William J. Cahill, Jr.
Vice President

Central File

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Telephone (212) 460-3819

December 31, 1975

Re: Indian Point Unit No. 2
AEC Docket No. 50-247

Mr. James P. O'Reilly, Director
Office of Inspection and Enforcement, Region 1
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. O'Reilly:

Table 13.3-1 in Section 13.3 of the Unit No. 2 Final Facility Description and Safety Analysis Report lists those tests to be performed from the initial core loading to rated power. The results of those tests relating to reactor physics were previously submitted to the Commission via our letter to Mr. Edson G. Case, dated March 21, 1975. In accordance with the requirements of Section 6.9.1 of our Technical Specifications and Regulatory Guide 10.1 (Revision 2), we are herewith submitting twenty-five (25) copies of the results of the remaining tests required by Table 13.3-1.

In accordance with prior committments to your Mr. Anthony Fasano, we are also submitting the results of two special tests not included in the above listing. These were the "Generator Load Trip Test" and "RCFC Condensate Measuring System Functional Test".

Very truly yours,

William J. Cahill

William J. Cahill, Jr.
Vice President

cc/ Mr. Donald F. Knuth, Director (2 copies)
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. William G. McDonald, Director (2 copies)
Office of Management Information and Program Control
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

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Note 1 - The results of these tests were previously submitted to the Commission as part of the Indian Point Unit No. 2 Startup Physics Test Report.

Control Rod Drop Test (IPP-SU-4.10.2)

The purpose of this test was:

1. To determine the drop time of each full length control rod under four plant conditions: cold, no flow; cold, full flow; hot, no flow; hot, full flow.

2. To determine ten additional drop times for the rod having the fastest drop time and the rod having the slowest drop time.

3. To measure withdrawal speed at maximum stepping rate and compare to speed specified in the operating manual at hot full flow conditions.

All control rod drop times to the dashpot were less than the 1.8 second maximum permitted by the Technical Specifications for full flow and operating temperature conditions. Drop times for all other test flow and temperature conditions were also less than the 1.8 second maximum. A final compilation of the drop times is as follows:

RC System Parameters	Time in Seconds Initiation of Event to Dashpot Entry	Time in Seconds Initiation of Event to Bottom of Dashpot
Cold, No Flow	1.09 min., 1.18 max.	2.35 min., 2.52 max.
Cold, Full Flow	1.35 min., 1.45 max.	2.94 min., 3.32 max.
Hot, No Flow	1.02 min., 1.07 max.	1.92 min., 2.05 max.
Hot, Full Flow	1.02 min., 1.29 max.	2.17 min., 2.45 max.

The fast and slow rod selection was based on total drop time. On this basis, control rod K-2 was selected as the slowest rod and control rods G-3 and E-9 were selected as the fastest rods. Each of these rods was dropped an additional ten times with the following results:

Control Rod	K-2	G-3	E-9
Avg. Drop Time*	2.52	2.31	2.34
Max. Drop Time*	2.54	2.34	2.36
Min. Drop Time*	2.47	2.25	2.30

* Time in seconds from initiation of event to bottom of dashpot.

The withdrawal speed of the control rods at the maximum stepping rate was measured and found to be not in excess of the speed specified in the operating manual.

Thermocouple/RTD Intercalibration (IPP-SU-4.11.1)

This procedure provided for the functional checkout and cross calibration data for the in-core thermocouples and reactor coolant RTD's during hot functional testing. Specifically the objectives of the test were:

1. Verification of expected resistance versus temperature characteristics of RTD's.

2. Verification of expected millivolt versus temperature characteristics for thermocouples.

3. Determination of isothermal corrections for individual thermocouples.

The test was conducted by disconnecting the RTD's from their normal readout terminals and connecting them to multiposition low contact resistance switches. A calibrated precision decade box was connected to a multiposition switch to providing a reference for checking any drift in the readout instrumentation during each temperature run. The output of the multiposition switches was connected to a calibrated ohms converter/DVM or resistance bridge. At discrete temperature intervals between 250°F and the 547°F hot zero power coolant temperature, the RTD resistances were measured and recorded when the Reactor Coolant System reached thermal equilibrium. During the RTD measurements, in-core thermocouple readouts were obtained and recorded.

The average of all RTD readings was determined and was considered to be the true temperature for each run. Any RTD reading differing from the average by more than 2°F was excluded from the average. A comparison was then made between the true temperature and the individual RTD average temperatures and the difference recorded as the RTD installed correction. The installed corrections were then utilized as appropriate for calibration of the downstream process instrumentation.

A comparison was made between individual in-core thermocouples and the average RTD temperature and a set of correction factors determined. These correction factors were applied to the Honeywell direct reading thermocouple monitor readouts and the P-250 process computer.

Automatic Reactor Control Test

IPP-SU-5.10.1

I. Introduction

The Automatic Reactor Control test was completed on April 1, 1974 and verified the ability of the automatic control system to maintain the average coolant temperature within 1.5°F of the programmed reference temperature during steady-state operation and within acceptable limits during load changes. Also, the ability of the automatic control system to respond to step changes in the average coolant temperature was verified.

II. Ramp Changes in Load

With the reactor initially at equilibrium conditions at 50% rated power, the power was increased to 75% with the reactor in automatic control. As the load on the generator was increased at the rate of approximately .5% per minute, the controlling bank (Bank D) moved automatically and continuously from 127 steps to 167 steps withdrawn. Plant parameters including pressurizer level control, pressurizer pressure, steam generator level were continuously monitored, and these systems were found to perform satisfactorily with changes in plant load. Also during automatic control, an adequate margin to trip existed for the overtemperature ΔT trip and the overpower ΔT trip. The test was repeated satisfactorily for a power descension from 75% to 50% power.

