DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

August 4, 1986

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. B. J. Youngblood, Project Director
PWR Project Directorate No. 4

Re: Catawba Nuclear Station, Unit 2
Docket No. 50-414

Dear Sir:

In accordance with License Condition 3 of Facility Operating License NPF-52 and 10 CFR 50.59(b), please find attached the description of a change that has been made to the Initial Startup Test Program for Catawba Unit 2. This change would delete the Doppler Only Power Coefficient Verification tests as was previously done on McGuire Unit 2.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

ROS/06/slb

Attachment

xc: Dr. J. Nelson Grace, Regional Administration
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

NRC Resident Inspector
Catawba Nuclear Station

A001
1/1

Form 34634 (R8-85)

DUKE POWER COMPANY
NUCLEAR SAFETY EVALUATION CHECKLIST

(1) STATION: Catawba UNIT: 1 X 2 X 3 ____

OTHER: ____

(2) EVALUATION APPLICABLE TO (DESCRIPTION AND NUMBER OF NSM, PROCEDURE, PROCEDURE CHANGE, OR TEST/EXPERIMENT): Clarification of descriptions of Unit startup testing in chapter 14 of FSAR.

(3) SAFETY EVALUATION — PART A

The item to which this evaluation is applicable represent:

☑ Yes ☐ No A change to the station or procedures as described in the FSAR: or a test or experiment not described in the FSAR? Affected FSAR Section(s) are: chapter 14 (see attachments 1-3)

If the answer to the above is "Yes," identify the **affected** section(s) of the FSAR. Attach additional sheets as necessary.

(4) SAFETY EVALUATION — PART B

☐ Yes ☑ No Will this item require a change to the station Technical Specifications? Affected Tech. Specs. Section(s) are: no technical specification changes needed

If the answer to the above is "Yes," identify the specification(s) **affected** and/or attach the applicable page(s) with the change(s) indicated. Tech. Spec. changes require NSRB and NRC approval prior to use.

(5) SAFETY EVALUATION — PART C

As a result of the item to which this evaluation is applicable:

☐ Yes ☑ No Will the probability of an accident previously evaluated in the FSAR be increased? Explain: see pages 3 & 4

☐ Yes ☑ No Will the consequences of an accident previously evaluated in the FSAR be increased? Explain: see pages 3 & 4

Form 34634 (R8-85)

DUKE POWER COMPANY
NUCLEAR SAFETY EVALUATION CHECKLIST

☐ Yes ☑ No May the possibility of an accident which is different than any already evaluated in the FSAR be created? Explain:

see pages 3 & 4

☐ Yes ☑ No Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased? Explain:

see pages 3 & 4

☐ Yes ☑ No Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased? Explain:

see pages 3 & 4

☐ Yes ☑ No May the possibility of malfunctions of equipment important to safety different than any already evaluated in the FSAR be created? Explain:

see pages 3 & 4

☐ Yes ☑ No Will the margin of safety as defined in the bases to any Technical Specification be reduced?

Explain:

see pages 3 & 4

Justification for the answers above (Yes or No) must be provided in the above spaces (attach additional sheets as necessary).

An unreviewed safety question **is** involved if any answer to Part C above is "Yes" and NRC authorization is required.

(6) Prepared by: Peter LeRoy Date: July 22, 1986

(7) Reviewed by: SW Brown Date: 7/23/86
(Qualified Reviewer)

(8) Page 2 of 4

Duke Power Company

MEMORANDUM

Form 00846 (10-80)
Formerly 847L

DATE 7-22-86

TO file

ADDRESS

FROM Peter LeRoy

SUBJECT Safety Evaluation (10 CFR 50.59)

This safety evaluation is intended to supplement the evaluation performed for change 10 to TP/2/A/2100/01. The previous evaluation establishes why deleting the Doppler Only Power Coefficient Verification does not constitute an unreviewed safety question. This procedure revision did not explain that other clarifying changes were also made to FSAR Figure 14.2.11-1 and additional changes need to be made to Table 14.2.12-2 (Page 23) and Figure 14.2.11-1.

