FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE NUCLEAR POWER PLANT

UNIT 1

DOCKET NO. 50-335

LICENSE NO. DPR-67

SUPPLEMENTARY STARTUP TEST REPORT

MAY, 1977

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This report fulfills the Requirements of Regulatory Guide 1.16 which states that a Startup Test Report will be submitted to the NRC within 9 months of initial criticality and every three months thereafter until the unit has been declared commercial and startup testing is completed. Initial Criticality was April 22, 1976. The initial Startup Report, submitted January 21, 1977, covered the Start Test Program, thru the 50% power level of power ascension testing. As described in that report, testing was stopped in July, 1976, due to a flux anomaly which required replacement of the burnable poison pins in the fuel assemblies. This report covers the core reload, return to 50% power, the remainder of power ascension and completion of startup testing. Also, the unit was declared commercial at 0001, December 21, 1976. Startup testing was essentially completed on March 24, 1977 with the completion of the Nuclear Steam Supply System warranty run.

The Startup Test Program was organized and administered by Florida Power & 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) Core Reload.
(2) Post Core Load Hot Functional Tests.*
(3) Approach to Criticality.
(4) Low Power Physics Tests.
(5) Escalation to Power Tests - 50% Plateau.
(6) Escalation to Power Tests - 80% Plateau.
(7) Escalation to Power Tests - 100% Plateau.
(8) Post 100% Plateau (Transient) Tests.

* None included in this report.

1.0 INTRODUCTION AND SUMMARY (cont)

1.1 Introduction (cont)

Maximum licensed reactor core power level (100%) is 2560. Mwt. The Startup Test Program recommenced November 4, 1976, with the loading of the first fuel assembly into the reactor vessel and was completed March 24, 1977. FP&L requested and received permission from Inspection and Enforcement, Region II, to submit this report by May 30, 1977, rather than April 22, 1977. This allowed time to include in this report all testing not previously reported and allows both FPL and the NRC to close out the startup phase of St. Lucie #1 without another supplementary report.

1.2 Summary

1.2.1 Initial Fuel Load

Fuel loading commenced on November 4, 1976, and was completed on November 15, 1976. A sizable portion of this time was spent in non-fuel loading activities. The largest period of non-fuel loading time (approximately 41% of total time) was associated with inspecting fuel elements B-069 and A-015, and A-049 and B-001, which came into physical contact with each other (B-069 into A-015, etc.). 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 assemblies per shift.

1.2.2 Post CORE LOAD Hot Functional Tests

All post core load hot functional testing was completed during initial startup and reported in the first Startup Report.

1.2.3 Initial Approach to Criticality

The initial approach to criticality commenced on December 2, 1976. The reactor was declared critical on December 4, 1976. The CEA's were withdrawn, followed by a slow RCS dilution to criticality. Measured RCS soluble boron concentration at criticality was in close agreement with that which was predicted and well within the acceptance criteria. The only problems of consequence encountered were with the CEDM System. CEA 1-29 "dropped" four different times during dilution. Dilution was secured and CEA 1-29 was retrieved each time. This problem has been resolved and has not reoccurred.

1.0 INTRODUCTION AND SUMMARY (cont)

1.2 Summary (cont)

1.2.4 Low Power Physics Test

The Low Power Physics Test (LPPT) phase commenced on December 4, 1976. The LPPT phase was completed on December 9, 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

Power Ascension Testing began December 9, 1976. The program consisted of a 2-week ascent to 50% power, about 2 weeks at 50% power, ascent to 80% power, ascent to 90% power, and when the license was amended, ascent to 100% power. The slow ascent to 50% and the hold at that power were to allow ample time for monitoring of in-core flux distribution to verify that the flux distribution anomaly no longer existed. The results of this monitoring were satisfactory. After initial operation at 100% power, the two 100% trip tests were performed. Testing was then performed at 100% steady state followed by plant response (transient) testing. The last test was the 200 hour NSSS acceptance run.

2.0 CORE RELOAD

At 0010 hours on November 4, 1976 fuel assembly number B-009 containing neutron source number 1, step number 1 (of Appendix A of Operating Procedure 1600022, Revision 3), was loaded into core location X-11. Fuel loading was completed at 2215 hours on November 15, 1976 when fuel element number C-010, step number 217, was loaded into core location A-8.

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. It should be noted that in the original initial core load (cycle 1) fuel assembly C-112 containing CEA 64 was in core location J-20, and fuel assembly C-107 was in core location G-20. In this initial core load (cycle 1A) fuel assembly C-107 containing CEA 64 is in core location J-20, and fuel assembly C-112 is in core location G-20. This change was done per Combustion Engineering Company's request because fuel assembly number C-112 contained a slightly depressed guide tube. The core design characteristics are the same as for cycle 1, and are available in Table 2.0-2 of the original Startup Test Report (dated December, 1976).

Neutron count rate was monitored during fuel loading on two separate detectors, Wide Range Log Channel C and Wide Range Log Channel D. 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 fuel loading.

Fuel loading was conducted with the spent fuel pool full of borated water. A refueling boron concentration of $\geq$ 1720 ppm boron was maintained with shutdown cooling flow through the core in accordance with the Technical Specifications.

Fuel loading took approximately 290 hours. A few problems occurred, two of which will be addressed at this time. The fuel handling equipment malfunctioned numerous times. Florida Power and Light Company's maintenance department repaired the upender, refueling machine and new fuel machine as required. Equipment down time resulted in a loss of approximately 12 hours. During movement of the fuel bundles within the core area fuel assembly B-069 came into physical contact with fuel assembly A-015, and fuel assembly A-049 came into physical contact with fuel assembly B-001. The subsequent removal, inspection and replacement of these four assemblies resulted in an approximate 120-hour delay. Miscellaneous other delays accounted for another 11 hours.

TABLE 2.0-1

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
1	B009	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	C104	31	X-9
9	C007	-	Y-8
10	C114	-	X-7
11	A002	25	W-7
12	B042	-	V-7
13	A010	30	V-9
14	B017	-	V-11
15	A038	39	V-13
16	B068	-	V-15
17	A004	47	W-15
18	C106	-	X-15
19	C023	-	Y-14
20	C027	-	X-6
21	B080	-	W-6
22	A013	18	V-6
23	C016	-	X-16
24	B020	-	W-16

TABLE 2.0-1 (Cont.)

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
25	A015	51	V-16
26	C008	-	X-5
27	C212	14	W-5
28	B046	-	V-5
29	C001	-	X-17
30	C208	55	W-17
31	B037	-	V-17
32	C012	-	W-4
33	C015	-	V-3
34	C202	67	V-4
35	C034	-	W-18
36	C036	-	V-19
37	C205	69	V-18
38	C013	-	T-20
39	C204	63	T-19
40	B049	-	T-18
41	A019	H	T-17
42	B069	-	T-16
43	A048	46	T-15
44	B010	-	T-13
45	A049	C	T-11
46	B001	-	T-9
47	A008	24	T-7
48	B012	-	T-6

TABLE 2.0-1 (Cont.)

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
49	A018	F	T-5
50	B043	-	T-4
51	C002	-	T-2
52	C210	6	T-3
53	C035	-	S-2
54	B074	-	S-3
55	A014	10	S-4
56	B013	-	S-5
57	A054	17	S-6
58	B045	-	S-7
59	A055	-	S-9
60	B026	-	S-11
61	A061	-	S-13
62	B035	-	S-15
63	A053	50	S-16
64	B071	-	S-17
65	A020	59	S-18
66	B039	-	S-19
67	C033	-	S-20
68	C109	-	R-20
69	A028	62	R-19
70	B078	-	R-18
71	A031	54	R-17
72	B007	-	R-16

TABLE. 2.0-1 (Cont.)

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
73	A037	45	R-15
74	B004	-	R-13
75	A030	34	R-11
76	B038	-	R-9
77	A059	23	R-7
78	B040	-	R-6
79	A056	13	R-5
80	B041	-	R-4
81	A022	5	R-3
82	C102	-	R-2
83	C021	-	P-1
84	C116	2	N-2
85	B029	-	N-3
86	A040	9	N-4
87	B064	-	N-5
88	A064	-	N-6
89	B056	-	N-7
90	A034	29	N-9
91	B051	-	N-11
92	A057	38	N-13
93	B065	-	N-15
94	A060	-	N-16
95	B077	-	N-17
96	A009	58	N-18

TABLE 2.0-1 (Cont.)

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
97	B008	-	N-19
98	C111	65	N-20
99	C004	-	P-21
100	C028	-	M-21
101	B023	-	L-20
102	A026	73	L-19
103	B006	-	L-18
104	A043	D	L-17
105	B044	-	L-16
106	A068	44	L-15
107	B054	-	L-13
108	A032	33	L-11
109	B025	-	L-9
110	A041	22	L-7
111	B053	-	L-6
112	A044	A	L-5
113	B036	-	L-4
114	A024	70	L-3
115	B079	-	L-2
116	C011	-	M-1
117	C030	-	K-1
118	C005	-	H-1
119	C110	1	J-2
120	B047	-	J-3

TABLE 2.0-1 (Cont.)

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
121	A011	8	J-4
122	B059	-	J-5
123	A058	-	J-6
124	B058	-	J-7
125	A050	28	J-9
126	B055	-	J-11
127	A045	37	J-13
128	B028	-	J-15
129	A063	-	J-16
130	B061	-	J-17
131	A006	57	J-18
132	B076	-	J-19
133	C107	64	J-20
134	C025	-	K-21
135	C022	-	H-21
136	C112	-	G-20
137	A005	61	G-19
138	B018	-	G-18
139	A042	53	G-17
140	B031	-	G-16
141	A051	43	G-15
142	B033	-	G-13
143	A052	32	G-11
144	B070	-	G-9

TABLE 2.0-1 (Cont.)

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
145	A039	21	G-7
146	B005	-	G-6
147	A035	12	G-5
148	B060	-	G-4
149	A033	4	G-3
150	C103	-	G-2
151	C040	-	F-2
152	B066	-	F-3
153	A016	7	F-4
154	B052	-	F-5
155	A065	16	F-6
156	B072	-	F-7
157	A067	-	F-9
158	B011	-	F-11
159	A066	-	F-13
160	B015	-	F-15
161	A069	49	F-16
162	B063	-	F-17
163	A021	56	F-18
164	B034	-	F-19
165	C039	-	F-20
166	C024	-	E-20
167	C211	60	E-19
168	B019	-	E-18

TABLE 2.0-1 (Cont.)

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
169	A062	G	E-17
170	B016	-	E-16
171	A036	42	E-15
172	B003	-	E-13
173	A012	B	E-11
174	B032	-	E-9
175	A047	20	E-7
176	B002	-	E-6
177	A027	E	E-5
178	B030	-	E-4
179	C014	-	E-2
180	C207	3	E-3
181	C029	-	D-3
182	C203	66	D-4
183	B024	-	D-5
184	A029	15	D-6
185	B014	-	D-7
186	A046	27	D-9
187	B048	-	D-11
188	A007	36	D-13
189	B073	-	D-15
190	A017	48	D-16
191	B021	-	D-17
192	C038	-	D-19

TABLE 2.0-1 (Cont.)

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
193	C201	68	D-18
194	C020	-	C-18
195	C206	52	C-17
196	B062	-	C-16
197	A001	41	C-15
198	B075	-	C-13
199	A023	71	C-11
200	B022	-	C-9
201	A003	19	C-7
202	B027	-	C-6
203	C019	-	C-4
204	C209	11	C-5
205	C003	-	B-5
206	C017	-	B-6
207	C108	-	B-7
208	C115	26	B-9
209	B050	Source 2	B-11
210	C009	-	B-17
211	C037	-	B-16
212	C113	-	B-15
213	C101	35	B-13
214	C006	-	A-14
215	C026	-	A-12
216	C018	-	A-10
217	C010	-	A-8

