Surry
Unit 1 Cycle 16
Startup Physics
Tests Report

Nuclear Analysis and Fuel
Nuclear Engineering & Services

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SURRY UNIT 1, CYCLE 16
STARTUP PHYSICS TESTS REPORT

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PREFACE

This report presents the analysis and evaluation of the physics tests, which were performed to verify that the Surry Unit 1, Cycle 16 core could be operated safely, and makes an initial evaluation of the performance of the core. It is not the intent of this report to discuss the particular methods of testing or to present the detailed data taken. Standard testing techniques and methods of data analysis were used. The test data, results and evaluations, together with the detailed startup procedures, are on file at the Surry Power Station. Therefore, only a cursory discussion of these items is included in this report. The analyses presented include a brief summary of each test, a comparison of the test results with design predictions, and an evaluation of the results.

The Surry Unit 1, Cycle 16 startup physics test results and evaluation sheets are included as an appendix to provide additional information on the startup test results. Each data sheet provides the following information: 1) test identification, 2) test conditions (design), 3) test conditions (actual), 4) test results, 5) acceptance criteria, and 6) comments concerning the test. These sheets provide a compact summary of the startup test results in a consistent format. The design test conditions and design values (at design conditions) of the measured parameters were completed prior to the startup physics testing. The entries for the design values were based on the calculations performed by Virginia Electric and Power Company's Nuclear Analysis and Fuel Group[1]. During the tests, the data sheets were used as guidelines both to verify that the proper test conditions were met and to facilitate the preliminary comparison between measured and predicted test results, thus enabling a quick identification of possible problems occurring during the tests.

SECTION 1

INTRODUCTION AND SUMMARY

On October 19, 1998 Unit No. 1 of the Surry Power Station shutdown for its sixteenth refueling. During this shutdown, 57 of the 157 fuel assemblies in the core were replaced with 56 fresh assemblies and one once-burned assembly. The Cycle 16 core consists of 6 sub-batches of fuel: two once burned batches from Cycle 15 (batches 17A and 17B); two twice-burned batches from Cycle 14 (batches 16A and 16B); and two fresh batches (batches 18A and 18B). The single once-burned assembly is from Cycle 14 (batch 16A).

The core loading pattern and the design parameters for each sub-batch are shown in Figure 1.1. Beginning of cycle fuel assembly burnups are given in Figure 1.2. The incore thimble locations available during startup physics testing are identified in Figure 1.3. Figure 1.4 identifies the location and number of burnable poison rods and flux suppression insert locations for Cycle 16, while figure 1.5 identifies the control rod locations.

The cycle 16 core achieved initial criticality at 0527 on November 19, 1998. Prior to and following criticality, startup physics tests were performed as outlined in Table 1.1. A summary of the physics test results follows:

1. The measured drop time of each control rod was within the 2.4 second limit of Technical Specification 3.12.C.1.

2. The reference control rod bank was measured with the dilution method, and the result was within 0.2% of the design prediction. Individual control rod bank worths were measured using the rod swap technique[2,3] and the results were within –2.1% of the design predictions. The sum of the individual measured control rod bank worths was within –0.8% of the design prediction. All results were within the design tolerance of ±15% for individual bank worths (±10% for the rod swap reference bank worth) and the design tolerance of ±10% for the sum of the individual control rod bank worths.

3. Measured critical boron concentrations for two control bank configurations were within 13 ppm of the design predictions. The all-rods-out (ARO) result was within the 50 ppm design tolerance, and met the Technical Specification 4.10.A criterion that the overall core reactivity balance shall be within ±1% Δk/k of the design prediction. The reference bank in critical boron concentration was within its design tolerance.

4. The boron worth coefficient measurement was within 3.4% of the design prediction, which is within the design tolerance of ±10%.

5. The measured isothermal temperature coefficient (ITC) for the all-rods-out configuration was within 0.92 pcm/°F of the design prediction. This result is within the design tolerance of ±3 pcm/°F. The measured ITC was –0.83 pcm/°F. When the Doppler temperature coefficient (-1.70 pcm/°F) and a 0.5 pcm/°F uncertainty are accounted for in the +6.0 pcm/°F MTC limit of Core Operating Limits Report (COLR) 2.1, the MTC acceptance criteria is satisfied as long as the ITC is less positive than +3.80 pcm/°F.

6. Measured core power distributions were within established acceptance criteria and COLR limits. The average relative assembly power distribution measured/predicted percent difference was 1.4% or less for the three initial power ascension flux maps. The heat flux hot channel factors, F-Q(Z), and enthalpy rise hot channel factors, F-DH(N), were within the limits of COLR Sections 2.3 and 2.4, respectively.

In summary, all startup physics test results were acceptable. Detailed results, specific design tolerances and acceptance criteria for each measurement are presented in the following sections of this report.

Table 1.1

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
CHRONOLOGY OF TESTS

Test	Date	Time	Power	Reference Procedure
Hot Rod Drop-Hot Full Flow	11/18/98	1150	HSD	1-NPT-RX-014
Zero Power Testing Range	11/19/98	0640	HZP	1-NPT-RX-008
Reactivity Computer Checkout	11/19/98	0729	HZP	1-NPT-RX-008
Boron Endpoint – ARO	11/19/98	0800	HZP	1-NPT-RX-008
Temperature Coefficient - ARO	11/19/98	1027	HZP	1-NPT-RX-008
Bank B Worth	11/19/98	1114	HZP	1-NPT-RX-008
Boron Endpoint - B in	11/19/98	1515	HZP	1-NPT-RX-008
Bank D Worth - Rod Swap	11/19/98	1550	HZP	1-NPT-RX-008
Bank C Worth - Rod Swap	11/19/98	1613	HZP	1-NPT-RX-008
Bank A Worth - Rod Swap	11/19/98	1630	HZP	1-NPT-RX-008
Bank SB Worth - Rod Swap	11/19/98	1700	HZP	1-NPT-RX-008
Bank SA Worth - Rod Swap	11/19/98	1735	HZP	1-NPT-RX-008
Flux Map – 28% Power Peaking Factor Verification & Power Range Calibration	11/21/98	1512	28%	1-NPT-RX-002 1-NPT-RX-005 1-NPT-RX-008
Flux Map – 69% Power Peaking Factor Verification & Power Range Calibration	11/26/98	0300	69%	1-NPT-RX-002 1-NPT-RX-005 1-NPT-RX-008
Flux Map – 100% Power Peaking Factor Verification & Power Range Calibration	12/02/98	1000	100%	1-NPT-RX-002 1-NPT-RX-005 1-NPT-RX-008

Figure 1.1

SURRY UNIT 1 - CYCLE 16
CORE LOADING MAP

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						16B 57A	16A 21A	16B 37A							1
				16A 03A	16B 42A	18B 36C	17B 51B	18B 54C	16B 52A	16A 15A					2
			16B 55A	18A 24C	18B 45C	17A 01B	18A 13C	17A 13B	18B 56C	18A 32C	16B 51A				3
		16B 63A	16B 38A	18B 55C	17A 09B	18A 27C	17A 07B	18A 08C	17A 16B	18B 40C	16B 45A	16B 47A			4
	16A 33A	18A 21C	18B 34C	17B 35B	18A 11C	17A 17B	18A 07C	17A 31B	18A 16C	17B 57B	18B 48C	18A 30C	16A 22A		5
	16B 53A	18B 43C	17A 18B	18A 15C	17B 54B	17B 42B	17A 03B	17B 34B	17B 40B	18A 23C	17A 11B	18B 38C	16B 39A		6
16B 41A	18B 49C	17A 28B	18A 26C	17A 27B	17B 53B	17B 41B	17B 48B	17B 46B	17B 52B	17A 30B	18A 04C	17A 05B	18B 44C	16B 58A	7
16A 12A	17B 49B	18A 29C	17A 04B	18A 20C	17A 12B	17B 36B	16A 16A	17B 47B	17A 19B	18A 31C	17A 32B	18A 01C	17B 43B	16A 11A	8
16B 62A	18B 51C	17A 06B	18A 05C	17A 26B	17B 39B	17B 56B	17B 58B	17B 44B	17B 45B	17A 25B	18A 12C	17A 20B	18B 50C	16B 44A	9
	16B 40A	18B 53C	17A 22B	18A 19C	17B 38B	17B 33B	17A 08B	17B 55B	17B 37B	18A 14C	17A 21B	18B 46C	16B 48A		10
	16A 09A	18A 06C	18B 47C	17B 60B	18A 22C	17A 24B	18A 02C	17A 29B	18A 28C	17B 59B	18B 35C	18A 09C	16A 17A		11
		16B 46A	16B 64A	18B 39C	17A 23B	18A 25C	17A 15B	18A 03C	17A 10B	18B 42C	16B 54A	16B 56A			12
			16B 50A	18A 17C	18B 37C	17A 14B	18A 18C	17A 02B	18B 33C	18A 10C	16B 61A				13
				16A 31A	16B 60A	18B 52C	17B 50B	18B 41C	16B 43A	16A 26A					14
						16B 59A	16A 27A	16B 49A							15

☐ ===> Batch
===> Assembly ID

FUEL ASSEMBLY DESIGN PARAMETERS

	SUB-BATCH					
	16A	16B	17A	17B	18A	18B
INITIAL ENRICHMENT (W/O U-235)	3.81	4.01	3.81	4.01	4.11	4.26
BURNUP AT BOC 16 (MWD/MTU)	37996	38168	21779	21040	0	0
ASSEMBLY TYPE	15x15	15x15	15x15	15x15	15x15	15x15
NUMBER OF ASSEMBLIES	13	28	32	28	32	24
FUEL RODS PER ASSEMBLY	204	204	204	204	204	204

