FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE NUCLEAR POWER PLANT

UNIT I

DOCKET NO. 50-335

LICENSE NO. DPR-67

STARTUP TEST REPORT

DECEMBER, 1976

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This report fulfills the requirement of Regulatory Guide 1.16 which states that a Startup Test Report will be submitted to the NRC within 9 months of initial criticality. Initial criticality was April 22, 1976. Due to a flux distribution problem,completion of our test program has been delayed. Therefore, this report covers only testing through the 50% plateau (see below for description of test program) then discusses the flux distribution problem and resolution. A supplementary report will be submitted on the remainder of the test program as required by Regulatory Guide 1.16.

The Startup Test Program was organized and administered by Florida Power and Light Company (FP&L) personnel assisted by Combustion Engineering (CE) Startup Engineers on-site and home office personnel in Windsor, Connecticut (CE, Windsor). The Startup Test Program consisted of several phases. CE commented on the test results from each phase. Then, the Facility Review Group (FRG) reviewed the results of each phase. Composition of the FRG is defined in our Technical Specifications, Section 6.5.1. Any test results falling outside of acceptance criteria were resolved prior to beginning the next test phase. The test phases were as follows:

(1) Initial Fuel Load

(2) Post Core Hot Functional Tests

(3) Initial Approach to Criticality

(4) Low Power Physics Tests

(5) Escalation to Power Tests - 20% Plateau

(6) Escalation to Power Tests - 50% Plateau

*(7) Escalation to Power Tests - 80% Plateau

*(8) Escalation to Power Tests -100% Plateau

Maximum licensed reactor core power level (100%) is 2560 MWt. The Startup Test Program began March 3, 1976 with the loading of the first fuel assembly into the reactor vessel and was terminated due to the flux distribution anomaly July 10, 1976.

* Not included in this report. Results of these tests and any repeated during our Startup and ascent to power after resolution of the flux distribution problem will be included in our Supplementary Startup Report (s).

1.0 INTRODUCTION AND SUMMARY (Cont.)

1.1 Introduction (Cont.)

Since this unit is identical in core characteristics to other CE 2560 MW_t plants, the testing performed by Calvert Cliffs Unit No. 1 and Millstone Point Unit No. 2 formed a basis for elimination of some physics testing at St. Lucie Unit No. 1.

1.2 Summary

1.2.1 Initial Fuel Load

Fuel loading commenced on March 3, 1976 and was completed on March 11, 1976. A sizable portion of this time was spent in non-productive activities. The largest non-productive time consumption (approximately 15% of total time) was associated with fuel handling equipment malfunction and maintenance. There was a 2 hour delay while inspecting fuel elements A047 and B032, which may have come into physical contact as A047 was being lowered into the core. None of the inspected fuel assemblies showed visible signs of damage. Fuel loading was done 24 hours per day by three crews working 8 hour shifts. Without any delays or equipment problems, an experienced crew could load 12 to 14 fuel assemblies per shift.

1.2.2 Post Core Load Hot Functional Tests

Post Core Load Hot Functional Testing (PCHF) commenced with plant heatup on March 26, 1976. PCHF was completed on April 20, 1976. In addition to the tests required by the FSAR and Regulatory Guides, other tests were performed due to maintenance and previous inability to complete tests. These tests were Reactor Coolant System (RCS) Expansion, RCS and Steam Generator instrument calibration checks, RCS Heat Loss, Pressurizer Controls Functional Test and Main Generator Windage test. All the tests met acceptance criteria with the exception of RCS Flow and CEDM 44 performance testing. RCS flow was measured by ΔP to be slightly (5%) below the required value. The Operating License currently limits power level to 90% of full power and the problem is still under evaluation. CEDM 44 is a full length Control Element Assembly (CEA) in Shutdown Group A. It would not operate properly (withdraw) during cold testing, but did trip properly under all conditions. CEDM 44 has been replaced and was tested upon return to operation after resolution of the flux distribution anomaly.

1.2.3 Initial Approach to Criticality

The initial approach to criticality commenced at 0815 on April 20, 1976. The reactor was declared critical at 0830 on April 22, 1976. The only problems of consequence encountered were with the CEDM control system and the digital data processing system (DDPS). The CEDM control system gave unwarranted "deviation" and "out of sequence" alarms. The DDPS was reading CEA height in error. New values of CEA levels were entered into the DDPS and the situation was corrected. A slow RCS dilution followed CEA withdrawal. Measured RCS soluble boron concentration at criticality was in close agreement with that which was predicted and well within the acceptance criteria.

1.2.4 Low Power Physics Tests

The Low Power Physics Test (LPPT) phase commenced on April 22, 1976. The LPPT phase was completed on April 30, 1976. There were no significant delays or occurrences. Most LPPT measurements were in close agreement with predictions and all were within acceptable limits.

1.2.5 Power Ascension Testing

The Power Ascension Testing began on May 4, 1976. The testing progressed through the completion of the 50% power plateau. Power was increased to 80% before it was decided to withdraw from power ascension testing due to the development of a reactivity anomaly in the form of an unacceptable azimuthal tilt and increased axial peaking. The reactor was shut down on July 10, 1976. A second LPPT program (mini-LPPT) was run in mid-July 1976 to evaluate the flux distribution anomaly. (See Section 7).

2.0 INITIAL FUEL LOAD

At 0410 hours on March 3, 1976, fuel assembly No.1 containing neutron source No.1 was loaded into core location X-11. Fuel loading was completed at 1426 hours on March 11, 1976 when fuel assembly No.217 was loaded into core location V-7.

Table 2.0-1 and Figure 2.0-1 show the fuel loading sequence. Figures 2.0-2 and 2.0-3 show fuel assembly location and CEA location by their respective serial numbers. Table 2.0-2 gives core design characteristics.

Neutron count rate was monitored during loading on four separate detector channels, Wide Range Log Channels B and D plus two temporary detectors. Temporary Detector A in location V-7 and Temporary Detector B in location V-15 were installed prior to core loading. In step 140-B detector B was moved to location D-15. Independent plots of inverse count rate versus the number of fuel assemblies loaded were maintained to ensure the reactor remained subcritical at all times during loading.

Fuel loading was conducted with the spent fuel pool dry, refueling cavity full to the top of the fuel transfer tube flange and the reactor vessel filled to above the vessel nozzles but below the internals support ledge. A refueling boron concentration of ≥1720ppm boron was maintained with shutdown cooling flow through the core in accordance with the Technical Specifications at all times.

No major problems were encountered during fuel loading. Numerous minor problems were encountered with fuel handling equipment, resulting in a total loss of approximately 2 days. These problems were various in nature and all were corrected by plant maintenance personnel under the direction of the vendor representative on site.

Three Reportable Occurances were generated relating to initial core loading and are described as follows and in Licensee Event Reports Nos. 335-76-1, 335-76-2, and 335-76-4.

335-76-1
During initial core loading, the water level in the Refueling Cavity was found to be approx. 2 inches below the top of the Fuel Transfer tube. This was in conflict with spec.-3.9.4.c, which requires no direct access from containment to outside atmosphere incapable of automatic isolation. The immediate corrective action was to suspend core alterations until the level was restored and to require more frequent surveillance of the water level in the Refueling Cavity.

After a thorough investigation, two possible causes for the event were found. The first is that during electrical checkouts on the refueling canal sump pump, more water may have been pumped than had been realized. The corrective action was to cancel the blue tag (startup test) clearance on the pump and clearance tag the pump to the Nuclear Plant Supervisor. The second possible cause is an incorrect valve lineup, causing a gradual decrease in the water level.

During subsequent refilling operations, this lineup error may have been corrected. The corrective action was to re-verify the valve lineups affecting refueling cavity water level. The water that was lost was identified to have ended up in the Equipment and Chemical Drain Tanks. Due to the limited number of pathways to transmit this water, the two above situations are considered the only probable causes.

335-76-2
During initial core loading, the spent fuel crane overload interlock setpoint was found to have drifted above 2000 pounds. This is contrary to the surveillance requirements of Technical Specification 4.9.7. The immediate corrective action was to stop core loading.

The cause of the occurrence was malfunction of the crane interlock. The interlock was malfunctioning such that it would occasionally prevent a fuel assembly from being lifted. Conversely, when tested, the interlock would occasionally permit loads in excess of 2000 pounds to be lifted. This indicated an intermittent condition. The immediate action was to propose that the requirement for the interlock be temporarily suspended to allow fuel loading to continue. The proposal included administratively limiting the permissible crane load to 2000 pounds by limiting the objects which could be lifted to a fuel element assembly or a control element assembly, neither of which weighs in excess of 2000 pounds. The temporary suspension, effective until midnight, March 19, was granted on March 5, 1976, by letter from the NRC Division of Project Management. The malfunction was later found to be due to improper assembly of the load sensor which caused inconsistent operation. The interlock has been repaired and functions properly.

335-76-4
During initial core loading, a containment purge fan was started and containment pressure became subatmospheric causing a Containment Vacuum Relief Valve to open. This was in conflict with the wording of Technical Specification 3.9.4 which requires that, during refueling operations, there be no direct access from containment to the outside atmosphere which is incapable of automatic isolation. In order to protect the containment structure from excessive vacuum during all modes of operation, the Containment Vacuum Relief Valves, by design, do not receive a Containment Isolation Signal (CIS). They do close again when containment pressure approaches atmospheric; however, they are not capable of being closed automatically by CIS. The immediate corrective action was to suspend core loading until the valves were closed.

Specification 3.9.4 does not consider the unique function of the Containment Vacuum Relief Valves. Even though containment integrity is not exactly as described in the specification when one of these valves opens, it should be noted that the valves and their associated check valves do close automatically when containment pressure approaches atmospheric. Thus, actuation of the relief valves for the purpose of performing their design function (protection of the containment from excessive vacuum) does not violate the concept of containment integrity because there is no outflow of air from containment to the outside atmosphere through the relief valves. A request for license ammendment (Appendix A, Technical Specifications) to allow for correct operation of the Containment Vacuum Relief Valves has been submitted and is awaiting NRC action.

TABLE 2.0-1

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
1	B009	Neutron Source 1	X-11
2	C031	-	Y-10
3	C032	-	Y-12
4	C105	40	X-13
5	B067	-	W-13
6	A025	72	W-11
7	B057	-	W-9
8	C-104	31	X-9
9	C-007	-	Y-8
10	C-114	-	X-7
11	A002	25	W-7
12	A010	30	V-9
13	B017	-	V-11
14	A038	39	V-13
15	A004	47	W-15
16	C106	-	X-15
17	C023	-	Y-14
18	C016	-	X-16
19	B020	-	W-16
20	A015	51	V-16
21	B069	-	T-16
22	A048	46	T-15
23	B010	-	T-13
24	A049	C	T-11

TABLE 2.0-1 (CONT'D.)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
25	B001	-	T-9
26	A008	24	T-7
27	B012	-	T-6
28	A013	18	V-6
29	B080	-	W-6
30	C027	-	X-6
31	C008	-	X-5
32	C212	14	W-5
33	B046	-	V-5
34	A018	F	T-5
35	B013	-	S-5
36	A054	17	S-6
37	B045	-	S-7
38	A055	-	S-9
39	B026	-	S-11
40	A061	-	S-13
41	B035	-	S-15
42	A053	50	S-16
43	B071	-	S-17
44	A019	H	T-17
45	B037	-	V-17
46	C208	55	W-17
47	C001	-	X-17
48	A031	54	R-17

TABLE 2.0-1 (CONT'D.)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
49	B007	–	R-16
50	A037	45	R-15
51	B004	–	R-13
52	A030	34	R-11
53	B038	–	R-9
54	A059	23	R-7
55	B040	–	R-6
56	A056	13	R-5
57	B064	–	N-5
58	A064	–	N-6
59	B056	–	N-7
60	A034	29	N-9
61	B051	–	N-11
62	A057	38	N-13
63	B065	–	N-15
64	A060	–	N-16
65	B077	–	N-17
66	A043	D	L-17
67	B044	–	L-16
68	A068	44	L-15
69	B054	–	L-13
70	A032	33	L-11
71	B025	–	L-9
72	A041	22	L-7

TABLE 2.0-1 (CONT'D)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
73	B053	-	L-6
74	A044	A	L-5
75	B059	-	J-5
76	A058	-	J-6
77	B058	-	J-7
78	A050	28	J-9
79	B055	-	J-11
80	A045	37	J-13
81	B028	-	J-15
82	A063	-	J-16
83	B061	-	J-17
84	A042	53	G-17
85	B031	-	G-16
86	A051	43	G-15
87	B033	-	G-13
88	A052	32	G-11
89	B070	-	G-9
90	A039	21	G-7
91	B005	-	G-6
92	A035	12	G-5
93	B052	-	F-5
94	A065	16	F-6
95	B072	-	F-7
96	A067	-	F-9

TABLE 2.0-1 (CONT'D.)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
97	B011	–	F-11
98	A066	–	F-13
99	B015	–	F-15
100	A069	49	F-16
101	B063	–	F-17
102	A062	G	E-17
103	B016	–	E-16
104	A036	42	E-15
105	B003	–	E-13
106	A012	B	E-11
107	B032	–	E-9
108	A047	20	E-7
109	B002	–	E-6
110	A027	E	E-5
111	B024	–	D-5
112	A029	15	D-6
113	B014	–	D-7
114	A046	27	D-9
115	B048	–	D-11
116	A007	36	D-13
117	A017	48	D-16
118	B021	–	D-17
119	C206	52	C-17
120	B064	–	C-16

TABLE 2.0-1 (CONT'D.)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
121	A001	41	C-15
122	B075	-	C-13
123	A023	71	C-11
124	B022	-	C-9
125	A003	19	C-7
126	B027	-	C-6
127	C209	11	C-5
128	C003	-	B-5
129	C017	-	B-6
130	C108	-	B-7
131	C115	26	B-9
132	B050	NEUTRON SOURCE 2	B-11
133	C101	35	B-13
134	C113	-	B-15
135	C037	-	B-16
136	C009	-	B-17
137	C006	-	A-14
138	C026	-	A-12
139	C018	-	A-10
140	C010	-	A-8
141	B068	-	V-15
142	C012	-	W-4
143	C202	67	V-4
144	B043	-	T-4

TABLE 2.0-1 (CONT'D.)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
145	A014	10	S-4
146	B041	-	R-4
147	A040	09	N-4
148	B046	-	L-4
149	A011	08	J-4
150	B060	-	G-4
151	A016	07	F-4
152	B030	-	E-4
153	C203	66	D-4
154	C019	-	C-4
155	C029	-	D-3
156	C207	03	E-3
157	B066	-	F-3
158	A033	04	G-3
159	B047	-	J-3
160	A024	70	L-3
161	B029	-	N-3
162	A022	05	R-3
163	B074	-	S-3
164	C210	06	T-3
165	C015	-	V-3
166	C002	-	T-2
167	C035	-	S-2
168	C102	-	R-2

TABLE 2.0-1 (CONT'D.)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
169	C116	02	N-2
170	B079	-	L-2
171	C110	01	J-2
172	C103	-	G-2
173	C040	-	F-2
174	C014	-	E-2
175	C005	-	H-1
176	C030	-	K-1
177	C011	-	M-1
178	C021	-	P-1
179	C034	-	W-18
180	C205	69	V-18
181	B049	-	T-18
182	A020	59	S-18
183	B078	-	R-18
184	A009	58	N-18
185	B006	-	L-18
186	A006	57	J-18
187	B018	-	G-18
188	A021	56	F-18
189	B019	-	E-18
190	C201	68	D-18
191	C020	-	C-18
192	C038	-	D-19

TABLE 2.0-1 (CONT'D.)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
193	C211	60	E-19
194	B034	-	F-19
195	A005	61	G-19
196	B076	-	J-19
197	A026	73	L-19
198	B008	-	N-19
199	A028	62	R-19
200	B039	-	S-19
201	C204	63	T-19
202	C036	-	V-19
203	C013	-	T-20
204	C033	-	S-20
205	C109	-	R-20
206	C111	65	N-20
207	B023	-	L-20
208	C112	64	J-20
209	C107	-	G-20
210	C039	-	F-20
211	C024	-	E-20
212	C022	-	H-21
213	C025	-	K-21
214	C028	-	M-21
215	C004	-	P-21
216	B073	-	D-15
217	B042	-	V-7

TABLE 2.0-2

FIRST CYCLE CORE DESIGN CHARACTERISTICS

Nuclear Characteristics

Fuel Management	3-Batch, Mixed Central Zone
Average First Cycle Burnup, MWd/MTU	15,400
U-235 Enrichment, w/o	
Batch A (69 assemblies)	1.93
Batch B (80 assemblies)	2.33
Batch C (68 assemblies)	2.82

Mechanical Characteristics

Fuel Assemblies

Batch	No. of Assemblies	Fuel Rods No./Assy.	Poison Rods No./Assy.	Poison Rods No./Batch
A	69	176	0	0
B	80	164	12	960
C	40	176	0	0
C-(Low Concentration B_4C loading)	12	164	12	144
C+(High Concentration B_4C loading)	16	164	12	192
	217			1296

Fuel Rod Array, Square	14 x 14
Fuel Rod Pitch, Inches	0.580
Spacer Grid	
Type	Leaf Spring
Material	Zircaloy-4
Number per Assembly	8
Retention Grid	
Type	Leaf Spring
Material	Inconel
Number per Assembly	1

TABLE 2.0-2 (Cont'd)

Weight of Contained Uraniun, kg U - per assembly	
Batch A	395
Batch B	368
Batch C (poisoned)	368
Batch C (unpoisoned)	395

Outside Dimensions

Fuel Rod to Fuel Rod, Inches 7.980 x 7.980

Fuel Rod	BATCH A	BATCH B	BATCH C
Fuel Material (Sintered Pellets)	UO_2	UO_2	UO_2
Pellet Diameter, inches	0.3805	0.3795	0.3765
Pellet Dish Depth, inches	0.029	0.015	0.029
Pellet Dish Diameter, Inches	0.2725	0.2915	0.2685
Pellet Length, inches	0.450	0.650	0.450
Pellet Density, g/cc	10.41	10.193	10.41
Pellet Theoretical Density, g/cc	10.96	10.96	10.96
Pellet Density (% theoretical)	95.0	93.0	95.0
Stack Height Density, g/cc	10.05	10.05	10.05
Clad Material	Zircaloy-4		
Clad ID, inches	0.388	0.388	0.384
Clad OD, (nominal) inches	0.440	0.440	0.440
Clad Thickness, (nominal) inches	0.026	0.026	0.028
Diametral Cap, (cold, nominal), inches	0.0075	0.0085	0.0075
Active Length, inches	136.7	136.7	136.7
Plenum Length, inches	8.6	8.6	8.6

Burnable Poison Rod	
Active Length, inches	122.7
Material	$B_4C-Al_2O_3$
Pellet Diameter, inches	0.376
Clad Material	Zircaloy-4
Clad ID, inches	0.388
Clad OD, inches	0.440
Clad Thickness, (nominal) inches	0.026
Diametral Gap, (cold, nominal), inches	0.012

Control Element Assembly (CEA)	Full Length	Part Length
Number	73	8
Number of Absorber Elements per Assembly	5	5
Type	Cylindrical Rods	Cylindrical Rods
Clad Material	Inconel 625	Inconel 625
Clad Thickness, inches	0.040	0.040
Clad OD, inches	0.948	0.948
Poison Material	(1)	(1)
Total Element Length	161	161

(1) Poison material is primarily $B_4C-Al_2O_3$
Several CEA's finger's have a combination of Al_2O_3 and $B_4C-Al_2O_3$

TABLE 2.0-2 (Cont'd)

Control Element Drive Mechanisms (CEDM)	Single CEA	Dual CEA	PLCEA
Number of drive mechanism	49	12	8
Stroke, inches	137	137	137
Speed, inches per minute	40	20	30
Drop time, seconds (90% insertion)	2.5	2.5	N.A.

Core Arrangement	
Number of Fuel Assemblies in Core, Total	217
Number of CEA's	81
Number of Fuel Rods	36,896
Number of Poison Rods	1,296
CEA Pitch, Min., inches	11.57
Spacing between Fuel Assemblies, Fuel Rod Surface to Surface, inches	0.200
Spacing, Outer Fuel Rod Surface to Core Shroud, inches	0.204
Hydraulic Diameter, Nominal Channel, feet	0.0444
Total Flow Area (excluding Guide Tubes) (Sq. Ft.)	53.5
Total Core Area, sq. ft.	101.1
Core Equivalent Diameter, inches	136
Core Circumscribed diameter, inches	142.5
Core Volume, liters	32,600
Total Fuel Loading, kg U	82,850
Total Fuel Weight, pounds UO_2	207,200
Total Weight of Zircaloy Clad, pounds	44,700
Total Heat Transfer Area, sq. ft.	48,420
Fuel Volume (including pellet dished ends), cu. ft.	330.2

Florida Power & Light Company
REACTOR FUEL LOCATION
St. Lucie Plant Unit No. 1

N

FIGURE 2.0-1
CORE LOADING SEQUENCE

Y X W V T S R P N M L K J H G F E D C B A

21							215	214	213	212							
20					203	204	205	206	207	208	209	210	211				
19				202	201	200	199	198	197	196	195	194	193	192			
18			179	180	181	182	183	184	185	186	187	188	189	190	191		
17		47	46	45	44	43	48	65	66	83	84	101	102	118	119	136	
16		18	19	20	21	42	49	64	67	82	85	100	103	117	120	135	
15		16	15	141	22	41	50	63	68	81	86	99	104	216	121	134	
14	17																137
13		4	5	14	23	40	51	62	69	80	87	98	105	116	122	133	
12	3																138
11		1	6	13	24	39	52	61	70	79	88	97	106	115	123	132	
10	2																139
9		8	7	12	25	38	53	60	71	78	89	96	107	114	124	131	
8	9																140
7		10	11	217	26	37	54	59	72	77	90	95	108	113	125	130	
6		30	29	28	27	36	55	58	73	76	91	94	109	112	126	129	
5		31	32	33	34	35	56	57	74	75	92	93	110	111	127	128	
4			142	143	144	145	146	147	148	149	150	151	152	153	154		
3				165	164	163	162	161	160	159	158	157	156	155			
2					166	167	168	169	170	171	172	173	174				
1							178	177	176	175							

Florida Power & Light Company
REACTOR FUEL LOCATION
St. Lucie Plant Unit No. 1

N

FIGURE 2.0-2
FUEL ASSEMBLY SERIAL NUMBERS AND CORE LOCATIONS

Y X W V T S R P N M L K J H G F E D C B A

21							C004	C028	C025	C022							
20					C013	C033	C109	C111	B023	C112	C107	C039	C024				
19				C036	C204	B039	A028	B008	A026	B076	A005	B034	C211	C038			
18			C034	C205	B049	A020	B078	A009	B006	A006	B018	A021	B019	C201	C020		
17		C001	C208	B037	A019	B071	A031	B077	A043	B061	A042	B063	A062	B021	C206	C009	
16		C016	B020	A015	B069	A053	B007	A060	B044	A063	B031	A069	B016	A 017	B 062	C037	
15		C106	A004	B068	A048	B035	A037	B065	A068	B028	A051	B015	A036	B073	A001	C113	
14	C023																C006
13		C105	B067	A038	B010	A061	B004	A057	B054	A045	B033	A066	B003	A007	B075	C101	
12	C032																C026
11		B009	A025	B017	A049	B026	A030	B051	A032	B055	A052	B011	A012	B048	A023	B050	
10																	C018
9	C031																
		C104	B057	A010	B001	A055	B038	A034	B025	A050	B070	A067	B032	A046	B022	C115	
8	C007																C010
7		C114	A002	B042	A008	B045	A0 59	B056	A041	B058	A039	B072	A047	B014	A003	C108	
6		C027	B080	A013	B012	A054	B040	A064	B053	A058	B005	A065	B002	A029	B027	C017	
5		C008	C212	B046	A018	B013	A056	B064	A044	B059	A035	B052	A027	B024	C209	C003	
4			C012	C202	B043	A014	B041	A040	B036	A011	B060	A016	B030	C203	C019		
3				C015	C210	B074	A022	B029	A024	B047	A033	B066	C207	C029			
2					C002	C035	C102	C116	B079	C110	C103	C040	C014				
1							C021	C011	C030	C005							

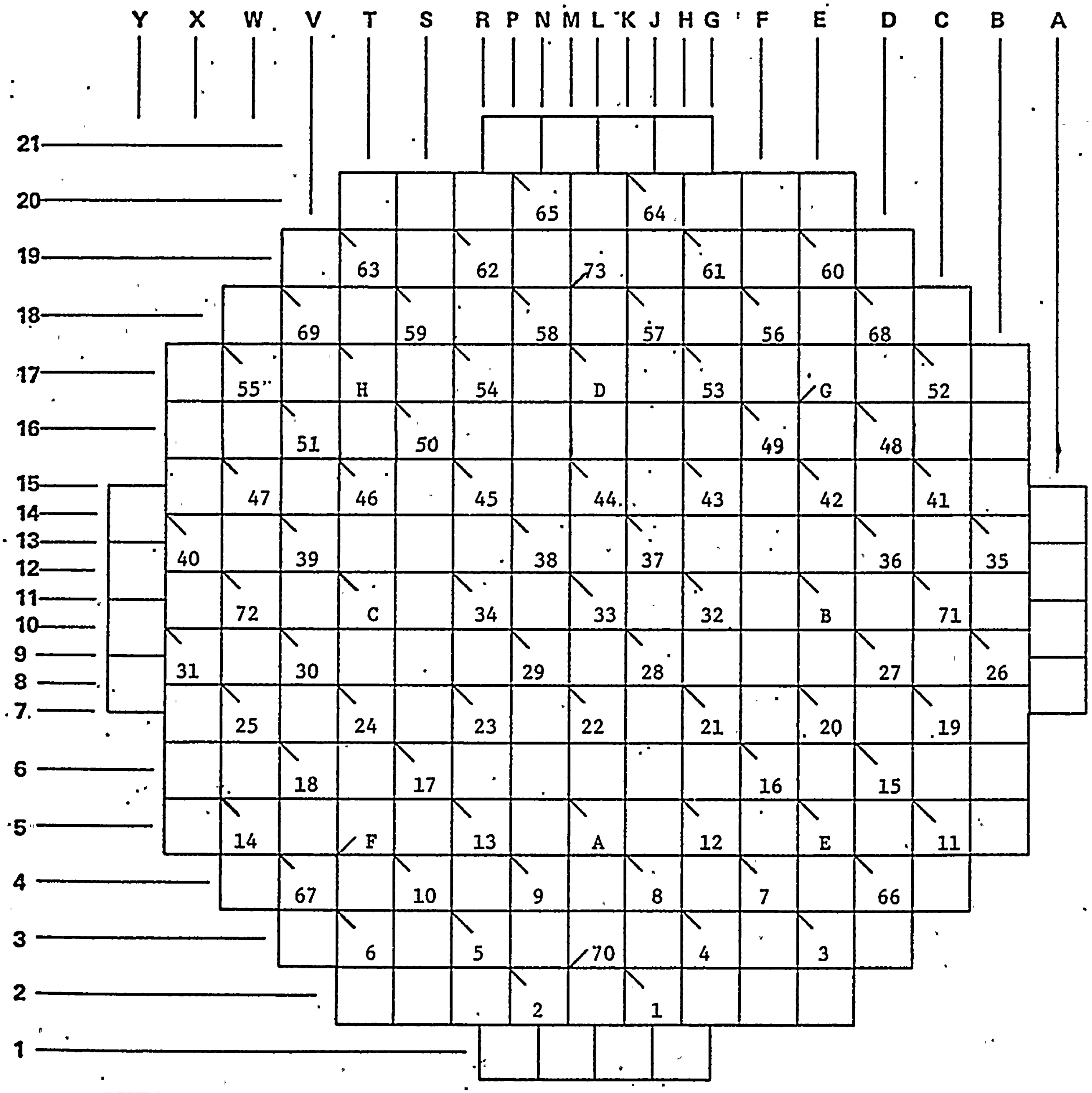
Florida Power & Light Company

REACTOR FUEL LOCATION

St. Lucie Plant Unit No. 1

Figure 2.0-3

CONTROL ELEMENT ASSEMBLY SERIAL NUMBERS AND CORE LOCATIONS



Normal CEA orientation: Serial # on NW web

Exception: CEA's 70, 73, F and G will have their serial # on SW web

3.0 POST CORE HOT FUNCTIONAL TESTS (PCHF)

Several of the tests required prior to initial criticality require installation of the fuel and all reactor internals as a prerequisite. These tests (Post Core Hot Functional Tests) are conducted after initial fuel loading and complete the prerequisites for initial criticality. A list of the required tests follows:

(1) Mechanical and instrument tests on Control Element Drive Mechanisms (CEDM) and Control Element Assembly (CEA) position indicators. This includes rod drop time measurements (cold and hot) under various Reactor Coolant System Flow conditions;

(2) Reactor protective trip circuit and manual scram tests;

(3) Final leak tests of RCS;

(4) Primary and Secondary Water Chemistry;

(5) Neutron response check of source range monitors;

(6) Mechanical and electrical tests of incore instrumentation, including incore detector resistance readings and testing of moveable incore instruments;

(7) RCS flow determination tests;

(8) Pressurizer effectiveness tests; and

(9) Vibration monitoring per Regulatory Guide 1.20

The above tests except Items (2), (5), (8) and (9) are described in Sections 3.1 through 3.5. The instrumentation portions of Item (2) Reactor trip circuit were performed as a prerequisite of the Initial Core Loading procedure and actual CEA trips and manual scram tests were included in the CEDM testing of Item (1). In addition, this testing was repeated as a prerequisite to the Initial Approach to Criticality (IAC). Item (5) Neutron Response Check of Source Range Monitors was performed as a prerequisite to Core Loading and functionally checked during IAC. Item (8) Pressurizer Effectiveness is discussed with the Pressurizer Controls Functional Test, Item 3, listed below. Item 9 Vibration Monitoring for this plant was completed during pre-core load hot functional testing. We are taking internal vibration measurements with installed equipment, but these are not the same type of measurements as discussed in Regulatory Guide 1.20.

