U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-286/85-25

Docket No. 50-286

License No. DPR-64 Priority Category C

Licensee: Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Facility Name: Indian Point 3 Nuclear Power Plant

Inspection At: Buchanan, New York

Inspection Conducted: October 21-25, 1985

Inspectors: Unte Cheh — 11/20/85
Unte Cheh, PE, Ph.D., Reactor Engineer — date

Approved by: P W Eselgroth — 11/27/85
Pete W. Eselgroth, Chief — date
Test Programs Section, OB, DRS

Inspection Summary:
Inspection on October 21-25, 1985 (Report No. 50-286/85-25)

Areas Inspected: Routine, unannounced inspection of cycle 5 refueling/start-up testing up to the 50% of the rated thermal power level. This included precritical tests, power escalation tests, administrative controls for test procedures, test programs and QA involvement in the start-up testing. The inspection involved 45 hours on site by a region-based NRC inspector.

Results: No violations were identified.

1.0 Persons Contacted

*J. E. Russel, Superintendent of Power
*F. W. Gumble, Site Reactor Engineer
*J. J. Anderson, Assistant Plant Engineer
L. Kelly, Performance and Reliability Supervisor
*F. X. Pindar, QA Superintendent
J. Gillen, General Chemistry Supervisor
G. Tasick, QA Supervisor
M. Morrissey, Performance Supervisor
J. Somrai, I&C General Supervisor
*W. D. Hamlin, Assistant to Resident Manager
*S. L. Munoz, Technical Services Superintendent

U. S. NRC

*P. Eselgroth, Chief, Test Program
P. Koltay, Senior Resident Inspector
*U. Cheh, Reactor Engineer

The inspector also contacted other licensee employees in the course of the inspection.

*Denotes those present at the exit interview on October 25, 1985.

2.0 Cycle 5 Start-up Physics Test Program

The start-up physics test program was conducted according to Indian Point No. 3 Cycle 5 Start-up Physics Test Program, RA-7, Revision 3, Approved August 30, 1985. The test program outlined the steps in the testing sequence, set initial conditions and prerequisites, specified calibration or surveillance procedures at appropriate points, and referenced detailed test procedures and data collections in attachments. Initial criticality of Cycle 5 was achieved on October 2, 1985. Upon completion of the Zero Power Physics Test, the unit experienced a water chemistry problem and held at lower power of approximately 50% rated power. The Power Ascension Tests will depend on water chemistry.

The inspector independently verified that the predicted values and acceptance criteria were obtained from "Plant Operations Package for the Indian Point Unit 3 Power Plant Cycle 5", WCAP-10873 dated July 1985. The inspector reviewed test results and documents described in this report to ascertain that the start-up testing was conducted in accordance with technically adequate procedures and as required by Technical Specifications (TS). The details and findings of the review are described in Sections 3 and 4.

3.0 Cycle 5 Start-up Physics Testing Precritical Tests

The inspector reviewed calibration and functional test results to verify the following:

-- Procedures were provided with detailed instructions;
-- Technical contents of the procedures were sufficient to result in satisfactory components and tests;
-- Instruments and calibration equipment used were traceable to the National Bureau of Standards;
-- Acceptance and operability criteria were observed in compliance with TS.

The following tests were reviewed:

3.1 Control Rod Checks and Tests

The rod drop measurement was performed in accordance with the Procedure, Full Length Rod Drop Time Test, 3 PT-R4, Rev.4 approved October 17, 1985. The inspector verified by review of the test results performed on September 27, 1985, that Rod Cluster Control Assemblies (RCCA) were tested for drop times and the individual RCCA drop times were all less then 2.4 seconds as required by the TS. The inspector also reviewed several visicorder traces and verified that the drop times had been interpreted correctly.

No noncompliance was identified.

3.2 Incore Thermocouple, Wide Range RTD and Narrow Range RTD Measurement

The Indian Point Unit 3 plant computer system of Westinghouse P-250 are currently being replaced to Perkin-Elmer 3240 and the thermocouple readings are available, but are not ready for calibration against the Narrow Range RTD readings. The thermocouples are not used for any safety systems at Indian Point Unit 3. Westinghouse P-250 Operator's Console Reference Manual for Nuclear Power Plant Supervision TP044-P dated August, 1972 will still be used to take all the P-250 operator functions on Perkin-Elmer 3240,

No noncompliance was identified.

