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Gentlemen:

Re: Turkey Point Unit 4
Docket No. 50-251
<u>Startup Report</u>

In accordance with Technical Specification 6.9.1.1, the enclosed Startup Report is provided for Florida Power and Light Company Turkey Point Unit 4. The Unit 4 Cycle XIV Startup Report documents the first use of axial (natural uranium) blankets and snag-resistant spacer grids at the top and bottom of the fuel assemblies.

If you have any questions, please contact us.

Very truly yours,

T. F. Plunkett
Vice President
Turkey Point Nuclear

TFP/RJT/rt

Attachment

cc: S. D. Ebneter, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Nuclear

IE26
/11

ATTACHMENT

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT NUCLEAR PLANT

UNIT 4 CYCLE XIV

STARTUP REPORT

INTRODUCTION

This report contains the official summary of the Startup Physics Tests performed on Turkey Point Unit 4 at the beginning of Cycle XIV. The testing program was conducted in accordance with Turkey Point Plant Procedures, and meets the requirements of ANSI/ANS 19.6.1, Revision 0 (12/13/85), "Startup Physics Tests for Pressurized Water Reactors".

Withdrawal of Shutdown banks commenced May 23, 1992 at 0242 and initial criticality was achieved 6 hours and 24 minutes later.

WCAP-13682, "The Nuclear Design and Core Management of the Turkey Point Unit 4 Nuclear Power Plant, Cycle 14", was the design source for verifying that acceptance criteria as specified in ANSI/ANS 19.6.1 were met. All tests performed for nuclear design verification meet their acceptance criteria.

The contents of this report provide the documentation required by Technical Specification 6.9.1.1.

TABLE OF CONTENTS

1.0 UNIT 4 CYCLE XIV CORE

1.1 Fuel Design Changes

Unit 4 Cycle 14 fuel is essentially the same as Cycle 13 fuel with the exception that Cycle 14 fuel includes axial blankets and additional snag-resistant grids.

Axial blankets, previously used in Turkey Point Unit 3 Cycle 13 core design are new to Unit 4. Axial blankets consist of a nominal 6 inches of natural UO_2 pellets at the top and bottom of the fuel pellet stack. Axial blankets are designed to reduce neutron leakage and therefor improve uranium utilization.

Anti-snag mid-grids were included in the Unit 4 Cycle 13 design. The Unit 4 Cycle 14 design adds top and bottom anti-snag grids to the fuel assembly design. This addition will reduce the possibility of assembly damage during fuel handling.

1.2 Loading Pattern

This section presents the as-loaded core configuration (Figure 1, page 5).

1.3 Rod Pattern and Rod Drop Times

This section presents the Control and Shutdown Rod pattern and the Rod Drop Times for all rods as measured per Procedure 4-PMI-028.3, "RPI Hot Calibration, CRDM Stepping Test, and Rod Drop Test" (Figure 2, page 6). All rods meet the drop time limit of 2.4 seconds as per Technical Specification 3.1.3.4.

FIGURE 1
TURKEY POINT UNIT 4 CYCLE 14
CORE LOADING

NORTH

						RR23 HF23	RR30 HF16	RR15 HF06						
				PP26	SS35 R52	TT38	RR49	TT40	SS33 R54	PP55				
			RR19	TT46	TT19 4W	SS20 R53	TT22 4W	SS18 R51	TT24 4W	TT48	RR07			
		RR04	SS48	SS41	RR46 R57	TT03 16W	RR27 R56	TT06 16W	RR47 R55	SS29	SS47	RR06		
	PP40	TT49	SS40	SS14 R61	TT08 16W	RR39 R59	TT30 8W	RR33 R60	TT14 16W	SS11 R58	SS39	TT50	PP33	
	SS38 R66	TT31 4W	RR41 R65	TT15 16W	RR11	SS01	SS17 R64	SS07	RR10	TT02 16W	RR51 R63	TT18 4W	SS37 R62	
RR08 HF07	TT41	SS24 R70	TT16 16W	RR36 R68	SS10	TT33 8W	SS03	TT34 8W	SS09	RR34 R69	TT04 16W	SS25 R67	TT42	RR20 HF13
RR29 HF15	RR50	TT32 4W	TT16 16W	TT20 8W	SS27 R74	SS16	RR25 R73	SS12	SS19 R72	TT27 8W	RR24 R71	TT28 4W	RR48	RR32 HF05
RR09 HF20	TT43	SS21 R77	TT11 16W	RR35 R78	SS02	TT35 8W	SS13	TT36 8W	SS15	RR40 R80	TT12 16W	SS26 R76	TT37	RR14 HF02
	SS30 R84	TT25 4W	RR43 R83	TT09 16W	RR26	SS08	SS28 R82	SS06	RR01	TT10 16W	RR52 R81	TT26 4W	SS43 R79	
	PP51	TT51	SS32	SS04 R89	TT05 16W	RR37 R86	TT21 8W	RR38 R87	TT07 16W	SS05 R85	SS44	TT45	PP45	
		RR12	SS46	SS34	RR42 R90	TT01 16W	RR28 R91	TT13 16W	RR44 R88	SS31	SS45	RR03		
			RR13	TT52	TT23 4W	SS23 R92	TT17 4W	SS22 R95	TT29 4W	TT47	RR05			
				PP30	SS42 R101	TT44	RR45	TT39	SS36 R93	PP54				
						RR22 HF11	RR31 HF10	RR21 HF01						

