PHILADELPHIA ELECTRIC COMPANY

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(215) 841-4000

May 31, 1989

Docket Nos.: 50-352
50-353

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

SUBJECT: Draft Limerick Generating Station, Units 1 and 2 FSAR Revisions to Incorporate the Extended Load Line Region, Increased Core Flow, and Partial Feedwater Heating

Gentlemen:

This letter provides markup of current FSAR pages (Enclosure 1) which are made to incorporate the extended load line region (ELLR), increased core flow (ICF), and partial feedwater heating (PFH) into the Limerick FSAR. These changes were discussed with NRC staff on May 11, 1989 and are consistent with the Final Draft version of the Unit 2 Technical Specifications transmitted by NRC letter dated May 19, 1989. The analyses supporting these operational conditions for Unit 2 are included as Enclosures 2 and 3. Enclosures 4 and 5 contain the corresponding General Electric affidavits to support a proprietary finding in accordance with 10CFR2.790 for the reports of Enclosures 2 and 3. The Unit 2 analyses are identical to the Unit 1 analyses which the NRC has previously accepted by safety evaluations dated February 17, 1987 (ICF and PFH), and August 14, 1987 (ELLR). The attached FSAR markup also updates the power flow map of Section 14. These proposed revisions will be incorporated in a future FSAR amendment.

If any additional information is required, please let me know.

Sincerely,

G. A. Hunger, Jr.

G. A. Hunger, Jr.
Director, Licensing Section

Enclosures

MAM/mv/05128901-A

cc: W. T. Russell, USNRC, Administrator - Region I
T. J. Kenny, USNRC, Senior Resident Inspector - LGS
R. J. Clark, USNRC, LGS - Project Manager

A001 1/1 Change LPDR Ltr 1 Encl 1NP Portion
NRC PDR 1 1NP Portion

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	S. J. Kowalski	S25-1
	E. J. Bradley	S23-1
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	J. M. Madara, Jr.	S7-1
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	A. S. MacAinsh	LGS MC SB3-4
	M. S. Iyer	BPC/SF
	DAC	NG-8

GENERAL ELECTRIC COMPANY

AFFIDAVIT

I, Robert C. Mitchell, being duly sworn, depose and state as follows:

1. I am Manager, Nuclear Products Licensing, General Electric Company, and have been delegated the function of reviewing the information described in paragraph 2 which is sought to be withheld and have been authorized to apply for its withholding.

2. The information sought to be withheld is contained in NEDC-31577P (Class III), "General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Limerick Generating Station Unit 2, Cycle 1," GE Nuclear Energy, March 1989.

3. In designating material as proprietary, General Electric utilizes the definition of proprietary information and trade secrets set forth in the American Law Institute's Restatement of Torts, Section 757. This definition provides:

   "A trade secret may consist of any formula, pattern, device or compilation of information which is used in one's business and which gives him an opportunity to obtain an advantage over competitors who do not know or use it.... A substantial element of secrecy must exist, so that, except by the use of improper means, there would be difficulty in acquiring information.... Some factors to be considered in determining whether given information is one's trade secret are: (1) the extent to which the information is known outside of his business; (2) the extent to which it is known by employees and others involved in his business; (3) the extent of measures taken by him to guard the secrecy of the information; (4) the value of the information to him and to his competitors; (5) the amount of effort or money expanded by him in developing the information; (6) the ease or difficulty with the which the information could be properly acquired or duplicated by others."

4. Some examples of categories of information which fit into the definition of proprietary information are:

   a. Information that discloses a process, method or apparatus where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information consisting of supporting data and analyses, including test data, relative to a process, method or apparatus, the application of which provide a competitive economic advantage, e.g., by optimization or improved marketability;

c. Information which if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product;

d. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of General Electric, its customers or suppliers;

e. Information which reveals aspects of past, present or future General Electric customer-funded development plans and programs of potential commercial value to General Electric:

f. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection;

g. Information which General Electric must treat as proprietary according to agreements with other parties.

5. Initial approval of proprietary treatment of presentation information is typically made by the Subsection manager of the originating component, the person who is most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within the Company is limited on a "need to know" basis and such documents are clearly identified as proprietary.

6. The procedure for approval of external release of such presentation information typically requires review by the Subsection Manager, Project Manager, Principal Scientist or other equivalent authority, by the Subsection Manager of the cognizant Marketing function (or delegate) and by the Legal Operation for technical content, competitive effect and determination of the accuracy of the proprietary designation in accordance with the standards enumerated above. Disclosures outside General Electric are generally limited to regulatory bodies, customers and potential customers and their agents, suppliers and licensees then only with appropriate protection by applicable regulatory provisions or proprietary agreements.

7. The presentation information mentioned in paragraph 2 above has been evaluated in accordance with the above criteria and procedures and has been found to contain information which is proprietary and which is customarily held in confidence by General Electric.

8. The information to the best of my knowledge and belief has consistently been held in confidence by the General Electric Company, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties have been made pursuant to regulatory provisions of proprietary agreements which provide for maintenance of the information in confidence.

9. Public disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of the General Electric Company and deprive or reduce the availability of profit making opportunities because it would provide other parties, including competitors, with valuable information

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

Robert C. Mitchell, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 24 day of MAY, 1989.

Robert C. Mitchell
Robert C. Mitchell
General Electric Company

Subscribed and sworn before me this 24th day of May 1989.

Mary L. Kendall
NOTARY PUBLIC, STATE OF CALIFORNIA

OFFICIAL SEAL
MARY L KENDALL
Notary Public-California
SANTA CLARA COUNTY
My Comm. Exp. Mar. 26, 1993

GENERAL ELECTRIC COMPANY

AFFIDAVIT

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2. The information sought to be withheld is contained in NEDC-31578P (Class III), "Increased Core Flow and Partial Feedwater Heating Analysis for Limerick Generating Station Unit 2, Cycle 1," GE Nuclear Energy, March 1989.

3. In designating material as proprietary, General Electric utilizes the definition of proprietary information and trade secrets set forth in the American Law Institute's Restatement of Torts, Section 757. This definition provides:

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d. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of General Electric, its customers or suppliers;

e. Information which reveals aspects of past, present or future General Electric customer-funded development plans and programs of potential commercial value to General Electric;

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STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

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That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 5 day of MAY, 1989.

Robert C. Mitchell

Robert C. Mitchell
General Electric Company

Subscribed and sworn before me this 5th day of May 1989.

Mary L. Kendall

NOTARY PUBLIC, STATE OF CALIFORNIA

OFFICIAL SEAL
MARY L KENDALL
Notary Public-California
SANTA CLARA COUNTY
My Comm. Exp. Mar. 26, 1993

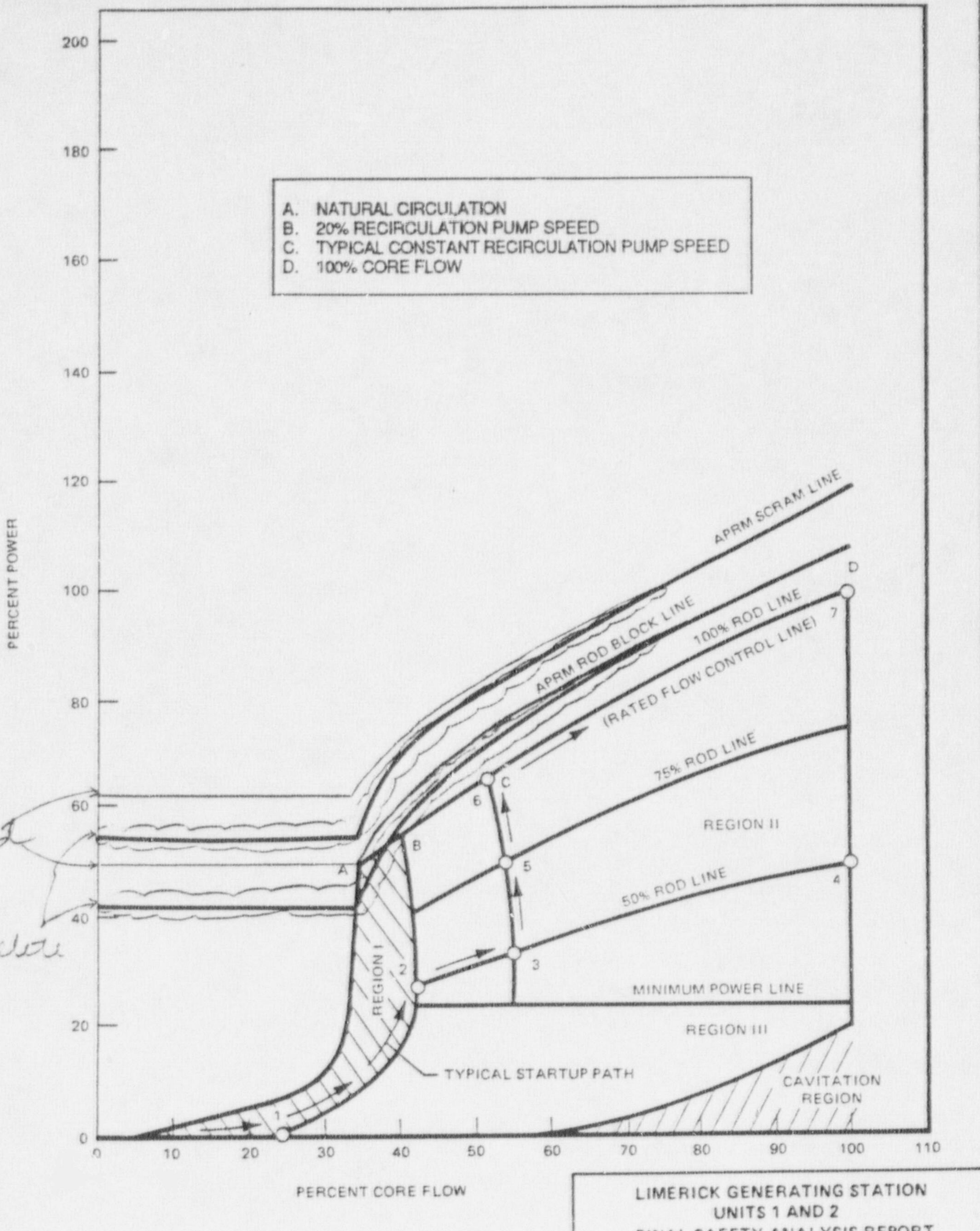
LDCN PS-1506

TABLE 4.3-7

CALCULATED INITIAL CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C

	Unit 1	Unit 2
Beginning of Cycle-1, K-effective		
Uncontrolled	1.1047	1.1051
Fully Controlled	0.9230	0.9330
Strongest Control Rod Out (26-55)	0.9826	0.9799
R, Maximum Increase in Cold Core Reactivity with Exposure Cycle-1, Δk	0.0	0.0

