NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT UNIT 2

STARTUP REPORT FOR CYCLE 4 WITH GE 11 FUEL

PURPOSE

This report presents a summary of the results from the plant startup and power escalation testing which was conducted for Cycle 4 at Nine Mile Point Unit 2. This report is submitted pursuant to Technical Specification 6.9.1.1 "Routine Reports-Startup Report" because of the use of a new fuel design at Nine Mile Point Unit 2.

Each of the tests identified in FSAR Section 14, "Initial Test Program" were reviewed. In general, a description is included of the measured values during the test compared with design predictions. Any corrective actions or specific details required in license conditions are also included.

CONTENTS

1.0 GE-11 BUNDLE DESIGN

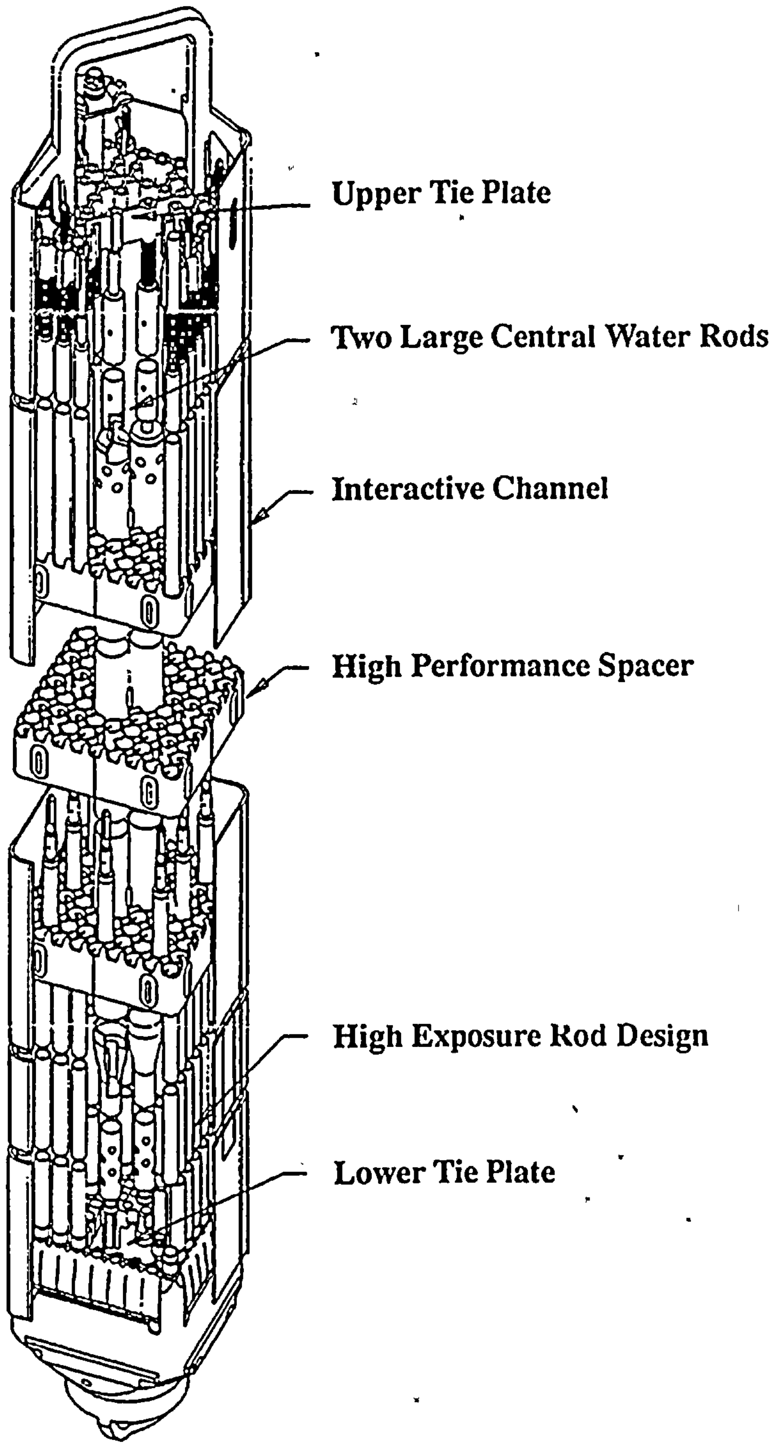
The refueling outage included the replacement of 228 fuel bundles. Of these, 196 were of the new GE-11 fuel design, Figure 1. The remaining 32 bundles were reinserted GE6B design fuel previously discharged after 2 cycles of exposure. The following General Electric fuel designs are used in Cycle 4:

