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Donald C. Cook Nuclear Plant Unit 2
Docket No. 50-316
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DONALD C. COOK NUCLEAR PLANT UNIT 2, CYCLE 8 STARTUP REPORT

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

Attn: T. E. Murley

February 22, 1991

Dear Dr. Murley:

In accordance with the requirements in Donald C. Cook Nuclear Plant Technical Specifications 6.9.1.1 through 6.9.1.3, attached is the Unit 2, Cycle 8 Startup Report. Unit 2 Cycle 8 is the first Unit 2 reload supplied by Westinghouse since Cycle 3. Westinghouse has replaced Advanced Nuclear Fuels (ANF), which supplied the previous four reloads.

The Unit 2, Cycle 8 core consists of 193 fuel assemblies, 77 of which were newly fabricated by Westinghouse. The remaining 116 assemblies are ANF assemblies that have been used in previous Unit 2 cycles. Unit 2, Cycle 8 began on November 8, 1990 and the Startup Test Program was completed on November 23, 1990.

The Startup Report is required by Technical Specification 6.9.1.1(3) in the case of "installation of fuel that has a different design or has been manufactured by a different fuel supplier." The Startup Report is being submitted within 90 days following completion of the startup test program as required by Technical Specification 6.9.1.3(1). The contents of the report are identified in Technical Specification 6.9.1.2, which requires in part that "the startup report shall address each of the tests identified in the FSAR. . . ." The tests identified in the FSAR are tests which were to be performed at the beginning of Unit 2 Cycle 1. Not all of these tests need to be performed on a reload cycle. Those FSAR tests not required to be performed for Unit 2 Cycle 8 are addressed in the Unit 2 Cycle 1 Startup Report. The tests that were performed for Unit 2 Cycle 8 are addressed in detail in the attached Startup Report.

IE26 /1

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Sincerely,

M. P. Alexich
Vice President

ldp

cc: D. H. Williams, Jr.
A. A. Blind
J. R. Padgett
G. Charnoff
A. B. Davis - Region III
NRC Resident Inspector - Bridgman
NFEM Section Chief

ATTACHMENT TO AEP:NRC:10710

DONALD C. COOK NUCLEAR PLANT

UNIT 2, CYCLE 8 STARTUP REPORT

DONALD C. COOK

UNIT 2 CYCLE 7/8

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

February 11, 1991

TABLE OF CONTENTS

I. INTRODUCTION

The Unit 2 Cycle 7/8 outage began with shutdown of the reactor on June 30, 1990 after six days of power coastdown. The total Cycle 7 burnup was 17,854.7 MWD/MTU. Cycle 8 began with initial criticality on November 8, 1990 and 100% reactor thermal power (RTP) was reached on November 21, 1990. Startup testing was completed on November 23, 1990 following the analysis of Flux Map 208-06.

The fuel movement sequence for this outage was accomplished by using the total core unload and reload technique. The Unit 2 Cycle 8 core consists of 116 Advanced Nuclear Fuels (ANF) and 77 Westinghouse VANTAGE-5 fuel assemblies and is the transitional cycle to VANTAGE-5 fuel. The unload took place over July 26-29, 1990 and the reload occurred over September 2-4, 1990. The refueling campaign had no fuel handling sequence errors; a video inspection of the core confirmed proper fuel loading as compared to design. In addition, the binocular inspection performed during unload/reload revealed no structural damage to any fuel assemblies.

The approach to criticality began with the withdrawal of the Shutdown Banks at 1521 hours on November 7, 1990. Rod Position Indicators (RPIs) and Bank Demand Counter differences caused a minor delay in the approach to criticality. Dilution began at 2010 hours, and the reactor was critical on November 8, 1990 at 0236 hours.

After the reactor reached steady state conditions, data was taken to determine the Point of Adding Nuclear Heat, Zero Power Physics Testing Range, and All Rods Out (ARO) Critical Boron Concentration. The Zero Power Physics Testing Program began at 0340 hours on November 8, 1990; the program included ARO Isothermal Temperature Coefficient (ITC) Determination and Rod Worth Measurements. The ARO ITC was +2.31 pcm/°F as compared to a design value of +1.54 pcm/°F. The Moderator Temperature Coefficient (MTC) calculated from the ARO ITC was +4.89 pcm/°F which is less than the Technical Specification Limit of +5.0 pcm/°F below 70% RTP. All rod worth and boron endpoint measurements compared favorably with design values and excess shutdown margin was verified.

Power Ascension Testing began on November 9, 1990, and was completed on November 23, 1990. The testing program included Flux Maps at approximately 34%, 47%, 66%, 88%, 86%, and 99% RTP. The testing also included an Incore/Excore Cross Calibration using the One Point Methodology at approximately 47% RTP. Power Ascension Testing went smoothly, with the hot channel factors calculated from the incore flux maps being within the required Technical Specification Limits. The 86% RTP Flux Map was taken due to the indication of a Nuclear Instrumentation System Quadrant Power Tilt Ratio in excess of 2%. The Quadrant Power Tilt Ratio obtained from the analysis of this Flux Map satisfied the Technical Specification Limit. The power range drawers were recalibrated to remove the instrument tilt.

In general, all startup tests were relatively routine. The tests were conducted in a timely and expedient manner and resulted in accurate startup information. The collected data and test results compared well with design predictions and satisfied all of the test's Acceptance Criteria and all Technical Specification Limits.

As stated in Section 6.9.1.2 of Unit 2 Technical Specifications, the tests identified in the FSAR shall be addressed in the Startup Report. The tests in the FSAR are tests which were to be performed at the beginning of Unit 2 Cycle 1. Not all of these tests need to be performed on a reload cycle. Those FSAR tests not required to be performed on this reload core are addressed in the Unit 2 Cycle 1 Startup Report. The tests that were performed on this reload core are addressed in detail in this report.

Section II

FUEL RELATED ACTIVITIES

II.1 CORE UNLOAD/RELOAD AND FUEL INSPECTION

The Unit 2 Cycle 7 core unload sequence began at 1035 hours on July 26, 1990 and was completed at 0027 hours on July 29, 1990. The Cycle 8 reload sequence commenced on September 2, 1990, at 0305 hours and was completed on September 4, 1990, at 0649 hours.

The Unit 2 Cycle 7 core consisted of 26 Region 7, 87 Region 8, and 80 Region 9 assemblies. Regions 7, 8, and 9 were all ANF fuel assemblies. The new core loading of Cycle 8 consists of 8 Region 7, 28 Region 8, 80 Region 9, 36 Region 10A, 40 Region 10B, and 1 Region 10C assemblies. Twenty-six (26) Region 7 and 59 Region 8 ANF assemblies were replaced by 77 fresh Region 10 Westinghouse VANTAGE-5 assemblies and 8 twice-burned Region 7 ANF assemblies which were last used during Cycle 6. A more detailed description of the VANTAGE-5 fuel assembly is provided in Appendix II.1. Core loading diagrams for Unit 2 Cycle 7 and Cycle 8 are shown in Figure II.1 and II.2, respectively. Figures II.3 and II.4 illustrate the differences of the core designs for both cycles.

No fuel handling sequence errors (ie., assemblies put in the wrong location) were reported during this campaign, and only 19 Fuel Movement Deviation Reports (FMDRs) were generated. The majority of the FMDRs were to resolve difficulties with fuel assembly bowing and the transfer system (upender). For the first time, a "horseshoe" was used at Cook Plant during the reload. The horseshoe proved its worth by greatly reducing the number of fuel assembly "boxes" necessary to position assemblies in their final core location.

A binocular inspection of the fuel took place at the Spent Fuel Pit (SFP) area for the unload. During the reload, however, the binocular inspection took place at the SFP as well as inside containment. As each assembly was unloaded, all four sides were inspected for any structural damage or abnormalities. A final check for damage was completed as the assemblies were reloaded into the core. The binocular inspection of the fuel assemblies revealed no structural damage or abnormalities.

At the conclusion of the core reload, a video inspection of the core and a visual inspection of the SFP were performed, as required by 12 THP 4040 SNM.302 and SNM.304, respectively. No discrepancies were found during either inspection.

FIGURE II.1

D.C. Cook Unit 2 Cycle 7 Core Map

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
15					233 T56	SS09 V06	222 V11	254 T28	219 V17	247 V30	252 T75				
14			206 T33	R155 U27	X4PV05 V21	R128 U83	16PV35 V03	R135 U10	16PV17 V18	R147 U22	X4PV18 V29	R118 U85	268 T20		
13		251 T03	X4PV08 V56	12PV01 V74	R106 U33	16PV03 V28	R105 U70	SS10 U11	R138 U67	16PV06 V33	R127 U63	12PV07 V79	X4PV16 V59	241 T16	
12		R153 U80	12PV06 V78	R122 U40	16PV09 V39	265 U54	16PV23 V51	R110 U25	16PV38 V62	273 U71	16PV30 V45	R119 U37	12PV03 V75	R103 U88	
11	317 T60	X4PV15 V50	R40 U61	16PV24 V48	270 T80	16PV28 V66	218 U49	20PV02 V43	207 U42	16PV40 V72	209 T73	16PV10 V41	R116 U81	X4PV06 V23	258 T69
10	313 V10	R114 U28	16PV08 V37	283 U64	16PV32 V70	R15 T84	224 U15	R144 U26	266 U03	R01 T91	16PV27 V67	271 U58	16PV05 V31	R18 U77	217 V63
9	256 V15	16PV01 V25	R108 U47	16PV20 V58	257 U02	263 U05	246 U05	240 U36	314 U09	215 U06	264 U62	16PV18 V57	R104 U75	16PV14 V05	245 V12
8	216 U53	R109 U18	235 U19	R137 U55	20PV04 V53	R126 U23	253 U31	R113 U51	210 U35	R136 U24	20PV03 V52	R112 U60	228 U21	R143 U20	234 U57
7	260 V13	16PV15 V08	R141 U66	16PV25 V55	243 U50	230 U04	267 U08	250 U34	249 U12	202 U07	237 U43	16PV12 V60	R146 U69	16PV21 V24	220 V14
6	223 V40	R129 U17	16PV19 V32	231 U56	16PV29 V68	R117 T82	293 U16	R134 U29	244 U01	R111 T52	16PV31 V69	225 U73	16PV07 V36	R140 U32	203 V07
5	232 T53	X4PV07 V26	R115 U84	16PV33 V42	307 T74	16PV39 V71	269 U48	20PV01 V20	211 U52	16PV26 V65	275 T77	16PV13 V47	R121 U79	X4PV09 V44	214 T68
4		R131 U87	12PV04 V76	R145 U30	16PV11 V46	302 U74	16PV22 V61	R123 U44	16PV16 V49	204 U72	16PV37 V38	R130 U41	12PV05 V77	R125 U86	
3		262 T71	X4PV12 V64	12PV08 V80	R124 U38	16PV36 V35	R139 U46	SS07 U14	R133 U45	16PV04 V27	R102 U82	12PV02 V73	X4PV13 V54	277 T22	
2			242 T24	R150 U59	X4PV20 V34	R101 U78	16PV02 V22	R120 U13	16PV34 V02	R142 U68	X4PV14 V04	R107 U76	255 T32		
1					226 T79	248 V19	238 V16	292 T11	261 V09	SS08 V01	205 T64				

234 <--- INSERT NO.

Y56 <--- FUEL ASSY. NO.

