ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE

STEAM ELECTRIC STATION

UNIT TWO

STARTUP REPORT

TO THE

U.S. NUCLEAR REGULATORY COMMISSION

LICENSE NUMBER NFP-6

DOCKET NUMBER 50-368

FOR THE

PERIOD ENDING July 31, 1979

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7909180316

Foreword

This Startup Report for Arkansas Nuclear One Unit 2 covers the period from the issuance of an operating license by the Nuclear Regulatory Commission, July 18, 1978, until completion of 20% power testing. It is being submitted in accordance with Unit 2 Technical Specification 6.9.1.1 and Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications." The latter requires a startup report to be submitted within 90 days following completion of the startup test program or within 9 months following initial criticality, whichever is earliest.

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PREFACE

Arkansas Nuclear One (ANO) - Unit 2 is located adjacent to Arkansas Nuclear 1 on a peninsula in the Dardanelle Reservoir on the Arkansas River in Pope County, Arkansas. The plant is about six miles West-Northwest of Russellville, Arkansas, and about two miles Southeast of the village of London, Arkansas. The Nuclear Steam Supply System for both units is of the pressurized water reactor design. Unit 1 is rated at 2568 Mwt and was supplied by Babcock and Wilcox. Unit 2 is rated at 2815 Mwt and was supplied by Combustion Engineering, Inc. Bechtel Corporation was the Engineer Constructor for both units. Major design parameters are listed below:

Design Thermal Power	2815 Mwt
Design Electrical Power	912 Mwe
Average Temperature Operating (100% Power)	583°F
Normal Operating Pressure (Primary)	2250 psia
Reactor Coolant Flow Rate	120.4×10^6 lb/hr
Normal Operating Pressure (Secondary)(100%)	900 psia
Steam Flow (Total Both Steam Generators at 100%)	12.64×10^6 lb/hr

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1. INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The Startup Test Program was organized by Arkansas Power and Light (AP&L) personnel and administered by Combustion Engineering (CE) consulting Startup Engineers assisted by CE site Startup Engineers and home office personnel in Windsor, Connecticut. The Startup Test Program consisted of several phases. The test results from each phase were reviewed by the Test Working Group consisting of an AP&L Startup Supervisor, CE Lead Startup Engineer, CE Site Manager, Bechtel Project Startup Engineer, CE Chief Test Engineer, AP&L Nuclear Engineer, and others as required. Test results falling outside of acceptance criteria received an additional review by the Plant Safety Committee and were resolved prior to beginning the next test phase. The test phases are as follows:

A. Initial Fuel Load

B. Post Core Hot Functional

C. Initial Approach to Criticality

D. Low Power Physics Tests

E. 0% thru 20% Power Plateaus

F. 20% thru 50% Power Plateau

G. 50% thru 80% Power Plateau

H. 80% thru 100% Power Plateau

The maximum licensed reactor core power level (100%) is 2815 MWth. The Startup Program began at 0102 7/23/78 with the loading of the first fuel assembly into the reactor vessel and was completed (Later).

1.2 SUMMARY

1.2.1 Initial Fuel Load

Fuel loading was preceded by a response checkout of all neutron detectors that were utilized during the core loading. Two temporary incore detectors were used as well as the two permanently installed start-up range channels. Operability of the detectors was verified by positioning a neutron source near the detectors. Background count rate was also determined at this time. Fuel loading commenced on 7/23/78 and the final fuel assembly was installed on 7/28/78. During the evolution, minor delays were experienced due to equipment malfunctions and in one instance a fuel assembly was lowered on a previously loaded fuel assembly. A visual inspection of the involved assemblies revealed no damage.

1.2.2 Post Core Hot Functional Tests

Post Core Hot Functional testing commenced on 9/5/78 and was completed on 12/3/78. The primary purpose of this testing was to verify that all required plant systems were operable prior to initial criticality. The Post Core Hot Functional Tests were conducted prior to bringing the reactor critical at selected pressures and temperatures ranging from ambient to zero-power, no-load conditions (545°F, 2250 psia).

A number of significant problems occurred during Post Core Hot Functionals that considerably lengthened the testing. These major problems included Control Element Assembly Control System failures, Coil Stack replacement for CEDM #39, failure of #2 Diesel Generator, failure of a Reactor Coolant pump flow DP sensing line root valve, and unit inverter problems. Each instance required a plant cooldown before repairs could be effected.

1.2.3 Initial Criticality

The approach to initial criticality was initiated on 12/4/78 at 1650 with the commencement of CEA withdrawal. Minor problems were encountered with the control element drive system that required adjustment of operating voltages and component replacement. A slow dilution of the Reactor Coolant System at approximately one ppm/minute was established on 12/5/78 at 0900. The reactor was declared critical at 1455 on 12/5/78. The measured boron concentration for criticality was in excellent agreement with the predicted value.

1.2.4 Low Power Physics Testing

Low Power Physics testing commenced on 12/5/78 at 2330 and was completed on 12/16/78 at 1050. All test results agreed favorably with predictions and were within acceptance criteria. Numerous minor delays were caused by problems with the Process Computer System and the Control Element Assembly Position System requiring additional grooming of the systems.

1.2.5 POWER ASCENSION TESTING

The power ascension phase of testing commenced on 12/16/78 at 0800 and the 20% power test plateau was completed allowing escalation to 50% power on June 24, 1979. Testing is to be conducted at 3 additional major plateaus (50, 80 and 100%). The purpose of this test program is to verify as-built plant characteristics are acceptable and to verify the assumptions used in the FSAR.

2.0 INITIAL FUEL LOAD

Initial fuel loading of Arkansas Nuclear One - Unit II commenced at 0102 hours on July 23, 1978, and was completed at 0620 hours on July 28, 1978. The fuel loading sequence is shown in Table 2.0.1 and Figure 2.0.1. Figure 2.0.2 shows fuel assembly location and CEA location by their respective serial numbers. As indicated by Figure 2.0.1, the core was loaded in a seven assembly wide slab from the west face to the east face. Once the slab was completed the south side of the core was loaded after which the north side was loaded.

Neutron multiplication during the fuel loading was monitored using four detector channels, the two permanently installed startup channels and two temporary channels. Inverse multiplication plots were maintained for each detector channel throughout the loading to provide assurance that the reactor remained subcritical at all times.

Following completion of the fuel loading, a loading verification and alignment check was performed to provide the final position, identification, and orientation check of fuel assemblies, CEA's and sources. No major problems were encountered during the fuel loading.

TABLE 2.0.1

FUEL ASSEMBLY LOADING SEQUENCE

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
1	AKC303	Neutron Source 1	A-8
2	AKC306	65	A-7
3	AKBD42	-	B-7
4	AKA014	51	B-8
5	AKC312	64	A-9
6	AKBT04	-	B-9
7	AKC414	-	A-10
8	AKC407	-	A-6
9	AKB007	-	C-8
10	AKA007	32	C-7
11	AKB043	-	C-6
12	AKC216	-	B-5
13	ACA021	44	C-5
14	AKA040	31	C-9
15	AKB039	-	C-10
16	AKC205	-	B-11
17	AKA026	43	C-11
18	AKB054	-	D-11
19	AKA022	23	D-10
20	AKB034	-	D-9
21	AKA001	13	D-8
22	AKB033	-	D-7
23	AKA104	24	D-6
24	AKB041	-	D-5

Table 2.0.1 (CONT'D)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
25	AKA047	F	E-5
26	AKB011	-	E-6
27	AKA010	16	E-7
28	AKB020	-	E-8
29	AKA028	15	E-9
30	AKB051	-	E-10
31	AKA018	E	E-11
32	AKB052	-	F-11
33	AKA108	10	F-10
34	AKB038	-	F-9
35	AKA009	7	F-8
36	AKBT01	-	F-7
37	AKA024	11	F-6
38	AKB009	-	F-5
39	AKA015	17	G-5
40	AKB023	-	G-6
41	AKA019	3	G-7
42	AKB026	-	G-8
43	AKA039	2	G-9
44	AKB028	-	G-10
45	AKA012	14	G-11
46	AKB022	-	H-11
47	AKA033	6	H-10
48	AKB008	-	H-9

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Table 2.0.1 (CONT'D)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
49	AKA005	1	H-8
50	AKB050	-	H-7
51	AKA107	8	H-6
52	AKBT03	-	H-5
53	AKA109	18	J-5
54	AKB049	-	J-6
55	AKA042	4	J-7
56	AKB027	-	J-8
57	AKA034	5	J-9
58	AKB024	-	J-10
59	AKA036	21	J-11
60	AKB019	-	K-11
61	AKA016	13	K-10
62	AKB025	-	K-9
63	AKA037	9	K-8
64	AKB010	-	K-7
65	AKA003	12	K-6
66	AKB021	-	K-5
67	AKA004	G	L-5
68	AKB035	-	L-6
69	AKA046	19	L-7
70	AKB048	-	L-8
71	AKA008	20	L-9
72	AKB036	-	L-10

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Table 2.0.1 (CONT'D)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
73	AKA017	H	L-11
74	AKBT02	-	M-11
75	AKA020	28	M-10
76	AKB015	-	M-9
77	AKA038	D	M-8
78	AKB037	-	M-7
79	AKA110	27	M-6
80	AKB029	-	M-5
81	AKA043	47	N-5
82	AKB002	-	N-6
83	AKA023	35	N-7
84	AKB045	-	N-8
85	AKA031	36	N-9
86	AKB012	-	N-10
87	AKA051	48	N-11
88	AKC210	-	P-11
89	AKC105	60	P-10
90	AKB005	-	P-9
91	AKA011	58	P-8
92	AKB013	-	P-7
93	AKC203	-	P-5
94	AKC404	-	R-6
95	AKC310	70	R-7
96	AKC305	Neutron Source 2	R-8

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Table 2.0.1 (CONT'D)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
97	AKC104	56	B-6
98	ACK304	71	R-9
99	AKC401	-	R-10
100	ACK402	-	P-12
101	AKC201	-	N-12
102	AKA027	41	M-12
103	AKB018	-	L-12
104	AKA106	29	K-12
105	AKB014	-	J-12
106	AKA035	A	H-12
107	AKBT05	-	G-12
108	AKA013	22	F-12
109	AKB016	-	E-12
110	AKA002	38	D-12
111	AKC206	-	C-12
112	AKC416	-	B-12
113	AKC502	63	C-13
114	AKC207	-	D-13
115	AKA029	42	E-13
116	AKB046	-	F-13
117	AKA032	30	G-13
118	AKB040	-	H-13
119	AKB044	-	J-13
120	AKA045	-	K-13

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Table 2.0.1 (CONT'D)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
121	AKA045	49	L-13
122	AKC208	-	M-13
123	AKC504	72	N-13
124	AKC403	-	M-14
125	AKC212	-	L-14
126	AKC101	61	K-14
127	AKB047	-	J-14
128	AKA025	50	H-14
129	AKB004	-	G-14
130	AKC102	54	F-14
131	AKC209	-	E-14
132	AKC415	-	D-14
133	AKC405	-	F-15
134	AKC302	62	G-15
135	AKC309	-	H-15
136	AKC301	73	J-15
137	AKC406	-	K-15
138	AKC410	-	P-4
139	AKC204	-	N-4
140	AKA049	40	M-4
141	AKBT06	-	L-4
142	AKA030	26	K-4
143	AKB032	-	J-4
144	AKA103	C	H04

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Table 2.0.1 (CONT'D)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
145	AKB030	-	G-4
146	AKA048	25	F-4
147	AKB017	-	E-4
148	AKA006	39	D-4
149	AKC215	-	C-4
150	AKC408	-	B-4
151	AKC503	66	C-3
152	AKC213	-	D-3
153	AKA041	45	E-3
154	AKB031	-	F-3
155	AKA050	33	G-3
156	AKB001	-	H-3
157	AKA101	34	J-3
158	AKB053	-	K-3
159	AKA102	46	L-3
160	AKC214	-	M-3
161	AKC501	69	N-3
162	AKC412	-	M-2
163	AKC211	-	L-2
164	AKC103	58	K-2
165	AKB003	-	J-2
166	AKA044	52	H-2
167	AKB006	-	G-2
168	AKC106	57	F-2

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Table 2.0.1 (CONT'D)

STEP NO.	FUEL ASSEMBLY NO.	CEA NO.	CORE LOCATION
169	AKC202	-	E-2
170	AKC413	-	D-2
171	AKC409	-	F-1
172	AKC308	67	G-1
173	AKC311	-	H-1
174	AKC307	68	J-1
175	AKC411	-	K-1
176	AKC107	59	P-6
177	AKC108	55	B-10

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FUEL ASSEMBLY INDEXING

CORE LOADING SEQUENCE

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
1						171	172	173	174	175					
2				170	169	168	167	166	165	164	163	162			
3			151	152	153	154	155	156	157	158	159	160	161		
4		159	149	148	147	146	145	144	143	142	141	140	139	138	
5		12	13	24	25	38	39	52	53	66	67	80	81	93	
6	8	97	11	23	26	37	40	51	54	65	68	79	82	176	94
7	2	3	10	22	27	36	41	50	55	64	69	78	83	92	95
8	1	4	9	21	28	35	42	49	56	63	70	77	84	91	96
9	5	6	14	20	29	34	43	48	57	62	71	76	85	90	98
10	7	177	15	19	30	33	44	47	58	61	72	75	86	89	99
11		16	17	18	31	32	45	46	59	60	73	74	87	88	
12		112	111	110	109	108	107	106	105	104	103	102	101	100	
13			113	114	115	116	117	118	119	120	121	122	123		
14				132	131	130	129	128	127	126	125	124			
15						133	134	135	136	137					

FIGURE 2.0-1

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ARKANSAS NUCLEAR ONE
UNIT 2 CORE INVENTORY MAP

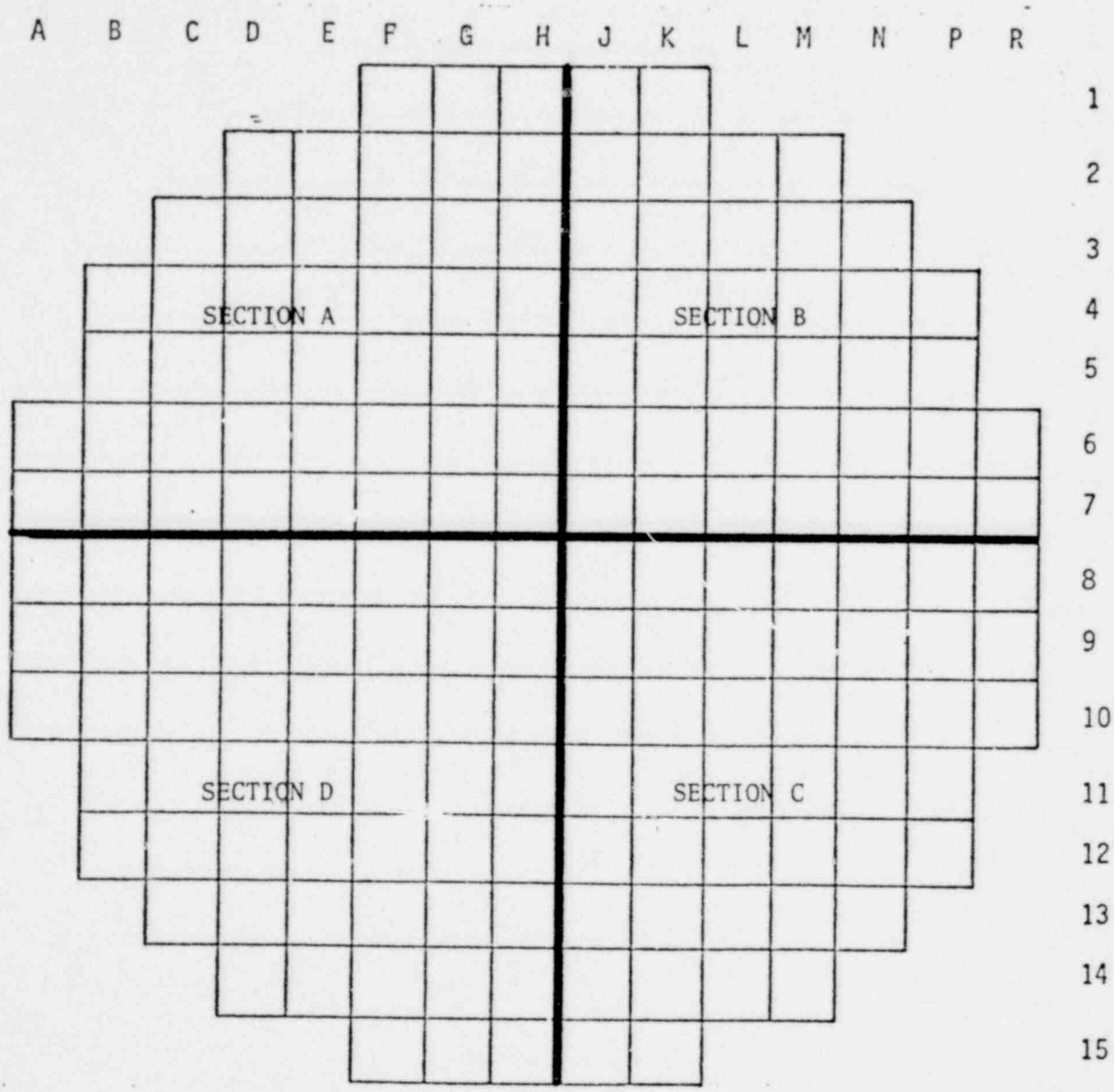

FIGURE 2.0-2 (Page 1 of 5)

ARKANSAS NUCLEAR ONE
UNIT 2 CORE INVENTORY MAP
SECTION A

	A	B	C	D	E	F	G	H
1						AKC409	AKC308 67	AKC311
2				AKC413	AKC202	AKC106 57	AKB006	AKA044 52
3			AKC503 66	AKC213	AKC041 45	AKB301	AKA050 33	AKB001
4		AKC408	AKC215	AKA006 39	AKB017	AKA048 25	AKB030	AKA103 C
5		AKC216	AKA021 44	AKB041	AKA047 F	AKB009	AKA015 17	AKBT03
6	AKC407	AKC104 56	AKB048	AKA104 24	AKB011	AKA024 11	AKB023	AKA107 8
7	AKC306	AKB042	AKA007 32	AKB033	AKA010 16	AKBT01	AKA019 3	AKB050

FIGURE 2.0-2 (Page 2 of 5)

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ARKANSAS NUCLEAR ONE
UNIT 2 CORE INVENTORY MAP
SECTION B

J	K	L	M	N	P	R	
ACK307 68	ACK411						1
AKB003	AKC103 58	AKC211	AKC412				2
AKA101 34	AKB053	AKA102 46	AKC214	AKC501 69			3
AKB032	AKA030 26	AKBT06	AKA049 40	AKC204	AKC410		4
AKA109 18	AKB021	AKA004 G	AKB029	AKA043 47	AKC203		5
AKA049	AKA003 12	AKB035	AKA110 27	AKB002	AKC107 59	AKC404	6
AKA042 4	AKB010	AKA046 19	AKB037	AKA023 35	AKB013	AKC310 70	7

FIGURE 2.0-2 (Page 3 of 5)

ARKANSAS NUCLEAR ONE
UNIT 2 CORE INVENTORY MAP
SECTION C

J	K	L	M	N	P	R	
AKB027	AKA037 9	AKB048	AKA038 D	AKB045	AKA011 35	AKC305	8
AKA034 5	AKB025	AKA008 20	AKB015	AKA031 36	AKB005	AKC304 71	9
AKB024	AKA016 13	AKB036	AKA020 28	AKB012	AKC105 60	AKC401	10
AKA036 21	AKB019	AKA017 H	AKBT02	AKA051 48	AKC210		11
AKB014	AKA106 29	AKB018	AKA027 41	AKC201	AKC402		12
AKA105 37	AKB044	AKA045 49	AKC208	AKC504 72			13
AKB047	AKC101 61	AKC212	AKC403				14
AKC301 73	AKC406						15

FIGURE 2.0-2 (Page 4 of 5)

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ARKANSAS NUCLEAR ONE
UNIT 2 CORE INVENTORY MAP
SECTION D

A	B	C	D	E	F	G	H	
AKC303	AKA014 51	AKB007	AKA001	AKB020	AKA009 7	AKB026	AKA005	8
AKC312 64	AKBT04	AKA040 31	AKB034	AKA028 15	AKB038	AKA039 2	AKB008	9
AKC414	AKC108 55	AKB039	AKA022 23	AKB051	AKA108 10	AKB028	AKA033 6	10
	AKC205	AKA026 43	AKB054	AKA018 E	AKB052	AKA012 14	AKA022	11
	AKC416	AKC206	AKA002 38	AKB016	AKA013 22	AKBT05	AKA035 A	12
		AKC502 63	AKC207	AKA029 42	AKB046	AKA032 30	AKB040	13
			AKC415	AKC209	AKC102 54	AKB004	AKA025 50	14
					AKC405	AKC302 62	AKC309	15

FIGURE 2.0-2 (Page 5 of 5)

3.0 POST CORE HOT FUNCTIONAL TEST

INTRODUCTION

The Post Core Hot Functional Test was composed of those tests required to be carried out prior to initial criticality which required the presence of the fuel and all reactor internals for their performance. The Post Core Hot Functional Test (HFT) phase was a period of hot operations and integrated testing of the Reactor Coolant System (RCS) and associated auxiliary systems following the initial fuel load of the reactor.

The major objectives of the Post Core Hot Functional Test were to verify that all necessary plant systems were operable, that the operations personnel were familiarized with the operation of the integrated systems, and that the initial conditions for criticality were met. The test program essentially took the plant from the cold shutdown condition at the end of the core loading to hot standby at 545°F, 2250 psia, and then to 260°F, 460 psia for initial criticality. In addition to the major objectives, numerous specific objectives were satisfied during the Post Core Hot Functional Test program and are described in this section (see Table 3.0.1).

The successful completion of the Post Core Hot Functional Test program ensured that the plant was ready for initial criticality.

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POST CORE HOT FUNCTIONAL TESTS

RCS Temperature/Pressure Plateau vs. Tests

TABLE 3.0.1

TESTS	120°F ~50 psia	160°F 50 psia	200°F 50 psia	260°F 460 psia	290°F 460 psia	320°F 460 psia	360°F 460 psia	400°F 460 psia
Intercomparison of PPS, CPCs & Proc. Comp. Inputs				X			X	
Reactor Coolant System Flow Measurements								
CEDM/CEA Testing	X			X				
Primary and Secondary Water Chemistry Data	X			X				
Fixed Incore Instrumentation Verification				X	X	X	X	X
Movable Incore Instrumentation Operation Verification	X							
Pressurizer Spray Valve and Control Adjustment								
Reactor Coolant System Leakage Measurement								
Reactor Coolant System Expansion Measurement	X	X	X	X		X	X	X
Emergency Feedwater System/Waterhammer								
Reactor Coolant System Heat Loss Measurement								
Emergency Feedwater System Flow Settings								

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POST CORE HOT FUNCTIONAL TESTS

RCS Temperature/Pressure Plateau vs. Tests

TABLE 3.0.1 (cont'd)

TESTS	120°F ∿50 psia	160°F 50 psia	200°F 50 psia	260°F 460 psia	290°F 460 psia	320°F 460 psia	360°F 460 psia	400°F 460 psia
Reactor Coolant System Cold Leg Restraint Gap Meas.	X			X			X	
Pipe/Component Hot Deflection Preoperational Checks	X			X	X	X		
Steady State Vibration Test	X							
Safety Injection System Check Valve Retest								
Chemical and Volume Control System Integrated Test								
Core Protection Calculator/Reactor Trip Response Time	X							
CEA Exercise Tests								
Main Steam Safety Valve Piping Dynamic Transient								
Emergency Feedwater Pump Turbine Dynamic Transient								
Secondary Hydrostatic Test	X*							

*RCS Pressure 175 to 375 psia, Temperature >90°F

POST CORE HOT FUNCTIONAL TESTS

RCS Temperature/Pressure Plateau vs. Tests

TABLE 3.0.1 (cont'd)

TESTS	450°F 1100 psia	470°F 1550 psia	500°F 1750 psia	530°F 2000 psia	540°F 2250 psia	545°F 2250 psia	260°F 460 psia
Intercomparison of PPS, CPCs, & Proc. Comp. inputs	X	X	X	X	X	X	
Reactor Coolant System Flow Measurements						X	
CEDM/CEA Testing						X	
Primary and Secondary Water Chemistry Data	X					X	
Fixed Incore Instrumentation Verification	X	X	X	X		X	
Movable Incore Instrumentation Operation Verification						X	
Pressurizer Spray Valve and Control Adjustment						X	
Reactor Coolant System Leakage Measurement						X	
Reactor Coolant System Expansion Measurement	X	X		X		X	X
Emergency Feedwater System/Waterhammer	X					X	
Reactor Coolant System Heat Loss Measurement						X	
Emergency Feedwater System Flow Settings						X	

POST CORE HOT FUNCTIONAL TESTS

RCS Temperature/Pressure Plateau vs. Tests

TABLE 3.0.1 (cont'd)

TESTS	450°F 1100 psia	470°F 1550 psia	500°F 1750 psia	530°F 2000 psia	540°F 2250 psia	545°F 2250 psia	260°F 460 psia
Reactor Coolant System Cold Leg Restraint Gap Meas.	X					X	
Pipe/Component Hot Deflection Preoperational Checks	X					X	X
Steady State Vibration Test						X	
Safety Injection System Check Valve Retest						X	
Chemical and Volume Control System Integrated Test						X	
Core Protection Calculator/Reactor Trip Response Time							
CEA Exercise Tests						X	
Main Steam Safety Valve Piping Dynamic Transient						X	
Emergency Feedwater Pump Turbine Dynamic Transient						X	
Secondary Hydrostatic Test							

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3.1 NSSS TESTS

3.1.1 INTERCOMPARISON OF PPS, CPC'S, MAIN CONTROL BOARD AND PROCESS COMPUTER INPUTS TESTS

3.1.1.1 Purpose

The PPS, CPC's, Main Control Board Process Instruments, Core Operating Limits Supervisory System (COLSS), and the Plant Computer have many parameters with common inputs. This procedure verifies the agreement between inputs as specified by the acceptance criteria.

3.1.1.2 Test Method

The comparisons were performed at various temperature and pressure plateaus as the primary plant was brought from cold shutdown to a hot standby mode. (Refer to Table 3.1.1 for plateaus.)

3.1.1.3 Test Results

The data taken at each plateau demonstrated that the inputs to the CPC's, PPS, Main Control Board Process Instruments, COLSS, and the Computer were in agreement as required.

TABLE 3.1.1

Intercomparison Plateaus

Test No.	Temperature (°F)	Pressure (psia)
1	260	460
2	360	460
3	450	1100
4	470	1550
5	500	1750
6	530	2000
7	540	2250
8	545	2250

3.1.1.4 Conclusion

All acceptance criteria were met showing accurate and consistent comparisons of the inputs to the PPS, CPC's, Main Control Board Process Instruments, COLSS, and the computer.

3.1.2 REACTOR COOLANT SYSTEM FLOW MEASUREMENTS TESTS

3.1.2.1 Purpose

A. The Purpose in performing this test was to determine the characteristics of and collect data for the Reactor Coolant System (RCS) as follows:

a. Determine the post-core RCS flow rates and pressure drops.

b. Determine the Reactor Coolant Pump (RCP) coastdown flow characteristics.

c. Collect data on the operation of the flow related portions of the Core Operating Limits Supervisor System (COLSS) and the Core Protection Calculators (CPC's) for steady state and transient flow conditions and adjust the applicable constants as necessary to reflect operating conditions.

d. Verify that the CPC's provide a reactor trip signal consistent with the trip times predicted by the CPC Fortran model.

B. In addition to the RCS flow-related objectives stated above, one portion of this test was designed to verify the response time associated with a Low Departure from Nucleate Boiling Ratio (DNBR) CPC trip as described in the Technical Specifications (Section 3/4.3.1, Table 3.3-2, Page 3/4 3-6).

3.1.2.2 Test Method

The test was performed at Hot Zero Power conditions of 545°F $\pm$ 0.5°F and 2250 psia, pressurizer level was maintained at 34% $\pm$ 1%. Data was obtained for both transient and steady state conditions for 26 various RCP configurations. These configurations are summarized in Table 3.1.2.1 and are given in more detail in Table 3.1.2.2.

For the purpose of verifying the conservative operation of the flow-related algorithms of the CPC's as well as to determine the DNBR trip times at various power levels, special Floppy test discs were loaded into the CPC's prior to performing the test runs. These discs simulated zero power ARO conditions on CPC Channel A; 50% power, ARO on Channel B; 80% power, ARO, on Channel C; and 100% power, ARO, on Channel D.

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CPC trip times were recorded by monitoring the RCP breaker status and CPC DNBR trip signals. All continuous data collection was taken via BRUSH 260 strip chart recorders and a PDP 1104 mini-computer. In addition to continuous monitoring, data was also obtained from the plant computer and Main Control Board periodically as required.

The portion of the test which measured CPC trip response time from a low DNBR signal at simulated hot full power conditions was performed after run #26. Hot full power (HFP) conditions were simulated by loading the CPC channels with special floppy test discs, different from those used previously. The test was initiated by simultaneously tripping two diametrically opposed RCP's. The time delay was measured using a high speed Visicorder. The above sequence was conducted 6 times, once for each CPC channel, each time alternating two RCP's.

The difference between the two portions of the test which measured low DNBR CPC trip response times is that the portion performed during the main body of the test was necessary to verify Fortran modeling of the CPC's. The portion performed following Run #26 was necessary to demonstrate that the CPC's would generate a low DNBR trip signal within the time specified by the Technical Specifications.

3.1.2.3 Test Results

The four-pump steady state volumetric flow rate was measured to be 361,468 GPM or 112% of design. The corresponding mass flow rate was determined to be 137 x 10^6 lbm/hr or 114% of design.

Flow coastdown characteristics were recorded and determined from COLSS and CPC data acquisition. Also, RCP ΔP's, speeds, and RCS temperatures were recorded on the BRUSH 260 chart recorders. Baseline data for RCS pressure drops were recorded on the BRUSH 260 recorders from reactor vessel ΔP signals.

An initial set of CPC flow constants FC1 and FC2 were supplied by CE Windsor in order to ensure the conservative operation of the CPC's through the 50% power testing. Further extensive analysis of the RCP coastdowns yielded a final set of CPC flow constants to be incorporated prior to leaving the 50% plateau.

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Furthermore, the RCP coastdowns have been examined and compared to those assumed in the safety analyses to assure conservatism.

CPC operation for both steady state and transient conditions was verified via the data acquisition and subsequent reduction of such parameters as mass flow rate, DNBR trip times, and DNBR margin versus time. COLSS flow related operations were verified via analysis of the COLSS snapshots collected throughout the test. COLSS addressable constants were adjusted to reflect actual measured four-pump steady-state and three-pump steady-state RCS flow, inclusive of the reverse flow algorithm.

The CPC's provided a trip signal on low DNBR within the acceptable times in all cases, (see Table 3.1.2.3).

The CPC Low DNBR Trip response times were acceptable in all cases, (see Table 3.1.2.4).

3.1.2.4 Conclusions

All objectives and acceptance criteria for this test have been met.

RCS flow for steady state four pump operation is 361,468 GPM or 112% of design flow. The CPC's have demonstrated acceptable operation with regard to RCS flow considerations and have initiated low DNBR trip signals well below the time intervals given in the Technical Specifications.

TABLE 3.1.2.1: RCP CONFIGURATION SUMMARY

# OF RUNS THIS CONFIGURATION	RCP CONFIGURATION	DATA COLLECTED
8	All 4 RCP's Operating	Steady State
1	All 4 RCP's Stopped	Steady State
1	4-lose-4 RCP Coastdown	Transient
2	4-lose-2 RCP Coastdown	Transient
4	4-lose-1 RCP Coastdown	Steady State
4	3 RCP's Operating, 1 Stopped	Steady State
2	2 RCP's Operating, 2 Stopped	Steady State
4	1 RCP Operating, 3 Stopped	Steady State
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TABLE 3.1.2.2: RCS FLOW & OCASTDOWN TEST SEQUENCE

Test Run	RCPs Running*	RCPs Tripped	Static Data	Noise Data	Transient Data
1	A,B,C,D	None	X	X	
2		A,B,C,D			X
3	None	A,B,C,D	X		
4	A,B,C,D	None	X		
5		A			X
6	B,C,D	A	X	X	
7	A,B,C,D	None	X		
8		B			X
9	A, C,D	B	X		
10	A,B,C,D	None	X		
11		B,C			X
12	A, D	B,C	X	X	
13	A	B,C,D	X		
14	D	A,B,C	X		
15	A,B,C,D	None	X		
16		C			X
17	A,B, D	C	X		
18	A,B,C,D	None	X		
19		D			X
20	A,B,C	D	X		
21	A,B,C,D	None	X		
22		A,D			X
23	B,C	A,D	X	X	
24	B	A, C,D	X		
25	C	A,B, D	X		
26	A,B,C,D	None	X		

*A corresponds to 2P32A, B to 2P32B, C to 2P32C, D to 2P32D

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TABLE 3.1.2.3: CPC TRIP TIMES FOR FLOW COASTDOWNS

Coastdown & Channel	Power	Expected Range (sec.) Min.	Max.	Actual Trip[1] Time (sec.)	Error (sec.)
Run 2, 4/4					
A	0	1.413	1.962	1.428	0
B	50	1.236	1.443	1.243	0
C	80	.297	.443	.370	0
D	100	.206	.355	.27[illegible]	0
Run 11, 2/4					
A	0	1.979	3.205	[illegible]3	0
B[2]	50	1.373	1.631	1.400	0
C	80	.638	.840	.655	0
D	100	.288	.445	.289	0
Run 22, 2/4					
A	0	1.963	3.114	2.646	0
B[2]	50	1.332	1.628	1.358	0
C	80	.616	.825	.646	0
D[3]	100	.296	.453	.265	-.031
Run 5, 1/4					
A	0	No Trip	No Trip	No Trip	
B[2]	50	1.322	1.594	1.424	0
C[2]	80	1.32[illegible]	1.594	1.384	0
D	100	.394	.570	.476	0
Run 8, 1/4					
A	0	No Trip	No Trip	No Trip	
B[2]	50	1.386	1.660	1.450	0
C[2]	80	1.386	1.660	1.475	0
D	100	.456	.660	.475	0
Run 16, 1/4					
A	0	No Trip	No Trip	No Trip	
B[2]	50	1.393	1.667	1.422	0
C[2]	80	1.393	1.667	1.455	0
D[3]	100	.478	.666	.409	-.069
Run 19, 1/4					
A	0	No Trip	No Trip	No Trip	
B[2]	50	1.384	1.638	1.446	0
C[2]	80	1.384	1.638	1.491	0
D	100	.497	.702	.510	0

1 Actual trip times plus 0.0833 seconds for breaker delay time.
2 90% RCP speed trip times.
3 Even though these are outside of the acceptance range, they are acceptable because they are conservative.