The changes to Table 14.2.12-2 (Pages 35 & 23), (attachment 1 & 1a) are to clarify in the abstract the tests will be performed on Unit 1 only.

The changes to Figure 14.2.11-1, (attachment 2 & 3) are to clarify the ~~f~~ wording of the figure to agree with the test abstracts.

Duke Power Company

MEMORANDUM

Form 00846 (10-80)
Formerly 847L

DATE 7-22-86

TO File

ADDRESS

FROM Peter LeRoy

SUBJECT Safety Evaluation (10CFR 50.59)

No unreviewed safety question is created by these clarifying FSAR Changes nor are any changes to Catawba's Technical Specifications necessary. These FSAR changes will not change any safety analysis or create any new accidents.

attachment 1

Table 14.2.12-2 (Page 23)

DOPPLER ONLY POWER COEFFICIENT VERIFICATION ← (Unit 1 Only)

Abstract

Purpose

To verify the nuclear design predictions of the doppler only power coefficient.

Prequisites

The reactor is at a stable power condition with rods in the specified maneuvering band. The instrumentation necessary for collection of data is installed, calibrated and operable.

Test Method

Initial data is taken. With the turbine and reactor controls in manual, the turbine load is decreased then increased. Data is recorded during and after the load maneuver and used to infer a measured doppler coefficient verification factor. This factor is compared to a vendor supplied predicted doppler verification factor.

Acceptance Criteria

The inferred measured doppler coefficient verification factor agrees with predicted values as specified by the vendor.

attachment 1a

Table 14.2.12-2 (Page 35)

NATURAL CIRCULATION VERIFICATION TEST ← (Unit 1 Only)
Abstract

Purpose

To demonstrate the capability of the NSSS to remove sensible heat by natural circulation flow in the primary loop. To verify that pressurizer pressure and level control systems can respond automatically to a loss of forced circulation and can maintain reactor coolant pressure within acceptable limits. To verify that steam generator level and feedwater flow can be maintained under natural circulation conditions in order to maintain effective heat transfer from the reactor coolant system. To provide operator training to satisfy NUREG 0737 requirements.

Prerequisites

The reactor is critical at a power level of approximately 3% full power with all reactor coolant pumps in operation. Rod control is in manual with Bank D positioned to maintain a slightly negative isothermal temperature coefficient. Pressurizer pressure and level control are in automatic. Steam dump control is in the pressure control mode. Steam generator level is being maintained through use of the auxiliary feedwater header.

The intermediate and power range (low setpoint) high level reactor trips have been reduced to approximately 7% rated thermal power. UHI isolation valves have been gagged. Overtemperature and overpower ΔT reactor trip signals have been blocked.

Various Technical Specifications test exemptions are required for the conduct of this test. These special test exemptions are provided in Technical Specifications. Special operator action guidelines are provided by the test procedure to compensate for the blocking of various safety injection functions and reactor trips. The test is required to be performed at core burnups which ensure that no signifigant core decay heat levels are present.

Test Method

The test will be initiated by tripping all operating reactor coolant pumps. The establishment of natural circulation will be verified by observing the response of wide range hot and cold leg temperatures as well as core exit thermocouples. The response of pressurizer level and pressure will be observed. Steam generator level and pressure response will be monitored. During the performance of this test on Catawba Unit 1 only, the test will be repeated for each operating shift at Catawba or suitable simulator facility, for the purpose of initial operator training. Each RO and SRO will observe or participate in the initiation, detection and maintenance of natural circulation conditions during at least one of the test runs.