<---- NORTH

0

270 --------- 90

180

<--- FUEL ASSY. ORIENTATION

FIGURE II.2

D.C. Cook Unit 2 Cycle 8 Core Map

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
15					201 U76	210 U83	231 V22	258 V73	254 V02	227 U22	212 U59				
14			225 T30	R139 W03	347D W49	R128 W52	339D W15	R104 W14	332D W17	R134 W60	369D W53	R126 W08	228 T25		
13		257 T19	313 V58	374D W57	R138 V06	222 V11	R102 V33	SS10 V74	R131 V28	219 V17	R124 V30	366D W58	253 V57	272 T17	
12		R109 W01	340D W41	R112 V49	378D W27	266 V48	356D W73	R113 U07	342D W61	205 V41	370D W26	R143 V61	334D W42	R115 W07	
11	204 U86	372D W54	R120 V10	358D W29	246 U05	379D W67	230 V66	349D W28	271 V72	380D W65	314 U09	363D W23	R135 V63	352D W37	274 U87
10	275 U32	R136 W45	256 V15	217 V39	361D W70	R130 U01	283 V04	R114 V43	292 V34	R106 U16	350D W69	233 V45	245 V12	R146 W43	251 U17
9	268 V05	348D W16	R18 V32	365D W72	242 V70	206 V44	247 V56	SS16D W36	224 V59	203 V26	250 V67	360D W71	R125 V36	338D W11	263 V25
8	244 V77	R142 W20	SS09 V78	R117 U04	368D W32	R155 V53	336D W35	R133 W77	335D W34	R123 V52	344D W21	R40 U06	SS08 V75	R105 W09	232 V76
7	207 V24	383D W19	R150 V37	353D W64	202 V68	255 V23	214 V64	SS15D W33	317 V54	240 V50	252 V69	371D W76	R129 V31	351D W18	264 V08
6	218 U77	R147 W48	260 V13	248 V46	375D W62	R127 U03	215 V29	R140 V20	273 V21	R122 U15	359D W75	226 V38	220 V14	R111 W47	269 U28
5	211 U88	331D W51	R145 V40	373D W31	267 U08	346D W74	262 V71	381D W30	234 V65	364D W68	249 U12	357D W25	R153 V07	355D W50	277 U80
4		R137 W06	341D W40	R144 V62	382D W24	223 V42	377D W66	R141 U02	376D W63	209 V47	384D W22	R103 V51	354D W59	R118 W05	
3		208 T27	293 V55	343D W38	R107 V19	238 V16	R119 V27	SS07 V80	R110 V35	261 V09	R101 V01	367D W56	237 V60	243 T36	
2			307 T26	R154 W04	345D W55	R01 W46	333D W13	R121 W12	362D W10	R116 W44	337D W39	R108 W02	270 T31		
1					241 U85	221 U78	216 V03	302 V79	235 V18	288 U68	265 U27				

234 <--- INSERT NO.
Y56 <--- FUEL ASSY. NO.

<---- NORTH

0
270 --------- 90
180

<--- FUEL ASSY. ORIENTATION

Figure II.3
D. C. Cook Unit 2 Cycle 7
Core Plan

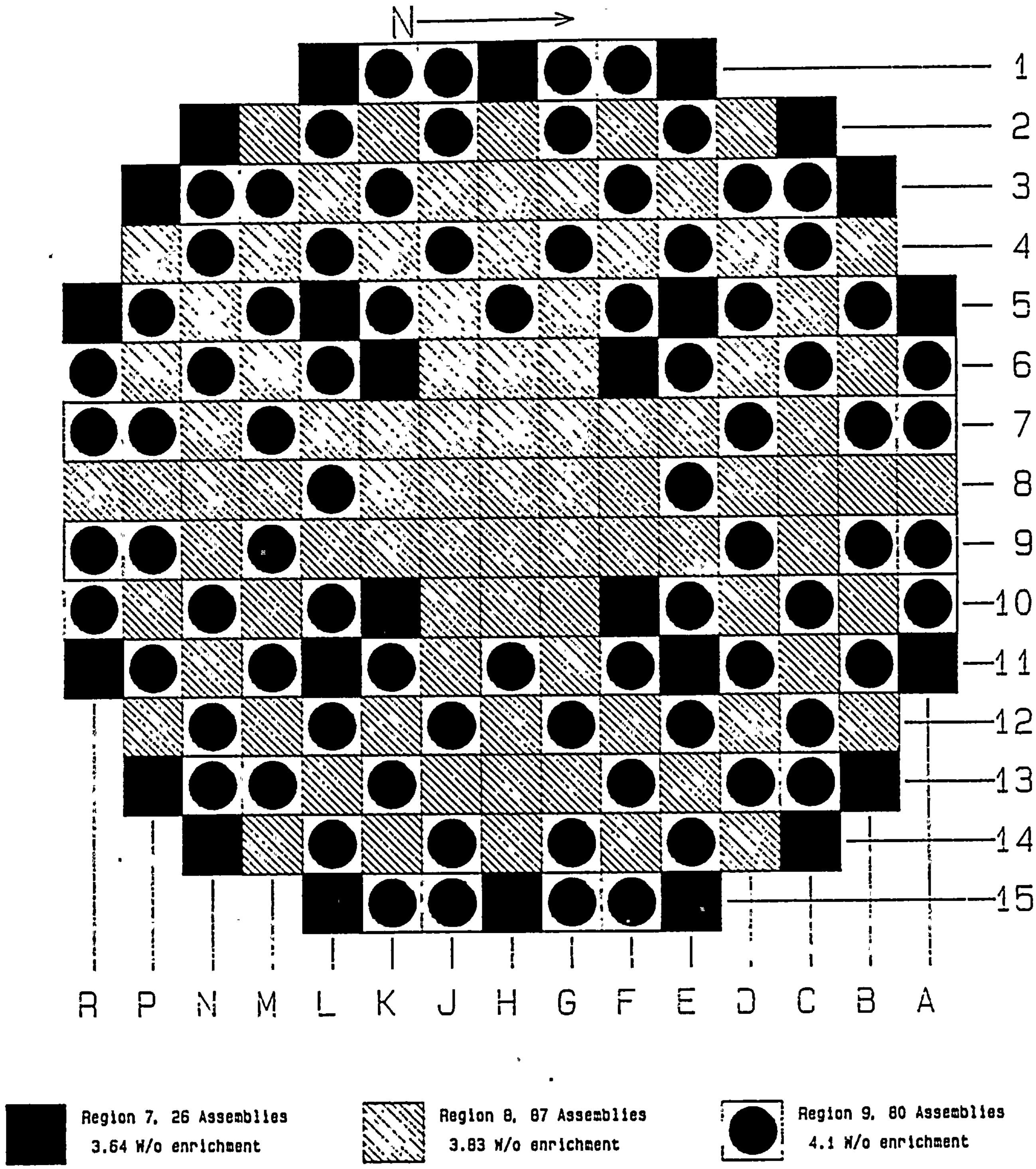
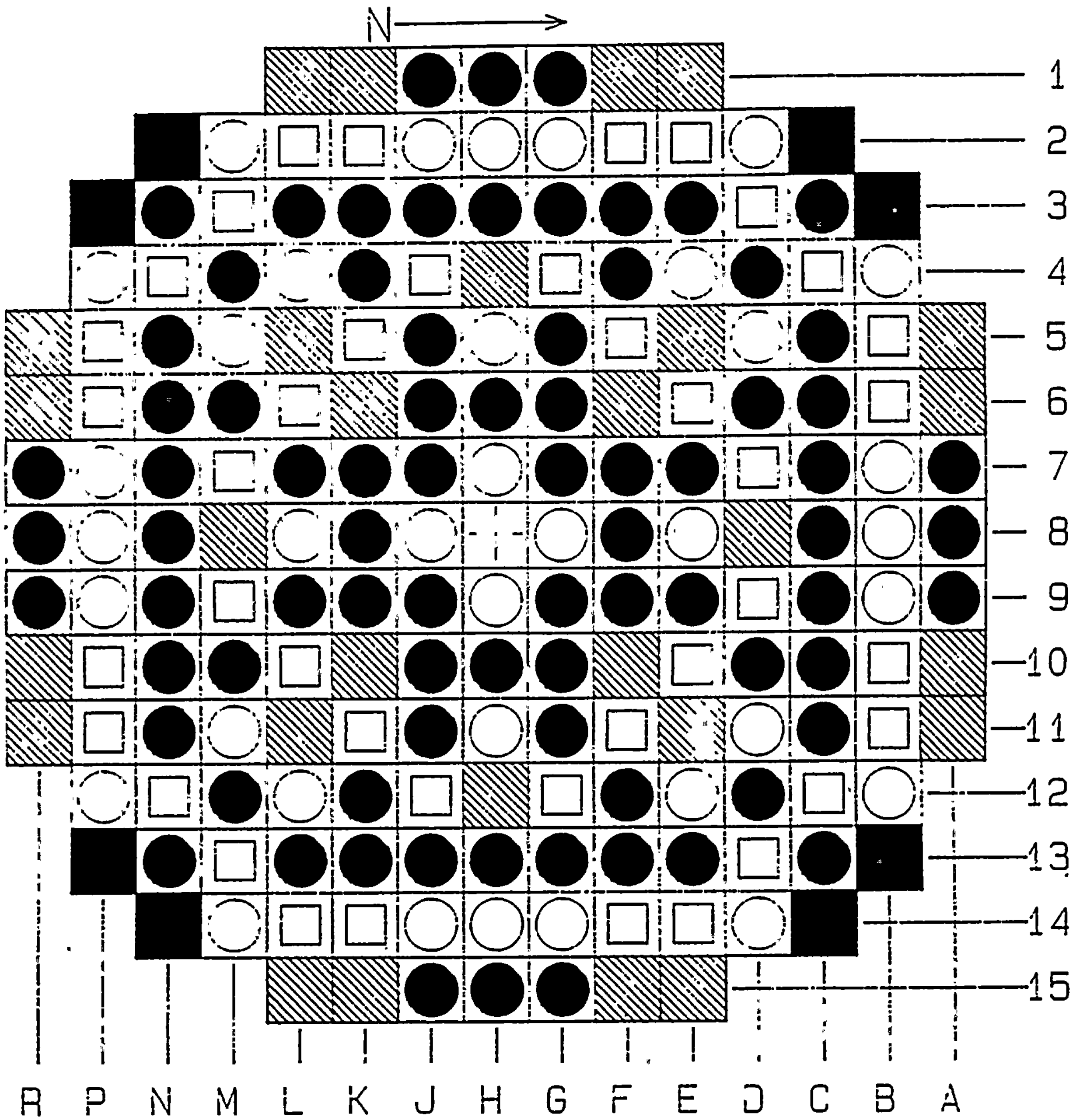


Figure II.4
D. C. Cook Unit 2 Cycle 8
Core Plan



Region 7, 8 Assemblies
3.64 W/o enrichment

Region 8, 28 Assemblies
3.83 W/o enrichment

Region 9, 80 Assemblies
4.11 W/o enrichment

Region 10A, 36 Assemblies
3.602 W/o enrichment

Region 10B, 40 Assemblies
4.2 W/o enrichment

Region 10C, 1 Assembly
1.49 W/o enrichment

WESTINGHOUSE 17 × 17 VANTAGE-5 FUEL ASSEMBLY

Unit 2 Cycle 8 core presently consists of 77 Westinghouse and 116 ANF fuel assemblies. Westinghouse's safety evaluation and analyses concluded that the VANTAGE-5 and ANF designs are mechanically and hydraulically compatible with each other.

Table II.1 is a list of the design features for the ANF and Westinghouse fuel assembly. Figure II.5 is a schematic of the two assembly's dimensions. The major differences in the Westinghouse VANTAGE-5 design as compared to the ANF fuel assembly consist of the following:

- **Integral Fuel Burnable Absorber (IFBA):**

  The IFBA is a fuel pellet that has its surface coated with a thin layer of zirconium diboride. IFBAs are used to control power peaking factors and reduce the beginning of cycle moderator temperature coefficient.

- **Intermediate Flow Mixer (IFM) Grids:**

  The VANTAGE-5 design incorporates three Zircaloy-4 Intermediate Flow Mixing (IFM) grids. The three IFM grids are located in the uppermost spans between the three Zircaloy-4 structural grids. Increased DNB margin is realized by the mid-span flow mixing in the hottest fuel assembly spans. The increased DNB margin permits an increase in the design basis $F^N_{\Delta H}$ and F_Q

- **Reconstitutable Top Nozzle/Extended Burnup Capability:**

  The VANTAGE-5 top nozzle consists of a design feature which facilitates easy removal of the nozzle from the fuel assembly for fuel rod examination or replacement. Changes in the design of the top and bottom nozzles have allowed for a slightly longer fuel rod. The increased length extends the burnup margins by providing additional plenum space for fission gas accumulation.