Figure 1.2

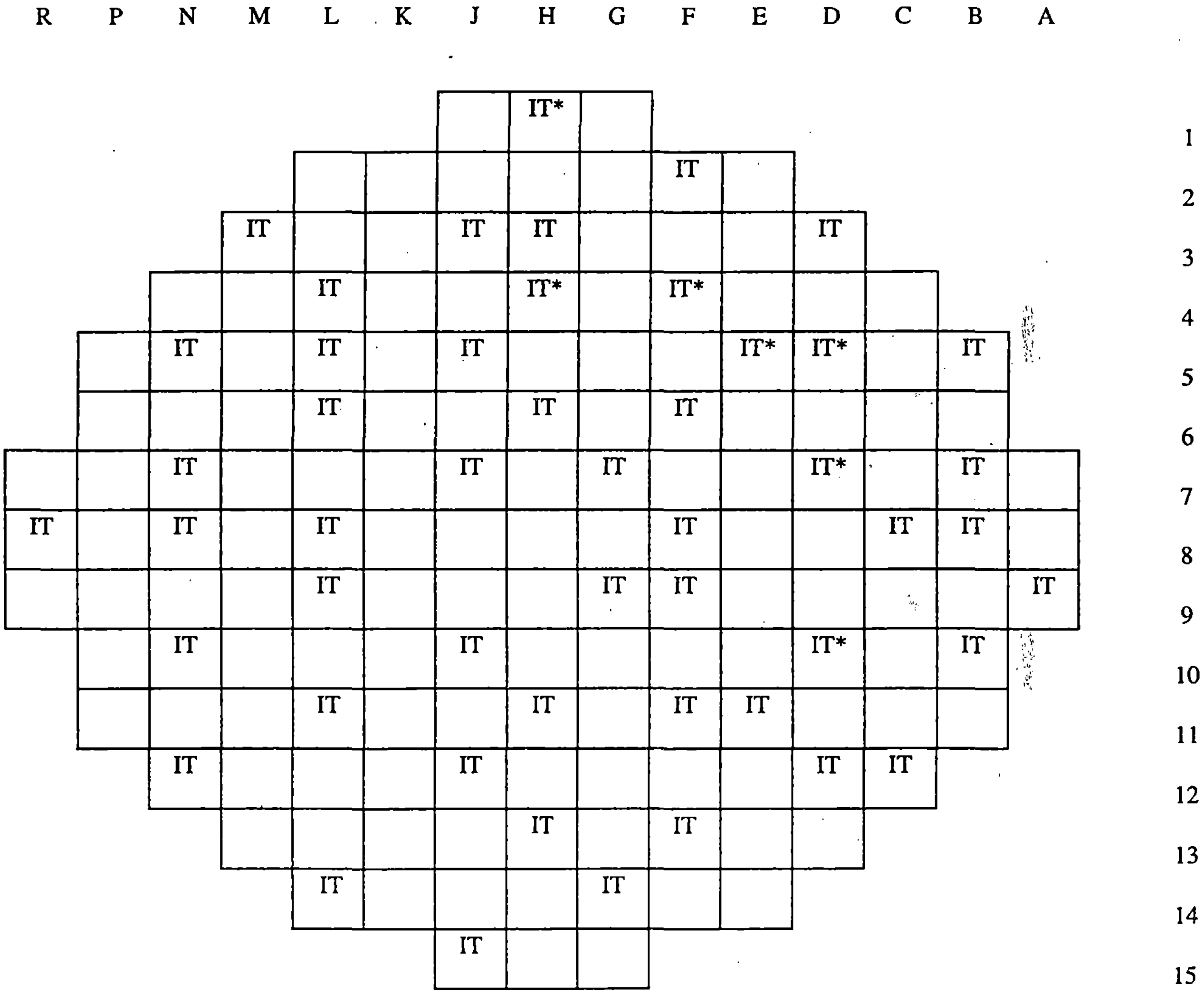
SURRY UNIT 1 - CYCLE 16
BEGINNING OF CYCLE FUEL ASSEMBLY BURNUPS

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						57A 38.61	21A 40.18	37A 38.08							1
				03A 38.84	42A 32.81	36C 0.00	51B 21.77	54C 0.00	52A 33.33	15A 38.46					2
			55A 42.96	24C 0.00	45C 0.00	01B 17.59	13C 0.00	13B 17.99	56C 0.00	32C 0.00	51A 42.37				3
		63A 42.27	38A 37.17	55C 0.00	09B 23.37	27C 0.00	07B 22.83	08C 0.00	16B 23.40	40C 0.00	45A 37.33	47A 42.38			4
	33A 38.39	21C 0.00	34C 0.00	35B 22.13	11C 0.00	17B 23.79	07C 0.00	31B 23.64	16C 0.00	57B 21.57	48C 0.00	30C 0.00	22A 38.25		5
	53A 33.90	43C 0.00	18B 23.38	15C 0.00	54B 21.83	42B 18.46	03B 22.59	34B 18.27	40B 21.61	23C 0.00	11B 22.80	38C 0.00	39A 33.04		6
41A 38.53	49C 0.00	28B 18.20	26C 0.00	27B 23.54	53B 18.64	41B 21.96	48B 22.57	46B 21.65	52B 18.45	30B 23.41	04C 0.00	05B 17.50	44C 0.00	58A 38.65	7
12A 40.43	49B 21.75	29C 0.00	04B 22.56	20C 0.00	12B 22.54	36B 23.32	16A 23.56	47B 22.61	19B 21.99	31C 0.00	32B 22.67	01C 0.00	43B 22.01	11A 40.53	8
62A 38.33	51C 0.00	06B 17.60	05C 0.00	26B 23.55	39B 18.43	56B 21.36	58B 22.52	44B 21.97	45B 18.57	25B 23.68	12C 0.00	20B 18.01	50C 0.00	44A 38.52	9
	40A 33.20	53C 0.00	22B 23.26	19C 0.00	38B 21.88	33B 18.46	08B 22.37	55B 18.37	37B 21.21	14C 0.00	21B 23.20	46C 0.00	48A 33.77		10
	09A 38.41	06C 0.00	47C 0.00	60B 21.75	22C 0.00	24B 23.31	02C 0.00	29B 23.50	28C 0.00	59B 22.04	35C 0.00	09C 0.00	17A 38.26		11
		46A 42.90	64A 37.89	39C 0.00	23B 22.98	25C 0.00	15B 22.80	03C 0.00	10B 23.25	42C 0.00	54A 37.37	56A 42.26			12
			50A 42.22	17C 0.00	37C 0.00	14B 17.64	18C 0.00	02B 17.61	33C 0.00	10C 0.00	61A 42.84				13
				31A 38.27	60A 33.69	52C 0.00	50B 21.62	41C 0.00	43A 33.45	26A 38.70					14
						59A 38.65	27A 40.36	49A 38.44							15

===> Assembly ID
===> Assembly Burnup (MWD/MTU)

Figure 1.3

SURRY UNIT 1 - CYCLE 16
INCORE THIMBLE LOCATIONS



R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
							IT*								1
									IT						2
			IT			IT	IT				IT				3
				IT			IT*		IT*						4
		IT		IT		IT				IT*	IT*		IT		5
				IT			IT		IT						6
		IT				IT		IT			IT*		IT		7
IT		IT		IT					IT			IT	IT		8
				IT				IT	IT					IT	9
		IT				IT					IT*		IT		10
				IT			IT		IT	IT					11
		IT				IT					IT	IT			12
							IT		IT						13
				IT				IT							14
						IT									15

IT – Incore Thimble

* - Unavailable Location

Figure 1.4

SURRY UNIT 1 - CYCLE 16
BURNABLE POISON AND FLUX SUPPRESSION INSERT LOCATIONS

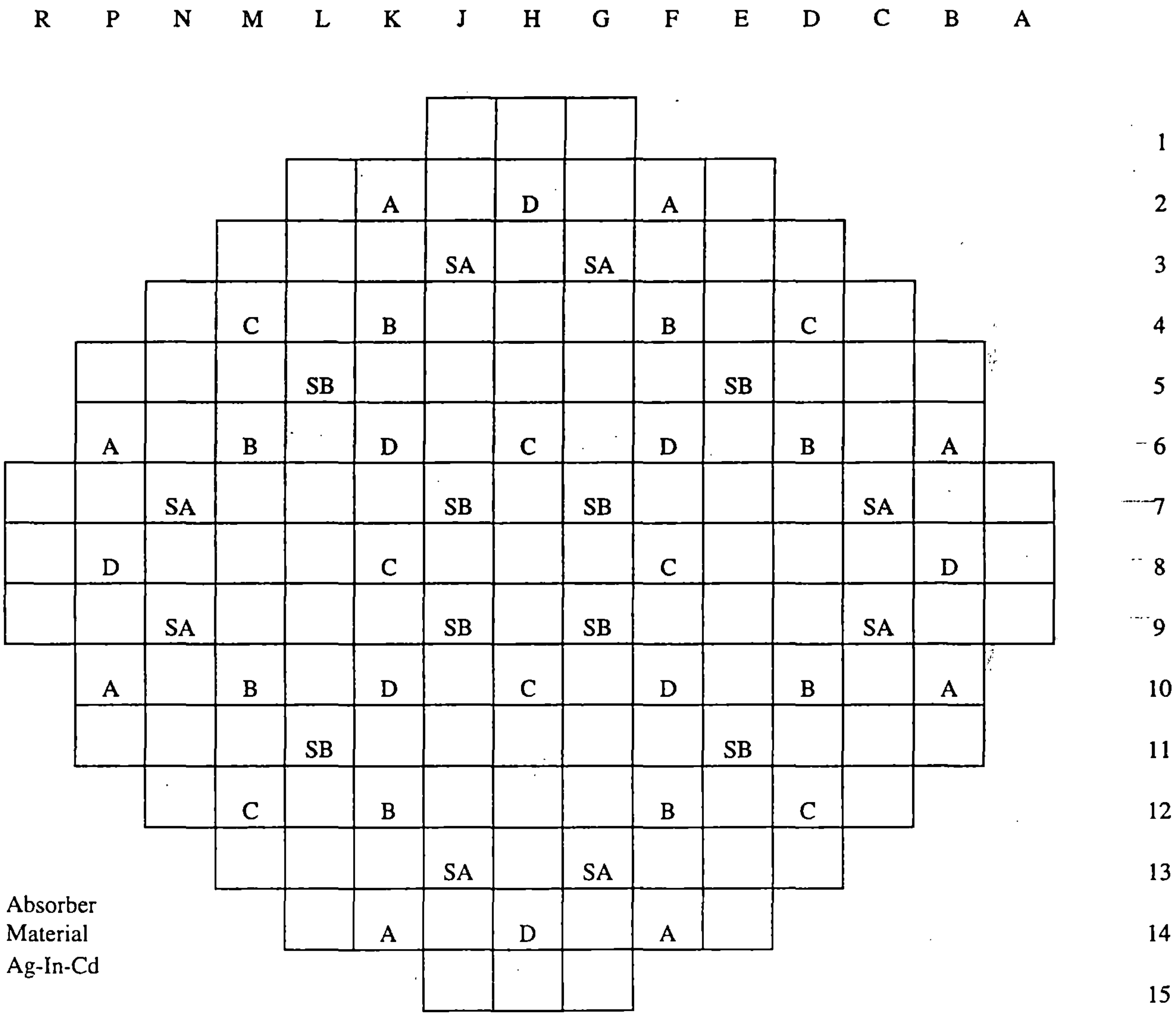
R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
							FSIS FS005								1
						3P BP643		3P BP650							2
			FSIL FS014	5P BP676	20P BP659		20P BP631		20P BP666	5P BP675	FSIL FS007				3
		FSIL FS008		20P BP660		20P BP638		20P BP635		20P BP665		FSIL FS009			4
		5P BP678	20P BP661		20P BP654		20P BP630		20P BP657		20P BP664	5P BP682			5
		20P BP662		20P BP655						20P BP656		20P BP663			6
	3P BP644		20P BP642								20P BP641		3P BP649		7
FSIS FS002		20P BP634		20P BP629						20P BP628		20P BP633		FSIS FS003	8
	3P BP645		20P BP639								20P BP637		3P BP648		9
		20P BP674		20P BP658						20P BP653		20P BP667			10
		5P BP677	20P BP673		20P BP651		20P BP627		20P BP652		20P BP668	5P BP681			11
		FSIL FS010		20P BP672		20P BP636		20P BP640		20P BP669		FSIL FS011			12
			FSIL FS012	5P BP680	20P BP671		20P BP632		20P BP670	5P BP679	FSIL FS013				13
						3P BP646		3P BP647							14
							FSIS FS004								15

3P – 3 BURNABLE POISON ROD CLUSTER
5P – 5 BURNABLE POISON ROD CLUSTER
20P – 20 BURNABLE POISON ROD CLUSTER
FSIL – FLUX SUPPRESSION INSERT (LONG)
FSIS – FLUX SUPPRESSION INSERT (SHORT)

xxP	-	#OF BP RODS
BP###, FS###	-	BP ASSEMBLY ID, FLUX SUPPRESSION INSERT ID

Figure 1.5

SURRY UNIT 1 - CYCLE 16
CONTROL ROD LOCATIONS



	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2						A		D		A					
3							SA		SA						
4				C		B				B		C			
5					SB						SB				
6		A		B		D		C		D		B		A	
7			SA				SB		SB				SA		
8		D				C				C				D	
9			SA				SB		SB				SA		
10		A		B		D		C		D		B		A	
11					SB						SB				
12				C		B				B		C			
13							SA		SA						
14						A		D		A					
15															

Absorber
Material
Ag-In-Cd

Function	Number of Clusters
Control Bank D	8
Control Bank C	8
Control Bank B	8
Control Bank A	8
Shutdown Bank SB	8
Shutdown Bank SA	8

SECTION 2

CONTROL ROD DROP TIME MEASUREMENTS

The drop time of each control rod was measured at hot full-flow reactor coolant system (RCS) conditions (Tavg of 547 +/- 5^0 F) in order to verify that the time from initiation of the rod drop to the entry of the rod into the dashpot was less than or equal to the maximum allowed by Technical Specification 3.12.C.1.

The rod drop times were measured by withdrawing three banks to their fully withdrawn position and dropping all 24 control rods within the three banks by opening the reactor trip breakers. This allowed the rods to drop into the core as they would during a plant trip with the exception that during a plant trip all six banks would drop simultaneously. The Individual Rod Position Indication (IRPI) secondary coil voltage signals were recorded for each rod in the bank to determine each rod's drop time. A stationary gripper coil voltage was also measured as confirmation of the initiation of the reactor trip breaker opening. This procedure was repeated for the remaining three banks.

As shown on the sample rod drop trace in Figure 2.1, the initiation of the rod drop is indicated by the decay of the stationary gripper coil voltage when the reactor trip breakers reopened. As the rod drops, a voltage is induced in the IRPI secondary coil. The magnitude of this voltage is a function of control rod velocity. As the rod enters the dashpot region of the guide tube, its velocity slows causing a voltage decrease in the IRPI coil. This voltage reaches a minimum when the rod reaches the bottom of the dashpot. Subsequent variations in the trace are caused by rod bouncing.

The measured drop times for each control rod are recorded on Figure 2.2. The slowest, fastest, and average drop times are summarized in Table 2.1. Technical Specification 3.12.C.1 specifies a maximum rod drop time from loss of stationary gripper coil voltage to dashpot entry of 2.4 seconds with the RCS at hot, full flow conditions. The test results satisfy this limit.