attachment 2

Figure 14.2.11-1

TESTING FOLLOWING INITIAL FUEL LOADING

Fuel Loading	Hot Precritical Testing	Initial Criticality	Zero Power Physics Test	0% - 5% Power Post-Physics Testing	10% - 25% Power
Initial Fuel Loading	1. Moveable Incore Detector Functional Test	1. Initial Criticality	1. Controlling Proccedure for Zero Power Physics Testing:	1. Radiation Shielding Survey	1. Loss of Control Room Test (Note 1)
	2. Incore Thermocouple Functional Test		(a) Nuclear Instrumentation Overlap Verification	2. Natural Circulation Verification (Note 3)	2. Station Blackout Test (Note 1)
	3. Incore Thermocouple and RTD Cross Calibration (Optional)		(b) Onset of Nuclear Heat	3. Unit Load Steady-State Test	3. NIS Initial Calibration
	4. Rod Position Indication Check		(c) All Rods Out Critical Boron	*4. Process and Effluent Radiation Monitor Test	4. Steam Generator Water Hammer Test
	5. Rod Control Cluster Assembly Drop Time Test		(d) Isothermal Temperature Coefficient Test	5. NIS Initial Calibration	
	6. Rod Control System Alignment Test		(e) Differential and Integral Worth of Sequenced Control Banks		
	7. Full Length Rod Drive Mechanism Timing Test		(f) Differential Boron Worth at Hot Zero Power		
	8. Reactor Coolant System Flow Test		(g) Integral Control Rod Worth With One Stuck Rod (Note 3)		
	9. Reactor Coolant System Flow Coastdown Test		(h) Pseudo-Ejected RCCA worth at Hot Zero Power (Note 3)		
	10. RTD Bypass Flow Verification				
	11. Pressurizer Functional Test				

* The completion of this test is not required before initial escalation to the next power testing plateau.

NOTE 1: Tests will be completed prior to exceeding the 30% testing plateau.

NOTE 2: Test will be completed prior to exceeding the 75% testing plateua.

Note 3: Test will be performed on Unit 1 Only.

attachment 3

~30% F. P.	~50% F. P.	~75% F. P.	~90% F. P.	~100% F. P.
1. Unit Load Steady State	1. Unit Load Steady State Test	1. Unit Load Steady State Test	1. Unit Load Steady State Test	1. Unit Load Steady State Test
2. Radiation Shielding Survey	2. Radiation Shielding Survey	2. Radiation Shielding Survey	2. NIS Initial Calibration	2. Radiation Shielding Survey
3. Rod Control System at Power Test	3. NIS Initial Calibration	3. NIS Initial Calibration	3. Core Power Distribution	3. NIS Initial Calibration
4. NIS Initial Calibration	4. Core Power Distribution Test	4. Core Power Distribution Test	*4. Feedwater Temperature Variation Test (Note 3)	4. Core Power Distribution Test
5. Core Power Distribution	5. ~~Power Coefficient and Power Defect Measurement~~	5. ~~Power Coefficient and Power Defect Measurement~~	5. Doppler only Power Coefficient Verification (Note 3)	5. Unit Load Transient Test
6. Psuedo Ejection Rod Test (Note 3)	6. Unit Load Transient Test	6. Unit Load Transient Test		6. Unit Loss of Electrical Load Test
7. ~~Power Coefficient and Power Defict Measurement~~	7. Below Bank Test (Note 3)	7. Incore and Nuclear Instrumentation System Detector Correlation		7. Process and Effluent Radiation Monitor Test
8. Unit Load Transient	8. Process and Effluent Radiation Monitor Test	8. Turbine Trip Test (power just below P-9 setpoint) (Note 2)		8. Support Systems Verification Test
9. Pressurizer Level and Pressure Control Test	9. Support Systems Verification Test			
~~10 Steam Generator Water Hammer Test~~				

Doppler only Power Coefficient Verification (Note 3)

Form 35283 (R8-85)