3.3 Reactivity Computer Setup/Verification

The reactivity computer was set up and calibrated according to the Procedure STS-126, Rev. 4 on May 7, 1985. The reactivity computer was adjusted with the correct inputs of delayed neutron fractions (betas) and decay constants (lambdas). An exponential test signal was fed into the reactivity computer. The dynamic response was then completed with predicted values which were derived from point reactor kinetics. The results of this calibration check were satisfactory.

The reactivity computer was further checked when the reactor reached critical. Comparisons of predicted and measured reactivities based on doubling time measurement were accepted.

No noncompliance was identified.

3.4 Cycle 5 Reload Safety Evaluation and Core Verification

The inspector reviewed the Procedure SOP-RP-1, Rev. 4, Preparation for and Return from Refueling approved July 25, 1985, and the reload was conducted per the procedure.

The Cycle 5 reactor core is comprised of 193 fuel assemblies. During the cycle 4/5 refueling, 76 fresh fuel assemblies (batch T) were loaded into the core. The remaining 117 fuel assemblies were from previous cycles operation. The reload safety evaluation (RSE) along with the required Technical Specifications (TS) change was submitted to the NRC for review. This reload submittal was found acceptable (Letter from John D. Neighbors (NRC) to John C. Brons (NYPA) dated August 27, 1985). The basic assumption used in the RSE was Cycle 4 burnup of 14,090 ± 500 MWD/MTU. The inspector verified the actual cycle 5 burnup to be 14713 ± 500 MWD/MTU. The highpoint burnup of 15200 MWD/MTU was used for the RSE. The assumption is thus valid.

The inspector reviewed one half of the core verification videotape #1 dated August 19, 1985 and verified that the core loading agreed with the intended core loading plan.

4.0 Cycle 5 Startup Physics Testing - Post Critical Tests

The inspector reviewed selected test programs to verify the following:

-- The test programs were implemented per the Cycle 5 Startup Physics Test Program;

-- Step-wise instructions of the test procedures were adequately provided including Precautions, Limitations and Acceptance Criteria in conformance with the requirements of the TS;

-- Provisions for recovering from anomalous conditions were provided.

-- Methods and calculations were clearly specified and the tests were performed accordingly.

-- Review, approval and documentation of the results were in conformance with the requirements of the TS and the licensee's administrative controls.

The following tests were reviewed.

4.1.1 Zero Power Physics Test

The licensee measured the just critical boron concentration per the Procedure RA-2, Rev. 3, Initial Criticality approved August 30, 1985 and control rod worth per the Procedure RA-04, Rev. 3, Control Rod Worth Measurements approved August 30, 1985. The inspector reviewed the data and noted the following results:

Rod Position	Predicted Value (PPM)	Test Value (PPM)
All Rods Out (ARO) (RA-2, Rev. 3, Initial Criticality)	1500 ± 50	1496
D IN	1380 ± 207	1402
D+C IN	1301 ± 195	1307
D+C+B IN	1225 ± 183	1234
D+C+B+A IN (RA-4, Rev. 3, Control Rod Worth Measurements)	1092 ± 163	1115

4.1.2 Isothermal Temperature Coefficient

Isothermal temperature coefficients were measured and documented per the Procedure RA-3, Rev. 4, Isothermal Temperature Coefficient. The inspector noted the following results:

Rod Position	Predicted Value (pcm/°F)	Measured Value (pcm/°F)
ARO	-1.30±3	-1.401
D IN	-2.66±3	-2.88
D+C IN	-4.72 ± 3	-4.85

The isothermal temperature coefficient (ITC) is defined as the change in reactivity for a unit change in the moderator, clad and fuel pellet temperatures. Thus, the ITC can be interpreted as the sum of the moderator and Doppler coefficient. The doppler coefficient is difficult to measure in normal operation. A value of -2.03 pcm/°F was obtained from Westinghouse Report, "The Nuclear Design and Core Management of the Indian Point Unit No. 3 Cycle 5" WCAP-10839. Thus, during zero power physics testing, the ARO Moderator Temperature Coefficient (MTC) was determined to be 0.629 pcm/°F. Therefore, a withdrawal curve was generated to meet the TS requirement of 3.1.C.1.