III. Transient Recovery of the Reactor Using Automatic Control

With the reactor control system initially in manual operation, T_{avg} was increased to approximately 6°F above T_{ref} by control bank withdrawal. When switched to automatic, the system responded very well reaching within 1.5°F of the T_{ref} value in approximately 2 minutes. No oscillation in the amplitude of the T_{avg} signal was observed. The pressurizer pressure response was found to be acceptable in maintaining the pressure at its setpoint value of 2235 psig. The pressurizer level was found to be within a few percent absolute of the setpoint value during the transient.

The transient test was repeated for a decrease in T_{avg} relative to T_{ref}. In this case, T_{avg} stabilized within 1.5°F of the T_{ref} signal in 2 minutes and 5 seconds.

IV. Conclusions

The excellent performance of the automatic control system precluded making any setpoint changes to improve plant performance.

Therefore, the performance of the automatic reactor coolant system in maintaining T_{avg} within steady-state limits was verified.

Load Swing Test (IPP-SU-9.1)

The purpose of the Load Swing Tests was to verify the nuclear plant transient response, including automatic control system performance, when a 10% step load change was introduced at the turbine generator.

Step load changes of 10% were initiated from steady state conditions at approximately 50%, 75% and 100% of rated licensed power. Various plant parameters were recorded during the resultant transient to determine system response. In addition, minimum and maximum values were noted for comparison with test acceptance criteria.

As a result of an examination of the data recorded and the visicorder traces obtained, it was concluded that the various minima and maxima requirements were not exceeded as well as all acceptance criteria were met. It was, therefore, concluded that the test was satisfactory.

Plant Trip Test (IPP-SU-9.5)

Plant trip tests were performed at 35% and 100% of rated licensed power. The purpose of these tests was as follows:

1. To verify the ability of the primary and secondary plant to sustain a trip from 35% and 100% power and to bring the plant to stable conditions following the transient.
2. To determine the overall response time of the reactor coolant hot leg resistance temperature detectors.
3. To evaluate the data resulting from this test to determine possible changes in control system setpoints in order to improve transient response based on actual plant operation.

Both plant trips were initiated by manually tripping the turbine. Several variables, as specified in the test procedure, were recorded on high speed test recorders during the transient. In addition, pertinent plant parameters from normal plant instrumentation were recorded before and after the transient with maximum and minimum values noted during the transient.

Acceptance criteria used to determine successful test completion was applied to the data from the plant trip at 100% power only and per approved test procedure did not apply to the plant trip test conducted at 35% power. An evaluation of the results of the 100% power plant trip test indicated that all six acceptance criteria were met. The pressurizer and steam generator safety valves did not lift. Safety injection was not initiated. The overall RTD response time was 4.8 seconds (versus an upper limit for acceptance of 7.3 seconds). The nuclear flux dropped to the 15% level in 1.17 seconds (versus an upper limit of 2.0 seconds). And finally, the full length rods did release and drop.

Thus on the basis of the above, it was concluded that the test was satisfactory.

Pressurizer Spray Flow Verification

(IPP-SU-4.1.9)

Test results indicated that continuous spray into the pressurizer was adjusted to prevent excessive temperature differentials between reactor coolant system, surge line, spray line and pressurizer nozzle during steady state conditions of plant operation. The spray control valves 455A and 455B performed satisfactorily. The opening of these valves did indeed cause a pressure reduction in the pressurizer at the required rate. The effectiveness of these valves was demonstrated by the test.

Flow Coastdown Measurement (IPP-SU-9.3)

The purpose of this test was to measure the rate at which reactor coolant flow rate changes subsequent to to various reactor coolant pump stops and starts and to measure various delay times associated with the loss of flow accident. The test was performed by recording the following parameters on a high speed strip chart recorder as various reactor coolant pump(s) are tripped:

1. Elbow tap differential pressure from each of the four reactor coolant pumps.
2. Position (on/off) of each reactor coolant pump motor breaker.
3. Output of low flow trip relays for all four loops.
4. Rod position indication for at least one of the withdrawn control rods.

