BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT

UNIT 1

DOCKET NO. 50-317

LICENSE NO. DPR-53

STARTUP TEST REPORT

AUGUST 29, 1975

1

86

B A L T I M O R E G A S A N D E L E C T R I C C O M P A N Y

C A L V E R T C L I F F S N U C L E A R P O W E R P L A N T

U N I T I

Docket No. 50-317

License No. DPR-53

S T A R T U P T E S T R E P O R T

August 29, 1975

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

This report fulfills the requirement of Technical Specification 6.12.1 which states that a Startup Test Report will be submitted to NRC within 90 days of completion of the Startup Test Program. The test program was completed on 5/30/75.

The Startup Test Program was organized and administered by Baltimore Gas and Electric Company (BG&E) 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. The test results from each phase were reviewed by a Test Data Evaluation Group (TDEG) consisting of the BG&E Startup Test Coordinator and the CE Chief Test Engineer and others as required. Test results falling outside of acceptance criteria received an additional review by the Plant Operations and Safety Review Committee (POSRC) and were resolved prior to beginning the next test phase. The test phases were as follows:

(1) Initial Fuel Load

(2) Post Core Hot Functional Tests

(3) Initial Approach to Criticality

(4) Low Power Physics Tests

(5) Escalation to Power Tests - 20% Plateau

(6) Escalation to Power Tests - 50% Plateau

(7) Escalation to Power Tests - 80% Plateau

(8) Escalation to Power Tests - 100% Plateau

Maximum licensed reactor core power level (100%) is 2560 MWth. The Startup Test Program began at 2100 on 8/4/74 with the loading of the first fuel assembly into the reactor vessel and was completed at 1600 on 5/30/75.

The test program was rather extensive and the results will form the basis and justification for significantly reducing the required testing on Calvert Cliffs Unit 2. The design of most safety related systems on Unit 2 are very similar and generally identical to Unit 1.

1.2 Summary

1.2.1 Initial Fuel Load

Fuel loading commenced on 8/4/74 and was completed on 8/16/74. Approximately 25% (see Section 1.2.6) of this time was spent in the non-productive activities of fuel handling equipment maintenance, hand manipulation of fuel assemblies, and inspection of fuel assemblies which gave momentary indications of hanging up on adjacent fuel assemblies while being lowered into the core. None of the inspected fuel assemblies showed visible signs of damage. An evaluation by CE, Windsor, of the mechanical forces involved in the several incidents of fuel assembly hangup indicated that observed forces were below those required to cause damage. Fuel handlers worked three (3) eight (8) hour shifts per day and an experienced crew could load twelve (12) to fourteen (14) fuel assemblies per shift providing there were no maintenance problems.

1.2.2 Post Core Hot Functional Tests

Post Core Hot Functional (PCHF) Tests commenced 9/3/74 and were completed on 10/5/74. In addition to those tests required by the FSAR, performance testing of the turbine bypass valves, auxiliary feedwater pumps, and secondary safety valves was also performed during this phase. All test results met acceptance criteria with the exception of Reactor Coolant System (RCS) flow rate for the Reactor Coolant Pump (RCP) 11A and 11B combination and the performance of Control Element Drive Mechanism (CEDM) 15. RCP 11A and 11B combination yielded a flow rate slightly below that required by present Technical Specifications. That combination is presently administratively restricted from use during power operation. CEDM 15 is a Part Length Control Element Assembly (CEA) in Part Length Group 2 (PLR 2). Performance testing revealed a defective upper gripper coil and/or latch in the CEDM. PLR 2 is presently fully withdrawn and administratively restricted from use while the reactor is critical.

1.2.3 Initial Approach to Criticality

The Initial Approach to Criticality commenced at 2130 on 10/5/74 and the reactor was declared critical approximately forty-eight (48) hours later. The approach was relatively uneventful with the exception of some minor CEA Group interlock malfunctions which occurred during the CEA withdrawal sequence. A slow RCS dilution followed CEA withdrawal. Measured RCS soluble boron concentration at the time of criticality was in good agreement with that which was predicted and well within the acceptance criteria.

1.2.4 Low Power Physics Tests

The Low Power Physics Test (LPPT) Phase commenced on 10/7/74 and was finally completed on 12/27/74. This phase was marred by two (2) major incidents which required approximately eight (8) weeks of non-test time to resolve. The incidents are discussed in more detail in Section 7.0. All LPPT measurements were in good agreement with predictions and well within acceptance limits.

1.2.5 Escalation to Power Tests

The Escalation to Power Test (EPT) Phase began 12/27/74. All required test measurements were performed and results reviewed and evaluated on-site by the TDEG with assistance as necessary from CE, Windsor with the following exceptions:

(1) A test of axial xenon oscillation dampening using Part Length CEA's (PLR) was deferred until such time as a decision is made to use PLR's. Later in first cycle life, an axial xenon oscillation dampening test will be performed using full length CEA's.

(2) A part loop operation power distribution measurement was deferred until such time as a decision is made to operate at power using less than four (4) Reactor Coolant Pumps (RCP). Administrative restrictions prevent power operation with less than full RCS flow (4 RCP's). In addition, the Reactor Protective System (RPS) will cause a reactor trip whenever reactor power is greater than 10^{-4}% and less than 4 RCP's are operating.

(3) A performance test of the automatic CEA control features of the Reactor Regulating System (RRS) has been deferred until such time as a decision to use this feature is made. The principal reason for deferring the test was to decrease the possibility of fuel failures which could be exacerbated by relatively rapid and near continuous CEA motion, a characteristic of automatic CEA control.

Administrative restrictions prevent part loop operation and use of automatic CEA control and PLR's until such time as tests are performed and results reviewed and approved by the POSRC.

The off-site analysis by CE, Windsor of two (2) tests, Psuedo Ejected CEA and Dropped CEA continues. A preliminary on-site analysis of the results of the Psuedo Ejected CEA test by the TDEG indicated that those test results were well within the power peaking acceptance criteria and on that basis, full power operation was continued. However, the acceptance criteria also require completion of a more detailed analysis by CE, Windsor. The evaluation of the dropped CEA test results by the TDEG was inconclusive. In addition, acceptance criteria require the completion of analysis by CE, Windsor. Should a dropped CEA occur during power operation, plant operating procedures specify that the reactor will be manually tripped if at greater than 50% power. The off-site analysis will determine the power level to which the turbine generator should runback in response to a dropped CEA. On that basis, the requirement to manually trip the reactor may be removed from the plant operating procedures.

Results of the Shielding Survey indicated higher than expected radiation levels in the containment. Investigation and evaluation revealed that the primary source was gamma and neutrons streaming out of the annulus between the reactor vessel and the primary shield wall. Temporary shielding has reduced dose rates to acceptable levels (see Section 7.0 for more detail). Planning and engineering for a permanent shield design continues.

Following an unplanned trip from 100% power, the feedwater rings for both steam generators were uncovered, as is the usual case. While in the process of regaining normal steam generator water levels, a water hammer was experienced in the main feedwater lines. Some damage to pipe supports and to motor operated stop valves resulted. Investigation and evaluation indicated that filling at a slower rate when level was in the vicinity of the feedwater ring precluded water hammer. Feedwater ring design changes are also being considered. See Section 7.0 for more detail.

These and other minor equipment problems resulted in forced outages which delayed completion of EPT by about twelve (12) weeks. Figure 1.2-1 presents an as originally planned and as experienced power history for EPT. EPT and the Startup Test Program were concluded on 5/30/75. Unit 1 had been previously declared in commercial operation on 5/8/75.

1.2.6 Evaluation of Startup Test Program Critical Path Hours

As a guide to future planning, particularly in connection with the Initial Startup of Unit 2, a detailed evaluation was made of the as experienced Unit 1 Startup Test Program critical path. The results of that analysis are summarized in Table 1.2-1. In order to assist the reader in an evaluation of the significance of the various numbers in Table 1.2-1, the following explanatory remarks are made.

Each chronological hour from 2100 on 8/4/74 to 1600 on 5/30/75 was investigated to determine which plant activity was controlling the critical path for completion of the Startup Test Program. Control room logs and the test organization's records were the primary sources of information and the evaluation was performed in increments as the test program evolved.

Each critical path hour was assigned to one of four (4) activities: startup testing; operations; primary system maintenance; or secondary system maintenance. The startup test and operations categories were further divided into productive and non-productive categories. Productive hours would be those which were consumed in the process of completing scheduled test program evolutions. Examples of non-productive startup test hours would be those spent: repeating test measurements previously determined to be invalid; repeating test measurements not fully developed before being interrupted by a forced maintenance activity; and repeating test measurements not properly performed originally. Examples of non-productive operations hours would be those required: to put the plant in the proper condition for conduct of a forced maintenance activity; to return the plant to the last previous test condition after a forced maintenance activity; and operator error resulting in loss of proper test conditions. All maintenance hours were considered to be non-productive with the exception of post-fuel loading reactor vessel reassembly. In some cases several activities were jointly responsible for a critical path hour and it was divided among them.

The original estimates of scheduled critical path hours are provided for comparison purposes. Those estimates included some allowance for minor forced maintenance. When compared with the actual productive hours, including the non-productive, the estimates were much too optimistic in the case of most every test phase.

TABLE 1.2-1

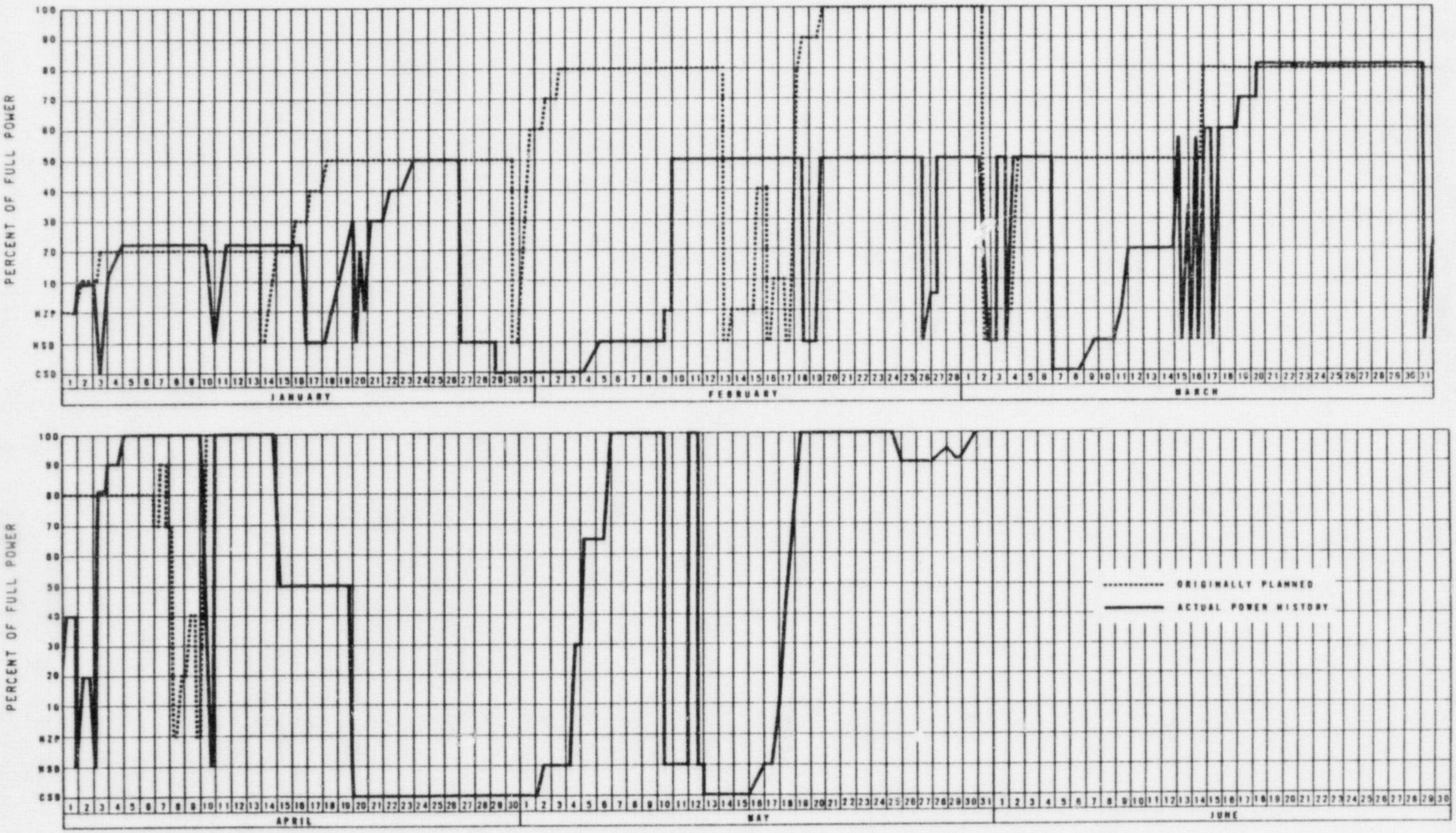
START-UP TEST PROGRAM SUMMARY OF CRITICAL PATH HOURS

DATE	TEST PHASE	AVERAGE POWER LEVEL	CRITICAL PATH HOURS: START-UP TEST PROD.	START-UP TEST NON-PROD.	OPERATIONS PROD.	OPERATIONS NON-PROD.	MAINT PRI.	MAINT SEC.	TOTAL PROD.	TOTAL NON-PROD.	TOTAL TOTAL	ORIG. EST.	PROD.(3) RV REASS.
8/4-16/74	Fuel Load	RSD (1)	21.5	20.25	108.5	80.5	46.25	-	130	147	277	168	-
8/16-9/3/74	RV Reass.	RSD (1)	-	-	24	-	410 (3)	-	434	-	434	336	-
9/3-10/5/74	PCHF	HSD (2)	310	19.5	70	138.5	217.5	23	380	398.5	777.5	336	-
10/5-7/74	IAC	0	41.5	-	-	-	8	-	41.5	8	49.5	40	-
10/7-12/27/74	LPPT	0	269	108	-	424	1089	43	269	1664	1993	336	-
12/27/74-1/17/75	EPT 20%	11.9	171.5	66	-	109	49	114.5	171.5	338.5	510	312	-
1/17-3/14/75	EPT 50%	26.4	364	315	-	72	34.5	585.5	364	970	1334	408	-
3/14-4/3/75	EPT 80%	59.6	333.5	7	58	45.5	-	47	391.5	99.5	491	432	-
4/3-5/30/75	EPT 100%	51.5	449	22	170	291	99	334	619	746	1365	600	-
8/4/74-5/30/75	TOTAL	37.8 (4)	1960	558	431	1161	1543	1110	2801	4370	7171	2968	410
	PERCENT		27	8	6	16	22	15	39	61	100		6
			35		22		37				100		6

(1) RSD ≡ Refueling Shutdown
(2) HSD ≡ Hot Shutdown
(3) Reactor Vessel Reassembly is figured into the total as productive critical path hours.
(4) During EPT

CALVERT CLIFFS UNIT 1

ORIGINALLY PLANNED & ACTUAL POWER HISTORY FOR EPT



HSP HOT ZERO POWER
HSD HOT SHUT DOWN
CSD COLD SHUT DOWN

FIGURE 1.2-1

2.0 INITIAL FUEL LOAD

Fuel loading commenced on 8/4/74 and was completed on 8/16/74. Table 2.0-1 and Figure 2.0-1 give the fuel loading sequence for initial core load. Figures 2.0-2 and 2.0-3 show fuel assembly core location by serial number and CEA core location by serial number respectively.

Neutron count rate was monitored during core assembly on four separate detector channels. In addition to Wide Range Log Channels 1 and 3, two temporary neutron detectors were used. The temporary neutron detectors were installed prior to core load with detector A installed in core location V-7 and detector B installed in core location V-15. In Step 140 B of the loading sequence detector B was shifted to core location B-15. Initial locations and movement step numbers are shown in Figure 2.0-4. Independent plots of inverse neutron count rate versus the number of fuel assemblies loaded were maintained to ensure that the reactor remained subcritical at all times.

The fuel loading was conducted with the refueling and spent fuel pools dry and borated water in the reactor vessel to within approximately eight (8) inches of the top of the reactor vessel outlet nozzles. This dry loading allowed access to refueling equipment for maintenance, improved visibility of fuel assemblies during manipulation and allowed access to the reactor vessel flange during repositioning of temporary neutron detectors.

Several problems with the fuel handling equipment occurred during fuel load. A description of these problems and their resolution are given below:

Fuel Transfer Machine and Upender

(1) On several occasions lights and interlocks on the transfer machine carriage and upender would not function properly. Filling with water and venting the limit switches restored the interlocks and lights to proper operation.

(2) An open phase on the fuel transfer machine brakes and two faulty relays caused two separate delays. Replacement of the parts restored the transfer machine to service.

Spent Fuel Handling Machine

The spent fuel handling machine festooned cable hung up. A shim was placed in a cable hanger to eliminate drag.

Refueling Machine

(1) On several occasions the hoist on the refueling machine hung up. The problem was eventually traced to a bushing binding up. Initially the bushing was cleaned and extra lubricant added to minimize the problem. Two days later, the drive shaft and a bushing of different material were installed and fuel loading continued.

(2) The hoist position indication malfunctioned. The hoist position read out device was replaced to correct the problem.

(3) The switch controlling the refueling machine bridge would only move the bridge in the reverse direction about half the time. Freeing up the jammed finger switch and lubricating the gears returned the machine to normal operation.

Fuel Insertion Problems

Fuel insertion problems were experienced with fourteen (14) fuel assemblies resulting in approximately 14 hours of non-productive time. The insertion problems were generally an interaction between grids of the fuel assembly being inserted and grids of adjacent fuel assemblies. These problems were resolved as follows:

(1) An observer was stationed at the upender to ensure that the fuel assembly orientation was parallel to the sides of the upender as it was lifted by the refueling machine.

(2) After several assemblies were removed and inspected and no damage was found, adequate insertion clearance was obtained by making small changes to the refueling machine position index, pushing on the refueling machine spreader or pushing on the upper end fittings of adjacent fuel assemblies with a long handled pole as required.

Modifications to fuel handling equipment are being made to minimize these problems in the future.

Except for the maintenance problems with the refueling equipment, the fuel loading was conducted in an efficient manner.

TABLE 2.0-1

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
1	1B015	(Neutron Source)	X - 11
2	1C035	-	Y - 10
3	1C040	-	Y - 12
4	1C206	0J	X - 13
5	1B067	-	W - 13
6	1A017	3A	W - 11
7	1B019	-	W - 9
8	1C203	0F	X - 9
9	1C003	-	Y - 8
10	1C201	-	X - 7
11	1A061	2D	W - 7
12	1A014	02	V - 9
13	1B028	-	V - 11
14	1A026	15	V - 13
15	1A031	0K	W - 15
16	1C210	-	X - 15
17	1C015	-	Y - 14
18	1C030	23	X - 16
19	1B063	-	W - 16
20	1A058	0G	V - 16
21	1B025	-	T - 16
22	1A040	12	T - 15
23	1B010	-	T - 13
24	1A055	40	T - 11
25	1B023	-	T - 9
26	1A027	05	T - 7

TABLE 2.0-1 (Cont'd)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
27	1B006	-	T - 16
28	1A025	1E	V - 6
29	1B049	-	W - 6
30	1C020	10	X - 6
31	1C011	-	X - 5
32	1C102	2C	W - 5
33	1B071	-	V - 5
34	1A063	-	T - 5
35	1B003	-	S - 5
36	1A052	43	S - 6
37	1B002	-	S - 7
38	1A050	-	S - 9
39	1B038	-	S - 11
40	1A049	-	S - 13
41	1BT03	Surveillance Holder Center Guide Tube	S - 15
42	1A033	45	S - 16
43	1B065	-	S - 17
44	1A019	-	T - 17
45	1B034	-	V - 17
46	1C103	14	W - 17
47	1C009	-	X - 17
48	1A028	OE	R - 17
49	1B058	-	R - 16
50	1A044	29	R - 15
51	1B008	-	R - 13
52	1A048	24	R - 11

TABLE 2.0-1 (Cont'd)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
53	1B057	-	R - 9
54	1A046	2F	R - 7
55	1B001	-	R - 6
56	1A069	0C	R - 5
57	1B037	-	N - 5
58	1A009	-	N - 6
59	1B030	-	N - 7
60	1A064	0D	N - 9
61	1B061	-	N - 11
62	1A045	1B	N - 13
63	1B033	-	N - 15
64	1A062	-	N - 16
65	1B068	-	N - 17
66	1A054	41	L - 17
67	1B029	-	L - 16
68	1A047	18	L - 15
69	1B036	-	L - 13
70	1A016	1C	L - 11
71	1B077	-	L - 9
72	1A043	1A	L - 7
73	1B009	-	L - 6
74	1A056	3J	L - 5
75	1B066	-	J - 5
76	1A065	-	J - 6
77	1B032	-	J - 7
78	1A003	1F	J - 9

TABLE 2.0-1 (Cont'd)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
79	1B018	-	J - 11
80	1A042	39	J - 13
81	1B004	-	J - 15
82	1A038	-	J - 16
83	1B064	-	J - 17
84	1A066	2J	G - 17
85	1B044	-	G - 16
86	1A001	2A	G - 15
87	1B059	-	G - 13
88	1A002	09	G - 11
89	1B055	-	G - 9
90	1A015	37	G - 7
91	1BT01	Surveillance Holder Center Guide Tube	G - 6
92	1A057	2B	G - 5
93	1B011	-	F - 5
94	1A068	42	F - 6
95	1B024	-	F - 7
96	1A004	-	F - 9
97	1B045	-	F - 11
98	1A051	-	F - 13
99	1BT02	Surveillance Holder Center Guide Tube	F - 15
100	1A053	44	F - 16
101	1B051	-	F - 17
102	1A060	-	E - 17
103	1B040	-	E - 16

TABLE 2.0-1 (Cont'd)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
104	1A029	01	E - 15
105	1B026	-	E - 13
106	1A005	3K	E - 11
107	1B012	-	E - 9
108	1A039	21	E - 7
109	1B007	-	E - 6
110	1A010	-	E - 5
111	1B052	-	D - 5
112	1A024	34	D - 6
113	1B046	-	D - 7
114	1A021	36	D - 9
115	1B035	-	D - 11
116	1A059	2E	D - 13
117	1A067	0H	D - 16
118	1B070	-	D - 17
119	1C105	2G	C - 17
120	1B031	-	C - 16
121	1A030	0A	C - 15
122	1B048	-	C - 13
123	1A036	3B	C - 11
124	1B043	-	C - 9
125	1A008	33	C - 7
126	1B016	-	C - 6
127	1C111	2H	C - 5
128	1C032	-	B - 5
129	1C028	25	B - 6
130	1C211	-	B - 7

TABLE 2.0-1 (Cont'd)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
131	1C213	1G	B - 9
132	1B020	Neutron Source	B - 11
133	1C209	38	B - 13
134	1C205	-	B - 15
135	1C034	03	B - 16
136	1C036	-	B - 17
137	1C038	-	A - 14
138	1C014	-	A - 12
139	1C002	-	A - 10
140	1C006	-	A - 8
140B	Detector B		D - 15
141	1C060	-	V - 15
142	1C024	-	W - 4
143	1C104	3C	V - 4
144	1B047	-	T - 4
145	1A013	1D	S - 4
146	1B042	-	R - 4
147	1A023	08	N - 4
148	1B050	-	L - 4
149	1A041	35	J - 4
150	1B075	-	G - 4
151	1A012	11	F - 4
152	1B079	-	E - 4
153	1C106	3D	D - 4
154	1C017	-	C - 4
155	1C022	-	D - 3

TABLE 2.0-1 (Cont'd)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
156	1C107	1K	E - 3
157	1B013	-	F - 3
158	1A007	22	G - 3
159	1B078	-	J - 3
160	1A018	3E	L - 3
161	1B080	-	N - 3
162	1A034	1E	R - 3
163	1B072	-	S - 3
164	1C101	16	T - 3
165	1C021	-	V - 3
166	1C018	-	T - 2
167	1C023	26	S - 2
168	1C204	-	R - 2
169	1C215	17	N - 2
170	1B022	-	L - 2
171	1C208	19	J - 2
172	1C202	-	G - 2
173	1C027	31	F - 2
174	1C033	-	E - 2
175	1C007	-	H - 1
176	1C001	-	K - 1
177	1C013	-	M - 1
178	1C039	-	P - 1
179	1C029	-	W - 18
180	1C109	3F	V - 18
181	1B069	-	T - 18
182	1A035	27	S - 18

TABLE 2.0-1 (Cont'd)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
183	1B039	-	R - 18
184	1A020	04	N - 18
185	1B056	-	L - 18
186	1A037	1J	J - 18
187	1B027	-	G - 18
188	1A022	20	F - 18
189	1B017	-	E - 18
190	1C110	3G	D - 18
191	1C026	-	C - 18
192	1C016	-	D - 19
193	1C108	2K	E - 19
194	1B074	-	F - 19
195	1A011	30	G - 19
196	1B062	-	J - 19
197	1A006	3H	L - 19
198	1B005	-	N - 19
199	1A032	28	R - 19
200	1B053	-	S - 19
201	1C112	13	T - 19
202	1C025	-	V - 19
203	1C012	-	T - 20
204	1C005	32	S - 20
205	1C212	-	R - 20
206	1C207	07	N - 20
207	1B021	-	L - 20
208	1C214	0B	J - 20

TABLE 2.0-1 (Cont'd)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
209	1C216	-	G - 20
210	1C031	06	F - 20
211	1C019	-	E - 20
212	1C008	-	H - 21
213	1C037	-	K - 21
214	1C004	-	M - 21
215	1C010	-	P - 21
215B	Temporary Detector B	-	Remove from Core
216	1B041	-	D - 15
216A	Temporary Detector A	-	Remove from Core
217	1B014	-	V - 7

CALVERT CLIFFS UNIT 1
CORE LOADING SEQUENCE
BOL, 1st CYCLE

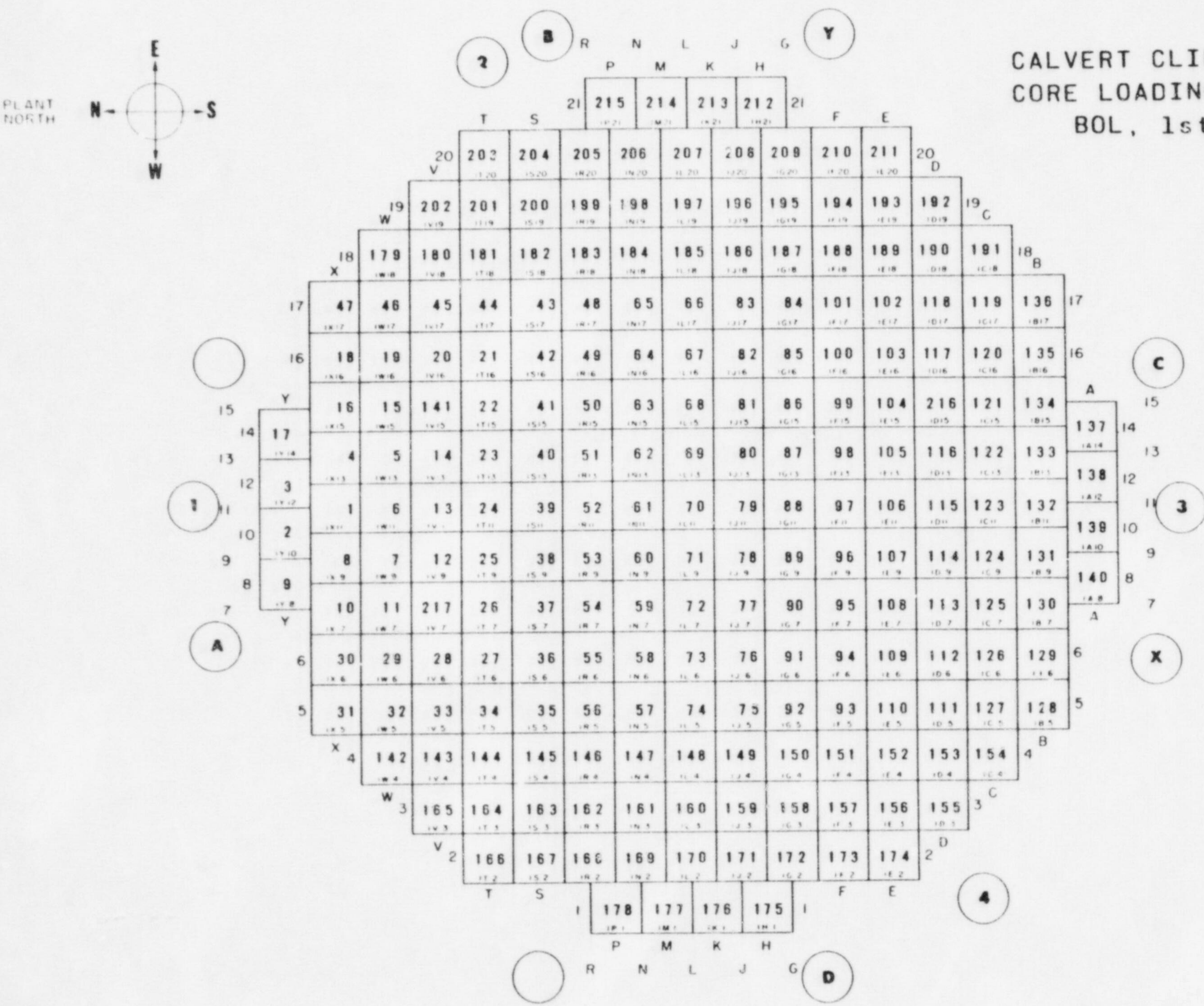


FIGURE 2.0-1

CALVERT CLIFFS UNIT 1
FUEL ASSEMBLY SERIAL NUMBERS AND CORE LOCATIONS
BOL, 1st CYCLE

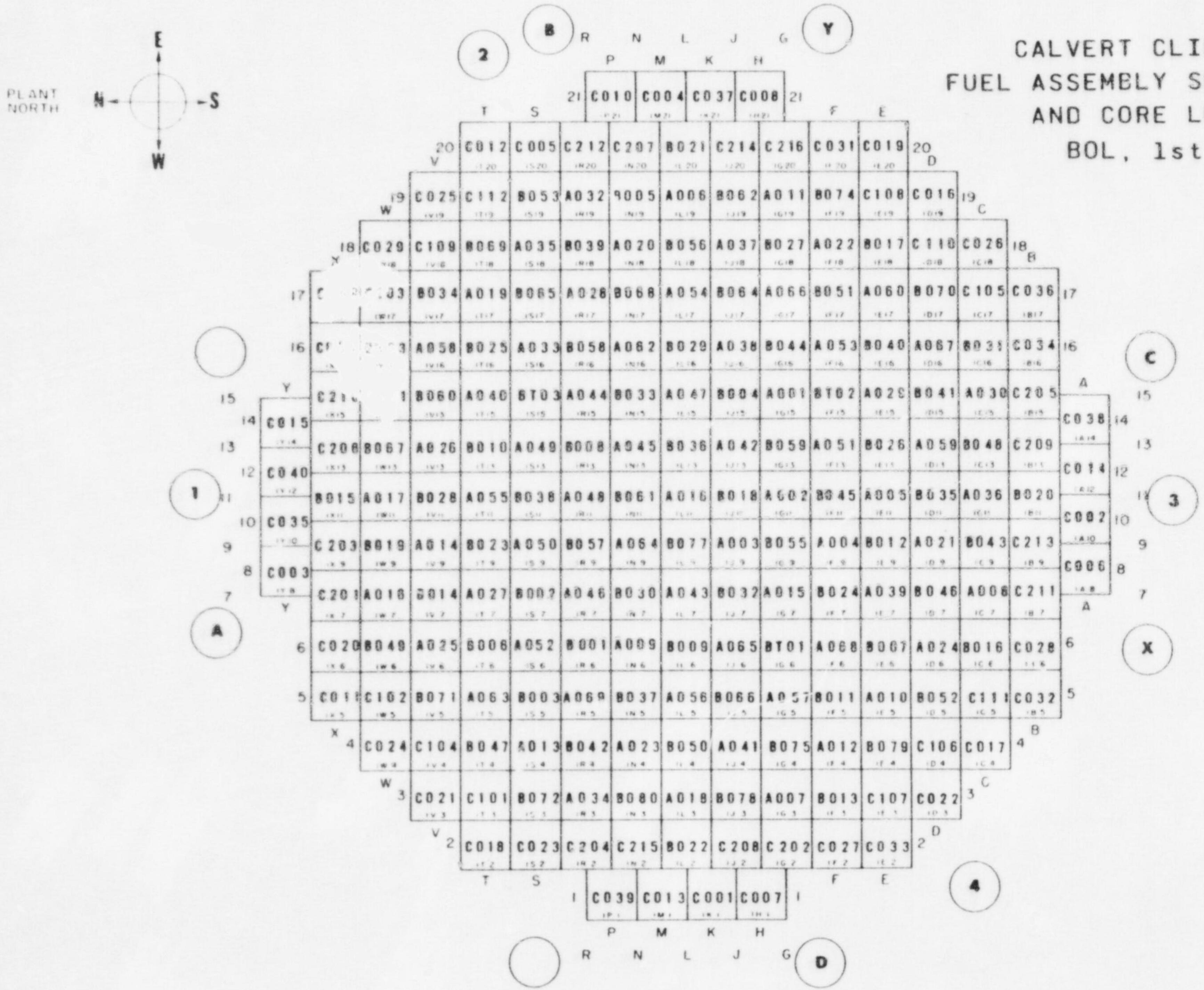


FIGURE 2.0-2

CALVERT CLIFFS UNIT 1
CONTROL ELEMENT ASSEMBLY SERIAL NUMBERS AND CORE LOCATIONS
BOL, 1st CYCLE

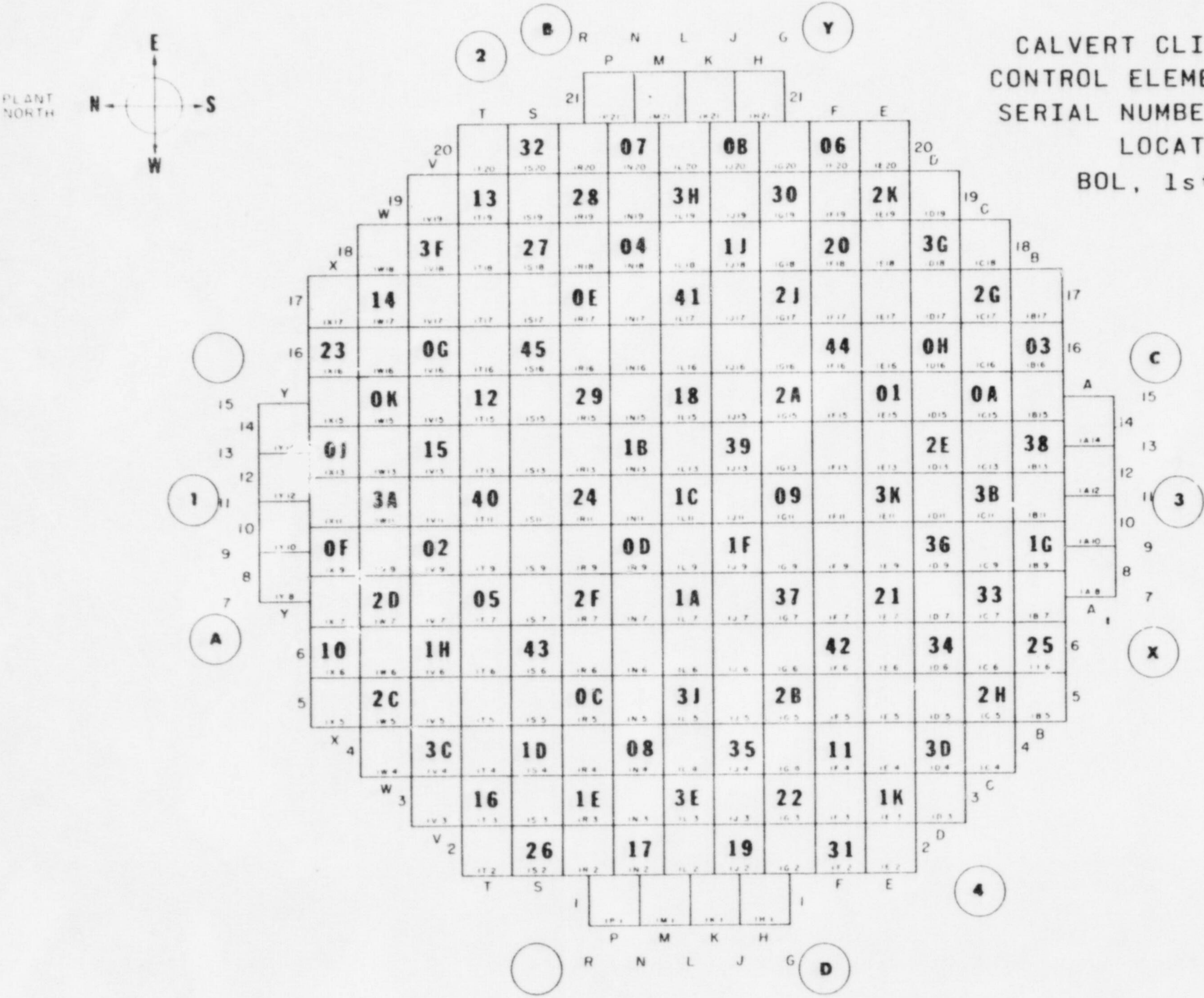


FIGURE 2.0-3

CALVERT CLIFFS UNIT 1
TEMPORARY FUEL LOADING
DETECTOR LOCATION
AND MOVEMENT STEP NUMBER
BOL, 1st CYCLE

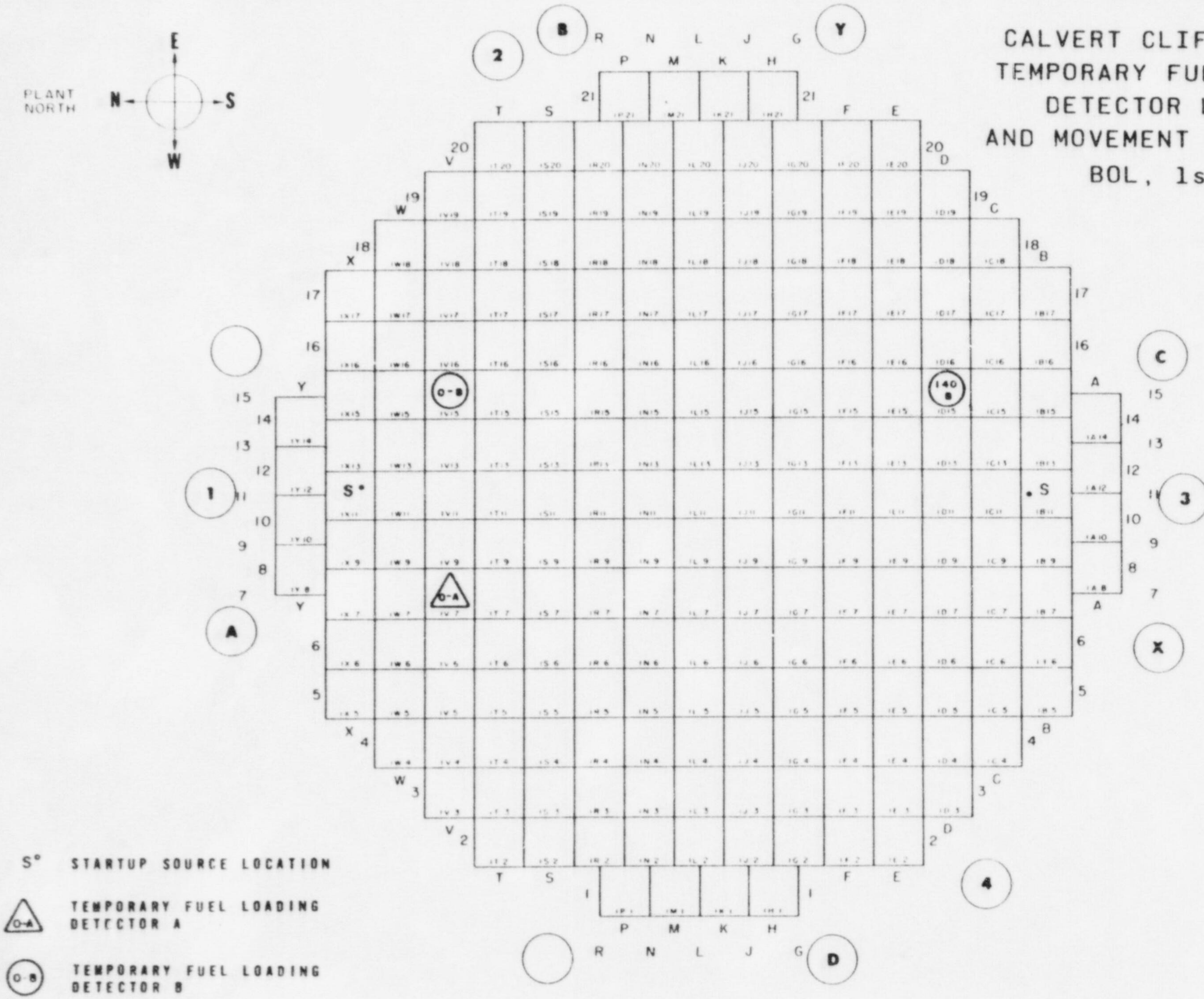


FIGURE 2.0-4

3.0 POST CORE HOT FUNCTIONAL TESTS (PCHF)

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

(1) Mechanical and instrument tests on Control Element Drive Mechanisms (CEDM) and Control Element Assembly (CEA) position indicators;

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

(3) Rod drop time measurements (cold and hot) at rated Reactor Coolant System (RCS) flow and with no RCS flow;

(4) Final leak tests of RCS;

(5) Chemistry and radiochemistry tests for water quality;

(6) RCS flow determination tests;

(7) Pressurizer effectiveness tests; and

(8) Neutron response check of source range monitors.

All of the above tests except items (2) and (8) are described in Sections 3.1 through 3.6. Item (2), Reactor protective trip circuit and manual scram tests, was performed as an initial condition to the Initial Approach to Criticality (IAC). Item (8), Neutron response check of source range monitors, was verified as an initial condition to the IAC Procedure by comparing as observed signal levels with those observed prior to fuel loading, after fuel loading, and during PCHF and noting no significant difference other than an expected increase due to RCS heatup during PCHF.

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

(1) Turbine Bypass Valves test;

(2) Checking the Turbine Generator roll to maximum RPM;

(3) Taking measurements of RCS expansion;

(4) RCS and Steam Generator instrument calibration checks;

(5) Auxiliary Feedwater System test;

(6) RCS heat loss test;

(7) In-Core Detector resistance readings; and

(8) Secondary Safety Valves test.

All test results met their acceptance criteria with the exception of RCS flow rate for the Reactor Coolant Pump (RCP) 11A and 11B combination, the performance of CEDM 15, and proper operation of the Turbine Bypass Valves. RCP 11A and 11B combination yielded a flow rate slightly below that required by present Technical Specifications. That combination is presently administratively restricted from use during power operation. CEDM 15 is a Part Length CEA in Part Length Group 2 (PLR 2). Performance testing revealed a defective upper gripper coil and/or latch in the CEDM. PLR 2 is presently administratively restricted from use while the reactor is critical. Discussion of problems with Turbine Bypass Valves is contained in Section 6.1.

3.1 CEDM/CEA Performance Tests

3.1.1 Purpose

The Control Element Drive Mechanism/Control Element Assembly (CEDM/CEA) Performance Tests were performed to accomplish the following four objectives.

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

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

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

(4) To verify that all Regulating and Shutdown CEA's have drop times for 90 percent insertion in accordance with Technical Specification 3.10.

3.1.2 Test Results

The CEDM/CEA Performance Tests were conducted at RCS temperatures and pressures of 260°F/450 psia and 532°F/2250 psia. The tests consisted of:

(1) A measurement of coil resistance.

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

(3) Timing of rod drop from full out to 90 percent insertion of all regulating and shutdown CEA's with full RCS flow (4 RCP's).

(4) Ten (10) additional drops of the fastest and slowest CEA's.

(5) Measurement of CEA withdrawal speed and rod drop time for all regulating and shutdown CEA's with zero RCS flow. Performed at RCS temperature and pressure of 532°F/2250 psia only.

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

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

The results of the CEDM speed and drop time test were found to be acceptable and are as follows:

<u>260°F/450 psia</u>

<u>CEDM Speed</u>

Fastest single CEA - CEDM 37 - 32.79 inches/min.
Slowest single CEA - CEDM 12 - 29.89 inches/min.
Fastest dual CEA - CEDM 39 - 20.40 inches/min.
Slowest dual CEA - CEDM 6 - 19.15 inches/min.

<u>Drop Time to 90% Full Insertion (2 RCP's)</u>

Slowest single CEA - CEDM 59 - 2.23 sec.
Fastest single CEA - CEDM 34 - 2.00 sec.
Slowest dual CEA - CEDM 51 - 2.03 sec.
Fastest dual CEA - CEDM 7 - 1.88 sec.

<u>532°F/2250 psia</u>

<u>CEDM Speed</u>

Fastest single CEA - CEDM 35 - 32.66 inches/min.
Slowest single CEA - CEDM 28, 58 - 31.52 inches/min.
Fastest dual CEA - CEDM 39 - 20.56 inches/min.
Slowest dual CEA - CEDM 8 - 19.84 inches/min.

<u>Drop Time to 90% Full Insertion (Full Flow)</u>

Slowest single CEA - CEDM 57 - 2.36 sec.
Fastest single CEA - CEDM 35 - 2.10 sec.
Slowest dual CEA - CEDM 48 - 2.22 sec.
Fastest dual CEA - CEDM 42 - 2.11 sec.

<u>Drop Time to 90% Full Insertion (Zero Flow)</u>

Slowest single CEA - CEDM 21 - 2.06 sec.
Fastest single CEA - CEDM 5, 35 - 1.94 sec.
Slowest dual CEA - CEDM 49, 51 - 1.88 sec.
Fastest dual CEA - CEDM 38 - 1.78 sec.

3.1.3 Conclusions

CEDM 15 has an apparent defective upper gripper coil or latch and was not fully tested. It was placed at the upper electrical limit (fully withdrawn) after initial testing and deenergized. Part Length CEA's (PLR) including CEDM 15 will be left at the upper electrical limit as present operating plans do not envision use of the PLR's. Operating procedures prevent use of PLR's during power operation.

All other test results were evaluated to be acceptable.

3.2 Reactor Coolant System Flow Tests

3.2.1 Purpose

The test was conducted to determine Reactor Coolant System (RCS) flow rates and pressure drops around the reactor coolant loops for various Reactor Coolant Pump (RCP) combinations.

3.2.2 Test Results

RCS flow measurements were taken at RCS temperatures and pressures of 260°F/450 psia and 532°F/2250 psia using permanently installed and temporary instrumentation. At 260°F/450 psia flow data was collected for one (1) and two (2) RCP combinations only. Measurements at 532°F/2250 psia were taken for the RCP combinations listed in Table 3.2-1.

3.2.3 Conclusions

Flow with two (2) RCP's in the same loop (RCP's 11A and 11B) was measured to be 185,255 gpm. This is 1.5% below the 49.7% of full flow specified in Technical Specification 3.1. Until such time as a part loop operation test is satisfactorily performed to determine power distributions for less than four (4) RCP operation and results are approved, administrative restrictions on plant operations prevent power operation with less than four (4) RCP's.

All other RCP combinations meet acceptance criteria and exceed Technical Specification requirements for minimum flow.

TABLE 3.2-1

REACTOR COOLANT PUMP FLOW DATA (1)

RCP's RUNNING	RCP FLOW (GPM)(3) 11A	11B	12A	12B	TOTAL FLOW (GPM)	% of MIN. TOTAL FLOW(2)
11A, 11B, 12A, 12B	96,939	99,656	104,035	103,239	403,871	105.06
11A, 11B, 12A	102,279	105,949	135,130	-37,950	305,408	79.45
11A, 11B, 12B	102,850	105,150	-37,150	133,150	304,000	79.08
11A, 12A, 12B	129,052	-37,200	109,545	108,829	310,226	80.70
11B, 12A, 12B	-36,700	129,400	109,750	108,050	310,500	80.77
11A, 11B	107,243	110,012	-16,750	-15,250	185,255	48.19
12A, 12B	-15,250	-17,000	113,400	112,250	193,400	50.31
11A, 12A	133,240	-31,250	139,953	-32,750	209,193	54.42
11A, 12B	134,600	-31,500	-30,750	138,100	210,450	54.74
11B, 12A	-33,900	134,937	140,817	-33,300	208,554	54.25
11B, 12B	-33,000	133,600	-31,350	137,250	206,500	53.72
11A	136,947	-27,750	-7,000	-6,750	95,447	24.83
11B	-29,500	137,222	-7,500	-5,750	94,472	24.57
12A	-8,800	-10,000	144,056	-29,650	95,606	24.87
12B	-8,750	-9,000	-27,850	140,550	94,950	24.70

(1) RCS temperature and pressure of 532°F/2250 psia

(2) Loss of Coolant Safety Analysis assumes a single RCP flow of 92,500 gpm. Due to measurement uncertainties a flow of 96,100 gpm must be measured to assure that the 92,500 gpm minimum flow used in Safety Analysis is met. Total measured minimum flow is therefore 384,400 gpm.