Table 2.1

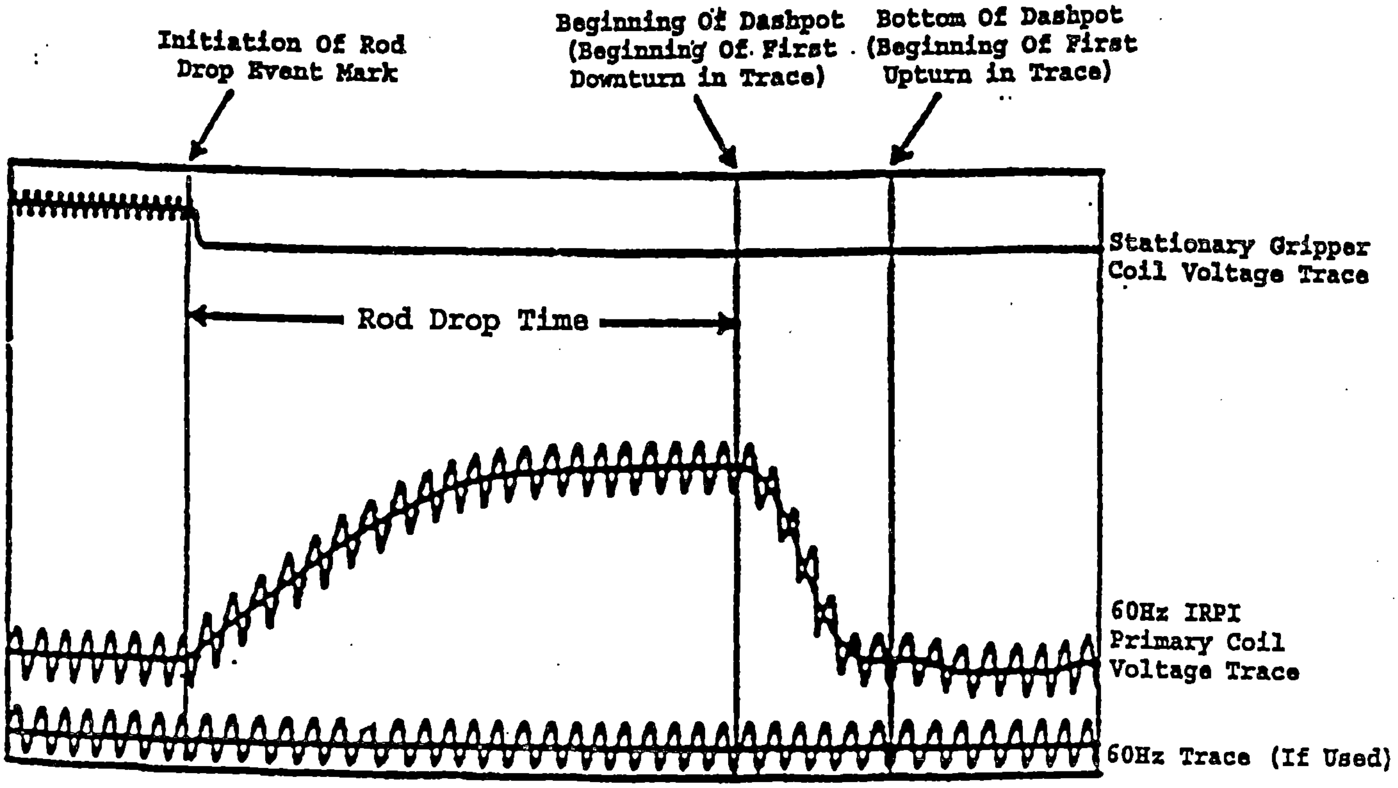
SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
HOT ROD DROP TIME SUMMARY

ROD DROP TIME TO DASHPOT ENTRY

SLOWEST ROD	FASTEST ROD	AVERAGE TIME
F-8 1.39	C-9/F-2 1.23 sec.	1.29 sec.

Figure 2.1

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
TYPICAL ROD DROP TRACE



ROD DROP TIME MEASUREMENT

Figure 2.2

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ROD DROP TIME - HOT FULL FLOW CONDITIONS

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
															1
					1.30		1.27		1.23						2
						1.28		1.28							3
			1.27		1.26				1.29		1.31				4
				1.29						1.29					5
	1.26		1.26		1.28		1.30		1.27		1.28		1.36		6
		1.27				1.31		1.27				1.28			7
	1.29				1.31				1.39				1.32		8
		1.31				1.28		1.30				1.23			9
	1.27		1.29		1.28		1.31		1.31		1.30		1.30		10
				1.27						1.29					11
			1.27		1.26				1.28		1.32				12
						1.25		1.29							13
					1.32		1.27		1.35						14
															15

x.xx ===> Rod drop time to dashpot entry (sec.)

SECTION 3

CONTROL ROD BANK WORTH MEASUREMENTS

Control rod bank worths were measured for the control and shutdown banks using the rod swap technique[2,3]. The initial step of the rod swap method diluted the predicted most reactive control rod bank (hereafter referred to as the reference bank) into the core and measured its reactivity worth using conventional test techniques. The reactivity changes resulting from the reference bank movements were recorded continuously by the reactivity computer and were used to determine the differential and integral worth of the reference bank. For Cycle 16, Control Bank B was used as the reference bank.

After the completion of the reference bank reactivity worth measurement, the reactor coolant system temperature and boron concentration were stabilized with the reactor near critical and the reference bank near full insertion. Initial statepoint data for the rod swap maneuver were obtained by moving the reference bank to its fully inserted position with all other banks fully withdrawn and recording the core reactivity and moderator temperature. From this point, a rod swap maneuver was performed by withdrawing the reference bank several steps and then one of the other control rod banks (i.e., a test bank) was inserted to balance the reactivity of the reference bank withdrawal. This sequence was repeated until the test bank was fully inserted and the reference bank was positioned such that the core was just critical or near the initial statepoint reactivity. This measured critical position (MCP) of the reference bank with the test bank fully inserted was used to determine the integral reactivity worth of the test bank. The core reactivity, moderator temperature, and the differential worth of the reference bank were recorded with the reference bank at the MCP. The rod swap maneuver then was repeated in reverse such that the reference bank again was fully inserted with the test bank fully withdrawn from the core. This rod swap process was then repeated for each of the other control and shutdown banks.

A summary of the test results is given in Table 3.1. As shown in this table and the Startup Physics Test Results and Evaluation Sheets given in the Appendix, all of the individual measured bank worths for the control and shutdown banks were within the design tolerance ($^{\pm}$10% for the reference bank, $^{\pm}$15% for test banks of worth greater than 600 pcm, and $^{\pm}$100 pcm for test banks of worth less than or equal to 600 pcm.). The sum of the individual measured rod bank worths was within –0.8% of the design prediction. This is well within the design tolerance of $^{\pm}$10% for the sum of the individual control rod bank worths.

The integral and differential reactivity worths of the reference bank (Control Bank B) are shown in Figures 3.1 and 3.2, respectively. The design predictions and the measured data are plotted together in order to illustrate their agreement. In summary, the measured rod worth values were satisfactory.

Table 3.1

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS CONTROL ROD BANK WORTH SUMMARY

BANK	MEASURED WORTH (PCM)	PREDICTED WORTH (PCM)	PERCENT DIFFERENCE (%) (M-P)/P X 100
B-Reference Bank	1374	1371	0.2
D	1070	1077	-0.6
C	712	727	-2.1
A	286	290	-1.4*
SB	1129	1129	0
SA	1010	1032	-2.1
Total Worth	5581	5626	-0.8

* Difference is less than 100 pcm.

Figure 3.1

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
CONTROL BANK B INTEGRAL ROD WORTH - HZP
ALL OTHER RODS WITHDRAWN

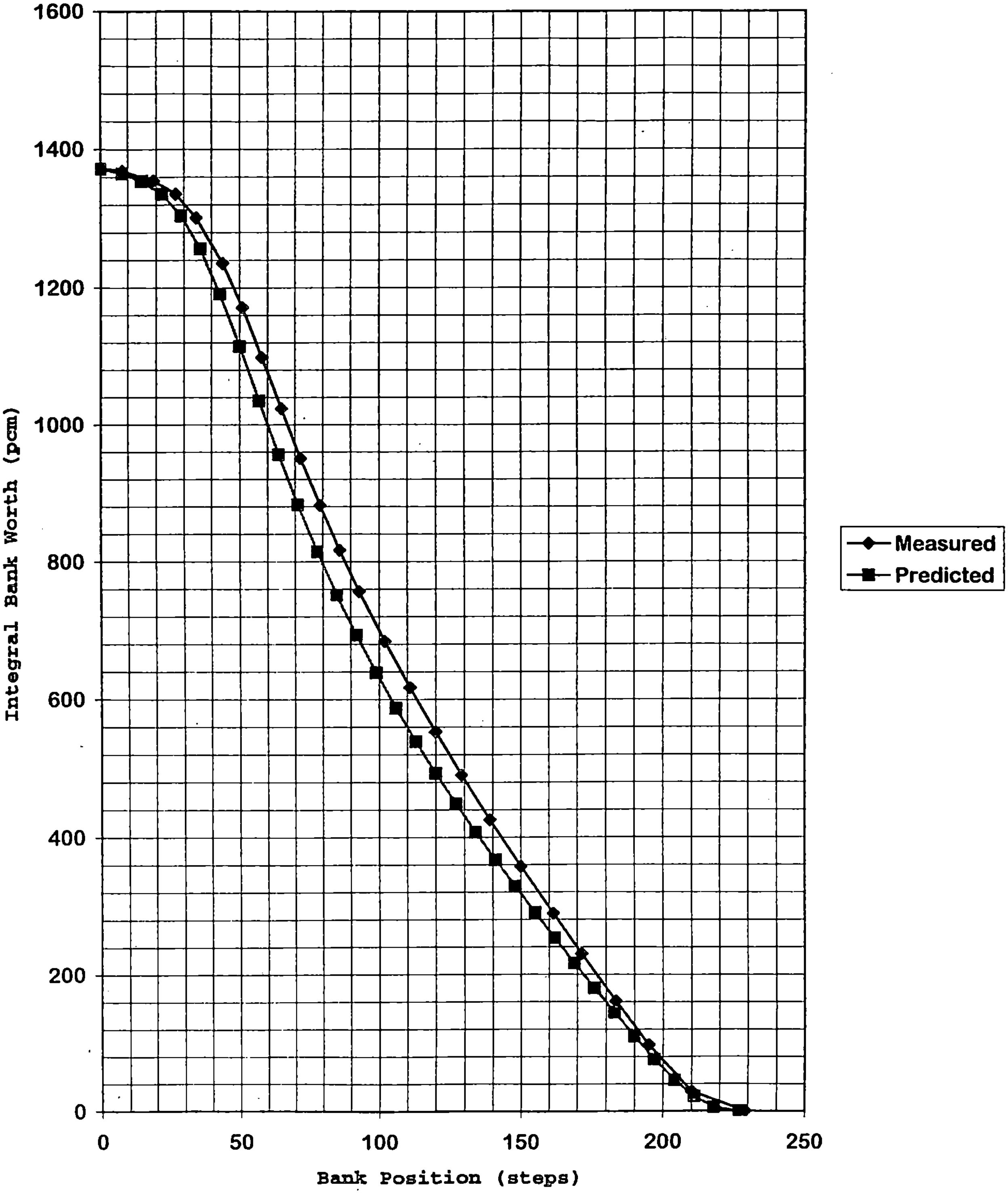
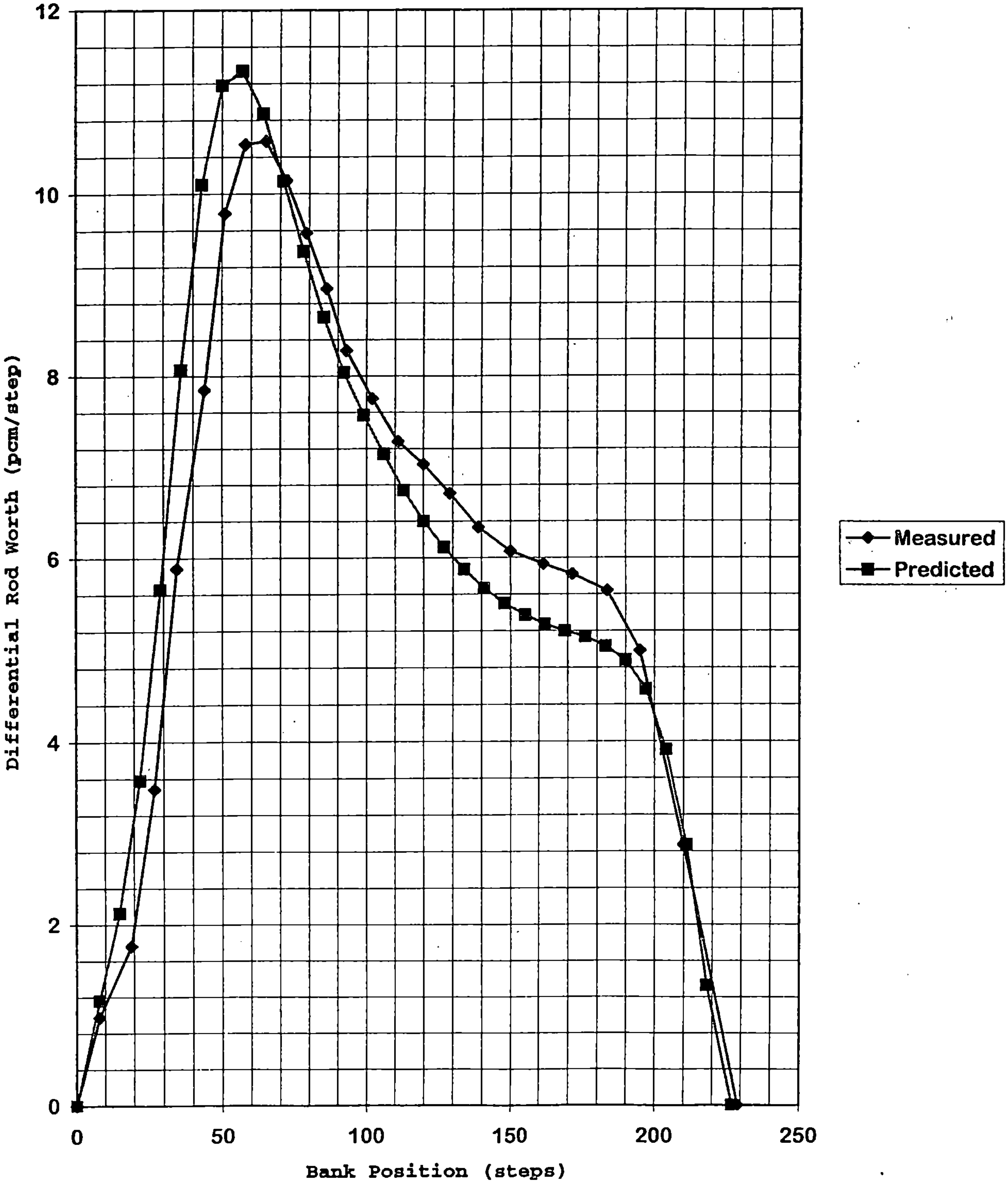


Figure 3.2

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
CONTROL BANK B DIFFERENTIAL ROD WORTH - HZP
ALL OTHER RODS WITHDRAWN



SECTION 4

BORON ENDPOINT AND WORTH MEASUREMENTS

Boron Endpoint

With the reactor critical at hot zero power, reactor coolant system (RCS) boron concentrations were measured at selected rod bank configurations to enable a direct comparison of measured boron endpoints with design predictions. For each critical boron concentration measurement, the RCS conditions were stabilized with the control banks at or very near a selected endpoint position. Adjustments to the measured critical boron concentration values were made to account for off-nominal control rod position and moderator temperature, if necessary.