CORE LOADING SEQUENCE

Figure 2.0-1

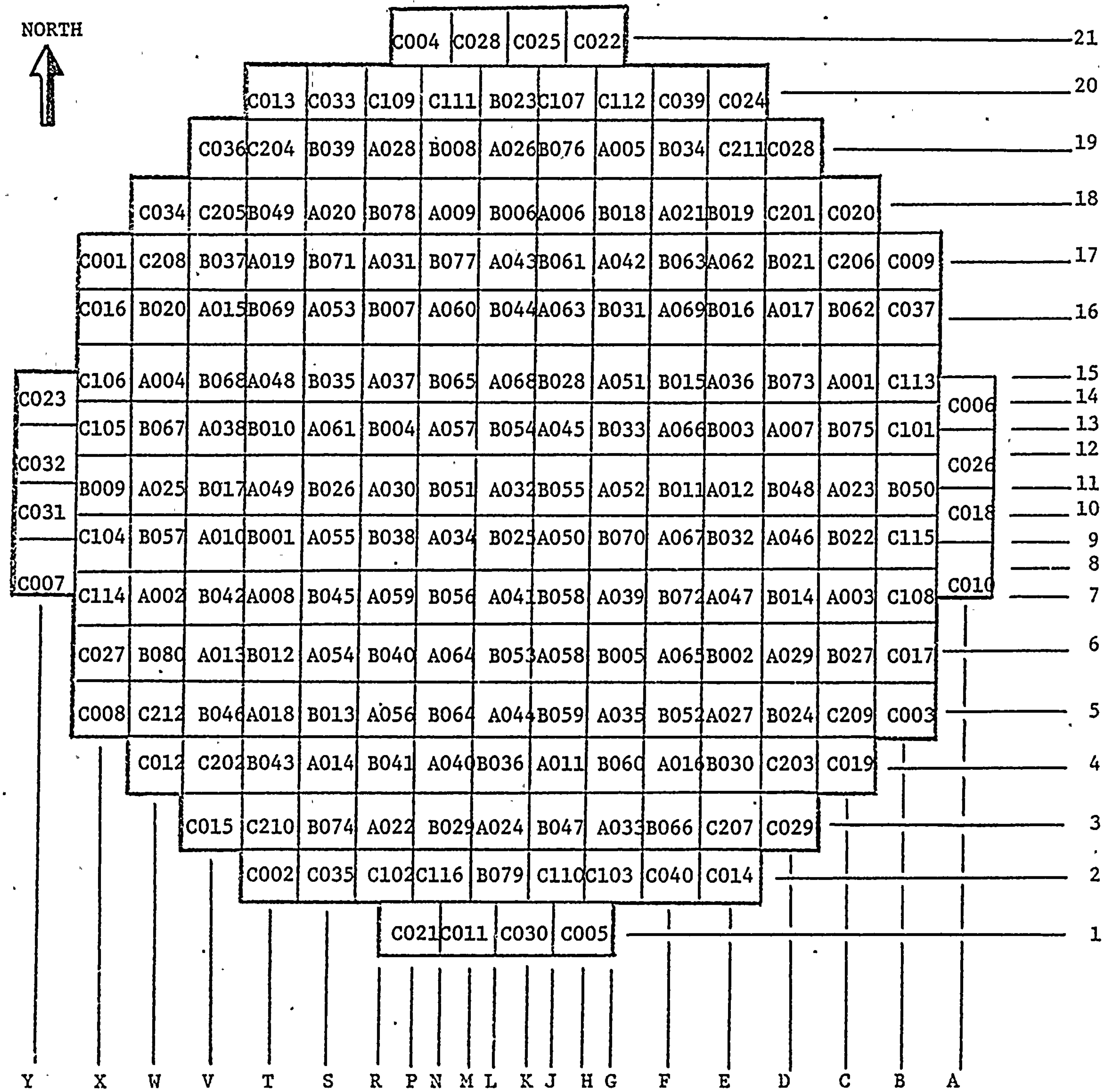
NORTH

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

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

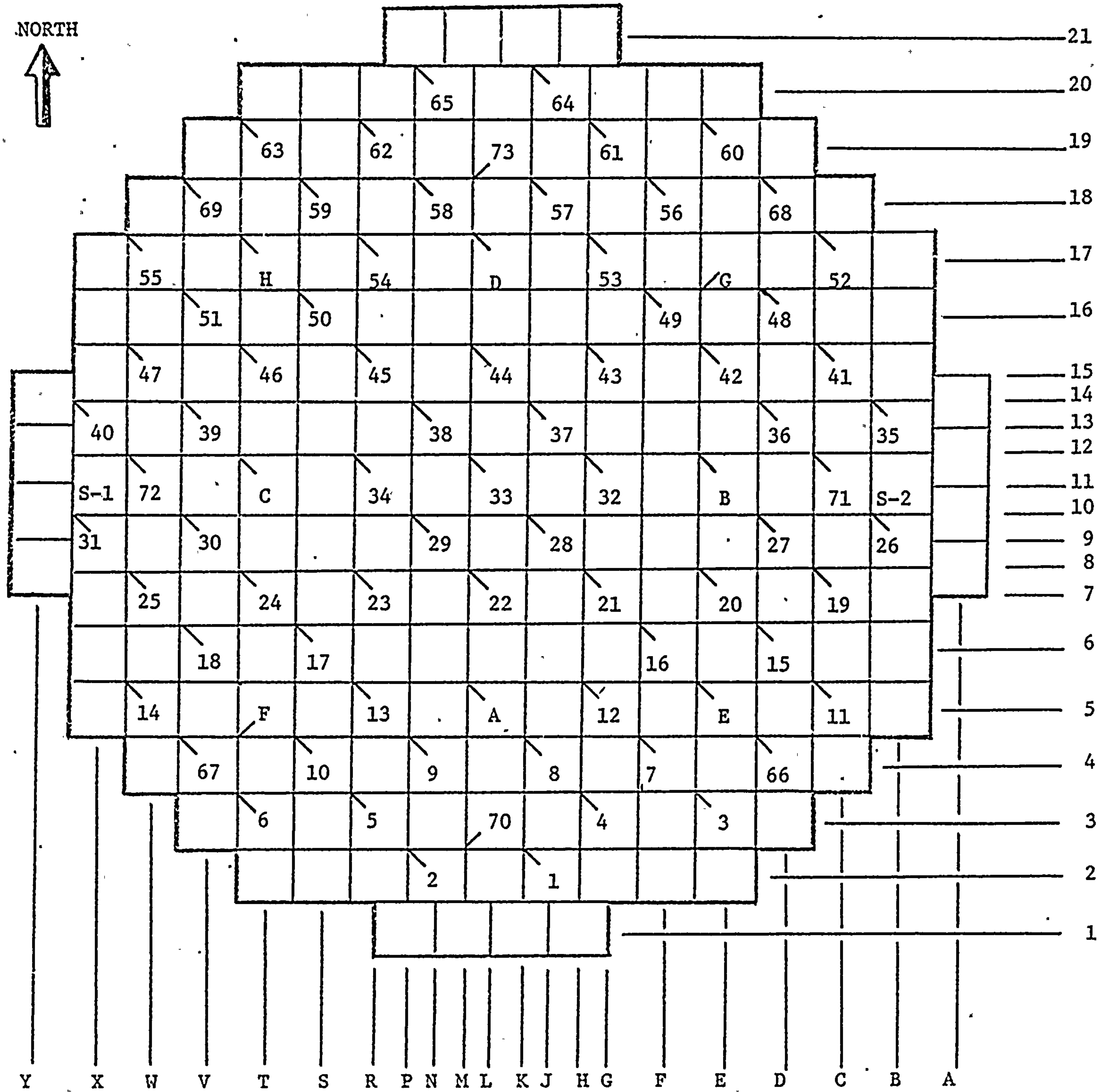
FUEL ASSEMBLY SERIAL NUMBERS
AND CORE LOCATIONS

Figure 2.0-2



ST. LUCIE PLANT, UNIT NO. 1
CONTROL ELEMENT ASSEMBLY SERIAL
NUMBERS AND CORE LOCATIONS

Figure 2.0-3



Normal CEA Orientation: Serial # on NW Web

Exception: CEA's 70, 73, F and G have Serial # on SW Web

4.0 APPROACH TO CRITICALITY FOLLOWING CORE RELOAD

Criticality was achieved on December 4, 1976 at Reactor Coolant System (RCS) conditions of 533°F and 2263 psia. The initial RCS boron concentration was 1760 ppm. The Approach to Criticality 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 890 ppm. The Initial Criticality Following Refueling procedure limits RCS boron concentration dilution below 800 PPM. The procedure cautions that "criticality shall be anticipated whenever CEA's are being withdrawn or boron dilution operations are in progress".

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 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 4.0-4 was recorded and compared with predicted values.

In summary, initial criticality following core reload was achieved in a safe and orderly fashion. There was acceptable agreement between the measured and predicted critical boron concentrations. The predicted boron concentration at criticality was 837 ppm. The actual value of 890 ppm was well within the PSL acceptance tolerance of $\pm$ 100 ppm.

TABLE 4.0-1

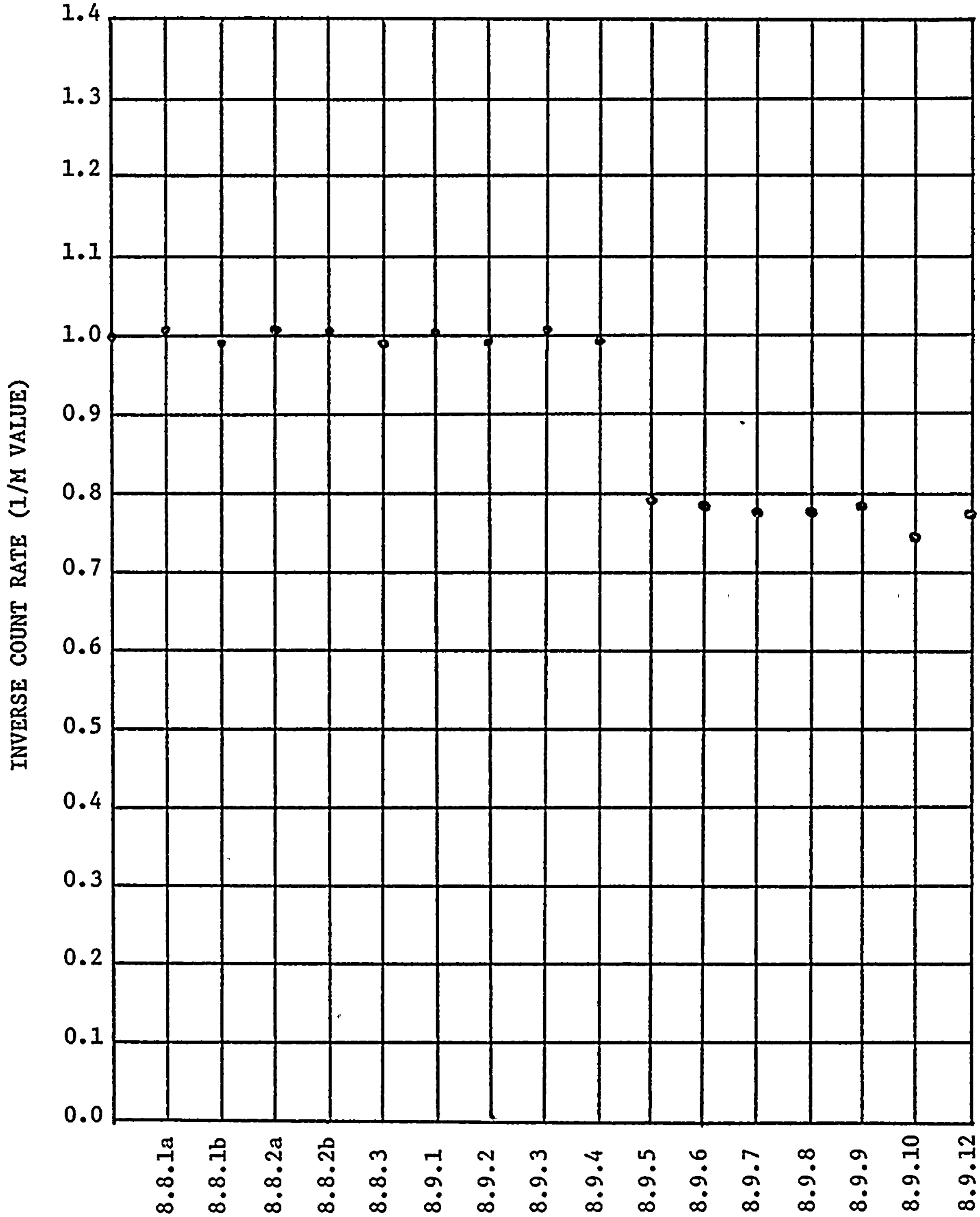
CEA WITHDRAWAL SEQUENCE

*STEP	CEA GROUP	INCHES WITHDRAWN	B 1/M	D 1/M
8.8.1a	A	68	1.040	0.992
8.8.1b	A	136	0.980	1.020
8.8.2a	B	68	1.050	1.050
8.8.2b	B	136	1.010	0.980
8.8.3	P	>132	0.970	1.030
8.9.1	1	68	1.010	1.010
8.9.2	1 2	UEL ≤54	0.980	1.010
8.9.3	2 3	122 ≤40	1.020	1.000
8.9.4	2 3 4	UEL ≤107 ≤26	0.990	0.980
8.9.5	3 4 5	UEL ≤93 ≤11	0.790	0.820
8.9.6	4 5	UEL ≤79	0.780	0.780
8.9.7	5 6	UEL ≤54	0.770	0.790
8.9.8	6 7	122 ≤40	0.770	0.770
8.9.9	6 7	UEL 68	0.780	0.760
8.9.10	7	UEL	0.740	0.790
8.9.12	7	68	0.772	0.801

*Steps IAC with O.P. 0030221
NOTE: UEL = Upper Electrical Unit

FIGURE 4.0-1

ST. LUCIE UNIT 1
INITIAL APPROACH TO CRITICALITY
CYCLE 1A, 532°F, 2250 PSIA
WIDE RANGE LOG CHANNEL B
RCS BORON ≥ 1720 PPM



CEA WITHDRAWAL STEPS IAC WITH O.P. 0030221

FIGURE 4.0-2

ST. LUCIE UNIT 1
INITIAL APPROACH TO CRITICALITY
CYCLE 1A, 532°F, 2250 PSIA
WIDE RANGE LOG CHANNEL D
RCS BORON $\geq$ 1720 PPM

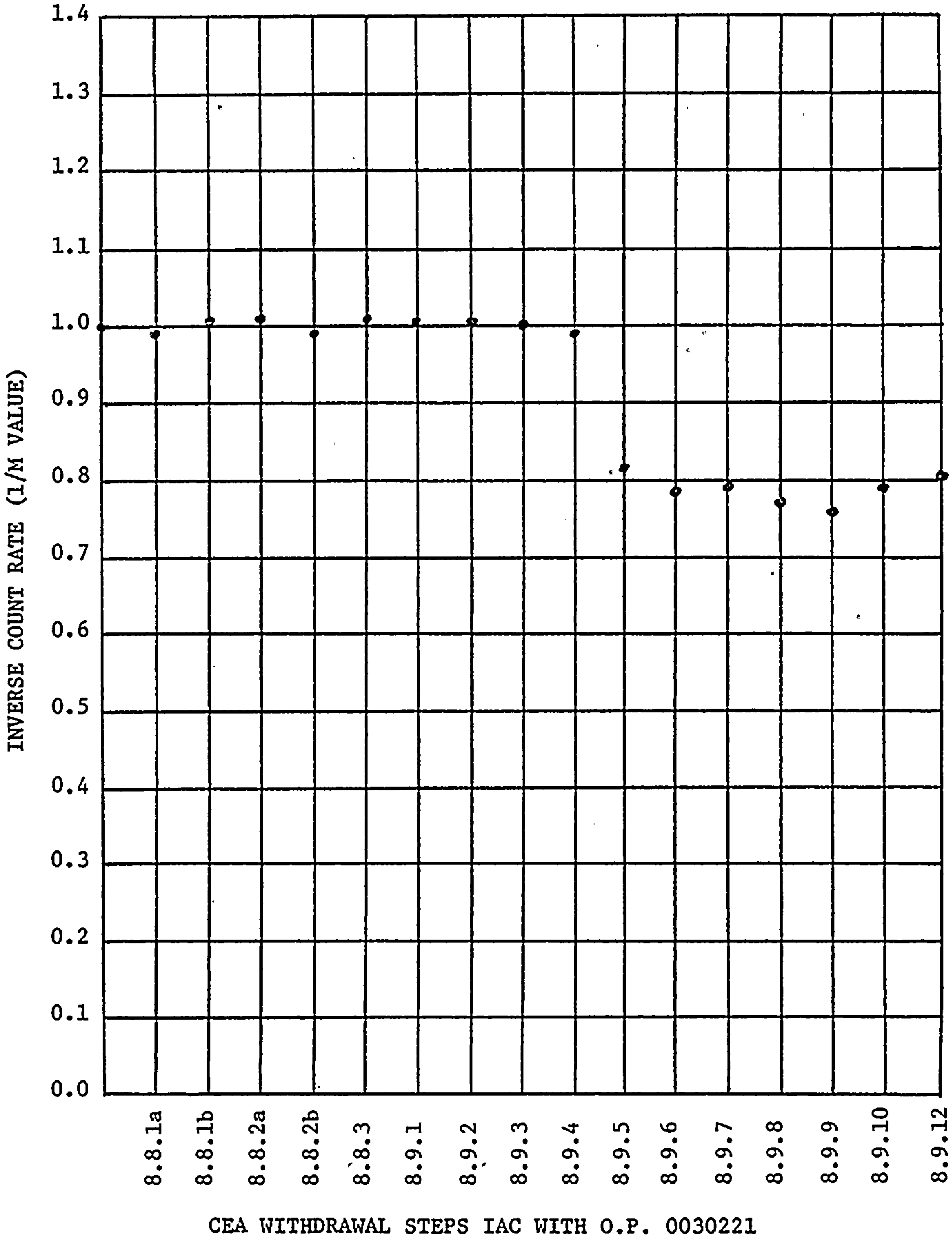


TABLE 4.0-2

DILUTION TIME (MINUTES)	R.C.S. BORON CONCENTRATION	CHANNEL B 1/M VALUE	CHANNEL D 1/M VALUE
0	1770	1.000	1.000
20	1776	0.968	0.945
50	1668	0.885	0.917
75	1603		
80		0.851	0.844
105	1493	0.792	0.790
135	1470	0.759	0.699
165	1395	0.675	0.685
195	1338	0.608	0.612
225	1290	0.563	0.576
255	1181	0.494	0.526
285	1179	0.481	0.478
315	1157	0.420	0.414
345	1061	0.369	0.371
357	1070		
369	1080	0.385	0.387
400	1054	0.366	0.367
430	1002	0.346	0.343
444		0.336	0.328
449	1021		
474		0.312	0.308
479	1009		
494		0.286	0.283
504	992		
519	980	0.270	0.266
539	967	0.259	0.248
559	950	0.237	0.193
579	937	0.194	
599	934	0.163	0.165
619	903	0.100	0.108
629		0.052	0.055
639	894	0	0
659	890	0	0
679	890	0	0

FIGURE 4.0-3

ST. LUCIE UNIT 1
INITIAL APPROACH TO CRITICALITY
CYCLE 1A, 532°F, 2250 PSIA
WIDE RANGE LOG CHANNEL B
CEA GROUP 7 AT 68 INCHES