(3) A minus sign in front of a flow value indicates reverse flow through RCP; ie, short circuiting core.

3.3 Reactor Coolant System Flow Coastdown Tests

3.3.1 Purpose

The purpose of this test was to determine the Reactor Coolant System (RCS) flow coastdown characteristics by tripping Reactor Coolant Pumps (RCP) in various combinations.

3.3.2 Test Results

Testing was conducted at RCS conditions of 532°F/2250 psia. Recorder traces of RCS flow versus time after trip were made of each RCP combination listed in Table 3.3-1. Comparison of recorder traces for the four (4) RCP trips with predictions revealed that the measured coastdown curve was less conservative than prediction based on flow fractions versus time after trip. However, total RCS flow versus time after trip was more conservative than prediction. Subsequent evaluation of measured results by CE, Windsor confirmed the acceptability and conservatism of measured results when predictions of flow fraction versus time were revised to reflect as built RCS flow characteristics.

3.3.3 Conclusions

Measured results from the four (4) RCP trip were evaluated to be acceptable. Other results were used for evaluating the method of calculating Low Flow Trip setpoints for the Reactor Protective System (RPS).

TABLE 3.3-1

REACTOR COOLANT PUMP FLOW COASTDOWN COMBINATIONS

PUMPS INITIALLY RUNNING	PUMPS TRIPPED
11A, 11B, 12A, 12B	11A, 11B, 12A, 12B
11A, 11B, 12A, 12B	11A
11A, 11B, 12A, 12B	11B
11A, 11B, 12A, 12B	12A
11A, 11B, 12A, 12B	12B
11A, 11B, 12B - lowest total Mass Flow	12B
11A, 11B - lowest total Mass Flow	11B - results in fastest RCS flow coastdown
11B, 12B - lowest total Mass Flow	12B - results in fastest RCS flow coastdown

3.4 Primary and Secondary Water Chemistry

3.4.1 Purpose

To establish, monitor, and control primary and secondary water chemistry during plant heatup and conduct of Post Core Hot Functional (PCHF) Tests. Baseline data to support the Low Power Physics and Escalation to Power Test phases will be obtained.

3.4.2 Test Results

All primary and secondary water chemistry results during PCHF were either acceptable or when acceptance criteria were not met, corrective action was instituted to achieve acceptable conditions. During baseline testing for Reactor Coolant System (RCS) particulate level and soluble corrosion products, no unusual or unexpected results were obtained.

3.4.3 Conclusions

Overall the data collected indicated that good chemistry control was maintained even with the variable plant conditions required by the test program.

3.5 Pressurizer Effectiveness Test

3.5.1 Purpose

The objectives of the pressurizer effectiveness test were to prove that air assisted pressurizer spray valves will open and close fully when pressurizer pressure is 2250 psia $\pm$ 15 psi, Reactor Coolant System (RCS) temperature is 532 $\pm$ 5°F and four (4) Reactor Coolant Pumps (RCP) are in operation and to determine the effectiveness of the pressurizer sprays and heaters in controlling RCS pressure during plant transients.

Acceptance criteria for this test was as follows:

(1) Following an increase in pressure transient spray valves open fully and decrease pressurizer pressure at 82 $\pm$ 20 psi/min.

(2) Following a decrease in pressure transient backup heaters increase pressurizer pressure at 17.5 $\pm$ 3.5 psi/min.

3.5.2 Test Results

The pressurizer spray valves did indicate that they would fully open and close with pressurizer pressure at 2250 psia, RCS temperature at 532°F and four (4) RCP's in operation. Effectiveness of the pressurizer sprays and heaters were determined by measuring the rate of pressurizer pressure decrease with sprays only and pressure increase with backup heaters only. Results of these tests are shown on Table 3.5-1.

3.5.3 Conclusions

During runs number 1 - 6, one or both pressurizer spray valves were leaking. While it is not evident from the data, leaking spray valves could increase rate of pressure decrease and decrease rate of pressure increase. Run number 7 was conducted during Escalation to Power Testing (EPT) after maintenance on the spray valves was completed. At which time test results satisfied the acceptance criteria.

TABLE 3.5-1

PRESSURIZER SPRAY AND HEATER EFFECTIVENESS TEST

RUN	PRESSURE DECREASE (PSI/MIN)	PRESSURE INCREASE (PSI/MIN)
1	70	18
2	78	17
3	90	15
4	74	12
5	73	13
6	104	16
7	72	15

3.6 Reactor Coolant System Leak Test

3.6.1 Purpose

A leak test of the Reactor Coolant System (RCS) was performed to check for indications of abnormal leakage from the primary system.

3.6.2 Test Results

The leak test of the RCS was conducted at 2300 (+25, -0) psia and covered the following areas:

(1) Reactor Coolant Pump (RCP) Seal Area

(2) Reactor Vessel Head Seal

(3) Steam Generator Manways

(4) Control Element Drive Mechanism (CEDM) In-Core Instrument Penetrations

(5) Stem leakage on all valves

(6) Pressurizer Heater Penetrations

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

3.6.3 Conclusions

The RCS leak test showed that the primary system was tight after reactor vessel reassembly following fuel load and no abnormal leakage should be expected during the Startup Test Program.

4.0 INITIAL APPROACH TO CRITICALITY

Initial criticality was achieved on 10/7/74 at Reactor Coolant System (RCS) conditions of 260°F and 475 psia. The initial RCS boron concentration was 1990 ppm. The Initial Approach to Criticality (IAC) began by withdrawing all CEA's in specified increments with count rate data taken after each increment. Criticality was subsequently achieved by deborating the RCS to a boron concentration of 1033 ppm at a rate of 44 gpm.

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. At the end of each reactivity addition, 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 reactivity addition was the value plotted.

The CEA withdrawal sequence and intervals are show in Table 4.0-1. The inverse count rate versus CEA position points for each WRLC are shown in Figures 4.0-1 through 4.0-4. The inverse count rate versus RCS dilution time in hours is shown in Figures 4.0-5 through 4.0-8. WRLC's 1 and 3 detectors were located closest to startup sources. The relatively greater scatter in data from WRLC's 2 and 4 was due to their greater distance from the startup sources. Figure 4.0-9 shows the change in RCS boron concentration versus dilution time.

During the approach to criticality, response of the WRLC's was monitored. The degree of response is recorded in Table 4.0-2 and indicates that a greater than one (1) decade "overlap" existed between the proportional counters and the fission chamber of each WRLC.

After achieving initial criticality, Control Element Assembly (CEA) Group 5 was used to control neutron flux. Conditions were stabilized at 10^{-4}% power and the critical data shown in Table 4.0-3 was recorded and compared with predicted values.

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

TABLE 4.0-1

CEA WITHDRAWAL SEQUENCE

CEA POSITION POINT	CEA GROUP	INCHES WITHDRAWN
1	Shutdown A	67.0
2	Shutdown A	Full Out
3	Shutdown B	67.0
4	Shutdown B	Full Out
5	Shutdown C	67.0
6	Shutdown C	Full Out
7	Part Length CEA's (PLR)	Full Out
8	Regulating 1	33.5
9	Regulating 1	67.0
10	Regulating 1	100.5 (2@ 11.25)
11	Regulating 1	136.0 (2@ 44.75)
12	Regulating 2	85.5 (3@ 3.5)
13	Regulating 2	119.0 (3@ 29.5)
14	Regulating 3	70.5 (2@ 136.0)
15	Regulating 3	104.0 (4@ 14.75)
16	Regulating 4	55.5 (3@ 136.0)
17	Regulating 4	89.0 (5@ 3.75)
18	Regulating 4	122.5 (5@ 33.25)
19	Regulating 5	52.0 (4@ 136.0)

TABLE 4.0-2

WIDE RANGE LOG CHANNEL RESPONSE[1]

Signal Source [2]	CHANNEL 1	CHANNEL 2	CHANNEL 3	CHANNEL 4
Proportional Counters and Fission Chambers				
CPS	2×10^3	2×10^3	2×10^3	2×10^3
Percent Power	7×10^{-5}	7×10^{-5}	9×10^{-5}	8×10^{-5}
Fission Chamber Only				
Percent Power	10^{-6}	10^{-6}	10^{-6}	10^{-6}

(1) Each Wide Range Log Channel (WRLC) signal is a combination of a signal from several proportional counters and a signal from a fission chamber.

(2) At approximately 2000 counts per second (cps) increasing, proportional counters are automatically deenergized and the WRLC signal consists of the fission chamber signal only.

TABLE 4.0-3

INITIAL CRITICALITY DATA

PARAMETER	MEASURED	INITIAL CONDITION
RCS Temperature (°F)	260	260
RCS Pressure (psia)	475	460
RCP's Operating	11A and 11B	11A and 11B
WRLC 1 (% power)	2×10^{-4}	1×10^{-4}
WRLC 2 (% power)	4×10^{-4}	1×10^{-4}
WRLC 3 (% power)	1×10^{-4}	1×10^{-4}
WRLC 4 (% power)	2×10^{-4}	1×10^{-4}
CEA GROUPS (inches)		PREDICTED
Shutdown A	135	135
Shutdown B	135	135
Shutdown C	135	135
Reg 1	135	135
Reg 2	135	135
Reg 3	135	135
Reg 4	134	134
Reg 5	47	52
PLR 8	135	135
RCS Boron Concentration (ppm)	1033	1050 ± 100

CALVERT CLIFFS UNIT 1

INITIAL APPROACH TO CRITICALITY

BOL, 1st CYCLE, 260°F, 480 PSIA

WIDE RANGE LOG CHANNEL 1

ZERO POWER - CEA GROUP 5 @ 50"

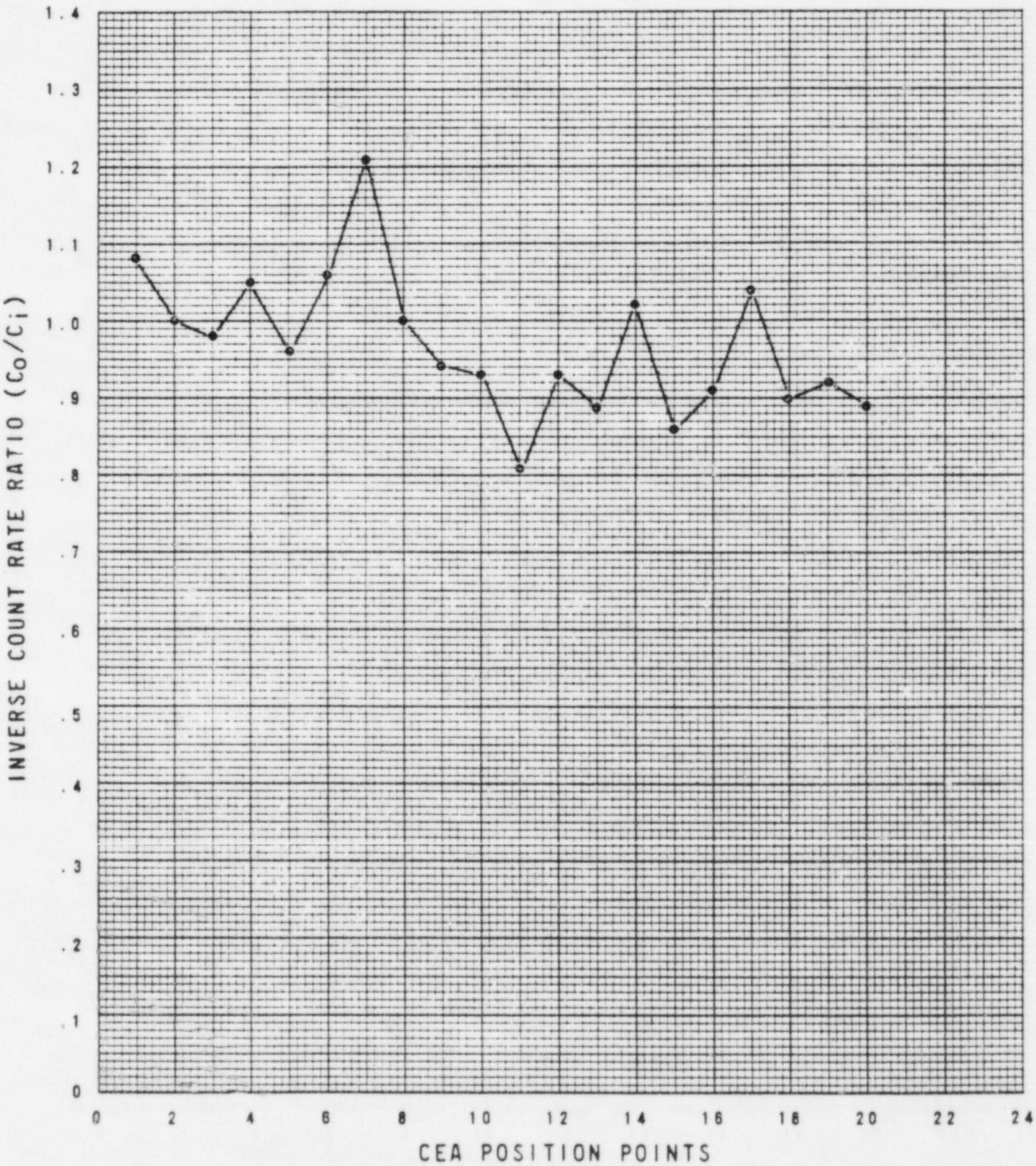


FIGURE 4.0-1

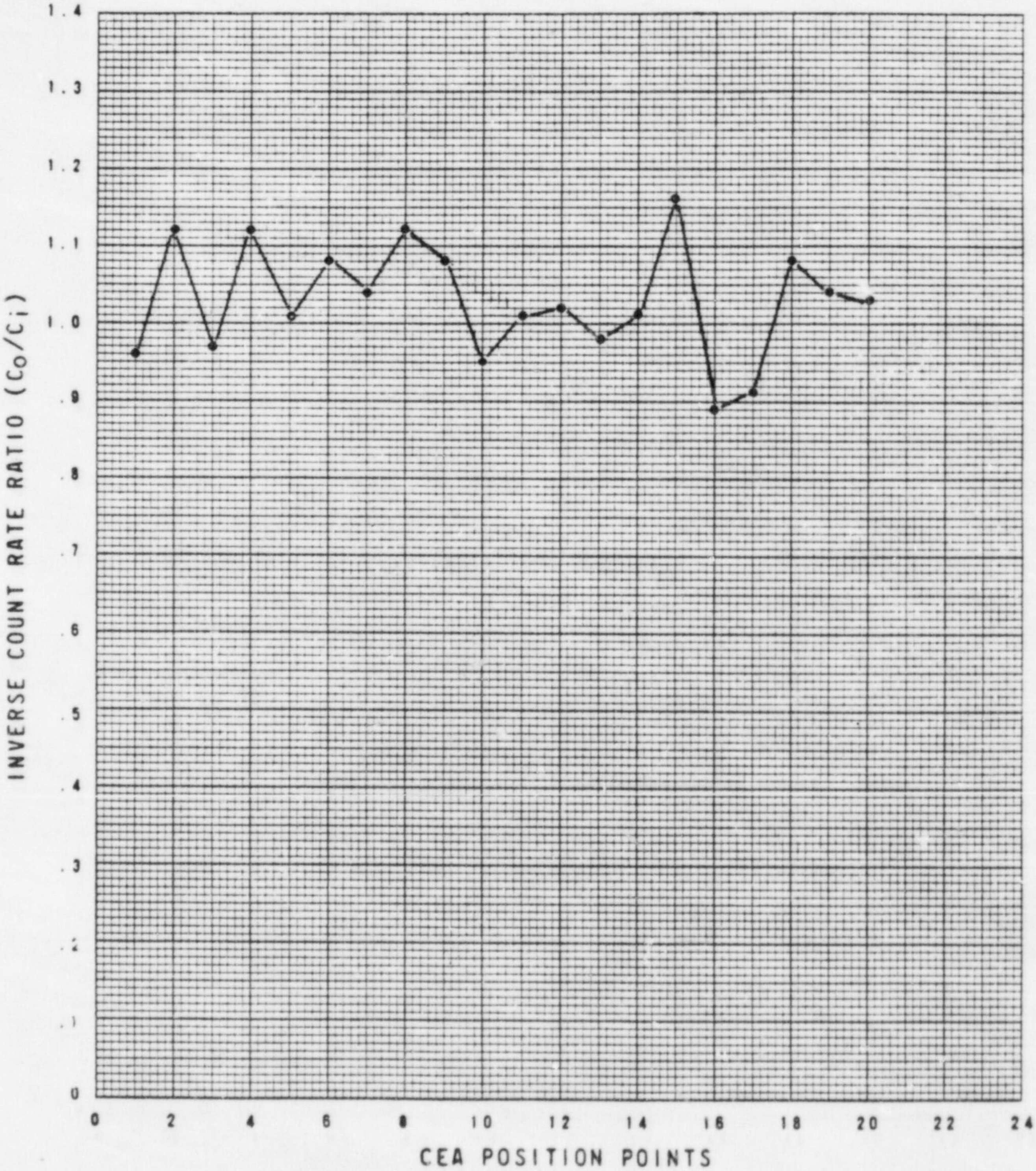
CALVERT CLIFFS UNIT 1

INITIAL APPROACH TO CRITICALITY

BOL, 1st CYCLE, 260°F, 480 PSIA

WIDE RANGE LOG CHANNEL 2

ZERO POWER - CEA GROUP 5 @ 50"



CALVERT CLIFFS UNIT 1

INITIAL APPROACH TO CRITICALITY

BOL, 1st CYCLE, 260°F, 480 PSIA

WIDE RANGE LOG CHANNEL 3

ZERO POWER - CEA GROUP 5 @ 50"

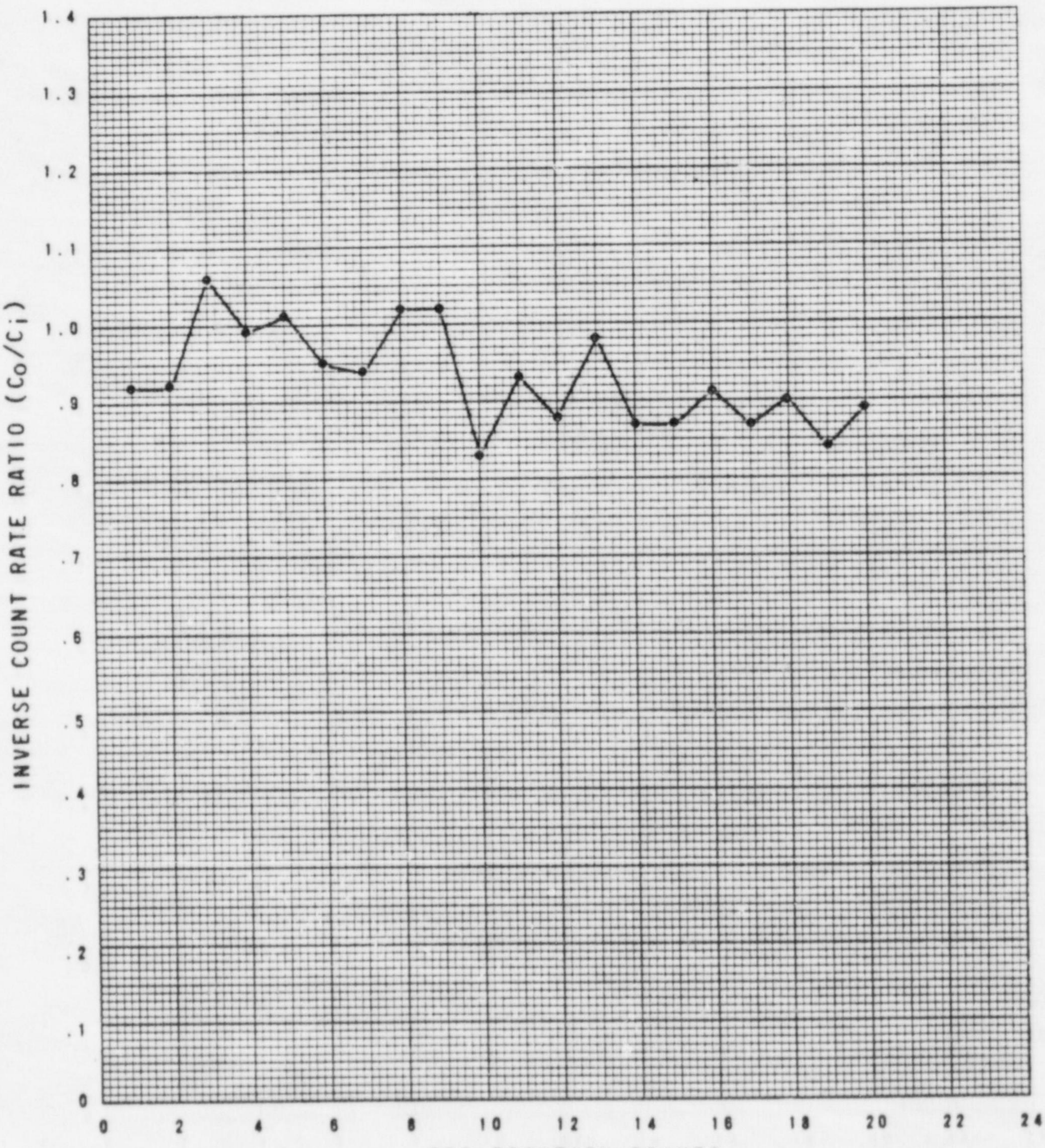


FIGURE 4.0-3

CALVERT CLIFFS UNIT 1

INITIAL APPROACH TO CRITICALITY

BOL, 1st CYCLE, 260°F, 480 PSIA

WIDE RANGE LOG CHANNEL 4

ZERO POWER - CEA GROUP 5 @ 50"

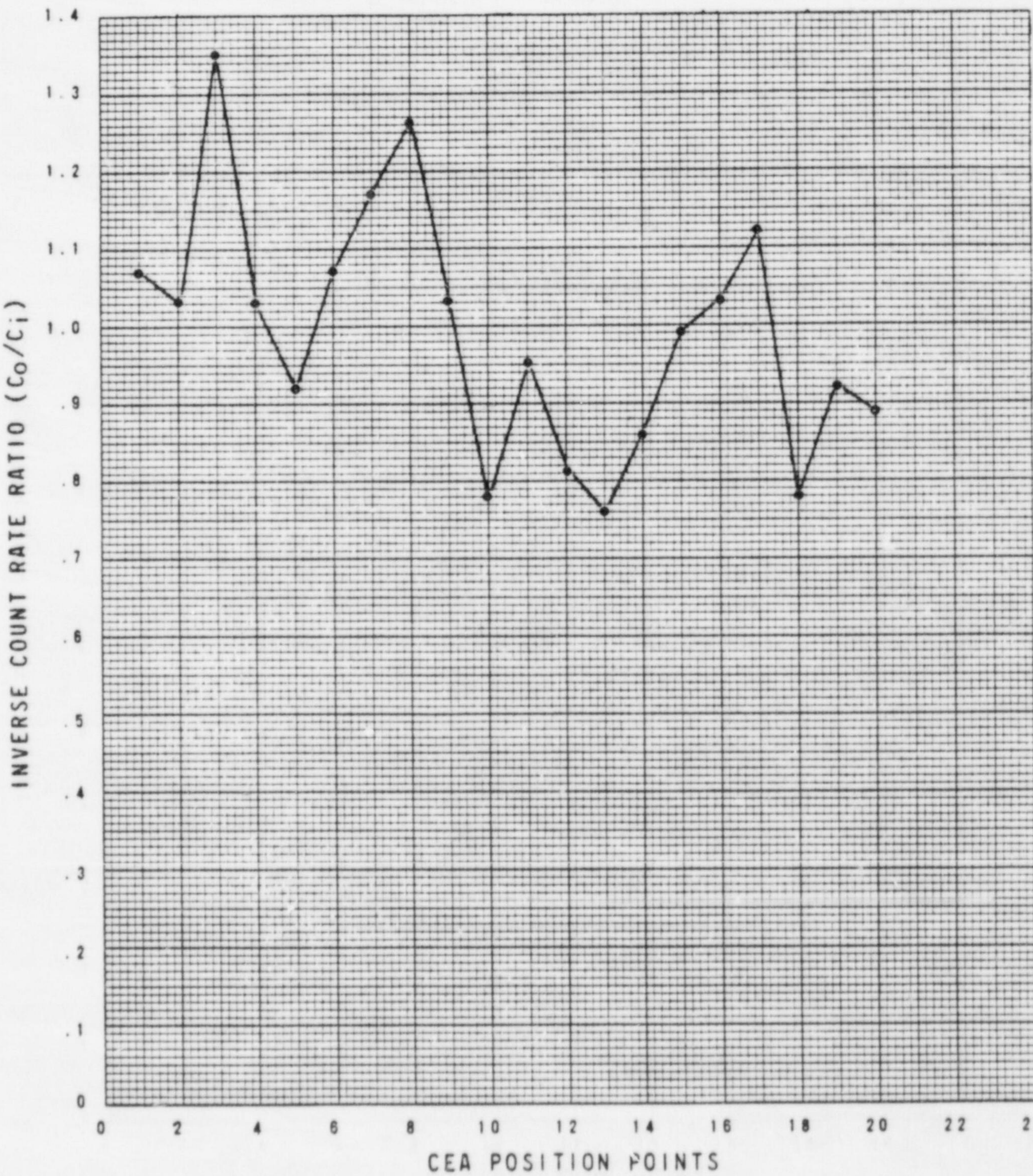


FIGURE 4.0-4

CALVERT CLIFFS UNIT 1

INITIAL APPROACH TO CRITICALITY

BOL, 1st CYCLE, 260°F, 480 PSIA

WIDE RANGE LOG CHANNEL 1

ZERO POWER - CEA GROUP 5 @ 50"

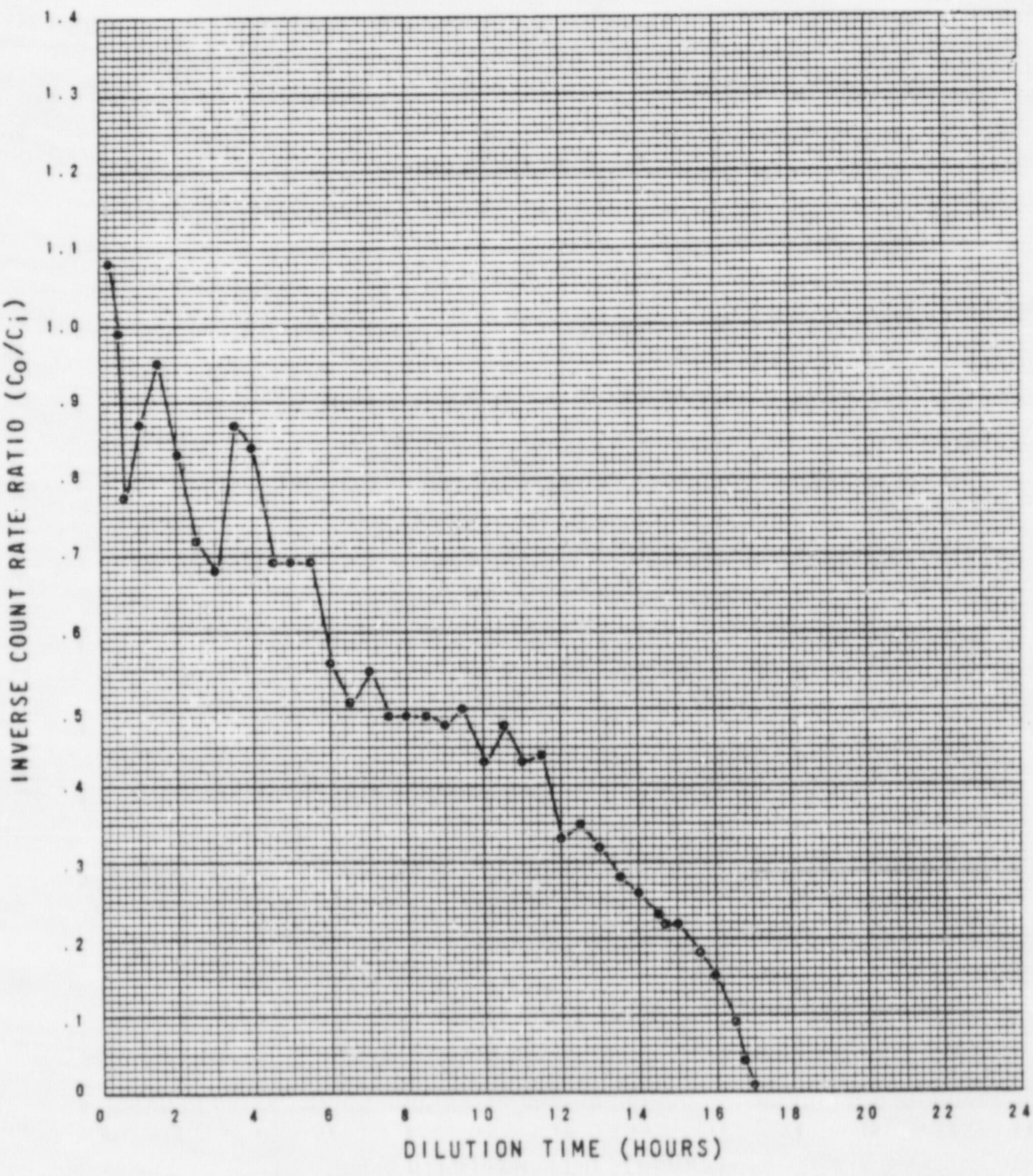


FIGURE 4.0-5

CALVERT CLIFFS UNIT 1

INITIAL APPROACH TO CRITICALITY

BOL, 1st CYCLE, 260°F, 480 PSIA

WIDE RANGE LOG CHANNEL 2

ZERO POWER - CEA GROUP 5 @ 50"

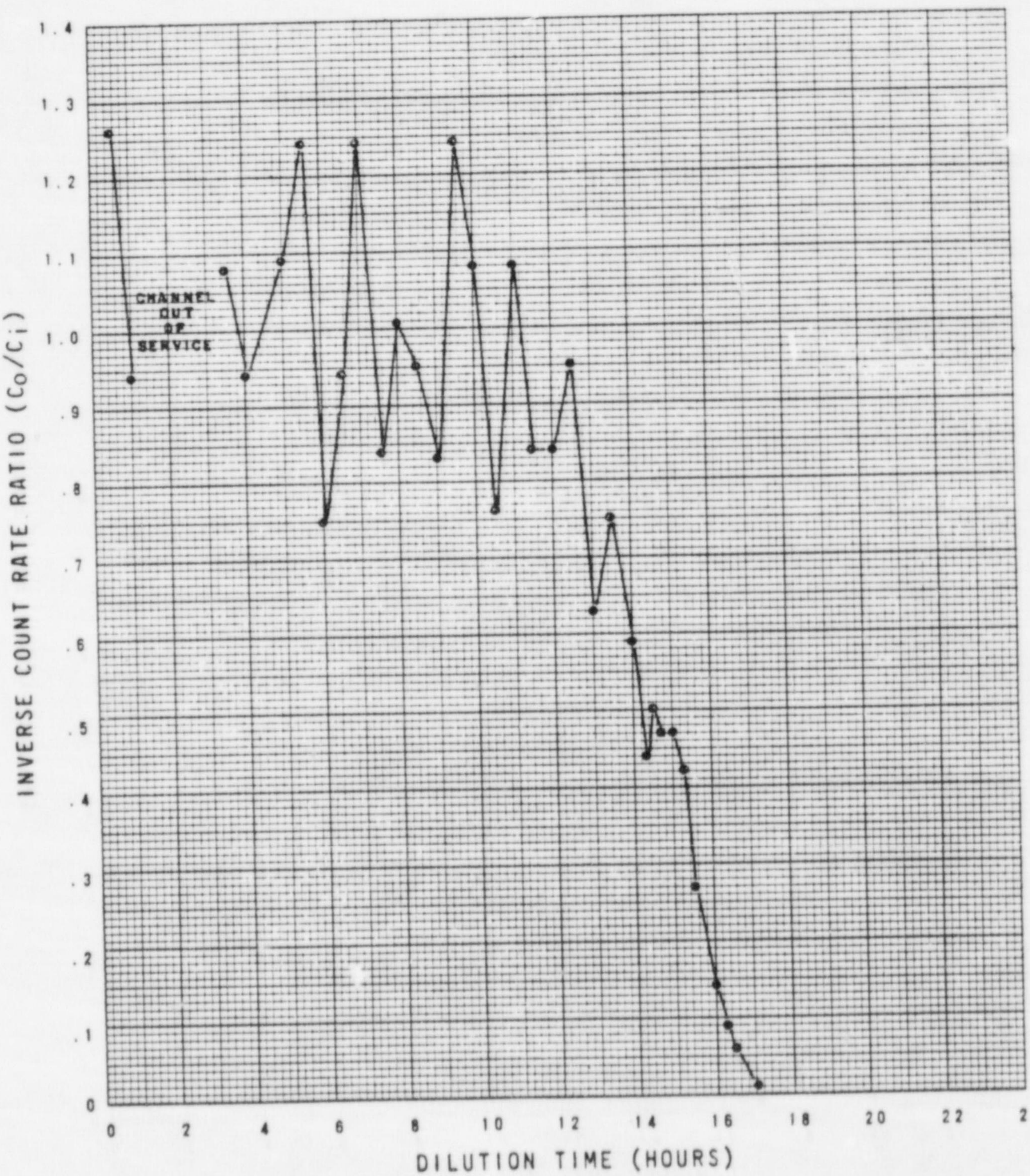


FIGURE 4.0-6

CALVERT CLIFFS UNIT 1

INITIAL APPROACH TO CRITICALITY

BOL, 1st CYCLE, 260°F, 480 PSIA

WIDE RANGE LOG CHANNEL 3

ZERO POWER - CEA GROUP 5 @ 50"

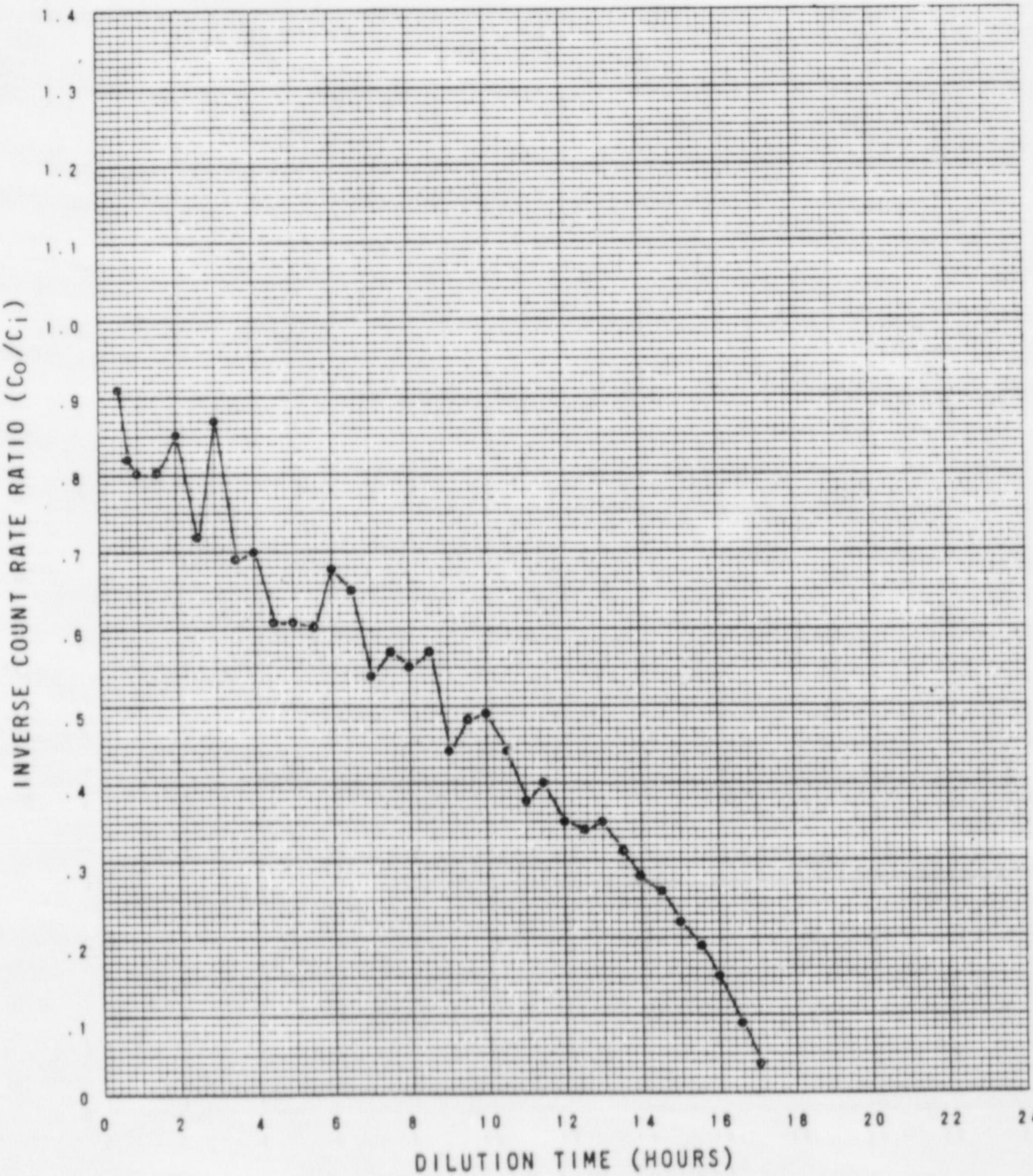


FIGURE 4.0-7

CALVERT CLIFFS UNIT 1

INITIAL APPROACH TO CRITICALITY

BOL, 1st CYCLE, 260°F, 480 PSIA

WIDE RANGE LOG CHANNEL 4

ZERO POWER - CEA GROUP 5 @ 50"

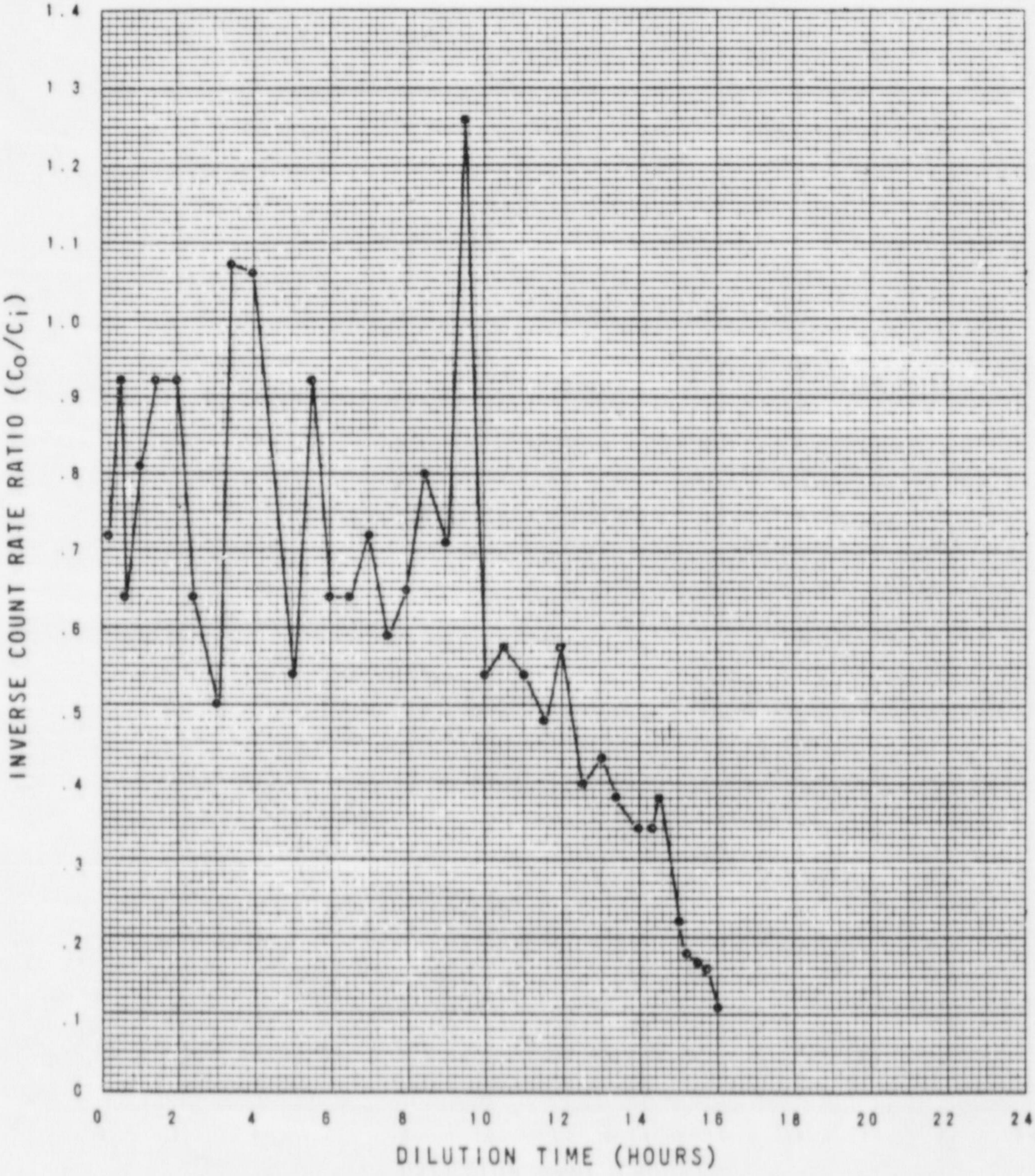


FIGURE 4.0-8

CALVERT CLIFFS UNIT 1

BOL, 1st CYCLE

REACTOR COOLANT SYSTEM BORON CONCENTRATION

VS DILUTION TIME

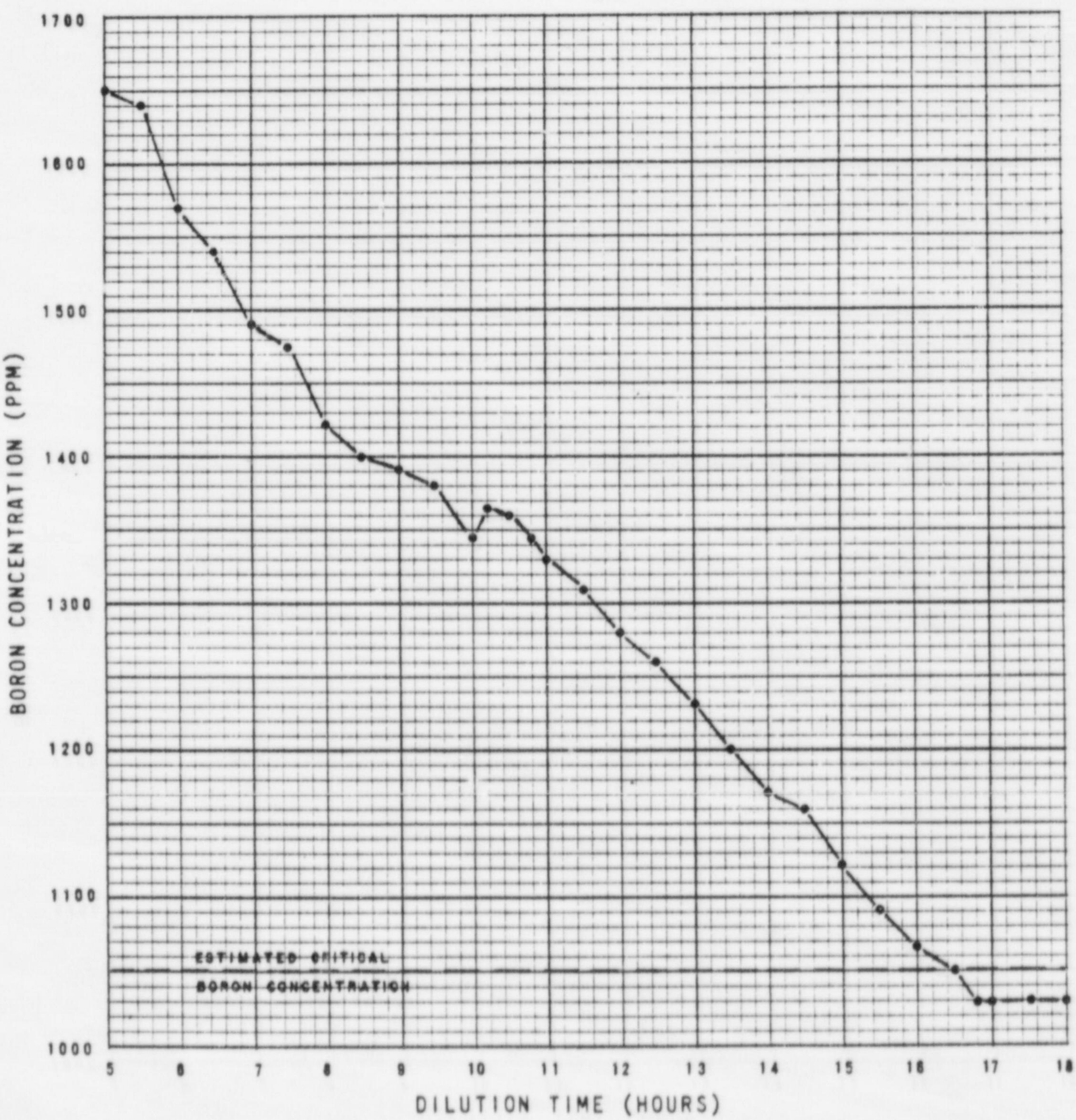


FIGURE 4.0-9

5.0 LOW POWER PHYSICS TESTS (LPPT)

The Calvert Cliffs Unit 1 initial core consists of two hundred seventeen (217) fuel assemblies each containing one hundred seventy-six (176) fuel rods/burnable poison rods and five (5) Control Element Assembly (CEA) guide tubes. Fuel assemblies are divided into three (3) distinct groups by enrichment, Type A, B, and C. Twelve (12) fuel rods in all Type B and several Type C fuel assemblies are replaced with burnable poison rods. Table 5.0-1 tabulates this and other important core design characteristics.

In addition to soluble boron in the Reactor Coolant System (RCS), reactivity control is provided by eighty-five (85) CEA's. CEA's are inserted into and withdrawn from the core by means of sixty-five (65) Control Element Drive Assemblies (CEDM's). Twenty (20) CEDM's are attached to dual CEA's. Figure 5.0-1 shows the core location of the CEA's. Note that dual CEA's have a single serial number designation corresponding to the serial number of their respective CEDM. The CEDM's are arranged into ten (10) CEA Groups. Those Groups are further defined by function. CEA Groups A, B, and C are Shutdown Groups. CEA Groups 1 through 5 are Regulating Groups. CEA Groups PLR 1 and PLR 2 are Power Shaping Groups. PLR 1 and PLR 2 may be combined into a single CEA Group designated PLR 8. Figure 5.0-2 displays the relative core location of the CEA Groups.

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

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

During the conduct of LPPT, several unusual events occurred. A relatively rapid increase in RCS pressure drop across the reactor vessel was noted. This necessitated cooling down, removing the reactor vessel head and some internals, and thoroughly inspecting the internal reactor vessel cavity and fuel for a cause of the differential pressure buildup. When that situation was resolved, the reactor vessel was reassembled. However, as discovered during subsequent testing, several CEA's were not properly connected to their CEDM's. This required another cooldown and reactor vessel head removal and reassembly. A more detailed description of these events is included in Section 7.0.