In addition, those items or systems which required maintenance or had testing deferred from pre-core hot functional tests were tested during PCHF. This included:

(1) Taking measurements of RCS expansion;

3.0 POST CORE HOT FUNCTIONAL TESTS (PCHF) (Cont.)

(2) RCS and Steam Generator instrument calibration checks;

(3) Pressurizer Controls functional test;

(4) Checking Main Generator air flow and

(5) RCS heat loss test.

The above tests are described in Section 3.6 through 3.10.

All test results met their acceptance criteria with the exception of RCS flow rate and the performance of CEDM 44. After consideration of instrument and measurement errors and uncertainties, RCS flow rate was slightly (5%) below that required by Technical Specifications. An Interim Operating License change allowed operation up to 60% rated thermal power. Further analysis resulted in an Operating License change allowing operation up to 90% rated thermal power. Flow was later evaluated using a plant power calorimetric approach and this indicated RCS flow is, in fact, above the required value. As of December, 1976, this subject has not been fully resolved.

CEDM 44 is a shutdown rod in Group A. Initial performance testing revealed that this CEA could not be withdrawn at cold conditions. Further testing demonstrated that this CEA functioned satisfactorily at hot (operating) conditions. A Technical Specification change deleting permission to go critical below 515°F RCS temperature for testing was approved, allowing criticality and further testing. Considerable extra testing was done and evaluated. Since all testing indicated proper performance at elevated temperatures (> 400°F) and the CEDM never failed to trip, FP&L did not feel the CEDM required replacement. However, in compliance with License Amendment 4, this CEDM was replaced during the shutdown for resolution of the power distribution anomaly (See Section 7.0). It has been retested satisfactorily as applicable during startup and return to the Power Ascension Test Program. A request for license amendment will be submitted requesting reapproval of the exception allowing criticality below 515°F for testing.

3.1 CEDM/CEA Performance Tests

3.1.1 Purpose

The Control Element Drive Mechanism/Control Element Assembly (CEDM/CEA) performance tests were performed to accomplish the following objectives:

(1) To demonstrate proper functioning of the CEA's and CEDM's under various Reactor Coolant System (RCS) temperature, pressure and flow conditions.

(2) To provide measured CEA withdrawl, insertion and drop time data which will serve as comparison standards for future performance tests.

(3) To perform a check of the position indication system and to establish proper functioning of the CEA operating and interlock lights.

(4) To verify proper operation of the upper gripper coil power supply paralleling switches.

(5) Verification that PLCEA's (part length CEA's) do not drop.

3.1.2 Test Results

The CEDM/CEA performance tests were conducted at RCS temperatures and pressures of 260°F/475 psia and 532°F/2250 psia. These tests consisted of:

(1) Verification that all single, dual, and non-tripping coil current traces indicate proper operation.

(2) A check of CEDM/CEA withdrawl speed.

(3) Timing of rod drops from full out to 90% insertion of all regulating and shutdown CEA's under various RCS flow conditions.

(4) Additonal drops of the fastest and slowest CEA's.

(5) A check of all CEDM/CEA position indication, operating and interlock lights.

3.1 CEDM/CEA Performance Tests (Cont'd)

3.1.2 Test Results (Cont'd)

(6) A check of the upper gripper coil paralleling switch circuit for regulating and shutdown CEA's.

CEDM withdrawal and insertion traces were analyzed and adjustments were made to Coil Power Programmers during the test to ensure acceptable operation of the CEDM's. Discrepancies in rod position indication were corrected by adjusting computer setpoints and programming and replacing defective reed switch stacks.

The results of the CEDM drop time test were found to be acceptable and are given in Table 3.1-1.

3.1.3 Conclusions

The results of the CEA/CEDM test were acceptable with the exception of CEDM 44 during cold testing. CEDM 44 had an apparent defective lower gripper coil or latch and would not work consistently during cold testing. CEDM 44 was replaced and testing completed with all results evaluated to be acceptable. (See Section 7). This testing demonstrated proper operation of the CEDM's and verified they would perform their control and shutdown functions.

TABLE 3.1-1
ROD DROP TIME RESULTS

260°F/475 psia

Drop Time to 90% Full Insertion

			3 pump flow	2 pump flow	0 pump flow
Fastest	#1	46	2.08 sec.	1.90 sec.	2.00 sec.
	#2	8	2.15 sec.	1.96 sec.	1.90 sec.
Slowest	#1	51	2.36 sec.	2.24 sec.	2.17 sec.
	#2	69	2.44 sec.	2.35 sec.	2.24 sec.

532°F/2250 psia

Drop Time to 90% Full Insertion (Full Flow) -Specification $\leq$3.3 Seconds

Fastest CEA	#1	46	2.02 sec.
	#2	9	2.08 sec.
Slowest CEA	#1	6	2.36 sec.
	#2	69	2.36 sec.

Drop Time to 90% Full Insertion (Zero Flow)

Fastest CEA	#1	46	1.71 sec.
	#2	9	1.74 sec.
Slowest CEA	#1	61	2.07 sec.
	#2	69	2.07 sec.

3.2 Reactor Coolant System, Pump Flow and Coastdown Test

3.2.1 Purpose

This test was conducted to determine the following Reactor Coolant System (RCS) characteristics:

(1) Reactor Coolant System flow rates and pressure drops around the reactor coolant system.

(2) Reactor Coolant Pump Coastdown flow characteristics.

(3) Reactor Coolant System flow input to Reactor Protection System (RPS) Low Flow trip unit noise characteristics.

(4) Pressurizer Spray characteristics for various 2, 3, and 4 pump combinations.

3.2.2 Test Results

RCS flow measurements were taken at a RCS temperature and pressure of 532F and 2250 psia respectively using installed and temporary instrumentation. All instruments were calibration checked or calibrated at least twice, once before the test and once after. Measurements were taken for RCP combinations as listed in Table 3.2-1. All data collected was corrected for pressure and temperature and zero flow conditions. The flow coastdown portion of the test consisted of monitoring pump D/P, Vessel D/P and Steam generator D/P. Coastdowns were coordinated by turning off the common timing pulse for a brief interval and restarting it shortly before the pump trip. Recorder traces of RCS flow versus time after pump trips were made for each RCP combination listed in Table 3.2-1 as indicated under transient data. Figure 3.2-1 shows a typical flow coastdown curve computed from RPS flow input.

Spray Data was taken as per Table 3.2-2. Manual control was taken of pressurizer spray valves so as to demand a full open signal. Flow through one or the other spray line was verified by decreased pressurizer pressure and increased spray line temperatures. Spray lines are equipped with check valves which prevent reverse flow from the pressurizer to the RCS loops. Loop temperature increases which would indicate reverse flow were not observed. Table 3.2-2 and Figure 3.2-2 are the results of the pressurizer spray test.

3.2.3 Conclusion

Evaluation of test data indicated that all test data were satisfactory although the resultant flow was about 5% less than that necessary to guarantee the 370,000 gpm required

3.2 Reactor Coolant System, Pump Flow and Coastdown Test, Continued

3.2.3 Conclusion, Continued

by the Technical Specifications. An interim License amendment was issued limiting operation to 90% power based on this apparent low flow. The safety analysis was modified to cover a minimum acceptable RCS flow rate of 354,000 gpm pending final verification of flow through calorimetric determination. A calorimetric RCS flow determination was performed at 80% power. The flow determination produced a measured RCS flow of 399,000 gpm corresponding to 123.1$\pm$ 4.6% of design flow. This flow rate is above the minimum technical specification value of 113.9% design flow even with allowance for measurement uncertainties. This data, with supporting information, has been submitted to the NRC with a request for license amendment allowing operation at 100% power. It should be noted that, although RCS low flow trip setpoints are derived from this ΔP instrumentation, they are based on a specified relative change (decrease) in the output signal of the instruments and thus are independent of RCS flow measurements. See Section 7.3 for further discussion.

TABLE 3.2-1

REACTOR COOLANT PUMP FLOW AND COASTDOWN TEST SEQUENCE

RCPs RUNNING	STATIC DATA	NOISE DATA	SPRAY DATA	TRANSIENT DATA
1A1 1A2 1B1 1B2	X	X		X
NONE	X			
1A1	X			
1A1 1A2	X	X	X	
1A1 1A2 1B1	X	X	X	
1A1 1A2 1B1 1B2	X	X		X
1A2 1B1 1B2	X	X	X	
1B1 1B2	X	X	X	
1B1				
1A1 1B1	X	X	X	
1A1 1B1 1B2	X	X	X	
1A1 1A2 1B1 1B2	X			X
1A1 1A2 1B2	X	X	X	
1A1 1B2	X	X	X	
1B2	X			
1A2 1B2	X	X	X	
1A2	X			
1A2 1B1	X	X	X	
1A1 1A2 1B1				
1A1 1A2 1B1 1B2	X			X
1A1 1B1 1B2			X	
1A1 1A2 1B1 1B2	X			X
1A1 1A2 1B1				
1A1 1A2 1B1 1B2	X			

FIGURE 3.2.1

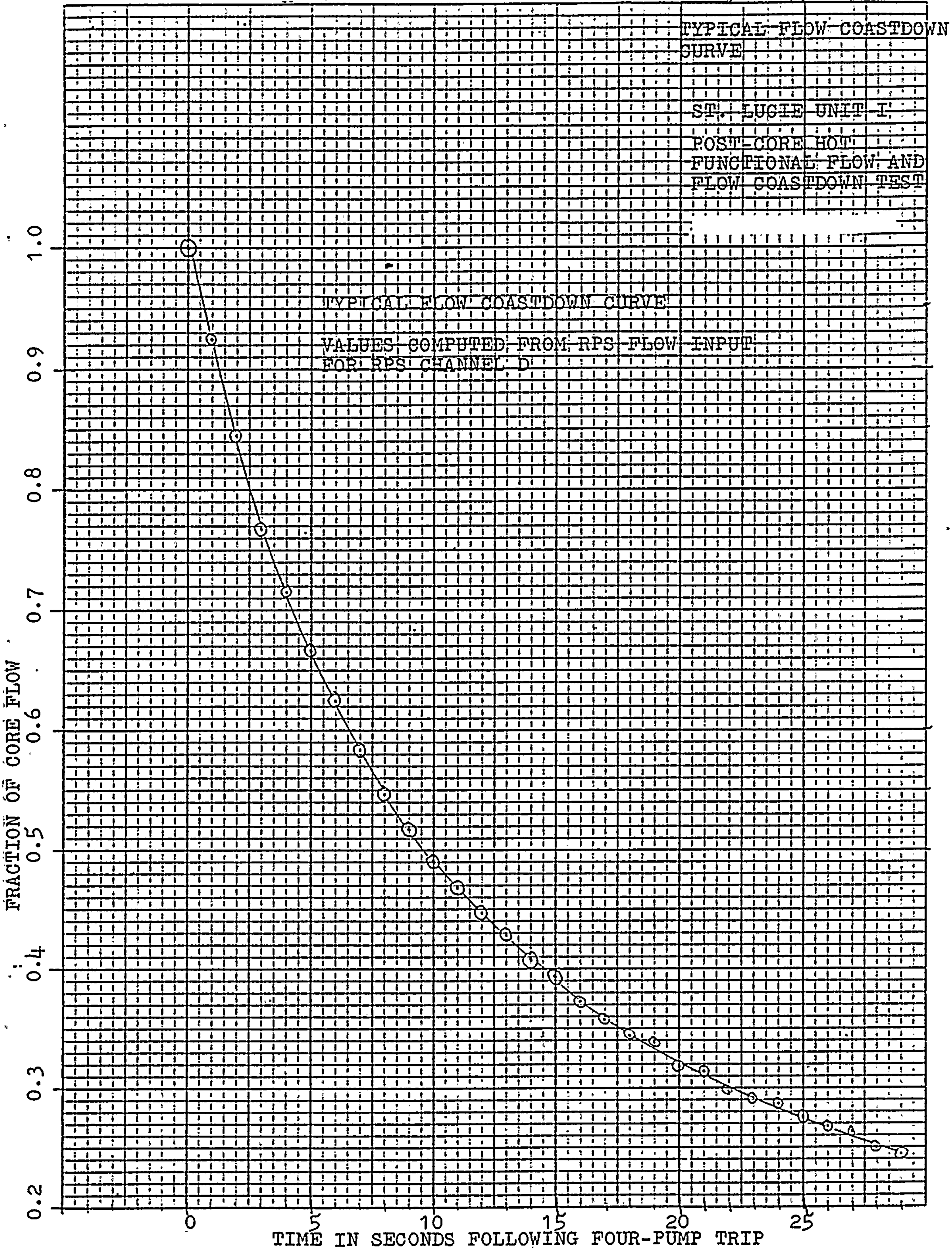

TYPICAL FLOW COASTDOWN CURVE
ST. LUCIE UNIT I
POST-CORE HOT FUNCTIONAL FLOW AND FLOW COASTDOWN TEST
TYPICAL FLOW COASTDOWN CURVE
VALUES COMPUTED FROM RPS FLOW INPUT FOR RPS CHANNEL D
FRACTION OF CORE FLOW
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0
5
10
15
20
25
TIME IN SECONDS FOLLOWING FOUR-PUMP TRIP

FIGURE 3.2.2
TRANSIENT RESPONSE - SPRAY VALVE FULL OPEN
(4 RCPS OPERATING)

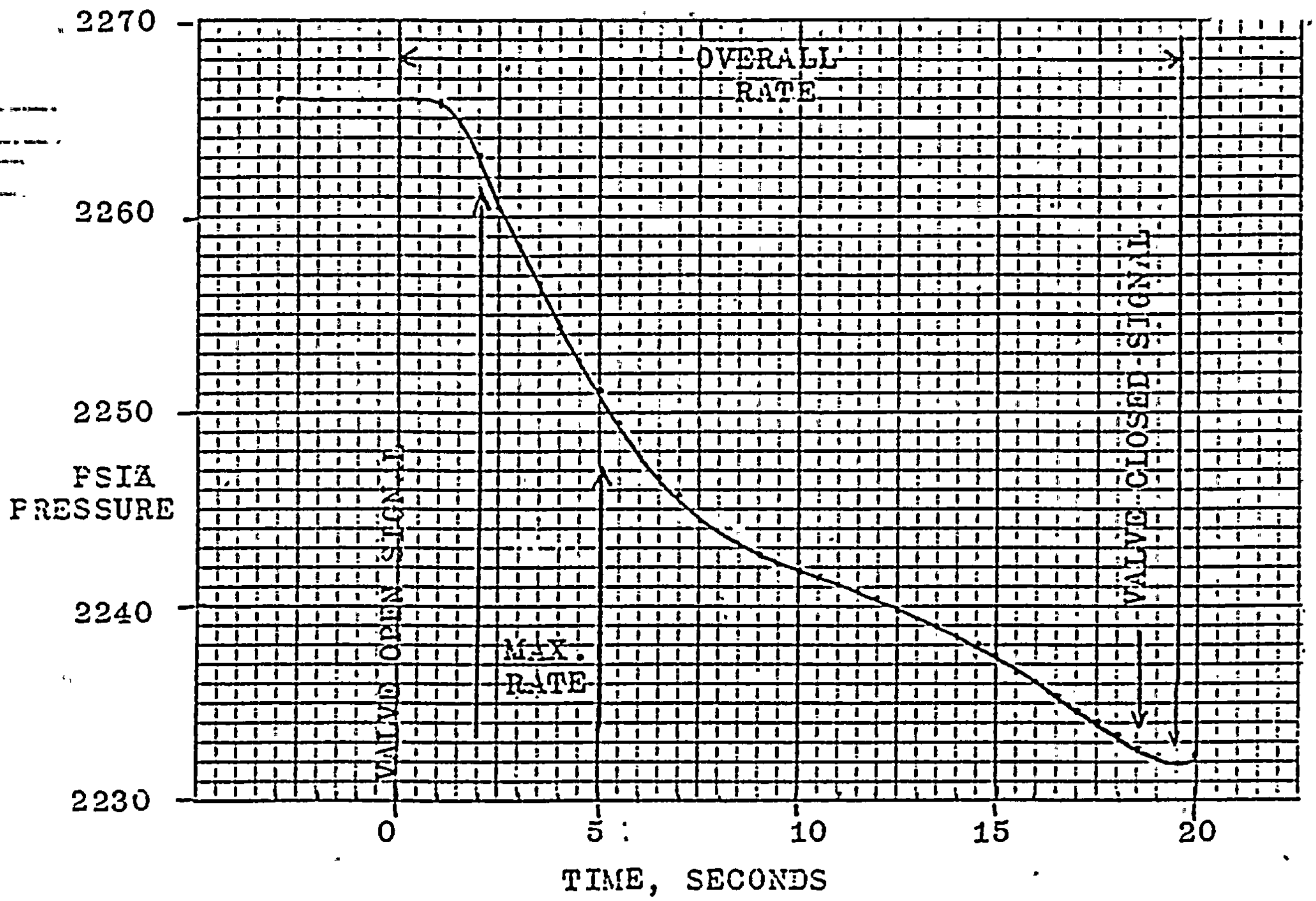


TABLE 3.2-2

SPRAY TEST RESULTS

	OPERATING PUMPS				MAXIMUM RATES PSI/MINUTE	OVERALL RATE PSI/MINUTE	°F SPRAY LINE TEMPERATURES INITIAL 1B1	1B2	FINAL 1B1	1B2
4 Pumps	1A1	1A2	1B1	1B2	225	156	480	520	510	530
3 Pumps	1A1	1A2	1B1		160	87	487	500	520	520
		1A2	1B1	1B2	200	120	475	485	510	510
	1A1		1B1	1B2	340	174	425	405	470	470
	1A1	1A2		1B2	150	62	445	460	440	510
2 Pumps	1A1	1A2			none	none	No change			
	1A1		1B1		none	none	No change			
	1A1			1B2	47	23	506	495	525	495
		1A2	1B1		none	none	No change			
		1A2		1B2	none	none	No change			
			1B1	1B2	300	81	475	480	500	500

3.3 RCS Leak Tests

3.3.1 Purpose

A leak test of the Reactor Coolant System (RCS) was performed to check for indications of abnormal leakage from the primary system and a leak rate test was performed to demonstrate sensitivity of the water inventory balance procedure.

3.3.2 Test Results - Leak Test

The leak test of the RCS was performed at 2330 (± 20) PSIA and consisted of a visual inspection of the entire RCS, with particular emphasis on the following areas:

(1) Reactor Coolant Pump (RCP) Seal Area

(2) Reactor Vessel Head Seal

(3) Steam Generator Manways

(4) Control Element Drive Mechanisms (CEDM's)

(5) In-Core Instrument Penetrations

(6) Valve stem packing and body to bonnet joints

(7) Pressurizer heater penetrations

The inspection of these areas showed no abnormal leakage and the test was considered satisfactory.

3.3.3 Test Results - Leak Rate Test

A water inventory balance (over 2 hours) was performed to determine RCS leak rate. Immediately afterward, a 1 gpm known leak was initiated through the sample system and another 2 hour inventory balance test run. Close agreement (within 1/10 gpm) between the two (disregarding the 1gpm known leak) indicated the accuracy of the method and that this method will detect leaks in the range of ¼gpm or less.

3.3.4 Conclusions

The RCS leak test showed that the primary system was tight after reactor vessel reassembly following fuel load and no abnormal leakage should be expected. The RCS leak rate test demonstrated the water inventory balance method works satisfactorily and will detect leaks in the range of ¼gpm or less.

3.4 Primary and Secondary Water Chemistry

3.4.1 Purpose

To establish, monitor, and control primary and secondary water chemistry during plant heatup and conduct of Post Core Hot Functional (PCHF) Tests using normal chemistry operating procedures. Baseline data to support the Low Power Physics and Escalation to Power Test phases was obtained. Also, Baseline Corrosion data was obtained and a chemical shock treatment was performed to aid in evaluating apparently low RCS flow.

3.4.2 Test Results

3.4.2.1 Chemistry Control

All primary and secondary water chemistry results during PCHF were either acceptable or when acceptance criteria were not met, corrective action was instituted to achieve acceptable conditions.

3.4.2.2 Baseline Corrosion.

(1) Suspended Solids (S.S.) in the RCS were generally <.01 ppm. The highest S.S. value observed was 0.25 ppm after a heatup from 260° to 530°F.

(2) A five gallon filter sample of reactor coolant was analyzed by X-ray diffraction. The results are as follows: (Relative %)

	Top Filter		Bottom Filter	
	Element	Oxide	Element	Oxide
Silica	44.2	50.1	15.7	17.7
Aluminum	21.2	21.1	67.7	67.3
Phosphorus	2.4	2.9	ND	
Sulfur	3.6	4.7	4.7	6.2
Chlorine	2.9	1.5	ND	
Potassium	1.0	0.65	ND	
Calcium	0.2	0.13	ND	
Chromium	ND			
Titanium	1.7	1.5	ND	
Iron	20.6	15.6	11.8	8.9
Nickel	0.5	0.34	ND	
Molybdenum	1.7	1.4	ND	

No foreign materials (organics) were indicated.

(3) During PCHE the Steam generators were intermittently fed with deaerated water from the Condensate Storage Tank (CST) and chemicals were injected manually to control pH and a hydrazine residual. The main feed and condensate systems were not operated. Consequently there were no S.S. problems. S. S. values were generally in the 0.1-0.2 ppm range. Highest value of S. S. observed was 0.5 ppm. Blowdown was run at maximum permissible rate dependent upon makeup water supply.

3.4.2.3 Chemical Shock Treatment

This evolution was performed to aid in evaluation of possible causes for the lower than expected RCS flow.

1) Prior to shock test: S.S. were generally <.01 ppm. On 4-15-76 @ 1700 the S.S. value was 0.25 ppm resulting from heatup from 260°F to 530°F.

2) Chemical Addition: On 4-16-76 @ 1340 six (6) gallons of Hydrazine were added yielding 40 ppm N_2H_4, and 1627 grams of LiOH was added increasing Lithium from 0.8 to 2.0 ppm. An increase in pH from 6.05 to 6.82 was observed.

3) Resulting S.S.: Ten minutes after chemical addition S.S. were 0.24 ppm. This result is believed to be erroneous. It appeared that part of the bottom filter had stuck to the drying tray thereby increasing the relative weight of the top filter. Visual inspection of the top filter indicated no discoloration and under magnification there appeared to be no significant particles. About three hours after chemical addition a four (4) liter filter showed 0.03 ppm.

   About twenty-two (22) hours after chemical addition a 5 gallon filter showed 0.038 ppm.

4) It was concluded that the shock test produced no significant crud burst.

3.4.3 Conclusions

Overall the data collected indicated that good chemistry control was maintained even with the variable plant conditions required by the test program. During baseline testing for Reactor Coolant System (RCS) particulate level and soluble corrosion products, no unusual or unexpected results were obtained. The lack of a crud burst from the chemical shock test implied that RCS preconditioning had been effective and demonstrated that there was no excessive buildup of crud (corrosion products) within the RCS.

3.5 Incore Instrumentation Functional Tests

3.5.1 Purpose

These tests were performed to:

(1) Obtain fixed incore thermocouple data
(2) Measure fixed incore detector leakage resistance
(3) Verify proper installation of the moveable incore detector transfer and drive machines and detector thimbles
(4) Verify proper operation of the moveable incore detector drive system
(5) Obtain path length measurements for use during future operation and
(6) Verify proper operation of incore detector drive system when controlled by the Digital Data Processing System (DDPS).

These tests will verify the system is ready for actual use for operations during LPPT and power ascension.

3.5.2 Test Results

3.5.2.1 Fixed Detectors

The thermocouple data was compared to other plant instrumentation and was satisfactory. The minimum leakage resistance of the fixed incore detectors was a factor of 2 higher than required. One detector string cable connector of 45 was faulty. In addition, one detector string was apparently damaged during installation and never functioned during initial startup and ascent to power. It has since been replaced.

3.5.2.2 Moveable Detectors

Proper operation of the system was verified by performing all evolutions the system was capable of, including measuring full path lengths and the satisfactory completion of this not only verifies proper operation but also proper installation.

NOTE: The thermocouples, by design, are encapsuled and isolated from RCS flow. Therefore, at power they are subject to preferential gamma heating and are not an accurate representation of RCS coolant temperatures.

3.5 Incore Instrumentation Functional Tests (cont.)

3.5.2.2 (cont.)

DDPS control of the moveable incore drive system was demonstrated by traversing all paths and ordering the DDPS to attempt to traverse impossible paths. The system functioned properly, including the printout of evolution in progress and "error" messages when improper commands were given.

3.5.3 Conclusions

These tests demonstrated that the thermocouples were satisfactory for indication of actual incore temperature at zero power and that the leakage resisteance of the fixed incore detectors was well above specification and should present no operational problems due to shorting out.

Also verified was the ability of the drive system to properly move the detectors to the various required locations in either manual control or when automatically controlled by the DDPS. Satisfactory completion of all these evaluations, including verifying all fixed detectors to have leakage resistance greater than 1×10^7 ohms, demonstrated the incore instrumentation to be ready for use during Low Power Physics Testing and Power Ascension.

3.6 Reactor Coolant System Piping Thermal Expansion and Restraint

3.6.1 Purpose

This test was repeated during PCHF as many restraints and snubbers had been added or modified since original performance of the test. The purposes were to verify that:

(1) piping systems were free to expand thermally as designed;

(2) spring hangers were not bottomed out or unloaded;

(3) hydraulic snubbers and pipe rupture restraints did not unduely restrict thermal movements;

(4) mechanical snubbers did not lock-up; and

(5) vent and drain lines did not vibrate excessively.

3.6.2 Test Results

The components described in Items 1 through 4 were observed and/or measurements taken by the Contractor at ambient, 260°F, 360°F, 470°F and 532°F and after cool-down to ambient. Also, mechanical snubbers were checked during the heatup to those temperatures. Vent and drain lines were checked during system operation to verify the (new) restraints prevented excessive vibration. Early in the heatup FP&L observers noted that 5 mechanical assemblies were locked up. (Licensee Event Report 335-76-9, 4/12/76). This prompted addition of Item 4 to the official test program. The cause was confirmed to be corrosion and all mechanical snubbers were replaced with snubbers of different design from a different vendor which are not susceptible to this problem. (Follow-up LER 335-76-9, 6/10/76). The locked snubbers were the only problem of any significance noted during this test.