- **Axial Blankets:**

  The axial blankets are a nominal six inch stack of natural UO_2 pellets at each end of the fuel stack. The purpose of the blankets are to reduce neutron leakage and improve uranium utilization.

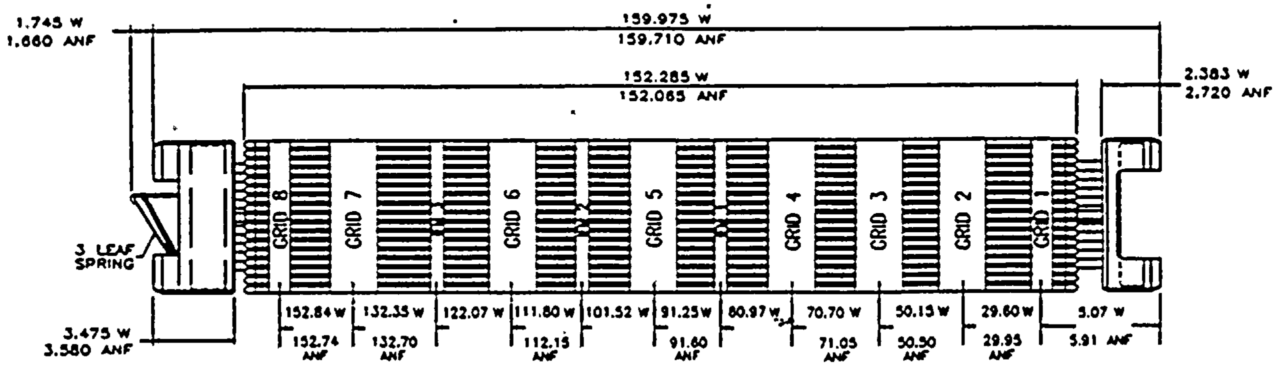
TABLE II.1

Comparison of Westinghouse VANTAGE-5 and ANF Assembly Design

Parameter	VANTAGE-5 Design	ANF Design
Fuel Assy Length, in	159.975	159.710
Fuel Rod Length, in	152.285	152.065
Assy Envelope (width), in	8.426	8.426
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in	0.496	0.496
Number of Fuel Rods/Assy	264	264
Guide Thimble Tubes/Assy	24	24
Instrumentation Tube/Assy	1	1
Fuel Tube Material	Zircaloy-4	Zircaloy-4
Fuel Rod Clad OD, in	0.360	0.360
Fuel Rod Clad Thickness, in	0.0225	0.0250
Fuel/Clad Radial Gap, mil	3.1	3.5
Fuel Pellet Diameter, in	0.3088	0.3030
Fuel Pellet Length:		
Enriched Fuel, in	0.370	0.348
Unenriched Fuel, in	0.500	N/A
Guide Thimble Material	Zircaloy-4	Zircaloy-4
Guide Thimble OD, in (above dashpot)	0.474	0.480

<u>FIGURE II.5</u>

COMPARISON OF WESTINGHOUSE VANTAGE-5 and ANF FUEL ASSEMBLY DIMENSIONS



W – WESTINGHOUSE 17x17 VANTAGE 5 FUEL ASSEMBLY DIMENSION+
ANF – ADVANCED NUCLEAR FUEL 17x17 FUEL ASSEMBLY DIMENSION+

<u>ANF GRID WIDTH</u>
- 2.50 (valley to valley)
- 2.96 (peak to peak)

<u>WESTINGHOUSE GRID WIDTH</u>*
- top and bottom grid, 1.522
- mid grids, 2.25
- IFM grids, 0.475

---

* based on inner strap width
+ dimensions are in inches (nominal)

II.2 ULTRASONIC EXAMINATION OF IRRADIATED FUEL ASSEMBLIES

Advanced Nuclear Fuels (ANF) provided fuel Ultrasonic Testing (UT) services. The UT began at 1100 hours on August 17, 1990 and was completed at 1230 hours on August 19, 1990. The 116 irradiated fuel assemblies designated for the Cycle 8 core were tested. Predictions of two to five failed fuel pins were made during the Cycle, with an end of Cycle prediction of four failures. The UT campaign revealed five failed fuel rods; all were found in assembly T-24. No core redesign was necessary based on the results of the UT campaign.

The UT System works by a probe transceiver sending a high frequency sound wave into a fuel pin and measuring the strength of the returning signal, or "ring back". A fuel pin can be determined to have water in it by monitoring the relative strength of the ring back. A fuel assembly is tested from two sides; however, all four sides of an assembly may be tested if an indication of a failed rod exists.

During the UT campaign, fuel assemblies V16, V13, V12, and V70 had suspicious indications of failed rods due to their low returning signals. As a result, these assemblies were retested later in the campaign from all four sides at slightly different elevations and they were determined to be sound. The initial results of the low returning signals were believed to be caused by poor probe alignment. The final UT results showed no failures in all 116 ANF fuel assemblies.

In addition to the 116 reload assemblies, the discharged assembly T-24 was also tested. Failed rods in the assembly were detected in locations G01, G03, K10, H16, and H17. This assembly had also been tested during the Cycle 6/7 UT campaign in 1988, and rods G01, K10, H16, and H17 were found to be failed at that time. The UT results on assembly T-24 were close to the end of Cycle 7 prediction. Visual inspections of two sides of assembly T-24 were performed and video taped with the underwater camera mounted on the UT system. The failed rods on the periphery of assembly T-24 showed signs of hydride blisters.

II.3 RCCA EXAMINATION

All 53 Rod Cluster Control Assemblies (RCCAs) of Unit 2 were inspected during the Cycle 7/8 outage by Westinghouse and Echoram personnel. There were 3.5 days of delay prior to the start of the RCCA examination due to repairs and tests on the Spent Fuel Handling Crane. Eddy current inspection of the Unit 2 RCCAs began at 1940 hours on August 5, 1990, and was completed at 0718 hours on August 8, 1990.

The RCCA eddy current inspection equipment consists of an eddy current coil assembly guide fixture and a data acquisition/analysis system. The eddy current coil assembly guide fixture is placed on top of the spent fuel pool racks. The data acquisition/analysis system is set up on the operating deck of the spent fuel pool.

The eddy current inspection is based on the principle of Electromagnetic Induction. When a metal object is placed in a varying electromagnetic field, electromagnetic currents are induced in the object which tend to oppose the external field. The strength of these induced electromagnetic fields depends on the amount of magnetic permeability, conductivity, and shape of the metal object to be tested. The strength of the induced fields causes the electrical impedance of the exciting electromagnetic field source to change. If an anomaly such as a reduction in cross-section or a crack enters the electromagnetic field, the impedance of the system would change. Thus, anomalies such as areas with worn metal or cracks in the cladding can be measured by lowering a RCCA through the eddy current coil assembly guide fixture.

The results showed normal wear patterns on the rodlets near the guide card region and longitudinal cracks on several rod tips. These wear patterns were expected based on tests at other plants utilizing Westinghouse RCCAs and will not affect the absorbing capability of the control rod. The results also indicated that the wear had penetrated the cladding of rodlet E5 of control rod R15. A new control rod was used to replace R15 prior to reloading the Cycle 8 core.

Section III

FLUX MAPPING SYSTEM ACTIVITIES

III.1 THIMBLE TUBE CLEANING

APEX Technologies provided services to clean the Unit 2 incore thimble tubes. The cleaning commenced with demineralized water flushing and vacuum drying of the thimble tubes on July 14, 1990. Thimble cleaning was completed on July 21, 1990.

Problems were encountered during the cleaning due to the C-7 thimble leak and ten-path indexer leaks. The indexer leaks allowed oil to be released from their reservoirs, which settled on the bottom of the ten-path housings. When the C-7 thimble leak occurred, oil was flushed from the bottom of the ten-paths and down into the thimble tubes. Fifteen thimble tubes were completely plugged. Also, the remaining thimbles showed signs of the oil and water mixture. All thimble tubes were cleaned successfully and were verified passable by use of a test cable. The excessive amount of water also filled the five-path transfer boxes.

Apex assisted in pumping out the water from the five-path transfer boxes. They also cleaned the interconnecting tubing runs from the ten-paths to the seal table and from the five-paths to the ten-paths.

The general arrangement of the reactor, thimbles, and seal table is presented in Figure III.1. Figure III.2 shows the equipment layout for demineralized water flushing and air drying of thimble bores, and Figure III.3 shows the isolation valve and thimble orientation of the seal table.