The results of these measurements are given in Table 4.1. As shown in this table and in the Startup Physics Test Results and Evaluation Sheets given in the Appendix, the measured critical boron endpoint values were within their respective design tolerances. The all-rods-out (ARO) endpoint comparison to the predicted value met the requirements of Technical Specification 4.10.A regarding core reactivity balance. In summary, the boron endpoint results were satisfactory.

Boron Worth Coefficient

The measured boron endpoint values provide stable statepoint data from which the boron worth coefficient or differential boron worth (DBW) was determined. By relating each endpoint concentration to the integrated rod worth present in the core at the time of the endpoint measurement, the value of the DBW over the range of boron endpoint concentrations was obtained.

A summary of the measured and predicted DBW is shown in Table 4.2. As indicated in this table and in the Appendix, the measured DBW was well within the design tolerance of +/-10%. In summary, the measured boron worth coefficient was satisfactory.

Table 4.1

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS BORON ENDPOINTS SUMMARY

Control Rod Configuration	Measured Endpoint (ppm)	Predicted Endpoint (ppm)	Difference M-P (ppm)
ARO	1952	1939	13
B Bank In	1766	1760*	6

* The predicted endpoint for the B Bank In configuration was adjusted for the difference between the measured and predicted values of the endpoint taken at the ARO configuration as shown in the boron endpoint Startup Physics Test Results and Evaluation Sheet in the Appendix.

Table 4.2

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS BORON WORTH COEFFICIENT

Measured Boron Worth (pcm/ppm)	Predicted Boron Worth (pcm/ppm)	Percent Difference (%) (M-P)/P x 100
-7.38	-7.14	3.4

SECTION 5

TEMPERATURE COEFFICIENT MEASUREMENT

The isothermal temperature coefficient (ITC) at the all-rods-out condition is measured by controlling the reactor coolant system (RCS) temperature through varying the steam generator blowdown flow, establishing a constant heatup or cooldown rate, and monitoring the resulting reactivity changes on the reactivity computer. This test sequence includes a cooldown followed by a heatup.

Reactivity was measured during the RCS cooldown of 3.0°F and RCS heatup of 3.0°F. Reactivity and temperature data were taken from the reactivity computer and strip chart recorders. Using the statepoint method, the temperature coefficient was determined by dividing the change in reactivity by the change in RCS temperature. An X-Y plotter, which plotted reactivity versus temperature, confirmed the statepoint method in calculating the measured ITC.

The predicted and measured isothermal temperature coefficient values are compared in Table 5.1. As can be seen from this summary and from the Startup Physics Test Results and Evaluation Sheet given in the Appendix, the measured isothermal temperature coefficient value was within the design tolerance of ±3 pcm/°F. Accounting for the Doppler temperature coefficient (-1.70 pcm/°F) and a 0.5 pcm/°F uncertainty, the moderator temperature coefficient was 0.87 pcm/°F, which meets the requirement of Core Operating Limits Report Section 2.1. In summary, the measured results were satisfactory.

Table 5.1

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS ISOTHERMAL TEMPERATURE COEFFICIENT SUMMARY

BANK POSITION (STEPS)	TEMPERATURE RANGE (°F)	BORON CONCENTRATION (ppm)	ISOTHERMAL TEMPERATURE COEFFICIENT (PCM/°F)				
			C/D	H/U	AVE MEAS	PRED	DIFFER (M-P)
D/209	544.3 to 547.3	1948	-0.83	-0.83	-0.83	-1.75	0.92

SECTION 6

POWER DISTRIBUTION MEASUREMENTS

The core power distributions were measured using the moveable incore detector flux mapping system. This system consists of five fission chamber detectors, which traverse fuel assembly instrumentation thimbles depicted in Figure 1.3. For each traverse, the detector voltage output is continuously monitored on a strip chart recorder, and scanned for 61 discrete axial points by the PRODAC P-250 process computer. Full core, three-dimensional power distributions are determined from this data using the CECOR code[4]. CECOR couples the measured voltages with predetermined analytic signal-to-power conversions, pin-to-box factors, and average coupling coefficients in order to determine the power distribution for the whole core.

A list of the full-core flux maps taken during the startup test program and the measured values of the important power distribution parameters are given in Table 6.1. A comparison of these measured values with their Technical Specification limits is given in Table 6.2. Flux map 2 was taken at approximately 28% power to verify the radial power distribution (RPD) predictions at low power. Figure 6.1 shows the measured RPDs from this flux map. Flux maps 3 and 4 were taken near 70% and 100% power, respectively, with different control rod configurations. These flux maps were taken to check at-power design predictions and to measure core power distributions at various operating conditions. The radial power distributions for these maps are given in Figures 6.2 and

6.3. These figures show that the average relative assembly power distribution measured/predicted percent difference was 1.4% or less for the three maps. The measured F-Q(Z) and F-DH(N) peaking factor values for all flux maps were within the limits of the Core Operating Limits Report (COLR) Sections 2.3 and 2.4, respectively. All three flux maps were used to recalibrate the power range excore detectors.

In conclusion, the power distribution measurement results were considered to be acceptable with respect to the design tolerances, the accident analysis acceptance criteria, and the COLR limits. It is therefore anticipated that the core will continue to operate as designed throughout Cycle 16.

Table 6.1

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
INCORE FLUX MAP SUMMARY

Map Description	Map No.	Date	Burn up MWD/MTU	Pwr %	Bank D Steps	F-Q(Z) Hot (1) Channel Factor			F-DH(N) Hot Channel Factor		Core F(Z) Max		(2) Core Tilt		Axial Off Set (%)	No. of Thimbles
						Assy	Axial Point	F-Q(Z)	Assy	F-DH(N)	Axial point	F(Z)	Max	Loc		
Less thn 30% Pwr	2	11/21/98	8	28	171	F11	30	2.142	F11	1.523	26	1.302	1.0104	NE	0.715	44
Btwn 65% and 75%	3	11/26/98	35	69	193	F11	30	1.929	F11	1.482	30	1.205	1.0095	NE	-0.354	45
Grt than 95% Pwr	4	12/02/98	175	99	227	F11	32	1.852	F11	1.470	30	1.164	1.0056	NE	-0.893	45

NOTES: Hot spot locations are specified by giving assembly locations (E.G. H-8 is the center-of-core assembly) and core height (in the "Z" direction the core is divided into 61 axial points starting from the top of the core).

(1) F-Q(Z) includes a total uncertainty of 1.08.

(2) CORE TILT - defined as the average quadrant power tilt from CECOR.

(3) MAPS 2, 3, and 4 were used for power range detector calibrations.

Table 6.2

SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
COMPARISION OF MEASURED POWER DISTRIBUTION PARAMETERS WITH THEIR CORE OPERATING LIMITS

Map	Peak F-Q(Z) Hot Channel Factor*			F-Q(Z) Hot Channel Factor** (At Node of Minimum Margin)				F-DH(N) Hot Channel Factor		
No.	Meas.	Limit	Node	Meas.	Limit	Node	Margin (%)	Meas.	Limit	Margin (%)
2	2.142	4.628	30	2.130	4.582	26	53.5	1.523	1.898	19.8
3	1.929	3.373	31	1.896	3.297	21	42.5	1.482	1.707	13.2
4	1.852	1.153	32	1.852	2.341	30	20.9	1.470	1.565	6.1

*The Core Operating Limit for the heat flux hot channel factor, F-Q(Z), is a function of core height and power level. The value for F-Q(Z) listed above is the maximum value of F-Q(Z) in the core. The COLR limit listed above is evaluated at the plane of maximum F-Q(Z).

**The value for F-Q(Z) listed above is the value at the plane of minimum margin. The minimum margin values listed above are the minimum percent difference between the measured values of F-Q(Z) and the COLR limit for each map.

The measured F-Q(Z) hot channel factors include 8% uncertainty as defined in Tech. Spec 3.12.B.

Figure 6.1
SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
28% POWER

Predicted
Measured
Pct Difference

Predicted
Measured
Pct Difference

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						0.247 0.250 0.992	0.235 0.236 0.406	0.246 0.248 0.600							1
				0.301 0.302 0.488	0.563 0.568 0.928	1.073 1.085 1.129	0.824 0.826 0.197	1.069 1.076 0.651	0.561 0.567 1.052	0.299 0.304 1.560					2
			0.271 0.280 3.355	1.090 1.094 0.317	1.314 1.326 0.894	1.233 1.256 1.868	1.288 1.282 -0.450	1.230 1.239 0.747	1.310 1.328 1.391	1.084 1.104 1.851	0.268 0.286 6.512				3
		0.268 0.275 2.445	0.630 0.634 0.724	1.325 1.313 -0.899	1.272 1.277 0.360	1.325 1.338 0.938	1.245 1.255 0.821	1.322 1.340 1.361	1.270 1.292 1.731	1.320 1.338 1.385	0.631 0.633 0.343	0.271 0.270 -0.360			4
	0.299 0.307 2.422	1.083 1.124 3.732	1.316 1.337 1.584	1.294 1.312 1.389	1.345 1.353 0.612	1.211 1.217 0.540	1.315 1.333 1.415	1.212 1.237 2.057	1.346 1.382 2.685	1.296 1.311 1.118	1.324 1.300 -1.791	1.091 1.084 -0.652	0.302 0.308 2.051		5
	0.561 0.562 0.073	1.310 1.316 0.422	1.267 1.271 0.264	1.340 1.344 0.295	1.161 1.161 -0.068	1.206 1.208 0.190	1.151 1.176 2.156	1.214 1.248 2.781	1.166 1.236 6.031	1.344 1.372 2.117	1.272 1.273 0.089	1.317 1.314 -0.239	0.565 0.567 0.352		6
0.246 0.248 0.597	1.070 1.061 -0.875	1.233 1.205 -2.256	1.325 1.310 -1.124	1.211 1.195 -1.317	1.210 1.196 -1.183	1.150 1.128 -1.907	1.108 1.111 0.242	1.154 1.163 0.763	1.215 1.247 2.640	1.213 1.231 1.460	1.326 1.329 0.185	1.238 1.231 -0.533	1.080 1.087 0.619	0.249 0.249 0.306	7
0.236 0.254 7.626	0.829 0.828 -0.099	1.293 1.294 0.029	1.247 1.231 -1.337	1.317 1.275 -3.192	1.155 1.138 -1.465	1.113 1.103 -0.829	1.045 1.044 -0.070	1.113 1.119 0.512	1.156 1.182 2.208	1.318 1.330 0.949	1.248 1.244 -0.341	1.294 1.261 -2.542	0.830 0.817 -1.509	0.236 0.233 -1.004	8
0.249 0.250 0.761	1.080 1.072 -0.687	1.238 1.225 -1.029	1.326 1.313 -0.981	1.213 1.208 -0.413	1.215 1.210 -0.437	1.155 1.157 0.201	1.111 1.109 -0.190	1.153 1.143 -0.902	1.214 1.215 0.081	1.213 1.218 0.424	1.326 1.326 -0.003	1.234 1.225 -0.738	1.071 1.063 -0.767	0.247 0.243 -1.359	9
	0.565 0.556 -1.615	1.317 1.286 -2.366	1.272 1.256 -1.247	1.344 1.337 -0.574	1.167 1.166 -0.094	1.216 1.242 2.133	1.155 1.156 0.039	1.214 1.215 0.076	1.166 1.177 0.934	1.344 1.353 0.678	1.269 1.276 0.498	1.312 1.317 0.355	0.562 0.564 0.328		10
	0.302 0.297 -1.370	1.091 1.079 -1.147	1.324 1.311 -0.959	1.297 1.288 -0.666	1.348 1.333 -1.096	1.215 1.198 -1.381	1.319 1.296 -1.727	1.215 1.218 0.188	1.349 1.395 3.367	1.297 1.307 0.706	1.319 1.333 1.059	1.085 1.099 1.266	0.300 0.303 0.887		11
		0.271 0.282 3.931	0.631 0.626 -0.873	1.321 1.302 -1.441	1.271 1.241 -2.364	1.325 1.266 -4.455	1.248 1.218 -2.413	1.329 1.316 -0.980	1.275 1.273 -0.220	1.328 1.330 0.142	0.631 0.646 2.315	0.269 0.288 7.054			12
			0.269 0.264 -1.694	1.086 1.066 -1.760	1.312 1.282 -2.317	1.232 1.196 -2.922	1.291 1.260 -2.401	1.236 1.216 -1.627	1.317 1.273 -3.392	1.093 1.076 -1.502	0.271 0.270 -0.375				13
				0.300 0.297 -0.974	0.562 0.549 -2.226	1.070 1.046 -2.323	0.826 0.813 -1.599	1.076 1.083 0.685	0.564 0.555 -1.590	0.302 0.297 -1.599					14
						0.247 0.241 -2.312	0.236 0.232 -1.448	0.248 0.248 0.273							15