Bundle Type	Enrichment % U - 235	# of Cycles	LHGR (KW/ft)	Number of Bundles
GE11	3.32	0	14.4	196
GE9B	3.20	1	14.4	248
GE9B	2.99	2	14.4	196
GE6B	2.19	3	13.4	92
GE6B	2.19	2	13.4	32

The 196 GE-11 bundles use a 9x9 matrix of fuel pins as opposed to the 8x8 matrix already in use. Seven pin locations in the center of these new bundles are occupied by large central water rods. Eight pins are only partial length (90"), to allow more water moderation in the top of the bundle. The individual pins are smaller in diameter and the tie plate has been changed to increase pressure drop, thus keeping the overall bundle pressure drop the same as previous designs to ensure compatibility in a mixed core.

FIGURE 1
GE 11 FUEL DESIGN

GE11 Fuel Bundle Design



2.0 STARTUP TEST PROGRAM

During refueling operations and the subsequent return to power, activities were controlled under normal administrative programs rather than a separate, formally defined post-refueling startup program. These administrative programs cover areas of normal operation such as:

Design Changes/Post-modification Testing
Technical Specification Surveillances
Special Nuclear Material Control
Computer Software Modification
Post-Maintenance Testing
Inservice Inspection
Periodic and Special Tests
Radiation Control
Radiochemical Surveillances

3.0 TESTS IMPACTED BY NEW FUEL DESIGN

As required by Technical Specification, this report addresses the startup tests identified in FSAR Subsection 14.2.12.2, "INITIAL TEST PROGRAM". These tests were evaluated to determine whether they were impacted by the new fuel design:

TEST	STATUS	*TEST SECTION
CHEMICAL AND RADIOCHEMICAL	No Impact	Not Applicable
RADIATION MEASUREMENT	No Impact	Not Applicable
FUEL LOADING	Impact	Core Verification (Section 9.0)
FULL CORE SHUTDOWN MARGIN	Impact	Shutdown Margin Demonstration (Section 5.0)
CONTROL ROD DRIVE SYSTEM	Impact	Control Rod Drive Scram Test (Section 4.0)
SOURCE RANGE MONITOR PERFORMANCE	No Impact	Not Applicable
INTERMEDIATE RANGE MONITOR PERFORMANCE	No Impact	Not Applicable
LPRM CALIBRATION	No Impact	Not Applicable
APRM CALIBRATION	No Impact	Not Applicable
NSSS PROCESS COMPUTER	No Impact	Not Applicable
RCIC SYSTEM	No Impact	Not Applicable
SELECTED PROCESS TEMPERATURES	No Impact	Not Applicable
WATER LEVEL REFERENCE AND VARIABLE LEG TEMPERATURES	No Impact	Not Applicable
SYSTEM EXPANSION	No Impact	Not Applicable
TIP UNCERTAINTY	Impact	TIP Uncertainty (Section 8.0)
CORE PERFORMANCE	Impact	Surveillance Comparison (Section 7.0)

*Refer to the section described below for an evaluation of impact.

3.0 (Cont)

TEST	STATUS	*TEST SECTION
STEAM PRODUCTION	No Impact	Not Applicable
PRESSURE REGULATOR	No Impact	Not Applicable
WATER LEVEL SET POINT, MANUAL FEEDWATER FLOW CHANGES	No Impact	Not Applicable
LOSS OF FEEDWATER HEATING	No Impact	Not Applicable
FEEDWATER PUMP TRIP	No Impact	Not Applicable
MAXIMUM FEEDWATER RUNOUT CAPABILITY	No Impact	Not Applicable
TURBINE VALVE SURVEILLANCE	No Impact	Not Applicable
MAIN STEAM ISOLATION VALVES FUNCTIONAL TESTS	No Impact	Not Applicable
FULL REACTOR ISOLATION	No Impact	Not Applicable
RELIEF VALVES	No Impact	Not Applicable
TURBINE TRIP AND GENERATOR LOAD REJECTION	No Impact	Not Applicable
SHUTDOWN FROM OUTSIDE THE MAIN CONTROL ROOM	No Impact	Not Applicable
RECIRCULATION FLOW CONTROL, VALVE POSITION CONTROL	No Impact	Not Applicable
RECIRCULATION FLOW LOOP CONTROL	No Impact	Not Applicable
RECIRCULATION SYSTEM, ONE-PUMP TRIP	No Impact	Not Applicable
RECIRCULATION SYSTEM, TWO-PUMP TRIP	No Impact	Not Applicable
RECIRCULATION SYSTEM PERFORMANCE	No Impact	Not Applicable
RECIRCULATION PUMP RUNBACK	No Impact	Not Applicable
RECIRCULATION SYSTEM CAVITATION	No Impact	Not Applicable
LOSS OF TURBINE GENERATOR AND OFFSITE POWER	No Impact	Not Applicable