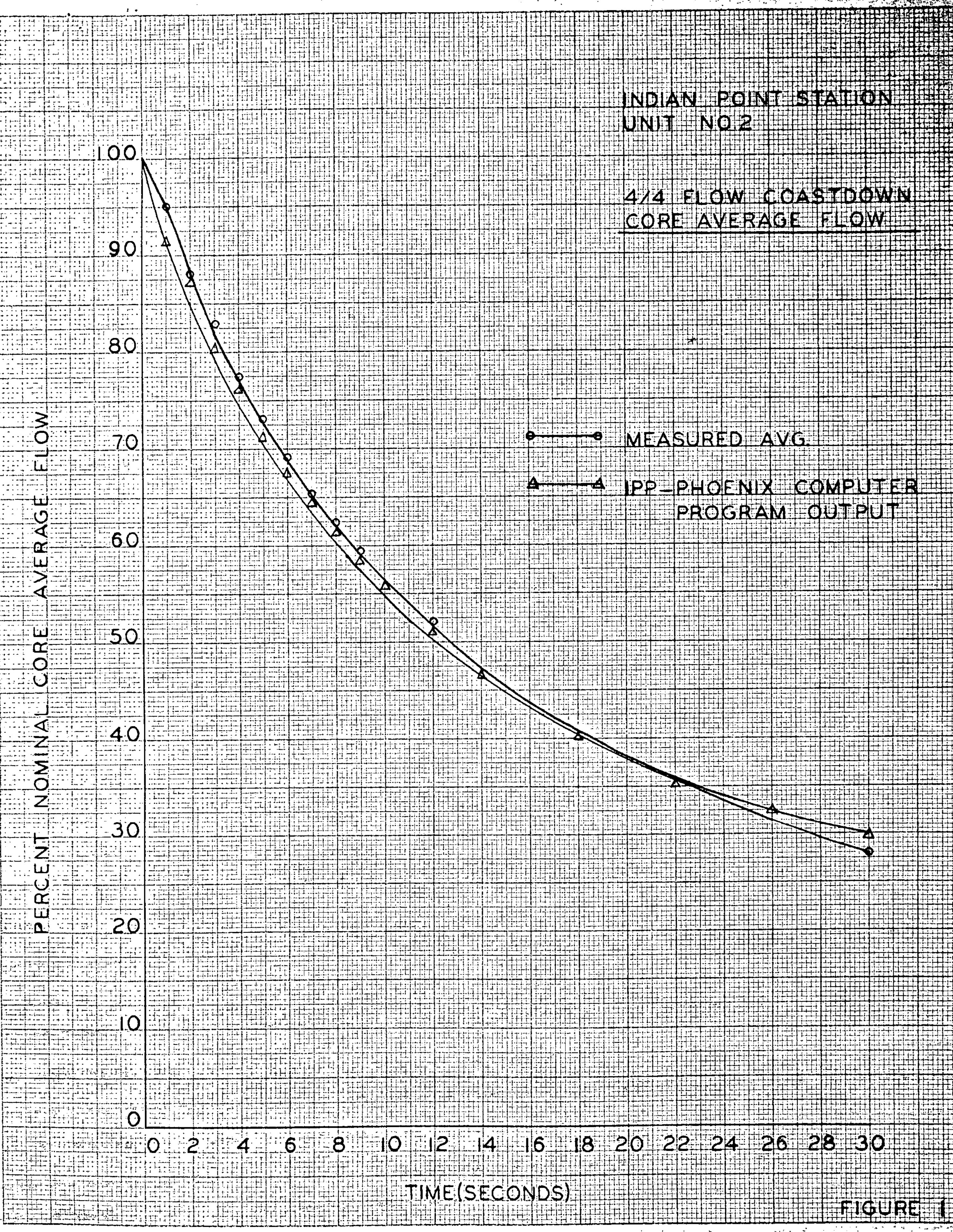
The test was conducted with the reactor coolant system at hot shutdown conditions with all RCC assemblies fully inserted except as modified in the procedure. Elbow differential pressure readings were reduced to coolant flow values (fraction of initial differential pressure value as full loop flow). These fractional flow values for one out of four and four out of four flow coastdown were then plotted against calculated curves using the Phoenix Computer code and design system parameters. The measured curves (See Figures 1 thru 3) were in close agreement with the calculated curves. It was noted there was a small deviation from the FSAR predicted flow coastdown curve for the loss of one out of four reactor coolant pumps. The accident analysis in the Unit No. 2 Fuel Densification Report dated January, 1973 included Section 6.6.2 covering the locked rotor loss of flow case. Calculations for the instantaneous seizure of one out of four pumps show a minimum DNB ratio of 1.353 as shown in Figure 6.6 of this report. Since the measured curve lies above the assumed locked rotor flow coastdown curve, it was considered acceptable.

Figures 4 and 5 show core average flow for loss of one out of three pumps and three out of three pumps respectively.

Measured time delays as compared to FSAR values were as follows:

	Measured	FSAR
Low flow time delay	1.93 sec.	1.967 sec.
Under voltage trip delay	0.274 sec.	1.2 sec.
Under frequency trip delay *	0.450 sec.	0.6 sec.

* Time set point reached to initiation of rod motion.


INDIAN POINT STATION
UNIT NO.2
4/4 FLOW COASTDOWN
CORE AVERAGE FLOW
MEASURED AVG.
IPP-PHOENIX COMPUTER PROGRAM OUTPUT
PERCENT NOMINAL CORE AVERAGE FLOW
100
90
80
70
60
50
40
30
20
10
0
0 2 4 6 8 10 12 14 16 18 20 22 24 26 28 30
TIME (SECONDS)
FIGURE 1