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

TABLE 5.0-1

FIRST CYCLE CORE DESIGN CHARACTERISTICS

Nuclear Characteristics

Fuel Managment	3-Batch, Mixed Central Zone
Average First Cycle Burnup, MWd/MTU	15,400
U-235 Enrichment, w/o	
Batch A (69 assemblies)	2.05
Batch B (80 assemblies)	2.45
Batch C (68 assemblies)	2.99
H_2O/UO_2 Volume Ratio, Unit Cell (Cold)	1.63

Mechanical Characteristics

Fuel Assemblies

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

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

TABLE 5.0-1 (cont'd)

Weight of Contained Uranium, kg U		
Batch A	395	
Batch B	368	
Batch C (poisoned)	368	
Batch C (unpoisoned)	395	
Outside Dimensions		
Fuel Rod to Fuel Rod, inches	7.980 x 7.980	
Fuel Rod		
Fuel Material (Sintered Pellets)	UO_2	
Pellet Diameter, inches	.3795	
Pellet Dish Depth, inches	0.015	
Pellet Dish Diameter, inches	0.2915	
Pellet Length, inches	0.650	
Pellet Density, g/cc	10.193	
Pellet Theoretical Density, g/cc	10.96	
Pellet Density (% theoretical)	93.0 ± 1.5	
Stack Height Density, g/cc	10.054	
Clad Material	Zircaloy-4	
Clad ID, inches	0.3880	
Clad OD, (nominal) inches	0.440	
Clad Thickness, (nominal) inches	0.026	
Diametral Gap, (cold, nominal), inches	0.0085	
Active Length, inches	136.7	
Burnable Poison Rod		
Active Length, inches	122.7	
Material	B_4C - Al_2O_3	
Pellet Diameter, inches	0.376	
Clad Material	Zircaloy-4	
Clad ID, inches	0.388	
Clad OD, inches	0.440	
Clad Thickness, (nominal) inches	0.026	
Diametral Gap, (cold, nominal), inches	0.012	
Control Element Assembly (CEA)		
	Full Length	Part Length
Number	77	8
Number of Absorber Elements per Assembly	5	5
Type	Cylindrical Rods	Cylindrical Rods
Clad Material	Inconel 625	Inconel 625
Clad Thickness, inches	0.040	0.040
Clad OD, inches	0.948	0.098
Poison Material	(1)	(1)
Total Element Length	161.31	161.31

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

CALVERT CLIFFS UNIT 1
CORE LOCATION OF THE CEA'S
BOL, 1st CYCLE

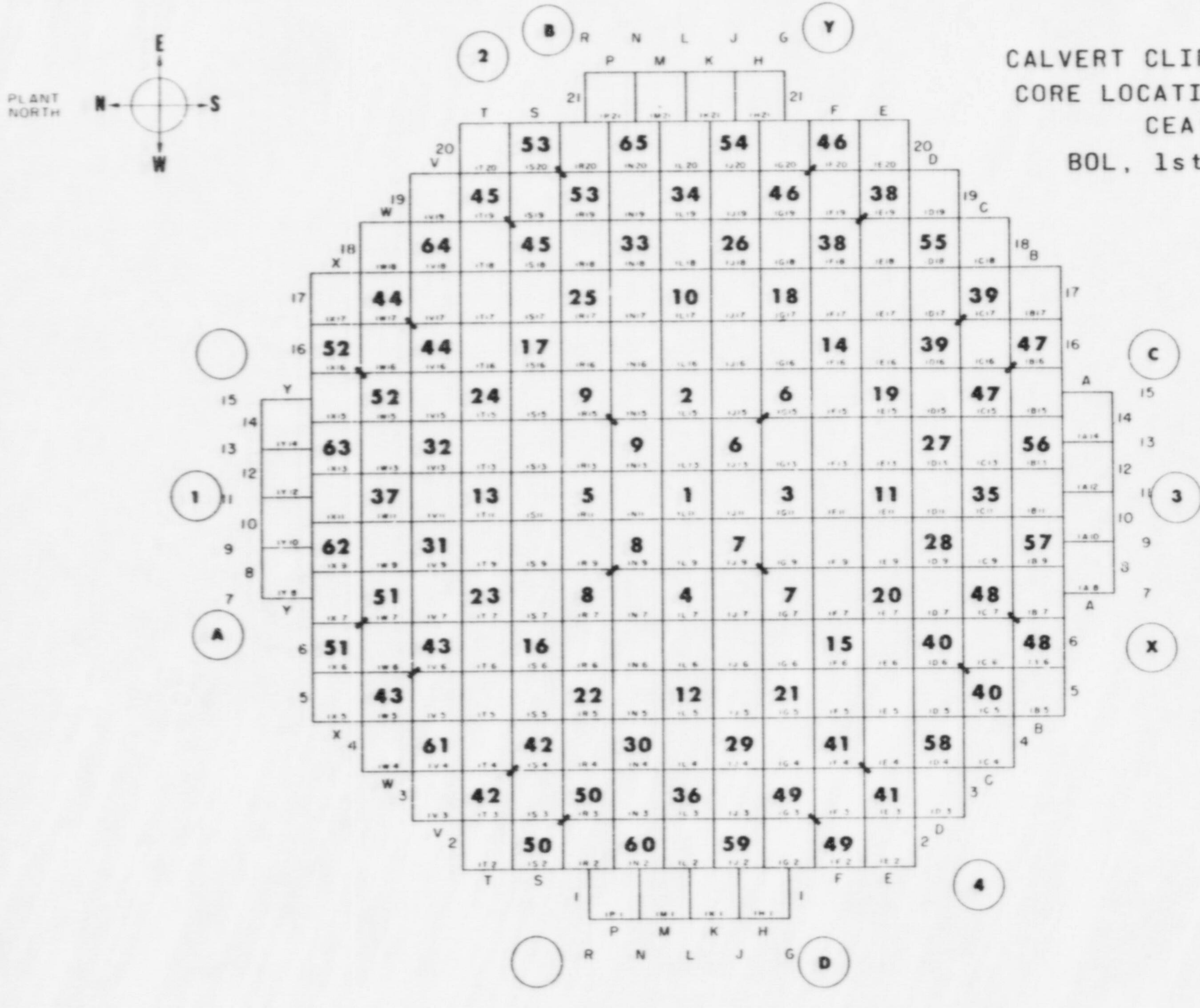


FIGURE 5.0-1

CALVERT CLIFFS UNIT 1
RELATIVE CORE LOCATION
OF THE CEA GROUPS
BOL, 1st CYCLE

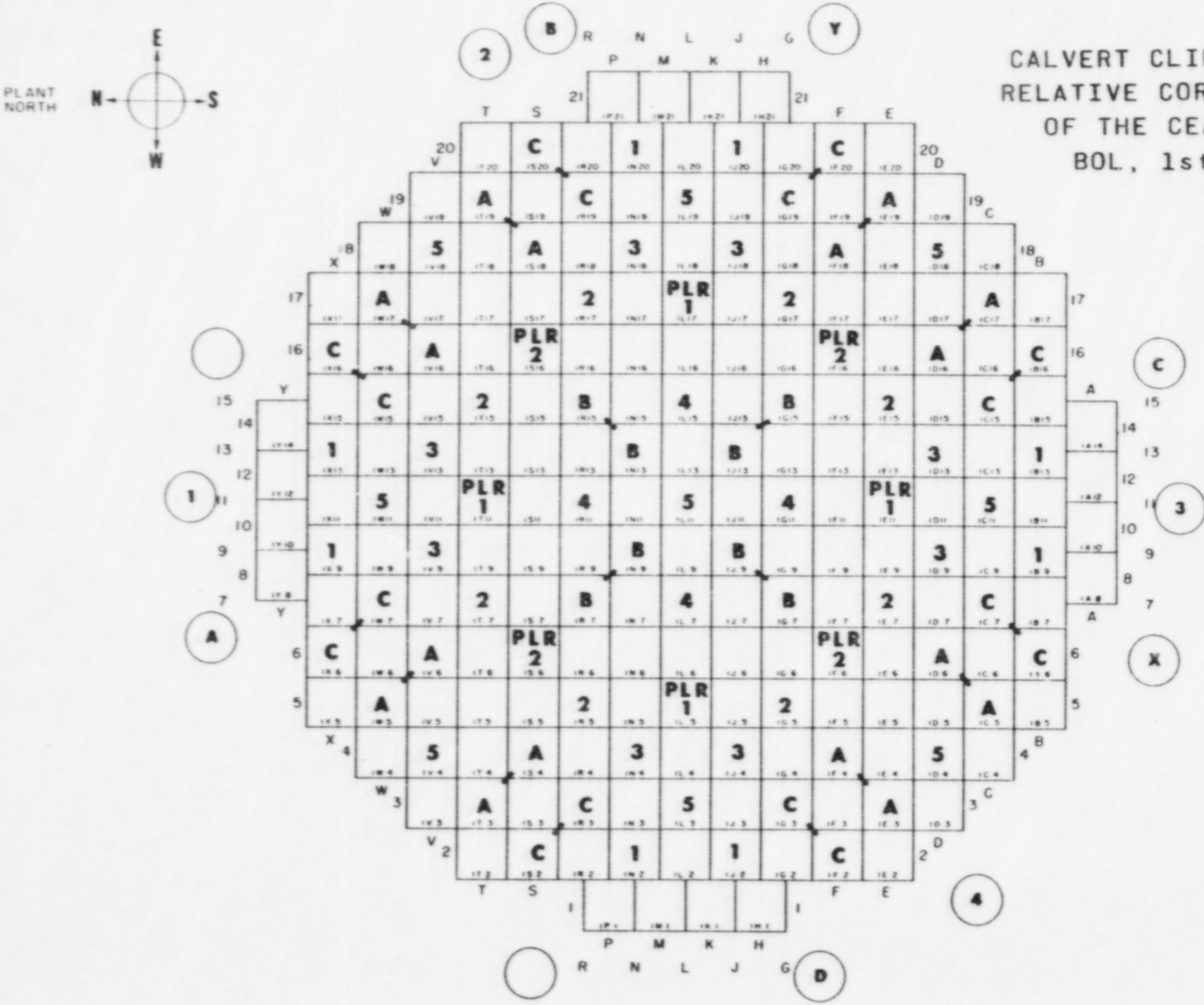


FIGURE 5.0-2

5.1 Shielding Effectiveness and Plant Radiation Level Measurements

5.1.1 Purpose

A comprehensive survey of radiation levels around the plant in general and the biological shield in particular was performed. This survey was the basis for determining later buildup of radioactivity in particular piping and components. It was also used to predict radiation levels at higher power levels and to evaluate efficiency of shielding. Design bases dose rates for the several zones are listed in FSAR Section 11.2.1.

5.1.2 Results

General area gamma dose rates and all shielding point gamma and neutron dose rates were less than 0.1 mrem/hr for all Survey Points outside of Unit 1 Containment. All except six of the TLD's placed outside of Unit 1 Containment indicated dose rates attributable to natural background radiation. One TLD mounted near the boronometer showed a measurable but small (0.05 mrem/hr average) dose from the boronometer source, and five TLD's mounted on the Units 1/2 barrier fence in the turbine building showed some exposure above natural background attributable to radiography from Unit 2 construction.

Measurable gamma and neutron dose rates were obtained at various locations within Unit 1 Containment during routine radiological surveys throughout LPPT. It was difficult to extrapolate readings obtained into meaningful full-load dose rates due to sensitivity of measurements and the uncertainties in the power measurement. However, based on TLD results and neutron survey meter results, it was estimated that at 100 percent power the total exposure rate just inside the containment personnel hatch may be 10 - 100 rem/hr and 30 - 300 rem/hr near 12B Safety Injection Tank with gamma and neutron each contributing about one-half the exposure.

5.1.3 Conclusions

The maximum general area dose rates at all survey locations when extrapolated to 100 percent power were generally consistent with the criteria presented in FSAR, Section 11.2.1.

5.2 Effluent Radiation Monitors Calibration

5.2.1 Purpose

Gaseous and liquid effluent in process radiation monitors were initially calibrated using manufacturers detection efficiency data. During initial radioactive fluid releases, grab sample analysis is compared with in process monitor response to establish valid correlations for future reference.

5.2.2 Test Results

Fluid releases to the environment were of such low radioactivity levels that collection of meaningful data for a valid comparison of monitor responses and grab sample analyses was not practical.

5.2.3 Conclusions

Effluent radiation monitor calibrations were deferred until generation of higher Reactor Coolant System liquid and gaseous radioactivity levels expected during Escalation to Power Testing.

5.3 Critical Boron Concentration Measurements

5.3.1 Purpose

Critical boron concentration measurements were performed at various Reactor Coolant System temperatures and pressures. The purpose of these measurements was to obtain an as measured value for excess reactivity loaded in core and to provide bases for verification of predicted reactivity worths.

5.3.2 Test Results

Boron concentration values are averages of multiple chemical analysis measurements made during periods of stable Reactor Coolant System (RCS) boron concentration. As indicated in Table 5.3-1 below, measurement points were also independently analyzed by the reactor vendor (CE).

TABLE 5.3-1

COMPARISON OF PREDICTED AND MEASURED CRITICAL BORON CONCENTRATION

NOMINAL RCS TEMPERATURE AND PRESSURE	CEA GROUP 5 POSITION (INCHES WITHDRAWN)	PREDICTED VALUE (ppm)	MEASURED VALUE (ppm)		ACCEPTANCE LIMITS (ppm)
			BG&E	CE	
260°F, 460 psia	121.5"	1062	1048	1060	± 100
532°F, 2250 psia	124.5"	1078	1089	1102	± 100

5.3.3 Conclusions

Results indicate that measured boron concentration are in adequate agreement with predictions and well within the acceptance criterion of ± 100 ppm.

5.4 Temperature Coefficient of Reactivity Measurements

5.4.1 Purpose

The moderator temperature coefficient of reactivity can be either negative or positive, depending upon the magnitude of the Reactor Coolant System boron concentration. The moderator temperature coefficient cannot be measured directly but it can be derived from a measurement of the isothermal temperature coefficient. The assumption being that the measured reactivity change due to a change in measured moderator temperature is actually a function of a like and simultaneous change in the temperature of all core components including fuel and moderator. The moderator temperature coefficient at full power condition as derived and extrapolated from the zero power measurement of isothermal temperature coefficient must be less positive than that required by Technical Specifications.

5.4.2 Test Results

Isothermal temperature coefficient measurements were conducted at several different Reactor Coolant System temperatures 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 improves with increase in Reactor Coolant System temperature and shows good agreement at 532°F.

Extrapolation of the zero power all rods out isothermal temperature coefficient resulted in a full power moderator coefficent of $-0.26\times10^{-4}\Delta k/k/^{o}F$. Technical Specification 3.10.I.1 specifies that the moderator temperature coefficient shall not be greater than $+0.2\times10^{-4}\Delta k/k/^{o}F$ at full power.

5.4.3 Conclusions

For all cases, the measured values of isothermal temperature coefficient are within the acceptance criterion of $+0.5\times10^{-4}$ $\Delta k/k/^{o}F$ of the predicted value. The extrapolated value of the zero power isothermal temperature coefficient to the full power condition is less positive than the limit specified in Technical Specifications and is therefore acceptable.

TABLE 5.4-1

SUMMARY OF ISOTHERMAL TEMPERATURE COEFFICIENT MEASUREMENTS

NOMINAL RCS TEMPERATURE AND PRESSURE	RCS BORON CONCENTRATION (ppm)	CEA GROUP POSITION (INCHES WITHDRAWN)	ISOTHERMAL TEMPERATURE COEFFICIENT ($x10^{-4}$ $\Delta k/k/°F$) MEASURED	PREDICTED
260°F, 460 psia	1022	5@0" 4@0 to 36" (3)	+0.125	-0.030
260°F to 360°F, 460 psia	1071 (1)	5@68"	+0.211	+0.140
360°F to 450°F, 1100 psia	1084 (2)	5@68"	+0.240	+0.230
532°F, 2250 psia	1087	5@105"	+0.257	+0.300
532°F, 2250 psia	949	5@0" 4@0" 3@18"	-0.289	-0.260
532°F, 2250 psia	720	1-5@0" C@10 to 30" (3)	-0.809	-0.680

(1) @ RCS temperature of 310°F

(2) @ RCS temperature of 405°F

(3) CEA Group position was changed during measurement, however, data reduction took account of this movement.

5.5 Non-Overlapped Regulating and Shutdown CEA Group Worth Measurements

5.5.1 Purpose

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

5.5.2 Test Results

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

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

The integral worths of CEA Groups 4 and 5 were measured at a Reactor Coolant System temperature of 260°F. The integral worths of all Shutdown and Regulating CEA Groups were measured at 532°F. These results are compared with predicted valves in Table 5.5-2. In addition, the integral reactivity worth curves developed at 532°F for all Shutdown and Regulating CEA Groups are displayed in Figures 5.5-1 through 5.5-8.

5.5.3 Conclusions

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

TABLE 5.5-1

REACTIVITY CONTROL FUNCTION OF CEA GROUPS

CEA GROUP NUMBER	NUMBER OF CEA's	CONTROL FUNCTION
A	16	Safety
B	8	Safety
C	16	Safety
1	8	Power Regulating
2	8	Power Regulating
3	8	Power Regulating
4	4	Power Regulating
5	9	Power Regulating
PLR 1	4	Axial Power Shaping
PLR 2	4	Axial Power Shaping

TABLE 5.5-2

COMPARISON OF MEASURED AND PREDICTED CEA GROUP INTEGRAL REACTIVITY WORTHS

I. Reactor Coolant System Temperature at 260°F

CEA GROUP	NUMBER OF CEA'S	MEASURED WORTH (% $\Delta k/k$)	PREDICTED WORTH (% $\Delta k/k$)	ACCEPTANCE LIMITS (% $\Delta k/k$)
4	4	0.135	0.120	$\pm$ 0.05
5	9	0.453	0.430	$\pm$ 0.11

II. Reactor Coolant System Temperature at 532°F

CEA GROUP	NUMBER OF CEA'S	MEASURED WORTH (% $\Delta k/k$)	PREDICTED WORTH (% $\Delta k/k$)	ACCEPTANCE LIMITS (% $\Delta k/k$)
A	16	3.327	3.200	$\pm$ 0.85
B	8	0.997	0.990	$\pm$ 0.220
C	16	1.303	1.400	$\pm$ 0.350
1	8	0.953	1.040	$\pm$ 0.260
2	8	0.778	0.740	$\pm$ 0.190
3	8	0.932	0.900	$\pm$ 0.230
4	4	0.353	0.350	$\pm$ 0.090
5	9	0.552	0.540	$\pm$ 0.140
Total	77	9.195	9.160	

CALVERT CLIFFS UNIT 1

INTEGRAL CEA GROUP WORTH

BOL, 1st CYCLE, 532°F, 2250 PSIA

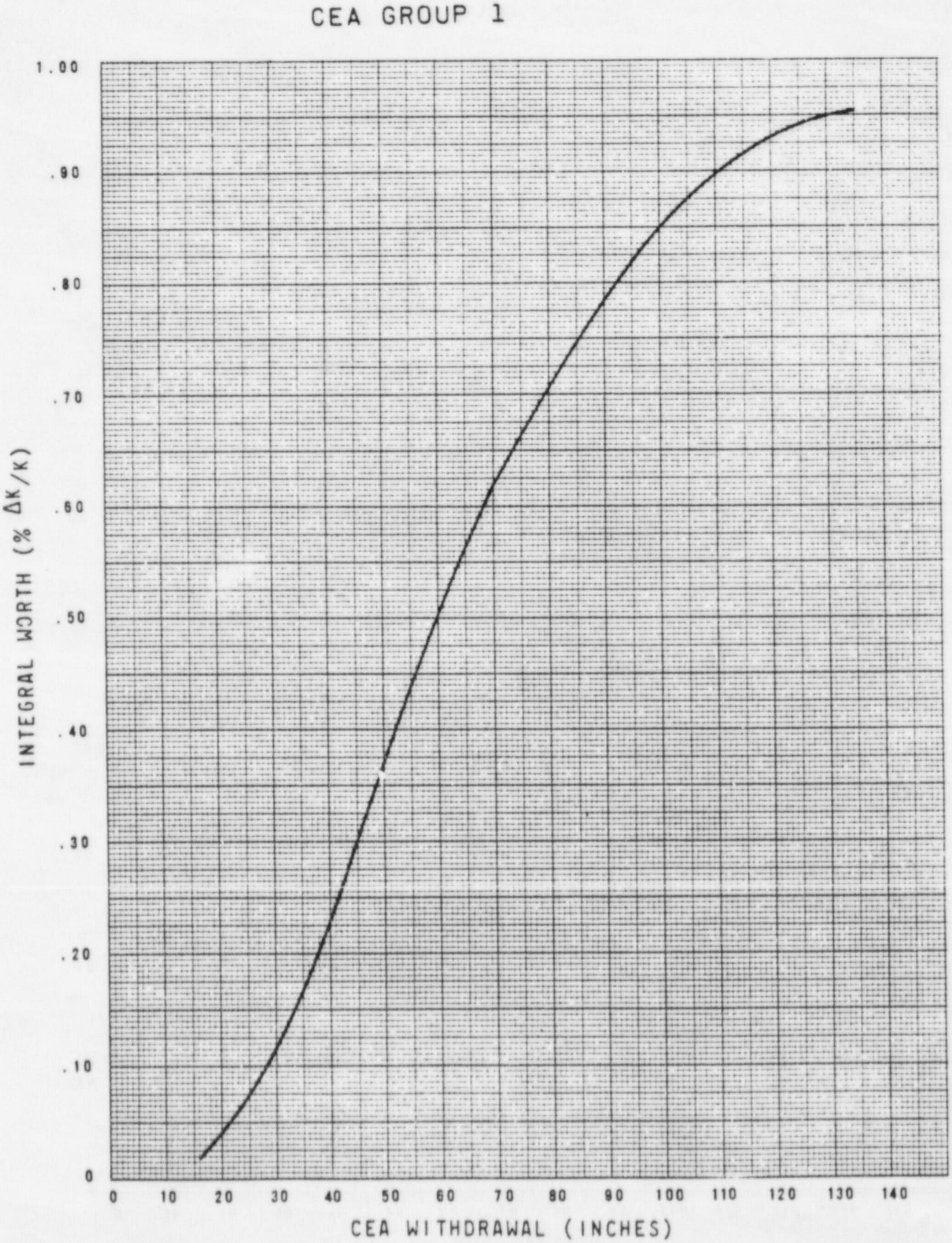


FIGURE 5.5-1

CALVERT CLIFFS UNIT 1

INTEGRAL CEA GROUP WORTH

BOL, 1st CYCLE, 532°F, 2250 PSIA

CEA GROUP 2

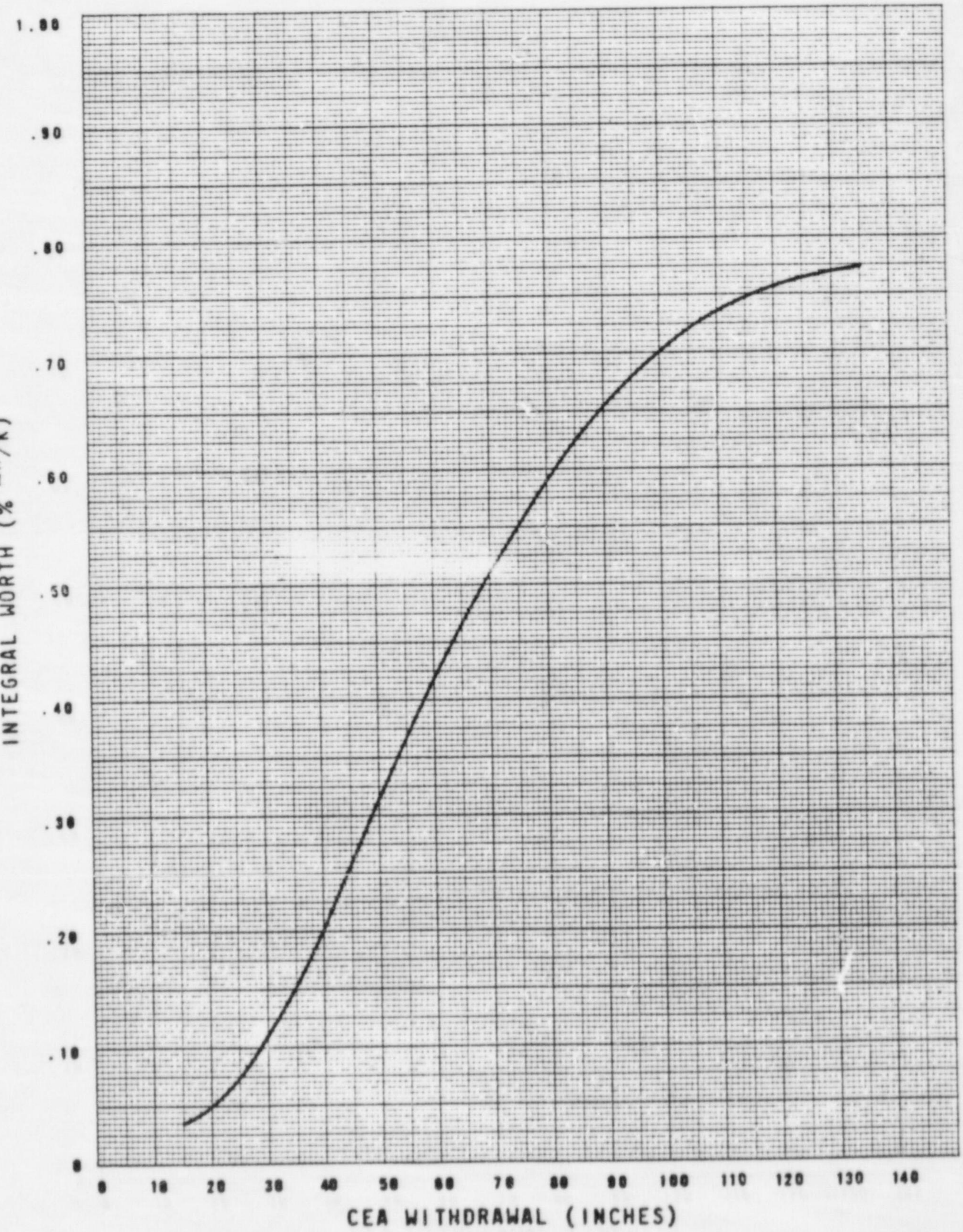


FIGURE 5.5-2

CALVERT CLIFFS UNIT 1

INTEGRAL CEA GROUP WORTH

BOL, 1st CYCLE, 532°F, 2250 PSIA

CEA GROUP 3

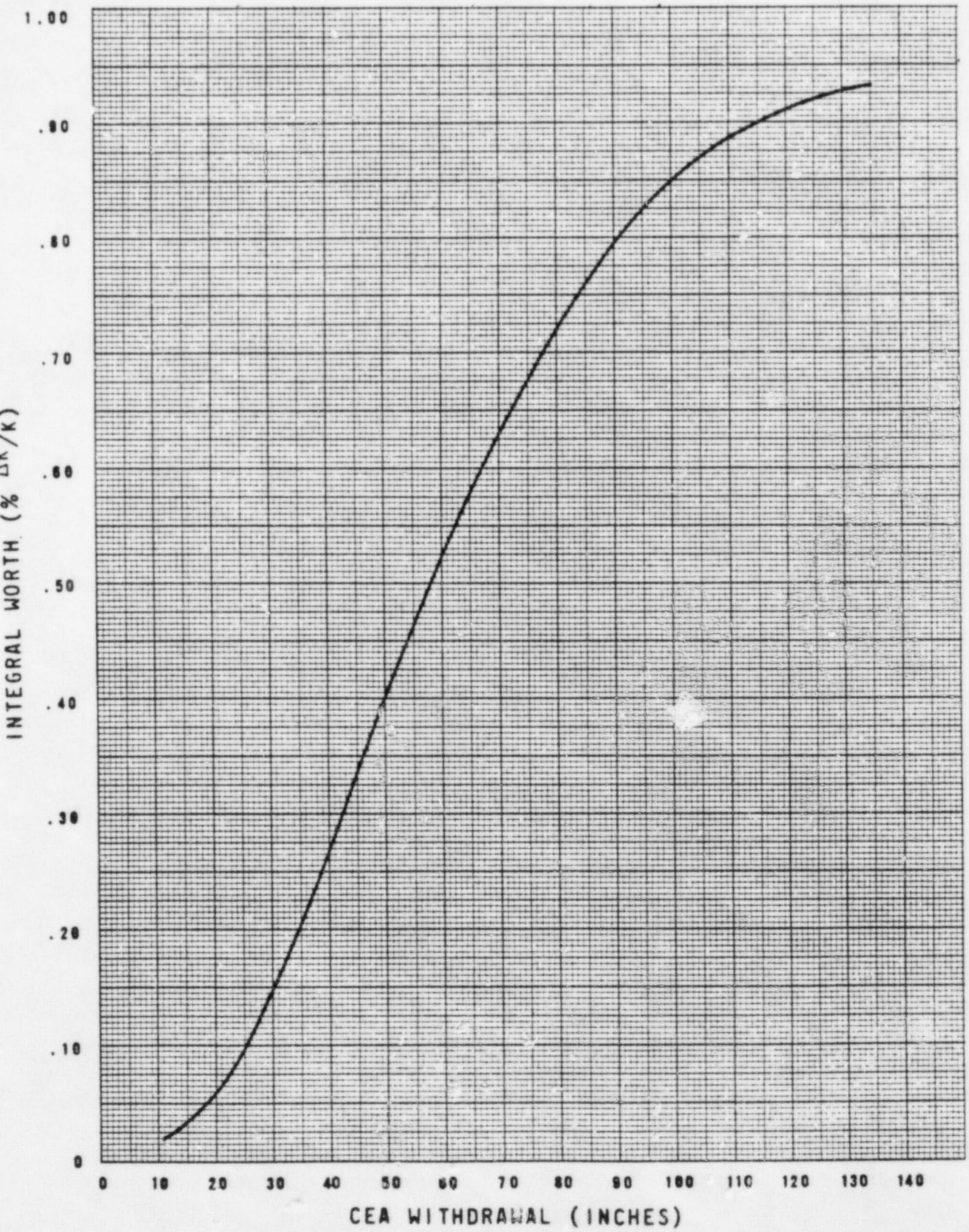


FIGURE 5.5-3

CALVERT CLIFFS UNIT 1

INTEGRAL CEA GROUP WORTH

BOL, 1st CYCLE, 532°F, 2250 PSIA

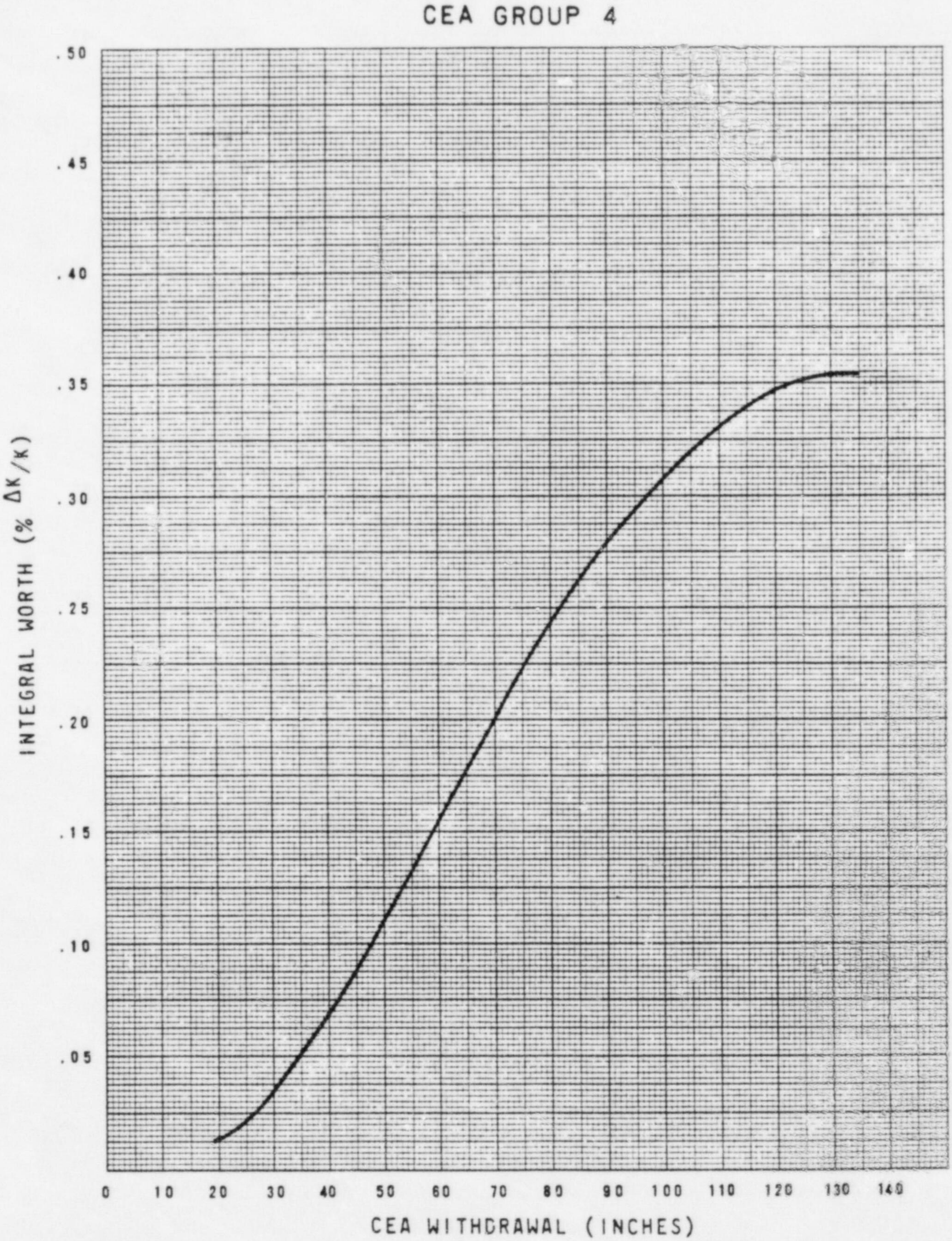


FIGURE 5.5-4

CALVERT CLIFFS UNIT 1

INTEGRAL CEA GROUP WORTH

BOL, 1st CYCLE, 532°F, 2250 PSIA

CEA GROUP 5

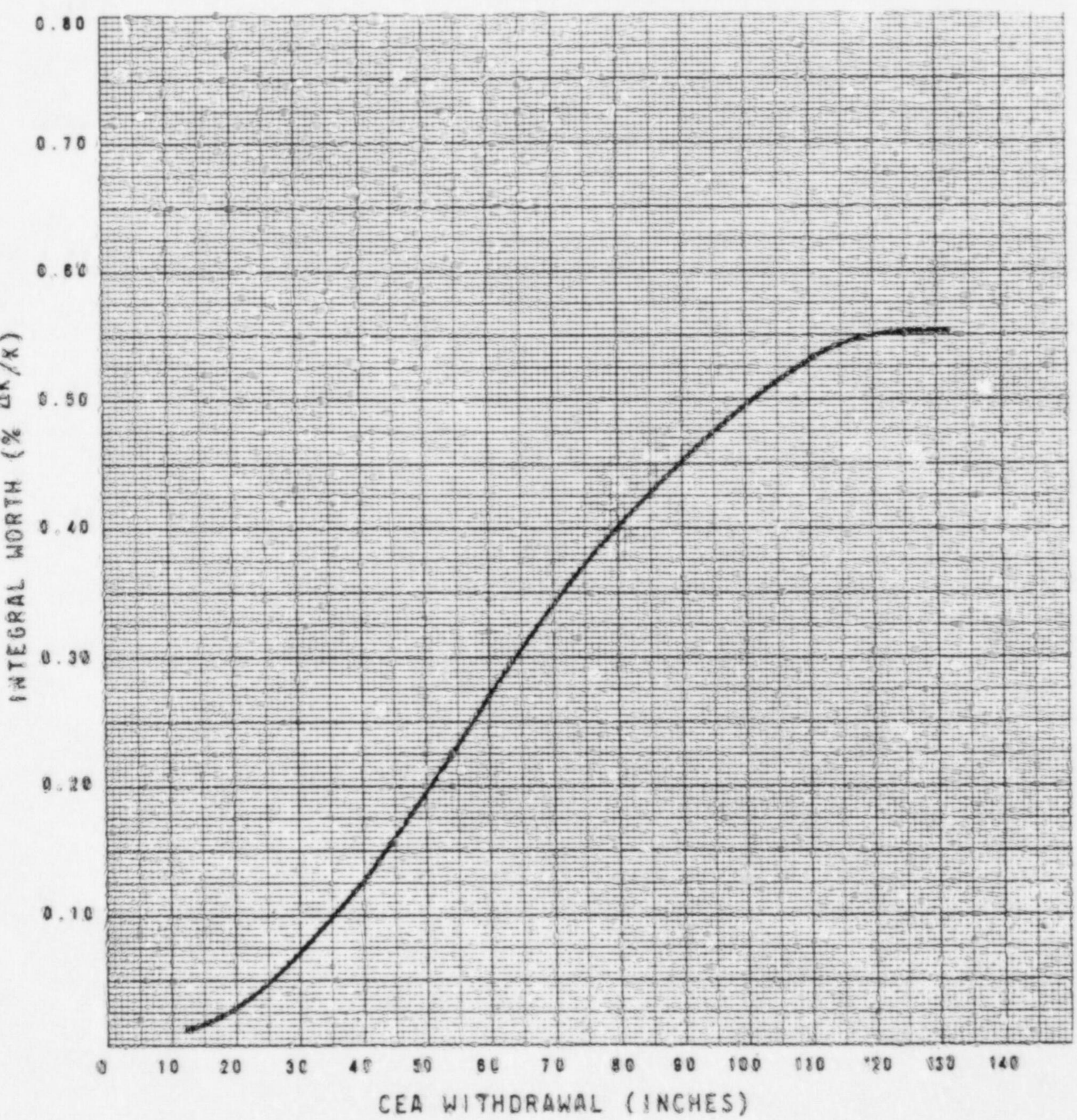


FIGURE 5.5-5

CALVERT CLIFFS UNIT 1

INTEGRAL CEA GROUP WORTH

BOL, 1st CYCLE, 532°F, 2250 PSIA

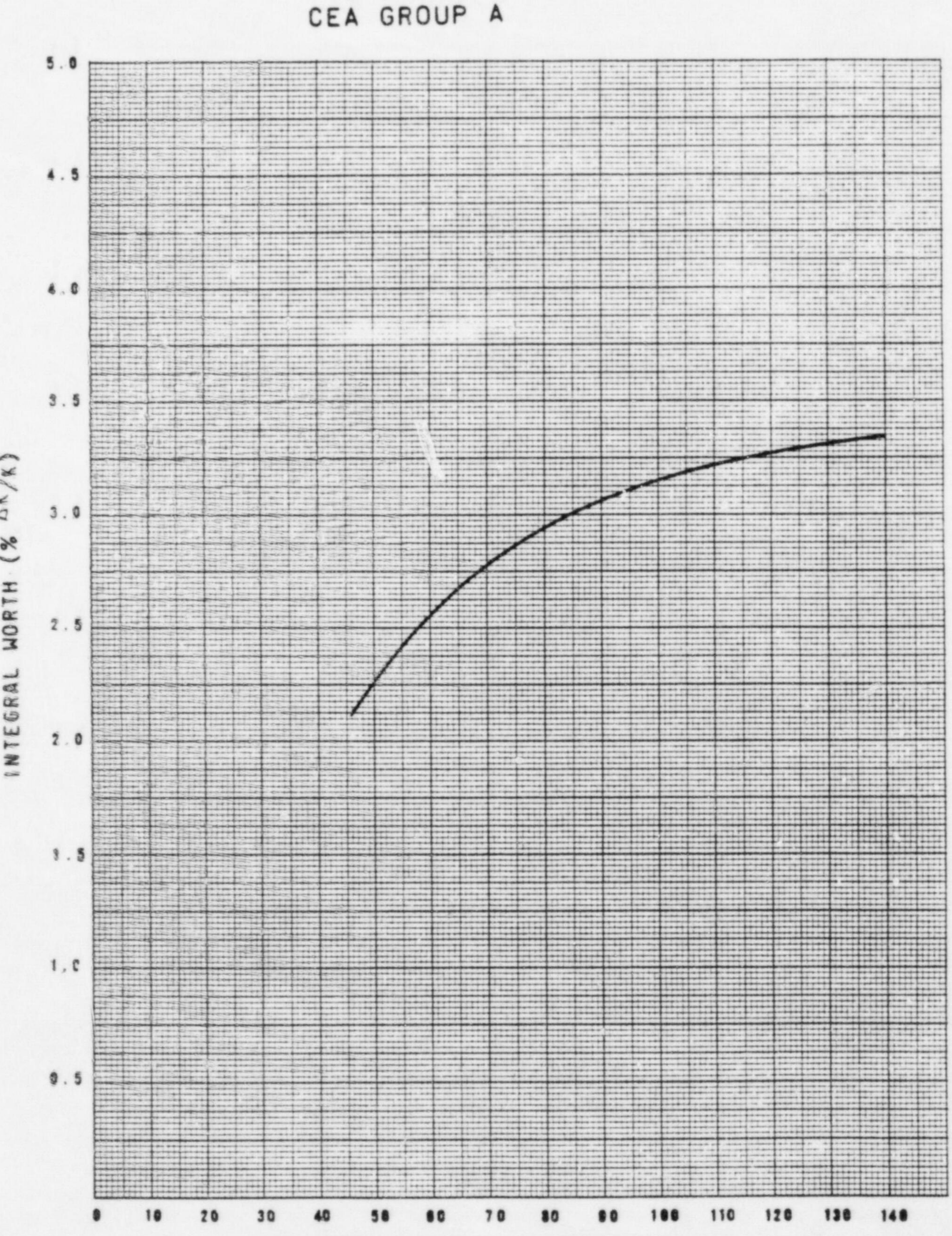


FIGURE 5.5-6

CALVERT CLIFFS UNIT 1

INTEGRAL CEA GROUP WORTH

BOL, 1st CYCLE, 532°F, 2250 PSIA

CEA GROUP B

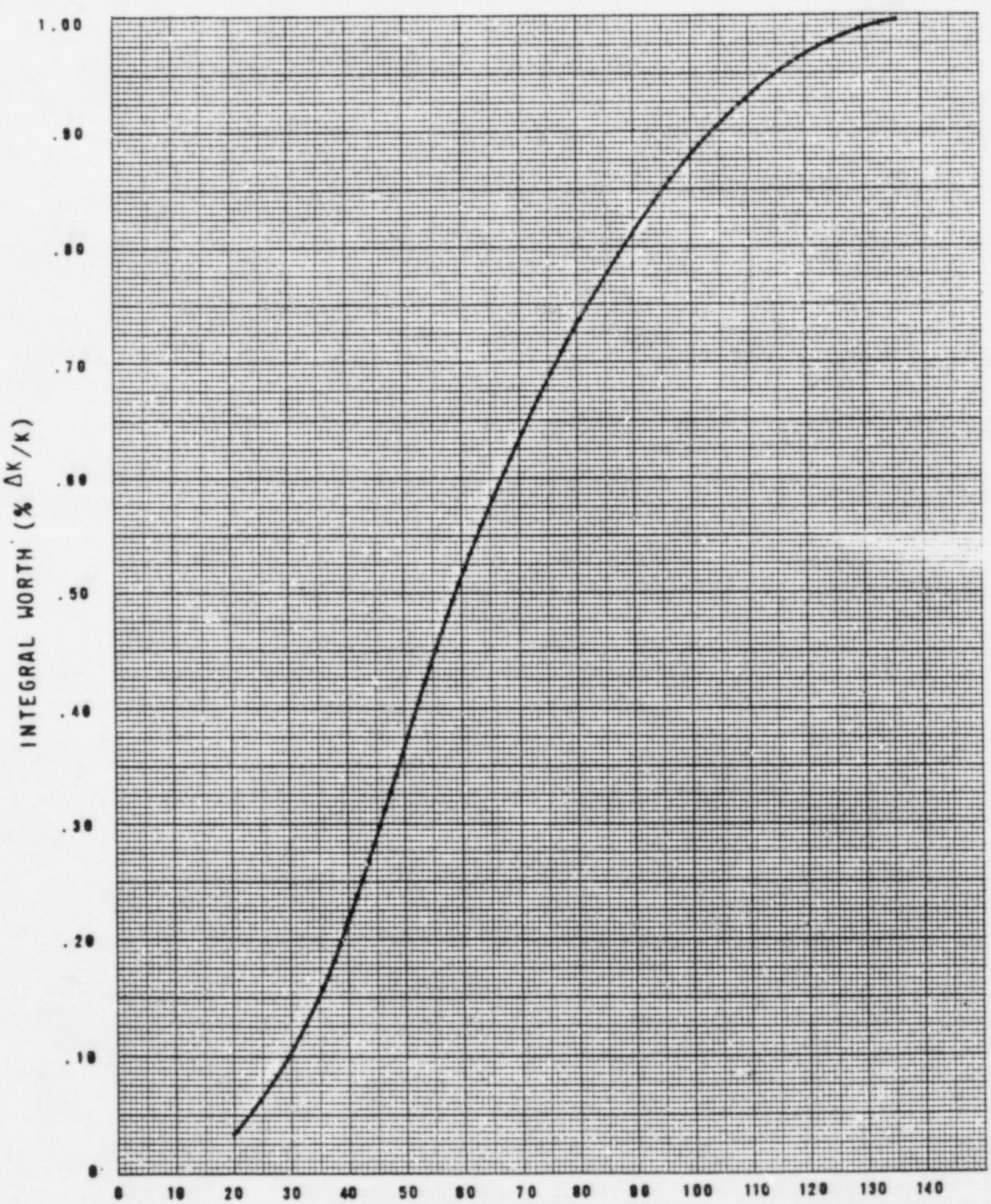


FIGURE 5.5-7

CALVERT CLIFFS UNIT 1

INTEGRAL CEA GROUP WORTH

BOL, 1st CYCLE, 532°F, 2250 PSIA

CEA GROUP C

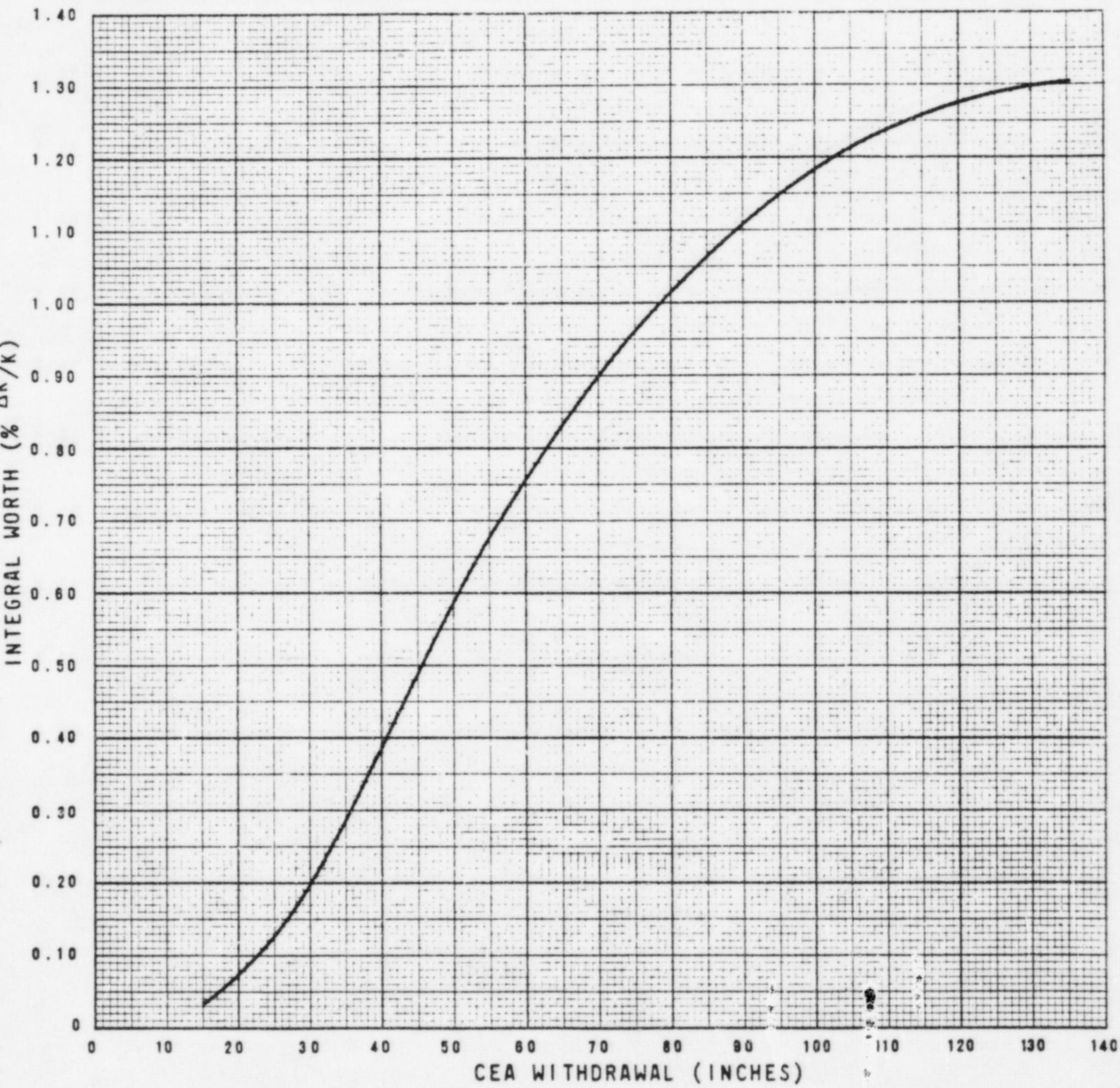


FIGURE 5.5-8

5.6 Overlapped Regulating CEA Group Worth Measurements

5.6.1 Purpose

Reactor Power level may be controlled by sequential insertion or withdrawal of Regulating Control Element Assemblies (CEA). Percent of overlap is selected so as to insure a relatively constant insertion rate of positive or negative reactivity over the full range of CEA Group movement. Technical Specifications allow CEA Group insertion as a function of reactor power level. Maximum insertion occurs at zero power and amounts to approximately 1.5%$\Delta k/k$ negative reactivity inserted due to CEA's. The integral reactivity worth curve for Regulating CEA Groups 5, 4 and 3 in an overlapped mode was measured. Maximum allowed insertion at zero power being at approximately 60 inches withdrawal on CEA Group 3.