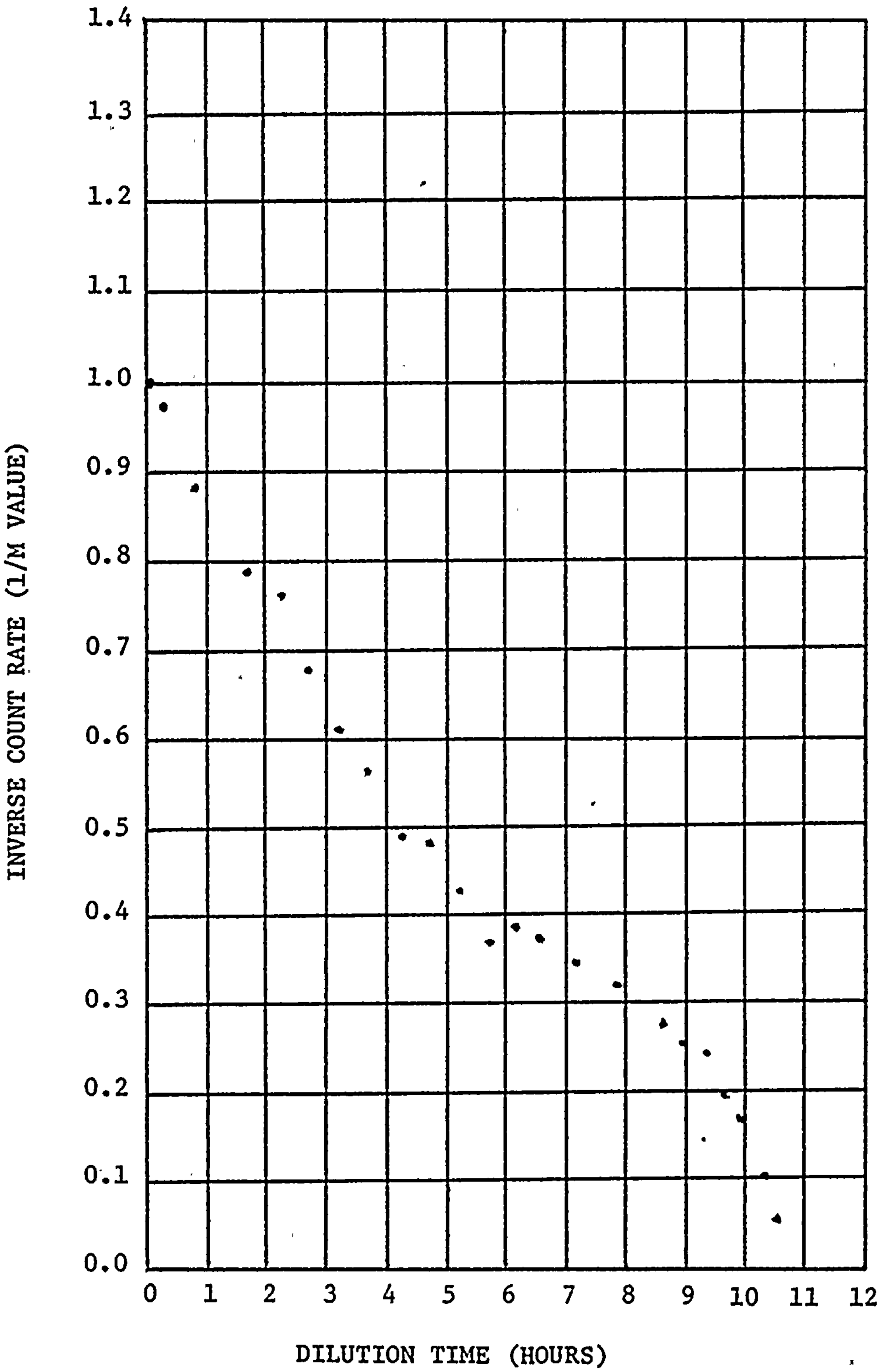


FIGURE 4.0-4

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

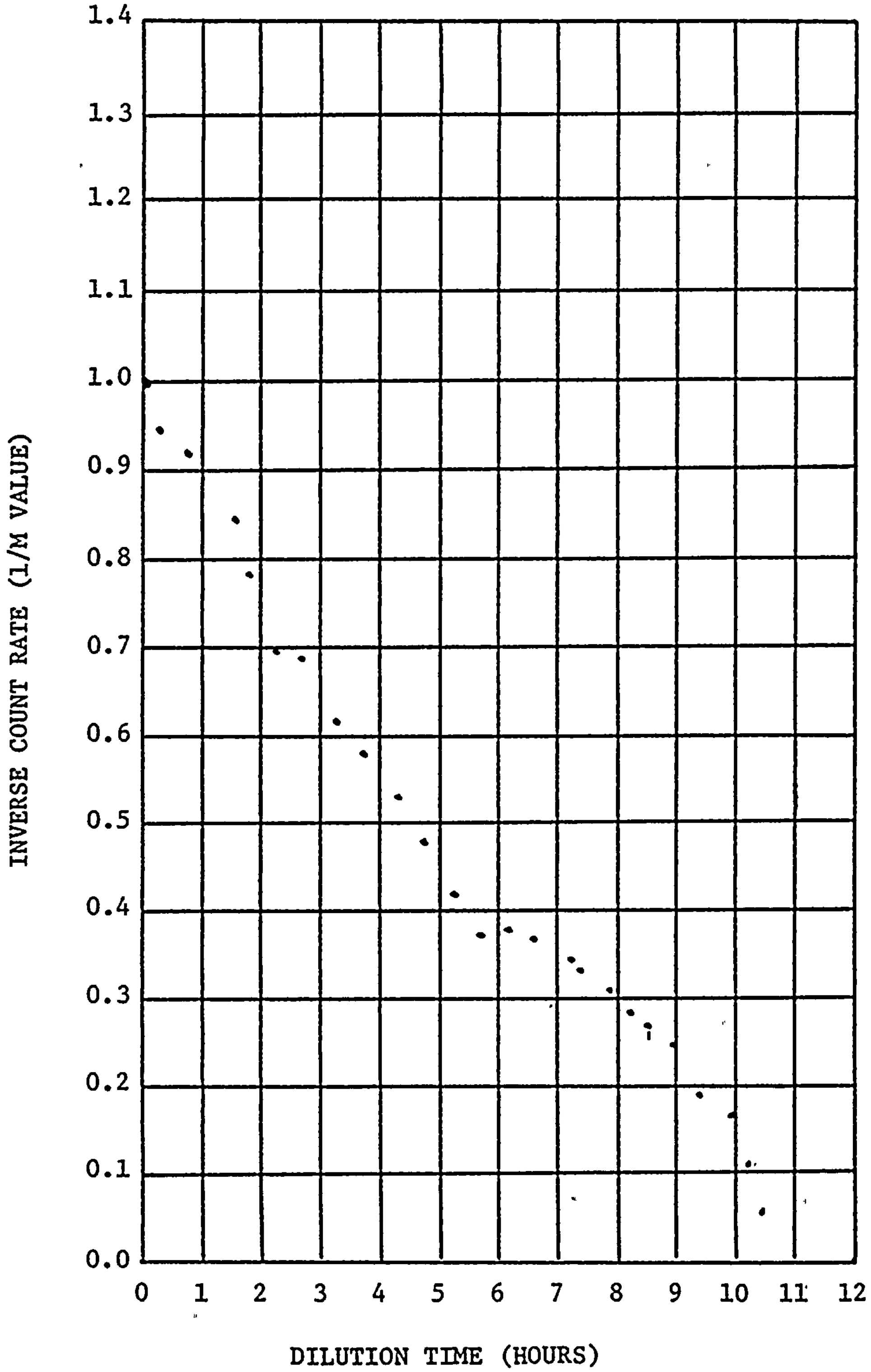


TABLE 4.0-3

ST. LUCIE UNIT 1
CYCLE 1A
RCS BORON CONCENTRATION
VS
DILUTION TIME

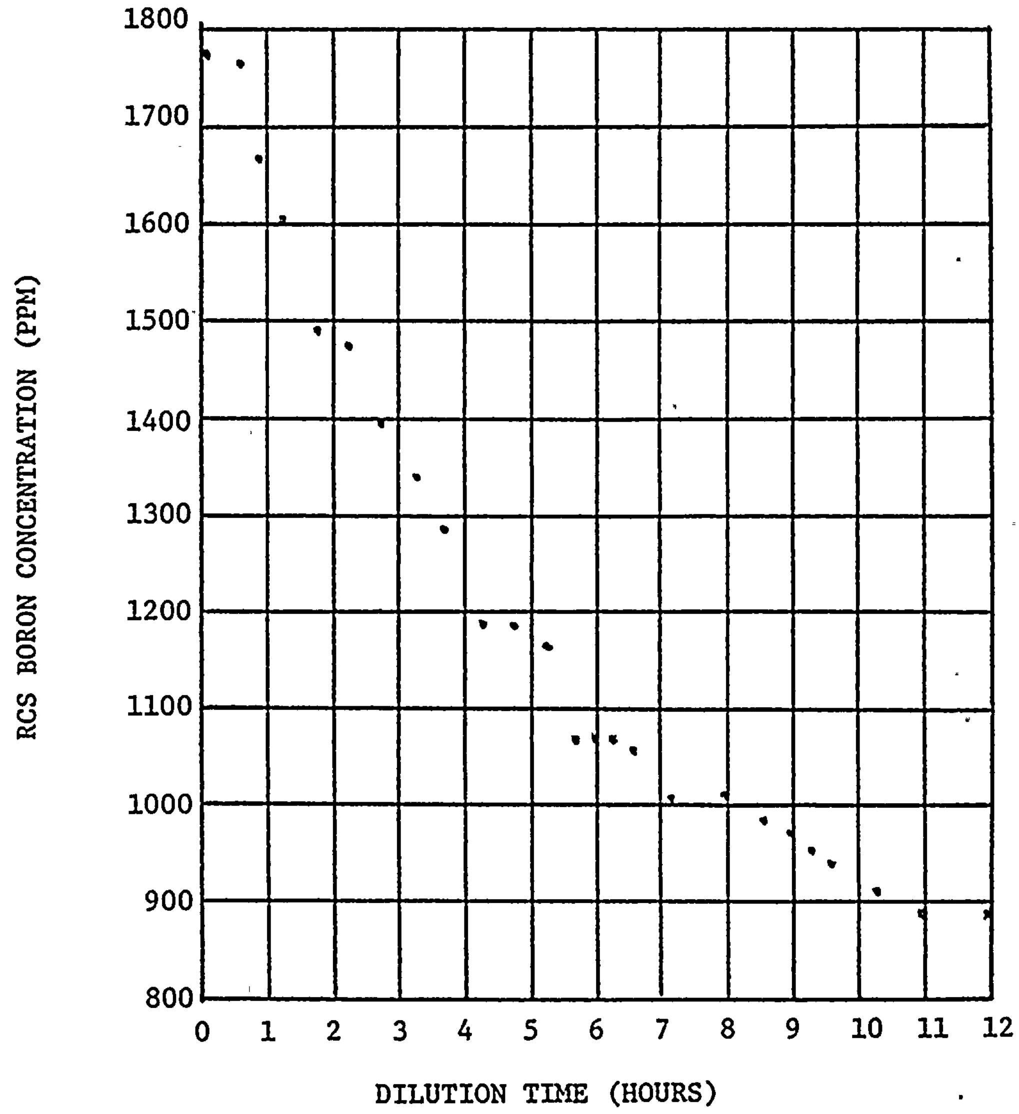


TABLE 4.0-4

PARAMETER	INITIAL CONDITION	MEASURED VALUE
RCS TEMPERATURE (^{o}F)	532	533
RCS PRESSURE (PSIA)	2250	2262.5
RCP'S OPERATING	4	4

CEA GROUPS WITHDRAWN	INCHES	
A	136	136
B	136	136
P-1	136	136
P-2	136	136
1	136	136
2	136	136
3	136	136
4	136	136
5	136	136
6	136	136
7	68	68

	PREDICTED	MEASURED
RCS BORON CONCENTRATION (PPM)	837	890

5.0 LOW POWER PHYSICS TESTS (LPPT)

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.

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 data analyses.

LPPT was completed satisfactorily during the original plant startup. Due to the flux distribution anomaly and fuel assembly poison pin replacement described in the original report, it was deemed prudent to repeat selected portions of LPPT. This testing and the flux monitoring (Section 6.1) performed during the slow ascent to power and throughout the test program confirmed that the flux distribution anomaly had been resolved satisfactorily.

TABLE 5.0-1

LOW POWER PHYSICS TEST SCHEDULE	
A*	Reactivity Computer Response Check
B	CEA Symmetry
C	Unrodded Critical Boron Concentration
D	Isothermal Temperature Coefficient (CEA Group 7 approximately 115 inches)
E	CEA Group 7, 6 and 5 worth (non overlap)
F	Isothermal Temperature Coefficient (CEA Group 5 approximately 10 inches)
G	CEA Group 4, 3 and 2 worth (non overlap)
H	Isothermal Temperature Coefficient (CEA Group 1 approximately 126 inches)
I	Sequential Worth of CEA Groups 2, 3, 4, 5, 6 and 7 (overlap mode)
J	Center CEA Worth Measurement, GP7≃100"

*Reference: Zero Power Physics Tests After Reload Preoperational Procedure No. 0110097

5.1 REACTIVITY COMPUTER RESPONSE CHECK

Purpose

The reactivity computer is an analog computer that calculates reactivity by solving the kinetics equations of reactivity. This exercise was not necessarily a test, but rather a verification of appropriate constants, and of equipment functioning properly. It was also necessary to know what flux (power) levels corresponded to the operability range of the scale on the reactivity computer, and this correlation check was performed at this time, in the Low Power Physics Test Schedule.

Test Results

The reactor was critical at a power level $\leq 10^{-2}\%$. Reactivity was being controlled by CEA group 7. A small amount of reactivity was introduced into the core, by group 7 withdrawal. The reactor period was calculated using the equation $T = \Delta t \div \ln (P/Po)$. This was repeated a couple of times, and the measured reactivity (from the trace generated on the chart of the reactivity computer) was compared to the design reactivity (using calculated T and design curve $\Delta\rho$).

The Lower Power level was determined by decreasing power until the trace on the pen on the chart recorder on the reactivity computer was "Noisier" than 0.3¢, peak to peak. The upper limit was below the point of adding nuclear heat.

Conclusions

The reactivity computer did respond properly. The acceptance criteria (correction factor = 1.00 ± 0.10) was satisfied. The lower and upper power levels were defined.

TABLE 5.1-1

REACTIVITY COMPUTER REACTIVITY RESPONSE CHECK

REACTOR PERIOD**	216 Seconds	222 Seconds	345 Seconds
STRIP CHART $\Delta\rho$	4.550	4.560	3.000
DESIGN CURVE $\Delta\rho$	4.800	4.650	3.200
CORRECTION FACTOR*	1.055	1.020	1.067

* CF = DESIGN $\Delta\rho$ / Recorder $\Delta\rho$

** T = Δt / ln (P/Po)

TABLE 5.1-2

REACTIVITY COMPUTER LOWER, MIDRANGE AND UPPER POWER LEVEL

WIDE RANGE RPS AVERAGE POWER (%)	0.500×10^{-2}	1.500×10^{-2}	2.250×10^{-2}
KIETHLY INPUT NI-9 (AMPS)	0.020×10^{-6}	0.058×10^{-6}	0.100×10^{-6}
KIETHLY INPUT NI-10 (AMPS)	0.016×10^{-6}	0.042×10^{-6}	0.070×10^{-6}
REACTIVITY COMPUTER STRIP CHART	30	65	100
CEA GP 7 POSITION (INCHES WITHDRAWN)	93	93	91

5.2 CEA SYMMETRY

Purpose

After fuel loading and reactor vessel reassembly, it is necessary to insure CEA's are coupled to their respective CEDM shafts. During this test, the reactivity worths of the symmetric FLCEA's (full length CEA's) were compared to each other. That is, one CEA's worth was compared to the worth of another CEA that is symmetric to it. This was done to insure uniform reactivity control within a CEA group.

The worthiest (according to design calculations) CEA was measured at this time, also.

The PLCEA's (partial length CEA's) were tested for operability only.

Test Results

The initial conditions before starting this test were reactor critical, with CEA group 7 approximately 95 inches withdrawn. The first designated CEA was diluted to LEL (lower electrical limit) and CEA group 7 adjusted to zero reactivity. By rod swapping, that is inserting another CEA and withdrawing the CEA that was in the core, keeping reactivity and reactor power within limits, without moving CEA group 7, the relative worth of the newly inserted CEA can be compared to the relative worth of the newly withdrawn CEA. As different symmetrical CEA groups are compared, the first CEA is inserted to LEL, and reactivity zeroed using CEA group 7. The remaining CEA's in that group are then swapped and compared to each other.

The worthiest, by design calculations, CEA was B-7. The worth was measured by diluting B-7 to LEL, measuring reactivity, as it was being stepped into the core. The worth was also measured by borating B-7 out. The two values were averaged, and the resultant worth was used as the test result.

Each individual PLCEA was inserted to 120 inches and withdrawn. This was done to verify that negative reactivity was being inserted into and withdrawn from the core.

Conclusions

The acceptance criteria for the test were met with one exception. The CEA symmetry check revealed 2 CEA's with relative worth more than specified when compared to its symmetric "sisters." CE has evaluated these results and verifies that their relatively lower worth is due to their location in the area of greatest fuel depletion. (They are in the high power area of the previous core tilt which led to fuel reconstitution.) The FRG concurs with this analysis and accepts all the results of this test. FP&L Reactor Engineering also agrees with CE's analysis. The highest worth CEA, B-7, was worth 0.158 $\%\Delta\rho$, (acceptance criteria calls for equal to or less than 0.231 $\%\Delta\rho$).

5.3 Critical Boron Concentration Measurements

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.

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).

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 5.3-1

CRITICAL BORON CONCENTRATION AT VARIOUS CEA POSITIONS

CEA CONFIGURATION	ARO	5 LEL	2 LEL	7 100"
MEASURED CBC (PPM)	912	788	540	907
PREDICTED CBC (PPM)	866	742	504	-
RCP'S OPERATING	4	4	4	4
RCS TEMPERATURE (^{o}F)	532	532	533	533
RCS PRESSURE (PSIA)	2259	2265	2260	2260
CEA GROUP	POSITION			
P1	UEL	UEL	UEL	UEL
P2	UEL	UEL	UEL	UEL
A	UEL	UEL	UEL	UEL
B	UEL	UEL	UEL	UEL
1	UEL	UEL	UEL	UEL
2	UEL	UEL	LEL	UEL
3	UEL	UEL	LEL	UEL
4	UEL	UEL	LEL	UEL
5	UEL	LEL	LEL	UEL
6	UEL	LEL	LEL	UEL
7	UEL	LEL	LEL	100"
TIME	2104	1701	1247	1113
DATE	12/6/76	12/8/76	12/8/76	12/9/76

5.4 Temperature Coefficient of Reactivity Measurements

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.

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 be less positive than $+0.5 \times 10^{-4}$ $\Delta k/k/^{o}F$ whenever power $\leq$ 70%. The FSAR accident analysis assumes various values within the range of +0.5 to -1.4×10^{-4} $\Delta k/k/^{o}F$ for beginning of core life.

Conclusions

For all cases, the measured values of isothermal temperature coefficient are within the FSAR and Technical Specifications acceptance criteria, and are therefore acceptable.

TABLE 5.4-1

ISOTHERMAL TEMPERATURE COEFFICIENTS AT VARIOUS CEA POSITIONS

RCS Boron, PPM	909	780	542
CEA Configuration (Desired)	7@ 115"	5@ 10"	1@ 126"
Measured MTC*	+0.26	-0.25	-0.90
Technical Specification MTC Limit*	<0.50	<0.50	<0.50
RCS Temperature, °F	532	532	532
RCS Pressure, PSIA	2262	2260	2258
CEA Group	Position **		
P1	U	U	U
P2	U	U	U
A	U	U	U
B	U	U	U
1	U	U	U
2	U	U	12"
3	U	U	L
4	U	U	L
5	U	15"	L
6	U	L	L
7	115"	L	L
TIME	0226	2300	1516
DATE	07 Dec 76	07 Dec 76	08 Dec 76

* X 10^{-4} $\Delta K/K$ /°F

** L = Lower Electrical Limit, U = Upper Electrical Limit

5.5 NON-OVERLAPPED REGULATING CEA GROUP WORTH MEASUREMENTS

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). The CEA group worths were measured over the range of group 7 UEL to group 2 LEL, at the various corresponding Reactor Coolant System boron concentrations.

Test Results

All CEA Group reactivity worths were measured using a soluble boron swap method, either dilution or boration, to maintain criticality while inserting 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. The results of this test were compared to the results of the identical test performed on cycle 1 (Ref. Start Up Report dated December 1976, Section 5.5) to insure similar CEA worths.

Conclusions

Amendment 10 to the Facility License requires that "individual CEA group worths will be measured for CEA groups 7, 6, 5, 4, 3, and 2. The measurement of the reactivity worth for regulating CEA group 1 and shutdown CEA group B will be made only if the previously measured worths of CEA groups 7, 6, 5, 4, 3, and 2 vary from the design calculations by more than the acceptance criteria. The acceptance criteria are that the measured individual group worth varies from the calculated worth by less than either 10% or 0.1 %ΔK/K (whichever limit is larger) <u>and</u> that the measured cumulative worth of CEA groups 7 through 2 is within 10% of the calculated cumulative worths."