FIGUR[illegible]1

General Arrangement of Reactor, Thimbles and Seal Table Area

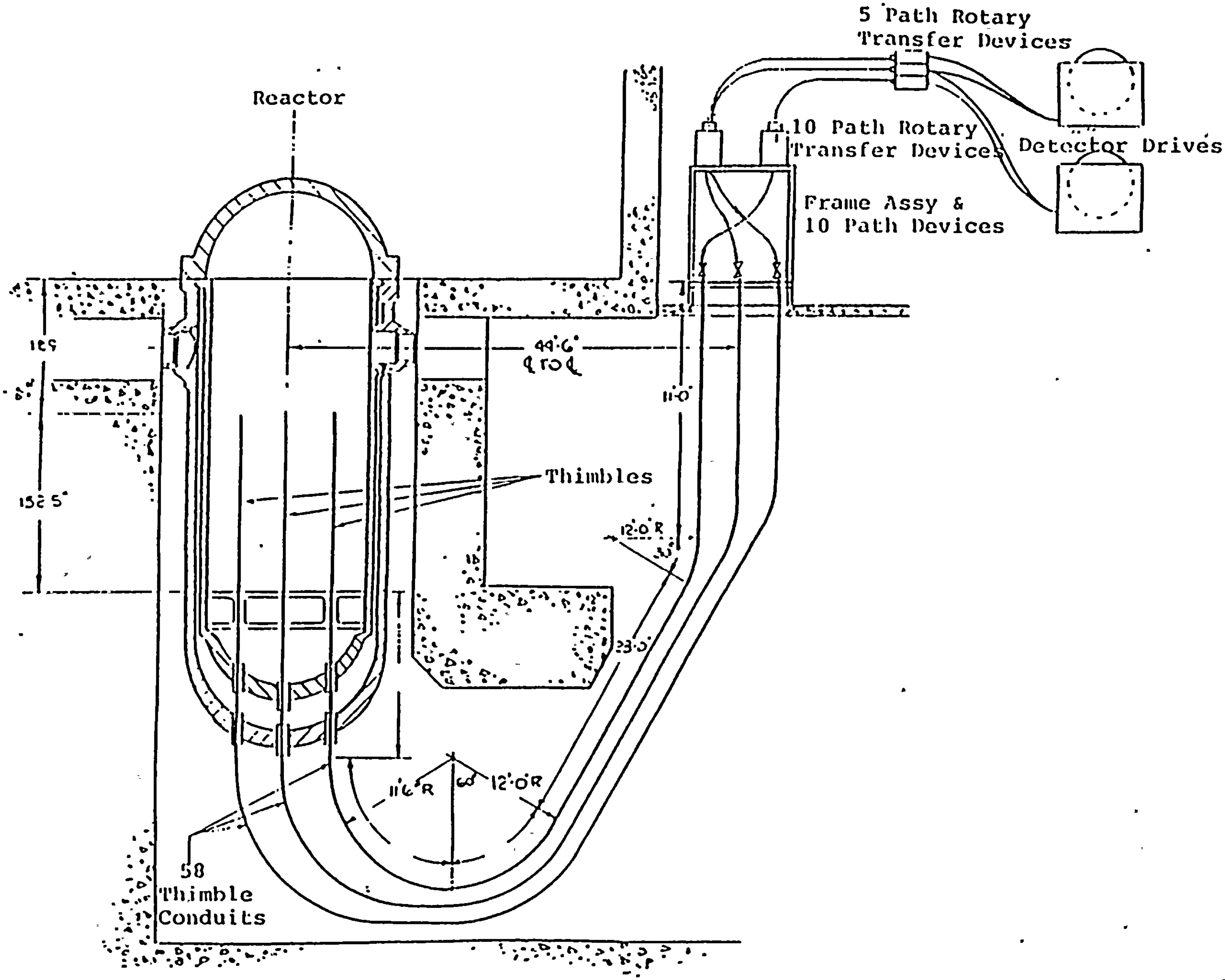


FIGURE III.2

Demineralized Water Flush and Air Drying of Thimble Bore

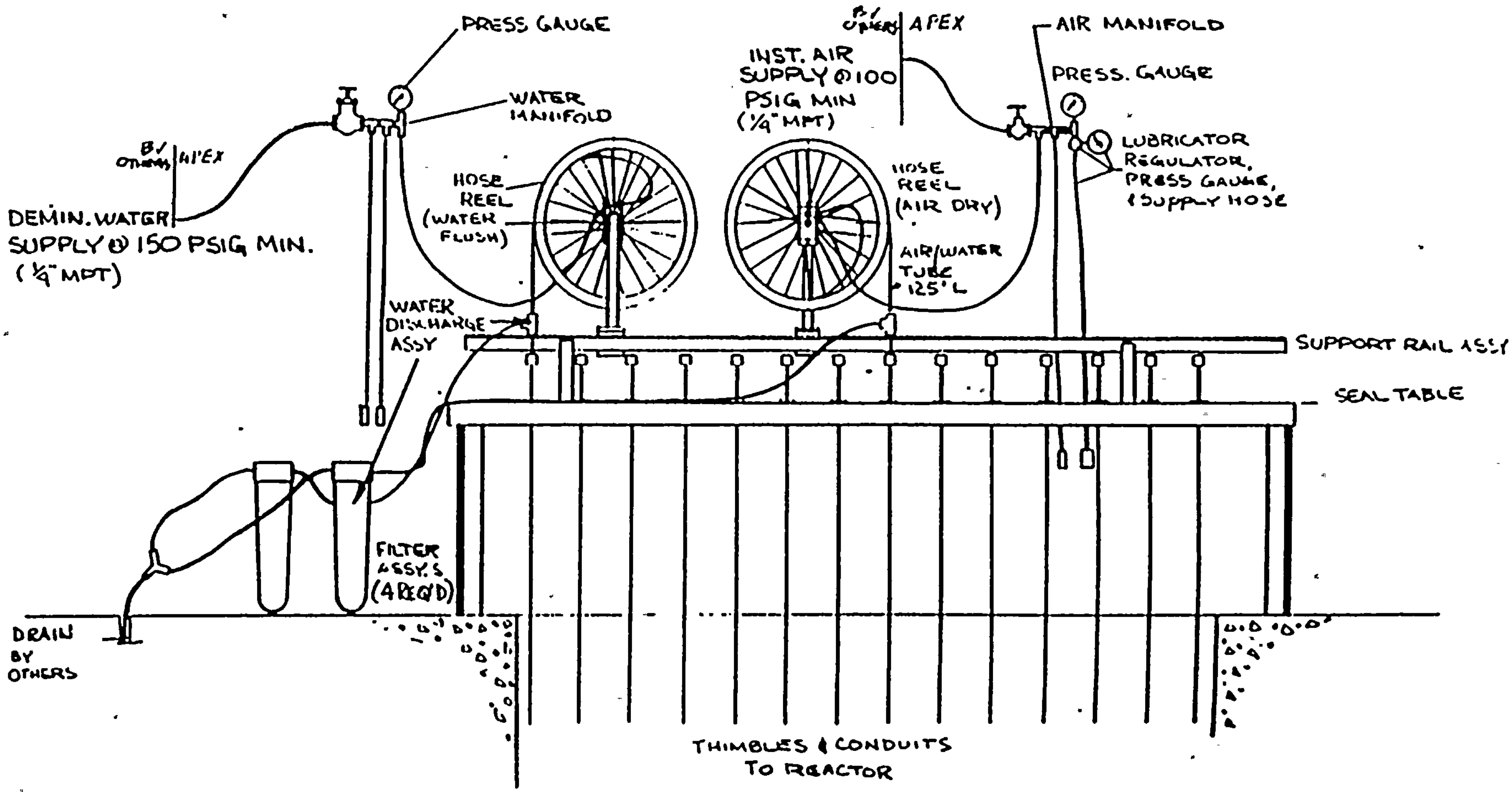
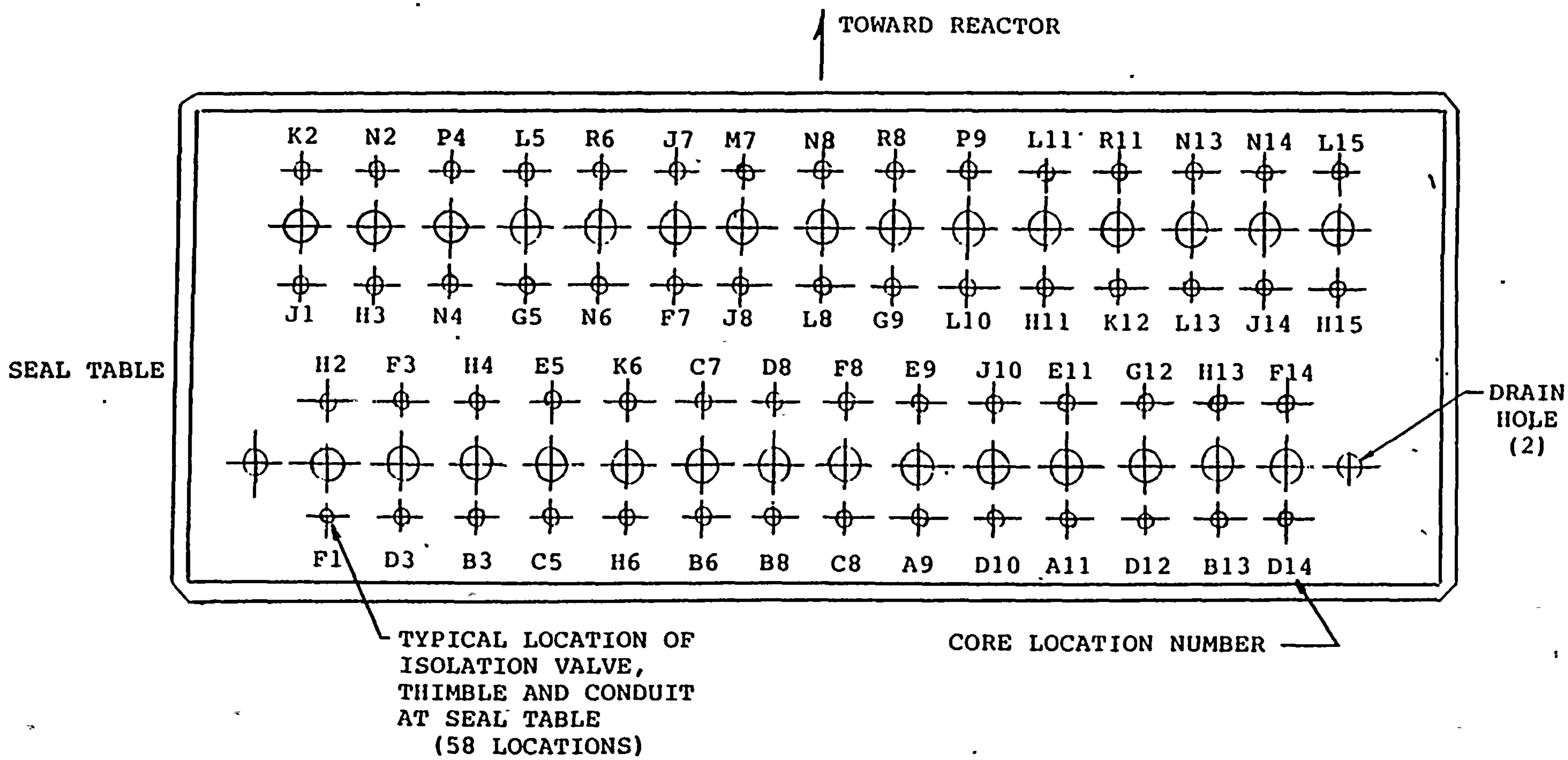


FIGURE III.3

Isolation Valve and Thimble Orientation at Seal Table



III.2 EDDY CURRENT EXAMINATION OF INCORE THIMBLE TUBES

Cramer and Lindell Engineers, Inc. provided services for the Eddy Current Examination of Unit 2's 58 Incore Thimble Tubes. The Eddy Current exam of the Incore Thimble Tubes commenced at approximately 1530 hours on July 18, 1990 and was completed at approximately 0026 hours on July 19, 1990. Bubble Hoods were used during the examination due to the need for respiratory protection and effective communication during the exam. No major delays were encountered.

The setup of the eddy current equipment was quite simple. An eddy current probe that is welded to a helix cable was manually fed to the top each thimble. During withdrawal of the probe, eddy current signals were passed to the data aquisition station located just outside of containment. Eddy current data of all 58 thimbles was collected and analyzed by the vendor.

Twenty-nine thimble tubes were determined to have greater than 30% wall loss. Nine of those thimble tubes had a wall loss of greater than 60%, including the leaking thimble tube, C-7. The majority of the wear was determined to be located at the fuel assembly bottom nozzle. Due to the severity of the thimble tube wear, ten thimble tubes were replaced and 19 thimbles tubes were repositioned.

III.3 THIMBLE TUBE REPLACEMENT PROJECT

The Thimble Tube Replacement Project was performed by Apex Technologies in three different phases at the Seal Table and Reactor Cavity. An additional activity of preparing the new thimbles was completed on the Turbine Deck. Ten thimbles were replaced, and 19 thimbles were repositioned. The project was completed in the order as follows:

- **Phase I: Seal Table Preparation with Reactor Cavity Drained**

  Phase I had two objectives. The first objective was to cut off approximately 15 feet of the "cold" (non-irradiated) end of the thimbles to be replaced. The second objective was to prepare for the removal of the thimbles through the Reactor Cavity. This was accomplished with slide seal valve assemblies which were installed at the seal table to act as low pressure seals during the removal of the thimbles. The function of the slide seal valve assembly was to allow personnel at the seal table to push the thimbles up through the lower core plate to personnel stationed at the Reactor Cavity while the cavity was filled.

  Phase I was started on August 8, 1990 and was completed in approximately two hours. There were no major delays.

- **New Thimble Tube Preparation - Cut and Clean**

  This activity was completed on the Unit 1 Turbine Deck. The new thimbles were unpackaged, measured, cut to length, and cleaned. The same cleaning technique was used as if the thimbles were installed in the reactor. The thimbles were first flushed with demineralized water and vacuum dried. A fine coating of neolube was then applied and allowed to air dry.

  This phase was started on August 17, 1990 and was completed on August 18, 1990. Again, there were no major delays.

- **Phase II: Removal of Spent Thimbles from Reactor with the Reactor Cavity Flooded**

  The second phase of the project involved the actual removal of the thimbles from the Reactor Cavity. The basic procedure for the removal involved pushing the thimbles up through the core plate and grappling onto them. After this was accomplished, the "hot end" was

moved underwater to the upender and the "cold end" was grappled. The "cold end" was then taken out of the water and cut into pieces until the radiation fields would not permit anymore cutting above water. At this point the "hot end" was cut underwater and placed into a canister (two were used) which was positioned in the upender. The canisters were then moved to the Spent Fuel Pool for storage. The leaking thimble, C-7, and the next most worn thimble, A-9, were placed in a special canister for later retrieval and analysis. The wear scars on those two thimbles were video taped with an underwater camera.

Phase II was started on August 26, 1990 and was completed on August 29, 1990. Major delays were encountered in this phase mainly due to extra protective measures required for the radiological conditions.

- **Phase III: New Thimble Installation and Thimble Repositioning**

The final phase involved the installation of the 10 new thimbles and the repositioning of 19 others. Also, the guide tubes of the new thimbles were vacuum cleaned.