4.1.3 Control Rod Worth Measurement

The control rod reactivity worth measurements were performed per the Procedure RA-4, Rev. 3, Control Rod Worth Measurements. The following results were noted:

Configuration	Predicted Worth (pcm)	Measured Worth (pcm)
Control Bank D	1043 ± 156	1064.5
Control Bank C (D IN)	699 ± 104	718
Control Bank B (D+C IN)	667 ± 100	633
Control Bank A (D+C+B IN)	1186 ± 178	1205.3
Total	3595 ± 359.5	3620.8

No noncompliance was identified.

4.2 Power Ascension Tests

4.2.1 Core Power Distribution

The procedure and method used by the licensee to verify that the plant is operating within the power distribution limits defined in TS were reviewed and discussed with cognizant licensee personnel. The data taken by the Movable Incore Detector System was digitized and stored by the plant computer. This information was then fed into a time-shared large scale CDC-7600 computer owned by United Information Services, Inc. which performed the core power distribution calculation using the licensee's version of the Westinghouse "Incore" code.

Flux maps taken and analyzed per the Procedure RA-17, Rev. 2, Flux Map Analysis approved August 30, 1985 to support the cycle 5 startup and power operation are tabulated below for the available power level up to 49% of the rated power.

		Fq		$F^N_{\Delta H}$	
Date (Flux Map)	Power Level (%)	Measured Value	TS Limit	Measured Value	TS Limit
10-2-85	0.3	2.6490	4.26	1.6276	2.015

4.2.2 Core Thermal Power

The inspector reviewed the Procedure SOP-RPC-6, Rev. 9, Reactor Thermal Power Calibration and the calculation results of October 22, 1985 through October 23, 1985, and verified that the Core Thermal

Power was determined adequate for the procedure. The final readings of the Feedwater Flow Power calculations were all within ±0.5% of the rated core thermal power of 3025 Mwt.

The inspector was informed that the calorimetric calculations employing the Leading Edge Flow Measurement (LEFM) System were the most accurate method since the LEFM could measure the feedwater flow accurately and avoid the inherent uncertainty involved in measuring steamflow.

During the Startup Testing Period at approximately 31%, 41% and 50% power plateaus, the licensee performed heat balance comparisons between the plant computer outputs and hand calculations. The inspector also performed an independent calculation. The hand-calculated core thermal power of 1509.67 Mwt is based on the inspector's observation of control room instrument readings. All comparisons were in good agreement as shown below.

Test Date	Power Level	Method	Results (Mwt)
10-22-85	31%	Procedure SOP-RPC-6, Rev. 9	934.73
		Licensee Hand Calculation	934.86
10-22-85	41%	Procedure SOP-RPC-6, Rev. 9	1247.51
		Licensee Hand Calculation	1247.35
10-23-85	50%	Procedure SOP-RC-6, Rev. 9	1501.98
		Licensee Hand Calculation	1502.00
		Inspector Calculation	1509.67

No noncompliance was identified.

4.2.3 Target Axial Flux Difference Determination

Target Axial Flux Difference Determination was performed in accordance with the Procedure RA-10, Rev. 4 approved April 6, 1984. The inspector reviewed the latest calculation performed on October 25, 1985 and verified that this new delta flux target was entered into the plant process computer. The inspector toured the control room and verified that this required information was in use by the reactor operators.

No noncompliance was identified.

5.0 Training and Qualification

The inspector reviewed the qualification of the two members of the reactor engineering staff and noted that both of them were engineering school graduates. The site reactor engineer had a Master's degree and 7 years experience and the Assistant Plant Engineer had a Bachelor's degree and 2½ years experience at Indian Point Unit 3. The personnel met the training and qualification guidelines of ANSI 18.1-1971.