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TABLE 3.1.2.4: CPC RESPONSE TIME TEST RESULTS

PARAMETER	REQUIRED * RESPONSE TIME	MEASURED RESPONSE TIME
Channel A, Low DNBR, RCPs A & D	≤0.80 sec.	0.58 sec.
Channel B, Low DNBR, RCPs B & C	≤0.80 sec.	0.53 sec.
Channel C, Low DNBR, RCPs A & D	≤0.80 sec.	0.53 sec.
Channel D, Low DNBR, RCPs B & C	≤0.80 sec.	0.53 sec.

* Technical Specification

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3.1.3 CEDM/CEA TESTS

3.1.3.1 Purpose

The purpose of this test was to demonstrate that the Control Element Drive Mechanism Control System (CEDMCS) and its associated peripherals, i.e., the Plant Computer, CPC/CEAC's, and the Operator Console, control the motion of the CEDM/CEA's and provide consistent position indication between devices. In particular the following objectives were defined:

A. Check the position indication systems and verify the proper functioning of the CEDM limits and alarms;

B. Verify that all individual CEA's (81) have the proper drop time from a fully withdrawn position to the 90% insertion position;

C. Verify the proper operation of the CEDM holding bus for each subgroup; and

D. Determine the CPC and PPS integrated reactor trip response time for both DNBR and LPD trip parameters. These were limited to trip conditions caused by a dropped CEA.

3.1.3.2 Test Method

CEDM/CEA testing was executed at three stable plateaus of the HFT controlling procedure 2.650.01:

TEST CONDITION	RCS FLOW	RCS PRESSURE	RCS TEMPERATURE
1	Shutdown Cooling	~ 50 psia	≤ 200°F
2	One pump each loop	460 psia	260°F
3	Four Pumps	2250 psia	545°F

The CEDM Holding Busses were tested between the ≤ 200°F and 260°F RCS temperature plateaus, and the CPC response time testing was performed after the final rod drop sequence at the 545°F plateau.

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All CEA movements during this test were executed in the Manual Individual mode. A Honeywell Visicorder was used throughout the test to monitor the 5 motor coil (i.e., upper gripper, lower gripper, lift, load transfer, and pulldown) current traces. Startup nuclear instrumentation channels 1 and 2 were monitored throughout the test for any unexpected reactivity changes associated with CEA motion. Whenever possible, all 81 CEA's were withdrawn to approximately 6 inches to verify the longevity of the CEDM power supply/ switch operation. All CEA disconnect breakers were de-energized except for the CEA under test.

A. CEA Testing at Cold Shutdown (0 RCP's Running):

At this plateau, a preliminary operational check was performed on each CEA before actual testing. The CEA was energized and withdrawn to approximately 10 inches, then inserted to its lower electrical limit. The rod was subsequently tripped and re-energized at the disconnect breaker. After each preliminary check, each CEA was individualy withdrawn to its upper electrical limit. Periodic stops were made at 12 in., 36 in., 60 in., 84 in., 108 in., and 132 in. during withdrawal to take coil traces and verify appropriate position indication. The CEA was then inserted to its lower electrical limit with periodic holds at the aforementioned points to complete the data aquistion. After tripping and re-energizing the CEA, a continuous withdrawal to the upper electrical limit was performed to determine CEA withdrawal rate. After connecting the appropriate reed switch position transmitter (RSPT) signal to the visicorder, and initializing the computer rod drop program, the CEA was dropped to the bottom by opening its associated disconnect breaker. The 90% and 100% insertion times were then determined for the CEA by examining the visicorder trace and the computer output. Upon completion of the tests described above, for all 81 CEA;s, the two fastest and two slowest rods were retested a total of 3 times to determine the drop characteristics of the most limiting CEA's.

B. Hold Bus Test:

The Hold Bus Test was performed during the heat up to the 260°F plateau. Each CEA Subgroup (20) was placed on it appropriate holding bus. This was done by withdrawing each CEA in the subgroup to 6 inches and energizing the corresponding subgroup maintenance switch at the CEDMCS cabinets.

It was then verified that rod motion for that subgroup was not possible from the operators console. The system was also tested to verify that only one subgroup could be assigned to a holding bus at a time.

C. CEDM Power Supply Burn In Test:

Whenever possible, all 81 CEA's were energized and withdrawn 6 inches to verify proper operation of the CEDM power supplies.

D. CEA Drop Time Tests (260°F, 2 RCP's Operating):

At this plateau, only CEA drop time testing was performed. Each CEA was withdrawn to its upper electrical limit and dropped. The drop traces were analyzed for 90% and 100% insertion times using the same method. The two fastest and two slowest CEA's were retested.

E. CEDM/CEA Testing at 545°F, 2250 psia, (4 RCP's Running):

Testing at this plateau was identical to testing with the RCS <200°F with the following exceptions:

a. Fewer rod motion verification traces were taken, and

b. Since the various alarms checked during cold shutdown were not temperature dependent, they were not verified at this plateau.

F. CPC/PPS DNBR/HLPD Trip Response Time Testing:

Each CPC and CEAC channel were reprogrammed through interactive commands via a test disc and the teletype. This was to simulate full power operating conditions to the calculators, thus bypassing certain level inputs.

	CPC A		CPC B		CPC C		CPC D	
TEST CHANNEL	DNBR	HLPD	DNBR	HLPD	DNBR	HLPD	DNBR	HLPD
*CPC-A (DNBR)	L	B	T	T	B	B	B	B
CPC-B (DNBR)	T	T	L	B	B	B	B	B
CPC-C (DNBR)	B	B	T	T	T	T	B	B
CPC-D (DNBR)	B	B	B	B	T	T	L	B

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TEST CHANNEL	CPC A DNBR	CPC A HLPD	CPC B DNBR	CPC B HLPD	CPC C DNBR	CPC C HLPD	CPC D DNBR	CPC D HLPD
CPC-A (HLPD)	B	L	T	T	B	B	B	B
CPC-B (HLPD)	T	T	B	L	B	B	B	B
CPC-C (HLPD)	B	B	T	T	B	L	B	B
CPC-D (HLPD)	B	B	B	B	T	T	B	L

* L = Live
* T = Tripped
* B - Bypass

The above test matrix was employed to ensure that only the channel under test tripped, with the required two out of four PPS logic. In each case the test channel was placed in bypass long enough to withdraw a target CEA for that channel to its upper electrical limit. The reed switch position transmitter upper electrical limit. The reed switch position transmitter (RSPT) for that CEA was then used as a live input to the test channel. A visicorder was connected to the following signals:

a. The target CEA RSPT,

b. The target CEA Coil Monitor,

c. The CEA #1 Coil Monitor, and

d. The Test Channel DNBR/HLPD trip output contact.

The test sequence was initiated by tripping the target CEA at its disconnect breaker, and monitoring the transient with the visicorder. The total CPC/PPS response time was determined from the visicorder trace by measuring the time between the target CEA trip, and the interruption of current to CEA #1. All four CPC channels were tested in the above manner for the DNBR and HLPD trip parameters.

3.1.3.3 Test Results

General:

All 81 CEA's were tested per procedure at the three specified plateaus (<200°F, 260°F, and 545°F).

Few CEA's were tested successfully without some form of hardware trouble shooting required. Although problems experienced with rod motion were diverse, three general types of components were responsible for a majority of the failures. These were:

1) RCA CMOS (Complementary Metal Oxide Semiconductor) chips.

2) Timer board strip switches, and

3) Monsanto Opto Isolators.

Due to the repetitive nature of the aforementioned CEDMCS component failures, all CEDM Time Board strip switches were returned to the vendor for repair. In addition, all power switch opto isolators were replaced as well as approximately 3500 other CMOS logic chips.

The above components are all part of the CEDMCS logic cicuitry i.e., upstream of the coil power switches. Only one CEA drive motor was found defective. This was CEDM #39 which developed a low resistance condition on its upper gripper coil. The coil stack assembly was replaced and CEDM-39 was successfully tested.

It was intended that the computer drop time program be used at all three test plateaus to verify its calculational accuracy. However, a review of cold rod drop data showed that the computer program gave inaccurate and nonconservative results. This was due to a combination of software errors and program loading priority. As a result, use of the computer program was discontinued.

RCS < 200°F:

The following table summarizes the data taken at 160°F:

PARAMETER	FASTEST DATA/CEA	SLOWEST DATA/CEA	RANGE	UNITS
90% Drop Time	2.19/#2	2.48/#80	.29	sec.
100% Drop Time	2.45/#1	2.78/#80	.33	sec.
90% Drop Time(PL)*	2.17/#26	2.22/#25, #27	.05	sec.
100% Drop Time(PL)*	2.42/#26	2.51/#23	.09	sec.
CEA Speed	29.64/#18	30.5/#12	.86	in/min
CEA Speed(PL)*	19.85/#25	20.34/#26	.49	in/min

* (PL) = Part Length CEA

RCS 260°F

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The following table summarizes the data taken at 260°F:

PARAMETER	FASTEST DATA/CEA	SLOWEST DATA/CEA	RANGE	UNITS
90% Drop Time	2.30/#1	2.58/#71	.28	sec.
100% Drop Time	2.55/#1	2.89/#71	.34	sec.
90% Drop Time(PL)*	2.22/#26	2.29/#28	.07	sec.
100% Drop Time(PL)*	2.42/#24	2.55/#23	.13	sec.
CEA Speed	-------	--------	---	in/min
CEA Speed(PL)*	-------	--------	---	in/min

* (PL) = Part Length CEA

RCS 545°F:

The following table summarizes the data taken at 545°F:

PARAMETER	FASTEST DATA/CEA	SLOWEST DATA/CEA	RANGE	UNITS
90% Drop Time	2.42/#2	2.77/#71	0.35	sec.
100% Drop Time	2.69/#2	3.03/#71	0.34	sec.
90% Drop Time(PL)*	2.31/#26	2.42/#23, 25	0.11	sec.
100% Drop Time (PL)*	2.55/#26	2.81/#28	0.26	sec.
CEA Speed	30.4/#56	28.8/#31	1.60	in/min
CEA Speed(PL)*	20.11/#25	20.0/#28	0.11	in/min

* (PL) = Part Length CEA

CPC DNBR and HLPD Response Time Test:

All DNBR and HLPD test trips were conducted successfully per procedure. The results of these tests are as follows:

TEST CHANNEL	RESPONSE TIME (SEC) DNBR	HLPD
CPC-A	.94	1.20
CPC-B	.82	1.28
CPC-C	.91	1.29
CPC-D	.85	1.32

Two test runs were made for each CPC channel trip parameter. The above table includes the most conservative response time.

3.1.3.4 Conclusion

The following applies to the CEDM/CEA's and their associated control systems:

1) All CEA's have a characteristic 90% insertion drop time of less than three seconds;

2) All regulating and shutdown CEA's move at 30 ± 2 in/min;

3) All part length CEA's move at 20 ± 2 in/min.

4) All CEA position indication systems and alarms operate properly;

5) The CEDM holding busses operate properly for each CEA subgroup; and

6) All CEA's exhibit a dashpot effect as verified by the drop time and recorder traces between 90% and 100% insertion.

Each CPC/PPS safety channel was tested for its response time to trip for the low DNBR and high LPD trip parameters. In each case, the CPC/PPS DNBR and HLPD trip times were ≤1.58 sec. respectively.

3.1.4 PRIMARY AND SECONDARY WATER CHEMISTRY TESTS

3.1.4.1 Purpose

The purpose of this test was to verify proper water chemistry control for the primary and secondary cooling systems during post core hot functional testing. Further, to provide the necessary sampling frequency to comply with the Technical Specifications and Chemistry Manual (CENPD-28) and to provide assurance that chemistry limits are not exceeded. Also, baseline chemistry data was to be established.

3.1.4.2 Test Method

Both reactor coolant and secondary system samples were obtained per the sampling frequency specified in CENPD-28. The results of these samplings were recorded on the appropriate plant data sheets.

A. RCS chemistry was maintained with the limits of CENPD-28 by chemical addition, feed and bleed, by the use of purification filters (2F3A and 2F3B), and by the use of ion exchangers (2T36A, 2T36B, 2T70). These processes were carried out in accordance with plant operating procedures and instructions from the Radiochemistry Department.

B. Secondary chemistry was maintained with the use of chemical additions, in accordance with plant operating procedures and Chemistry Department instructions, and by performing Steam Generator Blowdowns.

3.1.4.3 Test Results

A. Primary water chemistry results taken from the hot lab water report for the period of August 30th to November 9th, 1978, are listed below. The figures represent average values for the time period, and are reported in ppm unless otherwise indicated.

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PARAMETER	AVERAGE VALUE
PH	6.26
Cl^-	0.07
F^-	0.04
Suspended Solids	0.01
Li^-	0.57
Total Gas	51.4 ml/liter
H_2	12.7%
N_2	82.2%

The values for Cl^- and F^- were typically less than .01 ppm, except during plant transients. All the values above are well within those specified in CENPD-28.

B. The secondary plant was started up and shutdown numerous times during post core hot functional testing. These transients prevented the gathering of sufficient data to draw representative conclusions as to the adequacy of secondary chemistry control.

3.1.4.4 Conclusion

A. This test showed that primary chemistry can be controlled within the specified limits and that the sampling frequency required to verify this is consistent with CENPD-28 specifications.

B. Secondary water chemistry data provided inconclusive results due to varing plant conditions. However, further testing was performed during power ascension to verify proper secondary chemistry control.

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3.1.5 FIXED INCORE INSTRUMENTATION TESTS

3.1.5.1 Purpose

The purpose of this test was to verify that the leakage resistance of each fixed incore detector and its associated cabling is equal to or greater than 1×10^7 ohms, and to provide a permanent record of fixed incore thermocouple data during plant startup. A leakage resistance of 1×10^7 ohms is necessary to ensure proper detector response and to indicate proper electrical continuity between the fixed incore detector, its cabling, and amplifier system.

3.1.5.2 Test Method

Incore detector leakage resistances were measured at the 545°F/2250 psia plateau i.e. Hot Standby. During the test, incore detector signal leads were disconnected at the incore amplifier input, one detector at a time, and leakage resistance measured for each rhodium and background detector associated with the chosen instrument assembly. A megohmbridge using an applied voltage of 10.0 volts was used for all resistance measurements.

Core exit temperatures, as measured by the thermocouple in each incore detector assembly at various temperature/pressure plateaus, were recorded via computer printout during the initial heatup to Hot Standby.

3.1.5.3 Test Results

All incore detector leakage resistance values were greater than 1×10^7 ohms. However, two rhodium detector strings, specifically Rhodium #3 of detector R-10 and Rhodium #4 of detector N-12, had measured leakage resistances three orders of magnitude greater than all the other detector strings (approximately 1×10^{10} ohms). These two detector strings were examined for continuity with a time domain reflectometer, but no indication of open connections was discovered. The incore thermocouple data was collected and the results tabulated and included with the test data.

3.1.5.4 Conclusion

The acceptance criteria requiring fixed incore detector leakage resistance to be greater than or equal to 1×10^7 ohms for all rhodium and background detectors was verified by the test results. The fixed incore thermocouple readings were collected for baseline data only and no specific acceptance criteria was required.

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3.1.6 MOVABLE INCORE INSTRUMENTATION OPERATION VERIFICATION TESTS

3.1.6.1 Purpose

The purpose of this test was to:

A. Obtain movable incore detector path length measurements for input to the plant computer.

B. Verify the proper operation of the moveable incore detector system (MICDS) when controlled by the plant computer.

C. Verify that the detectors can be inserted into all assigned incore locations.

3.1.6.2 Test Method

A. Path Length Measurements

The movable incore detector path lengths were physically measured at ambient temperatures ($\sim$ 120°F) using dummy detector cabling. The path lengths were remeasured using the encoder and portable control box to verify the physical measurement data at Hot Standby to verify that the effects of expansion were negligible.

B. Computer Control

Computer control of the movable incore detector system (MICDS) was tested in the manual, semi-automatic, and automatic modes at Hot Standby.

3.1.6.3 Test Results

During the testing, several system problems were identified:

A. Encoder and computer interface problems were found and identified.

B. The original rhodium detectors were oversized and would not fit into the calibration tubes at hot conditions.

C. Computer software problems were encountered.

D. Cable kinking problems prevented insertion of the detector into several drive paths.

E. Two drive machine motors failed.

The encoder and computer interface problems were corrected. New Rhodium detectors were ordered but were not present for testing. The MICDS was tested with cabling representative of the actual cable to be used with the incore detectors. The two drive machine motors were replaced and/or rebuilt and a cable straightening tool was obtained and used on both drive cables to correct the kinking problems

Manual, semi-automatic, and automatic computer control modes were verified for selected paths for both MICD A and MICD B. A software modification was instituted to allow automatic mode to be restarted in any given path.

3.1.6.4 Conclusion

The movable incore detector path lengths were measured using two different methods and the results were in agreement within the acceptance criteria. The performance of the manual, semiautomatic, and automatic MICDS computer control modes was demonstrated.

3.1.7 REACTOR COOLANT SYSTEM LEAKAGE MEASUREMENT TESTS

3.1.7.1 Purpose

Three methods were used to measure leakage from the RCS. Monitoring equipment in containment was installed to continuously monitor the environmental conditions via remote observation in the Control Room. Instrumentation was also installed to monitor containment sump water level. The third method of leakage detection was a daily operator's calculation of RCS water inventory balancing losses against additions. This procedure verified that the RCS leakage rate operating procedure gives accurate results for RCS water inventory.

3.1.7.2 Test Method

With the RCS at 545°F, 2250 psia, the following parameters were monitored over a 4 hour period in order to determine RCS leakage:

A. Containment sump level,

B. Safety injection tank level and pressure.

C. RCS pressurizer safeties and quench tank level and pressure,

D. Holdup tanks and vacuum degasifier levels,

E. Temperature in reactor vessel gasket telltale drain line,

F. Flow to sample sink from RCS sample lines,

G. RCS and CVCS test connections, drains and vents,

H. Heat exchangers having an RCS or CVCS interface,

I. Volume control tank level, and

J. Charging flow rate.

Data was collected before and after the 4 hour hold period. Data reduction was performed by two methods: (1) the graphical reduction used in the operating procedure was compared to, (2) the more precise reduction method as detailed in the test procedure.

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*TECHNICAL SPECIFICATION ALLOWED LEAKAGE RATE	TEST PROCEDURE LEAKAGE RATE	OPERATING PROCEDURE LEAKAGE RATE	% DIFFERENCE
$\leq$1.0 gpm	0.346 gpm	0.342 gpm	1.2

*unidentified source

3.1.7.3 Conclusion

The operating procedure gives results that are consistent with the leak rate determined by the test procedure method and thereby meets the acceptance criteria. The total RCS leakage rate was less than the allowed leakage rate as stated in the Technical Specifications. (Section 3.4.6.2).

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3.1.8 REACTOR COOLANT SYSTEM EXPANSION MEASUREMENT TESTS

3.1.8.1 Purpose

This test was a repetition of the one performed during Pre-Core HFT because of the many restraints, shims, and snubbers that have been added, or modified, since the original performance of the test. The purpose was to demonstrate that the shims installed as a result of the Pre-Core RCS Expansion Measurement Test were correctly sized, and to verify that RCS components (Steam Generators and Reactor Vessel) are free to expand and contract during plant heat up and cooldown.

3.1.8.2 Method

A. During the initial heatup, inspections and/or measurements were made at the following locations.

   a. Reactor Vessel lower shear keys

   b. Reactor Vessel upper lateral restraints

   c. Steam Generator sliding bases

   d. Steam Generator snubber assemblies

During the initial heatup, visual inspections were made at 160°F, 200°F, 320°F, 400°F, 480°F and 520°F to ensure that the components were free to expand without binding. Measurements were taken at 120°F, 260°F, 360°F, 450°F and 545°F. Due to an unscheduled cooldown, measurements required at 260°F and 120°F, during cooldown, were not taken. The test was run a second time in order to obtain all required data.

B. The following acceptance criteria were developed for the hot standby plateau.

   a. The sum of the RV lower shear key dimensions shall be $\geq$ 0.030 in. with any one dimension $\geq$ 0.030 in. and $\leq$ 0.230 in.

   b. The sum of the RV upper lateral restraint dimension shall be $\geq$ .030 in. with any one dimension $\geq$ 0.030 in. and $\leq$ 0.040 in.

   c. The steam generator sliding base dimensions shall be $\geq$ 0.200 in. and $\leq$ 0.230 in.

d. The steam generator snubber pin to pin distance shall not be greater than 36.750 in.

3.1.8.3 Test Results

A. During the initial RCS heatup, none of the components were binding nor did any of the gaps being monitored constrict to the point of requiring shim adjustment. All required measurements were completed with the exception of the final 545°F and 260°F stable readings.

B. RCS freedom to expand and contract was verified during the first heatup. During the second heatup, all required measurements at stable conditions were performed with one exception of the final 260°F and 120°F ambient readings. These measurements were omitted with the concurrence of Combustion Engineering. All measurements were within the required tolerance with the following exceptions:

a. Steam Generator A sliding base

b. Plant 0° reactor vessel upper lateral restraint

c. Plant 120° upper lateral restraint

3.1.8.4 Conclusion

No points of contact were observed throughout this test. The RCS expanded and contracted within the tolerances established in the test procedure, with the above exceptions. These were documented and transmitted to Combustion Engineering to expedite an engineering review of the out of tolerance gaps. C.E's response established the acceptability of the gaps as measured.

3.1.9 RCS COLD LEG RESTRAINT GAP MEASUREMENT (SHIMMING VERIFICATIONS) TESTS

3.1.9.1 Purpose

This test was repeated during PCHF Testing since many restraints, shims and snubbers had been added or modified since original performance of the test during precore hot functional testing. The purposes were to:

A. Measure RCS cold leg restraint - restraint lug hot gaps for shim installation verification.

B. Measure RCS cold leg lateral restraint shim surface impact pad hot gap for shim installation verification.

C. Verify RCS cold leg angular restraint cold gaps are within tolerance as specified by the Test Procedure.

D. Verify RCS cold leg piping and insulation do not structurally interfere with restraints prior to and during RCS heatup to hot standby no load conditions.

3.1.9.2 Test Method

After the RCS was stabilized for six hours at 120°F $\pm$ 5°F, initial ambient measurements were taken at all gap locations described in A through C above. Final acceptance criteria for the cold leg angular restraint (C), requires ambient measurements only, therefore, these were not measured at any further temperature plateaus.

During heatup, visual inspections of all gap locations were made at 260°F, 360°F, 450°F and 545°F to assure that cold leg piping and insulation did not structurally interfere with restraints. Certain gaps suspect of contact were also measured at these plateaus to keep a close watch on their movement.

After the RCS was stabilized for six hours at 545° $\pm$ 5°F, measurements were taken at all locations, with the exception of the angular restraints. In order to measure any long term thermal expansion of the RCS, these measurements were repeated just prior to cooldown following hot functional testing.

Due to an unscheduled plant cooldown during PCHF, the measurements required just prior to cooldown, were obtained following the second heatup.

3.1.9.3 Test Results

The initial heatup to 545°F yielded several gaps either in physical contact or exceeding clearance requirements. Deficiencies were written against all gaps not meeting acceptance criteria and work was done to correct them during the shutdown period.

Measurements taken following the second heatup showed that the contact points had been cleared but several of the gap clearances did not come within tolerance. Ten of the twenty-four RCS cold leg restraint - restraint lug gaps did not meet the original acceptance criteria of .0625" + .0625" - 0". Two of the thirty-two cold leg lateral restraint shim surface-impact pad hot gaps did not meet the acceptance criteria of 0.375" + 0.125" - 0.172".

3.1.9.4 Conclusions

All measurement data was forwarded to Bechtel and CE project offices for analysis. Each contractor reviewed the data and following analysis, determined the data was acceptable and would meet the design criteria of both CE and Bechtel.

3.1.10 CORE PROTECTION CALCULATOR/REACTOR TRIP RESPONSE TIME TESTS

3.1.10.1 Purpose

The objective of this test was to measure the response times associated with the following Core Protection Calculator Trips:

A. Departure from Nucleate Boiling Ratio

a. Th
b. Tc
c. Pzr Pressure
d. Neutron Flux Level

B. High Local Power Density

a. Neutron Flux Level

3.1.10.2 Test Method

The CPC response time tests measured the overall CPC system response to a single input parameter change. Two test discs were employed in order to load Response Time Test Software. Test Disc #1 was used to load Ch. A CPC, Ch. B CPC, and Channel B CEAC. Test Disc #2 was used to load Ch. C CPC, Ch. D CPC, and Channel C CEAC. The test software was identical to that which resides in memory during normal system operations with the exception that the test software allowed one or more live inputs with the rest simulated by the software.

The functional unit tests for the CPC response time testing were divided into two main groups, High Local Power Density (LPD) trips, and Departure from Nucleate Boiling Ratio (DNBR) trip tests. The test methods were as follows.

A. LPD Trips

a. Neutron Flux Power from Excore Neutron Detectors

The neutron flux power test was a step change in all excore detector outputs to 116.5% of their initial values. The test apparatus as shown in Figure 3.1.10.1 was utilized. Total response time measured was from signal initiation, via the current generator, to the pre-amplifiers until the Trip Circuit Breakers opened deenergizing the CEDM coils. The response time was measured with a high speed visicorder.

B. DNBR Trips

a. Neutron Flux Power from Excore Neutron Detectors

The DNBR Trip neutron flux power test methodology was identical to the test producing the LPD trip. In this test case, the LPD trip was jumpered out so as to produce a DNBR trip only.

b. Cold Leg Temperature

The cold leg temperature test was a step change in one cold leg temperature input to the CPC from 553.5°F to 583.5°F. The step temperature change between the sensor and CPC's was simulated with the use of decade boxes, in conjunction wiht the test apparatus as shown in Figure 3.1.10.2. For the total response times calculation, the difference (DT) between the computed total response time and the step response time was added to the results of the procedure. This DT accounts for the calculated time difference between a step change and a ramp input to the sensor. Since the test was performed using step changes, the DT was calculated and added to yield accurate time response data.

Measured response time was from signal initiation to CEDM coil deenergization. A high speed visicorder was used to measure the response time.

c. Hot Leg Temperature

The hot leg temperature test was a step change in one hot leg temperature input to the CPC from 612.5°F to 642.5°F. The hot leg temperature response time methodology was the same as that used in the cold leg response time test.

D. Reactor Coolant Pressure from the Pressurizer

The PZR pressure response time test consisted of a step decrease in pressure input from 2175 psia to 2125 psia. The test rig shown in Figure 3.1.10.3 was employed for this test sequence. Test initiation was by means of a fast acting solenoid operating a test valve that allowed accumulator pressure to be applied to the pressure transmitter, thereby decreasing indicated pressure to below the DNBR trip setpoint. Response time measured was from the moment of solenoid actuation to CEDM coil - deenergization.

3.1.10.3 Test Results

The results of all response time tests are tabulated in Table 3.1.10.1. For each parameter tested, for all channels, the response times were within those specified in the ANO - 2, Technical Specifications, Table 3.3-2.

The major problem encountered during the conduct of this test sequence was with noise during the performance of High Neutron Flux Power initiated trips. The amount of electrical noise initially contributed to inaccurate response times of the NI Safety channels. These problems were methodically traced down and eliminated. Suspect response time tests were re-performed and the results yielded accurate and consistent data.

3.1.10.4 Conclusion

The intent and objectives of all response time tests conducted were met successfully. The total response times were within the limits prescribed as shown by the tabulated results of Table 3.1.10.1.

TABLE 3.1.10.1

CPC Response Time Test Results

Parameter	Required Response Time	$MT^{(1)}$	DT	$TT^{(2)}$
Ch. A Low DNBR, NI Safety Ch. A	≤0.39 sec.	0.240	0	0.240
Ch. B Low DNBR, NI Safety Ch. B.	≤0.39 sec.	0.355	0	0.355
Ch. C Low DNBR, NI Safety Ch. C	≤0.39 sec.	0.344	0	0.344
Ch. D Low DNBP, NI Safety Ch. D	≤0.39 sec.	0.229	0	0.299
Ch. A High LPD, NI Safety Ch. A	≤2.58 sec.	2.533	0	2.53
Ch. B High LPD, NI Safety Ch. B	≤2.58 sec.	2.475	0	2.48
Ch. C High LPD, NI Safety Ch. C	≤2.58 sec.	2.375	0	2.38
Ch. D High LPD, NI Safety Ch. D	≤2.58 sec.	1.527	0	1.53
Ch. A Low DNBR, TH1-4610-1	≤1.54 sec.	0.191	1.26	1.45
Ch. B Low DNBR, TH1-4610-2	≤1.54 sec.	0.148	1.26	1.41
Ch. C Low DNBR, TH1-4610-3	≤1.54 sec.	0.149	1.26	1.41
Ch. D Low DNBR, TH1-4610-4	≤1.54 sec.	0.130	1.26	1.39
Ch. A Low DNBR, TH2-4710-1	≤1.54 sec.	0.147	1.26	1.41
Ch. B Low DNBR, Th2-4710-2	≤1.54 sec.	0.154	1.26	1.41
Ch. C Low DNBR, TH2-4710-3	≤1.54 sec.	0.148	1.26	1.41
Ch. D Low DNBR, TH2-4710-4	≤1.54 sec.	0.134	1.26	1.39
Ch. A Low DNBR, TC1-4611-1	≤3.79 sec.	0.175	3.49	3.67
Ch. B Low DNBR, TC1-4611-2	≤3.79 sec.	0.190	3.49	3.68
Ch. C Low DNBR, TC1-4611-3	≤3.79 sec.	0.188	3.49	3.68
Ch. D Low DNBR, TC1-4611-4	≤3.79 sec.	0.181	3.49	3.67

TABLE 3.1.10.1 (cont'd)

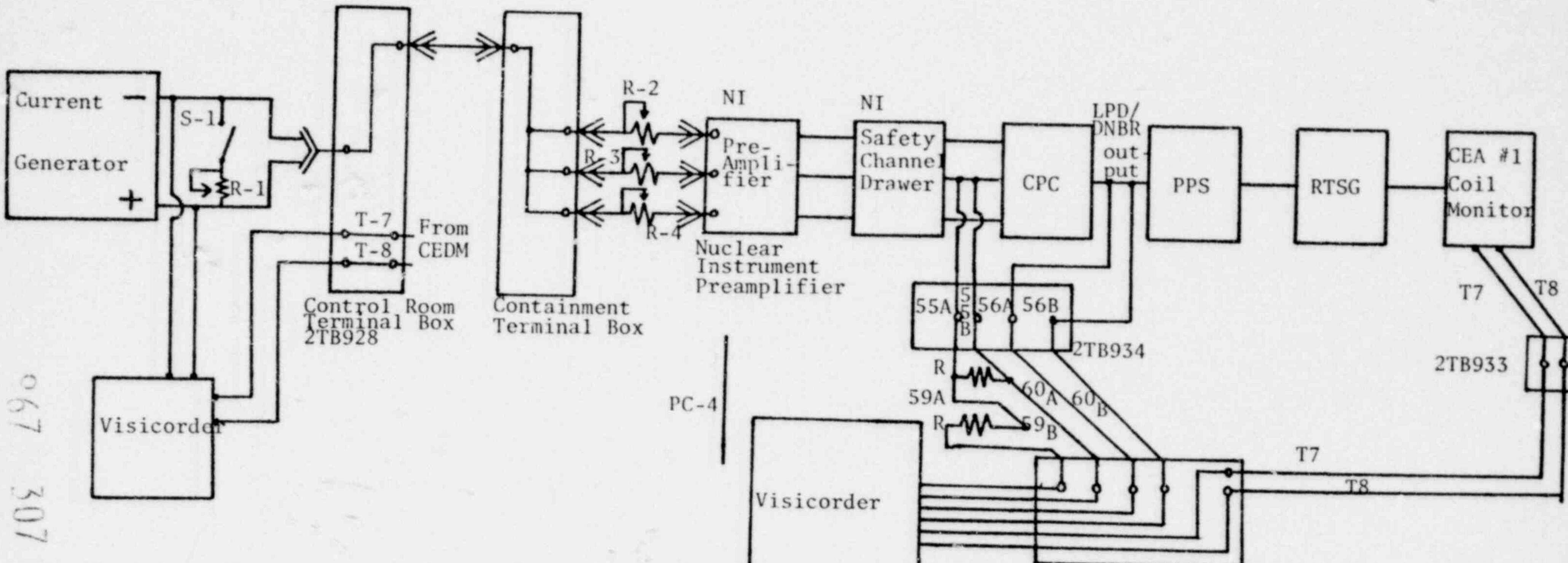
Parameter	Required Response Time	MT(1)	DT	TT(2)
Ch. A Low DNBR, TC1-4711-1	≤3.79 sec.	0.171	3.49	3.66
Ch. B Low DNBR, TC1-4711-2	≤3.79 sec.	0.186	3.49	3.68
Ch. C Low DNBR, TC1-4711-3	≤3.79 sec.	0.179	3.49	3.67
Ch. D Low DNBR, TC1-4711-4	≤3.79 sec.	0.168	3.49	3.66
Ch. A Low PZR Press., PT-4601-1	≤3.19 sec.	0.257	0.011	0.268
Ch. B Low PZR Press., PT-4601-2	≤3.19 sec.	0.204	0.011	0.215
Ch. C Low PZR Press., PT-4601-3	≤3.19 sec.	0.356	0.012	0.368
Ch. D Low PZR Press., PT-4601-4	≤3.19 sec.	0.261	0.010	0.271

(1) Via a Visicorder

(2) TT = DT + MT
DT = Differential Time
TT = Total Time
MT = Measured Time

NOTE: 1) All times are in seconds.

2) Differential time is the difference between the computed total response time and the step response time.