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

5.6.2 Test Results

The overlapped integral reactivity worth of CEA Groups 5, 4, and 3 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 for CEA Groups 5, 4, and 3 from all rods out to less than 60 inches withdrawn on CEA Group 3.

5.6.3 Conclusions

The Overlapped Integral CEA Worth curve derived from this measurement has been adequate for use as an operational tool.

CALVERT CLIFFS UNIT 1

BOL, 1st CYCLE, 532°F, 2250 PSIA

OVERLAPPED CEA GROUPS 3, 4, & 5

ZERO POWER - PDIL TO FULLOUT

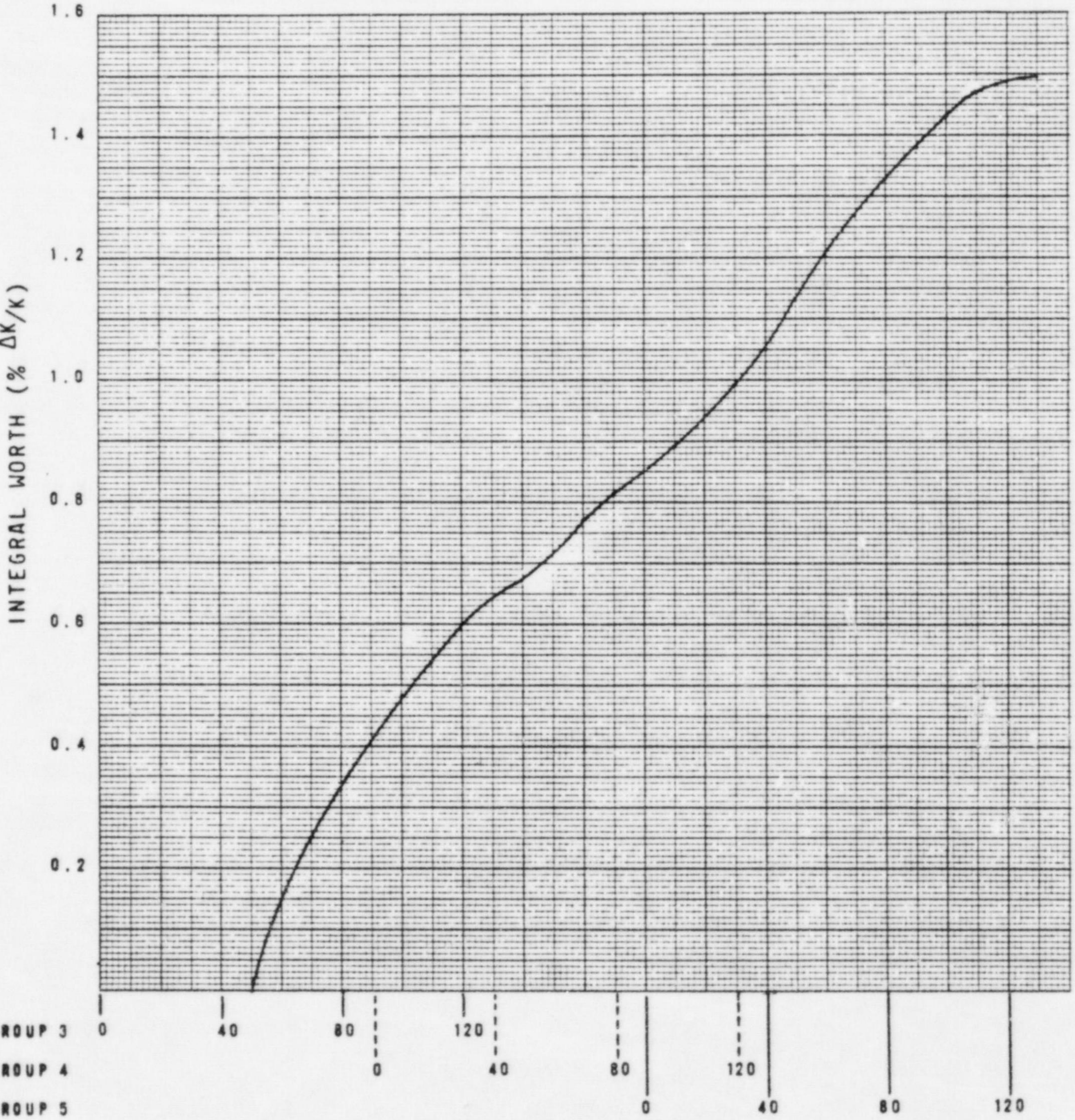


FIGURE 5.6-1

5.7 Pressure Coefficient of Reactivity Measurements

5.7.1 Purpose

The pressure coefficient of reactivity can be either negative or positive depending upon the magnitude of the Reactor Coolant System boron concentration. While small and relatively insignificant, the measured value of the pressure coefficient of reactivity is of interest.

5.7.2 Test Results

The pressure coefficient of reactivity was measured at two different Reactor Coolant System (RCS) temperatures over two different pressure ranges. The results are present in Table 5.7-1 below.

TABLE 5.7-1

MEASURED PRESSURE COEFFICIENTS OF REACTIVITY

NOMINAL RCS TEMPERATURE(°F)	RCS PRESSURE RANGE (psia)	PRESSURE COEFFICIENT OF REACTIVITY ($X10^{-7} \Delta k/k/psi$)
360	460 to 1100	-4.81
450	1100 to 2250	-5.03

5.7.3 Conclusions

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

5.8 Dropped CEA Worth Measurements

5.8.1 Purpose

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

5.8.2 Test Results

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

The most worthy dropped CEA from a full power CEA configuration was CEA B-8. Its measured integral reactivity worth is compared with predicted worth in Table 5.8-1 below.

TABLE 5.8-1

COMPARISON OF PREDICTED AND MEASURED DROPPED CEA INTEGRAL REACTIVITY WORTHS

NOMINAL RCS TEMPERATURE AND PRESSURE	REGULATING CEA GROUP POSITION (INCHES WITH-DRAWN)	MEASURED WORTH (B-8) (%Δk/k)	PREDICTED WORTH (GROUP B CEA) (%Δk/k)	ACCEPTANCE LIMITS (%Δk/k)
532°F, 2250 psia	1-4 @ ARO(1) 5 @ 110"	0.151	0.140	±0.050

(1) ARO ≡ All Rods Out

5.8.3 Conclusions

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

5.9 Ejected CEA Worth Measurements

5.9.1 Purpose

Technical Specifications state that the maximum reactivity worth of any one CEA in the core shall not be more than 0.38%Δk/k at full power at beginning of life. In addition, the maximum worth of any one CEA in the core shall not be greater than 1.06%Δk/k at hot zero power at beginning of life. A pseudo ejected CEA reactivity worth measurement was made under full power and zero power CEA configurations in order to verify safety analysis calculations related to the hypothetical CEA Ejection Incident.

5.9.2 Test Results

5.9.2.1 Full Power CEA Configuration. Under full power conditions, CEA Group 5 may be inserted to approximately 79 inches withdrawn. All other CEA Groups are at all rods out configuration. With respect to reactivity worth, CEA Group 5 consists of three (3) different types of CEA's. The integral reactivity worth of each of these three (3) CEA types with a predicted CEA worth typical of others of its same symmetrical core position was measured. The measurement technique was one of boration of the first CEA to full out followed by a swap between succeeding CEA's to measure integral reactivity worth of each CEA from approximately 79 inches withdrawn to full out. The most reactive CEA was 5-1. Its measured results are compared with predicted in Table 5.9-1.

5.9.2.2 Zero Power CEA Configuration. Under zero power conditions CEA Groups 5 and 4 may be inserted to full in with CEA Group 3 at approximately 60 inches withdrawn. All other CEA Groups are at All Rods Out configuration. Certain CEA's from all three (3) inserted CEA Groups were selected and their reactivity worths measured in accordance with the method outlined in Section 5.9.2.1 above. In this instant, there were five (5) different CEA types with respect to reactivity worth. The most reactive CEA was 5-36. Its measured results are compared with predicted in Table 5.9-1.

5.9.3 Conclusions

The measured values for both pseudo ejected CEA conditions are less than the predicted reactivity worths and in addition, are well within the acceptance criteria. Measured values are also approximately an order of magnitude less than those assumed in performance of the Safety Analyses.

TABLE 5.9-1

COMPARISON OF MEASURED AND PREDICTED PSEUDO EJECTED CEA REACTIVITY WORTHS

I. FULL POWER CEA CONFIGURATION

CEA NUMBER	NOMINAL RCS TEMPERATURE AND PRESSURE	REGULATING CEA GROUP POSITION (INCHES WITHDRAWN)	MEASURED WORTH (% Δk/k)	PREDICTED WORTH (% Δk/k)	ACCEPTANCE LIMIT (%Δk/k)
5-1	532°F, 2250 psia	1-4 @ ARO (1) 5 @ 79"	0.030	0.050	± 0.050

II. ZERO POWER CEA CONFIGURATION

CEA NUMBER	NOMINAL RCS TEMPERATURE AND PRESSURE	REGULATING CEA GROUP POSITION (INCHES WITHDRAWN)	MEASURED WORTH (% Δk/k)	PREDICTED WORTH (% Δk/k)	ACCEPTANCE LIMIT (%Δk/k)
5-36	532°F, 2250 psia	1-2 @ ARO (1) 3 @ 54" 4-5 @ 0"	0.114	0.120	± 0.050

(1) ARO ≡ All Rods Out

5.10 Stuck CEA Worth Measurement

5.10.1 Purpose

Technical Specifications state that available shutdown margin shall not be less than 2.4%Δk/k with the highest worth CEA stuck out whenever the reactor is critical. The reactivity worth of the most reactive stuck CEA was measured in order to verify the validity of predicted stuck CEA reactivity worths.

5.10.2 Test Results

A Group A CEA was predicted to be the most worthy stuck CEA. CEA's from Group A (A-44) and from Group B (B-9) were selected for measurement of stuck CEA reactivity worth. The measurement technique consisted of CEA Groups A and B combined trips with and without the "stuck" CEA stuck in the full out position. The reactivity data from these measurements was then used in combination with data from the CEA Group A reactivity worth measurement described in Section 5.5.2 to extrapolate an integral reactivity worth for CEA's A-44 and B-9. This preliminary on-site evaluation resulted in a measured maximum stuck CEA reactivity worth of no greater than 1.98%Δk/k. The worth of CEA B-9 being slightly greater than that for A-44.

A subsequent off-site analysis of the measured data by CE, Windsor has resulted in a final stuck CEA reactivity worth of 1.70%Δk/k.

5.10.3 Conclusions

The preliminary on-site analysis of measured data indicated that CEA B-9 was the most reactive "stuck" CEA. Its reactivity worth being within the bounds of the acceptance criteria of 1.58 $\pm$ 0.40%Δk/k. Subsequent off-site analysis of data by CE, Windsor has verified this on-site conclusion.

5.11 Part Length CEA Group Measurements

5.11.1 Purpose

The purpose of these measurements was to collect Part Length Control Element Assembly (PLR) Group reactivity worth data.

5.11.2 Test Results

The reactivity worth of PLR 1 (see Figure 5.0-2) was measured over its full travel from full out to full in for two different Regulating CEA Group configurations. A soluble boron swap with PLR 1 technique was used to generate reactivity worth versus CEA position information. Figure 5.11-1 displays the integral reactivity worth curve of PLR 1 for a 50% (Power Dependent Insertion Limit) PDIL Regulating CEA Configuration (i.e., CEA Group 5 @ 0 inches and Group 4 @ 76 inches). Figure 5.11-2 displays similar information for a Full PDIL Regulating CEA Configuration (i.e., CEA Group 5 @ 79 inches). The reactivity worths of PLR 2 and a combination of PLR 1 and PLR 2 referred to as PLR 8 were not measured.

5.11.3 Conclusions

Measured PLR 1 integral worth curve data was collected.

BOL, 1st CYCLE, 532°F, 2250 PSIA

INTEGRAL PARTLENGTH CEA GROUP WORTH

50% POWER - PDIL

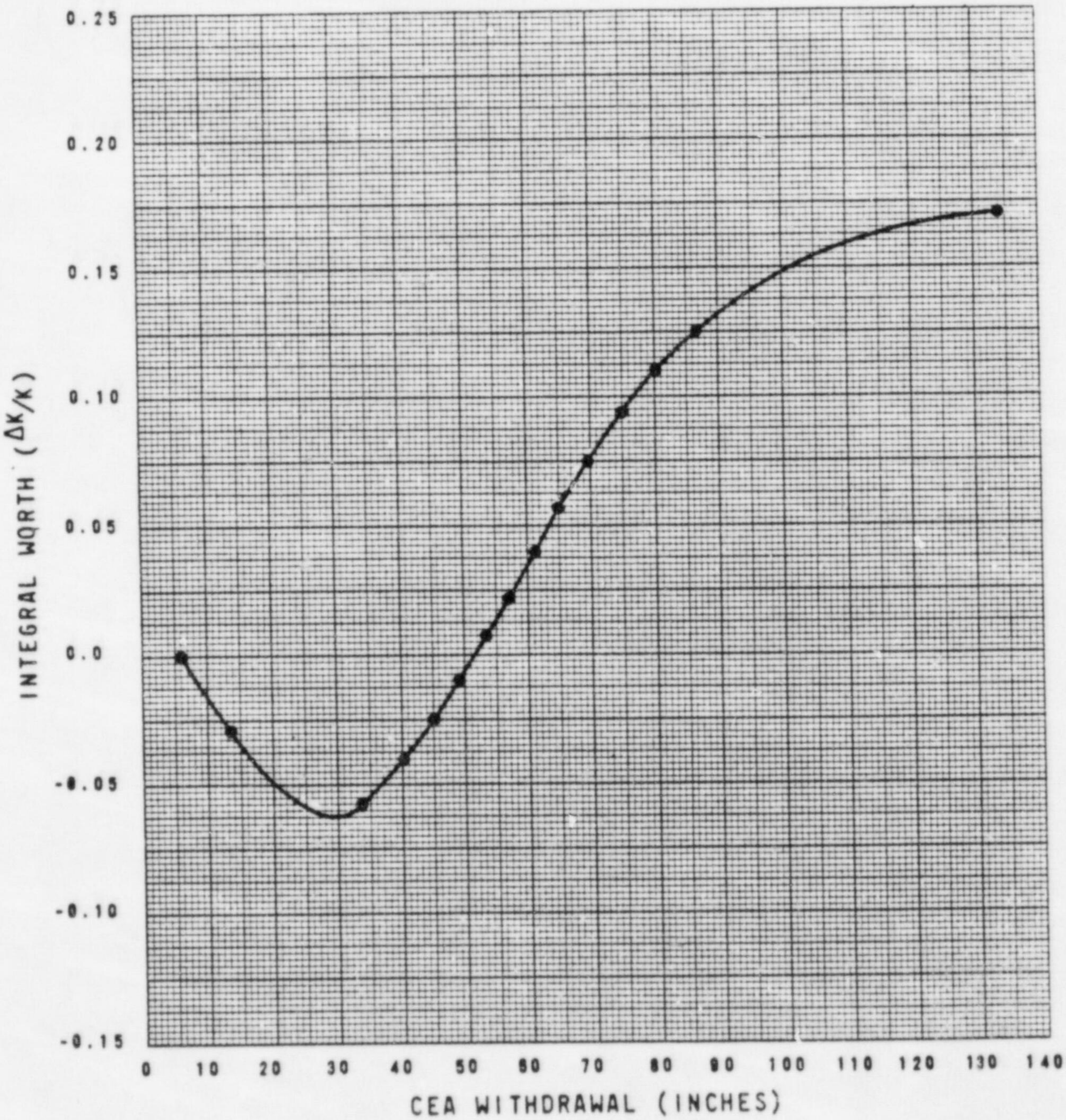


FIGURE 5.11-1

CALVERT CLIFFS UNIT 1

BOL, 1st CYCLE, 532°F, 2250 PSIA

INTEGRAL PARTLENGTH CEA GROUP WORTH

FULL POWER - PDIL

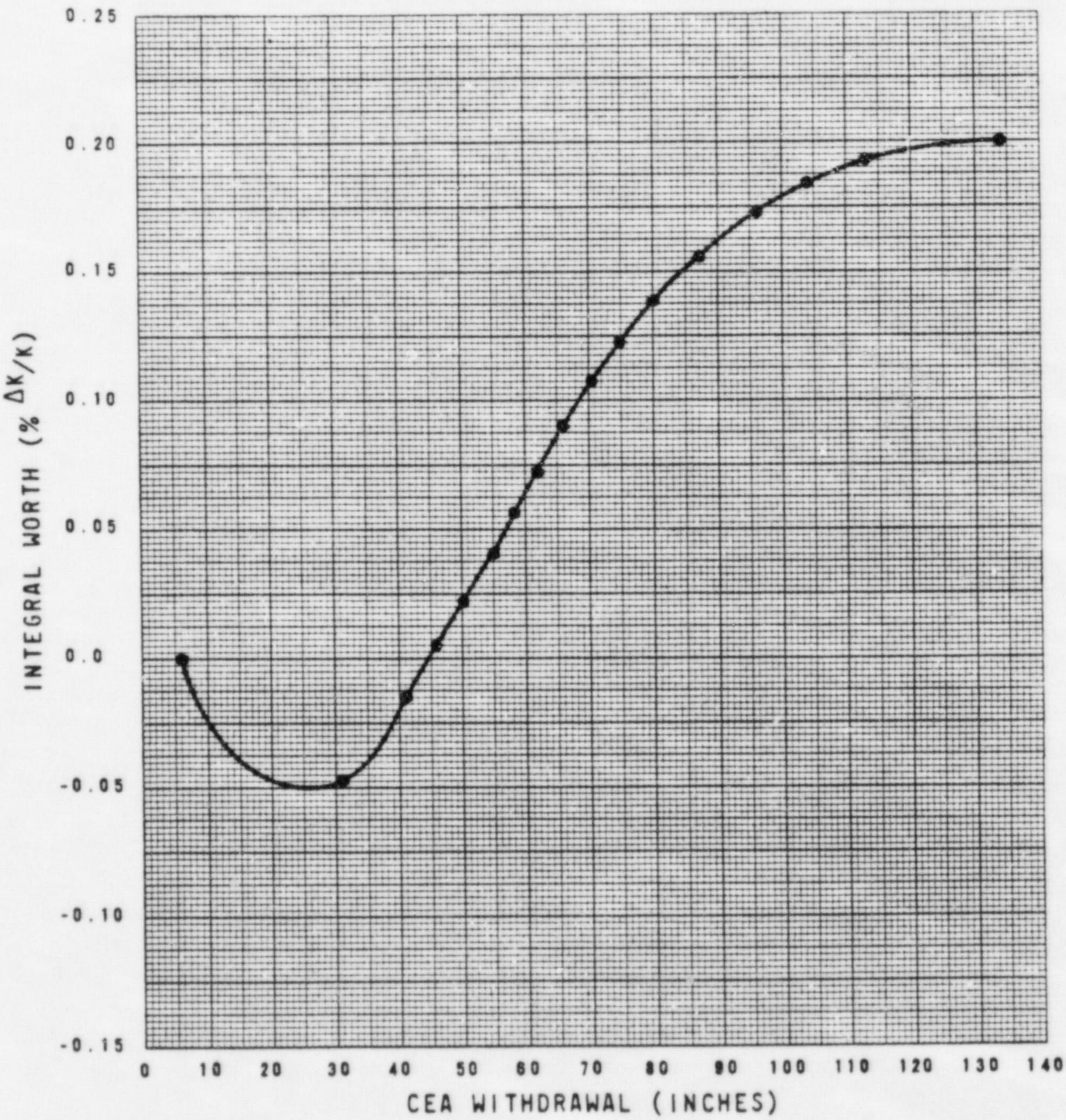


FIGURE 5.11-2

5.12 Critical Boron Concentration and Soluble Boron Worth Measurements

5.12.1 Purpose

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

5.12.2 Test Results

CEA Group integral reactivity worths were measured using a soluble boron swap technique. In addition, the soluble boron concentration at the end point of several of those CEA configurations was also measured. Soluble boron samples were independently analyzed by BG&E and by CE, Windsor. A comparison of measured with predicted critical boron concentrations for these several CEA configurations is presented in Table 5.12-1.

A relationship between reactivity change, CEA configuration, and critical boron concentration was developed over a range of Reactor Coolant System (RCS) boron concentrations. An average soluble boron worth was developed from that data and is compared with the predicted values in Table 5.12-2 below.

TABLE 5.12-2

COMPARISON OF MEASURED AND PREDICTED SOLUBLE BORON WORTH

NOMINAL RCS TEMPERATURE AND PRESSURE	RANGE OF RCS SOLUBLE BORON CONC. (1)		MEASURED SOLUBLE BORON WORTH (ppm/%Δk/k)		PREDICTED SOLUBLE BORON WORTH (ppm/%Δk/k)	ACCEPTANCE LIMITS (ppm/%Δk/k)
	BG&E	CE	BG&E	CE		
532°F, 2250 psia	628-1087	555-1102	77.3	77	83	± 15

(1) CEA configurations for BG&E and CE ranges are not the same.

5.12.3 Conclusions

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

TABLE 5.12-1

COMPARISON OF MEASURED AND PREDICTED CRITICAL BORON

CONCENTRATIONS FOR VARIOUS CEA CONFIGURATIONS

NOMINAL RCS TEMPERATURE AND PRESSURE	CEA GROUPS INSERTED (INCHES WITHDRAWN)	MEASURED CRITICAL BORON CONC. (ppm)		PREDICTED CRITICAL BORON CONC. (ppm)	ACCEPTANCE LIMITS (ppm)
		BG&E	CE		
260°F, 460 psia	ARO (1)	1048	1060	1062	± 100
260°F, 460 psia	5-4 @ 0"	1022	--	1023	± 100
532°F, 2250 psia	ARO (1)	1089	1102	1078	± 100
532°F, 2250 psia	5-3 @ 0"	949	--	929	± 100
532°F, 2250 psia	5-1 @ 0" C @ 0" B @ 120"	720	726	665 (2)	± 100

(1) ARO ≡ All Rods Out

(2) Predicted Critical Boron concentration for CEA Groups 5, 4, 3, 2, 1, and C fully inserted and CEA Groups A and B fully withdrawn.

5.13 Chemical and Radiochemical Tests

5.13.1 Purpose

Chemical and radiochemical analysis of the Reactor Coolant System (RCS) and the steam side of the steam generators can give clues as to the metallurgical condition of critical system components including fuel. The purpose of these tests was to determine baseline corrosion data, fission product activity levels and buildup, tramp uranium contributions to activity levels, and early detection of failed fuel.

5.13.2 Test Results

5.13.2.1 Baseline Corrosion Study. A number of crud samples were taken during Low Power Physics Test (LPPT). Most of these were taken as a result of investigations into the cause of the reactor vessel differential pressure situation. The crud samples were analyzed by emission spectroscopy and X-ray diffraction. Those results are summarized in Table 5.13-1.

The following observations were made:

(1) Weight of particulates (ppm) - In most cases all crud samples were less than .01 ppm insoluble.

(2) Microscopic Examination of Particulates - Specific examination of crud samples collected 10/23/74 and 11/5/74 showed greater than 99% crud-type material, and less than 1% fibers and resin fines.

(3) Soluble corrosion products - During LPPT, the following maximum and minimum values were observed:

	Max	Min
Fe, ppm	.17	less than .01
Cr, ppm	less than .02	less than .02
Ni, ppm	.34	less than .04
Cu, ppm	.04	less than .01

5.13.2.2 Fission and Activiation Product Buildup Study. Iodine - 133, Cs - 137 and Xe - 135 were the only fission products positively identified during LPPT. The maximum and minimum values of the following were observed:

	Max	Min
Xe-135, μci/ml	3.07×10^{-7}	3.07×10^{-7}
I-133, μci/ml	2.84×10^{-7}	2.05×10^{-8}
Cs-137, μci/ml	2.31×10^{-8}	2.01×10^{-8}
Gross Beta Gamma, μci/ml	1.40×10^{-4}	1.25×10^{-7}
Total activity, μci/ml	1.10×10^{-4}	4.73×10^{-7}

It is felt that the source of Cs - 137 may be from the contaminated equipment supplied by CE for the core inspection rather than from the fuel. The reactor was not critical when Cs - 137 was first observed.

Other activation products identified in liquid or crud samples include:

F-18	Cr-51	Zr-95
Na-24	Ni-65	Zr-97
Mn-56	Np-239	Co-60
Co-58	Mo-99	Cu-64
Ar-41	W-187	

Generally Mn-56 and Ar-41 were the largest contributors to total activity during LPPT in liquid samples. In crud samples Mn-56, W-187 and Co-58 were generally the largest contributors to crud activity.

Since only sporadic occurrences of I-131 were observed in liquid samples, no effort was made to determine an I-131/I-133 ratio value. The generally low activities observed and poor counting statistics obtained would make any ratio inaccurate at best.

5.13.2.3 Lithium Buildup. During LPPT lithium buildup was not appreciable. Typically, Li was less than .1 ppm throughout LPPT when not added deliberately. Lithium was added to the RCS on 11/9/74 to bring RCS concentration to 2.0 ppm for pH control. CE recommended the higher lithium concentration as a change in chemistry control of the RCS.

5.13.2.4 Purification System D.F.

(1) Crud samples from filter inlet and outlet were collected when crud levels were greater than .01 ppm. The D.F. was approximately 5. Crud levels were typically less than .01 ppm from filter inlet and outlet, and it was not possible to obtain really meaningful filter D.F. data during LPPT samples.

(2) Due to the limited operation of the ion exchangers, sample analysis of Ni, Fe, and Cr on effluent were not performed.

(3) During LPPT, activity levels in the RCS were near minimum detectable. Due to the minimum activity and only limited operation of the Chemical and Volume Control System (CVCS) ion exchangers, gamma scans and gross beta-gamma analysis were not taken. Any data taken under these conditions could not have been considered meaningful.

5.13.2.5 Volume Control Tank (VCT) Gas. Due to the fact that fission gases were at or below their minimum detectable activity, no buildup of fission gases were observed in the VCT and gamma scans were not conducted weekly. Samples of reactor coolant collected during LPPT showed no fission gas.

5.13.3 Radiochemistry Results during Evaluation and Resolution of Reactor Vessel Pressure Differential Increase Incident. During the conduct of LPPT, a relatively rapid increase in RCS pressure drop across the reactor vessel was noted (see Section 7.0). As a result of that observation and its aftermath, the reactor vessel was disassembled and reassembled twice. Additional radiochemistry evolutions were required during this period and their results are summarized as follows:

5.13.3.1 pH Increase Tests.

(1) Ammonia addition. At 1345 on 11/8/74 approximately 3.25 gallons of a 28% NH_3 solution was added to the RCS to raise NH_3 concentration. pH rose to about 7.3, conductivity to about 130 ppm and ammonia to about 25 ppm. Suspended solids collected every 15 minutes during the test all showed less than 10 ppb.

(2) Lithium addition. At 1630 on 11/9/74 approximately 3 kilograms of $^7Li\ OH\text{-}H_2O$ was added to the RCS. This was done in order to raise the high temperature pH about 1 pH unit into the alkaline region. Room temperature pH was essentially unaffected by the addition. Conductivity showed a slight increase and lithium reached 2.0 ppm. Suspended solids collected every 15 minutes during the test all showed less than 10 ppb.

5.13.3.2 pH Decrease Tests.

Following a preservice rinse to the Waste Processing System (WPS) to insure that chlorides would not be expelled from the resin, Purification Ion Exchanger 12 was placed in service at 1710 on 11/10/74. Lithium was removed from the RCS at a rate dependent on the purification half-life. pH showed a slow decrease to that of boric acid alone. By the end of the test, following shut-down and feed and bleed operations, pH fell to about 5.7. This pH was about the same as the pH before testing began. Suspended solids collected every 15 minutes continued to show less than 10 ppb.

5.13.3.3 Startup Data Following Reactor Vessel Differential Pressure Investigation.

Prior to hydrazine addition, suspended solids were measured routinely to be about 70 ppb with a high value measured at 125 ppb. Following the addition of hydrazine, suspended solids fell to less than 10 ppb and remained there with the exception of one sample at 50 ppb during the subsequent heatup. Oxygen levels dropped to within specification after 12 hours following start of hydrazine addition.

Fe and Ni were analyzed routinely. Maximum and minimum values were:

	Max	Min
Fe, ppm	.080	.020
Ni, ppm	.335	.040

5.13.3.4 Subsequent Shutdown Information.

Shortly after criticality, it became apparent that several of the CEA's were uncoupled and a shutdown was inevitable. Since H_2 overpressure had been established, the shutdown involved degassing the RCS to 5 cc/Kg. The H_2 supply to the VCT was isolated and to hasten the degassing process the pressurizer was vented through the NSSS sample sink. The RCS H_2 concentration showed only gradual decrease (VCT vapor still about 60% H_2) until the VCT was purged with N_2 and H_2 concentration in VCT vapor decreased to about 9% H_2. At 1315 on 12/12/74 the RCS H_2 was found to be 4.8 cc/Kg allowing the RCS to be opened to atmosphere.

5.13.3.5 Subsequent Startup Information.

The startup to complete LPPT was uneventful. It should be noted that 2.2 Kg of Li $OH\text{-}H_2O$ was added to the RCS with a corresponding rise in pH (@ 25°C) to about 6.7.

5.13.4 C clusions

Chemical and radiochemical data was reviewed throughout LPPT and all results were either acceptable or, when appropriate, corrective action was instituted to achieve acceptable conditions. Little meaningful data was obtained on the performance of CVCS ion exchangers and filters due to infrequent operation of the ion exchangers and low levels of suspended solids and radioactivity.

Lithium exceeded its acceptance limits due to intentional additions of ^7Li for pH control. The acceptance limit had been previously changed to allow the RCS to operate on ^7Li pH control. The new limit being 2 ppm instead of 0.5 ppm.

TABLE 5.13-1

SPECIAL CRUD SAMPLE RESULTS

DATE	TIME	TYPE OF SAMPLE	ANALYSIS RESULTS AND REMARKS
11/5	2330	50-liter crud	46 ppb - microscopic exam showed greater than 99% crud
11/6	1531	100-liter crud	less than 10 ppb - first showed significant Cu - 64 on gamma scan
11/7	0430	462-liter crud	less than 10 ppb
			Fe, Ni, Cu, Pb (major)
			B, Zn, Cu (minor)
11/8	0700	577.5 liter crud	less than 10 ppb
11/9	1000	630-liter crud	major - Cu
			large - Fe, Ni, Na (questionable)
			minor - Ci, Pb, Ca, Mg, Al, Ag, Ti, Si, B
11/11	2035	30-liter crud	less than 10 ppb - CE analysis
		(2 filter papers)	

Metal	micro grams/30 liter	Metal	micro grams/30 liter
Fe	2.4 (1)	Zn	1.2
	1.1 (2)		1.0
Cu	4.7 (1)	Pb	26.4
	3.9 (2)		less than 2.0

(1) Top

(2) Bottom

TABLE 5.13-1 (cont'd)

DATE	TIME	TYPE OF SAMPLE	ANALYSIS RESULTS AND REMARKS
11/11	2035	continued	<u>Metal</u> <u>micro grams/30 liter</u> <u>Metal</u> <u>micro grams/30 liter</u>
			Ni non-detectable (1) Ci non-detectable (1)
			non-detectable (2) non-detectable (2)
			Al not completed
			CE remarks - unusual color would expect gray to black;
			Cu and Pb very unusual
11/13	1800	#11 Purifi. Filter Deposits	1. Yellowish soft wax - grease
			2. white material - 2 types
			<u>elastic</u> - organic polymer
			<u>particulate</u> - titanium dioxide (TiO_2)
			3. black particulate
			Zr - 60-70%
			Ni, Ci, Fe - less than 30% total
			Ti - about 5%
			SiO_2 - less than 1%
			Cu - trace
			Visual inspection of filter indicated greater than 99% black particulate. The grease and white material were scattered at random.

(1) Top
(2) Bottom

TABLE 5.13-1 (cont'd)

DATE	TIME	TYPE OF SAMPLE	ANALYSIS RESULTS AND REMARKS
11/15	0900	Residue from hot leg sample point	Sample collected from drain line off of hot leg sample line upstream of pneumatic isolation valve in containment - 28 mg collected. Analysis Zr (major) Fe, Cr, Ni (large)
11/19	Unknown	Swipes from upper guide structure	Fe_3O_4 - Fe_2O_3 50-50% Fe (major) Ni, Ti, Zr, Cr (large) Cu (trace)
11/20	0000	Swipes from fuel assembly	End fitting - Cr (major) Fe, Ni, Ti (large) Zr (minor) Cu (trace) Fuel pins - Zr (major) Fe, Ni, Ti, Cr (large)

6.0 ESCALATION TO POWER TESTS (EPT)

The Escalation to Power Tests were conducted to determine as-built plant characteristics during steady state and transient operations from 0% to 100% power and to demonstrate, with reasonable assurance, that the plant is capable of withstanding the accidents and transients analyzed in the FSAR. Tests requiring steady state power were performed at major plateau's of 20%, 50%, 80% and 100% power. Several minor tests were performed at 30%, 40%, 60%,70% and 90% power.

During EPT, two unusual events occurred which interrupted or delayed testing until they were temporarily or permanently resolved. During the 20% power test plateau, radiation measurements taken outside Unit 1 containment structure equipment hatch indicated that dose rates would exceed the 0.5 mrem/hr limitation specified in the FSAR for the outside surface of the containment structure. In addition, higher than anticipated radiation levels were indicated at a number of locations inside the containment, where access on an infrequent basis is required during operation. Analysis indicated that the high radiation levels were primarily coming from the annulus between the reactor vessel flange and the primary shield wall. Temporary shielding was installed over the annulus and outside the equipment hatch to reduce radiation levels. A detailed discussion of this problem is given in Section 7.3. Late in EPT, an unplanned trip from 100% power occurred which resulted in uncovering of the feedring in both steam generators. During the refilling of the steam generators, a water hammer occurred in the main feedwater piping resulting in damage to the feedwater regulating valves. Subsequently, operating procedures were revised, such that, between -85 inches and -30 inches, auxiliary feedwater is utilized to restore water level at a maximum rate of 1.2 inches per minute. This casualty resulted in a seven (7) day delay in the test program. A more detailed description of the water hammer incident is included in Section 7.4.

Three EPT program tests covered below were deferred with approval of the Nuclear Regulatory Commission.

(1) A test of axial xenon oscillation dampening using Part Length CEA's (PLR) was deferred until such time as a decision is made to use PLR's. Later in first cycle life, an axial xenon oscillation dampening test will be performed using full length CEA's.

(2) A part loop operation power distribution measurement was deferred until such time as a decision is made to operate at power using less than four (4) Reactor Coolant Pumps (RCP). Administrative restrictions prevent power operation with less than full RCS flow (4 RCP's). In addition, the Reactor Protective System (RPS) will cause a reactor trip whenever reactor power is greater than 10^{-4}% and less than 4 RCP's are operating.

(3) A performance test of the automatic CEA control features of the Reactor Regulating System (RRS) has been deferred until such time as a decision to use this feature is made. The principal reason

for deferring the test was to decrease the possibility of fuel failures which could be exacerbated by relatively rapid and near continuous CEA motion, a characteristic of automatic CEA control.

Administrative restrictions prevent part loop operation and use of automatic CEA control and PLR's until such time as tests are performed and results reviewed and approved by the POSRC.

The off-site analysis by CE, Windsor of two (2) tests, Pseudo Ejected CEA and Dropped CEA continues. A preliminary on-site analysis of the results of the Psuedo Ejected CEA test indicated that they were well within acceptance criteria. A detailed on-site evaluation could not be accomplished as the In-Core Analysis (INCA) program is not capable of analyzing asymmetric power distributions.

6.1 Turbine Generator Startup and Atmospheric Dump/Turbine Bypass Valves Test

6.1.1 Purpose

The test was designed to accomplish several objectives:

(1) To verify the proper operation of the turbine bypass/atmospheric dump valves control system;

(2) To verify the proper operation of turbine controls, generator controls and support systems;

(3) To bring the turbine generator to about 10% power; and

(4) To transfer feedwater control to automatic.

6.1.2 Test Results

The test of the turbine bypass/atmospheric dump valves was conducted concurrently with preparations for startup of the turbine generator. The turbine bypass/atmospheric dump valves were used to maintain Reactor Coolant System (RCS) temperature at 532°F. The steam dump valves operated normally during this test. All the turbine bypass valves were mechanically limited to about half (1/2) stroke. 1-CV-3944 and 1-CV-3946 were isolated from their automatic signal and operated in manual local upon determining that bypass control became unstable and flow through these valves insignificant when operated in automatic or manual from the control room. Evaluation of the turbine bypass problems continued throughout EPT. In addition to instability problems, the valves would leak past the seat and required manual isolation to prevent unwanted cooldown. Evaluation has indicated that a possible solution will be the installation of hydraulic dampeners on all valve operators and installing new plugs made of 420 C steel. The hydraulic dampeners would increase the stiffness of the valve to operate and dampen out valve motion in the unstable region. The original plugs were 440 C material which was brittle and was fracturing. The change to the more ductile 420 C material plugs should eliminate the valve cracking problem. The corrections for the turbine bypass valve were only partially installed by the end of the Startup Test Program.

Testing of the turbine generator control circuits prior to initial operation consisted of the completion of Surveillance Test Procedures. These procedures covered the following:

(1) Test of opening and closing circuits on main stop valves, combined intercept valves and bleeder trip test valves;

(2) Checking the phase angle of vibration;

(3) Hydraulic Thrust Wear Detector test;

(4) Master Trip Solenoid Valve test;

(5) Backup Speed Control Amplifier test;

(6) Automatic start of emergency bearing oil pump;

(7) Automatic start of hydraulic pump (EHC hydraulic power unit); and

(8) Oil Trip test.

No major problems were detected during the testing of turbine generator controls.

6.1.3 Conclusions

No major problems were encountered during startup of the turbine generator. Problems with the turbine bypass valves continued throughout FPT. Corrective measures are partially complete. Final evaluation of the problems will be completed after testing the valves upon completion of the maintenance activity.

6.2 Reactivity Coefficient Measurements

6.2.1 Purpose

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

6.2.2 Test Results

Variable T avg Tests were conducted at each major power plateau (20%, 50%, 80% and 100%) with the Control Element Assemblies (CEA) inserted to approximately 100 inches on CEA Group 5. The test was conducted with the Power Coefficient and the Isothermal Temperature Coefficient (ITC) as separate tests. During the ITC test, ΔT power was held constant and Reactor Coolant System (RCS) T cold was varied. T cold was decreased 10°F below original temperature, conditions stabilized, data recorded and temperature increased to 5°F above original temperature, conditions stabilized, and data recorded. This cycle was repeated twice with the exception that T cold was returned to the original temperature at the end of the final cycle.

The Power Coefficient Test was conducted by holding T cold constant and decreasing gross electrical power by approximately 34 MWe at a rate of 1%/min, conditions stabilized, data recorded. Then gross electrical power was increased to the original power level at 1%/min, conditions stabilized and data recorded. This cycle was repeated two (2) more times. The final power coefficient and ITC values were the average value of the runs conducted. The measured and predicted values for the temperature and power coefficient for each power plateau are shown in Table 6.2-1.

6.2.3 Conclusions

The measured values for Isothermal Temperature Coefficient compared well with predicted values. The Power Coefficient was predicted to decrease in value with increasing power level. It was found that the measured Power Coefficient was reasonably constant with increasing power. However, results were all within acceptance limits.

TABLE 6.2-1

COMPARISON OF MEASURED AND PREDICTED ISOTHERMAL TEMPERATURE AND POWER COFFICIENTS

NOMINAL REACTOR POWER (%)	ISOTHERMAL TEMPERATURE COEFFICIENT ($\Delta k/k/^{\circ}F$)		POWER COEFFICIENT ($\Delta k/k/\%$ power)	
	PREDICTED	MEASURED	PREDICTED	MEASURED
20	$+0.02 \pm 0.5x10^{-4}$	$+0.05x10^{-4}$	$-1.99 \pm 1.0x10^{-4}$	$-1.34x10^{-4}$
50	$-0.17 \pm 0.5x10^{-4}$	$-0.11x10^{-4}$	$-1.42 \pm 1.0x10^{-4}$	$-1.04x10^{-4}$
80	$-0.28 \pm 0.5x10^{-4}$	$-0.18x10^{-4}$	$-1.10 \pm 1.0x10^{-4}$	$-1.04x10^{-4}$
100	$-0.37 \pm 0.5x10^{-4}$	$-0.21x10^{-4}$	$-1.02 \pm 1.0x10^{-4}$	$-0.99x10^{-4}$

6.3 Plant Power Calibration

6.3.1 Purpose

The purpose of the test was to:

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

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

(3) Perform when necessary a calibration of the Safety and/or Control Power Range Channels followed by a calibration check using a current source standard.

6.3.2 Test Results

Secondary Calorimetrics using hand caluclations were conducted at the 20, 50 and 80% test plateaus. The calorimetrics were used to calibrate nuclear instrumentation and to verify the plant computer core thermal power calculations. The Power Range Safety Channels and ΔT Power Reference Calculators were adjusted to agree within 0.5% of the Secondary Calorimetric calculations. These adjustments were performed at the 20, 40, 50, 70, 80, 90 and 100% power test plateaus.

Initial calibration of the Power Range Safety Channels was conducted during the 20% test plateau using the Keithley pico-ammeters as a standard for subchannel calibration. Adjustment of both the Power Range Safety and Control Channels was completed at the 50, 80 and 100% test plateaus.

6.3.3 Conclusions

Hand calculations of core thermal power pointed out some minor discrepancies in the computer calculations. After correction of the deficiences, computer calculations proved to be reliable and accurate. Calibration of the Power Range Safety and Control Subchannels was accomplished acceptably at each major test plateau. The intent of the Ex-Core Nuclear Instrument Calibration was to adjust nuclear power, ΔT power and the calorimetric to within 0.5% of each other. Due to noise on the ΔT power channels, nuclear power and ΔT power did vary by 0.7% while both were within 0.5% of the calorimetric. An investigation to determine the source of noise on the ΔT power channels and eliminate it is under way. Plant operating procedures contain instructions for hand calculations in case of computer failure and/or to verify computer calculations.

6.4 Shielding Effectiveness and Plant Radiation Levels

6.4.1 Purpose

The test was conducted to accomplish the following four objectives:

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

(2) Evaluate the adequacy of plant radiation shielding.

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

(4) Determine radioactivity buildup in specified piping and components.

6.4.2 Test Results

Radiation surveys were conducted prior to initial criticality and at 0%, 50%, 80% and 100% power. After initial criticality, all measurements taken in and adjacent to the containment structure included neutron and gamma readings. During the initial phase of the power range testing program, radiation measurements taken outside Unit 1 containment equipment and personnel hatches indicated that dose rates would exceed the 0.5 mrem/hr limitation specified in the acceptance criteria for the outside surface of the containment structure. In addition, higher than anticipated radiation levels were indicated at a number of locations inside the containment, where access on an infrequent basis is required during operation.

Analysis of the measured radiation levels and a review of the existing shield arrangement indicated that the high radiation levels were caused by neutron and gamma streaming out of: (1) the annulus between the reactor vessel flange and the primary shield wall; (2) the annulus around the Reactor Coolant System (RCS) piping where it penetrates the primary shield wall and (3) to a lesser extent, an access opening through the lower part of the primary shield.

Neutron and gamma dose rates were measured at 20% power and extrapolated to 100% power. Neutron dose rates at the 69' elevation ranged from 1000 mrem/hr to more than 25000 mrem/hr; gamma dose rates ranged from 250 mrem/hr to 10000 mrem/hr. The highest dose rates occurred at the north end of the refueling pool and over 12B Reactor Coolant Pump (RCP) removal hatch. At the 45' elevation, neutron dose rates ranged from 75 to 1500 mrem/hr. Gamma dose rates ranged from 50 to 300 mrem/hr. The dose rates outside the equipment hatch were 400 mrem/hr from neutrons and 150 mrem/hr from gammas. At the 10' elevation, dose rates of 5000 mrem/hr from neutrons and 3000 mrem/hr from gamma were found.

To reduce the radiation levels, temporary shielding was installed in the following locations:

(1) Temporary neutron shielding consisting of bagged crystalline boric acid and polyethylene sheets were stacked on support grating specially erected to span the annular gap between the reactor vessel flange and the primary shield.

(2) A shield consisting of concrete blocks was fitted inside the access hole located at the bottom of the primary shield to a thickness of 48 inches.

(3) Polyethylene shielding, approximately 12 inches thick was added to RCP hatches 11A, 11B and 12A. Two layers, approximately 12 inches of boric acid bags were placed on 12B hatch.

(4) A 24-inch concrete block shield was installed outside of equipment hatch between the hatch and the rolling metal door reduced the total gamma and neutron dose rate to less than 0.5 mrem/hr outside the equipment hatch.

(5) Shielding was also installed outside the emergency personnel hatch.

After shielding installation, a neutron radiation survey at approximately 4% power showed that a dose rate reduction factor for the 69' elevation was from 30 to 100. The dose rate reduction factor for the 45' elevation was 5 to 40. The neutron dose rate reduction factor for the 10' elevation was 2 to 10.

The gamma dose rate reduction factor for the 69' elevation was 10 to 15. On the 45' elevation, the gamma dose rates were less than 28 mrem/hr. On the 10' elevation the gamma dose rates near the containment wall were reduced by factors of 1.2 to 8, whereas the dose rates near the primary shield did not change significantly.

6.4.3 Conclusions

The total dose rate at the 45' elevation inside containment is now less than 100 mrem/hr, which is believed to be an acceptable level on a permanent basis to allow access during power operation. A further reduction by a factor of 5 to 10 in dose rates at the 69' elevation would be necessary to reach a nominal target level of 100 mrem/hr. Design of permanent shielding for the reactor vessel annulus and RCP motor hatches is underway. This shielding is expected to bring radiation levels down to or below 100 mrem/hr in normally accessible areas.

6.5 Turbine Runback/Step Load Change Test

6.5.1 Purpose

The objectives of the test were to determine the following:

(1) Plant power levels at which TRAC I (negative startup rate), TRAC II (dropped CEA reed switch), and applicable turbine-generator undercurrent relays arm and disarm.

(2) Time between receipt of turbine runback signal and arrival at required power.

(3) That the Feedwater Regulating System maintains steam generator levels above the reactor pretrip setpoints and below the turbine trip setpoint.

6.5.2 Test Results

The turbine runback was initiated from a load limiting condition by placing a temporary jumper across one of the turbine runback control element assembly reed switch contacts. This caused a turbine runback to the point at which the runback circuit disarmed. The temporary jumper remained in place until turbine control using the load limiter had been established.

A turbine runback was attempted on 5/6/75. This runback test was unsuccessful because the turbine generator undercurrent relays were improperly adjusted. The relays were set for 70% of turbine full load instead of the equivalent of 70% of 2611 MWth as specified in the Safety Analysis. The current setting for 70% of turbine full load exceeded the actual current value at 1828 MWth and the runback circuit was not armed. The relays were adjusted to the equivalent of 70% of 2611 MWth and the test was rerun on 5/18/75.

The arming and disarming setpoints as determined by slowly varying reactor power are shown in Table 6.5-1. The runback was initiated with reactor power about 71% by jumpering the dropped CEA reed switch for CEA 5-1. A runback resulted to 1574 MWth or 61.5% reactor power and occurred at a rate of 0.75% power/sec. The turbine reached 61.5% reactor power at 27 seconds after runback was initiated. During the runback, steam generator levels varied between +12 inches and -16 inches. Variations of plant parameters for a 35 minute period following initiation of turbine runback are shown in Figures 6.5-1 through 6.5-5. The RCS Letdown Excess Flow Check Valve shut at two (2) minutes and five (5) seconds after initiation of the runback.

6.5.3 Conclusions

The test adequately demonstrated the performance of the turbine runback circuitry. The steam generator levels remained in a relatively small band centered around a normal level of 0 inches. However, the runback rate was slower than specified in the acceptance criterion. As the turbine runback is a scheme for recovering from a dropped CEA incident, CE, Windsor will consider the as measured runback rate in their analysis of the dropped CEA test results (see Section 6.6).

