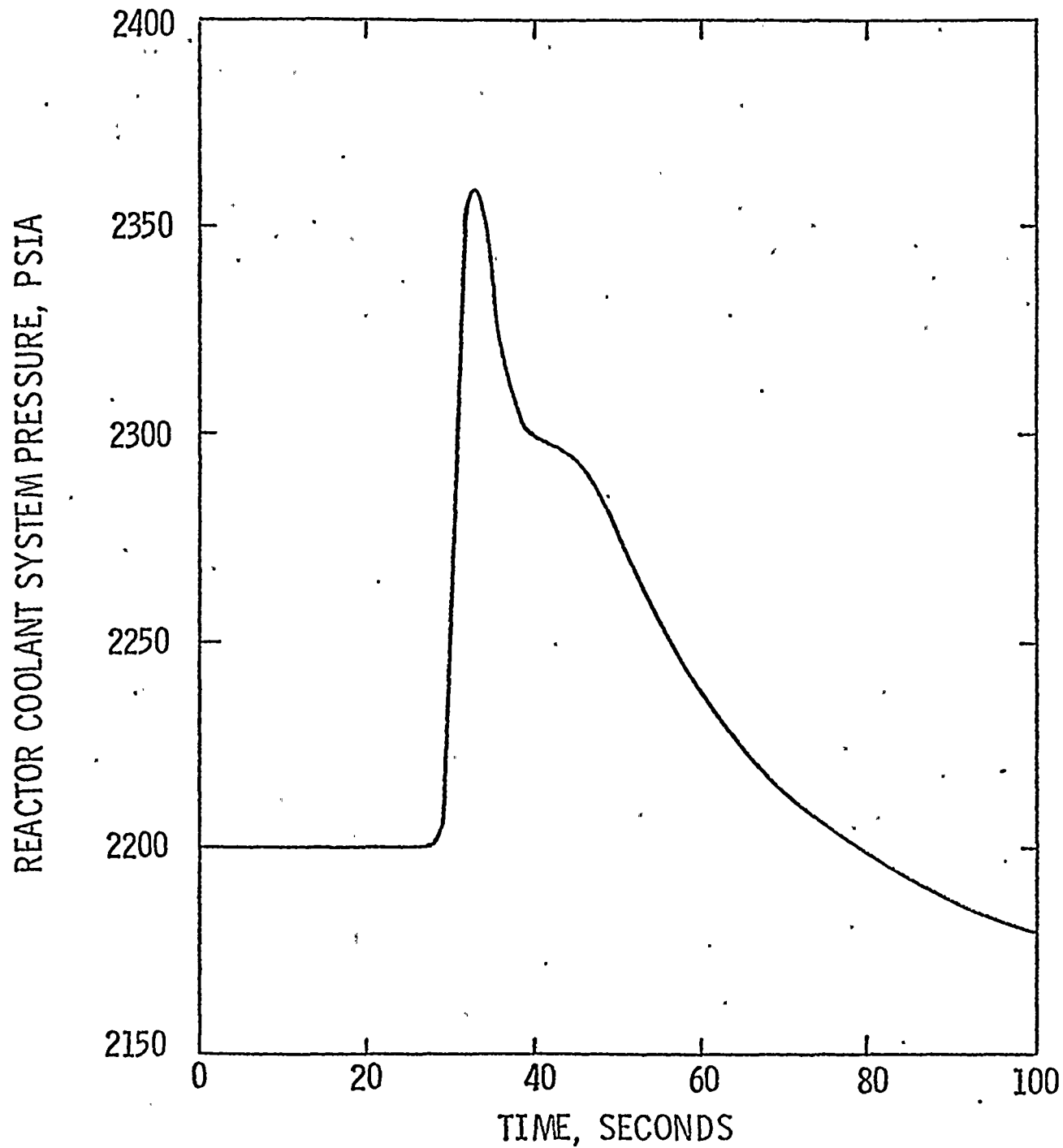
Figure
7.1-2



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CEA WITHDRAWAL
REACTOR COOLANT SYSTEM TEMPERATURES VS TIME

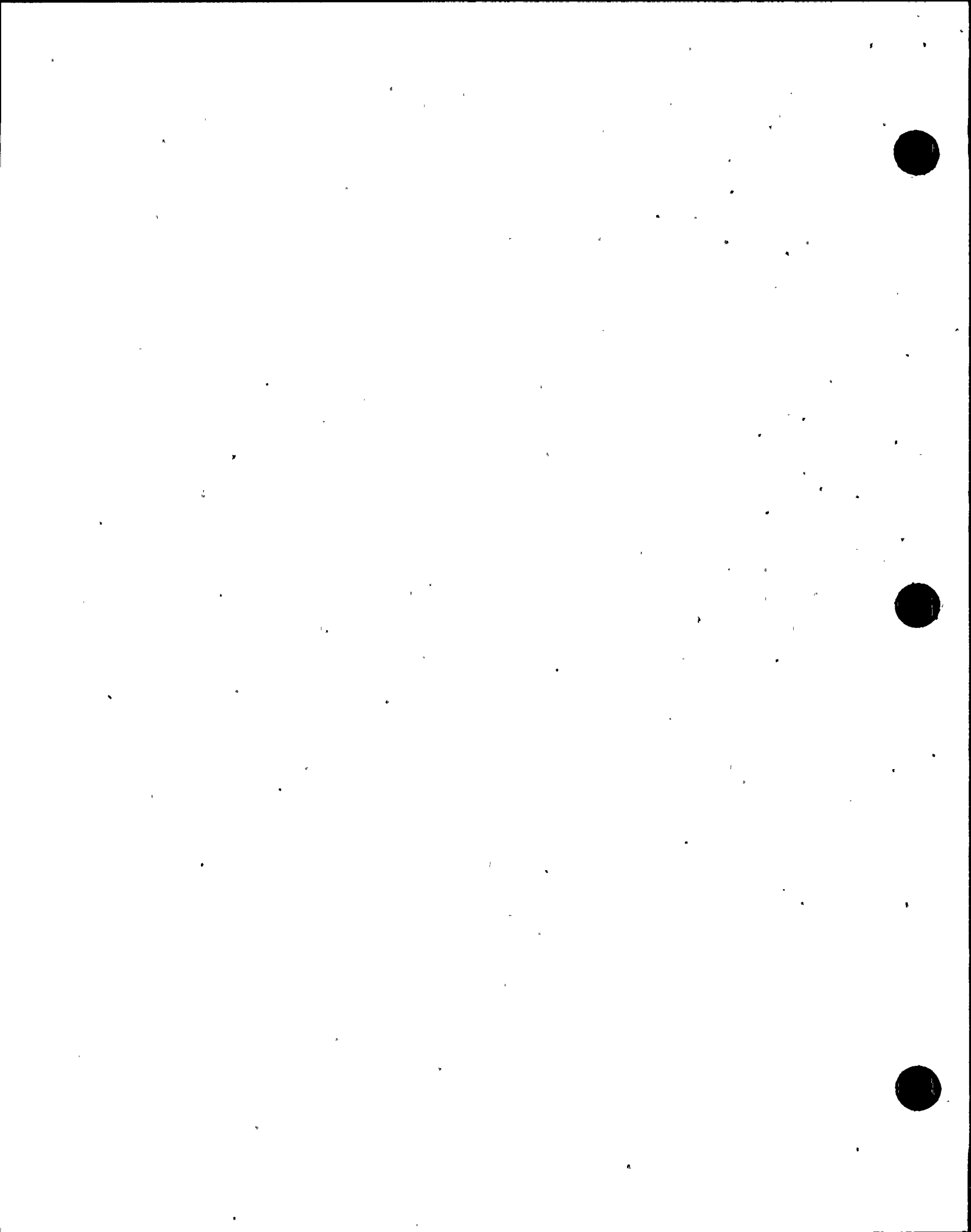
Figure
7.1-3



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Unit 2

CEA WITHDRAWAL
REACTOR COOLANT SYSTEM PRESSURE VS TIME

Figure
7.1-4



7.2 RCS DEPRESSURIZATION EVENT

The RCS Depressurization event was reanalyzed for Cycle 3 due to a more conservative CEA drop time to the 90% insertion value of 3.1 seconds for Cycle 3 in comparison to the Cycle 2 value of 3.0 seconds. As stated in CENPD-199-P (Reference 1), this event is one of the DBEs analyzed to determine a bias term input to the TM/LP trip. Hence, this event was analyzed for Cycle 3 to obtain a pressure bias factor. This bias factor accounts for measurement system processing delays during this event. The trip setpoints incorporating a bias factor at least this large will provide adequate protection to prevent the DNBR SAFDL from being exceeded during this event.

The assumptions used to maximize the rate of pressure decrease and, consequently, the fastest approach to DNBR SAFDLs are:

- 1) The event is assumed to occur due to an inadvertent opening of both pressurizer relief valves while operating at rated thermal power. This results in a rapid drop in the RCS pressure and, consequently, a rapid decrease in DNBR.
- 2) The initial axial power shape and the corresponding scram worth versus insertion used in the analysis of both cases is a bottom peaked shape. This power distribution maximizes the time required to terminate the decrease in DNBR following a trip.
- 3) The charging pumps, the pressurizer heaters, and the pressurizer backup heaters are assumed to be inoperable. This maximizes the rate of pressure decrease and, consequently, maximizes the rate of approach to DNBR SAFDL.

The analysis of this event shows that a pressure bias factor of 30.0 psia is adequate. This is less than that required by the CEA Withdrawal event. Hence, the use of the pressure bias factor determined by the CEA Withdrawal event will prevent exceeding the SAFDLs during an RCS Depressurization event.

7.3 LOSS OF COOLANT FLOW EVENT

The Loss of Coolant Flow event was reanalyzed for Cycle 3 due to an increase in the radial peaking factor, the use of a more conservative CEA drop time to 90% insertion, and changes in the axial power distributions which included increases in axial peaks for some of the distributions analyzed. The purpose of the reanalysis is to demonstrate the Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during this event.

The methodology employed for Cycle 3 is identical to that used for Cycle 2 and is discussed in detail in Reference 1, "C-E Setpoint Methodology".

The 4-pump loss of flow was initiated at the Limiting Conditions for Operation (LCO) since this produces the most adverse transient. All the key input parameters for Cycle 3 are listed in Table 7.3-1. In addition, Table 7.3-1 contains a comparison between the input parameters for Cycle 3 and Cycle 2. The loss of flow event is caused by an assumed loss of power to all four reactor coolant pumps. At 0.9 seconds, a low flow signal is generated at 93% of minimum guaranteed flow and at 2.05 seconds, the CEAs begin to drop into the core to mitigate the transient.

In Table 7.3-2, the NSSS and RPS responses are shown for an $I_p = -0.23$. Figures 7.3-1 through 7.3-5 present the time dependent NSSS parameters of core flow fraction, core power, core heat flux, RCS pressure, and coolant temperatures for $I_p = -0.23$. This shape index is conservative with respect to the most negative shape index ($I_p = -0.21$) allowed by the LCOs at 100% power. Figure 7.3-6 shows the DNBR as a function of time for this limiting case ($I_p = -.23$) which results in a minimum hot channel DNBR = 1.31 based on the W-3 correlation.

Similar calculations were performed for the Loss of Coolant Flow event over a range of axial power shapes, core burnups, and CEA configurations to determine the DNBR margin requirements as discussed in CENPD-199. The case presented here is the most adverse of those allowed at the negative extreme of the shape index LCO limits.

For the case of the loss of coolant flow event arising from the simultaneous loss of power to all four reactor coolant pumps, the low flow trip in conjunction with the initial overpower margin built into the Technical Specifications LCO limits maintain the minimum DNBR greater than or equal to 1.30 during this event.

TABLE 7.3-1

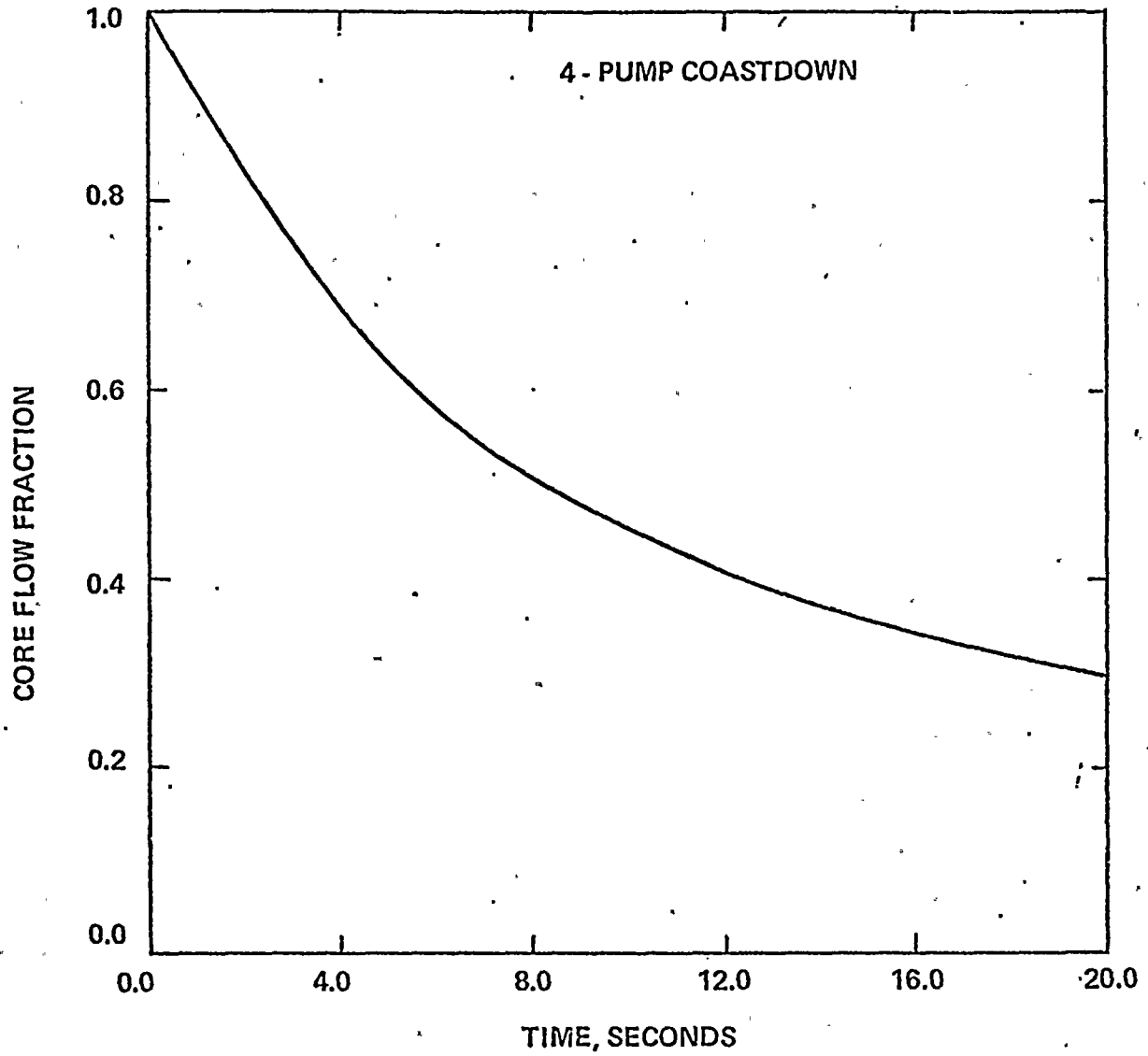
KEY PARAMETERS FOR THE LOSS OF COOLANT FLOW INCIDENT ANALYSIS

<u>Parameter</u>	<u>Units</u>	<u>Unit 1, Cycle 2</u>	<u>Unit 1, Cycle 3</u>
Initial Core Power Level	(MWt)	102% of 2560	102% of 2560.
Core Inlet Coolant Temperature	(°F)	544	544
Core Mass Flow Rate	(10 ⁶ lbm/hr)	134.9	134.9
Reactor Coolant System Pressure	(psia)	2200	2200
Moderator Temperature Coefficient	(10 ⁻⁴ Δρ/°F)	+ .5	+ .5
Doppler Coefficient Multiplier	--	.85	.85
LFT Response Time	(sec)	0.65	0.65
CEA Holding Coil Delay	(sec)	0.4	0.5
CEA Time to 90% Insertion (Including Holding Coil Delay)	(sec)	3.0	3.1
CEA Worth Available at Trip from FP	(10 ⁻² Δρ)	-5.41	-5.41
Total Unrodded Radial Peaking Factor, F _r ¹		1.58	1.64
4-Pump RCS Flow Coastdown		Figure 7.3-1 of Reference 9	Figure 7.3-1

TABLE 7.3-2

LOSS OF FLOW SEQUENCE OF EVENTS

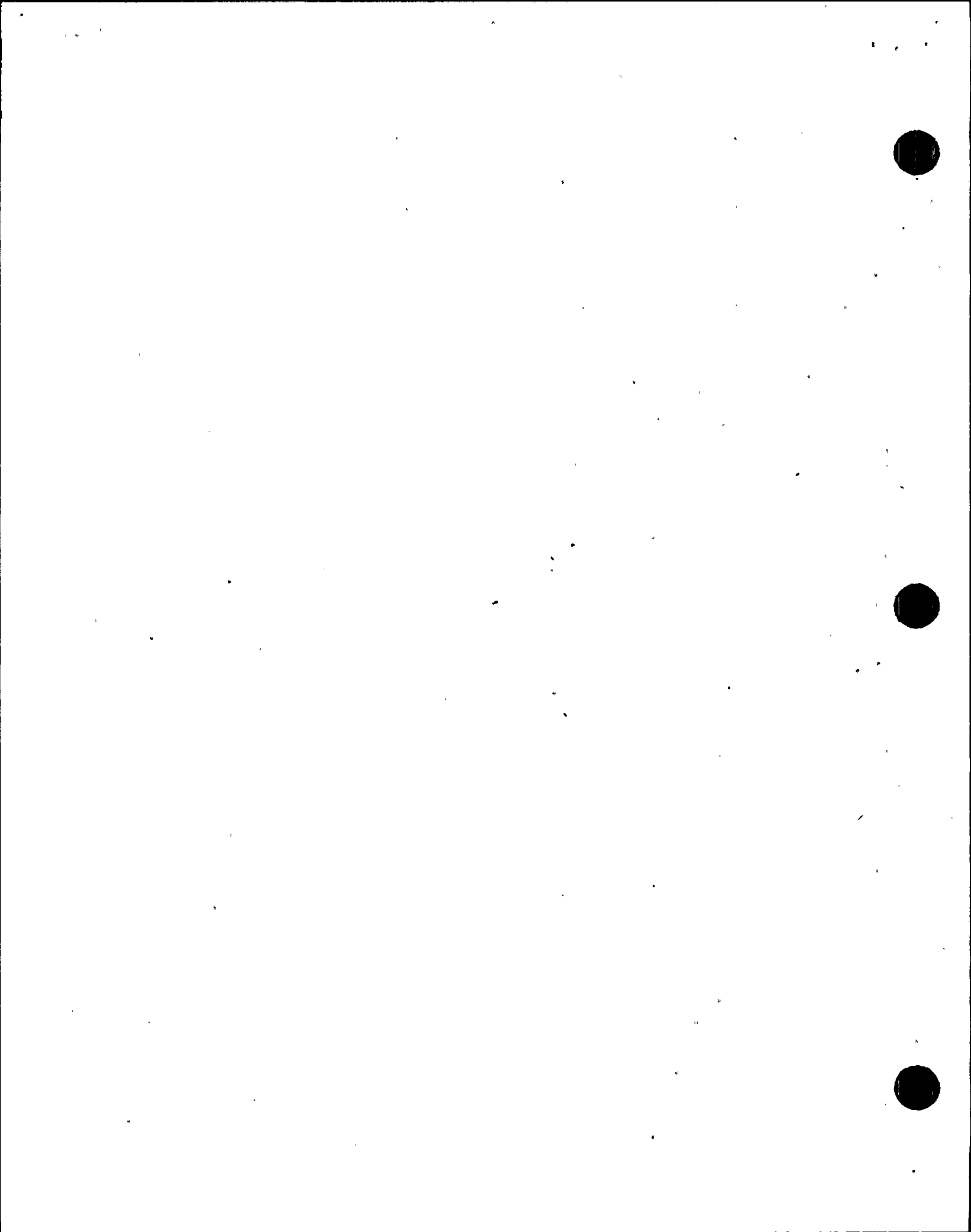
<u>Time (Sec)</u>	<u>Event</u>	<u>Value</u>
0.0	Loss of Power to all Four Reactor Coolant Pumps	--
0.9	Low Flow Trip	93% of 134.9×10^6 lbm/hr
1.55	Trip Breakers Open	
2.05	Shutdown, CEAs Begin to Drop into Core	--
2.64	Minimum W-3 DNBR	1.31
4.5	Maximum RCS Pressure, psia	2261