*Refer to the section described below for an evaluation of impact.

3.0 (Cont)

TEST	STATUS	*TEST SECTION
DRYWELL PIPING VIBRATION	No Impact	Not Applicable
RECIRCULATION SYSTEM FLOW CALIBRATION	No Impact	Not Applicable
REACTOR WATER CLEANUP SYSTEM	No Impact	Not Applicable
RESIDUAL HEAT REMOVAL SYSTEM	No Impact	Not Applicable
OFF-GAS SYSTEM	No Impact	Not Applicable
DRYWELL COOLING SYSTEM	No Impact	Not Applicable
ESF AREA COOLING	No Impact	Not Applicable
BOP PIPING VIBRATION	No Impact	Not Applicable
BOP SYSTEM EXPANSION	No Impact	Not Applicable
REACTOR INTERNALS VIBRATION	No Impact	Not Applicable
EMERGENCY RECIRCULATION VENTILATION	No Impact	Not Applicable
DRYWELL HIGH ENERGY PENETRATIONS	No Impact	Not Applicable

*Refer to the section described below for an evaluation of impact.

4.0 CONTROL ROD DRIVE SCRAM TIME TESTS

4.1 Control Rod Drive Scram Time Test Abstract

Following a major refueling outage, it is necessary to verify that the control rods fully insert upon receiving a scram signal within the time interval specified in the Technical Specifications.

The general procedure is to individually withdraw and scram each control rod until all rods have been scrammed. Scram times are captured on a computerized system designed to time and record plant data.

The control rod drive scram time testing shall be considered acceptable if Technical Specification 3.1.3.2, 3.1.3.3 and 3.1.3.4 are met (See Table 4.2).

4.2 Control Rod Drive Scram Time Test Results

Table 4.1 contains the results of the control rod drive scram time tests. Results of the test are within the values specified by Technical Specification 3.1.3.2, 3.1.3.3 and 3.1.3.4. (see Table 4.2).

TABLE 4.1

ROD SCRAM TIMES AFTER 1993 OUTAGE (SECONDS)

Rod	Notch 45	Notch 39	Notch 25	Notch 5
59-18	0.252	0.547	1.268	2.326
59-22	0.246	0.555	1.304	2.474
59-26	0.254	0.539	1.264	2.330
59-3∅	0.258	0.569	1.318	2.474
59-34	0.252	0.551	1.290	2.396
59-38	0.254	0.561	1.294	2.364
59-42	0.260	0.577	1.350	2.519
55-14	0.244	0.539	1.264	2.362
55-18	0.246	0.537	1.284	2.412
55-22	0.236	0.523	1.250	2.354
55-26	0.205	0.490	1.195	2.315
55-30	0.258	0.557	1.274	2.364
55-34	0.258	0.553	1.290	2.368
55-38	0.248	0.535	1.268	2.366
55-42	0.301	0.589	1.288	2.322
55-46	0.244	0.529	1.294	2.472
51-10	0.254	0.557	1.284	2.356
51-14	0.256	0.563	1.324	2.496
51-18	0.254	0.599	1.408	2.557
51-22	0.260	0.553	1.286	2.374
51-26	0.248	0.537	1.248	2.308
51-30	0.254	0.543	1.272	2.416
51-34	0.246	0.551	1.272	2.314
51-38	0.260	0.567	1.312	2.482
51-42	0.238	0.517	1.165	2.130
51-46	0.258	0.567	1.312	2.424
51-50	0.252	0.541	1.232	2.284
47- 6	0.250	0.541	1.256	2.322
47-10	0.246	0.545	1.278	2.422
47-14	0.234	0.507	1.169	2.200
47-18	0.254	0.545	1.256	2.370
47-22	0.256	0.553	1.302	2.396
47-26	0.256	0.539	1.250	2.366
47-30	0.252	0.537	1.274	2.336
47-34	0.246	0.537	1.296	2.354
47-38	0.252	0.541	1.256	2.342
47-42	0.250	0.529	1.228	2.266
47-46	0.260	0.559	1.300	2.390
47-50	0.252	0.565	1.374	2.512
47-54	0.236	0.517	1.246	2.360
43- 2	0.250	0.561	1.316	2.442
43- 6	0.258	0.561	1.294	2.406
43-10	0.244	0.535	1.260	2.330
43-14	0.254	0.573	1.360	2.480
43-18	0.208	0.515	1.268	2.366
43-22	0.256	0.563	1.310	2.418
43-26	0.260	0.577	1.330	2.440
43-30	0.250	0.547	1.256	2.328
43-34	0.254	0.545	1.272	2.372