LDCN FS-1596



LIMERICK GENERATING STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
POWER–FLOW OPERATING MAP
FIGURE 4.4-2 REV. 54, 10/88

TABLE 6.3-4 (Page 1 of 3)

MAPLHGR, MAXIMUM LOCAL OXIDATION & PEAK CLAD TEMPERATURE VS. EXPOSURE [1,2,3]

Exposure MWD/ST	Exposure MWD/MT	MAPLHGR[4] KW/FT	P.C.T. DEG F	Oxide Fraction
A. Fuel Type: P8CIB071-No Gd-150 - ~~100M~~ (P8CRB071)[5]				
200	220.5	11.5	1960	.011
1,000	1,102.3	11.4	1921	.009
5,000	5,511.5	11.4	1854	.007
10,000	11,023.0	11.5	1838	.007
15,000	16,534.5	11.5	1835	.007
20,000	22,046.0	11.1	1787	.006
25,000	27,557.5	10.4	1720	.004
30,000	33,069.0	9.8	1649	.003
35,000	38,580.5	9.1	1575	.002
40,000	44,092.0	8.5	1500	.002
45,000	49,603.5	7.8	1423	.001
B. Fuel Type: P8CIB094 - No Gd - 150 - ~~100M~~ (P8CRB094)[5]				
200	220.5	10.7	1954	.011
1,000	1,102.3	11.0	1952	.011
5,000	5,511.5	11.6	1925	.009
10,000	11,023.0	11.9	1905	.009
20,000	22,046.0	11.3	1815	.006
25,000	27,557.5	10.5	1725	.005
30,000	33,069.0	9.8	1645	.003
35,000	38,580.5	9.2	1575	.002
40,000	44,092.0	8.5	1505	.002
45,000	49,603.5	7.9	1432	.001
C. Fuel Type: P8CIB163 - 2GZ - 150 - ~~100M~~ (P8CRB163)[5]				
200	220.5	11.8	2031	.014
1,000	1,102.3	11.8	2031	.014
5,000	5,511.5	12.4	2047	.014
10,000	11,023.0	12.8	2042	.013
15,000	16,534.5	12.9	2067	.015
20,000	22,046.0	12.9	2067	.015
25,000	27,557.5	12.2	1970	.011
30,000	33,069.0	11.2	1829	.007
35,000	38,580.5	10.6	1748	.005
40,000	44,092.0	10.1	1686	.004
45,000	49,603.5	9.4	1626	.003

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TABLE 6.3-4 (Cont'd)

D. Fuel Type: P8CIB248 - 4GZ - 150 - 100M (P8CRB248)(5)

Exposure MWD/ST	Exposure MWD/MT	MAPLHGR(4) KW/FT	P.C.T. DEG F	Oxide Fraction
200	220.5	12.1	2090	.017
1,000	1,102.3	12.1	2080	.017
5,000	5,511.5	12.3	2028	.013
10,000	11,023.0	12.2	1987	.012
15,000	16,534.5	12.1	1998	.012
20,000	22,046.0	11.9	1985	.012
25,000	27,557.5	11.2	1911	.009
30,000	33,069.0	10.7	1822	.007
35,000	38,580.5	10.0	1743	.005
40,000	44,092.0	9.4	1674	.004
45,000	49,603.5	8.7	1612	.003

E. Fuel Type: P8C1B278 - 3G3.0 - 150 - 100M (P8CRB278)(5)

Exposure MWD/ST	Exposure MWD/MT	MAPLHGR(4) KW/FT	P.C.T. DEG F	Oxide Fraction
200	220.5	11.7	1998	.013
1,000	1,102.3	11.8	1997	.013
5,000	5,511.5	12.4	2010	.012
10,000	11,023.0	12.5	2002	.012
15,000	16,534.5	12.4	1996	.012
20,000	22,046.0	12.2	1983	.011
25,000	27,557.5	11.5	1903	.009
30,000	33,069.0	10.8	1817	.006
35,000	38,580.5	10.2	1732	.005
40,000	44,092.0	9.5	1661	.004
45,000	49,603.5	8.9	1596	.003

(1) The analyses contained herein were performed with the assumption that all lower tie plates are fully drilled.

(2) This analysis is valid for operation at all points on the power-flow map bounded by the most restrictive of the following:

a) Less than the 100% rated power line
b) Less than the APRM rod block line
c) Less than the 100% rated core flow line

(3) The corewide metal-water reaction has been calculated using method 1 described in NEDO-20566. The calculation was done using the standard model power distribution. The value is as follows:

Corewide metal-water reaction % = 0.11

LDCNFS-1506

TABLE 6.3-4 (Cont'd)

(4) Maximum average planar linear heat generation rate

(5) GESTAR II Nomenclature as used on Figure 4.3-31.

(6) Unit 1 initial core fuel is 100 mil while Unit 2 initial core fuel is 80 mil.

channel thickness

channel thickness

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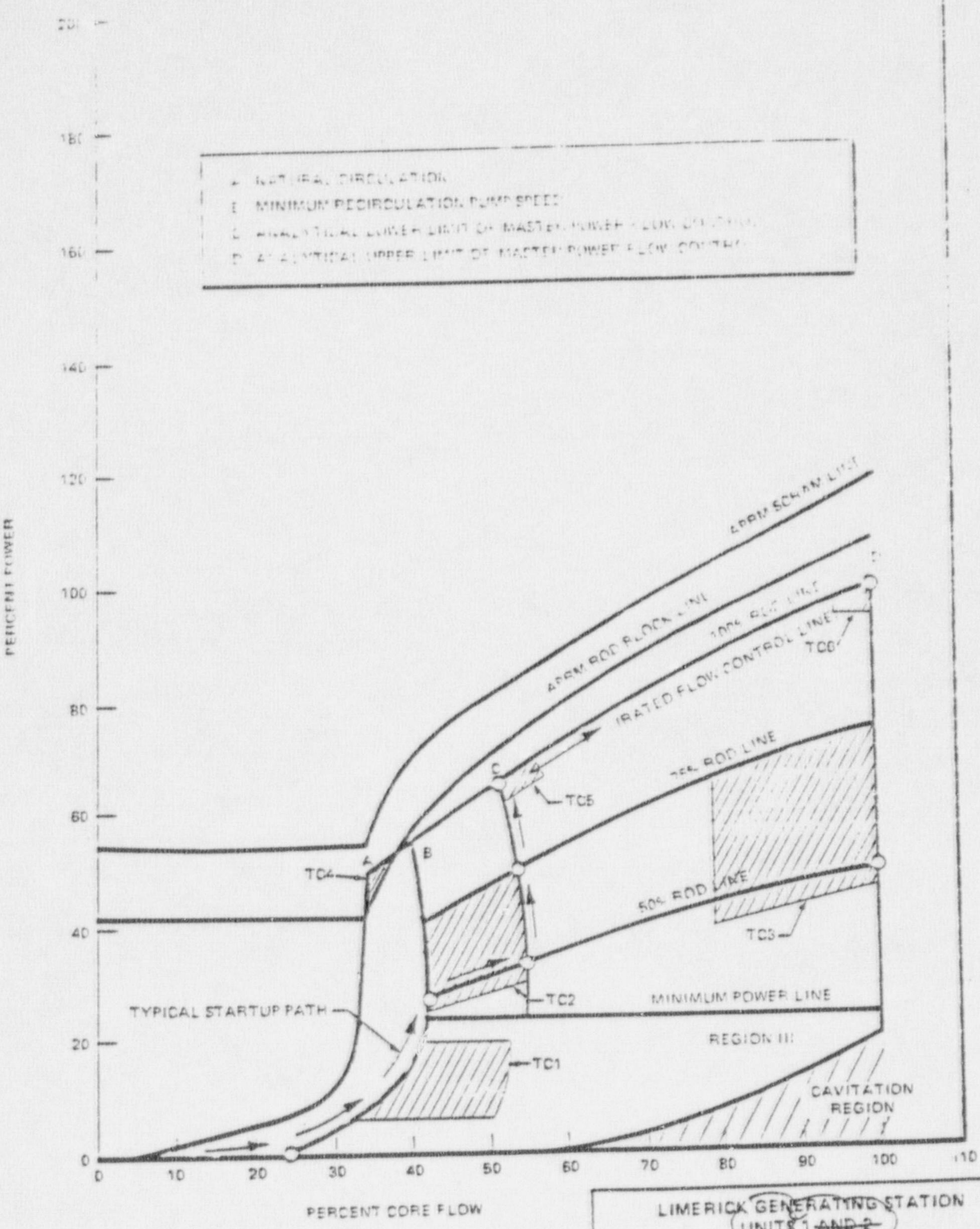


LIMERICK GENERATING STATION
UNITS 1 ~~AND 2~~
FINAL SAFETY ANALYSIS REPORT

POWER–FLOW OPERATING MAP
(SHEET 1 of ~~2~~ 4)

FIGURE 14.2-9 REV. 28, 01/84

LDCN FS-1506

TEST CONDITION (TC) REGION DEFINITIONS

TEST CONDITION NO.	POWER–FLOW MAP REGION AND NOTES
1	BEFORE OR AFTER MAIN GENERATOR SYNCHRONIZATION BETWEEN 5 AND 20% THERMAL POWER-WITHIN +10, –0% OF M-G SET MINIMUM OPERATING SPEED LINE IN LOCAL MANUAL MODE.
2	AFTER MAIN GENERATOR SYNCHRONIZATION BETWEEN THE 45 AND 75% POWER ROD LINES BETWEEN M-G SET MINIMUM SPEEDS FOR LOCAL MANUAL AND MASTER MANUAL MODES.
3	FROM 45 TO 75% CONTROL ROD LINES–CORE FLOW BETWEEN 80 AND 100% OF ITS RATED VALUE.
4	ON THE NATURAL CIRCULATION CORE FLOW LINE–WITHIN +0, –5% OF THE INTERSECTION WITH THE 100% POWER ROD LINE.
5	WITHIN +0, –5% OF THE 100% CONTROL ROD LINE–WITHIN –0, +5% OF THE ANALYTICAL LOWER LIMIT OF MASTER FLOW CONTROL.
6	WITHIN +0, –5% OF RATED 100% POWER – WITHIN +0, –5% OF RATED 100% CORE FLOW RATE.