key:

	ASSY	INS.
ASSY INS.	PPxx RRxx SSxx TTxx	Rxx zzW HFxx

PP Reload Cycle 11
RR Reload Cycle 12
SS Reload Cycle 13
TT Feed Cycle 14
R Control Rod
W WABA Insert
HF Hafnium Inserts
xx Sequence number
zz Number of WABA fingers

FIGURE 2
TURKEY POINT UNIT 4 CYCLE 14
RCCA BANK PATTERN AND DROP TIMES

NORTH

					CB-B 1.37				CB-B 1.35					
						SB-A 1.36		SB-A 1.35						
					CB-C 1.35		CB-D 1.34		CB-C 1.32					
				SB-B 1.33		CB-A 1.37		CB-A 1.31		SB-B 1.32				
	CB-B 1.39		CB-C 1.32				SB-B 1.37				CB-C 1.33		CB-B 1.34	
		SB-A 1.34		CB-A 1.32						CB-A 1.33		SB-A 1.33		
			CB-D 1.35		SB-B 1.37		CB-D 1.35		SB-B 1.35		CB-D 1.34			
		SB-A 1.34		CB-A 1.35						CB-A 1.35		SB-A 1.35		
	CB-B 1.37		CB-C 1.34				SB-B 1.35				CB-C 1.32		CB-B 1.35	
				SB-B 1.35		CB-A 1.34		CB-A 1.32		SB-B 1.30				
					CB-C 1.34		CB-D 1.35		CB-C 1.34					
						SB-A 1.34		SB-A 1.36						
					CB-B 1.37				CB-B 1.40					

key:

RCCA TIME	RCCA	TIME
	SB-x	sec.
	CB-x	

SB Shutdown Bank
CB Control Bank
x Bank Identifier
sec. Drop Time to Dashpot

2.0 INITIAL CRITICALITY

2.1 INVERSE COUNT RATE RATIO (ICCR) vs DILUTION

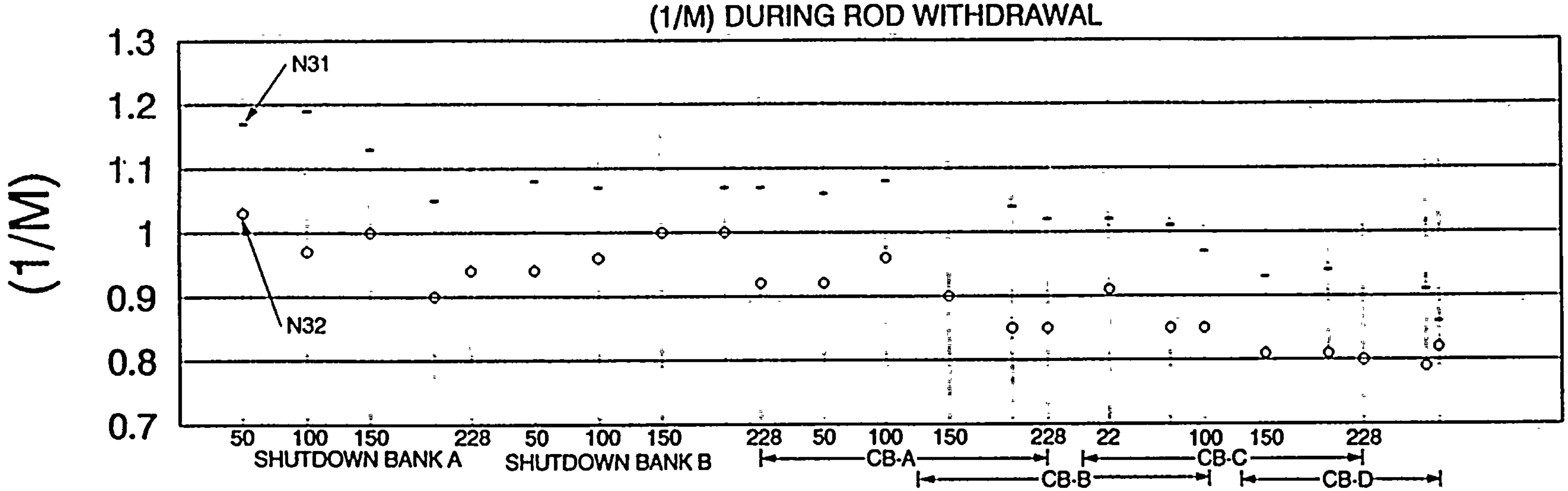
The approach to criticality began May 23, 1993 at approximately 0242 when the stepping of shutdown banks began in accordance with Procedure 0-OSP-040.6, "Initial Criticality After Refueling." Criticality was achieved approximately 6 hours and 24 minutes later on May 23, 1993 at 0906 by diluting 14,460 gallons of water with control bank D at 180 steps. Figure 3 (page 8) is a plot of the ICRR during the approach to criticality.