TABLE 6.5-1

TURBINE RUNBACK CIRCUIT

RELAY ARMING AND DISARMING SETPOINTS

DEVICE	ARMING		DISARMING	
	REACTOR POWER (%)	FIRST STAGE TURBINE PRESSURE (psig)	REACTOR POWER (%)	FIRST STAGE TURBINE PRESSURE (psig)
TRAC I	67.2	315	65.2	303
TRAC II	69.5	325	61.9	285
337/R3 GEN undercurrent relay	66 (12.8 KA)	308	63 (12.2 KA)	291

CALVERT CLIFFS UNIT 1 1ST CYCLE

TURBINE RUNBACK

AVERAGE STEAM GENERATOR PRESSURE

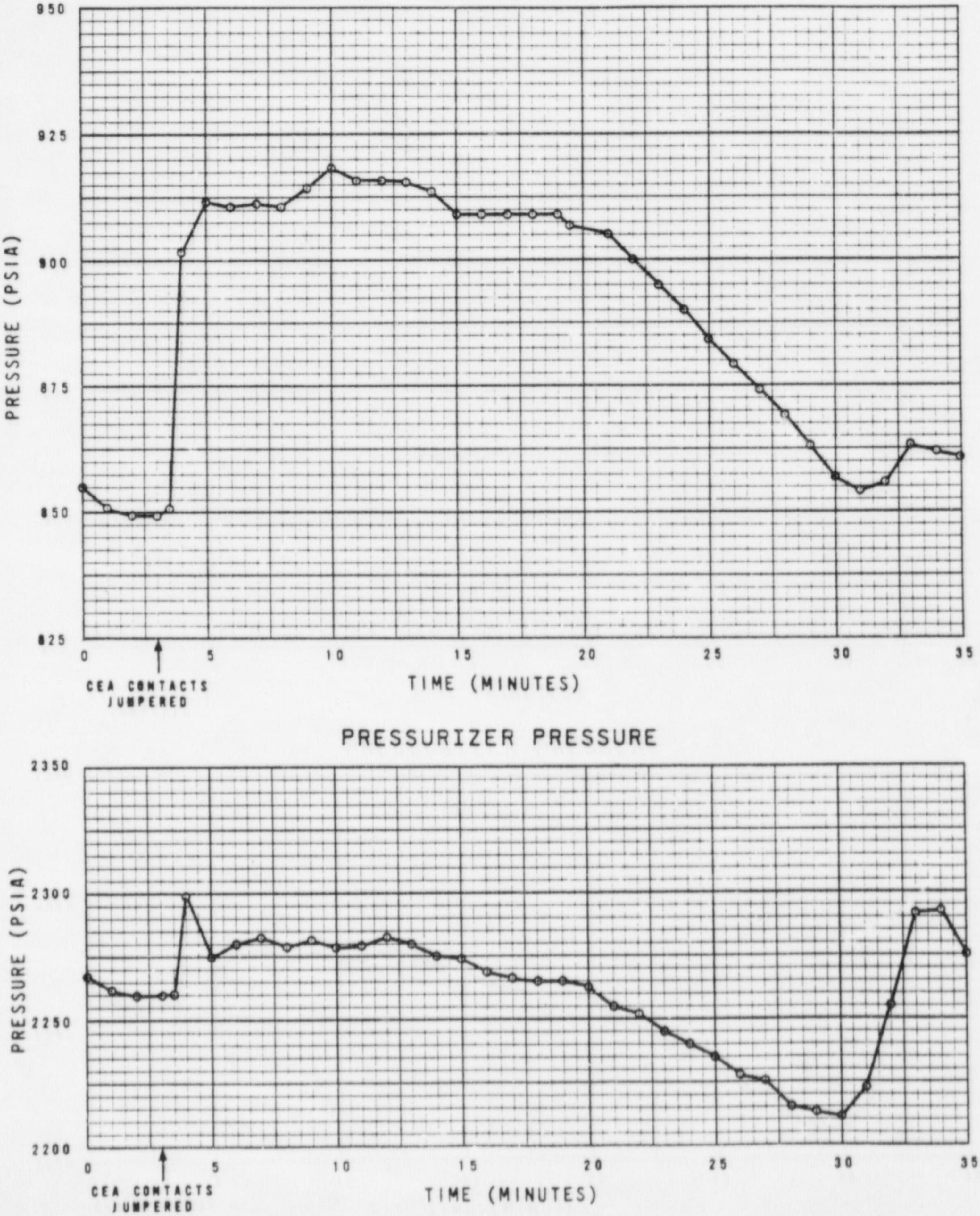
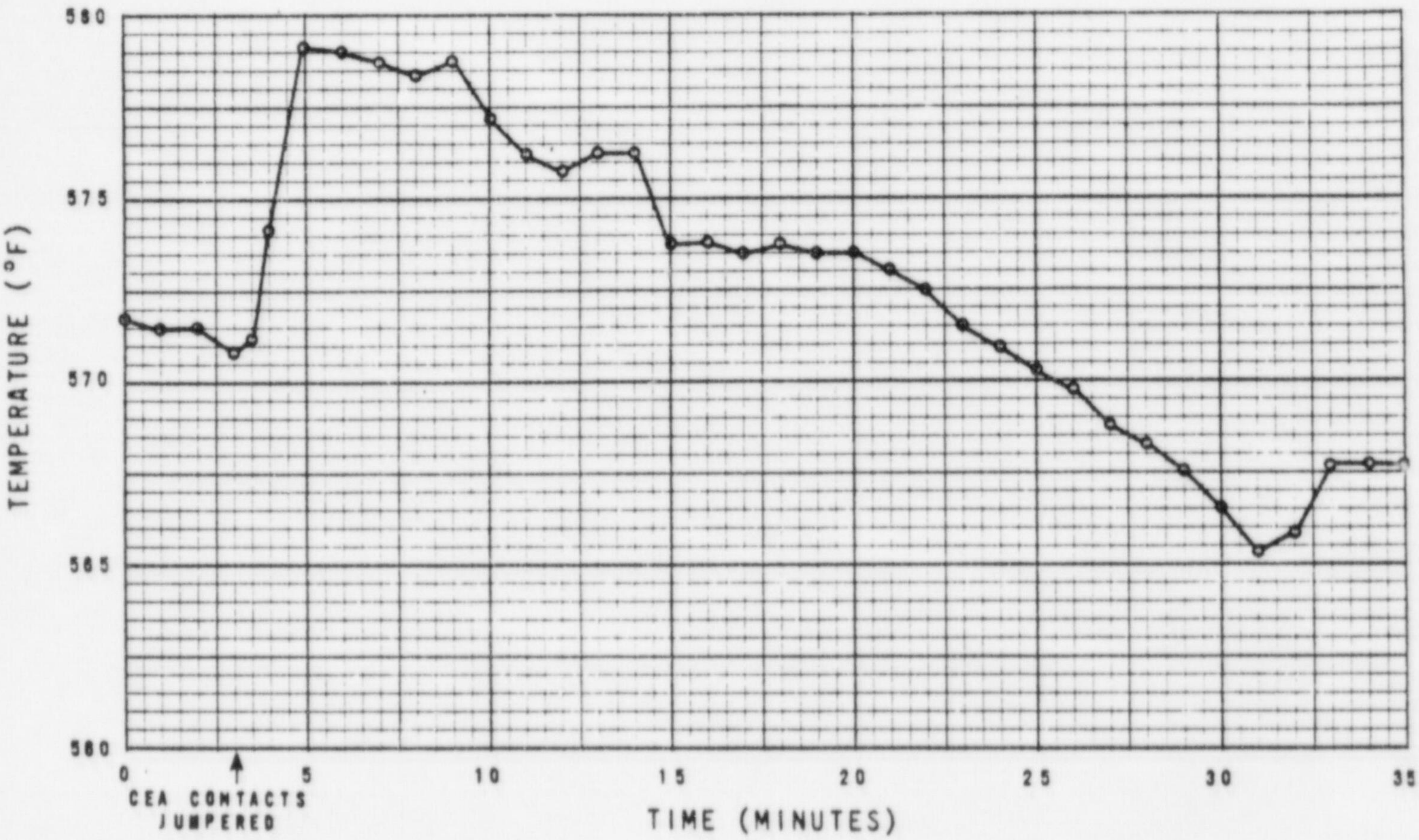


FIGURE ...

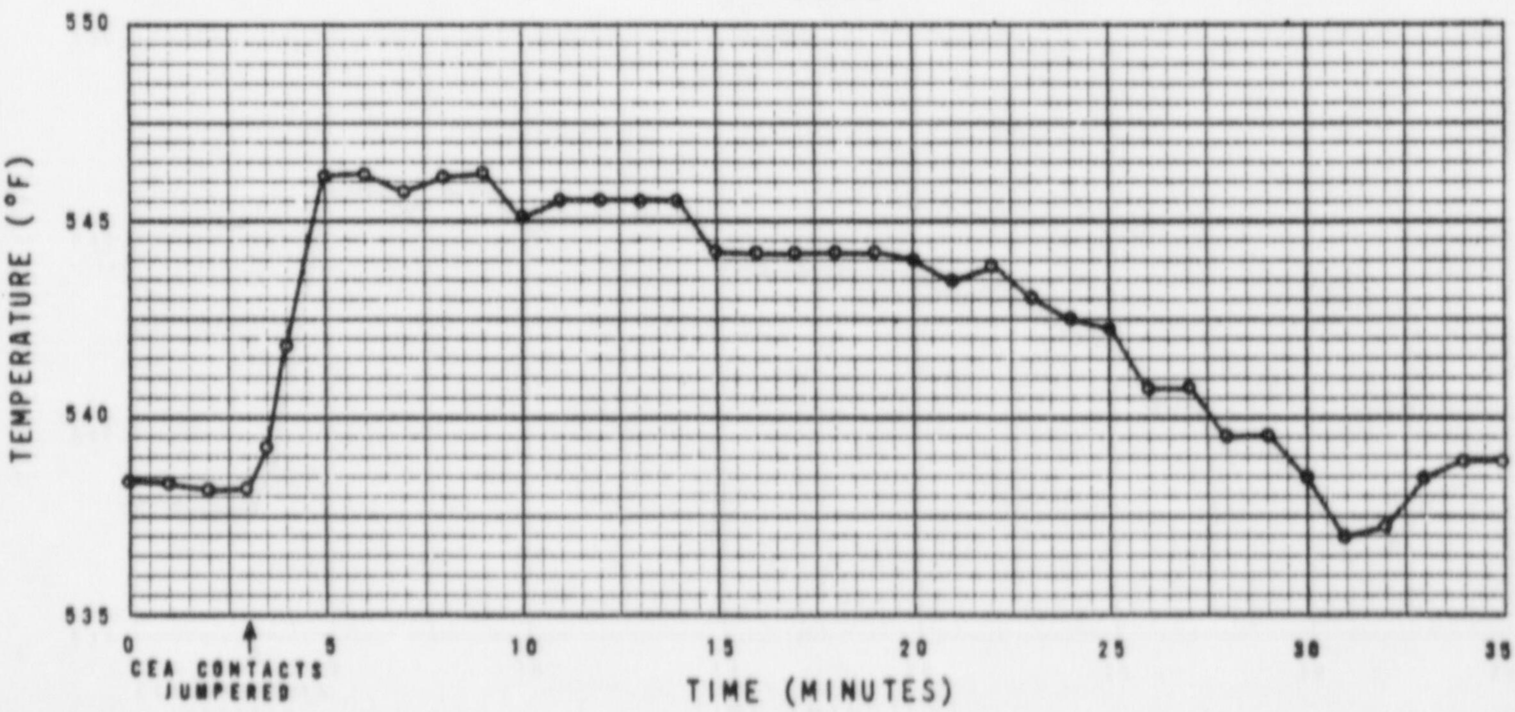
CALVERT CLIFFS UNIT 1 1ST CYCLE

TURBINE RUNBACK

T HOT



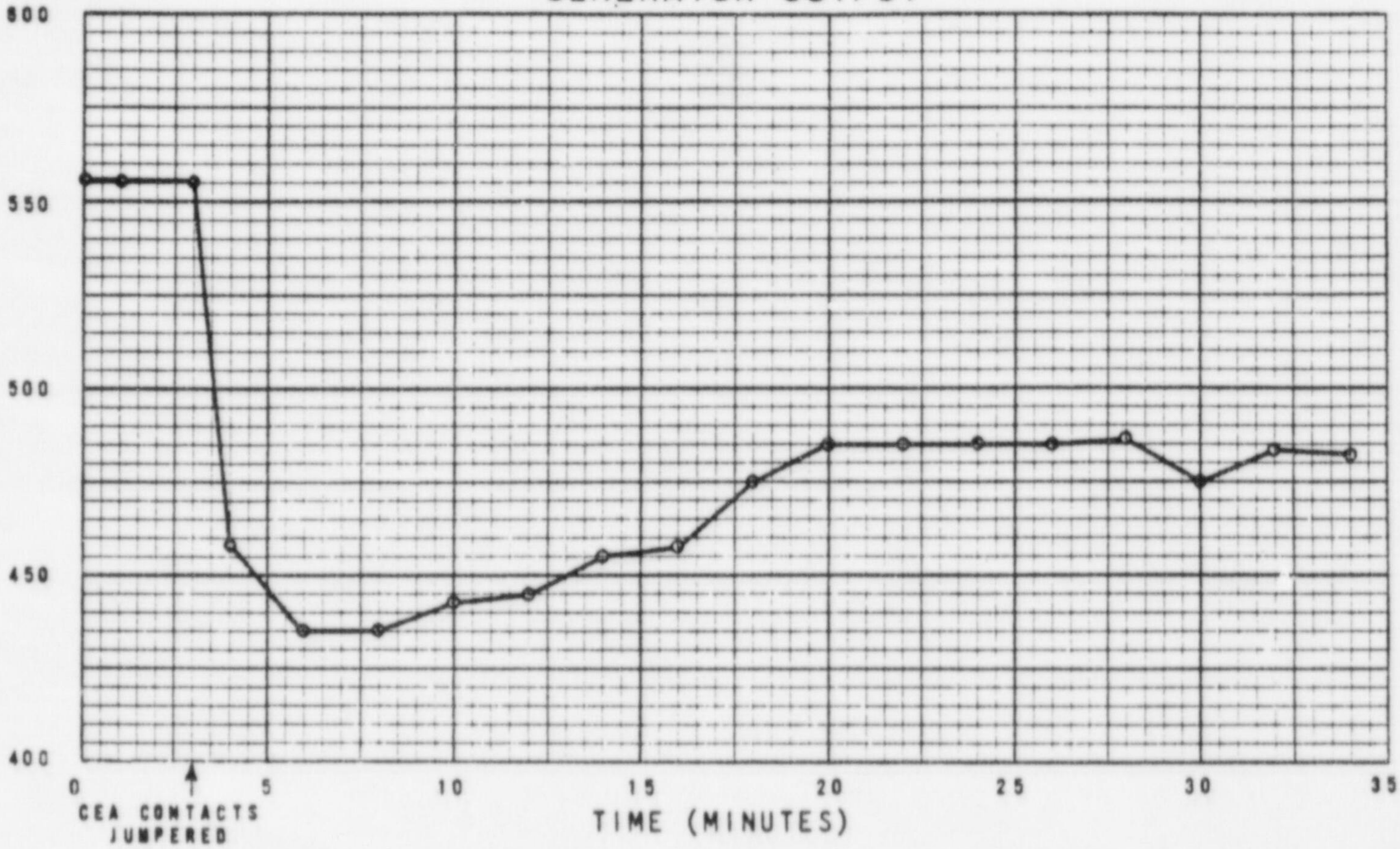
T COLD



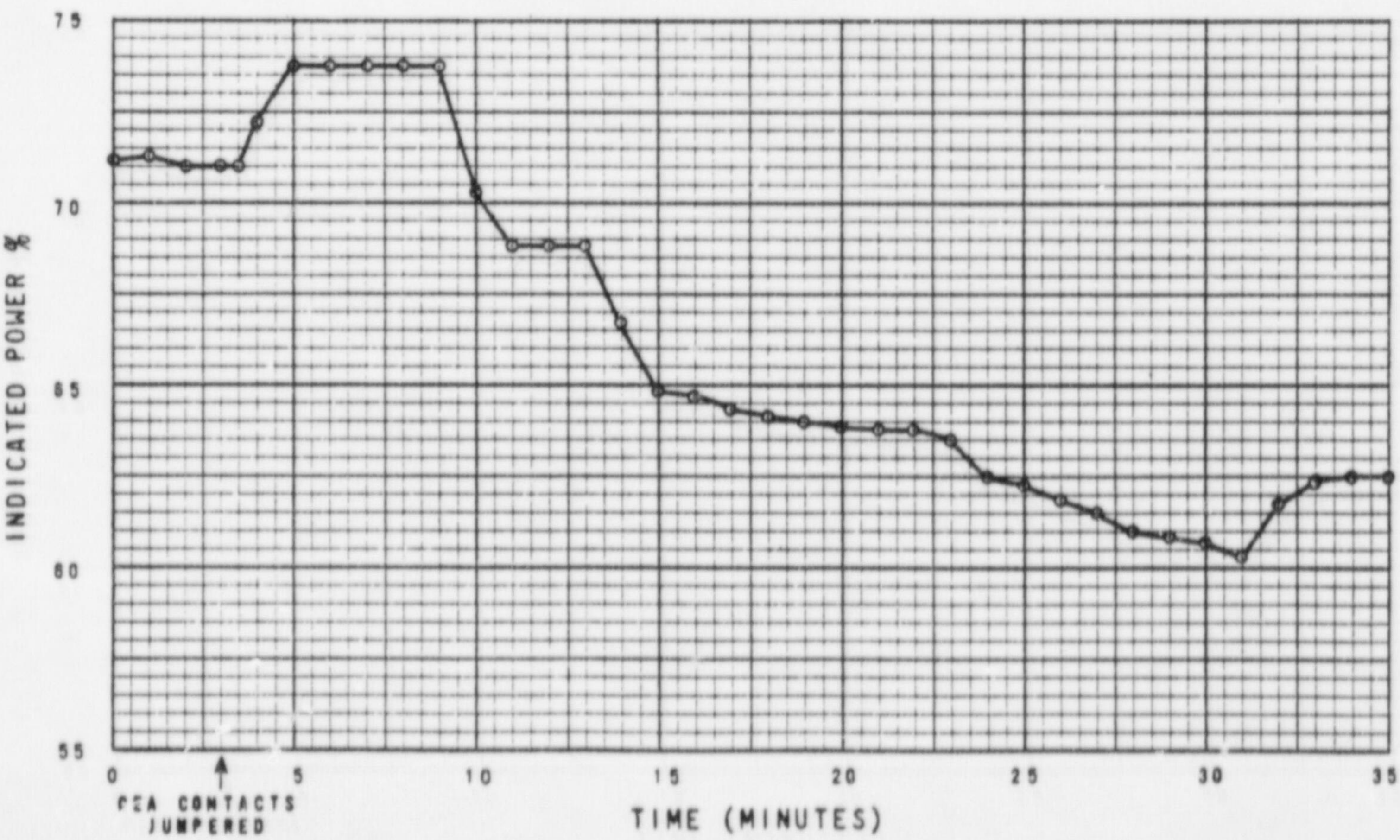
CALVERT CLIFFS UNIT 1 1ST CYCLE

TURBINE RUNBACK

GENERATOR OUTPUT



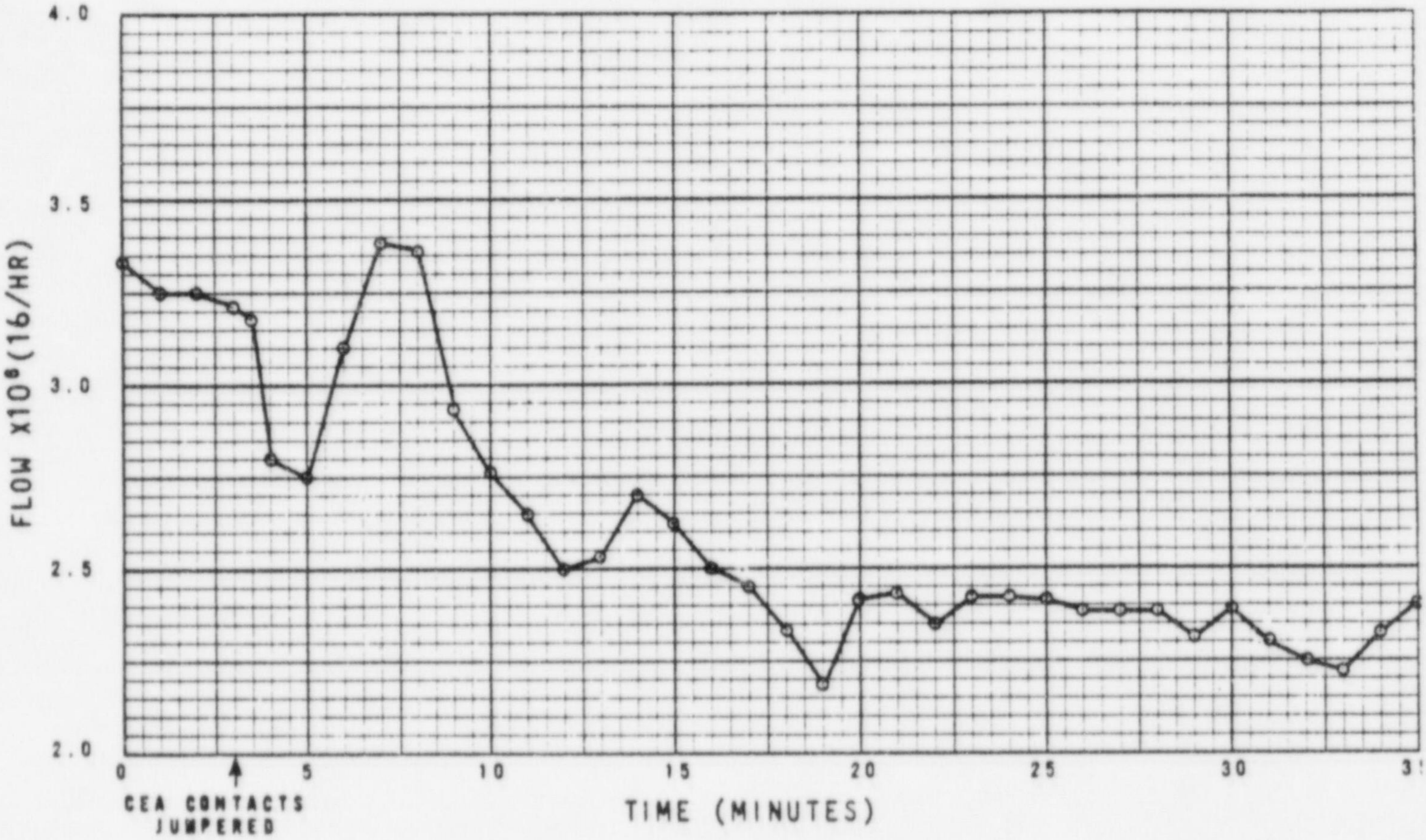
INDICATED NUCLEAR POWER



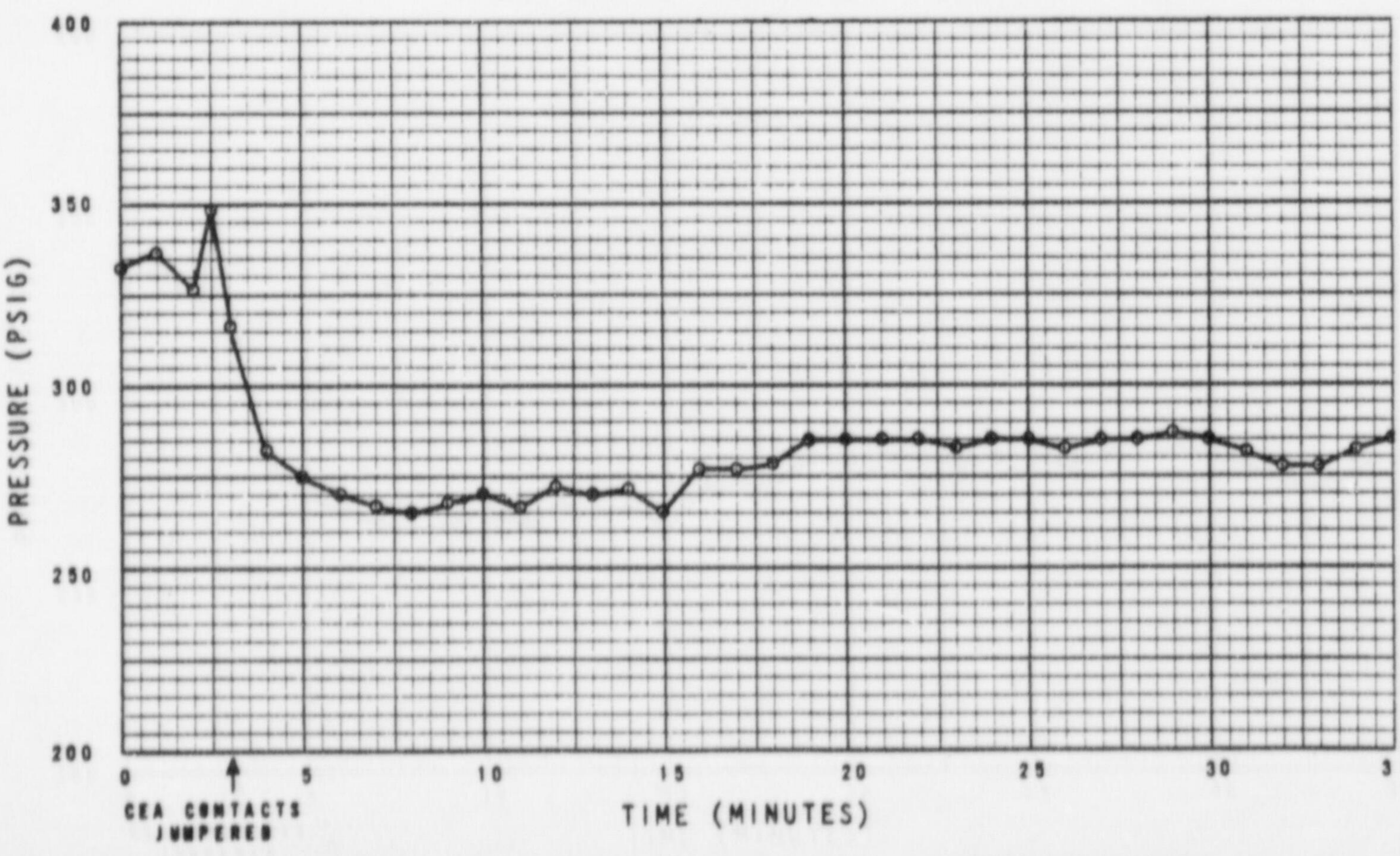
CALVERT CLIFFS UNIT 1 1ST CYCLE

TURBINE RUNBACK

AVERAGE FEEDWATER FLOW

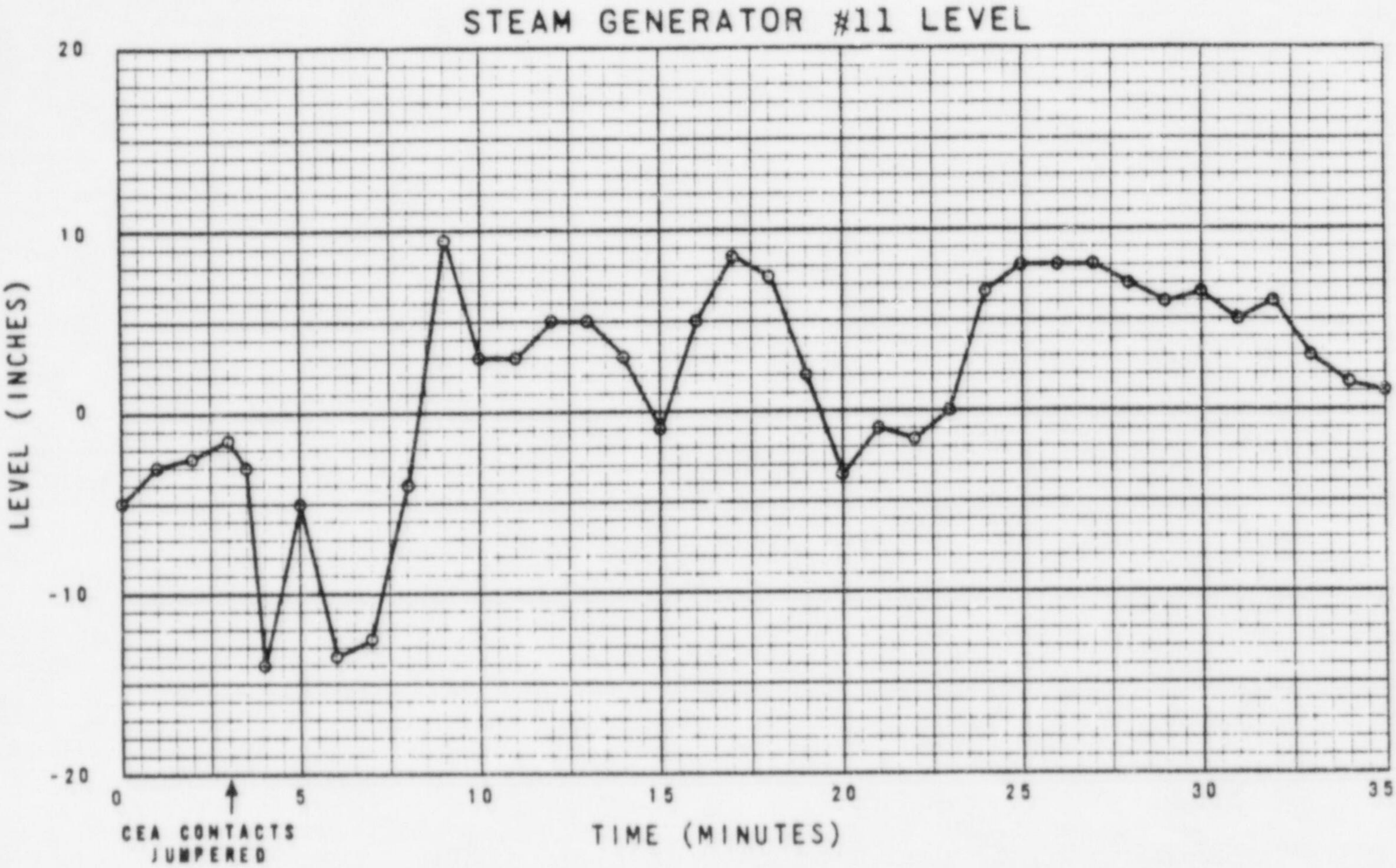


FIRST STAGE TURBINE PRESSURE



CALVERT CLIFFS UNIT 1 1ST CYCLE

TURBINE RUNBACK



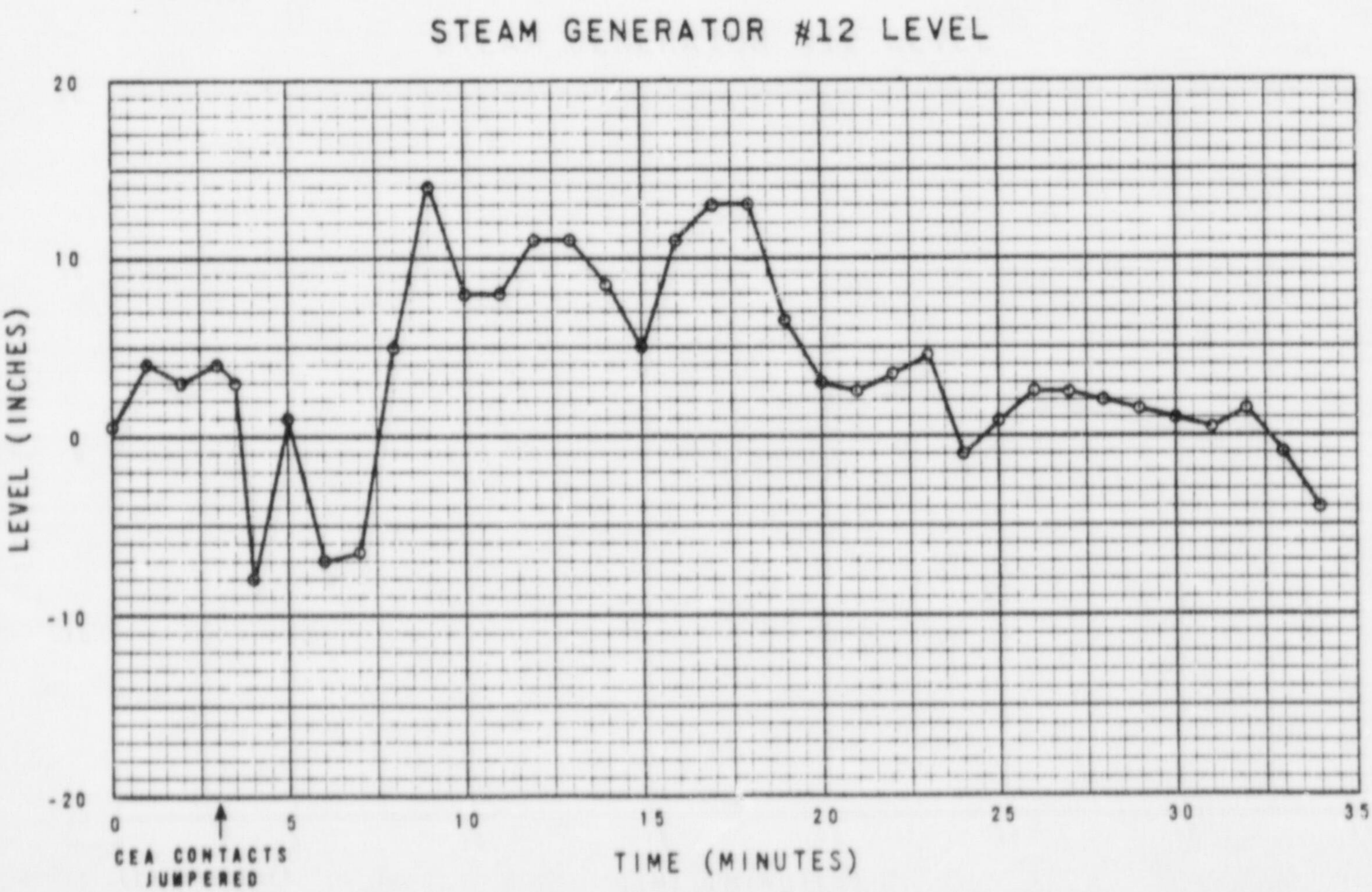


FIGURE 6.5-5

6.6 Dropped CEA, Power Asymmetry and Azimuthal Xenon Transient Tests

6.6.1 Purpose

The test was designed to accomplish the following objectives:

(1) Measure the asymmetric power distribution resulting from the drop of the most worthy Control Element Assembly (CEA).

(2) Determine the ability of the excore detectors to sense the drop of the most remote full length CEA.

(3) Determine the dampening characteristics of an azimuthal xenon oscillation.

6.6.2 Test Results

The dropped CEA test consisted of one static and two dynamic rod drop tests. The static test was conducted to observe the static power tilt and resulting azimuthal xenon oscillations. The static rod drop measurement consisted of diluting CEA B-8, a dual shutdown CEA, to its full in position while holding ΔT power constant. Power tilt indication at the start of the test was 0.8%. The maximum power tilt occurred after CEA B-8 had been inserted approximately 4 hours, and was 16.5%. CEA B-8 was then withdrawn and aximuthal xenon oscillations continued until dampened with time. See Figure 5.0-1 for relative locations of CEA's and excore detectors. Analysis of incore detector signals prior to and as close as possible to just after the static drop test indicated a greater than 12% change on at least three (3) incore rodium detectors.

Two dynamic rod drop tests were conducted. CEA 5-58, a low reactivity worth, single CEA on the core periphery was used for the first drop and the center core CEA 5-1 was used for the second drop.

When CEA 5-58 was dropped, excore nuclear instrumentation near the rod (Power Range Regulating Channel [PRRC] X) showed a drop of about 7% power, while instrumentation about 90 degrees away (PRRC Y) showed only about 1.5% power reduction. Power Range Safety Channels (PRSC) C and D showed dropped rod indication with CEA 5-58.

When CEA 5-1 was dropped, all four (4) PRSC's indicated a dropped rod almost immediately and PRRC's X and Y indicated a decrease of 9% and 8% power respectively.

6.6.3 Conclusions

Test results indicate that a dropped CEA would be detected and azimuthal power oscillations would converge.

A preliminary on-site analysis of incore detector data indicated that the acceptance criterion for the change in incore detector signals before and after the static CEA drop had been exceeded. The situation could not be adequately resolved on-site. Consequently, administrative restrictions currently prevent operation above 50% reactor power with a dropped CEA. Another acceptance criterion was that CE, Windsor complete a more detailed analysis of the data to determine maximum allowable power level with a dropped CEA. That analysis is in progress. The Safety Analysis reported in the FSAR was conducted at 100% power and assumed conditions other than those at which this test was conducted.

6.7 Trip of Main Generator Breaker

6.7.1 Purpose

The test was designed to evaluate system reliability during a loss of generator load from nominal 100% power.

6.7.2 Test Results

The trip was initiated by opening the generator output 500 KV breaker 552-23 from the main control panel. Reactor Coolant System (RCS) temperature and pressure indications were nominal for a reactor trip until about one minute after the trip. At that point, turbine bypass and atmospheric dump valve controllers called for zero demand and temperature continued to decrease. It was then realized that turbine bypass valves 3940 and 3944 were not fully closed and/or were leaking. Manual isolation valves ahead of valves 3940 and 3944 were closed to terminate the cooldown, while retaining turbine bypass capability. However, a shaft key was missing from one valve handwheel making the isolation extremely lengthy for 3940. During the cooldown RCS temperature reached a minimum of 465°F, pressurizer pressure a minimum of 1200 psia, and steam generator pressure a minimum of 520 psig. As a result of the RCS depressurization, safety injection was automatically initiated. Due to a leaky valve connecting the safety injection with the containment spray system several gallons of safety injection water were released into the containment. Reduction of secondary system pressure to 520 psig resulted in a Steam Generator Isolation Signal (SGIS) automatically shutting the main steam isolation valves. With the shutting of the main steam isolation valves, the cooldown was terminated and a return to Hot Shutdown commenced. RCS temperature, pressurizer pressure and steam generator level indication was back on scale or approaching Hot Shutdown conditions at the end of one hour from time of trip. Variations in plant parameters for one hour after the trip are shown in Figures 6.7-1 and 6.7-2.

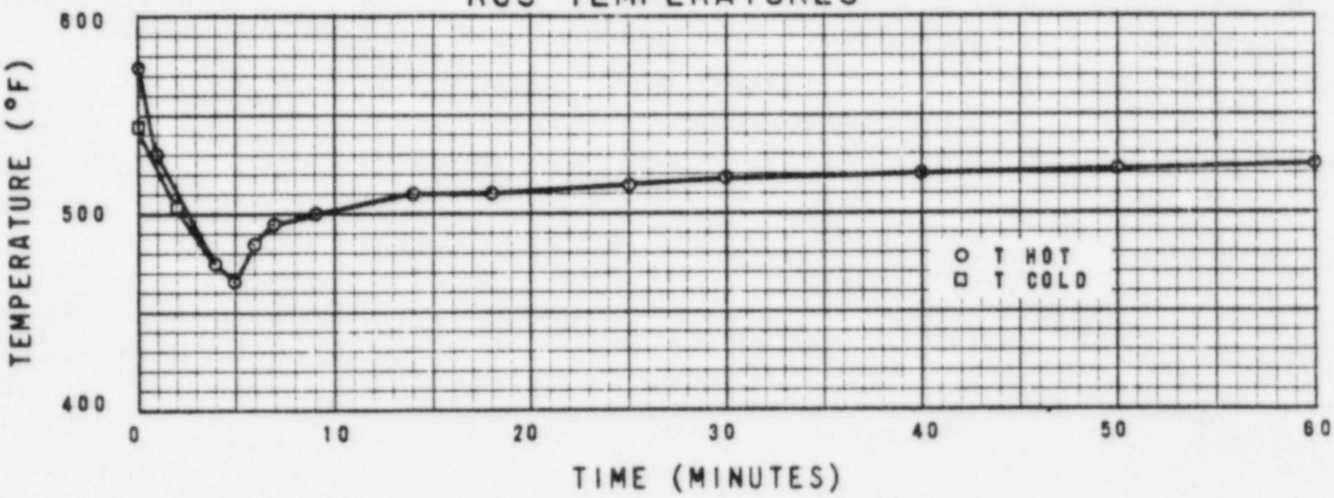
6.7.3 Conclusions

Evaluation of data indicates that the low levels in the pressurizer and steam generators did not cause any damage to plant systems and the plant was safely returned to Hot Shutdown. The failure of the turbine bypass valves to close was due to cracked plugs in the valves and the valves have been repaired. The turbine bypass valves have continued to be a problem throughout the test program. Discussion of design changes under consideration is contained in Section 6.1.

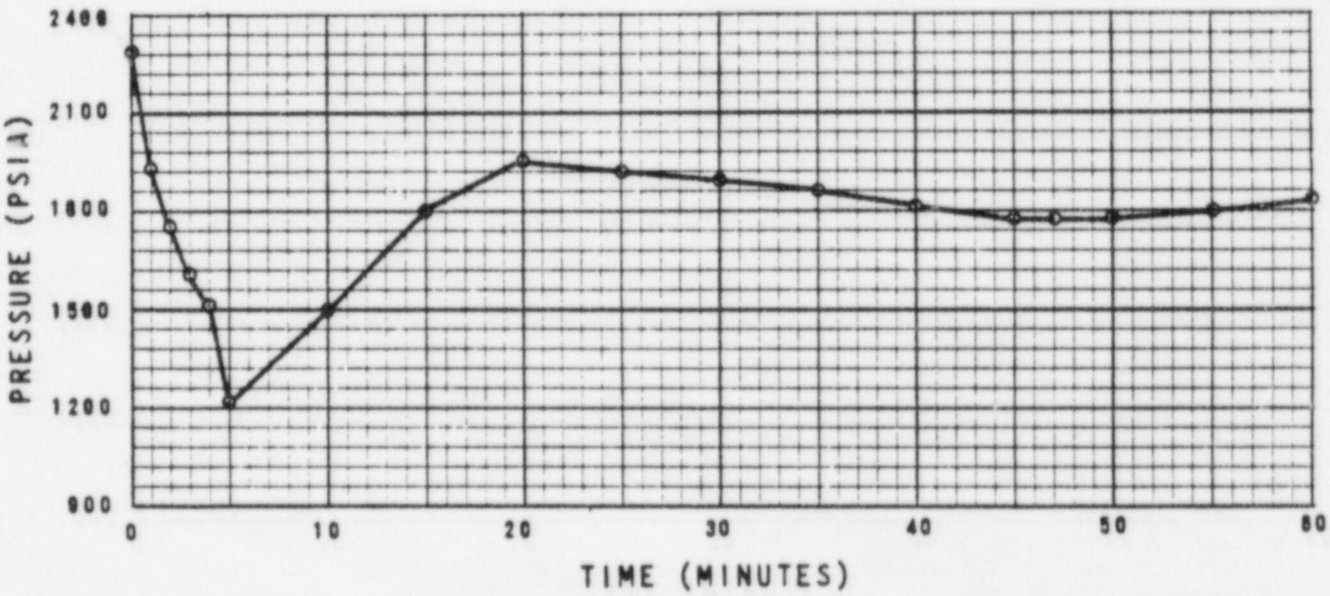
CALVERT CLIFFS UNIT 1 1ST CYCLE

100% POWER GENERATOR TRIP

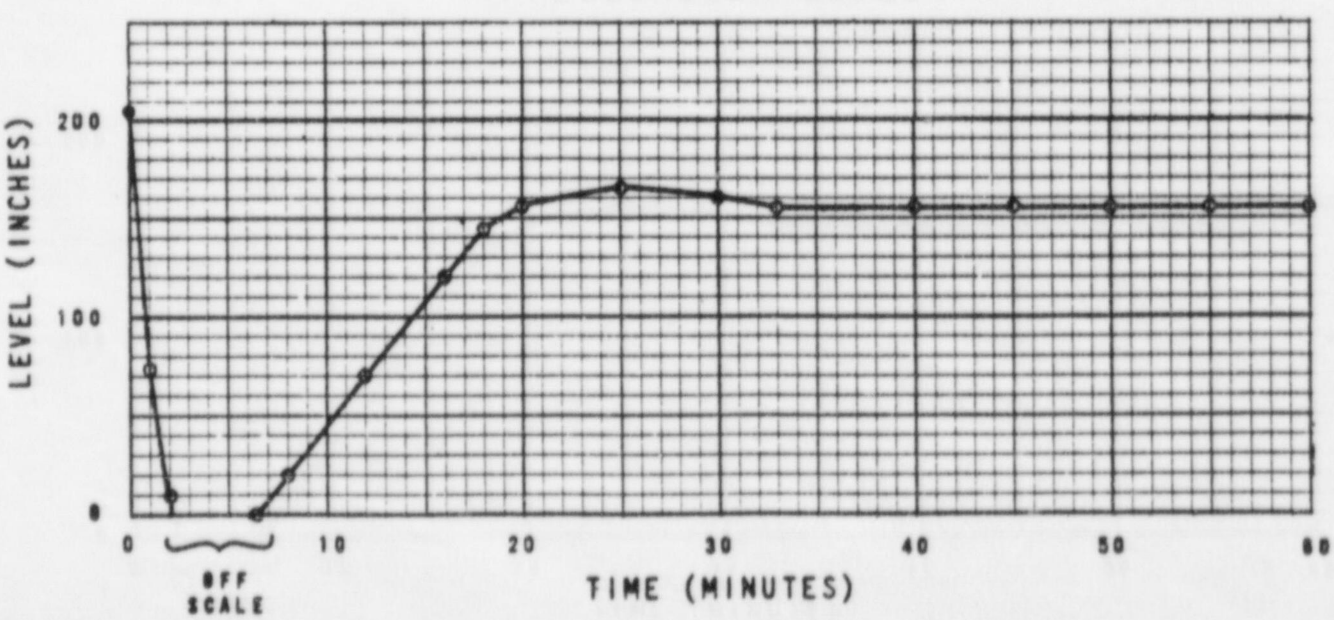
RCS TEMPERATURES



RCS PRESSURE



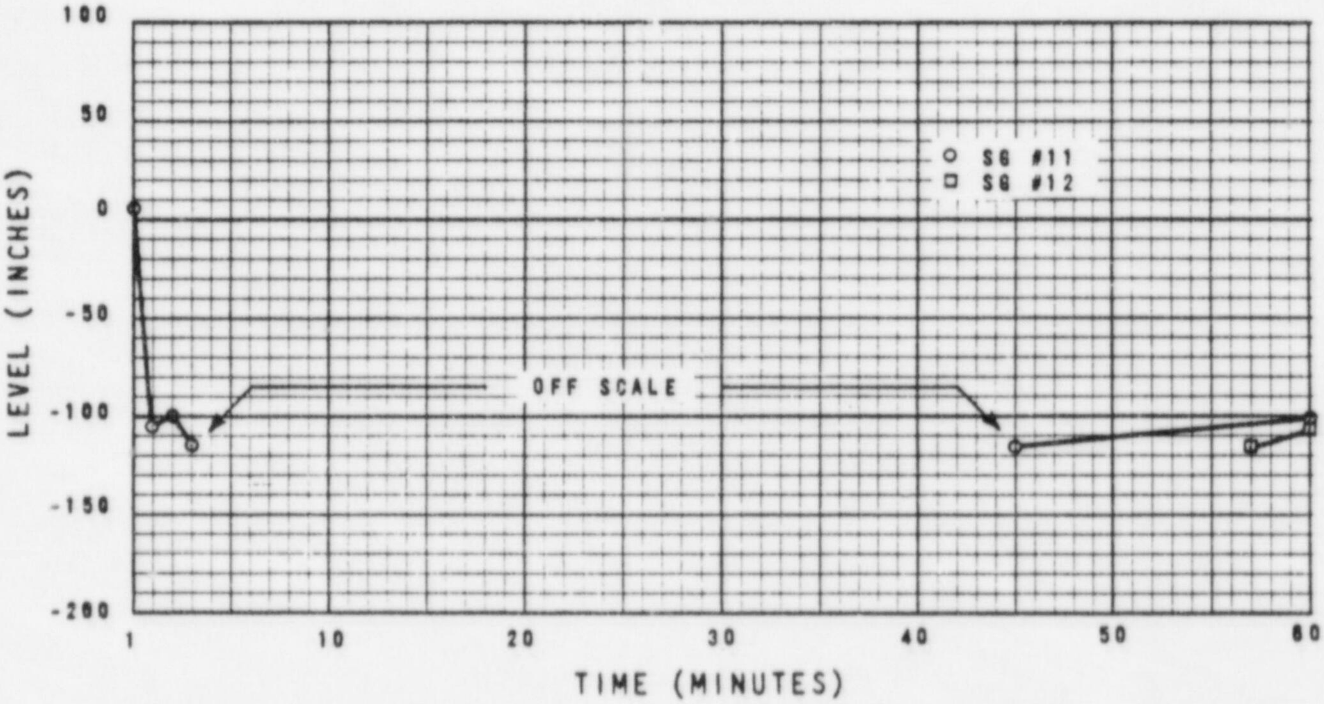
PRESSURIZER LEVEL



CALVERT CLIFFS UNIT 1 1ST CYCLE

100% POWER GENERATOR TRIP

STEAM GENERATOR LEVEL



STEAM GENERATOR PRESSURE

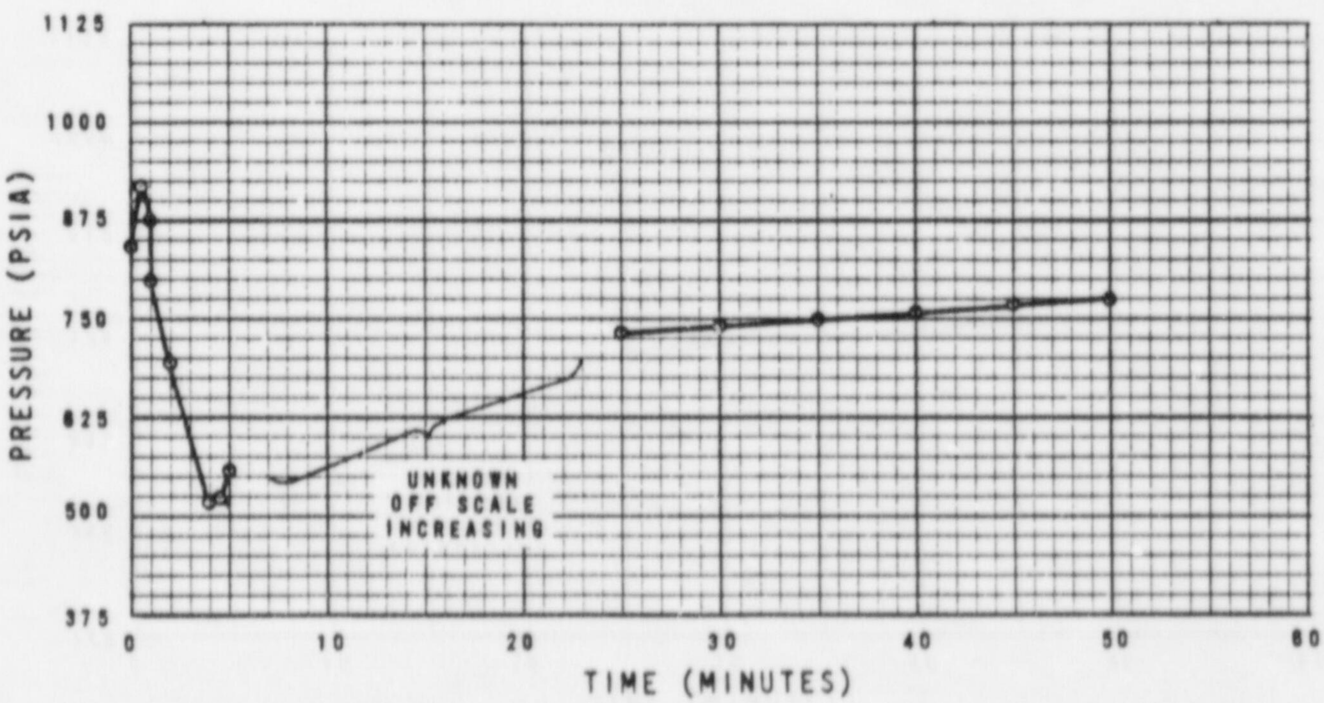


FIGURE 6.7-2

6.8 Xenon Follow Measurements

6.8.1 Purpose

The purpose of this test was to obtain transient test data at several test plateaus for the purpose of evaluating the Shape Annealing Factor (SAF) for each Power Range Safety Channel and to evaluate an induced free xenon oscillation performed at the 80% test plateau.