STANDARD
DEVIATION
= 1.268

AVERAGE
PCT DIFFERENCE
= 1.3

SUMMARY

MAP NO: S1-16-02

DATE: 11/21/98

POWER: 27.68%

CONTROL ROD POSITIONS:

D BANK AT 171 STEPS

F-Q(Z) = 2.142

F-DH(N) = 1.523

F(Z) = 1.302

BURNUP = 8.0

QPTR:

NW	1.0009	NE	1.0104
SW	0.9887	SE	1.0001

A.O. = 0.715%

Figure 6.2
SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
69% POWER

Predicted
Measured
Pct Difference

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						0.265 0.261 -1.251	0.256 0.253 -1.356	0.264 0.262 -0.596							1
				0.308 0.302 -1.761	0.572 0.565 -1.362	1.089 1.076 -1.214	0.874 0.861 -1.522	1.085 1.081 -0.400	0.571 0.578 1.371	0.307 0.311 1.423					2
			0.274 0.281 2.543	1.067 1.047 -1.903	1.281 1.264 -1.342	1.223 1.212 -0.929	1.280 1.246 -2.632	1.220 1.215 -0.409	1.277 1.289 0.882	1.062 1.078 1.515	0.271 0.284 4.689				3
		0.271 0.273 0.838	0.629 0.623 -0.876	1.283 1.239 -3.443	1.244 1.228 -1.316	1.299 1.293 -0.489	1.232 1.227 -0.386	1.297 1.304 0.536	1.243 1.259 1.344	1.279 1.298 1.471	0.629 0.636 1.163	0.274 0.275 0.481			4
	0.307 0.310 1.291	1.061 1.084 2.181	1.276 1.286 0.761	1.270 1.288 1.380	1.331 1.332 0.073	1.213 1.220 0.566	1.307 1.320 1.049	1.214 1.233 1.525	1.332 1.365 2.455	1.271 1.293 1.666	1.282 1.287 0.330	1.068 1.071 0.298	0.308 0.306 -0.888		5
	0.571 0.570 -0.136	1.277 1.278 0.034	1.240 1.236 -0.389	1.327 1.313 -1.080	1.217 1.211 -0.504	1.238 1.240 0.145	1.178 1.206 2.431	1.246 1.272 2.069	1.221 1.285 5.272	1.330 1.361 2.359	1.244 1.256 0.953	1.283 1.289 0.473	0.574 0.578 0.642		6
0.264 0.264 0.269	1.086 1.080 -0.562	1.223 1.206 -1.429	1.299 1.285 -1.074	1.213 1.194 -1.567	1.243 1.227 -1.265	1.190 1.168 -1.865	1.150 1.148 -0.144	1.194 1.181 -1.047	1.247 1.271 1.972	1.215 1.233 1.521	1.299 1.307 0.640	1.227 1.227 0.000	1.095 1.110 1.352	0.266 0.269 1.136	7
0.256 0.267 4.007	0.878 0.878 -0.034	1.284 1.290 0.501	1.233 1.220 -1.039	1.308 1.272 -2.739	1.181 1.168 -1.127	1.154 1.147 -0.550	1.090 1.090 0.010	1.154 1.157 0.247	1.182 1.206 2.012	1.309 1.324 1.118	1.234 1.233 -0.024	1.284 1.255 -2.255	0.878 0.882 0.377	0.257 0.257 0.232	8
0.266 0.266 0.055	1.095 1.085 -0.856	1.227 1.213 -1.092	1.299 1.290 -0.726	1.215 1.217 0.237	1.247 1.249 0.219	1.195 1.205 0.849	1.152 1.158 0.548	1.193 1.194 0.053	1.245 1.256 0.881	1.214 1.225 0.883	1.300 1.305 0.424	1.224 1.219 -0.393	1.087 1.084 -0.261	0.264 0.259 -1.673	9
	0.574 0.562 -2.110	1.283 1.241 -3.226	1.244 1.230 -1.112	1.330 1.332 0.128	1.221 1.232 0.903	1.247 1.286 3.058	1.181 1.196 1.291	1.245 1.256 0.877	1.220 1.237 1.319	1.329 1.344 1.143	1.242 1.255 1.071	1.278 1.286 0.570	0.571 0.569 -0.369		10
	0.308 0.303 -1.639	1.068 1.054 -1.286	1.283 1.276 -0.541	1.272 1.276 0.347	1.333 1.337 0.262	1.216 1.222 0.521	1.309 1.318 0.695	1.216 1.227 0.850	1.334 1.367 2.447	1.272 1.287 1.153	1.278 1.298 1.601	1.062 1.078 1.499	0.307 0.309 0.784		11
		0.274 0.286 4.529	0.629 0.628 -0.157	1.280 1.275 -0.420	1.243 1.233 -0.870	1.298 1.276 -1.723	1.233 1.218 -1.253	1.301 1.294 -0.587	1.246 1.244 -0.199	1.285 1.292 0.543	0.629 0.651 3.497	0.271 0.288 6.115			12
			0.272 0.269 -1.005	1.063 1.051 -1.115	1.278 1.260 -1.450	1.221 1.195 -2.196	1.281 1.239 -3.331	1.225 1.202 -1.878	1.283 1.242 -3.190	1.068 1.056 -1.180	0.274 0.275 0.351				13
				0.307 0.298 -2.920	0.571 0.560 -1.989	1.086 1.062 -2.213	0.875 0.857 -2.078	1.091 1.089 -0.139	0.573 0.563 -1.771	0.308 0.304 -1.447					14
						0.264 0.255 -3.451	0.256 0.251 -2.101	0.265 0.264 -0.526							15

STANDARD
DEVIATION
= 1.061

AVERAGE
PCT DIFFERENCE
= 1.3

SUMMARY

MAP NO: S1-16-03

DATE: 11/26/98

POWER: 68.61%

CONTROL ROD POSITIONS:

D BANK AT 193 STEPS

F-Q(Z) = 1.929

F-DH(N) = 1.482

F(Z) = 1.205

BURNUP = 35

QPTR:

NW	0.9931	NE	1.0095
SW	0.9945	SE	1.0029

A.O. = -0.354 %

Figure 6.3
SURRY UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
99% POWER

Predicted
Measured
Pct Difference

Predicted
Measured
Pct Difference

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						0.275 0.271 -1.201	0.271 0.267 -1.319	0.273 0.272 -0.678							1
				0.308 0.304 -1.493	0.572 0.565 -1.250	1.090 1.077 -1.152	0.915 0.902 -1.419	1.086 1.081 -0.487	0.571 0.577 1.065	0.307 0.311 1.218					2
			0.273 0.284 4.250	1.041 1.025 -1.580	1.251 1.236 -1.240	1.210 1.199 -0.902	1.272 1.241 -2.462	1.208 1.202 -0.508	1.248 1.256 0.629	1.036 1.050 1.347	0.270 0.291 7.577				3
		0.270 0.274 1.421	0.624 0.621 -0.507	1.254 1.214 -3.195	1.226 1.209 -1.332	1.283 1.274 -0.662	1.223 1.217 -0.472	1.280 1.285 0.351	1.224 1.235 0.902	1.250 1.259 0.730	0.625 0.624 -0.158	0.273 0.270 -0.981			4
	0.307 0.312 1.708	1.035 1.064 2.740	1.247 1.258 0.813	1.257 1.271 1.152	1.330 1.327 -0.205	1.220 1.219 -0.096	1.305 1.317 0.919	1.222 1.239 1.396	1.331 1.357 1.889	1.258 1.264 0.449	1.253 1.222 -2.491	1.041 1.028 -1.283	0.309 0.312 1.063		5
	0.571 0.571 -0.076	1.248 1.248 -0.006	1.222 1.214 -0.621	1.327 1.304 -1.703	1.280 1.271 -0.764	1.268 1.270 0.178	1.200 1.233 2.743	1.276 1.307 2.458	1.284 1.344 4.661	1.329 1.350 1.549	1.225 1.220 -0.421	1.252 1.243 -0.764	0.574 0.572 -0.332		6
0.273 0.275 0.679	1.087 1.081 -0.555	1.210 1.188 -1.794	1.282 1.264 -1.368	1.220 1.196 -1.946	1.272 1.257 -1.214	1.223 1.209 -1.083	1.182 1.192 0.825	1.226 1.237 0.922	1.276 1.308 2.501	1.222 1.238 1.316	1.282 1.282 -0.035	1.213 1.202 -0.941	1.095 1.093 -0.126	0.276 0.274 -0.502	7
0.271 0.287 5.869	0.918 0.921 0.246	1.276 1.282 0.493	1.224 1.209 -1.260	1.307 1.263 -3.336	1.204 1.191 -1.084	1.185 1.185 -0.049	1.126 1.135 0.769	1.186 1.202 1.374	1.205 1.241 3.013	1.307 1.324 1.300	1.225 1.222 -0.251	1.276 1.242 -2.680	0.919 0.902 -1.759	0.271 0.266 -1.763	8
0.276 0.277 0.645	1.094 1.088 -0.609	1.213 1.201 -0.980	1.282 1.273 -0.671	1.222 1.226 0.363	1.276 1.282 0.437	1.227 1.243 1.270	1.183 1.195 0.952	1.225 1.235 0.790	1.274 1.284 0.713	1.221 1.233 0.996	1.283 1.289 0.480	1.211 1.205 -0.453	1.087 1.079 -0.709	0.273 0.266 -2.642	9
	0.574 0.563 -1.886	1.252 1.214 -3.079	1.225 1.213 -0.939	1.329 1.333 0.306	1.285 1.299 1.135	1.277 1.323 3.630	1.203 1.215 0.984	1.275 1.286 0.864	1.284 1.302 1.401	1.328 1.349 1.561	1.223 1.240 1.415	1.249 1.263 1.146	0.571 0.579 1.288		10
	0.309 0.305 -1.309	1.041 1.032 -0.909	1.253 1.249 -0.289	1.259 1.264 0.451	1.332 1.336 0.304	1.223 1.224 0.124	1.308 1.289 -1.420	1.223 1.226 0.254	1.333 1.366 2.479	1.259 1.287 2.234	1.249 1.278 2.350	1.036 1.062 2.504	0.307 0.313 2.030		11
		0.273 0.290 6.485	0.625 0.627 0.321	1.251 1.249 -0.134	1.225 1.216 -0.722	1.281 1.257 -1.884	1.224 1.202 -1.794	1.284 1.273 -0.882	1.227 1.225 -0.216	1.255 1.267 0.941	0.625 0.649 3.945	0.270 0.297 9.904			12
			0.270 0.269 -0.410	1.037 1.031 -0.566	1.249 1.235 -1.085	1.209 1.186 -1.908	1.273 1.236 -2.909	1.212 1.189 -1.892	1.252 1.208 -3.532	1.042 1.031 -1.115	0.273 0.275 0.763				13
				0.307 0.305 -0.602	0.571 0.564 -1.299	1.087 1.069 -1.627	0.915 0.901 -1.586	1.091 1.091 0.000	0.573 0.562 -1.811	0.309 0.304 -1.359					14
						0.274 0.270 -1.459	0.271 0.267 -1.410	0.275 0.274 -0.321							15

STANDARD
DEVIATION
= 1.354

AVERAGE
PCT DIFFERENCE
= 1.4

SUMMARY

MAP NO: S1-16-04 DATE: 12/02/98 POWER: 98.8%

CONTROL ROD POSITIONS: F-Q(Z) = 1.852 QPTR:

D BANK AT 227 STEPS F-DH(N) = 1.470 NW 0.9934 | NE 1.0056

F(Z) = 1.164 SW 0.9967 | SE 1.0044

BURNUP = 175 A.O. = -0.893%

SECTION 7

REFERENCES

1. T.S. Psuik, "Surry Unit 1, Cycle 16 Design Report", Technical Report NE-1177, Revision 0, Virginia Power, November, 1998.

2. T. K. Ross, W. C. Beck, "Control Rod Reactivity Worth Determination By The Rod Swap Technique," VEP-FRD-36A, December, 1980.