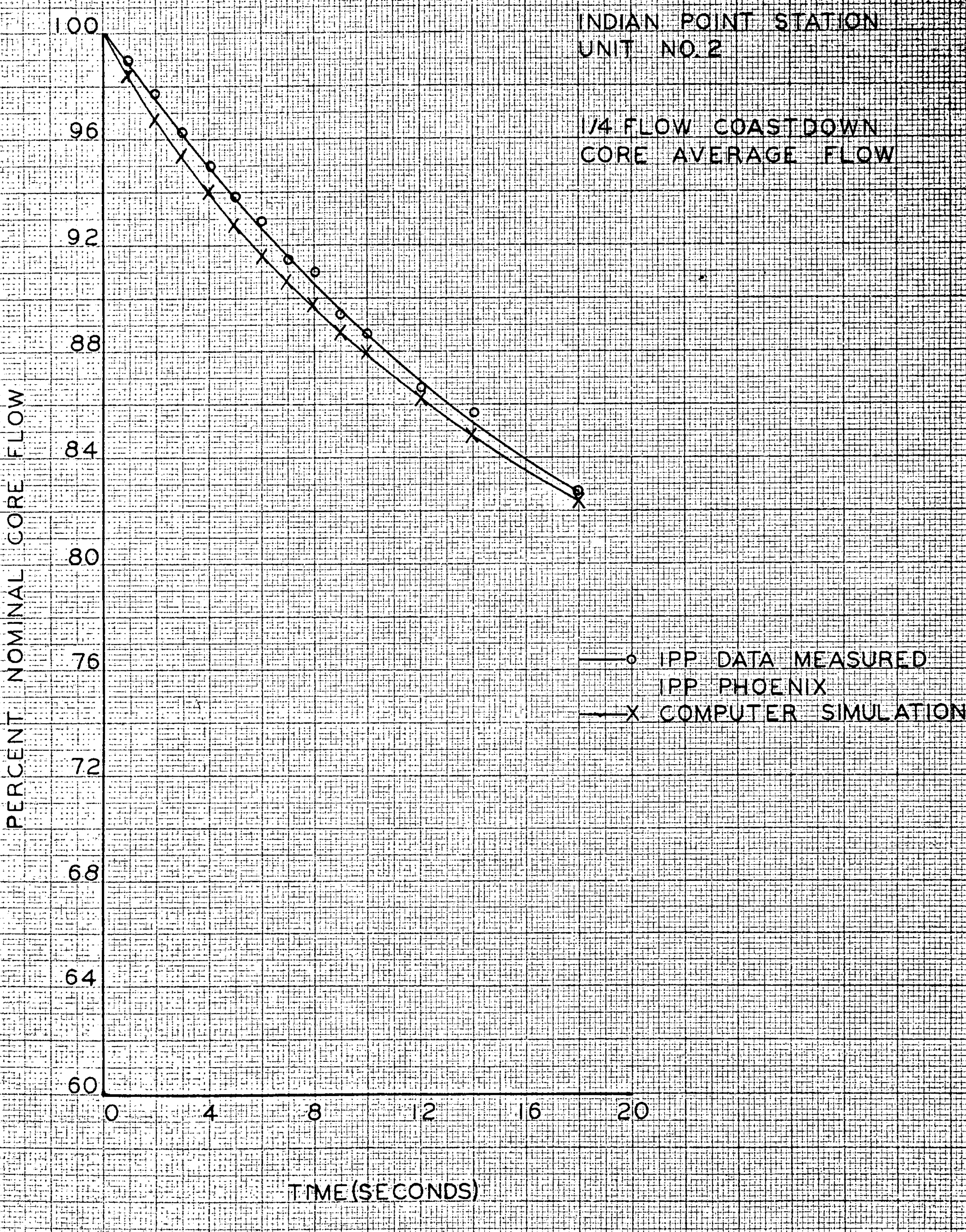

INDIAN POINT STATION
UNIT NO. 2
1/4 FLOW COASTDOWN
CORE AVERAGE FLOW
PERCENT NOMINAL CORE FLOW
100
96
92
88
84
80
76
72
68
64
60
0
4
8
12
16
20
TIME (SECONDS)
IPP DATA MEASURED
IPP PHOENIX
COMPUTER SIMULATION