The acceptance criteria was met. See chart on next page.

TABLE 5.5-1

NON-OVERLAPPING CEA GROUP WORTHS

		CYCLE 1, %ΔK/K			CYCLE 1A, %ΔK/K		
CEA GROUP	DESIGN WORTH	MEASURED	ACCEPTABLE LIMITS	LIMITS MET	MEASURED	ACCEPTABLE LIMITS	LIMITS MET
2	1.451	1.297	1.088 - 1.814[1]	Yes	1.470	1.306 - 1.596[2]	Yes
3	0.575	0.517	0.431 - 0.719[1]	Yes	0.500	0.475 - 0.675[3]	Yes
4	1.375	1.262	1.031 - 1.718[1]	Yes	1.300	1.238 - 1.513[2]	Yes
5	0.329	0.311	0.247 - 0.411[1]	Yes	0.395	0.229 - 0.429[3]	Yes
6	0.505	0.470	0.379 - 0.631[1]	Yes	0.548	0.405 - 0.605[3]	Yes
7	0.739	0.707	0.554 - 0.923[1]	Yes	0.709	0.639 - 0.839[3]	Yes
ACCUMULATIVE	4.974	NOT APPLICABLE			4.922	4.477 - 5.471[2]	Yes

1 = Design ± 25%   2 = Design ± 10%   3 = Design ± 0.1 %ΔK/K

5.6 OVERLAPPED REGULATING CEA GROUP WORTH MEASUREMENTS

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 2, 3, 4, 5, 6 and 7 in an overlapped mode was measured.

This measurement was made at a Reactor Coolant System (RCS) temperature of 532°F. Principal purpose of this test was to compare the reactivity worth of the overlap mode for this fuel loading (Cycle 1A) with the reactivity worth of the overlap mode for the original fuel loading (Cycle 1).

Test Results

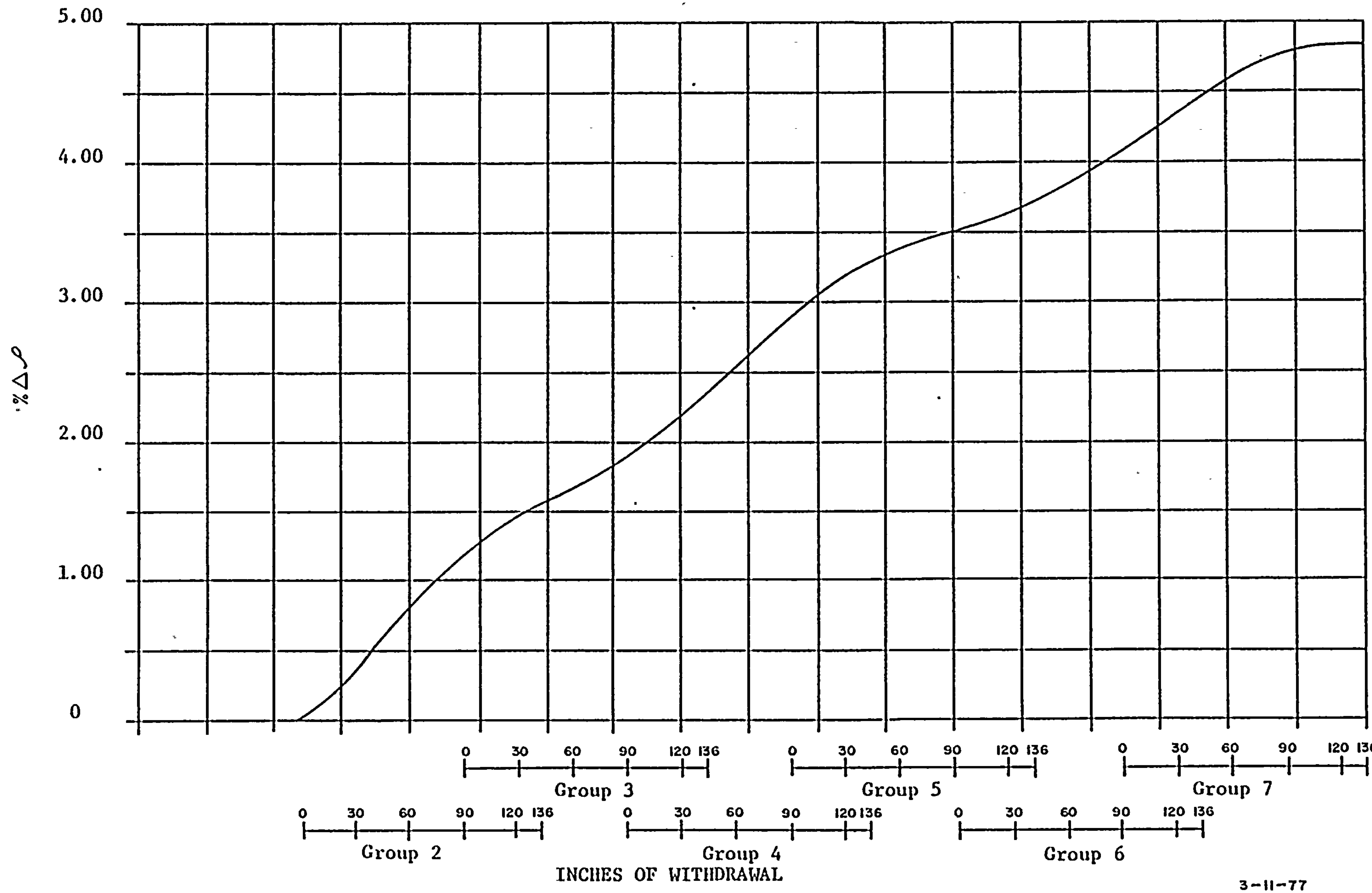
The overlapped integral reactivity worth of CEA Groups 2, 3, 4, 5, 6 and 7 was measured using a soluble boron swap method to maintain criticality while sequentially inserting CEA Groups in increments. The reactivity trace developed by this CEA Group movement was reduced to obtain the relationship between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

Conclusions

The overlap CEA Group worth measurements for Cycle 1A compared satisfactorily with overlap CEA Group measurements for Cycle 1. On this basis, and the conclusion reached in Section 5.5, it was judged not to generate new individual CEA group worth curves but to use those curves generated during Low Power Physics Tests in Cycle 1.

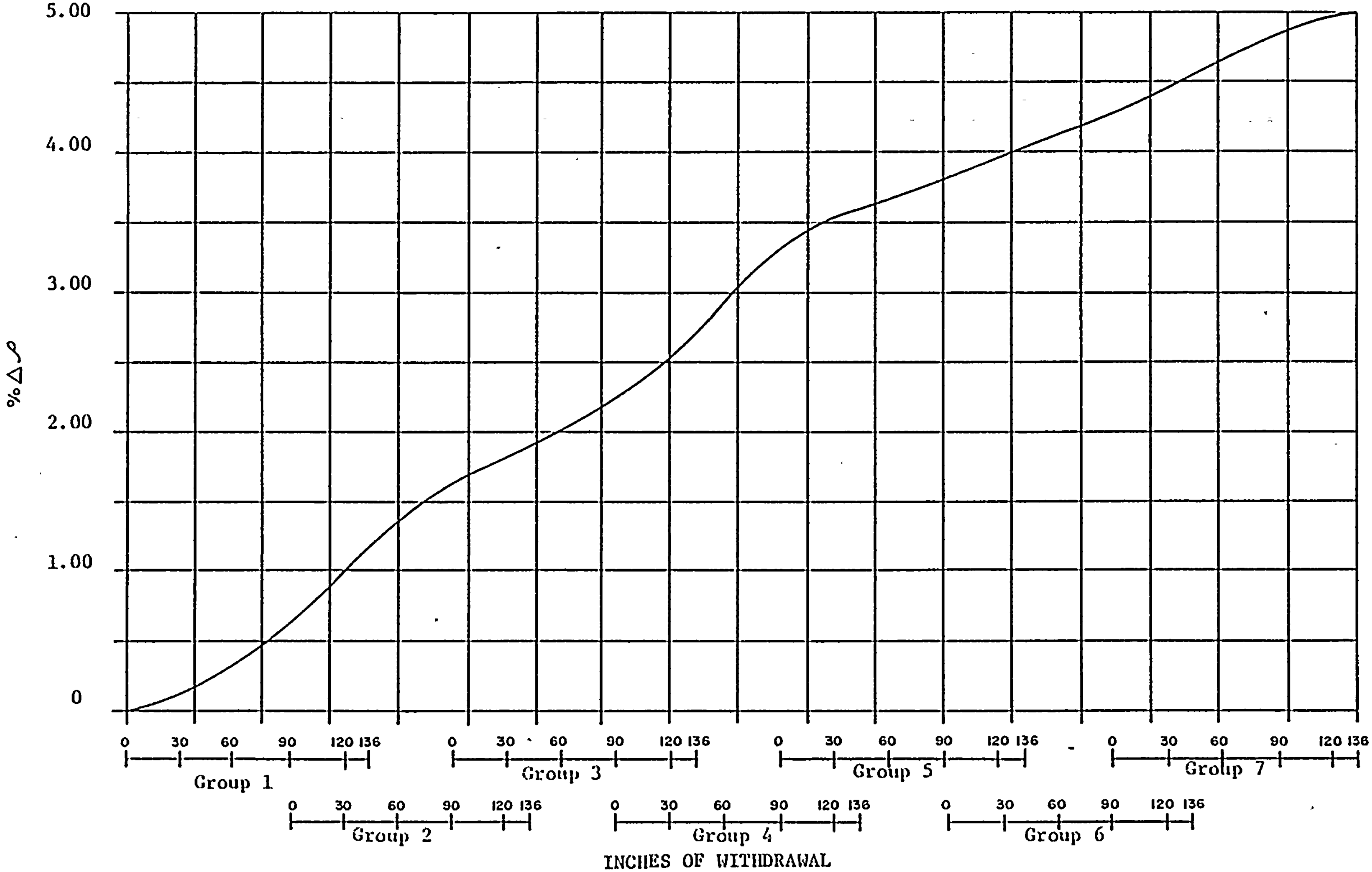
ST. LUCIE UNIT 1
CYCLE 1A, 532°, 2250 PSIA
SEQUENTIAL CEA WORTH

Figure 5.6-1



3-11-77

ST. LUCIE UNIT 1
CYCLE 1, 532°, 2250 PSIA
SEQUENTIAL CEA WORTH
Figure 5.6-2



3-11-77

5.7 CEA 7-1 WORTH MEASUREMENT, GROUP 7 ∿ 100"

Purpose

This measurement was done to get the differential worth of CEA 7-1, which will be used later in the Moderator Temperature Coefficient/Power Coefficient tests.

Test Results

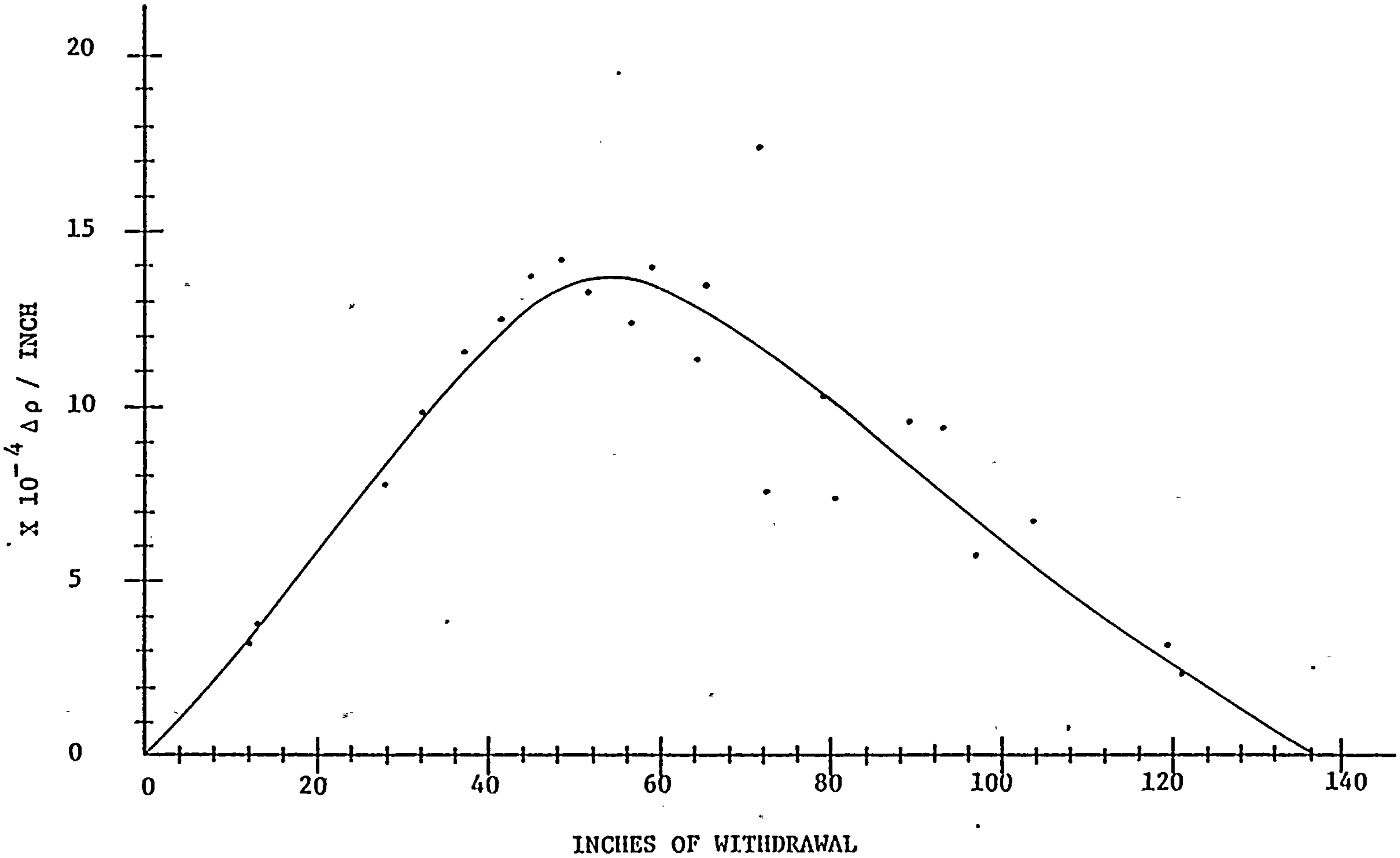
The reactor was critical with CEA group 7 approximately 100 inches withdrawn. CEA 7-1 was borated to UEL (upper electrical limit), diluted to LEL (lower electrical limit), then borated to UEL. As these steps were performed, the reactivity was being monitored on the reactivity trace on the reactivity computer. The final boration left CEA 7-1 back at CEA group 7 height.

Conclusions

Enough raw data was collected to generate an integral rod worth curve for CEA 7-1. The measured value of CEA 7-1 was 0.106$\Delta\%\rho$.

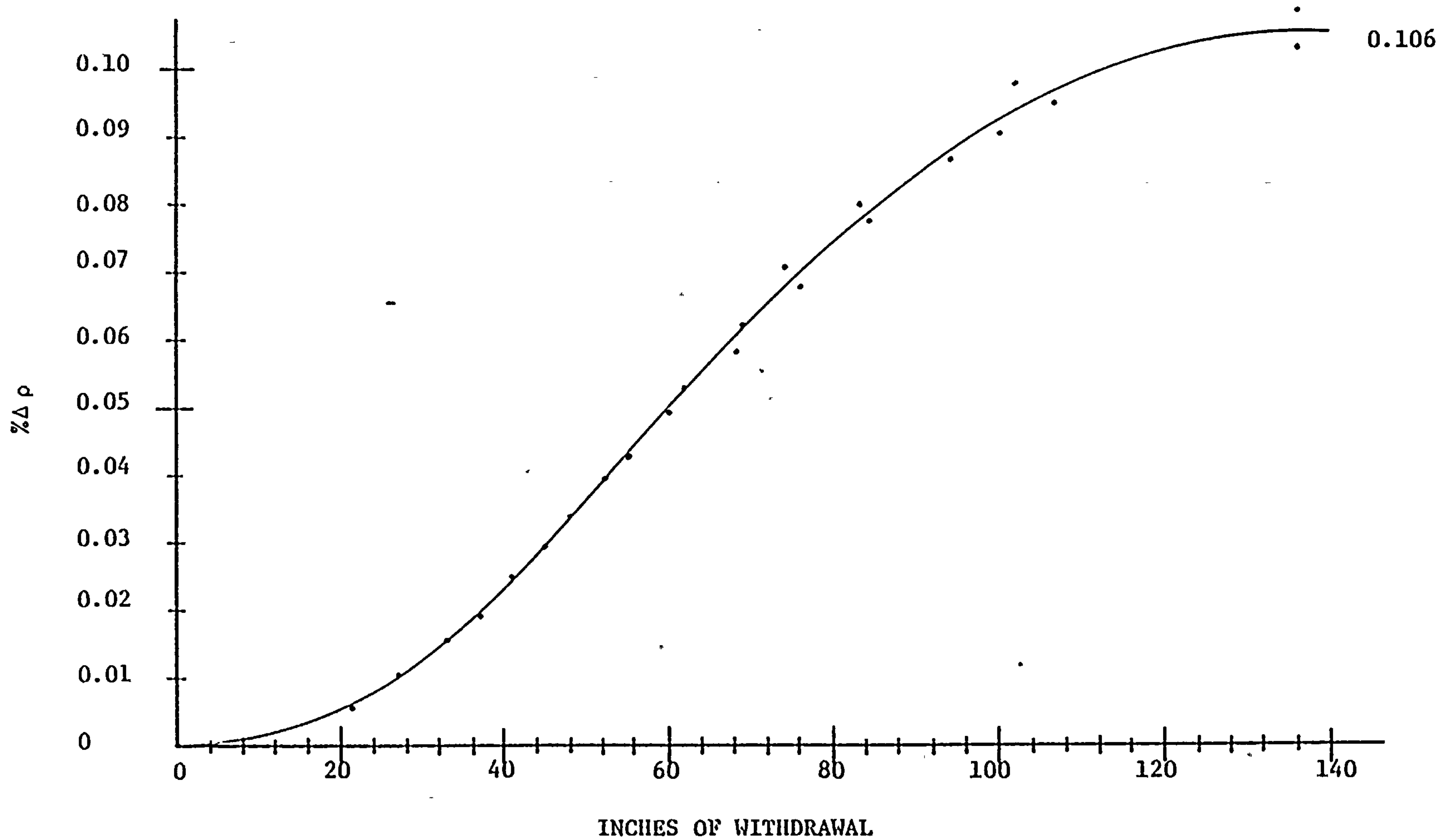
CEA 7-1 DIFFERENTIAL ROD WORTH
HZP, 532°F, 2260 PSIA

Figure 5.7-1



CEA 7-1 INTEGRAL ROD WORTH
HZP, 532°F, 2260 PSIA

Figure 5.7-2



5.8 SHUTDOWN MARGIN VERIFICATION

Purpose

The purpose of this test is to verify the shutdown margin at the zero power dependent insertion limits (ZPDIL) assuming the highest worth CEA is stuck out.