TABLE 4.1 (Cont)

ROD SCRAM TIMES AFTER 1993 OUTAGE (SECONDS)

Rod	Notch 45	Notch 39	Notch 25	Notch 5
43-38	0.258	0.563	1.340	2.510
43-42	0.248	0.547	1.292	2.362
43-46	0.242	0.537	1.244	2.298
43-50	0.250	0.559	1.312	2.408
43-54	0.244	0.551	1.328	2.514
43-58	0.279	0.581	1.314	2.456
39- 2	0.252	0.563	1.328	2.454
39- 6	0.248	0.555	1.338	2.510
39-10	0.234	0.543	1.310	2.478
39-14	0.256	0.559	1.294	2.454
39-18	0.258	0.569	1.318	2.452
39-22	0.240	0.540	1.280	2.440
39-26	0.262	0.561	1.278	2.340
39-30	0.252	0.539	1.238	2.292
39-34	0.268	0.593	1.398	2.565
39-38	0.256	0.559	1.280	2.360
39-42	0.254	0.569	1.362	2.573
39-46	0.236	0.537	1.244	2.320
39-50	0.250	0.547	1.272	2.396
39-54	0.228	0.535	1.266	2.338
39-58	0.262	0.575	1.318	2.458
35- 2	0.240	0.521	1.200	2.254
35- 6	0.258	0.565	1.316	2.452
35-10	0.234	0.531	1.266	2.330
35-14	0.260	0.561	1.296	2.434
35-18	0.258	0.593	1.482	2.777
35-22	0.248	0.541	1.258	2.366
35-26	0.262	0.567	1.298	2.388
35-30	0.260	0.557	1.294	2.406
35-34	0.245	0.625	1.240	2.305
35-38	0.248	0.533	1.258	2.334
35-42	0.258	0.567	1.312	2.430
35-46	0.236	0.509	1.228	2.318
35-50	0.238	0.517	1.202	2.226
35-54	0.254	0.565	1.374	2.607
35-58	0.248	0.551	1.302	2.432
31- 2	0.252	0.551	1.288	2.390
31- 6	0.254	0.539	1.310	2.340
31-10	0.250	0.551	1.288	2.414
31-14	0.254	0.543	1.264	2.358
31-18	0.258	0.561	1.278	2.344
31-22	0.260	0.555	1.270	2.392
31-26	0.244	0.575	1.416	2.591
31-30	0.264	0.549	1.264	2.322
31-34	0.254	0.553	1.282	2.360
31-38	0.242	0.525	1.214	2.246
31-42	0.256	0.567	1.296	2.380
31-46	0.250	0.545	1.235	2.230
31-50	0.246	0.551	1.292	2.440

TABLE 4.1 (Cont)

ROD SCRAM TIMES AFTER 1993 OUTAGE (SECONDS)