(1) ID No. TP/2/A/2100/01
Change No. ~~8~~ 10

Permanent/Restricted To

DUKE POWER COMPANY
PROCEDURE MAJOR CHANGE
PROCESS RECORD

(2) STATION Catawba

(3) PROCEDURE TITLE Controlling Procedure for Power Escalation

(4) SECTION(S) OF PROCEDURE AFFECTED 12.4.3.10, 12.6.3.6

(5) DESCRIPTION OF CHANGE: (Attach additional pages, if necessary).
(a.) Delete step 12.4.3.10. ~~and renumber succeeding steps in subsection 12.4.3~~ OSB
(b.) Delete step 12.6.3.6.
(These steps involve performance of TP/2/A/2150/04, Doppler Only Power Coefficient Verification. 50% and 90% plateau.

(6) REASON FOR CHANGE
see attached sheets

(7) PREPARED BY A. D. Rickman DATE 6/26/86

(8) SAFETY EVALUATION

This change:

(A) ☒ Yes ☐ No Represents a change to the station or procedures as described in the FSAR, or a test or experiment not described in the FSAR?

(B) ☐ Yes ☒ No Requires a change to the station Technical Specifications?

(C) ☐ Yes ☒ No Involves an unreviewed safety question?

(D) ☒ Yes ☐ No Requires completion of a **NUCLEAR SAFETY EVALUATION CHECK LIST** ?

If the answer to any of the above is **YES**, attach a detailed explanation. As appropriate attach a completed **NUCLEAR SAFETY EVALUATION CHECK LIST** form. If the answer to (B) or (C) is **YES** the change must be approved by the NSRB and NRC prior to implementation.

By [signature] Date 6/29/86

(9) REVIEWED BY Daniel A. Webb SWB DATE 7/7/86

Cross-Disciplinary Review By __________ N/R DAW

(10) TEMPORARY APPROVAL (If Necessary)

By __________ (SRO) Date __________

By __________ Date __________

(11) APPROVED BY [signature] DATE 7/8/86

(12) MISCELLANEOUS

Reviewed/Approved By __________ Date __________

Reviewed/Approved By __________ Date __________

(13) Page 1 of ~~4~~ 9 AHR

Form 34895 (6-82)
Formerly SPD - 1003 - 2A

DUKE POWER COMPANY
PROCEDURE MAJOR CHANGE
PROCESS RECORD CONTINUATION FORM

ID No: TP/2/A/2100/01
Change No: ~~1~~ 10 AGR

(6) Reason For Change:

From Enclosure 13.4 of TP/1/A/2150/04 and TP/2/A/2150/04 the predicted verification values (C_p) for the Doppler Only Power Coefficient Verification Test are defined by the equation:

(1) $C_p\,(°F/\%) = -\frac{\delta\rho}{\delta Q} \Big/ \alpha^{p}_{iso} + \delta$

where $\frac{\delta\rho}{\delta Q}$ = doppler only power coefficient

α^{p}_{iso} = predicted ITC

δ = correction factor

Below are average values taken from a minimum of six measurements at several power levels for the Unit 1 test. Errors in measured versus predicted errors were well within the acceptance criteria.

Power Level (%)	Predicted value, C_p (°F/%)	Measured value, C_m (°F/%)
30	−2.753	−2.771
50	−2.104	−2.182
75	−1.46	−1.46
90	−1.165	−1.065

Form 34895 (6-82)
Formerly SPD - 1003 - 2A

DUKE POWER COMPANY
PROCEDURE MAJOR CHANGE
PROCESS RECORD CONTINUATION FORM

ID No: TP/2/A/2100/01
Change No: ~~9~~ 10 AHR

Doppler Only Power Coefficients used in calculating the predicted values (C_p) were taken from the Nuclear Design Report WCAP 10422 and are as follows:

Power Level (%)	Unit 2 Doppler Coefficients (pcm /% power)
30	-12.0
50	-11.5
75	-10.7
90	-10.3

Substitution of measured values (C_m) for all measured cases at each power level yields the following average Doppler Coefficients:

Power Level (%)	Doppler Coefficients Based on C_m (pcm /% power)
30	-12.34
50	-11.83
75	-10.74
90	-9.35

These values are well within the bounds of the FSAR accident analysis. (FSAR Figure 15.0.4-2)

Power Level (%)	Upper Bound Coefficient	Lower Bound Coefficient
30	-8.5	-17.5
50	-8.0	-16.0
75	-7.0	-14.7
90	-6.7	-13.5

Form 34895 (6-82)
Formerly SPD - 1003 - 2A

DUKE POWER COMPANY
PROCEDURE MAJOR CHANGE
PROCESS RECORD CONTINUATION FORM

ID No: TP/2/A/2100/01
Change No: ~~8~~ 10 ABR

The Doppler Only Power Coefficients to be used in calculating the Unit 2 predicted values would be taken from the Nuclear Design Report, WCAP 10932 and are identical to those used for Unit 1 calculations. Since both units have essentially identical core designs, both measured and predicted values for Unit 2 would be similar to those calculated for Unit 1. Doppler coefficients calculated from measured values would be well within the margin defined by the FSAR accident analyses. Therefore, no impact on accident analysis assumptions would occur as a result of deleting the Doppler Only Power Coefficient Verification from the Unit 2 Power Escalation Testing. In addition, the deletion of this test for Catawba Unit 2 would not set a precedent since this test was also deleted from the McGuire Unit 2 Power Escalation Testing Program based on the same type of argument.

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Form 34634 (R8-85)

DUKE POWER COMPANY
NUCLEAR SAFETY EVALUATION CHECKLIST

(1) STATION: Catawba UNIT: 1 ____ 2 X 3 ____

OTHER: ____

(2) EVALUATION APPLICABLE TO (DESCRIPTION AND NUMBER OF NSM, PROCEDURE, PROCEDURE CHANGE, OR TEST/EXPERIMENT): TP/2/A/2100/01, Controlling Procedure for Power Escalation-Deletion of steps to perform Doppler Only Power Coefficient Verification at 50 and 90% Power.

(3) SAFETY EVALUATION — PART A

The item to which this evaluation is applicable represent:

☒ Yes ☐ No A change to the station or procedures as described in the FSAR: or a test or experiment not described in the FSAR? Affected FSAR Section(s) are: Table 14.2.7-1 (page 3), Figure 14.2.11-1 (marked up copies attached).

If the answer to the above is "Yes," identify the **affected** section(s) of the FSAR. Attach additional sheets as necessary.

(4) SAFETY EVALUATION — PART B

☐ Yes ☒ No Will this item require a change to the station Technical Specifications? Affected Tech. Specs. Section(s) are: This item does not require a change to the Station Technical Specifications.

If the answer to the above is "Yes," identify the specification(s) **affected** and/or attach the applicable page(s) with the change(s) indicated. Tech. Spec. changes require NSRB and NRC approval prior to use.

(5) SAFETY EVALUATION — PART C

As a result of the item to which this evaluation is applicable:

☐ Yes ☒ No Will the probability of an accident previously evaluated in the FSAR be increased? Explain: The performance of this test does not verify safety-related equipment functional capability. The deletion of this test does not degrade safety-related equipment functional capability as evaluated in the FSAR. The deletion of this test does not subject the plant to accident scenarios in the FSAR. Adequate margin exists between FSAR Accident Analysis values of Doppler Only Power Coefficients & those values taken from Design data if measurements were made.

☐ Yes ☒ No Will the consequences of an accident previously evaluated in the FSAR be increased? Explain: Deletion of Doppler Only Power Coefficient Measurements will not increase the severity of accidents previously evaluated in the FSAR since the transient analysis uses very conservative values that were never approached in Unit 1 testing. By virtue of essentially identical core design there is no reason for Unit 2 measurements to differ significantly from Unit 1 results.