Test Results

The reactivity worths of shutdown group A, shutdown group B, regulating groups withdrawn to the ZPDIL, and the highest worth individual CEA were compiled from test measurements taken during fuel cycle 1.

Conclusions

Section 3.1.1.1 of the Technical Specifications specifies that the shutdown margin should be greater than or equal to 2.45%$\Delta\rho$. The value that was measured was 5.18%$\Delta\rho$.

Table 5.8-1

Worth of Regulating CEA's Withdrawn at the ZPDIL	3.335 %$\Delta\rho$
Worth of Shutdown CEA's, Group A	4.520 %$\Delta\rho$
Worth of Shutdown CEA's, Group B	0.425 %$\Delta\rho$
Total Available Rod Worth at the ZPDIL	8.280 %$\Delta\rho$
Highest Worth Stuck CEA	3.100 %$\Delta\rho$
Actual Shutdown Margin	5.180 %$\Delta\rho$

6.0 POWER ASCENSION TESTS

The power ascension tests were conducted to determine the as-built 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 performed at the 20, 50, 80 and 100% power plateaus. Several other tests were performed at the 14, 25, 30, 40, 60, 70, 75 and 85% power levels.

Power ascension tests through 50% power on fuel cycle 1 was reported in the Startup Test Report, dated December, 1976.

The power ascension testing for fuel cycle 1A repeated some of the same tests below 50% power and completed the 50 to 100% spectrum, also. The Preoperational Test Procedure Number 0010180 titlted "POWER ASCENSION SEQUENCING DOCUMENT" was used as the guideline for this phase of testing. Two revisions were made to this procedure during testing to accommodate the re-ascent to power after fuel reconstitution, correct minor editorial errors, and revise the sequence of testing to best fit plant availability and system load demand.

TABLE 6.0-1

POWER ASCENSION TESTING, CYCLE 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
TCT. 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		

1 = Performed once at this power level
2 = Performed twice at this power level, usually once at the beginning and again just prior to leaving

*Reference - Page 87 of Startup Test Report for St. Lucie Unit 1, Cycle 1, dated December, 1976

TABLE 6.0-2

POWER ASCENSION TESTING, CYCLE 1A

PROCEDURE TITLE AND NUMBER		
NOTE: P=Preoperational O=Operating		
Effluent Monitoring	Chemistry 77	
Calorimetric	DDPS	
Shapshot	DDPS	
Load Cycle	P	0010195
Static CEA Drop	P	0110088
Dynamic CEA Insertion	P	0110089
10% Load Reduction	P	0110090
RCS Flow Det. by Cal. Proc.	O	0120051
Partial Loss of Flow	P	0120081
Total Loss of Flow/Nat Circ	P	0120084
S/G Feedwater Hammer	P	0700080B
APD & SA Baseline	O	1130021
Nuclear & ΔT Power Calib	O	1200051
Linear Power Range Calib.	O	1220052
Auto Control System	P	1400084
Secondary Sample	P	1730080
NSSS Acceptance Run	P	2100082
Generator Trip	P	2100089
Turbine Trip	P	2100090
Loss of Offsite Power	P	2100091
Fixed Incore Alarm Setpoint	O	3200050
Mod Temp Coef & Pwr Coef	O	3200051
Power Defect	P	3200084
Radiation Shielding Eval.	P	3300081
Chem & Radchem Analy.	P	3400081

TABLE 6.0-2 (Cont.)

POWER ASCENSION TESTING, CYCLE 1A

		POWER LEVEL (%)																				
		20	30	50	80	0	40	0	20	0	90	98	0	98	0	98	25	50	85	90	98	S*
Chem. 77																1						
Cal. DDPS		2	2	2	2						2	1				2						
Snp. DDPS		2	2	2	2						2	1				2						
P	0010195																		1			
P	0110088																	1				
P	0110089																	1				
P	0110090																		1			
O	.0120051			1																		
P	0120081				1																	
P	0120084						1															
P	0700080B						1															
O	1130021				1											1						
O	1200051	2	2	2	2						2	1				2						
O	1220052		1	1																		
P	1400084																1	1		1		
P	1730080																					1
P	2100082																				1	
P	2100089													1								
P	2100090											1										
P	2100091								1													
O	3200050			1							1											
O	3200051			1	1											1						
P	3200084					1																
P	3300081															1						
P	3400081				1											1						

1=Performed once at this power level 2=Performed twice at this power level

*S=To be completed sometime during Power Ascension Testing

6.1 Flux Distribution Monitoring

Due to the flux distribution anomaly and poison pin replacement a program of slow ascent to power with a long observation period at 50% was used for the return to power operations. Over a week was used in ascent to 50% power and the plant remained at 50% power for over three weeks. The monitoring program to verify absence of the anomaly is discussed below.

During Cycle 1, an azimuthal tilt developed due to the failure of burnable poison pellets. During the repair/down-time the fuel element assemblies were reconstituted. The reconstituted fuel was reloaded into the core, and, except for two fuel assemblies exchanging core locations, the core was identical to the initial clean fuel. To distinguish this reconstituted fuel and subsequent testing and operation from the original clean fuel, the designation "Cycle 1A" was used.

As Cycle 1A was returned to the 50% power plateau to resume testing in the Power Ascension Testing program, the azmuthal power tilt was closely monitored. Table 6.1-1 represents a few of the many snapshots that were taken. It should be noted that there was extensive time taken to return to the Power Ascension Testing program, to insure adequate time to observe that the fuel was "burning evenly." Although Table 6.1-1 ends on January 11, 1977, Tq was still closely monitored during the remainder of the Power Ascension Tests and no indication of any anomaly has been observed.

SELECTED Tq VALUES FROM CYCLE 1A

TABLE 6.1-1

DATE	TIME	% POWER	Tq	F_r^T	GROUP 7
09DEC76	2000	1	–	–	ARO*
10DEC76	1127	21	0.016	1.358	ARO
11DEC76	0630	30	0.013	1.287	ARO
12DEC76	0849	30	0.013	1.288	ARO
13DEC76	1106	31	0.012	1.296	ARO
14DEC76	0830	30	0.014	1.297	ARO
15DEC76	1820	32	0.011	1.295	ARO
16DEC76	0936	42	0.011	1.302	ARO
17DEC76	1631	50	0.009	1.302	ARO
18DEC76	1730	50	0.009	1.307	ARO
19DEC76	1230	51	0.009	1.302	ARO
20DEC76	0615	50	0.009	1.306	ARO
21DEC76	2020	51	0.008	1.300	ARO
22DEC76	0628	51	0.008	1.304	ARO
23DEC76	0810	51	0.008	1.314	ARO
24DEC76	1548	51	–	–	7 at 100
25DEC76	0802	51	0.008	1.349	7 at 100
26DEC76	0707	51	–	–	7 at 100
27DEC76	0600	51	0.008	1.350	7 at 100
28DEC76	1530	51	0.007	1.306	7 at 72
29DEC76	0728	51	0.006	1.302	
30DEC76	2142	51	0.006	1.299	7 at 72
31DEC76	0015	51	0.007	1.286	ARO
01JAN77	1116	51	0.006	1.302	ARO
02JAN77	1515	51	0.006	1.306	ARO
03JAN77	1116	51	0.006	1.300	ARO
04JAN77	–	–	–	–	
05JAN77	1930	51	–	–	
06JAN77	0300	51	0.006	1.286	ARO
07JAN77	1004	51	0.007	1.305	ARO
08JAN77	1230	51	0.007	1.300	ARO
09JAN77	1530	60	0.006	1.309	ARO
10JAN77	1048	61	0.007	1.307	ARO
11JAN77	0600	70	0.005	1.307	ARO

*All Rods Out

6.2 PLANT POWER CALIBRATION

Purpose

The purpose of the test was to:

(1) Determine core thermal power by means of a Reactor Coolant System heat balance.

(2) Adjust the Power Range Safety Channels and Δ T 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.

Test Results

Feed flow Calorimetrics were conducted through power ascension. The Calorimetrics were used to calibrate nuclear instrumentation. 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 periodically throughout power ascension.

Conclusions

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, and this was done.

It should be noted that the Calorimetric program had been slightly revised during the fuel reconstitution shutdown. An error was introduced which caused Calorimetric power to be 2.5% lower than actual (at 80% power). This was found and corrected before the plant exceeded 80% power. The rest of the program was checked to verify no further errors existed. (LER 335-77-4 dated February 18, 1977).

6.3 FIXED INCORE DETECTOR ALARM SETPOINTS

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.

Results

The procedure instructs that a set of data, containing 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. It was also done when the Peak Linear Heat Generation Rate interim limit due to STRIKIN II coding errors was lifted.

Conclusions

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 setpoint.

6.4 MODERATOR TEMPERATURE COEFFICIENT AND POWER COEFFICIENT

Purpose

The purpose of these tests were to determine the at-power moderator temperature coefficient (MTC) and power coefficient at the 50, 80 and 100% nominal power levels. The Technical Specifications have set limits for temperature coefficient. Combustion Engineering design values ($\pm$10%) were used for power coefficient limits.

Test Results

At the beginning of these (three) plateaus, the plant was steady state at the nominal power level with CEA group 7 ≈ 100 inches withdrawn, with CEA 7-1 in the manual individual (MI) control mode. The test at each plateau was done in two parts.

The moderator temperature coefficient (MTC) was calculated by holding power constant and lowering RCS Tave. The resultant reactivity addition was compensated by insertion of CEA 7-1. The RCS Tave was returned to nominal value with the resultant negative reactivity addition being compensated by CEA 7-1 withdrawal. This technique was repeated several times and the average temperature coefficient values were used.

The power coefficient was measured by holding the RCS Tave constant and lowering power. The resultant reactivity change was compensated by movement of CEA 7-1. The power was returned to nominal value and CEA 7-1 again was used to compensate for reactivity changes. This technique was repeated several times and the average power coefficient values were used.

Conclusions

The reactivity worth of CEA 7-1 is known. By measuring CEA 7-1 height change (Δh) we have, in effect, measured reactivity change ($\Delta\rho$). By comparing the two variables in each case ($\Delta\rho/\Delta^{o}F$) and ($\Delta\rho/\Delta\%$) we have the respective coefficients.

It should be noted that the reactivity change associated with the variable Tave process is actually the isothermal temperature coefficient (ITC). ITC is made up of two components, moderator temperature coefficient (MTC) and fuel temperature coefficient (FTC). ITC = MTC + FTC. FTC is also known as doppler feedback. ITC is what was measured. The FTC value that was used was taken from design values. MTC is calculated by subtracting FTC from ITC. MTC = ITC - FTC. Moderator temperature coefficient values and power coefficient values at different power levels are presented along with their respective limits, on the table on the next page. The measured values of MTC and Power Coefficient met all acceptance criteria.

TABLE 6.4-1

AT POWER DETERMINATION OF
MODERATOR TEMPERATURE COEFFICIENT AND POWER COEFFICIENT

Nominal Reactor Power	50%	80%	100%

Moderator temp. coef.	$\Delta k/k$ / $^{\circ}F$	$\Delta k/k$ / $^{\circ}F$	$\Delta k/k$ / $^{\circ}F$
Measured, x 10^{-4}	-0.35	-.022	-0.307
Tech-Spec Limit, x 10^{-4}	>-2.25 <+0.50	>-2.25 <+0.20	>-2.25 <+0.20
FSAR Limit, x 10^{-4}	>-1.40 BOL <+0.50	>-1.40 BOL <+0.50	>-1.40 BOL <+0.50

BOL = Beginning of Life

Power Coefficient	$\Delta k/k$ / %	$\Delta k/k$ / %	$\Delta k/k$ / %
Measured, x 10^{-4}	-1.06	-1.10	-0.965
C.E. design, x 10^{-4}	-1.00	-1.025	-0.98

Date	04 Jan. 77	25 Jan. 77	07 Mar. 77

6.5 TURBINE VALVE TESTS

Purpose

The purpose of performing this evolution was to verify capability of plant systems to perform this evolution at power.

Test Results

The test recommended by the turbine vendor is essentially a stroke test. That is, the valves are cycled closed and open to verify freedom of movement. There are two valves (Reheat and Interceptor) in each of 4 lines from the high pressure to the low pressure turbine via the Moisture Separator Reheaters. These were simply closed and slowly reopened, one set at a time. The throttle valves (4) which are primarily for startup and quick isolation of the turbine were tested in the same manner. The governor valves which directly control turbine loading are tested in single valve control mode with impulse (1st stage) pressure control in automatic. In single valve control mode the 4 governors act together in response to a single signal. When one valve was closed and reopened, the other three acted automatically to maintain turbine load essentially constant.

Conclusions

The evolution went smoothly. It demonstrated capability to test the valve at power and that our procedural requirement to reduce power to 90% is more than adequate to prevent exceeding power limits during the minor transients imposed by this testing. This evolution was performed during the initial ascent to 80% power.

6.6 PARTIAL LOSS OF FLOW TEST

Purpose

The purpose of this test is to observe plant response to a partial loss of reactor coolant flow while at power and verify that the Reactor Protective System low flow trip units initiate a reactor trip.

Test Results

With the plant at 80% power, one Reactor Coolant Pump was turned off, reducing flow to about 80% of normal full flow. The RPS initiated a low flow trip and all 4 of the low flow trip channels responded to the low flow condition. Overall plant response was as expected with no significant problems.

Conclusions

The plant responsed properly to the trip from 80% power. The RPS low flow trip units are capable of sensing reduced flow within the required time limits.

6.7 POWER DEFECT AND XENON WORTH AFTER SHUTDOWN

Purpose

Various plant and reactor parameters affect core reactivity changes. Temperature, RCS boron concentration, CEA position and other variables have a pronounced affect on reactivity. Power level also contributes to reactivity changes. The higher the power level, the more negative reactivity is inserted into the core. This phenomenon is called "power defect". One of the purposes of this test was to measure power defect.

When a reactor is critical, one of the by-products, both directly due to fission and due to decay of other fission produced elements, is xenon. Xenon has a very high neutron absorption cross section and its presence in a reactor has a direct affect on reactivity. Because xenon absorbs neutrons, xenon inventory in a core is a function of both power history and present power level. The severest effect of xenon follows a rapid reactor shutdown from power (i.e., a reactor trip). The second purpose of this test was to measure the xenon worth after a shutdown as a function of time.

Test Results

The reactor was at 80% steady state, equilibrium xenon and ARO when it was tripped. Within an hour, the reactor was brought critical and at zero power. The position of CEA group 7 was noted. The difference in CEA position from 80% power ARO partially represents the reactivity due to being at 80% power (i.e., power defect). As xenon production was being compensated by CEA withdrawal, the reactivity steps were measured on the reactivity computer. The amount of reactivity addition to return to ARO was calculated and used in the power defect measurement section of this test.

CEA group 7 was then diluted in to approximately 20 inches withdrawn. Negative reactivity insertion due to xenon production was compensated by group 7 withdrawal. A record of reactivity insertion and time after reactor trip was kept. A chart of these two items plotted against each other was made. This chart fulfills the xenon worth measurement section of this test.