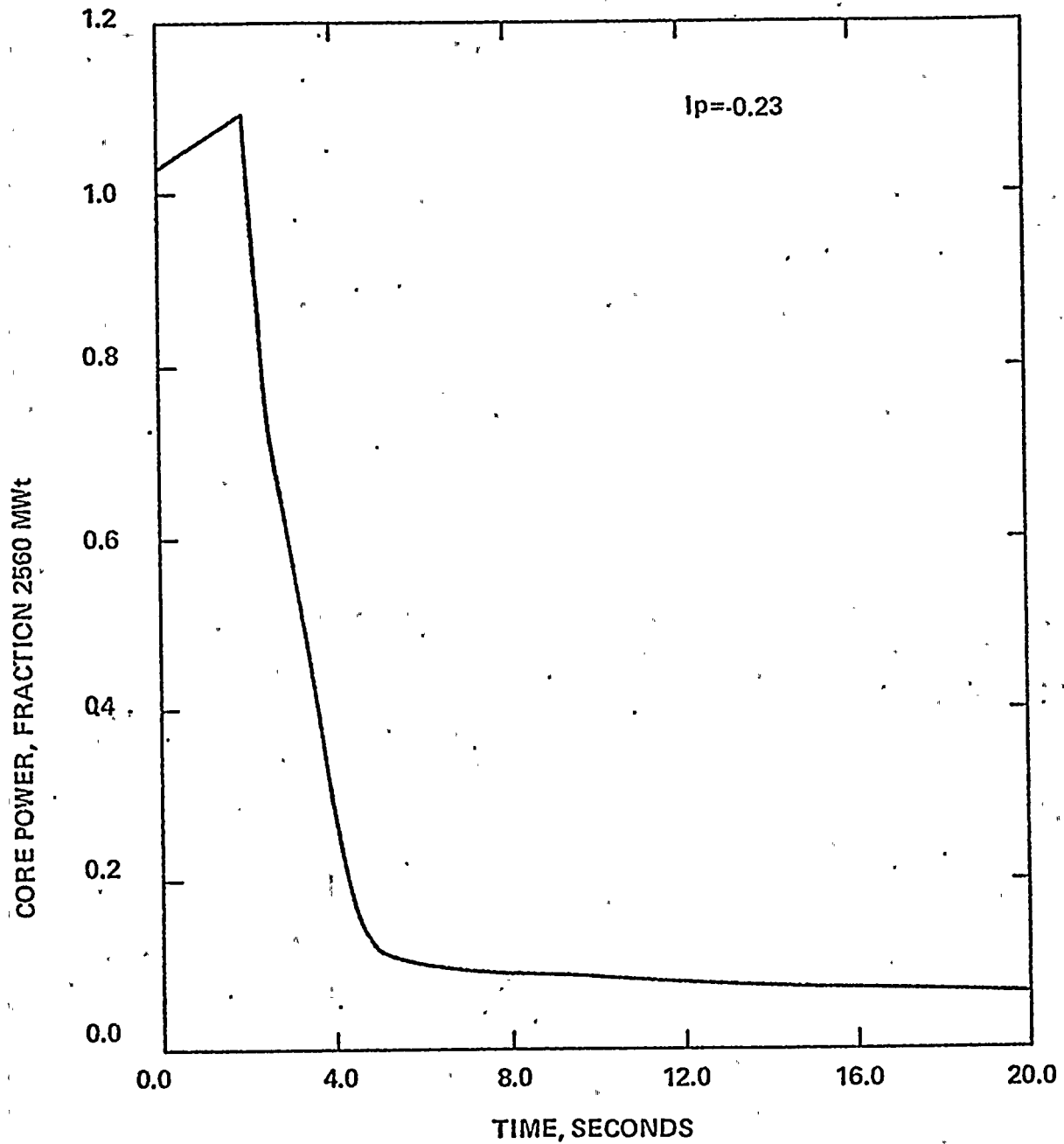


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LOSS OF COOLANT FLOW EVENT
CORE FLOW FRACTION VS TIME

Figure
7.3-1

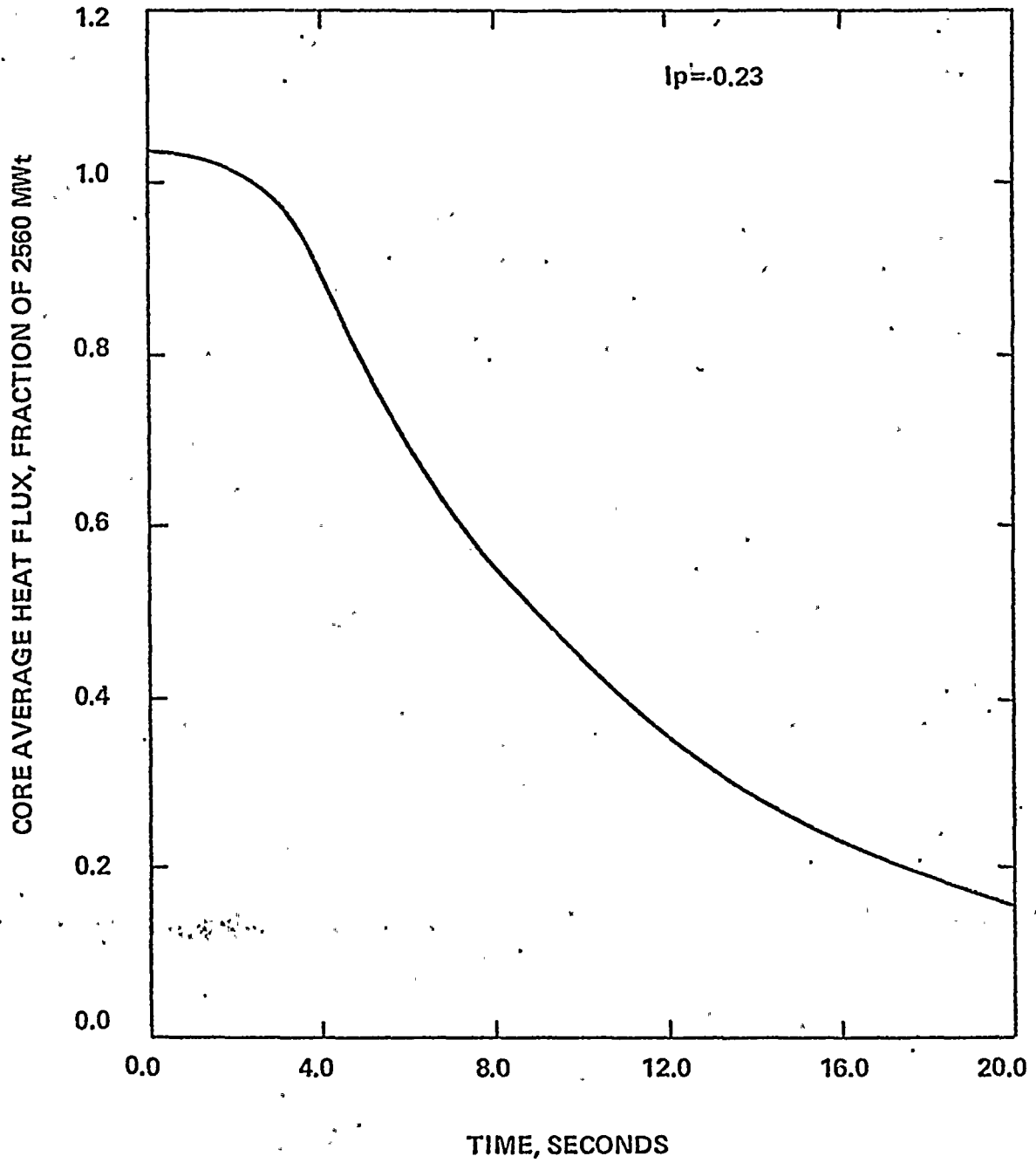




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LOSS OF COOLANT FLOW EVENT
CORE POWER vs TIME

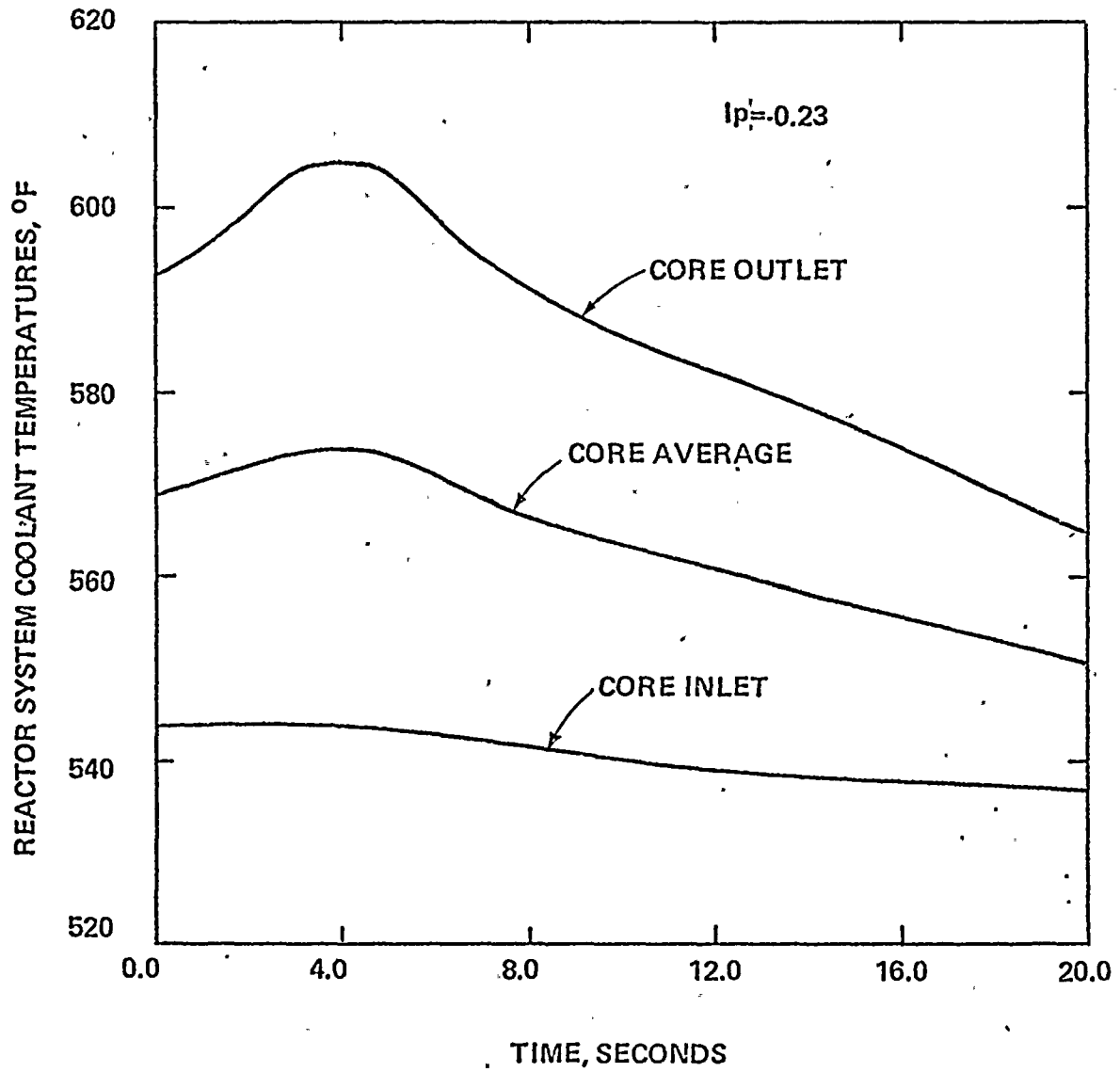
Figure
7.3-2



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LOSS OF COOLANT FLOW EVENT
CORE AVERAGE HEAT FLUX vs TIME

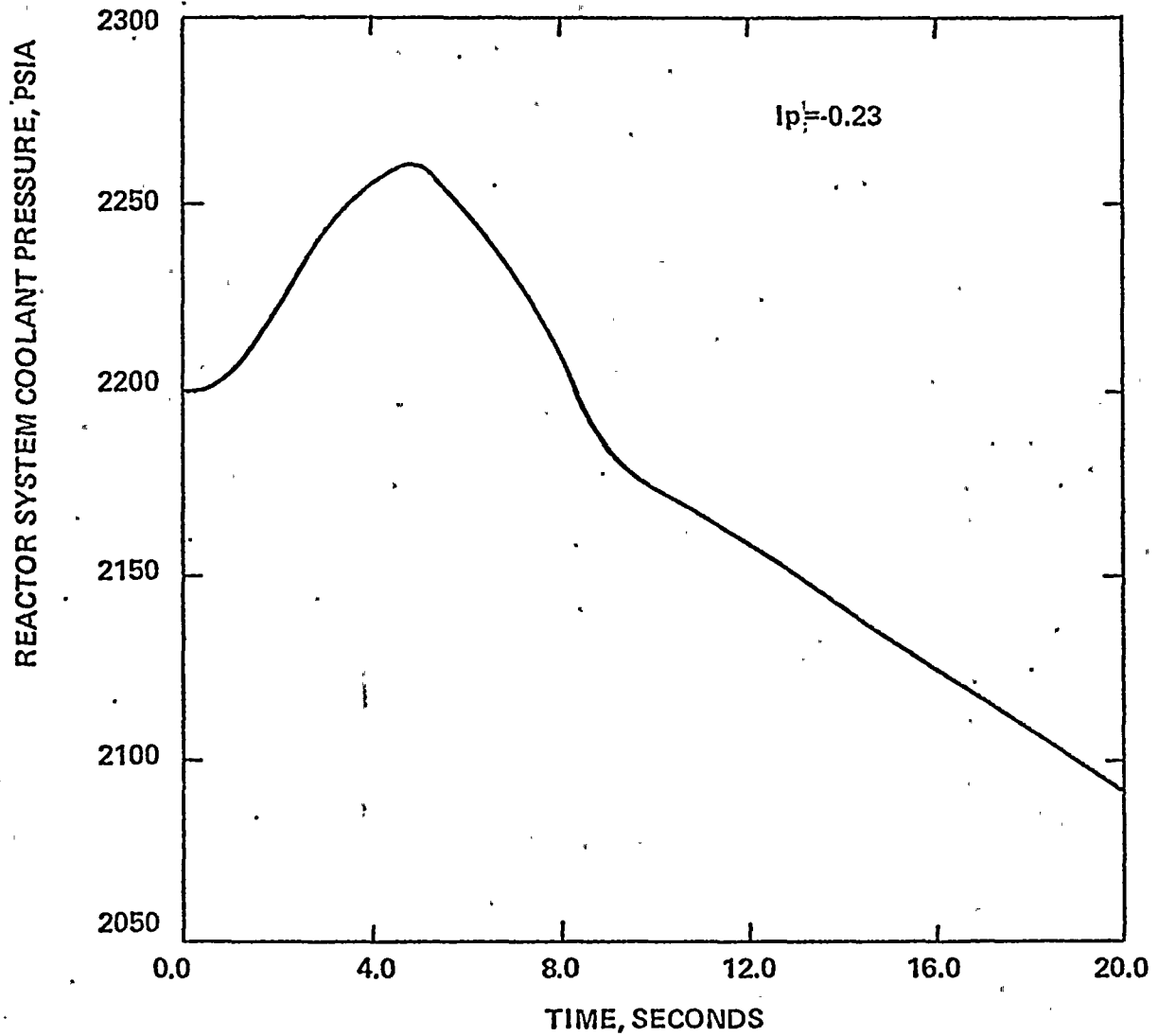
Figure
7.3-3



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LOSS OF COOLANT FLOW EVENT
REACTOR SYSTEM COOLANT TEMPERATURE VS TIME

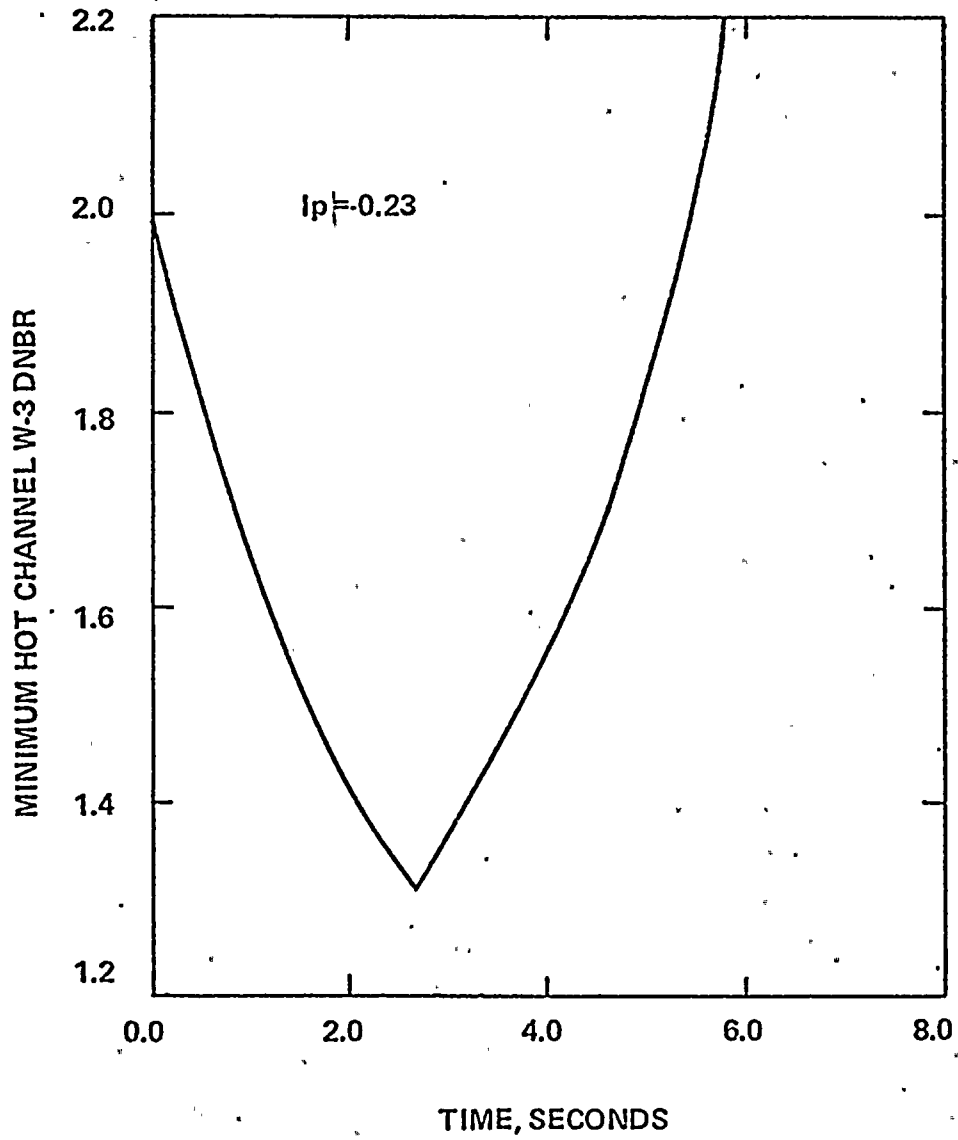
Figure
7.3-4



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LOSS OF COOLANT FLOW EVENT
REACTOR SYSTEM PRESSURE VS TIME

Figure
7.3-5



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LOSS OF COOLANT FLOW EVENT
MINIMUM HOT CHANNEL W-3 DNBR VS TIME

Figure
7.3-6

7.4 CEA EJECTION

The CEA Ejection event was reanalyzed for Cycle 3 because a more conservative CEA drop time to 90% insertion was assumed, and because of a more adverse pin census, and an increase in the azimuthal tilt allowance relative to Cycle 2. The purpose of this analysis is to demonstrate that the clad damage criterion is not exceeded for the zero power and the full power CEA ejection events for Cycle 3.