FIGURE 2

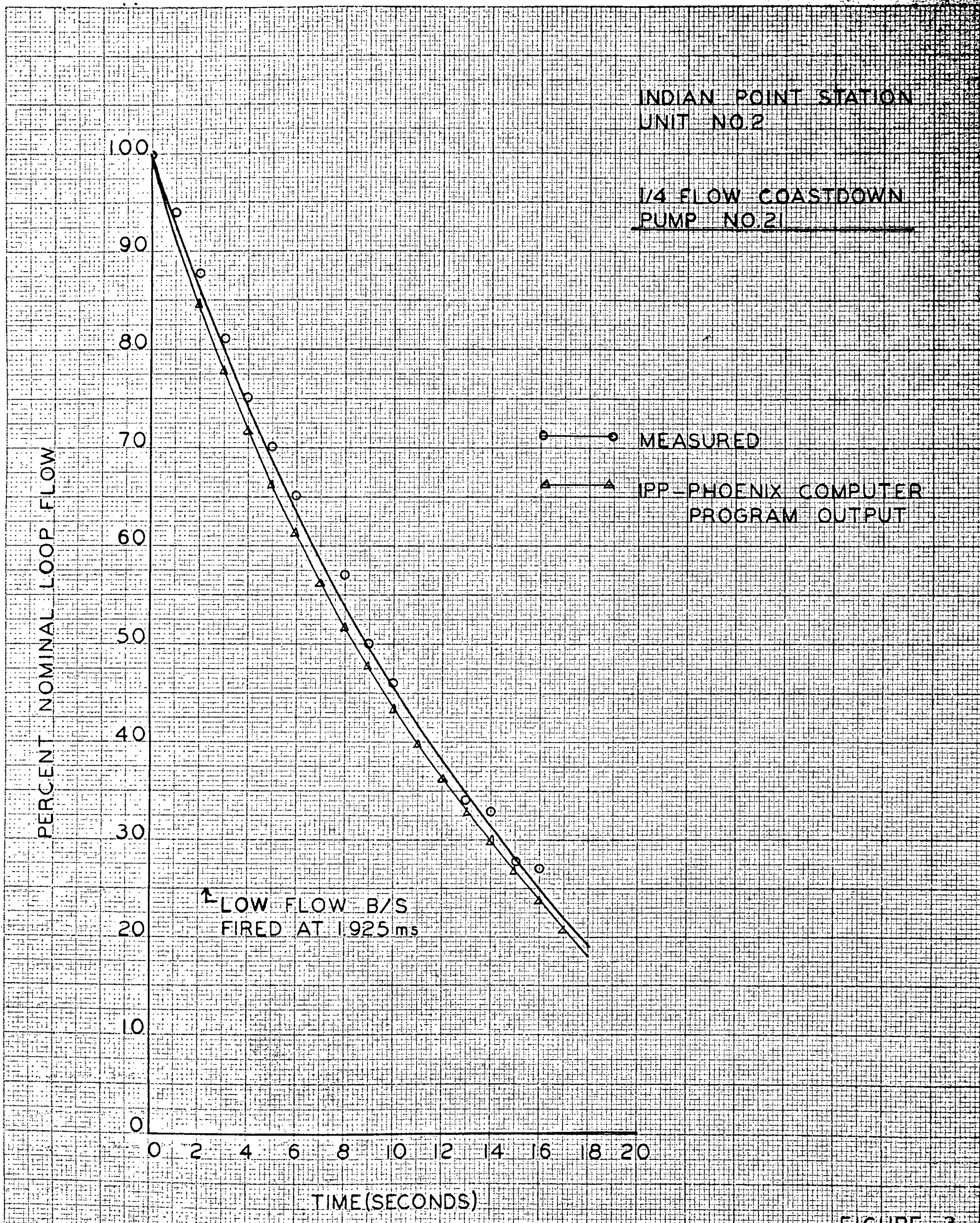

INDIAN POINT STATION
UNIT NO.2
1/4 FLOW COASTDOWN
PUMP NO.21
MEASURED
IPP-PHOENIX COMPUTER PROGRAM OUTPUT
PERCENT NOMINAL LOOP FLOW
100
90
80
70
60
50
40
30
20
10
0
0 2 4 6 8 10 12 14 16 18 20
TIME (SECONDS)
LOW FLOW B/S
FIRED AT 1.925 ms

FIGURE 3

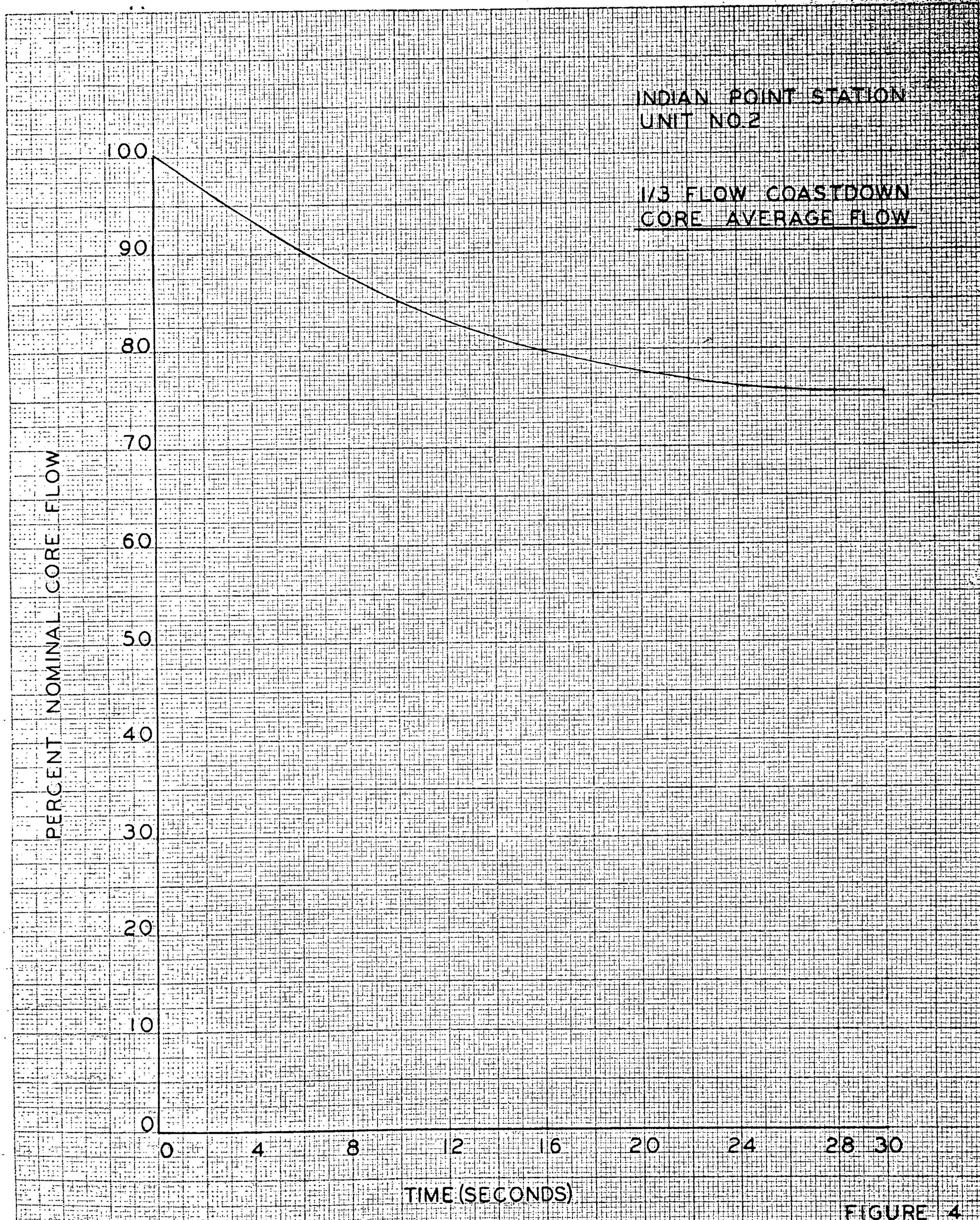

INDIAN POINT STATION
UNIT NO.2
1/3 FLOW COASTDOWN
CORE AVERAGE FLOW
PERCENT NOMINAL CORE FLOW
100
90
80
70
60
50
40
30
20
10
0
0
4
8
12
16
20
24
28
30
TIME (SECONDS)