	(1)	(2)
Channel A	R2R384J	R2R384C
Channel B	G2R385J	G2R385C
Channel C	Y2R386J	Y2R386C
Channel D	B2R387J	B2R387C

FIGURE 3.1.10.1

CPC TIME RESPONSE FROM HIGH LINEAR POWER INPUT (LPD TRIP)

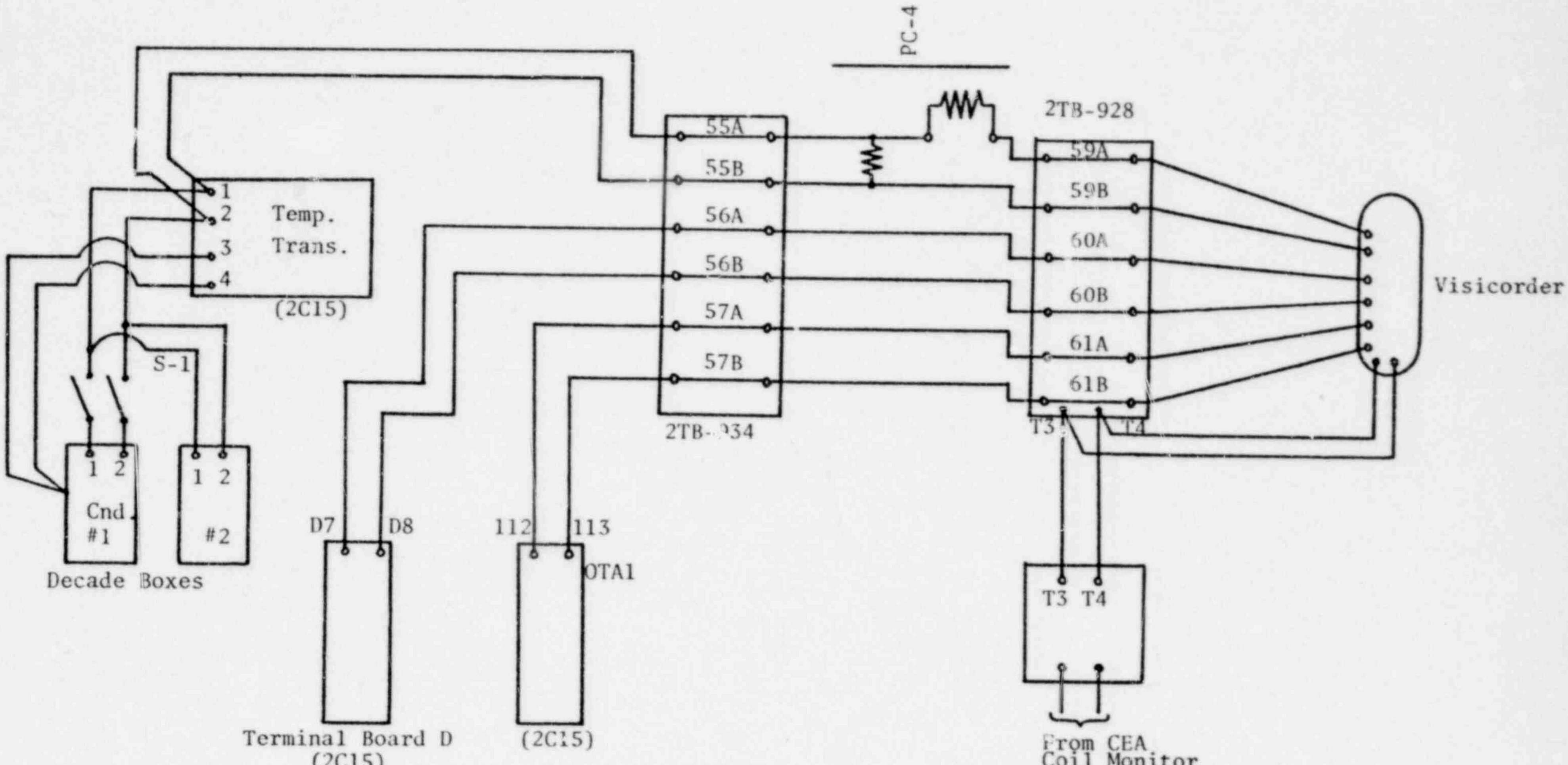


FIGURE 3.1.10.2

CPC TIME RESPONSE TEMPERATURE INPUT TEST APPARATUS (DNBR TRIP)

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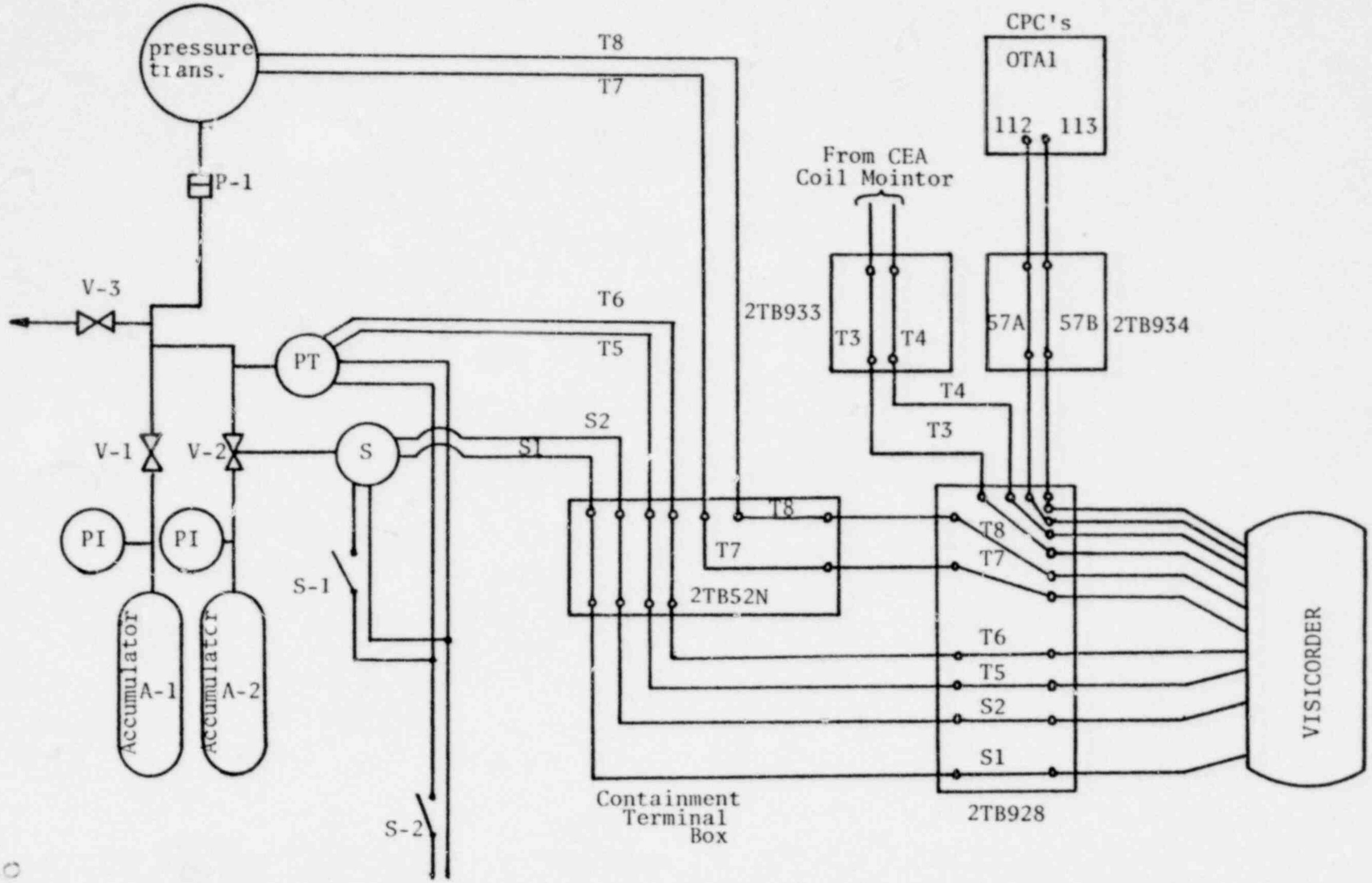

FIGURE 3.1.10.3

CPC TIME RESPONSE PZR PRESSURE INPUT TEST APPARATUS (DNBR TRIP)

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3.1.11 PRESSURIZER SPRAY VALVE CONTROL AND ADJUSTMENT TESTS

3.1.11.1 Purpose

The purpose of this test was to:

A. Establish the proper flow settings for the Pressurizer continuous spray valves (2RC-8A and 2RC-8B) to minimize thermal stresses across the pressurizer spray nozzle.

B. Measure the rate at which the pressurizer/reactor coolant system pressure can be reduced utilizing pressurizer spray flowing through the pressurizer spray valves (2CV-4651 and 2CV-4652) both individually and in parallel.

C. Verify that the pressurizer level control system is capable of automatically maintaining program level.

D. Verify proper operation of the pressurizer level alarms and the Reactor Coolant System high letdown flow alarm.

E. Verify the vibration of the pressurizer spray line is acceptable during pressurizer spray operation.

3.1.11.2 Test Method

A. Continuous Spray Valve Settings

A temporary thermocouple was placed on the spray line piping just before the horizontal run to the pressurizer. Reactor Coolant System temperature/pressure was established at 545°F/2250 psia per the HFT controlling procedure. The temperature of the spray line piping as indicated by the thermocouple was compared to the average reactor coolant system cold leg temperature. Both continuous spray valves (2RC-8A and 2RC-8B) were then positioned to adjust the spray line temperature to 25°F to 30°F lower than the average reactor coolant system cold leg temperature.

B. Pressurizer Spray Effectiveness Test

The Reactor Coolant System was stabilized at the 545°F/2250 psia plateau per the Post Core Hot Functional Controlling Procedure.

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Pressurizer Spray Valve 2CV-4651 and Pressurizer Pressure controller 2PIC-4626A were selected for automatic operation. The setpoint of 2PIC-4626A was then increased to 2300 psia and pressure was allowed to stabilize. Next, the setpoint of pressurizer pressure controller 2PIC-4626B was adjusted to 2000 psia. All pressurizer heaters were deenergized and 2PIC-4626B was then selected for pressure control. The resulting pressure transient was recorded on a strip chart recorder. Following the pressure transient, control was returned to 2PIC-4626A and pressure was allowed to stabilize at 2300 psia. Once again control was shifted to 2PIC-4626B and the resulting transient was recorded on a strip chart recorder. After the second pressure transient control was shifted to 2PIC-4626A and pressure was allowed to stabilize at 2300 psia. The above described pressure transients were repeated, first with only 2CV-4652 in automatic and then with both 2CV-4651 and 2CV-4652 in automatic. Upon completion of the final pressure transient, pressure was returned to 2250 psia and the pressurizer pressure control system was returned to normal. During the pressure transients, the vibration of the spray line piping was monitored.

C. Pressurizer Level Control Test

Reactor Coolant System was stabilized at the 545°F/2250 psia plateau per Post Core Hot Functional Controlling Procedure. Pressurizer level control, charging pump interlock setpoints, pressurizer heaters level interlock setpoints and level deviation alarm setpoints were verified utilizing an external test source to vary the pressurizer level setpoint above and below the actual pressurizer level. Pressurizer level was then manually decreased to verify the low-low level alarm setpoints. Following the level decrease both pressurizer level setpoint and pressurizer level were returned to 40% and the letdown controller placed in automatic. Letdown control setpoint was decreased below the actual pressurizer level. After the maximum

letdown flow for automatic operation was recorded, the letdown control valve controller was placed in manual and the letdown flow increased to the high letdown flow alarm setpoint, then to approximately 148 gpm where controller data was obtained. Pressurizer level and pressurizer level setpoint were matched, and with the letdown controller in automatic, the level setpoint was increased above the actual pressurizer level. After minimum letdowm flow for automatic operation was recorded, the letdown control valve controller was placed in manual and letdown flow decreased to approximately 16 gpm where controller data was obtained. Pressurizer level and pressurizer level setpoint were matched and returned to 40 percent and testing secured.

3.1.11.3 Test Results

A. Pressurizer Continuous Spray Valve

The pressurizer continuous spray valves (2RC-8A and 2RC-8B) were positioned at various valve settings to record piping temperatures. The temperatures yielded a temperature difference (delta-T) between the spray line and the average of the A & B RCS cold legs. The delta-T and pressurizer heater requirements were recorded. From these, an optimum continuous spray valve setting of one quarter (1/4) of one turn was selected. This valve setting resulted in a 38°F delta-T and required either one (1) bank of backup heaters or two (2) banks of proportional heaters to maintian the RCS pressure at 2250 psia.

B. Pressurizer Spray Effectiveness Test

The results of this test are listed below:

SPRAY		INITIAL PRESSURE (PSIA)	FINAL PRESSURE (PSIA)	TRANSIENT TIME (SECONDS)	RATE OF PRESSURE (PSI/MIN)
2CV-4651	Run 1	2300	2050	139.2	108
	Run 2	2300	2050	145.4	103
2CV-4652	Run 1	2300	2050	148.8	100.8
	Run 2	2300	2050	153.6	97.6
2CV-4651	Run 1	2300	2050	111.36	134
and 2CV-4652	Run 2	2300	2050	121.92	123.03

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All results of this test met design criteria specifications.

C. Pressurizer Level Control Test

The results of this test showed that the pressurizer level control system is capable of automatically maintaining the programmed level and that control and alarm setpoints operate properly.

3.1.11.4 Conclusion

A. Continuous Spray Valve Settings

The continuous spray valves were not set in accordance with the test because the required delta-T of 25°F - 30°F between the spray line nozzle and the average RCS cold leg temperature resulted in excessive use of the pressurizer heaters. Combustion Engineering reviewed the results of the continuous spray valve setting test and stated that a delta-T not exceeding 40°F is acceptable. Hence, the present setting yielding a 38°F delta-T is sufficient to ensure that the pressurizer spray nozzle will not undergo excessive thermal transients.

B. Pressurizer Spray Effectiveness Test

The pressurizer spray system operated as designed to reduce pressurizer pressure at the rates required in the acceptance criteria for all spray valve combinations. The pressurizer spray valve line piping vibration was verified acceptable by visual observation during spray operations.

C. Pressurizer Level Control Test

It was proven that the pressurizer level control system is capable of automatically maintaining the programmed level. Also, proper operation of the control and alarm setpoints were verified.

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3.1.12 RCS HEAT LOSS TESTS

3.1.12.1 Purpose

The purpose of this test was to determine the RCS heat loss under hot, zero power conditions (545°F, 2250 psia). The results are used:

A. As a measure of the effectiveness of the RCS insulation, and

B. as input to the secondary plant calorimetric which will be used to determine reactor power.

3.1.12.2 Test Method

A. The total RCS heat loss was determined using the steamdown method. The plant was initially stabilized with steam generators filled to a specified level. Blowdown and feedwater were then secured and the generators were steamed to a specified target level while RCS conditions were maintained. Energy inputs to the system were determined by monitoring reactor coolant pump and pressurizer heater parameters. Similarly, energy losses were determined by calculating energy transferred out of the steam generators and by monitoring charging flow and temperature . A heat balance was then performed to determine the heat loss.

B. Pressurizer heat loss was measured by maintaining the plant at essentially constant conditions. The energy input to the system by the pressurizer heaters was determined by measuring the power input over a specified time interval. The pressurizer heat loss was then equal to the pressurizer heater input. This value was measured for conditions of:

a. no spray to the pressurizer, and
b. bypass spray only to the pressurizer.

3.1.12.3 Test Results

A. The total RCS heat loss was determined as described above to be 3.80×10^6 BTH/hr. This was neglecting the heat loss of the Chemical and Volume Control System (CVCS) which was 3.09×10^6 BTU/hr. Including this term yielded a total heat loss of 6.89×10^6 BTU/hr.

B. Results of the pressurizer heat loss measurements were:

a. 5.96×10^5 BTU/hr. for the case of no pressurizer spray, and
b. 1.04×10^6 BTU/hr. with continuous bypass spray.

3.1.12.5 Conclusions

The measured value of the RCS heat loss satisfied the acceptance criteria. The measured pressurizer heat loss values, however, were not within the acceptance criteria. Recommendation of the vendor after subsequent evaluation was that the measured pressurizer heat loss values were acceptable.

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3.1.13 CHEMICAL AND VOLUME CONTROL SYSTEM INTEGRATED TESTS

3.1.13.1 Purpose

The Chemical & Volume Control System Integrated Test was performed to accomplish the following objectives:

A. Demonstrate the proper operation of the CVCS letdown system. This included the following:

  a. Back pressure controller and back pressure control valves can maintain pressure downstream of level control valves automatically during pressurizer level transients without lifting back pressure relief valve and stay within specified limits.

  b. Letdown flowrates can be maintained within specified limits.

  c. Letdown temperature controller can control letdown temperature in automatic within specified limits.

B. Record Purification Filter D/P, Letdown Strainer D/P, and Ion Exchanger D/P at various letdown flow rates for baseline information.

C. Verify proper flow rates to the Boronometer and process radiation monitor.

3.1.13.2 Method

The back pressure controller 2PIC-4812 in combination with the letdown flow control valves and the back pressure control valves were tested at 545°F and 2250 psia plateau by placing the controller in automatic with a specified pressure setpoint set and simulating a pressurizer level error by adjusting the manual setpoint 5% below level setpoint. This resulted in maximum transient on the back pressure control valves.

Letdown and Purification System operations were performed at the 545°F and 2250 psia plateau. Letdown flowrates of 29, 40, 80, 120 and 138 GPM were established with various combinations of letdown flow control valves, back pressure control valves, purification filters, and purification ion exchangers. During these tests, data was recorded for baseline information.

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3.13.3 Results

The back pressure controller restored the back pressure to the initial value during the performance of system transients, and the back pressure relief valve did not lift. The letdown temperature and pressure was regulated by their respective controllers at normal flowrates.

3.13.4 Conclusions

It was verified that the letdown system functions properly. The Boronometer and process radiation monitor flowrates were satisfactorily set.

3.1.14 CEA EXERCISE TESTS

3.1.14.1 Purpose

The purpose of this test was to verify:

A. Proper operation of the CEDMCS

B. Proper operation of the CEA related computer alarms and interlocks

C. CPC indicated group positions

D. CPC calculated peaking and penalty factors

E. CEAC calculated penalty factors

F. CPC calculated rod shadowing factors

G. Proper transmission of the CEAC penalty factors to the CPC's

3.1.14.2 Test Method

The CEA exercise check test was performed during the Post Core Hot Functional Test sequence at the 545°F/2250 psia plateau i.e. Hot Standby. Prior to increasing the neutron population, a determination of base count rate (Co) was performed for each of the two startup channels. The scaler output of each startup channel drawer was connected to an individual counter-scaler, thus allowing simultaneous count rates to be obtained from the two separate channels.

With the boron concentration at 2038 PPM, CEA Groups A, B, and P were withdrawn one group at a time in manual group mode. CEA group withdrawal was stopped at approximately equal reactivity insertion levels to monitor startup channels 1 and 2 from which count rates were obtained to construct inverse count rate ratio (1/M) plots. When group P reached the upper electrical limit, manual sequential mode was used to withdraw the regulation CEA groups. Again 1/M plots were maintained until all CEA groups were fully withdrawn to the upper electrical limit.

At that point, the CEA groups and individual CEA's were moved into various configurations to test the proper operation of the CEDMCS and its associated computer alarms and interlocks; and to verify the proper calculation of

CEA group positions, radial peaking factors, rod shadowing factors, and CEA deviation penalty factors by the CPC's and CEAC's.

3.1.14.3 Test Results

A. Computer Alarms and Interlocks

A total of 49 comparisons were made at various rod configurations to test computer setpoints, permissives, alarms, and motion inhibits. In all, 15 deficiencies were found. The Lower Sequential Permissive (LSP) and Upper Sequential Permissive (USP) actuation deviations were more than expected in 9 cases but all were conservative from a rod overlap point of view and thus deemed acceptable. The other deficiencies were reviewed and the results were found to be acceptable for all deficiencies except for one group minor deviation alarm. This deficiency was retested and the results found acceptable.

B. CEA Group Positions

Each CPC channel was examined to determine agreement between CPC CEA group positions and the corresponding lowest CPC subgroup position. Each subgroup indication was required to be within $\pm.75$ inches of the indicated group position. In all, 31 CEA group configurations were tested for a total of 125 comparisons.

The CPC indicated position of each subgroup was checked for agreement with the position of the target CEA as shown by the CEAC's. The agreement tolerance was $\pm.75$ inches if both CEAC and CPC receive input from the same reed switch position transmitter (RSPT) and ±3.75 inches if the CEAC and CPC receive input from different RSPT's. A total of 69 CEA group configurations were tested. All comparisons indicated that tolerances were satisfied.

C. CEA Related Factors

a. CPC Planar Radial Peaking Factors (PRPF)

Each CPC channel was checked for proper calculation of its CEA position dependent PRPF's. Two separate comparisons were made: 1) each CPC

channel PRPF was verified at all 20 axial core nodes for 6 CEA group configurations, and 2) each CPC channel PRPF was verified at 3 nodes for 26 CEA group configurations. In all, 6 PRPF's were found dificient.

b. CPC Rod Shadowing Factors (RSF)

Each CPC channel was checked for proper calculation of its CEA position dependent rod shadowing factors. Data was gathered for 10 separate CEA group configurations. Each RSF corresponding to its appropriate Excore Detector was verified for each CPC channel. In all, 120 comparisons were made. Three RSF's were found deficient, one each for CPC channels B, C, and D. Upon review of the data, it was discovered that the RSF deficiencies were due to a procedural error and retesting eliminated the deficiency.

c. CPC Penalty Factors

All four CPC channels were checked for proper CEA position dependent penalty factor calculations. In all, 32 CEA group configurations were tested. Values of TPEN (Total Penalty Factor) and GRPPEN (Group Penalty Factor) were recorded for above CEA positions. Out of 128 comparisons, one value of TPEN was out of tolerance compared to the expected value.

d. CEAC Penalty Factors

Both CEAC's were checked to verify proper calculation of CEA position related penalty factors. CEAC - 1 penalty factors (PF1 RAW) and CEAC - 2 penalty factors (PF2 RAW) were recorded for 17 CEA group configurations. Out of 34 total comparisons of expected vs. recorded values, 1 penalty factor was found out of tolerance, PF1 for CEAC - 1.

e. CEAC/CPC Penalty Factor Transmission

In addition to the aforementioned CEAC penalty factor verifications, each penalty factor input to the CPC's, PF1 RAW and PF2 RAW, was compared to the actual penalty factor output from the CEAC's (PF OUT).

Verification was made at 17 different CEA group configurations. The required deviation between the two variables was ≤ 0.001. Out of 34 total comparisons, one out of tolerance condition was found for each CEAC.

f. Deficiency Review

Discrepancies between the expected and recorded values of the CEA - related factors were attributable to small differences between the actual CEA positions used by the CPC's relative to the recorded positions from the CPC report obtained via the data link to the plant computer. Such differences between the positions used by the CPC's and the recorded positions arose from minor analog to digital conversion differences. These deficiencies were reviewed by Combustion Engineering and were found to be insignificant.

3.1.14.4 Test Conclusion

The deficiencies and Startup Field Reports issued against this procedure identified several CEDMCS inadequancies. Due to the frequency of problems associated with the CEDMCS, it was concluded that hardware changes were necessary in order to increase the CEA maneuvering reliability. In summary, a whosesale replacement of the opto-isolators were performed. The components were replaced with new, state-of-the-art opto-isolators which were more reliable.

In addition, large numbers of strip switches were replaced and/or repaired. These changes along with extensive troubleshooting solved the large majority of the CEDMCS problems.

During the performance of this procedure when CEA motion was occuring in group mode, the CPC DNBR and LPD margin analog indicators cycled in an oscillatory type motion when the shutdown groups or part length CEA's crossed penalty factor boundaries. Review of the situation revealed that this was due to failure to provide an adequate dead-band at the CEA exercise limit boundaries in CEA groups with more than one subgroup. Operating procedure precautions were added to avoid unnecessary protection system trips until a software change could be prepared and approved. This problem caused no non-conservatisms.

3.1.15 STEADY STATE VIBRATION TEST

3.1.15.1 Purpose

The purpose of this test is to monitor pipe vibrations of the systems listed below during all significant plant operating modes that are likely to cause vibration in the subject piping system.

A. Shutdown Cooling System.

B. Steam Generator Blowdown System.

C. Reactor Coolant System.

D. Charging System.

E. Letdown System.

3.1.15.2 Method

A walkdown and visual examination of each system was conducted at each specified test mode. Piping was observed for excessive or abnormal vibration. In addition to the visual inspection of the RCS piping, the reactor coolant pump vibration monitors were checked to verify that no alarm condition was present.

3.1.15.3 Results

No excessive or abnormal vibration was detected in any of the above listed piping systems. All acceptance criteria were met.

3.15.4.1 Conclusions

Vibrations of all piping systems are acceptable as determined by visual inspection.

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3.1.16 SAFETY INJECTION SYSTEM CHECK VALVE RETEST

3.1.16.1 Purpose

The Safety Injection System Check Valve tests were performed to accomplish the following objectives:

A. Verify that Safety Injection Tank (SIT) 2T2D discharge check valve 2SI16D will pass flow at normal operating pressure and temperature.

B. Verify that the Safety Injection loop check valve 2SI15B will pass flow to the Reactor Coolant System at normal operation pressure and temperature.

C. Verify proper operation of the ECCS hot leg injection check valves (2SI-26A, 2SI-27A, 2SI-28A, 2SI-26B, 2SI-27B and 2SI-28B) with normal operating back pressure.

3.1.16.2 Test Method

The following tests were performed under hot standby conditions (545°F/2250 psia):

A. The test of valve 2SI-16D was performed by routing flow from Safety Injection Tank 2T2D through the check valve to drain valve 2CV-5061-2 then to the RWT via return header valve 2CV-5082. The discharge rate from the SIT was controlled by throttling 2SI-17 (SIT drain to RWT).

B. Proper operation of SIT check valve 2SI-15B was verified by charging into the RCS via the High Pressure Safety Injection System. Normal charging flow was established into the flow then passes through 2SI-15B into the RCS.

C. The HPSI ECCS Hot Leg Injection Check Valve test was performed as follows. Normal charging flow was established with 2CV-115 (charging pump discharge to HPSI system) open. Flow through check valves 2SI-26A, 2SI-27A and 2SI-28A was established by opening 2CV-5101-1 (HPSI header to Shutdown Cooling suction). To establish flow to check valves 2SI-26B, 2SI-27B, and 2SI-28B, the crossover valves (2SI-30 and 2SI-31) from HPSI header #1 to HPSI were opened.

3.1.16.3 Test Results

A. Proper operation of check valve 2SI-16D was verified by noting a decrease in the SIT level and pressure.

B. Flow through 2SI-15B was verified by noting a decrease in normal charging flow (2FIS-4863) and an increase in HPSI header pressure (2PI-5020) to slightly above RCS pressure.

C. Proper operation of the Hot Leg Injection check valves was verified by noting a decrease in the charging flow as indicated on 2FIS-4863.

3.1.16.4 Conclusions

All valves tested, performed satisfactorily and passed flow under normal operating conditions.

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3.2 PLANT TESTS

3.2.1 EMERGENCY FEEDWATER SYSTEM S/G WATERHAMMER TESTS

3.2.1.1 Purpose

A. The objective of the test was to demonstrate that the emergency feedwater system (EFWS) will automatically supply water to the steam generator following an emergency feedwater actuation signal (EFAS).

B. The second objective of this procedure was to show that the system design is adequate to prevent a damaging waterhammer.

To ensure proper system performance, the following acceptance criteria were established.

a. The indicated flow rate from 2P7A to SG2E24A, and from 2P7B to SG2E24B was $\geq$ 575 gpm.

b. The EFW valves actuated by an EFAS moved to their proper positions.

c. EFW pump 2P7A starts and reaches $\geq$ rated speed within $\leq$ 20 seconds.

d. Following an EFAS, visual inspections, and evaluation of test data do not reveal any indications of damaging waterhammer.

3.2.1.2 Test Method

A. The RCS is maintained at 450°F $\pm$ 5°F and 1100 $\pm$ 15 psia. The EFW system is aligned for normal operating conditions. Both S/G levels are greater than 50% and the pressurizer level $\geq$ 45%. The levels in both Steam Generators were gradually reduced until S/G levels were at the trip point* and the PPS initiated an EFAS signal.2P7A was actuated first followed by 2P7B after an approximate 90 second time delay. After attaining a 50% level in the steam generators, 2P7A was stopped and flow from 2P7B was regulated to maintain levels in the operating band. The test was also performed at the 545°F test plateau.

* Temporarily reduced to 10% S/G level for the purpose of this test.

B. In order to check for indications of waterhammer occurence, the test was performed at various flow rates. These flow rate combinations were:

a. 2P7B to B S/G at 200 gpm flow.

b. 2P7B to B S/G at 400 gpm flow.

c. 2P7B to B S/G at full flow ($\geq$ 575 gpm).

d. 2P7A to A S/G at 250 gpm flow.

e. 2P7A to A S/G at full flow ($\geq$ 575 gpm).

f. 2P7A and 2P7B to A and B S/G's at full flow ($\geq$ 575 gpm to each S/G).

Following each test of the EFW system, a visual inspection and evaluation of test data was performed to ensure that no indication of damaging waterhammer was incurred.

3.2.1.3 Test Results

Several problems were encountered during initial attempts to perform this test. These included:

(1) 2P7A tripped on overspeed. Maintenance was performed upon the governor and throttle adjustments were made.

(2) During EFW actuation, the emergency suction valves from the service water system opened due to low suction pressure transients. The following maintenance activities were performed:

a. The pump suction startup strainers removed, cleaned and reinstalled.

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b. Air pockets existed in the high points of the pump suction piping and venting the pump suction piping was required.

(3) Additionally, in order to achieve the requirement that 2P7A deliver a total flow to both steam generators of 650 gpm, the turbine controller was adjusted.

(4) 2CV-1039 indication in the control room was inoperable. This was corrected by adjustment of limit switch settings.

Following completion of maintenance activities, the test was repeated successfully.

3.2.1.4 Conclusion

After initial problems were resolved through maintenance and adjustments, the system was retested to verify proper function.

The objectives of the test were met. The Emergency Feedwater System (EFWS) will automatically supply water to the steam generators following an Emergency Feedwater Actuation Signal (EFAS). Also, the system design was demonstrated adequate to prevent damaging waterhammer.

3.2.2 EMERGENCY FEEDWATER FLOW SETTINGS

3.2.2.1 Purpose

The purpose of this test was to flow balance the Emergency Feedwater System (EFWS). To assure design basis system performance, the following acceptance criteria were established:

A. EFW pumps 2P-7A and 2P-7B shall have a minimum recirculation flow of 50 gpm.

B. Given that steam generator secondary pressure is 1015 psia $\pm$ 5 psia; EFW pumps 2P-7A and 2P-7B shall develop 575 + 5 - 10 gpm flow through their corresponding discharge trains while operating independently.

3.2.2.2 Test Method

EFW Pump 2P-7B:

EFW pump 2P-7B was started to verify proper recirculation flow. The flow path to steam generator 2E24B was then lined up and EFW flow was established to 2E24B. The stroke on the control valve was then limited to produce the required flow rate with maximum controller demand. To verify train B system performance, an EFAS was simulated.

The flow path to steam generator 2E24A was lined up and EFW flow was established to 2E24A. The stroke on the control valve was then limited to produce the required flow rate with maximum controller demand. To verify train A performance, an EFAS was simulated.

To determine system performance with trains A and B operating simultaneously, both flow paths were opened and the associated flow rates recorded. Trains A and B from 2P-7B were then isolated and 2P-7B secured.

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EFW Pump 2P-7A:

EFW pump 2P-7A was started and proper recirculation flow verified.

Trains A and B were lined up to steam generators 2E24A and 2E24B. EFW pump 2P-7A speed was then increased until a combined reference flow rate of approximately 650 gpm was established. The demand on speed controller was recorded and 2P-7A secured.

System performance was verified by simulating an EFAS to 2P7A. This was accomplished by energizing steam admission valve and verifying that 2P-7A accelerated to the reference speed and flow rate. Train A performance was determined by isolating train B and recording the flow rate to 2E24A. Train B performance was determined by isolating train A and recording the flow rate 2E24B.

3.2.2.3 Test Results

The mass flow rate developed by the EFW System to the steam generators depends upon secondary pressure, pump speed, recirculation flow rate, system leakage, and control valve position as well as inherent system head losses. Initial testing of the EFWS revealed several component related deficiencies that prevented system performance from meeting the acceptance criteria. The following summarizes the deficiencies and their solutions:

a. EFW pump recirculation flow control valves would not hold their position while throttled. This caused fluctuations in pump discharge pressure and thus flow to the steam generator secondary. Flow Orifices were replaced with "Rodman" type variable restriction orifices. By increasing the head loss factor for these orifices, it was possible to open the recirc. valves to their backseats thus providing stable pump recirculation flow rates.

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b. Some discharge flow stop valves would not isolate their respective trains thus robbing flow from the other train under test. The valve seats and seals were inspected for these valves and reworked as necessary.

c. Speed control for the 2P-7A driver was erratic. The speed controller was reworked by the vendor and the turbine governor controller was adjusted.

d. A flow element was installed backwards. This increased the line resistance in train B from EFW pump 2P-7B. As a result, flow data recorded with the flow element in this condition was inaccurate. The flow element was reinstalled with the proper orientation and its corresponding flow transmitter was calibrated.

After performing the above maintenance, the following results were obtained:

1. Recirculation flow rates for EFW pumps 2P-7A and 2P-7B were successfully set to 50 gpm.

2. The maximum flow rate developed by 2P-7B to Steam Generator 2E24B was 570 gpm against a secondary pressure of 1020 psia.

3. The maximum flow rate developed by 2P-7B to Steam Generator 2E24A was 567 gpm against a secondary pressure of 1018 psia.

4. The maximum flow rate developed by 2P-7A to Steam Generator 2E24A was 575 gpm at a turbine speed of 3600 rpm. Secondary pressure was 1018 psia.

5. The maximum flow rate developed by 2P-7A to Steam Generator 2E24B was 582 gpm at a turbine speed of 3596 rpm. Secondary pressure was 1015 psia.

3.2.2.4 Conclusion

The Emergency Feedwater System performs adequately after an EFAS to mitigate the consequences brought on by reactor decay heat following a design basis accident.

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3.2.3 MAIN STEAM SAFETY VALVE PIPING DYNAMIC TRANSIENT TESTS

3.2.3.1 Purpose

The purpose of this test was to verify the adequacy of the piping restraints for the main steam line during blowdown of one main steam safety valve.

3.2.3.2 Method

Test instrumentation to measure the dynamic response of the system piping was installed on the main steam line and on the safety valve. Testing required increasing the system pressure until safety lift occured.

3.2.3.3 Results

The main steam header pressure was increased from normal operating pressure (∿ 980 psia) until the safety valve lifted @ 1080 psia. Blowdown continuted until system pressure was @ 676 psia. All monitored parameters were acceptable except the measured strain @ SG1 of 625μ in/in vs. the max. expected value of 300μ in/in.

Bechtel (S.F.) evaluated the measured strain of 625μ in/in and determined that it was well within the design limits of the system piping.

3.2.3.4 Conclusions

For the purpose of this test, the safety valve operation and piping restraint performance was satisfactory.

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3.2.4 EMERGENCY FEEDWATER PUMP TURBINE TRIP DYNAMIC TRANSIENT TEST

3.2.4.1 Purpose

The purpose of this test was to verify the adequacy of the piping restraints for the steam supply line to the emergency feedwater pump turbine during fast closure of the turbine trip and throttle valve.

3.2.4.2 Method

Test instrumentation was installed to monitor the dynamic response of the system piping during fast closure of the trip and throttle valve which included load cells, pressure transducers and displacement transducers. During the test, the emergency feedwater pump was brought up to speed to deliver 240 ± 5 gpm. After stable conditions were obtained, the turbine was manually tripped.

3.2.4.3 Results

All dynamic responses were within the expected values except valve closure time, which was .065 sec. (measured) vs. .050 sec. (calculated).

Bechtel, S.F., evaluated the measured valve closure time and determined that all affected transients were acceptable.

3.2.4.4 Conclusion

The dynamic response of the piping system was acceptable.

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3.2.5 SECONDARY HYDROSTATIC TESTS

3.2.5.1 Purpose

The purpose of this test was to perform a hydrostatic test on the non-isolable feed lines and other lines that had been reworked. The rework was required because sections of the feedwater piping failed to meet the material brittle fracture specifications.