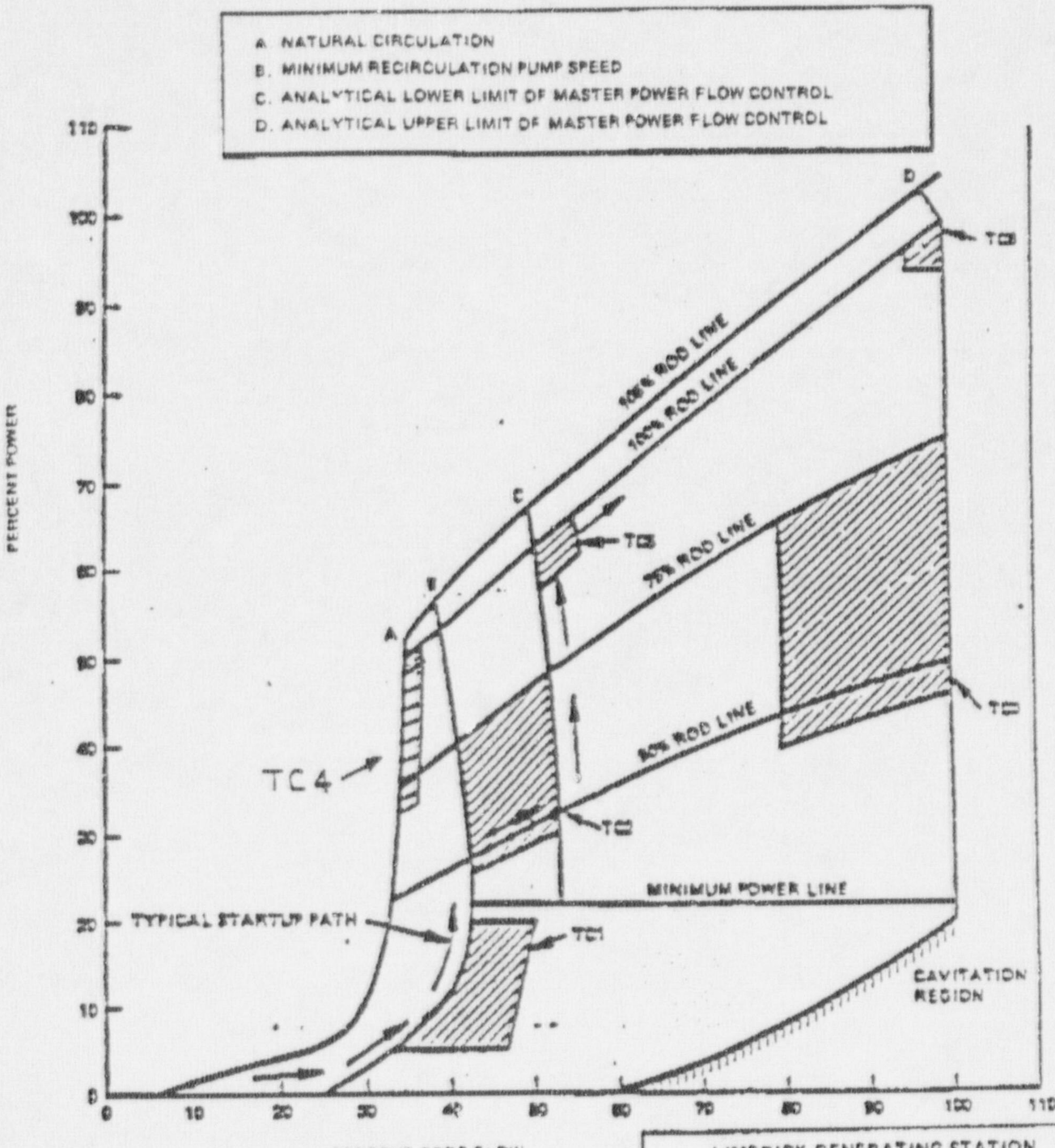
LIMERICK ~~GENERATING~~ STATION
UNITS 1 ~~AND 2~~
FINAL SAFETY ANALYSIS REPORT

POWER–FLOW OPERATING MAP
(SHEET 2 of 2)

4

FIGURE 14.2-9 REV. 28, 01/84

LDCN FS-1506



LIMERICK GENERATING STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

ADD THIS PAGE

POWER–FLOW OPERATING MAP
(SHEET 3 OF 4)

FIGURE 14.2-9

REV. 28, 01/84

LDCN FS-1506

TEST CONDITION (TC) REGION DEFINITIONS

TEST CONDITION NO.	POWER—FLOW MAP REGION AND NOTES
1	BEFORE OR AFTER MAIN GENERATOR SYNCHRONIZATION BETWEEN 5 AND 20% THERMAL POWER-WITHIN +10, –0% OF M-G SET MINIMUM OPERATING SPEED LINE IN LOCAL MANUAL MODE.
2	AFTER MAIN GENERATOR SYNCHRONIZATION BETWEEN THE 45 AND 75% POWER ROD LINES BETWEEN M-G SET MINIMUM SPEEDS FOR LOCAL MANUAL AND MASTER MANUAL MODES.
3	FROM 45 TO 75% CONTROL ROD LINES—CORE FLOW BETWEEN 80 ~~AND~~ (20) 100% OF ITS RATED VALUE.
4	ON THE NATURAL CIRCULATION CORE FLOW LINE—WITHIN +0, –5% OF THE INTERSECTION WITH THE 100% POWER ROD LINE.
5	WITHIN +0, –5% OF THE 100% CONTROL ROD LINE—WITHIN –0, +5% OF THE ANALYTICAL LOWER LIMIT OF MASTER FLOW CONTROL.
6	WITHIN +0, –5% OF RATED 100% POWER – WITHIN +0, –5% OF RATED 100% CORE FLOW RATE.

LIMERICK ~~GENERATING~~ STATION
UNITS ~~1 AND 2~~
FINAL SAFETY ANALYSIS REPORT

POWER—FLOW OPERATING MAP
(SHEET 2 of 2) 4 4

FIGURE 14.2-9

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LDCN FS-136

a. Those actions that could be performed by one person
b. Those actions that would have constituted a correct procedure had the initial decision been correct
c. Those actions that are subsequent to the initial operator error and have an effect on the designed operation of the plant, but are not necessarily directly related to the operator error.

Examples of single operator errors are as follows:

a. An increase in power above the established flow control power limits from control rod withdrawal in the specified sequences

b. The selection and complete withdrawal of a single control rod out of sequence

c. The incorrect calibration of an average power range monitor

d. Manual isolation of the main steam lines as a result of operator misinterpretation of an alarm or indicator.

15.0.3.2.1.3 Single Active Failure or Single Operator Error Analysis

a. The undesired action or maloperation of a single active component, or

b. Any single operator error as defined in Section 15.0.3.2.1.2.

15.0.3.3 Core and System Performance

15.0.3.3.1 Introduction

Section 4.4, Thermal and Hydraulic Design, describes the various fuel failure mechanisms. Avoidance of mechanisms a. and b. for incidents of moderate frequency is verified statistically with consideration given to data, calculating, manufacturing, and operating uncertainties. An acceptable criterion was determined to be that 99.9% of the fuel rods in the core would not be expected to experience boiling transition (see Ref 15.0-2). This criterion is met by demonstrating that incidents of moderate frequency do not result in a minimum critical power ratio (MCPR) less than 1.06 for the initial core. The reactor steady state CPR operating limit is derived by determining the decrease in MCPR for the most limiting transient, or accident. All other transients, or accidents, result in smaller MCPR decreases and are not reviewed in depth in this chapter. The MCPR during significant abnormal transient, or accident, is calculated using a transient core heat transfer analysis computer program. The computer program is based upon a multinode, single channel thermal hydraulic model that requires

TABLE 15.0-1

SUMMARY OF INITIAL CORE TRANSIENTS

~~RESULTS SUMMARY OF TRANSIENT EVENTS APPLICABLE TO BWRs~~

PARAGRAPH NO.	FIGURE NO.	DESCRIPTION	MAXIMUM NEUTRON FLUX % NBR	MAXIMUM DOME PRESSURE (psig)	MAXIMUM VESSEL PRESSURE (psig)	MAXIMUM STEAM LINE PRESSURE (psig)	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX % OF INITIAL	ΔCPR [1]	FREQUENCY CATEGORY [2]	NO. OF VALVES 1ST BLOWDOWN	DURATION OF BLOWDOWN (sec)
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1-2	Loss of Feedwater Heater, Manual Flow Control	127.7	1030.0	1069.0	1016.0	119.4	0.16	a	0	0.0
15.1.2	15.1-3	Feedwater Controller Failure, ~~Maximum Demand, 127% Flow [3]~~	~~156.3~~	~~1168~~	~~1194~~	~~1165~~	~~105.0~~	~~0.06~~	~~a~~	~~14~~	8.0
15.1.2	15.1-3A	Feedwater Controller Failure, Maximum Demand, Bypass Off, 127% Flow [3]	216.6	1204	1231	1200	110.2	0.11	b	14	
~~15.1.3~~	~~15.1-4~~	~~Pressure Regulator~~ Failure - Open	104.3	1149.0	1165.0	1148.0	100.3	<0.06 [4]	a	5	3.2
15.1.4	-	Inadvertent Opening of Safety or Relief Valve	See Text						b		
15.1.6	-	Inadvertent RHR Shutdown Cooling Operation	See Text						a		
15.2	-	INCREASE IN REACTOR PRESSURE									
15.2.1	-	Pressure Regulator Failure - Closed	See 15.2.2 and 15.2.3 (Bypass on)						a		
15.2.2	15.2-1	Generator Load Rejection, Trip Scram, Bypass, and RPT - On [5]	178.5	1169	1193	1164	101.2	0.03	a	14	8.0
15.2.3	15.2-2	Generator Load Rejection, Trip Scram, Bypass - Off, RPT - On [5]	222.5	1200	1225.0	1196.0	106.2	0.08	a [6]	14	12.7
15.2.3	15.2-3	Turbine Trip, Trip Scram, and RPT - On	163.3	1174.0	1196.0	1169.0	102.0	<0.16 [4]	a	14	5.8

LDCN FS-1506

LDCN FS-1506

TABLE 15.0-5

REQUIRED OPERATING LIMIT CPR VALUES FOR INITIAL CORE TRANSIENTS

Pressurization Events:

	CPR (Option A)(1)	CPR (Option B)(1)
Load Rejection Without Bypass	1.19	1.11
Turbine Trip Without Bypass	1.17	1.10
Feedwater Controller Failure (127% Flow)	1.17	1.14
Load Rejection	1.14	1.07
Feedwater Controller Failure Without Bypass	1.22	1.19
Feedwater Controller Failure Without Bypass and Without EOC RPT out of service	1.30	1.23

Nonpressurization Events:

	CPR
Rod Withdrawal Error(3)	1.21
Loss of Feedwater Heater	1.22(2)

(1) Includes adjustment factors as specified in Reference 15.0-5.