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Form 34634 (R8-85)

DUKE POWER COMPANY
NUCLEAR SAFETY EVALUATION CHECKLIST

☐ Yes ☒ No May the possibility of an accident which is different than any already evaluated in the FSAR be created? Explain: No new accidents not evaluated in the FSAR will become possible.

☐ Yes ☒ No Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased? Explain: Doppler Only Power Coefficient Measurements do not verify or affect performance of safety-related equipment. Therefore, deletion of these measurements will not increase probability of safety-related equipment malfunction.

☐ Yes ☒ No Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased? Explain: Deletion of Doppler Only Power Coefficient Measurements will not degrade safety-related equipment or further promote a previously degraded condition.

☐ Yes ☒ No May the possibility of malfunctions of equipment important to safety different than any already evaluated in the FSAR be created? Explain: No safety-related equipment malfunctions not evaluated in the FSAR will become possible as a result of measurement deletion

☐ Yes ☒ No Will the margin of safety as defined in the bases to any Technical Specification be reduced?
Explain: There are no bases in the Tech Specs which would be affected by the deletion of Doppler Only Power Coefficient Measurements.

Justification for the answers above (Yes or No) must be provided in the above spaces (attach additional sheets as necessary).

An unreviewed safety question **is** involved if any answer to Part C above is "Yes" and NRC authorization is required.

(6) Prepared by: [signature] Date: 7/7/86

(7) Reviewed by: Daniel A. Wells Date: 7/7/86
(Qualified Reviewer)

(8) Page 2 of 3

TABLE 14.2.7-1 (Page 3)

COMPLIANCE WITH REGULATORY GUIDES

Regulatory Guide	Compliance	Affected Section(s)	Exception Taken	Justification
1.68 Rev. 2	Partial	App. A 5	Tests and acceptance criteria will be developed to demonstrate the ability of major principal plant control systems to automatically control process variables within design limits around the nominal reference value.	Control system testing should verify proper control of process variables within the design control deadband, not over the range of design values of process variables. Proper control of process variables will be demonstrated during power escalation over the range of 0 to 100% F.P.
	Partial	App. A 5.a	Power coefficient measurements will not be performed at 100% power but will be performed at 90% power instead. (Unit 1 only). Power coefficient measurements will not be performed on Unit 2.	NSSS vendor does not recommend performing this test at 100% power due to potential of violating axial flux difference Technical Specification. Unit 2 has essentially identical fuel and core loading as Unit 1. Errors between measured and predicted power coefficients at 30%, 50%, 75% and 90% on Unit 1 were less than the acceptance criterion value of ±0.5%. There is no reason for Unit 2 measurements to be different from predicted.
		App. A 5.b	Departure from nucleate boiling ratio (DNBR), maximum average planar linear heat generation rate (MAPLHGR), and minimum critical power ratio (MCPR) will not be directly verified during power escalation testing.	Axial, Radial, and Total Peaking will be directly measured and verified during power escalation testing and will be used to verify DNBR and linear heat rate margin by analysis.
	Partial	App. A 5.f	Core thermal and nuclear parameters will not be demonstrated to be in accordance with predictions following a return of the rod to its bank position.	The reactor core will be under xenon transient conditions at this time. There would be insufficient time to gather data under transient conditions. There are no NSSS vendor predictions for this configuration.
		App. A 5.g	Special testing to demonstrate control rod sequencers/withdrawal block funtions operation will not be performed.	Refer to Q640.52 item 4.i response.
		App. A 5.h	Rod drop times will not be measured at power.	Measuring rod drop times at power would require disabling all position indication for the rods in violation of plant Technical Specifications.
		App. A 5.i	Test to demonstrate incore/excore instrumentation sensitivity to detect rod misalignment will not be performed at full power.	From vendor predictions the Xenon and power distributions at 50% and 100% are similar. The performance of this test at 50% should adequately demonstrate the capability and sensitivity of incore/excore instrumentation to detect control rod misalignments equal to or less than Technical Specifications.