6.8.2 Test Results

During the 80% test plateau, axial oscillations were induced in the core. These oscillations were monitored by the plant computer In Core Analysis (INCA) program and the Axial Shape Index (ASI) calculated by the Reactor Protective System (RPS). All axial oscillations were convergent and the minimum DNBR was 6.13 and the maximum measured Linear Heat Rate (LHR) was 9.47 kw/ft. When corrected for uncertainities, the maximum LHR was less than the acceptance criterion of 14.9 kw/ft. DNBR acceptance limit was 1.3.

The SAF for each Power Range Safety Channel was also measured. The SAF corrects the detector signal to account for the distance from the detector to the reactor core and corrects for the signal received by the upper detector from neutrons generated in the bottom of the core and the signal received by the lower detector from neutrons generated in the upper part of the core. The SAF is determined by plotting ASI (INCA) versus the ASI (EXT) as read from the Reactor Protective System (RPS) during a xenon oscillation with all CEA's full out. The slope of the line resulting from this plot is the Shape Annealing Factor. ASI is Axial Shape Index, a ratio of the difference in power generated in the lower and upper halves of the core to total core power. Table 6.8-1 summarizes the comparisons between measured and predicted results. As measured SAF's were incorporated into RPS set-points.

TABLE 6.8-1

COMPARISON OF MEASURED AND PREDICTED SHAPE ANNEALING FACTORS

RPS CHANNEL	MEASURED	PREDICTED
A	2.18	1.47
B	2.05	1.47
C	1.60	1.47
D	1.88	1.47

6.8.3 Conclusions

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

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

6.9 Remote Shutdown Test

6.9.1 Purpose

The purpose of the test was to demonstrate that the plant responds properly to a generator trip and that it can be safely maintained in the Hot Shutdown condition from outside the control room.

6.9.2 Test Results

The generator trip was initiated from 50% power by simulating a generator fault at the Unit Protection Panel in the Cable Spreading Room. An emergency crew was used to monitor the control panels and to provide emergency control should the need have arisen to abort the test. The regular shift crew left the control room after verifying the trip and went to the local panels to complete the test. RCS temperature was decreased to 527°F, held momentarily, then increased and maintained at approximately 537°F by local operation of the turbine bypass valves. Steam generator levels were maintained by local control of the Auxiliary Feedwater System. Variations of plant parameters for sixty (60) seconds following the trip are shown in Figures 6.9-1, 6.9-2A and 6.9-2B.

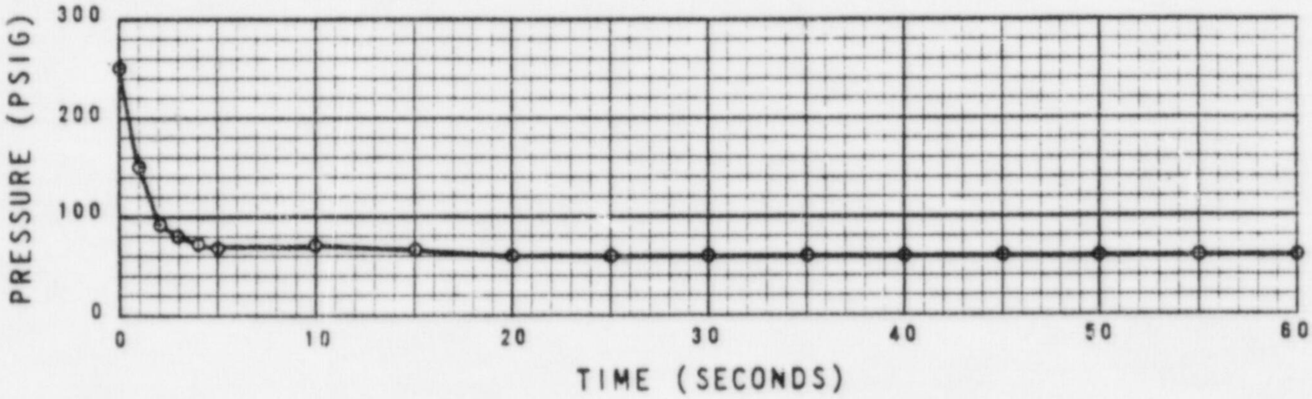
6.9.3 Conclusions

The ability to safely shutdown and maintain the plant in Hot Shutdown from local panels was successfully demonstrated. Both the trip and subsequent cooldown progressed smoothly with no equipment damage.

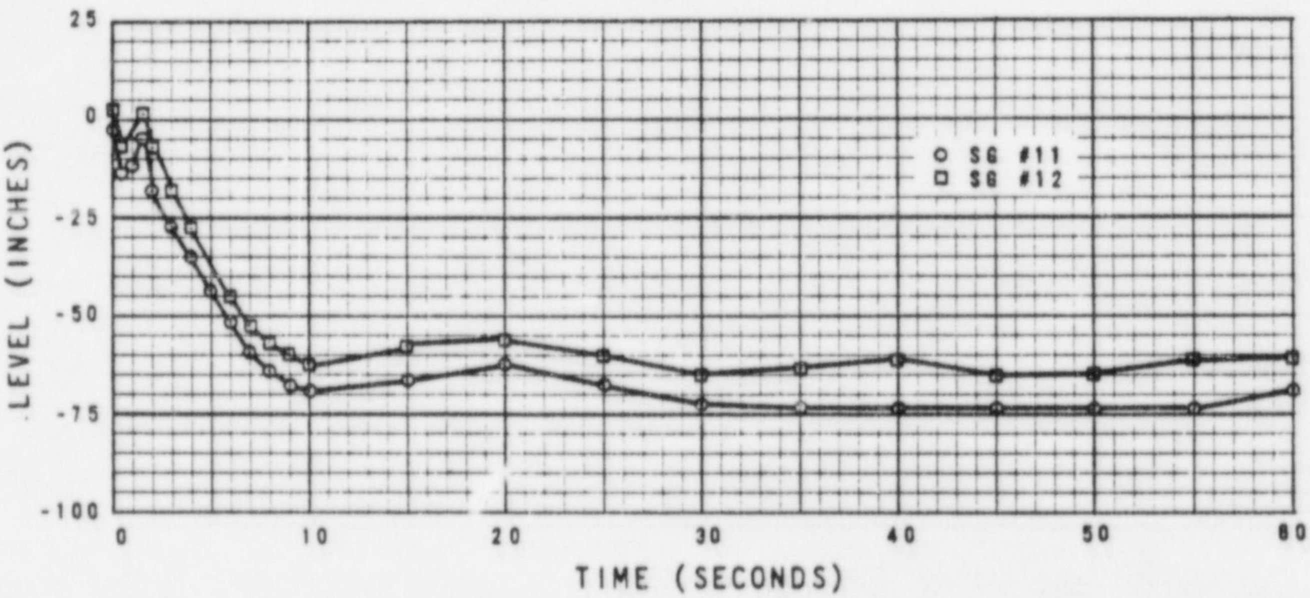
CALVERT CLIFFS UNIT 1 1ST CYCLE

50% POWER GENERATOR TRIP & S/D FROM OUTSIDE CONTROL ROOM

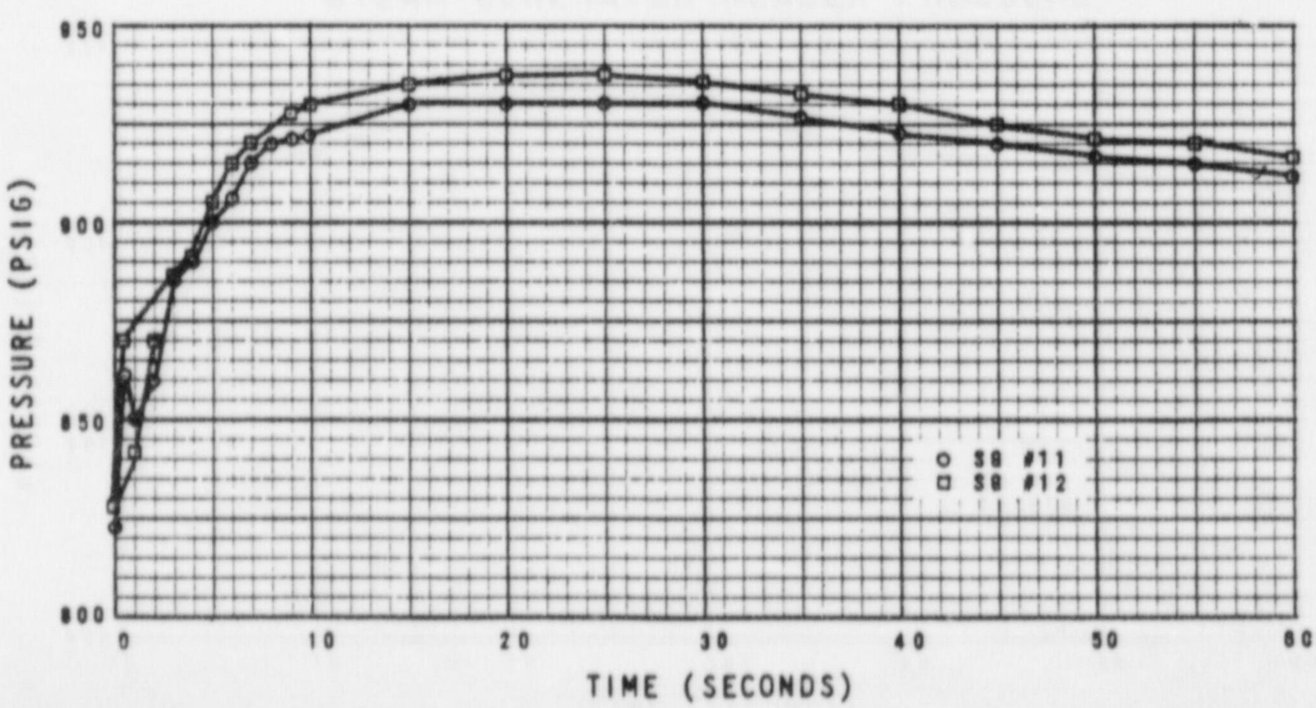
TURBINE FIRST STAGE PRESSURE



STEAM GENERATOR LEVELS



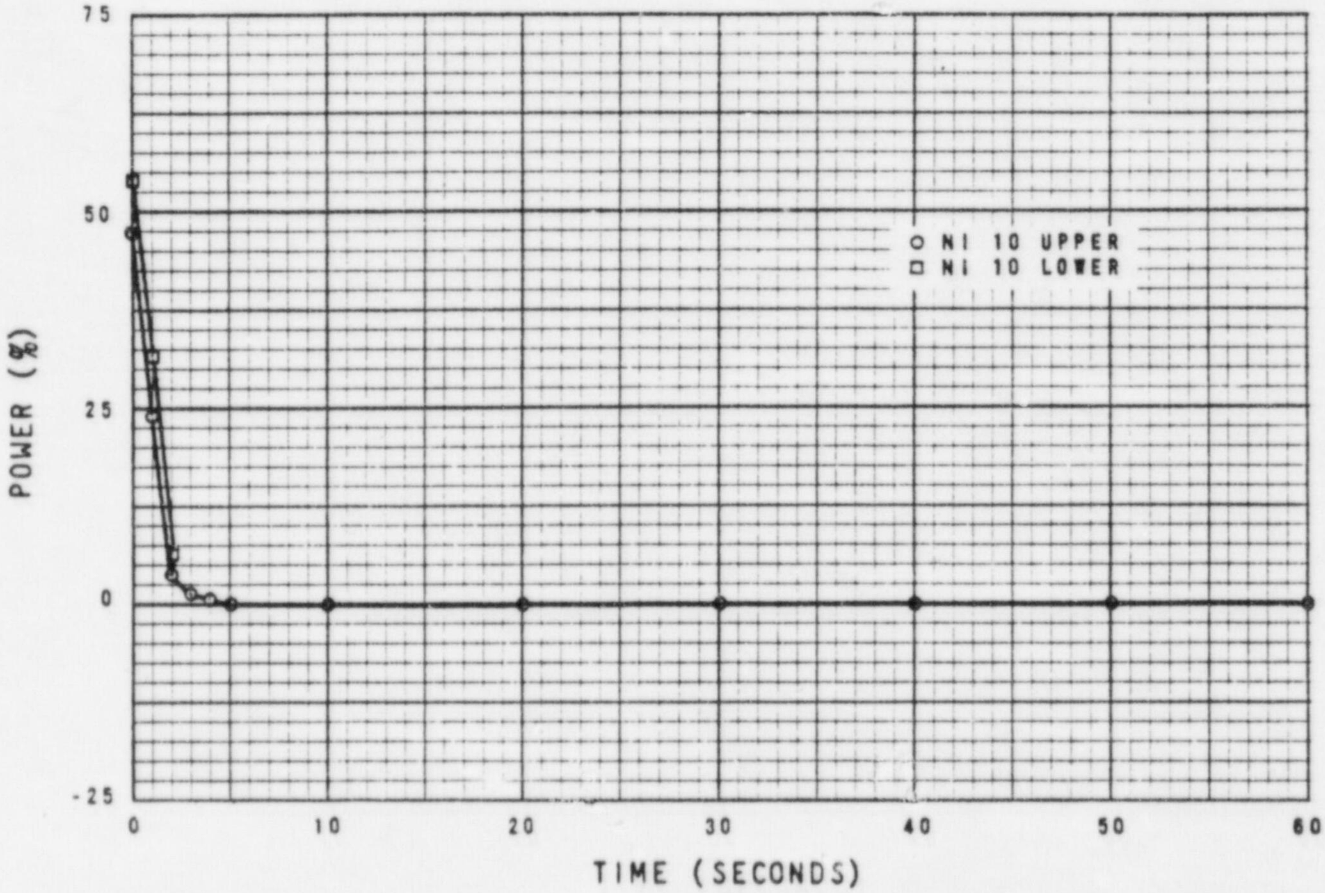
STEAM GENERATOR HEADER PRESSURE



CALVERT CLIFFS UNIT 1 1ST CYCLE

50% POWER GENERATOR TRIP & S/D FROM OUTSIDE CONTROL ROOM

NUCLEAR INSTRUMENTS



RCS COLD TEMPERATURES

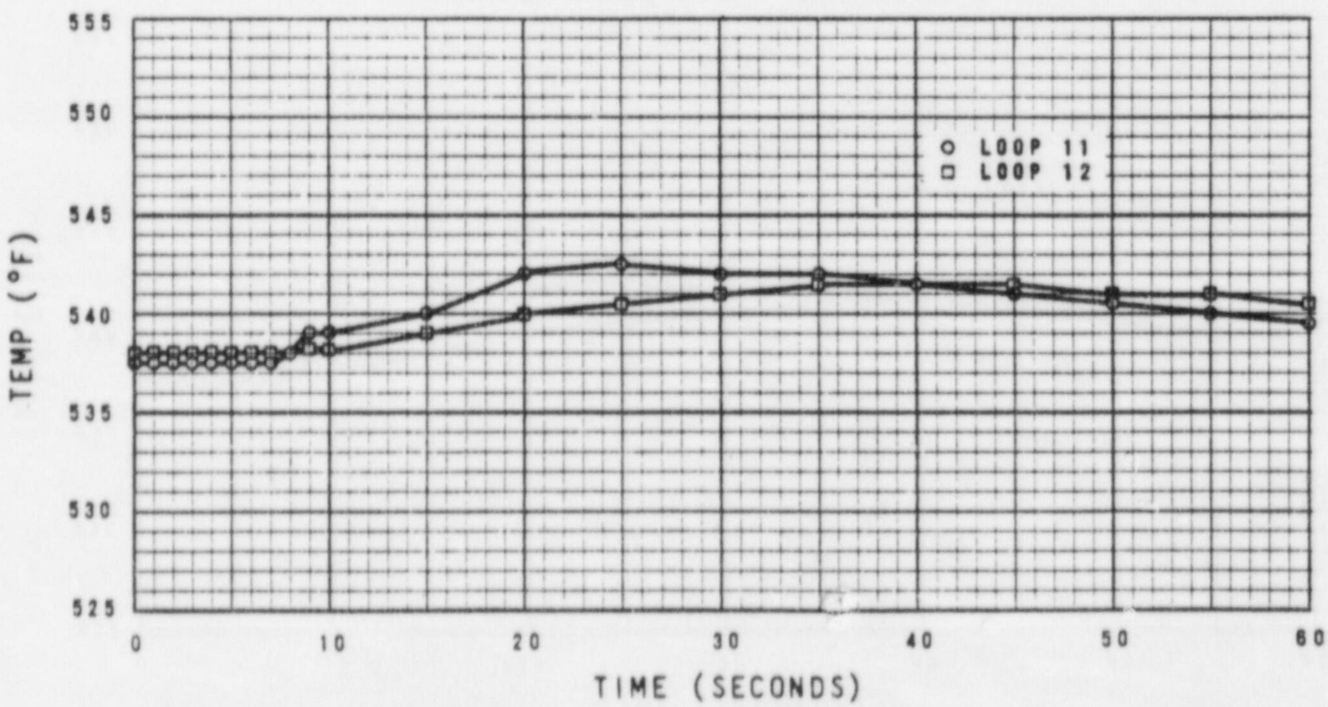
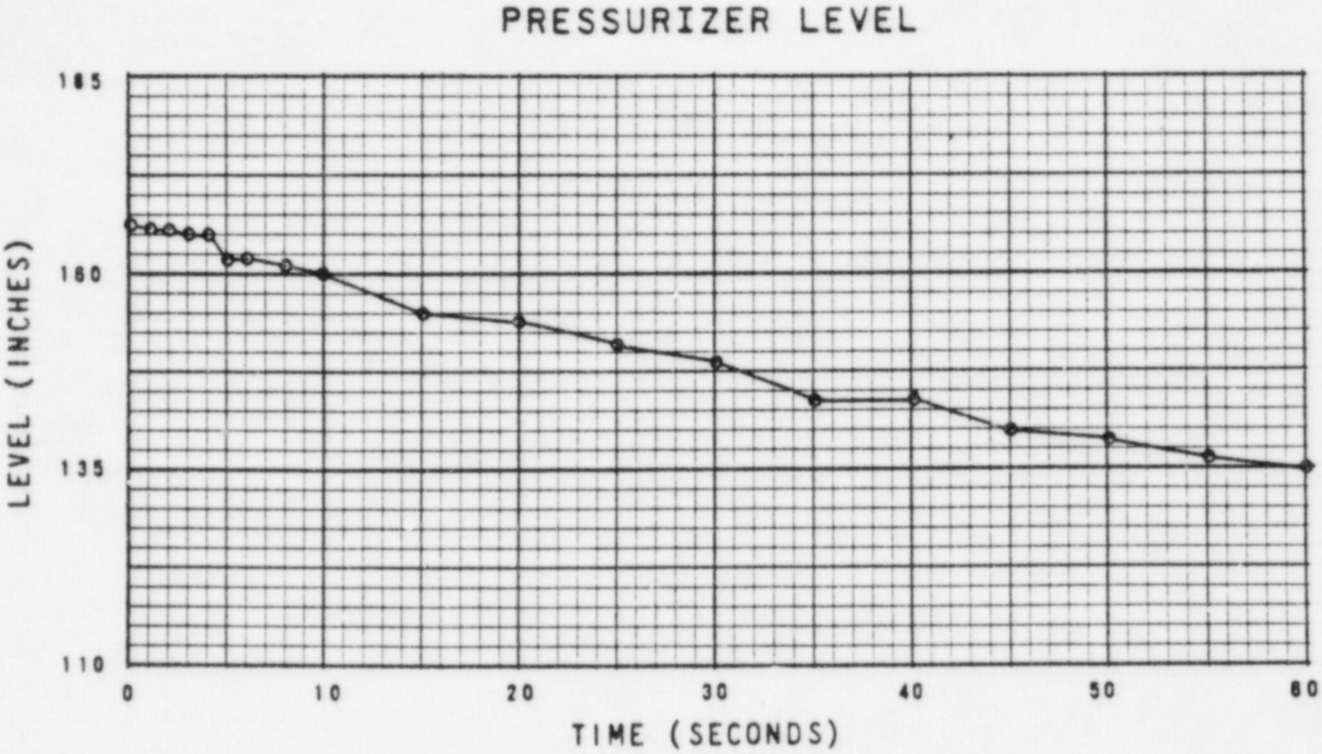


FIGURE 6.9-2A

CALVERT CLIFFS UNIT 1 1ST CYCLE

50% POWER GENERATOR TRIP & S/D FROM OUTSIDE CONTROL ROOM



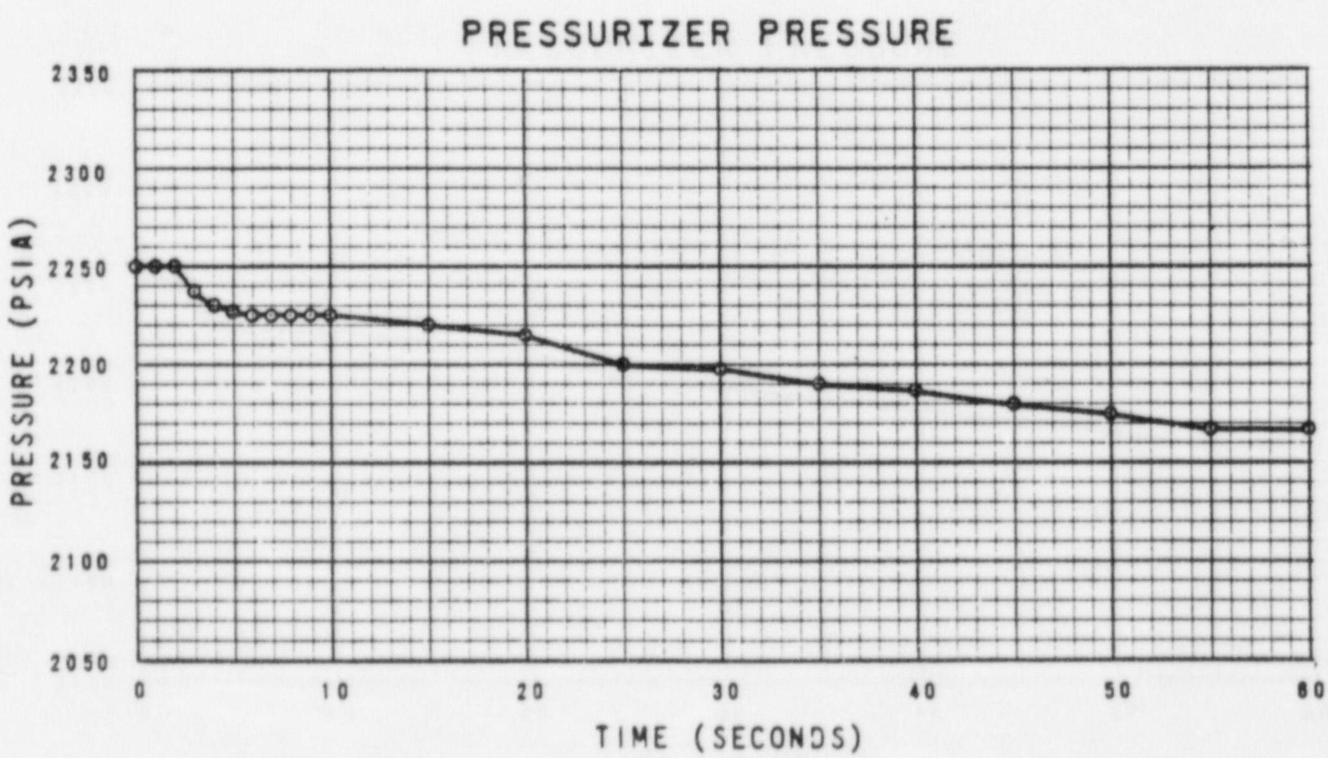


FIGURE 6.9-2B

6.10 Feedwater Regulating System Test

6.10.1 Purpose

The purpose of this test was to monitor the operation of the Feedwater Regulating System during power changes.

6.10.2 Test Results

Operations of the Feedwater Regulating System was monitored throughout EPT during reactor scrams; rod drops; steady state operations; ramp load changes; and the Turbine Runback/Step Load Change test. During the rod drop test, it was noted that the steam generator levels increased to approximately 20 inches when turbine load was reduced about 10% to control Reactor Coolant System (RCS) temperature. Investigation showed that feedwater flow was not decreasing as fast as steam flow resulting in increasing steam generator levels. The setting of the proportional bands on the feedwater regulator valve controller was changed to increase response time. Monitoring of the system during scrams and ramp load changes after the above modification indicated satisfactory operation of the system. During ramp changes of between 0.5%/min to 1.0%/min, the Feedwater Regulating System maintained steam generator levels with ± 10 inches of zero (normal) level.

6.10.3 Conclusions

Operation of the Feedwater Regulating System was acceptable or when discrepancies were noted, adjustments were made to provide acceptable results.

6.11 Loss of Off-Site Power with Coastdown

6.11.1 Purpose

The test was designed to accomplish the following objectives:

(1) To evaluate plant systems reliability during a total loss of AC power.

(2) To demonstrate correct operation of the turbine generator assisted Reactor Coolant Pump (RCP) coastdown circuitry.

(3) To obtain Reactor Coolant System (RCS) flow data during the coastdown.

6.11.2 Test Results

In order to defeat the automatic load transfer to live buses in case of a generator trip, Unit 1 house loads were transferred to the Unit 1 side of the plant output ring bus and the ring bus was split. When the outgoing 500 KV breakers were tripped during the test, Unit 1 was isolated from the grid and depended on internal sources for electrical power such as station batteries, diesel generators, and turbine generator assisted RCP coastdown.

The plant was manually tripped from 20% power using the trip pushbuttons on the Reactor Protective System Panels. Upon tripping of the turbine and generator, the electrically assisted coastdown of the RCP's did not occur as scheduled. During turbine generator (TG) assisted coastdown, the RCP's and other plant equipment are powered by the generator until coast down is terminated by either less than 80% generator output voltage or at 20 seconds following the trip whichever occurs first. TG assisted coastdown did not occur since the generator field breaker tripped immediately due to a miswired relay. The wiring of the relay has been corrected to allow TG assisted coastdown following a total loss of off-site power.

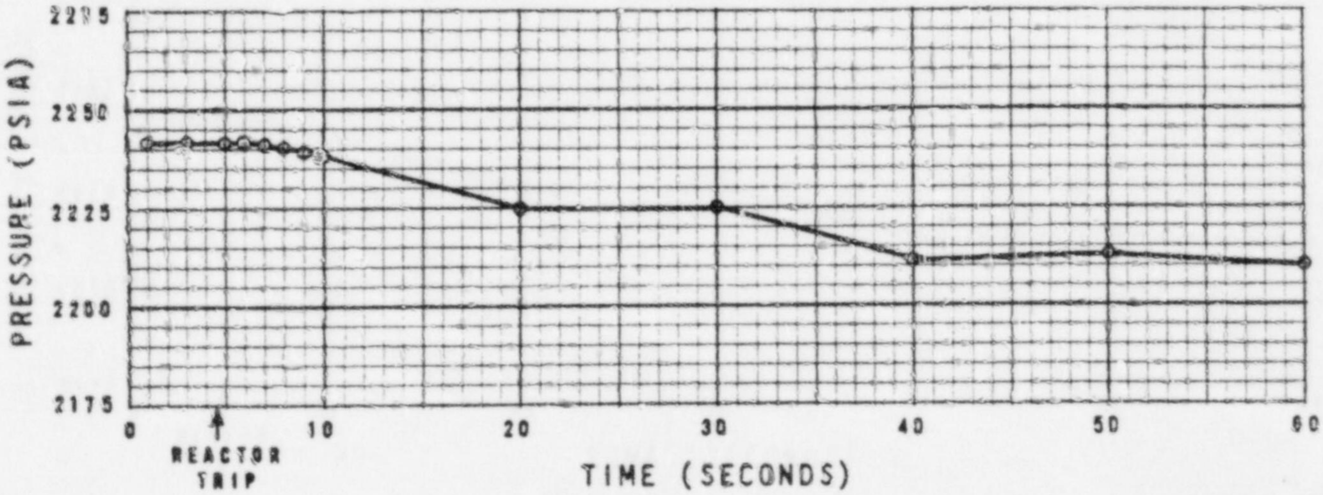
After loss of off-site power, diesel generators 11 and 12 started immediately and diesel generator 11 loaded within 10 seconds after the trip. From a post trip time of 2 seconds until approximately 10 seconds, test instrumentation recordings and observation indicated that the auxiliary boiler, instrument air compressor, atmospheric dump controller, turbine bypass valve controller and some control room instrumentation was lost.

Part of the indication, the instrument air compressor and the controller output signals were restored when diesel generator 11 picked up 11 4 KV bus. The remainder of the

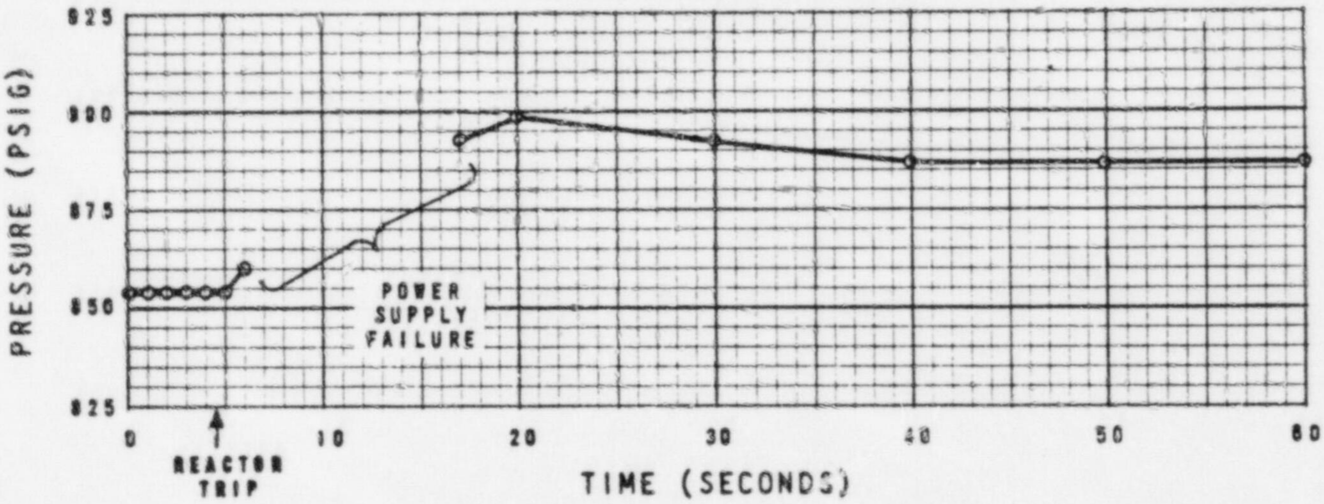
CALVERT CLIFFS UNIT 1 1ST CYCLE

20% POWER TOTAL LOSS OF OFFSITE POWER

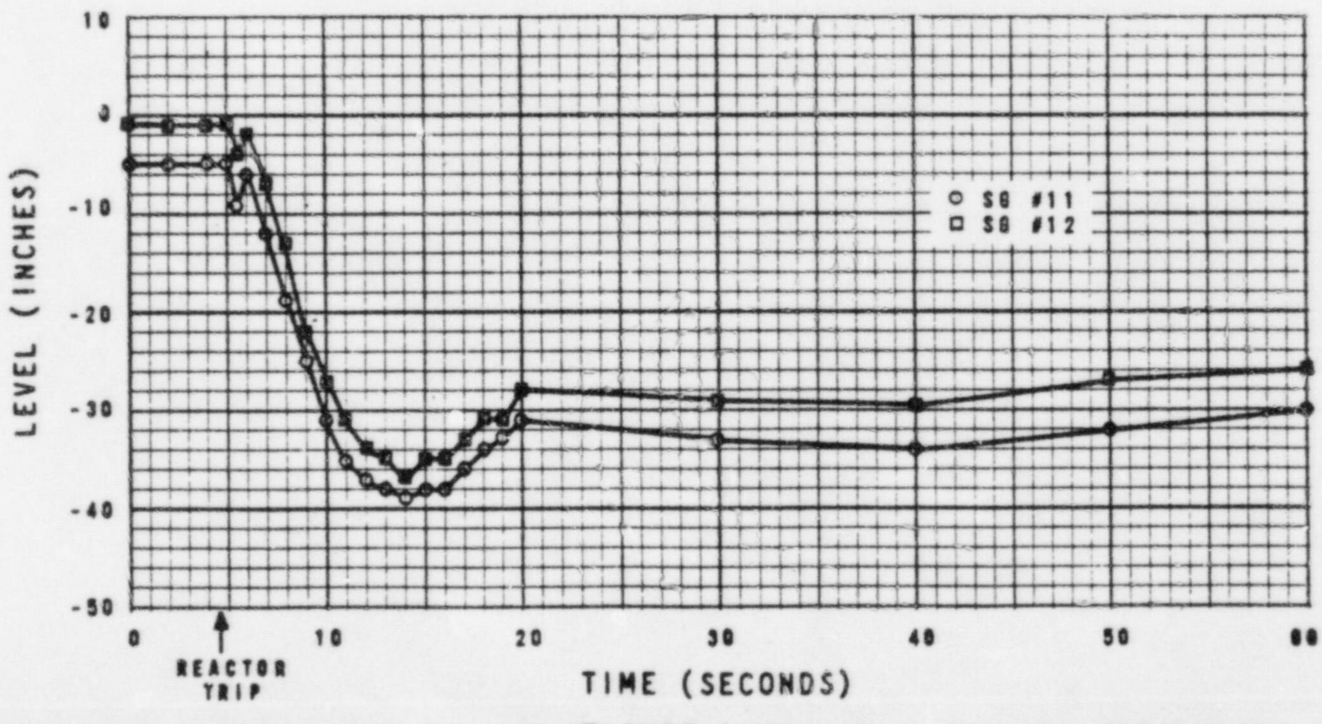
PRESSURIZER PRESSURE



STEAM GENERATOR PRESSURE



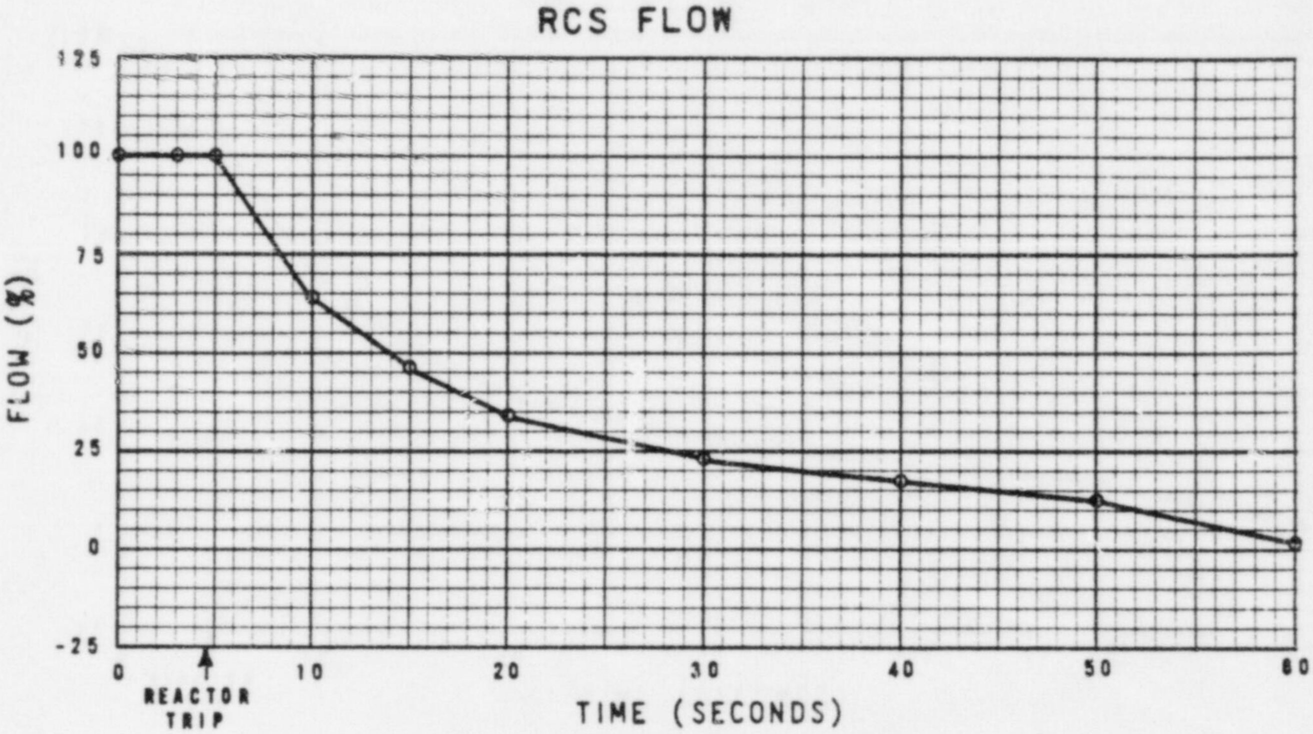
STEAM GENERATOR LEVELS

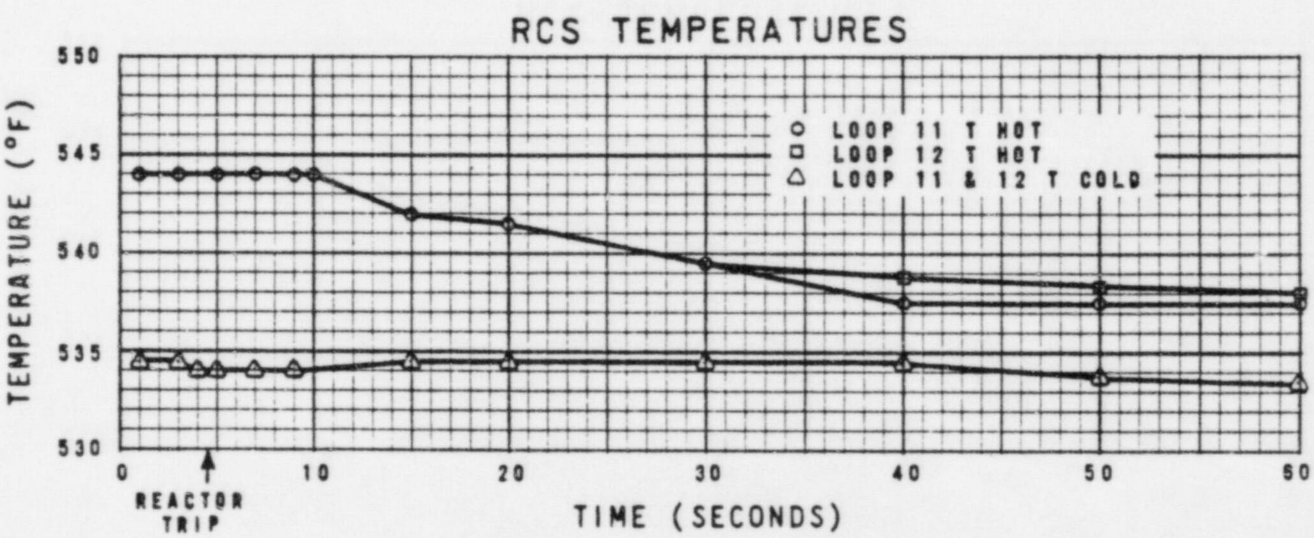


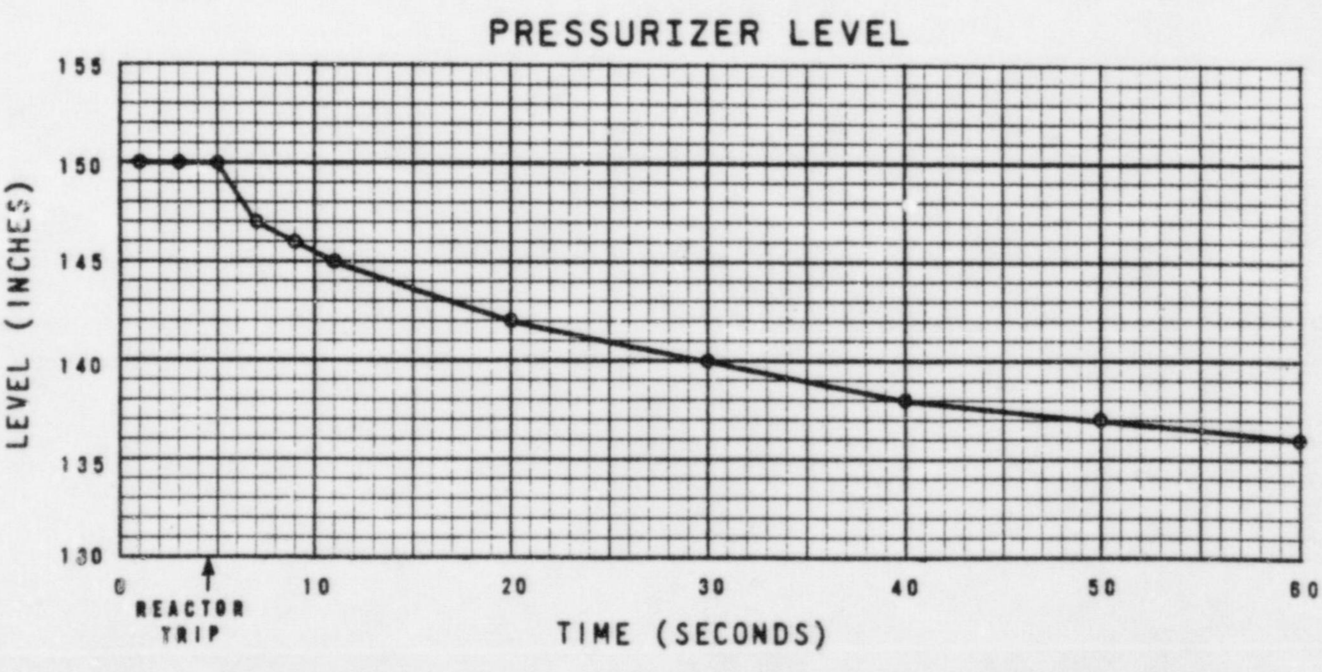
FIGURE

CALVERT CLIFFS UNIT 1 1ST CYCLE

20% POWER TOTAL LOSS OF OFFSITE POWER







6.12 Core Power Distributions

6.12.1 Purpose

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

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

(1) Fuel assembly power fraction: Fuel assembly power fraction is defined as the ratio of the average Linear Heat Rate (LHR) in a fuel assembly to the average LHR over the entire core. The measured value shall be within ± 20% of the predicted value.

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

(3) Peak linear heat rate: It shall always be less than 14.9 KW/ft.

(4) DNBR: It shall be greater than 1.3.

6.12.2 Test Results

A summary of core power distribution results is presented in Table 6.12-1.

6.12.2.1 Fuel Assembly Power Fractions. Steady state equilibrium xenon core power distribution measurements were performed at the major test plateaus of 20%, 50%, 80%, and 100% of full power. The definition of equilibrium xenon used in this section is:

(1) The change in critical boron concentration as determined from Reactor Coolant System grab sample analysis one hour apart over a four hour period is less than 5 ppm with no discernable trend; and

(2) The change in the Axial Shape Index (ASI) as determined by the incore neutron detectors, between four measurements made at one hour intervals is less than 0.001 ASI units. Figure 6.12-0 shows the location of the incore neutron detectors.

The analysis of the incore detector readings is performed by two plant computer programs. The first program automatically converts the voltage signal from the detector to the correct neutron flux level. The Incore Analysis (INCA) program converts the neutron flux levels and various other reactor parameters and, on demand, calculates several incore data. The INCA program assumes eighth core symmetry. The four cases reported in this section are compared with predictions in Figures 6.12-1 through 6.12-4. In no case does the difference exceed the acceptance criteria.

6.12.2.2 Axial Power Distributions. At steady state equilibrium xenon, the core average axial power distribution was determined at each of the four major test plateaus using the INCA program. Figures 6.12-5 through 6.12-8 are comparisons between the measured and predicted values. As can be seen from these figures, the comparison between measured and predicted distributions shows good agreement.

6.12.2.3 Peak LHR. The peak LHR is determined by the INCA program. As can be seen in Table 6.12-1, the peak LHR never exceeded the acceptance criteria of 14.9 KW/ft. If the worst case of 10.75 KW/ft is multiplied by 1.222 to account for uncertainties, the resultant 13.14 KW/ft is still acceptable. These uncertainties include measurement-calculational uncertainty, an engineering factor, effects of fuel densification and thermal expansion, and power measurement uncertainty.

6.12.2.4 DNBR. The INCA program calculates DNBR using the W-3 correlation. As noted in Table 6.12-1, DNBR did not go below the acceptance criteria of 1.3.

6.12.3 Conclusions

At steady state equilibrium xenon, the axial and radial core power distributions are within acceptable limits. The peak linear heat rate does not exceed that allowed by Technical Specifications and the minimum DNBR is greater than 1.3.

TABLE 6.12-1

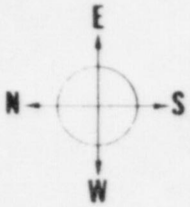
SUMMARY OF CORE POWER DISTRIBUTIONS AND CORE THERMAL CONDITIONS (AT EQUILIBRIUM XENON AND ARO[1])

	POWER PLATEAU (%)			
	20	50	80	100
Date	1/6/75	2/22/75	3/22/75	4/7/75
Time	2200	1014	0915	1715
Measured Power (%)	20.8	53.2	80.3	98.7
Core Burnup (MWD/MTU)	20.7	359.8	658.6	1024.9
Boron Concentration (ppm)	915	812	773	756
Minimum DNBR	31.29	11.57	6.61	4.73
Maximum Linear Heat Rate[2]	2.15	5.46	8.67	10.75
Maximum Peaks				
Axial	1.32	1.32	1.35	1.35
Radial	1.29	1.26	1.30	1.33

(1) ARO ≡ All Rods Out

(2) Not adjusted for uncertainty factors required by Technical Specifications.