6.7 POWER DEFECT AND XENON WORTH AFTER SHUTDOWN (cont)

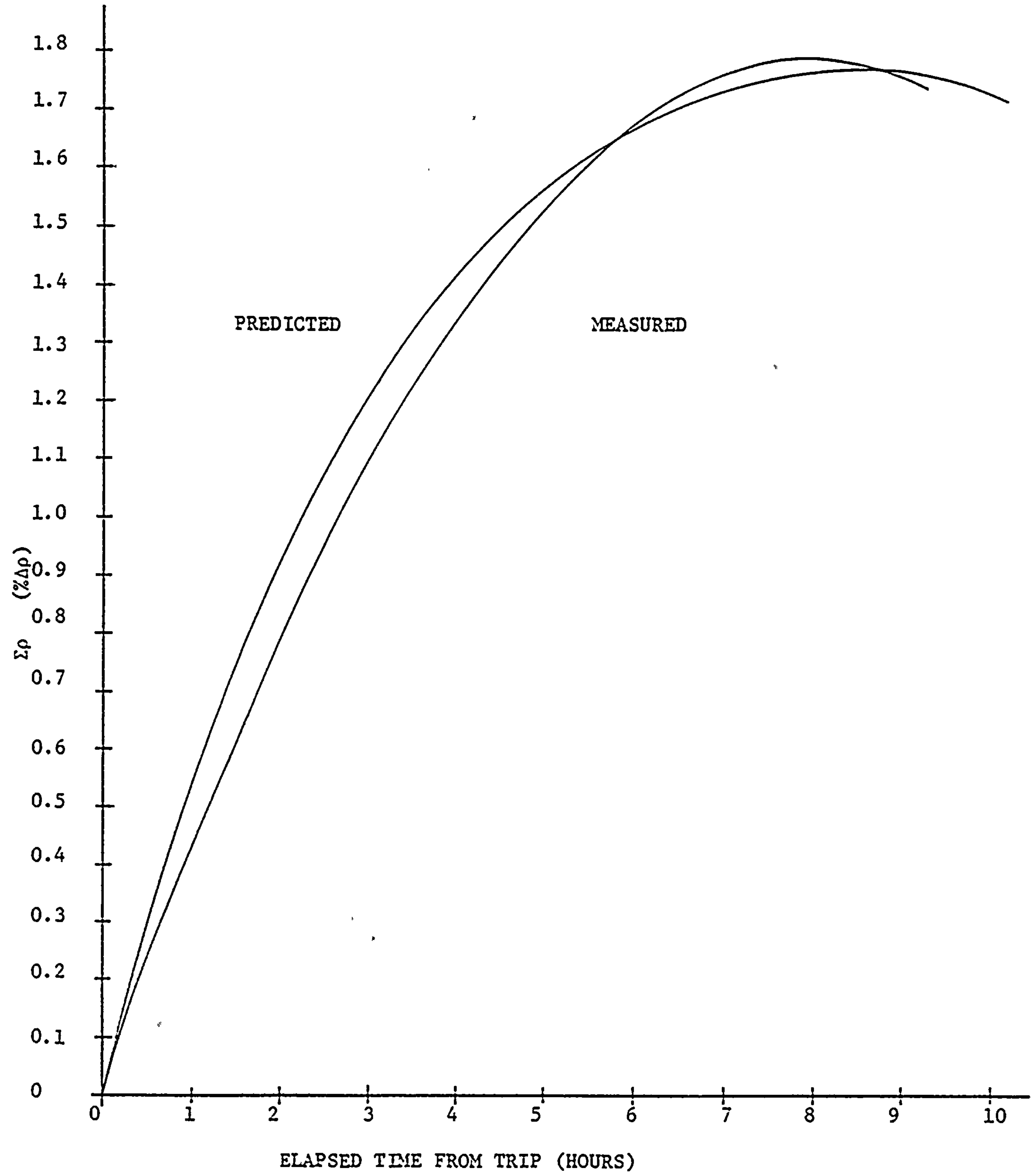
Conclusions

The power defect was actually in two parts. The first part was the negative reactivity that was inserted due to xenon production during the hour between the reactor trip and criticality. This value can be infered from the size,shape,magnitude,and time on the xenon worth curve (see next paragraph). The second part was the negative reactivity that was compensated for by withdrawal of CEA group 7 to ARO. The power defect from 80% power was 0.75 %$\Delta\rho$.

The peak value of xenon worth was 1.78 %$\Delta\rho$ eight hours following the reactor trip. The plot of the measured xenon worth versus time compared to the predicted xenon worth versus time is on Figure 6.7-1. From Figure 6.7-1, it can be seen that there is very good agreement between the predicted and actual value. For instance, the difference between predicted and actual peak xenon was less than 2%. The acceptance criteria were met.

FIGURE 6.7- 1

XENON WORTH



6.8 TOTAL LOSS OF FLOW/NATURAL CIRCULATION TEST

Purpose

The purpose of this test was to verify that the RPS would initiate a reactor trip on low flow after a total loss of RCS flow and to verify that natural circulation occurs.

Test Results

With the plant operating at 40% power, all 4 RCP's were turned off simultaneously. A reactor trip from low flow then occurred followed by turbine and generator trip. The plant was then taken to hot standby conditions except that the RCP's were not restarted until natural circulation flow was verified. Starting before the trip, ΔT power data was taken until one hour, 15 minutes after the trip. ΔT power initially dropped to nearly zero, returned to the pre-trip value and then started decreasing. For the last 45 minutes before pumps were restarted, ΔT power slowly but steadily decreased (as did T_{hot}), verifying that natural circulation was occurring. This allows controlled decay heat removal/cooldown without RCP flow.

Conclusions

The RPS low flow trip units initiated the trip and all 4 channels detected the trip condition within the required time. Verification of natural circulation flow and initiation of a low flow trip by the RPS satisfied the acceptance criteria given in the Purpose.

6.9 STEAM GENERATOR FEEDWATER HAMMER TEST

Purpose

The purpose of this test was to verify the absence of water hammer in the steam generator feedwater piping when the steam generator was drained below the feed ring and then refilled. This was repeated, as the method of preventing rapid draining of the feed ring was modified since performance of the test during the first ascent to power.

Test Results

Following a trip from 40% reactor power (after several days of operation at 80%), 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 of two motor-driven Auxiliary Feedwater Pumps (785 gpm). Level was again reduced and held below the feed ring for 2 hours. Then level was raised to normal at the design flow rate for one motor-driven Auxiliary Feedwater Pump (300 gpm). The NRC had requested decay heat equivalent to that 2 hours after shutdown from at least 2 days of operation at 30% power or greater. This corresponds to slightly less than 5 megawatts (Mw). Actual decay heat was calculated to be 7.2 Mw for the first test run and 6.9 Mw for the second. The behavior of the feedwater piping was monitored by: observers inside the containment, installed RCS noise monitoring equipment, a test pressure transducer and recorder, and by measurements of line and restraint positions before and after each test. No evidence of feedwater hammer was observed.

Conclusions

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

6.10 LOSS OF OFF-SITE POWER AND LOAD REJECTION

Purpose

The purpose of this test was to evaluate unit reliability during a load rejection and a loss of off-site AC power.

Test Results

With the unit at 20% power and off-site AC power supply breakers open, the unit was separated from the system distribution grid. This was done to evaluate response to this 120 MW load rejection and determine the unit capability regarding carrying its own auxiliary loads (about 40 MW) while divorced from the grid. This information was of interest to FP&L and did not adversely affect the main objective of the test which was response to loss of off-site power.

After the unit carried its own auxiliaries briefly, the generator was tripped. This demonstrated unit response to a trip and loss of off-site power. The general acceptance criteria was that systems required during loss of off-site power function as designed. The specific acceptance criteria were:

1. Decay heat is satisfactorily removed and Tavg is reduced to the no-load value (532°F) or less.
2. Steam generator levels can be manually restored and maintained within limits by feeding with the auxiliary feedwater pump.
3. At least one emergency diesel generator starts and supplies power.
4. At least one Intake Cooling Water pump starts and operates.
5. At least one Component Cooling Water pump starts and operates.
6. RCS pressure shall not exceed 2500 psia during the test.
7. The RCS shall achieve final conditions of pressure <2400 psia with pressurizer cooldown not exceeding 200°F in one hour and steam generator pressure <985 psig with RCS cooldown not exceeding 100°F in one hour.

6.10 LOSS OF OFF-SITE POWER AND LOAD REJECTION (Cont)

Conclusions

The unit was capable of handling the load rejection with the steam dumps controlling pressures and carrying its own auxiliaries thus providing data for evaluation and demonstrating ability to handle separation from the grid at low power levels. All the acceptance criteria (1-7 above) were met and the plant was returned to and maintained in hot standby without off-site AC power. The systems required during loss of off-site AC power did function properly as designed.

It should be noted that both diesels did start and supplied power and two ICW and CCW pumps started, as designed, thus exceeding the minimum acceptance criteria.

6.11 10% LOAD REDUCTION - TURBINE RUNBACK

Purpose

The purpose of this test was to verify turbine runback would occur on a (simulated) rod drop and to determine plant response resulting from a load reduction.

Test Results

An actual rod drop occurred with the plant operating at 90%. Turbine runback was properly initiated and plant performance was satisfactory. This met the intent of the procedure and documentation was gathered to support the proper performance of the systems involved.

Conclusions

Turbine runback is properly initiated by a dropped rod and plant response is satisfactory.

6.12 TURBINE TRIP TEST

Purpose

The purpose of this test was to verify control systems perform as designed to bring unit to hot standby conditions following a trip from 100% power.

Test Results

The test was to demonstrate the following items would occur following the turbine trip.

1. The generator and reactor tripped.

2. RCS pressure did not exceed 2500 psia.

3. Steam dump and bypass systems returned the plant to nominal hot standby conditions - RCS temperature about 532°F.

4. Steam generator pressure did not exceed 1025 psig.

5. Steam generator levels were manually returned to normal hot standby level.

These items did all occur properly after the turbine trip, verifying the controls maintained the unit within design parameters.

Conclusions

Satisfactory completion of the acceptance criteria 1-5 above demonstrated that the control systems performe as designed, and bring the unit to hot standby conditions following a trip from 100% power with minimum operator action required.

6.13 GENERATOR TRIP TEST

Purpose

This test demonstrated that the unit could accept design load rejection--that is, a loss of generator load from 100% power.

Test Results

The acceptance criteria for this test were:

1. RCS achieves stable conditions, <2400 psia, S/G pressure <985 psig.
2. The turbine does not exceed its design overspeed of 111%.
3. RCS pressure does not exceed 2500 psia during the test.
4. Steam generator pressure does not exceed 1025 psig during the test.

In addition, the turbine and reactor protective systems must terminate the transient before any limiting set-points are exceeded.

Conclusions

The reactor tripped and initiated turbine trip on high pressurizer pressure at the proper setpoint. Steam generator levels were restored to normal and maintained by normal operator action. The 4 acceptance criteria were satisfactorily met also. This demonstrates that the unit can accept a design (100%) load rejection and be brought to hot standby conditions with minimum operator action.

6.14 LOAD CYCLE TEST

Purpose

This test was conducted to evaluate plant and nuclear parameters in a typical load cycle operation. FSAR Table 4.4-2 provides limiting values for the three dimensional point peaking factor F_q, and the integrated radial peaking factor F_r (also identified as the total enthalpy rise factor, ΔH hot channel/ΔH core, $F_{\Delta H}$).

Test Results

The plant was initially at approximately 85% power, steady state. The power was decreased, at the nominal rate of 30%/hour, to approximately 40%. The power was increased, at the nominal rate of 30%/hour, to approximately 85%. Snapshot paper tapes containing incore detector readings, CEA positions and calorimetric powers, were taken at the 85%, 40%, and 85% plateaus. Various plant pressures, temperatures and flows were recorded on strip chart recorders.

Conclusions

From the data that was monitored on the strip chart recorders (see Table below) and the nuclear measured via snapshot/GINCA (see Table on next page), it was determined that all levels of acceptance criteria had been met.

Table 6.14-1

Parameters monitored by Strip Chart Recorders	
Pressurizer pressure	Pressurizer level
S/G 1A1 Pressure	S/G 1B1 Pressure
S/G 1A1 level	S/B 1B1 level
S/G 1A1 Feedwater flow	S/G 1B1 Feedwater flow
S/G 1A1 Steam Flow	S/G 1B1 Steam Flow
Loop 1A1 T_h (NR)	Loop 1B1 T_h (NR)
Loop 1A1 or 1A2 T_c (NR)	Loop 1B1 or 1B2 T_c (NR)
1st Stage Pressure	Generator MW
Nuclear Power NI-9	

TABLE 6.14-2

LOAD CYCLE TEST

(conducted March 19, 1977)

TIME	POWER (%)	F_q	$F_{\Delta H}$
0100	77.35		
0200	83.57		
0300	83.57		
0400	80.51		
0500	82.91		
0530	82.62	1.57	1.28
0600	72.48		
0700	45.19		
0730	37.28	1.61	1.30
0800	47.11		
0900	76.79		
0930	80.12		
1000	83.30	1.55	1.28
1030	82.27	1.55	1.28
1100	81.81	1.56	1.28
FSAR LIMIT		≤ 2.71	≤ 2.02

6.15 STATIC CEA DROP

Purpose

The purpose of this test was to observe the effects of a control element assembly (CEA) insertion and withdrawal on core power distribution. The subsequent azmuthal power tilt (Tq) oscillation and dampening, and the effects on linear heat rate was also observed.

Test Results

The test was conducted at 50% power. At this power level, the trends could be seen, but the fuel would not be subjected to as large power transients as would be encountered at higher power levels and margins from peaking would be greater.

Initial conditions were all rods out, 50% nominal power and equilibrium xenon. The static measurement consisted of diluting rod B-7, a dual shutdown CEA, to its lower electrical limit (LEL), while holding reactor power constant. The dilution/insertion took approximately one hour.
The azmuthal power tilt grew until it reached a maximum. This required approximately four hours after LEL. The CEA was then borated out, again holding reactor power constant. Boration/withdrawal took approximately two hours.

Conclusions

During the time the CEA was in the core, reactor protection system (RPS) Channels A and D showed an increase in nuclear power, while Channels B and C showed a decrease in nuclear power.

Tilt (Tq) indication at the beginning of the test (before the CEA was inserted) was 0.1%. The maximum value of Tq occurred $\sim$ 4 hours after the CEA reached LEL and was $\sim$20.2%. As the CEA was withdrawn, Tq decreased in value. The tilt oscillations continued, although dampened with time. The curve of reactor power, CEA B-7 position, and Tq plotted against time can be found on the next page.

The effects of the dropped CEA were measured and the ratio of afterdrop to before drop radial fuel peak is less than or equal to 1.154. Also, an acceptable value of linear heat rate was not exceeded.

F_r^T (after drop) / F_r^T (before drop) = 1.153 $\leq$ 1.154 (FSAR Limit)

Peak Linear Heat Rate = 8.2 KW/ft

TABLE 6.15-1

STATIC CEA DROP

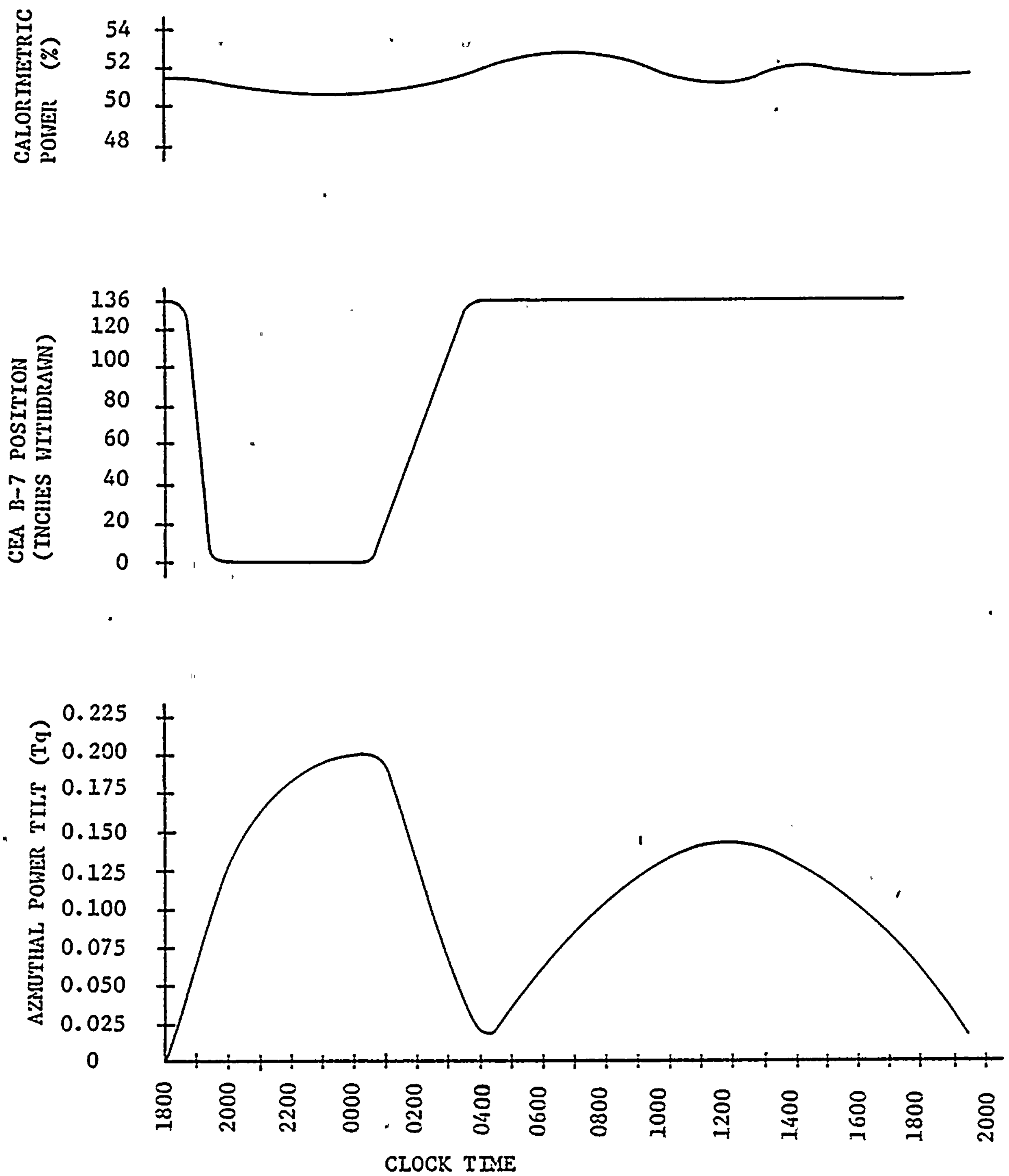
DATE	TIME	POWER	B-7	Tq*	RPS NUCLEAR POWER A	B	C	D
MARCH 15, 1977	1800	51.6	U	0.001	51.4	51.5	51.5	51.6
	1825		U					
	1830	51.9	109		52.6	52.1	52.0	53.0
	1900	51.7	50	0.075	55.3	50.1	48.9	54.9
	1930	51.6	10		58.3	49.7	47.7	57.2
	1935		L					
	2000	51.3	L	0.136	59.0	49.2	47.0	57.8
	2100	51.0	L	0.161	60.0	48.5	46.1	59.3
	2200	51.1	L	0.181	60.6	47.7	45.5	58.9
	2300	51.1	L	0.193	61.2	47.4	44.6	59.3
MARCH 16, 1977	0000	51.0	L	0.201	61.3	46.9	44.1	59.2
	0100	50.7	L	0.202	60.9	46.5	43.7	58.8
	0115		L					
	0130	51.5	30		60.9	47.6	45.0	58.9
	0200	51.3	58	0.137	58.6	48.5	46.5	57.1
	0300	51.9	117	0.060	54.3	49.5	50.3	54.7
	0314		U					
	0330	51.9	U		53.0	50.2	49.7	52.8
	0400	51.9	U	0.018	52.5	51.2	51.1	52.7
	0500	52.5	U	0.025	51.5	53.1	53.4	52.1
	0600	52.5	U	0.059	52.1	52.7	52.7	52.4
	0700	52.2	U	0.088	50.4	53.1	53.4	51.1
	0730	52.4	U	0.101	49.9	53.6	54.1	50.7
	0830	52.4	U	0.118	49.2	54.4	55.1	50.4
	0930	51.9	U	0.133	48.3	54.6	55.4	49.5
	1030	51.3	U	0.141	47.8	54.8	55.6	49.2
	1130	51.2	U	0.143	47.5	54.5	55.5	49.0
	1230	51.2	U	0.141	47.8	54.6	55.4	49-2
	1330	52.3	U	0.130	48.9	55.1	55.9	50.3
	1430	52.5	U	0.114	49.3	54.5		50.4
	1530	51.5	U	0.102	49.4	53.5	53.9	50.4
	1630	51.5	U	0.081	49.7	52.4	52.5	50.4
	1730	51.4	U	0.065	50.3	51.6	51.5	50.7
	1830	51.3	U	0.041	51.0	50.6	50.3	51.1
	1930	51.4	U	0.022	51.8	50.0	49.6	51.6