The new thimbles were passed through a containment penetration and loaded into their respective locations. New high pressure seals were installed on the thimbles and connected to the seal table.

The repositioning involved removing the old high pressure seal ferrule sets and cutting off a selected amount of thimble tube, to move the wear scare away from its original position. New high pressure seals were installed and the connections to the seal table were made. After the ten new thimbles were installed and the 19 others repositioned, an additional Eddy Current Examination was performed to provide baseline data for the next refueling outage.

Phase III work commenced on September 11, 1990 and was completed on September 12, 1990. Only minor delays were encountered during this phase.

III.4 Flux Mapping System Five-Path and Ten-Path Refurbishment

The Ten-path refurbishment of the Unit 2 Flux Mapping System was scheduled to be completed by the plant I&E Section to relieve the problem of indexer oil leaks. The indexers were originally installed with a plexiglass baseplate for refueling inspections of oil level. After years of operation, the plate and the sealing gasket began to show signs of wear and allowed small quantities of oil to leak from the indexer. The oil ran down the indexer and onto the small micro-switches which provide detector core location during a Flux Map. The oil made these switches sticky, and in some cases, give false indication of detector location on the core map. A replacement baseplate and gasket were installed to relieve the problem.

The inspection of the flux mapping system after the C-7 thimble leak showed that extensive repairs would have to be completed on the system. All 6 ten-paths had been flushed with water, and all 6 of the five-path had at least 5 to 6 inches of standing water inside them. Because the water damaged all electrical components inside of the five and ten-paths, they had to be replaced. Also, after further inspection by I&E, it was determined that a power supply would have to be replaced in Detector F's drive box.

The parts replaced during the refurbishment included:

- 12 Transfer Motors
- 12 Clutches
- 12 Baseplates and Gaskets
- 150 Micro-Switches
- 1 Power Supply
- 1 Volt Meter

The project started on July 13, 1990 with the ten-path removal off the support rack and was completed on October 6, 1990 with a successful system checkout.

IV. ROD DROP TESTING

Rod drop testing commenced at 0715 hours on October 13, 1990, with the briefing of all personnel involved in the test and was completed with the repeat drops of the slowest rod, H-14, at 1430 hours on October 13, 1990.

Rod drop times were all approximately the same, with the slowest drop time being 1.53 seconds to the dash pot for Rod H-14, and the fastest rod drop time being 1.34 seconds to the dash pot for Rod J-13. The slowest rod, H-14, was dropped four additional times to check for repeatability. The four drop times were 1.52, 1.50, 1.50, and 1.50 seconds.

The rod drop testing proceeded smoothly with no delays. The results and shapes of the rod drop traces were in agreement with the previous cycle and no discrepancies were identified. However, the rod drop times slightly increased for some of the rods. As expected, this was predominately seen in the VANTAGE-5 fuel assemblies, which have a smaller diameter guide tube than the ANF fuel assemblies. All rod drop times were well within Technical Specification 3.1.3.4 Limit of 2.7 seconds.

V. INITIAL CRITICALITY

Unit 2 Cycle 8 achieved Initial Criticality at 0236 hours on November 8, 1990. The All Rods Out Boron Concentration (C_B) was calculated to be 1654.9 ppm as compared to the design value of 1687 ppm and was well within the Acceptance Criteria of ± 75 ppm.

On October 22, 1990, Reactor Engineering Section Personnel were requested to start Low Power Physics Testing. At 1411 hours, **12 THP 6040 PER.357, "Initial Criticality, All Rods Out Boron Concentration and Nuclear Hearing Level", began with the briefing of Operations and Chemistry personnel. At 1435 hours withdrawal of Shutdown Bank "A" commenced followed by the remaining shutdown banks and control banks. With Control Bank "D" (CBD) at 194 steps, Reactor Engineering Section was notified to halt Low Power Physics Testing until the steam dump valves were completely repaired. The Shutdown and Control rods were reinserted to the bottom of the core and PER.357 was aborted.

The approach to criticality, again, began with the withdrawal of the Shutdown Banks at 1521 hours on November 7, 1990. Next, withdrawal of the Control Banks in overlap, as shown in Figure V.1, began at 1636 hours. Source Range Data was monitored throughout withdrawal, with the Control Banks stopped every 50 steps to plot the Inverse Count Rate Ratio (ICRR) as shown in Figure V.1. Withdrawal of the Control Banks to CBD at 194 steps (approximately 100 pcm of negative reactivity in the core) was completed at 1736 hours on November 7, 1990. On two rods, Rod Bank Demand Counter and Analog Rod Position Indicator differences of greater than 12 steps delayed dilution to criticality approximately two hours.

At 2010 hours, RCS dilution began with C_B approximately equal to 2077 ppm. Mode 2 was declared at 2342 hours with a boron concentration of 1807 ppm. At 0232 hours on November 8, 1990 dilution was stopped and subsequently, the Reactor was declared critical at 0236 hours. The Reactor's stable critical conditions were CBD at 185½ steps, flux level at 1×10^{-8} amps, and C_B equal to 1636 ppm. During the dilution to critical, ICRR vs. C_B, Primary Water, and Time were plotted and are shown in Figures V.2, V.3, and V.4, respectively.

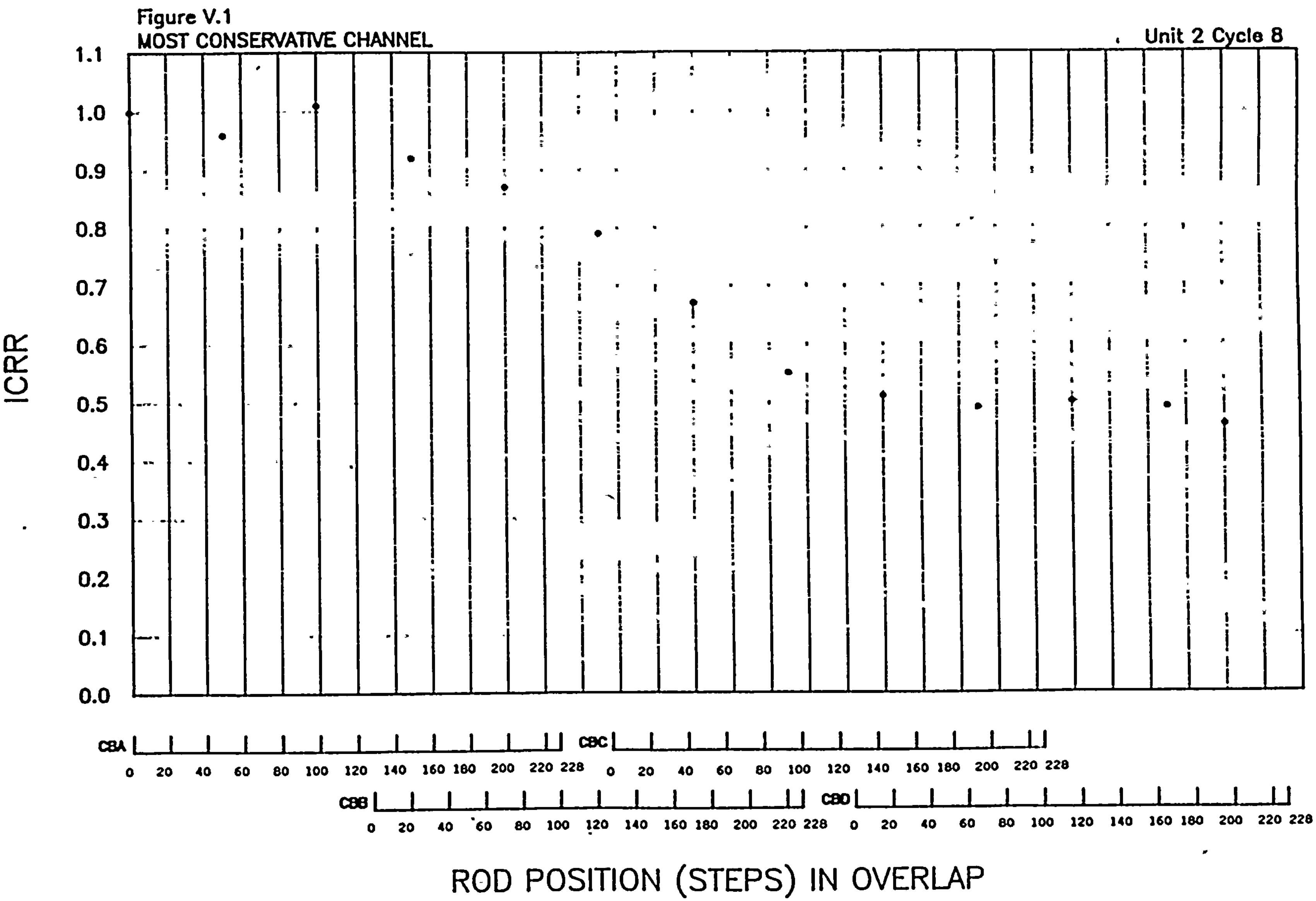
After the Reactor stabilized at 1×10^{-8} amps, data was obtained to determine the Nuclear Heating Level. The objective of determining Nuclear Heat was to set the upper limit of the Zero Power Testing Range. The Nuclear Heating Level was observed as a small decrease in reactivity due to feedbacks (Doppler) caused by an increasing neutron flux. At the same time, an increase in RCS temperature was also observed due to the addition of Nuclear Heat. The Nuclear Heating Level was determined to be 2.2×10^{-7} amps. Prior to withdrawing the Control Banks, γ background noise data was collected for determining the Zero Power Testing Range lower limit. Thus, the Zero Power Testing Range was set to obtain the least amount of error in reactivity measurements. The Zero Power Testing Range was determined to be 1×10^{-8} to 8×10^{-8} amps which was 1.17 decades above the γ background noise level (6.8×10^{-10}) and 0.44 decades below the Nuclear Heating Level. The Subcritical, Nuclear Heating Level, and Zero Power Testing Range test data is summarized in Table V.1.