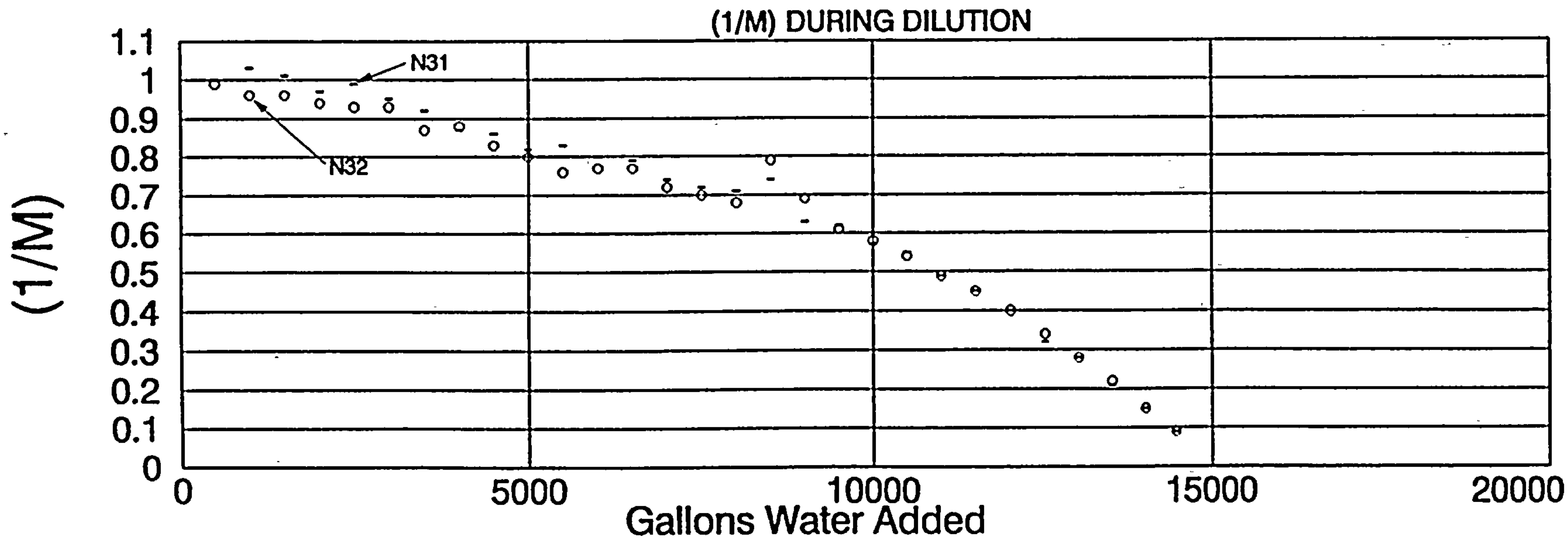
2.2 CRITICAL DATA

Upon attaining criticality, the flux level was increased to 1 x 10^{-8} amps on the reactivity computer to obtain critical data, as follows:

Tavg =	547.1°F
Control Bank D =	178 Steps
Reactor Coolant System (RCS) Boron =	1689 ppm
Picoammeter Flux =	1 x 10^{-8} A
N35 Flux =	1.1 x 10^{-8} A
N36 Flux =	1.8 x 10^{-8} A

FIGURE 3
Turkey Point Unit 4 (1/M) on Approach to Criticality





3.0 SUMMARY OF TESTS

This section provides a summary of the results of the low power physics tests for Unit 4, Cycle XIV along with the Westinghouse design data. For each test, the acceptance criteria is listed at the bottom of the table. This report compares design and measured data using Difference and Percent Difference.