U = UPPER ELECTRICAL LIMIT L = LOWER ELECTRICAL LIMIT

*Azmuthal Power Tilt from Excore Detectors

FIGURE 6.15-1

STATIC CEA INSERTION



6.16 DYNAMIC CEA INSERTION

Purpose

To demonstrate the ability of the reactor protective protection system (RPS) to detect a dropped control element assembly (CEA) under typical operating conditions. To demonstrate that the control element drive motor (CEDM) limit switches can independently detect a dropped rod and indicate this on the Core Mimic Display. To demonstrate that the associated annunciator circuits respond to a dropped CEA. To demonstrate that the digital data processor system (DDPS) will indicate a rod has been dropped on the CEA position log printout.

Test Results

This test was performed at a nominal power level of 50%, with all rods out (ARO) and the turbine under valve position limit. The control element drive system (CEDS) was placed in the manual individual (MI) control mode with CEA 7-1 selected. The CEA was dropped by opening the 240 volt breaker in the coil power programmer. CEA 7-1 was the CEA selected to meet the requirement regarding dropping the furtherest (from the excore detectors) detectable CEA. Reference FSAR section 15.2.3.1.

The following plant and nuclear parameters were monitored on recorders throughout the test:

Loop 1A Th	Loop 1B Th
Loop 1A Tc	Loop 1B Tc
Loop 1A Steam Flow	Turbine First Stage Pressure
SG 1A pressure	SG 1B pressure
SG 1A level	SG 1B level
Loop 1A feed flow	Generator Output (MW)
Pressurizer Pressure	Pressurizer Level
NI 9 upper	NI 9 lower
NI 10 upper	NI 10 lower

Conclusions

The RPS detected the rod tested, determined it to be dropped and indicated this on the RPS panel. The associated annunciator circuits K9, K14, K18, K24, K27, K30 and K33 responded to the dropped rod. The CEDM limit switches detected CEA 7-1 and verified on the Core Mimic Display that it was dropped. The DDPS indicated on the rod position log that CEA 7-1 had been dropped, thus all acceptance criteria were met.

6.17 AUTOMATIC CONTROL SYSTEM CHECKOUT, STEAM GENERATOR LEVEL CONTROL, CEA REGULATING SYSTEM, AUTOMATIC TURBINE CONTROL AND LOAD SWING TEST

Purpose

The purposes of this test were:

To demonstrate the capability of the CEA Regulating System to cause CEA insertion when required under steady state and normal transient operation.

To verify that the feedwater regulating system for 1A and 1B steam generators will give an adequate and stable response during steady state and expected transient conditions.

To verify smooth response of the turbine control system during load increases or decreases.

Test Results

Testing was conducted at reactor power levels of 30%, 50%, and 90%. Each series of tests at a given plateau consisted of a 2% turbine load reduction at a rate of 1/2%/min, and a 10% turbine load reduction at a rate of 1%/minute. Both CEA Regulating System channels were tested in this manner. In addition, at each test plateau, one channel was tested with a 10% turbine load reduction at a rate of 5%/minute, and the other channel tested by a 5% "step" reduction in turbine load at a rate of 200 MWE/minute followed by a second such step.

Each test at a given plateau demonstrated the following:

The CEA Regulating System has the capability to decrease reactor power to minimize reactor - turbine mismatch during load reductions.

Steam generator level can be controlled in the automatic mode without significant oscillations during steady state operation and following transients.

The turbine control system performed the load increases and decreases required by the test in a controlled manner with no undamped or divergent oscillations as indicated by generator megawatts or turbine first stage pressure.

Conclusion

Testing showed that the CEA Regulating System, steam generator level control system and turbine control system respond when in automatic mode to control plant conditions during normal steady state operation, minimize reactor-turbine mismatch during load reductions, and stabilize plant conditions subsequent to transients. Thus, all acceptance criteria as listed in the purpose, were satisfactorily met.

6.18 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) ACCEPTANCE RUN

Purpose

The purposes of the test were to:

(1) Verify reliable steady-state full power capability of the Nuclear Steam Supply System portion of the plant at or above the 200-hour initial warranted power level of 2450 MWt.

(2) Verify 100 hour continuous operation at the maximum warranted power level of 2560 MWt.

(3) Verify steam moisture at warranted power did not exceed 0.2%.

Test Results

The first 100 hours of the warranty run was satisfactorily conducted at a power level at or above 2450 MWt.

The second 100 hours of the warranty run was satisfactorily conducted at a power level of 2560 MWt. The actual test was conducted over a period of 111 hours due to a power reduction caused by a main condenser tube leak. This did not invalidate the 100 hour run as the cause was not related to the NSSS. Due to plant and test equipment availability we have not yet measured steam moisture.

Conclusions

The NSSS was demonstrated capable of sustained operation during the warranted period of 200 hours. Steam moisture will be measured as soon as practical but we will not issue a report unless significant problems are noted.

6.19 Shielding Effectiveness and Plant Radiation Level Measurements

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.

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%, 50% and 100%. In addition to personnel performing surveys, two special neutron monitoring systems were available allowing dose rate determinations to 20 Rem/hour (neutron only). Due to excessive exposure rates found at 50% power, surveys were not taken inside containment at 80% and 100% power.

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 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.

6.19 (Con't.)

Conclusion

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.

Surveys determined that areas above FSAR limits are those in and adjacent to the containment. Other than the containment itself, the area which presents the greatest exposure problem is the spent fuel handling 62' elevation which has exposure levels of 100 mRem/hr. (neutron plus gamma) where the FSAR specifies 2.5 mRem/hr. maximum combined neutron plus gamma.

Of primary concern are the high levels found at the 62' elevation of the containment. Exposures of 12 Rem/hr were found at 50% power. However, no single areas were found other than the refueling machine, which were not in the same range of exposure rate. This uniformity of dose rate on the 62' elevation is postulated to be from backscatter from the containment vessel walls and ceiling. The high dose rates in the containment severely limit and at most power levels prohibit personnel access to areas which require periodic or special inspection.

As noted in FSAR Section 12.1, the analytical results indicated the need for shielding and so the design effort for St. Lucie 1 preceded the measurement program. We have now submitted to the NRC a report entitled "Neutron Streaming Report" letter L77-126 dated 4-25-77. This report fully describes the neutron problem and our proposed solution(s) to correct it and gives more detailed neutron survey results.

6.20 CHEMISTRY AND RADIOCHEMISTRY TESTS AT POWER

Purpose

Chemical and Radiochemical analyses of the Reactor Coolant System (RCS) and the steam generators were performed to determine corrosion data, fission and activation product levels and buildup and to detect failed fuel and various impurities which could enhance corrosion. Also to ensure that primary and secondary water chemistry meet the criteria set forth in the St. Lucie Unit 1 Chemistry Manual for system protection.

Test Results

Chemical and Radiochemical tests were performed in accordance with the St. Lucie Unit 1 Chemistry Manual and Pre-Operational Test Procedure #3400081. This included samples at various power levels up to and including nominal 100%.

Primary (RCS)

1. During and following re-startup activities all parameters were within specified limits with few exceptions as described below. Iodine activities showed no new evidence of fuel failure.

2. Lithium - Lithium was added in the form of LiOH during heatup and prior to criticality to 1.2 ppm. Subsequent additions maintained Li concentrations at the high end of specifications to aid in building a protective corrosion film.

3. Dissolved Oxygen (D.O.) - Hydrazine ($N2H_4$) was added to the primary system during the fill and vent evolution to a maximum of 30 ppm N_2H_4 along with purges of the VCT scavenged oxygen in the system. Subsequent additions of N_2H_4 and VCT purges were performed until D.O. was generally <.005 ppm. D.O. was maintained at this level through power ascension and at no time did D.O. exceed .1 while >250°F.

4. Suspended Solids (S.S.) - S.S. were maintained generally < .01 ppm during startup and operation. Maximum reading obtained was 2.45 ppm following first Reactor Coolant Pump run during initial heatup after fuel reconstitution.

Secondary Systems

1. The moisture carryover test has not been performed due to small chloride problems and lack of sufficient equilibrium power levels greater than 99%.

6.20 CHEMISTRY AND RADIOCHEMISTRY TESTS AT POWER (cont)

Secondary Systems (cont)

2. Small condenser tube leaks caused the unit to vary power. Maximum concentration of chlorides in steam generators was 1 ppm. Through blowdown and isolation and repair of water boxes chlorides were quickly reduced and maintained below operating limits and generally <.05 ppm.

3. The Hotwell Cation Conductivity cells have proven to be very responsive to small tube leaks in the condenser. They have been very reliable for detecting chloride leaks and enabled us to isolate and repair the indicated water box normally before exceeding the chloride specification.

4. General chemistry on the secondary side was operated within specified parameters with few exceptions. These exceptions were generally predictable as in power transients and pump starts and were quickly reduced and returned to within specifications.

5. pH - pH additions again were not necessary to control feed or steam generator pH while maintaining a hydrazine residual of approximately 20-30 ppb.

6. S.S. - Suspended solids in the Feed system and steam generators were controlled by generator blowdown and were generally <1.0 ppm. Exceptions to this were during startup of pumps and when vacuum was broken on condenser causing increased oxygen. These periods were short in time and reduced quickly by going to full blowdown on generators.

7. Cl^-- Chlorides were generally <.05 ppm except on a few occasions when sea water leaks were detected in the condenser. Chlorides were quickly reduced by increased blowdown and isolation of faulty water box. The Hotwell Cation Conductivity cells were the mainstay of our leak detection program and functioned better than expected in keeping chloride from accumulating in the steam generators. Maximum Cl^- found in generators was ∿1 ppm, for a very short period.

Conclusions

Chemistry controls in primary and secondary systems as specified in the St. Lucie Unit 1 Chemistry Manual were implemented and found to provide adequate protection for plant system. Surveillances and test procedures gave sufficient indications of adverse trends to allow ample time to restore conditions normally prior to exceeding operational limits.

TABLE 6.20
RCS CHEMISTRY AT STEADY STATE POWER

ANALYSIS	LIMITS	1/17/77 80%	3/3/77 3/4/77 100%
pH	4.5-10.2	6.43	6.40
Conductivity	Varies	12.0	10.67
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	624	591
Lithium	.2-2.0 ppm	1.10	1.34
Diss. H_2	10-50 cc/kg	34.4	16.38
Gas Act.	--	$3.51E^{-}01$	$4.216E^{-}1$
Gross Act.	--	$3.0E^{-}1$	$8.34E^{-}1$
Crud Act.	--	$4.315E^{-}4$	$1.977E^{-}3$
Tritium	--	$7.47E^{-}2$	$1.03E^{-}1$
I Ratio	--	.517	1.085
DEQ I-131	$\leq$ 1 uCi/gm	$2.03E^{-}3$	$5.81E^{-}3$
Spectrum	--	Performed	Performed

6.21 TOTAL RADIAL PEAKING FACTOR

This measurement was taken at the 50% power level. Total radial peaking factor (F_r^T) is defined as the product of the unrodded planer peaking factor (F_r^P) and the quantity one plus the azmuthal tilt (T_q). $F_r^T = F_r^P * (1 + T_q)$. This necessitates a calculation of azmuthal power tilt. Both T_q and F_r^T are calculated by the GINCA computer incore analysis program (see section under DDPS Snapshot heading). F_r^T is also a required monthly surveillance item that is performed by the Reactor Engineering Department. See Table below for selected measurements taken during the power ascension testing program.

Table 6.21-1

TIME	DATE	% POWER	T_q **	F_r^T
1130	12/10/76	20	0.0157	1.328
0630	12/11/76	30	0.013	1.287
0230	12/17/76	42	0.010	1.301
0600	12/21/76	51	0.008	1.306
1512	01/01/77	51	0.006	1.300
1048	01/10/77	61	0.007	1.307
2200	01/15/77	80	0.004	1.310
1018	01/29/77	81	0.007	1.310
1412	02/10/77	89	0.004	1.309
0754	02/24/77	100*	0.004	1.313
0800	03/22/77	100*	0.004	1.300

* F_r^T limit is 1.36 at 100% power

** T_q limit is 0.02 per Technical Specifications

6.22 TURBINE OVERSPEED TRIP TEST

Purpose

The purpose of this test was to demonstrate that the Turbine overspeed trip mechanism would trip the turbine at a speed of 1998+0 -10 rpm and to verify that the overspeed trip weight will operate when the trip weight body is subjected to oil pressure and record this pressure.

Test Results

Holding the test handle in "Test" position, oil pressure was applied to the trip mechanism to verify it would actuate. The pressure was recorded for future use during periodic testing. Holding the test handle prevents the turbine from actually tripping while allowing an operability check of the trip mechanism. Then the turbine speed was raised until the turbine actually tripped. The speed at trip was low so adjustments were made and the trip repeated. Final overspeed trip value was 1971 rpm. This is lower than the specified value but was considered acceptable as it is conservative, will have no effects on reliable turbine operation, and the next adjustment would have been close to the upper rpm limit of 111% of design speed.

Conclusions

This procedure verified proper setting of the turbine overspeed trip. This was performed during Cycle 1 but was inadvertently omitted from the original startup report.

6.23 XENON FOLLOW

Purpose

This was not part of the scheduled power ascension testing, but rather a verification that the shape annealing factors (SAF) that were calculated and input as a gain adjustment in the linear power range subchannels during testing in Cycle 1 were still valid for Cycle 1A.

Results

The reactor was at equilibrium xenon with CEA group 7 at approximately 68 inches. CEA group 7 was borated out and the resultant axial shape index (ASI) oscillation was monitored for each power range channel. By correlating the GINCA ASI to the excore ASI, the SAF for each channel can be generated. Two methods were used to reduce the data that was generated. The first technique plotted GINCA ASI vs excore ASI and visually deriving the slope of the line (SAF) that was generated for each channel. See figure 6.23-1 for a typical geometric display. The second technique used a mathematical analysis known as least squares fit. The two techniques are compared with the reference values for each channel, in table 6.23-1.

This verification was done at the 50% power plateau and again at the 80% power plateau. The graphs of calorimetric power, CEA group 7 position, and average RPS ASI, vs time for the 50% power level and the 80% power level may be found on figures 6.23-2 and 6.23-3, respectively.

Conclusions

The reference (from fuel Cycle 1) SAF values were still valid and the gain adjustments for the linear power range subchannels were left as they were. Combustion Engineering confirmed this decision.

The oscillation for Cycle 1 began with equilibrium, all rods out (ARO). CEA group 7 was diluted to mid-core and as the swing reached a maximum value, CEA group 7 was borated to ARO. The oscillation for Cycle 1A verification began with equilibrium conditions and CEA group 7 at mid-core. GINCA ASI solution assumes oscillation beginning at ARO, hence there are some discrepancies.