3.2.5.2 Test Method

The main steam lines were filled by overflowing the Steam Generators. After the system was filled and vented, the Emergency Feedwater Pump was used to raise system pressure to approximately 500 psig. At 500 psig, the Hydro Pump was started and system pressure slowly raised. The Hydro Pump could not raise pressure above approximately 1100 psig. It was found that several Main Steam Safety valves were leaking as well as the "B" MSIV packing. System pressure was reduced to tighten the MSIV packing and safety valve gags. System pressure was then increased and held at 1360 + 25/-0 psig for 10 minutes followed by a reduction in pressure to 1100 $\pm$ 25 psig for final inspection. The system was depressurized and restored to its normal condition.

3.2.5.3 Test Results

The secondary system was inspected by the Bechtel Hydro Engineer and the ASME Boiler Code Inspector. No leakage was observed on the welds and piping that was required to be inspected.

3.2.5.4 Conclusions

The Main Steam and Feedwater systems met all acceptance criteria for this hydrostatic test and one hydro cycle was used.

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3.2.6 PIPE/COMPONENT HOT DEFLECTION PREOPERATIONAL TESTS

3.2.6.1 Purpose

The purpose of this test is to verify that the piping systems listed below respond to thermal expansion in accordance with the design intent.

A. Main steam.

B. Main steam bypass to condenser.

C. Main steam to emergency feedwater pump turbine driver.

D. Steam Supply to main feedwater pump turbine drivers.

E. Charging system.

F. Shutdown cooling.

G. Pressurizer relief valve discharge piping.

H. Pressurizer surge piping.

I. Pressurizer spray system.

J. Steam generator to blowdown tank.

K. Letdown line.

Note: The design intent is that:

a. The piping expands freely with constraints only at rigid restraints and anchors: i.e., the expansion is not contrained at spring hangers, snubbers, pipe whip restraints or any other obstructions.

b. The pipe returns to its approximate original position in the cold condition.

3.2.6.2 Method

A. Temperature Measurements: Type 'K' magnetic thermocouples were installed at all specified locations in the containment. Temperature was read by and printed by a Data Logger, usually @ 4 hr. intervals. Temperatures outside the containment were taken by magnetic dial indicators and by a hand held thermocouple read by a Fluke Digital Thermometer.

B. Pipe Displacement Measurement: Pipe displacement at selected points was measured by either of the below methods:

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a. Measured displacement of the spring in spring hangers.

b. Measured displacement of mechanical snubbers.

c. Measured displacement of pipe with respect to a fixed reference point. e.g., stanchion, wall, restraint or scribe mark.

d. Measured displacement of pipe as measured by scribe trace or a scratch gage or pencil trace on cardboard target.

The measured displacement of the pipe during heatup from ambient to various temperature plateaus was compared to an expected displacement range. During heatup, piping systems were walked down and visually checked for any interference or binding.

3.2.6.3 Test Results

During heatup, it was observed that the main steam lines were restrained by a whip restraint just downstream of the steam generators. Consultation with Bechtel Stress Group revealed that no excessive stress problem existed. During a subsequent cooldown, shims were removed from the restraints and main steam lines could expand freely. The major deficiency in pipe movement occured on the main steam lines @ Hangers 2EBB-1A-H7 and 2EBB-2A-H7 where the pipes deflected to the West instead of to the East as expected. The North-South movement of the main steam headers in the turbine building was only a fraction of what was expected due to the piping leads to turbine being only at ambient temperature.

Both main steam headers were up against their stops just downstream of the MSIVS.

Several hanger locations of the shutdown cooling system were inaccessible when pipes were hot. Several of the installed scratch gages failed due to the flexible arms not being able to absorb large pipe displacement. Also, several pencil gages had failed due to location problems. During subsequent heatups, failed scratch and pencil gages were replaced and new measurements were taken.

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Shims in the main steam header restraints just downstream of the MSIVS were machined down and during a subsequent hot measurement, an adequate gap was measured.

Bechtel evaluated all questionable pipe movement, including main steamline movement at Hangers 2EBB-1A-H7 and 2EBB-2A-H7 and determined that no stress problem exists.

3.2.6.4 Conclusion

All piping systems in the scope of this test were observed for free expansion during system heatup and no interference was observed. Evaluation of the test results by the Bechtel Stress Group revealed that no stress problems exist in the subject piping system and that the intent of the test has been met.

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4.0 INITIAL APPROACH TO CRITICALITY

4.1 Purpose

To provide a safe organized method for attaining the initial criticality of the Arkansas Nuclear One - Unit 2 Reactor.

4.2 Test Method

The approach to criticality commenced on 12/4/78 with the Reactor Coolant System at 260°F and 460 psia, all CEA groups fully inserted, and two reactor coolant pumps running. The Reactor Coolant System was at a boron concentration of 1996 ppm.

Prior to increasing the neutron population, a determination of base count (C_o) was performed for each of the two startup channels. The scaler output of each startup channel drawer was connected to an individual counter-scaler, thus allowing simultaneous count rates to be obtained from the two separate channels. A similar connection was made to two logarithmic power channels to provide instrumentation overlap and continuity in the event criticality was not achieved prior to the startup channel deactivation at 1×10^{-6}% power.

The neutron population was increased initially by withdrawing the CEA Groups in incremental steps resulting in an essentially unrodded core (CEA Group 6 was returned to ~ 75" withdrawn for later maneuverability when critical). The CEA withdrawal sequence, and intervals are shown in Table 4.1, and correspond to approximately equal reactivity insertion levels. Following each withdrawal increment, the count rate (C_i) on each startup channel was recorded and a ratio, C_i/C_o=M, calculated. From this, a plot of inverse count râte (1/M) versus CEA withdrawal was generated. Figure 4.1 shows this plot for each startup channel. Several CEDMCS related problems occurred during the CEA withdrawal sequence. Troubleshooting corrected these problems so that the withdrawal sequence could be continued.

After CEA withdrawal had been completed, the dilution of the RCS boron concentration was commenced at an initial rate of approximately 2 ppm/minute (88 gpm DMW). During the RCS dilution from 1996 to 1196 ppm, the RCS was sampled for boron concentration at 30 minute intervals and a plot of 1/M versus dilution time was maintained for each startup channel. After a 30 minute stabilization period at 1196 ppm, the RCS dilution was recommenced at a rate of approximately 1 ppm/minute (44 gpm DMW). The

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RCS was sampled for boron concentration at 15 minute intervals and plots of 1/M versus dilution time were continued. The plot was frequently extrapolated to provide estimates of the time remaining to initial criticality. Figure 4.2 shows 1/M versus dilution time curves for each startup channel, and Figure 4.3 shows a curve of RCS boron concentration versus dilution time. The RCS boron concentration was decreased until criticaity was achieved at a boron concentration of 980 ppm at $1 \times 10^{-7}\%$ power. CEA Group 6 was then used to increase reactor power to $1 \times 10^{-5}\%$ at which point it was stabilized. CEA Group 6 was at 82 inches withdrawn at criticality.

4.3 Test Results

Initial criticality of the Arkansas Nuclear One - Unit 2 reactor was achieved at 1455 on December 5, 1978. The startup was performed by first withdrawing CEAs and by diluting the reactor coolant system. Criticality was achieved with CEA Group 6 at 82 inches withdrawn and a boron concentration of 980 ppm. This was in good agreement with the predicted value of 996 ppm with CEA Group 6 at 75 inches withdrawn. No abnormal conditions were observed throughout the approach to criticality other than the CEDMCS problems previously mentioned.

4.4 Conclusion

Arkansas Nuclear One - Unit 2 was brought to criticality in a safe and organized manner. The critical CEA position and boron concentration agreed well with the predicted values.

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TABLE 4.1

CEA Withdrawal Sequence

Procedure Step #	CEA Group	Inches Withdrawn
7.9	All CEA's	Full in
7.11.1	Shutdown A	24
7.11.2	Shutdown A	34
7.11.3	Shutdown A	56
7.11.4	Shutdown A	Full out
7.11.5	Shutdown B	30
7.11.6	Shutdown B	52
7.11.7	Shutdown B	Full out
7.11.8	Part Length P	Full out
7.12.1	Regulating 1 *	78
7.12.2	Regulating 2 *	78
7.12.3	Regulating 3 *	99
7.12.4	Regulating 4 *	138
7.12.5	Regulating 6 *	146
7.12.13	All CEA's	Full out

*Withdrawal of regulating groups performed in manual sequential mode.

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FIGURE 4.1

INITIAL APPROACH TO CRITICALITY

BOL, 1st Cycle, 260^{o}F, 460 psia

POOR ORIGINAL

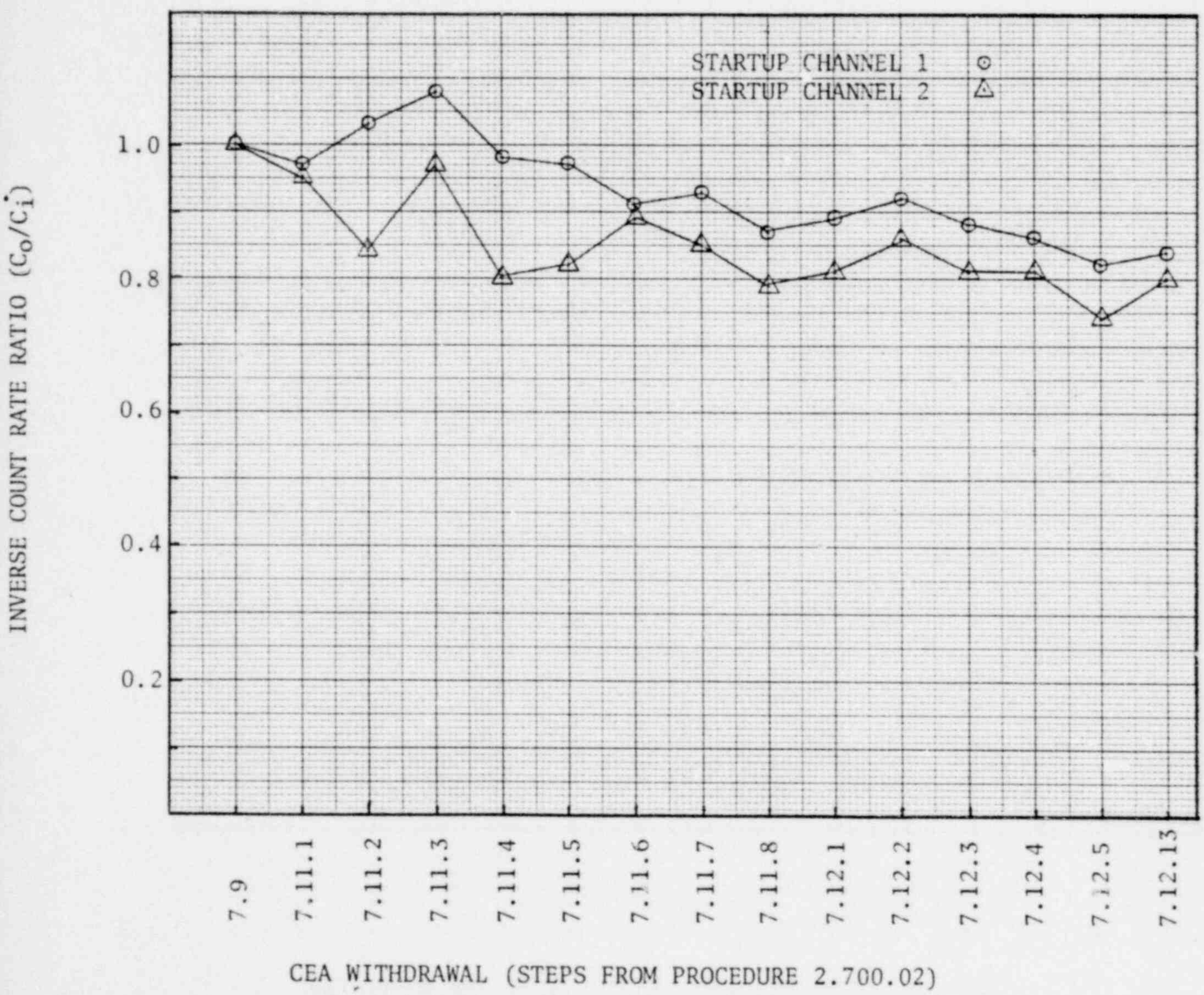

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FIGURE 4.2

ARkansas Nuclear One Unit 2

INITIAL APPROACH TO CRITICALITY

BOL, 1st Cycle, 260°F, 460 psia

POOR ORIGINAL

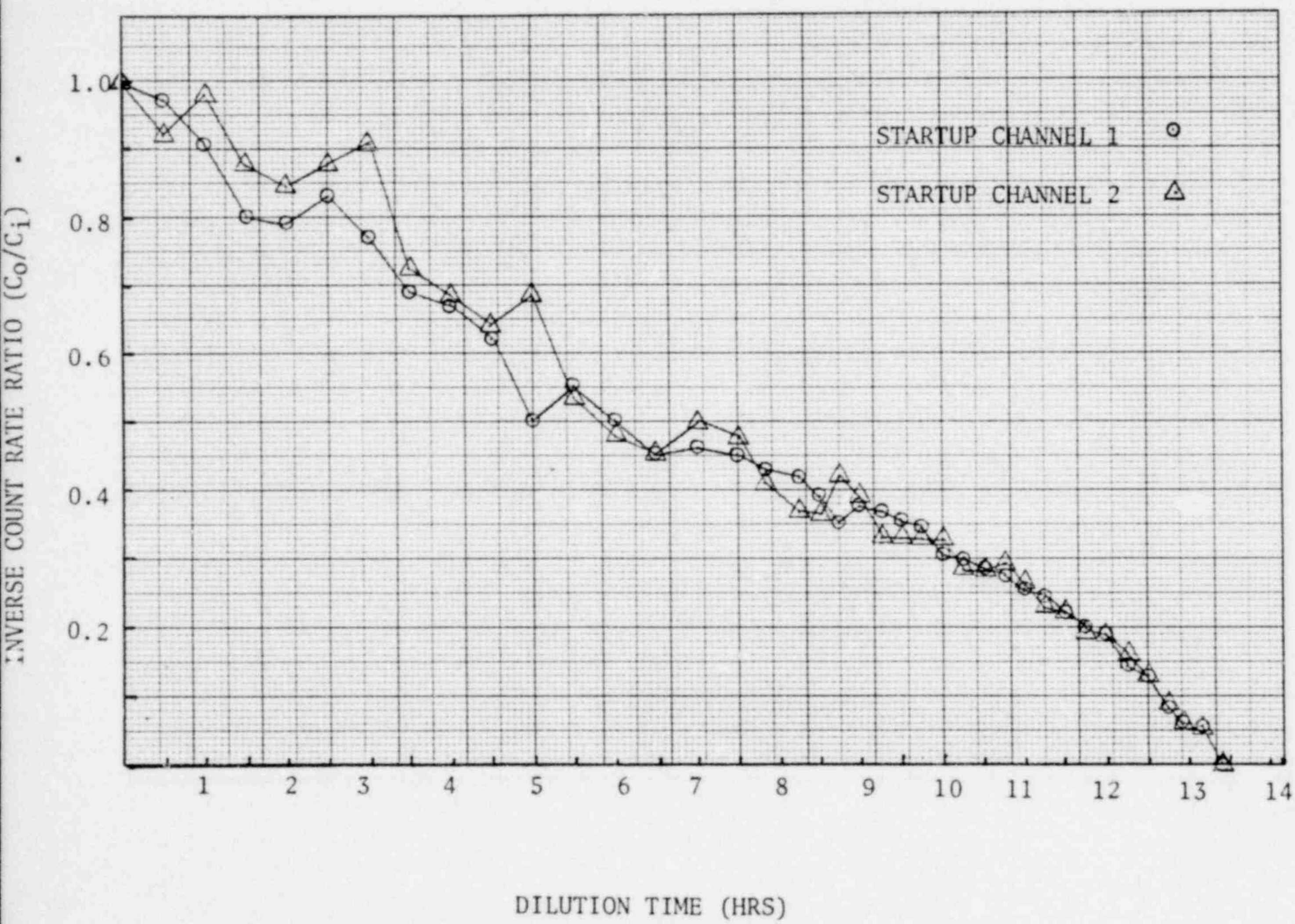

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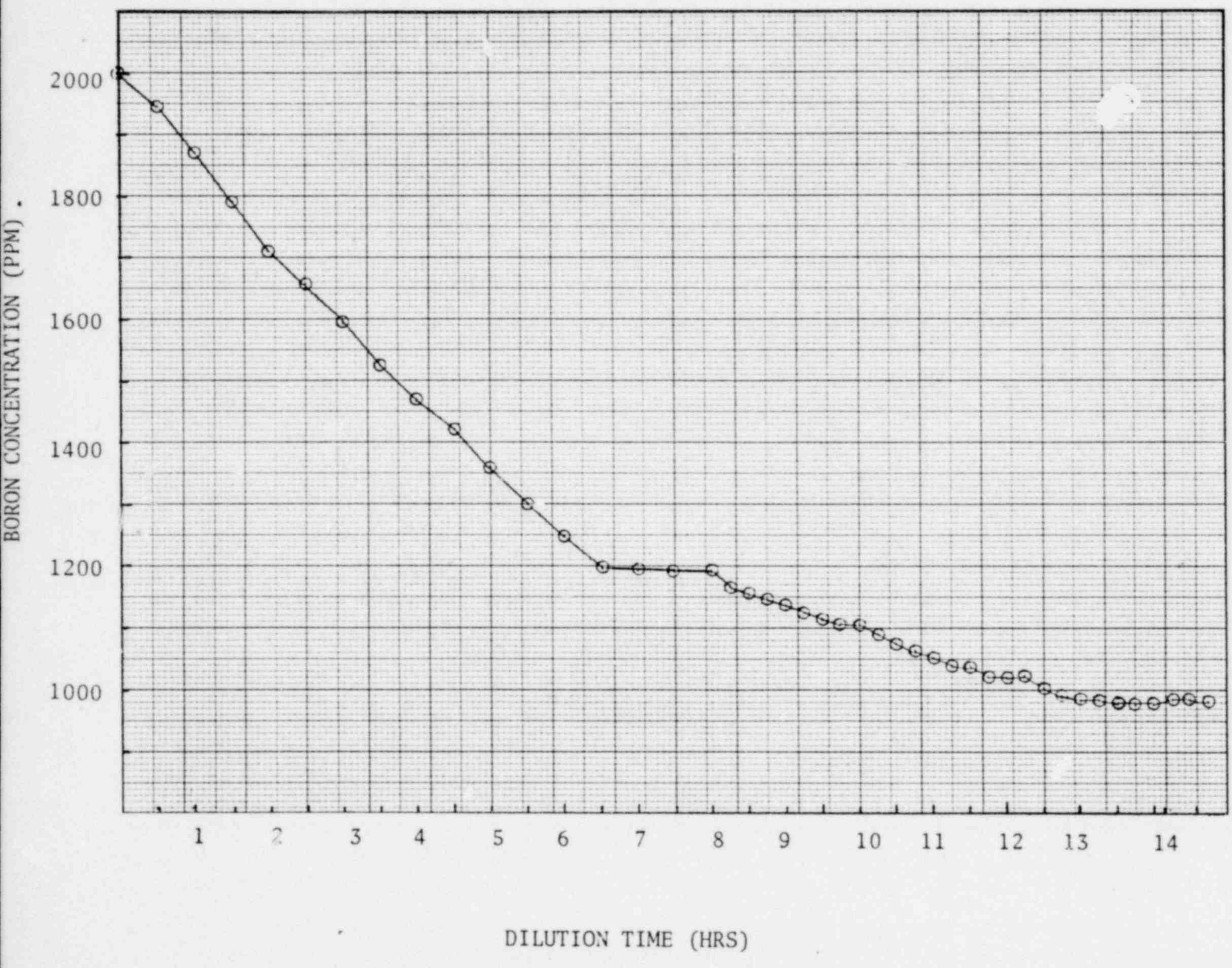
FIGURE 4.3

ARKANSAS NUCLEAR ONE UNIT 2

INITIAL APPROACH TO CRITICALITY

BOL, 1st CYCLE, 260°F, 460 psia

POOR ORIGINAL



NOTE: RCS boron concentration by chemical analysis

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5.0 LOW POWER PHYSICS TESTS

5.1 INTRODUCTION

The objectives of the Low Power Physics Test program were to measure the physics characteristics of the as-built core and to demonstrate conformance with applicable Technical Specifications. The measurements were conducted at 260°F, 460 psia and 545°F, 2250 psia with the reactor operating between 10^{-4} and 10^{-3}% of full power - high enough to provide a good signal to noise ratio but low enough to avoid sensible heat effects. A reactivity computer system was used for reactivity measurements with two dual ex-core uncompensated ion chambers being used as input. Measurements of CEA group worths, isothermal temperature coefficients, dropped, ejected, and stuck CEA worths, and of inverse boron worths were made. A summary of the results appears in Table 5.1. Refer to the applicable sections of this chapter for more detailed information. The Low Power Physics Test measurement results were in close agreement with the predicted values.

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TABLE 5.1

LOW POWER PHYSICS TEST RESULTS

I. 260°F, 460 PSIA

MEASUREMENT	PREDICTED	MEASURED
ARO Critical Boron Conc.	1006 ppm	999 ppm
EARO ITC	$-.076\times10^{-4}\Delta\rho/°F$	$+.08\times10^{-4}\Delta\rho/°F$
CEA Group 6 Worth	0.376%Δρ	0.404%Δρ
CEA Group 5 Worth	0.662%Δρ	0.624%Δρ
CEA Group 4 Worth	0.331%Δρ	0.373%Δρ
CEA Gps. 6, 5, 4 in Differential Boron Worth	65 ppm/%Δρ	64.89 ppm/%Δρ
CEA Gps. 6, 5, 4 ITC	$-0.22\times10^{-4}\Delta\rho/°F$	$0.08\times10^{-4}\Delta\rho/°F$
Sequential Worth of CEA Gps. 6, 5, 4	1.38%Δρ	1.39%Δρ
All Rods Out Differential Boron Worth	65 ppm/%Δρ	64.97 ppm/%Δρ

II. Heatup to 360°F, 460 PSIA

ITC ARO	$-.035\times10^{-4}\Delta\rho/°F$	$-.096\times10^{-4}\Delta\rho/°F$

III. Pressurization to 1100 PSIA

Pressure Coefficient *	$-.41\times10^{-6}\Delta\rho$/psia	$-.462\times10^{-6}\Delta\rho$/psia

IV. Heatup From 360°F to 450°F

ITC ARO	$-.018\times10^{-4}\Delta\rho/°F$	$.103\times10^{-4}\Delta\rho/°F$

V. Pressurization From 1100 PSIA to 2250 PSIA

Pressure Coefficient *	$-.38\times10^{-6}\Delta\rho$/psia	$-.58\times10^{-6}\Delta\rho$/psia

VI. Heatup From 450°F to 545°F

ITC ARO 450°F to 500°F *	N/A	$.069\times10^{-4}\Delta\rho/°F$
ITC ARO 500°F to 545°F *	N/A	$.016\times10^{-4}\Delta\rho/°F$

* For information only

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TABLE 5.1 (Cont'd.)

LOW POWER PHYSICS TEST RESULTS

VII. 545°, 2250 PSIA

MEASUREMENT	PREDICTED	MEASURED
Worst Dropped PLCEA(P-24)	-.027%Δρ	-.0342%Δρ
Worst Dropped PLCEA Sub-Group (P-1)	-.116%Δρ	-.133%Δρ
Worst Dropped CEA (6-1)	-.132%Δρ	-.127%Δρ
Next Worst Dropped CEA (6-47)	-.0717%Δρ	-.085%Δρ
ARO Critical Boron Conc.	1001 ppm	1012 ppm
CEA Group 6 Worth	0.557%Δρ	0.568%Δρ
CEA Group 5 Worth	0.514%Δρ	0.524%Δρ
CEA Group 4 Worth	0.766%Δρ	0.743%Δρ
CEA Group 3 Worth	0.778%	0.792%
Gps 6-3 @ LEL, Gp 2 @97.5°WD		
Differential Boron Worth	76.5 ppm/%Δρ	72.56 ppm/%Δρ
ITC, ZPDIL	$-.5\times10^{-4}$ Δρ/°F	$-.48\times10^{-4}$ Δρ/°F
MTC, ZPDIL	$-.34\times10^{-4}$ Δρ/°F	$-.32\times10^{-4}$ Δρ/°F
Worst Ejected CEA at ZPDIL (4-11)	.376%Δρ	.351%Δρ
Next Worst Ejected CEA at ZPDIL (6-1)	.348%Δρ	.322%Δρ
CEA Group 2 Worth	.942%Δρ	.916%Δρ
CEA Group 1 Worth	1.258%Δρ	1.275%Δρ
CEA Groups 6-1 in		
Differential Boron Worth	76.5 ppm/%Δρ	72.30 ppm/%Δρ
ITC, CEA Gps 6-1 in	0.81×10^{-4} Δρ/°F	-1.07×10^{-4} Δρ/°F
MTC, CEA Gps 6-1 in	$-.65\times10^{-4}$ Δρ/°F	$-.908\times10^{-4}$ Δρ/°F
PLCEA Group P Worth	.400%Δρ	.418%Δρ
CEA Group B Worth	3.112%Δρ	3.402%Δρ
CEA Group A Worth minus Stuck CEA A-52	1.99%Δρ	2.032%Δρ
Total Inserted Worth of CEA Gps 6-A (minus the Stuck CEA) and w/Gp. P	10.736%Δρ	10.665%Δρ
Sequential Worth of CEA Groups 1-6	4.85%Δρ	4.746%Δρ
ITC, ARO	0.0Δρ/°F	0.323×10^{-4} Δρ/°F
MTC, ARO	$.16\times10^{-4}$ Δρ/°F	$.1923\times10^{-4}$ Δρ/°F
CEA 6-01	.135%Δρ	.1207%Δρ
PLCEA Group P Integral Worth	.226%Δρ	.25%Δρ

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5.1.1 CEA COUPLING TEST

5.1.1.1 Purpose

To confirm that the CEAs are coupled to their respective CEA extension shafts.

5.1.1.2 Test Method

The test was conducted at 260°F and 460 psia. Each CEA was inserted one at a time until a reactivity decrease of $\geq$.5¢ had been achieved. The CEA was then returned to its original position.

5.1.1.3 Test Results

With each movement of a CEA, a resulting change in reactivity was seen. All CEAs exhibited this, thus indicating all were properly coupled.

5.1.1.4 Conclusion

Since all CEAs, when moved, caused an appropriate decrease or increase in reactivity, it was concluded that the CEAs were properly coupled.

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5.1.2 CEA/PLCEA SYMMETRY TESTS

5.1.2.1 Purpose

CEA/PLCEA symmetry tests were performed to verify that the CEAs and fuel were loaded in the core as designed.

5.1.2.2 Test Method

The test was performed with the NSSS at 545°F, 2250 psia, and CEA Groups A, B, 1, 2, 3, 4, 5 and I at the upper electrical limit (UEL). CEA Group 6 was initially at ~ 75" withdrawn.

The test program began by inserting the center CEA (CEA 6-1) to its lower electrical limit, (LEL). Reactivity and Power swings, as monitored on the reactivity computer, were compensated for by CEA Group 6 movement. Subsequent symmetric CEA subgroups were checked for symmetry as follows: the first CEA of a subgroup (the reference CEA) was inserted to its LEL while reactivity and power swings were compensated for with CEA Group 6 movement. (After conditions stabilized, Group 6 was not moved again until the first CEA of the next symmetric subgroup was inserted.) The second CEA was then inserted to its LEL while the reference CEA was withdrawn to its UEL. Any reactivity difference, as indicated on the reactivity computer, was noted. The third CEA was then inserted to its LEL while the second was withdrawn to its UEL. This trading process was continued until all the CEAs of a particular symmetric subgroup were traded. The last CEA was then withdrawn as the reference CEA was reinserted. This provided a means of evaluating for reactivity drift which may have occurred since the reference CEA was initially checked.

The above described procedure was carried out for each of the symmetric Groups of Table 5.1.2.1.

5.1.2.3 Test Results

Each CEA of Table 5.1.2.1 was checked within its symmetric CEA subgroup as described above. Data was recorded and analyzed per procedure and all symmetric CEAs were found to agree (within a subgroup) to within the acceptance criteria of ± 1.5¢ of the symmetric CEA subgroup average. Figure 5.1.2.1 summarizes the results of this test.

967 348

5.1.2.4 Conclusions

Since the acceptance criteria for this test (as described above) were satisfactorily met, it can be concluded that the Fuel and CEAs were correctly loaded.

967 349

350

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TABLE 5.1.2.1
SYMMETRIC CEA/PLCEA GROUPS

Symmetric CEA/PLCEA Groups are shown below. The R designation indicates the CEA/PLCEA that the group is referenced to for purposes of this test:

Group 1	Group 2	Group 3	Group 4	Group 5	Group 6	Group 7
R-6-46	R-5-58	R-4-10	R-3-62	R-2-71	R-2-6	R-1-38
6-47	5-59	4-11	3-63	2-74	2-7	1-39
6-48	5-60	4-12	3-64	2-77	2-8	1-40
6-49	5-61	4-13	3-65	2-80	2-9	1-41
			3-66			1-42
			3-67			1-43
			3-68			1-44
			3-69			1-45

Group 8	Group 9	Group 10	Group 11	Group 12	Group 13	Group 14
R-A-50	R-A-70	R-B-14	R-B-30	R-B-2	R-P-22	R-P-26
A-51	A-72	B-15	B-31	B-3	P-23	P-27
A-52	A-73	B-16	B-32	B-4	P-24	P-28
A-53	A-75	B-17	B-33	B-5	P-25	P-29
A-54	A-76	B-18	B-34			
A-55	A-78	B-19	B-35			
A-56	A-79	B-20	B-36			
A-57	A-81	B-21	B-37			

FIGURE 5.1.2.1

CEA/PLCEA SYMMETRY TEST RESULTS

N
↑

(X)

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
1							9 -0.38		9 -0.30						
2						4 -0.70		2 -0.74		4 -0.57					
3			5 -0.10		8 -0.46		7 -1.22		7 -1.45		8 -0.24		5 -0.05		
4				1 -0.31		11 -0.49		13 -0.19		11 -0.20		1 -0.06			
5			8 -0.24		14 -0.11		10 -0.48		10 -0.33		14 -0.07		8 -0.09		
6		4 -0.24		11 -0.20		3 -0.52		6 -0.37		3 -0.49		11 -0.05		4 -0.15	
7	9 +0.13		7 -0.35		10 -0.21		12 -0.18		12 -0.14		10 -0.12		7 -1.20		9 +0.15
8		2 +0.28		13 -0.03		6 -0.18				6 -0.17		13 -0.03		2 -0.48	
9	9 +0.13		7 +0.05		10 -0.12		12 R		12 +0.04		10 -0.01		7 +0.22		9 +0.15
10		4 -0.11		11 +0.05		3 R		6 R		3 -0.32		11 +0.11		4 -0.02	
11			8 +0.05		14 R		10 R		10 -0.01		14 +0.005		8 +0.06		
12				1 R		11 R		13 R		11 +0.02		1 +0.06			
13			5 R		8 R		7 R		7 +0.03		8 +0.07		5 0.00		
14						4 R		2 R		4 -0.14					
15							9 R		9 -0.02						

(X)

X Y	Symmetric Group # (See Table 5.1.2.1) Deviation from reference CEA (R) in ¢. (See Table 5.1.2.1)

Acceptance Criteria = ± 1.5¢

(X) = Uncompensated Ion Chamber at 10'8" Radius

967 351

5.1.3 ISOTHERMAL TEMPERATURE COEFFICIENT TESTS

5.1.3.1 Purpose

Isothermal temperature coefficient (ITC) tests were performed to determine the as-built values of ITC as a function of soluble boron concentration and CEA configuration. A 'measured' moderator temperature coefficient (MTC) was then derived to verify compliance with Technical Specifications and to ensure consistency with certain assumptions in the Safety Analysis.

5.1.3.2 Test Method

The ITC for a given temperature and core configuration was measured by changing the primary system temperature and recording the change in reactivity. Each particular measurement involved a series of temperature changes wherein the primary system temperature was changed and returned to the base temperature two to three times while reactivity was being calculated and recorded on the reactivity computer. From each temperature change, an individual ITC was calculated (change in reactivity divided by change in temperature), and the final ITC value was taken as the average of the individual ITCs. For ITC measurements performed at 260°F, the initial temperature change was an increase of 10°F. For ITC measurements at 545°F, the initial temperature change was a decrease of 10°F.

5.1.3.3 Test Results

Seven ITC measurements were performed during the Low Power Physics Testing for various base temperatures, CEA configurations, and soluble boron concentrations. These measurements and their predictions, acceptance criterion, and results are delineated in Table 5.1.3.1.

The corresponding moderator temperature coefficients (MTCs) for the ITCs of Table 5.1.3.1 at 545°F, are given in Table 5.1.3.2 along with their predictions, and acceptance criteria.

5.1.3.4 Conclusion

All measured Isothermal Temperature Coefficients as well as Moderator Temperature Coefficients are within the applicable acceptance criteria, and are therefore acceptable.

TABLE 5.1.3.1: ISOTHERMAL TEMPERATURE MEASUREMENT RESULTS

MEASUREMENT CONDITIONS	PREDICTED VALUE $(X10^{-4} \Delta k/k/°F)$	MEASURED VALUE $(10^{-4} \Delta k/k/°F)$	ERROR[4] $(X10^{-4} \Delta k/k/°F)$
260°F, EARO[1], 460 psia, ~996 ppm	-0.076(1006 ppm)	+0.080	-0.156
260°F, CEA Groups 6, 5 & 4 at LEL, 460 psia, ~907 ppm	-0.220(917 ppm)	-0.079	-0.141
Heatup (260°F to 360°F), Gp. 6 at 115" WD, 460 psia	-0.035	+0.096	-0.131
Heatup (360°F to 450°F), Gp. 6 at 115" WD, 1100 psia	-0.018	+0.103	-0.121
Heatup (450°F to 500°F), Gp. 6 at 115" WD, 2250 psia	[2]NA	+0.069	N/A
Heatup (500°F to 545°F), Gp. 6 at 115" WD, 2250 psia	[2]NA	+0.016	N/A
545°F, ZPDIL[3], 2250 psia, ~809 ppm	-0.500(N/A)	-0.479	-0.021
545°F, CEA Gps. 6 through 1 at LEL, 2250 psia, ~657 ppm	-0.810(638 ppm)	-1.068	+0.258
545°F, EARO[1], 2250 psia. ~1016 ppm	+0.000(1001 ppm)	+0.032	-0.032

NOTES:

1. EARO: Essentially All Rods Out, i.e., all CEAs at their UELs except Group 6 which is ≥ 130" withdrawn.
2. No acceptance criteria or predictions applicable.
3. ZPDIL: Zero Power Dependent Insertion Limit, i.e., CEA Groups 6 through 3 are at their LELs and CEA Group 2 is at 97.5" withdrawn.
4. ERROR = Predicted Value - Measured Value.

   The error is compared to the acceptance criteria given below.

   "The measured ITC shall be within ± 0.5 x $10^{-4} \Delta k/k/°F$ of the predicted ITC value for the test condition.