3.6.3 Conclusions

This test verified that the piping restraint systems met all acceptance criteria. That is, it demonstrated that: piping was restrained only by the installed restraints, snubbers or hangers; that insulation was not deformed excessively by interferences; that piping displacements were as predicted by design (within tolerances) and that mechanical snubbers remained free and moved properly during heatup. These results verified that piping is properly restrained for system heatup and operation but is not over-restrained which could lead to overstress and eventual failure.

3.7 Reactor Coolant System and Steam Generator Instrumentation Calibration Check

3.7.1 Purpose

The purpose of this test was to check the calibration of the named instrumentation by correlating indications for a given parameter (i.e. T_{cold}) between the Digital Data Processing System (DDPS) Reactor-Turbine Gageboards (RTGB), Reactor Protection System (RPS) and the Engineered Safeguards System (ESF).

3.7.2 Test Results

Readings were taken from the locations listed above for certain parameters at 7 different plant conditions, from ambient temperature with the RCS vented to normal operating temperature and pressure. The parameters taken were:

(1) RCS T_{cold}
(2) RCS T_{hot}
(3) RCS pressure
(4) RCS flow
(5) Charging flow
(6) Letdown flow
(7) Letdown temperature
(8) Containment Pressure
(9) Steam Generator Pressure
(10) Steam Generator Level

Although all these parameters are not monitored at all the locations in 3.7.1, all are monitored at two or more of those locations. Data was evaluated throughout the test to ensure acceptable results.

3.7.3 Conclusions

The acceptance criteria for this test were met. All the safety channel correlations agreed within instrument accuracy tolerances. This gives assurance that the instrumentation provides accurate signals to control and protection systems and gives proper indications to guide the operators.

3.8 Pressurizer Controls Functional Test

3.8.1 Purpose

The objectives of this test were as follows:

(1) Verify the operation of the power operated relief valves at 2400 +0 -25PSIA (simulated pressure signal) and demonstrate that the quench tank would condense and cool the discharge of these valves.

(2) Establish proper settings for the continuous spray valves.

(3) Verify the operation of the spray valves at design pressures with the RCS at 532 $\pm 2^{o}$F and 4 RCP's operating.

(4) Verify that system pressure can be reduced at the design rate using spray flow. Rate of increase previously had been verified and is part of 5 below also.

(5) Verify design alarm and control settings and that the level and pressure controls maintain the pressurizer within desired limits.

(6) Demonstrate cooldown of the pressurizer using auxiliary spray.

3.8.2 Test Results

All of the above were satisfactorily completed. The rate of pressure decrease using normal spray was 80 psi/minute which exceeded the minimum required value of 65 psi/minute. All valves and controls (spray and power operated reliefs) operated properly. And, the quench tank accepted the power operated relief valve discharge for 30 seconds without overpressurization or rupture disc problem. This discharge was the first opportunity for observation of thermal expansion for the involved lines and tank and this was completed satisfactorily.

3.8.3 Conclusions

Item 2, setting of continuous spray valves, was implicitly included in Item 6, verification of control within design limits, as well as documented complete within the body of the procedure. Items 1 and 3 through 6 were acceptance criteria for the test and were completed satisfactorily. Also, note that additional spray valve testing was performed in conjunction with RCS flow testing (Section 3.2)

3.9 Main Generator Air Flow Test

3.9.1 Purpose

The purpose of this procedure was to collect data for the generator vendor to use to verify that design changes in the generator ventilation (cooling) system would prevent an unstable aerodynamic condition (blower surging) during operation.

3.9.2 Test Results

Air pressure, air temperature and barometric data were taken at various locations in the generator at two different values of generator revolutions per minute. (25% and 38% of full speed). Steam generated by the Reactor Coolant Pump heat was used to roll the turbine generator. The data was extrapolated to full speed. Comparison with the expected values (acceptance criteria) verified that the ventilation (cooling) system would function properly.

3.9.3 Conclusions

This test verified the design of the ventilation system and demonstrated that the system would provide proper cooling gas flow without aerodynamic instabilities such as blower surging or other flow related problems.

3.10 Reactor Coolant System Heat Loss Measurements

3.10.1 Purpose

The purpose of this test was to determine the RCS (including pressurizer) heat loss to the environment (containment). This number is an input to the calorimetric calculation which determines reactor thermal power. Also, it provides a check of the efficiency of the thermal insulation installed on the system.

3.10.2 Test Results

The pressurizer heat loss was determined by establishing very stable RCS conditions, isolating pressurizer spray and measuring the power input to the pressurizer heaters and the time intervals during which they were energized over a given period of time. Initial and final data were taken to verify pressurizer operating conditions remained stable. The calculated energy input was then equal to the heat loss. This value was then used as part of the data to determine RCS heat loss. This value was somewhat higher than expected but did not pose a safety or operational concern.

RCS heat loss was determined by establishing stable RCS conditions with a known steam generator level (mass) and isolating feedwater flow. RCS temperature was maintained steady by dumping steam through the Atmospheric Steam Dumps until steam generator level (mass) had been reduced to a predetermined level. Knowing the mass and enthalpy of the steam and having collected data on all other sources of energy to and from the RCS, including pressurizer heat loss as determined above, it was possible to calculate the total RCS heat loss. This steam down test was performed twice.

In addition, two feed up tests were performed. The only differences were that no steam was removed and feedwater was added to control RCS temperature and increase steam generator level (mass) a given amount. It proved impossible to regulate feedwater precisely enough to maintain the very tight RCS temperature band (normal $T_{cold} \pm 1^{o}F$) required for the heat loss calculation. However, the calculated results were of the same order of magnitude and provided assurance that the steamflow heat loss determination method was correct.

3.10 Reactor Coolant System Heat Loss Measurements (cont.)

3.10.3 Conclusions

The results of the 2 steam down tests agreed very well with each other, with the vendor's predicted results and with results from similar plants. This assured that the RCS heat loss value was proper for inclusion in the thermal power calorimetric calculation. It also confirmed that the design and installation of the RCS thermal insulation were satisfactory.

4.0 INITIAL APPROACH TO CRITICALITY

Initial criticality was achieved on 4-22-76 at Reactor Coolant System (RCS) conditions of 532°F and 2250 psia. The initial RCS boron concentration was 1729 ppm. The Initial Approach to Criticality (IAC) began by withdrawing the CEA's in specified increments with count rate data taken after each increment. During this withdrawal CEA Group control and group interlocks were verified to be functioning properly. Criticality was subsequently achieved by deborating the RCS to a boron concentration of 935 ppm.

Throughout the approach to criticality, two (2) independent sets of inverse multiplication plots were maintained. Two plots of inverse count rate versus RCS boron concentration were maintained during the dilution phase. Periodically, count rates were obtained from each Wide Range Log Channel (WRLC). The ratio of initial average count rate to the count rate at the end of each time increment was the value plotted.

The CEA withdrawal sequence and intervals are shown in Table 4.0-1. The inverse count rate versus CEA position points for two WRLC are shown in Figures 4.0-1 and 4.0-2. The inverse count rate versus RCS dilution time in hours is shown in Figures 4.0-3 and 4.0-4.

After achieving initial criticality, Control Element Assembly (CEA) Group 7 was used to control neutron flux. Conditions were stabilized at 10^{-4}% power and the critical data shown in Table 4.0-3 was recorded and compared with predicted values.

In summary, initial criticality was achieved in a safe and orderly fashion. There was good agreement between the measured and predicted critical boron concentrations.

TABLE 4.0-1

CEA WITHDRAWAL SEQUENCE

STEP *	CEA GROUP	INCHES WITHDRAWN	B 1/M	D 1/M
12.8.1.a	A	68	1.019	1.00
12.8.1.b	A	136	1.022	0.971
12.8.2.a	B	68	1.055	1.044
12.8.2.b	B	136	1.118	0.952
12.8.3	P	> 132	1.085	0.981
12.9.1	1	68	1.00	0.975
12.9.2	1 2	UEL ≤ 54	1.036	0.931
12.9.3	2 3	122 ≤ 40	1.081	0.968
12.9.4	2 3 4	UEL ≤ 107 ≤ 26	0.821	0.750
12.9.5	3 4 5	UEL ≤ 93 ≤ 11	0.845	0.819
12.9.6	4 5	UEL ≤ 79	0.881	0.824
12.9.7	5 6	UEL ≤ 54	0.855	0.777
12.9.8	6 7	122 ≤ 40	0.813	0.726
12.9.9	6 7	UEL 68	0.852	0.786
12.9.10	7	UEL	0.833	0.750
12.9.12	7	68	0.790	0.763

*Steps from IAC Procedure, #3200086

NOTE: UEL = Upper Electrical (position) Limit

TABLE 4.02

DILUTION TIME (Minutes)	DILUTION TIME (Hours)	CLOCK TIME	RCS BORON CHEM. ANAL.	B 1/M	D 1/M
0		4-21-762000	1731	0.996	0.999
30		2030	1679	0.962	0.926
60	1	2100	1594	0.972	0.796
90		2130	1528	0.816	0.887
120	2	2200	1459	0.795	0.675
150		2230	1389	0.692	0.726
180	3	2300	1327	0.627	0.628
210		2330	1277	0.571	0.641
240	4	4-22-760000	1220	0.593	0.504
250		0010		0.550	0.527
260		0020		0.516	0.504
270		0030	1182	0.478	0.516
280		0040		0.464	0.468
290		0050		0.471	0.496
300	5	0100	1134	0.408	0.415
310		0110		0.466	0.444
320		0120		0.461	0.426
330		0130	1112	0.470	0.452
340		0140		0.433	0.405
350		0150		0.435	0.409
360	6	0200	1114	0.434	0.444
390		0230	1112	0.400	0.427
420	7	0300	1114	0.468	0.411
450		0330	1116	0.452	0.443
480	8	0400	1112	0.459	0.429
510		0430	1084	0.409	0.359
540	9	0500	1062	0.382	0.373
570		0530	1039	0.339	0.331
600	10	0600	1017	0.315	0.308
630		0630	1015	0.312	0.294
660	11	0700	998	0.268	0.282
670		0710		0.260	0.252
680		0720	986	0.245	0.229
690		0730		0.235	0.221
700		0740	974	0.201	0.197
710		0750		0.168	0.174
720	12	0800	960	0.146	0.136
730		0810		0.079	0.079
740		0820	944	0.018	0.019
750		0830	935	0	0

TABLE 4.0-3

PARAMETER	INITIAL CONDITION	MEASURED VALUE
RCS TEMPERATURE	532°F	532°F
RCS PRESSURE	2250 PSIA	2250 PSIA
RCP'S OPERATING	4	4
CEA GROUPS WITHDRAWN, IN INCHES		
A	UEL	UEL
B	UEL	UEL
P-1	UEL	UEL
P-2	UEL	UEL
1	UEL	UEL
2	UEL	UEL
3	UEL	UEL
4	UEL	UEL
5	UEL	UEL
6	UEL	UEL
7	68	68

	PREDICTED	MEASURED
RCS BORON CONCENTRATION (PPM)	936	935

ST. LUCIE UNIT 1
INITIAL APPROACH TO CRITICALITY
BOL, 1st CYCLE, 532°F, 2250 PSIA
WIDE RANGE LOG CHANNEL B

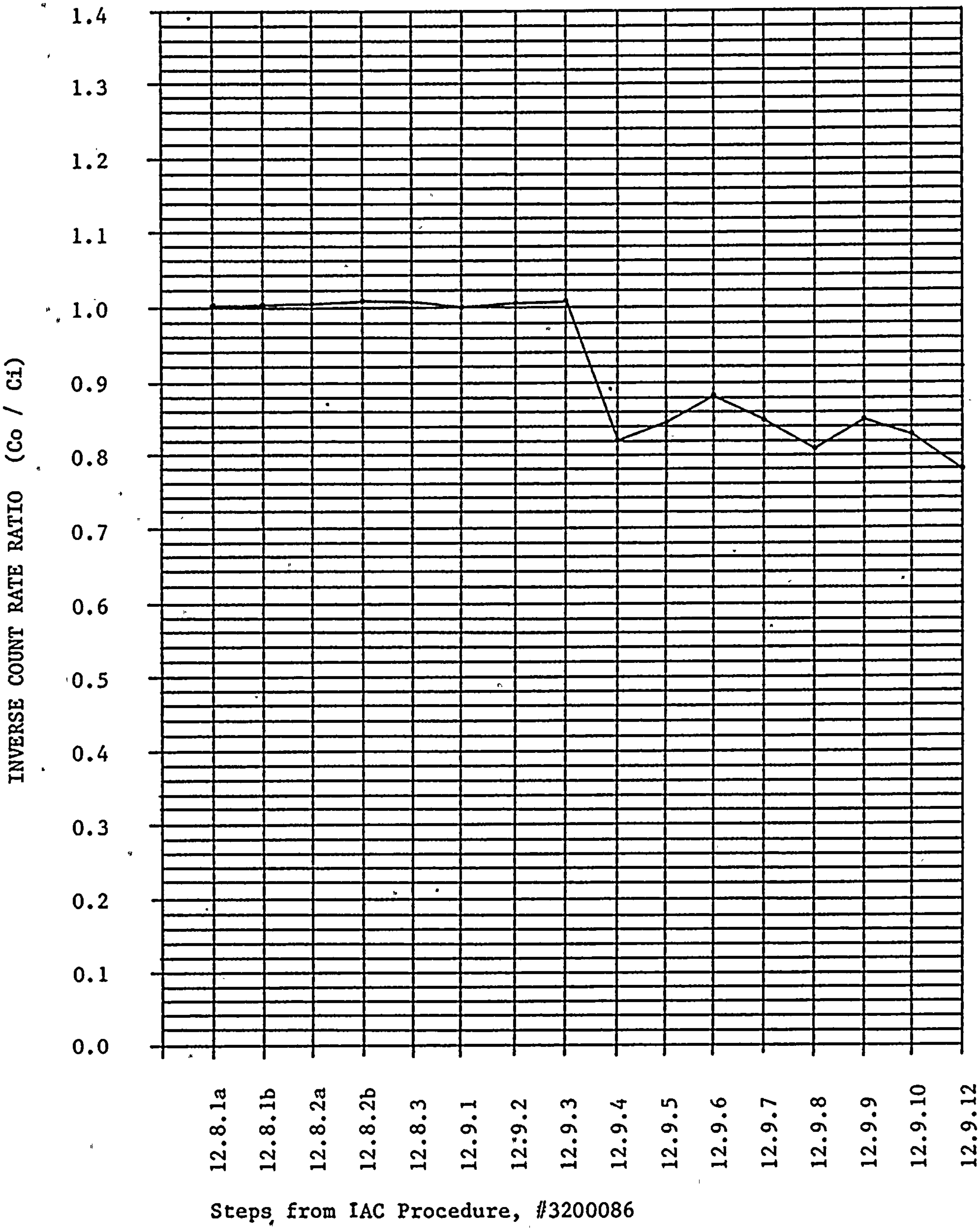
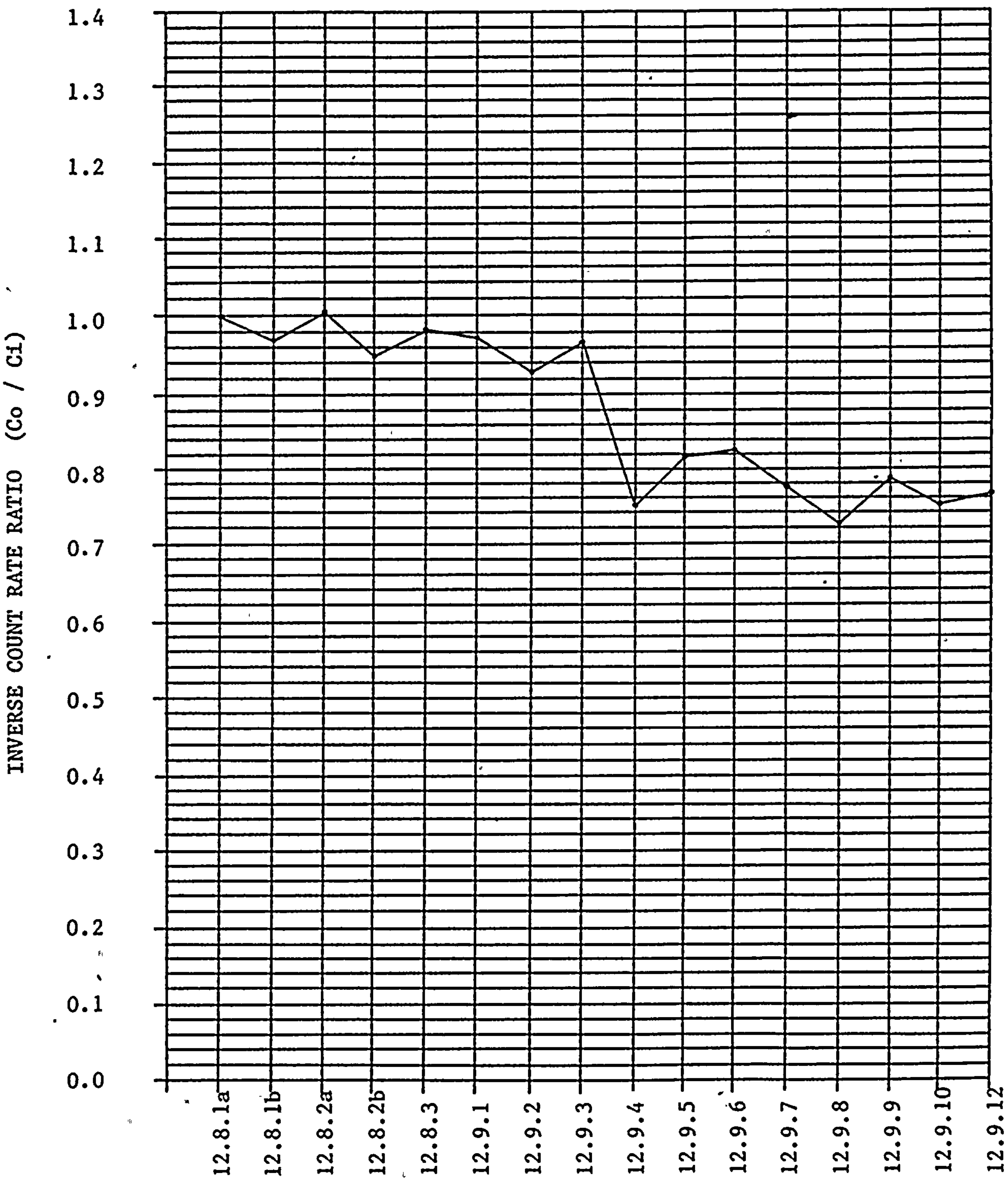


Figure 4.0-1

ST. LUCIE UNIT 1
INITIAL APPROACH TO CRITICALITY
BOL, 1st CYCLE, 532°F, 2250 PSIA
WIDE RANGE LOG CHANNEL D



Steps from IAC procedure, #3200086

Figure 4.0-2

ST. LUCIE UNIT 1
INITIAL APPROACH TO CRITICALITY
BOL, 1st CYCLE, 532°F, 2250 PSIA
WIDE RANGE LOG CHANNEL D

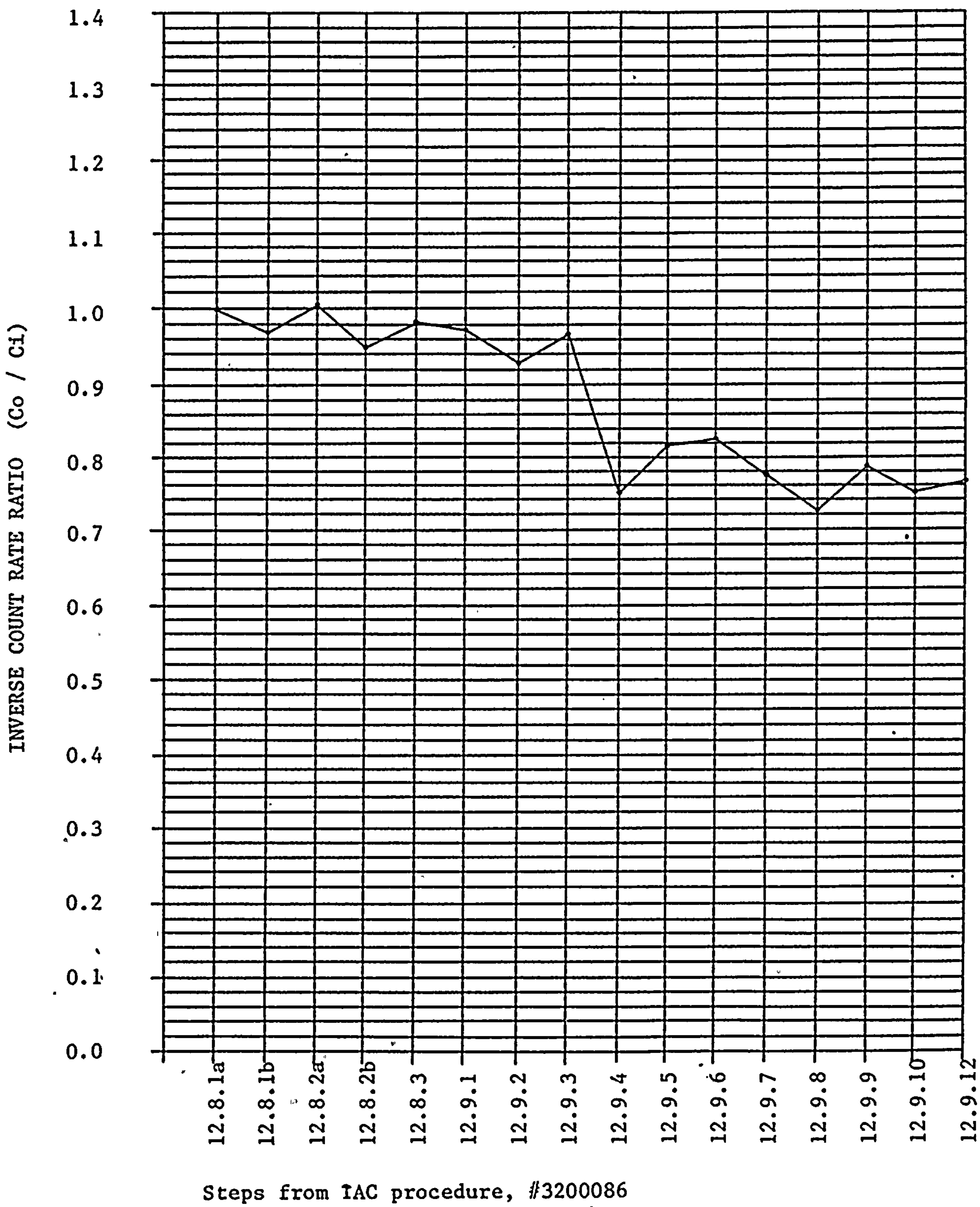


Figure 4.0-2

ST. LUCIE UNIT 1
INITIAL APPROACH TO CRITICALITY
BOL, 1st CYCLE, 532^{o}F, 2250 PSIA
WIDE RANGE LOG CHANNEL B
ZERO POWER; CEA GROUP 7 AT 68 INCHES

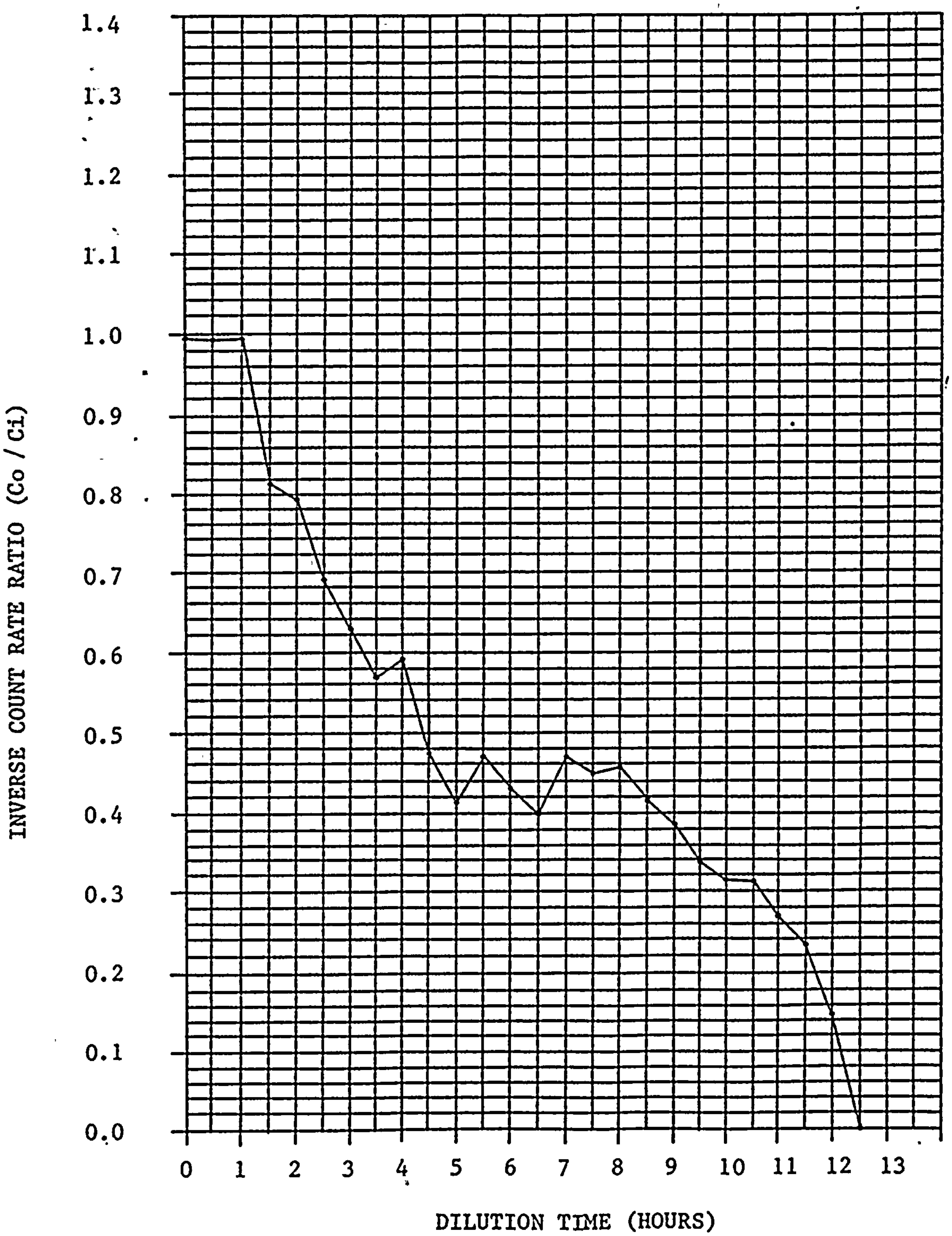


Figure 4.0-3

ST. LUCIE UNIT 1
INITIAL APPROACH TO CRITICALITY
BOL, 1st CYCLE, 532°F, 2250 PSIA
WIDE RANGE LOG CHANNEL D
ZERO POWER, CEA GROUP 7 AT 68 INCHES