6.0 QA/QC Interface in Cycle 5 Refueling/Startup Testing

The inspector interviewed NYPA QA/QC personnel on the subject of QA's role in Cycle 5 Refueling/Startup testing and reviewed QA surveillance report "Standard Audit Report No. 85-14" currently being prepared. Through discussion and documents review, the inspector verified that the licensee QA/QC organization played an active role in Cycle 5 Refueling/-Startup testing coverage. To further strengthen QA coverage in this area, a licensee QA representative stated that QA will verify test results and surveillances at appropriate power plateaus for the startup physics testing in progress.

No noncompliance was identified.

7.0 Control Room Observations and Facility Tours

The inspector observed control room operations for control room manning and facility operation in accordance with the Technical Specification requirements and administrative procedures. Inspection tours of the Turbine/Generator areas were conducted.

No noncompliance was identified.

8.0 Exit Interview

Licensee management was informed of the purpose and scope of the inspection at the entrance interview. The findings of the inspection were periodically discussed and were summarized at the conclusion of the inspection on October 25, 1985. The licensee indicated that the inspection information discussed at the exit meeting did not contain proprietary information. At no time during this inspection was written material given to the licensee or any licnesee representative.

1.0 Persons Contacted

*J. E. Russel, Superintendent of Power
*F. W. Gumble, Site Reactor Engineer
*J. J. Anderson, Assistant Plant Engineer
L. Kelly, Performance and Reliability Supervisor
*F. X. Pindar, QA Superintendent
J. Gillen, General Chemistry Supervisor
G. Tasick, QA Supervisor
M. Morrissey, Performance Supervisor
J. Somrai, I&C General Supervisor
*W. D. Hamlin, Assistant to Resident Manager
*S. L. Munoz, Technical Services Superintendent

U. S. NRC

*P. Eselgroth, Chief, Test Program
P. Koltay, Senior Resident Inspector
*U. Cheh, Reactor Engineer

The inspector also contacted other licensee employees in the course of the inspection.

*Denotes those present at the exit interview on October 25, 1985.

2.0 Cycle 5 Start-up Physics Test Program

The start-up physics test program was conducted according to Indian Point No. 3 Cycle 5 Start-up Physics Test Program, RA-7, Revision 3, Approved August 30, 1985. The test program outlined the steps in the testing sequence, set initial conditions and prerequisites, specified calibration or surveillance procedures at appropriate points, and referenced detailed test procedures and data collections in attachments. Initial criticality of Cycle 5 was achieved on October 2, 1985. Upon completion of the Zero Power Physics Test, the unit experienced a water chemistry problem and held at lower power of approximately 50% rated power. The Power Ascension Tests will depend on water chemistry.

The inspector independently verified that the predicted values and acceptance criteria were obtained from "Plant Operations Package for the Indian Point Unit 3 Power Plant Cycle 5", WCAP-10873 dated July 1985. The inspector reviewed test results and documents described in this report to ascertain that the start-up testing was conducted in accordance with technically adequate procedures and as required by Technical Specifications (TS). The details and findings of the review are described in Sections 3 and 4.

3.0 Cycle 5 Start-up Physics Testing Precritical Tests

The inspector reviewed calibration and functional test results to verify the following:

-- Procedures were provided with detailed instructions;
-- Technical contents of the procedures were sufficient to result in satisfactory components and tests;
-- Instruments and calibration equipment used were traceable to the National Bureau of Standards;
-- Acceptance and operability criteria were observed in compliance with TS.

The following tests were reviewed:

3.1 Control Rod Checks and Tests

The rod drop measurement was performed in accordance with the Procedure, Full Length Rod Drop Time Test, 3 PT-R4, Rev.4 approved October 17, 1985. The inspector verified by review of the test results performed on September 27, 1985, that Rod Cluster Control Assemblies (RCCA) were tested for drop times and the individual RCCA drop times were all less then 2.4 seconds as required by the TS. The inspector also reviewed several visicorder traces and verified that the drop times had been interpreted correctly.

No noncompliance was identified.