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TABLE 5.1.3.2: ISOTHERMAL TEMPERATURE MEASUREMENT RESULTS

MEASUREMENT CONDITIONS	PREDICTED VALUE ($X10^{-4}\Delta k/k/°F$)	MEASURED VALUE ($10^{-4}\Delta k/k/°F$)	ERROR[4] ($X10^{-4}\Delta k/k/°F$)
545°F, ZPDIL[1], 2250 psia, ~809 ppm	-0.340(N/A)	-0.320	-0.020
545°F, CEA Gps. 6 through 1 at LEL, 2250 psia, ~657 ppm	-0.650(638 ppm)	-0.908	+0.258
545°F, EARO[1], 2250 psia, ~1016 ppm	+0.160(1001 ppm)	+0.192	-0.032

NOTES:

1. ZPDIL: Zero Power Dependent Insertion Limit, i.e.; CEA Groups 6 through 1 at their LELs and CEA Group 2 is at 97.5" withdrawn.

2. EARO: Essentially All Rods Out; i.e., all CEAs at their UELS except for Group 6 which is $\geq$ 130" withdrawn.

3. ERROR: Predicted value - Measured value
   The error is compared to the acceptance criteria given below:

   "The measured MTC shall be within $\pm$ 0.5 x $10^{-4}\Delta k/k/°F$ of the predicted MTC value for the test condition. The measured MTC for the test condition at 545°F or greater shall be less positive than 0.5 x $10^{-4}\Delta k/k/°F$, (Technical Specification 3.1.1.4 - BOL condition)."

967 554

5.1.4 CEA GROUP WORTH TESTS

5.1.4.1 Purpose

This series of tests provided for the measurement of the reactivity worths of the CEA's, to verify the accuracy of models used for design and safety calculations by demonstrating that they accurately predict the measured CEA worths. The group worths which were measured are as follows:

A. RCS at 260°F/460 psia

1. CEA Groups 6, 5, 4 (no overlap)
2. CEA Groups 6, 5, 4 (with overlap)

B. RCS at 545°F/2250

1. CEA Groups 6 thru 1 (no overlap)
2. CEA Groups P, B, A
3. CEA Groups 6 thru 1 (with overlap)
4. CEA Group P (group 6 @ 120")

5.1.4.2 Test Method

All CEA group reactivity worths were measured by introducing a continuous change in the RCS boron concentration (boration/dilution) and maintaining reactor power by inserting or withdrawing CEA groups in increments. The resulting reactivity traces were then reduced to obtain integral reactivity worths.

During dilution of the CEA's into the core, the group sequence was 6-5-4-3-2-1-P-B-A. Hence, the core was in a different rodded state for each group worth measurement. Also, since safety considerations preclude the reactor being critical with all CEAs inserted, measurement of the worth of shutdown Group A was slightly modified. This was done by using a combination of boration/dilution, group trips, and extrapolating to find the total integral worth of Group A.

5.1.4.3 Test Results

Tables 5.1.4.1 and 5.1.4.2 present the results of the CEA group worth measurements in comparison to predicted worths. Also, the integral

967 355

group reactivity worths measured at 545°F are displayed in Figures 5.1.4.1 through 5.1.4.10. The worth of Group P was measured twice, once during the sequential insertion of all of the CEA's and again with ARO except Group 6 which was at 120" withdrawn. Figure 5.1.4.11 presents the overlapped Regulating CEA Group integral worth curve.

5.1.4.4 Conclusion

All of the measured CEA integral reactivity worths are in good agreement with predicted values and satisfy all acceptance criteria.

967 356

TABLE 5.1.4.1

CEA GROUP REACTIVITY WORTHS
260°F/460 psia

CEA GROUP	NUMBER OF CEA's	FUNCTION	PREDICTED WORTH (% Δk/k)	MEASURED WORTH (% Δk/k)	ACCEPTANCE LIMITS (% Δk/k)
6	5	Regulating	0.376	0.404	.276-.476
5	4	Regulating	0.662	0.624	.562-.762
4	4	Regulating	0.331	0.373	.231-.431

967 357

TABLE 5.1.4.2

CEA GROUP REACTIVITY WORTHS
545°F/2250 psia

CEA GROUP	NUMBER OF CEA's	FUNCTION	PREDICTED WORTH (% Δk/k)	MEASURED WORTH (% Δk/k)	ACCEPTANCE LIMITS (% Δk/k)
6	5	Regulating	0.557	0.568	0.457-0.657
5	4	Regulating	0.514	0.524	0.414- .614
4	4	Regulating	0.766	0.743	0.651- .881
3	8	Regulating	0.778	0.792	0.661- .895
2	8	Regulating	0.942	0.916	0.801-1.083
1	8	Regulating	1.258	1.275	1.069-1.447
P	8	Shaping	0.400	0.418	0.300-0.500
B	20	Safety	3.112	3.402	2.645-3.579
A	16	Safety	3.567	3.55	3.032-4.102
*A	16	Safety	1.99	2.032	1.692-2.288
**P	8	Shaping	0.226	0.25	0.15-0.35

*Worth of Group A without stuck CEA.
**Group 6 at 120" withdrawn.

967 358

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP 6

POOR ORIGINAL

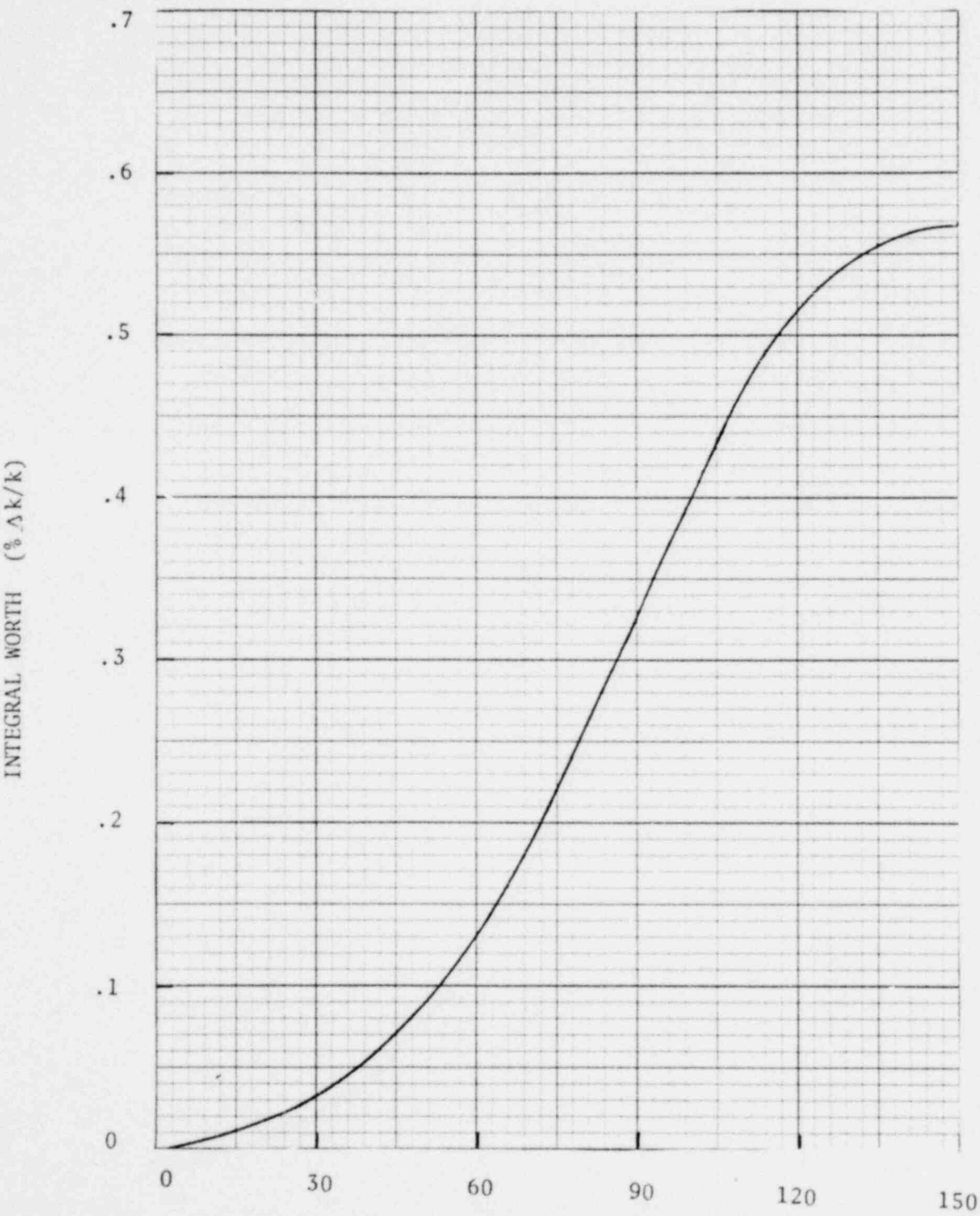


FIGURE 5.1.4.1

967 359

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP 5

POOR ORIGINAL

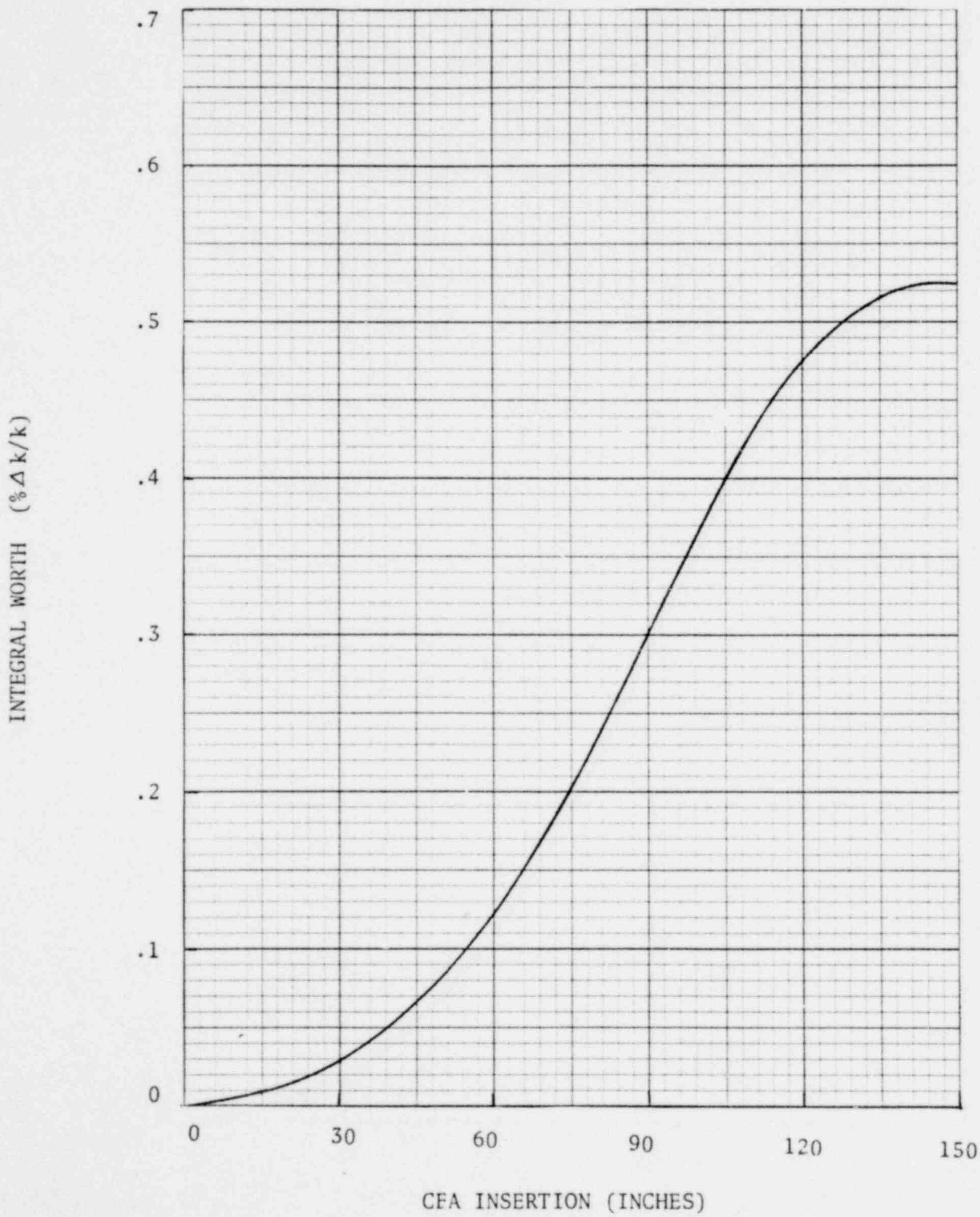


FIGURE 5.1.4.2

360

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP 4

POOR ORIGINAL

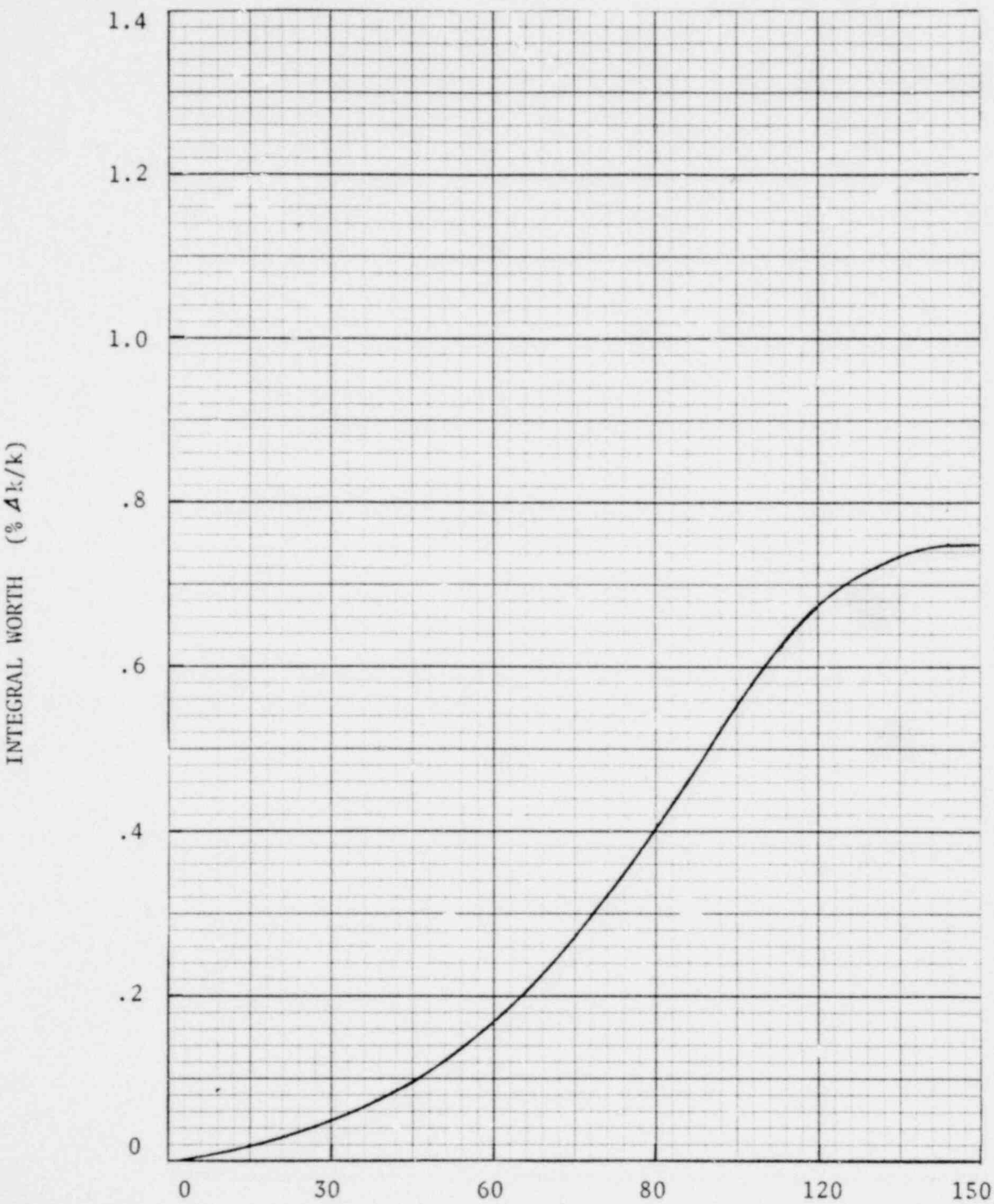


FIGURE 5.1.4.3

968 001

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP 3

POOR ORIGINAL

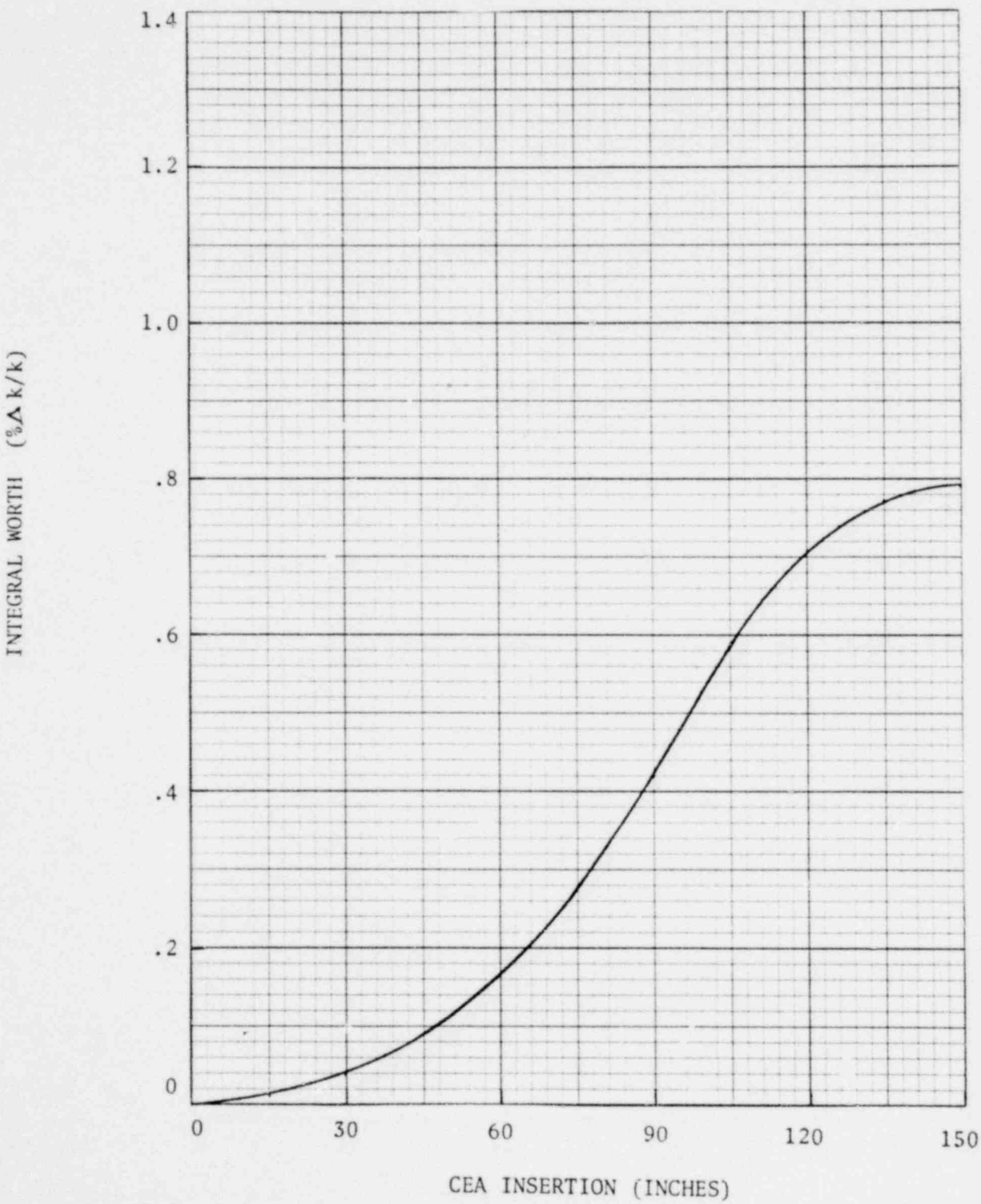


FIGURE 5.1.4.4

968 002

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP 2

POOR ORIGINAL

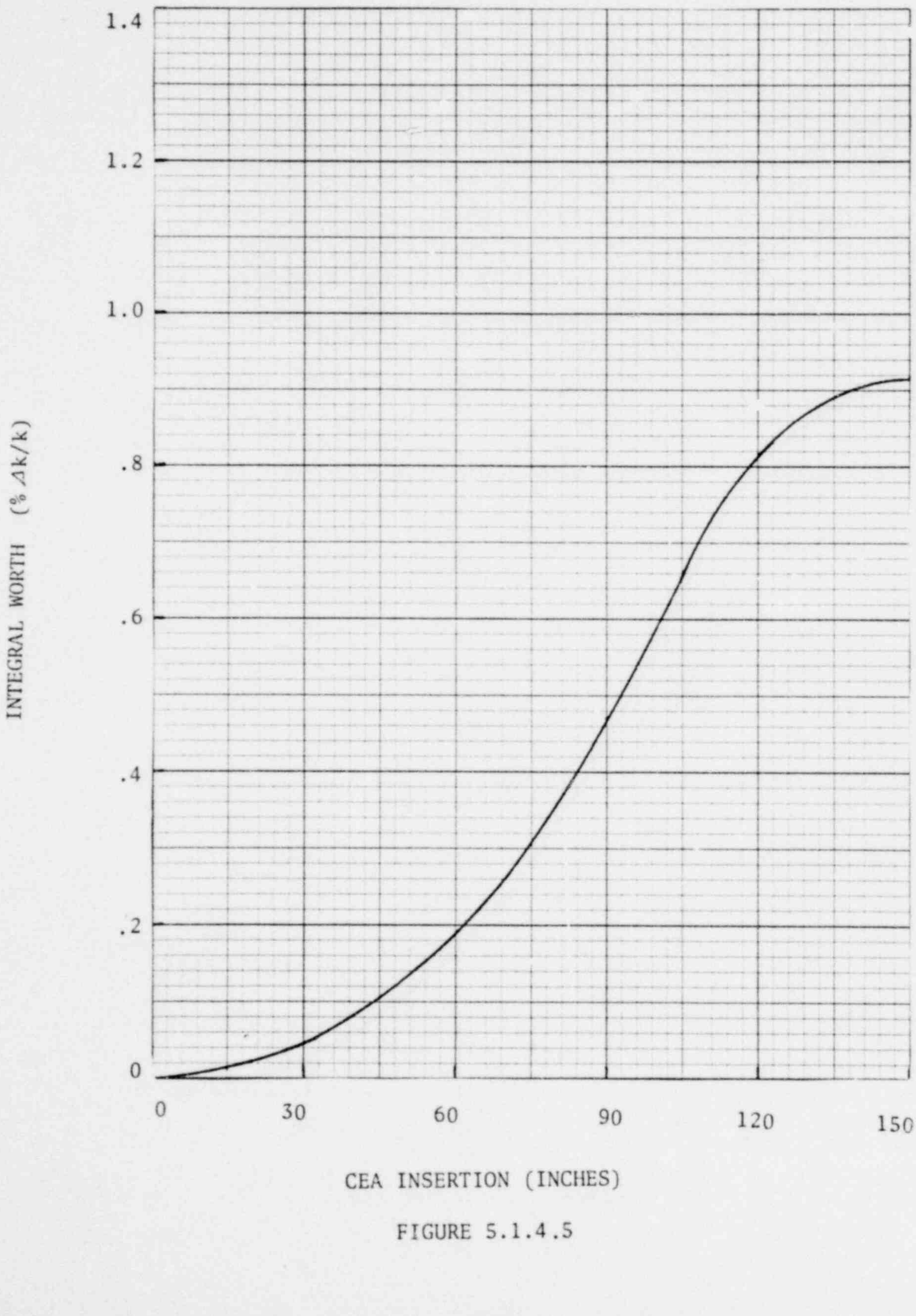


FIGURE 5.1.4.5

968 003

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP 1

POOR ORIGINAL

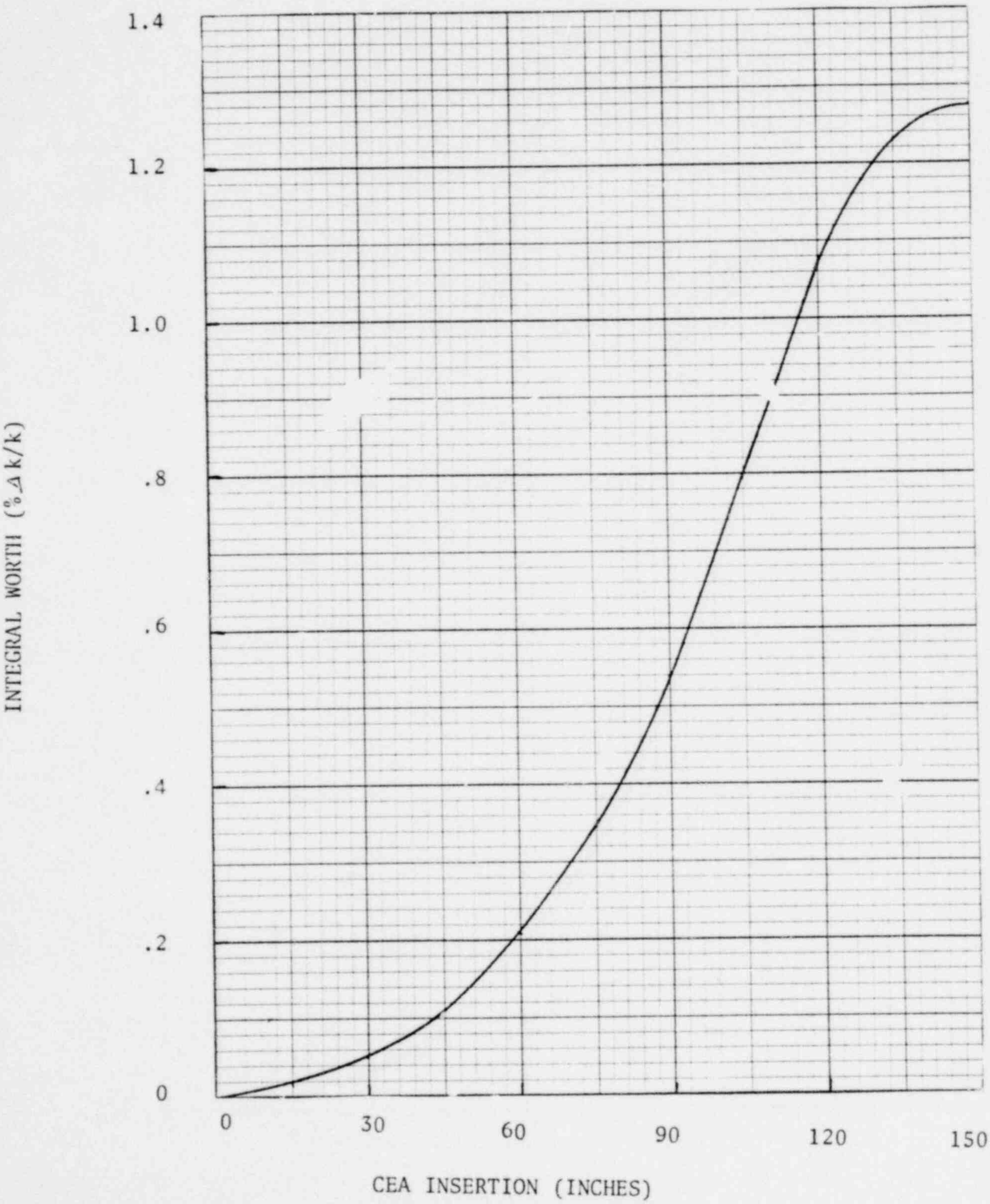


FIGURE 5.1.4-6

968 004

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP P
(Groups 6-1 Inserted)

POOR ORIGINAL

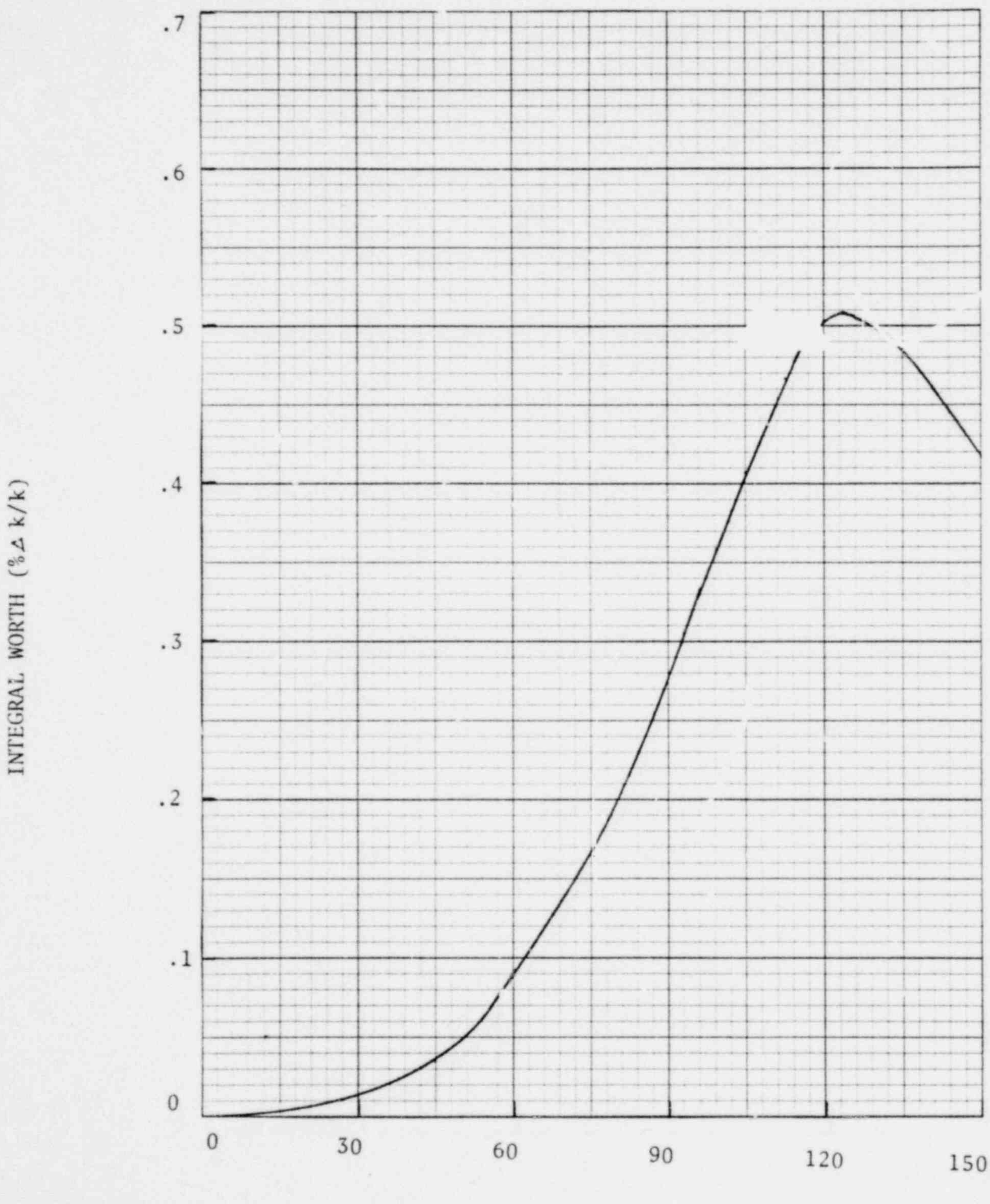


FIGURE 5.1.4.7

968 005

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP B

POOR ORIGINAL

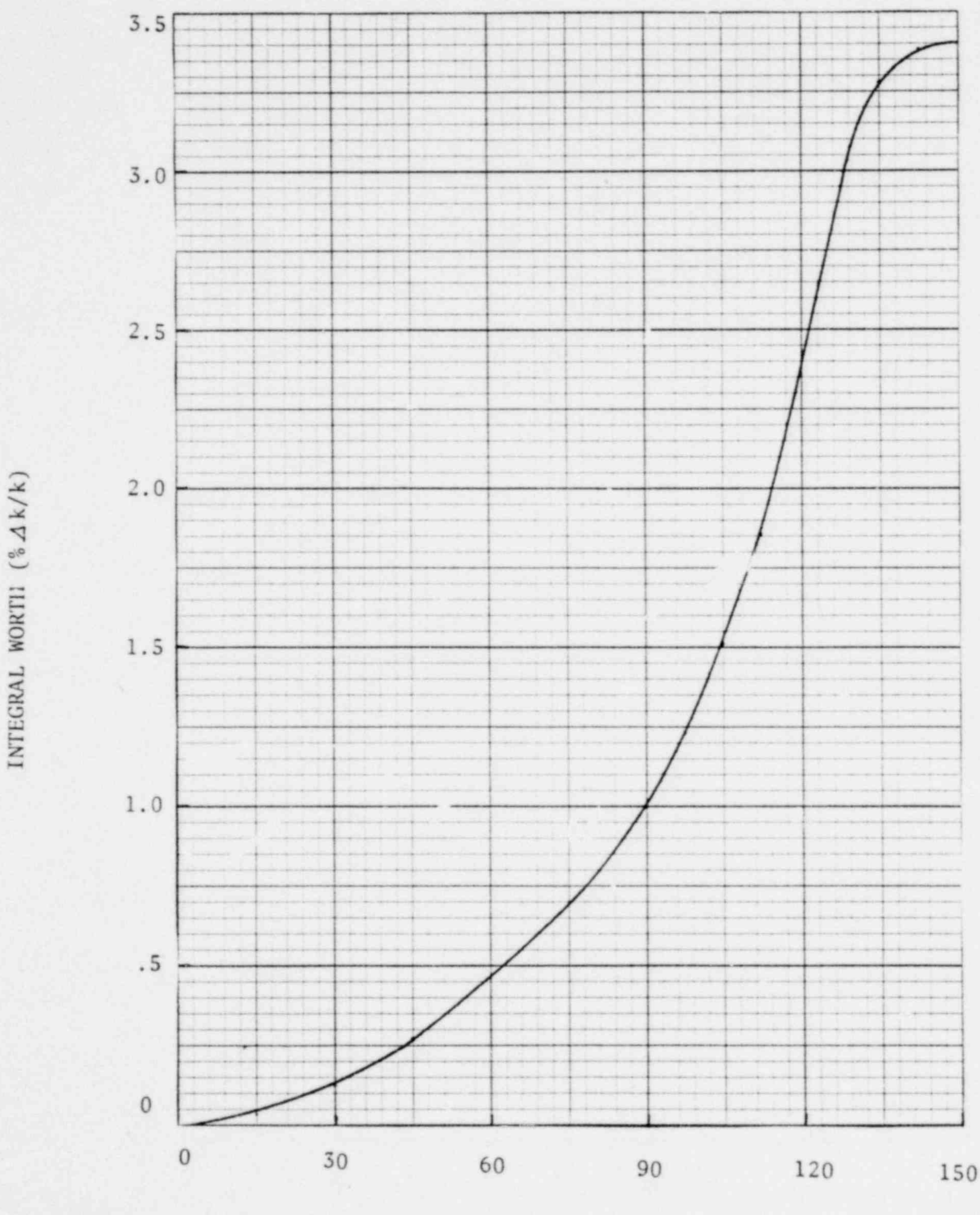


FIGURE 5.1.4.8

968 006

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP A

POOR ORIGINAL

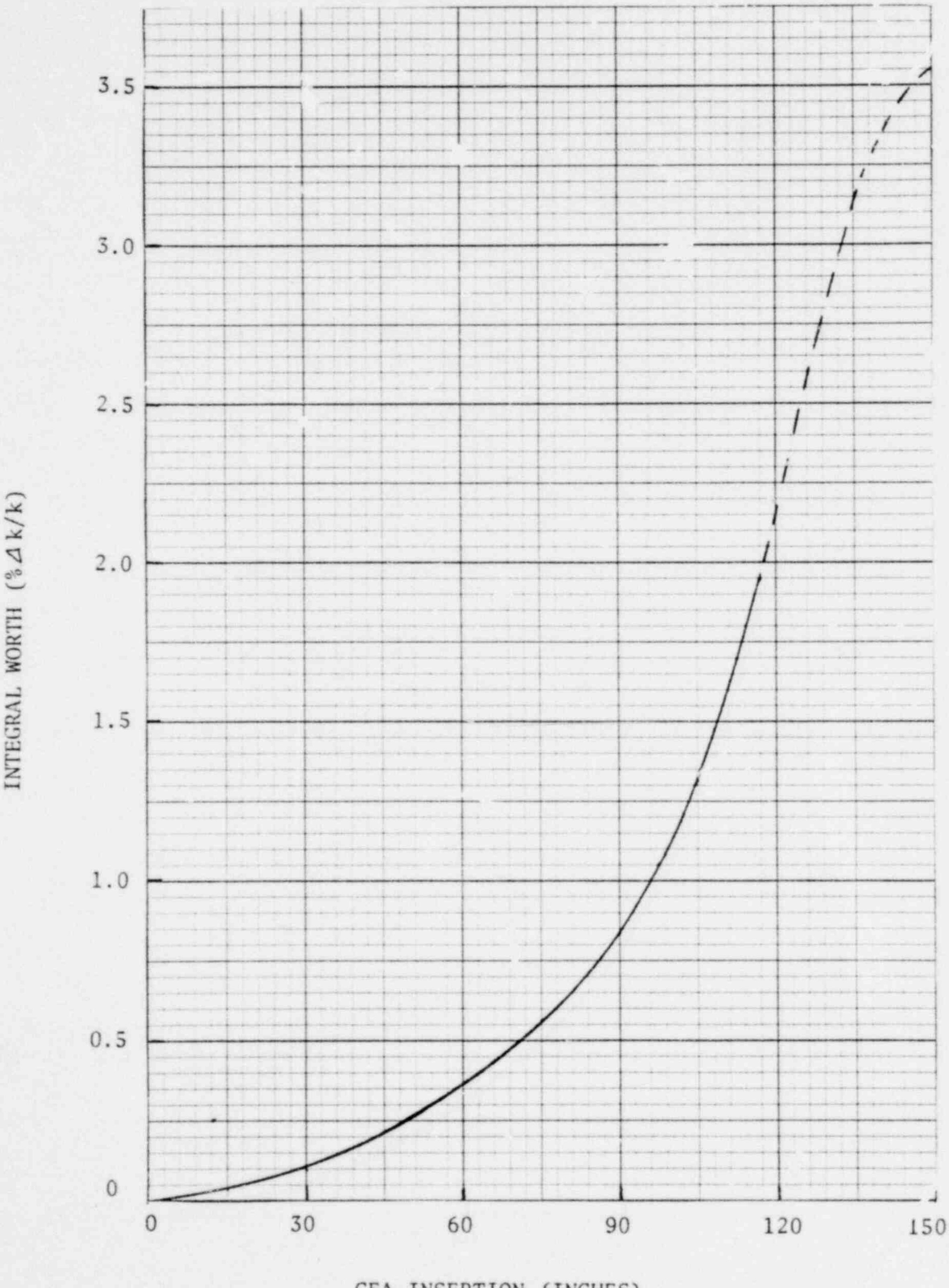


FIGURE 5.1.4.9

968 007

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

CEA GROUP P
(Group 6 @ 125")