To bound the most adverse conditions during the cycle, the most limiting of either the Beginning of Cycle (BOC) or End of Cycle (EOC) value was used in the analysis. A BOC Doppler defect was used since it produces the least amount of negative reactivity feedback to mitigate the transient. A BOC moderator temperature coefficient of $+0.5 \times 10^{-4} \Delta\rho/^\circ\text{F}$ was used which results in positive reactivity feedback with increasing coolant temperatures. A EOC Beta Fraction was used in the analysis to produce the highest power rise during the event. For the full power and zero power case, the most negative shape index allowed by the DNBR monitoring band (-0.21 for full power, -0.40 for zero power) was used. This is consistent since the power shifts to the top of the core after the CEA ejects.

In the analysis, the CEA is assumed to be ejected in 0.05 seconds (the same as for Cycle 2). At zero power, the core is assumed to be operating at 1 Mwt for conservatism. The Variable Overpower trip is conservatively assumed to initiate at 25% (15% + 10% uncertainty) of 2754 Mwt.

The full power case was initiated at 2754 Mwt and is terminated by the high power trip.

The analytical methods used in the analysis of this event have been demonstrated to be conservative relative to the approved Combustion Engineering CEA Ejection Method which is presented in Reference 8 (CENPD-190A).

The reactivity-forced power transient was simulated by a digital computer program, CHIC-KIN (Reference 3), which simultaneously solves the one group neutron point kinetics equations together with the time and space dependent thermal and hydraulics equations for heat generation and transport within a single channel. The kinetics model incorporates the standard six-delay group representation along with explicit reactivity contributions from: (a) CEA motion, (b) Doppler effect, and (c) moderator density variations. By simulating the core average channel, the CHIC-KIN code computes the core average integrated energy output during the course of the transient.

In the CEA ejection event, the principal reactivity feedback mechanism affecting the power transient is the Doppler feedback. In the point kinetics approach, utilized in CHIC-KIN, a spatial Doppler weighting factor (K) accounts for the fact that the Doppler feedback effect is a function of the

spatial flux distribution. In order to represent the radial Doppler effect in a conservative manner, a space-time analysis was performed in which point kinetics calculations for various radial slices were compared with time-dependent, two-dimensional diffusion theory results obtained with a C-E modified version of the TWIGL code (Reference 4). The results of the space-time analysis have demonstrated that the use of the static (non-Doppler flattened) radial fuel rod peaking factor, as obtained from two-dimensional diffusion theory calculations, in conjunction with the average hot spot energy releases, yield energy increases that are conservatively large. Radial Doppler weighting factors obtained as a function of the ejected CEA worth are defined such that CHIC-KIN and TWIGL results give the same total core energy release.

The average energy rise in the hottest fuel pellet is obtained from the following relationship:

$$\Delta E_H = [(P/A)_H \times \Delta E_{Ave} \times K] - E_{HT} \quad (7.4-1)$$

Where ΔE_{Ave} is the average core energy rise obtained from CHIC-KIN; $(P/A)_H$ (the three-dimensional fuel rod peaking factor) is the ratio of the hot spot power density to the core average power density obtained from static, non-Doppler flattened diffusion theory calculations; K is the reduction factor defined above. For the zero power case, it is conservatively assumed that E_{HT} , which accounts for heat transferred out of the fuel rod during the transient, is zero.

The average energy in the hottest fuel pellet at the beginning of the transient is added to the net average energy rise in the hottest fuel pellet as obtained from Equation (7.4-1) to determine the total average enthalpy in the hottest fuel spot in the core. A similar procedure is used to compute the total centerline enthalpy in the hottest spot. The initial energy is obtained by correlating the initial local fuel temperature with an empirical temperature-enthalpy relationship (see Reference 8).

The spatial variation of the core local-to-average power ratio results from the convolution of the axial power distribution with radial pin power census distributions for the post-ejection condition, which are based on static core physics calculations. Combining these results with the total average and centerline enthalpies in the hottest fuel spot yields the fractional number of fuel rods with specific total average and centerline enthalpies. The calculated enthalpy values are compared to threshold enthalpy values to determine the amount of fuel experiencing the various degrees of fuel damage. These threshold enthalpy values are (References 5, 6, and 7):

Clad Damage Threshold:

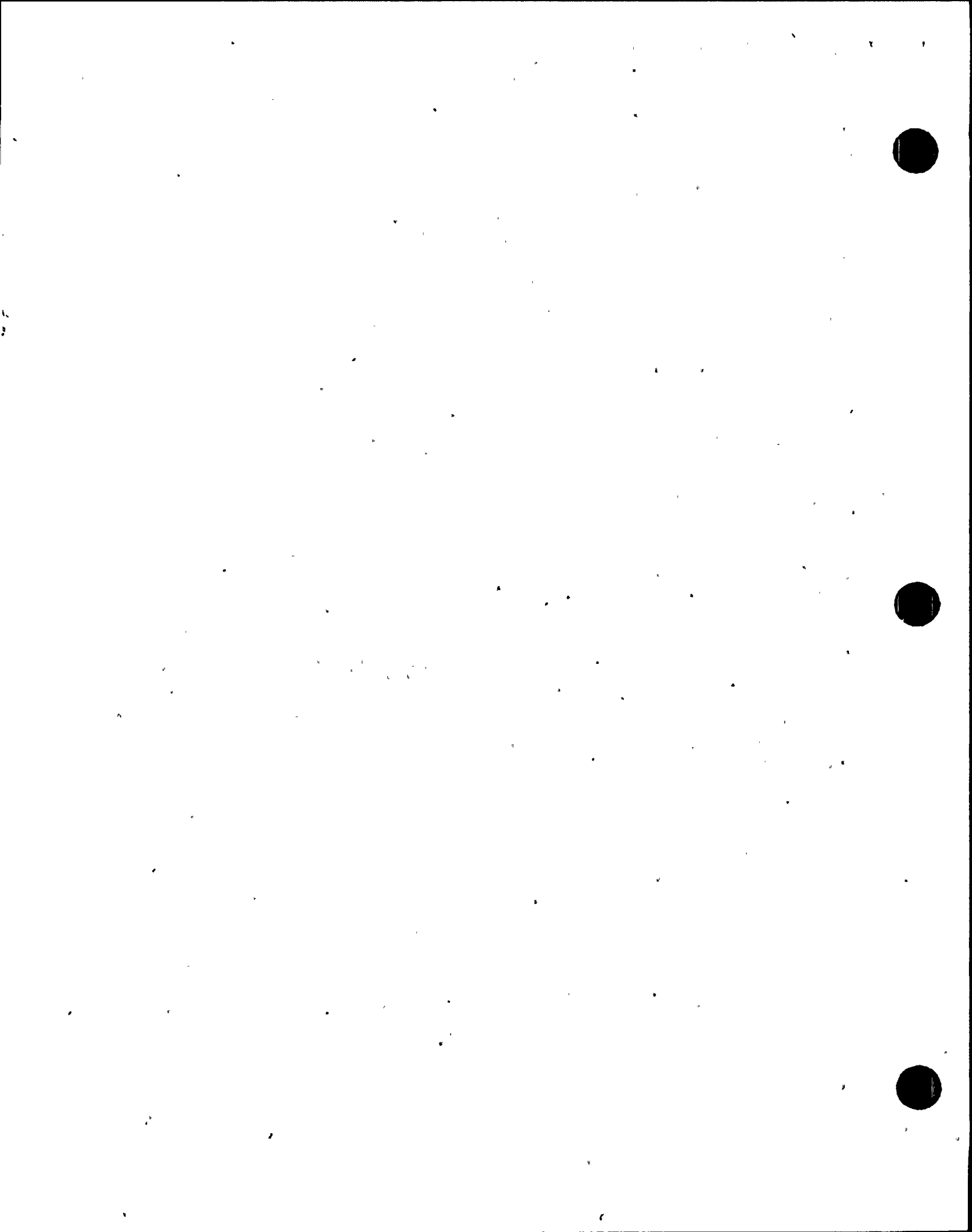
$$\text{Total Average Enthalpy} = 200 \text{ cal/gm}$$

Incipient Centerline Melting Threshold:

$$\text{Total Centerline Enthalpy} = 250 \text{ cal/gm}$$

Fully Molten Centerline Threshold:

$$\text{Total Centerline Enthalpy} = 310 \text{ cal/gm}$$



The criterion for determining the fraction of fuel rods that will release their radioactive fission products during a CEA ejection is the same as the one quoted above for determining clad damage. Thus, it is assumed that any fuel rod that exceeds a total average enthalpy of 200 cal/gm releases all of its gap activity. The gap activity corresponding to the hottest fuel rod during the core cycle is conservatively assumed for each rod that suffers clad damage.

The method described above is the same method that was used for Cycle 2, the reference cycle.

The key parameters for the full power and zero power CEA ejection events for Cycle 3 and Cycle 2 are listed in Table 7.4-1. All the ejected CEA worths and radial peaking factors include appropriate allowances for calculation uncertainties. The parameters shown in Table 7.4-1 produce the most limiting transients for both zero power and full power CEA ejection events. An augmentation factor of 1.071 was used instead of the 1.058, shown in Section 5.0, to bound later cycles. The higher augmentation factor is conservative.

The power transients produced by a CEA ejection initiated at full power is shown in Figure 7.4-1, and at zero power is shown at 7.4-2. In both cases, the power transient is terminated by the Doppler feedback and the event is terminated by the CEAs dropping into the core.

The results of the full power and zero power CEA ejections are shown in Table 7.4-2. For the zero power case, the total energy deposited during the event has increased due to an increase in the key parameters mentioned previously.

For the full power case, the total energy deposited has decreased for Cycle 3 in comparison to Cycle 2. This is due to the fact that the decrease in the post-ejected radial peaking factor offsets the more conservative CEA drop time and slightly larger azimuthal tilt allowance. The augmentation factor has only minor impact on the results. Although the total energy deposited has decreased, the CEA ejection pin census data has become more adverse. Thus, there are more pins in the threshold range. The number of fuel pins that experience incipient centerline melting has increased (from 1.3% to 2.8%) for Cycle 3.

For the zero power case, the total energy deposited has increased for Cycle 3, in comparison to Cycle 2, because of the more conservative CEA drop time. All other key input parameters for this event are bounded by Cycle 2 values as shown in Table 7.4-1.

Since the total energy deposited for both zero power and full power CEA ejection events is less than the criterion for clad damage (i.e., 200 cal/gm), no fuel pins will fail.

TABLE 7.4-1

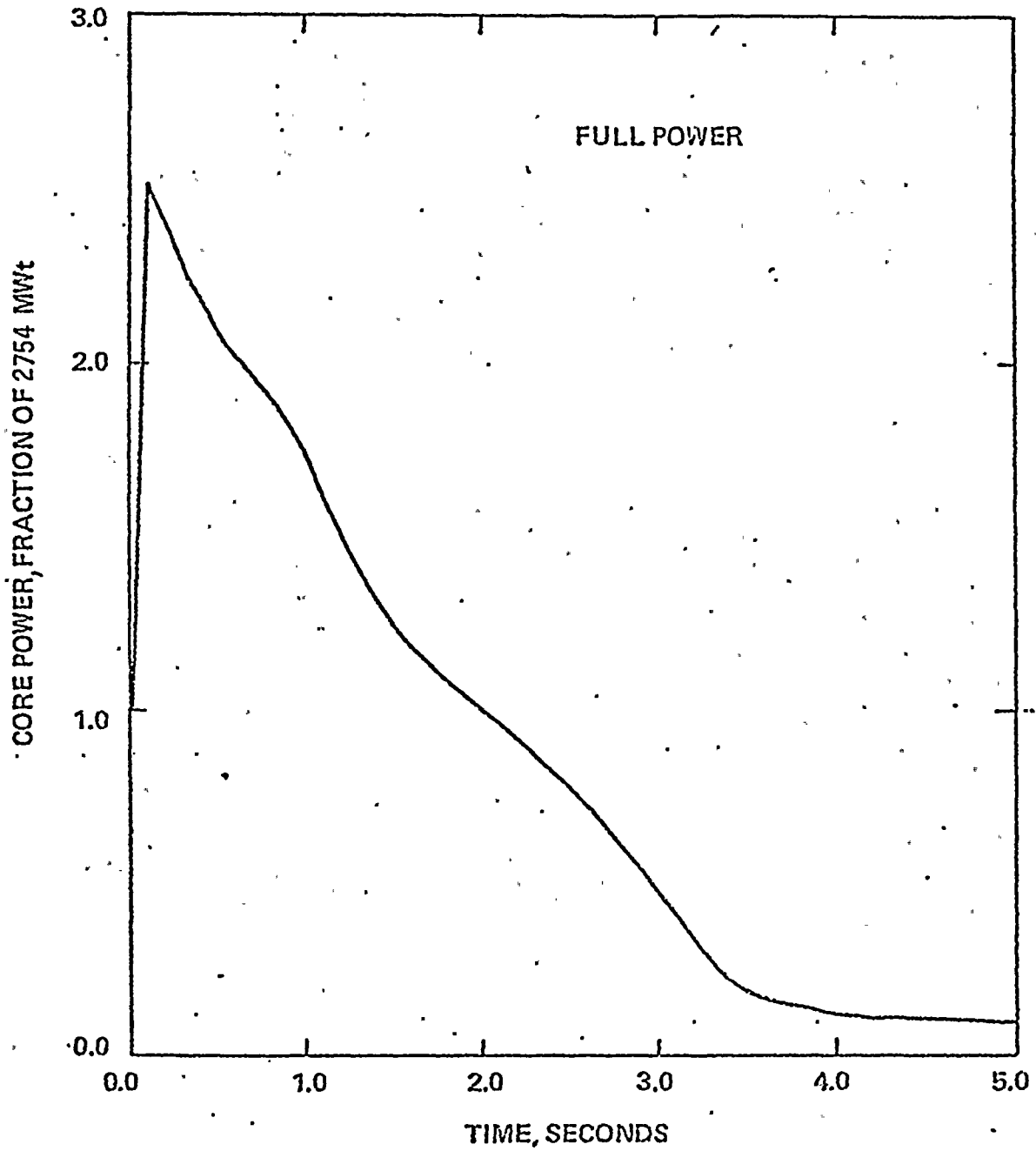
KEY PARAMETERS ASSUMED IN THE CEA EJECTION ANALYSES

<u>Parameter</u>	<u>Units</u>	<u>Reference Cycle 2</u>	<u>Cycle 3</u>
<u>Full Power</u>			
Core Power Level	MWt	2754	2754
Core Average Linear Heat Rate of Fuel Rod	kw/ft	6.41	6.29
Moderator Temperature Coefficient	$10^{-4} \Delta\rho/^{\circ}\text{F}$	+5	+5
Ejected CEA Worth	% $\Delta\rho$.29	.29
Delayed Neutron Fraction, β		.0047	.0047
Post-Ejected Radial Power Peak		3.9	3.6
Axial Power Peak		1.39	1.39
CEA Bank Worth at Trip	% $\Delta\rho$	-3.0	-3.0
Augmentation Factor		1.062	1.071
K-Factor		.92	.92
Tilt Allowance		1.02	1.03
<u>Zero Power</u>			
Core Power Level	MWt	1.0	1.0
K-Factor		.88	.88
Ejected CEA Worth	% $\Delta\rho$.65	.65
Post-Ejected Radial Power Peak		8.34	8.34
Axial Power Peak		1.59	1.59
CEA Bank Worth at Trip	% $\Delta\rho$	-1.47	-1.47
Tilt Allowance		1.10	1.10
CEA Drop Time		3.0	3.1

TABLE 7.4-2

CEA EJECTION ACCIDENT RESULTS

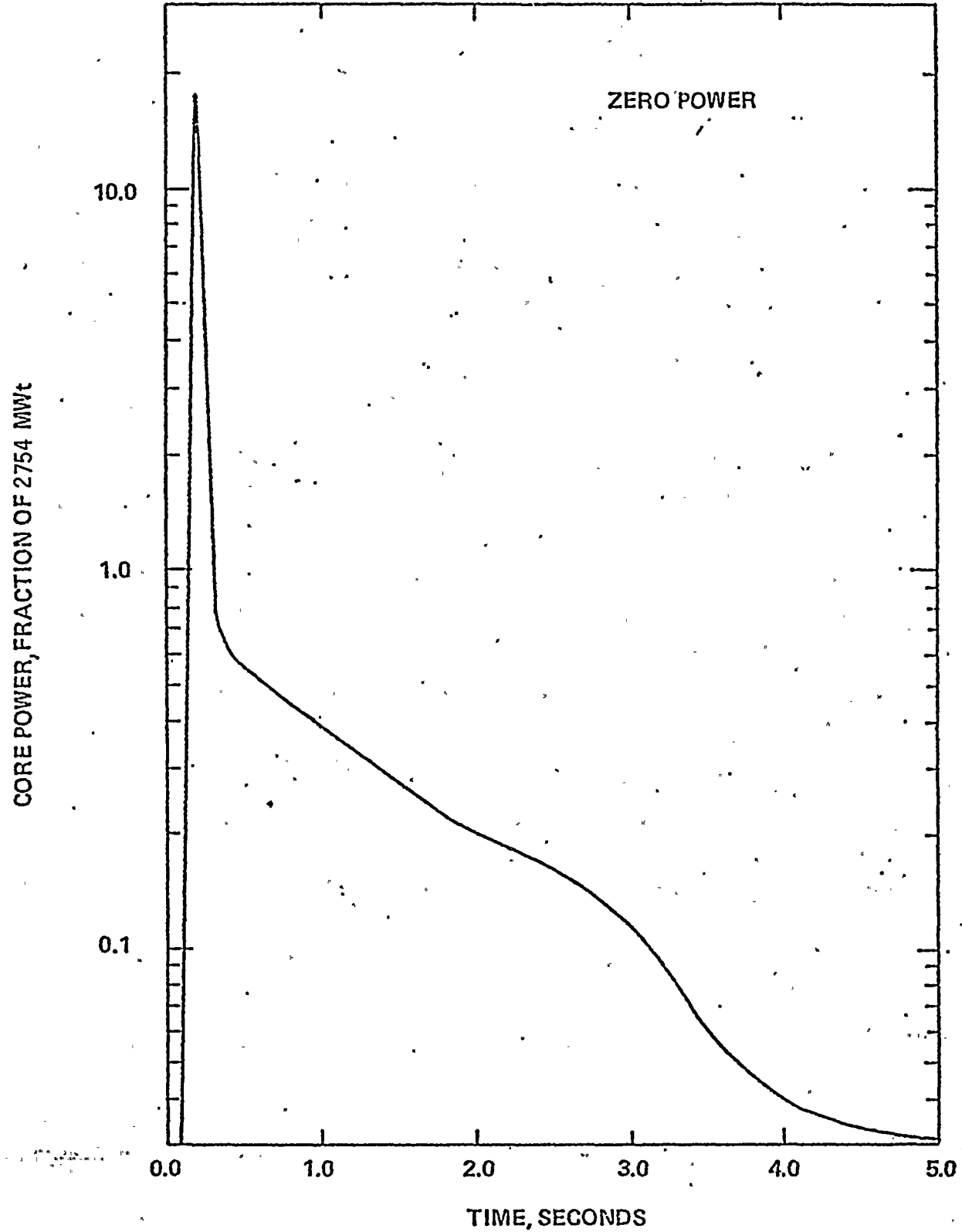
	<u>Reference Cycle Cycle 2</u>	<u>Cycle 3</u>
<u>Full Power</u>		
Total Average Enthalpy of Hottest Fuel Pellet (cal/gm)	198.0	194.0
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gm)	293.0	289.0
Fraction of Rods that Suffer Clad Damage (Average Enthalpy \geq 200 cal/gm)	0	0
Fraction of Fuel Having at Least Incipient Centerline Melting (Centerline Enthalpy \geq 250 cal/gm)	.013	.028
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy \geq 310 cal/gm)	0	0
	<u>Reference Cycle Cycle 2</u>	<u>Cycle 3</u>
<u>Zero Power</u>		
Total Average Enthalpy of Hottest Fuel Pellet (cal/gm)	177.7	186.0
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gm)	198.0	209.5
Fraction of Rods that Suffer Clad Damage (Average Enthalpy \geq 200 cal/gm)	0	0
Fraction of Fuel Having at Least Incipient Centerline Melting (Centerline Enthalpy \geq 250 cal/gm)	0	0
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy \geq 310 cal/gm)	0	0



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CEA EJECTION EVENT
CORE POWER VS TIME

Figure
7.4-1



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CEA EJECTION EVENT
CORE POWER vs TIME

Figure
7.4-2

7.5 SEIZED ROTOR EVENT

The Seized Rotor event was reanalyzed due to a more adverse pin census and an increase in radial peaking factors from Cycle 2 to Cycle 3. Hence, a reanalysis was performed for Cycle 3 to ensure that only a small fraction of fuel pins are predicted to fail during this event.