Difference = Predicted - Measured

For calculating Percent Difference, the equation is:

$$\%Diff = [\frac{PredictedValue}{MeasuredValue} - 1] x100$$

3.1 Nuclear Heating

The point of adding Nuclear Heat was determined in accordance with Procedure 0-OSP-040.6, "Initial Criticality After Refueling". This is performed by establishing a small positive startup rate and measuring the flux level at which T_{avg} departs from its established steady state value. Nuclear Heating was measured to first occur at values presented on Table 3.1.1.

TABLE 3.1.1: FLUX LEVEL (AMPS)

Picoammeter	N-35	N-36
1.5×10^{-7}	2.1×10^{-8}	3.4×10^{-8}

All physics tests were conducted at or below 1×10^{-7} amps on the picoammeter connected to N-44 to assure Nuclear Heating did not occur.

3.2 Reactivity vs. Period

Reactivity Computer checkout was done in accordance with Procedure 0-OSP-040.6, "Initial Criticality After Refueling." This checkout is performed by inserting small positive and negative reactivities using rod motion. The period of the flux change is used to calculate the design reactivity. The measured reactivity is taken directly from the reactivity computer. The results of this test are given in Table 3.2.1.

TABLE 3.2.1: MEASURED REACTIVITY VS. DESIGN

PERIOD (SEC)	MEASURED REACTIVITY (PCM)	DESIGN REACTIVITY (PCM)	% DIFF*
+151.2	+39.0	+38.8	+.5
-249.6	-33.0	-33.7	-2.1
+78.3	+65.0	+65.4	-.6
+129.1	+44.0	+44.2	-.5

*Acceptance Criteria is 4% for positive period.

3.3 Boron Endpoints (ppm)

The Boron Endpoint measurement is a way of measuring the steady state boron concentration of an under-rodded core (positive period in effect) or an over-rodded core (negative period in effect). In FPL's testing program the first case is an unrodded core and the second case is a core with the reference bank at the bottom. The Boron Endpoint is measured using Procedure 0-OSP-040.5, "Nuclear Design Verification." In this methodology a just-critical condition is established as near as practical to the required rod configuration. The rods are then moved into the desired configuration and back to equilibrium. The RCS boron concentration which was measured at equilibrium is then adjusted for the ppm worth of the rods. The results of the two boron endpoint measurements are given in Table 3.3.1.

TABLE 3.3.1: BORON ENDPOINTS (ppm)

	MEASURED (ppm)	WESTINGHOUSE (ppm)	DIFFERENCE* (ppm)
ARO	1698	1693	+5
SB-B	1552	1547	+5

*Acceptance Criteria is +/- 50 ppm.

3.4 Rod Worth (ppm), Most Reactive Bank

Rod Worth was measured per Procedure 0-OSP-040.5, "Nuclear Design Verification." The reference bank (highest predicted worth) was first inserted as the controlling bank was withdrawn. Then a dilution was used to adjust the reference bank to approximately 30 steps from the bottom. Finally a Boron Endpoint (see section 3.3) was performed. By graphing the rod worth, measured by the reactivity computer, versus rod insertion a differential rod worth curve is generated. By summing the differential worth an integral rod worth curve is generated and the bank worth determined. The total bank worth is presented in Table 3.4.1. The Integral and differential bank worth of the reference bank is displayed in Figure 4 (page 12).

TABLE 3.4.1: ROD WORTH (pcm)

	MEASURED (pcm)	WESTINGHOUSE (pcm)	% DIFF* (%)
SB-B	1173	1189	1.4

*Acceptance Criteria is less than 10%

3.5 Rod Worth (pcm), Remaining Banks

The remaining RCCA bank worth was measured per Procedure 0-OSP-040.5, "Nuclear Design Verification," using the rod swap technique. This technique involves swapping the negative reactivity of the bank being inserted with the positive reactivity from the bank being withdrawn. Each bank is sequentially swapped for the reference bank. The worth of each bank can then be determined from the integral rod worth curve. The results of this measurement are given in Table 3.5.1.

TABLE 3.5.1: ROD WORTH (pcm)