TABLE V.1

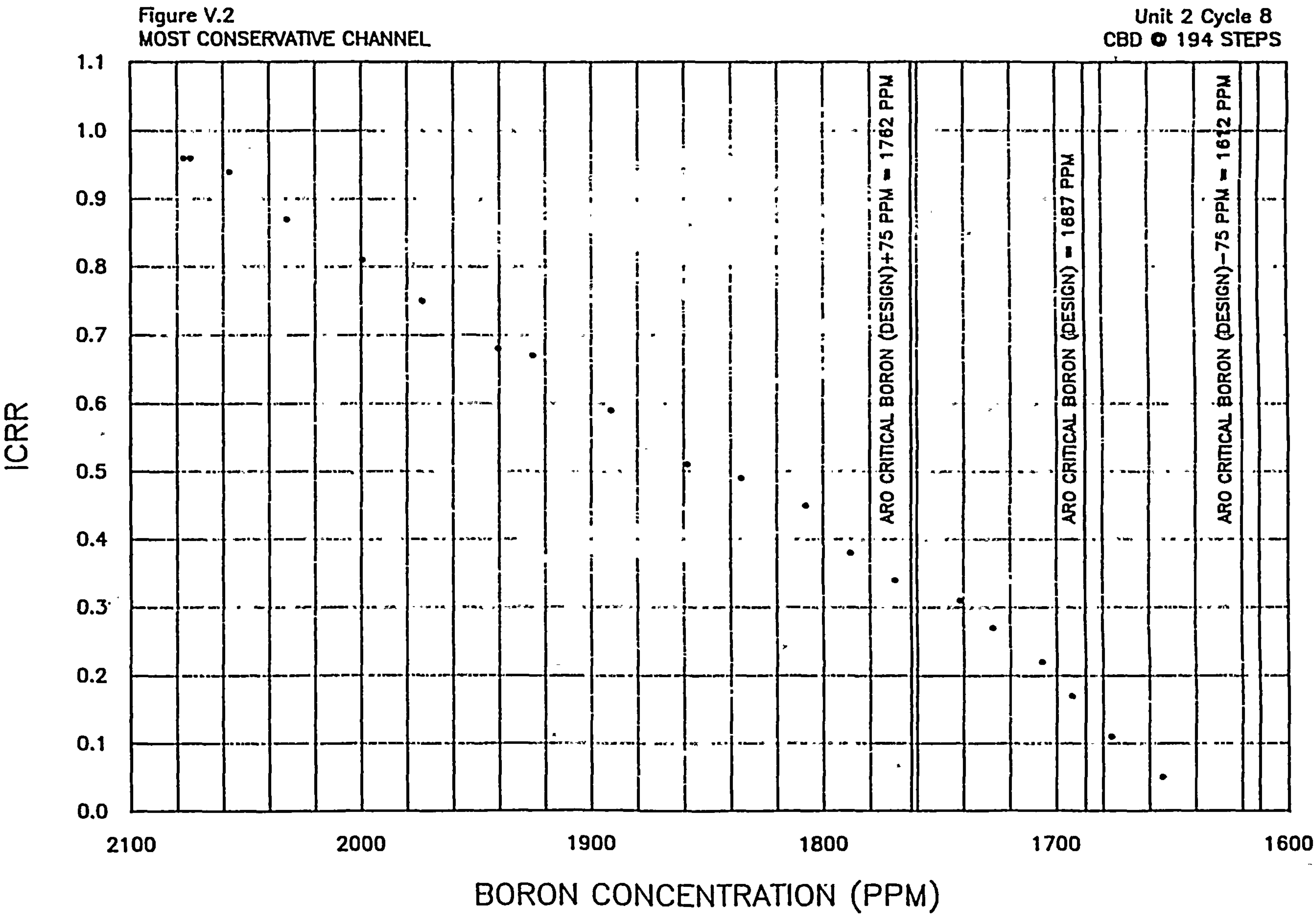
Subcritical Data **(Shutdown Banks Withdrawn, Control Banks Inserted)**
Induced current with 1000 V applied to Detector N41: Measured = 6.8×10^{-10} amps 90% value = 6.1×10^{-10} amps 50% value = 3.4×10^{-10} amps
Nuclear Heating Level
Flux Level = 2.2×10^{-7} amps
Zero Power Physics Testing Range†
Flux Range = 1×10^{-8} to 8×10^{-8} amps

† Due to the quality of the data obtained, no compensating current was required during Zero Power Physics Testing.

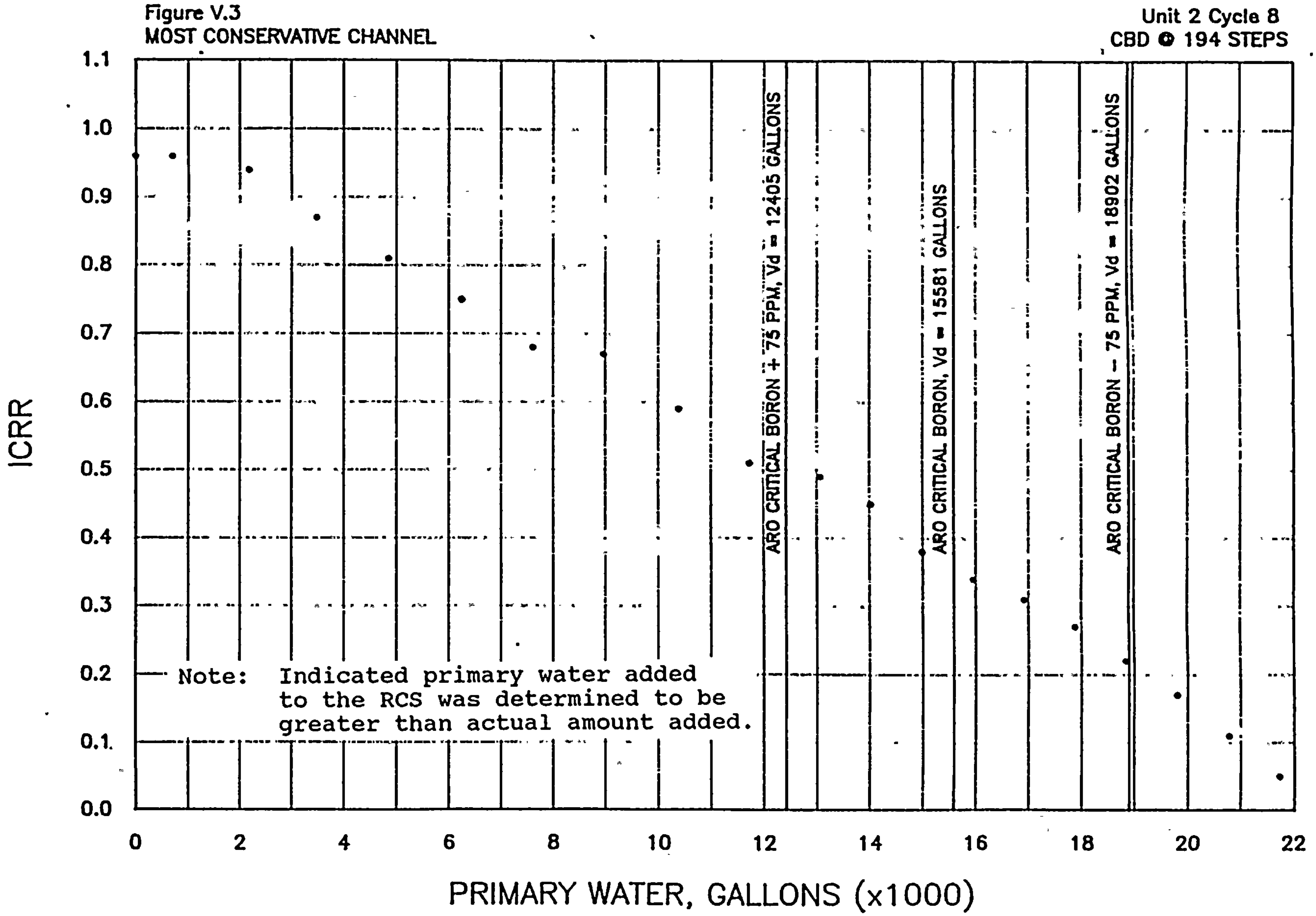
ICRR VS. CONTROL BANK WITHDRAWAL IN OVERLAP



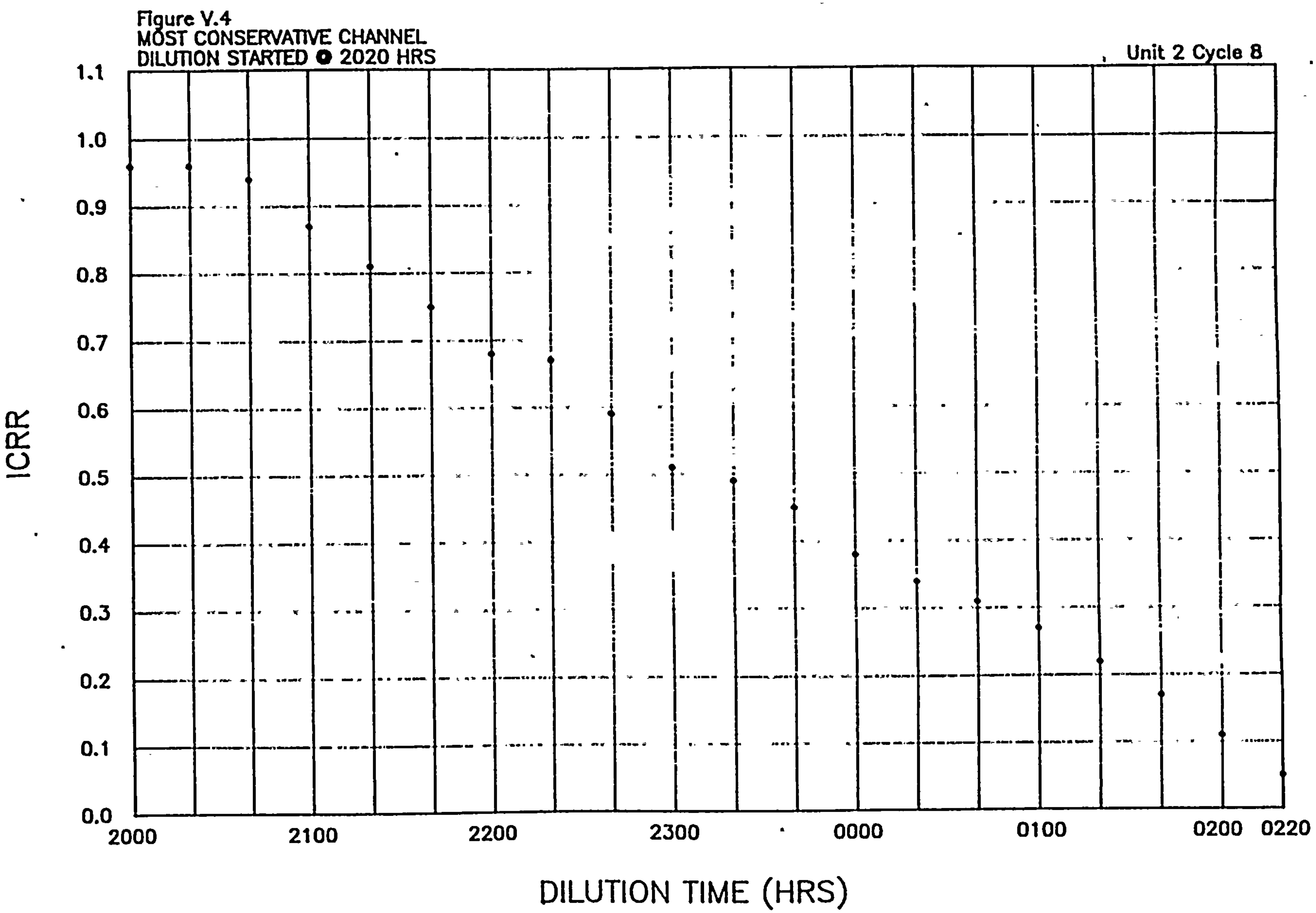
ICRR VS. BORON CONCENTRATION



ICRR VS. PRIMARY WATER



ICRR VS. DILUTION TIME



VI. ZERO POWER PHYSICS TESTING

Zero Power Physics Testing for Unit 2 Cycle 8 commenced at 0920 hours on November 8, 1990, and was completed at 0230 hours on November 9, 1990. Zero Power Physics Testing was performed for the following reasons:

1. To determine the Moderator Temperature Coefficient (MTC) and verify the MTC Technical Specifications are satisfied, and

2. To determine Control/Shutdown Bank Rod Worths and verify Shutdown Margin Technical Specifications are satisfied.

The Zero Power Physics Testing Program consisted of:

1. ARO Isothermal Temperature Coefficient (ITC) Measurement and ARO MTC Calculation, and

2. Control Rod Worth Measurement (Rod Swap) with ARO and CBD Inserted Critical Boron Concentration Measurements.

The Steam Dump Valves had to be taken out of service just prior to measurement of the ITC due to leak in the plant boiler. As a result, the Pressure Operated Relief Valves (PORVs) were used in performing the ITC Test. The RCS heatup and cooldown rates established with the PORVs were very constant (linear) and provided excellent test data for analysis.

All testing was completed with no anomalies, and results were well within the given Acceptance Criteria. The MTC was calculated to be +4.89 pcm/°F, less than the 70% RTP Technical Specification limit. However, the MTC was more positive than design so the measured value was normalized to Hot Full Power Conditions, which yielded a negative MTC. Therefore, no adjustments to rod withdrawal/boron limits were required to ensure an MTC of less than 5 pcm/°F at 70% RTP or a negative MTC at 100% RTP (Technical Specification 3.1.1.4). The results of this test are presented in Table VI.1

Rod worth measurements were completed using the "Rod Swap" methodology. The highest worth bank (CBD) was diluted into the core and the reactivity worth was directly measured from the reactivity computer strip chart. See Figure VI.1 for the plots of CBD Integral and Differential Rod Worth vs Rod Position. Following CBD measurement, the remaining seven banks, from predicted least to highest worth, were swapped with CBD and the previous Test Bank. The configuration of the rod banks after each swap would be: Test Bank fully inserted, CBD inserted at a discrete position, and all other banks withdrawn. Based on the position of CBD, the inferred rod worth of the Test Bank was calculated.