Rod	Notch 45	Notch 39	Notch 25	Notch 5
31-54	0.256	0.551	1.272	2.330
31-58	0.238	0.531	1.272	2.392
27- 2	0.250	0.547	1.296	2.408
27- 6	0.248	0.555	1.318	2.462
27-10	0.234	0.523	1.248	2.360
27-14	0.252	0.549	1.266	2.354
27-18	0.260	0.557	1.292	2.356
27-22	0.252	0.535	1.250	2.350
27-26	0.240	0.519	1.204	2.250
27-30	0.252	0.561	1.316	2.434
27-34	0.262	0.573	1.334	2.426
27-38	0.244	0.541	1.284	2.404
27-42	0.258	0.569	1.348	2.446
27-46	0.252	0.539	1.264	2.386
27-50	0.299	0.605	1.330	2.428
27-54	0.244	0.545	1.296	2.418
27-58	0.291	0.573	1.296	2.400
23- 2	0.240	0.537	1.268	2.408
23- 6	0.252	0.551	1.322	2.410
23-10	0.246	0.545	1.256	2.398
23-14	0.246	0.545	1.262	2.360
23-18	0.234	0.525	1.282	2.394
23-22	0.250	0.545	1.270	2.362
23-26	0.224	0.501	1.167	2.130
23-30	0.254	0.547	1.298	2.420
23-34	0.252	0.561	1.318	2.470
23-38	0.341	0.659	1.586	3.033
23-42	0.256	0.553	1.306	2.474
23-46	0.252	0.537	1.220	2.250
23-50	0.254	0.551	1.272	2.336
23-54	0.252	0.557	1.308	2.424
23-58	0.281	0.555	1.268	2.348
19- 2	0.194	0.489	1.290	2.466
19- 6	0.268	0.559	1.280	2.412
19-10	0.256	0.553	1.320	2.426
19-14	0.252	0.553	1.262	2.346
19-18	0.248	0.543	1.300	2.392
19-22	0.256	0.551	1.288	2.438
19-26	0.238	0.511	1.185	2.204
19-30	0.258	0.541	1.274	2.396
19-34	0.236	0.547	1.324	2.464
19-38	0.248	0.533	1.260	2.386
19-42	0.244	0.543	1.268	2.330
19-46	0.256	0.553	1.280	2.402
19-50	0.262	0.569	1.330	2.418
19-54	0.242	0.515	1.206	2.228

TABLE 4.1 (Cont)

ROD SCRAM TIMES AFTER 1993 OUTAGE (SECONDS)

Rod	Notch 45	Notch 39	Notch 25	Notch 5
19-58	0.244	0.537	1.260	2.358
15- 6	0.250	0.559	1.330	2.450
15-10	0.246	0.543	1.286	2.396
15-14	0.240	0.527	1.214	2.202
15-18	0.232	0.533	1.266	2.350
15-22	0.254	0.563	1.304	2.362
15-26	0.256	0.559	1.284	2.358
15-30	0.258	0.557	1.278	2.364
15-34	0.256	0.547	1.270	2.344
15-38	0.246	0.519	1.224	2.318
15-42	0.262	0.561	1.278	2.314
15-46	0.252	0.561	1.284	2.306
15-50	0.248	0.539	1.312	2.432
15-54	0.256	0.559	1.308	2.418
11-10	0.250	0.535	1.295	2.400
11-14	0.252	0.553	1.304	2.462
11-18	0.244	0.533	1.302	2.525
11-22	0.256	0.539	1.268	2.368
11-26	0.248	0.559	1.316	2.428
11-30	0.256	0.539	1.252	2.336
11-34	0.246	0.517	1.204	2.224
11-38	0.242	0.501	1.149	2.158
11-42	0.250	0.541	1.270	2.280
11-46	0.258	0.559	1.276	2.364
11-50	0.250	0.547	1.280	2.412
7-14	0.272	0.585	1.330	2.442
7-18	0.252	0.545	1.256	2.318
7-22	0.246	0.547	1.272	2.362
7-26	0.244	0.537	1.252	2.380
7-30	0.260	0.551	1.270	2.326
7-34	0.254	0.553	1.328	2.502
7-38	0.246	0.537	1.240	2.270
7-42	0.275	0.570	1.325	2.485
7-46	0.246	0.545	1.278	2.418
3-18	0.252	0.543	1.246	2.348
3-22	0.248	0.535	1.238	2.272
3-26	0.258	0.563	1.330	2.468
3-30	0.248	0.545	1.264	2.330
3-34	0.258	0.565	1.326	2.460
3-38	0.254	0.545	1.274	2.438
3-42	0.250	0.545	1.266	2.328

TABLE 4.2

Average Scram Insertion Time Comparisons

Notch From Fully Inserted	Average Scram Insertion After 1993 Outage	Times (SEC) Tech Spec Limit
45	0.2515	0.43
39	0.5491	0.86
25	1.2843	1.93
5	2.3872	3.49

5.0 SHUTDOWN MARGIN TEST

5.1 Shutdown Margin Test Abstract

The purpose of this test is to demonstrate that the reactor can be made subcritical with a shutdown margin of 0.38% delta K/K at any time in the cycle with the strongest operable control rod fully withdrawn. The shutdown margin demonstration was performed using the in-sequence critical method.