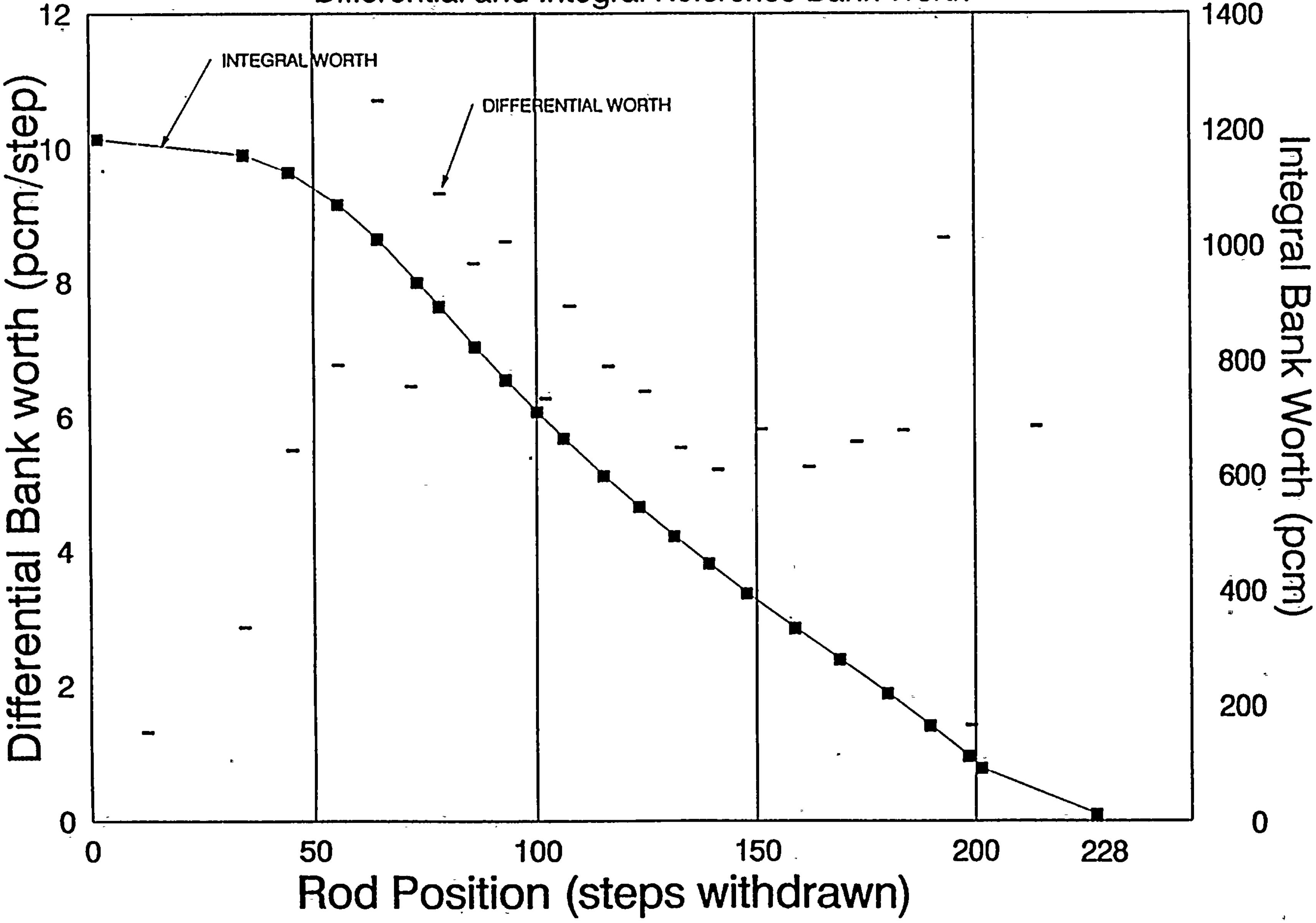
	MEASURED (pcm)	WESTINGHOUSE (pcm)	% DIFF* (%)
SB-A	1052.3	1098	4.3
CB-A	1085.6	1104	1.7
CB-B	435.3	484	11.2
CB-C	1093.0	1163	6.4
CB-D	635.9	664	4.4
TOTAL	5475.1	5702	4.1

NOTE: The total rod worth includes the reference bank.

* The acceptance criteria for rod worth measurements are: Individual banks within +/- 15% or +/- 100 pcm of design, whichever is greater and Total of all measured banks within +/- 10% of design.

FIGURE 4

Differential and Integral Reference Bank Worth



3.6 Temperature Coefficient

The isothermal and moderator temperature coefficients were determined using Procedure 0-OSP-040.5, "Nuclear Design Verification." The isothermal temperature is measured by varying the moderator temperature below the point of adding nuclear heat. The reactivity change is then simply divided by the temperature change to obtain the isothermal temperature coefficient. The moderator temperature coefficient is calculated from the isothermal temperature coefficient by subtracting the doppler coefficient. The values determined for this testing sequence are presented on Tables 3.6.1 and 3.6.2 below:

TABLE 3.6.1: ISOTHERMAL TEMPERATURE COEFFICIENT (pcm/°F)

MEASURED (pcm/°F)	WESTINGHOUSE (pcm/°F)	DIFF* (pcm/°F)
-1.44	-.43	1.01

*Acceptance Criteria is +/- 2 pcm/°F of design.