3.2 Incore Thermocouple, Wide Range RTD and Narrow Range RTD Measurement

The Indian Point Unit 3 plant computer system of Westinghouse P-250 are currently being replaced to Perkin-Elmer 3240 and the thermocouple readings are available, but are not ready for calibration against the Narrow Range RTD readings. The thermocouples are not used for any safety systems at Indian Point Unit 3. Westinghouse P-250 Operator's Console Reference Manual for Nuclear Power Plant Supervision TP044-P dated August, 1972 will still be used to take all the P-250 operator functions on Perkin-Elmer 3240,

No noncompliance was identified.

3.3 Reactivity Computer Setup/Verification

The reactivity computer was set up and calibrated according to the Procedure STS-126, Rev. 4 on May 7, 1985. The reactivity computer was adjusted with the correct inputs of delayed neutron fractions (betas) and decay constants (lambdas). An exponential test signal was fed into the reactivity computer. The dynamic response was then completed with predicted values which were derived from point reactor kinetics. The results of this calibration check were satisfactory.

The reactivity computer was further checked when the reactor reached critical. Comparisons of predicted and measured reactivities based on doubling time measurement were accepted.

No noncompliance was identified.

3.4 Cycle 5 Reload Safety Evaluation and Core Verification

The inspector reviewed the Procedure SOP-RP-1, Rev. 4, Preparation for and Return from Refueling approved July 25, 1985, and the reload was conducted per the procedure.

The Cycle 5 reactor core is comprised of 193 fuel assemblies. During the cycle 4/5 refueling, 76 fresh fuel assemblies (batch T) were loaded into the core. The remaining 117 fuel assemblies were from previous cycles operation. The reload safety evaluation (RSE) along with the required Technical Specifications (TS) change was submitted to the NRC for review. This reload submittal was found acceptable (Letter from John D. Neighbors (NRC) to John C. Brons (NYPA) dated August 27, 1985). The basic assumption used in the RSE was Cycle 4 burnup of 14,090 ± 500 MWD/MTU. The inspector verified the actual cycle 5 burnup to be 14713 ± 500 MWD/MTU. The highpoint burnup of 15200 MWD/MTU was used for the RSE. The assumption is thus valid.

The inspector reviewed one half of the core verification videotape #1 dated August 19, 1985 and verified that the core loading agreed with the intended core loading plan.

4.0 Cycle 5 Startup Physics Testing - Post Critical Tests

The inspector reviewed selected test programs to verify the following:

-- The test programs were implemented per the Cycle 5 Startup Physics Test Program;

-- Step-wise instructions of the test procedures were adequately provided including Precautions, Limitations and Acceptance Criteria in conformance with the requirements of the TS;

-- Provisions for recovering from anomalous conditions were provided.

-- Methods and calculations were clearly specified and the tests were performed accordingly.

-- Review, approval and documentation of the results were in conformance with the requirements of the TS and the licensee's administrative controls.

The following tests were reviewed.

4.1.1 Zero Power Physics Test

The licensee measured the just critical boron concentration per the Procedure RA-2, Rev. 3, Initial Criticality approved August 30, 1985 and control rod worth per the Procedure RA-04, Rev. 3, Control Rod Worth Measurements approved August 30, 1985. The inspector reviewed the data and noted the following results:

Rod Position	Predicted Value (PPM)	Test Value (PPM)
All Rods Out (ARO) (RA-2, Rev. 3, Initial Criticality)	1500 ± 50	1496
D IN	1380 ± 207	1402
D+C IN	1301 ± 195	1307
D+C+B IN	1225 ± 183	1234
D+C+B+A IN (RA-4, Rev. 3, Control Rod Worth Measurements)	1092 ± 163	1115

4.1.2 Isothermal Temperature Coefficient

Isothermal temperature coefficients were measured and documented per the Procedure RA-3, Rev. 4, Isothermal Temperature Coefficient. The inspector noted the following results:

Rod Position	Predicted Value (pcm/°F)	Measured Value (pcm/°F)
ARO	-1.30±3	-1.401
D IN	-2.66±3	-2.88
D+C IN	-4.72 ± 3	-4.85

The isothermal temperature coefficient (ITC) is defined as the change in reactivity for a unit change in the moderator, clad and fuel pellet temperatures. Thus, the ITC can be interpreted as the sum of the moderator and Doppler coefficient. The doppler coefficient is difficult to measure in normal operation. A value of -2.03 pcm/°F was obtained from Westinghouse Report, "The Nuclear Design and Core Management of the Indian Point Unit No. 3 Cycle 5" WCAP-10839. Thus, during zero power physics testing, the ARO Moderator Temperature Coefficient (MTC) was determined to be 0.629 pcm/°F. Therefore, a withdrawal curve was generated to meet the TS requirement of 3.1.C.1.