FIGURE 4

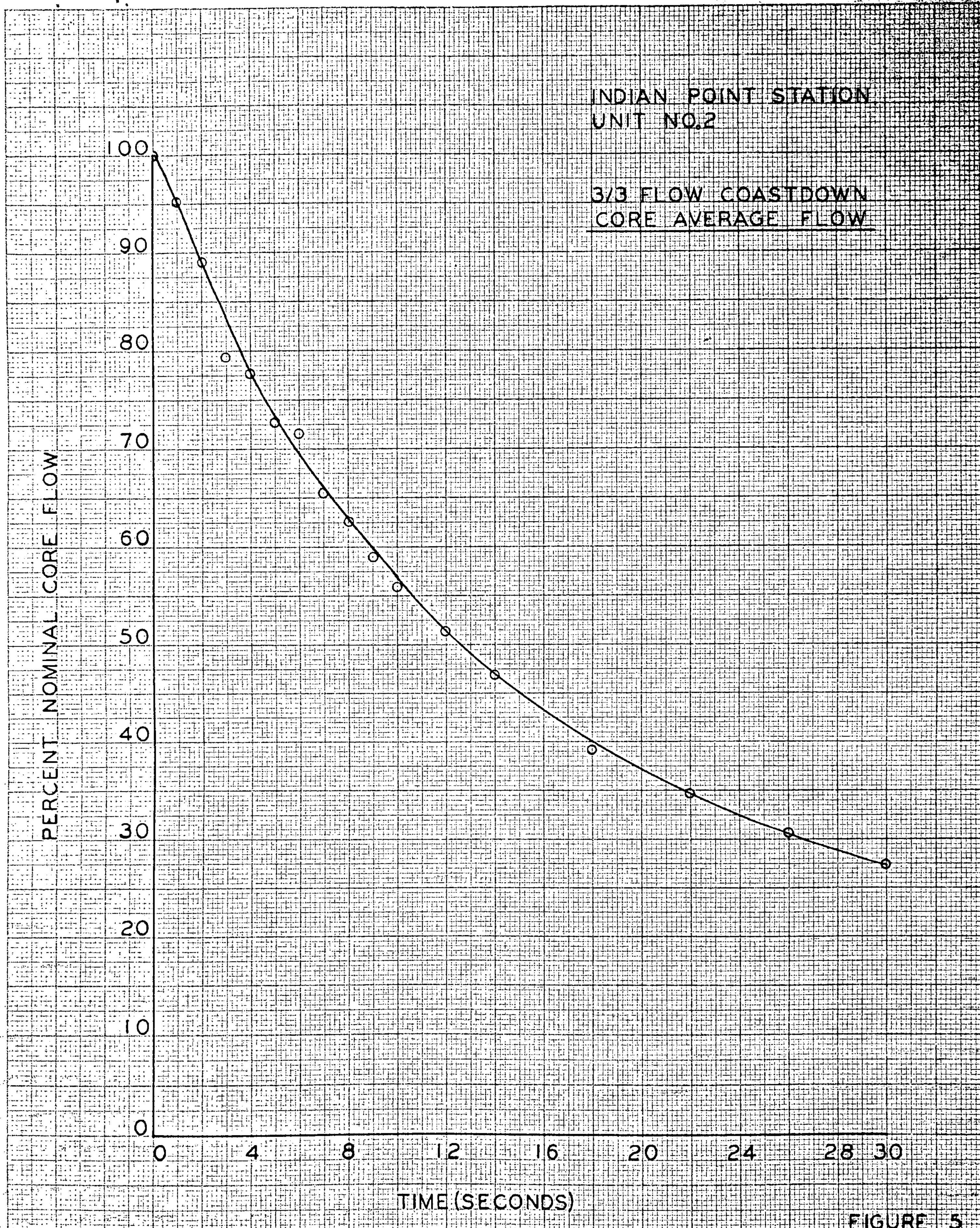

INDIAN POINT STATION
UNIT NO.2
3/3 FLOW COASTDOWN
CORE AVERAGE FLOW
PERCENT NOMINAL CORE FLOW
100
90
80
70
60
50
40
30
20
10
0
0
4
8
12
16
20
24
28
30
TIME (SECONDS)

FIGURE 5

Dynamic Rod Drop Test

IPP-SU-9.6

I. Introduction

The Dynamic Rod Drop Test was conducted on April 5, 1974 at a reactor power of approximately 75%. With the reactor control, turbine generator control and feedwater pump speed control in the automatic mode, RCC E-9 was dropped into the core initiating a turbine runback. The plant transient response during the turbine runback was followed by continuous recording of the following parameters: power range channels, trip bistables, T_{avg}, ΔT, pressurizer pressure and level, steam header pressure, steam generator levels, steam and feedwater flow, feedwater pump speed and control bank position.

II. Test Results and Conclusions

In the static rod drop test IPP-SU-8.3, RCC E9 was found to have the largest measured rod worth (149 pcm) and was therefore chosen as the rod to be dropped dynamically. A power reduction of 12.7% caused by the highest worth dropped rod was calculated using an average (measured) power coefficient of -11.7 pcm/% power. Thus the amount of the load cutback is set to be greater than or equal to 12.7%.

Test data indicated that the rod drop detection circuitry operated as expected and adjustments or setpoint changes were not required. The turbine runback rate was calculated to be 1.32% power per second. A safety analysis indicates that a 1% per second runback rate is sufficient for the dropped rod transient accident so that the 1.32% measured value is conservative even with an allowance being made for instrumentation inaccuracies. The turbine cutback time was initially set for 12.5 seconds so that a 16.5% reduction in power was achieved. This load cutback during the runback was greater than the power reduction caused by the highest worth dropped rod (RCC-E9). However, since a conservative load reduction of 30% was desired, the cutback time was increased to 23 seconds.

The automatic response of the primary and secondary plant parameters were found to be satisfactory. During the runback, the expected reduction in feedwater flow and steam flow (of approximately $.4 \times 10^6$ lbs/hr) took place. At no time did ΔT approach (within 2°F) the overtemperature ΔT or overpower ΔT setpoints during the transient, indicating that the core was fully protected from DNB.

The operation of the alarms, automatic reactor control system, and the blocking of automatic rod withdrawal all performed satisfactorily.