TABLE 6.23-1

SHAPE ANNEALING FACTORS[a]

50% POWER			
CHANNEL[b]	REFERENCE[c]	L.S.F.[d]	GEOMETRIC
A	3.46	3.282	3.385
B	3.77	3.597	3.678
C	3.92	3.588	3.676
D	3.07	2.918	2.946
9	3.21	3.186	-
10	3.68	3.448	-

80% POWER			
A	3.46	3.25	3.31
B	3.77	3.53	3.62
C	3.92	3.48	3.56
D	3.07	2.81	2.95
9	3.21	3.09	3.14
10	3.68	3.32	3.54

a SAF = (Ginca Axial Shape Index) ÷ (Excore Axial Shape Index)
b Reactor Protective Channel or Nuclear Instrument Control Channel
c Cycle 1 Forced Xenon Oscillation (Ref. Dec. '76 Startup Report, Sec. 6.11)
d Least Squares Fit

FIGURE 6.23-1

GEOMETRIC FIT FOR SAF
(CHANNEL A, 50% POWER)

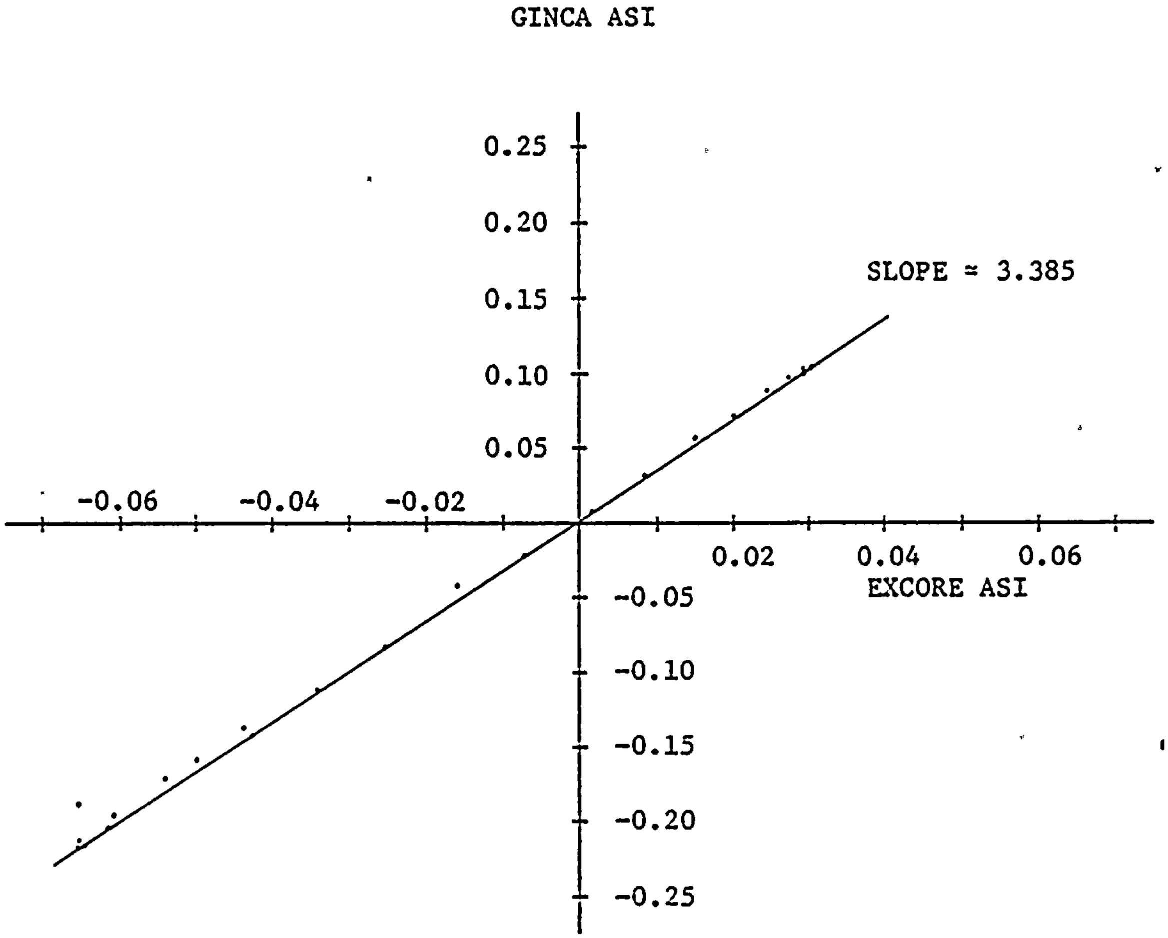


FIGURE 6.23-2

50% XENON FOLLOW

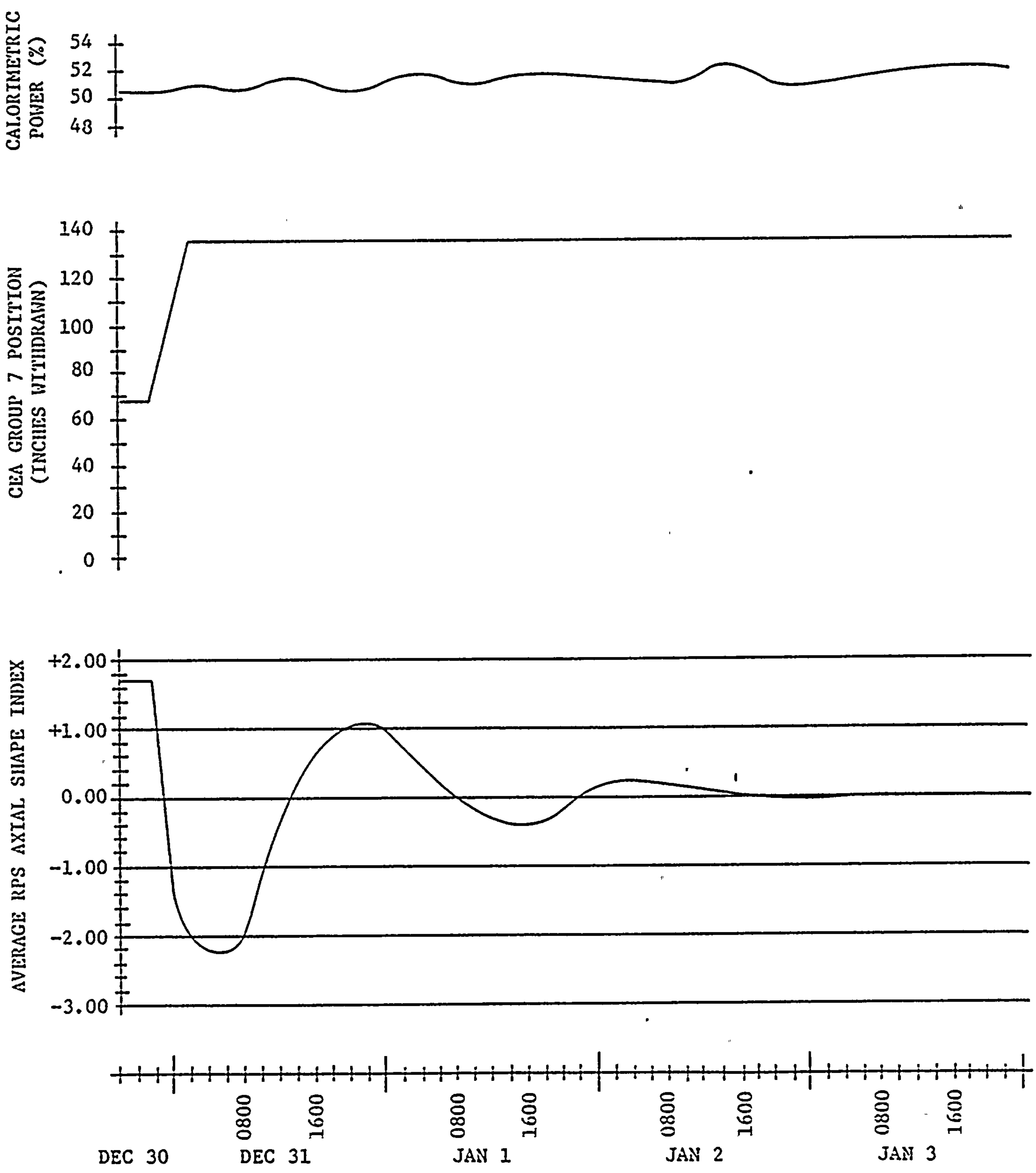
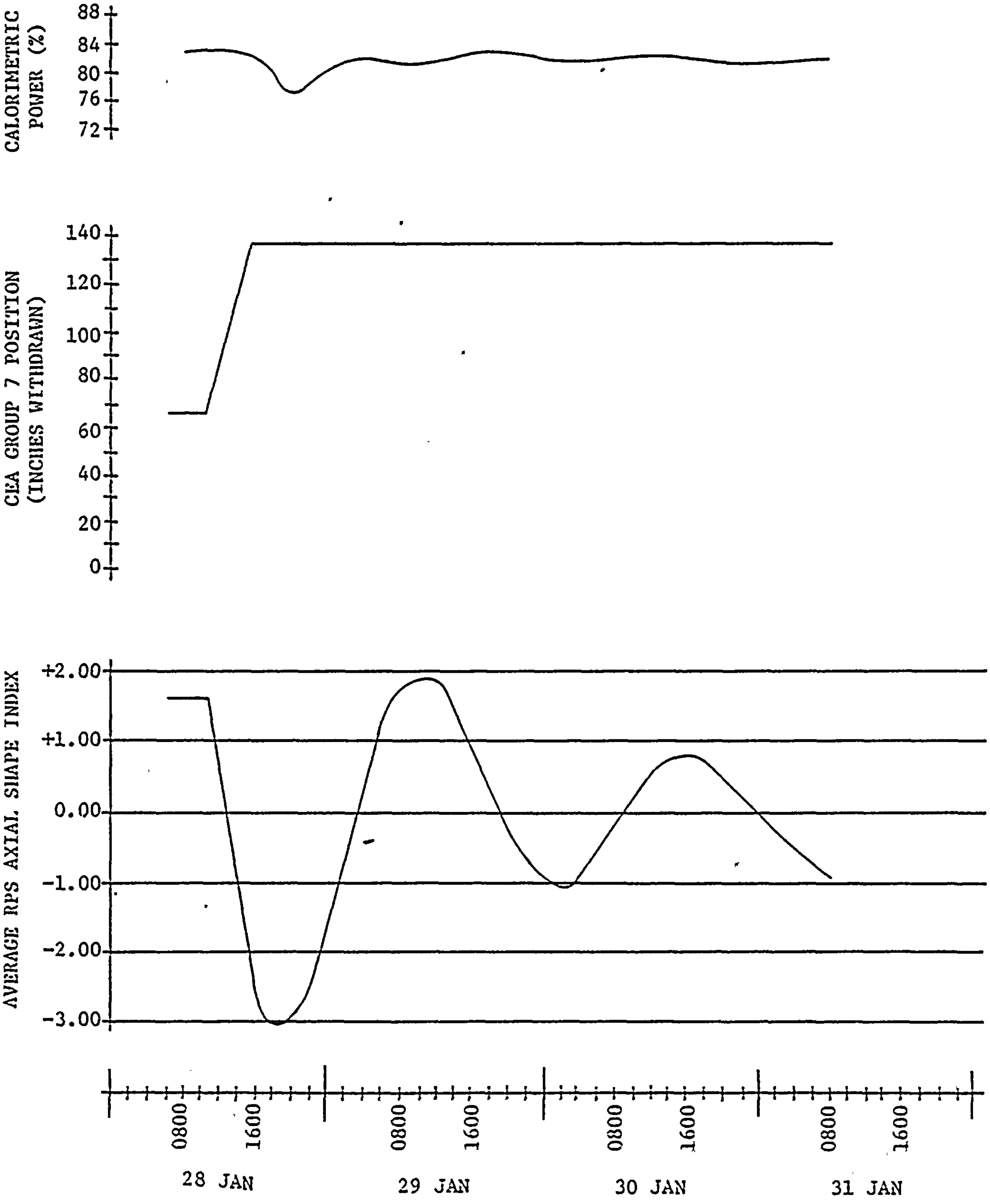


FIGURE 6.23-3

80% XENON FOLLOW



6.24 EFFLUENT MONITOR CORRELATION

Purpose

The purpose of this evolution was to compare the indicated count rate on the various monitors to actual effluent activity. This provided a check of monitor linearity and provided data for activity versus count rate curves for in-plant use.

Results

The following monitors were included in the correlation:

- Liquid Radioactive Waste
- Gaseous Radioactive Waste
- Plant Vent - Iodine, Particulate, Gaseous
- Air Ejector Vent
- Steam Generator 1A and 1B Blowdown

All monitors were checked at least twice--one of these checks was during nominal 100% power operation.

The first three listed above were done using effluent streams from which samples were taken for detailed analysis. Since we have no steam generator tube leaks there was no activity in the effluent streams for the last two. For these, representative samples were made up, traceable to the National Bureau of Standards. This allowed generation of curves comparing monitors count rate to actual activity from representative effluent streams. In all cases, samples and standards used were counted on a gamma analyzer calibrated to NBS standards using appropriate geometries.

Conclusions

All monitor readings were acceptably linear throughout the range of the instruments. Meaningful curves comparing monitor count rate to actual activity were drawn for each monitor for in-plant use.

6.25 DDPS CALORIMETRIC AND DDPS SNAPSHOT

At the beginning and end of every major power level, a digital data processor system (DDPS) calorimetric and snapshot was performed.

The reactor power, in million British thermal units (MBTU) was the main item of interest on the calorimetric printout. The reactor power in MBTU can be divided by 87.3728 to get reactor power in per cent of rated thermal power.

The snapshot is a paper tape punch out containing time, date, CEA positions, signal readings from each of the 45 vanadium incore neutron detectors and signal readings from each of the four levels of the rhodium incore neutron detectors. The snapshot, normally, was read via an on-site terminet to Combustion Engineering's computer in Connecticut, where a core performance analysis, GINCA, would be performed.

The GINCA result can be transmitted to FP&L Power Resources in Miami, in a "long form" format. The GINCA result can also be put on micro-fiche and the film mailed to PSL Reactor Engineering Department, in the "long form" format. Furthermore, the GINCA result, in a "short form" format can be printed out at the on-site terminet.

The table on the next page identifies the make-up of the GINCA short form that is returned to the terminet.

TABLE 6.25-1

INFORMATION ON GINCA SHORT FORM

Reactor Power
RCS Inlet and Outlet Temperatures
Inoperable Incore Neutron Detectors
Core Axial Shape Index
Core Average Axial Shape Index
Core Average Axial Peak Location
Core 3-D Power Peak
Core Average Burnup
Batch Burnup
Azmuthal Tilt Amplitude: Excores
Azmuthal Tilt Angle: Excores
Azmuthal Tilt Amplitude: Incores
Azmuthal Tilt Angle: Incores
Alarm Setpoint (level vs value)
Total Radial Peaking Factor
Integrated Radial Peaking Factor

Core Maps with Max. and Min. Values
Relative Power Density
Linear Heat Rate
F_z
Exposure (MWD/MTU)

7.0 COMMENTS ON ORIGINAL STARTUP REPORT

During our preparation of this followup Startup Report, we noted a few items in the initial report that were not completely discussed or were inadvertently omitted. Below is a disccusion of these items.

Section 3.1.2, Test Results, CEDM/CEA Performance Tests has a list of the tests performed. That list should include:

(7) Manual scram tests.
This verified that the CEDM/CEA system responded properly to a manual scram and it was completed satisfactorily. These manual scram tests were also performed as part of the initial Approach to Criticality procedure but were not reported in that section either.

Section 3.10 RCS Heat Loss. This section did not discuss the predicted value for heat loss and did not compare actual data to the predicted. The actual total value obtained was 2.84 Mw. The expected value was on the order of 3 Mw.

Section 6.9, Reactivity Coefficient Measurements compared the actual results to the Technical Specification or vendor numbers, but FSAR values were not given.

For MTC, the FSAR Accident Analysis assumes values from $^{-4}$ $+.5$ to -1.4×10^{4} $\Delta K/K/^{o}F$. The measured value is $-.1 \times 10^{-4}$ $\Delta K/K/^{o}F$ which is within the band. For Power Coefficient the FSAR gives -1.6×10^{-3} $\Delta K/K/Kw/ft$ which corresponds to $-.96 \times 10^{-4}$ $\Delta K/K/\%$. The measured value was -1.07×10^{-4} $\Delta K/K/\%$ and the vendor design value is $(-1.0 \pm .1) \times 10^{-4}$ $\Delta K/K/\%$.

Section 6.13, Core Power Distributions, stated that the radial and axial peaks measured (1.3414 and 1.336 respectively at 50% power) were acceptable. The accident analysis assumes values of 1.7 and 1.44 respectively.

Our measured values are conservative with respect to the FSAR values and, as originally stated in 6.13, are satisfactory. The Technical Specification limit for radial peaking is ≤ 1.36 which is above the measured value.

7.1 RCS FLOW COASTDOWN

In the original Startup Report, the flow coastdown results were given. In that report, they were not specifically compared to the FSAR flow coastdown curve, Figure 15.2.5-1. The attached figure has a reproduction of the FSAR curve (1) for total loss of flow and a plot of the actual test results (3). In addition, curve (2) is CE's revised coastdown curve, using as-built data, which was used in their final safety and setpoint analysis.

It should be noted that the test results are conservative with respect to both analytical curves and the acceptance criteria was met.

FLOW COASTDOWN