(2) ~~Required OLCPR using either Option A or Option B adjustment factor regardless of frequency category of the turbine-generator trip events with bypass failure.~~ with operable EOC RPT.

(3) OLCPR value is obtained for the 107% Rod Block setpoint, Control Cell Core analysis.

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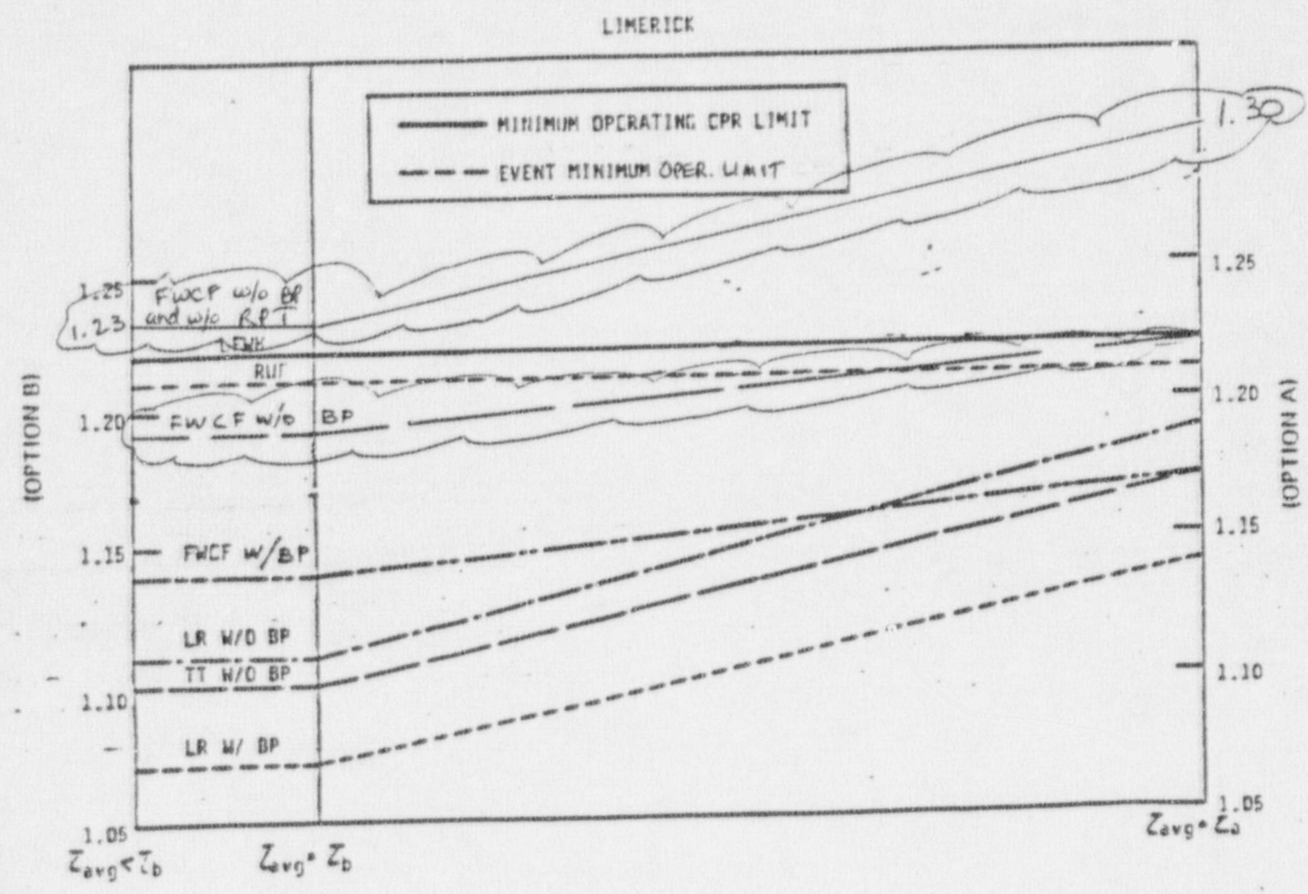

LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

MINIMUM OPERATING CPR LIMITS ~~VS. SCRAM SPEED~~ for Initial Core

FIGURE 15.0-3

REV. 22, 07/83

LDCN FS-1506

Consequently, the nuclear system process barrier pressure limit is not endangered.

The bypass valves subsequently close to re-establish pressure control in the vessel during shutdown. The level would gradually drop to the low level isolation reference point, activating the RCIC/HPCI systems for long-term level control.

15.1.2.3.4 Consideration of Uncertainties

All systems utilized for protection in this transient were assumed to have the most conservative allowable response (e.g., relief setpoints, scram stroke time, and reactivity characteristics). Expected plant behavior is, therefore, expected to lead to a less severe transient.

15.1.2.4 Barrier Performance

As noted above, the consequences of this transient do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.2.5 Radiological Consequences

While the consequence of this transient does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via MSRV operation. Because this activity is contained in the primary containment, there is no exposure to operating personnel. This transient does not result in an uncontrolled release to the environment, so the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established technical specifications and, at the worst, would only result in a small increase in the yearly integrated exposure level.

→ Insert D

15.1.3 PRESSURE REGULATOR FAILURE - OPEN

15.1.3.1 Identification of Causes and Frequency Classification

15.1.3.1.1 Identification of Causes

The total steam flow rate to the main turbine resulting from a pressure regulator malfunction is limited by a maximum flow limiter imposed at the turbine controls. This limiter is set to limit maximum steam flow to approximately 115% NBR.

If either the controlling pressure regulator or the backup regulator fails to the open position, the turbine control valves

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15.1.2.6 Additional Transients Evaluated

An additional transient has been considered for Limerick. The effects of an inoperative turbine bypass system in combination with the Feedwater Controller Failure, Maximum Demand transient described above were determined with an ODYN code analysis. The model, the initial conditions, and the operation of other systems is as described above, except that the operability of the turbine bypass is not taken credit for. The sequence of events with an inoperative bypass system is shown on Table 15.1-3A and the transient parameters are on Fig. 15.1-3A. The MCPR values for this transient are shown in Table 15.0-5.

TABLE 15.1-3A ~~15.1-2-2~~ SEQUENCE OF EVENTS FOR ~~FIGURE 15.1-2-2~~ Feedwater Controller Failure ~~FWCF W/O BP (ODTK)~~ Without Turbine Bypass (Figure 15.1-3A)

Time-sec	Event
0	Initiated simulated failure of 127 percent upper limit on feedwater flow.
27.2	L8 vessel level setpoint trips main turbine and feedwater pumps.
27.2(est)	Reactor scram trip actuated from main turbine stop valve position switches.
27.2	Recirculation Pump Trip (RPT) actuated by stop valve position switches.
27.3	Turbine bypass valves fail to open.
27.3	Main turbine stop valves closed.
27.4	Recirculation pump motor circuit breakers open causing recirculation drive flow to coast-down.
28.6 to 28.8	Relief groups 1 to 3 actuated due to high pressure.

Add this

~~* This table is for GE internal use only.~~

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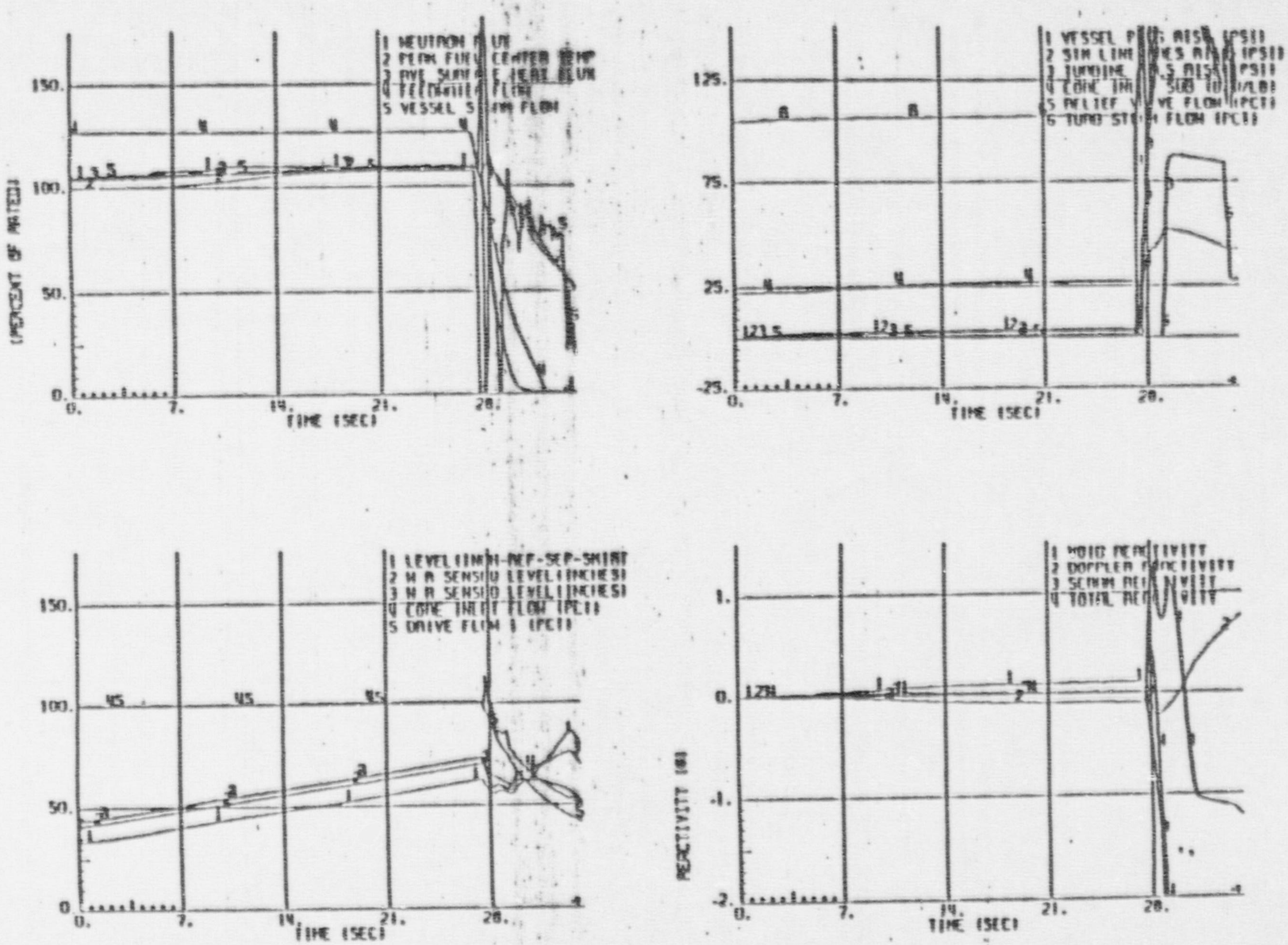