4.1.3 Control Rod Worth Measurement

The control rod reactivity worth measurements were performed per the Procedure RA-4, Rev. 3, Control Rod Worth Measurements. The following results were noted:

Configuration	Predicted Worth (pcm)	Measured Worth (pcm)
Control Bank D	1043 ± 156	1064.5
Control Bank C (D IN)	699 ± 104	718
Control Bank B (D+C IN)	667 ± 100	633
Control Bank A (D+C+B IN)	1186 ± 178	1205.3
Total	3595 ± 359.5	3620.8

No noncompliance was identified.

4.2 Power Ascension Tests

4.2.1 Core Power Distribution

The procedure and method used by the licensee to verify that the plant is operating within the power distribution limits defined in TS were reviewed and discussed with cognizant licensee personnel. The data taken by the Movable Incore Detector System was digitized and stored by the plant computer. This information was then fed into a time-shared large scale CDC-7600 computer owned by United Information Services, Inc. which performed the core power distribution calculation using the licensee's version of the Westinghouse "Incore" code.

Flux maps taken and analyzed per the Procedure RA-17, Rev. 2, Flux Map Analysis approved August 30, 1985 to support the cycle 5 startup and power operation are tabulated below for the available power level up to 49% of the rated power.

		F_Q		$F^N_{\Delta H}$	
Date (Flux Map)	Power Level (%)	Measured Value	TS Limit	Measured Value	TS Limit
10-2-85	0.3	2.6490	4.26	1.6276	2.015

4.2.2 Core Thermal Power

The inspector reviewed the Procedure SOP-RPC-6, Rev. 9, Reactor Thermal Power Calibration and the calculation results of October 22, 1985 through October 23, 1985, and verified that the Core Thermal

Power was determined adequate for the procedure. The final readings of the Feedwater Flow Power calculations were all within ±0.5% of the rated core thermal power of 3025 Mwt.

The inspector was informed that the calorimetric calculations employing the Leading Edge Flow Measurement (LEFM) System were the most accurate method since the LEFM could measure the feedwater flow accurately and avoid the inherent uncertainty involved in measuring steamflow.

During the Startup Testing Period at approximately 31%, 41% and 50% power plateaus, the licensee performed heat balance comparisons between the plant computer outputs and hand calculations. The inspector also performed an independent calculation. The hand-calculated core thermal power of 1509.67 Mwt is based on the inspector's observation of control room instrument readings. All comparisons were in good agreement as shown below.

Test Date	Power Level	Method	Results (Mwt)
10-22-85	31%	Procedure SOP-RPC-6, Rev. 9	934.73
		Licensee Hand Calculation	934.86
10-22-85	41%	Procedure SOP-RPC-6, Rev. 9	1247.51
		Licensee Hand Calculation	1247.35
10-23-85	50%	Procedure SOP-RC-6, Rev. 9	1501.98
		Licensee Hand Calculation	1502.00
		Inspector Calculation	1509.67

No noncompliance was identified.

4.2.3 Target Axial Flux Difference Determination

Target Axial Flux Difference Determination was performed in accordance with the Procedure RA-10, Rev. 4 approved April 6, 1984. The inspector reviewed the latest calculation performed on October 25, 1985 and verified that this new delta flux target was entered into the plant process computer. The inspector toured the control room and verified that this required information was in use by the reactor operators.

No noncompliance was identified.

5.0 Training and Qualification

The inspector reviewed the qualification of the two members of the reactor engineering staff and noted that both of them were engineering school graduates. The site reactor engineer had a Master's degree and 7