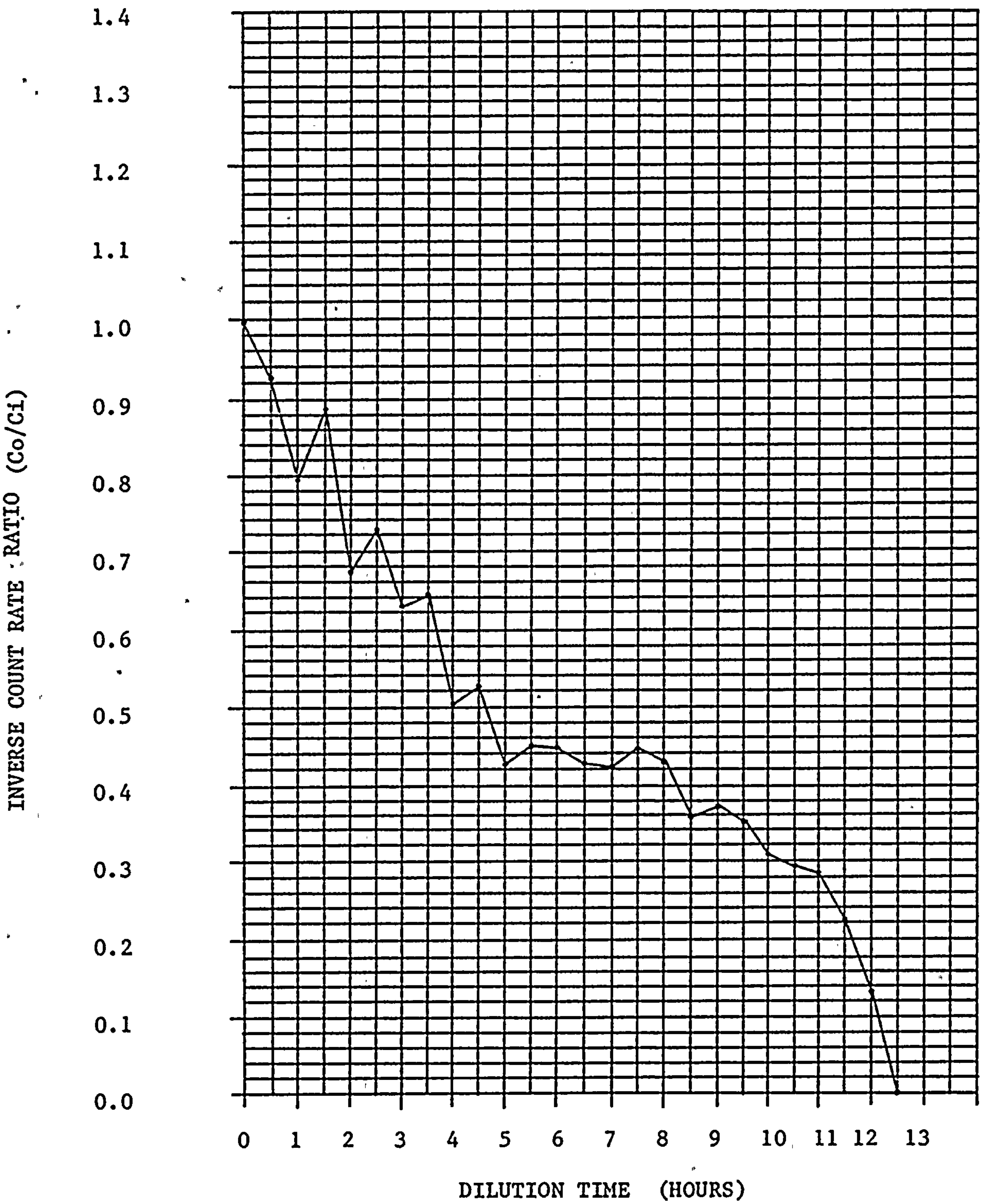


FIGURE 4.0-4

ST. LUCIE UNIT 1
BOL 1st CYCLE
REACTOR COOLANT SYSTEM BORON
CONCENTRATION VS DILUTION TIME

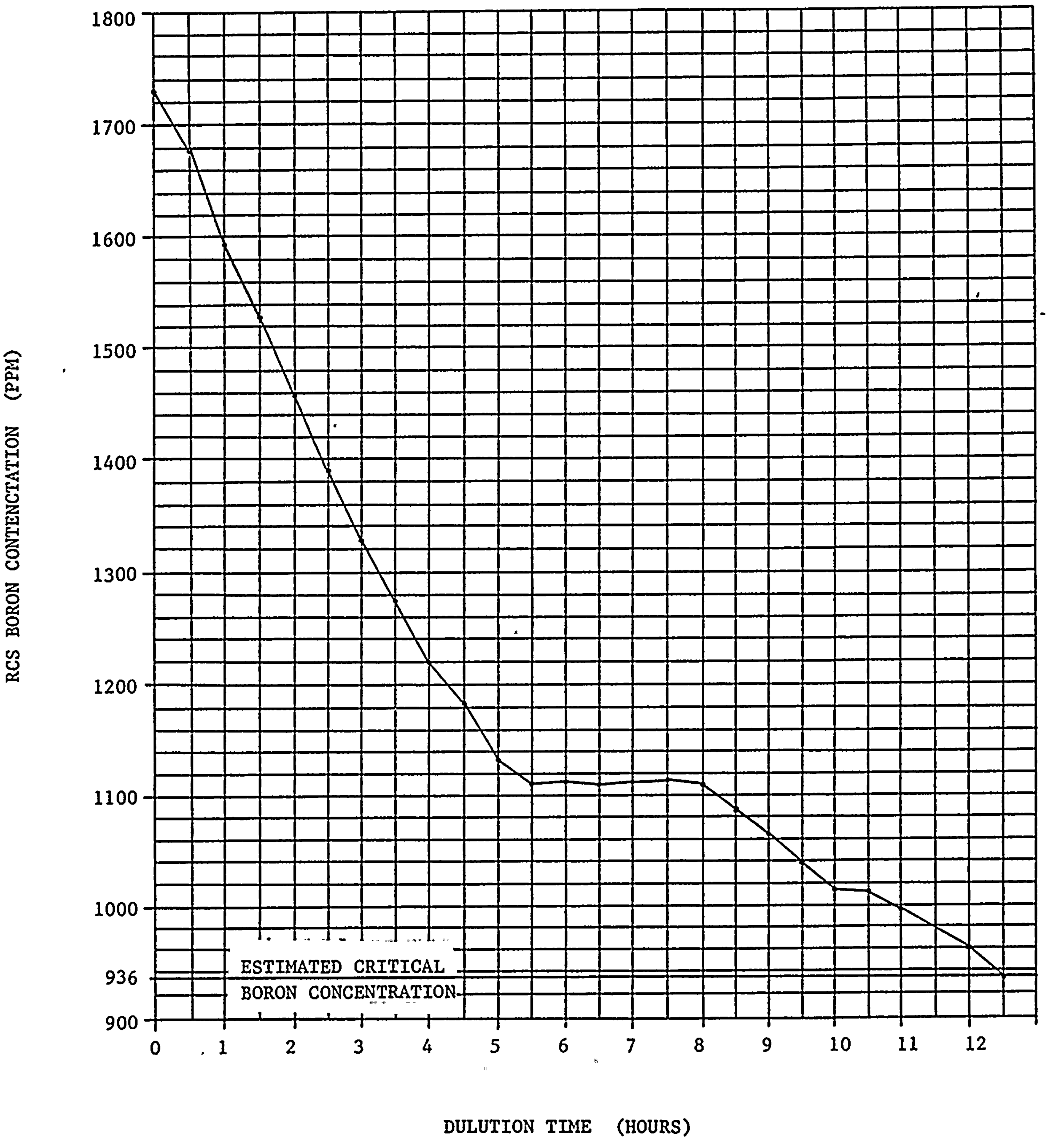


Figure 4.0-5

5.0 LOW POWER PHYSICS TESTS (LPPT)

The St. Lucie Unit 1 initial core consists of two hundred seventeen (217) fuel assemblies each containing one hundred seventy-six (176) fuel rods/burnable poison rods and five (5) Control Element Assembly (CEA) guide tubes. Fuel assemblies are divided into three (3) distinct groups by enrichment, Type A, B and C. Twelve (12) fuel rods in all Type B and twenty-eight (28) Type C fuel assemblies are replaced with burnable poison rods. Table 2.0-2 tabulates this and other important core design characteristics.

In addition to soluble boron in the Reactor Coolant System (RCS), reactivity control is provided by eight-one (81) CEA's. CEA's are inserted into and withdrawn from the core by means of sixty-nine (69) Control Element Drive Assemblies (CEDM's). Twelve (12) CEDM's are attached to dual CEA's. Figure 5.0-1 shows the core location of the CEA's. The CEDM's are arranged into eleven (11) CEA Groups. Those Groups are further defined by function. CEA Groups A and B are Shutdown Groups. CEA Groups 1 through 7 are Regulating Groups. CEA Groups P-1 and P-2 are Power Shaping Groups. Figure 5.0-2 displays the relative core location of the CEA Groups.

CEA Group movement is restricted as a function of power level in order to insure that CEA configurations not analyzed for in the safety analysis do not occur. The mechanism for this restriction is a so-called Power Dependent Insertion Limit (PDIL) curve found in the Technical Specifications. Automatic control features as well as operating instructions prevent insertion of CEA Groups into the core below this PDIL curve. The lower the reactor power, the greater the CEA insertion allowed.

LPPT consists primarily of the measurement of reactivity worths of phenomena which can vary the critical condition of the core. To speed the collection of this data, as well as to enhance its accuracy, an analog computer which solves the kinetics equation for reactivity was used. Several techniques were used in conjunction with this reactivity computer to measure CEA worths. The soluble boron swap technique consisted of a continuous or slug dilution or boration of the RCS simultaneous with small compensating reactivity changes in CEA position. The reactor was kept near critical during this evolution, and the reactivity computer provided a signal which could be trended and correlated with CEA position as a function of time. A CEA trip technique was also used in conjunction with the reactivity computer. The rapid change in reactivity caused by a CEA or CEA Group trip was correlated with reactivity change detected by the reactivity computer.

5.0 LOW POWER PHYSICS TESTS (LPPT) (Cont'd)

All raw test data was collected, reduced, and analyzed on site. In all cases, measured data met applicable acceptance criteria. CE, Windsor provided backup support for all measured data analyses.

FIGURE 5.0-1

ST. LUCIE UNIT 1

CORE LOCATION
OF THE CEDM's

BOL, 1st CYCLE

(CEA LOCATIONS ARE GIVEN IN FIGURE 2.0-3)

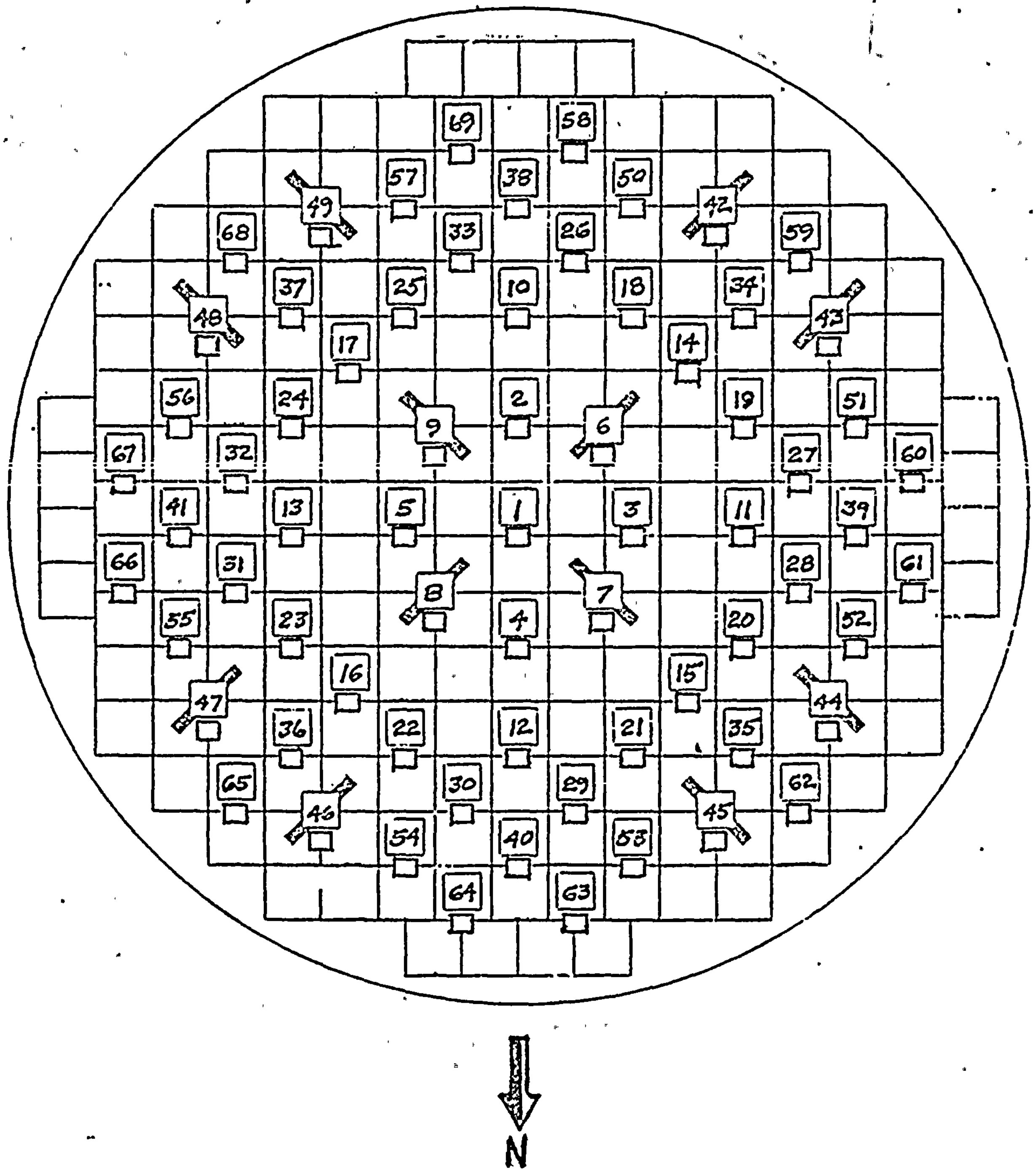
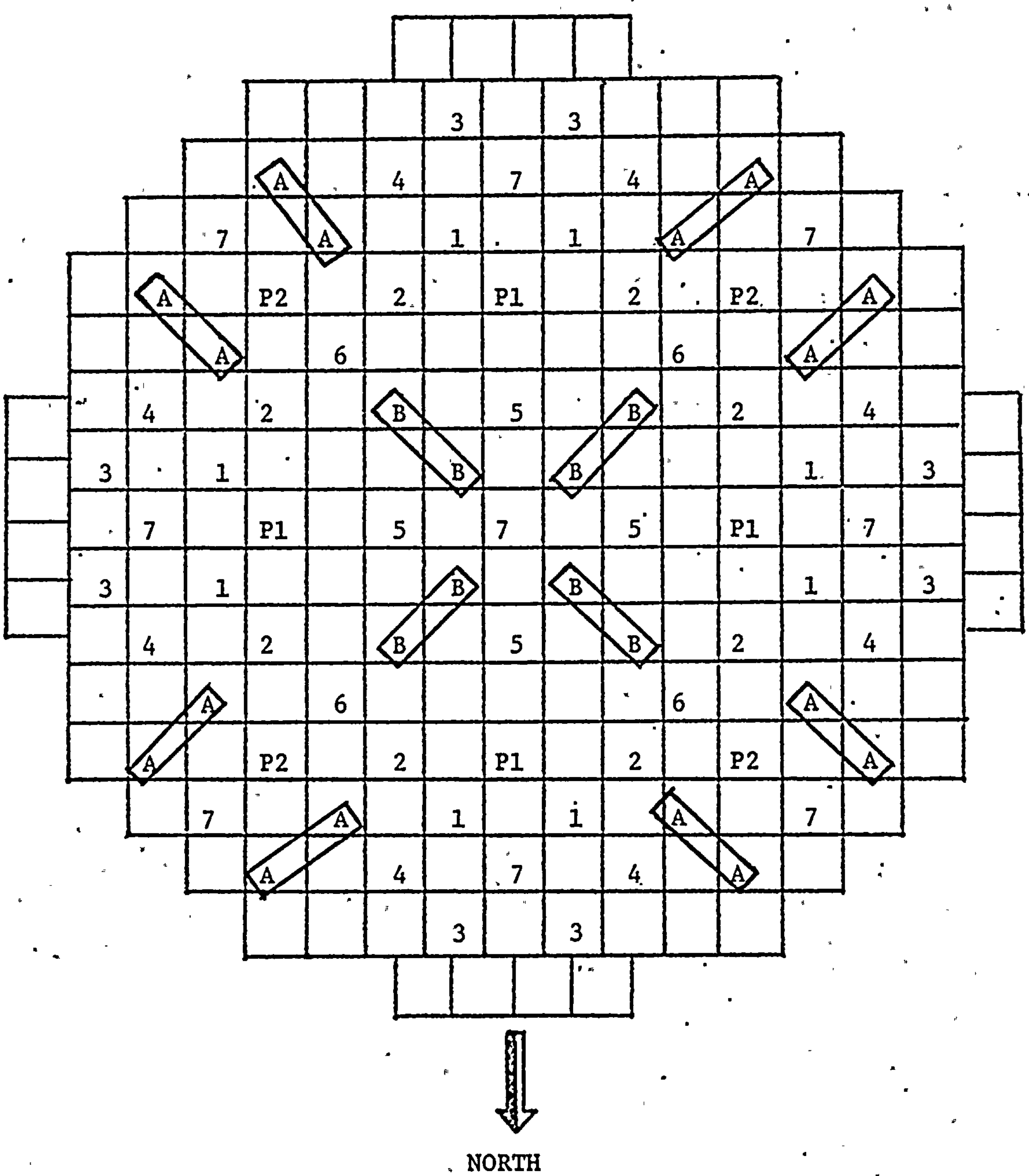


FIGURE 5.0-2

ST. LUCIE UNIT 1

RELATIVE CORE LOCATIONS
OF THE CEA GROUPS

BOL 1st CYCLE



5.1 Critical Boron Concentration Measurements

5.1.1 Purpose

Critical boron concentration measurements were performed at various CEA positions at relatively constant RCS temperature and pressure. The purpose of these measurements was to obtain an as-measured value for the excess reactivity loaded in the core and to provide basis for verification of predicted CEA Group reactivity worths.

5.1.2 Test Results

Boron concentration values were averages of multiple chemical analysis measurements made during periods of stable reactor coolant system (RCS) boron concentration. Boron end point technique was used when required. This method borates (dilutes) CEA's out near UEL* (in near LEL**). After RCS conditions stabilize, and the RCS boron concentration has been chemically analyzed, the CEA's are quickly moved to UEL (to LEL), reactivity stabilized, and CEA quickly moved back in (out) to their bite position. The reactivity change (reactivity being plotted on a recorder of a reactivity computer) is measured. The amount of reactivity added (subtracted) is converted, via boron worth, to an equivalent Δ PPM. This Δ PPM is added to (subtracted from) the measured boron concentration. This technique gives a safe, fast and accurate method of determining critical boron concentrations at hard to achieve CEA positions (relatively low reactivity worths at UEL's and LEL's).

5.1.3 Conclusions

Results indicate that measured boron concentration were in adequate agreement within predictions and well within the acceptance criteria of ± 100 PPM.

* UEL - Upper Electrical Limit
** LEL - Lower Electrical Limit

TABLE 5.1-1

CEA CONFIGURATION	ARO	7 LEL	5 LEL	4 LEL	2 LEL	B LEL
MEASURED CBC (ppm)	962	906	845	746	605	523
PREDICTED CBC (ppm)	963	906	841	733	579	---
RCS TEMPERATURE, ^{o}F	533.0	532.8	533.7	535.3	534.7	533.7
RCS PRESSURE, psia	2255	2255	2255	2256	2250	2250
CEA GROUP	POSITION					
P1	UEL	UEL	UEL	UEL	UEL	UEL
P2	UEL	UEL	UEL	UEL	UEL	UEL
A	UEL	UEL	UEL	UEL	UEL	UEL
B	UEL	UEL	UEL	UEL	UEL	LEL
1	UEL	UEL	UEL	UEL	UEL	LEL
2	UEL	UEL	UEL	UEL	LEL	LEL
3	UEL	UEL	UEL	UEL	LEL	LEL
4	UEL	UEL	UEL	LEL	LEL	LEL
5	UEL	UEL	LEL	LEL	LEL	LEL
6	UEL	UEL	LEL	LEL	LEL	LEL
7	UEL	LEL	LEL	LEL	LEL	LEL
TIME	1352	1602	2027	0818	2155	0600
DATE	4/24/76	4/24/76	4/24/76	4/25/76	4/25/76	4/27/76

5.2 Critical Boron Concentration and Soluble Boron Worth Measurements

5.2.1 Purpose

Soluble boron in the form of dissolved boric acid in the Reactor Coolant System provides variable reactivity control over the life of a core. It can supplement the reactivity control provided by CEA Groups. However, its principal function is to compensate for burnup of excess reactivity as core depletion proceeds. The critical boron concentration for various CEA configurations were measured in order to develop a relationship for determination of the soluble boron reactivity worth. CEA Group hold down values were also measured and are presented in Section 5.5

5.2.2 Test Results

CEA Group integral reactivity worths were measured using a soluble boron swap technique. In addition, the soluble boron concentration at the end point of several of those CEA configurations was also measured. Soluble boron samples were independently analyzed by FP&L and by CE-Windsor. A comparison of measured boron worths for these several CEA configurations/ reactor coolant system (RCS) boron concentrations is listed in Table 5.2-1.

5.2.3 Conclusions

The agreement between measured and predicted critical boron concentrations and between measured and predicted soluble boron worths are adequate and well within the acceptance criterion.

TABLE 5.2-1

COLUMN 1	COLUMN 2	COLUMN 3	COLUMN 4	COLUMN 5
MEASURED CRITICAL BORON CONCENTRATION	CEA CON-FIGURATION	CHANGE IN CBC	NON-OVERLAP CEA WORTH	MEASURED SOLUBLE BORON WORTH
TABLE 5.3-1			TABLE 5.5-1	COLUMN 3 ÷ COLUMN 4
PPM BORON		ΔPPM	% Δk/k	PPM / % Δk/k
962	ARO	---	---	---
906	7 LEL	56	0.707	79.21
845	5 LEL	61	0.781	78.10
746	4 LEL	99	1.262	78.45
605	2 LEL	141	1.814	77.73
			AVERAGE	78.37

5.3 Chemical and Radiochemical Tests

5.3.1 Purpose

Chemical and radiochemical analyses of the Reactor Coolant System (RCS) and Steam Generators were performed to determine baseline corrosion data, fission and activation product levels and buildup, to detect failed fuel and various impurities which could enhance corrosion.

5.3.2 Test Results

5.3.2.1 Baseline Corrosion

Data collection, commenced during PCHF, continued during LPPT.

(1) Suspended Solids (S.S. in the RCS were generally <.01 ppm. The highest S. S. value observed was 0.25 ppm after a heatup from 260° to 530°F.

(2) During Low Power Physics Testing the Steam generators were intermittently fed with deaerated water from the Condensate Storage Tank (CST) and chemicals were injected manually to control pH and a hydrazine residual. The main feed and condensate systems were not operated. Consequently there were no S.S. problems. S. S. values were generally in the 0.1-0.2 ppm range. Highest value of S. S. observed was 0.5 ppm. Blowdown was run a maximum permissible rate dependent upon makeup water supply.

5.3.2.2 Dissolved Oxygen (D.O.)

(1) Prior to heatup and LPPT, D.O. was scavenged by hydrazine and nitrogen purges of the Volume Control Tank (VCT).

(2) Prior to and during LPPT, D.O. was generally <.005 ppm and at no time did it exceed the limit of 0.1 ppm when >250°F.

(3) On two occasions prior to LPPT D.O. approached the limit reaching a maximum of .07 ppm. These increases were due to large volume dilutions and borations. In both cases hydrazine was immediately added.

(4) During LPPT, 0.01 ppm was the highest D.O. detected.

5.3 Chemical and Radiochemical Tests (cont)

5.3.2.3 Fission and Activation Product Buildup

(1) Fission Products (uci/ml)

	max.	min.
I 133	4.4×10^{-7}	2.7×10^{-7}
Xe 135	9.3×10^{-7}	2.6×10^{-7}

(2) VCT gases detected (uci/cc)

	max
Ar 41	1.8×10^{-5}
Kr 85M	1.49×10^{-6}
Kr 88	2.42×10^{-6}
Xe 133	9.1×10^{-7}
Xe 135	3.6×10^{-6}

(3) Activation Products (uci/ml)

	max.	min.
Ar 41	3.5×10^{-6}	7.9×10^{-7}
F 18	3.6×10^{-5}	3.0×10^{-6}
Na 24	7.8×10^{-7}	7.3×10^{-7}
Gross. Act.	8.2×10^{-5}	9.6×10^{-7}

5.3.2.4 Lithium: Approximately one week prior to LPPT lithium was added to 0.75 ppm, then to 1.5 ppm. Several subsequent additions were made to maintain >1.0 ppm.

Due to numerous dilutions and borations, lithium buildup was not evident until well into the power ascension program.

5.3.2.5 Demineralizers: RCS purity and lithium were maintained by use of a 1:1 LiOH borated mixed bed resin.

A second demineralizer was loaded with HOH borated resin. This was used on one occasion during Power Ascension for lithium removal. Due to the purity of reactor coolant, D.F.'s were low and no meaningful data was accumulated.

5.3 Chemical and Radiochemical Tests (cont.)

5.3.3 Conclusions

Evaluation of the test results showed no abnormal or unexpected conditions in the RCS or steam generators and provided good baseline data for future use. Also, it was verified that, for the hot standby conditions of the LPPT phase, the chemistry control methods worked quite satisfactorily.

5.4 Temperature Coefficient of Reactivity Measurements

5.4.1 Purpose

The moderator temperature coefficient of reactivity can be either negative or positive, depending upon the magnitude of the Reactor Coolant System boron concentration. The moderator temperature coefficient cannot be measured directly but it can be derived from a measurement of the isothermal temperature coefficient.

5.4.2 Test Results

Isothermal temperature coefficient measurements were conducted at several different CEA withdrawal configurations and boron concentrations. Measured values for each condition are the result of averaging data from several segments of the heatup and cooldown phases of the measurement. Throughout the measurements, reactor power was maintained below the point of adding nuclear heat to minimize the confusing effect of doppler feedback. Reactor Coolant System ramp temperature changes were affected by proper positioning of turbine bypass or atmospheric dump valves.

Table 5.4-1 summarizes the results of the measurements and comparisons with predicted valves. Agreement between measured and predicted values indicates acceptance criteria has been met. Technical Specification 3.1.1.4a specifies that the moderator temperature coefficient shall less positive than $+0.5X10^{-4}$ $\Delta k/k/^oF$ whenever power $\leq 70\%$.

5.4.3 Conclusions

For all cases, the measured values of isothermal temperature coefficient are within the acceptance criterion of $+0.5X10^{-4}$ $\Delta k/k/^oF$ of the predicted value, and is therefore acceptable.

TABLE 5.4-1

CEA CONFIGURATION	5 LEL	2 ∿ 10"	7 ∿ 120"
MEASURED MTC, $x10^{-4}$ $\Delta K/K$ $/^{o}F$	-0.25	-0.79	+0.25
TECHNICAL SPECIFICATION LIMIT MTC	<0.50	<0.50	<0.50
RCS NOMINAL TEMPERATURE, ^{o}F	532	532	532
RCS PRESSURE, psia	2250	2250	2250
CEA GROUP	POSITION		
P1	UEL	UEL	UEL
P2	UEL	UEL	UEL
A	UEL	UEL	UEL
B	UEL	UEL	UEL
1	UEL	UEL	UEL
2	UEL	∿ 10"	UEL
3	UEL	LEL	UEL
4	UEL	LEL	UEL
5	LEL	LEL	UEL
6	LEL	LEL	UEL
7	LEL	LEL	∿120"
TIME	0100	0200	0900
DATE	4-25-76	4-26-76	4-30-76

5.5 Non-Overlapped Regulating and Shutdown CEA Group Worth Measurements

5.5.1 Purpose

During reactor operations, nearly all excess reactivity is held down by soluble boron concentration in the Reactor Coolant System and burnable poison shim rods in the fuel assemblies. Additional hold down and reactivity control is provided by moveable Control Element Assemblies (CEA). These CEA's are arrayed in symmetrical groups about the core (see Figure 5.0-1). The number of CEA's in each Regulating and Shutdown CEA Group and the function of that group is described in Table 5.5-1. The CEA Group worths were measured in a non-overlapped mode over the full range of their movement at various Reactor Coolant System boron concentrations.