The Seized Rotor event is initiated by the assumed complete seizure of a pump shaft in one of the reactor coolant pumps. This causes the core coolant flow to decrease rapidly to the three pump flow. This is conservatively assumed to happen instantaneously. For additional conservatism, no credit is taken for the decrease in core heat flux, the increase in RCS pressure, or the decrease in core inlet temperature at the time of minimum DNBR.

The methodology used in determining the amount of fuel damage is the same as for Cycle 2. These methods are discussed in detail in Reference 2. A conservatively "flat" pin census distribution (a histogram of the number of pins with radial peaks in intervals of .01 in radial peak normalized to the maximum peak) was used to determine the number of pins that experience DNB.

A comparison of the key parameters assumed in this analysis for Cycle 3 (and for the reference cycle, Cycle 2) are presented in Table 7.5-1. The Seized Rotor event was initiated at the LCOs to maximize the number of predicted fuel pin failures for Cycle 3.

The NSSS and RPS response for Seized Rotor event initiated at an $I_p = -0.23$ is shown in Table 7.5-2. Figures 7.5-1 through 7.5-4 present the time dependent NSSS parameters at an $I_p = -0.23$ for core power, core heat flux, RCS pressure and coolant temperatures. An $I_p = -0.23$ is conservative with respect to the most negative shape index ($I_p = -0.21$) allowed by the LCOs.

The Seized Rotor analysis for Cycle 3 yields a minimum W-3 DNBR of 1.0 and the predicted number of pins that experience fuel failure is 0.99% in comparison to 0.7% in Cycle 2.

For the case of the loss of coolant flow resulting from a seizure of a reactor coolant pump shaft, a trip on low coolant flow is initiated to limit the predicted fuel pin failure to only a small fraction of the total number of fuel pins. Based on the low probability of this event, the small number of predicted fuel pin failures is acceptable.

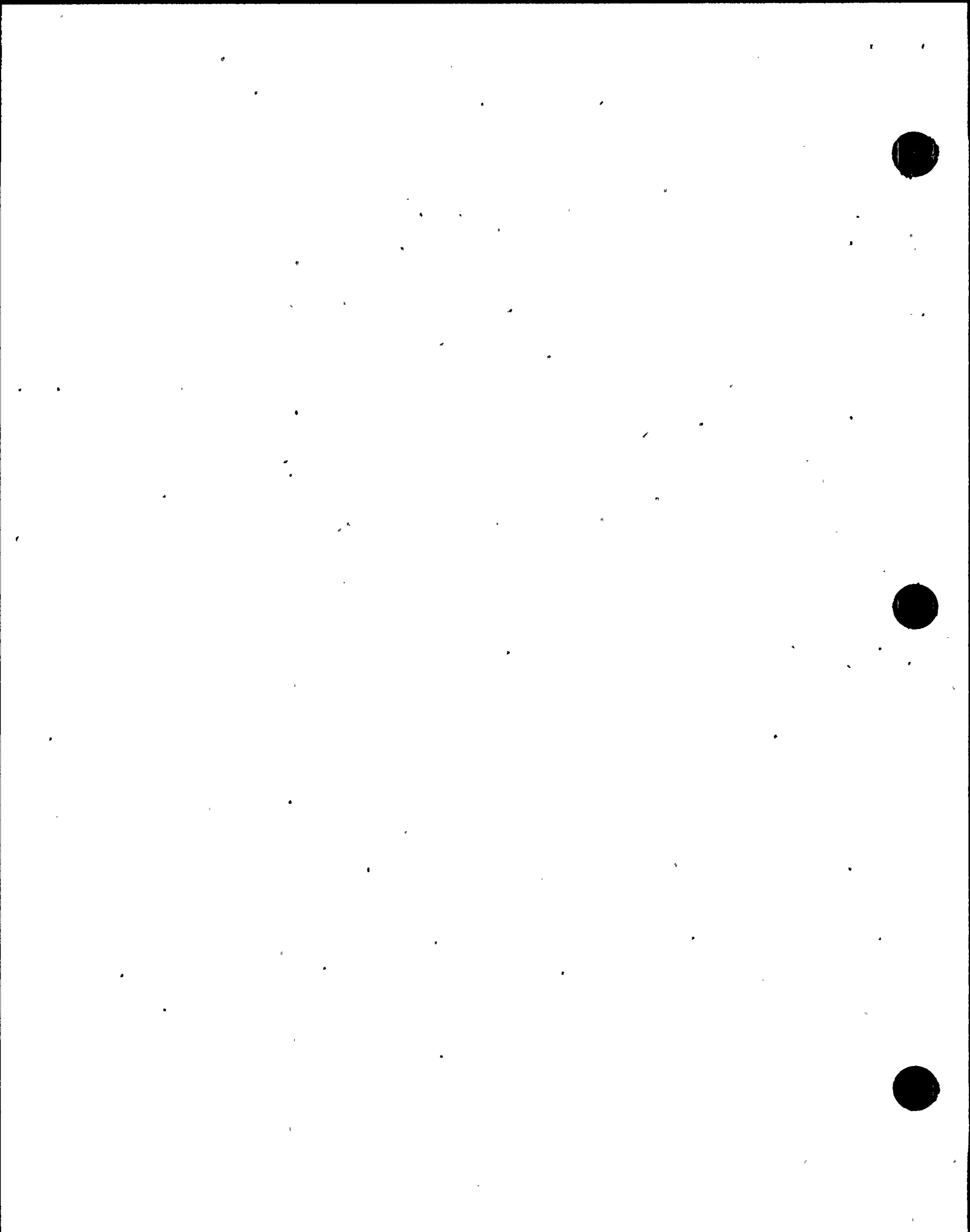


TABLE 7.5-1

ASSUMPTIONS FOR SEIZED ROTOR EVENT

<u>Parameter</u>	<u>Units</u>	<u>Unit 1, Cycle 2</u>	<u>Unit 1, Cycle 3</u>
Initial Core Power Level	MWt	2754	2754
Core Inlet Coolant Temperature	$^{\circ}\text{F}$	544	544
Core Mass Flow Rate	10^6 lbm/hr	134.9	134.9
Three Pump Core Mass Flow Rate	10^6 lbm/hr	104.1	104.1
Reactor Coolant System Pressure	psia	2200	2200
Moderator Temperature Coefficient	$10^{-4} \Delta\rho/^{\circ}\text{F}$	+ .5	+ .5
Doppler Coefficient Multiplier	--	.85	.85
CEA Worth at Trip	$10^{-2} \Delta\rho$	-5.41	-5.41
Local Unrodded Radial Peaking Factor, F_r		1.58	1.64

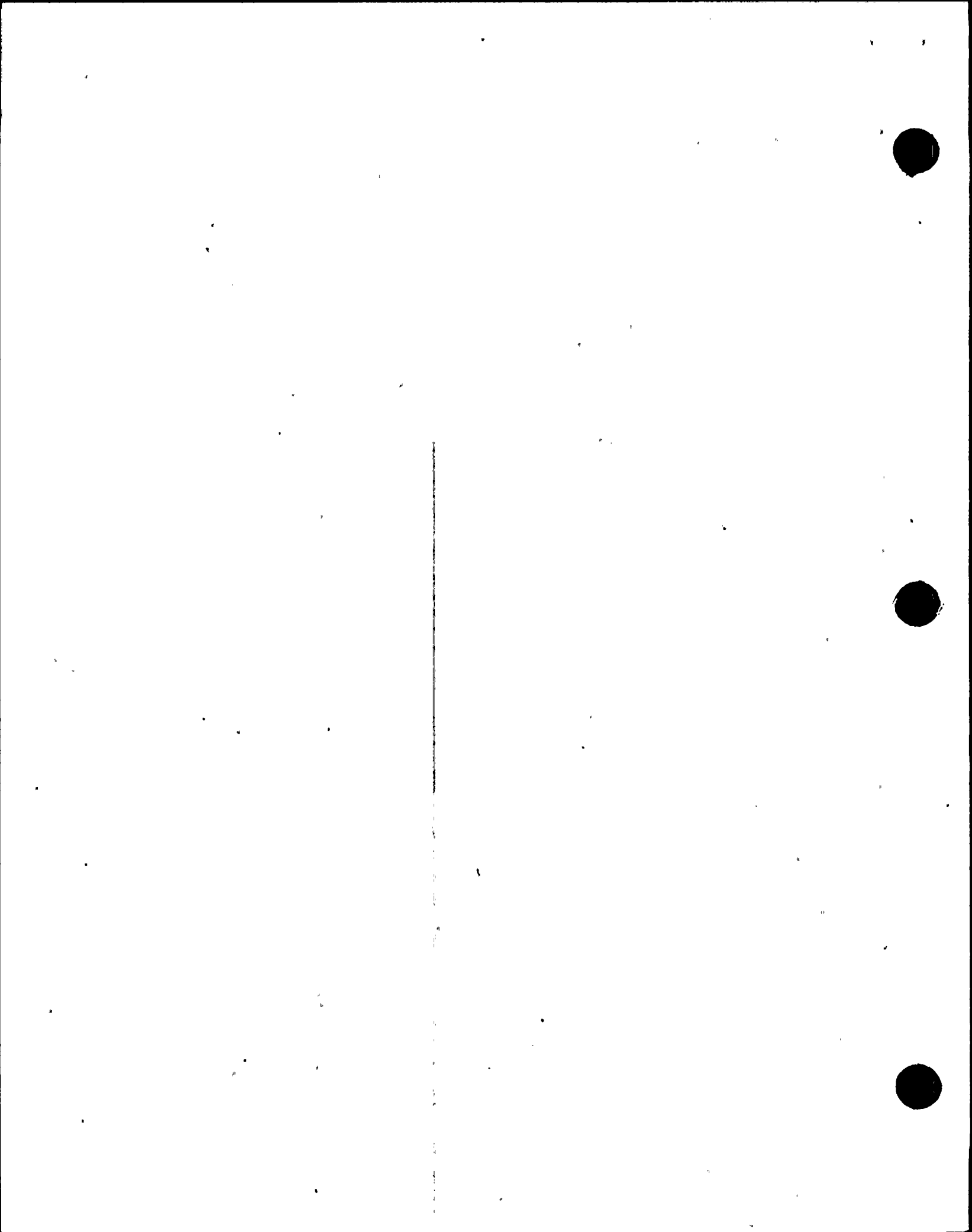
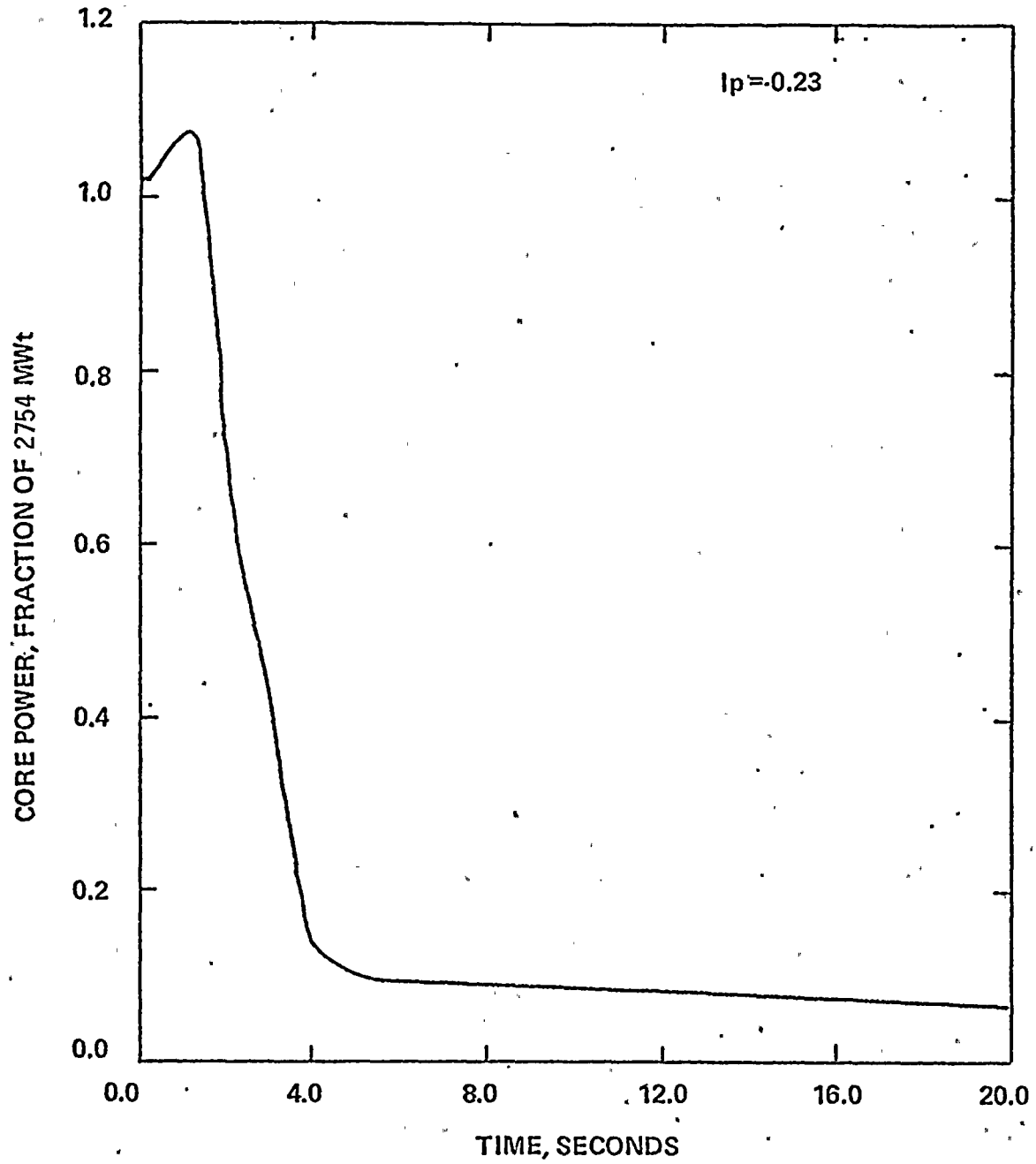


TABLE 7.5-2

SEIZED ROTOR SEQUENCE OF EVENTS

<u>Time (Sec)</u>	<u>Event</u>
0.0	Seizure of One Reactor Coolant Pump
0.0	Low Flow Trip
0.65	Trip Breakers Open
1.15	Shutdown CEAs Begin to Drop Into Core
3.3	Maximum RCS Pressure (2276)

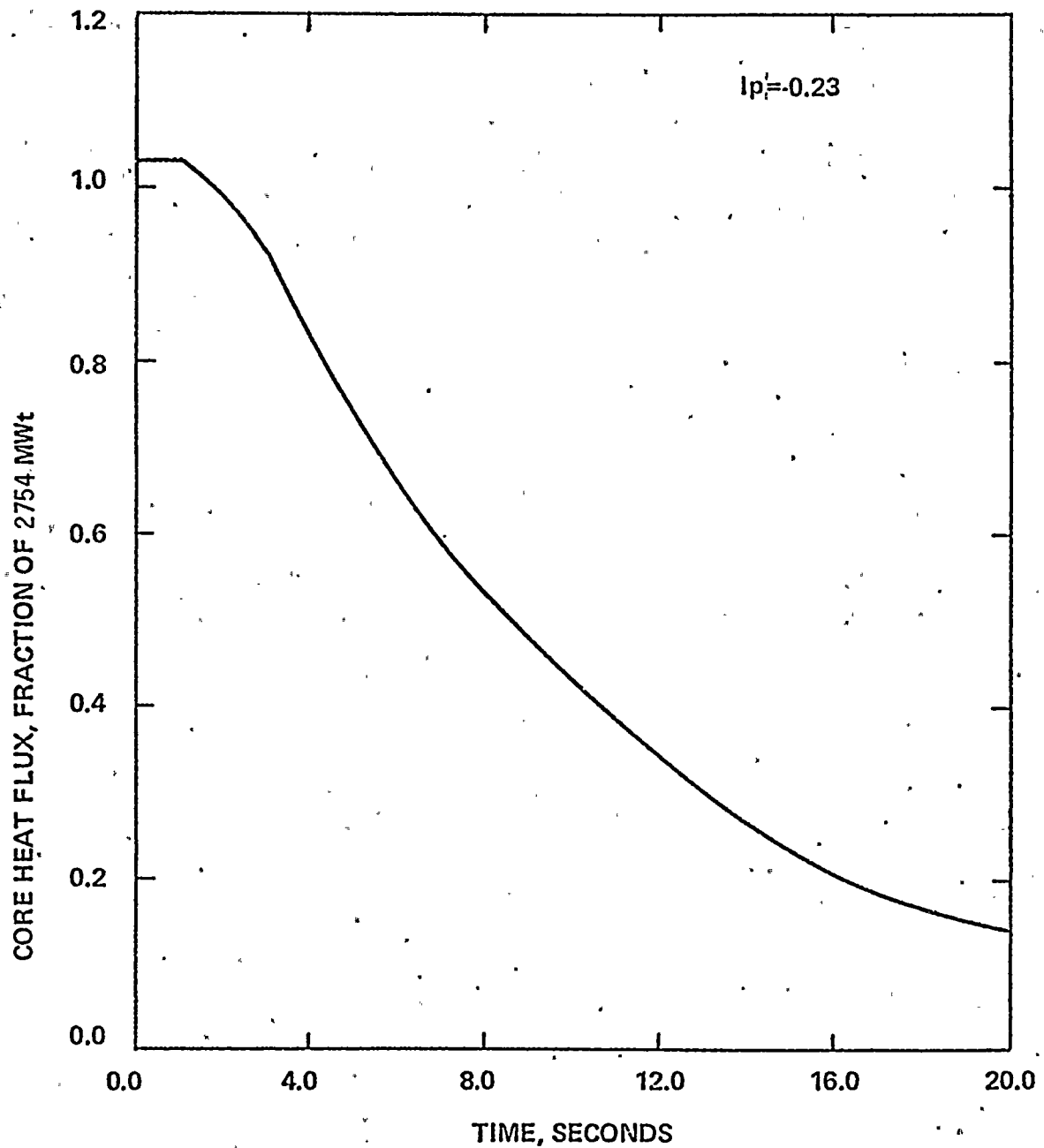


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SEIZED ROTOR EVENT
CORE POWER VS TIME