The shutdown margin test shall be considered acceptable if the amount of shutdown margin is greater than that required by Technical Specification.

5.2 Shutdown Margin Test Results

The Shutdown Margin test was performed during the initial criticality. Forty-four rods were fully withdrawn and 8 rods were withdrawn to position 4 to make the reactor critical. Values obtained from the Cycle Management Report (CMR) and actual plant data (See Figure 5.1) were used in the equation given below to determine the Shutdown Margin.

SDM = (K-Crit) - (K-SRO) + (K-Temp) - (K-Per) - (R)

Where;

SDM	= Shutdown Margin
K-Crit	= Predicted eigenvalue for the total number of rod notches withdrawn (CMR)
	= 1.0009 K effective
K-SRO	= Eigenvalue with strongest rod out (CMR)
	= .9801 K effective
K-Temp	= Moderator Temperature Correction factor (CMR)
	= -.00135 delta K (147°F)
K-Per	= Period Correction factor (CMR)
	= .00017 delta K (394.6 sec)
R	= Maximum decrease in SDM from Beginning of Cycle (CMR)
	= .0018 delta K

Inserting these values into the SDM equation resulted in a SDM of 1.748% delta K/K, well above the Technical Specification acceptance criteria of $\geq$.38% delta K/K.

FIGURE 5.1

SHUTDOWN MARGIN TEST ROD PATTERN

Blank = Position 00

	02	06	10	14	18	22	26	30
59						4		
55					48		48	
51								
47			48		48		48	
43		4						
39	48		48		48		48	
35								
31	48		48		48		48	

Moderator Temperature: 147°F

Number of Notches: 2144

Reactor Critical: 395 second period

6.0 COLD CRITICAL COMPARISON AND REACTIVITY ANOMALY

6.1 Cold Critical Comparison Test Abstract

The cold critical control pattern was checked against predicted conditions to ensure that a reactivity anomaly of greater than 1% did not exist. The predicted rod inventory at -1% ΔK and +1% ΔK around the critical position was calculated. The rod inventory was adjusted for actual moderator temperature and period. Figure 6.1 shows the actual rod pattern.

6.2 Cold Critical Comparison Test Results

Figure 6.1 contains the actual cold critical control rod pattern (2144 notches). The predicted notches to achieve criticality are 2208 notches. An additional one percent reactivity (1% Δk) was calculated to be 2304 notches and one percent less reactivity (-1% ΔK) was calculated to be 1344 notches. The difference between the observed and predicted control rod inventories is less than one percent in reactivity. Results of the test are within the criteria specified in Technical Specifications.

COLD CRITICAL COMPARISON

ACTUAL ROD INVENTORY	2144 notches
PREDICTED ROD INVENTORY -1%	1344 notches
PREDICTED ROD INVENTORY +1%	2304 notches

6.3 Reactivity Anomalies Test Abstract

The Reactivity Anomaly test was performed during full power operation. The actual control rod notch inventory was corrected for operating conditions including core thermal power, core flow, inlet subcooling and pressure. This corrected notch inventory was compared against the predicted rod inventory.

6.4 Reactivity Anomalies Test Results

Figure 6.2 shows the actual control rod pattern. The predicted notches at 681 MWD/st core exposure are 450. An additional 1% ΔK represents 946 notches while 1% ΔK less represents -46 notches. The actual notches were 476. The difference between the corrected and predicted control rod inventories is less than one percent in reactivity. Results of the test are within the criteria specified in Technical Specifications.