5.5.2 Test Results

All CEA Group reactivity worths were measured using a soluble boron swap method, either dilution or boration, to maintain criticality while inserting or withdrawing CEA Groups in increments. The reactivity trace generated by this evolution was reduced to obtain the relationship between CEA Group positions from full in to full out and integral reactivity worth at these positions.

For Shutdown CEA Group A, integral worth was measured using the soluble boron swap method in combination with a group trip method. The combination of methods allows total integral worth of Group A to be determined from extrapolation of measured data without decreasing shutdown margin below the Technical Specification limit.

The integral worths of all Shutdown and Regulating CEA Groups were measured at 532°F. These results are compared with predicted values in Table 5.5-1. In addition, the integral reactivity worth curves developed at 532°F for all Shutdown and Regulating CEA Groups are displayed in Figures 5.5-1 through 5.5-9.

5.5.3 Conclusions

The measured CEA Group integral reactivity worths are in good agreement with predicted values and are well within acceptance limits.

TABLE 5.5-1

CEA GROUP	NUMBER OF CEA'S	FUNCTION	REACTIVITY WORTH % ΔK/K		ACCEPTANCE LIMITS
			PREDICTED	MEASURED	
P1	4	SHAPING	--	--	--
P2	4	SHAPING	--	--	--
A	* 16	SAFETY	4.120	4.520	3.090-5.150
B	** 8	SAFETY	0.406	0.425	0.305-0.508
1	8	REGULATING	0.687	0.648	0.515-0.859
2	8	REGULATING	1.451	1.297	1.088-1.814
3	8	REGULATING	0.575	0.517	0.431-0.719
4	8	REGULATING	1.374	1.262	1.031-1.718
5	4	REGULATING	0.329	0.311	0.247-0.411
6	4	REGULATING	0.505	0.470	0.379-0.631
7	9	REGULATING	0.738	0.707	0.554-0.923

(* EIGHT SETS OF DUAL CEA'S)
(** FOUR SETS OF DUAL CEA'S)