PLANT NORTH



1-7: 4 FOLD SYMMETRIC
8-9: 3 FOLD SYMMETRIC
10: 2 FOLD SYMMETRIC
11: NON-SYMMETRIC

CALVERT CLIFFS UNIT 1
LOCATIONS OF THE INCORE NEUTRON DETECTORS
BOL, 1st CYCLE

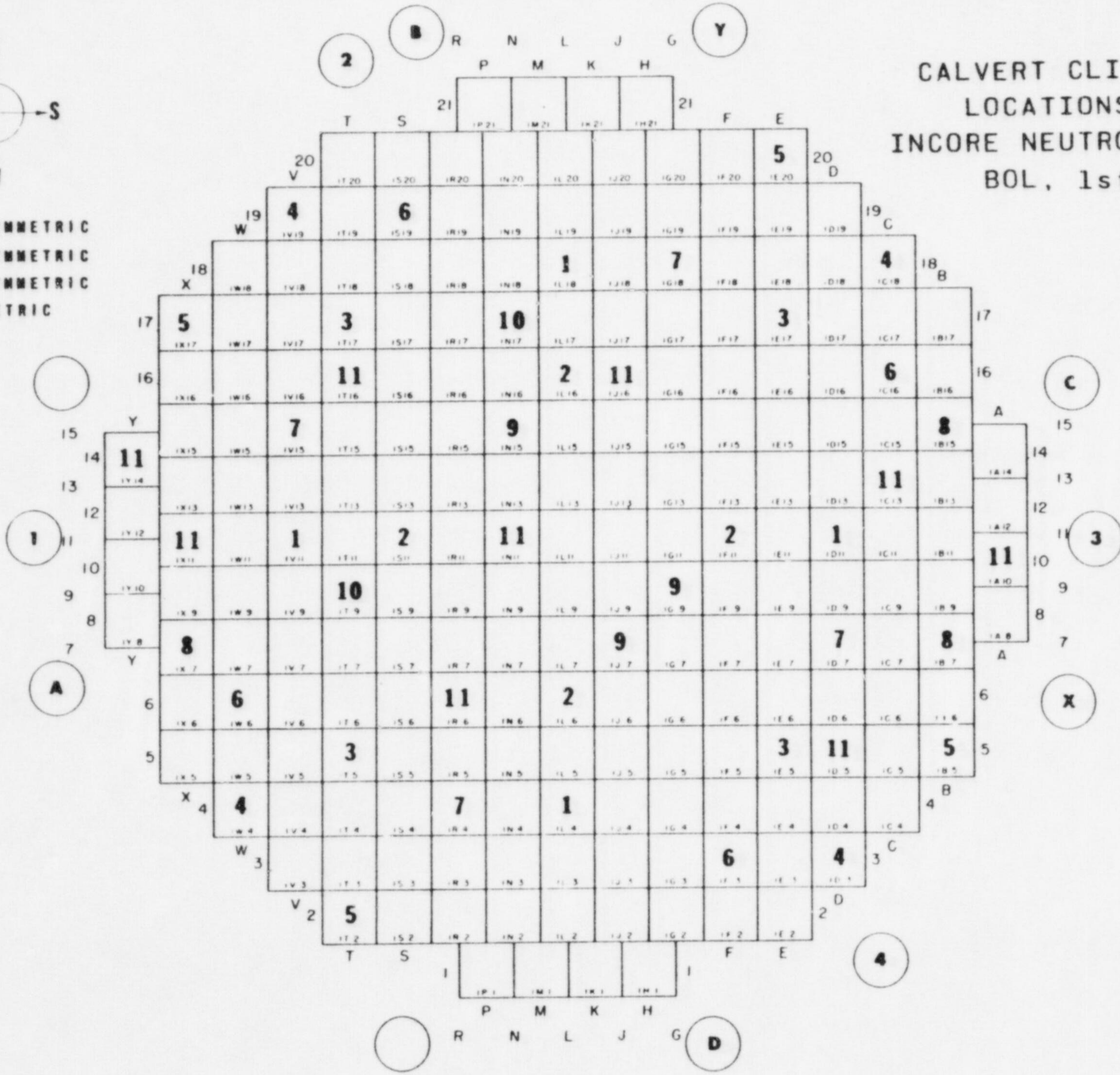


FIGURE 4.12-9

CALVERT CLIFFS UNIT 1 — 1ST CYCLE

FIGURE 6.12-1

POWER FRACTION

DATE/TIME 4-7-75/1715

POWER 2527.9 MWth

INCA BURN-UP 1024.9 MWD/MTU

ROD POSITION ARO

XENON EQUILIBRIUM

INLET TEMP. (Tc) 541.5°F

PRIMARY PRES. 2250.3 PSIA

BORONOMETER 756 PPM

BORON GRAB SAMP. 746 PPM

INITIAL

PEAK LHR 10.75 KW/FT

MIN. DNBR 4.73

INCA → X.XXX

100% X $\frac{\text{PREDICT-INCA}}{\text{PREDICT}}$ → ± XX.X

FUEL TYPE → XX

		0.659 +0.3 C	1.023 +0.3 C•					
	0.5723 +0.1 C	0.4884 +0.6 C•	0.994 -1.1 B	1.036 -0.1 A				
	0.784 +1.4 C	0.955 -0.2 B	1.022 +1.3 A	1.072 +2.2 B	1.099 +0.6 A			
	0.931 +1.1 C+	1.006 +1.2 A	1.090 -0.1 B	1.099 +1.3 A	1.171 -1.9 B	1.164 -1.2 A		
0.584 +0.8 C	1.070 +1.6 C+	1.051 +0.1 B	1.083 +1.4 A	1.142 +0.2 B	1.127 +2.5 A	1.204 -1.5 B	1.183 +0.1 A	
0.784 -1.2 C	0.934 +2.4 B	1.031 +2.0 A	1.114 +0.5 B	1.124 +1.3 A	1.183 -0.6 B	1.177 +0.2 A	1.210 -0.5 B	1.187 +0.6 A

CALVERT CLIFFS UNIT 1 1ST CYCLE

FIGURE 6.12-2

POWER FRACTION

DATE/TIME 3-22-75/0915

POWER 2056.4 MWth

INCA BURN-UP 658.6 MWD/MTU

ROD POSITION ARO

XENON EQUILIBRIUM

INLET TEMP. (Tc) 538.8°F

PRIMARY PRES. 2253.5 PSIA

BORONOMETER 773 PPM

BORON GRAB SAMP. 780 PPM

INITIAL

PEAK LHR 8.67 KW/FT

MIN. DNBR 6.61

INCA →	X.XXX
100% X $\frac{\text{PREDICT-INCA}}{\text{PREDICT}}$ →	± XX.X
FUEL TYPE →	XX

		0.662 +1.1 C	1.023 +1.3 C•					
	0.579 +0.3 C	C.992 +1.3 C•	0.991 -0.4 B	1.036 -0.1 A				
	0.792 +1.7 C	0.958 +0.2 B	1.023 +1.3 A	1.069 +2.0 B	1.098 -0.1 A			
	0.939 +1.3 C+	1.010 +1.3 A	1.086 +0.1 B	1.099 +0.7 A	1.164 -2.4 B	1.158 -1.9 A		
0.589 +1.6 C	1.075 +2.3 C+	1.048 +0.7 B	1.083 +1.0 A	1.137 -0.1 B	1.125 +1.6 A	1.190 -1.6 B	1.176 -0.7 A	
0.796 -0.9 C	0.9392 +2.9 B	1.033 +2.0 A	1.111 +0.4 B	1.122 +0.6 A	1.174 -0.9 B	1.170 -0.4 A	1.199 -0.9 B	1.181 -0.3 A

CALVERT CLIFFS UNIT 1 1ST CYCLE

FIGURE 6.12-3

POWER FRACTION

DATE/TIME 2-22-75/1014

POWER 1362.0 MWth

INCA BURN-UP 359.8 MWD/MTU

ROD POSITION ARO

XENON EQUILIBRIUM

INLET TEMP. (T_c) 536.7°F

PRIMARY PRES. 2250.9 PSIA

BORONOMETER 812 PPM

BORON GRAB SAMP. 822 PPM

INITIAL

PEAK LHR 5.46 KW/FT

MIN. DNBR 11.57

$$100\% \times \frac{\text{PREDICT-INCA}}{\text{PREDICT}}$$

INCA →	X.XXX
100% X (PREDICT-INCA)/PREDICT →	± XX.X
FUEL TYPE →	XX

		0.676 +1.7 C	1.038 +2.0 C•					
	0.593 +0.7 C	1.007 +2.1 C•	0.998 +0.1 B	1.038 -0.3 A				
	0.809 +2.3 C	0.970 +0.4 B	1.030 +0.9 A	1.066 +1.5 B	1.091 -1.0 A			
	0.956 +1.8 C+	1.019 +1.4 A	1.084 -0.1 B	1.093 -0.1 A	1.145 -2.9 B	1.137 -2.8 A		
0.608 +1.7 C	1.084 +4.0 C+	1.045 +1.8 B	1.081 +0.7 A	1.124 -0.6 B	1.110 +0.5 A	1.159 -2.1 B	1.149 -1.8 A	
0.813 +0.5 C	0.943 +4.7 B	1.035 +2.5 A	1.108 -0.1 B	1.113 -0.3 A	1.151 -1.6 B	1.145 -1.5 A	1.168 -1.9 B	1.155 -2.2 A

CALVERT CLIFFS UNIT 1 1ST CYCLE

FIGURE 6.12-4

POWER FRACTION

DATE/TIME 1-6-75/2130-2235

POWER 532.8 MWth

INCA BURN-UP 20.7 MWD/MTU

ROD POSITION ARO

XENON EQUILIBRIUM

INLET TEMP. (Tc) 534.6°F

PRIMARY PRES. 2259 PSIA

BORONOMETER 915 PPM

BORON GRAB SAMP. 934 PPM

INITIAL

PEAK LHR 2.15 KW/FT

MIN. DNBR 31.29

INCA →	X.XXX
100% X (PREDICT-INCA / PREDICT) →	± XX.X
FUEL TYPE →	XX

		0.692 +2.2 C	1.051 +3.2 C•					
	0.618 -0.4 C	1.026 +2.8 C•	1.005 +0.5 B	1.035 -0.2 A				
	0.834 +2.2 C	0.984 +0.5 B	1.031 +0.8 A	1.043 +2.6 B	1.070 -1.0 A			
0.637 +0.9 C	0.980 +2.1 C+	1.031 +1.3 A	1.077 +0.1 B	1.078 -0.4 A	1.115 -2.8 B	1.110 -3.5 A		
0.858 -0.6 C	1.119 +3.8 C+	1.051 +2.0 B	1.078 +0.1 A	1.102 -0.6 B	1.082 +0.3 A	1.119 -1.9 B	1.113 -2.3 A	
	0.992 +2.5 B	1.051 +1.7 A	1.094 +0.3 B	1.094 -0.7 A	1.111 -0.9 B	1.110 -1.8 A	1.127 -2.1 B	1.117 -2.3 A

CALVERT CLIFFS UNIT 1

1st CYCLE, EQUILIBRIUM XENON

AXIAL PEAKING FACTOR

VS ACTIVE CORE HEIGHT

20% POWER, ARO

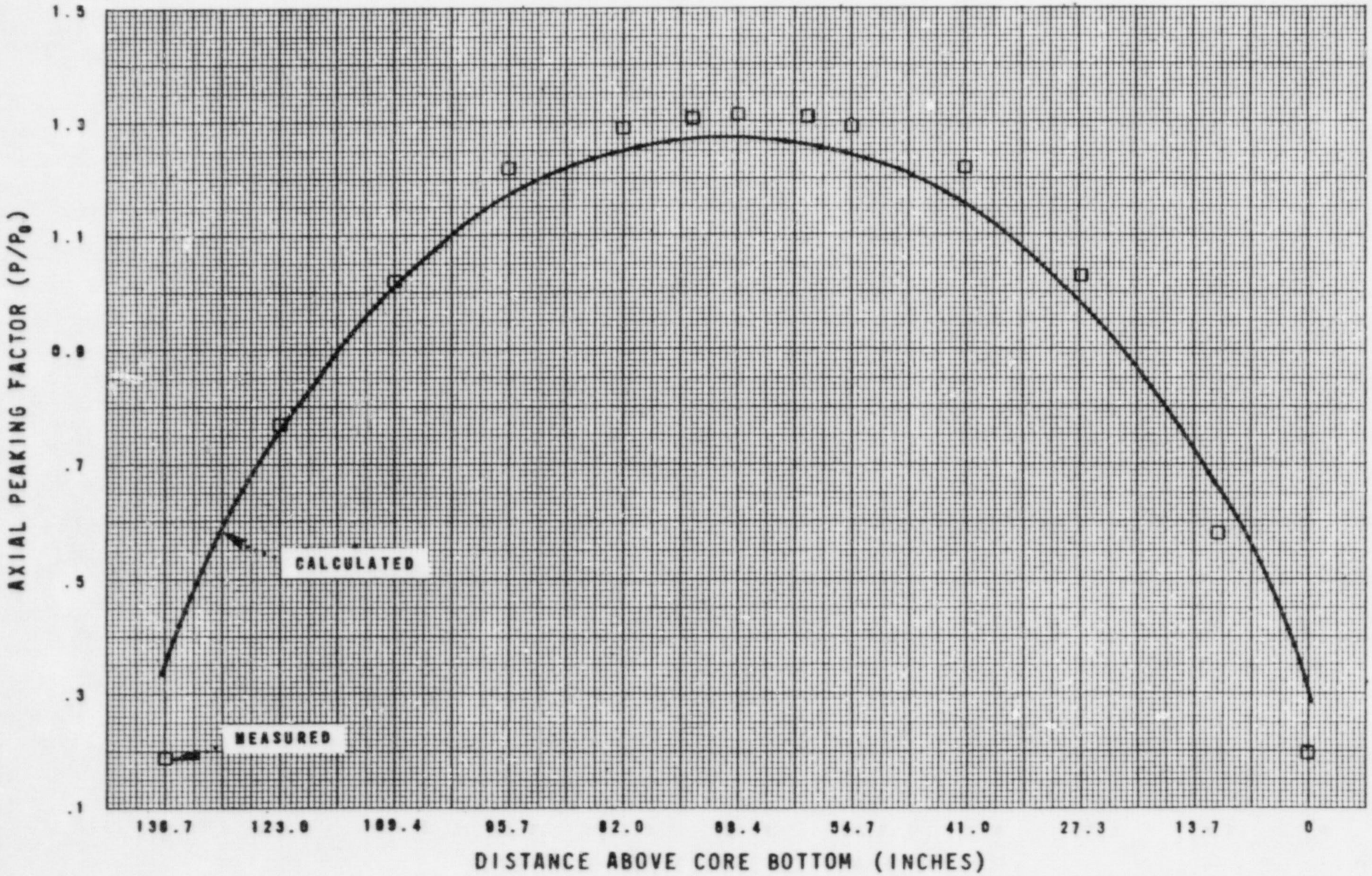


FIGURE [illegible]

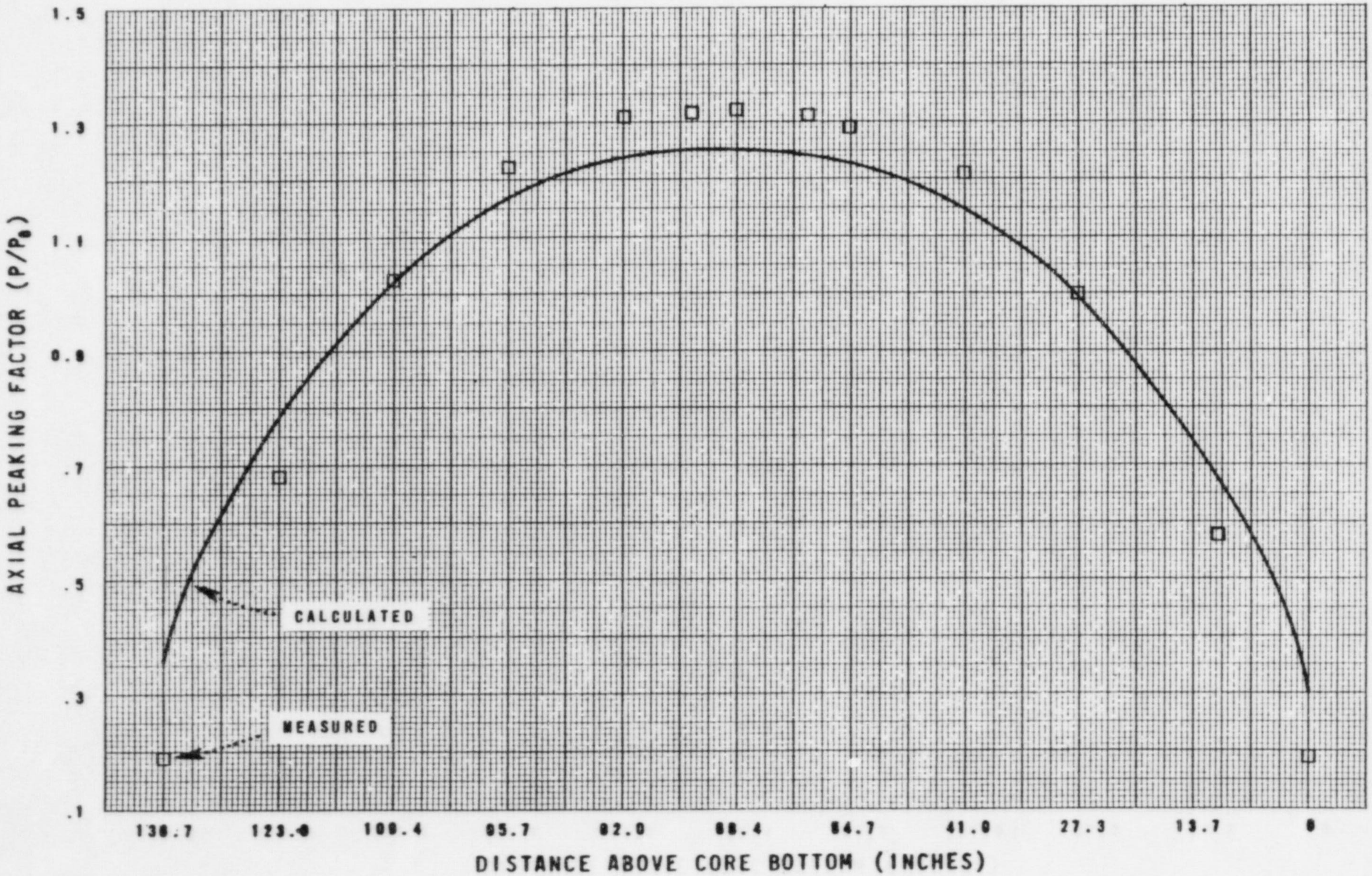
CALVERT CLIFFS UNIT 1

1st CYCLE, EQUILIBRIUM XENON

AXIAL PEAKING FACTOR

VS ACTIVE CORE HEIGHT

50% POWER, ARO



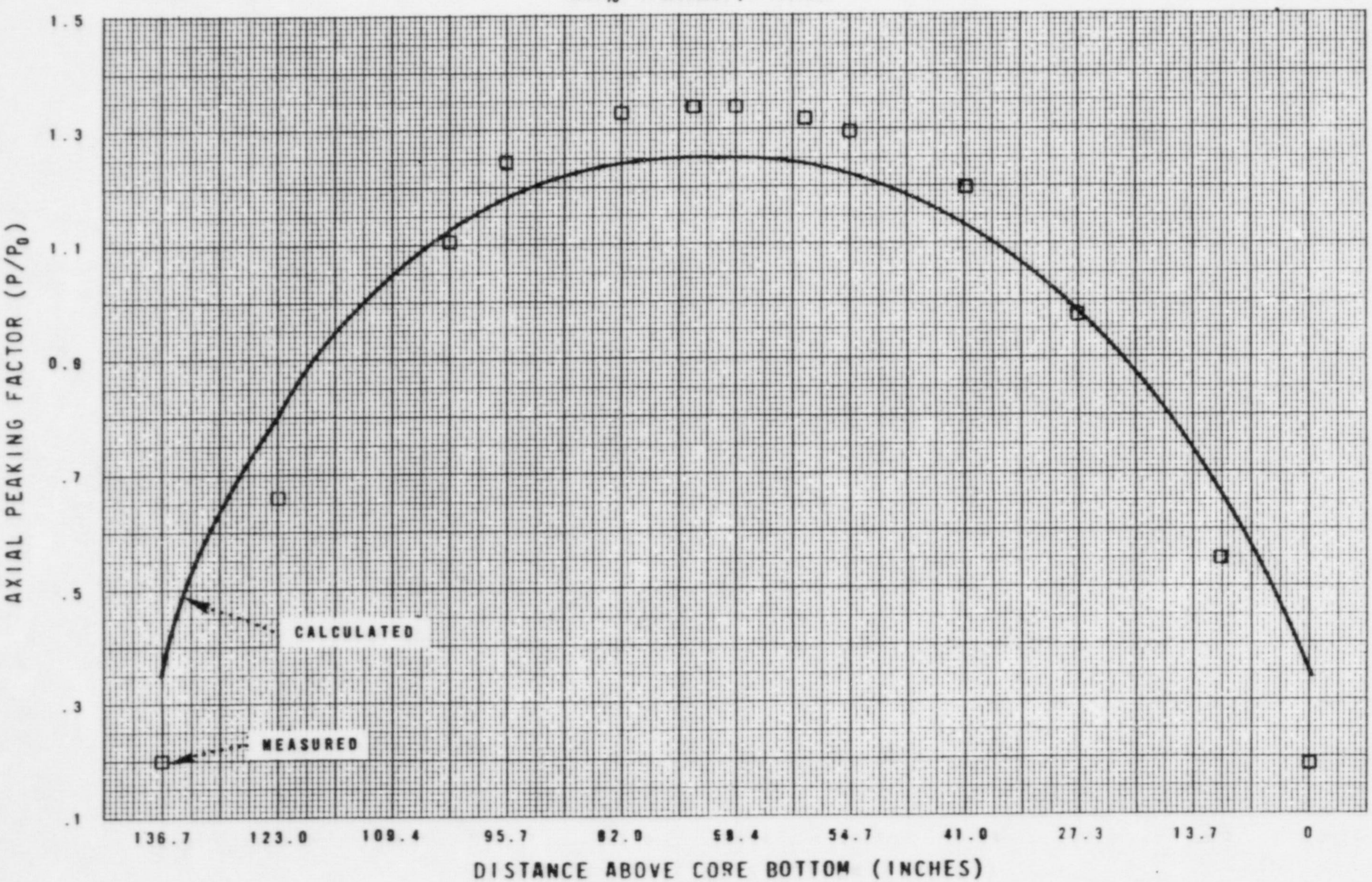
CALVERT CLIFFS UNIT 1

1st CYCLE, EQUILIBRIUM XENON

AXIAL PEAKING FACTOR

VS ACTIVE CORE HEIGHT

80% POWER, ARO



CALVERT CLIFFS UNIT 1

1st CYCLE, EQUILIBRIUM XENON

AXIAL PEAKING FACTOR

VS ACTIVE CORE HEIGHT

100% POWER, ARO

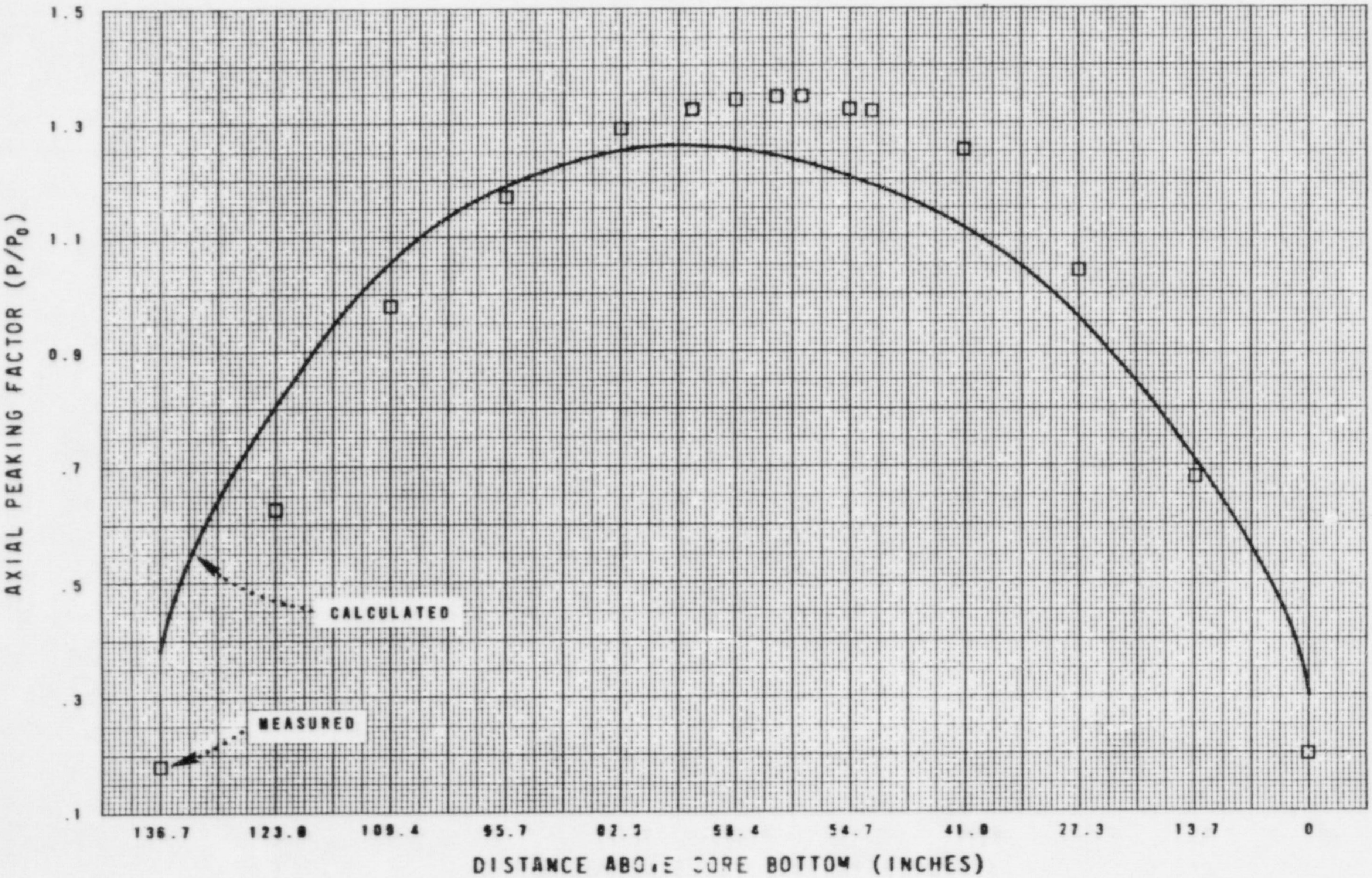


FIGURE 4.13-8

6.13 Psuedo Ejected CEA Power Distribution Measurement

6.13.1 Purpose

The purpose of this test is to measure the core power distribution resulting from a CEA being "ejected" from a full power CEA configuration inserted to the Power Dependent Insertion Limit (PDIL). The measurement results are compared with that predicted by Safety Analysis to verify its conservatism.

6.13.2 Test Results

When CEA's are inserted to the full power PDIL, only CEA Group 5 is in the core. With respect to relative reactivity worth, CEA Group 5 consists of three (3) types of CEA's. One of each of the three (3) types was "ejected" from the full power CEA configuration and appropriate power distribution information collected using the following technique. Equilibrium xenon was established at a nominal 20% power level, and then, core power distribution information gathered from the in-core neutron detectors. Each of the three (3) CEA Group 5 CEA's were "ejected" from the core; the first by a soluble boron swap technique; and the others by a rod swap with the preceding CEA. Power distribution information from the in-core detectors was gathered at the full out condition for each CEA. Power level was maintained at 20.0 ± 0.5% throughout the test.

A three-dimensional power peaking factor was developed from measured data by extrapolating in-core detector signals to the "ejected" CEA locations. As had been expected, this resulted in a power peaking factor considerably below that predicted in safety analysis. Safety analysis assumed an ejected CEA worth which was an order of magnitude greater than that measured during Low Power Physics Testing. Results of the on-site analysis for each CEA are presented in Table 6.13-1 below. In addition, measured data is receiving a more detailed analysis by CE, Windsor.

TABLE 6.13-1

3-D POWER PEAKING FACTOR RESULTING FROM PSUEDO EJECTED CEA (1)

"EJECTED" CEA NUMBER	MEASURED 3-D POWER PEAK	ACCEPTANCE LIMIT
5-1	1.64	less than 6.72
5-37	1.74	less than 6.72
5-58	1.27	less than 6.72

(1) Measurement performed at 20.0 ± 0.5% power.

6.13.3 Conclusions

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

6.14 Partial Loss of Flow Trip Test

6.14.1 Purpose

The objective of this test was to measure plant response to a partial loss of Reactor Coolant System (RCS) flow while at power, and to gather xenon reactivity worth measurement data for informational purposes only.

6.14.2 Test Results

The partial loss of flow trip was initiated by securing 12A Reactor Coolant Pump (RCP) with the reactor at 80% power. The time from pump trip to reactor trip was approximately 2.2 seconds. The trip was handled smoothly and without equipment damage. Variations in plant parameters for sixty (60) seconds following the trip are shown in Figures 6.14-1A, 6.14-1B and 6.14-2. Measurement of xenon worth was delayed because 12A RCP could not be restarted immediately due to low oil pressure. The oil pressure problem was traced to a faulty check valve. After the valve was cleaned and oil pressure restored, the RCP was restarted and the reactor taken critical for collection of xenon reactivity worth measurement data.

The incremental difference in reactivity worth between equilibrium and peak xenon was determined by following the xenon buildup until the peak value was reached. The xenon worth was calculated from RCS soluble boron concentrations at the peak and the equilibrium xenon conditions just prior to reactor trip.

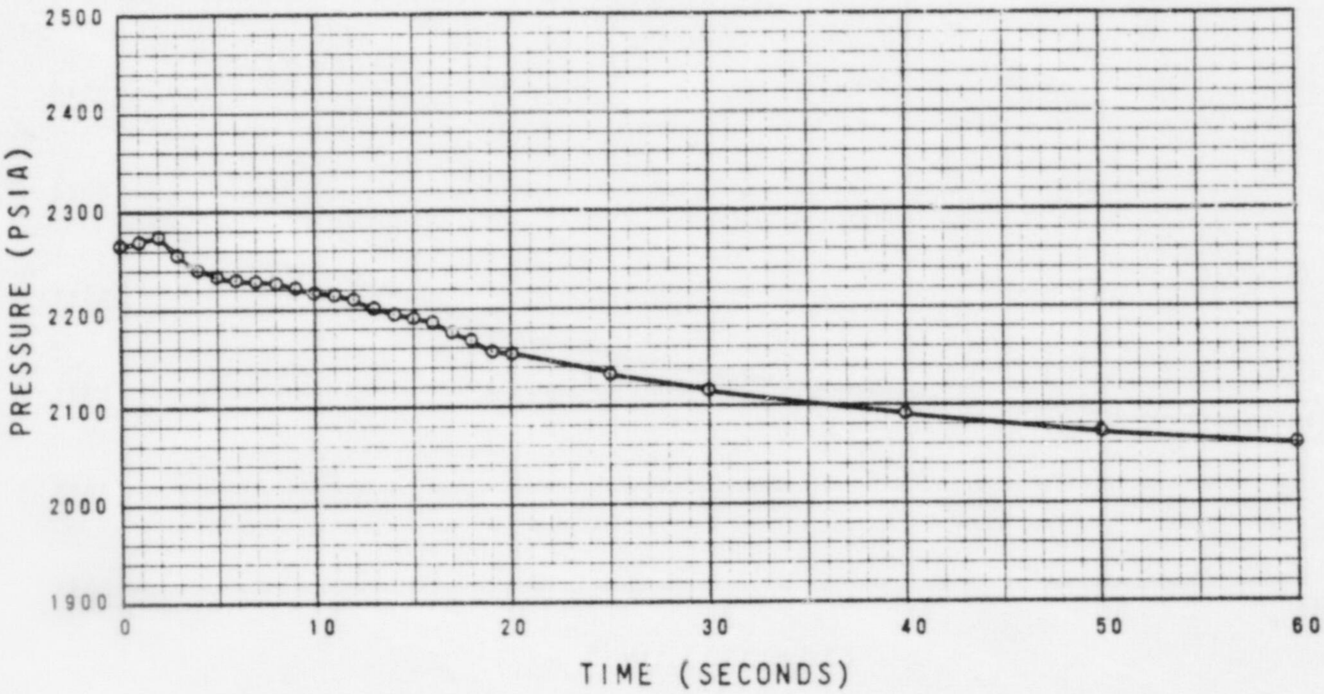
6.14.3 Conclusions

The partial loss of flow trip proceeded smoothly and satisfactorily without any equipment damage. Xenon reactivity worth measurement data was collected.

CALVERT CLIFFS UNIT 1 1ST CYCLE

80% PARTIAL LOSS OF FLOW TRIP

PRESSURIZER PRESSURE



PRESSURIZER LEVEL

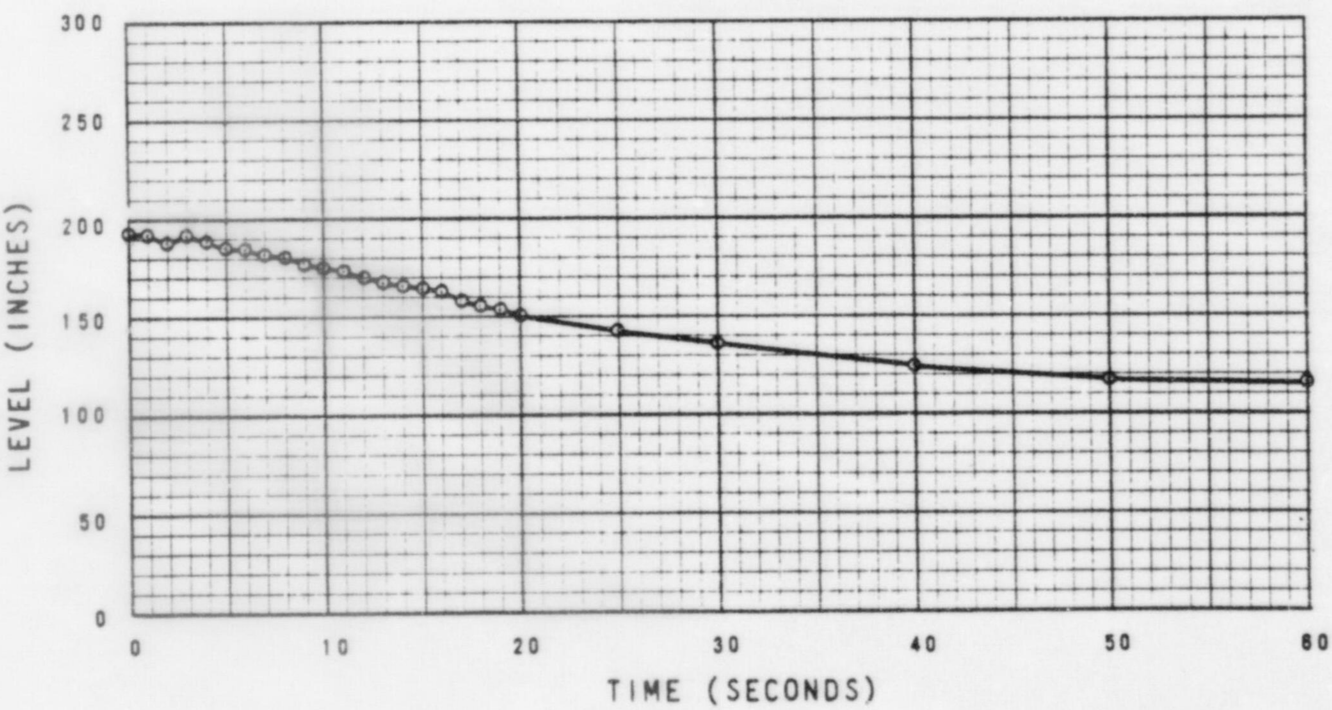
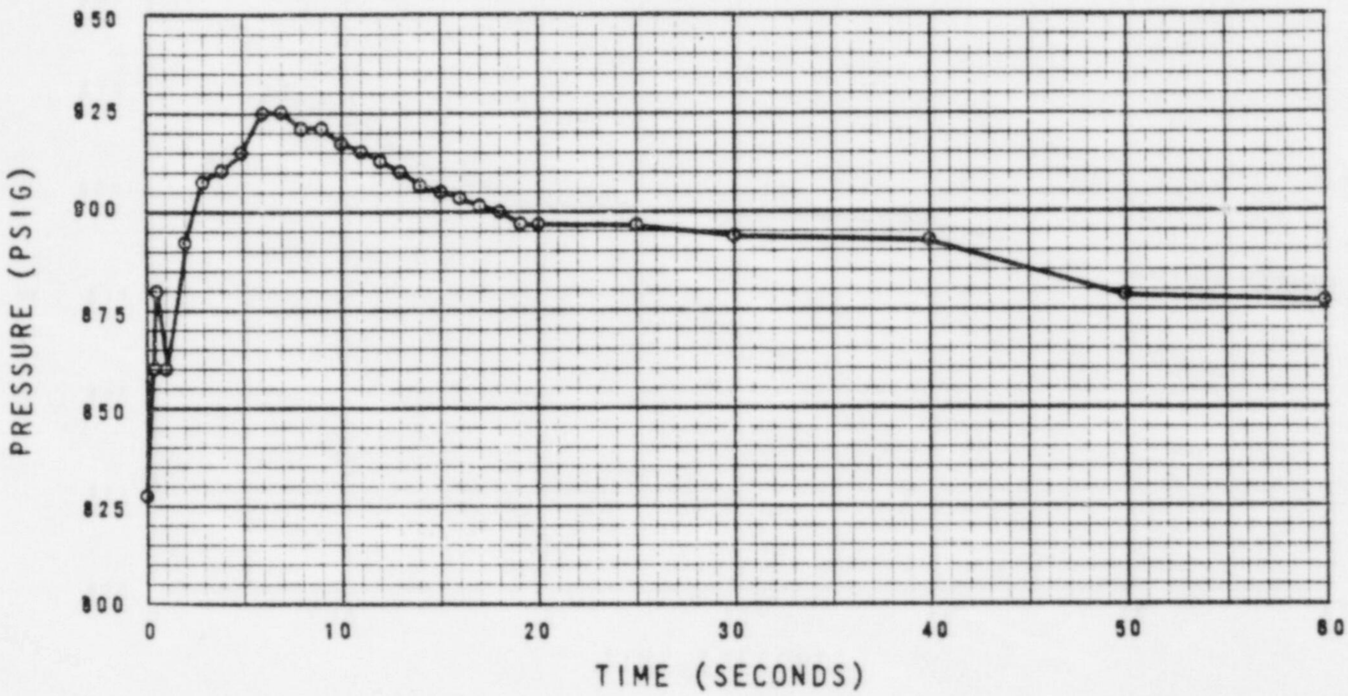


FIGURE 6.14-1A

CALVERT CLIFFS UNIT 1 1ST CYCLE

80% PARTIAL LOSS OF FLOW TRIP

SG #11 HEADER PRESSURE



STEAM GENERATOR LEVELS

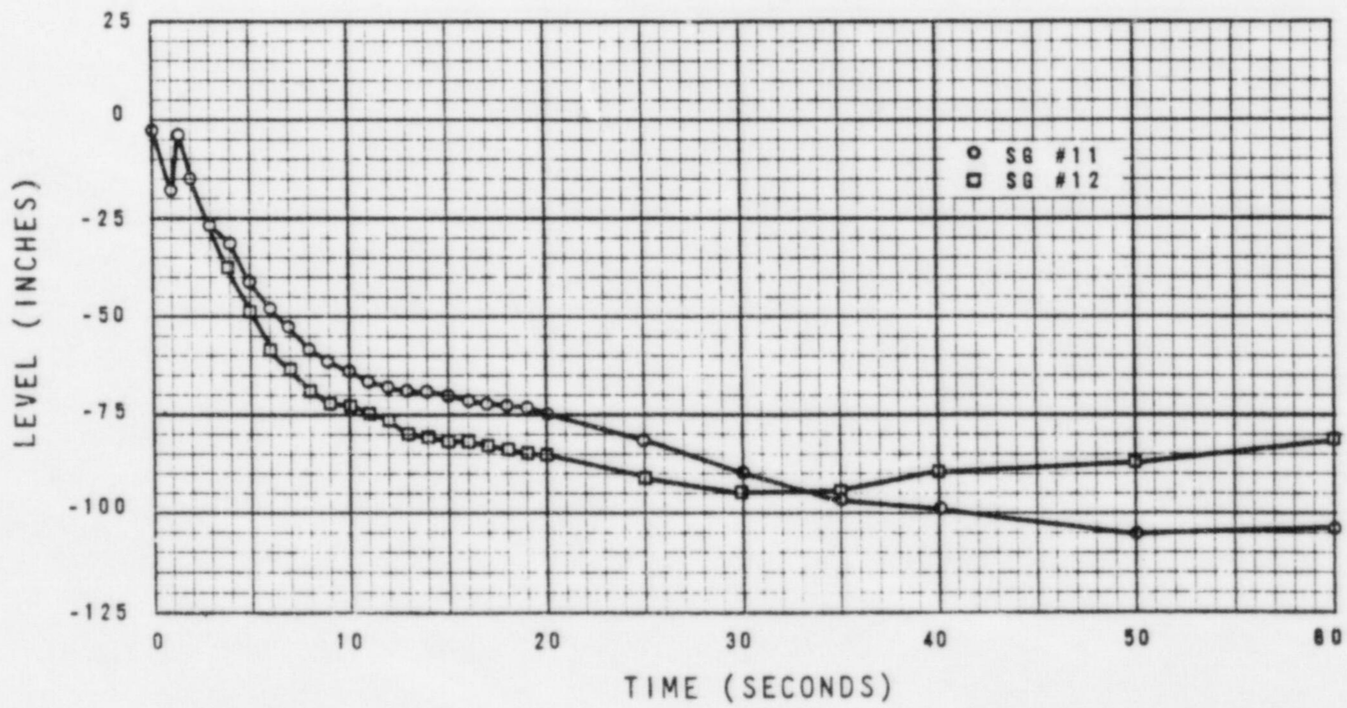
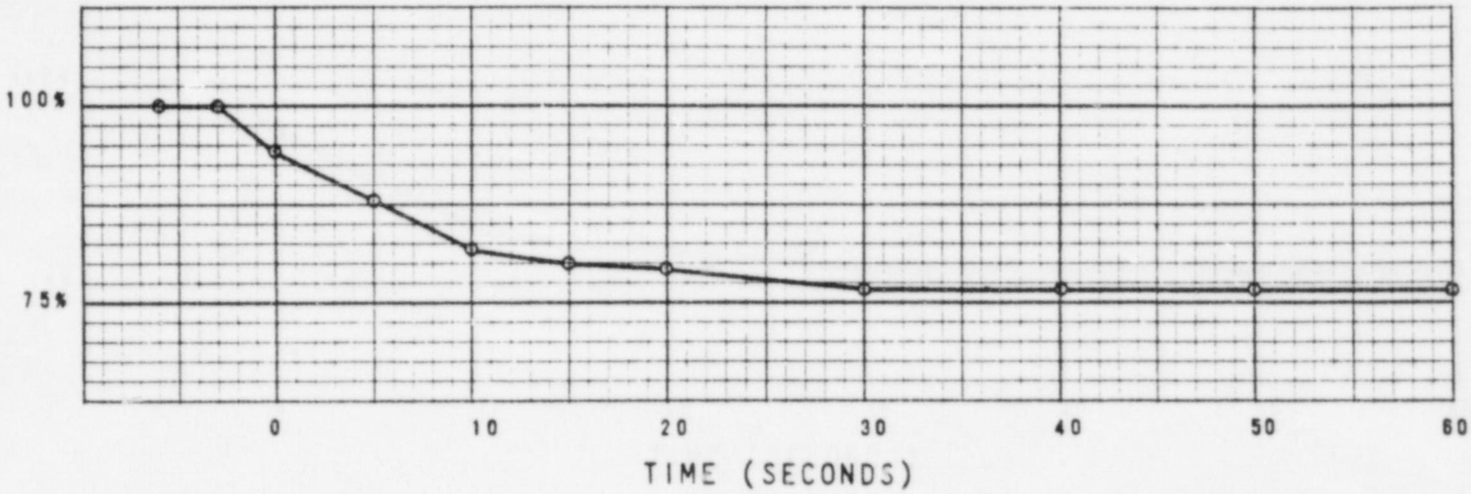


FIGURE 6.14-1B

CALVERT CLIFFS UNIT 1 1ST CYCLE

80% PARTIAL LOSS OF FLOW TRIP

RCS TOTAL FLOW



RCS LOOP TEMPERATURES

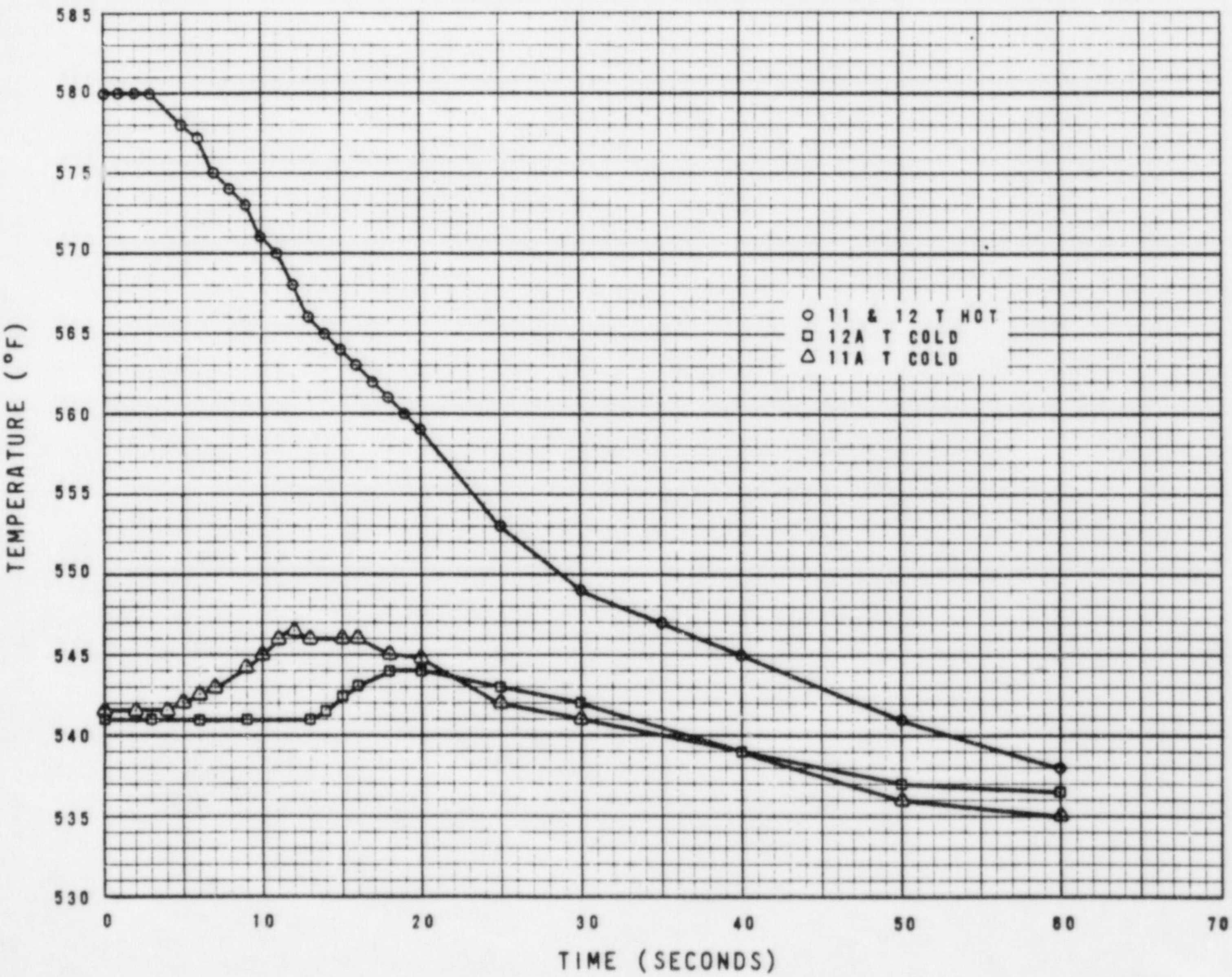


FIGURE 6.14-2

6.15 Total Loss of Flow/Natural Circulation Test

6.15.1 Purpose

The objective of the test was to monitor plant response to a total loss of flow while at power and to determine the power to flow ratio during natural circulation.

6.15.2 Test Results

The total loss of flow trip was conducted from 40% power. It was initiated by tripping the breaker supplying power to all four Reactor Coolant Pumps (RCP). The reactor trip proceeded smoothly without equipment damage. Variations of plant parameters for sixty (60) seconds following the trip are shown in Figures 6.15-1A, 6.15-1B, and 6.15-2.