3. Letter from W. L. Stewart (Virginia Power) to the U.S.N.R.C, "Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2: Modification of Startup Physics Test Program - Inspector Followup Item 280, 281/88-29-01", Serial No. 89-541, December 8, 1989.

4. T. W. Schleicher, "The Virginia Power CECOR Code Package", Technical Report NE-831, Revision 4, Virginia Power, August, 1998.

5. Surry Unit 1 and 2 Technical Specifications, Sections 3.1.E.1, 3.12.B.1, 3.12.C.1, 4.10.A, and 5.3.A.6.b.

6. R. W. Twitchell, "Surry 1, Cycle 16 TOTE Calculations", PM-777, Revision 0, November, 1998.

7. P. D. Banning, "Surry 1, Cycle 16 Flux Map Analysis", PM-781, Revision 0, and Addenda, November – December, 1998.

8. R.W. Twitchell, "Surry 1, Cycle 16 RSAC Calculations", PM-760 Revision 0, August 1998.

9. R.W. Twitchell, "Surry1, Cycle 16 Design Report Calculations", PM-774, Revision 0, November 1998.

APPENDIX

STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEETS

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Zero Power Testing Range Determination Proc No / Section: 1-NPT-RX-008	Sequence Step No:
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 229 CC: * CD: *	RCS Temperature (^{O}F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 229 CC: 229 CD: 117	RCS Temperature (^{O}F): 547.5 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 0640	
	Reactivity Computer Initial Flux Background Reading	0* amps * BUCKING CURRENT SET TO 1.4284×10^{-9} AMPS
	Flux Reading At Point Of Nuclear Heating	3.0×10^{-7} amps
	Zero Power Testing Range	1×10^{-8} to 10×10^{-8} amps
	Reference	Not Applicable
V Acceptance Criteria	FSAR/Tech Spec	Not Applicable
	Reference	Not Applicable
VI Comments	Design Tolerance is met** : ✓ YES ___ NO Acceptance Criteria is met** : ✓ YES ___ NO	
	* At The Just Critical Position ** Design Tolerance and Acceptance Criteria are met if ZPTR is below the Point of Nuclear Heating and above background.	

Prepared By: ____________ Reviewed By: ____________

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

<table>
<tr><td>I
Reference</td><td colspan="2">Test Description: Reactivity Computer Checkout
Proc No / Section: 1-NPT-RX-008 Sequence Step No:</td></tr>
<tr><td rowspan="2">II
Test
Conditions
(Design)</td><td>Bank Positions (Steps)</td><td rowspan="2">RCS Temperature (^{o}F): 547
Power Level (% F.P.): 0
Other (specify):
Below Nuclear Heating</td></tr>
<tr><td>SDA: 229 SDB: 229 CA: 229
CB: 229 CC: * CD: *</td></tr>
<tr><td rowspan="2">III
Test
Conditions
(Actual)</td><td>Bank Positions (Steps)</td><td rowspan="3">RCS Temperature (^{o}F): 547.0
Power Level (% F.P.): 0
Other (specify):
Below Nuclear Heating</td></tr>
<tr><td>SDA: 229 SDB: 229 CA: 229
CB: 229 CC: 229 CD: 117</td></tr>
<tr><td rowspan="5">IV
Test
Results</td><td>Date/Time Test Performed:
11/19/98 07:29</td></tr>
<tr><td>Measured Parameter
(Description)</td><td>ρ_c= Measured Reactivity using ρ-computer
ρ_t= Predicted Reactivity</td></tr>
<tr><td>Measured Value</td><td>ρ_c= -47.0 , +49
ρ_t= -47.8 , +49.6
%D= -1.7% , -1.2%</td></tr>
<tr><td>Design Value</td><td>%D= $\{(\rho_c - \rho_t)/\rho_t\} \times 100\% \leq 4.0\ \%$</td></tr>
<tr><td>Reference</td><td>WCAP 7905, Rev. 1, Table 3.6</td></tr>
<tr><td rowspan="2">V
Acceptance
Criteria</td><td>FSAR/Tech Spec</td><td>Not Applicable</td></tr>
<tr><td>Reference</td><td>Not Applicable</td></tr>
<tr><td rowspan="2">VI
Comments</td><td colspan="2">Design Tolerance is met : ✓ YES ____ NO
Acceptance Criteria is met : ✓ YES ____ NO</td></tr>
<tr><td colspan="2">* At The Just Critical Position
The allowable range will be set based on the above results, as well as results from the benchmark test.
Allowable Range = -47.0 to +49.0 pcm</td></tr>
</table>

Prepared By: ____________ Reviewed By: ____________

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Critical Boron Concentration - ARO Proc No / Section: 1-NPT-RX-008	Sequence Step No:
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 229 CC: 229 CD: 229	RCS Temperature (^{O}F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 229 CC: 229 CD: 229	RCS Temperature (^{O}F): 547.5 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 08:00	
	Measured Parameter (Description)	$(C_B)^{M}_{ARO}$; Critical Boron Concentration - ARO
	Measured Value (Design Conditions)	$(C_B)^{M}_{ARO}$ = 1952 ppm
	Design Value (Design Conditions)	C_B = 1939 ± 50 ppm
	Reference	Technical Report NE-1177, Rev. 0
V Acceptance Criteria	FSAR/Tech Spec	$\lvert \alpha C_B \times C_B^{D} \rvert \leq 1000$ pcm
	Reference	Technical Specification 4.10.A
VI Comments	Design Tolerance is met : ✓ YES ____ NO Acceptance Criteria is met : ✓ YES ____ NO	
	αC_B = -7.05 pcm/ppm $C_B^{D} = \lvert (C_B)^{M}_{ARO} - C_B \rvert$; C_B is design value	

Prepared By: Thomas S. P

Reviewed By:

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: HZP Boron Worth Coefficient Measurement Proc No / Section: 1-NPT-RX-008	Sequence Step No:
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: moving CC: 229 CD: 229	RCS Temperature (^{O}F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: moving CC: 229 CD: 229	RCS Temperature (^{O}F): 547.5 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 15:15	
	Measured Parameter (Description)	αC_B ; Boron Worth Coefficient
	Measured Value	αC_B = -7.38 pcm/ppm
	Design Value (Design Conditions)	αC_B = -7.14 ± 0.71 pcm/ppm
	Reference	Technical Report NE-1177, Rev. 0
V Acceptance Criteria	FSAR/Tech Spec	Not Applicable
	Reference	Not Applicable
VI Comments	Design Tolerance is met : ✓ YES ____ NO Acceptance Criteria is met : ✓ YES ____ NO	

Prepared By: ____________ Reviewed By: ____________

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Isothermal Temperature Coefficient - ARO Proc No / Section: 1-NPT-RX-008	Sequence Step No:
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 229 CC: 229 CD: 229	RCS Temperature (^{O}F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 229 CC: 229 CD: 209	RCS Temperature (^{O}F): 547.3 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
	Date/Time Test Performed: 11/19/98 10:27	
IV Test Results	Measured Parameter (Description)	$(\alpha_T^{ISO})_{ARO}$: Isothermal Temperature Coefficient - ARO
	Measured Value	$(\alpha_T^{ISO})_{ARO}$= -0.83 pcm/^{O}F (C_B= 1948 ppm)
	Design Value (Actual Conditions)	$(\alpha_T^{ISO})_{ARO}$= -1.75 ±3.0 pcm/^{O}F (C_B= 1948 ppm)
	Design Value (Design Conditions)	$(\alpha_T^{ISO})_{ARO}$= -1.83 ±3.0 pcm/^{O}F (C_B= 1939 ppm)
	Reference	Technical Report NE-1177, Rev. 0
V Acceptance Criteria	FSAR/COLR	$\alpha_T^{ISO} \leq 3.80$ * pcm/ ^{O}F α_T^{DOP} = -1.70 pcm/^{O}F
	Reference	COLR 2.1.1,Technical Report NE-1177, Rev. 0
VI Comments	Design Tolerance is met : ✓ YES ____ NO Acceptance Criteria is met : ✓ YES ____ NO	
	*Uncertainty on αT_{MOD} = 0.5 pcm/^{O}F (Reference: memorandum from C.T. Snow to E.J. Lozito dated June 27, 1980.)	

Prepared By: Thomas S.

Reviewed By:

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Control Bank B Worth Measurement, Rod Swap Ref. Bank Proc No / Section: 1-NPT-RX-008 Sequence Step No:	
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: moving CC: 229 CD: 229	RCS Temperature (^{O}F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: moving CC: 229 CD: 229	RCS Temperature (^{O}F): 547.3 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 11:14	
	Measured Parameter (Description)	I_B^{REF}; Integral Worth Of Control Bank B, All Other Rods Out
	Measured Value	I_B^{REF}= 1374 pcm
	Design Value (Design Conditions)	I_B^{REF}= 1371 ± 137 pcm
	Reference	Technical Report NE-1177, Rev. 0 And Engineering Transmittal NAF 98-0183, Rev. 0
V Acceptance Criteria	FSAR/Tech Spec	If Design Tolerance is exceeded, SNSOC shall evaluate impact of test result on safety analysis. SNSOC may specify that additional testing be performed.
	Reference	VEP-FRD-36A
VI Comments	Design Tolerance is met : ✓ YES ____ NO Acceptance Criteria is met : ✓ YES ____ NO	

Prepared By: ____________ Reviewed By: ____________

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Critical Boron Concentration - B Bank In Proc No / Section: 1-NPT-RX-008 Sequence Step No:	
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 0 CC: 229 CD: 229	RCS Temperature (^{O}F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 0 CC: 229 CD: 229	RCS Temperature (^{O}F): 547.5 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 15:15	
	Measured Parameter (Description)	$(C_B)^M_{B_I}$: Critical Boron Concentration, B Bank In
	Measured Value (Design Conditions)	$(C_B)^M_B$= 1766 ppm
	Design Value (Design Conditions)	CB = 1747 + ΔCB ^{Prev} ± (10 + 137.1/|αCB|) ppm CB = ~~1756~~ 1760 ± 29 ppm
	Reference	Technical Report NE-1177, Rev. 0
V Acceptance Criteria	FSAR/Tech Spec	Not Applicable
	Reference	Not Applicable
VI Comments	Design Tolerance is met : ✓ YES ____ NO Acceptance Criteria is met : ✓ YES ____ NO	
	αCB = -7.14 pcm/ppm ΔCB ^{Prev} = (CB)^{M}ARO - 1939 ppm	

Prepared By:

Reviewed By:

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Shutdown Bank A Worth Measurement, Rod Swap Proc No / Section: 1-NPT-RX-008	Sequence Step No:
II Test Conditions (Design)	Bank Positions (Steps) SDA: moving SDB: 229 CA: 229 CB: moving CC: 229 CD: 229	RCS Temperature (^{O}F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: moving SDB: 229 CA: 229 CB: moving CC: 229 CD: 229	RCS Temperature (^{O}F): 547.2 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 17:35	
	Measured Parameter (Description)	I_{SA}^{RS}; Integral Worth of Shutdown Bank A, Rod Swap
	Measured Value	I_{SA}^{RS}= 1010 (Adjusted Measured Critical Reference Bank Position = 149 steps)
	Design Value (Actual Conditions)	I_{SA}^{RS}= 1032 (Adjusted Measured Critical Reference Bank Position = 149 steps)
	Design Value (Design Conditions)	I_{SA}^{RS}= 1032 ± 155 pcm (Critical Reference Bank Position = 146 steps)
	Reference	Engineering Transmittal NAF 98-0183, Rev. 0, VEP-FRD-36A
V Acceptance Criteria	FSAR/Tech Spec	If Design Tolerance is exceeded, SNSOC shall evaluate impact of test result on safety analysis. SNSOC may specify that additional testing be performed.
	Reference	VEP-FRD-36A
VI Comments	Design Tolerance is met : ✓ YES ____ NO Acceptance Criteria is met : ✓ YES ____ NO	

Prepared By: ____________ Reviewed By: ____________

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Shutdown Bank B Worth Measurement, Rod Swap Proc No / Section: 1-NPT-RX-008	Sequence Step No:
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: moving CA: 229 CB: moving CC: 229 CD: 229	RCS Temperature (°F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: moving CA: 229 CB: moving CC: 229 CD: 229	RCS Temperature (°F): 547.8 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 17:00	
	Measured Parameter (Description)	I_{SB}^{RS}; Integral Worth of Shutdown Bank B, Rod Swap
	Measured Value	I_{SB}^{RS}= 1129 (Adjusted Measured Critical Reference Bank Position = 169 steps)
	Design Value (Actual Conditions)	I_{SB}^{RS}= 1129 (Adjusted Measured Critical Reference Bank Position = 169 steps)
	Design Value (Design Conditions)	I_{SB}^{RS}= 1129 ± 169 pcm (Critical Reference Bank Position = 164 steps)
	Reference	Engineering Transmittal NAF 98-0183, Rev. 0, VEP-FRD-36A
V Acceptance Criteria	FSAR/Tech Spec	If Design Tolerance is exceeded, SNSOC shall evaluate impact of test result on safety analysis. SNSOC may specify that additional testing be performed.
	Reference	VEP-FRD-36A
VI Comments	Design Tolerance is met : ✓ YES ____ NO Acceptance Criteria is met : ✓ YES ____ NO	