Figure 15.1-3A Feedwater Controller Failure Without Turbine Bypass

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APPENDIX 15.A

EXTENDED OPERATING DOMAIN AND PARTIAL FEEDWATER HEATING ANALYSES

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15.A.1 Introduction

This appendix presents the results of a safety impact evaluation to justify the operation of Limerick Units 1 and 2 in an extended operating domain and with partial feedwater heating for the initial fuel cycle. The extended operating domain consists of two separate regions: the extended load line region and the increased core flow region. The definition and benefits of these modes of operation are described below.

15.A.1.1 Extended Load Line Region

The extended load line region (ELLR) is defined as a power flow operating domain bounded by:

a) 100% rated power,
b) 108% Average Power Range Monitor (APRM) rod block load line which intercepts 100% rated power at 87% core flow, and
c) 100% rated load line.

This ELLR is shown in Figure 15.A-1.

The ELLR improves operational flexibility during power ascension by providing additional operating room above the rated load line. It also provides a lower core flow condition at rated power to allow for flow control reactivity compensation due to fuel burnup during an operating cycle and for fuel cycle economic purposes.

15.A.1.2 Increased Core Flow Region

The increased core flow region (ICFR) is defined as a power flow operating domain bounded by:

a) 100% rated power,
b) 105% rated core flow, and
c) 100% rated core flow.

The ICFR is shown in Figure 15.A-1.

Similar to the ELLR, the ICFR also improves operational flexibility during power ascension by providing additional flow range at rated power to compensate for xenon variations during power changes. The increased flow capability provides reactivity compensation due to fuel burnup during an operating cycle and for providing additional reactivity to extend an operating cycle to improve fuel cycle economics.

15.A.1.3 Partial Feedwater Heating Condition

Partial feedwater heating (PFH) is defined as:

a) Operation with a reduced feedwater heating condition that is 60°F less than the normal feedwater heating temperature at any given power condition.

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b) Partial feedwater heating applies to conditions during an operating cycle when some feedwater heaters are out of service due to equipment malfunctions and to conditions after the end of an operating cycle when some feedwater heaters are intentionally valved out of service to extend an operating cycle.

PFH provides potential operating availability by allowing continued plant operation at rated power with out-of-service feedwater heaters. It also provides fuel cycle economic benefits by cycle extension through reactivity gain from reduced feedwater heating.

15.A.1.4 Combined Operating Condition

The combined ELLR/ICFR extended operating domain is illustrated in Figure 15.A-1. The entire power flow operating map for the Limerick units is therefore bounded by the following:

a) 100% rated power,
b) 105% core flow,
c) 108% APRM rod block load line which intercepts rated power at 87% core flow, and
d) low power and flow recirculation pump and cavitation limitations.

In addition to the benefits described above for each of the individual operating modes, a combined ELLR/ICFR/PFH mode of operation enables the Limerick units to take advantage of the flow control "spectral shift" operation benefit to extend the operating cycle further to improve fuel cycle economics. The "spectral shift" mode of operation involves initial power operation at rated power at a reduced core flow (ELLR) to build up Pu-239 production and better utilize the U-238 fuel followed by subsequent increases of core flow (ICFR) at the end of the fuel cycle to take advantage of the Pu-239 and U-238 fuel reactivity and extend the operating cycle. This, combined with the additional reactivity from the lower feedwater temperature (PFH), provides further improved fuel cycle economics.

15.A.2 Summary

A safety impact evaluation has been performed to investigate the impact of Limerick plant operation with the extended operating domain and PFH on plant safety for the initial fuel cycle. The evaluations included in this appendix are:

a) Transient responses,
b) Overpressure protection,
c) Loss of coolant accident peak cladding temperature,
d) Containment pressure, temperature and hydrodynamic loads, and
e) Thermal-hydraulic stability.

The results of this evaluation show that the previous analyses (prior to the consideration of the extended operating domain and PFH) are bounding for the additional modes of operation defined in Section 15.A.1, except that certain limiting transient responses of Chapter 15 have

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become more severe. The results of the evaluations including the limiting transient responses are provided in this appendix.

15.A.3 Transient Responses

All transient events analyzed in Chapter 15 have been reexamined for the impact of ELLR/ICFR/PFH for the initial fuel cycle. Limiting events have been identified and reanalyzed. The evaluations were performed at 104.5% power, 105% core flow for ICFR; 100% power, 87% core flow for ELLR; and 360°F feedwater temperature (60°F reduction from 420°F), 100% power for PFH. In addition, the most limiting combination of ELLR/ICFR/PFH was evaluated for the limiting transients. The limiting combination was determined by sensitivity studies of these conditions affecting the transient performance. The limiting transients considered the turbine bypass (BP) system to be inoperable and also considered the end-of-cycle recirculation pump trip (RPT) system to be out of service.

Plant heat balance, core coolant hydraulics, and nuclear parameter data were developed and used in the transient analyses. ELLR/ICFR/PFH result in variations in the plant's initial steam void, power distribution, coolant flow rates, and feedwater temperatures which affect the scram, void and doppler reactivity characteristics of the plant during rapid pressurization and coolant temperature/inventory perturbation transient events. The 100°F loss of feedwater heating (LFWH) event is a relatively slow subcooling perturbation transient that has been shown to be insignificantly affected by these initial condition changes. The control Rod Withdrawal Error (RWE) event is mitigated by the Rod Block Monitor (RBM) system. This transient has also been shown to be insignificantly affected by ELLR/ICFR/PFH. The flow-dependent transients have also been shown to be insignificantly affected.

The limiting transient is determined to be the feedwater controller failure without turbine bypass for both the RPT system in and out-of-service cases. These events were analyzed using the computer model of Reference 15.A.10-1. The results of the limiting transients and their associated minimum critical power ratio (MCPR) limits for the initial fuel cycle are presented in Tables 15.A-1 and 15.A-2. The transient performance responses are shown in Figures 15.A-2 to 15.A-5. The details of all the analyzed transients are documented in References 15.A.10-2 to 15.A.10-5.

15.A.4 Overpressure Protection

The limiting transient to satisfy the ASME Code overpressure protection criteria (FSAR Chapter 5) is the main steam isolation valve (MSIV) closure event with secondary (flux) scram. PFH and ELLR result in lower initial steam flow and operating pressure that enables the peak vessel pressure to be lower than the values reported in Chapter 5 for this limiting transient. The ICF analysis produces a higher peak vessel pressure. Its peak pressure of 1273 psig is still well below the ASME Code limit of 1375 psig. The peak pressure results for the initial fuel cycle analyzed cases are presented in Table 15.A-3. The transient responses for the ICF case are shown in Figure 15.A-6.

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15.A.5 Emergency Core Cooling System Performance

The effect of the ELLR, ICFR, and PFH on the loss of coolant accident (LOCA) peak cladding temperature (PCT) calculation discussed in Chapter 6.3 (Emergency Core Cooling System Performance) has been examined for the initial fuel cycle.

When operating in the ELLR, a LOCA initiated from 87% core flow produces a slightly earlier (0.1 second) loss of nucleate boiling in the top portion of the limiting bundle when compared to the 100% core flow condition. However, it does not affect the dryout time of the high power node significantly where the PCT occurs. The reduced initial core flow also has little effect on the reflood phase following the LOCA. This period is dominated by the effect of counter current flow limit (CCFL) at the top of the bundles. Since the core power is not changed, the steam generation rate in the core and the liquid down flow rate through the fuel bundles will be similar. Therefore, the effect of the reduced initial core flow on reflooding time is small. It is concluded that the operation in the ELLR is bounded by the current Chapter 6.3 PCT calculations to satisfy the requirements of 10CFR50.46.

When operating in the ICFR, the LOCA PCT will not be significantly affected. This is because the parameters which most strongly affect the PCT, i.e., high power node boiling transition time and core reflooding time, have been shown to be not significantly affected. Results of the LOCA analysis performed for Limerick show a negligible impact on PCT at the ICF condition for the limiting break when compared to the rated flow condition. PCT changes throughout the remainder of the break spectrum are also negligible and thus will not alter the limiting break. Therefore, operation in ICFR is bounded by the current Chapter 6.3 PCT results.

For operation with PFH, the increased subcooling increases the total system mass and the mass flow rate during a LOCA break at a given vessel pressure. However, the lower initial steam flow with PFH reduces the vessel pressure which results in a net decrease in break flow rate at most times during the accident. The increased total system mass delays the time of lower plenum flashing. The increased system mass and the decreased break flow act together to result in later jet pump, break, and core uncovery times. These combined effects result in a lower PCT for the PFH condition compared to the normal feedwater heating condition.

Therefore, the calculated PCT reported in Chapter 6.3 is bounding for the extended operating domain and partial feedwater heating condition.