Figure
7.5-1

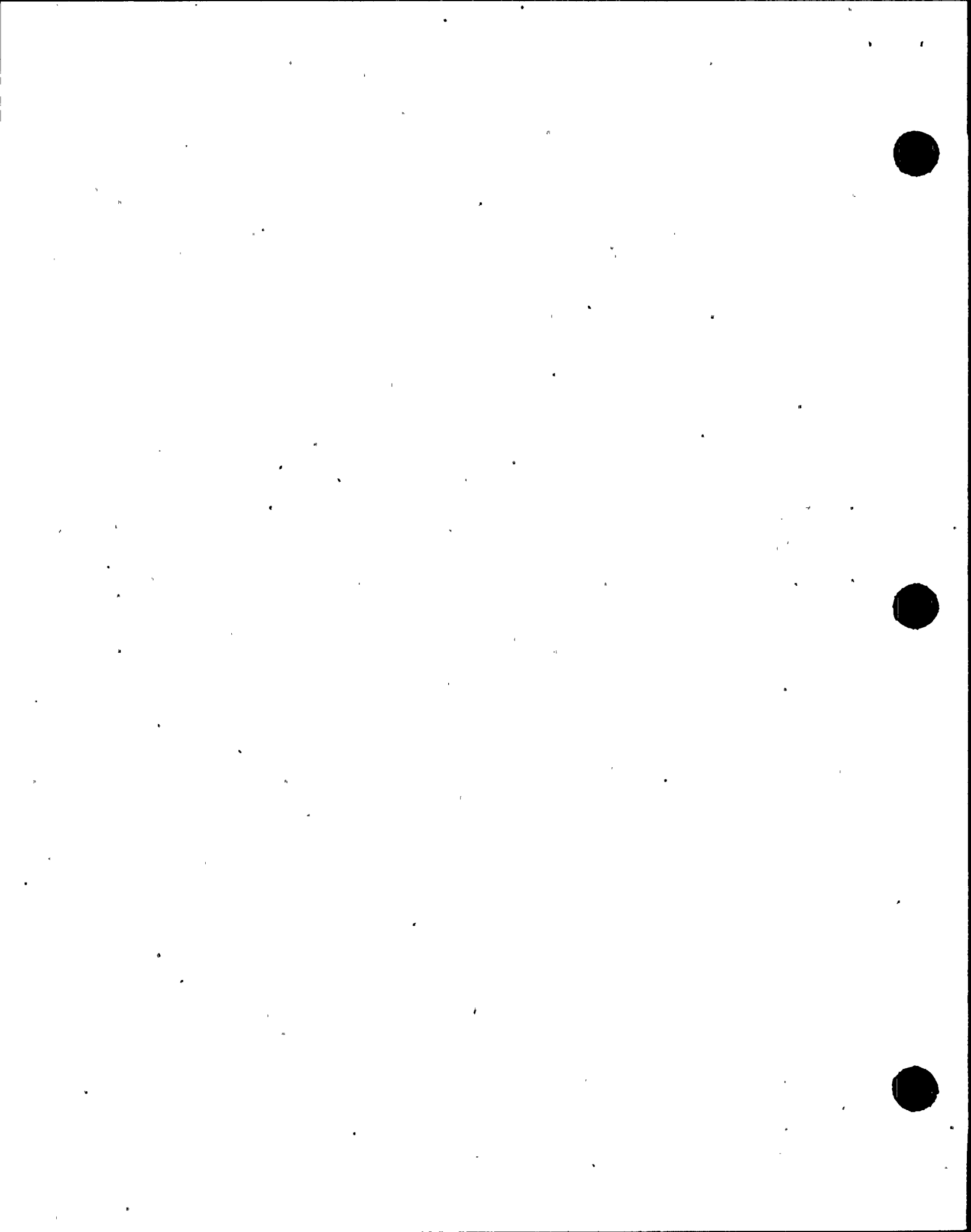


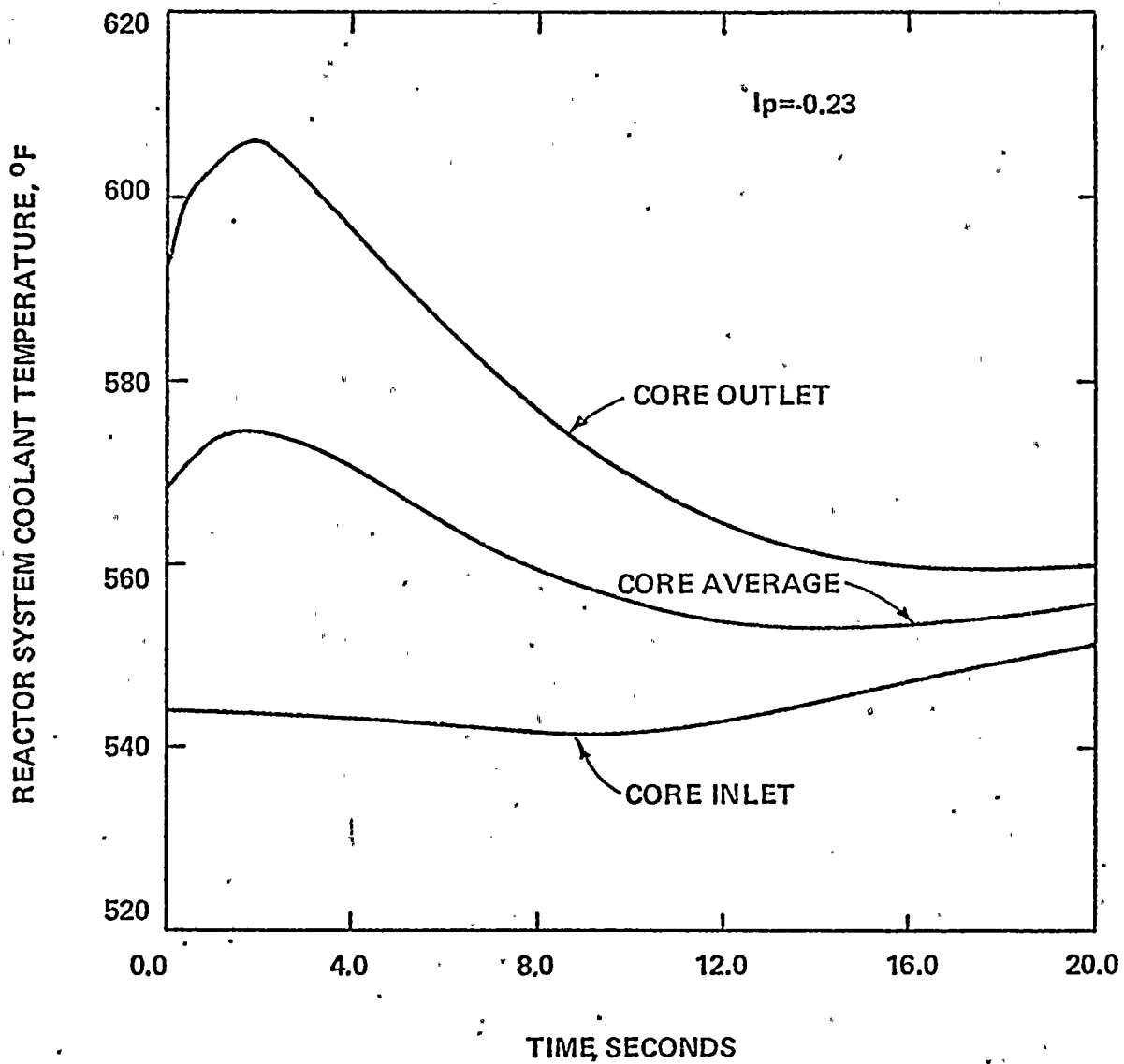


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SEIZED ROTOR EVENT
CORE AVERAGE HEAT FLUX VS TIME

Figure
7.5-2

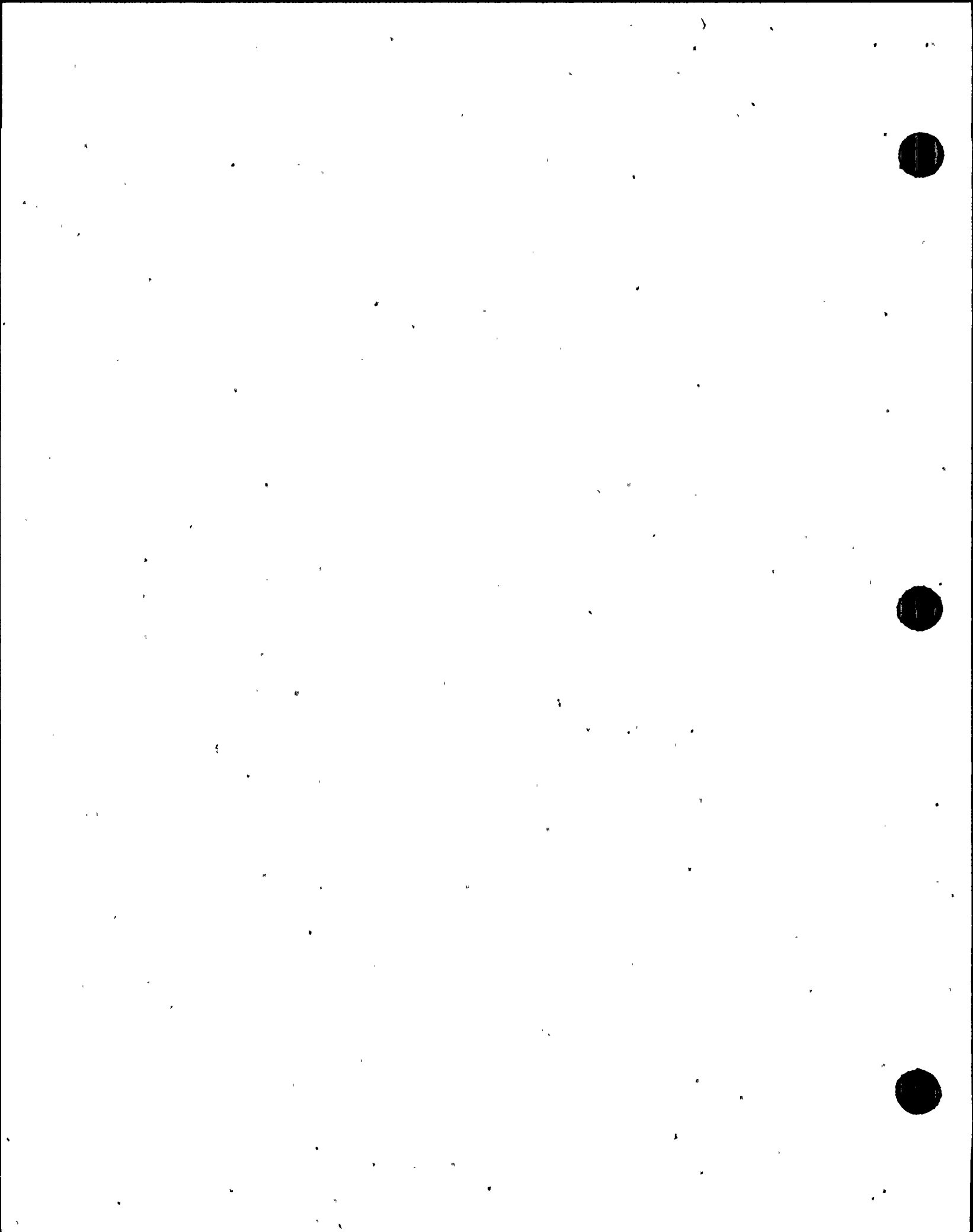


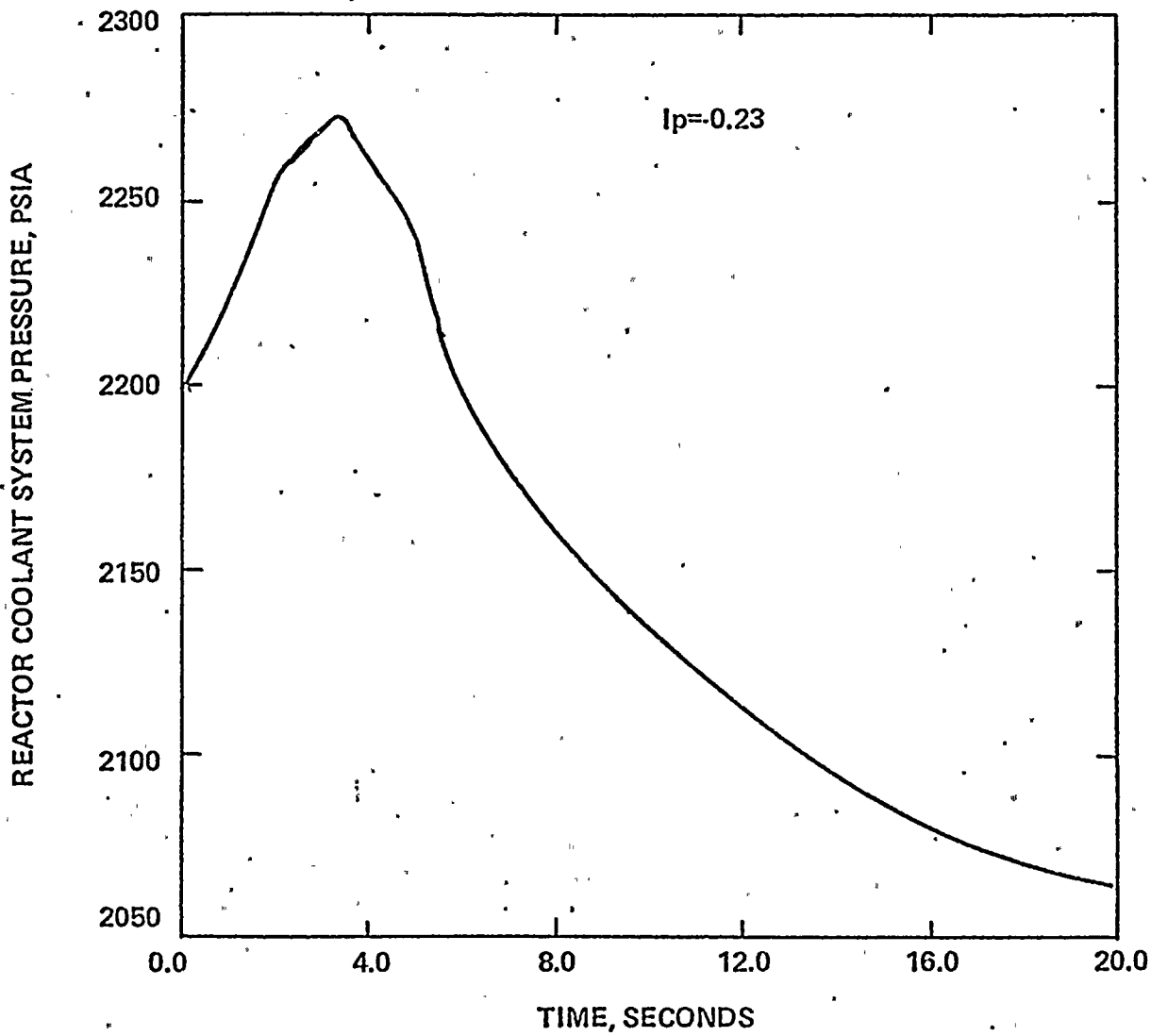


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SEIZED ROTOR EVENT
REACTOR SYSTEM COOLANT TEMPERATURE vs TIME

Figure
7.5-3

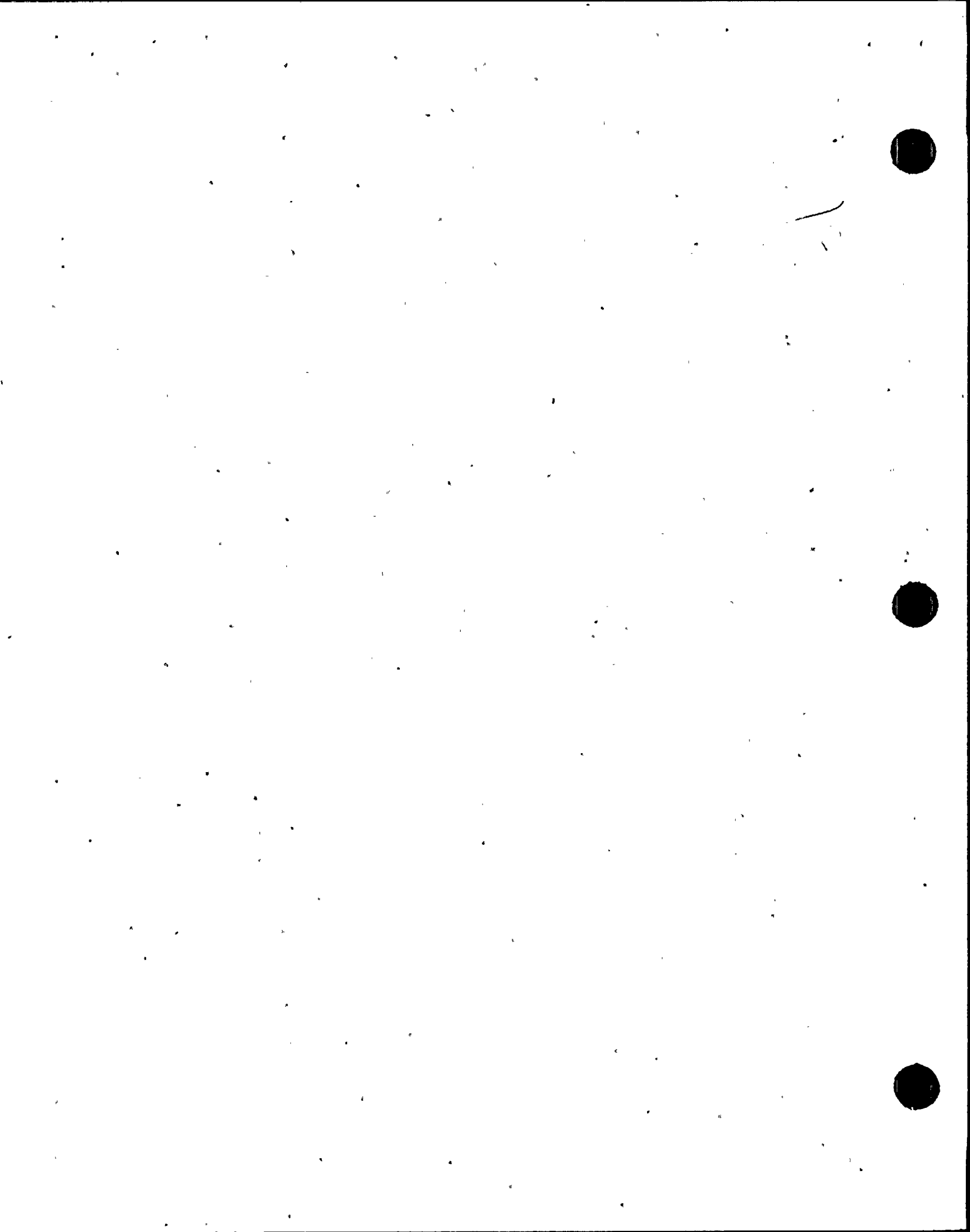




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SEIZED ROTOR EVENT
REACTOR SYSTEM PRESSURE VS TIME

Figure
7.5-4



REFERENCES FOR SECTION 7.0

- 1) "C-E Setpoint Methodology," CENPD-199, Topical Report, April, 1976.
- 2) "C-E Methods for Loss of Flow Analysis," CENPD-183, Topical Report, July, 1975.
- 3) WAPD-TM-479, J. A. Redfield, "CHIC-KIN - A Fortran Program for Intermediate and Fast Transients in a Water Moderator Reactor," January, 1965.
- 4) WAPD-TM-743, J. B. Yasinsky, M. Natelson, and L. A. Hageman, "TWIGL - A Program to Solve the Two-Dimensional, Two Group, Space-Time Neutron Diffusion Equations with Temperature Feedback," February, 1968.
- 5) GEMP-482, H. C. Brassfield, et. al., "Recommended Property and Reactor Kinetics Data for use in Evaluating a Light Water-Cooled Reactor Loss-of-Coolant Incident Involving Zircaloy-4 or 304-SS Clad UO_2 ," April, 1968.
- 6) Idaho Nuclear Corporation, Monthly Report, Ny-123-69, October, 1969.
- 7) Idaho Nuclear Corporation, Monthly Report, Hai-127-70, March, 1970.
- 8) CENPD-190A, "CEA Ejection, C-E Method for Control Element Assembly Ejection," July, 1976.
- 9) Letter, Robert E. Uhrig (FP&L) to Victor Stello (NRC), dated March 22, 1978, "St. Lucie Unit 1 Docket No. 50-335 Proposed Amendment to Facility Operating License DPR-67"

ATTACHMENT B

Re: St. Lucie Unit 1
Docket No. 50-335
Cycle 3 Operation

PROPOSED TECHNICAL SPECIFICATIONS

DEFINITIONS

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.27 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.28 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.29 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

LOAD FOLLOW OPERATION

1.30 LOAD FOLLOW OPERATION is repeated daily power level changes of more than 10% rated thermal power or daily insertion of control rods beyond the long term insertion limits.