Lastly, the ability to retrieve the dropped RCC manually was demonstrated.

Large Load Reduction Tests (IPP-SU-9.4)

The purpose of these tests was (1) to verify the ability of the primary and secondary plant and the automatic reactor control systems to sustain a 50% step load reduction from 75% and 100% of full power, (2) to evaluate the interaction between the control systems, and (3) to evaluate the test data to determine if possible setpoint changes are required in the control systems in order to improve transient response based on actual plant operation. The acceptance criteria used to determine successful test completion were as follows:

1. Reactor and turbine did not trip.
2. Safety injection was not initiated.
3. Pressurizer safety valves did not lift.
4. Steam generator safety valves did not lift.
5. No manual intervention was required to bring the plant conditions to equilibrium values following the transient.

For both of the load reductions described above, the acceptance criteria for successful test completion were met. During the performance of the 50% load reduction from 100% of full power, the heater drain tank pumps tripped automatically upon low tank level which in turn resulted in an overspeed trip of No. 21 main boiler feedwater pump. In order to reduce the potential for boiler feed pump trip under these conditions, a ten turn potentiometer was installed in the pump control circuit to allow finer control of the speed signal from the control room. In addition, the speed controller for each boiler feed pump was calibrated such that the pump speed was limited to approximately 4900 rpm for the 50 milliamp maximum signal from the controller. Adjustments were also made to the heater drain tank level control instrumentation. Some problems were also encountered during this particular test with the visicorder traces for certain operating parameters; however, since the acceptance criteria were met and there were no safety criteria applicable, it was judged that these problems did not warrant repeating the test.

Load Follow Test (IPP-S.U.-10.4)

The purpose of the test was to demonstrate the ability of the plant to follow programmed load changes both with and without part length rods and to verify that the reactor core is capable of performing programmed load changes. The test was divided into two parts as follows:

> Part I utilized full-length control rods only (part-length rods were fully withdrawn from the core at all times) and reactor coolant system boron concentration changes to accomplish the programmed load changes.
>
> Part II utilized both full-length and part-length control rods and adjustments of the reactor coolant system boron concentration to accomplish the programmed load changes.

Both parts of the test were organized to include daily load cycles resembling, to the best estimate of the Company's System Operations Department, the anticipated typical load-follow schedule for the plant. The tests also included demonstrations of high ramp rate change capability.

Part I of the test comprised four (4) daily load-follow cycles starting at 0000 hours of day 1, with the daily load cycle as follows:

1st Day

3 hours of linear power decrease (16.7%/hr) - 2 hours at 50% power - 3 hours of linear power increase (16.7%/hr) - 14 hours at full power (FP) - 2 hours of linear power decrease (16.7%/hr)

2nd Day

1 hour power decrease at 16.7%/hr - 5 hours at 50% power - 3 hours power increase at 16.7%/hr - 13 hours at FP - 2 hours of power decrease (16.7%/hr)

3rd Day

1 hour power decrease at 16.7%/hr - 5 hours at 50% power - 3 hours power increase at 16.7%/hr - 13.5 hours at FP - 1 hour power decrease at 50%/hr - 0.5 hr at 50% power

4th Day

7 hours at 50% power - 1 hour power increase at 50%/hr - FP (until end of Part I)

Based on analysis of data collected during the test, it was concluded that the boration and dilution systems were capable of handling the above load swings. In addition, it was demonstrated that the flux difference, ΔI, could be maintained well within the target band at all times and that all core variables (boron concentration, rod position, etc.) agreed well with predictions. The core average burnup at the time of the test was approximately 3000 MWD/MTU.

Part II of the test comprised three (3) daily load follow cycles starting at 0000 hours of day 1 with the daily load cycles as follows:

1st Day

power decrease from 100% at 1.6%/min - 6.5 hrs. at low power (55-60%) - power increase at 1.3%/min - 16.5 hours at full power (FP).

2nd Day

0.5 hr. at full power - power decrease at 1.8%/min - 6 hrs. at approximately 50% power - power increase at 2.0%/min - 16.5 hrs. at full power.

3rd Day

1 hr. at full power - power decrease at 2.8%/min - 5.75 hrs. at approximately 50% power - power increase at 4.1%/min - full power (up to the end of Part II).

Based on analysis of data collected during the test, it was concluded that the boration and dilution systems had ample capacity to handle the load changes throughout the duration of the above test and that power changes at rates up to 5%/minute could be accommodated without exceeding power distribution limits specified in the proposed Technical Specifications based on ECCS/FAC evaluation. It was also demonstrated that the flux difference (ΔI) could be maintained within the target band at all times. The core average burnup at the time of the test was approximately 4000 MWD/MTU.

It should be noted that load follow capability will be restricted during later core life due to capacity limitations of the chemical and volume control systems.

Initial Turbine Roll (IPP-T.P-4.31.2)

The purpose of this test was to perform an initial roll of the main turbine generator utilizing the reactor coolant pumps and the stored energy in the reactor coolant and secondary water as a source of steam supply. During the initial roll, the turbine was monitored closely for any signs of rubbing or other unusual conditions. Since the maximum speed attained was 1725 revolutions per minute, checks planned to be performed at synchronous speed were not accomplished. These checks were completed immediately prior to initial synchronization.

No unusual conditions were observed during the performance of this test and the results were found acceptable by the Manufacturer and Con Edison.

Turbine Generator Checkout - Hot and Cold

(IPP-T.P.-4.31)

The purpose of this test was to verify that the turbine, generator, exciter, controls and associated equipment and components were installed and checked out according to the Manufacturer's specifications prior to initial operation. The turbine generator checkouts were successfully accomplished under the direction of an on site Manufacturer's Representative. Data was collected on special check out forms provided by the Manufacturer. A copy of all completed forms was submitted to Con Edison for review and record purposes.