TABLE 3.6.2: MODERATOR TEMPERATURE COEFFICIENT (pcm/°F)

MEASURED* (pcm/°F)	WESTINGHOUSE (pcm/°F)
.26	1.27

*Acceptance Criteria is < 5 pcm/°F.

3.7 HZP Differential Boron Worth

The Hot Zero Power (HZP) Differential Boron Worth was measured using Procedure 0-OSP-040.5, "Nuclear Design Verification." The worth of the reference bank is divided by the boron change from ARO to the reference bank fully inserted. The value obtained for this test is presented on Table 3.7.1.

TABLE 3.7.1: HZP DIFFERENTIAL BORON WORTH (pcm/ppm)

MEASURED (pcm/ppm)	WESTINGHOUSE (pcm/ppm)	% DIFF* (pcm/ppm)
8.56	8.32	2.8

*Acceptance Criteria ≤ +/- 15%.

4.0 SHUTDOWN MARGIN

The Shutdown Margin was calculated prior to power escalation to verify adequate shutdown capability. For this calculation, the total of the design rod worth (minus the most reactive stuck rod) were reduced by 7%. The results show adequate shutdown margin at Beginning of Life (BOL) and End of Life (EOL). The following is a summary of the data used*:

TABLE 4.1: UNIT 4, CYCLE XIV SHUTDOWN DATA

	BOL	EOL
HZP Control Rod Worth Requirement (%$\Delta\rho$)		
All Rods Inserted Less Most Reactive Stuck Rod	6.28	6.72
(1) Less 7%	5.84	6.25
Hot Full Power (HFP) to HZP Reactivity Insertion (%$\Delta\rho$)		
Reactivity Defects (Doppler, T_{avg}, Void, Redistribution)	1.72	2.71
Rod Insertion Allowance	0.50	0.50
(2) Total Requirements	2.22	3.21
Shutdown Margin (1) - (2) (%$\Delta\rho$)	3.62	3.04
Required Shutdown Margin (%$\Delta\rho$)	1.00	1.77

*Source: WCAP 13682

5.0 POWER DISTRIBUTION MAPS

The core was mapped using incore instrumentation for power levels of 30%, 50% and 100%. A summary of the results are presented on pages 15 through 17.

FIGURE 5
SUMMARY OF 30% POWER FLUXMAP

MEASURED ASSEMBLY POWER AND PERCENT DIFF. TO EXPECTED POWER - INSTR. LOC. ONLY

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R							0.222	0.232 -1.8	0.223						
P					0.349	0.824	1.115	0.774	1.103	0.808	0.345				
N				0.393 3.7	1.110	1.328 1.4	1.262	1.315 -0.7	1.229 -2.7	1.298	1.100 0.9	0.366			
M			0.386	0.941	1.272	1.076	1.286	0.900	1.284	1.103	1.304	0.933	0.348 -8.0		
L		0.341 -0.3	1.083	1.198	1.196 -6.5	1.250	1.167 1.0	1.331 4.0	1.202	1.339 3.3	1.292 1.0	1.294	0.999	0.313	
K		0.817	1.337	1.084	1.292	0.988	1.223	1.255	1.235	0.996	1.273	1.056	1.261	0.786	
J	0.228 0.5	1.109	1.267	1.330 4.5	1.207	1.222 2.1	1.252	1.178	1.253 1.9	1.169	1.066 -7.7	1.208	1.218 -3.1	1.110	0.227
H	0.244	0.798	1.274 -3.8	0.908	1.352 5.7	1.275	1.184	0.820	1.181	1.223 -0.2	1.211	0.843 -5.5	1.326 0.1	0.785	0.238 0.7
G	0.240	1.195 6.8	1.294	1.278	1.164	1.209	1.257	1.194	1.285 4.5	1.199	1.101	1.199	1.217	1.140	0.231
F		0.865	1.384 5.7	1.122	1.238 -4.1	1.016	1.224 2.3	1.261 2.9	1.208	0.963 -3.6	1.243	0.996 -9.0	1.266	0.835 2.2	
E		0.361	1.162	1.337	1.368 7.0	1.299	1.163	1.310	1.151	1.285	1.294 1.2	1.246	1.045	0.338	
D			0.411	1.035 8.5	1.331	1.051 -4.1	1.213	0.837	1.206	1.109	1.304 1.9	0.947	0.361 -4.9		
C				0.411	1.087	1.255	1.227	1.258 -5.0	1.275	1.346	1.097	0.376			
B					0.327	0.779 -4.6	1.125	0.809 2.7	1.191 6.4	0.843	0.344 0.9				
							0.243 6.9	0.249	0.236						