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TABLE 2.2-1

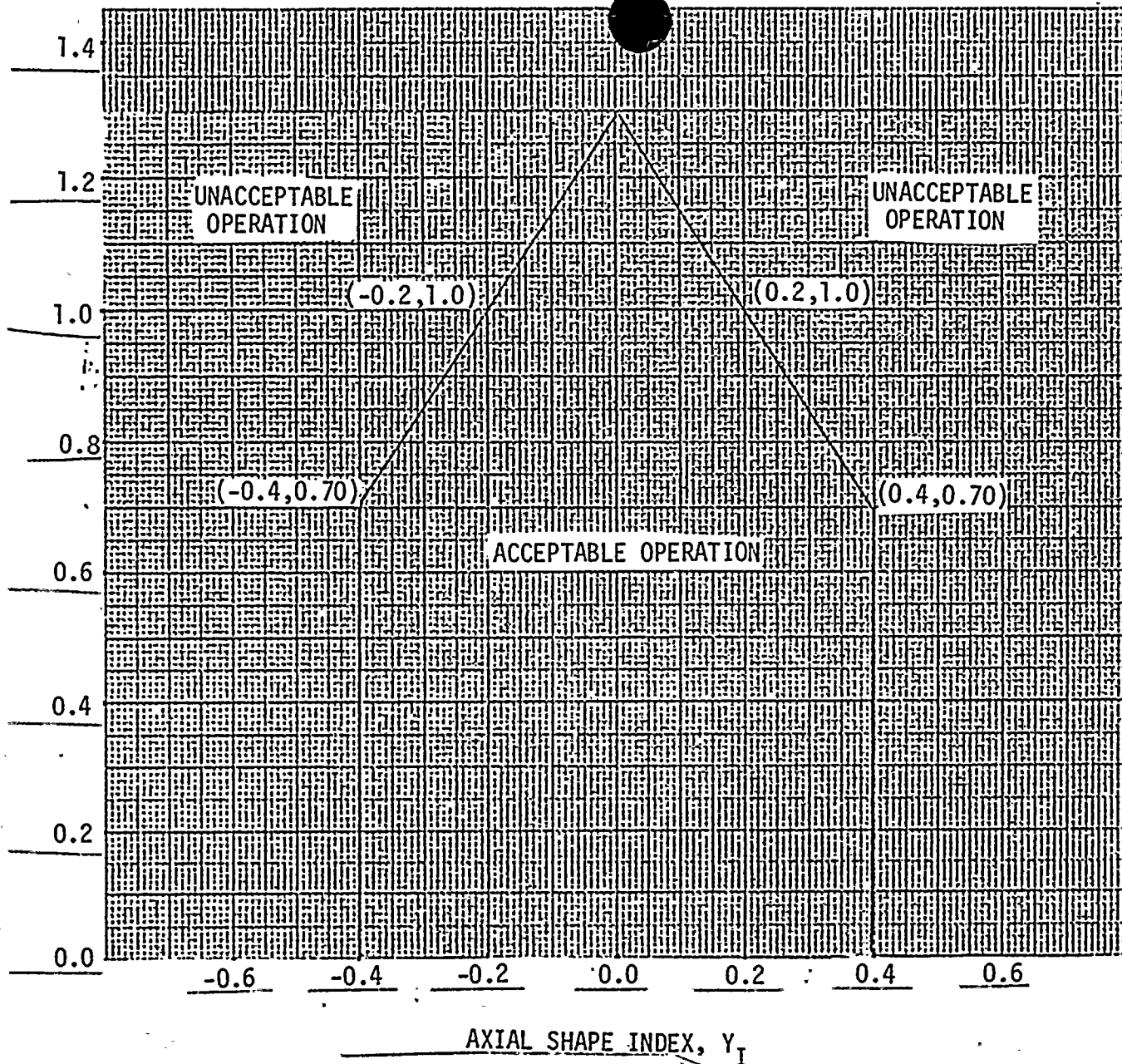
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level - High (1) Four Reactor Coolant Pumps Operating	\leq 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of \leq 107.0% of RATED THERMAL POWER.	\leq 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of \leq 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1) Four Reactor Coolant Pumps Operating	$>$ 95% of design reactor coolant flow with 4 pumps operating*	$>$ 95% of design reactor coolant flow with 4 pumps operating*
4. Pressurizer Pressure - High	\leq 2400 psia	\leq 2400 psia
5. Containment Pressure - High	\leq 3.3 psig	\leq 3.3 psig
6. Steam Generator Pressure - Low (2)	\geq 500 psia	\geq 500 psia
7. Steam Generator Water Level -Low	\geq 37.0% Water Level - each steam generator	\geq 37.0% Water Level - each steam generator
8. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

*Design reactor coolant flow with 4 pumps operating is 370,000 gpm.

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QR₂

AXIAL SHAPE INDEX, Y_I

FIGURE 2.2-2
Local Power Density-High Trip Setpoint
Part 2 (QR₂ versus Y_I)

REACTIVITY CONTROL SYSTEMS

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be ≤ 3.1 seconds from when electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

- a. $T_{avg} \geq 515^{\circ}\text{F}$, and
- b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

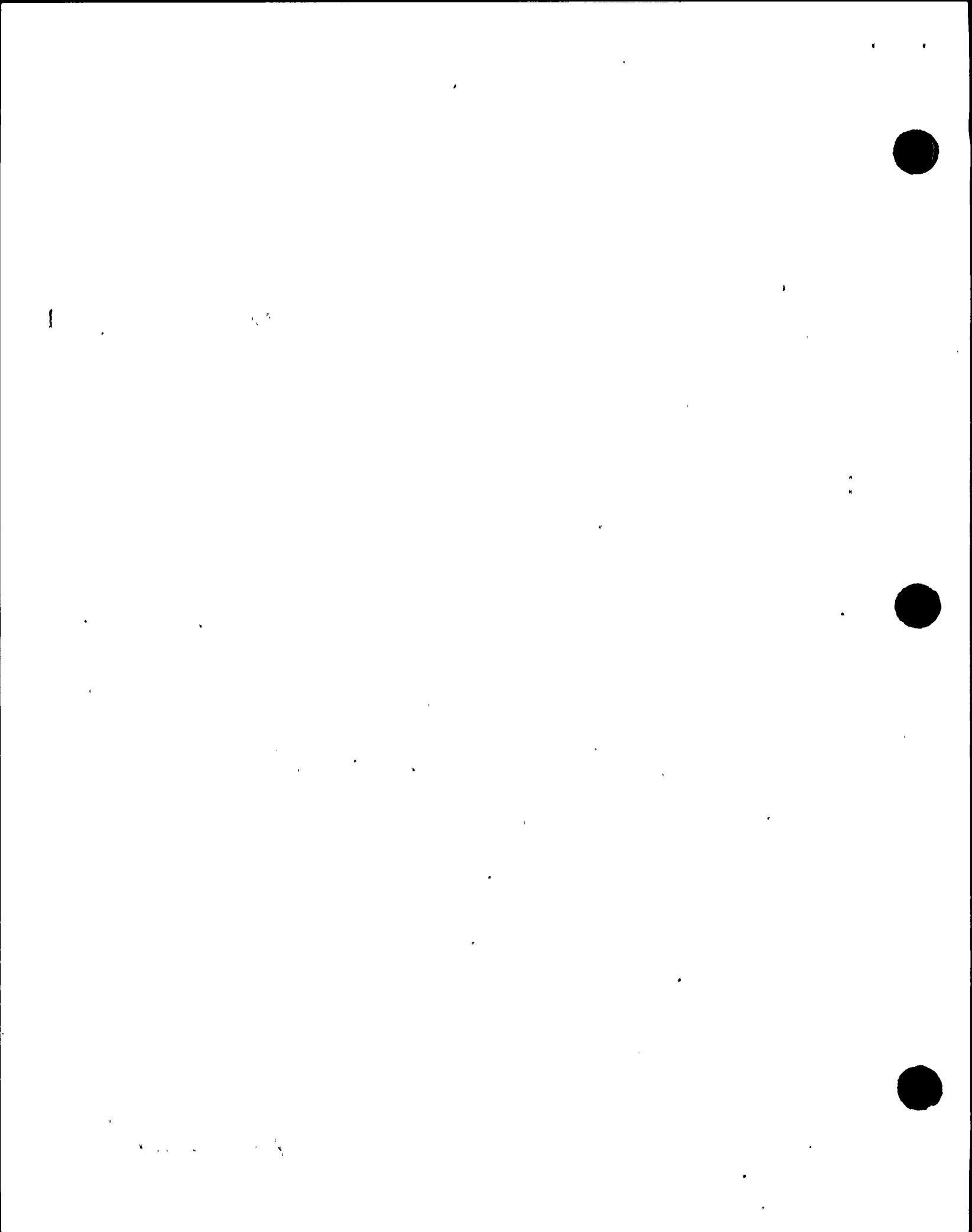
ACTION:

- a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.
- b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

- a. For all CEAs following each removal of the reactor vessel head,
- b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and
- c. At least once per 18 months.



3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

- a. Restore the linear heat rate to within its limits within one hour, or
- b. Be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

- a. Verifying that the full length CEA's are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.
- b. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on Figure 3.2-2.
- c. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2, where 100 percent of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

M x N

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.
2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F_{xy}^T curve shown in Figure 3.2-3a.

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

- a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.
- b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:
 1. Flux peaking augmentation factors as shown in Figure 4.2-1,
 2. A measurement-calculational uncertainty factor of 1.07 (when in the load follow operation mode, an uncertainty factor of 1.10 applies),
 3. An engineering uncertainty factor of 1.03,
 4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and
 5. A THERMAL POWER measurement uncertainty factor of 1.02.

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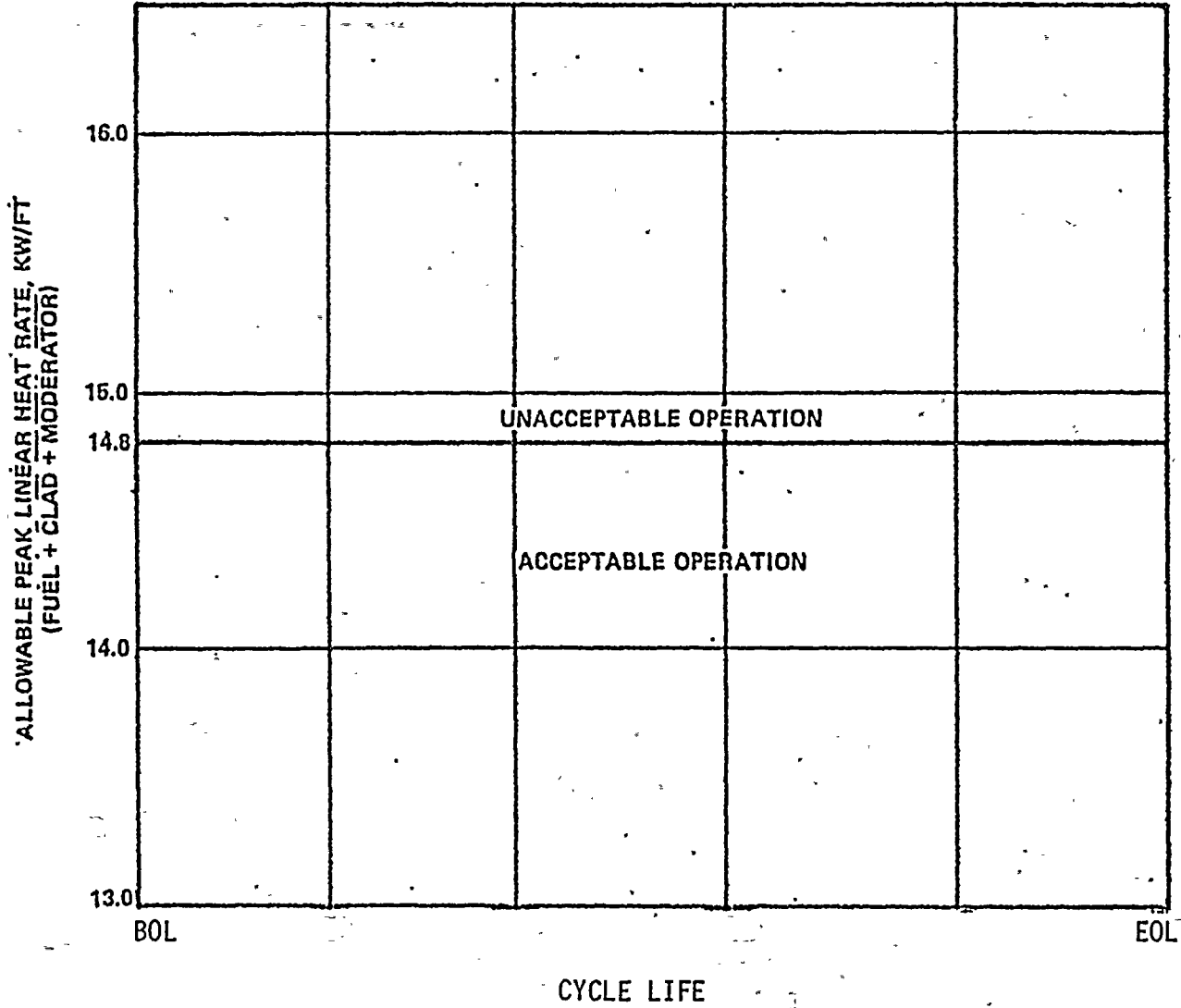


Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup



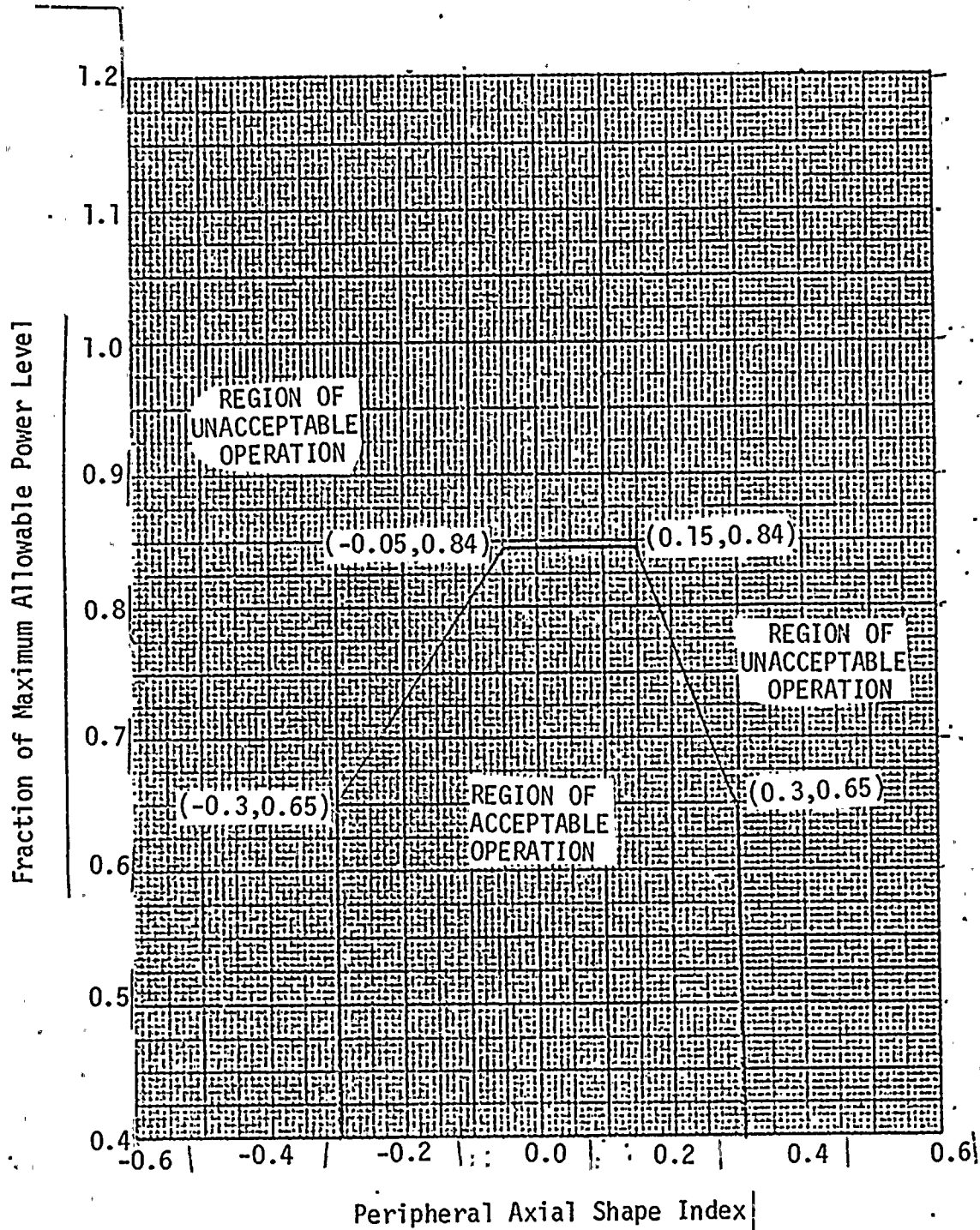


FIGURE 3.2-2
 AXIAL SHAPE INDEX vs. Fraction of Maximum
 Allowable Power Level per Specification 4.2.1.3