Turbine Generator Low Power Tests

There was no formal test of the turbine generator under low power conditions. Various equipment checks were made by the Manufacturer. These included verification of turbine overspeed protection set points. Minor problems were encountered and corrected during the testing and performance of the unit was found to be within the manufacturer's limitations.

Main Turbine Steam Stop Valve Test

(IPP-T.P.-4.28.2)

The purpose of this test was to verify that the turbine generator control valves and stop valves could be checked for freedom of movement with the Unit under load. The test was conducted at approximately 310 MWe gross and consisted of slowly closing the control valve on one side of the Unit by means of the valve test control switch. At the same time the control valve was being closed, the governor was ran in the open direction to open the other control valves so as to maintain the load constant. The corresponding stop valve was observed to close when the control valve closed position limit switches energized the stop valve test solenoid valve. The same testing sequence was utilized for the remaining three sets of control and stop valves. All four sets of control and stop valves were found to operate correctly when tested in the above manner.

Plant Reliability Demonstration and Heat Rate Determination Test (IPP-SU-10.1)

The purpose of this test was (1) to demonstrate that the plant will achieve a sustained net plant output (turbine load less essential station auxiliaries) of 872,890 KW, (2) to verify that when the plant is operating at the net plant output of 872,890 KW that the net plant heat rate does not exceed 10,790 BTU/KW-Hr. and (3) to provide the basis for plant acceptance by maintaining a net plant output of 872,890 KW for 100 continuous hours.

All three of the above objectives were successfully accomplished during a test which started on June 26, 1974 and was completed on June 30, 1974. Calculations showed the plant net heat rate to be no more than 10,724 BTU/KW-Hr. The plant achieved a sustained net plant output of approximately 865,000 KW for the entire test period of 100 hours except for a slight reduction in output for two days when No. 23 circulating water pump was out of service for environmental considerations. This output was well within the design net output of 872,890 KW $^{+0}_{-2}$%.

Generator Load Trip Test (IPP-SU-9.7)

The purpose of this test was to obtain a reference point of turbine overspeed for comparison with calculated levels and to determine the need for further overspeed protection that would satisfy the design and code limitations set for the unit. A trip test was performed on September 7, 1973 from approximately 50% of licensed full power. An analysis of the data from this test indicated very good agreement with the manufacturer's turbine overspeed response curve. The measured overspeed was 11.3%. When normalized to the design back pressure of 1.5 inches of mercury, the corrected overspeed was 12.0%. This value was in very good agreement with the predicted overspeed of 12.2%.

Although the test procedure required a generator load trip test from 100% of licensed full power, an analysis of data from several generator trips at or near full power indicated that the measured overspeed for all of these trips was very close to the predicted overspeed. In all cases, they were below the predicted values. On the basis of this analysis, it was concluded that the overspeed performance of the unit had been demonstrated thus satisfying the intent of the test and precluded the need for a formal generator load trip from 100% of full power. Documentation of this position was contained in a letter dated July 2, 1975 from Mr. William J. Cahill, Vice President, to Mr. Eldon J. Brunner, USNRC Region 1 Reactor Operations Branch.

Plant Shutdown from Outside the Control Room

(Special Instruction S-15)

The purpose of this test was to demonstrate that the plant can be safely shutdown and controlled from outside the control room in accordance with Emergency Procedure E-5 (Control Room Inaccessibility). The results of the test demonstrated that controls and information available in the local control stations were functioning properly and are sufficient to permit the operators to trip the plant, control heat removal, and borate in an orderly manner to reach and maintain the reactor in a hot shutdown status should the control room ever become uninhabitable.

RCFC Condensate Measuring System Functional Test

(IPP-SU 4.14.1)

The purpose of this test was to verify the capability of the condensate measuring system to detect a reactor coolant system leak of one gallon per minute per detector. The test was conducted by injecting raw steam (simulated leak) within the crane wall of the containment of Elevation 46' and observing the effects on the containment dew point recorder and the containment fan cooler/filter condensate flow measuring devices.

In order to provide a meaningful indication of steam injected into the containment, as measured by the house service boiler steam flow recorders, the rate of injection was increased to about 5,000 pounds per hour (about 10 gpm). As steam was being injected, containment dew point temperatures and condensate pot weir levels were recorded at fifteen minute intervals. An analysis of the data indicated reasonably good agreement between the rate of leakage as indicated by the containment dew point and condensate collection systems and the simulated leak. The dew point system indicated leakage was about 8 gpm and that indicated by the condensate collection system was about 7.6 gpm.

Reactor Coolant System Flow Measurement

(IPP-SU-4.1.13)

The purpose of this test was to provide a means of obtaining the necessary data to interrelate reactor coolant pump input power and elbow tap ΔP as an accurate measurement of absolute Reactor Coolant System (RCS) flowrate. The four reactor coolant pumps installed in the RCS are Model V1102A1 controlled leakage pumps manufactured by the Westinghouse Electromotive Division at Cheswick, Pennsylvania. Each is driven by a 6600 volt induction motor at a nominal speed of 1190 rpm with a power consumption at operating condition of about 4200 KW.

The basic strategy of the test was to measure the pump power and elbow tap differential pressure in one loop and de-energizing, in succession, the remaining pumps. Because of back flow in the de-energized loops, flow in the instrumented loop increases along with a corresponding decrease in pump head and required pump power. Four data points of pump power and relative loop flow were generated for each loop. These data were compared to the predicted performance curves for the pump-impeller combinations in the loops. By assuming various flow rates for the normal (4 pumps in operation) condition and calculating the increased flow for the other combinations, the four data points were filled to the shape of the performance curve. The assumed flow which gave the best fit was then taken as the best approximation to the actual flow rate.