POOR ORIGINAL

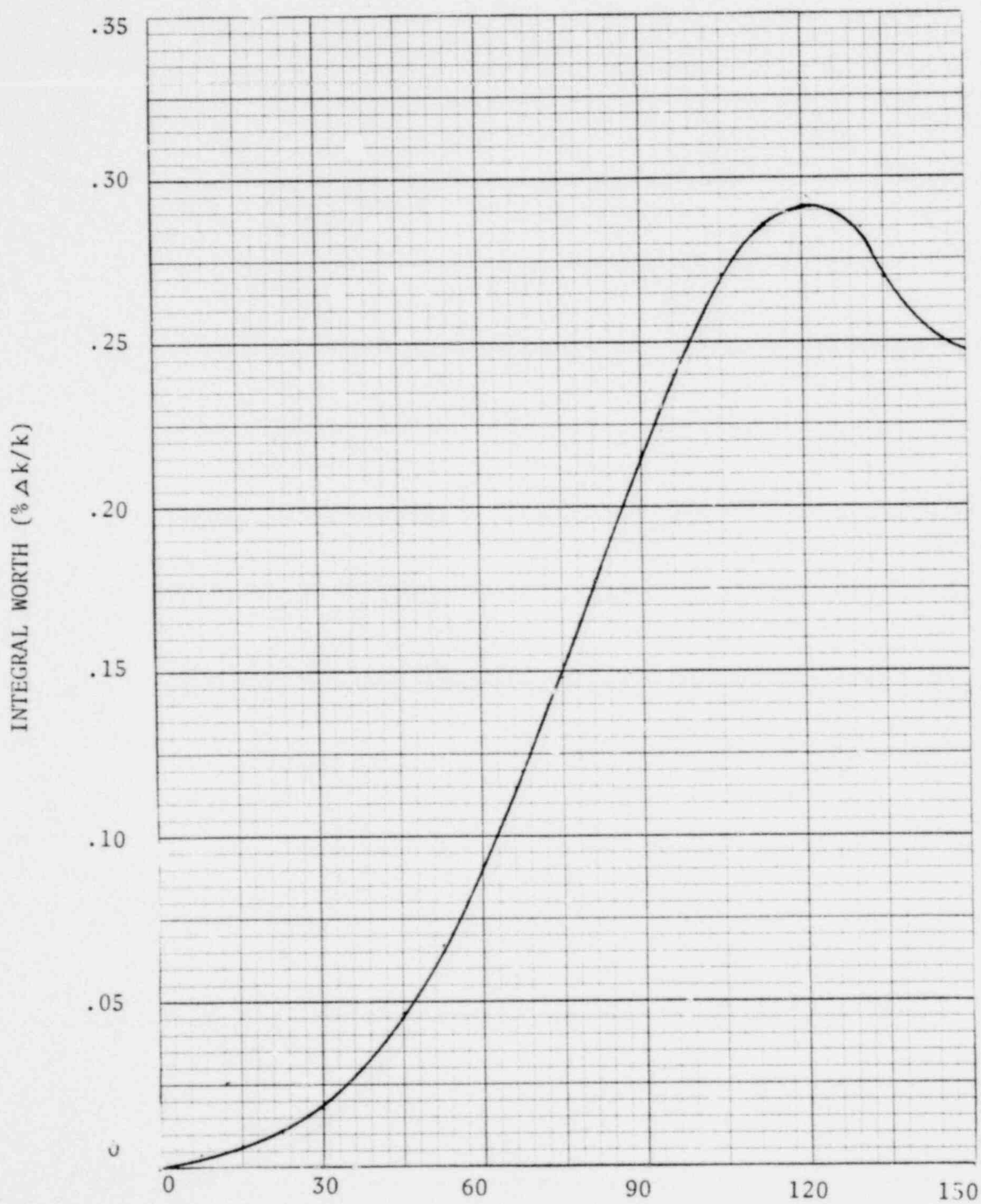


FIGURE 5.1.4.10

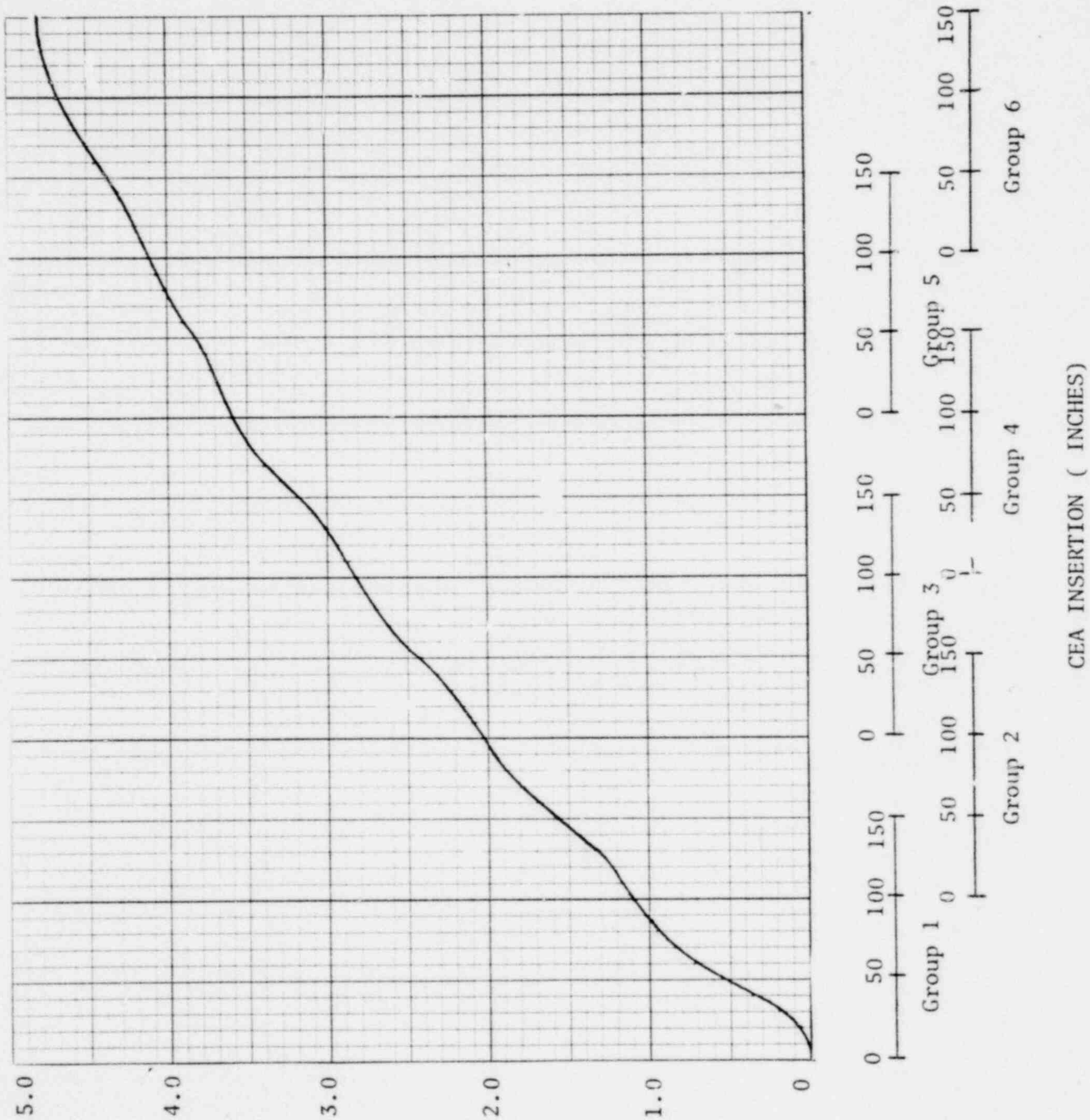
966 008

INTEGRAL CEA GROUP WORTH

BOL, FIRST CYCLE

Regulating Groups Overlap Worth

POOR ORIGINAL



INTEGRAL WORTH (% Δk/k)

FIGURE 5.1.4-11

968 009

5.1.5 DIFFERENTIAL BORON WORTH DETERMINATION

5.1.5.1 Purpose

The purpose in determining differential boron worth is to provide a means for predicting the resultant change in boron brought about by a change in reactivity, (or vice versa), for a varying set of core conditions.

5.1.5.2 Test Method

Differential boron worths were calculated by solving the following equation:

Differential Boron Worth (%Δk/k/100 ppm Boron) =

$$\frac{\text{Inserted CEA Worth (Initial)} - \text{Inserted CEA Worth (Final)}}{\text{CBC (Initial)} - \text{CBC (Final)}}$$

Where: CBC = Critical Boron Concentration corresponding to the particular (initial or final) core configuration.

The inverse of differential boron worth, IBW, which is often more convenient to use, was calculated as follows:

Inverse Boron Worth (IBW, ppm/%Δk/k) =

$$\frac{100}{\text{Differential Boron Worth}}$$

It should be noted that predictions of boron worths were provided in the IBW form as was the acceptance criteria.

The plateaus and core configurations for which boron worth calculations were carried out are summarized in Table 5.1.5.1.

5.1.5.3 Test Results

The results of boron worth calculations and corresponding data for initial and final cases are given in Table 5.1.5.1.

5.1.5.4 Conclusion

All inverse boron worths were compared to their appropriate predicted values and were found to be within the acceptance criteria of $\pm$ 15 ppm/%Δk/k of the predicted values.

968 010

TABLE 5.1.5.1: DIFFERENTIAL BORON WORTH DETERMINATION

	MEASUREMENT CONDITION	INSERTED REACTIVITY (%Δk/k)	BORON CONCENTRATION (ppm)	MEASURED BORON WORTH (%Δk/k/100 ppm)	*MEASURED IBW (ppm/%Δk/k)	PREDICTED IBW (ppm/%Δk/k)	ERROR
	260°F, 360 psia:						
	**						
1.	EARO, CBC (INITIAL)	0	999				
	CEA Gp 6,5,&4 at LEL with CEA			1.541	64.89	65.0	0.11
	Gp 3 at 146.3" WD (FINAL)	-1.4485	905				
2.	CEA Gp 6,5,&4 at LEL (INITIAL)	-1.4485	905	1.539	64.97	65.0	0.03
	CEA Gp 6 at ~ 115" WD (FINAL)	-0.0783	994				
	545°F, 2250 psia:						
	**						
3.	EARO, CBC (INITIAL)	0	1012				
	CEA Gp 6,5,4&3 at LEL with			1.378	72.56	76.5	3.94
	CEA Gp 2 at 97.5" WD (FINAL)	-2.7719	811				
4.	CEA Gp 6,5,4&3 at LEL with						
	CEA Gp 2 at 97.5" WD (INITIAL)	-2.7719	811				
	CEA Gp 6 through 1 at LEL with			1.38[illegible]	72.30	76.5	4.20
	CEA Gp B at 120.8" WD (FINAL)	-4.902	657				

*Measured IBW: Measured Inverse Boron Worth. This value must agree with the PREDICTED IBW within ±15 ppm/%Δk/k.

**EARO: Essentially All Rods Out. This refers to all CEAs at their UELs except for Group 6 which is ≥ 130 inches withdrawn.

5.1.6 CRITICAL BORON CONCENTRATION

5.1.6.1 Purpose

The purpose of this section was to determine the RCS boron concentration for a particular all rods out critical condition.

5.1.6.2 Test Method

A. At the 260°F plateau, boric acid was injected through the charging pumps with the reactor critical until all rods were fully withdrawn. Chemistry samples were then taken to determine the ARO critical boron concentration.

B. At the 545°F plateau, boric acid was injected through the charging pumps with the reactor critical until all rods were fully withdrawn except group six which stabilized between 130" and 140" withdrawn. Chemistry samples were then taken to determine critical boron concentration with the above control rod configuration. CEA group six was withdrawn to the fully withdrawn position (the reactivity computer was used to determine the residual worth of group six) and reactivity was allowed to stabilize. Group six was then returned to between 130" and 140" withdrawn to return power to its base level. The above manipulation of Group six was done two more times so that a more defined value for Group 6 residual worth could be obtained.

5.1.6.3 Test Results

A. At the 260°F plateau, the ARO critical boron concentration was measured and determined to be 999 ppm.

B. At the 545°F plateau, the ARO critical boron concentration was calculated to be 1012 ppm.

5.1.6.4 Conclusion

The acceptance criteria of within $\pm$ 100 ppm of predicted values was met at both temperature plateaus as summarized in Table 5.1.6.1.

968 012

TABLE 5.1.6.1

ARO CRITICAL BORON CONCENTRATION MEASUREMENT RESULTS

TEMPERATURE PLATEAU(°F)	TEST RESULTS (PPM)	ACCEPTANCE CRITERIA(PPM)	DPM DIFFERENCE
260	999	1006 ± 100	7 PPM
545	1012	1001 ± 100	11 PPM

968 013

5.1.7 PSEUDO DROPPED AND EJECTED CEA WORTH TESTS

5.1.7.1 Purpose

The purpose of these tests was to determine the reactivity worths of the worst case dropped and ejected CEA's. In addition, the worth of Part Length CEA subgroup P-1 was determined.

5.1.7.2 Test Method

"Dropped" and "Ejected" CEA worth measurements were performed at the 545°F, 2250 psia plateau. The initial CEA configurations for each case are given below.

To ensure the predicted worst case Regulating Group CEA was selected for each condition, the predicted second most reactive CEA was traded with the predicted most reactive CEA and their worths compared. Reactivity measurements were made with a Reactivity Computer.

A. Dropped CEA:

For the full-length CEA's, CEA 6-1 was predicted as the most reactive dropped CEA whereas, CEA 6-47 was predicted to be the second most reactive. For the Part Length CEA's, CEA P-24 was predicted to be the most reactive. Part Length CEA subgroup P-1 was predicted to be the most reactive Part Length subgroup. This was verified as follows.

With Regulating Group 6 between 130" and 150" withdrawn, Part Length CEA P-24 was diluted from its Upper CEA Limit (UCL) to its most negative worth insertion point. The CEA was then inserted to its Lower CEA Limit (LCL) to determine the residual (positive) worth of the rod. P-24 was subsequently borated to its UCL.

The worth of Part Length Subgroup P-1 was determined via the same method as CEA P-24 while using Regulating Group 6 to compensate for reactivity swings. Supgroup P-1 was also retrieved to its UCL after testing.

968 014

B. Ejected CEA:

In the case of the ejected CEA, CEA 4-11 was predicted to be the worst ejected CEA while CEA 6-1 was predicted as the second worst. This was verified as follows.

With the reactor stable and CEA Regulating Groups 3, 4, 5, and 6 at their LEL's and Group 2 at 97.5 inches withdrawn, CEA 6-1 was borated to its UEL to determine the ejected rod worth. CEA 6-1 was then traded with CEA 4-11 and Regulating Group 2 was adjusted such that CEA 6-1 was at its LEL and CEA 4-11 was at its UEL. CEA 4-11 was then realigned with group 4 using Group 2 to compensate for reactivity swings.

5.1.7.3 Test Results

Table 5.1.7.1 summarizes the "Dropped" and "Ejected" CEA worths as determined by this test. As can be seen in the Table, CEA 4-11 was determined to be more reactive than CEA 6-1 for the ejected CEA case, which was as predicted.

5.1.7.4 Conclusions

Each of the measured "Dropped" and "Ejected" CEA and subgroup CEA worths agrees with its predicted counterpart within the range of acceptance criteria as shown in Table 5.1.7.1.

968 015

TABLE 5.1.7.1

"DROPPED" AND "EJECTED" CEA WORTH MEASUREMENT RESULTS

CEA	MEASURED WORTH(%Δk/k)	PREDICTED WORTH(%Δk/k)	MEETS ACCEPTANCE CRITERIA ?(1)(2)(3)
DROPPED RESULTS:			
PLCEA P-24	-0.034	-0.027	Yes/Yes
Subgroup P-1	-0.133	-0.116	Yes/Yes
CEA 6-1	-0.127	-0.132	Yes/Yes
CEA 6-47	-0.085	-0.0717	Yes/Yes
EJECTED RESULTS:			
CEA 6-1	0.322	0.348	Yes/Yes
CEA 4-11	0.351	0.376	Yes/Yes

1. The acceptance criteria for the "Dropped" cases is: "The measured dropped CEA/PLCEA worth shall be within ± 25% of the predicted dropped worth of the CEA/PLCEA, or within 0.1% Δk/k of the predicted worth, whichever is larger."

2. The acceptance criteria for the "Ejected" cases is: "The measured ejected CEA worth shall be within ± 25% of the predicted ejected CEA worth, or within 0.1% Δk/k of predicted worth, whichever is larger.

3. The first entry here corresponds to the comparison for ± 25% of the predicted value. The second entry here corresponds to the comparison for 0.1% Δk/k of the predicted value.

968 016

5.1.8 STUCK CEA WORTH MEASUREMENT

5.1.8.1 Purpose

Technical Specifications require that available shutdown margin shall not be less than 5.0% Δk/k, with the highest worth CEA stuck out, whenever the reactor is critical. In order to verify the validity of the predictions, the shutdown worth with the predicted most reactive CEA withdrawn was measured.

5.1.8.2 Test Method

CEA A-52 was predicted to be the worst case stuck CEA.

The test was performed at 545°F and 2250 psia, with CEA groups 6 through B (including the part length CEAs) tripped; Group A was initially at ∿ 136" withdrawn. CEA A-52 was withdrawn to its UEL and secured. The remainder of Group A was inserted (diluted) to ∿ 10 inches withdrawn ± 5 inches. Conditions were allowed to stabilize and then the remainder of Group A was inserted to its LEL to determine its residual worth. The residual worth determination was repeated two more times so that a more accurate value could be obtained.

5.1.8.3 Test Results

The worth of all CEAs (full and part length) inserted except for the most reactive CEA, which was stuck out, was determined to be 10.67% Δk/k versus 10.74% Δk/k as predicted. This in turn, corresponds to a stuck (CEA A-52) worth of 1.52% Δk/k versus a predicted worth of 1.58% Δk/k. The worth of Group A minus the stuck CEA was determined to be 2.03% Δk/k versus 1.99% Δk/k as predicted. Stuck CEA results and evaluations are summarized in Table 5.1.8.1.

5.1.8.4 Conclusions

Acceptance criteria related to stuck CEA cases were applied to the following determinations:

1) Worth of Group A minus the most reactive stuck CEA (CEA A-52).

2) Worth of all CEAs inserted (including part length CEAs) minus the most reactive stuck CEA (CEA A-52).

These acceptance criteria were met satisfactorily (see Table 5.1.8.1).

TABLE 5.1.8.1: STUCK CEA RESULTS

CASE	MEASURED WORTH(%Δk/k)	PREDICTED WORTH(%Δk/k)	ERROR	MEETS ACCEPTANCE CRITERIA
Stuck CEA A-52	*1.52	1.58	3.8%	N/A
Group A Minus Stuck CEA A-52	2.032	1.99	2.07% /.042 %Δk/k	YES(1)
All CEAs Inserted (with PLRS) Minus Stuck CEA A-52	10.666	10.736	.65%	YES(2)

* Based on a Group A measured worth of 3.55% Δk/k.

(1) The acceptance criteria for this case is: "the measured worth for this case shall be within ± 15% of the predicted worth or within 0.1% Δk/k of the predicted worth, whichever is larger.

(2) The acceptance criteria for this case is: "the measured worth for this case shall be within ± 10% of the predicted worth".

968 018

6.0 POWER ASCENSION TESTS

The Power Ascension tests were conducted to determine the as built plant characteristics during steady state and transient operation from Hot Zero power to 100% power and to demonstrate that the plant is capable of withstanding the accidents and transients analyzed in the FSAR. The majority of the tests requiring steady state conditions occurs at the major plateaus of 20%, 50%, 80% and 100% power. Minor testing was also accomplished at other power plateaus.

Plant performance data was recorded during testing at the major plateaus (see Table 6.0.1). All test data for a given plateau was evaluated prior to increasing power to the next major plateau.

The Power Ascension test report is divided into four sections, each one corresponding to a major power plateau. Each section contains details of the tests performed during power ascension to a major plateau and all tests performed at the plateau.

968 019

POWER ASCENSION TESTS

Power Plateau vs Test

TABLE 6.0.1

TEST	0%	5%	10%	15%	20%	30%	40%	50%	60%	70%	80%	20%	50%	65%	80%	90%	95%	100%
NSS Plant Data Record		X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X
Transient Data		X	X	X	X	X	X	X	X	X	X					X	X	X
RCS ΔT Power Determination		X	X	X	X													
Nuclear and Thermal Power Calibration				X	X			X			X		X		X		X	X
SDBCS Capacity Checks								X										
NSSS Calorimetric Calculation				X	X			X			X							X
RCS Calorimetric Flow Measurement					X			X			X							X
Linear Power Subchannel Calibration					X			X										
Process Variable Intercomparison					X			X			X							X
Chemistry and Radiochemistry Tests					X			X			X							X
Core Performance Record					X			X			X							X
CPC/COLSS Verification	X				X			X			X							X
Variable T_{AVG} Tests					X			X			X							X
CEA Shadowing Factor Verification					X													
Unit Load Transient Test		X			X			X			X	X						X
Shape Annealing Matrix Measurement					X			X										
Temperature Decalibration Verification								X										

968 020

POWER ASCENSION TESTS

Power Plateau vs Test

TABLE 6.0.1 (cont'd)

TEST	0%	5%	10%	15%	20%	30%	40%	50%	60%	70%	80%	20%	50%	65%	80%	90%	95%	100%
Radial Peaking Factor Verification								X										
20% Rx Trip with S/D outside Control Room					X													
80% Total LOF/Natural Circ. Verification											X							
Loss of Offsite Power												X						
Ejected CEA													X					
Dropped CEA													X					
Load Rejection from 100%																		X
Fast Trip Recovery/XE .ollow								X										
PLCEA Xenon Control														X				
Incore Detector Signal Verification					X			X			X							X
Movable Incore Detector Checks					X			X			X							X
FWCS Post Trip Setting		X																
100% Turbine Trip																		X
Field Adjustment of Incore Dynamic Comp.								X										
50% Turbine Trip								X										

POWER ASCENSION TESTS

Power Plateau vs Test

TABLE 6.0.1 (cont'd)

TEST	0%	5%	10%	15%	20%	30%	40%	50%	60%	70%	80%	20%	50%	65%	80%	90%	95%	100%
Turbine/Generator Loading		X	X	X	X	X	X	X	X	X	X					X	X	X
Main and Reheat Steam Test		X*			X*			X	X	X	X					X		X
Condensate and Feedwater System Test					X			X			X							X
Main Turbine Electro Hydraulic Control					X			X			X							X
FW Vents, Drains and Water Induction					X			X			X							X
Vibration and Loose Parts Monitoring	X				X			X			X							X
HVAC Performance	X				X			X			X							X
Pipe/Comp Hot Deflection																		X
Piping Dynamic Transient					X			X			X							X
Steady State Vibration					X	X		X										X
Biological Shield Survey	X	X			X			X										X

* Performed during Power Ascension following 20% Rx Trip

6.1 0% thru 20% POWER PLATEAU

INTRODUCTION

Power Ascension testing commenced on December 16, 1978 with the plant at Hot Zero power and continued thru the 20% power plateau. A total of 27 individual tests were performed at the 20% power plateau. Testing was also accomplished at the 0%, 5%, 10% and 15% power plateaus.

All testing requiring power levels between Hot Zero power and 20% power has been performed and those major objectives of the Power Ascension Test program have been satisfied. Sections 6.1.1 thru 6.1.27 provide a detailed description of the 0% thru 20% power tests.

968 023

6.1.1 RCS ΔT POWER DETERMINATION TESTS

6.1.1.1 Purpose

The RCS ΔT Power Determination was performed in order to measure the reactor thermal output at low power plateaus (5%, 10%, 15% and 20%).

6.1.1.2 Test Method

A primary system calorimetric was completed after steady state conditions were achieved at each plateau. Temperature, pressure, and flow data were obtained from the CPC's. The core thermal output was then determined by multiplying the primary mass flow rate by the change in enthalpy across the reactor.

6.1.1.3 Test Results

RCS ΔT Power data was collected and a primary system calorimetric was calculated. The calculated thermal output is compared with the average values of calibrated neutron flux power, static thermal power, and core average power in Table 6.1.1.1.

6.1.1.4 Conclusion

The RCS ΔT Power Determination was completed successfully at the 5%, 10%, 15%, and 20% power plateaus.

968 024

TABLE 6.1.1.1

POWER PLATEAU		CALCULATED RCS ΔT POWER	AVG. NEUTRON FLUX POWER	STATIC THERMAL POWER	CORE AVG. POWER
5%	Run 1	5.025	* 1.55	5.3	5.66
	Run 2	4.76	5.31	5.06	6.07
10%		9.93%	* 3	9.91	10.52
15%		17.32%	14.51	14.99	15.9
20%		19.54%	19.58%	19.35%	19.56%

* Prior to adjustments on excore linear calibrate potentiometer.

968 025

6.1.2 NUCLEAR AND THERMAL POWER CALIBRATION TESTS

6.1.2.1 Purpose

To adjust the Excore Linear Power Calibrate potentiometers and the CPC addressable constants (KCAL and TPC) relating to the core power level to agree with the RCS ΔT power.

6.1.2.2 Test Method

The Nuclear and Thermal Power Calibration Test was performed at the 5%, 15%, and 20% power plateaus as part of the power ascension test sequence. For each safety channel, the input to the High Linear Power Bistable and the CPC values, PHICAL (calibrated neutron flux power) and BDT (static thermal power) were recorded and compared to the RCS ΔT power. Adjustment of the Excore Linear Power Calibrate potentiometers, or the addressable CPC constants KCAL or TPC was necessary if the High Linear Power, PHICAL or BDT readings varied from the RCS ΔT power by more than ± 0.2% of Rated Thermal Power.

For each safety channel (one at a time) the following adjustments were performed as necessary.

A. The Excore Linear Power Calibrate potentiometer was adjusted so that the input to the High Linear Power Bistable, as monitored by an external DVM at the PPS cabinet, equaled the following value:

$$\text{DVM Reading} = \%\ \frac{\text{Power x 5 Volts}}{100} \pm 0.005\text{V}$$

B. The CPC addressable constants KCAL and TPC were adjusted as follows:

$$\text{KCAL (NEW)} = \%\ \frac{\text{Power x KCAL (OLD)}}{\text{PHICAL (OLD)}}$$

$$\text{TPC (NEW)} = \%\ \frac{\text{Power x TPC (OLD)}}{\text{BDT (OLD)}}$$

968 026

After the initial adjustments were performed, readings from all 4 channels for High Linear Power, PHICAL, and BDT were taken and compared to the RCS ΔT power. If any of the readings varied from the RCS ΔT power by more than $\pm$ 0.2% of Rated Thermal Power, the adjustments were repeated until the $\pm$ 0.2% criteria could be met.

6.1.2.3 Test Results

Initial attempts at 5 and 10% power to calibrate the Excore Linear Power Signals because there was not enough adjustment range in the calibration potentiometers. The linear power subchannel gains were adjusted by changing taps on fixed resistors in the amplifier circuit and all four channels were adjusted using the previously described method and the acceptance criteria of agreement within $\pm$ 0.2% of Rated Thermal Power was met.

Appendix E was successfully completed at the 15% and 20% power plateaus without incident.

6.1.2.4 Conclusion

At the 5%, 15% and 20% power plateaus, the Excore Linear Power Calibrate potentiometers and the CPC addressable constants KCAL and TPC were adjusted using information obtained from the RCS ΔT power determination. The High Linear Power, PHICAL, and BDT readings for all safety channels agreed with the RCS ΔT power to within $\pm$ 0.2% of Rated Thermal Power.

6.1.3 NSSS CALORIMETRIC TESTS

6.1.3.1 Purpose

The purpose of this test was to:

A. Determine core thermal power by means of a secondary plant heat balance:

B. Verify the COLSS core thermal power calculations;

C. Verify that OP 2103.16 (Heat Balance Calculation) will provide a satisfactory indication of core power.

6.1.3.2 Test Method

Plant parameters were maintained essentially constant while steam generator data and reactor power information was collected over a 3 hour period. This data along with the energy input and loss terms measured during the RCS Heat Loss Test was used to calculate the net heat output.

The calculated core thermal power was compared to the COLSS secondary calorimetric power (BSCAL) to verify the accuracy of the algorithm and to the COLSS primary calorimetric power (BDELT). Adjustments were made as necessary to the ΔT Power Gain factor (in the BDELT algorithm) to provide agreement between BDELT and BSCAL. OP 2103.16 (Heat Balance Calculation) was completed concurrently and compared to the calculated core thermal power to verify its accuracy.

6.1.3.3 Test Results

Due to various difficulties which were encountered, this test was performed a total of three times. During the first test run, one of the feedwater flow transmitters was found to be out of calibration, causing minor errors in the secondary calorimetric results. Results of the second test run were satisfactory; however, OP 2103.16 was not performed. All results of the third test run were satisfactory. Table 6.3.1 reflects the results of each of the test runs.

6.1.3.4 Conclusions

The plant computer secondary calorimetric was found to be within the acceptable limits. Also OP 2103.16 (Heat Balance Calculation) was found to provide acceptable results.

968 028

968 029

TABLE 6.1.3.1

RESULTS OF NSSS CALORIMETRIC

	DATE PERFORMED	CALCULATED CORE THERMAL POWER	BSCAL (BEFORE ADJUSTMENT)	BDELT (BEFORE ADJUSTMENT)	RESULTS OF OP 2103.16	CALCULATED VALUE FOR ΔT POWER GAIN	BSCAL (AFTER ADJUSTMENT)	BDELT (AFTER ADJUSTMENT)
RUN #1	12/30/78	16.86%	16.91%	18.57%	(1)	(2)	(2)	
RUN #2	12/31/78	20.76%	20.69%	18.33%	(1)	1.132	19.32%	19.50%(3)
RUN #3	1/14/79	18.72%	18.99%	17.40%	18.53%	1.0916	19.05%	19.03%

NOTES:

(1) OP 2103.16 not performed. Deficiency written.

(2) Test aborted prior to this step due to miscalibrated flow transmitter.

(3) Actual power was adjusted slightly prior to this step.

6.1.4 RCS CALORIMETRIC FLOW MEASUREMENT TESTS

6.1.4.1 Purpose

The purpose of this test was to determine the reactor coolant flow rate based upon the computer secondary plant calorimetric and the measured primary pressure and temperatures (T_c and T_h) and to provide guidance for adjustment of the CPC and COLSS flow algorithm constants if necessary.

While this method yields more accurate results at higher power levels, it was performed at lower power levels as well to provide additional information. No adjustments are made below 80% of rated power.

6.1.4.2 Test Method

Calculation of the reactor coolant mass flow rate was based upon secondary plant calorimetric power and primary pressure and temperatures. Over a specified period, during which time plant conditions were maintained essentially constant, RCS data was recorded from both the CPC's and the plant computer. Following this collection period, the average enthalpy rise of the reactor coolant was determined and used with secondary calorimetric power to calculate the mass flow rate of the reactor coolant.