FIGURE 6.1
COLD CRITICAL CONTROL ROD PATTERN

Blank = Position 00

Predicted Notches: 2208

	02	06	10	14	18	22	26	30
59						4		
55					48		48	
51				4				4
47			48		48		48	
43		4				4		
39	48		48		48		48	
35				4				4
31	48		48		48		48	

Actual Notches: 2144

	02	06	10	14	18	22	26	30
59						4		
55					48		48	
51								
47			48		48		48	
43		4						
39	48		48		48		48	
35								
31	48		48		48		48	

FIGURE 6.2
REACTIVITY ANOMALIES MARGIN TEST ROD PATTERN

Blank = Position 48

	02	06	10	14	18	22	26	30
59								
55								16
51								
47								
43								
39						12		12
35								
31		16				12		

7.0 ROUTINE SURVEILLANCE COMPARISON WITH PREDICTED MEASUREMENTS

7.1 Routine Surveillance Comparison Description

Thermal limits in the core are monitored by the Process Computer. Off line predictive computer calculations/models were used to predict thermal limits corresponding to specific plant operating conditions throughout the cycle.

7.2 Routine Surveillance Comparison Test - Results

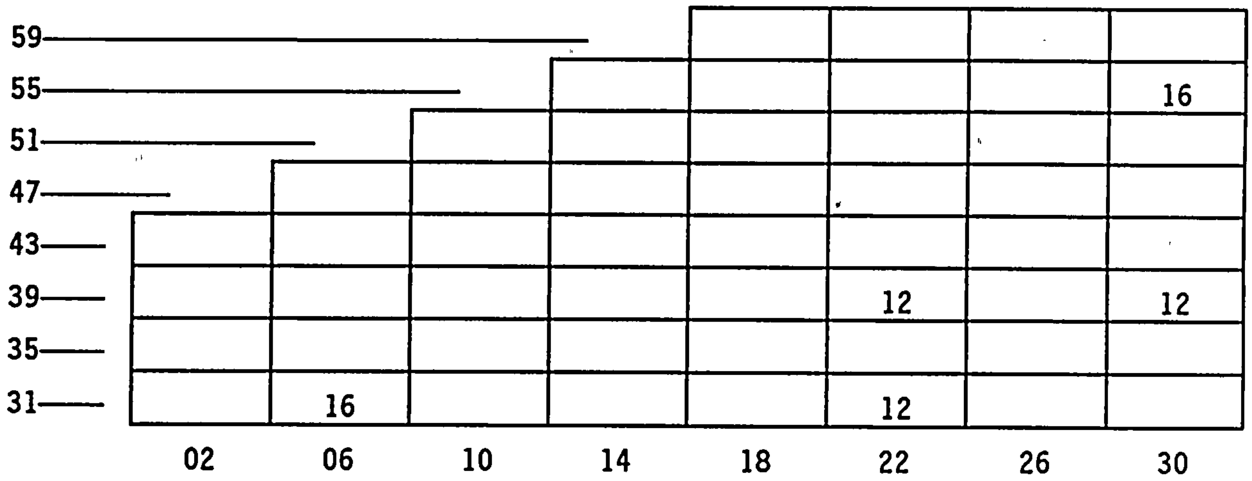
The predicted core performance is shown in Figure 7.1. The actual core performance is shown in Figure 7.2. Listed below is a comparison.

	Predicted	Actual	Units	Delta (%)
Power	3323	3321	MWt	-0.1%
Exposure	200	200.1	MWD/st	0.1%
Flow	108.5	107.1	Mlb/hr	-1.3%
MFLCPR	0.802	0.808	--	0.7%
MFLPD	0.817	0.777	--	-4.9%
MAPRAT	0.848	0.790	--	-6.8%

The actual axial power distribution in the core was found to be peaked higher than predicted. This results in core thermal limit margins for MAPRAT and MFLPD being greater than predicted. This deviation is conservative and not greater than that which is considered acceptable.