ST. LUCIE UNIT 1
INTEGRAL CEA GROUP WORTH
BOL, 1ST CYCLE, 532°F, 2250 PSIA
CEA GROUP 1

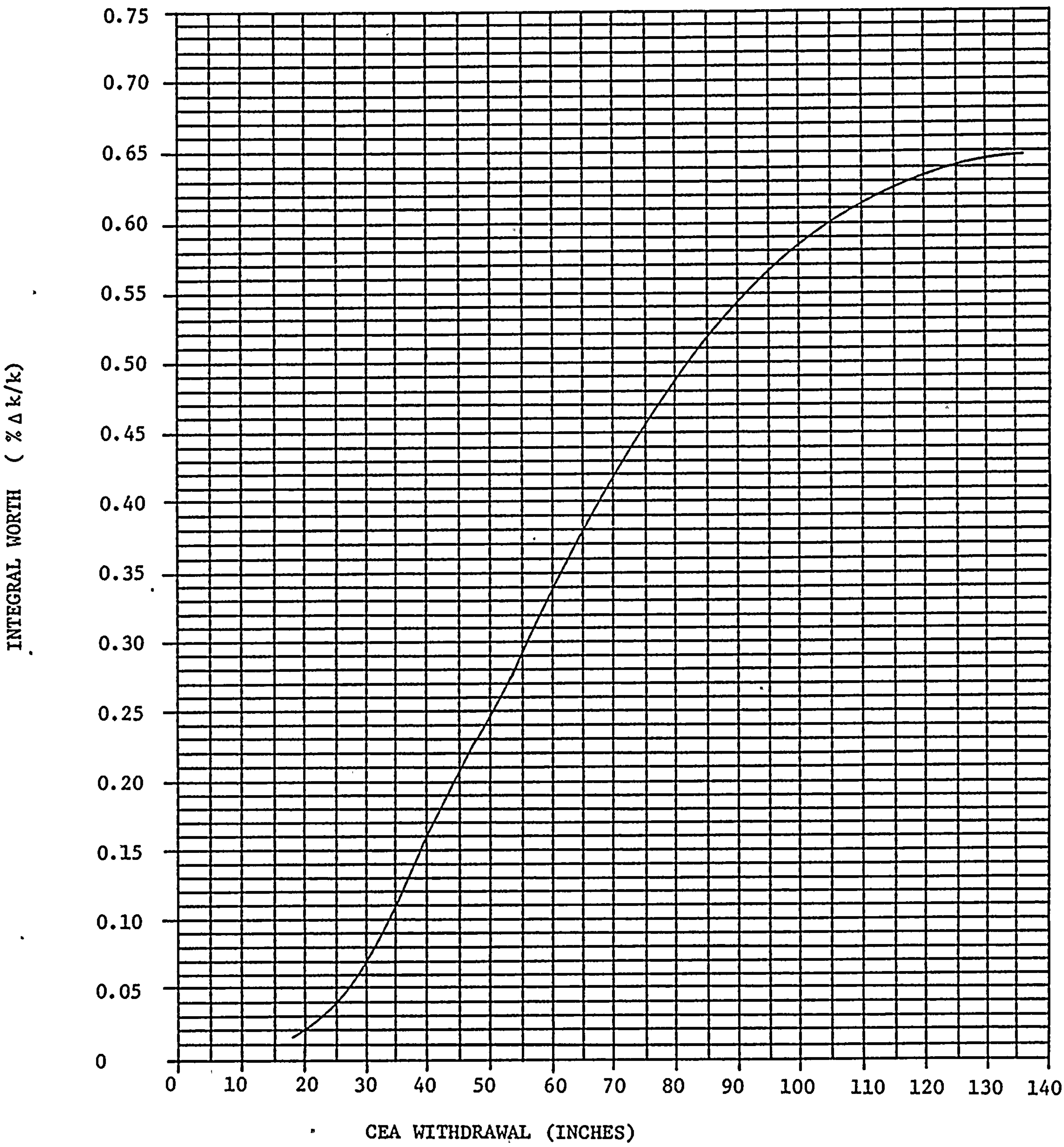


FIGURE 5.5-1

ST. LUCIE UNIT 1
INTEGRAL CEA GROUP WORTH
BOL, 1ST CYCLE, 532°F, 2250 PSIA
CEA GROUP 2

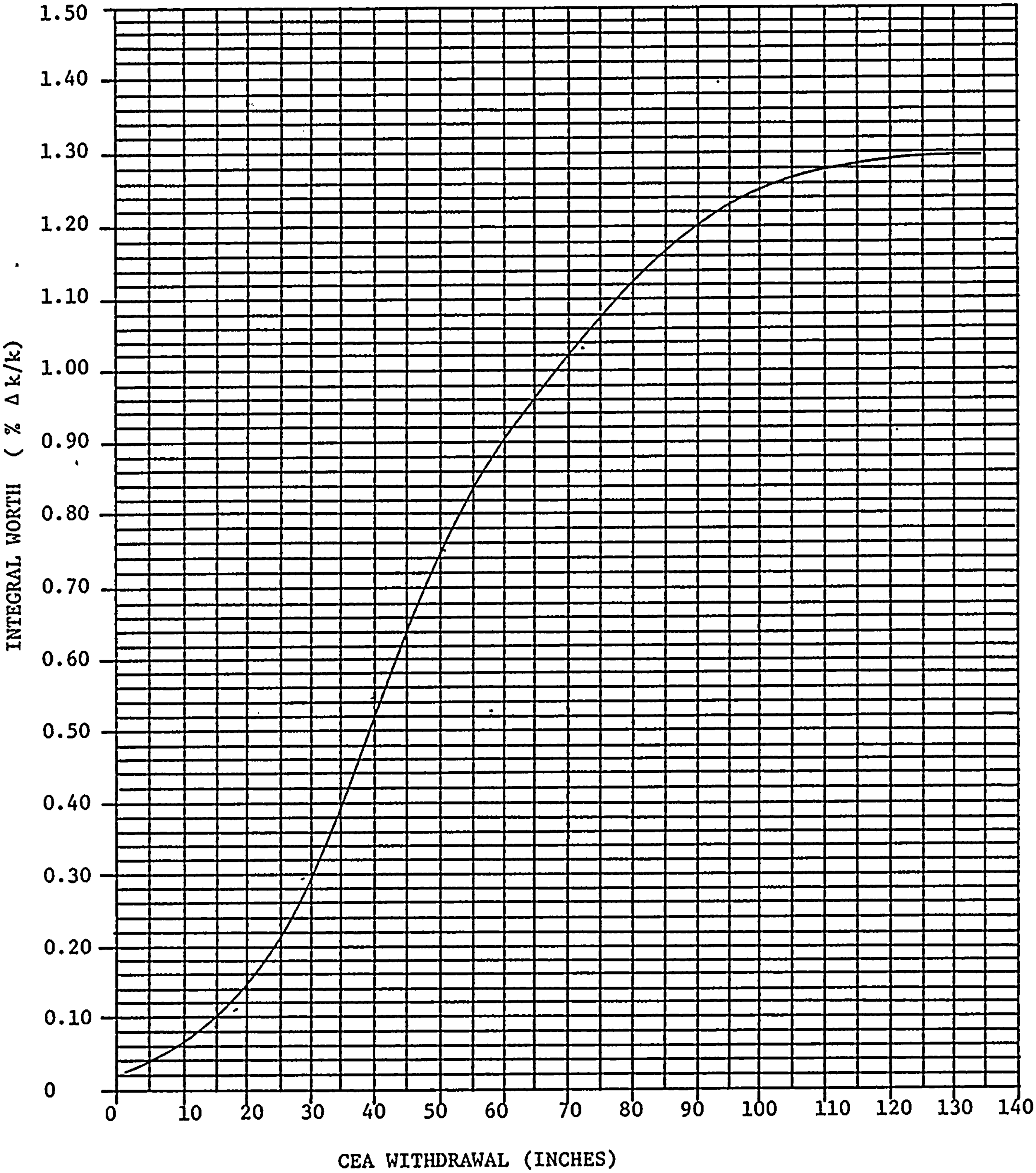


FIGURE 5.5-2

ST. LUCIE UNIT 1
INTEGRAL CEA GROUP WORTH
BOL, 1ST CYCLE, 532°F, 2250 PSIA
CEA GROUP 3

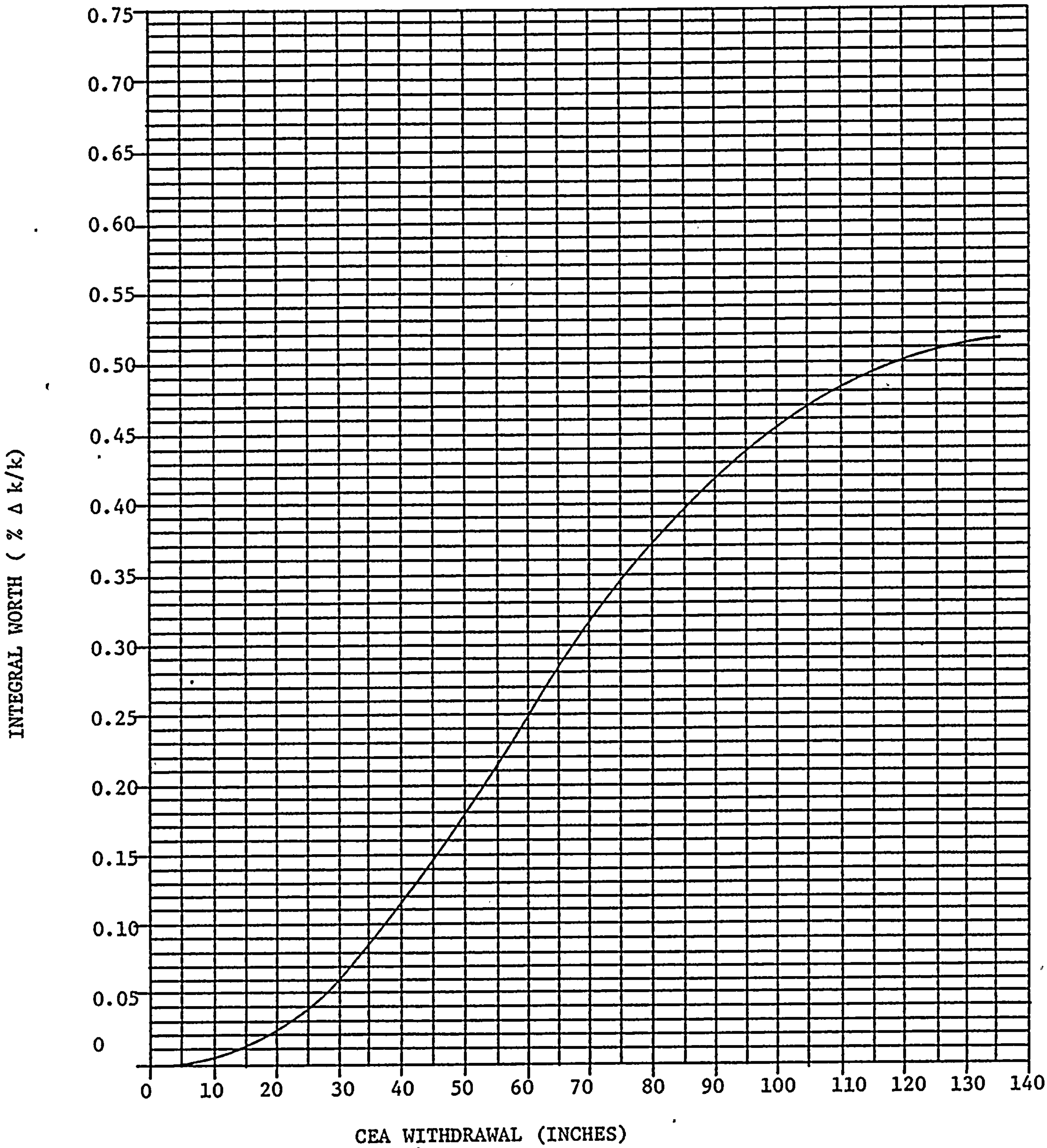


FIGURE 5.5-3

ST. LUCIE UNIT 1
INTEGRAL CEA GROUP WORTH
BOL, 1ST CYCLE, 532°F, 2250 PSIA
CEA GROUP 4

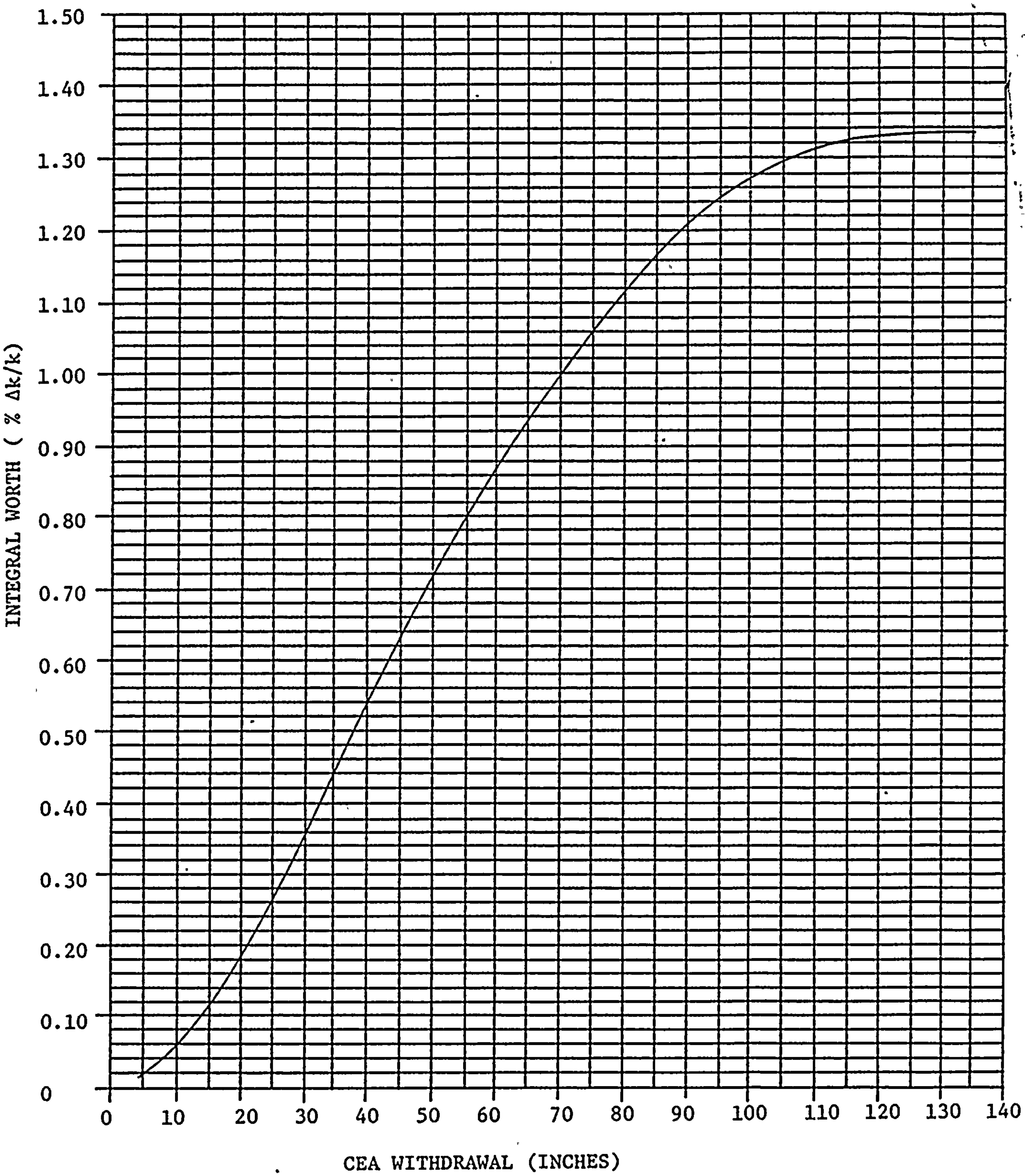


FIGURE 5.5-4

ST. LUCIE UNIT 1
INTEGRAL CEA GROUP WORTH
BOL, 1ST CYCLE, 532°F, 2250 PSIA
CEA GROUP 5

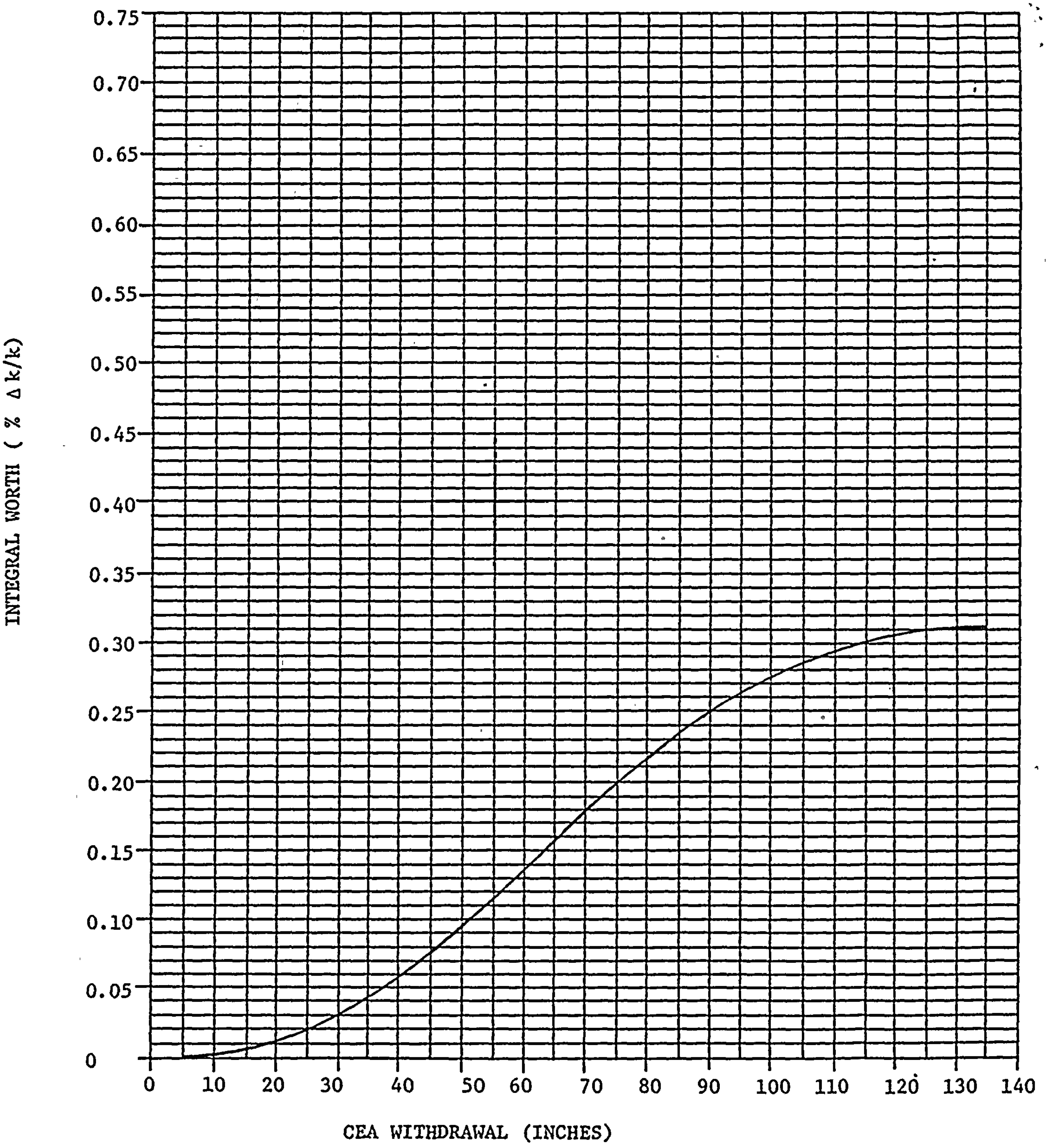


FIGURE 5.5-5

ST. LUCIE UNIT 1
INTEGRAL CEA GROUP WORTH
BOL, 1ST CYCLE, 532°F, 2250 PSIA
CEA GROUP 6

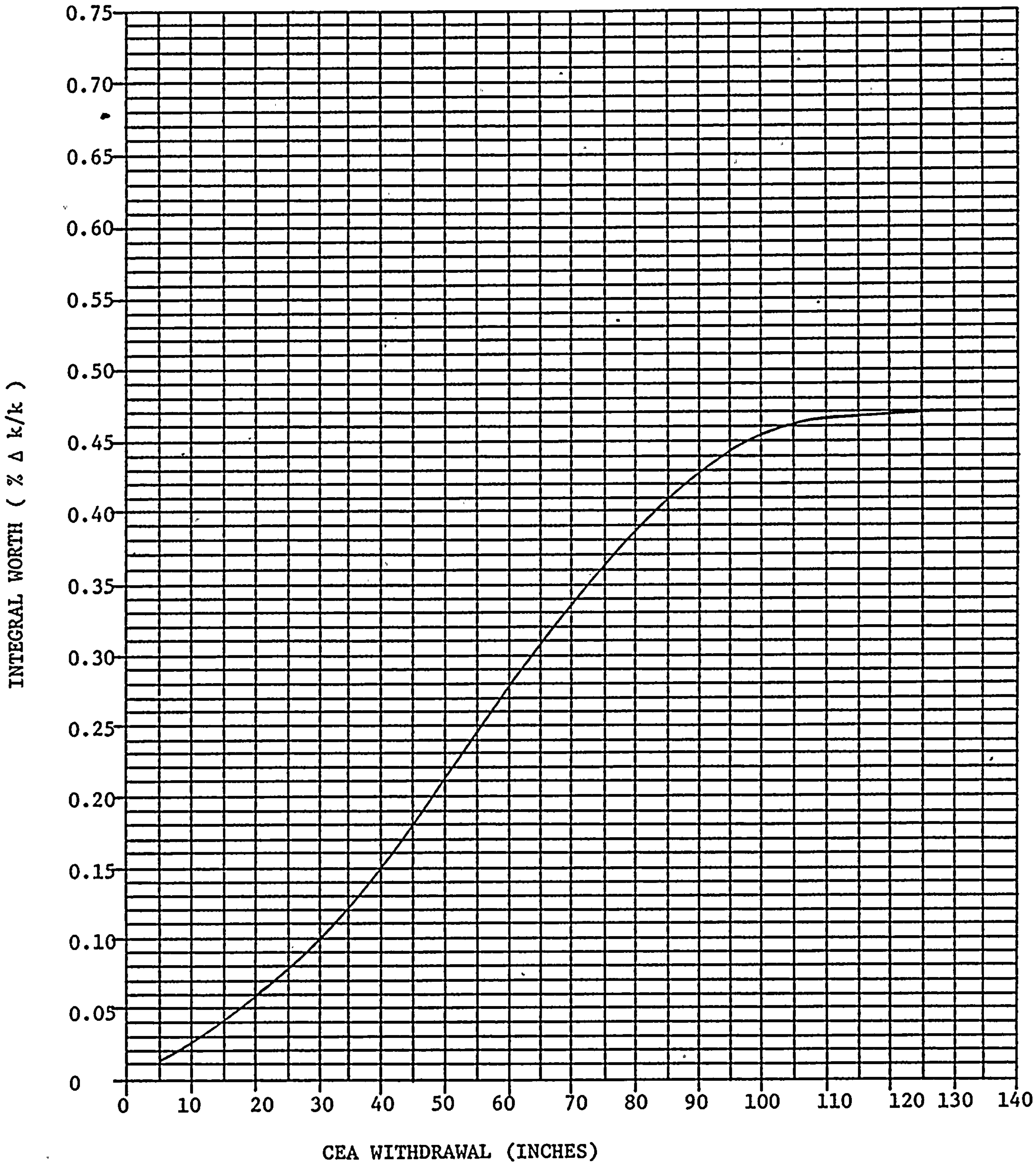


FIGURE 5.5-6

ST. LUCIE UNIT 1
INTEGRAL CEA GROUP WORTH
BOL, 1ST CYCLE, 532°F, 2250 PSIA
CEA GROUP 7

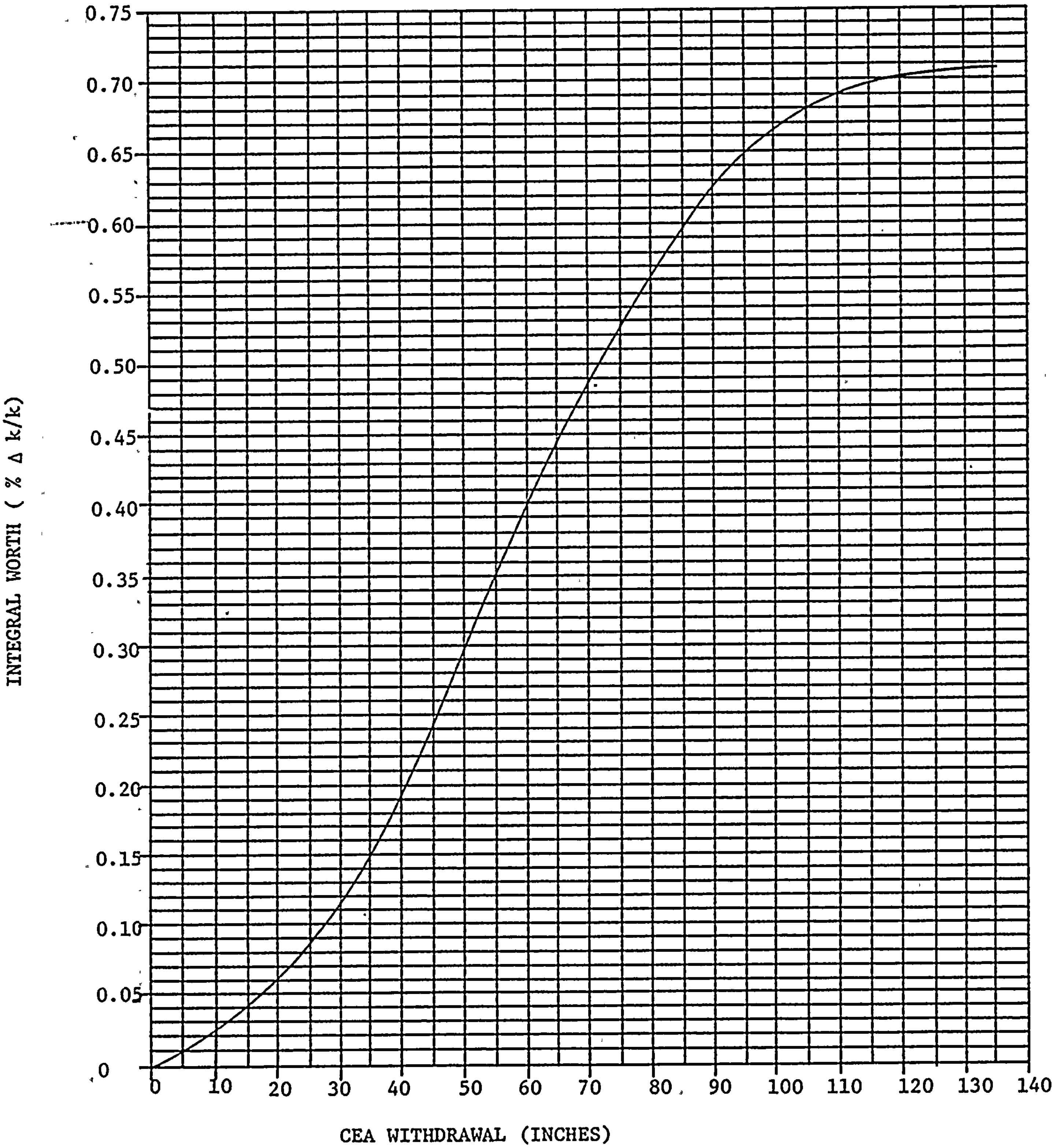


FIGURE 5.5-7

ST. LUCIE UNIT 1
INTEGRAL CEA GROUP WORTH
BOL, 1ST CYCLE, 532°F, 2250 PSIA
CEA GROUP B

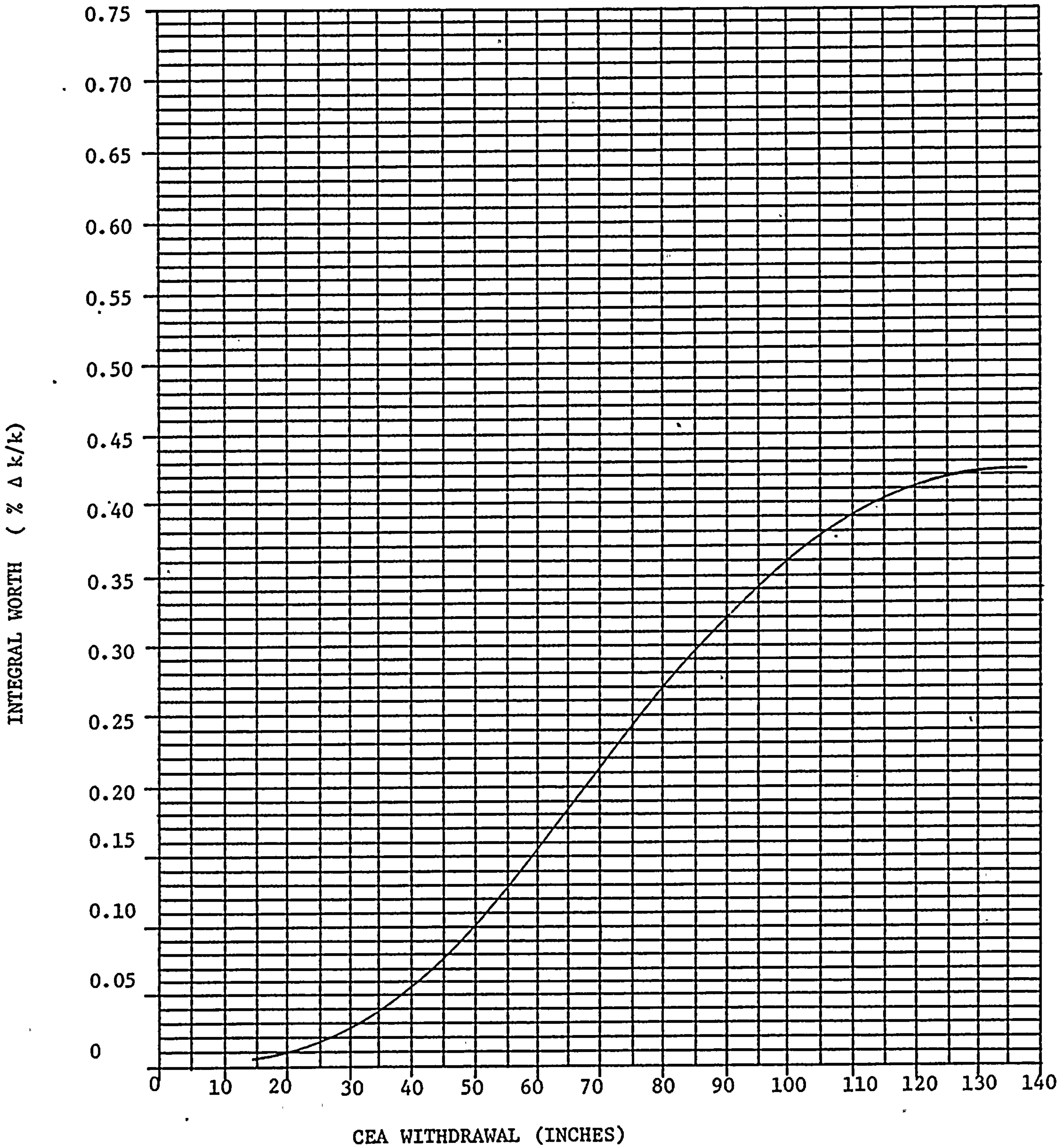


FIGURE 5.5-8

ST. LUCIE UNIT 1
INTEGRAL CEA GROUP WORTH
BOL, 1ST CYCLE, 532°F, 2250 PSIA
CEA GROUP A

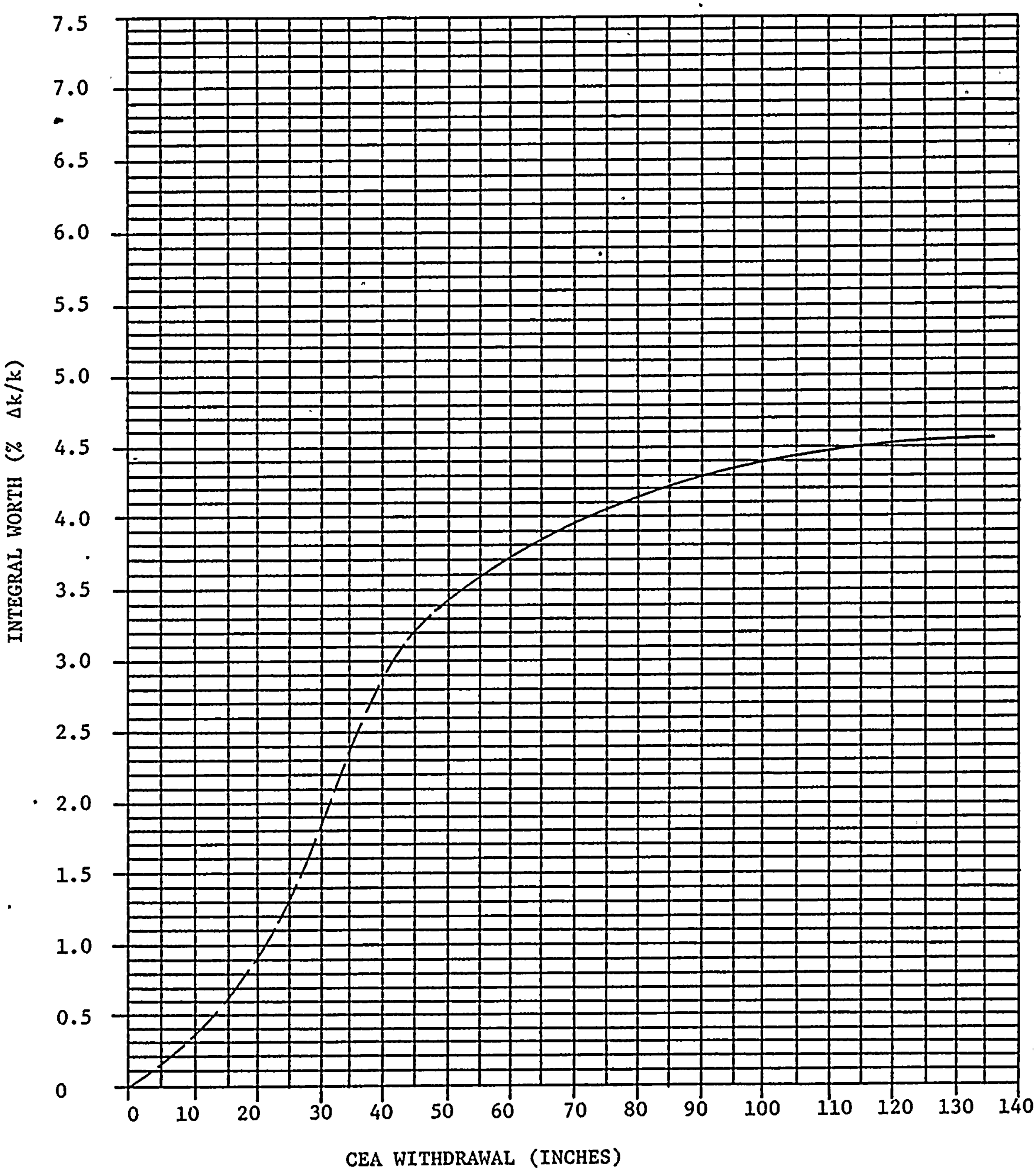


FIGURE 5.5-9

5.6 Overlapped Regulating CEA Group Worth Measurements

5.6.1 Purpose

Reactor Power level may be controlled by sequential insertion or withdrawal of Regulating Control Element Assemblies (CEA). Percent of overlap is selected so as to insure a relatively constant insertion rate of positive or negative reactivity over the full range of CEA Group movement. Technical Specifications allow CEA Group insertion as a function of reactor power level. The integral reactivity worth curve for Regulating CEA Groups 1,2,3,4,5,6, and 7 in an overlapped mode was measured. Maximum allowed insertion at zero power being at approximately 68 inches withdrawal on CEA Group 4.

This measurement was made at a Reactor Coolant System (RCS) temperature of 532°F. Principal purpose of the measurement being to develop an integral worth curve for use in making estimated critical condition calculations prior to a reactor startup. Reactor startup's are performed at a nominal RCS temperature of 532°F.

5.6.2 Test Results

The overlapped integral reactivity worth of CEA Groups 1,2,3,4,5,6 and 7 was measured using a soluble boron swap method to maintain criticality while sequentially withdrawing CEA Groups in increments. The reactivity trace developed by this CEA Group movement was reduced to obtain the relationship between CEA Group positions and integral reactivity worth at those positions. Figure 5.6-1 displays the overlapped integral reactivity worth curve.

5.6.3 Conclusions

The Overlapped Integral CEA Worth curve derived from this measurement has been adequate for use as an operational tool, and has been placed in the Plant Curve Book.

ST. LUCIE UNIT 1
BOL, 1ST CYCLE, 532°F, 2250 PSIA
SEQUENTIAL CEA WORTH

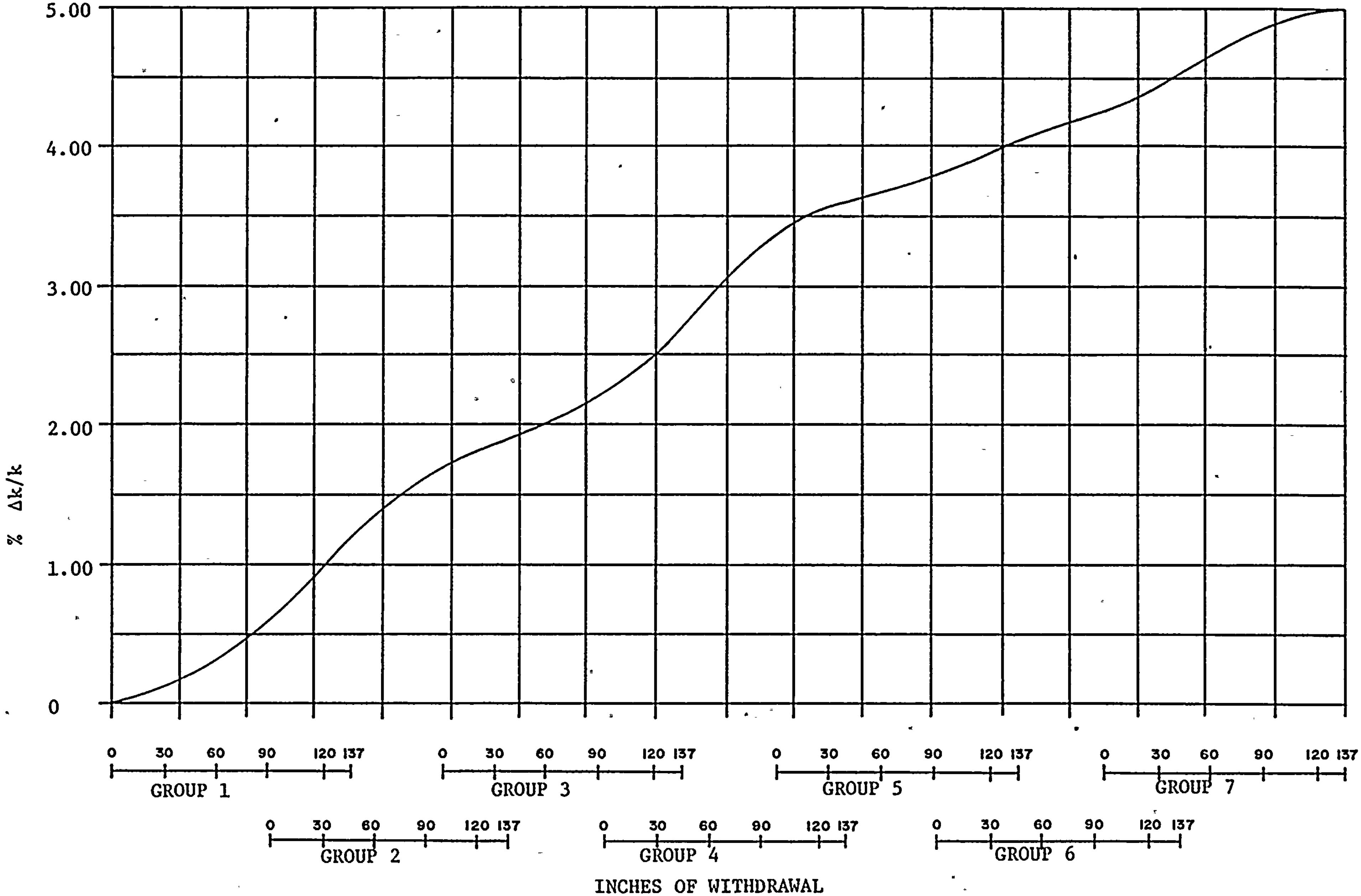


FIGURE 5.6-1

5.7 Pressure Coefficient of Reactivity Measurements

5.7.1 Purpose

The pressure coefficient of reactivity can be either negative or positive depending on the magnitude of the Reactor Coolant System boron concentration.

5.7.2 Test Results

The pressure coefficient of reactivity was not measured at PSL-1. However, the measurement was done at sister plants and the nominal value was -5×10^{-7} $\Delta k/k/psi$.

5.7.3 Conclusions

Results indicate that the pressure coefficient of reactivity is several orders of magnitude smaller than the temperature coefficient of reactivity and will be considered relatively insignificant.

5.8 Dropped CEA Worth Measurements

5.8.1 Purpose

A dropped CEA under power operation conditions will reduce reactor power level and distort the core power distribution. The reactivity worth of the most reactive CEA is measured from a full power CEA configuration in order to verify safety analysis.

5.8.2 Test Results

The dropped CEA integral reactivity worth measurement was performed simultaneously with a check of core symmetry. The integral reactivity worth of each CEA was measured using a CEA swap technique. This measurement was compared with that of all symmetric CEA's in its CEA Group in order to detect any unexpected core asymmetry. No significant asymmetry was noted and thereby gave additional assurance of proper assembly and core loadings.

The most worthy dropped CEA from a full power CEA configuration was CEA B-9. Its measured integral reactivity worth is compared with predicted worth in Table 5.8-1.

5.8.3 Conclusions

The integral reactivity worth of the most reactive dropped CEA from a full power CEA configuration was determined to be slightly more than predicted and was well within the acceptance criteria.

TABLE 5.8-1

SELECTED CEA WORTHS	
CEA SWAP TECHNIQUE	
CEA	REACTIVITY WORTH, ¢
7 - 1	10.80
7 - 38	6.71
7 - 68	5.92
6 - 14	11.85
5 - 3	11.74
4 - 50	9.55
3 - 69	5.62
2 - 20	9.38
1 - 33	10.32
B - 9	17.65
A - 49	10.90
P1 - 11	6.24
P2 - 35	6.41

	REACTIVITY WORTH					
CEA	BORATE	DILUTE	AVERAGE	AVERAGE	PREDICTED	LIMIT
B - 9	21.0¢	19.7¢	20.35¢	0.142 %Δρ	0.138 %Δρ	0.231%Δρ

5.9 Ejected CEA Worth Measurements

5.9.1 Purpose

F.S.A.R. safety analysis states that the maximum reactivity worth of any one ejected CEA in the core shall not exceed 0.92% $\Delta k/k$ at hot zero power at the beginning of core life and not exceed 0.23% $\Delta k/k$ at full power at beginning of life.

5.9.2 Test. Results

A pseudo ejected CEA reactivity measurement was made at the rounded zero power dependent insertion limit. CEA groups 7,6,5 and 4 were fully inserted; that is, at their lower electrical limits (LEL). The remaining CEA groups were fully withdrawn; that is, at their upper electrical limits (UEL). Certain CEA's from the three inserted CEA groups were selected and their respective reactivity worths measured. The measurement technique was of borating the first selected CEA to UEL, and then CEA "swapping" between the succeeding CEA's, to measure their integral reactivity worths. The most reactive CEA was 5-3. The results of this test are tabulated in Table 5.9-1.

A pseudo ejected CEA reactivity measurement was made at the full power dependent insertion limit. CEA group 7 was at approximately 74 inches. The other CEA groups were at their UEL's. CEA group 7 contains the center CEA and eight symmetrically located CEA's. The center CEA, and one other group 7 CEA, was measured for reactivity worth using the borating/swapping technique described in the preceding paragraph. The most reactive CEA was 7-1. The results of this test are tabulated in Table 5.9-2.

5.9.3 Conclusions

The measured values of both pseudo ejected CEA conditions were less than the predicted reactivity worths and were within the acceptable limits. Measured values were also less than the limit assumed (or set) by accident analysis.

	[redacted] TABLE 5.9-1	[redacted] TABLE 5.9-2
RCS TEMPERATURE, ^{o}F	532	532
RCS PRESSURE, PSIA	2250	2250
NUCLEAR POWER, %	<0.1	<0.1
POWER DEPENDENT INSERTION LIMIT	ROUNDED ZERO	FULL
CEA GROUP POSITION		
1	UEL *	UEL
2	UEL	UEL
3	UEL	UEL
4	LEL **	UEL
5	LEL	UEL
6	LEL	UEL
7	LEL	174 INCHES WITHDRAWN
CEA/WORTH (% $\Delta k/k$)	5-3 / 0.176	7-1 / 0.0322
CEA/WORTH (% $\Delta k/k$)	6-15 / 0.302	7/39 / 0.023
CEA/WORTH (% $\Delta k/k$)	7-1 / 0.083	--
PREDICTED WORTH (% $\Delta k/k$)	0.329	0.050
$\pm$ 25% PREDICTED WORTH	0.247 - 0.411	0.038 - 0.063
W/IN PREDICTED RANGE	YES	***
F.S.A.R. SAFETY ANALYSIS LIMIT (% $\Delta k/k$)	0.920	0.230
LESS THAN FSAR LIMIT	YES	YES

* Upper Electrical Limit
** Lower Electrical Limit
*** OUTSIDE OF RANGE, BUT STILL ACCEPTABLE PER C.E. LETTER F-SF-842 DATED 5/4/76

5.10 Stuck CEA Worth Measurement

5.10.1 Purpose

Technical Specifications state that available shutdown margin shall not be less than 2.45%, with the highest worth CEA stuck out, whenever the reactor is critical. The reactivity worth of the most reactive stuck CEA was measured in order to verify the validity of predicted stuck CEA reactivity worths.

5.10.2 Test Results

CEA A-48 was predicted to be the most worthy stuck CEA. The measurement technique consisted of CEA Group A combined trips with and without the "stuck" CEA stuck in the full out position. The reactivity data from these measurements was then used in combination with data from the CEA Group A reactivity worth measurement described in Section 5.5.2 to extrapolate an integral reactivity worth for CEA A-48. This preliminary on-site evaluation resulted in a measured stuck CEA reactivity worth of 3.109% $\Delta k/k$.

A subsequent off-site analysis of the measured data by CE-Windsor resulted in a stuck CEA reactivity worth of 3.146% $\Delta k/k$.

5.10.3 Conclusions

The preliminary on-site analysis of measured data indicated that CEA A-48 reactivity worth was within $\pm$ 25% of the design Hot Zero Power stuck CEA worth of of 2.7% $\Delta k/k$. Subsequent off-site analysis of data by CE-Windsor has verified this on-site conclusion.

5.11 Part Length CEA Group Measurements

5.11.1 Purpose

The two groups (P1 and P2)of part length CEA's are designed for axial power shaping purposes, for the control of quadrant tilt and for the control of offset. Their function is, as they are being withdrawn from LEL to UEL, to add negative reactivity, zero reactivity and finally positive reactivity.

5.11.2 Test Results

PSL-1 is administratively prohibited from moving the part length CEA's from UEL. The reactivity worths are not needed, and therefore, were not measured.

5.11.3 Conclusions

Measurements have been taken at other similar-design reactors and the results are available, if this information is ever needed.

6.0 Power Ascension Tests

The power ascension tests are conducted to determine the as-built plant characteristics during steady state and transient operations from 0% to 100% power and to demonstrate that the plant is capable of withstanding the accidents and transients analyzed in the FSAR. Tests requiring steady state power were to be performed at major plateau's of 20, 50, 80 and 100% power. Several other tests were to be performed at 14, 25, 30, 40, 60, 70, 75 and 85% power plateau's.

During power ascension testing an imbalance in the power pattern was discovered. At the beginning of the 80% plateau, it was decided to withdraw from power ascension testing.

The tests performed up through and including the 50% plateau are described in this start-up report, ref. Table 6.0-1. All other testing, including tests repeated due to the flux anomaly problem, will be described in one or more supplementing reports. See Section 7 for further discussion.

TABLE 6.0-1

PROCEDURE TITLE & NUMBER			% POWER						
(NOTE: P=Preoperational, O=Operating)			5	14	20	0	30	40	50
SIMULATED CEA EJECTION	P	0110087							1
STEAM BYPASS	P	0810080		1					
START UP TRANSFORMER	P	0910081			1				
GENERATOR EXCITATION	P	0910085		1					
NUCLEAR & ΔT POWER CALIB	O	1200051			2		2	2	2
GEN. TRIP OUTS. CONT. ROOM	P	1400093							1
PRIMARY CALORIMETRIC	O	3200020			2		2	2	2
FIXED INCORE ALARM SETPOINT	O	3200050				1			1
MOD TEMP COEF & PWR COEF	O	3200051							1
TOT. RAD. PEAK FACT. (F_r^T)	O	3200054							1
PWR RNG CONT SUBCH CALIB	P	3200080			1				1
FORCED XENON	P	3200087							1
RADIATION SHIELDING EVAL	P	3300081	1		1				1
CHEM & RADCHEM	P	3400081			1				1
S/G FEEDWATER HAMMER	P	0700080A					1		

6.1 Steam Bypass Valve Test and Turbine Generator Startup

6.1.1 Purpose

The purpose of the test was to verify the proper operation of the steam bypass valves and control system. This includes valve stroke time in the "quick-open" mode and automatic control of steam generator pressure. At the completion of this test the first turbine generator startup was then performed.

6.1.2 Test Results

All valves met or exceeded the specified 3 seconds maximum opening time. The valves operated satisfactorily in manual and then easily met the acceptance criteria of maintaining pressure within 25 psi of setpoint operating in the automatic mode. This portion of the test included using the steam bypass valves to control reactor power up to about 14% steam demand. The turbine generator startup was then performed satisfactorily. During this startup the steam bypass valves were used to control steam generator pressure.

6.1.3 Conclusions

The system met or exceeded all acceptance criteria. In addition, during its first actual operation it performed satisfactorily to control steam generator pressure during the initial Turbine Generator Startup. The turbine and generator performed properly although the generator was not loaded at this time due to other necessary testing (See Main Generator Excitation System Initial Operation, Section 6.2).

6.2 Main Generator Excitation System Initial Operation

6.2.1 Purpose

The purpose of this test was to perform initial energization of the Main Generator Excitation System and the Generator and verify that the system would:

(1) flash the main generator field

(2) increase and decrease voltage with the regulator out of service (base adjust) and in service (voltage adjust)

(3) transfer the regulator in and out of service and remove the generator field and excitation circuits from service.

6.2.2 Test Results

The test was performed after initial turbine generator startup and no problems of any significance were experienced. The Excitation System met all acceptance criteria (as listed above) and performed well in both manual and automatic modes of operation.

6.2.3 Conclusions

The Excitation System will function properly to maintain generator voltage (nominal 22,000 volts) under all expected normal and transient operational conditions.

6.3 20% Power Trip Test and Auxiliary to Startup Transformer Auto Transfer Test

6.3.1 Purpose

The purpose of this test was to measure plant response to a reactor trip from 20% power and to verify proper transfer of the 4160 and 6900 volt A.C. buses from the auxiliary (in plant) to the startup (off-site) transformer.

6.3.2 Test Results

The reactor trip initiated a turbine trip resulting in the interceptor steam valves and turbine valves closing and the generator field and output breakers opening. The automatic transfer functioned properly for the safety related buses but the non-safety related bus 1A1 breaker did not close to energize the bus from the startup transformer. A wiring error was discovered and corrected. On a later plant trip (2 days later) all buses transferred properly including the 1A1 bus. Required transfer time was 0.3 seconds or less and actual transfer times were 0.05 seconds or less. Decay heat was satisfactorily removed by the Steam Bypass Control System and the Feedwater Regulating System reduced feedwater flow to about 5% flow as desired. Also, Reactor Coolant System pressure was automatically controlled to less than 2500 psia (design pressure) throughout the transient.

6.3.3 Conclusions

The acceptance criteria (given above) were all properly met. The conditions at the completion of the transient were all as intended; RCS pressure less than 2400 psia, pressurizer cooldown less than 200°F per hour, Steam Generator pressure less than 985 psia, RCS cooldown less than 100°F/hour and electrical power being supplied by the startup transformers. This verifies that all involved systems performed properly to control a plant trip from 20% power.

6.4 Plant Power Calibration

6.4.1 Purpose

The purpose of the test was to:

(1) Determine core thermal power by means of a primary plant heat balance.

(2) Adjust the Power Range Safety Channels and ΔT Power Reference Calculators to agree with the thermal energy balance calculations.

(3) Perform, when necessary, a calibration of the Safety and/or Control Power Range Channels.

6.4.2 Test Results

Feed flow Calorimetrics using hand calculations were conducted at the 20 and 50% power plateaus. The Calorimetrics were used to calibrate nuclear instrumentation and to verify the plant computer core thermal power calculations. The Power Range Safety Channels and ΔT Power Reference Calculators were adjusted to agree within 0.5% of the Calorimetric calculations. These adjustments were performed at the 20, 30, 40 and 50% power plateaus.