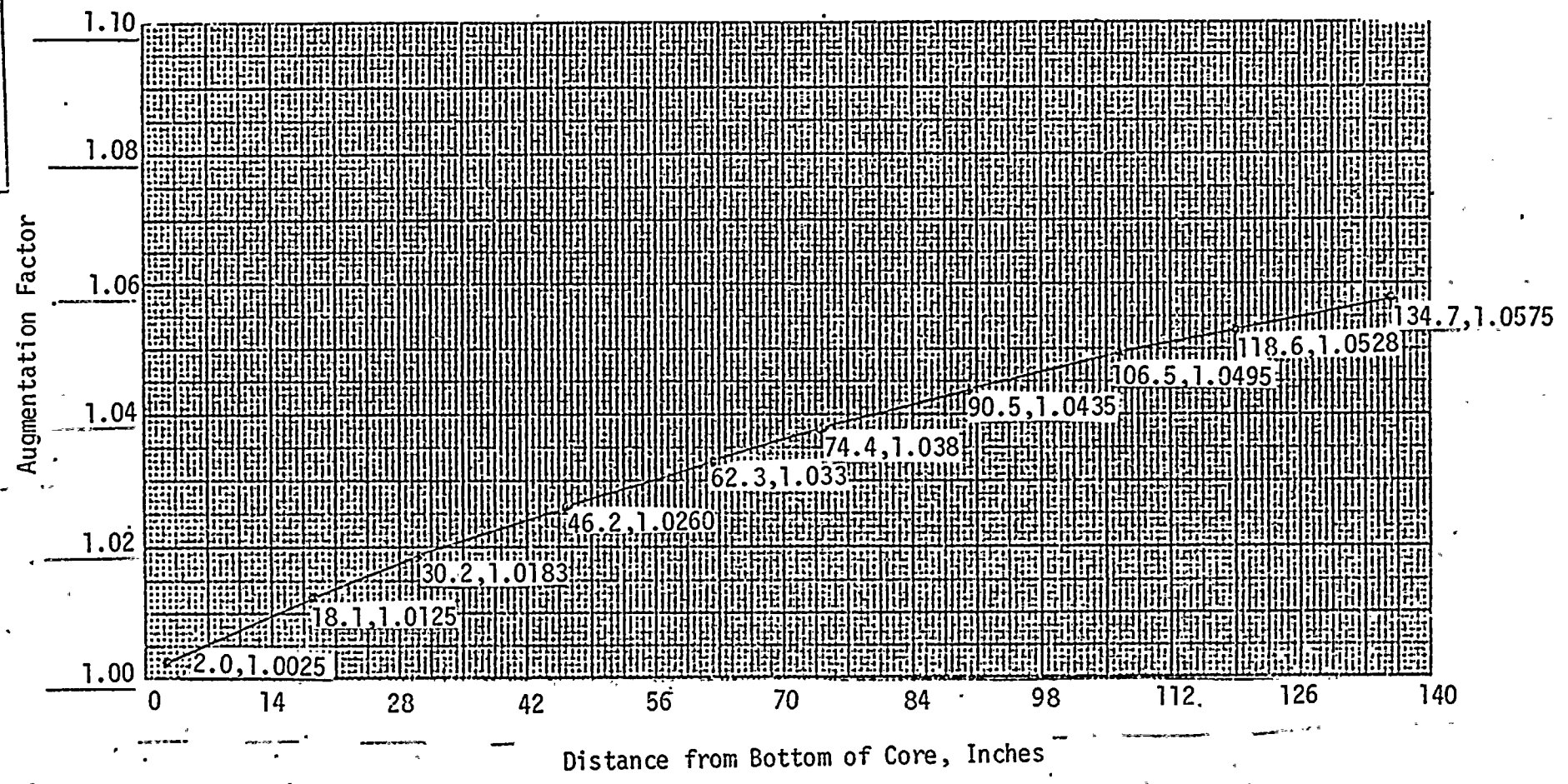


Figure 4.2-1 Augmentation Factors vs. Distance from Bottom of Core

POWER DISTRIBUTION LIMITS

TOTAL PLANAR RADIAL PEAKING FACTOR - F_{xy}^T

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F_{xy}^T , defined as $F_{xy}^T = F_{xy}(1+T_q)$, shall be limited to ≤ 1.627 .

APPLICABILITY: MODE 1*.

ACTION:

With $F_{xy}^T > 1.627$, within 6 hours either:

- a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_{xy}^T to within the limits of Figure 3:2-3a and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or
- b. Be in HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy}^T shall be calculated by the expression $F_{xy}^T = F_{xy}(1+T_q)$ when in a non-load following mode and by the expression $F_{xy}^T = 1.03 F_{xy}(1+T_q)$ when in a load following mode and F_{xy}^T shall be determined to be within its limit at the following intervals:

- a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,
- b. At least once per 31 days of accumulated operation in MODE 1, and
- c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is >0.03 .

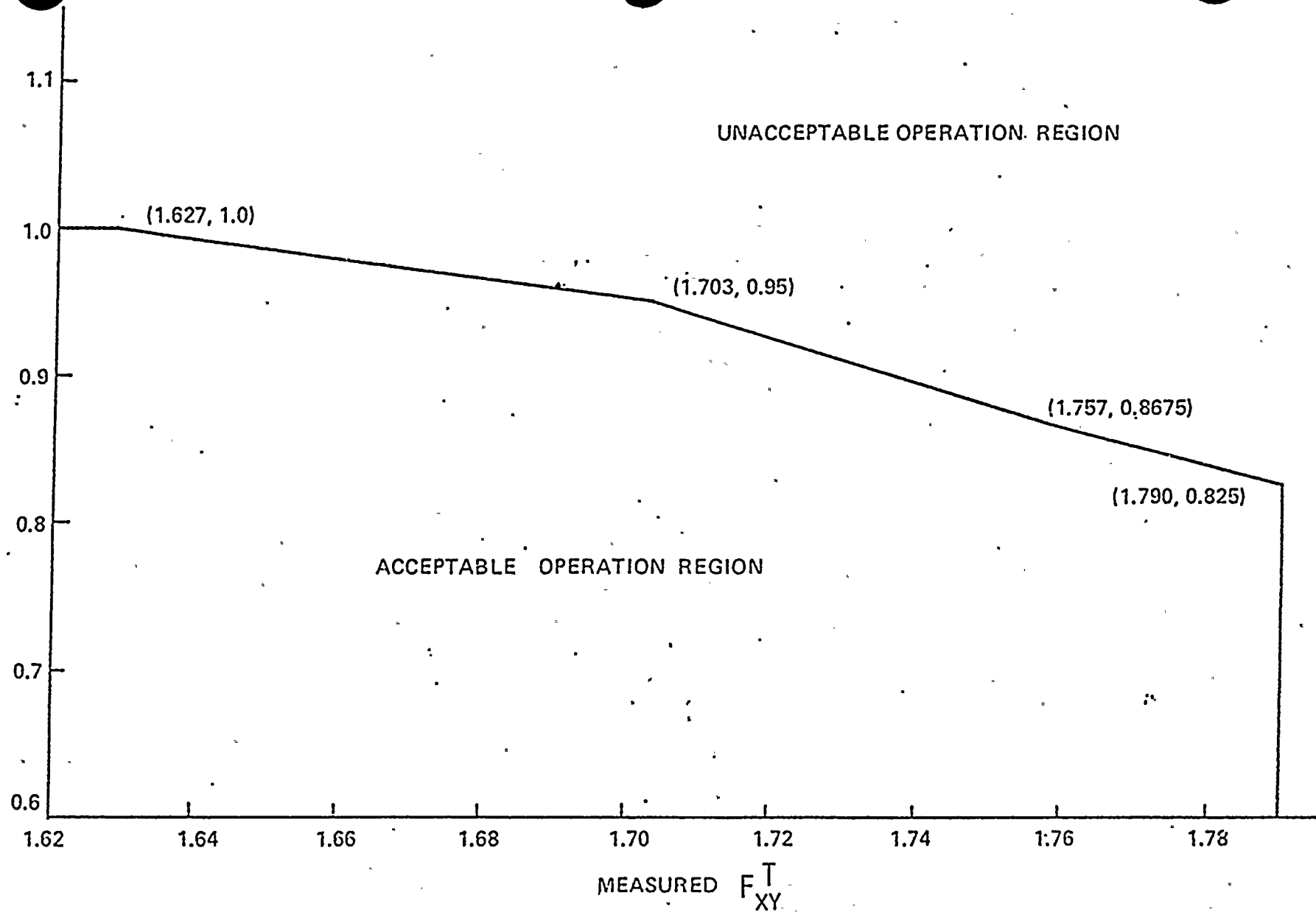
*See Special Test Exception 3.10.2.

ST. LUCIE UNIT 1

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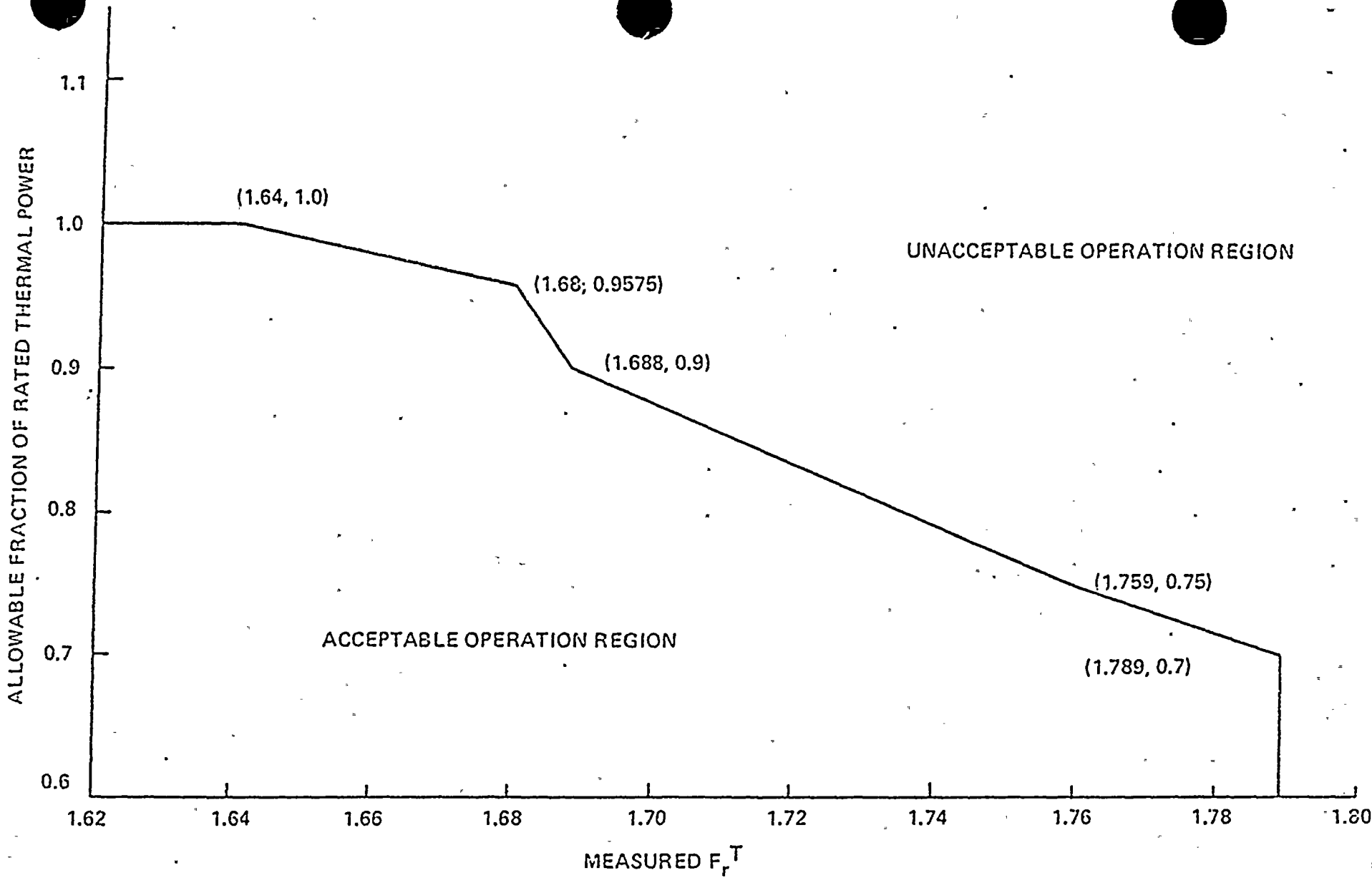
ALLOWABLE FRACTION OF RATED THERMAL POWER



MEASURED F_{xy}^T

FIGURE 3.2-3a

ALLOWABLE COMBINATIONS OF THERMAL POWER AND F_{xy}^T



MEASURED F_r^T

FIGURE 3.2-3b

ALLOWABLE COMBINATIONS OF THERMAL POWER AND F_r^T

POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T , defined as $F_r^T = F_r(1+T_q)$, shall be limited to ≤ 1.64 .

APPLICABILITY: MODE 1*.

ACTION:

With $F_r^T > 1.64$, within 6 hours either:

- a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3b and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or
- b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r(1+T_q)$ when in a non-load following mode and by the expression $F_r^T = 1.02 F_r(1+T_q)$ when in a load following mode and F_r^T shall be determined to be within its limit at the following intervals:

- a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,
- b. At least once per 31 days of accumulated operation in MODE 1, and
- c. Within four hours if the AZIMUTHAL TILT (T_q) is >0.03 .

*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.03.

APPLICABILITY: MODE 1*

ACTION:

- a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.030 but ≤ 0.10 , either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F_{xy}^T) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T) are within the limits of Specifications 3.2.2 and 3.2.3.
- b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10 , operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T) and TOTAL PLANAR RADIAL PEAKING FACTOR (F_{xy}^T) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to $\leq 20\%$ of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENT

- 4.2.4.1 The provisions of Specification 4.0.4 are not applicable.
- 4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:
 - a. Calculating the tilt at least once per 7 days when the Subchannel Deviation Alarm is OPERABLE,

*See Special Test Exception 3.10.2.

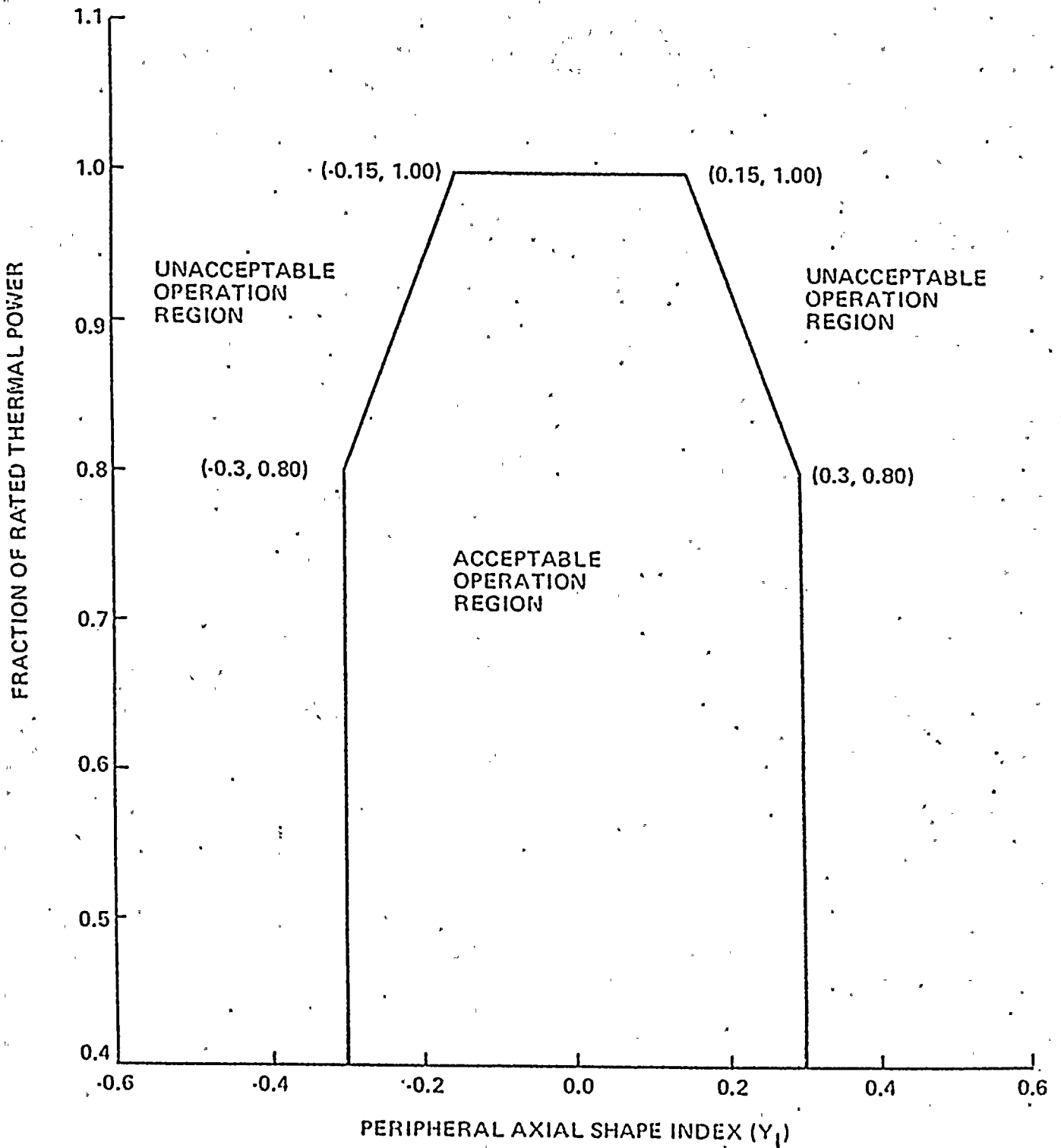


FIGURE 3.2-4

AXIAL SHAPE INDEX OPERATING LIMITS WITH 4 REACTOR COOLANT PUMPS OPERATING



2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Coolant Flow-Low (Continued)

reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.30 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.30 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

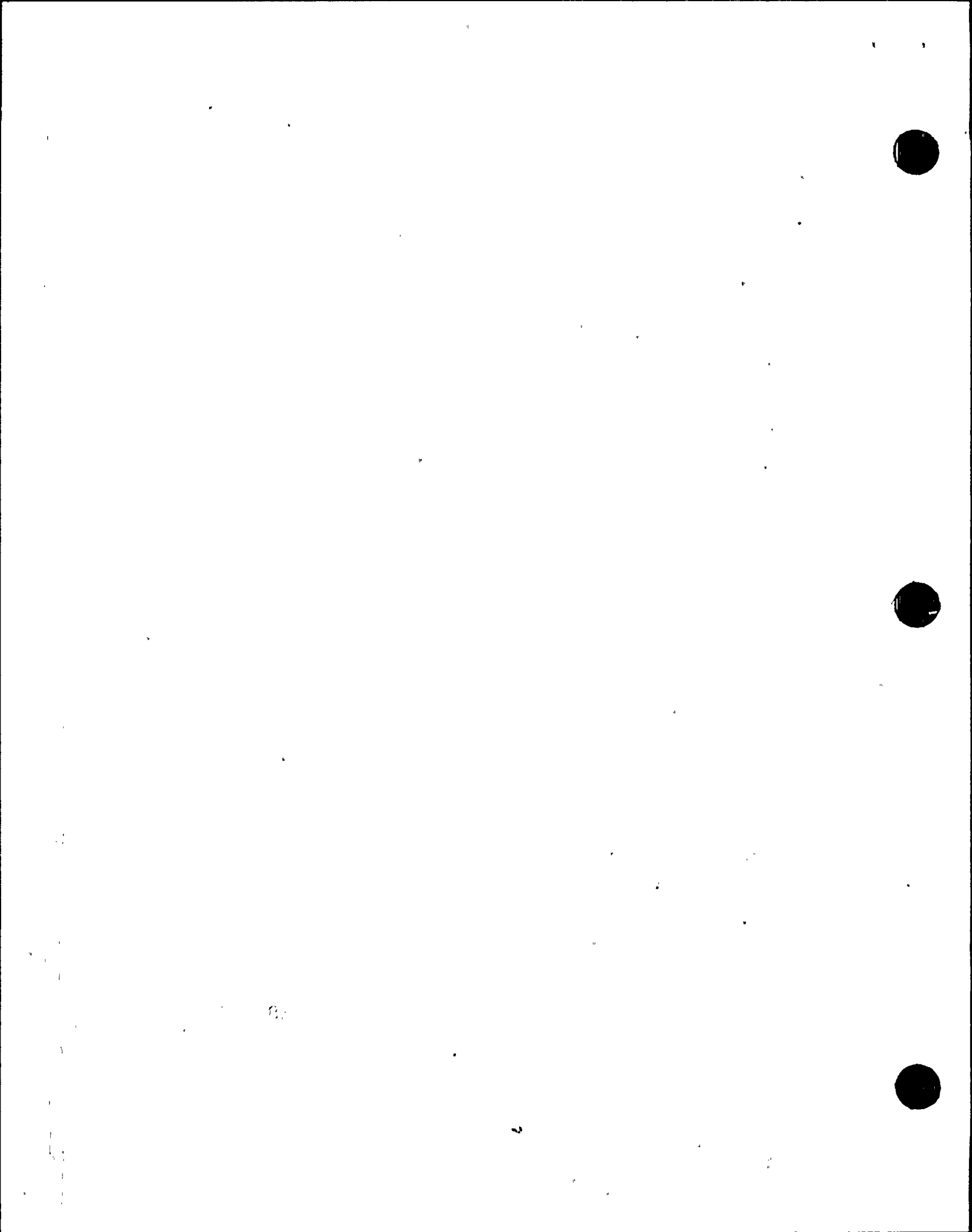
The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 500 psia is sufficiently below the full-load operating point of 800 psig so as not



REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time permitted by Specification 3.1.3.4 is the assumed CEA drop time of 3.1 seconds used in the safety analyses. Measurement with $T_{avg} \geq 515^{\circ}\text{F}$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

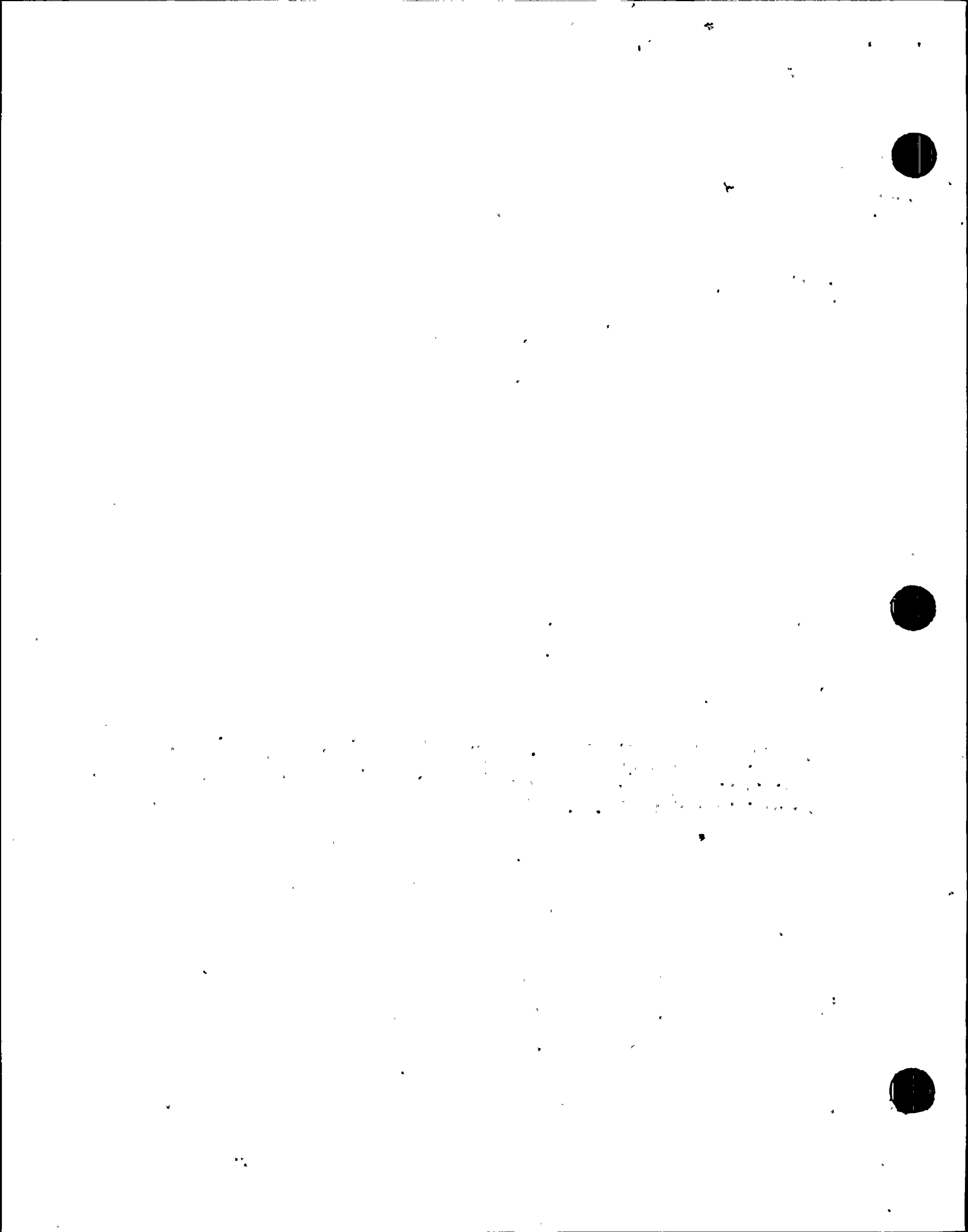
The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.07 for non-load following, 1.10 for load following, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F_{xy}^T AND F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitations on F_{xy}^T and T_q are provided to ensure that the assumptions used in the analysis^{xy} for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F_r^T and T_q are provided to ensure that the assumptions



CHAPTER 8

St. Lucie I - Cycle 3
Performance Results

1.0 Introduction and Summary

The ECCS performance evaluation for St. Lucie Unit I, Cycle 3, presented herein demonstrates appropriate conformance with 10CFR50.46 which presents the Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Reactors⁽¹⁾. The evaluation demonstrates acceptable ECCS performance for St. Lucie Unit I, during cycle 3, at a peak linear heat generation rate of 14.8 kw/ft. The method of analysis and results are presented in the following sections.

2.0 Method of Analysis

The method of analysis consisted of a comparison of the pertinent fuel parameters for the limiting fuels in cycles 2 and 3. Although the comparison of the fuel parameters (Table 8-1) shows that cycle 3 has the higher stored energy, there are additional considerations which establish that the previous analysis is conservative, and, hence, can also be applied to cycle 3. The bases for concluding that the ECCS performance for cycle 3 is bounded by the cycle 2 analysis⁽¹²⁾ are discussed in the results section below. As a result, the peak clad temperature and local clad oxidation (STRIKIN-II)⁽⁶⁾ calculations performed for cycle 2⁽¹²⁾ are also conservative for cycle 3. In addition, the blowdown (CEFLASH-4A)⁽⁴⁾, refill (COMPERC-II)⁽⁵⁾, and core wide oxidation (COMZIRC)^(4, sup. 1) analyses as used in evaluating the cycle 2 analysis remain valid for cycle 3. The comparison of the pertinent fuel parameters supporting this conclusion for cycle 3 is presented below.

3.0 Results

The cycle 3 core contains 21 Batch B low density depleted fuel assemblies, 128 partially depleted Batch C and D high density fuel assemblies, and 68 undepleted Batch E high density fuel assemblies.

Burnup dependent calculations were performed for the high density fuel assemblies with the FATES⁽⁷⁾ and STRIKIN-II⁽⁶⁾ codes. The results demonstrate that the most limiting fuel pin during cycle 3 is located in one of the undepleted Batch E assemblies.

Table 8-1 compares the fuel specific parameters which correspond to the limiting fuels in cycle 2 and cycle 3. As shown in the table, the limiting high density fuel in cycle 3 has a stored energy 13⁰F higher than the limiting fuel in the previous cycle 2 analysis.

Although the stored energy for the limiting fuel for cycle 3 is slightly higher than the limiting fuel for cycle 2, the ECCS performance for cycle 3 remains bounded by the previous cycle for the following reasons:

1. The cycle 2 analysis did not utilize the PARCH⁽¹⁰⁾ code to compute steam cooling heat transfer coefficients after the rod rupture is predicted to occur. Since the peak clad temperature is achieved during late reflood, use of the PARCH code will significantly reduce the peak temperatures calculated in the cycle 2 analysis. Use of the PARCH methodology should reduce the peak clad temperature by at least 100⁰F.
2. The radial power distribution for use in the thermal rod-to-rod radiation model is less limiting for cycle 3 than that used for the previous cycle. Use of the less limiting radiation enclosure for cycle 3 should reduce the peak clad temperature at least 50⁰F.

The above defined margins are far in excess of the 13⁰F increase in fuel stored energy for the cycle 3 limiting fuel relative to the previous cycle. It can, therefore, be concluded that the cycle 2 analysis is also applicable to cycle 3.

4.0 Conclusion

The comparison between the pertinent fuel parameters and analysis assumptions for the limiting fuels in cycles 2 and 3 demonstrates that the cycle 2 ECCS performance analysis conservatively bounds the performance for cycle 3. Therefore, the peak linear heat generation rate of 14.8 kw/ft which was demonstrated to be acceptable for cycle 2 is also an acceptable limit for cycle 3 operation. Conformance of this evaluation is the same as stated in References 11 and 12.

As presented in Reference 8, the small breaks are not limiting.

5.0 Computer Code Version Identification

Version 77036 of the STRIKIN-II code of Combustion Engineering's ECCS Evaluation Model was used to perform the burnup dependent calculations in evaluating the fuel data.

Table 8-1

St. Lucie I Cycle 3 Core Parameters

Quality

Value

Cycle 2
Batch D

Cycle 3
Batch E

Average Linear Heat Rate (102% of Nominal)

6.2126

6.0956

kw/ft

Gap Conductance at PLHGR

1552

1525

BTU/hr-ft²-°F

Fuel Centerline Temperature at PLHGR

3484

3512

°F

Fuel Average Temperature at PLHGR

2184

2197

°F

Hot Rod Gas Pressure

1047.8

1031

psia

Hot Rod Burnup (Minimum HGAP)

820

820

MWD/MTU

IV. References

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.

2. CENPD-132, "Calculative Methods for the CE Large Break LOCA Evaluation Model", August 1974 (Proprietary).

CENPD-132, Supplement 1, "Updated Calculative Methods for the CE Large Break LOCA Evaluation Model", December 1974 (Proprietary).

3. CENPD-132, Supplement 2, "Calculation Methods for the CE Large Break LOCA Evaluation Model", July 1975, (Proprietary).

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

CENPD-133, Supplement 2, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis (Modification)", December 1974 (Proprietary).

5. CENPD-134, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core", April 1974 (Proprietary).

CENPD-134, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modification)", December 1974 (Proprietary).

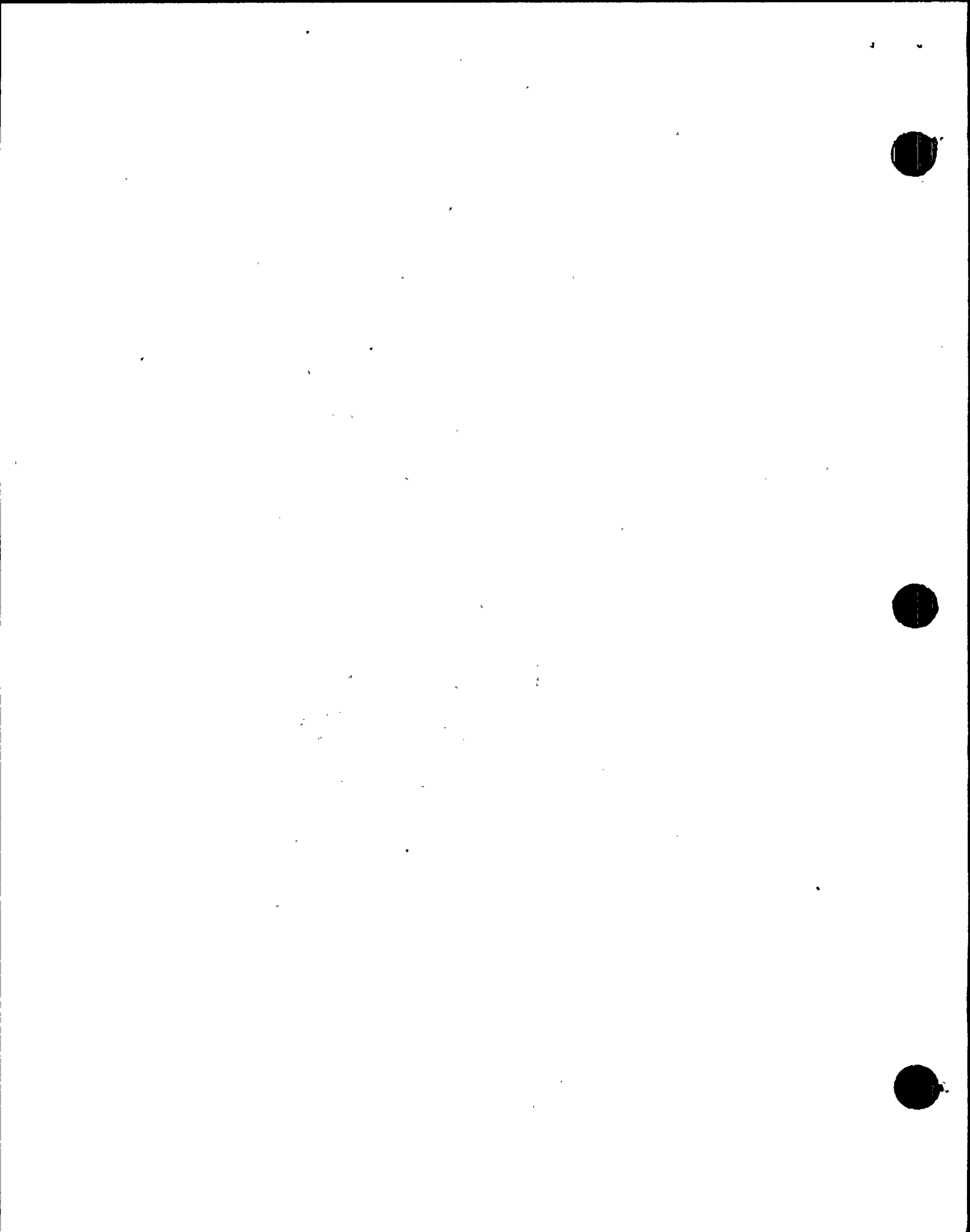
6. CENPD-135, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program, April 1974 (Proprietary).

CENPD-135, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)", February 1975 (Proprietary).

CENPD-135, Supplement 4, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program", August 1976, (Proprietary).

CENPD-135, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977 (Proprietary).

7. CENPD-139, "CE Fuel Evaluation Model", July 1974 (Proprietary).
8. CENPD-137, "Calculative Methods for the CE Small Break LOCA Evaluation Model", Combustion Engineering Proprietary Report, August 1974 (Proprietary).
9. St. Lucie Nuclear Power Plant (Formerly Hutchinson Island) Unit One, Final Safety Analysis Report in Support of Docket No. 50-335, License No. DPR-67.
10. CENPD-128, "PARCH, A FORTRAN IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup", Safety Analysis Group, August, 1974.-
11. Cycle II Analysis
12. Cycle II Re-Analysis



9. TECHNICAL SPECIFICATIONS

In this section, changes are described that must be made to the Technical Specifications in order to make them valid for operation of Cycle 3. A summary description of the recommended change is presented for each page that must be modified. The pages themselves, with modifications included, are shown in Attachment B.

Page 1-6

New Specification 1.30 adds a definition for LOAD FOLLOW OPERATION. The definition has been added to accommodate an additional set of physics uncertainties as requested by the NRC.

Page 2-4

Table 2.2-1 setpoint limits (Reactor Protective Instrumentation) are revised for the "Containment Pressure - High" and "Steam Generator Pressure - Low" trips in Table 2.2-1. Refer to FPL letter L-79-5 of January 8, 1979 for additional information of the "Containment Pressure - High" setpoint revision.

Page 2-7

Figure 2.2-2 is revised based on the Cycle 3 analysis.

Page 3/4 1-26

A more conservative CEA drop time limit of 3.1 seconds is incorporated in Specification 3.1.3.4.

Page 3/4 2-1

New Specification 4.2.1.3.a is added to require verification of proper full length CEA withdrawal. The current Specifications 4.2.1.3.a and 4.2.1.3.b are renumbered 4.2.1.3.b and 4.2.1.3.c, respectively.

Page 3/4 2-2

The wording on how to determine the factor "N" in Specification 4.2.1.3 is revised.

The uncertainty factor in Specification 4.2.1.4.b.2 is revised to 1.07 for non-load-follow operation and 1.10 for load-follow operation.

Page 3/4 2-3

Figure 3.2-1 is revised to extend its applicability beyond Cycle 2.

Page 3/4 2-4

Figure 3.2-2 is revised based on the Cycle 3 analysis.



The following information was obtained from the records of the
 Department of the Interior, Bureau of Land Management, on
 the subject of the above-captioned matter.

The Bureau of Land Management has advised that the
 land described in the above-captioned matter is
 owned by the United States of America.

The Bureau of Land Management has advised that the
 land described in the above-captioned matter is
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 land described in the above-captioned matter is
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The Bureau of Land Management has advised that the
 land described in the above-captioned matter is
 situated in the City of Los Angeles.

The Bureau of Land Management has advised that the
 land described in the above-captioned matter is
 situated in the City of Los Angeles.

Page 3/4 2-5

Figure 4.2-1 is revised based on the Cycle 3 analysis.

Page 3/4 2-6

The total planar radial peaking factor limit of Specification 3.2.2 is revised. The revised limit is consistent with Figure 3.2-3a, and is consistent with the radial peaking factor inputs for the Cycle 3 analysis.

A separate equation, applicable to load-follow operation, is added to Specification 4.2.2.2.

The allowable Azimuthal Power Tilt of Specification 4.2.2.2c is revised.

Page 3/4 2-8a

A separate Figure 3.2-3a is provided to show the allowable combinations of thermal power and total planar radial peaking factor, based on the Cycle 3 analysis.

Page 3/4 2-8b

A separate Figure 3.2-3b is provided to show the allowable combinations of thermal power and total integrated radial peaking factor, based on the Cycle 3 analysis.

Page 3/4 2-9

The total integrated radial peaking factor limit of Specification 3.2.3 is revised. The revised limit is consistent with Figure 3.2-3b, and is consistent with the integrated peaking factor inputs for the Cycle 3 analysis.

A separate equation, applicable to load-follow operation, is added to Specification 4.2.3.2.

The allowable Azimuthal Tilt of Specification 4.2.3.2.c is revised.

Page 3/4 2-11

The azimuthal power tilt limit of Specification 3.2.4 is revised to 0.03.

Page 3/4 2-15

Figure 3.2-4 is revised based on the Cycle 3 analysis.

Page B 2-5

The revised setpoint of 500 psia is included in the Bases section for the Steam Generator Pressure - Low reactor trip.

Page B 3/4 1-4

The revised CEA drop time limit of 3.1 seconds is included in Bases Section 3/4.1.3.

Page B 3/4 2-1

Bases section 3/4.2.1 is revised to include separate uncertainty factors for load-follow and non-load-follow operation.