FIGURE 7.1
ROUTINE SURVEILLANCE COMPARISON
PREDICTED CORE PERFORMANCE

Blank = Position 48



CORE POWER 3323 MWt (100%)

EXPOSURE 200 MWD/st

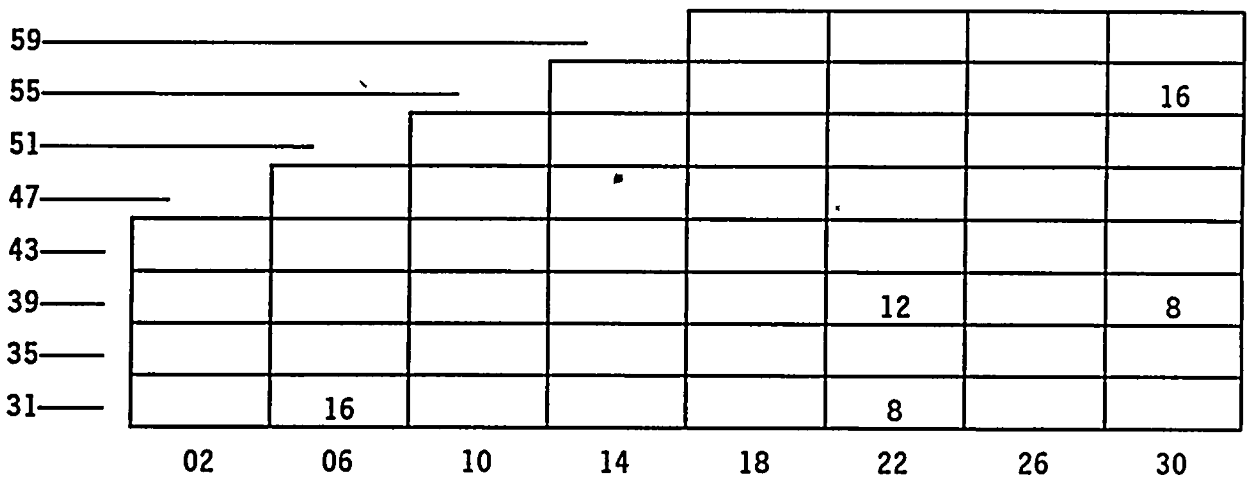
CORE FLOW 108.5 Mlb/hr (98%)

THERMAL PERFORMANCE

MAXIMUM FRACTION LIMITING CRITICAL POWER RATIO	0.802
MAXIMUM FRACTION LIMITING POWER DENSITY RATIO	0.817
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO	0.848

FIGURE 7.2
ROUTINE SURVEILLANCE COMPARISON
ACTUAL CORE PERFORMANCE

Blank = Position 48



DECEMBER 16, 1993	DATA	
CORE POWER	3321 MWt (99.9%)	
EXPOSURE	200.1 MWD/st	
CORE FLOW	107.1 Mlb/hr (98.7%)	
THERMAL PERFORMANCE		
MAXIMUM FRACTION LIMITING CRITICAL POWER RATIO		0.808
MAXIMUM FRACTION LIMITING POWER DENSITY RATIO		0.777
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO		0.790

8.0 TRAVERSING INCORE PROBE UNCERTAINTY

8.1 Traversing Incore Probe (TIP) Uncertainty Abstract

The control rod pattern is adjusted so that there are eleven pairs of symmetrical TIP locations in the core. TIP data is collected and the total uncertainty of the TIP data is calculated. A high level of uncertainty indicates that the core power distribution is not uniform or that the TIP tubing is not symmetrically located.

8.2 Traversing Incore Probe (TIP) Uncertainty Test Results Comparison

The TIP uncertainty test was performed under the core operating conditions shown on Figure 8.1. The results of the test showed a total TIP uncertainty of 3.05%. This is well within the Acceptance Criteria of 6%. This is to be expected for a plant with thermal TIP's.

Date/Time: December 16, 1993
Core Power: 3320 MWt
Recirc. Flow: 106.7 Mlb/hr

	Actual	Acceptance Criteria
TIP Uncertainty	3.05%	<6.0%

FIGURE 8.1
ROUTINE SURVEILLANCE COMPARISON
TIP UNCERTAINTY

Blank = Position 48

	02	06	10	14	18	22	26	30
59								
55								16
51								
47								
43								
39						12		8
35								
31		16				8		

CORE POWER 3320 MWt (99.9%)

EXPOSURE 200 MWD/st

CORE FLOW 106.7 Mlb/hr (98.4%)

9.0 CORE VERIFICATION

Core verification consisted of videotaping the loaded reactor core and verifying that the orientation and location of the fuel bundles was consistent with the final core loading map. Three passes with a television camera were recorded for the core verification. During the first pass the orientation was confirmed. During the second pass, the bundle identification numbers were checked. During the third pass, the seating was checked.

Core loading was found to be acceptable with zero deficiencies.