The calculated coolant mass flow rate was compared to CPC's and COLSS values for RCS flow. New values were calculated for the constants in the CPC and COLSS algorithms to provide the desired agreement.

6.1.4.3 Test Results

Average core thermal power during this test was 19.80% (COLSS secondary plant calorimetric power). The average enthalpy rise of the reactor coolant across the core as determined from CPC data was 13.330 Btu/lbm. Hence, the reactor coolant mass flow rate was calculated to be 1.4266×10^8 lbm/hr. This translates to 118.49% of the base mass flow rate (120.4×10^6 lbm/hr). By comparison, all four CPC channels indicated approximately 111% of base flow and COLSS indicated 111.68% of base flow. More detailed results are shown in Table 6.1.4.1.

968 030

New values were calculated for the COLSS flow bias constants and for the CPC flow constants and for the CPC thermal power (BDT) scaling constants (TPC). These values are shown in Table 6.1.4.2 and are the values required to make the CPC and COLSS flow rates agree with the measured coolant flow rate and to offset the change to CPC ΔT power caused by changing CPC calculated flow. Since this test was performed for information only at this power level, none of the new constants were entered.

6.1.4.4 Conclusions

The calculated RCS flow was slightly higher than expected but was within acceptable limits.

TABLE 6.1.4.1

REACTOR COOLANT MASS FLOW VALUES

INDICATION	VALUE (1)
Calculated (2)	118.49%
CPC A	111.07%
CPC B	111.16%
CPC C	111.04%
CPC D	111.06%
COLSS	111.68%

(1) All values are given as percent of base flow (120.4×10^6 lbm/hr.).

(2) As calculated using COLSS secondary calorimetric power and coolant enthalpy rise across the core.

TABLE 6.1.4.2

COLSS AND CPC FLOW ADJUSTMENT FACTORS

	CPC VALUES			COLSS VALUES			
	Flow Constants		Thermal Power Constants				
	FC-1	FC-2	TPC	D15(1)	D15(2)	D15(3)	D15(4)
Previous Values:							
CPC A	1.1054	-.022094	.99929				
CPC B	1.1049	-.022094	.95633				
CPC C	1.1044	-.022094	1.1503				
CPC D	1.1044	-.022094	1.1106				
COLSS				0	0	0	0
Calculated* Values:							
CPC A	1.1792	-.023569	.92839				
CPC B	1.1778	-.023552	.89118				
CPC C	1.178	-.023576	1.0670				
CPC D	1.1783	-.023573	1.0373				
COLSS				-5434.85	-5484.85	-5484.85	-5484.85

* Values calculated at 20% power for information only (not input into CPCs).

968 033

6.1.5 LINEAR POWER SUBCHANNEL CALIBRATION TESTS

6.1.5.1 Purpose

The purppse of this test was to adjust the Linear Power Subchannel gains, the Excore Linear Power Potentiometers, the 200% Linear Calibrate potentiometer, and the CPC addressable constants (KCAL and TPC) relating to the core power level.

6.1.5.2 Test Method

The reactor was stabilized at 20% power. An NSSS Calorimetric Measurement was performed. Following the completion of the NSSS calorimetric, baseline power data was obtained from all four Core Protection Calculator channels and the Plant Protection System (PPS) channels. The selected PPS channel (A, B, C or D) High Linear Power and Low DNBR trips were bypassed. The Excore Linear Subchannel amplifier for each of the three (3) detectors for the selected channel was adjusted to the calorimetric power. At the completion of the Linear Subchannel amplifier adjustment, the 200% Linear Calibrate potentiometer was adadjusted and a neutron power signal was simulated to each of the three Linear Subchannel amplifiers to verify proper amplifier operation. The Excore Linear Power calibration potentiometer was adjusted such that the indicated Linear Power was in agreement with the computer secondary calorimetric power. Upon completion of adjustment to the Excore Linear channel, KCAL and TPC were adjusted as necessary, thereby adjusting CPC Calibrated Nuclear Power (PHICAL) and CPC Delta Temperature Power (BDT) respectively. The above process was performed on the remaining three (3) protection channels and as left power data was obtained from all four CPC and PPS channels.

6.1.5.3 Test Results

All Linear Power Subchannel amplifiers were adjusted to the NSSS calorimetric value. The 200% Linear Calibrate potentiometer and the Excore Linear Power potentiometers were successfully adjusted for all four channels. KCAL and TPC adjustments were performed as described in the body of the test.

6.1.5.4 Conclusions

All necessary adjustments were made to the Linear Subchannel gains, the Excore Linear Power Potentiometers, the 200% Linear Calibrate Potentiometers, and the CPC addressable constants relating to core power level (KCAL and TPC).

6.1.6 PROCESS VARIABLE INTERCOMPARISON TESTS

6.1.6.1 Purpose

The purpose of this test was to compare Process Instrumentation readings obtained from the Plant Computer, Plant Protection System, Core Protection Calculators, and various console meters to verify proper agreement between systems.

6.1.6.2 Test Method

This test was performed at the 20% power plateau. After establishing steady state RCS conditions (not necessarily equilibrium Xenon), data was recorded for the following variables:

1. RCS cold leg temperature,
2. RCS hot leg temperature,
3. RCP differential pressure,
4. RCP speeds,
5. RCS pressure,
6. Pressurizer level,
7. Steam Generator levels, and
8. Steam Generator pressures.

Common process variable readings for each system were then intercompared against preset criteria to assure the accuracy of process loop calibrations and system signal processing.

6.1.6.3 Test Results

All process variable intercomparisons were within specified tolerance at the 20% power plateau with the exception of one RCS cold leg temperature indicator, three pressurizer pressure indicators, one pressurizer level indicator, and five steam generator level indicators.

Following the 20% power plateau trip and subsequent return to power, a hold was included in the power ascension at 20% power and retesting was completed which resolved all outstanding deficiencies.

6.1.6.4 Conclusion

All process variable intercomparisons were within the specified tolerances at the 20% power plateau.

6.1.7 CHEMISTRY AND RADIOCHEMISTRY TESTS

6.1.7.1 Purpose

The purpose of this test was to conduct chemistry tests with the intent of establishing baseline corrosion data and activity buildup with power level. As a result of this, procedures for sample collection analysis were verified. Also, this test was used to verify the calibration of the process radiation monitor.

6.1.7.2 Test Method

A. Primary System

Sample and analysis procedures were performed using the CE Chemistry Manual (CENPD-28) as a guide. Three sets of RCS Chemistry Analyses were performed at the 20% plateau. The RCS Chemistry Analyses included the following tests:

a. pH

b. Conductivity

c. Cl

d. F^-

e. Dissolved Oxygen

f. Suspended Solids

g. Boron

h. Lithium

i. Dissolved Hydrogen

j. Gamma Spec. Analysis (gas)

k. Degased Gross Beta

l. Crud Activity

m. Tritium

n. Iodine Ratio

o. Iodine Dose Equivalent

p. Gamma Spec. Analysis (liquid)

q. Total Gas (primary coolant)

B. Secondary System

Sampling and analysis procedures were performed using CENPD-28 as a guide. Five sets of secondary chemistry analyses were performed at the 20% plateau. Each set of chemistry analyses included the following tests:

a. pH

b. Conductivity

c. Cation Conductivity

d. Dissolved Oxygen

e. Hydrazine

f. Ammonia

g. Silica

h. Sodium

i. Iron

j. Copper

C. Process Radiation Monitor

A sample was to be taken downstream of the Process Radiation Monitor. Laboratory results of the Gross Gamma Coolant Analysis were to be compared to the Process Radiation Monitor analysis for verification of proper Process Radiation Monitor function. Agreement within ± 20% was necessary to verify proper calibration of the Process Radiation Monitor.

6.1.7.3 Test Results

The required radiochemistry and secondary samples were obtained and analyzed. The required process radiation monitor readings were not obtained due to inoperability of this system. Baseline activities for the 20% plateau were established.

6.1.7.4 Conclusion

It was proven that primary and secondary sampling and analysis can be performed in accordance with Technical Specifications and CENPD-28. Baseline activities for the RCS were recorded. The Process Radiation Monitor calibration was not verified at the 20% power plateau.

6.1.8 CORE PERFORMANCE RECORD TESTS

6.1.8.1 Purpose

The purpose of this test was to record core performance data from incore detectors, and to specify the acceptance criteria for comparison of the measured results with predicted core operating parameters.

6.1.8.2 Test Method

A. While the reactor was being maintained at 20% steady state power, with equilibrium Xenon, incore detector data was collected for analysis.

B. The measured results were then compared to predicted values in the following manner:

   a. The comparison of the measured power distribution with the predicted power distribution is a root mean squared statistical comparison of the relative power density distribution for each of the 177 fuel assemblies.

   b. The comparison of the measured axial power distribution with the predicted axial power distribution is a root mean squared statistical comparison of the relative axial power distribution for each of the 100 axial nodes.

   c. The measured values of total planar radial peaking factor (Fxy), total integrated radial peaking factor (Fr), core average axial peak (Fz), and core 3-D power peak (Fq) were compared to predicted values.

6.1.8.3 Test Results

A. Results of the statistical comparisons and peaking factors are summarized in Tables 6.1.8.1 and 6.1.8.2.

968 038

TABLE 6.1.8.1

	Measured Results (RMS)	Acceptance Criteria (RMS)
Power Density Distribution	1.903	≤ 5
Axial Power Distribution	2.120	≤ 5

TABLE 6.1.8.2

	Measured	Predicted	% Difference	Acceptance Criteria
Fxy	1.4145	1.3607	3.95	≤ 10%
Fr	1.3786	1.3607	1.31	≤ 10%
Fz	1.30111	1.30	0.08	≤ 10%
Fq	1.79745	1.769	1.61	≤ 10%

6.1.8.4 Conclusions

All acceptance criteria have been met for the comparisons between predicted values and measured results. As shown by the above results, the computer model predictions were adequate for determining core operating parameters.

968 039

6.1.9 CPC/COLSS VERIFICATION TESTS

6.1.9.1 Purpose

The CPC/COLSS Verification Tests were performed to accomplish the following objectives:

A. Verify that the CPC/COLSS DNBR and LPD calculations are correct.

B. Evaluate the effect of process input noise on the CPC/COLSS system.

6.1.9.2 Test Method

A. At hot zero power, the process input noise on one CPC channel was recorded on FM tape, as a zero power data base. The CPC addressable constants and slowly varying input parameters were input into the CEDIPS* computer code. The CPC output parameters were compared to the CEDIPS* output in order to evaluate the effect of noise.

Detailed verification of CPC/COLSS DNBR and LPD calculations were not performed at this plateau.

B. At 20% power with ARO and Xenon equilibrium, the process input noise was measured. Plant computer reports containing information on the CEAC, CPC's, and COLSS were obtained for use in the verification of the CPC/COLSS DNBR and LPD calculations. The CPC/COLSS data was compared to the results of the CEDIPS* computer code and the incore detector analysis results.

6.1.9.3 Test Results

A. The process noise data from hot zero power was recorded and sent to CE-Windsor.

B. The process noise data from the 20% plateau was recorded. The data required for verification of CPC/COLSS DNBR and LPD calculations was collected and compared to the results of the CEDIPS* computer code. All data was transmitted to CE-Windsor for review.

6.1.9.4 Conclusions

The CPC output parameters were compared to the CEDIPS* code and were found to be acceptable. The COLSS DNBR and LPD related calculations were reviewed by CE-Windsor and found to be adequate.

968 040

*CEDIPS is a FORTRAN program for statistical analysis of effects of process inputs upon the CPC system.

6.1.10 VARIABLE T_{AVG} TESTS

6.1.10.1 Purpose

The objective of this test was to determine the Isothermal Temperature Coefficient (ITC) and Power Coefficient.

6.1 10.2 Test Method

Two methods were used to determine the Isothermal Temperature and Power Coefficients; one method was performed with no CEA movement and the other was performed with center CEA movement. These two approaches are described in more detail below:

A. No CEA Movement

With the reactor at steady state and equilibrium or near equilibrium xenon and CEA group 6 at 120 inches withdrawn, a small step change in the turbine control valve position is made and then adjusted to establish a new coolant inlet temperature. This change produces a small turbine load-reactor power mismatch. The temperature change results in a reactivity feedback and a resultant power change. The power change produces an opposite reactivity feedback and the reactor settles out at a new power and temperature condition. The cycle is then reversed by making a small step change in the turbine control valve position in the opposite direction. The ITC is calculated iteratively using the resultant power and temperature changes along with an assumed power coefficient. The Moderator Temperature Coefficient (MTC) is then calculated by subtracting the predicted Fuel Temperature Coefficient (FTC) from the measured Isothermal Temperature Coefficient.

B. With Center CEA Movement

a. Isothermal Temperature Coefficient

With the reactor at steady state and equilibrium xenon and CEA group 6 at 120 inches withdrawn, a small step change in the turbine control valve position is made and then adjusted to establish a new coolant inlet temperature. This change produces a small turbine load-reactor power

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mismatch. The temperature change results in a reactivity feedback. This reactivity is matched with equal and opposite reactivity by movement of the center CEA (holding reactor power constant). The ITC is calculated iteratively knowing the power and temperature changes along with the center CEA integral worth curve and by using the test predictions as initial guesses for the Isothermal Temperature and Power Coefficients. The MTC is calculated as described previously.

b. Power Coefficient

A reactivity insertion is made using the center CEA, resulting in a change in reactor power. Average coolant temperature is held constant by changing turbine load to match reactor power. The reactor settles out at a new power when the reactivity feedback due to change in power is equal and opposite to the CEA reactivity insertion. The Power Coefficient is calculated iteratively in a manner similar to the ITC calculation.

6.1.10.3 Test Results

The Variable T_{AVG} Test was performed at the 20% power plateau as part of the power ascension test program. During the ITC measurement with no CEA movement, T_{cold} was swung approximately $\pm$ 5°F about the programmed T_{cold} at 20% power or 546.5°F. However, during the ITC measurement with center CEA movement, T_{cold} had to be lowered from the programmed T_{cold} in order to accommodate a complete 10°F temperature swing and still maintain the center CEA position in an area of relatively high worth.

The Power Coefficient measurement with center CEA movement was performed by withdrawing CEA 6-1 to 135" withdrawn and noting the change in reactor power in moving CEA 6-1 from 120" withdrawn (the group 6 average position) to 135" withdrawn. The reactor power was then decreased approximately twice the amount determined above by inserting CEA 6-1. This cycle was performed three times.

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The final ITC and power coefficient values were the average value of the runs conducted. The measured value, test predictions, and acceptance criteria for the 20% power plateau are shown in Table 6.1.10.1. The physics test prediction for the ITC was corrected for the measured RCS boron concentration.

6.1.10.4 Conclusion

The measured values for the Isothermal Temperature Coefficient and Power Coefficient compared well with the predicted values. Agreement between measurement and prediction was well within the uncertainties associated with each parameter.

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TABLE 6.1.10.1

Nominal Reactor Power	20%
Boron Concentration (RCS)	828 ppm
Isothermal Temperature Coefficient	
Measured	-0.200×10^{-4} $\Delta\rho/°F$
Predicted	-0.17×10^{-4} $\Delta\rho/°F$
Acceptance Criteria	$\pm 0.5 \times 10^{-4}$ $\Delta\rho/°F$
Power Coefficient	
Measured	-1.189×10^{-4} $\Delta\rho/\%$ Power
Predicted	-1.17×10^{-4} $\Delta\rho/\%$ Power
Acceptance Criteria	$\pm 0.2 \times 10^{-4}$ $\Delta\rho/\%$ Power

6.1.11 CEA SHADOWING FACTOR VERIFICATION TESTS

6.1.11.1 Purpose

The purpose of this test was to verify that the CEA shadowing factors used in the CPCs are valid.

6.1.11.2 Test Method

The test was performed at 20% power with initial conditions of equilibrium Xenon, All Rods Out (ARO) and T_{cold} at 545.5°F ± 0.5°F. Data was taken for numerous CEA configurations (see Table 6.1.11.1) and reduced via procedure. Reactivity was monitored on a Reactivity computer. Power was monitored on the CPCs, COLSS, and Turbine-Generator MWe.

The CEA shadowing factors were calculated as follows:

For configurations without Part Length CEAs inserted:

$$F_x = \frac{\sum_{i=1}^{3} D_i^x \text{ (with CEA's inserted)}}{\sum_{i=1}^{3} D_i^x \text{ (ARO)}} * \frac{\text{Power (ARO)}}{\text{Power (with CEA's inserted)}}$$

For configurations with Part Length CEAs inserted:

$$F = \frac{D_2^x \text{(with CEA's inserted)}}{D_2^x \text{ (ARO)}} * \frac{\text{Power (ARO)}}{\text{Power (with CEA's inserted)}}$$

Where: F_x = CEA shadowing factor for CPC Channel x (x = A, B, C, D)

D_1^x = upper excore detector, Channel x

D_2^x = middle excore detector, Channel x

D_3^x = lower excore detector, Channel x

Power = reactor power (COLSS calorimetric value)

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After the CEA shadowing factors are calculated, they are compared to appropriate acceptance criteria. If any shadowing factor does not agree with its acceptance criteria, new correction factors (ASM2, ASM3, ASM4) are calculated. Revised acceptance criteria are then generated. If any shadowing factor does not agree with the revised acceptance criteria, new power uncertainty factors (BERR1 and BERR3) are calculated to compensate for the additional error difference.

Any revised correction and power uncertainty factors are then loaded into the CPC's.

6.1.11.3 Test Results

The channel-wise CEA shadowing factors were calculated for the CEA configurations listed in Table 6.1.11.1 and are presented in Table 6.1.11.2. The results of calculations indicated that these measured factors were not in agreement with the appropriate acceptance criteria. The channel-wise CEA shadowing factors were averaged and using the average value, new correction and uncertainty factors were calculated. These values were then loaded into the CPC's via addressable constants.

6.1.11.4 Conclusions

Upon adjustment of the correction and uncertainty factors (CPC constants ASM2, ASM3, ASM4, and BERR1, BERR3) the CPC's now accurately compensate for the effects of CEA shadowing.

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TABLE 6.1.11.1

CEA CONFIGURATIONS USED IN VERIFYING
CEA SHADOWING FACTORS / CORRECTION FACTORS

CEA CONFIGURATION	CORRECTION FACTOR
Group 6 @ LEL	
Group 6 and 5 @ LEL	ASM2
Groups 6, 5 and 4 @ LEL	
Groups 6, 5, 4 @ LEL, P @ 37.5" WD	ASM4
Groups 6, 5 @ LEL, P @ 37.5" WD	
Group 6 @ LEL, P @ 37.5" WD	ASM3
Group P @ 37.5" WD	

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TABLE 6.1.11.2

MEASURED CEA SHADOWING FACTORS

CEA/GROUP/POSITION	CHANNEL A	CHANNEL B	CHANNEL C	CHANNEL D
6/LEL	1.0531	1.0643	1.0605	1.0513
6/LEL, 5/LEL	0.8517	0.8534	0.8454	0.8485
6/LEL, 5/LEL, 4/LEL	1.0386	1.0497	1.04[illegible]	1.0438
6/LEL, 5/LEL, 4/LEL, P/37.5"	1.0906	1.1015	1.0861	1.0956
6/LEL, 5/LEL, P/37.5"	0.8808	0.8838	0.8844	0.8856
6/LEL, P/37.5"	1.0848	1.0957	1.1096	1.1037
[illegible]/37.5"	1.0545	1.0604	1.0600	1.0638

6.1.12 UNIT LOAD TRANSIENT TEST

6.1.12.1 Purpose

The purpose of this test was to:

A. Demonstrate that the following systems operate satisfactorily in the automatic mode to maintain plant parameters within acceptable limits during steady state power operations, 5% per minute power down ramps, 1% per minute up ramps and 10% down step change in plant power:

a. Reactor Regulating System (RRS)

b. Feedwater Control System (FWCS)

c. Steam Dump and Bypass Control System (SDBCS)

d. Megawatt Demand Setter (MDS)

e. Pressurizer Level Control System (PLCS)

f. Pressurizer Pressure Control System (PPCS)

B. Monitor the response of the RRS, FWCS, PLCS, PPCS and SDBCS to plant trips.

6.1.12.2 Test Method

This test was performed at the 3% and 20% power plateaus. The tests which occured at each plateau are listed below:

A. 3% Plateau

The reactor was stabilized at 3% power and the control systems verified operational with PLCS and PPCS in automatic operation and steam pressure being maintained by the SDBCS in automatic controlling the 5% dump valve (2CV-0303). 2CV-0303 was placed in manual and slowly closed. As 2CV-0303 closed, 2CV-0302 automatically opened. When stable conditions were achieved (with 2CV-0303 fully closed), 2CV-0303 was returned to automatic mode (without balancing automatic and manual control signals). Plant conditions were allowed to stabilize with 2CV-0303 controlling steam pressure automatically. Various parameters were recorded on strip chart recorders and computer trends during the performance of the test. The test data was reduced and analyzed to verify proper operation of the SDBCS.

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6.1.12.2 (cont'd) B. 20% Power Plateau

a. Automatic Steady State Operation

The reactor was stabilized at 20% power and the control systems verified to be in the automatic mode of operation. Strip chart recorders and computer trends were established as required and a 30 minute steady state run was performed. Following the 30 minute run, the test data was collected, reduced and analyzed to determine the acceptability of the control systems operations. Control system setpoint adjustments were performed as necessary based on the results of the test data analysis. The above described process was performed until no further setpoint changes were required.

b. SDBCS Test

The reactor was stabilized at 20% power and the control systems verified to be in the automatic mode of operation. The turbine load was decreased at approximately 1/2% per minute. As the turbine load was decreased, the SDBCS automatically opened 2CV-0303 to control steam pressure. When 2CV-0303 was 50% open, the down power transient was secured and turbine load was increased at approximately 1/2% per minute until 2CV-0303 was fully closed and the steam pressure was stable at its normal level. Strip chart recorders and computer trends were used to monitor the transient, the data was analyzed to determine the acceptability of the control systems operations. Control system setpoint adjustments were performed as necessary based on the results of the test data analysis. The test was rerun as necessary until no further adjustments of the SDBCS were necessary.

c. FWCS Tests

The reactor was stabilized at 20% power and the control systems verified to be in the automatic mode of operation. Steam Generator level transients were initiated by changing the setpoint at the master controller.

6.1.12.2 (cont'd)

Master Controller No. 1 controlled level in Steam Generator A and Master Controller No. 2 controlled level in Steam Generator B. After each of the transients listed below, strip chart recorder traces and computer trends were analyzed and the FWCS setpoints adjusted as required. The transient was repeated until no further adjustments were required. The following transients were completed first on FWCS No. 1 then on FWCS No. 2:

	INITIAL STEAM GENERATOR LEVEL	FINAL STEAM GENERATOR LEVEL	RATE OF CHANGE
1)	70%	60%	10% per min.
	60%	70%	10% per min.
2)	70%	60%	1% per sec.
	60%	70%	10% per min.
3)	70%	80%	10% per min.
	80%	70%	10% per min.
4)	70%	80%	1% per sec.
	80%	70%	10% per min.

d. RRS Tests

The reactor was stabilized at 20% power, CEA Group 6 between 113" and 135" withdrawn, the CEDMCS in manual, sequential, all other control systems in automatic. Using RRS #1 (#2) for temperature control T_{AVG} was decreased 4.5°F less than T_{REF}, the CEDMCS was placed in Automatic Sequential and the resultant transient recorded on strip chart recorders and computer trend groups. The CEDMCS was returned to the Manual Sequential mode, the results analyzed and RRS setpoints adjusted as required. Either or both transients were repeated as necessary until no further adjustments were necessary.

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6.1.12.2 (cont'd)

e. MDS Test

The reactor was stabilized at 20% power, CEA GR 6 between 113" and 135" withdrawn, the CEDMCS in Manual Sequential, the MDS in the Ready Mode and all other control systems in automatic. Turbine load was decreased by 20 MWe from the turbine control panel, control was transferred to the MDS. The MDS was placed in the Operator Set mode and the turbine load was increased 20 MWe at 1% per minute. The MDS was placed in the Ready Mode and turbine control was returned to the turbine control panel where load was increased by 20 MWe. The MDS was placed in the Operator Set mode and the turbine load was decreased 20 MWe at 5% per minute. Both transients were recorded using strip chart recorders and computer trends. The test data was analyzed and the MDS setpoints adjusted as necessary. The transients were repeated until no further setpoint adjustments were necessary.

f. Reactor Trip Test

The reactor was stabilized at 20% power. Strip chart recorders were set up to monitor the parameters specified in the test procedure. Reactor was tripped in accordance with the controlling procedure and data taken to record the transient.

6.1.12.3 Test Results

A. 3% Power Plateau

The test was performed as described, however, the SDBCS Master Controller was in local setpoint rather than remote setpoint. When in local setpoint, the setpoint is manually set using the thumbswitch mounted on the face of the controller. The 13% steam dump valve 2CV-0302 was automatically opened to maintain steam pressure at the setpoint as the 5% steam dump valve (2CV-0303) was closed. When 2CV-0303 was returned to automatic operation, 2CV-0302 closed as 2CV-0303 opened to maintain steam pressure at the required setpoint.

6.1.12.2 (cont'd) B. 20% Power Plateau

a. Automatic Steady State Operation

The results of this test showed the following:

1. No spurious CEA motion

2. Steam generator levels were maintained at 70.5 $\pm$ 5%

3. Pressurizer level maintained at the programmed level $\pm$ 5%

4. Pressurizer pressure maintained at 2250 psia $\pm$ 15 psi

5. Reactor power was maintained at 20 $\pm$ .5%

6. T_{AVG} was maintained at $\pm$ 1°F of the desired value

7. Steam generator pressure was maintained at $\pm$ 15 psi of desired value.

b. SDBCS Test

The test was performed as required except the main feed pump turbine speed control was not in automatic as required. As the turbine load was decreased, the SDBCS controlled pressure around setpoint. The maximum value of steam pressure was 1035 psia (by plant computer). This value was reduced to and maintained at the setpoint approximately five minutes after this maximum was reached. Computer trends and brush pen recordings showed no unusual transients.

c. FWCS Test

The brush pen recorder data of steam generator level and flow show that proper feed water control was maintained. The graphs indicate that the level demanded by the FWCS #1 (#2) would be achieved in steam generator A (B) while the level in the remaining steam generator was relatively unaffected. During the transient, an overshoot of the demanded set-point was seen with the level finally settling down to $\pm$ 2% in a fairly short period of time.

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6.1.12.2 (cont'd)

d. RRS Test

The RRS tests were performed for RRS #1 and RRS #2. Analysis of the data for this test revealed proper CEA motion for each transient. The artificially created power defect was damped quickly with very little overshoot.

e. MDS Test

The 1% per minute up power test was performed as described using the MDS in the operator set mode. However, due to the initial CEA position, (114.9 inches withdrawn) power could not be increased 20 MWe prior to reaching an all CEA's out configuration as required by the test procedure. Therefore, power was increased until the group 6 CEA's were 146 inches withdrawn and then the 5% per minute down transient was performed. Data was collected as required.

f. 20% Reactor Trip

The required data was monitored on the brush recorders, thus meeting the requirements of this test. After the trip, the feed flow was excessive, the Reactor Trip Override bias potentiometers in FWCS #1 and #2 were adjusted to provide proper feedwater flow following a reactor trip.

6.1.12.4 Conclusion

A. 3% Power Plateau

The SDBCS operated satisfactorily to maintain steam pressure at the desired setpoint during the test.

B. 20% Power Plateau

a. Steady State Operation

The ability of the various control systems when in automatic mode, to maintain steady state operations was demonstrated.

6.1.12.2 (cont'd)

b. SDBCS Test

The SDBCS operated satisfactorily to control and return steam pressure to the prescribed setpoint.

c. FWCS

The FWCS has been shown to operate as expected in the Automatic Control mode. The ability of FWCS #1 and #2 to achieve demanded setpoints at various rates has been demonstrated.

d. RRS Test

Both RRS #1 and #2 operated satisfactorily to maintain T_{AVG} within the T_{REF} control band as designed. No adjustments to the RRS setpoints were required.

e. MDS Test

The MDS operated as designed during increasing and decreasing power transients. Oscillation of $\pm$ 3 MWe around the setpoint were observed during steady state operation between transients. Adjustments were made to decrease the magnitude of the oscillation and improved steady state performance was achieved.

f. Reactor Trip Test

All the required data was taken during the reactor trip test at the 20% power plateau. Adjustment of the reactor trip override bias potentiometers provided proper flow signals which provides proper feedwater flow following reactor trip. All control systems functioned properly during the reactor trip to maintain system parameters within their specified bands.

968 055

6.1.13 SHAPE ANNEALING MATRIX AND BOUNDARY CONDITION MEASUREMENT TESTS

6.1.13.1 Purpose

The objective of this test was to measure the Shape Annealing Matrix (SAM) and to verify the Boundary Point Power Correlation (BPPC) constants for the CPC's. The primary purpose for performance of this test at the 20% power plateau was to evaluate the test method and to provide additional data which could be used to verify the adequacy of the SAM and BPPC constants to be measured at the 50% plateau.

6.1.13.2 Test Method

The SAM coefficients and BPPCs are determined from a least squares of the measured excore detector readings and corresponding axial power distribution determined from the incore detectors signals. Since these values must be representative for rodded and unrodded cores throughout life, it is desirable to use as wide a range of core axial shapes as are available to establish their values. This is done by initiating an axial xenon oscillation. Data is periodically gathered during the oscillations so that the data will be representative of as wide a range of axial shapes as possible. Incore, excore and related data is recorded. This data is input to the CECOR incore code which relates the incore detector signals to power distribution and summarizes the necessary power distribution and excore detector data in a form and format which can be easily input to programs used to perform the least squares fitting. The output from CECOR includes:

1) the excore detector fractional responses for each CPC,

2) the core peripheral power fraction for the upper, middle, and lower third of the core,

3) the core average power fractions for the upper, middle, and lower third of the core, and

4) the upper and lower core boundary average power.

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The above CECOR output is then used to determine a "best set" of SAM coefficients and BPPC constants by using least squares analysis. The results of these calculations are then used to adjust the power uncertainty factors (BERR1, BERR3) used by the CPC's in the LPD and DNBR calculations.

6.1.13.3 Test Results

While it was intended to monitor the xenon oscillation for at least 60 hours, sufficient self-dampening had occurred within 50 hours to terminate the data collection. One hundred CECOR cases were run, of which approximately 20% reflected the core in a rodded configuration. The core was unrodded for the remainder of the cases.

The measured SAM is presented in Table 6.1.13.1. Contrary to the initial intent, "measured" 20% SAM values were input to the CPC's prior to escalating to 50% power. This was done because the original predicted values were judged to be inadequate to support the 50% testing. Due to the fact that the BPPC acceptance criteria were developed for the 50% measurement, no evaluation of these results were performed.

6.1.13.4 Conclusions

Performance of this test verified the computational methods for determining the SAM and BPPC coefficients and provided a satisfactory Shape Annealing Matrix for the 20% plateau. Two areas were identified for possibly improving the data to be collected during the 50% plateau measurement. First, the xenon oscillation during this test was initiated from an equilibrium xenon condition with CEA Group 6 at 120" withdrawn. By starting from an ARO equilibrium Xe condition, a "larger" oscillation would be obtained. Secondly, the withdrawal of CEA Group 6 to an ARO configuration during this test was performed very slowly. This caused some dampening of the oscillation. A more rapid withdrawal of Group 6 during the 50% plateau test should also result in a "larger" oscillation.

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TABLE 6.1.13.1

SHAPE ANNEALING MATRIX COEFFICIENTS

(20% Power)

SAM COEFFICIENT	CPC CHANNEL A	B	C	D
S(1,1)	5.1452	6.8268	6.6279	7.0577
S(1,2)	1.7594	-1.0973	-.7470	-1.8424
S(1,3)	-4.1878	-2.6112	-2.7996	-2.0400
S(2,1)	-1.0207	-1.1478	-.6828	-.4026
S(2,2)	4.7274	4.9342	4.3819	3.8429
S(2,3)	-.3914	-.4685	-.3377	.0100
S(3,1)	-1.1244	-2.6790	-2.9451	-3.6550
S(3,2)	-3.4868	-.8368	-.6349	.9995
S(3,3)	7.5792	6.0796	6.1374	5.0300

968 058

6.1.14 REACTOR TRIP WITH SHUTDOWN OUTSIDE THE CONTROL ROOM TESTS

6.1.14.1 Purpose

This test was performed to demonstrate that the plant could be taken from 20% power to hot standby from outside the control room following a reactor trip.

6.1.14.2 Test Method

The test was initiated from 20% power by tripping the reactor manually from the remote shutdown station. One operating crew performed the shutdown from the remote stations while a second crew stood by in the control room to take action if needed. The remote crew performed all immediate and follow-up actions of the remote shutdown operating procedure. To monitor the trip, 47 parameters were recorded on brush recorders and 12 parameters were placed on a 1 second computer trend. Watchstanders were stationed to verify proper EHC indication of the turbine trip.

6.1.14.3 Test Results

The main events were recorded as follows: Reactor trip breakers 2, 3, 6 and 7 were manually opened. The CEDM main power bus undervoltage relays 1 and 2 deenergized, tripping the main turbine in 1.143 seconds. Start-up transformer #3 auto transferred to supply vital busses 2H-1, 2A-1, and 2A-2. The generator output breaker (500 KV) tripped within 1.572 seconds.

Table 6.1.14.1 and Figures 6.1.14.1 thru 6.1.14.3 show the results of the computer trends.

6.1.14.4 Conclusion

As indicated in Table 6.1.14.1, all parameters were established for hot standby conditions by 240 seconds following the reactor trip. All control actions were conducted from outside the control room.