Prepared By:

Reviewed By:

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Control Bank A Worth Measurement, Rod Swap Proc No / Section: 1-NPT-RX-008 Sequence Step No:	
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: moving CB: moving CC: 229 CD: 229	RCS Temperature (°F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: moving CB: moving CC: 229 CD: 229	RCS Temperature (°F): 547.2 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 16:30	
	Measured Parameter (Description)	I_A^{RS}: Integral Worth of Control Bank A, Rod Swap
	Measured Value	I_A^{RS}= 286 (Adjusted Measured Critical Reference Bank Position = 59 steps)
	Design Value (Actual Conditions)	I_A^{RS}= 290 (Adjusted Measured Critical Reference Bank Position = 59 steps)
	Design Value (Design Conditions)	I_A^{RS}= 279 ± 100 pcm (Critical Reference Bank Position = 52 steps)
	Reference	Engineering Transmittal NAF 98-0183, Rev. 0, VEP-FRD-36A
V Acceptance Criteria	FSAR/Tech Spec	If Design Tolerance is exceeded, SNSOC shall evaluate impact of test result on safety analysis. SNSOC may specify that additional testing be performed.
	Reference	VEP-FRD-36A
VI Comments	Design Tolerance is met : ✓ YES ____ NO Acceptance Criteria is met : ✓ YES ____ NO	

Prepared By: ____________ Reviewed By: ____________

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Control Bank C Worth Measurement, Rod Swap Proc No / Section: 1-NPT-RX-008	Sequence Step No:
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: moving CC: moving CD: 229	RCS Temperature (°F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: moving CC: moving CD: 229	RCS Temperature (°F): 547.3 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 16:13	
	Measured Parameter (Description)	I_c^{RS}; Integral Worth of Control Bank C, Rod Swap
	Measured Value	I_c^{RS}= 712 (Adjusted Measured Critical Reference Bank Position = 105 steps)
	Design Value (Actual Conditions)	I_c^{RS}= 727 (Adjusted Measured Critical Reference Bank Position = 105 steps)
	Design Value (Design Conditions)	I_c^{RS}= 725 ± 109 pcm (Critical Reference Bank Position = 98 steps)
	Reference	Engineering Transmittal NAF 98-0183, Rev. 0, VEP-FRD-36A
V Acceptance Criteria	FSAR/Tech Spec	If Design Tolerance is exceeded, SNSOC shall evaluate impact of test result on safety analysis. SNSOC may specify that additional testing be performed.
	Reference	VEP-FRD-36A
VI Comments	Design Tolerance is met :	✓ YES ____ NO
	Acceptance Criteria is met :	✓ YES ____ NO

Prepared By: ____________ Reviewed By: ____________

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Control Bank D Worth Measurement, Rod Swap Proc No / Section: 1-NPT-RX-008	Sequence Step No:
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: moving CC: 229 CD: moving	RCS Temperature (°F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: moving CC: 229 CD: moving	RCS Temperature (°F): 547.5 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/17/98 15:50	
	Measured Parameter (Description)	I_D^{RS}; Integral Worth of Control Bank D, Rod Swap
	Measured Value	I_D^{RS}= ~~1077~~ 1070 rws ~~1046~~ rw (Adjusted Measured Critical Reference Bank Position = ~~155~~ 159 rws steps)
	Design Value (Actual Conditions)	I_D^{RS}= ~~1078~~ 1077 rws (Adjusted Measured Critical Reference Bank Position = ~~155~~ 159 rws steps)
	Design Value (Design Conditions)	I_D^{RS}= 1076 ± 161 pcm (Critical Reference Bank Position = 154 steps)
	Reference	Engineering Transmittal NAF 98-0183, Rev. 0, VEP-FRD-36A
V Acceptance Criteria	FSAR/Tech Spec	If Design Tolerance is exceeded, SNSOC shall evaluate impact of test result on safety analysis. SNSOC may specify that additional testing be performed.
	Reference	VEP-FRD-36A
VI Comments	Design Tolerance is met : Acceptance Criteria is met :	✓ YES ____ NO ✓ YES ____ NO

Prepared By:

Reviewed By:

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: Total Rod Worth, Rod Swap Proc No / Section: 1-NPT-RX-008	Sequence Step No:
II Test Conditions (Design)	Bank Positions (Steps) SDA: moving SDB: moving CA: moving CB: moving CC: moving CD: moving	RCS Temperature (^{O}F): 547 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
III Test Conditions (Actual)	Bank Positions (Steps) SDA: moving SDB: moving CA: moving CB: moving CC: moving CD: moving	RCS Temperature (^{O}F): 547.1 Power Level (% F.P.): 0 Other (specify): Below Nuclear Heating
IV Test Results	Date/Time Test Performed: 11/19/98 11:14	
	Measured Parameter (Description)	I_{Total}; Integral Worth of All Banks, Rod Swap
	Measured Value	I_{Total}= 5581 pcm
	Design Value (Actual Conditions)	I_{Total}= 5626 pcm
	Design Value (Design Conditions)	I_{Total}= 5611 ± 561 pcm
	Reference	Engineering Transmittal NAF 98-0183, Rev. 0, VEP-FRD-36A
V Acceptance Criteria	FSAR/Tech Spec	If Design Tolerance is exceeded, SNSOC shall evaluate impact of test result on safety analysis. Additional testing must be performed.
	Reference	VEP-FRD-36A
VI Comments	Design Tolerance is met : [✓ YES] ____ NO Acceptance Criteria is met : [✓ YES] ____ NO	

Prepared By: ____________ Reviewed By: ____________

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference

Test Description: M/D Flux Map - Banks D,C at Insertion Limits

Proc No / Section: 1-NPT-RX-008 ,002 Sequence Step No:

II Test Conditions (Design)

Bank Positions (Steps)

SDA: 229 SDB: 229 CA: 229

CB: 229 CC: * CD: *

RCS Temperature (^{O}F): TREF ± 1

Power Level (% F.P.): ≤ 30

Other (specify):

Must have ≥ 38 thimbles**

III Test Conditions (Actual)

Bank Positions (Steps)

SDA: 229 SDB: 229 CA: 229

CB: 229 CC: 229 CD: 171

RCS Temperature (^{O}F): NOM 27.7%

Power Level (% F.P.): 27.7%

Other (specify):

IV Test Results

Date/Time Test Performed:

11/21/98 15:12

	Measured Parameter (Description)	Maximum Relative Assembly Power %DIFF (M-P)/P	Nuclear Enthalpy Rise Hot Channel Factor FΔH(N)	Total Heat Flux Hot Channel Factor $F_Q(Z)$	Maximum Positive Incore Quadrant Power Tilt
IV Test Results	Measured Value	6.0 $P_i \geq .9$ 7.6 $P_i < .9$	1.523	2.142	1.0104
	Design Value (Design Conditions)	±10% for $P_i \geq 0.9$ ±15% for $P_i < 0.9$ (P_i = assy power)	N/A	N/A	≤ 1.0206
	Reference	WCAP-7905, Rev. 1 NE-1177, Rev. 0	None	None	WCAP-7905,Rev.1 NE-1177,Rev. 0
V Acceptance Criteria	FSAR/COLR	None	$F\Delta H(N) \leq 1.56(1+0.3(1-P))$	$F_Q(Z) \leq 4.64{*}K(Z)$	None
	Reference	None	COLR 2.4	COLR 2.3	None

VI Comments

Design Tolerance is met : ✓ YES ____ NO

Acceptance Criteria is met : ✓ YES ____ NO

* As required

Prepared By: ____

Reviewed By: Pamela D. Banning

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference	Test Description: M/D Flux Map - At Power Proc No / Section: 1-NPT-RX-008 ,002 Sequence Step No:				
II Test Conditions (Design)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 229 CC: 229 CD: *		RCS Temperature (^{O}F): TREF ± 1 Power Level (% F.P.): $65 \leq P \leq 75$ Other (specify): Must have ≥ 38 thimbles**		
III Test Conditions (Actual)	Bank Positions (Steps) SDA: 229 SDB: 229 CA: 229 CB: 229 CC: 229 CD: 193		RCS Temperature (^{O}F): Non 68.61% Power Level (% F.P.): 68.61% Other (specify):		
	Date/Time Test Performed: 11/26/98 03:00				
IV Test Results	Measured Parameter (Description)	Maximum Relative Assembly Power %DIFF (M-P)/P	Nuclear Enthalpy Rise Hot Channel Factor FΔH(N)	Total Heat Flux Hot Channel Factor $F_Q(Z)$	Maximum Positive Incore Quadrant Power Tilt
	Measured Value	5.3 $P_i \geq .9$ 6.1 $P_i < .9$	1.482	1.929	1.0095
	Design Value (Design Conditions)	±10% for $P_i \geq 0.9$ ±15% for $P_i < 0.9$ (P_i = assy power)	N/A	N/A	≤ 1.0204
	Reference	WCAP-7905, Rev. 1 NE-1177, Rev. 0	None	None	WCAP-7905,Rev.1 NE-1177,Rev. 0
V Acceptance Criteria	FSAR/COLR	None	$F\Delta H(N) \leq 1.56(1+0.3(1-P))$	$F_Q(Z) \leq 2.32/P^{*}K(Z)$	None
	Reference	None	COLR 2.4	COLR 2.3	None
VI Comments	Design Tolerance is met : ✓ YES ____ NO Acceptance Criteria is met : ✓ YES ____ NO				
	* As required ** Must have at least 16 thimbles for quarter core maps for multi-point calibrations				

Prepared By: ____________

Reviewed By: ____________

SURRY POWER STATION UNIT 1 CYCLE 16
STARTUP PHYSICS TEST RESULTS AND EVALUATION SHEET

I Reference

Test Description: M/D Flux Map - At Power
Proc No / Section: 1-NPT-RX-008 ,002 Sequence Step No:

II Test Conditions (Design)

Bank Positions (Steps)
SDA: 229 SDB: 229 CA: 229
CB: 229 CC: 229 CD: *
RCS Temperature (°F): TREF ± 1
Power Level (% F.P.): 95 ≤ P ≤ 100
Other (specify):
Must have ≥ 38 thimbles**

III Test Conditions (Actual)

Bank Positions (Steps)
SDA: 229 SDB: 229 CA: 229
CB: 229 CC: 229 CD: 227
RCS Temperature (°F): NORMAL
Power Level (% F.P.): 98.8%
Other (specify):

IV Test Results

Date/Time Test Performed: 12/2/98 1000

Measured Parameter (Description)	Maximum Relative Assembly Power %DIFF (M-P)/P	Nuclear Enthalpy Rise Hot Channel Factor FΔH(N)	Total Heat Flux Hot Channel Factor $F_Q(Z)$	Maximum Positive Incore Quadrant Power Tilt
Measured Value	4.7% for $P_i \geq 0.9$ 9.9% for $P_i < 0.9$	1.470	1.852	1.0070
Design Value (Design Conditions)	±10% for $P_i \geq 0.9$ ±15% for $P_i < 0.9$ (P_i = assy power)	N/A	N/A	≤ 1.0204
Reference	WCAP-7905, Rev. 1 NE-1177, Rev. 0	None	None	WCAP-7905,Rev.1 NE-1177,Rev. 0

V Acceptance Criteria

FSAR/COLR	None	$F\Delta H(N) \leq 1.56(1+0.3(1-P))$	$F_Q(Z) \leq 2.32/P*K(Z)$	None
Reference	None	COLR 2.4	COLR 2.3	None

VI Comments

Design Tolerance is met : ✓ YES ___ NO
Acceptance Criteria is met : ✓ YES ___ NO

* As required
** Must have at least 16 thimbles for quarter core maps for multi-point calibrations

Prepared By: DC Lawrence

Reviewed By:

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

February 12, 1999

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 99-029
NL&OS/GDM R0
Docket Nos. 50-280, 281
50-339
License Nos. DPR-32, 37
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
NORTH ANNA POWER STATION UNIT 2
ASME SECTION XI RELIEF REQUESTS

North Anna Power Station Unit 2 is presently in the second ten year inservice inspection interval, and examinations are conducted to the requirements of the 1986 Edition of ASME Section XI. Surry Power Station Units 1 and 2 are presently in the third ten year inservice inspection interval, and examinations are conducted to the requirements of the 1989 Edition of ASME Section XI. Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested from certain requirements of the ASME Section XI Code associated with Code required examinations.

The Code requirements for the Code Editions referenced above require system hydrostatic testing and associated VT–2 visual examination of all Class 1 pressure retaining piping and valves. However, small diameter (≤ 1 inch), Class 1, reactor coolant system (RCS) pressure boundary vent and drain, sample, and instrumentation connections are equipped with valves that provide for double isolation of the reactor coolant system (RCS) pressure boundary. These valves are maintained closed during normal operation and the piping outboard of the first isolation valve is, therefore, not normally pressurized. Therefore, relief is requested from performing the hydrostatic testing and associated VT–2 visual examination for these small diameter lines because imposition of Code requirements would cause a burden that would not be compensated by an increase in quality and safety. The basis for the relief is provided in the attached relief requests.