The measured/inferred rod worths for each bank were all within the Acceptance Criteria. The results of the measured/inferred rod worth, design rod worth, and Percent Error are presented in Table VI.2. Also, Table VI.3 shows the results of Boron Endpoint Measurements which were determined during this test. The results of this test were used to verify that excess Shutdown Margin (SDM) would exist for beginning and end of cycle conditions. The beginning of cycle excess SDM was 1477.5 pcm, and the end of cycle excess SDM was 1377.5 pcm. The results of the calculations satisfied Surveillance Requirement 4.1.1.1.1d of Technical Specification 3.1.1.1.

D.C. Cook Unit 2 Cycle 8 Zero Power Physics Testing Results†

TABLE VI.1
ISOTHERMAL TEMPERATURE COEFFICIENT

Measured ITC	Doppler Coefficient	MTC=(ITC-DTC)	Design MTC
+2.31 pcm/°F	-2.58 pcm/°F	+4.89 pcm/°F	+4.12 pcm/°F

TABLE VI.2
ROD WORTH MEASUREMENT

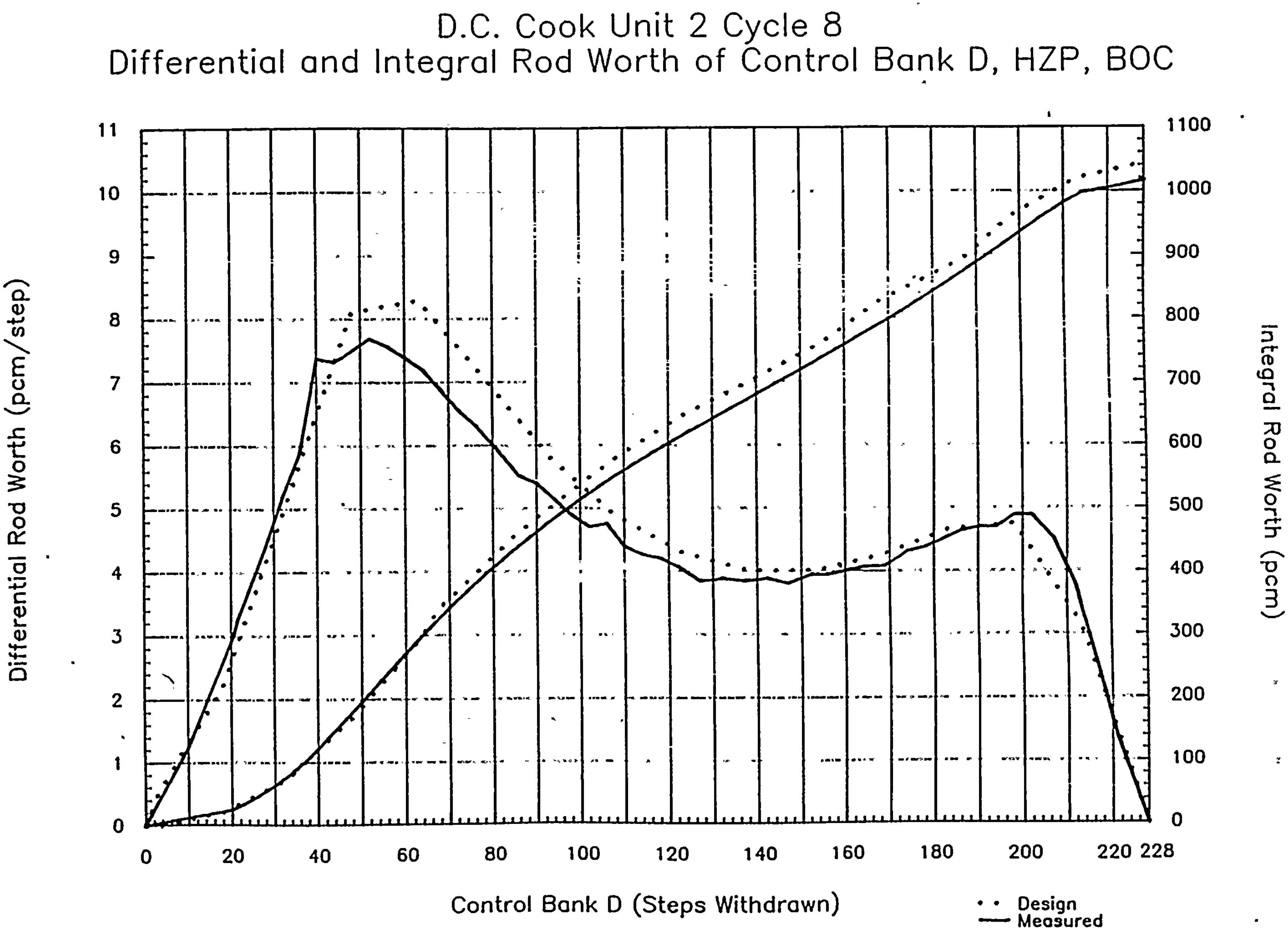
Bank	Measured Worth (pcm)	Predicted Worth (pcm)	% Error (Meas.-Pred.) x100 / Pred.
CBD	1016.2	1043	-2.6
SBA	380.2	351	+8.3
CBA	318.5	363	-12.3
SBC	466.5	456	+2.3
SBD	471.2	456	+3.3
CBB	716.4	647	+10.7
CBC	736.1	759	-3.0
SBB	897.4	889	+0.9
TOTAL	5002.5	4964	+0.8

TABLE VI.3
BORON ENDPOINTS
(from **2 THP 6040 PER.352)

Bank	Measured (ppm)	Design (ppm)
ARO	1654.3	1687
CBD in	1538.0	1566

† The design data has been obtained from Westinghouse's Nuclear Design Report.

FIGURE VI.1



D.C. Cook Unit 2 Cycle 8
Differential and Integral Rod Worth of Control Bank D, HZP, BOC
Differential Rod Worth (pcm/step)
Integral Rod Worth (pcm)
0
1
2
3
4
5
6
7
8
9
10
11
0
100
200
300
400
500
600
700
800
900
1000
1100
0
20
40
60
80
100
120
140
160
180
200
220
228
Control Bank D (Steps Withdrawn)
Design
Measured

VII.1 POWER ASCENSION TESTING

Unit 2 Cycle 8 Power Ascension Testing commenced with entry into Mode 1 on November 9, 1990, and was completed on November 23, 1990, when Full Core Flux Map (FCFM) 208-06 was obtained (at ~ 99% RTP) and processed. Figure VII.1 shows the progress of the Power Ascension Testing Program over time and consisted of the following:

1. Intermediate Range Trip Setpoint Verification
2. Core Power Distribution Measurements
3. Incore/Excore Detector Cross Calibration

During the Power Ascension to 30% RTP, intermediate range detector data was collected. The data was used to ensure the calculated, installed High Level Trip (HLT) setpoints were set conservatively, less than 25% RTP. The HLTs occurred at 23.2% and 24.3% ΔT Power for N35 and N36, respectively. Also, calculations were performed and the results indicated that the setpoints would remain conservative over the entire cycle.

Flux maps were taken at various power level plateaus during the ascension. FCFMs were taken at approximately 34%, 47%, 66%, 88%, 86%, and 99% RTP. The power distribution in each case was calculated using the DETECTOR Code (Version DETECT27) and the appropriate Engineering Data Set. All of the flux maps had peaking factors well within Technical Specification Limits, and the power distribution over the entire core met all of the Acceptance Criteria. The maximum incore quadrant power tilt ratio of all maps was less than 1.009 (0.9%). A summary of the peaking factors measured at various power levels is given in Table VII.1.

At approximately 47% RTP, FCFM 208-02 was obtained and used to calculate Incore/Excore Cross Calibration Data. After I&E began the Nuclear Instrumentation System (NIS) calibration, it was discovered that the power range drawers could not be calibrated with the provided data. Upon investigation, it was discovered that due to the very low leakage loading pattern, the expected output currents from the detectors were too small to complete the calibration. Also, it was known that the Unit 1 Power Range Drawers were previously modified to handle this situation. Since Unit 1 was shutdown for a refueling outage, it was decided to replace the Unit 2 drawers with the ones from Unit 1 (Unit 1 drawers were subsequently modified). As each drawer was replaced, the calibration was completed with no problems.

The 86% RTP FCFM was taken due to an indicated NIS Quadrant Power Tilt Ratio in excess of 2%. The flux map confirmed that no incore tilt existed (~ 0.2%) so data was provided to I&E to calibrate the power range drawers. Upon completion of the calibration, the "instrument tilt" disappeared, as expected.

TABLE VII.1

FLUX MAP DATA

MAP #	Power %	A.O. %	QPTR	$F_{\Delta H}^{M}$	$F_{\Delta H}^{L}$	$F_{Q}^{M}(z)$	$F_{Q}^{L}(z)$	APL %
208-01	33.55	1.5831	1.0076	1.4599	1.6880	1.8680	4.0764	98.54
208-02	46.65	3.9144	1.0088	1.4555	1.6490	1.9047	4.0970	97.67
208-03	65.64	1.7377	1.0067	1.4098	1.5924	1.7643	3.0972	106.06
208-04	88.30	1.0506	1.0048	1.3884	1.5249	1.7172	2.2791	106.28
208-05	86.45	2.2327	1.0029	1.3958	1.5304	1.7254	2.3458	107.60
208-06	98.72	0.4110	1.0006	1.3849	1.4938	1.6800	2.0386	108.64

A.O. = Axial Offset

QPTR = Maximum Quadrant Power Tilt Ratio

$F_{\Delta H}^{M}$ = Measured Total Enthalpy Rise Hot Channel Factor

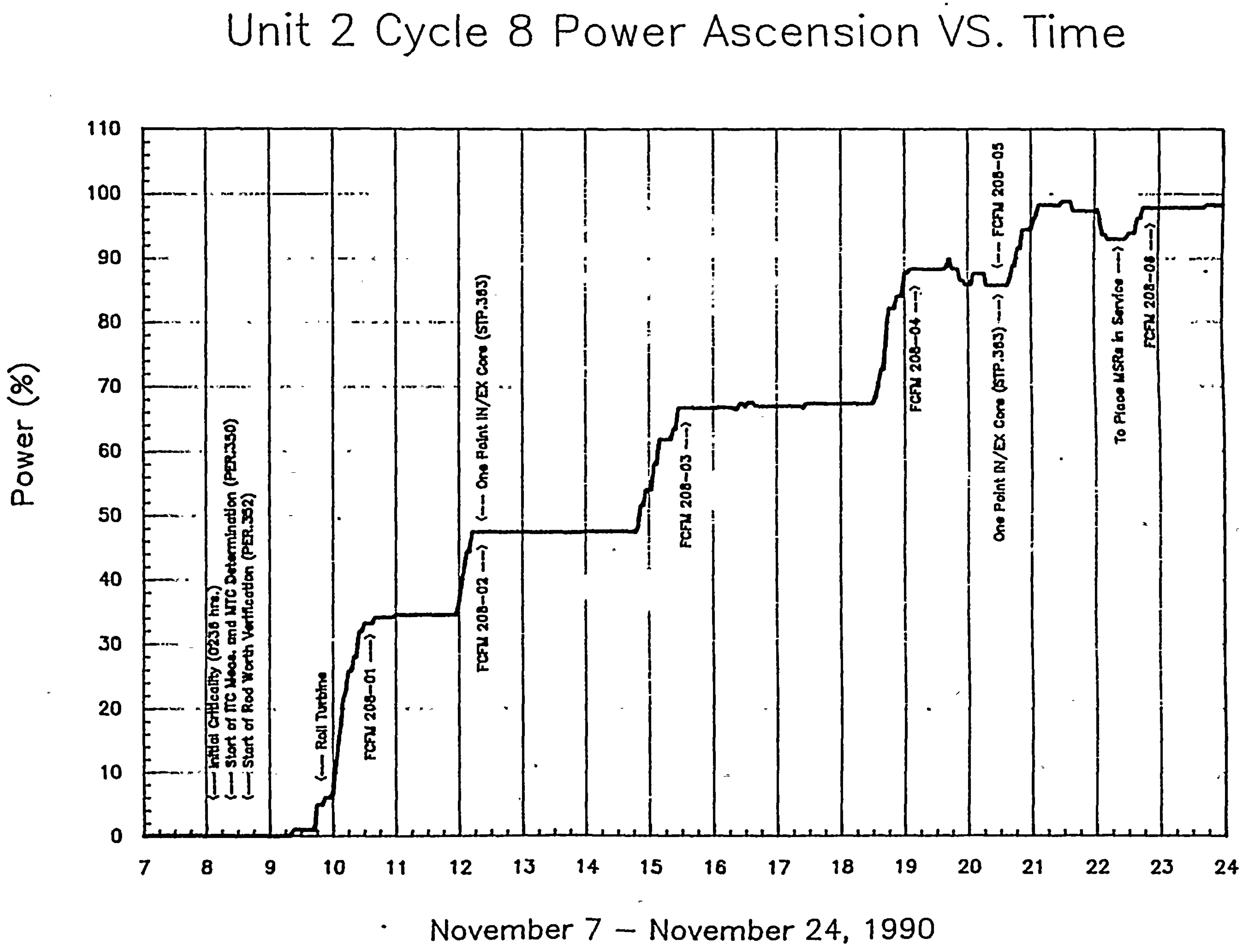
$F_{\Delta H}^{L}$ = Technical Specification Enthalpy Rise Limit