Initial calibration of the Power Range Safety Channels was conducted during the 20% test plateau using the Keithley pico-ammeters as a standard for subchannel calibration. Adjustment of the Power Range Control Channels was also completed at the 50% power plateau.

6.4.3 Conclusions

Hand calculations of core thermal power pointed out some minor discrepancies in the computer calculations. After correction of the deficiencies, computer calculations proved to be reliable and accurate. Calibration of the Power Range Safety and Control Subchannels was accomplished acceptably at each major test plateau. The intent of the Ex-Core Nuclear Instrument Calibration was to adjust nuclear power, ΔT power and the Calorimetric to within 0.5% of each other. Plant operating procedures contain instructions for hand calculations in case of computer failure and/or to verify computer calculations.

FIGURE 6.4-1

DDPS CALORIMETRIC FOR NUCLEAR & ΔT POWER CALIB. O.P. 1200051		
DATE	NOMINAL REACTOR POWER, %	CALORIMETRIC POWER %
5-8-76	20	20.76
5-9-76	20	22.55
5-12-76	20	19.23
5-13-76	30	30.06
5-14-76	30	30.12
5-15-76	40	39.76
5-20-76	50	50.63
5-20-76	50	48.73
5-21-76	50	49.00
5-23-76	50	47.69
5-24-76	50	49.80
5-25-76	50	49.80
5-26-76	50	50.07
5-27-76	50	50.19

PRIMARY CALORIMETRIC, OPERATING PROCEDURE 3200020		
NOMINAL REACTOR POWER, %	20	50
Qc, BTU / HR	1.971 e 9	4.110 e 9
ACTUAL REACTOR POWER, %	23.7	49.6
DATE	5-9-76	5-20-76

6.5 Power Range Safety and Control Subchannel Calibration

6.5.1 Purpose

The purpose of this procedure was to calibrate the upper and lower power range excore detectors and provide (input) Shape Annealing Factors (SAF's) to be used by the Subchannels to calculate the Axial Shape Indices (ASI's).

The excore power range detectors must be accurately calibrated in order to monitor the in-core flux profiles as determined by the incore flux detectors. At 20% power, conservatively assumed values of SAF were input to allow operation until 50% power. At 50%, actual observed SAF's were input. In both cases the ASI's calculated from the Safety Subchannels signals to the Reactor Protection System were required to be within 5% of the computer calculated values based on the incore detector signals.

6.5.2 Test Results

All Subchannels calibrated properly at both 20% and 50% power although it was noted that detector signals were somewhat lower than expected requiring amplifier gain to be increased for proper operation and calibration. Using the assumed values of SAF, the RPS calculated values of ASI were well with the 5% limit when compared to the actual computer calculated ASI's. And, using the observed SAF values, the RPS calculated values of ASI agreed quite well with the actual ASI's (well within the 5% tolerance).

6.5.3 Conclusions

This procedure is used periodically to calibrate the subchannels and verify that the outputs are correct and the results confirmed that the procedure and equipment will properly calibrate the Power Range Subchannels. Also this procedure assured safe operation between 20% and 50% power and at 50% power verified that the ASI's calculated by the Power Range Excore Subchannels are in fact an accurate representation of the actual in-core flux profiles.

6.6 Shielding Effectiveness and Plant Radiation Level Measurements

6.6.1 Purpose

The test was conducted to accomplish the following objectives:

(1) Determine background radiation levels prior to plant startup.

(2) Evaluate the adequacy of plant radiation shielding.

(3) Determine radiation levels throughout the plant at various power levels.

6.6.2 Test Results

A comprehensive series of gamma and neutron dose rate level surveys was performed during initial startup, low-power-physics testing and power escalation. Dose rates at selected points, both inside and outside of the Radiation Controlled Area, were determined. Dose rates at each point were determined at power levels of 0% (Background), $1x10^{-3}$%, 5%, 20% and 50%. The final survey planned for the 100% power level has not been completed, since the facility has not operated at full power as of December, 1976.

Radiation dose rate levels at each point were compared for different power levels to verify that a linear relationship existed. This was done to ensure an extrapolation of dose rates to 100% power could be considered valid allowing for identification of potential problem areas. It was also felt that it would be in keeping with ALARA* to limit dose for personnel performing surveys in areas where access during full-power operation would be highly restricted due to high and possibly variable dose rates.

In addition to personnel performing surveys, two special neutron monitoring systems were available allowing dose rate determinations to 20 Rem/hour (neutron only).

Design dose rates utilized for comparison are specified in the St. Lucie FSAR, Chapter 12, Section 12.1.1.

General area gamma dose rates for all areas around the containment were less than 0.1 mrem/hour and neutron levels were less than 0.5 mrem/hour with the exception of the personnel hatch and the two entrance doors to the containment annulus. These measurements do not include electrical and mechanical penetration areas in the reactor auxiliary

* ALARA - As Low As Reasonably Achievable

6.6 <u>Shielding Effectiveness and Plant Radiation Level Measurements</u> (Cont.)

6.6.2 Test Results (Cont.)

building or the fuel handling building or spent fuel pool area. These enclosed equipment spaces may experience variable dose rates. Therefore, in addition to the initial measurements which verified that these areas were not Radiation Areas, they are surveyed periodically to ensure proper control on a continuous basis.

Dose rates extrapolated to 100% power indicated approximately 3 mrem/hour and 10 mrem/hour contact with the southwest and northeast annulus doors respectively (combined n,γ dose rates). At the containment entrance hatch combined neutron and gamma dose rates extrapolated to 100% power indicated 1 mrem/hour outside the airlock, 22 mrem/hour in the airlock, and 80 mrem/hour in the containment at the airlock exit. Neutron/gamma dose rate ratios in containment were variable from approximately 1 to as high as 20 at some points along the cavity edge.

6.6.3 Conclusions

Maximum general area gamma and neutron dose rates as determined by shielding effectiveness surveys at the St. Lucie Plant were generally consistent with criteria presented in Section 12.1.1 of the FSAR for areas outside the exterior containment wall and outside the Radiation Controlled Area boundary. The presence of possible projected neutron streaming problems in the containment, RAB and personnel emergency escape hatch areas was reinforced by the results of the surveys performed to date.

As noted in FSAR Section 12.1, the analytical and empirical results to date indicate the need for shielding and so the design effort for St. Lucie 1 has preceded the measurement program. Following completion of the St. Lucie 1 neutron streaming measurement program, the adequacy of the proposed shield design will be evaluated against the actual measured data. The proposed reactor cavity neutron shield consists of nylon-neophrene covered bags holding ordinary light water (non-borated). The water bags will be fabricated by the B. F. Goodrich Company. The bags are vertically installed by prefolding them in a manner such that no gaps will exist between bags as the bags are filled.

The 100% power level shielding effectiveness survey will be performed as soon as practicable after the unit reaches full power operation and will be repeated following installation of neutron shielding currently being considered to reduce neutron streaming from the gap around the reactor vessel.

6.7 Chemistry and Radiochemistry at Power

6.7.1 Purpose

1) To ensure that the primary and secondary systems water chemistry meet the criteria set forth in the St. Lucie I Chemistry Procedures Manual for system protection.

2) To correlate corrosion data and fission product buildup data to power levels.

6.7.2 Test Results:

Chemical and Radiochemical tests were performed as specified in the Chemistry Procedures Manual and Preoperation Test Procedure No. 3400081 (Chemical and Radiochemical Analyses 20, 50, 80, 100% power). The plant reached 80% power for a short period of time and was shutdown due to a flux tilt problem. Meaningful data was obtained for steady state powers of 20 and 50%. (Table 6.16-1).

6.7.2.1 Primary

1) All parameters were within limits except Lithium which was intentionally high for better filming during initial power operations.

2) A small fuel failure was suspected shortly before plant shutdown. On 6-22-76 the iodine ratio began increasing. Several days passed before a significant increase in Dose Equivalent Iodine 131 was seen. On 6-30-76 at 80% power the iodine ratio reached 0.4 and Dose Equivalent I131 reached 1.04×10^{-3} uc/ml. Also an increase in VCT fission gases was noted. The reactor was shutdown soon after so conclusive data is not yet available.

6.7.2.2 Secondary

1) Moisture carryover test was not performed due to premature plant shutdown.

2) Maximum steam generator blowdown was maintained when possible, for the major part of power testing.

6.7 (cont.)

6.7.2.2 (cont.)

3) Condensate and Feed Systems were flushed until solids were as low as possible prior to feeding the steam generators.

4) It was found that pH additives were not necessary to control steam generator or feed pH while maintaining a feed system hydrazine residual of 30-40 ppb.

5) Suspended solids in the Feed System and steam generators were lower than expected until the mechanical shock of the High Volume Steam Dump Valves drove Feed Solids to 0.8 ppm and Steam Generators 1A to 44 ppm and 1B to 51 ppm. Maximum blowdown was maintained resulting in a decrease in solids to specification.

6) At one point silica increased significantly and steam generator sample water became frothy. This is believed to have resulted from initial condensate cleaning chemicals that could not be flushed during flush evolutions at that time. Maximum blowdown was used to remedy this problem.

6.7.3 Conclusions

Chemical Testing, both primary and secondary, showed that the plant was in a generally good condition. It also showed that chemistry could be controlled within specified limits and that out of specification parameters could be remedied in a timely manner.

TABLE 6.7-1

RCS CHEMISTRY RESULTS AT STEADY STATE POWER

Analysis	Limits	5-10-76 20%	5-26-76 50%
pH	4.5-10.2	6.28	6.7
Conductivity	Varies	8.3	16
Cl^-	$\leq$.15 ppm	<.05	<.05
F^-	$\leq$.1 ppm	<.05	<.05
D.O.	$\leq$.1 ppm	.005	<.005
S.S.	<.5 ppm	<.01	<.01
Boron	Varies	822	714
Lithium	.2-2.0 ppm	.73	1.49
Diss. H_2	10-50 cc/kg	31.8	47.2
Gas Act.	---	$7.32e^{-4}$	$6.99e^{-3}$
Gross Act.	---	$1.81e^{-2}$	$4.3e^{-2}$
Crud Act.	---	$5.2e^{-6}$	$2.1e^{-4}$
Tritium	---	$5.6e^{-3}$	$1.7e^{-2}$
I ratio	---	.07	.08
DEq I-131	$\leq$1 uci/gm	$4.4e^{-4}$	$2.5e^{-4}$
Spectrum	---	performed	performed

6.8 Fixed Incore Detector Alarm Setpoints

6.8.1 Purpose

This was not a test, but more of an instrumentation setpoint procedure. The purpose was to calculate and adjust the fixed incore detector alarm setpoints. The core is considered to be divided into four axial regions, each approximately one fourth the core height, and each encompassing the axial region monitored by one segment level of the incore detector.

6.8.2 Results

The procedure instructs that a set of readings of the measurement of the incore detector readings, various temperatures, and CEA heights be taken. The plant computer has a program called "SNAPSHOT" that will do this. Via hand calculations or a computer program called "GINCA", the Nodal overpower ratios are calculated. Through various factors and constants, an incore detector alarm setpoint is generated for each of the 180 incore detectors. This was repeated periodically as power level was increased to ensure up to date setpoints.

At the beginning of the 80% plateau, the setpoints were readjusted due to discovery of STRIKIN II computer code errors (See Section 6.13). An incorrect adjustment factor was used in the setpoint calculations resulting in nonconservative setpoints. This was noted about 8 hours later and corrected. Review of the data verified no actual setpoints were violated. (LER 335-76-34, August 6, 1976)

6.8.3 Each of the alarm setpoints was entered into the Digital Data Processor. The operability of the Digital Data Processor was checked, and each of the fixed incore detectors was determined to have a valid alarm set point.

6.9 Reactivity Coefficient Measurements

6.9.1 Purpose

A test commonly referred to as a Variable T avg Test was conducted to determine the Power Coefficient and the Isothermal Temperature Coefficient (ITC).

6.9.2 Test Results

Variable T avg Tests were conducted at the 50% power plateau with the Control Element Assemblies (CEA) inserted to approximately 100 inches on CEA Group 7. The test was conducted with the Power Coefficient and the Isothermal Temperature Coefficient (ITC) as separate tests. During the ITC test, ΔT power was held constant and Reactor Coolant System (RCS) T avg was varied. T avg was decreased $10^{o}F$ below original temperature, conditions stabilized, data recorded and temperature increased to original temperature, conditions stabilized, and data recorded. This cycle was repeated twice.

The Power Coefficient Test was conducted by holding T avg constant and decreasing gross electrical power by approximately 5%, conditions stabilized, data recorded. Then gross electrical power was increased to the original power level, conditions stabilized and data recorded. This cycle was repeated twice. The final power coefficient and ITC values were the average value of the runs conducted. The measured value and limits for the temperature and power coefficient for the 50% power plateau are shown in Table 6:9-1.

6.9.3 Conclusions

The measured value for Isothermal Temperature Coefficient was well below its limit. The Power Coefficient was within the tolerance of its limit.

TABLE 6.9-1

NOMINAL REACTOR POWER	50%

MODERATOR TEMPERATURE COEFFICIENT	$\Delta k/k$ / ^{o}F
MEASURED	-0.10×10^{-4}
LIMIT	$<+\ 0.50 \times 10^{-4}$

POWER DEFECT COEFFICIENT	$\Delta k/k$ / %
MEASURED	-1.07×10^{-4}
LIMIT	$(-1.0+0.1)\times10^{-4}$

6.10 Total Radial Peaking Factor

6.10.1 Purpose

This was done at 50% power to measure the total radial peaking factor, F_r^T. F_r^T is defined as the product of the unrodded planar peaking factor and the quantity one (1) plus the azmuthal tilt. $F_r^T = F_r^P \times (1 + Tq)$.

6.10.2 Test Results

This was more of a measurement than it was a test. As stated in Section 6.0, a power imbalance became apparent during the power ascension testing sequence. This imbalance grew very apparent during testing at power levels greater than 50%. Selected data is presented in Table 6.10-1.

6.10.3 Conclusion

The total radial peaking factor measurement done at the 50% power level proved that the plant was operating below the Technical Specification limit of 1.36. However, as previously stated, the azmuthal tilt/ total radial peaking factor at higher power levels was serious enough to warrent withdrawal from the power ascension testing program.

TABLE 6.10-1

TOTAL RADIAL PEAKING FACTOR

DATE	5/21/76
POWER, %	50
Tq	0.006
F_r^T	1.32

6.11 Xenon Follow·Measurements

6.11.1 Purpose

The purpose of this test was to obtain transient test data at the 50% power plateau for the purpose of evaluating the Shape Annealing Factor (SAF) for each Power Range Safety Channel and to evaluate an induced free xenon oscillation.

6.11.2 Test Results

During the 50% test plateau, axial oscillations were induced in the core. These oscillations were monitored by the In Core Analysis GINCA computer program and the Axial Shape Index (ASI) calculated by the Reactor Protective System (RPS). All axial oscillations were convergent.

The SAF for each Power Range Safety Channel was also measured. The SAF corrects the detector signal to account for the distance from the excore detector to the reactor core and corrects for the signal received by the upper detector from neutrons generated in the bottom of the core and the signal received by the lower detector from neutrons generated in the upper part of the core. The SAF is determined by plotting ASI GINCA versus the ASI (EXT) as read from the Reactor Protective System (RPS) during a xenon oscillation with all CEA's full out. The slope of the line resulting from this plot is the Shape Annealing Factor. ASI is Axial Shape Index, a ratio of the difference in power generated in the lower and upper halves of the core to total core power. Table 6.11-1 summarizes the comparisons between F.P.&L. and C.E. results. As measured SAF's were incorporated into RPS setpoints.

6.11.3 Conclusions

The induced xenon oscillation test proved that induced xenon oscillations were self dampening to a stable power distribution and that resultant DNBR and LHR were both well within acceptance limits.

The ASI's calculated by the RPS as well as GINCA were continuously observed and evaluated. Observation verified the adequacy of SAF's previously determined from measured data.

TABLE 6.11-1

CHANNEL	SHAPE ANNEALING FACTOR (SAF)		INTERCEPT	
	F. P. & L.	C. E.	F. P. & L.	C. E.
A	3.45	3.46	-0.0004	-0.0012
B	3.76	3.77	+0.0019	+0.0009
C	3.89	3.92	+0.0090	-0.0102
D	3.06	3.07	-0.0010	-0.0015
9	3.21	3.21	+0.0010	0
10	3.67	3.68	+0.0070	+0.0060

TABLE 6.11-2

DATE	TIME	A. S. I.	CEA GP 7	DATE	TIME	A. S. I.	CEA GP 7
29MAY76	2130	+0.011	ARO	30MAY76	2314	-0.149	ARO
30MAY76	0624	+0.011	133	31MAY76	0020	-0.146	ARO
	0721	+0.168	87		0116	-0.136	ARO
	0813	+0.281	68		0200	-0.123	ARO
	0914	+0.312	68		0300	-0.107	ARO
	1016	+0.334	68		0400	-0.087	ARO
	1113	+0.345	68		0500	-0.067	ARO
	1220	+0.337	73		0600	-0.044	ARO
	1317	+0.265	88		0700	-0.024	ARO
	1413	+0.133	116		0800	-0.004	ARO
	1516	+0.056	ARO		0900	+0.017	ARO
	1645	+0.005	ARO		1000	+0.034	ARO
	1715	-0.033	ARO		1100	+0.048	ARO
	1817	-0.075	ARO		1200	+0.061	ARO
	1915	-0.102	ARO		1600	+0.079	ARO
	2020	-0.125	ARO	01JUN76	1236	-0.010*	ARO
	2110	-0.135	ARO	02JUN76	0812	+0.017**	ARO
	2214	-0.145	ARO	* MIN. VALUE		** MAX. VALUE	

ST. LUCIE UNIT 1
B.O.L., 1ST CYCLE, 532^{o}F, 2250 PSIA
A.S.I. vs TIME

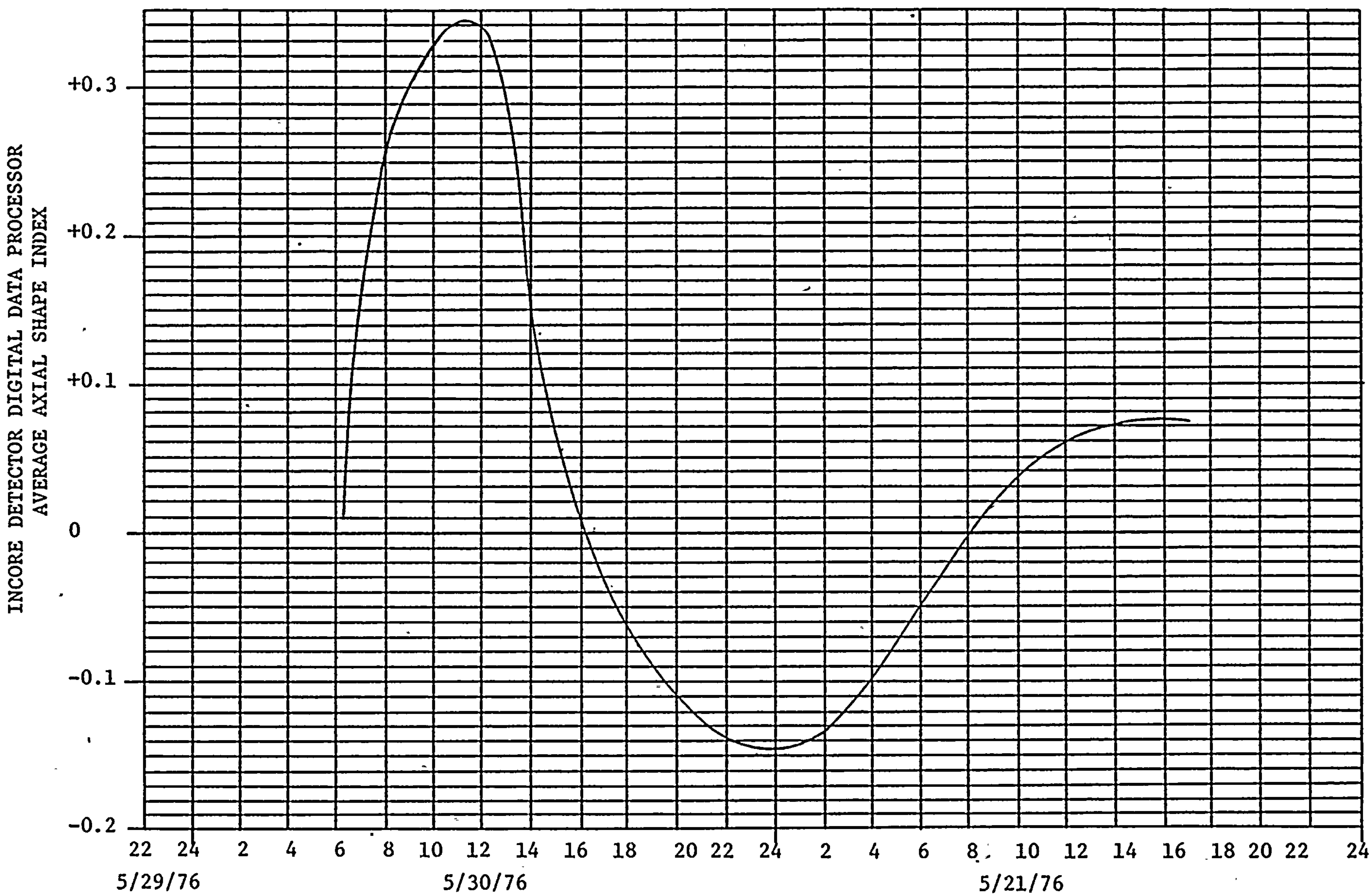


FIGURE 6.11-1

6.12 Pseudo Ejected CEA Power Distribution Measurement

6.12.1 Purpose

The purpose of this test is to measure the core power distribution resulting from a CEA being "ejected" from a full power dependent insertion limit (FPDIL) CEA configuration. The measurement results are compared with that predicted by Safety Analysis to verify its conservatism.

6.12.2 Test Results

When CEA's are inserted to the full power PDIL, only CEA Group 7 is in the core. With respect to relative reactivity worth, CEA Group 7 consists of three (3) types of CEA's. The center CEA and one peripheral CEA was ejected from the full power configuration and appropriate power distribution information collected using the following technique. Equilibrium xenon was established at a nominal 50% power level, and then, core power distribution information gathered from the in-core neutron detectors. Two of CEA Group 7 CEA's were "ejected" from the core; the first by a soluble boron swap technique; and the other by a rod swap with the preceding CEA. Power distribution information from the in-core detectors was gathered at the full-out condition for each CEA. Power level was maintained at 50% throughout the test.

A three-dimensional power peaking factor was developed from measured data by extrapolating in-core detector signals to the "ejected" CEA locations. As had been expected, this resulted in a power peaking factor considerably below that predicted in safety analysis. Safety analysis assumed an ejected CEA worth which was an order of magnitude greater than that measured during Low Power Physics Testing. Results of the onsite analysis for each CEA are presented in Table 6.12-1.

6.12.3 Conclusions

A preliminary on-site analysis of data indicates that the 3-D power peaks resulting from pseudo ejected CEA power distribution measurements are considerably smaller than those calculated in the Safety Analysis. The results of this on-site evaluation when coupled with the results of ejected CEA worth measurements made during LPPT confirm that the Safety Analysis of the CEA Ejection Incident is very conservative.

TABLE 6.12-1

DATE	JUNE 11, 1976
NOMINAL RCS TEMPERATURE	548.7 °F
NOMINAL RCS PRESSURE	2253.4 PSIA
NOMINAL NUCLEAR POWER	49.8 %
3-D FUEL ROD PEAK, Fq	
CEA 7-1	1.99
CEA 7-59	1.97
LIMIT	$\leq$ 5.10

6.13 Core Power Distributions

6.13.1 Purpose

Detailed core power distribution measurements were performed under steady state conditions during PAT to verify that fuel assembly power fractions, axial power distributions and peak linear heat rates were within acceptable limits.

The specific acceptance criteria applied to the measured core power distributions are listed below.

(1) Fuel assembly power fraction: Fuel assembly power fraction is defined as the ratio of the average Linear Heat Rate (LHR) in a fuel assembly to the average LHR over the entire core.

(2) Axial power distribution: The measured core average axial power distribution shall be compared with the predicted distribution for general agreement.

(3) Peak linear heat rate: It shall be less than 12.7 KW/ft.

6.13.2 Test Results

A summary of core power distribution results is presented in Table 6.13-1.

6.13.2.1 Fuel assembly power fraction: Steady state equilibrium xenon core power distribution measurements were performed at 20% and 50% power plateaus. The definition of equilibrium xenon used in this section is:

> The change in critical boron concentration as determined from Reactor Coolant System chemical analysis, shall be less than 1% deviation from the average, of 3 continuous samples, taken 15 minutes apart.

The analysis of the incore detector readings is performed by two computer programs. The first program automatically converts the voltage signal from the detector to the correct neutron flux level. The Incore Analysis(GINCA) program converts the neutron flux levels and various other reactor parameters and, on demand, calculates several incore data. The GINCA program assumes eighth

core symmetry. The two cases reported in this section are shown in Tables 6.13.1 through 6.13.3.

6.13.2.2 Axial power distributions: At steady state equilibrium xenon, the core average axial power distribution was determined at each of the major test plateaus using the GINCA program. Figures 6.13-2 and 6.13-3 display the measured values of core average axial power distribution.

6.13.2.3 Peak LHR: The peak LHR is determined by the GINCA program. As can be seen in Table 6.13-1 the peak LHR never exceeded the acceptance criteria of 12.7 KW/ft. If the worst case of 5.33 KW/ft is multiplied by 1.2991 to account for uncertainties, the resultant 6.9242 KW/ft is still acceptable. These uncertainties include measurement-calculational uncertainty, an engineering factor, effects of fuel densification and thermal expansion, and power measurement uncertainty.

6.13.3 Conclusions

At steady state equilibrium xenon, the axial and radial core power distributions are within acceptable limits. The peak linear heat rate does not exceed that allowed by Technical Specifications. It should be noted that the limiting peak LHR of 12.7 KW/ft is an interim limit imposed due to the discovery of coding errors in the vendor's STRIKIN II computer code. Results of the first corrected analysis (worst case LOCA) indicate that, when full reanalysis is complete, the permanent limit for peak LHR will be higher thus making these results even more conservative.