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NSEC Pg. 3 of 5

Figure 14.2.11-1

TESTING FOLLOWING INITIAL FUEL LOADING

Fuel Loading	Hot Precritical Testing	Initial Criticality	Zero Power Physics Test	0% - 5% Power Post-Physics Testing	10% - 25% Power
Initial Fuel Loading	1. Moveable Incore Detector Functional Test 2. Incore Thermo-couple Functional Test 3. Incore Thermo-couple and RTD Cross Calibration (Optional) 4. Rod Position Indication Check 5. Rod Control Cluster Assembly Drop Time Test 6. Rod Control System Alignment Test 7. Full Length Rod Drive Mechanism Timing Test 8. Reactor Coolant System Flow Test 9. Reactor Coolant System Flow Coastdown Test 10. RTD Bypass Flow Verification 11. Pressurizer Funct-tional Test	1. Initial Criticality	1. Controlling Proc-cedure for Zero Power Physics Testing: (a) Nuclear Instru-mentation Over-lap Verification (b) Onset of Nuc-lear Heat (c) All Rods Out Critical Boron (d) Isothermal Temperature Coefficient Test (e) Differential and Integral Worth of Se-quenced Con-trol Banks (f) Differential Boron Worth at Hot Zero Power (g) Integral Con-trol Rod Worth With One Stuck Rod (NOTE 3) (h) Pseudo-Eject-ed RCCA worth at Hot Zero Power (NOTE 3)	1. Radiation Shielding Survey 2. Natural Circulation Verification 3. Unit Load Steady-State Test *4. Process and Effluent Radiation Monitor Test 5. NIS Initial Calibra-tion	1. Loss of Control Room Test (Note 1) 2. Station Blackout Test (Note 1) 3. NIS Initial Calibration 4. Steam Generator Water Hammer Test

* The completion of this test is not required before initial escalation to the next power testing plateau.

NOTE 1: Tests will be completed prior to exceeding the 30% testing plateau.

NOTE 2: Test will be completed prior to exceeding the 75% testing plateua.

NOTE 3: Test will be performed on Unit 1 only

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NSEC Pg. 5 OF 5

~30% F. P.	~50% F. P.	~75% F. P.	~90% F. P.	~100% F. P.
Unit Load Steady State	1. Unit Load Steady State Test	1. Unit Load Steady State Test	1. Unit Load Steady State Test	1. Unit Load Steady State Test
Radiation Shielding Survey	2. Radiation Shielding Survey	2. Radiation Shielding Survey	2. NIS Initial Calibration	2. Radiation Shielding Survey
Rod Control System at Power Test	3. NIS Initial Calibration	3. NIS Initial Calibration	3. Core Power Distribution	3. NIS Initial Calibration
NIS Initial Calibration	4. Core Power Distribution Test	4. Core Power Distribution Test	*4. Feedwater Temperature Variation Test (NOTE 3)	4. Core Power Distribution Test
Core Power Distribution	5. Power Coefficient and Power Defect Measurement (NOTE 3)	5. Power Coefficient and Power Defect Measurement (NOTE 3)	5. Doppler only Power Coefficient Verification (NOTE 3)	5. Unit Load Transient Test
Psuedo Ejection Rod Test (NOTE 3)	6. Unit Load Transient Test	6. Unit Load Transient Test		6. Unit Loss of Electrical Load Test
Power Coefficient and Power Defict Measurement (NOTE 3)	7. Below Bank Test	7. Incore and Nuclear Instrumentation System Detector Correlation		7. Process and Effluent Radiation Monitor Test
Unit Load Transient	8. Process and Effluent Radiation Monitor Test	8. Turbine Trip Test (power just below P-9 setpoint) (Note 2)		8. Support Systems Verification Test
Pressurizer Level and Pressure Control Test	9. Support Systems Verification Test			
10 ~~Steam Generator Water Hammer Test~~				