POWER TILT IN UPPER HALF OF CORE		POWER TILT IN LOWER HALF OF CORE		CORE AVERAGE AXIAL OFFSET
(-,+)	(+,+)	(-,+)	(+,+)	
0.9974	0.9665	1.0087	1.0129	0.208
1.0486	0.9875	0.9817	0.9966	
(-,-)	(+,-)	(-,-)	(+,-)	

TOP TEN NUCLEAR F-DELTA-H

245	G14MI	FDHN=1.5422
309	B 7IC	FDHN=1.5359
260	F13LC	FDHN=1.5285
274	E12LF	FDHN=1.5262
273	E13MB	FDHN=1.5020
287	D11JD	FDHN=1.4936
185	M 5FL	FDHN=1.4909
293	D 5FD	FDHN=1.4907
233	H11ML	FDHN=1.4859
198	L 4DJ	FDHN=1.4787

LIMITING 3 FQ ELEVATIONS (TOP 15%, BOTTOM 15%, AND MIDDLE 70% OF THE CORE)

AXIAL POINT	FQ(Z) LIMIT	MEAS. FQ(Z)	PERCENT TO LIM.	SOURCE NO. ID
28	4.6052	2.2654	50.81	402 G14XX
10	4.3964	1.7459	60.29	417 F13XX
52	4.6400	1.6294	64.88	322 P 7XX

FIGURE 6
SUMMARY OF 50% POWER FLUXMAP

MEASURED ASSEMBLY POWER AND PERCENT DIFF. TO EXPECTED POWER - INSTR. LOC. ONLY

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R							0.225	0.236	0.226						
								-0.2							
P					0.342	0.801	1.111	0.774	1.103	0.773	0.324				
N				0.386	1.086	1.289	1.245	1.318	1.195	1.242	1.031	0.379			
				2.0		-1.6		-0.4	-5.3						
M			0.382	0.962	1.282	1.080	1.253	0.870	1.253	1.075	1.264	0.937	0.378		
													-0.1		
L		0.340	1.081	1.280	1.277	1.277	1.136	1.264	1.158	1.308	1.261	1.236	1.065	0.340	
		-0.6			-0.2		-1.7	-1.3		0.8	-1.5	-3.5			
K		0.811	1.319	1.105	1.307	0.991	1.177	1.214	1.197	0.995	1.272	1.067	1.262	0.810	
J	0.226	1.133	1.278	1.296	1.182	1.221	1.206	1.157	1.232	1.182	1.136	1.249	1.240	1.112	0.225
				1.8					0.0		-1.7				
H	0.239	0.794	1.332	0.902	1.304	1.264	1.206	0.815	1.164	1.210	1.259	0.880	1.305	0:785	0.236
			0.7		1.8					-1.3			-1.3		0.0
G	0.231	1.143	1.276	1.290	1.178	1.228	1.283	1.194	1.269	1.211	1.147	1.243	1.234	1.113	0.225
							4.2		3.0						
F		0.832	1.339	1.120	1.323	1.031	1.226	1.255	1.221	1.015	1.283	1.060	1.277	0.799	
			2.3				2.4			1.5		-3.3		-2.1	
E		0.349	1.120	1.319	1.313	1.321	1.179	1.299	1.170	1.296	1.264	1.253	1.060	0.335	
					2.7										
D			0.393	0.990	1.314	1.106	1.288	0.899	1.280	1.091	1.266	0.942	0.375		
				3.8		1.0			0.9		-1.2				
C				0.394	1.103	1.312	1.269	1.345	1.279	1.312	1.073	0.376			
								1.7							
B					0.339	0.809	1.150	0.809	1.145	0.821	0.339				
						-0.9			2.4		-0.5				
A							0.240	0.246	0.231						
							5.8								

POWER TILT IN UPPER HALF OF CORE		POWER TILT IN LOWER HALF OF CORE		CORE AVERAGE AXIAL OFFSET
(-,+)	(+,+)	(-,+)	(+,+)	
1.0010	0.9787	0.9978	0.9865	-0.272
1.0284	0.9920	1.0128	1.0028	
(-,-)	(+,-)	(-,-)	(+,-)	

TOP TEN NUCLEAR F-DELTA-H

274	E12LF	FDHN=1.5074
287	D11JD	FDHN=1.5016
307	B 9GC	FDHN=1.4814
273	E13HB	FDHN=1.4814
260	F13LC	FDHN=1.4800
309	B 7IC	FDHN=1.4773
245	G14MI	FDHN=1.4747
215	J14HG	FDHN=1.4671
179	M11JL	FDHN=1.4653
297	C11NC	FDHN=1.4650

LIMITING 3 FQ ELEVATIONS (TOP 15%, BOTTOM 15%, AND MIDDLE 70% OF THE CORE)