TABLE 6.13-1

MEASURED POWER, %	19.38	49.64
CORE BURNUP, EFPH	13	200
BORON CONCENTRATION, PPM	827	730
LINEAR HEAT RATE, KW/FT	2.18	5.33
RADIAL PEAK, F_r^T	1.3257	1.3414
AXIAL PEAK, F_z	1.296	1.336
NOMINAL RCS TEMP, ^oF	537	538
NOMINAL RCS PRESSURE, PSIA	2250	2245
CEA CONFIGURATION	ARO	ARO
TIME	2100	2130
DATE	5-11-76	5-29-76

POWER	19.38%
TIME	2100
DATE	5-11-76

TABLE 6.13-2

Fuel Type Relative Power Density Linear Heat Rate	I-C-0 0.662 1.619

		I-C-0 0.662 1.619	I-C-2 1.109 1.969					
	I-C-0 0.563 1.459	I-C-2 1.060 1.900	I-B-0 1.054 1.800	I-A-0 1.006 1.669				
	I-C-0 0.767 1.616	I-B-0 0.993 1.726	I-A-0 0.992 1.648	I-B-0 1.122 1.856	I-A-0 1.059 1.742			
	I-C-1 1.021 2.007	I-A-0 0.980 1.628	I-B-0 1.104 1.827	I-A-0 1.050 1.729	I-B-0 1.166 1.906	I-A-0 1.084 1.779		
I-C-0 0.588 1.522	I-C-1 1.149 1.925	I-B-0 1.084 1.796	I-A-0 1.033 1.701	I-B-0 1.153 1.885	I-A-0 1.081 1.755	I-B-0 1.187 1.919	I-A-0 1.097 1.800	
I-C-0 0.766 1.606	I-B-0 1.009 1.714	I-A-0 1.001 1.646	I-B-0 1.125 1.843	I-A-0 1.063 1.744	I-B-0 1.174 1.970	I-A-0 1.097 1.800	I-B-0 1.193 1.935	I-A-0 1.088 1.785

POWER	49.64%
TIME	2130
DATE	5-29-76

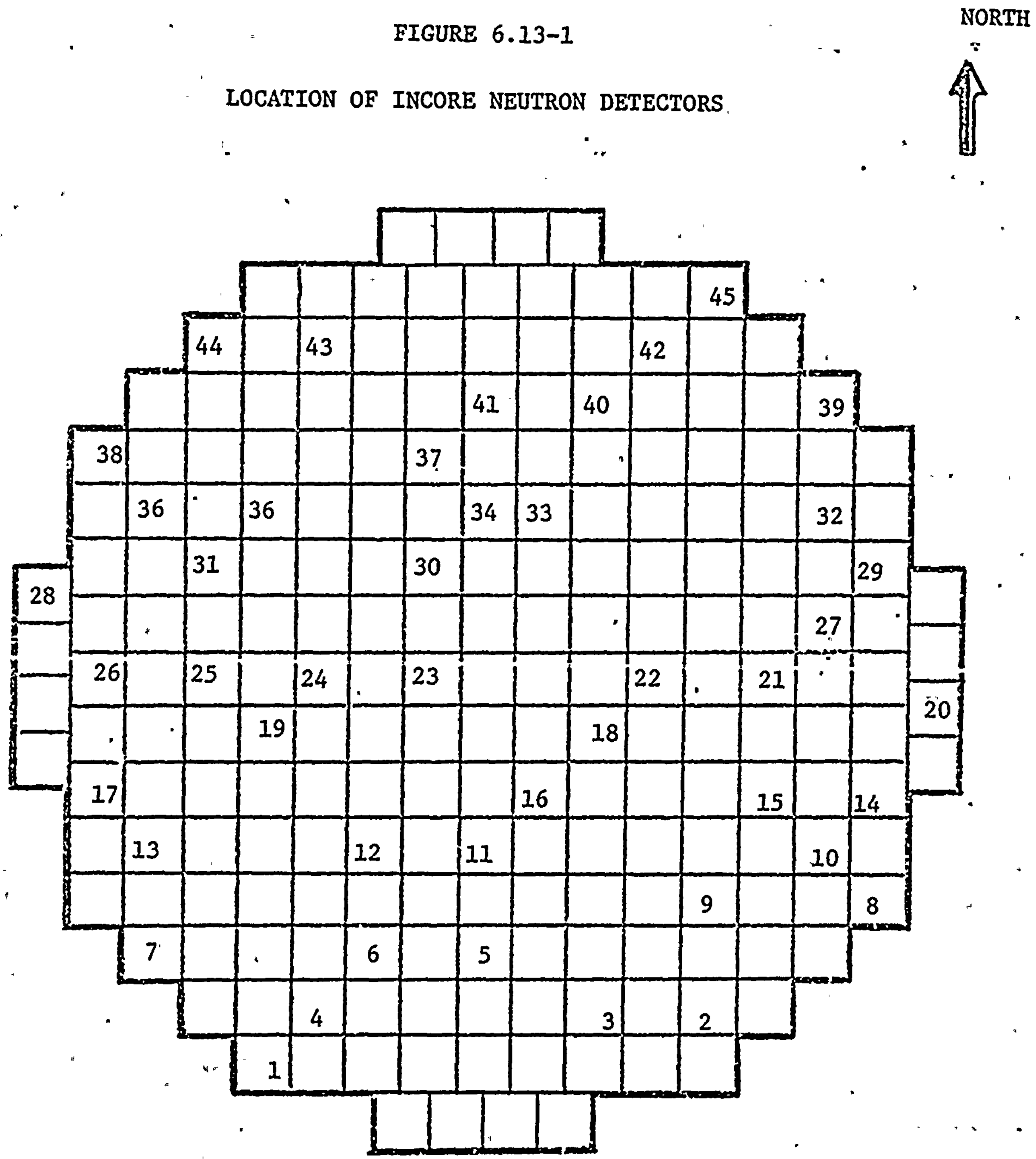
TABLE 6.13-3

Fuel Type	I-C-0
Relative Power Density	0.691
Linear Heat Rate	4.240

		I-C-0 0.691 4.240	I-C-2 1.067 4.712					
	I-C-0 0.596 3.891	I-C-2 1.024 4.654	I-B-0 1.004 4.370	I-A-0 1.049 4.376				
	I-C-0 0.829 4.460	I-B-0 0.952 4.199	I-A-0 1.033 4.318	I-B-0 1.081 4.490	I-A-0 1.092 4.524			
I-C-0 0.623 4.125	I-C-1 1.027 5.108	I-A-0 1.038 4.357	I-B-0 1.065 4.465	I-A-0 1.086 4.501	I-B-0 1.114 4.622	I-A-0 1.112 4.615		
I-C-0 0.814 4.318	I-C-1 1.132 4.913	I-B-0 1.044 4.426	I-A-0 1.071 4.459	I-B-0 1.097 4.519	I-A-0 1.086 4.458	I-B-0 1.122 4.637	I-A-0 1.124 4.647	
	I-B-0 0.995 4.665	I-A-0 1.049 4.455	I-B-0 1.080 4.517	I-A-0 1.091 4.526	I-B-0 1.114 4.620	I-A-0 1.118 4.633	I-B-0 1.138 4.660	I-A-0 1.131 4.663

FIGURE 6.13-1

LOCATION OF INCORE NEUTRON DETECTORS



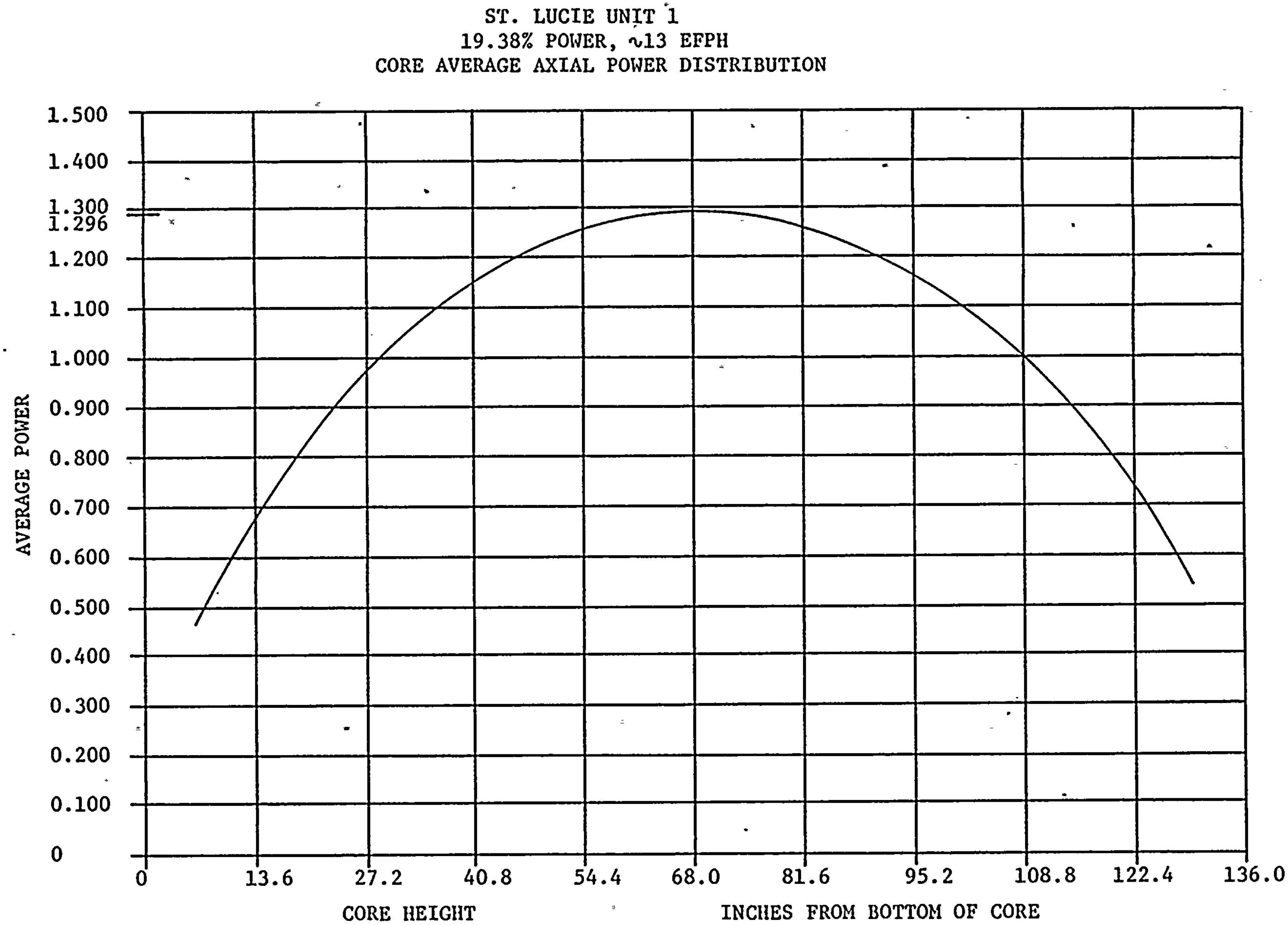

ST. LUCIE UNIT 1
19.38% POWER, ∿13 EFPH
CORE AVERAGE AXIAL POWER DISTRIBUTION
AVERAGE POWER
1.500
1.400
1.300
1.296
1.200
1.100
1.000
0.900
0.800
0.700
0.600
0.500
0.400
0.300
0.200
0.100
0
0
13.6
27.2
40.8
54.4
68.0
81.6
95.2
108.8
122.4
136.0
CORE HEIGHT
INCHES FROM BOTTOM OF CORE

FIGURE 6.13-2

ST. LUCIE UNIT 1
49.64% POWER, ~200 EFPH
CORE AVERAGE AXIAL POWER DISTRIBUTION

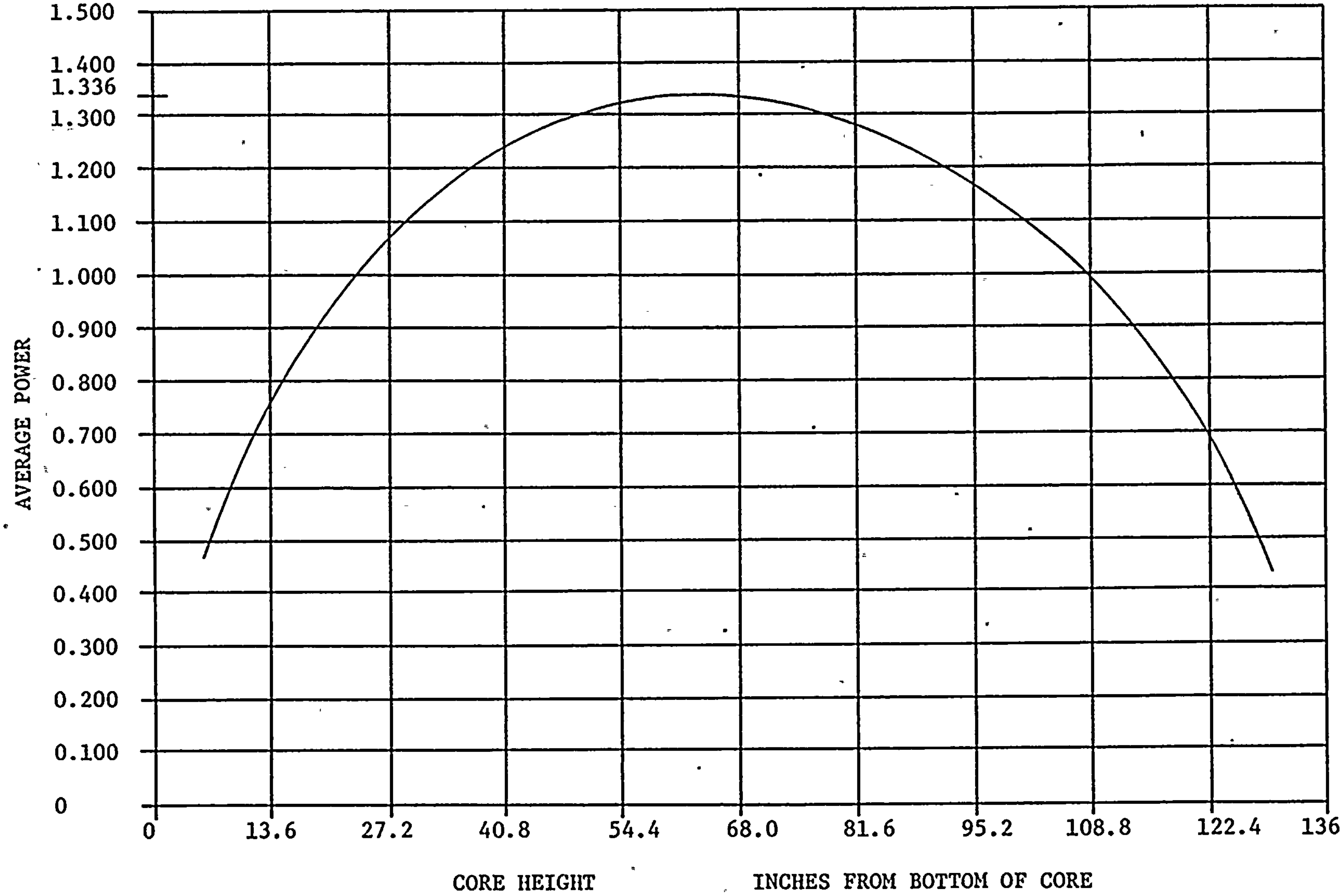


FIGURE 6.13-3

6.14 GENERATOR TRIP WITH SHUTDOWN OUTSIDE CONTROL ROOM

6.14.1 Purpose

The purpose of this test was to demonstrate that the plant responds properly to a generator trip from 50% power and that it can be safely shutdown to hot shutdown conditions from outside the control room without exceeding any safety limits.

6.14.2 Test Results

The acceptance criteria were:

(1) Successful shutdown of the plant from outside the control room by the normal complement of plant operators without exceeding any safety limits, including a check that all automatic trip related functions did, in fact, occur.

(2) Emergency communications successfully established between the Hot Shutdown Control Panel and various local operating stations .

(3) Successful boration (at least 10 ppm) from the remote operating station .

(4) Satisfactory removal of decay heat using the steam dump and maintaining reactor T avg at or below the hot standby temperature.

(5) Satisfactory control of steam generator level by manual control of the auxilliary feedwater sys - tem.

All the acceptance criteria were met with no significant problems.

6.14.3 Conclusions

Satisfactory completion of this test demonstrates that the plant can be safely controlled and shutdown should the control room become inaccessible.

6.15 Steam Generator Feedwater Hammer Test

6.15.1 Purpose

The purpose of this test was to verify the absence of any water hammer in the steam generator feedwater piping when the steam generator was drained below the feed ring and then refilled.

6.15.2 Test Results

Following a trip from 33% reactor power, level in one steam generator was reduced below the feed ring and held at that level or lower for 2 hours. Then the level was raised to normal at the maximum flow rate (300 gpm) of one Auxiliary Feedwater pump. The behavior of the feedwater piping was monitored by: observers inside containment, installed RCS noise monitoring equipment and by measurements of line and restraint positions before and after the test. No evidence of feedwater hammer was observed.

6.15.3 Conclusions

The absence of any evidence of steam generator feedwater hammer indicated that St. Lucie Unit 1 was not susceptible to the problem.

Since performance of this test, the method of preventing the feedring from draining has been modified in accordance with 10CFR50.59. Therefore this test will be repeated upon return to power operations.

7.0 UNUSUAL EVENTS

During the time interval covered by Startup Testing, several major problems occurred which significantly affected testing. The first 4 problems failed to delay testing only because the fifth, and most important one, created a delay of approximately six months. The problems were:

(1) Higher than predicted cooling water discharge canal levels
(2) Inability to operate CEDM 44 when cooled down
(3) Apparent lower than predicted Reactor Cooling Pump Flow
(4) Higher than predicted Containment radiation levels
(5) A power distribution anomaly

These problems and their resolution are discussed in greater detail in the following Sections 7.1 through 7.5 respectively.

7.1 Higher Than Predicted Cooling Water Discharge Canal Level

When all cooling water systems began to operate simultaneously, discharge canal levels were found to be higher than expected. When this was observed to threaten spill-over from the canal banks (especially at high tide), evaluation was begun.

Flow determinations for the main circulating water pumps were performed by FP&L's Power Resources Test Group to see if the pumps produced greater than design flow. This was, in fact, true so the pump discharge valves were throttled to just above design flow rate. However, this was not the only problem as canal level still threatened to spill over the banks during high tide.

More extensive investigation/evaluation was undertaken. This included having divers photograph the interior of the canal outfall pipe. This 12 foot concrete pipe extends from the canal's end, under the dune line and beach for 1200 feet out to sea. This pipe had been completed for nearly a year with virtually no flow through it. The photographs and TV video tape proved that there was extensive marine growth on the interior surfaces. In places this fouling was up to an inch deep. Under normal operating conditions, flow through the pipes and the small residual chlorine level allowed would help reduce this fouling although it cannot be entirely eliminated.

Although it was felt that this fouling was not the only problem, it was obvious that the growth must be removed to allow proper evaluation of the system. Therefore, Florida Power & Light Personnel designed and had fabricated a large pipe cleaning machine. During the early part of the shutdown for resolution of the power distribution anaomaly, the pipe was cleaned. Once this was completed, the main circulating water pumps (turbine condenser cooling water) were run as much as possible to help prevent recurrence of the fouling. There were no cases of threatened spill-over observed for 2 months.

The LPPT and initial ascent to power (December 1976) coincided with a period of higher than average tides compounded by a weather front also causing higher tides. Under these conditions, it again proved necessary to reduce cooling water flow to avoid spill over. Again divers photographed the interior of the pipe. A thin layer of fouling was evident on most of the surface. It is believed that the fouling reduces flow not by reducing pipe cross sectional area but, by its non-uniformity, inducing turbulent flow and thus increasing the friction factor.

7.1 Higher Than Predicted Cooling Water Discharge Canal Level (cont.)

This problem is still not resolved. In the future during certain high tides, it will be necessary to reduce turbine load and cooling water flow. Various alternatives are being evaluated but no final decision has been made.

NOTE: This problem raised concern that a "reverse" problem might arise for the intake canal. That is, at full flow and low tides the drawdown in the intake canal might be excessive. It has been verified that, although drawdown is greater than expected, it is not excessive and presents no safety or operational concerns.

7.2 CEDM 44

During cold rod testing, this CEDM was found to be inoperative (would not withdraw) at low temperatures. This CEDM was tested after heatup (as were all CEDM's) and found to operate very satisfactorily. It was, in fact, the second fastest rod in terms of the 90% insertion drop time. After a cooldown with this CEDM out, it dropped satisfactorily and would withdraw some, indicating slight improvement. This rod never failed to drop properly and never failed to operate hot satisfactorily.

Amendment No. 4 to St. Lucie License No. DPR-67 was issued April 16, 1976 . This deleted a special test exemption allowing low temperature criticality for physics tests. It also required repair or replacement of CEDM 44, at the first shutdown expected to last 2 weeks or longer. This allowed hot criticality and continuation of the test program.

This CEDM was tested each shutdown and cooldown during testing. It continued to show operability improvement after each heatup/ cooldown cycle and again, never failed to trip, which is its only safety related function.

During the early parts of the shutdown for resolution of the power distribution anomaly, this CEDM was inspected mechanically. No conclusive evidence was noted. As soon as it became apparent this shutdown would be lengthy it was decided to replace CEDM 44 even though it presented no safety concerns.

This CEDM has been replaced and was retested satisfactorily (cold and hot) upon return to operation. The NRC Division of Inspection and Enforcement has reviewed the work and retest documentation and considered it satisfactory. Assuming issuance of an Amendment to License deleting reference to CEDM 44 this matter is fully resolved and will not be discussed in the supplementary Startup Report(s).

7.3 Apparent Low Reactor Cooling Pump Flow

During Post Core Load Hot Functional Testing, Reactor Coolant System Flow was determined, using Reactor Cooling Pump ΔP.

The original nominal design 100% flow was 324,800 gpm total. The FSAR, safety analysis and Technical Specifications assume a total flow of 370,000gpm (113.92%). Actual measured flow was about 5% less than that needed to guarantee the Technical Specification required flow.

Considering this 5% reduction in flow, Florida Power & Light Company requested a license limit of 90% power (10% reduction) to allow continuation of testing during resolution of the problem. Amendment 5 to St. Lucie License DPR-67 allowed operation to 60% power as an interim measure. An Order for Modification of License, dated June 17, 1976, addressing vendor Strikin II computer code errors as well as the flow uncertainty, amended this limit to 90% power.

During Power Ascension Testing, a flow determination by calorimetric means was performed at 80% power. Actual flow was determined to be 123.1% (399,800 gpm) of nominal design 100% flow. Since measurement uncertainties are about the same using this method or using pump ΔP it is evident that actual flow is in fact proper to allow operation at 100% power. It should be noted that the ΔP method of flow determination is obviously dependent upon pump geometric configuration and can be significantly affected by minor variations in inner casing surfaces, test tap orientation and internal clearances. It is felt that this is the root cause of the initial low measured flow. A calorimetric determination is, of course, independent of pump geometry.

Based on the above information, a request for license amendment has been submitted to the NRC. The request also includes a detailed error analysis of the flow determination by calorimetric means, and other supporting information such as discussion of results from similar plants.

Florida Power & Light Company feels this matter is now resolved. Assuming favorable response from the NRC on the request for license amendment, this item will not be discussed in the supplementary Startup Report(s).

7.4 Higher Than Predicted Containment Radiation Levels.

Dose rates at selected points, both inside and outside the Radiation Controlled Area, were determined during Power Ascension Testing. These results indicated that the 50% levels could be extrapolated to 100% to give preliminary verification of an expected dose rate problem. 100% power dose rates have not yet been determined as the unit has not yet operated at 100% (see section 7.5) A dose rate problem was expected due to problems at similar plants.

Extrapolated results indicated 3 potential problem areas. These were the doors to the shield building/containment annulus and the containment personnel airlock. Dose rates (combined neutron and gamma) were 3 mrem/hr and 10 mrem/hr at the southwest and northeast annulus doors respectively. Dose rates were 1 mrem/hr outside the airlock, 22 mrem/hr inside the airlock and 80 mrem/hr inside containment at the airlock. In addition, within containment, certain areas of the operating deck area (which should be accessible if necessary during operation) had estimated dose rates of up to 60 rem/hr (extrapolated).

This neutron streaming is primarily due to the large annular gap between the reactor vessel and the primary shield wall. This gap is to provide a vent path for a postulated LOCA but also provides a means for neutrons to scatter from the core midplane area. These then scatter from the containment ceiling causing the excessive dose rates.

As this problem was anticipated, a preliminary sheilding design has been prepared (see section 6.7). However, until 100% power dose rates are known, final evaluation of this design and a proposal to the Nuclear Regulatory Commission cannot be performed. Final resolution of this item will be discussed in our followup Startup Report(s).

7.5 Power Distribution Anomaly

On June 30, 1976, with the reactor at 80% power and all control rods out, a routine power distribution map gave the first indication of a small azimuthal power tilt. This was attributed at that time to detector errors or failure. It should be noted that at this time Technical Specifications for tilt and total radial peaking factor (F_r^T) were suspended for physics testing in accordance with the special test exceptions of the Technical Specifications.

Within the next week a few incore alarms were received. During evaluation of these, it was found that the calculated alarm setpoints were in error (LER 335-76-34, August 6, 1976) and it was also determined that the previously indicated tilt was still present. The alarms were corrected. On July 6, 1976, plant power was reduced to 50% for routine cleaning of a condensate pump strainer. While at 50% power, it was determined conclusively from the power distribution that an azimuthal tilt of approximately 4% was present along with an axial peaking value of 1.5, as compared to an expected value of <1.35. This tilt was verified using the moveable incore detector system. Technical Specifications for tilt and total radial peaking factor (F_r^T) were reinstated. It should be noted here that at no time was the plant in violation of any Technical Specification regarding azimuthal tilt or peaking. (LER 335-76-35, July 23, 1976)

On July 13, reactor power was reduced to about 10^{-2}% and a low power physics test program commenced. This program was a repeat of selected protions of the LPPT performed after initial startup. At this time two theories were offered as possible explanations: 1) a selective deposition of crud on the fuel leading to local flow maldistributions; and 2) early burnout of the burnable poison pins in the fuel assemblies. The results of these tests (available July 18) verified that the tilt was present, and that the core was more reactive (about .45%) than predicted. This second finding tended to support the early poison pin burnup theory. It was decided to open the reactor vessel for inspections and a shutdown/cooldown was commenced. Over the next week, many discussions were held, data was reduced and evaluated and theories postulated. None of this information could conclusively explain the existing phenomenon; therefore, on July 27, actual disassembly of the reactor began.

Representative fuel assemblies were removed from various areas of the vessel and inspected. The crud buildup theory was quickly dispensed with, as blisters and perforations were found on the poison pin cladding. More fuel assemblies were removed and inspected. Sufficient flaws were found to statistically demonstrate that there was a core wide problem with the cladding of the burnable poison pins. It should be noted here that no evidence was noted of any fuel pin anomalies. Due to the core-wide poison pin problem, the plant was defueled.

After the discovery of these poison pin cladding failures, a new theory was postulated. This was that the failure allowed the boron within the rods to wash out and be lost or to migrate and redistribute within the poison pins. This boron loss/redistribution theory correlated much better than any other theories previously considered.

It was then necessary to resolve two major concerns: 1) what caused the cladding failure and 2) what must be done to return the plant to power operation. To aid in resolving the first concern, poison pins were removed from selected fuel assemblies. These were submitted to on-site visual and eddy current testing. Then they were sent to research laboratories to determine the cause of the cladding failures, the mechanisms of boron loss and redistribution, and verification that loss of boron had occured in some pins and that it could cause the observed results.

As a result of these laboratory/test reactor inspections, the cause of the failure was confirmed to be hydriding of the zircalloy cladding of the pins. This was caused by excessive moisture content within the pins. Under incore conditions of high temperature and neutron flux the moisture produced free hydrogen which attacked the cladding. It was proven that the perforations did result in loss/redistribution of boron from the affected poison pins under incore conditions. And, it was confirmed that this loss/redistribution of boron could create the conditions observed at the St. Lucie Plant.

Then, regarding resolution of the second concern, it was determined that on site replacement of the poison pins with new ones of much lower moisture content was the appropriate solution. At the time this decision was made, some of the pin removal equipment had already been proven in removal of the pins for testing. So, there was reasonable assurance the job could be done even though it had to be done under water in the spent fuel pool. The vendor's specifications and controls on moisture content were significantly tightened to avoid repetition of the original problem. Replacement of the poison pins resulted in fuel assemblies virtually identical to the original ones except for minor fuel depletion (burnup).

Actual reconstitution (removal of old pins and installation of new ones) commenced on October 5, 1976. The basic procedure consisted of: drilling or cutting the flow plate webs above the poison pins; cleaning and deburring the newly machined surfaces; removing old pins using a template to ensure fuel pins were not removed; installing new pins; installing a retention assembly over the flow plate; and final QC and FP&L acceptance inspection. This process is described in greater detail in the FP&L submittals leading up to Ammendment 10 to St. Lucie License DPR-67, dated 3 December 1976, which allows resumption of power operations.

By November 3, 1976 this process was close to complete and core reload was commenced. By November 7, all but 2 assemblies were completed and on November 10, the last of the 108 assemblies had been reconstituted and core reload was continuing (supplementary LER 335-76-35, December 17, 1976).

We have now resumed power ascention testing and thus far have seen no evidence of any anomalies except those directly related to the uneven fuel depletion (burnup) which resulted from the power tilt/ peaking. These have been minor in magnitude and should be self-correcting as plant operation (and fuel depletion) continue. The activities after fuel reconsititution (fuel reload, initial criticality etc) will be described in our supplementary Startup Report(s).