Natural circulation was verified about five (5) minutes after the trip. Verification was based on the behavior of the Reactor Coolant System (RCS) hot and cold leg temperatures (T HOT and T COLD, respectively). Indication that cooler water from the cold leg was reaching the hot leg was signalled by the decrease in T HOT which was a good indication that sufficient natural circulation was indeed occurring.

The power to flow ratio was obtained by two different methods. The first method involved securing feed to the steam generators and adjusting the turbine bypass valve controller to provide constant T COLD for one hour while steaming down the steam generators. This data indicated that 13.09 MWth of decay heat was removed from the RCS. This amounted to 0.51% of rated power. The core flow calculated to remove the 13.09 MWth of heat was 9200 GPM or 2.29% of total flow. This produced a power to flow ratio of 0.22.

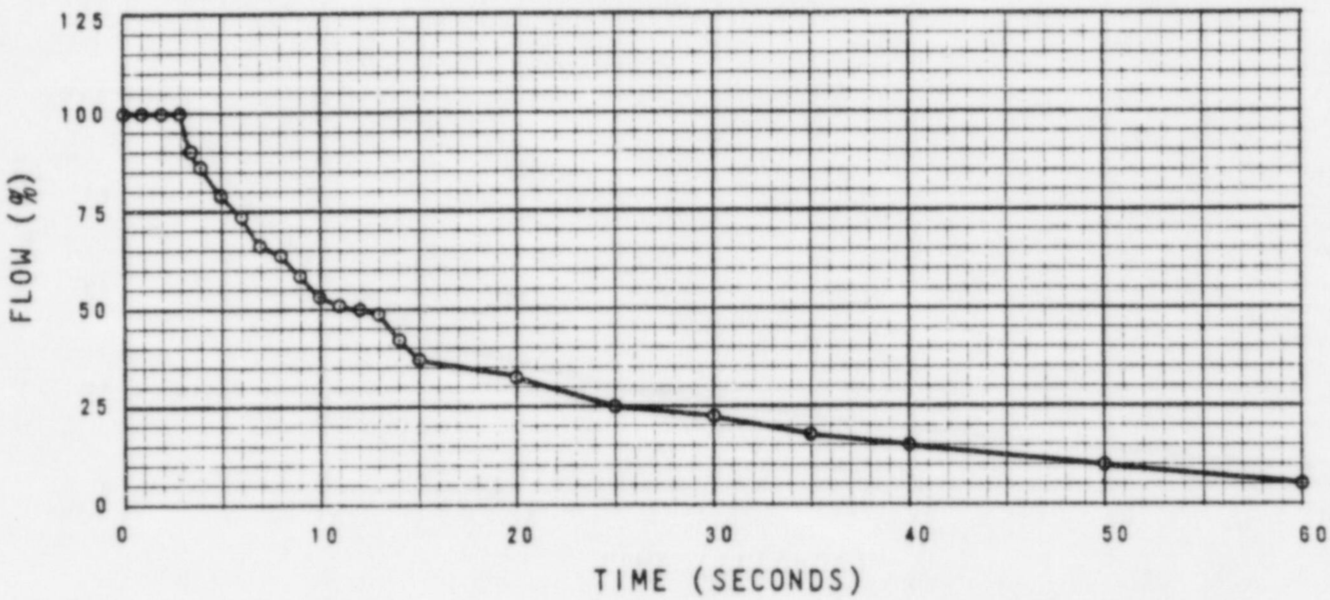
The second method of calculating the power to flow ratio involved the ΔT Power readout. Under steady-state conditions of natural circulation with significant flow, the ratio of power to flow can be directly obtained from the ΔT Power reading. During the time interval that natural circulation data was taken, ΔT Power read approximately 20.7%, so the power to flow ratio was .207. Variations of plant parameters during the natural circulation portion of the test are shown in Figures 6.15-3A and 6.15-3B.

6.15.3 Conclusions

The 40% power total loss of flow trip proceeded smoothly. Natural circulation was verified approximately five (5) minutes after the trip. A power to flow ratio of approximately 0.22 was measured, which is well within the acceptance criterion of less than 1.0.

40% TOTAL LOSS OF FLOW TRIP

RCS FLOW



RCS LOOP TEMPERATURE

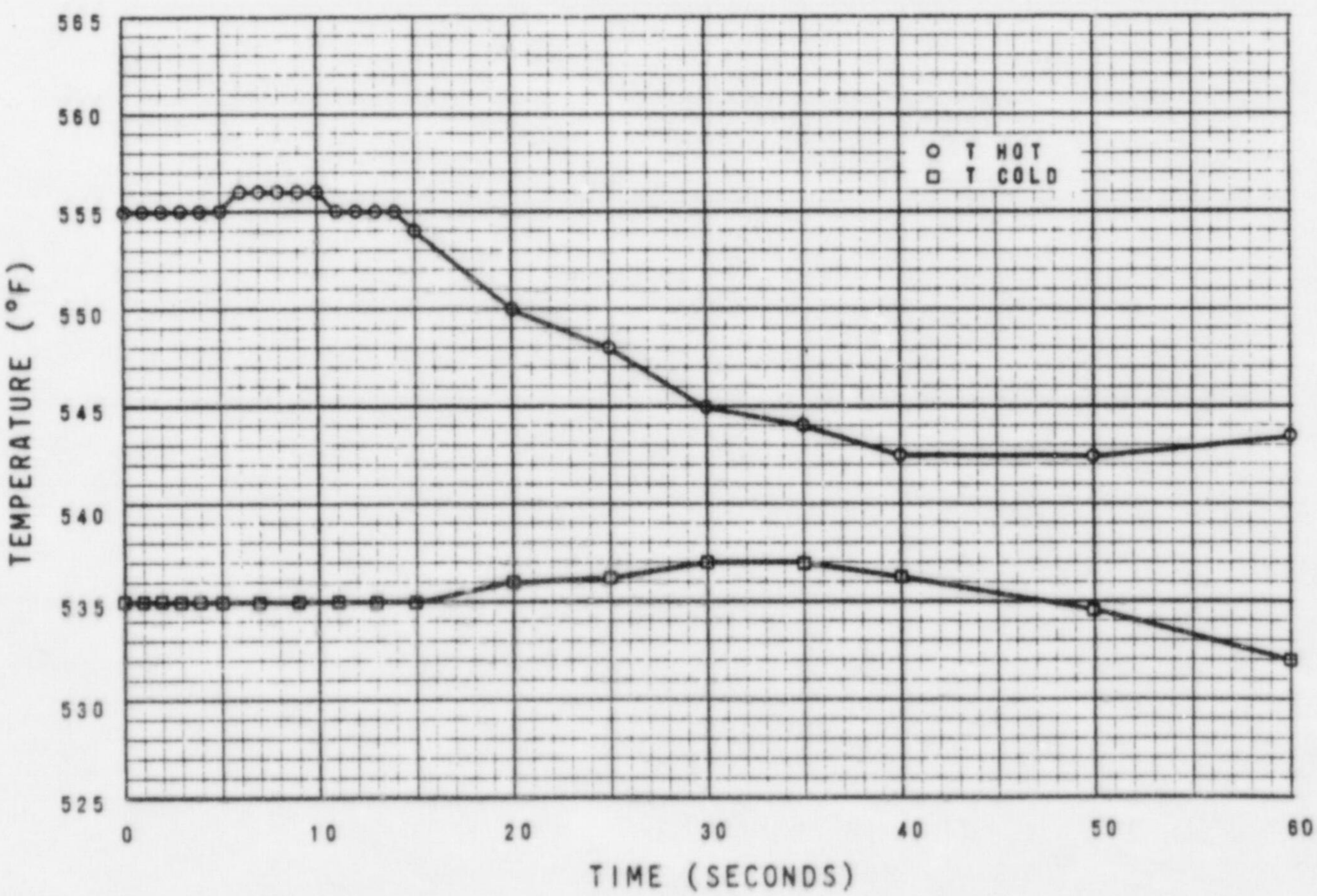
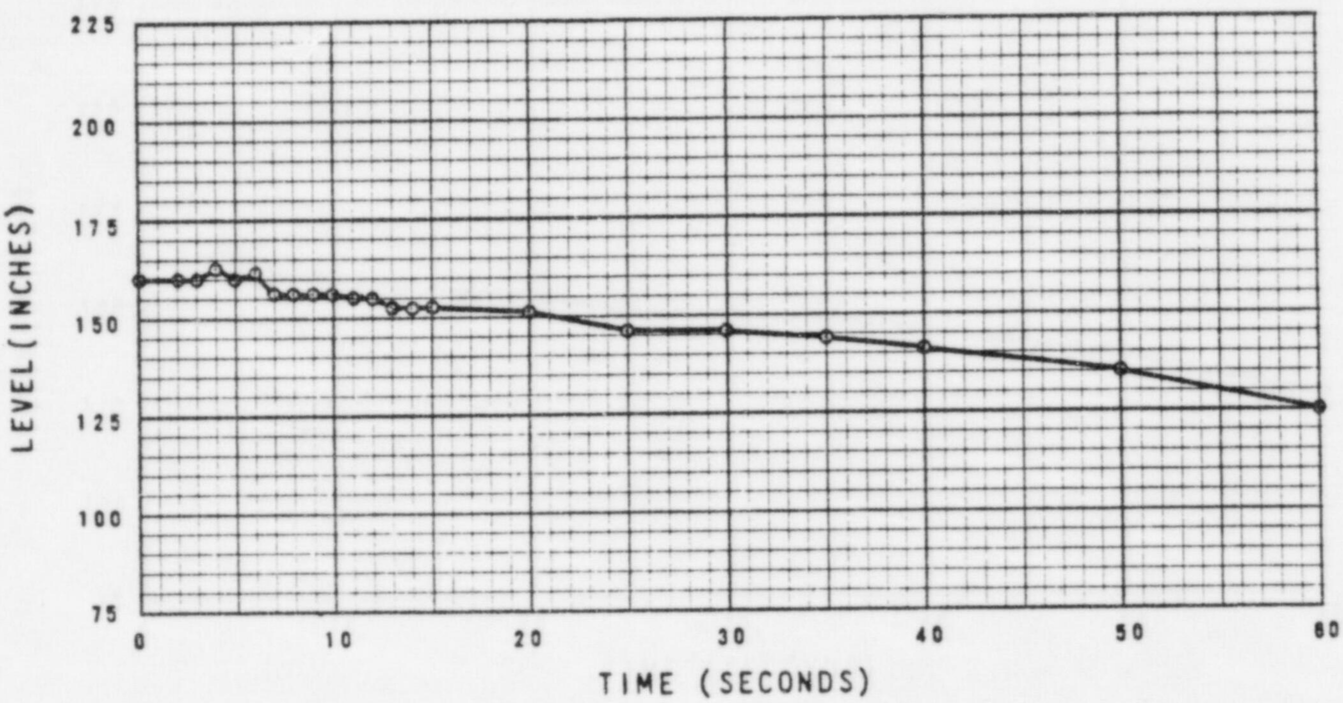


FIGURE 6.15-1A

40% TOTAL LOSS OF FLOW TRIP

PRESSURIZER LEVEL



PRESSURIZER PRESSURE

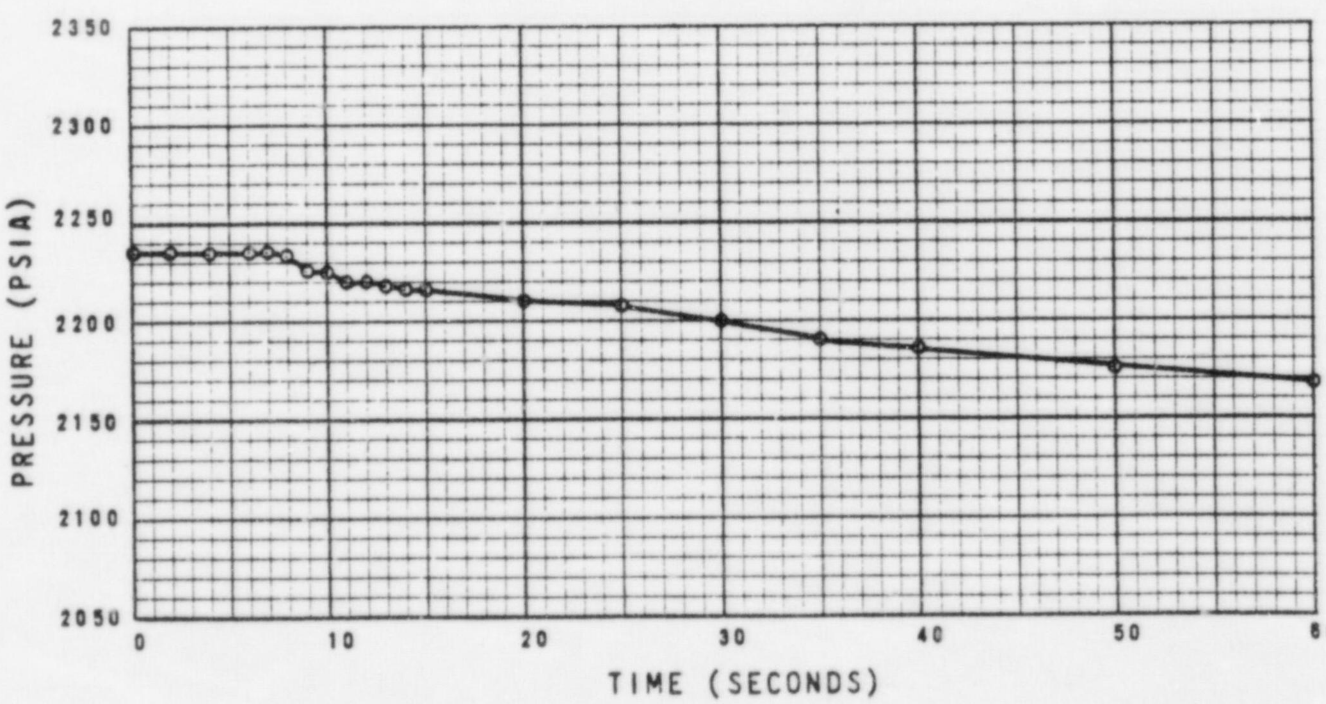
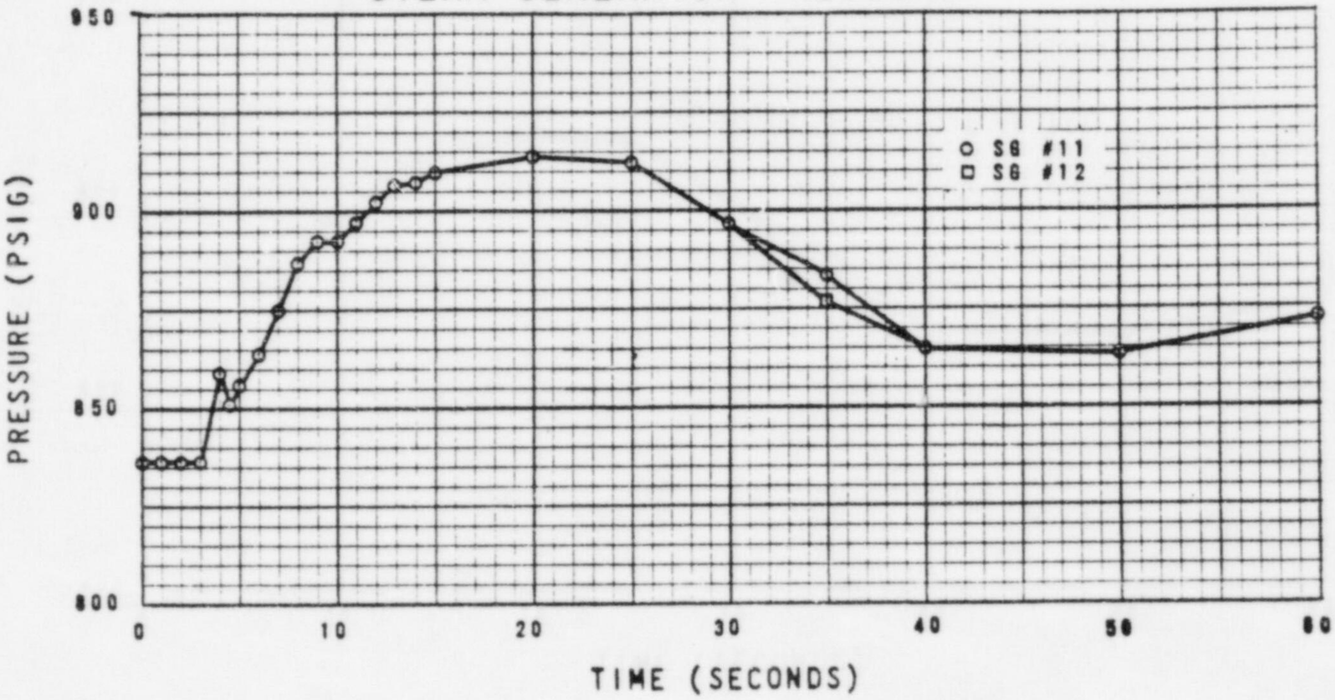


FIGURE 6.15-1B

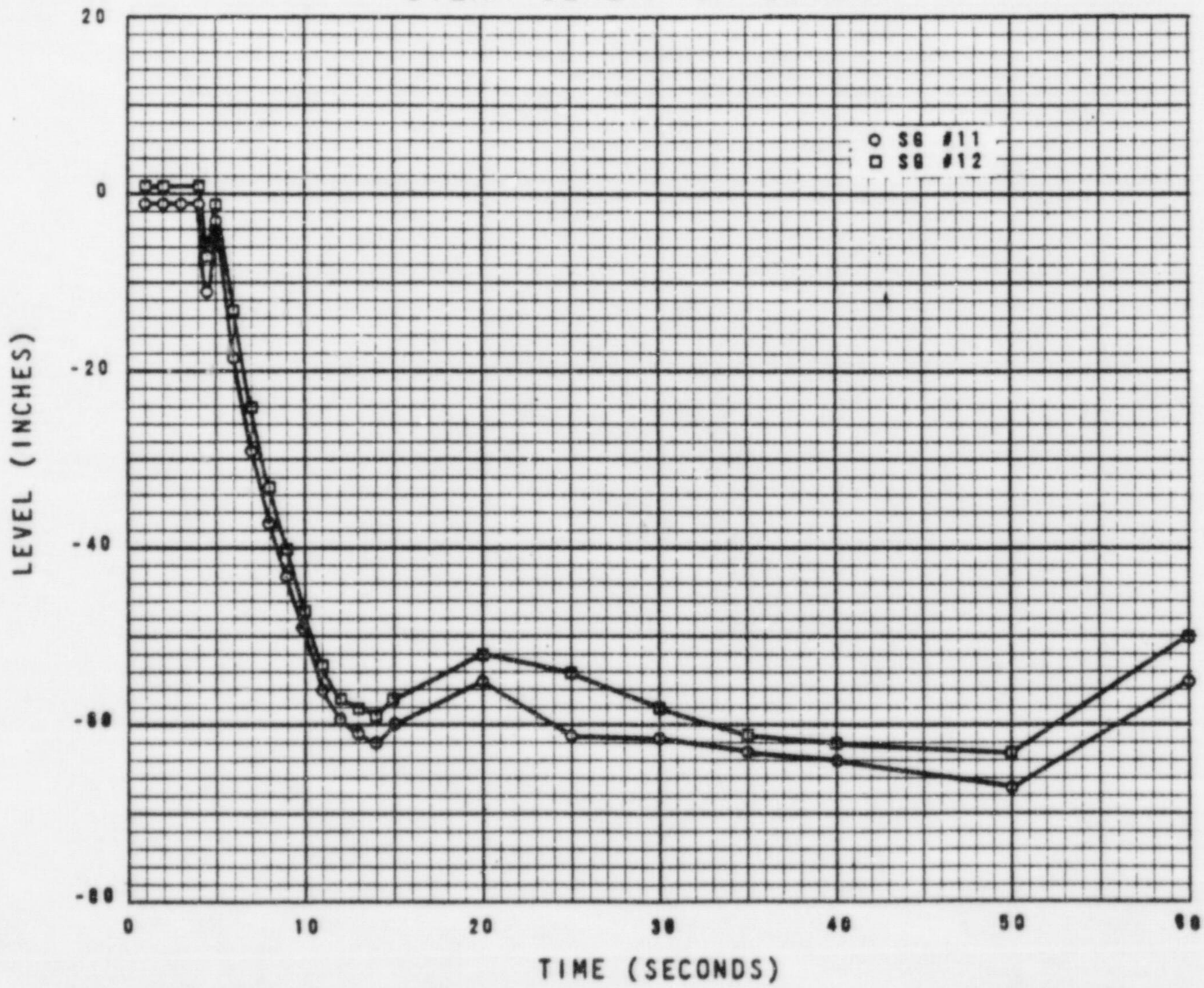
CALVERT CLIFFS UNIT 1 1ST CYCLE

40% TOTAL LOSS OF FLOW TRIP

STEAM GENERATOR PRESSURE



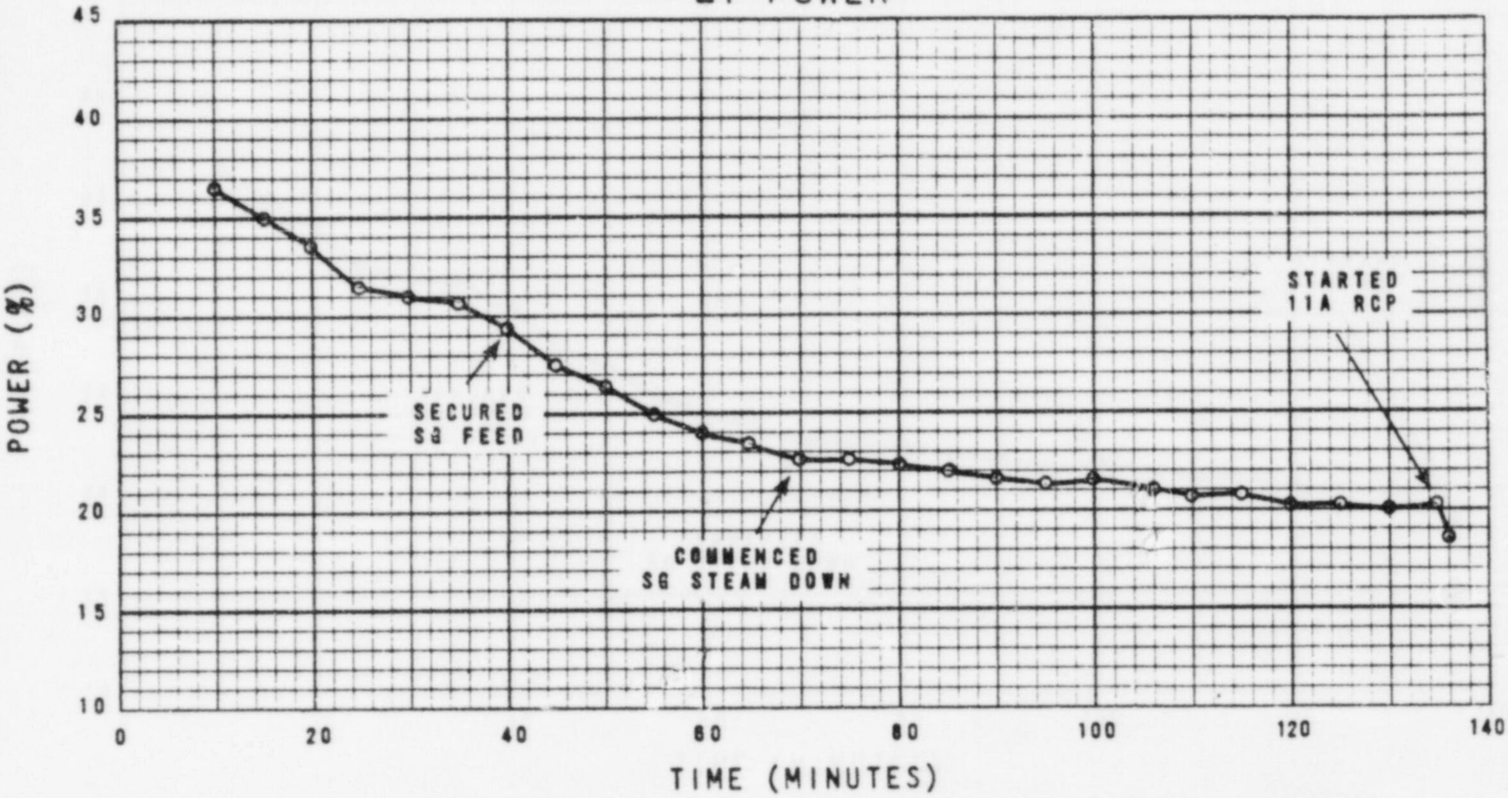
STEAM GENERATOR LEVELS



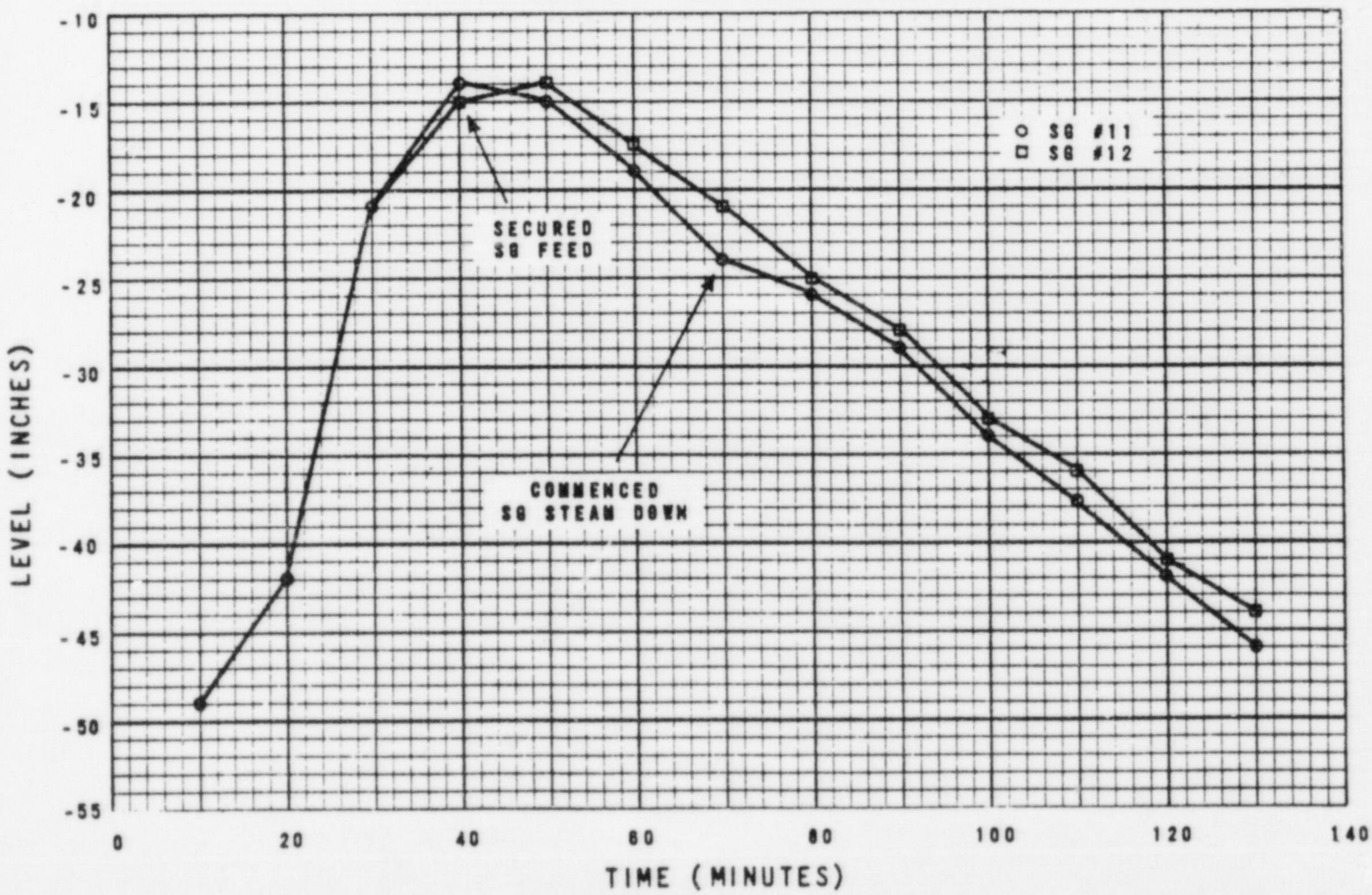
CALVERT CLIFFS UNIT 1 1ST CYCLE

NATURAL CIRCULATION TESTS

ΔT POWER



STEAM GENERATOR LEVELS



NATURAL CIRCULATION TESTS

RCS LOOP TEMPERATURE

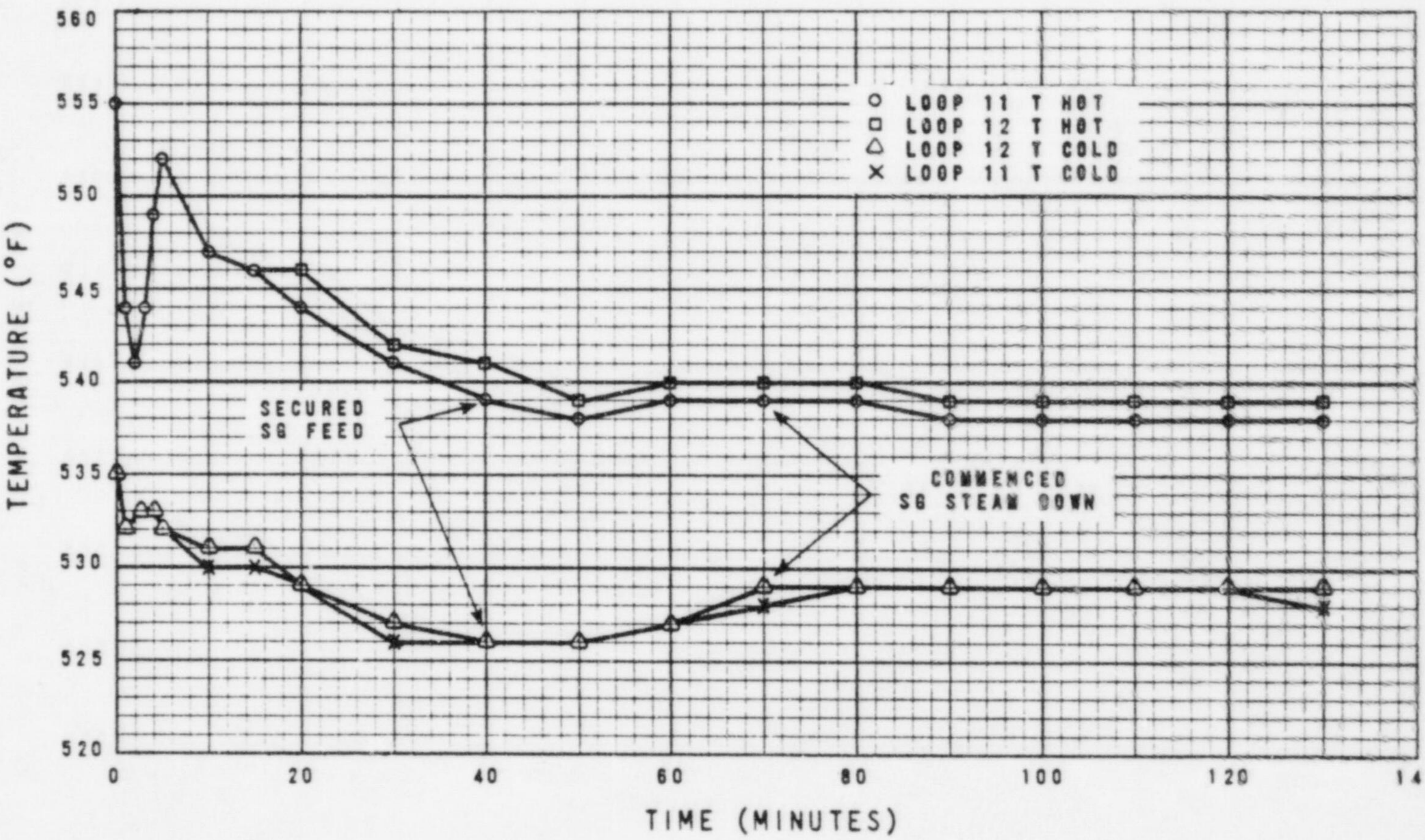


FIGURE 6.15-3B

6.16 Chemistry and Radiochemistry Tests at Power

6.16.1 Purpose

The objectives of the Chemistry and Radiochemistry Tests were as follows:

(1) To ensure that the reactor and secondary systems water chemistry met the criteria of Calvert Cliffs Radiation and Chemistry Procedures.

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

(3) To verify that moisture carryover from each steam generator is less than the design maximum of 0.2%.

6.16.2 Test Results

Chemistry and radiochemistry tests specified by the plant operating procedures were conducted at all power levels and at intervals specified in the procedure. The tests specified in objectives (2) and (3) above were conducted at power levels of 20%, 50%, 80%, and 100%, and during transients from 0-25%, 0-50%, 0-55%, and 0-80% power. Table 6.16-1 provides a detailed breakdown of the chemistry tests at various power levels.

The routine chemistry and radiochemistry results on the Steam Generators, Condensate, Feedwater, Main Steam and Reactor Coolant Systems were either acceptable or, when appropriate, corrective action was instituted to achieve acceptable conditions. The moisture carryover test was conducted at 80% and 95% power. Results show that carryover is less than 1%, which is about the lowest that can be detected using sodium tracers.

Corrosion data and fission product buildup tests were conducted for informational purposes only. Results indicate that during 100% power testing, lithium levels increased at about 0.1 ppm lithium per day and operation of a CVCS ion exchanger with the cation resin in the hydrogen form provided good control over lithium levels. Corrosion products remained at generally a low level during 100% power testing. Fission product buildup and in particular iodine levels remained low indicating defect free fuel until approximately one week before the end of EPT. At that time, the failed fuel detector and chemical analysis indicated that the first fuel had failed. Analysis of gas samples from the pressurizer and the VCT also indicated no fuel failure until the last week of EPT. Decontamination factors on the purification system were reviewed and found acceptable with the exception of gross beta-gamma which has been lower than expected.

6.16.3 Conclusions

Chemistry and radiochemistry tests normally indicated that water chemistry met specifications. When out of specification readings occurred, corrective action was taken to achieve acceptable conditions. No delays in the test program resulted from chemistry tests. Baseline data on radiochemistry was collected prior to the first fuel failure which occurred about one week prior to the end of the test program.

TABLE 6.16-1

CHEMISTRY AND RADIOCHEMISTRY TEST SCHEME

TEST	POWER LEVEL (%)					
	20	50	80	95	100	TRANSIENT
Corrosion Product Testing	X	X	X		X	
Fission Product Buildup			X		X	
Lithium Buildup			X		X	
Purification System Efficiency	X	X	X		X	
Moisture Carryover	X(baseline)		X	X		
VCT Gas Testing			Transient Tests Only			0-25%, 0-50% 0-55%, 0-80%
Pressurizer Gas Testing			Transient Tests Only			0-25%, 0-50% 0-55%, 0-80%

6.17 Effluent Monitoring System Test

6.17.1 Purpose

The purpose of the test was to compare the predicted response of the Radiation Monitoring System liquid and gas waste discharge monitors with the actual response during liquid and gaseous releases.

6.17.2 Test Results

The Radiation Monitoring System (RMS) response for liquid releases conducted early in the Escalation To Power Tests (EPT), when the amount of activity in the Reactor Coolant System (RCS) was low, was calculated using vendor calibration data and appeared to agree with actual responses. However, as the amount of activity in the RCS increased, it became apparent that the calculated response was lower than the actual RMS response. Investigation revealed that the calculated response, based on the vendor's data for Cobalt 58, was in error by about a factor of two (2). In subsequent releases, the response for Cobalt 58 calculated using vendor data was adjusted by a factor of two (2), giving a new predicted response. This new predicted response agreed favorably with actual release responses. Tables 6.17-1 and 6.17-2 show data taken during releases throughout EPT. Calculated responses are those obtained by using vendor calibration data. Predicted values are those obtained by applying the Cobalt 58 correction factor to the calculated value.

Gaseous release responses were predicted using vendor calibration curves. The agreement between predicted and actual responses for gaseous releases was found to be acceptable.

6.17.3 Conclusions

During EPT, agreement between RMS actual and predicted responses was found to be acceptable after a correction factor to the vendor supplied Cobalt 58 data was applied. Agreement between Radiation Monitoring System actual and predicted responses for gaseous releases was found to be acceptable.

TABLE 6.17-1

LIQUID WASTE RELEASES

PERMIT #	PREDICTED (CPM)	CALCULATED (CPM)	ACTUAL (CPM)	PERMIT	PREDICTED (CPM)	CALCULATED (CPM)	ACTUAL (CPM)
R-1-75	24±12	24	20	R-13-75	60±30	44	85
R-2-75	24±12	24	21	R-14-75	100±50	32	85
R-3-75	24±12	24	19	R-15-75	110±55	39	80
R-4-75	24±12	24	20	R-16-75	110±55	32	60
R-5-75	24±12	24	25	R-17-75	110±55	26	150
R-6-75	24±12	24	20	R-18-75	200±100	180	265
R-7-85	24±12	24	25	R-19-75	180±90	76	175
R-8-75	36±18	36	50	R-20-75	1660±830	664	1100
R-10-75	350±175	130	350	R-21-75	1000±500	250	575
R-11-75	350±175	130	300	R-22-75	424±212	169	390
R-12-75	350±175	125	300	R-23-75	250±125	104	275
				R-25-75	295±147	147	270
				R-27-75	169±85	84	180
				R-28-75	148±74	74	200
				R-29-75	110±55	55	125
				R-30-75	240±120	120	250
				R-31-75	480±240	240	500
				R-32-75	188±94	94	200

TABLE 6.17-2

GASEOUS WASTE RELEASE

PERMIT #	PREDICTED (CPM)	ACTUAL (CPM)	REMARKS
G-1-75	100±50	90	
G-2-75	30±15	30	containment purge
G-3-75	100±50	50	
G-4-75	150±75	175	
G-6-55	95±47	25	containment purge
G-7-75	100±50	90	containment purge
G-8-75	150±75	200	
G-9-75	200±100	200	containment purge
G-12-75	120±60	120	
G-13-75	250±125	200 max	

6.18 Process Computer Measured Values Check

6.18.1 Purpose

The purpose of this test was to compare process computer readings with meter readings for selected safety related plant parameters.

6.18.2 Test Results

Process computer readings were compared to meter readings at the 80% and 100% plateaus. The comparison included reactor power; Reactor Coolant System (RCS) temperatures, pressures, levels and flows; boronometer and CEA position. The comparison showed that the process computer values were generally always within acceptance criteria.

6.18.3 Conclusions

The process computer will record values for instruments it reads in close agreement with the meter value for the same plant parameter. The computer readouts can be used for all plant operating functions where critical instrument readouts are required.

7.0 UNUSUAL EVENTS

During the time interval covered by Startup Testing, several major problems occurred which interrupted and significantly delayed testing until they were either temporarily or permanently resolved. Those problems and their resolutions are discussed in more detail below.

7.1 Reactor Vessel Differential Pressure Increase

Differential pressure sensing devices are installed across the steam generator inlet and outlet nozzles; the purpose of which is to supply Reactor Coolant System (RCS) flow information to the Reactor Protective System (RPS). In addition, other components of the RCS are heavily instrumented with differential pressure (dp) sensing devices, including local dp cells and remote reading dp transmitters across all four (4) Reactor Coolant Pumps (RCP), reactor vessel and portions of the RCS hot and cold legs.

On 11/2/74 during Low Power Physics Testing (LPPT), dp instruments began to show a slow but steadily increasing pressure drop across the reactor vessel. By late on 11/5/74, it had increased approximately three (3) psi from a nominal value of about thirty-eight (38) psi. The increase was recorded on three (3) separate dp instruments; a fourth being out of service at the time. The reactor was shutdown at approximately 1800 on 11/5/74, and RCS flow reduced by selective securing of RCP's.

Over the next several days, increased radiochemical sampling and analyses were performed by BG&E and CE, Windsor personnel. See Section 5.13.3 for some additional details. Crud buildup was suspected, but RCS water samples did not indicate an increase in the already very low RCS particulate levels. On 11/12/74, the decision was made to cool down, disassemble the reactor vessel and visually inspect the reactor vessel cavity, internals, and fuel for mechanical flow blockage and crud buildup.

The inspection began on 11/19/74. Six (6) fuel assemblies were removed from the core to the spent fuel storage area. One (1) fuel assembly was inspected using remote system TV. Three (3) fuel assemblies were inspected visually from behind leaded glass shielding. Gross beta-gamma readings on contact were less than 100 rem/hr. Fuel assemblies were sufficiently cool thermally to allow the inspection to be conducted in air. Swipes were taken of fuel assembly surfaces and analyzed by BG&E and CE, Windsor.

A remote TV monitoring system was used to inspect the faces of the fuel assemblies surrounding the hole in the core left by the six (6) removed fuel assemblies. The same system was also used to inspect the downcomer annulus, the flow holes in the flow skit, the underneath surface of the core support plate, and the top surface of the core support plate in the vicinity of the hole in the core. There was no hint of mechanical blockage, and cleanliness of surface and clarity of water was excellent. Swipes and visual inspections of fuel assembly, upper guide structure, and reactor vessel head surfaces revealed a barely perceptable coating of iron and zirconium oxide less than one (1) mil thick.

Although the reason for the dp increase was and still is not completely resolved, it was evident that mechanical flow blockage had not occurred. On that basis, the reactor vessel was reassembled and LPPT recommenced on 12/7/74. During the latter part of 12/74 and during 1/75, reactor vessel dp slowly decreased to levels comparable to those experienced prior to the increase on 11/2/74. The dp has remained at that level to date. A preliminary evaluation is that the observed increase in dp was caused by ever so slight roughening of fuel rod and other reactor internal surfaces precipitated by thin oxide film buildup. Continued operation has tended to smooth the film, thereby decreasing dp to original levels.

7.2 Uncoupled CEA Incident

CEDM/CEA operational checks performed subsequent to reactor vessel reassembly after the reactor vessel dp increase incident revealed the possibility of several unlatched CEA's and CEA's not connected to their CEA extension shafts and, therefore, not capable of movement by their respective CEDM's. Subsequent core asymmetry checks on all CEA's confirmed that CEA's 5-1, 5-54, and PLR1-13 were uncoupled. Plant cooldown and reactor vessel disassembly began on 12/11/74.

During the earlier CEDM/CEA operational checks, which included CEA drop time determination, the CEA extension shafts had been unsuspectingly dropped onto the hub of their respective CEA's. The extension shafts were removed and visually inspected for damage. Some slight indentation of the grapple fingers was noted. A borescope was used to inspect the surfaces of the CEA shrouds which may have been contacted by a falling extension shaft. No damage was noted. A borescope inspection of the CEA hubs revealed some errupted metal and shavings on the inside surfaces of the hubs. A special tool was devised and used to perform in-place lapping of the hubs and removal of shavings. All three (3) extension shafts were replaced.

Simultaneous with the above, an extensive review of the procedure and technique for performing and verifying the grappling of CEA extension shafts to CEA's was performed and new procedures instituted. The reactor vessel was reassembled and LPPT recommenced on 12/22/74.

7.3 Higher Than Predicted Containment Radiation Levels

During the 20% power test plateau of the Escalation to Power Testing (EPT), radiation measurements taken outside Unit 1 containment structure equipment and personnel hatches indicated that dose rates would exceed the 0.5 mrem/hr limitation specified in the FSAR for the outside surface of the containment structure. In addition, higher than anticipated radiation levels were indicated at a number of locations inside containment, where access on an infrequent basis is required during operation. Shortly after finding that radiation levels were higher than expected, temporary shielding was placed outside of the containment structure equipment hatch to reduce the radiation level to less than 0.5 mrem/hr at power levels up to about 20% of full power. The temporary shielding consisted of concrete logs and drums filled with borated water. Areas which remained greater than 0.5 mrem/hr were roped off to limit access.

EPT continued while an investigation of the cause of and solution for the high radiation levels was conducted. A preliminary analysis of the measured radiation levels and a review of the existing shield arrangement indicated that the high radiation levels may have been caused by neutron and gamma radiation streaming out of: (1) the annulus between the reactor vessel flange and the primary shield wall, (2) the annulus around the reactor coolant piping where it penetrates the primary shield wall, and (3) to a lesser extent, the access opening through the lower part of the primary shield. The annulus between the reactor vessel and the primary shield is approximately 2-1/2 feet wide, and radiation in the vessel cavity scattered by the vessel wall and primary shield concrete streams out of the large gap.

Neutron dose rates of the magnitude measured in the containment were indicative of a neutron flux of sufficient magnitude to result in a significant source of gammas from capture of thermal neutrons in components made of steel. Conservative calculations indicated that a thermal flux of this magnitude incident on steel can result in a gamma dose from neutron capture of about 300 mrem/hr. Therefore, shield modifications directed toward reduction of the neutron dose rate were considered to be of primary importance; the necessity to provide shielding for gammas could be better ascertained from dose rates measured after the neutron shield modification was made.

On 1/27/75, EPT was suspended and the plant was shutdown and cooled down for installation of temporary neutron shielding. Shielding consisted of bagged crystalline boric acid (H_3BO_3) and polyethylene sheets stacked on support grating specially erected to span the 2-1/2 feet annular gap between the reactor vessel flange and primary shield. Basically, the shield consisted of approximately 750 bags of H_3BO_3 stacked five bags high directly over the gap and tapering down to a single layer at the outer edge of the shield. A 6-inch thick ring of polyethylene, consisting of six 1x24x36-inch sheets cut to form an annular segment, spanned the reactor vessel insulation and flange areas to attenuate neutrons which would otherwise stream out of the insulation gap. The polyethylene was arranged as four 1-inch layers of virgin polyethylene followed by two 1-inch layers of borated polyethylene.

To properly shield the gap, the shielding should have been installed directly on the seal plate. However, to avoid affecting the calculated reactor cavity pressure during a postulated pipe break, a less effective design was developed, with the shielding installed on a grating raised above the gap. Shielding of the access hole located at the bottom of the primary shield consisted of concrete blocks fitted inside the hole to a thickness of 40 inches. Approximately 12 inches of polyethylene/boric acid bags were stacked above the Reactor Coolant Pump (RCP) removal hatches at the 69' elevation. The Reactor Coolant System (RCS) piping penetrations through the primary shield were not shielded.

The temporary shielding modification was completed and EPT recommenced on 2/10/75. An extensive post-shield installation survey was conducted. The results of that survey and the direction of planning for a permanent shield can be summarized as follows:

(1) The temporary shield modification over the vessel gap reduced neutron dose rates by a factor of 30 to 100 at most locations, and by a factor of 50 at the equipment hatch. This was consistent with expectations based on extrapolation of experimental data and also on simple neutron removal theory.

(2) The total dose rate at the 45' elevation inside containment is now less than 100 mrem/hr, which is considered to be an acceptable level on a permanent basis to allow limited access during power operation.

(3) The radiation detected over the RCP removal hatches at the 69' elevation appears to be neutrons and reactor coolant N-16 gammas streaming up from within the secondary shield.

(4) It does not appear practical to achieve the 0.5 mrem/hr limit outside the equipment hatch by simply adding shielding over sources inside the containment. Rather, the currently installed shielding at the hatch is a more practical solution.

(5) The addition of shielding around the piping penetrations in the primary shield does not appear necessary. The dose rate at the RCP motors is within the limits specified by CE. In addition, long-term component activation from streaming neutrons is not expected to adversely affect maintenance and accessibility after shutdown.

Planning and work on the permanent shield design continues.

7.4 Water Hammer in Main Feedwater Lines

On 5/12/75 following an unplanned trip from 100% power, the feedwater rings of both steam generators were uncovered. Some forty minutes later, as water level was being restored prior to taking the reactor critical, a water hammer incident was experienced in the main feedwater lines. At the time of the incident, the water level was approaching the bottom of the feedwater rings at -57 inches.

Subsequently, operating procedures were revised, such that between -85 inches and -30 inches, auxiliary feedwater is utilized to restore water level at a maximum rate of 1.2 inches per minute.

The water hammer occurred as the water level approached the bottom of the feedwater ring (-57 inches) as cold water was introduced from the feedwater system into the feedwater ring. Hence, use of the auxiliary feedwater system between -85 inches and -30 inches will ensure that the feedwater ring is flooded prior to re-establishing feedwater flow through the main feedwater line. Further, turbulence in the system contributes to the potential for water hammer. Therefore, a level increase at a maximum rate of 1.2 inch per minute has been established for auxiliary feedwater

A test program on both steam generators was conducted to verify that water hammer did not occur when refilling a steam generator using this revised procedure.

The maximum rate was demonstrated by the test not to produce water hammer. Should it become desirable to refill at a faster rate, additional tests will be conducted.

The plant was returned to power operation and Escalation to Power Testing recommenced on 5/19/75.