$F_{Q}^{M}(z)$ = Measured Penalized Heat Flux

$F_{Q}^{L}(z)$ = Technical Specification Penalized Heat Flux Limit

APL = Allowable Power level

FIGURE VII.1



Unit 2 Cycle 8 Power Ascension VS. Time
Power (%)
110
100
90
80
70
60
50
40
30
20
10
0
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
November 7 – November 24, 1990
<--- Initial Criticality (0236 hrs.)
<--- Start of ITC Meas. and MTC Determination (PER.350)
<--- Start of Rod Worth Verification (PER.352)
<--- Roll Turbine
FCFM 208-01 --->
FCFM 208-02 --->
<--- One Point IN/EX Core (STP.363)
FCFM 208-03 --->
FCFM 208-04 --->
One Point IN/EX Core (STP.363) --->
<--- FCFM 208-05
To Place MSRs in Service --->
FCFM 208-06 --->

VII.2 REACTOR COOLANT FLOW MEASUREMENT

The primary purpose of this test was to determine the total reactor coolant flowrate independent of the reactor coolant flow transmitters. The reactor coolant flowrate was computed from a steam generator heat balance calculation utilizing steam generator secondary side parameters.

In addition, it was also the purpose of this test to recalibrate the reactor coolant flow elbow tap differential transmitters, as required, based on the computed reactor coolant flow rates and elbow tap differential pressure data.

The total coolant flowrate was determined at approximately 90% and 100% levels of reactor thermal power using the equations of Figure VII.2. Table VII.2 shows that the total reactor coolant system flowrate was above the minimum RCS flowrate of 366,400 GPM as defined in Technical Specification 3.2.5.

TABLE VII.2

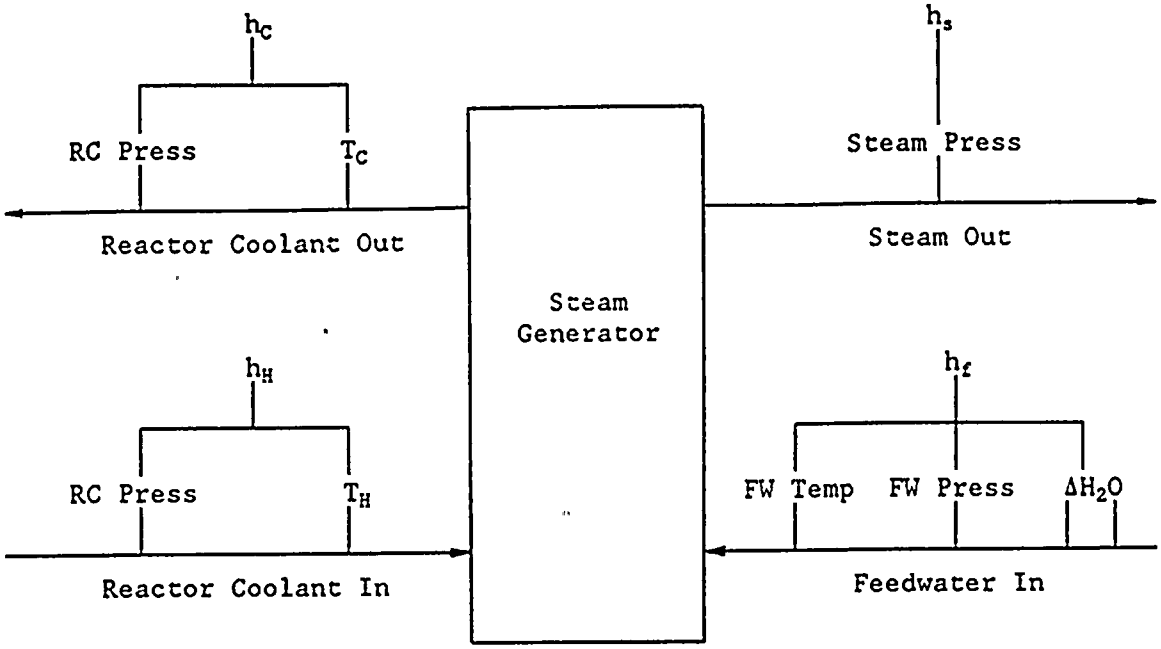
Computed Reactor Coolant System Flowrates (GPM)†

Loop #	90% RTP	100% RTP
1	93,860	93,600
2	98,260	97,570
3	93,280	92,780
4	98,330	98,040
Average	95,930	95,500
Total Flow Rate	383,730	381,990

† Flows are converted to GPM using RCS cold leg conditions.

FIGURE VII.2

Reactor Coolant Flow Determination



1. Feedwater Flowrate $= \dot{M}_{fw} = 359.1 \times d^2 \times C \times F_a \times \sqrt{\gamma \times \Delta H_2O/(1-(d/D)^2)}$

where:
- d = throat diameter of venturi (inches)
- C = coefficient of discharge
- F_a = venturi thermal expansion factor
- γ = specific weight of feedwater (lb/ft^3)
- ΔH_2O = differential pressure across venturi (inches)
- D = pipe diameter at inlet of pressure tap (inches)

2. Steam Generator Thermal Output = SGTO = $\dot{M}_{fw} \times (h_s - h_f)$

where:
- h_s = enthalpy of steam (BTU/lb)
- h_f = enthalpy of feedwater (BTU/lb)

3. Reactor Coolant Loop Flow = SGTO/(h_H − (h_C − 0.315 BTU/lb))

where:
- h_H = enthalpy of hot leg (BTU/lb)
- h_C = enthalpy of cold leg (BTU/lb)

4. Total Reactor Coolant Loop Flow = sum of the loop 1 through 4 flows (lb/hr)

VII.3 PLANT THERMAL POWER CALIBRATION

The purpose of this test was to determine reactor thermal power by measuring secondary system feedwater flow and steam parameters and to verify the accuracy of the following plant process computer outputs:

1. Reactor thermal power
2. Feedwater flows
3. Feedwater temperatures
4. Nuclear power range instrumentation

During the initial power ascension program the reactor thermal output was calculated at various power levels by measuring secondary side parameters. Data was collected at approximately 30%, 90%, and 100% RTP. The parameters that are measured for this calculation were feedwater flow, feedwater temperature, feedwater pressure, and steam pressure. With the measured values, thermal output was be calculated.

The power determinations were made by measuring the feedwater parameters before and after the steam generators. The amount of energy added by the steam generators was determined from these measurements. The energy gained by the steam side of the steam generator was the equivalent energy given off by the reactor coolant system. By knowing the heat transferred by each of the four steam generators, the total heat added to the secondary side was determined. The total heat added to the secondary side minus the heat added by RCP operation and the RCS system losses was the actual reactor power. The power determination data at 90% and 100% power is shown in Table VII.3.

All data taken for the thermal power measurement were from instruments calibrated for this test. The pressure measurements were made using 4–20 mA pressure transmitters. Feedwater flows were measured at the local transmitter for each loop. Feedwater temperatures were read using the installed thermocouples.

Before data taking commenced, steam generator blowdown was isolated and all plant parameters are allowed to stabilize. The computer thermal power was monitored during the actual thermal power test. Upon test completion, a comparison was made between the plant process computer value and the actual measured value, and no adjustments were necessary.

The primary purpose of the power determination at ~ 30% RTP was to adjust the NIS Power Range gains, if necessary, prior to the power escalation to 48%. The comparison of the measured power and NIS data proved that no adjustment to the NIS was necessary.

TABLE VII.3

THERMAL POWER CALIBRATION DATA

Calculated Power	88.35				98.80			
Computer Power	88.30				98.78			
Loop Number	1	2	3	4	1	2	3	4
Feedwater Pressure (psig)	841.74	836.29	835.96	844.27	808.0	831.9	830.9	833.6
Feedwater Temperature (°F)	410.8	410.5	410.7	410.7	421.5	421.3	421.5	421.7
Steam Pressure (psig)	806.46	799.34	797.35	804.74	793.6	784.7	783.4	782.2

Section VIII

PLANT CHEMISTRY HISTORY

VIII. PLANT CHEMISTRY HISTORY

The Unit 2 Chemistry cleanup efforts began June 29, 1990 during the reactor shutdown, with steam generator hideout analysis testing. Samples were taken at 50% RTP, 25% RTP, and 500°F with no delay in the cooldown rate. The cooldown was intentionally held for 12 hours at 325°F and 4 hours at 250°F for hideout sampling. The hideout sampling program only directly affected the system cooldown since the RCS degassing was a concurrent event.

Following the unit shutdown, reactor coolant system (RCS) dose equivalent I-131 spiked to a maximum of 0.129 μCi/cc. No problems in cleaning of the spike were noted. All compensatory Auxiliary Building Vent samples were analyzed with no problems noted.

Following the reactor coolant system degassing, the system was oxygenated by the addition of 30% hydrogen peroxide to solubilize the Co-58. RCS Co-58 activity spiked to a maximum of 0.81 μCi/cc following the addition of hydrogen peroxide. The methodology of additions for this evolution was changed per the request of AEPSC. A large initial addition (four gallons) was made to the system instead of several small initial additions, as done in previous outages. A smaller addition (1.25 gallons) was made after the iodine spike, as in previous outages. Cleanup was delayed several times due to RCS filter changeouts (with subsequent demineralizer isolation) due to high activities that were deposited on the filters. Approximately 70 hours after the initial Co-58 spike, a second unexplainable spike occurred. Reactor coolant system Co-58 activity during the second spike peaked at 0.70 μCi/cc. The cleanup progressed for another 30 hours after the second spike and was called complete at 0.095 μCi/cc. A "fresh" H-OH CVCS mixed bed (120 GPM) and the existing cation bed (75 GPM) were used for the cleanup effort. Approximately 112 Curies of Co-58 were removed from the system.

Based on the duration of outage, outage management decided not to have sludge lancing of the steam generators and the dye checking of the main and feedpump condensers performed.

During the aborted startup in October 1990, the Unit had to be degassed to repair leaking valves in containment. The Unit was restarted on November 8, 1990 and was at steady state power (~100% RTP) on November 23, 1990. At Full Power, the dose equivalent I-131 was 7.24×10^{-4} μCi/cc.