Similar relief has been previously granted to the Edwin I. Hatch Nuclear Plant, Units 1 and 2, in the Safety Evaluation for relief request RR-17 provided in the NRC letter from

000012
9902220356 990212
PDR ADOCK 05000280
Q PDR

A047 1/1

Mr. H. N. Berkow of the NRC to Mr. H. L. Sumner, Jr. of the Southern Nuclear Operating Company, Inc., dated September 3, 1998.

Relief request SPT-17 for North Anna Unit 2 is provided in Attachment 1. Relief request no. 13 for Surry Unit 1 and no. 7 for Surry Unit 2 are provided in Attachments 2 and 3, respectively. The relief requests have been approved by the applicable Station Nuclear Safety and Operating Committee.

If you have any questions concerning these requests, please contact us.

Very truly yours,

L. N. Hartz
Vice President - Nuclear Engineering and Services

Attachments

cc: U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30323

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

Mr. R. Smith
Authorized Nuclear Inspector
Surry Power Station

Mr. M. Grace
Authorized Nuclear Inspector
North Anna Power Station

ATTACHMENT 1

ASME SECTION XI RELIEF REQUEST NO. SPT-17
NORTH ANNA POWER STATION UNIT 2

North Anna Power Station Unit 2
Second 10 Year Interval
Request for Relief Number SPT-17

I. IDENTIFICATION OF COMPONENTS

Approximately 30, small diameter (≤ 1 inch), Class 1, reactor coolant system (RCS) pressure boundary vent and drain, sample, and instrumentation connections.

II. IMPRACTICABLE CODE REQUIREMENTS

Section XI, 1986 Edition, Examination Category B–P, Items B15.51 and B15.71 require system hydrostatic testing and associated VT–2 visual examination of all Class 1 pressure retaining piping and valves.

III. ISI BASIS FOR RELIEF REQUEST

These piping segments are equipped with valves, or valve and flange, that provide for double isolation of the reactor coolant system (RCS) pressure boundary. These components are generally maintained closed during normal operation and the piping outboard of the first isolation valve is, therefore, not normally pressurized. The proposed alternative provides an acceptable level of safety and quality based on the following:

1. ASME Section XI Code, paragraph IWA–4400, provides the requirements for hydrostatic pressure testing of piping and components after repairs by welding to the pressure boundary. IWA–4700(b)(5) excludes component connections, piping, and associated valves that are 1 inch nominal pipe size and smaller from the hydrostatic test. Visual examination of these ≤ 1 inch diameter RCS vent/drain/sampling connections once each 10–year interval is unwarranted considering that a repair weld on the same connections is exempted by the ASME XI Code.

2. The non–isolable portion of the RCS vent and drain connections will be pressurized and visually examined as required. Only the isolable portion of these small diameter vent and drain connections will not be pressurized.

3. All piping connections are typically socket–welded and the welds received a surface examination after installation. The piping and valves are nominally heavy wall (schedule 160 pipe and 1500# valve bodies). The vents, drains, and sample lines are not subject to high stresses or cyclic loads, and the design ratings are significantly greater than RCS operating or design pressure.

The Technical Specifications (TS) require RCS leakage monitoring (TS 4.4.6.2.1) during normal operation. Should any of the TS limits be exceeded, then appropriate corrective actions, which may include shutting the plant down, are required to identify the source of the leakage and restore the RCS boundary integrity.

During the 1998 North Anna Unit 1 refueling outage similar piping segments were pressurized by removing a flange and connecting a test rig. A majority of these piping segments are located in close proximity to the RCS main loop piping thus requiring personnel entry into high radiation areas within the containment. The dose associated with this testing was 1.5 man–Rem.

IV. ALTERNATE PROVISIONS

As an alternative to the Code required hydrostatic test of the subject Class 1 reactor coolant system pressure boundary connections the following is proposed:

1. The RCS vent, drain, instrumentation, and sample connections will be visually examined for leakage, and any evidence of past leakage, with the isolation valves in the normally closed position each refueling outage during the ASME XI Class 1 System Leakage Test (IWB–5221).

2. The RCS vent, drain, instrumentation, and sample connections will also be visually examined with the isolation valves in the normally closed position during the 10–year ISI pressure test (IWB–5222 and Code Case N–498–1). This examination will be performed with the RCS at nominal operating pressure and at near operating temperature after satisfying the required 4–hour hold time.

In addition, during modes 1 through 4 the RCS will be monitored for leakage at the following frequency pursuant to TS requirements:

1. Every 72 hours, during steady state operation, the reactor coolant system leak rate will be monitored to assure the limit of one gallon per minute unidentified leakage is maintained.

2. Every 12 hours the containment atmosphere particulate radioactivity will be monitored.

The proposed alternative stated above will ensure that the overall level of plant quality and safety will not be compromised.

V. IMPLEMENTATION SCHEDULE

This alternative to Code requirements will be implemented upon receiving NRC approval for the remainder of the second ten-year inspection interval.

By a letter dated September 3, 1998 the NRR approved a similar relief request for the Edwin I. Hatch Plant, Units 1 and 2.

ATTACHMENT 2

ASME SECTION XI RELIEF REQUEST NO. 13
SURRY POWER STATION UNIT 1

Surry Power Station Unit 1
Third Year Interval
Request for Relief Number 13

I. IDENTIFICATION OF COMPONENTS

Approximately 30, small diameter (≤ 1 inch), Class 1, reactor coolant system (RCS) pressure boundary vent and drain, sample, and instrumentation connections.

II. IMPRACTICABLE CODE REQUIREMENTS

Section XI, 1989 Edition, Examination Category B–P, Items B15.51 and B15.71 require system hydrostatic testing and associated VT–2 visual examination of all Class 1 pressure retaining piping and valves.

III. ISI BASIS FOR RELIEF REQUEST

These piping segments are equipped with valves, or valve and flange, that provide for double isolation of the reactor coolant system (RCS) pressure boundary. These components are generally maintained closed during normal operation and the piping outboard of the first isolation valve is, therefore, not normally pressurized. The proposed alternative provides an acceptable level of safety and quality based on the following:

1. ASME Section XI Code, paragraph IWA–4400, provides the requirements for hydrostatic pressure testing of piping and components after repairs by welding to the pressure boundary. IWA–4700(b)(5) excludes component connections, piping, and associated valves that are 1 inch nominal pipe size and smaller from the hydrostatic test. Visual examination of these ≤ 1 inch diameter RCS vent/drain/sampling connections once each 10–year interval is unwarranted considering that a repair weld on the same connections is exempted by the ASME XI Code.

2. The non–isolable portion of the RCS vent and drain connections will be pressurized and visually examined as required. Only the isolable portion of these small diameter vent and drain connections will not be pressurized.

3. All piping connections are typically socket–welded and the welds received a surface examination after installation. The piping and valves are nominally heavy wall (schedule 160 pipe and 1500# valve bodies). The vents, drains, and sample lines are not subject to high stresses or cyclic loads, and the design ratings are significantly greater than RCS operating or design pressure.

The Technical Specifications (TS) require RCS leakage monitoring (TS Table 4.1-2A, Item No. 10) during normal operation. Should any of the TS limits be exceeded, then appropriate corrective actions, which may include shutting the plant down, are required to identify the source of the leakage and restore the RCS boundary integrity.

The required pressure testing was recently performed during their 1998 refueling at North Anna Unit 1. Similar piping segments were pressurized by removing a flange and connecting a test rig. A majority of these piping segments are located in close proximity to the RCS main loop piping thus requiring personnel entry into high radiation areas within the containment. The dose associated with this testing was 1.5 man–Rem. Conditions at Surry would yield comparable exposure results, if the testing were performed.

IV. ALTERNATE PROVISIONS

As an alternative to the Code required hydrostatic test of the subject Class 1 reactor coolant system pressure boundary connections, the following is proposed:

1. The RCS vent, drain, instrumentation, and sample connections will be visually examined for leakage, and any evidence of past leakage, with the isolation valves in the normally closed position each refueling outage during the ASME XI Class 1 System Leakage Test (IWB–5221).

2. The RCS vent, drain, instrumentation, and sample connections will also be visually examined with the isolation valves in the normally closed position during the 10–year ISI pressure test (IWB–5222 and Code Case N–498–1). This examination will be performed with the RCS at nominal operating pressure and at near operating temperature after satisfying the required 4–hour hold time.

In addition the RCS will be monitored for leakage at the following frequency pursuant to TS requirements:

1. The reactor coolant system leak rate will be monitored daily to assure the limit of one gallon per minute unidentified leakage is maintained.

Additionally, TS 3.1.C.1 states the following:

> "Detected or suspected leakage from the Reactor Coolant System shall be investigated and evaluated. At least two means shall be available to detect reactor coolant system leakage. One of these means must depend on the detection of radionuclides in the containment."

The proposed alternative stated above will ensure that the overall level of plant quality and safety will not be compromised.

V. IMPLEMENTATION SCHEDULE

This alternative to Code requirements will be followed upon receiving NRC approval for the remainder of the third ten-year inspection interval.

By a letter dated September 3, 1998 the NRR approved a similar relief request for Edwin I. Hatch Plant, Units 1 and 2.

ATTACHMENT 3

ASME SECTION XI RELIEF REQUEST NO. 7
SURRY POWER STATION UNIT 2

Surry Power Station Unit 2
Third Year Interval
Request for Relief Number 7

I. IDENTIFICATION OF COMPONENTS

Approximately 30, small diameter (≤ 1 inch), Class 1, reactor coolant system (RCS) pressure boundary vent and drain, sample, and instrumentation connections.

II. IMPRACTICABLE CODE REQUIREMENTS

Section XI, 1989 Edition, Examination Category B–P, Items B15.51 and B15.71 require system hydrostatic testing and associated VT–2 visual examination of all Class 1 pressure retaining piping and valves.

III. ISI BASIS FOR RELIEF REQUEST

These piping segments are equipped with valves, or valve and flange, that provide for double isolation of the reactor coolant system (RCS) pressure boundary. These components are generally maintained closed during normal operation and the piping outboard of the first isolation valve is, therefore, not normally pressurized. The proposed alternative provides an acceptable level of safety and quality based on the following:

1. ASME Section XI Code, paragraph IWA–4400, provides the requirements for hydrostatic pressure testing of piping and components after repairs by welding to the pressure boundary. IWA–4700(b)(5) excludes component connections, piping, and associated valves that are 1 inch nominal pipe size and smaller from the hydrostatic test. Visual examination of these ≤ 1 inch diameter RCS vent/drain/sampling connections once each 10–year interval is unwarranted considering that a repair weld on the same connections is exempted by the ASME XI Code.

2. The non–isolable portion of the RCS vent and drain connections will be pressurized and visually examined as required. Only the isolable portion of these small diameter vent and drain connections will not be pressurized.

3. All piping connections are typically socket–welded and the welds received a surface examination after installation. The piping and valves are nominally heavy wall (schedule 160 pipe and 1500# valve bodies). The vents, drains, and sample lines are not subject to high stresses or cyclic loads, and the design ratings are significantly greater than RCS operating or design pressure.

The Technical Specifications (TS) require RCS leakage monitoring (TS Table 4.1-2A Item No. 10) during normal operation. Should any of the TS limits be exceeded, then appropriate corrective actions, which may include shutting the plant down, are required to identify the source of the leakage and restore the RCS boundary integrity.

The required pressure testing was recently performed during their 1998 refueling at North Anna Unit 1. Similar piping segments were pressurized by removing a flange and connecting a test rig. A majority of these piping segments are located in close proximity to the RCS main loop piping thus requiring personnel entry into high radiation areas within the containment. The dose associated with this testing was 1.5 man–Rem. Conditions at Surry would yield comparable exposure results, if the testing were performed.

IV. ALTERNATE PROVISIONS

As an alternative to the Code required hydrostatic test of the subject Class 1 reactor coolant system pressure boundary connections the following is proposed:

1. The RCS vent, drain, instrumentation, and sample connections will be visually examined for leakage, and any evidence of past leakage, with the isolation valves in the normally closed position each refueling outage during the ASME XI Class 1 System Leakage Test (IWB–5221).

2. The RCS vent, drain, instrumentation, and sample connections will also be visually examined with the isolation valves in the normally closed position during the 10–year ISI pressure test (IWB–5222 and Code Case N–498–1). This examination will be performed with the RCS at nominal operating pressure and at near operating temperature after satisfying the required 4–hour hold time.

In addition the RCS will be monitored for leakage at the following frequency pursuant to TS requirements:

1. The reactor coolant system leak rate will be monitored daily to assure the limit of one gallon per minute unidentified leakage is maintained.

Additionally, TS 3.1.C.1 states the following:

> "Detected or suspected leakage from the Reactor Coolant System shall be investigated and evaluated. At least two means shall be available to detect reactor coolant system leakage. One of these means must depend on the detection of radionuclides in the containment."

The proposed alternative stated above will ensure that the overall level of plant quality and safety will not be compromised.

V. IMPLEMENTATION SCHEDULE

This alternative to Code requirements will be followed upon receiving NRC approval for the remainder of the third ten-year inspection interval.

By a letter dated September 3, 1998 the NRR approved a similar relief request for Edwin I. Hatch Plant, Units 1