15.A.6 Containment Responses

The impact of operation with ELLR, ICFR, and PFH on the containment pressure, temperature and hydrodynamic loads of Chapter 6.2 has been evaluated for the initial fuel cycle.

Operation in the ELLR results in increased subcooling which may lead to higher LOCA blowdown flow rates for certain time periods following the line break. An analysis has been performed to show that the calculated peak drywell and wetwell pressures are bounded by the corresponding values in Chapter 6.2. The peak drywell flow differential pressure (28.6 psid) (down load) is bounded by the design value. A qualitative evaluation shows that all other containment parameters (e.g., drywell and suppression chamber temperatures and maximum allowable leak rates) are bounded by results reported in Chapter 6.2.

Operation with ICF results in lower subcooling that provides improved containment pressure/temperature responses. Operation with PFH results in increased subcooling that may result in a higher LOCA blowdown rate for certain time periods following the line break. Similar to the evaluation for ELLR, an evaluation for ICF and PFH has shown that the calculated peak drywell and wetwell pressures are bounded by the corresponding values reported in Chapter 6.2. The peak drywell flow differential pressure is bounded by the design values. All other containment parameters are bounded by those reported in Chapter 6.2.

All LOCA-related pool swell, condensation oscillation, and chugging loads for the ELLR, ICFR, and PFH are shown to be bounded by the corresponding design loads.

15.A.7 Thermal-Hydraulic Stability

Operation in the ELLR results in operating on a higher control rod line which could potentially reduce stability margin. Operation with PFH results in increased subcooling and power distribution changes that also can potentially reduce stability margin. ICF operation will improve stability margin. In all these conditions, the NRC has completed the generic review of thermal-hydraulic stability of BWR cores and fuel designs contained in General Electric standard application for reactor fuel (GESTAR) (Reference 15.A.10-6). In NRC evaluation reports (References 15.A.10-7 and 15.A.10-8), the NRC concluded that GE fuel design meets stability criteria set forth in General Design Criteria 10 and 12 of 10CFR50, Appendix A, provided that the plant has in place operating procedures and Technical Specifications which are consistent with the recommendation of Reference 15.A.10-9. The NRC approval reports considered operation with ELLR, ICFR, and PFH. Therefore, thermal-hydraulic stability compliance is satisfied for the initial fuel cycle. PECo will remain apprised of any new NRC improvements established in this area.

15.A.8 Fuel and Reactor System Performance

The fuel system performance and reactor internals described in Chapter 4 were examined to determine the impact of the extended operating domain and PFH operation for the initial fuel cycle. It is shown that the fuel design criteria as described in the generic GE fuel licensing report, GESTAR II (Reference 15.A.10-6), are applicable to the ELLR, the ICFR, and PFH.

It is also shown that the reactor internals most affected by ELLR, ICFR, and PFH operation are the core plate, shroud, shroud support, shroud head,

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steam dryer, control rod guide tube, control rod drive housing, and jet pump. The impact of these operating conditions on these components was evaluated. The analysis concluded that the design of these components contains enough margin to handle the increased loading.

15.A.9 Conclusion

The safety impact evaluation has demonstrated that the overpressure protection, LOCA and containment responses, thermal-hydraulic stability, and the fuel and reactor system performance as described in FSAR Chapters 4, 5, and 6 are acceptable in the extended operating domain and partial feedwater heating condition for the initial fuel cycle. For transient performance discussed in Chapter 15, new MCPR results have been determined for this first cycle. Additional details of the extended operating domain and partial feedwater heating evaluations are included in References 15.A.10-2 to 15.A.10-5.

15.A.10 References

15.A.10-1 NEDO-24154A, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," August 1976.

15.A.10-2 NEDC-31139, "General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Limerick Generating Station Unit 1, Cycle 1," April 1986.

15.A.10-3 NEDC-31323, "Increased Core Flow and Partial Feedwater Heating Analysis for Limerick Generating Station Unit 1, Cycle 1," October 1986.

15.A.10-4 NEDC-31577-P, "General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Limerick Generating Station Unit 2, Cycle 1," March 1989.

15.A.10-5 NEDC-31578-P, "Increased Core Flow and Partial Feedwater Heating Analysis for Limerick Generating Station Unit 2, Cycle 1," March 1989.

15.A.10-6 NEDE-24011-P-A-9-US, "General Electric Standard Application for Reactor Fuel," Supplement for United States, September 1988.

15.A.10-7 NRC letter from L. S. Rubenstein to D. Crutchfield, "Safety Evaluation of GE Topical Report NEDE-24011 (GESTAR) Amendment 8," April 17, 1985.

15.A.10-8 C. O. Thomas (NRC) to H. C. Pfefferlen (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011, Revision 0, Amendment 8, 'Thermal Hydraulic Stability Amendment to GESTAR II'," April 24, 1985.

15.A.10-9 "BWR Core Thermal Hydraulic Stability," SIL No. 380 Revision 1, February 10, 1984.

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Table 15.A-1

Limiting Transient Performance Results for ELLR/ICFR/PFH

Transient	Initial Power/Flow (% NBR)	Peak Neutron Flux (% NBR)	Peak Heat Flux (% Initial)	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)	Δ CPR (not adjusted with adder)	Figure
ELLR							
FWCF w/o BP	100/87	188.6	110.9	1191	1221	0.12	15.A-2
FWCF w/o BP w/o RPT	100/87	303.0	119.9	1199	1235	0.19	15.A-3
ELLR/ICFR/PFH							
FWCF w/o BP	100/87	224.7	114.3	1186	1222	0.16	15.A-4
FWCF w/o BP w/o RPT	104.5/105	394.9	122.6	1190	1244	0.24	15.A-5

Note:

FWCF - Feedwater Controller Failure, maximum demand

BP - Turbine Bypass

RPT - EOC Recirculation Pump Trip

ELLR - Extended Load Line Region

ICFR - Increased Core Flow Region

PFH - Partial Feedwater Heating

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Table 15.A-2

Limiting Transient MCPR Results for ELLR/ICFR/PFH

Transient	FSAR		ELLR		ELLR/ICFR/PFH	
	MCPR Operating Limit		MCPR Operating Limit		MCPR Operating Limit	
	Option B	Option A	Option B	Option A	Option B	Option A
FWCF w/o BP	1.19	1.22	1.20	1.23	1.24	1.27
FWCF w/o BP w/o RPT	1.23	1.30	1.23	1.30	1.28	1.36
RWE	1.21	1.21	1.21	1.21	1.21	1.21
LFWH	1.22	1.22	1.22	1.22	1.22	1.22

Note:

FWCF - Feedwater Controller Failure, maximum demand

RWE - Rod Withdrawal Error

LFWH - 100°F Loss of Feedwater Heating

BP - Turbine Bypass

RPT - EOC Recirculation Pump Trip

ELLR - Extended Load Line Region

ICFR - Increased Core Flow Region

PFH - Partial Feedwater Heating

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Table 15.A-3

Overpressure Protection Analysis Results for MSIV Closure (Flux Scram) Transient for ELLR/ICFR/PFH

Initial Power/Flow (%/%)	Maximum Steam Line Pressure (psig)	Maximum Vessel Pressure (psig)	Figure
104.3/100	1227	1260	-
104.3/105	1235	1273	15.A-6
100/87	1215	1250	-

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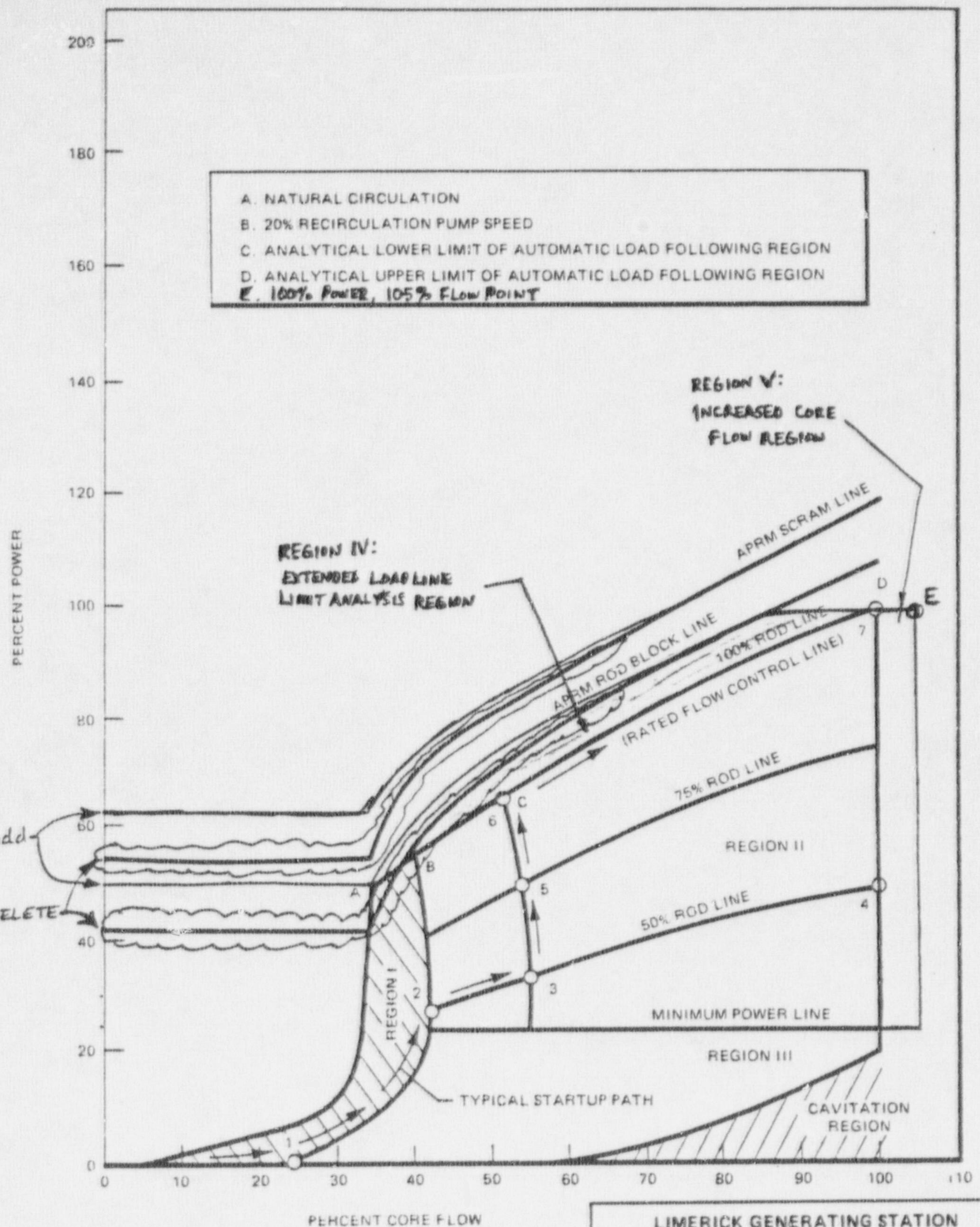