AXIAL POINT	FQ(Z) LIMIT	MEAS. FQ(Z)	PERCENT TO LIM.	SOURCE NO. ID
27	4.5936	2.0692	54.95	372 J14XX
52	4.6400	1.7941	61.33	384 J 2XX
10	4.3964	1.5974	63.66	431 E12XX

FIGURE 7
SUMMARY OF 100% FLUXMAP

MEASURED ASSEMBLY POWER AND PERCENT DIFF. TO EXPECTED POWER - INSTR. LOC. ONLY

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R							0.269	0.286 13.9	0.270						
P					0.359	0.742	1.119	0.860	1.184	0.919	0.380				
N				0.441 13.6	1.081	1.162 -7.4	1.207	1.351 2.8	1.269	1.148	1.146	0.425			
M			0.410	0.970	1.232	1.003	1.225	0.986	1.221	0.987	1.174	0.976	0.424 9.3		
L		0.340 -2.4	1.026	1.145	1.151 -7.4	1.174	1.095	1.224 -6.8	1.087	1.162 -8.7	1.226	1.218 -1.4	1.090	0.379	
K		0.864	1.226	1.001	1.243	1.002	1.206	1.240	1.183	0.995	1.216	1.059	1.235	0.871	
J	0.281 18.4	1.203	1.255	1.190	1.156	1.270 3.1	1.324	1.275	1.323 3.0	1.218	1.150 -2.2	1.259	1.236	1.163	0.268
H	0.281	0.857	1.335 1.6	0.982	1.226 -6.7	1.250	1.256	0.955	1.211	1.221 -3.6	1.274	1.012	1.319 0.3	0.849	0.286 13.6
G	0.248	1.138 5.0	1.237	1.242	1.141	1.203	1.283 -0.2	1.226	1.206 -6.1	1.178	1.155	1.281	1.263	1.161	0.260
F		0.797	1.189 -5.2	1.043	1.279 0.7	1.004	1.200 -2.6	1.325 4.7	1.209	0.989 -3.3	1.267	1.091 1.0	1.294	0.844 5.4	
E		0.330	1.063	1.230	1.174 -5.6	1.251	1.189	1.320	1.178	1.238	1.280 2.9	1.265	1.120	0.380	
D			0.419	1.012 7.9	1.250	1.092 1.1	1.239	0.971	1.264 -0.7	1.059	1.145 -7.3	0.966	0.439 13.0		
C				0.419	1.123	1.334	1.257	1.225 -6.8	1.194	1.259	1.073	0.411			
B					0.388	0.892 11.4	1.180	0.776	1.062 -2.0	0.837	0.388 11.6				
A							0.282 19.1	0.274	0.231						

POWER TILT IN UPPER HALF OF CORE		POWER TILT IN LOWER HALF OF CORE		CORE AVERAGE AXIAL OFFSET
(-,+)	(+,+)	(-,+)	(+,+)	
0.9843	1.0014	0.9939	1.0010	3.144
1.0128	1.0015	1.0005	1.0046	
(-,-)	(+,-)	(-,-)	(+,-)	

TOP TEN NUCLEAR F-DELTA-H

215	J14MG	FDHN=1.5287
165	P 7IM	FDHN=1.5057
307	B 9GC	FDHN=1.4991
227	J 2CG	FDHN=1.4787
257	G 2CI	FDHN=1.4773
245	G14MI	FDHN=1.4495
172	N 8ML	FDHN=1.4444
283	E 3CB	FDHN=1.4437
282	E 4DF	FDHN=1.4391
222	J 7BC	FDHN=1.4323

LIMITING 3 FQ ELEVATIONS (TOP 15%, BOTTOM 15%, AND MIDDLE 70% OF THE CORE)

AXIAL POINT	FQ(Z) LIMIT	MEAS. FQ(Z)	PERCENT TO LIM.	SOURCE NO. ID
20	2.2585	1.9780	12.42	372 J14XX
6	2.0692	1.7324	16.27	379 J 7XX
52	2.3223	1.9026	18.07	322 P 7XX

6.0 CRITICAL BORON CONCENTRATION

The critical boron concentration was calculated by adjusting a measured boron concentration to the equilibrium hot full power, all rods out condition, as per Operating Procedure 1009.6, "Critical Boron Concentration-Full Power." For Unit 4, Cycle XIV, this calculation was performed at 600 Megawatt - days/metric-ton-uranium (MWD/MTU). The following is a summary of the results.

TABLE 6.1: SUMMARY OF CRITICAL BORON CONCENTRATION (ppm)

MEASURED (ppm)	WESTINGHOUSE (ppm)	DIFF* (ppm)
1170	1187	17

*Acceptance Criteria +/- 50 ppm.