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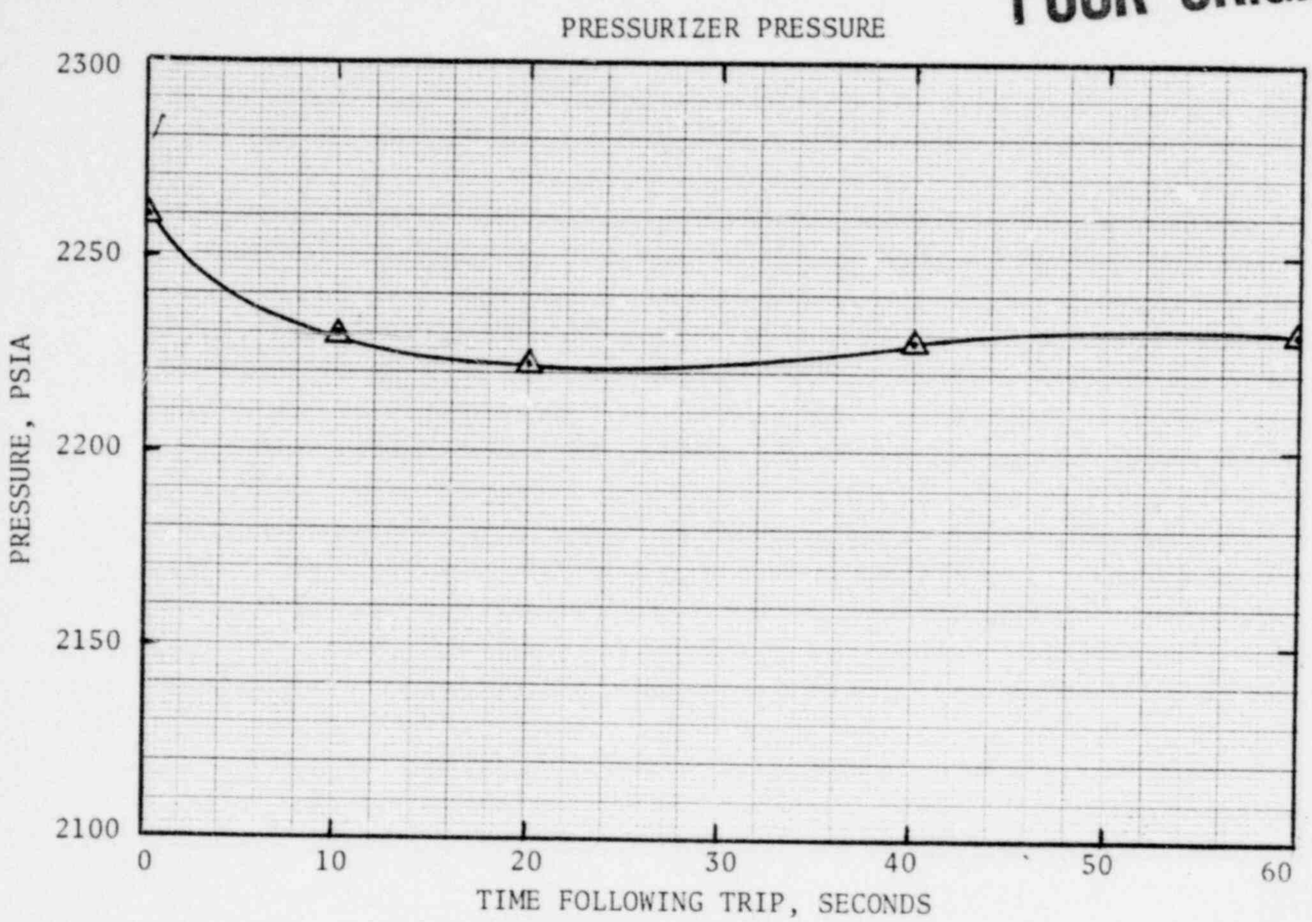
TABLE 6.1.14.1

POST TRIP REVIEW OF COMPUTER TREND CROUP DATA

20% Reactor Trip with Remote Shutdown

TIME AFTER TRIP	NEUTRON POWER	LEVEL			PRESSURE			TEMP.			
		S/GA	S/GB	PZR	S/GA	S/GB	RCS	TH_1	TH_2	TC_1	TC_2
0 sec	18.6	71.3	72.8	36.0	961.7	964.4	2261.4	556.0	556.4	546.7	546.4
10 sec	0.4	55.0	54.6	33.0	997.9	1000.9	2234.0	551.7	552.4	547.1	546.5
20 sec	0.2	60.0	61.6	33.0	1006.3	1009.4	2223.0	549.0	549.3	547.5	546.8
40 sec	0.1	62.2	62.8	32.6	1011.8	1015.0	2231.0	548.2	548.4	547.6	547.5
60 sec	0.05	62.9	63.0	32.2	1012.2	1015.3	2232.7	547.9	548.4	547.7	547.7
90 sec	0.01	64.4	63.1	31.3	1009.4	1012.4	2232.7	547.2	547.6	547.3	547.3
240 sec	0.01	68.2	68.4	30.9	996.0	998.9	2262.0	545.5	545.9	545.2	545.2



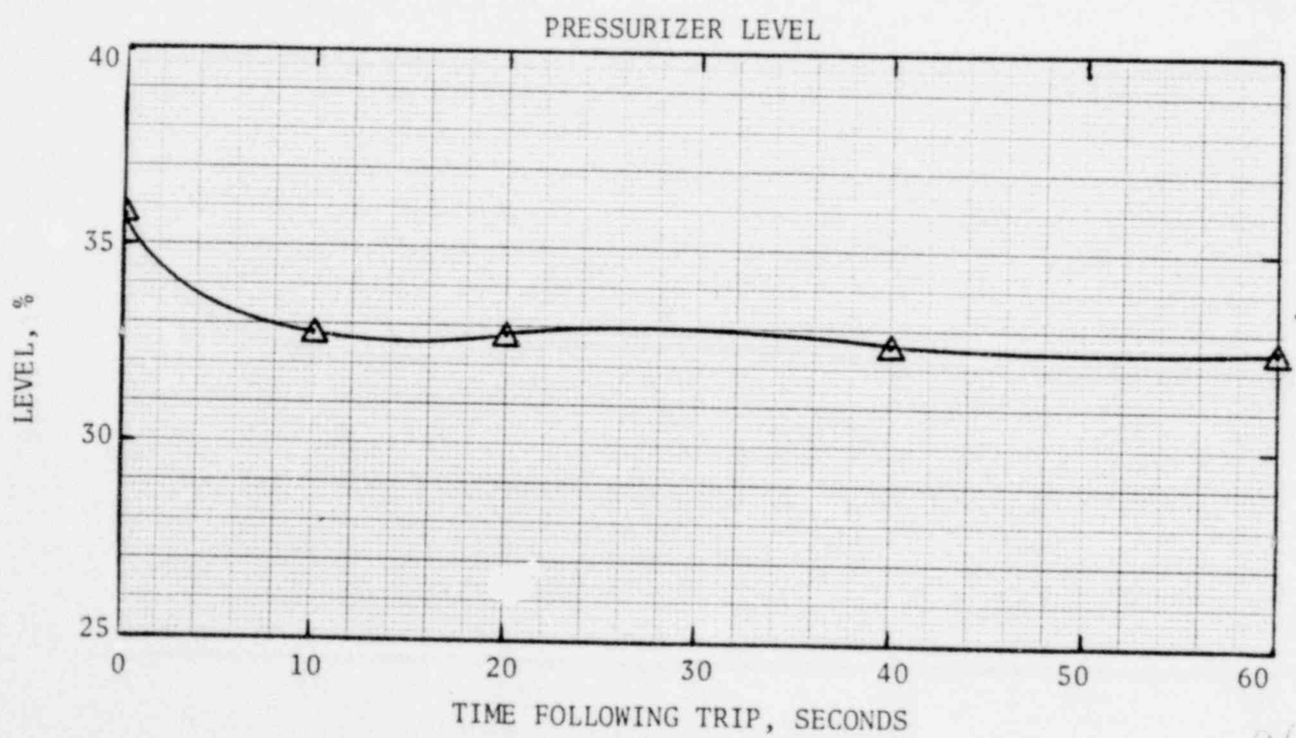


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FIGURE 6.1.14.1

POOR ORIGINAL

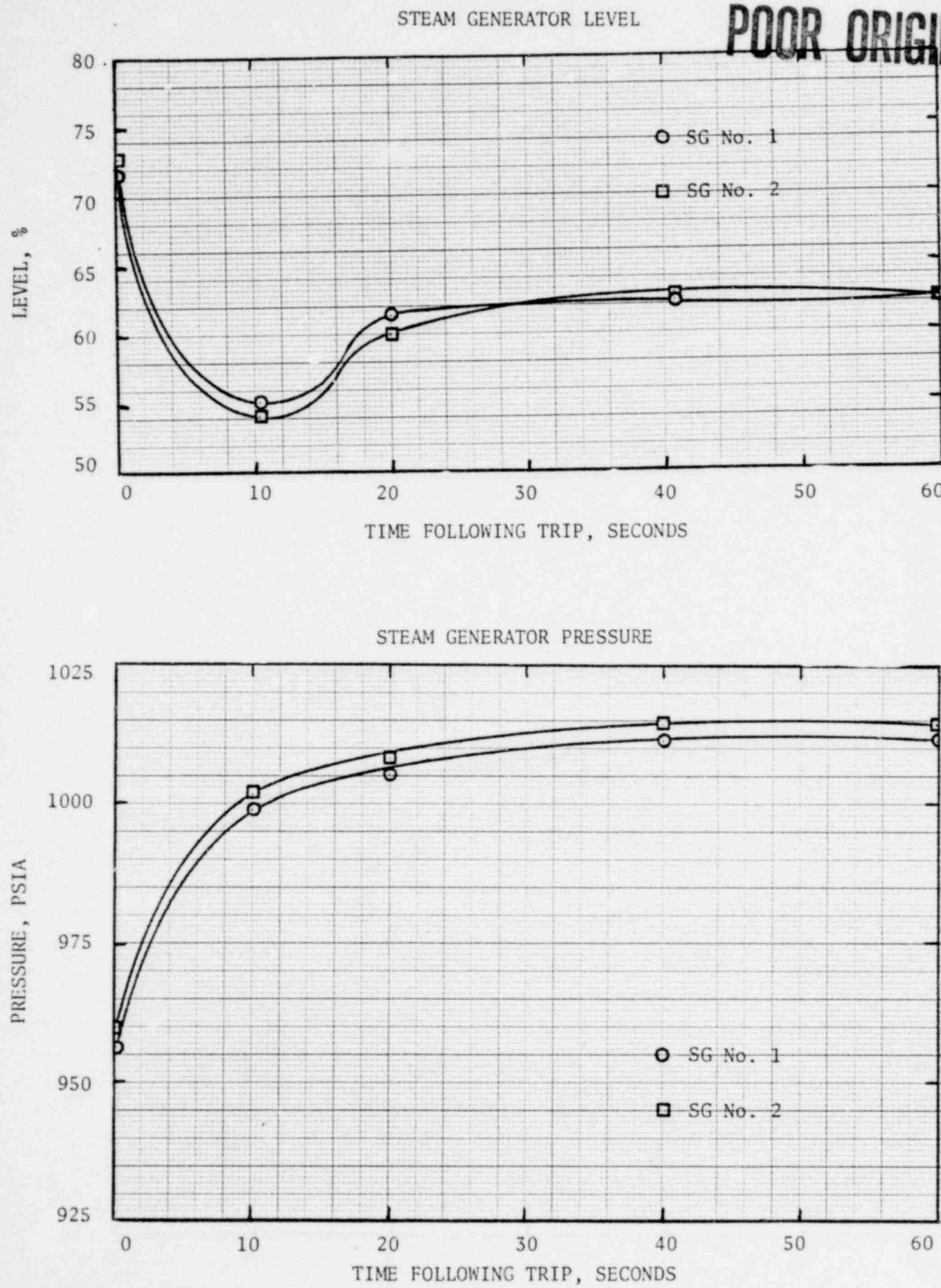

FIGURE 6.1.14.2

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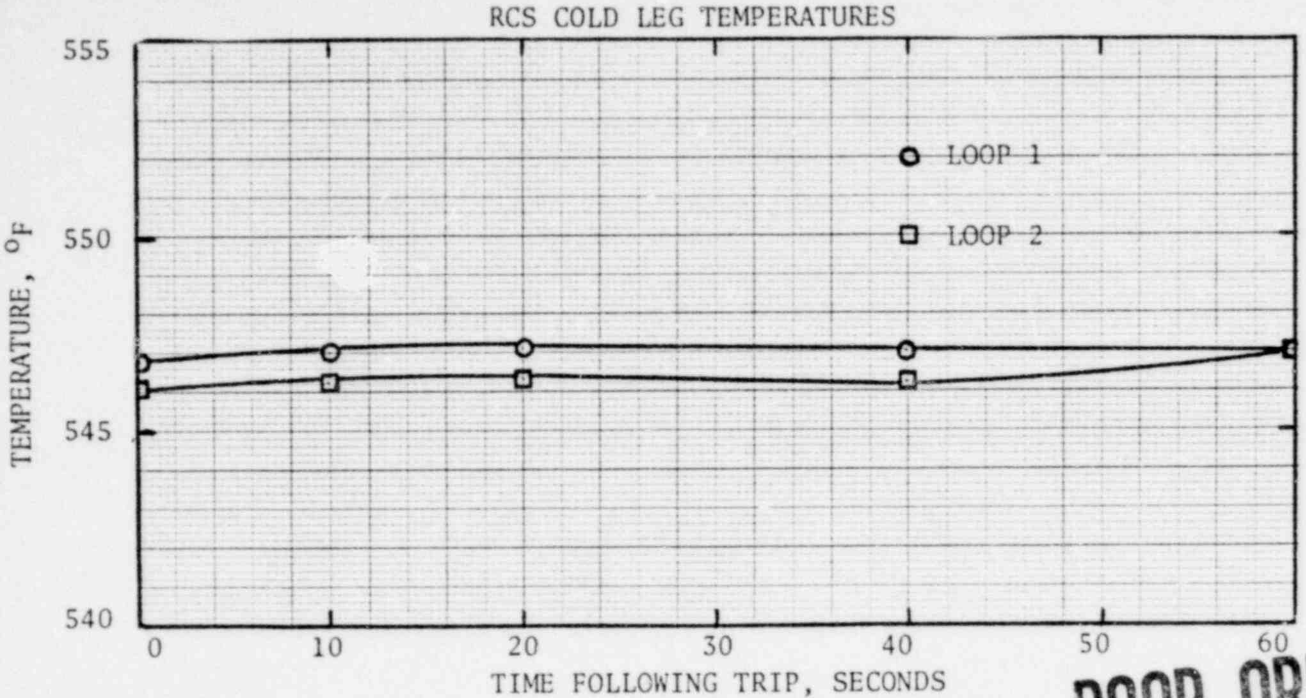

POOR ORIGINAL

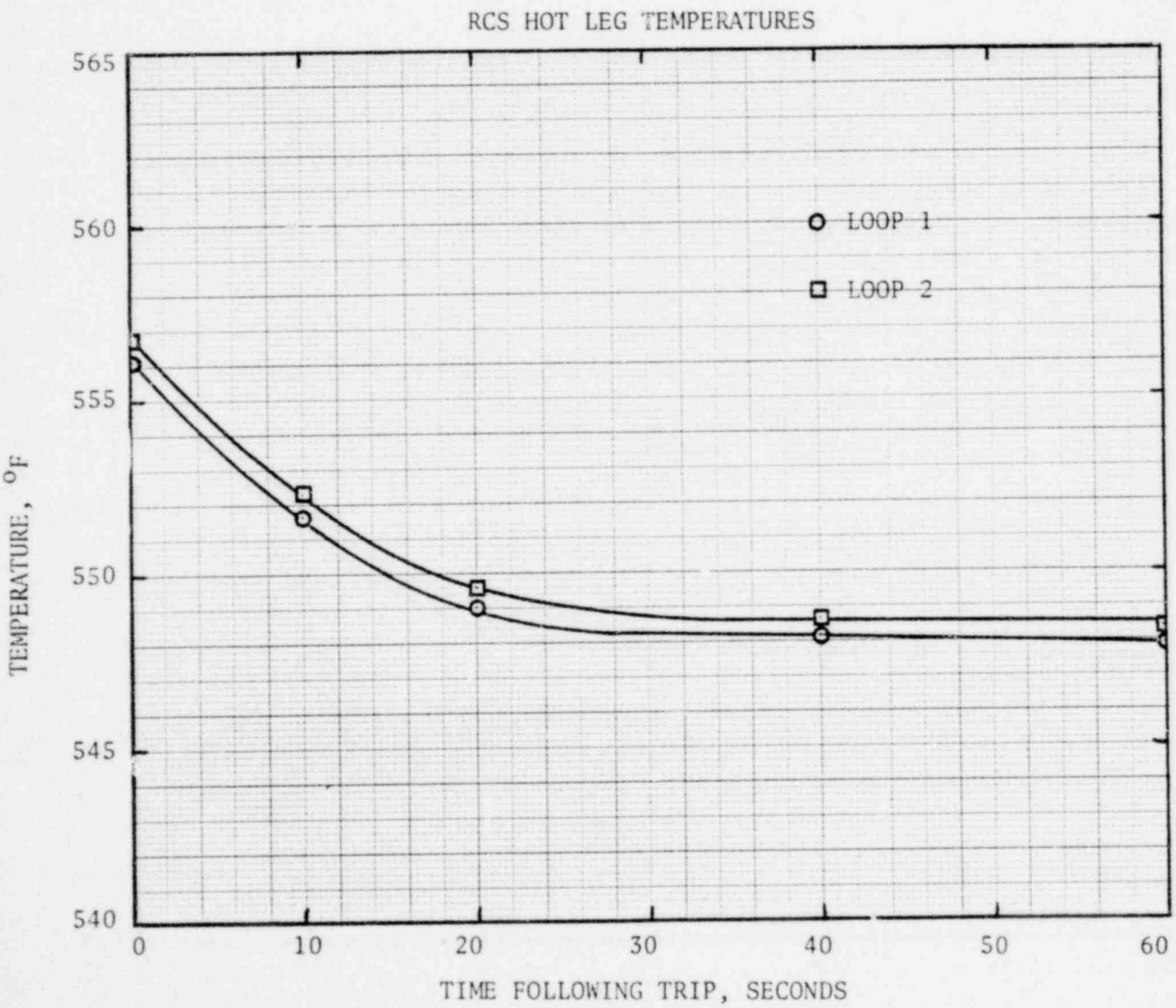

FIGURE 6.1.14.3

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6.1.15 INCORE DETECTOR SIGNAL VERIFICATION TESTS

6.1.15.1 Purpose

To verify the proper conversion of the signal from the incore detectors to voltage as read by the plant computer. This comparison of the signal generated by the incore detector to the voltage seen by the plant computer also verified the proper operation of the incore amplifier.

6.1.15.2 Test Method

The plant was maintained at a power level of approximately 20% ± 0.2%. The RCS pressure was at 2250 psia ± 15 psia. Following determination of the connector number for the core location desired, the amplifier associated with the connector was determined. The input connector was disconnected and a special test cable connected between the connector and the amplifier assembly. Using a pico ammeter, the current was measured and recorded for the level 1 detector while simultaneously recording the raw incore signal. This was repeated for the remaining detector levels for the incore string under test. Following completion of the string, the special test cable was disconnected and the input connector to the amplifier bin reconnected. This procedure was repeated for the remaining incore detectors.

6.1.15.3 Test Results

Problems with unstable signals caused poor agreement between detector signal and computer raw signals. Voltages as read on the computer were consistently low. Following an extensive investigation, the problems were determined to lie with the process computer input multipliers and not with incore detectors or the amplifiers. The incore detector system was adjusted and a retest performed at which time it was found that 37 detectors out of 220 were out of the acceptance criteria. Out of spec incores were checked, readjusted, and retested. Eventually all except 2 detectors were brought into agreement. These were determined to be failed in the detector assemblies.

6.1.15.4 Conclusion

The proper conversion of the signal from the incore detectors to voltage as read by the plant computer has been verified. Also, this comparison verified the proper operations of the incore amplifiers.

968 064

6.1.16 MOVEABLE INCORE DETECTOR TESTS

6.1.16.1 Purpose

This procedure was performed to provide baseline data on the moveable incore detector system (MICD). At the 20% power plateau, the MICD Manual Pause Interval Constant was to be determined and set into the computer.

6.1.16.2 Test Method

The reactor was operating at 20% power with temperature, pressure and level maintained constant. A brush recorder was set up to monitor the output of the moveable incore detector amplifier. A computer trend group was established on a one second trend monitoring the detector position and detector output.

With the pause interval set for 15 minutes, the manual mode of operation of the moveable incore drive system was selected.

Using the computer console, the #2 moveable incore drive machine was selected and the detector was driven to Level 3 of path #13.

After 15 minutes, the computer pulled the detector out of active core to the home position. The brush recorder tracing and the computer trends were analyzed for pause interval time. The detector output was considered stable when the detector build-up of current reached steady state.

6.1.16.3 Test Results

The procedure was commenced and aborted after several attempts.

Maintenance activities were initiated to correct the deficiencies and the procedure was recommenced. All steps were performed satisfactorily. The pause interval was determined to be 291 seconds.

6.1.16.4 Conclusion

The data collected, on the computer trend groups, correlated well with the data recorded on the brush recorders. MICDB machine operates as required and the new pause interval value has been entered into the computer.

968 065

6.1.17 FEEDWATER CONTROL SYSTEM POST - TRIP SETTING TESTS

6.1.17.1 Purpose

The purpose of this test was to adjust the Feedwater Control System (FWCS) Reactor Trip Override (RTO) bias voltage which corresponds to feed flow at 5% reactor power. The 5% override voltage automatically adjusts feed flow to prevent excessive cooldown of the RCS following a reactor trip.

6.1.17.2 Test Method

The FWCS valve lineup was set to duplicate that which would exist immediately following a reactor trip. While maintaining reactor power at 5% steady state, the flow demand voltages from FWCS's 1 and 2 were monitored to determine required flow for decay heat. After maintaining the required flow for fifteen minutes, the average flow was determined and the RTO bias potentiometer was adjusted to a value corresponding to the average flow demand signal.

6.1.17.3 Test Results

Several problems were incurred during the performance of this test:

A. At this low power level, steam generator levels are very sensitive to any feedwater control and the operators were unable to maintain levels within the band of 70.5% $\pm$ 10%.

B. There were problems with the feedwater pump recirculation breaker tripping and closing the valve when the controller was in automatic. In order to prevent tripping the plant, the controller was placed in manual.

C. The third problem was the FWCS #2 Master Controller. The procedure required both Master Controllers to be in manual, however, operators were having trouble controlling FWCS #2 in manual so it was put in automatic.

During performance of the fifteen minute flow test, FWCS #1 was in manual. The voltage output had large discrete changes rather than smooth transients which made it difficult to determine an average voltage. The average

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voltage from FWCS #2 as determined from the brush recorder charts was 1.5V. This value was then input to the Reactor Trip Override bias on both FWCS #1 and #2.

6.1.17.4 Conclusions

A. Although the FWCS was not operating satisfactorily during this test sequence, the voltages obtained were input as preliminary settings until a later time at which better values could be obtained. During the trip from 20% power, the FWCS operation and RCS cooldown rate were monitored to determine the validity of the results. The settings were slightly high and were appropriately adjusted at that time.

B. The test was not performed exactly as originally written. However, the intent, which was to set the RTO bias potentiometer, was accomplished satisfactorily.

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6.1.18 CONDENSATE AND FEEDWATER SYSTEM POWER ESCALATION TESTS

6.1.18.1 Purpose

The purpose of this test was to:

A. Obtain base operating data while demonstrating the ability of the Main Feedwater System to supply the steam generators at the required pressures, temperatures, and flows under all anticipated steady state conditions.

B. Verify the proper operation of the FWP recirc. valves.

6.1.18.2 Test Method

With the reactor at approximately 20% power, the Feedwater Control system is placed in Mode 1 (full auto) and flows are allowed to stabilize. Following flow stabilization, main feed pump data and flow valve position data is recorded from computer points and trend groups.

The 20% baseline data was also obtained from local readings and computer data points as noted in the procedure.

6.1.18.3 Test Results

The results confirmed that the main feedwater pump minimum flow valves maintain the required pump suction flow. Also, the base data obtained at the 20% power level was in agreement with guidelines per the GE heat balance diagrams.

6.1.18.4 Conclusion

At the 20% plateau test sequence, the base operating data was obtained and proper operation of the FWP recirc. valves was verified. Also, the ability of the Main Feedwater System to supply the steam generators at the required pressures, temperatures, and flow rates was demonstrated.

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6.1.19 MAIN TURBINE ELECTRO HYDRAULIC CONTROL TESTS

6.1.19.1 Purpose

This test was performed to verify that the Electrohydraulic Control System (EHC) functioned properly during the 20% trip.

6.1.19.2 Test Method

Baseline data was collected prior to the turbine trip. Data was obtained on each hydraulic fluid pump while the other pump was in standby. During the 20% trip, a watchstander observed the Main Stop Valves, Main Turbine Control Valves, and Main Turbine combined Intercept Valves.

6.1.19.3 Results and Conclusion

All the EHC controlled valve indications were verified as closed. The EHC system functioned properly during the 20% trip.

6.1.20 FEEDWATER HEATER VENTS, DRAINS AND WATER INDUCTION TESTS

6.1.20.1 Purpose

The purpose of this test was two fold:

A. To demonstrate the satisfactory operation of the Feedwater Heaters during steady-state conditions, and

B. To demonstrate the satisfactory operation of the Feedwater Heater and Heater Drain Tank dump and dump bypass valves to perform their function in the event of high heater shell and drain tank levels.

6.1.20.2 Test Method

Each individual Feedwater Heater shell and drain was instrumented with appropriate pressure gauges to allow test personnel to monitor the performance of the heaters.

Baseline data including process computer performance calculations to determine Feedwater Heater Terminal Temperature Difference and Drain Cooler Approach Temperatures.

6.1.20.3 Test Results

The required baseline feedwater heater data was obtained for this plateau. It was determined that the pressure transducers and transmitters installed on the #7, #6, and #5 feedwater heater shells were not ranged properly, i.e., they will not read sufficiently low enough to indicate the near vacuum conditions present in these heaters at low powers.

6.1.20.4 Conclusions

Since design data does not exist for the secondary plant at 20% power, the data gathered serves an information purpose only.

New pressure sensing equipment is presently on order for the #7, #6, and #5 heaters.

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6.1.21 VIBRATION AND LOOSE PARTS MONITOR (V&LPM) TESTS

6.1.21.1 Purpose

The purpose of this test was to provide baseline data for core vibration and loose parts monitoring at 0% and 20% of reactor power.

6.1.21.2 Test Method

This test consisted of basically two parts: Individual Reactor Coolant Pump (RCP) Baseline Data, and RCS V & LPM Baseline Operating Data. These are discussed below in more detail.

A. Individual RCP Baseline Data (< 0% Power)

For each individual RCP, a tape recording was made on the V & LPM capturing that RCP's start up as well as steady state operation. The recording was then aurally and spectroscopically (by frequencies) evaluated in order to identify any abnormal indications. This sequence was performed for each RCP with the other RCP's secured.

B. RCS V & LPM Baseline Operating Data

At reactor power levels of 0%, and 20% power, baseline V & LPM data was obtained for steady state operating conditions. For each area of the RCS which is monitored by the V & LPM, data was acquired via tape recordings and frequency/power spectrum plots. In addition, during these data acquisition runs, various parameters were trended for ∿ 5 minutes on the plant computer.

6.1.21.3 Test Results

The data as described above was obtained at the required power levels.

6.1.21.4 Conclusions

All data was obtained per procedure and acceptance criteria were satisfactorily met.

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6.1.22 HEATING, VENTILATING AND AIR CONDITIONING SYSTEMS PERFORMANCE TESTS

6.1.22.1 Purpose

The purpose in performing this test procedure was to:

1) Demonstrate the satisfactory performance of plant heating, ventilating and air conditioning (HVAC) systems under actual operating heat load.

2) Demonstrate that the HVAC system will satisfy the design criteria at cold shutdown conditions.

3) Provide baseline temperature and/or pressure data at selected points of the plant for future reference.

6.1.22.2 Test Method

This test was performed at cold shutdown and 20% power after plant conditions had stabilized for 24 hours. Cold shutdown data was taken during the winter months to ensure the coldest ambient conditions. The HVAC system status was verified to be in the correct operating mode. Data was taken at selected points in the plant. Temperatures outside of containment were taken using a hand held thermocouple and containment temperatures were read remotely, using installed RTD's.

6.1.22.3 Test Results

Temperatures were taken throughout the plant in accordance with the procedure.

6.1.22.4 Conclusion

The HVAC systems operate properly to maintain all plant temperatures within design tolerances.

6.1.23 BIOLOGICAL SHIELD SURVEY TESTS

6.1.23.1 Purpose

The test was conducted to accomplish the following objectives:

A. Determine background radiation levels prior to initial criticality.

B. Evaluate the adequacy of plant radiation shielding.

C. Determine radiation levels throughout the plant at various power levels.

6.1.23.2 Test Method

A comprehensive series of gamma and neutron dose rate level surveys were performed prior to initial criticality and at two low power level conditions. Low power shield tests were conducted at a steady state power level between 0% and 5% power (low power shield test #1) and between 15% and 20% power (low power shield test #2).

Dose rate surveys were taken at numerous locations which included but were not limited to the following areas:

A. Locations inside the Reactor Building.

B. Areas adjacent to the Reactor Building wall.

C. Around penetrations through the Reactor Building wall.

D. Selected points in the Turbine and Auxiliary Building.

Radiation dose rate levels at each measurement point were compared at different power levels to verify that a linear relationship existed. This was done to ensure that an extrapolation of dose rates to 100% power could be considered valid thus allowing for identification of potential problem areas.

6.1.23.3 Test Results

There were several areas where the design radiation levels were exceeded or were expected to be exceeded. These areas are listed in Table 6.1.23.1 along with suggested corrective actions

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6.1.23.4 Conclusion

Three major acceptance criteria were established to judge radiation dose rate levels.

A. Radiation levels should meet the radiation zoning criteria established by the FSAR.

   a. This criteria was satisfied for the Low Power Shield Test #1 (0% - 5% power) but not for the Low Power Shield Test #2 (15% - 20% power). See Table 6.1.23.1 for exceptions.

B. Radiation levels in unenclosed areas outside the Reactor Building should not be greater than 0.8 mrem/hr.

   a. This criteria was satisfied for Low Power Shield Tests #1 and #2 and is expected to be met at full power based on extrapolation.

C. Radiation resulting from streaming through penetrations, shielding defects, etc., will not cause a significant hazard to personnel.

   a. This criteria was satisfied for Low Power Shield Test #1 but not for Test #2. See Table 6.1.23.1 for exceptions.

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TABLE 6.1.23.1

Areas of Higher than Expected Dose Levels

AREA	PROBLEM	SUGGESTED CORRECTIVE ACTIONS
1. Reactor Building Elevation 424 East and West of Canal.	Dose Rate is expected to exceed 100 mrem/hr at 100% power.	These areas are not expected to require frequent or prolonged personnel access during power operation and therefore posting of these areas should be sufficient to ensure personnel protection.
2. Reactor Building Elevation 405 East and West of Canal	Dose rate exceeds 100 mrem/hr at 20% power	
3. Reactor Building Elevation 357 Penetrations 1, 5, and 7	Gamma streaming exceeds 100 mrem/hr at 20% power or is projected to exceed 100 mrem/hr at 100% power	These penetrations are greater than 6' above the floor and are thus noramlly considered inaccessible to personnel. However, special maintenance could require access to these areas. If measurements at higher power levels confirm that a dose rate in excess of 100 mrem/hr is anticipated at 100% power, then an engineering evaluation should be made to determine if a simple fix could be found to shield these penetrations. If a simple fix is not available, the permanent posting of these penetrations should be adequate.
4. Reactor Building Elevation 335 Penetration 4		
5. Auxiliary Buidling Elevation 335 Section A	The shield wall around the purification demineralizers does not go all the way up to the ceiling. When personnel stand above filter housings for 2F4A or B, work on overhead equipment in the area, or climb over the wall to the valve gallery for the ion exchangers, they are in a direct line with the ion exchangers. Also, a general area radiation problem is anticipated in the hallway due to scatter over the wall	An engineering analysis of the area concluded that no radiation dose rate problem existed at the present time. Further analysis may be required based on testing at higher power levels.

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6.1.24 STEADY STATE VIBRATIONS TESTS

6.1.24.1 Purpose

The purpose of this test was to monitor pipe vibrations of the systems listed below during all significant plant operating modes that were likely to cause vibration in the subject piping system.

A. Gaseous Waste System (Surge Tank 2T17 to CV-2428).

B. Penetration Room Ventilation System.

6.1.24.2 Test Method

A walkdown and visual examination of each system was conducted at each specified test mode. Piping was observed for excessive or abnormal vibration. In addition to the visual inspection of the RCS piping, the reactor coolant pump vibration monitors were checked to verify that no alarm condition was present.

6.1.24.3 Test Results

No excessive or abnormal vibration was detected in any of the above listed piping systems.

6.1.24.4 Conclusions

Vibrations of all piping systems are acceptable as determined by visual inspection.

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6.1.25 DYNAMIC TRANSIENT TESTS

6.1.25.1 Purpose

The purpose of this test was to verify the adequacy of the piping restraints for lines under transient loads as follows:

A. Gaseous Waste System (Surge Tank to CV-2428).

B. Penetration Room Vent System.

6.1.25.2 Test Method

A walkdown and visual examination of the piping and piping supports was conducted while the systems were repeatedly turned on and off to induce the transients.

6.1.25.3 Test Results

No excessive abnormal pipe or hanger movement was noted during the transients.

6.1.25.4 Conclusions

Dynamic response of the Gaseous Waste System and Penetration Room Vent System is acceptable as determined by visual inspection.

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6.1.26 TURBINE GENERATOR LOADING AT POWER TESTS

6.1.26.1 Purpose

The purpose of this test was to perform the initial generator loading, overspeed testing, exciter adjustments, and turbine balance as required. Turbine and generator baseline data w-- also collected for evaluation and future reference.

6.1.26.2 Test Method

The turbine generator was accellerated to 1800 RPM in accordance with the Turbine Startup Operating Procedure (OP 2106.09) at which time the following tests were performed:

A. Generator Excitation

B. Generator Synchronization

C. Loading to 10% Power

D. Overspeed Testing

6.1.26.3 Test Results

A. During the performance of the initial roll and subsequent generator loading, several unplanned turbine generator trips occurred. The trips were attributed to high vibrations of #2 Low Pressure Turbine. A maximum vibration of 21 mils was recorded on bearing T-2. However, the vibration problems were corrected with the addition of five balance shots.

B. Exciter checks were made in accordance with GEK 14870, "Off-Line Tests Generator Running," "Exciter Unloaded," during which time two turbine generator trips occurred, one from a high volts/Hz reading and one from an anti-motoring signal. Both were resolved and testing continued.

C. All generator synchronization tests were conducted with the exception of the Auto Synchronization which will be run during a subsequent start up. The transfer of auxiliaries to the startup transformer was performed during an actual reactor trip rather than simulated as initially planned.

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D. During each of the three overspeed trip tests, the turbine tripped at 1959 RPM. Although the overspeed trips were expected to occur between 1980 and 1998 RPM, the results were determined to be satisfactory.

6.1.26.4 Conclusions

All required tests were conducted, with the exception of the Auto Synchronization, as noted previously. Testing to this point was satisfactory and there are no problems related to this test procedure that would prohibit testing at higher power levels.

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7.0 CONCLUSION

February 1, 1979, the reactor was tripped completing the majority of the 20% test plateau. An extended maintenance outage was commenced to modify a design defect in the reactor coolant pump motors, repair and retest the main steam safety valves, implement a design change in the main condenser hot wells, install a cooling tower by pass line, replace a hydrogen seal on the main generator and repair the number 2 diesel generator engine. On June 6, 1979, the reactor was taken critical. Deficiencies from the 20% plateau were retested and all remaining testing required for escalation to 50% power was completed. 50% power was achieved June 24, 1979. The 50% power plateau testing commenced and is in progress at this time. Scheduled testing has been interrupted intermittently while investigating an anomaly with the RCS hot leg temperature. A supplementary report will be issued describing the post 20% power test as required by the Unit 2 Technical Specification and Regulatory Guide 1.16.

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