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FINAL SAFETY ANALYSIS REPORT

Extended POWER–FLOW OPERATING MAP

FIGURE 15.A-1

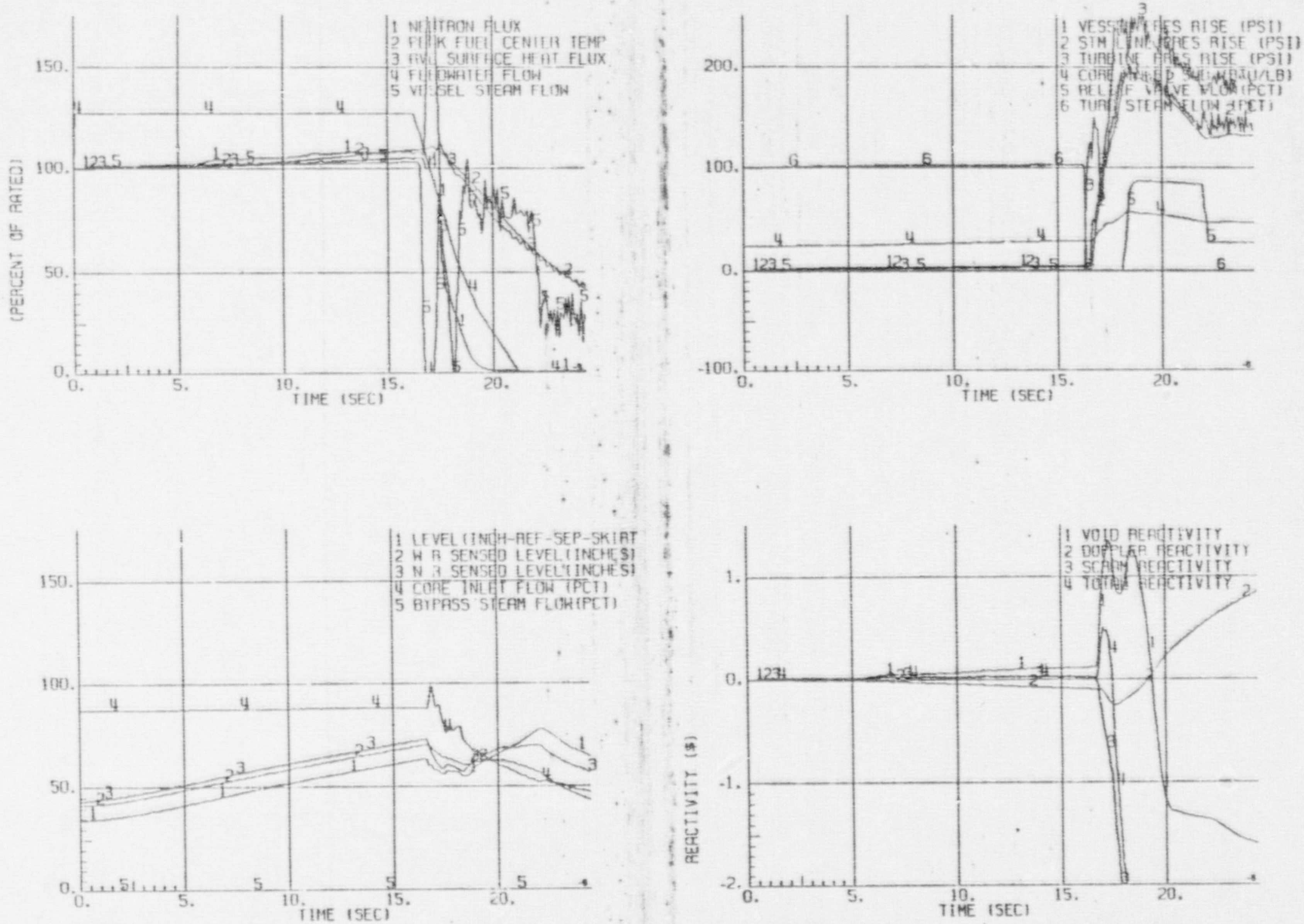

Figure 15.A-2 Feedwater Controller Failure Without Turbine Bypass at 100% power, 87% Flow

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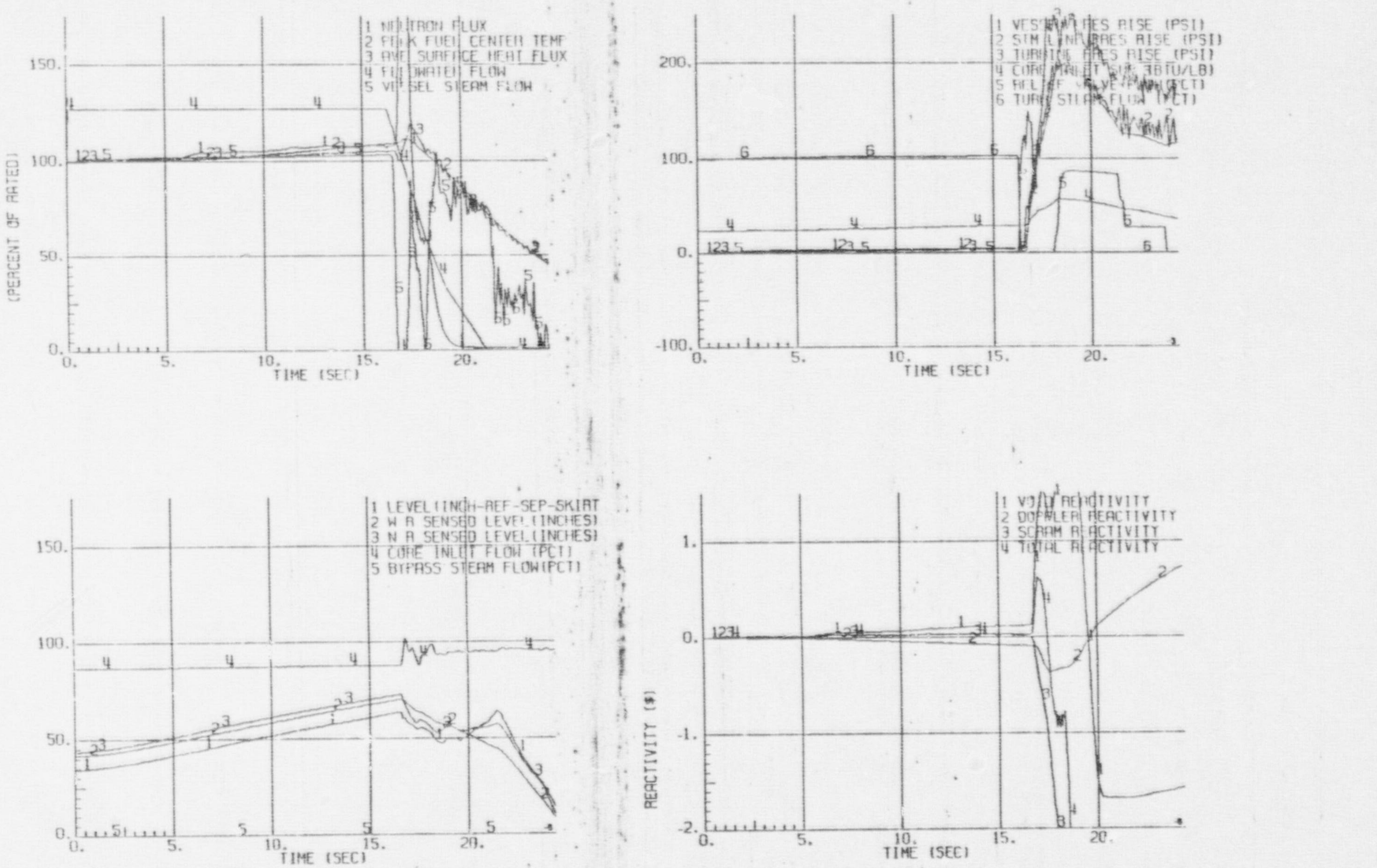

Figure 15.A-3 Feedwater Controller Failure Without Turbine Bypass and Without Recirculation Pump Trip at 100% Power, 87% Flow

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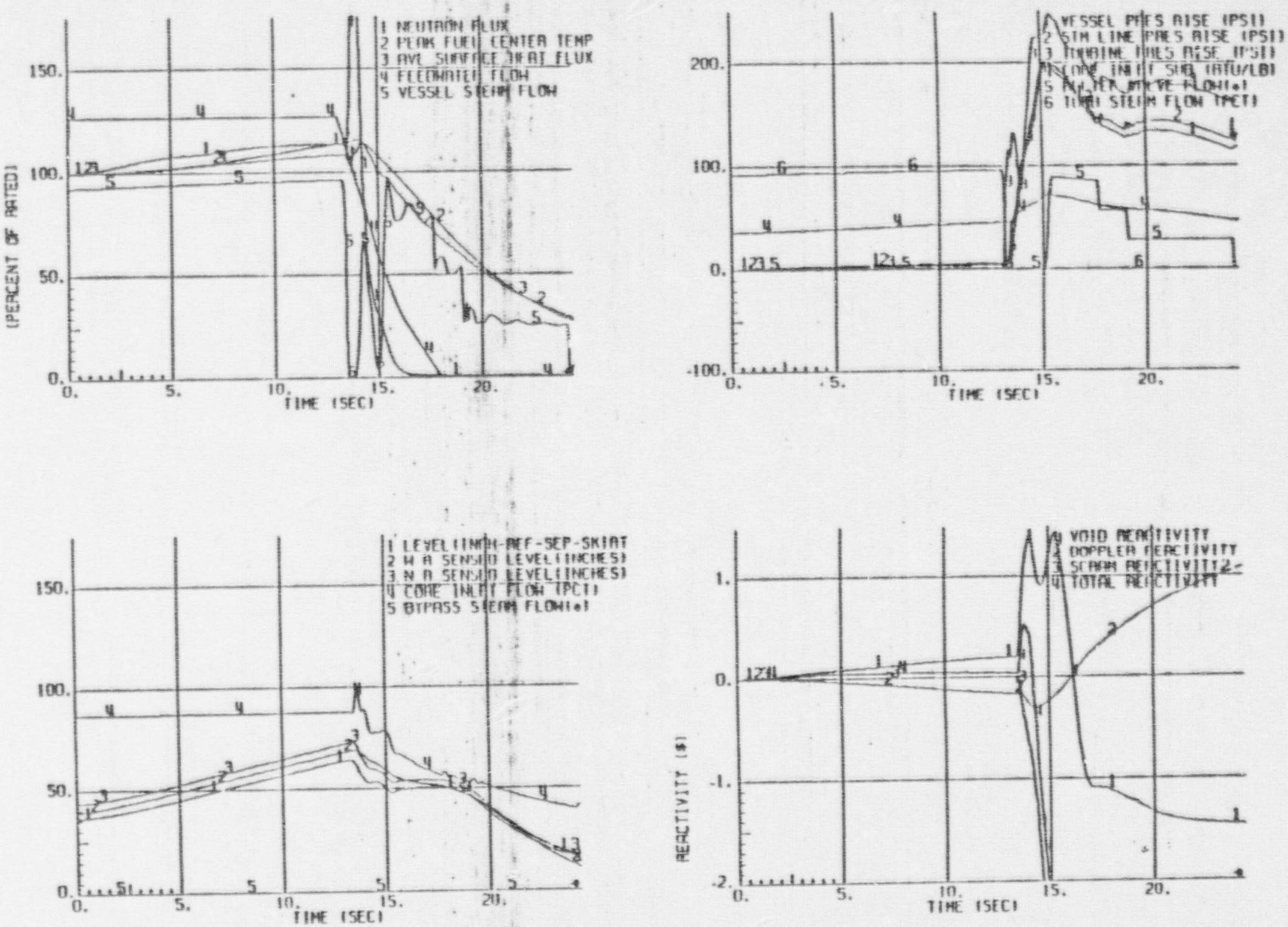

Figure 15.A-4 Feedwater Controller Failure Without Turbine Bypass at 100% power, 87% Flow with $60^{\circ}F$ Feedwater Temperature Reduction

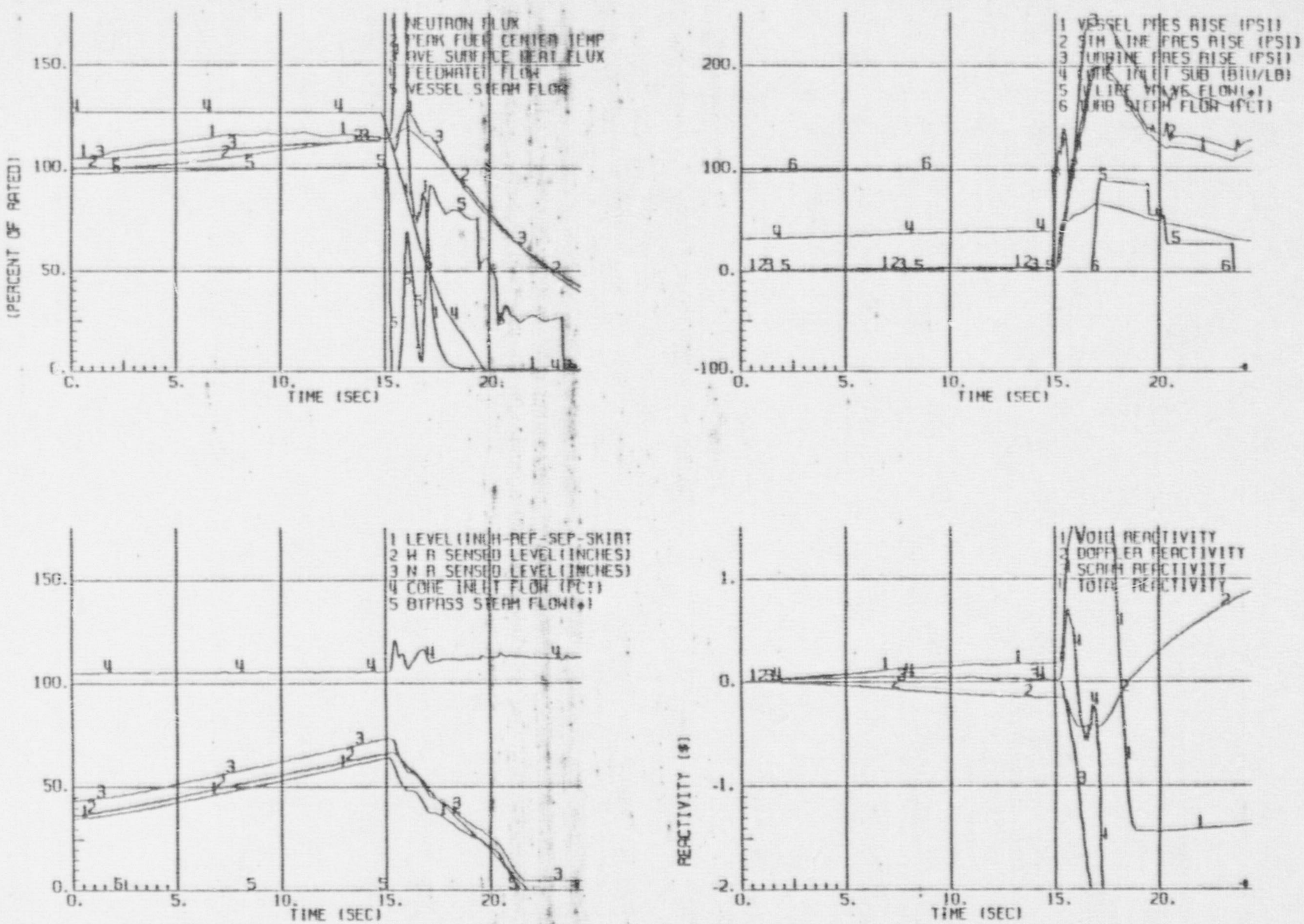

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Figure 15.A-5 Feedwater Controller Failure Without Turbine Bypass and Without Recirculation Pump Trip at 104.5% Power, 105% Flow with 60°F Feedwater Temperature Reduction

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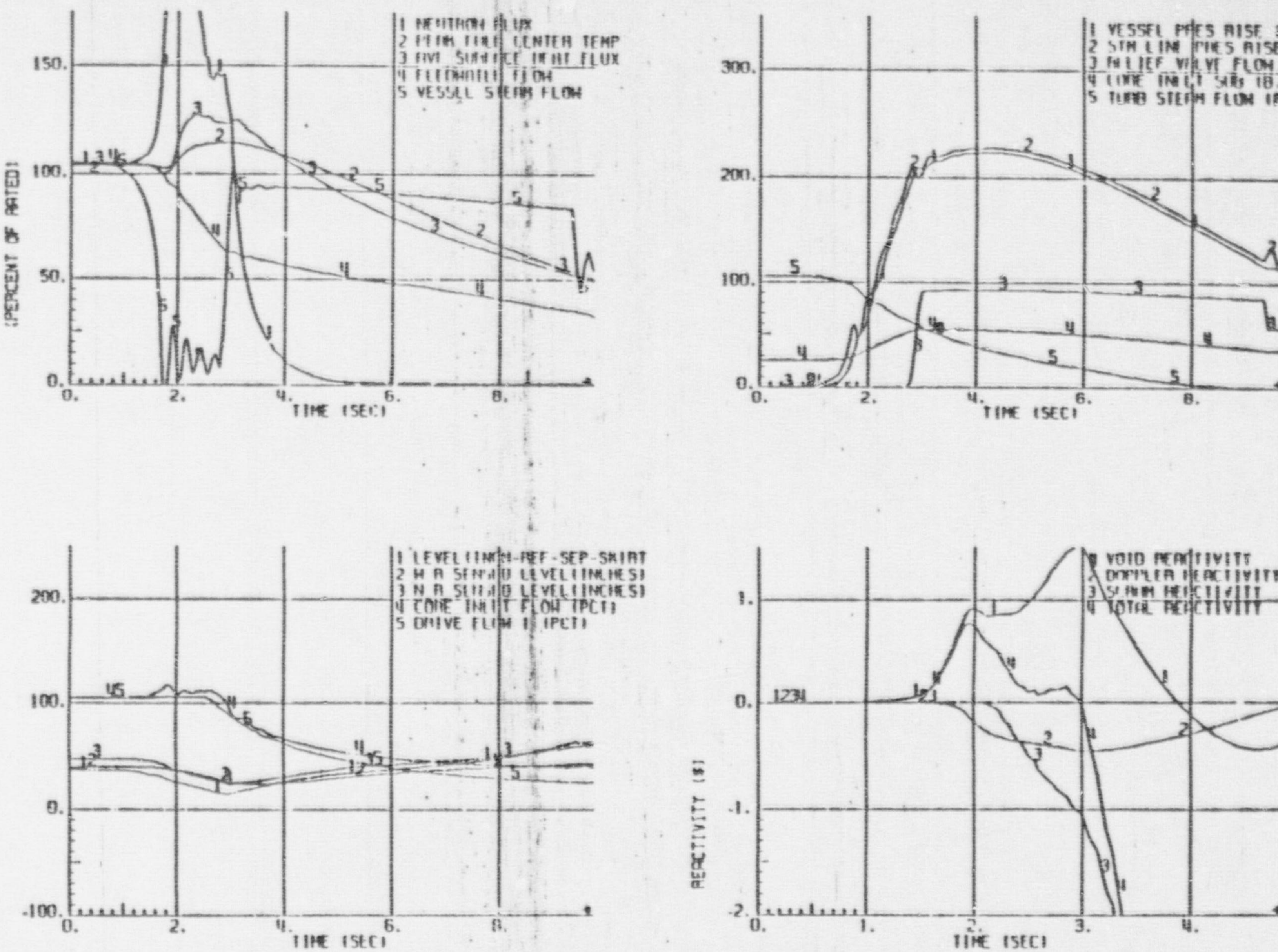

Figure 15.A-6 MSIV Closure, Flux Scram, 104.3% Power, 105% Flow